Exhibit 4 PDR

85

ACTIt ITEM LIST
FROM 6/9/85 Rx TRIP

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
1.	AFW Pump Turbine Governor Control				
a.	#1 AFWPT - 1. Establsh cause(s) of inadvertent trip (CAL Item 3a) 2. Determine and implement corrective action to prevent recurrence (CAL Item 3b)	Wilczynski	Hartigan Missig MPR		
b.	#2 AFWPT - 1. Establish cause(s) of inadvertent trip (CAL Item 3a) 2. Determine and implement corrective action to prevent recurrence (CAL Item 3b)	Wilczynski	Hartigan Missig MPR		
c.	Auto/Essential Control Problems	Jain	Yarger		
d.	Trip Throttle Valve Problem	Gradomski	Terry Turbine		
2.	SG Integrity/Cycle Impact Due To SUFP Initiation of Water • Evaluation of thermal shock considerations on both S/G's (CAL Item 5) • Maximum S/G shell differential temperature (CAL Item 5)	Chen	B&W	N/A	Preliminary info. from B&W indicates no problems. Draft report due from B&W on 6/19/85.

Rev. 0 - 6/11/85
Rev. 1 - 6/14/85
Rev. 2 - 6/18/85

ACTI‹ ITEM LIST
FROM 6/9/85 Rx TRIP

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
3.	Actions of Operators Adequacy of Procedures	O'Connor	Derivan		
4.	Classification of the Event Under the Emergency Plan (EAL Pg. 3 of 37) • Basis for event classification (CAL Item 5)	Scott-Wasilk	O'Connor		Complete 6/13/85 (A85-204H)
5.	SFRCS Half Trip On One Level Input	Yarger	Stalter		Problem appears to be analytical, new bistables and response to secondary pressure upset after turbine trip.
6.	SFRCS Alarms	Miller	Lingen-felter		
7.	MSIV/SFRCS Response • Establish cause(s) of unexpected valve closure (CAL Item 2a) • Determine and implement corrective action (CAL Item 2b) • Conduct testing to ensure operating as required (CAL Item 2c)	Miller	Jain Yarger		MSIV's may have properly responded to low level SFRCS trip which cleared quickly. Related to Item 5.

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Rev. 1 - 6/14/85
Rev. 2 - 6/18/85

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
8.	MFP Control System • Establish cause of inadvertent trip of #1 MFP (CAL Item 4a) • Determine and implement corrective action to prevent recurrence for both MFP's (CAL Item 4b)	Blay	J.Johnson Missig Topor Isley G.E.		Action plan ready for NRC FFT on 6/18/85.
9.a.	Damaged turbine bypass valve	Raynes	Hiss Lammon		
b.	Water hammer indication during transient				
10.	PORV Condition - Cycled 3 Times (3rd cycle looked inconsistent) # of Stress Cycles	Isley	Straube Marley		
11.	Discuss Event With Ottawa County Commissioners (Significance)	Scott-Wasilk		N/A	Completed - 6/11/85 at 1:00 p.m. (A85-203H)
12.	S/G Isolation Valves Capability To Open Against Large D/P (AFW 599 & 608)	Long	Bajestani	6/14-Action plan approved for troubleshooting.	Troubleshooting expected to begin 6/18/85.

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Rev. 2 - 6/18/85

ACTI ITEM LIST
FROM 6/9/85 Rx TRIP

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
13.	Service Water Effect on S/G	Briden			No detectable impact to S/G's was found - to be documented.
14.	Operator Error for Initiating AFW				
a.	HED on switches (FCR	Calcamuggio	Czuba Batch		
b.	Plastic covers	E.Johnson			
15.	Source Range NI's (RPS)				Action plan to be ready for NRC FFT on 6/18/85.
a.	NI-1	Borysiak			
b.	NI-2	Isley			
16.	MSSV's - Blowdown to 900 #'s	Huston	Mahoney		
17.	Problems With SPDS	Smith			Item removed from equipment freeze list.
18.	SP7A Problems (S/U Feedwater)	Uebbing	Bonner		Valve appeared to have opened during the transient. May however, be a panel light problem.
19.	Adequacy of Information as Originally Provided to NRC on June 9, 1985 (CAL Item 5)	MacDonald	Wideman		

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#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
20.	Complete Items 1-5 of CAL. Obtain concurrence of Regional Administrator Prior to Restart, Mode 2 (CAL Item 6)	Wood	Wideman		
21.	Perform Testing and Demonstrate that the AFP's Will Operate as Required (CAL Item 7a)	Topor	Blay G.E.		
22.	Perform Testing and Demonstrate that the MFP's Will Operate as Required (CAL Item 7b)	Gradomski	Hartigan Missig		
23.	Appropriate Test Results, Conclusions and Corrective Actions Shall be Reported to the NRC Resident Inspector (CAL Item 7c)	Wood	Wideman		
24.	Obtain Concurrence from Regional Administrator Prior to Exceeding 5% Reactor Power	Wood	Wideman		
25.	Main Steam Walkdown (Outside Containment) • Document any walkdown performed • prior to June 12 approx. 5:00 p.m.	Hiss	Raynes Dieterich		Completed - info. given to NRC.

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
25. Cont.	• QC - Conduct a damage inspection walkdown of the main steam system.	Rhodes			QC walkdown completed 6/14/85 per QPIC-037. Bechtel to review and disposition results started 6/18/85.
26.	Review service water transfer of suction to AFW pumps. • Determine if actions were appropriate. • Determine corrective actions if appropriate.	Czuba	Yarger		
27.	MS 106 - investigate seal in open circuit.	Bonner			

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ACTIC .TEM LIST
FROM 6/9/85 Rx TRIP

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
1.	AFW Pump Turbine Governor Control				
a.	#1 AFWPT - 1. Establsh cause(s) of inadvertent trip (CAL Item 3a) 2. Determine and implement corrective action to prevent recurrence (CAL Item 3b)	Wilczynski	Hartigan Missig		
b.	#2 AFWPT - 1. Establish cause(s) of inadvertent trip (CAL Item 3a) 2. Determine and implement corrective action to prevent recurrence (CAL Item 3b)	Wilczynski	Hartigan Missig		
c.	Auto/Essential Control Problems	Jain	Yarger		
d.	Trip Throttle Valve Problem	Gradomski			
2.	SG Integrity/Cycle Impact Due To SUFP Initiation of Water • Evaluation of thermal shock considerations on both S/G's (CAL Item 5) • Maximum S/G shell differential temperature (CAL Item 5)	Chen	B&W		

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
3.	Actions of Operators	O'Connor	Derivan		
4.	Classification of the Event Under the Emergency Plan (EAL Pg. 3 of 37) • Basis for event classification (CAL Item 5)	Scott-Wasilk	O'Connor		
5.	SFRCS Half Trip On One Level Input	Yarger	Stalter		
6.	SFRCS Alarms	Miller	Lingen-felter		
7.	MSIV/SFRCS Response • Establish cause(s) of unexpected valve closure (CAL Item 2a) • Determine and implement corrective action (CAL Item 2b) • Conduct testing to ensure operating as required (CAL Item 2c)	Miller	Jain Yarger		

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
8.	MFP Control System • Establish cause of inadvertent trip of #1 MFP (CAL Item 4a) • Determine and implement corrective action to prevent recurrence for both MFP's (CAL Item 4b)	Blay	J.Johnson Missig Topor Isley G.E.		
9.a.	Damaged turbine bypass valve	Raynes	Hiss Lammon		
b.	Water hammer indication during transient				
10.	PORV Condition - Cycled 3 Times (3rd cycle looked inconsistent) # of Stress Cycles	Isley	Straube Marley		
11.	Discuss Event With Ottawa County Commissioners (Significance)	Scott-Wasilk			Completed - 6/11/85 at 1:00 p.m. (A85-203H)
12.	S/G Isolation Valves Capability To Open Against Large D/P (AFW 599 & 608)	Long	Bajestani		
13.	Service Water Effect on S/G	Briden			

ACTIC .TEM LIST
FROM 6/9/85 Rx TRIP

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
14.	Operator Error for Initiating AFW				
a.	HED on switches (FCR	Calcamuggio	Czuba Batch		
b.	Plastic covers	E.Johnson			
15.	Source Range NI's (RPS)				
a.	NI-1	Borysiak			
b.	NI-2	Isley			
16.	MSSV's - Blowdown to 900 #'s	Huston	Mahoney		
17.	Problems With SPDS	Smith			
18.	SP7A Problems (S/U Feedwater)	Yarger	Bonner		
19.	Adequacy of Information as Originally Provided to NRC on June 9, 1985 (CAL Item 5)	MacDonald	Wideman		
20.	Complete Items 1-5 of CAL. Obtain concurrence of Regional Administrator Prior to Restart, Mode 2 (CAL Item 6)	Wood	Wideman		

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
21.	Perform Testing and Demonstrate that the AFP's Will Operate as Required (CAL Item 7a)	Topor	Blay G.E.		
22.	Perform Testing and Demonstrate that the MFP's Will Operate as Required (CAL Item 7b)	Grad.	Hartigan Missig		
23.	Appropriate Test Results, Conclusions and Corrective Actions Shall be Reported to the NRC Resident Inspector (CAL Item 7c)	Wood	Wideman		
24.	Obtain Concurrence from Regional Administrator Prior to Exceeding 5% Reactor Power	Wood	Wideman		
25.	Main Steam Walkdown (Outside Containment) • Document any walkdown performed • prior to June 12 approx. 5:00 p.m. • QC - Conduct a damage inspection walkdown of the main steam system.	 Hiss Rhodes	 Raynes Dieterich		

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
26.	Review service water transfer of suction to AFW pumps. • Determine if actions were appropriate. • Determine corrective actions if appropriate.	Marley			

(18)

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
1.	AFW Pump Turbine Governor Control				
a.	#1 AFWPT - 1. Establsh cause(s) of inadvertent trip (CAL Item 3a) 2. Determine and implement corrective action to prevent recurrence (CAL Item 3b)	Wilczynski	Hartigan Missig		
b.	#2 AFWPT - 1. Establish cause(s) of inadvertent trip (CAL Item 3a) 2. Determine and implement corrective action to prevent recurrence (CAL Item 3b)	Wilczynski	Hartigan Missig		
c.	Auto/Essential Control Problems	Jain	Yarger		
d.	Trip Throttle Valve Problem	Gradomski			
2.	SG Integrity/Cycle Impact Due To SUFP Initiation of Water • Evaluation of thermal shock considerations on both S/G's (CAL Item 5) • Maximum S/G shell differential temperature (CAL Item 5)	Chen	B&W		

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#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
3.	Actions of Operators	O'Connor	Derivan		
4.	Classification of the Event Under the Emergency Plan (EAL Pg. 3 of 37) • Basis for event classification (CAL Item 5)	Scott-Wasilk	O'Connor		
5.	SFRCS Half Trip On One Level Input	Yarger	Stalter		
6.	SFRCS Alarms	Miller	Lingen-felter		
7.	MSIV/SFRCS Response • Establish cause(s) of unexpected valve closure (CAL Item 2a) • Determine and implement corrective action (CAL Item 2b) • Conduct testing to ensure operating as required (CAL Item 2c)	Miller	Jain Yarger		

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
8.	MFP Control System • Establish cause of inadvertent trip of #1 MFP (CAL Item 4a) • Determine and implement corrective action to prevent recurrence for both MFP's (CAL Item 4b)	Blay	J.Johnson Missig Topor Isley G.E.		
9.a. b.	Damaged turbine bypass valve Water hammer indication during transient	Raynes	Hiss Lammon		
10.	PORV Condition - Cycled 3 Times (3rd cycle looked inconsistent) # of Stress Cycles	Isley	Straube Marley		
11.	Discuss Event With Ottawa County Commissioners (Significance)	Scott-Wasilk			
12.	S/G Isolation Valves Capability To Open Against Large D/P (AFW 599 & 608)	Long	Bajestani		
13.	Service Water Effect on S/G	Briden			

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
14.	Operator Error for Initiating AFW				
a.	HED on switches (FCR)	Calcamuggio	Czuba Batch		
b.	Plastic covers	E.Johnson			
15.	Source Range NI's (RPS)				
a.	NI-1	Borysiak			
b.	NI-2	Isley			
16.	MSSV's - Blowdown to 900 #'s	Huston	Mahoney		
17.	Problems With SPDS	Smith			
18.	SP7A Problems (S/U Feedwater)	Yarger	Bonner		
19.	Adequacy of Information as Originally Provided to NRC on June 9, 1985 (CAL Item 5)				
20.	Complete Items 1-5 of CAL. Obtain concurrence of Regional Administrator Prior to Restart, Mode 2 (CAL Item 6)	Wood	Wideman		

#	ISSUE/CONCERN	TED LEAD	SUPPORT	NRC EVAL/CONCUR	COMMENTS
21.	Perform Testing and Demonstrate that the MFP's Will Operate as Required (CAL Item 7a)	Topor	Blay G.E.		
22.	Perform Testing and Demonstrate that the MFP's Will Operate as Required (CAL Item 7b)	Gradomski	Hartigan Missig		
23.	Appropriate Test Results, Conclusions and Corrective Actions Shall be Reported to the NRC Resident Inspector (CAL Item 7c)	Wood	Wideman		
24.	Obtain Concurrence from Regional Administrator Prior to Exceeding 5% Reactor Power	Wood	Wideman		

PDR

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1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

Position

Licensees and applicants shall describe their program for ensuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely. A report describing the program for review and analysis of such unscheduled reactor shutdowns should include, as a minimum:

1. The criteria for determining the acceptability of restart.

2. The responsibilities and authorities of personnel who will perform the review and analysis of these events.

3. The necessary qualifications and training for the responsible personnel.

4. The sources of plant information necessary to conduct the review and analysis. The sources of information should include the measures and equipment that provide the necessary detail and type of information to reconstruct the event accurately and in sufficient detail for proper understanding (See Action 1.2).

5. The methods and criteria for comparing the event information with known or expected plant behavior (e.g., that safety-related equipment operates as required by the Technical Specifications or other performance specifications related to the safety function).

6. The criteria for determining the need for independent assessment of an event (e.g., a case in which the cause of the event cannot be positively identified, a competent group such as the Plant Operations Review Committee, will be consulted prior to authorizing restart) and guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event.

7. Items 1 through 6 above are considered to be the basis for the establishment of a systematic method to assess unscheduled reactor shutdowns. The systematic safety assessment procedures compiled from the above items, which are to be used in conducting the evaluation, should be in the report.

Documentation Required

Licensees and applicants shall submit a report describing their program addressing all the items in the position.

RESPONSE - ITEM 1.1

Toledo Edison (TED) has always placed a high priority on the review of reactor trips and unit transients. Procedures have existed at Davis-Besse Nuclear Power Station, Unit No. 1 (DB-1) to ensure that the cause of the reactor trip is known and corrected prior to a reactor startup. Station management has been routinely involved in the review of the cause of the trip and the decision to restart. A detailed review is performed, and a formal transient report is submitted for every unscheduled reactor trip. DB-1 is also an active participant in the B&W Owners Group Transient Assessment Program (TAP), which allows all the B&W units the opportunity to learn from the experience of any unit.

NUREG 1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant", provided guidance for a detailed method of performing the post trip review and determining the restart requirements. Toledo Edison has reviewed this guidance and is in the process of modifying the Station procedures to place the Station in compliance. A structured systematic method for post trip review will be added to the "Trip Recovery Procedure", PP 1102.03. This review will assure that prior to returning to power operation, Station personnel have reasonably determined the cause of the trip, verified the proper operation of safety related equipment, and ensured the trip did not have any significantly detrimental effect on the plant.

The following position has been adopted by the B&W Owners Group and forms a uniform philosophical basis by which each member utility will develop a post-trip review program description and procedure:

> To determine the plant's readiness to return to power operation following an unscheduled reactor trip, Station personnel must reasonably determine the cause of the trip, verify the proper functioning of safety-related and other important equipment during the trip, and ensure the trip did not have a detrimental effect on the plant.
>
> The on-shift Shift Supervisor (SS) has the overall responsibility for the safe and reliable operation of the plant during his/her shift. Therefore, the SS is responsible for ensuring a thorough and complete post trip review. An individual independent of the operating shift should be involved in the post trip review process. This individual should enhance the objectivity and thoroughness of the review process and will have qualifications specified by plant management, which should include experience or specific training in plant transient analysis.
>
> Sufficient data must be collected to perform a complete investigation and restart decision. The trip data should be collected as soon as feasible after the trip. It is important to recognize that the quality of the trip investigation and restart

decision is dependent upon the collected trip data. Specific data collection guidelines should be developed to ensure the ability to:

1. Determine the sequence of events,

2. Evaluate plant transient behavior, and

3. Evaluate the performance of safety systems and other important systems.

Computer records, strip charts, and operator interviews are examples of data that should be collected.

The trip event should be reconstructed, analyzed, and evaluated to determine the most probable cause of the trip, the proper functioning of safety related and other important equipment, and any detrimental effects of the trip on the plant. Experience gained through the B&WOG Transient Assessment Program can form a basis for determining acceptable plant response.

The post trip review will result in a recommendation for either a reactor restart or further investigation. If the immediate cause of the reactor trip cannot be determined, or if any unresolved safety issues exist, or if compliance with licensing requirements is in question, then the trip requires further review.

Each B&WOG utility will develop a structured and systematic method for performing post trip reviews. Having a structured method for post trip reviews will contribute to a consistently thorough investigation and decision making process. Additionally, the review process will produce a data package for each trip that can be used for subsequent review and evaluation through the B&WOG Transient Assessment Program.

Using the B&WOG guidelines, the post trip review at DB-1 will be divided into three steps:

1. Identifying plant pre-trip conditions.

2. Identifying plant post trip conditions.

3. Identifying any safety concerns.

The first step reviews and documents the plant conditions prior to the trip. The reactor power, status of the Integrated Control System (ICS), status of safety systems, and status of any other contributing abnormal plant condition will be reviewed. This is done to ensure any contributing abnormal plant conditions prior to the trip are identified and documented to allow further review.

The second step, the post trip review, provides a designated set of criteria for comparing the event data with the expected post trip response. In order to understand the criteria chosen, it is necessary to understand the normal post trip response at DB-1. DB-1 operates with a constant reactor coolant average temperature above approximately 28% of full power. The pressurizer level is normally 200 inches, and the normal RCS pressure is 2155 psig. After a reactor trip, a +145 psi bias is added to the normal header pressure setpoint to control the steam generator header pressure at 1015 psig. With the loss of the heat source from the reactor trip, the reactor coolant temperature rapidly approaches the saturation temperature for the steam generators (approximately 546°F). This cooling down of the primary coolant causes an increase in the coolant density which decreases the coolant volume. The pressurizer level will drop to below 40" (for trips near full power) to provide this volume. The steam generator pre-trip level will vary with the power level (higher level for higher powers), but the Integrated Control System (ICS) normally controls the post trip value at 35". For a normal trip, the power operated relief valve (PORV) will not operate, and the Safety Features Actuation System (SFAS) and the Steam and Feedwater Rupture Control System (SFRCS) will not actuate. Therefore, the following criteria have been designated:

1. Did the PORV actuate?
2. Did the pressurizer code safety valves actuate?
3. Did either steam generator level exceed 82.5%
4. Did SG level go below 18"
5. Was SFAS actuated?
6. Did pressurizer level decrease below 8"?
7. Did pressurizer level exceed 300"?
8. Was the Emergency Plan activated?
9. Did the SFRCS actuate?

The above list is capable of determining a wide variety of unusual post trip responses. Any excessive RCS pressure condition is detectable by the actuation of the PORV or code safeties (Criteria 1 and 2). Sonic flow detectors provide direct open indication to the Station annunciators and Station computer to allow positive verification of the valves actuating. A substantial feedwater transient would be detectable by observation of steam generator levels (Criteria 3 and 4). The Control Room recorders, indicators, Station computer, and annunciators provide adequate indication of operate range level and startup level. Any excessive feedwater would also be detectable by causing the RCS pressure to drop to the 1620 psig setpoint of the SFAS (Criteria 5).

Any failure of the RCS inventory control would be detectable by reviewing the pressurizer level response (Criteria 6 and 7). An overcooling or overheating transient could also be detected through the pressurizer response.

Criteria 8 detects any incident that initiated the Emergency Plan in the Unusual, Alert, Site Area, or General category. This is an extremely broad criteria covering circumstances from a personnel injury to a loss of safety related equipment. Any loss of offsite power, loss of main feedwater, low steam generator level, or low steam generator pressure is detectable by an SFRCS actuation (Criteria 9). The SFRCS at DB-1 actuates on loss of four reactor coolant pumps, low steam generator level (26"), low main steam line pressure (612 psig), loss of main feed pumps (sensed by steam generator to feedwater differential pressure), making it an excellent monitor of a deviance from normal post trip response. Any deviance from the expected post trip conditions requires that futher investigations be performed to determine the cause of the deviance. The cause of the deviance is required to be documented on the post trip review form.

In addition to the set of post trip criteria, the review of the post trip conditions (Step #2) also requires a review of the cause of the trip, the sequence of events which resulted in the trip, and a documentation of the action taken to prevent a recurrence. This requires the cause of the trip to be reasonably determined and that actions be taken to minimize the possibility of recurrence.

