AEC DISTRIBUTION FOR PART 50 DOCKET MATERIAL
(TEMPORARY FORM)

CONTROL NO: 5979

FROM: Carolina Power & Light Co. Raleigh, N.C. 27602 E.E. Utley	DATE OF DOC: 10-27-72	DATE REC'D 11-2-72	LTR X	MEMO	RPT	OTHER
TO: Mr. D. J. Skovholt	ORIG 1 signed	CC	OTHER	SENT AEC PDR ✓ SENT LOCAL PDR ✓		
CLASS: U/PROP INFO	INPUT	NO CYS REC'D 1		DOCKET NO: 50-261		

FILE

DESCRIPTION: Ltr re our 10-5-72 ltr...trans the following:

ENCLOSURES: REPORT-Description of Present Status of Steam Generators for H.B. Robinson Unit 2.....

(1 cy encl rec'd)

DO NOT REMOVE

ACKNOWLEDGED

PLANT NAMES: H.B. Robinson Unit 2

FOR ACTION/INFORMATION DL 11-2-72

BUTLER(L) W/ Copies	SCHWENCER(L) W/ Copies	✓SCHEMEL(L) W/ 9 Copies	KNIGHTON(E) W/ Copies
CLARK(L) W/ Copies	STOLZ(L) W/ Copies	ZIEMANN(L) W/ Copies	YOUNGBLOOD(E) W/ Copies
GOLLER(L) W/ Copies	VASSALLO(L) W/ Copies	CHITWOOD(FM) W/ Copies	REGAN(E) W/ Copies
KNIEL(L) W/ Copies	H. DENTON W/ Copies	DICKER(E) W/ Copies	W/ Copies

INTERNAL DISTRIBUTION

✓REG FILE	TECH REVIEW	✓VOLLMER	HARLESS	WADE E
✓AEC PDR	✓HENDRIE	DENTON		SHAFER F & M
✓OGC, ROOM P-506A	✓SCHROEDER	GRIMES	F & M	BROWN E
✓MUNTZING/STAFF	✓MACCARY	GAMMILL	SMILEY	G. WILLIAMS E
✓CASE	✓LANGE	KASTNER	NUSSBAUMER	E. GOULBOURNE L
✓GIAMBUSSO	✓PAWLICKI	BALLARD		A/T IND
BOYD-L(BWR)	✓SHAO	FINE	LIC ASST.	BRAITMAN
✓DEYOUNG-L(PWR)	✓KNUTH		SERVICE L	SALTZMAN
✓SKOVHOLT-L	✓STELLO	ENVIRO	MASON L	
P. COLLINS	✓MOORE	MULLER	WILSON L	PLANS
	✓HOUSTON	DICKER	MAIGRET L	MCDONALD
REG OPR	✓TEDESCO	KNIGHTON	SMITH L	DUBE
✓FILE & REGION (2)	✓LONG	YOUNGBLOOD	GEARIN L	
✓MORRIS	✓LAINAS	PROJ LEADER	DIGGS L	INFO
✓STELLE	✓BENAROYA		TEETS L	C. MILES
		REGAN	LEE L	

EXTERNAL DISTRIBUTION

✓1-LOCAL PDR Hartville, S.C.
✓1-DTIE(ABERNATHY)
✓1-NSIC(BUCHANAN)
1-ASLB-YORE/SAYRE WOODWARD/H. ST.
✓16-CYS ACRS [redacted]

SENT TO LIC. ASST. S. TEETS ON 11-2-72

(1)(5)(9)-NATIONAL LAB'S
1-R. CARROLL-OC, GT-B227
1-R. CATLIN, A-170-GT
1-CONSULANT'S NEWMARK/BLUME/AGABIAN

1-PDR-SAN/LA/NY
1-GERALD LELLOUCHE BROOKHAVEN NAT. LAB
1-AGMED(WALTER KOESTER, Rm C-427, GT)
1-RD...MULLER...F-309GT

50-261

Regulatory File Cy.

Carolina Power & Light Company

Raleigh, North Carolina 27602

October 27, 1972

File: Robinson 2-0-4

Mr. D. J. Skovholt
Asst. Director for Operating Reactors
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
INTEGRITY OF STEAM GENERATORS UNDER ACCIDENT CONDITIONS

Dear Mr. Skovholt:

We have reviewed your letter of October 5, 1972, regarding the existence of an unreviewed safety question in the operation of our H. B. Robinson Unit No. 2 plant, and are of the opinion that adequate assurance exists to ensure the continued operation of the plant in its present state with no increase in risk to the health and safety of the public.

The increase in the pressure differential between the primary and secondary side of the steam generators due to a steam line break or other similar accident will not cause catastrophic failure of the steam generator tubes in their present condition. Various mechanical tests on steam generator tubing, coupled with Robinson Plant experience, are summarized in the attached document, and show that this is indeed the case for the transient conditions and the flaw sizes under consideration. The integrity of the fuel clad, however, cannot be assured under the assumed accident condition of steam line break with collapsed clad areas. Thus, one of the two barriers to fission product release during this accident cannot be assumed to exist as in the Final Safety Analysis Report. However, the integrity of the steam generators is maintained and is sufficient to prevent an increase in the consequences of a steam line break accident.

In addressing the Commission's concerns specifically, the attached document shows:

1. The present operation of the Robinson steam generators shows no indication of any detectable primary-to-secondary leakage. Corrective actions to the previous mechanical and chemical operation of the plant were implemented after the tube plugging operations in May and June, 1972, and the absence of recurring leakage is indicative of the success of these actions in arresting further degradation of the tubes.

5979

2. The plugging operations involved all tubes which had eddy current indications. The lower limit of detectability of this method is approximately 25-30% penetration in the tube wall so that the tubes plugged had penetrations greater than this value. We are confident that all tubes with penetrations greater than 50% were found and plugged, based on the testing method and the pattern of tubes tested in each generator.

3. Mechanical tests at operating temperatures of new and service exposed tubes show that, for the maximum flaw length found in the Robinson Plant, the pressure differential required to cause plastic failure of a through-wall flaw is at least 500 psi above the maximum pressure experienced during a steam break accident. Test results also indicate that a flaw of this type and size would exhibit a primary-to-secondary leakage some four times greater than that allowed under present interim technical specification limits which require plant shutdown. The presence of wall thickness (or partial through-wall penetration of the flaw) serves to increase the margin between burst pressure and maximum pressure resulting from the accident.

