Stephen B. Bram
Vice President

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, NY 10511
Telephone (914) 737-8116

March 25, 1988

Re: Indian Point Unit No. 2
Docket No. 50-247

Mr. William Russell
Regional Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1498

SUBJECT: Response to NRC Bulletin No. 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"

Dear Mr. Russell:

Transmitted as Attachment I to this letter is our response to NRC Bulletin No. 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," dated February 5, 1988. This response summarizes the actions taken by Con Edison prior to the issuance of Bulletin No. 88-02 and the information provided to NRC in Con Edison letters dated November 23, 1987, December 17, 1987 and December 23, 1987. For completeness of reference, and as part of our response to Bulletin No. 88-02, these letters are resubmitted herewith as Attachment IV, III, and II, respectively.

In our letters of November 23, 1987 and December 17, 1987, we submitted our evaluations of the susceptibility of IP-2 steam generator U-bend tubing to fatigue failure of the type experienced at North Anna-1 on July 15, 1987. In those submittals we concluded that the tubes remaining in service are not expected to be susceptible to fatigue rupture at the top support plate in a manner similar to North Anna.

In our letter of December 23, 1987, we submitted our description of the primary-to-secondary leak rate determination procedural enhancements which were made before IP-2's startup from the previous refueling outage. These procedural enhancements were discussed with the NRC staff on December 18, 1987, are consistent with the leak-rate versus time curves supplied by the NRC and provide reasonable assurance that timely corrective action would be taken before the occurrence of fatigue-induced gross tube rupture similar to the type experienced at North Anna-1. Furthermore, we stated in the December 23, 1987 letter that these procedural enhancements will remain in effect pending resolution of concerns regarding the potential susceptibility of IP-2's steam generators to high-cycle fatigue tube failure. At such time, we will evaluate the need to continue these procedural enhancements.

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March 25, 1988

Re: Indian Point Unit No. 2
Docket No. 50-247

Our response is provided pursuant to the provisions of Section 182a, Atomic Energy Act of 1954, as amended. Should you or your staff have any questions, please contact us.

Very truly yours,

John A. Basile

Subscribed and sworn to
before me this 25th day
of March, 1988.

ANTHONY R. ARNONE
Notary Public, State of New York
No. 4883047
Qualified in Westchester County
Commission Expires January 26, 1989

Anthony R. Arnone
Notary Public

cc: Document Control Desk (Orig. & Copy)
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Washington, DC 20555

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Senior Resident Inspector
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50-247

RESPONSE TO NRC BULLETIN 88-002

w/ltr 3-25-88 #8804080139

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-NOTICE-

Attachment I

Response to NRC Bulletin No. 88-02
"Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March 25, 1988

NRC Bulletin 88-02, dated February 5, 1988, requested that licensees having Westinghouse steam generators with carbon steel support plates submit a written report detailing the status of Licensee compliance with actions provided therein for purposes of minimizing the potential for rapidly propagating fatigue failure such as occurred at North Anna-1. Our response to each action item is provided below.

NRC Action Item A:

The most recent steam generator inspection data should be reviewed for evidence of denting at the uppermost tube support plate. Inspection records may be considered adequate for this purpose if at least 3% of the total steam generator tube population was inspected at the uppermost support plate elevation during the last 40 calendar months. "Denting" should be considered to include evidence of upper support plate corrosion and the presence of magnetite in the tube-to-support plate crevices, regardless of whether there is detectable distortion of the tubes. The results of this review shall be included as part of the 45-day report. Where inspection records are not adequate for this purpose, inspections of at least 3% of the total steam generator tube population at the uppermost support plate elevation should be performed at the next refueling outage. The schedule for these inspections shall be included as part of the 45-day report and the results of the inspections shall be submitted within 45 days of their completion. Pending completion of these inspections, an enhanced primary-to-secondary leak rate monitoring program should be implemented in accordance with paragraph C.1. below.

Response:

The most recent steam generator inspection data was reviewed for evidence of denting at the uppermost tube support plate. The results of that review were transmitted to the NRC by Con Edison letters dated November 23, 1987 (Attachment IV) and December 17, 1987 (Attachment III). In those letters we stated that the analysis of eddy current measurements shows that denting is present at the top support plate and therefore all of the tubes in Rows 8 through 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.

The results of the steam generator tube in-service examination performed during the last refueling outage were reported to NRC by Con Edison letter dated January 22, 1988 (Attachment V). The 1987 steam generator tube in-service examination demonstrates that the Indian Point Unit No. 2 steam generators are acceptable for continued service at full power. Based on the results of this examination, the next steam generator examination has been scheduled for the next refueling outage.

NRC Action Item B:

For plants where no denting is found at the uppermost support plate, the results of future steam generator tube inspections should be reviewed for evidence of denting at the uppermost support plate. If denting is found in the future, the provisions of item C below should be implemented. Commitments to implement these actions shall be submitted when the results of A above are submitted.

Response:

As stated in response to NRC Action Item A, analysis of the most recent steam generator eddy current measurements shows that denting is present at the top support plate. As such, Item B is not applicable.

NRC Action Item C1:

For plants where denting is found, the NRC staff requests that the following actions be taken:

1. Pending completion of the NRC staff review and approval of the program described in C.2 below or completion of inspections specified in item A above to confirm that denting does not exist, an enhanced primary-to-secondary leak rate monitoring program should be implemented as an interim compensatory measure within 45 days of the date of receipt of this bulletin.* Implementation of this program shall be documented as part of the 45-day report. The enhanced monitoring program is intended to ensure that if a rapidly propagating fatigue crack occurs under flow-induced vibration, the plant power level would be reduced to 50% power or less at least 5 hours before a tube rupture was predicted to occur. The effectiveness of this program should be evaluated against the assumed time-dependent leakage curve given in Figure 1.

   This program should consider and provide the necessary leakage measurement and trending methods, time intervals between measurements, alarms and alarm setpoints, administrative limits for commencing plant shutdown, and time limitations for (1) reducing power to less than 50% and (2) shutting down to cold shutdown. Appropriate allowances for instrument errors should be considered. Finally, the program should make provision for out of service radiation monitors, including action statements and compensatory measures.

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*While this bulletin was being prepared, licensees for a few plants committed to an enhanced primary-to-secondary leak rate monitoring program at the staff's request. These plants had been identified on a preliminary basis by Westinghouse as being potentially susceptible to rapidly propagating fatigue cracks.

These enhanced programs should be upgraded as necessary to comply with this paragraph. However, no relaxation from current commitments should be made without prior approval by the NRC staff.

Response:

In our November 23, 1987 letter we stated that we had assessed our primary-to-secondary leak detection capability to assure that IP-2 systems and procedures are capable of producing leak-rate data that would allow for positive detection and timely corrective actions before any fatigue-induced gross tube failure were to occur. A description of our overall capability and the administratively imposed leak rate limits, together with their associated corrective actions, was provided in that letter.

Changes to our procedures, to reflect our enhanced primary-to-secondary leak rate monitoring program, were made before we started-up from our last refueling outage. Our December 23, 1987 letter (Attachment II) described these enhancements and resulted from our review of the new leak-rate versus time curves provided by the NRC staff. Furthermore, these enhancements were discussed with the NRC staff on December 18, 1987 and are consistent with the NRC provided leak-rate versus time curves.

NRC Action Item C.2:

A program should be implemented to minimize the probability of a rapidly propagating fatigue failure such as occurred at North Anna Unit 1. The need for long-term corrective actions (e.g., preventive plugging and stabilization of potentially susceptible tubes, hardware, and/or operational changes to reduce stability ratios) and/or long-term compensatory measures (e.g., enhanced leak rate monitoring program) should be assessed and implemented as necessary. An appropriate program would include detailed analyses, as described in subparagraphs (a) and (b) below, to assess the potential for such a failure. Alternative approaches and/or compensatory measures implemented in lieu of the actions in subparagraphs (a) or (b) below should be justified.

Although the 45-day report shall provide a clear indication of actions proposed by licensees, including their status and schedule, a detailed description of this program and the results of analyses shall be submitted subsequently, but early enough to permit NRC staff review and approval prior to the next scheduled restart from a refueling outage. Where the next such restart is scheduled to take place within 90 days, staff review and approval will not be necessary prior to restart from the current refueling outage. An acceptable schedule for submittal of the above information should be arranged with the NRC plant project manager by all licensees to ensure that the staff will have adequate time and resources to complete its review without adverse impact on the licensee's schedule for restart.

(a) The analysis would include an assessment of stability ratios (including flow peaking effects) for the most limiting tube locations to assess the potential for rapidly propagating fatigue cracks. This assessment would be conducted such that the stability ratios are directly comparable to that for the tube which ruptured at North Anna.

(b) The analysis would include an assessment of the depth of penetration of each AVB. The purpose of this assessment is twofold: (1) to establish which tubes are not effectively supported by AVBs and (2) to permit an assessment of flow peaking factors.

(Note: Most steam generators have at least two sets of AVBs. This applies only to the set that penetrates most deeply into the tube bundle.) The methodolgy used to determine the depth of penetration of each individual AVB shall be described in detail in the written report. The criteria for determining whether a tube is effectively supported by an AVB shall also be identified. (Note: An AVB that penetrates far enough to produce an eddy current signal in a given tube may not penetrate far enough to provide a fully effective lateral support to that tube.)

If addressees cannot perform this suggested approach or meet this suggested schedule, they should justify to the NRC their alternative approaches and schedules.

Response:

Upon receiving notification from Westinghouse that IP-2, based upon a very preliminary review, may be susceptible to the North Anna type generator tube failure mechanism (i.e. high cycle fatigue), we initiated several plant-specific analyses with Westinghouse to more accurately determine which tubes, if any, might be susceptible.

Thermal-hydraulic, fatigue and vibration analyses that incorporated an accurate determination of anti-vibration bars (AVB) insertion depths were performed for IP-2 steam generators Nos. 22 and 24. The results of these analyses were discussed at the November 12, 1987 meeting and reported in our November 23rd letter. Based on these analyses, we concluded that steam generators 22 and 24 are not susceptible to tube fatigue rupture at the top support plate similar to that which occurred at North Anna-1, and that accordingly no modification or preventative tube plugging was required.

By Con Edison letter dated December 17, 1987, we notified the NRC staff that the review of the eddy current measurements for steam generators 21 and 23 verified anti-vibration bars (AVB) position for all tubes except two due to copper deposit interference. These two tubes, R12C89 and R12C90 in steam generator 21, were preventively plugged to eliminate them from further consideration. All remaining tubes were shown to have relative stability ratios less than 0.9 and stress ratios less than 1.0, thereby meeting the established criteria for determining that there is no need for further corrective action.

In summary, our November 23, 1987, December 17, 1987 and December 23, 1987 letters describe the program we implemented to minimize the probability of a rapidly propagating fatigue failure such as occurred at North Anna-1.

Attachment II

Con Edison Letter to NRC
Dated December 23, 1987

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March 25, 1988

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, New York 10511-1099

December 23, 1987

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Susceptibility of Indian Point Unit No. 2 (IP-2) Steam Generators to High Cycle Fatigue Tube Failure

In our letters of November 23, 1987 and December 17, 1987 we submitted our evaluation of the susceptibility of steam generator U-bend tubing to fatigue failure of the type experienced at North Anna-1 on July 15, 1987. In those submittals we concluded that the tubes remaining in service are not expected to be susceptible to fatigue rupture at the top support plate in a manner similar to North Anna. In subsequent discussions with the NRC staff we indicated that our operating practices relative to primary-to-secondary leaks were more conservative than those permitted under the applicable Technical Specifications. We were asked to apprise the staff of the operating procedures which will be in place before unit startup from the current outage, including certain procedural enhancements intended to further confirm incipient indications of potential North Anna-type tube failures.

This letter describes our revised procedural enhancements to determine primary-to-secondary leak rate resulting from our review of new leak-rate versus time curves provided by the NRC staff. These enhancements were discussed with the NRC staff on December 18, 1987 and are consistent with the NRC provided leak-rate versus time curves. A description of these enhancements is contained in Attachment I to this letter. Changes to our procedures will be made before startup from the current refueling outage and provide reasonable assurance that timely corrective action would be taken before any fatigue-induced gross tube rupture similar to that at North Anna were to occur.

These procedure revisions will remain in effect pending resolution of concerns regarding the potential susceptibility of the unit's steam generators to high cycle fatigue tube failure. At such time, we will evaluate the need to continue these procedural enhancements.

If you have any questions on this matter, please contact us.

Very truly yours,

Stephen Bram
Vice President
Nuclear Power

attachment

8801070235 5pp.

cc: Mr. Robert Capra, Director
Project Directorate I-1
Division of Reactor Projects I/II
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. William Russell
Regional Administrator - Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
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Ms. Marylee M. Slosson, Project Manager
Project Directorate I-1
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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, NY 10511

ATTACHMENT I

Temporary Procedural Enhancements
Steam Generator Leakage Surveillance and Corrective Action

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
December 23, 1987

PROCEDURAL ENHANCEMENTS:

Chemistry procedure IPC-A-110 and operating procedure AOI 1.2 will be revised prior to startup from the current refueling outage to include the following provisions:

A. With Condenser Air Ejector Monitor R-15 Operable:

- If the primary-to-secondary leak rate is equal to or greater than 0.05 gpm and less than 0.2 gpm, log R-15 readings and graphically plot leak rate once every two hours. If three consecutive projected leak rate calculations indicate that leakage will be equal to or exceed 0.2 gpm within 24 hours, reduce unit load to 50% within 2 hours.

- If the primary-to-secondary leak rate is equal to or greater than 0.2 gpm, remove the unit from service and make appropriate notifications.

B. With Condenser Air Ejector Monitor R-15 Inoperable:

- If the primary-to-secondary leak rate is less than 0.05 gpm, sample and calculate the leak rate once every four hours.

- If the primary-to-secondary leak rate is equal to or greater than 0.05 gpm and less than 0.2 gpm, sample once every two hours and graphically plot the leak rate. If three consecutive projected leak rate calculations indicate that leakage will be equal to or exceed 0.2 gpm within 24 hours, reduce unit load to 50% within 2 hours.

- If the primary-to-secondary leak rate is equal to or greater than 0.2 gpm, remove the unit from service and make appropriate notifications.

Attachment III

Con Edison Letter to NRC
Dated December 17, 1987

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March 25, 1988

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, New York 10511-1099

December 17, 1987

Re: Indian Point Unit No.2
Docket No. 50-247

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Susceptibility of Indian Point Unit No. 2 (IP-2) Steam Generators to High Cycle Fatigue Tube Failure

This letter transmits a summary of the evaluation of Indian Point Unit No. 2 (IP-2) U-bend tubing for susceptibility to a fatigue failure of the type experienced at North Anna 1 on July 15, 1987. Our November 23, 1987 letter provided our completed evaluation regarding steam generators 22 and 24 and indicated that similar analyses for steam generators 21 and 23 would be performed after anti-vibration bars (AVB) insertion depths are verified for these generators by analyzing eddy current data taken during the current refueling outage. The AVB insertion depths are a key factor in the assessment of fatigue failure susceptibility.

The IP-2 eddy current (EC) testing program was recently completed and a review of the EC measurements for steam generators 21 and 23 verified AVB position for all tubes except two due to copper deposit interference. These two tubes, R12C89 and R12C90 in steam generator 21, were preventively plugged to eliminate them from further consideration. All remaining tubes were shown to have relative stability ratios less than 0.9 and stress ratios less than 1.0, thereby meeting the established criteria for determining that there is no need for further corrective action.

Attachment A presents our evaluation of fatigue susceptibility of U-bend tubes for all four steam generators. Based on this evaluation, it is concluded that the IP-2 tubes remaining in service are not expected to be susceptible to fatigue rupture at the top support plate in a manner similar to that which occurred at North Anna 1. Therefore, no additional modification, preventive tube plugging or other measure to preclude such an event is judged to be necessary.

Please contact us if you have any questions on this matter.

Very truly yours,

Vice President
Nuclear Power

Attachment

8712290039 14pp

cc: Ms. Marylee Slosson
Project Directorate I/I
Divison of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, DC 2055

Mr. William Russell
Regional Administrator - region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Senior Resident Inspector
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P.O. Box 38
Buchanan, NY 10511

ATTACHMENT A

Evaluation of Dented Tube Fatigue Susceptibility At Indian Point Unit 2

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

EVALUATION OF DENTED TUBE FATIGUE SUSCEPTIBILITY AT INDIAN POINT UNIT 2

INTRODUCTION

The Indian Point 2 steam generators have been evaluated for the susceptibility of unsupported U-bend tubing with denting at the top tube support plate to a fatigue rupture 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 when the tube experienced denting at the support plate. A combination of conditions were present that led to the rupture. The tube is not supported by an anti-vibration bar (AVB), has a higher flow field due to the uneven insertion depths of AVB's, has reduced damping because of denting at the top support plate, and has reduced fatigue properties because of the additional mean stress with the denting and in the presence of the all volatile treatment (AVT) chemistry of the secondary water.

CRITERIA FOR EVALUATION

The criteria established for determining the benefit of ameliorative action was a 10% reduction in stability ratio that is expected to provide a 58% reduction in stress amplitude ($t_o < 4.0$ 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 ruptured 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 these stress amplitudes. Therefore, the evaluation of Indian Point 2 tubing has 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 which eliminates it from further consideration 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 multiplier 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 of some tubes 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 1,2,3 and 4, 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. Twenth-eight tubes (not including symmetrical tube locations) are identified as unsupported and dented tubes with the highest potential for fluid elastic vibration. These tubes are listed in Table 1. The local flow peaking corresponding to the AVB insertion depth near each of these tubes has been factored into the evaluation for each of these tubes. The results of incorporating local flow peaking is shown in Table 2 in the relative stability ratios presented. Stress ratios are also presented in Table 2 for each of the tubes in Table 1. With the exception of two tubes in steam generator 21, all tubes have relative stability ratios less than 0.9 and stress ratios less than 1.0. The two tubes exceeding these criteria are R12C89 and R12C90 in steam generator 21. These two tubes have been plugged as a preventive measure to eliminate them from further consideration.

Of the remaining tubes that meet the criterion, the tube at R11C47 has the highest stress ratio, 0.75. The expected maximum stress ampltiude for this tube is 3.0 ksi. Based on the operating history of Indian Point Unit 2, which is presented in Figure 5 in the form of normalized stability ratio versus time in days, the fatigue usage to date is expected to be less than 0.046. Projecting this to 40 years total operation, the fatigue usage is expected to be less than 0.114. This is less than the ASME Code limit of 1.0. The fatigue usage calculation utilizes a lower bound fatigue curve that is consistent with the fatigue properties of the North Anna tube that ruptured.

CONCLUSION

Based on these evaluations, it is concluded that the Indian Point 2 tubes remaining in service are not expected to be susceptible to fatigue rupture at the top support plate in a manner similar to that which occurred at North Anna 1. Therefore, no additional modification, preventive tube plugging or other measure to preclude such an event is judged to be necessary.

TABLE 1

UNSUPPORTED AND DENTED TUBES WITH THE HIGHEST VIBRATION DISPLACEMENT POTENTIAL

Steam Generator	Row	Column
21	10	3
		16
		24
		62
	11	3
		4
		86
		87
		88
		89
	12	2
		89
		90
22	10	24
	11	3
		37
	12	2
23	10	3
		20
	11	2
		3
		4
		47
24	10	71
	11	3
		46
		51
	12	2

TABLE 2

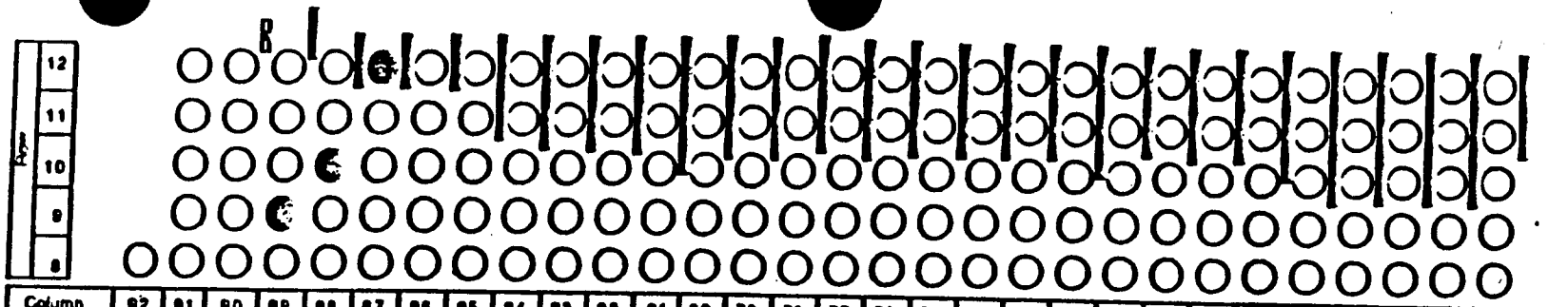
RESULTS OF EVALUATION OF UNSUPPORTED AND DENTED TUBES WITH THE HIGHEST VIBRATION DISPLACEMENT POTENTIAL

Steam Generator	Tube	Relative Stability Ratio(1)	Stress Ratio(1)
21	R10C3	0.70	0.21
	R10C16	0.68	0.19
	R10C24	0.74	0.31
	R10C62	0.75	0.33
	R11C3	0.84	0.55
	R11C4	0.82	0.50
	R11C86	0.77	0.34
	R11C87	0.79	0.40
	R11C88	0.81	0.45
	R11C89	0.82	0.50
	R12C2	0.79	0.35
	R12C89	1.10	>1.0(2)
	R12C90	1.14	>1.0(2)
22	R10C24	0.68	0.19
	R11C3	0.84	0.55
	R11C37	0.73	0.25
	R12C2	0.79	0.35
23	R10C3	0.70	0.21
	R10C20	0.74	0.31
	R11C2	0.58	0.06
	R11C3	0.84	0.55
	R11C4	0.82	0.50
	R11C47	0.88	0.75
24	R10C71	0.73	0.29
	R11C3	0.84	0.55
	R11C46	0.84	0.58
	R11C51	0.74	0.27
	R12C2	0.79	0.35

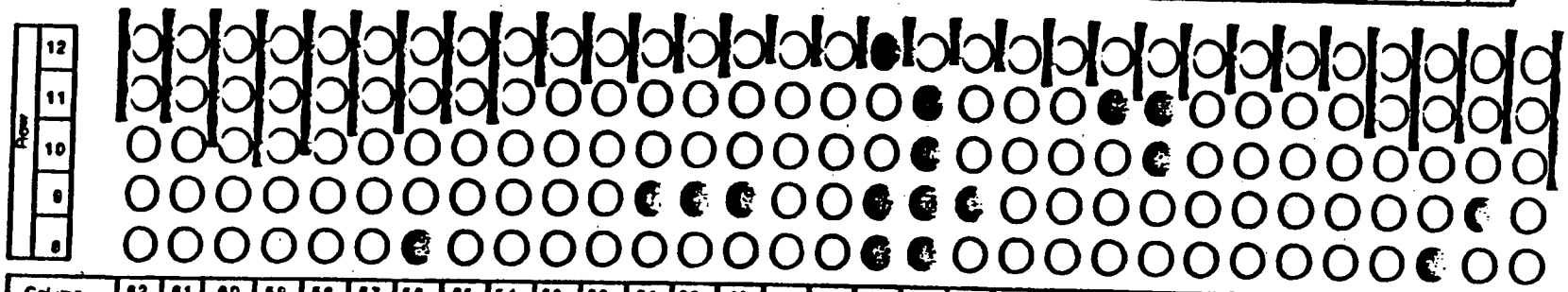
1) All ratios are in comparison to R9C51, North Anna 1, Steam Generator C.

2) This tube has been plugged as a preventive measure.

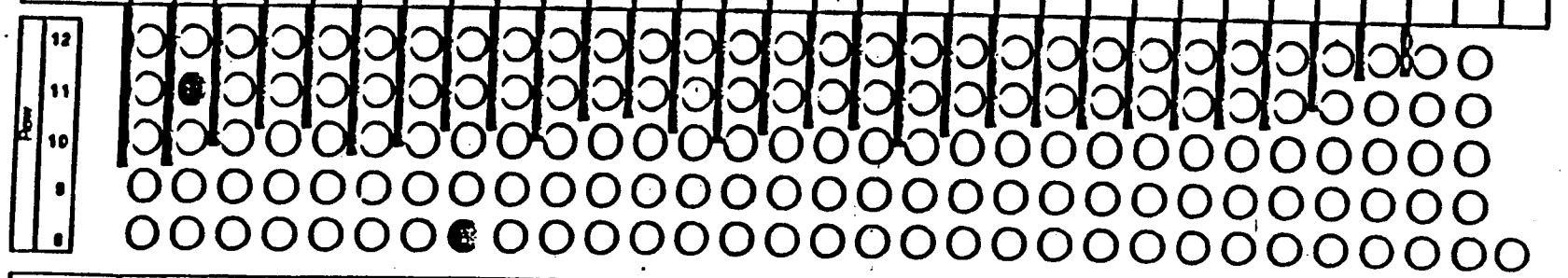
AVB Positions



Column	92	91	90	89	88	87	86	85	84	83	82	81	80	79	78	77	76	75	74	73	72	71	70	69	68	67	66	65	64	63
Projection				12.7	11.8	11.7	11.7	10.9	10.5	10.3	10.2	10.2	10	10.3	10.3	10.4	10.2	10.2	10.3	10.3	10	10.2	10.4	10.3	10.1	9.6	9.6	9.6	9.6	9.5
L1																														



Column	62	61	60	59	58	57	56	55	54	53	52	51	50	49	48	47	46	45	44	43	42	41	40	39	38	37	36	35	34	33	32
Projection	10.6	10.1	9.7	10	10.1	10.2	10.5	10.6	10.8	11.2	11.3	11.5	11.4	11.8	11.9	11.9	11.9	11.9	11.8	11.5	11.5	11.2	11.2	11.2	11.3	11.4	10.5	10.2	10.4	10.4	9.6
L1																															



Column	31	30	29	28	27	26	25	24	23	22	21	20	19	18	17	16	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
Projection	9.4	9.5	10	10.2	9.9	9.6	10	10.2	10.2	10.1	10.4	10.5	10.2	10.	10.2	10.3	10.1	10.1	10.2	10.5	10.5	10.4	10.4	10.5	10.5	10.6	11	11.7	11.7		
L1																															

NOTES:

Projections are based on AVB centerline relative to tube centerline.

