William J. Cahill, Jr.
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

REGULATORY DOCKET FILE COPY

February 14, 1978

Re: Indian Point Unit No. 3
Docket No. 50-286

RECEIVED FEB 22 1978 U.S. NUCLEAR

Director of Nuclear Reactor Regulation
ATTN: Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Reid:

Your letter of January 12, 1978 requested that further assurance be provided that the potential consequences of the postulated fuel handling accident inside the Vapor Containment Building are well within the guidelines of 10 CFR Part 100. We believe that our analyses, which were provided to you by letters dated March 21, 1977 and June 15, 1977, assure that the conservatively calculated offsite consequences satisfy the requirements of 10 CFR Part 100. However, in response to your request, we intend to take the further measure which is described in your letter and increase the minimum time after reactor shutdown before fuel movement can take place. As recommended by members of the Regulatory Staff, we will increase this minimum time after shutdown to 120 hours.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

cc: Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

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William J. Cahill, Jr.
Vice President

REGULATORY DOCKET FILE COPY

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

February 15, 1978

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Reid:

Forwarded herewith for your information is a copy of the Tenth Quarterly Report for the Seismic Monitoring Program for Indian Point covering the months of September 1977 through November 1977.

Very truly yours,

William J. Cahill, Jr.
Vice President

Copy to: Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

WC/nvg

docket

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

F 122

February 10, 1978

50-247
50-286

To All PWR Facility Licensees

Gentlemen:

By letter dated December 9, 1977, copy enclosed, we requested you and all other PWR facility licensees to complete and submit a questionnaire on steam generator operating history that was enclosed. The letter stated that the request for information was approved by GAO under a blanket clearance. Questions have been raised about the appropriateness of this request for information in light of the Federal Reports Act and about the referenced GAO blanket clearance. These questions have been discussed with representatives of GAO and it was determined that this clarifying letter should be sent to each recipient of our original letter. GAO has agreed that this request properly fits under the GAO blanket clearance for reports concerning possible generic problems and the applicable GAO clearance number should have been R0072 rather than R0071.

The request for additional information was prompted by the continuing degradation of tubes in all three vendors' steam generators. Such degradation is an important safety concern of the NRC because such tubes form part of the primary coolant pressure boundary. Several forms of degradation that have been observed in steam generators in recent months have included the wastage of tubes at Palisades and other facilities, stress corrosion at Ginna and other facilities, vibration cracking and "dinging" of tubes at the Oconee (B&W) facilities, antivibration bar fretting at San Onofre, and "denting" of tubes and associated support plate "hourglassing" and cracking at Surry, Turkey Point and about 15 other CE and W facilities. These events have resulted in many shutdowns of nuclear power stations and the safety significance of certain of these events have prompted the NRC to issue safety Orders. It is this need for important safety information that has dictated this request for additional information.

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Our previous letter acknowledged that selected portions of the information being requested may already be available to the NRC, but not in a convenient format which is readily accessible. We therefore requested that you assist us by returning a single completed copy of the enclosed questionnaire. We would like to clarify that an acceptable response to any item in the questionnaire would be to provide specific reference to any information previously submitted to the NRC, by an original response, or any combination thereof, whichever and for whatever reasons you elect to use.

Our previous letter further requested that you submit any changes or additions to your initial submittal to reflect the future operating experience with your steam generators. This would enable us to maintain the information current, which, as we stated, we will periodically publish and send copies to all participants. As we indicated, this would enable the NRC, you and others to draw from the operating experience of the entire nuclear industry on an ongoing basis when making safety and other decisions concerning steam generators in PWR plants. We are planning to prepare a submission to GAO for clearance of a request for reporting information regarding changes or additions to your initial submittal under this request.

We hope that the need for this clarification caused you no inconvenience. Because of the problems discussed above, we are extending the date for submitting the requested information to March 1, 1978.

Sincerely,

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Enclosures:
1. Letter dtd. 12/9/77
   to PWR Licensees
2. Questionnaire

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

December 9, 1977

TO ALL PWR FACILITY LICENSEES

Gentlemen:

The NRC staff has recently been engaged in a series of discussions with reactor vendors, EPRI, and the Steam Generator Owners Groups concerning steam generator operational problems. Central to these discussions is an accurate assessment of operational conditions and experiences as well as the programs aimed towards the resolution of these problems.

