GPU 2507

Records Center T1.2

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November 1, 1977

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BWT-1589
File: T1.2/128

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Toledo Edison Company
Power Engineering & Construction
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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-1578 and BWT-1579 dated October 5 and 7:

A. Description of the event

B. Evaluation of the reactor coolant components

C. Evaluation of RC pumps

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 impr. e on this date.

Section A describes the sequence of events as reconstructed from computer alarm print-out, 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 and 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.

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November 1, 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

JAL/hj

J. A. Lauer
Project Manager

Attachments

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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 21:35:13). 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. The steam in the containment also resulted in 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).

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 opeator 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.

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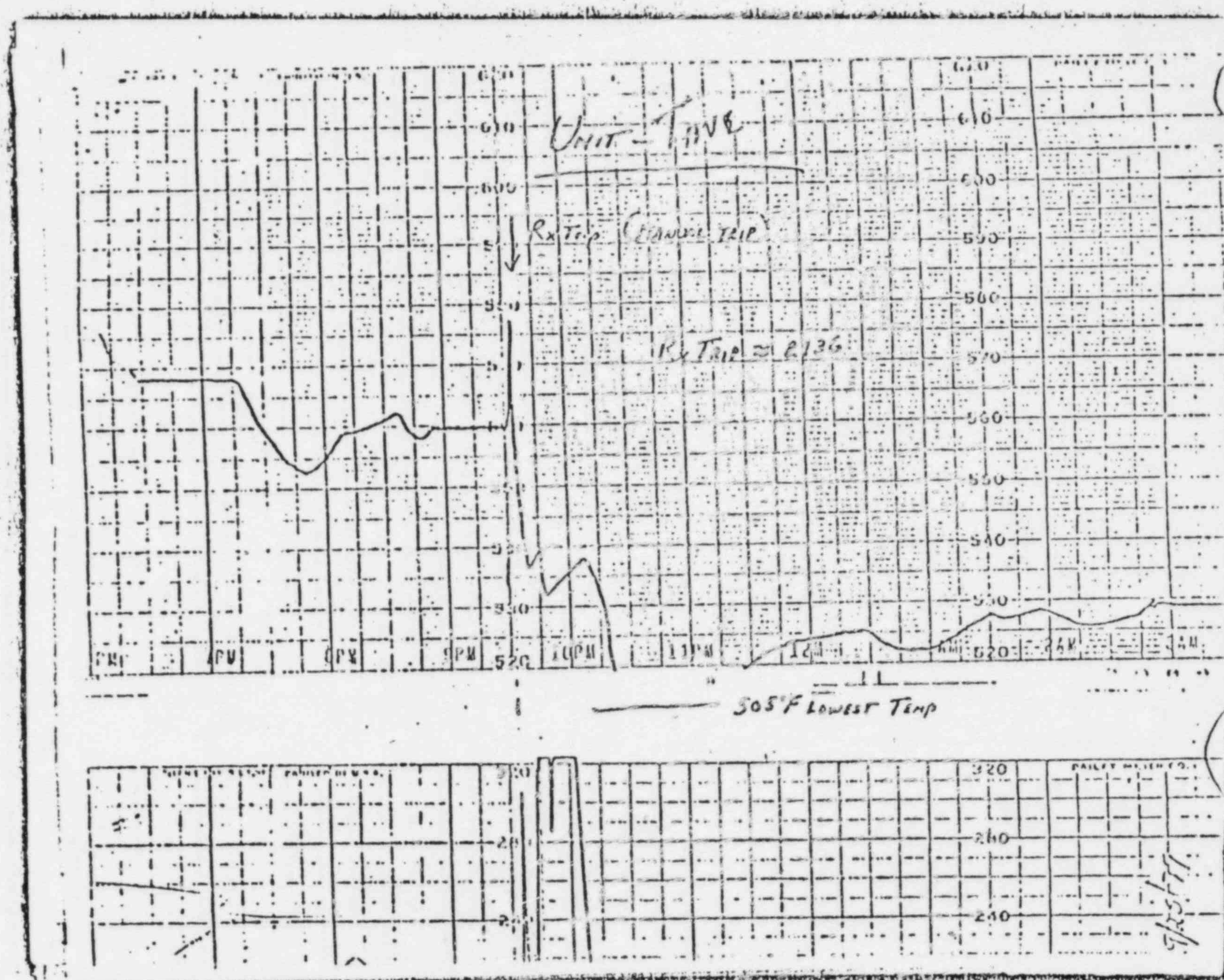

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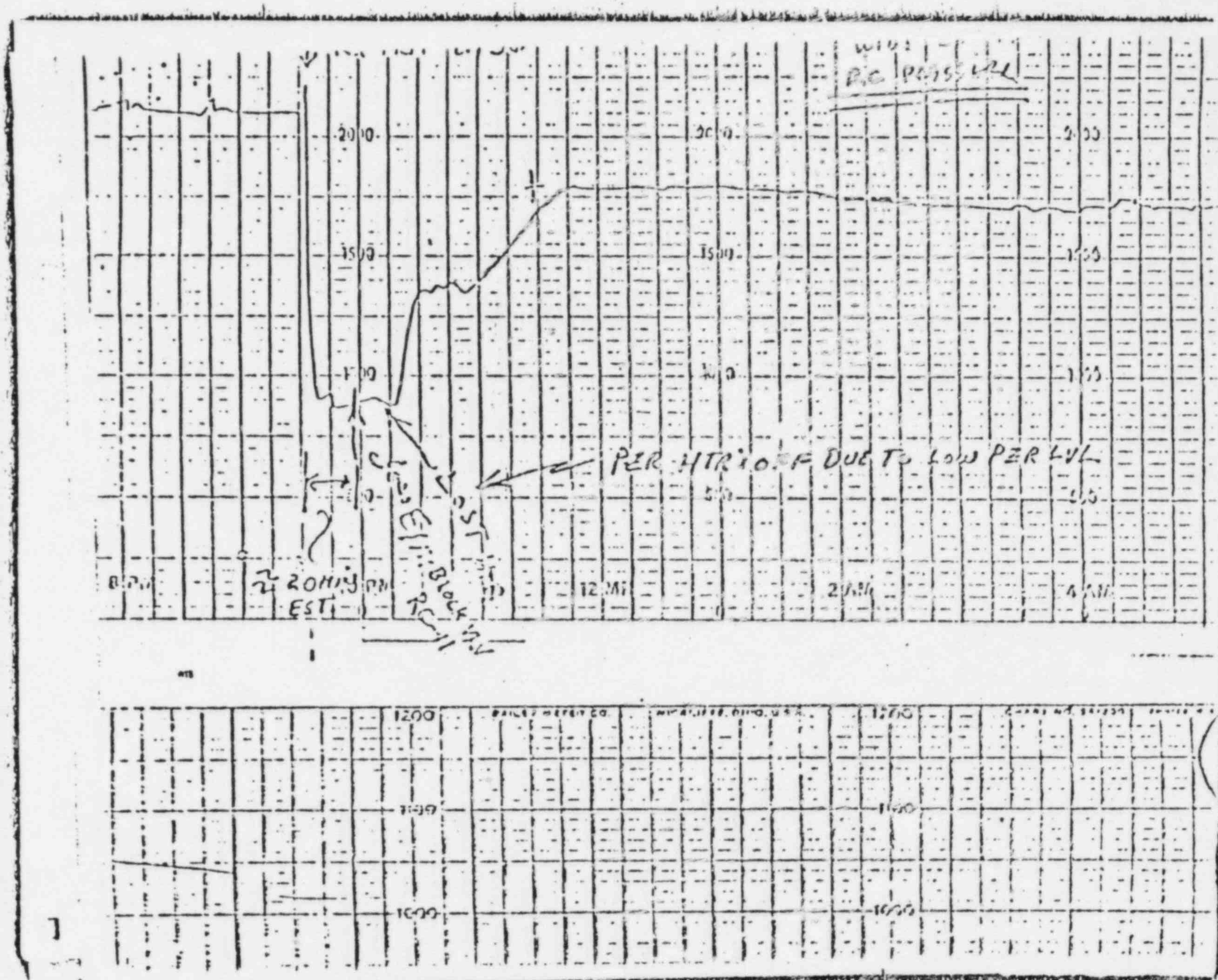
RC PRESSURE
PER HTR 10°F DUE TO LOW PZR LVL
~20 MIN EST
EST
BLOCK V.V
RC-11
2000
1500
1000
500
8 PM
12 MN
2 AM
4 AM
1200
1100
1000
BAILEY METER CO.

