GENERAL ELECTRIC

NUCLEAR ENERGY ENGINEERING DIVISION

ATTACHMENT I

RMS TRANSMITTAL
NUMBER 85-003

RADIOACTIVE MATERIALS SERVICES
DATA REPORT

METALLURGICAL EVALUATION OF RANCHO SECO STEAM GENRATOR VENT LINE CRACKING

JULY 1985

PREPARED BY: RA Carnahan
R.A. Carnahan, Engineer
Plant Materials Technology

PREPARED BY: JAHanson for D.M. Hoyt
D.M. Hoyt, Engineer
Radioactive Materials Services

APPROVED BY: GM Gordon
G.M. Gordon, Manager
Plant Materials Technology

APPROVED: LAHanson
L. A. Hanson, Manager
Radioactive Materials Services

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RAC025.11

1. SUMMARY

A short pipe segment from the Rancho Seco steam generator vent line, containing a through-wall circumferential crack, was sent to the General Electric Co., Vallecitos Nuclear Center, for determination of the cause of cracking. The crack was located on the pipe side of a pipe-to-tee weld and extended 120° along one side of the pipe. Scanning electron microscopy and metallographic examination revealed that the cracking was transgranular. The fracture surface did not exhibit any typical telltale features that could be used to determine the crack initiation location, nor did it reveal conclusive evidence of a fatigue cracking mechanism.

Due to the appearance of the crack (transgranular, little branching), its location in an area of typically high residual stress, and a reported source of cyclic loading, it appears that the most likely cause of cracking was high cycle fatigue. The lack of direct metallurgical evidence of fatigue crack propagation should not be used to argue that fatigue cracking did not occur because fine fracture surface features could have been eroded away by high temperature, high pressure steam, worn away by repeated crack opening and closing, or hidden by the adherent oxide layer.

2. INTRODUCTION

In order to determine the cause of a leaking, through-wall crack, a metallurgical evaluation has been performed on a segment of one inch diameter stainless steel piping from the steam generator vent line at the Rancho Seco Nuclear Generating Station. As agreed upon with the Sacramento Municipal Utilities District, the following tasks have been performed:

1. Visual Examination
2. Low Magnification Optical Microscopy
3. Scanning Electron Microscopy
4. Metallography
5. Hardness Measurements
6. Elemental Analysis (X-ray Flourescence)
7. Gamma Scan of Surface Smear

A complete chemical analysis of the failed piping segment is in progress and will be documented in a supplement to this report.

3. VISUAL EXAMINATION

Visual examination of the pipe-to-tee weld section received on Tuesday, June 25, revealed a through-wall crack on the pipe side of the weld. The crack extended approximately 120° around the pipe circumference and was located on the side of the pipe if the opening of the tee is called the top. On the OD surface the crack appeared to follow the weld fusion line except at the tips, where it veered slightly inward toward the base material. On the ID surface the crack was located approximately one eighth inch from the weld root, also on the pipe side of the weld. Typical photographs taken prior to sectioning the sample are shown in Figures 1-4. One end of the through-wall crack is shown in Figure 1, and Figure 2 reveals that the tee is stamped "304", apparently identifying the tee material as Type 304 stainless steel. Figures 3 and 4 show top and end views of the pipe-to-tee sample, respectively.

A second sample, reported to be the mating portion of the crack-containing pipe with part of the adjacent pipe-to-elbow circumferential weld, was received on Friday, June 28. This section, shown in Figure 5, was approximately 3/4 inch in length and did not contain any cracking visible to the naked eye. Excluding the weld material, the total length of the crack containing pipe appears to have have been 3/4 inch to 1 inch.

4. LOW POWER OPTICAL MICROSCOPY

Following sectioning of the pipe-to-tee section, the ID surface was examined with a low power optical microscope at magnifications of 5-30X. A series of axial (longitudinal) cracks were found located between the ID surface weld fusion line and the end of the pipe. All of the cracks were located in the pipe base material. No cracking was discovered in the weld metal, although there does appear to be a slight lack of fusion at some locations along the weld root. The lengths of the axial cracks are on the order of 0.050 inch. A

typical area on the ID of the pipe-to-tee section is shown in Figure 6. No cracking was discovered in the tee.

Low power optical microscopy of the mating pipe sample (2nd sample received) also revealed the presence of a large number of axial cracks in the pipe base material. Again, no cracking was observed in the weld metal. In addition to the axial cracks, small patches of intergranular attack (IGA) were also found on the pipe ID.

5. SCANNING ELECTRON MICROSCOPY

A section containing approximately 50% of the through-wall circumferential crack was removed from the pipe/tee sample using a band saw and a metallurgical cutoff machine. The fracture surface was pried open and placed in a scanning electron microscope (SEM) for examination. The SEM results reveal that the fracture surface is completely transgranular with a moderate degree of secondary cracking. The fracture surface is also heavily oxidized due to exposure to high temperature water. A thorough scan at high magnification (1000X) did not reveal any evidence of fatigue crack propagation (i.e. no fatigue striations were found). Typical SEM photomicrographs of the fracture surface are given in Figures 7 and 8.

Due to the heavy oxide layer, however, it is very possible that fatigue striations, if present, would be hidden. Unfortunately we know of no reliable technique for removing an adherent oxide layer without disturbing the underlying material. It is also possible that no striations were found because reported crack opening and closing due to a cyclic loading mechanism could have worn them away or that they could have been eroded away by the leaking high pressure steam.

It should be noted at this point that examination of the fracture surface at low magnification did not reveal any characteristic macroscopic features of fatigue cracking such as beach marks or river bed patterns. Also, from examination of the fracture surface alone, it is not possible to determine whether crack initiation occurred at

the inside or outside surface of the pipe. An assessment of the most likely location of crack initiation is contained in the subsequent discussion section.

6. METALLOGRAPHY

Metallographic examinations were performed on a longitudinal plane through the leaking crack, near one of the tips, and on a transverse plane located between the leaking crack and the weld, which revealed a pair of the above mentioned short axial cracks. A composite photomicrograph from ID to OD of the through-wall crack at 125X magnification is given in Figure 9. This crack is completely transgranular and located in the pipe base material, except near the OD surface where it passes through the weld cover pass. The portion of the crack located in the weld is trans-dendritic.

A cross-section of one of the axial cracks is shown in Figure 10. These cracks were also found to be transgranular. At this particular plane of examination the depth of the crack shown is 0.004 inch.

The pipe base material appears normal for welded Type 304 stainless steel and includes a rather extensive zone of sensitization in the vicinity of the weld. A third metallographic specimen prepared for the purpose of determining the extent of the sensitized weld heat affected zone (HAZ) revealed that the sensitization extends greater than 1/4 inch from the weld along the ID surface. A composite photograph of the microstructure along the ID surface HAZ is shown in Figure 11. It should be noted that the presence of sensitization is not considered relevant to the observed cracking because sensitization is a grain boundary phenomenon and the cracking is transgranular.

7. HARDNESS MEASUREMENTS

Two microhardness traverses were conducted on a polished axial cross section of the pipe-to-tee sample. The Knoop microhardness values were converted to the Rockwell B scale for interpretation. All values recorded are considered normal for Type 304 stainless steel. The hardness data is given below:

Radial Traverse		Axial Traverse*
ID	90 R_B	85 R_B
	90	86
	87	89
	86	86
	87	87
	88	89
OD	91	90
		90

*Proceeding from weld interface to approximately 0.2 inch into pipe base material at mid wall.

8. ELEMENTAL ANALYSIS

A longitudinal cross section containing portions of the pipe base material, the tee base material, and the weld was submitted for an X-ray flourescence analysis for the purpose of material identification. It should be noted that this type of X-ray analysis is not quantitative and does not detect light elements such as carbon or nitrogen.

It was determined that both the pipe base material and the tee base material consist of Type 304 stainless steel. The weld metal consists of Type 308 stainless steel.

9. GAMMA SCAN OF SMEAR

The following gamma-emitting radionucleides were detected in a smear sample taken from the pipe surface:

Radionucleide	Microcuries
Co-58	0.19
Co-60	0.11
Ag-110	0.026
Cr-51	0.020
Nb-95	0.013
Mn-54	0.010
Fe-59	0.008
Zr-95	0.005
Sn-113	0.002
Cs-137	0.0015
Co-57	0.001

10. DISCUSSION

Based on the results of the examinations performed, it appears that the most likely cause of the through-wall cracking in the steam generator vent line is high cycle fatigue. Although there is no conclusive metallurgical evidence to substantiate the operation of a fatigue cracking mechanism, such as striations or beach marks, there is a good deal of circumstantial evidence, such as:

- Transgranular crack propagation.
- Cracking in an area of presumably high residual tensile stress.
- A likely source of cyclic loading.

Transgranular cracking could also be caused by a stress corrosion mechanism, but this would require the presence of contaminants such as chlorides or caustic substances. Chloride cracking would be extensively branched, totally unlike the cracking observed in the vent line. Caustic cracking would require the presence of hydroxide ions and a mechanism for concentrating them on the ID surface of the

pipe. Also, caustic cracking would be expected to follow an intergranular path in the sensitized stainless steel.

If it could be verified that crack initiation occurred at the OD surface of the pipe it would be clear that the cracking was due to fatigue due to the lack of an aggressive environment on the outside of the pipe. Since an obvious crack initiation site has not been found and because the crack front is normal to the pipe wall, it cannot be determined whether initiation occurred at the ID or OD. An ID crack initiation location would also be consistent with a fatigue mechanism, however, and perhaps even more likely than OD initiation in this case due to the effect of weld residual stress. If the residual stress, which acts as a mean stress, is more tensile at the ID than the OD (which is typically the case) then a lower cyclic loading stress amplitude would be required to place the ID surface beyond the fatigue limit.

The origin of the axial cracks on the pipe ID has not been determined. It is possible that they occurred during manufacture of the pipe due to a cleaning process such as pickleing, or due to exposure to contaminants during storage. The pattern of axial cracking is not typical of fatigue. One could postulate, however, that the close proximity of the two circumferential welds at the ends of the pipe could have caused hoop tensile stresses great enough for cracking to occur during cyclic loading

Because a crack initiation site has not been located it cannot be determined whether the axial cracks aided initiation of the through wall crack. Judging from the roughness of the pipe ID (see Fig. 9) it does not appear that the presence of a prior crack would have been necessary to cause initiation of the through wall crack.

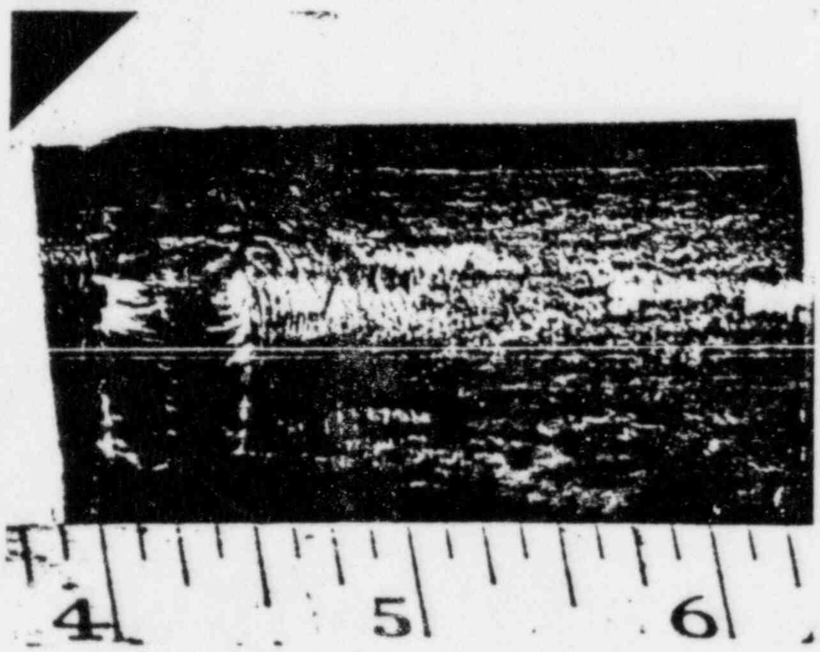


Figure 1. Photograph of Pipe-to-tee Sample Showing Through Wall Circumferential Crack Adjacent To Weld.

