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SUBJECT: Responds to NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes."

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Indiana Michigan
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AEP:NRC:1056

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
RESPONSE TO NRC BULLETIN 88-02, RAPIDLY PROPAGATING
FATIGUE CRACKS IN STEAM GENERATOR TUBES

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: A. B. Davis

March 31, 1988

Dear Mr. Davis:

This letter responds to NRC Bulletin No. 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes." NRC Bulletin 88-02 requests information regarding steam generator tube inspections and programs to minimize the probability of a rapidly propagating fatigue failure.

Attachment 1 to this letter contains our response to the NRC Bulletin 88-02 action items. Included are our evaluation of steam generator tube denting, a description of in-place leak detection methods, and a proposed long term corrective action program.

Sincerely,

M. P. Alexich
Vice President

cm

Attachment

cc: D. H. Williams, Jr.
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

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Milton P. Alexich, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the foregoing response to NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

M. P. Alexich
Vice President

Subscribed and sworn to before me this 31st day of March 1987.

NOTARY PUBLIC
Commission expires 3-9-91

Attachment 1 to AEP:NRC:1056

NRC Bulletin 88-02
Rapidly Propagating Cracks in
Steam Generator Tubes

BACKGROUND

NRC Bulletin 88-02 addresses the steam generator tube rupture incident that occurred at Virginia Electric and Power Company's North Anna Unit 1. North Anna's steam generators are Westinghouse (W) Series 51 with 0.875 inch O.D. by 0.050 inch nominal wall thickness Inconel 600 tubes. While operating at 100 percent power on July 15, 1987, the tube at R9C51 in SG C ruptured, with a resultant primary-to-secondary leak rate of between 550 and 637 gpm. The failure mechanism was subsequently classified as high cycle flexural fatigue superimposed on an elevated mean stress. Tube denting at the top support plate intersection has been identified as the source of elevated mean stress levels as well as "the most significant contributor to the occurrence of excessive vibration." Denting may also have contributed to fatigue stress concentration.

This report has been prepared in response to the "Action Requests" outlined in the bulletin. The NRC staff concluded that all three of the following conditions must be present for a North Anna type fatigue failure to occur:

(1) denting at the uppermost support plate;

(2) a fluid elastic stability ratio approaching that for North Anna's R9C51 tube;

(3) absence of effective AVB support.

The responses outlining compliance with each action request have been prepared in light of this condition.

Action Request A

Recent tube inspection data for Cook Units 1 and 2 has been reviewed for evidence of denting at the uppermost, or seventh, support plate. Eddy current records for Unit 1 inspections of May 1985 and July 1987 and Unit 2 inspections of September 1985, May 1986, March 1987 and September 1987 were reviewed.

The action request states that records may be considered adequate for this review "if at least 3% of the total steam generator tube population was inspected at the uppermost

support plate during the last 40 calendar months." All of the inspections cited above have, in fact, included 100% of the tubes, with the following extents[a]:

(1) Unit 1, May 1985
  (i) 100% inspected full length

(2) Unit 1, July 1987
  (i) Rows 1 and 2 inspected through and including the U-Bend[b]
  (ii) Rows 3 through 9 inspected through and including the 7th hot leg support plate
  (iii) Rows 10 through 46 inspected full length

(3) Unit 2, September 1985
  (i) 100% full length inspected in SG 22 and 23
  (ii) Rows 6 through 46 inspected full length in SG 21 and 24 except for R9C45 and R9C46 (both in SG 24)
  (iii) Rows 2 through 5 inspected through and including the U-Bend in SG 21 and 24 (Note - all Row 1 tubes were plugged in 1984)
  (iv) SG 24 tubes R9C45 and R9C46 inspected through 7th hot leg support plate

(4) Unit 2, May 1986
  (i) 100% inspected through and including the 7th hot leg support plate

(5) Unit 2, March 1987
  (i) 100% inspected through and including the 7th hot leg support plate

(6) Unit 2, September 1987
  (i) 100% inspected through and including the 7th hot leg support plate.

[a] See Table 1 for tabulated extents.
[b] U-Bend inspections include the 7th cold leg tube support plate intersection.

Furthermore, all the inspections cited have been reviewed by both a primary reviewer and an independent second party reviewer. Maps indicating the location of uppermost support plate dents are attached. No secondary side indications or imperfections have been reported at the uppermost tube

support plate. There are a total of 45 dents for Unit 1 and 45 dents for Unit 2 that have been reported in one outage or another:

Inspection	Number of Upper Support Plate Dents Reported Cold Leg	Hot Leg
U-1 5/85	13	16
U-1 7/87	13	21
U-2 9/85	1	16
U-2 5/86	0	19
U-2 3/87	0	22
U-2 9/87	0	27

The total number of dents at the uppermost support plates for the most recent outages do not add up to 45 on each of the units primarily because many of the previously reported dents were determined to be nonexistent. Since Unit 1 and Unit 2 were subject to essentially a 100% inspection as noted in Table 1 within the last 40 months, the possibility of undetected denting is highly unlikely.

Action Request B

This action request applies to plants that have not experienced denting at the uppermost support plate.

Action Request C, Part 1

The following paragraphs describe the leak rate monitoring program currently in place at Donald C. Cook Nuclear Plant.

Background

Due to the occurrence of primary-to-secondary leakage in the Unit 2 steam generators resulting first from Row 1-U-bend cracking and later from secondary side corrosion in the tubesheet crevice region, Indiana Michigan Power has taken several measures to become attuned to primary-to-secondary leak detection and trending and has always acted conservatively with respect to Technical Specification leak rate limits. An example of Indiana Michigan Power's commitment to leak rate monitoring is the installation and implementation of the Eberline SPING radiation monitoring system. The Eberline SPING Monitor is a state-of-the-art, self-contained microprocessor-based radiation detection system. The microcomputer performs the

tasks of data acquisition, history file creation, data management, operational status checking and alarm determination; operators communicate with this system through a central control terminal located in each control room. Indiana and Michigan demonstrated conservatism with respect to leak rate limits can best be exemplified by the administrative policy which initiates shutdown actions at less than 60 percent of the allowable leak rate (0.2 gpm in lieu of 0.347 gpm which is the Technical Specification value of 500 gal. per day per steam generator).

Leakage Measurement and Trending Methods

Primary-to-secondary leak rates are calculated from steam generator blowdown grab sample tritium at least once per day. Additional sampling and calculations based on tritium are performed if the chemical supervisor notes any significant change in the leak rates or questions the accuracy of an analysis. Grab sample xenon activity analyses are also performed on steam jet air ejector exhausts at least once per day and the leak rate is determined.

In addition to analyses performed on grab samples, the steam generator blowdown composite flow and steam jet air ejector (SJAE) and gland seal condenser exhauster (GSCE) discharges are continuously monitored and alarmed in the control room. Although all three of these readouts can be used to estimate primary-to-secondary leak rate, the primary means of estimating leak rate on line is accomplished by monitoring SJAE activity. The on-line blowdown composite flow is currently monitored by the Westinghouse Radiation Monitoring System which has a range sensitivity of 2.4×10^{-6} to 2.4×10^{-2} μCi/cc. SJAE and GSCE are both monitored by state-of-the-art Eberline SPING noble gas monitors which have a range sensitivity of 5.8×10^{-7} to 1.86×10^{4} μCi/cc.

