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June 13, 2000

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

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555-0001

Subject: Response to the Staff's Questions Regarding the Root Cause Evaluation for Steam Generator Tube Rupture Event of February 15, 2000 (TAC No. MA8219)

Reference: 1) Con Edison Letter to USNRC dated April 14, 2000
2) USNRC Letter to Con Edison dated April 28, 2000

Pursuant to 10 CFR 50.54(f), Consolidated Edison Company of New York, Inc. (Con Edison) hereby provides responses (Attachment A) to the staff's initial review of Con Edison's Root Cause Evaluation of the February 15, 2000 steam generator tube rupture event. This assessment was transmitted to the staff via Reference 1. Based upon its initial review, the staff determined that certain items were not addressed, and requested a meeting to allow the staff the opportunity to ask questions and provide comments as appropriate. Subsequent to the May 3, 2000 public meeting, it was requested that Con Edison provide written responses to the questions regarding the root cause evaluation. This letter provides Con Edison's responses to issues 4, and 15, which were identified in Reference 2. Additional responses will be forthcoming as they become finalized.

The commitments made in this correspondence are provided in Attachment B.

Should you or your staff have any concerns regarding this matter, please contact Mr. John McCann, Manager, Nuclear Safety & Licensing.

Sincerely,

Attachment

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Attachment A

Response to Root Cause Evaluation Issues Nos. 4, and 15

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
June 2000

Root Cause Evaluation–Issue No. 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, 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?

Response:

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, and whether there were any shortcomings in this program which prevented plant shutdown prior to the event.

Review of the primary to secondary leakage monitoring program and the 150 gpd administrative limit indicates that there were no shortcomings, which could have prevented a plant shutdown prior to the event. Less that 15 minutes prior to the event, primary to secondary leakage was confirmed to be less than 5 gpd as indicated by the N-16 radiation monitor. Although at the time the N-16 radiation monitor recorder was out of service, control room alarm and indication at the Accident Assessment Panel, and a local alarm were available. Industry experience has shown that the small amount of leakage prior to the event was not an indicator of an imminent tube failure. However, for the planned shortened operating cycle for the current steam generators, the administrative limit will be reduced from 150 gpd to 30 gpd. This is more conservative than the 75 gpd limit proposed by EPRI in the new guidelines which became effective February of 2000. Table 1 outlines the details of the program. Operations Procedure AOI 1.2, "Steam Generator Leak" will be revised prior to plant start up.

What was the alarm setpoint on the N-16 monitor?

Prior to the event, N-16 radiation monitor setpoints were 10 gpd, 25 gpd, and 150 gpd. The 150 gpd was the value specified in Operations Procedure AOI 1.2, "Steam Generator Tube Leak," that would require the plant to shut down. The Technical Specification value is 0.3 gpm (432 gpd) in any steam generator.

Discuss the operational status of the N-16 recorder prior to the event?

The N-16 monitor recorder has been out of service since April of 1999. However, as mentioned above, the N-16 monitor has control room indication via a common alarm on the Accident Assessment Panel. There is also a local alarm by the panel on the turbine floor. In the event that the common alarm were to activate an operator would be dispatched to investigate. The N-16 monitor recorder was repaired during the outage (Ref. WO # 99-08751).

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?

On February 15, 2000 at approximately 1915 hours, the N-16 monitor was indicating 3.4 gpd. The steam generator tube rupture event had initiated at approximately 1929 hours. A graph illustrating 24 steam generator primary to secondary leakage, as indicated by the N-16 monitor, is attached.

What were the air ejector rad monitor readings during the hours and minutes leading up to the event?

A graph illustrating the primary to secondary leakage rates, as derived by the air ejector radiation monitor readings from January 1, 1999 until just before the event, is attached.

To what leak rate was the alarm setpoint on the air ejector monitor set?