The "Sequence of Events" printout is the best source of information to determine the event sequence, but since the sequence of events points are also printed on the alarm typer, the alarm printout may be used. A "Post Trip Review" printout can also be reviewed to determine additional details of the transient (see response to Item 1.2 for details of available post trip information). This step ensures the event chronological order has been properly reconstructed.

The third step of the post trip review verifies that no safety concerns have been identified in the review of the trip. Three safety limits are specifically defined in Technical Specifications:

1. Reactor core pressure/temperature limits during Modes 1 and 2.

2. Reactor core power/imbalance limits during Modes 1 and 2.

3. Reactor Coolant System (RCS) pressure exceeding 2750 psig at any time.

Specific values for the reactor core limits are provided in the Technical Specifications to allow a comparison to the transient values. If any safety limit is exceeded, operation may not resume until authorized by the Nuclear Regulatory Commission as per 10 CFR 50.36 Section C. In addition to the safety limits, the third step requires a review of the plant transient to determine if a safety related system did not perform the design function for which it was intended.

The post trip review steps are completed as a joint effort by the Shift Technical Advisor (STA) and the Shift Supervisor. The Shift Supervisors at DB-1 maintain a Senior Reactor Operator License, which ensures they are knowledgeable on plant response, Station procedures, and Technical Specifications. At DB-1 in order for an individual to

qualify as an STA, he must possess the minimum of a bachelor's degree, or equivalent (as determined by the Station Superintendent) in science or an engineering discipline. In addition, he will have completed DB-1 Training Section approved courses in the following minimum subjects:

1. Thermodynamics
2. Heat Transfer Theory
3. Fluid Dynamics
4. Reactor Theory
5. Pressurizer Water Reactor Technology
6. Davis-Besse Technical Specifications
7. Babcock & Wilcox Simulator Training

This diversity in the completion of the post trip review enhances the objectivity and thoroughness of the review.

If the cause of the trip is not known, the sequence of events leading to the reactor trip not identified, or a safety concern raised, an independent assessment of the event will be performed by the Station Review Board (SRB) prior to restart. The SRB at DB-1 consists of the following members:

Chairman:	Assistant Station Superintendent
Member:	Station Superintendent
Member:	Technical Engineer
Member:	Chemist and Health Physicist
Member:	Operations Engineer
Member:	Maintenance Engineer
Member:	Lead Instrument and Control Engineer
Member:	Nuclear and Performance Engineer
Member:	Reliability Engineer

A quorum consists of the Chairman and at least four members, two of which may be the designated alternate. The SRB is a competent group of Station Management personnel which can perform a detailed review of any post trip response that deviates from the expected response.

In addition to the post trip review, permission for a plant restart must be granted by the Station Superintendent (or his designee) and the Operations Engineer (or his designee). This provides a verification from upper Station Management that adequate investigation and corrective action has been taken.

The post trip review process described in this response provides a detailed systematic method to assess unscheduled reactor shutdowns. No pre-implementation review of the post trip review process is requested. The process described is presently being implemented at DB-1 to address the concerns of Section 1.1 of Generic Letter 83-28. Implementation is scheduled to be completed by December 31, 1983. Enclosures 1 and 2 are the proposed modifications to the plant pro-

cedures to perform the post trip review. Although some minor changes may be made to the review forms during the approval process, no significant changes are expected.

1.2 POST TRIP REVIEW - DATA AND INFORMATION CAPABILITY

Position

Licensees and applicants shall have or have planned a capability to record, recall and display data and information to permit diagnosing the causes of unscheduled reactor shutdowns prior to restart and for ascertaining the proper functioning of safety-related equipment.

Adequate data and information shall be provided to correctly diagnose the cause of unscheduled reactor shutdowns and the proper functioning of safety-related equipment during these events using systematic safety assessment procedures (Item 1.1). The data and information shall be displayed in a form that permits ease of assimilation and analysis by persons trained in the use of systematic safety assessment procedures.

A report shall be prepared which describes and justifies the adequacy of equipment for diagnosing an unscheduled reactor shutdown. The report shall describe as a minimum:

1. Capability for assessing sequence of events (on-off indications).

    1. Brief description of equipment (e.g., plant computer, dedicated computer, strip chart).

    2. Parameters monitored.

    3. Time discrimination between events.

    4. Format for displaying data and information.

    5. Capability for retention of data and information.

    6. Power source(s) (e.g., Class IE, non-Class IE, non-interruptable).

2. Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdowns, and the functioning of safety-related equipment.

    1. Brief description of equipment (e.g., plant computer, dedicated computer, strip charts).

    2. Parameters monitored, sampling rate, and basis for selecting parameters and sampling rate.

3. Duration of time history (minutes before trip and minutes after trip).

4. Format for displaying data including scale (readability) of time histories.

5. Capability for retention of data, information, and physical evidence (both hardware and software).

6. Power source(s) (e.g., Class IE, non-Class IE, non-interruptable).

3. Other data and information provided to assess the cause of unscheduled reactor shutdowns.

4. Schedule for any planned changes to existing data and information capability.

Documentation Required

Licensees and applicants shall submit a report describing their data and information capability for unscheduled reactor shutdowns.

RESPONSE - ITEM 1.2

The capability to record and recall the plant information necessary to assist in the determination of the cause or causes of unscheduled reactor trips currently exists at DB-1. Digital indications (e.g., on/off, open/close, etc.) and key analog information are recorded by various transient monitoring systems during a reactor trip for subsequent analysis. The Plant Process Computer records and displays both digital and analog information. The Data Acquisition and Display System (DADS) located in the Technical Support Center also provides a means for recording and displaying analog information. An additional source of analog information used to support post trip efforts comes from the Control Room strip chart recorders. These systems provide the primary sources of information used for trip analysis at DB-1.

Plant Process Computer

The Plant Process Computer monitors digital and analog information from all major plant systems. Approximately 2,500 digital points and 2,000 analog points are fed into the computer. Some of this information is manipulated and stored for plant performance monitoring purposes, and all of the information is available to the Control Room operator in various display formats. Three functions of the Plant Process Computer provide information useful for transient analysis efforts. These functions include the Sequence of Events Monitor, the Post Trip Review, and the Alarm Printout.

The Sequence of Events (SOE) Monitor is designed to provide a sequential list of important plant events. All inputs to this function are digital. The list of monitored points is provided as Enclosure 3. A change of state of any of these digital points is recorded in the SOE file along with the time of occurrence. The time of occurrence listed with the event is based on computer clock time and recorded to the nearest five milliseconds. The SOE file can hold up to 256 records. Once the SOE file is filled, subsequent events replace the oldest recorded event in the file. The first event to be recorded in the file triggers an indicator to the operator that an SOE monitored event has occurred. This indication is cleared, and the SOE file is emptied when the operator requests a printout of the SOE file. Enclosure 4 illustrates the format of information presented in the Sequence of Events printout.

The Post Trip Review function is designed to record selected analog information for a period of time before and after a reactor trip. The list of parameters monitored by this function is provided in Enclosure 5. The most recent 15 minutes of historical values for these parameters is maintained in a rolling file. In the event of a reactor trip, this rolling file is frozen and data for the next 15 minutes is recorded. An indication that the Post Trip Review function has been initiated is provided to the operator. The operator may then request the Post Trip Review printout which clears the file. The Post Trip Review printout provides parametric data in engineering units given at 15 second intervals from 15 minutes prior to the trip until 15 minutes after the trip. Enclosure 6 provides a sample of one segment of a Post Trip Review printout. Note that some of the parameters monitored have scan intervals of more than 15 seconds. Consequently, some data may be repeated in successive 15 second records. The parameters monitored for the Post Trip Review function were chosen as a part of the original plant process computer design. The variables monitored are key parameters of the major primary and secondary systems which could indicate abnormal trends that may lead to, or result from, a reactor trip. Normally inoperative safety systems are not monitored by this function. The scan intervals selected for the parameters were based on the anticipated rates of change of the individual parameters, and multiplexing hardware and memory capacity limitations that existed at the time of the initial design.

The Alarm Printout function provides an historical listing of both digital and analog information recorded when the monitored parameters enter a predetermined alarm state. Essentially, all digital and analog input points are monitored for alarm status. Alarm messages are recorded as they occur on the alarm printer along with the time of occurrence. No operator action is required to initiate the alarm printout. All digital points are scanned once per second, and a change of point status is identified on the alarm printer. Analog points are scanned at varying intervals (either 1, 5, 15, 30, or 60 second intervals) and are compared at each scan to a predetermined

alarm value. Each time the parameter exceeds the alarm limit or returns to within limits, the event is recorded on the Alarm Printout. An example of a section of the Alarm Printout is provided in Enclosure 7.

The Plant Process Computer consists of redundant MODCOMP Classic 7870 CPUs. The CPUs are powered from separate uninterruptable instrumentation buses YAU and YBU. The uninterruptable buses are supplied from the station battery backed 250 volt DC power supply system through an inverter. Power can also be supplied to the bus from a nonessential regulated instrumentation bus through a static transfer switch within the inverter. The redundant CPUs were installed during the 1982 Refueling Outage as a part of the overall project to upgrade the Plant Process Computer system. The multiplexers providing inputs to the processors will be replaced in future outages. The multiplexers are currently supplied from YBU, consequently, a loss of YBU will interrupt all three transient monitor functions of the Plant Process Computer. The Data Acquisition and Display System will still be functional. As the multiplexers are replaced, they will be equipped with redundant power supplies.

Data Acquisition and Display System

The Data Acquisition and Display System (DADS), located in the Technical Support Center, was designed as a part of the emergency response facilities at DB-1. The primary function of the system is to provide information to emergency response personnel in the Technical Support Center to assist in evaluating plant status in an accident situation. Consequently, those variables important to determining the safety status of plant systems and the proper functioning of safety systems are inputs to the DADS.

While the DADS receives inputs from numerous sources, such as the Meteorological Tower and the Plant Process Computer, the inputs of importance to the transient monitoring function are supplied through a separate multiplexer. The list of parameters supplied by this multiplexer is provided in Enclosure 8. The scan rate for these variables is approximately once per second. Data is recorded at that rate for a period of 24 hours in a rolling file. Access to information in this data file is possible in several formats. Individual points or groups of points can be examined by a CRT or a line printer output. Additional output formats are being developed and will include the use of a printer/plotter to provide graphical trends.

The power supply for the multiplexer located in the station is YAU. The power supply for the DADS Computer System is independent of the station electrical system. The Davis-Besse Administration Building (DBAB) which houses the Technical Support Center and the DADS, is supplied from a construction feeder independent of the three 345 KV lines connected to the station grid. The DBAB electrical system supplies an emergency response facilities bus which can also be fed

by an emergency diesel generator through an automatic transfer switch. The emergency response facilities bus in turn feeds an uninterruptable distribution network. Power to the uninterruptable distribution network is backed up by a battery driven system through a static transfer switch which assures continuous operation of the DADS computer system. The emergency battery system is charged from the emergency response facilities bus.

Strip Chart Recorders

In the event that the Plant Process Computer and the Data Acquisition Display System are unable to perform their transient monitor functions, the Control Room strip chart recorders act as a backup source of information for transient analysis. Due to the compressed time scales of the strip chart recorders, the information cannot be used for sequence of events determination and the limited number of parameters recorded make determination of the cause of a transient very difficult. However, the parameters that are recorded are important major system parameters such as pressurizer level, Reactor Coolant System pressure, steam generator levels, feedwater flows, etc. The information available on the strip chart can be very useful in assuring that major system upsets did not occur as a result of the transient. Strip chart recorders are also useful in recognizing long term trends that may be indicative of problems leading to, or resulting from, a transient.

2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)

Position

Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement. In addition, for these components, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of these components should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply the information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reactor trip system reliability. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical

mailings. The program shall also define the interface and division of responsibilities among the licensees and the nuclear and non-nuclear divisions of their vendors that provide service on reactor trip system components to assure that requisite control of and applicable instructions for maintenance work are provided.

<u>Documentation Required</u>

Licensees and applicants should submit a statement confirming that they have reviewed the Reactor Trip System components and conform to the position regarding equipment classification. In addition, a summary report describing the vendor interface program shall be submitted for staff review. Vendor lists of technical information, and the technical information itself, shall be available for inspection at each reactor site.

<u>RESPONSE - ITEM 2.1</u>

A review of the reactor trip system and its components has been made for conformance to the position regarding equipment classification. The system listing consists of the following discrete components with their Q-List No. and Supplier. They are:

1. The Reactor Protection System (RPS), which includes remote sensors, furnished by B&W. (Q-List #6.10)

2. The Control Rod Drive Breakers, the Breaker's Undervoltage Trip devices, and the cubicle (Q-List #6.51) furnished by B&W.

3. a. Anticipatory Reactor Trip System (ARTS) furnished by Vitro. (Q-List #6.90)

   b. Remote sensors for ARTS furnished by Static-O-Ring (SOR) (Q-List #3.1280 and 3.1180).

4. Remote Reactor Trip button furnished by Diamond Power (B&W). (Q-List #6.51)

These components are all classified Class 1E.

A list of all technical manuals issued by B&W relative to Items 1, 2, and 4 above have been received and checked against the TED vendor log. <u>All</u> items are in TED possession. A request for a list of all updates pertinent to these manuals is presently being reviewed by B&W for submittal to TED. The vendor's list and the original technical manuals will be available at the site. Updates for technical manuals will be available at the site.

Items 3a and 3b, the ARTS system, was placed inservice in the 1982 Refueling Outage and all Technical manuals are current.

Vendor manuals have been used to develop station maintenance procedures and are referenced, where applicable, in the reference section of the procedure.

TED is currently developing additional review plans and schedules to fully verify the adequacy of the program and its administrative controls to meet the requirements of Item 2.1. The program outline has been developed. Fully developed plans and schedules will be submitted for your review by February 1, 1984.

The outline of the planned program to update and verify the Reactor Trip System Vendor Manuals is as follows:

1. Contact vendor for list of technical manuals relative to pertinent equipment supplied by vendor.

2. Verify against vendor log.

3. Acquire missing document (if any).

4. Request any addenda to update manuals to current status.

5. Provide verifiable feedback to vendor (with fully defined vendor interface and responsibility delineations).

6. Compare to station maintenance procedures (this would be part of a technical review).

7. Develop and implement administrative procedures to control flow of vendor documents from vendor thru TED document control, technical review, distribution and storage.

2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (PROGRAMS FOR ALL SAFETY-RELATED COMPONENTS)

Position

Licensees and applicants shall submit, for staff review, a description of their programs for safety-related* equipment classification and vendor interface as described below:

1. For equipment classification, licensees and applicants shall describe their program for ensuring that all components of safety-related systems necessary for accomplishing required safety functions are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders and replacement parts. This description shall include:

    1. The criteria for identifying components as safety-related within systems currently classified as safety-related. This shall not be interpreted to require changes in safety classification at the systems level.

    2. A description of the information handling system used to identify safety-related components (e.g., computerized equipment list) and the methods used for its development and validation.

    3. A description of the process by which station personnel use this information handling system to determine that an activity is safety-related and what procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10 CFR 50, Appendix B, apply to safety-related components.

    4. A description of the management controls utilized to verify that the procedures for preparation, validation and routine utilization of the information handling system have been followed.

*Safety-related structures, systems, and components are those that are relied upon to remain functional during and following design basis events to ensure: (1) the integrity of the reactor coolant boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100.

5. A demonstration that appropriate design verification and qualification testing is specified for procurement of safety-related components. The specifications shall include qualification testing for expected safety service conditions and provide support for the licensees' receipt of testing documentation to support the limits of life recommended by the supplier.

6. Licensees and applicants need only to submit for staff review the equipment classification program for safety-related components. Although not required to be submitted for staff review, your equipment classification program should also include the broader class of structures, systems, and components important to safety required by GDC-1 (defined in 10 CFR Part 50, Appendix A, "General Design Criteria, Introduction").

2. For vendor interface, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information for safety-related components is complete, current and controlled throughout the life of their plants, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of safety-related equipment should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reliability commensurate with its safety function (GDC-1). The program shall be closely coupled with action 2.2.1 above (equipment qualification). The program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgment for receipt of technical mailings. It shall also define the interface and division of responsibilities among the licensee and the nuclear and nonnuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided.

Documentation Required

Licensees and applicants should submit a report that describes the equipment classification and vendor interface programs outlined the position above.

RESPONSE - ITEM 2.2.1

For equipment classification at DB-1, the Q-list has been developed to identify and itemize those nuclear safety related structures, systems, and components that require Quality Assurance in accordance with the requirements of 10 CFR 50 Appendix B. Q-listed items require special consideration during design, procurement, fabrication, construction, preoperational testing, startup, and the operations phase. Using this list as a basis, the TED Nuclear Engineering Manager issues approved revisions as needed thus indicating those nuclear safety related structures, systems, and components which are within the TED Nuclear Quality Assurance Program.

The Q-list of itself is systems oriented; under specific defined systems, components are listed. Detailed "Q" limitations, or, in some instances, the actual physical boundaries of Quality Assurance coverage for those nuclear safety related structures, systems, and components itemized in the Q-list are defined in supportive approved documents such as engineering drawings, bills of material, design specifications, contracts, purchase orders, written procedures, and instructions.

Thus, by use of the Q-list and supportive approved documents, discrete information is provided to fully implement the scope of the TED Nuclear Quality Assurance Program, and maintain control and integrity of all safety related components.

RESPONSE - ITEM 2.2.1.1

Nuclear Safety Related (Q) components at DB-1 are defined in our "Nuclear Quality Assurance Manual" (NQAM), as follows:

> "Those facility features necessary to ensure the integrity of the reactor coolant pressure boundary, the capability to shut-down the reactor and maintain it in a safe condition, or the capability to prevent or mitigate the consequences of accidents which could result in off-site exposure comparable to the guidelines of the NRC Regulations 10 CFR Part 100."

The Manual further identifies items as Q in our "Q-List", as discussed above.

Components within Q systems are identified as Q on our drawings, such as Piping and Instrument Diagrams (P&ID's), Piping Isometric Drawings, Elementary Wiring Diagrams, Instrument Index, Electrical One-Line Diagrams, Electric Motor Lists, etc. Components are also identified as Q in our equipment specifications.

RESPONSE - ITEM 2.2.1.2

TED is developing an information handling system that can be used to identify safety-related components. A computerized data base is being developed and has been loaded with some limited information. An upgrade of the data base is planned for 1984.

RESPONSE - ITEM 2.2.1.4

At DB-1 the management controls associated with verifying procedural compliance with the information handling system, above the normal managerial review process, are those pertaining to the Quality Assurance audit process.

All Nuclear Safety related administrative procedures are reviewed and approved by the Quality Assurance Department and implementation of these procedures is verified during audits which are conducted in accordance with Section 18 of the Toledo Edison NQAM. Any deficiencies noted during the performance of these audits are documented in Audit Finding Reports and submitted to appropriate levels of management for prompt corrective action. Follow-up action including re-audits of deficiencies are conducted to verify implementation of corrective action.

The audit program at Toledo Edison Company meets the requirements of ANSI N45.2.12 as endorsed by NRC Regulatory Guide 1.144 for audit scheduling, preparation, performance and reporting. Auditors and Audit Team Leaders are qualified in accordance with the requirements of ANSI N45.2.23 as endorsed by NRC Regulatory Guide 1.146.

Audit reports are reviewed and approved by the TED Quality Assurance Director and distributed to management of the audited organization, the Toledo Edison President, the Vice President, Nuclear, and Company Nuclear Review Board Members. The CNRB reports to and advises The Toledo Edison Company President and Chief Operating Officer on matters concerning its review and audit of activities relative to the Davis-Besse Nuclear Power Station.

RESPONSE - ITEM 2.2.1.3, 2.2.1.5

Details of the process used to perform maintenance at DB-1 is controlled by Administrative Procedure AD 1844.00, "Maintenance". Per this procedure an individual (one who specializes in the appropriate maintenance craft) is assigned to prepare a Maintenance Work Order (MWO). The MWO is developed, defining and detailing the work to be performed. The Q-list is consulted along with its respective supporting lists and drawings to properly identify the safety related designation of the components to be worked on. Procedurally, MWO generation cannot be accomplished unless the component designation is complete.

A computerized information handling system is in the developmental stages at DB-1. The system has the capability to handle system and component classification, MWO preparation, control and tracking, as well as Technical Specification testing control.

Within the MWO process, if the component is designated as nuclear safety related, Enclosure 13 of AD 1844.00 must be completed and attached to the MWO. This document is included with this submittal as Enclosure 9. This attachment requires an evaluation of testing required to provide operability of the equipment be completed and documented. All nuclear safety related MWOs are approved by the Maintenance Engineer (or his designee) and reviewed by the Quality Control Department to determine inspection requirements. Other related quality control activities per 10CFR50 Appendix B are covered in the responses to Items 2.2.1.4 and 2.2.1.5.

If material is needed to complete the work, a Material Issue Ticket (MIT) is required to be completed. Administrative Procedure AD 1847.00, "Station Materials Control" covers the controls of materials at DB-1. This procedure requires the MIT designate if the material is Q-listed, ASME, or fire protection qualified. The MIT must be approved by a member of the management staff before the material can be removed from the storeroom. Each MIT has a designated serial number on all three copies, and all material used is listed on the MWO to allow for full material traceability.

Additional information on the performance of maintenance and material control is specified in detail in the Station Administrative Procedures. The above paragraphs were an attempt to summarize the applicable portions to position 2.2.1.3.

