Docket
50-3

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

February 10, 1978

GL-78-4

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.

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

Consolidated Edison Company of
New York, Inc.

cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

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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

Secondary Water

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

Typical Chemistry or Impurity Limits:

Feedwater

Typical Chemistry or Impurity Limits:

Condenser Cooling Water

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	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: