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POWER SYSTEMS

Docket No. STN-50-470F

June 29, 1983
LD-83-058

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Rapid Depressurization and Decay Heat Removal for System 80™

References: (A) Letter, R. L. Tedesco to A. E. Scherer, dated March 26, 1982

(B) Letter LD-82-046, A. E. Scherer to R. L. Tedesco, dated April 19, 1982

(C) Letter LD-83-013, A. E. Scherer to D. G. Eisenhut, dated February 28, 1983

Dear Mr. Eisenhut:

Reference (A) transmitted to Combustion Engineering (C-E) the Staff's request for information regarding the rapid depressurization and decay heat removal capability of the System 80™ Nuclear Steam Supply System. In Reference (B), C-E provided justification for the continued licensing of System 80 in the interim and committed to respond to the Staff's questions. A breakdown of the questions was provided in Reference (C) which identified those questions that would be addressed generically for System 80 (on the CESSAR-F docket) and the remaining questions to be addressed by referencing Applicants. In addition, C-E indicated that the majority of question responses would be prepared on a generic basis and provided by a C-E Owners Group (CEOG) report. The purpose of this letter is to submit that CEOG report on the CESSAR-F docket and provide a response to the additional questions that were considered applicable to the System 80 design.

Attachment (1) tabulates the location of responses to each of the NRC questions. Five (5) copies of the CEOG report "Depressurization and Decay Heat Removal, Response to NRC Questions", (CEN-239) are enclosed. In addition, separate copies of an executive summary are also included.

A review of the enclosed information on rapid depressurization and decay heat removal appears to demonstrate that there is no significant benefit to the health and safety of the public by adding Power Operating Relief Valves (PORVs) to the System 80 design. C-E recognizes that the NRC, in making its judgement, will also consider the information to be provided by Applicants referencing

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1 COPY ADVANCED NRR/DSI/RSB
1 COPY OF ENCL'S ADVANCED TO D. EISENHUT

CESSAR-F. In considering whether a design change will be required, we believe that the NRC should base its cost/benefit analysis on the existing System 80 plants. If the change is not cost/beneficial for these plants then the standard System 80 design should not be modified. Only in this way can the NRC and the industry be assured that all units of a standard design will remain identical - thus preserving the NRC's policy on standardization.

If I can be of any further assistance in this matter, contact me or Mr. G. A. Davis of my staff at (203) 688-1911, extension 5207.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Director
Nuclear Licensing

AES:las

Enclosures: (1) CEN-239, "Depressurization and Decay Heat Removal, Response to NRC Questions

(2) "Response to NRC Questions 13a, 13c and 13d on Rapid Depressurization and Decay Heat Removal", June 1983

(3) "Response to NRC Question 14 on Rapid Depressurization and Decay Heat Removal", June 1983

LD-83-058
Attachment (1)

REFERENCE LIST FOR RESPONSE TO NRC QUESTIONS ON RAPID DEPRESSURIZATION AND DECAY HEAT REMOVAL FOR SYSTEM 80™

NRC Question Number	Location of Response
1	CEN-239
2	CEN-239
3	CEN-239
4	CEN-239
5	CEN-239
6a, 6b	To be provided by referencing applicants
6c, 6d, 6e	CEN-239
7	CEN-239
8	To be provided by referencing applicants
9	To be provided by referencing applicants
10	To be provided by referencing applicants
11	To be provided by referencing applicants
12	To be provided by referencing applicants
13a, 13c, 13d	Enclosure (2) to LD-83-058
13b	CEN-239
14	Enclosure (3) to LD-83-058

ENCLOSURE 2 TO LD-83-058

RESPONSE TO NRC QUESTIONS 13A, 13C AND 13D
ON RAPID DEPRESSURIZATION AND DECAY HEAT REMOVAL

June 1983

Question 13:

One of the main reasons C-E has concluded that PORVs are not needed for emergency decay heat removal is that alternative water sources could be made available to the steam generators for decay heat removal purposes. An inherent assumption in this approach is that steam generator integrity will be maintained throughout the life of the plant. One method of assuring combined steam generator integrity is by inservice inspection and plugging of tubes excessively degraded. Please disuss the following:

a. What is the minimum allowable wall thinning that could exist in the steam generator tubes without plugging?

b. What is the probability that ISI will not detect a degraded tube? Provide the margin of error in eddy current measurements at various depths of degradation.

c. Given a steam generator with the maximum allowed tube thinning and degradation, confirm that those tubes will maintain their integrity by demonstrating they have been analyzed and shown to remain intact for all design basis loadings used for the steam generator design including seismic loads.

d. Describe the analytical and experimental justification for establishing a minimum acceptable steam generator tube wall thickness for the C-E System 80 steam generators in accordance with guidelines in Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes". The Justification should include the analyses to calculate the hydraulically induced loading on the steam generator and the thermal response of its tubes and shell to an assumed LOCA, MSLB, and an FWLB.

Response:

An evaluation was performed on the System 80™ steam generator to determine the allowable tube wall degradation. This evaluation is summarized in the attached report Permissible Steam Generator Tube Thinning. This report addresses parts a, c, and d of the above question. Part b to the above question will be addressed separately by the C-E Owners Group as part of CEN-239.

The attached report "Permissible Steam Generator Tube Thinning", shows that 43% tube wall degradation is acceptable at the most limiting tube locations. This value is determined by conservative comparisons to analyses performed on other C-E steam generator designs. The report also presents some tests results that substantiate the validity of the analytical methodology used by C-E to determine tube plugging limits.

SUPPLEMENTARY INFORMATION FOR

APPENDIX A

CAPABILITIES FOR THE DEPRESSURIZATION AND DECAY HEAT REMOVAL WITHOUT PORV'S

RESPONSE TO NRC QUESTION NO. 13

PERMISSIBLE STEAM GENERATOR TUBE THINNING

TABLE OF CONTENTS

LIST OF FIGURES

ABSTRACT

This report summarizes the results from a number of studies, addressing steam generator tube integrity, at six currently operating nuclear power plants. All of the evaluations were carried out in accordance with the provisions of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes".

The report also summarizes relevant work, which has been performed to date, on the lead System 80 NSSS. Since the response of System 80 steam generator tubes to a hypothetical loss of primary coolant accident (LOCA) has not previously been addressed in a rigorous manner, comparisons of geometry and operating parameters were made with previously performed studies for operating plants. Based on these comparisons, a conservative estimate of permissible tube thinning was made for the System 80 steam generators.

A brief summary of related experimental work is also included in this report to help support the conclusions drawn.

1. INTRODUCTION

Steam generator tubing for C-E nuclear power plants has traditionally been conservatively sized, such that a high degree of integrity persists, even after substantial attack from environmental agents. It has been previously demonstrated in the code stress reports, for six C-E power plants, that tube wall degradation ranging from 31% to 64% can be tolerated and still meet design basis criteria and the provisions of Regulatory Guide 1.121. The range is higher yet (50% to 64%) for those units which have not received a "rim-cut" modification to mitigate support plate denting.

The C-E System 80 steam generator tubes (see Figure 1) have also been evaluated for most design and pipe break accident criteria. Since most C-E steam generators are similar in design concept, an estimate of the permissible tube thinning for the System 80 steam generator units can be made based on previously performed work on other units and supporting experimental data. This report provides a comparative assessment of the System 80 steam generator tube plugging limits.

Sections 2 and 3 of this report will describe the C-E analysis used for Pre-System 80 plants. Sections 4 through 8 provide an evaluation for System 80.

2. LOSS OF PRIMARY COOLANT ACCIDENT (LOCA) PLUS SAFE SHUTDOWN EARTHQUAKE (SSE) ANALYSIS

The margin of safety against tube failure under a postulated LOCA accident concurrent with SSE should be consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the ASME Boiler and Pressure Vessel Code.

As a result of a postulated LOCA accident a steam generator U-tube will experience an inplane frame type deformation due to the rarefaction wave in the primary coolant which propagates away from the break location (see Figures 2 and 3). This loading, when combined with SSE, LOCA impulse and differential pressure, causes severe bending stress in the tube at its uppermost horizontal support.

SUMMARY OF RESULTS FOR PRE SYSTEM 80 STEAM GENERATORS

Geometries evaluated thus far sustain maximum tube bending stresses in healthy tubes of between 26.0 ksi and 52.1 ksi for the LOCA + SSE accident.

In addition, it has been determined that tubes having local uniform degradation at the worst possible locations of between 31% and 64% of the nominal tube wall can withstand this accident condition and still meet the criteria established in Appendix F of the ASME Code Section III for faulted conditions.

3. MAIN STEAM LINE BREAK ACCIDENT (MSLB) PLUS SAFE SHUTDOWN EARTHQUAKE (SSE) ANALYSIS

The margin of safety against tube failure under a postulated steam line break accident concurrent with SSE should be consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the ASME Boiler and Pressure Vessel Code.

In the event of a postulated main steam line break accident, the top of the tube bundle is subjected to extremely high velocity, high density crossflow of the secondary coolant. In a U-tube steam generator this loading, when combined with SSE, MSLB impulse and internal pressure, causes vertical bundle deflection with interaction among the various tube rows (see Figures 4 and 5). The resulting tube stress is highest at the top mid span position. The tube row of maximum stress is design dependent.

SUMMARY OF RESULTS FOR PRE SYSTEM 80 STEAM GENERATORS

Geometries evaluated thus far sustain maximum tube bending stresses of 27.2 ksi or less for the steam line break plus SSE Accident acting on healthy tubes. In addition, it has been determined that tubes having local uniform degradation at the worst possible locations of 63% or more of the nominal tube wall can withstand this accident condition and still meet the criteria established in Appendix F of the ASME Code Section III for faulted conditions.

4. FEEDWATER LINE BREAK ACCIDENT (FWLB) PLUS SAFE SHUTDOWN EARTHQUAKE (SSE) ANALYSIS FOR SYSTEM 80 NSSS

The margin of safety against tube failure under a postulated feedwater line break accident concurrent with SSE should be consistent with the margin of safety determined by the stress limits specified in NB-3225 of SECTION III of the ASME Boiler and Pressure Vessel Code.

The economizer divider plate, support cylinder, cold leg flow distribution plate and feedwater box are subjected to a hypothetical feedwater line break during 100% power operation. The CEFLASH model which was used to determine hydrodynamic loads is shown in Figure 6. The pressure distribution acting on the economizer divider plate during a postulated FWLB event was determined by applying the peak pressure differences between nodes. Reactive forces acting on the divider plate along the lugs which are attached to the support cylinder were applied to the support cylinder. These forces along with the pressure differential acting on the cylinder between the hot leg and cold leg comprised the active forces on the support cylinder. The peak pressure difference of 660 psi was assumed to act uniformly over the feedwater box. Differential pressure acting on the perforated portion of the distributor plate was taken to be the pressure difference between nodes 29 and 32.

SUMMARY OF RESULTS - (SEE FIGURE 7)

a. ECONOMIZER DIVIDER PLATE

The primary stresses of concern in the divider plate and blowdown assembly are membrane plus bending. The divider plate has maximum

4. a. CONTINUED

membrane plus bending stress of 34.2 ksi which is less than the allowable of 1.5 (0.7 S_u) = 73.5 ksi for the SA-515, GR 70 material. The blowdown duct has maximum membrane plus bending stress of 47.4 ksi and the allowable is 60.9 ksi for the A-500, GR B material.

b. SUPPORT CYLINDER

The membrane plus bending stress intensity at the base of the stay cap assembly is 14.5 ksi which is less than the allowable of 1.38 (0.7 S_u) = 77.3 ksi for the SA-508, CL 2 material. At the bimetal weld the membrane plus bending stress intensity is 9.4 ksi and the allowable is 67.6 ksi for the SA-516, GR 70 material.

c. COLD LEG FLOW DISTRIBUTION PLATE

The flow distribution plate has maximum ligament membrane plus bending stress intensity in the perforated region of 49.6 ksi and in the solid rim 34.8 KSI. The allowable for the SA-240, TY 405 material is 1.5 (0.7 S_u) = 58.7 ksi.

d. FEEDWATER DISTRIBUTION BOX

The inner cylinder of the feedwater distribution box has maximum membrane plus bending stress intensity of 38.4 ksi with the allowable for the SA-515, GR 70 material of 1.38 (0.7 S_u) = 67.6 ksi.

e. HEAT TRANSFER TUBES

The direct loading of the excaping fluid on the tubes is small (σ < 1.0 ksi). The danger to the tubes is that if one of the above four structures fails, it would put the adjacent tubes in jeopardy. However as noted above these structures are very conservatively designed, therefore they will have no impact on thinned tubes.

5. NORMAL OPERATION SAFETY FACTORS

(1) Yield Stress Limit

Tubes with detected defects of all types should not be stressed during the full range of normal reactor operation beyond the yield stress of the tube material.

Assuming the most conservative defect from a structural standpoint, uniform thinning about the circumference of the tube, the following relationship is taken from the ASME Code.

$$t_r = \frac{\Delta P \; R_i}{S_y - 0.5\,(P_1 + P_2)}$$

where: t_r - Required wall thickness

ΔP - $(P_1 - P_2)$

P_1 - Primary side pressure

P_2 - Secondary side pressure

R_i - Inside radius $(\frac{I.D.}{2})$

S_y - Minimum yield strength

Then the % allowable degradation is:

$$\% = \frac{t_n - t_r}{t_n}$$

where: t_n - Nominal wall thickness

5. CONTINUED

(2) Safety Factor Against Burst

Tubes with detected defects of all types should have a factor of safety against failure by bursting under normal operating conditions of not less than 3 at any tube location.

Assuming the most conservative defect from a structural standpoint, uniform thinning about the circumference of the tube, the following relationship is taken from the ASME Code (NB-3224.1):

$$t_r = \frac{3\Delta P \ R_i}{S_u - 0.5\ (P_1 + P_2)}$$

where: t_r - Required wall thickness

ΔP - $(P_1 - P_2)$

P_1 - Primary side pressure

P_2 - Secondary side pressure

R_i - Inside radius $(\frac{I.D.}{2})$

S_u - Ultimate Tensile Strength

Then the % allowable degradation is:

$$\% \ \frac{t_n - t_r}{t_n}$$

where: t_n - Nominal Wall Thickness

6. TUBE INTEGRITY EVALUATION

The following methodology was used to determine an allowable tube degradation for System 80:

(1) Data for six Pre System 80 plants was evaluated using information from specific stress reports.

(2) It was observed that the LOCA + SSE event was controlling in all cases; therefore, it should also be controlling for System 30.

(3) An approximate static loading was developed using the figure and equations at the bottom of Figure 3. A stress was developed

$$\sigma_B = \frac{M_p . C}{2I} .$$

This static-stress was compared with the actual stress (σ_{CALC}) from the structural dynamic "ANSYS" analysis. The ratio of static to dynamic stress was determined and was termed a "magnification factor" (M.F.).

(4) Based on the overall comparison of data, the following conservative assumptions were made concerning System 80:

(a) A magnification factor of 1.5. (A upper bound for plants without rim-cuts).

(b) The longest tube row won't bind in the vertical grids, (see Reference 1).

(c) A pressure drop to length ration of 1.0.

6. CONTINUED

(5) Using the above assumptions and the known parameters from the code stress report and Figure 9, the calc stress (σ_{CALC}) was determined. This value was used to determine an estimated permissible tube thinning of 43%.

While this method is not rigorous, it is a conservative approximation.

7. SIMULATED LOCA TUBE LOADING TEST

A research program, which was sponsored by EPRI (see Reference 1) provided considerable insight relative to steam generator tube response to LOCA.

OBJECTIVE:

The purpose of the program was to verify the CEFLASH computer code and the modeling techniques which are presently being used to predict thermal-hydraulic behavior in steam generator tubes during a LOCA transient. It was also an objective of this program to record the mechanical responses of the tubes to the LOCA event and compare these data to analytical predictions of bending stresses in the bend region of the tubes.

SCOPE:

The test loop (see Figure 10) simulated the primary side thermal-hydraulic conditions in an operational nuclear steam generator. The loop consisted of 5 full size double 90^o bend tubes and steam generator plena, a pressurizer, a reactor resistance simulator, a heater, a pump, and associated pipes and valves to complete the system. Cold leg guillotine breaks were simulated using quick opening valve and rupture disks. Break opening times ranged from less than 1 msec to as much as 67 milliseconds. The loop instrumentation was designed to measure the transient pressure history at various locations and monitor the structural response of the tube to the LOCA hydrodynamic loading.

A series of blowdown tests were performed for different operating and boundary conditions. The parameter variations included: fluid temperature, pre-blowdown flow rate, break opening time, break opening area, and break location. Both uniform and mixed length tube bundles were used.

RESULTS:

Analytically predicted transient pressure histories and the differential pressure history across the tube span were compared with the experimental data. Predicted structural responses in the bend region were also compared with the test data. The transient pressure histories as predicted by CEFLASH were in excellent agreement with the test data. The calculated structural responses of the tube also had good overall agreement with the test data (see Figure 11). Therefore the use of CEFLASH to predict steam generator tube response during a LOCA event is justified by the close agreement between the analytically prediction and the test data.

8. DEFECTED TUBE TESTING

Various organizations have conducted burst tests on tubes with real and simulated defects. The primary purpose of these tests was to demonstrate capability of Inconel 600 to withstand a wide variety of defects simulating damage received during steam generator operation. Figures 12, 13, and 14 plot burst pressure versus percent maximum defect depth for various defect types. The analytically predicted burst pressure line was developed using the relationship shown in Figure 12.

