UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OFFICE OF THE
COMMISSIONER

April 4, 1979

MEMORANDUM FOR: Edson Case, NRR
John Davis, I&E

FROM: John Ahearne

SUBJECT: RESTART OF DAVIS-BESSE UNIT 1

Attached for your information and review are copies of two analyses of the September 24, 1977, incident at Davis-Besse, one by Babcock and Wilcox, the other apparently by the licensee. I note that the B&W paper is labelled "Advance Copy," so there may have been a later version. I would appreciate any comments you may have on these analyses, particularly how they may relate to the Three Mile Island accident.

I understand that I&E has requested that Metropolitan Edison not restart Davis-Besse Unit 1 without first notifying I&E, and that this restart is tentatively scheduled for sometime in the April 9-10 time frame. Since the reports from the operators of B&W plants called for in the IE Bulletin 79-05, Nuclear Incident at Three Mile Island, sent on April 1 are due at about that same time, I strongly recommend that Metropolitan Edison be requested not to restart Unit 1 until NRC has received and reviewed these responses for whatever implications they may have on further operation of Unit 1.

Attachments

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
SECY
EDO

95004014

7906230093

Of these two transients the loss of feedwater results in the greater volumetric coolant contraction, because the forced coolant flow (RC Pumps operating) causes a faster rate of heat rejection to the steam generator.

1. Loss of Offsite Power

Preliminary calculations for a reactor trip following a loss of offsite power show that the pressurizer loses indication but does not empty. The assumptions used to derive this result included full runout auxiliary feedwater flow (~2400 gpm) resulting in a fill time to 120" of about 4 minutes. No net mass change to the primary coolant (no makeup, no letdown) was considered, even thou the makeup controls would respond to decreasing pressurizer level by increasing the net input to above 200 gpm. At the terminatio of the transient the pressurizer level is slightly above the outl into the surge line. Reactor coolant pressure reaches about 160C and high pressure injection may be automatically initiated.

Although the net makeup was not considered, it would in fact caus the pressurizer to refill to the normal level. At the same time compression of the steam would cause a partial repressurization o the system ensuring that the coolant remains subcooled. This tra presents no safety concerns.

2. Loss of Feedwater

This transient has a greater reactor coolant contraction than th loss of offsite power case, resulting in emptying of the pressur Consequently it will be described in greater detail.

A brief summary of the events is:

• Reactor trip	Time = 0
• Makeup control valve opens wide admitting full makeup to reactor coolant system	Time = 0^+
• AFW initiated	Time ≈ 40 se
• Pressurizer empties; RC system pressure slightly greater than 1800 psi	Time ≈ 2 min
• HPI initiated by SFAS; makeup isolated	Time ≈ 2^+_- mi
• Steam generator level = 10 ft; voids exist in reactor coolant	Time ≈ 4 min
• HPI inflow replaces volume occupied by voids; pressurizer level begins to be restored	Time ≈ 7-8 mi

The major concerns that evolve from this transient are the disposition of the steam voids and the approach to DNB. Both of the concerns are ameliorated by the reactor coolant pumps.

Reactor coolant pumps turned off @ TMI also, HPI turned off

95004015

Steam voids will not collect in reactor coolant piping and no flow blockage will occur because of dispersal and mixing by the forced flow. DNB acceptance criterion limit will be met because the power output of the core is at the decay heat level and all reactor pumps are operating, maintaining core heat removal. We conclude that no safety problem exists.

TABLE 1: STEAM AND FEEDWATER LINE RUPTURE CONTROL SYSTEM (SFRCS) ACTUATION PARAMETERS

Actuation Parameter		Accident
Station Variables	Setpoint	
1. Low Steam Line Pressure	$<$ 591.6 psig[1,2]	Steam Line Break Feedwater Line B
2. Low SG Level	$\leq$ 17 inches[1]	Loss of F/W
3. SG Pressure Minus Main Feedwater Line Pressure	$>$ 197.6 psi[1]	FWLB, LOMFW
4. Loss of All RC Pumps[3]		Loss of Off-Site

NOTES:

1. When actuated, SFRCS closes both main steam isolation valves, closes bo main FW control and stop valves, initiates AFW and controls AFW to main a 120 inch level in the SGs.

2. Alignment of AFW to a pressurized SG is provided for steam and feedwate line breaks.

3. AFW initiation but steam and feedwater line isolation does not occur.

95004016

III. Bounding Analysis of Loss of Feedwater Event With Failure of Operator to Control Feedwater Level at 35"

Introduction:

The following bounding analysis conservatively predicts the events occurring within the primary reactor coolant system and reactor following a loss of main feedwater from 100% power for the Davis-Besse Unit 1. Auxiliary feedwater control has been assumed at 10 feet within both steam generators.

Results:

Because of the conservative, bounding, nature of this calculation, the overcooling of the primary system due to auxiliary feedwater injection causes a contraction of coolant volume sufficient to create steam within the primary system. The steam is shown to be uniformly distributed within the RCS and the void fraction is 4%. The reactor coolant pumps maintain full capability. The DNB ratio is shown to exceed 2.0 and no return to criticality potential exists. Thus, during the course of the incident, no core problems develop. Further, following the time of maximum contraction, the system recovers to full pressure, pressurizer function is regained and the reactor coolant returns to a subcooled water configuration without operator action.

Analysis:

The following assumptions have been made to assure the bounding nature of the results:

Reactor Power:

100% until boiling stops in the steam generators; 0% after that time. This assumption is conservative as core heat would compensate for the cooling caused by the auxiliary feedwater.

Initial Coolant Inventories Water:

RCS = 11290 ft^3

Pressurizer = 864 ft^3

These assumptions are nominal operating values.

Initial Temperatures:

The whole system is taken to be at $T_{average} = 582^{\circ}F$.

This assumption is a reasonable average.

Initial System Mass: ~ 500,000 lbm

The mass is figured from the temperature and volumes above.

95004017

Makeup System:

No credit is taken for additional makeup flow which will occur as the pressurizer loses level., (In all likelihood, the makeup system will contribute approximately 200 ft^3 extra liquid volume).

Local Power (kw/ft): 18.4 kw/ft

This value is taken as the maximum allowed by Technical Specifications.

Secondary Side Volume At 10 Foot Level

711 ft^3 per generator, actual volume.

Auxiliary Feedwater Flow:

166.5 ft^3/min. per generator actual value.

Auxiliary Feedwater Enthalpy:

8 Btu/lbm lower bound for maximum cooling.

With the initiating event, loss of main feedwater, the reactor coolant system pressure will start to rise. Reactor trip will occur on high RCS pressure. Following trip, the RCS pressure will fall because core power has been reduced and boiling of residual main feedwater or auxiliary feedwater is occurring in the steam generators. These events are almost identical to those which occur in a main feed line break and are analyzed in detail in Section 15.2.8 of the FSAR.

In short order, the system will return to its initial configuration but, because the auxiliary feedwater heat absorption rate exceeds the decay heat generation rate, the RCS continues to depressurize. During this phase, residual main feedwater and injected auxiliary feedwater will be boiled and vented through the steam generator safety relief valves. The primary system average temperature will fall to the saturation temperature of water at the safety valve set pressure. At this time, primary and secondary conditions are expected to be approximately as follows:

	Primary	Secondary
Pressure	1800 psia	980 psia
Temperature	542 F	542 F
Mass	503344 lbm	0 lbm
Liquid Volume in Press.	400 ft^3	N.A.
Time into Transient	~ 2 min.	~ 2 min.

95004018

It is conservative to assume complete boiling of the secondary side water and complete equilibrium between primary and secondary sides, as these assumptions lead to the maximum follow on injection of auxiliary feedwater and therefore, maximum contraction. RCS pressure is held up by the steam bubble in the pressurizer.

The time has been estimated by calculating the necessary energy loss by the primary system from its initial conditions, the mass of auxiliary feedwater required to remove this energy and then dividing by the auxiliary feedwater flow rate.

$$\text{time} \simeq \frac{(586 - 542)\ 503344}{(1194-8)\ \ 353\ \ 62} \simeq 54 \text{ sec.}$$

Six seconds was used to estimate the initial pressurization portion of the transient.

In performing the remainder of the evaluation 10 feet of cooled (40 F) auxiliary feedwater is placed in each steam generator and the thermal equilibrium condition calculated. Because after a 10 foot level is obtained this auxiliary feedwater flow stops, this condition represents the maximum contraction possible. The state variables resulting are:

	Primary	Secondary
Pressure	560 psia	560 psia
Temperature	478 F	478 F
Enthalpy of Water	462 Btu/lbm	462 Btu/lbm
Specific Volume	.020 ft^3/lbm	.020 ft^3/lbm

From the specific volume, the primary liquid volume can be calculated:

$$Vol = MV_f = 10052 \ ft^3$$

As 10052 is smaller than the RCS minus pressurizer volume, the remaining volume must be filled with steam.

$$V_{st} = 10426 - 10552 = 374 \ ft^3 \cong 400 \ ft^3$$

400 ft^3 corresponds to a system void fraction of 3.8% $\simeq$ 4%, and as will be shown later, is of no consequence as far as core heating or system performance is concerned. This steam volume is larger than actually expected for two reasons: 1) some temperature difference would always exist between the primary and secondary systems, and 2) the effect of core decay heat has been ignored. Both of these would increase the primary side liquid temperature, thus increasing its volume and reducing the steam volume.

Following this state of maximum contraction, no further heat is removed from the RCS via the secondary side until the RCS rises in temperature due to decay heating; this will expand the liquid volume, compress the steam and repressurize the RCS. As no mass can be lost from the secondary system prior to achieving 980 psia the first reheating stage will end at a primary system pressure, temperature, and liquid volume of 980 psia, 542 F, 10832 ft^3. Subtracting 10426 from 10832 shows that about 400 ft^3 of fluid has been forced back into the pressurizer. Pressurizer function would then be restored (if not directly, then, by either the makeup or HPI system), the RCS subcooled and the transient ended.