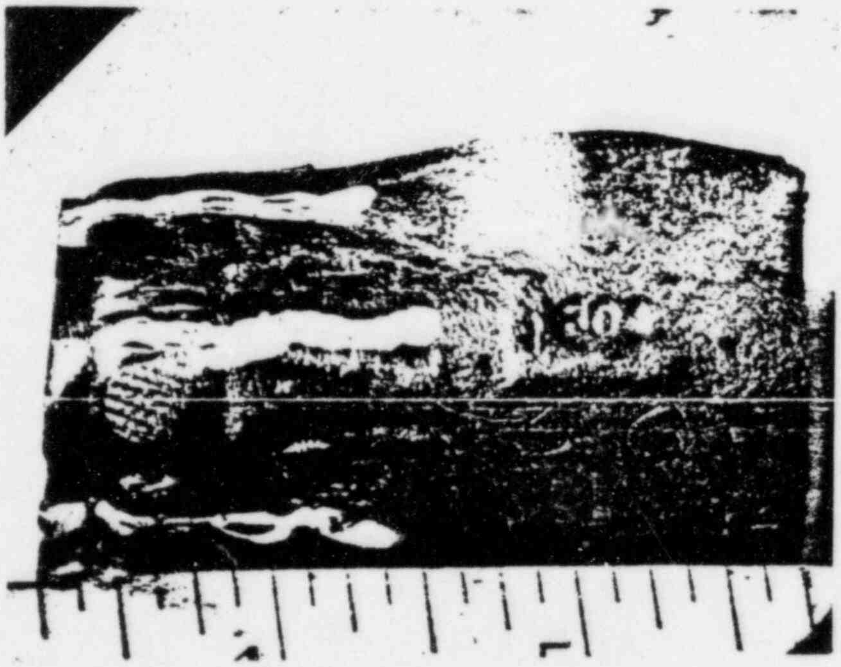
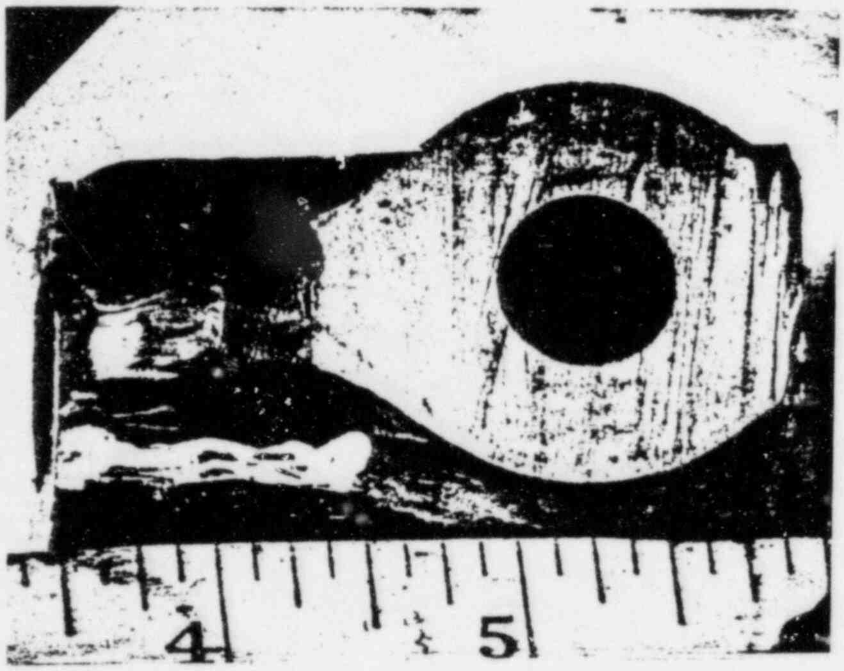
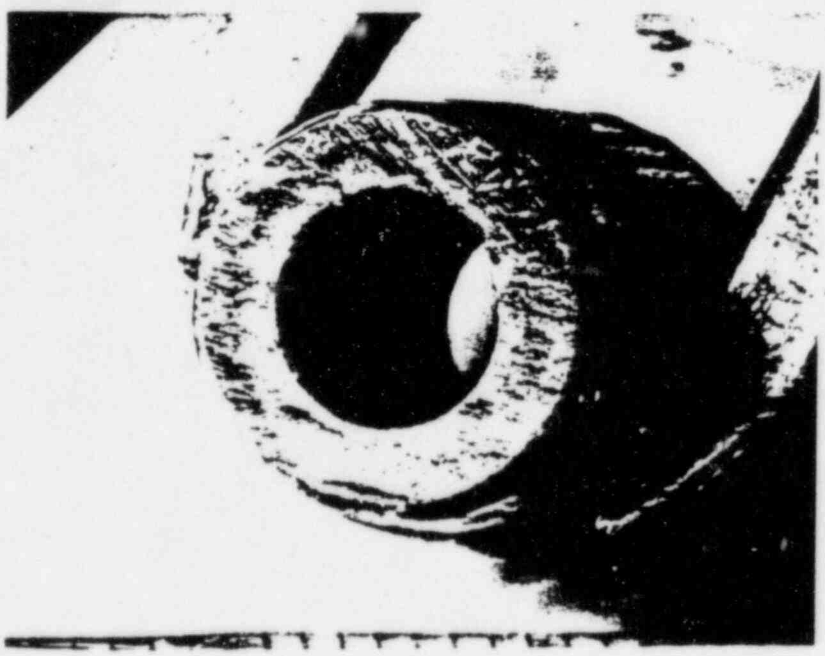


Figure 2. Side View of Pipe-to-tee Sample Showing "304" Stamp.





Figures 3 and 4. Top and End Views of the Pipe-to-tee Sample, Respectively.

Figure 5. Photograph of 2nd Steam Generator Vent Line Sample.

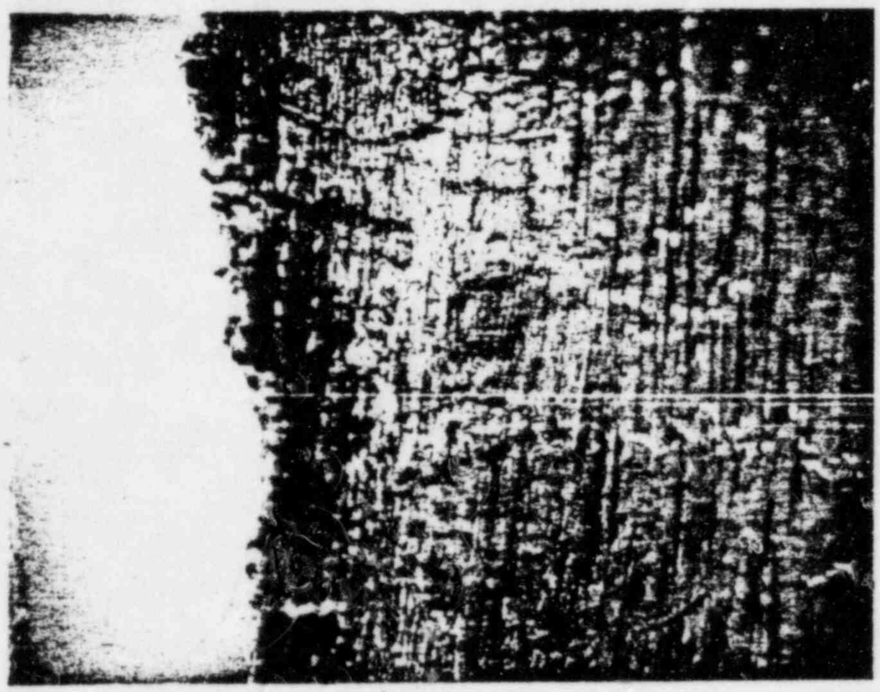
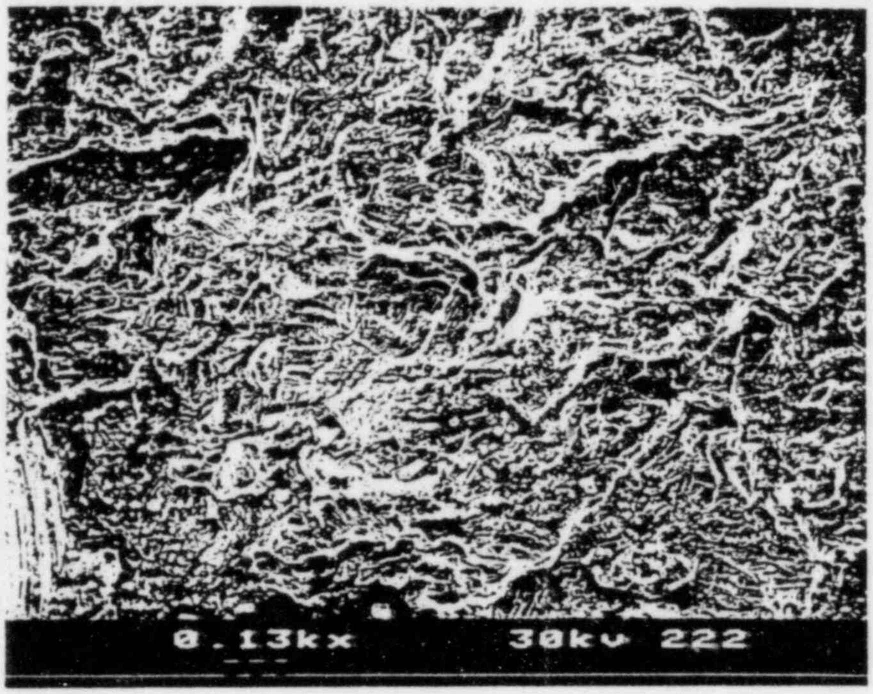


Figure 6. Photograph of ID Surface Adjacent to Through Wall Crack Showing Axial Cracks, 25X Magnification.



Figures 7 and 8. SEM Photographs of Pipe Fracture Surface Showing Transgranular Fracture Mode, 130X and 690X Magnification, Respectively.

