5473

Dated: January 26, 1988

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

before the

ATOMIC SAFETY AND LICENSING BOARD

In the matter of PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE, et al. (Seabrook Station, Units 1 and 2)	Docket Nos. 50-443-OL-1 50-444-OL-1 (Onsite Emergency Planning and Safety Issues)

APPLICANTS' RESPONSES TO NEW ENGLAND
COALITION ON NUCLEAR POLLUTION'S
SECOND SET OF INTERROGATORIES AND REQUEST
FOR PRODUCTION OF DOCUMENTS TO APPLICANTS
ON NECNP CONTENTION I.V.

INTERROGATORY NO. 1

Please identify all persons who participated in the preparation of answers to these interrogatories, and identify the portions of your response to which each person contributed.

RESPONSE NO. 1

See Attachment 1-1.

INTERROGATORY NO. 2

Identify and produce all documents referring or relating to preservice or subsequent testing of the steam generator

DS03

tubes by means of nondestructive examination by eddy current testing, including but not limited to the interpretations and evaluations of the results of such tests.

RESPONSE NO. 2

See documents identified in Attachment 2-1. These documents will be made available for inspection at the Seabrook Station site. Please contact Mr. William J. Daley, Jr. at (603) 474-9521, extension 2057 to arrange for inspection. It should be noted that at present there is no facility at Seabrook Station for reading the Conam Eddy Current Data recorded on magnetic tape.

INTERROGATORY NO. 3

With respect to the nondestructive examination by eddy current testing which is to be conducted at the Seabrook plant:

a) Identify and describe the estimated probability of a false positive error of such testing.

b) Identify and describe the estimated probability of a false negative error of such testing.

c) Produce all documents referring or relating to the responses to Interrogatory Questions 3a) and 3b).

RESPONSE NO. 3

The Applicant is not aware of any methodology utilized in the industry to estimate these probabilities.

INTERROGATORY NO. 4

Describe any and all diagnostic procedures which are to be used to assess whether primary to secondary side leakage exceeds the Tech Spec limit at the Seabrook plant.

RESPONSE NO. 4

Primary to secondary leak rate determinations are done in accordance with Chemistry Procedure CX 0920.05, "Primary to Secondary Leak Rate Calculations". See Attachment 4-1.

The method is based on tritium levels in the primary and secondary coolant. The change in tritium levels of the secondary coolant over time is correlated to the primary coolant tritium levels to derive the primary to secondary leak rate.

INTERROGATORY NO. 5

For each diagnostic procedure identified in the response to Interrogatory Question 4):

a) Describe its reliability.

b) Describe how its reliability is defined.

c) Describe how its reliability was established.

d) Describe what the estimated rate of false positive errors is for the diagnostic procedure.

e) Describe what the estimated rate of false negative errors is for the diagnostic procedure.

f) Describe any evidence both for and against the proposition that the leak rate test combined with the Tech Spec limit on leak rate predicts subsequent tube bursts.

g) Produce all documents referring or relating to the information requested in Interrogatory Questions 5a) through 5g).

RESPONSE NO. 5

In responding to Interrogatory 5 we assumed that NECNP was referring to the sensitivity of the methodology of the diagnostic procedure. The definition of reliability, as provided by NECNP in these interrogatories, cannot be applied to determine the effectiveness of the procedure to assess leakage because the purpose of the diagnostic procedure described in Response to Interrogatory 4 is not to identify specific tubes which may be leaking.

a) The sensitivity of the primary to secondary leak rate determination over a 24-hour measurement period is 2.3 gallons per day.

b) See Response to 5(c).

c) The sensitivity of the primary to secondary leak rate determination was established by the following calculation:

$$L = \frac{A_S}{A_P}\frac{V}{t} = \frac{(1 \times 10^{-5}\ \text{uc/ml})\ (228{,}000\ \text{gallons})}{(1.0\ \text{uc/ml})\ \ \ (1\ \text{day})}$$

$= 2.3$ gallons per day

A_S - tritium activity in secondary coolant at lower limit of detection 1×10^{-5} uc/ml

Ap - tritium activity in primary coolant 1.0 uc/ml from FSAR Table 11.1-1

V - secondary coolant volume 228,000 gallons from Table 11.1-3

t - 1 day

d) As noted above, the purpose of the diagnostic procedure is not to identify specific tubes which may be leaking. Therefore, it is not possible to identify a rate of false positive error. Rather, as provided in the basis to the Steam Generator Tube Inspection Technical Specification, upon exceeding 500 gallons per day, the plant would be shutdown and an unscheduled inspection would be performed to locate and plug the leaking tube.

e) See Response to Interrogatory 5(d).

f) The maximum permissible leakage during normal operation and Technical Specification limit is established by the NRC at a level that assures continued safe operation without the potential for tube rupture under postulated feedwater line and steam line break accident conditions. This limit corresponds to a crack length determined using the correlation of leak rate test data and crack length as described in Reference 1 to Response 5(g). Leakage from cracks greater in length than the crack corresponding to the Technical Specification limit would exceed that limit and plant shutdown for corrective action would be required.

