ATTACHMENT A

Evaluation of Susceptibility To High Cycle Fatigue Failure of Steam Generator Tubes

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
November, 1987

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EVALUATION OF DENTED TUBE FATIGUE SUSCEPTIBILITY FOR STEAM GENERATORS 22 AND 24 AT INDIAN POINT UNIT 2

INTRODUCTION

The Indian Point 2 steam generators 22 and 24 have been evaluated for the susceptibility of unsupported U-bend tubing with denting at the top tube support plate to a fatigue failure of the type experienced at Row 9 Column 51 (R9C51) of Steam Generator C, North Anna 1.

BACKGROUND

The initiation of the circumferential crack in the tube at the top of the top tube support plate at North Anna 1 was due to limited displacement, fluid elastic instability. The unstable condition prevailed in the R9C51 tube since the first fuel cycle when the tube experienced denting at the support plate. A combination of conditions were present that led to the failure. The tube was not supported by an anti-vibration bar (AVB), had a higher flow field due to the uneven insertion depths of AVB's, had reduced damping because of denting at the top support plate, and had reduced fatigue proportion because of the additional mean stress with the denting and because of the all volatile treatment (AVT) chemistry of the secondary water.

CRITERIA FOR EVALUATION

The criteria established for determining the benefit of corrective action was a 10% reduction in stability ratio that is expected to provide a 58% reduction in stress amplitude (to $<$ 4.0 ksi) for a Row 9 tube in the North Anna 1 steam generators. This reduction is expected to reduce the future fatigue usage to $<$ 0.021 per year for a Row 9 tube and to preclude growth of an existing through wall crack of up to 0.125 inch in length. This same criterion has been used as one of several criteria in the evaluation of Indian Point 2 tubing. With additional effects accounted for through a stress ratio criteria that permits the calculation of fatigue usage to demonstrate tube acceptability, the final criteria of cumulative fatigue usage to date plus future operation with current operating parameters has to be satisfied.

The stability ratios for Indian Point 2 tubing, the corresponding stress amplitude, and the resulting cumulative fatigue usage must be evaluated relative to the failed tube at Row 9 Column 51, North Anna 1, Steam Generator C, for two reasons. First, the local effect on the flow field due to various AVB insertion depths is not within the capability of available analysis techniques and must be determined by test. Second, an analysis and examination of the ruptured tube provided a range of initiating stress amplitudes but can only bound the possible stability ratios that correspond to those stress amplitudes. Therefore, the evaluation of Indian Point 2

tubing had been based on relative stability ratios, relative flow peaking factors and stress ratios.

The determination of stability ratio is the evaluation of a ratio of velocities, the effective velocity divided by the critical velocity. A value greater than unity (1.0) indicates instability. The stress ratio is the expected stress amplitude in an Indian Point 2 tube divided by the stress amplitude responsible for the North Anna 1 tube rupture.

The criteria for establishing that a tube has support from an AVB and therefore eliminate it from further considerations is that at least one sided support is present to at least the tube centerline. This has been established by analysis of eddy current (EC) measurements by either an EC indication of both legs of the AVB or by projecting the depth of insertion knowing the geometry of the AVB and EC indications that locate it on at least three larger radius U-bends in the same column of tubes.

RESULTS OF EVALUATION

The AVB insertion depths are a key factor in the assessment of fatigue failure susceptibility since the AVB positions determine the local flow peaking factors. The local flow peaking factor is a direct factor on the stability ratio and a small percentage change causes a significant change in stress amplitude. The relative flow peaking factors for Indian Point 2 tubing without direct AVB support have been determined by instability tests. These factors have been applied to relative stability ratios determined by 3-D tube bundle flow analysis and the combined relative stability ratio used in the stress ratio determination.

The analysis of eddy current (EC) measurements shows that denting is present at the top support plate and therefore all of the tubes in Rows 8 to 12 are treated as dented. The effect of denting on the fatigue life of the tube has been conservatively maximized by assuming the maximum effect of mean stress (based on 55 ksi) in the tube fatigue usage evaluation and by incorporating reduced damping in the tube vibration evaluation. A partial review of EC measurements also shows that none of the reviewed tubes in Row 8 through 12 have wall thinning indications at the AVB locations. Therefore, it is unlikely that those tubes have been unstable. Additionally, the eddy current data have been used to define AVB insertion depths and the extent of tube support for each column of tubes within the region. As shown in Figures 1 and 2, virtually all row 12 tubes are supported except for a few peripheral columns. Most row 11 and row 10 tubes are supported. Some row 9 and row 8 tubes are supported.

Nine tubes (not including symmetrical tube locations with identical flow conditions) are identified as the most critical of the dented and unsupported tubes. These are:

Steam Generator 22

Row	Column
12	2
11	3
11	37
10	24

Steam Generator 24

Row	Column
12	2
11	3
11	51
11	46
10	71

The local flow peaking corresponding to these AVB insertion depths has been factored into the evaluation for each of the above tubes. This results in the relative stability ratios and stress ratios shown below. (All ratios are in comparison to R9C51, North Anna 1, Steam Generator C.)

Steam Generator 22

Tube	Relative Stability Ratio	Stress Ratio
R12C2	0.79	0.35
R11C3	0.84	0.55
R11C37	0.73	0.25
R10C24	0.68	0.19

Steam Generator 24

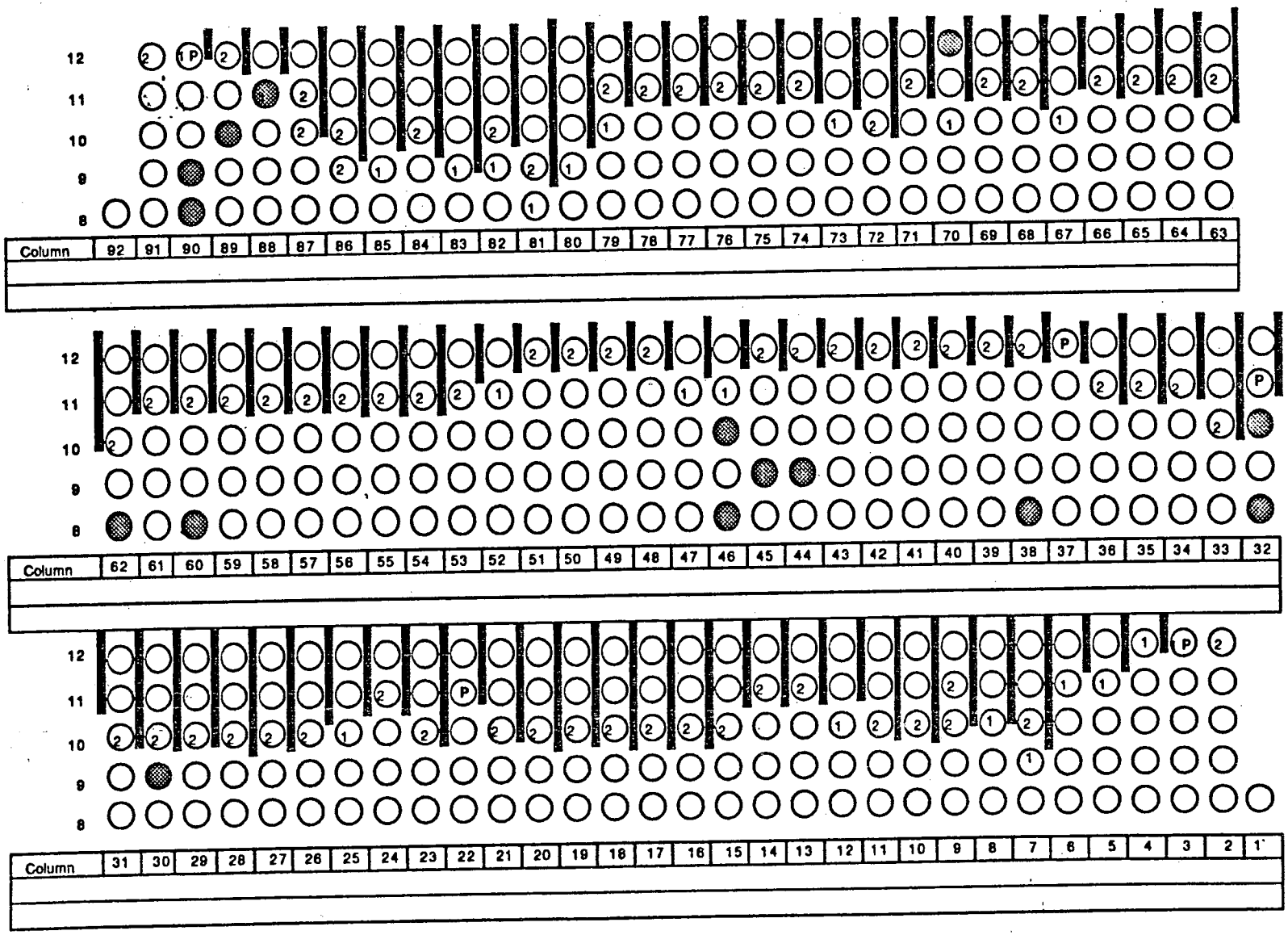
Tube	Relative Stability Ratio	Stress Ratio
R12C2	0.79	0.35
R11C3	0.84	0.55
R11C51	0.74	0.27
R11C46	0.84	0.58
R10C71	0.73	0.29

All of the above most critical tubes have relative stability ratios less than 0.9 and stress ratios less than 1.0. Therefore, all unsupported and dented tubes at Indian Point 2 meet these two criteria. The tube with the highest stress ratio is at Row 11 Column 46 with a ratio of 0.6. The stress amplitude for this tube is therefore 2.4 ksi. Based on the operating history of Indian Point 2 which is presented in Figure 3 in the form of normalized stability ratio versus time in days, the fatigue usage to date is 0.017 and the projected 40 year usage is 0.06. This is less than the limit of 1.0 and therefore meets the fatigue usage criteria. The fatigue usage calculation utilizes a lower bound fatigue curve that is consistent with the fatigue properties of the North Anna 1 tube that ruptured.

CONCLUSION

Based on these evaluations, it is concluded that the Indian Point Unit 2 steam generator 22 and 24 tubes are not susceptible to fatigue rupture at the top support plate similar to that which occurred at North Anna Unit 1. Therefore no modifications, preventive tube plugging or other measures to preclude such an event are required for these steam generators.

Indian Point - 2 : S/G - 22
AVB Insertion



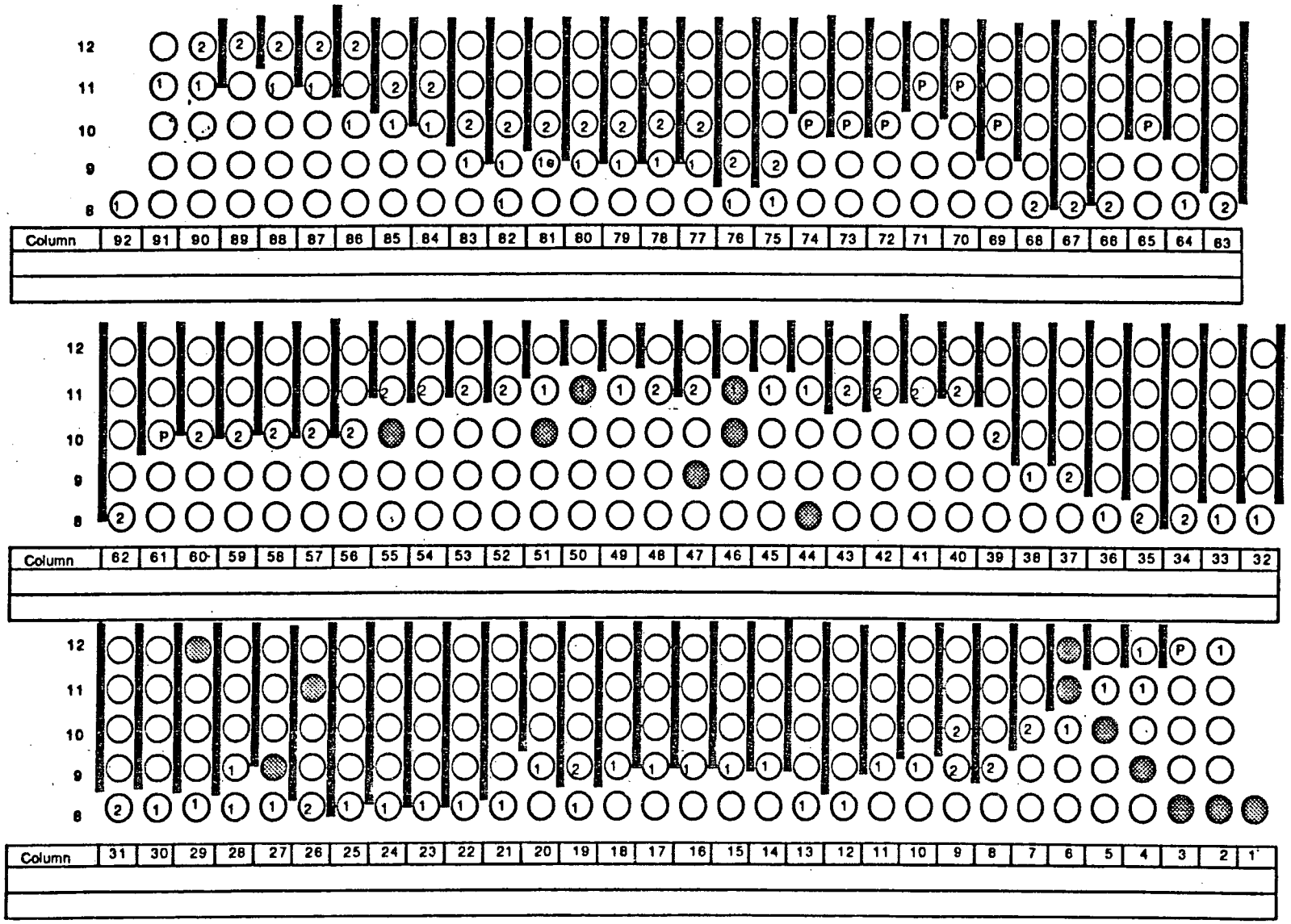
NOTES:
Numbers in tubes refer to AVBs visible
P - AVBs Located by Projection

Plugged Tube

11/11/87
hol

FIGURE 1

Indi[illegible]oint - 2 : S/G -24
AVB Insertion



NOTES:

Numbers in tubes refer to AVBs visible

P - AVBs Located by Projection

Indications in Columns 2 and 92 are retainer rings

Plugged Tube

11/10/87:a
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FIGURE 2

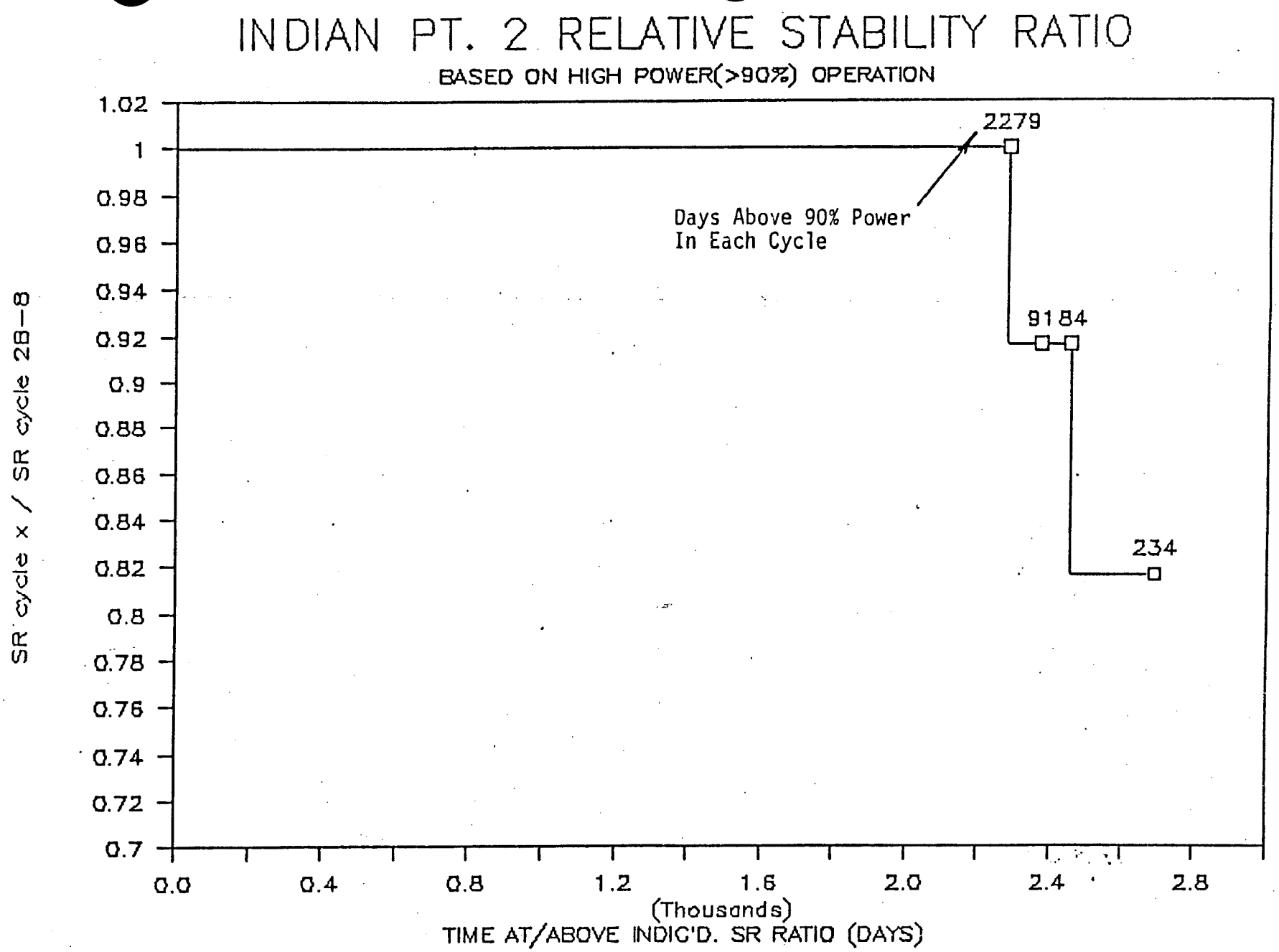

INDIAN PT. 2 RELATIVE STABILITY RATIO
BASED ON HIGH POWER(>90%) OPERATION
SR cycle x / SR cycle 2B-8
1.02
1
0.98
0.96
0.94
0.92
0.9
0.88
0.86
0.84
0.82
0.8
0.78
0.76
0.74
0.72
0.7
2279
Days Above 90% Power
In Each Cycle
9184
234
0.0
0.4
0.8
1.2
1.6
2.0
2.4
2.8
(Thousands)
TIME AT/ABOVE INDIC'D. SR RATIO (DAYS)

FIGURE 3

ATTACHMENT B

Primary to Secondary
Leak Detection Capability

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
November, 1987

1. LEAK DETECTION SYSTEMS:

The Indian Point Unit No. 2 (IP2) program for detection of primary system to secondary side leakage is based on Xenon-133 measured in the Condenser Air Ejector discharge. This method is preferred because Xenon is very volatile and does not have to be decay corrected to water mass turnover time. The Xenon-133 method is accurate to 0.001 gpm (1.5 gal/day). The monitoring of activity is either continuous as measured by R-15 (Condenser Air Ejector Radiation Monitor) or by grab sample and provides a rapid response to any leakage. In addition, the steam generator blowdown stream is monitored for radiation (R-19) and grab samples can be used for tracking Na-24, I-131 or I-133, relative to corresponding RCS activities to determine primary to secondary leakage. Both monitors (R-15 and R-19) are covered by existing Technical Specifications operability and surveillance requirements.