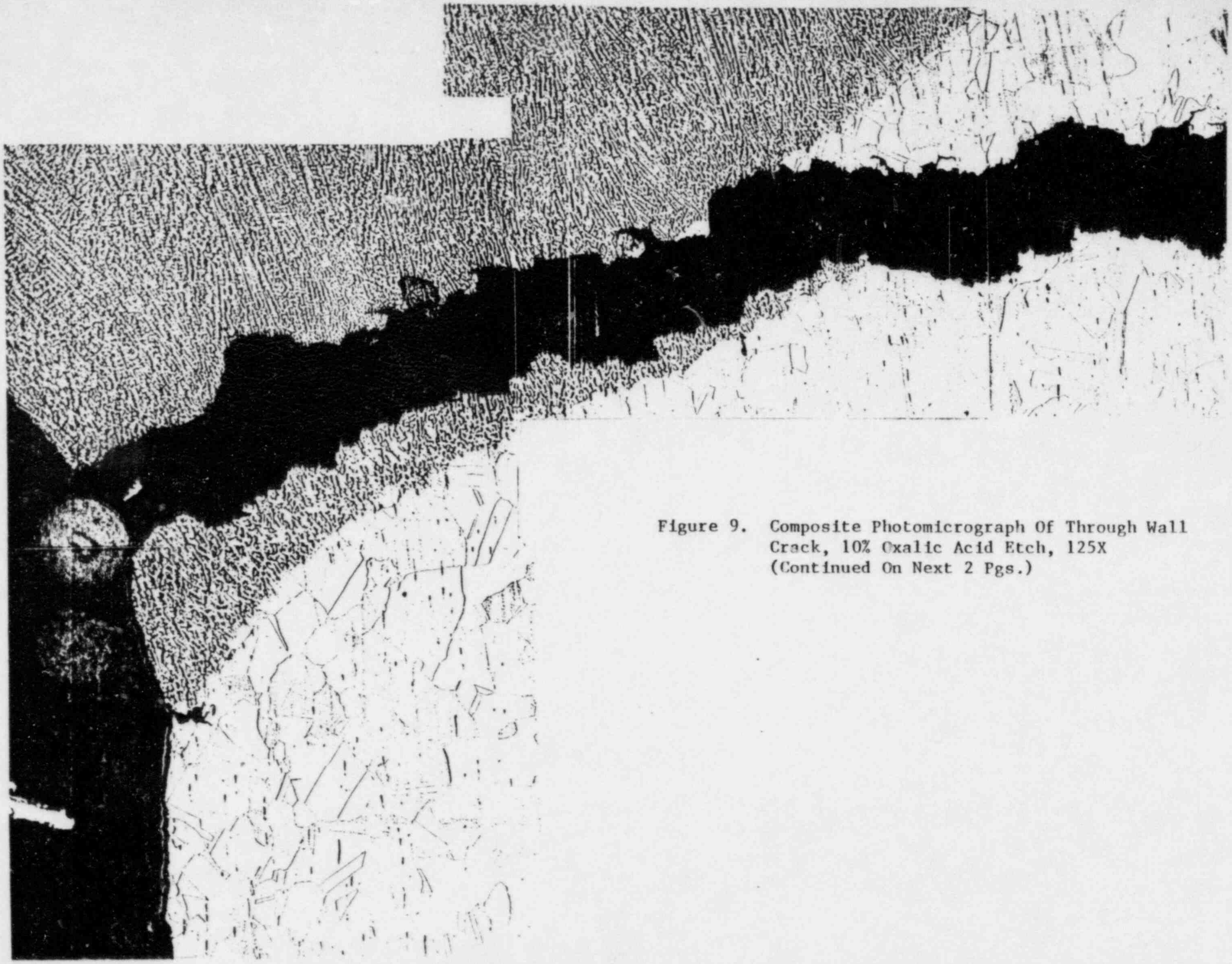


Figure 9. Composite Photomicrograph Of Through Wall Crack, 10% Oxalic Acid Etch, 125X
(Continued On Next 2 Pgs.)

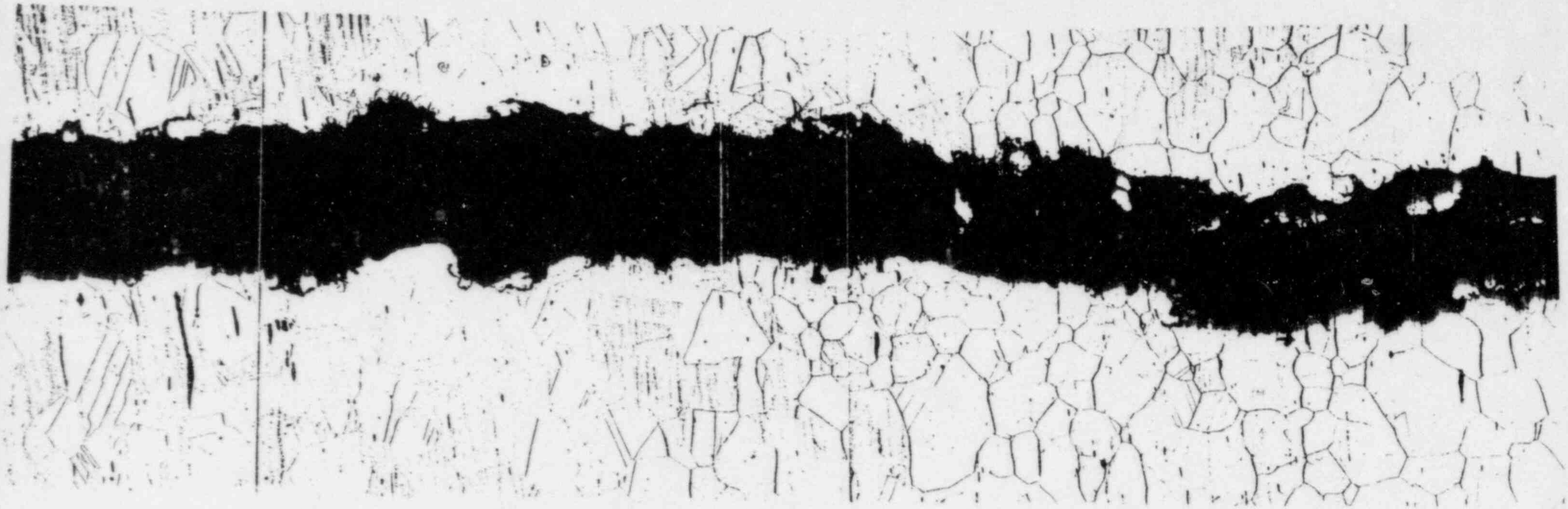


Figure 9. (Con'd) Mid-wall Portion Of Crack.

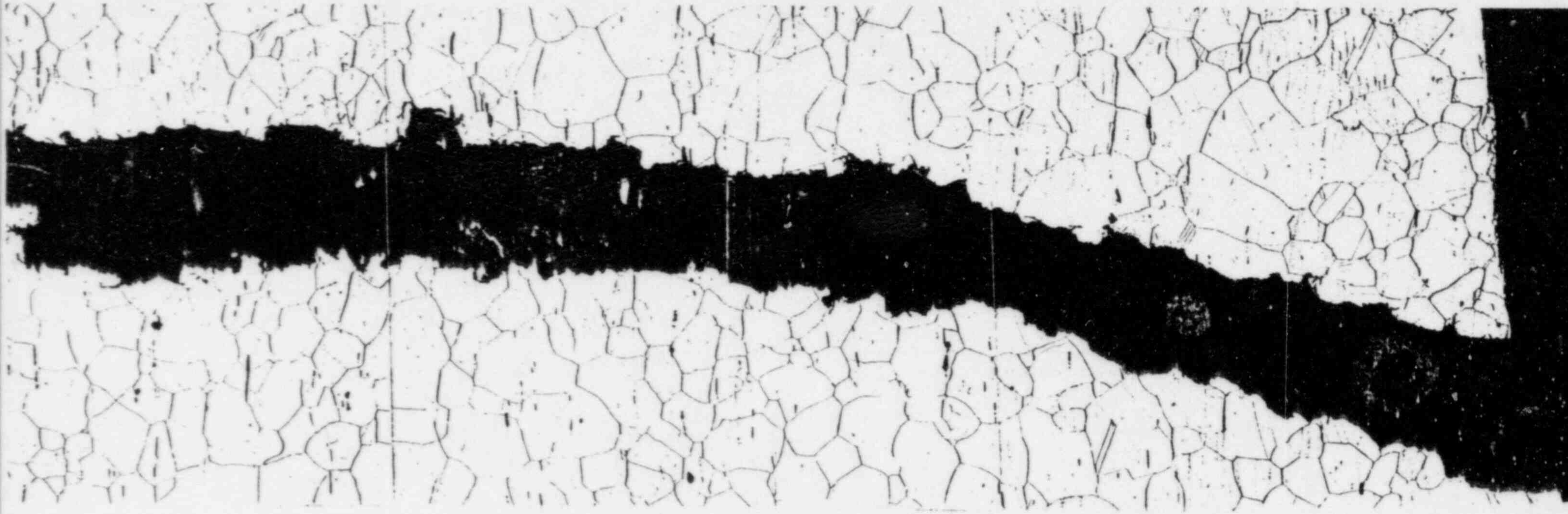


Figure 9. (Con'd) ID Surface Portion Of Crack.

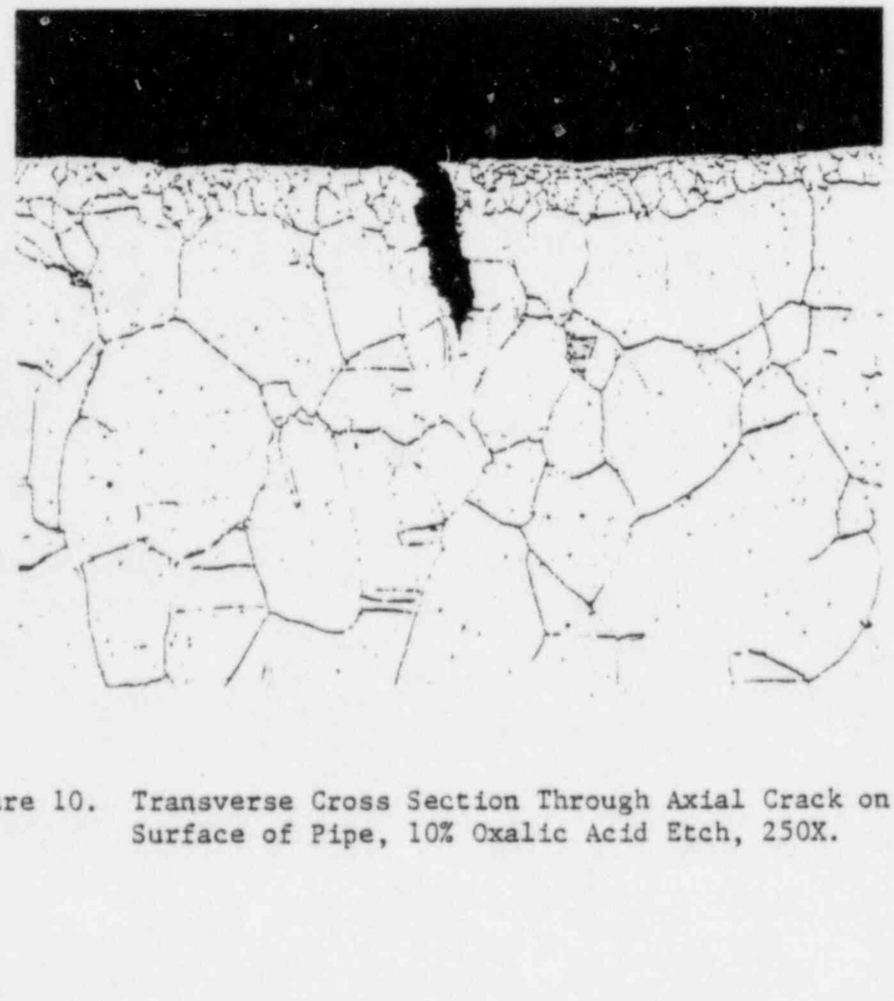


Figure 10. Transverse Cross Section Through Axial Crack on ID Surface of Pipe, 10% Oxalic Acid Etch, 250X.

