OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION

Steam Generator Tube Rupture Incident

Final Report

June 1984

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1.0 INTRODUCTION

At 4:50 p.m. on May 16, 1984, the Fort Calhoun Station experienced a tube failure in the "B" steam generator. At the time of tube failure, the plant was performing a Reactor Coolant System Leak Test as part of the normal startup procedure. The tube failure incident included a primary system depressurization with no release of radioactivity into the environment.

OPPD has performed extensive evaluations of the tube failure event. The evaluations were presented to the NRC staff on May 29, 1984, and documented by letters dated 5/31/84 (LIC-84-159, W. C. Jones to J. Miller and LIC-84-160, W. C. Jones to J. Collins). In addition, numerous telephone calls have been held between District personnel and NRC personnel. The above submittals addressed the following items: selective plant parameters and plant status immediately prior to, during, and following the incident; steam generator inspections performed and results obtained; visual inspections and laboratory metallurgical analysis of the failed tube; operation related activities including plant procedures and chemistry analysis; conclusions reached and the District's plans to return to power operation.

On June 5, 1984, the District received a letter signed by Mr. John Collins, Region IV Administrator, directing the District to perform and complete actions in addition to those stated in the District's May 31, 1984, letter. The letter dated June 5, 1984, also stated that NRC approval was required before the Station was taken out of refueling shutdown condition (Mode 5).

The purpose of this report is to summarize the District's activities to date, relating to the steam generator "B" tube failure incident. This report includes an update of the May 31, 1984, submittal referenced above and actions taken in response to the June 5, 1984, communication from the NRC-I&E Region IV Administrator.

This report is comprised of seven major sections. The first, Section 2, is a description of the event. Section 3 describes the steam generator inspections performed and the results obtained. Section 4 describes metallurgical examination performed and the results obtained. Section 5 describes operations related activities. Section 6 describes the District's corrective action to reduce the probability of the failure mechanism. Section 7 describes the District's tube plugging to eliminate suspected defects. The final major section is comprised of the District's 10 CFR 50.59 evaluation.

2.0 DESCRIPTION OF EVENT

2.1 Summary

At 4:50 p.m. on May 16, 1984, the Fort Calhoun Station experienced a tube failure in Steam Generator "B". During a routine plant startup from a refueling outage, the reactor coolant system (RCS) was being pressurized for a leak test. At approximately 1,800 psia, RCS leakage approached 110 gpm with indication of a tube rupture in RC-2B ("B" steam generator). A depressurization and cooldown of the RCS was initiated. RC-2B was isolated. NOTIFICATION OF AN UNUSUAL EVENT was declared. The unusual event was terminated when the RCS was placed in cold shutdown. No release of radioactivity to the environment occurred.

2.2 Operator Procedures and Actions

Various emergency procedures were used by plant personnel during the incident. Plant personnel performed actions dictated by procedures and safely placed the plant in a refueling shutdown condition (Mode 5). Operators' response and the written procedures used for mitigation of the incident were not only adequate, but performed as described and dictated by preplanned procedures for this type of incident.

2.3 Sequence of Events

Table 1 contains a sequence of events for the steam generator tube rupture incident

2.4 Time Sequence - Steam Generator Pressure and Level

Figure 1 contains a time sequence relating to steam generator level and pressure.

Table 1

Initial Conditions

Plant was being taken from Mode 4 to Mode 3

RCS boron approximately 2100 ppm

T_C = 398°F

Pressurizer level = 70%

Pressurizer pressure = 880 psia

Steam generator RC-2B level = 72%, pressure approximately 200 psig

Pressurizer fill in progress for RCS leak test; one charging pump in operation taking suction off of SIRWT

RC pumps RC-3A, RC-3B and RC-3C in operation

Letdown on minimum

Both MSIV's, HCV-1041A and HCV-1042A, open

Steam generator blowdown secured

Feeding both steam generators with FW-6 aux. feed pump; FW bypass valves HCV-1105 and HCV-1106 in AUTO

Atmospheric steam dump valve, HCV-1041, open slightly

The following is the sequence of events for the steam generator tube rupture (SGTR) of May 16, 1984.

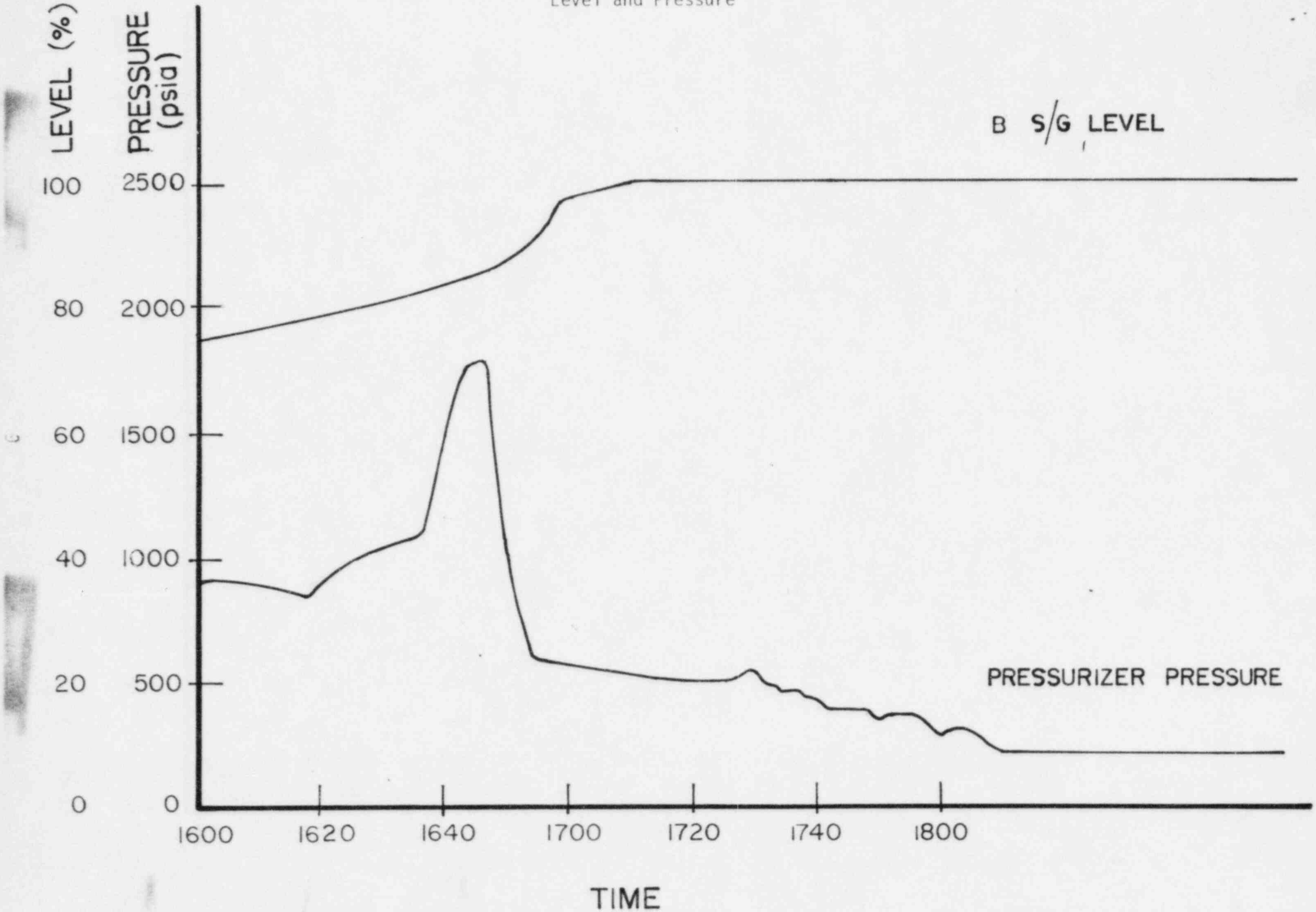
Time	Event
1618	Operator noted that pressurizer level was no longer increasing with single charging pump in operation; pressurizer pressure decreasing slowly; started other two charging pumps.
1636	Pressurizer pressure and level slowly increasing; however, charging flow rate only approximately 50 gpm versus expected flow rate of 120 gpm (probably due to inadequate NPSH with existing SIRWT level and three charging pumps); operator switched charging to VCT, flow rate increased to 120 gpm.
1639	PPLS reset at 1700 psia (automatic).
1641	Pressurizer solid; pressurizer pressure = 1800 psia and slowly increasing

Time	Event
*1642	Operator isolated letdown.
	Operator noted level increasing above setpoint in RC-2B, thought to be leakage through HCV-1106, operator closed block valve HCV-1385.
1645	VCT level approaching 0% despite blended makeup in progress; operator secured two charging pumps; pressurizer pressure = 1850 psia.
1646	PPLS blocked at 1700 psia (operator action).
1648	Pressurizer pressure dropping rapidly.
*1650	Operator noted continuing increase in RC-2B level; auxiliary FW pump FW-6 secured.
1654	Pressurizer pressure = 560 psia; RCS solid; operator opened letdown valve to draw pressurizer bubble.
1658	MSIV from RC-2B, HCV-1042A, closed by operator.
1659	Cooldown of RCS initiated using steam generator RC-2A and atmospheric dump valve HCV-1040.
1700	Reactor coolant pump RC-3C secured.
1701	Reactor coolant pump RC-3B secured.
1711	Notification of unusual event declared.
1717	NRC notified via red phone.
1718	RC-2B level off-scale high; secondary pressure approximately 200 psig.
1720	Steam generator blowdown sample lined up to radioactive waste system; blowdown monitor pegged high.
1730	Cooldown and depressurization of pressurizer initiated using auxiliary spray.
1830	Pressurizer pressure = 220 psia; T_C = 330°F; pressurizer level = 70%.
1841	VCT backfilled with N_2.
2005	Shutdown cooling initiated.

Time	Event
(May 17, 1984)	
0005	Terminated unusual event at 210°F.
*0730	Steam generator RC-2B solid.

* Time approximate based on interviews with operators; precise data unavailable.

Figure 1: Time Sequence Relating to Steam Generator Level and Pressure



3.0 STEAM GENERATOR INSPECTION HISTORY SUMMARY

3.1 Inspection Summary Prior to 1984

The Fort Calhoun Station utilizes two Combustion Engineering vertical U-tube steam generators, each of which contains 5,005 Inconel 600 tubes. The tubes are 0.75 inches outside diameter with 0.048 inch minimum wall thickness.

The Fort Calhoun Station has essentially always operated with a carefully maintained AVT secondary chemistry program. The periodic inspections utilizing visual and state of the art eddy current testing techniques of the steam generators have shown them to be in good condition. The District has endeavored to address operational problems in a timely manner. The results of all of the eddy current examinations prior to 1984 of the steam generator tubes have shown the generators to be in Technical Specification Category C-1.

