37
3-2-78

Spec of Dist

WAPD-TM-1309

DOE Research and Development Report

MASTER

Shippingport Atomic Power Station Steam Generator Tube Damage and Water Chemistry Control (1965 – 1975)

Bettis Atomic Power Laboratory
West Mifflin, Pennsylvania 15122

January 1978

Prepared for the
U.S. Department of Energy
By Westinghouse Electric Corporation
Under Contract No. EY-76-C-11-0014

DISCLAIMER

WAPD-TM-1309

SHIPPINGPORT ATOMIC POWER STATION
STEAM GENERATOR TUBE DAMAGE
AND
WATER CHEMISTRY CONTROL
(1965-1975)

W. J. Singley

January 1978

Contract No. EY-76-C-11-0014

Printed in the United States of America
Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

NOTE

This document is an interim memorandum prepared primarily for internal reference and does not represent a final expression of the opinion of Westinghouse. When this memorandum is distributed externally, it is with the express understanding that Westinghouse makes no representation as to completeness, accuracy, or usability of information contained therein.

BETTIS ATOMIC POWER LABORATORY WEST MIFFLIN, PENNSYLVANIA

Operated for the U. S. Department of Energy by
WESTINGHOUSE ELECTRIC CORPORATION

NOTICE

ABSTRACT

The four stainless steel tubed steam generators in the Shippingport Atomic Power Station were replaced in 1964 with larger, horizontal, NiCrFe Alloy 600 tubed units consistent with a power uprating of the plant. Each of the four Alloy 600 tubed units experienced tube leakage attributed to corrosion by water treatment chemicals (sodium phosphates) which concentrated locally on the tubes in inadequately flushed crevices. Corrective and preventive actions included tube plugging, flow blockage, conversion to all-volatile (hydrazine-morpholine) chemistry and replacement of the two steam generators with the straight tube design. Eddy current inspection after four years of all-volatile chemistry use showed a diminished rate of crevice corrosion and no tube denting.

INTRODUCTION

Background

The Shippingport Atomic Power Station, a pressurized water reactor plant, commenced power operations in December 1957, utilizing four steam generators tubed with Type 304 stainless steel. Tube leakage occurred in the original 1B stainless steel tubed steam generator following the initial operating period. The leakage was caused by stress corrosion cracking attributed(1) to the inadvertent presence of low levels (up to 10 ppm) of free caustic in the boiler water and local steam blanketed regions which permitted concentration of the caustic. Leaking tubes were plugged, additional steam risers were added, and operations continued with the stainless steel tubed steam generators until they were replaced in 1964 with larger units as part of plant refueling and power uprating. No additional leaks occurred in the period between 1958 and 1964, although examination of tubes from the removed 1B steam generator showed that there was extensive stress corrosion cracking. The post removal examination confirmed the prior conclusion that the stress corrosion cracking was attributed to the presence of free caustic during the initial operating period in 1957(2).

In conjunction with installation of a new core design in 1964, the stainless steel steam generators were replaced with four, larger units containing NiCrFe Alloy 600 tubes to accommodate an increased power rating for the Station. Plant operations resumed in 1965 and continued using the replacement steam generators until 1974 when the plant was shut down for modifications in support of the Light Water Breeder Reactor. During the plant shutdown, the two straight tubed heat exchangers were replaced with available spare units of identical design. Power operations with the Light Water Breeder Reactor were resumed in September 1977.

During the period 1965-1971, steam generator tube leakage occurred in each of the four NiCrFe Alloy 600 tubed replacement units. The history of tube leakage, the causes of the tube damage, the effects of secondary water chemistry and steam generator design, and corrective and preventive actions are described in this paper.

Steam Generator Design

The four replacement Alloy 600 tubed steam generators installed in 1964 were of two types. The major characteristics of each type are:

Steam Generator Designation	Tube Bundle Orientation	Tube Bundle Type	Support Plate Hole Geometry
1A, 1D	Horizontal	Straight-Tube	Cylindrical
1B, 1C	Horizontal	U-Bend	Hourglass (Double Chamfer - see Fig. 2)

The arrangement of the 1A and 1D straight tube units is shown schematically in Figure 1. The arrangement for the 1B and 1C steam generators is similar, except that the tubes are U-shaped. All steam generators are of the natural recirculation type and have separate steam drums with external risers and downcomers. Design parameters for the steam generators are given in Table I. An important difference in the design of the two types of steam generators is the geometry of the annular crevice formed between the support plate hole and the tube. The support plate crevice region designs for the two types of units are shown in Figure 2. Crevices and flow blocked regions are formed between the tubes of the U-bend steam generators (1B and 1C) and the loop end support straps, Figure 3.

Plant Operating History

The nominal plant operating conditions are given in Table I. Variations in primary plant temperature which occurred, primarily for test purposes, were as follows:

Period	Ave. Temp. °F
Nov. 1964-Aug. 1966	536
To September 1966	500
To January 1969	475
To February 1974	536

Variations in plant average power level also occurred because of changes in power demands. These average power level variations are summarized below:

Period	Plant Average Power Level	Approximate Equivalent Full Power Hours (EFPH) In Period
April 1965-Early 1968	45%	11,200
To March 1969	35%	2,400
To July 1970	40%	3,400
To May 1971	35%	2,000
October 1972 to 1974	20%*	4,200

Secondary Water Chemistry Control

Secondary water chemistry control for the Alloy 600 tubed steam generators, from their initial operations in April 1965 to August 1966, was maintained with coordinated phosphate plus hydrazine and morpholine(3). The chemistry specifications were:

Boiler Water	Specification	Chemical Additive
Phosphate	100 to 300 ppm	Disodium phosphate
pH	10.6 to 11.0	Trisodium phosphate
Chloride	0.5 ppm, max.	
Silica	25 ppm, max.	
Total Dissolved Solids	1000 ppm, max.	
Condensate		
pH	8.8 to 9.3	Morpholine
Conductivity	Less than 15 μmhos/cm.	
Chloride	0.3 ppm, max.	
Ammonia	2.0 ppm, max.	
Feedwater		
Hydrazine	1.5 x O_2 conc. (0.005 ppm min. O_2 conc. assumed)	Hydrazine

*During the period of low (20% average) power, the plant was operated normally at self sustaining power with power increased for about 14 hours each day to 40-67% and then reduced to self-sustaining power.

In order to minimize solids accumulation in the steam generators, the boiler water specifications were changed in August 1966 to reduce the phosphate concentration to a range of 5 to 100 ppm, and to use a pH of 9.5 or greater. A low level of phosphate was maintained in the boiler water to provide a buffer in the event of a condenser leak. pH was maintained at least 0.1 units below the Whirl-Purcell curve for trisodium phosphate to avoid free caustic(4). An additional specification was added for full wet layup to provide for a hydrazine residual of 50 to 100 ppm in the boiler water, and pH of 10.0 to 11.0. In April 1970 the minimum phosphate level was increased from 5 to 10 ppm, and the margin to the Whirl-Purcell curve was increased from 0.1 to 0.3 pH units. Chemistry control was maintained by this method until May 1971, when phosphate addition was discontinued because it was considered to be a contributor to steam generator tube corrosion within the cylindrical, drilled support plate holes in the straight tube units.

The control of secondary chemistry during the period of phosphate treatment was good. Sodium to phosphate molar ratios averaged 2.3 to 2.4 in the boiler water. The sodium to phosphate ratio of the treatment chemicals added was about 2.7. Occasional excursions of the Na/PO_4 ratio in the boiler water to values of 3.0 were observed, but these were quickly corrected. There was no occurrence of major condenser cooling water in-leakage. Immediate action was taken to correct the infrequent higher than specification chloride concentrations in the boiler water. Morpholine and hydrazine were continuously injected, and the morpholine concentration in the condensate ranged from 0.5 to 3.0 ppm. Hydrazine concentrations were monitored in the boiler water and typically ranged from 10 to 100 ppb.

All-volatile (hydrazine-morpholine) treatment was implemented in May 1971 in place of the phosphate-hydrazine-morpholine treatment. Secondary water chemistry control with all-volatile treatment is continuing during operations with the Light Water Breeder Reactor. A maximum boiler water conductivity limit of 10 μmho/cm was imposed to provide additional control against the presence of undesirable contaminants such as caustic, chlorides or hard scale forming materials. The conversion from the phosphate-hydrazine-morpholine to all-volatile treatment was accomplished initially in April 1971 by draining and refilling the steam generators. After three days of operation, phosphate additions were resumed. In May 1971 the steam generators were again converted to all-volatile chemistry, and phosphate was removed by blowdowns from hot standby. High concentrations of phosphate occurred in the boiler water during the first week following conversion. Blowdowns were performed as necessary to reduce phosphate concentrations. The range of phosphate concentrations for the one month period following conversion (May 1971 through June 1971) was:

Steam Generator	Range of Phosphate Concentration, ppm (May 1971-June 1971)
1A	Not Operating
1B	0.2 - 4.3
1C	0.7 - 80
1D	0.6 - 129

Low concentrations of phosphate, up to 4.6 ppm, persisted through October 1971; and from October 1971 through mid-November 1973, phosphate concentrations averaged 0.3 ppm with a maximum of 1.1 ppm. After November 1973, phosphates were no longer monitored.

Chloride concentrations in the boiler water were normally within specification (<0.5 ppm), but occasional out of specification values occurred. The average chloride concentration was 0.2 ppm in the boiler water during the 1971 to 1974 period with all-volatile treatment. Since resumption of operations in September 1977, chloride concentrations have been maintained at 0.1 ppm or less. The maximum chloride detected during the period of operation with all-volatile treatment was 1.5 ppm in the 1C steam generator. Chloride was removed by blowdown, but was higher than specification for about twenty-four hours. In general, the control of secondary chemistry during the period of all-volatile treatment was good.

The conversion of Shippingport steam generators from phosphate-hydrazine-morpholine to hydrazine-morpholine (all-volatile) chemistry did not result in severe tube corrosion or support plate corrosion after four years of additional operation based on eddy current inspection of the removed 1D unit in 1975. Cleaning of Shippingport steam generators was not performed prior to chemistry conversion. This conversion was done prior to the widespread corrosion problems observed in commercial plant steam generators following conversion from phosphate to all-volatile chemistry. In light of current corrosion concerns, particularly with plants having chloride in-leakage to the steam generators, the need for special precautions prior to conversion should be evaluated on a case basis for the individual steam generator design and operating history. For example, steam generator cleaning could be performed to minimize residual phosphate levels in the boiler water as experienced following Shippingport conversion.