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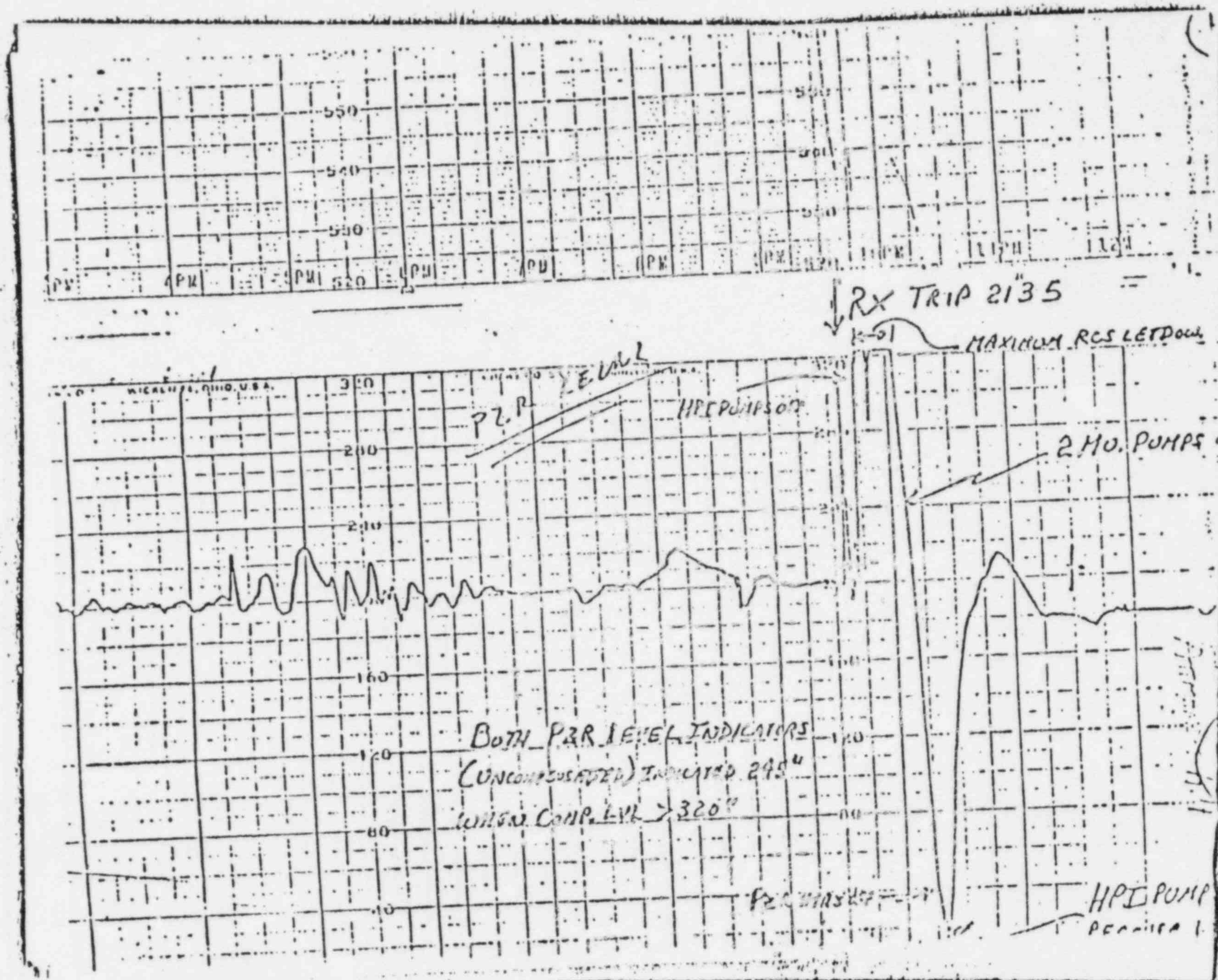

RX TRIP 21:35
MAXIMUM RCS LETDOWN
PZR LEVEL
HPI PUMPS ON
2 MO. PUMPS
BOTH PZR LEVEL INDICATORS
(UNCOMPENSATED) INDICATED 295"
WHEN COMP. LVL > 320"
HPI PUMP
550
540
530
320
200
240
160
120
80
40

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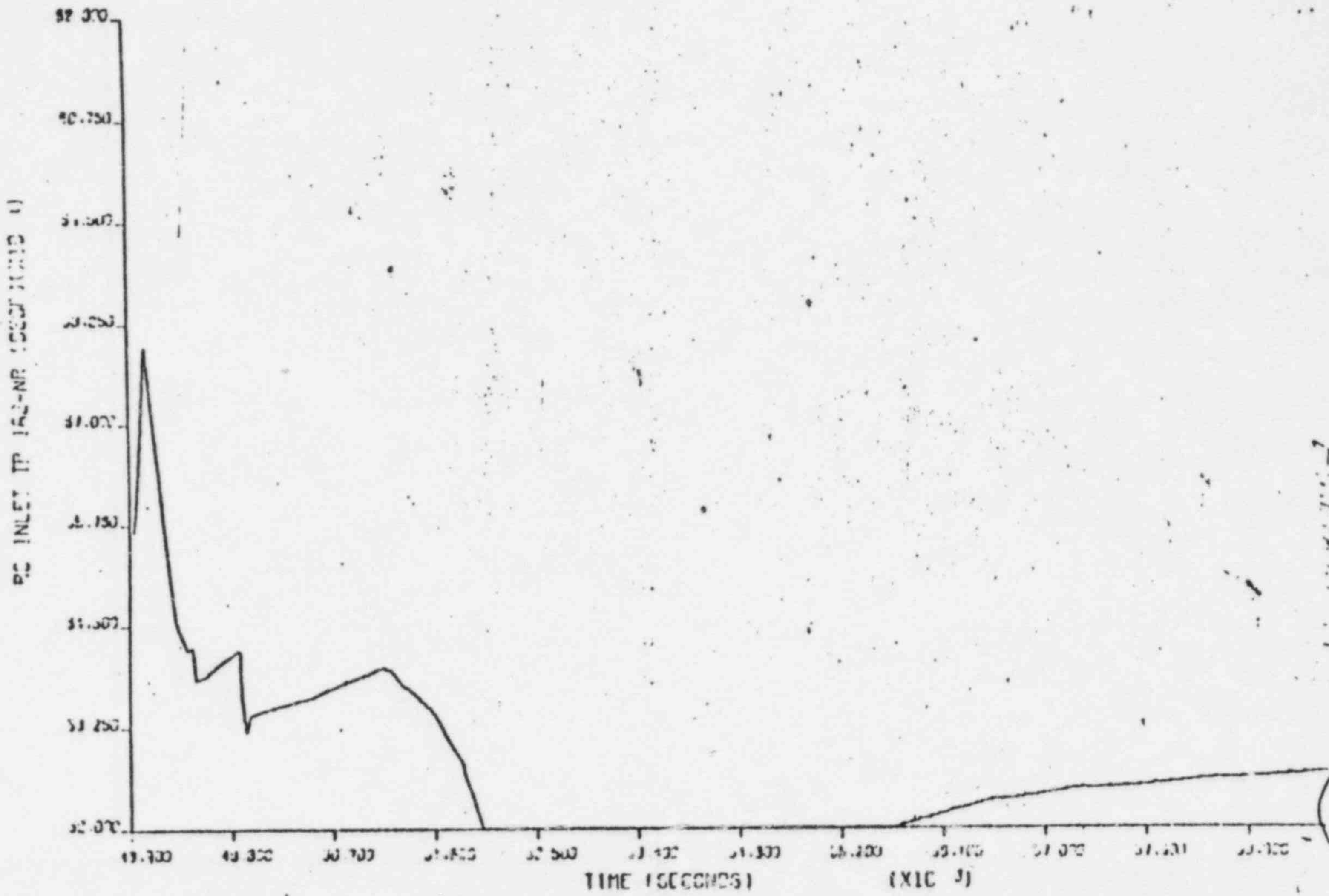

