MAINE YANKEE ATOMIC POWER COMPANY

EDISON DRIVE
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February 17, 1983
MN-83-29 JHG-83-33

Director fo Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
Division of Licensing

Reference (a) License No. DPR-36 (Docket No. 50-309)

Subject: Main Feedwater System

Dear Sir:

This letter transmits information for the docket relating to resolution of the recent feedwater system event at the Maine Yankee Atomic Power Station.

This material is essentially what was presented at the meeting held on February 9, 1983 in the Commission's Bethesda, Maryland offices to review the event and Maine Yankee's program of corrective actions. Minor revisions have been made for editorial purposes. Commentaries have been added to reflect some of the discussion which took place during the February 9, 1983 meeting. The description of the water hammer test has been revised to reflect discussions with the staff which followed the February 9 meeting.

The staff is to be commended for their prompt attention and expedited handling of this matter.

We trust this information is satisfactory. If there are any questions, please do not hesitate to call.

Very truly yours,

MAINE YANKEE ATOMIC POWER COMPANY

John H Garrity

John H. Garrity, Senior Director
Nuclear Engineering and Licensing

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Limited Dist

JHG:pjp

Enclosure (98 pages)

cc: Mr. Ronald C. Haynes
Mr. Paul A. Swetland

MAINE YANKEE ATOMIC POWER STATION FEEDWATER EVENT

TABLES OF CONTENTS

1) INTRODUCTION

On January 25, 1983, Maine Yankee experienced a feedwater line steam-water hammer event that caused the feedwater line adjacent to a steam generator to crack through and leak. This report describes the plant systems and components related to the event, its causes and consequences, the repairs and modifications that were made or planned, and provides a justification for continued power operation. Most of the material contained herein was presented to the NRC at a meeting held at their offices on February 9, 1983. It has been supplemented by commentary reflecting some of the technical comments made during the oral presentation.

2) GENERAL DESCRIPTION OF PLANT

- DESIGN CHARACTERISTICS
- CONTAINMENT PLAN
- CONTAINMENT ELEVATION
- STEAM GENERATOR VERTICAL SECTION AND KEY ELEVATIONS

TABLE 1.3.1

MAINE YANKEE DESIGN CHARACTERISTICS

Plant	
Net Electrical Power Output (MWe) @ 2,630 MWt	825
Gross Electrical Power Output (MWe) @ 2,630 MWt	864
Maximum Expected Gross Electrical Output (MWe)	864
Nuclear Steam Supply System	
Core Thermal Output (MWt)	2630
Operating Pressure (psig)	2235
Design Pressure (psig)	2485
Reactor Coolant Inlet Temperature (F)	532-550
Reactor Coolant Outlet Temperature (F)	532-600
Pipe Size: Outlet (ID, inches)	33-1/2
(Wall Thickness, inches)	3-1/4
Inlet (ID, inches)	33-1/2
(Wall Thickness, inches)	3-1/4
Flow per Loop (lb/hr)	44.87×10^6
Number of Loops	3
Number of Pumps	3
Type	Vertical, Centrifugal Mechanical Seals
Design Flow/Pump (gpm)	120,000
Core	
Total Heat Output (Btu/hr)	8.981×10^9
Heat Generated in Fuel (%)	97.5
DNB Ratio at Nominal Conditions	1.903
Minimum DNBR for Design Transients (W-3 Correlation)	1.30
Core Power Density (kW/liter)	80.86
Number of Fuel Assemblies	217
Number of Fuel/Poison Rods per Assembly	176
Fuel Rod Pitch (inches)	0.580
Fuel Clad Material	Zircaloy-4
Fuel Clad Minimum Thickness (inches)	0.024
Number of Control Rods	77
CEA Pitch (inches)	11.57
Poison Materials	B_4C/stainless steel
Control Road Drive Type	Magnetic Jack
Equivalent Core Diameter (inches)	136
Total Uranium (Mtu)	80-83

TABLE 1.3.1
(continued)

MAINE YANKEE DESIGN CHARACTERISTICS

Feedwater Heater Stages	6
Condensate Pumps - Number	3 Half-Capacity
Design Flow (gpm)	9060
Design Head (ft)	960
Feedwater Pumps - Electrical - Number	2 Half-Capacity
Design Flow (gpm)	14,000
Design Head (ft)	2038
Feedwater Pump - Steam Driven - Number	1 Full-Capacity
Design Flow (gpm)	28,000
Design Head (ft)	2200
Circulating Water Pumps - Number	4 Quarter-Capacity
Design Flow (gpm)	106,500
Design Head (ft)	26

Generator

Design Rating (Mva)	900
Power Factor	.90
Terminal Voltage (kV)	22

NSSS Auxiliary Systems

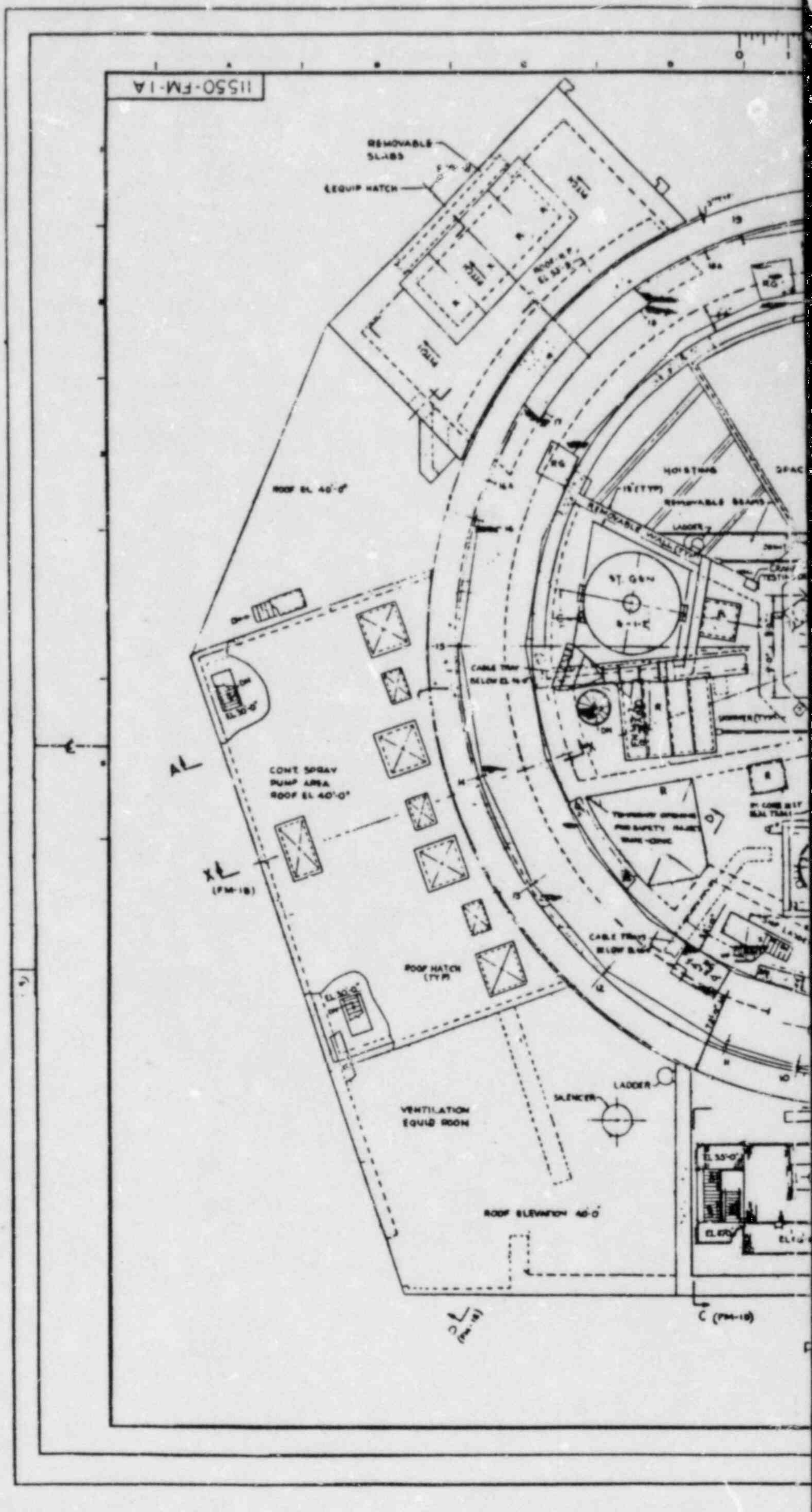
(a) Chemical and Volume Control System

Normal Letdown Flow Rate (gpm)	80
Maximum Letdown Flow Rate (gpm)	200
Charging Pumps - Number	3 Fixed-Capacity
Design Flow (gpm)	150
Design Pressure (psig)	2850
Auxiliary Charging Pump - Number	1 Variable Speed
Design Flow (gpm)	10 to 30
Design Pressure (psig)	3700
Regenerative Heat Exchanger - Number	1 Full-Capacity
Design Heat Transfer (Btu/hr)	10.0×10^6
Letdown Heat Exchanger - Number	1 Full-Capacity
Design Heat Transfer (Btu/hr)	7.82×10^6
Demineralizers - Number	3 Purification 1 Deborating
Nominal Rating (gpm)	80
Maximum Flow (gpm)	200
Resin Volume (ft^3)	32
Filter - Number	2
Type	Cartridge
Design Rating (gpm)	200
Filter Size (microns)	2

TABLE 1.3.1
(continued)

MAINE YANKEE DESIGN CHARACTERISTICS

(g) Spent Fuel Cooling System	
Spent Fuel Pool Capacity	953 assys.
Volume (ft^3)	59,116
Pumps - Number	2
Rating, Each (gpm)	772
Head (ft)	120
Heat Exchanger - Number	1
Rating (Btu/hr)	22.3×10^6
Filter - Number	1
Type	Cartridge
Rating (gpm)	200
Size (microns)	2
Demineralizer - Number	1
Resin Type	mixed bed
Bed Size (ft^3)	32
Nominal Flow (gpm)	200
Conventional Plant Auxiliary Systems	
(a) Service Water System	
Service Water Pumps - Number	4 Full-Capacity
Rating (gpm)	10,000
Head (ft)	66
(b) Compressed Air System	
Compressors - Number	3
Rating (scfm)	300
Discharge Pressure (psig)	100
Containment	
Type	Reinforced Concrete
Diameter (ft-inches)	135-0
Height (ft-inches)	169-6
Liner - Material Thickness (inches)	ASTM A516 Grade 60
Wall	3/8
Dome	1/2
Floor	1/4
Design Pressure (psig)	55
Design Temperature (F)	280
Leak Rate (percent per day)	0.1


11550-FM-1A
REMOVABLE SLABS
EQUIP HATCH
ROOF EL 40'-0"
HOISTING
REMOVABLE BEAMS
LADDER
ST. GEN
CABLE TRAY BELOW EL
CONT. SPRAY PUMP AREA ROOF EL 40'-0"
A
X
(FM-1B)
ROOF HATCH (TYP)
CABLE TRAY
LADDER
SILENCER
VENTILATION EQUIP ROOM
ROOF ELEVATION 40'-0"
C (FM-1B)

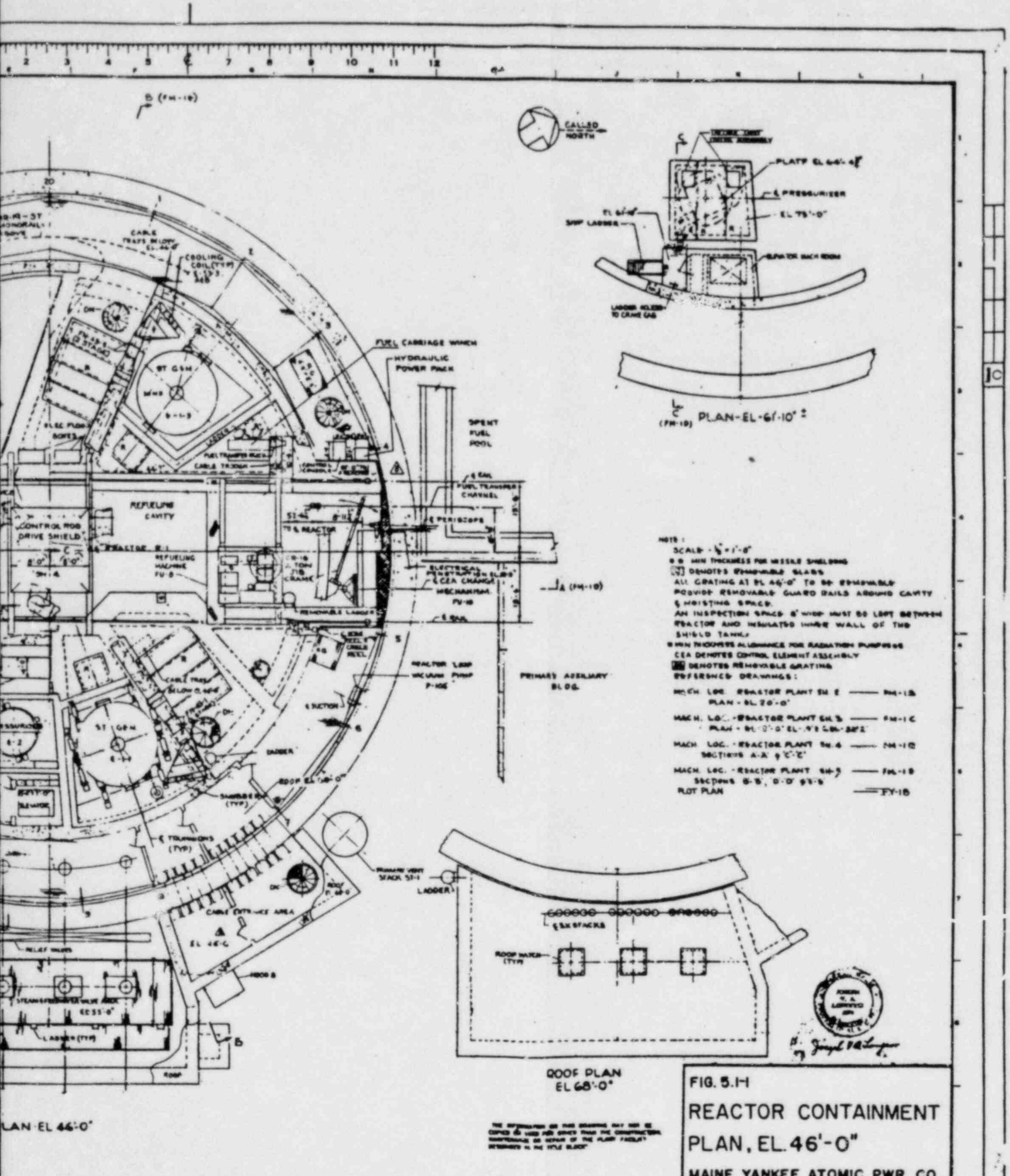
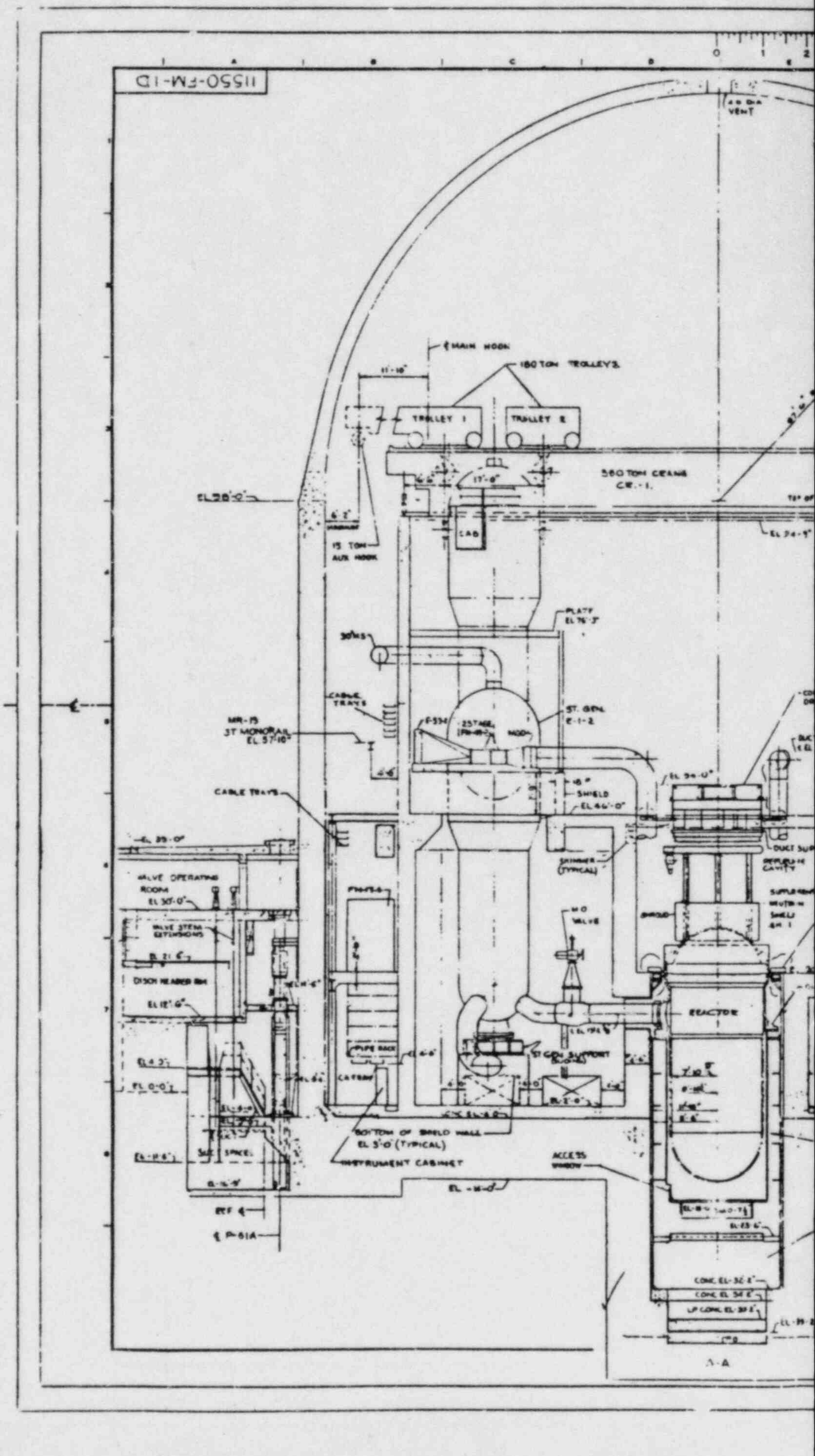


FIG. 5.1-1
REACTOR CONTAINMENT
PLAN, EL. 46'-0"
MAINE YANKEE ATOMIC PWR. CO.


MAIN HOOK
180 TON TROLLEYS
TROLLEY 1
TROLLEY 2
380 TON CRANE
CR.-1.
EL 98'-0"
15 TON
AUX HOOK
CAB
CABLE TRAYS
VALVE OPERATING ROOM
EL 30'-0"
VALVE STEM EXTENSIONS
DISCH HEADER RM
ST. GEN
E-1-2
SHIELD
EL 46'-0"
SKIMMER (TYPICAL)
PIPE RACK
REACTOR
ACCESS WINDOW
BOTTOM OF SHIELD WALL
EL 3'-0" (TYPICAL)
INSTRUMENT CABINET
VENT
A-A

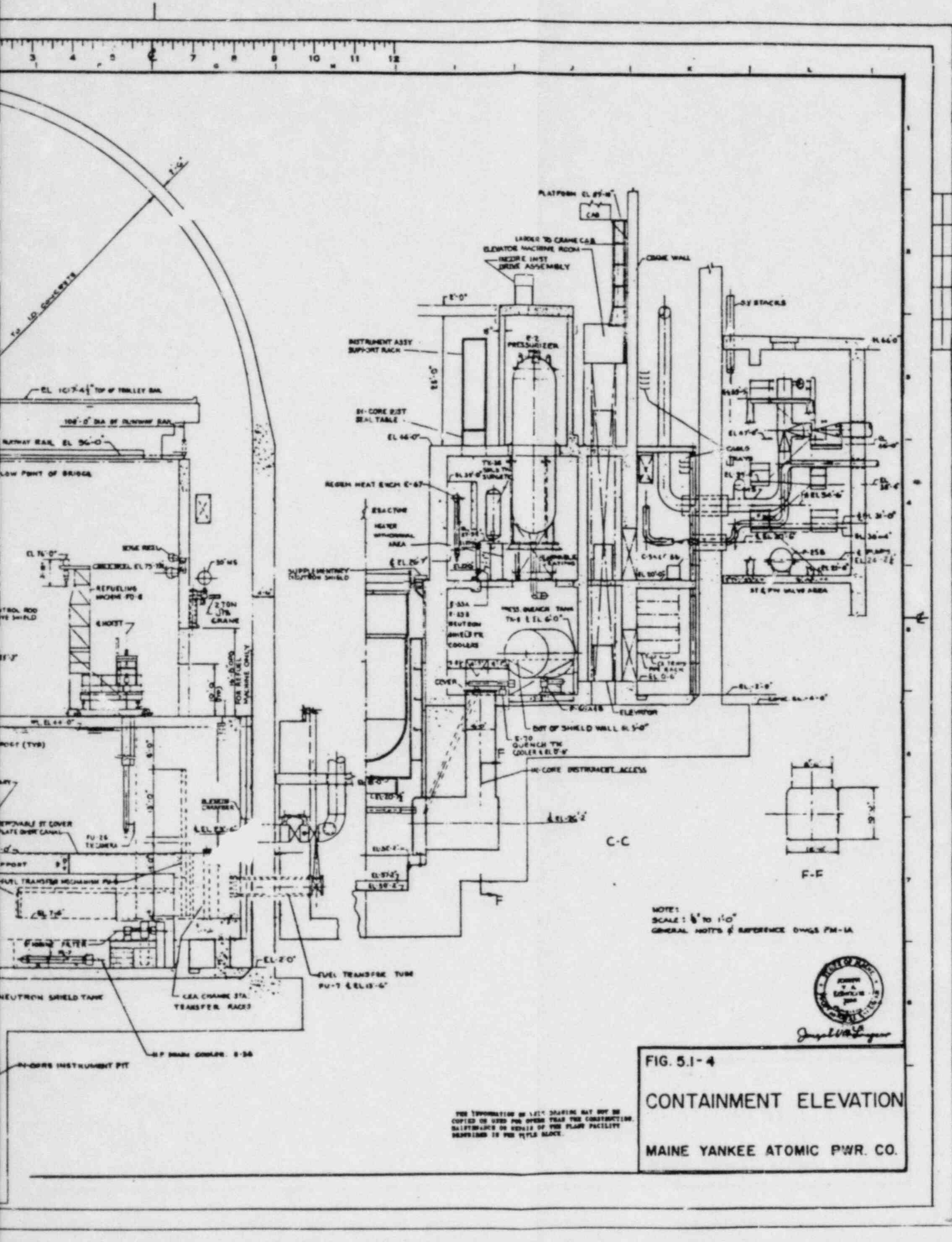


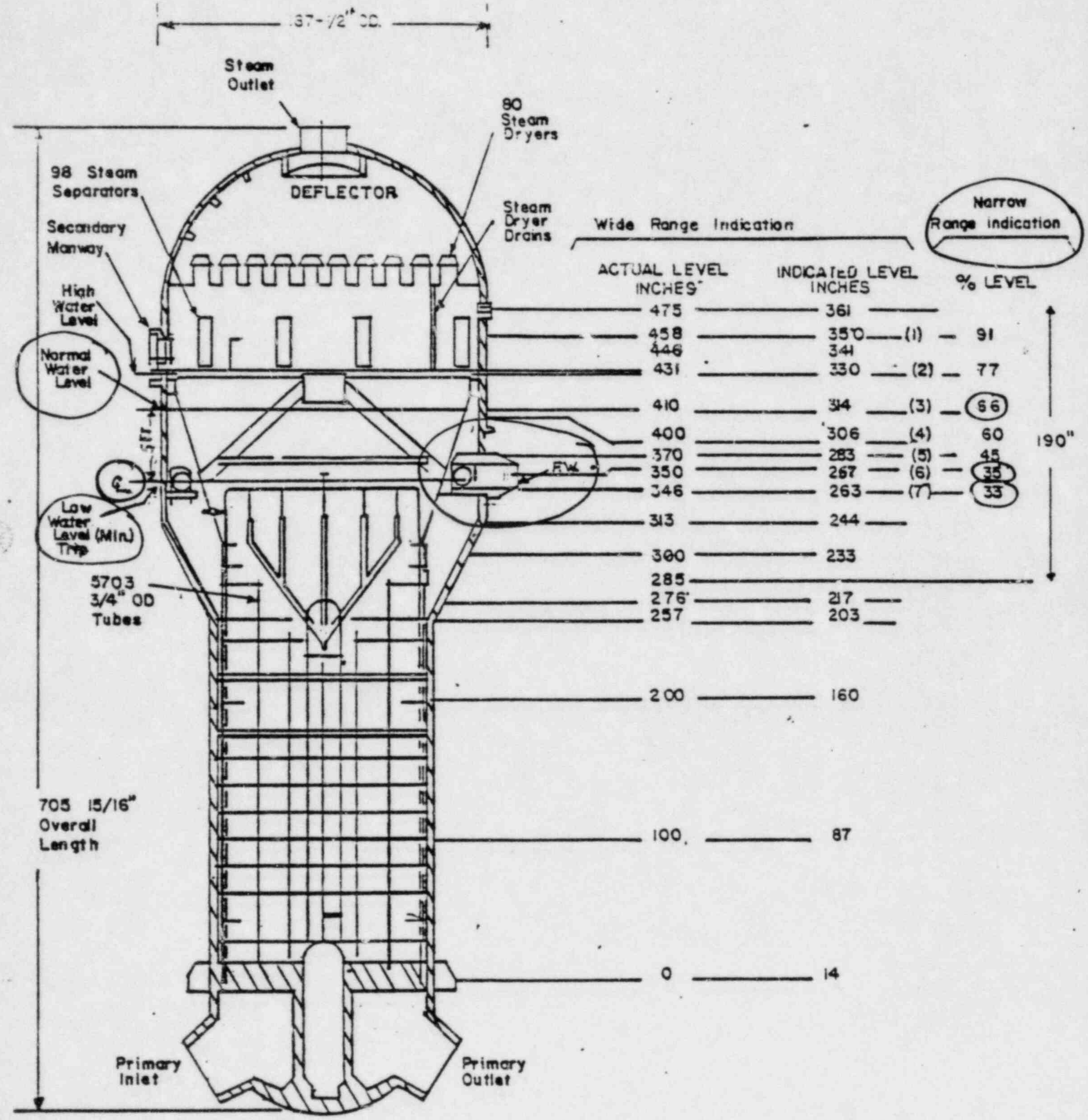
FIG. 5.1-4

CONTAINMENT ELEVATION

MAINE YANKEE ATOMIC PWR. CO.

REVIEWED BY [illegible] DATE [illegible] PREPARED BY [illegible]

ILLUSTRATION OF STEAM GENERATOR SHOWING ACTUAL VS. INDICATED WATER LEVELS AT T = 532°F (HZP) FOR WIDE RANGE

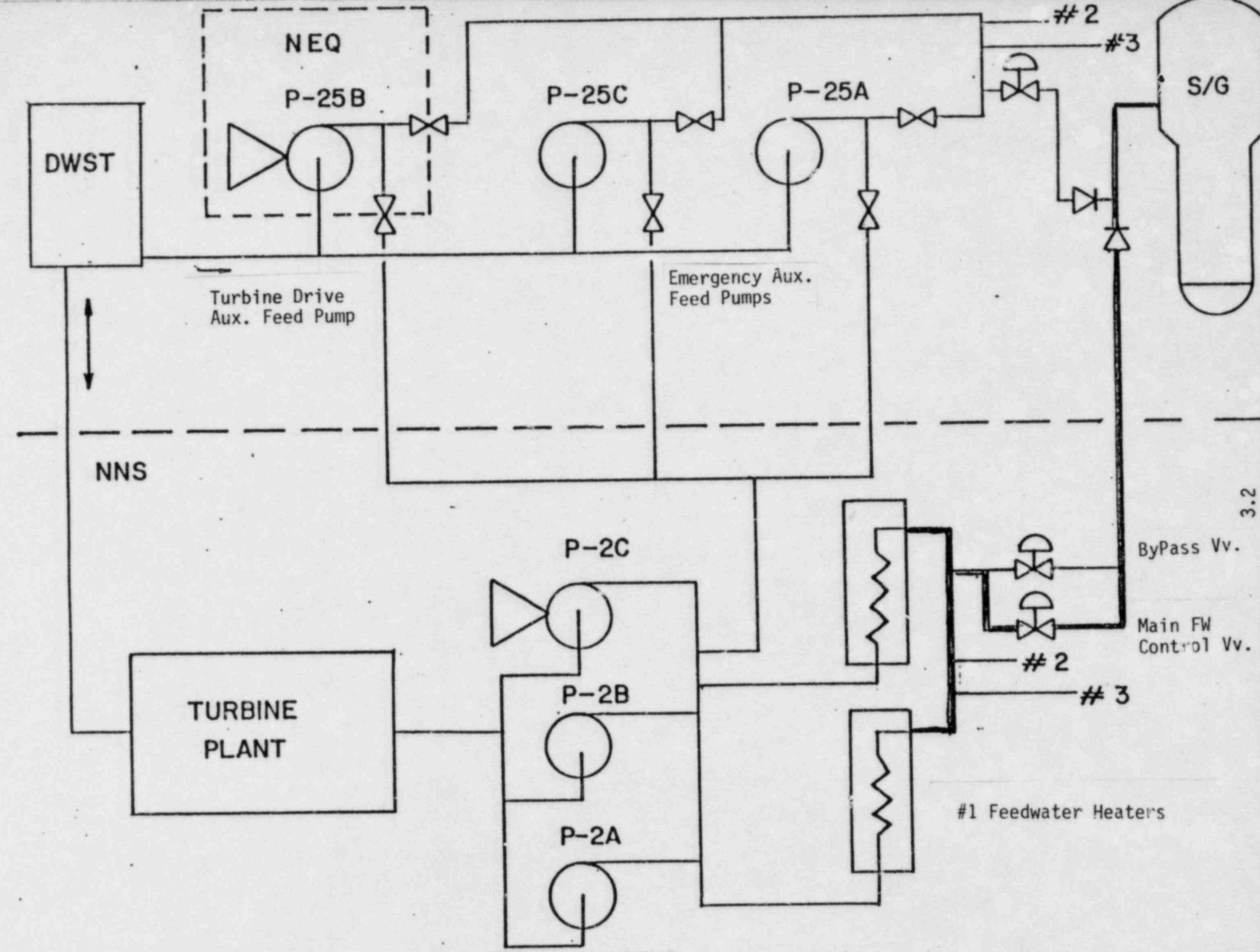


1. High Water Level Trip - 91%
2. High Water Level Alarm - 77%
3. Normal Water Level - 66%
4. Low Water Level Alarm - 60%
5. Low Water Level Pre-Trip - 45%
6. Low Water Level Trip - 35%
7. Top of Tube Bundle - 33%

3) DESCRIPTION OF FEEDWATER SYSTEM FLOW PATHS

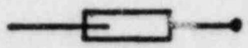
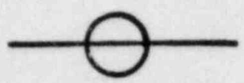
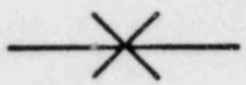
AND