From the figures it is clear that significant margin exists between analytically predicted and actual burst pressure for most defects. There are two principal reasons for this margin:

(1) Better material properties - actual tubes have higher yield and ultimate strengths than the ASME Code minimum values. Examination of mill test data verifies this statement.

(2) Reinforcement effect - for tubes that have a limited defective area, undefected material adjacent to the defect picks up redistributed load and the overall burst resistance of the tube is enhanced. Note from the figure that greater margins exist for the more limited defect sizes.

In summary the presented experimental data demonstrates additional conservatism beyond that contained in the analytical treatment in the previous section.

9. CONCLUSIONS

(1) Units which have had their upper support plates detached from the shell, in order to mitigate "denting" effects, have somewhat lower permissible tube thinning values in the upper tube bundle region (there is no effect near the tubesheet). To date, C-E plants have not experienced "denting" and tube attack in the same region of the steam generator.

(2) System 80 is comparable to plants which have been calculated to possess an allowable tube wall thinning of from 50% to 64%. System 80, allowable tube wall thinning limit is conservatively estimated to be 43%.

(3) Experimental results, from several sources, demonstrate that for degradation other than uniform thinning, additional conservatism is introduced by "reinforcement" supplied by the material surrounding the degradation. Further conservatism is introduced by the fact that most of the tests show a benefit from a greater than minimum ultimate strengths.

(4) Simulated full scale LOCA testing has verified the accuracy and conservatism of C-E's current methodology and Analytical computer codes in determining steam generator tube loading due to a hypothetical loss of primary coolant accident (LOCA). This event is controlling for tubing in C-E steam generators.

9. CONTINUED

(5) Analysis results show that the economizer divider plate, support cylinder, cold leg flow distribution plate and feedwater box are adequately designed to withstand a hypothetical feedwater line break accident. Thus, the tubes in the economizer region will not be damaged, because being lightly loaded hydraulically, only failure of an adjacent structure would harm the tubes.

10.0 REFERENCE

1. EPRI Research Project S144-1 Final Report, "Loads on Steam Generator Tubes During Simulated Loss-of-Coolant Accident Conditions", November 1982.

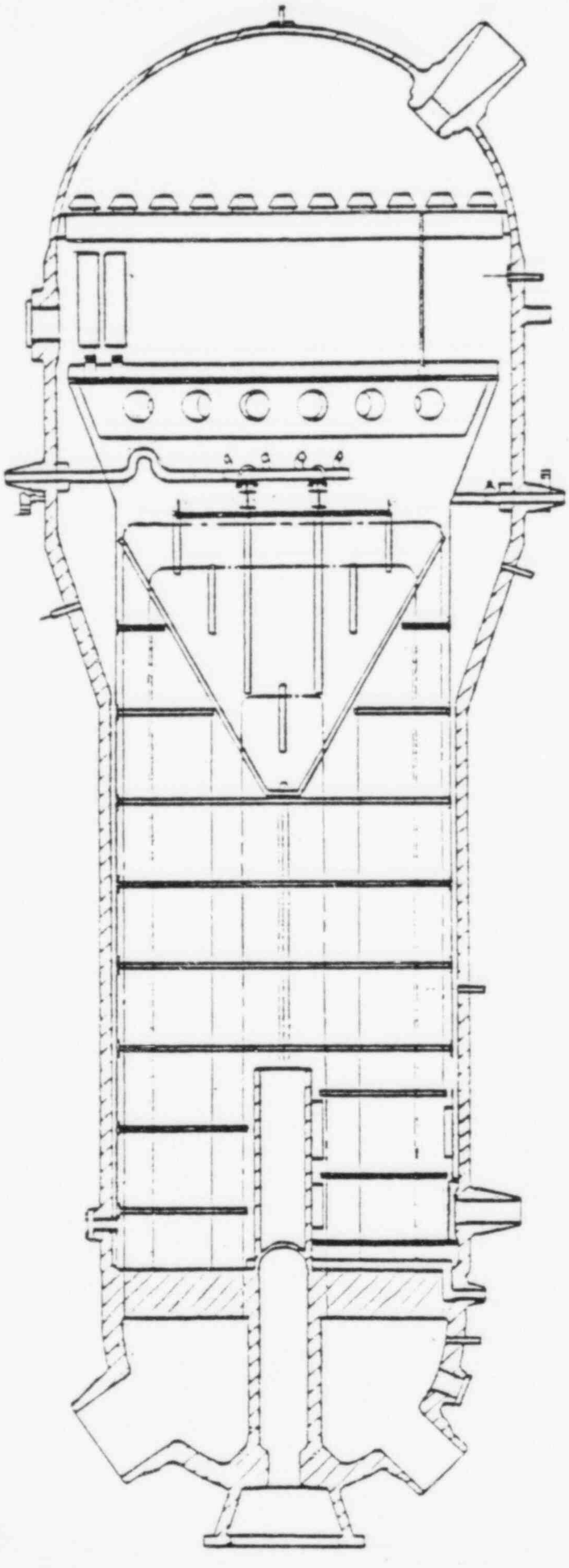


FIGURE 1.

C-E SYSTEM 80

STEAM GENERATOR

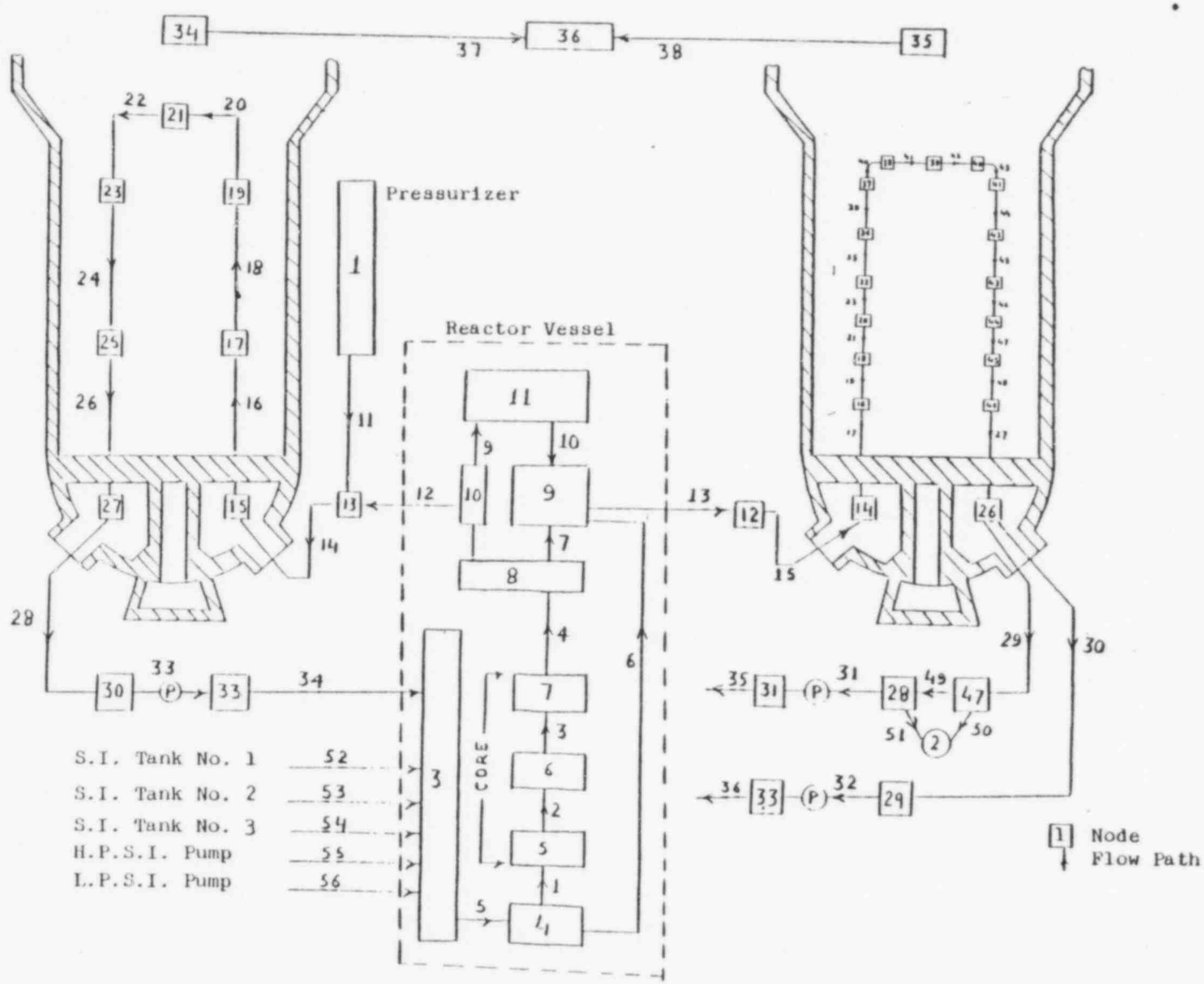
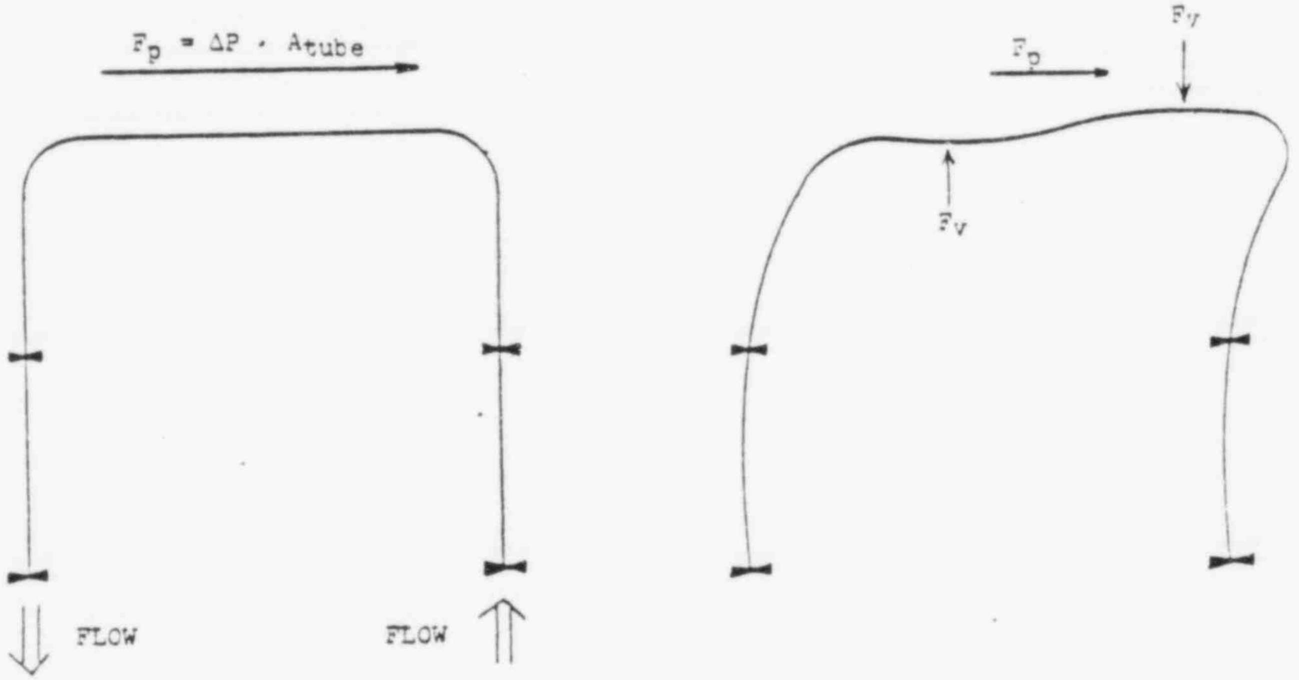


FIGURE 2. CEFLASH PRIMARY LOOP MODEL FOR LOCA

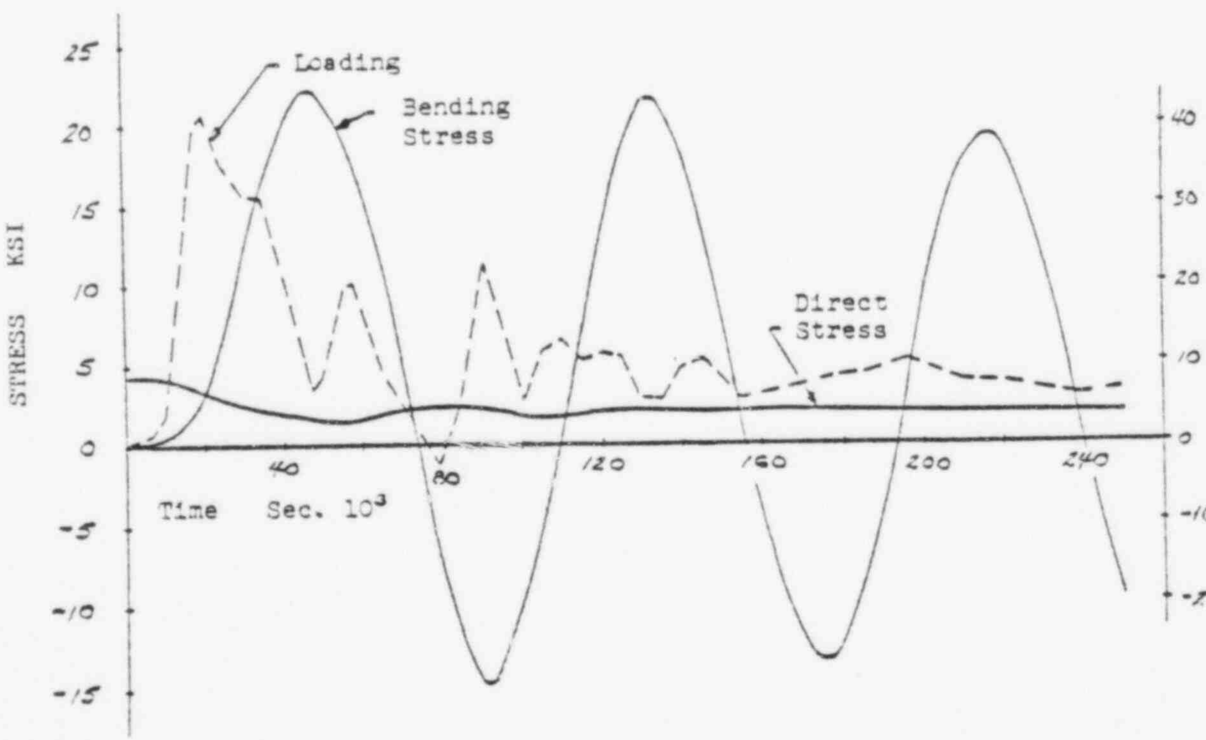
TUBE RESPONSE TO RAREFACTION LOAD CAUSED BY LOCA

(TYPICAL)



FLUID FORCES ACTING ON TUBE

EXAGGERATED DEFLECTION



RAREFACTION LOAD AND STRESS VS TIME

FIGURE 3.

