Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

May 31, 1984
LIC-84-159

Mr. James R. Miller, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Miller:

Steam Generator Tube Incident

The purpose of this letter and the attachments is to document information, commitments, and plans which were presented to the Commission during a meeting on May 29, 1984 concerning the Fort Calhoun Station's steam generator tube failure incident and inspections.

Attachment 1 contains information relating to selected plant parameters and plant status immediately prior to, during, and following a "B" steam generator tube incident on May 16, 1984.

Attachment 2 contains information relating to the steam generator tube inspections for the Fort Calhoun Station's steam generators. Data is provided for the inspection history of the steam generators, including the inspections performed during the 1984 refueling outage.

Attachment 3 contains information relating to the results of laboratory examinations performed on the section of the failed tube removed from "B" steam generator.

The above referenced Attachments 2 and 3 document a very comprehensive and thorough inspection and examination program which has been and is being conducted in order to establish a high level of confidence that the Fort Calhoun Station can be safely returned to service.

As discussed during the May 29, 1984 meeting, the District's current plans are as follows:

(1) Complete the inspection of approximately 300 tubes in "B" steam generator using the 1 x 8 and/or the 4 x 4 pancake array probes.

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(2) Analyze the data from the 1 x 8 and/or 4 x 4 probes inspection.

(3) Complete the examination of at least 3000 tubes in the hot leg side of "B" steam generator using the bobbin coil probes. (Subsequent to the May 29, 1984 meeting, the District has committed to completing the examination of all accessible hot leg side tubes in "B" steam generator using the bobbin coil probes.)

(4) Complete the analysis of the data from the bobbin coil probes testing referenced in item (3).

(5) Complete the re-review of the data from "A" and "B" steam generator tube inspections conducted in March, 1984.

(6) Continue laboratory examinations of the removed section of the "B" steam generator tube. The final report of these examinations will be submitted by June 30, 1984.

In the absence of any indication of significant flaws resulting from items (1) and (2) above, the District will begin reactor coolant system heatup in preparation for a reactor coolant system leak test at approximately 2200 psig. The reactor coolant system will be heated to a temperature of approximately 400°F for the test. After the satisfactory completion of the leak test, plant startup will continue. After items (4) and (5) are completed and in the absence of any indications of significant flaws, the plant will be returned to power operation. In the event any indications of significant flaws are identified, your office will be notified and a re-evaluation of the District's plans will be conducted.

Prior to returning the plant to power operation, the steam generator tube rupture emergency procedure will be reviewed to re-confirm adequacy and licensed operating personnel will receive refresher training on this emergency procedure. This review and training will assure that operating personnel maintain a high level of proficiency on emergency procedures, as demonstrated during the May 16, 1984 steam generator tube incident.

The Fort Calhoun Station operating manual will be revised to reflect an interim primary-to-secondary leakage through the steam generator tubes of 0.3 gpm total for both steam generators, as opposed to the existing Technical Specification limit of 1.0 gpm. If this leakage limit is exceeded, the action required by Technical Specification 2.1.4, paragraph (3), will be followed. This interim limit will serve to initiate corrective measures in a more timely manner in the unlikely event of additional steam generator tube leaks. In addition, the frequency of secondary side chemistry analyses related to detection of primary-to-secondary system leakage will be increased.

Upon completion of the items discussed above, the District will have taken reasonable and practical action to assure the continued safe operation of the Fort Calhoun Station and will have more than satisfied the station's Technical Specification and License requirements.

Sincerely,

W C Jones

W. C. Jones
Division Manager
Production Operations

WCJ/KJM:jmm

Attachments

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

Mr. E. G. Tourigny, Project Manager
Mr. L. A. Yandell, Senior Resident Inspector

ATTACHMENT 1

INITIAL CONDITIONS AND
TIME SEQUENCE RELATING
TO LEVEL AND PRESSURE

Initial Conditions - May 16, 1984

Plant heat up following core refueling

RCS boron approximately 2100 ppm

T_C = 398°F

Pressurizer level = 70%

Pressurizer pressure = Increasing

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 (40 gpm)