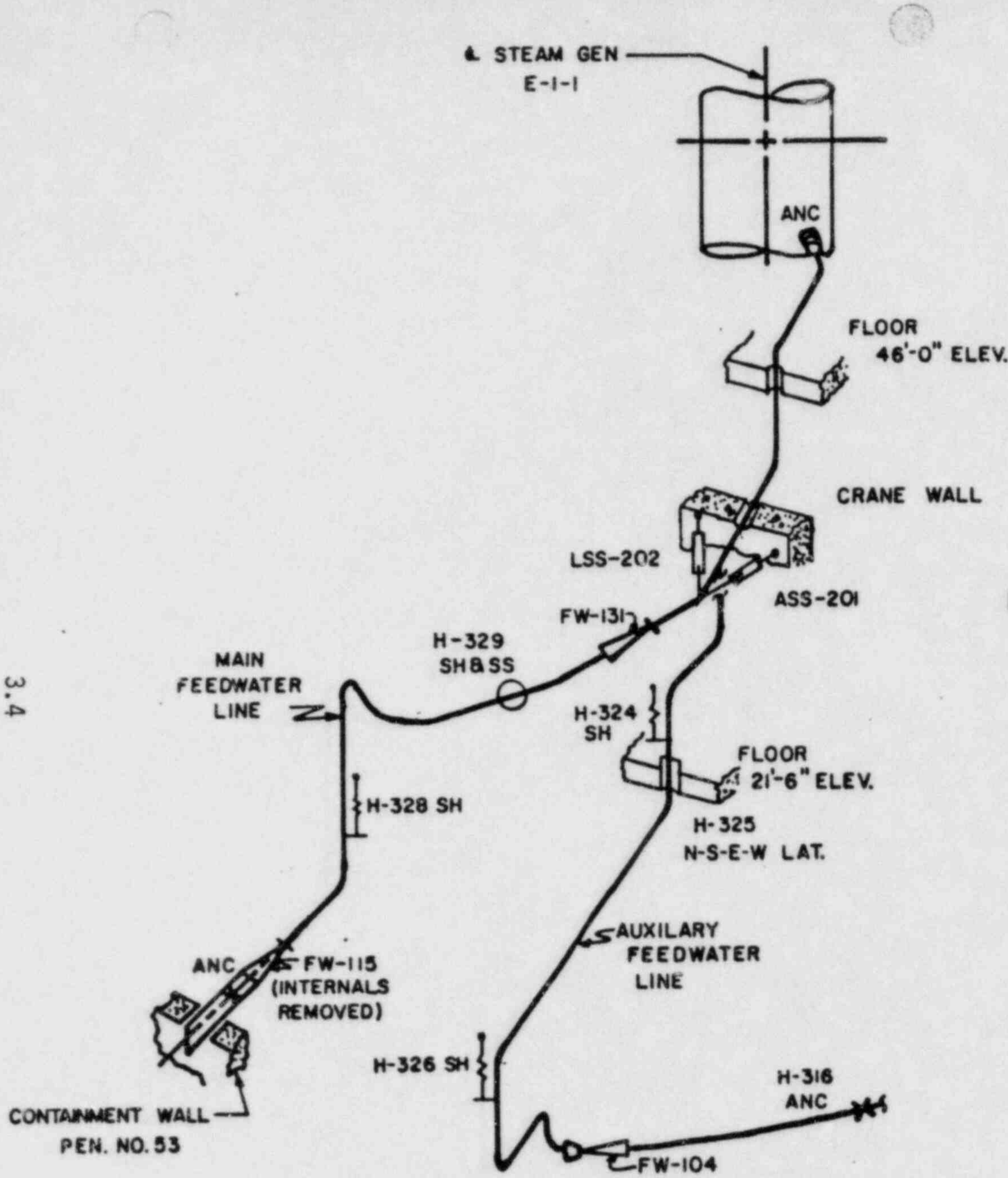
FEEDWATER PIPING, SAFE END, AND STEAM GENERATOR NOZZLE CONFIGURATION



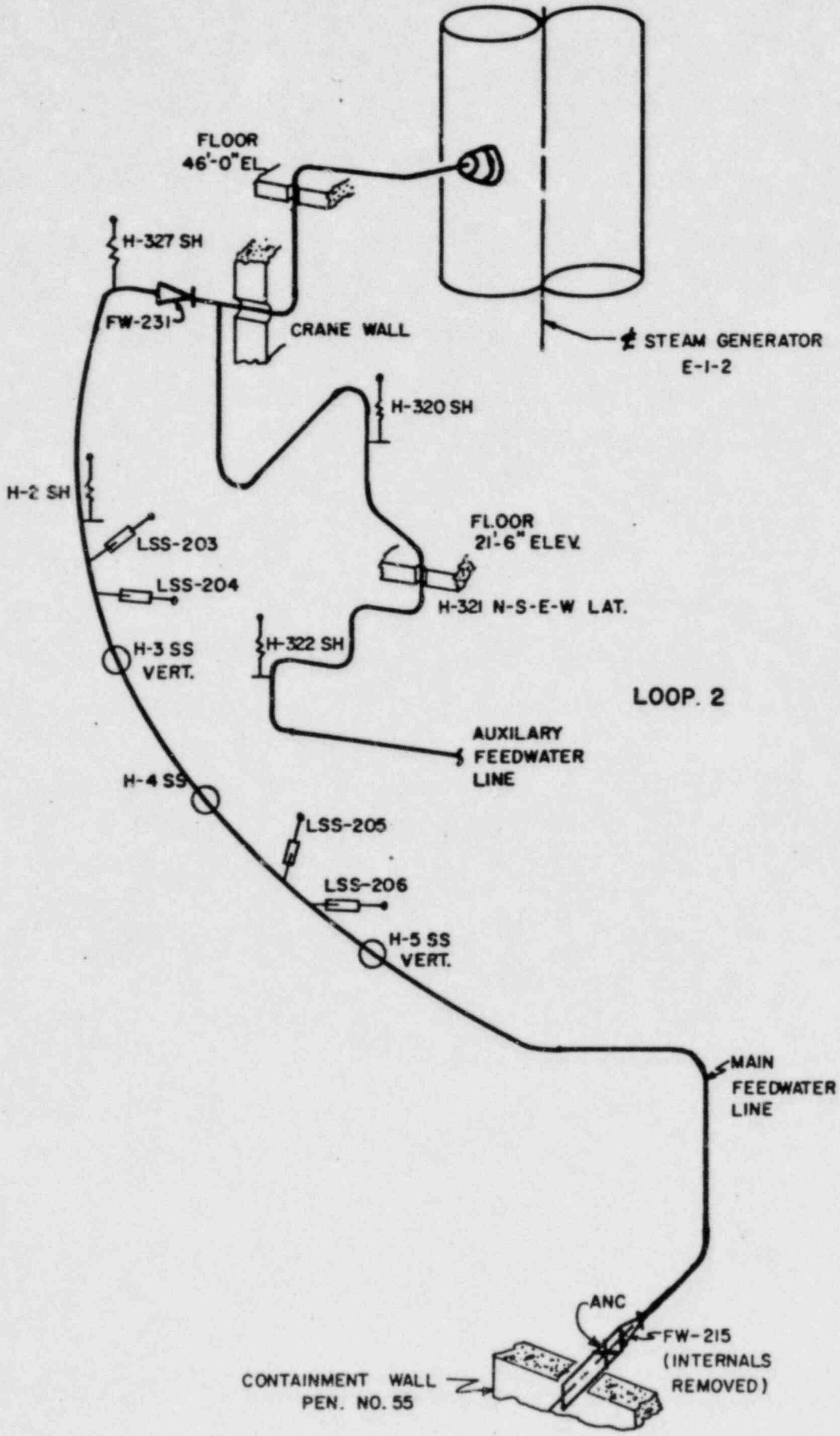
NEQ
P-25B
P-25C
P-25A
DWST
S/G
#2
#3
Turbine Drive
Aux. Feed Pump
Emergency Aux.
Feed Pumps
NNS
P-2C
P-2B
P-2A
TURBINE
PLANT
ByPass Vv.
Main FW
Control Vv.
#2
#3
#1 Feedwater Heaters
Main Feed Pumps
3.2

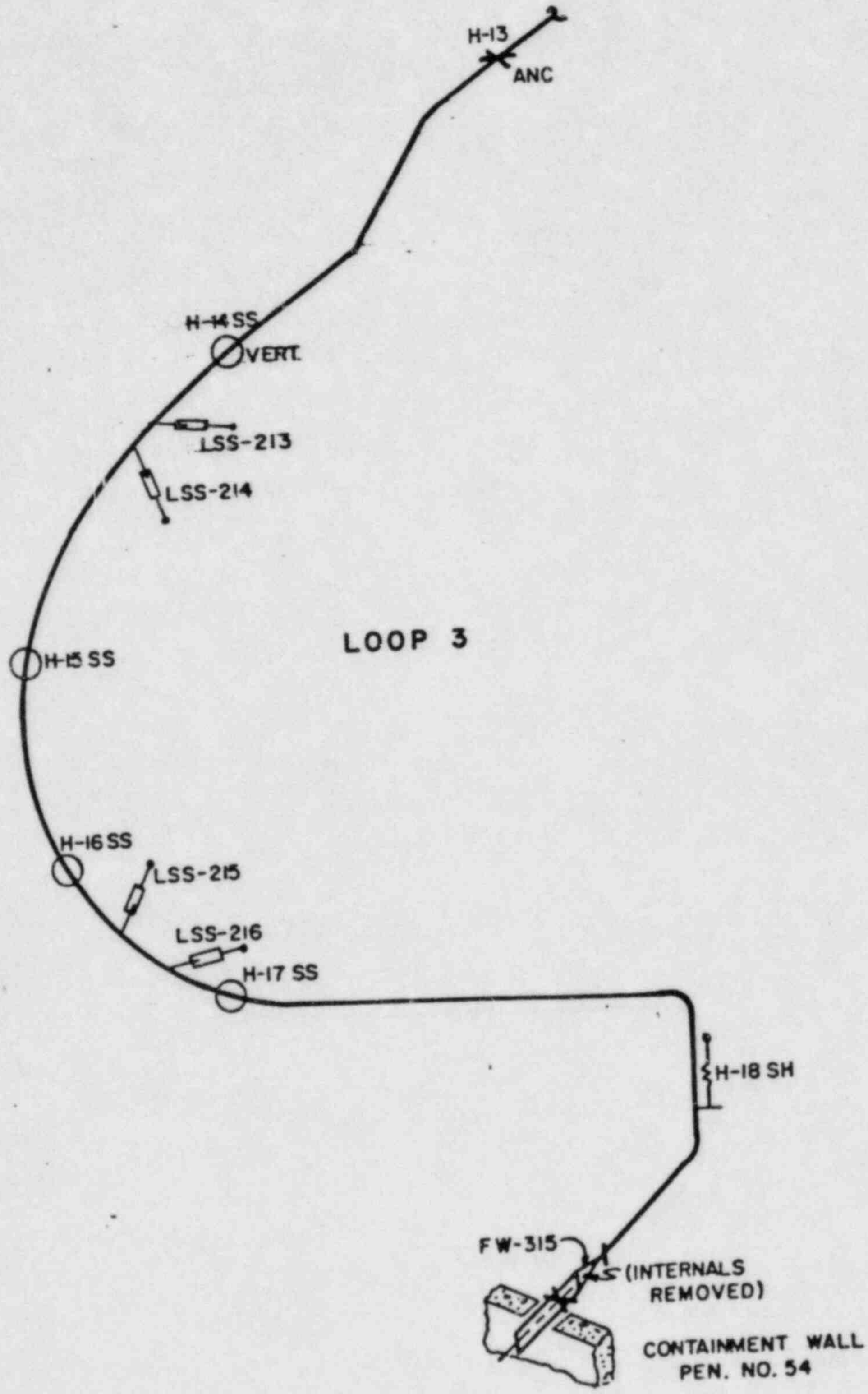
SUPPORTS

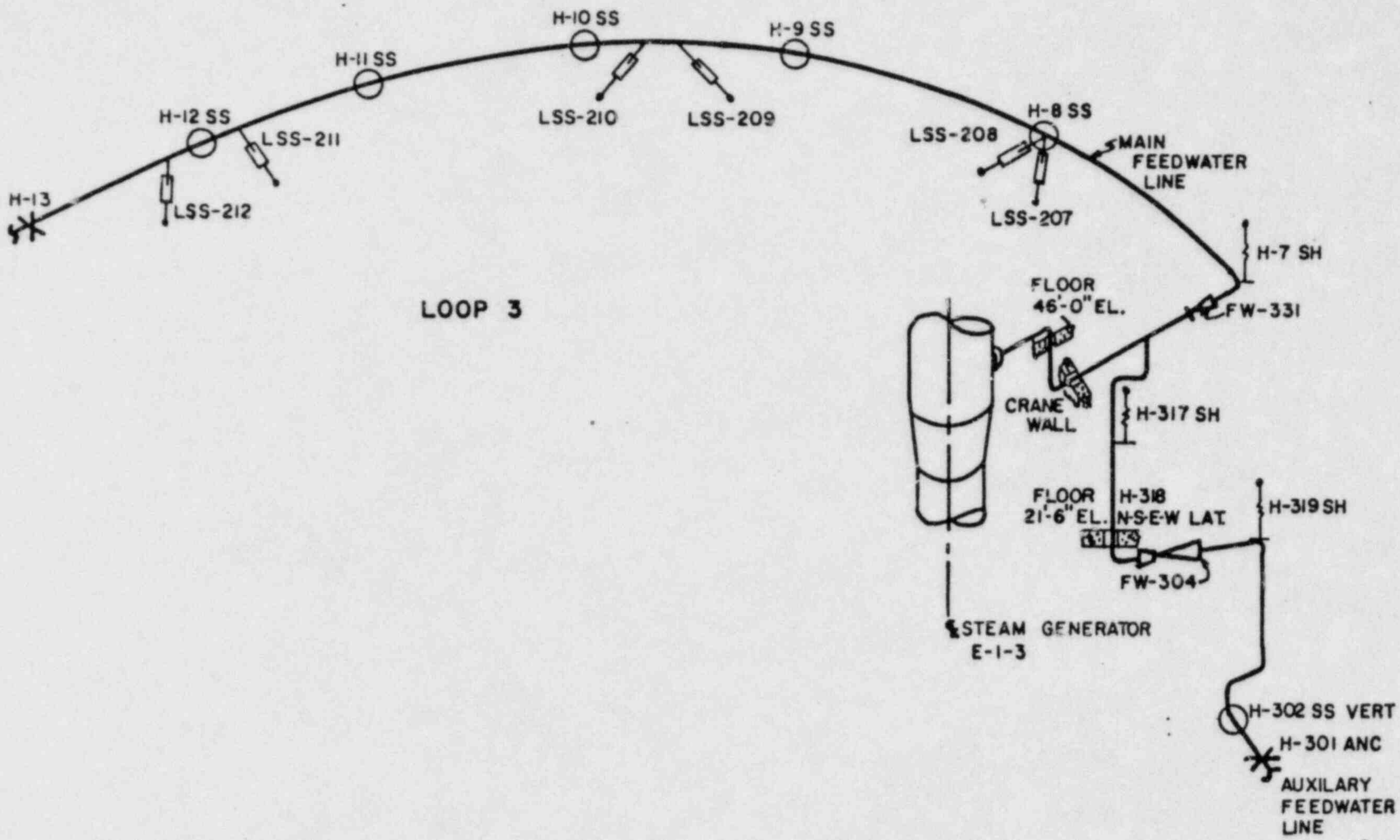
	SHOCK SUPPRESSOR
	SPRING HANGER
	SLIDING SUPPORT
	ANCHOR
HSS	HORIZONTAL SHOCK SUPPRESSOR
LSS	LATERAL " "
ASS	AXIAL " "
SS	SLIDING SUPPORT
SH	SPRING HANGER

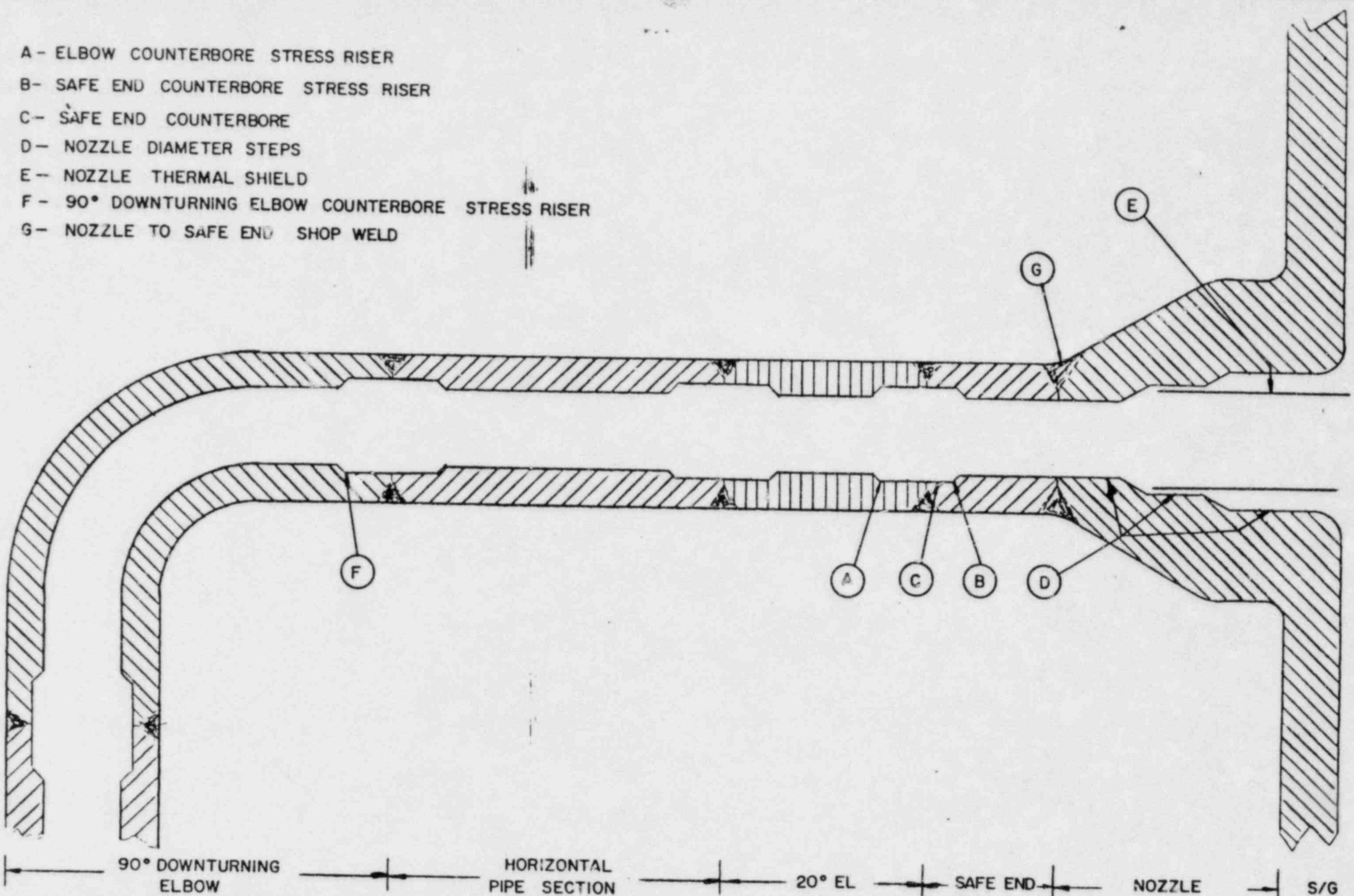

℄ STEAM GEN
E-1-1
ANC
FLOOR
46'-0" ELEV.
CRANE WALL
LSS-202
ASS-201
FW-131
H-329
SH&SS
MAIN
FEEDWATER
LINE
H-324
SH
FLOOR
21'-6" ELEV.
H-328 SH
H-325
N-S-E-W LAT.
AUXILARY
FEEDWATER
LINE
ANC
FW-115
(INTERNALS
REMOVED)
H-326 SH
H-316
ANC
CONTAINMENT WALL
PEN. NO. 53
FW-104

LOOP I


FLOOR
46'-0" EL
H-327 SH
FW-231
CRANE WALL
℄ STEAM GENERATOR
E-1-2
H-320 SH
H-2 SH
LSS-203
LSS-204
FLOOR
21'-6" ELEV.
H-321 N-S-E-W LAT.
H-322 SH
H-3 SS
VERT.
LOOP. 2
AUXILARY
FEEDWATER
LINE
H-4 SS
LSS-205
LSS-206
H-5 SS
VERT.
MAIN
FEEDWATER
LINE
ANC
FW-215
(INTERNALS
REMOVED)
CONTAINMENT WALL
PEN. NO. 55


H-13
ANC
H-14 SS
VERT.
LSS-213
LSS-214
LOOP 3
H-15 SS
H-16 SS
LSS-215
LSS-216
H-17 SS
H-18 SH
FW-315
(INTERNALS REMOVED)
CONTAINMENT WALL
PEN. NO. 54


H-13
LSS-212
H-12 SS
LSS-211
H-11 SS
H-10 SS
LSS-210
LSS-209
H-9 SS
LSS-208
H-8 SS
LSS-207
MAIN
FEEDWATER
LINE
H-7 SH
LOOP 3
FLOOR
46'-0" EL.
FW-331
CRANE
WALL
H-317 SH
FLOOR
21'-6" EL.
H-318
N-S-E-W LAT.
H-319 SH
FW-304
STEAM GENERATOR
E-1-3
H-302 SS VERT
H-301 ANC
AUXILARY
FEEDWATER
LINE


A - ELBOW COUNTERBORE STRESS RISER
B- SAFE END COUNTERBORE STRESS RISER
C- SAFE END COUNTERBORE
D- NOZZLE DIAMETER STEPS
E- NOZZLE THERMAL SHIELD
F- 90° DOWNTURNING ELBOW COUNTERBORE STRESS RISER
G- NOZZLE TO SAFE END SHOP WELD
E
G
F
A
C
B
D
90° DOWNTURNING ELBOW
HORIZONTAL PIPE SECTION
20° EL
SAFE END
NOZZLE
S/G

4) DESCRIPTION OF EVENT AND PHOTOGRAPHS

Maine Yankee Feedwater Line Leakage Event

At 1432 hours January 25, 1983, the Maine Yankee nuclear plant tripped from full load while isolating an electrical ground. Main feedwater flow was not available following the trip so steam generator level restoration was accomplished, as designed, by the auto start operation of the auxiliary feed pumps.

Approximately 15 minutes after the trip: a loud noise was heard in the plant machine shop which is located just below the main feed lines; a containment fire detector alarmed; and containment humidity began to rise. The containment was entered for inspection. The feed line was found to be leaking severely near the #2 steam generator inlet nozzle. Station cooldown was initiated to permit close access for inspection and to effect repairs.

Visual inspections conducted on January 26 revealed a feed line side crack had occurred adjacent to the pipe to steam generator nozzle weld on the upstream (feedwater) side of the weld, and that several feedwater line hangers exhibited deformation or other distress.

A program of inspection, analysis, repair and corrective modifications was initiated. The current status of each major aspect of the program follows:

Event Analysis

The leak occurred as a result of a water hammer event causing the ultimate failure of what was most probably an existing crack in the feed pipe. The crack was probably initiated by thermal cycling and non-uniform thermal effects working on a stress riser within the pipe. The water hammer event probably occurred when the outlet nozzle at bottom of the feed ring became submerged in the rising SG water level and the steam in contact with cold feedwater within the ring suddenly collapsed. The event and damage was typical of that experienced at other PWR's and as extensively described in NUREG's and other industry literature.

Plant records do not indicate the occurrence of any prior steam-water hammer events. This may be because all prior full load trips were followed by normal main feed flow providing warmer water for steam generator level restoration. Three prior trips during which loss of main feed occurred were at power levels of less than 50% so steam generator level shrinkage was far less. In two instances the auxiliary feed system was manually started and the flow rate was controlled by the operators. In one instance, the auxiliary feed system started automatically.

Feedwater Nozzles and Adjacent Pipe

All three steam generator configurations are the same. The nozzle is connected to a safeend, connected to a 20° el, connected to a 2' length of straight pipe, connected to a 90° down turning el. The connections were originally designed to minimize the horizontal run subject to possible steam-water hammer.

All pipe welds from the nozzle to and including both welds on the 90° el were radiographed (RT) on all steam generators (SGs). #2 SG's 20° el has been removed and examined. The external crack is located at the bottom of the el, right at the base of an internal stress riser. #3 SG radiograph indicated a similar crack existing at the same location propagating from the inside supporting the thermal cycle initiated crack postulation. #1 SG radiogrpah was inconclusive. Both #1 and #3 els were removed and the existance of similar cracks confirmed. All cracks are similar in location and magnitude extending from the 11 A.M. position to the 7 or 8 P.M. position looking into the SG.

The cracks occurred where one would expect: adjacent to the weld connecting the flexible pipe to the more rigid SG; at the bottom of the horizontal pipe section which is subject to the greatest amount of thermal cycling and distortion; at the base of an internal stress riser left in the pipes during installation, and close to the shock loads that occur with steam-water hammer.

Magnetic particle examination of the straight section of pipe internals where it attaches to the 20° el indicated a possible crack at the counterbore stress riser that was not evidenced by prior RT examination. All three pipe sections were removed to provide access to the 90° el horizontal end, which also contained a similar stress riser, and to the nozzles. The safe ends were examined by LP as part of the preparation for repair. In this examination two types of indications were observed. Circumferential indications were found at the counterbore stress riser. These were chased by grinding until they were clear as indicated by additional LP checks. Also, minor surface indications were found in the ID of the safe end counterbore. However, these indications were cleared by buffing the surface. Thus, the safe ends were determined to be satisfactory.

The steam generator nozzles expand into the steam generators in two steps. Visual and LP examination of the first step up in inner diameter revealed indications of cracking. These cracks were chased by grinding in accordance with CE material removal specifications, which allowed up to 1/8" of metal to be removed without necessitating repair by rewelding. The cracks were cleared before this metal removal limit was reached. The second step up in inner diameter is covered and protected by a thermal sleeve which precludes internal surface inspection.

All other feedwater piping welds back to the feedwater check valves were inspected by RT and found to be clear. In addition, the auxiliary feedwater connections to the feedwater piping (just downstream of the feedwater check valves) were inspected by mag particle and found to be clear.

On all three SG's, the 90° el horizontal end stress riser has been removed, the horizontal pipe and 20° el has been replaced with sections whose stress risers have been removed. The repair welds and transition will be radiographed, ultrasoncially tested to establish a baseline, and the entire pipe hydro tested.

Steam Generator Internals

SG feed rings and attachments were inspected. Only minor deformation of a support was noted. The numer two and three steam generator thermal sleeve appears to be somewhat expanded into the nozzle. The feed ring support will be repaired.

Each of the SG feed rings will be modified by closing off the 76 bottom 1" dia. nozzles and installing 28 top mounted J tubes (so-called) of 3" dia. This is a fix previously used at other PWR's that provides greater assurance that the feed ring will not empty and (300%) greater pressure equalization capability.

The expansion of the thermal sleeve into the nozzle will reduce the rate of ring draining in a loss of feed event which is helpful. However, it may reduce the number of lifetime thermal cycles permitted on the nozzle. An analysis will be completed prior to the next refueling outage and modifications made if warranted.

Feedwater Lines, Hangers & Support

Based upon support deformation and other evidence, it appears that the amount of line movement was not excessive. Support attachments to pipe were inspected and no evidence of overstress or deformation was found.

#2 & #3 feedwater line check valves have been opened and inspected. No damage was found.

Modification to Prevent Reoccurrence

1. Stress risers have been removed from the horizontal portions of the pipe, els and safe end.

2. J tubes have been installed on feed rings to prevent draining, reducing water hammer probability.

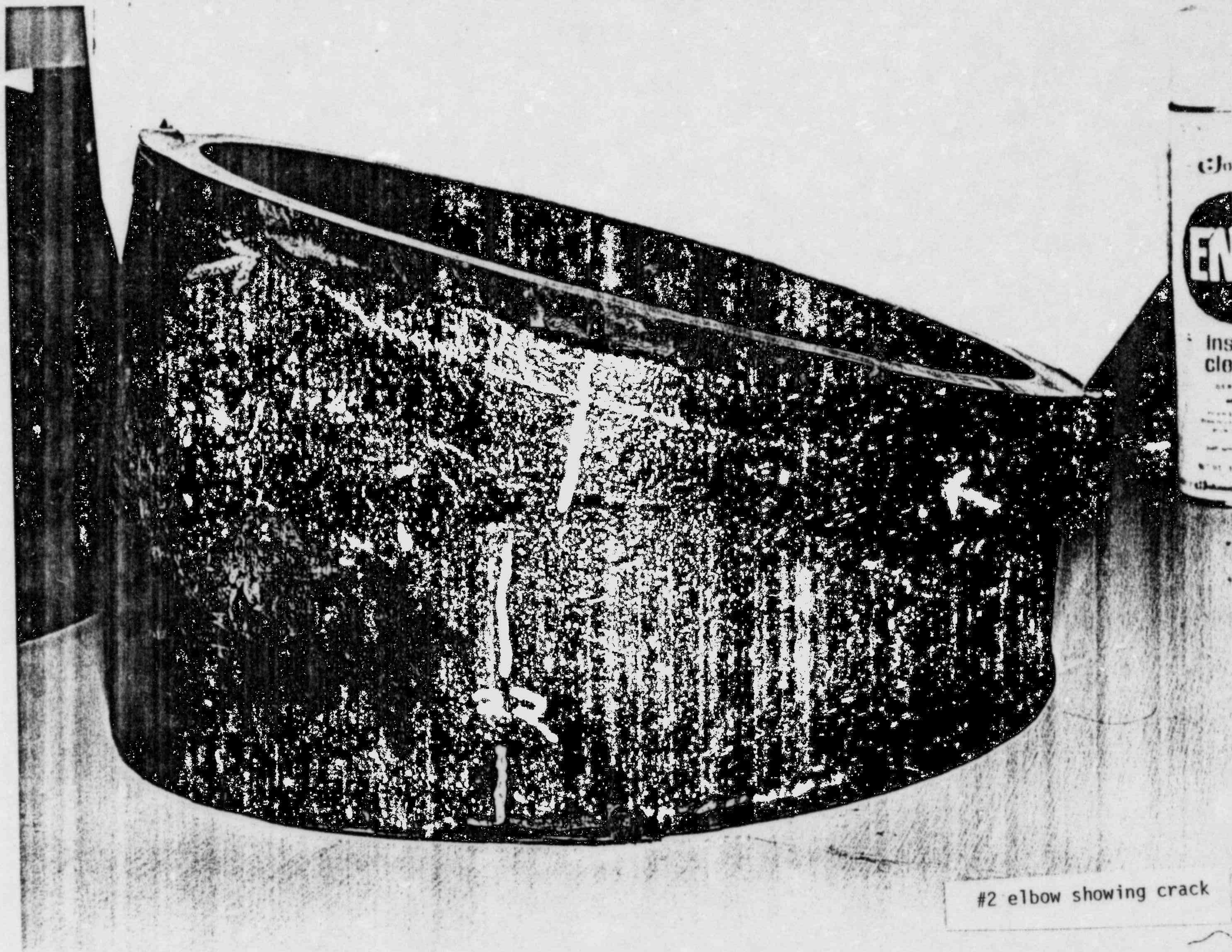
3. Operations instruction and guidance have been provided to reduce the probability of thermal cycling and feedwater hammer events.

PHOTOGRAPHS

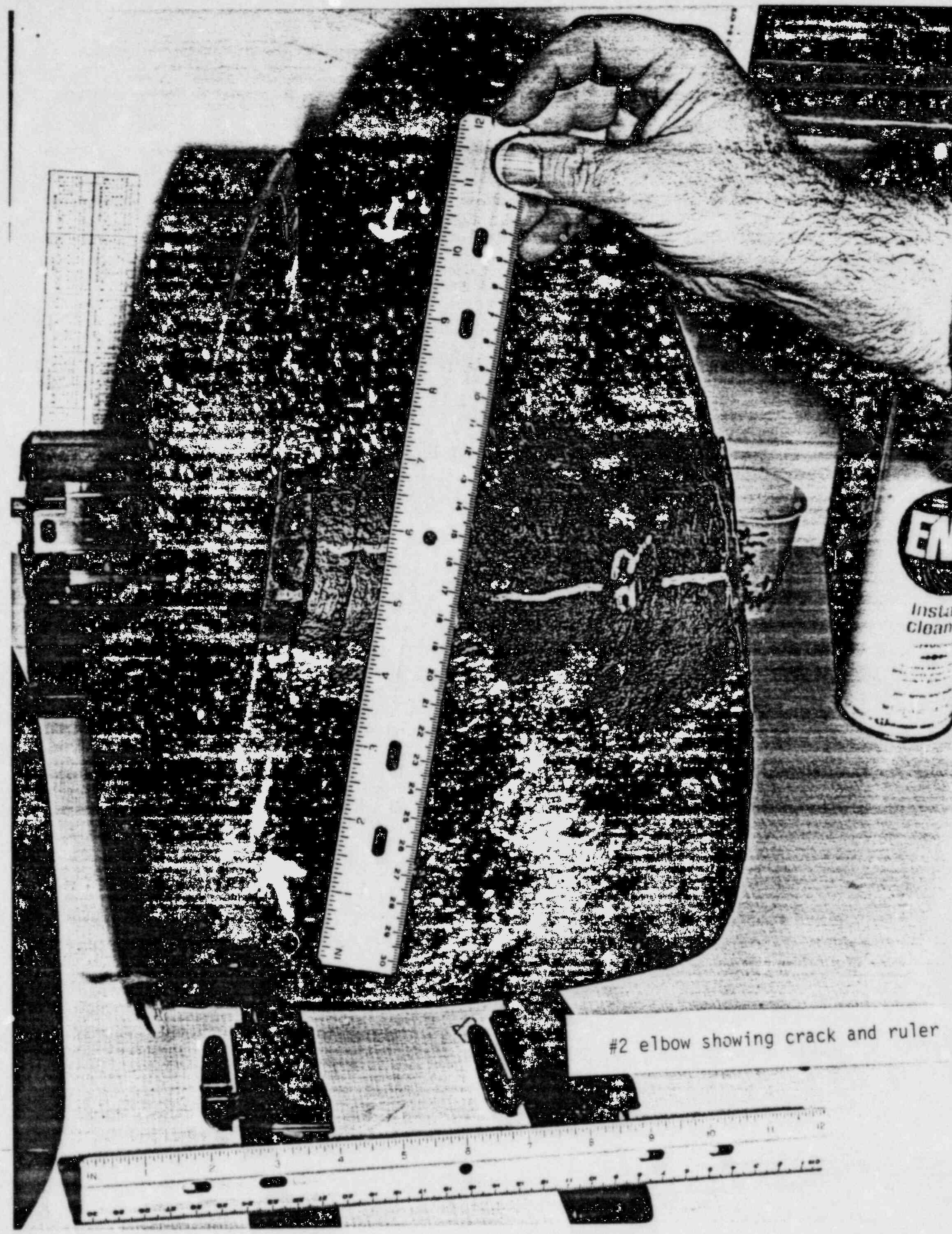
1. #2 Feedwater pipe elbow showing counterbore and crack.
2. #2 elbow showing through wall crack at outside.
3. #2 elbow showing crack.
4. #2 elbow showing crack and ruler
5. #2 SG nozzle, safe end, 20° elbow, short horizontal pipe section, and 90° elbow to vertical. Also shown: pipe cotter in place and chem addition line.
6. #3 Feedwater line horizontal support.