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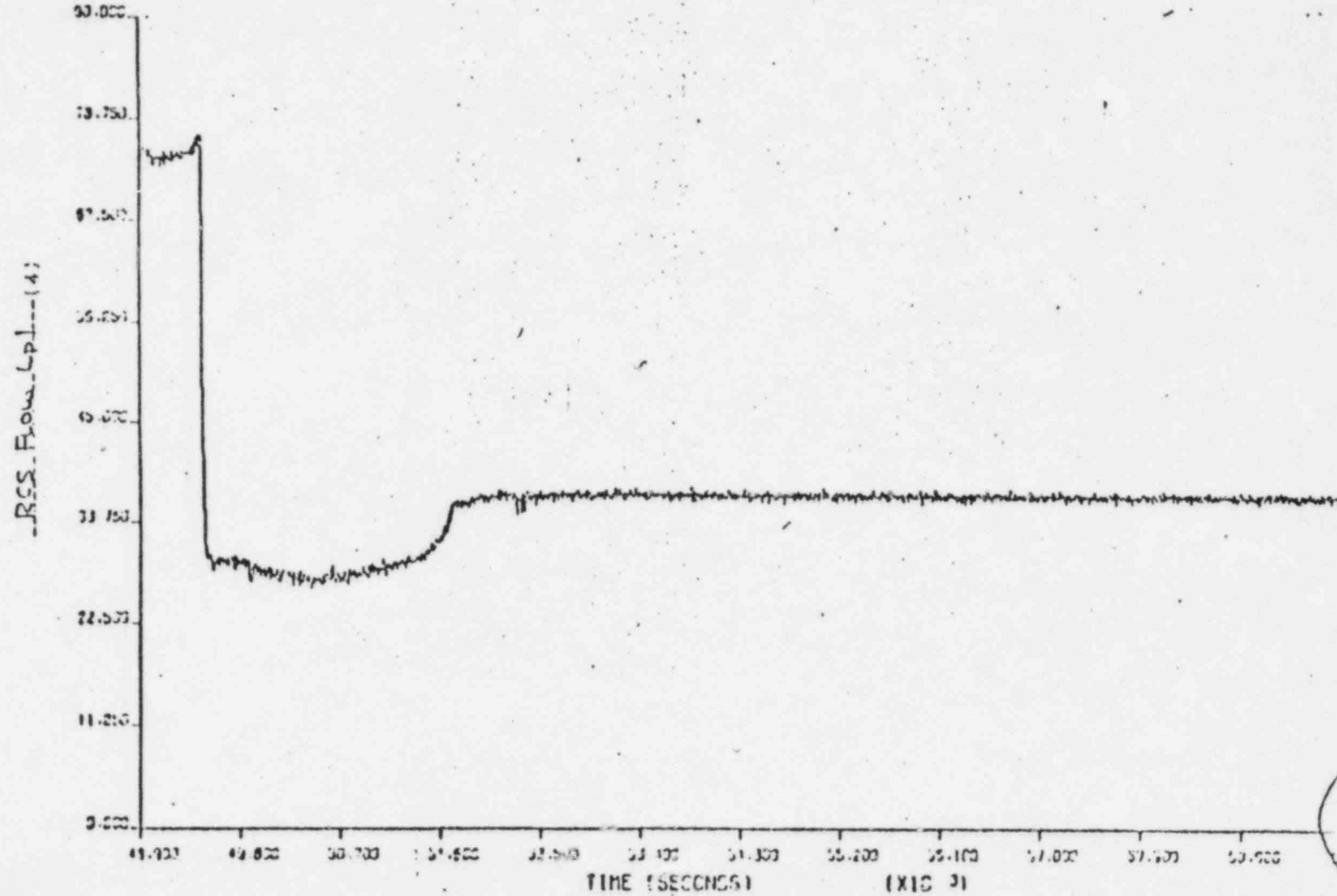

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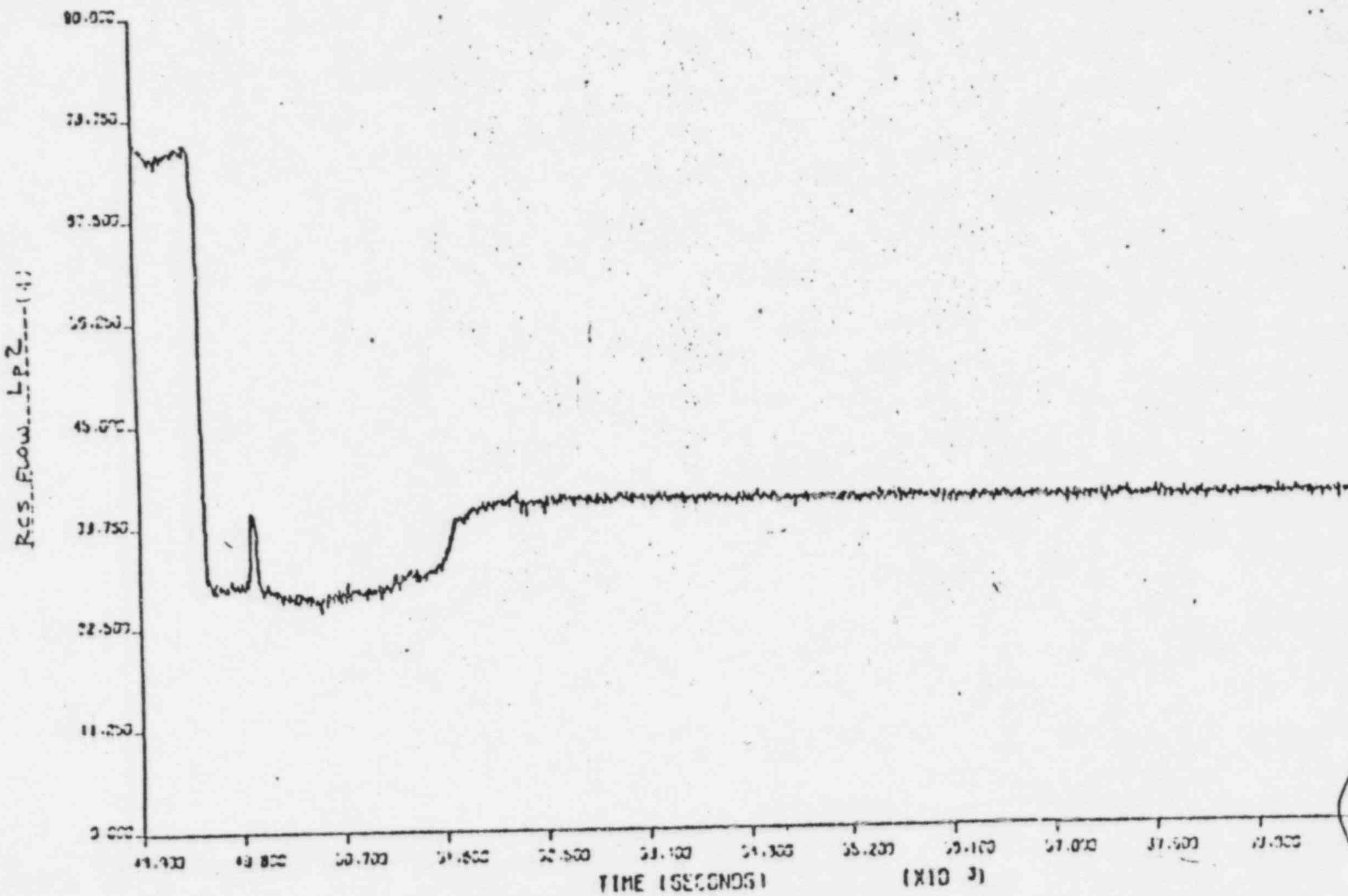

