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

MAR 24 1980

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Robert B. Minogue, Director
Office of Standards Development

Victor Stello, Jr., Director
Office of Inspection and Enforcement

FROM: Robert J. Budnitz, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER # 83 STEAM GENERATOR TUBE INTEGRITY

INTRODUCTION AND BACKGROUND

This Research Information Letter transmits the results of completed research dealing with the integrity of artificially defected Inconel 600 tubing typical of that found in present service in pressurized steam generators. This research was conducted in response to the NRR Users Request RSR 76-3 on Confirmatory Research for Experimental Determination of Failure Pressure of Steam Generator Tubes, dated April 8, 1976. The research was conducted to develop data to assist NRC regulatory staff in their duties of evaluating results of in-service inspections of nuclear power plants. The program description and results are given in NUREG/CR-0718 (enclosed).

For many years, a sodium phosphate treatment for PWR secondary coolant was widely used for recirculation type (U-tubed) steam generators. This treatment removed precipitated or suspended solids by blowdown. It was successful as a scale inhibitor. However, in early use, many PWR U-tubed steam generators with Inconel 600 tubing experienced stress corrosion cracking. The cracking was attributed to free caustic which can be formed when the Na/PO_4 ratio exceeds the recommended limit of 2.6. In addition, some of the insoluble metallic phosphates, formed by the reaction of sodium phosphates with the dissolved solids in the feedwater, were not adequately removed by blowdown. These precipitated phosphates tended to accumulate as sludge on the tube sheet and the tube supports at the central portion of the tube bundle where restricted water flow and high heat flux occur. Phosphate concentration at crevices in areas of the steam generator caused localized wastage resulting in thinning of the tube wall. The problem of stress corrosion was alleviated by maintaining the Na/PO_4 ratio between 2.3 and 2.6. Although the recommended Na/PO_4 ratio was maintained, it did not correct the phosphate concentration problem that caused the wastage of the Inconel 600 tubing. Largely to correct the wastage and caustic stress corrosion cracking encountered with the phosphate treatment, most PWR's with the recirculation-type steam generators using a phosphate treatment for the secondary coolant have now converted to an all-volatile chemistry (AVT). The adaption of AVT has, to a great extent, reduced the rapid degradation of the Inconel tubing which occurred during the phosphate treatment. Another problem, that of denting, has since occurred. This problem, which is affecting the continued operation of several U-tube steam generators with carbon steel-drilled hole tube support plates, does not form a principal part of the research described in this memorandum and will not be discussed herein.

Regardless of the cause, there are defects in Inconel 600 tubing in in-service PWR steam generators. This memorandum describes the initial results of the Steam Generator Tube Integrity (SGTI) Program being conducted by the Battelle-Pacific Northwest Laboratories, sponsored by the Metallurgy and Materials Research Branch, Division of Reactor Safety Research of NRC. The main objective of the SGTI Program is to establish a large data base to assist NRC in its regulatory duties as regards PWR steam generators. The main areas of investigation in this program are the behavior of defected steam generator tubing under various pressure test conditions, the assessment of the single frequency eddy current (EC) inspection method which is the presently accepted in-service inspection (ISI) method for examining and qualifying in situ PWR steam generator tubing and correlation of the EC flaw characterization results with the pressure test results. The pressure testing conducted on the defected tubing included burst, collapse and leak rate tests. All defected specimens used for the pressure tests were inspected using the single frequency EC method. This procedure developed significant data allowing comparison of the EC indications with the actual defect geometries and the burst pressure for these specimens.