Air ejector radiation monitor (R-45) Hi alarm setpoint was 1.4 E-3 uCi/cc and alarmed at approximately 1918 hours. This setpoint is based upon equations contained in the Offsite Dose Calculation Manual. The setpoint is not based upon a specific primary to secondary leakage rate. The setpoint relates to 1 percent of the site boundary dose rate per 10 CFR 20.

Chemistry performs a primary to secondary leak rate determination by obtaining a sample of the air ejector output and analyzing it in the lab. Nuclide distribution and condenser air in-leakage could cause the activity concentration, which correlates to a primary to secondary leak, to vary. Chemistry takes these variables into account during their calculations, however these types of changes could vary the correlation between the R-45 radiation monitor and a primary to secondary leak rate slightly. Although R-45 provides a good overall trend, the N-16 monitors are better suited to identify individual steam generator leakage.

Table 1

IP2 Action Level Table for Primary-to-Secondary Leakage[1]

	Leak Rate In Any One SG[3]	Increasing Leak Rate[3]	Action Responses
No Operable Continuous RMS	0-5 gpd	NA	• Repair RMS ASAP • Take grab samples at frequency in EPRI, Table 3-1, or observe portable RMS at frequency, in EPRI, Table 3-1
Increased Monitoring	≥ 5 gpd < 30 gpd	NA	• Identify leaking SG and requantify leakage • Repair out-of-service RMS ASAP • Increase grab sample frequency and RMS monitoring frequency, recorrelate RMS, adjust RMS • Review procedures • Trend and report leakage • When conditions stabilize, reset set points
Action Level 1	≥ 30 gpd < 75 gpd	<1.25 gph over 1 hour period[2,4]	• Be in Mode 3 in ≤ 24 hr • Monitor rate of increase in leak rate[2] • Identify leaking SG and quantify leak rate • Increase frequency of RMS monitoring • Contain systems to minimize spread of contamination
Action Level 2a	≥ 75 gpd sustained for ≥ 1 hr	≥ 1.25 gph over 1 hour period[2]	• Reduce power to ≤50% in 1 hr[5] • Be in Mode 3 in next 2 hr[5] • Identify leaking SG and quantify leakage • Increase frequency of RMS monitoring • Contain systems to minimize spread of contamination
Action Level 2b	≥ 150 gpd		• Be in Mode 3 in ≤ 6 hr[5] • Identify leaking SG and quantify leakage • Increase frequency of RMS monitoring • Contain systems to minimize spread of contamination

Table 1 Notes

1. During power maneuvers it is impractical to accurately determine leak rates and their rate of change. During power increases, radiation monitors should be trended. If unusual behavior is noted, such as an unexpected increase in radiation monitor readings, consideration should be given to suspending power ascension, stabilizing conditions and quantifying leakage. During power reductions, radiation monitors should also be trended for indication of a large increase in leakage that would justify accelerated shutdown. Plant shutdown based on exceeding action levels in these guidelines should not be suspended based on radiation monitor reading until conditions are stabilized in Mode 3.