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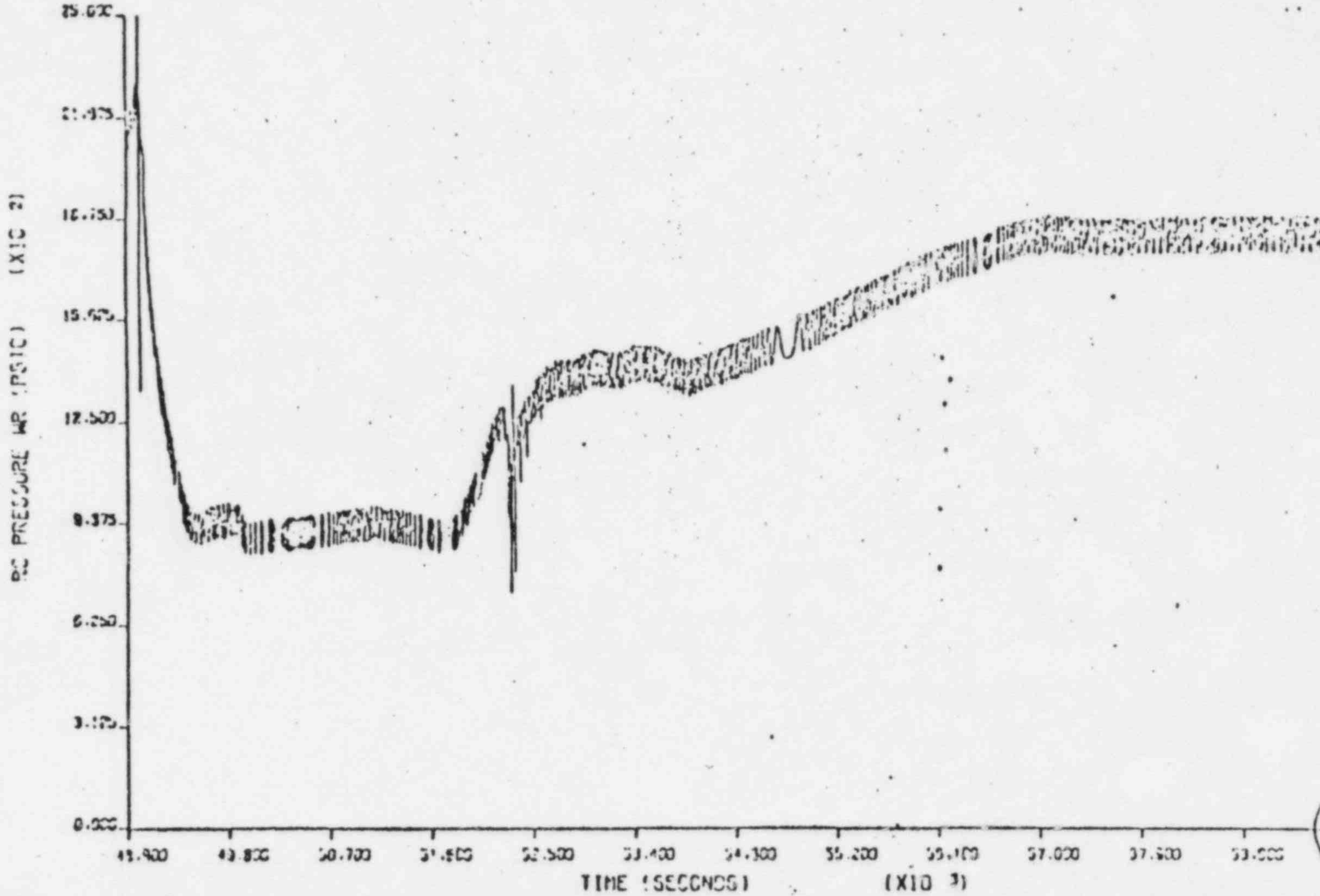

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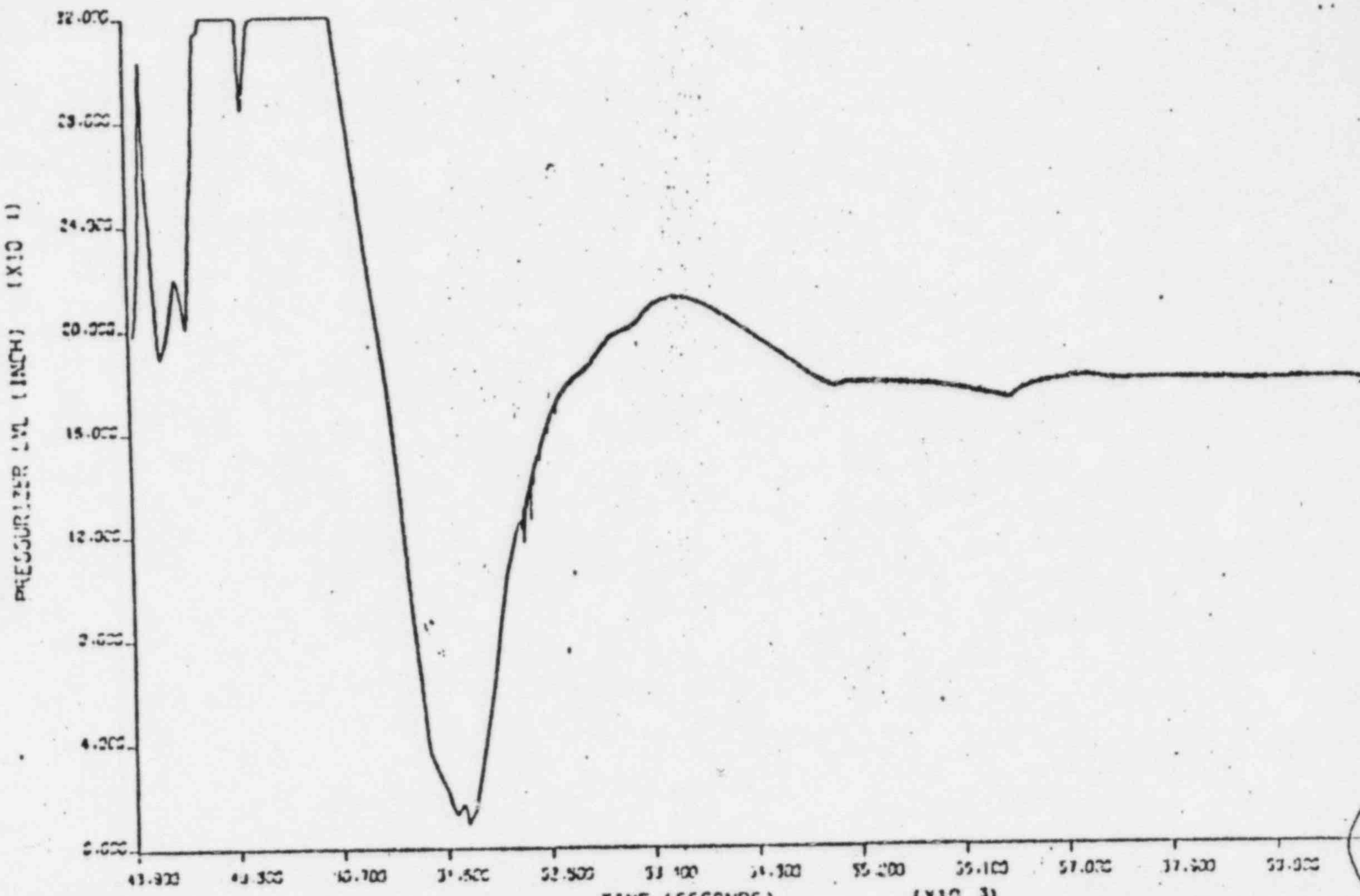

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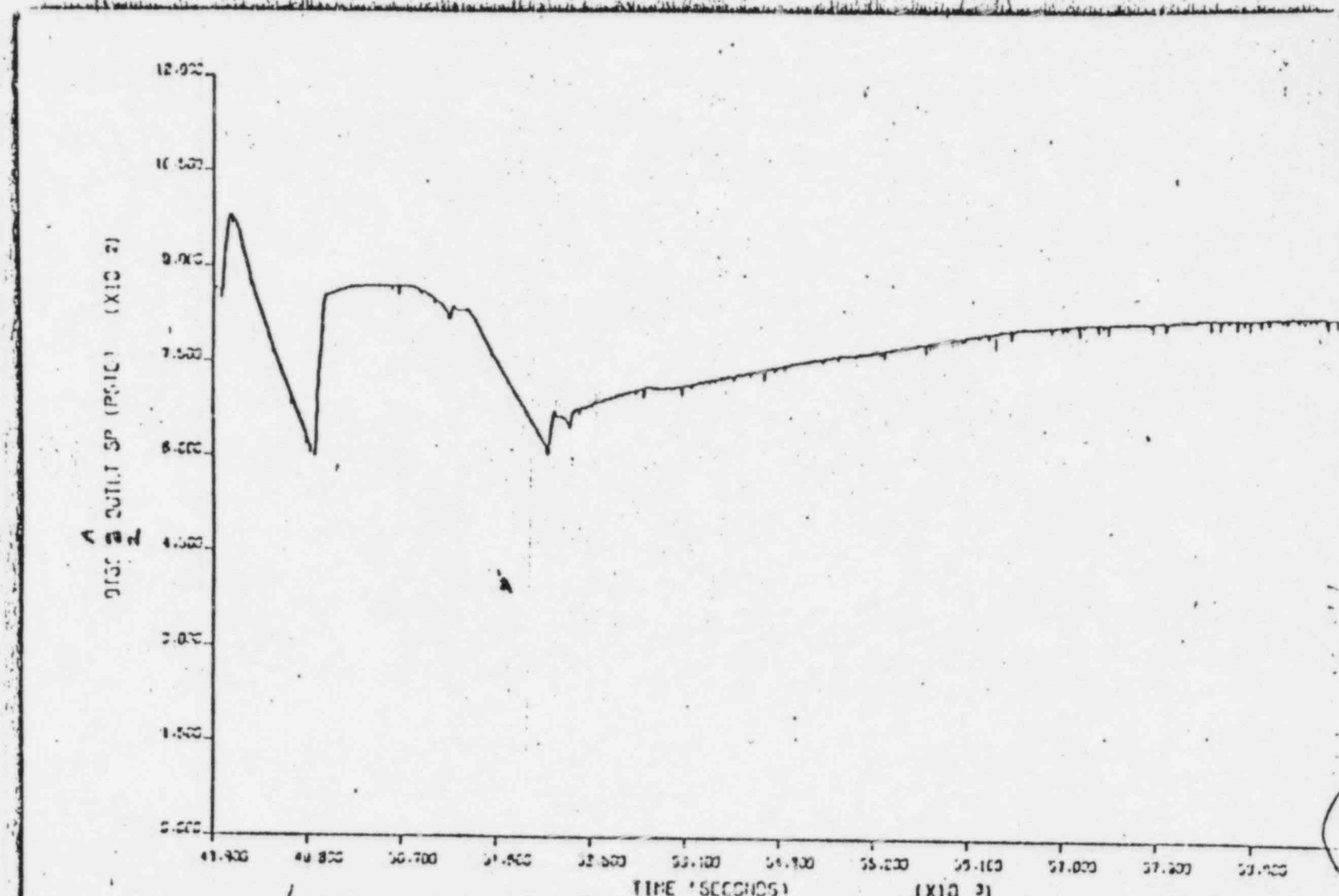