Purchase Orders for nuclear safety related components are identified as "Q" and as applicable, contain or make reference to Toledo Edison Company Engineering Specifications which contain: applicable codes and standards, service conditions, design and construction details and qualification testing and document submittal requirements. Vendor documentation including reports of environmental and seismic qualification testing are submitted for review by Toledo Edison, or its agents, Engineering and Quality Assurance Departments. The Toledo Edison Nuclear Quality Assurance Program requires that all such qualification test reports be appropriately reviewed and approved, or a safety review performed by Engineering in accordance with 10CFR50.59, prior to operation of the safety related component.

All procurement documents identified as Q-list, Fire Protection and/or ASME must be reviewed/approved by the Facility Engineering Department to ensure that the technical and quality requirements are acceptable and by the Quality Engineering Department to ensure that the quality requirements are sufficient, clear and adequately stated and that the supplier/contractor has the capability to comply with the quality requirements.

Quality Control Instruction (QCI) 3070 establishes the guidelines for the conduct of receipt inspection performed by the Toledo Edison Quality Engineering or Quality Control sections at DB-1. All Q-list, Fire Protection and/or ASME items are receipt inspected to ensure conformance with the procurement documents. Quality Assurance Instruction (QAI) 4042 establishes the guidelines for the evaluation and approval of supplier's/contractor's capabilities for furnishing a product to meet the design, manufacturing, and quality requirements prior to the initiation of a TED purchase order.

RESPONSE - ITEM 2.2.1.6

With respect to the equipment classification program in use at DB-1 for structures, systems and components important to safety, we are participating in the Utility Safety Classification Group and are seeking a generic resolution to the staff's concern in this regard through the efforts of this Group. We do not agree that plant structure and components important to safety constitute a broader class than the safety related set. We, nevertheless, believe that non-safety related plant structures, systems and components have been designed, and are maintained, in a manner commensurate with their importance to the safety and operation of the plant.

RESPONSE - ITEM 2.2.2

Toledo Edison is currently reviewing the administrative controls in place for ensuring the adequacy of the methods used to maintain up-to-date vendor manuals. TED will complete the implementation of a program adequately addressing to the extent possible, all the concerns of Item 2.2.2 of Generic Letter 83-28. TED is an active participant in the INPO-NUTAC that has been established to develop a Generic joint utility response to the subject Generic letter. It is our current understanding that the NUTAC position will be finalized in February 1984, with final printed issuance to follow in April, 1984. Following the receipt of the NUTAC position, TED will prepare its specific response to Item 2.2.2 as applicable to DB-1, incorporating all or part of the generic position as appropriate. We intend to submit this program for your review within three (3) months of the issuance of the NUTAC position. This will result in a submittal of a program plan and schedule to meet Item 2.2.2 by July 31, 1984.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

Position

The following actions are applicable to post-maintenance testing:

1. Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety-related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval. (Note that action 4.5 discusses on-line system functional testing.)

Documentation Required

Licensees and applicants should submit a statement confirming that actions 3.1.1 and 3.1.2 of the above position have been implemented.

RESPONSE - ITEM 3.1.1

All maintenance performed at DB-1 Station is controlled by a Maintenance Work Order (MWO) as delineated in Administrative Procedure AD 1844, "Maintenance" which was established to ensure that maintenance at the Station is planned, performed, and documented in accordance with the requirements of the Toledo Edison Nuclear Quality Assurance Manual, and Section XI of the ASME Code.

As per Administrative Procedure AD 1844.00, all MWOs on safety related equipment shall have an Enclosure 13, Work Requirements Check-off List, prepared, Enclosure 9 to this document. This was discussed in response to Item 2.2.1.3.

As per the Technical Specifications for the reactor trip system components, a review has verified that the surveillance tests that are performed to verify operability after maintenance do demonstrate that the equipment is capable of performing its safety functions before being returned to service. The review criteria supporting this demonstration includes the definitions of operability and the channel functional test (1.6 and 1.11 of the Technical Specifications).

The applicable Technical Specifications requirement portions include 3/4.3.1 for the Reactor Protection System (RPS), Control Rod Drive (CRD) trip breakers and manual reactor trip pushbuttons and the future 3/4.3.2.3 for the Anticipatory Reactor Trip System (ARTS).

RESPONSE - ITEM 3.1.2

All recommendations of G.E. Service Advice 9.3 including the 1983 supplement were included into the Station procedures for maintenance and testing of the CRD breakers. Modifications were verified to have been completed to the required surveillance tests per B&W Operating Plant Service Bulletin 25-005. Since no other vendor recommendations were discovered during the review, no other modifications were necessary to procedures governing reactor trip system components.

RESPONSE - ITEM 3.1.3

No post maintenance test requirements on reactor trip system components that degrade safety have been identified.

3.2 POST-MAINTENANCE TESTING (ALL OTHER SAFETY-RELATED COMPONENTS)

Position

The following actions are applicable to post-maintenance testing:

1. Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications where required.

3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which are perceived to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

Documentation Required

Licensees and applicants should submit a statement confirming that actions 3.2.1 and 3.2.2 of the above position have been implemented.

RESPONSE - SECTION 3.2

All maintenance performed at DB-1 is controlled by a Maintenance Work Order (MWO) as delineated in Administrative Procedure AD 1844.00, "Maintenance", which was established to ensure that maintenance at the Station is planned, performed, and documented in accordance with the requirements of the Toledo Edison Nuclear Quality Assurance Manual, and Section XI of the ASME Code.

As per Administrative Procedure AD 1844.00, Enclosure 5, all MWO's on safety related equipment shall have an Enclosure 13, Work Requirements Check-off List, prepared. The Enclosure 13 specifies what safety related equipment/systems are to be worked on and also the post maintenance testing requirements, which describes the method of proving operability, including surveillance tests.

TED is currently reviewing the level of effort required to develop the plans and schedules associated with meeting the Generic Letter Item 3.2.

Relative to Item 3.2.1, a review of test and maintenance procedures and Technical Specifications assuring that post-maintenance operability testing is performed is being planned. This effort is being integrated into the 3.2.2 response effort of the test and maintenance procedure and/or the Technical Specifications review for ensuring that vendor and engineering recommendations are included as appropriate test guidance.

The intent of the review program being developed is to:

1. Ensure that vendor and engineering recommendations for all safety related components have been added as appropriate test guidance to the test and maintenance procedures and/or Technical Specifications where required.

2. Ensure that no post-maintenance test requirements in existing Technical Specifications exist which are perceived to degrade rather than enhance safety.

3. Ensure all post maintenance testing of safety related components demonstrates the equipment capability of performing its safety functions.

4. Ensure all post maintenance testing of safety related components assures post-maintenance operability.

The development of this program will continue, however, due to the impact that the Item 2.2.2 response may have on this activity, TED intends to submit a detailed plan and schedule for this effort along with that of Item 2.2.2.

This will support a July 31, 1984 date for the submittal of this plan and schedule.

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

Position

All vendor-recommended reactor trip breaker modifications shall be reviewed to verify that either: (1) each modification has, in fact, been implemented; or (2) a written evaluation of the technical reasons for not implementing a modification exists.

For example, the modifications recommended by Westinghouse in NCD-Elec-18 for the DB-50 breakers and a March 31, 1983, letter for the DS-416 breakers shall be implemented or a justification for not implementing shall be made available. Modifications not previously made shall be incorporated or a written evaluation shall be provided.

Documentation Required

Licensees and applicants should submit a statement confirming that this action has been implemented.

RESPONSE - ITEM 4.1

At DB-1, General Electric Company type AK2A breakers are used for reactor trip breakers. A search was performed of any vendor recommended reactor trip breaker modifications.

Our review verifies that the only vendor recommended reactor trip breaker modification is the B&W field change package 128-00, which changed the undervoltage trip devices because of an assembly problem. This modification was completed and tested September 2, 1977.

4.2 REACTOR TRIP SYSTEM RELIABILITY (PREVENTATIVE MAINTENANCE AND SURVEILLANCE PROGRAM FOR REACTOR TRIP BREAKERS)

Position

Licensees and applicants shall describe their preventative maintenance and surveillance program to ensure reliable reactor trip breaker operation. The program shall include the following:

1. A planned program of periodic maintenance, including lubrication, housekeeping, and other items recommended by the equipment supplier.

2. Trending of parameters affecting operation and measured during testing to forecast degradation of operability.

3. Life testing of the breakers (including the trip attachments) on an acceptable sample size.

4. Periodic replacement of breakers or components consistent with demonstrated life cycles.

Documentation Required

Licensees and applicants should submit descriptions of their programs to ensure compliance with this action.

RESPONSE - ITEM 4.2.1

Maintenance Procedure MP 1405.05 (480V Type AK Circuit Breaker Maintenance and Testing) was written to provide instructions for the cleaning, inspecting, and testing of GE 480V circuit breakers. The CRD trip breakers have been incorporated into the Preventative Maintenance Program in which Maintenance Procedure MP 1405.05 is performed every refueling cycle or as necessary to ensure the design performance of the breakers. As per Maintenance Procedure MP 1405.05, Surveillance Test ST 5030.20 (CRD Trip Breaker Response Time Test) is performed to test the response time of contact opening after de-energization of CRD trip breaker undervoltage coil as required by Surveillance Test ST 5030.14 (RPS Overall Response Time Calculation). The Surveillance Test ST 5030.14 requires Surveillance Test ST 5030.20 to be performed any time corrective or preventive maintenance is performed that could affect the response time.

RESPONSE - ITEMS 4.2.2, 4.2.3, 4.2.4

The B&W Owners Group has reviewed the events at Salem with regard to the Reactor Trip Breaker failures, NUREG 1000, and other information pertinent to the situations leading to the breaker failures at Salem. In addition, we have reviewed the failures and performance history of GE AK2A breakers at the B&W facilities. A substantial amount of information has been gathered from numerous sources including GE, B&W and plant maintenance to determine problem areas with the breakers and susceptibility to problems similar to those that occurred at Salem.

From these investigations, it has been established that two factors are of primary importance to breaker functioning: undervoltage device pickup voltage setting and proper lubrication. Many other factors come into play in assuring the breakers are capable of performing their safety function and in evaluating their need for maintenance and/or replacement. We endeavored to develop a program which will account for these factors by monitoring breaker performance so as to forecast breaker problems/failures before they occur and thus permit appropriate preemptive action to be taken to preclude these problems.

A presentation was made to the NRC on August 25, 1983, outlining our conceptual program for a Reactor Trip Breaker (RTB) Reliability Program. Progress has been made in the refinement of that program as described below.

The RTB Reliability Monitoring Program will compile and analyze maintenance and surveillance data for the General Electric AK-2 Control Rod Drive System air circuit breakers. The program has the following objectives:

1. Maintena ce intervals will be identified to achieve reliable breaker operability without unnecessary maintenance efforts.

2. The comparative effectiveness of maintenance practices and personnel training between Owners will be shown through the presentation of data. Necessary revisions to maintenance practices should become evident to Owners, resulting in performance improvements.

3. Data will be collected from existing surveillance and maintenance programs to allow continuation of existing station schedules and to eliminate unnecessary device cycling to obtain data.

4. Establishment of an expected range of breaker trip response times for use as an indicator of a need for additional breaker maintenance.

5. A final reliability confirmation program report will be available to all Owners to provide justification of maintenance schedules and practices based on representative sample of AK-2 devices.

All operating B&W NSSS Owners are expected to participate in the data program. Data will be collected during maintenance and surveillance activities. Data from all AC and DC CRD breakers will be obtained (six breakers for each Oconee-series reactor and four breakers for each Davis-Besse-series reactor). This data includes breaker response times, trip bar torque values and undervoltage pickup setpoints, in both the as-found and as-left condition. The as-found data will be for the first trip actuation after the breaker is removed from service, and the as-left data will be collected post-maintenance prior to return to service.

In addition, selected operating units will be instrumented for on-line breaker response time testing during the performance of existing RPS Channel Functional Testing. The response time monitors will be capable of measuring time to the millisecond.

The BWOG will select the two operating units which will include not more than one unit per site. Engineering of each installation must take into account existing plant specific design. Lead time will be required for engineering, as well as for procurement of the response

time monitor and any required ancillary equipment. Installation would be made during a scheduled refueling outage. Response time testing would then proceed through and include the second refueling outage of the unit, in order to capture two sets of off-line data. As yet, the BWOG has not established criteria for choosing the units to be so instrumented.

All data will be sent by each utility in B&W who will act as the compiler. B&W will compile the data and issue periodic reports to the Utilities to enable each Utility to compare its breaker performance with the norm defined by the data base. The periodic report will also be used as a means of distributing information among Utilities on breaker operating experience.

The data collection and compilation program will run for approximately a two year period. At the conclusion of the program, a final report will be prepared documenting the results and providing recommendations for future breaker surveillance and maintenance.

We believe this program is the best possible solution, for our facilities, to the concerns that arose from the Salem event relating to trip breaker performance. The monitoring of key parameters and the comparisons among the B&W plants will allow us to develop a baseline for breaker performance and behavior and to determine when a breaker is deviating from expected and normal behavior. It will allow for evaluation of various maintenance techniques employed by the various Utilities, service conditions and varying preventive maintenance intervals.

After the approximate two year run of the data acquisition portion of the program, we believe that the need for preventive maintenance, refurbishment, and/or replacement can be positively identified on an as needed basis thus assuring our breakers are properly maintained on a frequency determined by the performance of each individual breaker as opposed to some generic criteria.

We believe this program exceeds the NRC positions for these items. The program not only trends important parameters; it provides a mechanism for modifying preventive maintenance frequencies as necessary based upon observed breaker parameters. This will help avoid breaker problems and/or failures that might otherwise occur.

4.3 <u>REACTOR TRIP SYSTEM RELIABILITY (AUTOMATIC ACTUATION OF SHUNT TRIP ATTACHMENT FOR WESTINGHOUSE AND B&W PLANTS)</u>

<u>Position</u>

Westinghouse and B&W reactors shall be modified by providing automatic reactor trip system actuation of the breaker shunt trip attach-

ments. The shunt trip attachment shall be considered safety related (Class IE).

Documentation Required

Licensees and applicants should submit a report describing the modifications.

RESPONSE - ITEM 4.3

The design process for modification of the reactor trip system is underway. The design is intended to address the concerns of the ATWS-SALEM event. The breaker shunt trip attachments will provide automatic system activation.

We anticipate that description of the TED/modification can be provided by December 1, 1983.

4.4 REACTOR TRIP SYSTEM RELIABILITY (IMPROVEMENTS IN MAINTENANCE AND TEST PROCEDURES FOR B&W PLANTS)

Position

Licensees and applicants with B&W reactors shall apply safety-related maintenance and test procedures to the diverse reactor trip feature provided by interrupting power to control rods through the silicon controlled rectifiers.

This action shall not be interpreted to require hardware changes or additional environmental or seismic qualification of these components.

Documentation Required

Licensees and applicants should submit a statement confirming that this action has been implemented.

RESPONSE - ITEM 4.4

DB-1 has four nuclear safety grade control-rod drive trip breakers designed to provide single failure protection against the failure of one of these breakers to open. Two of these breakers were added to the Station design prior to our obtaining an operating license. This was done so that we would not have to depend upon the silicon controlled rectifiers (SCRs) in the non-safety grade control rod drive control cabinets (CRDCC) to provide this single failure protection. The automatic shunt trip modification described in our response to Item 4.3 will be installed on all four of these breakers. This modification will further reduce the probability of a failure to trip the reactor. In addition, we do not have the capability, without modifications to the CRDCC, to test the SCRs ability to trip the reactor.

Therefore, for these reasons, we do not intend to apply safety-related maintenance and test procedures to the SCRs.

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

Position

On-line functional testing of the reactor trip system, including independent testing of the diverse trip features, shall be performed on all plants.

1. The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W (see Action 4.3 above) and CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants (see Action 4.4 above); and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

2. Plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. Alternatives to on-line testing proposed by licensees will be considered where special circumstances exist and where the objective of high reliability can be met in another way.

3. Existing intervals for on-line functional testing required by Technical Specifications shall be reviewed to determine that the intervals are consistent with achieving high reactor trip system availability when accounting for considerations such as:

    1. Uncertainties in component failure rates.
    2. Uncertainty in common mode failure rates.
    3. Reduced redundancy during testing.
    4. Operator errors during testing.
    5. Component "wear-out" caused by the testing.

Licensees currently not performing periodic on-line testing shall determine appropriate test intervals as described above. Changes to existing required intervals for on-line testing as well as the intervals to be determined by licensees currently not performing on-line testing shall be justified by information on the sensitivity of reactor trip system availability to parameters such as the test intervals, component failure rates, and common mode failure rates.

Documentation Required

For Item 4.5.1, licensees and applicants should submit a statement confirming that this action has been implemented.

For Item 4.5.2, licensees and applicants should submit a report describing the modifications for staff review.

For Item 4.5.3, licensees and applicants should submit proposed Technical Specification changes for staff review.

RESPONSE - ITEM 4.5.1

On line testing of the reactor trip system is performed monthly by Surveillance Test ST 5030.02 (RPS Monthly Functional Test), Surveillance Test ST 5030.19 (ARTS Monthly Functional Test), and Surveillance Test ST 5030.12 (Channel Functional Test of the Reactor Trip Module Logic and Control Rod Drive Trip Breakers). The CRD trip breakers are exercised monthly by each surveillance test by de-energization of the undervoltage coil. Independent testing of the shunt trip device will be performed on an eighteen month cycle.

RESPONSE - ITEM 4.5.2

System functional testing of the reactor trip system is being performed on line on a monthly basis as per response to Item 4.5.1. Independent testing of the shunt trip devices will be performed on an eighteen month cycle. To provide independent testing of the SCR trip functions would require hardware changes made to the system which are not required per Item 4.4.

RESPONSE - ITEM 4.5.3

The B&W Owners Group is sponsoring an analysis program with the objective of demonstrating that the current on-line test interval for the RTS is consistent with high RTS availability. The program will include sensitivity analysis to account for uncertainties in component and common mode failure rates, reduced redundancy and operator errors during testing and component wearout caused by testing. Consideration of changes to the current test interval will depend on the results of the analysis.

A plant-specific submittal will be provided by each Utility after completion of the analysis program.

It is anticipated that Toledo Edison will submit a plant specific report, together with any proposed Technical Specification changes deemed applicable, prior to February 15, 1985.

Checklist For Return to Power Following a Reactor Trip

The following checklist must be completed for each reactor trip from power prior to restart.

1. Attachment III, "Post Trip Review" has been completed or an SRB review of the trip has been performed.

   Verified________________Date________

2. Without actually filling out the checklist, look through Mode 2 and Mode 1 Startup Checklists in PP 1102.01, Prestartup Checklist. Verify the status of these systems is in a condition such that a startup can be made.

   Note: Refer to T.S. Table 4.3-1 for RPS surveillance requirements. Items 1, 10, 11, and 12 are required prior to each startup. Items 1, 10, 11, and 12 have a "Note 1" if not performed in previous 7 days.

   Notify the I&C Engineer or the I&C Shop Foreman that the following Surveillance Tests must be completed prior to plant startup:

   2.1 ST 5030.02 (Intermediate Range Only) must be done if required by the ST schedule.

   2.2 ST 5030.17 (Intermediate Range Prestartup Functional Test) if not performed within previous 7 days.

   2.3 ST 5091.01 (Channels 1 and 2, Source Range) if not performed within previous 7 days.

   2.4 ST 5030.12 (Functional Test of the Reactor Trip Module Logic and Control Rod Drive Trip Breakers) if not performed within previous 7 days.

   Operations personnel are to perform the following:

   2.5 ST 5030.13 (Functional Test of Manual Reactor Trip) if not performed within previous 7 days.

   Shift Supervisor______________Date________

3. Verify the unit is not in an ACTION statement of Technical Specifications which now would prevent re-entry into MODES 2 and 1.

   Shift Supervisor______________Date________

   Note: If 1, 2, or 3 cannot be verified, stop at this point in the checklist since the return to power cannot be made.

4. If the Operations Engineer and the Technical Engineer have concurred to extend the 16 hour limit on this checklist, so document this extension by filling in the time allowed in addition to the original 16 hours. If no concurrence was given, place N/A in the blank.

   Operations Engineer Notified By____________Date____ Extension____Hours

   Technical Engineer Notified By_____________Date____ Extension____Hours

5. At least two (2) licensed Reactor Operators must be present in the Control Room and at least one licensed Senior Reactor Operator must be present at the unit.

   NOTE: The SRO License may be one of the two individuals in the Control Room.

   Shift Supervisor______________Date________

6. The Operations Engineer (or his designee) and the Station Superintendent (or his designee) have given permission for restart.

   Station Superintendent Notified By_____________________Date________

   Operations Engineer Notified By________________________Date________

This form should be routed to the Operations Engineer for his review.

Reviewed By Operations Engineer____________________ Date______________

After the Operations Engineer completes his review, the completed form should be routed to the Technical Section for filing into the unit trip files.

Post Trip Review

The following review must be completed for each reactor trip (except normal tripping of CRD during heatups and cooldowns) even if a unit restart is not in progress.

1.1 Plant Pre-Trip Conditions (to be completed by the Shift Technical Advisor and operations personnel after the plant stabilization is complete)

(A) Reactor power prior to the trip: ________%

Note any runback that occurred: ____________________

____

____

____

(B) List any ICS stations in manual prior to the trip: ________

____

____

____

(C) List any testing in progress prior to the trip: ________

____

____

____

(D) List any safety systems inoperable prior to the trip: ________

____

____

____

(E) List any other abnormal plant conditions contributing to the plant trip (inoperable main feedwater pump, high condenser vacuum, etc.)