A pre-operational ECT baseline inspection of 200 tubes per steam generator was performed in July 1973. Some mechanical imperfections were noted in the "A" generator.

225 tubes in each steam generator were ECT inspected at the first refueling outage in February 1975. No evidence of degradation or magnetite denting was noted at that time. The same was true of the inspection of 408 tubes in the "B" steam generator in November 1976.

An ECT inspection of the "A" steam generator in November 1977 was performed in order to assess the imperfection indications which had been discovered in 1973 and 1975. This inspection was limited to 165 tubes and was not intended to meet the requirements of Regulatory Guide 1.83. There was no evidence of deterioration or denting of the type related to magnetite growth at the drilled hole support plates.

500 tubes in the "A" steam generator were inspected in October 1978, using single frequency ECT. Some dent-like indications were observed, but evaluation showed no change with regard to the 1977 indications. One tube showed 38% degradation and two tubes showed less than 20% degradation. Although none of these tubes exceeded the plugging criteria, they were plugged as a precautionary measure. During the 1984 inspection, it was discovered that two tubes had actually been plugged and one end each of two adjacent tubes. The open end of these two tubes have been plugged. The first indications which were reported to the District as magnetite denting resulted from the inspection of 328 tubes in the "B" generator in October of 1981. This was the first inspection which utilized multi-frequency ECT techniques. All previouis exams had been done with single-frequency ECT. One tube was reported as having 38% degradation. This tube was not plugged, and it was reinspected in 1982. Evaluation of the indication at that time showed a dent, but no defect, at the point in question.

In December 1982, 308 tubes in the "A" generator and 302 tubes in the "B" generator were examined using multi frequency ECT. This inspection showed the presence of moderate dent-like indications in both generators. One tube in steam generator "A" showed 20% degradation, and two tubes in steam generator "B" indicated less than 20% degradation.

In addition to the eddy current examinations which are conducted from the primary sides of the steam generators, detailed secondary inspections are conducted at each refueling outage. These inspections involve a detailed crawl-through of the secondary sides of the steam generators to ascertain that all components are properly secured and in good condition, sludge and scaling sampling and analysis, inspection of steam generator internals from the handholes, and photographic documentation. The secondary inspections which have been conducted have shown the Fort Calhoun steam generators to be in good condition and without excessive amounts of deposits.

3.2 1984 Inspection Summary

3.2.1 Planned Outage Inspection

Plans for the March 1984 inspection involved a nominal 1,000 tubes in each steam generator, primarily for assessment of the extent and growth of denting in the No. 8 partial drilled hole support plates as the primary input to a decision to perform the rim cut modification. The actual number of tubes which were examined full length during this inspection were 1,454 in steam generator "A" and 1,034 in steam generator "B". Additional part length examinations were conducted to measure sludge height, and some tubes restricted the passage of an ECT probe and are not included in these totals. The inspection showed further dent-like indications, primarily at the No. 8 partial drilled hole support plate and in the batwing areas. Based on evaluation of this data, the District decided to perform the rim cut modification on the No. 8 partial drilled hole support plate. At the time of this inspection, the evaluation of the data showed no degradation indications in the "A" steam generator and one previously detected indication in the "B" steam generator. Four tubes in steam generator "A" and five tubes in steam generator "B" were plugged due to restriction to passage of a 0.540 inch ECT probe, which is consistent with Combustion Engineering's plugging recommendations for restricted tubes.

Following the performance of the rim cut modification, 120 peripheral tubes in steam generator "A" and 111 peripheral tubes in steam generator "B" were retested to determine if there had been any damage resulting from the performance of the rim cut. One tube in steam generator

"A" was verified to have been damaged and was subsequently plugged. In addition to the peripheral inspections, 68 tubes in steam generator "A" and 69 tubes in steam generator "B", in the area of the No. 6 partial support plate/egg crate interface were examined to determine if any additional tubes were restricted in these areas. No additional restricted tubes were found. 118 tubes in steam generator "B" were examined in the steam-blanketed tight radius U-bend areas for the presence of indications such as have been found at other operating plants; no such indications were found. Also, approximately 50 tubes were examined with a profilometry probe in steam generator "A" in an effort to characterize the dent-like indications and the restriction at the No. 6 support elevation. This inspection was limited to vertical tube sections due to the type of probe that was used.

3.2.2 Post Tube Failure Inspections

The failed tube, L29R84, was eddy current tested in December of 1982. There were no defect or dent indications present in the tube at that time. The data tape from that inspection has been rereviewed subsequent to the failure, and certified analysts have again stated that there is no evidence of defect or dent indications in the tube at that time. This tube was included in the March 1984 inspection program. Reevaluation of the data tape from that March 1984 inspection shows a 99% through-wall defect at the location of the failure. This indication was missed on initial analysis of the data from the March 1984 inspection due to human error. Subsequent to identification of the leaking tube, the location of the failure was confirmed by eddy current testing. There are no defects in other portions of the failed tube.

Following discovery of the leaking tube, the District embarked on a test program which ultimately involved multifrequency eddy current testing of all accessible tubes in both steam generators. There were 24 tubes in Steam Generator "A" and 11 tubes in Steam Generator "B" which could not be inspected using the Zetec SM-4 polar positioner. The frequencies used for this inspection were as follows:

400 KHz differential
200 KHz differential
300 KHz absolute
100 KHz absolute

The 400 and 200 KHz signals were mixed to suppress the effects of the vertical support straps, and the 300 and 100 KHz signals were mixed to suppress the effects of the support plates and egg crates.

In addition, those tubes from the March, 1984, program which were not retested otherwise with bobbin coil or pancake array probe ECT were retested using a 100 KHz absolute test for enhanced defect sensitivity. The original program used 800 KHz instead of 100 KHz in order to mix out ID tube noise and allow better determination of denting in the No. 8 partial support plates.

In addition to the failed tube, B-L29R84, the following tubes showed degradation or defect indications at the hot leg vertical support. The degree of degradation is also indicated.

A-L85R80	<20%
A-L85R82	28%
A-L94R75	<20%
A-L101R80	<20%
B-L85R86	42%
B-L102R77	22%
B-L104R75	26%

The following tubes from the above group which showed indications at the hot leg vertical support were inspected in December, 1982, with results as noted:

A-L101R80 - No Detectable Defect (NDD) at hot leg vertical support, no known dent

B-L102R77 - Dent at hot leg vertical support, NDD
B-L104R75 - NDD at hot leg vertical support, no known dent

From all of the ECT work which was performed, only four tubes showed a defect (> 40%) indication, two in each steam generator. Tube A-L37R18 showed possible evidence of wastage in an area several inches above the tube sheet, with 27% and 53% indications in the wastage area. This tube has not been inspected previously. Because of concerns about the presence of these indications, the District has elected to cut and remove a tube section which contains the indications for future metallurgical analysis. There is no evidence to suggest that this problem is related to the failure of tube B-L29R84 which occurred at the top of the tube bundle. Tube A-L64R85 has a 44% indication just below the #7 hot leg support. This tube was intended to be plugged in 1978, but only the hot leg end was plugged. Reinspection showed progression of the indication. Tube B-L85R86 has a 42% indication at the hot leg vertical support strap. Tube B-L29R82 is the failed tube.

A summary of all degradations or defect indications from the 1984 inspection programs is presented below: Further details are given in Tables 3 and 4. Permeability variations (PV) are also included.

Steam Generator "A"	
< 20%	7
20 - 40%	8
> 40%	2
PV	6

Steam Generator "B"	
< 20%	18
20 - 40%	5
> 40%	2
PV	16

Specialized eddy current testing using 1 x 8 and/or 4 x 4 pancake array probes was performed on 300 tubes on Steam Generator "B". The only tube in this program which showed an indication in the vertical support areas was tube L85R86, which has been discussed previously.

Profilometry, using a 1x8 superflex profilometry probe, was performed and evaluated on 206 tubes in Steam Generator "B". 147 of these tubes are in the outer areas of the tube bundle and pass through all three vertical support straps. The test results for these 147 tubes were compared to the results of a bobbin coil exam performed on the same tubes. Of the dents detected, the largest were at the vertical support strap on the hot leg side of the generator. 74 of the 147 tubes had dent indications at this location. In comparing the two test methods, it was noted that the bobbin coil was only able to detect 59.5% of the dents at this vertical support strap. The overall results for all three vertical support straps showed that the bobbin coil detected 41.5% of the dents detected by profilometry. The bobbin coil also showed smaller dent indications than those that were observed with profilometry. This was not unexpected, however, due to the differences in the two test methods.

Profilometry was conducted on 59 tubes in inner areas of the tube bundle. These tubes have only a single, center, vertical support strap. 21 of these 59 tubes had dent indications at the vertical strap. These dent indications were of significantly lower magnitude than that were noted in tubes which pass through the three vertical support straps. Denting was noted with increasing frequency as the row number increased. From Row 49 outward, nearly all tubes had a dent indication in the vertical support strap. The profilometry data is currently being reduced to obtain strain measurements at the dent locations.

Further information regarding the bobbin coil ECT and profilometry testing is presented in Tables 2, 3 and 4. A summary of the tubes which have been plugged since initial operation is presented in Table 5.

In order to be assured that the probability of detecting degradation or defects is as high as possible, within the limits of eddy current testing, all data taken since the

tube failure has been analyzed and independently reviewed. The data from the March 1984 inspection was reanalyzed and independently reviewed.

3.3 Cycle 8 Leak Detection Program

In February 1984, approximately three weeks prior to a scheduled refueling shutdown, a very small primary-to-secondary leak was discovered in the "B" steam generator. This leak was confirmed two weeks prior to this scheduled shutdown. Based on comparison of primary and secondary coolant activities, the leakage rate was determined to be approximately 0.2 gallons per day. The estimated leak area to give this leak rate at normal operating temperatures and differential pressures is 2×10^{-7} square inches. In a concerted effort to locate the leaking tube, the District conducted two helium mass spectroscopy tests, one each before and after sludge lancing of the "B" steam generator during the 1984 refueling outage. These tests were unable to isolate the leaking tube. The District also conducted a hydrostatic test with a dye indicator as a further effort to locate the leaking tube. This test was also unsuccessful in locating the leaking tube. The failure was detected by adding water in known quantities to the steam generator and inspecting the primary channel heads for evidence of leakage at hold points in the procedure. The tube failure is located between the scallop bars in the vertical batwing support on the hot leg side of the generator, in the second peripheral row from the outside.

The District believes that it is highly likely that the tube which was leaking just prior to the refueling outage is the one which has now failed. This cannot be determined for certain, however, until additional chemical and radiochemical analyses can be conducted following the return of the unit to power operations.