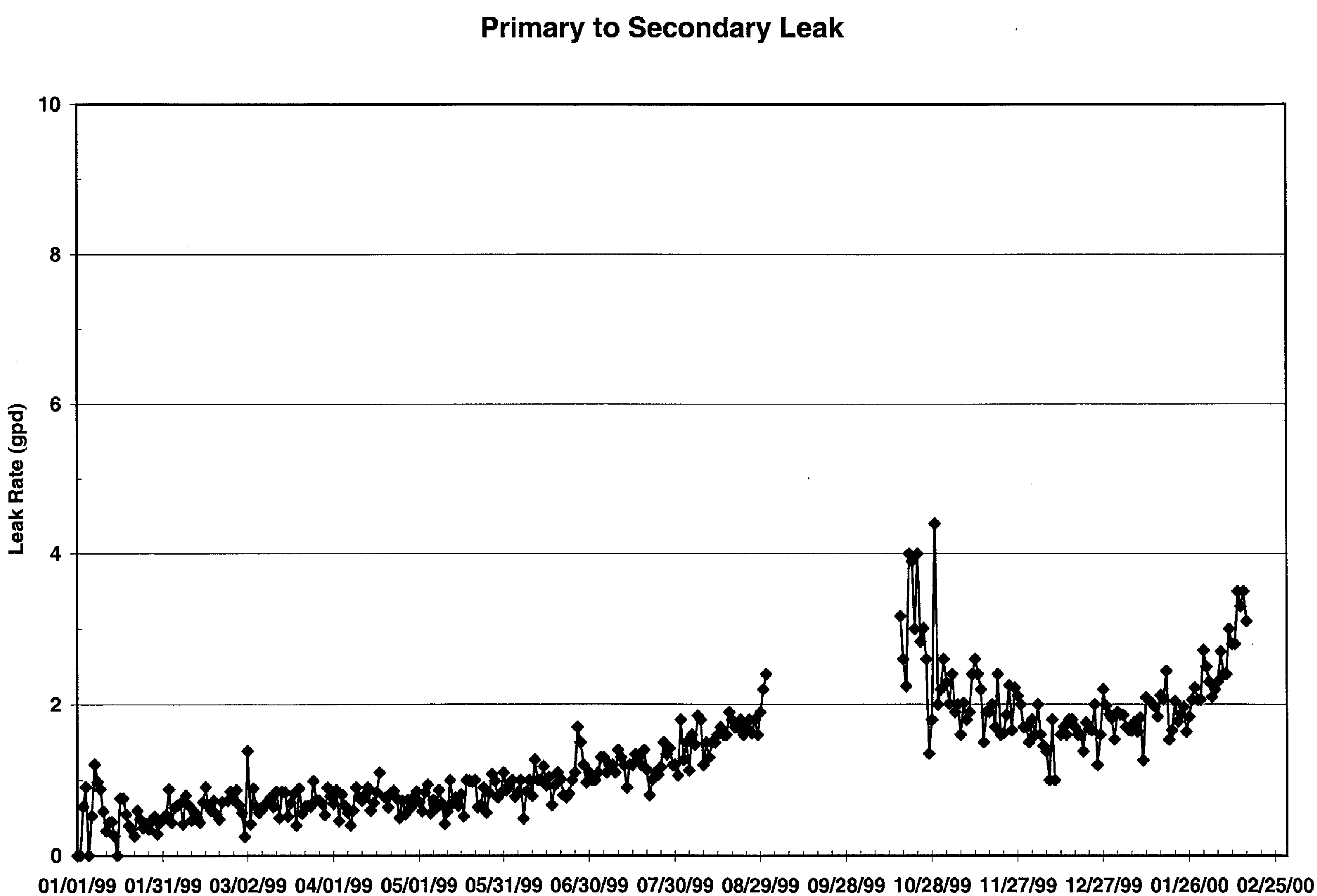
2. The rate of increase limit is provided to identify the potential need for a rapid power reduction to 50% power. This limit applies to progressively increasing leak rates and does not apply to leak rate spikes followed by decreasing leak rates. The rate of increase can be evaluated as leak rate changes over time intervals not exceeding 30 minutes. Alternate methods may be used to implement the rate of increase limit on a plant specific basis. Some examples of acceptable implementation include: 1) leak rate increase of 15 gpd above 75 gpd in a 30 minute period, 2) leak rate reaches 100 gpd within the 24 hour period allowed by Action 2 or 3) direct determination of leak rate slop changes taken from continuous monitor leak rate data showing that 30 gpd/hour has been reached and is not followed by a negative slope (i.e. is not due to a spike).

3. All leak rates and rates of increase in leak rates are in gallons based on room temperature measurements. 5 gpd equates to 1.7 lbm/hr, 30 gpd to 10.4 lbm/hr, 75 gpd to about 26 lbm/hr and 150 gpd to 52 lbm/hour.

4. Action Level 1 defines the action responses required when leakage is between 30 and 75 gpd and an increase in leakage rate that may be occurring but has not exceeded 1.25 gph over 1 hour period.