Operators are provided with updated correlations of primary-to-secondary leak rate as a function of SJAE activity levels so that leakage measurement is estimated continuously. These correlations are updated based on primary and secondary side activity grab sample analyses and SJAE flow. SJAE activity trending is extracted directly from the Eberline history management files. An updated hardcopy of these files is printed every 8 hours. In addition, blowdown sample analyses are recorded and trended daily.

Alarms

The steam generator blowdown radiation monitor is set to alarm in accordance with the Offsite Dose Calculation Manual (ODCM) at 3 X 10^{-4} μCi/cc. The Technical Specification limit on secondary side coolant activity is 0.10 μCi/cc. Receipt of a "High" alarm on the steam generator blowdown monitor initiates steam generator blowdown and sampling line isolation. Operators are required to verify this automatic function. Grab samples are analyzed to determine the extent of primary-to-secondary leakage.

Alarms on the SJAE and GSCE are also set in accordance with the ODCM; the "High" alarm is currently set at 4.77 X 10^{-4} μCi/cc while the "Alert" alarm is set at 60% of this value or 2.86 X 10^{-4} μCi/cc. Upon receipt of an "Alert" alarm signal from either a SJAE or GSCE monitor, a grab sample of the respective effluent is taken and analyzed. Upon receipt of a "High" alarm from either the SJAE and GSCE, additional steam generator blowdown analyses are performed to determine the extent of the primary-to-secondary leak. The current SJAE "High" alarm setting corresponds to roughly 1/10 the expected activity level expected for a 0.2 gpm primary-to-secondary leak.

Administrative Limits

Once primary-to-secondary leak rates approach 0.1 gpm, blowdown sampling frequency is increased to once per 8 hours. The current administrative limit set on leak rate is 0.2 gpm. When this limit is believed to be reached, the unit is brought down to 50% within approximately 1 hour. Cold shutdown is achieved within 24 hours.

Action with Inoperable Radiation Monitors

Current procedures require the following grab sample surveillance frequency in the event of an inoperable monitor:

Inoperable Monitor	Sample Frequency
Steam Generator Blowdown Liquid Monitor	(i) When blowdown specific activity $\leq$ 0.01 μCi/gram dose equivalent I-131, sampling frequency is at least once per 24 hours.
	(ii) When blowdown specific activity $\geq$ 0.01 μCi/gram, dose equivalent I-131, sampling frequency is at least once per 8 hours.

Inoperable Monitor	Sample Frequency
SJAE Noble Gas Monitor	Grab sample frequency is at least once per 8 hours and gross analysis is performed within 24 hours.
GSCE Noble Gas Monitor	Grab sample frequency is at least once per 8 hours and gross activity analysis is performed within 24 hours.
SJAE Flow Rate Monitor	Flow rate read locally every 4 hours.
GSCE Flow Rate Monitor	Flow rate read locally every 4 hours.

Effectiveness of Monitoring Program

The monitoring program in place at Cook compares favorably when evaluated against the assumed time dependent leakage curve presented in Figure 1 of the bulletin. The curve assumes that a leak will progress from roughly 10 to 300 gallons per day over a period of 5300 minutes (88 hours). Even assuming that the Eberline monitor on the SJAE was unavailable, SJAE effluent would be sampled at least every eight hours or roughly 11 times during that period. Steam generator blowdown samples would also be taken at least three times during this period. Furthermore, if significant radioactivity increases are noted in steam jet air ejector samples, additional blowdown samples will be taken and analyzed. Once the leak rate as determined by tritium level in blowdown samples approaches 0.1 gpm, the steam jet air ejector activity samples are taken and evaluated within thirty minutes. If the activity level continues to rise, SJAE samples will be monitored with sufficient frequency to indicate to plant management that either the 0.2 gpm limit will likely be exceeded or the activity level (and hence the leak rate) has stabilized. Blowdown samples will also be taken once a 0.1 gpm leak rate is exceeded, however, a tritium analysis does require a 90 minute turn around time. If the SJAE monitor is available and the leak rate exceeds 0.1 gpm, SJAE activity is closely monitored and verified by sampling. If it appears that the 0.2 gpm limit will be exceeded, the unit will be brought down. Again, the unit is brought down to 50% power within approximately one hour and to cold shutdown within 24 hours.

The plant has historically reacted to increasing leak rates in the above described manner. The plant's excellent record of detecting, confirming and reacting to primary-to-secondary leak rates demonstrates that procedures and instrumentation currently in place are adequate. When evaluating the monitoring program against the assumed tube fatigue failure leak rate vs. time curve, we believe that even with a disabled monitoring system, plant personnel will be able to discern the trend leading to fatigue failure and reduce the affected unit's power level well within the prescribed limits.

Action Request C, Part 2

Long Term Corrective Actions

(1) Due to the pervasiveness of the secondary side corrosion on Unit 2 at the tubesheet regions, the steam generators have been scheduled to be replaced beginning in the spring of 1988. The new generators in Unit 2 will incorporate stainless steel tube support plates, thus alleviating the denting and fatigue concerns resulting from carbon steel support plate corrosion. In addition, the new tube bundles will incorporate an enhanced AVB design which minimizes tube vibration in the U-bend region.

(2) To mitigate the effects of secondary side corrosion at the tubesheet region, the secondary side of the steam generators in both units is being treated with boric acid. In addition to retarding secondary side tube corrosion, boric acid treatments have been proven to inhibit the formation of magnetite and hence minimize denting. This boric acid treatment will continue on Unit 1.

(3) The previously described leak rate monitoring program will remain in effect.

(4) Due to hot leg secondary side corrosion concerns, the extent of future Unit 1 inspections will continue to be performed at levels well in excess of the Technical Specification requirements. Inspection of any tubes in rows potentially susceptible to North Anna type failures (specifically rows 8, 9 and 10) will include both the hot and cold leg 7th support plate intersections.

(5) If and when denting is realized within the potentially susceptible rows, AVB insertion depth will be assessed. Plugging and stabilization of potentially susceptible tubes or operational changes to reduce stability ratios will be implemented as required.

An alternative assessment of the potential for such a failure has been implemented in lieu of the actions described in subparagraph (a) and (b).

Subparagraph a

Subparagraph (a) suggests that "the analysis should 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." Westinghouse determined in their preliminary assessment (and reiterated in their final report October 1987) that "the ruptured tube is considered to have a worst case combination of loading conditions (i.e., highest flow peaking factor) and fatigue properties (i.e., denting)." Therefore, stability ratios for Cook have been calculated assuming that the worst flow peaking factor for Cook is the same as that for the failed tube at North Anna.

Both North Anna and the Cook units have Series 51 steam generators with 3388 mill annealed Inconel 600 (0.875 inch O.D. by 0.050 inch nominal wall thickness) tubes. For a given tube, the relative stability ratio is a function of flow conditions only (recirculation ratio, pressure, steam flow and flow peaking factor). For a R9C51 tube, the stability ratios relative to North Anna's failed tube are 0.91 and 0.92, respectively, for Cook Units 1 and 2 at 100% power. At the current administrative power limit of 90% and 80% on Unit 1 and Unit 2, the respective ratios are actually about 0.86 and 0.83.