CEFLASH SECONDARY SIDE MODEL FOR MSLB

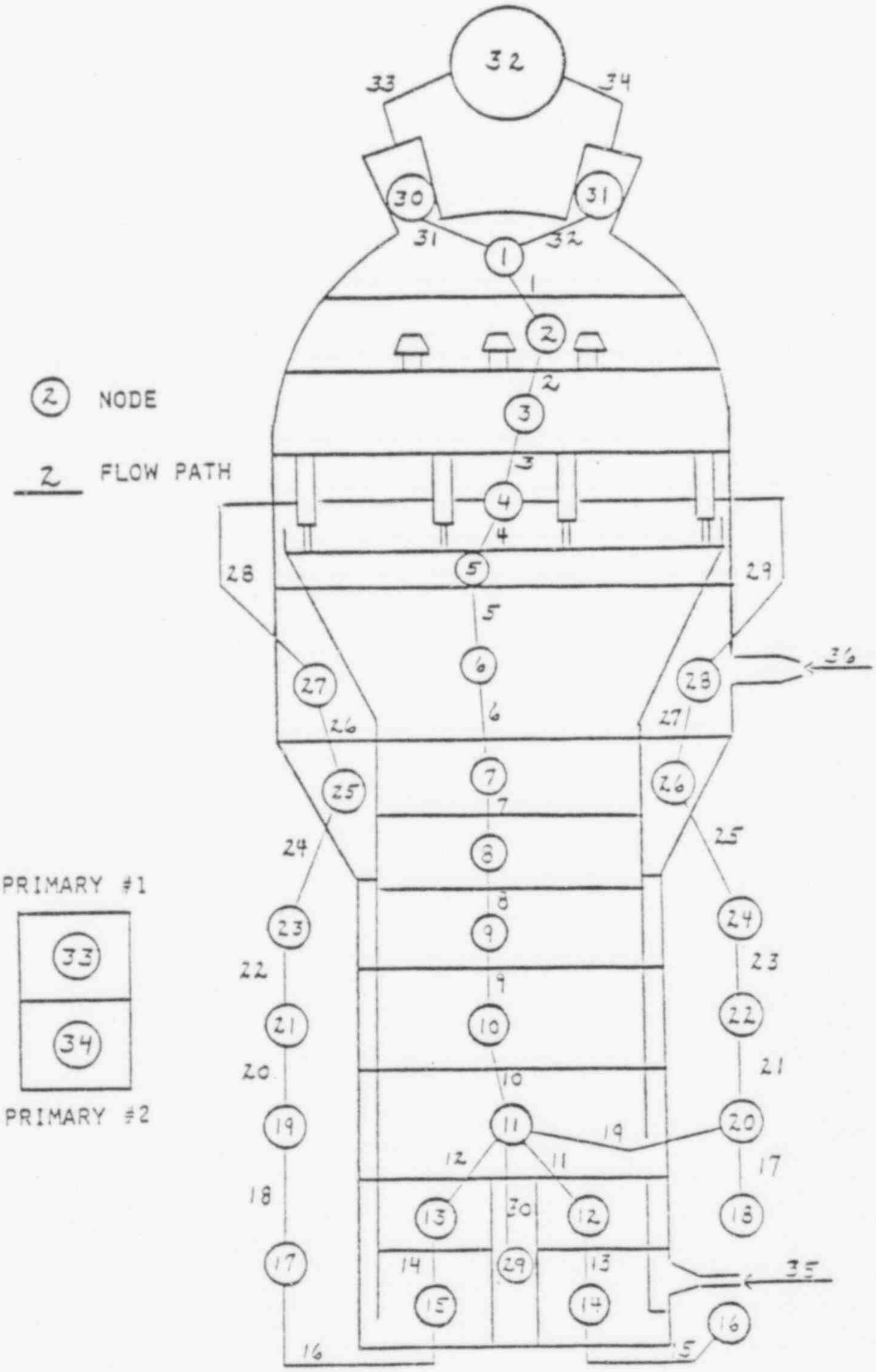
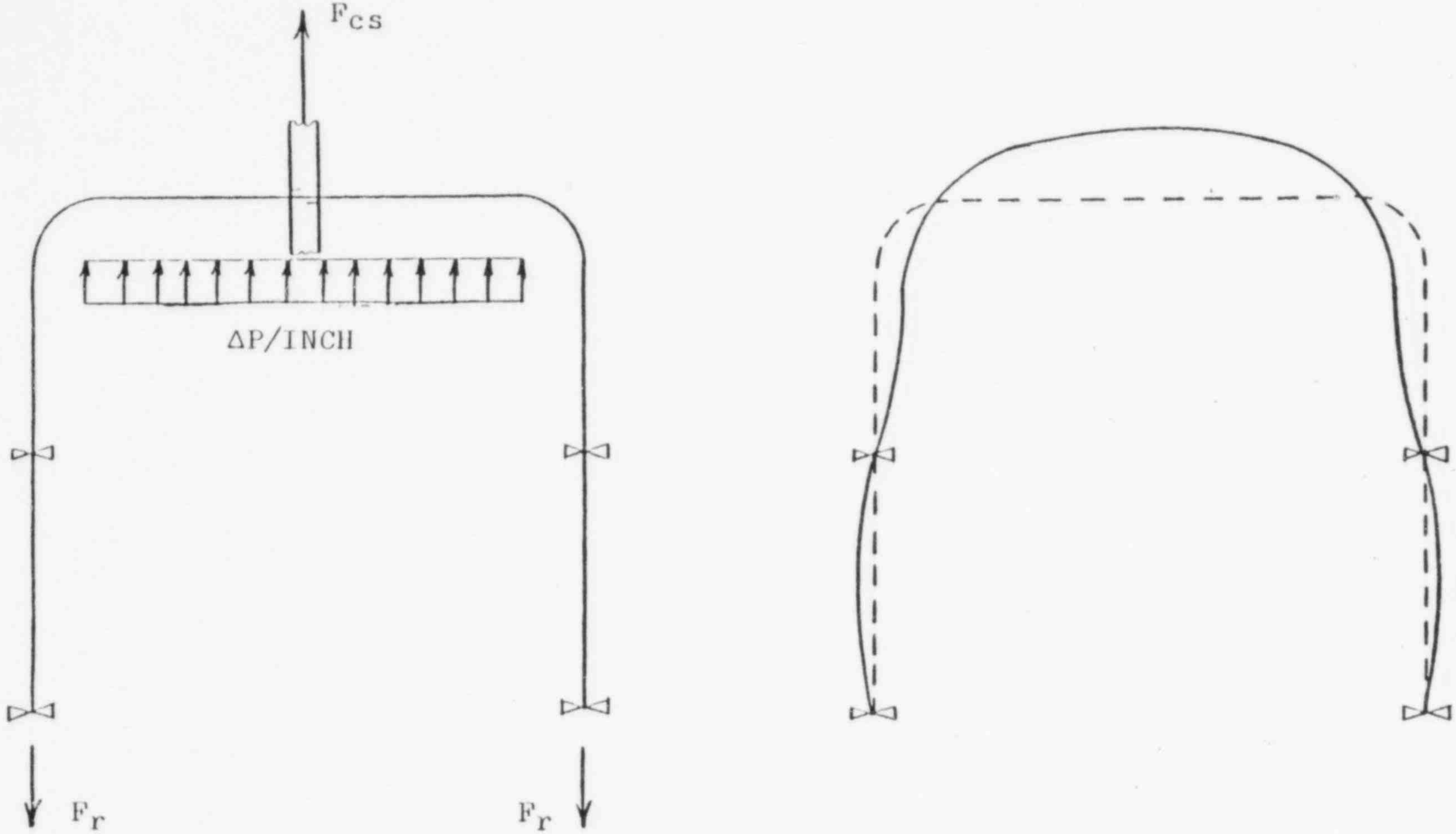


FIGURE 4.



SECONDARY FLOW AND
TUBE INTERACTION FORCES
ACTING ON TUBE

EXAGGERATED DEFLECTION

FIGURE 5. TUBE DEFLECTION DUE TO MSLB ACCIDENT FLOW

SECONDARY SIDE
CEFLASH MODEL FOR
FEEDLINE BREAK

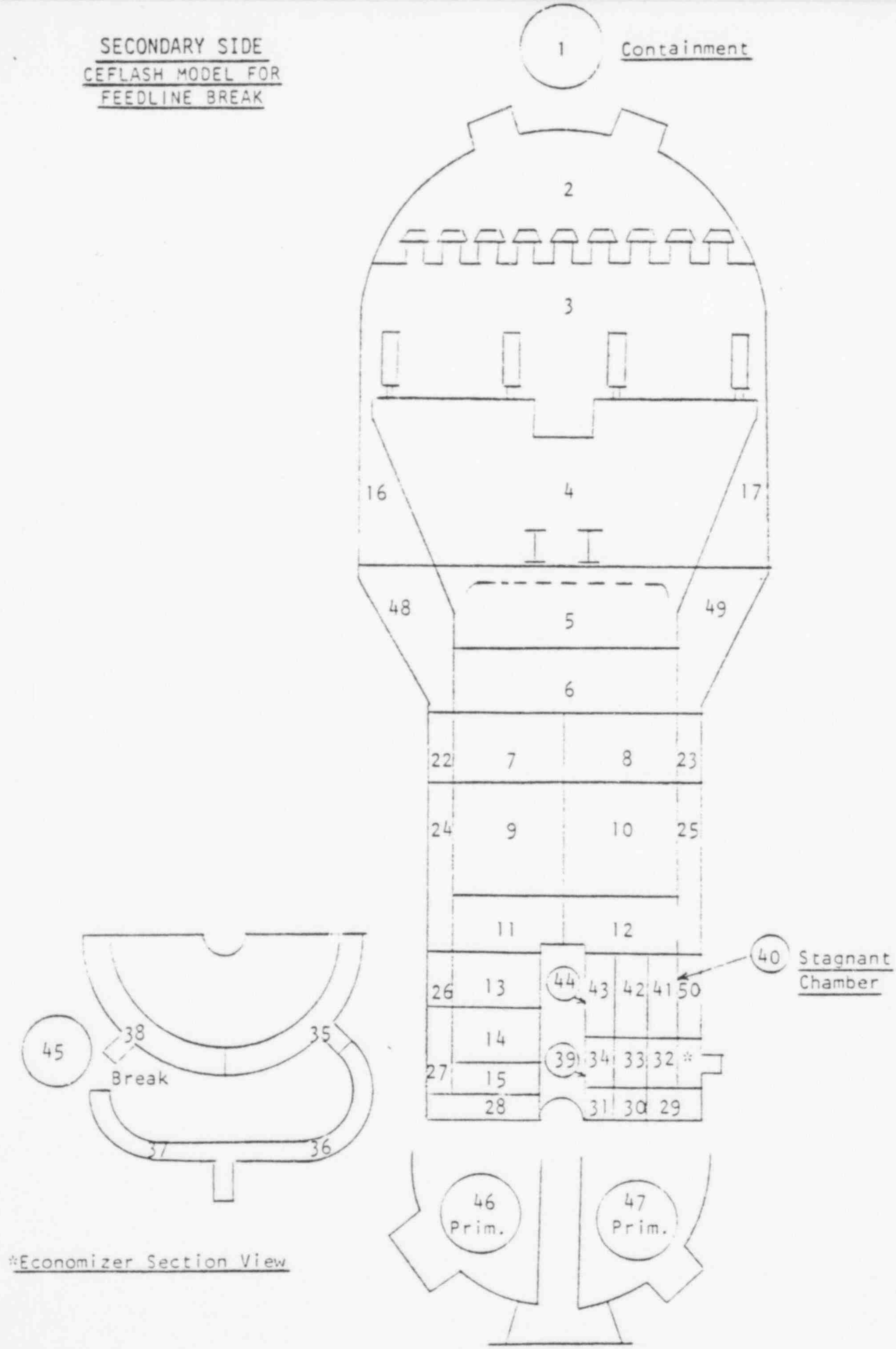


FIGURE 6.

ECONOMIZER ELEVATION VIEW

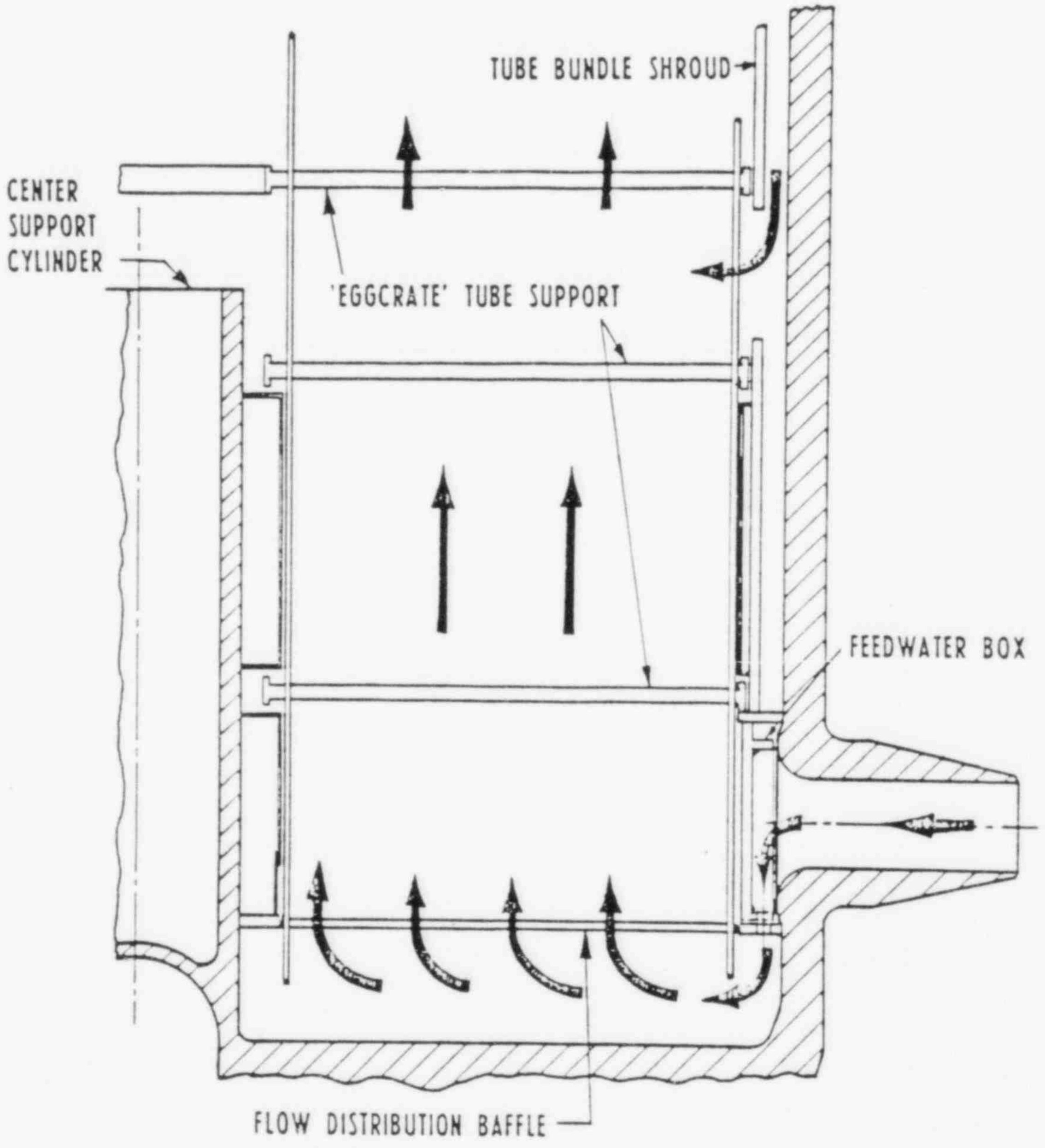


FIGURE 7.

TUBE WALL MARGINS FOR NORMAL OPERATION

P_1 (PSI)	P_2 100%	P_2 HOT STAND.	MAX ΔP	TUBE t_{WALL} (3/4 DIAM.) (IN)	t_{REQ} $\sigma \le S_y$ NORM OP.	t_{REQ} SF $\ge$ 3 BURST	$\frac{t_w - t_r}{t_w}$
2250	1070	1170	1180	0.042	.015	.015	64%

ALLOWABLE TUBE DEGRADATION

LOCA(1) +SSE	MSLB +SSE	FWLB $\pm$SSE	$\sigma \le S_y$ NORM OP.	SF $\ge$ 3 BURST		MIN. ALLOW DEGRAD
est. 43% (4)	83% (3)	Less than Hoop ΔP Stress	64% (2)	64%		43% (4)

FOR WORST CASE TUBE ROW

H (IN.)	L (IN.)	$(P_2-P_1)_{MAX}$ (PSI)	$\frac{(P_2-P_1)_{MAX}}{L}$ (P/IN^3)	$\sigma_B = \frac{M_p \cdot C}{2I}$ (KSI)	$\sigma_{CALC.}$ (KSI)	M.F.	ROW
58.5	163	163.0	Assume 1.00	28.6	42.9	Assume 1.50	Assume 159

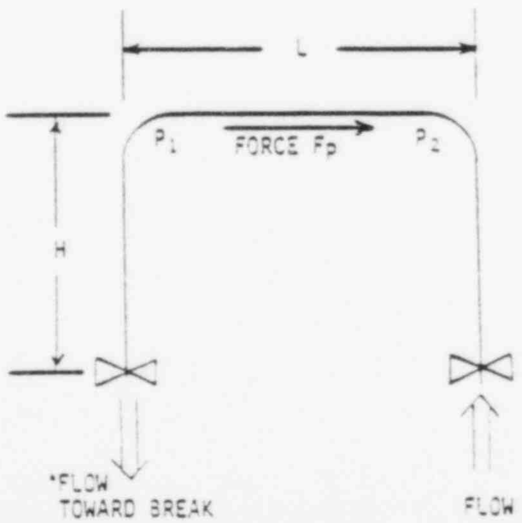
Notes:

(1) LOCA and SSE stresses conservatively added arithmetically rather than root mean sum of the squares.

(2) The combination of hydraulic flow load produced bending stress and axial pressure stress is less controlling than the hoop pressure stress.

(3) SSE stress not included.