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

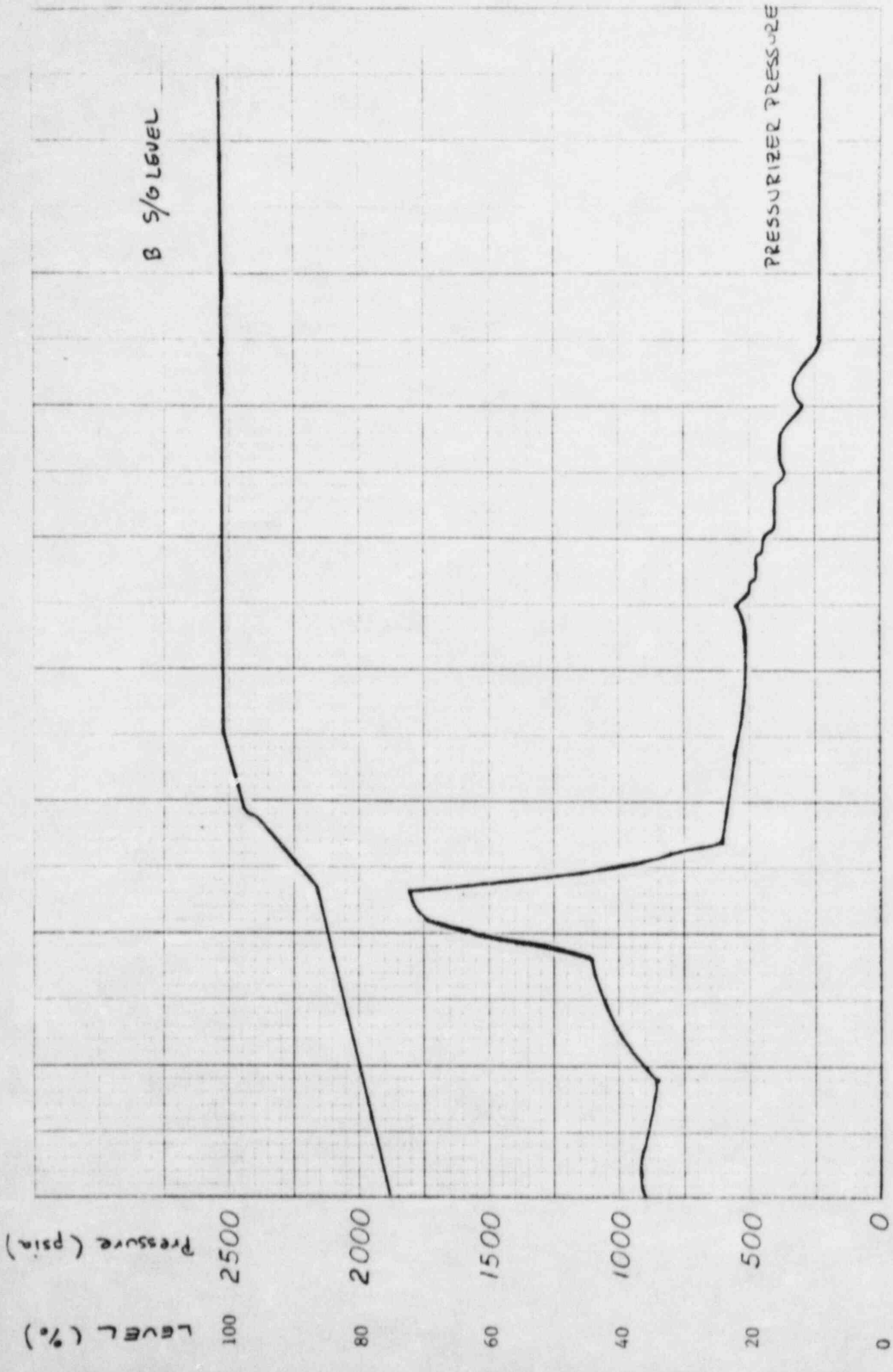
Letdown on minimum

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-1040, open slightly.


B S/G LEVEL
PRESSURIZER PRESSURE
TIME
1600
1620
1640
1700
1720
1740
1800
Pressure (psia)
2500
2000
1500
1000
500
0
LEVEL (%)
100
80
60
40
20
0

ATTACHMENT 2

STEAM GENERATOR INSPECTIONS

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 always operated with a carefully maintained AVT secondary chemistry program. The periodic inspections 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 of the steam generator tubes have shown the generators to be in Technical Specification Category C-1 except for the present inspection, which is Category C-2 due to the single failed tube in the "B" generator.

A pre-operational baseline inspection of 200 tubes per steam generator was performed in July 1973. Some mechanical imperfections were noted in the "A" generator. These are possibly the result of the generator being dropped several inches during erection.

225 tubes in each steam generator were 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 inspection of the "A" steam generator in November 1977 was performed in order to assess the imperfection indications which had been discovered previously. 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. 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. These two tubes which were plugged only on one end were reexamined in 1984; the indications had not changed, and the tubes were left in their present condition).

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. 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. 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.

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 St. Lucie 1 and Maine Yankee; 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. Analysis of this data is still in progress.

Following the tube rupture event, which occurred during a hydrostatic test as a normal part of the heatup process, all of the March 1984 inspection data has been reanalyzed. The results of the reanalysis have shown a 99% through-wall defect in the failed tube which was missed by the data analyst during the first evaluation. The reanalysis has also shown one tube in steam generator "A" which has been reevaluated as having degradation which is marginally greater than the reporting limit of 20%.