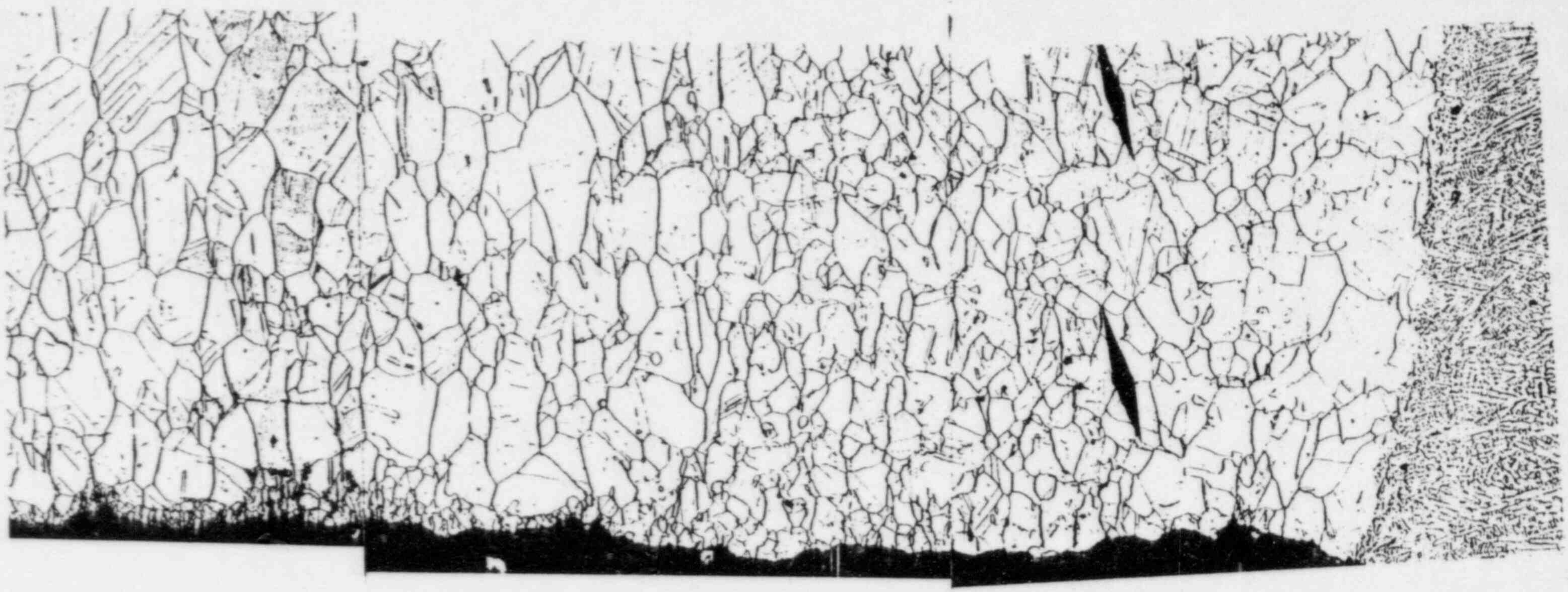


Figure 11. Composite Photomicrograph Of Weld HAZ Along Pipe ID Surface, 10% Oxalic Acid Etch, 125X. (Continued On Next 3 Pgs.)

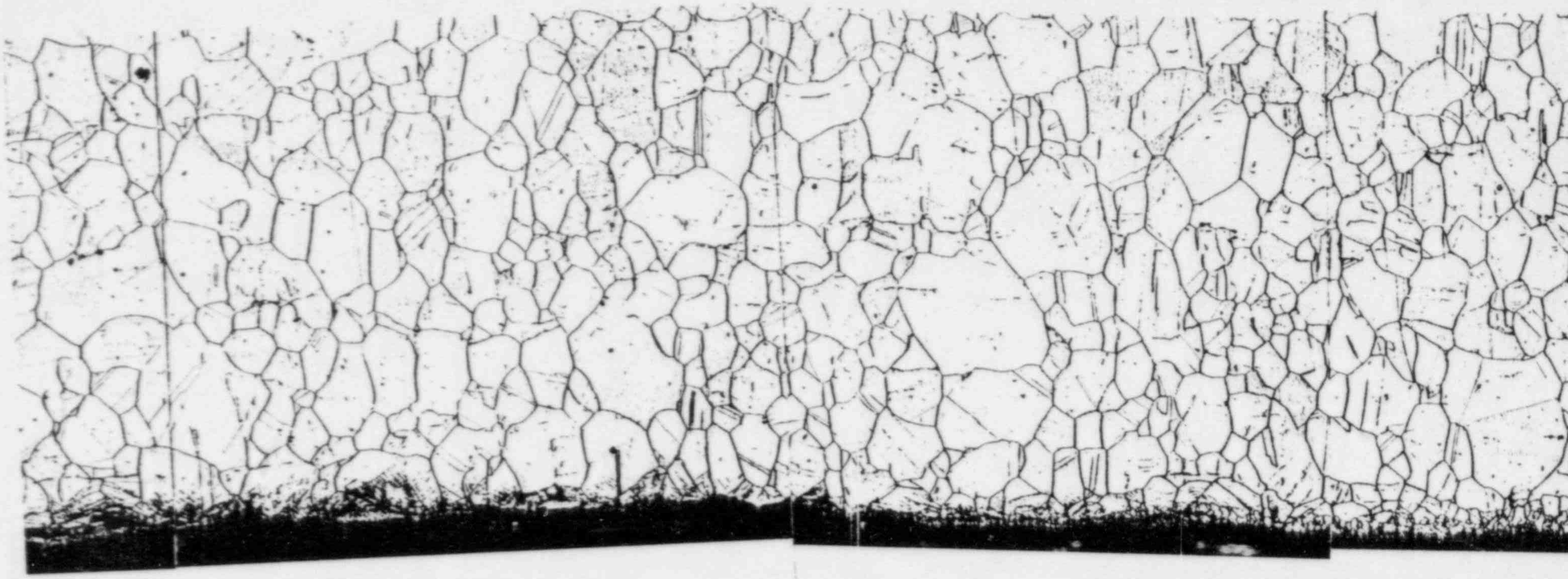


Figure 11, Sheet 2

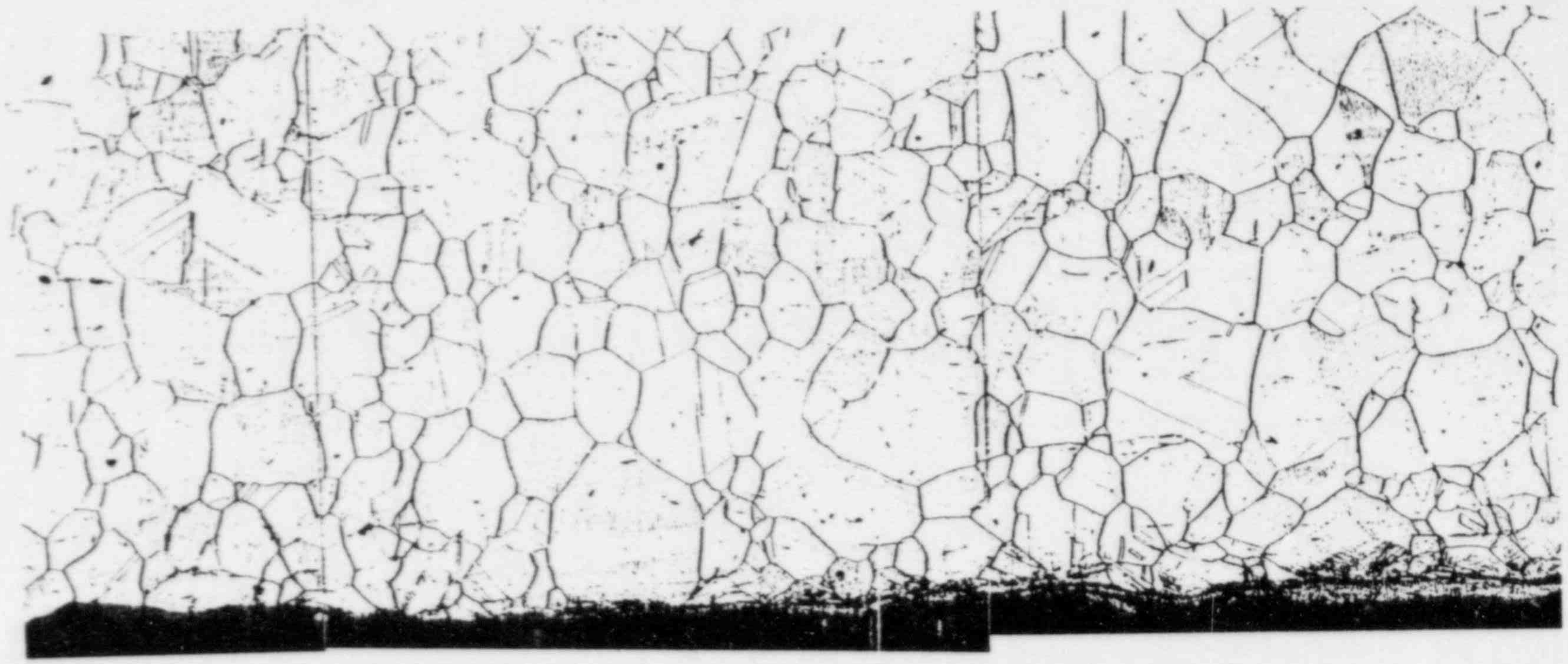


Figure 11, Sheet 3

Figure 11, Sheet 4

RANCHO SECO

STA TRANSIENT ASSESSMENT REPORT NO. 5

REACTOR COOLANT LEAK ON "B" HIGH POINT VENT LINE - JUNE 23, 1985

I. SUMMARY

At approximately 0227 on June 23, 1985 the Control Room Operators began to consistently receive RBDAT (Reactor Building Drain Accumulation Tank) dump annunciators. The RBDAT is a 120 gallon tank which is gravity fed from the "B" Reactor Building Normal Sump. The RBDAT dumps to the East DHR pump room sump which is pumped to the Spent Regenerant Tanks. Control Room operators confirmed a leak on the "B" hot leg via control room TV monitors using Reactor Building camera's.

The reactor was in a hot shutdown condition: RCS at 532°F, 2155 psig, 1271 ppmB, and CRA group 1 @ 100% withdrawn. Operators initiated a plant cooldown and depressurization per O.P., B.4, "Plant Shutdown and Cooldown." The Reactor Building was evacuated and an Unusual Event was declared per AP 501, "Recognition and Classification of Emergency," TAB 11, "Loss of Coolant."

Plant cooldown and depressurization occurred without problem. The Makeup Control Valve (LV-21503) was able to maintain pressurizer level and RCS pressure. An accurate leak rate could not be calculated due to the changing parameters. However, due to the consistent nature of the RBDAT dumps the leak rate could be estimated at approximately 17gpm.

A Reactor Building entry was made after RCS pressure was reduced significantly. The leak was confirmed to be on the "B" hot leg highpoint vent line (20564-1" CA) and could not be isolated (see Enclosure 7). Further inspection revealed an approximate 120° circumferential crack in a weld on the 1" schedule 160 pipe. The emergency high point vents, vent to R.B. atmosphere, vent to vent header, and nitrogen supply line are connected to this 1" pipe. The plant was further cooled down and depressurized so the RCS could be drained down and the weld repaired.

II. TRANSIENT ASSESSMENT

6/23/85	Sequence of Events
≈0100	Maintenance personnel unsuccessfully attempt to repair packing leak on SFV-70001 and leave R.B. Estimate packing leak to be 2-3 gpm.
0147	RBDAT dump; leak rate ≈.044 gpm.
0212	RBDAT dump; leak rate ≈4.8 gpm.
0227	RBDAT dump; leak rate ≈8 gpm.
0240	RBDAT dump; leak rate ≈9.2 gpm.
0251	RBDAT dump, leak rate ≈10.9 gpm.
0301	RBDAT dump; leak rate ≈12 gpm.
0212-1837 (to cold S/D)	RBDAT dumps 105 times. Avg. leak rate ≈12.65 gpm. Highest consistent leak rate ≈17.1 gpm.
≈0330	HP technician and two operators enter R.B. to hang clearance for continued maintenance on packing leak on SFV-70001. Notice R.B. more humid than normal. Before finding problem-contacted by Control Room and told to leave the building. Control Room operators had noticed steam leak in "B" D-ring" via TV monitor.
0351	Initiated normal plant cooldown in accordance with O.P. B.4, "Plant Shutdown & Cooldown." Secured "A" & C Reactor Coolant Pumps.
0356	Isolated Letdown per O.P., B.4; RCS depressurization started.
0404	Secured depressurization to drive group #1 control rods in, per O.P., B.4.
0405	Confirmed >10 gpm leak, Tech. Spec. LCO -T.S. 3.1.6.1 and 3.1.6.3. Confirmed a leak on "B" Hot Leg High Point Vent System via control room TV monitor. Declared Unusual Event. Made status notification to State OES, TRI-counties, and NRC in accordance with Emergency Plan AP 506, "Notification/Communication." Reactor Building evacuated.
0409	Tripped Group 1 rods per O.P., B.4.
0417	Bypassed HPI per O.P., B.4.

