Preliminary Evaluation of the Steam
Generator Localized Tube Diameter Reduction
Phenomenon and its Safety Impact

December, 1975

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I. Introduction

Localized steam generator tube diameter reductions have been observed at several operating Westinghouse nuclear plants which had operated with phosphate treatment of the steam generators for significant periods prior to switching to all Volatile Treatment (AVT). Westinghouse NSD Technical Bulletins 75-12 and 75-16 (attachments 1 and 2) first reported this phenomenon. The tube diameter reductions, which have been observed occur at locations where the tubes pass through the tube support plates. Review of eddy current inspection tapes has revealed that the tube diameter reductions occurred after the plants switched to AVT. There have been no occurrences of localized steam generator tube diameter reductions in plants, such as Prairie Island Unit 1, Takahama Unit 1, and Ringhals Unit 2, which had only limited phosphate treatment operation prior to conversion to AVT. The following paragraphs will discuss in greater detail the Turkey Point Unit 4 and Surry Unit 1 experiences, postulated mechanisms for the phenomenon, a safety evaluation of steam generators experiencing tube diameter reductions, planned diagnostic programs, and Westinghouse's conclusions regarding the phenomenon and its safety impact.

Considering the average reduction in tube diameter existing at the most recent inspection and the date of conversion to AVT operation, a linearly extrapolated, maximum rate of deformation may be determined. This rate is preliminary and conservative, in that the expected exponential decay with time is not considered. The average, apparent rate of deformation on the Turkey Point Unit 4 steam generators is 1.5 mils (diametral) per month of operation over the eight month interval which ends in May 1975.

For Surry Unit 1 the average, apparent rate of deformation is 1-1.2 mils/month over the (approximate) ten month interval which ends December 1975.

Attachment I-3 provides supplementary discussion of the apparent rate of deformation.

Nuclear Service Division

Technical Bulletin

An advisory notice of a recent technical development pertaining to the installation or operation of Westinghouse supplied Nuclear Plant equipment. Recipients should evaluate the information and recommendation, and initiate action where appropriate.

P.O. Box 2728, Pittsburgh, PA 15230

Subject: Occurrence of Denting in Steam Generator Tubes	Number: NSD-TB-75-12
System(s): Steam Generator	Date: 10/15/75
Affected Plants: All operating and construction	S.O.(s): 120
References: None	Sheet 1 Of 3

BACKGROUND

During the scheduled refueling outage of the Turkey Point Unit 4 plant in May 1975, eddy current inspections of the three steam generators were performed. The purpose of the inspection was to provide a measure of tube integrity following the steam generator chemistry change to AVT which occurred in September 1974. This insepction revealed a continuation of thinning due to the previous phosphate water chemistry treatment. The corrosion was limited to the area immediately above the tube sheet in the area of sludge accumulation. It has been attributed to residual phosphates retained in this sludge material which surrounds the tubes. These observations are identical to the findings at several other plants with a comparable operating history. The continued corrosion in the sludge pile has provided the basis for increased emphasis on sludge removal by lancing techniques in these units. Each of the steam generators was sludge lanced during the shutdown. Also during the May outage, steam generator mechanical modifications were accomplished.

INFORMATION

Examination of the eddy current tapes revealed additional signals at the intersection of the tube support plates. These signals were interpreted by Zetec and Westinghouse personnel to be due to physical deformation of the tube wall.

The unit was returned to full power in late June and operated until August 3, 1975 when shutdown was initiated to repair a primary to secondary leak which developed in the B steam generator. The leak was identified as a peripheral tube and this tube was plugged. The leak was between the second and sixth tube support plate on the hot leg side. The axial location could not be accurately determined since the standard eddy current probe could not pass beyond these locations.

Additional Information, if Required, may be Obtained from the Originator. Telephone 412 – 256-5413 or (WIN) 236-5413

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During the August shutdown, a tube sample was removed from the B steam generator for laboratory examination to more fully explore the eddy current denting signals. Examination of the tube is still in progress, and the findings thus far are enumerated below:

1. The tube has shown no significant loss in integrity.

2. Wall thinning to the extent of 4 percent (maximum) was evident on the OD surface adjacent to the tube support plate.

3. Tube wall inward deformation up to the extent of 7 mils on the radius was measured by profilometry. The denting is rather uniform, slightly oval, and is the area adjacent to the 3/4 inch thick tube support plate.

4. No gross "working" of the tube surface is evident from metallography.

5. There is no evidence of intergranular stress corrosion.

Both analytical and laboratory studies are in progress to pinpoint the possible cause for this finding. Eddy current tape reviews are also in progress to define any patterns to the denting which may exist.

Since no gross surface working or metal grain deformation is evident in the metallography, vibration forces such as those which might be induced by flow and turbulence do not appear a likely cause for denting. This is also consistent with analytical predictions of dynamic forces in the tube bundle. The definitive mechanism for denting, therefore, is not as yet readily explainable. It is clear that the denting is not related to steam generator mechanical modifications since the eddy current indications were evident prior to implementation of these modifications.

Since restart of the plant in August, another tube leak was detected in the B steam generator. The plant entered into another shutdown on September 21. The leakage was again identified as a peripheral tube and this tube was plugged in addition to several other tubes in this wedge region. It has not been ascertained as yet whether the tube leakage is related to the denting. A section of tubing from the hot and cold legs was removed during the outage for further investigation into the denting phenomena.

A review of eddy current inspection records is in progress for other plants. Some denting signals are also found in this review to an extent yet to be finalized. This review will cover Beznau I and II, Prairie Island , Ginna, Robinson, San Onofre, Point Beach 2, and others as inspections are performed in the future.

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It is important to note that no loss of tube integrity is indicated by this inspection. The eddy current signal from denting is relatively large compared to the usual amplitude of corrosion related phenomena, and calibration standards are being devised to better quantify the amount of physical deformation.

The specific mechanism involved is being pursued with expeditious effort by responsible division within Westinghouse.

Westinghouse Nuclear Service Division

Attachment I-2

Technical Bulletin

An advisory notice of a recent technical development pertaining to the installation or operation of Westinghouse-supplied Nuclear Plant equipment. Recipients should evaluate the information and recommendation, and initiate action where appropriate.

P.O. Box 2728, Pittsburgh, PA 15230

Subject: Localized Steam Generator Tube Diameter Reductions	Number: NSD-TB-75-16
System(s): Reactor Coolant System	Date: November 17, 1975
Affected Plants	S.O.(s) 120
References: NSD Technical Bulletin 75-12	Sheet 1 Of 4

Information

Localized steam generator tube diameter reductions have been observed at some operating plants. Evaluation of such anomalies with regard to the effects on the operation and safety analyses of such plants have been performed. The evidence available to date indicates that operation of affected units can continue without undue risk to public health and safety. Plans have been made to obtain additional data in order to better understand the causes and extent of such diameter reductions: the intent being to minimize their occurrence in the future.

The following elaborates on what has been observed, the evaluations performed to date, and the plans made to better characterize such localized steam generator tube diameter reductions.

Examination of eddy current inspection data obtained from the Turkey Point Unit 4 steam generators during the scheduled refueling in May, 1975 revealed the presence of anomalous signals at the tube/tube support plate intersections. These signals were interpreted as being a result of localized deformation of the tube wall in the area of the tube support plates.

Following refueling, Turkey Point 4 was returned to power in early June and operated until August 3, 1975 when the plant was shut down to repair a primary-to-secondary leak which developed in the "B" steam generator. The leak was in a peripheral tube on the hot leg side at approximately the 2nd tube support. This tube was plugged and another tube having a similar EC indication on the hot leg was removed for examination.

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The plant was restarted in August and operated until September 21 when it was shut down because of a primary-to-secondary leak. The leakage was again identified as being in a peripheral tube at the 2nd support plate, and this tube was plugged. During this shutdown, both a hot leg and cold leg tube exhibiting diameter reduction indications were removed from the unit for examination. Both tube sections extended from the primary face of the tubesheet to beyond the 5th tube support.

Laboratory examination of these tubes is still in progress. The results available to date indicate the following which are essentially similar to the results of the examination of the August tube:

1. Plastic deformation of the tube wall has occurred in the area adjacent to the 3/4" thick tube support plates, resulting in a constriction of the tube diameter in that area. Maximum reduction in diameter in the tubes examined in the laboratory is approximately 20 mils.

2. Metallographic examination and mechanical property tests indicate no loss in tube integrity.

3. Wall thinning to the extent of 4 percent (maximum) was observed on the O.D. of the tube surface adjacent to the tube support plate.

4. There is no evidence of stress corrosion cracking.

5. There is no evidence of extensive deformation of the surface layer within the dent region.

During September, the Surry 1 unit was shut down just prior to a scheduled refueling. At the time of shutdown, primary-to-secondary leakage was experienced in steam generator "A." Three leaking tubes were identified in this steam generator; all were in Row 2, which is one tube removed from the divider lane. The leaks were in the hot leg in the region of the 6th and 7th support plate.

Eddy current examination of this unit is still in progress. Initial results indicate prevalent localized diameter reduction. Plug gaging of the leaking tubes indicates they will not pass a standard EC probe beyond the 2nd tube support. Tubes have been pulled from this unit for detailed laboratory examination. The leaking tubes were plugged and were not removed due to inaccessibility and proximity to the divider plate.

Visual examinations of the tube support plates were made through the tube lane handholes. This examination indicated inplane hour-glassing of the flow slots in the center of the tube support plates. No other physical changes in the tube support plates were observed.

The support plate, tubes, and anti-vibration bars were examined in the U-bend region of the bundle. All components were intact and in the as-shipped condition.

Since the initial evidence of the localized diameter reduction problem, analytical and experimental programs have been underway to establish the mechanism responsible for the phenomena. Examination of affected tubes shows no evidence of surface deformation which would be expected if the problem was caused by flow induced tube vibrations. This is also consistent with analytical preductions of dynamic forces in the tube bundle. Analytical studies to explore the possibility of dia. reduction resulting from thermal ratchetting induced by local thermal/hydraulic effects in the tube support region also indicate that this is not a feasible mechanism. At the present time, it appears that the tube/tube support plate annulus, giving rise to forces sufficiently high to locally plastically deform the tubes.

Methods and procedures are being developed so that a section of tube within the adjacent tube support plate area can be secured for laboratory examination at some future date.

Additional EC examinations are planned to categorize the extent of the localized diameter reduction.

The eddy current inspections of those units which have converted to AVT after extended phosphate operation have indicated that the phenomenon is localized at the tube support plate locations since the AVT conversion. It has been observed in the past essentially on a random basis and to a minimum extent in these units during phosphate operation.

Eddy current inspections thus far on those plants which operated for a short period with phosphates and switched to AVT indicated no denting was taking place.

Laboratory testing is underway to simulate conditions of the proposed mechanism and to identify corrective measures which can be applied to prevent further wall deformation. This testing includes the exposure of tube/tube support simulations to aqueous environments containing contaminants corrosive to carbon steel. The intent is to determine if corrosion products will accumulate and expand on the tube wall. Also, means for neutralizing or removing deposits in the tube/support plate region are being investigated.

Possible implications of localized tube diameter reduction have been considered with respect to normal plant operation, expected transients, and postulated accident conditions.

Regarding normal operation with expected transients and based upon experience at several plants which have operated with tubes in this condition, if primary-to-secondary leakage from affected tubes occurs, it can be detected at low, controlled levels and progresses slowly. If necessary, shutdown and repairs can be accomplished in an orderly manner and with no undue risk to the public health and safety.

With respect to postulated design basis accidents, the criterion of importance is that significant tube damage or ruptures will not occur during the course of the accident as a result of tube diameter reduction. Our evaluation, based on analyses previously conducted, indicates that limiting stresses will not be exceeded, and hence, tube failures are not expected to occur. Also, collapse strength of tubes is considered to be adequate even with significant diameter reduction, based upon available test data.

In summary, the assessment of postulated accidents as presented in the safety analysis report is not expected to change as a result of localized tube diameter reduction.

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SEPARATE COVER

II. Turkey Point Unit 4 Experience

During the scheduled refueling outage of Turkey Point Unit 4 in May 1975, eddy current inspections of the steam generators were performed in accordance with Regulatory Guide 1.83. The inspections revealed some continuation of the "thinning" which had been observed in earlier inspections. The "thinning" was in the area just above the tube sheet and has been attributed to residual phosphates in areas of sludge surrounding the tubes.

In addition to the "thinning", the eddy current data revealed a new phenomenon; anomalous signals at locations where the tubes pass through the tube support plates. These signals were interpreted as localized diameter reductions of the tube wall. All tubes inspected on the hot leg side showed some diameter reduction at most support plates. In addition, tubes on the cold leg side showed some diameter reduction in the region of the top support plate. It was also noted that the standard 700 mil (OD) eddy current probe would not pass through some of the tubes in the periphery of the steam generator. (See Figure 1) In most cases the probe was stopped at the second support plate entering from the hot leg and at the fifth support plate on the hot leg side entering from the cold leg.

Following refueling, Turkey Point Unit 4 returned to full power in late June. On July 28, 1975, a primary to secondary leak of 50 gallons per day was detected in steam generator B. On August 23, 1975, the plant was shut down to repair the leak, at which time the measured leak rate was 72 gallons per day. The leak was located in a peripheral tube (R44C55) on the hot leg side somewhere between the second and fifth tube support plates. (See Figure 2) This tube was plugged. Another hot leg tube (R33C58) having an indication of localized diameter reduction of the tube wall was removed to just above the first

T.1/1

support plate for laboratory examination. (See Figure 2) Limited eddy current examination around the leaking tube revealed several tubes through which the standard 700 mil (OD) eddy current probe could not pass.

The plant resumed operation in August after the steam generator repairs were completed. On August 23, 1975, a 60 gpd primary to secondary leak was detected in steam generator B. On September 1 the measured leak rate was 84 gpd; on September 12, 84 gpd, and on September 17, 96 gpd. The plant was shut down for steam generator repairs on September 21, 1975. The leak was located in a peripheral tube (R44C56) on the hot leg side adjacent to the leaking tube plugged in August. (See Figure 3) The leak was in the region of the second support plate, and the tube was plugged. During the shutdown a tube (R25C72) exhibiting diameter reduction indications was removed to just above the fifth support plate on both the hot and cold leg sides for laboratory examination. (See Figure 3) A limited eddy current inspection of B steam generator using a cross shaped inspection pattern confirmed widespread localized tube diameter reductions in both the hot and cold legs. In addition, the diameter reductions appeared to have progressed to the regions of additional support plates on the cold leg side. The eddy current inspection was performed to obtain a baseline reading on the size of the diameter reductions for comparison to future measurements. The unit resumed operation in late September and has been in operation since.

Laboratory examination of the tubes removed in August and September has revealed the following results:

1. Plastic deformation of the tube wall has occurred in the area adjacent to the 3/4" thick tube support plates, resulting in a constriction of the tube diameter in that area. The maximum reduction in diameter of the tubes examined in the laboratory is approximatelv 20 mils.

2. Metallographic examination and mechanical property tests indicate no loss in tube strength (See Section IV, Page 5).

3. Wall thinning to a maximum of 4 percent (2 mils) was observed on the OD of the tube surface adjacent to the tube support plate.