4. The pressure increase during the accident takes place initially over a period of some ten seconds; this is not considered "rapid" in the sense of a shock wave or pressure pulse producing a "rapid" increase, or spike, in pressure. The time difference between the accident pressure transient and the pressure transients used in the mechanical tests is therefore of little significance.

Based on the above evidence, it is concluded that the steam generator tubes are not substantially degraded from their condition as of June 9, 1972, and that the existing steam generator leakage limitation and its frequency and accuracy of measurement provides assurance that the structural integrity of the steam generator tube is unchanged from that assumed in the H. B. Robinson Unit No. 2 Final Safety Analysis Report, that no increase in surveillance or special hydrostatic tests are required to substantiate this position and that no unreviewed safety problem exists.

Yours very truly,

E. E. Utley
Vice President
Bulk Power Supply

DBW/za

Attachment

cc: Mr. C. D. Barham
Mr. N. B. Bessac
Mr. B. J. Furr
Mr. S. Grant

DESCRIPTION OF PRESENT STATUS OF STEAM GENERATORS

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

A. STEAM GENERATOR INSPECTIONS

On May 14, 1972 the H. B. Robinson II plant was shutdown because of an increase in primary to secondary leakage in steam generator "A". The leakage in steam generator "A" increased from 55 gallons per day, which had existed since October 1971, to approximately 10 gallons per minute while returning to hot standby conditions following a maintenance shut-down.

Steam Generator "A" was visually inspected following a normal plant cool-down and after draining the secondary side of the generator to within 4 inches above the tube sheet. The inspection revealed a single leaking tube on the inlet side of the generator in row 7, column 43. No other leaks were noted.

An eddy current (EC) test program was initiated in steam generator "A" to characterize the defect in the subject tube and to determine the condition of surrounding tubes.

A thorough EC (Eddy Current) inspection of the leaking tube, row 7, column 43, revealed a through-wall defect just above the secondary side of the tube sheet (approximately 3 inches). No other EC indications were observed in the inspection of this tube which covered the length of tube from the inlet end up through the first tube support.

The sensitivity of this non-destructive testing procedure is such that indications greater than 25-30% penetration in the tube wall can be detected. It is therefore possible that indications less than this lower sensitivity level may exist in the unit, however, as discussed in following sections, these tubes have substantial strength.

The EC test program for steam generator "A" provided for inspection of 2029 inlet side tubes up to and through the first tube support. The area of the tube sheet tested is shown graphically in Figure 1. Of the total 2029 tubes tested, 23 tubes were found to contain eddy current indications of varying degrees of outside-diameter originated wall penetration. Table 1 lists the EC indications observed during the inspection of steam generator "A".

In addition to the testing of "A" generator tubes, an inspection program was carried out in the "C" and "B" units. The inspection patterns were established to give a comprehensive inspection of those areas of the generator where indications have been observed during previous inspections and also to avoid unnecessary deliberate exposure of personnel. These patterns are shown in Figures 2 and 3. A total of 1588 tubes were inspected in "C" inlet with 2 EC indications observed and 864 tubes inspected in "B" with four indications noted. These indications are tabulated in Table 1.

In summary, a total of 4,481 tubes were inspected with the eddy current technique in the three steam generators and 29 tubes were found to have positive indications. Three tubes were removed from steam generator "A" for further examination and all tubes with indications and the three locations where tubes were removed were plugged. The unit was subsequently leak tested from the secondary side at 800 psi.

During the primary side inspection and repair, modifications were made to the secondary side. These consisted of raising the downcomer resistance plates in each generator to increase and restore the original recirculation flow in each generator and the installation of new orifice plates in the swirl vane assemblies to improve their moisture separation efficiency.

Subsequent metallographic examination of the tubes removed from the unit revealed that they had been subjected to intergranular attack. The cracks were very tight and ran longitudinally along the axis of the tube similar to indications observed in tubes removed from this area in other units.

The unit was subsequently returned to service on June 9 and has been in operation since that date at an operating pressure differential across the tubes of approximately 1400 psi. The primary to secondary leak rate since these repairs were made has been at the limit of detectability (0-6 gpd). This is the most positive assurance that the action taken to correct further degradation of the tubes has been effective.

B. PROBABLE CAUSES AND CORRECTIVE ACTION.

Detailed examination of the tube samples removed from the Robinson steam

generators revealed an intergranular attack originating on the OD surface of the tubes. This attack was most probably a form of caustic stress corrosion.

A review of the generator secondary side operating history identified several factors which may have enhanced the environment contributing to the intergranular attack on the tubes. These include:

a) reduced generator circulation ratio
b) operation with condenser in-leakage without concurrent generator blowdown
c) operation with frequent out-of-specification chemistry conditions
d) phosphate concentration (10 ppm) which was inadequate in the presence of a), b) and c) above.

The corrective action that was implemented included:

a) plugging of all generator tubes which had positive eddy current indications
b) return of the steam generator circulation ratio to original design conditions
c) increase the minimum generator phosphate level to 20 ppm
d) continuous feed of phosphate, balanced by continuous blowdown

C. MECHANICAL TESTING PROGRAM

The objective of the mechanical test program conducted by Westinghouse has been to determine to what extent service exposure has changed the mechanical properties of Inconel 600 material and to interpret any such changes in terms of the integrity of service tubes in the presence of possible cracking.

A series of mechanical tests were conducted with new and service exposed Inconel 7/8" diameter steam generator tubing to compare properties. The service exposed tubing had intergranular attack similar to that removed from Robinson.

Through-Wall Slot Tests

In materials of high toughness the mechanism of failure of tubes containing through-wall defects is a plastic bulge of the area adjacent to the crack.

This is essentially a limit load or plastic instability failure. In this event, the bulge pressure is only a function of crack length, tubing diameter, thickness and material flow properties.

Pressure tests were conducted on tubes with through-wall slots of varying lengths by using an internal pressure seal. The seal consisted of a 1/16" thick neoprene bladder supported in the slot region by a very thin steel foil. This permitted pressure buildup in the tube were the pressure provided an effective seal on the bladder. The bladder did not provide any significant strengthening to the tube. Pressure was increased in the tube to the point where the tubes bulged sufficiently to cause extrusion of the pressure seal. Generally, the crack opening at this point was on the order of .10 inches. Figure 4 is a picture of the deformation of new and service exposed inconel tubing upon reaching the bulge pressure.

The results of these tests on the 7/8" diameter tubing indicated that the bulge pressure as a function of through-wall defect length for both new and service exposed tubes exhibited the same behavior, as was expected. Figure 5 is a plot of bulge pressure versus through-wall crack length. It is noted from this plot that the bulge pressure increased with decreasing crack length. A crack 0.6 inches long could be supported without bulging of the tubing at a pressure above 2300 psi with minimum allowable material properties at 600°F. With nominal properties of the material a pressure of 3000 psi can be supported without bulging.