#2 Feedwater pipe elbow showing
counterbore and crack

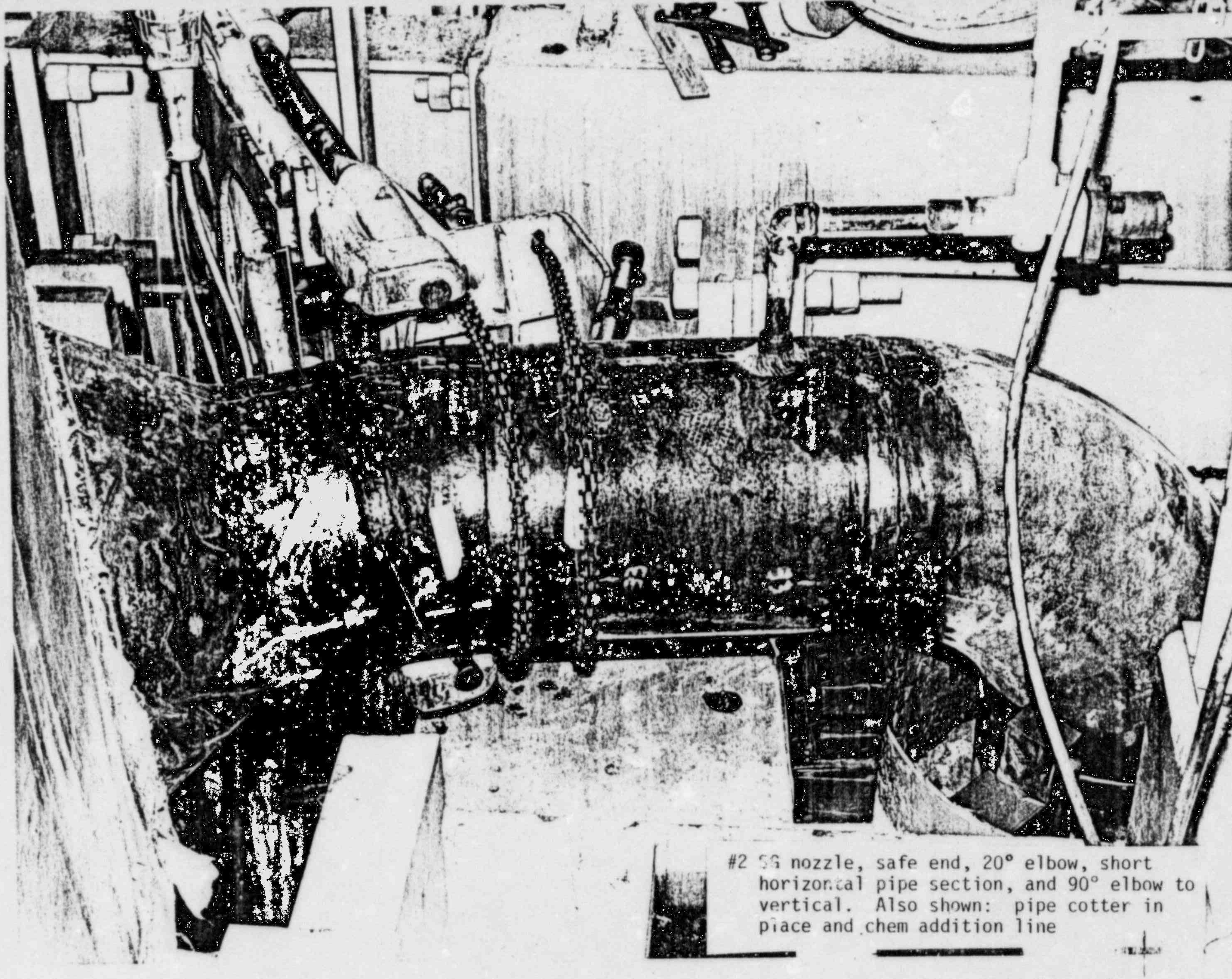
#2 elbow showing through wall crack at outside



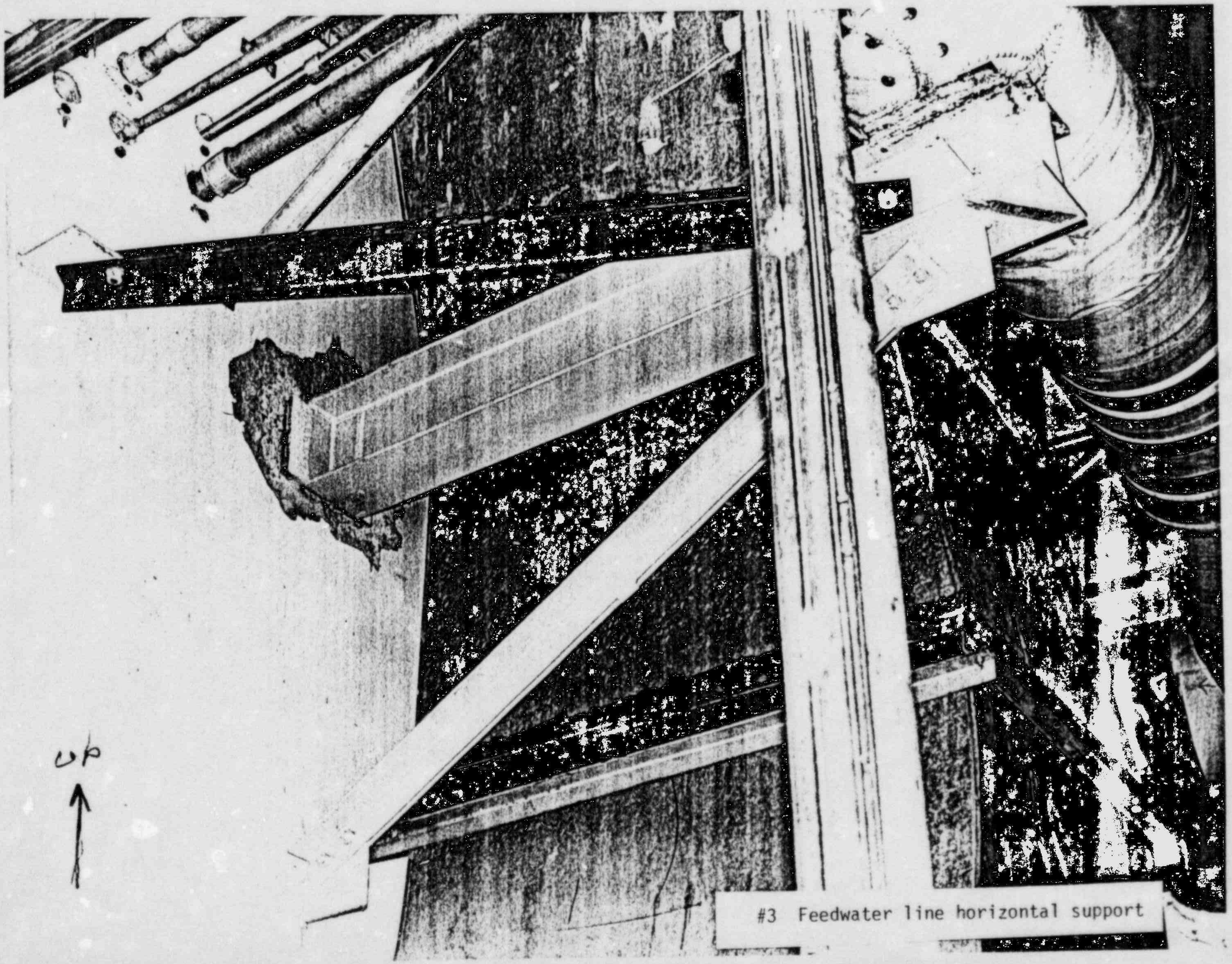
#2 elbow showing crack



#2 elbow showing crack and ruler



#2 SG nozzle, safe end, 20° elbow, short horizontal pipe section, and 90° elbow to vertical. Also shown: pipe cotter in place and chem addition line



#3 Feedwater line horizontal support

5) MAINE YANKEE FEEDWATER PIPE

DAMAGE MECHANISM

1. CAUSES OF CRACKING

   A. CRACK INITIATION

   B. CRACK PROPAGATION

2. STEAM CONDENSATION/WATERHAMMER INCIDENT

   A. HOW IT HAPPENED

   B. EXPLANATION OF DAMAGE

COMMENTARY:

1. Cracks are initiated by thermal cycling. Crack progression in excess of 0.1" is unlikely except where accompanied by non-uniform thermal condition. Non-uniform thermal conditions frequently occur in the steam generator nozzle and attached horizontal pipe sections during periods of low flow of cold feedwater.

2. The water hammer incident is similar to that experienced at other PWR's and probably caused the thermal sleeves in S.G. #2 and #3 to expand.

3. The expanded thermal sleeves are probably still 75% effective and adequate until the next refueling outage. A more detailed evaluation will be conducted prior to that outage.

POWER SYSTEMS

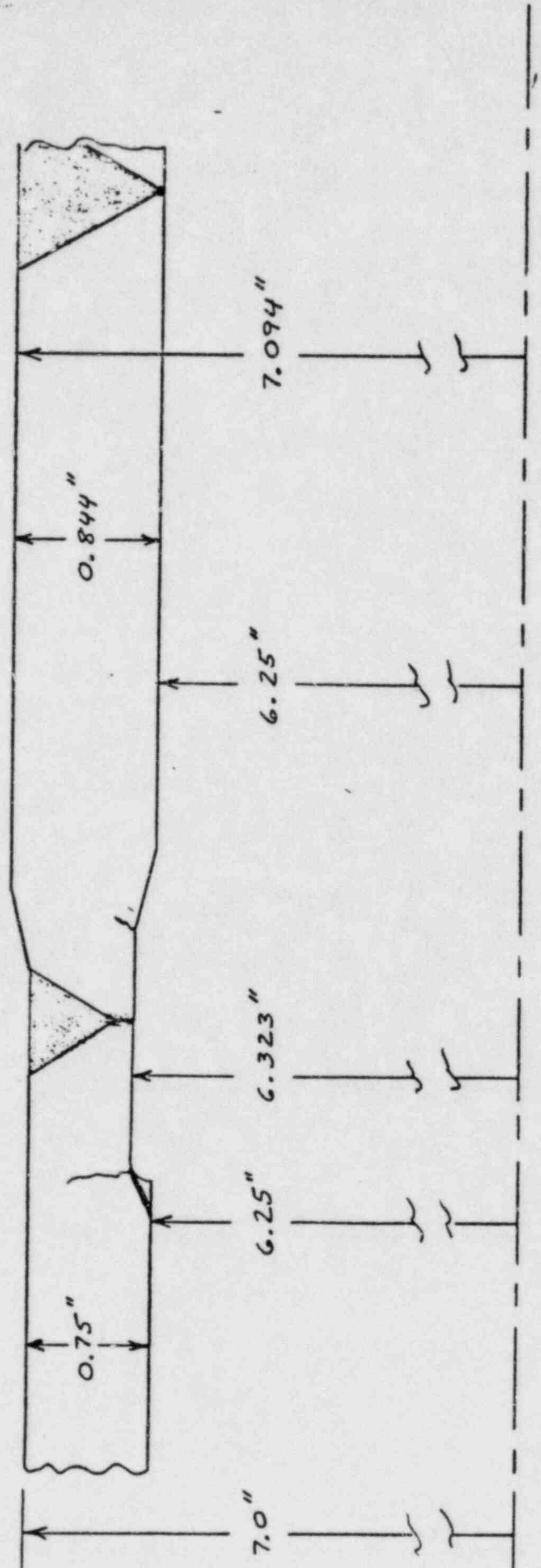
NUCLEAR POWER SYSTEMS
CHATTANOOGA NUCLEAR OPERATIONS

CAUSES OF CRACKING

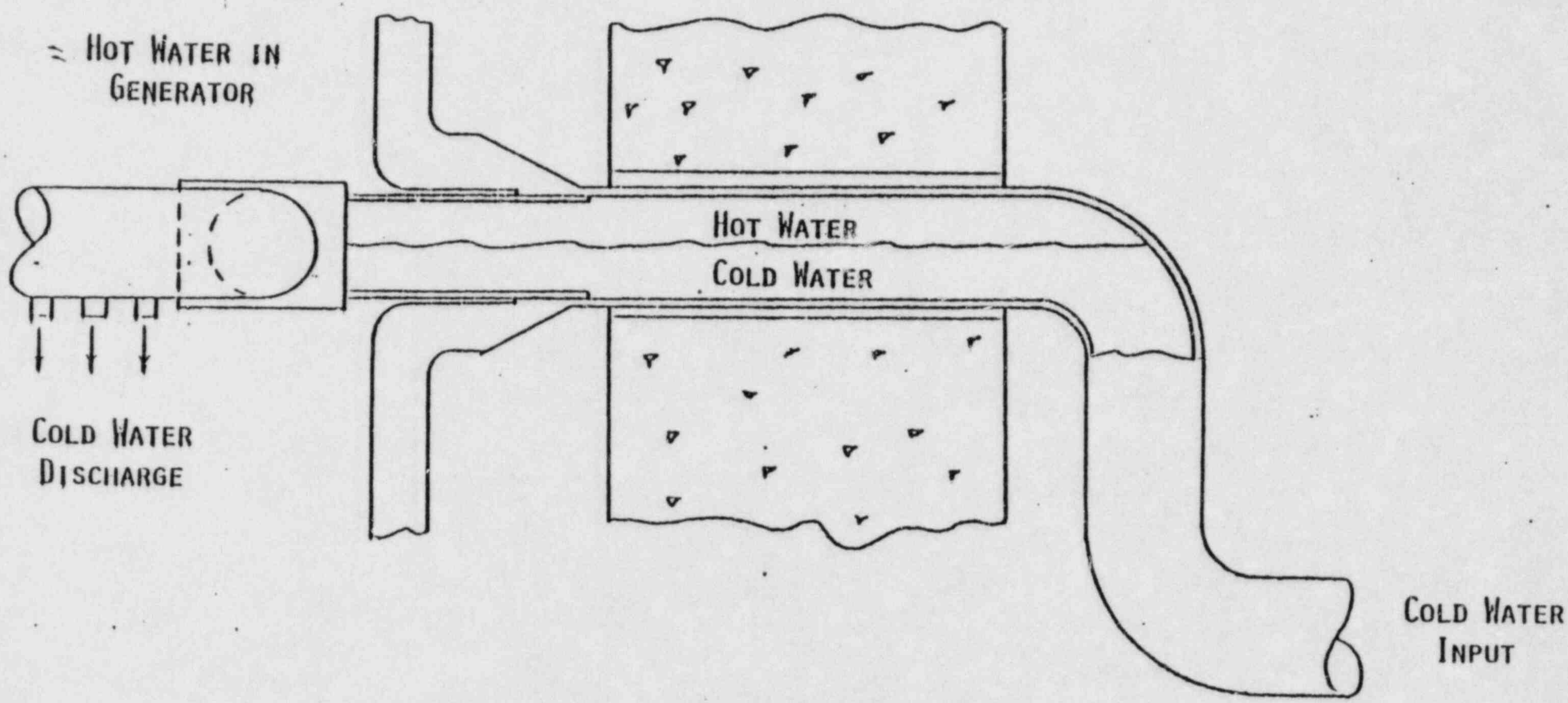
A. CONFIGURATION OF PIPE TO NOZZLE JUNCTURE.

B. INJECTING SMALL FLOW RATES OF COLD WATER THROUGH PIPE AND SUBSEQUENT 'STRATIFICATION' OF FLOW,

C. TRADITIONAL THERMAL STRESS AND PIPE LOAD STRESS.



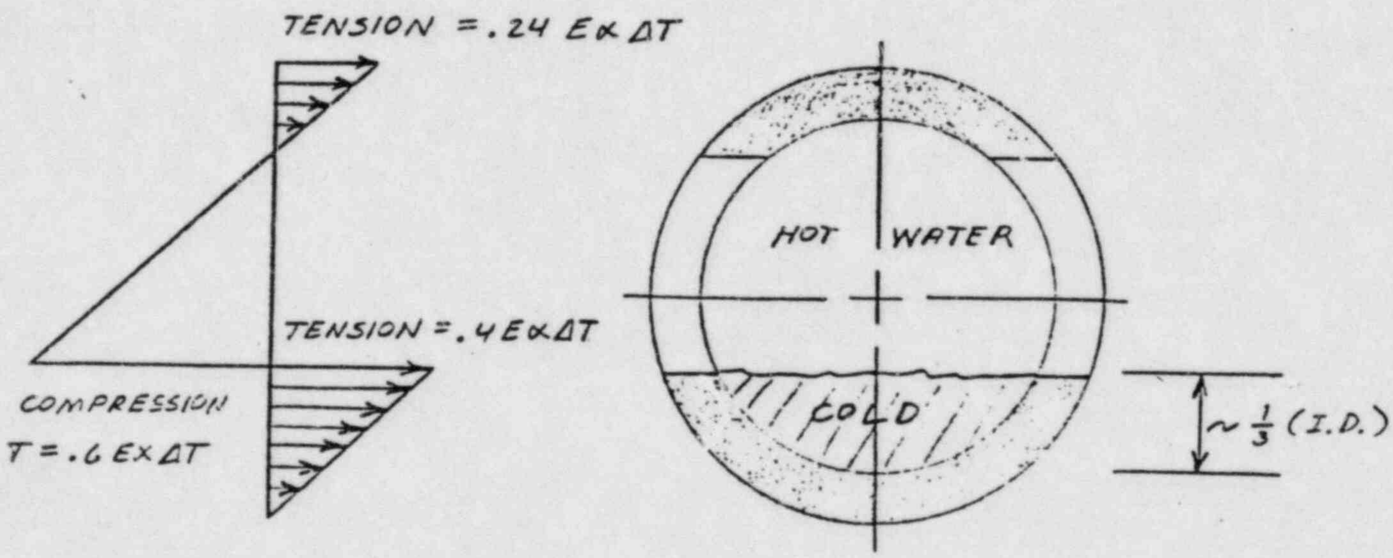
MAINE YANKEE S.G. PIPE WELD GEOMETRY



.STRATIFICATION IN FEEDWATER PIPE

5.4

MAINE YANKEE FEEDWATER PIPE
POTENTIAL STRESS DISTRIBUTION



AREAS OF TENSILE STRESS HAVE THE CAPACITY TO PROPAGATE INITIATED CRACKS

TRADITIONAL THERMAL AND PIPE LOAD STRESSES

1. THERMAL STRESSES

   A. RADIAL GRADIENT "SKIN" STRESS-SLUG FEEDING

   B. FLUCTUATING HOT/COLD FLUIDS DUE TO UNSTABLE MIXING, STREAMING AND LEVEL FLUCTUATION

2. PIPE LOAD STRESSES

   A. THERMAL EXPANSION - HEATUP

   B. CHANGE IN THERMAL EXPANSION DUE TO:

      (1) POWER CHANGES

      (2) COLD FEEDING

      (3) SUPPORT BINDING

STEAM CONDENSATION/WATERHAMMER INCIDENT

SEQUENCE OF EVENTS

1. REACTOR TRIP

2. TURBINE TRIP/COASTDOWN OF STEAM DRIVEN PUMPS

3. LOSS OF FEEDWATER TO S.G./LOW WATER LEVEL

4. INITIATION OF AUXILIARY FEEDWATER/REFILL

5. WATER LEVEL REACHES BOTTOM DISCHARGE NIPPLES

6. RAPID STEAM CONDENSATION INSIDE SPARGER

7. RAPID CONDENSATION IN PIPE

8. OPPOSING WATER COLUMNS ACCELERATE AND COLLIDE

9. WATERHAMMER CAUSES PRESSURE SPIKE AND PRESSURE WAVE PROPAGATES THROUGH SYSTEM

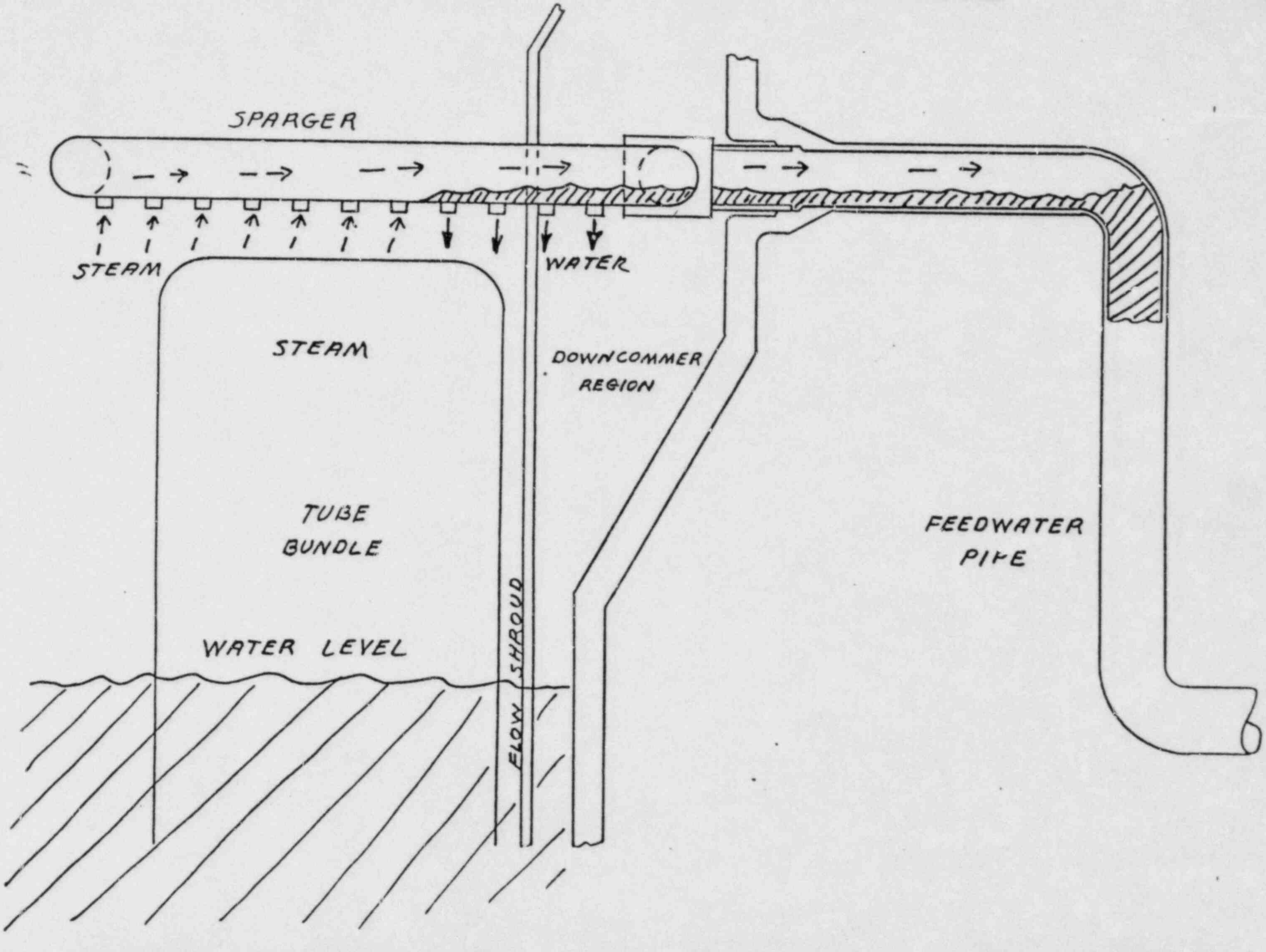
10. DAMAGE OCCURS:

    A. THERMAL LINER YIELDS AND EXPANDS

    B. PIPE CRACK PROPAGATES THROUGH WALL (S.G.2)

    C. PIPE HANGERS DAMAGED (S.G.3)

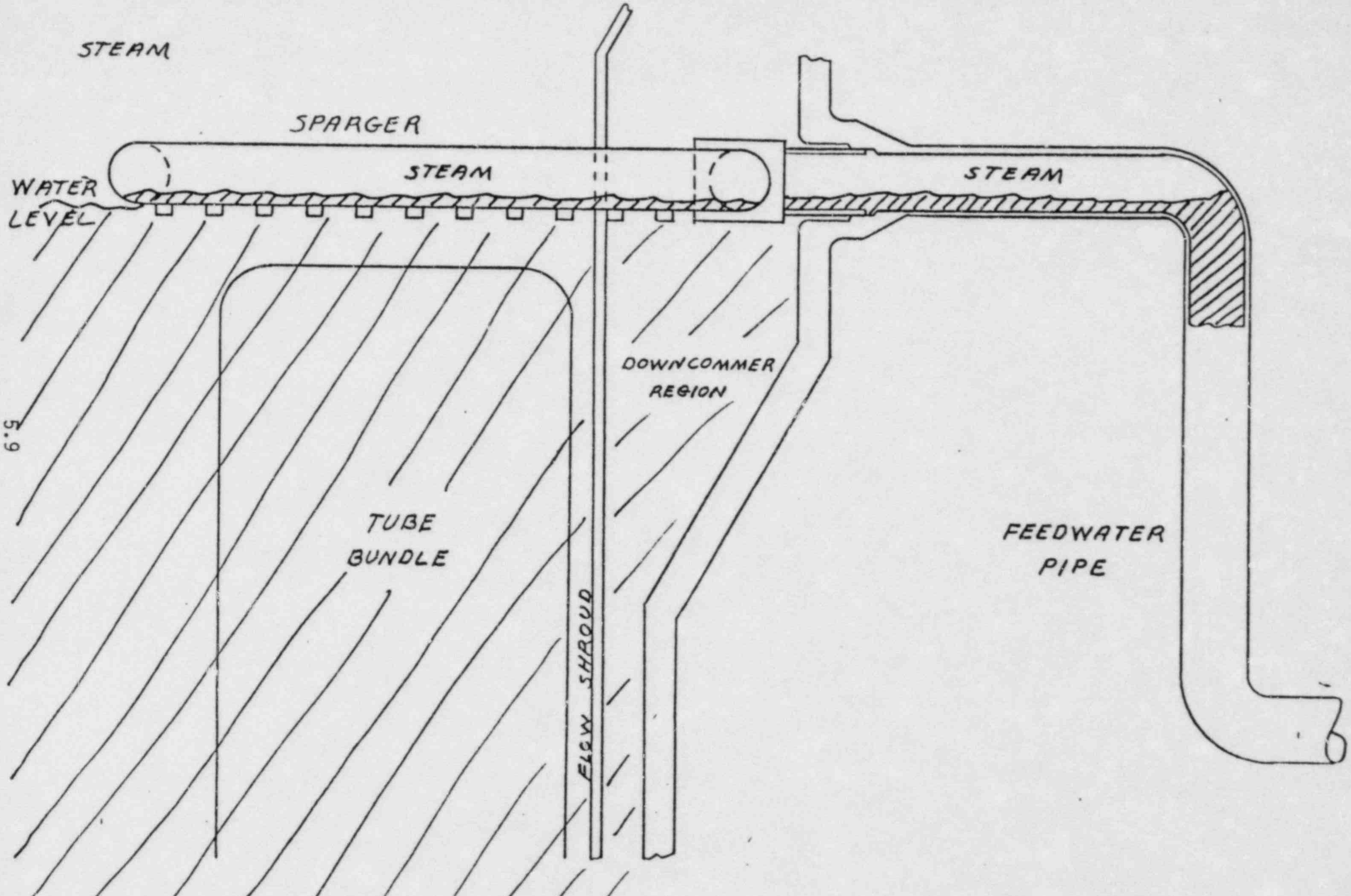
SCHEMATIC - MAINE YANKEE STEAM GENERATOR
REFILL - COLD AUXILIARY FEEDWATER FLOW



5.8

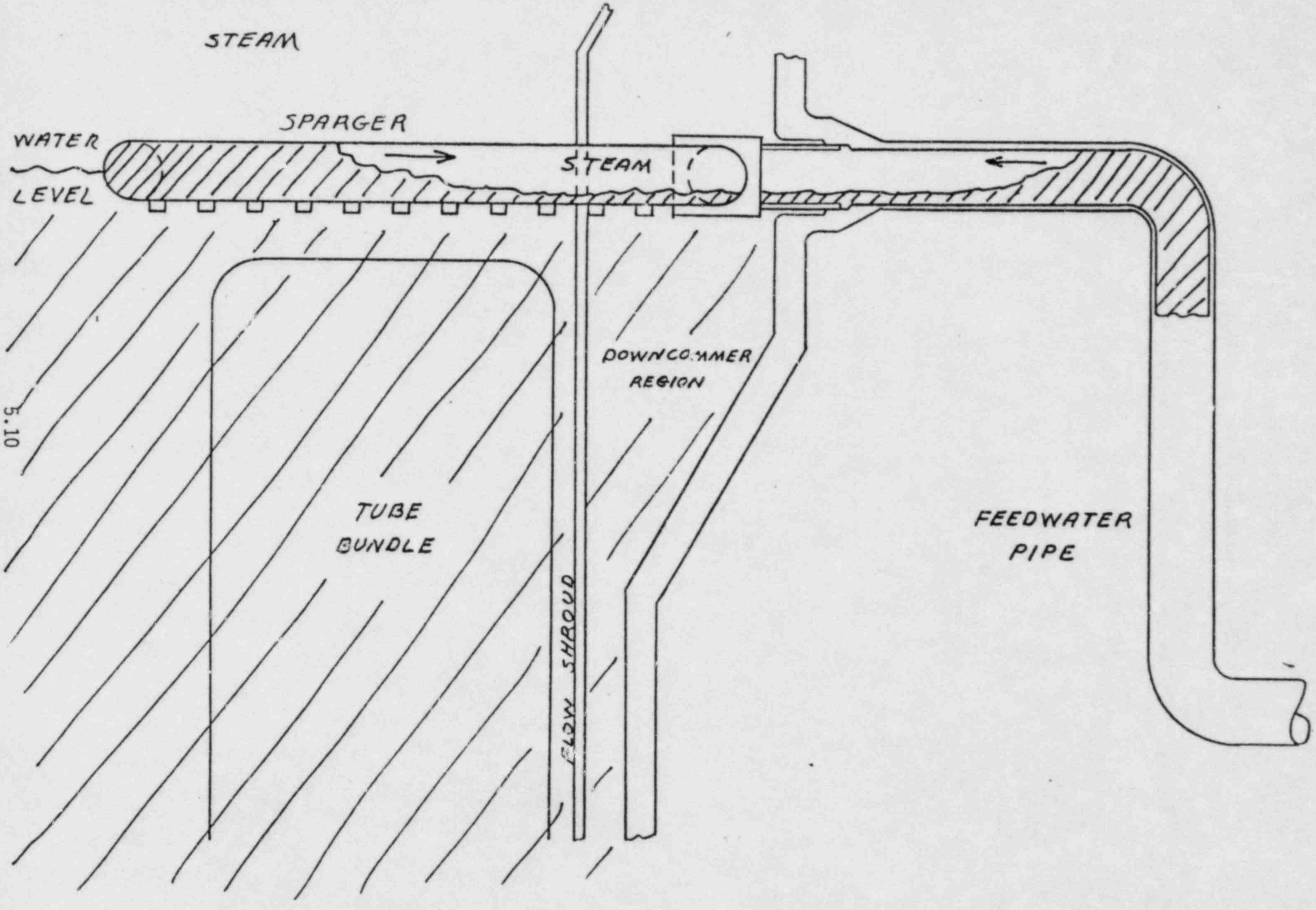
SCHEMATIC - MAINE YANKEE STEAM GENERATOR