Since the failed portion of the tube was reasonably accessible, the District decided to remove the failed section of tube for metallurgical analysis. The failed section was excised with a TIG torch after removing an equivalent portion of an adjacent tube for access. A brief onsite visual examination of the failed tube section was conducted, and the tube was packaged and shipped to Combustion Engineering's laboratory for analysis. The results of this analysis is documented in Section 4.0 of this report.

TABLE 2

1984 Eddy Current Testing Summary and Results
Steam Generators A and B

Steam Generator A:

Number of tubes - 5,005

Number plugged prior to initial startup - 24 (to hold orifice plate)

Number plugged between initial startup and 1984 refueling outage - 2 on both ends, 2 on one end each

Number of tubes inspected using bobbin coil multi-frequency ECT in 1984 - 4,955 (includes 2 plugged on one end)

Number inaccessible - 24

Results

Number of tubes with < 20% indications - 7

Number of degraded tubes - 8

Number of defective tubes - 2

Number of defective tubes plugged - 2

Number of other tubes plugged - 11

1 due to rim cut damage

4 due to restriction

4 due to indications in hot leg vertical support

1 due to indication approaching plugging limit

1 electively plugged HL end of partially plugged tube

Number of tubes profiled - 150

Results - analysis in progess

TABLE 2 (Continued)

Steam Generator B:

Number of tubes - 5,005

Number plugged prior to initial startup - 24 (to hold orifice plate)

Number plugged between initial startup and 1984 refueling outage - 0

Number of tubes inspected using bobbin coil multi-frequency ECT in 1984 - 4,970

Number inaccessible - 11

Results

Number of tubes with < 20% indications - 18

Number of degraded tubes - 5

Number of defective tubes - 2

Number of defective tubes plugged - 2

Number of other tubes plugged - 10

5 due to restriction

1 cut for access to failed tube

1 electively plugged due to proximity to failed tube

1 mis-plugged during process of plugging failed tube

2 due to indications in hot leg vertical support

Number of tubes tested with 1x8 and/or 4x4 pancake array probes - 300

Results - only tube L85R86 showed an indication in a vertical support area

Number of tubes profiled and analyzed - 206

Results - Denting/ovalization present in vertical supports, predominantly on the hot leg side

Number of additional tubes profiled - 70

Results - analysis in progress

Table 3

Summary of 1984 ECT Indications

Steam Generator A

Imperfection Indications

Tube	Depth	Location
L60R55	< 20%	2.2" above HL tube sheet
L70R43	< 20%	3.5" above HL tube sheet
L70R43	< 20%	4.0" above HL tube sheet
L76R51	< 20%	2.5" above HL tube sheet
L85R80	< 20%	Hot leg vertical support
L94R75	< 20%	Hot leg vertical support
L101R80	< 20%	Hot leg vertical support

Degradation Indications

Tube	Depth	Location
L32R43	28%	15" above #3 support
L45R34	26%	9.1" above HL tube sheet
L58R87	27%	1.0" above #8 HL support
L64R27	23%	25" above #4 CL tube sheet
L72R23	39%	10" above CL tube sheet
L85R22	22%	4" above HL tube sheet
L85R42	23%	4" above HL tube sheet
L85R82	28%	Hot leg vertical support

Defect Indication

Tube	Depth	Location
L37R18	27% & 53%	4" wastage area beginning 3" above CL tube-sheet, 2 defects within that area
L64R85	44%	Just below #7 HL support

Permeability Variations - Six Reported

Steam Generator B

Imperfection Indications

Tube	Depth	Location
L32R25	< 20%	4.3" above HL tube sheet
L39R30	< 20%	5.3" above HL tube sheet
L52R39	< 20%	6.6" above HL tube sheet,
L54R49	< 20%	4" above HL tube sheet
L57R64	< 20%	#3 CL tube support
L63R38	< 20%	6.5" above HL tube sheet
L65R36	< 20%	5.0" above HL tube sheet
L65R40	< 20%	3.0" above HL tube sheet
L66R41	< 20%	6.2" above HL tube sheet
L66R51	< 20%	3.5" above HL tube sheet
L67R40	< 20%	7.2" above HL tube sheet
L67R50	< 20%	4.0" above HL tube sheet
L65R52	< 20%	5.2" above HL tube sheet
L69R40	< 20%	4.0" above HL tube sheet
L71R48	< 20%	4.0" above HL tube sheet
L74R47	< 20%	2.5" above HL tube sheet
L83R40	< 20%	3.5" above HL tube sheet
L89R52	< 20%	4.5" above #3 HL support

Degradation Indications

Tube	Depth	Location
L36R29	25%	3.0" above HL tube sheet
L43R38	20%	5.0" above HL tube sheet
L62R57	31%	12.8" above #1 CL support
L102R77	22%	Hot leg vertical support
L104R75	26%	Hot leg vertical support

Defect Indications

Tube	Depth	Location
L29R84	100%	Hot leg vertical support
L85R86	42%	Hot leg vertical support

Permeability Variations - Sixteen Reported

Table 4 - Profilometry Data
Steam Generator B

The profilometry data reduction for the horizontal tube runs was done manually to read the maximum radius change at a given vertical support strap. The numbers recorded were in terms of signal amplitude (strip chart divisions), and, although the correlation is not strictly linear, signal amplitude to dent size in mils is roughly 1:1. The following table presents the data in terms of relative dent size at each vertical support location.

Tubes with Three Vertical Supports
(147 tubes profiled)

Location (Number of Dents)	Approximate Size of Dents (Number of Dents)		
	0-10 mils	10-20 mils	> 20 mils
Hot Leg (74)	(21)	(8)	(45)
Center (47)	(39)	(4)	(4)
Cold Leg (24)	(17)	(5)	(2)

Tubes with Center Support Only
(59 tubes profiled)

Number of Dents	Approximate Size of Dents (Number of Dents)		
	0-10 mils	10-20 mils	> 20 mils
21	(18)	(3)	(0)

Table 5 - Summary of Plugged Tubes

Steam Generator "A"

Tubes plugged to hold orifice plate (pre-operation) - 24

Tubes plugged - 1978 Refueling Outage
L79R98
L80R97
L64R85 - Hot leg side only
L63R84 - Cold leg side only

Tubes plugged - April, 1984
L16R67 - Rim cut damage
L72R83 - Restriction at #6 tube support
L72485 - Restriction at #6 tube support
L75R92 - Restriction at #6 tube support
L86R81 - Restriction at #6 tube support

Tubes plugged - June, 1984
L37R18 - Wastage and defect indications
L63R84 - Plugged hot leg to eliminate concern about partially plugged tube
L64R85 - Defect indication - plugged cold leg
L72R23 - Indication approaching plugging limit
L85R80 - Indication at hot leg vertical support
L85R82 - Indication at hot leg vertical support
L94R75 - Indication at hot leg vertical support
L101R80 - Indication at hot leg vertical support

Steam Generator "B"

Tubes plugged to hold orifice plate (pre-operation) - 24

No operational plugging prior to 1984 Refueling Outage

Tubes plugged - April, 1984
L51R78
L53R98
L55R94
L65R100
L83R82

All of these tubes were plugged due to restriction at the #6 tube support

Tubes plugged - June, 1984
L29R82 - Electively plugged due to proximity to failed tube
L29R84 - Failed tube
L29R86 - Removed for access to failed tube
L30R83 - Misplugged
L85R86 - Defect indication at hot leg vertical support
L102R77 - Indication at hot leg vertical support
L104R75 - Indication at hot leg vertical support

4.0 VISUAL INSPECTION AND LABORATORY ANALYSIS

4.1 Steam Generator B - Tube L29R84 Section

The following describes the results of the destructive examination of a section of tube L29R84 from the Fort Calhoun "B" steam generator and the probable failure mechanism of this tube.

4.1.1 Receipt Inspection

Upon receipt at the CE lab, the two tube specimens labeled 23B and 23C were visually inspected. Two cracks were observed on piece 23B. The first was a large, axial (1-1/4") "fishmouth" type crack, while the second was a series of small (approximately 1/4") length fissures which made an acute angle (45°) relative to the axis of the tube. One end from each tube section was removed to allow the eddy current probes to pass. Tube section 23B was the length of steam generator tube L29R84 from inboard of the first vertical tube support to outboard of the hot leg batwing tube support. The tube section labeled 23C was the length of the same tube from inboard of the first vertical tube support to outboard of the middle vertical tube support.

A. <u>Eddy Current Testing</u>

The Combustion Engineering (CE) field/laboratory Miz 12 eddy current test equipment was calibrated using an inline calibration standard with mix frequencies of 400 and 100 kHz. A bobbin probe was used for the laboratory inspection of the tube sections.

A 100% throughwall signal was identified at the location of the "fishmouth" failure on tube specimen 23B. One end of the defect signal was not clearly resolved due to probe interference at the torch cut end of the tube section.

Approximately 1/4 of an inch from the hot leg end of the first defect, a second O.D. initiated defect signal was observed which corresponded to the second crack. A kink in the tube distorted the signals from the small defect, rendering depth estimates impossible.

Significant dent signals were noted at the general location of the defects in 23B. These signals could not be quantified due to bending of the tube during removal from the steam generator. Several small dings were seen along the remaining portion of the tube section. These were not observed within the steam generator and, consequently, were probably caused during tube removal from the steam generator.

No defect signals were observed in the tube section labeled 23C.

These results are comparable to the reanalysis of the June 1984 in-service steam generator ECT inspection data, wherein two defect signals approximately 1/4" apart were identified. The first was approximately 100%, while the second was estimated at 50% throughwall.

B. Visual Inspection - Macro Photography, Video Taping

The first step of visual inspection consisted of documenting the as-received condition by videography. Subsequently, photomacrographs were taken to document the appearance of the tube section, including defect areas and areas of deposits. In particular, photographs were taken to illustrate the lower and upper scallop bar deposits, the overall appearance of the defects, the area between the two defects, closeups of each defect, and finally the appearance of the fracture surface. The large crack was located at the 6 o'clock position in the steam generator, as confirmed by the relative position of the scallop bar contact areas.

C. Dimensional Measurements

Figure 2 illustrates the dimensional measurements around the defect region. These measurements were taken before descaling and, as such, include the thickness of residual deposits. The measurement data indicate that the tube was ovalized. The major axis (6-12 o'clock) was elongated by 0.046-0.122 inch, while the minor axis (3-9 o'clock) was compressed by 0.045-0.070 inch diametrically.

4.1.2 Sectioning

Cutting of the tube sections labeled 23B and 23C is shown in Figure 2, along with relative lengths and disposition of each piece.