5 Action Level 2 defines the action response for two discrete events. The first addresses a sustained leak ≥ 75 gpd and an increasing leak rate greater than 1.25 gph over a 1-hr period. In this case the plant shall be in Mode 3 in 3 hrs. The second case addresses a significant leak ≥ 150 gpd that is not associated with a rapidly increasing leakage rate. This situation is considered less urgent and requires the plant to be in Mode 3 within 6 hrs.

Notes 1 - 3 from Table 3-2, EPRI PWR Primary-to-Secondary Leakage Monitoring Guidelines, Revision 2, February 2000


Primary to Secondary Leak
Leak Rate (gpd)
10
8
6
4
2
0
01/01/99
01/31/99
03/02/99
04/01/99
05/01/99
05/31/99
06/30/99
07/30/99
08/29/99
09/28/99
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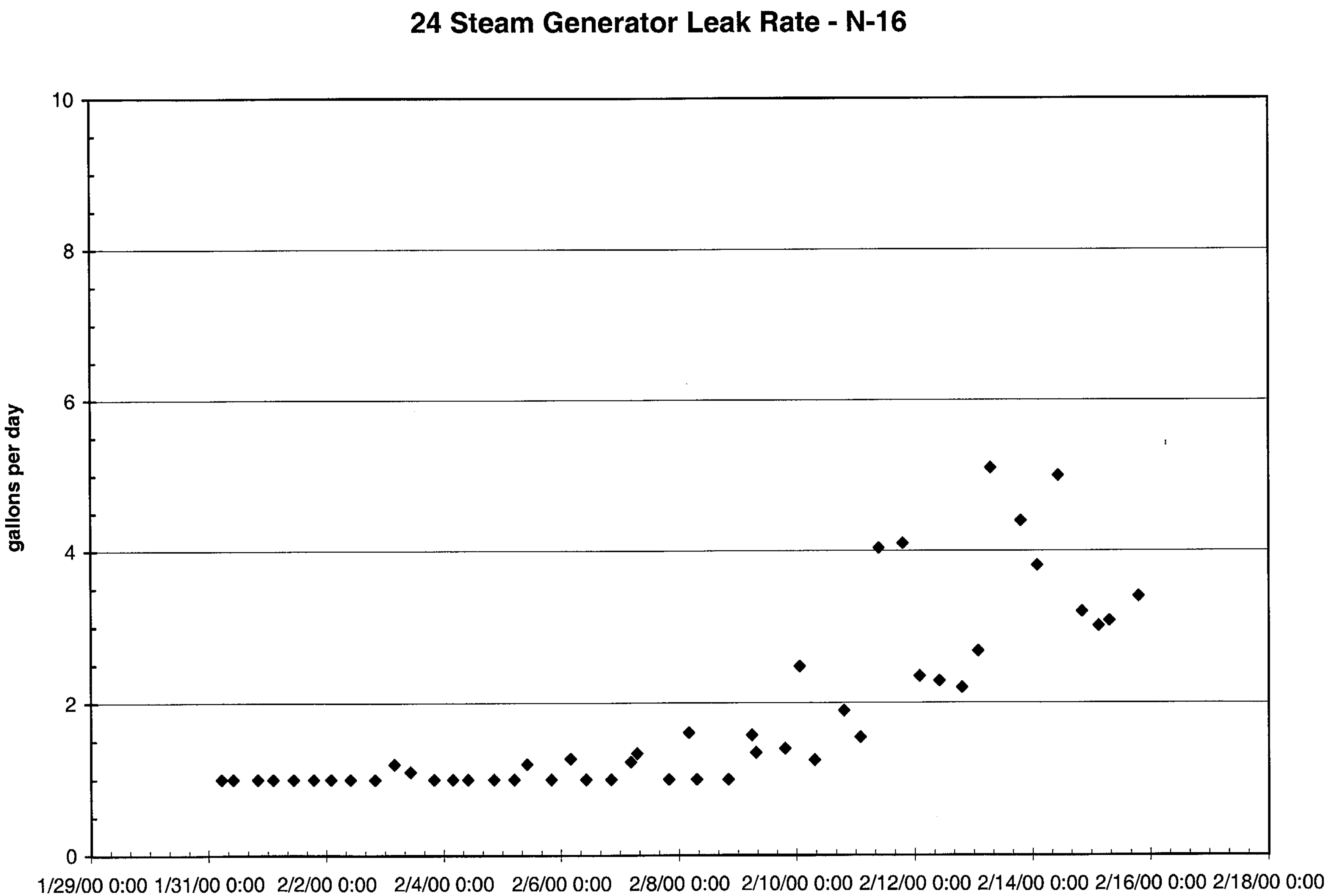
12/27/99
01/26/00
02/25/00