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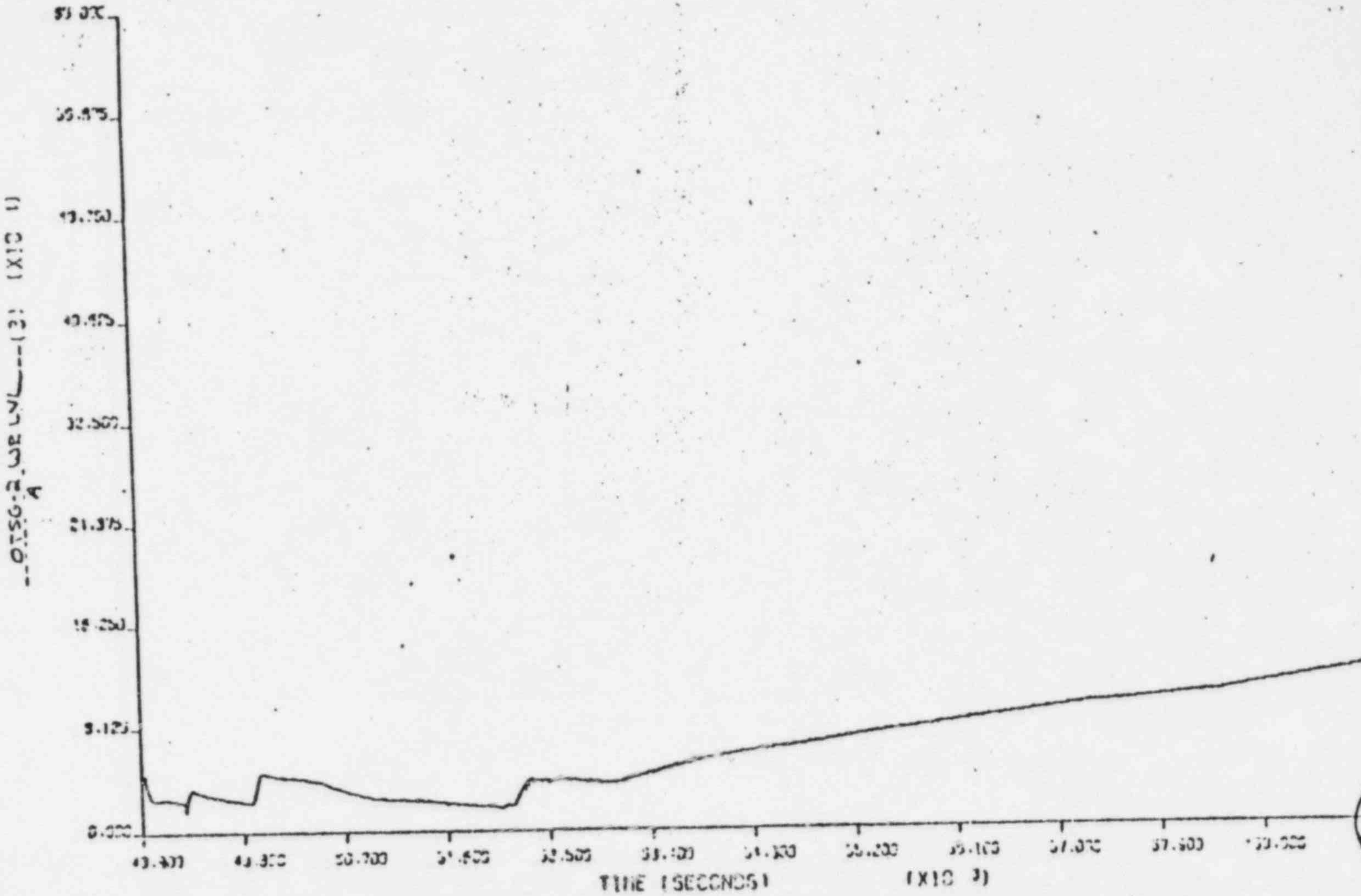

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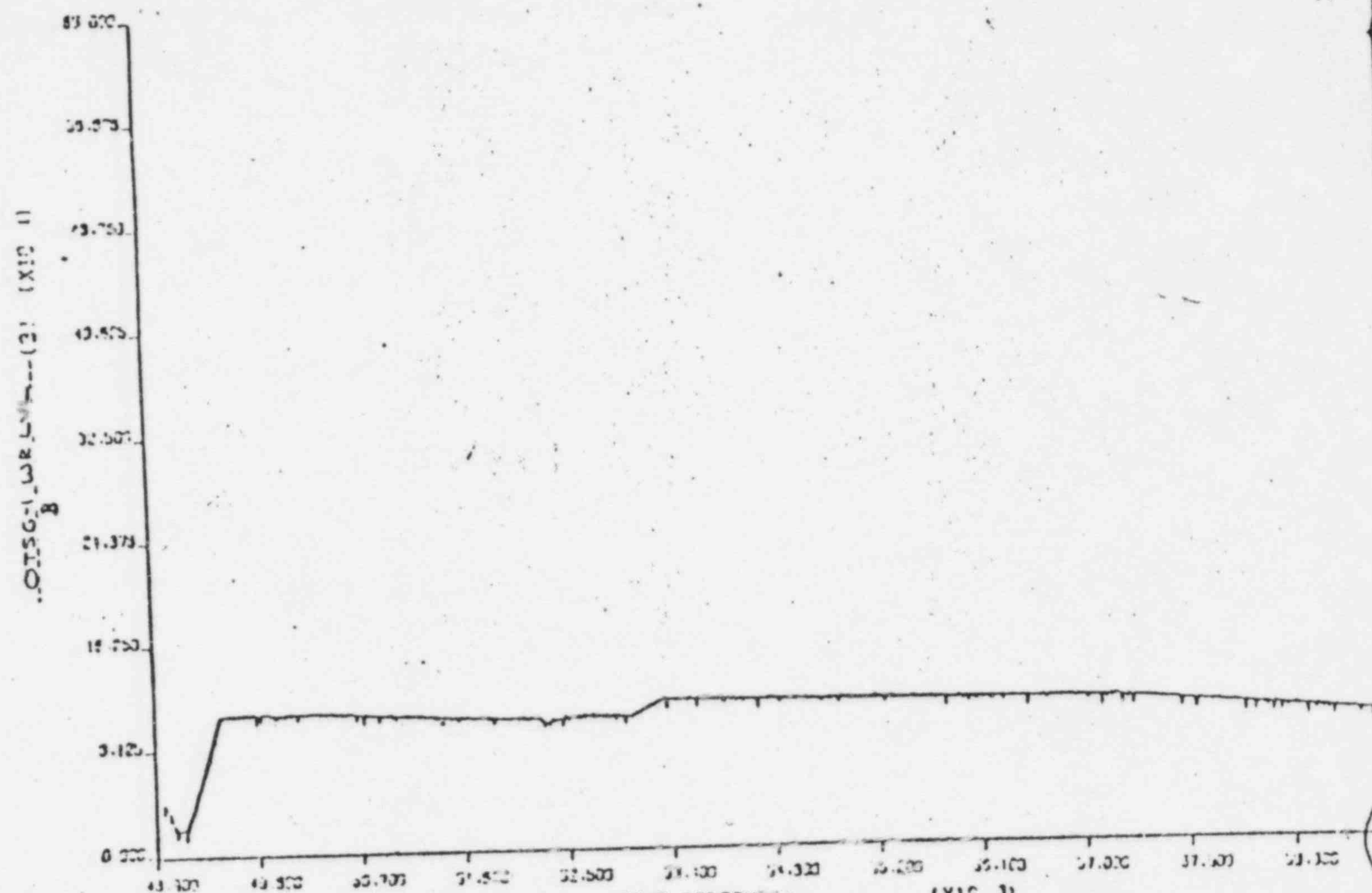