Subparagraph b

Subparagraph (b) suggests that "the analysis would include an assessment of the depth of penetration of each AVB. The purpose of the assessment is two fold: (1) to establish which tubes are not effectively supported by AVB's and (2) to permit an assessment of flow peaking factors." Westinghouse's design of the series 51 steam generators at

Cook specified AVB insertion to and including row 11. Since the AVB insertion depth for the Cook steam generators includes rows 11 through 46, the only tubes of concern in terms of a North Anna type fatigue failure are those tubes in rows 1 through 10. Westinghouse's assessment further narrowed the rows of concern to rows 8 through 10. Of all the eddy current inspections carried out on both units, only two tubes were identified within rows 8 through 10 as having experienced denting. The two tubes detected were:

Unit	S.G.	Row	Column	Leg	Outages
1	11	10	19	H	5/85
1	12	10	58	H	5/85

Although these two potentially unsupported tubes were reportedly dented in May 1985, they were not found to be dented in the July 1987 inspection. Even though the 1987 inspection revealed that the denting observed in 1985 did not in fact exist, AEP for conservatism requested Westinghouse to review the eddy current tapes to determine whether these tubes were supported by AVB's. Westinghouse reported the following:

Cook Plant Anti-Vibration Bar Location Analysis

S/G	Row	Column	Number AVB's	Location	1st Seen	2nd Seen	7th C-L
11	10	19	2	#1 AVB	16.0*	19.5	
				#4 AVB	28.0	31.0	
				7-CL			50.1
12	10	58	2	#1 AVB	19.3	20.1	
				#4 AVB	27.9	29.0	
				7-CL			52.4

* Distances given in table represent axial distance (inches) from the 7th hot leg support plate.

This not only verifies that these tubes are supported by the lower set of AVB's, but, it also verifies that these tubes are supported on both sides of the tubes. (See attached Figure 1).

Verification of bilateral AVB support constitutes our criteria for determining whether a tube is effectively supported. This criteria shall be applied in the future analyses when a row 8, 9, or 10 tube is dented at the uppermost support plate.

Summary of Evaluation

A complete review of eddy current inspection data for evidence of denting at the 7th support plate has been completed. A total of two tubes were identified as dented in steam generators 11 and 12 in the May 1985 inspection that fall into one of the three rows sensitive to a North Anna type fatigue failure. These dents did not show up in the July 1987 inspection. The apparent explanation for the revised call is that recent signals are "clearer", thus allowing more definitive interpretation. The clearer data may be attributed to the removal of deposits by continuing secondary side boric acid treatment. Nevertheless, Westinghouse has verified that both these tubes are in fact effectively restrained by AVB's.

Bundle flows (both design and administrative operation) for steam generators in both the Cook units are below 90% of North Anna's. The relative stability ratio (for the R9C51 failed tube configuration) is below 90% for both the Cook units at current administrative power levels, and is only marginally above for full load operation (0.91 and 0.92, respectively).

Additional confidence in safe operation is gained by the knowledge that a 100% inspection on Unit 1 and a virtually 100% inspection on Unit 2 have been performed within the last 40 months, far in excess of Technical Specification requirements.

Long term corrective action will include:

- Replacement of the Unit 2 steam generators (with stainless steel support plates and improved AVB design);

- Continued secondary side boric acid treatment on Unit 1;

- Commitment to perform eddy current inspections through the 7th support plate on the cold leg within the rows of concern and continued eddy current inspections in excess of Technical Specification requirements on Unit 1.

- If denting is realized within the three critical rows, AVB insertion depth will be assessed. Plugging and stabilization or operational changes to reduce stability ratios will be instituted, if required.

Table 1

Inspection Extents

Unit	Outage	F/L	U-Bend[a]	7H	Total[b]
1	5/85	13493	0	0	13493
1	7/87	10147	716	2602	13465
2	9/85	12576	543	2[c]	13121
2	5/86	3823	77	9040	12940
2	3/87	447	173	12167	12787
2	9/87	1	113	12565	12679

a U-Bend inspections include the 7th cold leg tube support plate intersection.

b In all cases, this also represents the total number of tubes available for inspection.

c Tubes R9C45 and R9C46 were inspected through and including 7H. They have not been full length inspected in subsequent outages.