24 Steam Generator Leak Rate - N-16
gallons per day
0
2
4
6
8
10
1/29/00 0:00
1/31/00 0:00
2/2/00 0:00
2/4/00 0:00
2/6/00 0:00
2/8/00 0:00
2/10/00 0:00
2/12/00 0:00
2/14/00 0:00
2/16/00 0:00
2/18/00 0:00

Root Cause Evaluation – Issue No. 15

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

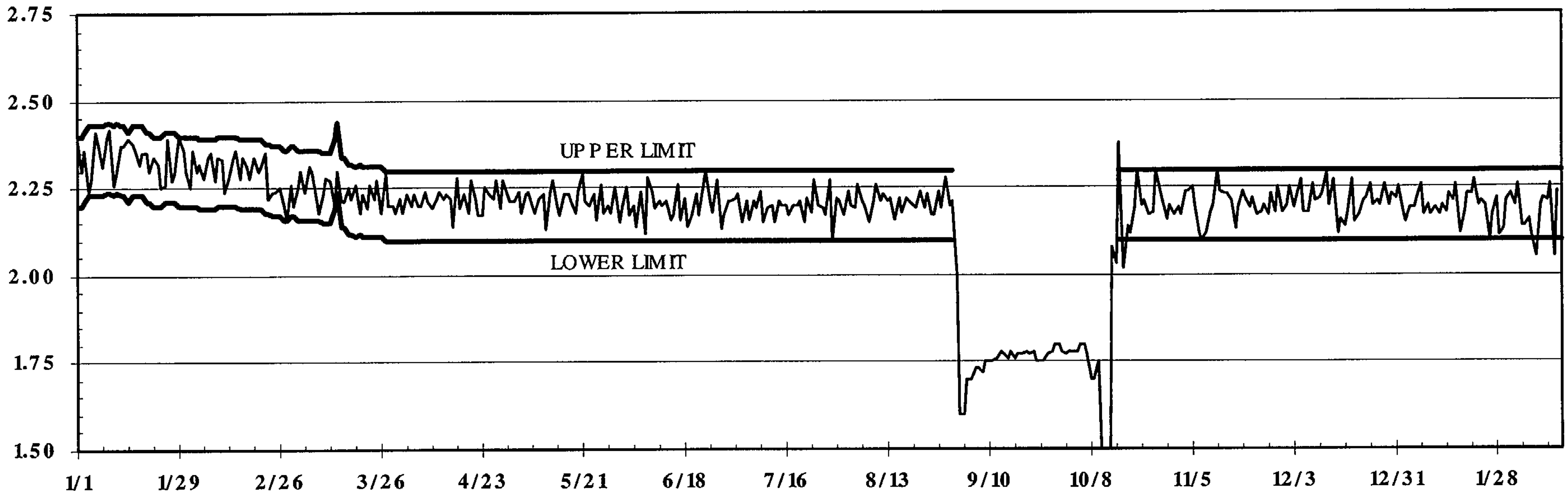
Response:

Indian Point's Primary Chemistry program has been developed in accordance with EPRI Guidelines, PWR Primary Water Chemistry Guidelines, Revision 4.