(4) Was conservatively estimated based on previous work and comparison of geometric parameters



$F_p = (P_2-P_1) \cdot A_{TUBE}$

Moment $M_p = F_p \cdot H$

$\sigma_B = \frac{M_p \cdot C}{2I}$ or $\frac{M_p}{2Z}$

$(P_2-P_1)_{MAX} \sim L$

SYSTEM 80 UPPER TUBE BUNDLE GEOMETRY

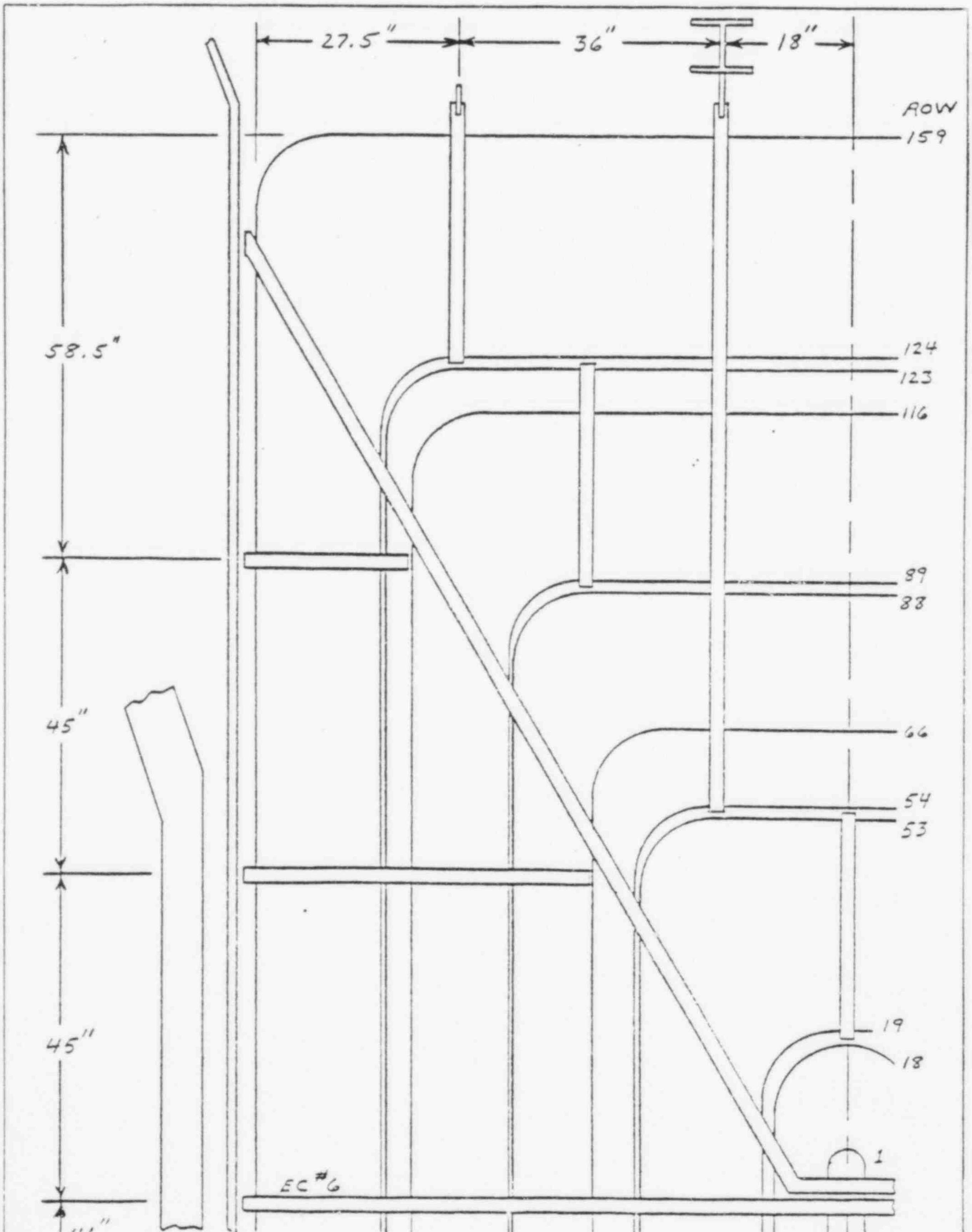


FIGURE 9.

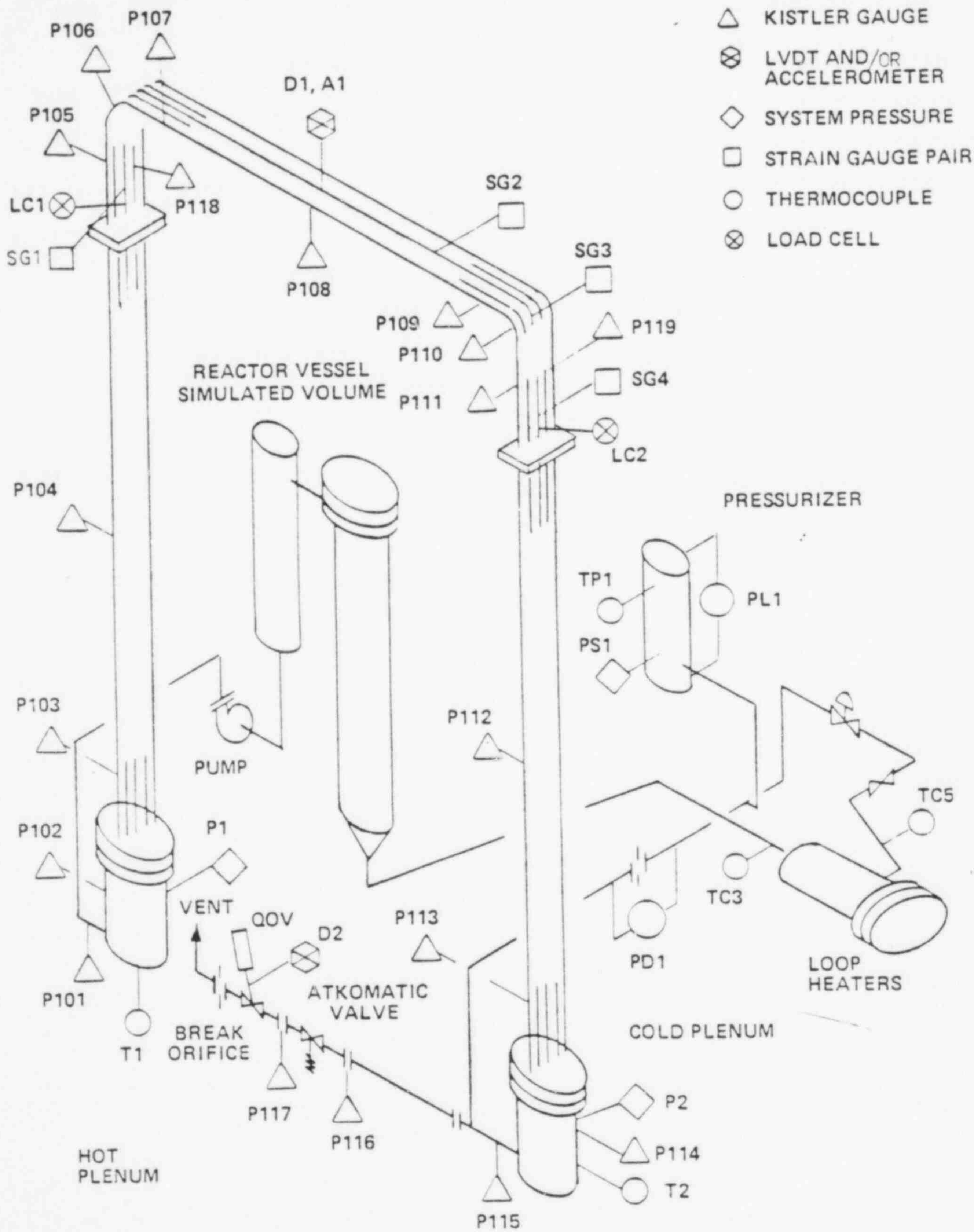


Figure 10. Steam Generator LOCA Blowdown Loop Schematic - Base Line Tests

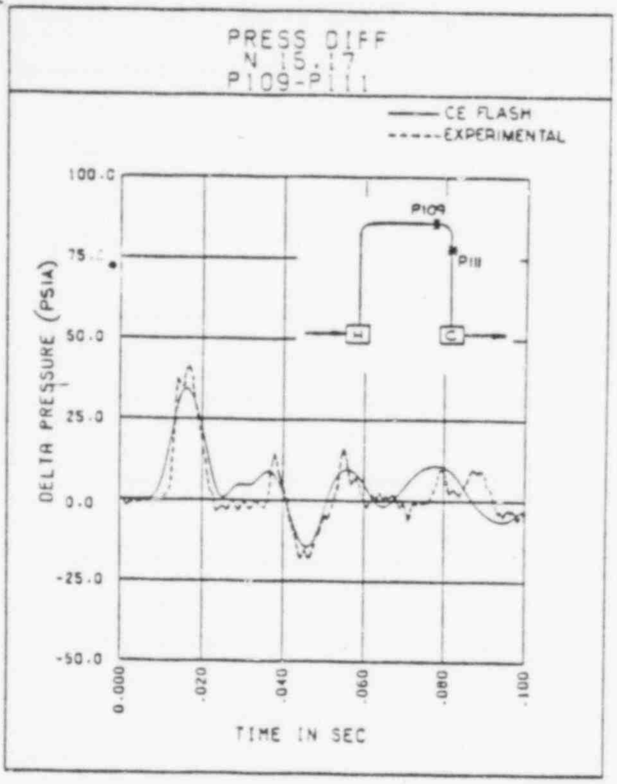


Figure 17a. Measured and Predicted Pressure Differentials Around the Cold Leg Bend, Lower Initial Flow

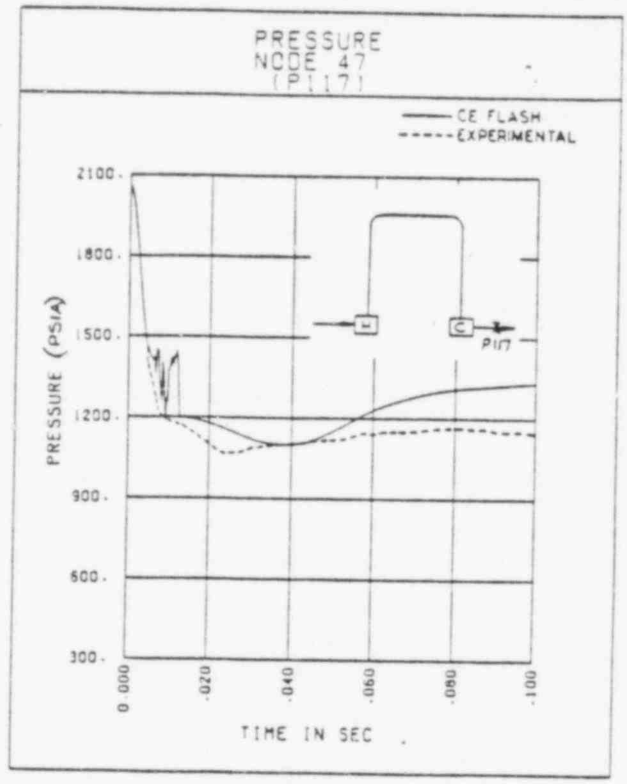


Figure 17b. Measured and Predicted Pressure Histories at Kistler 117 (Node 47), No Initial Flow

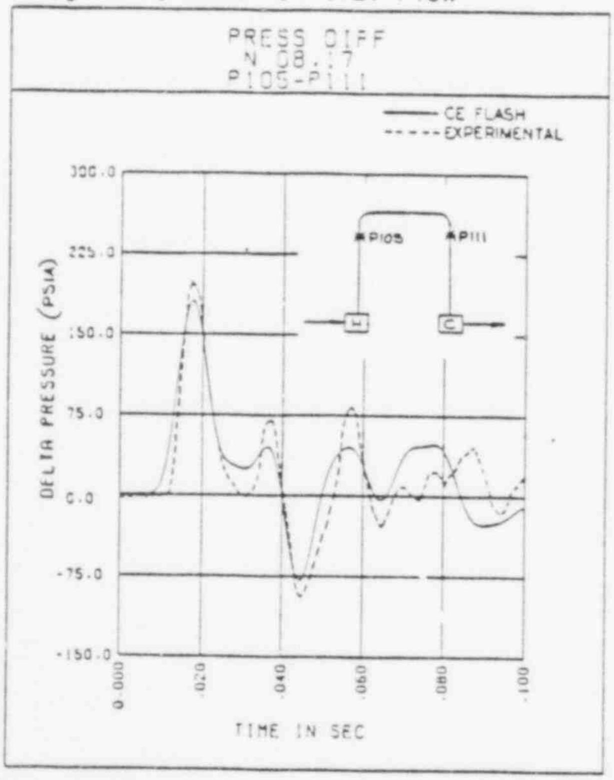


Figure 17c. Measured and Predicted Pressure Histories Across the Horizontal Span, No Initial Flow

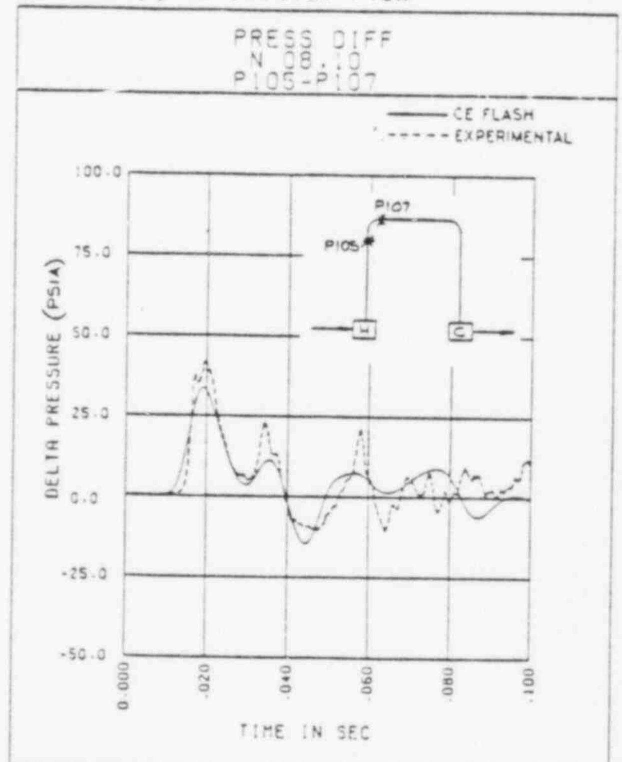


Figure 17d. Measured and Predicted Pressure Histories Around the Hot Leg Bend, No Initial Flow

TYPICAL ANALYSIS VS. TEST DATA COMPARISON

FIGURE 11.

FIGURE 12.