____

____

____

Completed By______________________________Date__________

1.2 Plant Post Trip Conditions (to be completed by the Shift Technical Advisor and Shift Supervisor after the plant stabilization is complete)

(A) Did any of the following occur? (Use Control Room recorders, computer information, or operator observations.)

	No
Did the PORV actuate?	___
Did the pressurizer code safety valves actuate?	___
Did either steam generator level exceed 82.5%	___
Did SG level go below 18"?	___
Was SFAS actuated?	___
Did pressurizer level decrease below 8 inches?	___
Did pressurizer level exceed 300 inches?	___
Was the Emergency Plan activated?	___
Did the SFRCS actuate?	___

If any of the above did occur, determine the cause and describe below:

____

____

____

____

____

(B) Write a short description of the cause of the trip, the reactor trip sequence of events which resulted in the trip, and any actions taken to prevent recurrence. (Review the Post Trip Review, Alarm Printout, and Sequence of Events Printouts, if available.)

____

____

____

____

____

Shift Technical Advisor___________________________ Date___________

Shift Supervisor________________________________ Date___________

1.3 Safety Review of Transient (to be completed by Shift Technical Advisor and Shift Supervisor)

(A) Verify no safety concerns* have been identified in the review of the trip.

*A safety concern is defined as a safety related system not performing the design function for which it was intended.

Shift Supervisor_________________________ Date__________

Shift Technical Advisor___________________ Date__________

(B) Verify no safety limit was exceeded during the transient (see Technical Specification 2.1). If any safety limit has been exceeded, operation shall not be resumed until authorized by the Commission as per 10CFR50.36 Section C.

Shift Supervisor_________________________ Date__________

Shift Technical Advisor___________________ Date__________

If the cause of the unit trip cannot be determined, or the Sequence of Events for the reactor trip cannot be determined, or any safety concern identified, a unit restart cannot proceed until a Station Review Board review of the transient has been completed.

After this form is completed, it should be routed to the Operations Engineer for his review.

Operations Engineer__________________________________ Date______________

After the Operations Engineer review, this attachment should be routed to the Technical Section to be included in the trip files.

SEQUENCE OF EVENTS POINTS LIST

Auxiliary Transformer 11 Trouble

Bus A Electrical Fault

Bus B Electrical Fault

Bus A to Transformer AC Breaker

Bus B to Transformer BD Breaker

Bus C2 Trouble

Bus D2 Trouble

Control Rod Drive (CRD) Trip Confirm

CRD Channel AC Any Trip Device

CRD Channel BD Any Trip Device

Electro Hydraulic Control Emergency Trip System Low Pressure

Emergency Diesel Generator 1 Trouble

Emergency Diesel Generator 2 Trouble

Essential Bus C1 Trouble

Essential Bus D1 Trouble

Essential Transformer CE 1-1 Trouble (typical CE 1-1, DF 1-1, CE 1-2, DF 1-2)

Generator and Main Transformer Overall Differential Trip

Generator Overcurrent Trip

Generator Reverse Current Power Trip

Generator Field Failure

Generator Out of Step

Generator Underfrequency

Generator Differential

Generator Ground Current Trip

Main Feed Pump Turbine (MFPT) 1 Trip (typical MFPTs 1 and 2)

Main Transformer Sudden Pressure Change

Moisture Separator Reheater 1 High Level Turbine Trip

Moisture Separator Reheater 2 High Level Turbine Trip

Reactor Protection System (RPS) Channel 1 Flux/Delta Flux/Flow Trip (typical Channels 1 through 4)

RPS Channel 1 High Flux/Number of Reactor Coolant Pumps (RCPs) Running Trip (typical Channels 1 through 4)

RPS Channel 1 Reactor Coolant (RC) Pressure/Temperature (typical Channels 1 through 4)

RPS Shutdown Bypass High Pressure Trip

RPS Channel 1 Containment High Pressure Trip (typical Channels 1 through 4)

RPS Channel 1 RC High Pressure Trip (typical Channels 1 through 4)

RPS Channel 1 RC Low Pressure Trip (typical Channels 1 through 4)

RPS Channel 1 Channel Trip (typical Channels 1 through 4)

RPS Channel 1 High Flux Trip (typical Channels 1 through 4)

RPS Channel 1 RC High Temperature Trip (typical Channels 1 through 4)

RPS Startup Rate Rod Withdrawal Inhibit

RC Pressurizer Low Level Heater Interlock

RCP 1-1 Motor Trouble (typical RCPs 1-1, 1-2, 2-1, and 2-2)

Safety Features Actuation System (SFAS) Channel 1 Borated Water Storage Tank (BWST) Level Low (typical Channels 1 through 4)

SFAS Channel 1 Containment Pressure > 38.4 psia (typical Channels 1 through 4)

SFAS Channel 1 Containment Pressure > 18.4 psia (typical Channels 1 through 4)

SFAS Channel 1 RC Pressure < 1650 psig (typical Channels 1 through 4)

SFAS Channel 1 RC Pressure < 450 psig (typical Channels 1 through 4)

SFAS Channel 1 Containment Radiation High (typical Channels 1 through 4)

Steam and Feedwater Rupture Control System (SFRCS) Full Trip

SFRCS Differential Pressure Half/Full Trip Steam Generator (SG) 1 (typical SGs 1 and 2)

Startup Transformer 01 Trouble

Startup Transformer 02 Trouble

Switchyard Oscillograph Started

Switchyard Bus J Differential

Switchyard Breaker 34563 Open/Closed (typical five breakers)

Turbine Generator Mechanical Trip Solenoid Turbine Trip

Turbine Generator Master Turbine Trip

Turbine Generator Mechanical Trip Valve Trip

Turbine Generator Master Trip Solenoid Trip

Turbine Bypass Valve 1-1 Open/Closed (typical six valves)

Unit Seismic Instrumentation Started

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Enclosure 4
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:D DAVIS-BESSE UNIT 1

!ØUP 10 SEQUENCE ØF EVENTS REVIEW
1:30:27 10/15/83

1:26:13:170	X027	SWYD ACB 34562	ØPEN
1:26:31:495	P702	SFRCS DP HALF/FULL TRIP ,SG 2	TRIP
1:26:31:505	P702	SFRCS DP HALF/FULL TRIP ,SG 2	NØRM
1:26:41:895	Q963	SFRCS FULL TRIP	TRIP
1:26:41:905	X038	T-G MASTER TURB TRIP	TRIP
1:26:41:940	X030	T-G MASTER TRIP SØLENØIDS	TRIP
1:26:41:945	Q181	CRD CH B/D ANY TRIP DEVICE	TRIP
1:26:41:950	Q180	CRD CH A/C ANY TRIP DEVICE	TRIP
1:26:42: 45	Q266	CRD TRIP CØNFIRM	TRIP
1:26:42:165	X033	T-G MECH TRIP SØLENØID TURB TRIP	TRIP
1:26:42:185	X032	T-G MECH TRIP VLV	TRIP
1:26:44:420	P382	EHC EMER TRIP SYS LØW PRESS	TRIP
1:27:11:540	J428	GEN REVERSE PWR	TRIP
1:27:11:565	X026	SWYD ACB 34561	ØPEN
1:27:11:565	X025	SWYD ACB 34560	ØPEN
1:27:13:630	J428	GEN REVERSE PWR	NØRM
1:27:32: 95	Y060	TURB BYPASS VLV 1-1	NC
1:27:32:445	Y060	TURB BYPASS VLV 1-1	CLØS
1:27:50:630	Y063	TURB BYPASS VLV 2-1	NC
1:27:52: 5	Y063	TURB BYPASS VLV 2-1	CLØS
1:33: 1:815	Q963	SFRCS FULL TRIP	NØRM
1:33:18:730	Q841	RPS SU RATE RØD WTHDRWL INHIBIT	INHB
1:34: 3:165	Q841	RPS SU RATE RØD WTHDRWL INHIBIT	NØRM
1:13:27:755	X026	SWYD ACB 34561	CLØS
1:13:32:820	X025	SWYD ACB 34560	CLØS

POST TRIP REVIEW POINT LIST

Auxiliary Feed Pump Turbine 1 Speed (typical Pumps 1 and 2)

Channel 1 Power Range Flux (typical Channels 1 through 4)

Channel 1 Power Range Delta Flux (typical Channels 1 through 4)

Condensate Pump Flow

Control Rod Drive Group 5 Position (typical Groups 5 through 8)

Deaerator 1 Storage Tank Level (typical Deaerators 1 and 2)

Generator Gross Megawatts

High Pressure Condenser Pressure

High Pressure Condenser Hotwell Level

High Pressure Turbine First Stage Turbine End Pressure

High Pressure Turbine First Stage Generator End Pressure

High Pressure Turbine Side 1 Inlet Temperature

High Pressure Turbine Side 2 Inlet Temperature

Low Pressure Condenser Pressure

Main Feedwater Average Flow Loop 1 (typical Loops 1 and 2)

Main Feedwater Temperature (typical Loops 1 and 2)

Main Feedwater Compensated Flow (typical Loops 1 and 2)

Main Feedwater Pump Turbine 1 Speed (typical Pumps 1 and 2)

Pressurizer Average Level

Pressurizer Pressure

Reactor Coolant Makeup Tank Level

Reactor Coolant Makeup Flow

Reactor Coolant Pump (RCP) Seal Injection Flow

RCP 1-1 Discharge Cold Leg Narrow Range Temperature (typical RCPs 1-1 and 2-1)

Reactor Coolant System (RCS) Loop 1 Hot Leg Narrow Range Temperature (typical Loops 1 and 2)

RCS Average Temperature

RCS Loop 1 Hot Leg Narrow Range Pressure (typical Loops 1 and 2)

RCS Average Hot Leg Total Flow

RCS Letdown Boron Concentration

Safety Features Actuation System (SFAS) Channel 1 Containment Pressure

SFAS Channel 1 Containment Radiation Core Power

SFAS Channel 3 Borated Water Storage Tank Level

Steam Generator (SG) 1 Full Range Level (typical SGs 1 and 2)

SG 1 Startup Level (typical SGs 1 and 2)

SG 1 Operate Level (typical SGs 1 and 2)

SG 1 Outlet Temperature (typical SGs 1 and 2)

SG 1 Outlet Pressure (typical SGs 1 and 2)

SG 1 Feedwater Pressure (typical SGs 1 and 2)

TED DAVIS-BESSE UNIT NO. 1

GROUP 3 POSTTRIP REVIEW
19:52:23 1/15/1983

1/15/1983

REACTOR PROTECTION SYSTEM (RPS)

TIME	CH1 PR NI6 FLUX% R795	CH2 PR NI5 FLUX% R804	CH3 PR NI8 FLUX% R814	CH4 PR NI7 FLUX% R820	CH1 PR NI6 OFLUX% R794	CH2 PR NI5 OFLUX% R803	CH3 PR NI8 OFLUX% R813	CH4 PR NI7 OFLUX% R819
16:44: 9	101,0	100,6	100,8	100,3	-12,888	-12,415	-12,796	-13,040
16:44:24	100,8	100,6	100,9	100,6	-12,964	-12,521	-12,857	-12,979
16:44:39	100,8	100,6	101,0	100,6	-12,964	-12,521	-12,918	-13,086
16:44:54	101,0	100,6	101,0	100,6	-12,964	-12,521	-12,918	-13,086
16:45: 9	100,9	100,4	100,7	100,5	-13,086	-12,521	-12,918	-13,086
16:45:24	101,0	100,7	101,0	100,3	-13,086	-12,613	-12,918	-13,086
16:45:39	101,2	100,6	101,0	100,5	-13,086	-12,613	-13,010	-13,147
16:45:54	100,4	100,5	100,2	99,8	-13,834	-13,345	-13,757	-14,017
16:46: 9	100,6	100,3	100,5	99,9	-13,925	-13,422	-13,757	-13,910
16:46:24	100,6	100,1	100,3	100,0	-13,925	-13,422	-13,834	-14,002
16:46:39	100,7	100,1	100,6	100,1	-14,047	-13,605	-13,834	-14,002
16:46:54	100,8	100,5	100,6	100,2	-14,047	-13,605	-13,925	-14,108
16:47: 9	100,8	100,3	100,6	100,2	-14,047	-13,605	-13,925	-14,108
16:47:24	100,7	100,4	100,6	100,3	-14,139	-13,712	-14,002	-14,215
16:47:39	101,0	100,3	100,8	100,4	-14,139	-13,635	-14,078	-14,215
16:47:54	100,8	100,7	100,5	100,2	-14,200	-13,696	-14,078	-14,292
16:48: 9	100,9	100,5	100,8	100,3	-14,200	-13,696	-14,078	-14,292
16:48:24	100,8	100,4	100,7	100,2	-14,353	-13,803	-14,139	-14,292
16:48:39	100,8	100,5	100,8	100,3	-14,353	-13,803	-14,139	-14,368
16:48:54	101,0	100,5	100,8	100,3	-14,276	-13,803	-14,200	-14,368
16:49: 9	100,9	100,4	100,7	100,4	-14,398	-13,864	-14,200	-14,368
16:49:24	101,0	100,6	100,9	100,3	-14,398	-13,864	-14,261	-14,429
16:49:39	101,0	100,6	100,9	100,4	-14,398	-13,925	-14,261	-14,429
16:49:54	100,8	100,4	100,7	100,4	-14,490	-14,002	-14,353	-14,429
16:50: 9	100,9	100,6	100,8	100,4	-14,490	-14,002	-14,353	-14,429
16:50:24	101,1	100,6	100,9	100,4	-14,490	-14,002	-14,353	-14,429
16:50:39	100,9	100,5	101,0	100,4	-14,490	-14,002	-14,353	-14,429
16:50:54	101,1	100,7	101,0	100,3	-14,490	-14,002	-14,429	-14,429
16:51: 9	101,2	100,6	100,9	100,3	-14,490	-14,093	-14,429	-14,429
16:51:24	101,1	100,4	100,9	100,5	-14,582	-14,093	-14,505	-14,521
16:51:39	100,9	101,0	100,9	100,5	-14,688	-14,200	-14,505	-14,521
16:51:54	101,3	100,5	100,8	100,6	-14,688	-14,108	-14,582	-14,627
16:52: 9	101,1	100,9	101,0	100,5	-14,826	-14,246	-14,490	-14,627
16:52:24	101,2	100,7	101,0	100,6	-14,826	-14,246	-14,582	-14,627
16:52:39	101,0	100,8	101,1	100,6	-14,826	-14,337	-14,673	-14,688
16:52:54	100,3	100,9	101,1	100,6	-15,894	-15,436	-14,673	-14,688
16:53: 9	100,4	99,9	100,1	99,8	-15,894	-15,436	-15,741	-16,184
16:53:24	100,6	100,2	100,6	100,3	-15,955	-15,436	-15,741	-16,184
16:53:39	100,8	100,4	100,7	100,2	-15,955	-15,543	-15,848	-16,184
16:53:54	100,8	100,5	100,6	100,2	-16,077	-15,619	-15,924	-16,275

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TED DAVIS-BESSE UNIT NO. 1

GROUP 3 POSTTRIP REVIEW
19:52:23 1/15/1983

1/15/1983

REACTOR PROTECTION SYSTEM (RPS)

TIME	CH1 PR NI6 FLUX% R795	CH2 PR NI5 FLUX% R804	CH3 PR NI8 FLUX% R814	CH4 PR NI7 FLUX% R82^	CH1 PR NI6 DFLUX% R794	CH2 PR NI5 DFLUX% R803	CH3 PR NI8 DFLUX% R813	CH4 PR NI7 DFLUX% R819
16:54: 9	100,6	100,4	100,6	100,4	-16,077	-15,619	-15,924	-16,275
16:54:24	100,9	100,6	100,9	100,3	-16,199	-15,619	-15,986	-16,275
16:54:39	101,0	100,6	100,9	100,3	-16,199	-15,680	-15,986	-16,337
16:54:54	101,0	100,5	100,7	100,3	-16,306	-15,680	-16,092	-16,398
16:55: 9	100,9	100,6	100,8	100,5	-16,306	-15,818	-16,092	-16,398
16:55:24	101,1	100,6	101,0	100,5	-16,306	-15,818	-16,184	-16,398
16:55:39	101,0	100,7	101,0	100,5	-16,398	-15,970	-16,184	-16,398
16:55:54	101,0	100,4	100,9	100,6	-16,398	-15,970	-16,275	-16,550
16:56: 9	101,1	100,6	100,7	100,4	-16,474	-15,970	-16,275	-16,550
16:56:24	100,9	100,4	100,8	100,5	-16,535	-16,047	-16,352	-16,550
16:56:39	100,9	100,5	101,0	100,8	-16,535	-16,108	-16,352	-16,550
16:56:54	100,9	100,6	100,8	100,4	-16,535	-16,047	-16,352	-16,550
16:57: 9	100,7	100,6	100,6	100,7	-16,535	-16,123	-16,459	-16,611
16:57:24	100,8	100,4	100,6	100,6	-16,535	-16,123	-16,459	-16,688
16:57:39	101,0	100,4	100,8	100,6	-16,535	-16,214	-16,459	-16,688
16:57:54	100,9	100,5	100,7	100,6	-16,642	-16,214	-16,565	-16,688
16:58: 9	101,1	100,6	100,9	100,4	-16,642	-16,214	-16,565	-16,688
16:58:24	100,4	100,1	100,1	99,8	-17,252	-16,825	-17,054	-17,420
16:58:39	100,7	100,1	100,4	100,3	-17,252	-16,886	-17,161	-17,420
16:58:54	100,7	100,5	100,5	100,3	-17,374	-16,962	-17,267	-17,420
	******************************** TRIP ********************************							
16:59:12	100,7	100,3	100,5	100,2	-17,374	-17,023	-17,267	-17,496
16:59:26	4,3	3,2	100,5	100,5	-0,954	-0,847	-0,893	-17,496
16:59:41	1,9	1,5	3,0	2,4	-0,374	-0,465	-0,572	-0,832
16:59:56	1,1	0,9	1,6	1,2	-0,267	-0,359	-0,481	-0,557
17: 0: 9	0,7	0,9	1,0	0,7	-0,206	-0,298	-0,481	-0,450
17: 0:26	0,5	0,3	0,7	0,3	-0,206	-0,298	-0,420	-0,450
17: 0:39	0,3	0,3	0,4	0,1	-0,130	-0,298	-0,420	-0,389
17: 0:55	0,2	0,0	0,3	0,0	-0,130	-0,237	-0,420	-0,389
17: 1: 9	0,1	0,0	0,2	-0,1	-0,130	-0,237	-0,343	-0,389
17: 1:27	0,1	-0,1	0,1	-0,1	-0,130	-0,237	-0,343	-0,328
17: 1:39	0,1	-0,1	0,1	-0,2	-0,130	-0,237	-0,343	-0,328
17: 1:54	0,0	-0,1	0,0	-0,2	-0,130	-0,237	-0,343	-0,328
17: 2: 9	0,0	-0,1	0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 2:24	0,0	-0,1	0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 2:39	0,0	-0,1	0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 2:54	0,0	-0,1	0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 3: 9	0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 3:24	0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 3:39	0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 3:54	0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328

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TED E -BESSE UNIT NO. 1

GROUP 3 POSTTRIP REVIEW
19:52:23 1/15/1983

1/15/1983

REACTOR PROTECTION SYSTEM (RPS)

TIME	CH1 PR NI6 FLUX% R795	CH2 PR NI5 FLUX% R804	CH3 PR NI8 FLUX% R814	CH4 PR NI7 FLUX% R820	CH1 PR NI6 OFLUX% R794	CH2 PR NI5 OFLUX% R803	CH3 PR NI8 OFLUX% R813	CH4 PR NI7 OFLUX% R819
17: 4: 9	-0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 4:25	-0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 4:39	-0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 4:54	-0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 5: 9	-0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 5:24	-0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 5:39	-0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 5:54	-0,0	-0,2	-0,0	-0,3	-0,130	-0,237	-0,343	-0,328
17: 6: 9	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 6:24	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 6:39	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 6:54	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 7: 9	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 7:24	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 7:39	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 7:54	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 8: 9	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 8:24	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 8:39	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 8:54	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 9: 9	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 9:24	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 9:39	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17: 9:54	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:10: 9	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:10:24	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:10:39	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:10:54	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:11: 9	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:11:24	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:11:39	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:11:54	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:12: 9	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:12:24	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:12:39	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:12:54	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:13: 9	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:13:24	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:13:39	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328
17:13:54	-0,0	-0,2	-0,1	-0,3	-0,069	-0,237	-0,343	-0,328