The crack length for which the maximum postulated faulted condition pressure differential will cause a crack to grow rapidly is referred to as the critical crack length. It is determined using the materials property data for the Inconel 600 steam generator tubing material and the correlation of burst pressure and crack length test data as described in Reference 1 to Response 5(g).

For Seabrook, as discussed in Reference 1 to Response 5(g), the critical crack length is greater than the crack length corresponding to the Technical Specification limit. As a consequence, in the event that a through-wall crack were to develop, primary-to-secondary leakage would occur which is readily detectable with the leak rate test, i.e., by monitoring primary-to-secondary leakage, and corrective action would be taken before the crack could grow to the critical crack length beyond which rupture would occur.

The fact that only five tube burst events have occurred in domestic PWRs in the hundreds of thousands of steam generator tubes in operation demonstrates that the leak rate test in combination with the Technical Specification limit has been used successfully to help maintain steam generator tube integrity.

The record of the five tube burst events in domestic PWRs is an argument against the proposition that the leak rate test in combination with the Technical Specification limit on leak rate does not predict subsequent tube bursts. These events, their causes, and implications for the Seabrook Station are discussed in the affidavits of John N. Esposito filed with Applicants' Memorandum In Support Of Low Power Operation (January 4, 1988) (Attachments 5-1 and 5-2).

g) Villasor, A.P., Jr., "Steam Generator Tube Plugging Margin Analysis For The Seabrook Nos. 1 & 2 Nuclear Power Plants". WCAP 10413, Westinghouse Nuclear Energy Systems. Pittsburgh, Pa., November 1983.

Applicants object to producing this document on the grounds that it contains information proprietary to Westinghouse Electric Corporation. It is not to be reproduced, transmitted, disclosed, or used otherwise, in whole or in part, without prior authorization from Westinghouse. This document will be produced to NECNP upon NECNP's signature to an appropriate protective order.

INTERROGATORY NO. 6

If no diagnostic procedure for assessing whether primary to secondary side leakage exceeds the Tech Spec limits was identified in the response to Interrogatory Question 4, explain why no such diagnostic procedure exists.

RESPONSE NO. 6

See response to Interrogatory No. 4.

INTERROGATORY NO. 7

Identify what portion of the FSAR describes the composition of the Seabrook plant's water chemistry.

RESPONSE NO. 7

Information on Seabrook's plant water chemistry can be found in the FSAR Chapter 9.3, 10.3, 10.4, and in RAI Responses 282.2 and 282.3.

INTERROGATORY NO. 8

State whether the applicants have evaluated whether there are other nuclear power plants with similar water chemistry to that of the Seabrook plan which have indicated critical areas for possible leaking tubes. Describe applicants' evaluation.

RESPONSE NO. 8

Yes. Applicants maintain close contact with Westinghouse and EPRI's Steam Generator Reliability Project and therefore have access to up-to-date information on operating experience in Model F steam generators. That experience was summarized in the Affidavit of John N. Esposito On The Ginna Tube Rupture Event And The Design Of And Experience With Domestic Model F Steam Generators filed with Applicants' Memorandum In Support Of Low Power Operation (January 4, 1988) (Attachment 5-1). Applicants have evaluated that experience and concluded that it is such that an In-Service Inspection Program consistent with the Standard Technical Specifications of NUREG-0452 will be appropriate for complying with the recommendations of Regulatory Guide 1.83.

As to Answers:

Ted C Feigenbaum

Ted C. Feigenbaum, Vice President
New Hampshire Yankee Division of
Public Service Company of New
Hampshire

State of New Hampshire
Rockingham County, ss.

Then appeared before me the above subscribed Ted C. Feigenbaum and made oath that he is the Vice President of New Hampshire Yankee Division, authorized to execute the foregoing responses to interrogatories on behalf of the Applicants, that he made inquiry and believes that the foregoing answers accurately set forth such information as is available to the Applicants.

