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Carolina Power & Light Raleigh, NC E. E. Utley	7-10-74	7-18-74	X			

TO:	ORIG	CC	OTHER	
Karl R. Goller	2 signed	38		SENT AEC PDR XXX SENT LOCAL PDR XXX

CLASS	UNCLASS	PROP INFO	INPUT	NO CYS REC'D	DOCKET NO:
	XXX			40	50-261

DESCRIPTION:

Ltr trans the following re second fueling outage.......

PLANT NAME: H. B. ROBINSON UNIT #2

ENCLOSURES:

Results of i nspection (leaking tube).....

ACKNOWLEDGED

DO NOT REMOVE

(40 cys encl rec'd)

FOR ACTION/INFORMATION 7 -20-74 GMC

BUTLER (L)	SCHWENCER (L)	ZIEMANN (L)	REGAN (E)
W/ CYS	W/ CYS	W/ CYS	✓W/ CYS
CLARK (L)	STOLZ (L)	DICKER (E)	✓LEAR
W/ CYS	W/ CYS	W/ CYS	W/ 7 CYS
PARR (L)	VASSALLO (L)	KNIGHTON (E)	
W/ CYS	W/ CYS	W/ CYS	W/ CYS
KNIEL (L)	PURPLE (L)	YOUNGBLOOD (E)	
W/ CYS	W/ CYS	W/ CYS	W/ CYS

INTERNAL DISTRIBUTION

✓REG FILE	TECH REVIEW	DENTON	LIC ASST	A/T IND
✓AEC PDR	✓HENDRIE	GRIMES	DIGGS (L)	BRAITMAN
✓OGC	SCHROEDER	GAMMILL	GEARIN (L)	SALTZMAN
✓MUNTZING/STAFF	✓MACCARY	KASTNER	GOULBOURNE (L)	B. HURT
✓CASE	✓KNIGHT	BALLARD	KREUTZER (E)	
GIAMBUSSO	✓PAWLICKI	SPANGLER	LEE (L)	PLANS
BOYD	✓SHAO		MAIGRET (L)	MCDONALD
MOORE (L)(LWR-2)	✓STELLO	ENVIRO	REED (E)	CHAPMAN
DEYOUNG (L)(LWR-1)	✓HOUSTON	MULLER	SERVICE (L)	DUBE w/input
✓SKOVHOLT (L)	✓NOVAK	DICKER	SHEPPARD (L)	E. COUPE
✓GOLLER (L)	✓ROSS	KNIGHTON	SLATER (E)	
P. COLLINS	✓IPPOLITO	YOUNGBLOOD	SMITH (L)	✓D. THOMPSON (2)
DENISE	✓TEDESCO	REGAN	✓TEETS (L)	✓KLECKER
✓REG OPR	✓LONG	PROJECT MGR	WILLIAMS (E)	✓EISENHUT
FILE & REGION (3)	✓LAINAS		WILSON (L)	
✓MORRIS	✓BENAROYA	HARLESS		
✓STEELE	✓VOLLMER			

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✓1 - TIC (ABERNATHY)	(1)(2)(10)-NATIONAL LABS	1-PDR-SAN/LA/NY
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1 - ASLB	1-W. PENNINGTON, Rm E-201 GT	1-G. ULRIKSON, ORNL
1 - P. R. DAVIS	1-B&M SWINEBROAD, Rm E-201 GT	1-AGMED (RUTH GUSSMAN) Rm B-127 GT
✓16 - ACRS SENT TO LIC ASST TEETS 7-22-74	1-CONSULTANTS NEWMARK/BLUME/AGBABIAN	1-RD..MUELLER, Rm F-3[illegible] GT

50-261

CP&L

Carolina Power & Light Company

July 10, 1974

File: NG-3514

Serial: NG-74-797

DOCKETED USAEC JUL 18 1974 REGULATORY MAIL SECTION DOCKET CLERK

JUL 13 1974

Mr. Karl R. Goller
Assistant Director for Operating Reactors
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Goller:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
STEAM GENERATOR TUBE INSPECTIONS

During the second refueling outage at the H. B. Robinson Unit No. 2 Plant, a thorough inspection of each of the three steam generators was conducted to determine if any additional tube degradation had occurred since the last inspections in 1973. The results of these inspections are discussed below and are submitted for your information.

Prior to the outage, which began on May 6, 1974, a small primary-to-secondary leak of the order of 0.05 gpm was tolerated in the "A" steam generator. This leak was first identified on February 18, 1974, and was reported for your information on March 20, 1974, (letter E. E. Utley to J. F. O'Leary, Serial No. NG-74-335). Subsequent inspections during the outage identified the leaking tube as occurring in Row 42, Column 34 (hereafter tubes will be identified only by row and column number, such as 42-34) of the inlet side of the generator, in the U-bend region approximately two inches above the top (No. 6) tube support. Refer to Figures 1 and 3 for the location of the tube. Additional inspection of approximately 200 tubes in the peripheral portion of the inlet side revealed tube 43-33 with an 88% through-wall penetration, at the same axial location. No other tube degradations in this area were identified.