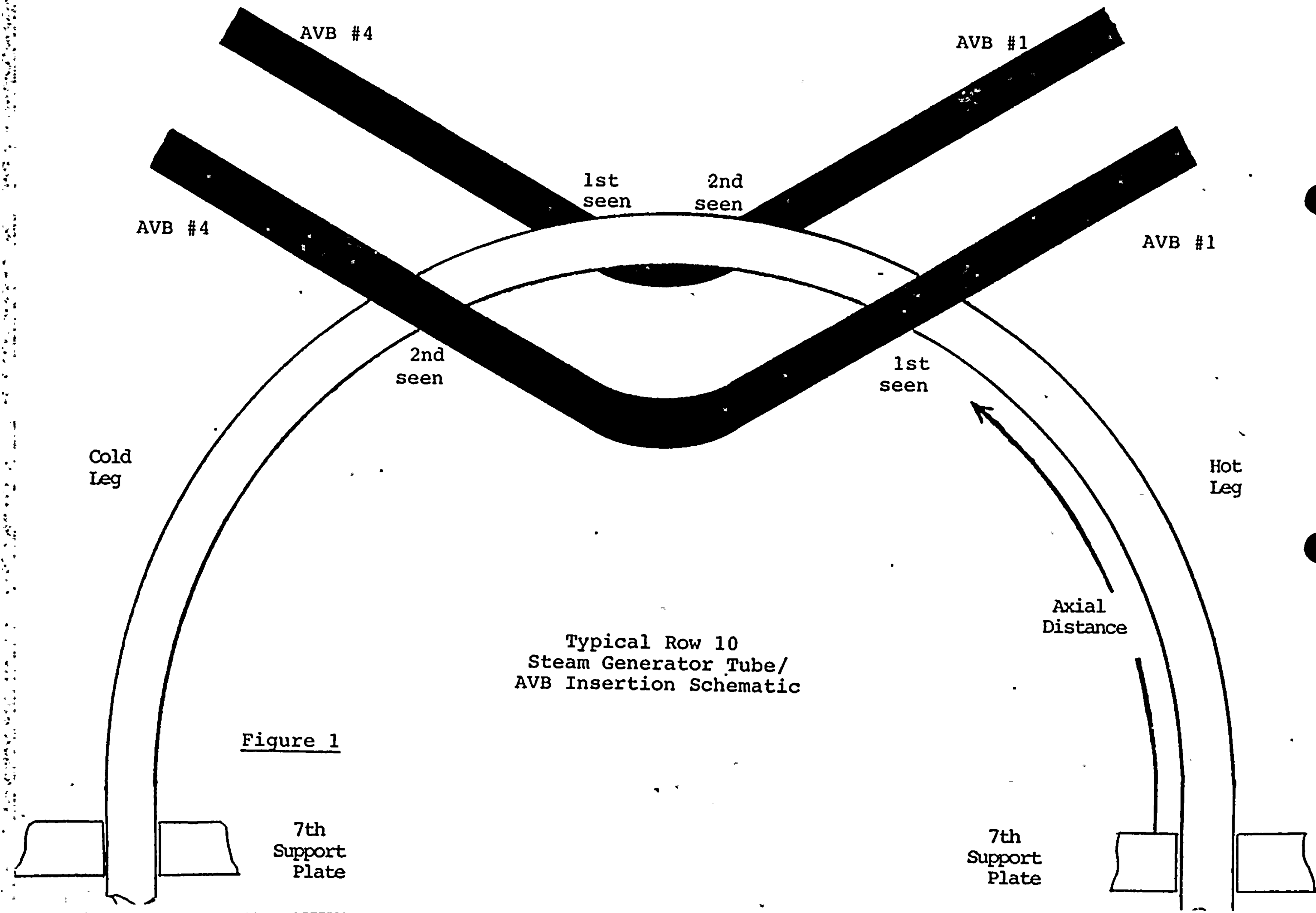

AVB #4
AVB #1
1st seen
2nd seen
AVB #4
AVB #1
2nd seen
1st seen
Cold Leg
Hot Leg
Axial Distance
Typical Row 10
Steam Generator Tube/
AVB Insertion Schematic
7th Support Plate
7th Support Plate

Figure 1

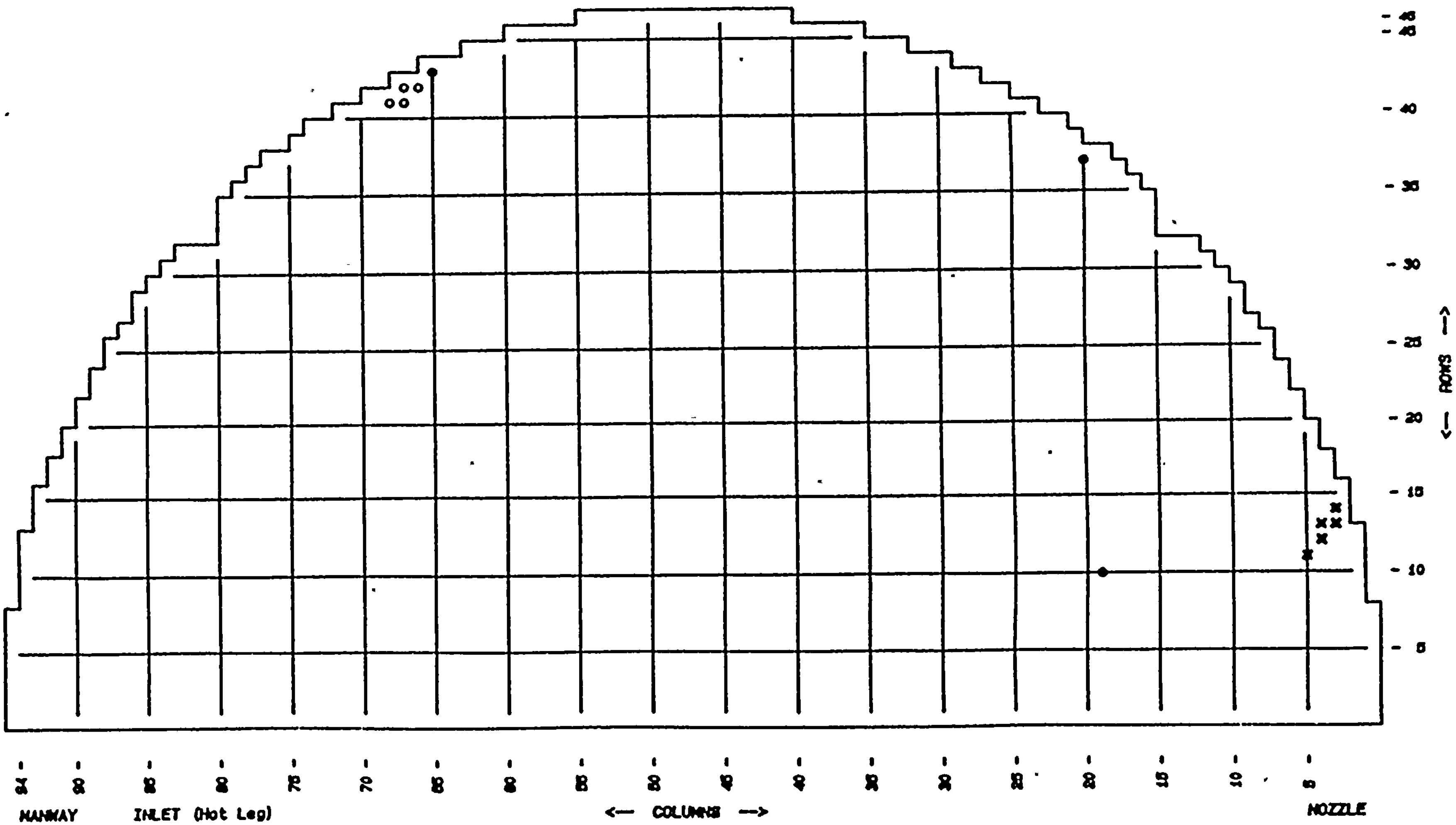
UPPERMOST TUBE SUPPORT PLATE DENTS (RESPONSE TO NRCB 88-02)

PLANT: DC COOK UNIT 1 GENERATOR: 11 TOTAL TUBES: 3388

X - DENTS - COLD LEG TOP TSP (5)

O - DENTS - HOT LEG TOP TSP (7)