4. There is no evidence of stress corrosion cracking.

5. There is no evidence of deformation of the surface layer within the deformed region.

6. The green corrosion deposit found adhering to the removed tubes (Figure 4) has been found to contain sodium, chromium, nickel, iron, and phosphates as principal constituents. Detailed analyses are given in Table 1 along with the analyses of deposits removed from steam generators and model boilers operating with phosphate treatment only. The analyses support the conclusion that the source of the deposit is previous operation with sodium phosphate water chemistry. The analytical results are quite similar except for the higher sodium level for those samples taken from sodium phosphate treatment.

ANALYSIS OF GREEN DEPOSIT

A deposit has been-found on various tube samples removed from steam generators and model boilers. The composition of the greenish-white material follows:

	After AVT Operation		(Values in Percent) Removed After Phosphate Operation						
Analysis	FLA-4 R33-C58 *	FL'-4 R25-C72	NBK	MB	MB	MB	MB	MB	MB
PO_4	82	51.4	30.0	35.4	45.9	38.4	42.9	39.6	43.5
Ni	12	9.6	15.0	17.8	11.4	5.3	8.9	18.2	26.9
Fe	5	5.4	4.3	6.1	4.1	1.5	1.3	2.3	4.3
Cr	6	9.5	4.2	3.3	2.5	1.1	1.5	3.5	3.9
Na	3	4.5	16.0	10.1	15.6	17.7	15.5	10.2	12.0
Si	0	~0.9	1.2	8.1	10.0	2.4	1.9	3.6	2.5
Cu	0	0.2	0.3	0.6	1.1	0.3	0.5	0.2	0

*Best estimate from relative values of EDX analysis.

TABLE 1

TURKEY POINT UNIT 4
SG B HOT LEG
X - TUBES THROUGH WHICH THE 700 MIL (OD)
EC PROBE WOULD NOT PASS

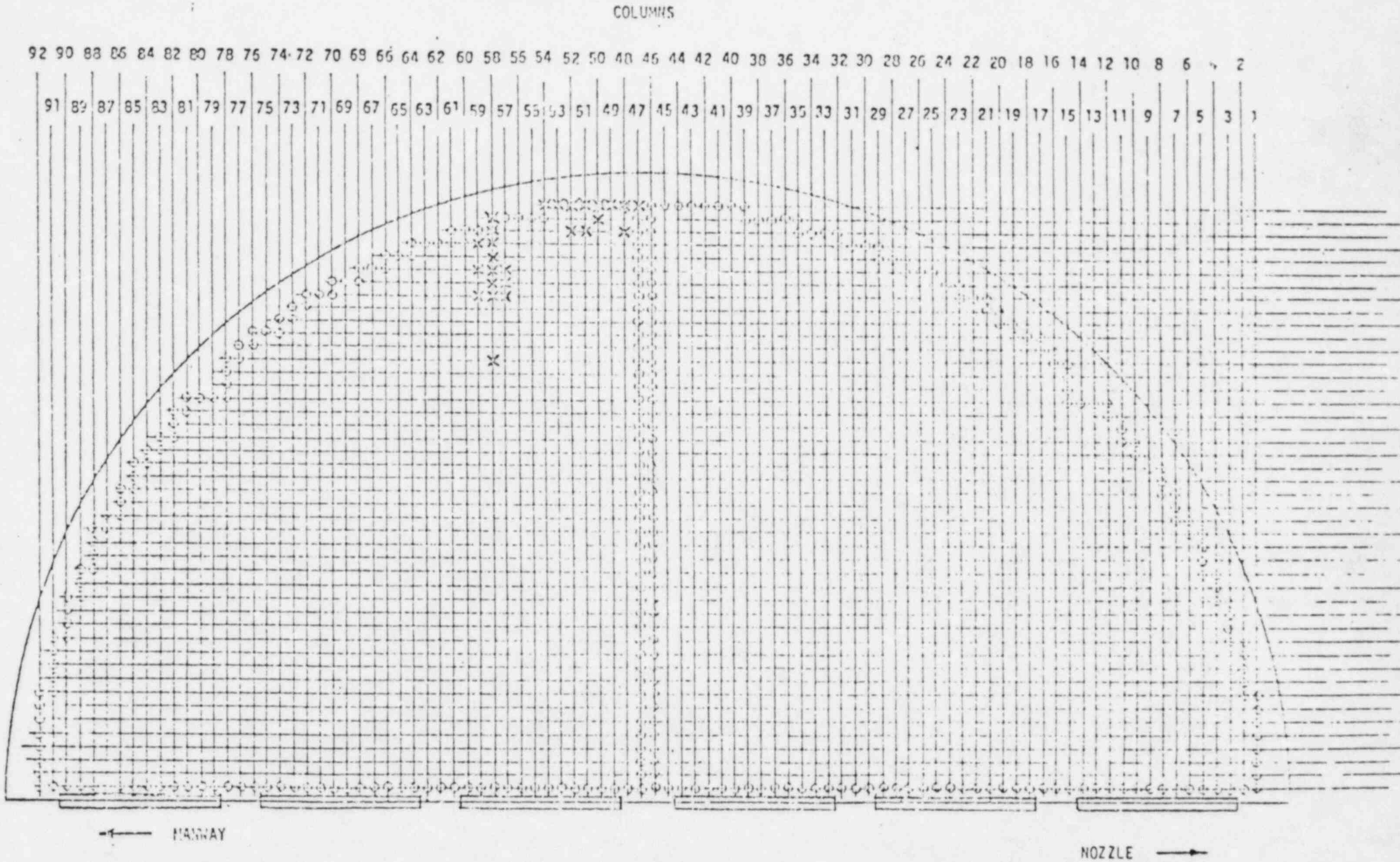


FIGURE 11

44 SERIES STEAM GENERATOR

TURKEY POINT UNIT 4
SG B HOT LEG
X - LEAKING TUBE 8/75
O - REMOVED TUBE 8/75

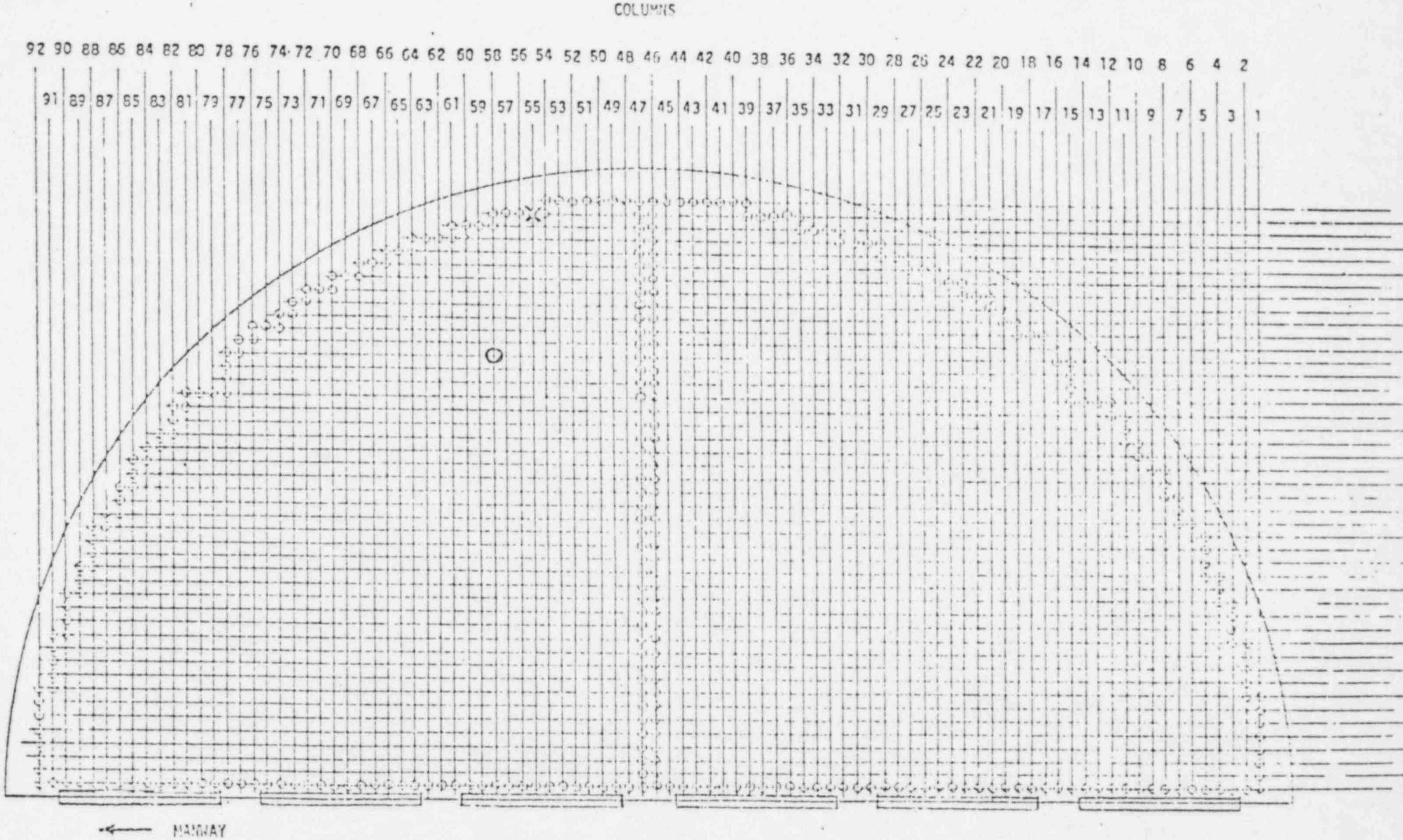


FIGURE 2

44 SERIES STEAM GENERATOR

TURKEY POINT UNIT 4
SG B HOT LEG
X - LEAKING TUBE 9/75
O - REMOVED TUBE (ALSO REMOVED ON COLD LEG) 9/75

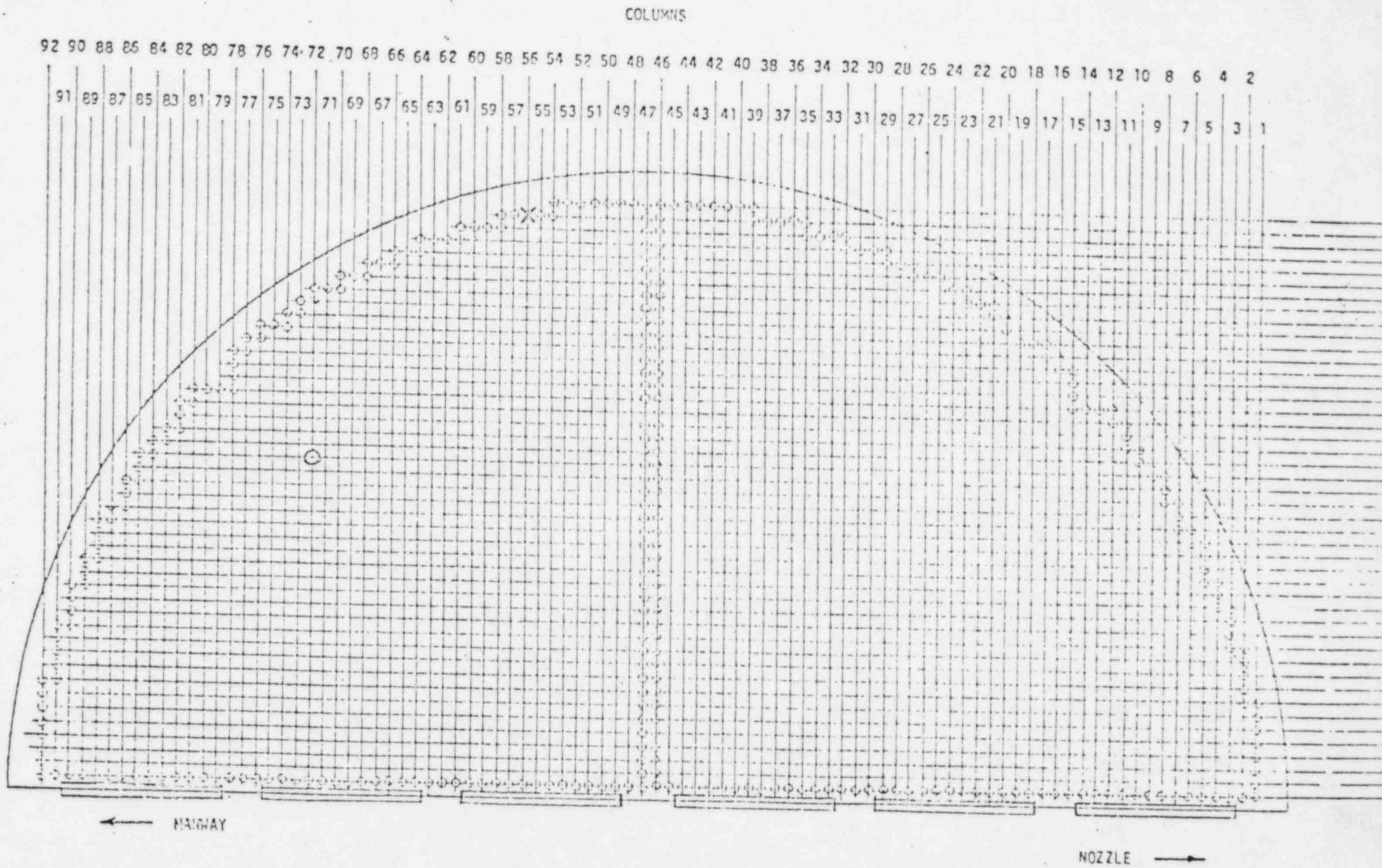
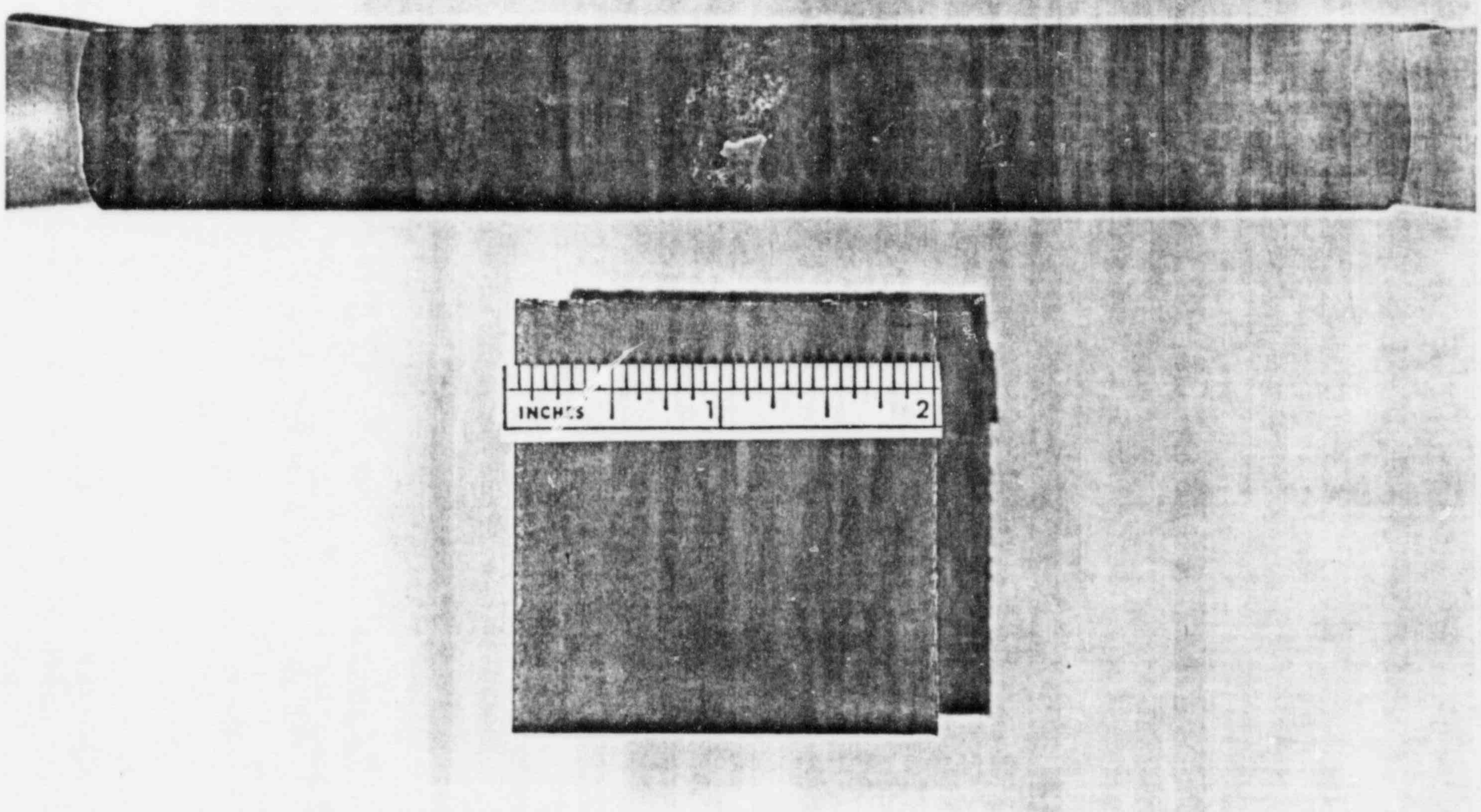