DESCRIPTION OF EXPERIMENTS

Defect Geometries and Tubing

All tubing used in this program was fabricated from Inconel 600 representative of that currently in use in PWR steam generators and was supplied by a vendor who normally supplies such tubing to the industry (the Westinghouse Blairsville, Pennsylvania mill). Four sizes of tubing were examined: 0.875 in. x 0.050 in.; 0.750 in x 0.048 in.; 0.750 in. x 0.043 in.; and 0.625 in x 0.034 in., where the first number in each set represents the outer diameter and the second number, the nominal wall thickness. All tubing, upon receipt, was carefully examined for defects and ovality using EC and ultrasonic inspection techniques. The tubes were then cut into 12-inch specimen lengths following which the defects were machined into them. After machining, each specimen defect was replicated using a silicone casting compound. Thus, precise knowledge of each defect geometry was obtained. The types of geometries were selected to emulate known or expected defects in PWR steam generators. These defects, shown in Figures 1 through 4 of NUREG/CR-0718, are Electro Discharge Machining (EDM) slots, elliptical wastage, elliptical wastage plus through-wall EDM slots, and uniform thinning. The range of defect lengths, depths and geometry combinations was quite extensive as shown in Table 1. All told, close to 600 tube specimens were prepared and tested.

Burst Tests

Burst testing was conducted in an autoclave assembly. Pressurization of both the tube (or primary side) and the autoclave (or secondary side) was with water, chemically simulating PWR steam generator feedwater. During burst testing, the autoclave and its load of tubes were simultaneously pressurized to 2250 psig, then heated to 600°F. The autoclave (secondary side) was then slowly depressurized to 1000 psi. When thermal and pressure equilibrium was achieved, the tubes (primary side) were individually pressurized at a rate of 1000 to 2000 psig/min. Exact burst pressures were determined by sudden loss of pressure in the tubes.

Collapse Tests

For the collapse tests, each tube specimen was placed inside a small pressure vessel and this assembly was then placed in the autoclave. These tests were carried out by first pressurizing the tube and the surrounding small pressure vessel simultaneously to 2250 psig, using the

TABLE 1

RANGE OF DEFECT DIMENSIONS

EDM SLOTS

DEPTH	25-30%, 55-60%, 85-90%
LENGTH	1/4 IN., 1/2 IN., 1 1/2 IN.
WIDTH	0.003 IN. — 0.010 IN.
END RADIUS	0.01 IN. — 0.02 IN.

ELLIPTICAL WASTAGE

DEPTH	25-30%, 55-60%, 85-90%
LENGTH	≈ 1 1/2 IN.
CUTTER RADIUS	6 IN., 12 IN., 24 IN.
WRAP ANGLE	0°, 45°, 135°

UNIFORM THINNING

DEPTH	25-30%, 55-60%, 75-80%
LENGTH	3/16 IN., 3/4 IN., 1 1/2 IN.
END RADIUS	1/16 IN.

same water chemistry as for the burst tests. The autoclave was then pressurized to 2250 psig and heated to 600°F. After thermal and pressure equilibrium occurred, the tubes (primary side) were vented to 1600 psig, thus establishing an external pressure differential of 650 psi across the tubes. The small pressure vessel (secondary side) was then pressurized at the rate of 1000 to 2000 psig/min. until collapse occurred. Collapse was determined by a marked decrease in pressure in the pressure vessel (secondary side), usually accompanied by a noticeable clicking sound emanating from the test assembly.

Leak Rate and Bulging Tests

In the leak rate experimental setup the tube was internally pressurized with 600°F water taken from an autoclave which served as a heated pressurized reservoir and was maintained at 3000 psig through use of He gas overpressure. Pressurization of the tube was through a flow regulator valve and flow meter. Leak rates, pressure and temperature for each specimen were recorded every second using the data logger as the pressure to the specimen was increased by opening the flow control valve until maximum capacity of the system was reached. Since critical flow through the slots should be reached at about 1000 psi differential, any increase in flow rate for pressure differentials above 1000 psi would indicate widening or lengthening of the slots.

Eddy Current Tests

The single frequency EC examination of the defected tubing was performed according to the summer 1976 addenda to Section XI, Appendix IV, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. A commercially available EC instrument was used in performing the tube examinations. This instrument, Automation Industries EM 3300, contains the electronic circuitry to drive the inspection coil, the signal separation circuitry, and a cathode ray tube (CRT) display for viewing the signal patterns generated by the defect. The differential bobbin-style probes were also commercial units purchased with the EC instrument. In addition, strip chart and magnetic tape recorders, tube support tray and other test apparatus were assembled from existing laboratory equipment.