The all-volatile chemistry specifications for Shippingport, and typical operating values for the boiler water, are shown below.

BOILER WATER	Specification	Average Measured Value	Range of Values
Cold Wet Layup			
Hydrazine, ppm	50 to 100	80	22-320
pH	9.5 to 10.5	9.7	9.2-10.7
Specific Conductivity, μmho/cm	30 max.	18	2-30
Chloride, ppm	0.5 max.	0.12	$<$0.05-0.25
Steaming/Hot Standby			
Hydrazine, ppm	Detectable	0.06	0.001-0.4
pH	8.0 min.	8.7	7.1-9.8
Specific Conductivity, μmho/cm	10 max.	7.6	3.6-17.3
Chloride, ppm	0.5 max.	0.2	$<$0.05-0.5
SiO_2, ppm	25 max.	2.4	0.56-7.2
FEED WATER			
Morpholine, ppm	0.5 to 3.0		
Hydrazine	1.5 x O_2 conc.		
pH	8.5 to 9.3		
Conductivity, μmhos	7.5 max.		
Chloride, ppm	0.1 max.		

STEAM GENERATOR TUBE DAMAGE

U-Tube Steam Generators

Primary to secondary leakage occurred in the 1B and 1C, U-tube steam generators, during 1966 and 1967. Ultrasonic inspection showed the tube leakage to be located at the apex of the U-bend where the loop-end support straps have maximum lineal contact with the tubes. The ultrasonic inspection showed that the defects were oriented longitudinally, and were up to about five inches long. For example, the defect in the leaking tube in the 1B steam generator in August 1966 had an opening 40 to 60 mils wide and about 4½ inches long. At first, the defects and leaks found at the loop end support straps in the 1B and 1C units were attributed to flow induced vibration. A summary of the leakage that occurred and the number of tubes plugged is shown in Table II. No tubes were removed from the 1B and 1C steam generators for inspection of loop-end region damage. When tube damage subsequently occurred in the 1A and 1D units in 1969-1971 (discussed later), these earlier 1966-1967 inspection results from the 1B and 1C loop end support regions were reviewed. It was observed that the tube damage was rotated circumferentially from the point of contact between the loop end support strap and the tube, which was not consistent with tube vibration damage. Further, the characteristics of the damage appear similar (except for the tube bundle orientation) to those reported for the loop end support region of the Mihama 1 steam generator. As a result, it was concluded that corrosion due to concentrated phosphates was the most probable cause of the 1B and 1C loop end tube damage.

Repairs of the 1B and 1C steam generators were achieved initially by tube plugging (1966 and 1967) and ultimately by installation of flow blocking plates (1968) in the outlet plenum to eliminate primary water flow through those tubes having long lineal contact between the apex of their U-bends and the loop-end support straps. The presence of the flow blocking plate prevents heat transfer from these tubes and concentration of any chemicals on the secondary side in this region. No tube leakage occurred in the 1B and 1C units after installation of the flow blocking plates.

An eddy current inspection was also performed on the 1C steam generator in September 1970. Seven indications of possible defects were reported, but these appeared only as distortions of the eddy current signal at the first support plate on the inlet leg of six tubes and at the third support plate of the inlet leg of the seventh tube. Three tubes were removed from the 1C steam generator to determine the cause of the distorted eddy current signals. Only one defect, with a maximum depth of 3 mils, was found on the removed tubes at a support plate location, and it was concluded that no severe phosphate corrosion had occurred in these hourglass support plate crevices.

STRAIGHT TUBE STEAM GENERATORS

In the period from August 1969 to January 1971, leakage occurred in the 1A and 1D straight tube steam generators. Eddy current inspections showed that defects occurred only in the tube to tube support plate crevice regions. The majority of the defects were found at the four support plates nearest the primary water inlet, with the maximum concentration of defects being at the third and fourth support plates. Support plates are numbered 1 through 11 starting at the primary water inlet end of the steam generator (see Figure 1). Figure 4 shows the distribution of defects found at the third support plate of the 1D steam generator during the 1969-1970 eddy current inspections. A summary of the tube leakage that occurred in the 1A and 1D units is given in Table III. Leaking tubes were plugged.

Tubes were removed from the 1D steam generator to determine the cause of the damage. Inspection of the removed tubes confirmed that defects occurred only within the support plate crevice region. The visual appearance and orientation of defects found at the first through fourth support plates for the 1D steam generator is shown in Figure 5. A typical defect, with about 20 mil depth is shown in Figure 6. The defects on removed tubes were examined in detail by metallography and by scanning electron microscopy. In general, the defects appeared to be large diameter (up to 1/4 inch), irregularly shaped, steep-walled pits, centrally located within the support plate region. The defect surface was usually covered by a thin layer of a whitish deposit having a high elemental phosphorus content, presumably as phosphate. Where the deposits were thin and corroded metal surfaces were exposed, it was found that grain boundary attack and intragrain pitting were visible as shown in Figures 7 and 8. The grain boundary attack was shallow based on metallographic examinations, and the penetrations were less than a few tenths of a grain in depth.

Metallographic examinations performed on defected tube sections showed no unusual microstructure. Metallographic sections through a leaking and non-leaking defect are shown in Figure 9. The metallurgical attributes and the chemical and physical properties of the Alloy 600 tubing were all within normal ranges. Hence, it was concluded that defective material was not an assignable cause of the tube damage. The presence of shallow grain boundary attack within the defect suggested that corrosion was responsible for the damage. Reviews of secondary water treatment records did not show the presence of any excessive contaminants, such as chlorides, that could have contributed to the damage. Further, the high concentrations of phosphorous found within the defect suggested that phosphate was the chemical species causing the corrosion. The concentration of defects at support plates near the primary water inlet end of the steam generators suggested that heat flux was an important parameter in creating the damage. Heat flux was believed to provide a mechanism for concentrating phosphate in the support plate crevice regions.

Calculated heat flux variations along the length of the straight tube steam generators are shown in Figure 10. Also, superimposed on Figure 10 are the numbers of defects found at each support plate location of the 1D steam generator during the 1970 inspection. The highest heat flux (63,000 BTU/hr-ft^2) occurs at the support plate nearest the primary inlet (No. 1). A large number of defects (172) occurred at the No. 1 support plate, but the maximum number of defects (475) were found at the third support plate. There is not, at present, an explanation for the maximum number of defects occurring at the third support plate. It is possible that occurrence of the maximum number of defects at the third support plate may be related to the local thermal and hydraulic conditions, which could have caused the highest phosphate concentrations in this region of the steam generator.

LABORATORY TESTS

A type of tube damage similar in characteristics, but not in extent to that found in the 1A and 1D steam generators was reproduced in laboratory tests performed using horizontal, heated tubes, on which were mounted simulated support plates, installed in an autoclave. The water chemistry was maintained within the control band specified for Shippingport steam generators using coordinated phosphate-hydrazine-morpholine treatment. Heat transfer was accomplished using electrical heaters inserted in the tubes, with power adjusted to obtain the heat flux corresponding to that at the first support plate in the Shippingport steam generators. Temperature and pressure were controlled to simulate conditions existing in Shippingport. Defects which visually appeared similar (including the shallow grain boundary attack and intragrain pitting) to those on the Shippingport tubes were produced with a maximum depth up to about 4 mils in 3400 hours of heat transfer operation. The defects in the test did not sustain this initial rate of corrosion penetration, unlike the defects in the operating steam generators. The test assembly as-removed from the autoclave is shown in Figure 11, and the appearance of the tubes after removal of the mockup support plates is shown in Figure 12. Note the similarity in appearance of defects with those shown in Figure 5 for a tube removed from the 1D steam generator. The surface of defects found on the test tubes showed grain boundary attack and intragrain pitting (Figure 13) similar to that observed on removed Shippingport steam generator tubes. Metallographic sections through a typical defect from the heated tube test are shown in Figure 14.

Other laboratory testing under non-heat transfer conditions showed that NiCrFe Alloy 600 had high initial corrosion rates in concentrated phosphate solutions at elevated temperature. For example, at 550°F the corrosion rate of Alloy 600 during a seven day electrochemical test in a coordinated sodium phosphate solution with a sodium to phosphate ratio of 2.4 and containing 50,000 ppm phosphate is on the order of 10 to 20 mils per year. Examination of the corrosion specimens from these tests showed that their surface appearance was similar to Shippingport steam generator tube defects (see Figure 15). Long term autoclave tests using weight change specimens in concentrated phosphate solutions have shown that the high initial corrosion rates of Alloy 600 (2 mpy after 250 hours) diminish to low values (0.2 mpy after 4000 hours), and the applicability of the initial high corrosion rate to Shippingport steam generator tube damage is not understood. It is not known by what mechanism the initial high corrosion rate was sustained in the Shippingport steam generators. The tests showed that high corrosion rates of Alloy 600 only occurred in the presence of concentrated sodium phosphate solutions, and that other variables had only small effects. It was concluded, therefore, that concentrated sodium phosphate was the probable cause of the Alloy 600 corrosion. The corrosion rate of about 9 mils per year at 550°F, 50,000 ppm phosphate concentration and a sodium to phosphate ratio of 2.4 is comparable to that required for the leakage observed in the Shippingport 1D steam generator after $4\frac{1}{2}$ years of operation. The surface appearance of NiCrFe Alloy 600 specimen after seven days of test is shown in Figure 15. Grain boundary attack similar to that found in defects of tubes removed from the Shippingport 1D steam generator is present. Figure 15 shows more intragrain attack than the Shippingport defect surface (Figure 8) or the heated tube test defect surface (Figure 13). Variations in the amount of intragrain surface attack have been observed among the various defects examined. These variations in the extent of intragrain surface attack may be caused by differences in phosphate concentrations at the defect surfaces.

Based on the laboratory test results, the mechanism by which the Shippingport tube defects were formed appears to be a direct reaction between concentrated phosphate and the tube metal. The concentration of phosphate occurs in the support plate crevice regions as a result of the heat transfer conditions existing during power operation. This concentration mechanism also appears to be responsible for the damage (initially believed to be caused by flow-induced vibration) that occurred in the loop-end support region of the 1B and 1C steam generator. As shown in Figure 3, the presence of the wide loop-end straps creates a flow blocked region which permits phosphate to concentrate and attack the tubing. The hourglass support plate design (Figure 2) used on the straight tube lengths of the 1B and 1C steam generator provides better crevice flushing than the cylindrical hole design of the 1A and 1D units. The concentration of phosphate is minimized with the 1B and 1C hourglass design; and, hence, only minimal tube damage might be expected, as was shown by the 1970 eddy current inspection of the 1C unit.