II. TRANSIENT ASSESSMENT

6/23/85	Sequence of Events
0442	Subcooling margin is 95°F.
0502	Withdrew Group 1 control rods and high flux trip reset to 4.9%.
0734	Site personnel enter R.B. to inspect leak and manually isolate if possible. Leak is on 1" line in the high point vent system and is not isolable.
1535	Started "A" DHR pump.
1814	Secured "B" RCP.
1838	Plant in cold shutdown condition. Unusual Event is terminated.
1840	RBDAT dump; leak rate is still approximately 8.6 gpm.
2315	Secured last RCP. T_{cold} = 145°F.
2347	RBDAT dumped; leak rate is still approximately 7 gpm.
*(cold S/D-0700)	RBDAT dumps 38 times. Avg. leak rate ≈ 6.1 gpm.
0700	RBDAT dump; leak rate is now approximately 2gpm.
2115	Started draining RCS per O.P., A.1 Reactor Coolant System.

Pre-Transient Review

The Reactor was in a hot shutdown condition; RCS at 532°F, 2155 psig. 1271 ppmB and CRA group 1 @ 100% withdrawn. "C" and "D" Reactor Building Emergency Coolers were in service. The makeup pump was running and the "A" and "B" HPI pumps were operable and in standby. Maintenance personnel were preparing to enter the Reactor Building to repair a packing leak on SFV-70001 (pressurizer liquid sample valve). This packing leak also somewhat "masked" the RCS vent line leak rate, Maintenance personnel had estimated the packing leak to be approximately 2-3 gpm.

Initiating Event

The initiating event of this transient was an approximate 120° circumferential crack of a weld on the "B" hot leg 1" vent line. The preliminary investigation of the weld failure by SMUD Engineering revealed that 3 piping supports and a dummy spool piece, which were considered in the original stress analysis, were not installed. Due to the

II. TRANSIENT ASSESSMENT

missing supports, excessive vibration may have led to the failure of the weld. The investigation is still in progress at the time of this writing.

Plant Transient Response

(see Enclosures 1-4). The plant cooldown and depressurization were performed smoothly and in accordance with O.P., B.4. The makeup control valve (LV-21503) was able to maintain pressurizer level and RCS pressure. There were no radiation alarms received in the control room.

Operator Actions/Procedural Adequacy

The operators involved in this transient performed in a competent and safe manner. The operators wisely used the control room camera to locate the leak and other indications to verify the RCS leak (i.e., RBDAT dumps, Makeup tank level and Makeup flow). Operating Procedure O.P., B.4, "Plant Shutdown and Cooldown" was used to shutdown the plant and no problems were noted. An Unusual Event was declared using Emergency Plant Procedure AP 501, "Recognition and Classification of Emergency" TAB 11, "Loss of Coolant." Notifications were made using AP 506, enclosures 7.1 and 7.2. The Reactor Building was evacuated for personnel safety. An Occurrence Description Report (AP.22) was written.

Safety Considerations

This transient did not adversely affect the safety of the plant. Pressurizer level, RCS pressure and subcooling margin were always maintained. T_{avg} did not increase.

Even if the 1" highpoint vent line had sheared off completely, the Safety Features Actuation System would have been able to handle this size LOCA. (See Enclosure5) 1" schedule 160 piping corresponds to a .815" inside diameter which would be a .00362 ft^2 break size. HPI alone will handle a .04ft^2 break size.

It has been calculated that a .00362 ft^2 break at a system pressure of 2155 psig and 532°F temperature would be a leak of 610 gpm. <u>Without operator action,</u> this would cause an RCS depressurization and reactor trip at 1900 psig and would lead to an SFAS trip at 1600 psig. At 1600 psig, 3 HPI pumps would be able to deliver 625 gpm which would be enough to overtake the leak and cause the RCS to refill and re-pressurize. If the RCS was a power and T_{avg} @ 582°F and again with no operator action, a 610 gpm leak would undoubtedly lead to an SFAS initiation because of the 1900 psig reactor trip and the subsequent cooldown causing a pressurizer outsurge. Again in time, high pressure injection would overcome

II. TRANSIENT ASSESSMENT

the leak rate and refill and re-pressurize the RCS.

Several scenarios were performed at the simulator with operator action. These preliminary results indicated that SFAS would not automatically be initiated. A known leak of 17 gpm was increased to 610 gpm with the operators anticipating the increase. This scenario was performed at 532°F and at 582°F/100% power. With the leak @ 532°F, letdown was isolated and both HPI pumps were started in addition to the makeup pump running. All four HPI valves were opened. Pressurizer level was maintained and the lowest RCS pressure reached was ≈2050 psig.

At 100% power, 582°F, and with the known leak, the plant was runback. "A" HPI pump was started along with one inject valve open to maintain pressurizer level. At 160" in pressurizer, the reactor was tripped in accordance with Casualty Procedure C.3, "Small Reactor Coolant Leak." The "B" HPI was started and all four inject valves were then opened. The lowest pressurizer level reached was ≈50 inches and the lowest RCS pressure ≈1900 psig.

The above scenarios were performed again with one HPI pump unavailable. There was no significant change with the scenarios at 532°F. With the leak at 582°F, the lowest pressurizer level reached was 38" and lowest RCS pressure ≈1830 psig. These scenarios were preliminary analysis and more analysis will be performed as warranted.

It must be noted that as the RCS pressure decreases, the leak will also decrease but the HPI flow will increase. Enclosure 6 shows HPI flow vs. system pressure. It should also be noted that this curve shows pre-throttled HPI flow and more HPI flow is actually available. The dashed curve represents leak rate vs. system pressure as calculated by B & W Engineering at several pressures and extrapolated. The intersection of the solid and dashed curves indicates at what pressure the RCS would be at when the leak would be in equilibrium with the HPI flowrate.

Areas for Further Review

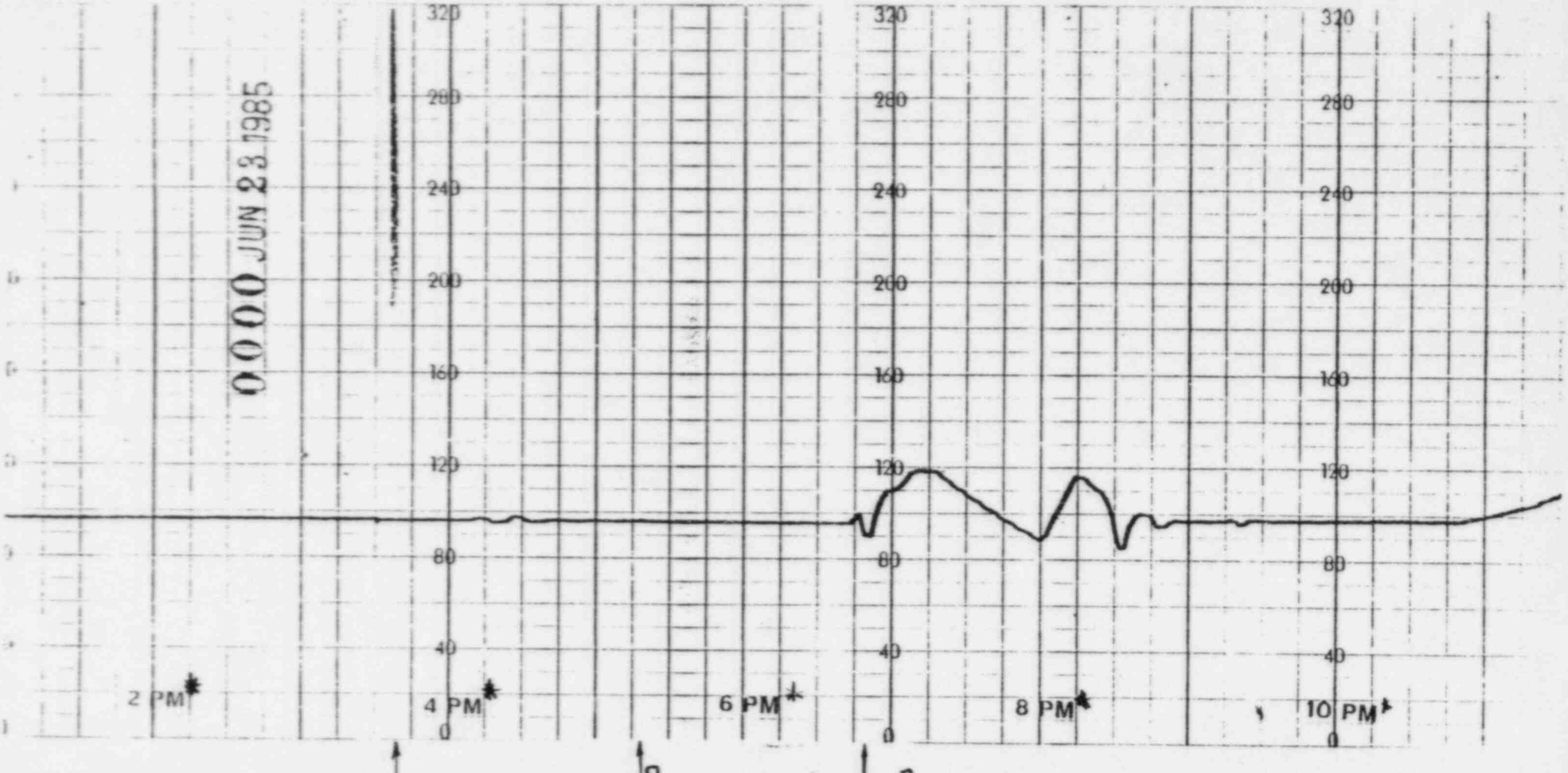
The failed weld has been removed and sent away for analysis. Radiography of additional welds in the high point vent system was performed. A walkdown and inspection of additional Class 1 piping system hangers were also performed. The results of this investigation are still pending.