In our letters of December 3, 1973, and January 17, 1974, we supplied information to the AEC on a primary-to-secondary leak that developed on November 22, 1973, in the "C" generator. The leaking tube was 43-33 on the inlet side, two inches above the sixth support, or the same location as the 88% penetration in generator "A" noted above. Investigation of drawings of these areas revealed solid, wedge-shaped metal areas as shown in Figure 2, which are designed to support and locate the tube support plates. Since these could possibly be areas of flow stagnation which could lead to corrosive attack in the event of adverse chemistry, additional eddy current inspections were conducted in the U-bend area of the tubes adjacent to these areas. The

inspections revealed no other tubes with indications in generator "A" and none in generator "B." However, tube 42-35 in generator "C" had indications of greater than 50% degradation. Upon the advice of the vendor, Westinghouse, all tubes with identified eddy current indications in the U-bend region were plugged.

Concurrent with the investigations in the U-bend area, essentially all the tubes in the inlet side of each steam generator were inspected in the area from the tube sheet surface up through the first tube support plate. These inspections revealed a number of tubes in each generator with indications of greater than 20% through-wall penetration. The locations of these indications are shown in Figures 3, 4 and 5 for generators "A," "B" and "C," respectively. The breakdown as to the magnitude of the wall thinning and the associated number of tubes is shown in Table 1.

Justification was provided by the vendor, Westinghouse, for plugging only those tubes with indications greater than 50%. This justification is attached as Appendix I, and considers the transient effects of the steam break and loss of coolant accidents on primary-to-secondary differential pressures. In all, thirty-five tubes were plugged as a result of the eddy current inspections.

The eddy current results and removal of a tube section for laboratory examination indicated that tube degradation was a result of generalized wall thinning over short ($\sim$1") sections of the tubes, as opposed to the intergranular stress corrosion cracking that was characteristic of the degraded tubes found in 1972 and the 1973 refueling outage. This was the first indication of this mode of corrosive attack, since a complete inspection of the same tubes in the 1973 refueling outage revealed only two tubes with indications of intergranular stress corrosion cracking.

An explanation for the number of tube indications found in this outage was obtained by examining the history of the steam generator chemistry over the past several years. Prior to the May, 1972, outage in which the stress corrosion cracking effects were observed, the secondary chemistry was maintained with a Marcy-Halstead ratio of sodium to phosphate (Na/PO_4) greater than 2.8 for a significant fraction of the time. This allowed the formation of free caustic, and led to caustic attack of the tubes in low flow regions of the steam generator just above the tube sheet. After the outage, the chemistry specifications were changed to require that steam generator chemistry be maintained with a Marcy-Halstead Na/PO_4 ratio of 2.0 to 2.6. Chemistry data reveals that an actual ratio of 2.2 to 2.4 was maintained for the majority of the time up to the 1973 refueling outage. A measure of the success of this modification was the lack of further tube degradation when the tubes were inspected. Subsequent to the 1973 outage, records show that a ratio less than 2.2 was maintained for greater than 50% of the time up to December of 1973; the specifications were then adjusted to require a maintenance of the ratio between 2.3 to 2.6 on the recommendation of Westinghouse. The raising of the lower limit of the Marcy-Halstead ratio was deemed necessary because of the discovery of a chemical invariant point at a ratio of 2.18; below this point, a corrosive phosphate

solution was discovered to form when concentrated within regions of low flow. It is felt that the long period of operation in 1973 with ratios sometimes well below the invariant point of 2.18 led to the type of attack discovered during the last outage, both at the tube sheet location and also in the U-bend areas.

Carolina Power & Light Company is continuing to work closely with Westinghouse on the development of means to arrest continued tube degradation. We are hopeful that the maintenance of a Marcy-Halstead Na/PO_4 ratio between 2.3 and 2.6 will arrest further tube degradation and feel that our chemistry history on the steam generators and tube inspections during refueling outages support this thesis. More tube inspections will be conducted at other Westinghouse plants during the next several months. These plants have had controlled Na/PO_4 ratios in the range of 2.3 to 2.6 for a longer period of time, and we expect these results to provide assurance that the mechanism for significant tube degradation has been discovered and eliminated. If that is not the case, we will reevaluate the chemistry requirements for our steam generators and will keep you informed of any significant developments.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

DBW:mvp

Attachments
cc: Messrs. N. B. Bessac
T. E. Bowman
B. J. Furr
W. E. Graham
D. V. Menscer
D. B. Waters
R. A. Watson

TABLE 1

SUMMARY OF STEAM GENERATOR TUBE DEGRADATION FOUND IN 1974 OUTAGE

INDICATIONS BY EDDY CURRENT EXAMINATION

	20 - 29%	30 - 39%	40 - 49%	>50%
Steam Generator "A"	55	13	7	7
Steam Generator "B"	57	39	21	22
Steam Generator "C"	38	13	6	6