RESULTS

Burst Tests

The results showed that the two primary factors governing the burst pressures for the defected tubes studied in this program are the defect depth and the defect length. This is true for all types of defects with the possible exception of the elliptical wastage type defect, where, because of the unique geometry, depth alone appears to be the predominant parameter. Figure 43 of NUREG/CR-0718 shows the burst pressure of various defect depth and length combinations of EDM slots in 0.875 x 0.050 in. tubes as a function of Maximum Degradation (i.e., defect depth/wall thickness). As can be seen, the data are fairly consistent, showing definite trends. Figure 44 of NUREG/CR-0718 shows the same data as a function of defect length. Again definite trends are discernible. It is quite clear that for any given EDM slot depth, the burst pressure approaches an asymptotic value as the defect length exceeds approximately 1.0 inch in length. The data developed for the burst pressures of EDM slots in the other tube sizes were quite similar to that shown in Figures 43 and 44 of NUREG/CR-0718. Figures 10 and 11 of NUREG/CR-0718 show similar plots for the uniform thinning specimens and Figure 30 of NUREG/CR-0718 shows the burst pressure as a function of depth for the elliptical wastage specimens. In this latter case, the length of the defect was not considered.

All the data from burst tests were put into nondimensional form by dividing the burst pressure of each defect specimen by the burst pressure of the same tube with no defects. The depth of defect was divided by the wall thickness and the length of defect was divided by the square root of the product of the inner radius of the tube and the wall thickness. These data were used to develop a curve fit using a least square methodology.

The curve fit equations developed were:

For EDM slots

$$\frac{\Delta P}{\Delta P_0} = 1 - \frac{h}{t} + \frac{h}{t} e^{-0.373\, L/\sqrt{Rt}} \qquad (1)$$

For Uniform Thinning

$$\frac{\Delta P}{\Delta P_o} = (1 - \frac{h}{t})^{(1 - e^{-.13\ L/\sqrt{R(t-h)}})} \qquad (2)$$

For Elliptical Wastage

$$\frac{\Delta P}{\Delta P_o} = (1 - \frac{h}{t})^{0.604} \qquad (3)$$

where

$\frac{\Delta P}{\Delta P_o}$ = Ratio of defected to undefected burst pressures

h = Defect depth

t = Wall thickness

R = Inner radius of tube

L = Defect length

It should be noted that these equations hold for all sizes of Inconel 600 tubes tested. Figures 60 and 61 of NUREG/CR-0718 show predicted burst pressure ratios for all EDM slots as a function of defect, length and depth, respectively. Figures 28 and 29 of NUREG/CR-0718 are similar plots for the uniform thinning specimens, and Figure 42 of NUREG/CR-0718 is the plot of the burst predictions for the elliptical wastage. Figures 61 and 29 of NUREG/CR-0718 clearly show the asymptotic nature of the effect of length of EDM slots and uniform thinning defects. Figures 27, 41 and 59 of NUREG/CR-0718 show the comparison of predicted burst parameters to actual burst parameters for all tubes tested.

<u>Collapse Tests</u>

Figures 81, 94 and 105 of NUREG/CR-0718 present the collapse data for the EDM, uniform thinning and elliptical wastage defects, respectively, in the 0.875 x 0.050 in. tube specimens. These data are typical of that achieved for the other size tubes. As can be seen, the severity of the EDM defect had little effect upon the collapse pressures. This is as one would expect since the external pressure would tend to close the defect rather than to open it as in the case of internal pressure in the tube. The other types of defects do show a strong relation between defect depth and length and collapse pressure. However, it should be noted that even with defect depths of 75 to 90 percent of wall for the uniform thinning and elliptical wastage specimens, the collapse pressures were considerably above the external pressure differential than one might expect to occur in the event of a loss-of-coolant accident (LOCA), approximately 1000 psi.

Examination of the collapse specimens after test showed that in the case of several severely degraded tubes, 75 percent of wall thickness or greater, some of the tubes did crack open during the collapse mode. If this had occurred at differential pressures of 1000 psig or less, in-leakage from the secondary to primary side might occur. However, it must be strongly emphasized that in all cases tested the collapse of these tubes and the subsequent loss of pressure integrity occurred at pressures considerably higher than could occur under the most credible accident condition, LOCA.