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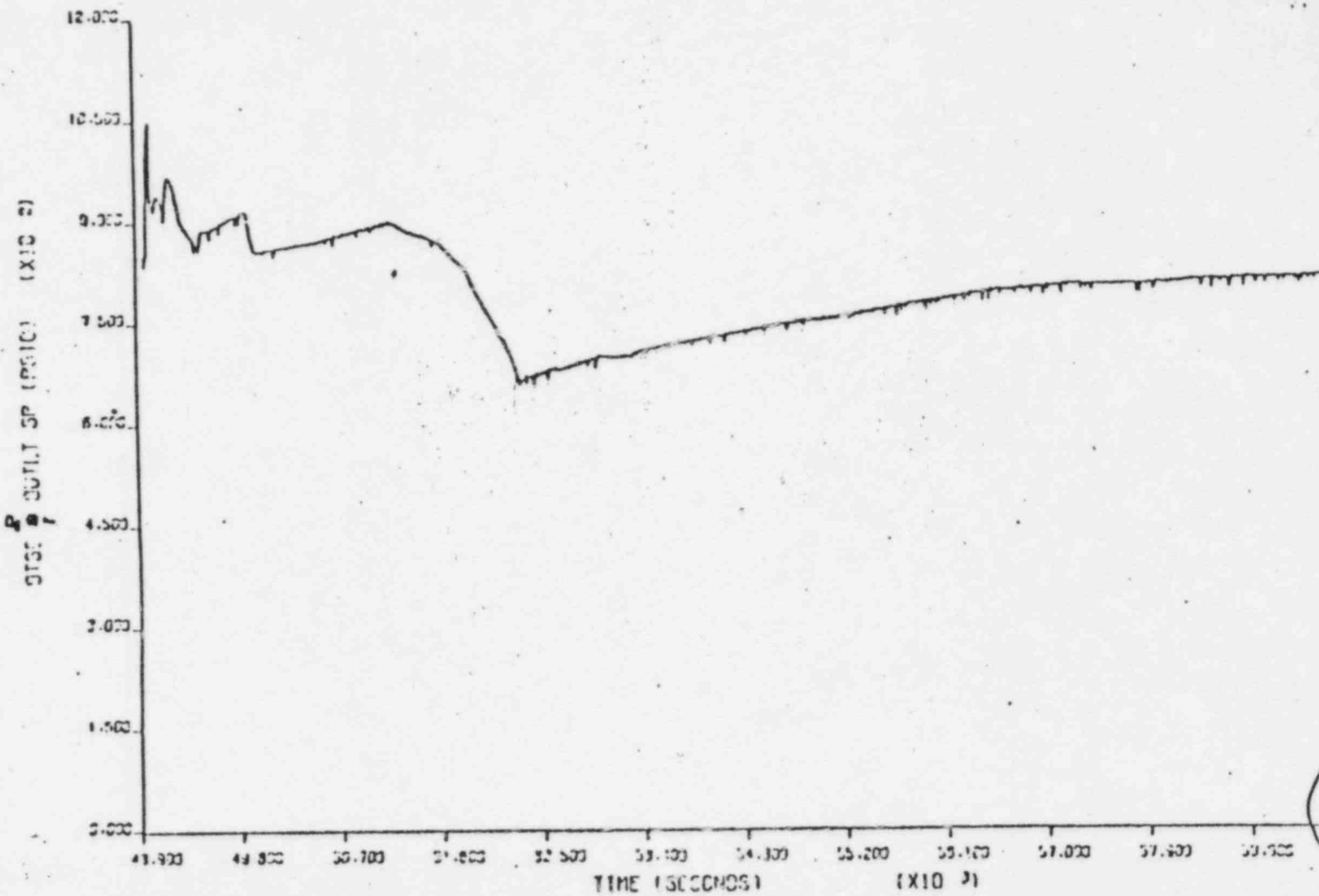

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RC PUMPS

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 mil peak-to-peak amplitude.

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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

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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 ESS 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.

      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.
      After approx. 11 sec., reduce strip chart speed.

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

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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.

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STATUS OF CHECKOUT OF REACTOR COOLANT PUMPS: 10/7/77 Attachment C.4 p1

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

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

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 } <.8 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.

<u>RCP 2-1</u>

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

<u>RCP 1-2</u>

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

<u>RCP 1-1</u>

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

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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 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.

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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 - 528 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, B&W feels that all four pumps are in good operating condition and require nothing more at this time than periodic monitoring.

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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 depressurization 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 significant fission gas production had occurred and none was released. During the 60 minute time period in which the indicated 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 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 BOL 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 hot, near zero power conditions at a coolant average temperature of 540°F. On the

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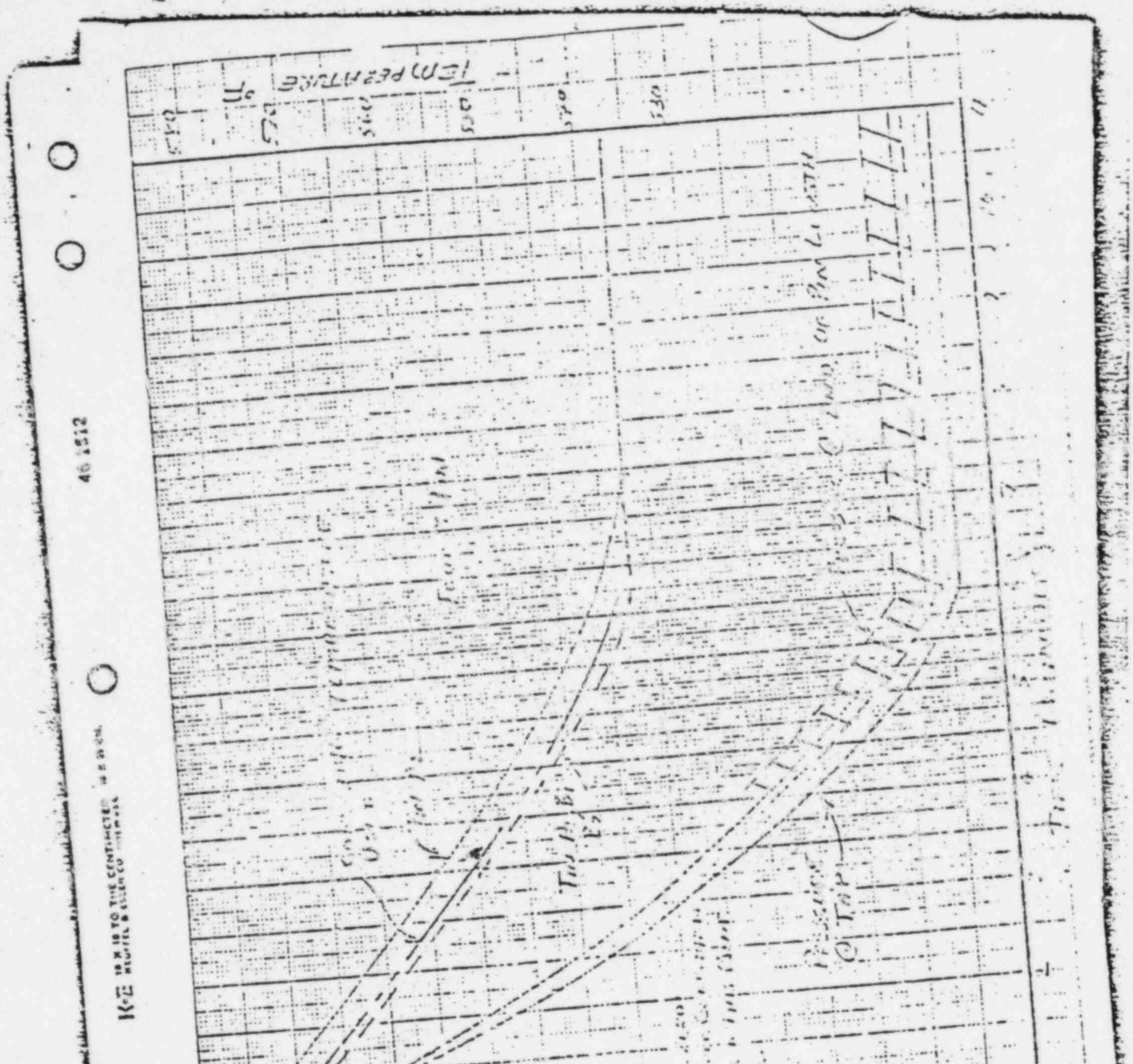
basis of this analysis it is concluded that the fuel rod cladding was not subjected to any significant level of tensile stress during the subject depressurization event.