WATER LEVEL REACHES NIPPLES - AUX. FLOW

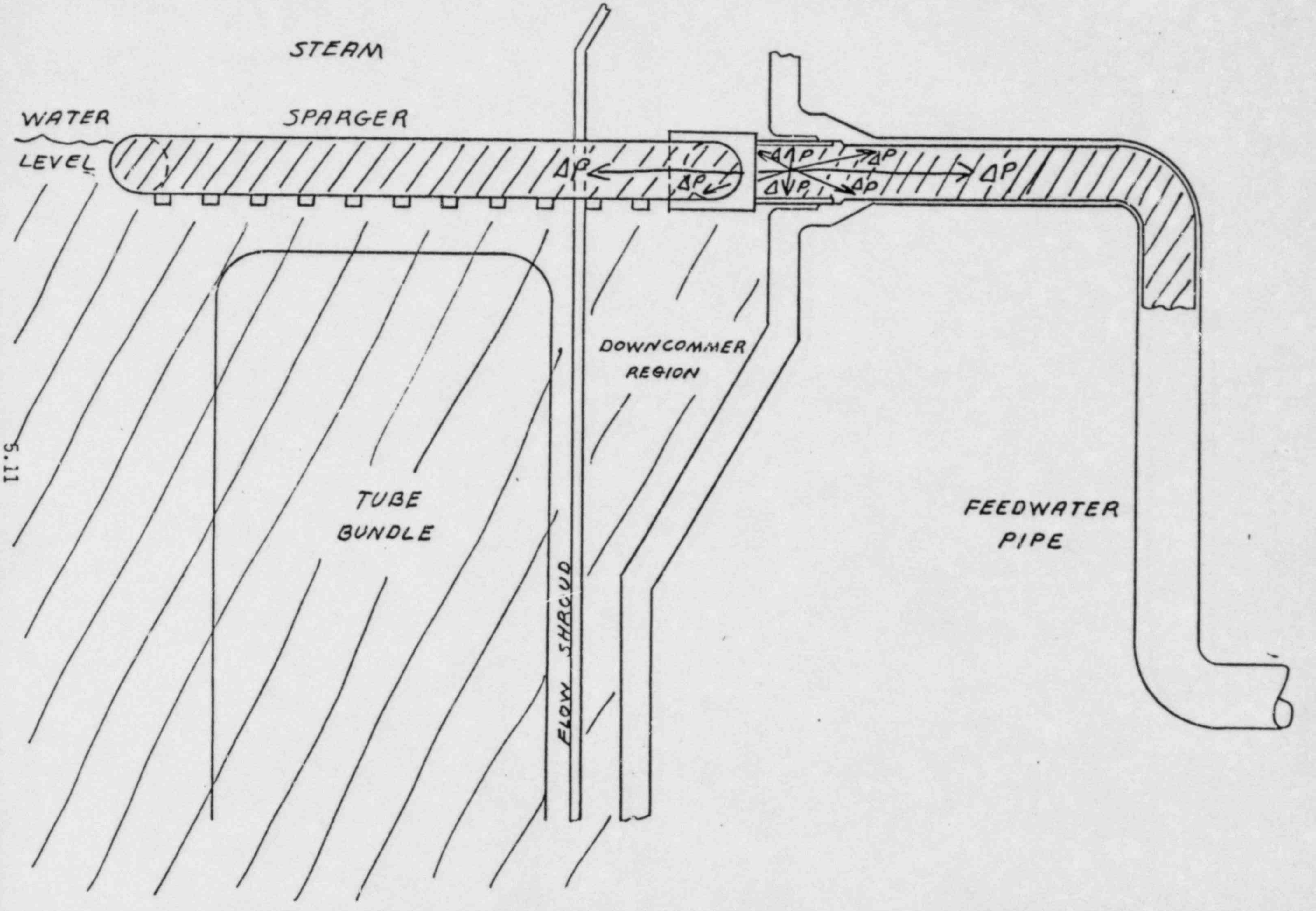


SCHEMATIC - MAINE YANKEE STEAM GENERATOR

RAPID CONDENSATION INSIDE SPARGER



SCHEMATIC - MAINE YANKEE STEAM GENERATOR
WATER HAMMER - PRESSURE SPIKE INSIDE SPARGER



6) INSPECTION AND REPAIR

- FEEDWATER PIPING
- SUPPORTS AND HANGERS
- NOZZLES
- STEAM GENERATOR INTERNALS

INSPECTION PLAN

I. Visual Inspection of Pipe Supports

a. Inspect feedwater supports between the steam generators and the main feed valves.
b. Inspect all auxiliary feedwater supports from the feedwater line tee, back to and including the containment penetrations.
c. Inspect pipe in vicinity of supports for damage or distress.

II. Functional Test of Shock Suppressors

a. Test the two suppressors that failed the visual inspection.
b. Test an additional sample if one of the first two suppressors fails to operate properly.

III. Surface Examinations (MP, LP)

a. Examine the pressure boundary welds at each anchor point in the feedwater system.
b. Examine the pressure boundary welds attaching the distressed shock suppressors.
c. Examine the inside diameter of the steam generator nozzle-to-safe end welds. (Includes safe end I.D. surfaces.)
d. Weld prep surfaces.

IV. Visual Inspection of Check Valve Internals

a. Feedwater valves FW-113, 213, and 313.

V. Radiographs of Feedwater Lines in Containment

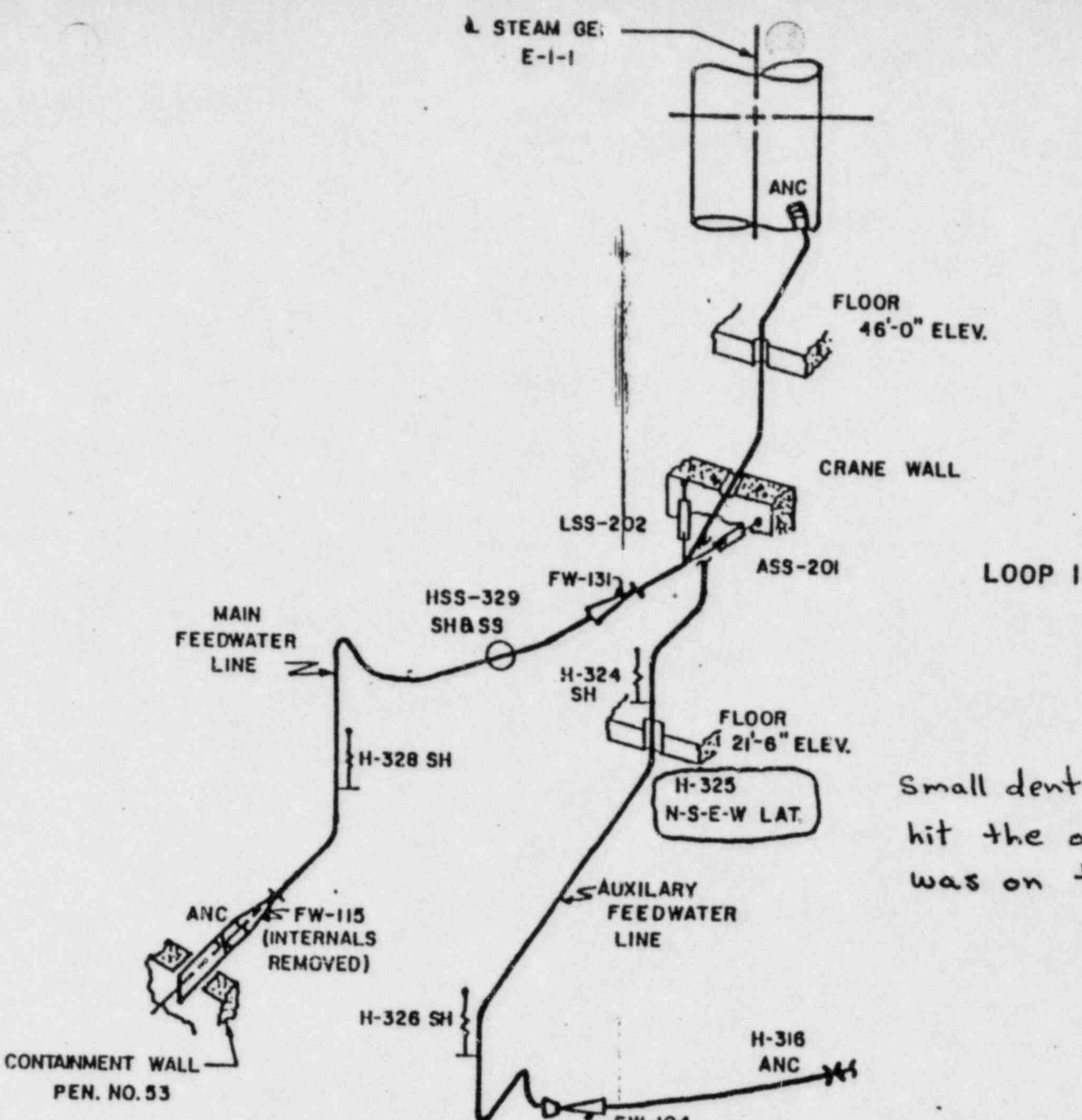
a. All butt welds, between the steam generators and the check valves.
b. Welds that had previous indications.

VI. Class "B" Leak Test of Feedwater Penetrations

VII. Four Hour Hydro of the Steam and Feed System

a. Steam generator and steam lines up to the excess flow check valves.
b. Blowdown lines to the first isolation valves.
c. Feedwater lines to the flow control valves.
d. Auxiliary feedwater line to the first check valve.

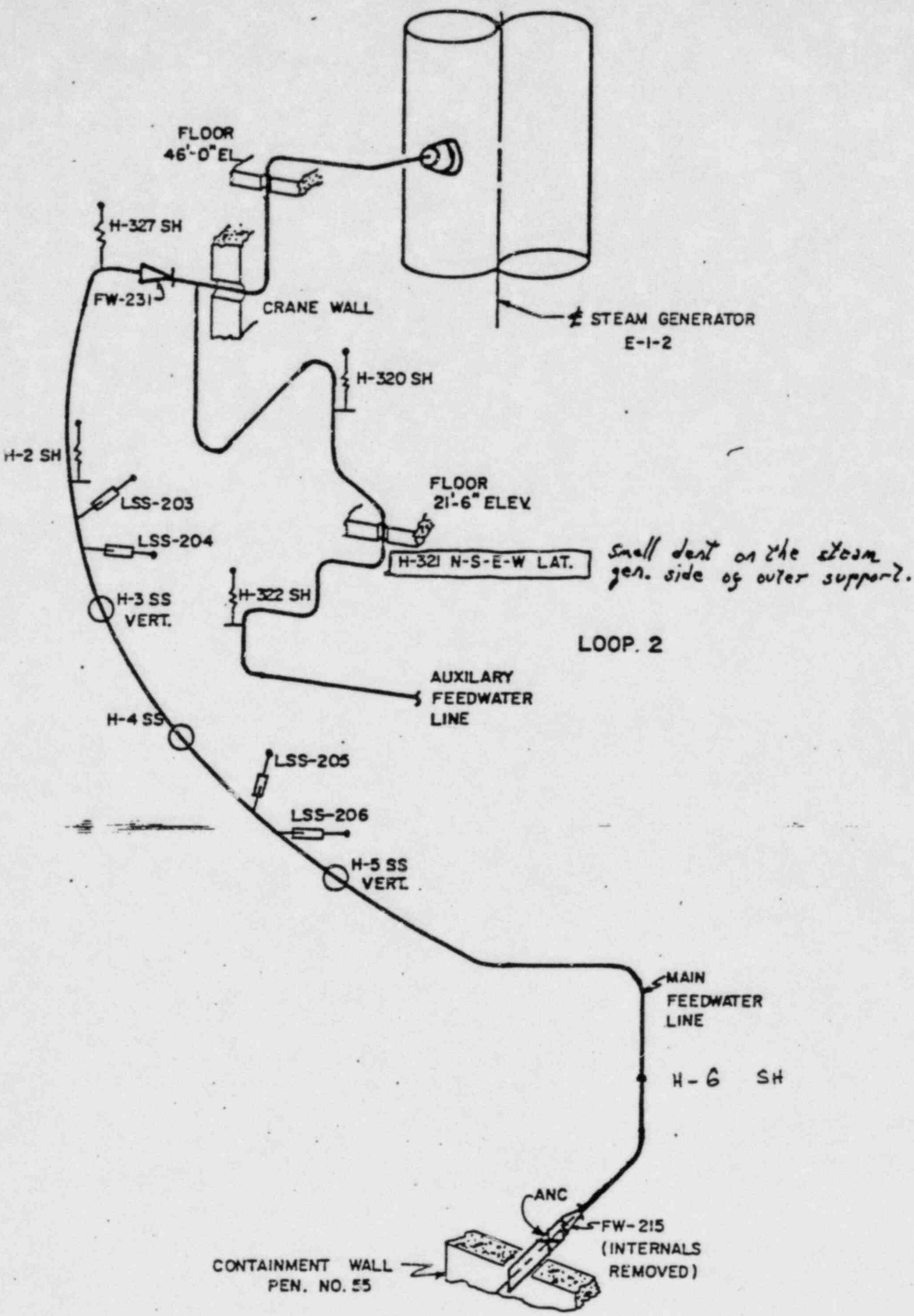
VIII. Inspection for Wetted Insulation on S.S. Piping

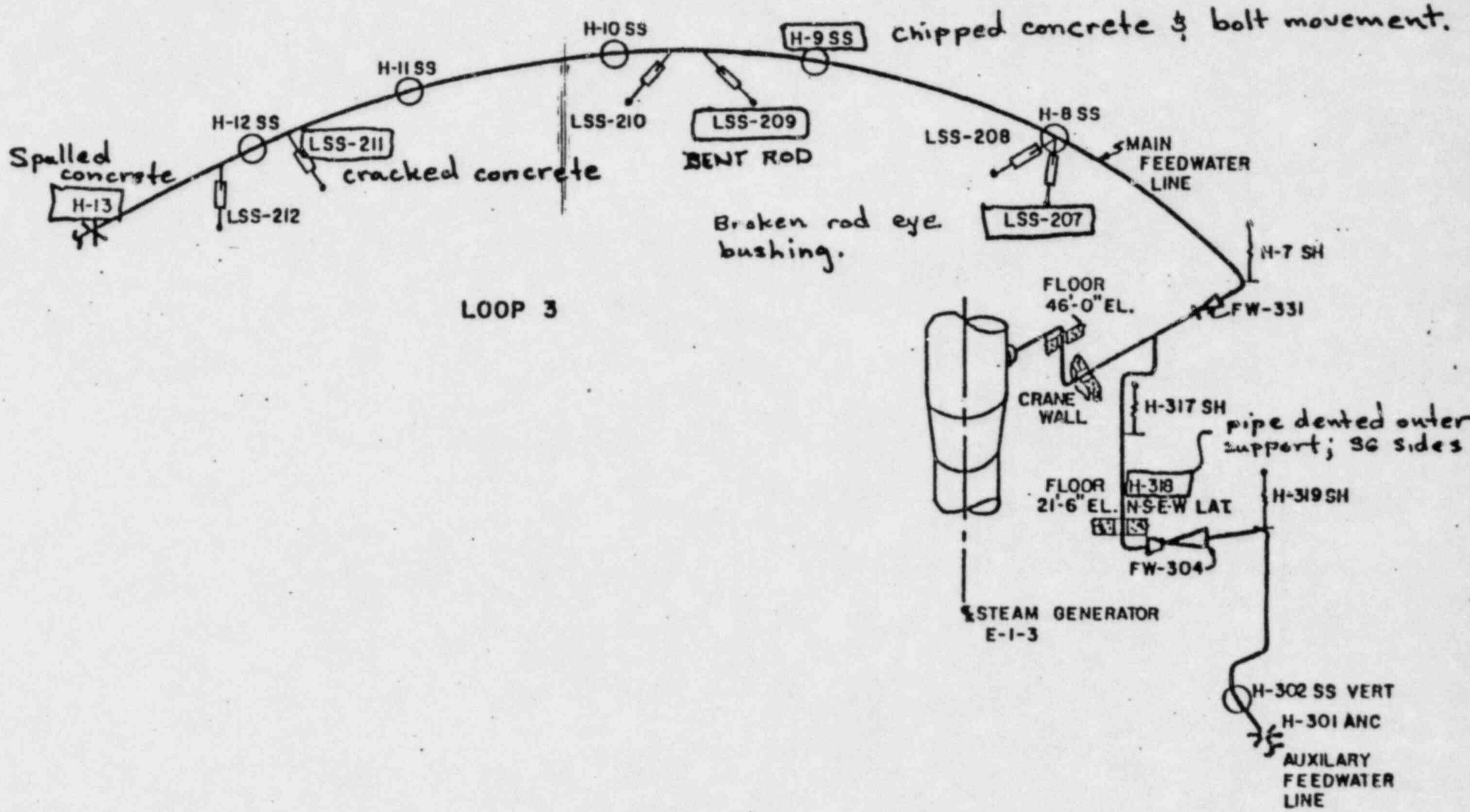


Small dent where the pipe lug hit the outer support. The dent was on the steam generator side.

6.3

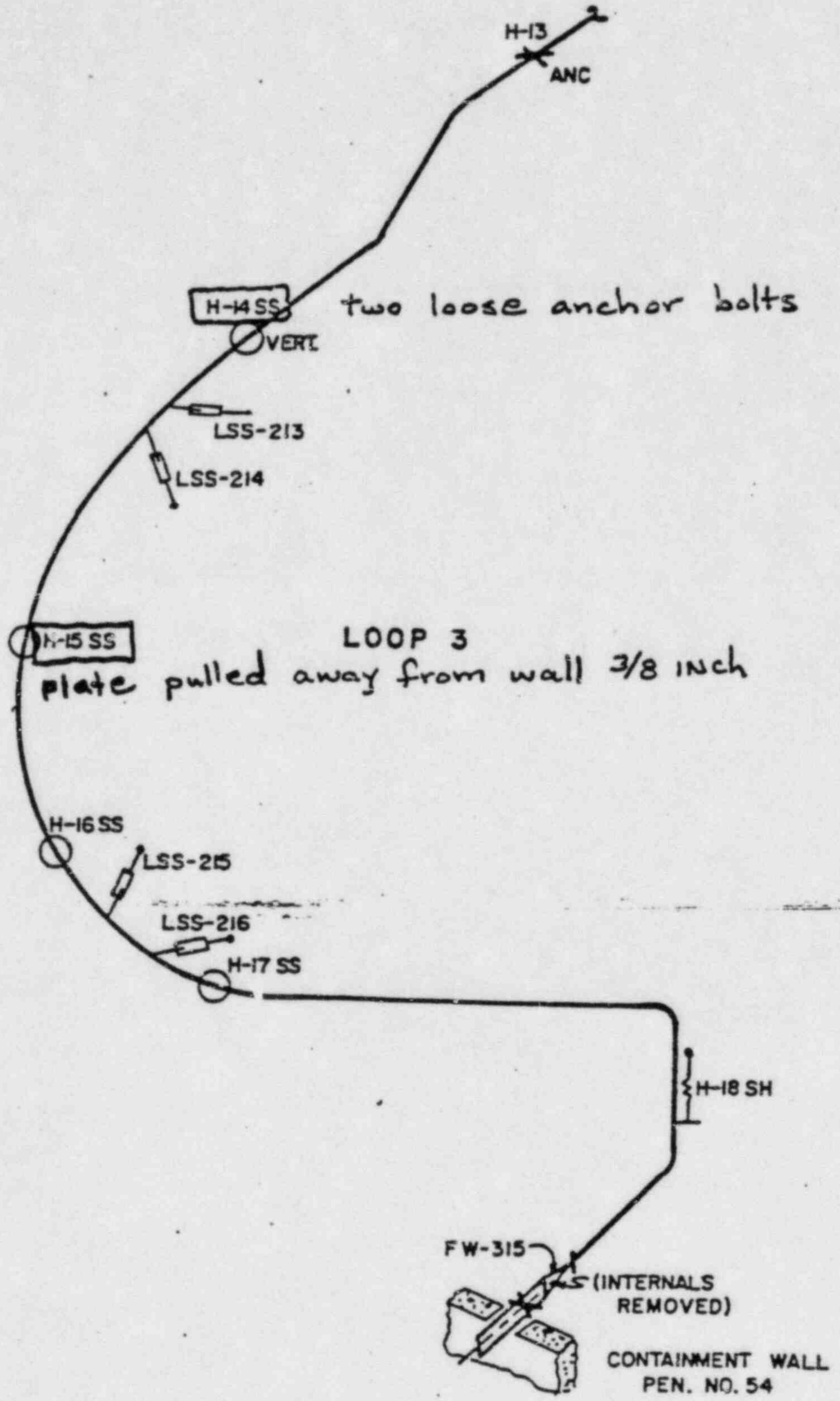
5.2


FLOOR
46'-0" EL
H-327 SH
FW-231
CRANE WALL
℄ STEAM GENERATOR
E-1-2
H-320 SH
H-2 SH
LSS-203
FLOOR
21'-6" ELEV.
LSS-204
H-321 N-S-E-W LAT.
Small dent on the steam gen. side of outer support.
H-322 SH
H-3 SS
VERT.
LOOP. 2
AUXILARY
FEEDWATER
LINE
H-4 SS
LSS-205
LSS-206
H-5 SS
VERT.
MAIN
FEEDWATER
LINE
H-6 SH
ANC
FW-215
(INTERNALS
REMOVED)
CONTAINMENT WALL
PEN. NO. 55


H-10 SS
H-9 SS
chipped concrete & bolt movement.
H-11 SS
H-12 SS
LSS-211
LSS-210
LSS-209
H-8 SS
LSS-208
MAIN FEEDWATER LINE
Spalled concrete
cracked concrete
BENT ROD
H-13
LSS-212
Broken rod eye bushing.
LSS-207
H-7 SH
LOOP 3
FLOOR 46'-0" EL.
FW-331
CRANE WALL
H-317 SH
pipe dented outer support; SG sides.
FLOOR 21'-6" EL.
H-318
N-S-E-W LAT
H-319 SH
FW-304
STEAM GENERATOR E-1-3
H-302 SS VERT
H-301 ANC
AUXILARY FEEDWATER LINE

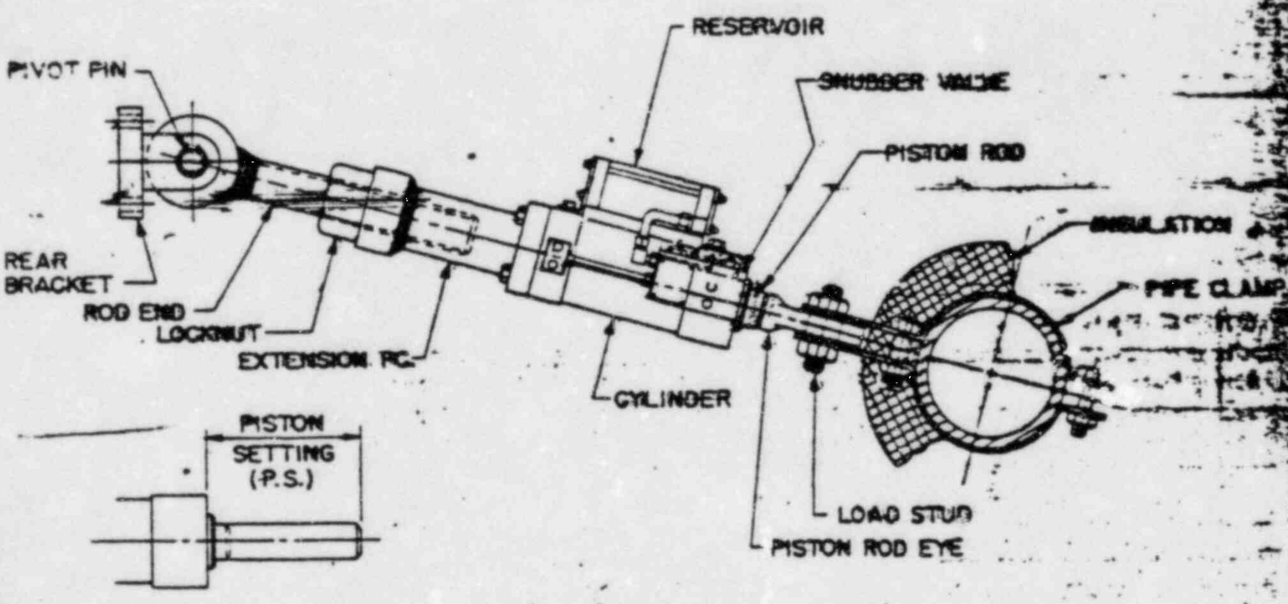
6.5

5.4


H-13
ANC
H-14 SS
two loose anchor bolts
VERT.
LSS-213
LSS-214
LOOP 3
H-15 SS
plate pulled away from wall 3/8 inch
H-16 SS
LSS-215
LSS-216
H-17 SS
H-18 SH
FW-315
(INTERNALS REMOVED)
CONTAINMENT WALL
PEN. NO. 54

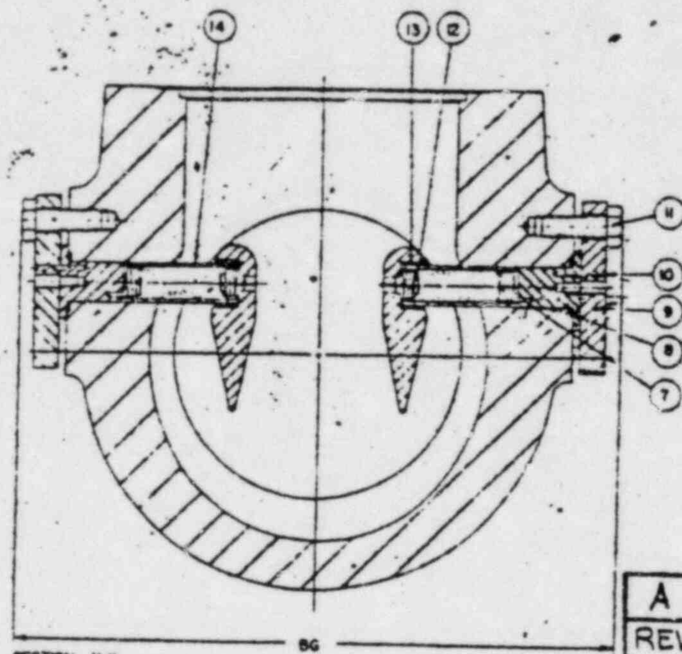
5.5

fig. 201



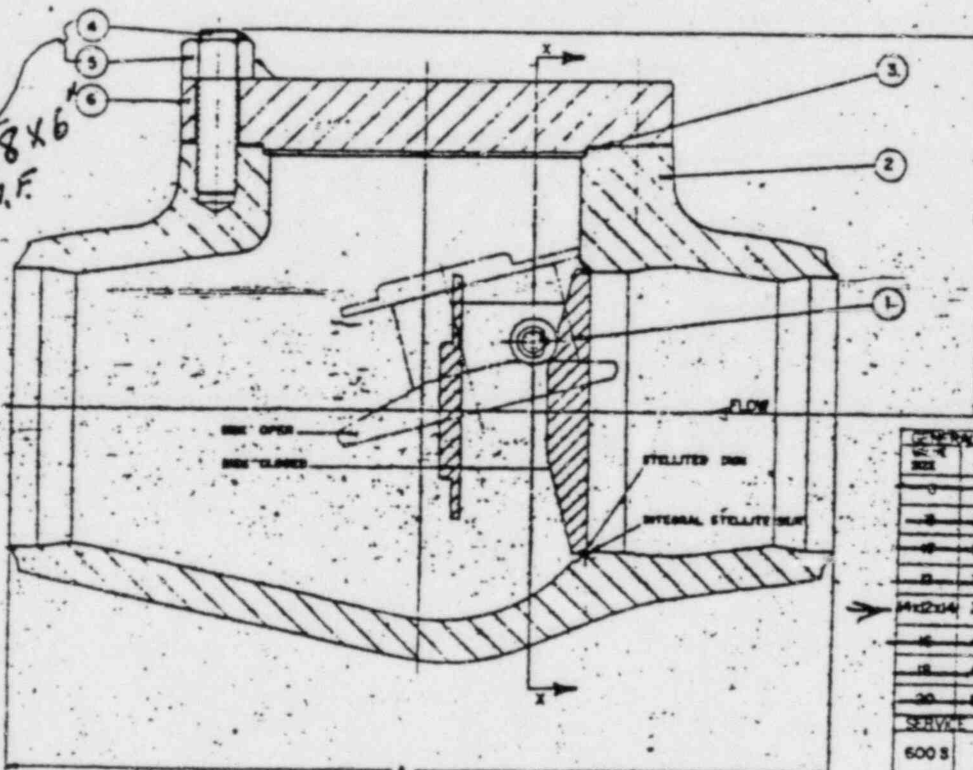
ph-136

ITEM	NAME OF PIECE	QTY	MATERIAL	SPECIFICATIONS	
1	DISC	1	FORGED ALLOY STEEL	A.S.T.M. A182, GRADE F[illegible]	[illegible]
2	BODY	1	CAST CARBON STEEL	A.S.T.M. A216, GRADE WCB	[illegible]
3	GASKET, BONNET	1	SPIRAL WOUND	[illegible]	[illegible]
4	STUD	[illegible]	CR-MOLYBDENUM STEEL	A.S.T.M. A-193 GRADE B7	[illegible]
5	NUT	[illegible]	HEAVY HEX NUT	A.S.T.M. A-194 GRADE [illegible]	[illegible]
6	COVER	1	CARBON STEEL PLATE	A.S.T.M. A-[illegible] GRADE [illegible]	[illegible]
7	SPRING, TORSION	[illegible]	[illegible]	A.S.T.M. A-[illegible]	[illegible]
8	GASKET	1	SPIRAL WOUND	[illegible]	[illegible]
9	RETAINER, HINGE PIN	2	CARBON STEEL PLATE	A.S.T.M. A-[illegible] GRADE [illegible]	[illegible]
10	PIN, ROLL	2	STEEL		[illegible]
11	BOLT	[illegible]	ALLOY STEEL	A.S.T.M. A-[illegible] GRADE [illegible]	[illegible]
12	GUIDE, [illegible]	2	STAINLESS STEEL	[illegible]	[illegible]
13	INSERT, [illegible]	2	STELLITE [illegible]	A.S.T.M. A-[illegible] GRADE [illegible]	[illegible]
14	PIN, HINGE	1	STELLITE [illegible]	A.S.T.M. A-[illegible] GRADE [illegible]	[illegible]



BOLT 7/8-9 x 3"
HEAD 1 7/16 A.F.

A	ADD W.E PER. S.W. LTR 10-29	1-27-70
REV.	DESCRIPTION	DATE



1 3/8" x 8 x 6"
[illegible] SIZE - 1 3/8"
[illegible] SIZE - 2 1/4" A.F.

GENERAL DIMENSIONS					
VALVE SIZE	A	E	BG		WEIGHT IN LBS.
~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~		~~[illegible]~~
~~8~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~		~~[illegible]~~
~~10~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~		~~[illegible]~~
~~12~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~		~~[illegible]~~
14x12x14	34 1/2	8 1/2	25		1530
~~16~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~		~~[illegible]~~
~~18~~	~~[illegible]~~	~~[illegible]~~	~~34~~		~~[illegible]~~
~~20~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~		~~[illegible]~~

SERVICE RATINGS	
600 S	CAST CARBON STEEL WCB 1440 P.S.I. AT 100°F 800 P.S.I. AT 850°F

VISUAL EXAM OF VALVE BODY, BOLTING, WELDS, DISC, SEAT, ETC.