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10:14:16 CONT Q392  EHC PWR DISCH FLT [illegible]                        NORM
10:14:16 CONT Q418  RPREM BUS CI CTRL PWR                                 NORM
10:14:20 BAD  V661  MFPT 2 BRP END BRG VIB (MILS)            -0.18            0.00
10:14:24 CONT Z464  HP FW HTR 1-6 HI LVL DRN VLV                          CLOS
10:14:26 CONT Q627  MFPT 1 MN OIL PMP 2                                   ON
10:14:29 CONT L466  HP FW HTR 2-4 LVL                                     HIGH
10:14:30 NORM P931  SG 1 OUT STM PRESS+PT12B1               901.06          901.93
10:14:43 NORM T968  DEAR HTR 1 TERM TEMP DIFF                 0.16            3.00
10:14:48 CONT L466  HP FW HTR 2-4 LVL                                     NORM
10:14:58 CONT Q905  SFAS CTMT AIR CLR LOGIC L223                          TRIP
10:14:59 CONT L762  RC MU TK LVL+MU16-2                                   HILO
10:14:59 FLAG X258  SFAS RRRR PWR LOGIC L263                              TRIP
10:15: 1 CONT L760  RC MU TK LVL+MU16-1                                   HILO
10:15:08 CONT Q905  SFAS CTMT AIR CLR LOGIC L223                          NORM
10:15:10 CONT Z732  RC DIVERTING VLV OPEN TO MU TK                        NO
10:15:10 CONT Q001  ARTS IN FROM MFPT                                     TRIP
10:15:12 LOW  A864  RPS CH 3 IR SU RATE DR14 (DPM)           -0.11            0.00
10:15:17 HIGH L761  RC MU TK LVL+MU16-1 (IN)                  [illegible]     87.00
10:15:18 CONT Z733  RC DIVERTING VLV OPEN TO RD SYS                       CLOS
10:15:20 NORM V661  MFPT 2 BRP END BRG VIB (MILS)            -0.17            2.00
10:15:23 CONT L760  RC MU TK LVL+MU16-1                                   NORM
10:15:27 CONT L762  RC MU TK LVL+MU16-2                                   NORM
CURRENT DATE IS: OCT., 19, 1983
10:15:43 NORM A871  RPS CH 4 IR SU RATE DR13 (DPM)            0.14            1.00
10:15:46 CONT Q853  SFAS CC WTR LOGIC L233                                TRIP
10:15:58 CONT Q853  SFAS CC WTR LOGIC L233                                NORM
10:16:02 NORM L761  RC MU TK LVL+MU16-1 (IN)                 84.75           88.00
10:16: 6 CONT L466  HP FW HTR 2-4 LVL                                     HIGH
10:16:13 NORM A864  RPS CH 3 IR SU RATE DR14 (DPM)            0.13            1.00
10:16:27 CONT L466  HP FW HTR 2-4 LVL                                     NORM
10:16:35 CONT Z464  HP FW HTR 1-6 HI LVL DRN VLV                          NO
10:16:43 LOW  T968  DEAR HTR 1 TERM TEMP DIFF                -0.03            0.00
10:16:43 LOW  A871  RPS CH 4 IR SU RATE DR13 (DPM)           -0.12            0.00
10:16:58 CONT Q953  SFAS SW LOGIC L243                                    TRIP
10:17:12 LOW  A864  RPS CH 3 IR SU RATE DR14 (DPM)           -0.10            0.00
10:17:13 CONT Q953  SFAS SW LOGIC L243                                    NORM
10:17:16 CONT Q001  ARTS IN FROM MFPT                                     NORM
10:17:20 BAD  V661  MFPT 2 BRP END BRG VIB (MILS)            -0.18            0.00
