

# ARGONNE NATIONAL LABORATORY

9700 SOUTH CASS AVENUE, ARGONNE, ILLINOIS 60439

Telephone: 312/972-6144

June 13, 1988

Mr. E. Murphy  
Office of Nuclear Reactor Regulation  
U. S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Dear Mr. Murphy:

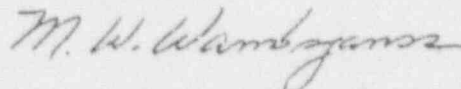
Subject: Westinghouse Method for Evaluation of Steam Generator Tube  
Vibration-Induced Fatigue

Argonne National Laboratory has been providing consultation to the U.S. NRC in the review of the North Anna Unit 1 steam generator tube failure assessment and the evaluation methodology developed by Westinghouse. The current status of the Westinghouse method is given in WCAP-11799 which documents the application of the method to evaluate the steam generator tubes of Beaver Valley Unit 1 nuclear station.

The review took place over a period of several months (September 1987 to date) that included the time period during which the Westinghouse method was under development. Among other things, as part of the review Argonne called attention to a number of uncertainties related to the inherent complexities of the physical situation and to associated inadequacies in state-of-the-art modeling and analysis techniques.

Upon review of the recent assessment of the Beaver Valley Unit 1 steam generator tubing (WCAP-11799), it is concluded (based on documentation, a laboratory visit, and correspondence) that Westinghouse has satisfactorily identified and appropriately addressed the major uncertainties associated with the test and analysis program; as an example, See Sections 8.2 - 8.7 of WCAP-11799. As a result, it is further concluded that the Westinghouse method as presently developed represents a sound engineering approach that, if properly implemented, will provide reasonable assurance that future steam generator tube failures of the type experienced at North Anna Unit 1 are not likely to occur.

Sincerely,

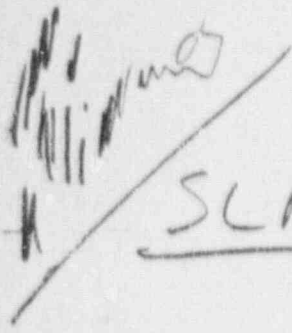


M. W. Wambsganss  
Materials and Components Technology Div.

MWW:cnl

9201280193 910621  
PDR FOIA  
WILLIAM91-106 PDR

C/19



SLB

Row 9

Row 8



Assian Line

100 gallons/day

$$\frac{\theta}{\Sigma} = 2.3$$

SLB  $\rightarrow$  225  $\Delta p$  early in

$\delta = .4$  in 6th direction

6.9  $\rightarrow$  24  $\rightarrow$  leak increases to  
325 g/d

12.4  $\rightarrow$  26  $\rightarrow$  355 g/d

SLB  $\rightarrow$  2600

$\frac{\theta}{\Sigma}$  150  $\rightarrow$  tube burst for row 9  
critical tube

... to ...  
...  
C/20

R4-C8 (2H) 720 DI Volt = 1.61  
 ← Remedy of May 87 date  
 RPC DI .7 v

R4-C8 3H (8x1) PI 7.11 v (on coil response) plugged  
 RPC PE 2.7 v May 1987

R37-C25 1H 720 NDB 5.28 v (on coil)  
 8x1 PI 2.08 v

R11-C45 1H 8x1 PE .89 (on coil)  
 RPC PE .92

R46 C47 3H+0 8x1 PE .66 (on coil)  
 +.2 RPC PE .55

4H+0 8x1 PE 5.20 (on coil)  
 +.2 RPC PE .25

R11-C51 1H + 1.22/21 720 DE 2.51 v May 1987 remount  
 - 1.01/21 720 DE 3.57 v May 1987 remount  
 + .72 8x1 PE 12.54 v (on coil)  
 +.2 RPC DI 2.92 v

R16-C71 2H -.77/21 720 DI 2.53 v  
 8x1 NDB  
 RPC DI 2.32 v

holy shit?!?

R12 - C27 TSH + 0 8x1 PE 37.23 v (8 coils)  
RPC PE 6.99 v

R22 - C31 TSH + 0 720 NDD  
8x1 PE 5.94 v (one coil)  
RPC PE 2.43 v

~~R5 - C32~~

R9 - C32 TSH + .8 720 TI 2.00 v  
8x1 NDD  
RPC TI .86 v

R21 - C32 TSH + 0 8x1 PE 18.9 v (3 coil response)  
RPC PE 8.27 v

R19 - C34 TSH + 0 720 NDD (~~27.17~~)  
8x1 PE 7.50 v (1 coil)  
RPC PE 2.92 v

R13 - C36 TSH + 1.4 720 TI 1.72 v  
+ 0 8x1 PE 5.92 v (eight coils)  
+ 1.2 RPC TI 1.34 v  
+ 1.0 RPC TE .96 v