In order to ensure that both the NRC and the nuclear industry have available a comprehensive collection of operating data for steam generators to permit informed, timely decisions and actions, DOR is establishing a steam generator information system. The system will collect appropriate information from all PWR licensees which will periodically be published. It is presently anticipated that the initial publication of information will be in the early part of 1978. You will be sent a copy of this and all future publications.

This information system will enable the NRC and each Licensee to draw from the operational experiences of the entire nuclear industry when making any decisions concerning steam generators. This should result in both safety and economic benefits.

Enclosed is a questionnaire which we request that you complete for each of your operating PWR units. We believe that the questionnaire is self explanatory, however, if questions arise or any clarifications are necessary, please do not hesitate to contact your NRC Project Manager. Please include with your response any diagrams you may have available which illustrate the tube plugging and/or the tube degradation patterns.

To enable us to maintain the information current, you are further requested to submit in the same format indicated by the questionnaire, any changes or additions to your initial submittal to reflect the future operating experience with your steam generators, i.e., the results of future steam generator inspections. The questionnaire should be completed to the extent applicable and appropriate at this time, i.e. regardless of operating experience.

The information being requested is quite extensive and will require a diligent effort on your part and ours to assure accurate and timely completion. Also, we realize that parts of the information may already be available to the NRC, but not in a convenient format which is readily accessible. Therefore, we request that you assist us by returning a single completed copy of the enclosed questionnaire to the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, within 60 days of receipt of this letter. Please include any comments or suggestions for improving this information system which you may have.

This request for generic information was approved by GAO under a blanket clearance number R0071; this clearance expires September 30, 1978.

Sincerely,

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Enclosure:
Steam Generator
Operating History
Questionnaire

cc w/enclosure:
See next page

ENCLOSURE
STEAM GENERATOR OPERATING
HISTORY QUESTIONNAIRE

NOTE: All percentages should be reported to four significant figures.

I. BASIC PLANT INFORMATION

Plant:
Startup Date:
Utility:
Plant Location:
Thermal Power Level:
Nuclear Steam Supply System (NSSS) Supplier:
Number of Loops:
Steam Generator Supplier, Model No. and Type:
Number of Tubes Per Generator:
Tube Size and Material:

II. STEAM GENERATOR OPERATING CONDITIONS

Normal Operation

Temperature:
Flow Rate: Allowable Leakage Rate:
Primary Pressure:
Secondary Pressure:

Accidents

Design Base LOCA Max. Delta-P:
Main Steam Line Break (MSLB) Max. Delta-P:

III. STEAM GENERATOR SUPPORT PLATE INFORMATION

Material:
Design Type:
Design Code:
Dimensions:
Flow Rate:
Tube Hole Dimensions:
Flow Hole Dimensions:

IV. STEAM GENERATOR BLOWDOWN INFORMATION

Frequency of Blowdown:

Normal Blowdown Rate:

Blowdown Rate w/Condenser Leakage:

Chemical Analysis Results

Results	Parameter Control Limits

V. WATER CHEMISTRY INFORMATION

<u>Secondary Water</u>

Type of Treatment and Effective Full Power (EFP) Months of Operation:

Typical Chemistry or Impurity Limits:

<u>Feedwater</u>

Typical Chemistry or Impurity Limits:

<u>Condenser Cooling Water</u>

Typical Chemistry or Impurity Limits:

Demineralizers - Type:

Cooling Tower (open cycle, closed cycle or none):

VI. TURBINE STOP VALVE TESTING (applicable to Babcock & Wilcox (B&W) S.G. only)

Frequency of Testing

Actual:

Manufacturer Recommendation:

Power Level At Which Testing Is Conducted

Actual:

Manufacturer Recommendation:

Testing Procedures (Stroke length, stroke rate, etc.)

