UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

FEB 8 1982

MEMORANDUM FOR: Distribution

FROM: R. W. Starostecki, Director, Division of Resident and Project Inspection, Region I

SUBJECT: PRELIMINARY REVIEW OF THE STEAM GENERATOR TUBE RUPTURE EVENT AT R. E. GINNA NUCLEAR POWER STATION

On January 25, 1982, a Steam Generator Tube Rputure (SGTR) event occurred at Ginna Station, Ontario, New York. This forwards a draft of the results of Region I's preliminary review of the event.

The review was based on independent analysis of the event and on interviews with operators and licensee management and considered the guidelines of the Westinghouse generic procedure for SGTR events (April 1980 Revision).

The attached contains:

1. Brief Description of the Event.
2. Review of Instrument Responses.
3. Review of Equipment Operation and Failures.
4. Review of Operator Performance.
5. Independent Procedure Review.
6. Event Chronology.
7. Tables and Figures Containing Parameter Data.
8. Copy of Ginna SGTR Procedure - E-1.4.

The attached is provided to the Steering Group for their use.

for R. W. Starostecki, Director
Division of Resident and Project
Inspection

Attachment: As Stated

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PDR FOIA
WILLIAM91-106 PDR

ENCLOSURE D-1

FEB 8 1982

Distribution w/ Attachment

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DRAFT NO. 2
Wiggins/cgl
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2/8/82

I. Introduction

Brief Description of the Event

At 9:25 a.m., January 25, 1982, a plant transient indicative of a tube rupture in the "B" Steam Generator occurred at Robert E. Ginna Nuclear Power Station, Ontario, New York. Within three minutes the plant tripped and Safety Injection initiated in response to low system pressure. All safety systems including Safety Injection operated as required during the event. In accordance with 10 CFR 50.72, the licensee informed the NRC about the event via the Emergency Notification System (ENS) five minutes after the plant trip and declared an Unusual Event. The severity of the rupture caused a rapid decrease in system pressure and pressurizer level. Eventually, the system stabilized at about 1200 - 1300 psig with the Pressurizer empty. The licensee then declared an Alert.

Following the applicable Station Emergency Procedure, one modeled after the Westinghouse generic procedure for Steam Generator Tube Rupture (SGTR), both Reactor Coolant Pumps (RCP) were secured. Securing the RCP's resulted in natural circulation flow with the core as the heat source and the "A" Steam Generator (S/G) as the sink. The combined effects of low flow in the Reactor Vessel Head, existent Head metal temperature and reduced system pressure resulted in the formation of a steam void in the Head.

ENCLOSURE D-2

The operators identified and isolated the faulted "B" S/G and commenced equalizing pressure between the RCS and the "B" S/G to reduce the leakrate. Because the RCP was unavailable, a Pressurizer Power Operated Relief Valve (PORV) was used to reduce RCS pressure. After two successful cycles of the PORV, a third attempt resulted in the PORV becoming stuck open. Pressurizer level rose rapidly and the operators shut the respective block valve. Before the block valve could reach its fully shut position, Pressurizer level went offscale high. The rapid rise in Pressurizer level gave clear indication of void existence and possible growth in the Head.

The "B" S/G pressure and level continued to increase because of the existent leak and SI flow to the RCS. Eventually the "B" S/G code safety lifted and reseated relieving S/G pressure and resulting in an untreated release. All SI pump were secured to avoid further lifting of the safety. The licensee declared a Site Area Emergency.

Without SI flow, RCS pressure dropped to about 800 psig. However, incore thermocouples continued to indicate the core remained subcooled. One SI pump was restarted to insure adequate RCS inventory; the "B" S/G safety again lifted and reseated.

The "A" RCP was restarted to provide forced circulation flow. The RCP forced cooler water into the Head region and swept out and collapsed the steam void. A steam bubble was reestablished in the Pressurizer and the plant cooldown was continued. To cooldown the faulted S/G, a controlled feed and bleed operation was used: the "B" S/G was fed by AFW which maintained S/G pressure greater than RCS pressure and bled through the tube rupture back to the RCS.

At 7:17 p.m., January 25, 1982, the licensee downgraded the Site Area Emergency to an Alert. At 10:45 a.m., January 26, 1982, the licensee downgraded the Alert to the Recovery Phase. At 6:35 p.m., the plant was in Cold Shutdown.

II. Plant Performance

A. Instrument Responses

Instrument responses were reviewed to determine the sequence of events during the transient, to assess plant performance, to assess the usefulness of the instruments to the operators and to analyze the event and its effects. Data were obtained from hard-copy printouts from the P-250 Process Computer from control room recorder traces and from manual records. A summary of the parameters reviewed along with references to various Tables and Figures in this report appear below:

Parameter	Table	Figure
Pressurizer Level	I	1
RCS Pressure and TSAT	II	1,
Steam Generator Level and Pressure	III	2, 3
Upper Head Thermocouple Temperature	IV	4, 6
Core Exit Thermocouple (AVG)	V	4, 6
Cold Leg Temperatures	VI	4
Pressurizer Relief Tank Pressure and Temperature	VII	None
Pressurizer Water, Steam and Surge Line Temperature	VIII	5

Several problems associated with parameter records existed. The Process Computer was out of service from about 11:19 a.m. to about 11:47 a.m. January 25. This failure resulted in a loss of automatic data transfer to the TSC and EOF and a loss of hard copy records for this time period. Additionally, some chart recorders either failed to ink at certain periods duri·· the event or failed to respond to the equipment monitored. These failures did not, however, hinder the response of the operators and are therefore considered insignificant.

It should be noted that Reactor Vessel Upper Head Thermocouple temperatures were used to verify and monitor the steam void in the Head area. These thermocouples mounted in the upper vessel head area provided vital data during the event.

B. Equipment Operation

During and after the event, the operation of relevant plant equipment was reviewed to determine the adequacy of equipment response and to identify areas of potential concern. This review included a comparison of the equipment used with the Emergency Procedure equipment requirements; an evaluation of the data accumulated during the event, personal observations in the Control Room and included discussions with the Ginna Operations Staff. As a brief summary, plant equipment generally performed its intended function as required throughout the event. Some specific failures were identified and are discussed below.

1. Reactor Protection System (RPS)

The plant tripped on low pressure as required. The setpoint for the Ginna RPS trip on low pressure is 1873 psig with a rate factor adjustment. It appears the trip occurred at an acceptable pressure level. No failures in this system were identified.

2. Engineering Safety Features

Safety Injection (SI) initiated in response to low system pressure with a coincident low pressurizer level. In connection with this initiation, the three SI pumps started, the RHR pumps started, the containment isolated and the motor driven Auxiliary Feedwater Pumps started. No failures were identified.

Operation of the SI pumps during the event maintained RCS inventory to keep the core covered and prevent the establishment of saturation conditions in the core region. Despite their effects, however, a steam void formed in the Vessel Head as a result of the existent Head temperature and reduced flow through the Head area under natural circulation conditions. In addition, the continued leakage of RCS inventory into the faulted Steam Generator (S/G) increased the pressure in this generator to the point where a code safety lifted. Until the Reactor Coolant Pump in the non-faulted loop was restarted and the Head void collapsed, the SI pumps caused S/G safety lifts and contributed to the overall radiological release during the event.

3. Pressurizer Power Operated Relief Valve (PORV) and Block Valve Operation

In accordance with the Ginna Emergency Procedure and the Westinghouse generic procedure for Steam Generator Tube Rupture (SGTR) events, RCS pressure is reduced to the faulted S/G pressure by use of a Pressurizer PORV if spray flow cannot be used. Due to the initial pressure transient, both Reactor Coolant Pumps were secured in accordance with the SGTR procedure. Therefore use of the PORV was mandated.

After two successful cycles of PORV PCV-430, a third use resulted in it failing to fully close. Its respective block valve was closed to arrest the RCS pressure drop. Before the block valve could be fully closed, pressurizer level rose until it indicated offscale high. During the depressurization resulting from the PORV sticking, the Vessel Head void apparently grew; however, incore thermocouples continued to indicated adequate subcooling existed in the core regions.

The PORV is a Copes-Vulcan 3" Reverse Acting Valve.

4. Letdown System Relief Valve and Seal Injection Return Relief Valve Operation

The letdown containment penetration is isolated outside containment by valve 371 which closed on containment isolation. Inside containment there is a parallel arrangement of three letdown orifice isolation valves in series with an additional valve LCV-(427) which is downstream of the regenerative heat exchanger. Neither the three orifice isolations LCV-(200A, 200B and 202) nor LCV-427 receive containment isolation signals; however, they each close on low pressurizer level. In addition, on loss of instrument air, 200A, 200B and 202 fail closed and 427 fails open.

During the event, 371 closed on containment isolation LCV-427 failed open on loss of instrument air which resulted from the containment isolation; and, the inservice orifice isolation closed as a result of low pressurizer level. Upon regaining of instrumentation and sufficient pressurizer level, 427, the inservice orifice isolation opened; 371 remain closed because its isolation signal was not rese. at the Containment Isolation Reset panel. The letdown penetration line then pressurized, lifting the letdown relief and adding to the RCS inventory loss and Pressurizer Relief Tank inventory increase.

The above actions were previously described in IE Inspection Report 50-244/81-09. The Inspection Report also indicates an engineering work request has been instituted to review and evaluate modifying the containment isolation signal logic to include letdown orifice valves 200A, 200B and 202.