TOTAL TUBES ASSIGNED: 12



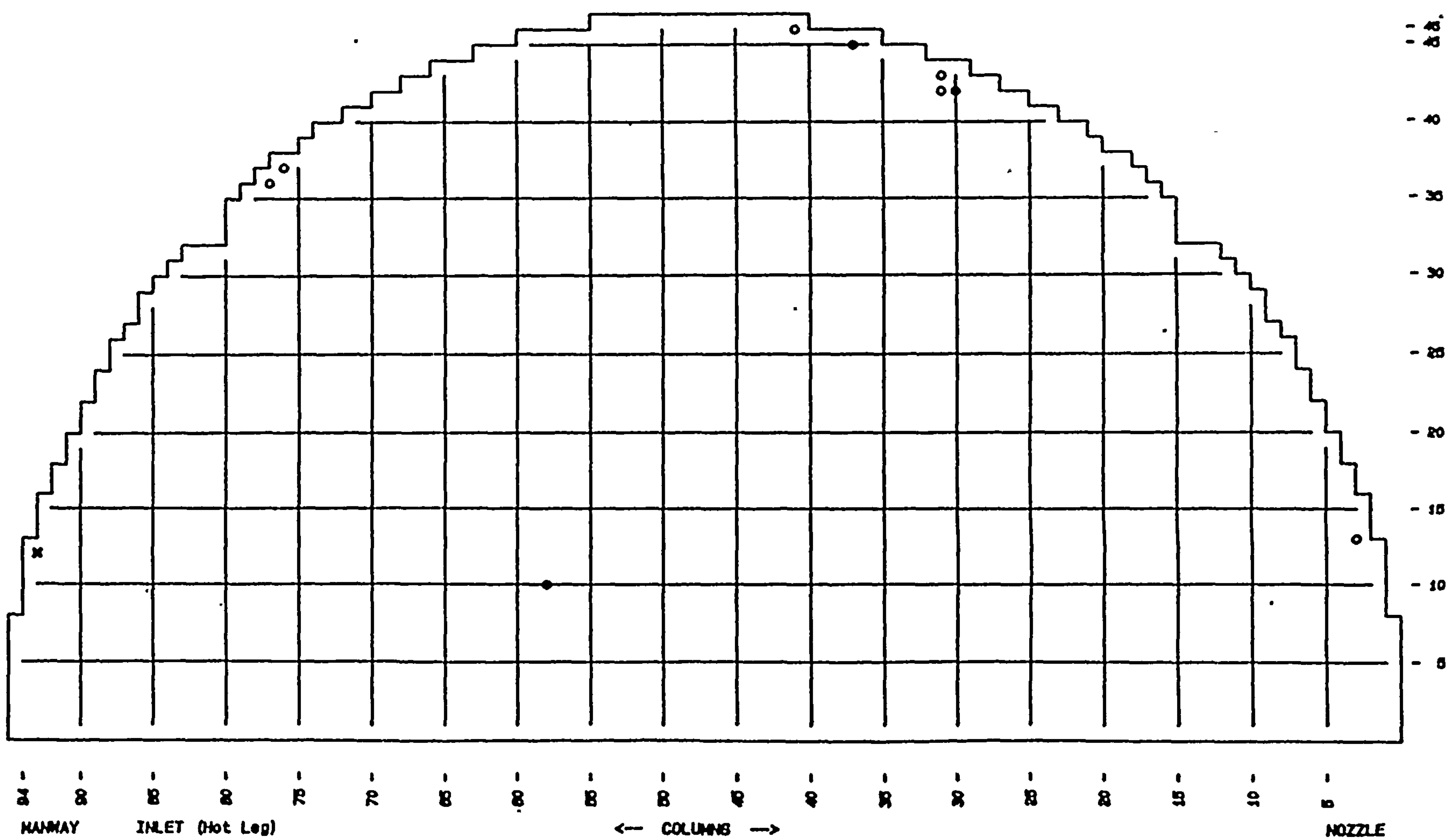
AEP TUBAN (TM) VER. 1.0 Thursday March 30, 1989 11:33 AM

UPPERMOST TUBE SUPPORT PLATE DENTS (RESPONSE TO NRCB 88-02)

PLANT: DC COOK UNIT 1 GENERATOR: 12 TOTAL TUBES: 3388

X = DENTS - COLD LEG TOP TSP (1) O = DENTS - HOT LEG TOP TSP (9)

TOTAL TUBES ASSIGNED: 10



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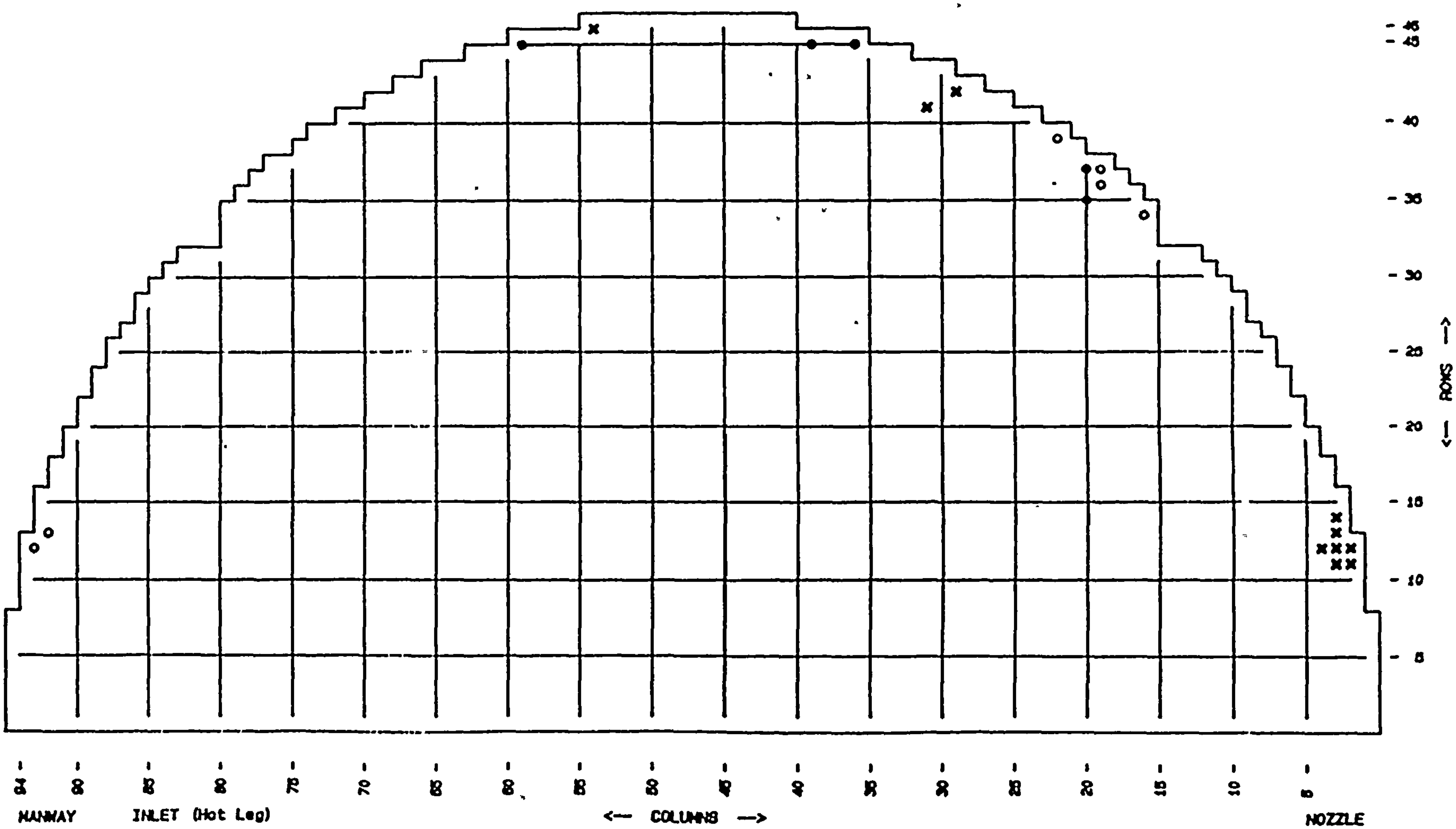
UPPERMOST TUBE SUPPORT PLATE DENTS (RESPONSE TO NRCB 88-02)

PLANT: DC COOK UNIT 1 GENERATOR: 13 TOTAL TUBES: 3388

X = DENTS - COLD LEG TOP TSP (10) O = DENTS - HOT LEG TOP TSP (11)