Partial Through-Wall Defect Tests

In order to further characterize the strength of Inconel 600 tubes in the presence of flaws, burst tests were conducted on tubes having electric discharge machined (EDM) defects part way through the wall. In general, the slots were performed with 7/8" O.D. tubing. Figure 6 is a plot of burst pressure versus reciprocal crack lengths. It will be noted from this data that tubing with 80 and 90% through-wall penetration has substantial additional strength as compared to the through-wall data. For comparison the through-wall defect data is plotted on this figure also.

In addition, pressure tests were conducted with service exposed Inconel tube sections which had eddy current indications. One tube was pressurized

to 4300 psi at which point it developed a small leak by opening up of two small adjacent cracks measuring about 1/16" and 1/8" in length on the O.D. surface. A second tube was pressurized to 5000 psi and held for ten minutes without rupturing or leaking. This specimen was subsequently slotted through wall and retested; it failed at a pressure slightly higher than that of new tubes containing a similar slot. The burst pressure of an unflawed tube is approximately 10-12,000 psi.

The results of the mechanical testing programs on service exposed material and new inconel showed that service conditions had not significantly degraded the mechanical properties. Further, the tubing had substantial strength even with through-wall or part through-wall cracks or slots.

D. LEAK RATE TESTS

A series of leak rate tests were performed at the Westinghouse Forest Hills test facility to determine the leakage rates through various crack configurations over a wide range of pressures. The experiments were conducted by growing tight fatigue cracks through 7/8" diameter inconel tubing of various lengths up to .8" long.

The sections of tubing were attached to a large autoclave which operated at 5550°F. The pressure in the autoclave was varied over a range of 1100 to approximately 2000 psi. Steam was discharged through the crack in the tubing to the atmosphere. The flow rates through the crack were measured by a calibrated flow meter in the line going to the piece of tubing.

The results of these tests for a .6" and .7" crack are shown on Figures 7 and 8 respectively. It will be noted that in both cases, at 1400 psi which is the operating pressure differential across the steam generator tubes at power, the leak rates are in excess of 1 gpm. In addition for all the tubes tested (.5" through .8" long cracks) the tubes retained their original round shape after the maximum pressure (2000 psi) was relieved from the system. The results confirm the results of the through-wall pressure testing previously discussed in the Mechanical Testing Program Section and further indicate the ability of the tube to withstand this pressure differential.

E. SAFETY EVALUATION

It is our conclusion that caustic corrosion will not develop into a guillotine or equivalent double ended break. This conclusion is based on both in-plant experience and on material and leak rate testing.

The experience with leaking tubes in a plant has shown that the process is an orderly one occurring over a time interval ranging from many hours to days. Thus, when leakage has occurred it was detected readily by the operators and remained well within values which precluded the use of any emergency or abnormal operating procedures. Maximum leakages which have occurred during power operation were on the order of 1 gpm.

Examination of tubes with intergranular attack which were removed from operating units has shown that the corrosion process is such that corrosion proceeds through the tube wall without achieving substantially long lengths. The affected area was within an axial band of approximately 1 inch and no cracks were this long. The longest crack area observed in the Robinson tubes was approximately 9/16" long.

The failure mode of a through-wall defect is shown in Figure 4 for both new and service-exposed tubes. The failure is a plastic deformation or bulge of the area adjacent to the crack. This deformation mechanism is in direct support of the in-plant experience of the orderliness of the leakage. Note from the figure that there is essentially no difference in the appearance of the deformation between new and service-exposed tubing. Bulge pressure is plotted as a function of through-wall slot length in Figure 5 for operating temperatures. For a given crack length, as long as the pressure differential across the tube is less than the bulge pressure, the crack opening or displacement will be elastic and the opening will return to its original dimensions as the pressure is reduced. If the bulge pressure is exceeded the material will yield and the crack opening will take a permanent set.

At a pressure differential of 1400 psi, which corresponds to the normal operating pressure differential, with a .6" through wall tight fatigue crack, a leak rate of approximately 1.4 gpm has been measured. For the

tubing in the CP&L plant a .6" through wall crack would have a bulge pressure of 2300 psi based on minimum material properties (Figure 5) and 3000 psi based on nominal properties. This leak rate is well in excess of that at which it is currently required[1] (.3 gpm) that the plant be shutdown and the leak repaired.

Based on the evidence which establishes that tubing retains substantial strength with through-wall or partial through-wall defects, on the experimentally derived leakage which can be associated with defects at operating conditions and on operational experience regarding the capability to detect and respond to the leakage as an indicator of tube deterioration, it is concluded that the existing steam generator leakage limitation[1] provides assurance that the structural integrity of the steam generator tubes is unchanged from that assumed in the H. B. Robinson Unit 2 Final Safety Analysis Report and that there is no unreviewed safety problem.

[1]AEC letter dated September 12, 1972 - "Interim Conditions for Operation".

TABLE 1

CPL STEAM GENERATOR "A"

(Tested May, 1972)

	Tubes with Indications	Wall Penetration %		Elevation (Above Tube Sheet)
1.	5-41	< 50	OD	0-3"
2.	6-20	90	OD	0-2"
3.	7-41	60	OD	0-3"
4.	7-21	90	OD	0-3"
5.	7-27	90	OD	0-3"
6.	7-43	100 Leaker	OD	0-3"
7.	7-45	85	OD	0-3"
8.	7-46	50	OD	4"
9.	7-47	90	OD	0-3"
10.	8-25	80	OD	0-3"
11.	8-29	85	OD	0-3"
12.	8-30	85	OD	0-3"
13.	8-42	75	OD	3"
14.	8-45	85	OD	0-3"
15.	8-48	75	OD	2"
16.	8-50	75	OD	0-3"
17.	8-51	90	OD	0-3"
18.	8-52	60	OD	0-3"
19.	9-47	80	OD	0-3"
20.	15-41	50	OD	
21.	16-40	75	OD	0-3"
22.	18-46	90	OD	0-3"
23.	19-48	< 50	OD	0-3"

All the above tubes were plugged.
In addition, the following tubes were also plugged.