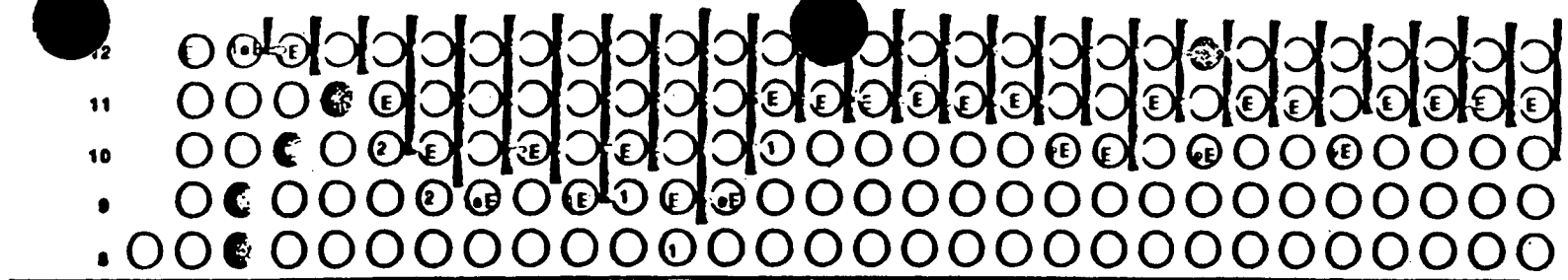
Column 3 and 89 AVB projections are based on single data points; not verified.

Plugged Tube

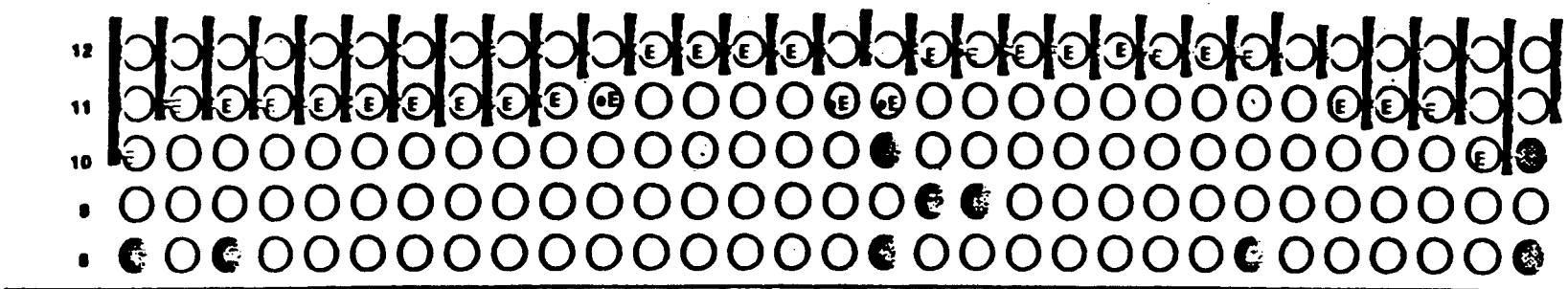
11/24/87
bel

FIGURE 1

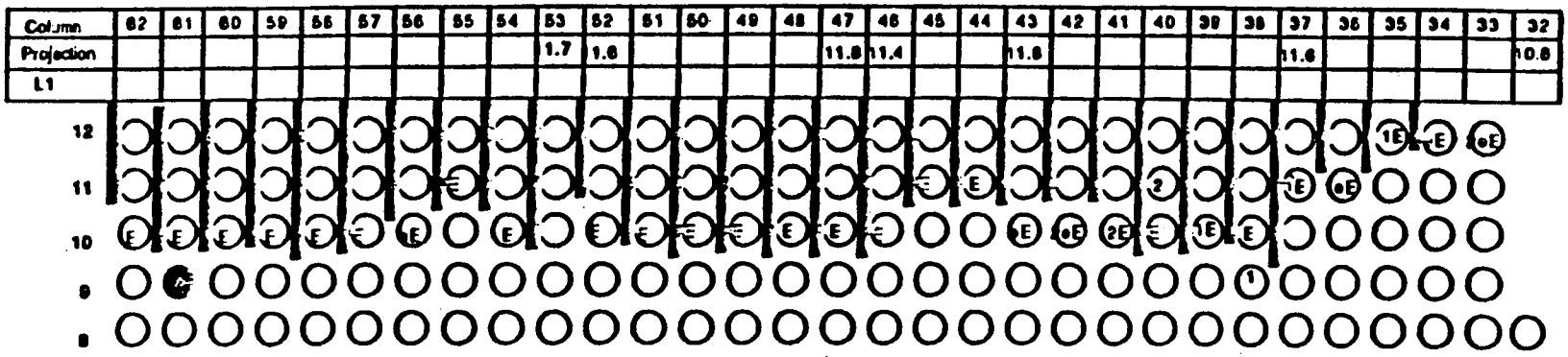
AVB Insertion



Column	92	91	90	89	88	87	86	85	84	83	82	81	80	79	78	77	76	75	74	73	72	71	70	69	68	67	66	65	64	63
Projection			12.1		11.6			10.6					10.3							10.6			10.8			11				
L1			2.8		1.8			1.1		2.8	2.3	7.6	.9	3.2																



Column	62	61	60	59	58	57	56	55	54	53	52	51	50	49	48	47	46	45	44	43	42	41	40	39	38	37	36	35	34	33	32
Projection										1.7	1.8					11.8	11.4			11.8						11.6					10.8
L1																															



Column	31	30	29	28	27	26	25	24	23	22	21	20	19	18	17	16	15	14	13	12	11	10	9	8'	7	6	5	4	3	2	1'
Projection	11.1						10.7			10.8										10.8	10.8						12		12.1		
L1																								3.6	1.4	1.8	0.9	4.0			

NOTES:

E - EPRI calls of last tube with observable AVB Indication

e - EPRI calls of AVB contacts ≤ 2.0 Inches

Numbers in tubes refer to Westinghouse Interpretation of AVBs visible

L1 - length of single AVB contacts read by Westinghouse

Projections by Westinghouse

Plugged Tube

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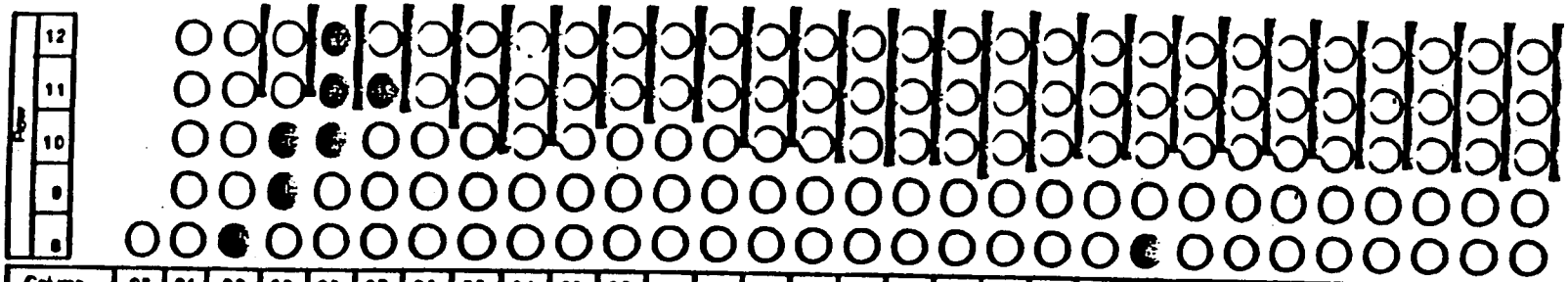
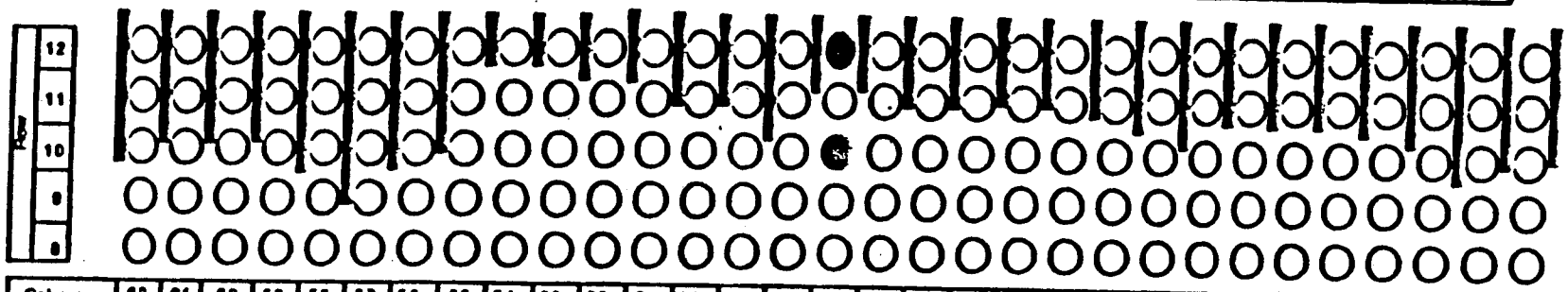
bel

FIGURE 2

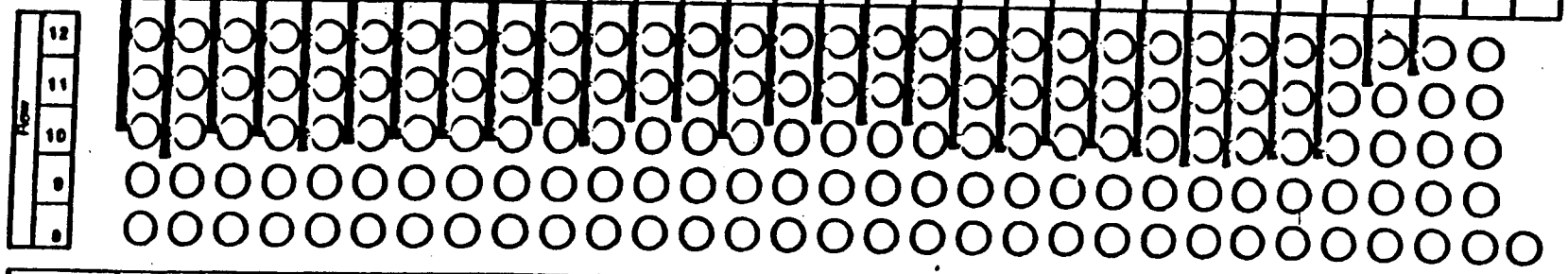
AVB Positions



Column	92	91	90	89	88	87	86	85	84	83	82	81	80	79	78	77	76	75	74	73	72	71	70	69	68	67	66	65	64	63
Projection			11	11	10.7	10.5	10.2	9.9	9.8	10	10.3	10.4	10.1	10.1	9.8	9.7	9.7	9.3	9.6	9.9	9.9	9.9	10	10	10	9.9	9.9	9.8	9.5	9.6
L1																														



Column	62	61	60	59	58	57	56	55	54	53	52	51	50	49	48	47	46	45	44	43	42	41	40	39	38	37	36	35	34	33	32
Projection	9.6	10	9.9	9.3	9.9	9.9	9.3	9.9	1.7	11.2	11.2	11.1	11	10.2	10.2	11.2	11	11	11	11	10.9	10.9	10.6	10.2	10.6	10.6	10.5	10.3	9.7	9.9	10
L1																															



Column	31	30	29	28	27	26	25	24	23	22	21	20	19	18	17	16	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
Projection	9.6	9.6	10	9.9	9.8	9.9	10.1	10.1	10	10.2	10	10.4	10.1	10.1	10.3	10.4	10.4	10	9.9	9.9	10	9.9	9.7	9.8	9.7	9.9	9.9	11.3	11.6		
L1																															

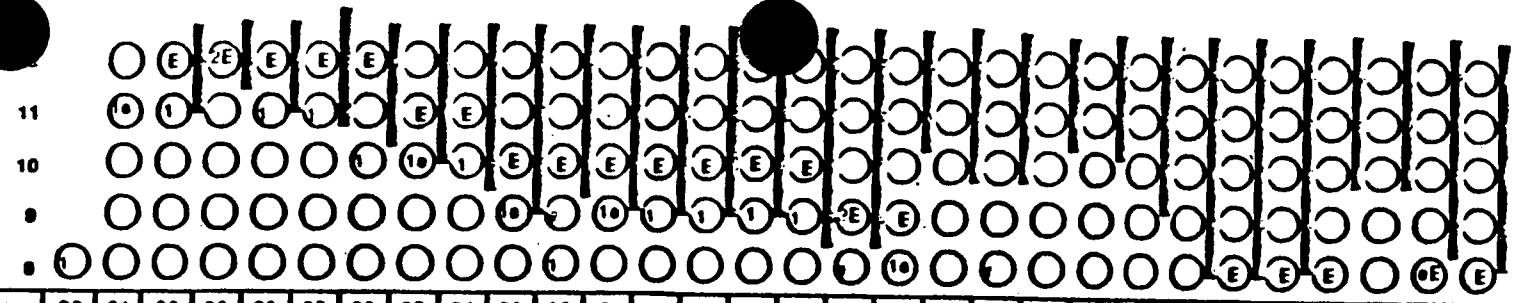
NOTES
Projections are based AVB centerline relative to tube centerline.

Plugged Tube

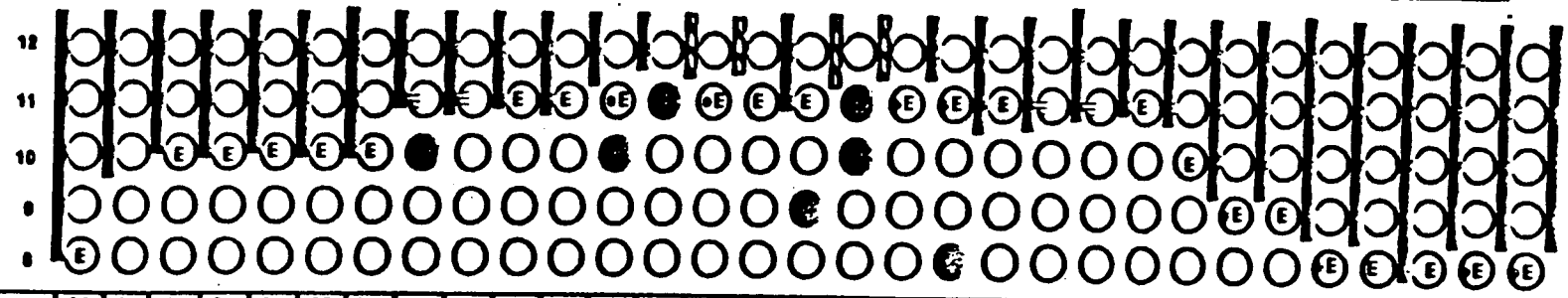
11/24/87
bol

FIGURE 3

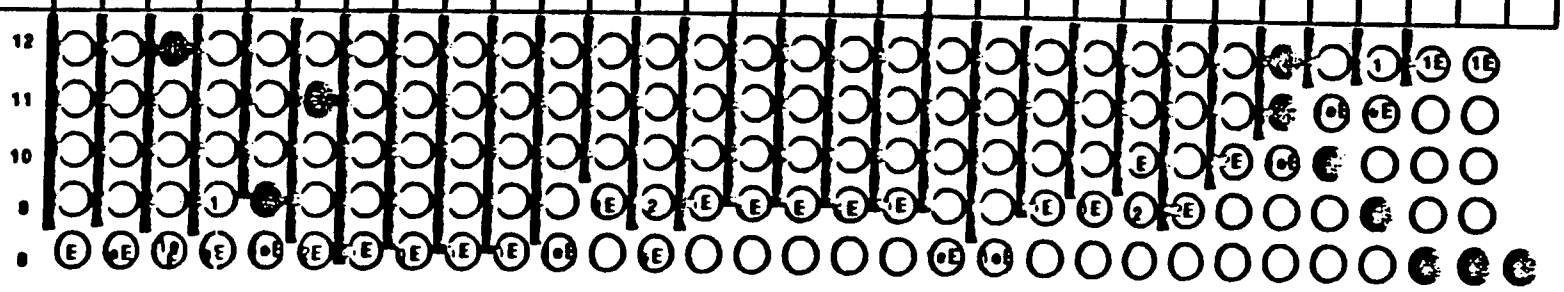
AVB Insertion



Column	92	91	90	89	88	87	86	85	84	83	82	81	80	79	78	77	76	75	74	73	72	71	70	69	68	67	66	65	64	63
Projection								11.1		10.5	10.9	11.7					10.4	10.1	10.2	9.9	9.8	10.5	10.3	9.3				9.7	10.6	
L1	2.0	1.0	3.9		3.7	4.5	1.2	3.5	4.6	2.8	0.9	2.5	2.4	3.0	3.3	4.8		2.4												



Column	62	61	60	59	58	57	56	55	54	53	52	51	50	49	48	47	46	45	44	43	42	41	40	39	38	37	36	35	34	33	32
Projection		9.5								11.		12	12						10.4	10.7					11.7		9.8			10.7	11
L1																															



Column	31	30	29	28	27	26	25	24	23	22	21	20	19	18	17	16	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1'
Projection		10.7	10.2	11.1	11.3						10.7		10.4						10.6	11						11.5	11.7		1.6		
L1			1.0	2.7	1.7		2.8	2.2	3.2	3.2	2.0	1.9		1.5	2.0	2.5	3.2	3.1	1.1	1.6	4.0	2.0				1.9	1.2	2.3	1.6	1.4	

NOTES:

- e - EPRI calls of AVB contacts ≤ 2.0 inches
- E - EPRI call of last tube with observable AVB indication
- Numbers in tubes refer to Westinghouse interpretation of AVBs visible
- L1 - length of single AVB contacts read by Westinghouse
- Projections by Westinghouse
- Indications in Columns 2 and 92 are retainer rings

Plugged Tube

11/11/87
bel

FIGURE 4

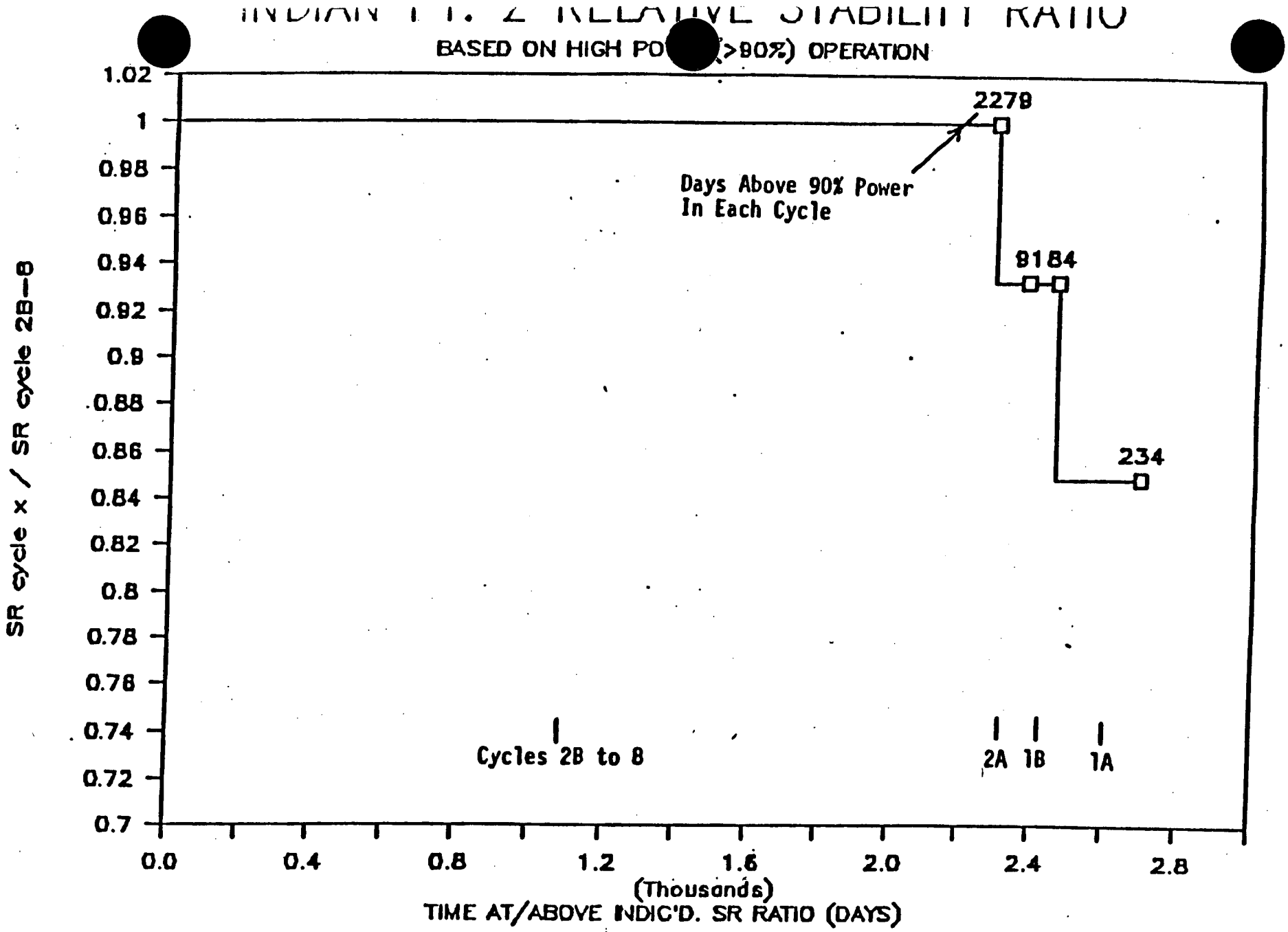

INDIAN PT. 2 RELATIVE STABILITY RATIO
BASED ON HIGH POWER (>90%) OPERATION
SR cycle x / SR cycle 2B-8
1.02
1
0.98
0.96
0.94
0.92
0.9
0.88
0.86
0.84
0.82
0.8
0.78
0.76
0.74
0.72
0.7
2278
Days Above 90% Power
In Each Cycle
9184
234
Cycles 2B to 8
2A
1B
1A
0.0
0.4
0.8
1.2
1.6
2.0
2.4
2.8
(Thousands)
TIME AT/ABOVE INDIC'D. SR RATIO (DAYS)

FIGURE 5

Attachment IV

Con Edison Letter to NRC
Dated November 23, 1987

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March 25, 1988

Murray Selman
Vice President

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, NY 10511
Telephone (914) 737-8116

November 23, 1987

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Susceptibility of Indian Point Unit No. 2 (IP-2) Steam Generators to High Cycle Fatigue Tube Failure

The purpose of this letter is to transmit information and material presented during a meeting on November 12, 1987 between NRC staff, Con Edison and representatives from Westinghouse. At that meeting we presented the results of the investigation and evaluations we have undertaken to determine whether the Indian Point Unit No. 2 (IP-2) steam generators may be susceptible to a tube failure similar to the one that occurred at North Anna-1 on July 15, 1987. Upon receiving notification from Westinghouse that IP-2, based upon a very preliminary review, may be susceptible to the North Anna type generator tube failure mechanism (i.e. high cycle fatigue), we initiated several plant-specific analyses with Westinghouse to more accurately determine which tubes, if any, might be susceptible. Thermal-hydraulic, fatigue and vibration analyses that incorporated an accurate determination of anti-vibration bars (AVB) insertion depths were performed for IP-2 steam generators Nos. 22 and 24. The results of these analyses were discussed at the November 12, 1987 meeting and a report is provided in Attachment A. Based on these analyses, we have concluded that steam generators 22 and 24 are not susceptible to tube fatigue rupture at the top support plate similar to that which occurred at North Anna 1, and that accordingly no modification or preventative tube plugging is required.

Similar analyses for steam generator Nos. 21 and 23 are ongoing and will be completed after AVB insertion depths are verified for these generators by analyzing eddy current data. Therefore, the eddy current testing program planned during the current refueling and maintenance outage has been expanded to allow us to determine the AVB location in steam generators 21 and 23. The final results of our evaluation of steam generators 21 and 23 will be provided to the NRC by a separate letter following the conclusion of the testing program and analyses, along with a description of any modifications and corrective actions, which may be indicated by our conclusions.

We have also assessed our primary to secondary leak detection capability to assure that IP-2 systems and procedures are capable of producing leak rate

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data that would allow for positive detection and timely corrective actions before any fatigue-induced gross tube failure were to occur. A description of our overall capability and the administratively imposed leak rate limits, together, with their associated corrective actions, are provided in Attachment B.