R13 - C40 TSH + .89 720 TI 2.49 v  
+ 1.0 8x1 PE 2.28 v (eight coils)  
+ 1.5 RPC TI .6 v  
+ 1.7 RPC TI 71 v

R23 - C43 TSH + 6.75 720 22% 2.12 v (100% coils)  
8x1 NDD

R 12 - C41 TSH + 1.53 720 TI\* 2.61 v

8x1 NDD

+ 1.5

RPC TI 5.15 v

R 11 - C43 TSH + .82 720 TI .89 v

8x1 NDD

+ 1.1

RPC TZ 1.61 v

R 12 - C43 TSH + 1.56 720 TI 2.95 v

8x1 NDD

+ 1.4

RPC TI 1.65 v

R 13 - C43 TSH + .9 720 TI 2.91 v

8x1 NDD

RPC TI .57 v

R 13 - C44 TSH + 1.32 720 TI 1.64 v

+ .85

720 TI 2.57 v

8x1 NDD

+ 1.2

RPC TI 1.2 v

R 14 - C44 TSH + .72 720 TI

8x1 NDD

+ .3

RPC TI

R 13 - C46 TSH + 1.23 720 TI

+ .85

720 TI

8x1 NDD

+ .6

RPC TI

~~R 12 - C44 TSH~~

R 13 - C47 TSH + 57 720 TI

+ 1.12 720 TI

8x1 NDD

+ .7 RPC TI

R 14 - C47 TSH + .76 720 TI

8x1 NDD

+ .5 RPC TJ

R 24 - C47 TSH + 6 8x1 PI (5 coils)

+ 0 RPC PI

R 32 - C47 TSH + 1.08 720 TI

+ 0 8x1 PI

+ 0 RPC PI

$\sqrt{2} \text{ C.C.S.}$  (Remains 7 May 87 date)

R 13 - C47 TSH + .88 720 TI

+ .55 720 TJ

8x1 NDD

+ .6 RPC TI

Remains 7 May 87 date

R 13 - C49 TI/PL/TI

R 31 - C49 TI/NDD/?

R 31 - C53 <20% @ 6.2" ATS with ~~8~~ 720 (normal)

R 21 - C57 <20% @ 4.52" ATS " " "

R 31 - C67 <20% @ 2.77" ATS " " "



R9-C32 7C + .80/3 720 DE 3.76V (AVB visible)

8x1 NDD

T.7/0 RPC DE .86V

R9-C51 Ruptured N6

R10-C51 7C +  $\odot$  720 37% .82V (normal indic.)

R19 - C23 1C + 17.45 720 23% .52 volts ✓  
8x1 NDD

R34 - C24 2C 710 420% 7.5V volts ✓  
8x1 NDD

R39 - C26 6C 720 NDD ✓  
8x1 PV V = 153.7 V ✓  
APC NDD

R42 - C46 1C + 30.43 720 26% V = .73 V  
8x1 NDD



R 9 - C 27	TSC + .8	720	TI	1.04 v
		RPL	55%	.52 v

R 4 - C 28	TSC + .5	720	TI	2.55 v
		8x1	PL	5.88 v (one coil)
		RPL	TI	1.53 v

R 4 - C 29	TSC + .10	720	TI	1.96 v
		8x1	PL	3.00 v (one coil)
		RPL	81%	1.26 v

R 15 - C 30	TSC + .5	720	TI	.85 v
		8x1	NDD	
		RPL	51%	.84 v

R 5 - C 31	TSC + .6	720	TI	1.98 v
		8x1	PL	2.71 v (one coil)
		RPL	50%	.85 v

R 5 - C 32	TSC + .16	720	TI	2.07 v	
		+ 1.17	8x1	PL	1.86 v (five coils) !!
		+ 1.3	RPL	59%	.75 v

R 20 - C 36	TSC + .9	720	TI	1.64 v
		8x1	NDD	
		RPL	77%	.88 v

J/R/A ASSOCIATES  
Regulatory information & Support Systems  
1407 Marco Drive  
Mitchellville, MD 20721  
301/249-9672

March 11, 1991  
OGW-062

Mr. Donnie H. Grimsley, Director  
Division of Freedom of Information and  
Publications Services  
U.S. Nuclear Regulatory Commission  
Washington, DC 20555

FREEDOM OF INFORMATION  
ACT REQUEST

FOIA-91-106  
Rec'd 3-15-91

SUBJECT: FREEDOM OF INFORMATION ACT REQUEST

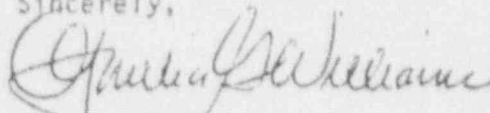
Dear Mr. Grimsley:

Pursuant to the Freedom of Information Act, 5 U.S.C. §552 and NRC's regulations, I am herein requesting that all documents, not previously released to the public via the Public Document Room, re the North Anna Unit 1 and Ginna steam generator tube rupture events, be placed in the Public Document Room.