2. EXPERIENCE:

Historically, IP-2 has demonstrated a rapid response to the identification of steam generator leakage and subsequent expeditious removal of the unit from service. We recognize that any size leak can be a serious problem with respect to in-plant contamination and cleanup, or could develop into a more serious problem, and therefore action is taken before the technical specifications limit of 0.3 gpm is exceeded. As can be seen in the attached leakage trend graph and table of events for February 1984, the 0.3 gpm Technical Specification limit was never reached. It is noteworthy that the elapsed time from the time of identifying the increase in leakage to 0.01 gpm (0405 hrs) until the time the unit was removed from service was 11 hours. This clearly demonstrates IP2's leak detection capabilities and responsiveness to minimizing the extent of the excursion.

3. PROCEDURE ENHANCEMENT:

Following careful scrutiny of existing procedures, we are prepared to change our operating and chemistry procedures prior to startup from the current refueling and maintenance outage to enhance our procedures for the monitoring of tube leaks and the taking of appropriate preventive actions. Chemistry procedure IPC-A-110 and Operating procedure AOI 1.2 will be revised to include the following provisions:

A. With Condenser Air Ejector Monitor (R-15) Operable

- If leak rate is $>$ 0.1 gpm and $<$ 0.2 gpm, sample once per two hours and notify General Manager NPG and graphically plot leakage. If two consecutive rate calculations indicate that leakage will be equal to or exceed 0.2 gpm within 24 hours, the unit is to be removed from service using normal operating procedures.

- If leak rate is $\geq$ 0.2 gpm, remove unit from service and make appropriate notifications.

B. <u>With Condenser Air Ejector Monitor (R-15) Inoperable</u>

- With primary to secondary leakage $<$ 0.1 gpm, sample and calculate leak rate once every four hours.

- If leak rate is $>$ 0.1 gpm and, $<$ 0.2 gpm, sample once per two hours and notify General Manager NPG and graphically plot leakage. If two consecutive rate calculations indicate that leakage will be equal to or exceed 0.2 gpm within 24 hours, the unit is to be removed from service using normal operating procedures.

- If leak rate is $\geq$ 0.2 gpm, remove unit from service and make appropriate notifications.

Our review of plant leak detection capabilities and procedures indicates that to administratively impose a power reduction (e.g. 50% of full power) based upon a leak rate limit is not required and would, in fact, adversely impact our ability to calculate primary-to-secondary leak rates using Xenon-133. A reduction in power level changes the Xenon equilibrium condition in the primary side. Since leak rate limit calculations are optimized for Xenon equilibrium condition on the primary side, any changes caused by power reduction would affect our ability to perform these calculations rapidly. More importantly, by establishing the administrative limits above which would require a unit shutdown based on projected leak rates ensures that the fatigue crack will no longer propagate and the unit is shutdown in advance of a tube rupture.

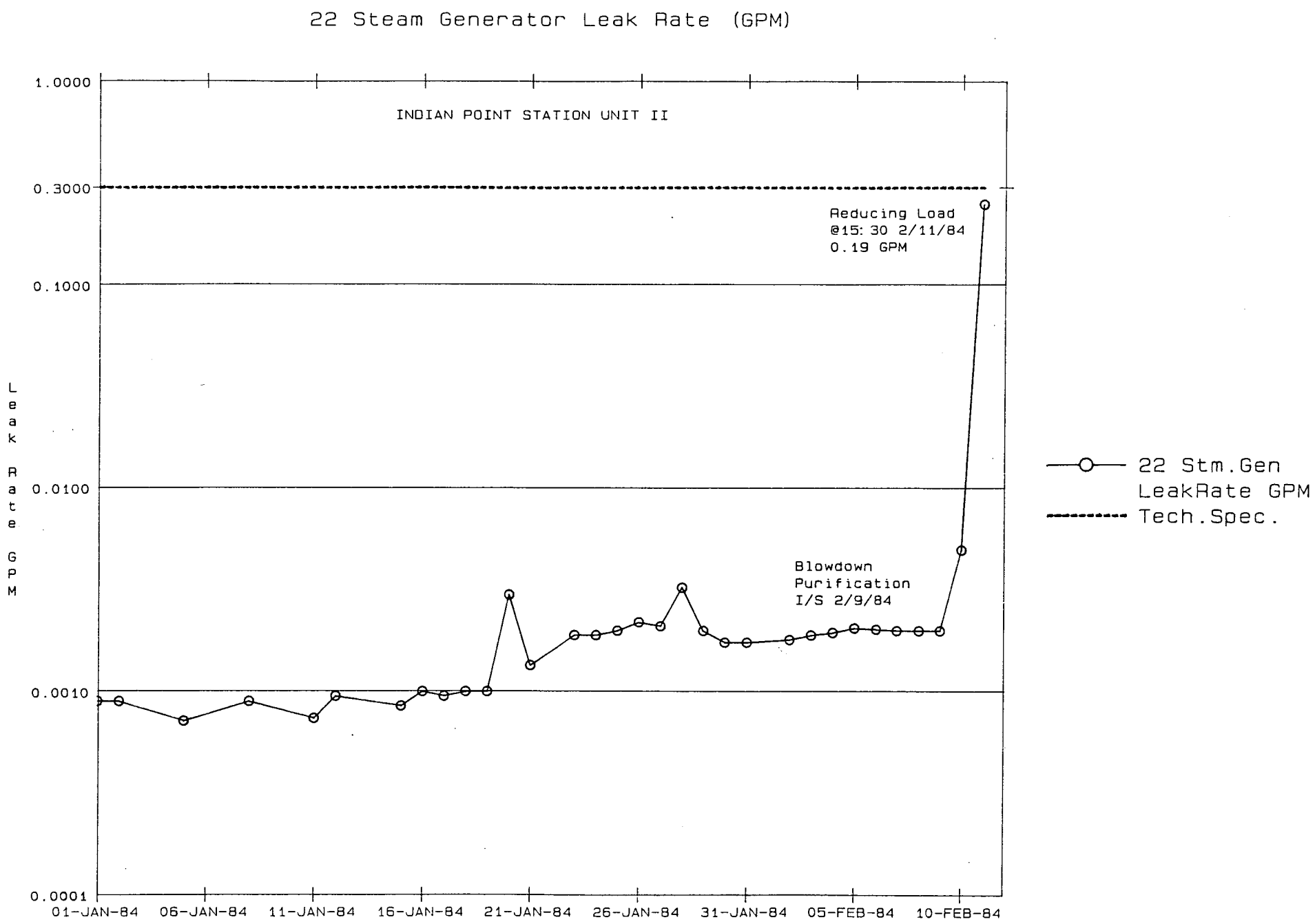

22 Steam Generator Leak Rate (GPM)
INDIAN POINT STATION UNIT II
1.0000
0.3000
0.1000
0.0100
0.0010
0.0001
Leak Rate GPM
Reducing Load
@15: 30 2/11/84
0.19 GPM
Blowdown
Purification
I/S 2/9/84
01-JAN-84
06-JAN-84
11-JAN-84
16-JAN-84
21-JAN-84
26-JAN-84
31-JAN-84
05-FEB-84
10-FEB-84
22 Stm.Gen
LeakRate GPM
Tech.Spec.

TABLE OF EVENTS

DATE	TIME	CAE CPM	CAE ICI/CC XE-133	LEAK RATE GPM
2/11/84	0015	100	2.0E-5	0.002
2/11/84	0405	2,000	2.0E-4	0.010
2/11/84	0714	5,500	1.1E-3	0.197
2/11/84	1042	5,000	1.1E-3	0.210
2/11/84	1300	REDUCING POWER		
2/11/84	1522	5,500	1.7E-3	0.250

ATTACHMENT C

Response to Questions
Requested by NRC Region I

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
November, 1987

INDIAN POINT UNIT NO. 2 (IP-2)
RESPONSE TO QUESTIONS REGARDING
STEAM GENERATORS AS A RESULT OF THE
NORTH ANNA TUBE RUPTURE INCIDENT

1. What is the Technical Specification primary to secondary steam generator leak rate limit?

Response:

Plant Technical Specification provision Section 3.1.F.2.a specifies that primary to secondary leakage through the steam generator tubes shall not exceed 0.3 gpm in any steam generator. Leakage in excess of 0.3 gpm for any steam generator will require plant shutdown within 24 hours and the leaking tube(s) to be located and plugged. Furthermore, the Technical Specifications require that if leakage from two or more steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours.

2. What are the administrative primary to secondary steam generator leak rate limits (and) the associated administrative action statements?

Response:

Plant administrative/procedural leak rate limits and associated correction actions reflect and implement the provisions of Technical

Specification Section 3.1.F.2.a described above. Although there are currently no administrative leakage limits lower than the 0.3 gpm of Technical Specification, we recognize that any size leaks can be a serious problem with respect to in-plant contamination and cleanup or it could develop into a more serious problem, and therefore corrective action is taken before the Technical Specifications limit of 0.3 gpm is exceeded. For example, on February 11, 1984 the Unit was taken out of service before a primary to secondary leak rate of 0.25 gpm was reached. Similarly, on August 21, 1981 a leak in Steam Generator No. 23 increased from 0.1 gpm to 0.205 gpm and again the Unit was repaired and returned to service.

As a result of our review of existing leak detection systems, procedures and practices, we are planning to revise plant operating and chemistry procedures to establish primary-to-secondary leak rate limits that will allow for positive detection and timely corrective action e.g. notification, trending and unit shutdown prior to tube failure. These procedures will be revised before startup from the current refueling and maintenance outage.

3. What leak rate detection system(s) are used and the frequency of use, and what is the sensitivity of these systems?

Response:

At IP2 the primary method to determine primary to secondary leak rates is the isotopic analysis of Xenon-133 in the Condenser Air Ejector (CAE) offtake compared to the reactor coolant, Xenon-133 activity. This method is preferred because Xenon is very volatile and does not have to be decay corrected to water mass turnover time. The

Xenon-133 method is accurate to 0.001 gpm (1.5 gal/day). With the CAE radiation monitor in service and a known primary to secondary leak, leak rates are calculated daily. With the CAE radiation monitor not in service, leak rates are calculated every four hours and could, if necessary, be performed hourly. Backup methods are available to identify the primary to secondary leak rate. These include:

Sodium - 24 in S/G blowdown

Iodine - 131 in S/G blowdown

Boron in S/G blowdown

Tritium in S/G blowdown

These backup methods will detect leakage greater than 0.1 gpm (144 gal/day) but generally take longer to perform than the primary method.

4. Please furnish a map of each steam generator bundle cross section (hot and cold legs) at the top support plate elevation. The map should indicate:

   A. Which tubes are dented
   B. Dent magnitudes (or range of dent magnitudes in various areas of the support plate) based on the most recent data.
   C. Which tubes contact antivibration bars.
   D. Which tubes are plugged.

Response:

Response to Question 4A & 4B: Attachment 1 contains maps of the steam generator bundle cross section at the top support plate elevation, showing for each tube examined, the eddy current measurement of the dents. Note that maps titled "Support Plate 6" are maps of the uppermost support plate hot leg; maps titled "Support Plate 7" are

maps of the same uppermost support plate cold leg. Locations marked "x" on the maps titled "Probe Size 700" are locations that did not pass the 700 mil diameter probe and were subsequently examined by the 610 mil diameter probe. The latter results are indicated on the maps titled "Probe Size 610". Locations marked "B" are plugged tubes.

Response to Question 4C: Attachment 2 shows for Steam Generator Nos. 22 and 24 the lowest row in each column in which the anti-vibration bar has been identified by review of previous (1984 and 1986) eddy current tapes. Tubes in these locations plus the tubes in the higher number rows contact anti-vibration bars. For Steam Generator Nos. 21 and 23, the current eddy testing program planned for the outage has been expanded to determine the AVB location with respect to the tubes in Steam Generator Nos. 21 and 23.

Response to Question 4D: Attachment 3 contains maps of plugged tubes in each steam generator.

5. What are the average and maximum rates of denting in the top tube support plate during the last two inspection cycles for each steam generator?

Response:

The average dent (in mils) based on eddy current data for the last two examinations for all four generators is:

S.G.	1984 Average Dent		1986 Average Dent	
	Hot Leg	Cold Leg	Hot Leg	Cold Leg
21	4.3	3.6	3.5	3.6
22	1.5	1.6	1.7	2.1
23	3.7	3.1	2.8	2.9
24	3.4	4.0	3.5	3.7

It is Con Edison's opinion that differences between 1984 and 1986 data appear to be within error of measurement.

6. When was the last inspection of each steam generator? Identify the tube/upper tube support plate intersections that were inspected in the last inspection. What inspection techniques were used i.e. ECT, profilometry or other?

Response:

The last inspection of each steam generator occurred during the 1986 refueling outage using Eddy Current Testing (ECT). Attachment 1 identifies the tube/upper tube support plate intersections that were inspected in the last inspection.

7. Have any ECT indications been observed at the antivibration bars or top support plate locations? If so, which tubes, when, and what was the nature of the indication?

Response:

Based on 1984 and 1986 Eddy current data no indications were observed at the anti-vibration bars or at top support plate locations.

8. What is the cross sectional shape of the antivibration bars (round, square or other)?

Response:

The cross sectional shape of the antivibration bars is square.

9. What is the current steam generator ratio at normal full power operation?

Response:

The circulation ratio at the current operating conditions is 4.4. This corresponds to a steam flow rate of 2,926,000 lb/hr, steam pressure of 690 psia and the downcomer flow resistance plate at the upper, withdrawn position (least flow resistance).

10. What is the current range of recirculation ratios for the range of power operations above 20%?

Response:

As operating load decreases from full load, the steam flow rate decreases. Since the total (steam and water) flow rate through the bundle does not vary significantly between 50% and 100% load, the

circulation ratio will increase proportionately. However, the loading on tubes in the U-bend, ρV^2, will decrease with power. At very low power levels the load on tubes will be even smaller than that at 50% power. The circulation ratio at 20% load would be about four times greater than that at full load.

11. What is the history of recirculation ratios under operating conditions?

Response:

Based on the historical operating data, we estimate that the full load circulation ratio has not varied much over the life of the plant. It has remained in the range of 4.3 to 4.4.

12. Describe any modifications that have been made to the steam generators e.g. feedwater flow restrictor plates, moisture separators, etc. or changes in operations that could have affected the steam generator recirculation ratio. When were these changes made?

Response:

Changes to the steam generator operation that could have affected the recirculation ratio are changes in the primary Tavg. The history of Tavg reduction at Indian Point 2 from a design Tavg of 569.5°F is as follows:

Dates			Tavg °F
5/22/73 (Initial Criticality)	to	6/25/75	566-569.5
6/25/75	to	2/28/77	554
2/28/77	to	Present	549

13. What secondary side loose parts monitoring system is used? Describe the system and its operation.

Response:

During a refueling/maintenance outage secondary side examination is performed. A video camera or a borescope is passed around the annulus between the tube bundle and the shell and down the tube lane between the hot and cold legs to search for foreign objects in all four steam generator. During plant operation a Metal Impact Monitoring System (MIMS) provides detection of loose metallic parts impacting within the primary and secondary side of the steam generators. The MIMS provides continuous display of the information to plant operation. A detailed description of this system and its operation is provided in Attachment 4.

14. What tubes are not supported by Antivibration Bars?

Response:

Refer to response to Item 4C and Attachment 1 to this letter.

15. Please provide any other information you feel is important to this issue.

Response:

The present plant Emergency Operating Procedures (EOPs) were developed using Westinghouse Owners Group Emergency Response Guidelines (ERGs) which address steam generator tube failures including multiple tube failures. Plant operators are extensively trained on the EOPs.

Elements of our operator training program on EOPs and in particular tube rupture events include:

1. Generic Westinghouse ERGs training program, constituting four separate seminars.

2. Generic Westinghouse Steam Generator Tube Rupture (SGTR) training program.

3. Indian Point 2 specific training in new EOPs (6 weeks per operator) with special emphasis on E-3 series procedures (SGTR).

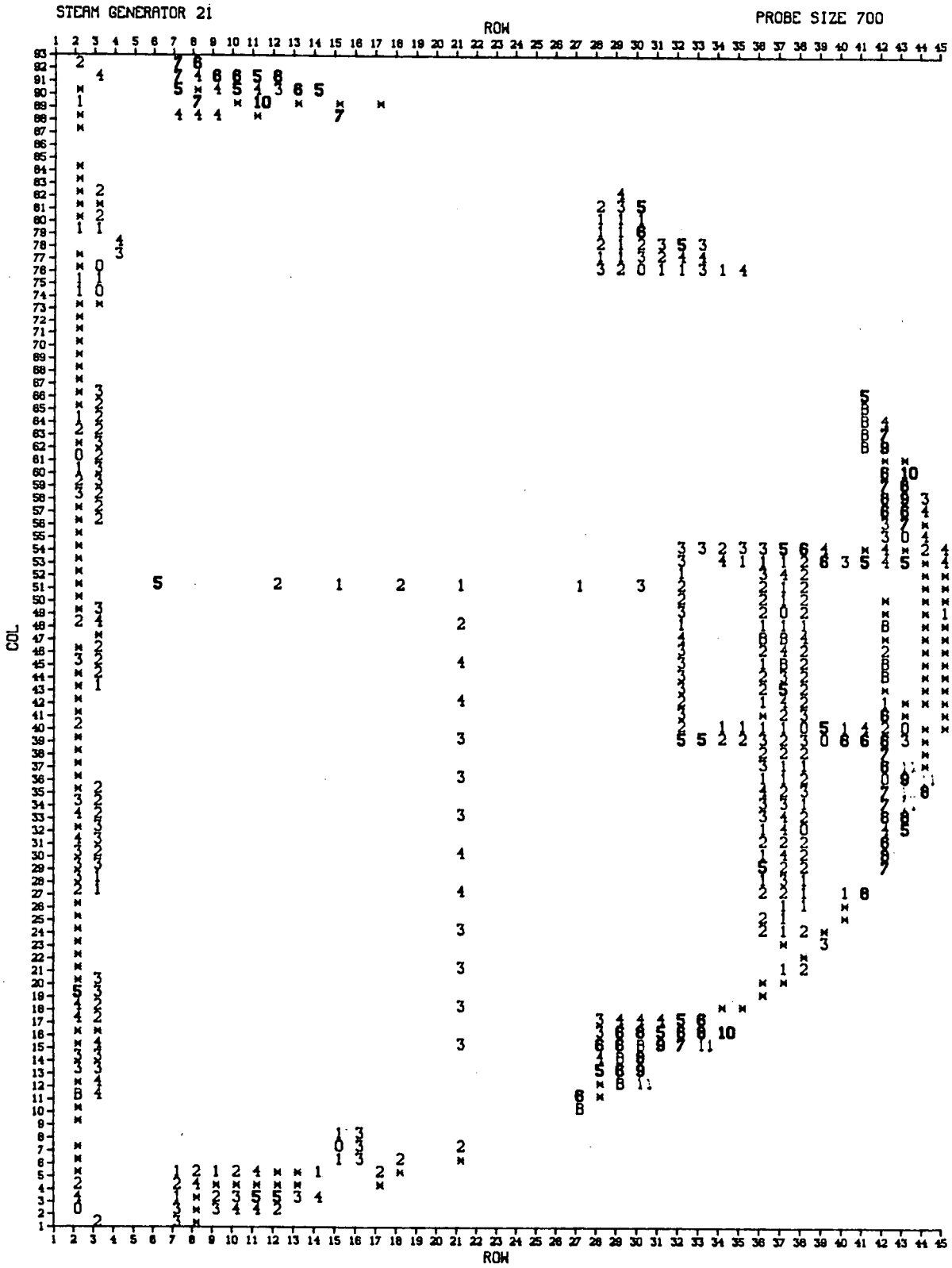
4. Requalification training occurring every two years. SGTR is a required scenario.