Additionally, Attachment C provides our updated response to fifteen (15) questions requested by the NRC Region I office.

Based upon the foregoing inquiries and actions undertaken to date, we are unaware of any information which would suggest that the steam generators of IP-2 are particularly susceptible to the tube failure mechanism experienced at North Anna 1.

Please contact us if you have any questions on this matter.

Very truly yours,

Murray Selman

Attachments

cc: Ms. Marylee M. Slosson
U.S. Nuclear Regulatory Commission
Project Directorate I-1
Division of Reactor Projects I/II
Washington, DC 20555

Mr. William Russell
Regional Administrator - Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, NY 10511

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		Steam Generator 24	
Row	Column	Row	Column
12	2	12	2
11	3	11	3
11	37	11	51
10	24	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.)

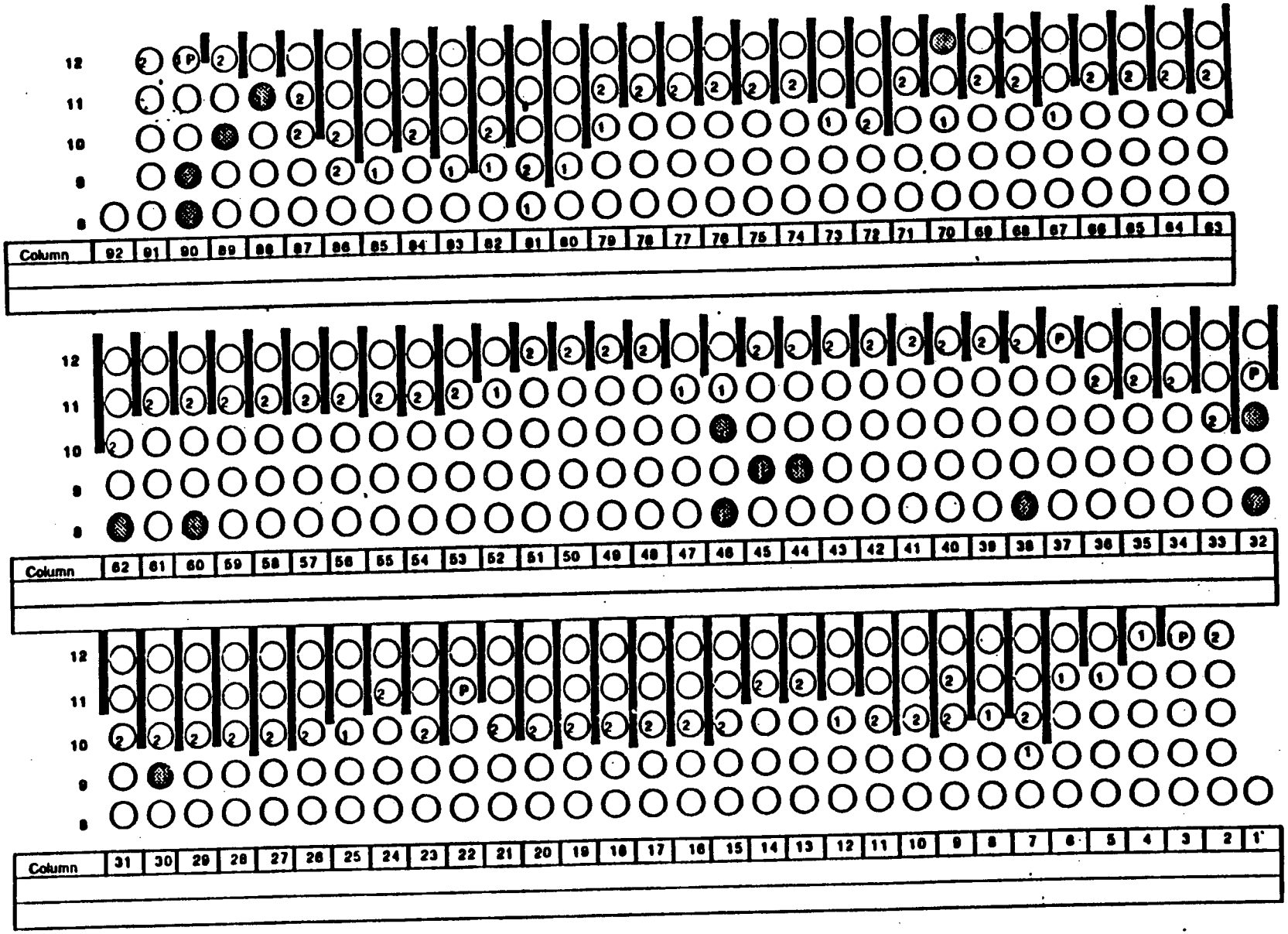
Steam Generator 22			Steam Generator 24		
Tube	Relative Stability Ratio	Stress Ratio	Tube	Relative Stability Ratio	Stress Ratio
R12C2	0.79	0.35	R12C2	0.79	0.35
R11C3	0.84	0.55	R11C3	0.84	0.55
R11C37	0.73	0.25	R11C51	0.74	0.27
R10C24	0.68	0.19	R11C46	0.84	0.58
			R10C71	0.73	0.29

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

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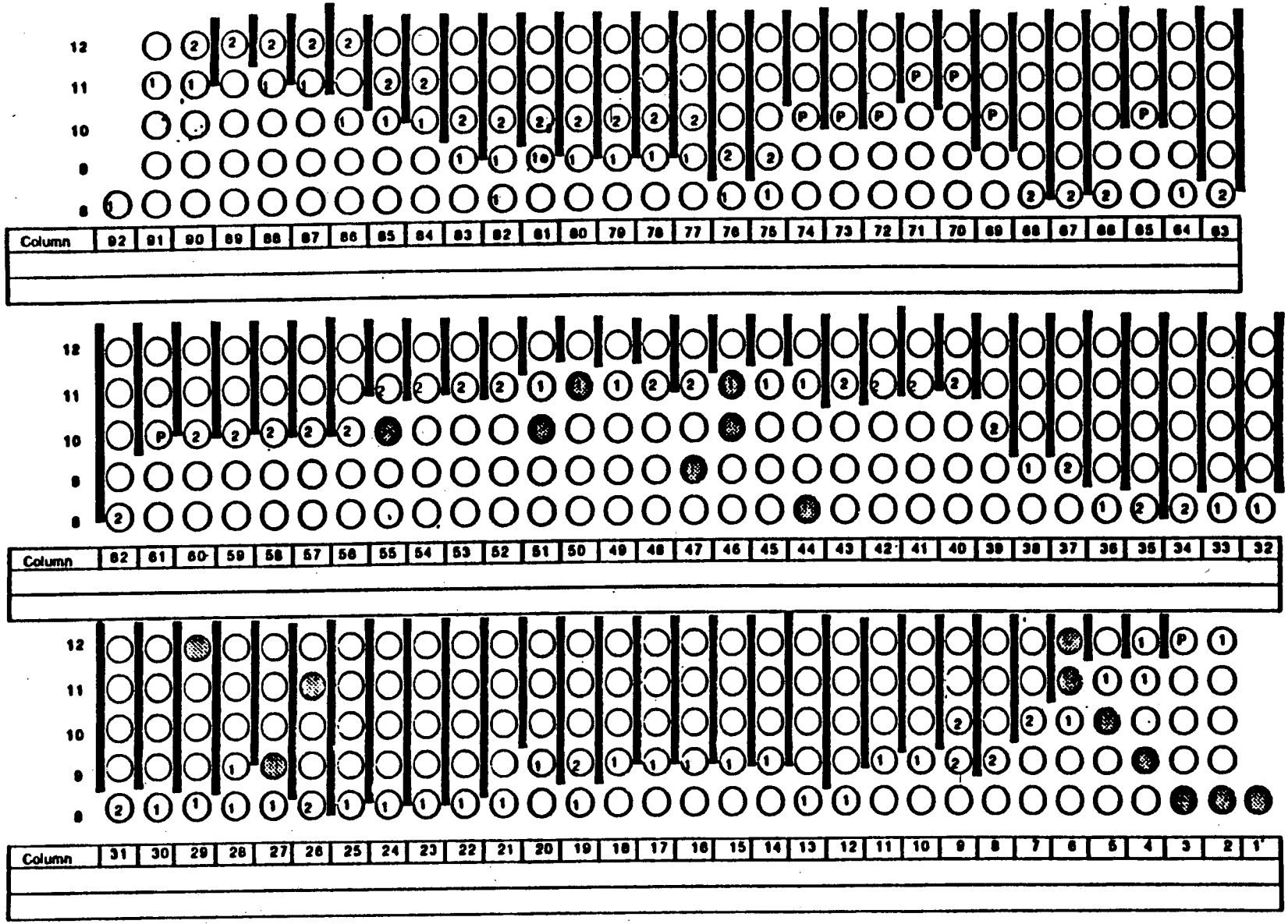
bcl

FIGURE 1

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

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FIGURE 2

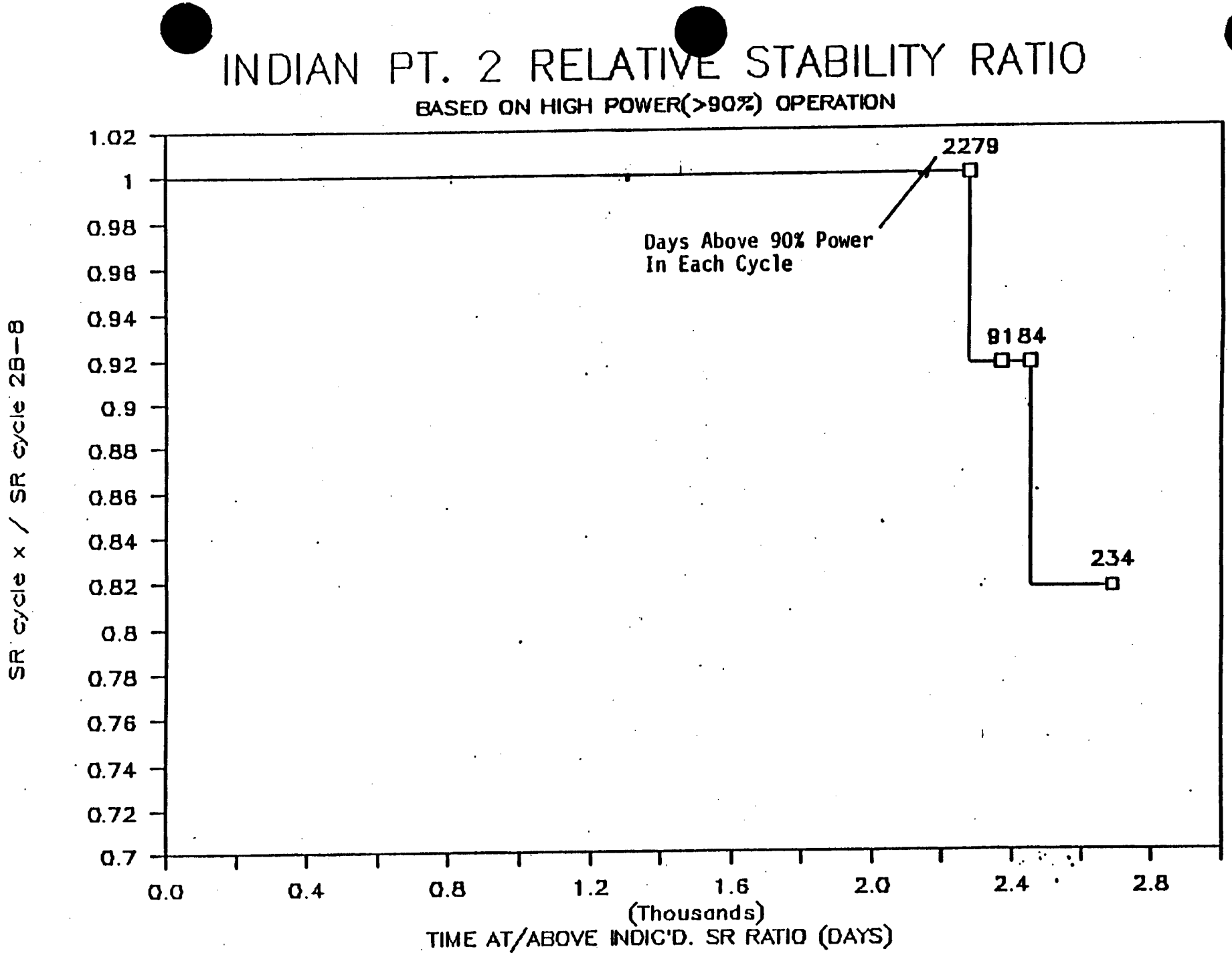

INDIAN PT. 2 RELATIVE STABILITY RATIO
BASED ON HIGH POWER(>90%) OPERATION
SR cycle x / SR cycle 2B-8
1.02
1
0.98
0.96
0.94
0.92
0.9
0.88
0.86
0.84
0.82
0.8
0.78
0.76
0.74
0.72
0.7
0.0
0.4
0.8
1.2
1.6
2.0
2.4
2.8
(Thousands)
TIME AT/ABOVE INDIC'D. SR RATIO (DAYS)
2279
Days Above 90% Power
In Each Cycle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

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)

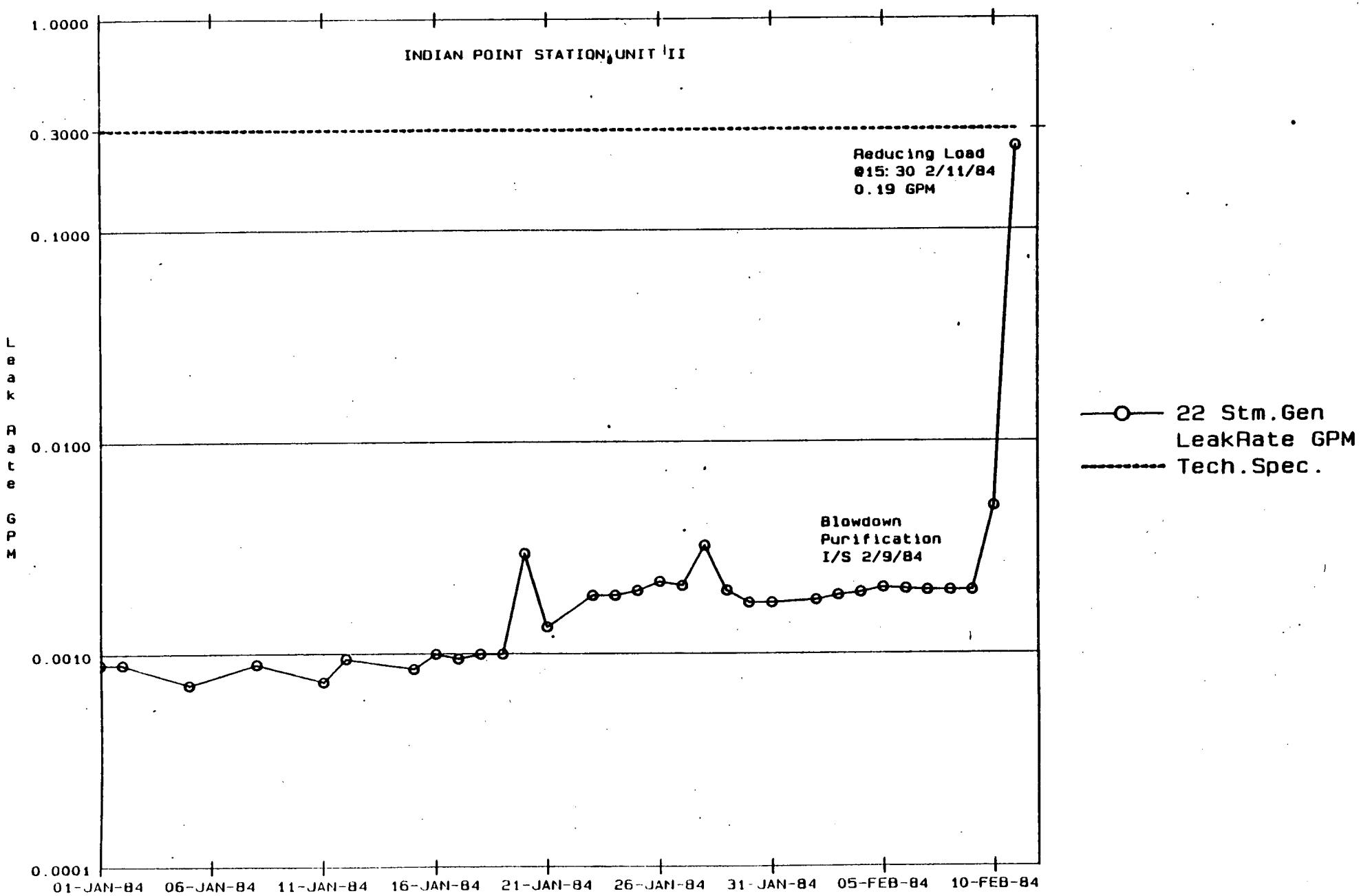


TABLE OF EVENTS

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).

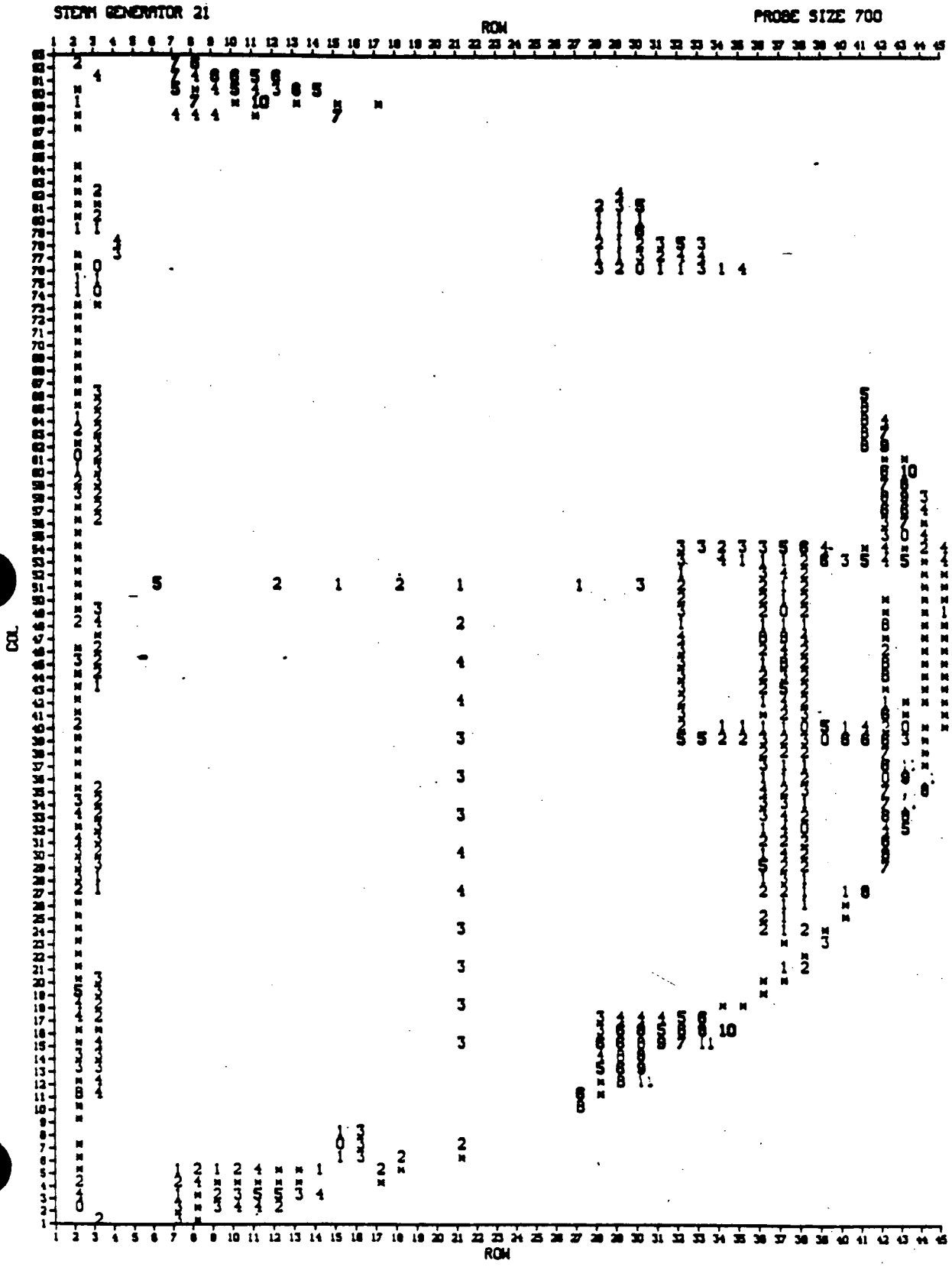
4. Requalification training occurring every two years. SGTR is a required scenario.

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

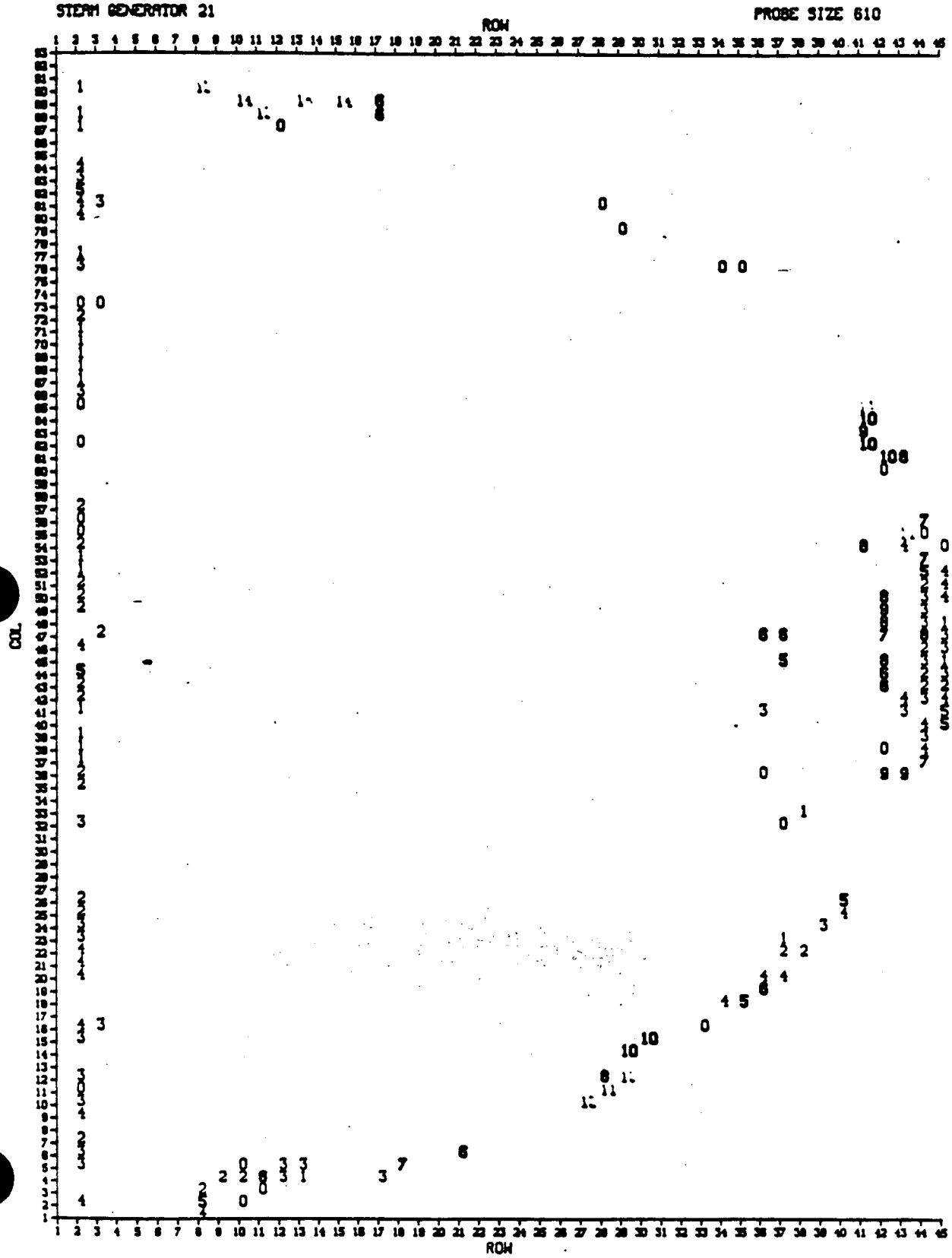
INSPECTION DATE 860221 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 21 PROBE SIZE 700



INSPECTION DATE 860221 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 21 PROBE SIZE 610