Several questions exist about the transient:

I. How is the 400 ft^3 dispersed within the primary system and can that volume collect in one location? From the auxiliary feedwater flow rate, over 4 minutes are required to fill the generators. As the pressurizer has 400 ft^3 in it at 980 psia and the RCS has 400 ft^3 in it at maximum contraction, approximately 2 minutes are used to eject steam from the pressurizer to the RCS. Because this steam will be superheated when it enters the RCS it will first desuperheat and then condense at a rate governed by its expanding pressure compared to the contraction of the liquid coolant. In the time of 2 minutes the reactor coolant will have made about 8 complete circles of the primary system and the steam can be considered well mixed. As the flow velocity in the RCS will remain normal, about 25 ft/sec, steam water separation will tend not to occur. Some limited steam accumulation may occur in the upper head of the reactor vessel as in that specific location of the RCS, velocity is low.

II. How well will the pumps work? Experiments performed on steam carry over capability show that for void fractions up to 10% no loss of pump capability is observed. This is documented in Figure 5-47 of BAW-10104, "B&W's ECCS Evaluation Report With Specific Application to 177 FA Class Plants With Lower Loop Arrangement." Actually pump capability increases for the first 5% of void introduced into the system.

III. Will any return to power be encountered because of the low RCS temperature? A return to power can occur for a non-borated core at 490F. This temperature includes the assumption of the most reactive rod stuck out of the core; if that rod were taken as inserted the critical temperature would fall to at or below 400F. Although no credit was taken for HPI in calculating the RC steam volume below 1600 psia, the HPI will be injecting borated water and, therefore, preventing any return to power condition. If the primary system were to stabilize at 1600 psia and thus prevent the HPI from providing boron the RCS temperature would be at least 511F and, therefore, no return to power would be expected.

IV. Will DNB be encountered in the core? The maximum contraction condition is again:

P = 560 psia
T = 478F
a = 4%,

95004020

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 2450[illegible]

Telephone: (804) 384-5111

October 28, 1977

BWT-1589

File: T1.2/12B

cc: J. D. Lenardson
J. C. Lewis
D. J. DeLaCroix
P. P. Anas/4c
E. C. Novak/1c

RECEIVED NOV 03 1977

Mr. C. R. Domeck
Nuclear Project Engineer
Toledo Edison Company
Power Engineering & Construction
300 Madison Avenue
Toledo, Ohio 43652

(Check these against final copy from Alice) 11/3: ADVANCE Copies to

ADVANCE COPY

Subject: Toledo Edison Company
REPORT ON DEPRESSURIZATION EVENT
Davis-Besse Unit 1
B&W Reference NSS-14

Dear Mr. Domeck:

By telecon of October 10, you have requested B&W input for a report to NRC regarding the depressurization event of September 24. The NRC exit interview notes dated October 7 summarized the necessary content of the report. B&W is providing write-ups in the following areas in order to substantiate the conclusions of BWT-1573 and BWT-15[illegible] dated October 5 and 7:

Report Section

4A A. Description of the event

4B(1) B. Evaluation of the reactor coolant components

4B(2) C. Evaluation of RC pumps

4B(3) D. Evaluation of the fuel

In order to expedite submittal of your report, we are sending Sections A, C and D at this time, as agreed in our telecon of October 24. We expect to forward Section B by November 7, and we will try to improve on this date.

Section A describes the sequence of events as reconstructed from computer alarm printout, reactimeter plots, and control room recorders (Attachment A.1). We have attached pertinent recorder charts of T_{ave}, RC pressure, pressurizer level (Attachments A2, A3 & A4) and reactimeter plots of RC inlet temperature, RCS flow in each loop, RC pressure, pressurizer level, and water level and outlet pressure of each steam generator (Attachments A5 through A13).

Section B will include evaluations of stresses in the pressure boundary, the depressurization transient, boiling the SG dry, jet impingement on the SG, and effect upon fatigue life.

95004021

Babcock & Wilcox

October 28, 1977
BWT-1589

Section C explains the evaluation which was performed to verify that there was no significant damage to RC pump bearings, seals, or impellers (attachment C1). The transient as it affected the pumps is summarized in Attachment C2. Attachment C3 defines the instrumentation and operational checks applied to the pumps. The results of the operational checks are tabulated in Attachment C4.

Section D evaluates the effect upon the core to determine (1) whether steam was produced in the core (2) the maximum internal fuel rod pressure, and (3) whether maximum lift force exceeded the limit (Attachment D.1). Reactimeter plots are attached for reference Attachments D.2 through D.6.

Very truly yours,

A. H. Lazar
Senior Project Manager

J. A. Lauer
Project Manager

JAL/hj

Attachments

95004022

Sequence of Events

The event started at time 21:34:20 on September 24, 1977. The plant was in Mode 1 with Power (MWT) = 263. The turbine had been shutdown earlier in the evening to repair a leak in the main steam line at an instrument connection between the turbine stop valves and the high pressure turbine. At this time a half trip of the Steam and Feedwater Rupture Control System (SFRCS) was initiated by an unknown cause. This trip shut the startup feedwater valve to #2 steam generator and stopped all feedwater to this generator (because of the low power level the main feedwater block valve was already shut, isolating the main feedwater control valve). The low level alarm was reached in #2 steam generator at 21:34:44. Before the operator could identify and correct the problem, the low level in #2 steam generator produced a full trip of the SFRCS. This trip shut the main steam isolation valves and feedwater isolation valves in both steam generators (time 21:35:18). SFRCS also started both auxiliary feedwater pumps. The number one pump performed as intended, however, number two auxiliary feedwater pump only came up to 2600 RPM, insufficient to feed its steam generator (#2).

The loss of feedwater, first to one and then both steam generators, caused an increase in primary water temperature, which resulted in an increase in pressurizer level and thus reactor coolant system pressure. At 2255 PSIG the pressurizer electromatic relief valve received an open signal. During the next 40 seconds, it received nine different open and close signals. After one of those signals the valve stuck open. This provided a continuous 2½" vent path from the pressurizer to the quench tank. When pressurizer level got to 290", the operator manually tripped the reactor (time 21:36:07). Energy escaping from the electromatic relief valve and three main steam relief valves caused a rapid cooldown and depressurization of the reactor coolant system. Reactor coolant system pressure dropped to 1600 PSIG (time 21:37:17) initiating the Safety Features Actuation System (SFAS). This started high pressure injection and closed numerous containment isolation valves, including the quench tank cooling lines.

With the electromatic relief valve still open and cooling water isolated to the quench tank, the quench tank rupture disc ruptured (time 21:40) relieving water/steam to the containment building. This discharge damaged a nearby ventilation duct, was deflected off this duct and directed onto #2 steam generator. The steam tore off approximately a 10' high x 20' circumferential section of insulation from #2 steam generator. The paint from the then exposed area of the steam generator was blasted away. ~~Other indications of systems interaction due to~~ the steam in the containment include two fire alarms (one near RCP 2-2 and one near the pressurizer) and a single channel RPS trip on high reactor building pressure (4 PSIG).

When the main steam relief valves reseated the decrease in reactor coolant system temperature stopped and the high pressure injection pumps started to raise pressurizer level. At time 21:40:34 the operator stopped the high pressure injection pumps. (The operators had been heavily involved before this time in regaining seal injection flow to the reactor coolant pumps. This flow had been stopped by the SFAS actuation. By 21:39:40 the appropriate SFAS signals had been overriden and normal flows restored to the seals of the pumps). Reactor coolant system pressure continued to decrease until saturation pressure was reached and steam began to form in the RCS (approximate time 21:42). This caused an insurge of water into the pressurizer and pressurizer level went off scale high at 320 inches. During this level increase the operator, seeing average reactor coolant system temperature and pressurizer level increasing, stopped one reactor coolant pump in each loop (time 21:43:11).

95004023

Due to decreasing pressure in #2 steam generator, the SFRCS system gave a low pressure block permit signal at time 21:48:33. This alerted the operator to the low level and feed condition of #2 steam generator. He blocked the low pressure trip (time 21:49:38), took manual control of the speed of #2 auxiliary feedwater pump and fed #2 generator (time 21:50). The operator saw the rapid addition of cold feedwater dropping the reactor coolant system temperature and stopped the feedwater addition to this generator.

At approximately 21:55 the operator shut the block valve for the electromatic relief valve on the pressurizer and stopped the venting of the reactor coolant system to the quench tank. At 22:05 pressurizer level came back on scale. At 22:15 the operator started a second makeup pump to try and stop the pressurizer level decrease. This additional cold water started the reactor coolant system on a slow decreasing temperature transient. At 22:17 pressurizer level reached the low level interlock and cut off the pressurizer heaters. At 22:23 the operator started a high pressure injection pump to try and stop the decreasing pressurizer level.