FIGURE 3

44 SERIES STEAM GENERATOR



VIEW OF SECTION FROM TUBE R25C72 SGB TURKEY POINT UNIT 4
COLD LEG 4TH SUPPORT LOCATION

FIGURE 4

III. Surry Unit 1 Experience

On September 23, 1975, a 125-150 gpd primary to secondary leak was detected in steam generator A of Surry Unit 1. The plant was shutdown 3 days later with the leakage rate holding steady at 125 gpd. Three leaking tubes were identified in Row 2 (one row away from and parallel to the divider lane) on the hot leg side. The leaking tubes (R2C31, R2C47, R2C63) were all in the area of the solid webs between the flow slots in the tube support plate. (See Figure 5) The leaks were located in the region of either the sixth or seventh (top) tube support plate. Two non leaking tubes (R20C74, R21C74) on the cold leg side were removed for laboratory examination. (See Figure 6) One was removed to just above the fifth support plate, the other was removed to just above the sixth support plate.

Extensive examinations of the Surry Unit 1 steam generators were conducted in November 1975 in an effort to obtain better information on the tube diameter reduction phenomenon. In addition to eddy current testing of the tubes, a visual inspection of the U-bend region of one steam generator was conducted. Additional visual inspections of the tubes and tube support plates were conducted through hand holes and "hillside" ports (non-radial penetrations) on all three steam generators.

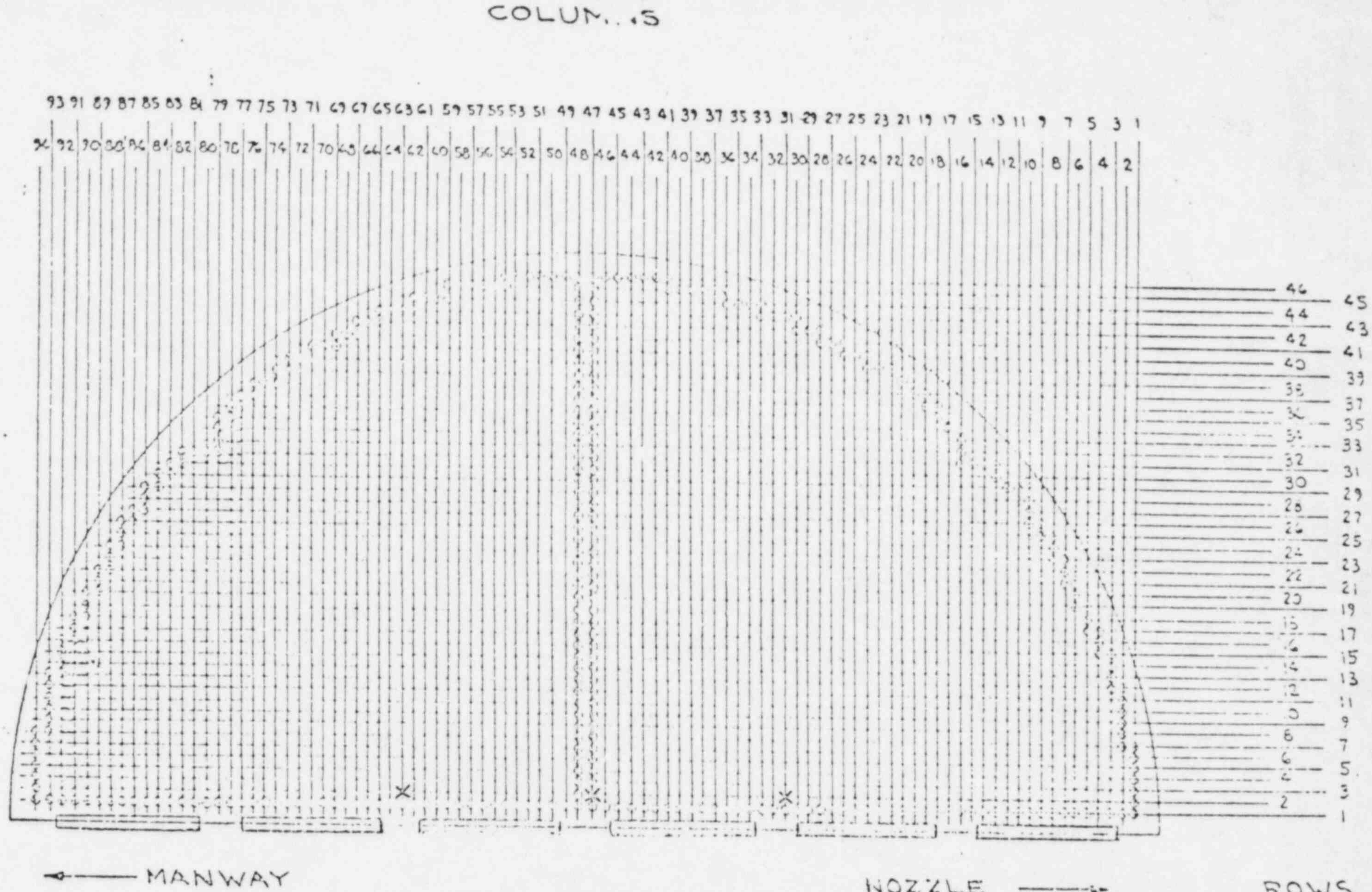
The eddy current testing showed that all inspected tubes on both the hot and cold leg sides exhibit some local diameter reduction at most support plates. The extent of this deformation is mostly in the range of less than 20 mils reduction in diameter, but some signals indicate as much as approximately 50 mils reduction in diameter. Deformation tended to be greatest in those rows closest to the divider lane (rows 1 through 3) where gauging of the leaking tubes indicated on inside diameter reduction to 0.550 inch (at the second support plate). No significant differences between the hot and cold legs were noted.

Visual inspection of the U-bend region and top of the tube bundle of C steam generator revealed no unusual conditions near the anti-vibration bars (AVB's) such as shifting, visible corrosion, or tube damage. There was a hard deposit noted at the intersection of the AVB's and the tubes. There was no apparent shifting of the uppermost (7th) tube support plate, with normal clearance between the support plate and wrapper evident. The spaces between the tubes and the tube support plates appeared to be filled with a deposit as no visible gaps were apparent. The tube bundle appeared to be in its as-built condition with no visible bending, bowing, or misalignment of tubes in the U-bend region.

Additional visual inspections through hand holes and "hillside" ports of A, B and C steam generators were conducted. In-plane "hour-glassing" of the rectangular flow slots in the support plates in all three steam generators was noted with the least amount occurring in steam generator B. (See Figures 7, 8, and 9). No signs of abnormal surface scale, deformation, cracks, or out of plane distortion were evident. Deposits were evident in the spaces between the tubes and tube support plates with no clearance between the tube and tube supports indicated. Looking down the divider lane, the tubes appear straight from the top of the tube sheet to approximately 15 to 20 inches below the first support plate with gradual bowing from there to follow the contour of "hour-glassing" in the support plate flow slots. (Figure 9) Uniform spacing was observed between adjacent columns of tubes.

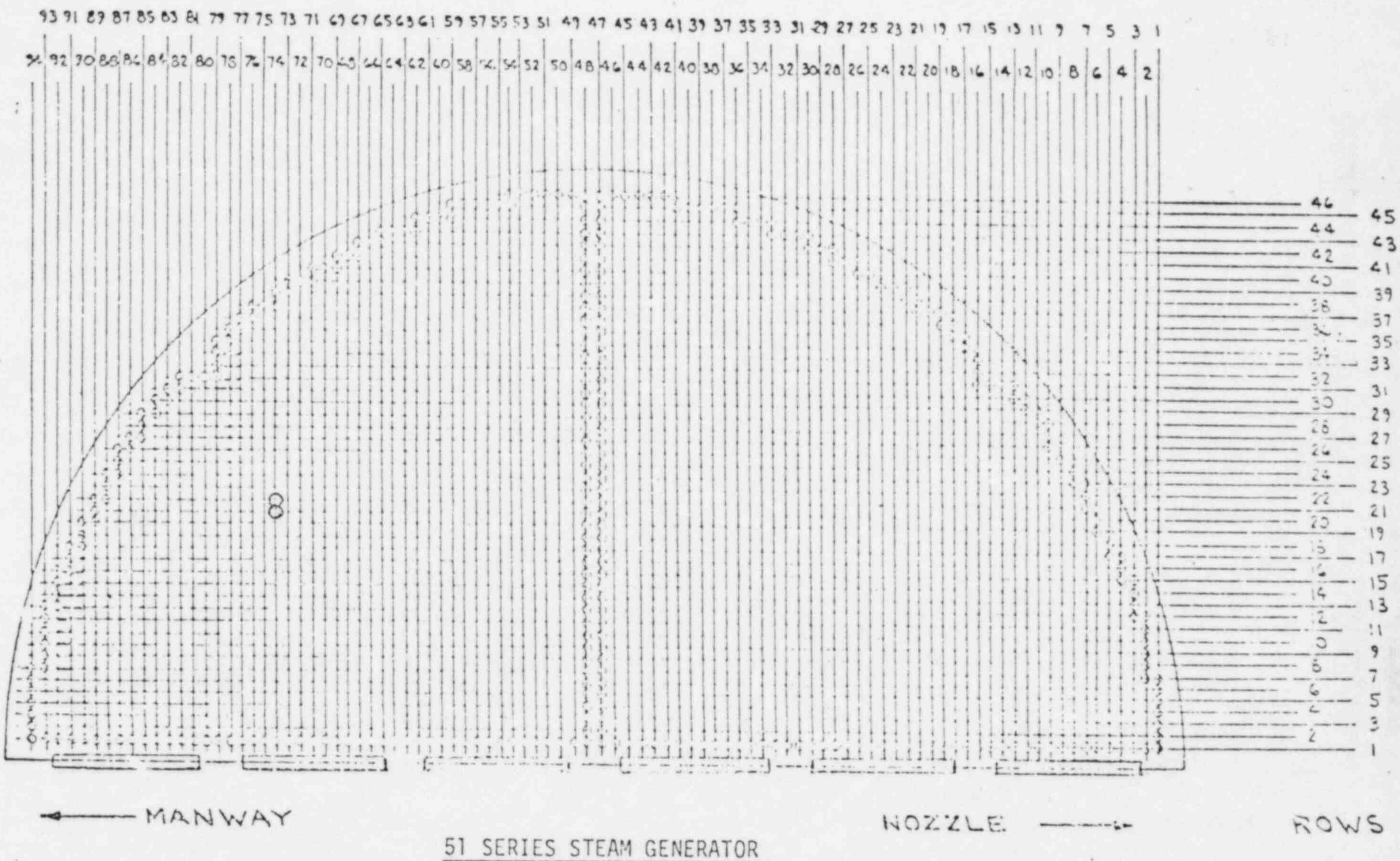
Surry Unit 1 resumed operation on December 8, 1975. A low level (less than 1 gpm) primary to secondary leak was detected in steam generator A on December 10, 1975 and the plant was shut down for repairs on that date. The leaking tube (R2C48) was adjacent to one of the tubes which was leaking and plugged in September. (See Figure 10) The leak was located in the region of the second tube support plate. Plant operation resumed following completion of steam generator repairs, with power level escalated to 100% on December 17, 1975.