STEAM GENERATOR PERFORMANCE SINCE CONVERSION TO ALL-VOLATILE CHEMISTRY

The corrective action taken in May 1971 to minimize the damage to the Shippingport 1A and 1D steam generator was to convert chemistry to all-volatile treatment. Installation of flow blocking plates in the 1B and 1C units was done earlier (in 1968) to prevent loop-end region damage. No tube leaks occurred in the 1B, 1C, and 1D steam generator during the period from the inception of all-volatile chemistry in May 1971 until the shutdown in February 1974 for plant modification. The 1A steam generator developed a small leak of about 0.8 to 1.5 gal. per hour immediately after its return to service in October 1972. The leak was suspected to be caused by a defective weld associated with a tube plug rather than by a new tube leak. The leak rate of the 1A unit increased slowly to about 5 gph in October 1973 when the steam generator was removed from service. Plant average power levels were about 20 percent lower during the all-volatile treatment period than during operation with phosphate treatment.

The 1A and 1D heat exchangers were replaced in 1975 (with available spare units of identical design) to provide additional assurance for continued steam generator tube integrity during future Shippingport operations for the Light Water Breeder Reactor. An eddy current inspection was performed in October 1975 on the 1D steam generator after its removal from the plant. Results from this eddy current inspection showed that no new defects had formed during the period of operation with all-volatile chemistry. Some growth of existing defects was detected for the period of operation with volatile chemistry with more defect growth occurring in the lower portions of the tube bundle. The defect growth rate per 1000 effective full power hours of operation was lower during the all-volatile treatment period (0.5 to 3% of tube wall thickness) than during the phosphate treatment period (5 to 20% of tube wall thickness).

The eddy current inspection results for the 1D steam generator showed no evidence of tube denting.

The small growth of defects during the period of operation with all-volatile treatment is believed to be related to the presence of low concentrations of phosphate in the boiler water during the period following conversion to volatile chemistry. The larger growth rate of defects in the lower region of the tube bundle may be attributable to the heavier sludge deposits found on these tubes. The thicker sludge deposits in the lower portions of the bundle would be expected to retain more phosphate than the thinner deposits found on the upper tubes.

CONCLUSIONS

Based on the Shippingport steam generator inspections, examinations of removed tubes, and laboratory tests, the major observations and conclusions are as follows:

1. All four Alloy 600 tubed steam generators had leaks.
2. The two U-tube steam generators had tube leaks in the loop end support strap crevice region, but not in the hourglass hole support plate crevice region.
3. The two straight-tube steam generators had tube leaks in the cylindrical hole support plate crevice region.
4. Plugging of leaking tubes adequately corrected leaks.
5. For the two U-tube steam generators, flow blockage of tubes having loop end support strap crevices succeeded in stopping further leaks.
6. For the two straight-tube steam generators new leaks occurred despite selective tube plugging and blockage. Conversion to all-volatile chemistry minimized further tube damage in crevices. These two units were ultimately replaced.
7. All four Alloy 600 tubed steam generators were converted from phosphate-hydrazine-morpholine to hydrazine-morpholine (all-volatile) chemistry, and no tube denting occurred as a result of this conversion based on eddy current inspection of the 1D steam generator after four years using all-volatile chemistry.
8. Examination of tubes removed from straight-tube steam generators showed high levels of residual phosphorous in the deposits on tubes in defected, crevice surfaces, and these deposits are believed to be metal phosphates.

9. Laboratory tests reproduced a similar type of Alloy 600 tube corrosion in both heated tube tests with nominal phosphate chemistry (as in Shippingport when tube leaks occurred) and in electrochemical tests and autoclave tests with artificially concentrated phosphate solutions. The tests produced initial high corrosion rates similar to rates which were comparable in Shippingport steam generator tubes. However, laboratory tests did not sustain the initial high corrosion rates as occurred in service.
10. The cause of the tube leaks in the Shippingport steam generators is considered to be sodium phosphate which concentrated in inadequately flushed crevices during steaming and corrode Alloy 600 tube material. It is not understood by what mechanism the high rates of corrosion seen in Shippingport steam generator tubes were sustained.
11. Variations in tube corrosion propensity from one crevice to another as observed in Shippingport steam generators is believed to be due to local thermal hydraulic condition variations.

ACKNOWLEDGEMENT

The investigation of Shippingport tube damage was performed in cooperation with and under the technical direction of the U.S. Department of Energy, Division of Naval Reactors. The author is indebted to the numerous persons at Bettis Atomic Power Laboratory who contributed significantly to this investigation.

REFERENCES

1. Singley, W.J.; Welinsky, I.H.; Whirl, S.F. and Klein, H.A., "Stress Corrosion of Stainless Steel and Boiler Water Treatment, Shippingport Atomic Power Station", Proc. Am. Power Conf. 21, pp. 748-768 (1959)

2. Welinsky, I.H.; Brindley, W.F.; Berry, W.E.; and Whirl, S.F, "Final Evaluation of Shippingport Power Stations Stainless Steel Steam Generator After Five Years", Proc. Nat. Assoc. Corrosion Engrs. Conference on High Purity Water Corrosion of Metals, pp. 1-13 (1968)

3. Kemmet, L.H. and Werling, J.A. editor, "Shippingport Operations from Initial PWR Core 2 Power Operation To Power Operation After First Refueling", WAPD-332, pp. 5-12, June 1973

4. Whirl, S.F. and Purcell, T.E., "Protection Against Caustic Embrittlement by Coordinated Phosphate pH Control", Proc., Annual Water Conf. Eng. Soc. of W.Pa. 3 pp. 45-60B (1942)

TABLE I

Shippingport Steam Generator Design Parameters

Type	Straight Tube (1A and 1D)	U-Tube (1B and 1C)
Steam Pressure, Operating range, psia	600-930	600-930
Heat Transfer load at full power, BTU/hr	4.3×10^8	4.3×10^8
Steam Flow per unit at full power, lb/hr	481,000	481,000
Recirculation Ratio	12:1	10:1
Temperatures (Full Load)		
Primary Inlet, °F (T_H)	564	564
Primary Outlet, °F (T_C)	508	508
Primary Average °F (T_{ave})	536	536
Steam	486	486
Feedwater	338	338
Heat Transfer Area, $ft.^2$	11,670	13,387
Tubes		
Outside diameter, inches	1/2	5/8
Minimum wall thickness, inches	0.042	0.054
Number	3050	1692
Support Plates		
Number	11	5
Thickness, inches	0.5	0.375
Hole Geometry	Cylindrical	Hourglass
Materials		
Tubes	NiCrFe Alloy 600	NiCrFe Alloy 600
Support Plates	Carbon steel	Carbon steel
Tube sheets	Carbon steel	Carbon steel

TABLE II

Steam Generator Tube Leakage and Repair
1B and 1C U-Tube Units

Date of Leakage or Repair	Aug. 1966	Jan. 1967	May 1967	1968	
Steam Generator	1B	1B	1C	1B	1C
EFLH(1) at Time of Leak	5960	6460	9870	--	--
No. of tubes leaking	1	1	1	0	0
Max. Est. Leak Rate	100 gph	150 gph	9 gpm	--	--
No. of Tubes Plugged/Blocked	1	32(3)	15(3)	189(2)	189(2)

(1) EFLH - Effective full load hours for the steam generator.
(2) Flow Blocker installed.
(3) Includes non-leaking tubes plugged because of excessive wall thinning.

TABLE III

Steam Generator Tube Leakage and Repairs 1A and 1D Straight Tube Steam Generators

Date of Leakage or Repair	Aug. 1969	July 1970	Jan. 1971	Feb. 1971	June 1971	Aug. 1971	Oct. 1973
Steam Generator	1D(6)	1A	1A	1D	1A(5)	1A	1A
EFLH at Time of Leak	16,120	19,300	--	17,820	22,412	22,412	24,396
No. of Tubes Leaking	1	2	--	1	5	2	--
Max. Est. Leak Rate(1), gph	N/A	N/A	5-7	N/A	N/A	N/A	5
No. of Tubes Plugged/Blocked	20(3)/4(4)	4	0	5/36(7)	15	37	--(2)

NOTES:

(1) N/A - Leak Rate not available.

(2) Steam Generator 1A removed from service.

(3) Includes nine tubes removed for examination.

(4) Partially flow blocked. Holes drilled in plugs to establish predetermined flow rates.

(5) 1A Steam Generator out of service May 1971 to September 1972.

(6) 1D Steam Generator out of service August 1969 to July 1970.

(7) Flow blocked tubes - plugged on one end only.