The seal return relief lifted as a result of RCS leakage through the seal pressurizing the seal return line after the containment isolation. This action is expected during containment isolations and its impact on the event was minimal.

5. Pressurizer Relief Tank (PRT)

The Pressurizer Relief Tank filled as a result of the combined effects of Pressurizer PORV operation, the letdown relief valve lift and, to a minor extent, the seal return relief valve lift. Eventually, PRT level and pressure were such that the PRT's rupture disc ruptured. Water from the PRT flowed to the "A" Containment Sump. The sump filled to about 1320 gallons.

6. Saturation Meters

One Saturation Meter uses input from Thot and RCS pressure; the other uses Th and Pressurizer pressure. The process computer uses average incore temperature and Pressurizer Pressure.

The Saturation Meters and the computer indicated adequate core subcooling throughout the event. However, the steam void in the Vessel Head was not detected by the meters or by the Process Computer. Manual calculations performed by the operators during the event and based on RCS pressure and Upper Head temperatures clearly showed the existence of the void. In addition, the accuracy of the meter using inputs from the faulted loop was adversely affected by loop flow to the break.

7. Computer Logs

From about 11:19 a.m., to about 11:47 a.m., during the initial stages of the event, the P-250 Process Computer failed. This failure required the operators to take thermocouple temperatures manually and compute subcooling manually. In addition, the failure hindered the communication of plant parameters to the Technical Support Center and the Emergency Operations Facility. Also, during the computer outage permanent records of the event, with the exception of control room logs and instrument strip charts, were lost. Accurate reconstruction of the event was hindered because some significant actions such as Reactor Coolant Pump restart and "B" S/G safety lift(s) occurred during this time frame.

Overall, however, the operators were able to manage the event successfully despite the computer failure. Core subcooling was adequately monitored and maintained throughout the event.

8. "A" Containment Sump Indication

The "A" Containment Sump level is monitored with two channels of instrumentation indicating in the Control Room.

After the PRT rupture disc ruptured, Channel 1 indicated a sump level of about 5.3 feet (about 1900 gallons) and channel 2 indicated 9.3 feet (about 8000 gallons). The licensee reported sump level at about 8000 gallons based on the Channel 2 indication. Upon further investigation, the licensee's Instrumentation and Controls personnel verified the current signals from each level transmitter agreed; the voltage signals to the Control Room indicators also agreed. A static charge on the Channel 2 meter movement affected its reading. After this charge was removed, Channel 2 indicated 5.3 feet, in agreement with Channel 1. The actual volume of water in the sump determined after the sump was pumped dry was 1320 gallons.

9. "B" Steam Generator Safety Valve Trend Recorder

At Ginna, a chart recorder is assigned to each S/G for the purpose of monitoring S/G PORV and Safety lifts. These recorders automatically start on 30 mr/hr in the respective steam line. During the event, when it became obvious to the operators that the "B" S/G safety would lift, and steam line radiation was <30 mr/hr they manually started the "B" S/G recorder. The valve lifted at least two times; the total lift time was estimated to be two minutes. The recorder failed to indicate any valve lifts. The failure of this recorder to indicate valve lifts impaired the post-event calculation of the radiological release through the "B" S/G safety valve.

10. <u>The Following Equipment-Related Items Require Further Review By The Licensee And The NRC</u>:

   a. The effects of thermal stresses on the "B" Reactor Coolant Pump casing and the "B" Steam Generator lower head resulting from the rapid drop in "B" T-cold during the event.

   b. The effects of thermal stresses on the Reactor Vessel Head and the Vessel internals resulting from the Vessel's rapid temperature reduction during the event.

   c. The effects of thermal stresses on the Pressurizer internals resulting from their dryout and subsequent quenching during the event.

d. An inspection of "B" Steam Generator Safety Valves to determine the effects of any two phase flow which occurred during valve lifts.

III. Operator Performance

A. Response

Operator response to the event was reviewed based on a comparison of the event chronology with the Ginna Steam Generator Tube Rupture Procedure (E-1.4), on observations in the Control Room and on interviews with the operators.

The operators appeared to have followed the guidance of E-1.4 and to have managed the event well. It should be noted that the onshift crew was augmented during the event by members of other onsite departments, namely, technical support and training. In addition, after the TSC was manned, added assistance was provided to the Control Room staff which affected the event's outcome in a positive manner.

B. Operator Comments/Failures

The following items were identified during the review:

1. The operators felt the procedure was difficult to follow because, in attempting to handle a spectrum of SGTR events, it uses many detailed steps, some of which were not applicable to plant conditions during the event.

2. The operators were unsure of the validity of the Safety Injection Termination criteria with a steam void in the Vessel Head.

3. The operators needed better guidance on handling a SGTR on natural circulation and better guidance regarding Reactor Coolant Pump restarting during SGTR events.

4. The operators felt constrained to follow the emergency procedure verbatum even though the procedure did not adequately address the existent plant conditions. This feeling affected their decisions concerning Safety Injection termination and Reactor Coolant Pump restart.

5. The operators recognized the need to start a second Control Rod Drive Mechanism (CRDM) Shroud Fan, but were initially unable to do so because of an existing interlock. The interlock prevents starting the second CRDM fan when one is running. They were able to start this second fan after serving the first one and starting both simultaneously.

DRAFT NO. 2
Wiggins/dmg
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February 5, 1982

Procedure Review

Ginna Station Emergency Procedure E-1.4, Steam Generator Tube Rupture, Revision 10 and the Westinghouse generic procedure for SGTR were reviewed to assess their suitability for this specific event. The two procedures, reviewed generally, agree with respect to the overall strategy for handling an SGTR event. However, in two significant areas, these procedures disagree, i.e., the RCS pressure at which the RCP's should be tripped, and the method for cooling down the faulted steam generator. It should be noted that, basically, the Ginna procedure functioned adequately during this specific event. Core cooling was maintained throughout the event, and the plant was stabilized shortly after the event's initial transient effects.

Comments

The following comments are based on inspector observations and analysis during and following the event and on interviews with the plant operators. Comments are segregated into two sets: those generically applicable to both the Ginna and the Westinghouse procedures, and those specifically applicable to the Ginna procedures.

Generic Comments

1. The procedures cover the spectrum of break sizes from small leaks to major ruptures. In addition, the procedures address an SGTR with and without offsite power, with and without RCP's available, and with and without an onscale Pressurizer level. Inclusion of all these options in one procedure overly encumbers the procedure and makes it more difficult to follow.

2. More guidance could be provided for SGTR events where Pressurizer level is lost. The rapid loss of level and system pressure results in the securing of the RCP's subsequently forming a steam void in the Reactor Vessel Head.

3. More guidance could be provided for managing the event under natural circulation conditions with a steam void in the Vessel Head.

4. The procedures could more clearly address the effects of the steam void on Pressurizer level when actions are taken to reestablish this level.

5. More guidance could be provided regarding Safety Injection termination criteria when a steam void is present in the Vessel Head. Pressurizer level may not be an accurate measure of RCS inventory. The possible effects of void growth after SI termination should be discussed in the procedure.

6. RCP's are secured during the initial transient for large SGTR events. Restart of an RCP does not occur until after SI has been terminated. A review should be undertaken to establish criteria to allow RCP restart earlier in the event. RCP restart will collapse the void in the Vessel Head and provide a better cooldown method.

7. More guidance could be provided for cooling down the faulted S/G after a large break. Ginna used a feed and bleed operation, not described in the procedures, which minimized the total release. The Westinghouse procedure discussed venting the faulted S/G to atmosphere if 10 CFR 20 release limits would not be violated, but is silent if these limits would be exceeded.

8. Both procedures could provide clearer guidance regarding the control of the faulted S/G pressure. Use of the faulted S/G's atmospheric relief could be considered, rather than reliance on the S/G code safety valves to prevent overpressure conditions.

9. The procedures should include steam flow-feed flow mismatch as an indication of an SGTR.

Specific Comments

1. The Ginna procedure requires securing RCP's at $\leq$ 1715 psig vice a lower pressure calculated using the algorithm provided by Westinghouse.

The reason for selection of this higher pressure is the lack of a qualified pressure instrument capable of measuring pressures at the Westinghouse level.

2. The Ginna procedure is silent regarding the mechanism for cooling down the faulted S/G.

3. The Ginna procedure should specify the use of both Control Rod Drive Mechanism fans to cool the Vessel Head area to prevent or mitigate the formation of a steam void.

4. The Ginna procedure should address the use of the Upper Head thermocouples to detect the presence of a Vessel Head void.