Following discovery of the leaking tube, 38 tubes around the leak area were examined for evidence of steam erosions or other indications resulting from the tube failure. No indications were found. In addition to this effort, all tubes which are accessible from the hot leg side of the steam generator and which were not previously inspected during 1984 have now been tested using a standard bobbin coil probe. Also, a minimum of 300 tubes will be inspected utilizing 1 x 8 and/or 4 x 4 pancake arrays. The analysis of the data from these inspections is in progress. Also, 156 of the 300 tubes were examined utilizing 1 x 8 superflex profilometry to characterize the denting in the vertical batwing straps. Analysis of this data is also in progress.

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.

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. Unfortunately, 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.

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.

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 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. 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.

The failed tube 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 inspection shows a 99% through-wall defect at the location of the failure. This indication was missed on initial evaluation due to human error.

Since the failed portion of the tube was reasonably accessible, the District and Combustion Engineering 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.

ATTACHMENT 3

VISUAL INSPECTION

AND LABORATORY ANALYSIS

The following describes the activities undertaken during the period of May 26-28, 1984 to perform a destructive examination of tube L29R84 from the Fort Calhoun "B" steam generator.

I. RECEIPT INSPECTION

Upon receipt, the two tube specimens labeled 23B and 23C were visually inspected. Two cracks were observed on piece 23B. The first was a large, axial (1¼") "fishmouth" type crack, while the second was a series of small (approximately ¼") 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. Eddy Current Testing

The 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 the 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 ¼ 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 in-steam generator ECT inspection data, wherein two defect signals approximately ¼" 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 1 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.

II. 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 is 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 chemical analysis are expected shortly.

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.

III. 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. No conclusions could be drawn regarding possible aggressive species that could promote intergranular stress corrosion cracking.

IV. 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 failure are intergranular stress corrosion cracking as a result of concentration of caustic species, from condenser cooling water in-leakage, or "Coriou" cracking in the secondary side AVT environment.