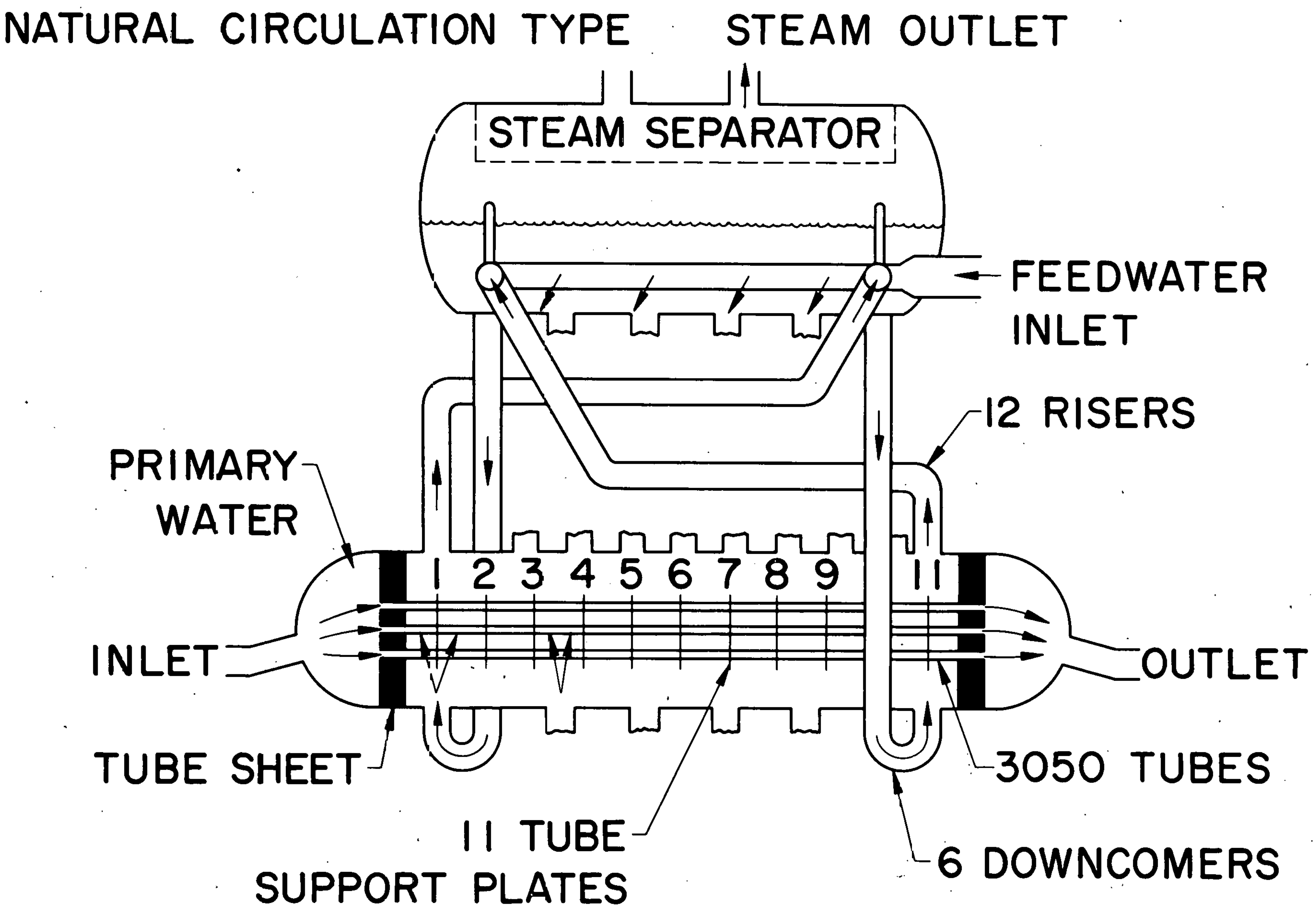
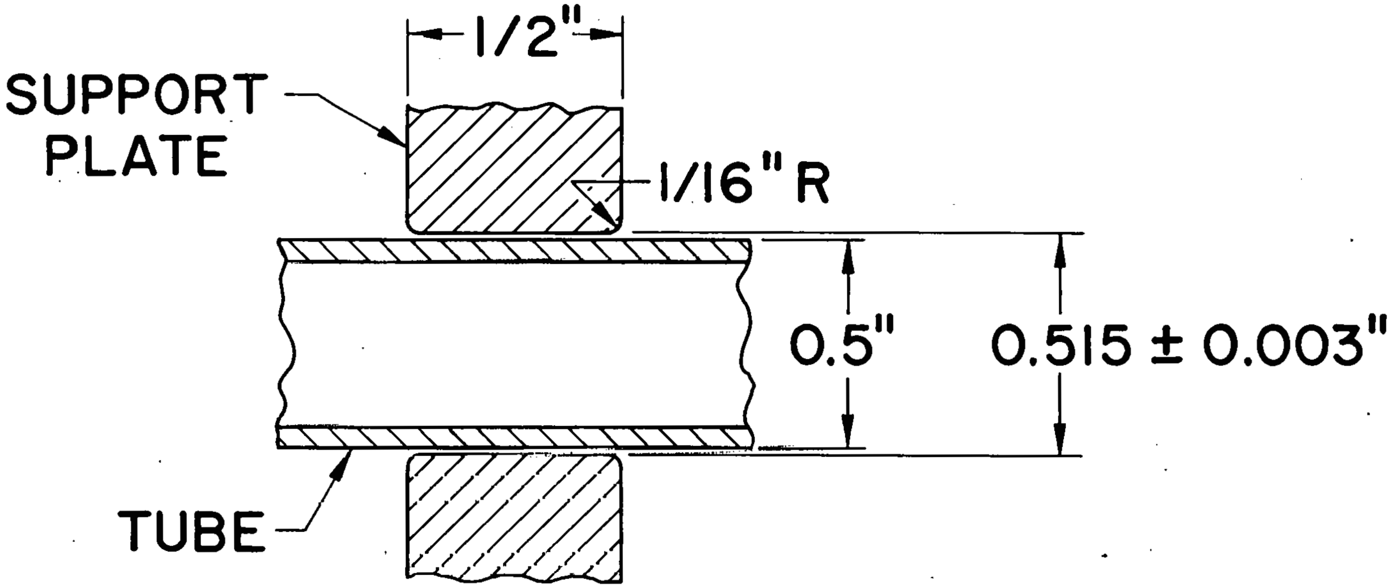
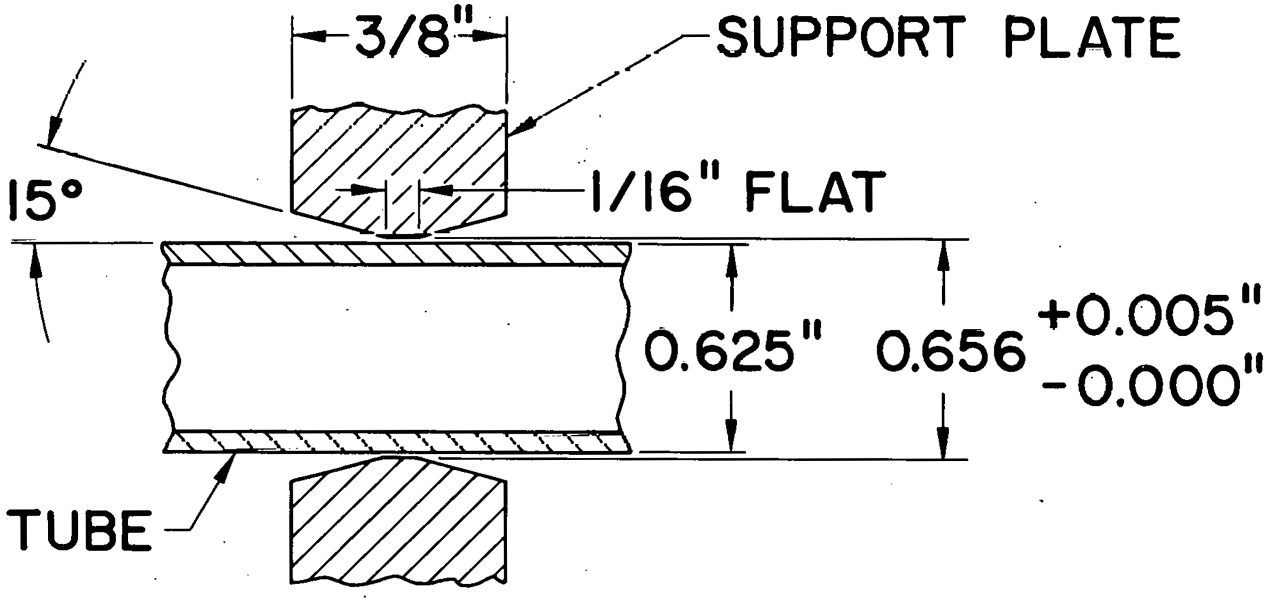


Fig. 1 Schematic of Shippingport Horizontal Straight Tube Steam Generator



SECTION THROUGH CYLINDRICAL HOLE SUPPORT PLATE, IA AND ID UNITS



SECTION THROUGH HOURGLASS HOLE SUPPORT PLATE, IB AND IC UNITS

Fig. 2 Support Plate Hole Geometry For Straight Lengths of Tubing (Top of Steam Generator is toward top of illustration.)

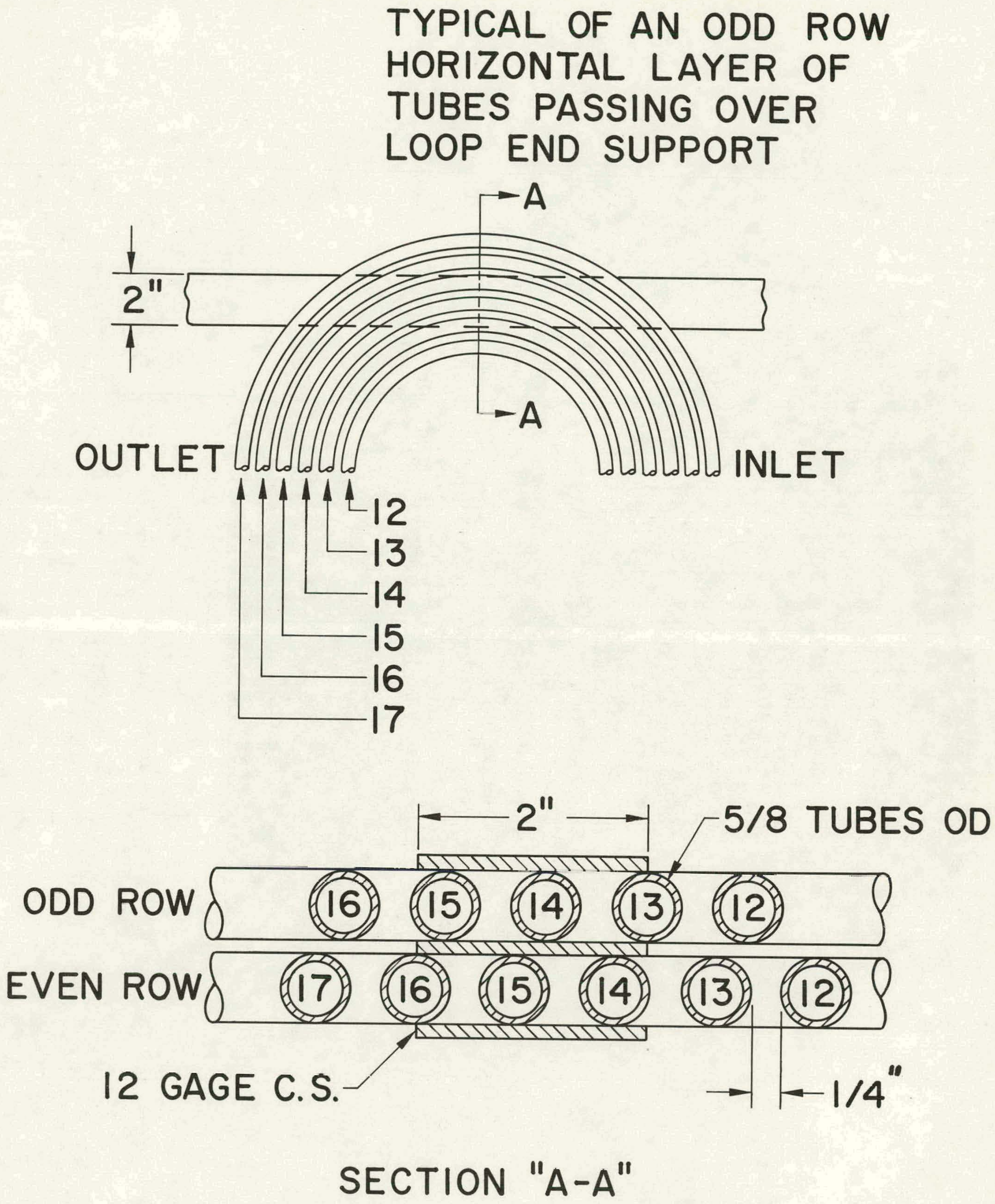


Fig. 3 1B and 1C Steam Generators View of Tubes in Loop End Support Area
(For upper illustration the vertical axis of the Steam Generator is perpendicular to the plane of the tubes shown. For the lower Figure, the top of the Steam Generator is toward the top of the illustration.)