Before me,

Beverly E. Silloway

Beverly E. Silloway , Notary Public
My Commission Expires: March 6, 1990

As to objections:

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ATTACHMENT 1-1

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ATTACHMENT 2-1

IDENTIFICATION NO.	TITLE/DESCRIPTION/SUBJECT
EX 1807.002	"Steam Generator Eddy Current Inspection", NHY Procedure
	Conam "PSI & ISI Program Plans"
	Conam Eddy Current Exam April & May 1985, Final Report
	Conam Eddy Current Exam July 1985, Final Report
	Conam Eddy Current Data - Magnetic Tapes
SBY-2	PSNH-NES Preservice Inspection Contract
SBY-2, EWS-7	PSNH-NES Contract Amendment, re: Eddy Current PSI
SBY-2, EWA-9	PSNH-NES Contract Amendment, re: Eddy Current re-exam
MRS 4.4 NAH-3	Westinghouse Field Service Report, "Tube to Tubesheet Expansion and Mechanical Plugging"
	YAEC QA Surveillance Reports
	Conam Letter dated 07/02/85, "Eddy Current Final Report"
	Conam Letter dated 07/09/85, "Eddy Current Procedure Transmittal"
5562-169	NES Letter dated 02/04/85, "Identification of Conam as Contractor"
5562-039	NES Letter dated 03/22/85, "Submittal of SG PSI Program Plan"
5562-045	NES Letter dated 04/21/85, "SG Schedule of PSI Exam Activity"

IDENTIFICATION NO.	TITLE/DESCRIPTION/SUBJECT
5562-180	NES Letter dated 04/17/85, "Contract Change Proposal 131"
5562-211	NES Letter dated 11/13/85, "EWA for July 1985 Eddy Current"
SB-15217	PSI Service Contract SBY-2, Cover Letter
SB-18445	Comments on SG PSI Program Plan
SB-19137	Eddy Current Procedure Acceptance
SB-20077	Conam Inspection Report
MSG-408/85	Cover Letter, Eddy Current Final Report
MSM 39/85	"Eddy Current Report SBS Steam Generators"
STD 85-099	SG Wet Layup Preps
STDINT 85-015	Arkwright Boston Insurance - PSI Contract
NAH 3246	Model "F" SG AVB Performance

STATION OPERATING PROCEDURE COVER FORM

A. IDENTIFICATION

NUMBER CKC920.05 REVISION 00

TITLE PRIMARY TO SECONDARY LEAK RATE CALCULATION

ORIGINATOR L. D. Rabideau

B. TECHNICAL REVIEW

TITLE	SIGNATURE	DATE
Working Foreman	[signature]	2/1/86
Chem Supv	[signature]	3/3/86

C. DEPARTMENT SUPVR./MGR. APPROVAL

TITLE	SIGNATURE	DATE
Chem Dept Supv	W. R. [signature]	3/4/86

D. QUALITY ASSURANCE REVIEW

TITLE	SIGNATURE	DATE
LQCI Technical	[signature]	5MAR86

E. SORC APPROVAL SORC MEETING NO. 86-14

F. APPROVAL AND IMPLEMENTATION

[signature]
STATION MANAGER 3/28/86 APPROVED DATE 4-4-86 EFFECTIVE DATE

PRIMARY TO SECONDARY LEAK RATE CALCULATION

Page No.	Revision
1	00
2	00
3	00
4	00
5	00
6	00
7	00

Form No.	Revision

1.0 OBJECTIVE

This procedure describes the method for determination of primary to secondary leak rates.

2.0 REFERENCES

2.1 Babcock and Wilcox Radiochemistry Manual, BAW-140, Rev. 1, April 1980

2.2 Babcock and Wilcox Radiochemistry Training Course, TRG-7815, August 1978

3.0 DISCUSSION

If a steam generator tube develops a leak, the primary coolant will be forced into the secondary system. This is of concern because the secondary system is not designed to handle radioactivity and, in addition, the radioactive releases to the environment are more difficult to control. Corrective actions will differ depending upon the severity of the leak. This procedure is designed to estimate the primary to secondary leak rate. Figure 10.1 shows leak rate sensitivity for tritium as a function of primary activity and time from leak.

4.0 ACCEPTANCE CRITERIA

4.1 Steam generater activity levels significantly above background ($>2\sigma$) indicates a possible leak.

4.2 The use of tritium as a leak monitor is viable when tritium, in the primary, is >1 E-2 μCi/ml, and time allows for tritium to reach a detectable level in the secondary system.

5.0 PRECAUTIONS

Not applicable to this procedure.

6.0 PREREQUISITES

6.1 Procedures

6.1.1 CX0920.07, Tritium Analysis by Liquid Scintillation Using the LS-1800

6.1.2 CS0911.01, Feedwater Sampling - CP148

6.1.3 CX0910.01, NSSS-CP-166, SS-CP-166 Sampling

6.1.4 CX0930.01, Gamma Spectroscopy System Operation and Calibration

6.1.5 CX0910.06, Routine Steam Generator Blowdown Recovery Sampling

6.1.6 CX0920.01, Determine of Reactor Coolant Radiogas

6.2 Special Equipment

Not applicable to this procedure.

6.3 Reagents

Not applicable to this procedure.

7.0 INITIAL CONDITIONS

Not applicable to this procedure.