10:17:24 CONT Q004  ARTS TEST TRIP                                        NORM
10:17:28 CONT L466  HP FW HTR 2-4 LVL                                     HIGH
10:17:43 CONT L466  HP FW HTR 2-4 LVL                                     NORM
10:17:43 NORM T968  DEAR HTR 1 TERM TEMP DIFF                 0.06            3.00
10:17:43 NORM A871  RPS CH 4 IR SU RATE DR13 (DPM)            0.10            1.00
10:17:50 NORM V661  MFPT 2 BRP END BRG VIB (MILS)            -0.18            2.00
10:18: 4 CONT L466  HP FW HTR 2-4 LVL                                     HIGH
10:18:15 HIGH P931  SG 1 OUT STM PRESS+PT12B1               902.89          [illegible]
10:18:19 CONT L466  HP FW HTR 2-4 LVL                                     NORM
10:18:24 CONT Z464  HP FW HTR 1-6 HI LVL DRN VLV                          CLOS
10:18:2[illegible] CONT Q931  SFAS CS VLV LOGIC L213                      TRIP
10:18:26 FLAG X206  SFAS CTMT IRO 2 LOGIC L213                            TRIP
10:18:30 NORM P931  SG 1 OUT STM PRESS+PT12B1               899.60          901.91
10:18:40 CONT L466  HP FW HTR 2-4 LVL                                     HIGH
10:18:42 NORM A864  RPS CH 3 IR SU RATE DR14 (DPM)            0.12            1.00
10:18:43 LOW  A871  RPS CH 4 IR SU RATE DR13 (DPM)           -0.01            0.00
10:18:43 CONT Q931  SFAS CS VLV LOGIC L203                                NORM
10:18:5[illegible] CONT L466  HP FW HTR 2-4 LVL                           NORM
10:19:13 CONT L466  HP FW HTR 2-4 LVL                                     HIGH
10:19:13 CONT Q929  SFAS RRRR PWR LOGIC L263                              TRIP
10:19:14 CONT Q004  ARTS TEST TRIP                                        BYPS
10:19:14 FLAG Q072  CTMT H2 DILUTN OUT ISO VLV, 5038                      NO
10:19:14 FLAG Q072  CTMT H2 DILUTN OUT ISO VLV, 5038                      NORM
10:19:30 CONT L466  HP FW HTR 2-4 LVL                                     NORM
10:19:36 LOW  [illegible]  MFP 1 DISCH PRESS                 44.01          [illegible]
10:19:42 LOW  A864  RPS CH 3 IR SU RATE DR14 (DPM)           -0.08            0.00
10:20:13 NORM A871  RPS CH 4 IR SU RATE DR13 (DPM)            0.[illegible]   1.00
10:20:20 BAD  V661  MFPT 2 BRP END BRG VIB (MILS)            -0.17            0.00
```

OCT 19 1983

DATA ACQUISITION AND DISPLAY SYSTEM RECORDED POINTS

Auxiliary Feedwater Flow to Steam Generator (SG) 1 (typical SGs 1 and 2)

Auxiliary Feed Pump 1 Discharge Pressure (typical Pumps 1 and 2)

Auxiliary Feed Pump Turbine 1 Speed (typical Pumps 1 and 2)

Containment Hydrogen Concentration

Containment Spray Pump 1 Discharge Flow (typical Pumps 1 and 2)

Containment Normal Sump Level

Containment Wide Range Level

Containment Wide Range Pressure

Containment Atmosphere Particulate Radiation

Containment Atmosphere Iodine Radiation

Containment Atmosphere Noble Gas Radiation

Containment Atmosphere Noble Gas Mid to High Range Radiation

Containment Wide Range Radiation

Unit Vent Particulate Radiation

Unit Vent Iodine 131 Radiation

Unit Vent Xenon 133 Radiation