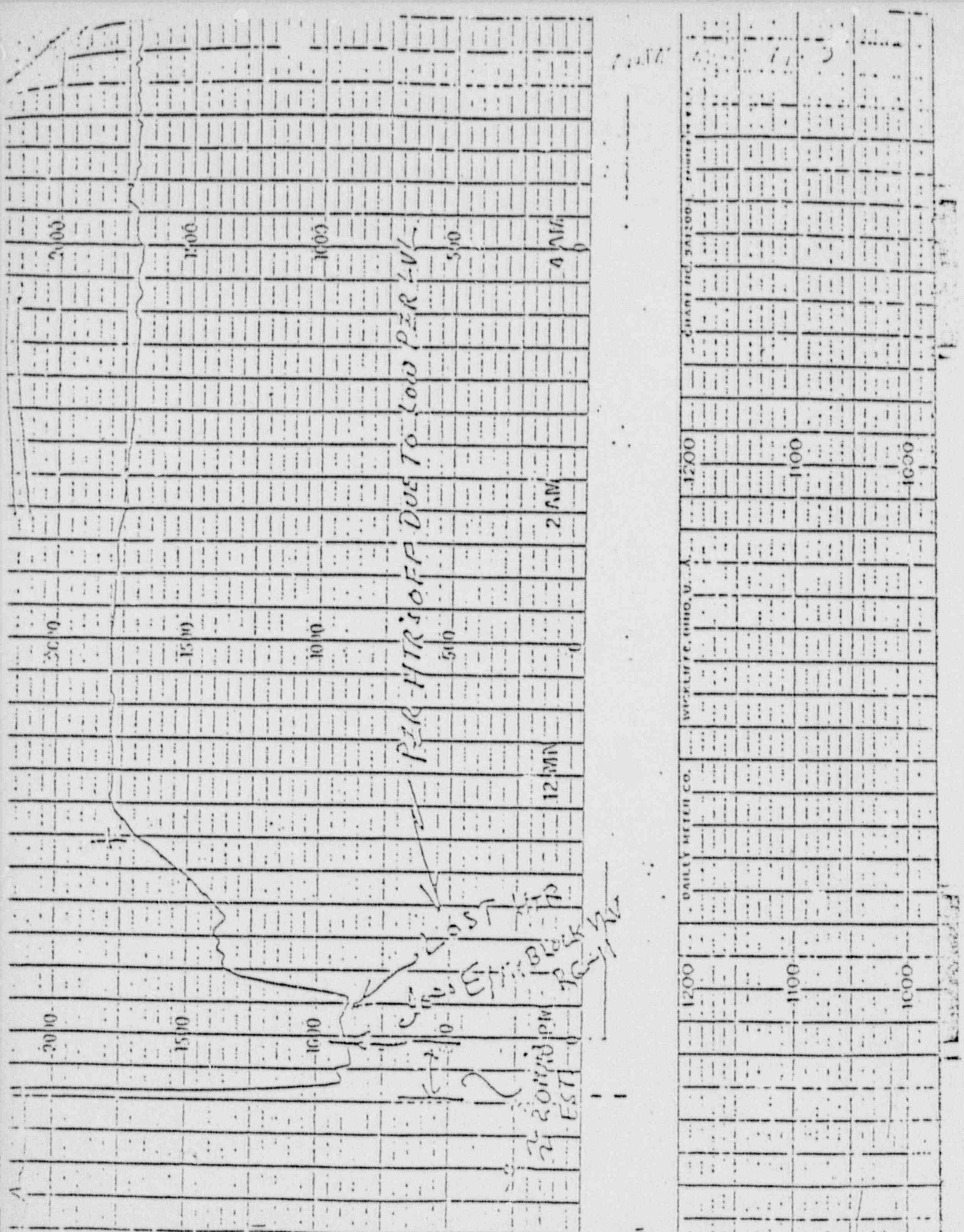
The ~~level and~~ pressure in #2 steam generator again decreased to the point where the SFRCS gave a low pressure block permit signal. The operator again blocked the trip and, through manual speed control of its auxiliary feedwater pump, restored level and pressure in #2 steam generator (time 22.25).

With pressurizer level well on its way to recovering, the operator stopped the high pressure injection pump (time 22:27:44). At time 22:31 he restored RC makeup flow to normal. This stopped the slow decreasing RC temperature transient started at time 22:15. All plant parameters were now fully under control and the plant was brought to a steady state condition and a normal plant cooldown started.

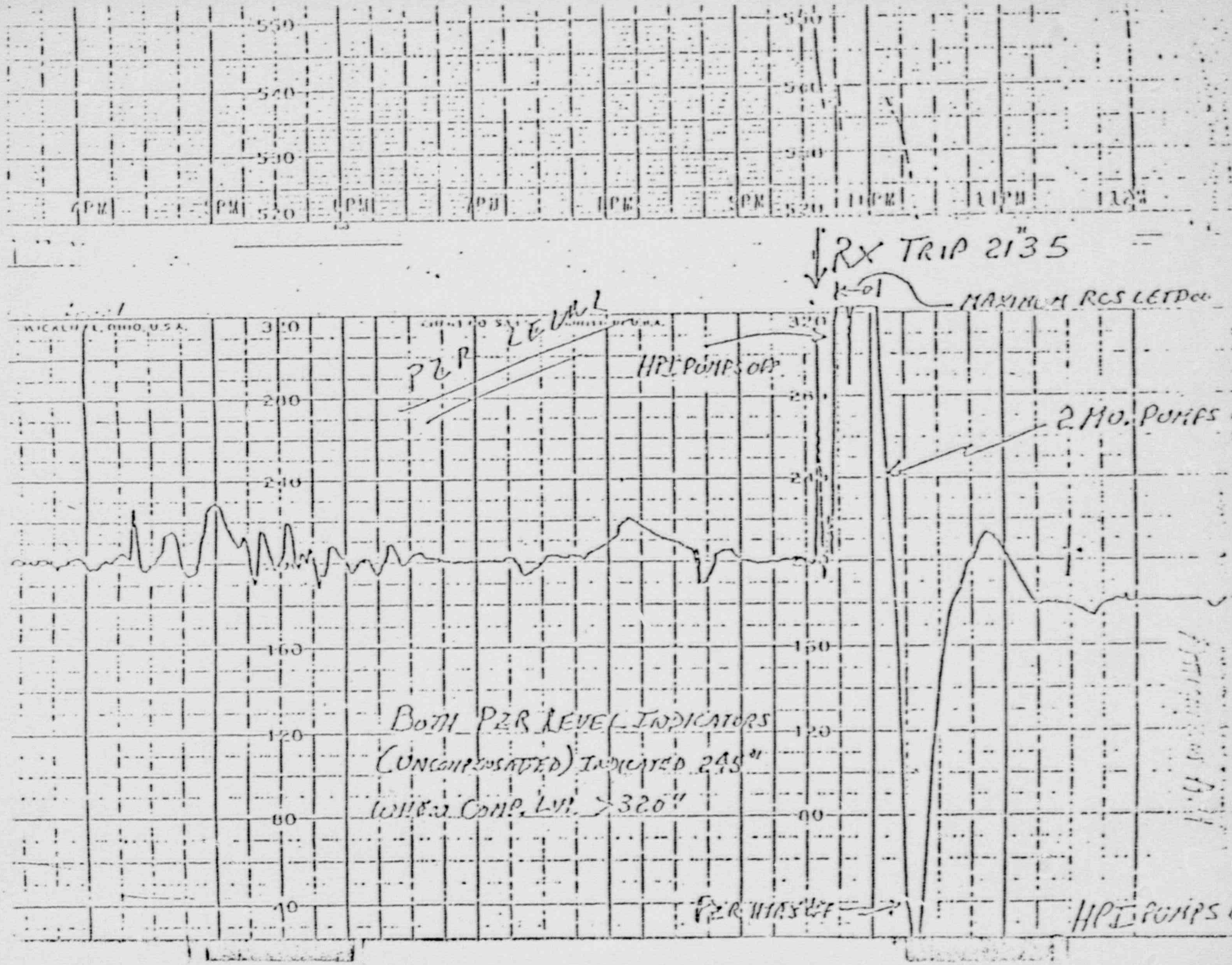
95004024


Unit Tave
Rx Trip (Manual Trip)
Rx Trip ≈ 2136
505°F Lowest Temp
610
600
590
580
570
560
550
540
530
520
9PM
10PM
11PM
12M
1AM
2AM
320
280
240