A. Dual Etch Microstructures

Two samples for dual etch microstructure evaluation were obtained: one for piece 23B and one for piece 23C. The 2% Nital etch revealed the grain boundaries, while the orthophosphoric acid was used to determine presence and location of carbides. The results identified that the material had a typical mill annealed Alloy 600 microstructure.

B. Modified Huey

One piece from each of 23B and 23C was cut and tested using the modified Huey procedure. Specifically, the test pieces were exposed to boiling 25% nitric acid for 48 hours. After the exposure, the pieces were scrubbed and reweighed. The weight losses of 0.1% for each specimen indicated that the tube material was in the mill annealed condition. Mill annealed material typically exhibit weight losses of 0.5% or less, while sensitized material exhibit weight losses in excess of 5.0%.

C. Bulk Chemical Analysis of Tubing

Confirmation of the tubing as being Alloy 600 was being pursued through analysis of the base metal composition. One piece from each specimen, 23B and 23C, were chemically descaled using a nitric-hydrofluoric acid solution. After all activity was removed from the tubing, the pieces were submitted for chemical analysis. Results of the bulk chemical analysis of tube L29R84 are shown in Table 6 below, and compared with the SB-163 specified values. No discrepancies in the chemical composition of tube L29R84 were indicated.

Table 6

ELEMENTAL ANALYSIS
of
Tube L29R84

Ni	Cr	Fe	Mn	Cu	Si	C	S	Co	
72.0 (min)	14.00-17.00	6.00-10.00	1.0 (max)	0.5 (max)	0.5 (max)	0.15 (max)	0.015 (max)	-	SB-163
76.08	14.50	8.16	0.248	0.325	0.190	0.025	0.001	0.035	23B
76.04	14.44	8.26	0.245	0.324	0.195	0.024	0.001	0.035	23C

D. pH Measurements

Measurements of the pH of the residual deposits on the steam generator tubing were attempted with drops of deionized water and litmus paper. The litmus paper was capable of detecting pH's in the range of 9-12, with different colors at each .5 pH unit. The paper registered no reading (below 9) when wetted by deionized water.

Some of the deposits were removed from the tube surface and crushed to form a slurry. When the pH of the slurry was checked, no change in color of the litmus paper was registered. This suggests the pH was below 9.0.

Finally, drops of water were placed at several locations along section 23B. In general, the pH paper did not register any color change at these locations. However, one spot along section 23B did have a color change, suggesting a pH of 10.0.

4.1.3 Visual and SEM Inspection Results

A. Major Crack - Transverse Mount

One end of the "fishmouth" failure surface was mounted and polished using conventional metallographic techniques. It was subsequently etched using 2% Nital and later glyceregia. The metallographic examination revealed the presence of intergranular stress corrosion cracking (IGSCC). There was no evidence of the presence of a network of intergranular attack between the fissures.

B. Fracture Surface

One face of the fracture surface was removed from the tube surface and chemically cleaned using a two step APAC descaling procedure. The descaled specimen was then evaluated by scanning electron microscopy (SEM) to determine the relative amounts of IGSCC and ductile failure on the fracture surface.

Approximately 95% of the wall thickness exhibited a distinct intergranular appearance. Only a small amount of ductile tearing, approximately 5% of the wall thickness, was evident at the I.D. surface. The "fishmouth" fracture was most probably formed from a series of essentially throughwall axially oriented intergranular penetrations, followed by ductile tearing of the material between the penetrations and the remaining tube wall thickness. There was no evidence of tube wall thinning as a result of corrosion or plastic deformation.

C. Minor Crack

The piece from the smaller of the two cracks was cut, mounted, and polished "dry" to prevent the elution of contaminate species during specimen preparation. The intergranular nature of the cracks was apparent in the as-polished cross section. The bakelite mounting material penetrated several of the fissures, although the crack tips were free of bakelite.

SEM energy dispersive spectrometry failed to reveal the presence of chemical deposits, even in the regions of the crack tips, which are known to be capable of the production of IGSCC in Alloy 600. Concentration of species identified (i.e., potassium, sodium, sulfur) were at or near background levels. The small quantity of silicon detected is attributed to handling and mounting contamination. One small particle rich in copper was observed. Analyses of several areas around the crack tip region were completed. In general, only Ni, Cr, and Fe, typical of Alloy 600, were found. However, at one location weak indications of potassium and sulfur were present. X-ray dot mapping showed no indications of concentrations of these elements. In another area there were weak indications of calcium, chloride, copper, magnesium, and aluminum along with silica. No conclusions could be drawn regarding possible aggressive species that could promote intergranular stress corrosion cracking.

D. Scale Analysis

Scrapings of the deposits from tube L29R84 were removed from the scallop bar region from the free length of tubing. Only light deposits were present within the scallop bar region, adjacent to the large "fishmouth" fracture. Ion chromatography detected 1793 ppm SO_4 and 833 ppm NO_3 , although the error was ± 50% due to the sample size. Atomic absorption techniques did not detect metal cations such as potassium, sodium, calcium, or magnesium. The threshold detection level was 1050 ppm because of the limited sample size.

E. Residual Strain Analysis

Testing of the specimen to determine the strain in the failed tube was not included in the initial CE evaluation program. Potential analyses to determine this strain are being reviewed and will be conducted if they are within the capability of the CE laboratory.

4.1.4 Causative Mechanism

Laboratory analysis has identified OD initiated IGSCC as the cause of the steam generator tube failure at Fort Calhoun. The elements required for IGSCC include (a) a susceptible material condition, (b) a significant tensile stress, and (c) an agressive environment. All elements must be present for IGSCC to occur.

Stress corrosion cracking (SCC) of Alloy 600 will occur under the appropriate condition in all metallurgical conditions, including the "mill annealed" condition. Material used in the steam generator tubes at Fort Calhoun is typical of high temperature mill annealed Alloy 600. This material is resistant to IGSCC in some but not all environments.

Normal operating stresses in straight lengths of steam generator tubes are relatively low. Additional stresses may be imposed through support-tube interactions. At Fort Calhoun, there was evidence that the failed tube was constrained by the vertical support member to the extent that deformation of the tube occurred, probably the result of corrosion product build-up between the tube and vertical support. Deformation of this type will provide additional stress at the point where failure occurred.

Three different environments are capable of producing IGSCC in Alloy 600. These environments are (a) caustic (caustic stress corrosion cracking), (b) relatively pure water (Coriou stress corrosion cracking), and (c) sulfur containing environments. Of these environments, it has been determined that a caustic environment was the most likely cause of the observed failure.

A caustic environment may have occurred in steam blanketed areas at Fort Calhoun as a result of periodic low level condenser in-leakage. When concentrated, the cooling water (Missouri River) tends to become alkaline, thereby producing a caustic condition. Deposits in the steam blanketed area, which may have contained alkaline species such as Na, K, etc., may have redissolved during the plant shutdown prior to the failure. This could explain the absence of these elements in the small cracks adjacent to the failure. Although some deformation of the tube occurred during service, the total deformation was relatively small (less than one percent). Caustic SCC has been produced in the laboratory in Alloy 600 at strain levels as low as 0.5% (elastic plus plastic). Also, caustic SCC has occurred in relatively short times at temperatures of 600°F or less, which approximates the tube wall temperature at Fort Calhoun. These observations, coupled with the fact that the failures occurred in a steam blanketed area where caustic species could concentrate, leads to the conclusion that the failure was probably the result of caustic stress corrosion cracking.

It has been determined that pure water stress corrosion cracking (Coriou cracking) is significantly less likely a mechanism for the tube failure. Coriou cracking has been identified as the failure mechanism in numerous steam generator tube failures in both domestic and foreign PWRs. No chemical contaminate( ) have been associated

with this type of SCC. Field and laboratory failures attributed to this particular mechanism generally occur in either highly deformed tubes (strains greater than 14%) or in tubes with distinct mechanical and/or micro-structured characteristics (high strength and intragranular carbides). Most, although not all, of the Coriou type failures have been I.D. initiated.

Although the failed tube at Fort Calhoun was deformed, the total strain was relatively low (probably less than one percent). Furthermore, the tubing used at Fort Calhoun was relatively low strength and the microstructure was relatively resistant to Coriou cracking, i.e., intergranular carbides with few intragranular carbides. Similar tubing has been severely deformed as a result of support plate and/or eggcrate denting in other CE supplied steam generators. Non-destructive and post-service destructive examinations of removed tubes confirmed the absence of Coriou type cracking in these steam generators. Furthermore, Coriou cracking is strongly temperature dependent and thus tends to occur when temperature is the highest; i.e., on the tube I.D.. The Fort Calhoun failure was O.D. initiated.

IGSCC induced by a sulfur containing compound is the least likely of the three postulated failure mechanisms. There was no apparent source of S bearing compounds at Fort Calhoun, other than the condenser cooling water. Furthermore, the condenser cooling water becomes alkaline when concentrated, not acidic. The various forms of S induced corrosion (IGSCC, wastage, intergranular attack, pitting) all occur at acidic values of pH. In addition, analysis of the intergranular cracks in the failed tube did not produce evidence of the presence of S, although some S compounds (ex. NiS) that form in high temperature aqueous environments are insoluble.

4.1.5 Laboratory Analysis Conclusions

A. The failure was O.D. initiated intergranular stress corrosion cracking (IGSCC). There was no evidence of general intergranular attack.

B. The material, Alloy 600, is in the mill annealed condition, based on microstructural examination and modified Huey testing.

C. The tube was significantly ovalized. The tube diameter increased approximately 46 to 122 mils in the plane of the "fishmouth" fracture. At 90° rotation, the tube diameter was reduced by approximately 45 to 70 mils. There was no change in the nominal wall thickness.

D. Chemical species which could have caused the observed intergranular stress corrosion cracking were not identified during this examination.

E. The most probable causes of the intergranular stress corrosion cracking are ranked in the following order of relative probability:

1. Concentration of caustic species, possibly as a result of condenser cooling water in-leakage.

2. "Coriou" cracking in the secondary side AVT environment.

3. Sulfur-induced corrosion.

Concentration of caustic species is the most likely causative agent.

4.2 Steam Generator B - Tube L29R86 Section

The following describes the actions undertaken to perform a destructive examination of tube L29R86. A section of this tube was removed for access to the failed tube, L29R84.