I would appreciate your prompt response within ten working days of the receipt of this letter, as provided by the Code. I will be responsible for costs associated with processing this request, however, please provide an estimate if the fees exceed \$50.00. I can be reached at 301/249-9672, if there are questions.

Thank you.

Sincerely,



Ophelia G. Williams

~~9149190072~~



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20545

APR 4 1981

MEMORANDUM FOR: Thomas H. Novak, Assistant Director  
for Operating Reactors  
Division of Licensing

FROM: V. S. Noonan, Assistant Director  
for Materials & Qualifications Engineering  
Division of Engineering

SUBJECT: GINNA STATION STEAM GENERATOR TUBES INSERVICE INSPECTION  
PROGRAM FOR THE 1980-1989 INTERVAL (TAC #43196)

Plant Name: R. E. Ginna Unit No. 1  
Suppliers: Westinghouse; Gilbert Associates  
Docket Number: 50-244  
Responsible Branch and Project Manager: ORB#5; R. P. Snaider  
Reviewer: D. T. Huang  
Description of Task: Review of Ginna Station's Steam Generator Tubes  
Inservice Inspection Program for the 1980-1989 Interval  
Review Status: Additional Information Needed

The Inservice Inspection Section of the Materials Engineering Branch, Division of Engineering has reviewed that portion of Rochester Gas and Electric Corporation's submittal dated November 6, 1980 regarding the Ginna Station Steam Generator Tubes Inservice Inspection Program for the 1980-1989 interval. We conclude on the basis of our review that the Ginna Station Steam Generator Tube Inservice Inspection Program for the 1980-1989 Interval is acceptable if the following two conditions are met:

- 1) Plugging limit for the ten test sleeves be established.
  - 2) Two typographical errors mentioned in our Safety Evaluation be corrected.
- Our Safety Evaluation is attached.

Vincent S. Noonan, Assistant Director  
Materials & Qualifications Engineering  
Division of Engineering

Contact: D. T. Huang  
X27377

cc: R. H. Vollmer  
D. G. Eisenhut  
R. A. Purple  
V. S. Noonan  
T. H. Novak  
S. S. Pawlicki  
V. Benardys

G. Johnson  
W. S. Hazelton  
R. P. Snaider  
E. L. Murphy  
D. T. Huang

ch

8144130434 (pp)

R. E. GINNA NUCLEAR POWER PLANT  
REVIEW OF THE STEAM GENERATOR TUBE INSERVICE INSPECTION PROGRAM  
FOR THE 1980 TO 1989 INTERVAL  
SAFETY EVALUATION REPORT

MATERIALS ENGINEERING BRANCH  
INSERVICE INSPECTION SECTION

INTRODUCTION

By letter dated November 6, 1980, Rochester Gas and Electric Corporation (the licensee) submitted the "Ginna Station Inservice Inspection Program for the 1980-1989 Interval" for review. Changes have been incorporated into Paragraph 5.7.1.1 and 6.5 of the Ginna Inservice Inspection Program in order to permit the installation of a maximum of ten test sleeves in steam generator tubes which would otherwise require plugging. These changes permit sleeving instead of plugging of tubes with unacceptable defects.

EVALUATION

We have reviewed the licensee's submittal dated November 6, 1980 regarding the Ginna Inservice Inspection Program. Based upon our review, we conclude that this inspection program meets the recommendations of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," Revision 1 and the requirements of Section XI of ASME Code with respect to the inspection methods to be used, provisions for a baseline inspections, selection and sampling of tubes, and inspection intervals. However, the Ginna Inservice Inspection Program is incomplete with respect to the installation of ten test sleeves in steam generator tubes, since it does not contain actions to be taken in the event defects are identified in the test sleeves, e.g. plugging limit for the ten test sleeves. Furthermore, two typographical errors should be revised to convey the same meaning regarding scope of

steam generators inservice inspection as originally intended. They are as follows:

1. The reference number in Paragraph 3.5 should be changed to 5 in order to be consistent with the reference list.
2. Paragraph 5.7.1.1 should be revised to indicate that the plant may resume operation only when both conditions (a) and (b) are met.

In conclusion, we find that the steam generator tube inservice inspection portion of the "Ginna Station Inservice Inspection Program for the 1980-1989 Interval" is acceptable with the condition that plugging limit for the ten test sleeves be established and the two typographical errors mentioned above be corrected.