TOTAL TUBES ASSIGNED: 21



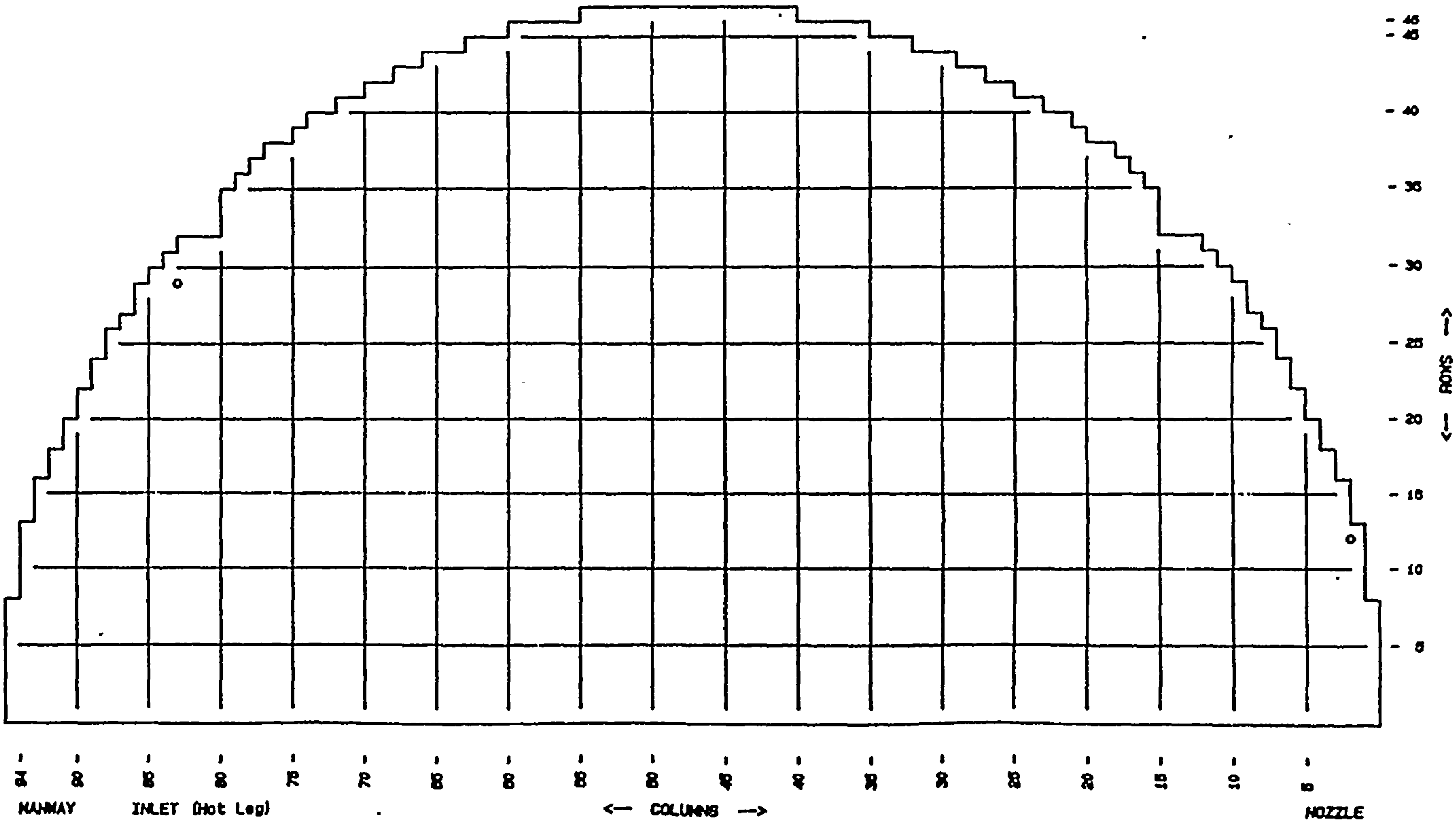
AEP TUBAN (TM) VER. 1.0 Thursday March 10, 1988 11:30 AM

UPPERMOST TUBE SUPPORT PLATE DENTS (RESPONSE TO NRCB-8802)

PLANT: DC COOK UNIT 1 .GENERATOR: 14 TOTAL TUBES: 3388

X = DENTS - COLD LEG TOP TSP (0) O = DENTS - HOT LEG TOP TSP (2)

TOTAL TUBES ASSIGNED: 2



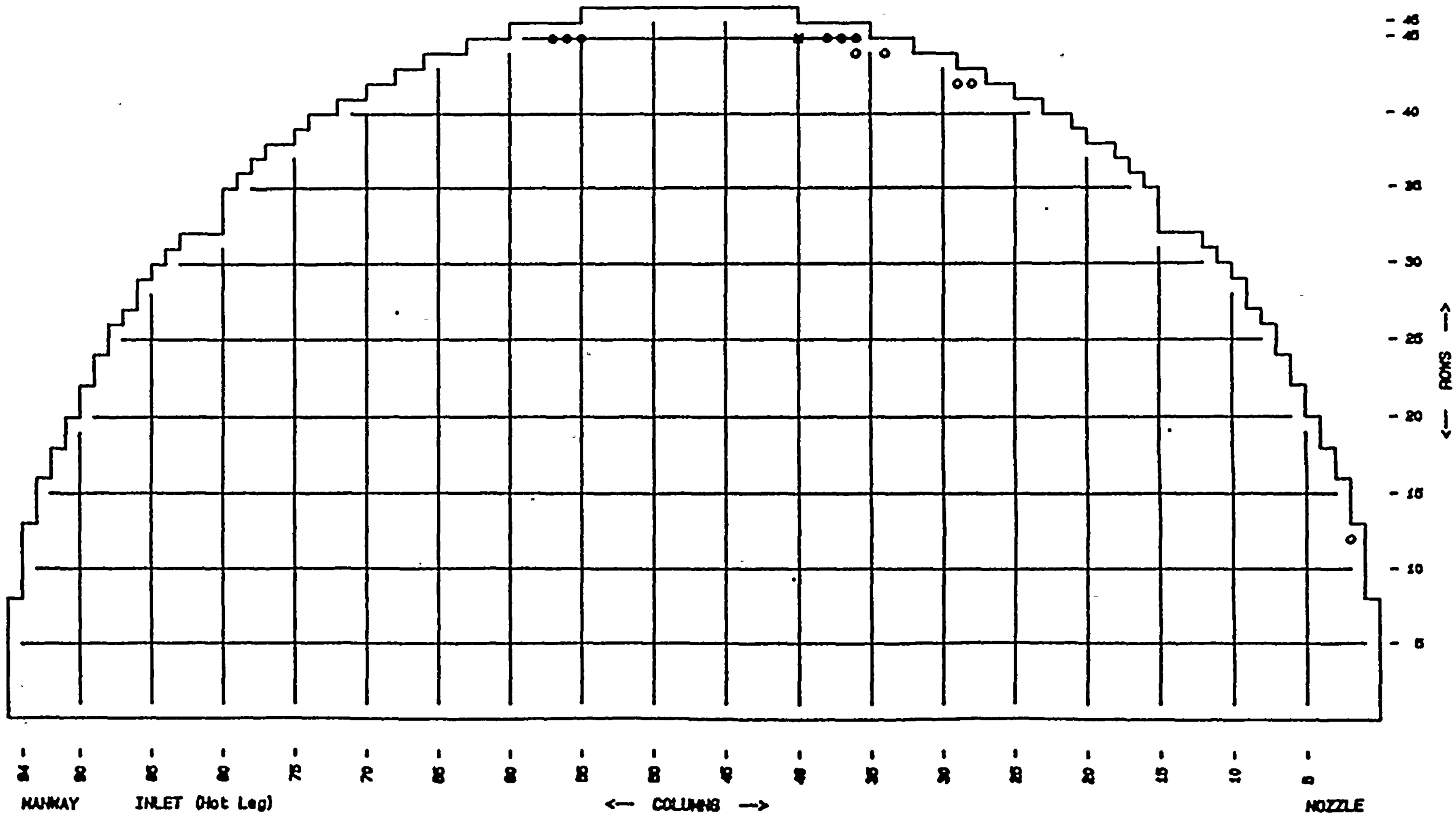
AEP TUBAN (TM) VER. 1.0 Thursday March 10, 1988 11:45 AM