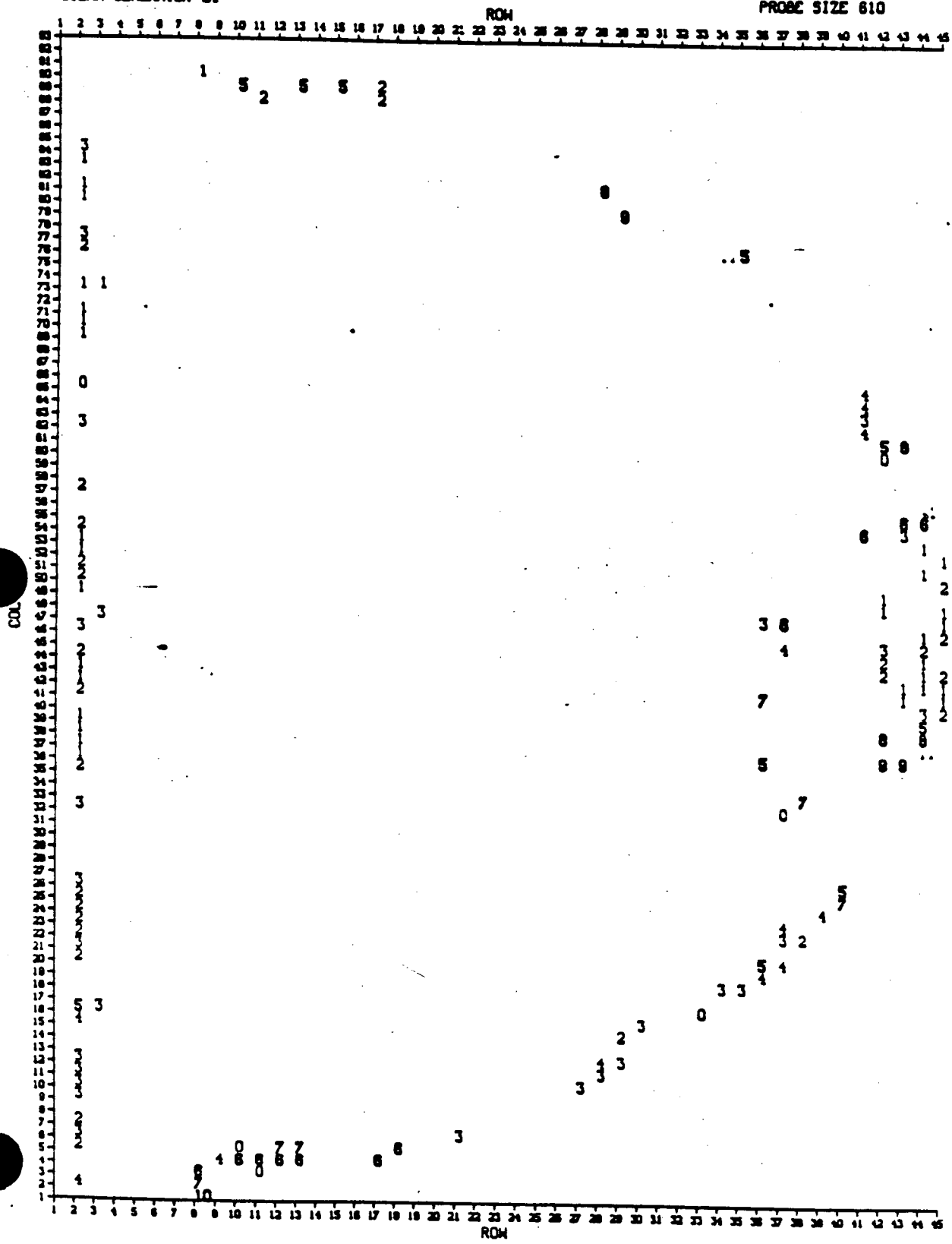
INSPECTION DATE 860221

STEAM GENERATOR 21

COLD LEG
DENTING LAYOUT

SUPPORT PLATE 7

PROBE SIZE 610



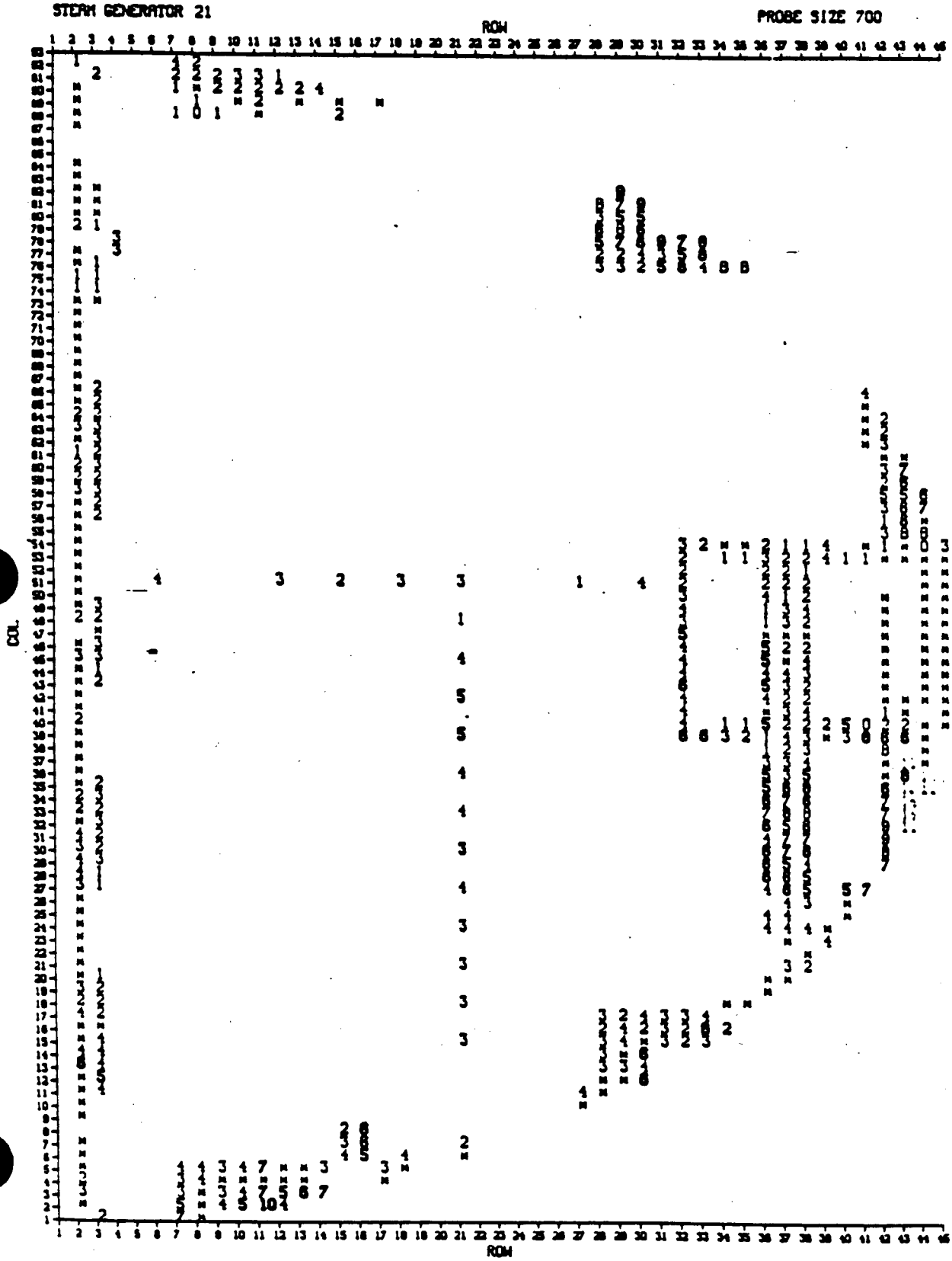
INSPECTION DATE 860221

STEAM GENERATOR 21

COLD LEG
DENTING LAYOUT

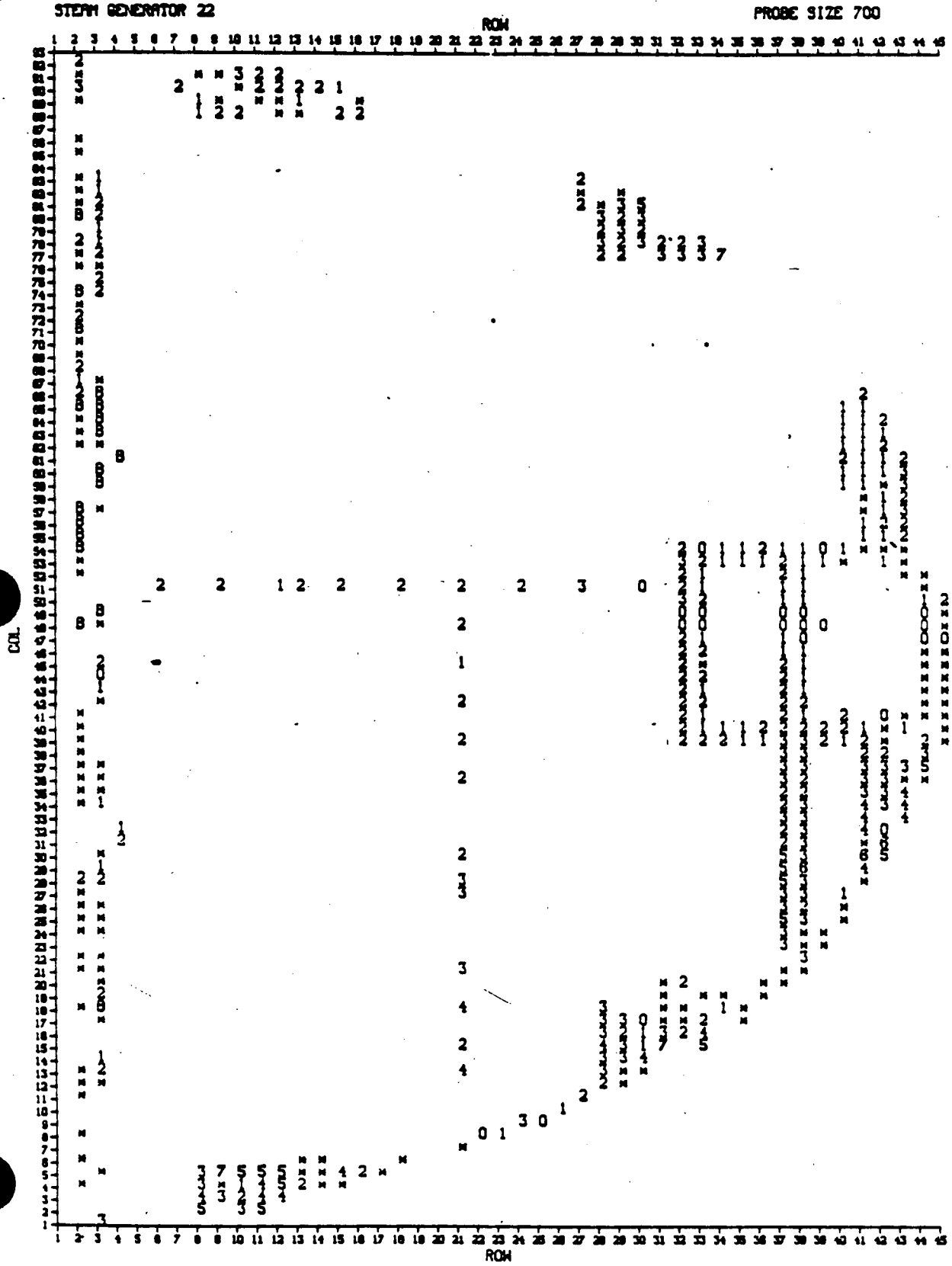
SUPPORT PLATE 7

PROBE SIZE 700



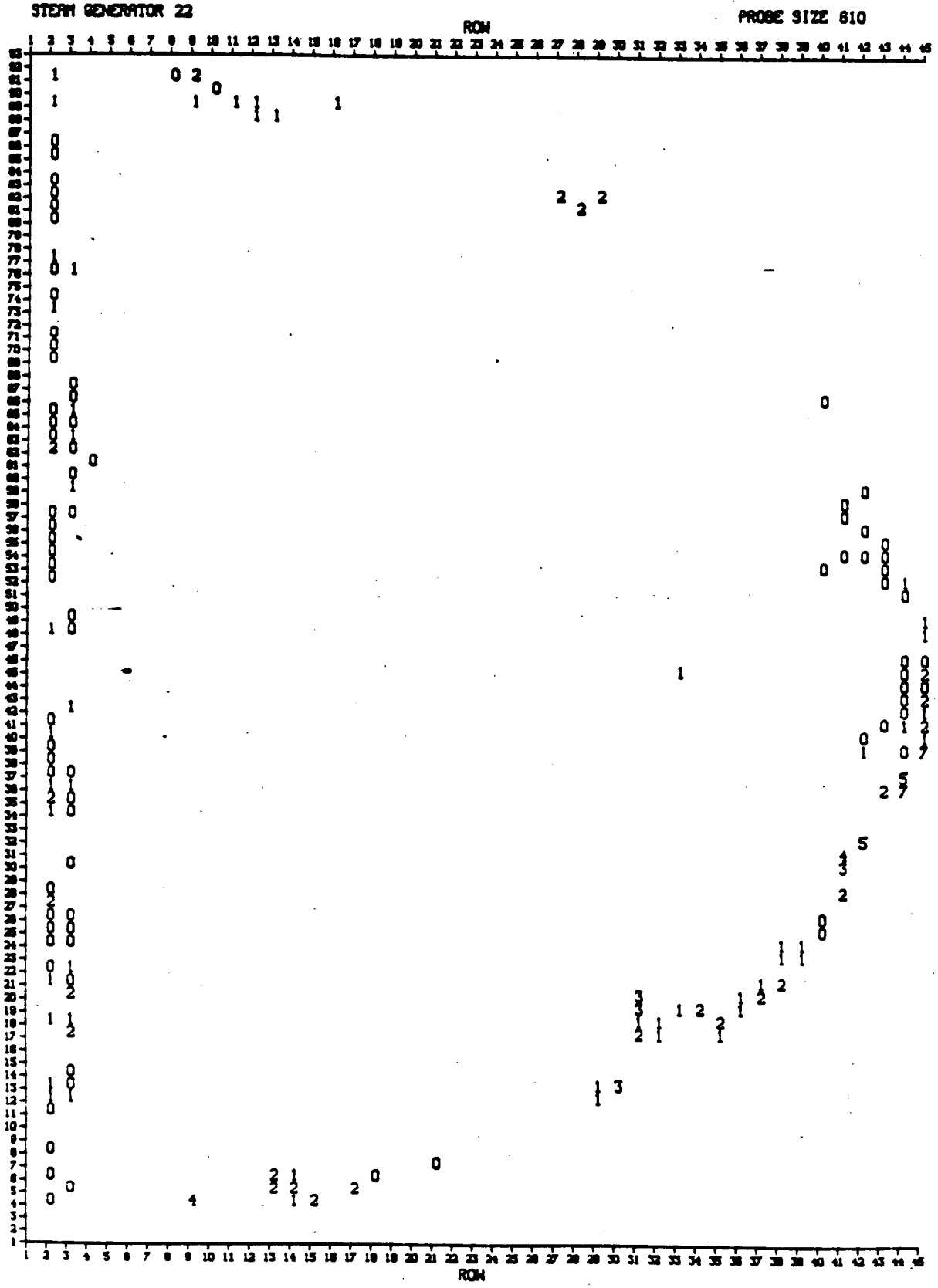
INSPECTION DATE 860222 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 22 PROBE SIZE 700



INSPECTION DATE 860222 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

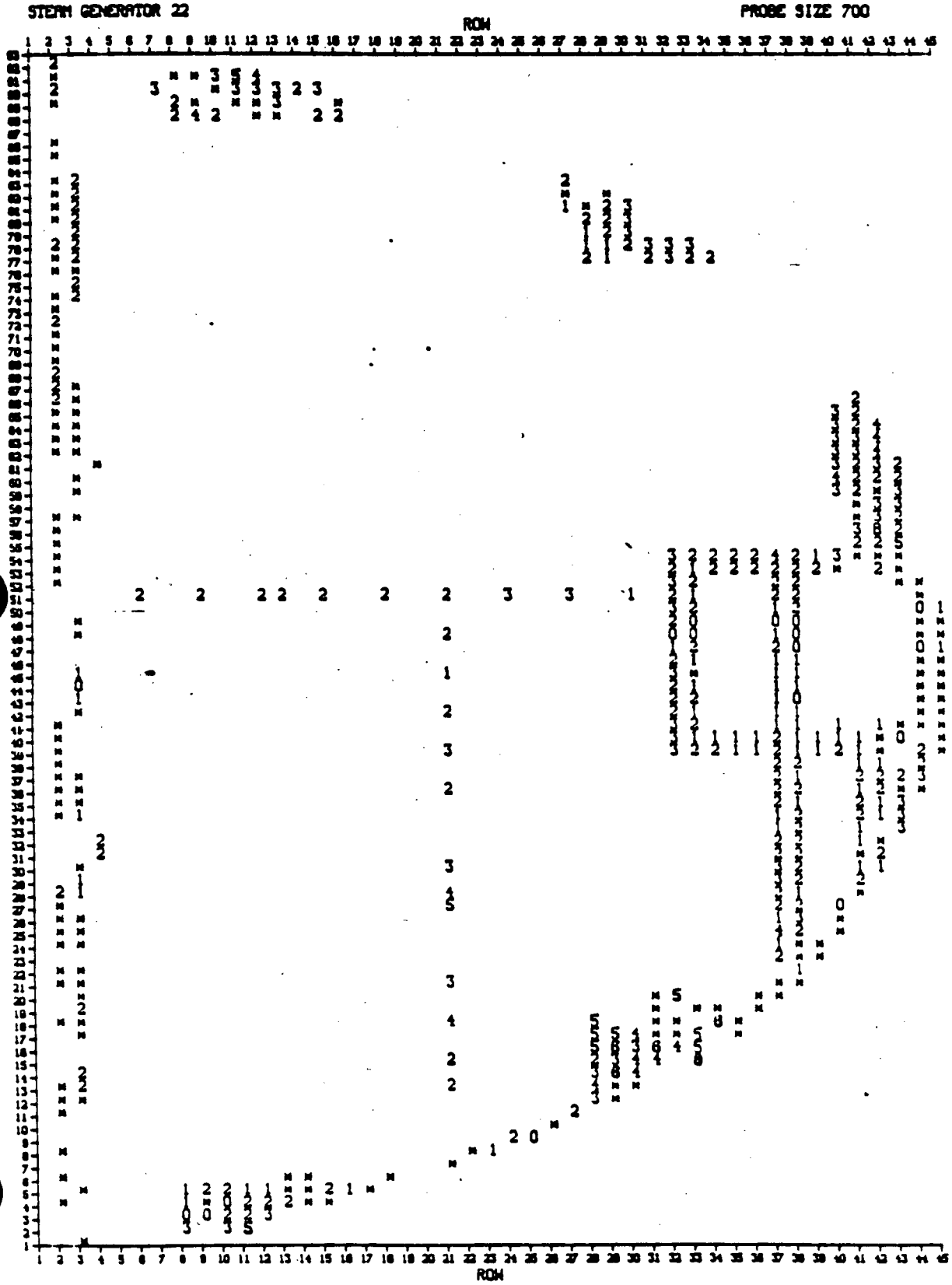
STEAM GENERATOR 22 PROBE SIZE 610



COLD LEG

INSPECTION DATE 880222 | DENTING LAYOUT | SUPPORT PLATE 7

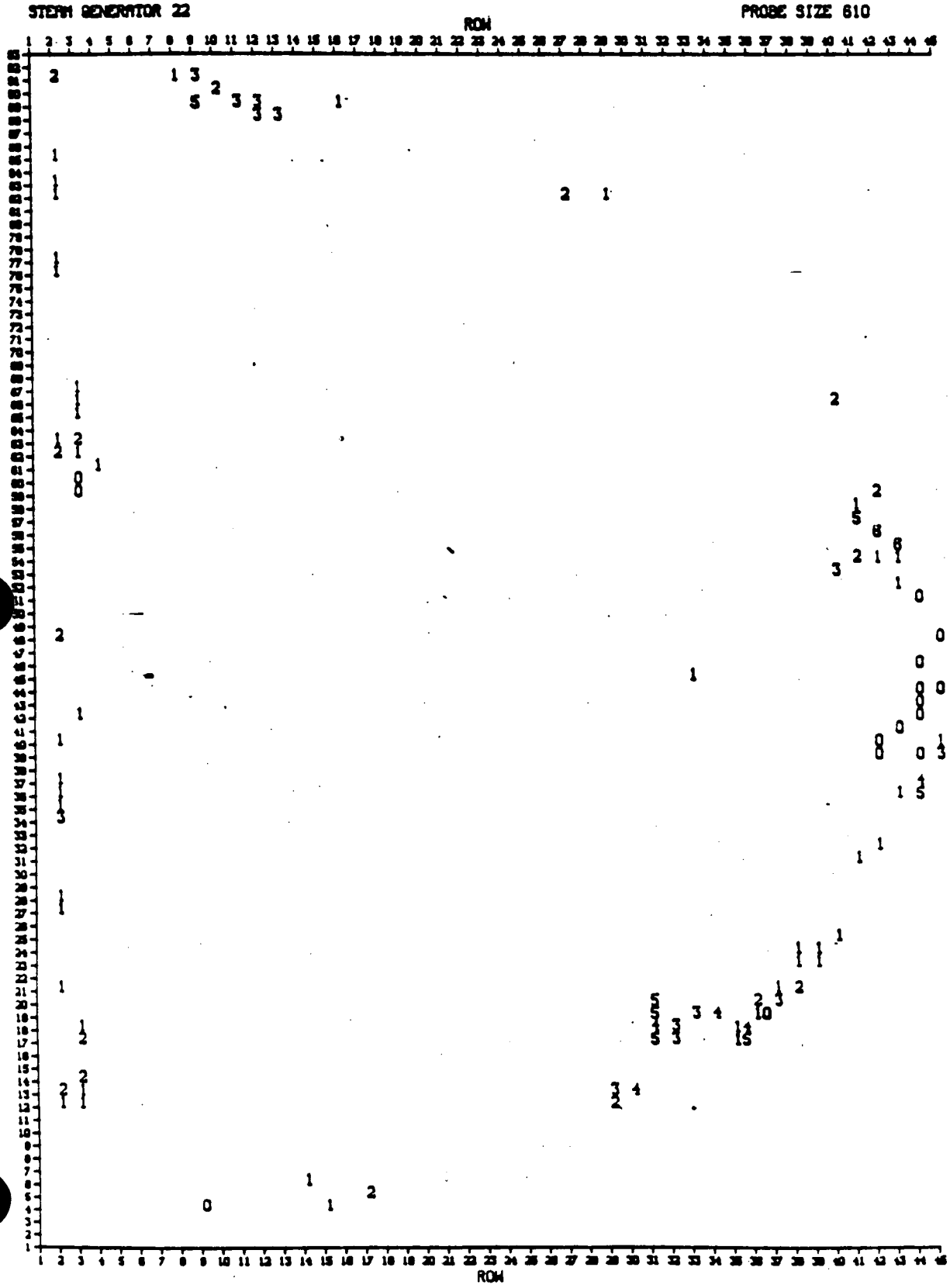
STEAM GENERATOR 22 | PROBE SIZE 700



COLD LEG

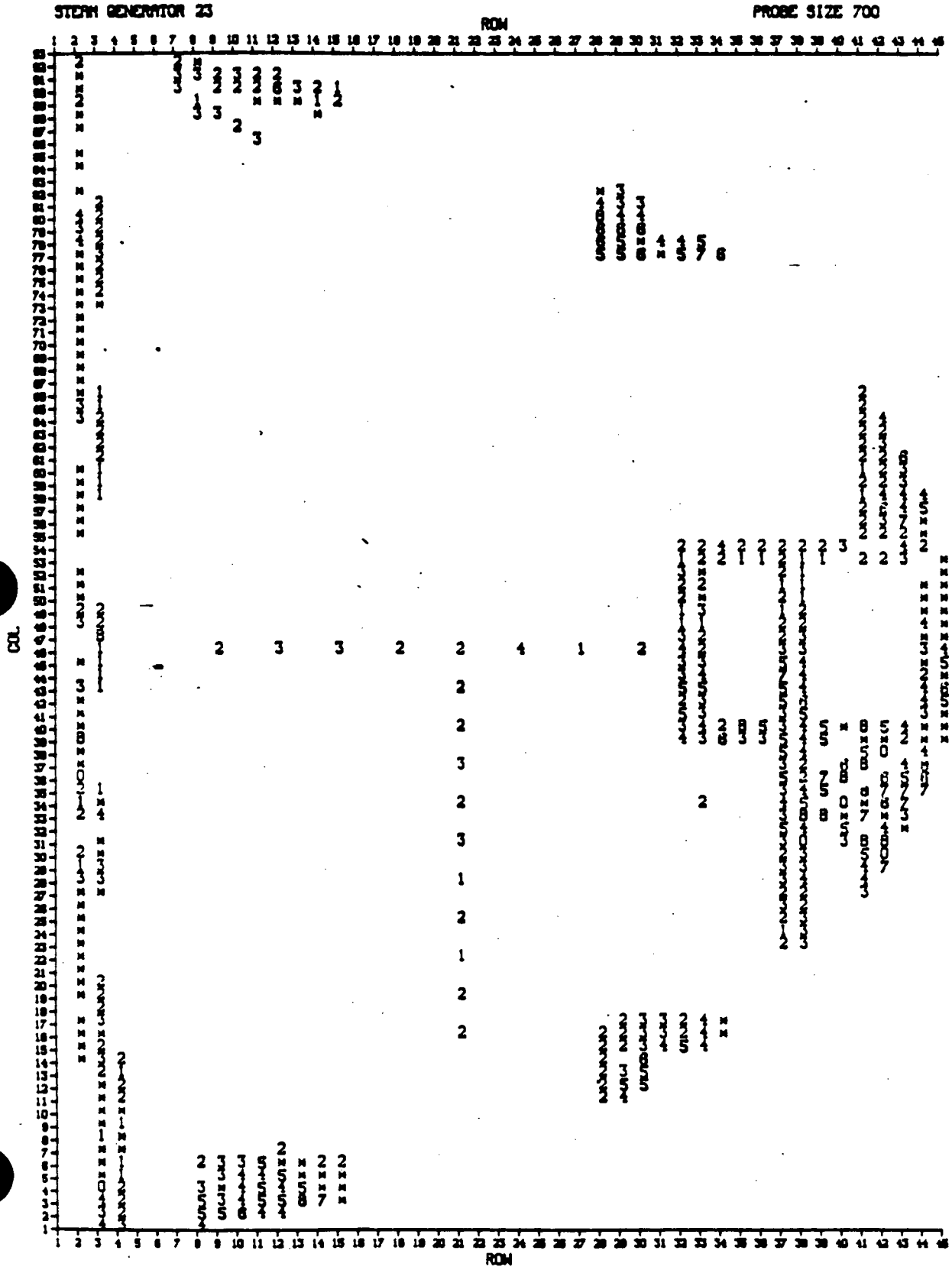
INSPECTION DATE 860222 | DENTING LAYOUT | SUPPORT PLATE 7