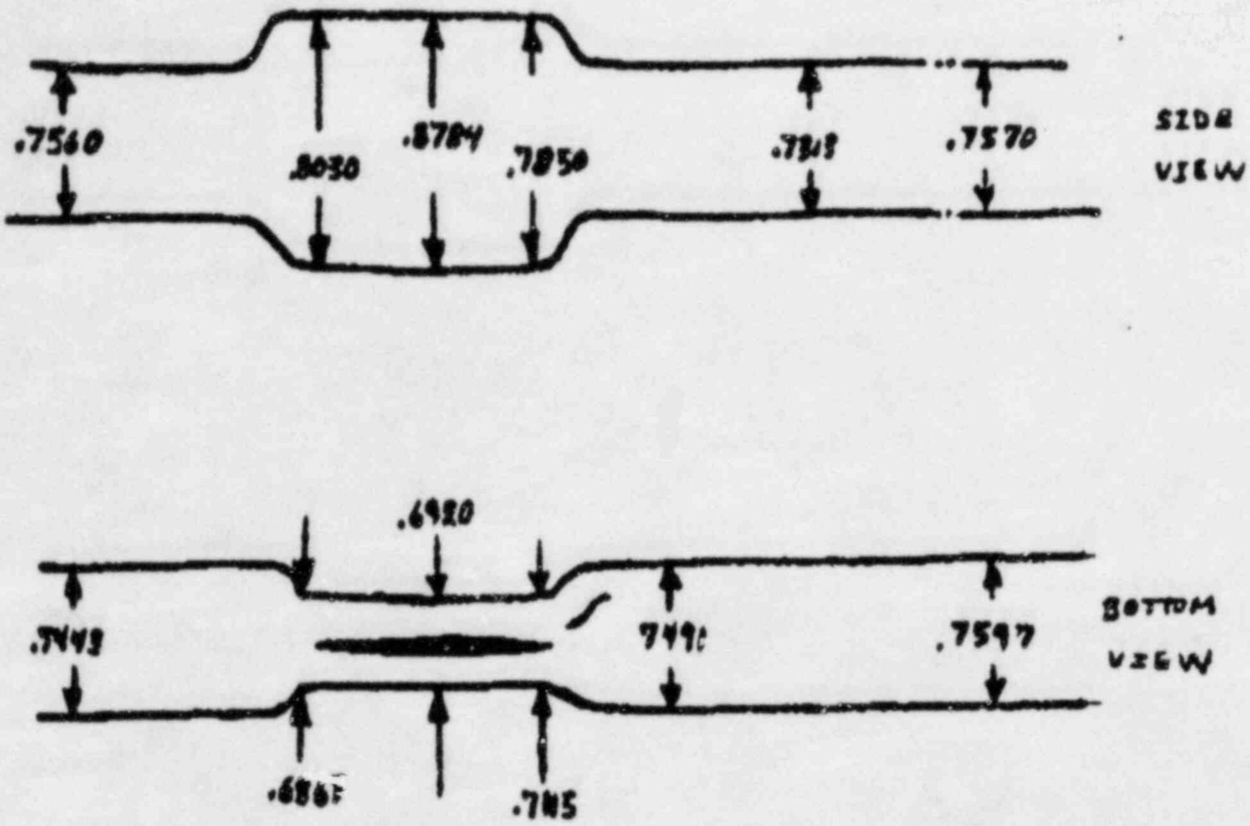


FIGURE 1. Dimensional Measurements

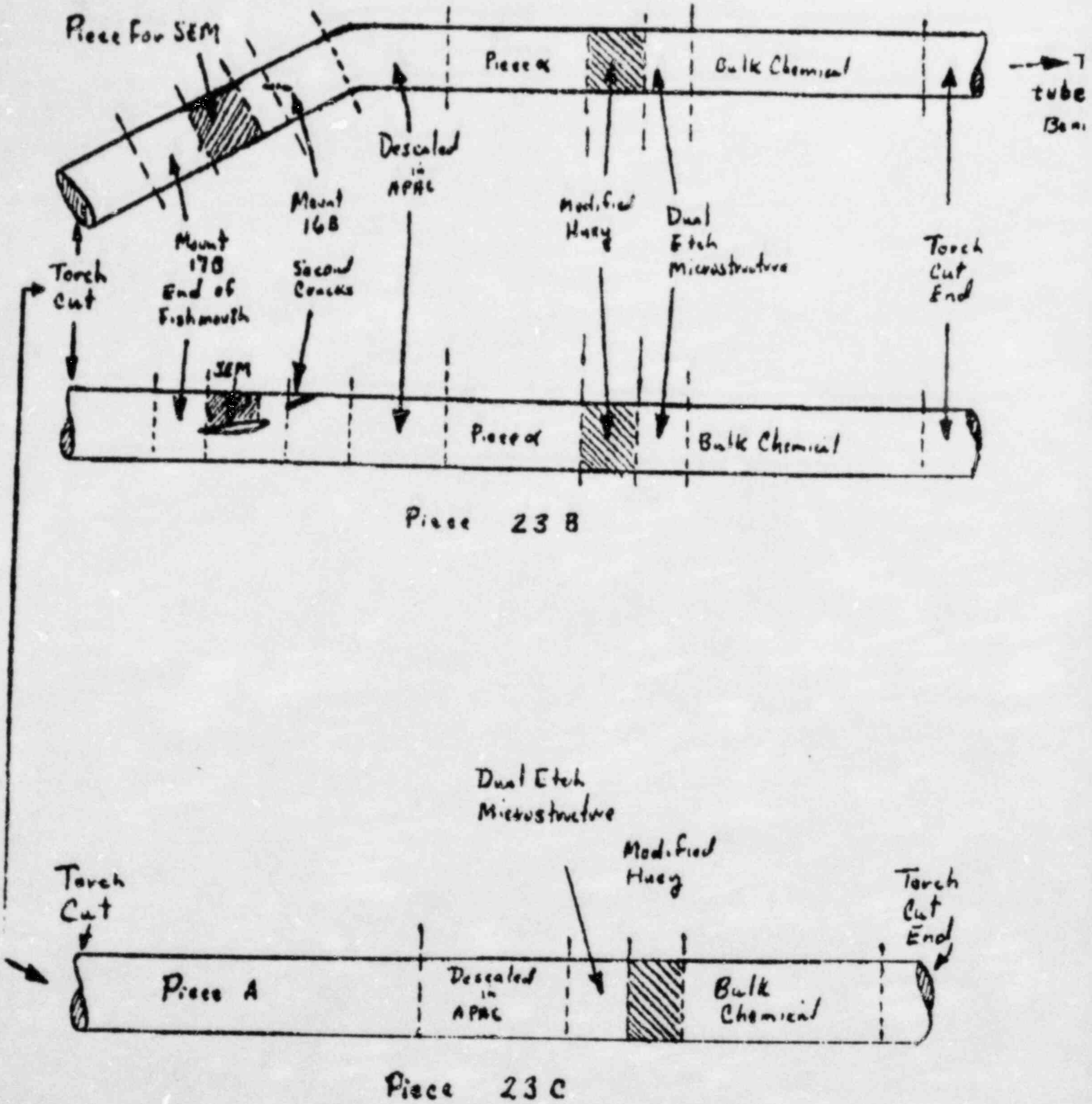


FIGURE 2. Sectioning Diagram