BLUED DISC & SEAT

M : CHECK ON HINGE PIN & PIN BORE

E			
D			
C			
B			
A			
REV.	DESCRIPTION OF CHANGE	DATE	BY

ROCKWELL MANUFACTURING COMPANY
RALEIGH PLANT - RALEIGH, NORTH CAROLINA

CAST STEEL HORIZONTAL AND VERTICAL TILTING DISK CHECK VALVE GENERAL ASSEMBLY
FIG 570BY

P-419253

TAG: VCW-60A
J.O. Nº 11550. P.O Nº MY-07

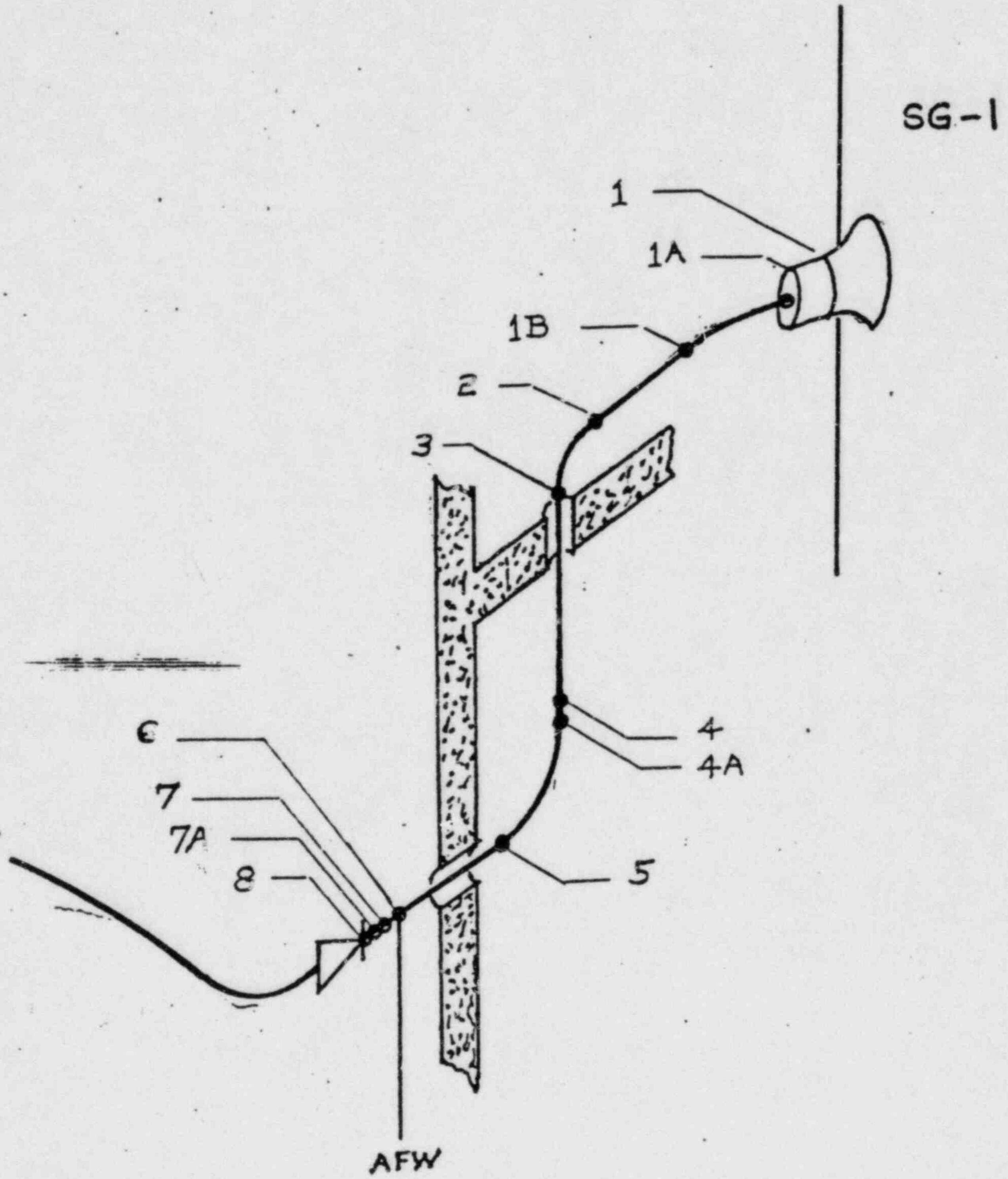
P-419253 REV. A
SHT 1 OF 2

5.7

FEBRUARY 1983

RADIOGRAPH

ID. NO.



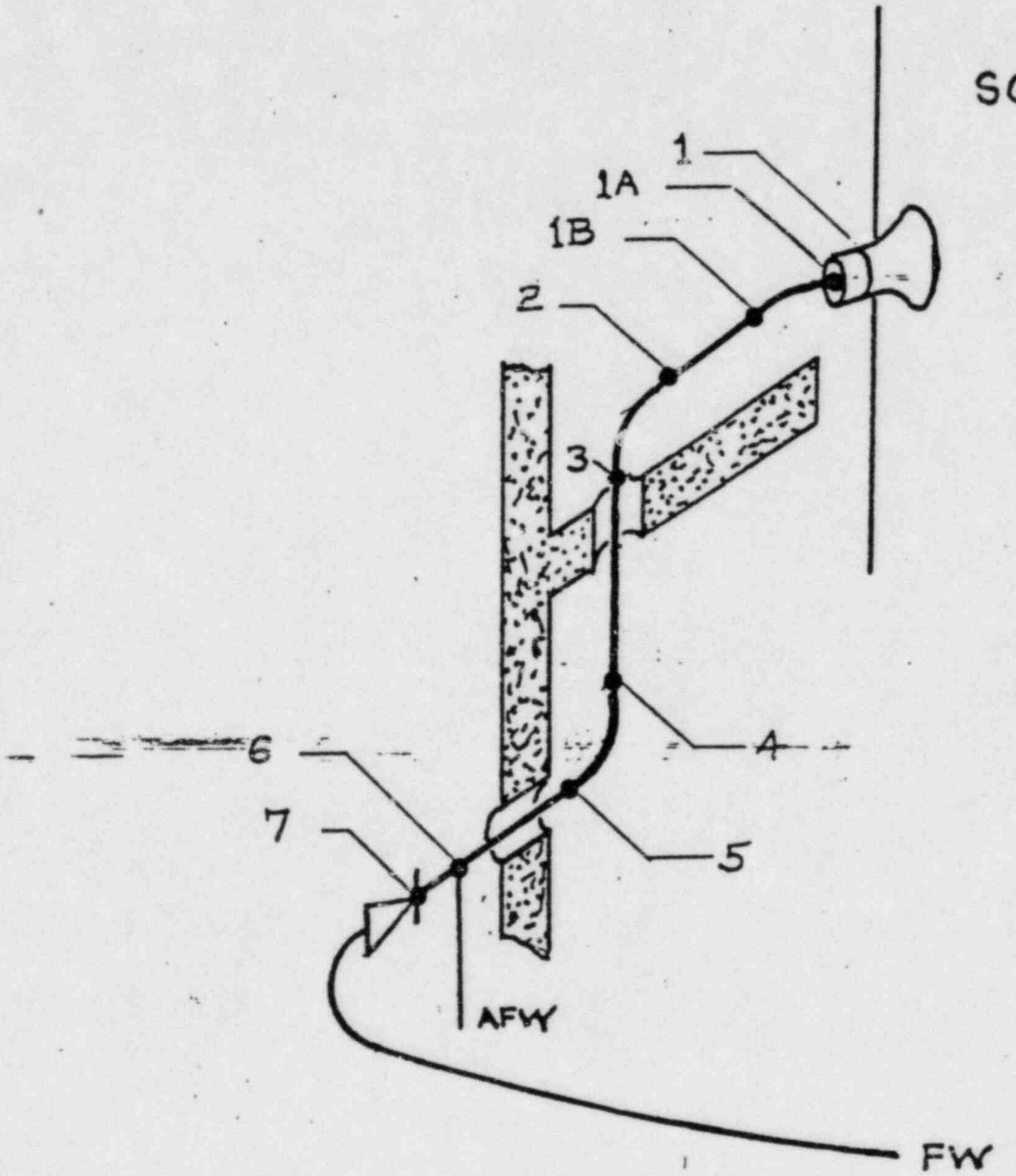
#6 MP ONLY

5.8

RADIOGRAPH

ID. NO.

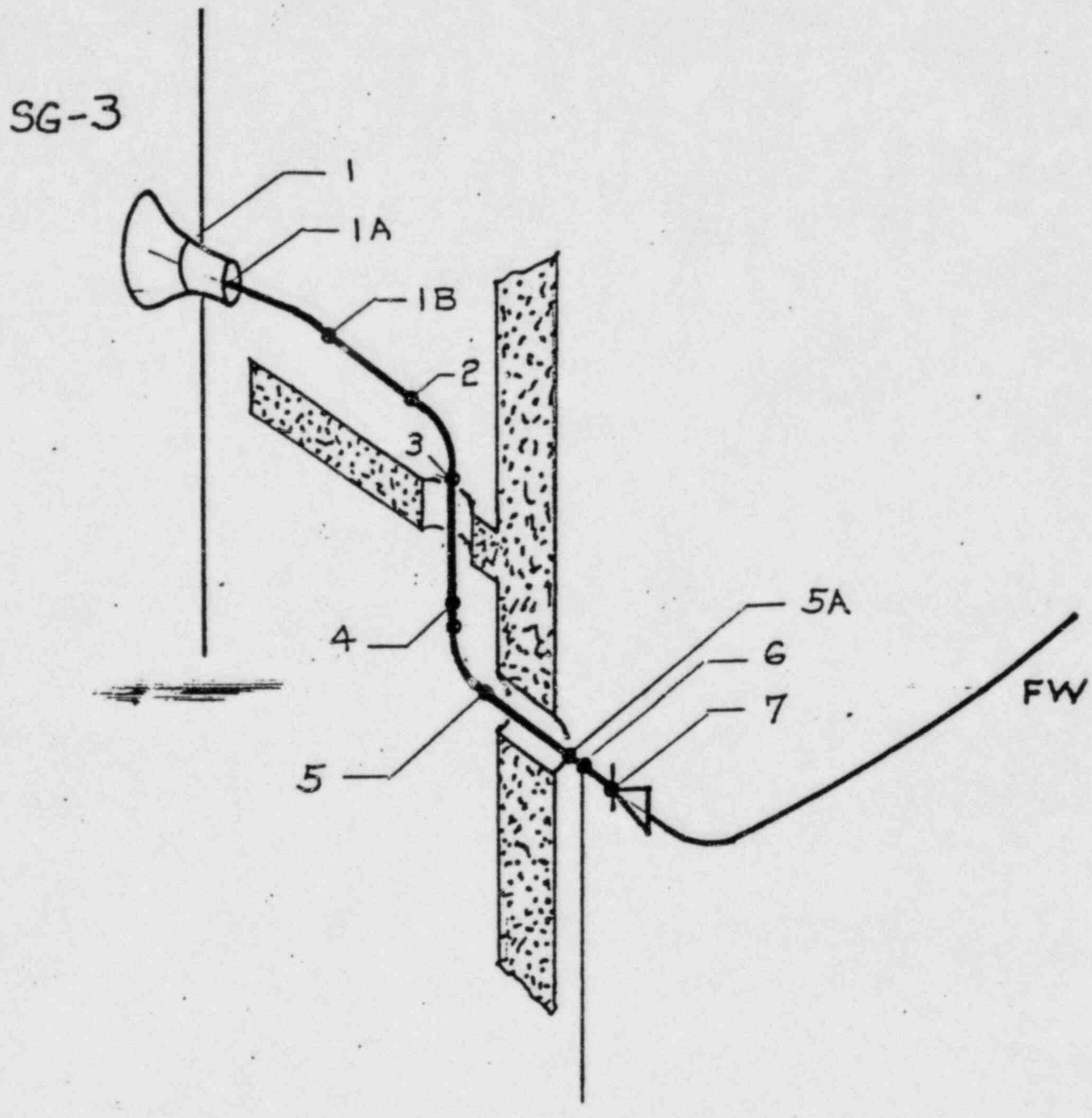
FEBRUARY 1983



#6 MP ONLY

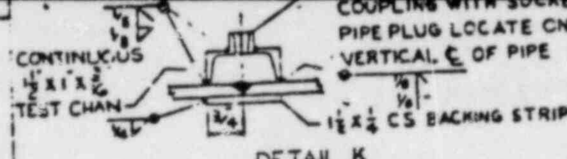
59

RADIOGRAPH
I.D. NO.
FEBRUARY 1983

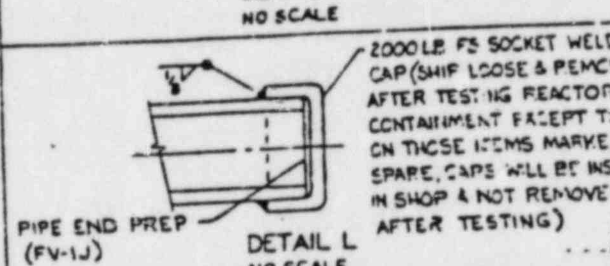


#6 MP ONLY.

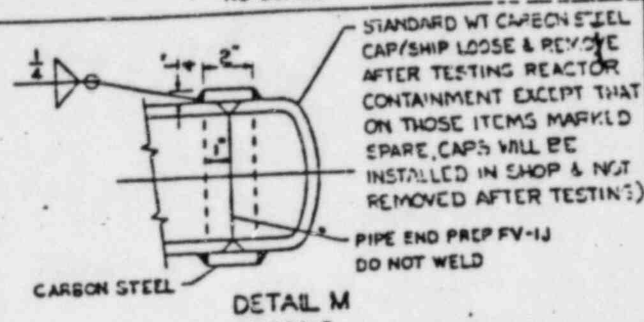
5/10



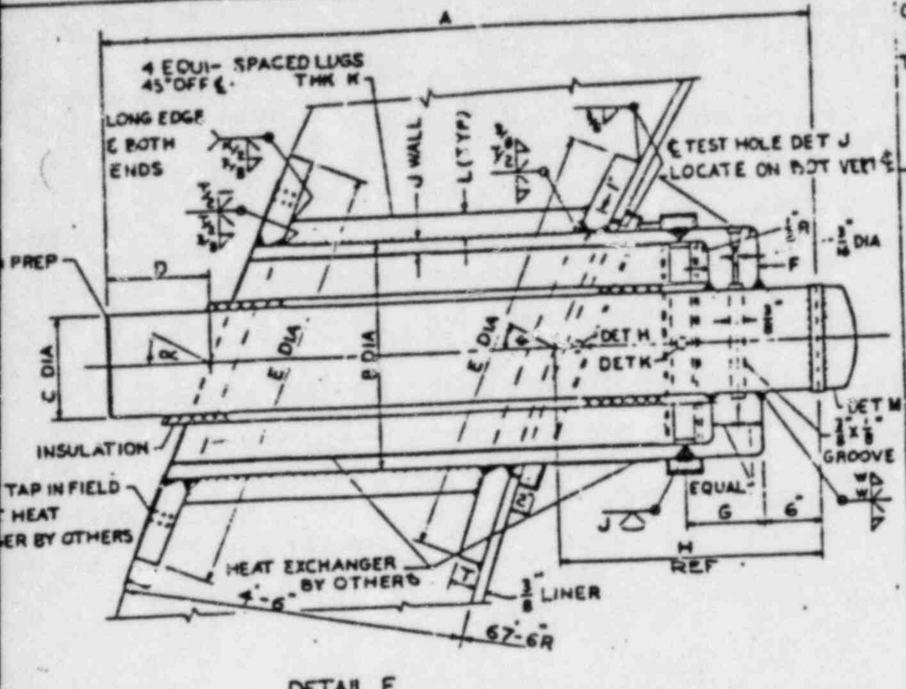
DETAIL K
NO SCALE



DETAIL L
NO SCALE



DETAIL M
NO SCALE



DETAIL E
NO SCALE

ITEM NO. FV1J	A	B	C	D	E	F	G	H	J	K	L	T	W	α	θ
66	10'-4 1/2"	3'-6"	30"	2'-5 5/8"	7'-0"	7"	8 1/2"	3'-5 5/16"	3/4"	1 1/2"	3"	2"	1 1/4"	12°1'	12°50'
53	8'-0 1/2"	2'-6"	14"	13 1/2"	5'-0"	5 1/2"	7 1/4"	2'-2 1/8"	1/2"	3/4"	2"	1"	1"	7°11'	7°40'
55	9'-2 3/16"	2'-6"	14"	2'-3 1/4"	5'-0"	5 1/2"	7 1/8"	2'-3 3/16"	1/2"	3/4"	2"	1"	1"	12°1'	12°50'
64	9'-2 1/2"	3'-6"	30"	16 7/16"	7'-0"	7"	8 1/2"	3'-3 5/16"	3/4"	1 1/2"	3"	2"	1 1/4"	7°11'	7°40'
65	8'-7 1/2"	3'-6"	30"	10 3/16"	7'-0"	7"	8 1/2"	3'-3 1/2"	3/4"	1 1/2"	3"	2"	1 1/4"	2°23'	2°33'
54	7'-4 1/2"	2'-6"	14"	9 5/16"	5'-0"	5 1/2"	7 1/2"	2'-0 1/2"	1/2"	3/4"	2"	1"	1"	2°29'	2°35'

NOTES:

MATERIAL:

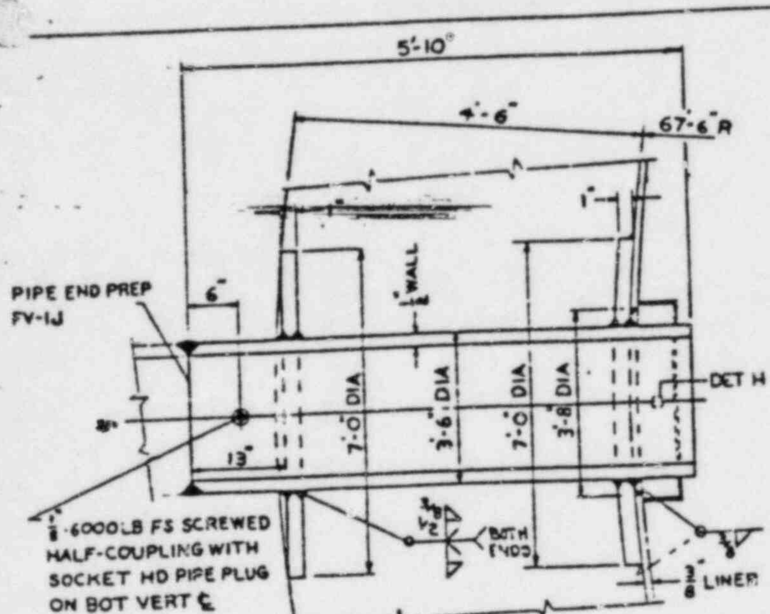
CARBON STEEL PLATE-ASTM-A516-GR60 NORMALIZED (MIN NDT -20°F)
CARBON STEEL FORGINGS ASTM-A350-GR LF 2
CARBON STEEL PIPES ✱ ASTM-A333-GR1
CARBON STEEL SLEEVES ✱ ASTM-A333-GR1
STAINLESS STEEL FORGINGS ASTM-A182-F304
STAINLESS STEEL PIPES ✱ ASTM-A312-TYPE 304 & 316 (FV-1J)
TEST ANGLES & CHANNELS ASTM-A131 GR C
ALL SLEEVES & FORGED CAPS TO BE CARBON STEEL

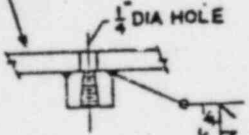
WELDING ELECTRODES:
CARBON STEEL TO CARBON STEEL ASTM-E7018
STAINLESS STEEL TO STAINLESS STEEL ASTM-E308
CARBON STEEL TO STAINLESS STEEL ASTM-E309

✱ WHERE PENETRATION SLEEVES HAVE A LONGITUDINAL WELDED SEAM, THAT PORTION OF THE SEAM EXTENDING INTO THE CONTAINMENT SHALL BE COVERED WITH A TEST CHANNEL SIMILAR TO THAT SHOWN IN DET K. WELDED SEAMS SHALL BE LOCATED ON BOTTOM VERTICAL ℄ OF THE PENETRATION.
✱ SEAMLESS PIPE SHALL BE SUPPLIED FOR ALL PENETRATIONS EXCEPT ITEM NO. 52, 64, 65, 66, 67 FOR WHICH BUTTWELDED FULLY RADIOGRAPHED PIPE MAY BE SUPPLIED.

REFERENCE DWG'S:
PENETRATION ITEM NUMBERS FV-1F
FV-1I
WELD END DETAILS FV-1J

FOR FINAL ARRANGEMENT & DETAILS SEE MFR DRAWINGS (SEE FV-1F)





DETAIL G
PURGE AIR-ITEM 67 FV-1J
NO SCALE

THE INFORMATION ON THIS DRAWING MAY NOT BE COPIED OR USED FOR OTHER THAN THE CONSTRUCTION, MAINTENANCE OR REPAIR OF THE PLANT FACILITY DESCRIBED IN THE TITLE BLOCK.

WALTER C. WOLFMAN (seal)

REACTOR CONTAINMENT-SH7
PIPING PENETRATIONS

ATOMIC POWER STATION
MAINE YANKEE ATOMIC POWER COMPANY
WISCASSET, MAINE

STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.

DRAWING NUMBER 11550-FV-1G

TABLES FOR DET C, D, E, F, & ... 2" PENET DET A, 1G TO DET B						
1	ORIGINAL ISSUE					
DESCRIPTION	CHKD	CORR	APPR	DATE	ISSUE	DESCRIPTION

5.11

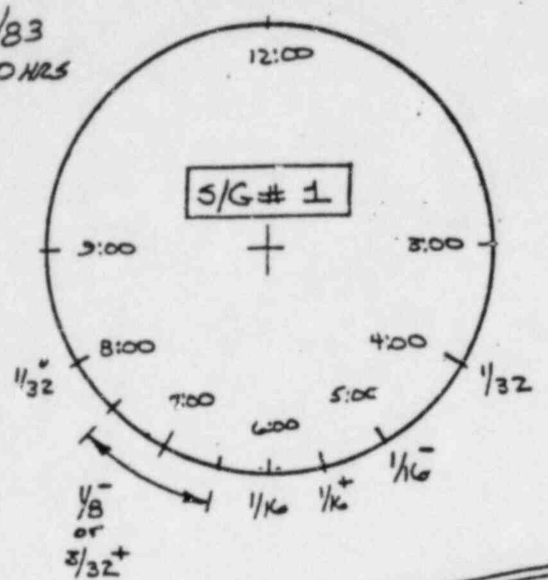
Designed by
Drawn by
Ch'd by

PIPE INSPECTION AND REPAIRS

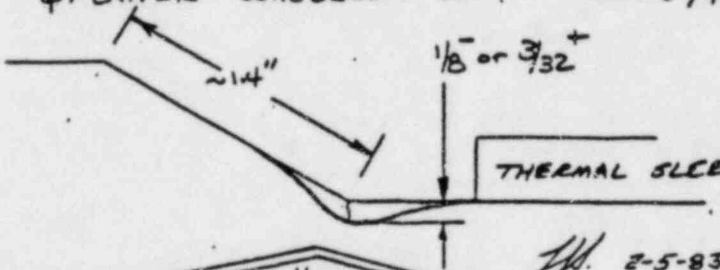
* Piping System Configuration
* Inspections
  - Selection criteria for weld joints
* Findings of Inspections
  - 79-13
  - Post event
* Repairs

STEAM GENERATOR FEED WATER NOZZLES AFTER REMOVAL OF PT INDICATIONS

2/4/83
100 HRS

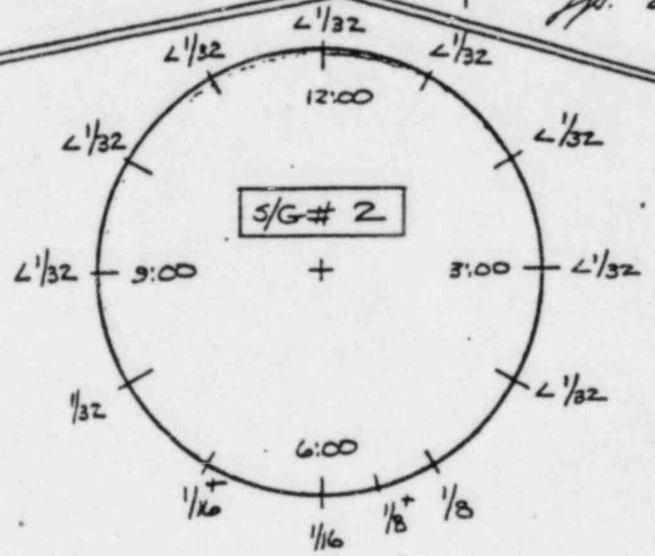


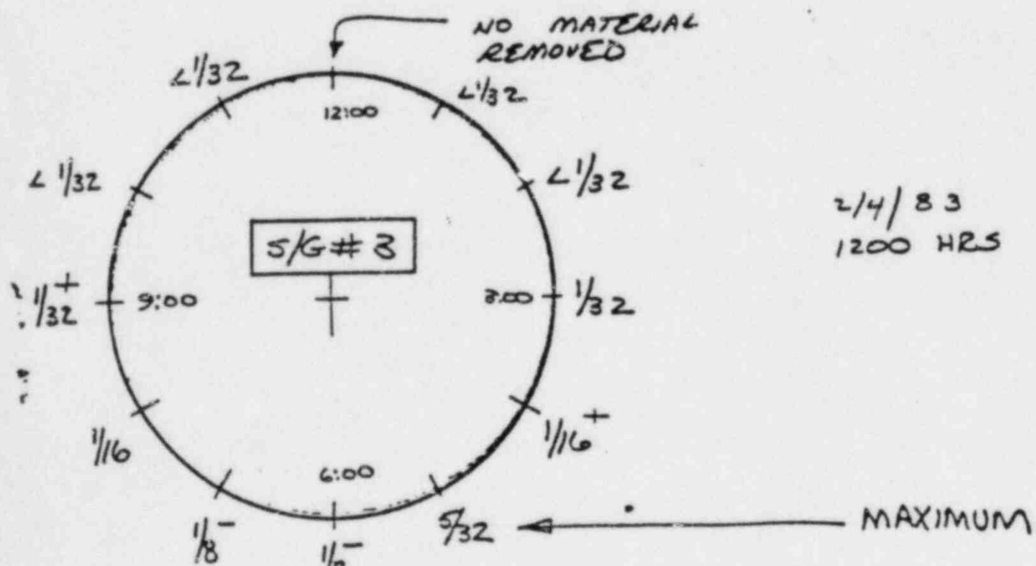
1. NO APPRECIABLE METAL REMOVAL FROM 9:00–3:00
2. POROSITY PER ATTACHED SHEET
3. APPEARS DEEPEST BETWEEN 6:30 & 7:30. GROOVE BLENDED & FLAPPER WHEELED. DEEPEST PART ≈ 1/4" RAD.



2-5-83

2/4/83
1300 HRS





2/4/83
1200 HRS

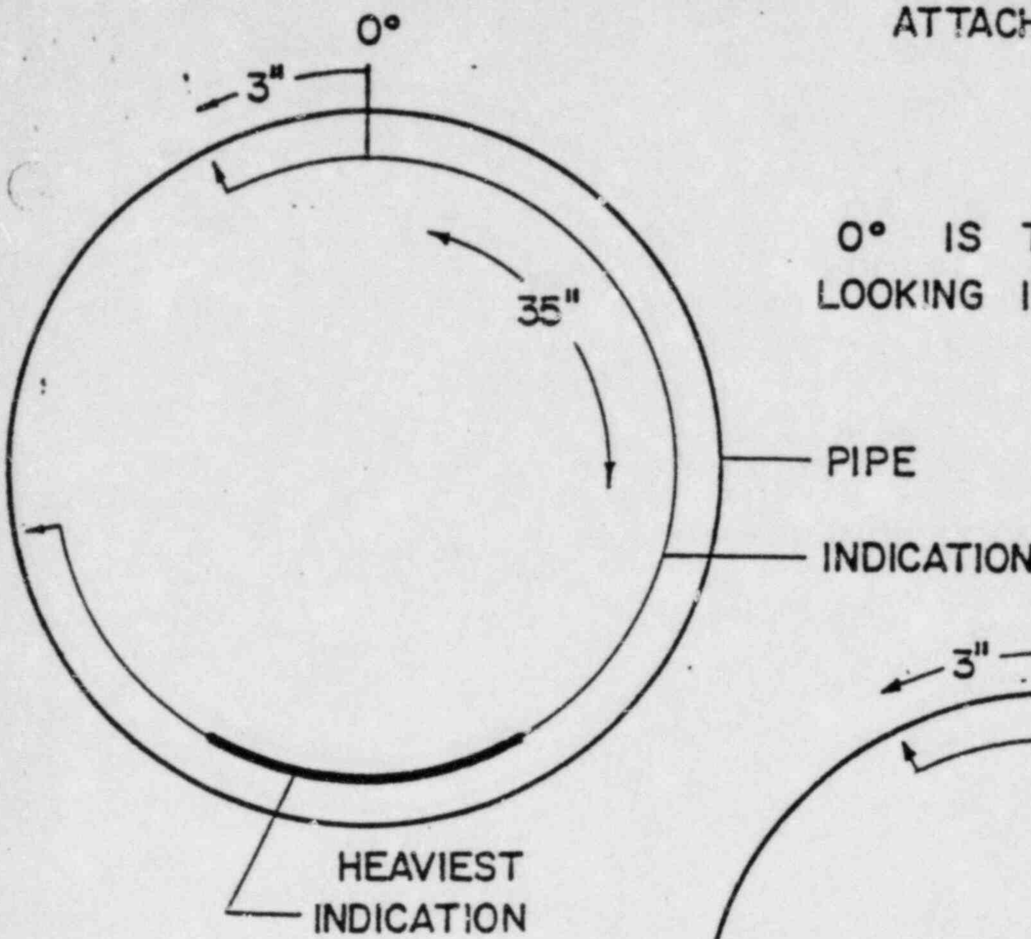
513

NOZZLE INSPECTION & REPAIRS

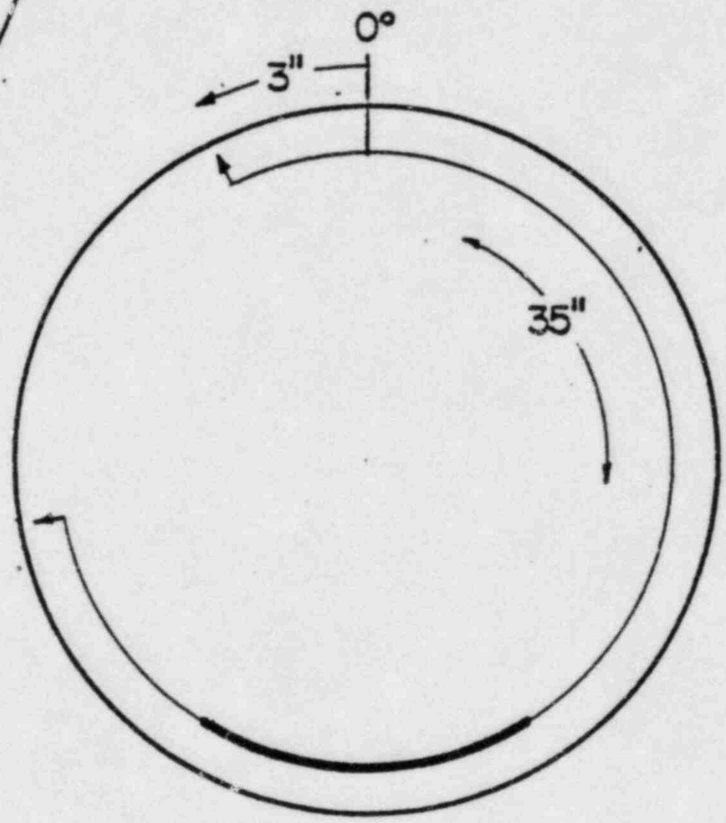
* PRE-EVENT CONFIGURATION -
    - NOZZLE DETAILS
    - WELD 1A JOINT CONFIGURATION
    - WELDING PROCESS - ORIGINAL
    - MATERIALS -
* INSPECTIONS -
    - CONSTRUCTION RT
    - 79-13 INSPECTION
    - POST EVENT INSPECTIONS
        - RT RESULTS
        - SURFACE EXAMS
* INSPECTION FINDINGS
    - SKETCHES OF CRACK EXTENT
    - AREA OF THRU CRACK
* REPAIRS
    - COUNTERBORE CONFIGURATION
    - WELDING

5.14

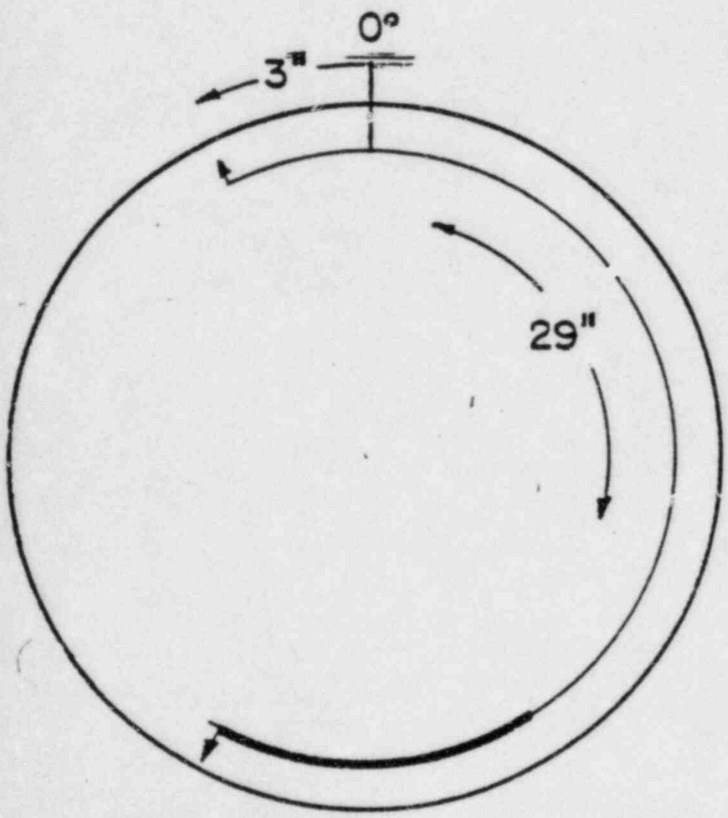
0° IS TOP DEAD CENTER LOOKING INTO STEAM GENERATOR.