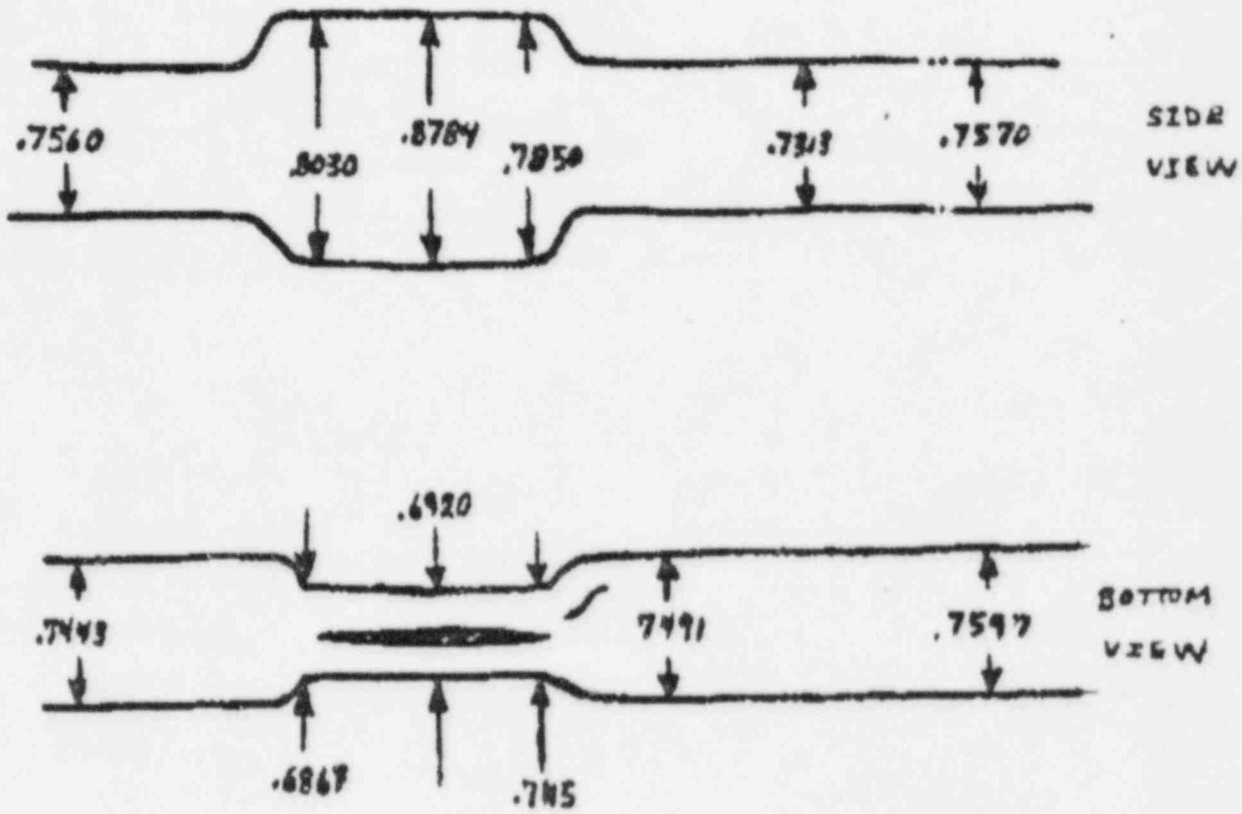
4.2.1 Receipt Inspections

Two additional tube sections labeled 13A and 13C from tube L29R86 were visually inspected after receipt in Windsor. A kink and other marks associated with the tube removal were present on the tube sections. Deposits were more abundant and homogeneous on these sections, compared to L29R84. There was no evidence of significant corrosion attack to either tube section.

A. Eddy Current Testing

Two small defect signals were detected on section 13A during the receipt inspection. However, tube deformations from the removal operations created interferences which made it impossible to accurately assess the nature of these two defects. The best interpretation of the data was that there were one or more kinks (creases) in the tube section as a result of deformation during the removal operation. (This was subsequently confirmed during the destructive examination).

This laboratory ECT data used MIZ-12 equipment which had been calibrated with an on-line standard. The mix frequencies were 400 and 100 KHz. A bobbin probe was used.



O.D. Dimensions in Inches

FIGURE 2. Dimensional Measurements of Tube L29R84

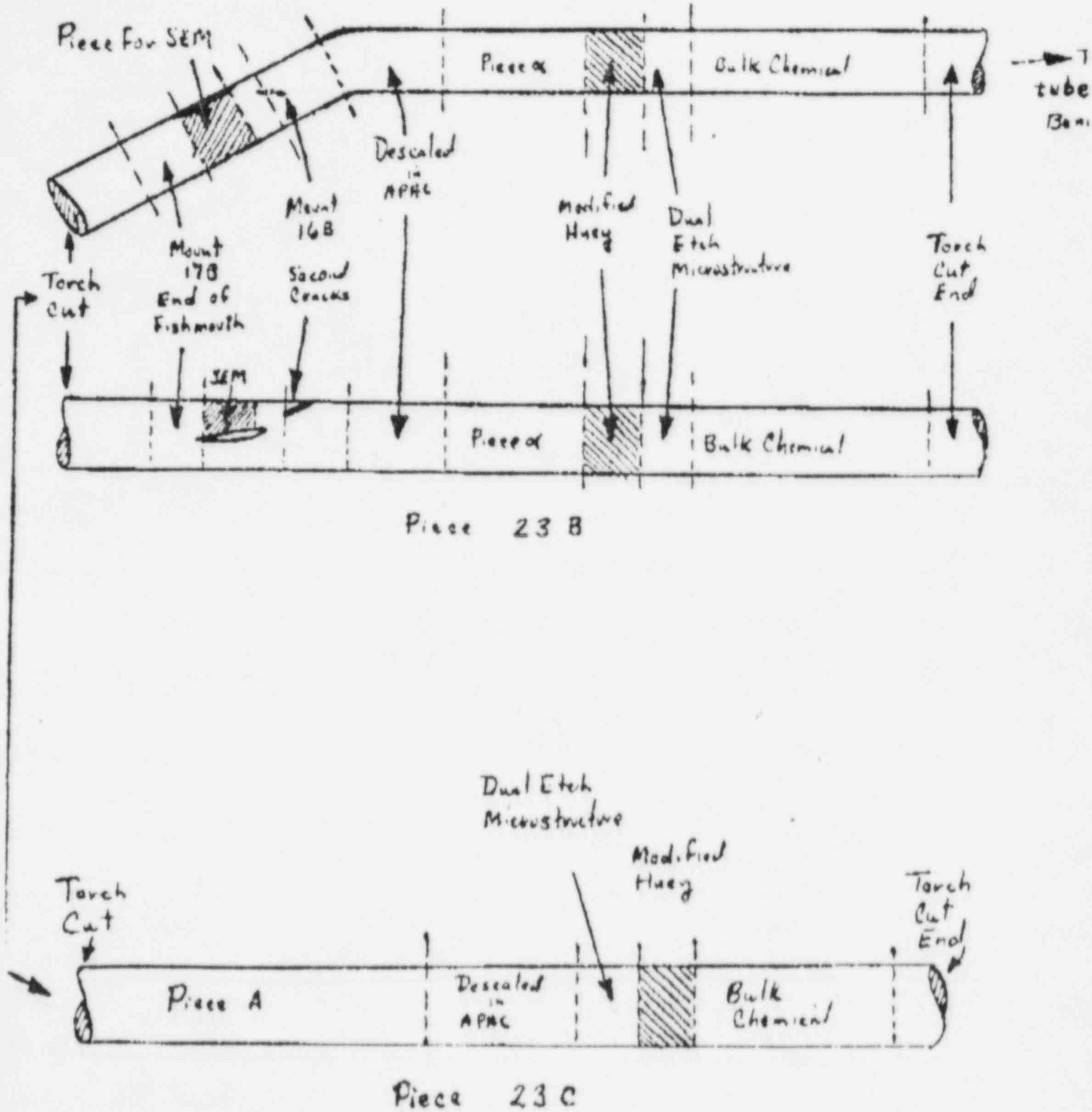


FIGURE 3. Sectioning Diagram of Tube L29R84

B. Visual Inspection

After eddy current testing was completed, the as-received condition of sections 13A and 13C was documented by video tape and macrophotography. The location of one scallop bar in section 13A was indicated by the presence of "rust" colored deposits. The location of the second scallop bar was not discernible. The crease in the tube, which occurred during the removal from the steam generator, also helped establish the orientation of the tube section within the steam generator.

C. Dimensional Measurements

Figure 4 illustrates the dimensional measurements taken for tube sections 13A and 13C from L29R86. These measurements were taken before descaling, and as such include the thickness of the residual deposits. The measurements suggested only slight ovalization of this tube. At the end of one scallop bar, the tube was indented .023 inches in the 3:00-9:00 position. In contrast, the removal of the tube caused a crease which reduced the 6:00-12:00 diametrical measurement by 0.033 inches and increased the 3:00-9:00 measurement by 0.009 inches.

4.2.2 Sectioning

Figure 5 illustrates the sectioning of tube section 13A to provide samples for subsequent examinations. These samples were removed for dual etch microstructure characterization, modified Huey testing, and surface examination. Prior to the sectioning, samples of the deposits were collected for chemical analysis. The locations of scrapings for chemical analysis are also illustrated in Figure 5.

Section 13C was not sectioned for any further destructive examination since laboratory eddy current testing did not indicate any flaws, and visual examination of the surface likewise did not identify any areas for further investigation. Only scrapings of the deposits for chemical analysis were made.

A. Dual Etch Microstructure

One specimen of L29R86 was selected for a dual etch microstructural evaluation. The 2% Nital etch revealed grain boundaries, while the orthophosphoric acid etch determined the presence and location of carbides. The grain size for this particular tube was noted to be significantly finer than observed

on L29R84. Extensive carbides were noted, particularly along the grain boundaries. This microstructure may have resulted from slow cooling during the fabrication process.

B. Modified Huey Testing

A single specimen from tube L29R86, piece 13A, was tested using a modified Huey procedure to assess the degree of sensitization. In this test, the weight loss of a specimen is determined after 48 hours exposure to 25% boiling nitric acid. Mill annealed material typically exhibits weight losses of 0.5% or less, while sensitized materials exhibit weight losses in excess of 5.0%. When the L29R86 specimen was tested, a weight loss of approximately 2.1% was recorded. In comparison, L29R84 had a weight loss of 0.1%. Although the weight loss was higher than for typical mill annealed tubing used in CE steam generators, it was not as great as expected for fully sensitized tubing. As indicated above, slow cooling during fabrication may have resulted in minor sensitization of the tube.

C. Visual Examination

One section from tube specimen 13A was chemically descaled using APAC, a standard two step descaling solution. The piece chosen was from the middle of one scallop bar to just past the kink in the tube, as shown in Figure 5. The purpose was to establish whether defects other than the kink/crease existed, and if so to correlate them to the previously observed eddy current indications. Once descaled, the specimen was examined at 5-40X under a microscope. Whereas only one crease was visible on the exterior surface of the tube, two creases were observed on the inside diameter. The shallower of the two creases corresponded to the location of the scallop bar. The distance between the two creases also matched the distance between the two eddy current signals. No other externally or internally visible defects were identified.