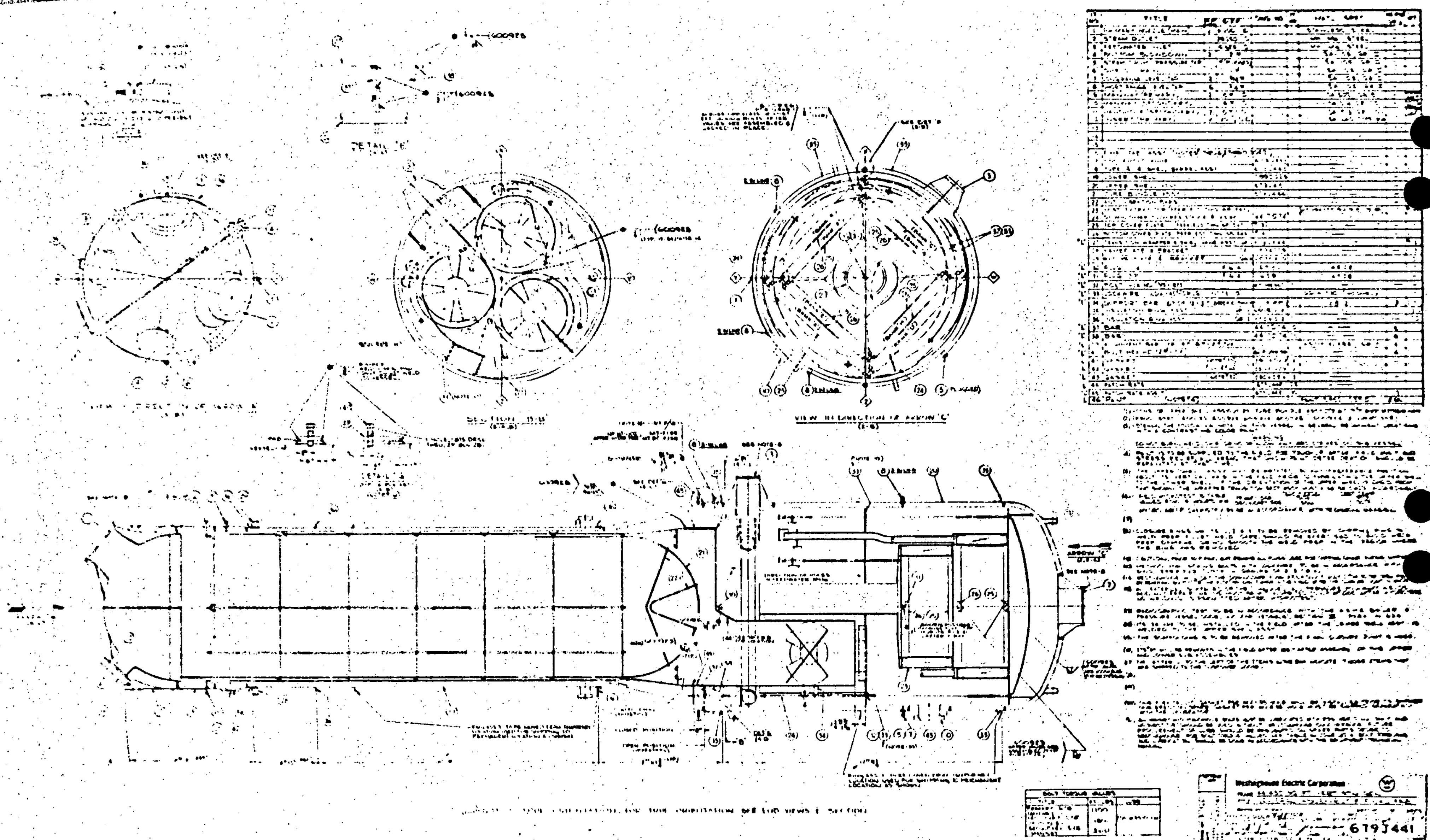


Figure 1

FIGURE 1-2 – General Assembly