#1 EL



2 EL



#3 EL

5·15

DETAILS OF STEAM

GENERATOR INSPECTIONS

AT

MAINE YANKEE

1/28/83, 1/29/83, 1/30/83 & 1/31/83

LIST OF FIGURES

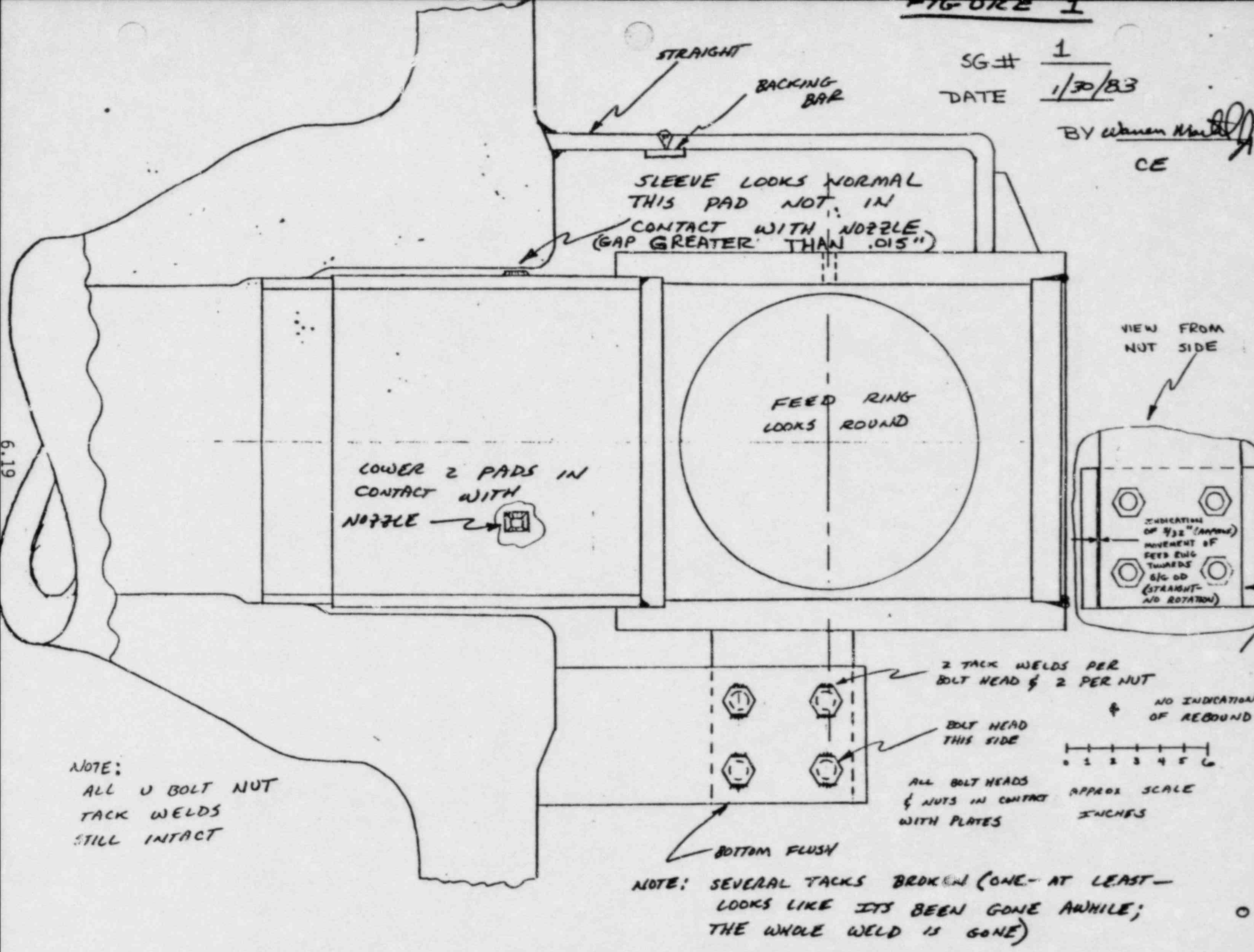


FIGURE 1
SG # 1
DATE 1/30/83
BY Warren Mart[illegible]
CE
STRAIGHT
BACKING BAR
SLEEVE LOOKS NORMAL
THIS PAD NOT IN CONTACT WITH NOZZLE (GAP GREATER THAN .015")
FEED RING LOOKS ROUND
LOWER 2 PADS IN CONTACT WITH NOZZLE
VIEW FROM NUT SIDE
INDICATION OF 1/32" (APPROX) MOVEMENT OF FEED RING TOWARDS S/G OD (STRAIGHT- NO ROTATION)
2 TACK WELDS PER BOLT HEAD & 2 PER NUT
& NO INDICATION OF REBOUND
BOLT HEAD THIS SIDE
0 1 2 3 4 5 6
APPROX SCALE INCHES
ALL BOLT HEADS & NUTS IN CONTACT WITH PLATES
BOTTOM FLUSH
NOTE: ALL U BOLT NUT TACK WELDS STILL INTACT
NOTE: SEVERAL TACKS BROKEN (ONE- AT LEAST- LOOKS LIKE ITS BEEN GONE AWHILE; THE WHOLE WELD IS GONE)

FIGURE 2

SG# 2

DATE 1/28/83

BY Warren [signature] CE

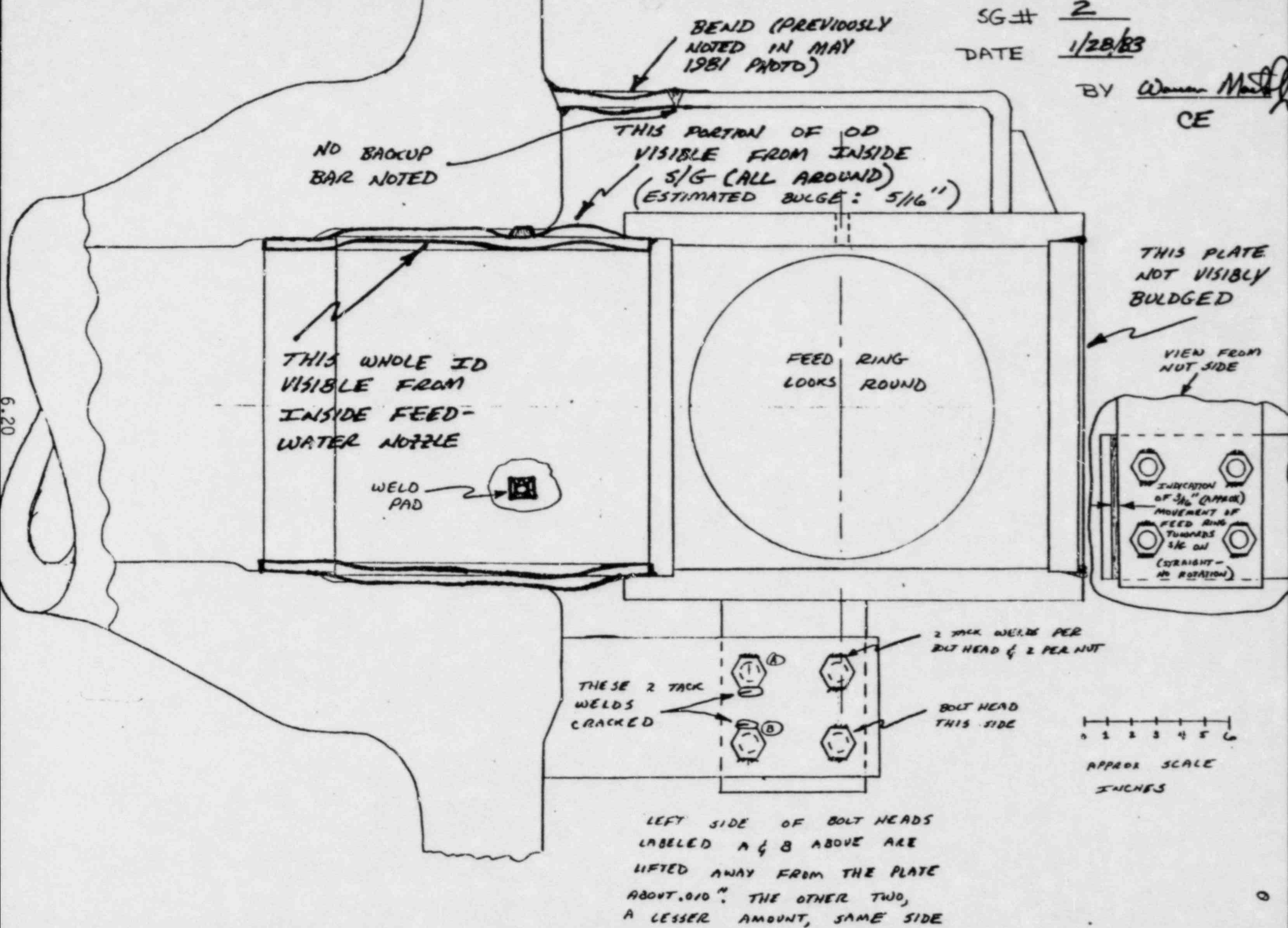


FIGURE 3

SG# 3

DATE 1/28/83

BY

CE

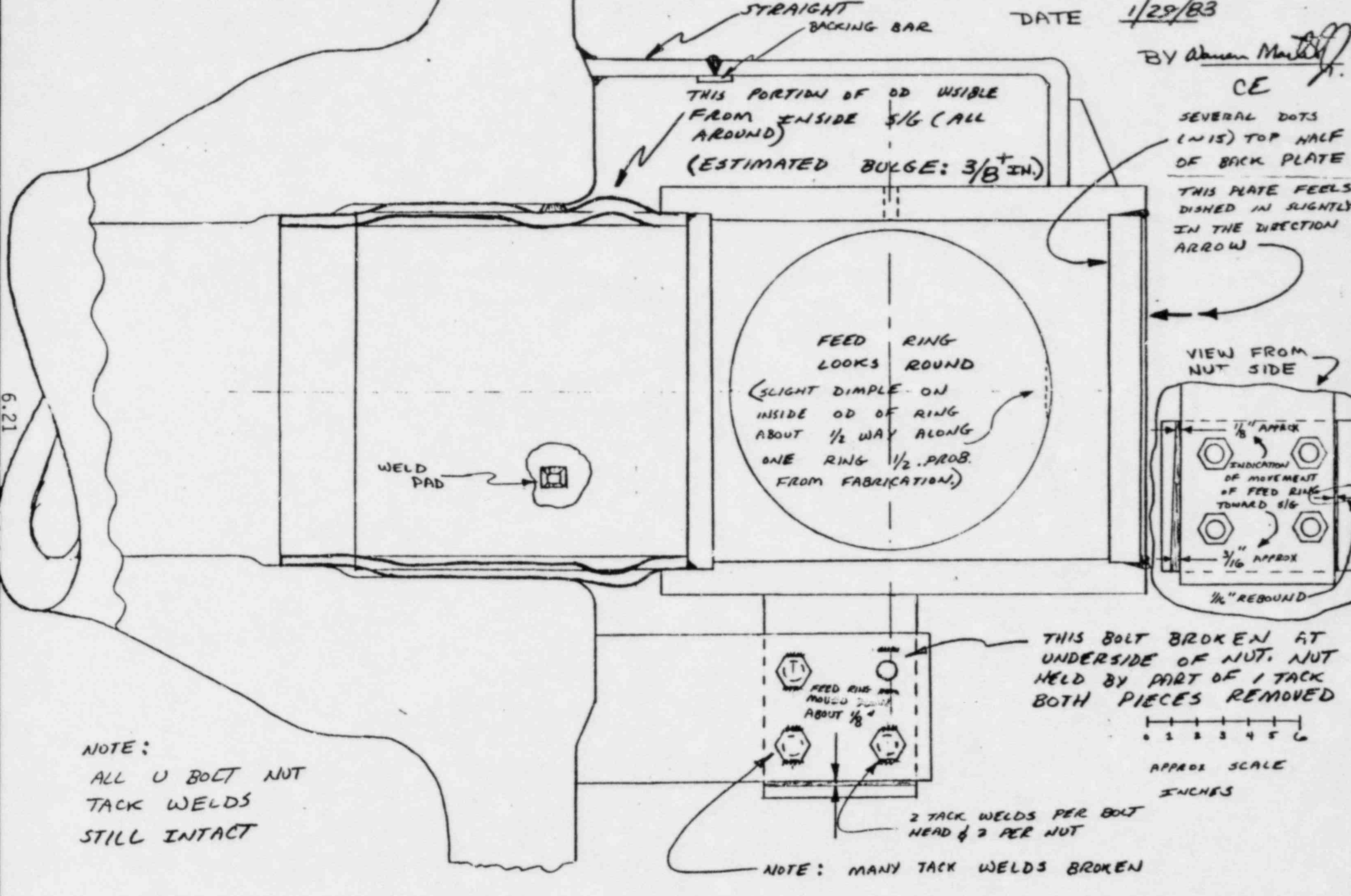
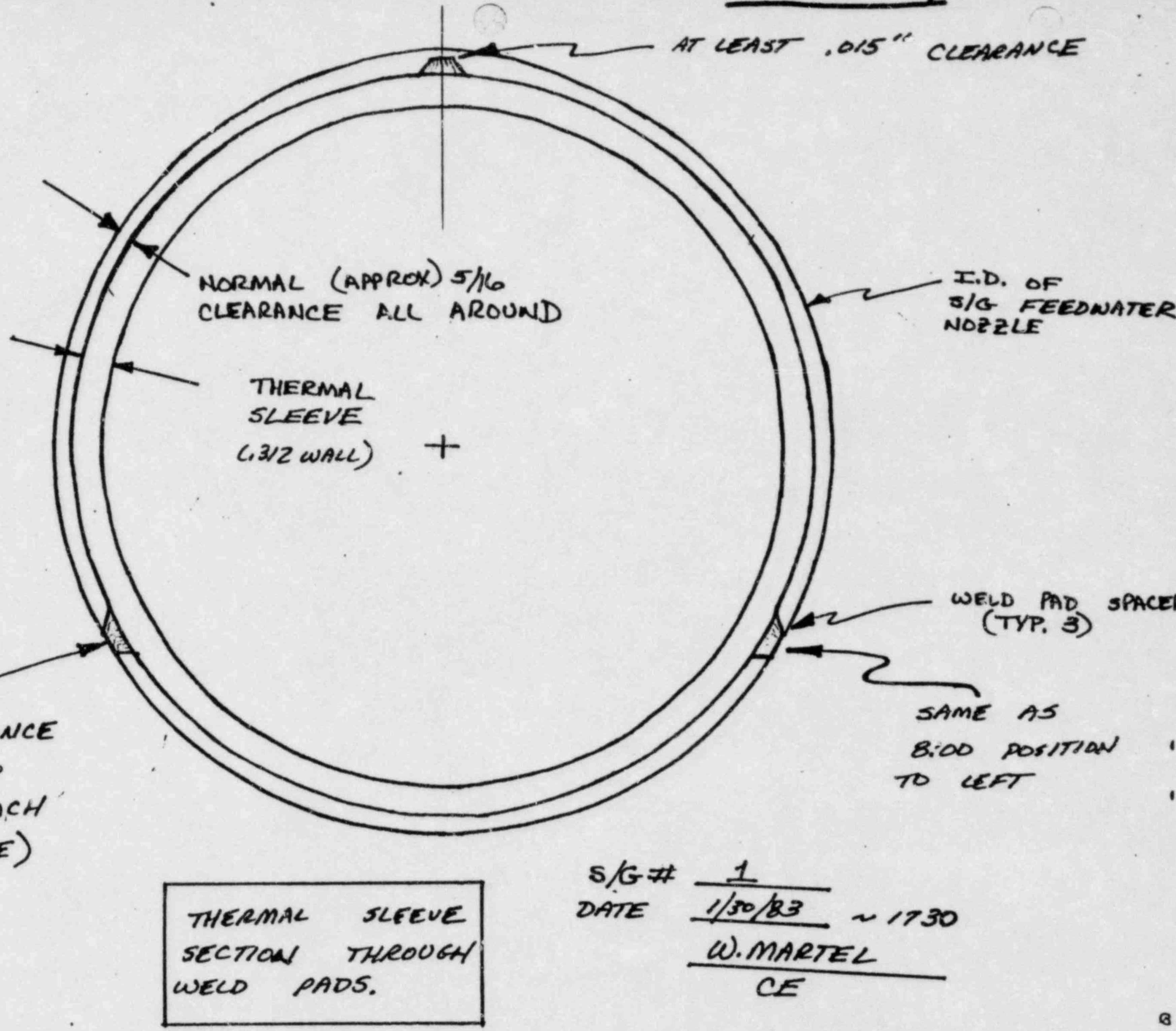


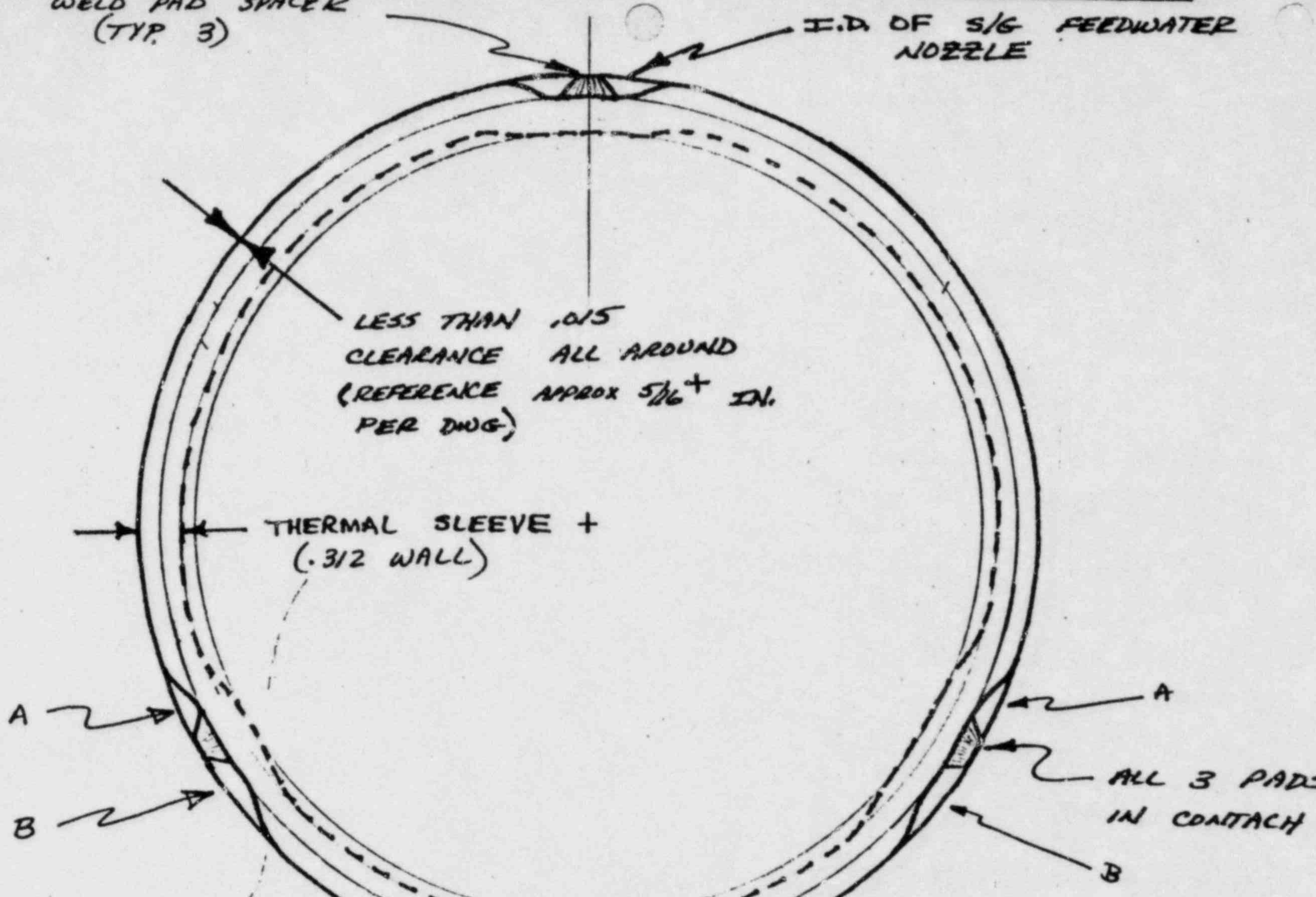
FIGURE 4



THERMAL SLEEVE SECTION THROUGH WELD PADS.

S/G # 1
DATE 1/30/83 ~ 1730
W. MARTEL
CE

FIGURE 5

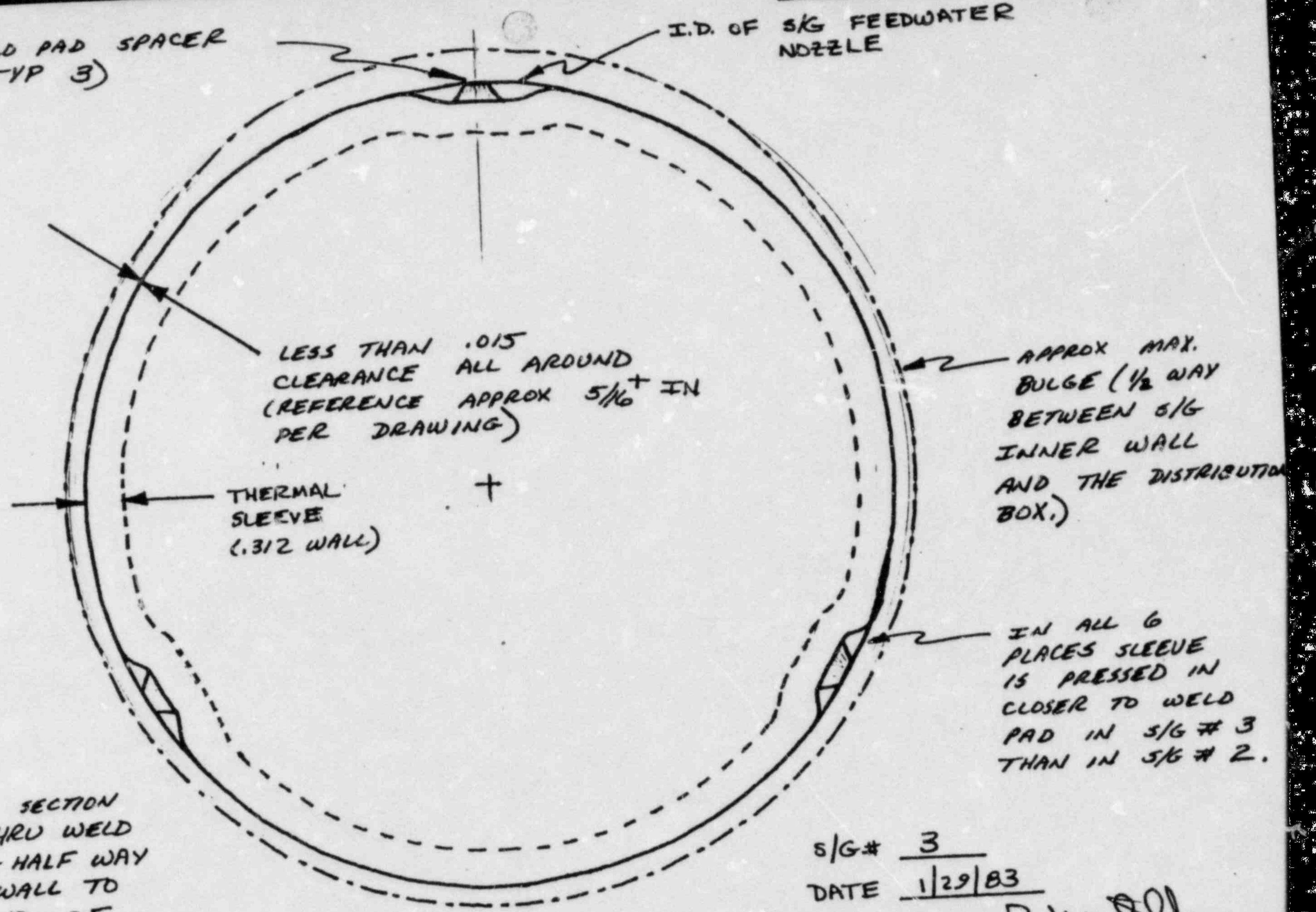


NOTE: In both cases, the sleeve is pressed out against the nozzle closer to the weld pad on side A than side B.

S/G# 2
DATE 1/28/83 (~ 1450)
BY W. MARTEL
CE

THERMAL SLEEVE SECTION THROUGH WELD PADS

FIGURE 6



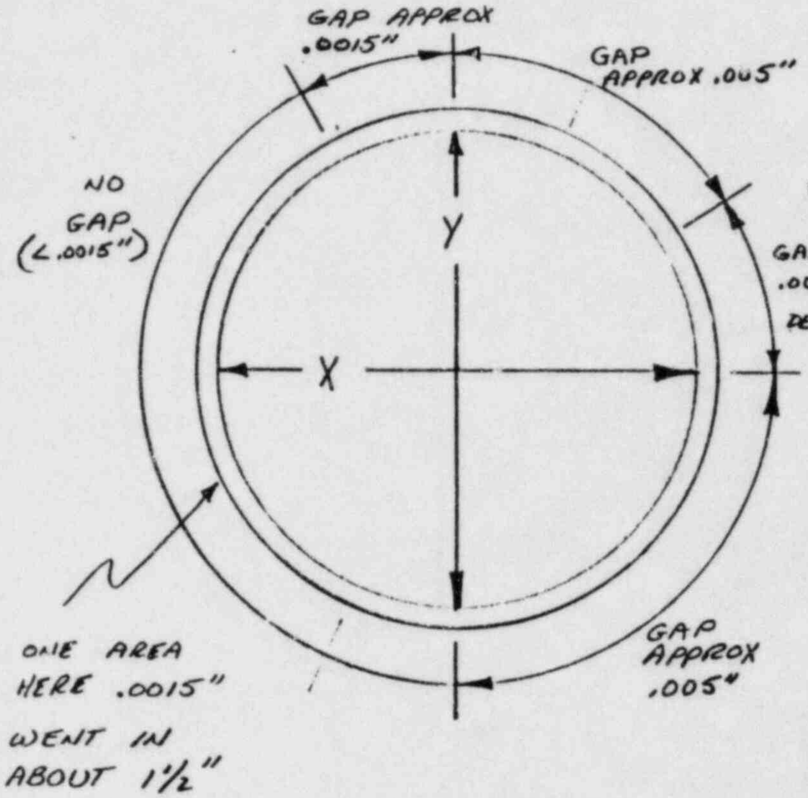
NOTE- THIS SECTION TAKEN THRU WELD PADS. ONE-HALF WAY FROM S/G WALL TO INSIDE END OF SLEEVE, THE BULGE WAS GREATER THAN THE (PROJECTED) ID OF THE FEEDWATER NOZZLE. SEE ELEVATION VIEW & OUTER DASHED (—·—·—) LINE ON THIS SKETCH.

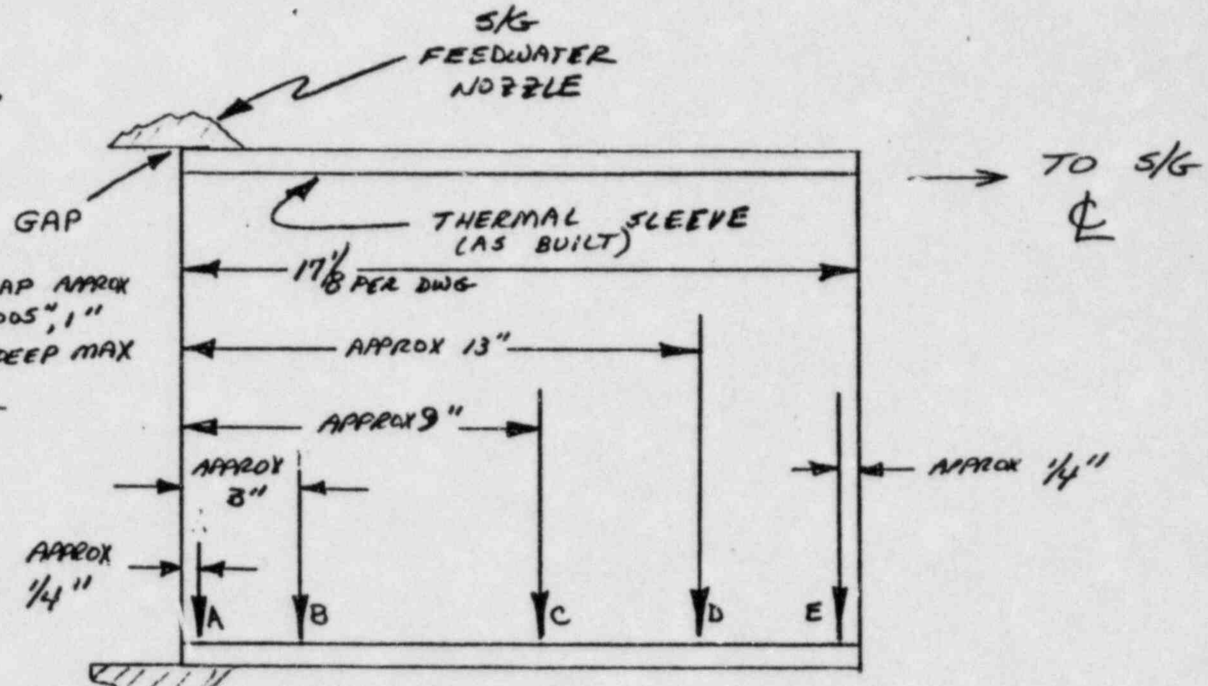
S/G# 3
DATE 1/29/83
BY Warren R. Martin [signature]
CE

THERMAL SLEEVE SECTION THROUGH WELD PADS

FIGURE 7

S/G # 1
DATE 1/31/82
BY W. Mantel Jr.
CE





MEASUREMENT	APPROX LOCATION				
	A	B	C	D	E
X (HORIZONTAL)	13.181	13.169	13.194	13.144	13.128
Y (VERTICAL)	13.120	13.133	13.138	13.151	13.191

NOTE FOR GAPS ABOVE: USED .0015, .005 & .015" FEELER GAGES. DUE TO DIFFICULT CONDITIONS GAP MEASUREMENTS ARE APPROX.