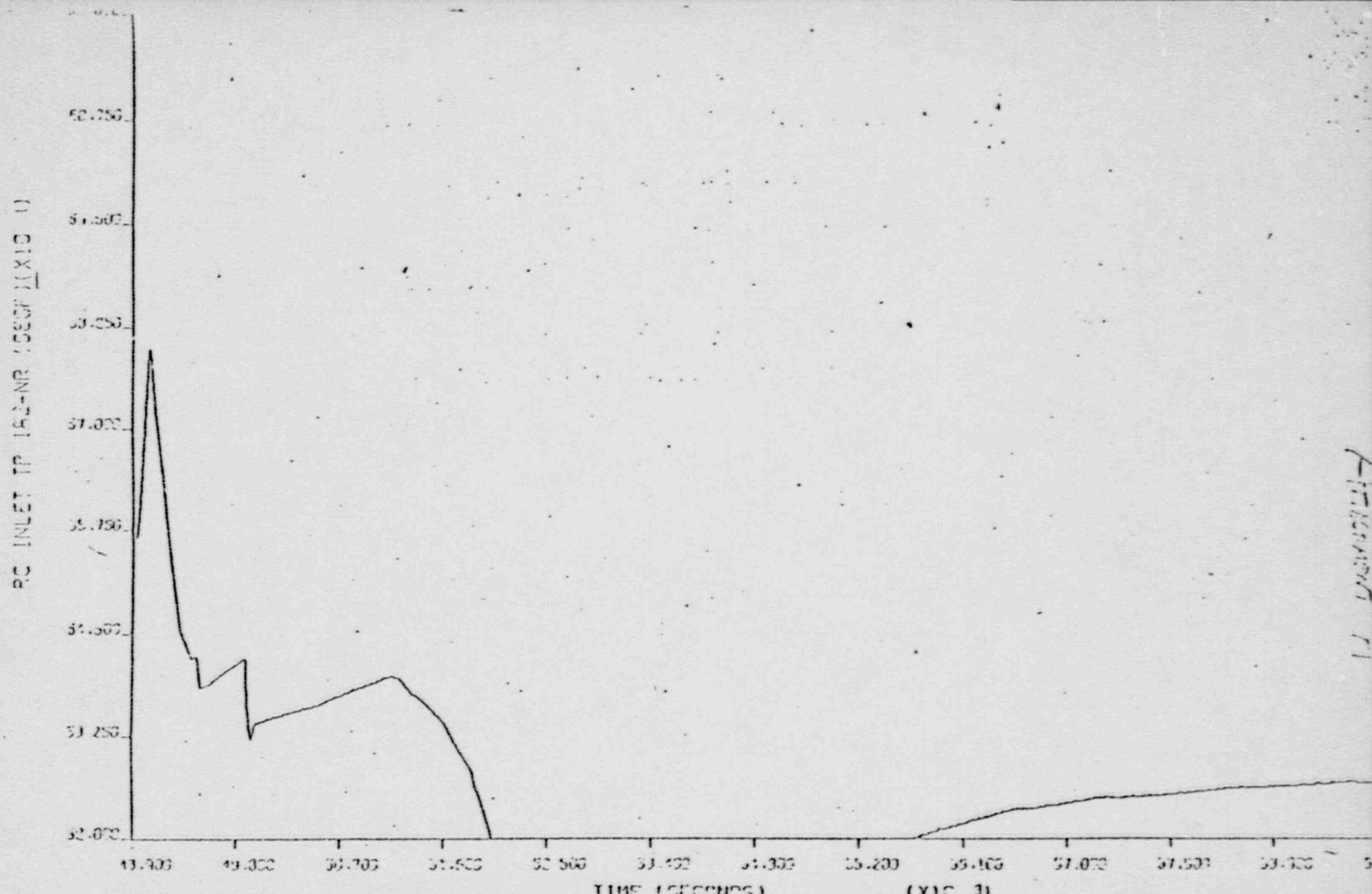
95004025


PZR HTR'S OFF DUE TO LOW PZR LVL
2000
1500
1000
500
0
4 AM
2 AM
12 MN
1200
1100
1000
BAILEY METER CO.

95004026


RX TRIP 2135"
MAXIMUM RCS LETDOWN
PZR LEVEL
HPI PUMPS OFF
2 MU. PUMPS
BOTH PZR LEVEL INDICATORS
(UNCOMPENSATED) INDICATED 245"
WHEN COMP. LVL >320"
PZR INLET
HPI PUMPS