STEAM GENERATOR 22 | PROBE SIZE 610



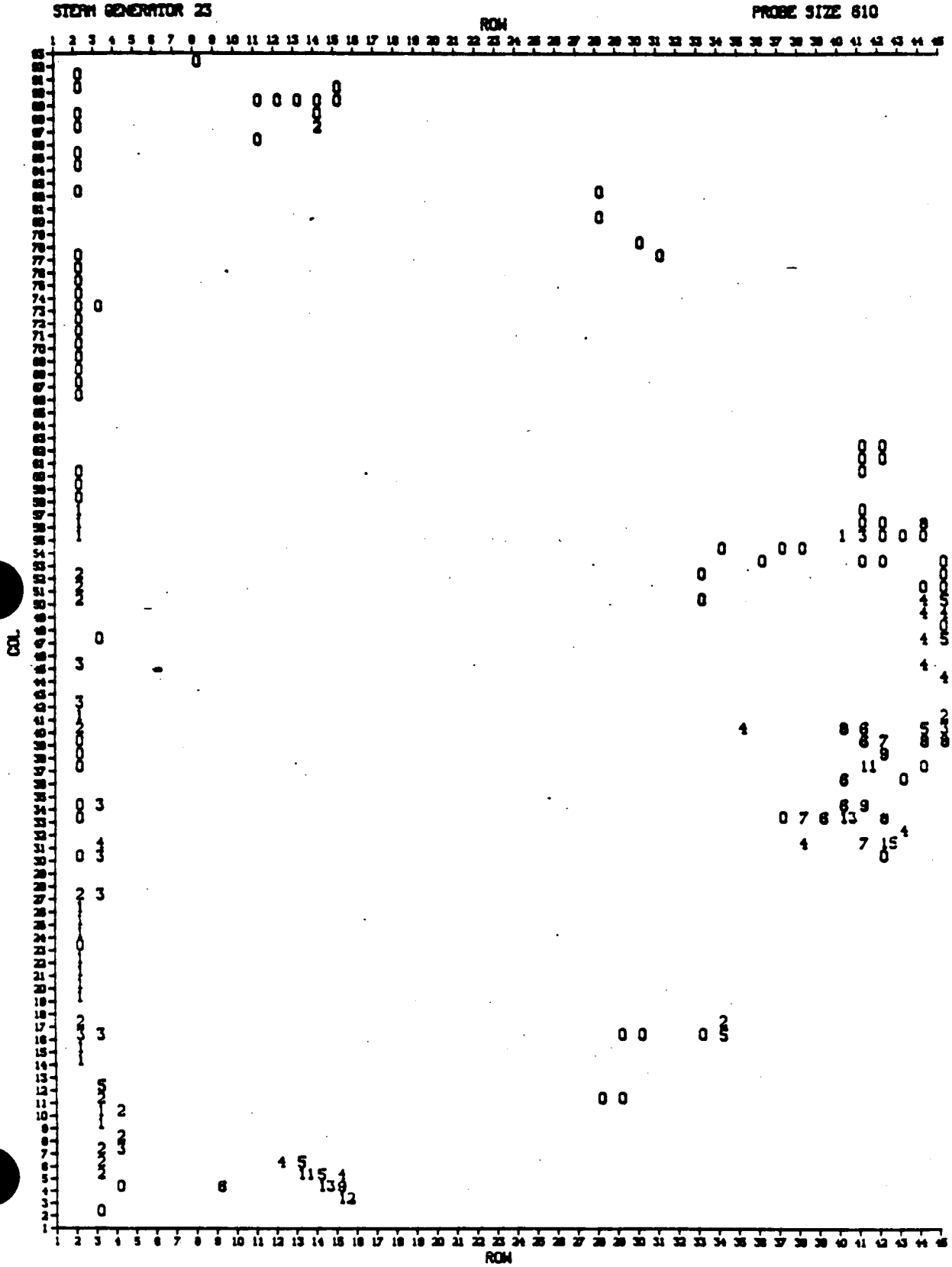
INSPECTION DATE 860223 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 23 PROBE SIZE 700



INSPECTION DATE 860223 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 23 PROBE SIZE 610



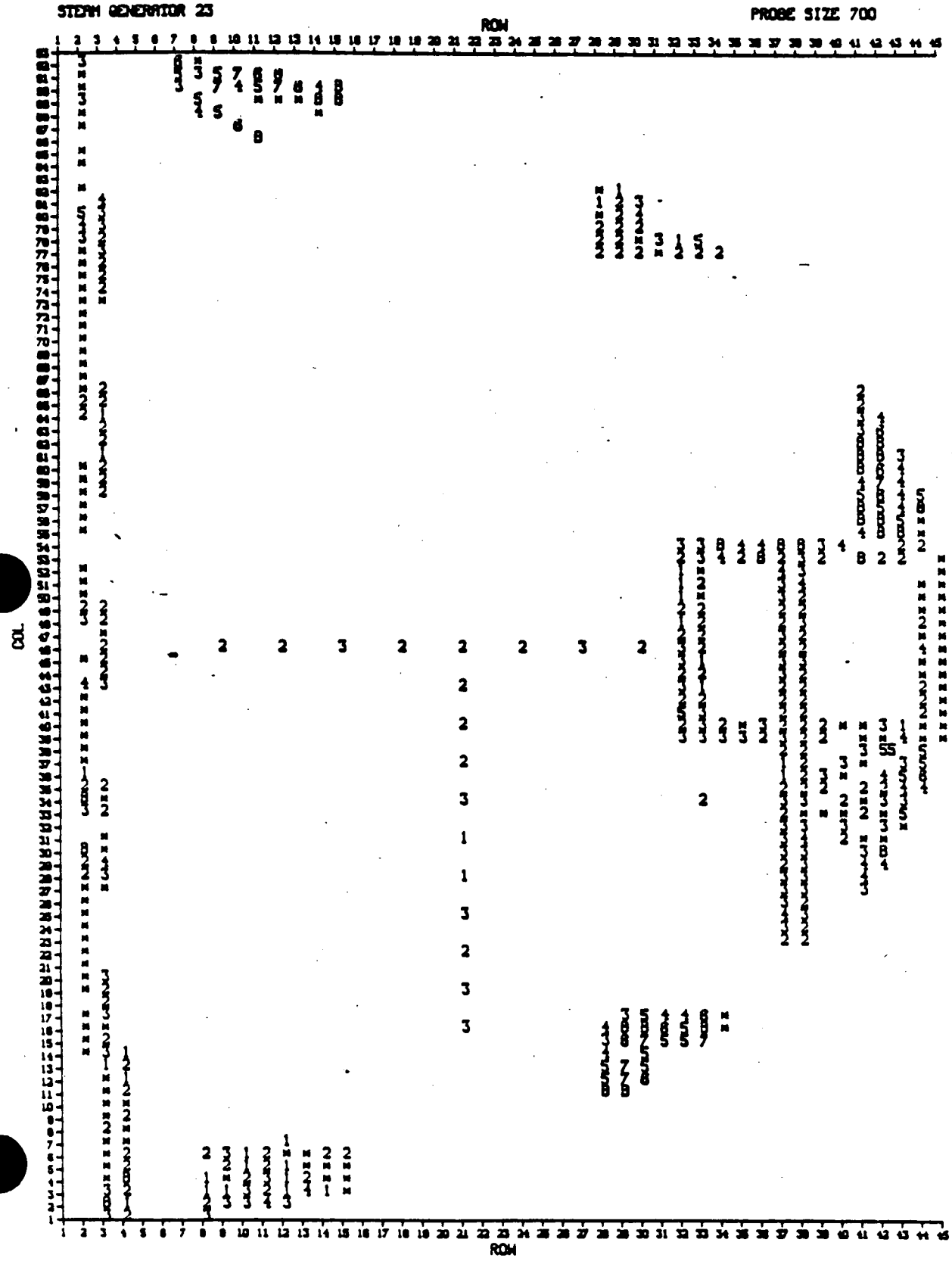
INSPECTION DATE 860223

STEAM GENERATOR 23

COLD LEG
DENTING LAYOUT

SUPPORT PLATE 7

PROBE SIZE 700



COLD LEG

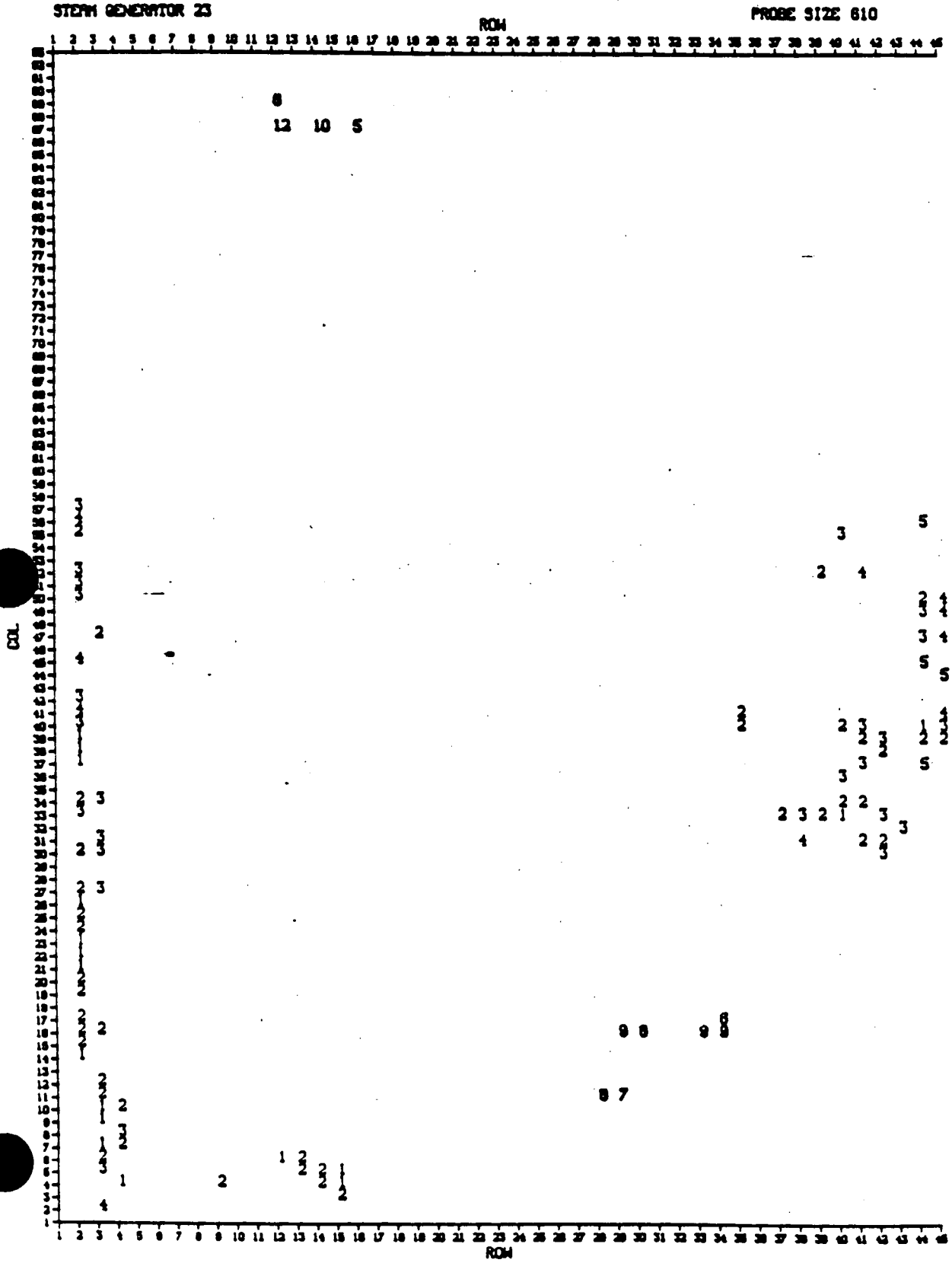
INSPECTION DATE 860223

DENTING LAYOUT

SUPPORT PLATE 7

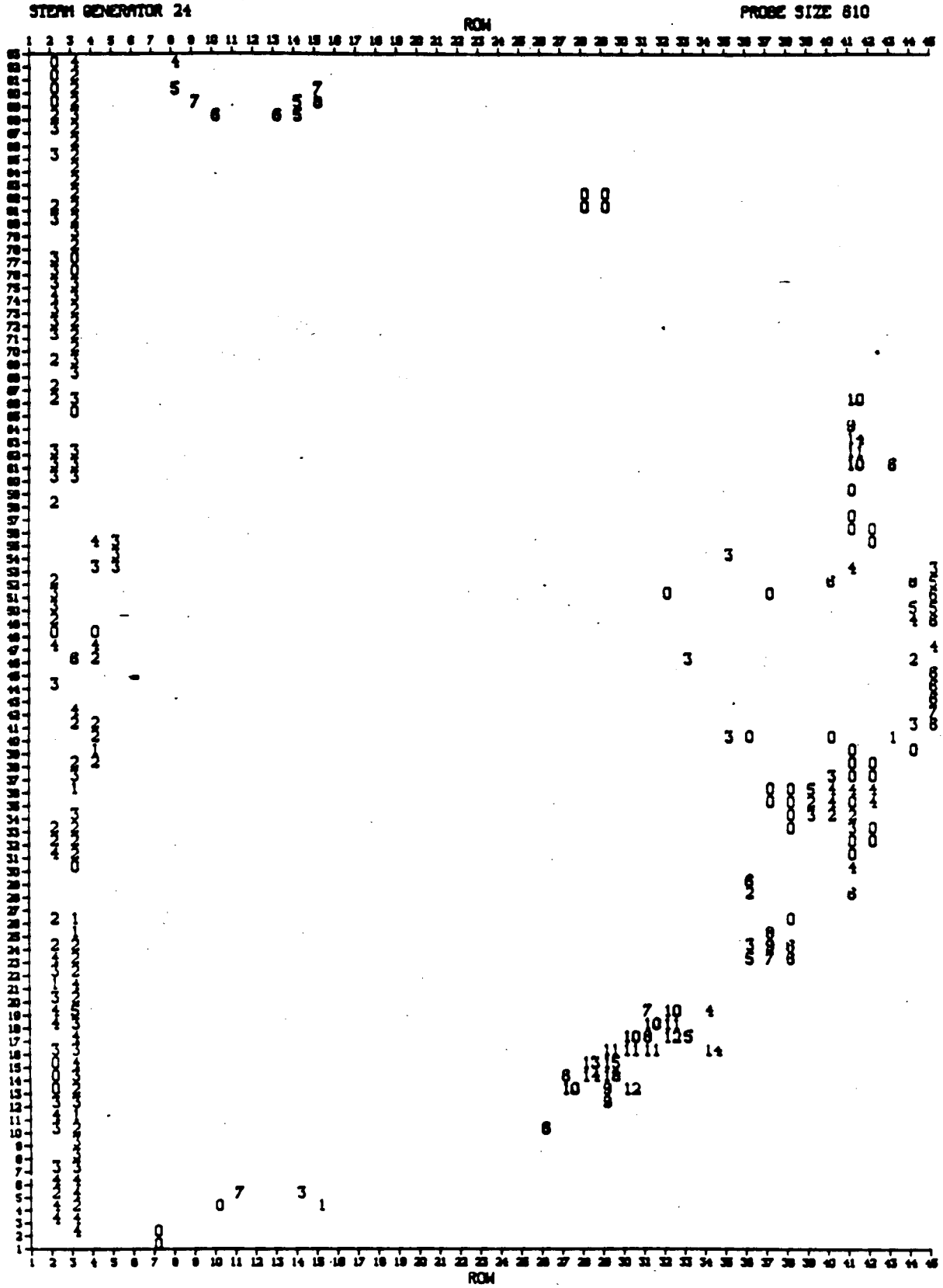
STEAM GENERATOR 23

PROBE SIZE 610



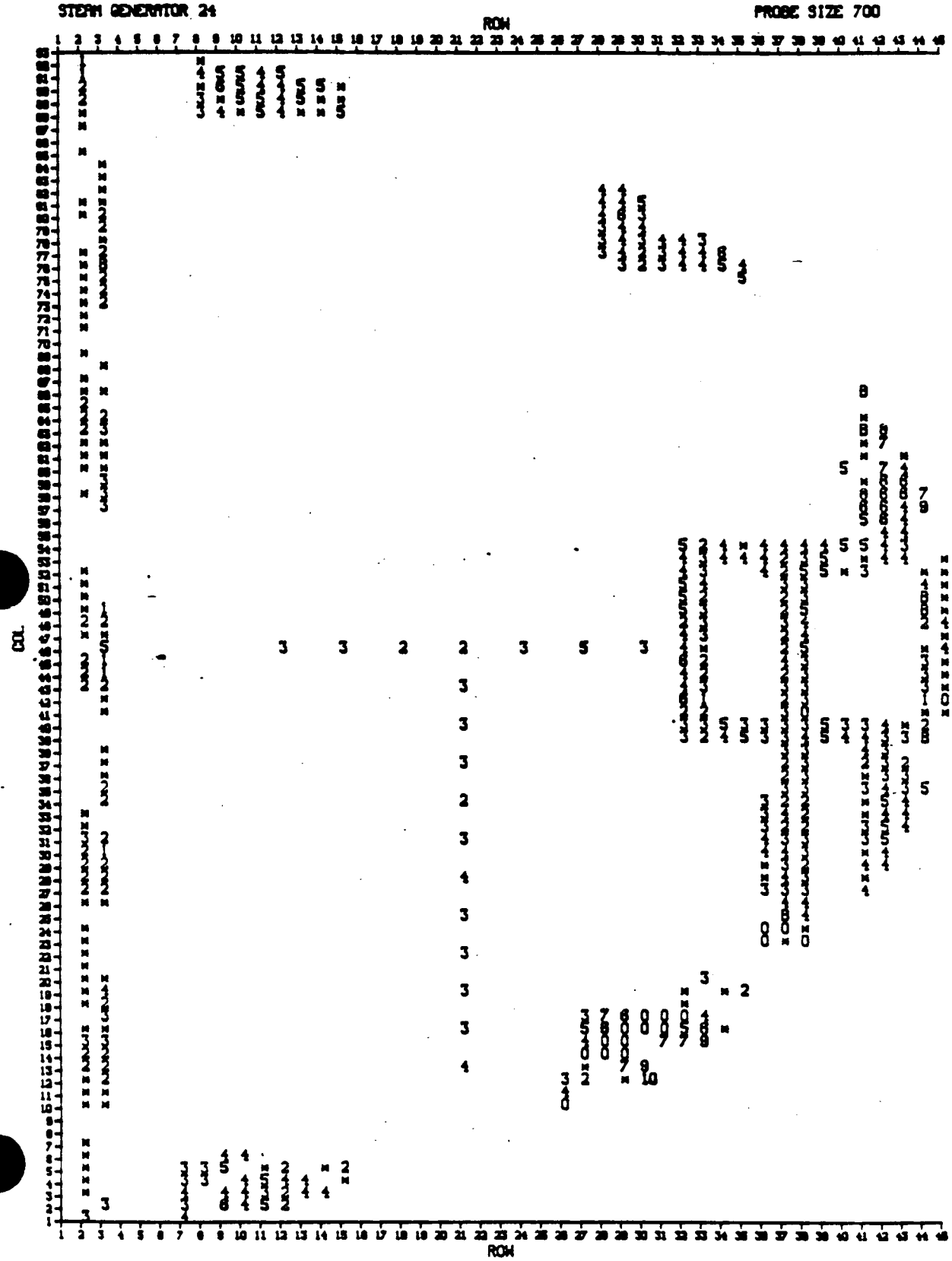
INSPECTION DATE 860224 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 24 PROBE SIZE 610