FIGURE 5



51 SERIES STEAM GENERATOR

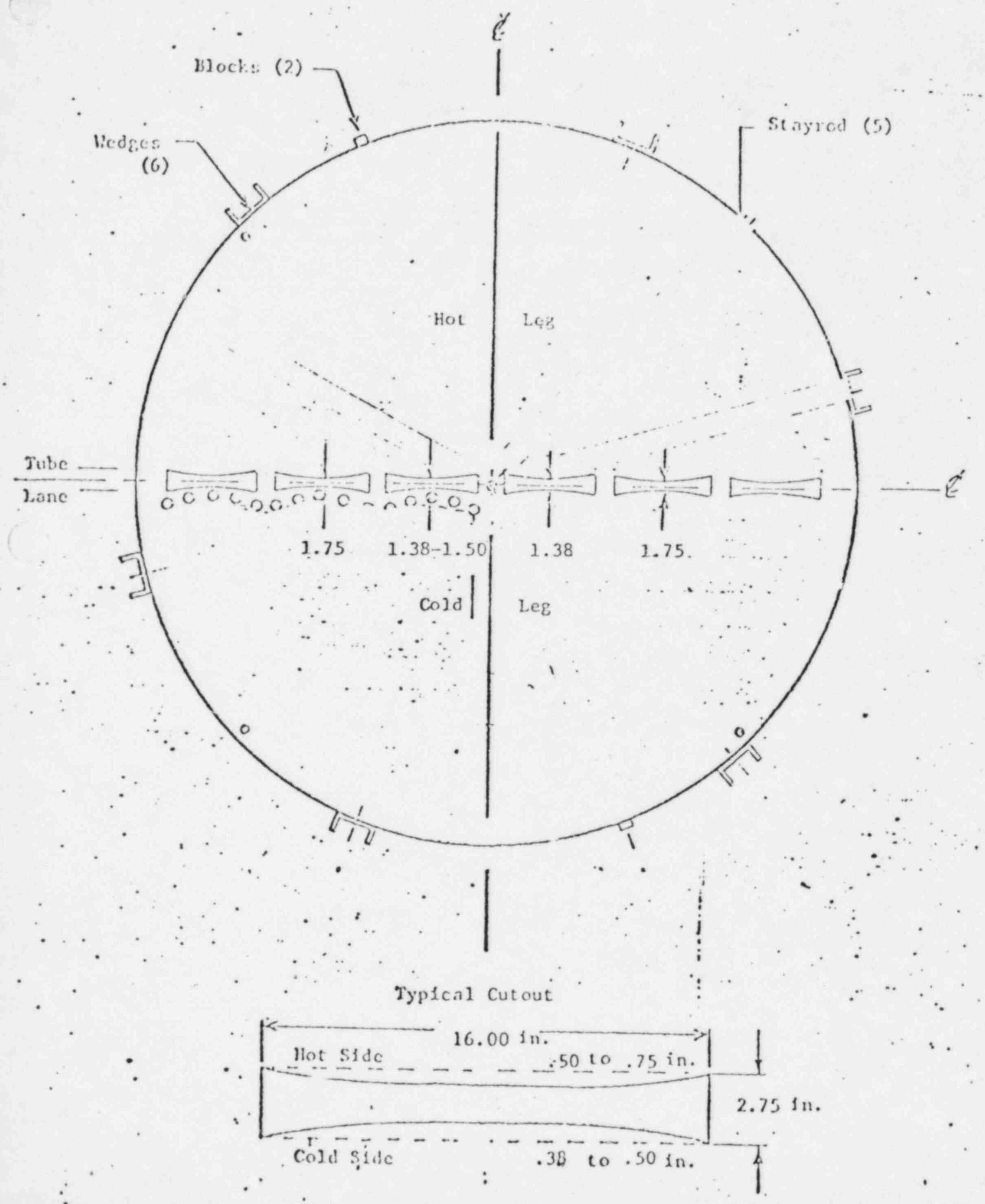
SURRY UNIT 1
SG A HOT LEG
✕ - LEAKING TUBES 9/75

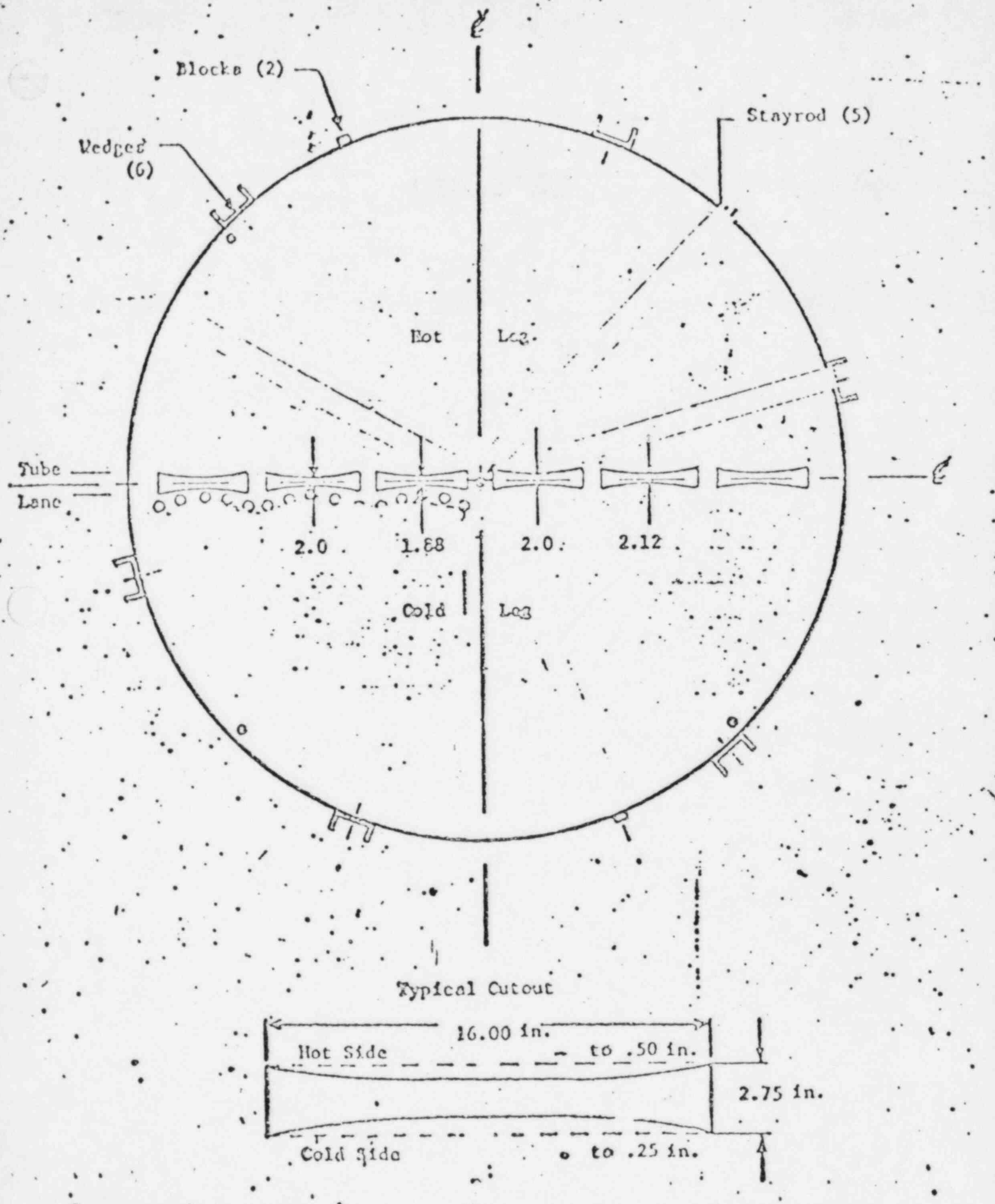


SURRY UNIT 1
SG A COLD LEG
O = REMOVED TUBES 9/75

FIGURE 6

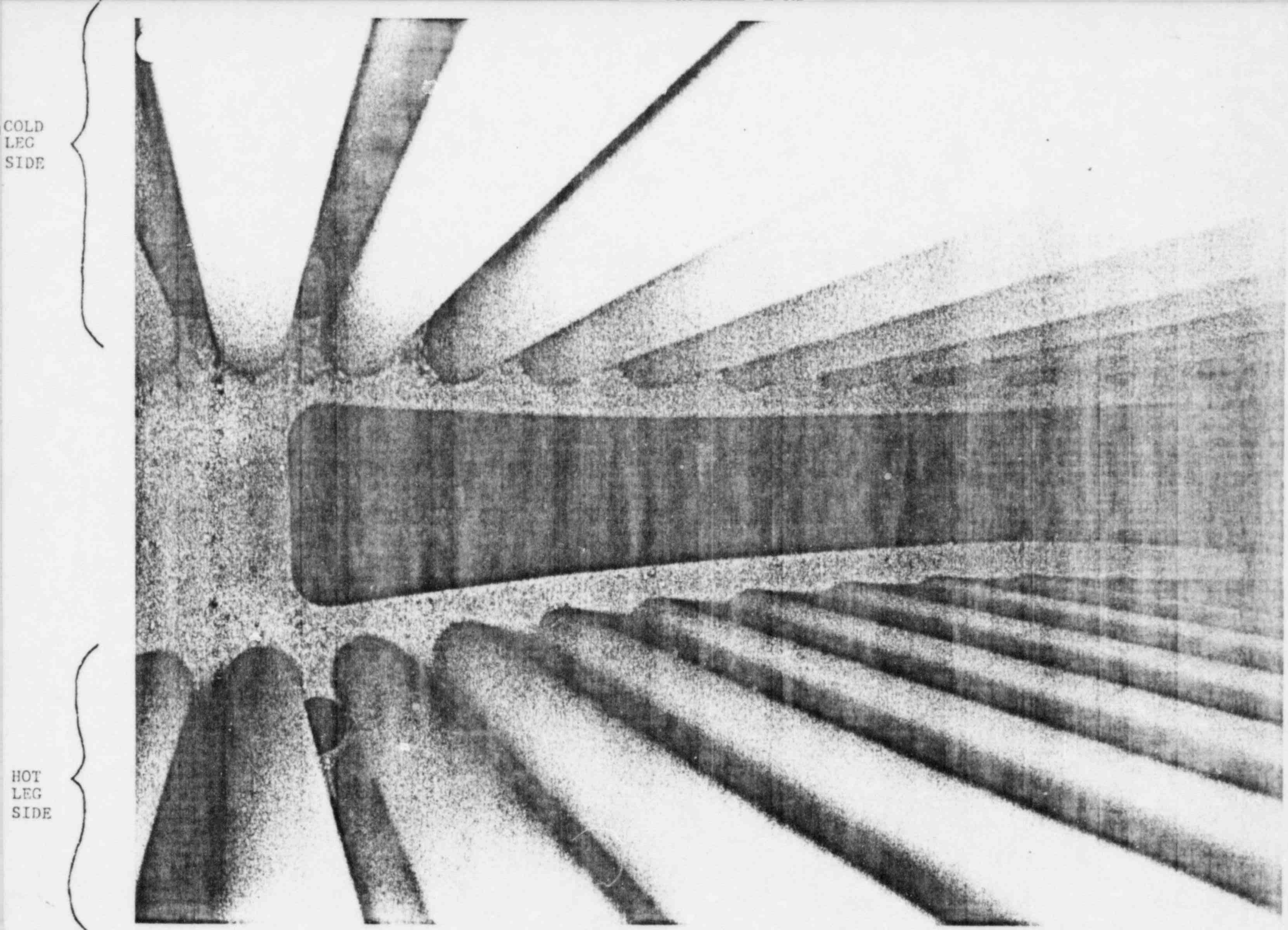
Surry 1 - S/G 'A' Cutouts





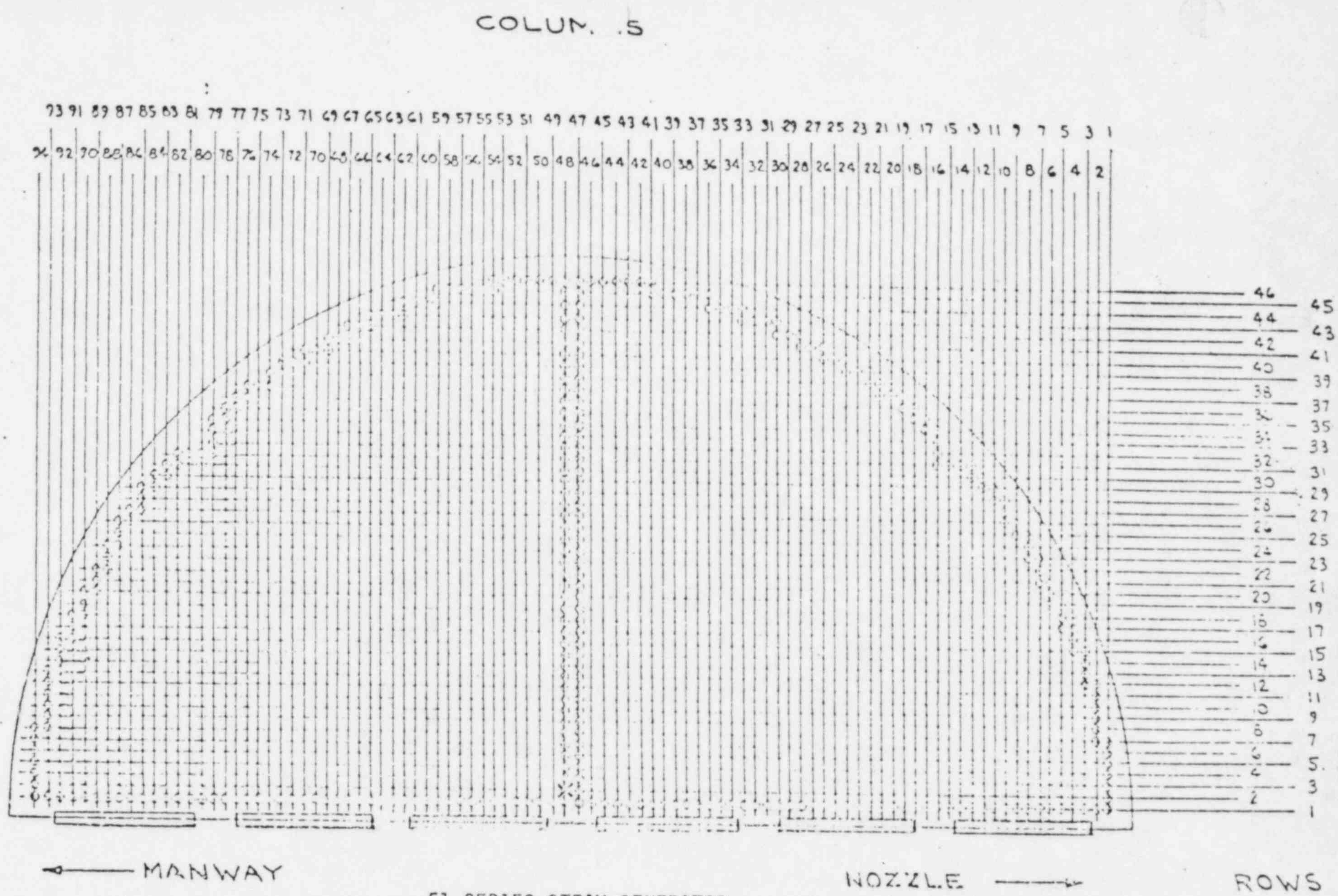
Blocks (2)
Stayrod (5)
Wedges (6)
Hot Leg
Tube Lane
2.0
1.88
2.0
2.12
Cold Leg
Typical Cutout
16.00 in.
Hot Side
to .50 in.
2.75 in.
Cold Side
to .25 in.

FIGURE 8



VRA S/G A OCTOBER 31, 1975
FIRST SUPPORT PLATE AND INNERMOST TUBE ROWS

FIGURE 9



51 SERIES STEAM GENERATOR

FIGURE 10

SURRY UNIT 1
SG A HOT LEG
X - LEAKING TUBE 12/75

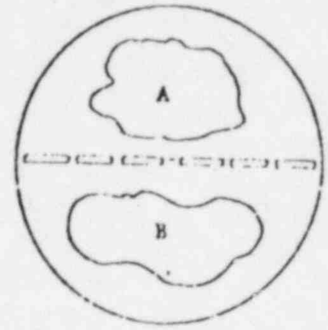
IV. Postulated Mechanism

Introduction

This section summarizes an oral presentation to the NRC on November 21, 1975, concerning the Steam Generator tube deformation (diameter reduction) phenomenon. The evaluation and tentative conclusions as to the nature of this phenomenon are, necessarily, preliminary and are based upon currently available information. Additional data is required to substantiate, or possibly modify, our postulated mechanism for tube deformation. Essential information, which is not yet available, includes extensive eddy current data, and the results of a detailed examination of an intact support plate sector (which contains undisturbed deformed tubes) removed from an operating unit.