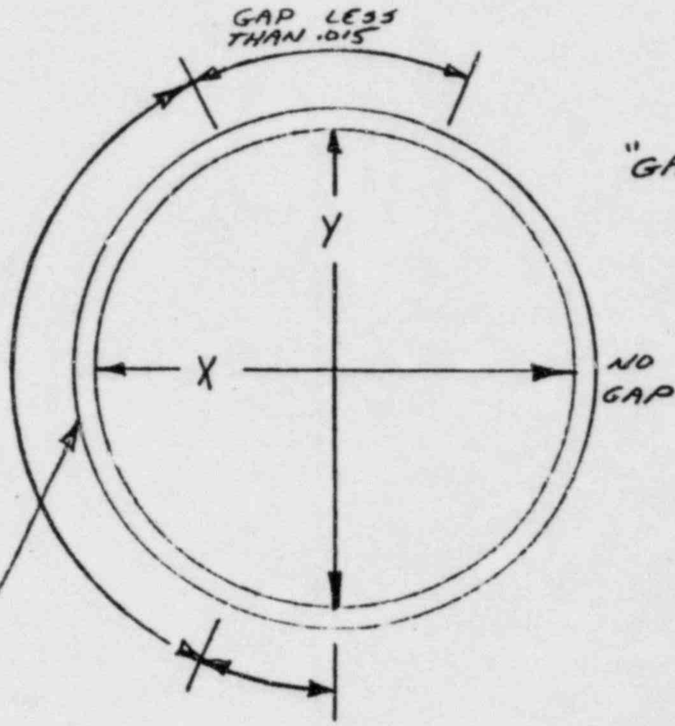
(PHOTOS TAKEN: 35 MM)

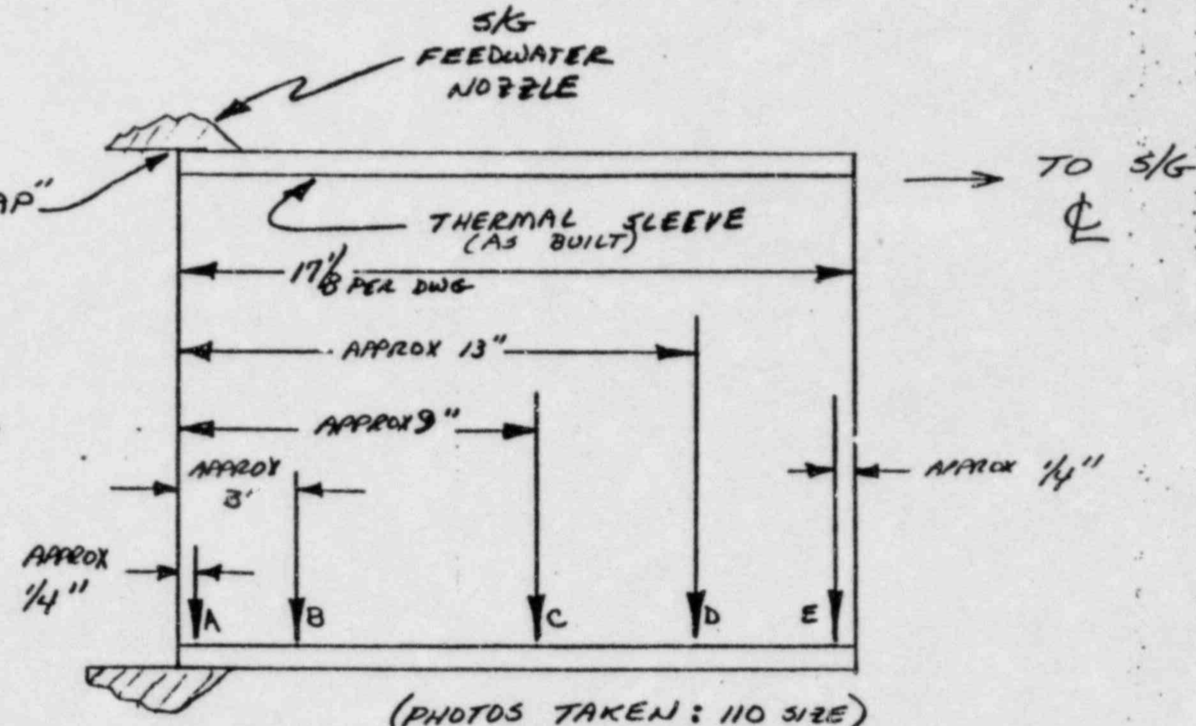
INSIDE MIC. # 61-038

6.25

FIGURE 8

S/G # 2
DATE 1/30/83
BY Warren R. Martel Jr.
CE





(PHOTOS TAKEN: 110 SIZE)

MEASUREMENT*	APPROX LOCATION				
	A	B	C	D	E
X (HORIZONTAL)	13.143	13.769	13.596	13.638	13.030
Y (VERTICAL)	13.136	13.717	13.727	13.778	13.330

NOTE:
THE MOUNDS CAUSED BY EACH WELD PAD ARE CLEARLY VISIBLE ON SLEEVE ID.

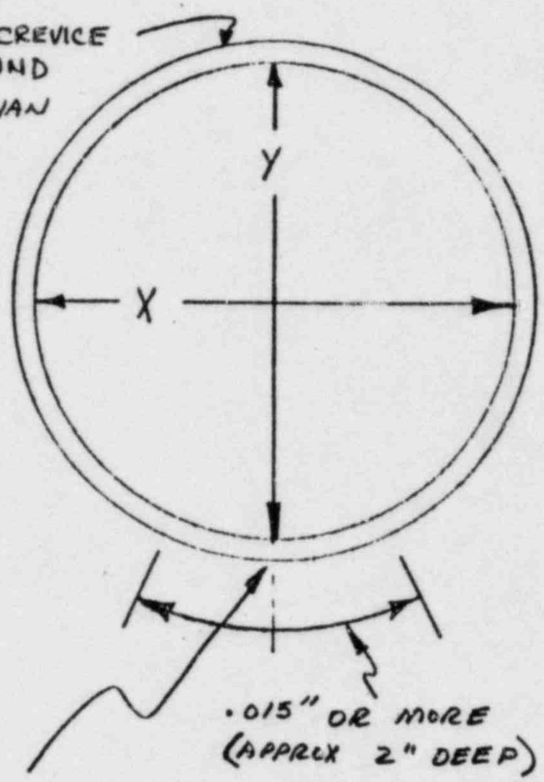
* ESTIMATED ACCURACY ±.005" DUE TO SURFACE CONDITION

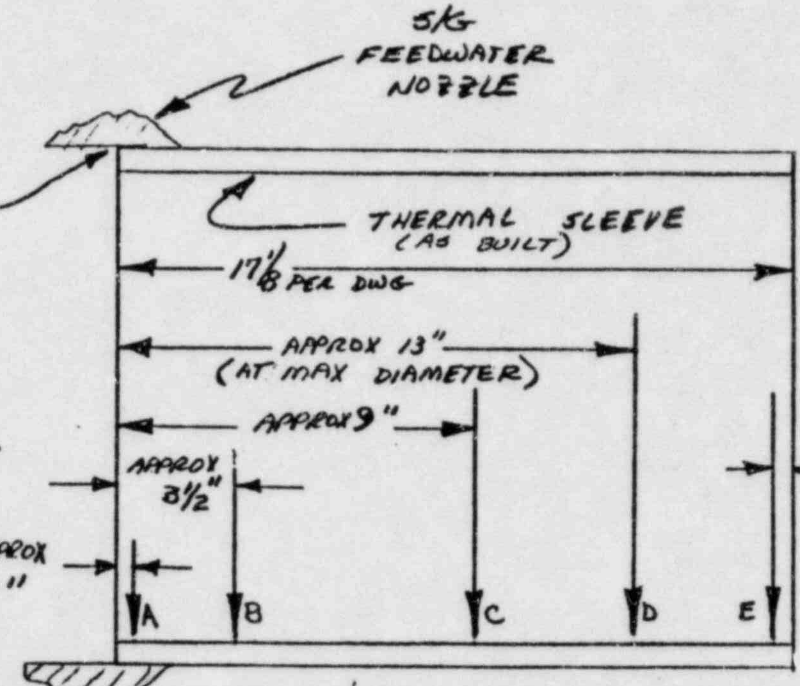
INSIDE MIC. # 61-038

FIGURE 9

S/G # 3
DATE 1/31/83
BY Warren R. Martin Jr.
CE

NOTE: MOST OF THE OUTWARD BULGE MORE THAN THE (PROJECTED) I.D. OF THE NOZZLE (I.E. AT LOCATION D) IS AT THE TOP





NOTE:
THE MOUNDS CAUSED BY WELD PADS ARE CLEARLY VISIBLE ON SLEEVE I.D.

(PHOTOS TAKEN: 35 MM)

MEASUREMENT	APPROX LOCATION				
	A	B	C	D	E
X (HORIZONTAL)	13.136	13761	13792	13.963	12.862
Y (VERTICAL)	13.120	13739	* 13716	** 14.275	13.492

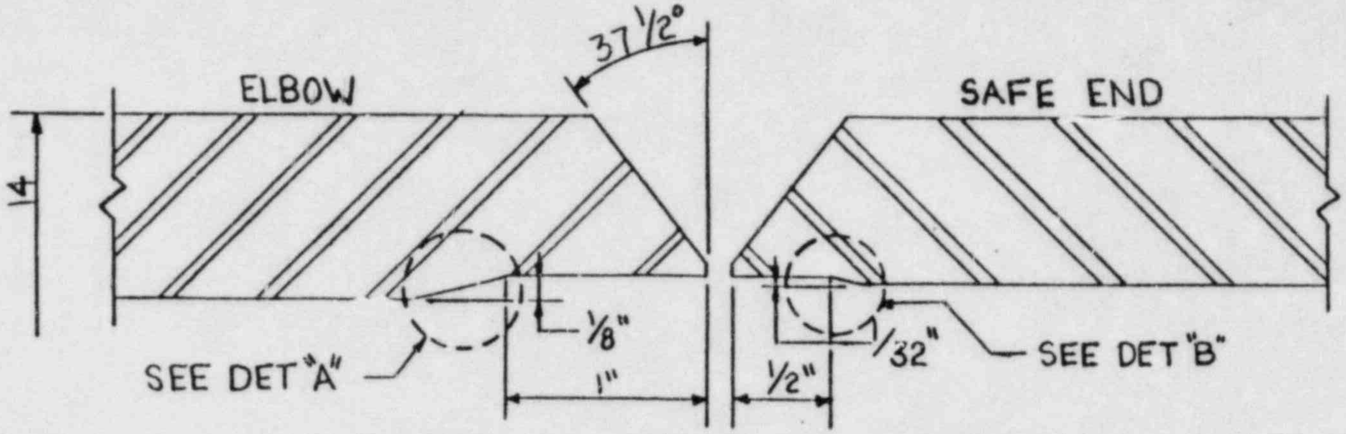
* This dimension taken from 11:00 to 5:00 vs. exactly vertical due to the weld pad bulge at the 12:00 position.

** This dimension approx. (and minimum). Was beyond capacity of I.D. mic. Est. range 14.275 $^{+.050"}_{-.000"}$

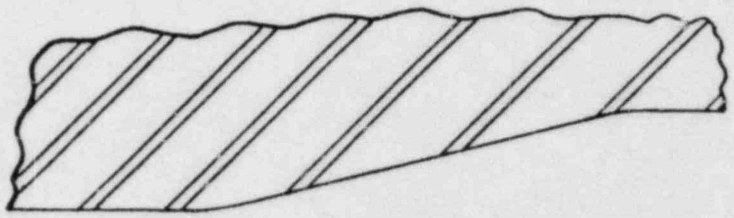
INSIDE MIC. # 61-038

7) <u>MODIFICATIONS</u>

- BLENDING TO REMOVE STRESS RISERS
- J VENTS
- OPERATIONAL GUIDANCE



ALL DIMENSIONS ARE APPROXIMATE



DETAIL "A"
NOT TO SCALE

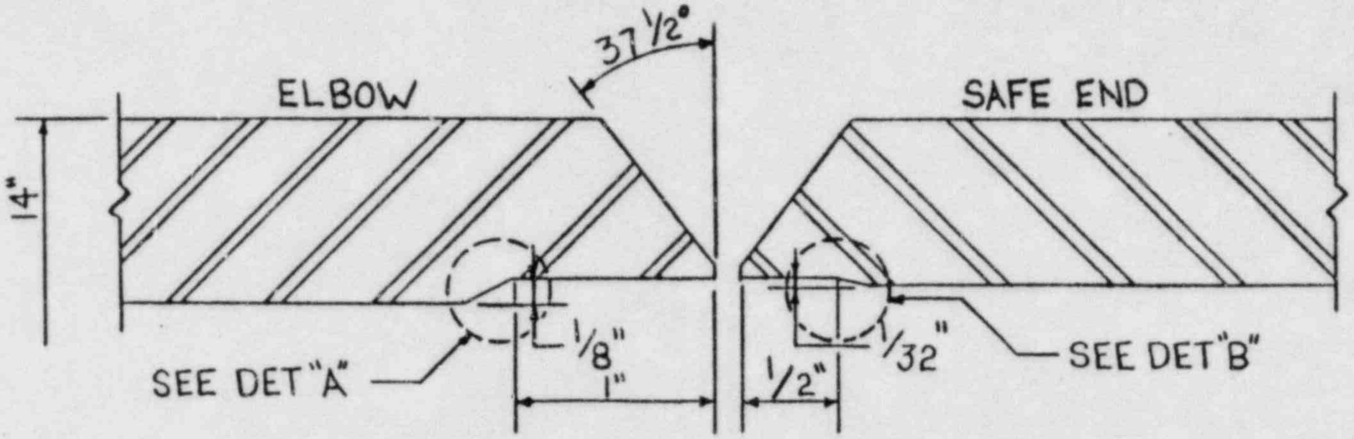
DETAIL "B"
NOT TO SCALE

NOTE

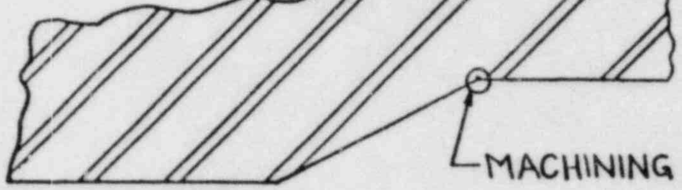
ALL COUNTERBORES TO BE GROUND OUT SMOOTH WITH A MINIMUM TAPER OF 4 TO 1.

SKETCH # 2

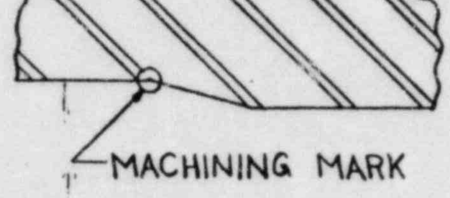
AS-LEFT CONDITIO



ALL DIMENSIONS ARE APPROXIMATE



DETAIL "A"
NOT TO SCALE

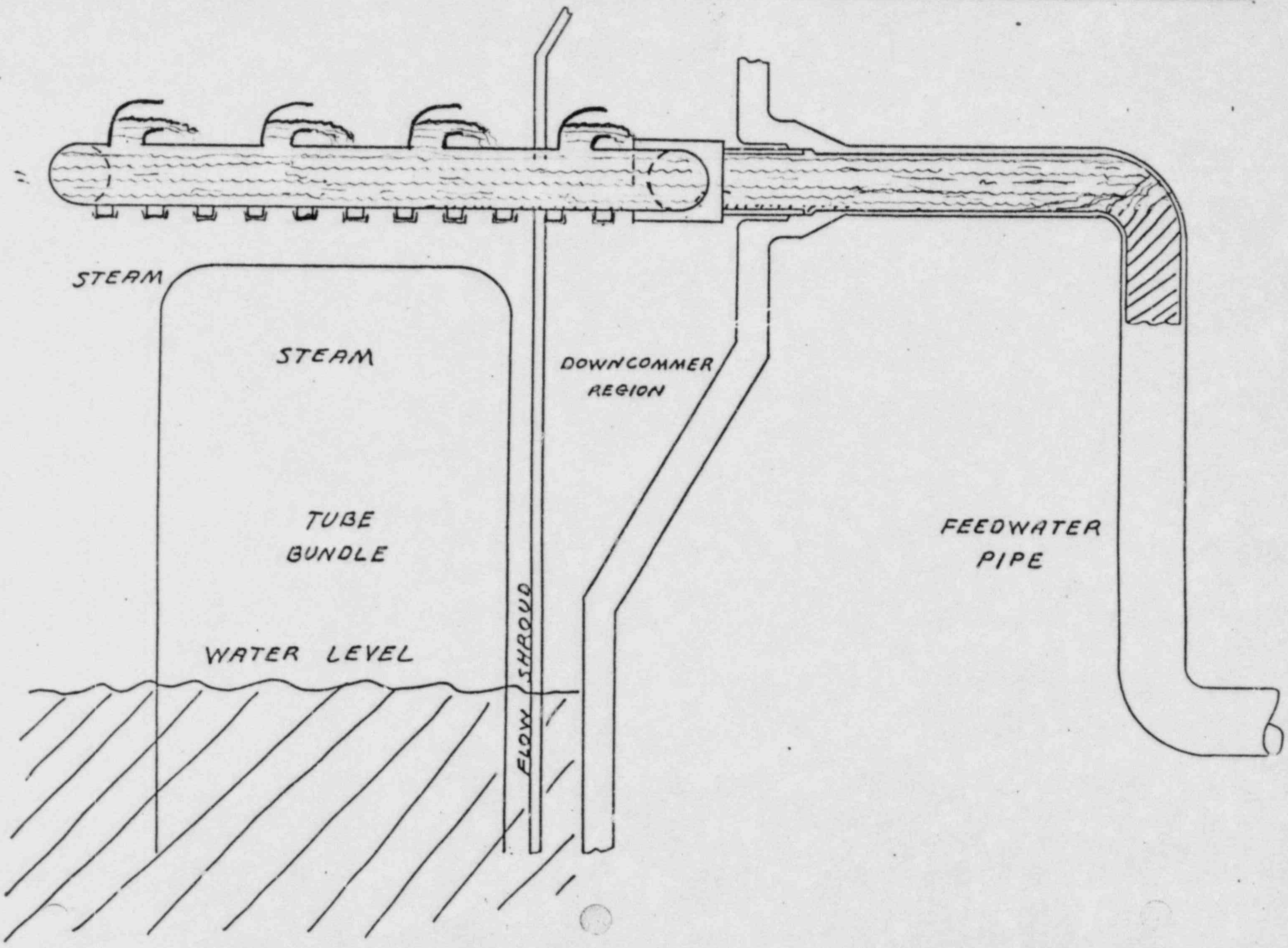


DETAIL "B"
NOT TO SCALE

SKETCH # 1

AS-FOUND CONDITION

SCHEMATIC - MAINE YANKE STEAM GENERATOR
REFILL - COLD AUXILIARY FEEDWATER FLOW - WITH J-TUBES



FEEDWATER SYSTEM OPERATION GUIDANCE

DISCUSSION:

Steam generator feedwater rings and adjacent piping have experienced damage as a result of thermal cycling, non uniform thermally induced stresses, and water hamering. The most adverse conditions often exists during post trip recovery but some thermal cycling may occur during normal hot shutdown or standby conditions.

Post Trip Recovery

Following a trip from power, steam generator water level drops and feedwater flow is reduced to about 5% by main and by-pass feedwater regulator valve operation. Feedwater temperture drops as the main feedwater pumps refill the generator with the residual heated feedwater in the system. The tempeature cycle is comparatively slow and uniform within the feedwater ring and adjacent pipe. Auxiliary feed, if automatically initiated during the event, will inject colder water into the feed flow making the thermal transient somewhat more severe. Appropriate immediate operator action would be to assure main feed is operating and securing the auxiliary feed. While SG level is being restored, auxiliary feed should be lined up through the #1 feedwater heaters and auxiliary steam to the heater placed in operation. After the SG level has been restored to above the feed ring (40%), heated auxiliary feed flow should be started and the main feed pumps shut down.

If main feed is not operating following a trip from power, the auxiliary feed flow should be maintained to keep the feed pipe and ring full. While the thermal cycle is greater, maintaining flow reduces the probability of more damaging non-uniform thermal stress and steam-water hammering. While SG level is being restored, auxiliary feed should be lined up through the #1 feedwater heaters and auxiliary steam to the heater placed in operation. After the SG level has been restored above the feed ring (40%), auxiliary feed flow may be directed thru the #1 heater.

If all feedwater flow is lost for more than several minutes, the ring and acjacent pipe may drain via the thermal sleeve clearance and fill with 520° steam. Subsequent initiation of cold feedwater could chill the bottom of the feed pipe causing non-uniform thermal stress. In addition, the steam void within the ring and pipe could collapse suddenly causing steam-water hammer shock.

Since the emergency auxiliary feed system is powered from the safeguards bus, loss of all feedwater flow could occur if a loss of both on-site and off-site power occurred which is an event of extremely low probability. However, appropriate operator action under such circumstances is to restore feed flow slowly ramping to 200 gpm per steam generator over several minutes using the turbine drive aux. feed pump or one of the motor driven pumps when power is restored. Again, routing auxiliary feed flow through #1 feedwater heter is a preferred path when time permits.

If a total loss of condensate pump flow occurs, some of the entrained feedwater in the system will flash to steam. Injecting coldwater into the system could cause the steam water. Do not reroute aux flow through the #1 heater until one hour after stable secondary plant conditions have been re-established. Auxiliary feed pressure should be applied to the system gradually to slowly collapse any steam voids that may be present.

<u>Normal Power Operation</u>

During normal power operation with less than two motor driven main feed pumps in operation, one MDFP should be maintained in the auto-start mode. This will ensure that the residual heated water in the feed train is used to the best advantage. To ensure that aux. feed supply is as warm as practical, maintain Demineralized Water Tank temperature between 80°F and 100°F.

Hot Standby or Hot Shutdown Condition Guidance

During these periods feedwater requirements are low. If the aux. feed system is taking water from the DWST thermal cycling ranging betwen 90°F and 520°F may occur in the S.G. nozle, feed ring and adjacent pipe when flow is intermittently stopped then restored. The magnitude of cycling can be reduced by maintaining some minimum flow when possible and by routing flow through the #1 FW heaters (with aux. steam supplied).

GUIDE OUTLINE

Post Trip

A. Condition - Reactor Trip from Power
Main Feed Flow System Operating
Aux. Feed Flow System Starts Auto

Action - 1. Trip off both Aux. Feed Pumps
2. Restore normal level via Bypass Regulating Valves
3. Transfer to Aux. Feed via #1 Heater when time permits

B. Condition - Reactor Trip from Power
Main Feed Flow System not Operating
Aux. Feed Flow System Starts Auto

Action - 1. Trim to one Aux. Feed Pump
2. Control Flow at 200 to 300 gpm per SG until level exceeds 40%
3. If Condensate System is operating, transfer aux. feed thru #1 Heater when time permits

C. Condition - Reactor Trip from Power
Loss of On-Site and Off-Site Power

Action - 1. Switch Aux. Feed to Manual; Ramp controllers closed
2. Start Turbine Driven Aux. Feed Pump (or Motor Driven Aux. Feed Pump is power is Restored)
3. Increase Aux. Feed Flow gradually over 5 minute period to between 200 and 300 gpm per steam generator
4. When SG Level exceeds 40%, adjust Aux. Feed Flow to maintain same constant flow to each SG

Normal Operation

A. Condition - Plant power level in excess of 2%

Action - 1. Maintain one MDFP in operating or auto start mode
2. Maintain DWST between 80°F and 100°F
3. Maintain aux. feed routing through #1 FW heater and aux. steam supply to #1 FW heater in a standby operable status
4. Maintain aux. feed in emergency auto start direct feed mode

Hot Standby Operations

A. Condition - RCS above 400°F
Power level less than 2%

Action - 1. Maintain some aux. feed flow to each SG is possible
2. Route aux. feed flow through #1 FW with steam controller set at about 10 psi.

8) RESPONSES TO NRC STAFF QUESTIONS

QUESTIONS FOR MAINE YANKEE
RE: FEEDLINE CRACK EVENT OF 1/25/83

1. Q. Have there been any other water hammer events since June 1979 in Main or Aux Feedwater system?
   A. Not to our knowledge.

2. Q. Any evidence of failure of feedwater check valves to perform their function?
   A. No.

3. Q. Was there anything unusual about plant operating conditions before the trip?
   A. Yes. Both motor-driven main feedwater pumps were out of service, so normal feedwater flow was stopped at the time of the trip.

4. Q. Is there relevant data in the following parameters:

a)	Steam Generator Water Level	A:	See attached level traces.
b)	Steam Generator Pressure	A:	See tabulations.
c)	Containment Temperature	A:	See tabulations.
d)	Containment Pressure	A:	See tabulations.
e)	Containment Humidity	A:	See tabulation (dew pt.)
f)	Containment Sump Water Level	A:	See tabulations.
g)	RCS Parameters	A:	Normal post-trip. See tabulations.

5. Q. Describe operation of aux. feed system during event. What are temperatures, flowrates? How long was aux. feed operated, and what manual operations were taken after initiation?

   A. Immediately after the trip SG levels dropped to below the aux. feed pump start initiation level of 35% and both P25-A and P25-C started (see attachment Q5). The initial flow rate was 400 gpm per SG which was soon cut back by operator action to 200 gpm/SG. The pumps draw water from the Demineralized Water Storage tank which was at 60°F.

   Seven minutes after the trip P25-C was shut off, discharge rerouted thru the #1 feedwater heater, and restarted. About that time water level was up to the feed ring in #2 and a pipe noise was heard in the plant.

   Indicated feedwater temperature decreased from 420°F to 100°F about 22 minutes after the trip.

   P25-A was realigned to supply water through the first point heater 43 minutes after the trip. Feed to #2 SG was stopped 87 minutes after the trip.

6. Q. Describe the normal mode of operation of the main feedwater system with the new turbine driven main feedwater pump following a reactor trip/turbine trip at power. Compare system operation during the event with the normal mode (normal mode is electric pump on standby).

   A. Normal mode of operation is to have one motor-driven feed pump in the auto start mode. Following a trip, after assuring main feed pump is on and level is rising, auxiliary feed pumps are turned off.

7. Q. Describe the rupture of the main feedwater line (e.g. crack location, crack description, effective break flow area). Describe the damage to piping supports and hangers.

   A. The crack in the main feed line occurred in the counterbore transition of the first field weld of the main feed system upstream of the #2 SG nozzle. The joint is a standard 37^{o} weld prep from the transition ring on the steam generator nozzle to a 20^{o} elbow of A234WPB specification.

   Visual, magnetic particle, and RT examination of the failed joint indicate a precracked condition existed in the transition area of the counterbore. The break occurred when the crack was continued through the wall of the pipe. The ID crack on steam generator #2 is approximately 35 inches in length. The location is shown on attached figure Q.7. The through-wall crack is 11 inches in length. The average width of the through-wall crack is approximately 1/16-inch with a 2-inch length in the center of the crack about 3/32 wide.

   The counterbore was machine cut using a single point tool, parallel to the OD surface. It was then ground on a taper to the original ID of the pipe. The maximum depth of the counterbore in the through-wall direction is approximately 1/8 inch. It was noted that there was a definite tool mark at the transition area from straight to tapered section.

8. Q. Discuss your evaluation of the cause of the rupture.

   A. In the fabrication of the Maine Yankee Feedwater pipe to Steam Generator safe-end weld, a stress riser was left in the pipe when it was counterbored to accommodate the I.D. of the safe-end. Thermal and mechanical cycling over the years has caused cracks to originate at this stress riser. The presence of these cracks was difficult to detect with conventional NDE techniques because of the artifacts caused by the stress riser itself.

   The water hammer event of January 25, 1982, caused the failure of the #2 feedwater line at the location of this cracking.

9. Q. Did you analyze impact of thermal stresses alone on piping and piping supports?

A. Normal thermal analysis, as required by Power Piping Code (ANSI/ASME B31.1), has been performed on this system and these thermal loads are included in the design of the system's pipe supports.

10. Q. Describe modifications intended to prevent event from reoccurring (Procedural and hardware modifications). Explain how modifications cover remaining uncertainties in the understanding of the event.

A. 28 top-mounted J tubes of 3" diameter have been installed on the Steam Generator feed rings in place of the 76 bottom-mounted 1" nozzles. The tubes will provide greater assurance that the feed ring will remain full, minimizing both the probability of steam-water hammer and the occurrence of the non-uniform thermal condition in the horizontal pipe when auxiliary feed water is introduced. The J tube location will provide assurance that a trapped steam condition will not exist. The J tubes will also increase the pressure equalization area about 300%, minimizing any pressure surge that may occur. Procedural modifications will provide operator guidance to minimize the occurrence of thermal cycling and feed ring draining.

11. Q. Explicitly describe results of reviews of all prior NDE inspections on safe-end to elbow and safe-end to pipe welds on all three steam generators. Include details on test techniques. Also for safe-end to nozzle welds. Describe results of current UT inspection on these welds.

A. The review of NDE data on steam generator #2 weld 1A (safe-end to elbow) consisted of a review of the original construction radiograph, the radiograph taken as part of the USNRC Bulletin 79-13 program, and the radiography performed prior to removal of the elbow.

The original radiograph was shot using a S&W procedure which was equivalent to ASME Section I, Criteria, about 1968 Edition. The procedure required an IR-192 source, allowed AA type film, with a qualifying sensitivity of 4T.

The radiographs taken in 1979 and 1983 were to ASME Section III, NB (Section V RT with Section III Penetrameter requirements). The source was IR-192, the film EKC type T, with 2T sensitivity on the #15 penetrameter and 4T sensitivity on the #12 penetrameter.

The review of the construction radiograph shows a dull line at the transition area of the counterbore on steam generator #2. The line was not described on the original reader sheet. There were no other significant indications.

11. A. (Con't)

A review of the 79-13 radiographs indicated a darker, somewhat sharper area in the counterbore transition. There was no branching and the indication was very straight. Review of the 79-13 radiograph against the original construction suggested that the higher sensitivity associated with the type T film only enhanced the "tool mark" at the counterbore transition.

The 1983 radiographs show the indication of a crack which is approximately 35 inches in length on the ID of the pipe (length determined by mag. particle inspection on ID surface), originating in the counterbore transition. Except for areas of local tearing associated with the failure, all cracking was confined to this counterbore transition area.

The review of welds in steam generators #3 and #1 shows less density in the counterbore transition area in the 79-13 radiographs. The welds are, in fact, cracked in the same areas at steam generator #2 and at about the same magnitude, i.e., lengths of about 29 to 35 inches.

UT examination of safe-end to nozzle welds was performed as an expeditious means of gathering data on further cracking. It was not performed as part of the formal evaluation of these joints. There was no UT performed on these joints as part of the ISI program. The Class 2 portion of the ISI program is only 3 years old at MY, and to date these welds were not part of the schedule.

12. Q. Describe the conditions that lead to the previous installation of load spreaders on feedwater pipe supports.

A. During inspections of the feedwater system in 1973, hanger no. H-14 on #3 Steam Generator Feedline was observed to have spalled concrete behind the baseplate. An evaluation of the integrity of the support was attempted but it was not possible to determine the condition of the anchor bolts which are located behind the baseplate and inaccessible for NDE. It was decided to augment the existing anchor bolts with additional anchors having equivalent shear and pull-out load capability. The load spreaders provided this capability.

13. Q. Question Deleted

14. Q. Supply mechanical property (including toughness) information on the safe-ends, 15° elbows, and straight pipe (on No. 3 S.G.).

A. The safe-ends were supplied by C.E. The safe-end to S/G nozzle weld was a shop weld performed by C.E. The safe-end to 15° elbow weld is a field weld performed by S&W. The safe-end material is to Mat'l Spec. SA-508, Class 1. The 15° elbow is to Mat'l Spec. SA-234, Gr. WPB (SMLS).

14. A. (Con't)

<u>Safe Ends</u> - SA 508 - Class 1

Tensile Strength	76,000 psi
Yield Strength	51,000 psi
Elongation in 2"	35%
Reduction in Area	74.5%
Charpy Impact	+10°F

Three Specimen ft-Lb (137/121/142)

<u>Elbows</u> - SA 234 Gr WPB (SMLS)

Tensile Strength	60,000 psi (min)
Yield Strength	35,000 psi (min)
Elongation in 2"	22% (min)
Reduction in Area	38% (min)

<u>Pipe</u> - SA 106 Gr B

Tensile Strength	60,000 psi (min)
Yield Strength	35,000 psi (min)

15. Q. Is there any other pressure or temperature time history data available to support the "water hammer" event.

A. No.

16. Q. Show results of your stress displacement calculations on feedwater piping. (Suggested presentation is curves of stress levels vs. length). Do the force levels inferred by the damage to supports and snubbers correspond to the model results.

A. The attached tables Q.16.1 - Q.16.3 list the calculated thermal displacements at the various support points, the corresponding stresses at these points (B31.1, eq. #14) and the observed displacements recorded subsequent to the feedwater system event. As illustrated by these tables, the observed deformations are consistently below those displacements expected due to normal thermal growth. The damage reported to supports and snubbers is consistent with the relatively small displacements observed.

This review was conducted primarily to aid in identifying areas requiring detailed inspection.

17. Q. Does your inspection and repair program adequately address all potential causes of this event: water hammer, thermal stresses, fatigue cracking, as-built defects.

A. Yes. The stress risers have been blended out of the replacement els. See also response to #10.

18. Q. Was any pressure surge or transient associated with realignment of aux. feed water to first point heaters?

A. Industry experience and literature regarding steam-water hammer indicates that it is probably caused by rapidly condensing steam in the ring and/or adjacent pipe sections causing a rapid pressure change, first negative then positive, where the surging water impacts. We think this occurred at Maine Yankee just as the feed ring nozzles became covered by the rising level. The expanded thermal sleeve sections are evidence of the positive pressure surge which probably extended the existing crack in the pipe.