PITTING DEFECT BURST TESTING

TEST CONDUCTED AT ROOM TEMPERATURE

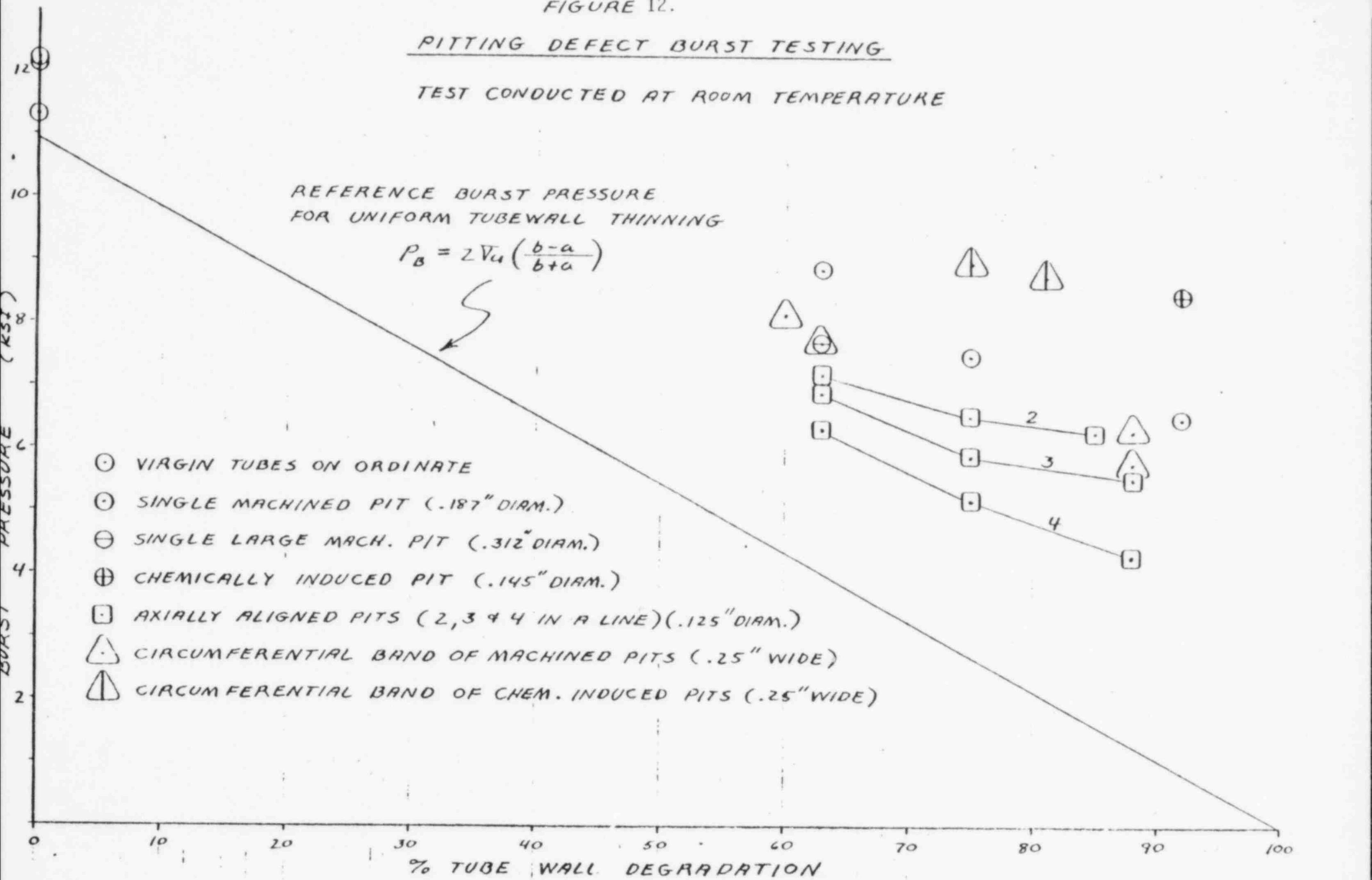
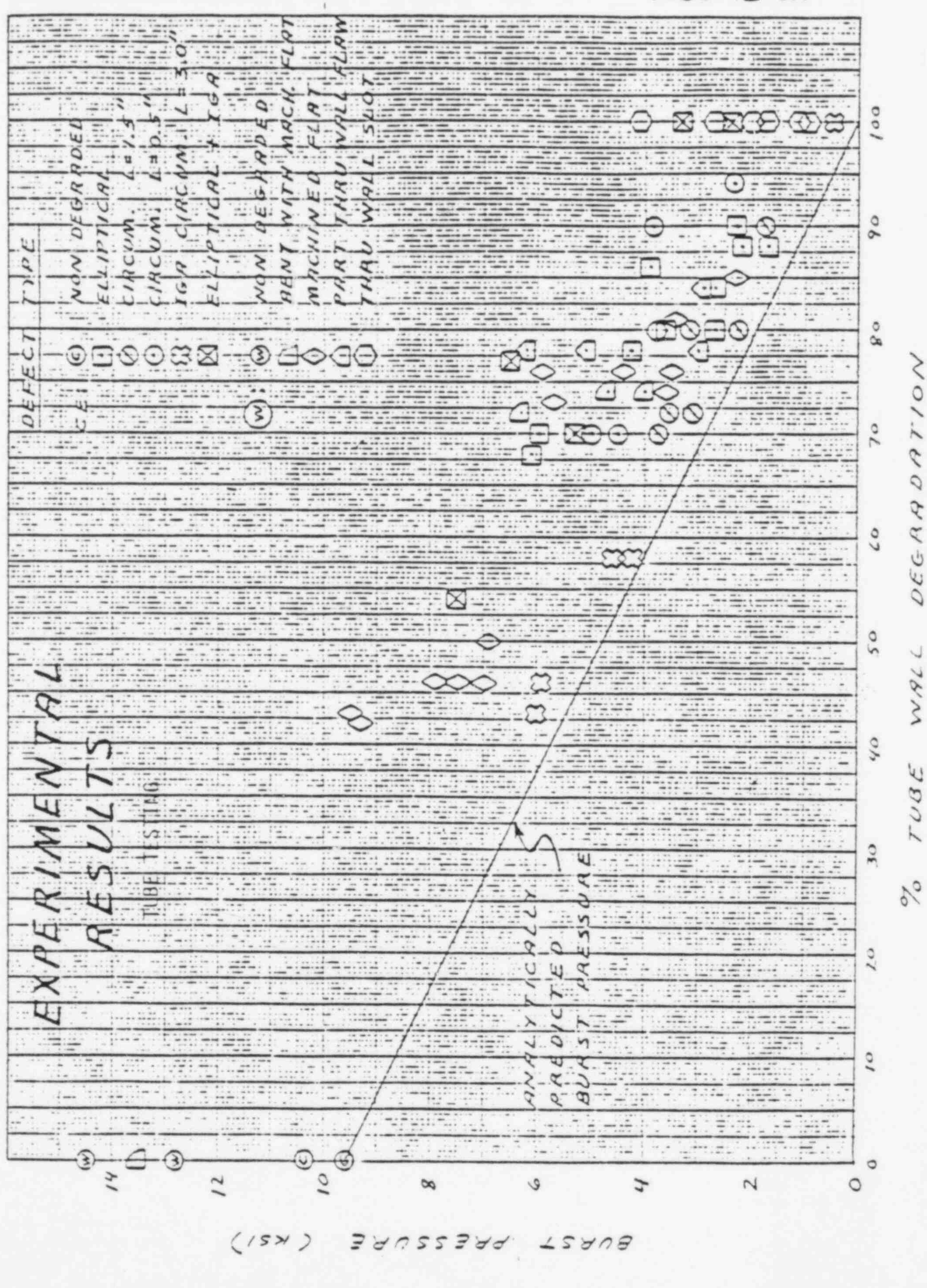


FIGURE 13.


EXPERIMENTAL RESULTS
TUBE TESTING
DEFECT TYPE
CE
NON DEGRADED
ELLIPTICAL
CIRCUM. L=1.5"
CIRCUM. L=0.5"
IGA CIRCUM. L=3.0"
ELLIPTICAL + IGA
W
NON DEGRADED
BENT WITH MACH. FLAT
MACHINED FLAT
PART THRU WALL FLAW
THRU WALL SLOT
ANALYTICALLY PREDICTED BURST PRESSURE
0
10
20
30
40
50
60
70
80
90
100
% TUBE WALL DEGRADATION
0
2
4
6
8
10
12
14
BURST PRESSURE (KSI)

Based on work carried out on NRC/RSR program "PWR Steam Generator Tube Integrity Program", NRC Contract Number B2097 for the Metallurgy and Materials Research Branch, Dr. Joseph Muscara, Program Manager.

by

MILTON VAGINS

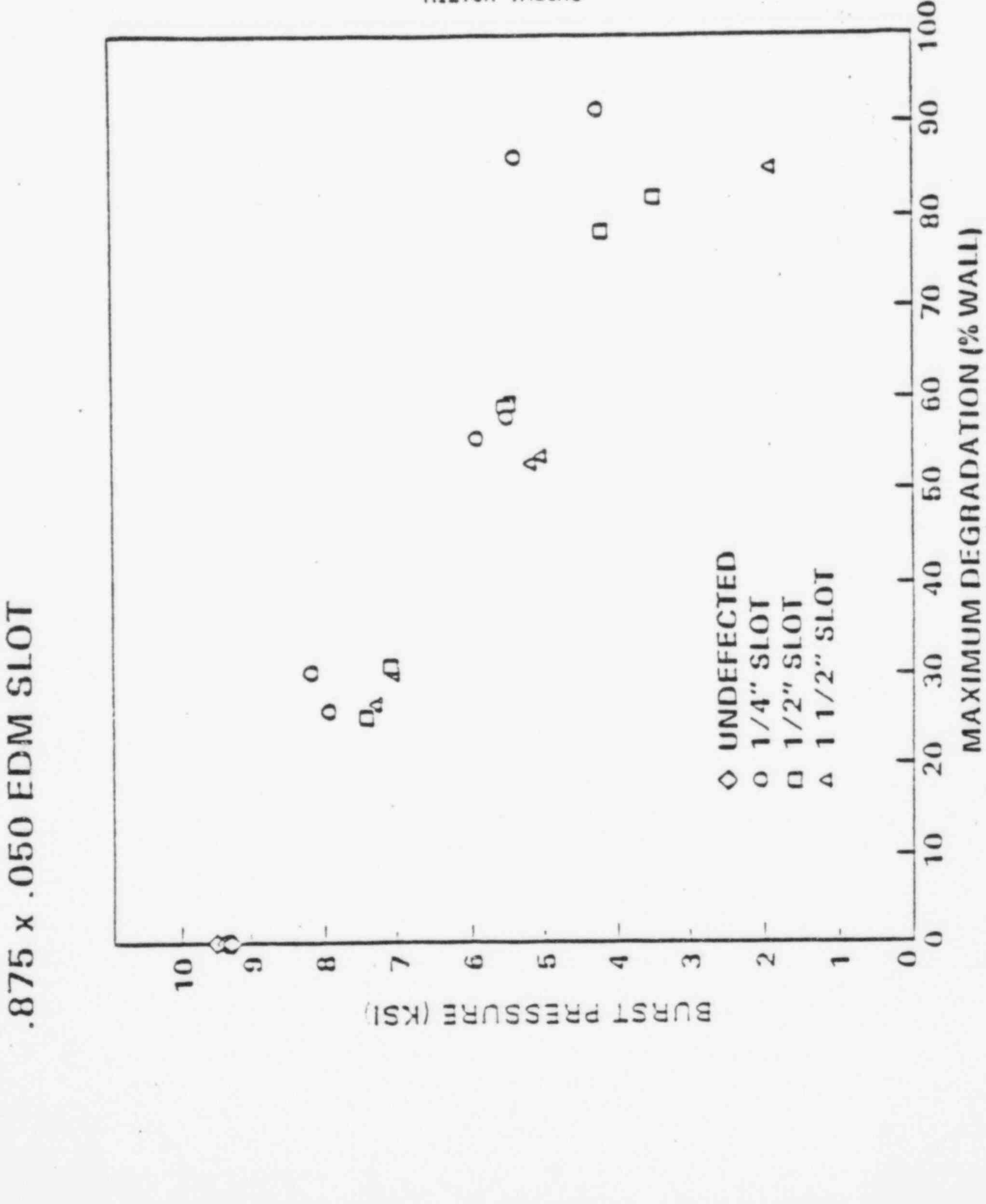


FIGURE 14a. Burst Pressure as a Function of Defect Depth - EDM Slots

.875 x .050 UNIFORM THINNING

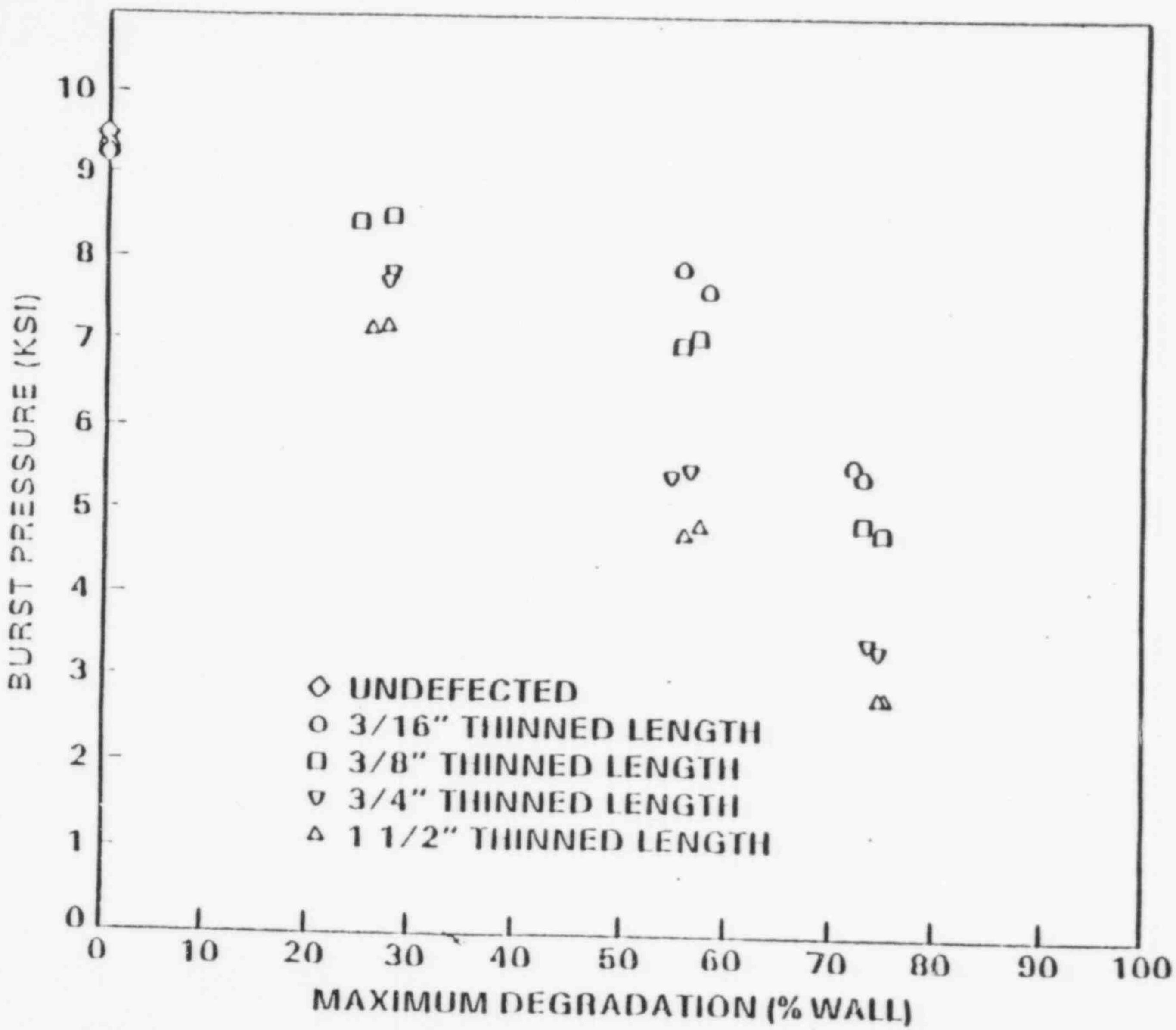


FIGURE 14b. Burst Pressures as a Function of Defect Depth - Uniform Thinning

by

MILTON VAGINS

Based on work carried out on NRC/RSR program "PWR Steam Generator Tube Integrity Program", NRC Contract Number B2097 for the Metallurgy and Materials Research Branch, Dr. Joseph Muscara, Program Manager.

.875 x .050 ELLIPTICAL WASTAGE

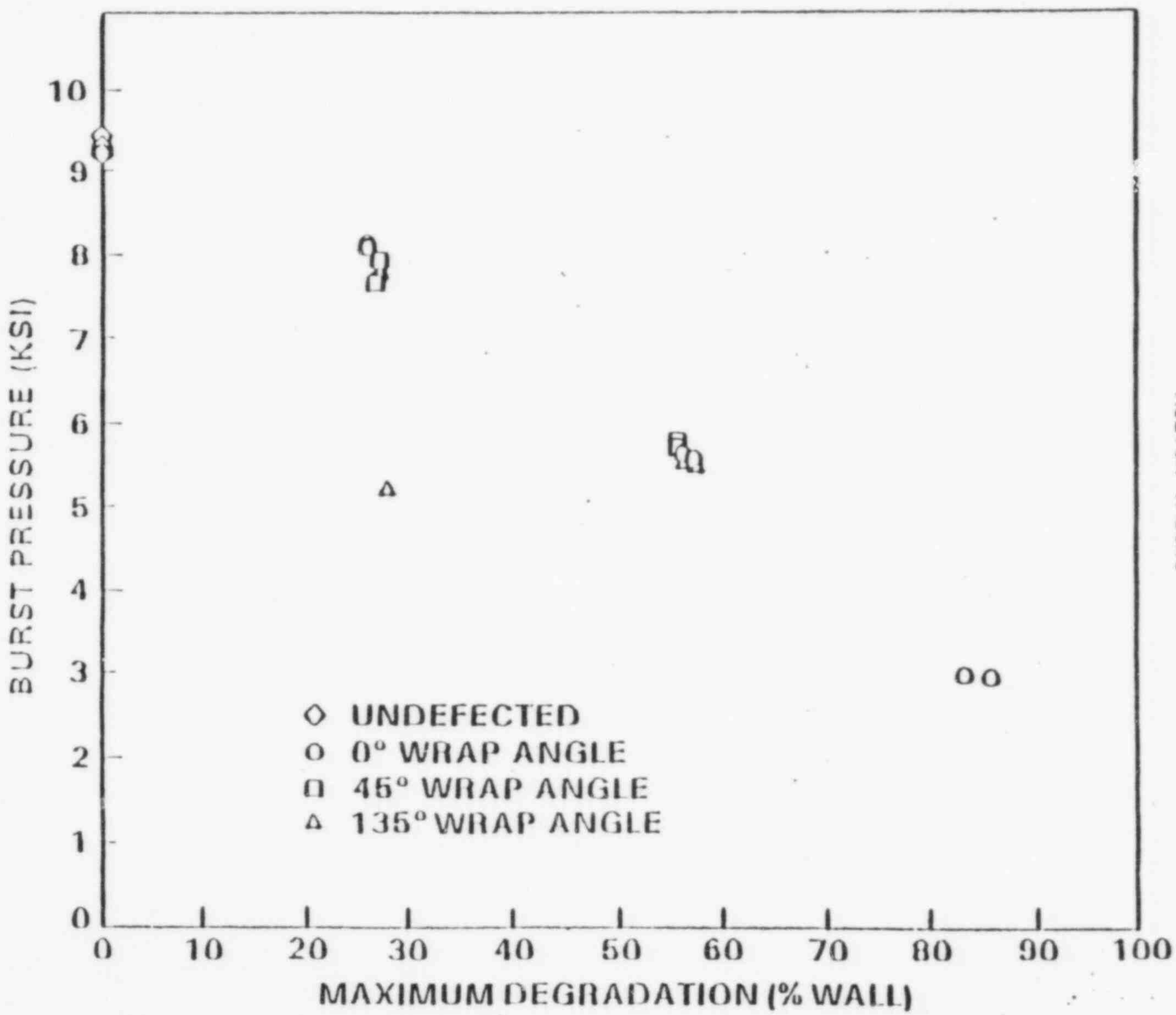


FIGURE 14c. Burst Pressures as a Function of Defect Depth - Elliptical Wastage

by

MILTON VAGINS

Based on work carried out on NRC/RSR program "PWR Steam Generator Tube Integrity Program", NRC Contract Number B2097 for the Metallurgy and Materials Research Branch, Dr. Joseph Muscara, Program Manager.