Leak Rate Tests

The results of leak rate tests are described on pages 95 and 96 of NUREG/CR-0718 and tabulated in Appendix E. These results are limited and inconclusive; hence, no firm conclusions can be drawn from them concerning rate of tube leakage as a function of defect size. A number of additional leak rate tests are being performed in FY 1980. These results will be transmitted to the staff promptly upon completion and will simply be considered as supplementary to this Research Information Letter.

Eddy Current Tests

The results of the signal interpretation showed significant error in the EC estimation of defect depth. The complex shape of several signals made determination of the phase angle difficult. Signals from nearly identical geometries produced very different signal patterns and depth indications. Figures 130, 131 and 132 of NUREG/CR-0718 show the comparison between the EC indicated depth of flaw and the actual depth of flaw for all tube sizes tested for the EDM slots, uniform thinning and the elliptical wastage defect specimens, respectively. In these figures, if the EC inspection procedures result in exact indications of defect length, all data points would fall along the 45° line. Any points below this line are conservative since the actual degradation is less than the indicated degradation and the defect depth is overestimated. Any points above this line are unconservative since the defect depth is underestimated. As can be seen, considerable error was found, particularly in the EDM slot and uniform thinning specimens. The former tended toward unconservative results while the results for the latter tended to be uniformly conservative.

As far as present practice is concerned, it is necessary to assess the results of the EC tests against the actual burst pressures of the tubes inspected. Figure 1 of this Research Information Letter presents the depth indications of the EC tests and the resultant burst pressure of each tube tested regardless of the geometry of the specimens. The cross hatch section on this figure represents that region where present plugging practice is employed. That is, accepting a plugging criteria of an indicated defect depth of 55 percent of wall thickness, there are no tube failures below a pressure differential of 4000 psi across the tube. Extension of the 55 percent indicated line across the plot and examination of those points that are above the line and to the right of the 4000 psi pressure vertical line indicates that many tubes would presently be plugged that have burst pressures well in excess of 4000 psi. Part of the reason for this is that present plugging practice does not consider defect length. It was shown previously that defect length is an important parameter affecting the burst pressure of the tubes.

RECOMMENDATIONS

The work reported above deals with pressure bearing characteristics and EC evaluation of carefully mechanically defected Inconel 600 steam generator tubing. It is still to be shown that these defects and the test results do, in fact, reflect the reality of actual in-service defected tubes. This work is in progress and will be reported on at a later date. However, the work done to date does represent the first significant data base generated and the first data base generated by others, besides steam generator vendors. Assuming that the mechanically defected tubes do emulate the performance characteristics of in-service defected steam generator tubes, then the following conclusions can be stated:

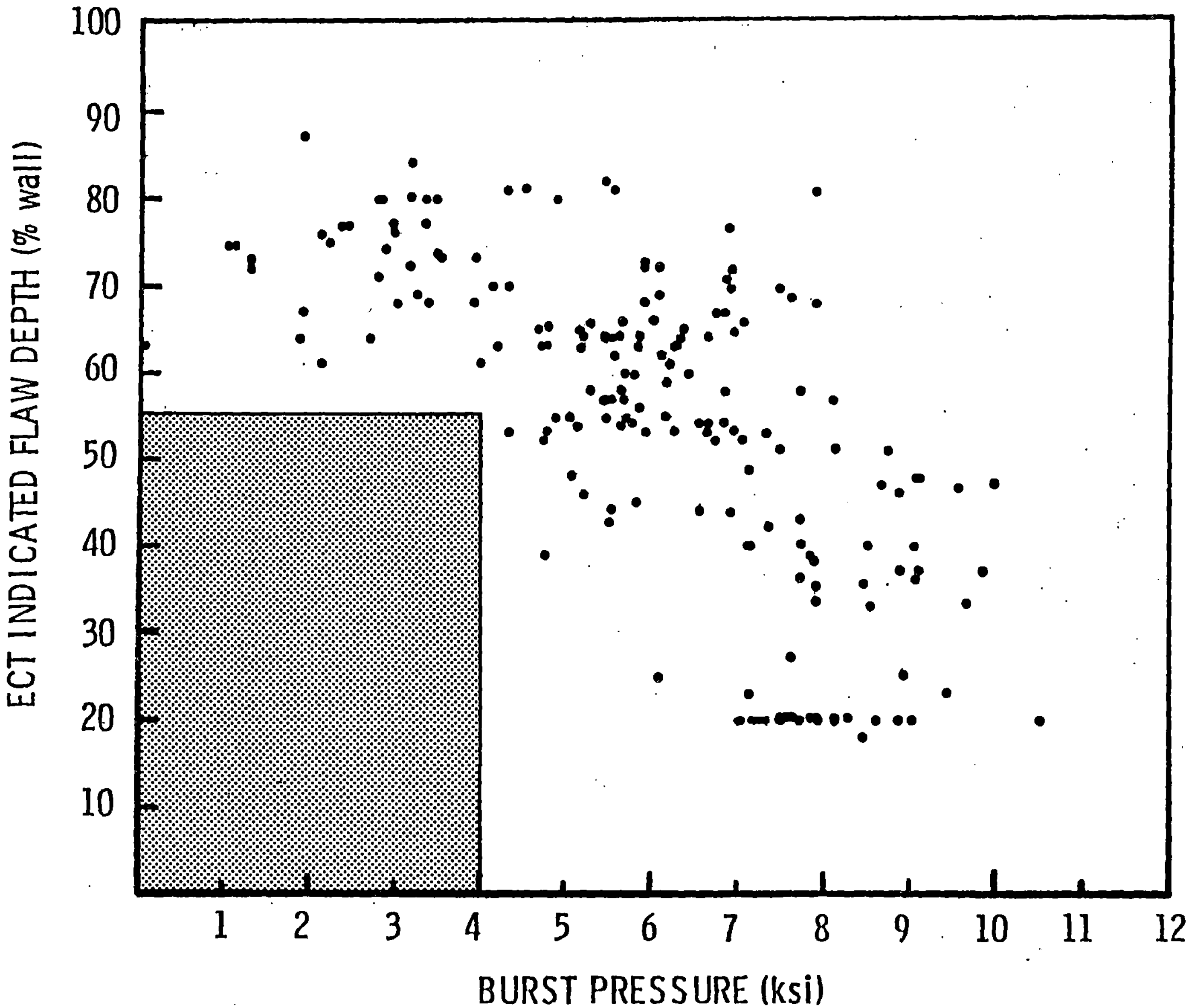

FIGURE 1

Depth indications of eddy current tests versus burst pressure for all tube geometries.

- Burst and collapse pressure relations were shown to exist which are highly dependent upon exact defect geometry, length as well as depth of defect being the important parameters. Present regulations, as defined in U.S. NRC Regulatory Guide 1.21, do not require the measurement or assessment of defect length in the judgement of a tube's ability to sustain its pressure boundary integrity.

- In all cases tested, the collapse of these tubes and the subsequent loss of pressure integrity occurred at pressures considerably higher than could occur under the most credible accident condition, LOCA.

- The tests conducted showed that the currently used method for steam generator tube ISI (the single frequency EC inspection) is capable of significant inaccuracies in defect measurement and has a low probability of detecting small volume defects in straight section tubes under optimum test conditions.

- The analysis of the burst data has led to a set of formulae which clearly define the residual strength (margin-to-burst) of defected tubes where the defect geometries are known.

- On the whole, due to the large margin of safety built into the choice of steam generator tubing wall thickness dimensions and the inherent toughness of Inconel 600 material, present inspection and plugging criteria appear to be conservative in nature, as evidenced by the high burst pressures in tubes where defect depth measured by EC exceeded the plugging depth.

It is clear, however, that significant improvement in steam generator ISI methodology is needed to clearly and accurately identify and define tubing defects. Until this is accomplished, no accurate judgements can be made as to a specific tube's life expectancy taking into account its chemical, thermal and pressure environment and any postulated accident conditions.