These EOPs along with our integrated training practices assures that the plant operators are prepared to safely handle a major event such as a tube rupture.

ATTACHMENT 1

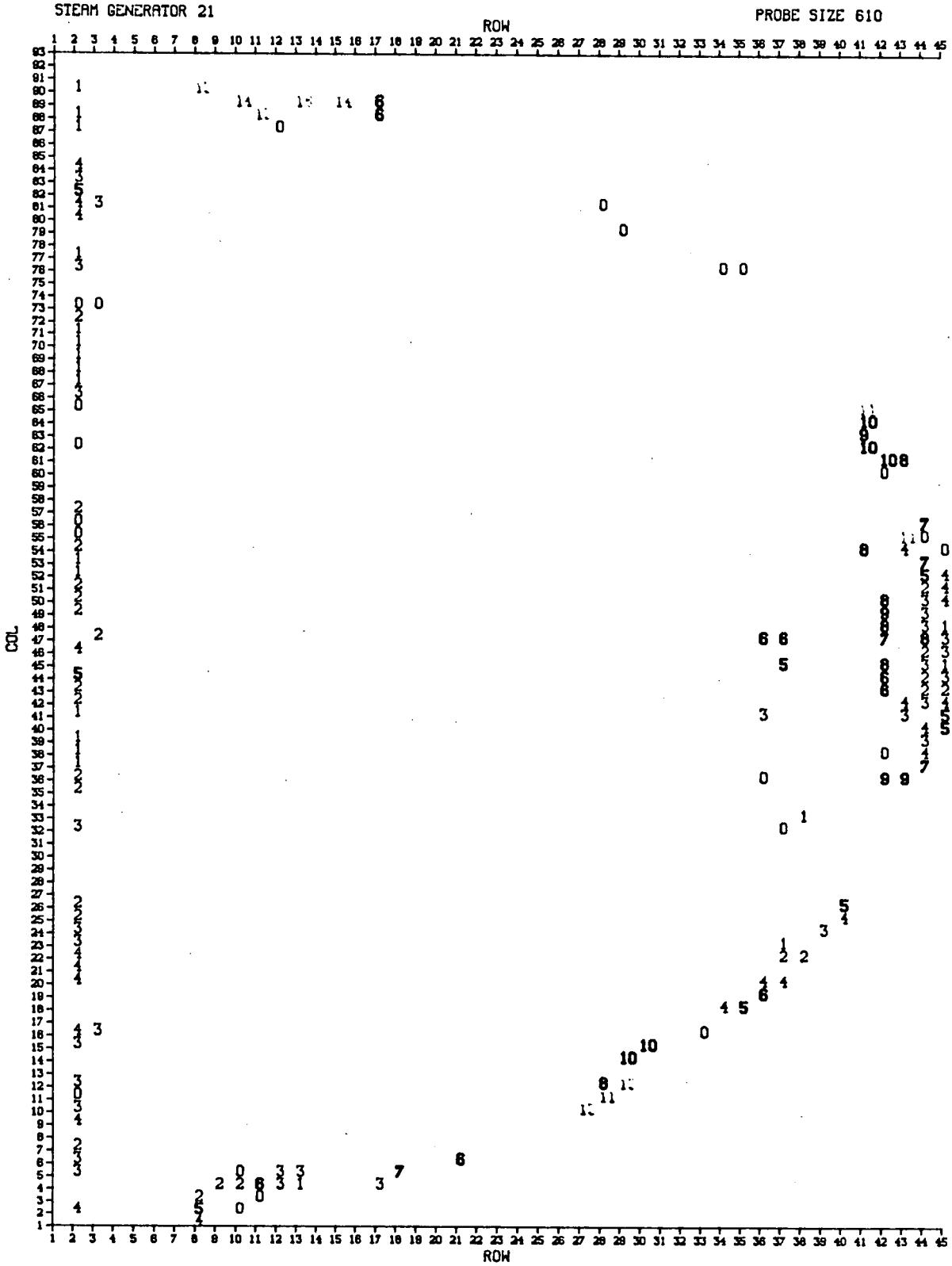
INSPECTION DATE 860221 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 21 PROBE SIZE 700



INSPECTION DATE 860221 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 21 PROBE SIZE 610



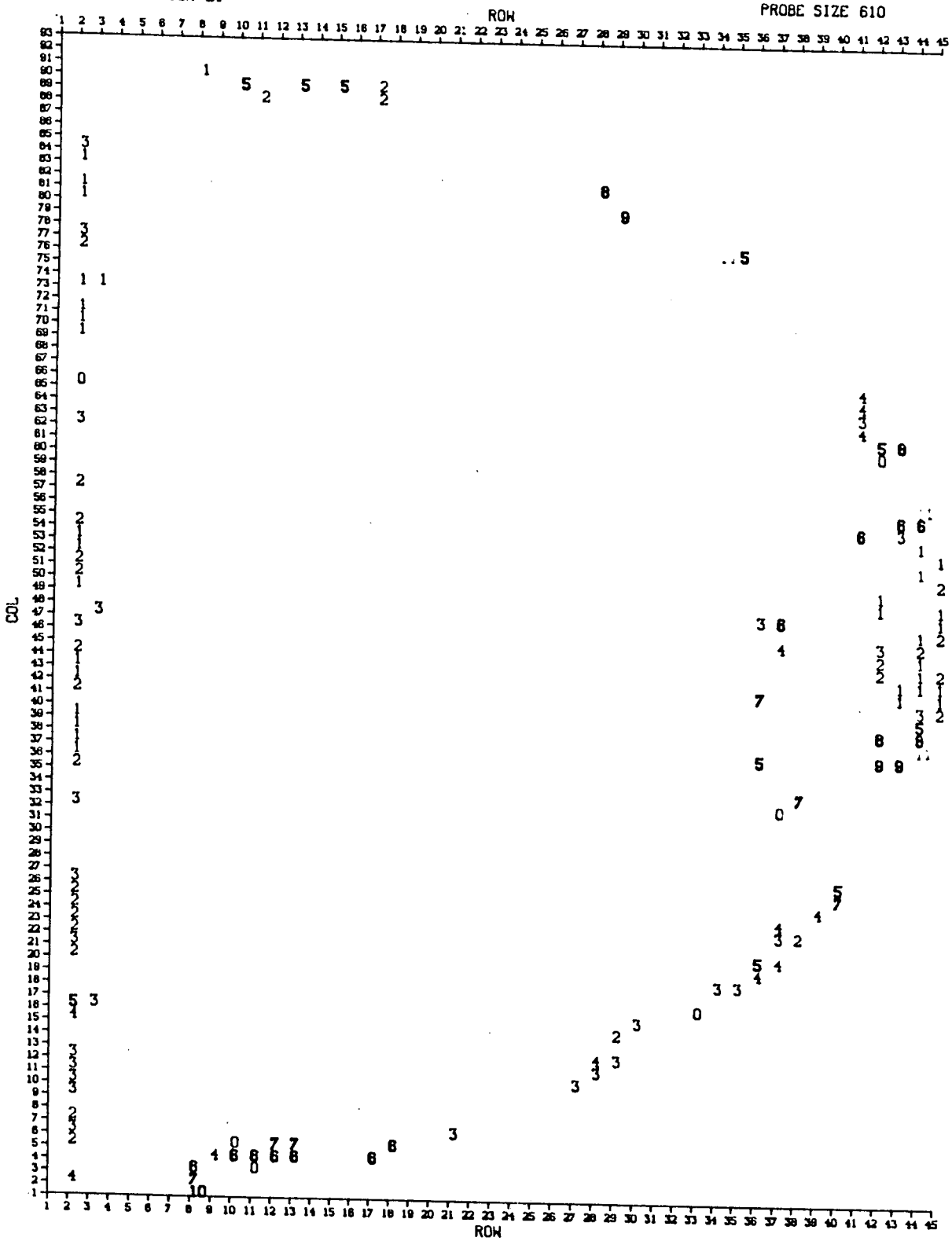
INSPECTION DATE 860221

STEAM GENERATOR 21

COLD LEG
DENTING LAYOUT

SUPPORT PLATE 7

PROBE SIZE 610



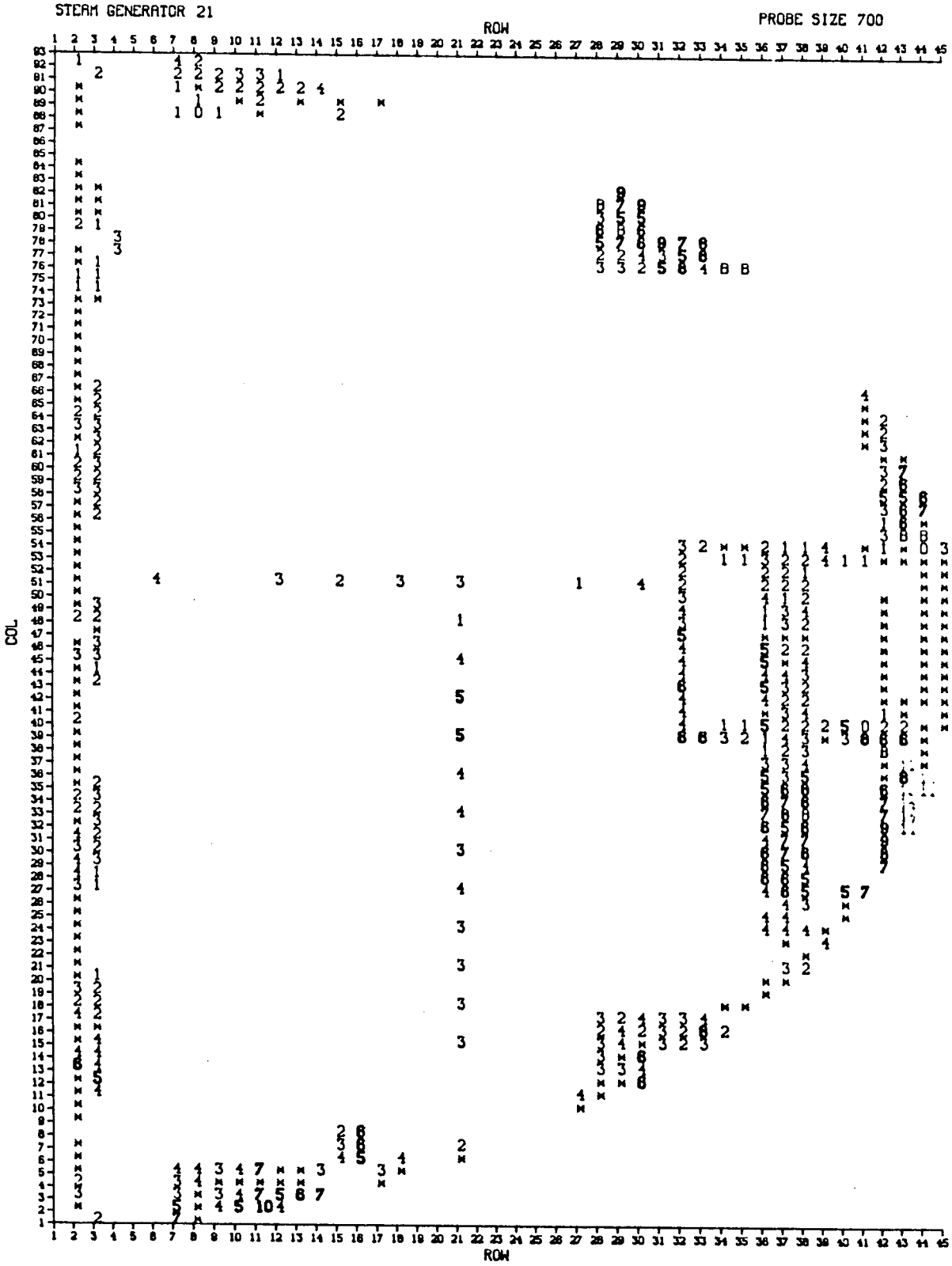
INSPECTION DATE 860221

STEAM GENERATOR 21

COLD LEG
DENTING LAYOUT

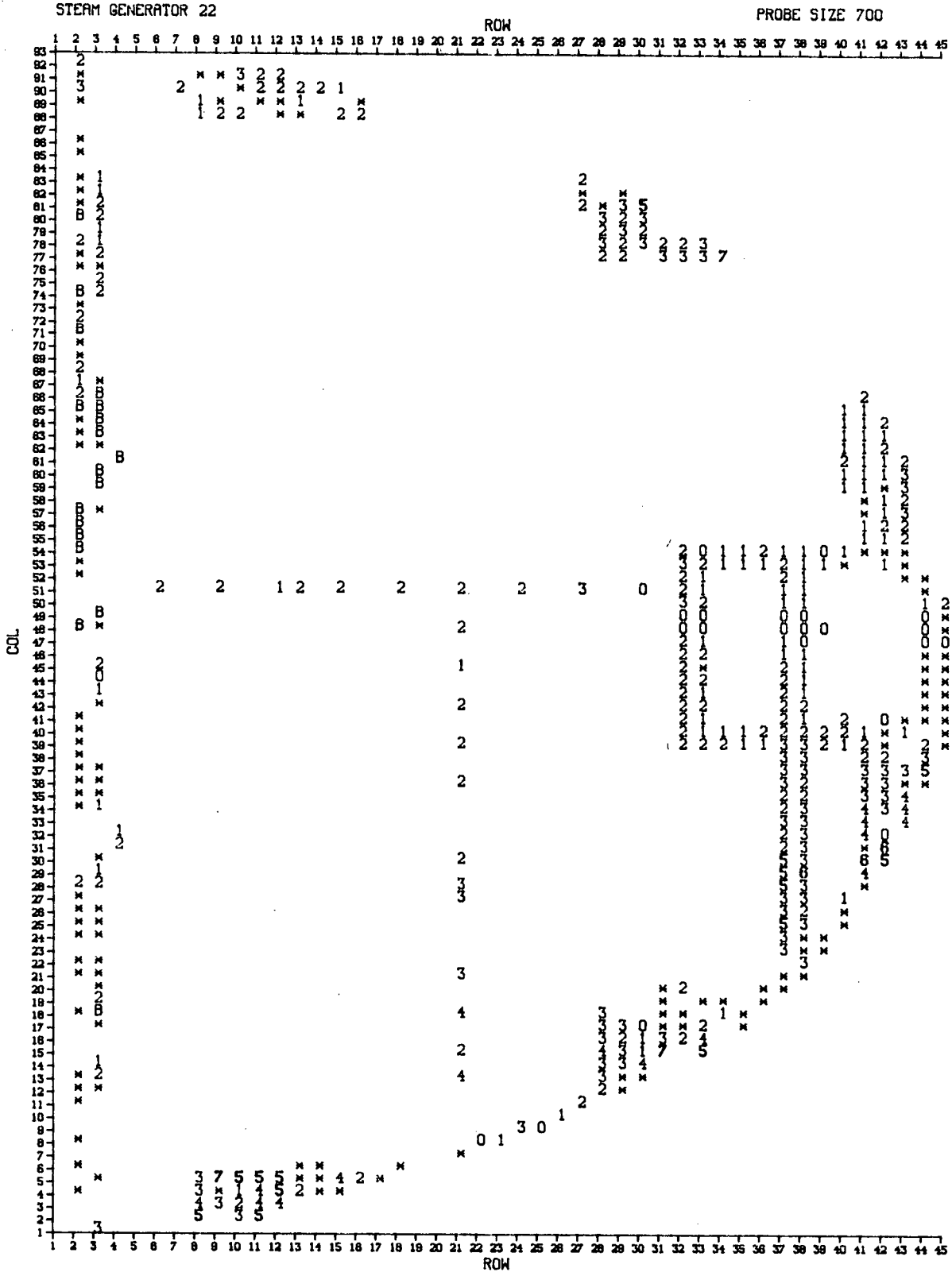
SUPPORT PLATE 7

PROBE SIZE 700



INSPECTION DATE 860222 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 22 PROBE SIZE 700