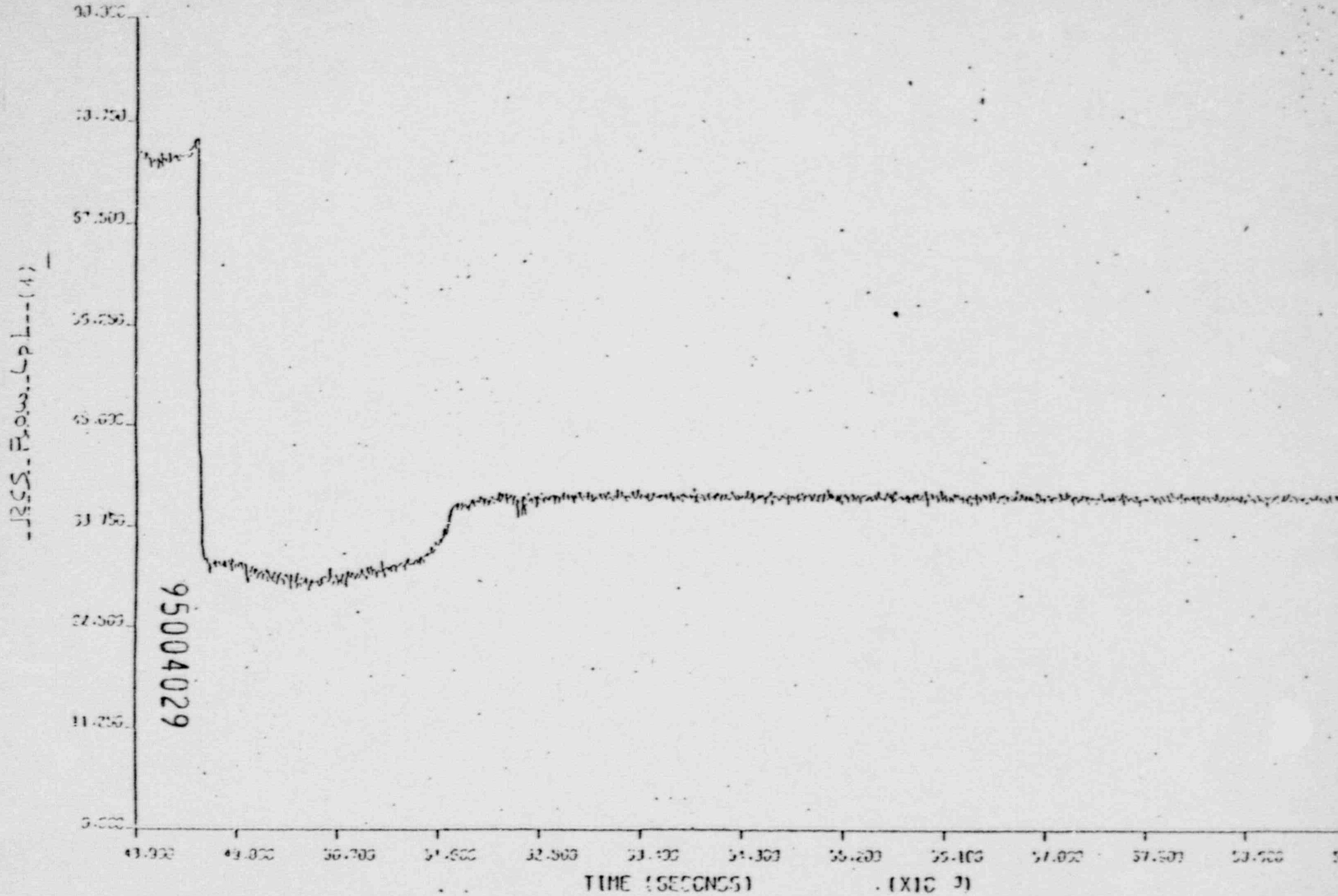
95004027



REACTIMETER PLOT   TSN=71

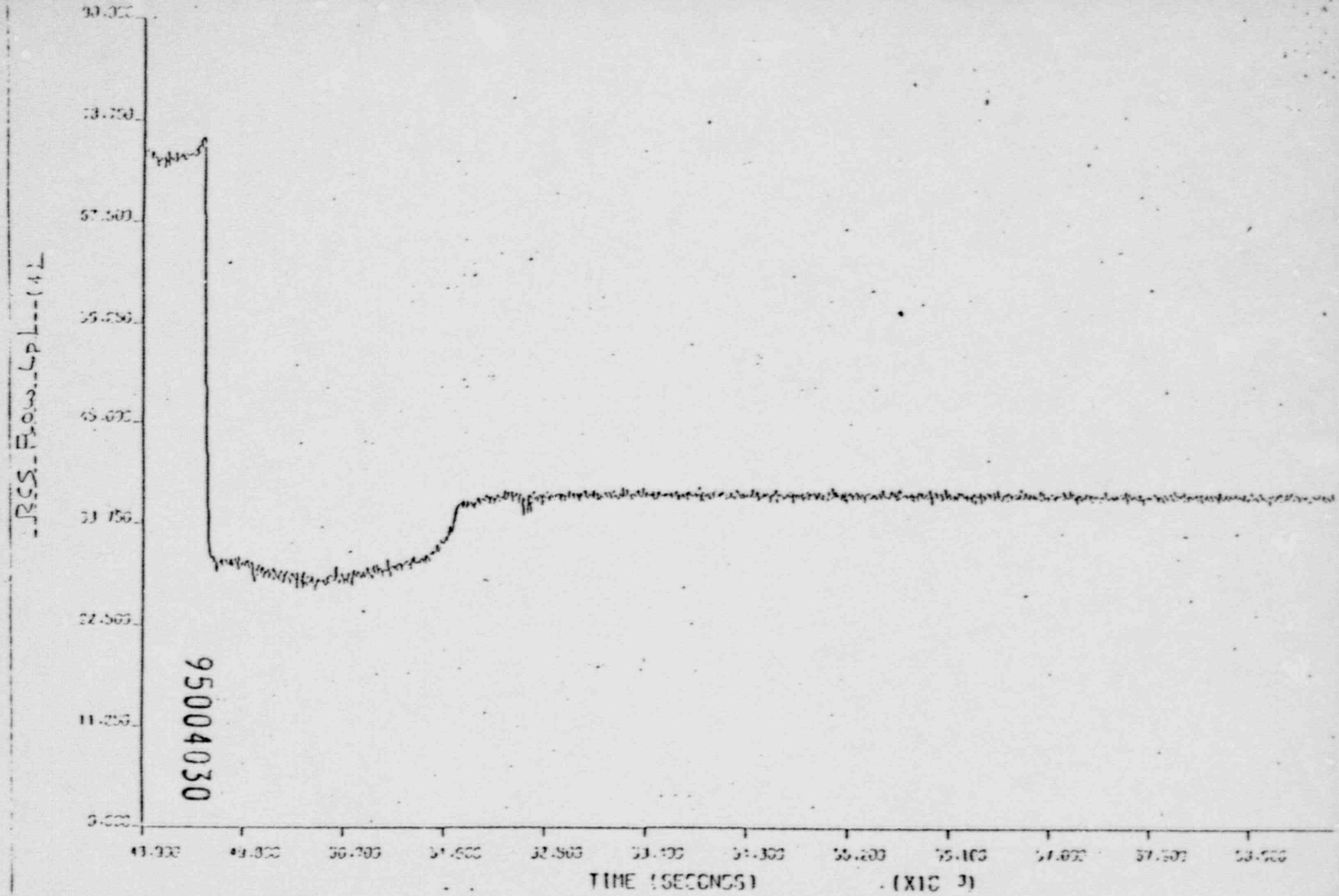
Figure 4-2

95004028



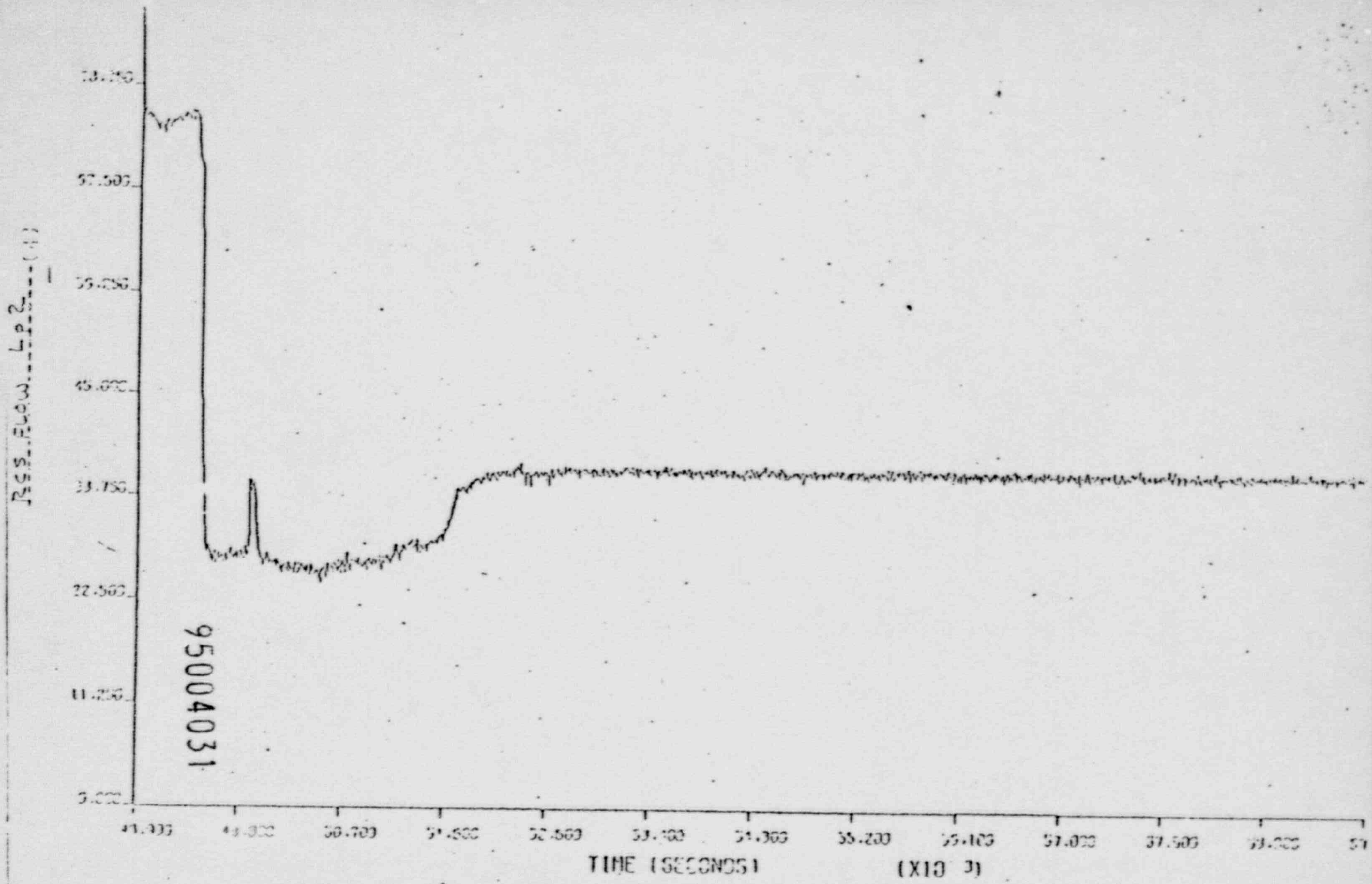
REACTIMETER PLOT TSN=71

95004029

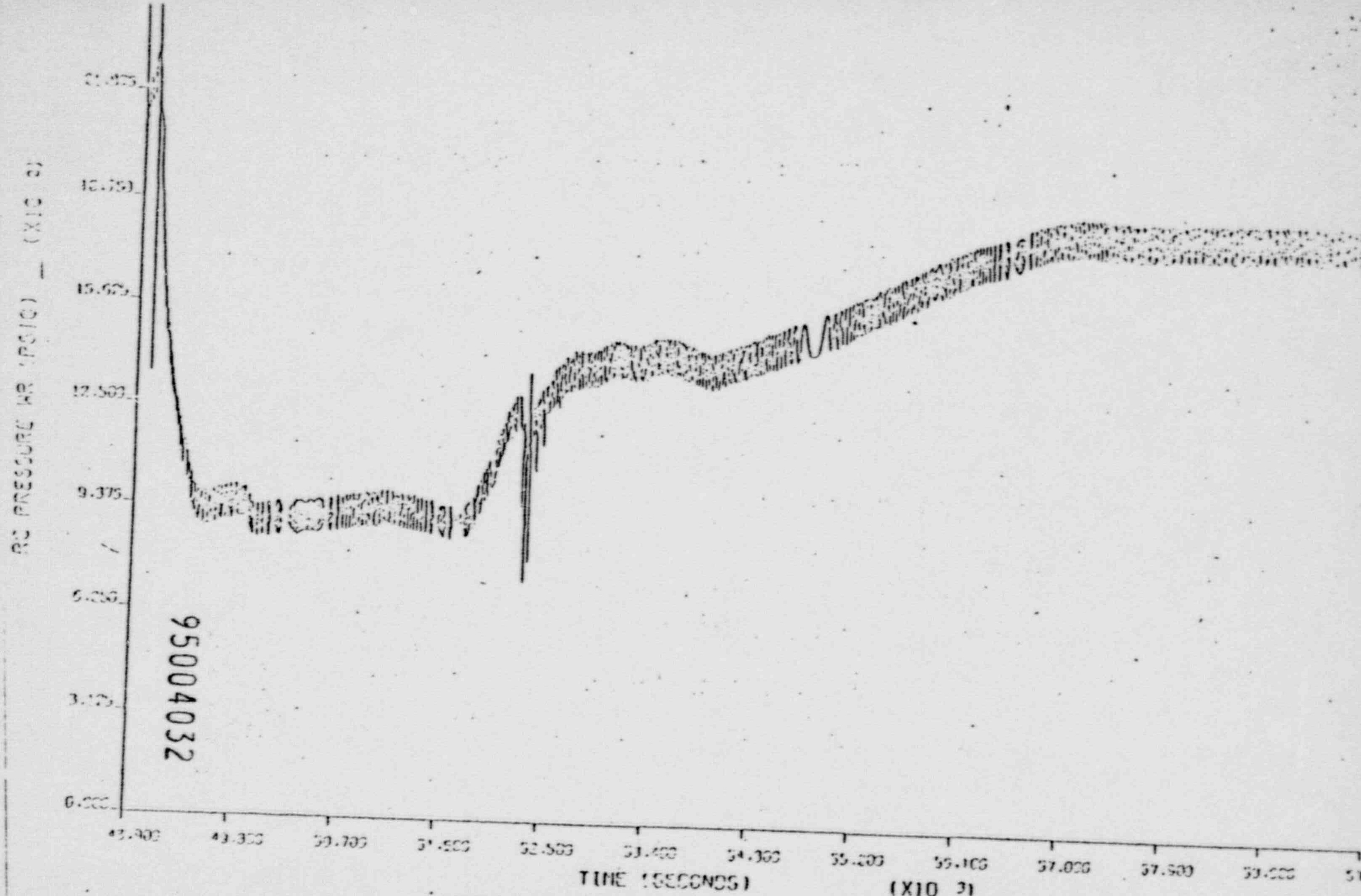


95004030

REACTIMETER PLOT TSN=71

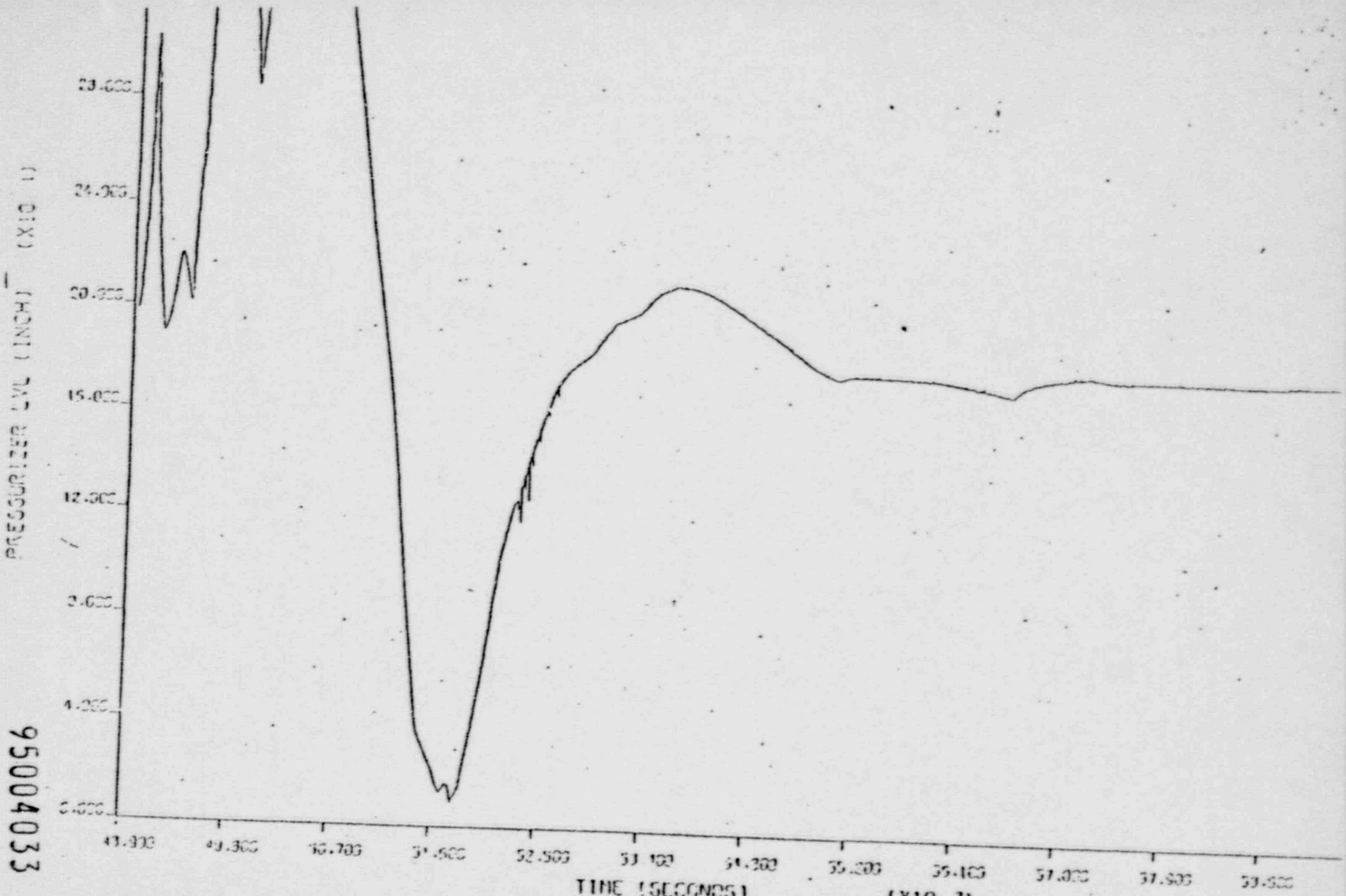


REACTIMETER PLOT TSN=71



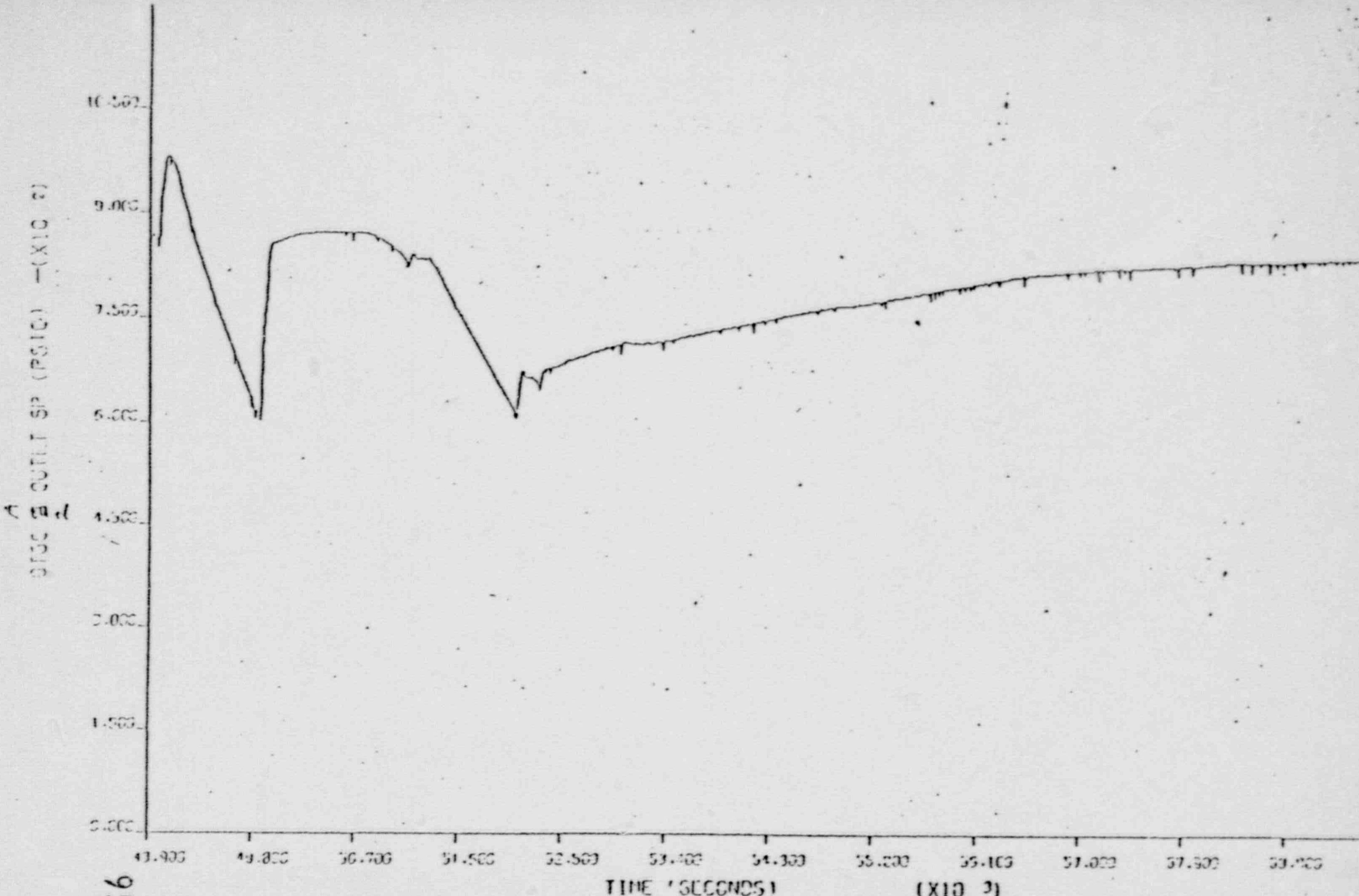
REACTIMETER PLOT TSN=71

Figure 4-1



REACTIMETER PLOT TSN=71

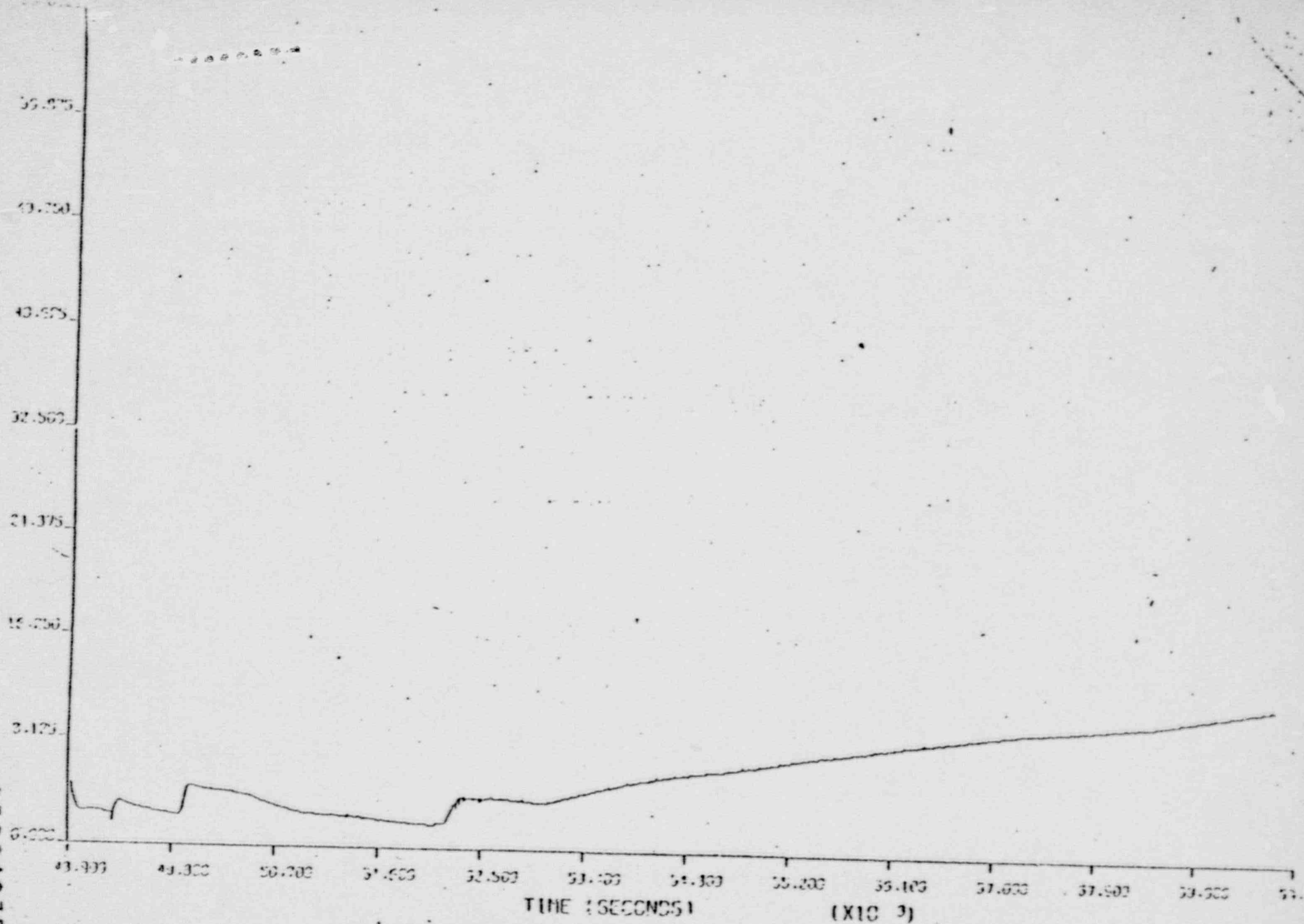
95004033



REACTIMETER PLOT TSN=71

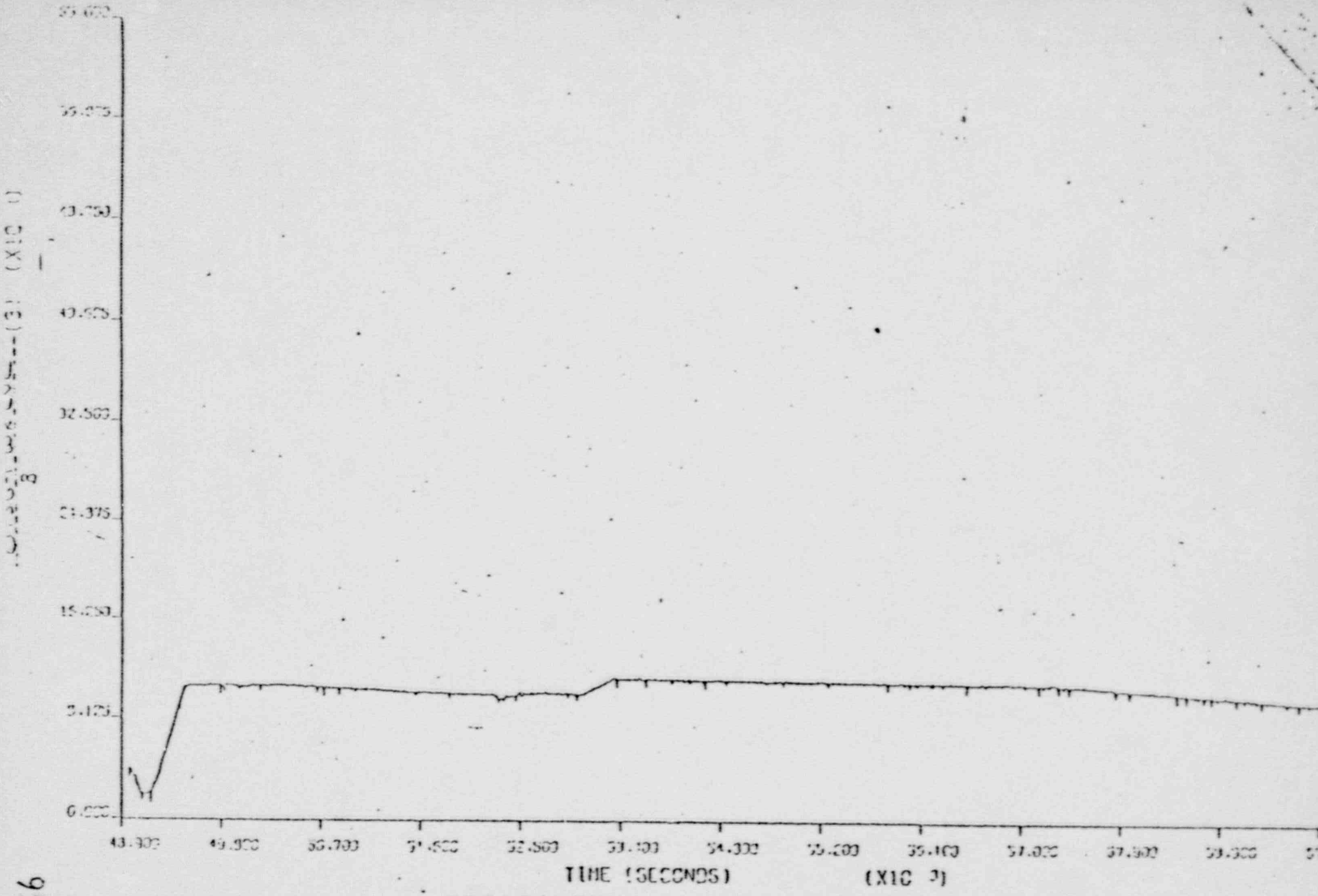
Figure 4-4

95004034



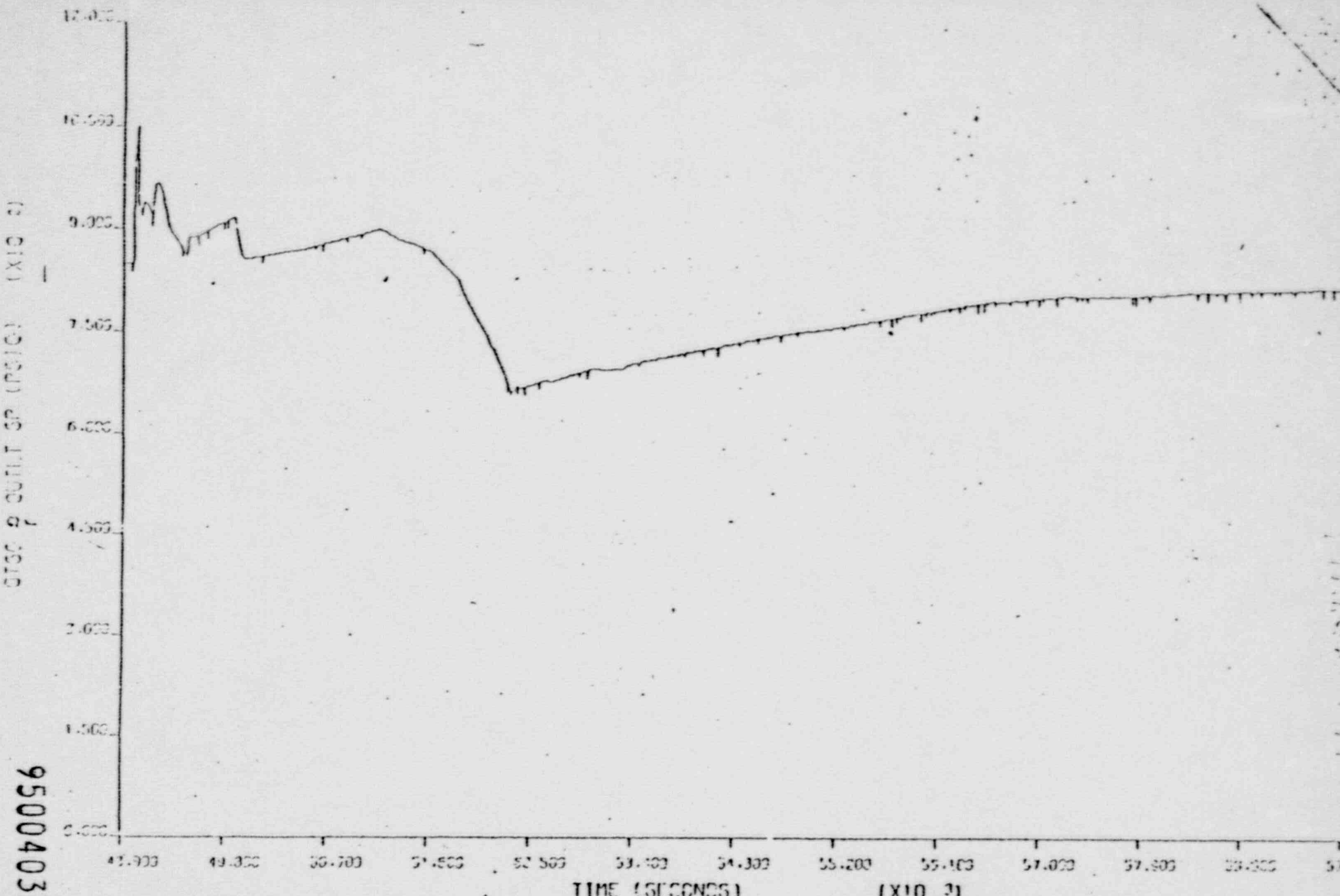
REACTIMETER PLOT TSN=71

95004035



REACTIMETER PLOT   TSN=71

95004036



REACTIMETER PLOT TSN=71

Figure 4-3

95004037

RC PUMPS

As a result of the September 24 abnormal system transient, the reactor coolant pumps experienced the conditions outlined in Attachment C.2. In order to demonstrate that there was no serious damage to the pumps, a series of operational checks were performed as outlined in Attachment C.3. The results of the operational checks are described in Attachment C.4.