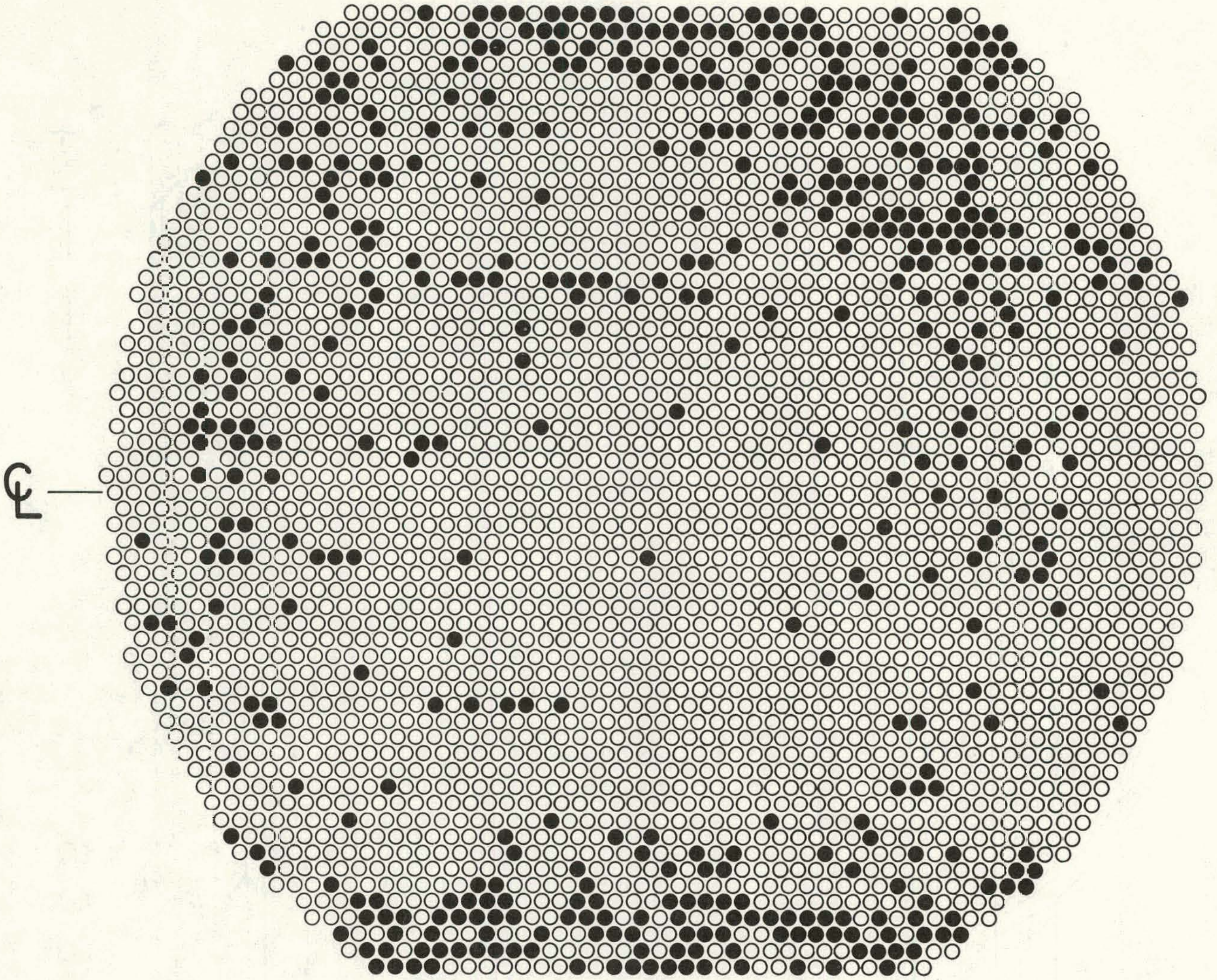


Fig. 4 Shippingport 1D Steam Generator Location of all defects at 3rd Support Plate (1969-1970 Inspection)
NOTE: The top of the steam generator is toward the top of the figure.

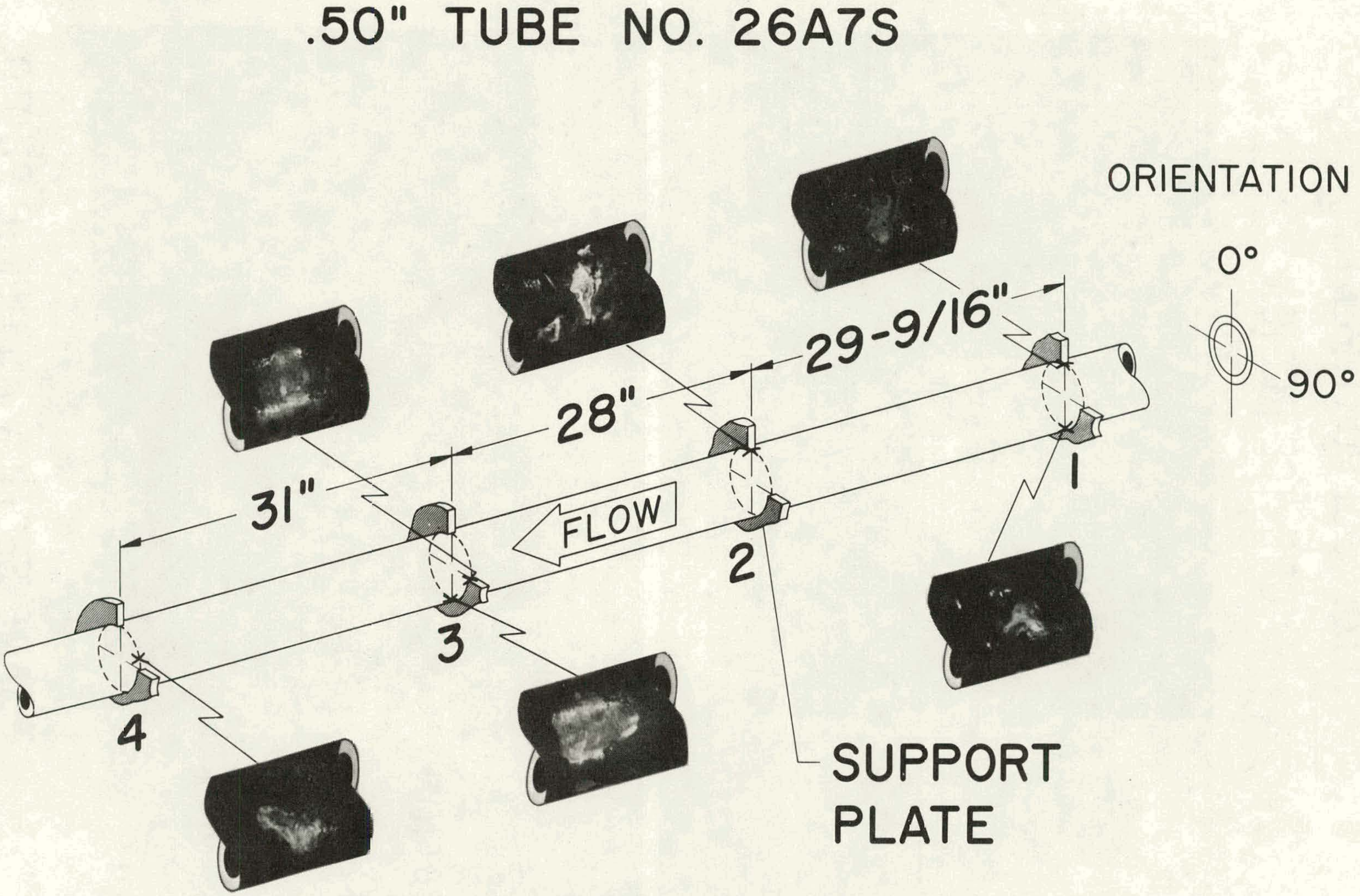


Fig. 5 Shippingport 1D Steam Generator - Major Defect (leak) at Support Plate #1, Minor Defects at Support Plates #2 and #4. Burnishing mark at support plate #2 and #4.
NOTE: 0° is toward top of steam generator.