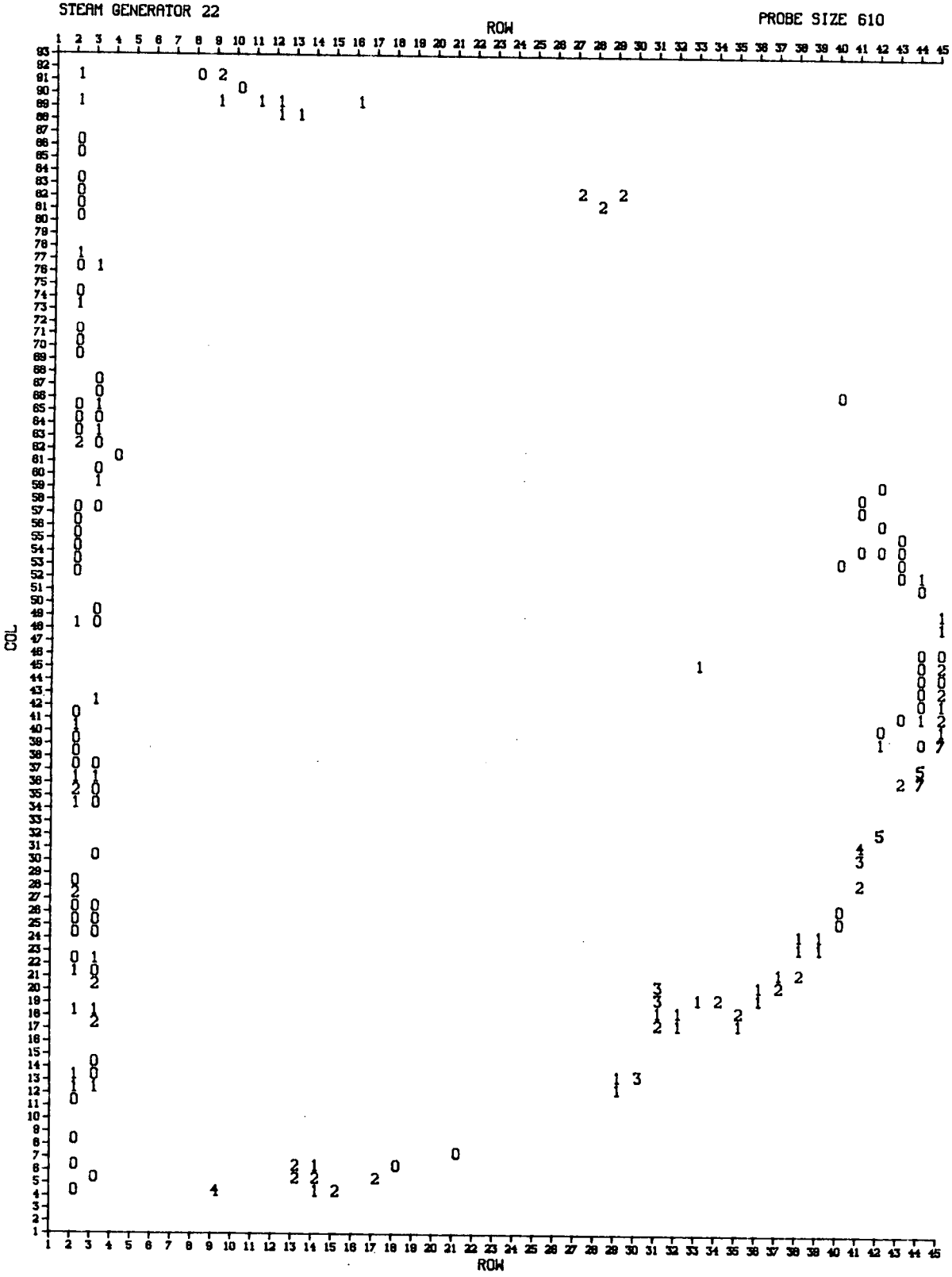
INSPECTION DATE 860222

HOT LEG DENTING LAYOUT

SUPPORT PLATE 6

STEAM GENERATOR 22

PROBE SIZE 610



COLD LEG

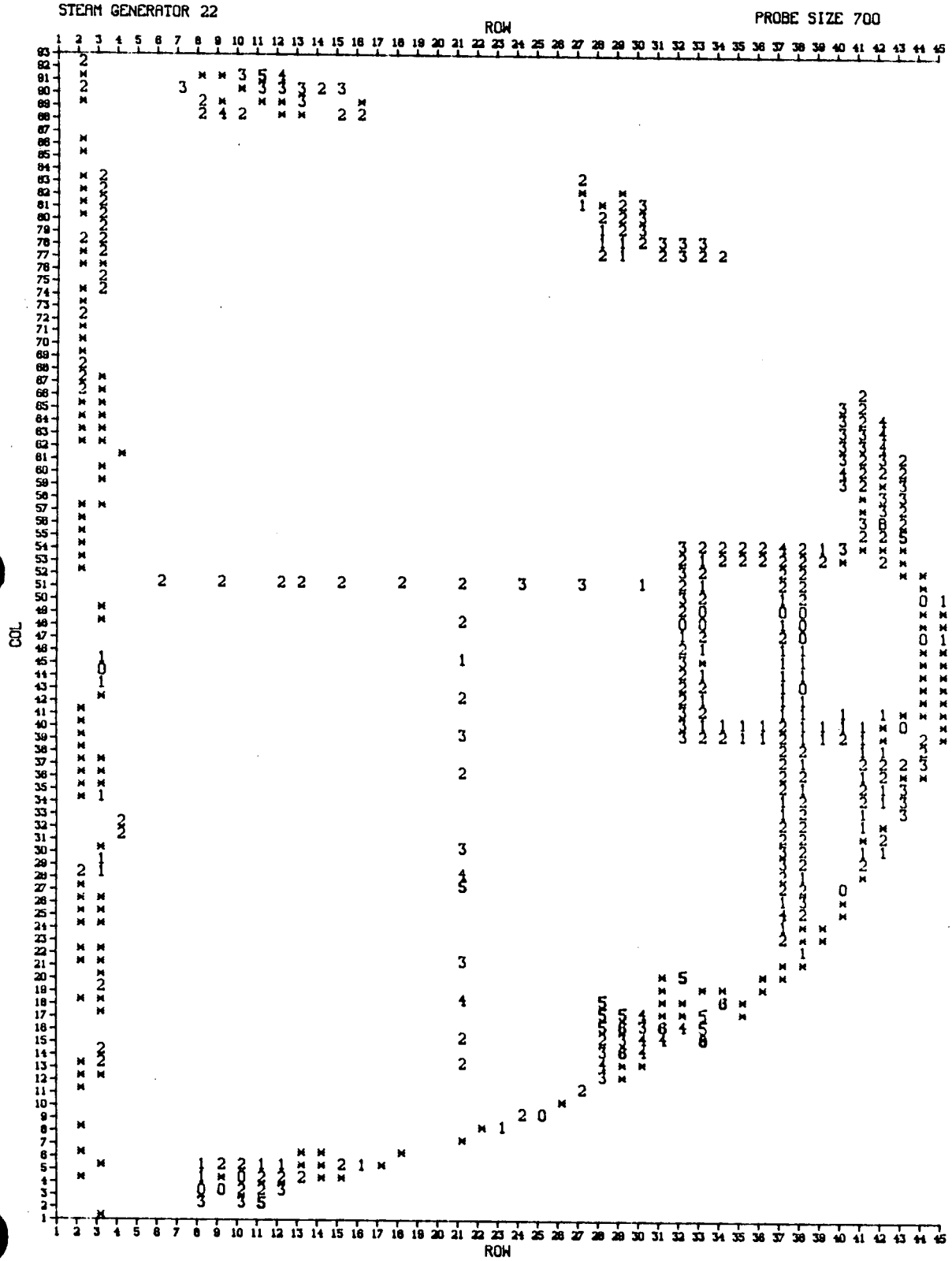
DENTING LAYOUT

INSPECTION DATE 860222

STEAM GENERATOR 22

SUPPORT PLATE 7

PROBE SIZE 700



COLD LEG

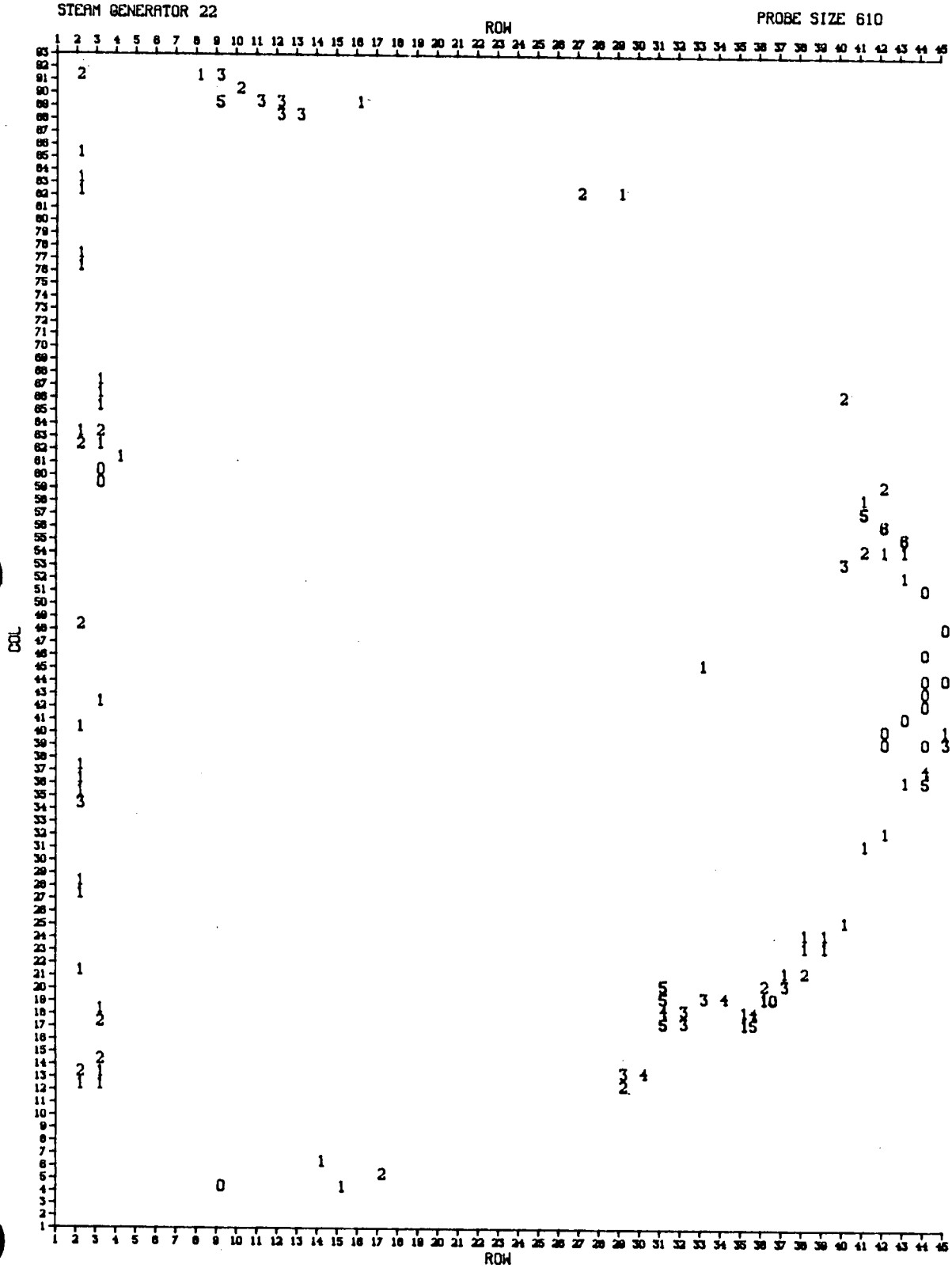
DENTING LAYOUT

...INSPECTION DATE 860222

STEAM GENERATOR 22

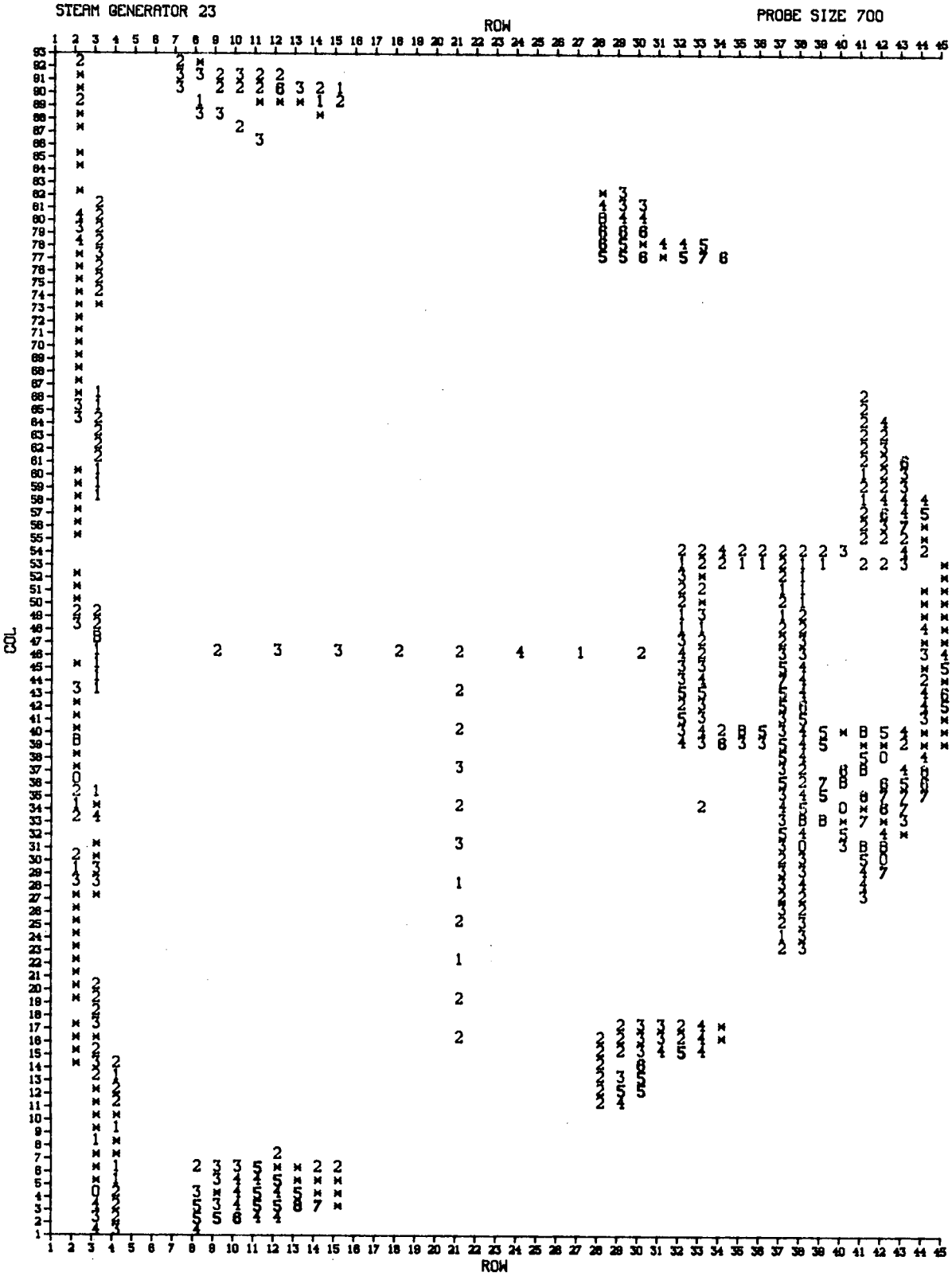
SUPPORT PLATE 7

PROBE SIZE 610



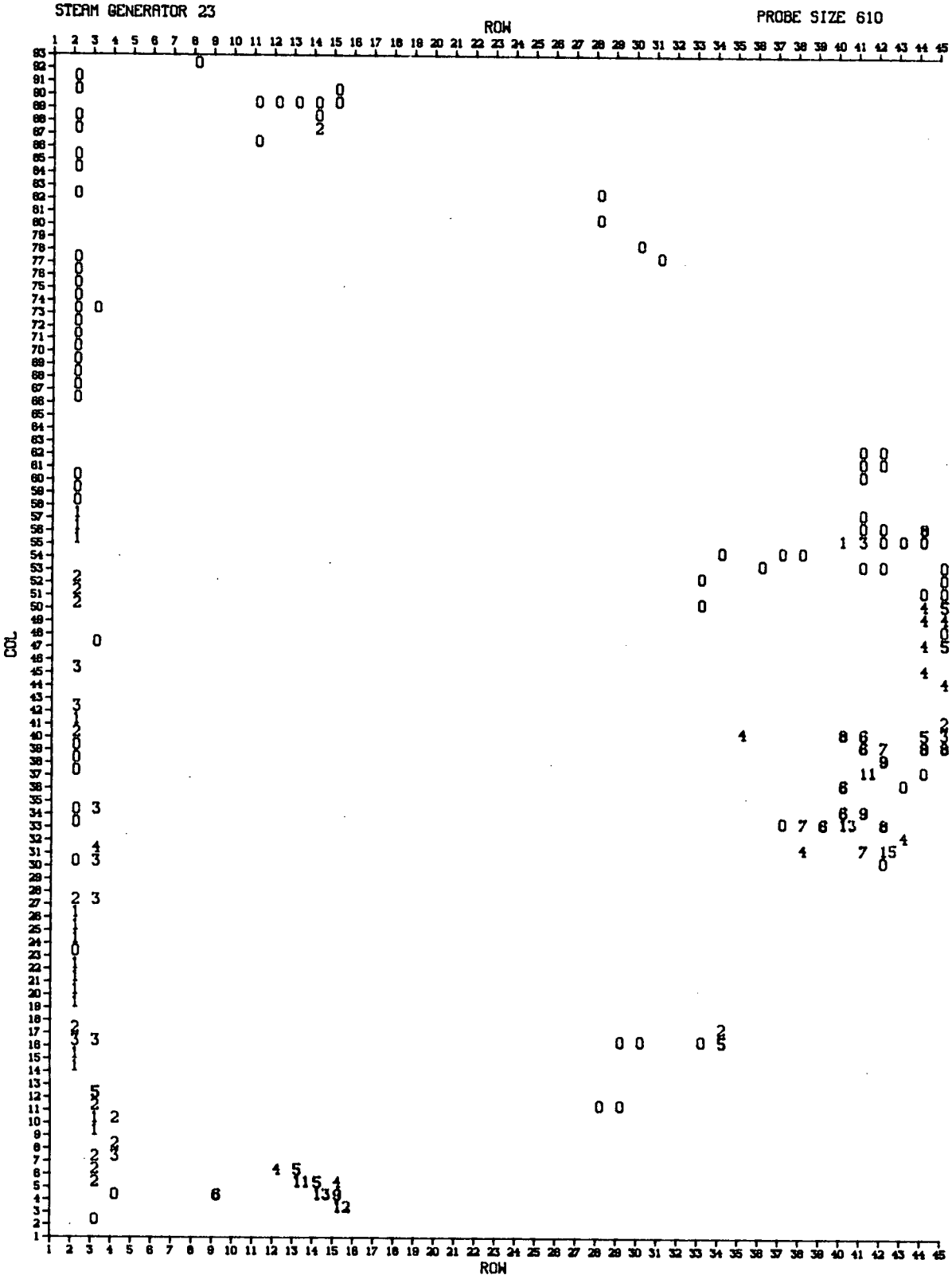
INSPECTION DATE 860223 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 23 PROBE SIZE 700



INSPECTION DATE 860223 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 23 PROBE SIZE 610