24	7-44	Indication free tube removed for examination.
25	7-28	Required plugging due to error in plugging outlet end of tubes 7-27, 8-25 and 8-30.
26	8-26	
27	8-31	

CPL STEAM GENERATOR "B"

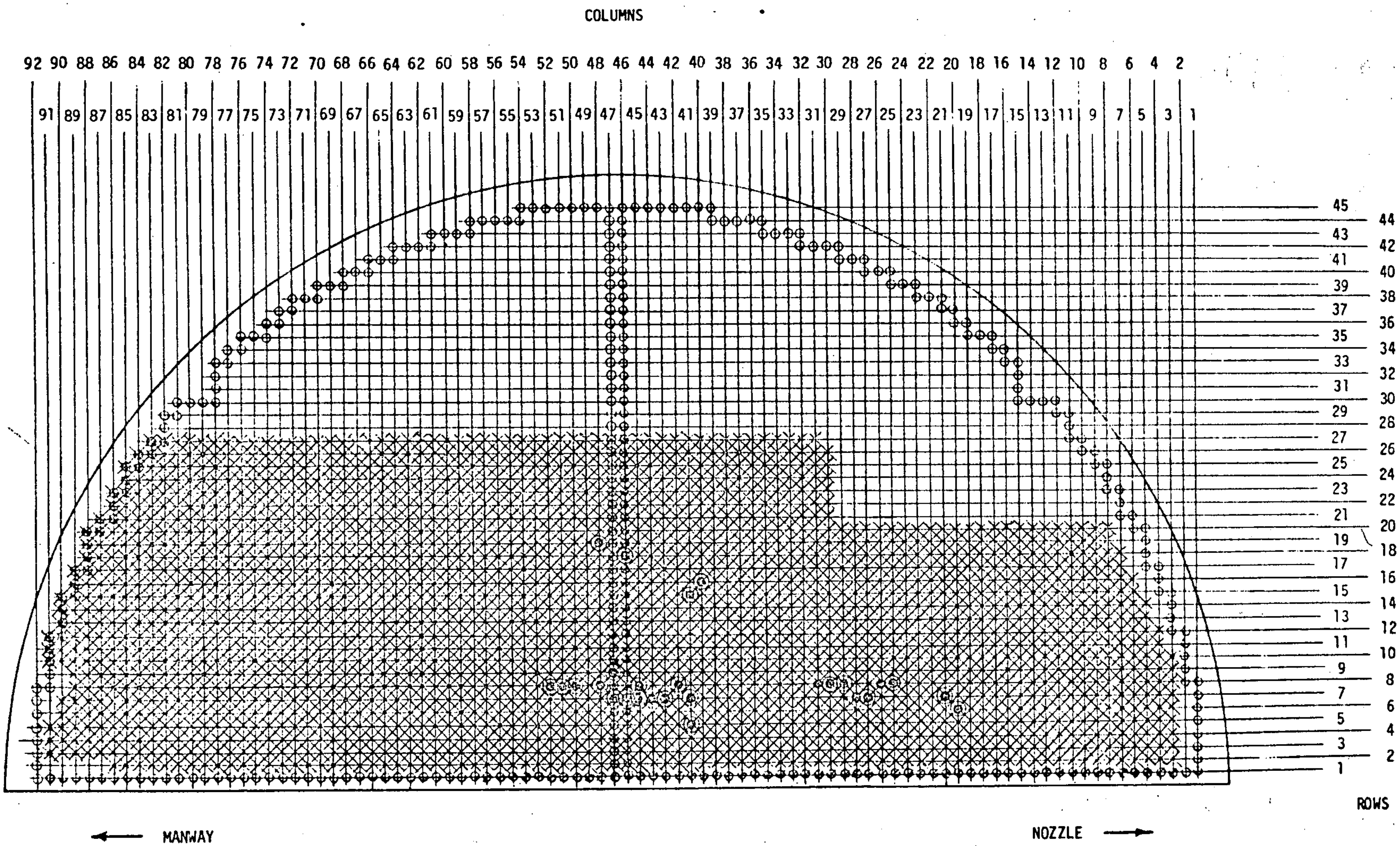
	Tubes with Indications	Wall Penetration-%		Elevation (Above Tube Sheet)
1.	18-41	75	OD	1/2"
2.	20-44	60	OD	At Top of Tube Sheet
3.	20-45	75	OD	At Top of Tube Sheet
4.	20-46	85	OD	At Top of Tube Sheet

All four tubes plugged.

CPL STEAM GENERATOR "C"

1.	19-45	75	OD	0-2"
2.	15-48	75	OD	0-2"

Both tubes plugged.



CPL STEAM GENERATOR "A"