Generator Gross Megawatts

High Pressure Injection 1-1 Flow (typical Lines 1-1, 1-2, 2-1, and 2-2)

Incore Outlet Temperature (typical 16 sensors)

Low Pressure Injection Pump 1 Flow (typical Pumps 1 and 2)

Main Feedwater Temperature to Integrated Control System

Main Feedwater Control Valve Position Loop 1 (typical Loops 1 and 2)

Main Feedwater Startup Control Valve Position Loop 1 (typical Loops 1 and 2)

Main Feedwater Compensated Flow Loop 1 (typical Loops 1 and 2)

Main Feedwater Startup Flow Loop 1 (typical Loops 1 and 2)

Reactor Coolant Makeup Tank Level

Reactor Coolant System (RCS) Hot Leg Flow Loop 1 (Loops 1 and 2)

RCS Pressurizer Compensated Level

RCS Pressurizer Quench Tank Level

RCS Pressurizer Quench Tank Pressure

RCS Hot Leg Wide Range Pressure Loop 1 (typical Loops 1 and 2)

RCS Average Narrow Range Temperature

RCS Calculated Hot Leg Subcooled Margin Channel A

RCS Calculated Hot Leg Subcooled Margin Channel B

RCS Hot Leg Wide Range Temperature Loop 1 (typical Loops 1 and 2)

RCS Pressurizer Temperature

RCS Pressurizer Power Operated Relief Valve Position

RCS Pressurizer Pressure Relief Valve Position (typical Valves 1 and 2)

Reactor Coolant Pump (RCP) 1-1 Discharge Cold Leg Wide Range Temperature (typical RCPs 1-1, 1-2, 2-1, and 2-2)

Reactor Protection System (RPS) Auctioneered Average Power

RPS Channel 1 Power Range Flux (typical Channels 1 through 4)

RPS Channel 1 Source Range Flux (typical Channels 1 and 2)

RPS Channel 3 Intermediate Range Flux (typical Channels 3 and 4)

Safety Features Actuation System (SFAS) Channel 1 Borated Water Storage Tank Level

Steam Generator (SG) 1 Outlet Steam Temperature (typical SGs 1 and 2)

SG 1 Operate Level (typical SGs 1 and 2)

SG 1 Startup Range Level (typical SGs 1 and 2)

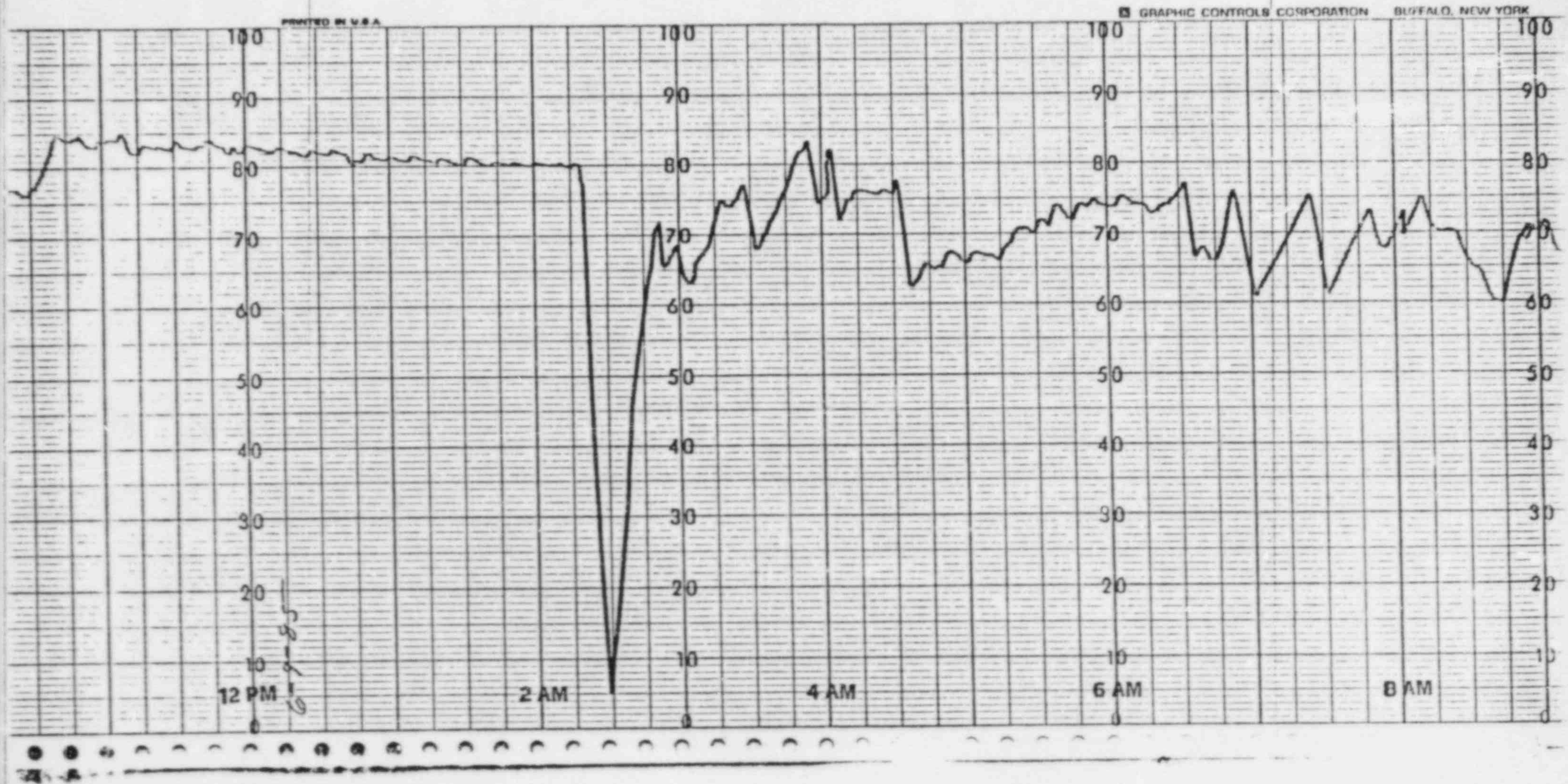
SG 1 Outlet Pressure (typical SGs 1 and 2)

109 PDR

Strip Charts for Makeup Tank

LR MU 16

6-9-85



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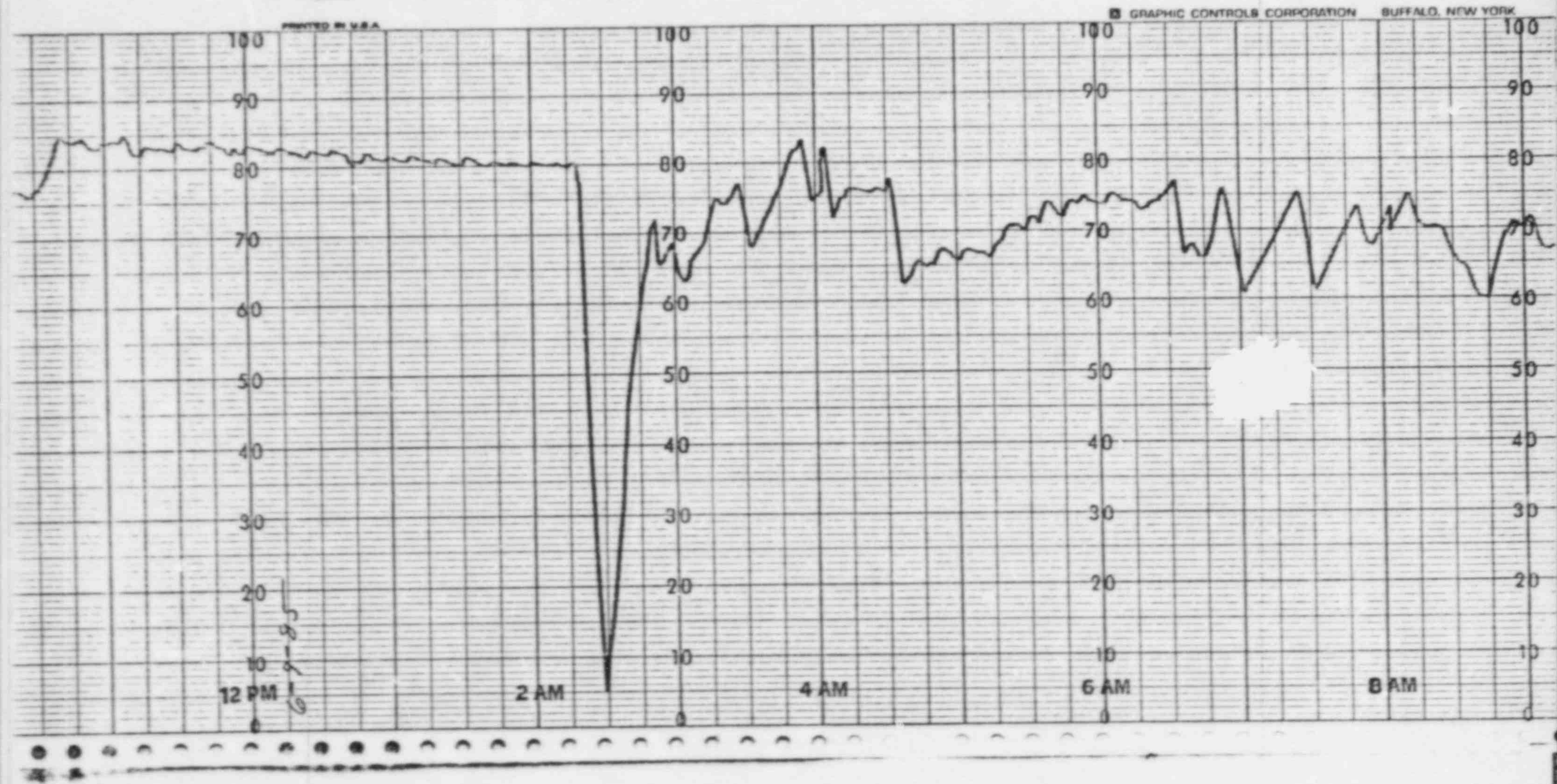
LR MU 16

6-9-85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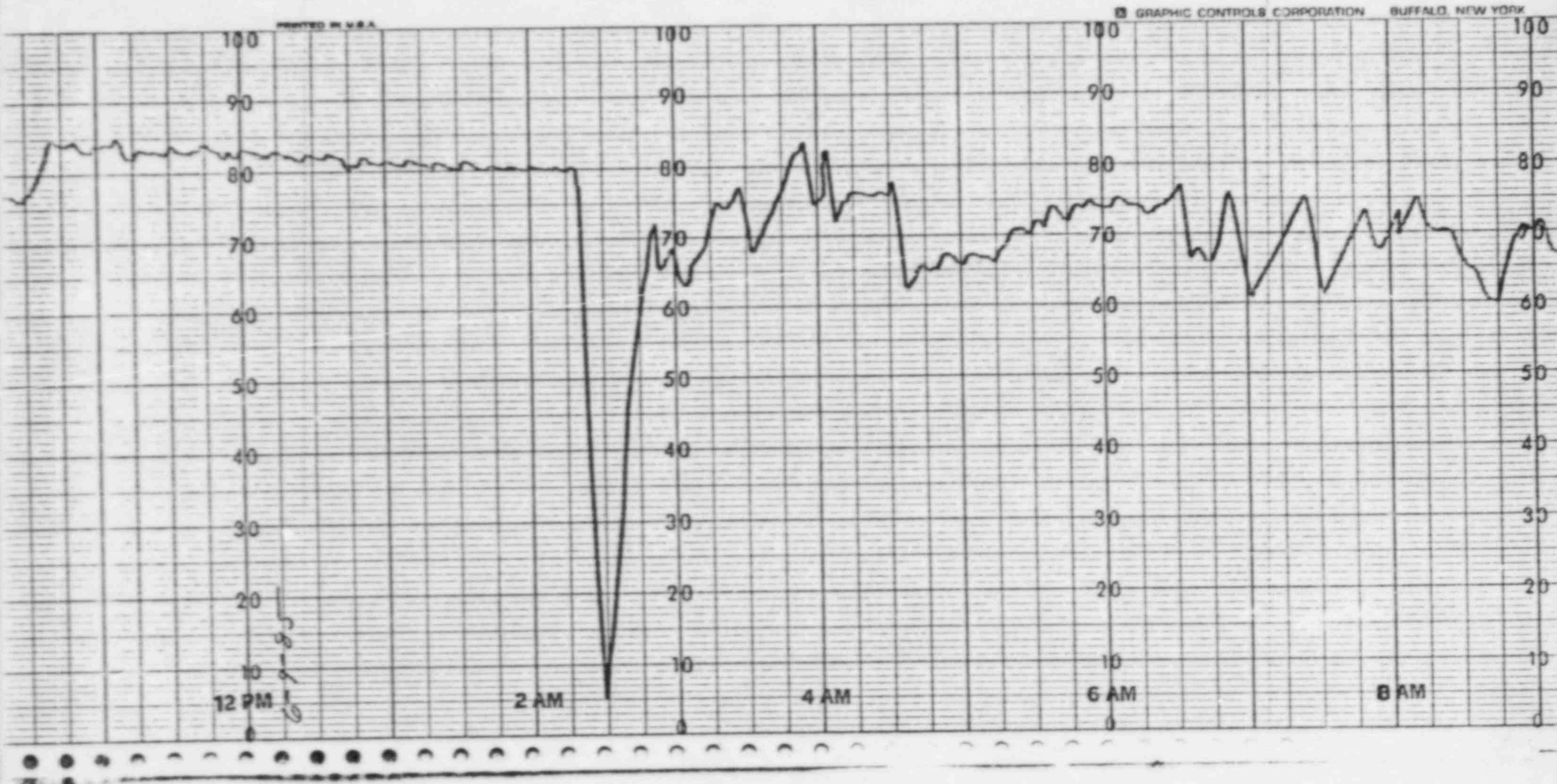
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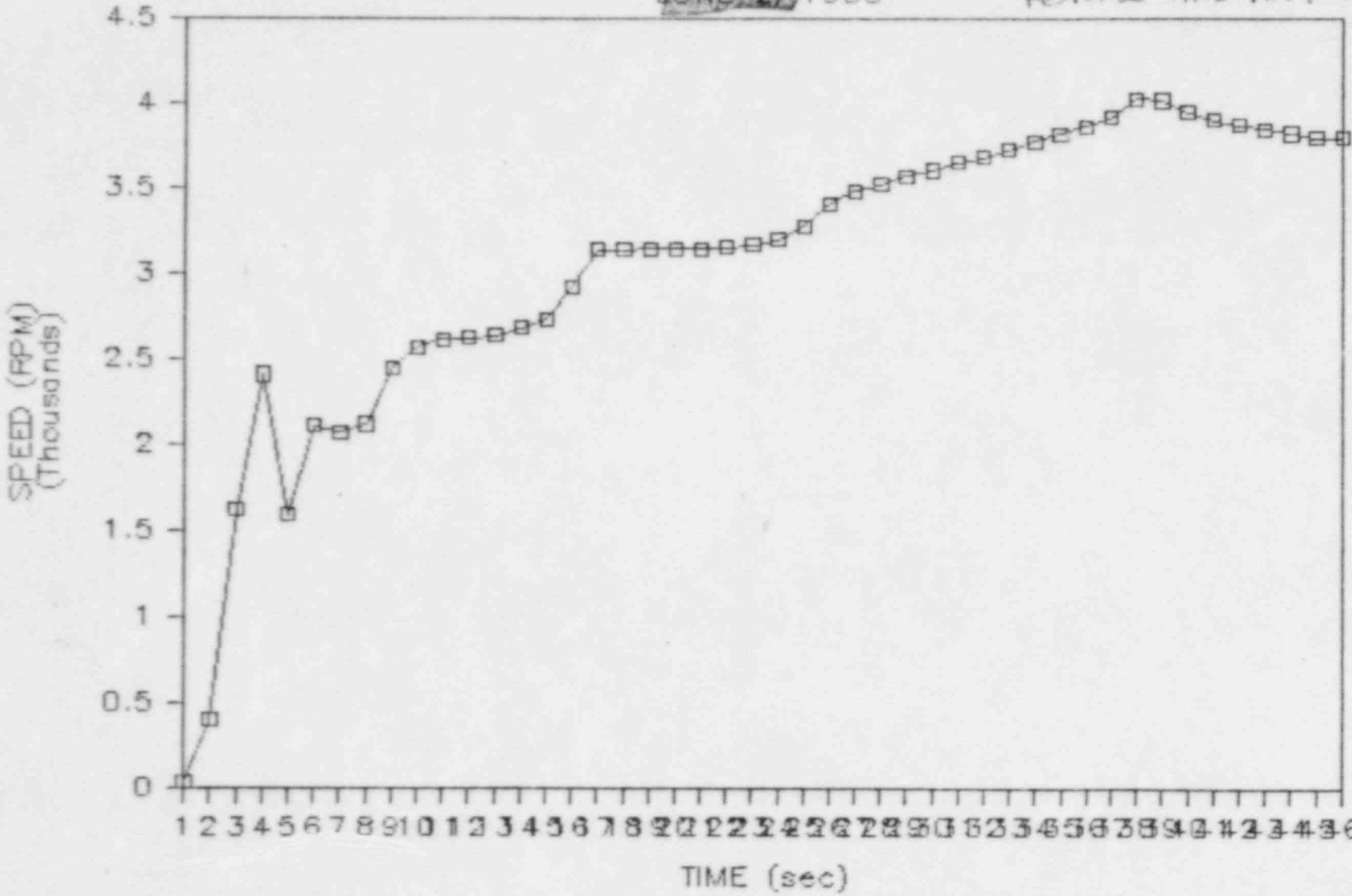
(25)

AUX FEED PUMP [redacted] SPEED vs. TIME

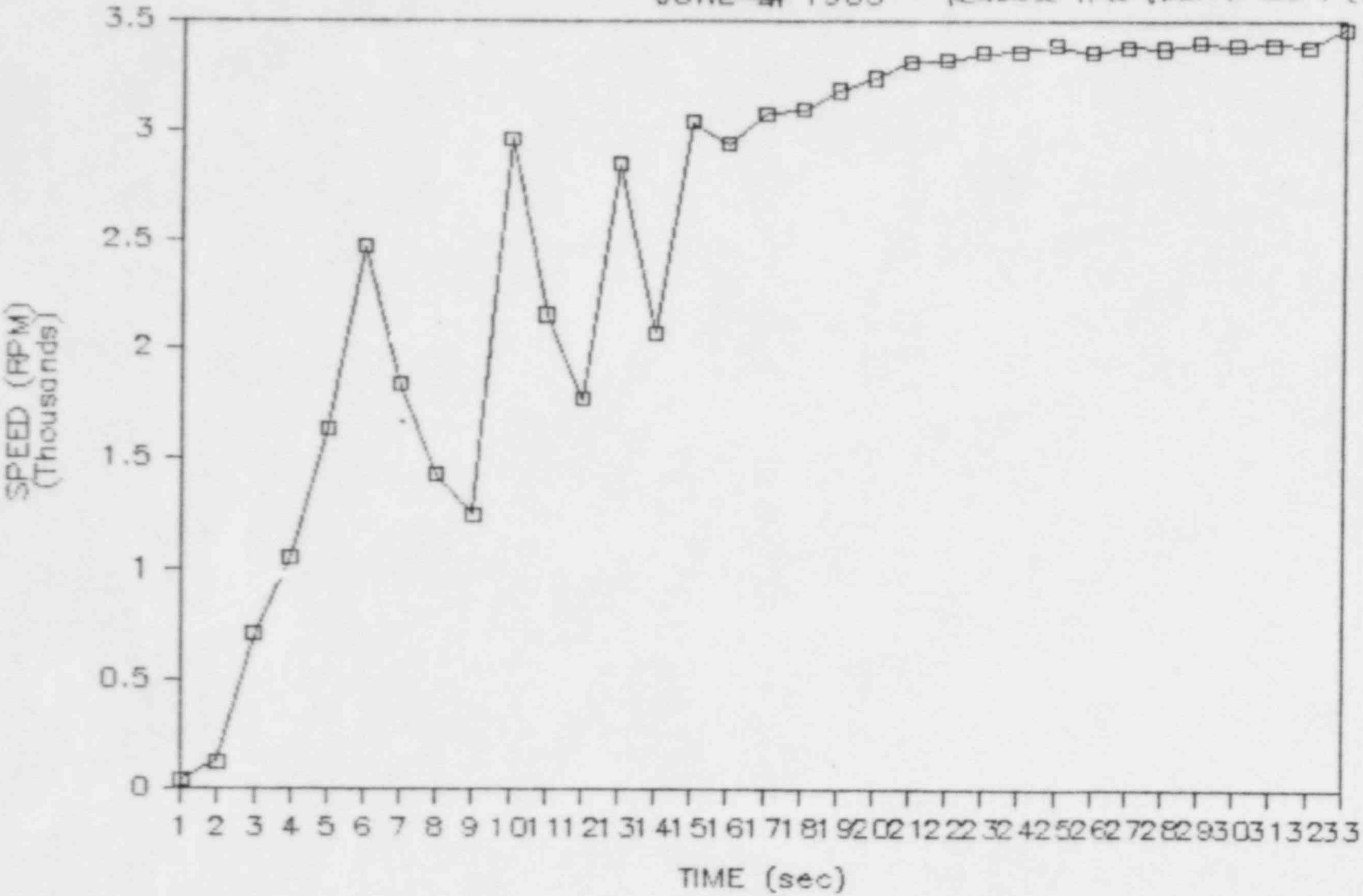
JUNE 2, 1985

RESPONSE TIME PRIOR TO EVENT (POSSIBLY JUNE ...)



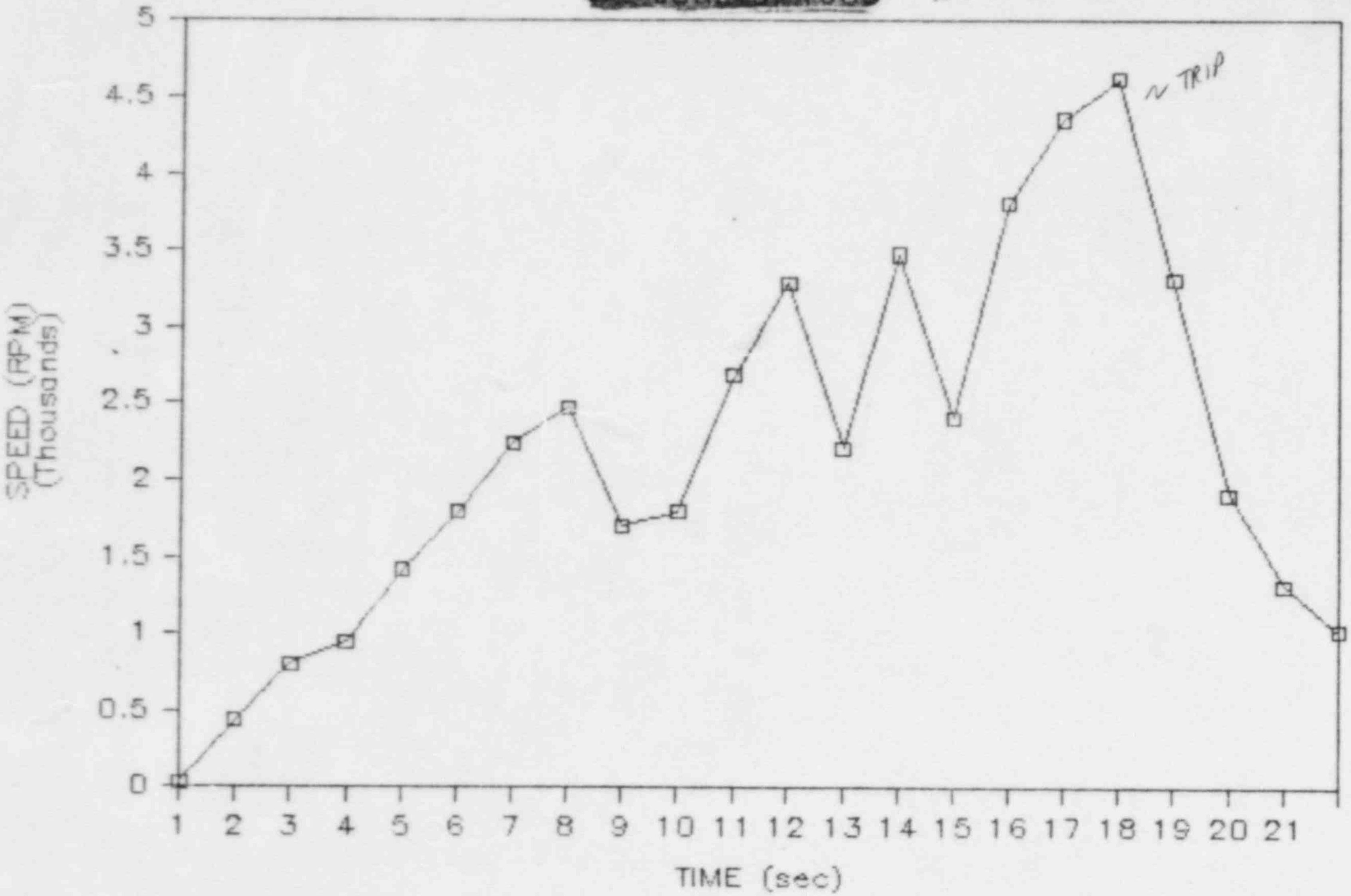
AUX FEED PUMP [redacted] SPEED vs. TIME

[redacted] 1985 RESPONSE TIME PRIOR TO EVENT (POSSIBLY JUNE 2 TRIP)



AUX FEED PUMP [redacted] SPEED vs. TIME

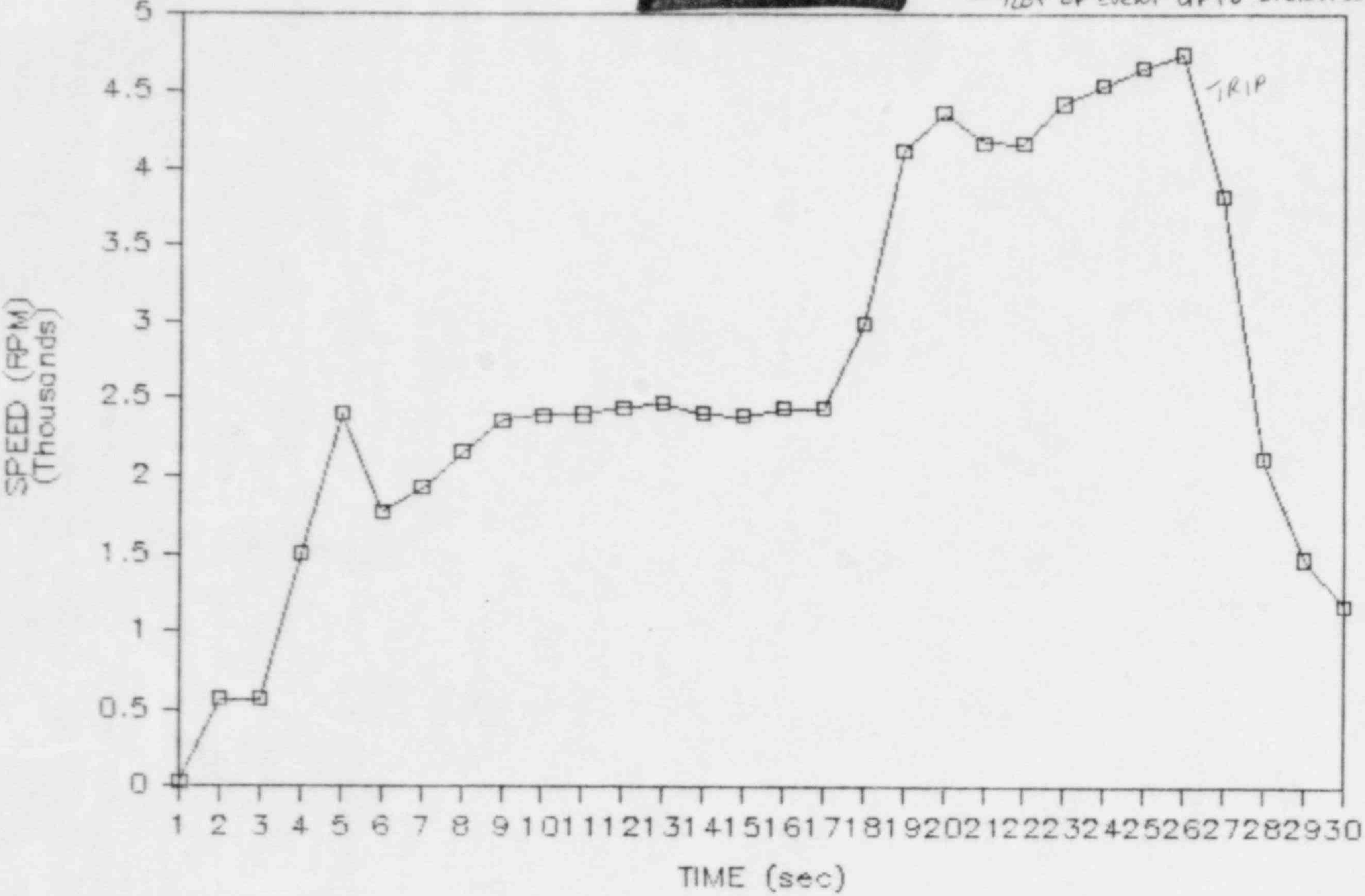
[redacted] ROT OF EVENT UP TO OVERSPEED TRIP



AUX FEED PUMP [redacted] SPEED vs. TIME

[redacted]

—PLOT OF EVENT UP TO OVERSPEED TRIP

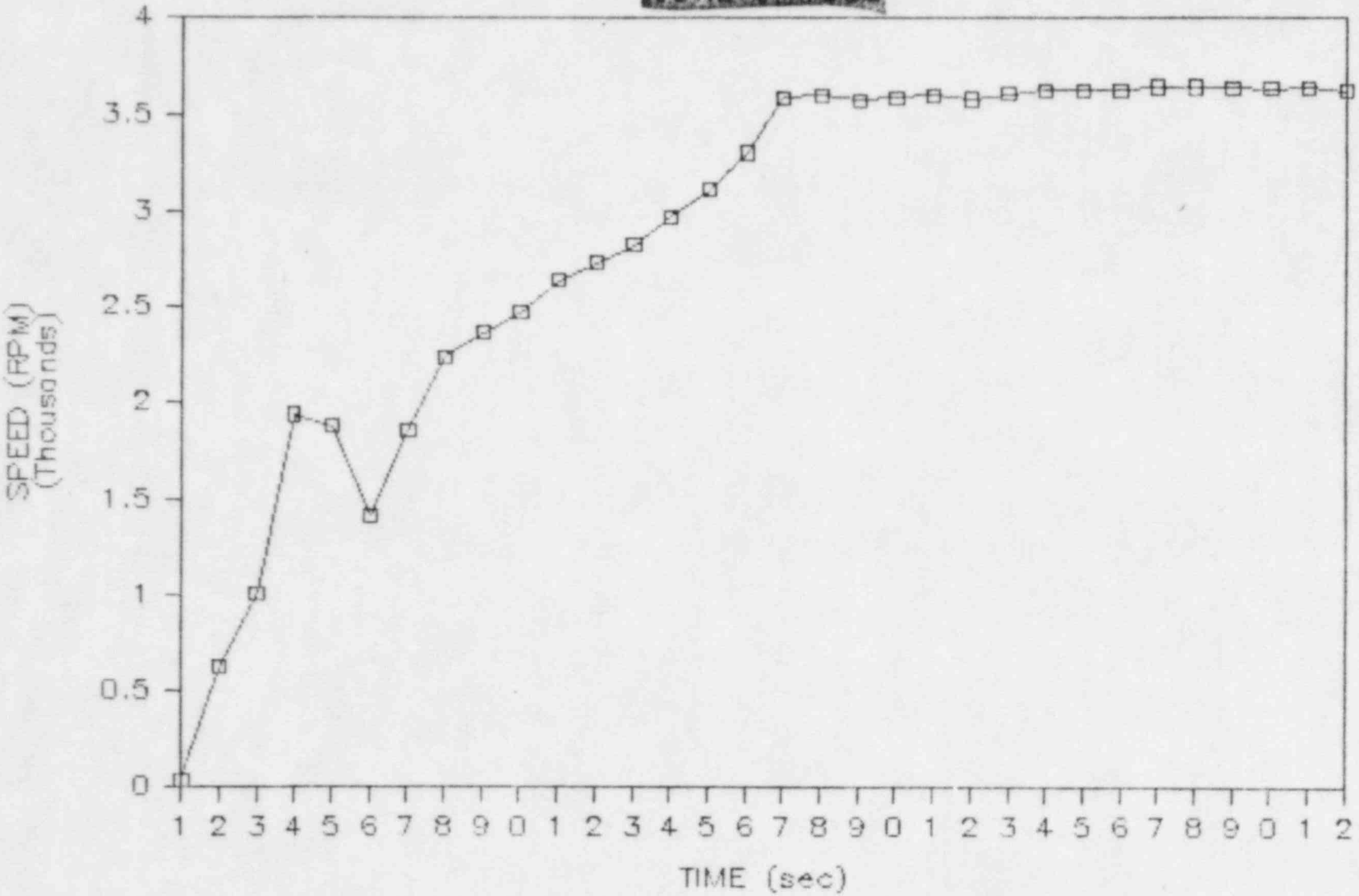


TEST

PERFORMED AFTER EVENT

AUX FEED PUMP [redacted] SPEED vs. TIME

[redacted]

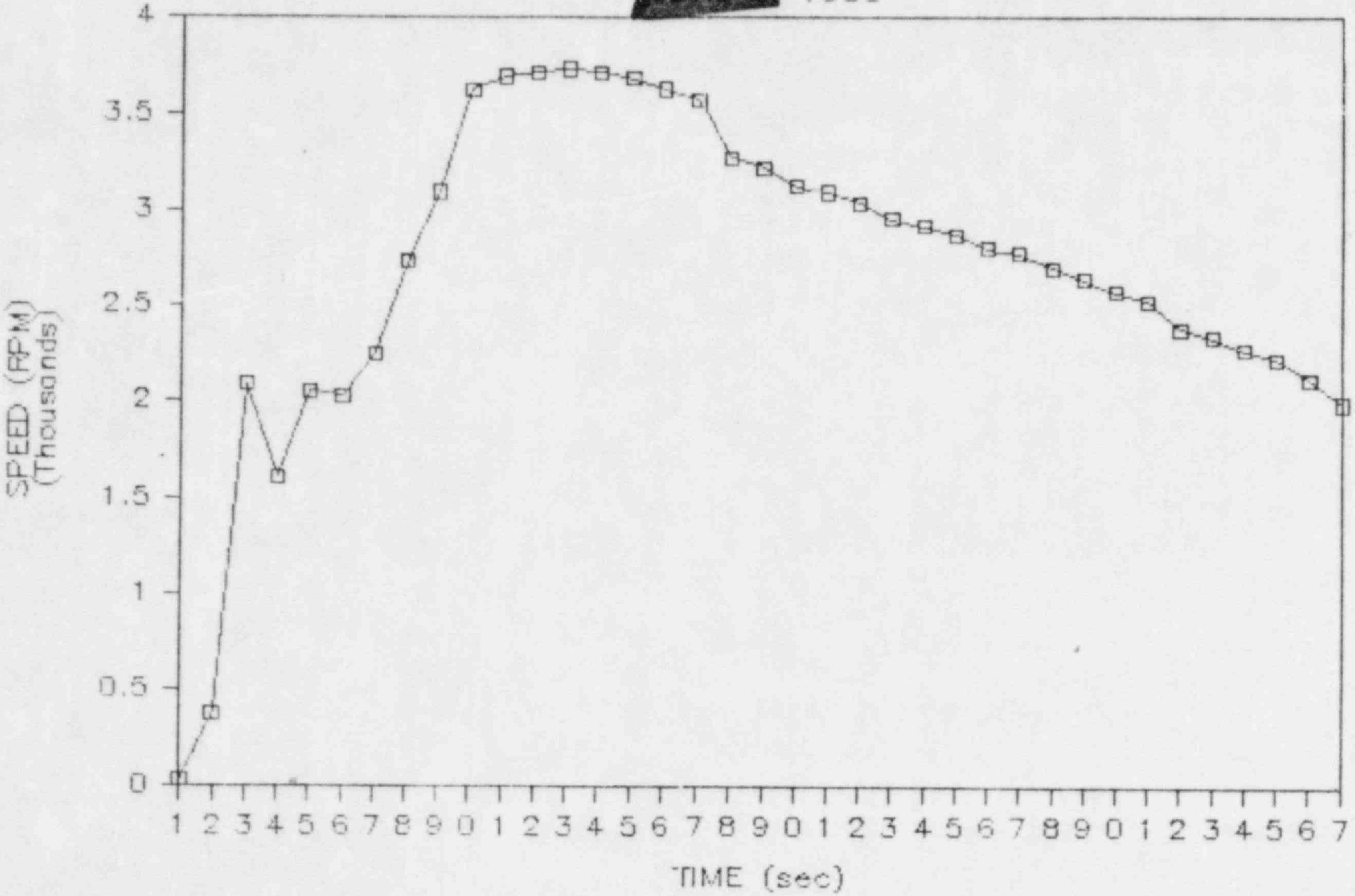


TEST

PERFORMED AFTER EVENT

AUX FEED PUMP [redacted] SPEED vs. TIME

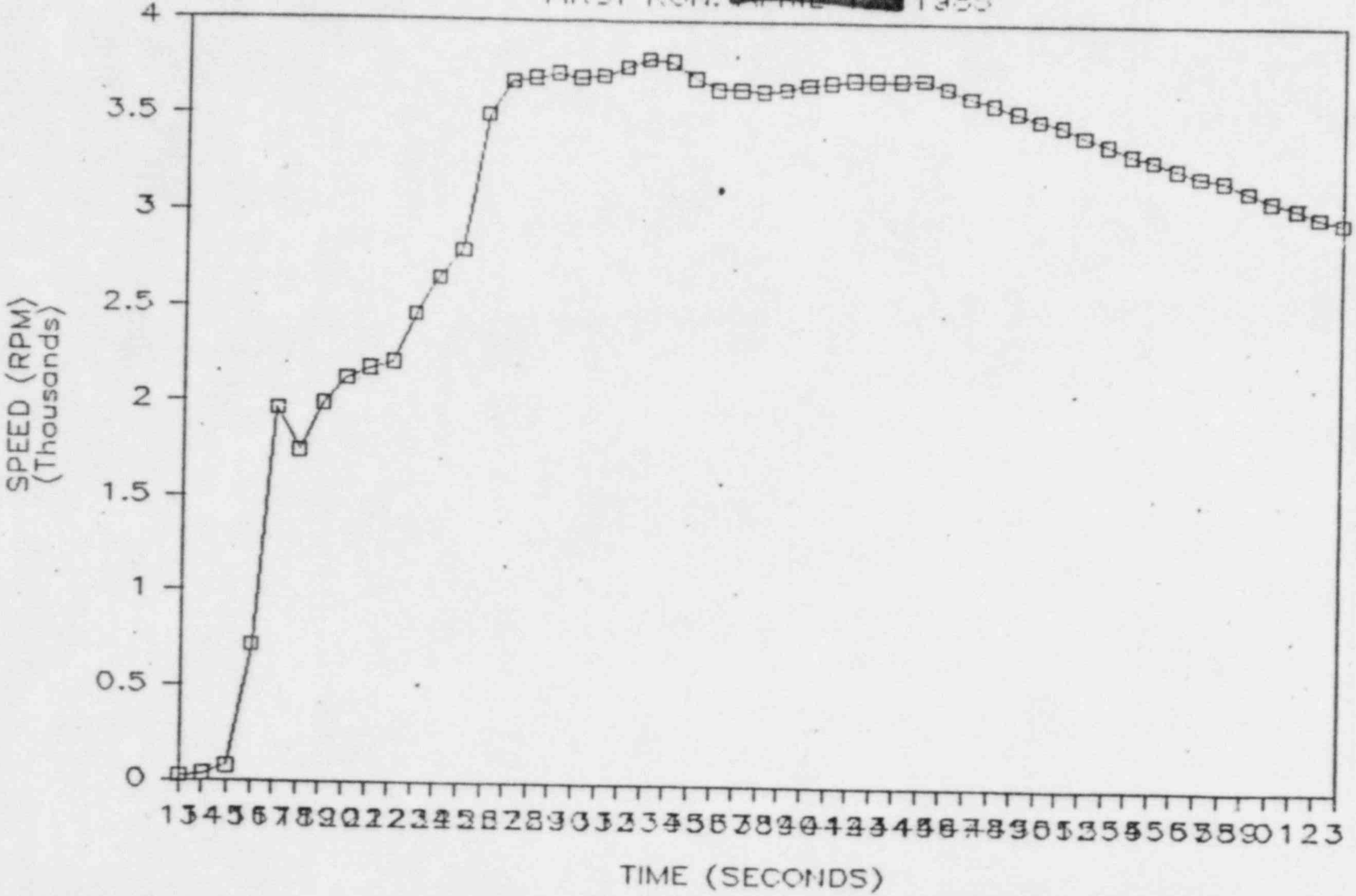
[redacted] 1985