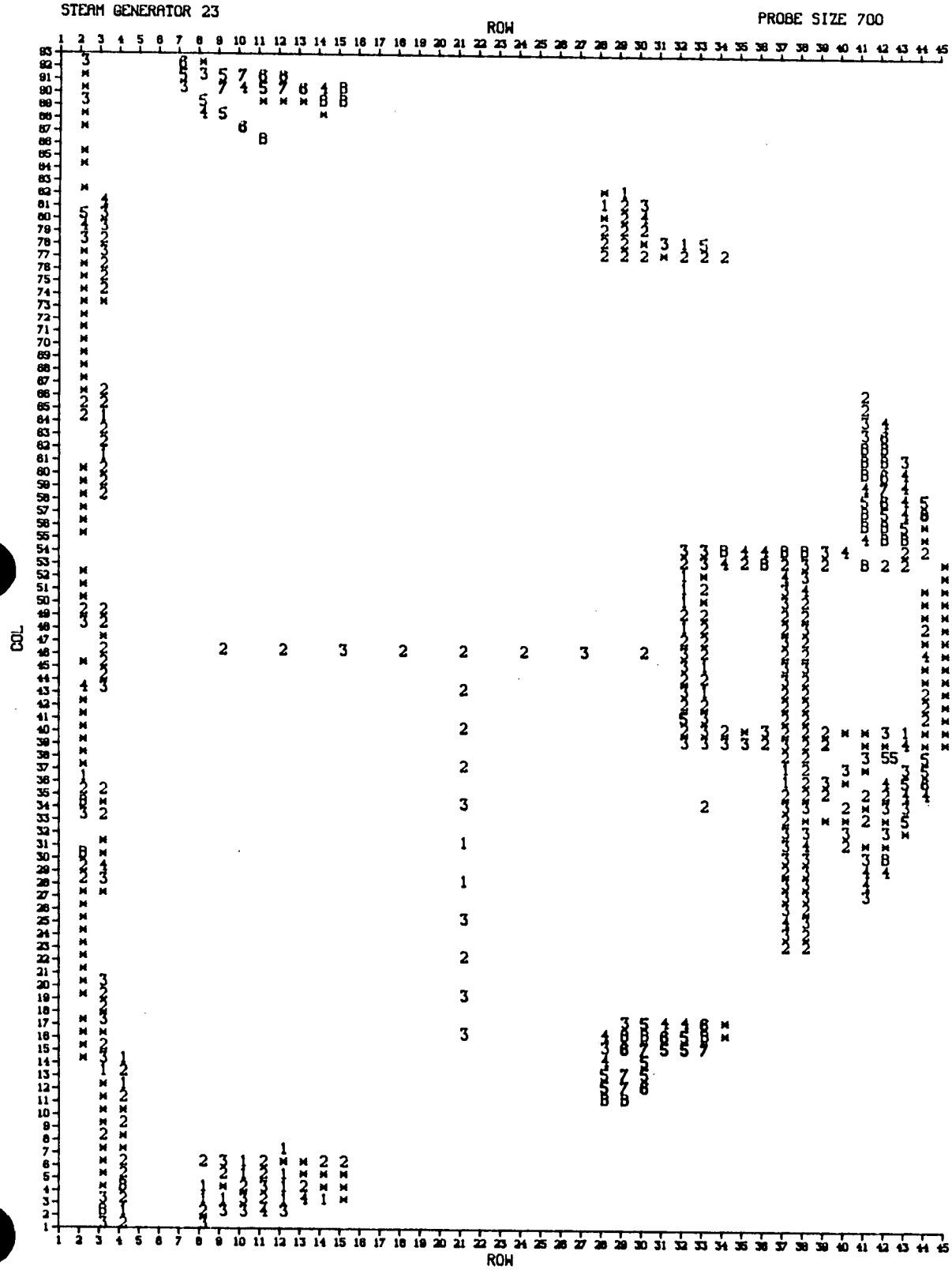
INSPECTION DATE 860223

STEAM GENERATOR 23

COLD LEG
DENTING LAYOUT

SUPPORT PLATE 7

PROBE SIZE 700



COLD LEG

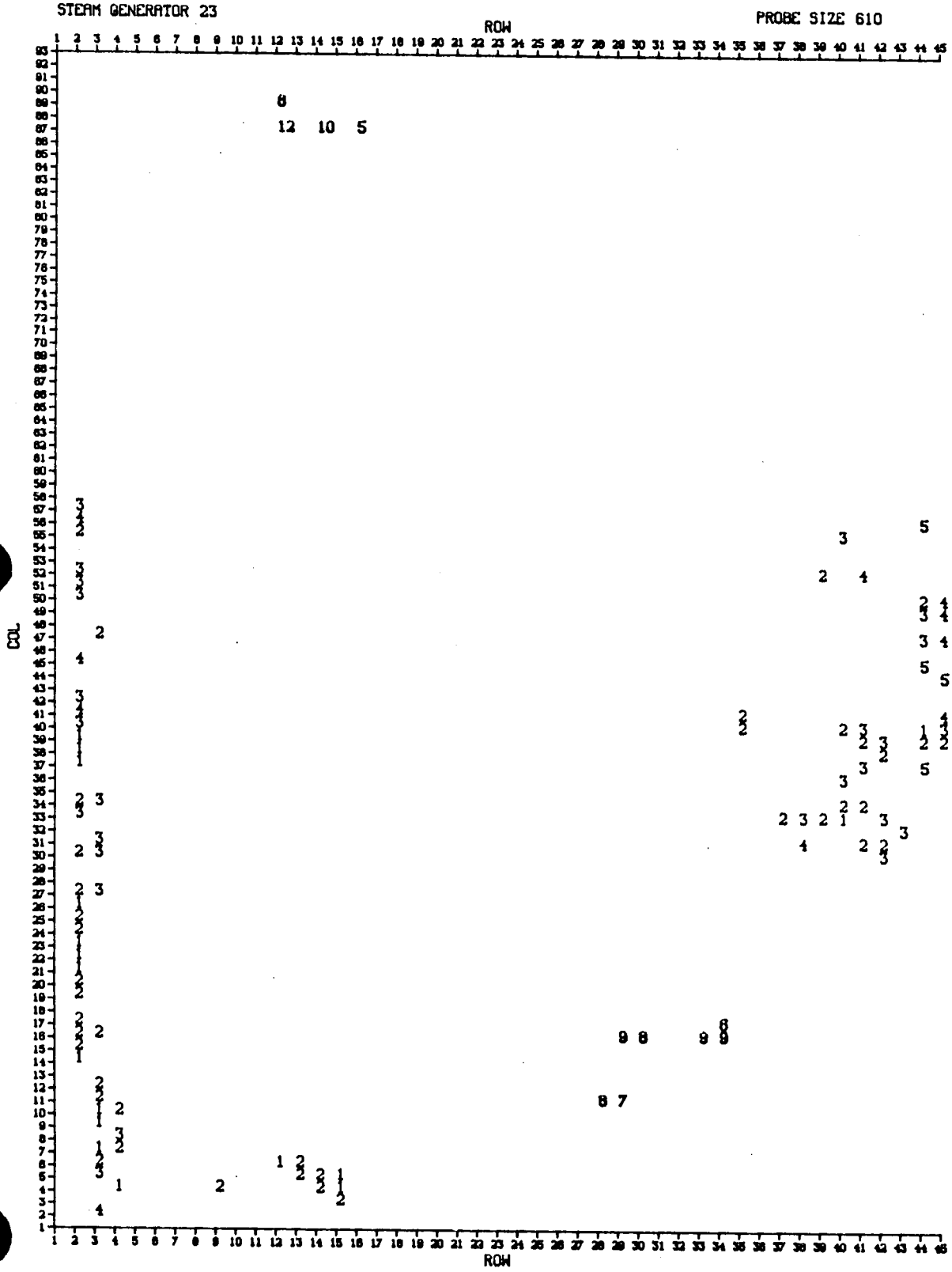
INSPECTION DATE 860223

DENTING LAYOUT

SUPPORT PLATE 7

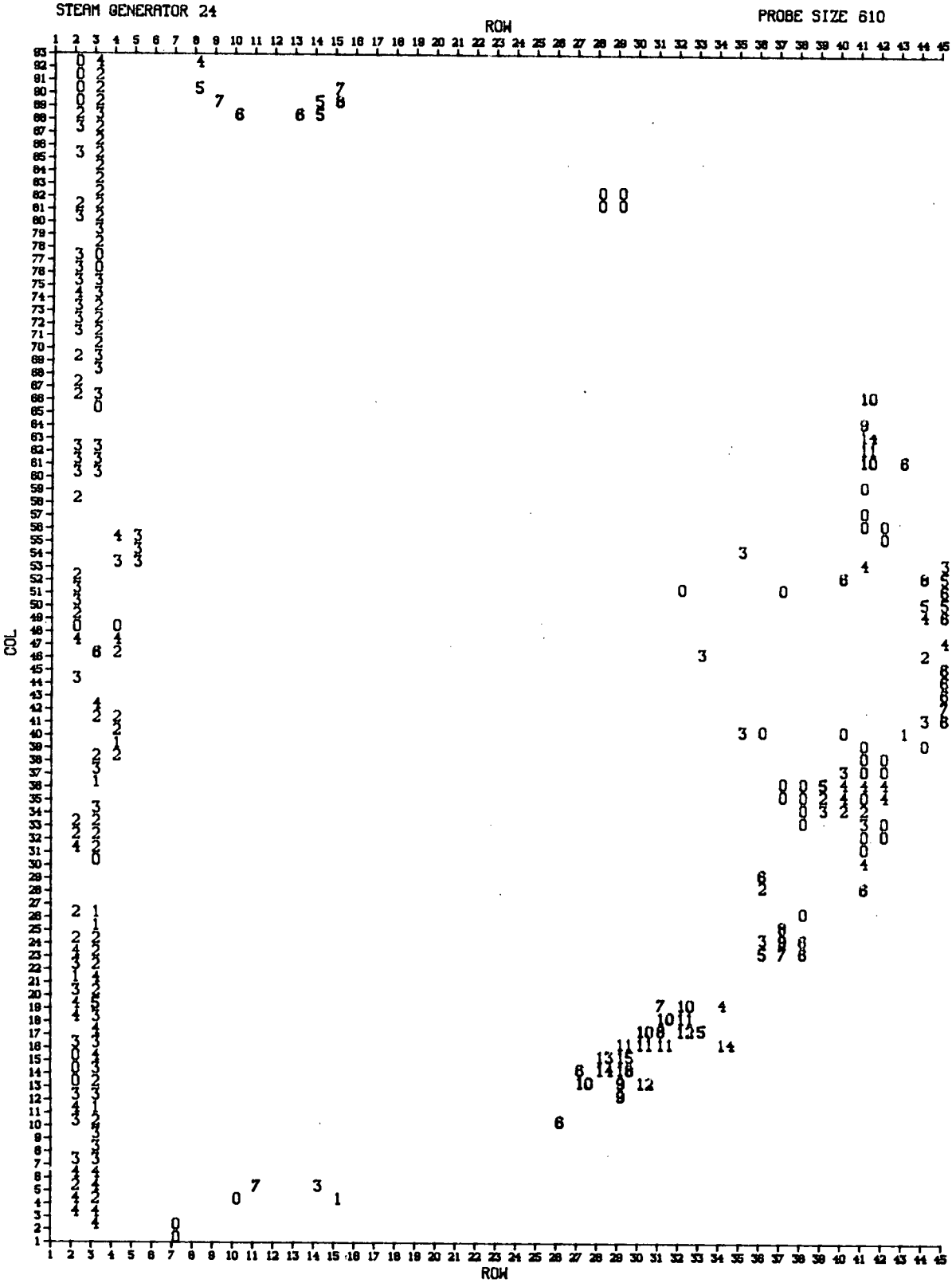
STEAM GENERATOR 23

PROBE SIZE 610



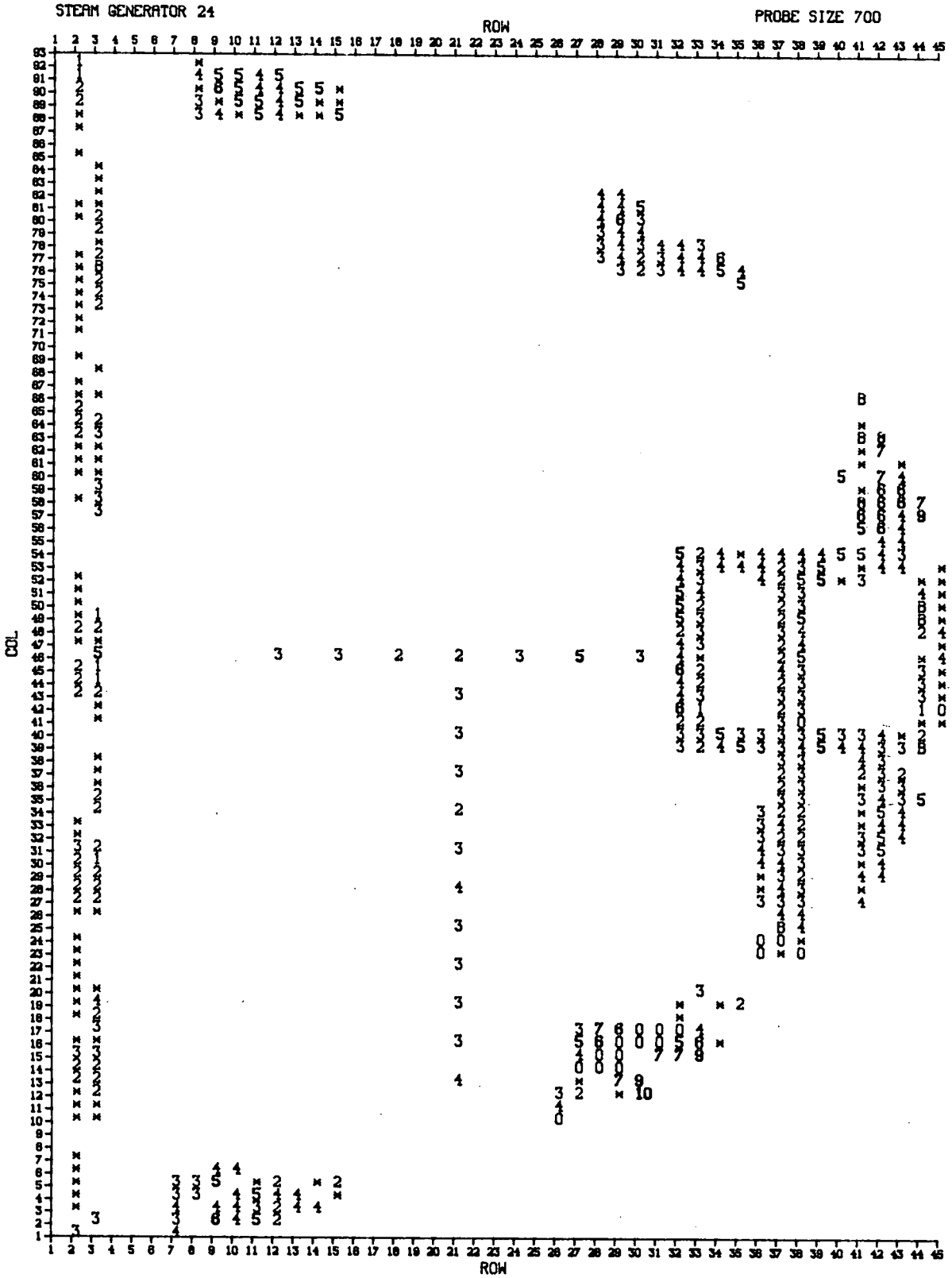
INSPECTION DATE 860224 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 24 PROBE SIZE 610



INSPECTION DATE 860224 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 24 PROBE SIZE 700