Table I

Pressurizer Level (%) vs. Time

Normal	48
9:26:20	32.5
9:26:52	30.5
9:27:00	30.2
9:27:16	28.9
9:27:30	23.5
9:27:46	17.9
9:27:55	14.9
9:28:12	9.0 (Rx Trip)
9:29:00	4.9
9:31:00	[illegible]5
9:33:29	2.2
9:35:44	2.5
9:37:44	1.7
9:40:36	1.2
9:41:57	1.5
9:43:25	1.7
9:50:21	1.9
9:52:29	1.8
9:56:05	2.0
10:01:57	2.0
10:03:01	2.1
10:05:04	3.0
10:09:09	25.9
10:15:36	103.0
10:33:58	103.0
10:34:57	103.0
10:40:52	103.0
10:42:49	103.0
10:44:56	103.0
10:46:07	103.0
10:52:40	103.0
10:53:48	103.0
11:00:32	103.0
11:04:58	103.0
11:07:16	103.0
11:12:31	103.0
11:15:02	103.0
11:19:18	103.0
11:47:42	103.0
11:52:12	102.5
11:53:54	101.8
11:58:41	98.8

12:04:34	87.8
12:12:36	49.1
12:15:35	60.4
12:19:28	48.9
12:24:36	66.6
12:29:46	69.3
12:34:19	76.3
12:37:56	77.1
12:46:24	71.5
12:55:18	67.5
13:02:11	61.4
13:04:24	59.6
13:14:32	56.0
13:16:31	53.4
13:27:21	48.2
13:40:16	46.5

Table II

Reactor Coolant Pressure and T Sat vs. Time

Time	Pressure (PSIG)	T Sat (° F)
9:26:17	2102	644
9:28:09	1912	630
9:33:29	1296.2	578
9:35:44	1291.6	
9:37:44	1212.9	
9:39:01	1142.4	
9:39:52	1138.7	556
9:40:36	1156.1	
9:41:57	1195.5	568
9:43:25	1228.5	
9:50:21	1263.2	
9:51:25	1264.1	
9:52:29	1266.9	
9:56:05	1282.5	
9:59:12	1293.5	
10:01:57	1298.0	
10:03:01	1299.9	
10:05:04	1324.6	580
10:08:25	1182.6	
10:09:09	1048.1	
10:10:53	872.4	530
10:13:56	1158.0	
10:15:36	1261.4	
10:20:34	1379.5	595
10:22:04	1379.5	
10:25:04	1378.6	
10:32:50	1369.5	
10:34:57	1369.5	586
10:40:52	952.0	
10:43:49	931.9	
10:44:10	933.6	
10:46:07	941.0	
10:51:16	963.9	
10:52:40	965.7	
11:00:32	988.6	
11:04:58	998.6	
11:06:25	989.5	
11:07:16	984.9	
11:12:31	1012.4	
11:14:18	1023.4	
11:15:02	1029.7	
11:18:29	1045.4	

Table II

Time	Pressure (PSIG)	T Sat (° F)
11:19:18	1052.6	
11:47:42	984.9	
11:51:21	996.9	
11:52:12	1001.4	
11:53:54	1006.0	
11:59:13	1029.7	
12:04:34	1044.4	
12:11:51	1002.41	
12:12:36	988.6	
12:15:35	974.0	
12:16:53	947.4	
12:19:28	909.9	
12:24:36	934.6	
12:29:46	937.4	
12:33:37	943.7	
12:37:56	933.6	
12:39:12	934.6	
12:45:00	941.0	
12:46:24	944.6	
12:55:18	922.6	
13:02:11	899.7	
13:03:36	897.0	
13:04:24	894.2	
13:14:32	876.9	
13:15:49	875.0	
13:16:31	873.2	
13:27:21	848.5	

Table III

Steam Generator Level and Pressure vs. Time

Time	Level A	Level B	Pressure A	Pressure B
9:26:17	52.0	60.1	763.4	763.5
9:27:21	46.1	53.4	881.1	880.6
9:28:17	42.5	50.1	913.3	912.8
9:29:37	12.1	14.9	941.4	939.7
9:31:05	15.4	23.9	923.9	922.5
9:33:29	27.2	37.7	880.3	878.1
9:35:44	38.1	43.2	854.3	852.6
9:37:44	55.2	47.7	839.7	837.5
9:40:36	76.1	51.0	792.2	788.2
9:41:57	77.8	54.7	762.5	765.5
9:44:28	79.2	62.4	642.8	748.1
9:52:29	73.4	93.0	535.8	841.7
9:57:13	66.1	93.0	528.7	894.4
10:03:01	57.8	102.6	487.1	955.2
10:11:40	48.5	102.4	469.7	978.4
10:16:44	43.4	102.3	425.3	1042.8
10:20:34	44.8	102.7	404.4	1069.0
10:23:01	49.6	102.6	359.7	1064.0
10:26:23	54.5	102.7	331.6	1058.1
10:34:57	56.3	102.8	292.3	1024.7
10:40:52	52.9	102.8	278.3	841.0
10:42:49	51.9	102.8	279.2	863.5
10:46:07	49.5	102.8	287.3	885.3
10:53:48	43.9	102.8	300.8	917.9
10:57:33	41.5	102.8	305.0	929.1
11:00:32	42.3	102.8	309.0	939.7
11:04:58	44.2	102.9	298.6	952.1
11:10:49	41.1	102.8	268.3	966.7
11:12:31	41.0	102.9	273.1	978.1
11:19:18	46.9	102.9	236.7	1035.1
11:47:42	35.6	103.0	207.0	922.4
11:53:54	36.0	103.0	195.7	868.9
11:57:30	36.4	102.9	189	885.4
11:59:13	38.6	103.0	183	891.0
12:04:34	37.4	103.0	169.0	849.5
12:15:35	37.1	103.0	148.2	854.0
12:17:41	37.5	102.9	146.7	835.3
12:19:28	36.9	103.0	147.2	840.3
12:24:36	39.6	103.0	136.3	854.4
12:29:46	42.8	103.0	132.9	962.5

Table III 2

Time	Level A	Level B	Pressure A	Pressure B
12:35:21	45.9	102.9	131.6	958.3
12:37:56	47.6	103.0	129.8	955.2
12:50:48	46.3	102.9	134.9	961.2
12:55:18	42.5	103.0	130.9	939.7
13:02:11	38.9	103.0	130.1	920.1
13:09:19	33.9	102.9	133.7	913.1
13:[illegible]:32	35.0	103.0	130.4	895.6
13:19:20	32.7	102.9	132.1	886.1
13:22:06	31.9	103.0	130.4	867.9
13:40:16	31.6	103.9	129.9	858.5

Table IV

Upper Head Average Thermocouple Temperature (° F)

Normal	590.5
9:53:56	559.0
10:47:31	540.8
11:03:07	526.4
11:55:25	397.7
12:13:44	382.4
12:30:22	371.4
12:47:28	369.9
13:07:32	369.2
13:17:31	369.0
13:47:36	366.9

Table V

Core Exit Average Thermocouple Temp. (° F)

Representative Avg.	598.2
9:39:01	531.8
9:39:52	523.7
9:51:25	504.6
9:53:56	501.3
9:59:12	496.6
10:08:25	483.9
10:10:53	465.4
10:13:56	457.1
10:25:04	459.1
10:32:50	447.7
10:44:10	451.9
10:47:31	452.8
10:51:41	454.2
11:03:07	457.6
11:06:25	457.1
11:14:18	449.7
11:18:29	448.0
11:51:21	398.0
11:55:25	394.8
12:11:51	381.6
12:13:44	379.4
12:16:53	376.1
12:31:35	369.8
12:33:37	368.6
12:39:12	367.9
12:45:00	368.6
12:47:28	369.2
13:03:36	368.4
13:07:32	368.7
13:15:49	368.1
13:17:31	368.2
13:45:53	366.1

Table VI

Cold Leg Temperatures vs. Time

Time	A Tc	B Tc	
9:26:18	551.1	546.1	
9:27:14	559.0	553.9	
9:28:10	558.8	554.1	
9:29:46	547.8	543.6	
9:31:06	545.1	541.6	
9:33:29	535.2	535.7	
9:35:44	527.3	532.4	
9:37:44	513.0	530.6	
9:39:01	492.0	529.7	
9:39:52	485.7	528.2	
9:40:36	483.4	524.2	
9:41:57	485.6	494.9	
9:51:25	491.1	385.6	
9:52:29	488.2	374.8	
9:59:12	483.6	361.2	
10:03:01	478.6	353.9	
10:08:25	473.3	318.2	1182
10:10:53	472.7	286.0	
10:13:56	460.7	285.1	872 PSI
10:20:34	455.2	348.8	
10:22:04	454.6	345.3	
10:25:04	446.2	344.6	
10:32:50	432.7	329.2	
10:34:57	430.7	325.0	
10:40:52	425.9	381.0	
10:42:49	424.2	371.5	
10:44:10	423.5	370.4	
10:46:07	423.9	369.3	
10:51:41	427.2	370.6	
10:53:48	427.5	369.7	
11:00:32	429.9	376.1	
11:04:58	429.7	371.9	
11:06:25	429.7	374.6	
11:12:31	420.0	366.7	
11:14:18	420.2	369.5	
11:18:29	417.8	365.1	
11:19:18	415.4	362.9	
11:47:42	405.4	402.4	
11:51:21	402.8	399.3	
11:53:54	400.6	397.7	
11:59:13	396.6	393.5	
12:04:34	390.4	387.1	
12:11:51	386.3	383.0	

Time	A Tc	B Tc
12:15:35	381.4	377.6
12:16:53	380.5	377.0
12:19:28	380.1	376.8
12:24:36	374.4	371.0
12:29:46	372.8	369.9
12:33:37	373.2	369.1
12:37:56	372.1	368.4
12:39:12	372.6	368.6
12:45:00	372.8	368.9
12:55:18	373.0	368.6
13:02:11	372.2	368.2
13:03:36	372.2	368.6
13:14:32	372.8	369.5
13:15:49	372.1	368.9
13:27:21	371.7	368.6
13:40:16	371.0	367.8