ENCLOSURE 3 TO LD-83- 058

RESPONSE TO NRC QUESTION 14
ON RAPID DEPRESSURIZATION AND DECAY HEAT REMOVAL

June 1983

Question 14

Fretting wear type damage of steam generator tubes in the vicinity of the feedwater inlet has been observed in certain preheat type steam generators of design similar to the C-E System 80™ steam generators. This damage is attributed to flow induced vibrations originating in the economizer of the steam generator. Provide a description of vibration analyses and model flow testing performed during the design of the C-E System 80 steam generators to assure that no damaging flow induced vibrations would occur in these steam generators.

Response

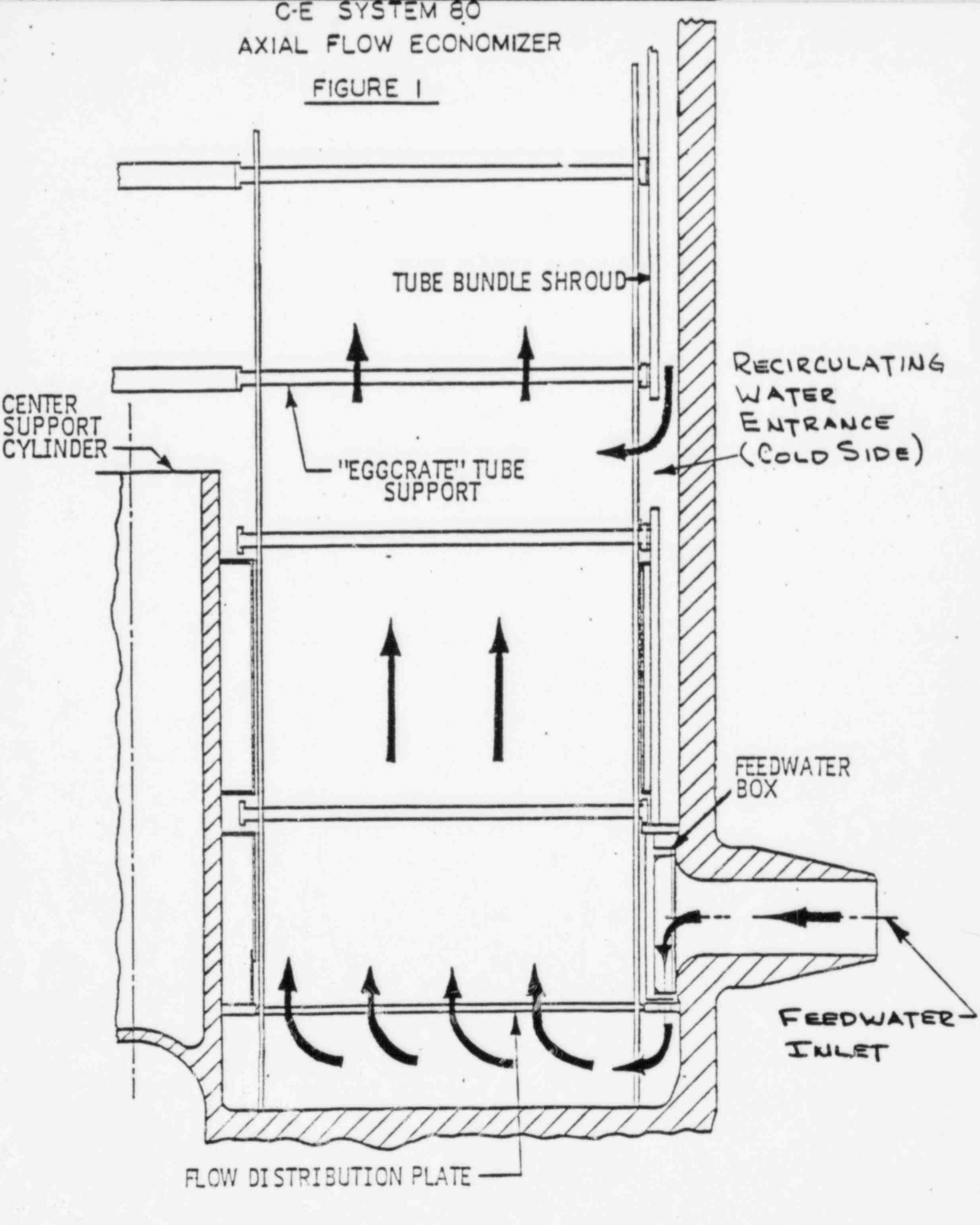
In the C-E axial flow economizer, the feedwater entrance flow is directed down toward the tubesheet by the feedwater box and then upward through the flow distribution baffle. The feedwater entering the economizer is then moving parallel with the economizer tubes (see Figure 1). In addition, recirculating water in the System 80 steam generator cold side enters just above the economizer section. In the economizer design which exhibited tube wear problems (Ringhals), the feedwater entrance flow passes through two perforated plates with non-aligned holes and impinges directly against (perpendicular to) the outermost tubes. Thus, the tube wear problems which occurred at Ringhals cannot occur in the region of the System 80 steam generator economizer entrance. As shown in Figure 1, recirculating water in the System 80 steam generator enters just above the economizer section.

In order to confirm the acceptability of the System 80 econimizer, a full scale flow test was conducted. This report, "Supplementary Information for Appendix A - Capabilities for Depressurization and Decay Heat Removal Without PORV's", is attached. The results of the test confirmed that the vibration at the feedwater entrance was extremely small and that the highest vibration amplitude was in the area of the cold side recirculating entrance. In the worst case, the tests show that the tubes experience no potentially harmful vibrational motion.

C-E SYSTEM 80

AXIAL FLOW ECONOMIZER

FIGURE 1



SUPPLEMENTARY INFORMATION FOR
APPENDIX A
CAPABILITIES FOR THE DEPRESSURIZATION AND DECAY HEAT REMOVAL WITHOUT PORV'S

TABLE OF CONTENTS

LIST OF FIGURES

ABSTRACT

This report summarizes the most recent flow induced vibrations testing performed specifically for the economizer region of the Combustion Engineering System 80 steam generator. Results of these economizer tests confirm that the tubes experience no potentially harmful vibrational motion.

1.0 INTRODUCTION

Combustion Engineering (C-E) has conducted experimental investigations of flow induced vibrations in the economizer region of the C-E System 80 steam generator. Scoping tests were first conducted with a 30° sector of a full-scale model and no tube vibrations of consequence were measured. Recent occurrences of fretting and wear of tubes in economizers of different design prompted C-E to expand the scope of its economizer tests to more thoroughly assess the susceptibility of the Sytem 80 steam generator design to similar damage mechanisms. Test results confirmed that the tubes experience no potentially harmful vibrational motion.

2.0 STEAM GENERATOR DESIGN - ECONOMIZER REGION

The System 80 steam generator design incorporates an integral axial flow economizer on the cold leg side of the tube bundle as shown in Figure 1. The economizer region is formed by a divider plate located in the tube lane and attached to the support cylinder and shell extending to a height of 100 inches above the tubesheet. There are two locations in this region where water enters the tube bundle shown in Figure 2. At the tubesheet, feedwater enters from the feedwater distributor below the flow distribution baffle and flows upward through the bundle. At the top of the economizer, auxiliary feedwater mixed with the cold leg recirculated water enters from the downcomer through an opening in the shroud.

3.0 TEST MODEL AND INSTRUMENTATION

The region of the steam generator which was modelled is shown in Figure 2 and includes both the feedwater and cold leg downcomer inlets to the tube bundle. Tubes, tube supports, tube support spacing, and shell side inlet openings are the same as for the System 80 steam generator. The model is rectangular in shape and constructed from structural steel with plexiglas sides to permit visual studies. It consists of 144 tubes, each 175 inches long which are arranged in a 7 line pattern as shown in Figure 3. The tube array is representative of a bundle with a depth of 20 rows of tubes from the periphery.

Selected tubes near the flow inlets are nstrumented as shown in Figures 3 and 4 with semi-conductor strain gages and bi-directional accelerometers. Penetrations through the plexiglas side are provided at eight elevations downstream of the two inlet openings for insertion of a pitot probe which can be moved horizontally for measuring velocities at positions across a section.

The test model is installed in a loop which consists of a holding tank, a centrifugal pump, flow control valves, flow meters, and orifice plates.

Inlet flow may be admitted to both economizer and downcomer inlet regions. System control valves are manipulated to achieve predetermined axial and radial mass fluxes through the tube bundle.

4.0 TEST CONDITIONS

Hydraulic testing was performed at room temperature with nominal flow rates equivalent to 100% power and for downcomer flows up to 200% nominal. Modeling similitude was based on equality of dynamic pressure. For the 100% case, the specified System 80 feedwater flow was used. The cold leg downcomer flow was determined from the ATHOS (Reference 1) analysis of the System 80 steam generator. The test conditions are summarized in Table 1.

TABLE 1. ECONOMIZER TEST CONDITIONS

Region	Nominal Case 100%	150%	200%
Downcomer Flow, (GPM)	652	978	1304
Economizer Flow, (GPM)	215	215	215
Cross Flow (Span 4), (GPM)	385	525	668

5.0 RESULTS

A. Hydraulics

Velocity distributions of the shell side fluid downstream of the two inlet openings were established from measurements made at the eight vertical and four horizontal intersecting locations shown in Figures 3 and 4. A two-dimensional "wedge" pitot probe was used for measuring the direction and magnitude of flow velocity at each grid point. In Figures 5 through 8, resultant velocity distributions are presented in vector form with the horizontal velocity component shown plotted vs opening height.

B. Tube Vibration

The maximum tube vibration amplitudes occurred in tube no. 4 (Figure 3). Figure 9 gives the measured deflections and Figure 10 shows the vibration amplitude profile. In addition, the following observations are made:

(1) The tube motion was elliptical with the major axis in the transverse direction.

(2) The largest observed vibration amplitudes occurred in the span above the cold side downcomer fluid entrance region (Span 5, Figure 10).

(3) The level of vibration in the tube span subjected to cold side downcomer fluid (Span 4) was relatively constant at 0.4 mil up to approximately 150% flow. The bending stress is less than 1 ksi for 100% flow.

(4) No vortex shedding induced vibration was observed for two reasons: (1) the fluid approaching the bundle was too turbulent, and (2) the triangular pitch tube array is so tightly packed that vortices cannot be sustained.

(5) When the velocity profiles shown in Figures 5 through 8 were examined using the methodology described in Reference 2, it was concluded that there is at least a 50% margin to instability at 100% power.

(6) Vibration of tubes in the feedwater entrance region of the tube bundle are extremely small as was predicted. All of C-E's operating steam generators have higher levels of vibration at the bundle entrance regions than will exist at the System 80 feedwater entrance region, due to the greater velocity of the recirculating fluids.

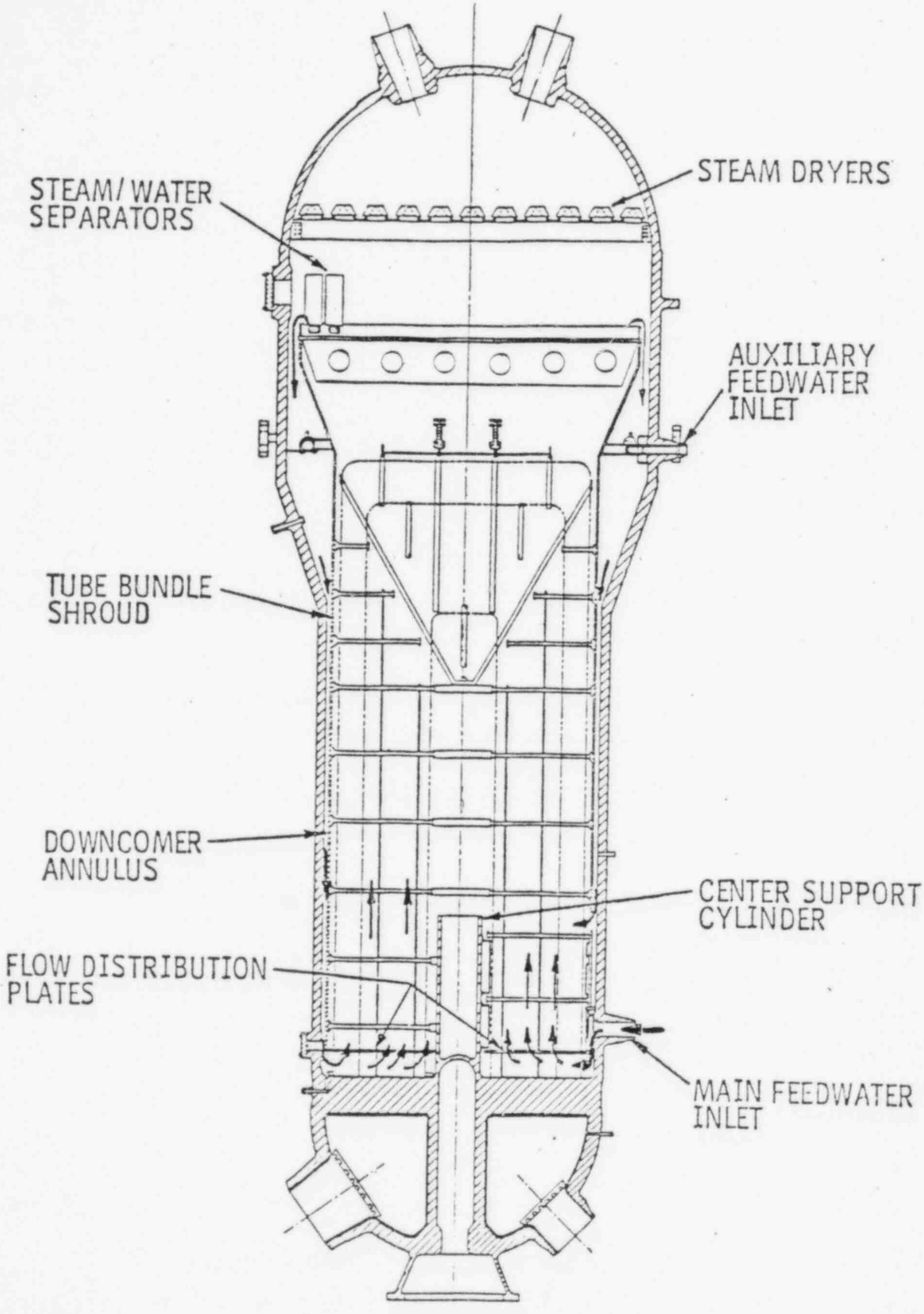
6.0 CONCLUSION

A full scale test of the System 80 steam generator economizer region was performed to investigate the vibrational response of tubes when subjected to cross flow due to water issuing from inlet openings. Both the feedwater inlet at the tubesheet and the recirculated water inlet at the top of the economizer region were included in the model. Test runs were made for nominal prototypic flow conditions and for recirculated water flow up to 200% nominal. It is concluded from results of the tests that tubes in the System 80 economizer region will experience no detrimental vibrational motion during normal operation.

7.0 REFERENCES

1. ATHOS - A Computer Program for Thermal Hydraulic Analysis of Steam Generators, December 1981, A. K. Singhal, L. W. Keeton, A. J. Przekwas, and J. S. Weems, CHAM of North America, Inc., Huntsville, Alabama.

2. "Fluidelastic Vibration of Heat Exchanger Tube Arrays", Journal of Mechanical Design, Volume 100, 1978, pp 347-353, by H. J. Connors, Jr.