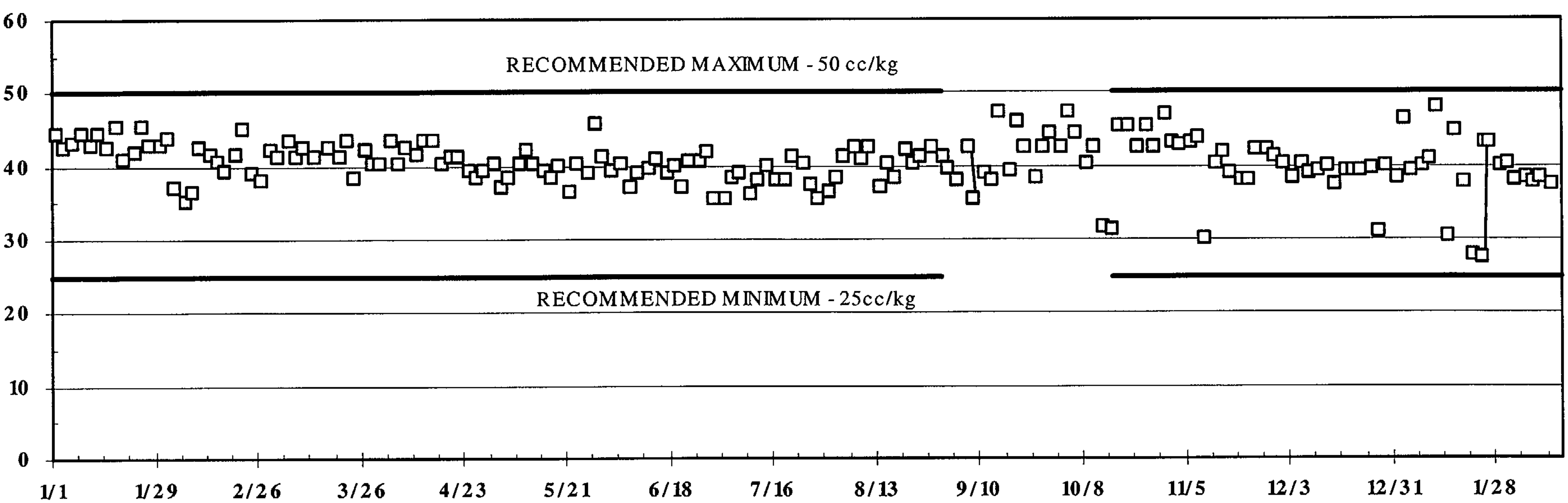
Based upon a review of the primary water chemistry history during the past operating cycle, Con Edison believes that water chemistry did not contribute to the occurrence of primary water stress corrosion cracking (PWSCC), which is the identified root cause for the tube leak. The primary system chemistry parameters reviewed for the operating cycle were boron/lithium concentration and hydrogen concentration. Boron/lithium concentration, which controls the pH, was maintained with procedural limits. Hydrogen concentration was also maintained within an approved band of 25- 50 cc/kg.

Attached graphs illustrate both of the above mentioned chemistry parameters to have been maintained within acceptable limits during the previous operating cycle.

REACTOR COOLANT LITHIUM



REACTOR COOLANT DISSOLVED HYDROGEN



Attachment B

List of Commitments

The following list identifies those actions committed to by Con Edison in this document. Any other actions discussed in the submittal represent intended or planned actions by Con Edison. These other actions are described to the NRC for NRC's information and are not regulatory commitments. Please notify Mr. John McCann, Manager, Nuclear Safety & Licensing of any questions regarding this document or any other associated regulatory commitments.

Commitment	Due Date
Implement new primary to secondary leakage administrative limits and actions level responses as detailed in Table 1 of response to Root Cause Evaluation-Issue No. 4. These requirements will be contained in Operations Procedure AOI 1.2, "Steam Generator Leak."	Operations Procedure AOI 1.2, "Steam Generator Leak" will be revised prior to plant start up.