INSPECTION DATE 860224 HOT LEG DENTING LAYOUT SUPPORT PLATE 6

STEAM GENERATOR 24 PROBE SIZE 700



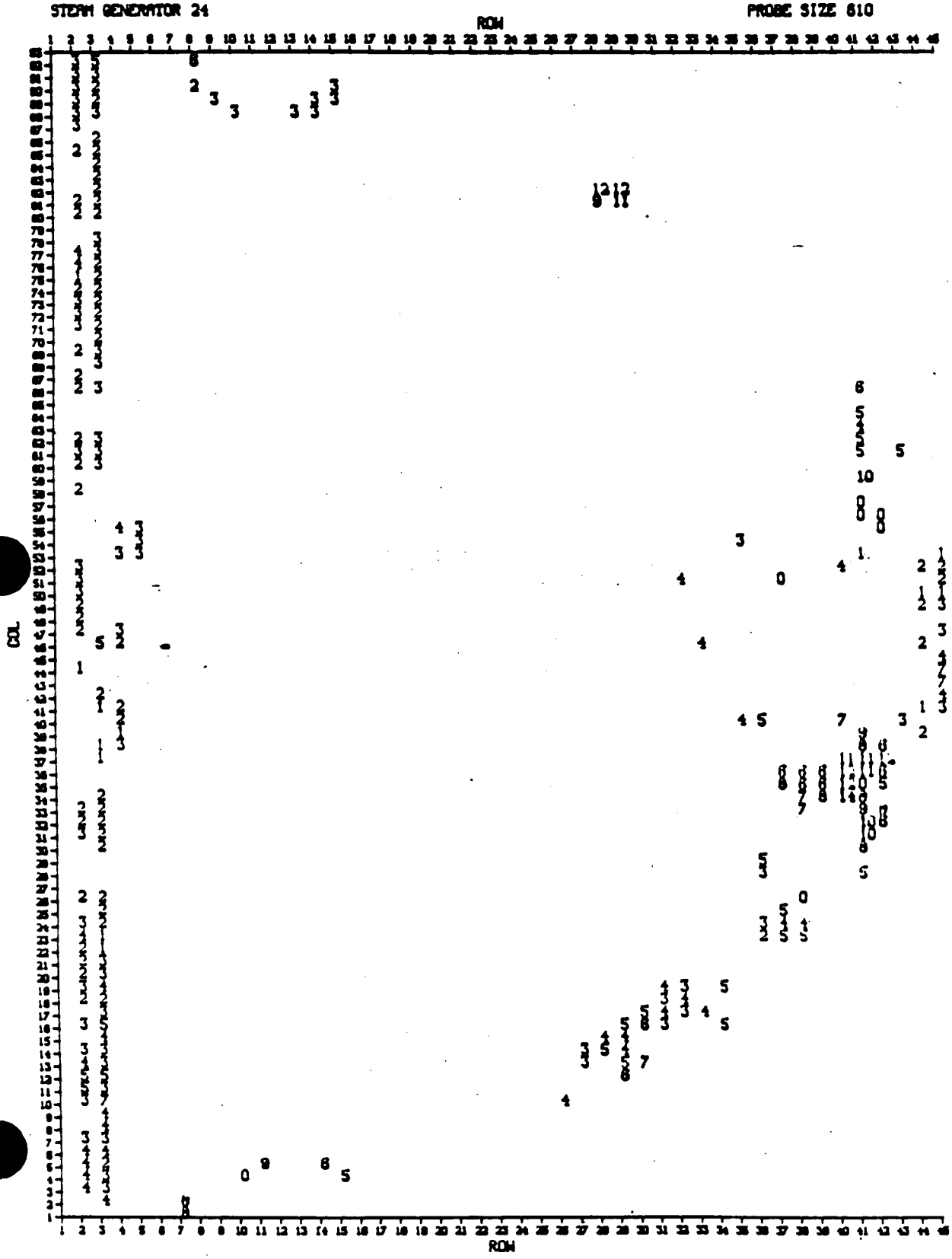
INSPECTION DATE 860224

COLD LEG
DENTING LAYOUT

SUPPORT PLATE 7

STEAM GENERATOR 24

PROBE SIZE 610



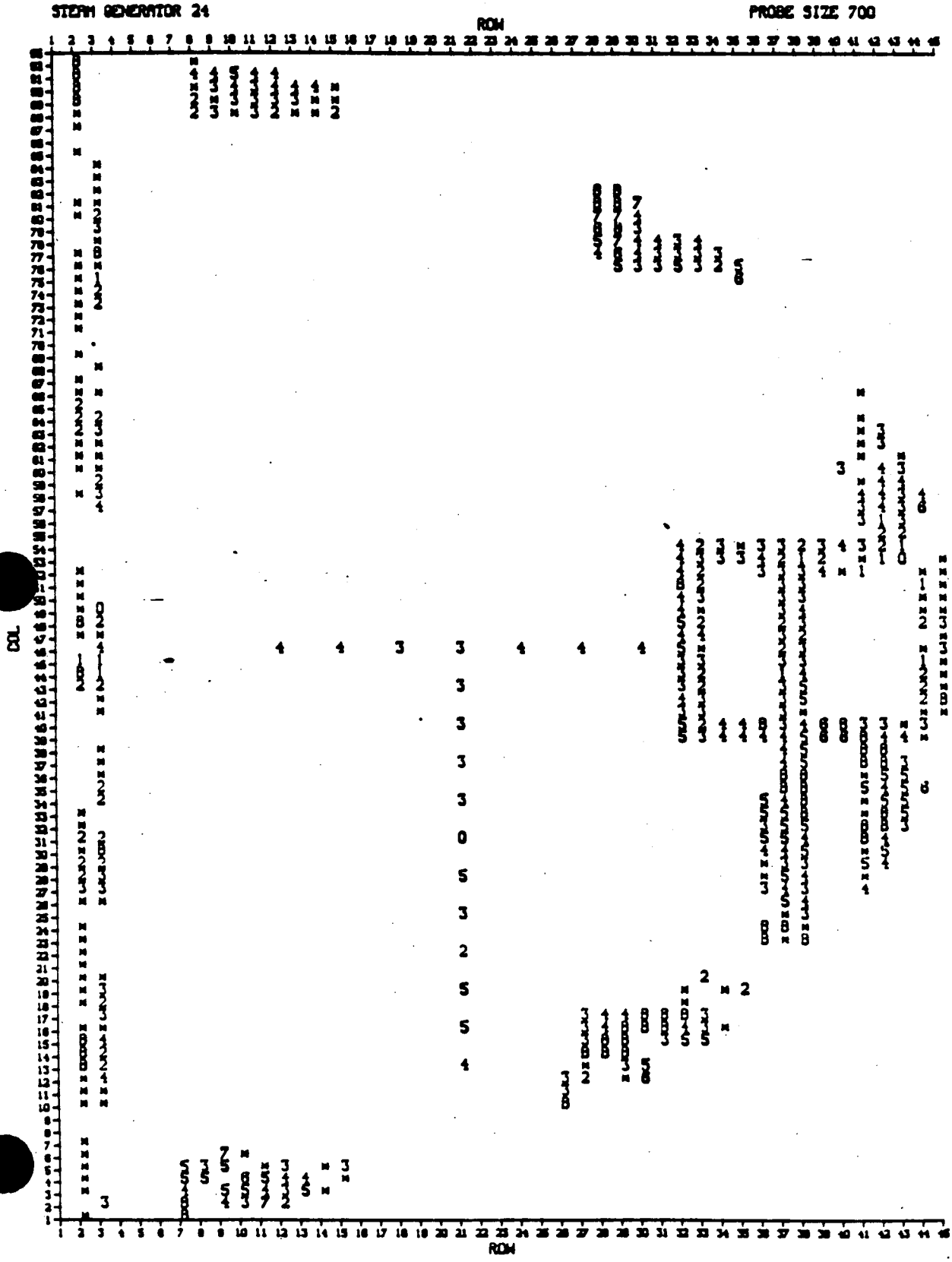
INSPECTION DATE 860224

COLD LEG
DENTING LAYOUT

SUPPORT PLATE 7

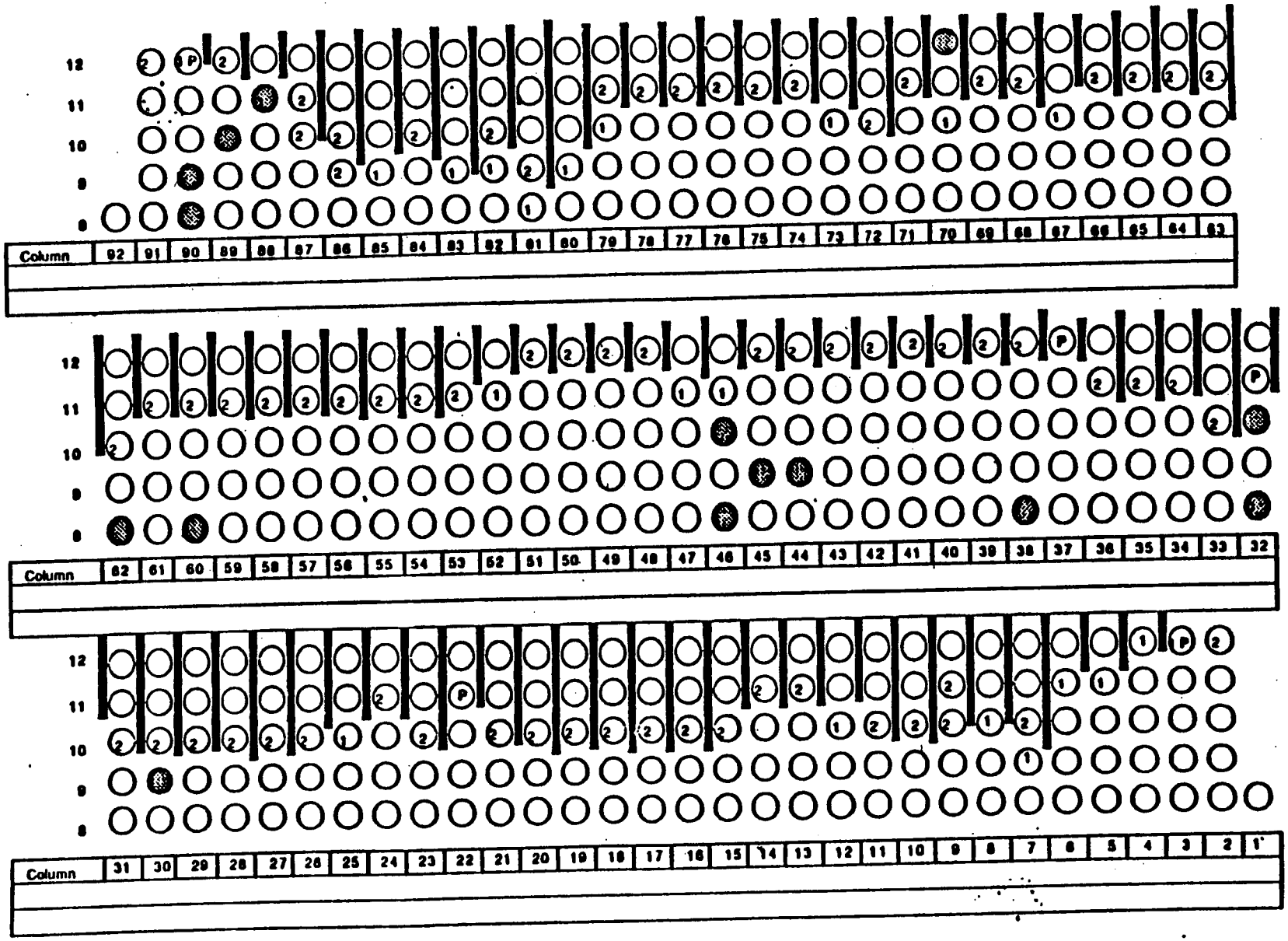
STEAM GENERATOR 24

PROBE SIZE 700



ATTACHMENT 2

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

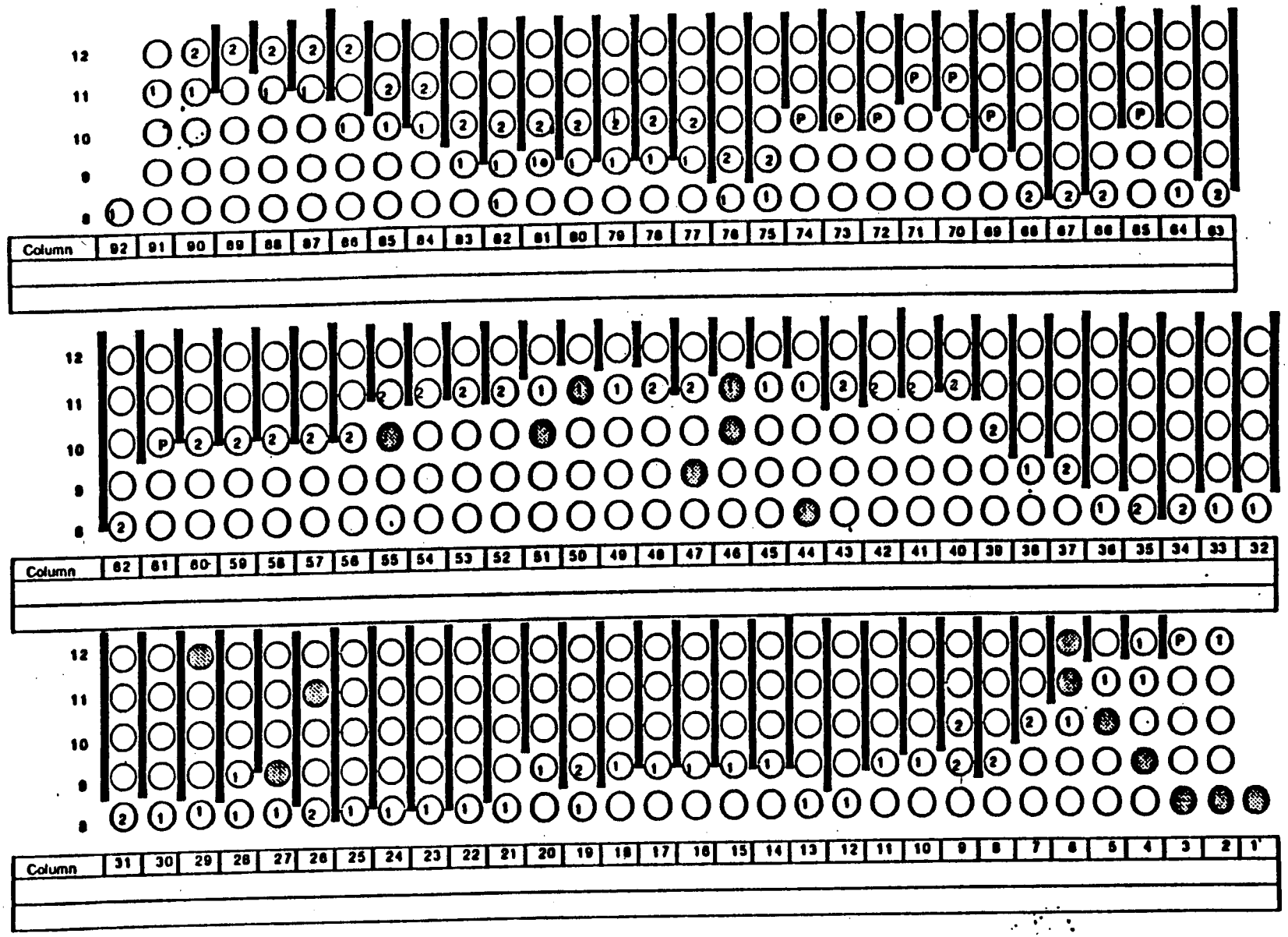


NOTES:
Numbers in tubes refer to AVBs visible
P - AVBs Located by Projection

Plugged Tube

11/11/97
bel

Indi[illegible]Point - 2 : S/G -24
[illegible]/B Insertion



NOTES:

Numbers in tubes refer to AVBs visible

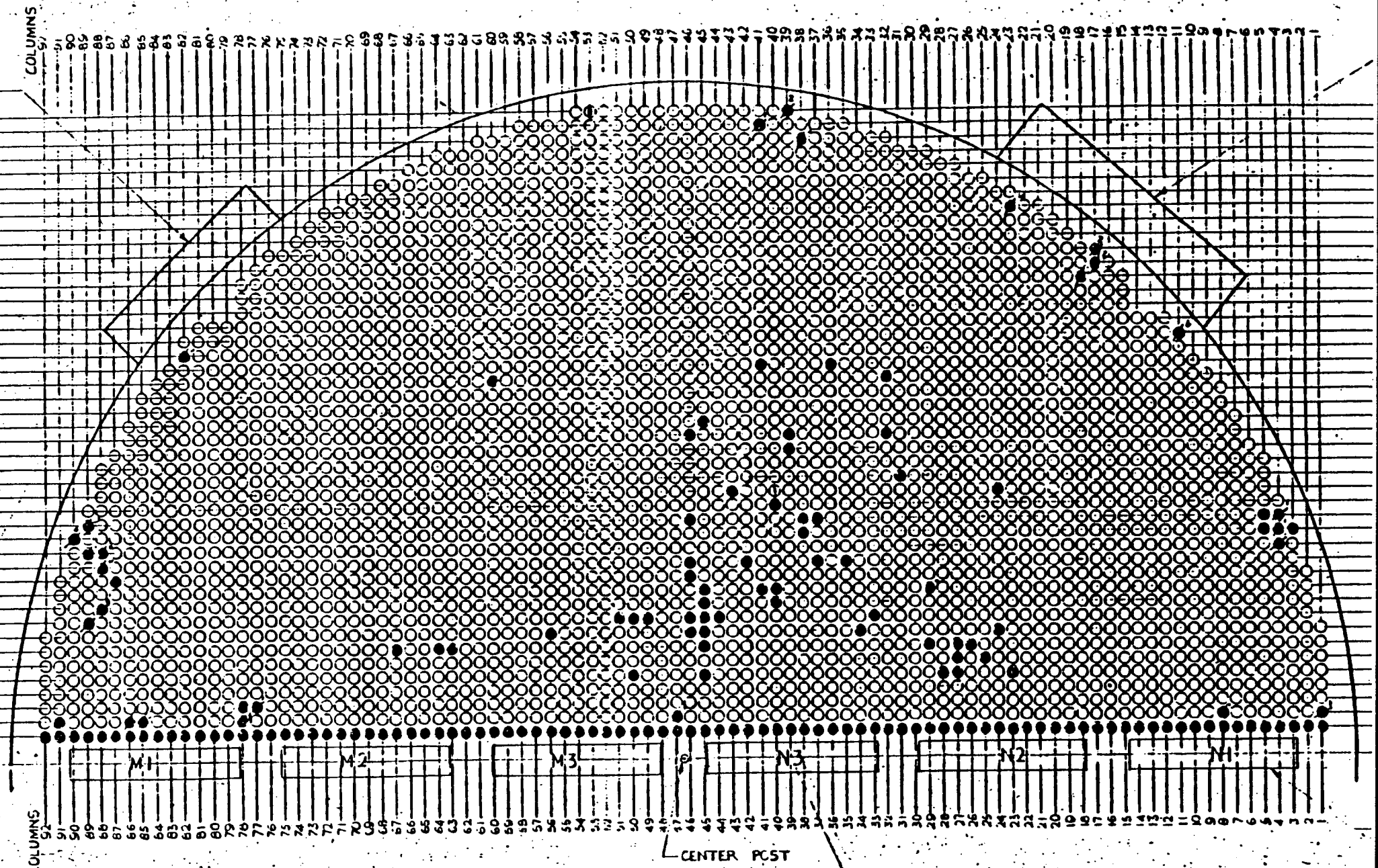
P - AVBs Located by Projection

Indications in Columns 2 and 92 are retainer rings

Plugged Tube

11/10/87:a
hol

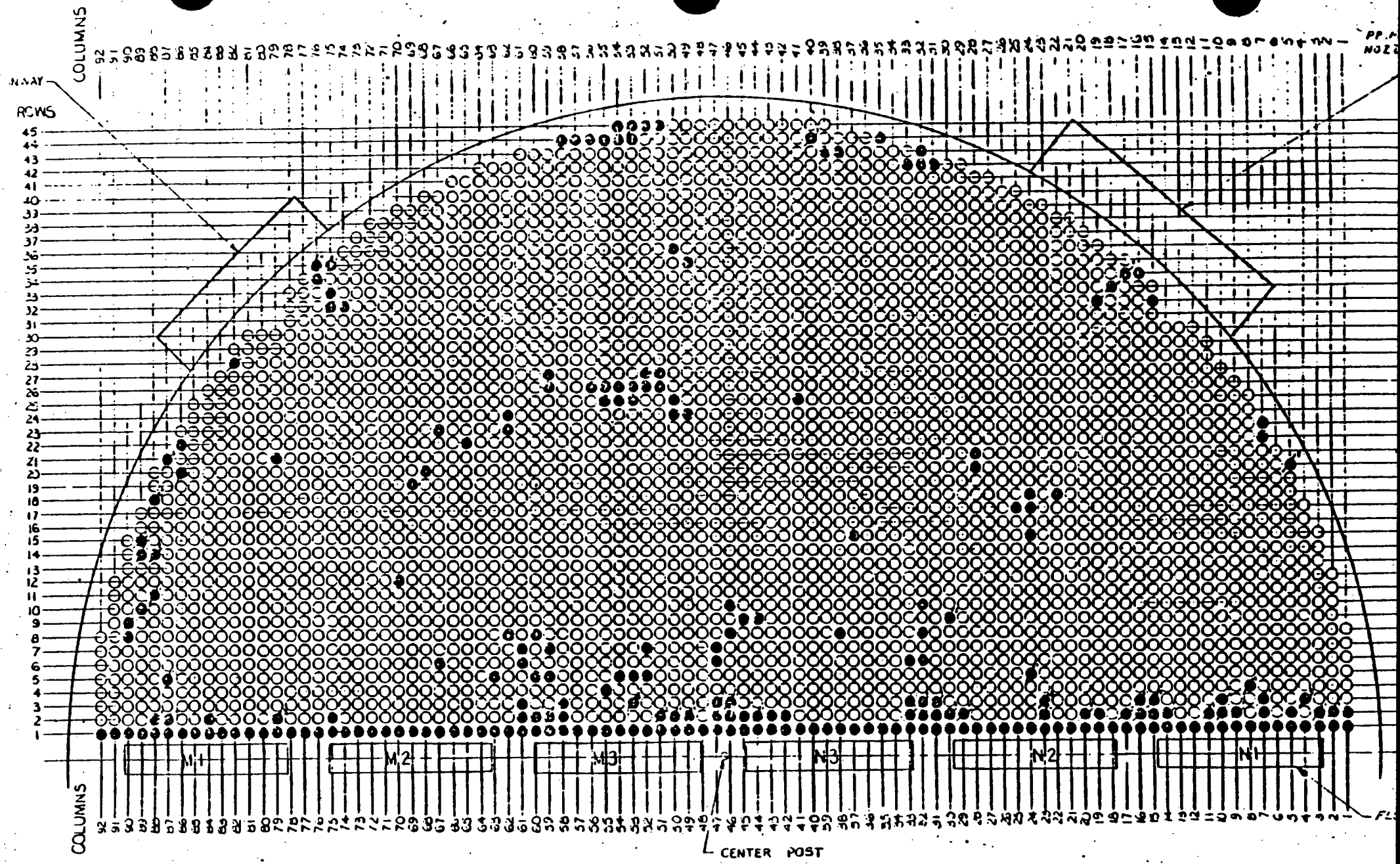
ATTACHMENT 3



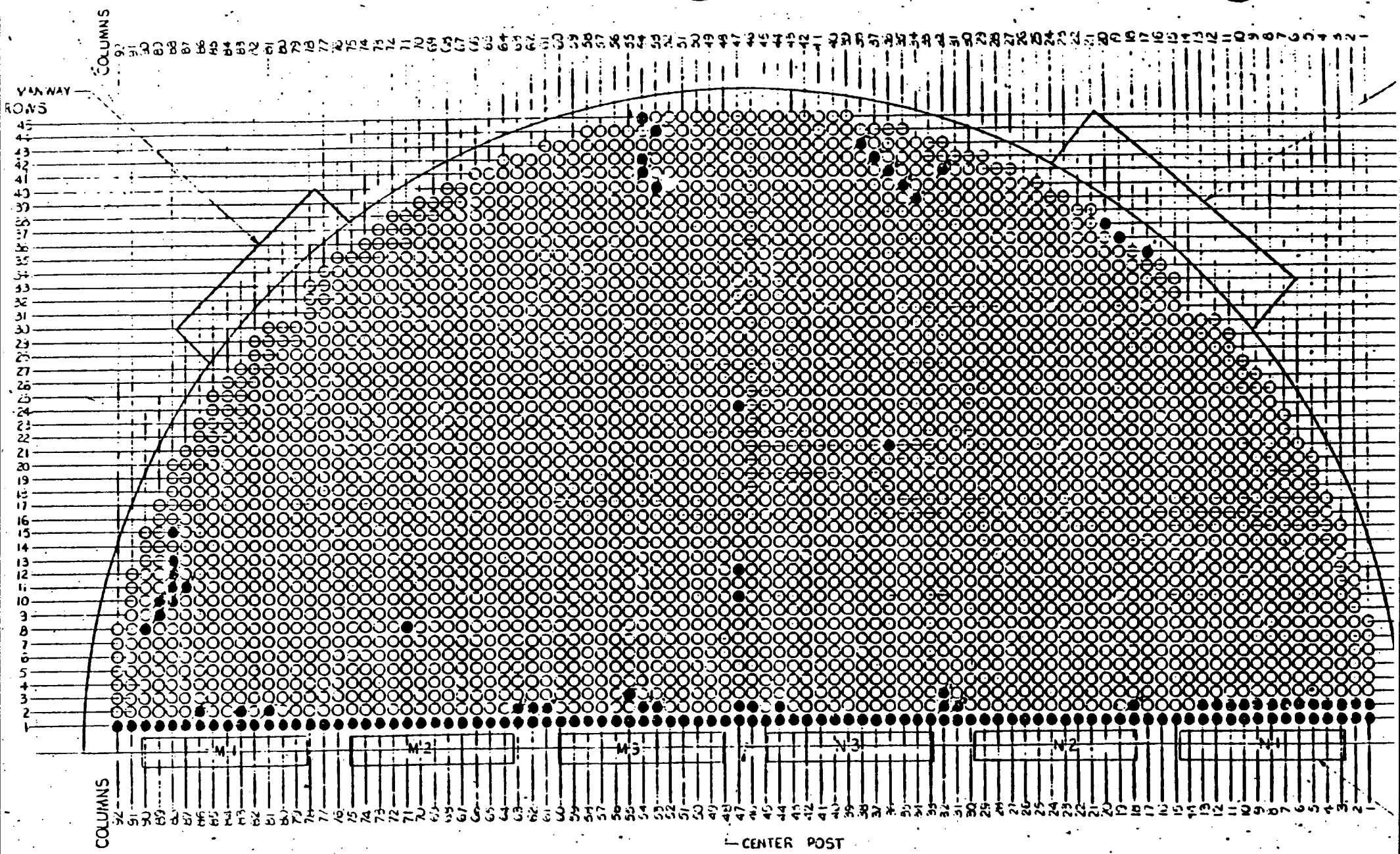
#21 STEAM GENERATOR TUBE IDENTIFICATION LAYOUT

○ = UNPLUGGED TUBE

● = PLUGGED TUBE



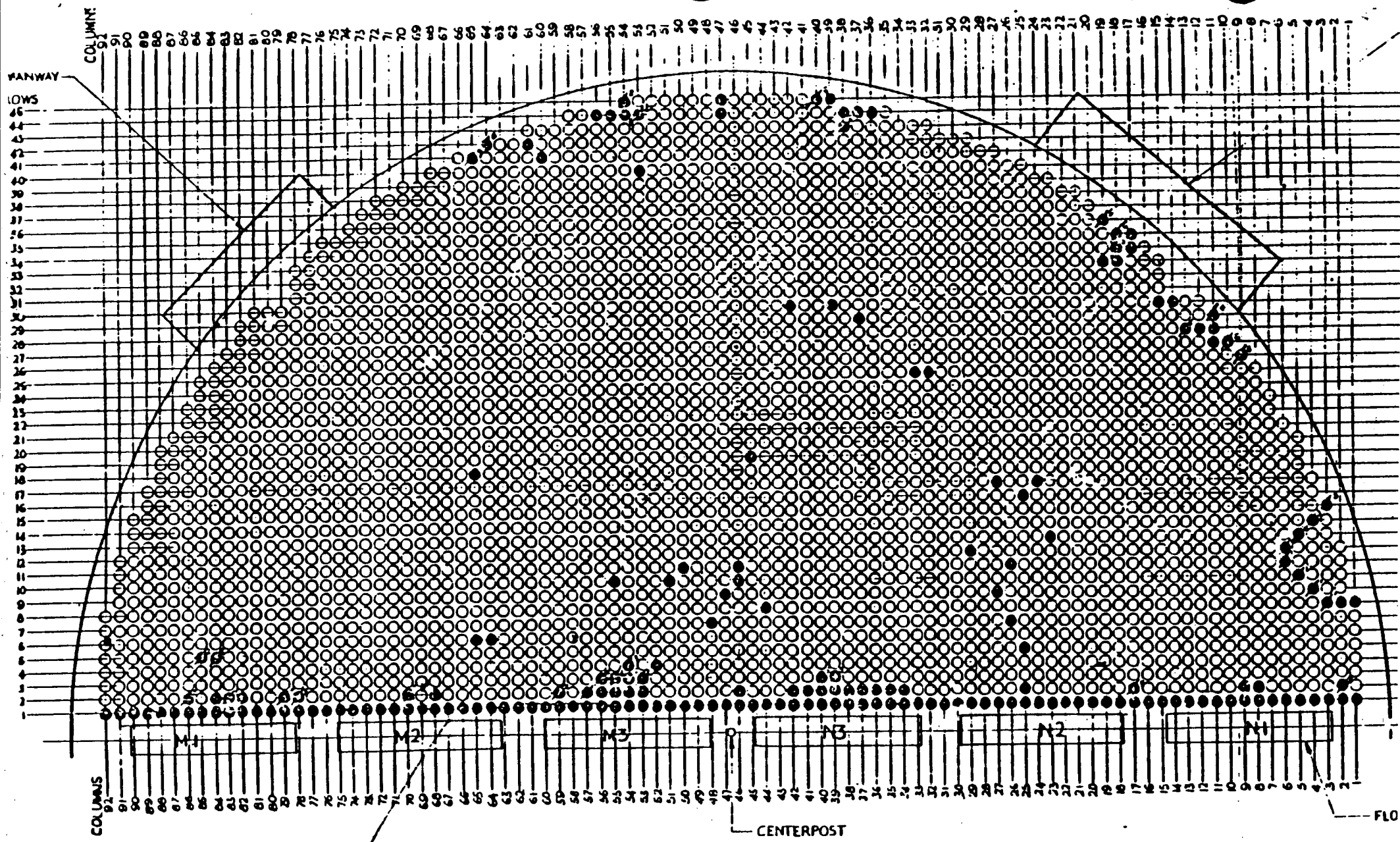
#22 STEAM GENERATOR TUBE IDENTIFICATION LAYOUT



#23 STEAM GENERATOR TUBE IDENTIFICATION LAYOUT

○ • UNPLUGGED TUBE

● • PLUGGED TUBE



#24 STEAM GENERATOR TUBE IDENTIFICATION LAYOUT

O • 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.