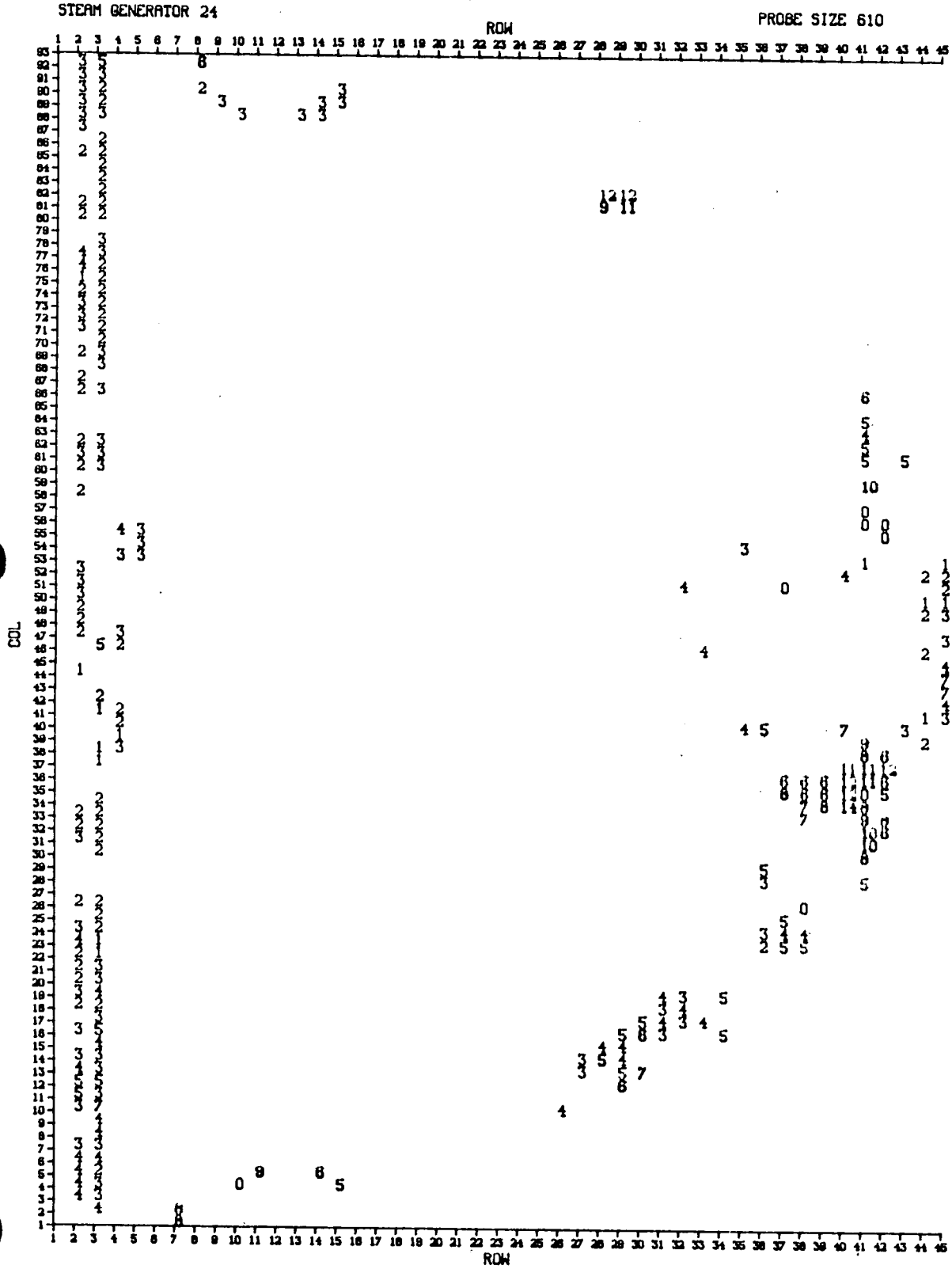
INSPECTION DATE 860224

STEAM GENERATOR 24

COLD LEG
DENTING LAYOUT

SUPPORT PLATE 7

PROBE SIZE 610



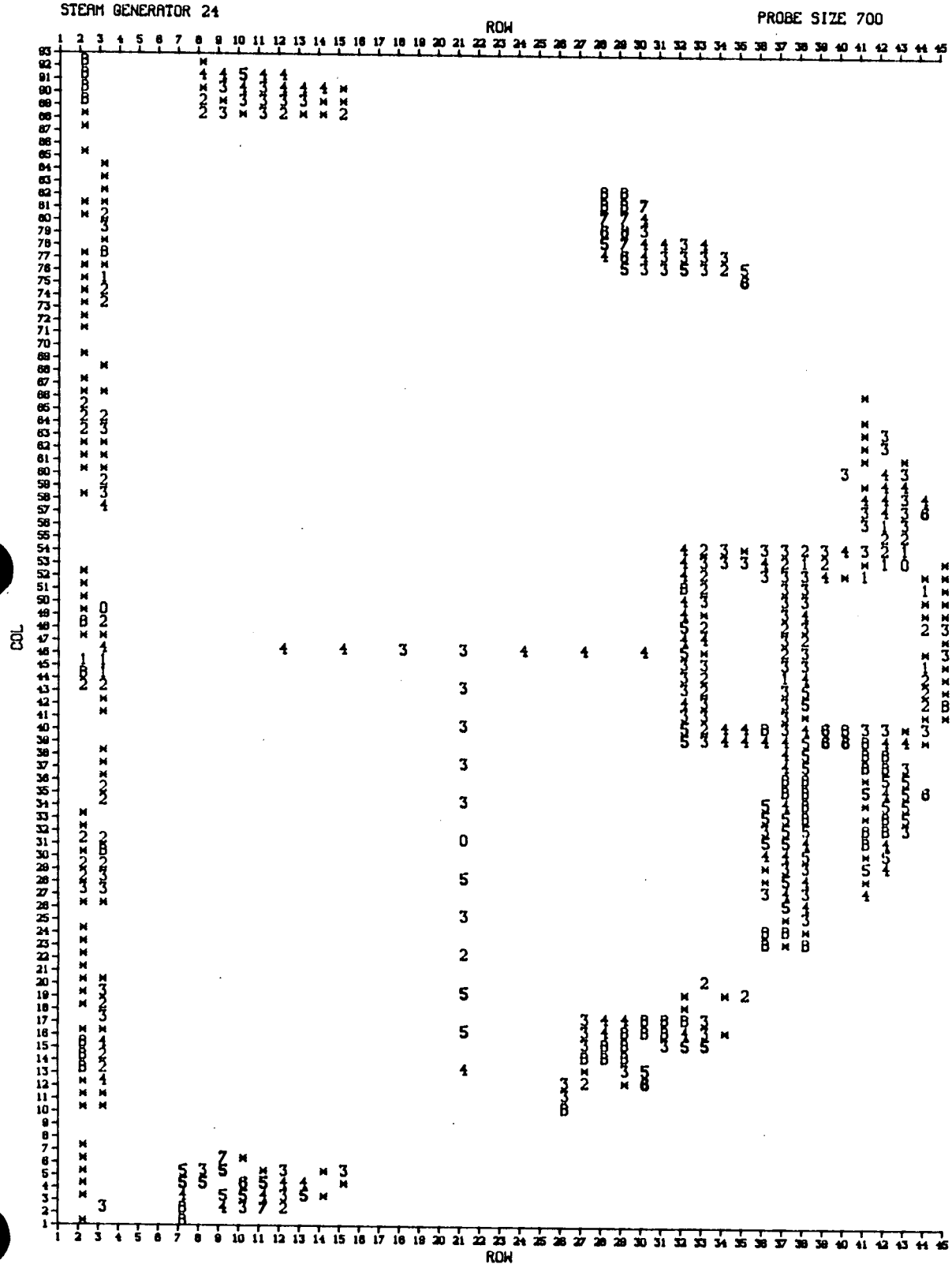
...INSPECTION DATE 860224

COLD LEG
DENTING LAYOUT

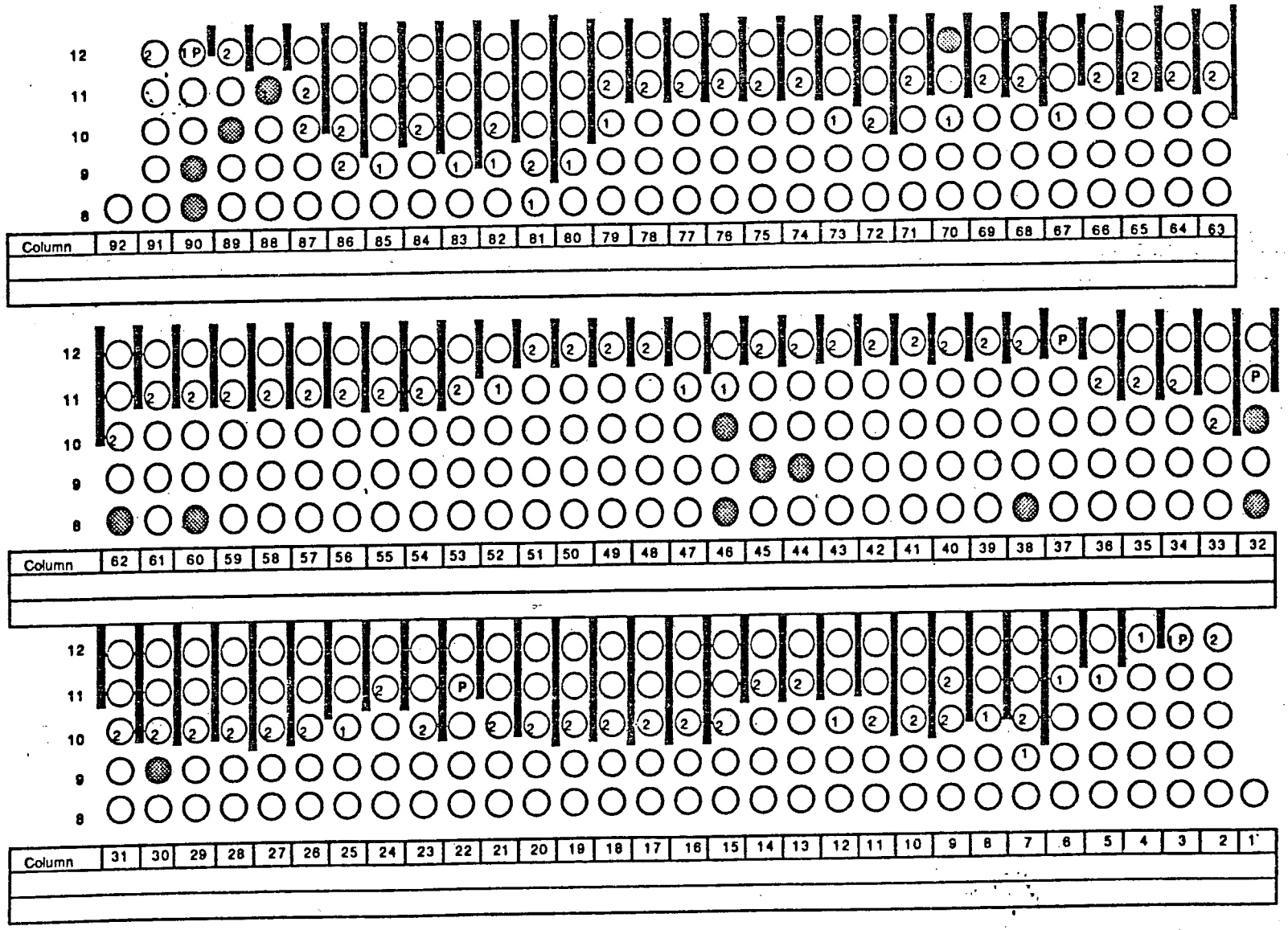
SUPPORT PLATE 7

STEAM GENERATOR 24

PROBE SIZE 700



Indian Point - 2 : S/G - 22
AVB Insertion

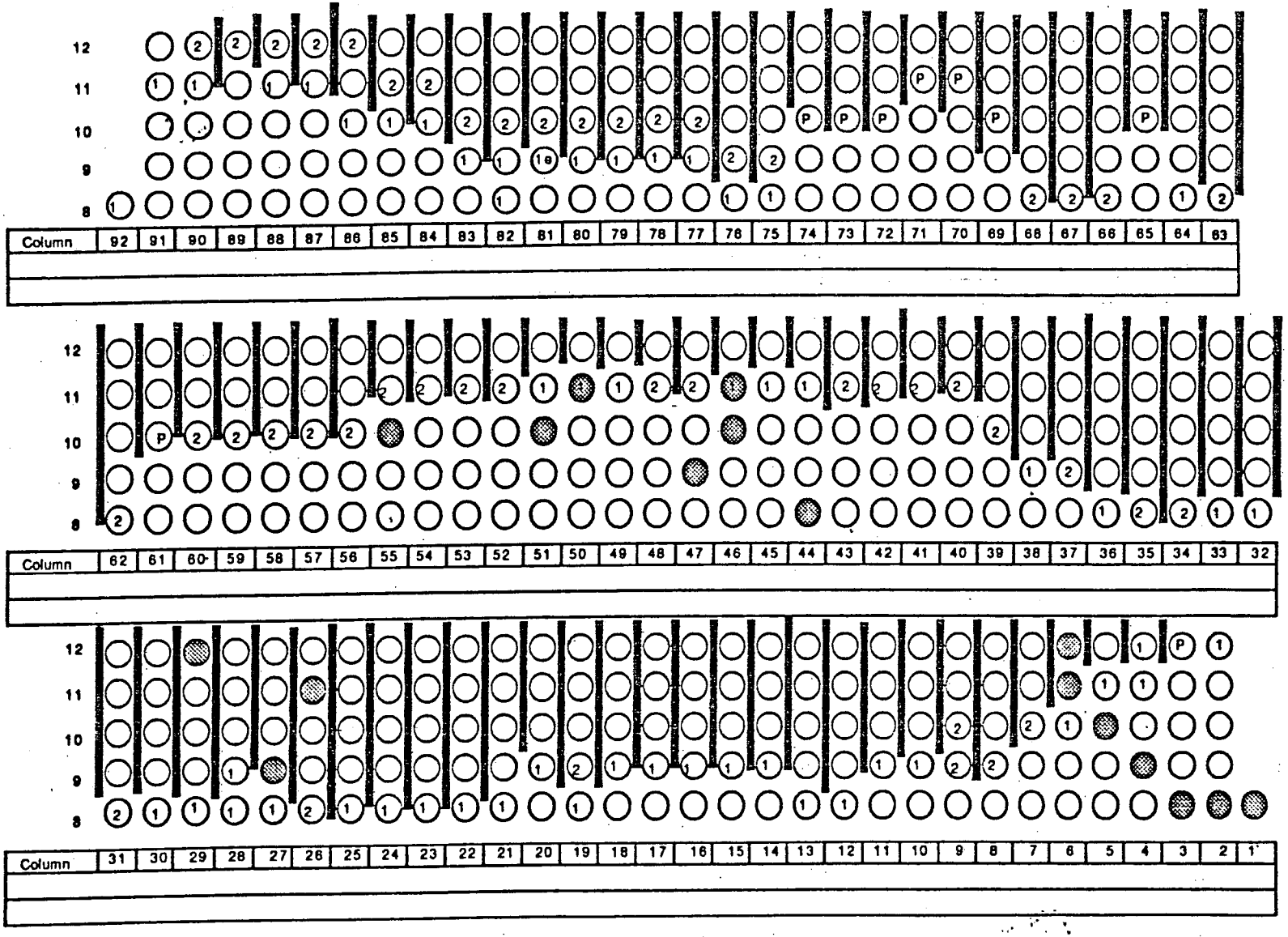


NOTES:
Numbers in tubes refer to AVBs visible
P - AVBs Located by Projection

Plugged Tube

11/11/87
bol

Indian Point - 2 : S/G -24
AVB Insertion



NOTES:

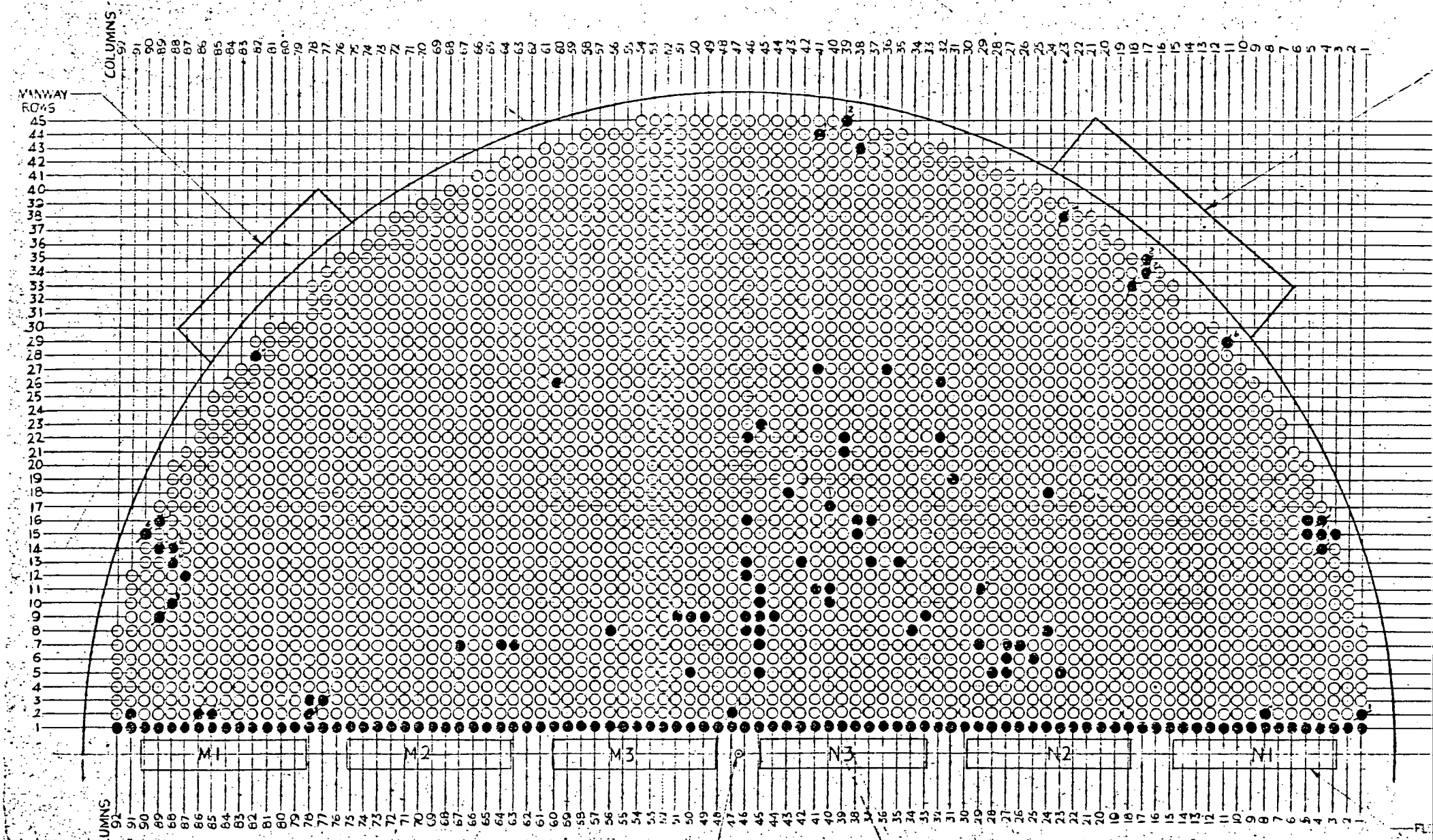
Numbers in tubes refer to AVBs visible

P - AVBs Located by Projection

Indications in Columns 2 and 92 are retainer rings

Plugged Tube

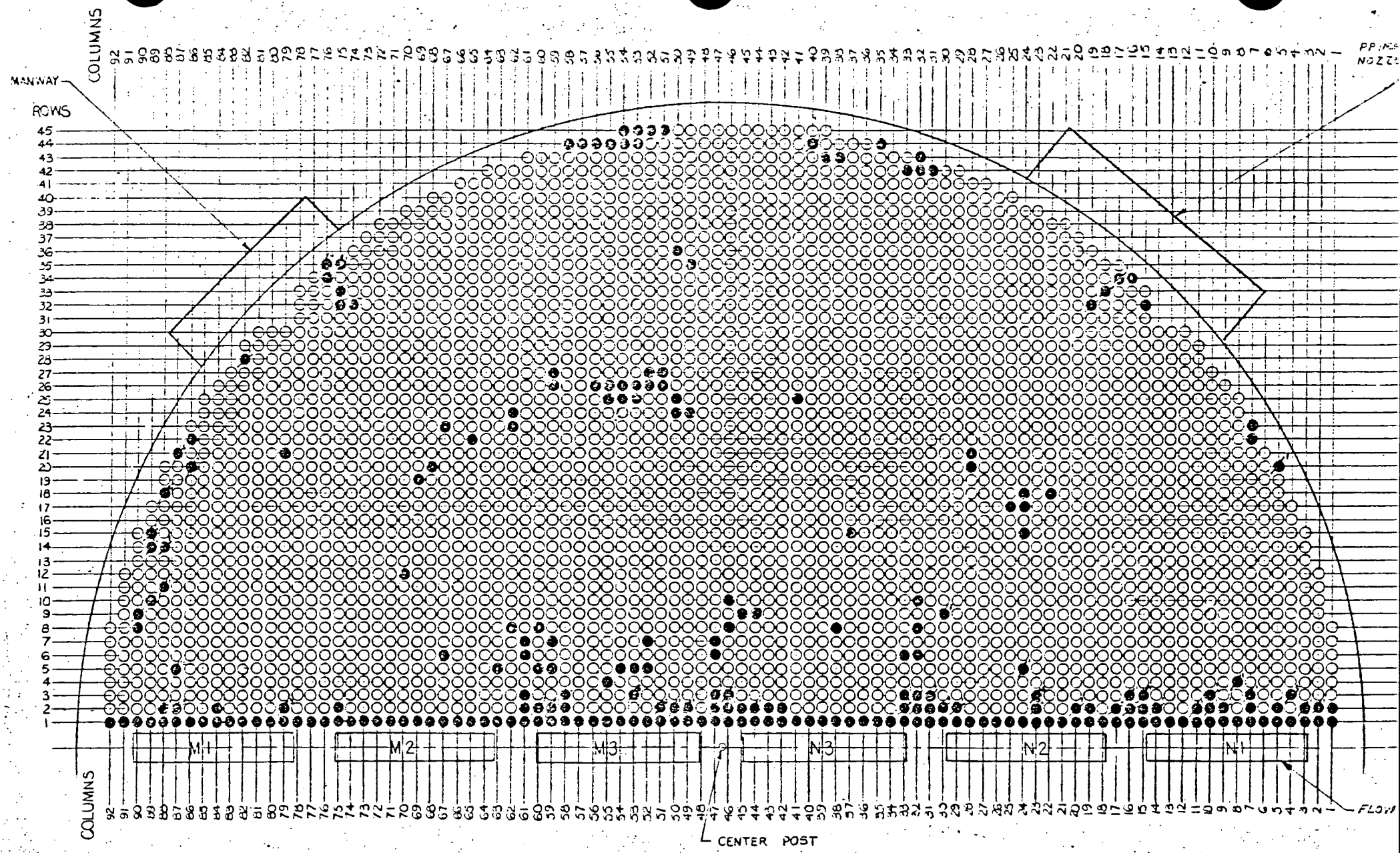
11/10/87:a
hol



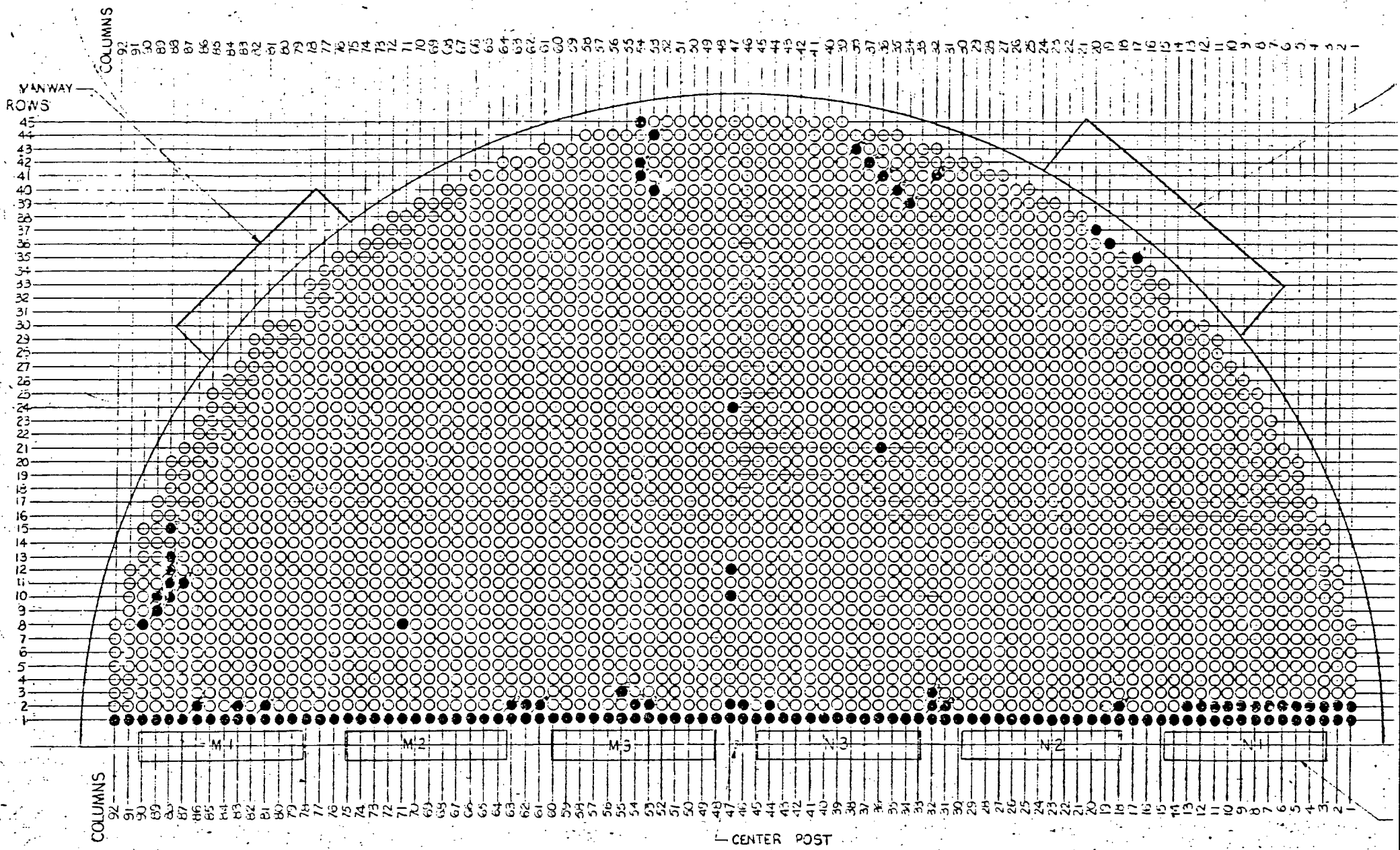
#21 STEAM GENERATOR TUBE IDENTIFICATION LAYOUT