UPPERMOST TUBE SUPPORT PLATE DENTS (RESPONSE TO NRCB 88-02)

PLANT: DC COOK UNIT 2 GENERATOR: 21 TOTAL TUBES: 3388

X = DENTS – COLD LEG TOP TSP (1) O = DENTS – HOT LEG TOP TSP. (11)

TOTAL TUBES ASSIGNED: 12



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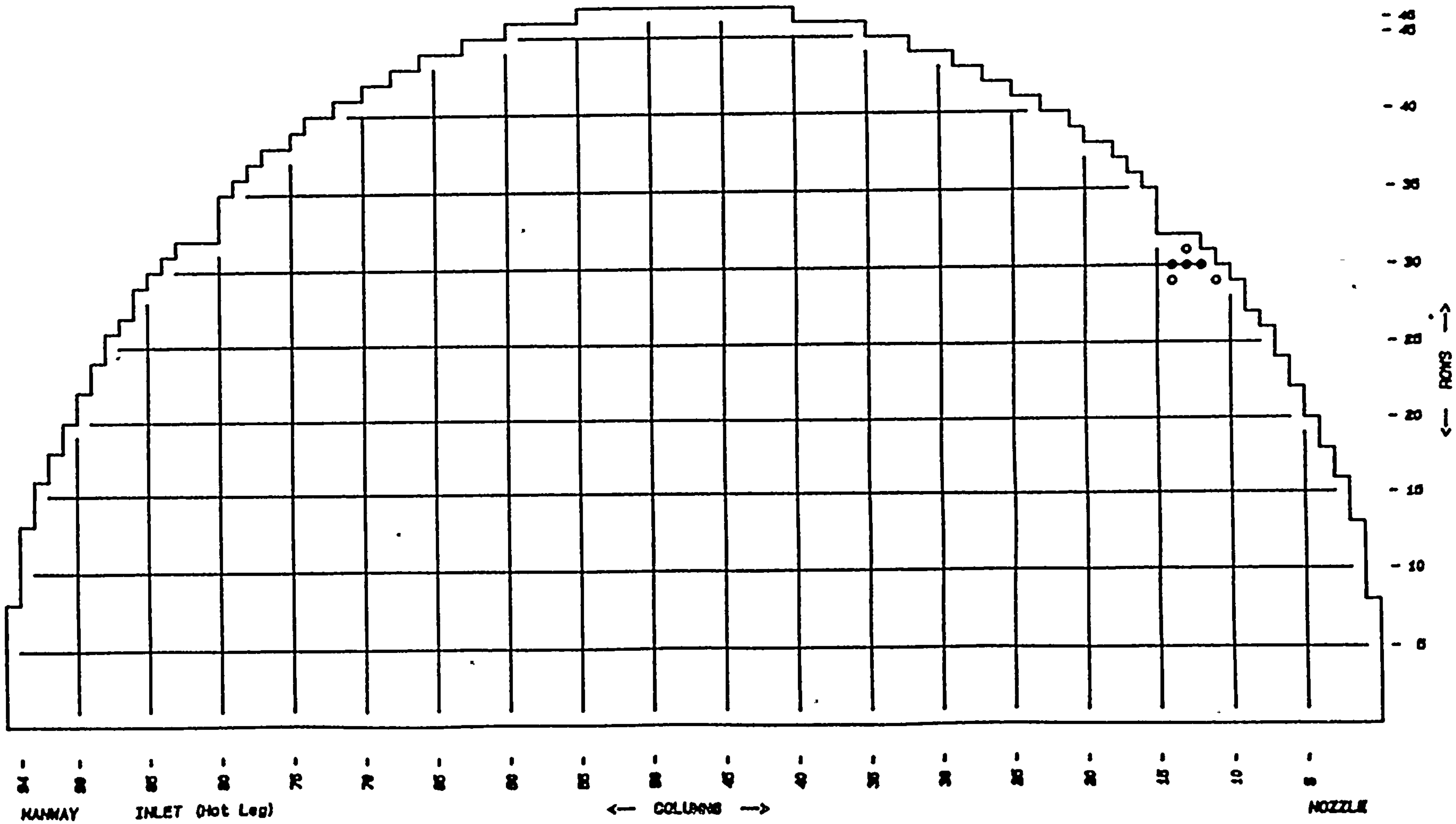
UPPERMOST TUBE SUPPORT PLATE DENTS (RESPONSE TO NRCB 88-02)

PLANT: DC COOK UNIT 2 GENERATOR: 22 TOTAL TUBES: 3388

X = DENTS – COLD LEG TOP TSP (0) O = DENTS – HOT LEG TOP TSP (6)