Actual:

Manufacturer Recommendation:

VII. STEAM GENERATOR TUBE DEGRADATION HISTORY

(The following is to be repeated for each scheduled ISI)

Inservice Inspection (ISI) Date:

Number of EFP Days of Operation Since Last Inspection:

(The following is to be repeated for each steam generator)

Steam Generator Number:

Percentage of Tubes Inspected At This ISI:

Percentage of Tubes Inspected At This ISI That Had Been Inspected At The Previous Scheduled ISI:

Percentage of Tubes Plugged Prior to This ISI:

Percentage of Tubes Plugged At This ISI:

Percentage of Tubes Plugged That Did Not Exceed Degradation Limits:

Percentage of Tubes Plugged As A Result of Exceedance of Degradation Limits:

Sludge Layer Material Chemical Analysis Results:

Sludge Lancing (date):

Ave. Height of Sludge Before Lancing:

Ave. Height of Sludge After Lancing:

Replacement, Retubing or Other Remedial Action Considered: (Briefly Specify Details)

Support Plate Hourglassing:

Support Plate Islanding:

Tube Metalurgical Exam Results:

Fretting or Vibration in U-Bend Area (not applicable to B&W S.G.) AS OF (4)

Percentage of Tubes Plugged	Other Preventive Measures

Wastage/Cavitation Erosion AS OF (4)

Hot Leg: (Repeat this information for the cold leg on Combustion Engineering (C.E.) and Westinghouse (W) S.G.)

Area of Tube Bundle (1)	a	b	c	d	e
% of Tubes Affected by Wastage/Cavitation Erosion					
% of Tubes Plugged Due to Exceedance of Allowable Limit (2)					
% of Tubes Plugged That Did not Exceed Degradation Limit					
Location Above Tube Sheet (3)					
Max. Wastage/Cavitation Erosion Rate for Any Single Tube (Tube Circum. Ave) (Mills/Month)					
Max. Wastage/Cavitation Erosion in Any Single Unplugged Tube (Tube Circum. Ave) (Mills)					

Cracking AS OF (4)

Caustic Stress Corrosion Induced in C.E. and W S.G.

Flow Induced Vibration Caused in B&W S.G.

Cracking (Con't)

Hot Leg: (Repeat this information for the cold leg on C.E. and W S.G.)

Area of Tube Bundle (1)	a	b	c	d	e
% of Tubes Affected By Cracking					
% of Tubes Plugged Due to Cracking					
% of Tubes Plugged That Did Not Exceed Degradation Limit					
Location Above Tube Sheet (3)					
Rate of Leakage From Leaking Cracks (gpm)					

Denting (Not applicable to B&W S.G.) AS OF (4)

Hot Leg: (Repeat this information for the cold leg on C.E. and W S.G.)

Area of Tube Bundle (1)	a	b	c	d	e
% of Tubes Affected by Denting					
% of Tubes Plugged Due to Exceedance of Allowable Limit (2)					
% of Tubes Plugged That Did Not Exceed Degradation Limit					
Rate of Leakage From Leaking Dents (gpm)					
Max. Denting Rate for Any Single Tube (Tube Circum. Ave) (Mills/Month)					
Max. Denting in Any Single Unplugged Tube (Tube Circum. Ave) (Mills)					

Denting (Con't)

Support Plate Levels	Max. Denting in Any Single Tube in Bundle Area (Tube Ave) (Mills) (1)					% of Tubes Affected By Denting in Bundle Area				
	a	b	c	d	e	a	b	c	d	e
1										
2										
3										
4										
5										
6										
7										
8										
9										
10										
11										
12										

TABLE KEY

NOTE: All percentages refer to the percent of the tubes within a given area of the tube bundle.

(1)

Area of the Tube Bundle	No. of Tubes Within the Area
a. Periphery of Bundle (wi/20rows for B&W; wi/10 rows for C.E. and W)	
b. Patch Plate (wi/4 rows)	
c. Missing Tube Lane (B&W only) (wi/5 rows)	
c. Flow Slot Areas (C.E. and W only) wi/10 rows)	
d. Wedge Regions (C.E. and W only) (wi/8 rows)	
e. Interior of Bundle (remainder of tubes)	

(2)

Allowable Limit for Wastage/Cavitation Erosion:

Allowable Limit For Denting:

(3)

1. Specifies area between the tube sheet and the first support plate
2. Specifies in the following locations: (list the additional locations)

   Wastage/Cavitation Erosion:

   Cracking:

(4)

Specify the date of the inspection for which results have been tabulated.