○ = UNPLUGGED TUBE

● = PLUGGED TUBE



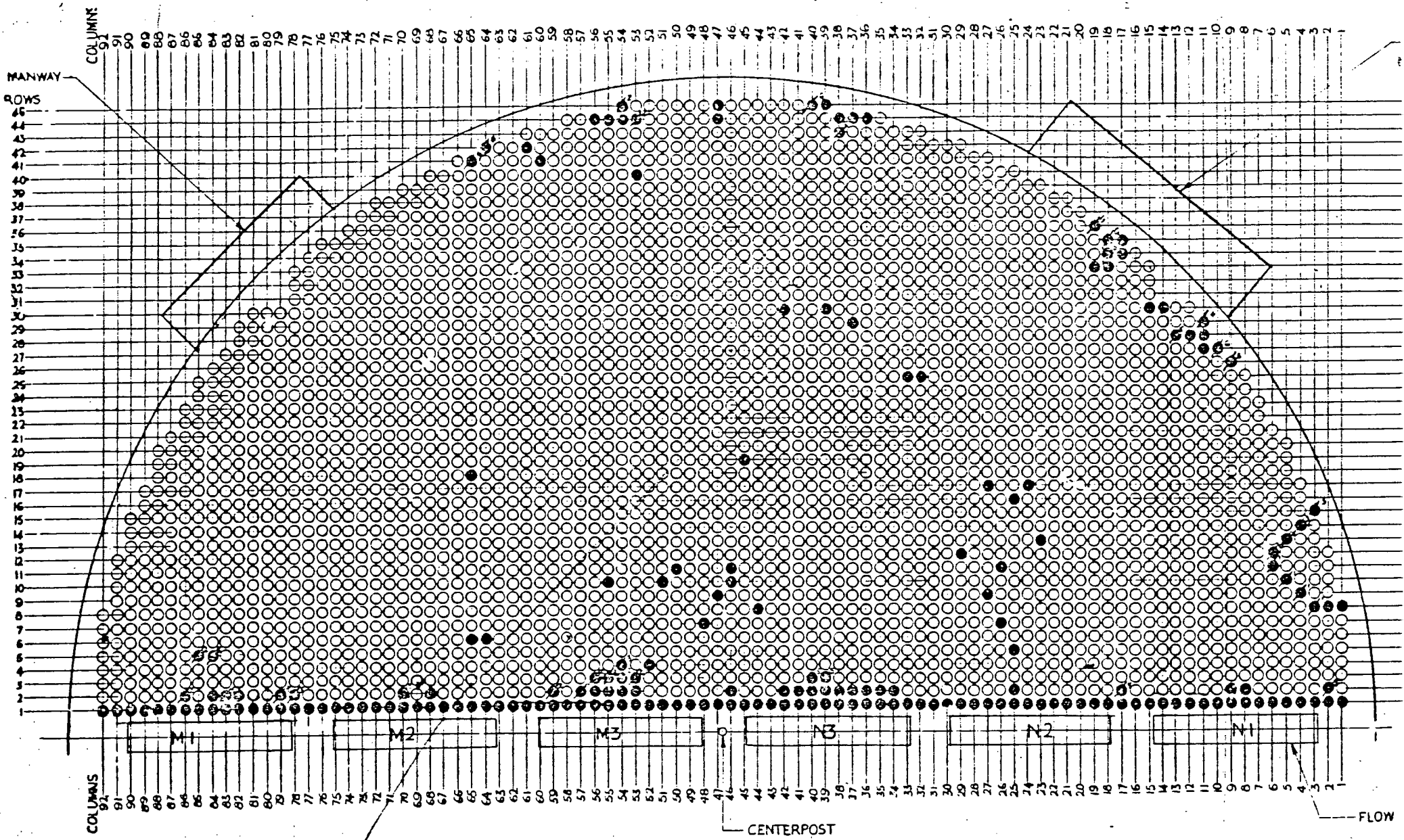
#22 STEAM GENERATOR TUBE IDENTIFICATION LAYOUT



#23 STEAM GENERATOR TUBE IDENTIFICATION LAYOUT

○ = UNPLUGGED TUBE

● = PLUGGED TUBE



#24 STEAM GENERATOR TUBE IDENTIFICATION LAYOUT

○ = UNPLUGGED TUBE

● = PLUGGED TUBE

ATTACHMENT 4

6.0 Metal Impact Monitoring System

6.1 Function

The Metal Impact Monitoring System (MIMS) provides reliable detection of loose metallic parts impacting within the reactor coolant system (reactor vessel and primary side of steam generator) and secondary side of the steam generators. By utilizing micro-processor technology, the MIMS provides continuous and automatic data reduction as well as display all essential information to plant operators.

6.2 Introduction

The MIMS consists of specially designed high sensitivity transducers at the natural collection points of the reactor vessel, reactor head and steam generators, amplifiers, signal conditioners, digital data processing and analysis, alphanumeric displays, speaker, printer, adjustable alarms, and optional tape recorder. A block diagram of the System is shown in Figure 18.

Impacts cause a small electrical charge proportional to acceleration to be generated by the piezoceramic accelerometers. The charge is transmitted by noise resistant, mineral-insulated hardline cable to the charge preamplifier inside the containment where it is converted to a voltage signal. The voltage signal is transmitted by normal shielded cable to the signal conditioner in the MIMS cabinet outside

of the containment and then to the Metal Impact Monitor Board where the analog signals are converted to digital form for processing by the central processor.

The transducers can be mounted in different modes. As shown in Figure 19, a screw provides the basic connection point. For the reactor vessel head and reactor vessel bottom, two transducers are mounted at each monitoring point for redundancy since they are inaccessible during normal operation. For the reactor Vessel bottom, transducers located at stud 33, are magnetically attached with clamps. As shown in Figure 20. the transducer for the reactor vessel head are tapped into its lifting lugs.

Only one active transducer per steam generator is necessary to detect metallic impacts on both the primary and secondary side of the steam generator due to the especiall high sensitivity of the transducers. The location of the tapped in steam generator transducers is shown in Figure 21.

6.3 Impact Measurement and Discrimination

Upon the occurrence of a metallic impact, resultant minute accelerations are generated in the reactor system component material and detected by the accelerometer (transducers). The acceleration signal frequencies are significant since they are proportional to the energy imparted by the impact

and provide a measure of the mass of the object. For this reason, acceleration resulting from the impact is the parameter chosen to monitor.

Extensive developmental testing by the vendor has determined that most metallic impacts in the reactor coolant system will occur within a frequency range of 2kHz to 10 kHz. The Metal Impact Monitor System employs accelerometers which have a linear response between 5 Hz and 20 kHz.

The MIMS has a potential noise limit of better than 0.002 g peak-to-peak within a temperature of 0°F to 650°F. The system has the capability to detect impact of about 0.02 ft-lb at 2 feet from the transducer. The sensitivity of impact measurements is determined by the impact energy which is a function of mass, impact velocity, the geometry of the impacting particle, the surface being impacted, and the distance from the transducer.

Signals from the preamplifiers located at 29'El in containment are carried through containment penetrations to the system electronics cabinet. The cabinet includes signal conditioners and amplifier units, a metal impact monitor drawer, and an interconnection panel.

The heart of the system is the metal impact monitor drawer illustrated in Figure 22. The drawer contains a metal impact monitor board, a utility card, and a control processor, all mounted in a suitable card cage with power supplies.

Each metal impact monitor board contains up to six independent microprocessor-based circuits. Each circuit has the capability of monitoring either of two redundant transducers, preamplifier, and amplifier chains. Switching between chains is accomplished under operator control (manually) or, if a channel failure is detected, by the central processor (automatically).

Digital processing allows the utilization of novel descrimination techniques to discriminate between signals generated by metallic impacts from mechanical noise and electrical impluses due to normal plant operation of pumps, valves, and other equipment. By carefully analyzing the characteristic patterns of each signal, rather than relying solely on impact amplitude, the MIMS is able to virtually eliminate false alarms. Since the alarm level for each channel is fixed at some point above the maximum normal background noise level, there can at times be a significant gap between the alarm level and the actual background level. This is especially true during plant startup when most loose parts originate. Therefore, in addition to individual fixed alarm setpoints for each channel the MIMS provides a floating threshold

which rises and falls with the background noise level as shown in Figure 23. Small impacts below the alarm level and above background noise levels are detected and recorded for investigation. This ability to detect small impacts can provide invaluable time to mitigate damage before it becomes severe.

6.4 System Operation

The design of the metal impact monitor places a great deal of emphasis on simplicity of operation; the use of a microprocessor results in a system which is both easy to use and capable of presenting a great deal of information. Furthermore, the careful use of digital electronics results in a system which needs little or no periodic recalibration; the only adjustments on the processing electronics drawer are for the output voltages of the DC power supplies.

The operator communicates with the system almost entirely through the front panel display and keyboard, so that we can describe system operation by explaining how the various keyboard keys are used. The keyboard is physically and functionally divided in two sections; the smaller, on the left hand side of the front panel, contains those keys (mostly numeric) which are used to respond to system requests for information, while the larger section is used to enter system commands. We shall now discuss some of the keys in

this larger section. Note that the keylock switch on the front panel can be used to prevent any parameter modifications from being performed ("OPERATE" position), and to disable control board alarms during system testing("TEST" position).

6.4.1 Modifying Operating Parameters

The left hand column of system command keys is entirely concerned with displaying and modifying system operating parameters. Depressing any of the top four keys in this column results in the presentation of the display panel of the system status display shown in the following Figure 24. The accelerometer locations are displayed across the top row of the display (Reactor Vessel Upper, Reactor Vessel Lower, and Steam Generators 1 through 4). The second row shows, by means of the transducer tag number, which of the two transducers at each location is currently being monitored. The third row shows either the assumed gain setting of the related signal conditioners (these values are used by the system to calculate the factors for converting volts to g; it is the operator's responsibility to ensure that the values match the settings on the signal conditioners themselves), or the impact logging threshold. The logging threshold is that impact amplitude below which

the system will entirely ignore impacts. It would normally be set to zero, thus causing all impacts to be logged, but experience has shown that in the event of frequent minor impacts, such as might be caused by the presence of a small loose part it is desirable to be able to prevent the logging of low amplitude impacts. In this manner the occurrence of larger impacts is more rapidly observed. The fourth row of the display shows the channel alarm setpoints and the fifth an approximate measure of the coolant flow induced background signal level. The background level figure is intended for observing long term trends in signal amplitude; it should remain essentially constant during normal plant operation.

Depressing one of the four upper buttoms in the left hand column of the command section of the keyboard allows the corresponding parameters to be modified. In all cases the system status display appears and the final line of the display is used to request the necessary change information (the channel number for which a parameter is to be modified and the new parameter value) from the operator. If at any time the operator enters an impossible value for a parameter

he is prompted to repeat the entry. As changes are made the status display is modified to reflect the new system operating parameters.

In this manner the operator can alter the transducers selected for monitoring, the full scale settings of the signal conditioners, the logging thresholds and the alarm setpoints. The fifth key in the left hand column allows him to set the system clock and calender.

6.4.2 Obtaining Events Reports

The upper keys in the second column allow the operator to display collected information concerning events which have occured in the last two days. All impact related information is collected on a daily basis, and impact counts, rates, etc., are reset at midnight each night. Two events on a single channel within a one minute period are required before an events report will be formed or a control board alarm generated. Isolated events (i.e., more than a minute apart) are assumed to be spurious indications unless an events report is already in existence for the channel in question; a separate report is complied to retain isolated event data.

For each processing channel three reports are available: today's yesterday's and isolated events. All event report are automatically printed at midnight. A typical events report is shown in Figure 25. All information is continuously updated. When an events report is requested by the operator, a list of those channels on which events have occured is displayed. He then selects the channel number for the report which he wishes to view.

The list of channels on which events have occurred is also a default display, which appears whenever the keyboard has not been activated in the last ten minutes. Events reports for single events and for yesterday's events are very similar to the one shown in Figure 24.

6.4.3 Other Functions

Other functions which can be performed from the front panel keyboard include the selection of a transucer signal for aural monitoring, system self test, and system printer and (optional) tape recorder control.

System self test is divided into functional units, including display, keyboard, program and data memory, impact detection modules and channel integrity. The

channel integrity test makes use of the known characteristics of the signals between the charge preamplifiers (inside containment) and the signal conditioners (in the equipment rack) to verify that the cable between the two units is neither short nor open circuited, and that the charge preamplifier is functional. A further semi-automatic test allows the operator to verify calibration of the signal conditioners and impact detection modules, and the correctness of complied events reports. Fault reports, together with a simple set of instructions provided in the system manual, permit the isolation of faults to a specific circuit board. Self test is performed automatically at midnight each night; if a fault is detected a message is dispalyed which instructs the operator to repeat the test to determine the cause of the failure.

The final keyboard column allows the operator to request that either the date and time or the current front panel display be transmitted to the system printer. Since these oprations can be performed even while other transactions with the system are occurring it is possible to maintain a complete log of all operator interactions with the system. Another key controls operation of the optional system tape recorder.

6.4.4 MIM Module

The main purpose of the MIM Module is to distinguish valid metal impact signals from noise and background signals which are normally present on the input from the Signal Conditioner. A valid metal impact signal is defined to have all the following four characteristics:

1. Frequency: equal to the predominant resonant frequency of the monitored component as determined by the initial frequency determination by a spring loaded punch.

2. Amplitude: at least twice the maximum amplitude of the normal backgroud signal.

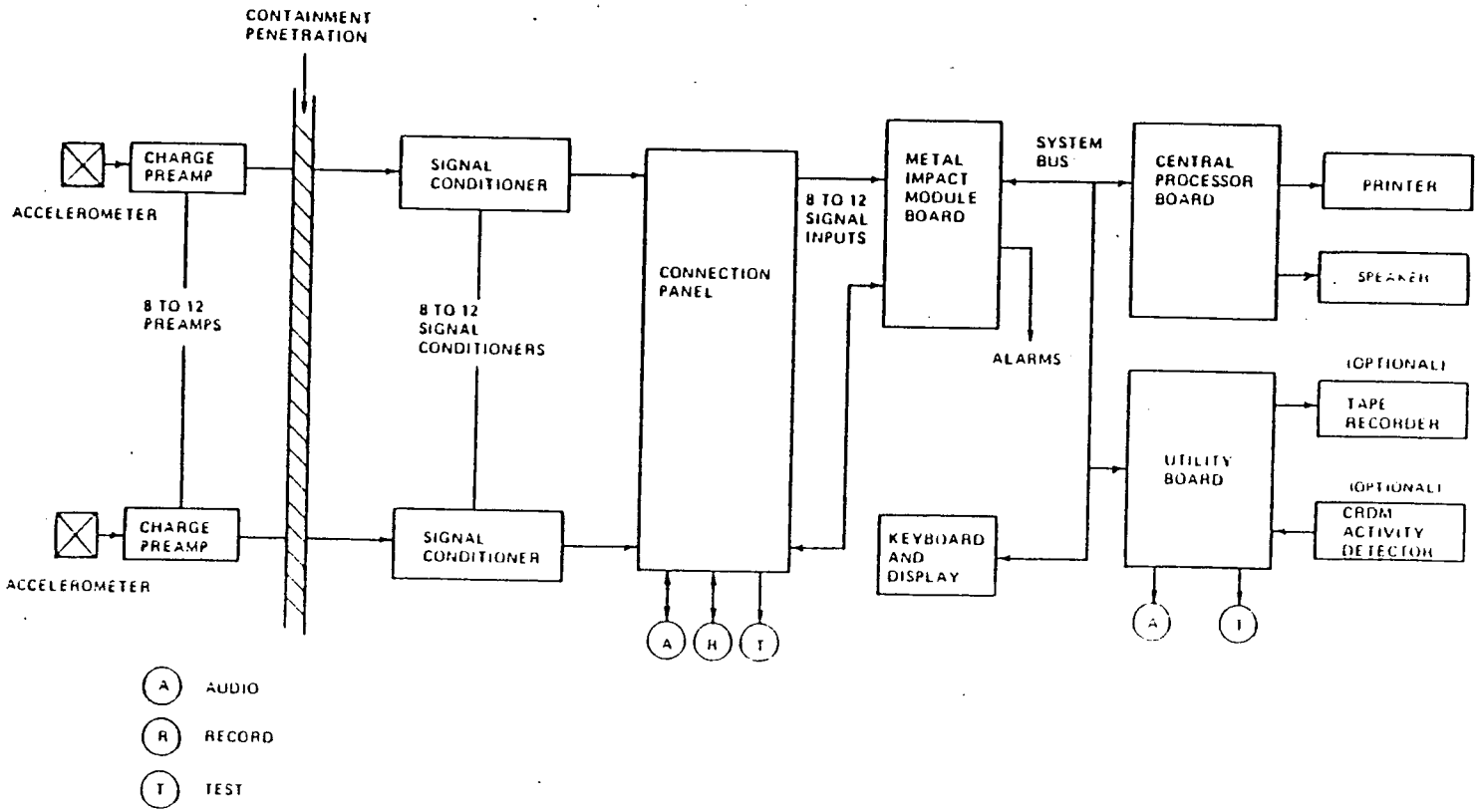
3. Duration: greater than two milliseconds. Normal electrical noise spikes are less than two milliseconds duration.