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.



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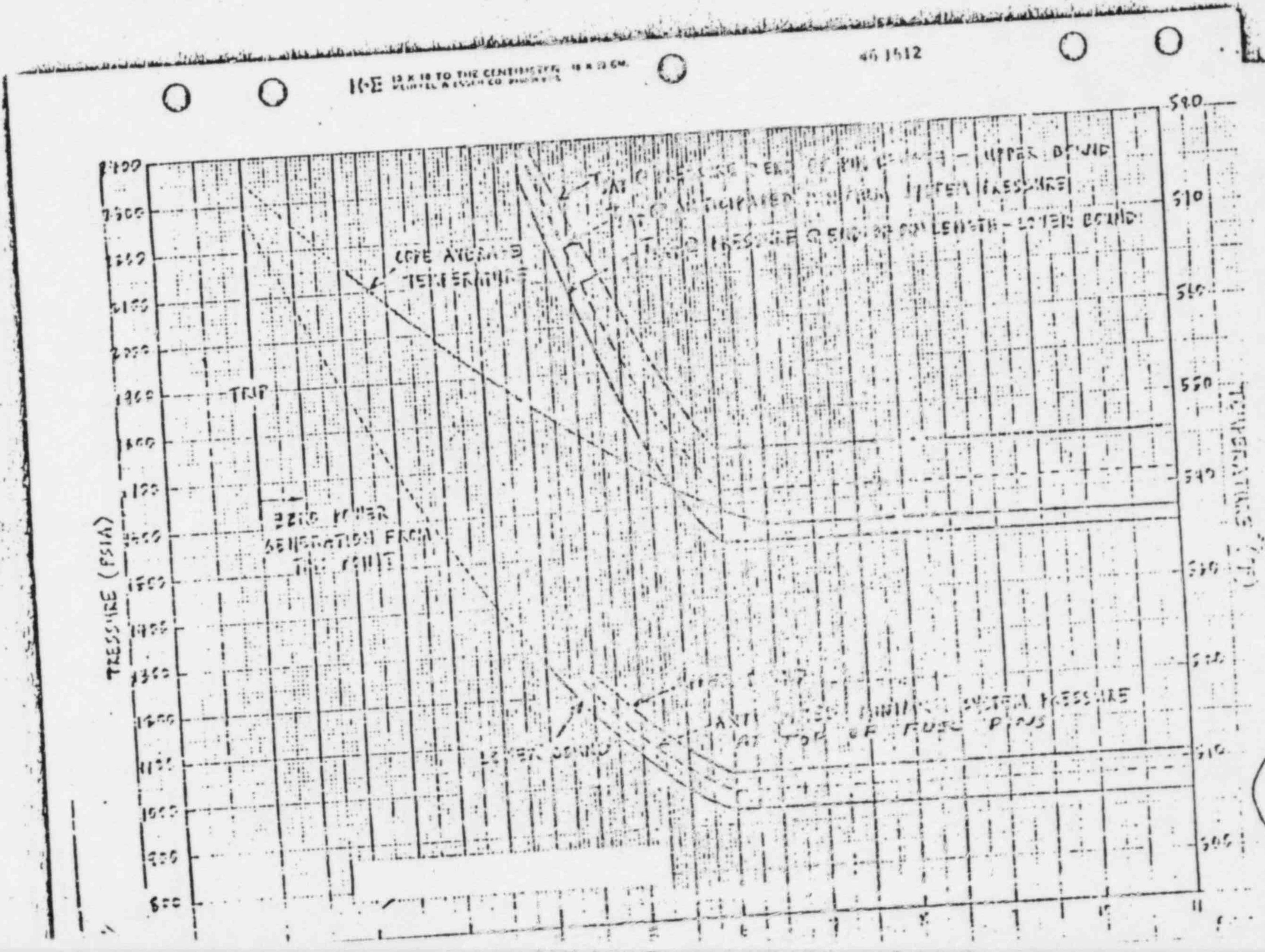
K+E 10 X 10 TO THE CENTIMETER
46 1512
TEMPERATURE (°F)
PRESSURE (PSIA)
TRIP
CORE AVERAGE TEMPERATURE
LOWER BOUND
580
570
560
550
540
530
510
500
2400
2300
2200
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1000
900
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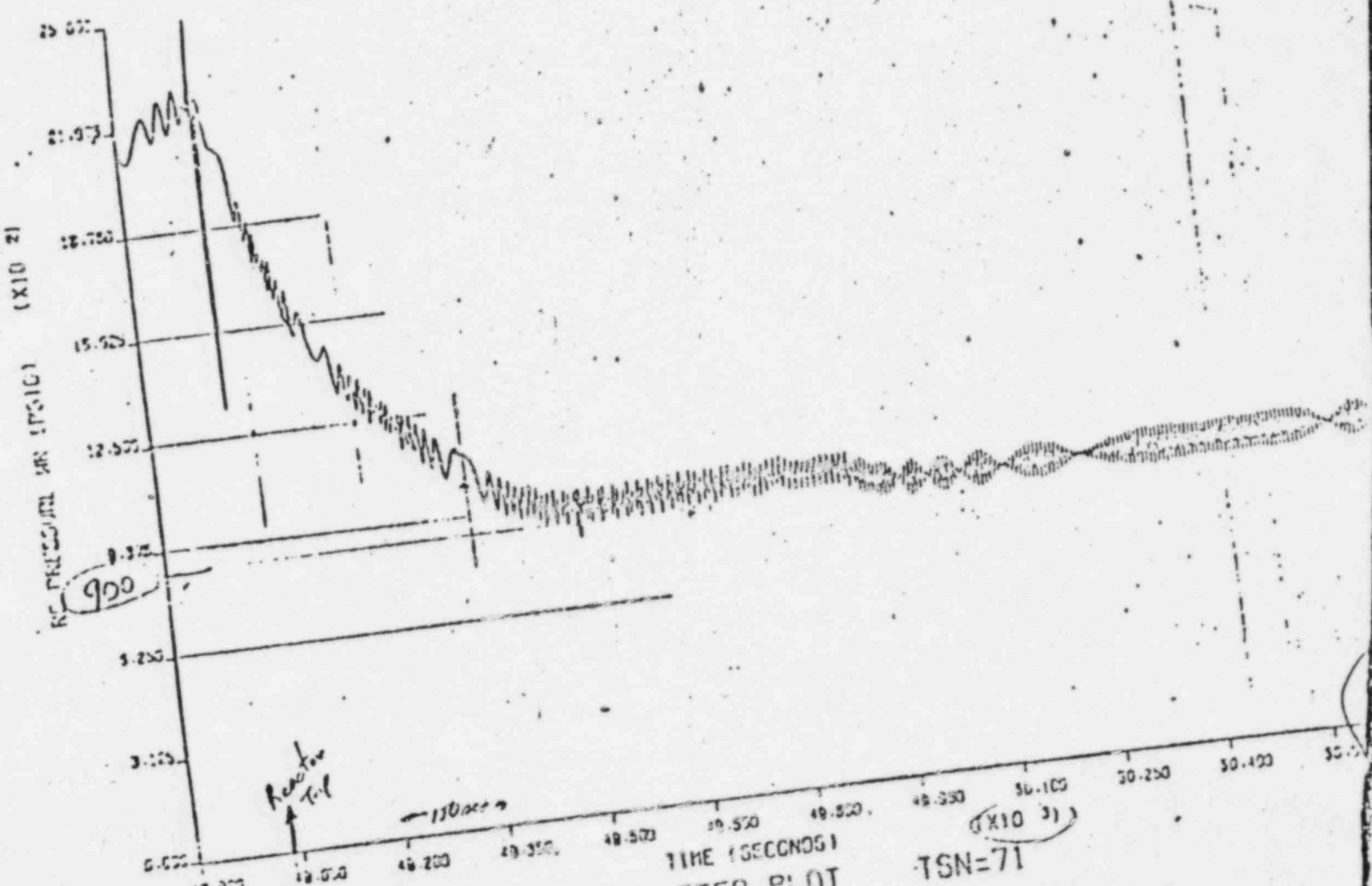

