April 28, 2000

Mr. A. Alan Blind
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (IP2) - TOPICS OF DISCUSSION FOR THE MAY 3, 2000, MEETING (TAC NO. MA8219)

Dear Mr. Blind:

By letter dated March 14, 2000, the U.S. Nuclear Regulatory Commission (NRC) staff requested information pertaining to your proposed steam generator tube examination program. The letter stated that the NRC staff is conducting a formal review of your steam generators. The focus of the review is on the steam generator inspections, problem identification, root cause analysis, and corrective actions. You provided the root cause analysis to the staff by letter dated April 14, 2000. The staff has performed an initial review of your evaluation and has determined that certain items were not addressed. A meeting is needed to allow the staff the opportunity to ask questions and provide comments as appropriate. The meeting is scheduled for May 3, 2000, at the U.S. Nuclear Regulatory Commission's offices at 11545 Rockville Pike, Two White Flint North, Room T 3 B 45, Rockville, Maryland 20852. Enclosed are topics you should be prepared to discuss. In addition, you may take the opportunity to share any additional information.

A response to these topics should be provided in subsequent submittals. If you have any questions, please contact me at (301) 415-1421.

Sincerely,

/RA/

Jefferey F. Harold, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosure: May 3, 2000, Topics for Discussion

cc w/encl: See next page

Mr. A. Alan Blind
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

April 28, 2000

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (IP2) - TOPICS OF DISCUSSION FOR THE MAY 3, 2000, MEETING (TAC NO. MA8219)

Dear Mr. Blind:

By letter dated March 14, 2000, the U.S. Nuclear Regulatory Commission (NRC) staff requested information pertaining to your proposed steam generator tube examination program. The letter stated that the NRC staff is conducting a formal review of your steam generators. The focus of the review is on the steam generator inspections, problem identification, root cause analysis, and corrective actions. You provided the root cause analysis to the staff by letter dated April 14, 2000. The staff has performed an initial review of your evaluation and has determined that certain items were not addressed. A meeting is needed to allow the staff the opportunity to ask questions and provide comments as appropriate. The meeting is scheduled for May 3, 2000, at the U.S. Nuclear Regulatory Commission's offices at 11545 Rockville Pike, Two White Flint North, Room T 3 B 45, Rockville, Maryland 20852. Enclosed are topics you should be prepared to discuss. In addition, you may take the opportunity to share any additional information.

A response to these topics should be provided in subsequent submittals. If you have any questions, please contact me at (301) 415-1421.

Sincerely,

/RA/
Jefferey F. Harold, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosure: May 3, 2000, Topics for Discussion

cc w/encl: See next page

DISTRIBUTION:

PDI-1 R/F	S. Black	J. Harold	OGC
PUBLIC	M. Gamberoni	G. Wunder	ACRS
B. Sheron	S. Little	L. Berry	M. Oprendek, Region I
E. Sullivan	J. Zwolinski	J. Strosnider	E. Murphy
S. Coffin	S. Long	D. Screnci (E-Mail)	

DOCUMENT NAME: G:\PDI-1\IP2\0503meeting topics.wpd
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		SC:PDI-1					
NAME	JHarold		SLittle		MGamberoni					
DATE	4/28/00		4/28/00		4/28/00					

Official Record Copy

TOPICS OF DISCUSSION

PUBLIC MEETING MAY 3, 2000

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

ISSUES IDENTIFIED FROM CON ED LETTER OF APRIL 14, 2000

DOCKET NO. 50-247

1. Section 2, second paragraph states that excessive noise prevented detection of R2C5 precursor signal in 1997. Weren't there other more fundamental contributing factors? For example, could not a correct calibration setup during the 1997 inspection have permitted the precursor signal in R2C5 to have been detected at that time? Could not a site-specific performance demonstration program in accordance with Electric Power Research Institute (EPRI) guidelines have alerted the licensee to the significant limitations of the generically qualified mid-range plus point for detecting primary water stress corrosion cracking (PWSCC) in the Indian Point 2 (IP2) small radius u-bends due to large amount of noise associated with surface deposits? Could not absence of adequate noise and data quality criteria in the data analysis procedures have also been an important contributing factor?