NRC reports issued prior to this memorandum giving background information are:

- NUREG-0359, "Steam Generator Tube Integrity Program, Quarterly Report, January 1 - March 31, 1977."

- NUREG/CR-0277, "Steam Generator Tube Integrity Program, Annual Progress Report, January 1 - December 31, 1977."

- NUREG/CR-0086 (PNL-2653-1), "Reactor Safety Research Programs, Quarterly Report, January 1 - March 31, 1978."

- NUREG/CR-0341 (PNL-2653-2), "Reactor Safety Research Programs, Quarterly Report, April 1 - June 30, 1978."

Robert J Budnitz

Robert J. Budnitz, Director
Office of Nuclear Regulatory Research

Enclosure: NUREG/CR-0718

H. R. Denton
R. B. Minogue
V. Stello

NRC reports issued prior to this letter giving background information are:

- NUREG-0359, "Steam Generator Tube Integrity Program," Quarterly Report, January 1 - March 31, 1977.
- NUREG/CR-0277, "Steam Generator Tube Integrity Program," Annual Progress Report, January 1 - December 31, 1977.
- NUREG/CR-0086 (PNL-2653-1), "Reactor Safety Research Programs," Quarterly Report, January 1 - March 31, 1978.
- NUREG/CR-0341 (PNL-2653-2), "Reactor Safety Research Programs," Quarterly Report, April 1 - June 30, 1978.

R. J. Budnitz, Director
Office of Nuclear Regulatory Research

Enclosures:
1. Table 1
2. Figure 1
3. NUREG/CR-0718

DIST.
Subj
Circ
Chron
JM rf
Branch rf
JM cy
Serpan cy
Kenneally cy
Shao cy
Larkins cy
Budnitz cy

GRSR:MMRB	DSS:MTEB	DSS:AD/E	RSR:D
JMuscara:bl	SPawlicki	JPKnight	TEMurley
2/12/80	2/26/80	2/ /80	2/ /80

OFFICE ▶	GRSR:MMRB	NRR	NRR	GRSR	RES:PCB	RES:D
SURNAME ▶	CZSerpan	VSNoonan	RJBosnak	Kenneally/Shao	JTLarkins	RJBudnitz
DATE ▶	2/12/80	2/15/80	2/19/80	2/ /80	2/ /80	2/ /80

NRC reports issued prior to this memorandum giving background information are:

- NUREG-0359, "Steam Generator Tube Integrity Program, Quarterly Report, January 1 - March 31, 1977."
- NUREG/CR-0277, "Steam Generator Tube Integrity Program, Annual Progress Report, January 1 - December 31, 1977."
- NUREG/CR-0086 (PNL-2653-1), "Reactor Safety Research Programs, Quarterly Report, January 1 - March 31, 1978."
- NUREG/CR-0341 (PNL-2653-2), "Reactor Safety Research Programs, Quarterly Report, April 1 - June 30, 1978."

Robert J. Budnitz, Director
Office of Nuclear Regulatory Research

Enclosures:
1. NUREG/CR-0718
2. Table 1
3. Figure 1

DIST.
Subj
Circ
Chron
Muscara rf
Branch rf
Muscara cy
Serpan cy
Kenneally cy
Shao cy
Murley cy
Larkins cy
Budnitz cy
Noonan cy
Bosnak cy
Pawlicki cy
Knight cy

*See previous yellow for concurrence.

Good RJC

RMK 3/3/80

GRSR:MMRB	GRSR:MMRB	NRR	RES:D
JMuscara:bl*	CZSerpan*	VSNoonan*	RJBudnitz
2/12/80	2/12/80	2/15/80	3/ /80

OFFICE	NRR	DSS:MTEB	DSS:AD/E	GRSR	RSR:D	RES:PCB
SURNAME	RJBosnak*	SPawlicki*	JPKnight*	Kenneally/Shao	TEMurley	JTLarkins
DATE	2/19/80	2/25/80	2/26/80	3/4/80	3/14/80	3/ /80

NRC reports issued prior to this memorandum giving background information are:

- NUREG-0359, "Steam Generator Tube Integrity Program, Quarterly Report, January 1 - March 31, 1977."
- NUREG/CR-0277, "Steam Generator Tube Integrity Program, Annual Progress Report, January 1 - December 31, 1977."
- NUREG/CR-0086 (PNL-2653-1), "Reactor Safety Research Programs, Quarterly Report, January 1 - March 31, 1978."
- NUREG/CR-0341 (PNL-2653-2), "Reactor Safety Research Programs, Quarterly Report, April 1 - June 30, 1978."