Prepared by J. Delrue date 7/3/85

J. Delrue

ENCLOSURE 1

Rancho Seco Pressurizer Level LR-21503 June 23, 1985



0030 ◻

0212 ◻
Beginning of
RBDAT Dumps

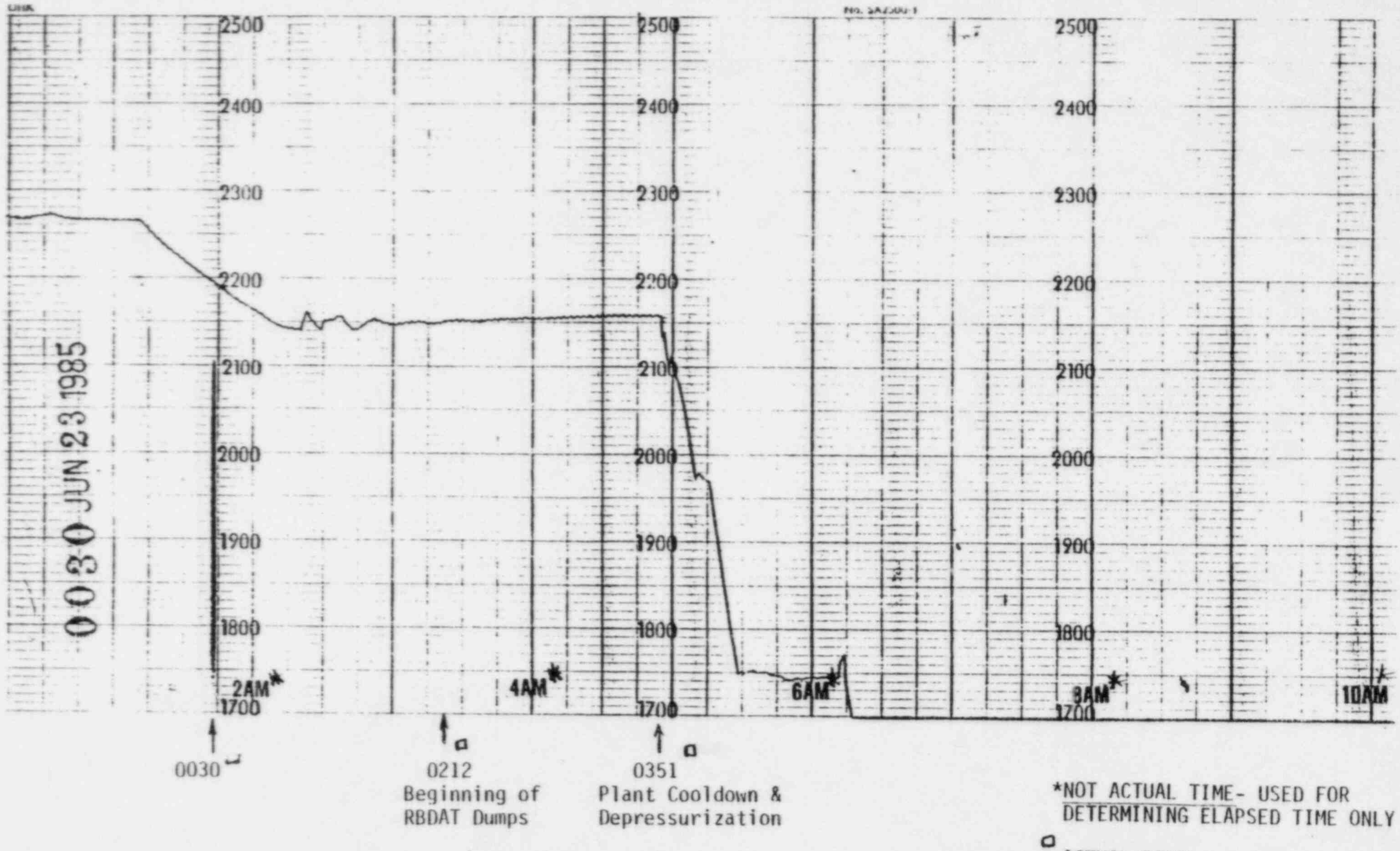
0351 ◻
L/D Isolated; Plant
C/D & Depressurization