8.0 PROCEDURE

8.1 Instantaneous Leak Rate Using Tritium Activity

8.1.1 Obtain a condensate pump discharge sample per reference 6.1.2.

8.1.2 Obtain a reactor coolant sample at approximately the same time as the condensate sample per 6.1.3.

8.1.3 Analyze the condensate and reactor coolant samples for tritium per reference 6.1.1.

8.1.4 If the time of the start of leakage is known, calculate the leak rate using the following equation.

$$L = \frac{A_s V}{A_p t}$$

L = Leak rate, gallons/minute.
A_s = Tritium activity in the secondary, µCi/ml.
V = Volume of secondary, gallons.
A_p = Tritium activity in the primary, µCi/ml.
t = Duration of leak, minutes.

8.1.5 If the time of the start of leakage is unknown, the leak rate can be calculated from the increase in count rate between two intervals. Two secondary samples will be required at different times (normally a 12 hour wait time is sufficient). Calculate the leak rate from the following equation.

$$L = \frac{V\,(As_2 - As_1)}{A_p\,(t_2 - t_1)}$$

L = Leak rate, gallons/minute.
As_2 = Tritium activity in the secondary at time 2, µCi/ml
As_1 = Tritium activity in the secondary at time 1, µCi/ml
Vs = Volume of secondary, gallons.
Ap = Tritium activity in the primary, µCi/ml.
$t_2 - t_1$ = elapsed time between secondary samples, minutes.

8.2 Instantaneous Leak Rate Using Radioactive Gaseous Activity

8.2.1 Collect a condenser air removal pump effluent sample by connecting a 1 liter gas marinelli to valve AR-V-150 in the air removal combined header, 53 foot turbine building, condenser side of the radiation monitor.

8.2.2 Determine the air removal flow rate (SCFM) by summing the flow from the meters on the 3 air removal pumps.

8.2.3 Obtain a reactor coolant gaseous sample at approximately the same time per reference 6.1.6.

8.2.4 Analyze the air removal effluent and reactor coolant for Xe^{133} per reference 6.1.4.

8.2.5 Calculate the leak rate from the following equation:

$$L = \frac{(A_A)(F)(2.83\times10^4)}{(A_P)(3.78\times10^3)}$$

L = leak rate, gpm
F = air removal flow rate, SCFM
A_A = Xe^{133} activity, µCi/ml, in the air removal header
A_P = Xe^{133} activity, µCi/ml, in the primary
2.83×10^4 = conversion from SCF to ml
3.78×10^3 = conversion from ml to gallons

8.3 Instantaneous Leak Rate Using Particulate Activity

8.3.1 Obtain a steam generator blowdown sample on the appropriate steam generator per reference 6.1.5.

8.3.2 Obtain a reactor coolant sample at approximately the same time as the condensate sample per reference 6.1.3.

8.3.3 Perform a gamma analysis on the samples per reference 6.1.4.

8.3.4 Use a non-volatile nuclide of high concentration in the primary for determination of leak rate. Normally sodium 24 is adequate.

8.3.5 Calculate the leak rate from the following equation:

$$L = \frac{AsV\left(\lambda + \frac{R}{V}\right)}{Ap}$$

L = Leak rate, gallons/minute
As = Activity of selected nuclide in steam generator
V = Volume of the steam generator
$\lambda = \dfrac{0.693}{T1/2(\text{minutes})}$
R = Blowdown flowrate for the generator sampled in gpm.
Ap = Activity of selected nuclide in the reactor coolant in μ Ci/ml.

> NOTE
>
> THE ABOVE CALCULATION IS FOR EQUILIBRIUM CONDITIONS. FOR SHORTLIVED NUCLIDES SUCH AS Na^{24} THIS WILL OCCUR SEVERAL HOURS AFTER THE LEAK HAS OCCURRED.

8.3.6 If the leak has occurred less than several hours ago, calculate the leak rate as follows:

$$L = \frac{AsV\left(\lambda + \frac{R}{V}\right)}{Ap\left(1 - e^{-\left[\left(\lambda + \frac{R}{V}\right)t\right]}\right)}$$

L = Leak rate, gallons/minute
As = Activity of selected nuclide in the steam generator
V = Volume of the steam generator
$\lambda = \dfrac{0.693}{T1/2(\text{minutes})}$
R = Blowdown flowrate for the steam generator sampled in gpm
Ap = Activity of selected nuclide in the reactor coolant in μ Ci/ml
t = Time since start of leak occurred in minutes

9.0 FINAL CONDITIONS

Not applicable to this procedure.

10.0 FIGURES

10.1 Leak rate sensitivity for tritium.

11.0 FORMS

Not applicable to this procedure.

FIGURE 10.1

LEAK RATE SENSITIVITY FOR TRITIUM