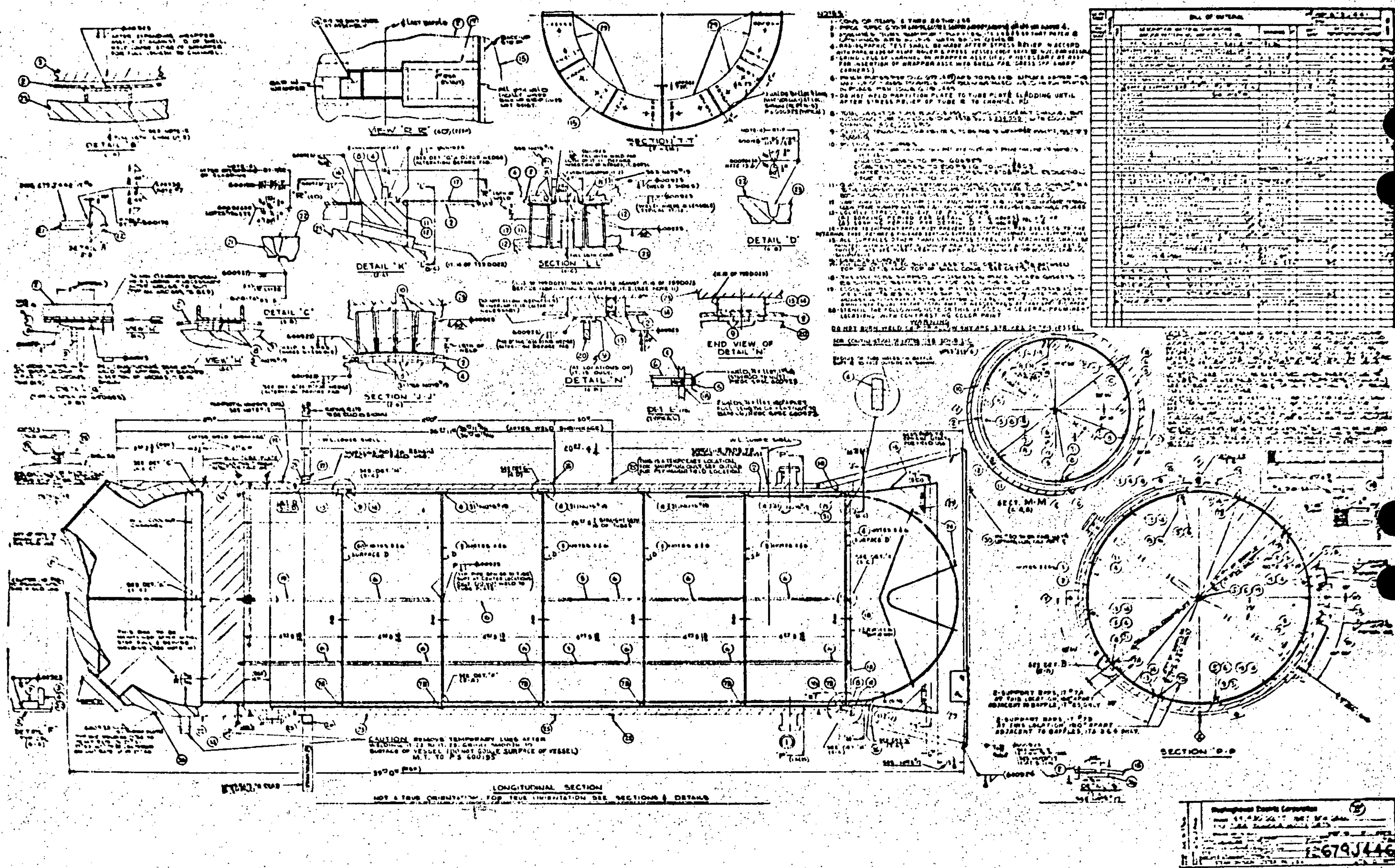
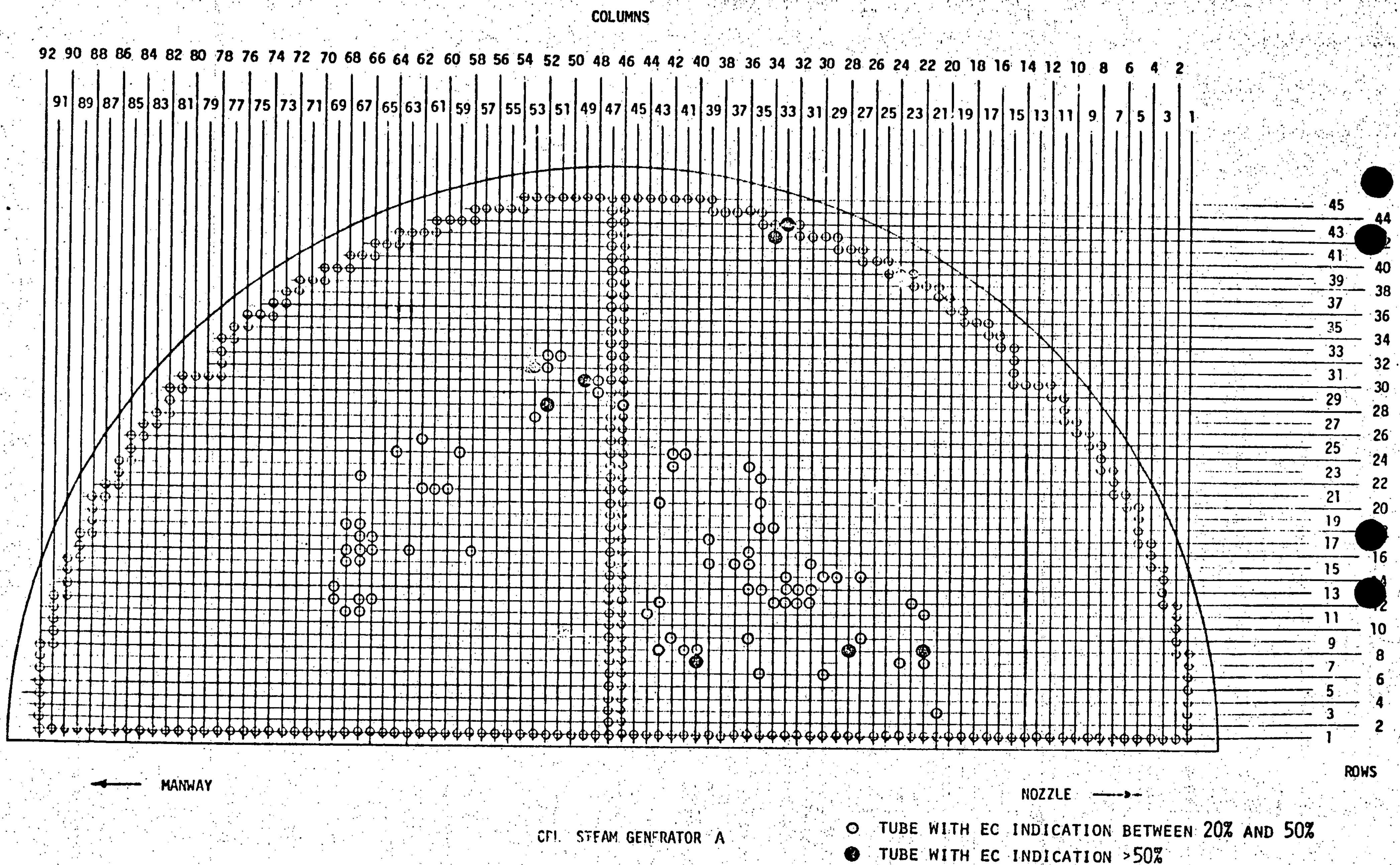