Table VII

PZR Relief Tank Level, Press, and Temp vs. Time

Time	Level	Press	Temp
9:43:25	68.5	1.7	89.7
9:50:21	68.7	1.7	89.7
9:56:05	69.0	2.0	89.7
10:01:57	69.2	2.0	89.7
10:05:04	69.2	1.7	89.7
10:09:09	71.0	12.0	103.0
10:15:36	75.5	30.5	140.5
10:33:58	85.0	50.5	140.0
10:44:56	90.0	50.5	139.5
10:52:40	90.2	-.5	132.5
11:07:16	83.5	-.5	132.0
11:15:02	87.2	-.5	131.7
11:52:12	90.5	.2	129.7
12:12:36	91.0	.2	128.7
12:34:19	90.7	.2	128.0
12:46:24	90.7	.2	127.2
13:04:24	90.7	0.0	126.7
13:16:31	91.0	.2	126.2
13:46:35	91.0	0.0	125.2
14:14:16	91.0	0.0	124.5
14:40:54	91.0	0.0	123.5
15:15:20	91.2	0.0	122.7

Table VIII

PZR Water, Steam, and Surge Line Temp

Time	Water	Steam	Surge Line
9:33:29	594	612.2	
9:35:44	600.4	611.6	
9:37:44	602.4	612.0	
9:40:36	[illegible]01.7	611.6	
9:41:57	605.8	611.6	
9:43:25	608	610.9	542.3
9:50:21	611.3	610.3	518.0
9:52:29	611.1	610.3	
9:56:05	612.2	610.5	509.4
10:01:57	611.8	610.5	501.7
10:03:01	610.9	610.5	
10:05:04	588.9	610.1	501.1
10:09:09	516.9	609.6	
10:15:36	505.8	556.9	484.2
10:20:34	507.5	558.6	
10:33:58	512.2	548.5	474.6
10:34:57	512.4	547.9	
10:40:52	513.7	541.7	
10:42:49	513.5	537.2	
10:44:56	514.2	536.2	496.0
10:46:07	515.7	535.5	
10:52:40	527.8	537.7	496.2
10:53:48	528.2	537.7	
11:00:32	527.4	540	
11:04:58	532.1	542.3	
11:07:16	534.2	542.1	488.7
11:12:31	539.6	539.3	
11:15:02	542.1	538.3	461.6
11:19:18	544.3	537.2	
11:47:42	553.9	541.7	
11:52:12	555.2	542.6	544.1
11:53:54	555.8	543.0	
11:59:13	557.9	544.9	
12:04:34	560.3	546.8	
12:12:36	556.7	542.8	547.7
12:15:35	554.9	540.8	
12:19:28	548.7	534.9	
12:24:36	551.1	536.6	
12:29:46	548.3	534.9	
12:34:19	550.2	536.4	393.0
12:37:56	549.4	535.9	
12:46:24	550.5	536.4	533.8
12:55:18	549.0	534.2	

Table VIII

Time	Water	Steam	Surge Line
13:02:11	546.4	532.3	
13:04:24	546.4	531.4	535.7
13:14:32	544.5	530.2	
13:16:31	543.8	529.7	534.4
13:27:21	540.6	526.5	
13:40:16	538.9	525.2	
13:41:40	539.3	525.0	
13:46:40	538.7	524.8	528.0

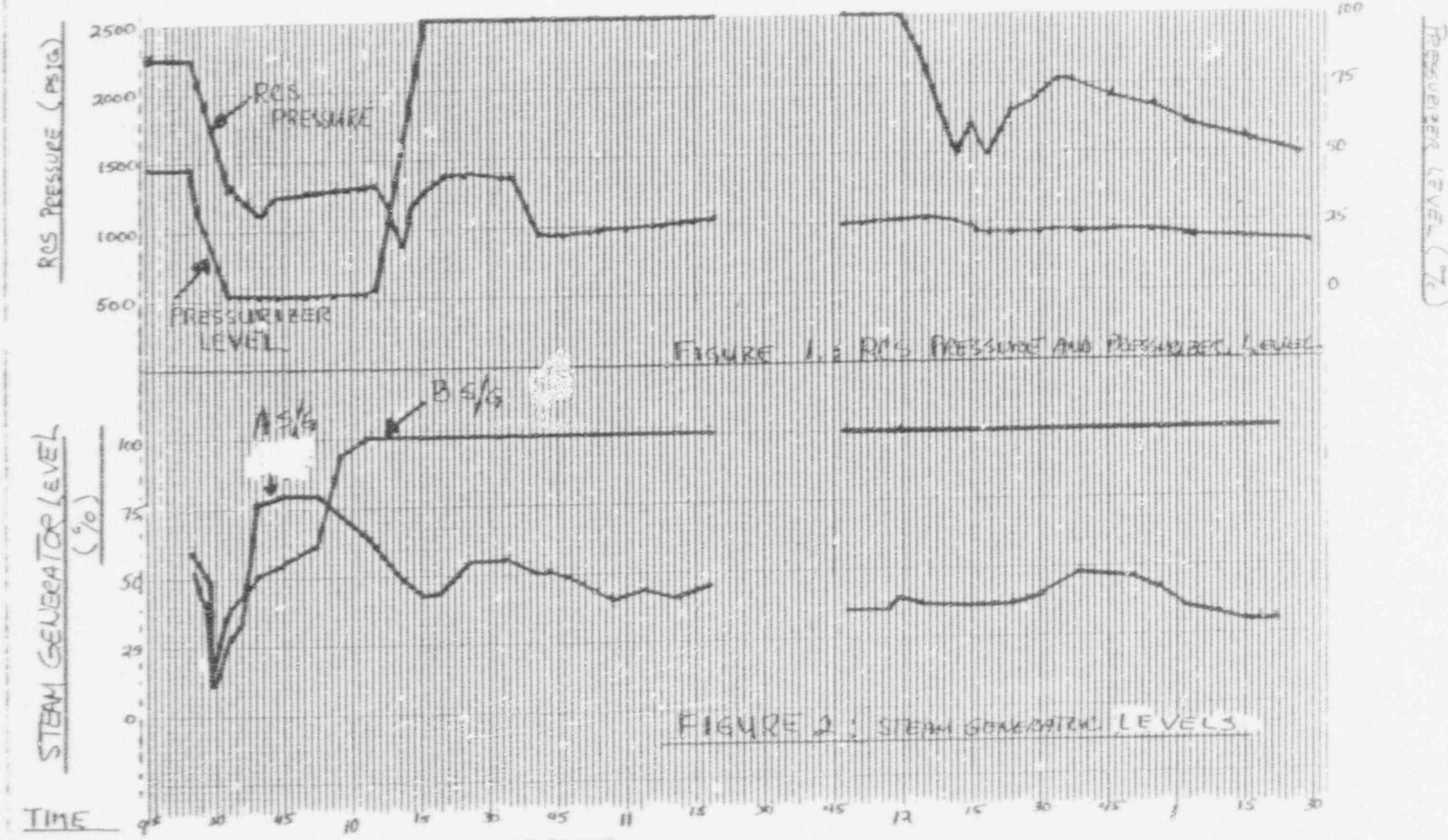
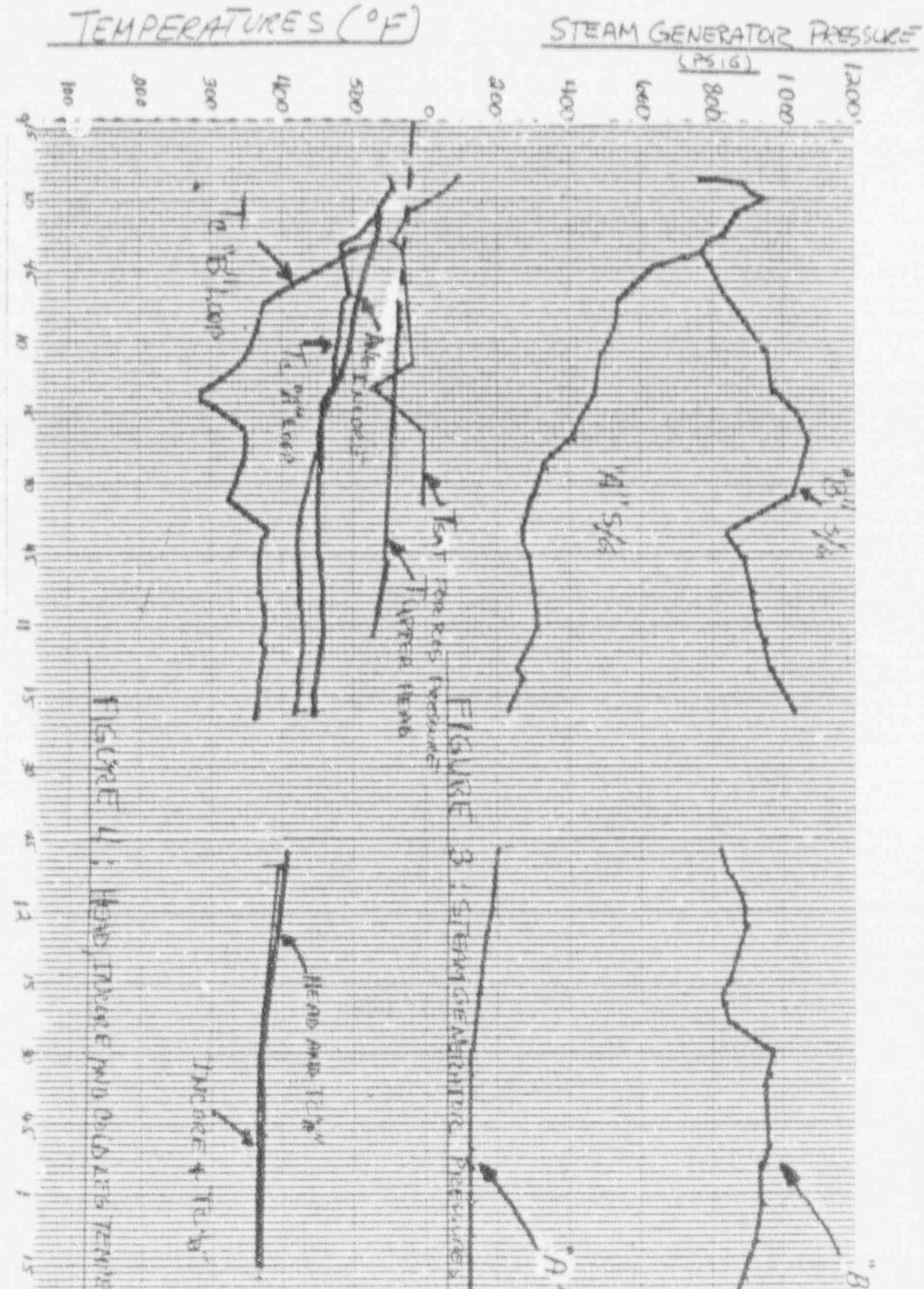