Summary

The current postulate which best satisfies the physical evidence is as follows: A substance fills the gap between the tube and tube support plate hole, denting the tube and dilating the hole (mainly through stretching of the ligaments which surround the tube hole). The pressure necessary to do this appears to be about 6,000 to 7,000 psi. As every tube hole dilates, the entire projected area of the tube support plate increases. At rigid regions of the support plate, the in-plane displacement of the plate is locally constrained. The rigid regions, termed "hard spots" are shown in Figure 11. "Hard spots" do not contain the array of interstitial flow holes found elsewhere in the support plate. The effect of constraint at the "hard spots" is to concentrate the deformation of tubes, tube holes, interstitial flow holes and flow ports (located at the center of the tube bundle). The deformation observed at "hard spots" is more extensive and of greater magnitude than it is elsewhere in the support plate.



At each instant in time, the distribution in area A of tube deformation is anticipated to be Gaussian, with some Standard Deviation. The "hard spot" locations will also reflect this pattern, but in addition there will be extensive deformation. Similarly for the cold leg, but area B will probably have a different mean and Standard Deviation.

As more data is gathered about the angular profile of a tube deformation, this distribution is also expected to be Gaussian.

Background and Initial Hypotheses

Immediately after the discovery of local tube deformations in the area of the tube support plates, an extensive analytical and experimental investigation was started. These analyses were based on several mechanisms, all of which were believed to locally reduce the outside diameter of the tube as shown in Figure 12. Of these mechanisms, the two most probable were thermal ratcheting and crud deposition. The thermal ratcheting was assumed to occur with an internal to external pressure difference of 1500 psi and an alternating tube OD temperature from approximately 590° to 490° (ΔT varies from 0 to 100°F as shown in Figure 13). This mechanism alone, however, was shown not to be credible both experimentally and analytically. The crud deposition model, however, did show credibility in that significant deformation could be obtained in relatively few cycles. The loading condition for one form of this model is presented in Figure 14. T[illegible] crud deposit in the tube to tube support hole annulus was assumed to be incompressible and with a zero coefficient of thermal expansion would continue to fill the gap after each cycle. The total radial deformation of each cycle was 1.7×10^{-4} inch. This or a similar mechanism will result in a net pressure on the tube outside diameter and the tube hole inside diameter.

Determination of External Pressure to Yield Tube and Support Plate

The tube ΔP can be approximated with the following relationship:

$$\Delta P = \frac{\text{(Yield Strength)}}{\left[\frac{\text{Mean Radius}}{\text{Wall Thickness}} + \frac{1}{2}\right]}$$

$$\Delta P = \frac{(44,660)}{\left[\frac{.4125}{.045} + \frac{1}{2}\right]} = 4620 \text{ psi}$$

This pressure difference will cause yielding and a permanent deformation of the tube. With the internal (Reactor Coolant) pressure at 2,235 psi, the outer pressure for a ΔP of 4,620 is 6,855 psi. The 6,855 psi represents the total "hydrostatic" pressure acting on the O.D. surface of the tube and would include the possible effect of secondary side pressure. When 6,855 psi is considered to be applied to the outside of the tube, it is also applied to the I.D. surface of the hole in the support plate under the assumptions that the crevice is filled with an incompressible substance.

The stress across the minimum width membrane in the support plate is calculated to be 35,718 psi, this is greater than the ASME code value at 500°F for the support plate material of 24,500 psi. Therefore, the tube support plate will receive a permanent distortion. Actual in-plane distortion of the tube support plates has been observed which tends to support the latter hypotheses. (Figure 9) Further details of the analysis are presented in Attachment 1 together with a discussion of further analysis to be conducted in near future of additional data as obtained from inspections of steam generator.

Laboratory Tests of Tubes

Mechanical tests are being run to simulate severely deformed tubes and to evaluate the hydrostatic collapse characteristics of tubes with given configurations of deformation which are representative of tube conditions observed in steam generators at Surry Unit 1 and Turkey Point Unit 4. The results of tests performed to date, and the test program outline is contained in Attachment V-3.

Mechanical tests have also been performed on a tube removed from Turkey Point Unit 4 which exhibited deformation. These results indicate typical mechanical properties and good elongation. Results for tube R24 C72, Section 2B4 (cold leg) follow:

yield strength (0.2%)	42,753 psi
ultimate strength	89,930 psi
elongation	35.4%

Tentative Conclusion

The deformation of tubes and support plates observed to date is the result of the growth of deposits and/or corrosion product within the support plate in those steam generators which have operated for extended periods on phosphate water chemistry and have switched to All Volatile Treatment (AVT). A substance accumulates in the annular gap between tube and support plate hole. The substance exerts force directed radially inward on the tube and outward on the support plate when the gap is filled. The result is a reduction of tube diameter and an increase of support plate hole diameter. At rigid areas of the support plate, the deformation is most extensive and least symmetrical (about the axes of the tube and of the hole).

Future Planned Actions

In an effort to obtain more factual information on the tube deformation phenomenon, measurement and sampling programs have been proposed and are discussed in Section VI, Diagnostic Programs. Briefly, they consist of an extensive eddy current program using a low gain, small diameter (if required) 400 KHZ probe. This program concentrates the examination to the areas of the hard spots and all support plates. There has also been a procedure developed for the removal of a section of tube support plate with two sections of tubes and a circulation holes in tact. This section would be removed from the area of the largest U-bends and on the rim of the tube support plate.

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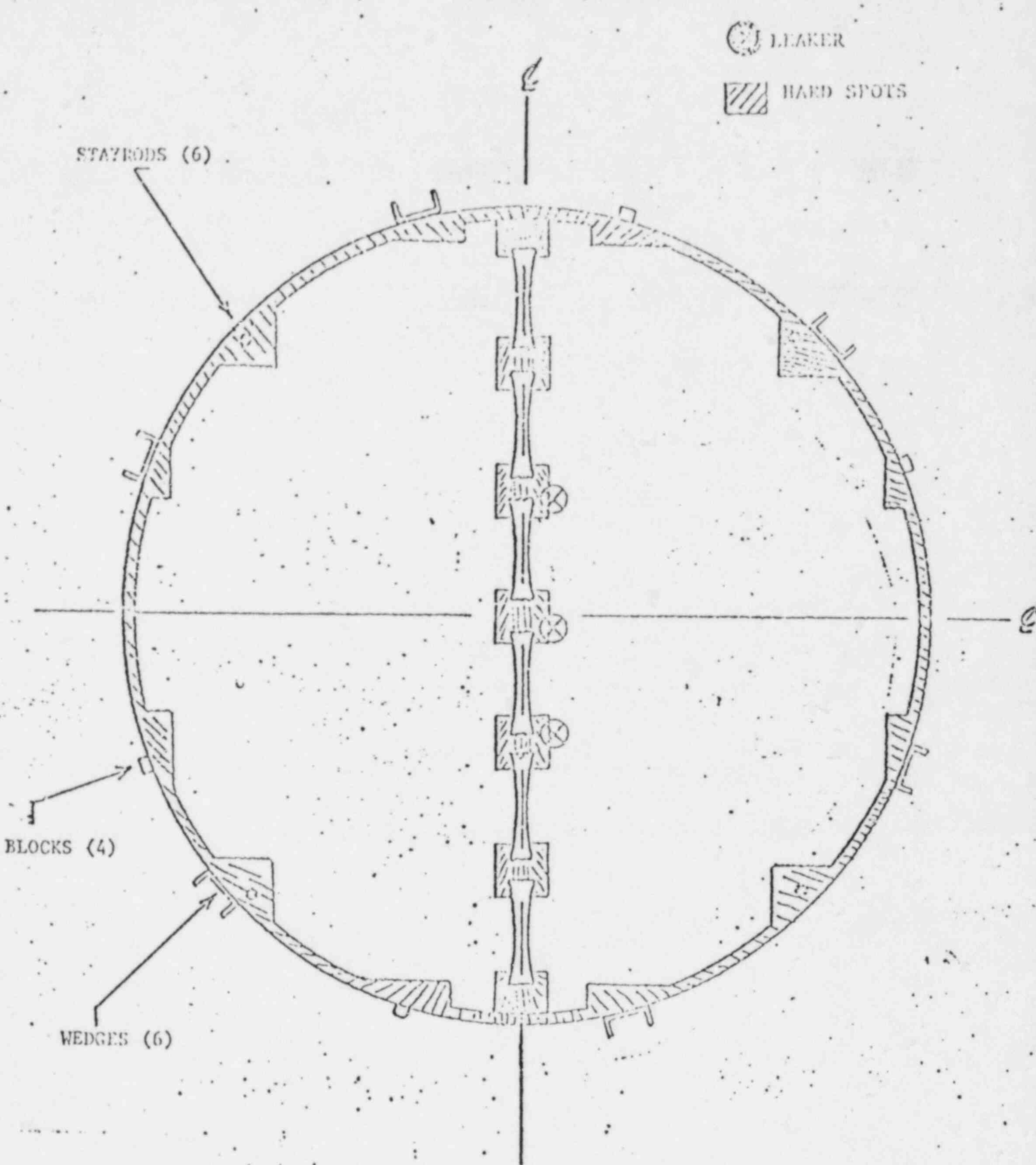
SURRY #1:



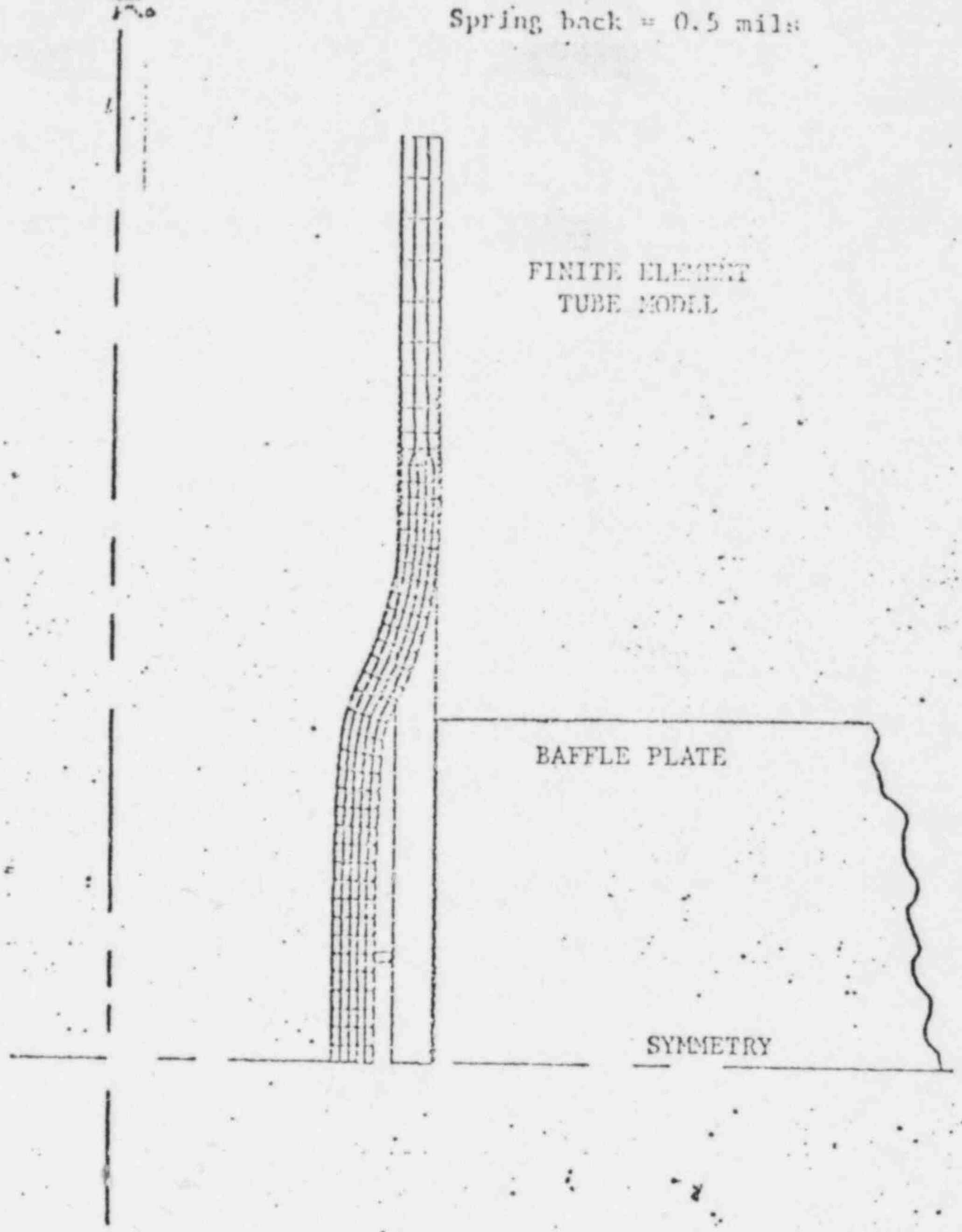
TYPICAL TUBE SUPPORT PLATE HARD SPOTS

FIGURE 11

Free radial growth of tube = 2.0 mils

Compressive stress at interface = 5000 psi

Spring back = 0.5 mils

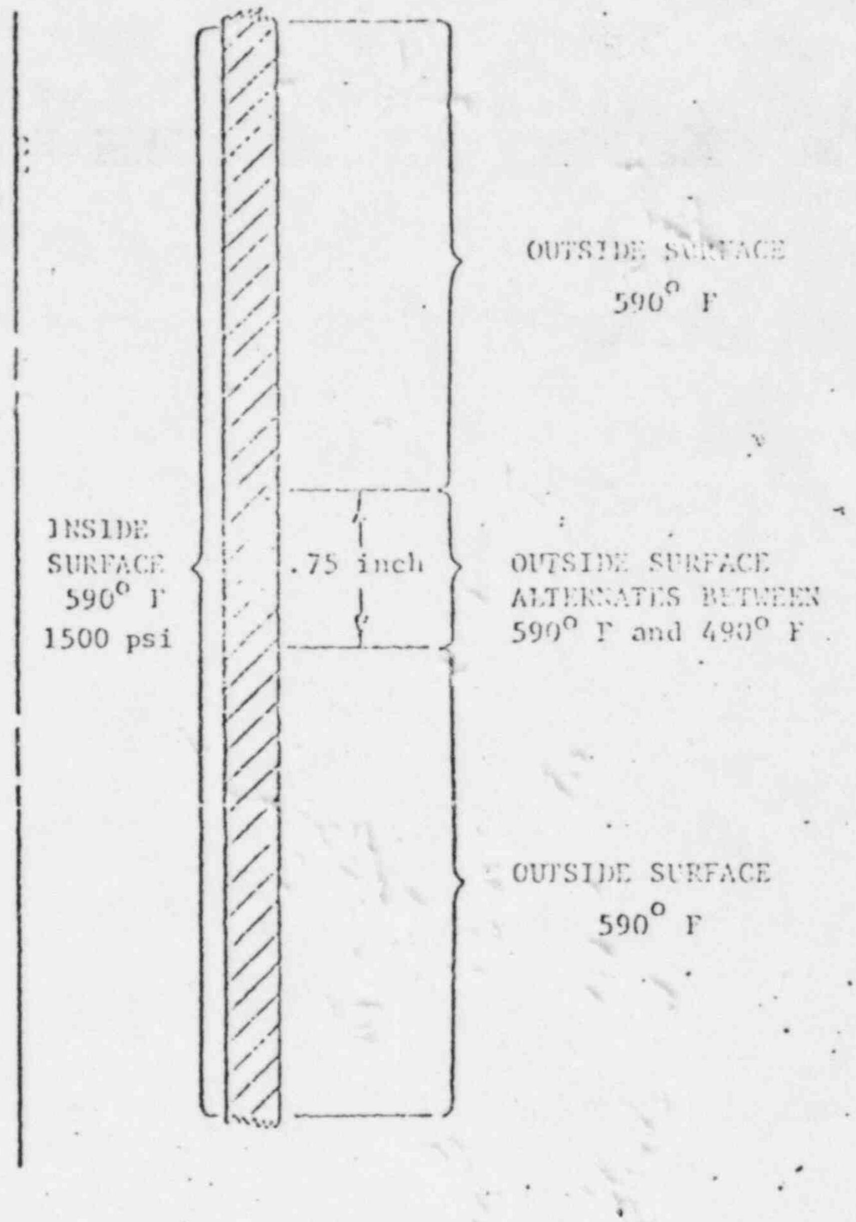


TUBE/BAFFLE INTERACTION CYCLE

FINAL TUBE POSITION AT ROOM TEMPERATURE

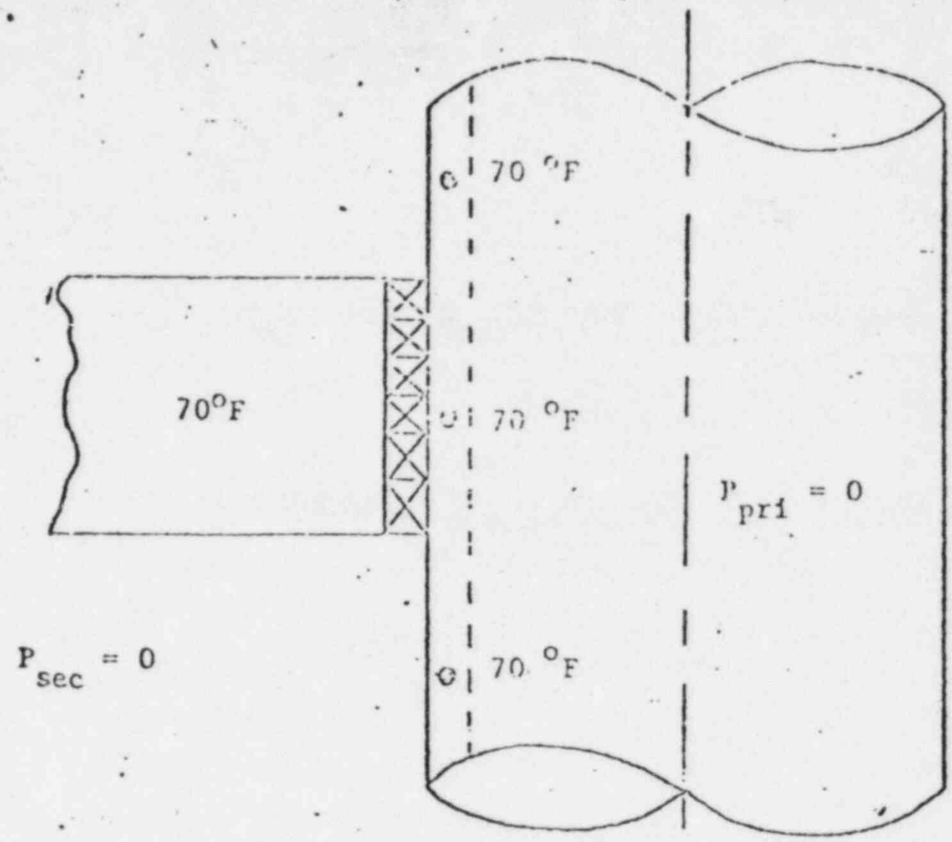
- 1 cycle -

FIGURE 12



Tube Wall Temperature History

FIGURE 13



TUBE/BAFFLE INTERACTION CYCLE

ASSUMED HOTLEG CONDITIONS

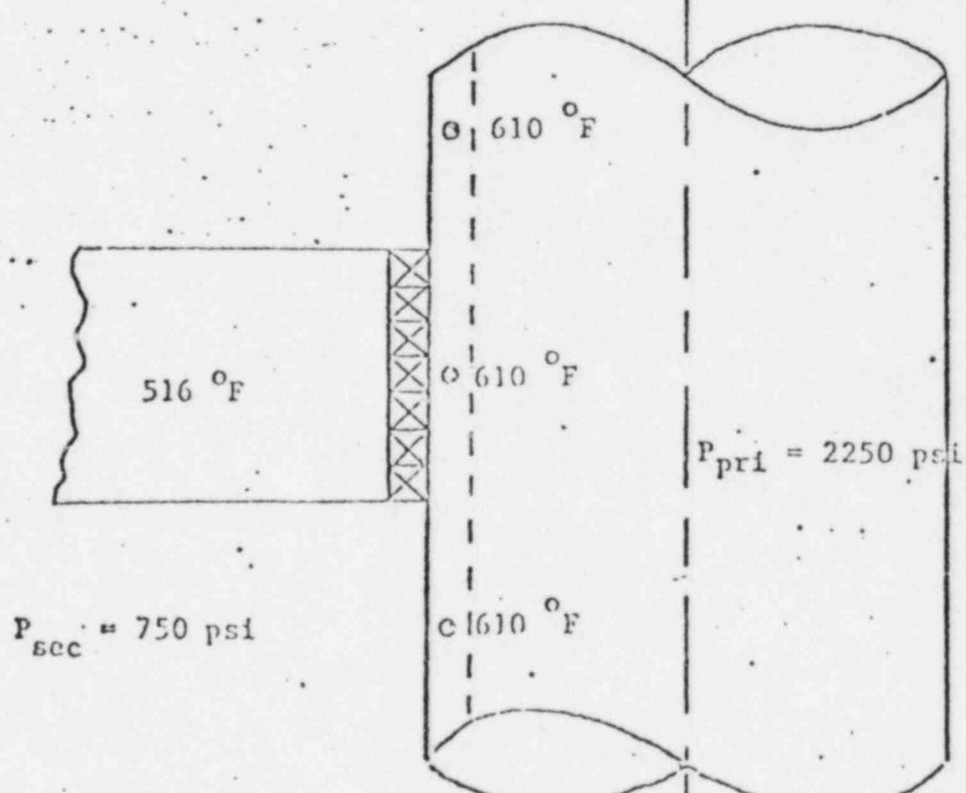


FIGURE 14

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AND IS UNDER

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V. Safety Evaluation

A. NORMAL OPERATION

Leak Detection

Primary to secondary leakage through defects in the Steam Generator tubes is detected by means of radiation monitors located at the condenser air ejector (sensitive to Xenon) and the blowdown radiation monitors (sensitive to Iodine and other radionuclides). Once detected by these monitors, batch sampling is performed to accurately determine the leakage rate, as derived from the radioanalysis of the sample and an analysis of the reactor coolant for the same radionuclides. Usually, Iodine-131 and Iodine-133 are analyzed in the short term. Tritium buildup may be used for longer term confirmation of the leakage rate. Normally, primary to secondary leakage is detected at a low rate and plant operations are continued until the next scheduled shutdown or another convenient time which may involve operation for extended periods with leakage within the Technical Specification limits.

Recent events at Turkey Point Unit 4 and Surry Unit 1 discussed in Sections II and III illustrate typical examples of leak detection and orderly shutdown for repair.

Calculated Stresses With Deformed Tubes for Normal And Upset Conditions

The conditions of loading for which the tube is analyzed are lumped into 100% load operation plus the seismic Operating Basis Earthquake (OBE) loading. The analysis performed is applicable to 44 and 51 Series steam generators both with and without secondary side mechanical modifications.

In the determination of hoop stress due to internal pressure, the effect of tube deformation has not been addressed, directly, because the reinforcing effect of the substance within the crevice causes the local hoop stress to be negligible. The effects of diameter reduction and wall thinning on bending stress have been addressed in the choice of dimensions used for this analysis.

Under Normal and Upset Conditions, the range of stress classified as primary plus secondary plus peak is evaluated according to the rules of Section III of the ASME Code.

These stresses considered result from the following loadings: pressure, mismatched thermal growth of the tube legs, radial and "hot spot" thermal gradients, and geometric discontinuities.

Each of the contributing factors was calculated for its own limiting case and then all cases were superimposed, thus resulting in an all encompassing case. Stresses were calculated for 100% load (ΔP = 1465 psi). For cases involving internal pressure, tube diameter reductions were considered to be symmetrical about the tube axis, unless otherwise noted.

For pressure the tube wall was considered to be thinned to 0.030 inch from the 51 Series tube nominal dimension of 0.050 inch. The "fixity" of the tube in a support hole increases the bending stress for certain loading conditions. The resulting bending stress, due to pressure, is $\pm$34.5 Ksi (for the largest OD tube) and is classified as a secondary stress. The shear stress is 1.6 Ksi and is negligible by comparison.

The calculated stresses are tabulated in Table 2-1 of Attachment V-1. The maximum stress intensity for the worst combination of events considered includes OBE loads superimposed on 100% load normal operating stresses. This conservative combination results in a stress (130.2 Ksi) which exceeds the ASME Code Section III, 3 S_m value (based on minimum specified properties) for SB-163 Inconel, of 69.9 Ksi. Therefore, in accordance with Para. NB-3228.3 of the code, a simplified elastic-plastic approach was taken. The alternating stress thus determined was used in the fatigue evaluation of a tube with diameter reduction. The result of these assumptions is a conservative estimate of the fatigue life of such a tube at the worst case location in the bundle (smallest U-Bend Tube) of 180 full cooldown cycles, and including the Operation Basis Earthquake seismic loading. This is 90% of the specified number of Heatup and Cooldown cycles given in the equipment specification (200 cycles). While at first, the result may appear to reflect a loss of usable fatigue life, the assumptions upon which this result is based should be considered. The tube wall is considered thinned to 0.030 inches (60% of the nominal) as well as reduced in diameter. Under these assumptions the fatigue life is conservatively demonstrated to be 90% of the original design requirement. Certainly, the above discussed result does not give rise to short term concern for its fatigue life. A detailed discussion of the analyses of Normal Operating and Upset Conditions is contained in Attachment V-1. "ASME Section III Evaluation Of Design, Normal Upset and Test Conditions of Tube Resulting From A Deformation Phenomenon."

B. FAULTED CONDITIONS

Loss of Coolant Accident

The LOCA plus SSE faulted conditions are analyzed in WCAP 7832 (and additional information, October 1974) for an as-manufactured steam generator. Therein, it is shown that the most significant stress (bending) results from the rarefaction wave traversing the U-bend section of the tube bundle, early in the LOCA event. Section 6 of Attachment V-2 discussed the changes in the strain configuration of the U-bend region of the bundle as a result of hard deposits in the tube/support plate annulus. The tubes which have deformed are assumed to fixed to the support plate holes as opposed to the analysis of WCAP 7832 wherein they were assumed to be simply supported and free to rotate within the holes. On Page 6-4 of Attachment V-2, the bending moment at the top support plate is shown to exceed the collapse moment of the section. Thus all rotation is considered to occur at the top support plate as a plastic hinge is formed in the tube. The strain is calculated to be 0.067 in/in (Page 6-5 of Attachment V-2). The ASME Section III strain limit for this condition is 0.7 S_u (65.4 Ksi). At temperature, 65.4 Ksi corresponds to a strain of 0.075 in/in. Thus, the calculated strain for a LOCA plus SSE is less than the code limit for a faulted condition.

Main Feedline Break

For steam generators which might experience tube diameter reduction (feedwater ring 44 and 51 series) the most severe secondary break accident is the main steamline break. In attachment V-2, the most severe feedwater break conditions were analyzed and found to be less severe than the Main Steam Line Break.

Main Steamline Break

Assumptions used in this analysis are a reduced secondary side level resulting from an instantaneous loss of load (from 100%). Thus the initial condition is no-load and water level below the top of the tube bundle. Initial temperature is assumed at 547°F and pressure is assumed to be 1020 psia above the water level on the secondary side just prior to the Steam Line Break.

A modified hydraulic analysis was performed to take into account the plugging of the annuli around each tube at each support as well as a reduction in the flow area of the interstitial flow holes and flow ports. During blowdown, uniform pressure drops occur across each support plate. The maximum ΔP is 9.3 psi across the top support plate. The results are discussed in detail in Sections 2 and 6 of Attachment V-2. The stresses on the top support plate due to blowdown loading, as well as the stresses on the wrapper hold down attachments are analyzed in Sections 4 and 5 of Attachment V-2 and found to be acceptable. The stresses on the tubes resulting from the blowdown effects on the support structure are found in Section 6 to be within Code requirements for a Faulted Condition.

C. EVALUATION OF COLLAPSE PRESSURE

During the latter stages of a loss of coolant accident, the pressure differential across the tube wall reverses as the primary system blows down. Eventually, a quasi-steady state equilibrium is reached with a secondary to primary differential of 1000 psi, or less. The effect of diameter reduction or tube deformation on collapse strength was evaluated.