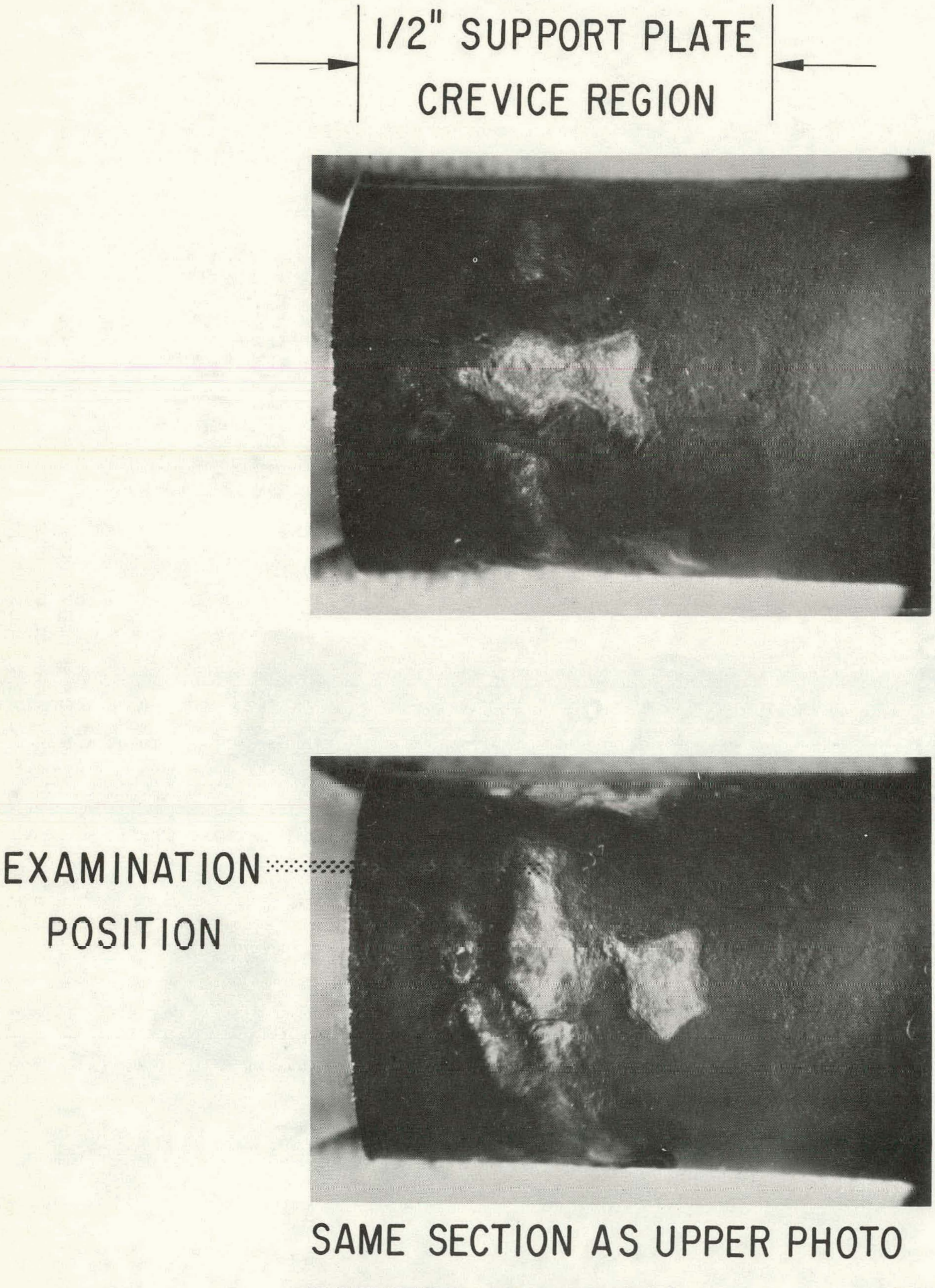


Fig. 6 Defected Section of Alloy 600 Tubing From Shippingport 1D Steam Generator

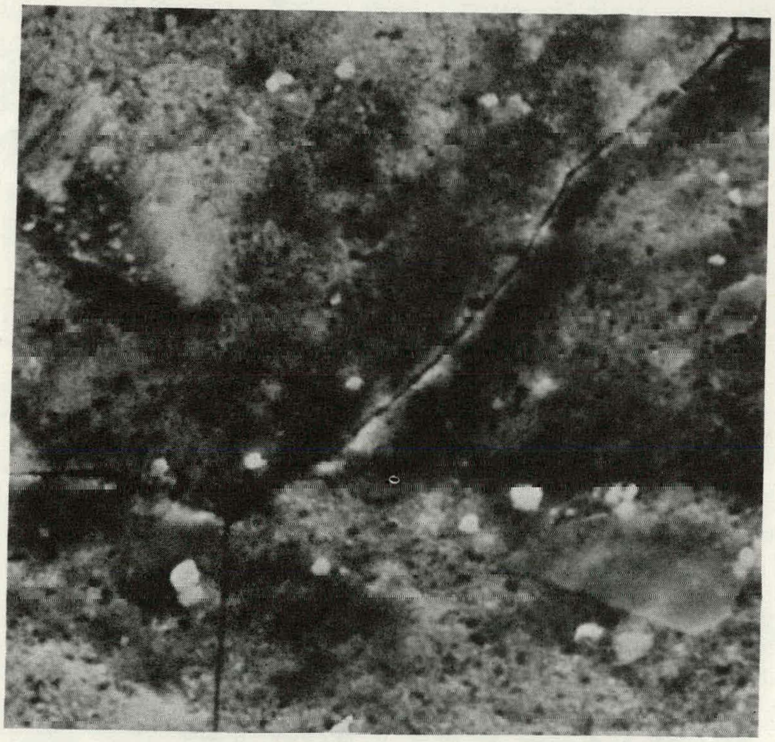
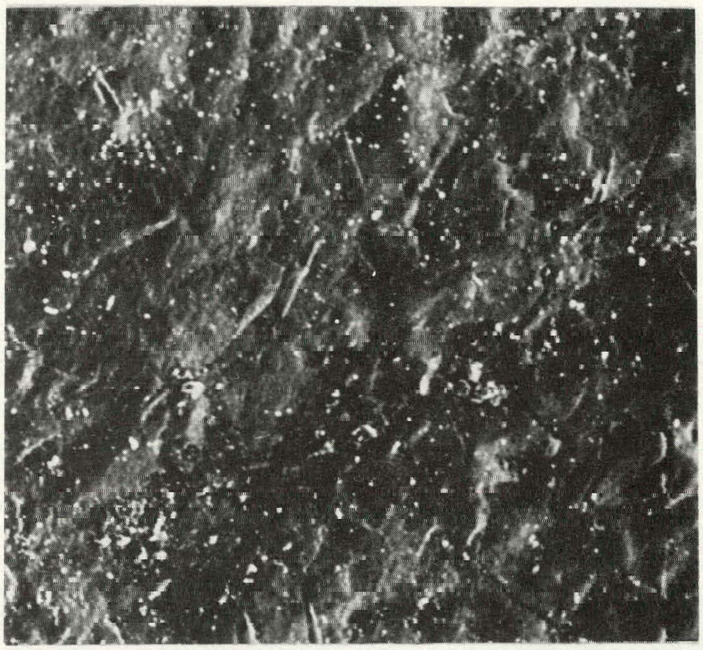
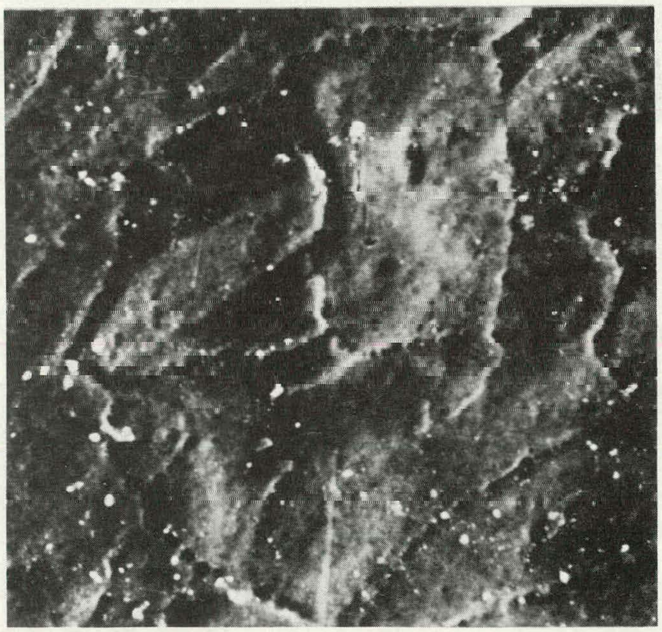


Fig. 7 Photograph of Defect Zone from Examination Position of Figure 1.

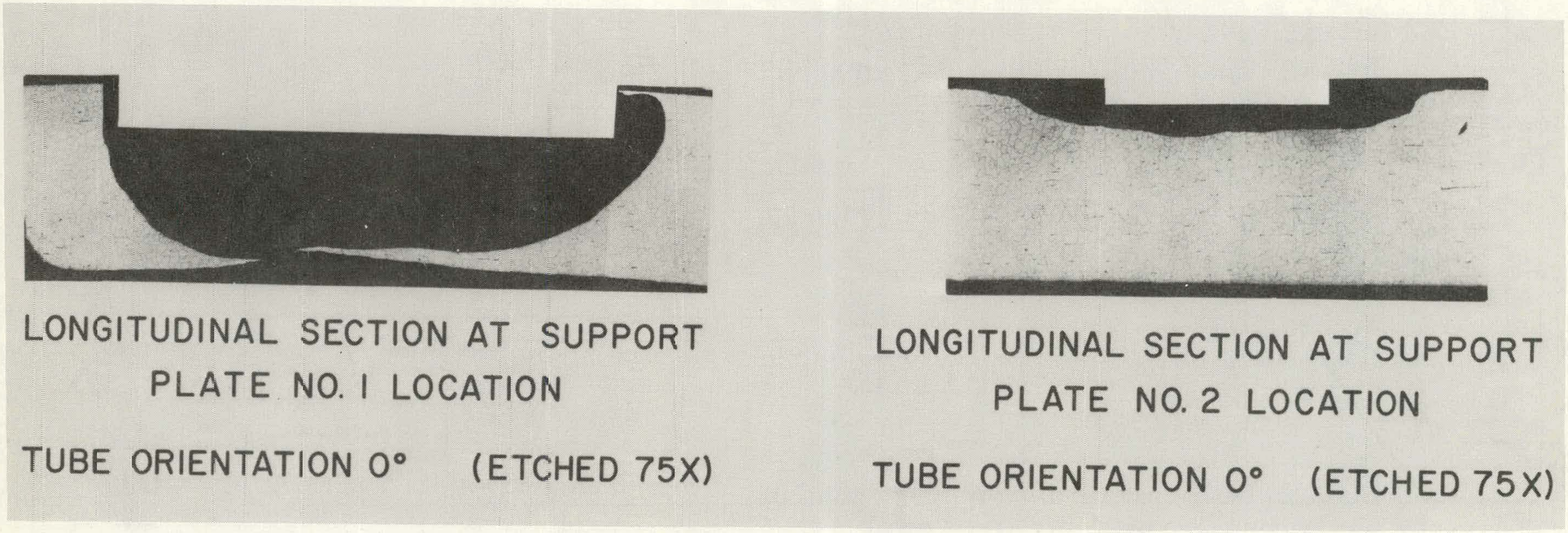


300X



CENTER OF 300X PHOTO
ABOVE 1000X

Fig. 8 Defect Surface from 1D Steam Generator at Third support plate.