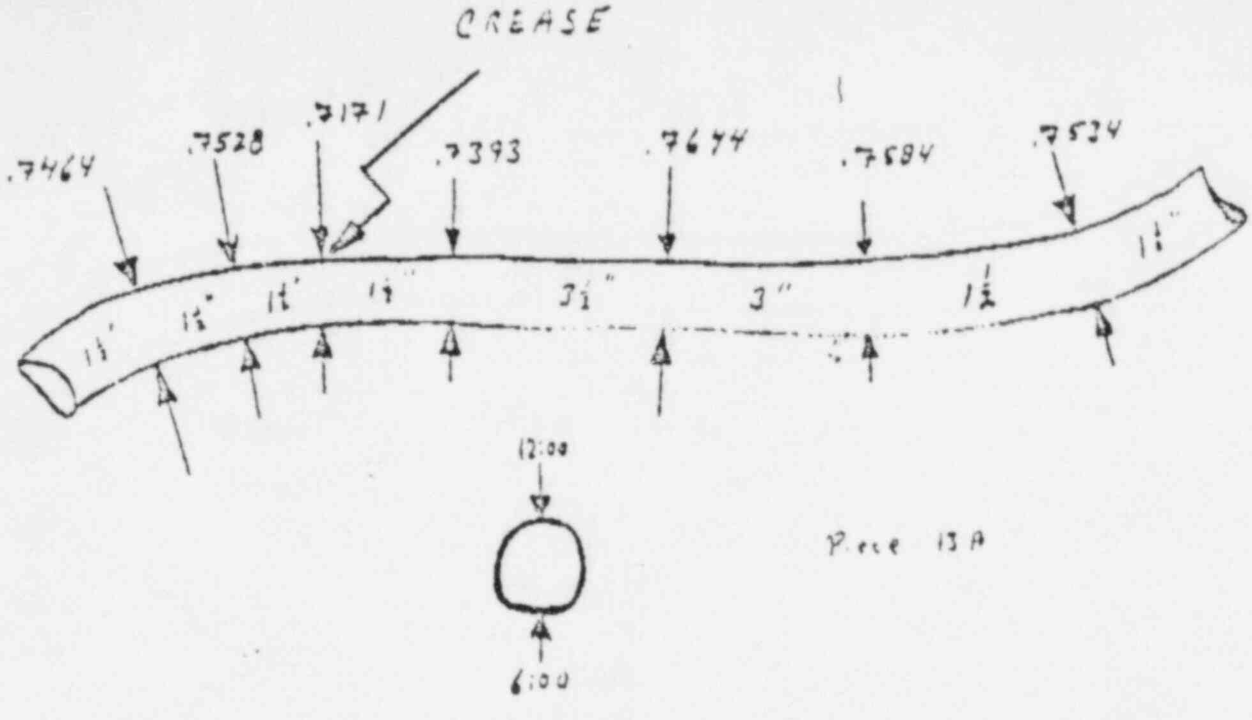
4.2.3 Additional Work in Progress

All easily removable deposits on tube specimen L29R86 were collected and labeled for analysis. Of particular interest were the scrapings at the location of the scallop bar for piece 13A. Analysis by several techniques were attempted. These include X-ray fluorescence, atomic absorption, and emission spectrophotometry. X-ray fluorescence was attempted to identify the presence of various species, including the alkaline metals. The presence of

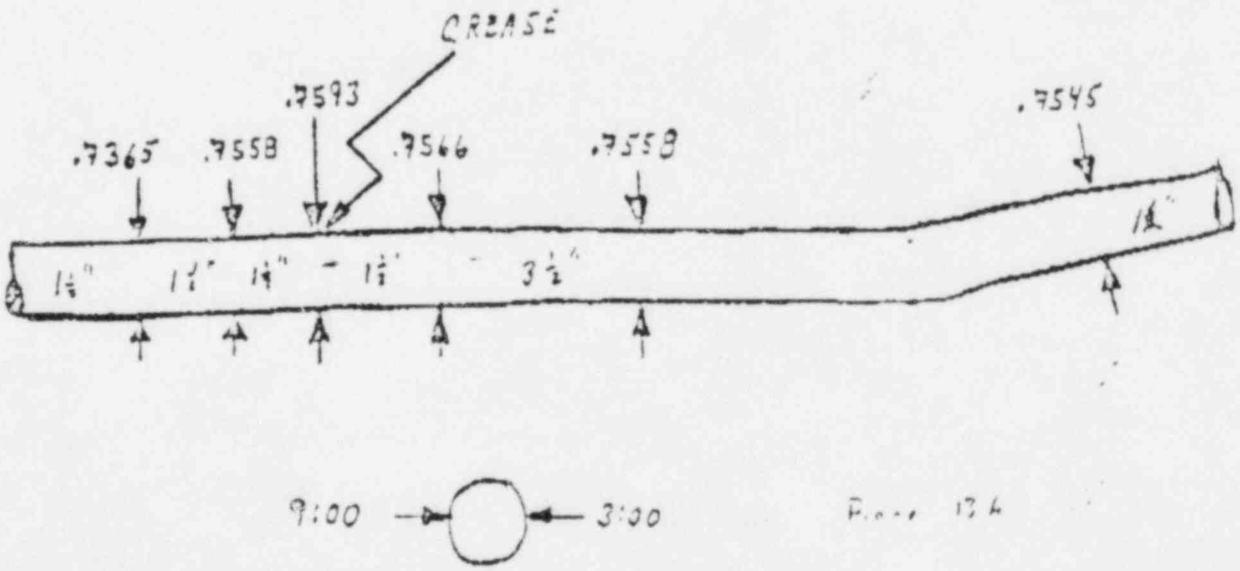
calcium, potassium and sulfur was suggested, in addition to firm indications of Fe, Cu and Ni, although the minimum detectable concentrations were unknown.

Numerous difficulties were encountered in trying to dissolve the deposit scrapings for tube L29R86. This is a necessary step prior to either wet chemical analysis or atomic absorption procedures. Thus far, the scrapings have not been dissolved in aqua regia, aqua regia with hydrogen peroxide, EDTA, or EDTA with hydrazine.

Emission spectrophotometry appears to offer the most promising method for analysis of the deposit scrapings. Presently, standards have been prepared, and data collection is in progress.