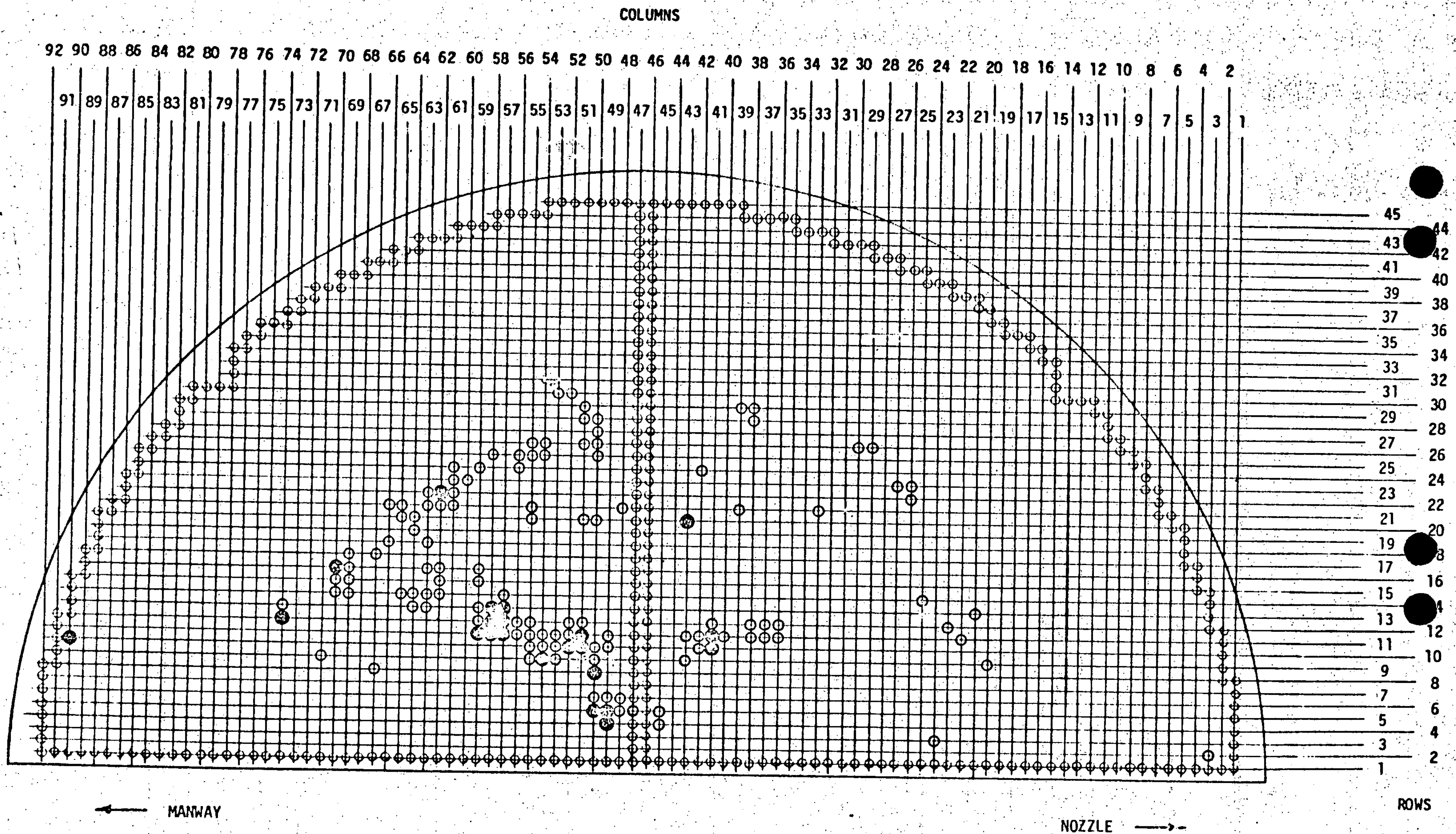
Figure 2

FIGURE 7–4 – Tube Bundle Assembly and Details



CFL STEAM GENERATOR A

FIGURE 3



O TUBE WITH EC INDICATION BETWEEN 20% AND 50%

● TUBE WITH EC INDICATION >50%

CPL STEAM GENERATOR B

FIGURE 4

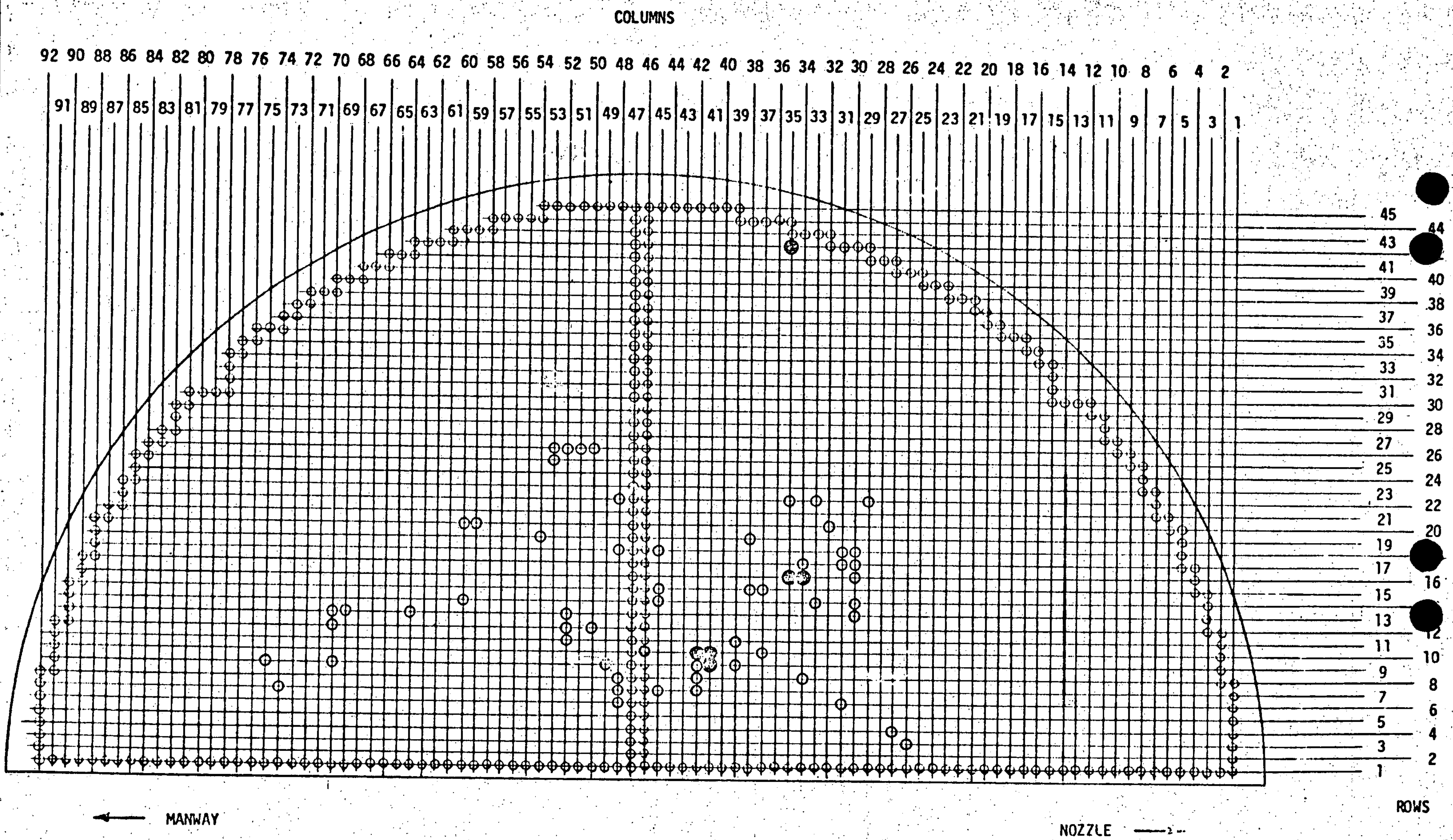


FIGURE 5

O TUBE WITH EC INDICATION BETWEEN 20% AND 50%

● TUBE WITH EC INDICATION >50%

CPL STEAM GENERATOR C

ATTACHMENT I

THE EFFECT OF LOCAL WALL THINNING ON THE INTEGRITY OF INCONEL STEAM GENERATOR TUBING

I. Summary

The allowable limit for localized wall reduction of Inconel 600 tubing used in steam generators has been established through both an experimental and analytical program. Production tubing was tested with simulated defects similar to, or more severe than that observed in plant tubing to define the relationship of strength to the configuration of the defect. Various factors were considered in the analysis including the properties of the tubing and the combined loadings occurring during accident conditions. It was concluded that a tube with a localized minimum wall thickness of 25% of the original wall (75% reduction) was sufficient to withstand the worst combination of loading conditions for a main steam line break or a LOCA. Accordingly, a criteria has also been established to plug all tubes in the steam generators with eddy current indications which show penetration greater than 50%. It is intended that the tubes, where indications have been observed, will be reinspected at a later date.