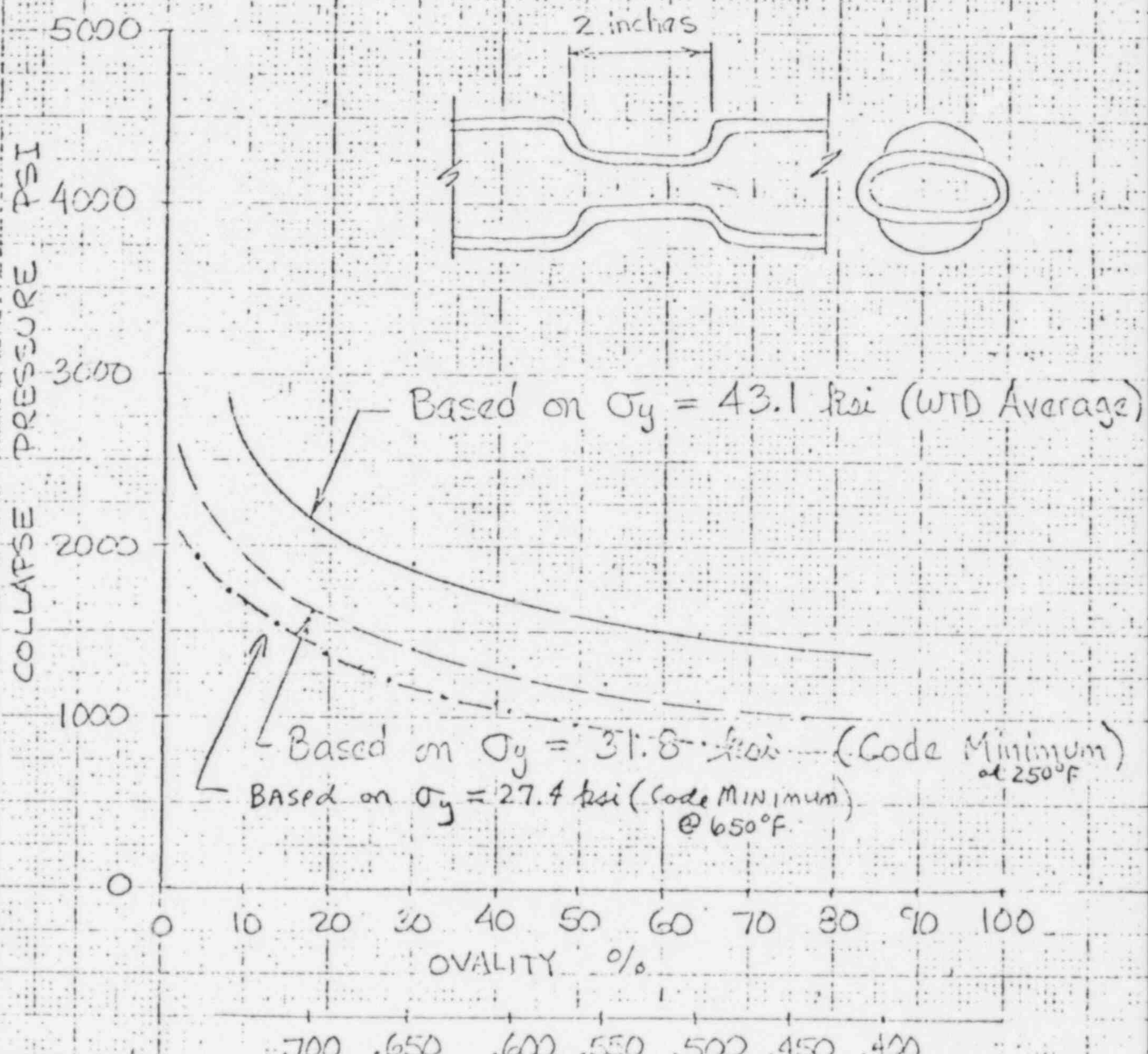
Figure 15 is a plot of hydrostatic collapse pressure versus ovality for the illustrated configuration of severe tube deformation. Data was obtained for values of ovality from 4 to 76 per cent. The additional curves in the plot were added for (1) measured average mechanical properties and (2) ASME Code minimum specified properties at 250°F and at 650°F. As shown on Figure 18 at 650°F, even the most severely ovalized tubes tested did withstand greater than 1000 psi. For minimum specified mechanical properties at 650°F, the collapse pressure approached 1000 psi at 40% ovality. Data from materials testing performed at the Tampa manufacturing facility indicated that approximately 92% of the tubes exhibit properties which fit the average curve rather than the minimum specified properties.

Attachment V-3, "Preliminary results of tests to evaluate the Mechanical Integrity of Dented Steam Generator Tubes" describes preliminary tests to study the mechanism of denting and outlines the test program concerned with hydraulic collapse strength and the effects of bending deformed tubes. The latter test sequence is designed to simulate the effect of the rarefaction wave in The Loss of Coolant Accident (LOCA) followed by a collapse pressure test to simulate primary blowdown.

A straight section of a tube removed from Turkey Point #4 was tested in the collapse mode as described in attachment V-3. This tube which was deformed 0.017 to 0.026 below original diameter withstood substantially greater external pressure than it would see in a postulated LOCA blowdown (greater than 3500 psi with no failure).

Based on these results, it is apparent that the tubes can be severely distorted and still maintain sufficient collapse strength to withstand the external pressure differential which develops in the latter stages of a LOCA.

FIGURE 15



Collapse Pressures of Dented 7/8" Tubes

Temperature	250°F & 650°F
Wall Thickness	.045 inches
Nominal ID	.775 inches

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VI. Diagnostic Programs

Several different diagnostic programs are planned or in progress in order to obtain better definition of the tube wall diameter reduction phenomenon and the required corrective actions. These diagnostic programs include laboratory test programs, expanded eddy current testing, special visual inspections, and removal of a tube/tube support plate sample.

Laboratory Programs

An extensive laboratory test program is underway in order to evaluate the possible causes for localized tube wall diameter reductions. The work is directed toward replicating the diameter reduction phenomenon in accelerated tests using aggressive corrodent solutions, and toward identification of various contaminants which may serve as the corrodent. As these studies imply, the "working hypothesis" involves the presence of a contaminant, trapped in the crevice between tube and tube support plate, which is progressively corroding the carbon steel. The consequential corrosion products, being larger in volume than the parent steel, press against the tube wall, leading to permanent strain.

Preliminary work has shown that carbon steel will undergo a significant change in volume when converted to its oxide. Carbon steel specimens exposed to deliberately concentrated synthetic sea water, and its vapor, in an autoclave at steam generator conditions produced the effect of metal-to-oxide volume change to illustrate the point. Subsequently, a large number of capsule and autoclave tests have been initiated in order to further define the working hypothesis by actually deforming a tube.

The most expeditious testing is being performed in a capsule configuration, wherein an Inconel-600 tube, fitted with end caps, contains a machined plug of carbon steel and the capsule is partially filled with one of a variety of potential corrodents. The capsule and contents are held at steam generator conditions for the test period of several weeks then opened. Gauge measurements will indicate the degree to which corrosion products have led to permanent deformation of the tube wall.

Typical corroding solutions for the capsule tests will include concentrated sea water, sodium phosphate solutions (at low Na/PO_4 ratios), and slurrys of synthetic "green" deposit, which simulate the metal phosphate deposit found on steam generator tubes at support plate locations.

Another test design will be performed in autoclaves which more exactly reproduce the physical geometry of the tube-tube support plate. In this, an Inconel-600 tube, heated internally, will be immersed in an appropriate solution of candidate corrodents. The tube OD surface will be fitted with carbon steel support plate sections--duplicating the diametral clearances as exist in the steam generators. This configuration will be operated with the intent of duplicating the diameter reduction process and contributing, thereby, to definition of the specific mechanisms involved.

Another series of tests are planned, and in progress, utilizing the model boiler concept, where the support plate-tube configuration will be used. (See Figure 16). The model boilers will be operated with phosphate chemistry, initially, in an attempt to generate some of the green deposit in the crevice region, as has been found on steam generator tubes. Once having accomplished this, two separate operational paths will be followed: 1) one boiler will be converted to AVT chemistry and the presence and/or progression of diameter reduction followed as a function of operating time, and 2) another boiler will be used for chemical cleaning trials.

As an overall part of this program, a detailed examination is being made of the deposits found at tube support plate regions in order to

.1/2

better synthesize this material for use in the various tests noted above. Thus far, the green material has been identified as a metal phosphate, primarily containing 30 weight percent or more phosphate, as given in Table 1. It has a low solubility, even in water at steam generator operating conditions, and is believed to be indigenous to the area where found, and may participate in the diameter reduction phenomenon. This thought, however, must be borne out by test.

The program described above is underway and results are expected to be collected during the next six months. Until then, unless an especially meaningful result is obtained, only fragmentary results and conclusions are expected.

Previous model boiler testing has been reviewed for a possible contribution to the question of localized diameter reduction. A number of tubes from the model boiler tests were examined by profilometry at regions adjacent to external devices which created a crevice with the tube. No evidence of diameter reduction was found by these measurements, which includes tests with phosphate, AVT, and phosphate to AVT chemistries. It is recognized, however, that the true steam generator configuration was not present in these earlier tests; namely, the external devices did not have clearance dimensions equal to the support plate, and none of the devices were fabricated of carbon steel. Much more meaningful information is expected from the present series of tests, as pertains to localized diameter reductions.

Eddy Current Tests

In an effort to better characterize and define the extent of the tube diameter reduction phenomenon, the Westinghouse standard eddy current inspection plan is being expanded in applicable units to include those areas near the divider lane and in the periphery of the steam generator where the tube diameter reductions have been the greatest and where leaking tubes have been located. (See Figure 17). These expanded eddy current tests are diagnostic in nature for the purpose of better defining the pattern of tube deformation at each support plate.

Visual Inspections

Extensive visual inspections of the tube support plate flow slots, such as those conducted at Surry Unit 1, will be conducted. These inspections will include at least the tubes adjacent to the divider lane and the first tube support plate. These inspections will help better define the effect of the diameter reduction phenomenon on the tube support plates and the tubes away from the region of the support plates.

Tube/Tube Support Plate Sample Removal

It is felt that to be able to formulate a definite mechanism for the tube diameter reduction phenomenon, a portion of support plate, with two deformed tubes intact, and with the deposits in place between the tube and support plate, should be obtained for laboratory analysis. Therefore, it is intended that such a sample will be obtained from a top support plate in the periphery of a steam generator at the earliest opportunity, contingent on plant availability. A procedure and special tooling have been developed and personnel have been trained in a mock-up to perform this sample removal operation.

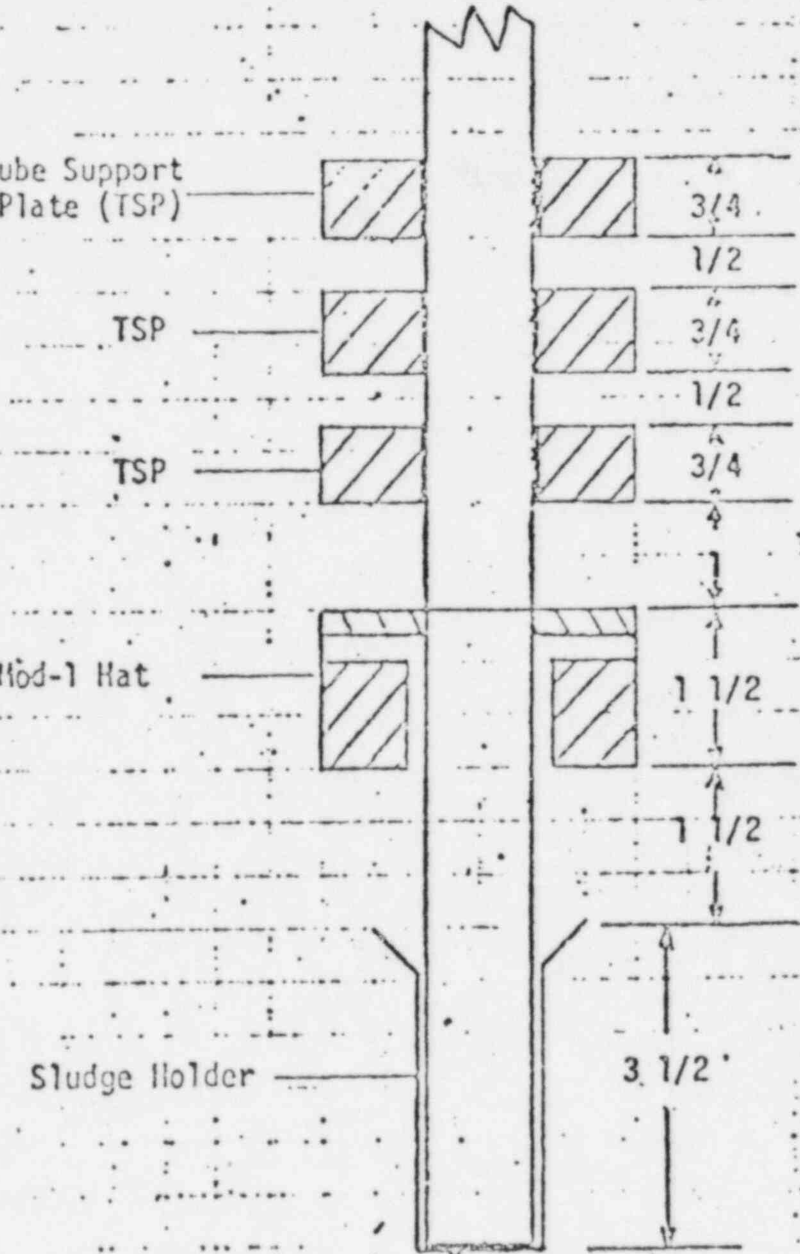
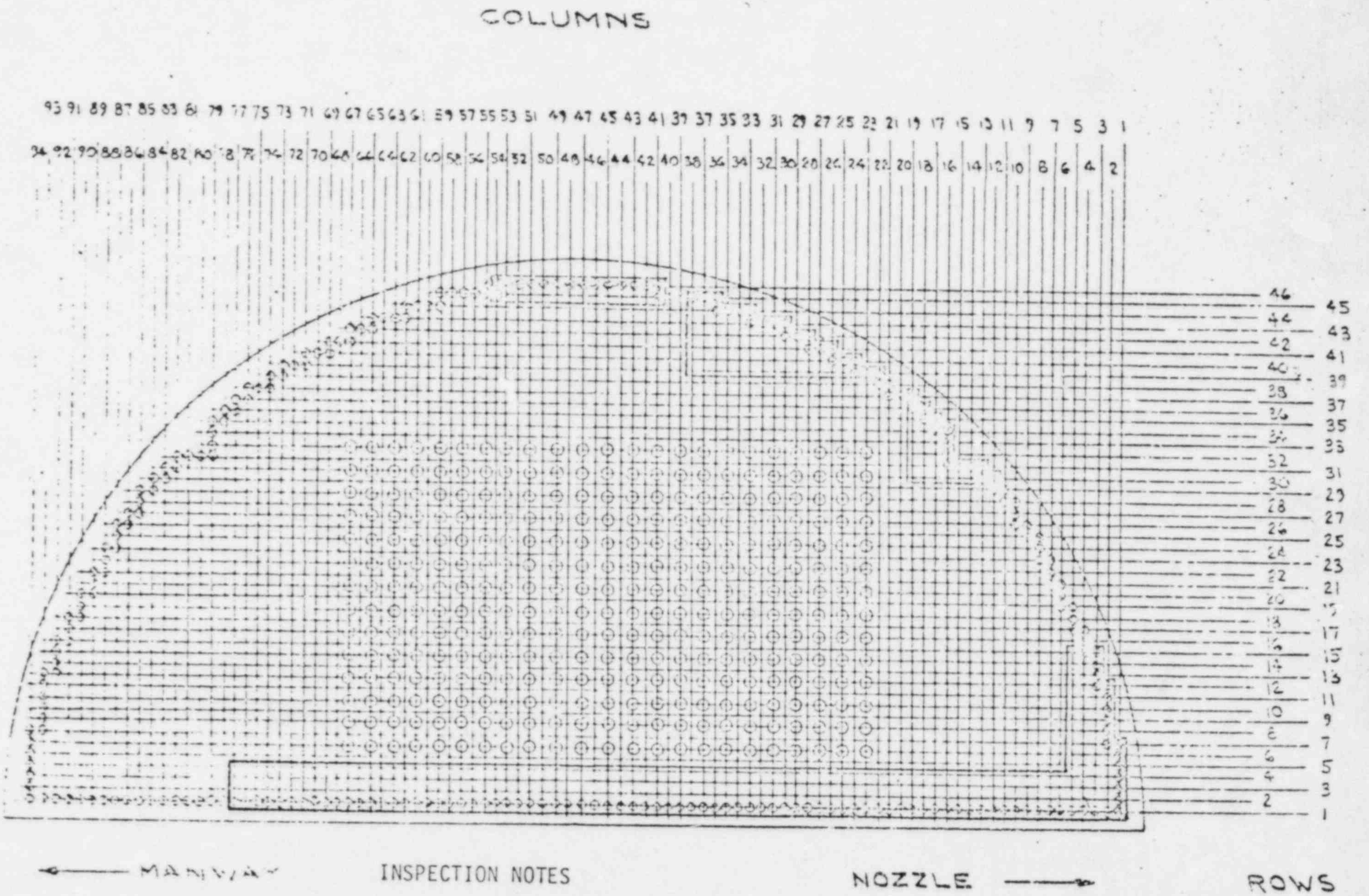


FIGURE 16 Boiler Tube Modification



INSPECTION NOTES

HL & CL
Through Top Support
Low Gain

FIGURE 17

VII. Conclusions

Based on all the information on the tube diameter reduction phenomenon available at this time, summaries of which have been presented above, Westinghouse has reached the following conclusions:

1. The phenomenon appears to be restricted to plants which had extensive operation with phosphate treatment prior to conversion to AVT.