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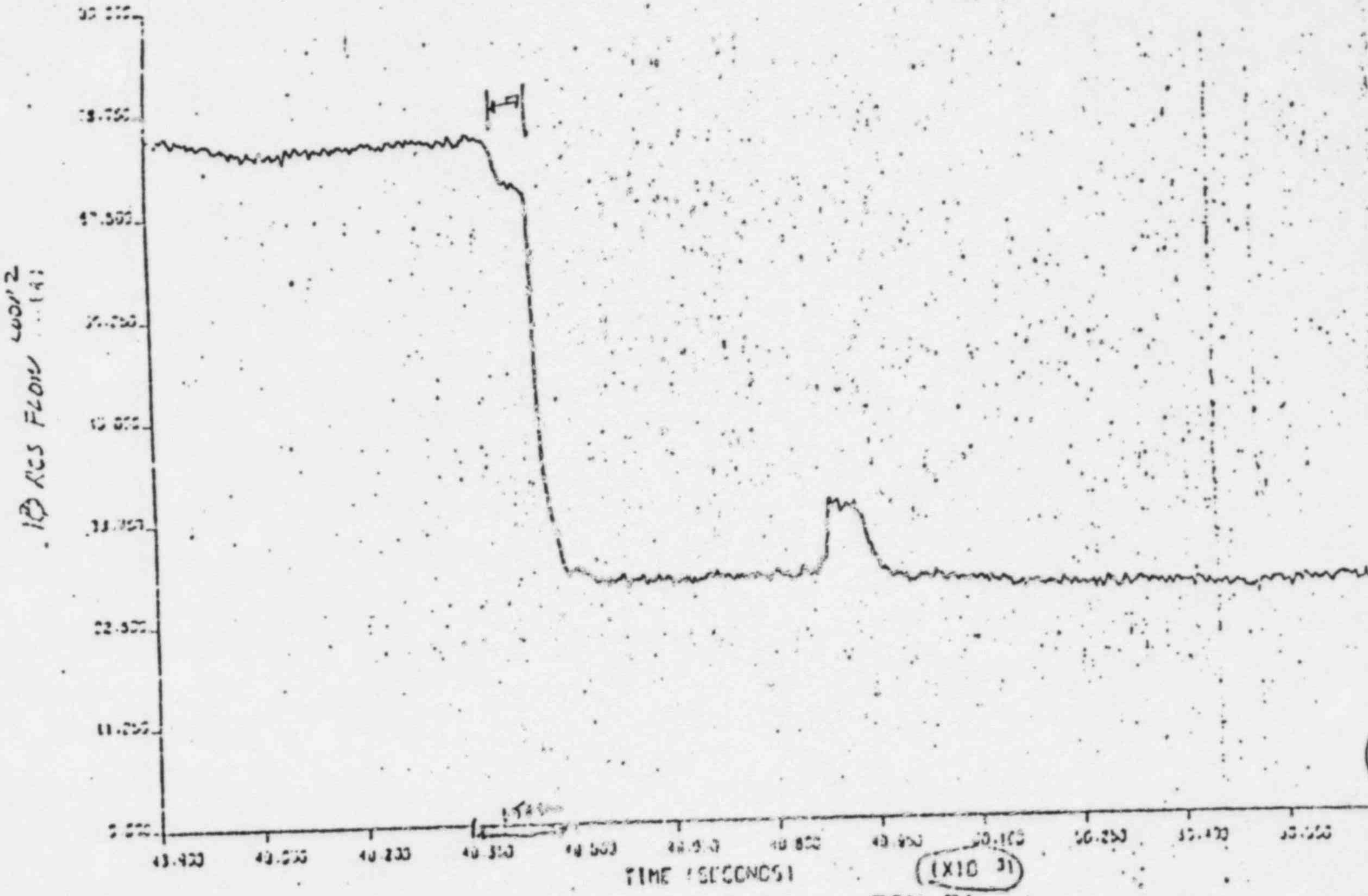

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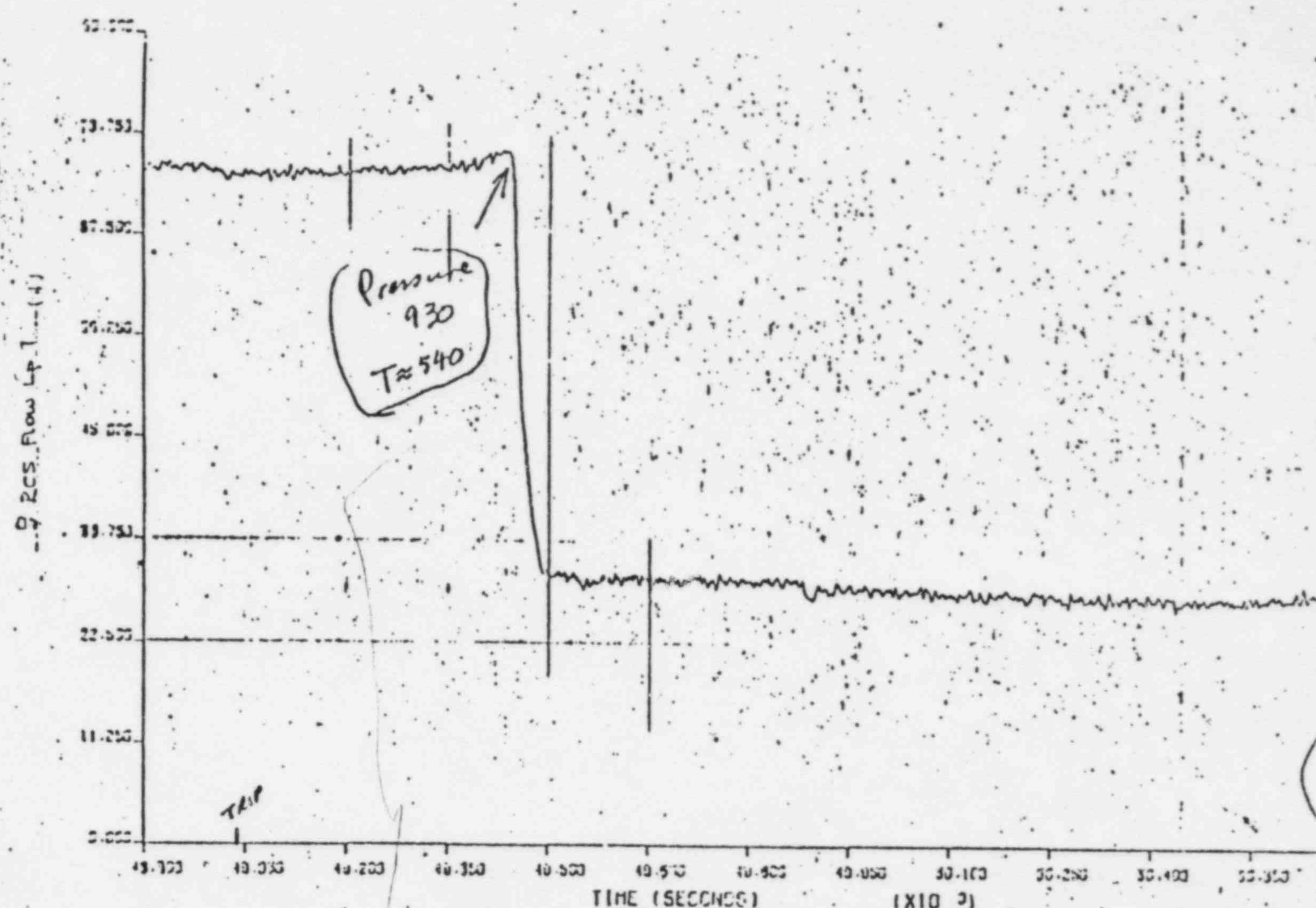

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