II. Description of Defects

With few exceptions, the eddy current indications in the H. B. Robinson Unit No. 2 steam generators are located just above the tube sheet. Tubes with similar eddy current indications removed from other units indicate that local wall thinning on the O.D. surface has occurred at similar locations. Laboratory standards and correlation with measurements of tubes removed from several plants have been the basis of establishing the accuracy of the eddy current inspection technique. In the range of 40% to 50% penetration the tendency is generally to over estimate the amount of the penetration of the tube in the inspections performed in the field.

III. Mechanical Test Program

The ASME Code provides techniques for analyzing the capability of the tubing for a uniform wall thickness. In order to determine the effect of localized corrosion on the strength of tubes a comprehensive series of mechanical burst tests have been performed on straight sections of 7/8 inch diameter Inconel-600 tubing. The tubing was standard production grade Inconel material which was obtained from Westinghouse Specialty Metals Division. In this series of tests a flat was machined on the tube samples to simulate the effect of broad thinning of the tube. These flats (defects) were of varying lengths, i.e., approximately 1/2, 1, 1-1/2, 2, and 9 inches long, to permit a determination of burst strength as a function of defect length.

The tests on the 7/8 inch tubing (0.867 inch x 0.048 inch wall) were performed at room temperature and with approximately 12 or 25 mils of wall remaining. These remaining wall thicknesses correspond to approximately 25% and 55% of the original no-defect wall thickness.

The results of this series of tests are presented in Figure 1 as burst pressure versus reciprocal flaw length. The slope of these curves is essentially the same as those obtained from a similar program where partial through-wall slots were machined in the tubing. The results of this series of tests on 7/8 inch tubing were as expected and they tend to confirm the results of the previous tests.

It will be noted from this data that the burst pressures at room temperature are in the range of 3700 to 4200 psi for tubing with 1 to 2 inch long flaws and with approximately 25% of the original wall remaining. The burst pressure increases to about 6000 psi when the flaws are in the 1/2 inch long range. For comparison the burst pressure for normal tubing is approximately 11,000 psi.

The effect of service temperatures and minimum code allowable properties are also indicated on Figure 1 for 25% wall remaining. It will be noted that the strength of the tube to withstand internal pressure is substantial even with as little as 25% of the tube wall remaining.

IV. Transient Loadings - Main Steam Line Break

The normal operating internal pressure across the tube varies somewhat as a function of power level, in the range of 1200 to 1400 psi.

The large steam line break accident produces a potential condition where the steam side pressure can go to nearly atmospheric conditions when the entire contents of the steam generator are discharged. Coincidental with this condition on the steam side, the reactor pressure will also decrease due to the cooling of the reactor coolant. The pressure differential increases shortly after the break and decreases after the pressurizer empties. In the limit, with atmospheric pressure on the steam side, the reactor pressure would ultimately be limited by the code safety valves on the Reactor Coolant System to approximately 2500 psi, which is the design pressure of the system. A sequence of events to produce this condition would be extremely difficult to produce in the system. For purposes of this analysis, however, a 2500 psi differential will be assumed as a limiting maximum condition.

Referring to Figure 1, the curve indicated as minimum tube properties at service temperature with 25% wall, it will be noted that with a 2-inch long flaw the tubing will withstand 2500 psi internal pressure. Shorter defect lengths and nominal material properties would increase the allowable pressure loading on the tube.

For this limiting condition it is concluded that the minimum acceptable wall would be equivalent to 25% of the original tube wall based on a two-inch long broad-area defect, considering that the tubing has minimum mechanical properties. A review of material test reports for steam generator tubing has indicated that the tubing is consistently produced with nominal properties providing additional margin for this analysis.

V. Loss of Coolant Accident

WCAP 7832 "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plant Under Combined LOCA plus SEE Conditions" was recently submitted to the AEC. This analysis considered the possible pressure, hydraulic, and seismic loadings in the tubes and the tube bundle during the combined loss of coolant accident and seismic occurrence. The analysis presented in this report was specifically for a 3/4 inch O.D. tube, and the conclusion was, for this condition, a uniform minimum wall of .026 inches would maintain stress levels below yield and code allowable limits. The corresponding values for the 7/8 inch diameter tubing used in the Robinson steam generators has been determined to be .021 inches (40% tube wall). The limiting stress condition was determined to occur in the U-bends due to the bending moment imposed by the discharing fluid.

Referring to the analysis in this report for various nodal points in the tube bundle, it should be noted that the bending moments in the region of the tube sheet are so small as to be negligible. At the tube sheet the primary stresses in the tube are caused by the initial pressure differential between primary and secondary sides which is at its highest value at time t = 0. This would correspond to the normal differential pressure across the tubes at full load of approximately 1400 psi. Accordingly, in this region where the corrosion has actually occurred the results provided in Figure 1 could similarly be applied and the 25% wall thickness is fully sufficient.