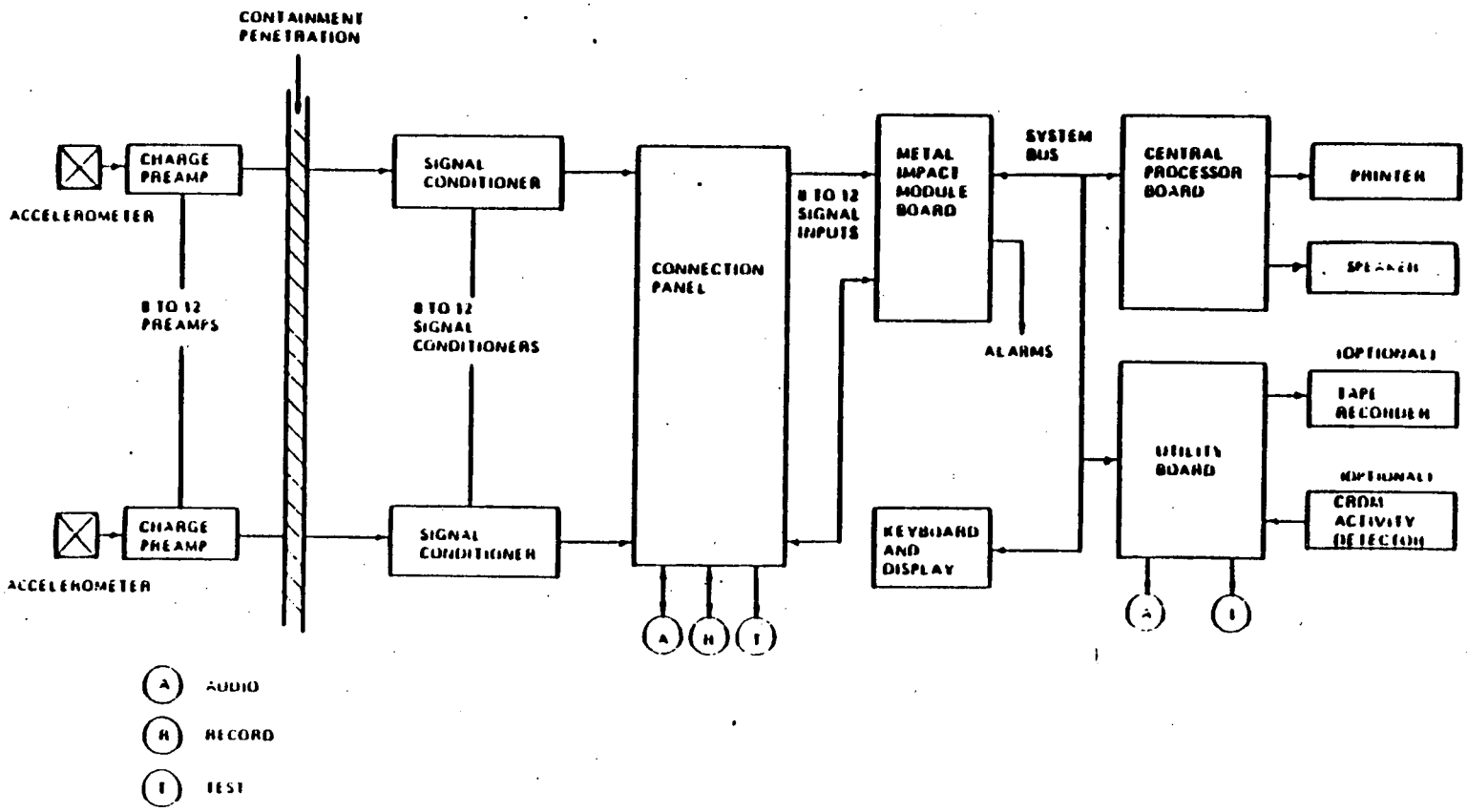
3. Duration: greater than two milliseconds. Normal electrical noise spikes are less than two milliseconds duration.

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

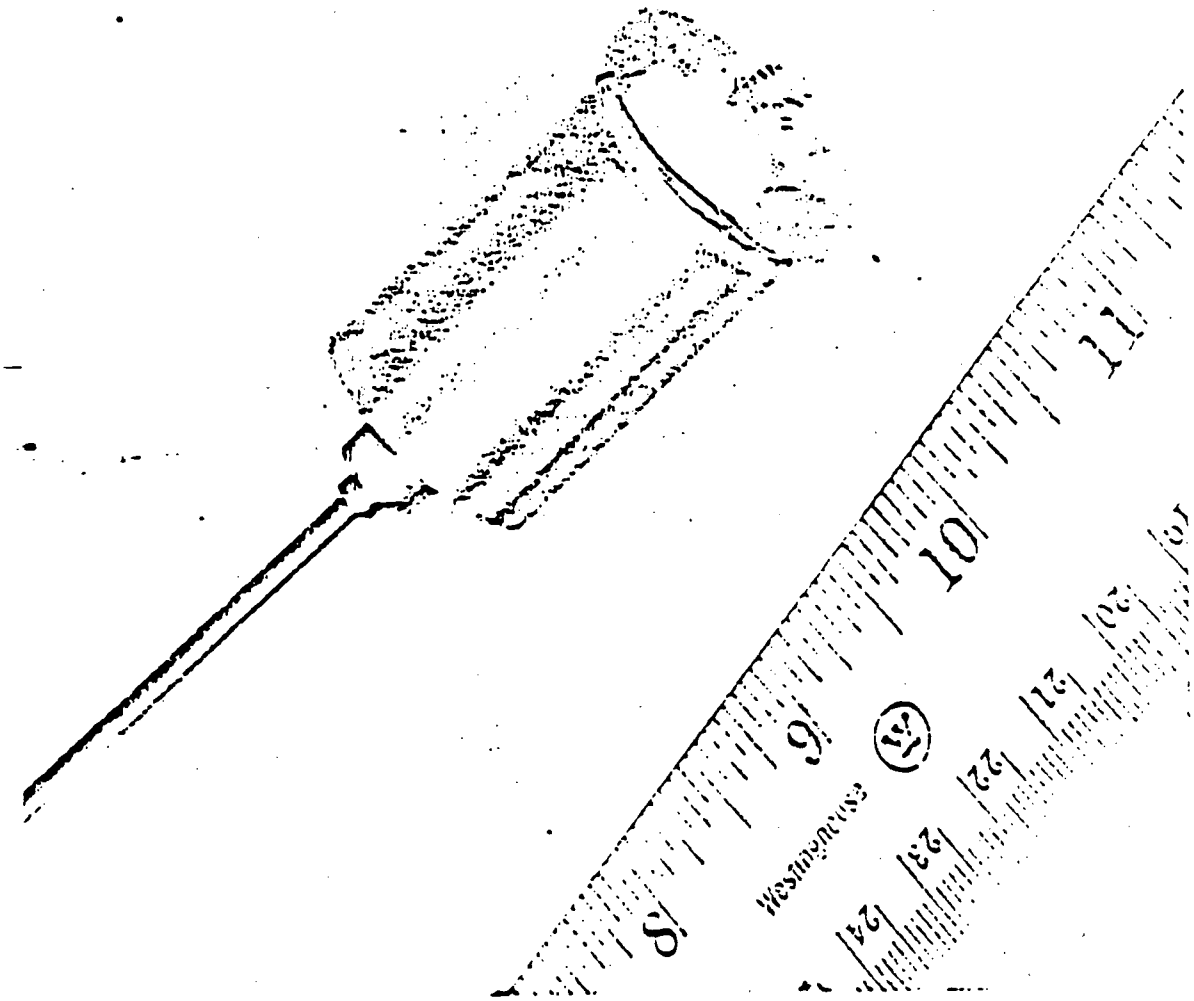
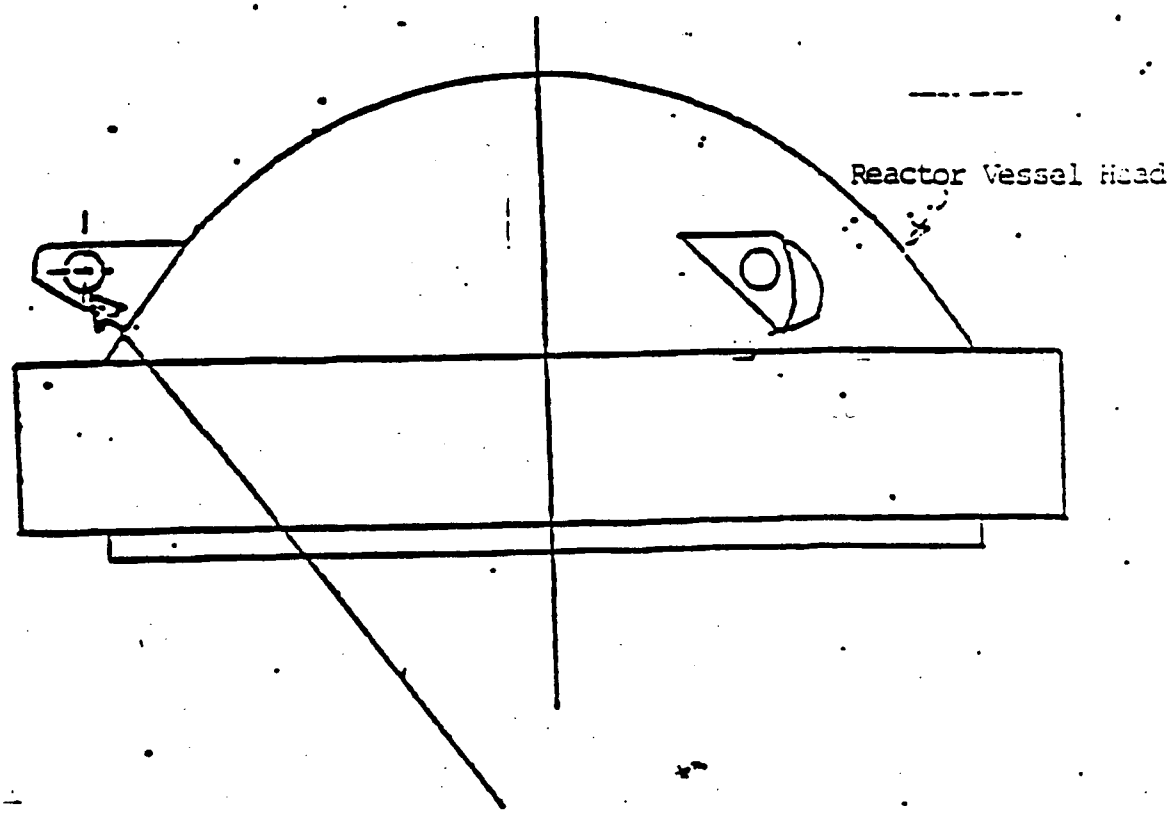


Figure 19

Transducer



Accelerometer Location
For The Reactor Vessel Head

Figure 20

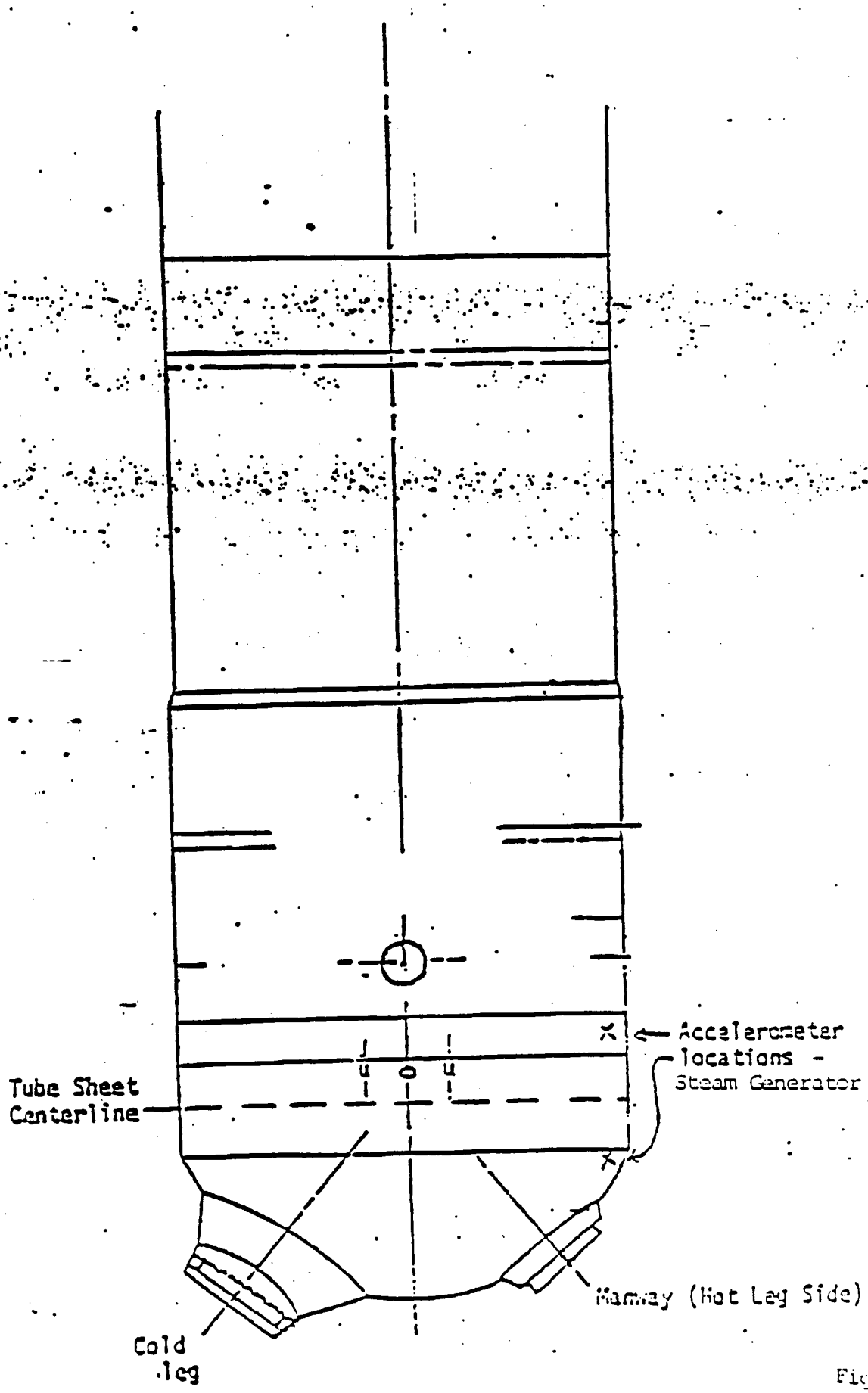
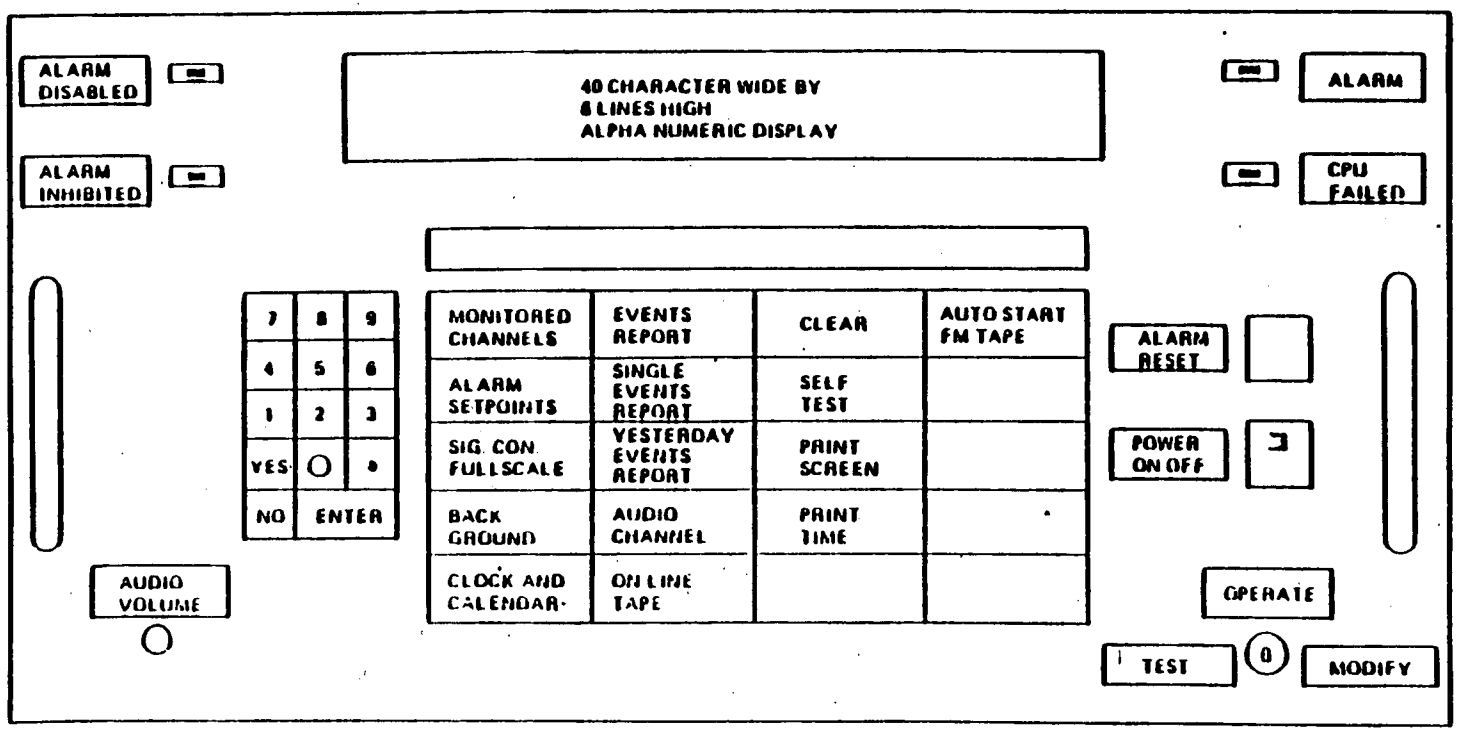
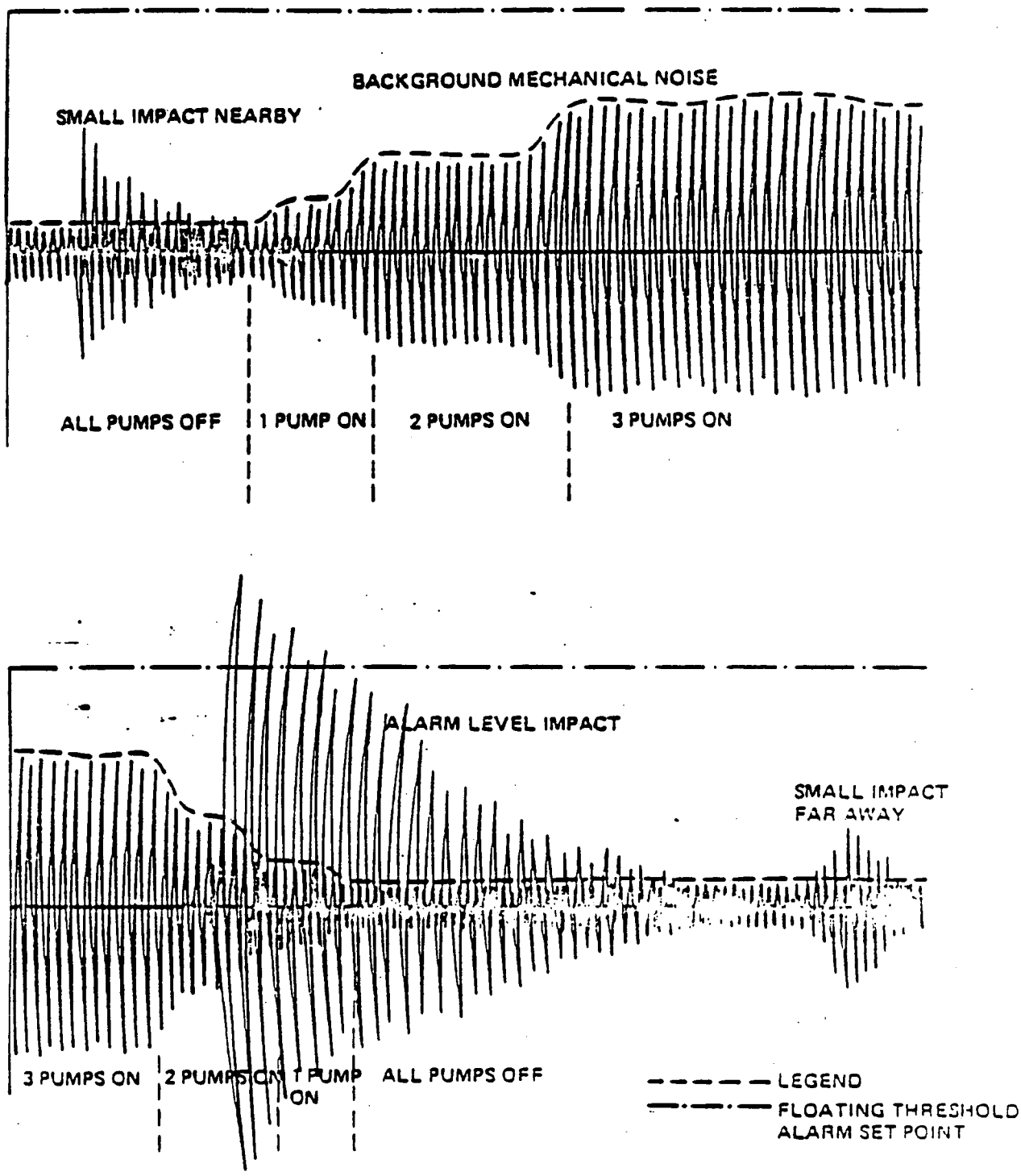

Accelerometer
locations -
Steam Generator
Tube Sheet
Centerline
Manway (Hot Leg Side)
Cold
leg

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

Attachment V

Steam Generator Tube Inservice Inspection Program
1987 Inspection Program Results

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March 25, 1988

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, New York 10511-1099

January 22, 1988

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Steam Generator Tube Inservice Inspection Program
1987 Inspection Program Results

The results of steam generator tube inspections as conducted at Indian Point Unit No. 2 (IP-2) during the recently concluded refueling outage are reported in Attachment I to this letter pursuant to IP-2 Technical Specification 4.13. The inspection program was described in our July 20, 1987 letter and approved by NRC by letter dated October 6, 1987. Additional information on the program was transmitted via a letter dated October 30, 1987.

Based upon the results of the most recent inspections, we have concluded that the next steam generator tube examinations should be conducted at the time of the next refueling outage. We request that you review the enclosed report, and approve our proposed examination schedule in accordance with Technical Specification Section 4.13.C.5

Should you or your staff have any questions, please contact us.

Very truly yours,

Stephen Bram
Vice President
Nuclear Power

Attachment

8802030145 27pp.

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ATTACHMENT I

Steam Generator Examination Program and Results
1987 Refueling Outage

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
January 22, 1988

1. Examination Program Description

Details of the steam generator tube inservice examination program planned for IP-2's eighth refueling outage were submitted to the NRC by Con Edison letter dated July 20, 1987.

NRC approval of the program was received by letter dated October 6, 1987. This NRC letter requested additional information, which we transmitted in our October 30, 1987 letter.

The scope, results and conclusions of our steam generator tube inservice examination program, as conducted during the eighth refueling outage just completed are discussed below.

2. Scope of the Examination

a. Steam Generator Tube Eddy Current Examination

One hundred percent of the tubes in service in the four steam generators were eddy current examined from the tubesheet to the first tube support plate on both the hot-leg and cold-leg sides.