FIGURE 1.2 RCS PRESSURE AND PRESSURIZER LEVEL

FIGURE 2 : STEAM GENERATOR LEVELS


TEMPERATURES (°F)
STEAM GENERATOR PRESSURE (PSIG)
100
200
300
400
500
0
200
400
600
800
1000
1200
Tc "B" LOOP
Tc "A" LOOP
AVG INCORE
TSAT FOR RCS PRESSURE
T UPPER HEAD
"A" S/G
"B" S/G
FIGURE 3: STEAM GENERATOR PRESSURES
FIGURE 4: HEAD, INCORE AND CALCULATED TEMPERATURES
HEAD AND Tc "A"
INCORE + Tc "B"
"A" S/G
"B" S/G
TIME

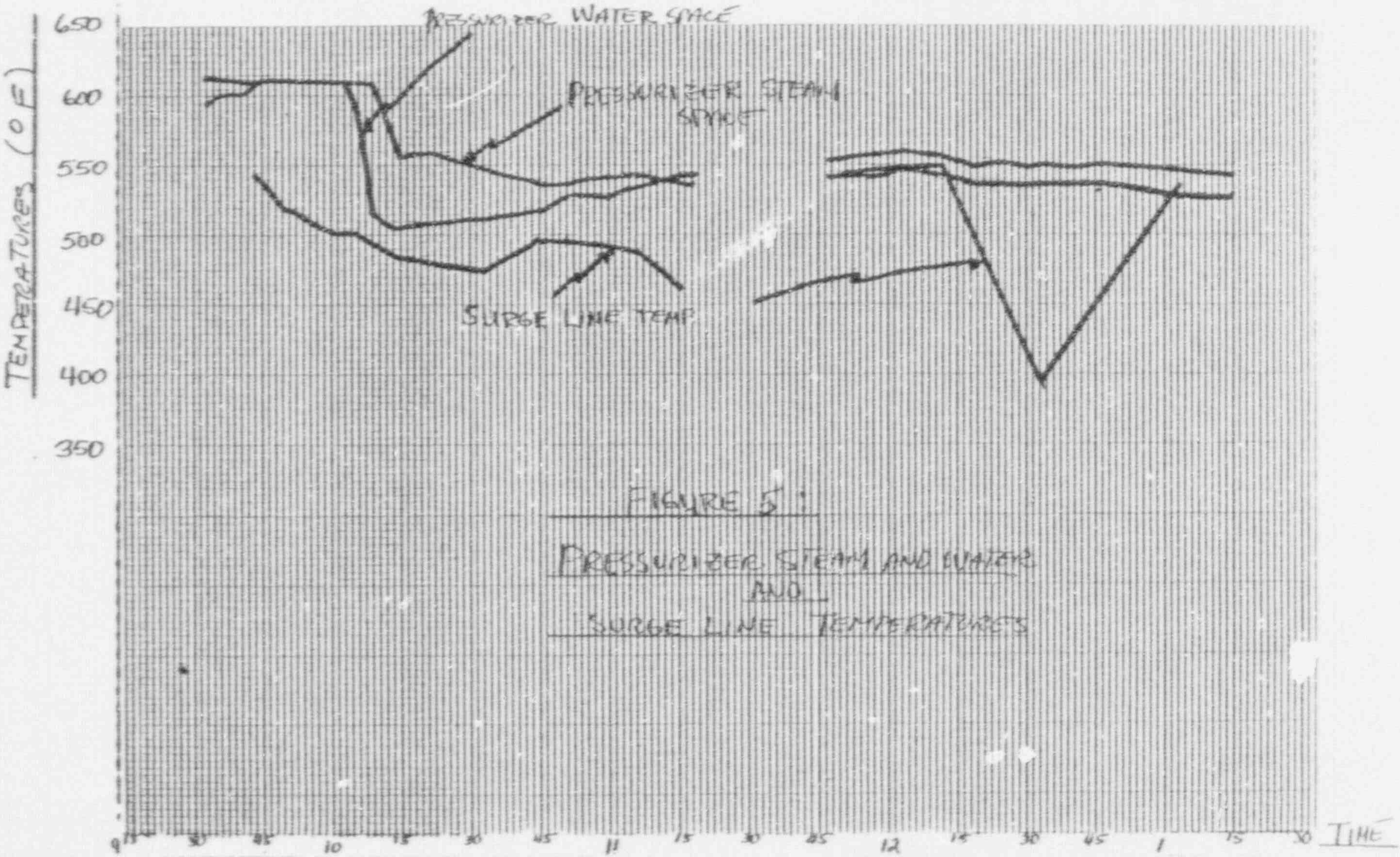


FIGURE 5:
PRESSURIZER STEAM AND WATER
AND
SURGE LINE TEMPERATURES

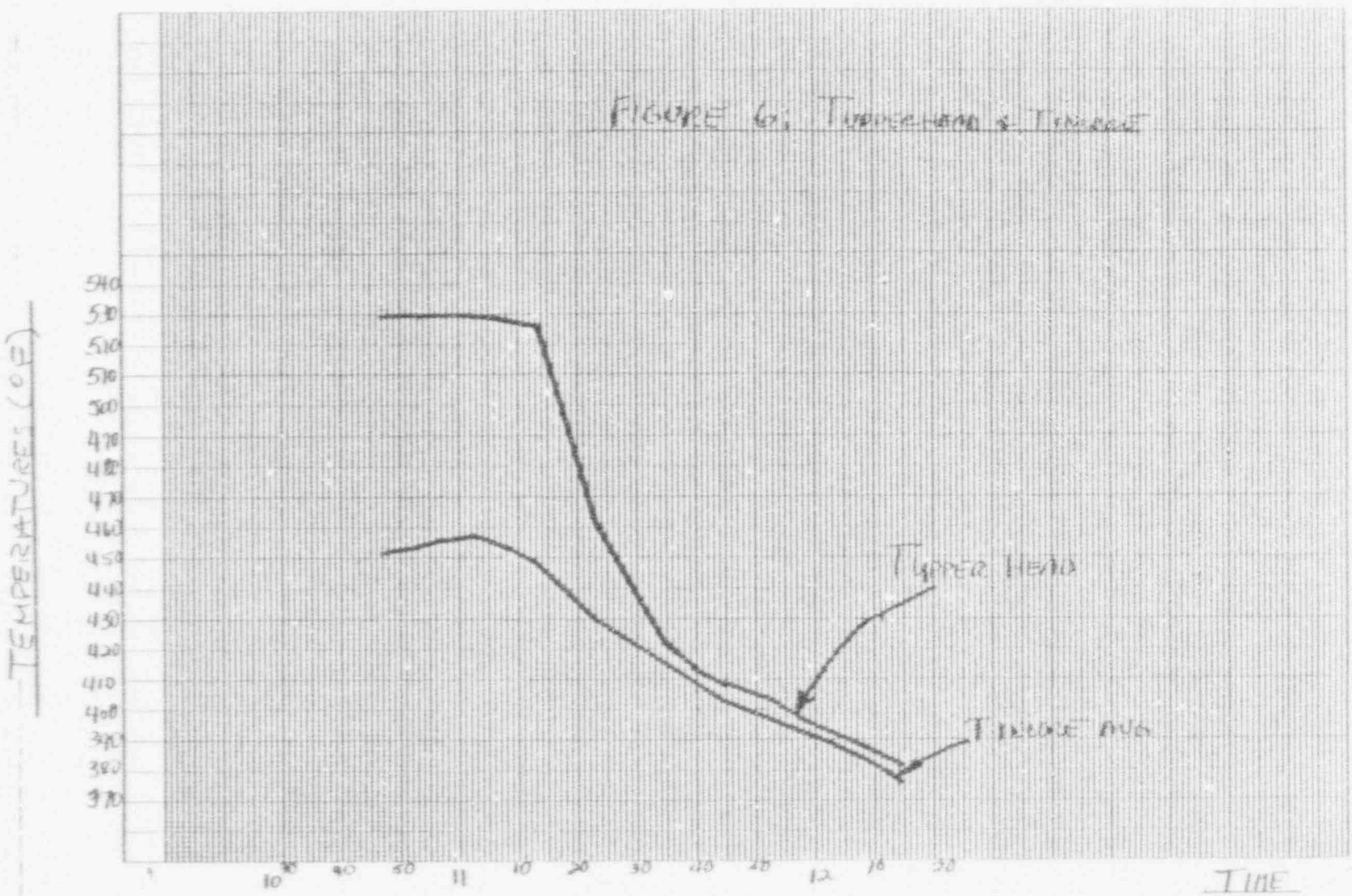
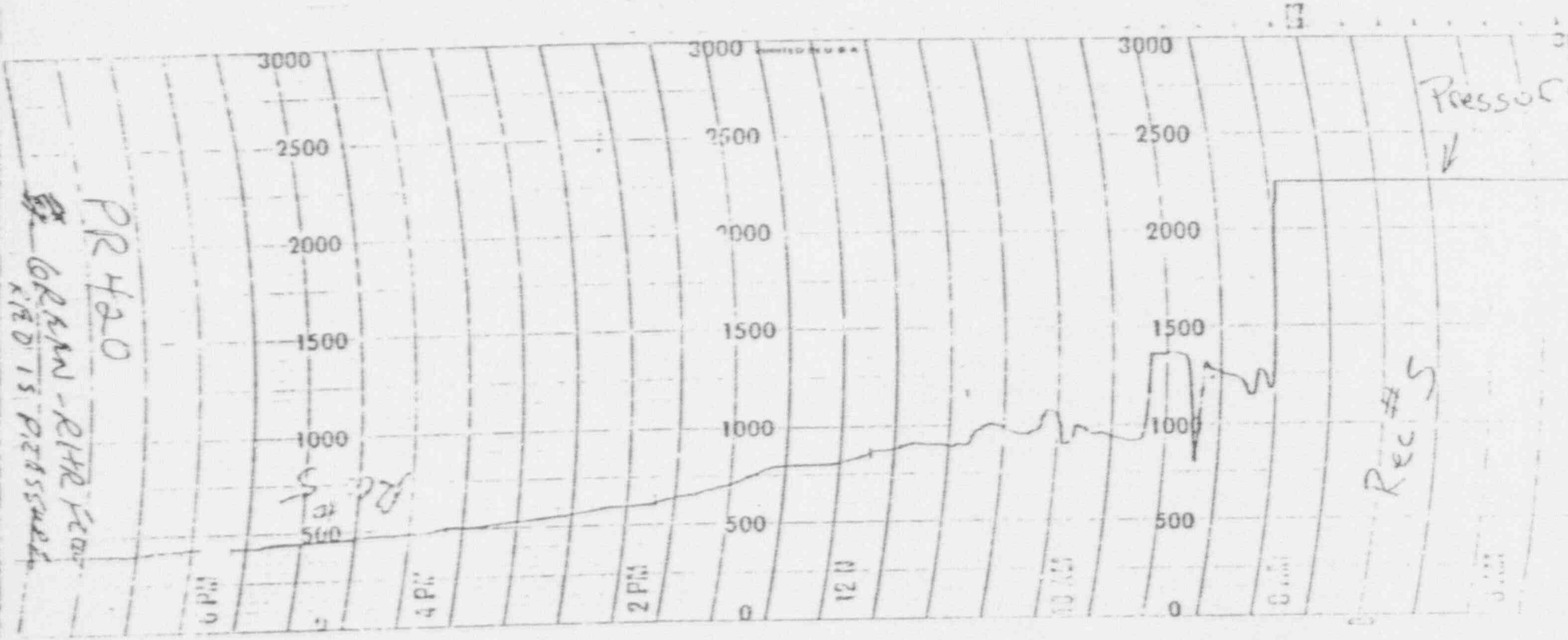
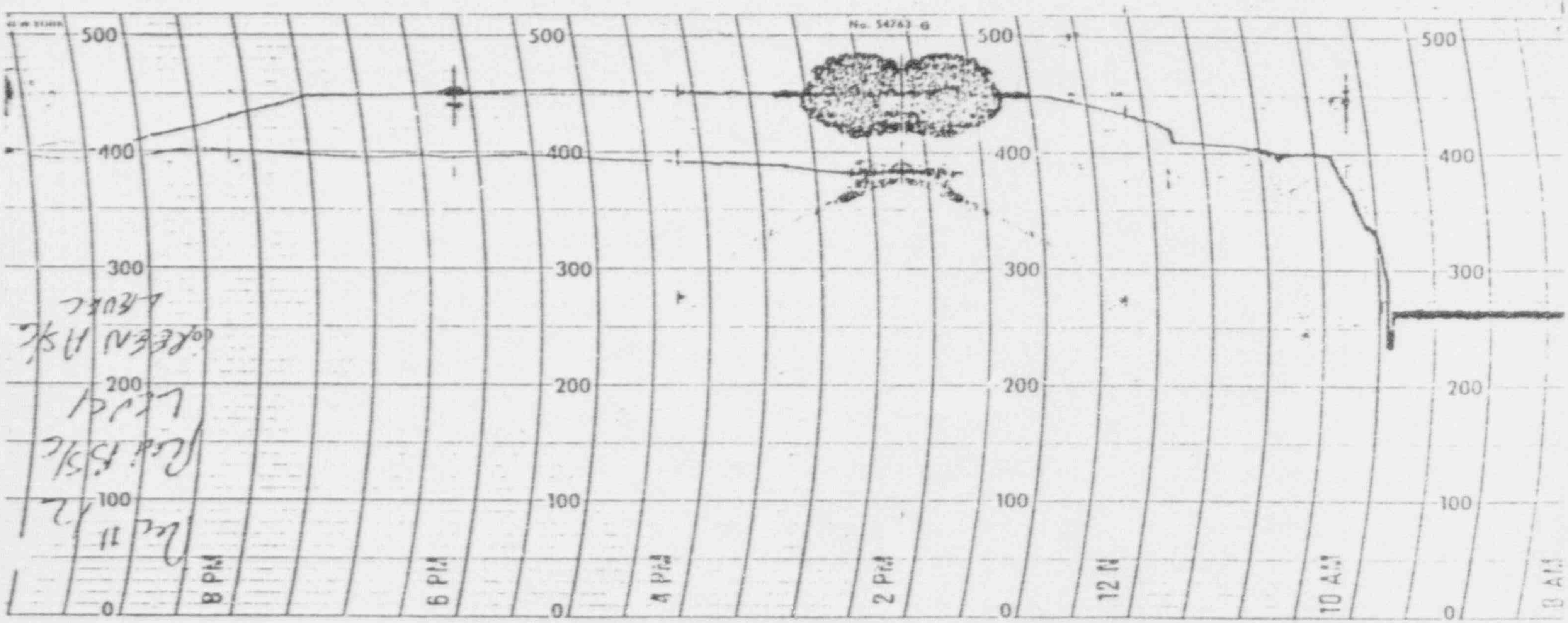
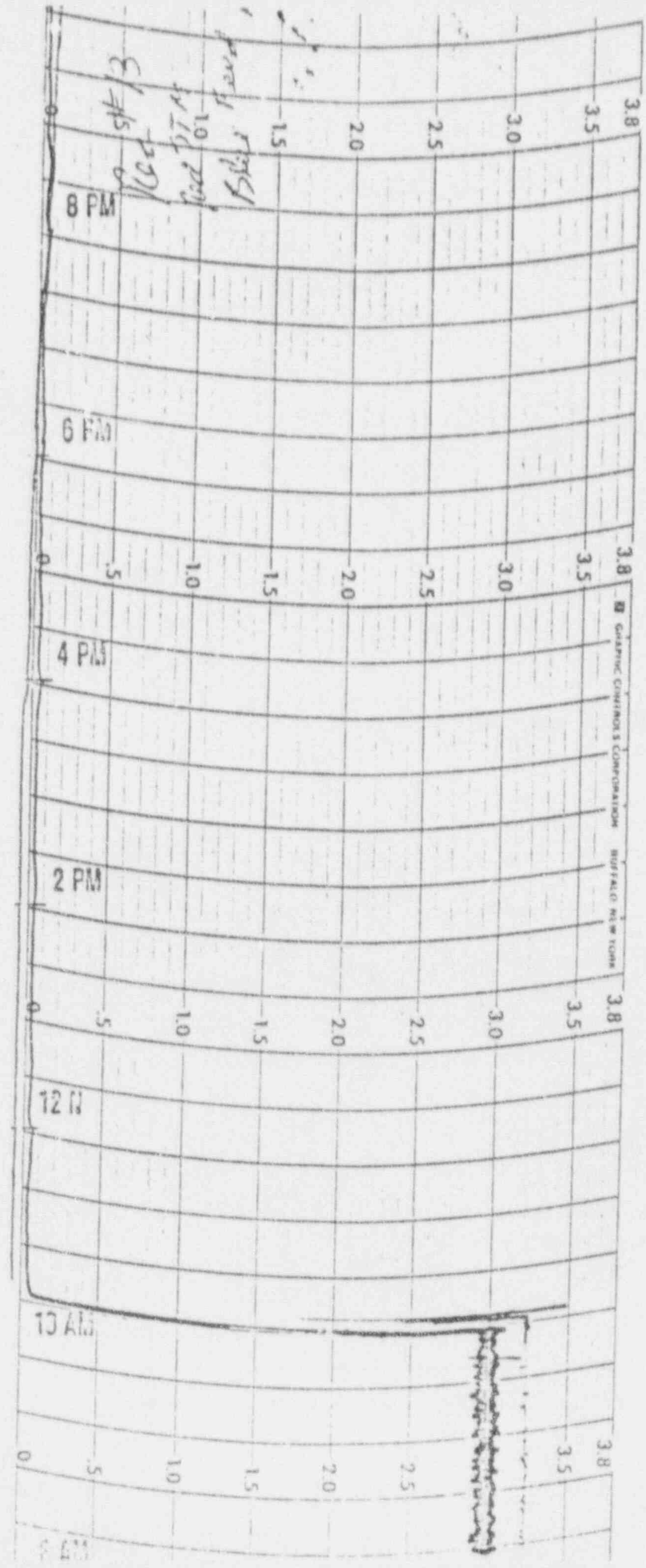