VIII. SIGNIFICANT STEAM GENERATOR ABNORMAL OPERATIONAL EVENTS

DATE	SUMMARY
	(Include event description; unscheduled ISI results, if performed; and subsequent remedial actions)

IX. CONDENSER INFORMATION

Condenser Material	Tube Leakage Date	Tube Leakage Rate (gpm)	Detectable Limit	Detection Method

X. RADIATION EXPOSURE HISTORY WITH RESPECT TO STEAM GENERATORS

Date	Exam Dosage (Man-Rem)	Repair Dosage (Man-Rem)	Comments

XI. DEGRADATION HISTORY FOR EACH TYPE OF DEGRADATION EXPERIENCED FOR TEN REPRESENTATIVE, UNPLUGGED TUBES FOR WHICH THE RESULTS OF TWO OR MORE ISI'S ARE AVAILABLE

If the results for ten tubes are not available, specify this information for all those tubes for which results are available.

(repeat the following information for each tube and degradation type)

Steam Generator No:
Tube Identification:
Type of Degradation: (specify denting, wastage, cavitation erosion, caustic stress corrosion cracking, or flow induced vibration cracking)

(repeat the following information chronologically for each ISI for which results are available)

ISI Date:
Amount of Degradation: (specify amount and units)
EFP Months of Operation Since Last ISI for Which Results are Given:

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

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New York, New York 10019

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

Docket file

February 2, 1978

50-247/286

All PWR Licensees (except for Trojan)

F 122

Gentlemen:

During the course of responding to the staff's review of an application for license amendment on the Trojan Nuclear Plant, the licensee informed the NRC that the reactor cavity annulus seal ring (used as a water seal during refueling operations, and not removed during normal operations) and associated biological shielding over the reactor vessel cavity could become missiles in the event of a loss of coolant accident (LOCA) pipe break inside the reactor vessel cavity. At the Trojan Nuclear Plant, these missiles could affect the ability of the control rods to shut down the reactor. From our preliminary evaluation of the information provided to the NRC staff by the licensee, the Portland General Electric Company and by Westinghouse, Combustion Engineering, Babcock & Wilcox and Bechtel in telephone discussions on January 25 and 26, 1978, it appears that this problem could occur in other PWR facilities such as yours and could potentially pose a threat to the health and safety of the public in the event of a LOCA.

Therefore, pursuant to 10 CFR 50.54(f) of the Commission's regulations, you are hereby requested to deliver to the Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, Washington, DC 20555, within 20 days of the date of this letter, i.e., February 22, 1978, the following information: (a) a statement as to whether the cavity annulus seal ring in your facility is left in place during normal operation or if biological shielding is installed in the reactor cavity annulus and, if the answer to (a) is yes; (b) when you will determine whether the cavity annulus seal ring or biological shielding could become a missile in your facility, and (c) a description of what you plan to do, and when, if the problem is found at your facility and (d) justification for continued operations until the problem has been resolved, such justification to support why continued operation will not create undue risk to the health and safety of the public.

A copy of this letter is being provided to each licensee's current service list.

Sincerely,

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

mail

William J. Cahill, Jr.
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

REGULATORY DOCKET FILE COPY

January 13, 1978

RECEIVED
JAN 17 1978
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Re: Indian Point Units 2 &
Docket Nos. 50-247
and 50-286

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Reid:

Your letter of September 2, 1977 requested information concerning our experience with waterhammers resulting from the rapid condensation of steam in the feedwater lines of the steam generators. Following the waterhammer incident which occurred at Indian Point Unit No. 2 on November 13, 1973, a detailed program was begun by Con Edison and Westinghouse to determine the causes of this incident and to find means of preventing a recurrence. This program is described in detail in our submittals to the Commission of January 14, 1974, March 12, 1974 and August 30, 1974. As a result of this program, we installed "J tubes" on the feedwater spargers in the Indian Point Unit 2 and 3 Steam Generators. This modification, which is discussed in our submittals of March 12, 1974 and August 30, 1974, effectively prevents the rapid draining of the feedwater spargers and lines. Since the installation of the " J tubes", no further waterhammers have been experienced in the feedwater systems of Indian Point Unit 2 or 3.

Your September 2, 1977 letter also requested that any future damaging waterhammer event at Indian Point Unit 2 or 3 be reported to the Commission using the form that was attached to your letter. Should such an event occur, we will comply with your request.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

cc: Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State of New York
10 Columbus Circle
New York, N. Y. 10019

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