B&W has reviewed the results of the operational checks and concluded that no detectable damage has occurred to the pump components. B&W finds the pumps to be serviceable for sustained full operational conditions with no immediate requirement for maintenance.

It should be noted that a step increase in vertical vibration of 2-2 pump was observed during the initial low pressure checkout runs. This indication was later assessed to be spurious instrument noise as a result of a loose connector on an instrument line. After the connector was tightened, vertical vibrations remained less than one quarter mils peak-to-peak amplitude.

RC PUMPS

SEPTEMBER 24 TRANSIENT @ DB-1

All four RC pumps were subjected to the following:

0:00	Reactor trip
1:10	SFAS trip
1:12	Seal return valves shut for 1:16
1:13	Seal injection valves shut for 1:52
	all four pumps operated for 1:15 with no seal injection and no seal return flow during an RCS de-pressurization
2:28	Seal return valves open
3:05	Seal injection valves open
~ 6:00	Steam formation
	pressure oscillating near P_{SAT} for ~30 to 45 minutes
36:07	Total seal injection flow low alarm

Pump 1-1:

7:04	Pump tripped
7:45	Shaft stopped
36:07	About one minute of low seal injection flow (near 2 gpm)
	flow imbalance starved seal injection
36:30	Seal return valve shut
1:12:55	Standpipe level high
1:17:07	Standpipe level normal

Pump 2-2:

4:20	High vibration
7:04	Pump tripped
36:07	Lost seal injection for about one minute
36:22	Seal return valve shut for about 40 seconds

95004039

ATTACHMENT C.3 p1

CHECKOUT OF REACTOR COOLANT PUMPS

PURPOSE:

Assess whether maintenance is required of RC pumps as a result of abnormal transient of 9/24/77. Operational checks will be required to demonstrate that no significant damage has occurred to the pump bearings, shaft and seals. First series of tests will be performed in Mode 5 due to operational restrictions by NRC. Later on operational checks will be performed in Mode 3. Each pump will be operated individually for a duration not to exceed ten (10) minutes, providing all defined parameters remain within limits established in this procedure.