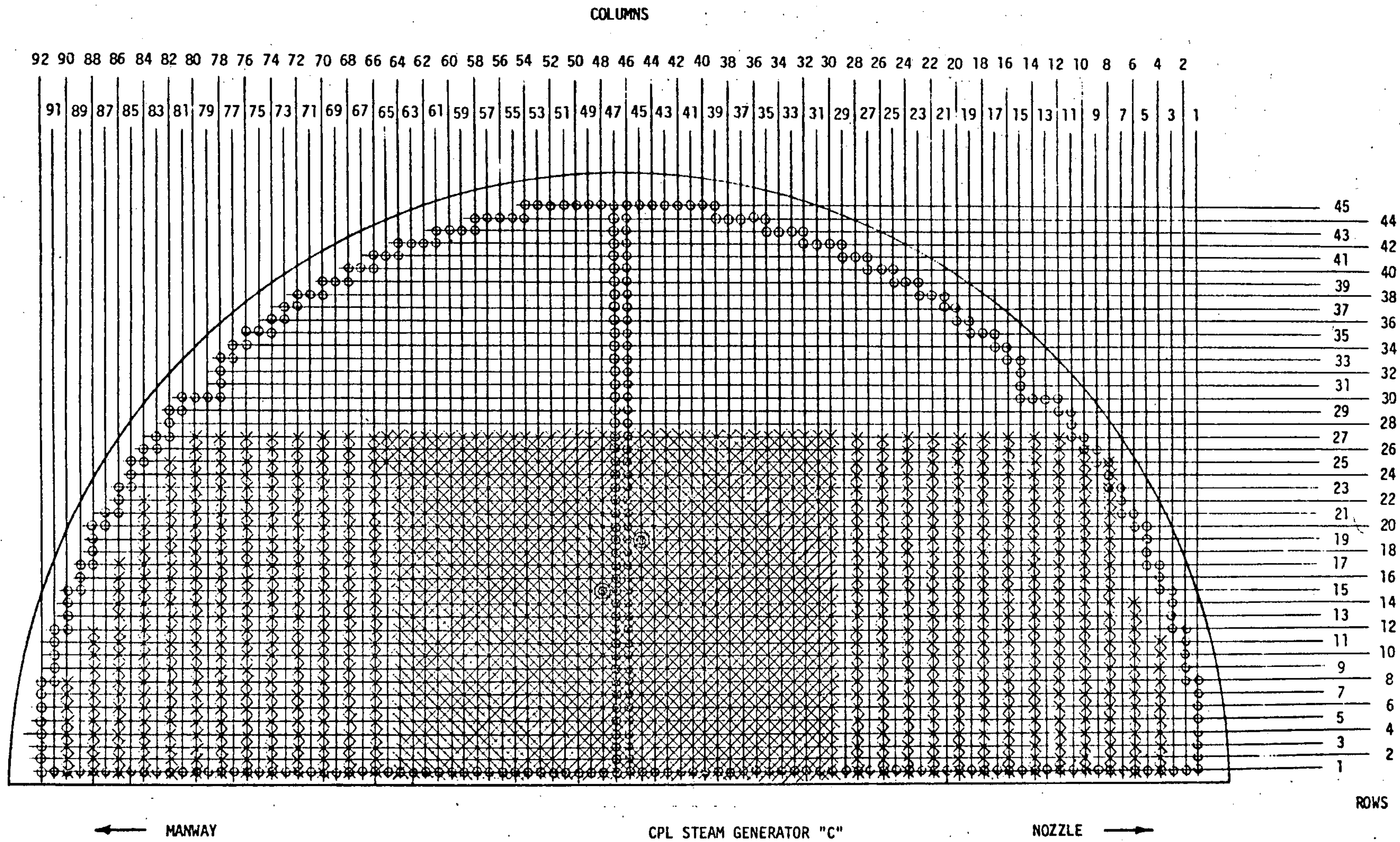
7-6 9-72 Rev.0 September 1, 1972

X - Indicates tube eddy current probed May, 1972

● - Indicates tube plugged May, 1972

O - Eddy current indications of wall penetration

FIGURE 1



CPL STEAM GENERATOR "C"