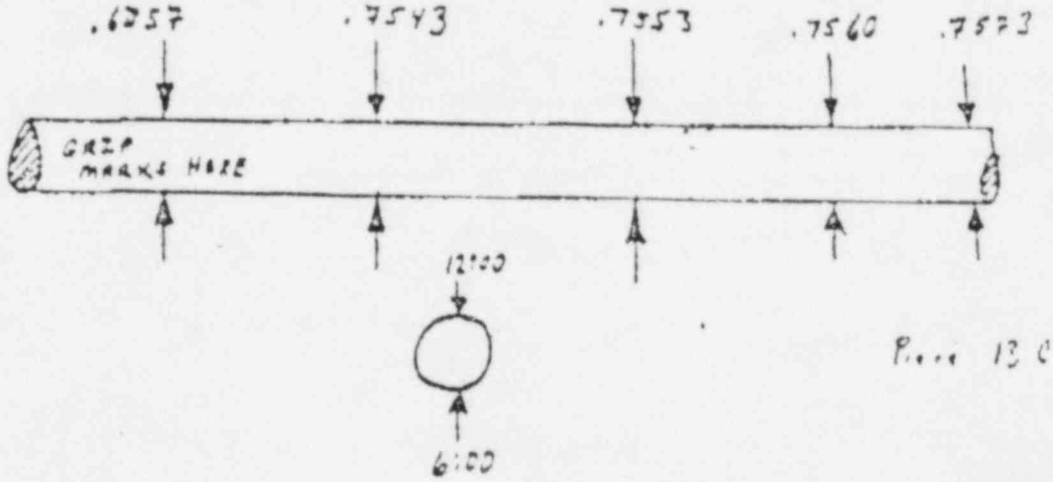
Position



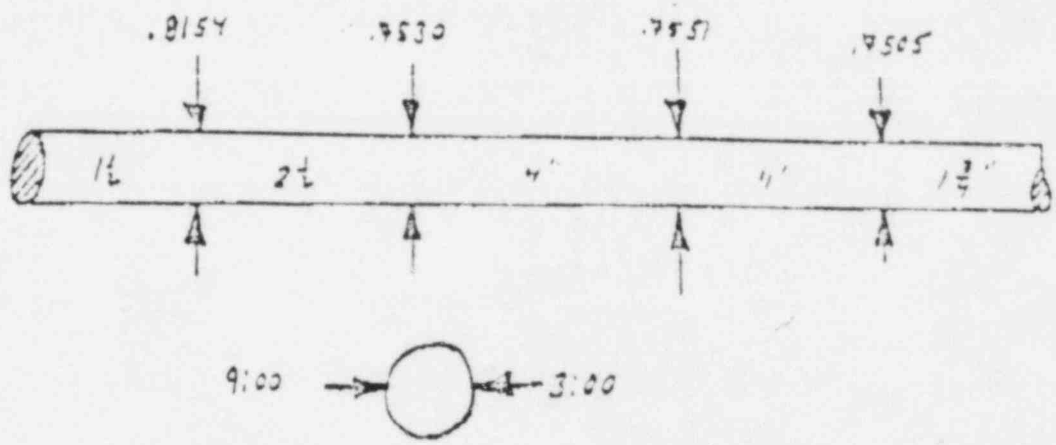
Position

Figure 4 - Dimensional Measurements of Tube L29R86, Section 13A

Dimensions are O.D. and in inches



Position



Position

Figure 4 (Continued) - Dimensional Measurements of Tube L29R86, Section 13C

Dimensions are in O.D. and in inches

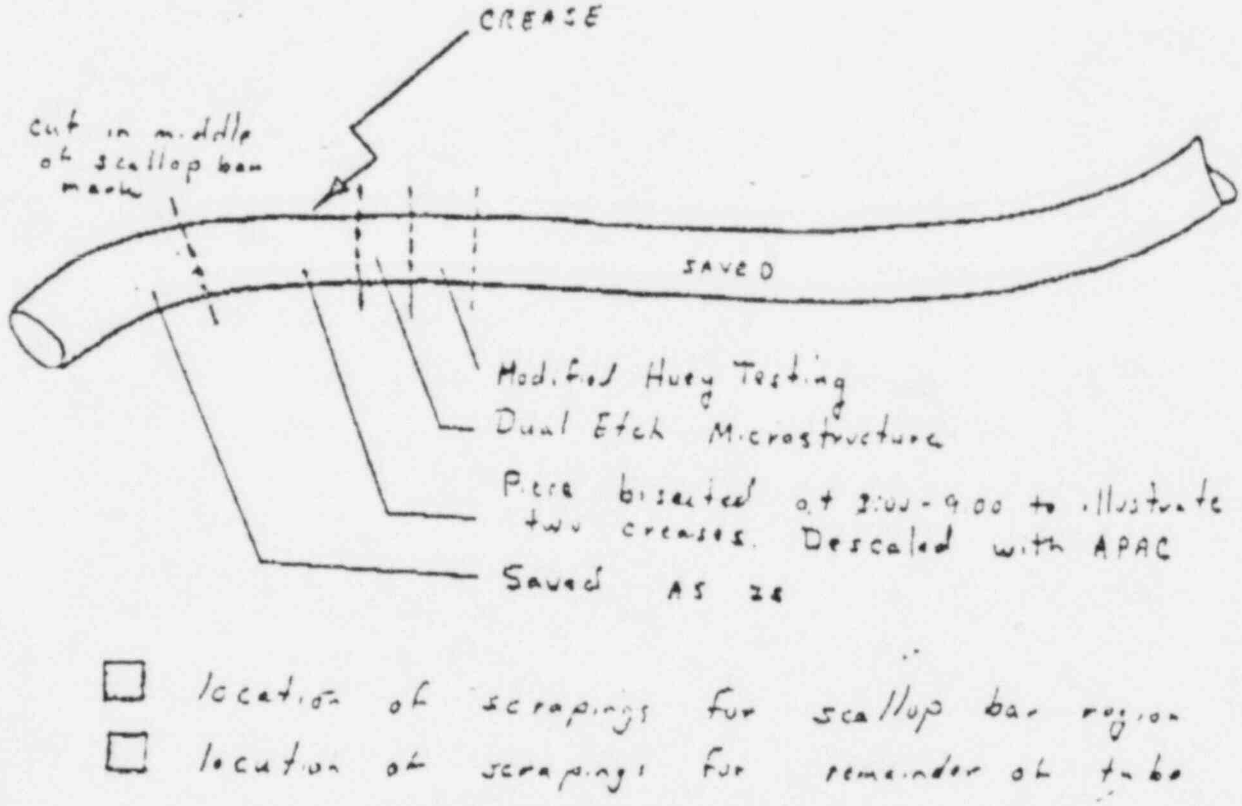


Figure 5 - Disposition of Pieces from Tube L29R86, Section 13A

5.0 OPERATION RELATED ACTIVITIES

In the event that any future steam generator tube leaks are experienced, several measures have been taken to provide additional assurance that leaks are detected early, that procedures are adequate and that licensed operations personnel are thoroughly trained in implementing these procedures. These measures are described in the following paragraphs.

5.1 Leakage Detection Improvements

An investigation of Fort Calhoun laboratory capabilities for determining primary-to-secondary leakage rates was conducted using station experience from several weeks of operation with a leak prior to the end of the last cycle (Cycle 8). Fort Calhoun Station gamma isotopic equipment is supplied by Canberra Industries and consists of Series 80 and Series 90 multi-channel analyzers connected to a DEC PDP/11/34 computer with Ge(Li) detectors. Typical 2000 second count times of 4 liter Marinelli beakers are used to obtain sensitivities and Lower Limits of Detectability necessary to support detection of low level leakage. All laboratory personnel have been properly trained in the use of the equipment and data evaluation and have gained considerable experience through daily demonstration of those skills.

It was determined using routine laboratory capabilities, typical reactor coolant boron and radionuclide concentrations and typical steam generator blowdown rates that the smallest leak detectable using boron in hot shutdown is 0.03 gpm and using Cs-137 in hot shutdown after refueling is 0.002 gpm. The leak prior to shutdown was detected at the 0.001 gpm rate. It was also determined that time to achieve a 90% equilibrium with typical blowdown is 1720 minutes and time to 10% equilibrium is 79 minutes. The sensitivity of analysis in the presence of typical operating short-lived fission products is such that a leak rate one to two orders of magnitude lower is possible.

5.2 Sampling Frequency Improvements

In order to assure early detection of leakage at low leakage rates, Fort Calhoun Station analytical frequency for gamma isotopic analysis of steam generator blowdown will be increased from weekly to daily. Boron analysis of blowdown will also be performed on a once per shift basis beginning in Mode 4 and continuing until 10 days after reaching Mode 1. New procedure CMP-4.68, "Steam Generator Primary-to-Secondary Leak Rate Determination," will be performed whenever activity or boron is detected. Steam generator blowdown monitors RM-054A and B will continue to be used to identify all but the smallest leaks during the intervals between sampling periods.

Special Order No. 35 entitled, "Allowable Primary-to-Secondary Reactor Coolant System Leak Rate" has been issued to establish an interim primary-to-secondary leakage limit through the steam generator tubes of 0.3 gpm total for both steam generators.

5.3 Procedure Reviews

In the District's letters of May 31, 1984, to the NRC, commitments were made to review the tube rupture emergency procedures to reconfirm adequacy, to provide refresher training on these emergency procedures prior to returning the plant to power operation, and to establish an interim primary-to-secondary leakage limit through the steam generator tubes of 0.3 gpm total for both steam generators.

The District was requested by the NRC to review the tube rupture emergency procedures using as guidance NUREG 0909, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at RE Ginna Nuclear Power Plant," Sections 9.0 and 10.0 and NUREG 0916, "Safety Evaluation Report related to the restart of RE Ginna Nuclear Power Plant," Sections 1.4.1, 1.4.2, 4.1, 4.2, 4.3 and 7.4.

The review team consisted of the following personnel. Reactor Engineer (SRO), two licensed operators (1SRO, 1RO) and a training coordinator (SRO). The completed procedure has been reviewed by the Plant Review Committee and the Reactor and Computer Technical Services Department.

The following procedures have been reviewed using the above listed NUREG sections:

1. EP-30, "Steam Generator Tube Rupture (PPLS Unblocked);"
2. EP-30A, "Steam Generator Tube Rupture (PPLS Blocked);"
3. OI-RC-11, "RCS Natural Circulation Cooldown;"
4. OP-6, "Hot Standby to Cold Shutdown;" and
5. EP-35, "Reset of Engineered Safeguards."

The existing procedures were found to adequately describe the appropriate operator action to be taken in the case of a steam generator tube rupture or leak. Emergency Procedures EP-30 and EP-30A have been revised to clarify and to improve the format of the procedure.

5.4 Licensed Operator Refresher Training

All licensed operating personnel will be given refresher training on these emergency procedures (EP-30 and EP-30A) prior to returning the plant to power operation. In the case of absence of any licensed operator due to illness or vacation, he will be trained prior to standing shift while the plant is at power operation.

As part of the regular training sequence, the Ginna tube rupture is currently being reviewed and discussed in detail. The Fort Calhoun steam generator tube failure event will also be reviewed.

6.0 CORRECTIVE ACTION TO REDUCE THE PROBABILITY OF THE FAILURE MECHANISM

The intergranular stress corrosion cracking (IGSCC) initiated on the outer diameter which lead to the tube failure was initiated by the simultaneous presence of three conditions:

A. A susceptible material condition

B. A significant tensile stress

C. An aggressive environment

(Refer to Section 4.1.4 for more information on IGSCC and the causative mechanism).

To reduce the likelihood of the simultaneous presence of these conditions, immediate corrective actions are being or have been taken to appropriately institute operational programs and techniques as briefly described below:

A. Material Condition

It may be possible to reduce the susceptibility of Alloy 600 through changes in the operating environment. The District will review available information to determine if changes in the physical operation of the station can provide an environment which is more resistant to IGSCC. This study will be completed in 6 months.

B. Tensile Stress

Development of procedures to arrest the dent growth rate is in progress. Results from industry experience with boron and hydrazine pacification treatments are being reviewed with other utilities and vendors of the Fort Calhoun turbine generator and NSSS. A chemistry program to arrest denting will be initiated upon completion of this review if it is determined that such a program will not produce undesirable secondary effects, e.g., increasing the likelihood of initiation of a turbine missile due to bucket or wheel failures.

During inspections this outage, profilometry data was collected to characterize dents and evaluate the strain in selected dented tubes in both steam generators. This data will be used in conjunction with repeated profilometry data from some of these same tubes to monitor the dent growth rate.

C. Chemical Environment

Several actions have been taken to control and monitor the chemical environment of the steam generator tubes:

1. Condenser Integrity Program

The chief source of impurities to the secondary system has been periodic low level condenser in-leakage. More restrictive administrative limits with the purpose of eliminating in-leakage during power operation are being adopted. In addition, programs involving surveillance of condenser tube degradation and failure mechanisms are being initiated. In conjunction with this assessment, the District will remove

several previously failed and plugged tubes from the condenser to determine the mechanisms leading to condenser tube failures for stainless steel tubing passing Missouri River water. ECT of condenser tubes is being considered to supplement the District's surveillance of condenser tube mechanical errosion of the inner diameter. ID wear has been monitored since 1974 and found to be nearly zero.

A survey of utility practices in condenser integrity is underway to assist the District in adopting practices to eliminate condenser in-leakage as a result of tube failure.

2. Chemistry Improvement Program

Fort Calhoun Station specifications for secondary system water quality have been reviewed and compared with those of Combustion Engineering. Specifications will be lowered for those parameters observed to be normally lower than the existing specifications. In many cases they will become equivalent to the CE specifications. For other parameters, the District has no measured chemical history (sulphates for example). In such cases, analytical data will be collected and specifications established reflecting the best water quality achievable with existing plant equipment. Additional analytical equipment of state-of-the-art sensitivity is being purchased to assure identification of trace contaminants. A program of corrective actions for varying levels of out-of-specification conditions is being established. District resources are being devoted to limit and correct mechanical malfunctions that could increase levels of secondary system contaminants. An aggressive program for system leak detection is being expanded. The District is expediting previously approved on-line monitoring instrument purchases for measurement of selected chemical parameters indicative of system leakage and is evaluating purchase of additional instrumentation.

3. Temperature Soaks During Heatups

Although the exact chemical causative agent or agents remain unknown, the District is reviewing operational procedures which will hold the steam generator temperatures at a plateau for a period of time to maximize secondary side chemical impurity soluability for blowdown removal and result in a reduction in impurity concentrations in crevices. This program will be in place for the coming plant heatup and assessed for its effectiveness prior to making it a standard practice during all heatups.

7.0 TUBE PLUGGING TO ELIMINATE SUSPECTED DEFECTS

In order to prevent reoccurrence of a steam generator tube rupture, all tubes showing an ECT indication in the hot leg vertical support region, which could be characteristic of OD intergranular cracking have been plugged, irregardless of the size of the indication. This is a total of seven tubes whose bobbin coil indications ranged from less than 20% to 42%. All of these indications were detected during ECT inspections using a frequency of 100 KHz mixed with other frequencies. Essentially all tubes in both steam generators have been tested using this frequency mix.