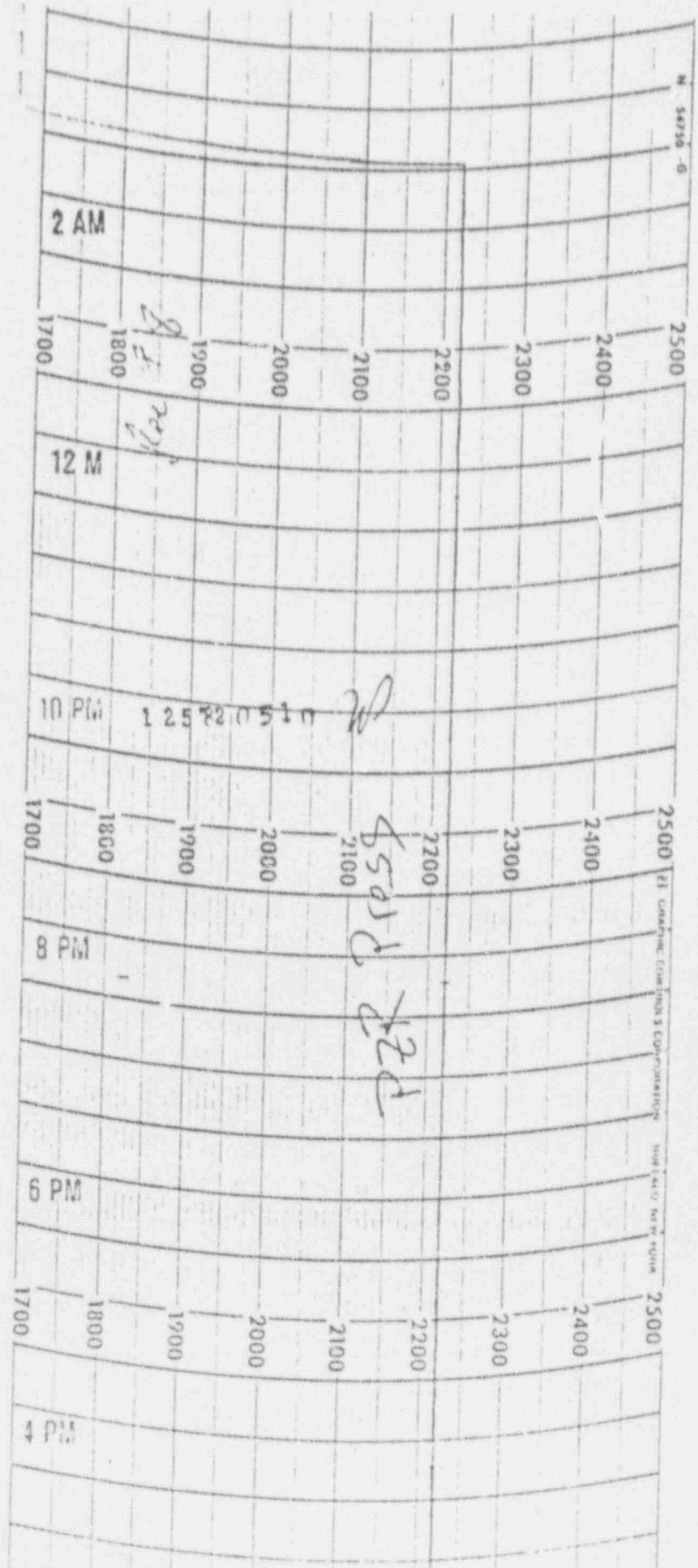
FIGURE 6: TUPPERHEAD & TINCORE

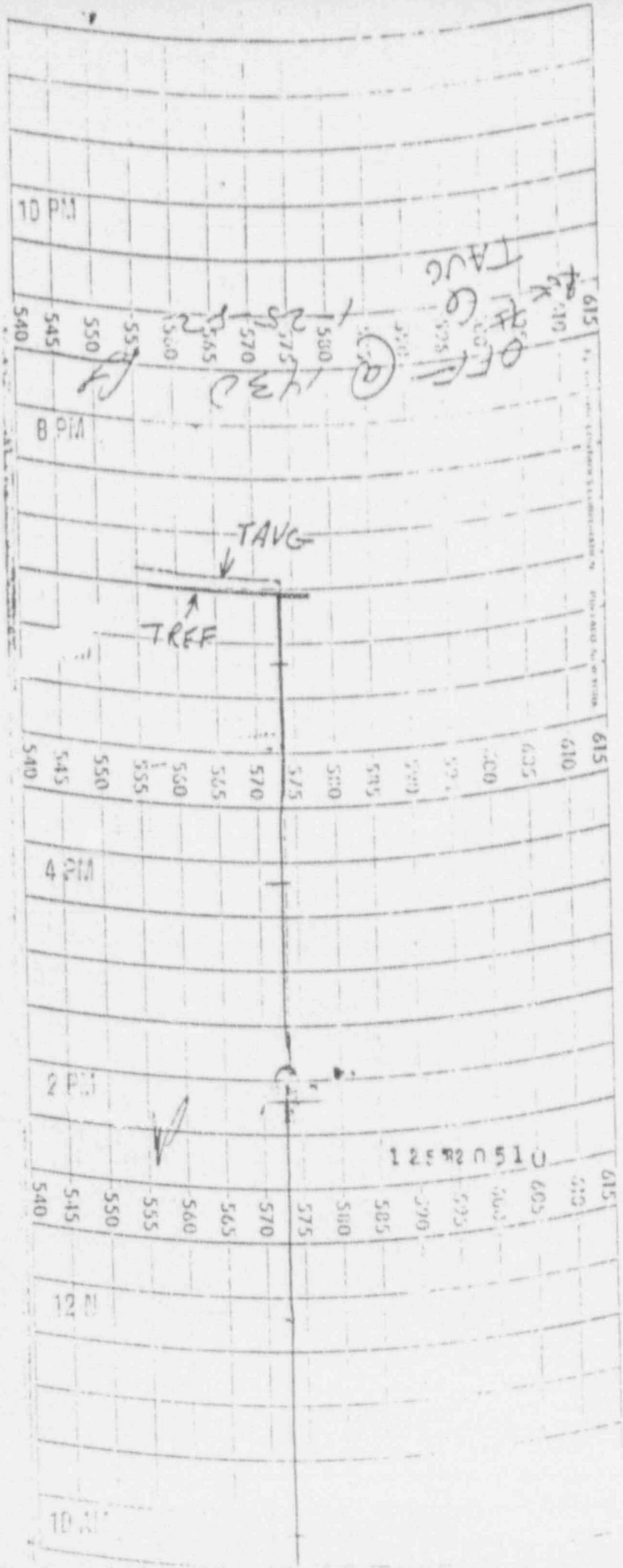

PR H2O
3000
2500
2000
1500
1000
500
0
6 PM
4 PM
2 PM
12 N
Rec #5
Pressur

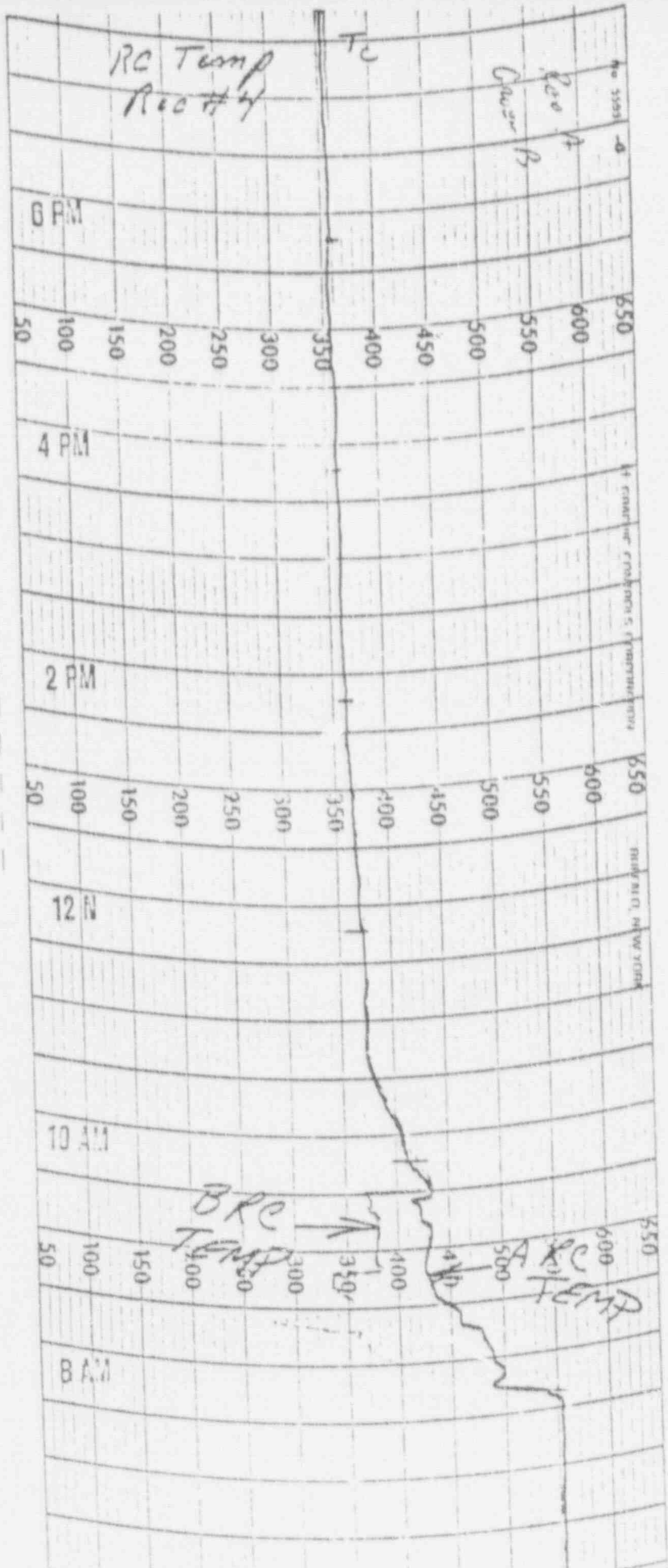

No. 54763-G
500
400
300
200
100
0
8 PM
6 PM
4 PM
2 PM
12 N
10 AM
8 AM



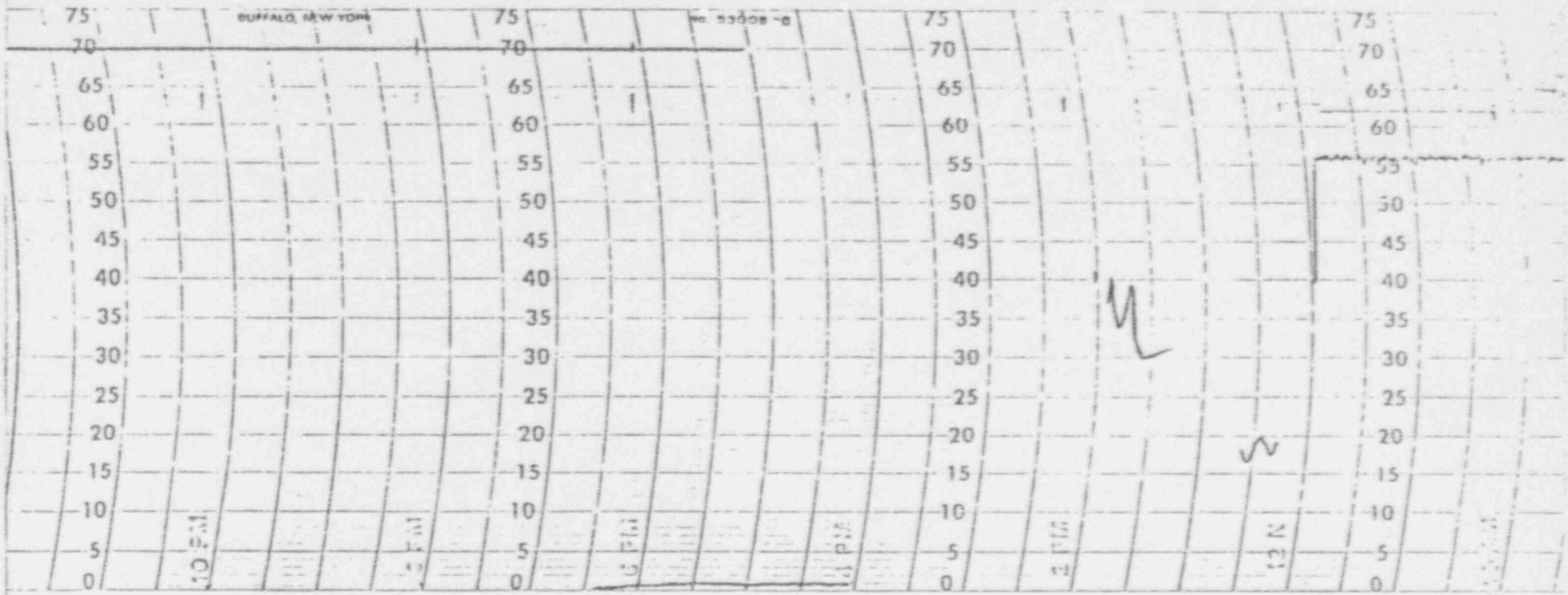
0
.5
1.0
1.5
2.0
2.5
3.0
3.5
3.8
8 PM
6 PM
4 PM
2 PM
12 N
10 AM
GRAPHIC CONTROLS CORPORATION BUFFALO NEW YORK

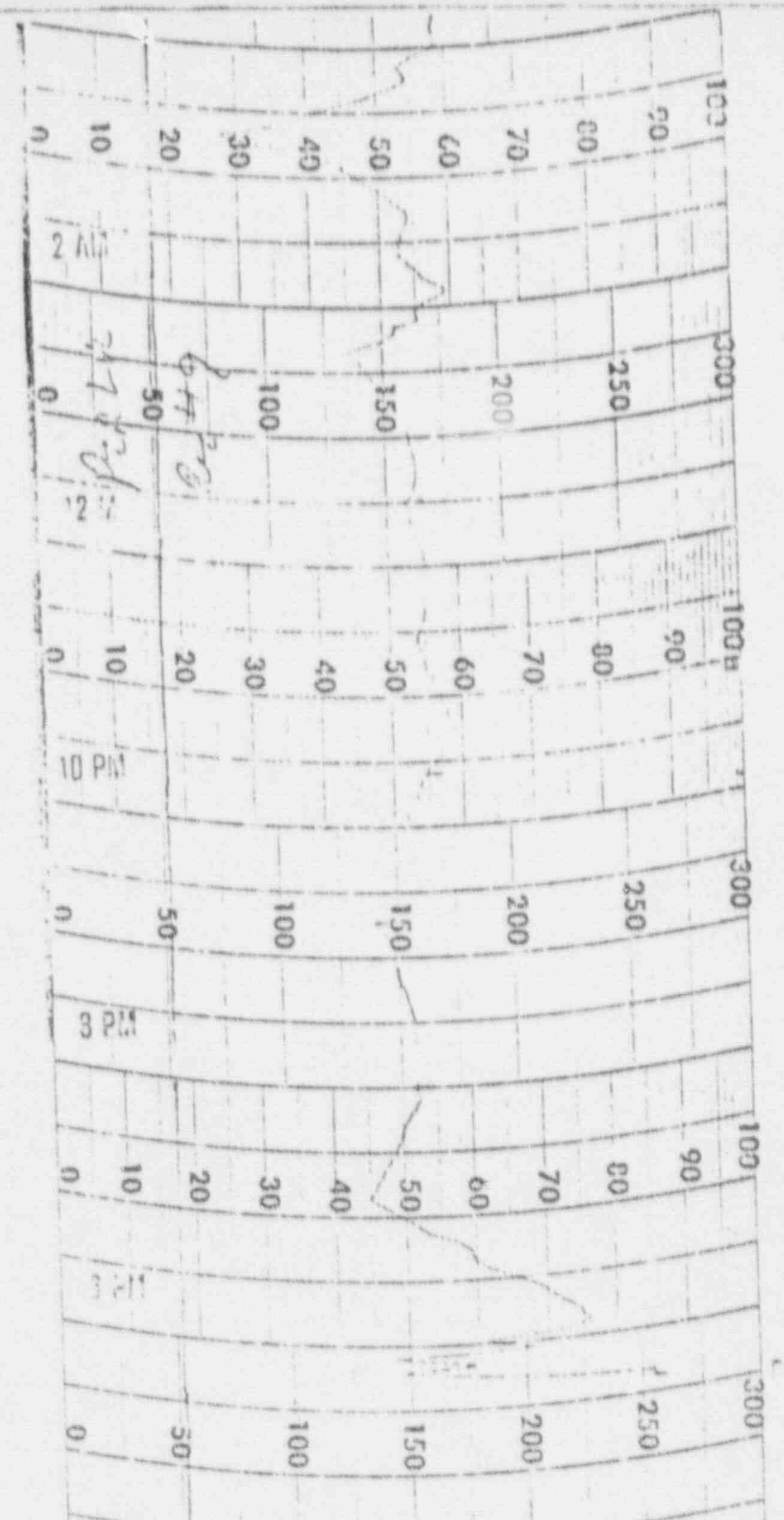

2 AM
12 M
10 PM
8 PM
6 PM
4 PM
1700
1800
1900
2000
2100
2200
2300
2400
2500


10 PM
8 PM
TAVG
TREF
540 545 550 555 560 565 570 575 580 585 590 595 600 605 610 615
4 PM
2 PM
12 N
10 AM


RC Temp
Rec #4
Tc
6 PM
50 100 150 200 250 300 350 400 450 500 550 600 650
4 PM
2 PM
50 100 150 200 250 300 350 400 450 500 550 600 650
12 N
BUFFALO, NEW YORK
10 AM
B RC TEMP
A RC TEMP
50 100 150 200 250 300 350 400 450 500 550 600 650
8 AM
6 AM

A Δt BAUMACK




2 AM
12 M
10 PM
3 PM