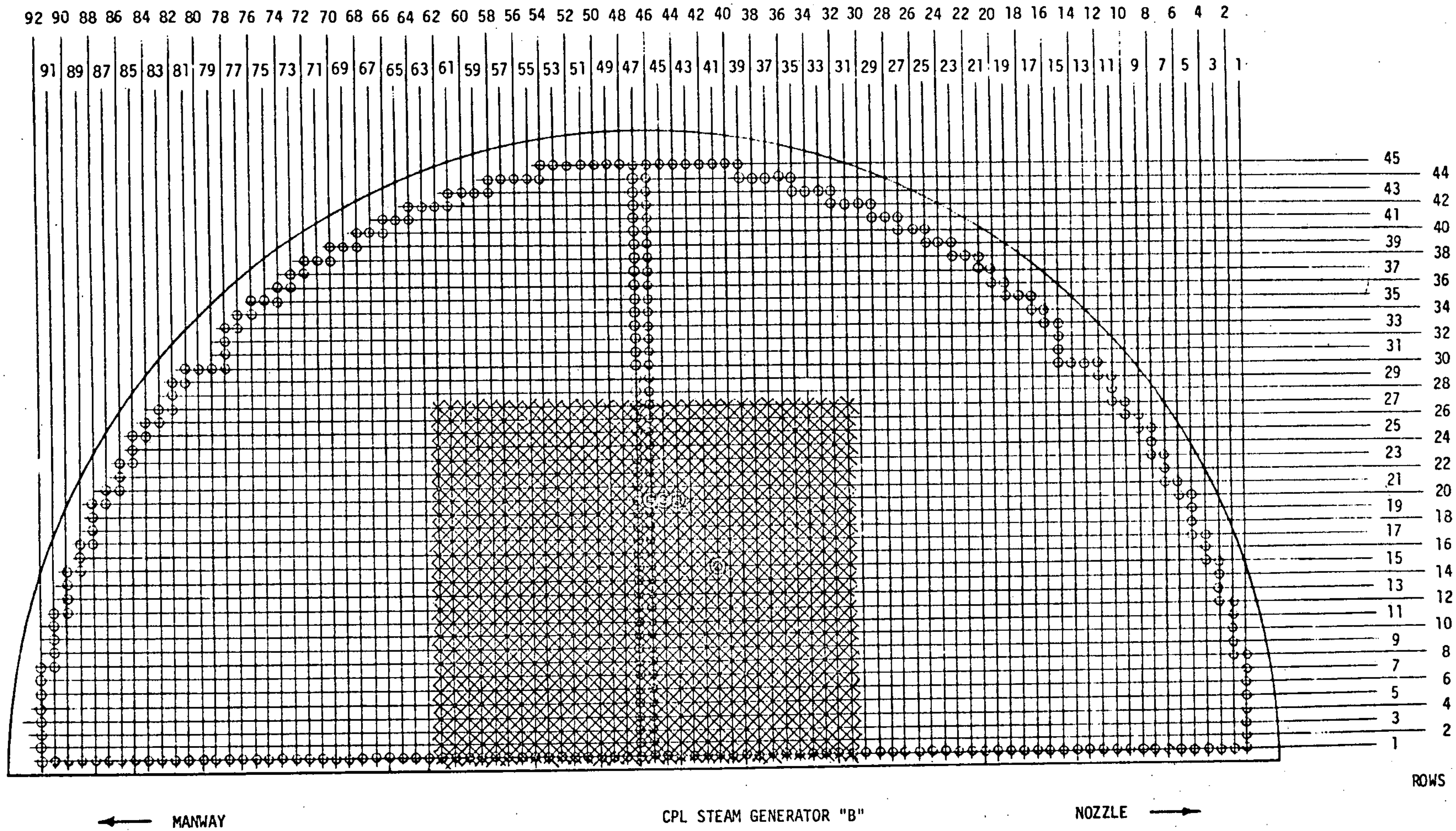
September 1, 1972

X - Indicates tube eddy current probed May, 1972

● - Indicates tube plugged May, 1972

○ - Eddy current indications of wall penetration

FIGURE 2



September 1, 1972

X - Indicates tube eddy current probed May, 1972

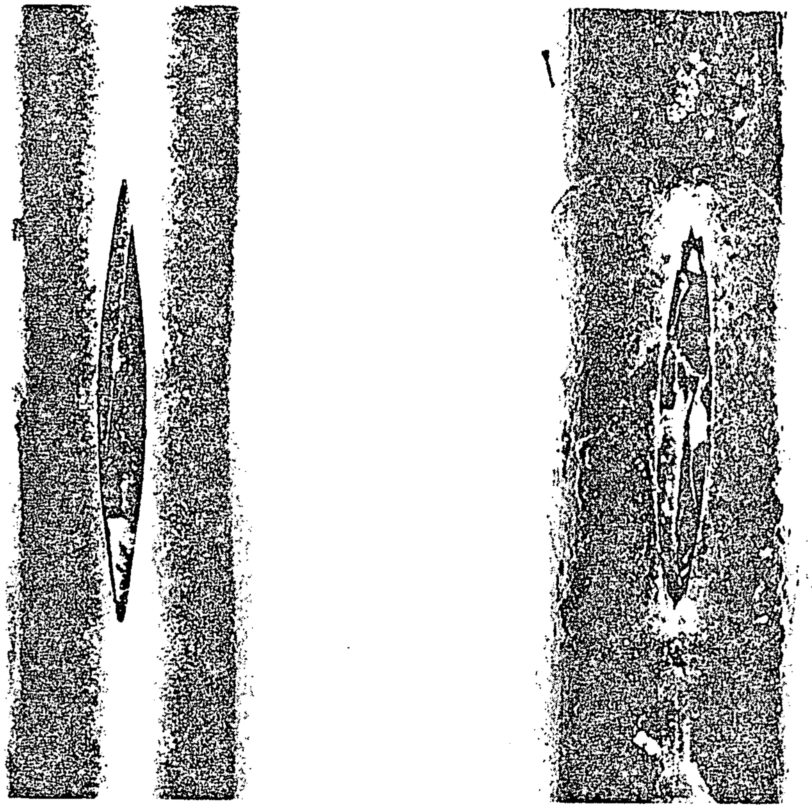
● - Indicates tube plugged May, 1972

○ - Eddy current indications of wall penetration