*NOT ACTUAL TIME-USED FOR
DETERMINING ELAPSED TIME ONLY

◻ACTUAL TIME

ENCLOSURE 2

Rancho Seco   Reactor Coolant System Pressure PR-21038   June 23, 1985

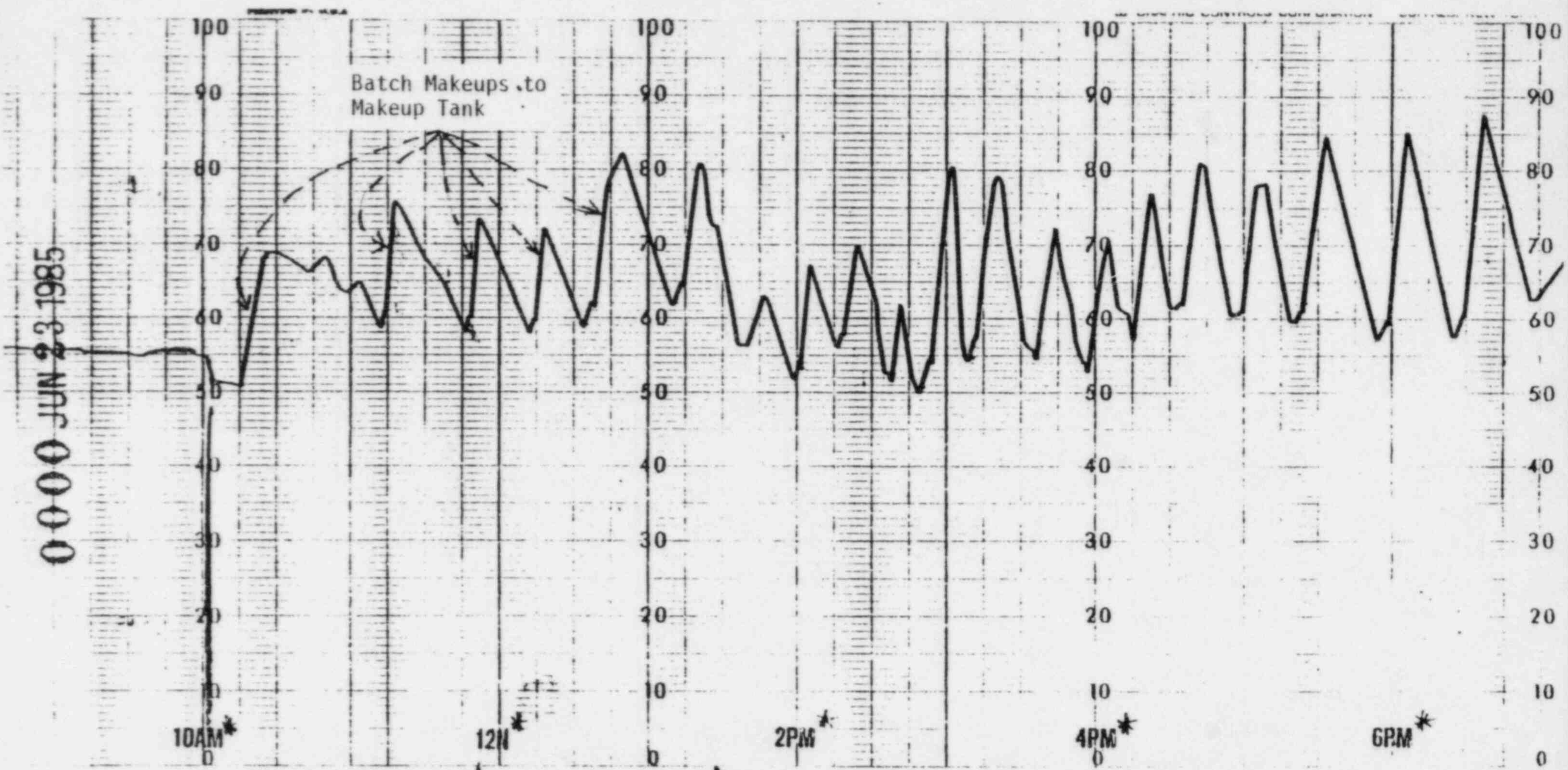


*NOT ACTUAL TIME- USED FOR DETERMINING ELAPSED TIME ONLY

□ ACTUAL TIME

ENCLOSURE 3

Rancho Seco Makeup Tank Level LR-23502 June 23,1985



0212□
Beginning of
RBDAT Dumps

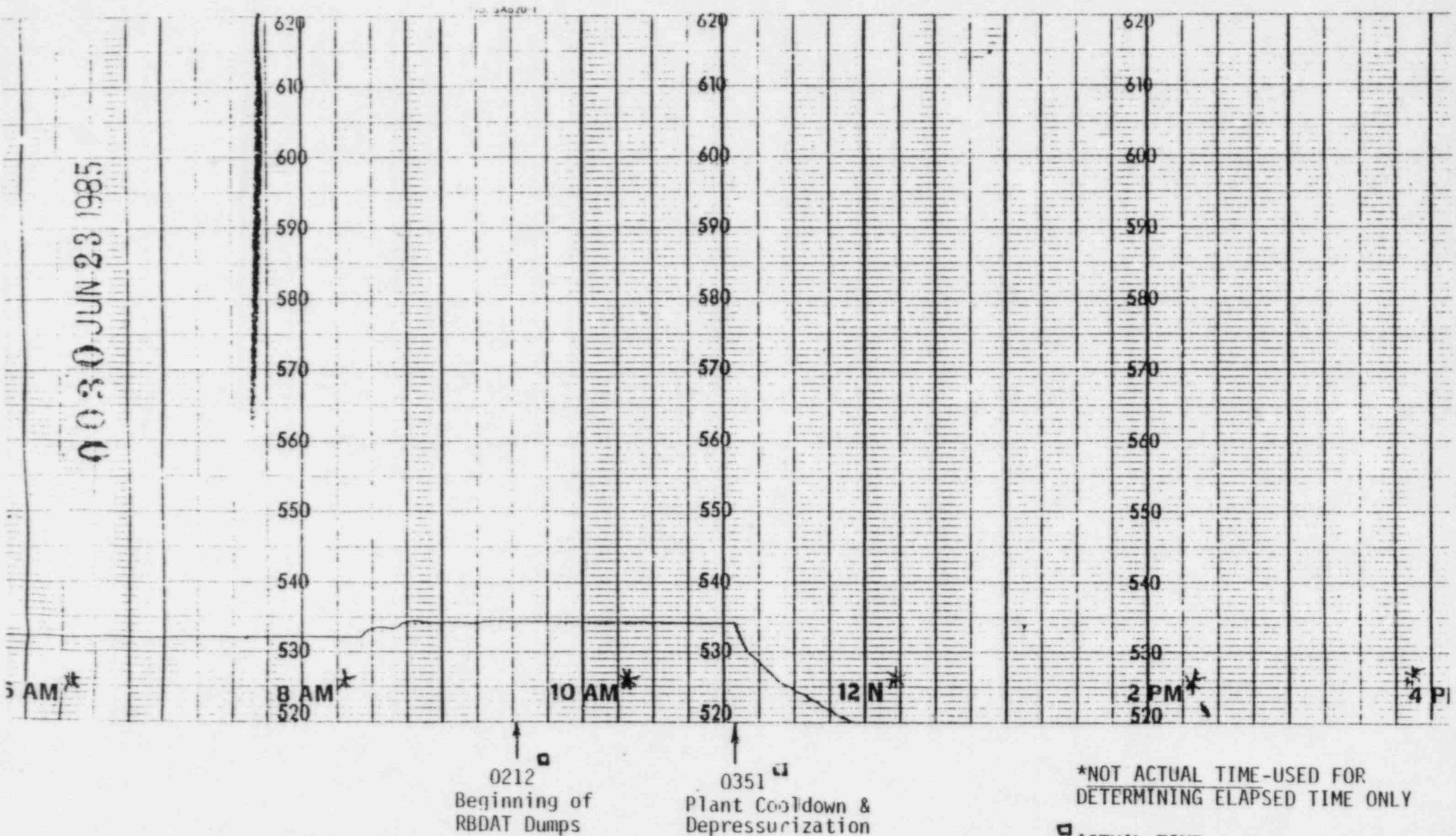
0351□
Plant Cooldown &
Depressurization

*NOT ACTUAL TIME- USED FOR
DETERMINING ELAPSED TIME ONLY

□ACTUAL TIME

ENCLOSURE 4

Rancho Seco $T_{average}$ TR-21025 June 23,1985

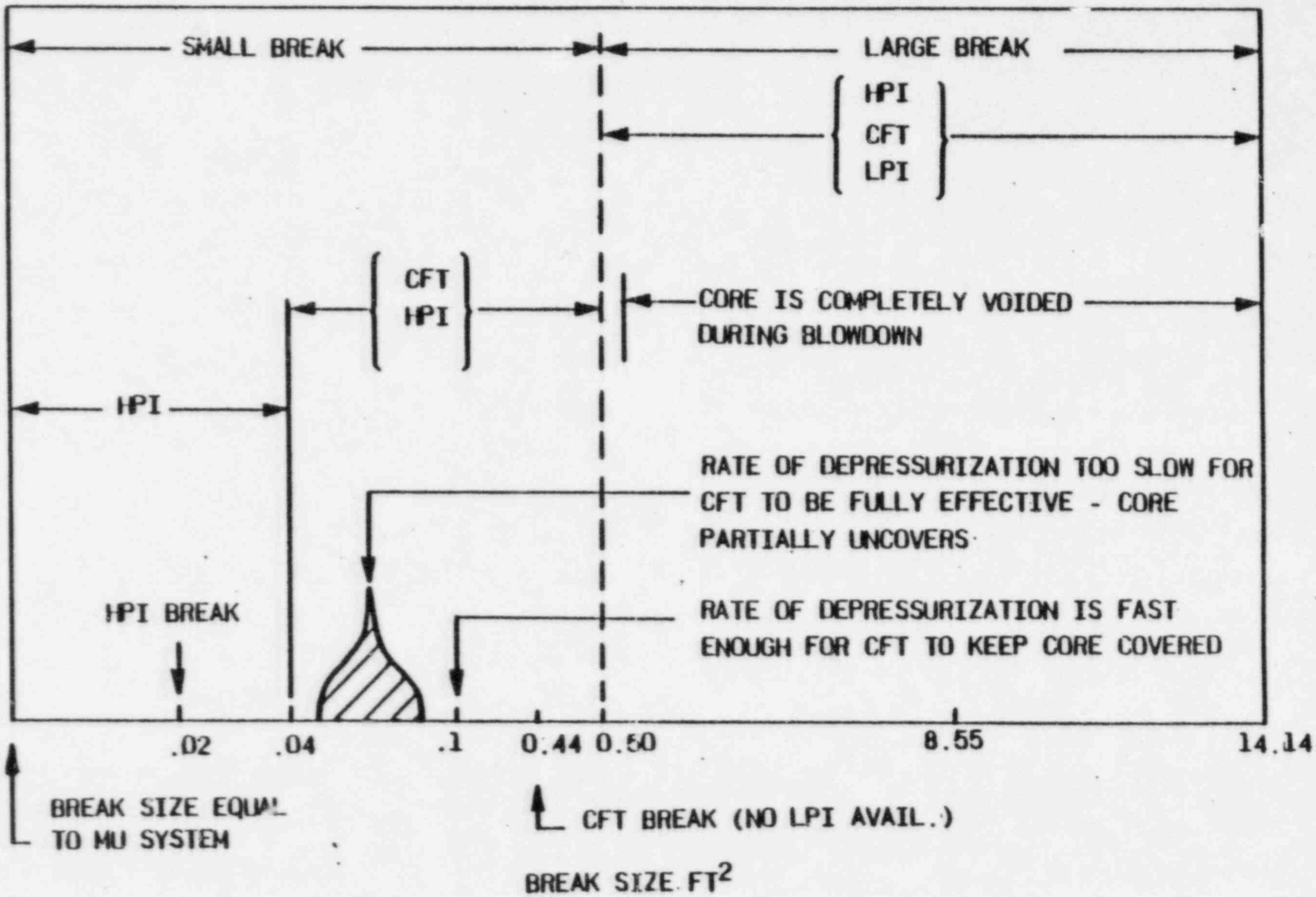


*NOT ACTUAL TIME-USED FOR DETERMINING ELAPSED TIME ONLY

□ ACTUAL TIME

ENCLOSURE 5

ECCS Vs Break Size



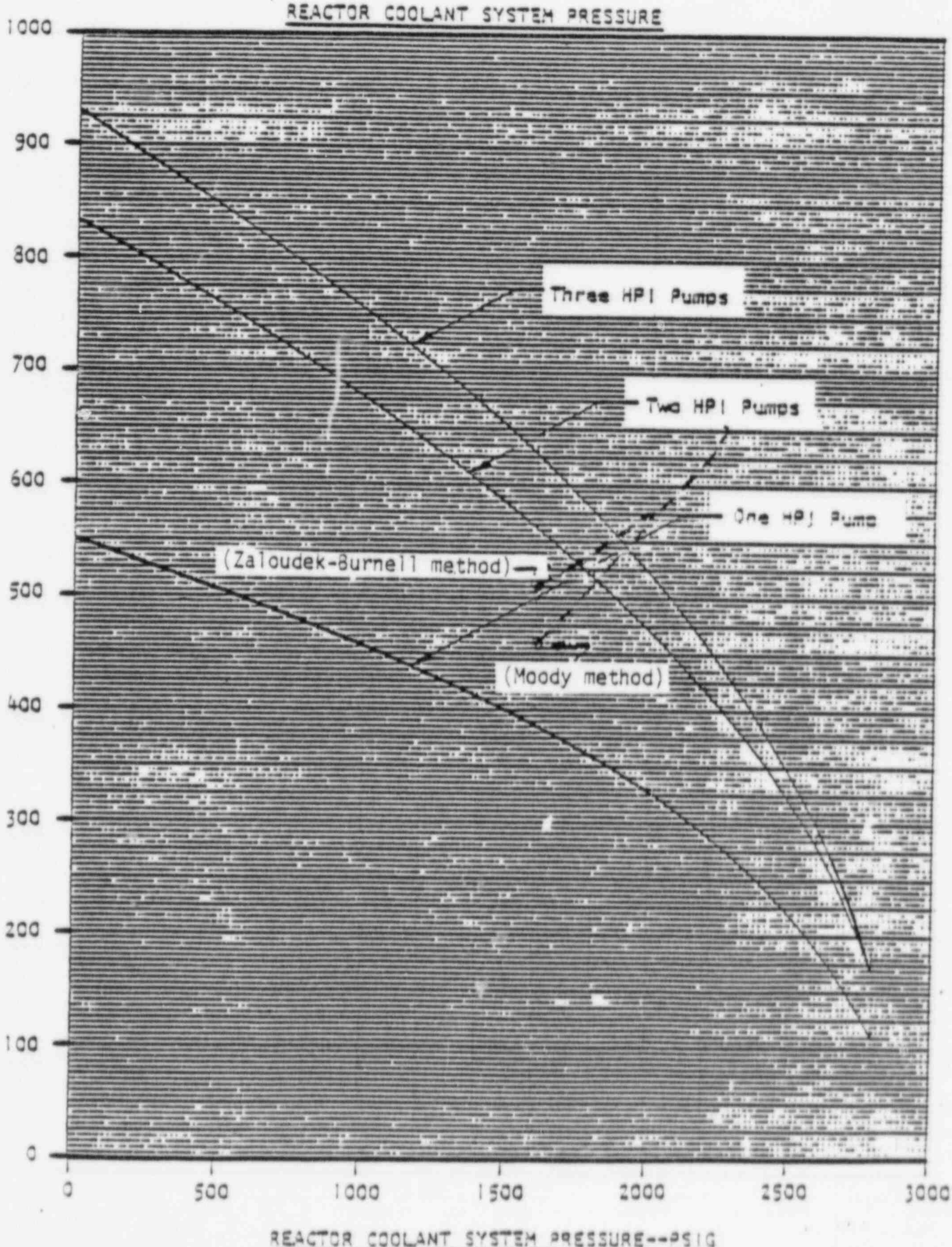
SBLOCA(177 FL)-37

ENCLOSURE 6

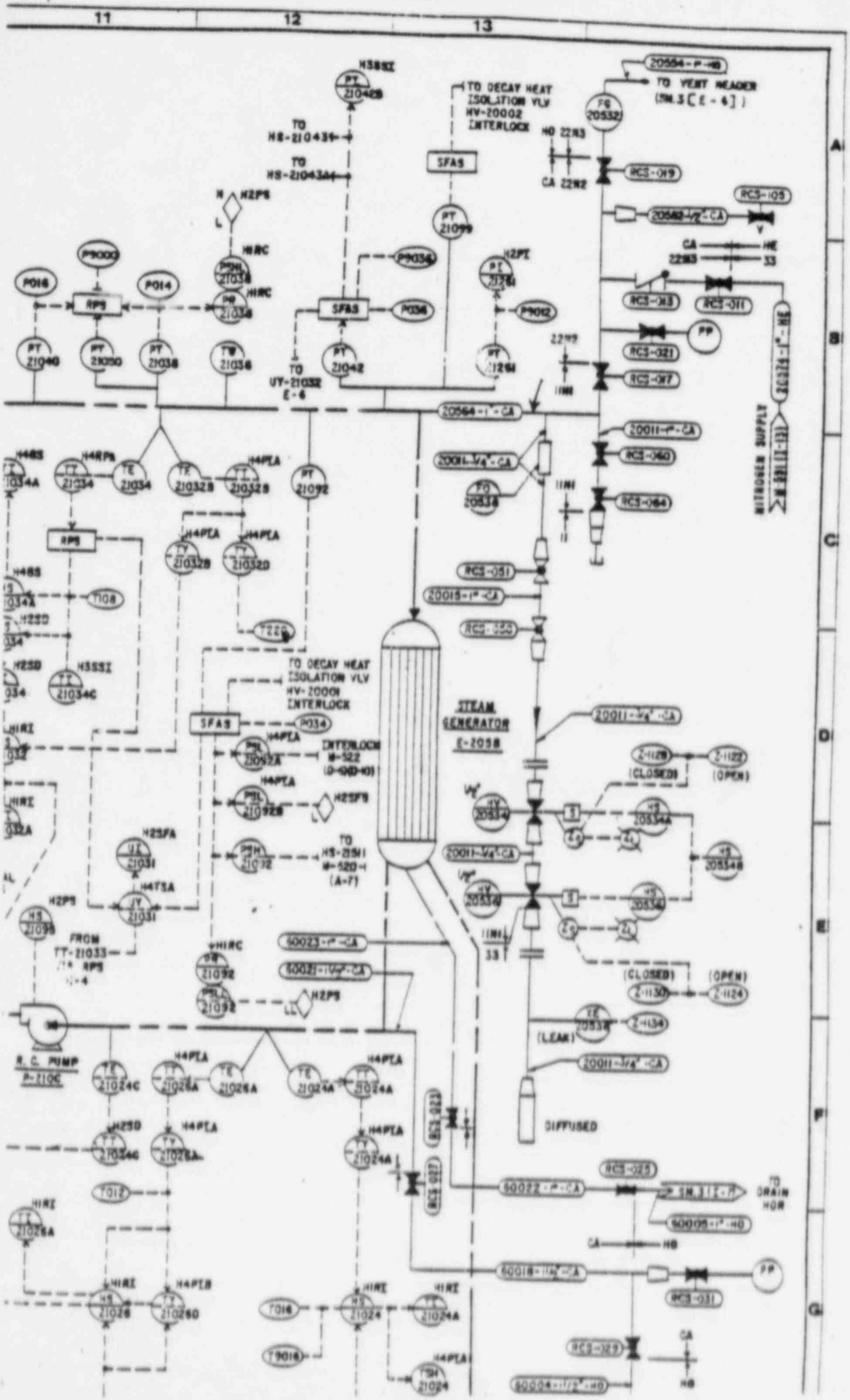
HIGH PRESSURE INJECTION FLOW

VS.

REACTOR COOLANT SYSTEM PRESSURE



ENCLOSURE 7



11
12
13
A
B
C
D
E
F
G
TO DECAY HEAT ISOLATION VLV HV-20002 INTERLOCK
TO VENT HEADER
SFAS
RPS
TO DECAY HEAT ISOLATION VLV HV-20001 INTERLOCK
STEAM GENERATOR E-2058
NITROGEN SUPPLY
(CLOSED)
(OPEN)
(LEAK)
DIFFUSED
R. C. PUMP P-210C
TO DRAIN HDR
FROM TT-21033
RCS-019
RCS-105
RCS-013
RCS-011
RCS-021
RCS-017
RCS-060
RCS-064
RCS-051
RCS-050
RCS-025
RCS-031
RCS-129

Simulator Run #1

Initial Conditions:

100% Full Power. 582°F 2155 psig
1 HPI pump not available.
NO OPERATOR ACTION FOR 10 MINUTES.

Sequence of Events/Results:

Min:Sec	Comments (Note: All times approximate)
0:58	Inserted 53.3 lbm/sec leak.
9:58	Pressurizer level at 70" and decreasing.
10:21	Reactor trip. Escalated leak to 86.6 lbm/sec.
10:28	Safety features actuation. Pressurizer level off-scale low.
11:06	Tripped RC pumps.
13:12	RCS pressure 1155 psig and decreasing.
13:59	RCS pressure reaches a minimum value of about 1106 psig.
16:00	Incore thermocouples tracing above and to the left of the variable subcooling margin curve on SPDS. "A" OTSG level at 68% on the operating range, "B" 81%. Tcold appears to be coupled with OTSG Tsat. Pressurizer level increasing.
18:15	RCS pressure 1231 psig. Incore thermocouples indicate 536°F. Tincore - Tcold equals 10°F. Two HPI/MU pumps running with 230 gpm/injection line.
22:25	RCS pressure 1310 psig. Incore thermocouples indicate 530°F. Pressurizer level 60" and increasing. Throttled down on HPI to 200 gpm/injection line.
23:04	RCS pressure 1345 psig. Incore thermocouples indicate 529°F. Pressurizer level 85" and increasing. Throttled down on HPI to 175 gpm/injection line.
23:54	Throttled down on HPI to 150 gpm/injection line.