Selected tubes in the hot-legs and cold-legs of Steam Generators 21, 22, 23 and 24 were further eddy current examined for both dents and defects. The selection of these tubes was based primarily on the results obtained from inspections during earlier outages. The examinations were performed at 10, 200, 400 and 600 KHz.

Examination was conducted from the hot-leg side of the channel head. Tubes identified as "hot-legs" were examined from the tube sheet, around the U-bend, to the uppermost support plate on the cold-leg side. Tubes identified as "cold-legs" were examined full-length from tubesheet to tubesheet.

A standard 700 mil or 720 mil diameter eddy current probe was used to perform the eddy current testing. If any tube did not permit passage of this standard 700 mil probe, the tube was eddy current tested with a 610 mil probe. If any tube did not permit passage of the 610 mil probe, it was plugged. Furthermore, the tubes immediately adjacent to any tube that did not pass the 610 mil probe were subjected to eddy current examination.

Identification of the hot-leg and cold-leg tubes in Steam Generators 21, 22, 23 and 24 which were eddy current examined for dents and defects are given in Tables 1, 2, 3, and 4. In Steam Generator 21, 766 tubes, or 24.8 percent of the active (in service) tubes, were examined. In Steam Generator 22, 736 tubes, or 24.5 percent of the active tubes, were examined. In Steam Generator 23, 637 tubes, or 20.5 percent of the active tubes, were examined. In Steam Generator 24, 662 tubes, or 21.7 percent of the active tubes, were examined.

In each steam generator, the eddy current examination from the hot-leg included tubes in the patch plate and peripheral "hard spot" areas, a sample of tubes in the interior section of the bundle, and all active tubes in Rows 2 and 3. Tubes in Row 1 were not examined because these tubes in the four steam generators were plugged during construction.

b. Flow Slot and Lower Support Plate Inspections

Using the hand holes above the tubesheet on the four steam generators, visual and photographic examinations of the flow slots in the lower tube support plates were made. Where feasible, higher support plates were also photographed through the flow slots in the lower support plates.

Using the "hillside" inspection ports in Steam Generators 22 and 23, visual and photographic examinations were made of the flow slots in the uppermost support plates.

c. Secondary Side Examination

A video camera was passed around the annulus between the tube bundle and the shell, and down the tube lane between the hot-legs and cold-legs to search for foreign objects in the steam generators.

d. Steam Generator Sludge Analysis

The sludge removed from the steam generator tube sheets during lancing operations was sampled and chemically analyzed.

e. J-Tubes Examination

Ultrasonic examination of selected J-tubes from Steam Generator 22 was performed to determine whether any unusual wear patterns or metal loss has occurred.

3. Results of Examination

a. Results of the Tube Examination

A total of 146 tubes with eddy current indications were left in service in 1986. In 1987, the indications were not found in 28 of these. In 45 of these tubes the indications were reported to be larger than reported in 1986. In 62 of these tubes the indications were reported to be approximately the same size (±10%) and in 11 tubes, the indications were reported to be smaller. Of the 45 tubes in which the eddy current indications were reported to be larger in 1987 than in 1986, the maximum single increase was 37%, and the average increase was 15%. Of the 39 tubes in which the indications were reported to be smaller (or to have disappeared) in 1987, the maximum single decrease was 39%, and the average decrease was 24%. Hence it is not possible to derive a "growth rate" for these indications.

The number of tubes with eddy current indications during the 1987 inspection and the distribution of the indications are listed in Table 5. Tubes with indications evaluated at 40% of the wall thickness or greater were plugged. The locations of the indications and the eddy current probe voltage of each are listed in Tables 6A through 6D.

Average dent size in the tubes in each steam generator was not substantially different from that measured during previous steam generator examinations

The number of tubes that did not permit passage of a 610 mil diameter probe and their distribution are listed in Table 7.

The number of tubes plugged this outage, tubes previously plugged, tube plugging totals, and the current percentage plugged are listed in Table 8.

b. Results of Flow Slot and Lower Support Plate Examination

The photographs taken of the lower support plate flow slots indicated essentially no change in the "hour-glassing" in the flow slots in the lower support plates in all four steam generators from that found in previous outages.

During previous steam generator examinations, photographs revealed cracks in the tube support plates at some flow slots in the steam generators. The cracks are in the ligaments between the flow slot and the first row tube holes. Examination of these flow slots during the 1987 steam generator examination program showed essentially no change in the cracks.

c. Top Support Plate Examination of Steam Generators 22 and 23

The top support plates in Steam Generators 22 and 23 were visually examined as was done in previous examinations. Examination is accomplished by utilizing a borescope inserted through the one inch diameter (nominal) "hillside port" provided in the steam generator shells. No "hour-glassing" of the flow slots in the top support plates was observed.

d. Results of Secondary Side Examination

Two new small foreign objects were observed on the secondary side of the tubesheet in the steam generators in addition to the objects observed during the previous examination. The objects could not be moved. A safety analysis, performed for these remaining objects concluded that Indian Point Unit No. 2 can safely return to power and operate with the foreign objects remaining in the steam generators until at least the next refueling outage, at which time the integrity of the steam generator tubes in the vicinity of these objects will be reassessed.

e. Sludge Analysis

Sludge was removed from each of the steam generators by lancing. Quantities of sludge removed are listed in Table 9. The sludge consisted of approximately 57% iron, 33% copper, and 10% insolubles.

4. Conclusions

The 1987 steam generator tube inservice examination demonstrates that the Indian Point Unit No. 2 steam generators are acceptable for continued service at full power.

Based on the results of this examination, the next steam generator examination has been scheduled for the next refueling outage, which is currently planned for the spring of 1989.

This scheduling for the next examination is appropriate for the following reasons:

- Our experience with steam generator tubes has been good. Since the initial operation of Indian Point Unit No. 2, there have been only seven instances of primary-to-secondary leakage, which in all cases has been very small. One leak occurred in the first quarter of 1975. The leak was in one tube near the tubesheet in Steam Generator 22. The second leak occurred in the fourth quarter of 1976. The leak was in two tubes near the top tube support plate in Steam Generator 24. The third leak occurred in Steam Generator 22 during the second quarter of 1979. Due to its extremely small size (i.e., 0.06 gallons per day), testing did not identify the exact location of the leak, but the leaky tube was included in those preventively plugged. The fourth leak occurred in two tubes on the cold-leg in Steam Generator 23 during startup hydrostatic testing in April 1981, and the fifth leak occurred in another cold-leg tube in Steam Generator 23 after a turbine trip in August 1981. The sixth leak occurred in a tube on the hot-leg in Steam Generator 22 in February 1984. The seventh leak occurred in one tube on the hot-leg of Steam Generator 23 during startup hydrostatic testing in February 1986.

- In 1978 we plugged tubes which did not pass a 610 mil probe. Profilometry was used as an added examination technique during the 1980, 1982 and 1984 examinations. If the maximum strain was found to be acceptable in tubes which did not pass a 610 mil probe, they continued in service. Profilometry was not performed in 1986 and 1987; we plugged tubes which did not pass a 610 mil probe including those previously left in service based on profilometry. No onset of increased tube degradation accompanied by tube leaks occurred when the unit was returned to service following any of the refueling/maintenance period steam generator examinations.

- During the 1987 examination, tubes in which eddy current signals indicated wall loss equal to or greater than 40%, and tubes which did not pass a 610 mil probe were plugged. No onset of increased tube degradation accompanied by tube leaks is expected to occur during future operation of IP2.

- Should any tube leak occur, our Technical Specification 3.1.F requires a thorough assessment of its significance and specifies applicable limiting conditions of operation.

- As an additional conservatism we have voluntarily updated our tube leak assessment program to track detectable leaks at incipient stages to assure that any sudden increases will be properly analyzed and appropriate mitigative action taken.

TABLE 1-A

HOT-LEG TUBES EDDY CURRENT EXAMINED
STEAM GENERATOR NO. 21

ROW	COLUMN	NO. OF TUBES
2	2-7,9-46,48-77,79-84,87-90,92	85
3	1-76,79-92	90
4	1,2,23,27,28,45,50,77,78,91,92	11
5	1,2,22,24-26,29,44,46,49,51,91,92	13
6	1,2,24,26,28,45,50,51,63,64,67,91,92	13
7	1-5,24,25,28,30,34,44,46,56,62, 88,90,91,92	18
8	1-5,23,25-27,29,33,35,44,47,49-51,55 57,88-92	24
9	2-5,32,34,43,47,52,56,63,64,67,88, 90,91	16
10	2-5,29,33,39,41,44,46,49-51,56,87, 89,90,91	18
11	2-5,28,30,39,42,44,46,87,88,89,90,91	15
12	2-5,29,35,37,40-42,45,47,51,86,88, 90,91	17
13	3-5,34,36,38,41,43,45,47,87,89,90	13
14	3,5,6,35,37,38,42,46,87,90	10
15	6-8,37,39,46,51,88,89	9
16	6-8,36,39,45,47,88	8
17	4,5,24,39,41,43,46,88,89	9
18	5,6,23,25,31,40,42,44,51,87,88	11
19	5,6,87,88	4
20	5,6,39,87,88	5
21	6,7,15,18,21,24,27,30,33,42,45,48, 51,86,87	15
22	7,8,31,33,38,40,85,86	8
23	7,8,32,39,44,46,85,86	8
24	8,9,84,85	4
25	8,9,32,60,84,85	6
26	9,10,31,33,36,41,59,61,83,84	10
27	10,11,32,35,37,40,42,51,82,83	10
28	11-17,36,76-81	14
29	12-17,76-82	13
30	12-17,51,76-81	13
31	15-17,76-78	6
32	15-18,39-54,76-78	23
33	15-17,19,39,54,76-78	9
34	16,18,39,40,53,54,76,77	8
35	18,39,40,53,54,75,76	7

TABLE 1-A (continued)

36	19,20,24,25,27-54,73,74	34
37	20,21,23-54,72,73	36
38	21,22,24,26-54,71,72	34
39	23,24,39,40,53,54,69,70	8
40	25-28,39,40,53,67,68	9
41	27,28,39,40,53,54,62-66	11
42	29-50,53-64	34
43	32-37,39-42,53-61	19
44	35-40,42-58	23
45	40-54	15
	Total Number of Hot-Leg tubes	766

TABLE 1-B

COLD-LEG TUBES EDDY CURRENT EXAMINED
STEAM GENERATOR NO. 21

ROW	COLUMN	NO. OF TUBES
7	1-5,88,90,91,92	9
8	5,88,91,92	4
9	5,88	2
10	2,3,5,90,91	5
11	2,3,5,89,90,91	6
12	90,91	2
18	6	1
29	13	1
31	16	1
33	16,39	2
36	24,25,27,28,29,30,31,35,36,37,38,39, 42,43,44,45,48,49,50,51,52	21
37	24-27,30,32,34-44,46,48-52	23
38	24,26-30,36-46,49-52	21
39	23,40,53,54	4
40	27,39,40,53	4
41	27,40,53	3
42	32-35,39,40,53,54,57-61,64	14
43	32,34,36,37,39,40,56-60	11
44	35,36,54,57,58	5
45	50,54	2
	Total Number of Cold-Leg Tubes	141

TABLE 2-A

HOT-LEG TUBES EDDY CURRENT EXAMINED
STEAM GENERATOR NO. 22

ROW	COLUMN	NO. OF TUBES
2	4,6,8,11-13,18,21,22,24-28,34-41,48, 52-57,62-74,76,77,78,80-83,85,86,89-92	55
3	1,2,3,5,6,8,9,11-14,17-22,24-30,34-45, 48-52,54-57,59,60,62-92	78
4	1,2,7,9,10,15,16,23,24,31,32,33,46,47, 52-54,56,58-60,63,87,91,92	25
5	1,2,8,23,25,32,33,47,51,55,58,61,62,64, 67,86,88,91,92	19
6	1,2,24,31,34,39,46,48,51,52,60,62,63,66, 68,87,91,92	18
7	1,2,32,33,38,46,48,51,53,58,60,62,67,90, 91,92	16
8	1,2-5,30,31,33,37,39,44,45,47,52,59,61, 63,88,89,91,92	21
9	2,3-5,29,31,32,38,43,46,51,60,62,88,89,91	16
10	2-5,30,31,33,45,47,88,90,91	12
11	2-5,32,46,70,90,91	9
12	2,3-5,51,69,71,88-91	11
13	3,4-6,51,70,88-90	9
14	3,4-6,24,37,87,90	8
15	3,4,5,23,25,36,38,51,88,90	10
16	4,5,24,25,37,88,89	7
17	4,5,22,23,26,88,89	7
18	5,6,21,23,25,51,69,87	8
19	5,6,22,24,28,68,70,86,87,88	10
20	6,27,29,67,69,79,85,87,88	9
21	6,7,13,15,18,21,27,29,30,36,39,42,45,48, 51,65,68,78,80,86	20
22	8,28,62,64,66,67,79,85	8
23	8,49,50,61,63,65,66,68,85,86	10
24	8,9,41,48,51,53-55,61,63,67,84,85	13
25	8,9,40,42,49,51,52,59,62,84,85	11
26	9,10,41,50,57,58,60,83,84	9
27	10,11,50,53-56,58,60,81-83	12
28	11,12-18,77-81	13
29	11,12-17,77-82	13
30	12,13-17,51,78-81	11
31	15-20,74,75,77,78	10
32	16-18,20,39-54,73,76,77,78	24
33	15-17,19,39-54,74,76,77,78	24
34	18,19,39,40,49,53,54,75,77	9
35	17,18,39,40,48,53,54,74	8

TABLE 2-A (continued)

36	19,20,39,40,49,51,53,54,73,74	10
37	20,21,23-54,73	35
38	21-54,71,72	36
39	23,24,39,40,53,54,69,70	8
40	25,26,27,39,40,41,53,54,67,68	10
41	27,28-40,54-66	27
42	29,30,34-41,53-64	22
43	33-37,40,41,52-61	17
44	36-39,41-52	16
45	39-50	12
	Total Number of Hot-Leg Tubes	736

TABLE 2-B

COLD-LEG TUBES EDDY CURRENT EXAMINED
STEAM GENERATOR NO. 22

ROW	COLUMN	NO. OF TUBES
3	58,59,60,63	4
8	2-5,88,89	6
9	3,4,5,51,88,91	6
10	2-5,88,90,91	7
11	2-5,90,91	6
12	3,4,5,51,89,90,91	7
13	51	1
18	51	1
21	15,18	2
35	17	1
36	19	1
37	23-52	30
38	23-52	30
39	23,39,40,53	4
40	27,39,40,41,54	5
41	29,30,32-40,55,56,59-66	21
42	30-32,34-41,53,55-64	22
43	33-35,37,40,55-61	12
44	37-39,43,47-50	8
45	44,47,48,50	4
	Total Number of Cold-Leg tubes	178

TABLE 3-A

HOT-LEG TUBES EDDY CURRENT EXAMINED
STEAM GENERATOR NO. 23

ROW	COLUMN	NO. OF TUBES
2	14-17,19-30,33-43,45,48-52,55-60, 64-80,82,84,85,87-92	65
3	1-31,33-54,56-92	90
4	1-14,32,55,91,92	18
5	1,2,91,92	4
6	1,2,91,92	4
7	1,2,90-92	5
8	1-4,6,70,72,88,89,91,92	11
9	2-6,46,47,71,88,90,91	11
10	2-6,46,48,87,90,91	10
11	2-6,47,86,89-91	10
12	2-7,46,48,62,87,89,90,91	13
13	3-6,47,86-90	10
14	3-6,88-90	7
15	3-6,46,87,89,90	8
16	4,5,88,89	4
17	4-6,88,89	5
18	5,6,46,87,88	5
19	5,6,87,88	4
20	5,6,36,87,88	5
21	6,7,16,19,22,25,28,31,34,37,40, 43,46,86,87	15
22	7,8,36,85,86	5
23	7,8,47,85,86	5
24	8,9,46,48,84,85	6
25	8,9,47,84,85	5
26	9,10,83,84	4
27	10,11,46,82,83	5
28	11-16,77-82	12
29	11-13,15-17,77-82	12
30	12-17,46,77-81	12
31	15-17,77,78	5
32	15-17,39-54,77,78	21
33	15-17,34,39-54,77,78	22
34	16,17,39,40,53,54,76,77	8
35	18,19,39,40,53,54,75,76	8

TABLE 3-A (continued)

36	20,39,40,53,54,73,74	7
37	21,23-54,72,73	35
38	21,22,23-54,71,72	36
39	23,24,33,35,36,39,40,53,54,69,70	11
40	25,26,31-34,36,37,40,54,67,68	12
41	27-31,33-35,37-40,53,55-66	25
42	29-36,38-40,53,55-64	22
43	32-37,39,40,53-61	17
44	35-51,52,54-58	23
45	39-53	15
	Total Number of Hot-Leg Tubes	637

TABLE 3-B

COLD-LEG TUBES EDDY CURRENT EXAMINED
STEAM GENERATOR NO. 23

ROW	COLUMN	NO. OF TUBES
8	1,2,4,6,88,89	6
9	2,3,5,46,88,90,91	7
10	2-6,90,91	7
11	2-6,90,91	7
37	23-54	32
38	23-52	30
39	35,36,39,40,53,54	6
40	31,32,34,37	4
41	27-30,35,38,64,65,66	9
42	29,32,34,35,36,40,53,57,58,64	10
43	34-37,40,53,54,56-60	12
44	41,43,46,57,58	5
	Total Number of Cold-Leg Tubes	135

TABLE 4-A

HOT-LEG TUBES EDDY CURRENT EXAMINED
STEAM GENERATOR NO. 24

ROW	COLUMN	NO. OF TUBES
2	1,3-7,10-16,18-24,26-33,43,44,45, 47-52,58,60-67,69,71-77,80,81,85,87-92	63
3	1-38,41-52,57-92	86
4	4-9,15-18,25,37-41,47,48, 51,53,55,56,82,84,85	25
5	24,26,37-41,52-54,56,57,65,82,83,86	16
6	25,26,48,63,66,84,85	7
7	1-5,25,27,44,47,49,64,65	12
8	4,5,26,27,43,45,47,48,88,89,90,91,92	13
9	2,3,5,6,26,44,46,48,51,55,88,89,90,91	14
10	2,3,4,6,88,89,90,91	8
11	2-5,7,25,27,29,45,47,49,51,55,88-91	17
12	2-5,7,23,26,28,30,46,50,88-91	15
13	3,4,6,22,24,29,88-90	9
14	3,5,23,88-90	6
15	4,5,25,46,88-90	7
16	24,26,27,28	4
17	23,25,28,65	4
18	24,27,44,45,46,64	6
19	44,46,65	3
20	45	1
21	13,16,19,22,25,28,31,34,37,40,43-47	15
22	43,44,45,46,47	5
23	43,44,45,46,47	5
24	32,33,43,44,45,46,47	7
25	9,31,43,44,45,46,47	7
26	10,11,12,32,33	5
27	12-17,46	7
28	14-17,37,77-82	11
29	12-17,36,38,39,42,76-82	17
30	12,13,16,17,37,38,40,41,43,46	10
31	15-17,39,42,76-78	8
32	15-19,39-54,76-78	24
33	15-17,20,39-54,76-78	23
34	16,19,20,39,40,53,54,76,77	9
35	19,20,39,40,53,54,75,76	8

TABLE 4-A (continued)

36	20,23,24,27-34,39,40,52-54	16
37	23-54	32
38	23-54	32
39	39,40,52,53,54	5
40	39,40,52,54,60,65	6
41	27-40,52,53,54,56-59,61-64,66	26
42	29-40,53-60,62,63	22
43	32-37,39,40,52,53-61	18
44	35,39-46,48-52,57,58	16
45	41-46,48-53	12
	Total Number of Hot-Leg Tubes	662

TABLE 4-B

COLD-LEG TUBES EDDY CURRENT EXAMINED
STEAM GENERATOR NO. 24

ROW	COLUMN	NO. OF TUBES
7	3-5	3
8	4,5,88,89,91	5
9	2,3,88,90,91	5
10	2,3,4,88,90,91	6
11	2,3,4,89,90,91	6
12	2-5,46,88-91	9
36	24,27-34	9
37	26-30,33,34,37-51	22
38	25-28,30,31,38,40,42-51	18
39	39,40,52,53,54	5
40	39,54	2
41	35,40,52-54,56-58	8
42	29,30,34-36,39,40,53-58,60,62,63	16
43	32-37,39,53,54,56-60	14
44	35,40,42-46,48,57,58	10
45	46,48	2
	Total Number of Cold-Leg Tubes	140

TABLE 5

NUMBER OF TUBES WITH EDDY CURRENT INDICATIONS
(PERCENT WALL THINNING)

S/G NO.	LEG	20-29%	30-39%	40-49%	50-59%	60-69%	>70%
21	Hot	2	2	0	0	1	0
21	Cold	25	34	9	4	3	0
22	Hot	14	8	4	0	1	0
22	Cold	26	33	4	5	2	0
23	Hot	18	19	4	3	1	0
23	Cold	3	10	6	7	6	1
24	Hot	24	9	0	0	0	0
24	Cold	16	31	13	8	8	0

TABLE 6-A

LOCATION OF INDICATIONS ≥ 40%
STEAM GENERATOR NO. 21

ROW	COLUMN	INCHES ABOVE TUBESHEET	VOLTAGE
4	53	0.4	0.40
6	56	2.9	0.50
8	44	0.4	1.38
8	53	7.1	0.83
8	55	4.0	1.09
8	60	3.9	0.93
9	38	12.6	0.77
9	43	12.8	0.31
10	53	7.3	0.72
10	60	7.5	0.42
11	51	9.7	1.22
11	60	7.3	0.57
15	69	0.4	0.39
16	35	11.2	0.62
17	24	1.0	0.77
26	33	2.8	0.43
28	58	0.4	0.49

TABLE 6-B

LOCATION OF INDICATIONS ≥ 40%
STEAM GENERATOR NO. 22

ROW	COLUMN	INCHES ABOVE TUBESHEET	VOLTAGE
5	35	1.1	0.43
6	29	1.8	0.46
6	68	1.7	0.53
7	63	3.0	0.65
9	72	1.2	0.59
11	71	1.0	0.50
11	73	0.8	0.51
13	28	4.9	0.72
18	46	11.1	0.36
18	47	6.4	0.42
19	27	1.9	0.55
20	26	0.4	0.77
26	58	2.1	0.86
27	50	1.4	0.45
28	54	6.3	1.35
42	30	0.1	0.59

TABLE 6-C

LOCATION OF INDICATIONS $\geq$ 40%
STEAM GENERATOR NO. 23

ROW	COLUMN	INCHES ABOVE TUBESHEET	VOLTAGE
7	22	0.0	1.14
7	38	2.4	0.56
7	68	1.1	0.77
8	37	2.3	0.91
8	38	2.3	0.66
8	68	0.9	0.31
9	31	1.6	1.20
9	38	1.8	1.32
9	39	2.4	0.91
10	39	2.8	0.83
11	46	9.3	0.51
12	40	1.0	0.62
12	57	4.0	0.40
14	28	2.3	0.36
14	36	3.0	0.81
15	71	0.7	0.98
15	72	0.6	0.82
18	49	4.0	0.67
20	35	2.0	0.84
20	36	1.6	0.94
20	49	3.0	0.53
21	34	0.9	0.91
21	35	1.3	0.67
21	47	5.1	0.43
22	41	2.1	0.68
25	59	1.2	0.86
28	33	0.5	1.47
28	38	1.9	0.57

TABLE 6-D

LOCATION OF INDICATIONS ≥ 40%
STEAM GENERATOR NO. 24

ROW	COLUMN	INCHES ABOVE TUBESHEET	VOLTAGE
6	24	0.8	0.98
6	25	0.7	0.96
6	40	0.8	0.76
6	66	0.2	3.93
7	20	0.4	0.77
7	38	5.7	0.83
7	67	1.4	2.27
8	27	1.1	0.86
9	20	0.9	0.72
9	24	0.6	1.03
9	48	11.5	0.97
9	57	5.8	0.70
10	26	0.8	1.16
11	23	0.5	0.99
11	24	0.6	1.05
11	59	7.8	1.07
12	30	3.2	1.35
12	70	1.0	0.70
13	69	1.0	0.55
14	55	9.4	1.16
14	68	1.9	0.69
14	69	0.7	0.91
20	47	9.8	0.64
23	60	1.7	0.57
26	34	1.3	0.35
28	37	1.1	0.80
31	40	1.4	1.10
44	52	8.7	2.82
45	49	11.7	1.43

TABLE 7

NUMBER OF TUBES PLUGGED IN 1987
BECAUSE OF TUBE RESTRICTIONS

S/G 21	3
S/G 22	0
S/G 23	11
S/G 24	5

TABLE 8

SUMMARY OF PLUGGED TUBES

S/G NO.	Plugged in 1987	Previously Plugged	Total	Percent Now Plugged
21	22*	177	199	6.1
22	16	249	265	8.1
23	39	149	188	5.8
24	34	201	235	7.2
Total	111	776	887	6.8

* Includes 2 tubes plugged with sentinel plugs because of North Anna-related concerns.

TABLE 9

SLUDGE REMOVAL IN 1987

S/G NO.	AMOUNT OF SLUDGE
21	306 lbs
22	201 lbs
23	316 lbs
24	226 lbs
Total	1049 lbs