FIGURE 3

New Tube

Service Exposed Tube



Bulge Pressure 1230 psi

Bulge Pressure 1230 psi

Figure 4. Deformation of new and service exposed tubing upon reaching the bulge pressure. Tubes contained machined through wall slots of 1.5 inches.

Figure 5 BULGE PRESSURE VERSUS THROUGHWALL DEFECT LENGTH

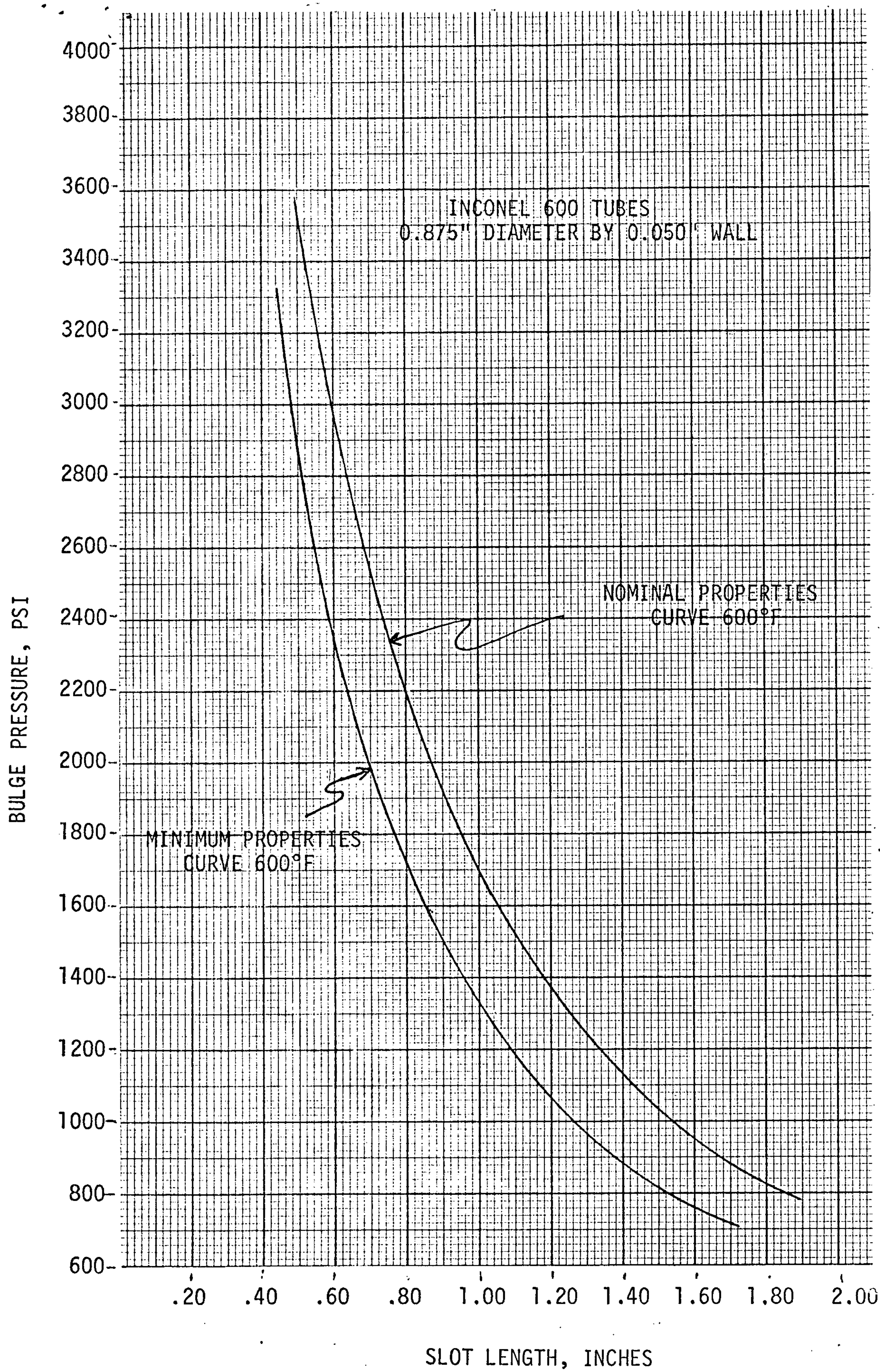
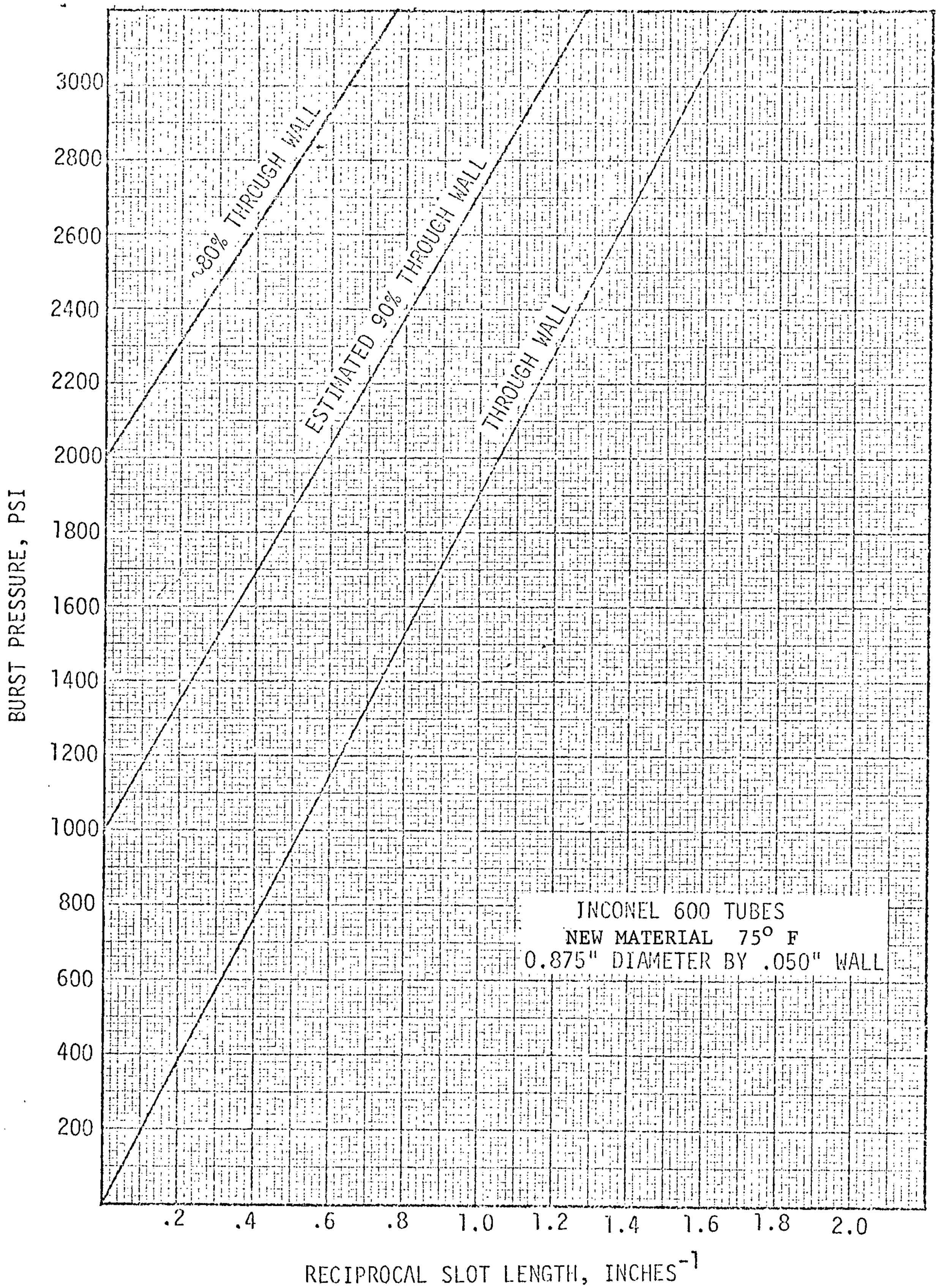