However, it is possible to postulate that the introduction of auxillary feed into the feed line just prior to the first point heater could cause a water-hammer type event or initiate a preliminary surge that precipitates the larger, more classical surge. The mechanism follows.

At the time of the trip, feedwater downstream of the first point heater is 1200 psi and 440°F. Loss of flow would trap the feed in the line between pump discharge check and the S.G. check. The pressure in the line would drop due to a 1" orificed warming line to saturation for 440°F, (367 psi) or the discharge head of the condensate pumps which is greater than 400 psi. If the condensate pumps were off or because of elevation differences the effective condensate pump pressure just below the last check was below saturation then a steam bubble would form in the line just below the last check. As aux. feed flow, 1200 psi, is introduced it would compress the existing 440°F steam bubble into a super saturated condition and force it thru the last check where it would contact the cold aux. feed. The collapsing super saturated steam would cause feedwater flow to surge, possibly initiating steam-water hammer in the ring and adjacent pipe.

The condensate pumps were on in this case and the elevation difference is inadequate. Nonetheless, our revised operating instructions will address this possibility to prevent its occurrence.

19. Q. When was last walk through of feedlines and supports in the containment?

A. During the 1982 refueling outage, the shock suppressors and some hangers and supports were walked through as part of the 1982 service inspection.

A complete walk through was conducted during the 1980 refueling outage.

20. Q. Were snubbers damaged by rapid application of load in excess of their load capacity, or because of the pipe displacement in excess of the capacity of the snubber displacement?

A. It appears that the failure modes for displacement and excessive load are identical. Our belief, based on reported pipe movement, is that failure was due to excessive load. The component that deformed apppears to be the weakest link in the assembly.

21. Q. Does any data on the No. 3 steam generator support a more severe event on that feedline?

A. The #3 steam generator feedwater loop could have experienced a more severe event than the other loops from pipe support perspective due to the following geometric and support differences:

1) Horizontal length of feedwater line in the annulus area is approximately 224 feet in loop #3, 69 feet in loop #2 and 25 feet in loop #1.

2) Only loop #3 has a mid-span anchor (H-13).

3) Only loop #3 has a change in elevation as it runs along the annulus. Either a pressure wave or water slug passing along this line and through the pair of back-to-back 30° bends just upstream of the anchor could account for the observed distress. However, it cannot be proven that this occurred.

t=0 @ 1433^02

QA

t, MIN	Stm Gen Pressure, PSIG #1	#2	#3	Containment Pressure PSIA	IN HG	Cmt Sump Level, ft	Cmt Dew Pt Temps, °F MPX 314	MPX 350	MPX 738	Cmt Vap P, PSI
0	840.0	838.8	837.1	16.03	32.52	0.710	73.7	71.2	73.7	0.84
1	894.9	892.2	894.5	–	–	–	–	–	–	–
2	882.7	879.9	882.9	–	–	–	–	–	–	–
3	875.4	872.5	874.9	–	–	–	–	–	–	–
4	869.9	867.6	870.0	–	–	–	–	–	–	–
5	866.2	864.0	866.4	–	–	–	–	–	–	–
9	864.4	861.6	864.0	16.01	32.53	0.729	73.3	71.3	73.3	0.83
19	876.7	873.8	876.2	16.49	33.24	2.615	76.8	83.3	76.8	0.93
22	876.4	874.0	876.2	16.55	33.44	2.780	80.1	89.8	80.1	1.03
28	875.6	872.9	875.3	16.73	33.58	2.980	85.5	98.5	85.5	1.23
43	871.8	868.9	871.3	16.74	33.39	3.281	90.5	98.7	90.5	1.44
47	873.6	870.9	873.3	16.74	33.32	3.333	90.3	98.1	90.3	1.43
51	875.6	872.8	875.2	16.68	33.27	3.416	90.1	97.1	90.1	1.42
63	873.0	870.1	873.1	16.61	33.01	3.490	90.2	94.3	90.2	1.43

Containment Zone Temperatures, °F

t, MIN	ZN #1, 28'	ZN #2, 28'	ZN #3, 28'	ZN #4, 30'	ZN #5, 30'	ZN #6, 30'	ZN #7, 30'	ZN #8, 30'
0	98.3	94.4	99.4	101.4	104.2	94.3	107.0	101.8
1	–	–	–	–	–	–	–	–
2	–	–	–	–	–	–	–	–
3	–	–	–	–	–	–	–	–
4	–	–	–	–	–	–	–	–
5	–	–	–	–	–	–	–	–
9	98.3	94.5	99.3	101.5	104.1	94.1	106.9	101.8
19	99.0	95.8	100.2	101.7	105.1	95.5	118.7	101.8
22	100.0	97.2	101.2	101.7	106.2	96.3	124.4	102.2
28	101.6	99.2	102.5	102.1	107.7	97.7	130.6	103.3
43	103.6	98.8	104.1	104.0	109.2	99.6	118.9	105.9
47	103.3	98.7	103.6	104.1	109.2	99.1	124.1	105.9
51	102.9	98.6	102.9	104.2	109.3	99.5	117.4	106.1
63	101.4	99.9	102.2	104.2	109.0	100.2	111.1	106.2

Q 4

8.10

t, MIN	CONTAINMENT ZONE TEMPERATURES, °F ZN #9, 60'	ZN #10, 60'	ZN #11, 60'	ZN #12, 60'	ZN #13, 110' (OFFSCALE HIGH)	CNMT AIR TEMP °F
0	100.7	102.8	105.3	82.0	150.1	101.9
1	–	–	–	–	–	–
2	–	–	–	–	–	–
3	–	–	–	–	–	–
4	–	–	–	–	–	–
5	–	–	–	–	–	–
9	100.6	103.0	105.6	81.7	150.1	102.0
19	102.2	103.5	125.4	90.1	150.1	107.5
22	103.5	104.3	133.8	94.0	150.1	110.3
28	105.1	105.8	122.7	93.9	150.1	109.0
43	106.5	108.3	117.6	95.1	150.1	108.9
47	106.4	108.3	117.5	94.7	150.1	108.9
51	106.1	108.2	117.2	94.0	150.1	108.6
63	105.4	106.4	115.1	92.4	150.1	107.4

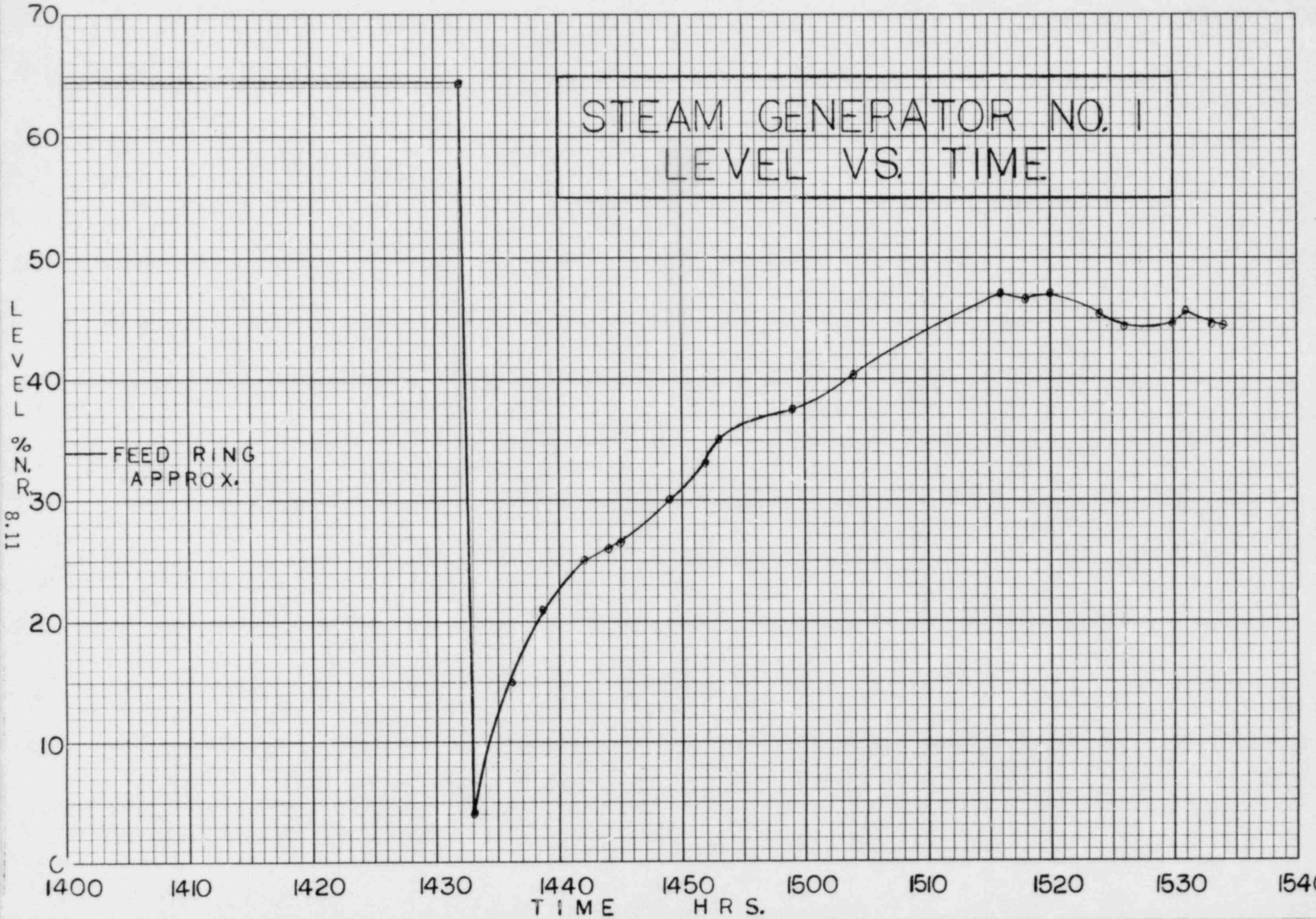
t, MIN	STM GEN FEED WATER T, °F #1	#2	#3	PRESSURIZER P, PSIG	RCS LOOP T_c, °F #1	#2	#3
0	439.5	444.3	439.7	2232.9	551.2	546.7	546.7
1	–	–	–	1904.3	541.7	634.9	535.3
2	–	–	–	1880.2	536.5	531.3	531.7
3	–	–	–	1886.3	535.4	530.2	530.6
4	–	–	–	1906.7	534.3	529.4	529.8
5	–	–	–	1930.8	534.0	528.7	529.1
9	438.3	440.4	439.0	1986.6	534	532	532
19	437.4	439.0	434.9	2032.2	534	532	532
22	434.9	431.2	433.1	2015.7	534	532	532
28	428.7	394.5	423.0	2085.8	534	532	532
43	428.1	201.1	379.7	2217.8	534	532	532
47	428.6	186.5	377.9	2218.7	534	532	532
51	426.2	163.2	333.8	2220.2	534	532	532
63	139.6	91.5	104.7	2237.2	534	532	532

RCS loop values for t = 9 to 63: FROM T_c TRACES (APPROX.)

K+E 10 X 10 TO THE INCH • 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

46 0780

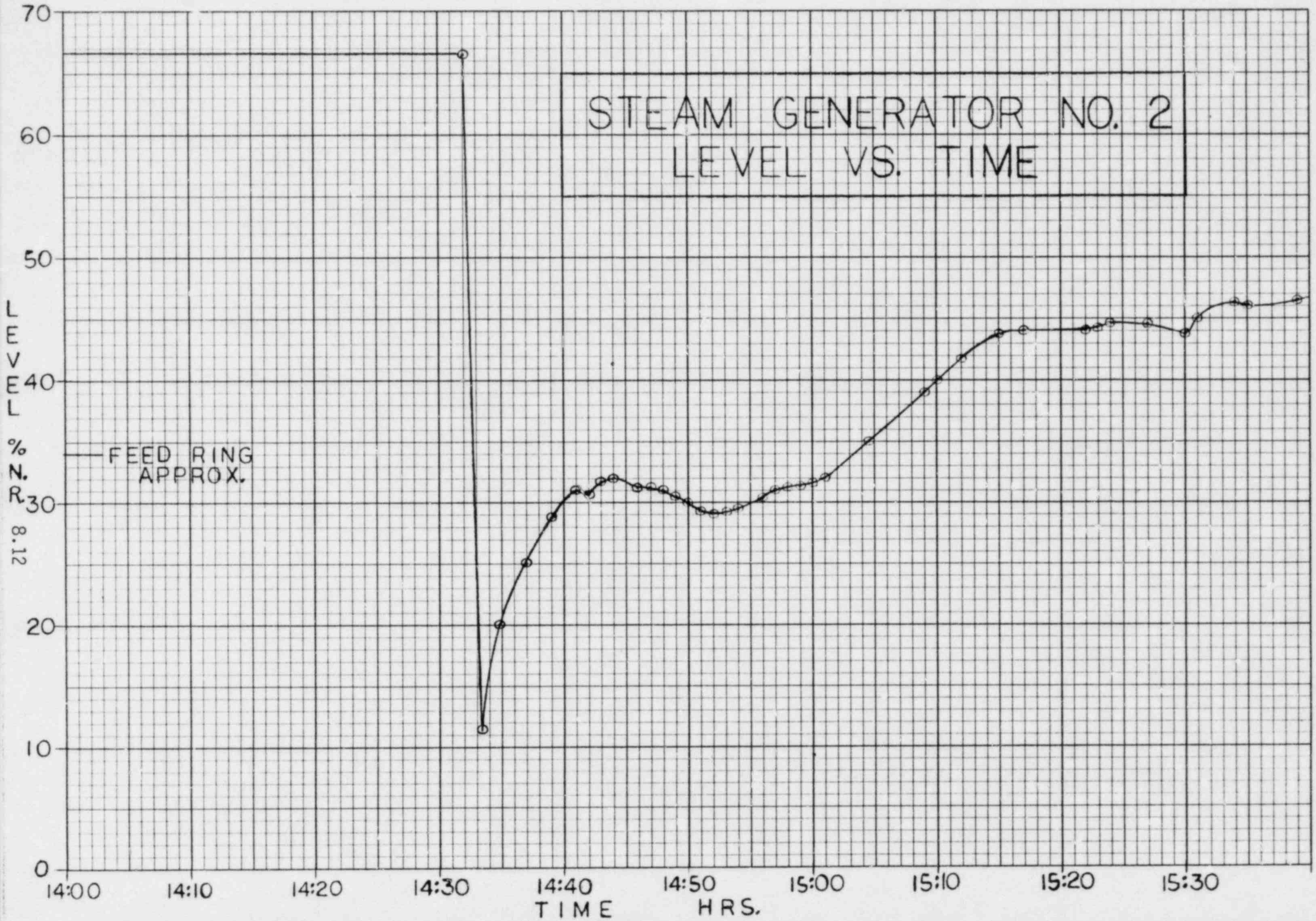
Q5



8.11

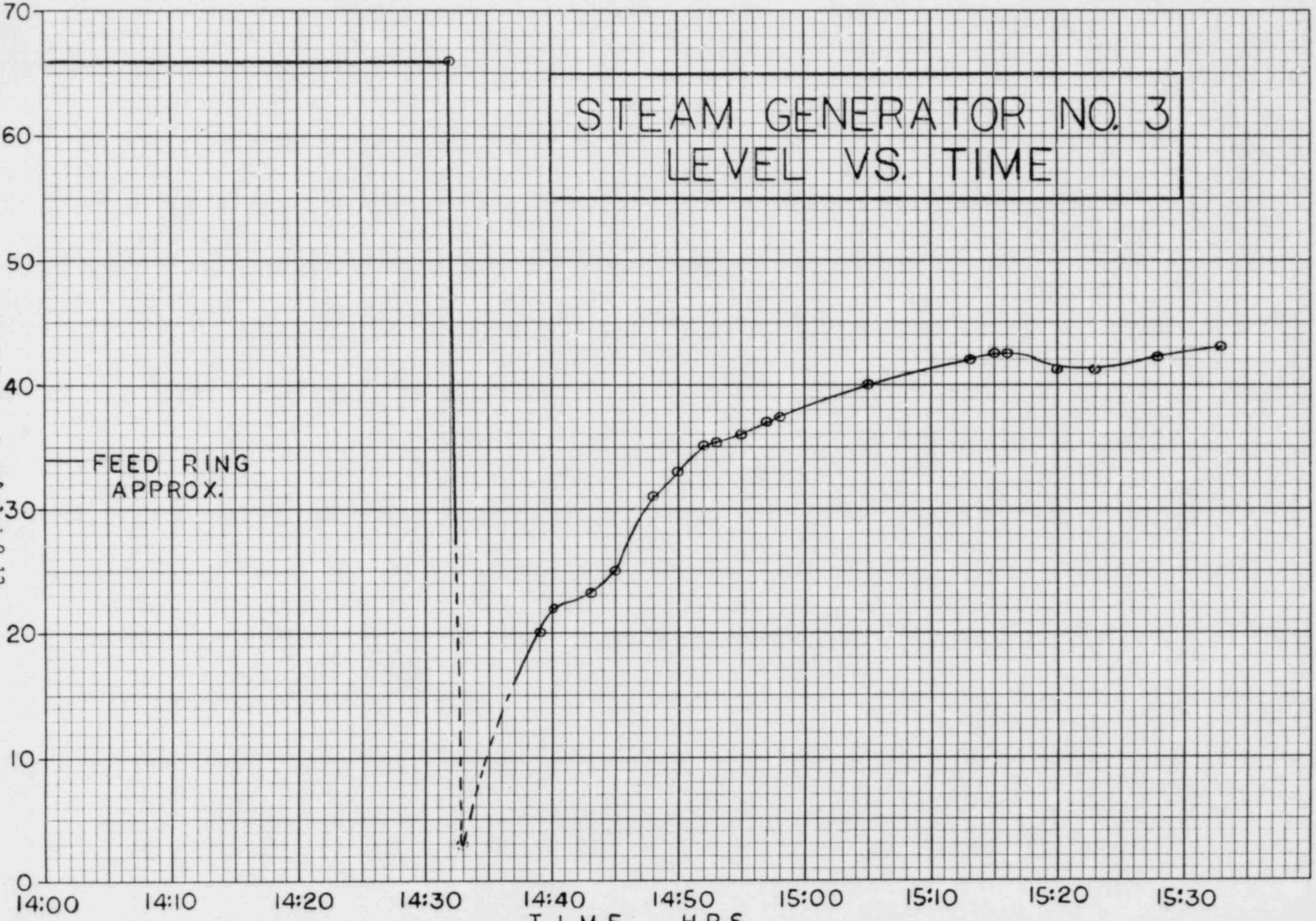
46 0780

Q5


STEAM GENERATOR NO. 2
LEVEL VS. TIME
FEED RING APPROX.
LEVEL % N.R.
8.12
70
60
50
40
30
20
10
0
14:00
14:10
14:20
14:30
14:40
14:50
15:00
15:10
15:20
15:30
TIME HRS.

46 0780

Q5



8.13

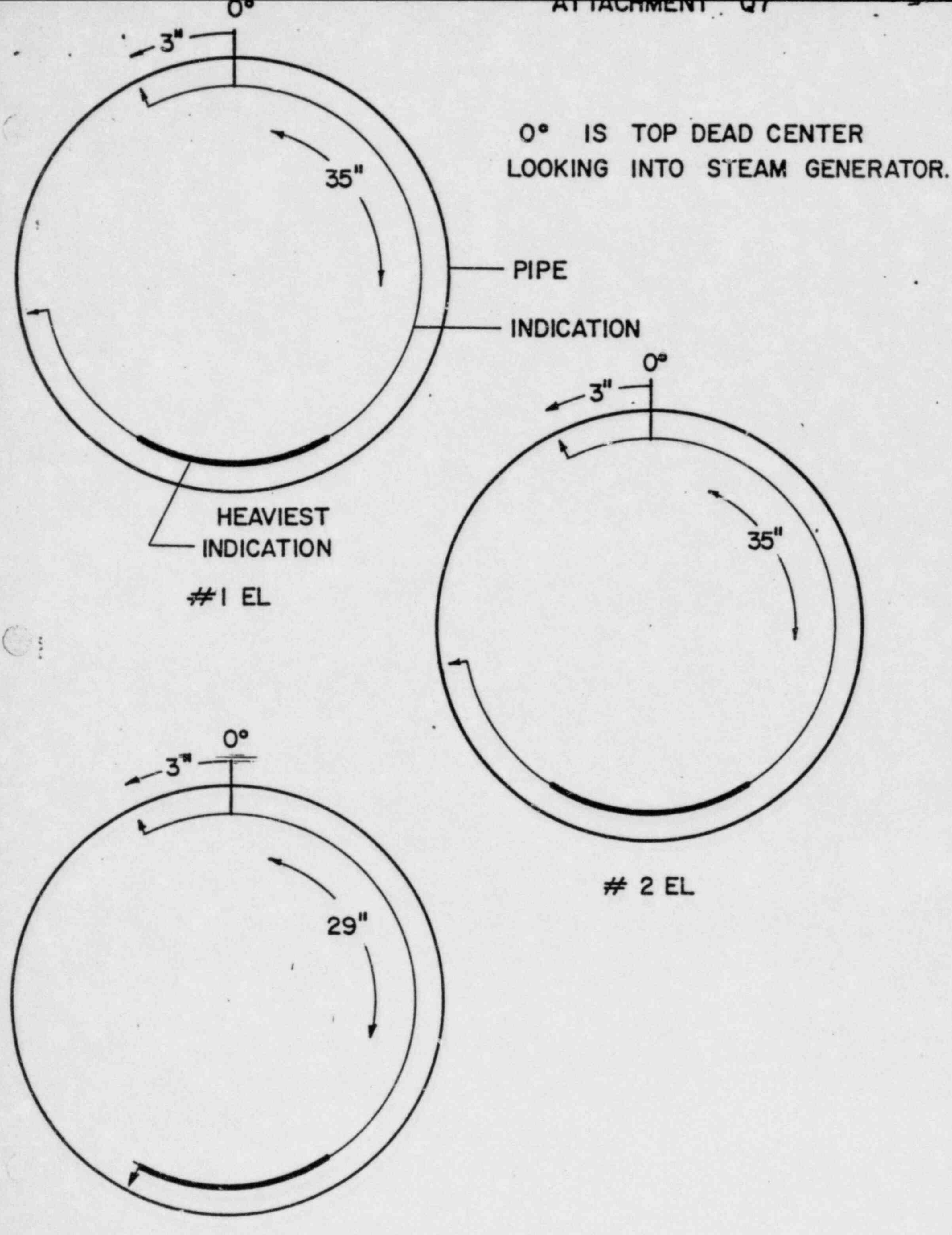

0°
3"
35"
0° IS TOP DEAD CENTER LOOKING INTO STEAM GENERATOR.
PIPE
INDICATION
HEAVIEST INDICATION
#1 EL
0°
3"
35"
2 EL
0°
3"
29"
3 EL

TABLE 16.1

STEAM GENERATOR FEEDWATER - LOOP #1
14"-WFDD-4-601 MKS-102A1-3

SUPPORT NO.	NODE	STRESSES (PSI) EQ. 14 ALLOW. = 37,500	THERMAL DEFLECTIONS *(IN.)			OBSERVED DEFLECTIONS
			ΔX	ΔY	ΔZ	
PEN. #53	54	23,138.	0.050	0.150	0.350	No observed deformation
M-328 S.W	50	12,600.	0.086	0.506	0.165	Slight gap on wall plate (< 1/8") (Appears old)
H-329 S.H. 155.	45	9,973.	0.890	0.699	1.686	Sliding plate indicates 1/2"-1" movement
LS3-202 SNUBBER	41	5,833.	0.850	1.528	1.739	No observed deformation
ASS-201 AXIAL SNUB.	62	15,329.	0.890	1.594	1.732	No observed deformation
STM. GEN. NOZZLE	59	22,146.	1.000	2.490	2.180	Offset piece indicated cracks, being replaced

SOURCE: CYGNA CALCULATION SET PI-007 FILE 82006-17F, REV. 0

*THERMAL DEFLECTIONS INCLUDE THERMAL ANCHOR MOVEMENTS

TABLE 16.2

STEAM GENERATOR FEEDWATER - LOOP #2

14"-WFDD-8-601 MKS-102E1-3

SUPPORT NO.	NODE	STRESSES (PSI) EQ. 14 ALLOW = 37,500	THERMAL DEFLECTIONS *(IN.) ΔX	ΔY	ΔZ	OBSERVED DEFLECTIONS
PEN. 55	215	9,974.	-0.080	0.156	0.349	No observed deformations
H-6 (S.H.)	183	8,441.	-0.193	0.301	0.068	No observed deformations
H-5 (Vert.)	145	7,759.	-0.903	0.209	0.616	No observed deformations
LSS-206	135	7,939.	-1.451	0.213	0.668	Loose bushing, minor surface spalling of concrete
LSS-205	125	5,833.	-1.610	0.214	0.649	No observed deformation
H-4 (S.S)	115	9,402.	-1.723	0.209	0.623	No observed deformation
H-3 (S.S)	100	9,421.	-2.269	0.209	0.205	No observed deformation
LSS-203 LSS-204	90	8,020.	-2.332	0.378	0.005	Loose bushings
N-2 (S.H.)	80	7,925.	-2.291	0.915	-0.396	No observed deformation
H-327 (S.H.)	70	10,018.	-2.209	1.176	-0.534	No observed deformation
110 ZELE @ E-1-2	5	15,170.	-2.160	2.078	-0.157	Offset piece cracked, being replaced

SOURCE: CYGNA CALCULATION SET PI-008 FILE 82006-17F, REV. 0

*THERMAL DEFLECTIONS INCLUDE THERMAL ANCHOR MOVEMENTS

916

TABLE 16.3

STEAM GENERATOR FEEDWATER - LOOP #3

14"-WFDD-10-601 MKS-102NI-3

SUPPORT NO.	NODE	STRESSES (PSI) EQ. 14 ALLOW = 37,500	THERMAL DEFLECTIONS *(IN.) ΔX	ΔY	ΔZ	OBSERVED DEFLECTIONS
PEN. 54	2	12,187.	-0.016	0.146	0.549	No observed deformation
H-18 (S.H.)	14	10,403.	-0.078	0.290	0.133	No observed deformation
H-19 (S.S)	28	8,369.	-0.695	0.183	1.444	1/4" (TANG.); 1" (Radial Sliding observed support ok
LSS 216	32	9,593.	-1.103	0.168	1.804	No deformation observed
LSS 215	36	9,549.	-1.353	0.170	1.956	No deformation observed
H16 (S.S)	40	10,840.	-1.842	0.183	2.091	No deformation observed
H-15 (S.S)	46	9,041.	-2.143	0.183	1.447	Small slide (1/4" TANG.); 1/4-3/8" Gap on right side of base plate
LSS-213 LSS-214	52	8,062.	-1.778	0.193	0.950	No deformation observed
H-14 (VERT-S.S.)	56	7,471.	-1.254	0.183	0.539	Bottom slide 1/4"-1/2" (TANG.) Top slide ≈1/2" anchor bolts only finger tight
H-13 (Anc)	68	13,054.	-0.311	0.199	-0.082	Spalled concrete (<1/2" deep) at W10 imbedment

SOURCE: CYGNA CALCULATION SET PI-009 FILE 82006-17F, REV. 0

*THERMAL DEFLECTIONS INCLUDE THERMAL ANCHOR MOVEMENTS

916

TABLE 16.3 (Continued)

STEAM GENERATOR FEEDWATER - LOOP #3

14"-WFDD-10-601 MKS-102-J1-3

SUPPORT NO.	NODE	STRESSES (PSI) EQ. 14 ALLOW = 37,500	THERMAL DEFLECTIONS *(IN.) ΔX	ΔY	ΔZ	OBSERVED DEFLECTIONS
H-12 (S.S.)	4	7,455.	-0.250	0.209	-0.715	No observed deflection
LSS 211 LSS 212	4	7,433.	-0.250	0.209	-0.715	LSS 212 shows small quantity of spalled concrete
H-11 (S.S)	7	7,507.	-0.134	0.209	-1.452	Slide Tang. 1/4" (approximately)
H-10 (S.S)	10	7,882.	0.256	0.209	-2.023	No observed deflection
LSS 209 LSS 210	11	7,591.	0.317	0.213	-2.065	LSS-209 Slight buckling of snubber eye bolt
H-9 (S.S.)	13	9,486.	0.824	0.209	-2.172	No observed deflection
LSS 207 LSS 208	16	9,979.	1.368	0.209	-1.869	No observed deflection
H-8	16	9,979.	1.368	0.209	-1.869	Slight slike (1/8") in opposite direction?
H-7 (S.H.)	20	10,567.	1.676	0.956	-1.332	No observed deflection
Nozzle @ E-1-3	29	16,974.	1,424	2.078	-1.584	Offset piece indicated

SOURCE: CYGNA CALCULATION SET PI-010 FILE 82006-17F, REV.0

*THERMAL DEFLECTIONS INCLUDE THERMAL ANCHOR MOVEMENTS

9) STARTUP TESTS

- HYDROSTATIC TEST

- WATER HAMMER TEST

MAIN STEAM & FEED SYSTEM HYDROS

ASME SECTION XI, SUBSECTION IWC-5000, HYDRO

The following systems will be pressurized for four hours and inspected for evidence of leakage:

Main Feed - From reg. values to the Steam Generators

Main Steam - SG's to the excess flow checks

Blowdown - SG's 1 & 2 to the CIS valves

SG 3 to the first isolation valve

Aux. Feed - SG's to the check valves

Pressure: 1188 to 1288 psig

Temperature: greater than or equal to 100°F

STEAM-WATER HAMMER TEST PROGRAM

OUTLINE

Prestart Conditions:

1. Plant in Hot Shutdown Condition.

2. Feedwater supplied using motor driven aux feed pump P-25-A via #1 heater, with steam supply on.

3. #2 Steam Generator Feed Line monitored for temperature, movement, sound.

4. #2 Steam Generator Aux Feed Control Valve open, #1 and #3 Aux Feed Control Valve Closed.

Test:

1. Reduce #2 Steam Generator level by blowdown and steaming to below auto start set point for auxiliary feed pumps (35% NR) while maintaining minimum flow.

2. Stop heated feed flow to #2 Steam Generator.

3. Start Motor Driven Aux Feed Pump P-25-C and establish 300 gpm max flow to #2 Steam Generator.

4. Record temperature, pressure, level, noise, pipe movement.

10) JUSTIFICATION FOR RETURN TO POWER

- SHORT TERM
- LONG TERM

10) JUSTIFICATION FOR OPERATION

SHORT TERM - RETURN TO POWER

1. Problem has been identified:

   a) Stress risers in SG nozzle and adjacent horizontal pipe.

   b) Thermal cycling and distortion causing cracks at stress risers.

   c) Steam-water hammer causes existing crack to fail.

2. Problem has been adequately addressed:

   a) Cracks repaired in nozzle and safe end.

   b) Horizontal pipe and 20° els replaced with new pipe without stress risers.

   c) Stress risers ground out of horizontal end of 90° el and safe ends.

   d) All pipe upstream of check valves has been NDE inspected and base line data recorded.

   e) Internal damage to feed ring repaired.

   f) Expanded thermal sleeves have been evaluated as adequate for restart.

   g) J tubes have been installed on feed rings to reduce the likelihood of steam-water hammer and somewhat mitigate non-uniform thermal stresses.

   h) Damaged hangers and supports have been repaired.

   i) Support attachments to pipe has been examined and found undamaged.

   j) Highest stress weld joint adjacent to support has been RT and is satifactory.

   k) Pipe movement has been evaluated and determined not to be excessive.

   l) Pipe returned to original position indicating no probable yielding at any point.

   m) Feedwater check valves inspected and found to be distortion free.

   n) Feedwater system from feed control station through SGs will be given a four-hour 120% pressure hydro test.

   o) Revised feedwater system operational guidance has been issued. Reduces the propability of steam-water hammer and reduces thermal effects.

   p) Operators have been trained in new guidance.

   q) Pre-startup operations will include a steam-water hammer test.

LONG TERM

1. Further measures will be taken prior to startup from the 1984 refueling outage to assure continued safe operation:

   a) All modified pipe welds and sections will be UT and compared to present baseline data.

   b) The inservice inspection program will be modified to revise the frequency and scope of inspection applied to the SG nozzle and horizontal pipe.

   c) All remaining pipe between feedwater check valves and steam generators will be modified to remove any stress risers that may exists.

   d) A more detailed evaluation of the expanded feed ring thermal sleeves shall be made. If warranted, the thermal sleeves shall be repaired or replaced.

   e) Further moficiations to the system or its operations will be investigated.