Ultimately for the LOCA condition the pressure in the reactor coolant system would approach containment pressure and the tubes would experience external pressure due to the remaining steam in the steam generator. Extensive experimental work has been performed on the collapse pressure of 7/8 inch O.D., .050 wall inch thickness tubes to determine an analytical correlation to calculate collapse pressure for this size tubing. Applying this analysis, a thinned 7/8 inch tube with 40% wall remaining (0.021 inches) is capable of sustaining an external pressure at design temperature of approximately 1300 psi for 1% ovality of the tubing as compared to approximately 850 psi full load steam pressure. These results would correspond to a tube which is uniformly thinned over its entire length and not for the localized thinning observed in the plant. The collapse pressure would be higher with a locally thinned area of the same remaining wall.

VI. Conclusion

Based on the testing program performed and analysis presented in WCAP 7832 it is concluded that a minimum wall thickness of 25% (75% penetration) in the region of the tube sheet will withstand all the loading conditions imposed on the tubing by the LOCA and steam line break conditions.

To provide additional margin criteria has been established to plug all tubes with eddy current indications in excess of 50% wall penetration in the region of the tube sheet. This technique, from past experience, often over predicts the amount of wall reduction. The margin between the allowable wall thickness of 25% and 50% indicated penetration from the eddy current inspection provides an additional margin.

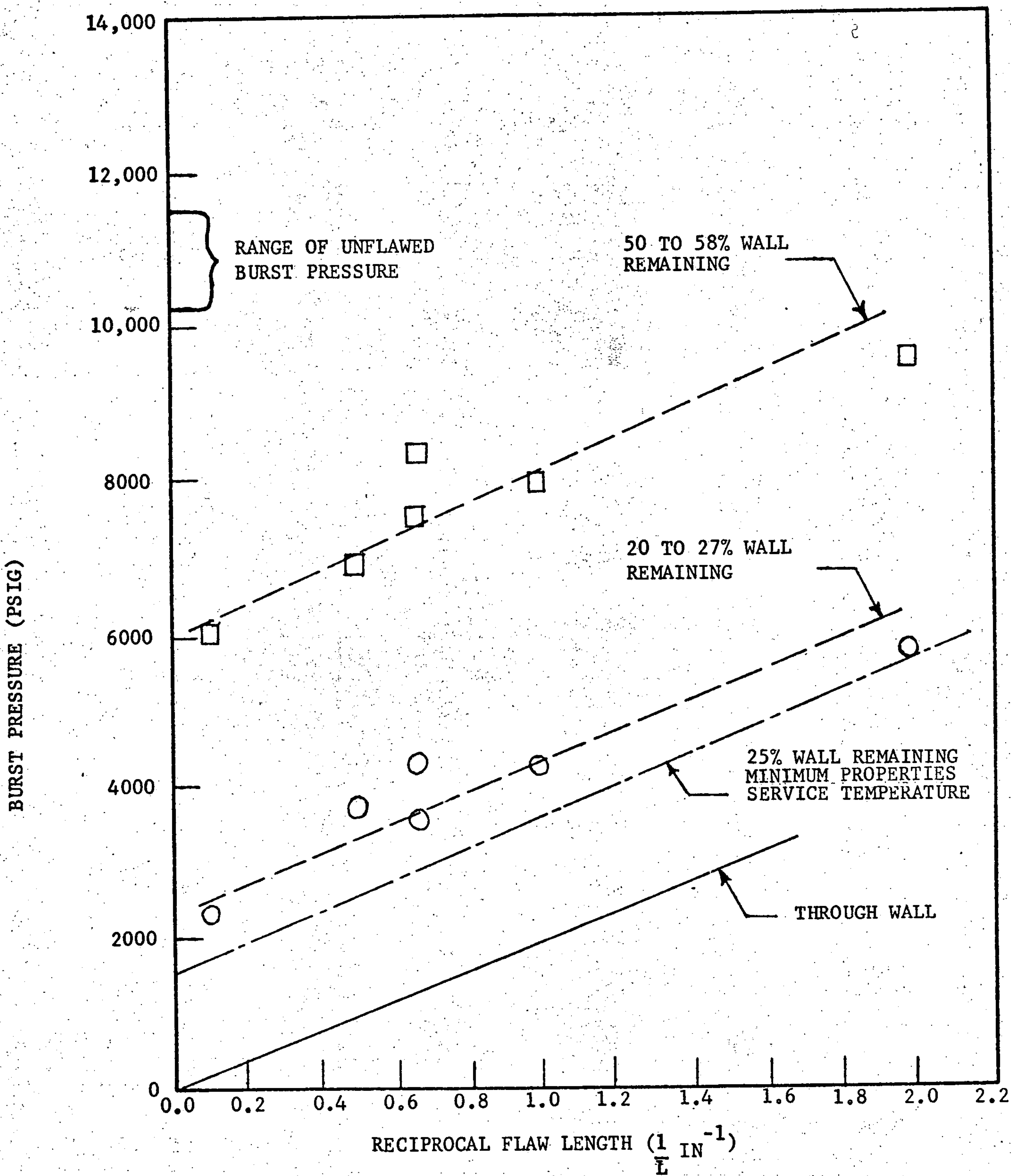


Figure 1. Room Temperature Burst Pressure of 7/8" Tubes with Machined Flats