Operational sequence will be as follows:

1. Lift pumps will be started and pump shafts rotated by hand. Torque values are not to exceed 200 ft-lbs. A stethoscope will be provided to detect any unusual mechanical noises in seal housing area. (This has been satisfactorily completed on 10/3/77).

2. Mode 5 testing 225 psig.

   2.1 Instrumentation Required - see attached (1A).

   2.2 Computer Data -
   Printout NSS special summary trend for running RCP every 15 seconds.

   2.3 Following limits shall not be exceeded:

   A. Shaft vibration - 15 mills peak to peak.

   B. Total standpipe leakage (upper seal leakage) plus seal return should not exceed 0.6 gpm. If, during the test this limit is exceeded, the possibility exists of an open seal. In no case will total seal leakage be allowed to exceed 1.5 gpm. If this limit is exceeded, maintenance will be required before further pump operation.

   C. All other normal plant limits and precautions prevail.

   2.4 Sequence of Operation:

   A. Secure standpipe flush.

   B. Establish seal injection in accordance with plant operating procedure.

   C. Measure and record standpipe leakage and return flow, confirm that total leakage limits are not exceeded.

   D. Assure communication between control room and personnel stationed at RCP standpipe leakage drain line.

95004040

   E. Countdown from 10 to 0

   Start strip chart recorders at high speed;
   Start Reactor Coolant Pump 2-2 in accordance with plant op. procedure.

F. Run pump for two (2) minutes unless any above limits are exceeded.

G. Data taken will be assessed by B&W and B-J representatives.

H. Following assessment of data, pump may be run for an additional five (5) minutes to allow for venting procedure requirements.

I. Follow above sequence on 2-1, 1-2 and 1-1.

J. Assessment of this data will determine whether any maintenance is required before higher pressure operation is allowed.

3. ~~Above~~ SIMILAR test will be repeated with system pressure at greater than 1300 psig before final determination on condition of the pumps is completed.

CCE:mlf
10/5/77

95004041

(1A)

LIST OF POINTS TO BE INSTRUMENTED FOR RC PUMP START UP:

1. Upper and lower cavity pressures - all four pumps.

2. Both horizontal B/N Vibration Probes - all four pumps.

3. WR System Pressure or suction pressure.

4. Vertical probe on 2-2 pump.

5. Standpipe leakage will be collected and measured during the test.

NOTE: All of above should be recorded on an 8 channel brush recorder located in the control room.

RFS:nlf
10/5/77

95004042

STATUS OF CHECKOUT OF REACTOR COOLANT PUMPS: 10/7/77 ATTACHMENT C.4

All four Reactor Coolant Pumps were run on 10/5/77, per the attached procedure, with the following results:

RCP 2-2 10/5/77 Run (2 min.):

System pressure 225 psig
2nd Seal cavity pressure 165 psig
3rd Seal cavity pressure 123.9 psig
Horizontal vibration 5 - 7.5 mills
Vertical vibration .25 mills

3rd Seal leakage plus seal return flow } <.4 gpm

After the two minute run, the pump was run for ten minutes for system venting. About 30 seconds before the pump was shutdown, there was a step increase in vertical vibration to 2.5 mills. The pump was run again on 10/6/77 for 10 minutes to checkout this phenomenon. The vertical vibration was again .25 mills until about 5 seconds before shutdown where it increased to 2.5 mills. To allow a longer run time, 2-1 and 2-2 pumps were run together for 10 minutes, then 2-2 was run alone for 10 minutes. The vertical vibration stayed at .25 mills for the entire run. This will continue to be monitored during pump runs for plant heat up.

RCP 2-1

System pressure 225 psig
2nd Seal cavity pressure 132 psig
3rd Seal cavity pressure 70 psig
Horizontal vibration 5 - 7.5 mills

3rd Seal leakage plus return flow } <.4 gpm

RCP 1-2

System pressure 225 psig
2nd Seal cavity pressure 40.29 psig
3rd Seal cavity pressure 81.3 psig
Horizontal vibration 5 - 7.5 mills

3rd Seal leakage plus return flow } <.4 gpm

RCP 1-1

System pressure 225 psig
2nd Seal cavity pressure 77.98 psig
3rd Seal cavity pressure 89.27 psig
Horizontal vibration 5 - 7.5 mills

3rd Seal leakage plus return flow } <.4 gpm

95004043

ATTACHMENT C.4

The apparent discrepancy on seal cavity pressures on 1-1 and 1-2 was checked on 10/6/77 by installing pressure gauges at the pressure transmitters. The gauges read as follows:

1-1:

184-2nd cavity
111-3rd cavity

1-2:
184-2nd cavity
112-3rd cavity

The readings indicate the seals are staging properly.

Based on the above performance, B&W sees no concern which would justify maintenance at this time.

Further Testing to be Done:

1. During heatup, contact B&W (~~R. F. Smith or C. C. England~~) whenever TECo plans to start a RCP, so additional data can be taken at B&W's discretion.
2. At system pressure > 1300 psig, 3 pumps running, data will be taken on all four pumps.

CCE:nlf
10/7/77

95004044

Attachment C.4

STATUS OF CHECKOUT OF REACTOR COOLANT PUMPS: 10/13/77

All four RC Pumps have been run at system pressure greater than 1300 psi. RC Pumps 2-1 and 2-2 have continued to run from the initial cold pump starts. Below is a typical line of data from each pump.

RCP 2-1

System Pressure - 1650 psig
2nd Seal Cavity Pressure - 1034 psig
3rd Seal Cavity Pressure - 500 psig
Horizontal Vibration - 3 mils

RCP 2-2

System Pressure - 1650 psig
2nd Seal Cavity Pressure - 1075 psig
3rd Seal Cavity Pressure - 588 psig
Horizontal Vibration - 3.5 mils

RCP 1-1

System pressure - 1650 psig
2nd Seal Cavity Pressure - 1056 psig
3rd Seal Cavity Pressure - 540 psig
Horizontal Vibration - 4 mils

RCP 1-2

System Pressure - 1650 psig
2nd Seal Cavity Pressure - 920 psig
3rd Seal Cavity Pressure - 520 psig
Horizontal Vibration - 3 mils

Based on the above data, I feel that all four pumps are in good operating condition and require nothing more at this time than periodic monitoring.

RJS:nlf
10/13/77

R. J. Smith

95004045

DB 1 CORE

ANALYSIS OF SEPTEMBER 24 DEPRESSURIZATION EVENT

A more detailed analysis was done to assess core thermal conditions during the September 24 depressurization event at Davis-Besse 1. Core conditions were analyzed to (1) determine if steam was produced in the core, (2) determine the maximum internal fuel rod pressure during the transient, and (3) determine if maximum lift force exceeded the limit.

CORE COOLANT CONDITIONS

Attachment D.2 shows transient thermal conditions as monitored by the reactimeter. The system pressure is measured at the pressure tap, which is approximately 65 feet above the top of the core. The RC pressure at the top of the core is approximately 50 psi higher than the measured pressure because of unrecoverable and elevation pressure losses. As shown in Attachment D.3, the predicted core coolant temperature is slightly higher than the minimum saturation temperature (based upon measured pressure), however, there is some uncertainty in both the measurement and the prediction, therefore, it is possible that some vapor bubble formation (steam bubbles in water) could have occurred within the core. An examination of the reactimeter data (attachment D.4) indicates that the RCS pressure level was near the saturation pressure for less than one hour and that during this time period the pressure oscillated with a variation of ± 50 psi. Therefore, the maximum time period during which the core could have been subjected to bubbly flow was less than one hour. ~~Approximately fifteen minutes after reactor trip the coolant temperature dropped below the minimum estimated saturation temperature, therefore, the bubbly flow, if it existed at all, occurred for no more than ten minutes.~~ If bubbles were formed during this period, the formation would be in the liquid as well as on the surface, as opposed to formation from a hot surface. With the temperatures, time duration, and type of formation, no significant effect on the components would be predicted.

FUEL ROD PRESSURE

Prior to the depressurizati n event the reactor had been operating at 15% power for approximately one week. Immediately prior to reactor trip the power level was 9% of rated power. The core burnup was 1 EFPD, therefore no signficant fission gas production had occurred and none was released. ~~The maximum initial backfill pressure of this fuel was 465 psia at 70°F.~~ During the 60 minute time perio. in which the indicate RCS pressure was estimated to vary from 900 to 1000 psia at the top of the core the average coolant temperature was less than 540°F and no significant heat generation occurred in the fuel. An initial evaluation (~~reference~~) had predicted tensile stresses in the cladding based upon a maximum pressure differential across the cladding of 200 to 300 psi. This evaluation had been based upon a DOL TAFY analysis with an arbitrary safety factor added to ensure that actual conditions would be bounded by the prediction. A more recent analysis, again using TAFY, has resulted in a predicted maximum internal fuel rod pressure of 1000 psia. This analysis considered as-built fuel properties and at, near zero power conditions at a coolant average temperature of 540°F. On the

95004046

basis of this analysis it is concluded that the fuel rod cladding was not subjected to any significant level of tensile stress during the subject depressurization event.

Since the cladding was not subjected to a large, long term tensile stress, no significant long term effects on the cladding resulted. The tensile stresses which could have occurred would have little effect on the cladding due to the small stress level and the short duration of the tensile stress.

CORE LIFT

Assuming a coolant temperature of 537 F and 150 X 10^6 lb/min system flow (per Attachments D.5 and D.6) the net lift force will be less than 375 lb. The maximum allowable lift force is 472 lb., therefore fuel assembly lift-off ~~is not predicted.~~ did not

We conclude that