Robert Budnitz

Robert J. Budnitz, Director
Office of Nuclear Regulatory Research

Enclosure: NUREG/CR-0718

DISTRIBUTION:
Subj
Circ
Chron
Muscara rf
Branch rf
Muscara cy
Serpan cy
Kenneally cy
Shao cy
Murley cy
Larkins cy
Budnitz cy
Noonan cy
Bosnak cy
Pawlicki cy
Knight cy

*See previous yellow for concurrence.

GRSR:MMRB	GRSR:MMRB	NRR	NRR
JMuscara:bl*	CZSerpan*	VSNoonan*	RJBosnak*
2/12/80	2/12/80	2/15/80	2/19/80

OFFICE	DSS:MTEB	DSS:AD/E	GRSR	RSR:D	~~RES:PCB~~	RES:D
SURNAME	SPawlicki*	JPKnight*	Kenneally*/Shao*	TEMurley*	VTL JTLarkins	RJB RJBudnitz
DATE	2/25/80	2/26/80	3/3/80 3/4/80	3/14/80	3/21/80	3/24/80

U.S. GOVERNMENT PRINTING OFFICE: 1979 — 294-644

FROM:	DATE OF DOCUMENT	DATE RECEIVED	NO.:
P. G. SHEWMON, CHAIRMAN, ACRS	5/21/80	5/23/80	002013
	LTR.: X	MEMO: REPORT:	OTHER:
TO: L. SHAO	ORIG.: X	CC: OTHER:	
	ACTION NECESSARY ☒ / NO ACTION NECESSARY ☐	CONCURRENCE ☐ / COMMENT ☐	DATE ANSWERED: / BY:
CLASSIF.: OUO / POST OFFICE / REG. NO.:	FILE CODE:		

DESCRIPTION: (Must Be Unclassified)	REFERRED TO	DATE	RECEIVED BY	DATE
RIL #83 STEAM GENERATOR INTEGRITY	SHAO	5/23		
	CY: BUDNITZ			
ENCLOSURES:	LARKINS			
	SLATER			
	MURLEY			
	SERPAN			
	FILE			
REMARKS:				

U. S. NUCLEAR REGULATORY COMMISSION

MAIL CONTROL FORM

FORM NRC-326S (6-76)

UNITED STAT' S
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

FOIA EXEMPTION (b)5

file w/ RIL # 83

May 21, 1980

MEMORANDUM FOR: L. Shao, Assistant Director, General Reactor Safety Research, RES

FROM: P. G. Shewmon, Chairman, ACRS Subcommittee on Metal Components

SUBJECT: RIL #83 STEAM GENERATOR INTEGRITY

In reviewing the RES programs, one all too often gets the impression from the contractor that almost no one else is or has worked on the problem before. I happen to know some of the people at Westinghouse who ruptured hundreds of Inconel 600 tubes to convince themselves and the NRC of the validity of the conclusions drawn by the referenced report. It's quite probable that someone else in the world has looked at the problem too.

This report makes no reference to any existing data base in NRC's hands or previous work on the subject. It strikes me as unscholarly, but more to the point as unprofessional and discomforting. Since the NRC is continuing to sink a lot of money into this project, I hope it's done and monitored by people who can and will learn from the work of others -- and admit it.

From a management viewpoint, there is the broader question of how the NRC can get effective professional review and counsel on their continuing programs. I think it's a problem worth working on though it's unclear how you motivate the reviewers without making them your contractors. I suspect vendor employees could and should be used more than they are, but I'm not familiar enough with your restraints to say.

E. U. [illegible] for

P. G. Shewmon, Chairman
ACRS Subcommittee on Metal Components

cc: C. Serpan, RSR
R. Budnitz, RES