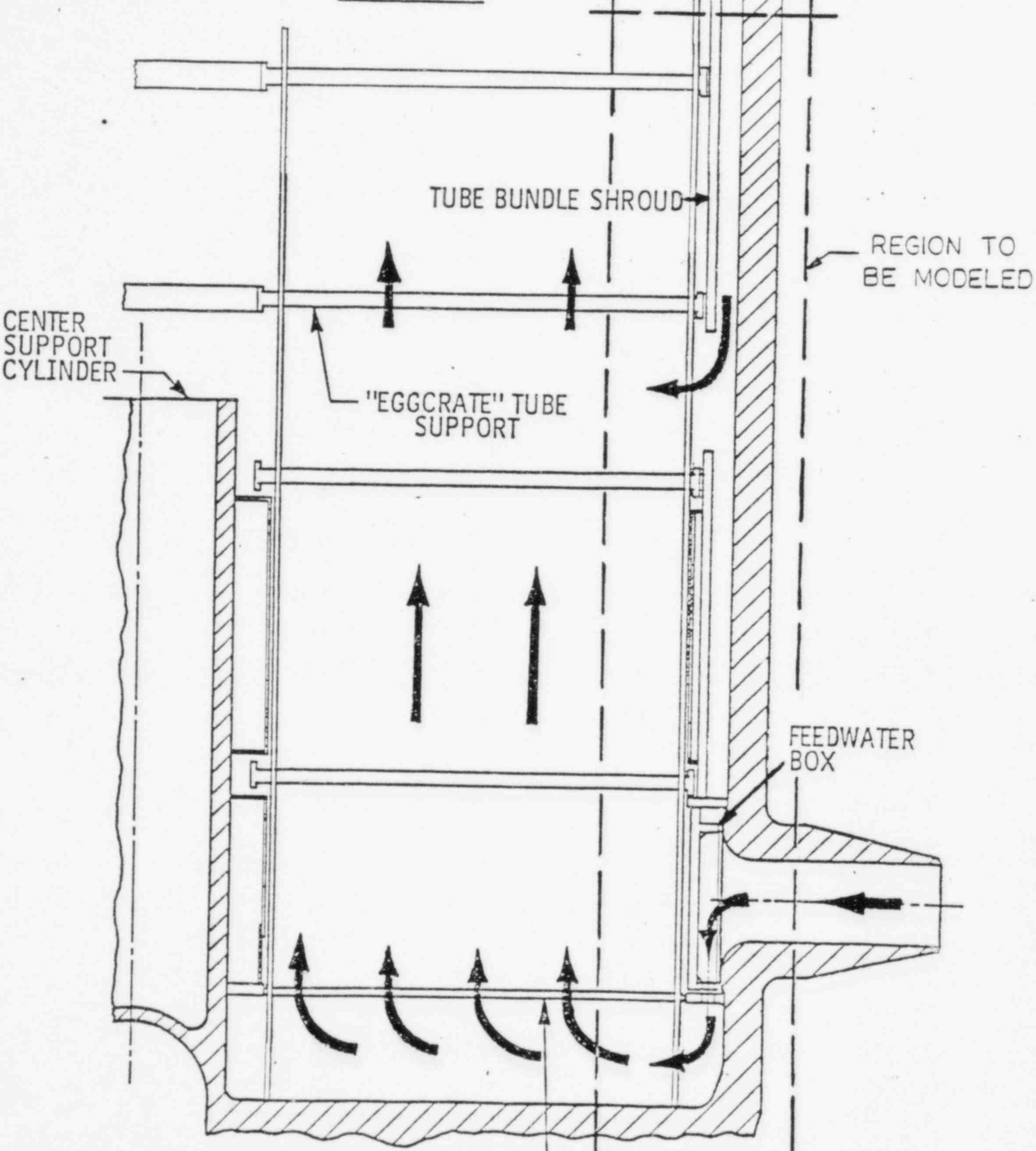


INTEGRAL ECONOMIZER STEAM GENERATOR

AXIAL FLOW

FIGURE 1

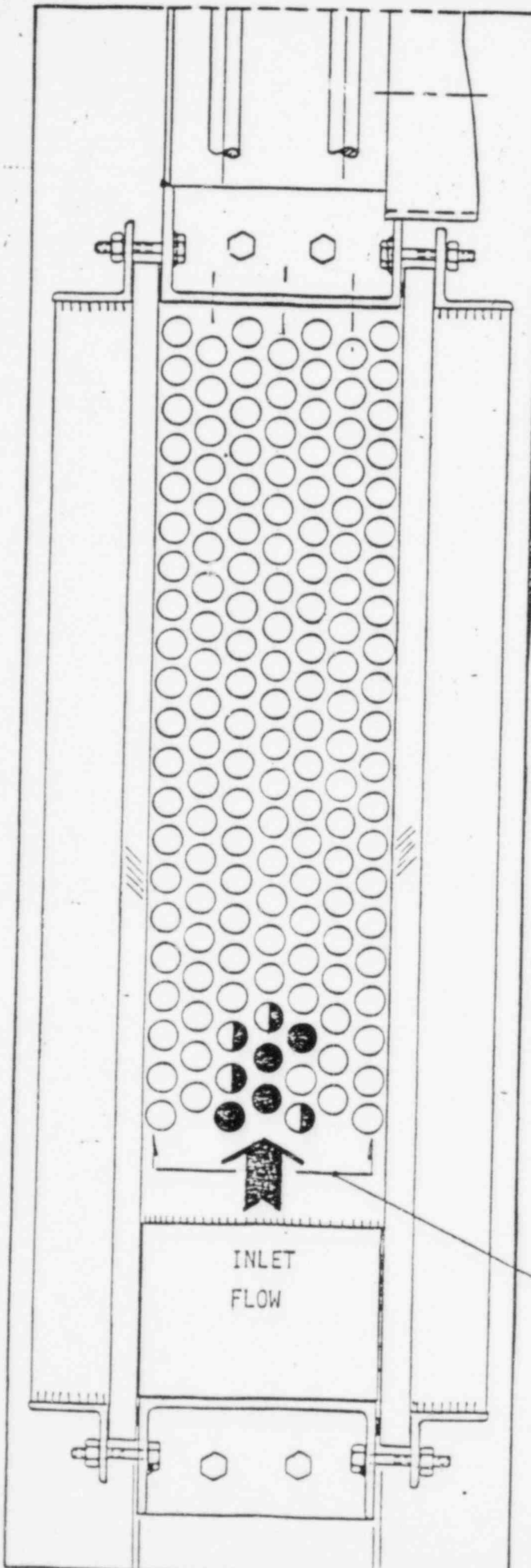
C-E SYSTEM 80
AXIAL FLOW ECONOMIZER
FIGURE 2

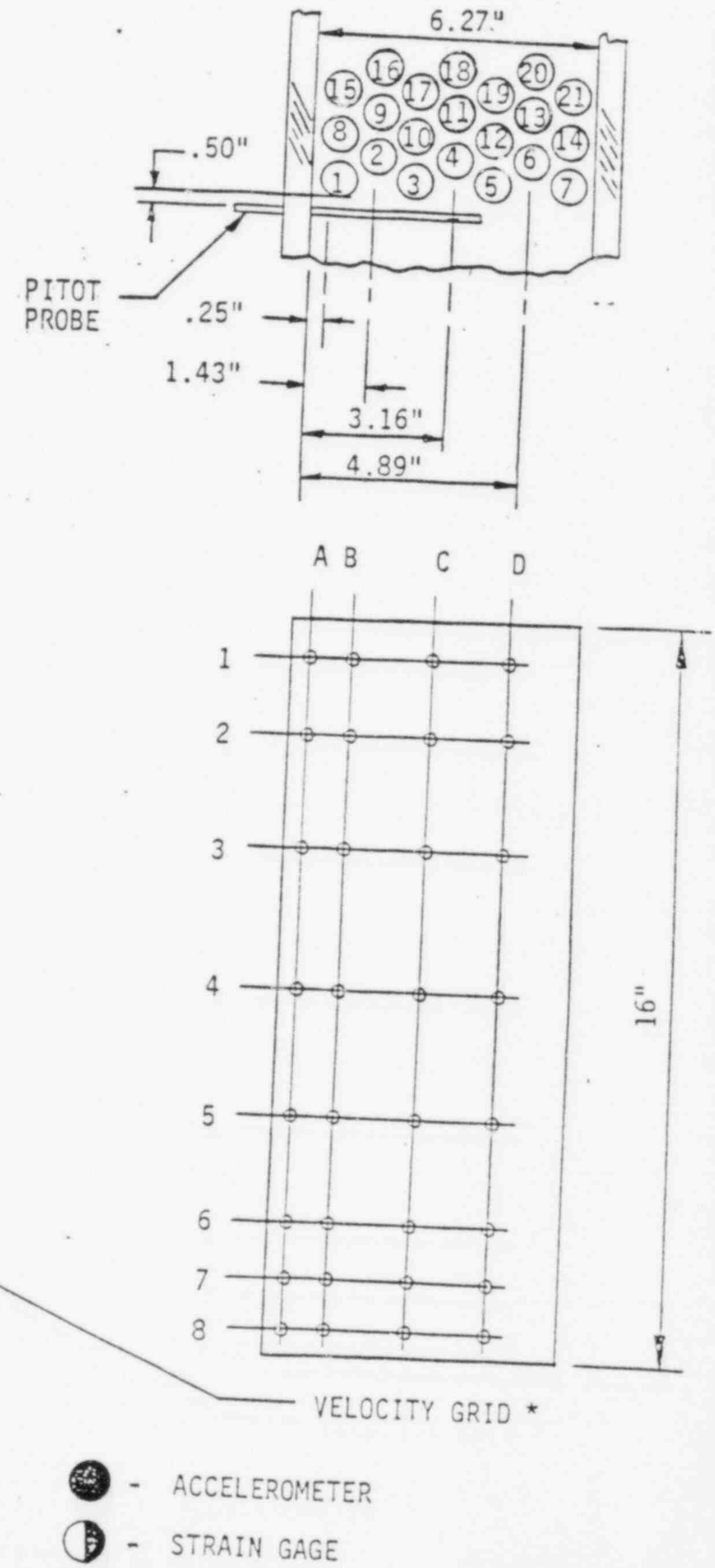


MODEL INSTRUMENTATION AND MEASUREMENT POSITIONS (HORIZONTAL)

(TYPICAL AT TWO INLET ELEVATIONS)

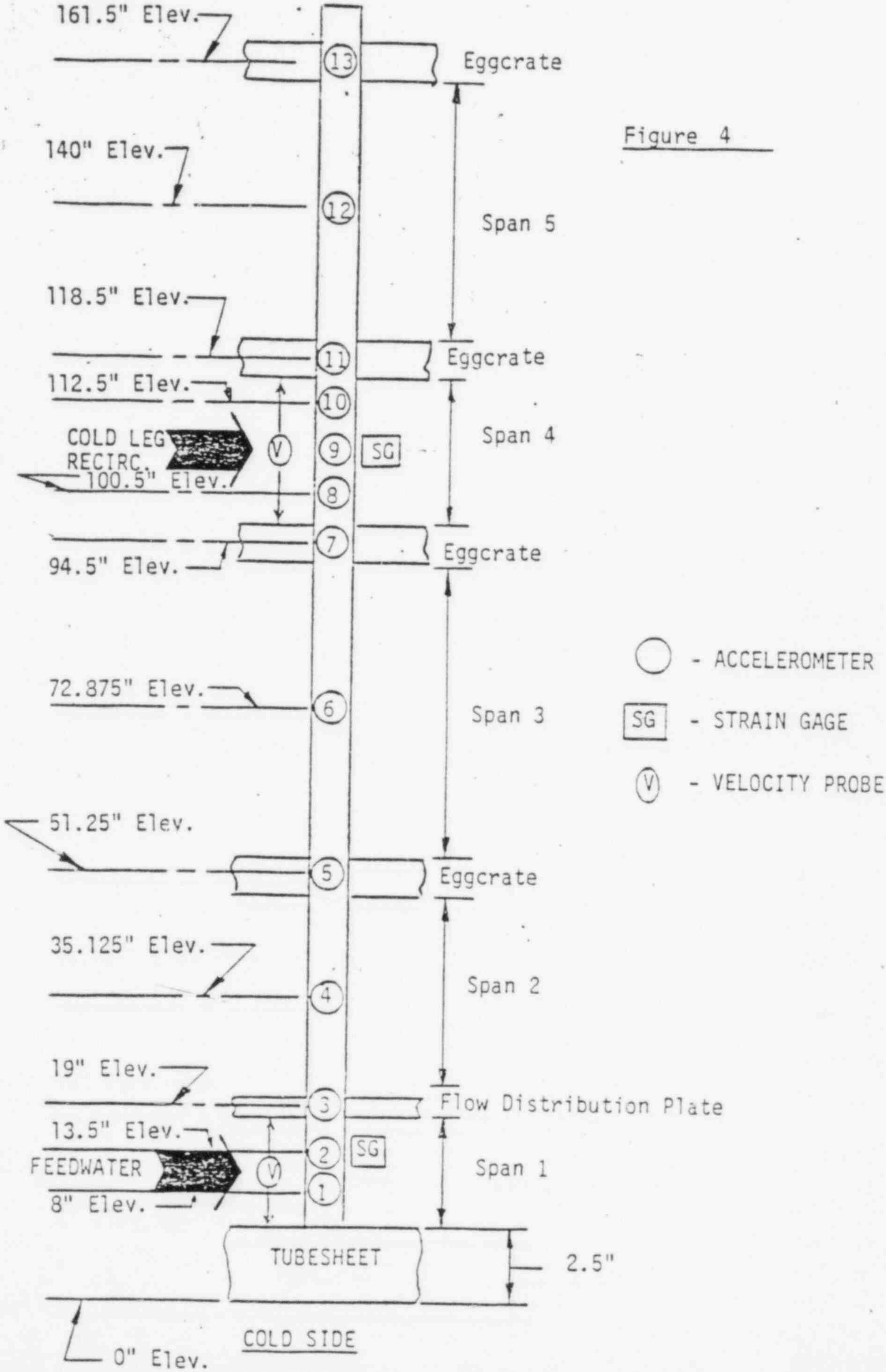
FIGURE 3





MODEL INSTRUMENTATION AND MEASUREMENT POSITIONS (VERTICAL)

Figure 4



VELOCITY DISTRIBUTION AT ECONOMIZER INLET 100% FLOW

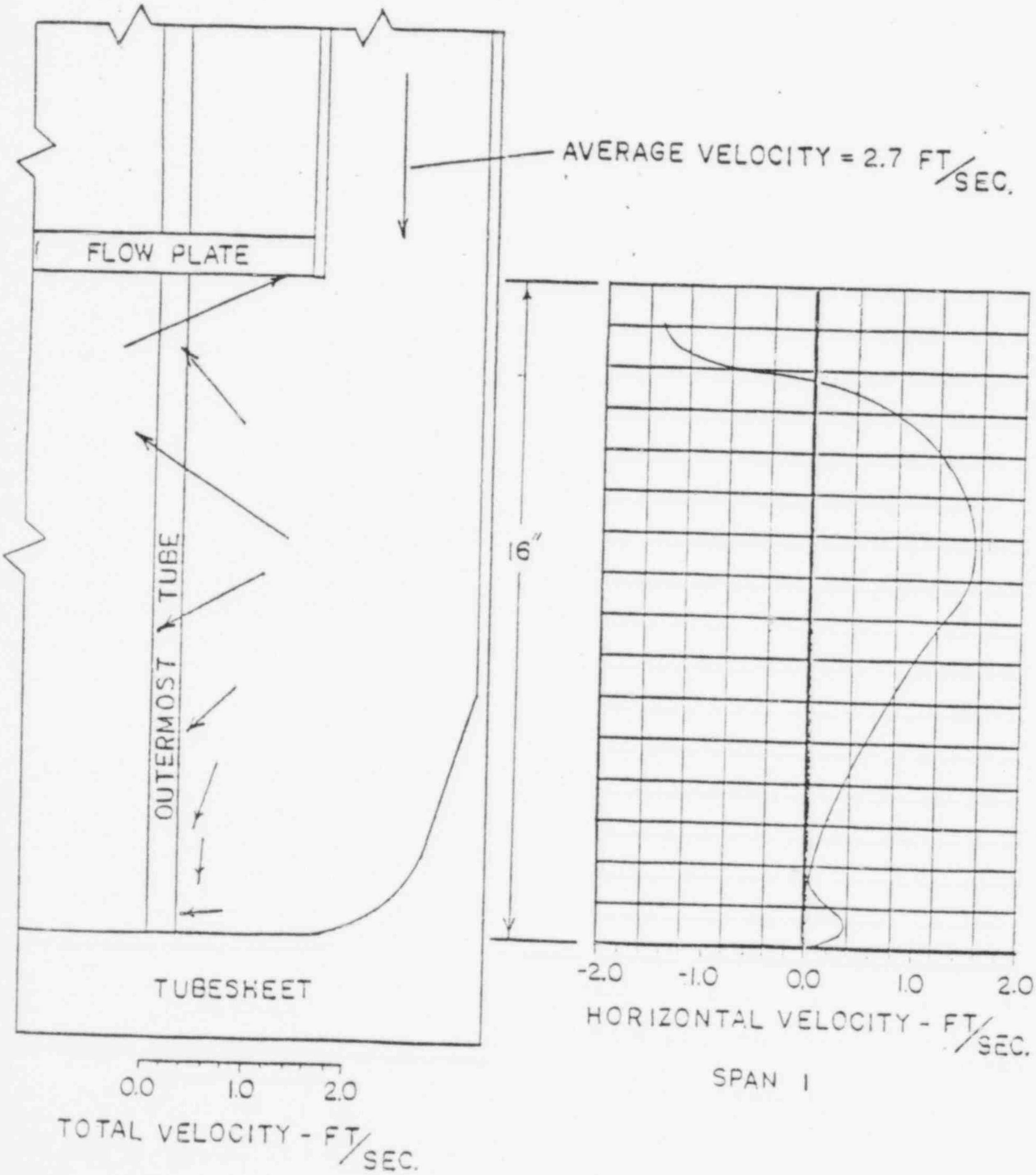
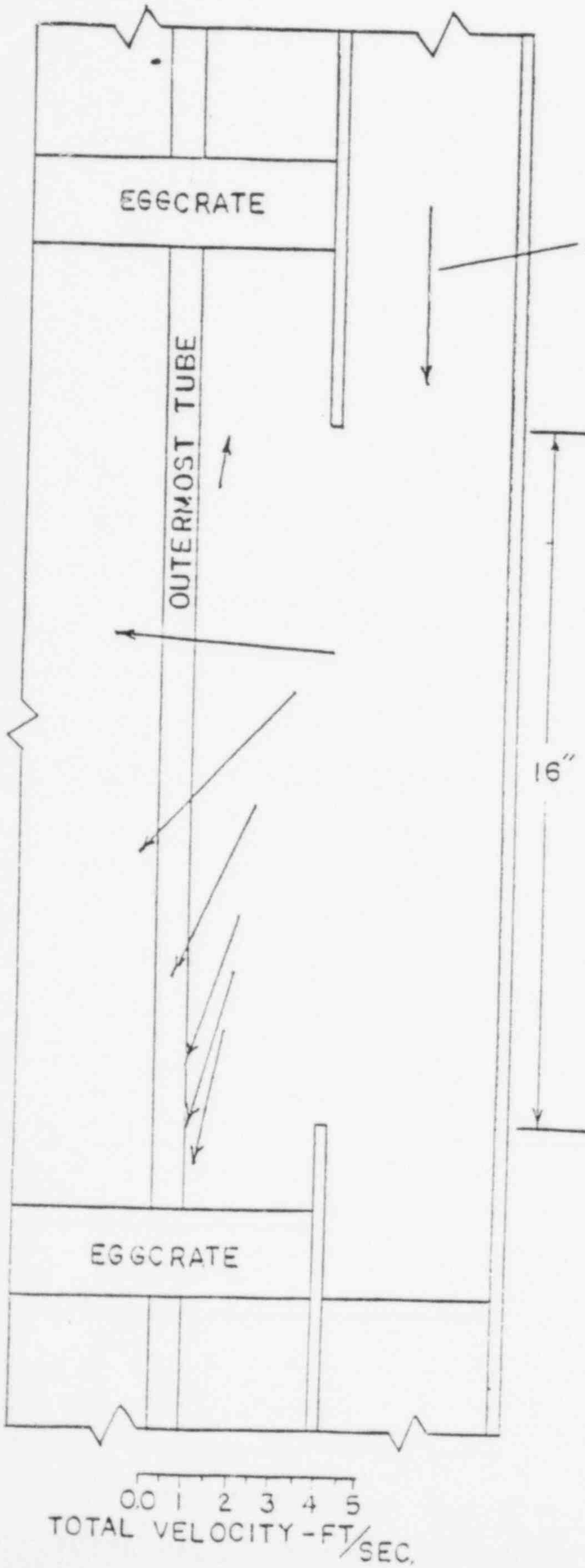