24:27	Throttled down on HPI to 125 gpm/injection line.
25:37	Terminating transient. RCS pressure 1364 psig. Incore thermocouples indicate 531°F. Tincore - Tcold equals 16°F. OTSGs controlled at 50% on the operating range. Pressurizer level 95".

The plant was initialized at 100% full power with one HPI pump unavailable. A 53.3 lbm/sec leak was inserted. This resulted in a reactor trip 9 minutes 23 seconds later. Safety features actuated about 7 seconds after the reactor trip. Upon the reactor trip, the leak was escalated to 86.6 lbm/sec. Operator actions were not performed until after 10 minutes into the transient. Actions included tripping RC pumps, maximizing and balancing HPI flow and controlling AFW flow to the OTSGs. Subcooling margin was lost and subsequently regained about 5 minutes later. Once subcooling margin was regained, HPI was throttled to control RCS pressure and restore pressurizer level. The transient was terminated with RCS pressure, pressurizer level and subcooling margin under control. OTSG heat removal and HPI flow were being controlled in an attempt to maintain the cooldown rate to less than 100°F/hr.

Procedures Used:

Emergency Operating Procedures

- Section E.01 (Immediate Actions)
- Section E.02 (Vital System Status Verification)
- Section E.03 (Loss of Subcooling)
- CP.103 (Transient Termination Following An Occurrence That Leaves The RCS Saturated With OTSG(S) Removing Heat)
- Rules Section

Simulator Run #2

Initial Conditions:

100% Full Power. 582°F 2155 psig
1 HPI pump not available.

Sequence of Events/Results:

Min:Sec	Comments (Note: All times approximate)
0:00	Inserted 17 gpm leak. Pressurizer level about 225" initially.
5:54	Makeup valve opening. Pressurizer level about 220".
7:21	Isolated letdown.
8:36	Escalated leak to 53.3 lbm/sec.
10:10	Makeup flow alarm (160 gpm and increasing). Starting a second HPI/MU pump and opening the "A" HPI injection valve.
11:02	RCS pressure 2132 psig. Pressurizer level at 200" and increasing. Two HPI/MU pumps running with 340 gpm through the "A" HPI valve.
12:12	Reducing power.
12:20	Pressurizer level reaches a minimum of 194".
13:35	Pressurizer level at 220". Tave at 586°F - shutting down on boron.
15:06	RCS pressure reaches a minimum of 2054 psig.
15:45	Terminating transient. RCS pressure 2061 psig. Tave at 581°F. Pressurizer level at 212". Two HPI/MU pumps running with 360 gpm through the "A" injection valve. 110 gpm makeup flow. Tsat meters indicate 44°F subcooling margin. Continuing power reduction.

The plant was initialized at 100% full power with one HPI pump unavailable. A 17 gpm leak was inserted. When the leak became recognizable, letdown was isolated. At the 8 minute 36 second mark, the leak was escalated to 53.3 lbm/sec. Operator response

included starting a second HPI/MU pump and opening the "A" injection valve and makeup flow were sufficient to overcome the inventory loss out of the break. A reactor trip was prevented and a plant shutdown was in progress when the transient was terminated.

Procedures Used:

Casualty Procedures

- C.3 (Small Reactor Coolant Leak)

Plant Operating Procedures

- B.3 (Normal Operations) Section 6.0

Simulator Run #3

Initial Conditons:

Hot Shutdown. 535°F 2155 psig
1 HPI pump not available.
Decay heat: 100 hours shutdown from 100% power.
NO OPERATOR ACTION FOR 10 MINUTES.

Sequence of Events/Results:

Min:Sec	Comments (Note: All times approximate)
6:00	Inserted 86.6 lbm/sec leak.
8:55	Pressurizer level at 5" and decreasing. Tsat meters indicate 96°F subcooling margin. 160 gpm makeup flow.
9:33	Reactor trip.
10:11	Pressurizer level off-scale low. 160 gpm makeup flow. Tsat meters indicate 89°F subcooling margin and decreasing.
11:00	Subcooling margin decreasing rapidly.
11:33	Safety features actuation. Tsat meters indicate 44°F subcooling margin and decreasing rapidly. Auxiliary feedwater actuation. AFW bypass valves full open.
12:50	Loss of subcooling margin.
12:55	Incore thermocouples indicate RCS is saturated.
13:50	RCS pressure 697 psig. Two HPI/MU pumps running. All 4 HPI valves in their pre-throttled position. 170 gpm/injection line.
15:00	RCS pressure 608 psig.
15:17	Core flood tanks emptying. CFT low pressure alarm.
16:06	Commencing operator action. Shutting the AFW bypass valves and throttling the AFW control valves. Cooling down at a very high rate. Maximizing HPI flow.

16:44	RCS pressure reaches a minimum of 482 psig.
17:06	Cooling down at 530°F/hr. RCS pressure 502 psig and increasing. Incore thermocouples at 458°F. OTSG levels at about 50% on the operating range. Since RC pumps were running greater than 2 minutes after losing subcooling margin, forced circulation was maintained.
18:44	Tsat meters indicate 16°F subcooling margin.
19:24	Pressurizer level on scale at 25". Tsat meters indicate 22°F subcooling margin.
21:01	RCS pressure 636 psig. Incore thermocouples indicate 452°F. Tsat meters indicate 42°F subcooling margin. Pressurizer level at 80" and increasing. Throttling down on HPI to 200 gpm/injection line.
21:50	Terminating transient. RCS pressure 672 psig. Incore thermocouples indicate 451°F. Tsat meters indicate 46°F subcooling margin. 200 gpm/injection line with 2 HPI/MU pumps running.

The plant was initialized at Hot Shutdown with one HPI pump unavailable. Pressurizer level was initially about 120 inches. An 86.6 lbm/sec leak was inserted. This resulted in a reactor trip 3 minutes 33 seconds later. Safety features actuated about 2 minutes after the reactor trip. Upon safety features actuation, all four HPI valves opened to their pre-throttled position and one HPI pump started. The AFW system also actuated with the AFW bypass valves going full open. Without operator action for 10 minutes into the transient, a large cooldown rate was achieved. This was due to the break itself and excessive primary to secondary heat transfer. The plant depressurized sufficiently to cause core flood tanks to start emptying. Operator actions were commenced 10 minutes into the transient. These actions included maximizing and balancing HPI flow, shutting the AFW bypass valves and controlling AFW flow. Since RC pumps were running greater than 2 minutes after a loss of subcooling margin, they were left running for the duration of the transient. Subcooling margin was eventually regained. The transient was terminated when the operator had control of RCS pressure, pressurizer level and the cooldown rate.

Procedures Used:

Emergency Operating Procedures

- Section E.Ø1 (Immediate Actions)
- Section E.Ø2 (Vital System Status Verification)
- Section E.Ø3 (Loss of Subcooling)
- CP.1Ø1 (A Large LOCA Has Occurred And The Core Flood Tank Is Emptying)
- CP.1Ø3 (Transient Termination Following An Occurrence That Leaves The RCS Saturated With OTSG(S) Removing Heat)
- Rules Section

Simulator Run #4

Initial Conditions:

Hot Shutdown. 535°F 2155 psig
1 HPI pump not available.
Decay heat: 100 hours shutdown from 100% power

Sequence of Events/Results:

Min:Sec	Comments (Note: All times approximate)
6:53	Determination of 17 gpm leak.
7:24	Isolated letdown. Commenced cooldown.
8:59	Escalated leak to 86.6 lbm/sec.
9:27	Started a second HPI/MU pump. Opening the "A" HPI injection valve.
9:59	360 gpm through the "A" HPI valve. Tripped reactor manually due to 50" pressurizer level and decreasing. Casualty Procedure C.3 states to trip the reactor if pressurizer level decreases below 160". Since the initial conditions had pressurizer level at 100" we used judgement on tripping the reactor.
12:42	Pressurizer level at 25" and decreasing. Tsat meters indicate 107°F subcooling margin. 370 gpm through the "A" HPI valve with 120 gpm makeup flow.
13:14	Opening the "B" HPI valve. 270 gpm through the "A" and "B" HPI valves.
13:30	RCS pressure 1955 psig. Pressurizer level is 5" and slowly decreasing. Tsat meters indicate 107°F subcooling margin.
14:25	RCS pressure 1944 psig. Incore thermocouple temperature 524°F. OTSGs being controlled at low level limits.
15:56	RCS pressure 1937 psig and fairly steady. Pressurizer level maintaining at about 6". Two HPI/MU pumps running with 170 gpm flow through each (4) injection valve. Tsat meters indicate 109°F subcooling margin.

17:36	Depressurizing the RCS with normal spray in order to increase HPI flow. Cooling down at 92°F/hr.
20:30	Continuing to depressurize the RCS with normal spray. RCS pressure 1708 psig. Incore thermocouples 516°F. Pressurizer level increasing. Cooling down at 103°F/hr. Throttling down on HPI (180 gpm/injection line) to decrease the cooldown rate.
21:02	Pressurizer level increasing. Throttling down on HPI to 160 gpm/injection line.
21:50	RCS pressure at 1620 psig and under control. Therefore bypassing SFAS. (Safety features actuated while bypassing. Caused AFW bypass valves to open and increased the cooldown rate).
23:25	Increased HPI flow to 200 gpm/injection line. Pressurizer level increasing.
24:36	Pressurizer level increasing. Throttled down on HPI to 150 gpm/injection line.
25:24	Cooling down at 92°F/hr. Throttling down on HPI to 100 gpm/injection line. 160 gpm makeup flow.
26:43	RCS pressure 1194 psig. Incore thermocouples at 496°F. Secured the "D" RC pump.
28:19	Pressurizer heaters energized due to clearing the low level heater cutoff (80"). Secured pressurizer heaters. Continuing to depressurize with normal spray.
31:45	Terminating transient. RCS pressure 1034 psig. Incore thermocouples 487°F. Pressurizer level 100". Two HPI pumps running with 110 gpm/injection line.

The plant was initialized at Hot Shutdown with one HPI pump unavailable. A 17 gpm leak was inserted. When a leak was determined to exist, the operator isolated letdown. At the 8 minute 59 seconds mark, the leak was escalated to 86.6 lbm/sec. The operator started a second HPI/MU pump and opened the "A" HPI valve. The reactor was manually tripped about one minute after the leak was escalated. Tripping the reactor was based on not being able to maintain pressurizer level with two HPI/MU pumps running and the "A" injection valve full open. Casualty Procedure C.3 states to trip the reactor if pressurizer level cannot be maintained above 160 inches. Since the initial conditions had pressurizer level at about 100 inches, a judgment had to be made concerning when to trip the reactor. Pressurizer level and subcooling margin were never lost during this transient. However, the operator had to depressurize the RCS with normal spray in order to increase the available HPI flow and decrease the leak rate. The actuation of safety features while bypassing this system caused some minor control problems. With the AFW system actuating, the RCS cooldown rate increased. System contraction was compensated for by increasing HPI flow until AFW flow was controlled. The transient was terminated when the operator had control of pressurizer level, RCS pressure and temperature.

Procedures Used:

Casualty Procedures

- C.3 (Small Reactor Coolant Leak)

Emergency Operating Procedures

- Section E.01 (Immediate Actions)
- Section E.02 (Vital System Status Verification)
- Rules Section

Plant Operating Procedures

- B.4 (Plant Shutdown and Cooldown) Section 5.0