2. Section 2 states that inability to detect the precursor indication in R2C5 due to noise was the principle cause of the leakage event and that the growth rate of the indication between 1997 and 2000 was moderate and not a principle cause. Given the low voltage response of the 1997 signal and the high signal to noise, what level of accuracy can be assumed for the depth of the 1997 indication? On what basis? How does the inferred growth rate of this indication compare to expected growth rates based on comparable industry experience and laboratory crack growth data (e.g., NUREG/CR-5752)?

3. Section 3, "IP-2 SG Summary of Inspection History," provides a three-sentence description of the steam generators and an extremely brief description of the evolution of various degradation mechanisms in the IP2 steam generators (SGs). Additional historical details are requested as they may be considered precursors to the u-bend failure in the year 2000. For example, please discuss early (prior to the year 2000) evidence of flow slot hourglassing of the uppermost support in SG 23. When was this hourglassing first observed and how did the amount of hourglassing change over the years? What was the basis for not considering this hourglassing as significant (making it reportable in accordance with the technical specifications)? Were there criteria in place for considering hourglassing to be "significant" and thus reportable? What was the basis for not being concerned that more significant top support plate hourglassing might be present in SG 24 and why it wasn't found necessary to install hill side ports at this time to permit inspection of the upper tube support plate (TSP) flow slots in these SGs.

4. The root cause report should assess the leakage trends leading up to the failure event, a description and assessment of the effectiveness of the leakage monitoring program,

Enclosure

and whether there were any shortcomings in this program which prevented plant shutdown prior to the event. What was the alarm setpoint on the N-16 monitor? Discuss the operational status of the N-16 recorder prior to the event? What were the N-16 leakage measurements as a function of time in the hours and minutes leading up to the event? What was the time interval between the last reading and the failure event? What were the air ejector rad monitor readings during the hours and minutes leading up to the event? To what leak rate was the alarm setpoint on the air ejector monitor set?

5. Section 4.3 should address flow slot hourglassing data for all of the u-bend indications found in 1997 and 2000. This is necessary to demonstrate a relationship between the occurrence of hourglassing and u-bend cracks.

6. Section 5 states that an apex indication was found in SG 24, R2C67, in 1997 with a length of 0.4 inches. You elected not to perform an insitu pressure test of this location on grounds that the Westinghouse screening criteria were met. These screening criteria are intended to account for eddy current measurement error. What was the basis for the assumed measurement error? Was this assumption applicable to the very low signal to noise ratio existing for the subject tube? Describe the supporting qualification data for samples simulating the IP2 specific noise conditions. Apart from plugging the tube, you apparently took no further action at that time to assess the potential for significant flaws developing in the u-bend during the next operating cycle. Given the evidence of hourglassing of the uppermost support plates, the apex location of the R2C67 indication, and the quality of the eddy current inspection data for the inner row u-bends, and the experience from the Surry 2 tube rupture, why wasn't imminent failure of the inner row u-bends anticipated?

7. What was the basis for assuming the generic EPRI Appendix H qualification of the plus point mid-range probe for small radius u-bend inspection to be applicable to the site specific conditions at IP2 (high noise due to copper and magnitite surface deposits)? Why wasn't a site-specific qualification performed as called for in the EPRI guidelines?

8. Section 5 states that the 1997 precursor signal for R2C5 in SG 24 was not permitted to be seen because of the noise levels which were present. Why was this noise considered acceptable? Why weren't steps taken to reduce the noise?