FIGURE 5
ECONOMIZER INLET

VELOCITY DISTRIBUTION
AT
COLD LEG DOWNCOMER INLET
100% FLOW



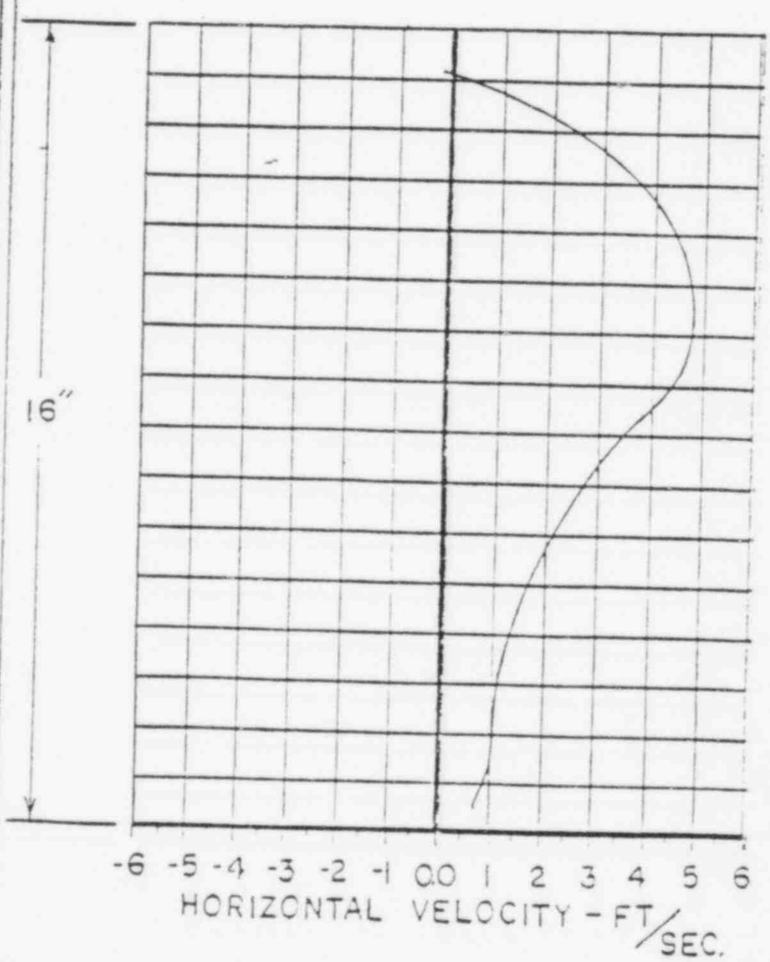


FIGURE 6 - 100% FLOW

DOWNCOMER INLET - COLD SIDE

SPAN 4

VELOCITY DISTRIBUTION AT COLD LEG DOWNCOMER INLET 150% DOWNCOMER FLOW

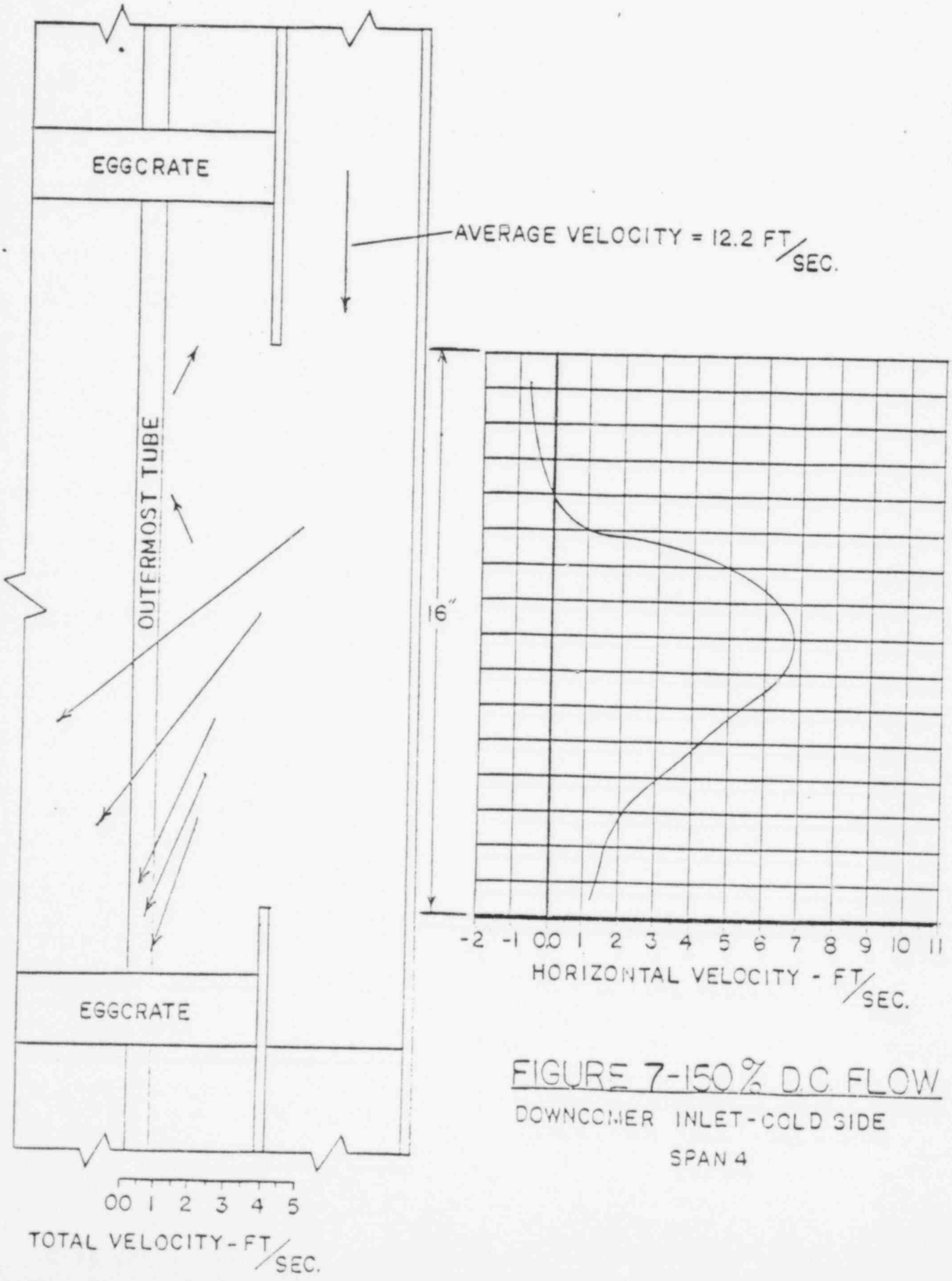


FIGURE 7-150% D.C. FLOW

DOWNCOMER INLET-COLD SIDE

SPAN 4

VELOCITY DISTRIBUTION AT COLD LEG DOWNCOMER INLET 200% DOWNCOMER FLOW

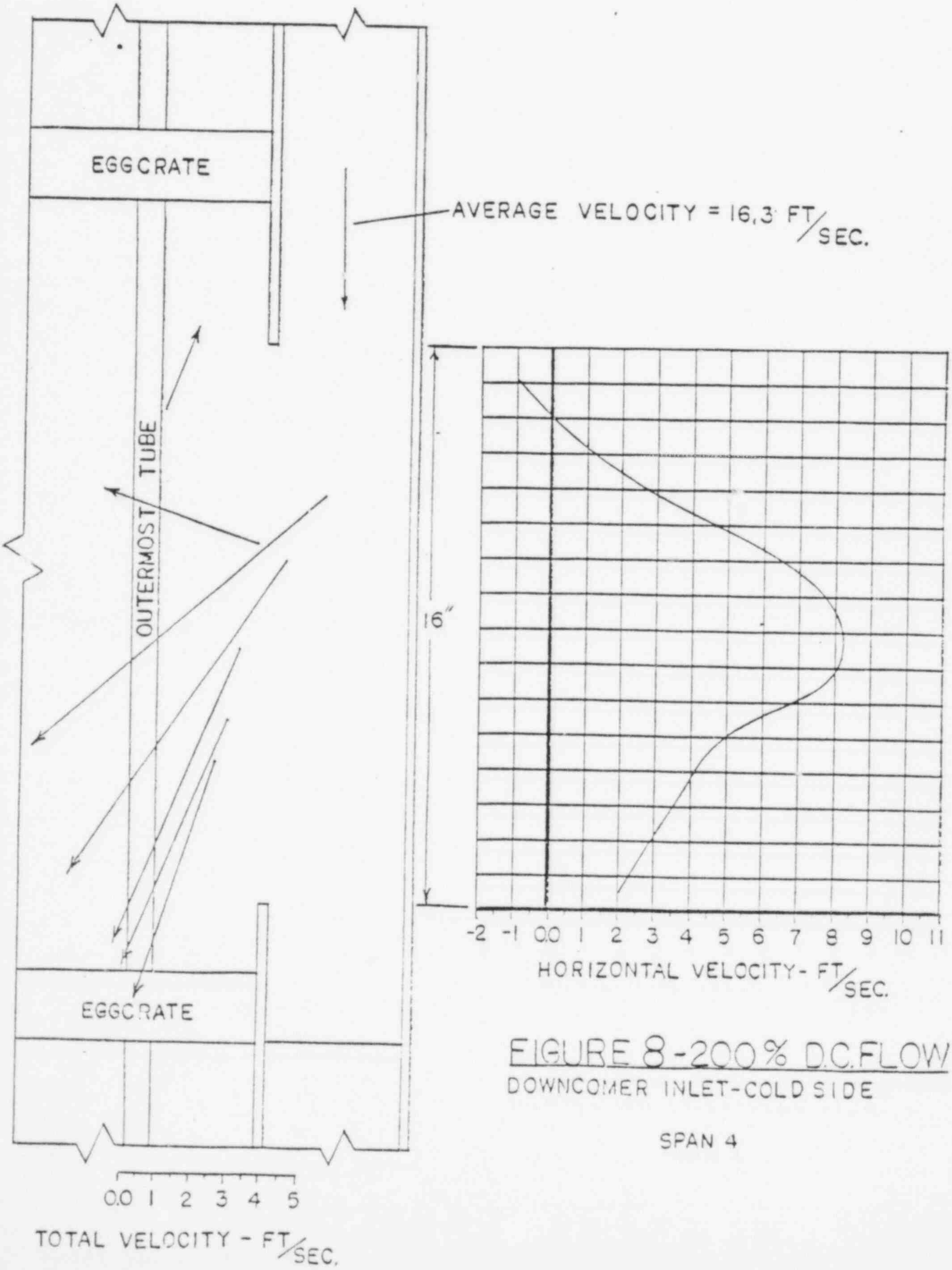


FIGURE 8 - 200% D.C. FLOW

DOWNCOMER INLET-COLD SIDE

SPAN 4

FIGURE 9

ECONOMIZER REGION FLOW VIBRATION TEST:

MEASURED DEFLECTIONS OF TUBE #4

	Probe Elevation*	Deflection, δ_R-mils 100% Flow	150% Flow	200% Flow
Feedwater Entrance	0	0.16	0.11	
	1	0.16	0.11	
	2	0.15	0.11	
	3	0.17	0.11	
	4	0.34	0.39	
	5	0.17	0.16	
	6	0.49	0.67	
Downcomer Entrance	7	0.29	0.15	
	8	0.27	0.22	
	9	0.41	0.35	1.12
	10	0.32	0.36	
	11	0.27	0.17	
	12	1.24	1.77	2.20
	13	0.34	0.36	

*Elevation numbers correspond to those shown in Figure 10.

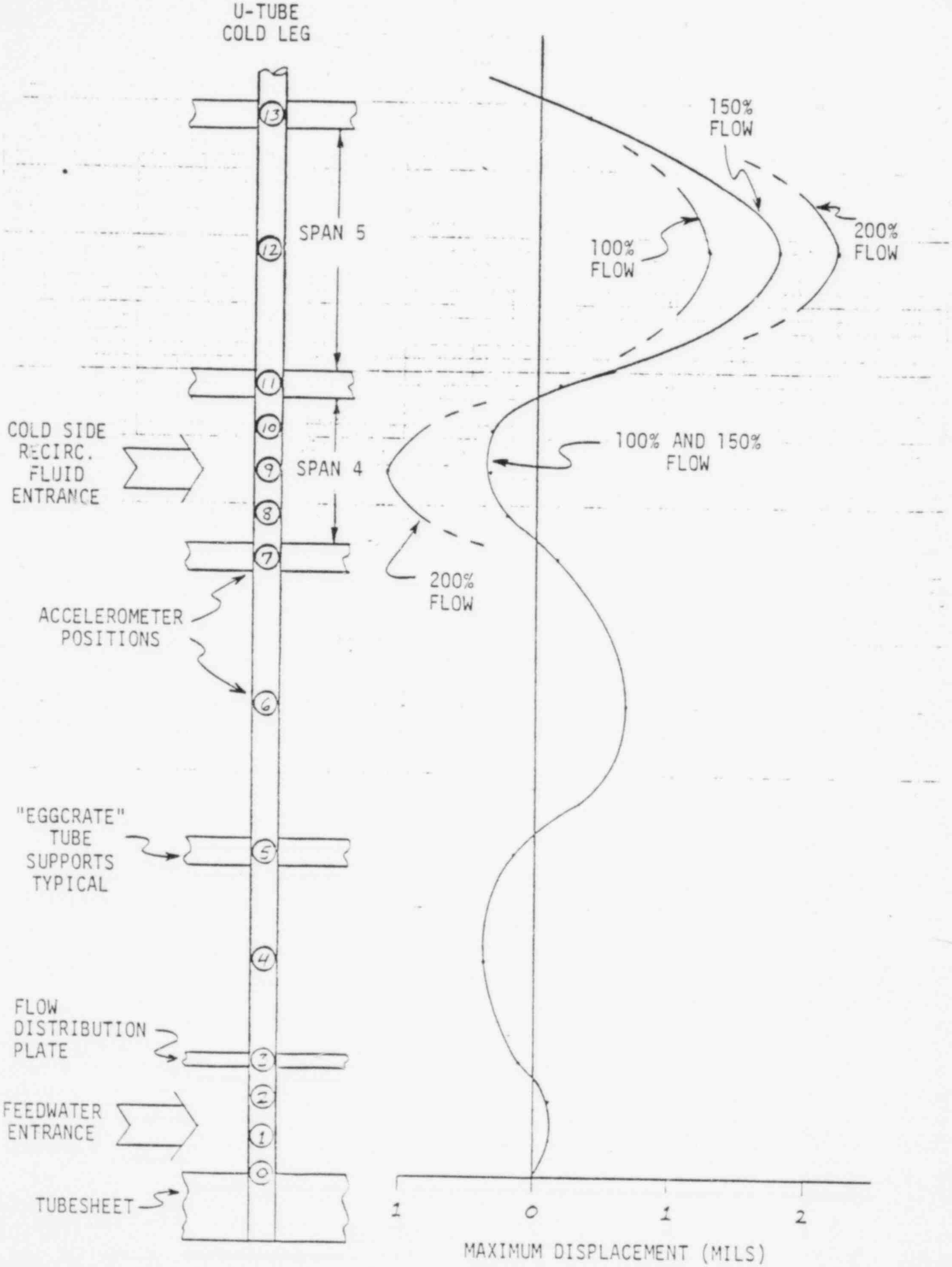


FIGURE 10. TUBE VIBRATION AMPLITUDE PROFILE