2. The phenomenon is hypothesized to be caused by a build up of a corrosion product in the space between the tube and the tube support plate.

3. There is no apparent change in the mechanical properties of the tube metal. Data from tests performed on tubes removed from steam generators indicate properties (e.g., ductility) which are typical of new tube material.

4. Plant operation can continue safely while further evaluation and investigation of the phenomenon proceeds.

Intensive efforts to better define and characterize the phenomenon are continuing as described in Section VI above. As more information is gained through the diagnostic programs, this information will be provided to the NRC Staff in a timely manner.

February 10, 1976

INSTRUCTIONS FOR SUPERSEDING PAGES TO WESTINGHOUSE PRELIMINARY EVALUATION OF THE STEAM GENERATOR LOCALIZED TUBE DIAMETER REDUCTION PHENOMENON AND ITS SAFETY IMPACT:

I. Supersede the following pages and figures with those attached.

A. Non-Proprietary Binder

1. Page IV-3
2. Page IV-5
3. Figure 12 (Section IV)
4. Figure 13 (Section IV)
5. Figure 15 (Section V)

B. Proprietary Attachment Binder

1. Page 1 (Attachment IV-1)
2. Page 2 (Attachment IV-1)
3. Page 2 - 4 (Attachment V-1)
4. Page 2 - 10 (Attachment V-1)
5. Page iii (Attachment V-2)

II. Add the following pages which were omitted from proprietary attachment V-2.

Pages 6 - 10, 6 - 11, 6 - 12, 7 - 1, 7 - 2, 8 - 1, and 9 - 1.

CORRECTIONS ARE NOTED IN THE MARGINS OF SUPERSEDING PAGES in relatively few cycles. The loading condition for one form of this model is presented in Figure 14. The crud deposit in the tube to tube support hole annulus was assumed to be incompressible and with a zero coefficient of thermal expansion would continue to fill the gap after each cycle. The total radial deformation of each cycle was 1.7×10^{-4} inch. This or a similar mechanism will result in a net pressure on the tube outside diameter and the tube support plate hole inside diameter. |

Determination of External Pressure to Yield Tube and Support Plate

The tube ΔP can be approximated with the following relationship:

$$\Delta P = \frac{(\text{Yield Strength})}{\left[\frac{\text{Mean Radius}}{\text{Wall Thickness}} + \frac{1}{2}\right]}$$

$$\Delta P = \frac{(44,660)}{\left[\frac{.4125}{.045} + \frac{1}{2}\right]} = 4620 \text{ psi}$$

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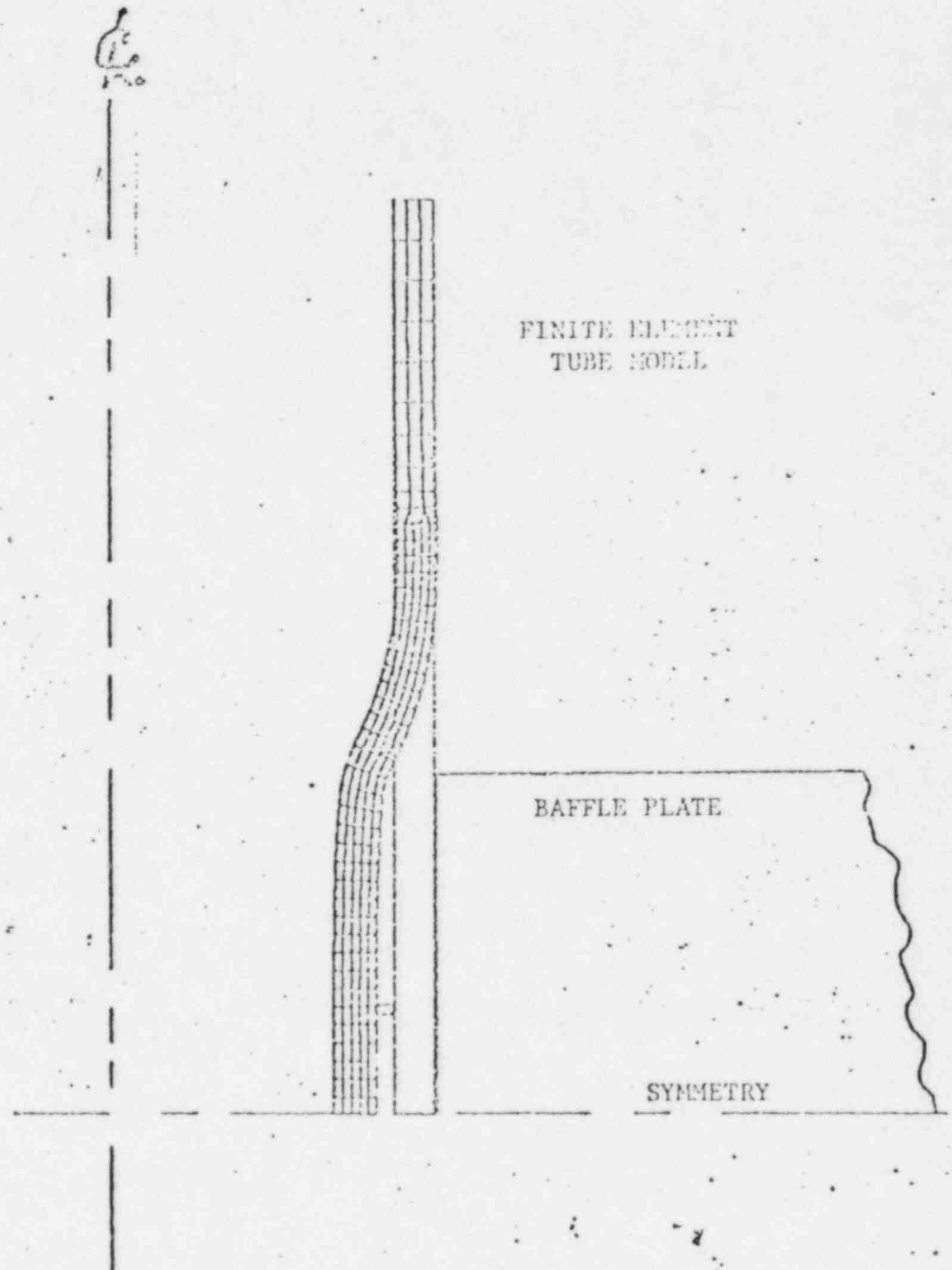
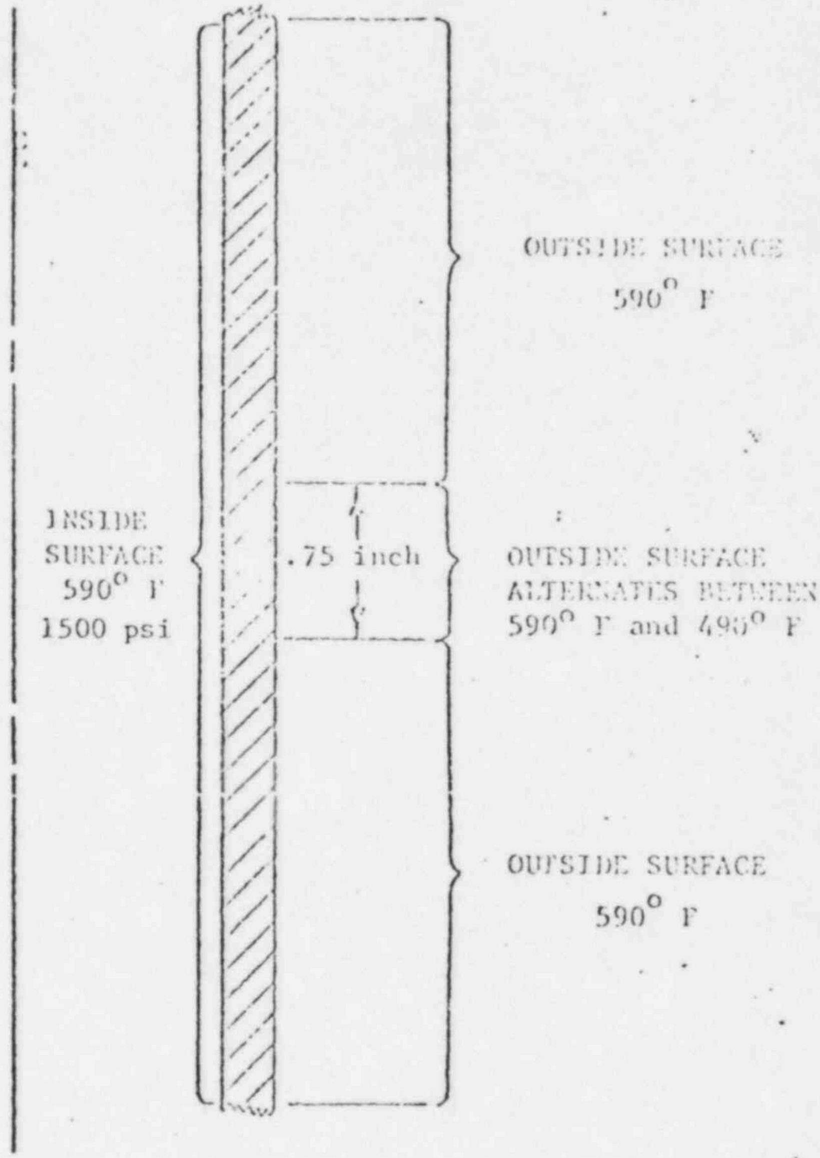


TUBE/BAFFLE INTERACTION CYCLE

FINAL TUBE POSITION AT ROOM TEMPERATURE

- 1 cycle -

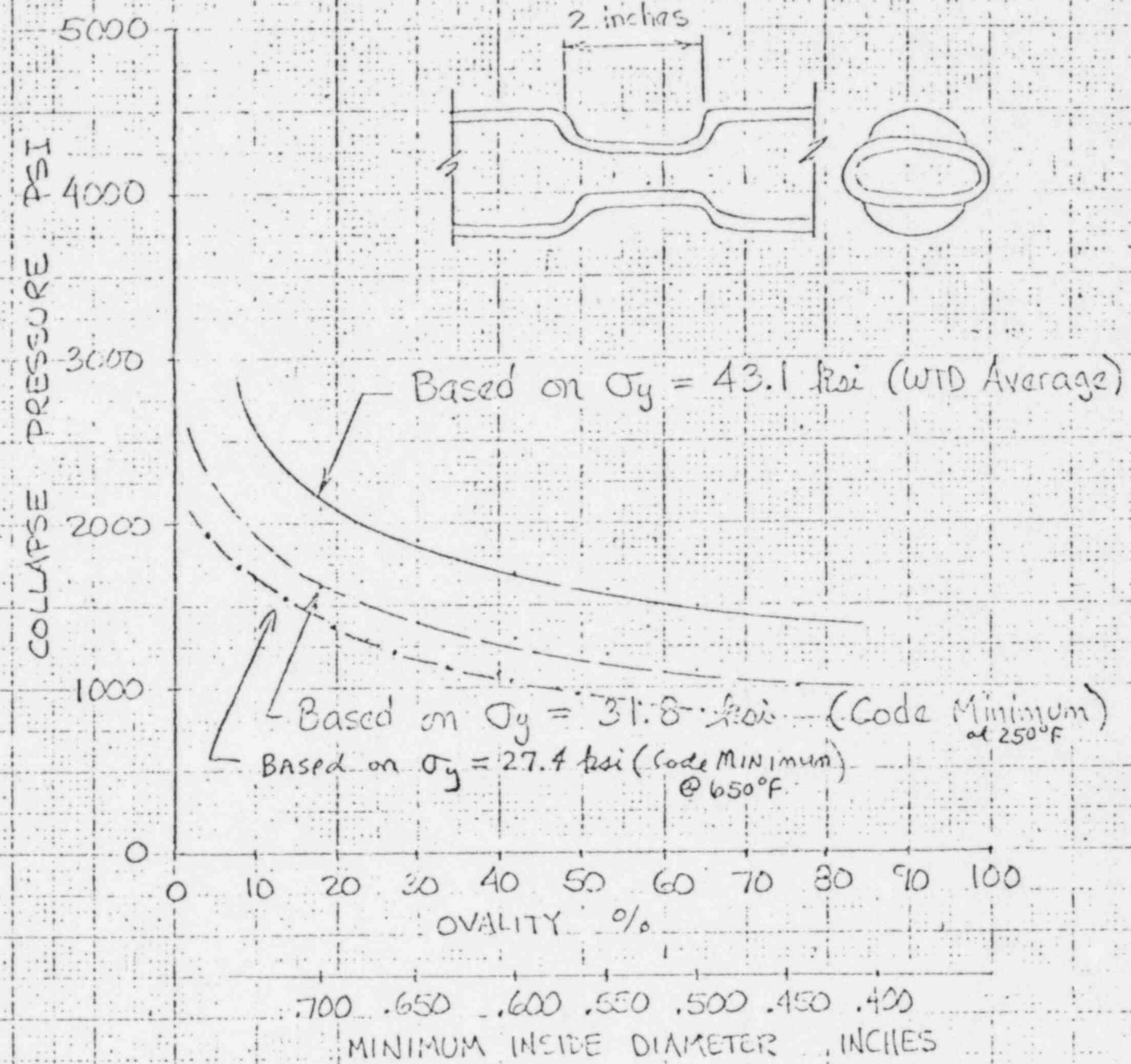
FIGURE 12



Tube Wall Temperature History

FIGURE 13

FIGURE 15



Collapse Pressures of Dented 7/8" Tubes

Temperature 250°F & 650°F
Wall Thickness .045 inches
Nominal ID .775 inches