This precludes failure during operation of all tubes with detectable indications and may be presumed to be in a comparable category to the failed tube B-L29R84.

8.0 10 CFR 50.59 SAFETY EVALUATION

8.1 Purpose

The purpose of this evaluation is to demonstrate that an unreviewed safety question does not exist with respect to the operation of Fort Calhoun Station following the steam generator tube rupture event and the subsequent inspections and analysis of the Fort Calhoun steam generators.

8.2 Method

The method used to determine that an unreviewed safety question does not exist is to examine the three questions contained in 10 CFR 50.59 and to make a negative declaration with respect to each question.

8.3 10 CFR 50.59 - Unreviewed Safety Question Evaluation

8.3.1 Will the probability of occurrences or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report be increased?

A. Events Considered

This question relates to accidents or malfunctions of equipment important to safety evaluated in the safety analysis report. This safety evaluation will concentrate on accidents since the steam generator tube rupture event and subsequent inspections do not relate to any equipment important to safety and, therefore, no occurrences or consequences of malfunctions of equipment important to safety would result from this event and inspections.

The accidents from the Safety Analysis Report which must be considered in this evaluation are the steam generator tube rupture and the main steam line break which could lead to a release of radioactivity through the steam generators.

B. Steam Line Break

The probability of a main steam line break accident would not be affected by the integrity of the steam generator tubes; however, the consequences of a steam line break accident could be affected by the amount of primary-to-secondary leakage. Since the primary-to-secondary leakage limit has been administratively lowered on an interim basis to 0.3 gpm, the consequences of a steam line break are decreased relative to the steam line break radiological consequences analyzed in the Safety Analysis Report.

C. Consequences of Steam Generator Tube Rupture

The consequences of the steam generator tube rupture event will not be increased for the following reasons: The activity in primary and secondary systems assumed in the analysis of the steam generator tube rupture will be unchanged during future operation. The equipment available to deal with the steam generator tube rupture event is unchanged from that considered in the Safety Analysis Report. The operator action assumed in the Safety Analysis Report is unchanged and operator action in the future should be improved due to the review and improvements made in the District's Emergency Operating Procedures and operator training. Therefore, it is concluded that the future operation of Fort Calhoun Station will not change the consequences of the steam generator rupture as discussed in the Safety Analysis Report.

D. Probability of Steam Generator Tube Rupture

The probability of a future steam generator tube rupture occurring at Fort Calhoun Station is related to the cause of the defective tube, current condition of the steam generators, actions which can be taken to reduce the probability of tube degradation, actions which can be taken to detect the precursor of a steam generator tube rupture and actions which can be taken to correct problems associated with the precursor of a steam generator tube rupture.

The cause of the tube rupture at Fort Calhoun Station on May 16, 1984, has been identified as intergranular stress corrosion cracking (IGSCC). The most probable environment which produced this IGSCC is a caustic environment. There is evidence that the failed tube was constrained by the vertical support member to the extent that deformation of the tube occurred, probably the result of corrosion product build-up between the tube and vertical support. Deformation of this type would have provided additional stress at the point where failure occurred. The metalographic exams revealed that there had been no general chemical attack such as intergranular attack on the steam generator tubes which could lead to overall degradation of the steam generators. Section 4.0 details the metalographic examinations which were performed on both the failed and unfailed tubes.

Eddy current testing has been performed on 100% of the accessible tubes in both steam generators. The inspection showed there were only 0.13% degraded tubes and only 0.04% of the tubes contained a de-

fect. 0.42% of the tubes showed some form of imperfection. The profilometry testing revealed significant denting of exterior tubes in the vertical support straps. This denting could provide additional stress on tubes similar to that seen in the ruptured tube.

The eddy current examination revealed additional defective tubes which are summarized in Table 3 of this report.

The cause of the defect in the vertical support strap is postulated to be the IGSCC based on the defect's location and type of eddy current indication. The cause of the IGSCC has been previously discussed. The cause of the defect near the tube sheet is postulated to be sludge pile pitting similar to that which has been seen in other CE steam generators. This conclusion is based on the fact that defects and degradation seen near the tube sheet of other CE steam generators has been attributed to sludge pile pitting. The cause of the sludge pile pitting is thought to be a galvanic cell type of corrosion in which the free copper in the sludge pile plays a role. To further evaluate the cause of the defect near the tube sheet, the District has removed a section of a tube containing these defects from the steam generator and examination of this section is currently underway.

The District has plugged all tubes which exhibit an imperfection greater than or equal to 40% of the nominal tube wall thickness. The degraded tube which exhibited a 39% imperfection was also plugged. The District has chosen to conservatively plug any tubes which exhibit an imperfection, irrespective of depth, within the hot leg vertical support strap structure. The District is utilizing a mechanical plug which can be removed from the tube. In the future, the District may analyze those tubes with imperfections less than the plugging limit of 40% of the nominal wall thickness utilizing the criteria given in Regulatory Guide 1.121 and determine that all or a portion of the plugged tubes are acceptable for operation. At that time, the plugs may be removed from the acceptable tubes.

Table 5 in Section 3 of this report contains the tubes plugged in the Fort Calhoun Station Steam Generators. These plugged tubes are widely dispersed throughout the steam generators and will not adversely effect the thermal hydraulic and mechani-

cal performance of the steam generators. The assumption of the number of tubes plugged in both steam generators used in the Safety Analysis also remains valid.

Based on the extensive examinations and limited plugging of selected steam generator tubes, there is an extremely low probability of steam generator leakage and an even lower probability of a steam generator tube rupture upon startup of Fort Calhoun Station.

The District will conduct the standard reactor coolant system integrity testing in accordance with Technical Specification 3.4 prior to returning the unit to hot shutdown. This testing will confirm the integrity of the steam generators.

The District will take actions which will reduce the probability of future intergranular stress corrosion cracking. A soak of the steam generators at an intermediate temperature will be performed during the upcoming heatup to remove the maximum amount of chemical species which may be involved in the intergranular stress corrosion cracking of the tubes from the steam generators. The temperature is chosen such that the maximum solubility of these species is obtained.

The District will take the necessary action to assure that concentrations of chemical species identified as environmentally causative agents in IGSCC will be maintained within the limits recommended by the steam generator manufacturer consistent with the current design capabilities of the Station. This includes a program to improve condenser performance to reduce raw water inleakage to a minimum level. In addition, the District is studying a boric acid neutralization of the denting process in the Fort Calhoun steam generators. This neutralization will be initiated once it is determined that the addition of boric acid to the steam will not significantly degrade the turbine rotor and will not increase the probability of the turbine rotor failure.

The District has taken and will take several corrective actions which should reduce sludge pile pitting if it is occurring in the Fort Calhoun steam generators. During the 1984 refueling outage, the District hydrolanced the sludge pile regions of both steam generators to remove the sludge pile material. The temperature soak of the steam generators during heatup will also help remove active chemical species from the sludge pile region. Finally, the District intends to remove the primary source of copper from the feedwater train.

The District has implemented programs which will increase the ability to detect a primary-to-secondary side leak, which is a potential precursor of a steam generator tube rupture. In our letter of May 31, 1984, the District committed to revising the operating manual to reflect an interim primary-to-econdary leakage of 0.3 gpm total for both steam generators, as opposed to the existing Technical Specification limit of 1.0 gpm. If the 0.3 gpm leakage limit is exceeded, the action required by Technical Specification 2.1.4 Paragraph 3 will be followed. Section 5.0 of this report details the District's secondary side chemistry program that will be undertaken upon startup. The program increases the frequency of testing for primary to secondary leakage and discusses the sensitivity of this program. This program can detect a leak of 0.002 gpm.

The District has undertaken a massive campaign to insure that there is no increase in the probability of a steam generator tube rupture at Fort Calhoun Station. The District has determined that the steam generators can be returned to service with an extremely low probability of a tube rupture during normal or transient conditions. The District has committed to taking actions to reduce any identified chemical and mechanical degradation of the steam generator tubes. The District will be able to detect a very small primary-to-secondary side leak and will take the appropriate action if such a leak occurs. Therefore, it is concluded that the probability of a steam generator tube rupture occurring is not increased.

8.3.2 Will the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report be created?

To answer this question, the District has surveyed the accidents currently analyzed in the Safety Analysis Report and multiple event scenarios which have been used for the generation of Emergency Operating Procedures and in probabilistic safety analyses. The two events of a different type than any evaluated previously in the Safety Analysis Report with the highest possibility of occurrence are a multiple steam generator tube rupture event, either rupturing more than one tube in a single steam generator, or rupturing tubes in both steam generators, and a steam line break with a concurrent steam generator tube rupture. The possibility of either of these events occurring upon startup is extremely low because the integrity of essentially all tubes is ensured due to the extensive eddy current testing and limited tube plugging.

During the next operating cycle, the probability of these events is extremely low because the cause of the steam generator tube rupture, intergranular stress corrosion cracking, is not indicative of an overall chemical or mechanical weakening of the steam generator tubes. The District is committed to taking corrective actions to reduce the chemical and mechanical causative agents of intergranular stress corrosion cracking and sludge pile pitting. In addition, the District has undertaken a program which will identify the potential precursor of these events and will take action prior to significant tube degradation.

The District concludes that the actions which have been taken to reduce the probability of intergranular stress corrosion cracking and sludge pile pitting and to detect a degraded tube, will prevent the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

8.3.4 <u>Will the margin of safety as defined in the basis of any Technical Specification be reduced?</u>

The applicable Technical Specifications are Technical Specification 2.1.4, primary-to-secondary leakage, Technical Specification 2.9, radioactive material release, Technical Specification 2.20, steam generator coolant radioactivity, and Technical Specification 3.3, steam generator tube inservice inspection program.

The margin to safety in Technical Specification 2.1.4 will be increased because of an interim limit on primary to secondary side leakage of 0.3 gpm. The margin of safety in Technical Specification 2.9 is unchanged because none of the radiological release limits are changed. The margin of safety of Technical Specification 2.20 is unchanged because the radiological limits on the steam generator coolant radioactivity is unchanged. The margin of safety of Technical Specification 3.3 is increased because the District has undertaken an inspection of 100% of all accessible tubes in both steam generators at the Fort Calhoun Station.

The District concludes that the margin of safety as defined in the basis for any Technical Specification will not be reduced and in two instances will be increased.

8.4 Conclusions

The District concludes that the actions described in the previous paragraphs will assure that there is not an unreviewed safety question associated with the restart of Fort Calhoun Station and its subsequent operation.