LONGITUDINAL SECTION AT SUPPORT PLATE NO. 1 LOCATION

TUBE ORIENTATION 0° (ETCHED 75X)

LONGITUDINAL SECTION AT SUPPORT PLATE NO. 2 LOCATION

TUBE ORIENTATION 0° (ETCHED 75X)

Fig. 9 Metallographic Sections of a Leaking Defect and a Typical Non-Leaking Defect - 1D Steam Generator

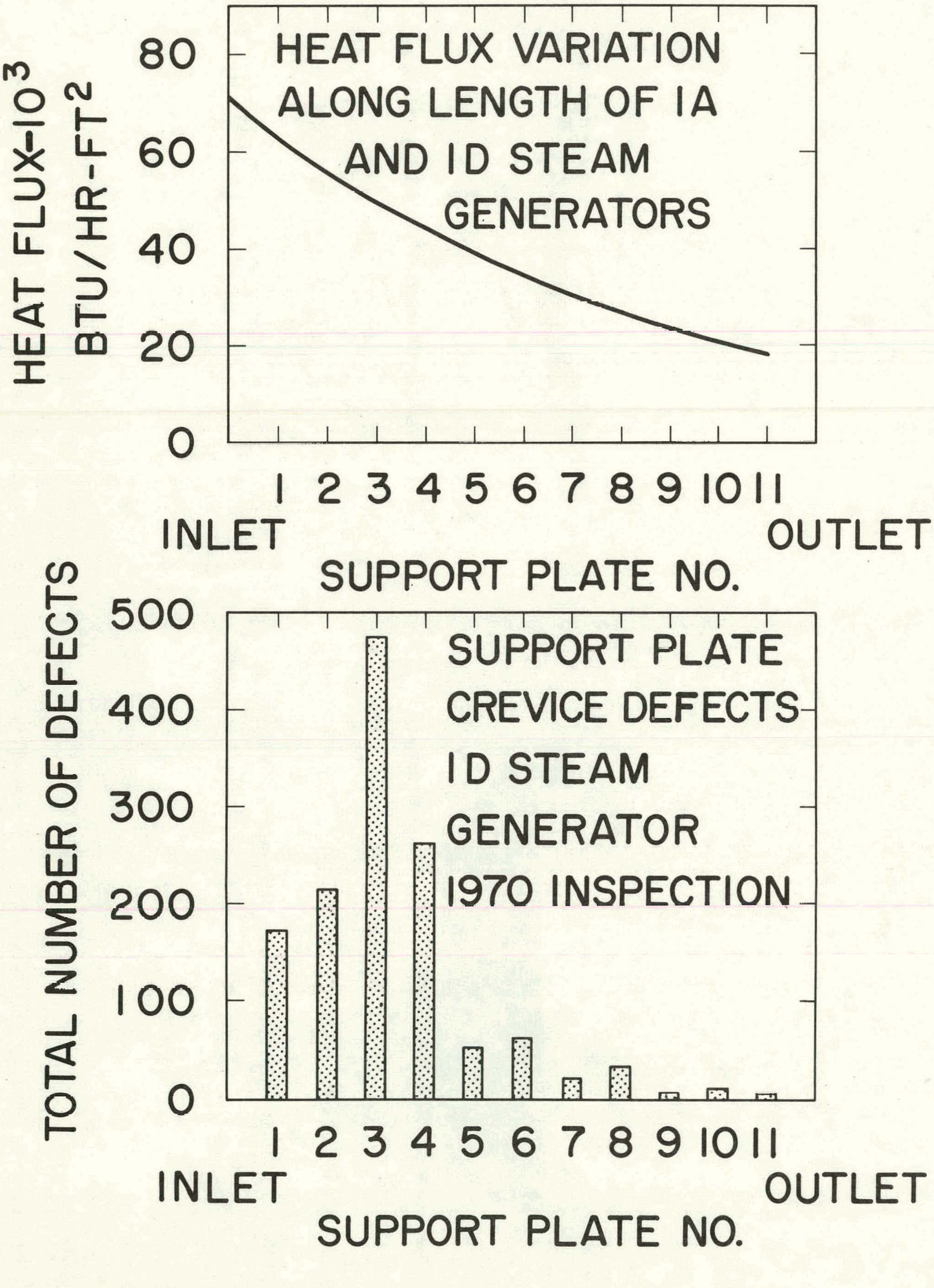


Fig. 10 Variation of Heat Flux and Number of Defects at Support Plate Crevices, 1D Steam Generator