9. Section 5 states that as a result of the tube failure investigation, a number of changes were incorporated into the analysis process. It is further stated that more stringent criteria were established for data quality. This implies that data quality criteria were employed during previous inspections. Please describe in detail the data quality criteria used previously. Were these documented? Were they for example, in the data analysis procedures? Were they addressed in the analyst training process? Please describe in detail the current data quality requirements and where they are documented. It is also stated in Section 5 that the analysis setup process was "changed" to achieve better resolution of the 20% ID calibration notch. Was the 1997 setup a correct setup in accordance with standard industry practice?

10. Describe the Appendix H qualification of the high frequency plus point for small radius u-bend inspections. What number of samples were included in the data set and what flaw

sizes were represented? Did the data set include representative copper and magnetite deposits and ovality? What kind of flaws were included in the data set (e.g., electrodischarge machining (EDM) notches, intergranular stress-corrosion cracking (IGSCC))? Did these flaws produce signals and exhibit signal to noise comparable to the situation at IP2? What were the results? What was the indicated probability of detection as a function of depth?

11. Section 6 states that only 22 tubes have been plugged due to tube restrictions since 1989, 20 of these were plugged in 1997 due to restrictions at, **or above**, TSP 6. The July 29, 1997, inspection report states that virtually all these restrictions involve restrictions in the u-bend. These restrictions included tubes in rows 2 through 4. Do these restrictions imply abnormal ovality in these u-bends? Describe your actions to characterize the current degree of ovalization and which tubes are affected. Is this ovality related to abnormal u-bend fabrication effects such as occurred at Doel Nuclear Plant and which led to a tube rupture at that plant? Might this be the root causal mechanism for the u-bend cracks at IP2? If not, why not? Alternatively, might this ovality have been induced by flow slot hourglassing?

12. Section 6 further states that the 1997 inspection was the first 100% inspection since startup and thus the noted plugging of restricted tubes in 1997 is believed to be the result of a larger inspection sample in 1997. Is this a reasonable explanation given the most affected SG (SG 22) received a 47% sample full length inspection in 1995 with the finding of no tube restrictions. Full length inspection samples in the other SGs in 1995 ranged from 67 to 100% with the finding of only one restricted tube.

13. Figure 5 is illegible. Please show each of the pictures on a full page. Pictures should indicate previously plugged tubes in rows 1 through 4 and at wedge supports. Also include pictures for inspection previous to 1995. Provide similar pictures for SG 22.

14. The root cause report should discuss the relative susceptibility of Alloy 600 mill-anneal tubing at IP2 to PWSCC relative to that for other plants. What was the range of mill anneal temperatures?

15. The root cause report should assess primary water chemistry as a potential contributing factor.

16. Section 7 doesn't address specified ovality limits on the small radius u-bends which may have been introduced during fabrication. What kind of post process inspections were performed to verify acceptable ovality, for example, ball gauge measurements? The data cited for Turkey Point isn't helpful here since the Turkey Point generators had experienced significant hourglassing in 1976.

17. Section 9.1 states that prior to the IP2 event, there have been no significant industry leakage events at the row 2 apex location. Have there been reported row 2 apex cracks? What were the circumstances? What about row 3? Apart from axial apex cracks and tangent point cracks, have there been other kinds of axial or circ ID or OD cracks affecting row 2 or row 3 u-bends? (NUREG/CR-5117 reported OD SCC at the apex of row 2 u-bends at Surry 2.)

Indian Point Nuclear Generating Station
Units 1/2

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. John McCann
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

Dave Lochbaum
Nuclear Safety Engineer
Union Concerned Scientists
1616 P Street, NW, Suite 310
Washington, DC 20036

Edward Smeloff
Pace University School of Law
The Energy Project
78 North Broadway
White Plains, NY 10603

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. Thomas Rose
Secretary - NFSC
Consolidated Edison Company
Of New York, Inc.
708 First Avenue
New York, NY 10017

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

Jim Riccio
Public Citizen's Critical Mass Energy Project
215 Pennsylvania Ave., SE
Washington, DC 20003

Michael Mariotte
Nuclear Information & Resources Service
1424 16th Street, NW, Suite 404
Washington, DC 20036