4. Time of Occurrence: does not occur when Inhibit input signal is present.

6.5 Power Supply

Lighting Panel 219, Crk. 20

Location Rack E-7 in Control Room



Block Diagram of the Metal Impact Monitoring System

Figure 18

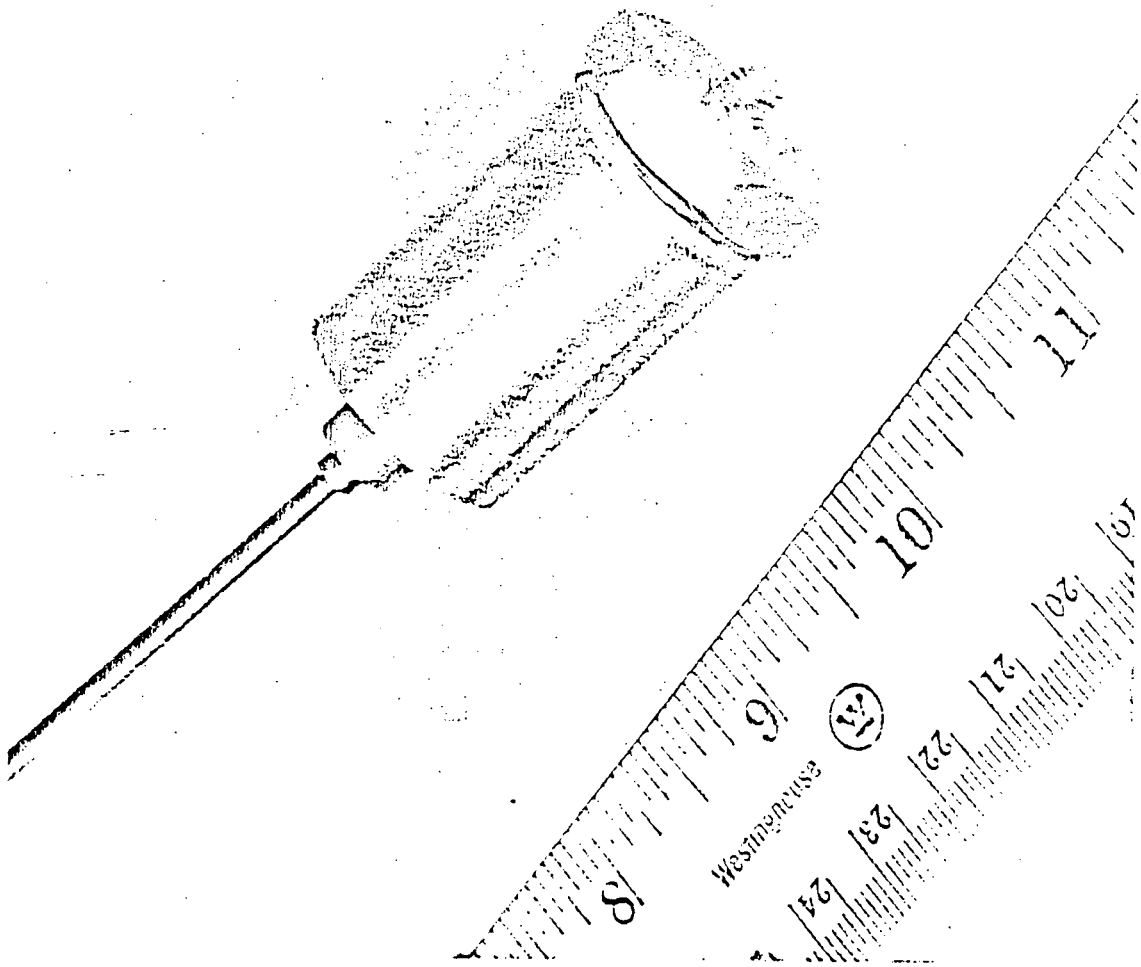
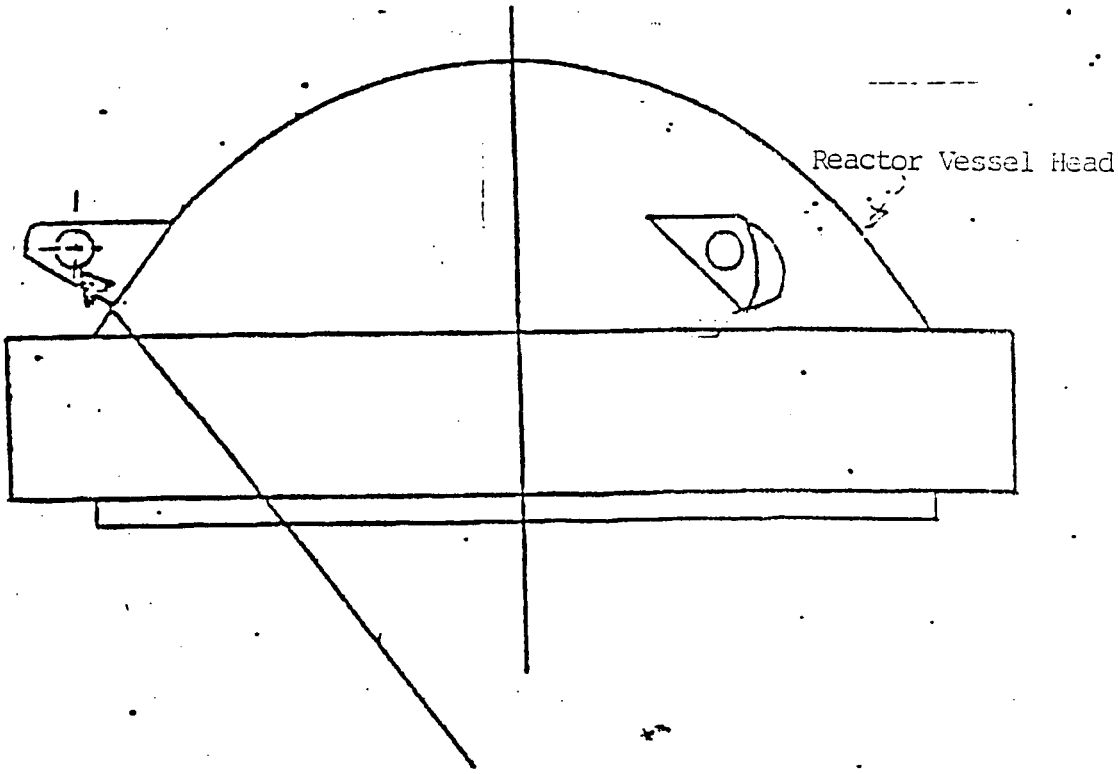


Figure 19

Transducer



Accelerometer Location
For The Reactor Vessel Head

Figure 20

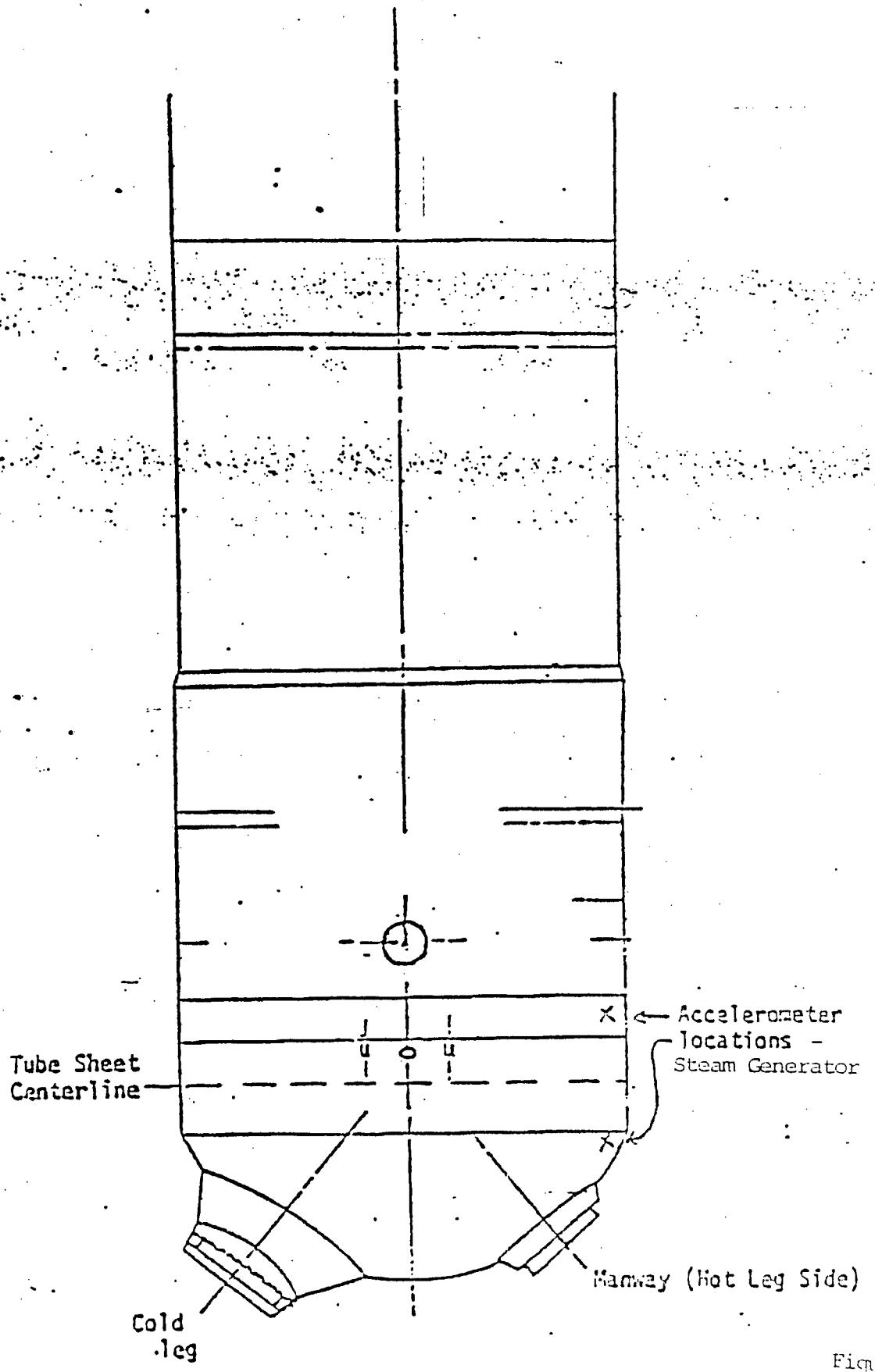
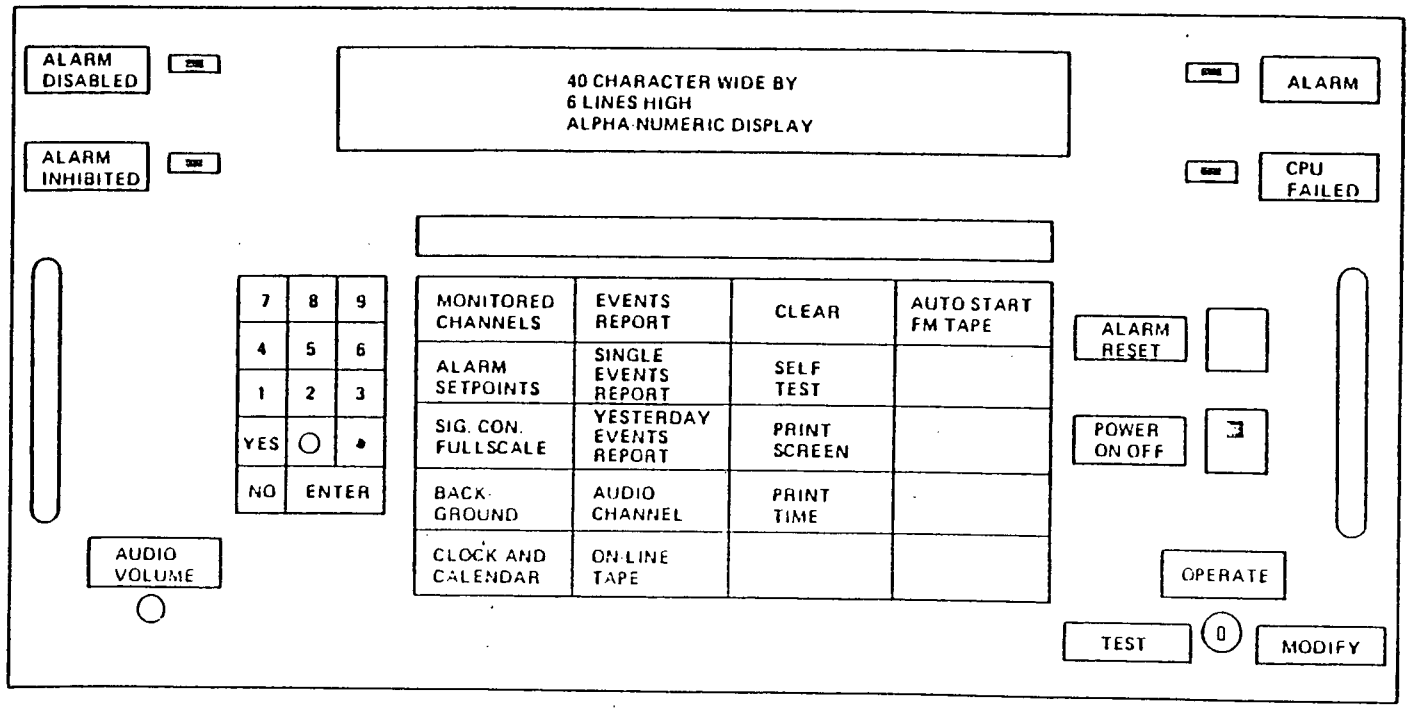
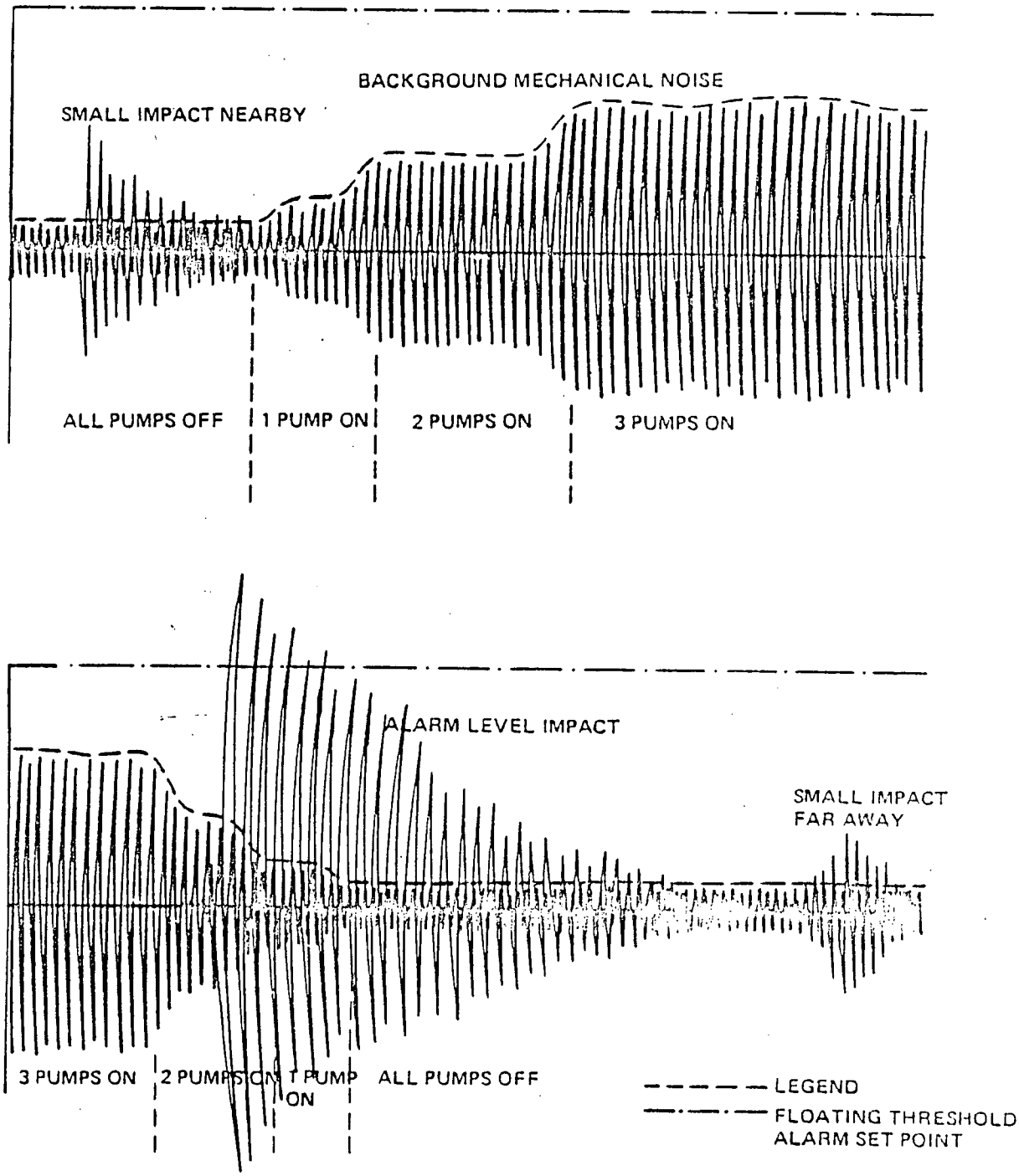

Accelerometer locations - Steam Generator
Tube Sheet Centerline
Manway (Hot Leg Side)
Cold leg

Figure 21



MIMS Drawer Front Panel

Figure 22



Floating Threshold Level
Figure 23

LOCATION	RVU	RVL	SG1	SG2	SG3	SG4
TRANSDUCER	750	752	754	756	758	760
S.C.F.S.	3.0	3.0	3.0	3.0	3.0	3.0
SETPOINT	1.5	1.5	2.0	2.0	2.0	2.0
BACKGROUND	0.6	0.7	0.8	0.7	0.8	0.7

Systems Status Display

Figure 24

LOCN:	RVU	AVG AMPL:	1.2	FIRST EVENT
CH NO.:	750	MAX AMPL:	2.6	07: 49: 13
SETP:	1.5	AVG RATE:	2.3	LAST EVENT
NO. >SP:	23	MAX RATE:	12	08: 19: 23
NO. <SP:	47	NO. SEC EV:	1	AMPL: 1.7
NO. OF 1 MINUTE PERIODS WITH IMPACTS: 12				

Events Report Display

Figure 25