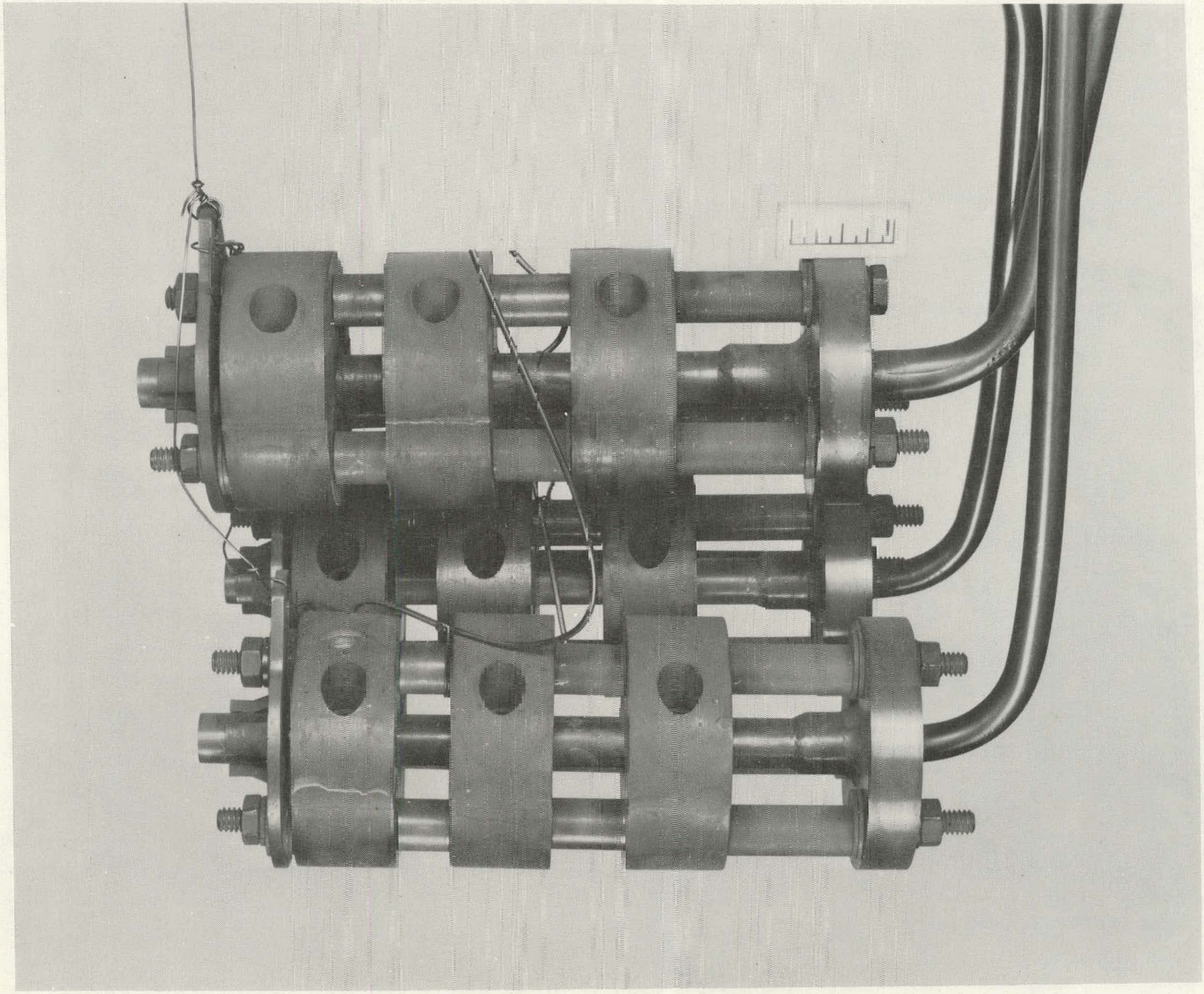


Fig. 11 Heated Tube Assembly as removed from test after 4000 hours

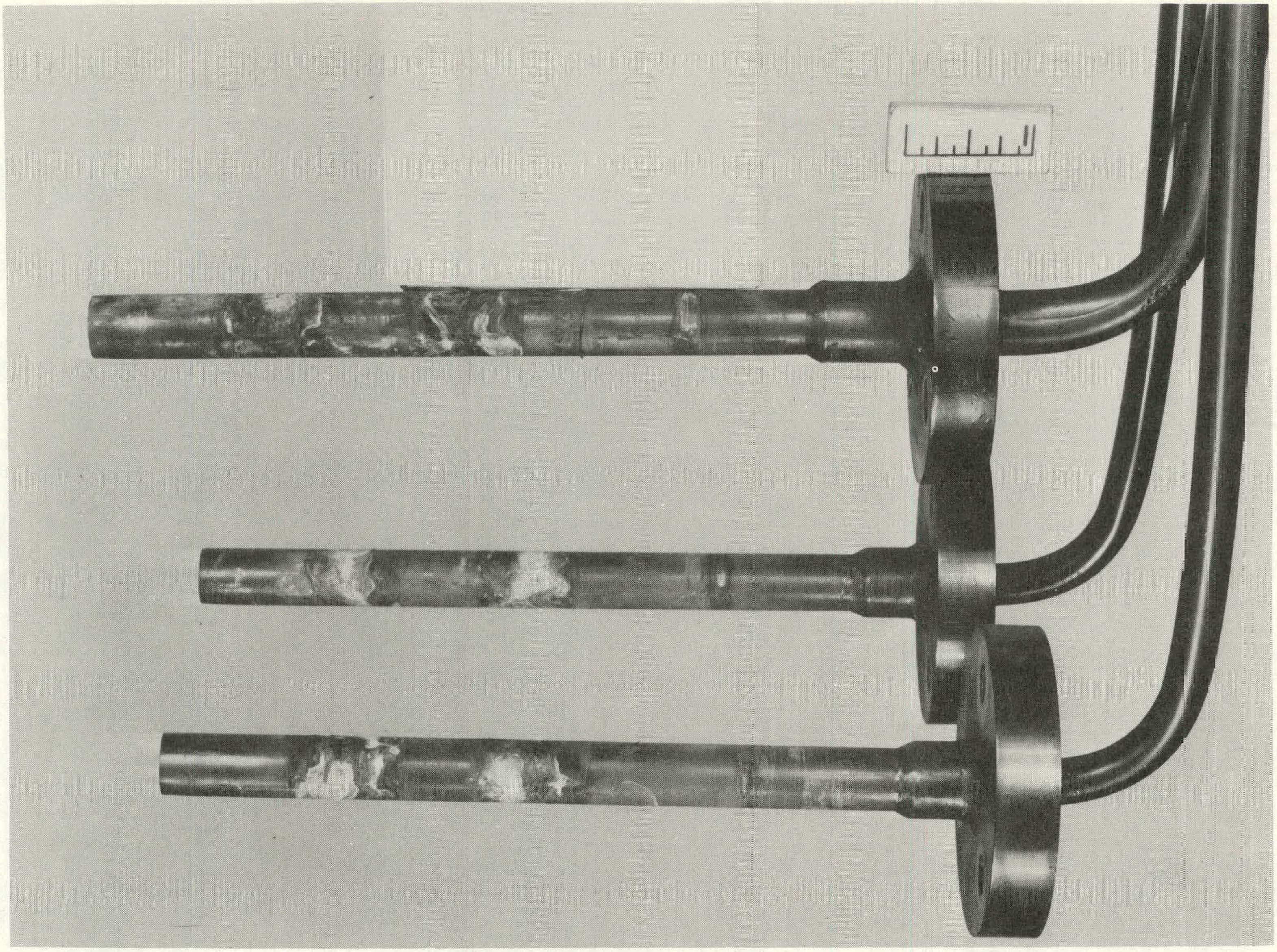
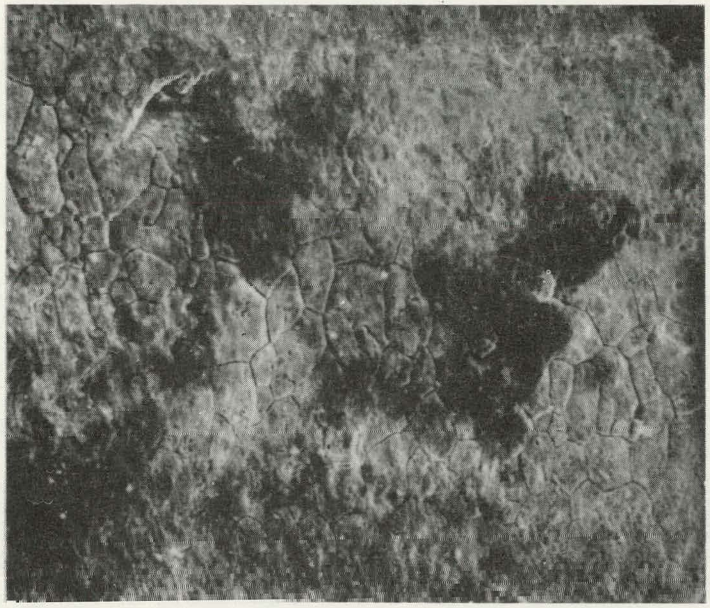
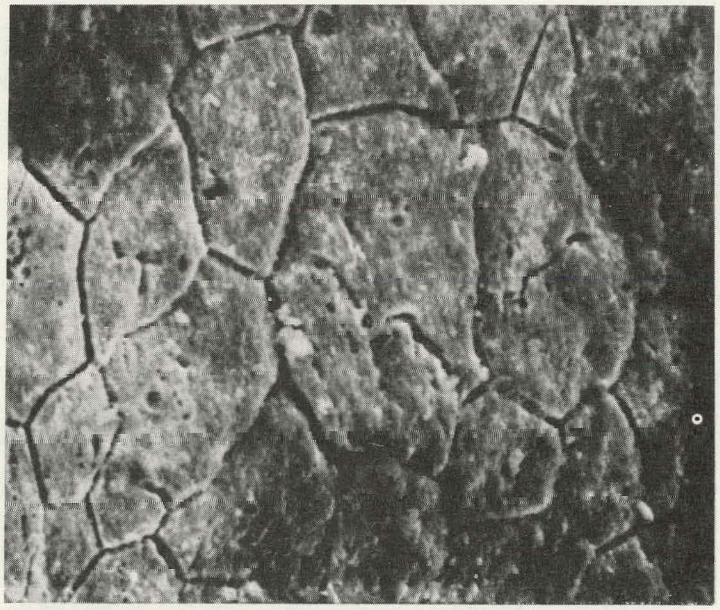


Fig. 12 Appearance of defects on tubes from Heated Tube Test After 4000 hours.
Note similarity with defects on tube removed from 1D Steam Generator (Fig. 5)

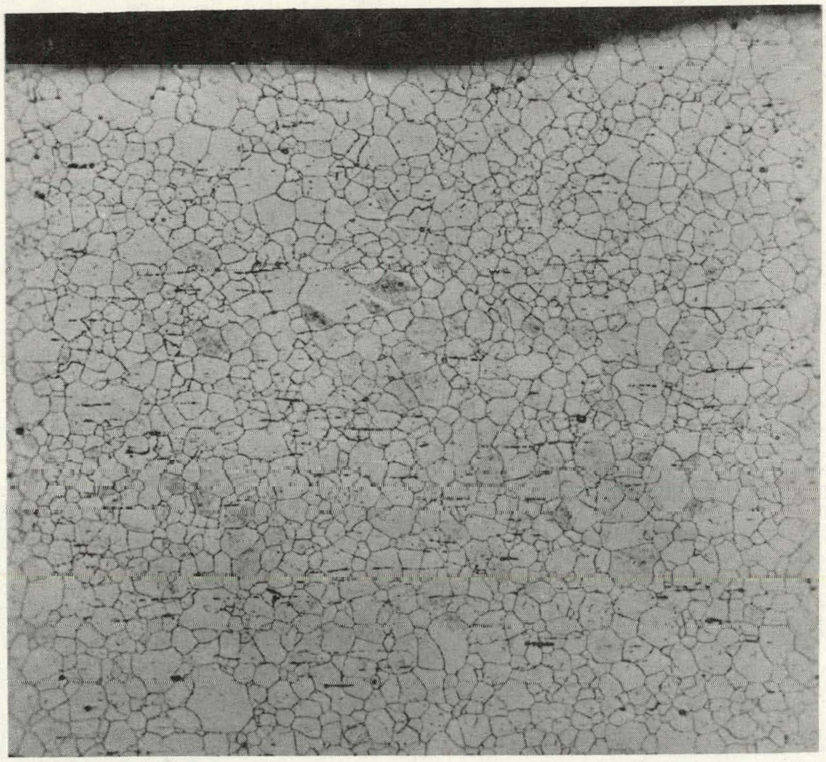


300 X

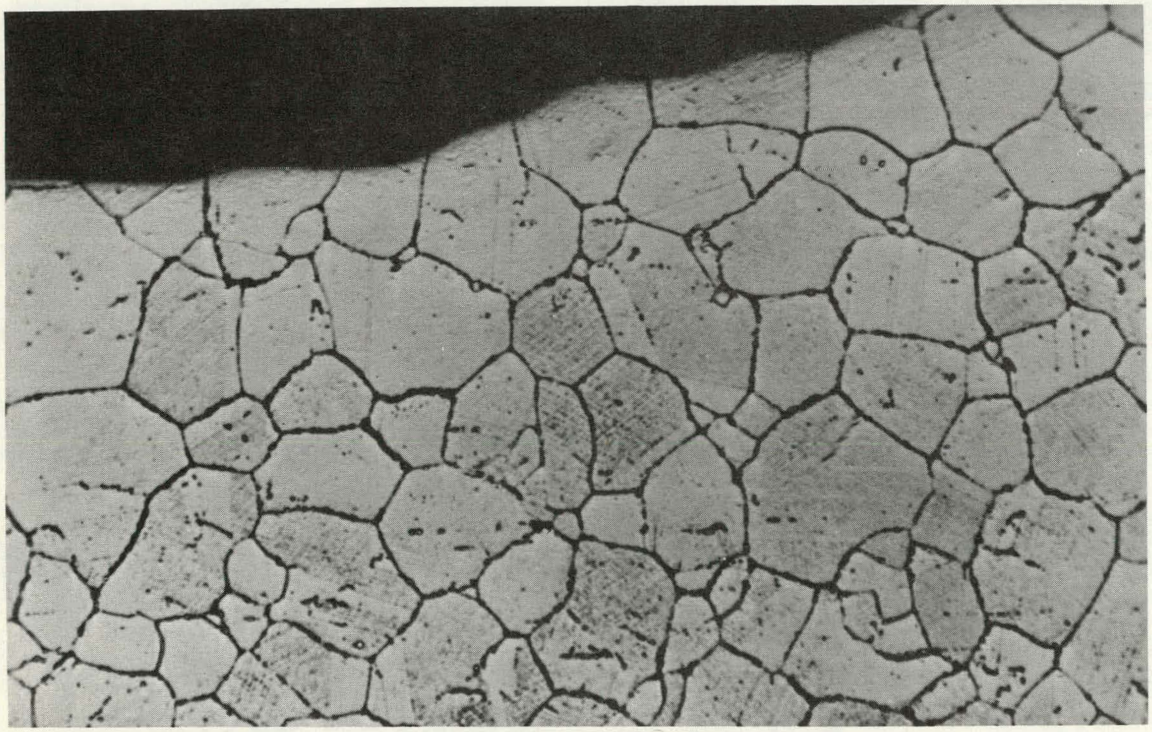


CENTER OF 300X PHOTO 1000X

Fig. 13 Heated Tube Test Defect Surface in Support Plate Crevice After 4000 hours of Test. Note similarity with 1D steam generator removed tube (Fig. 8)



ETCHED-PHOSPHORIC ACID 100X



ETCHED-PHOSPHORIC ACID 500X

Fig. 14 Metallographic Sections through defect in Support Plate Crevice of Heated Tube Test, after 4000 hours test.

Fig. 15 Defect Surface of NiCrFe Alloy 600 Specimen after 7 days at 550°F in 50,000 ppm phosphate (Na/PO_4 = 2.15) Vapor Phase of Specimen adjacent to Vapor-Liquid Interface.

☆U.S. GOVERNMENT PRINTING OFFICE: 1978-703-115/211