TOTAL TUBES ASSIGNED: 6

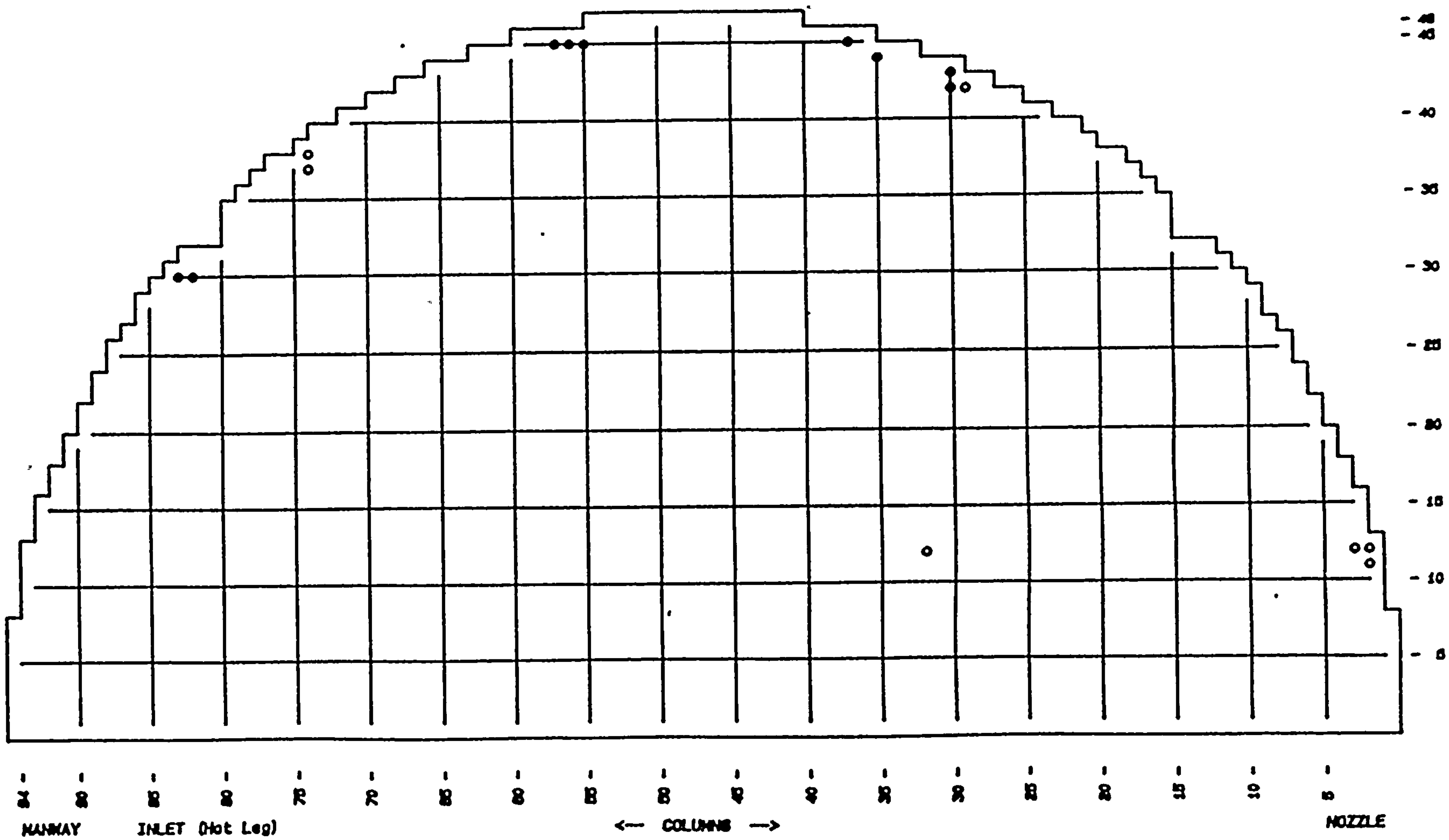


UPPERMOST TUBE SUPPORT PLATE DENTS (RESPONSE TO NRCB 88-02)

PLANT: DC COOK UNIT 2 | GENERATOR: 23 | TOTAL TUBES: 3388

X = DENTS - COLD LEG TOP TSP (0) | O = DENTS - HOT LEG TOP TSP (16)

TOTAL TUBES ASSIGNED: 16

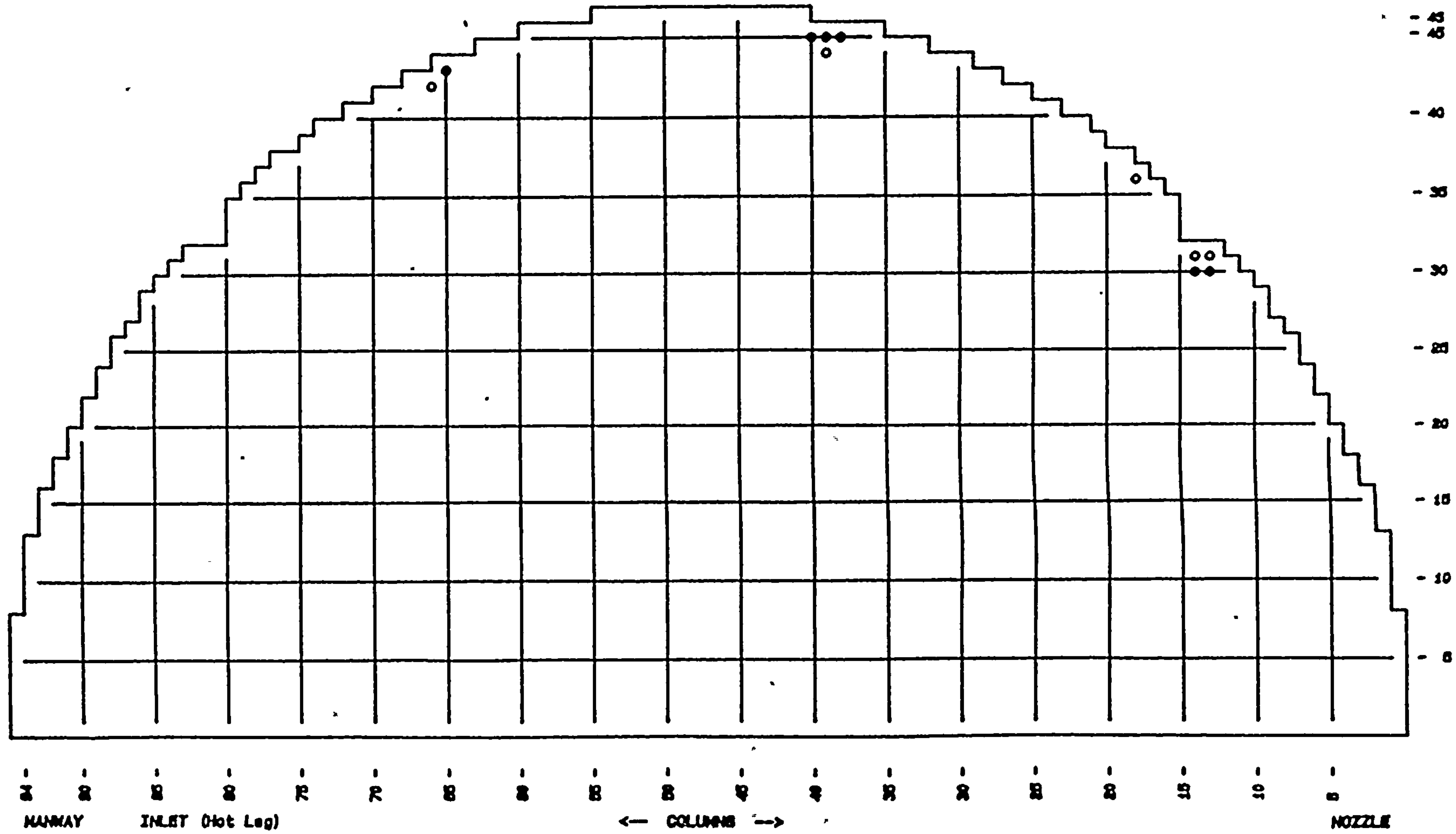


UPPERMOST TUBE SUPPORT PLATE DENTS (RESPONSE TO NRCB 88-02)

PLANT: DC COOK UNIT 2   GENERATOR: 24   TOTAL TUBES: 3388

X = DENTS - COLD LEG TOP TSP (0)   O = DENTS - HOT LEG TOP TSP (11)

TOTAL TUBES ASSIGNED: 11



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