Figure 6 BURST PRESSURE VERSUS RECIPROCAL SLOT LENGTH FOR VARIOUS SLOT DEPTHS



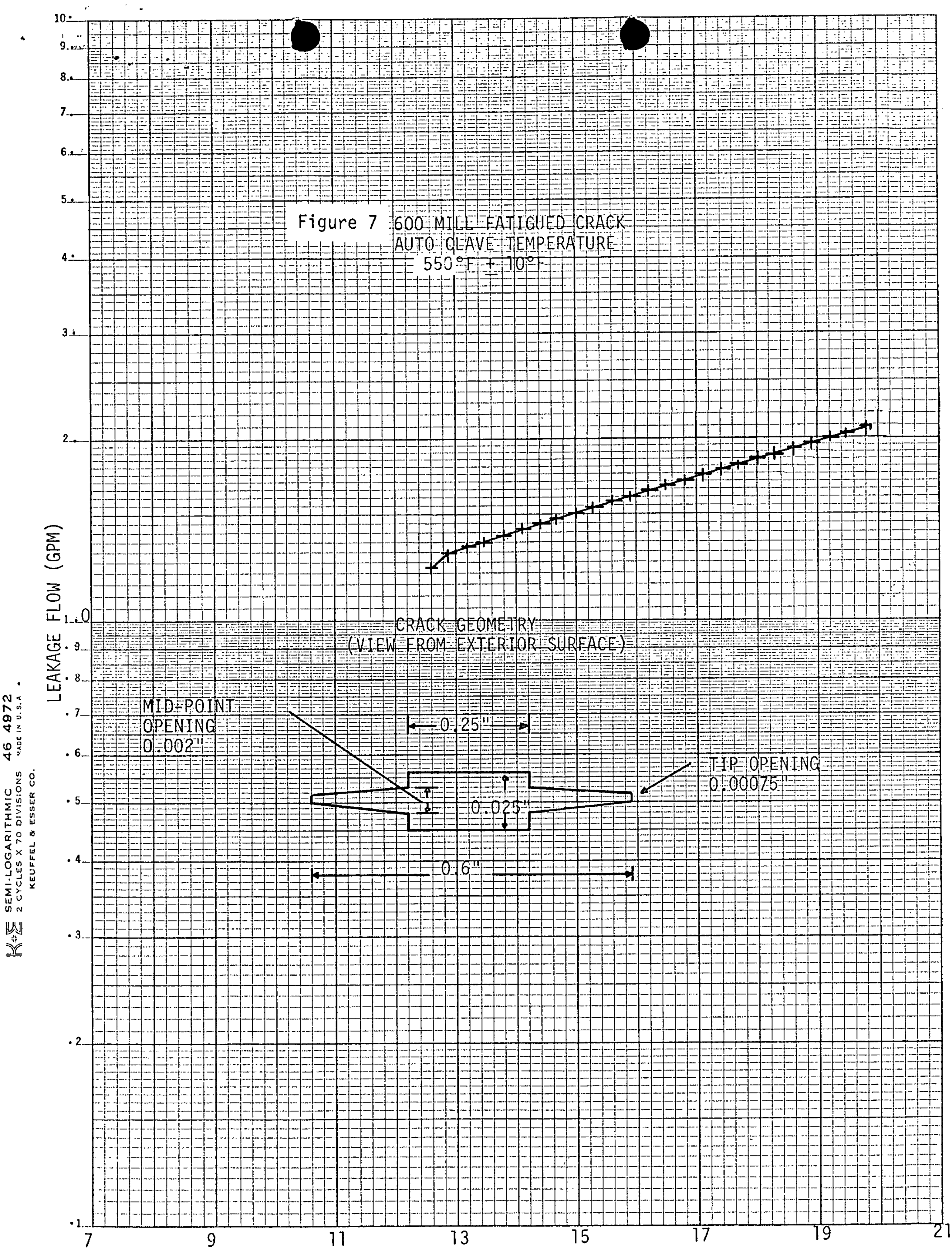


Figure 7 600 MILL FATIGUED CRACK
AUTO CLAVE TEMPERATURE
550°F ± 10°F
CRACK GEOMETRY
(VIEW FROM EXTERIOR SURFACE)
MID-POINT
OPENING
0.002"
0.25"
TIP OPENING
0.00075"
0.025"
0.6"
LEAKAGE FLOW (GPM)
ΔP x 10-2 (PSI)
10
9
8
7
6
5
4
3
2
1.0
.9
.8
.7
.6
.5
.4
.3
.2
.1
7
9
11
13
15
17
19
21
SEMI-LOGARITHMIC 46 4972
2 CYCLES X 70 DIVISIONS MADE IN U.S.A.
KEUFFEL & ESSER CO.

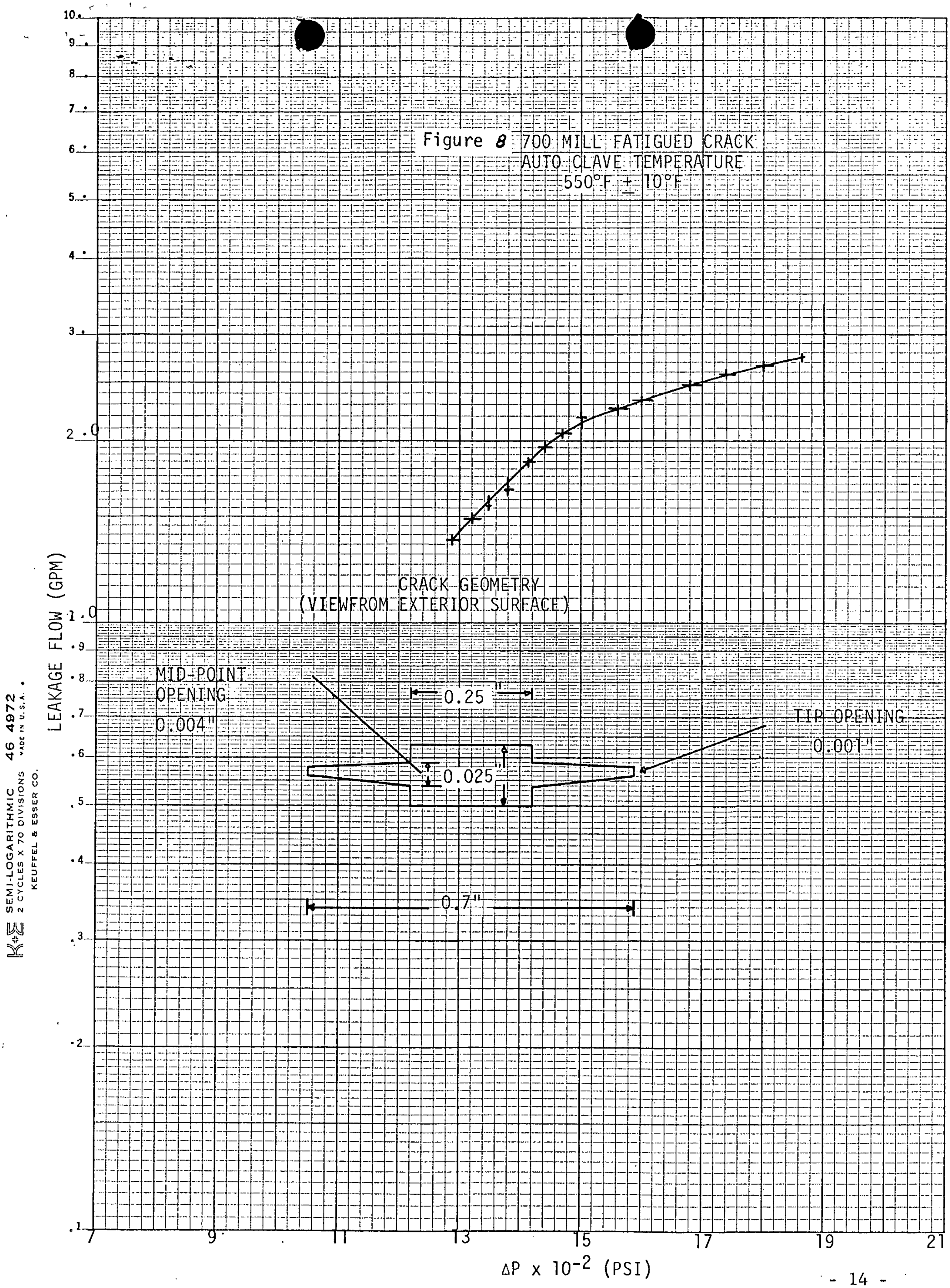


Figure 8 700 MILL FATIGUED CRACK
AUTO CLAVE TEMPERATURE
550°F ± 10°F
CRACK GEOMETRY
(VIEWFROM EXTERIOR SURFACE)
MID-POINT
OPENING
0.004"
0.25"
0.025"
TIP OPENING
0.001"
0.7"
LEAKAGE FLOW (GPM)
ΔP x 10-2 (PSI)
7
9
11
13
15
17
19
21
10
9
8
7
6
5
4
3
2.0
1.0
.9
.8
.7
.6
.5
.4
.3
.2
.1
SEMI-LOGARITHMIC 46 4972
2 CYCLES X 70 DIVISIONS MADE IN U.S.A.
KEUFFEL & ESSER CO.

SHAW, PITTMAN, POTTS, TROWBRIDGE & MADDEN
BARR BUILDING
910 17TH STREET, N.W.
WASHINGTON, D.C. 20006

WASHINGTON, D.C.
PM
5 MAY
1966

NATIONAL PARK SERVICE
50TH ANNIVERSARY

5c U.S. POSTAGE

Mr. Harold L. Price
Director of Regulation
Atomic Energy Commission
4915 St. Elmo Avenue
Bethesda, Maryland