AFPT 1[redacted] SPEED TREND DATA

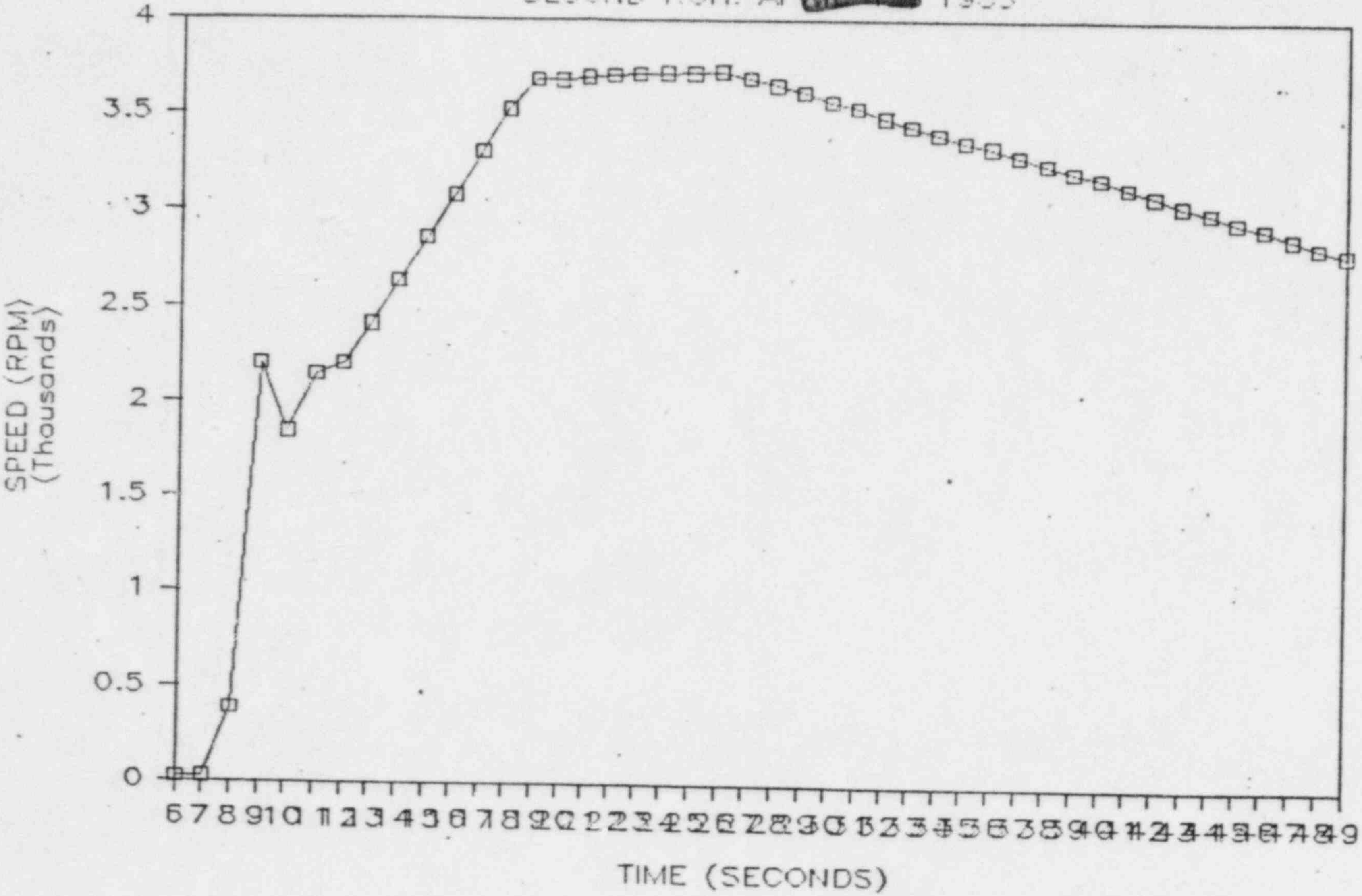
TESTING #2 PUMP FOR GOVERNOR REPLACEMENT

FIRST RUN: [redacted] 1985



AFPT [redacted] SPEED TREND DATA

SECOND RUN: APR[redacted] 1985



23 PDR

FR-SP4A	Steam Generator 2 Main Feedwater Flow
FR-SP4B	Steam Generator 1 Main Feedwater Flow
FRS-RC1	Reactor Coolant Hotleg Total Flow
JR 6013	Generator Wattage Recorder
LR-MU-16	Reactor Coolant Makeup Tank Level
LRS-SP-9A	Steam Generator 2 Operate Level
LRS-SP-9B	Steam Generator 1 Operate Level
LRS-RC-14	Pressurizer 1-1 Level
NR 3300B	Incore Monitor Recorder
NR-NI 1&3	Source Range Neutron Flux & Inter-range Neutron Flux
NR/NI 2&4	Source Range Neutron Flux & Inter-range Neutron Flux
NR/NI 3-1	Inter-range Neutron Flux
NR/NI 6	Power Range Neutron Flux
PR 530/541	High Pressure Condenser Pressure/Low Pressure Condenser Pressure
PRS-RC-2A1	Reactor Coolant Loop 2 Hotleg Wide Range SFAS Channel 2
PRS-RC-2A2	Reactor Coolant Loop 2 Hotleg Narrow Range Pressure
PRS-RC-2B	Reactor Coolant Hotleg Loop 1 Narrow Range Pressure
PRS-SP-16	Main Steamline Pressure
TJR 2508A	Low Pressure Turbine 1 G/E Exhaust Hood Temperature
TJR 2508B	Low Pressure Turbine 2 G/E Exhaust Hood Temperature
TJR 2508C	Main Turbine Differential Expansion
TJR 2508D	Main Turbine Differential Shell Expansion
TJR 2508E	First Stage Shell Lower Inner
TJR 2508F	First Stage Shell Lower Inner
TJR 2508G	First Stage Shell Lower Outer
TJR 2508M	Main Turbine Rotor Expansion
TJR 2508N	High Pressure Turbine Governor Valve 1 Inner Surface

TJR 2508P	High Pressure Turbine Governor Valve 1 Inner Surface
TR-RC-7	Reactor Coolant Unit Tave
TRS-RC-3	Reactor Coolant Loop 1 & 2 Hotleg Temperature
UJR 5023	Unit Vent Effluent Recorder
UJR 8443	Area Radiation Monitor Recorder #4
ZJR 2538	Turbine-Generator Vibration and Eccentricity

ms d/3

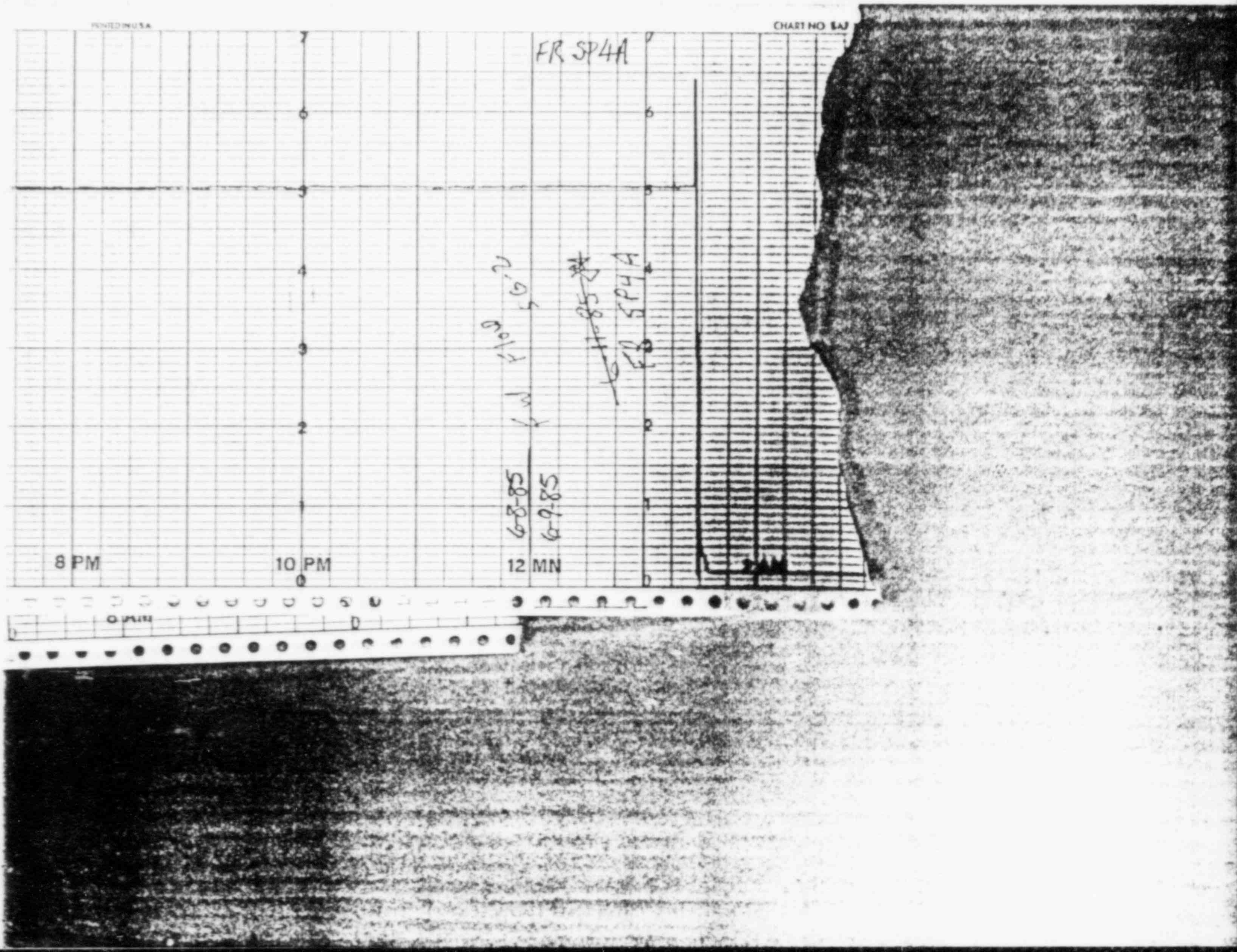


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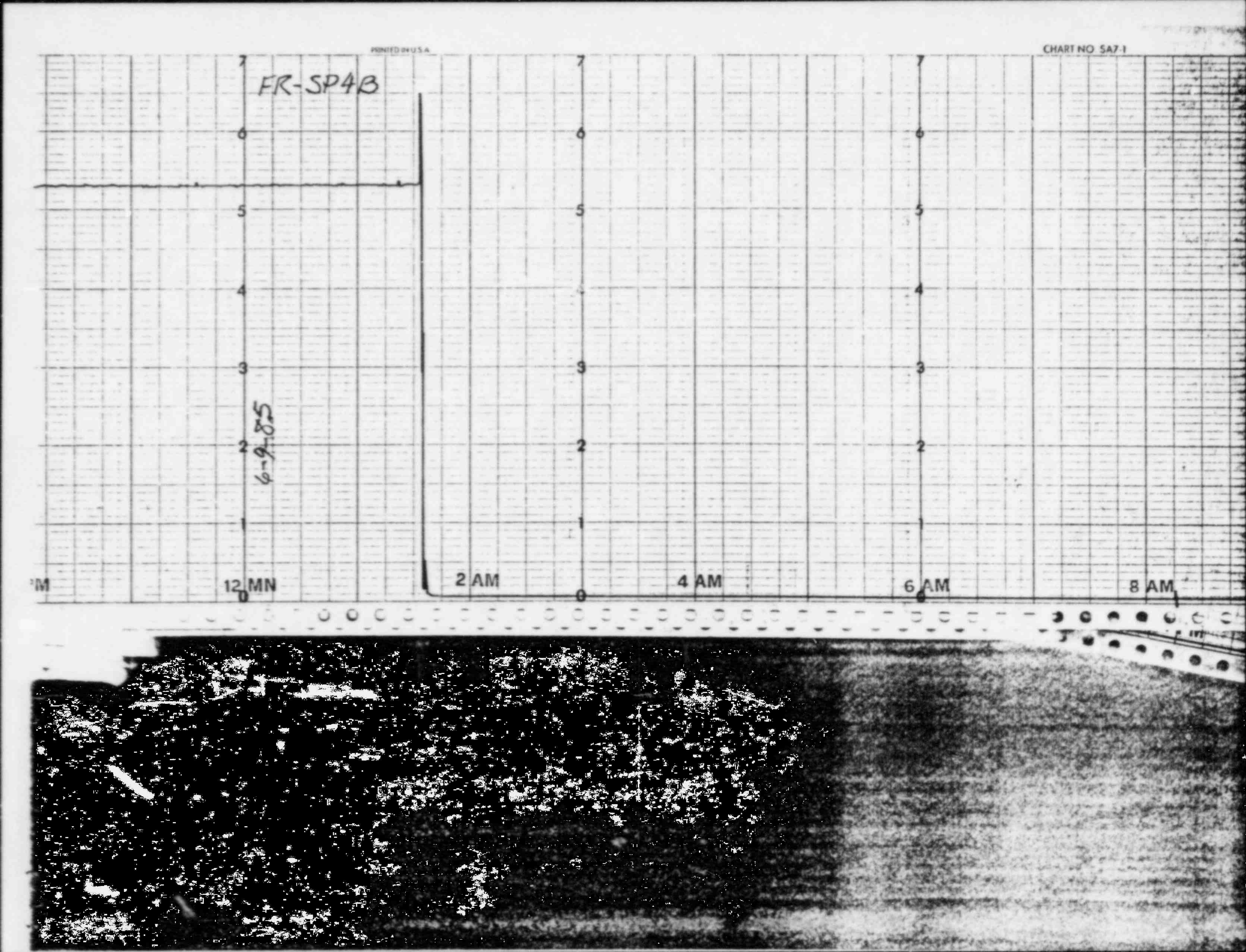
CHART NO
FR SP4A
6-8-85
6-9-85
FB SP4A
8 PM
10 PM
12 MN
8 AM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4 AM
6 AM
8 AM
7
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5
4
3
2
1
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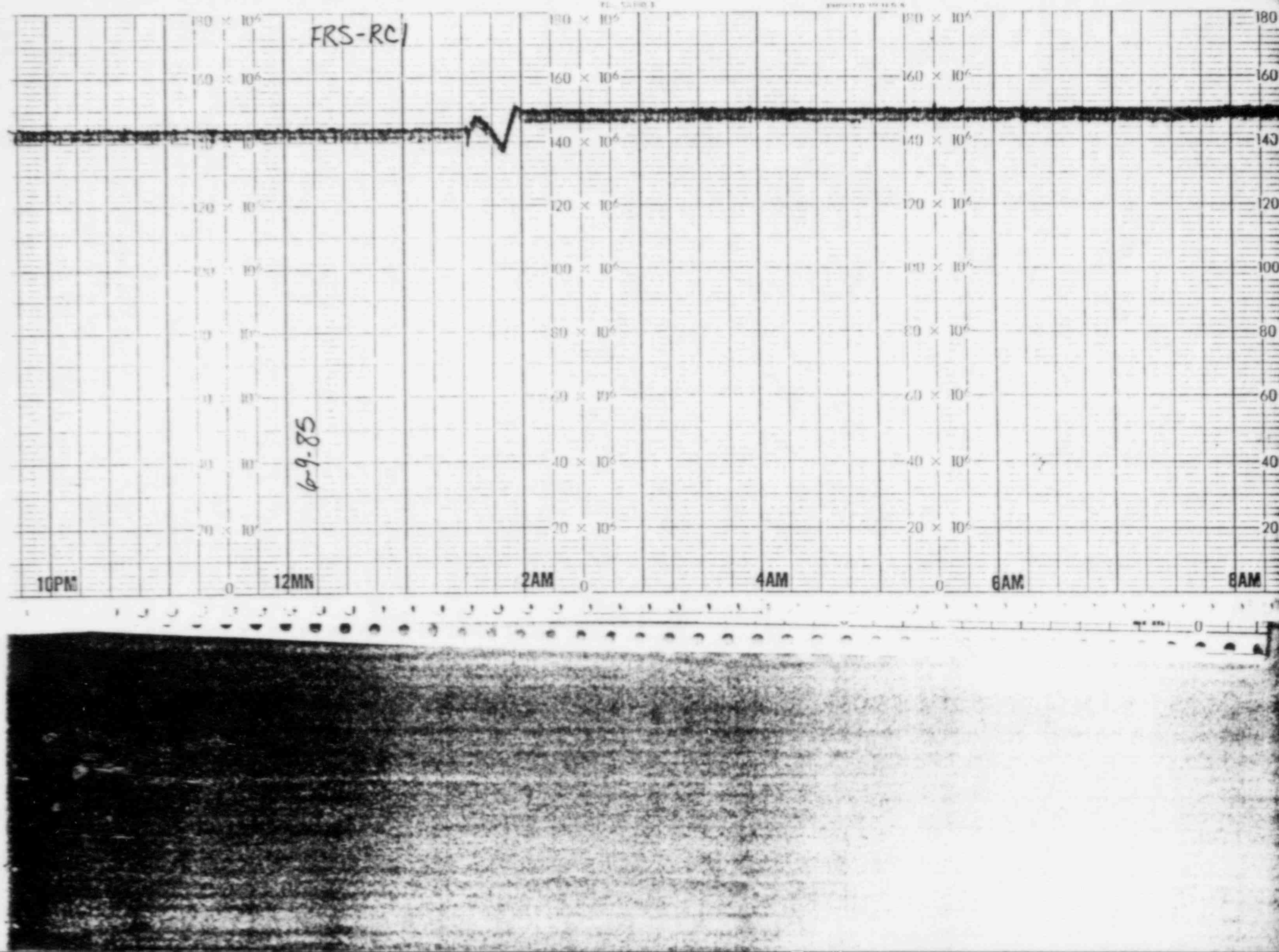
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40 × 10⁶
20 × 10⁶
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120
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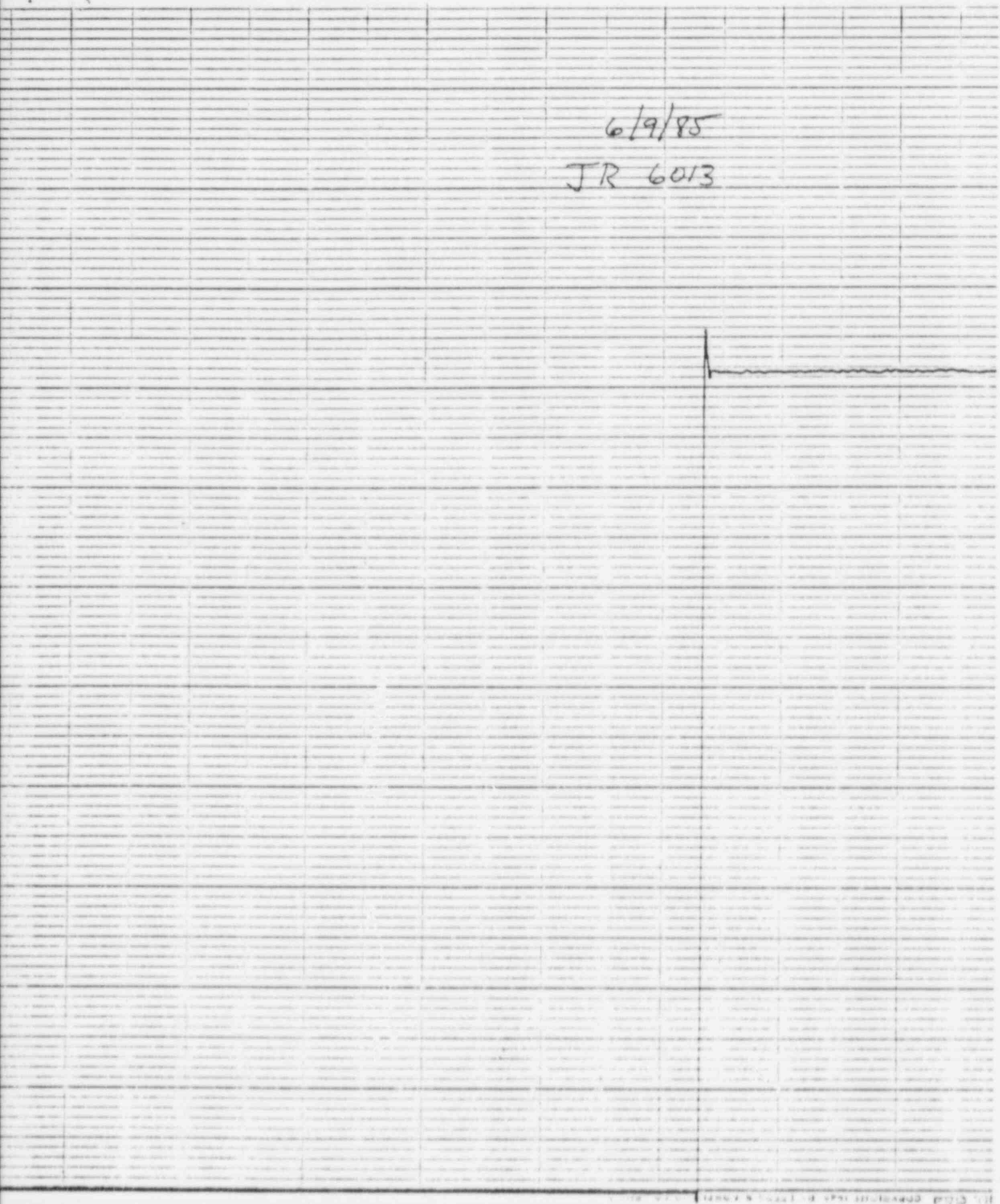
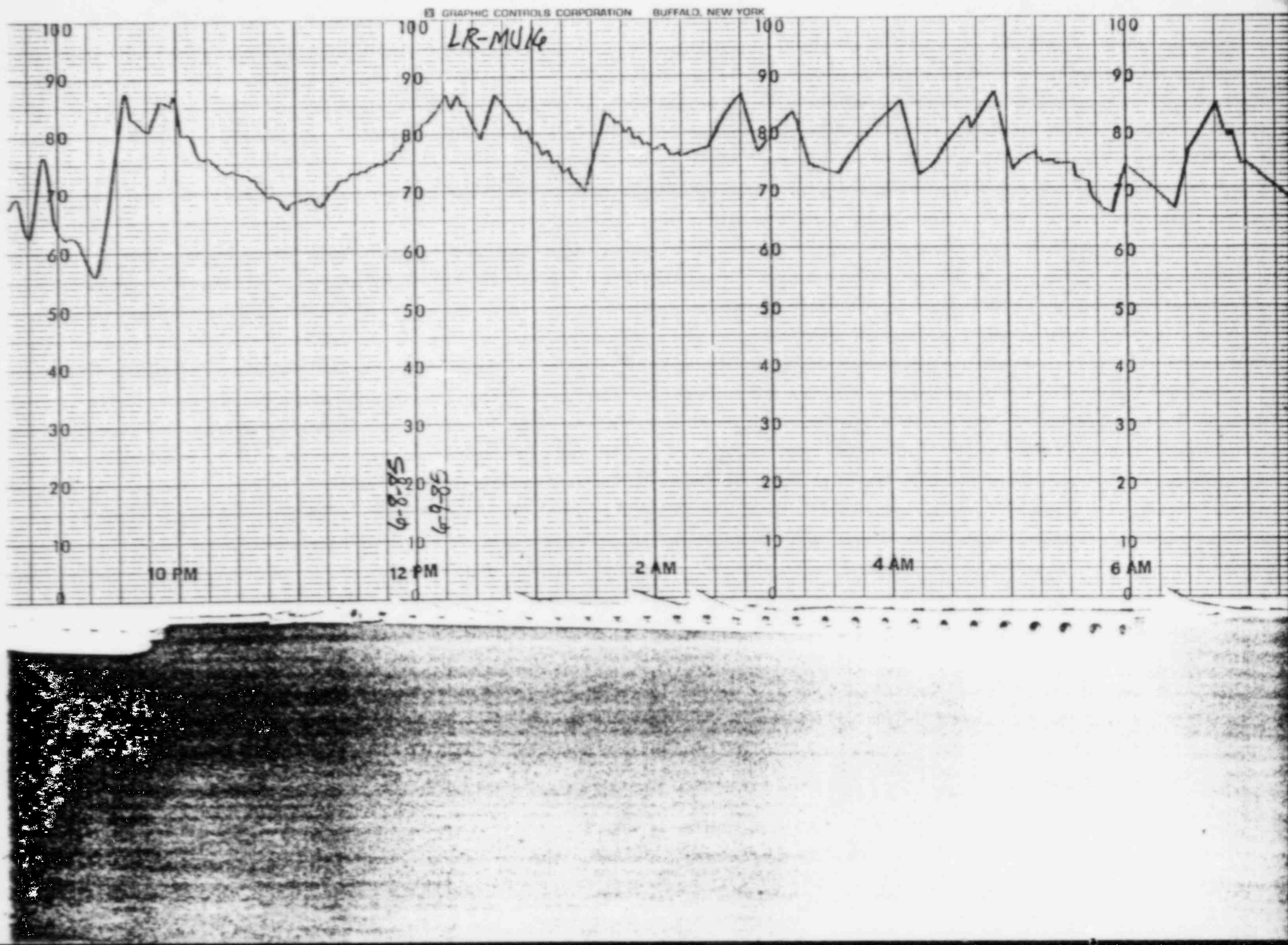
80
60
40
20
10PM
12MN
2AM
4AM
6AM
8AM


6/9/85
JR 6013


GRAPHIC CONTROLS CORPORATION BUFFALO, NEW YORK
LR-MUK6
6-8-85
6-9-85
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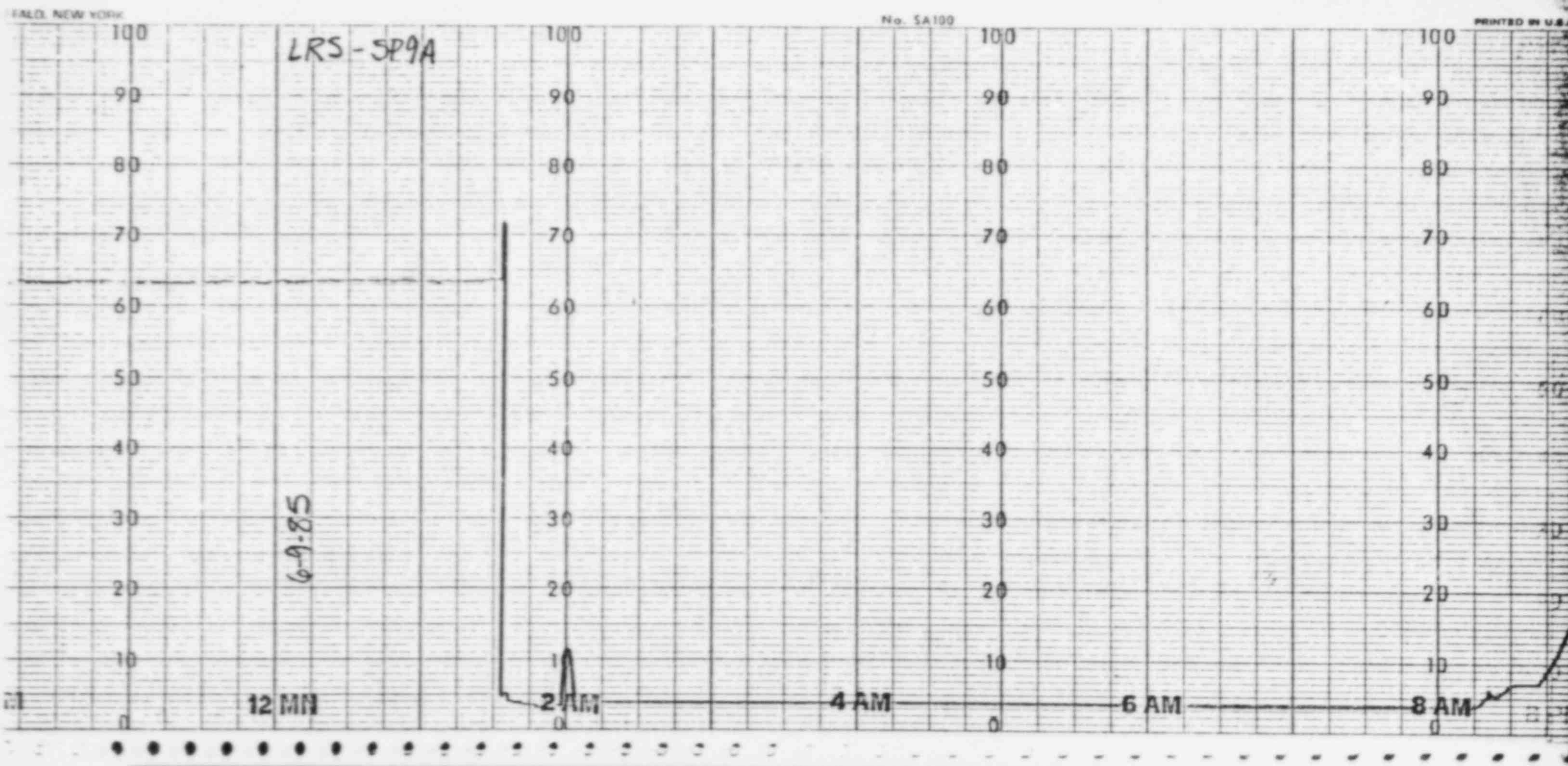
12 PM
2 AM
4 AM
6 AM
100
90
80
70
60
50
40
30
20
10
0



FALO, NEW YORK
No. SA100
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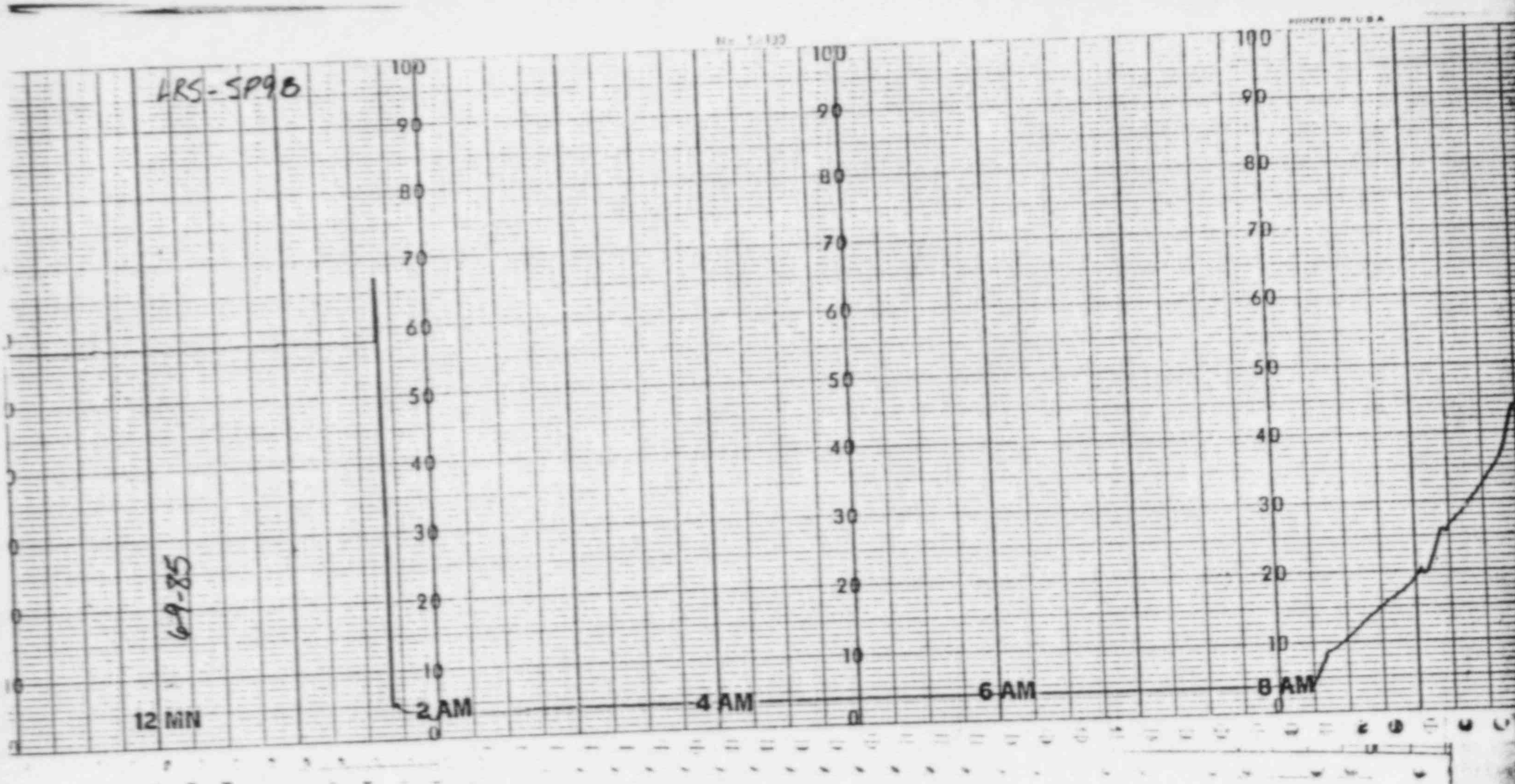
LRS - SP9A
6-9-85
12 MN
2 AM
4 AM
6 AM
8 AM



ARS-SP98
6-9-85
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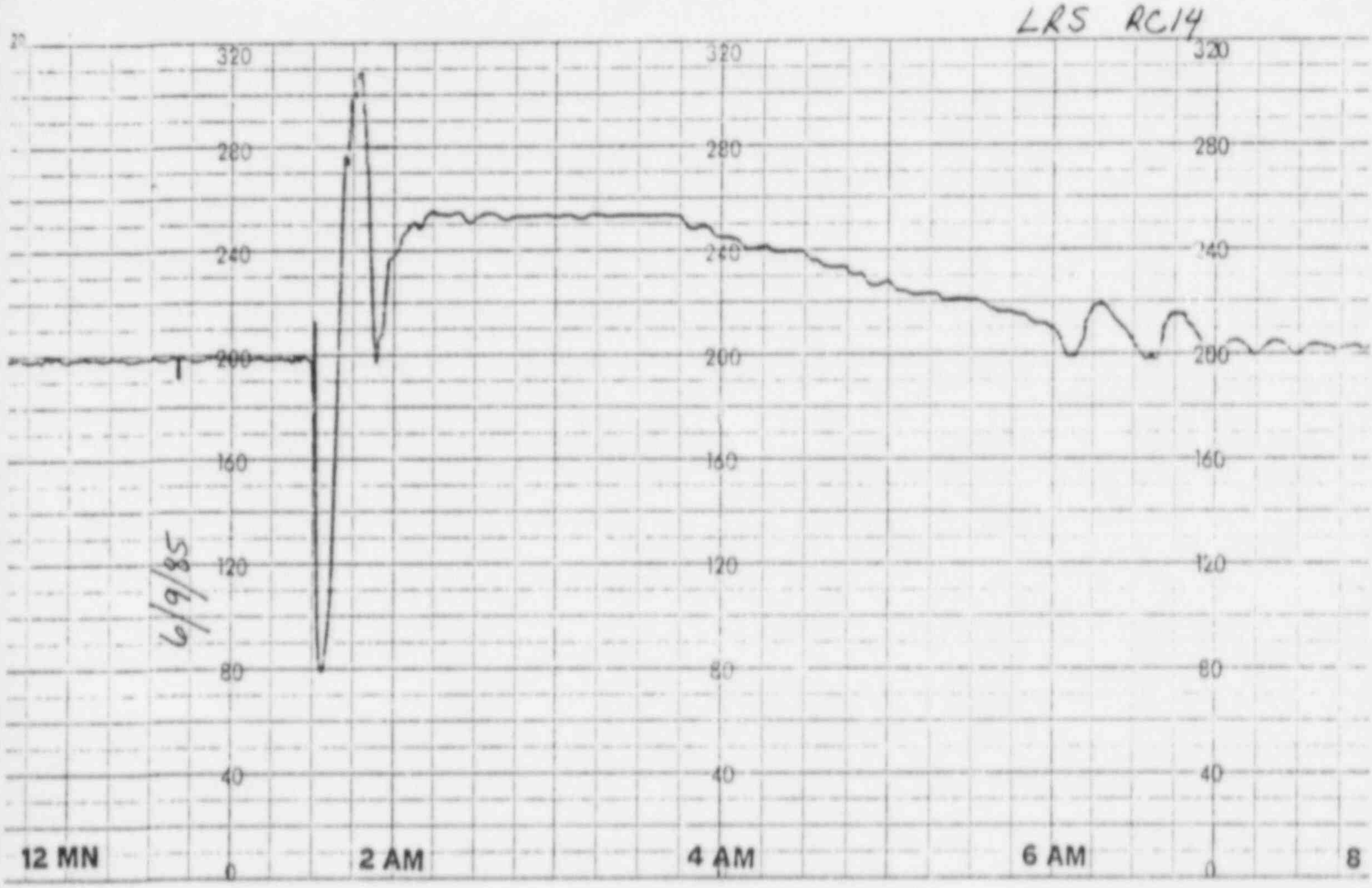
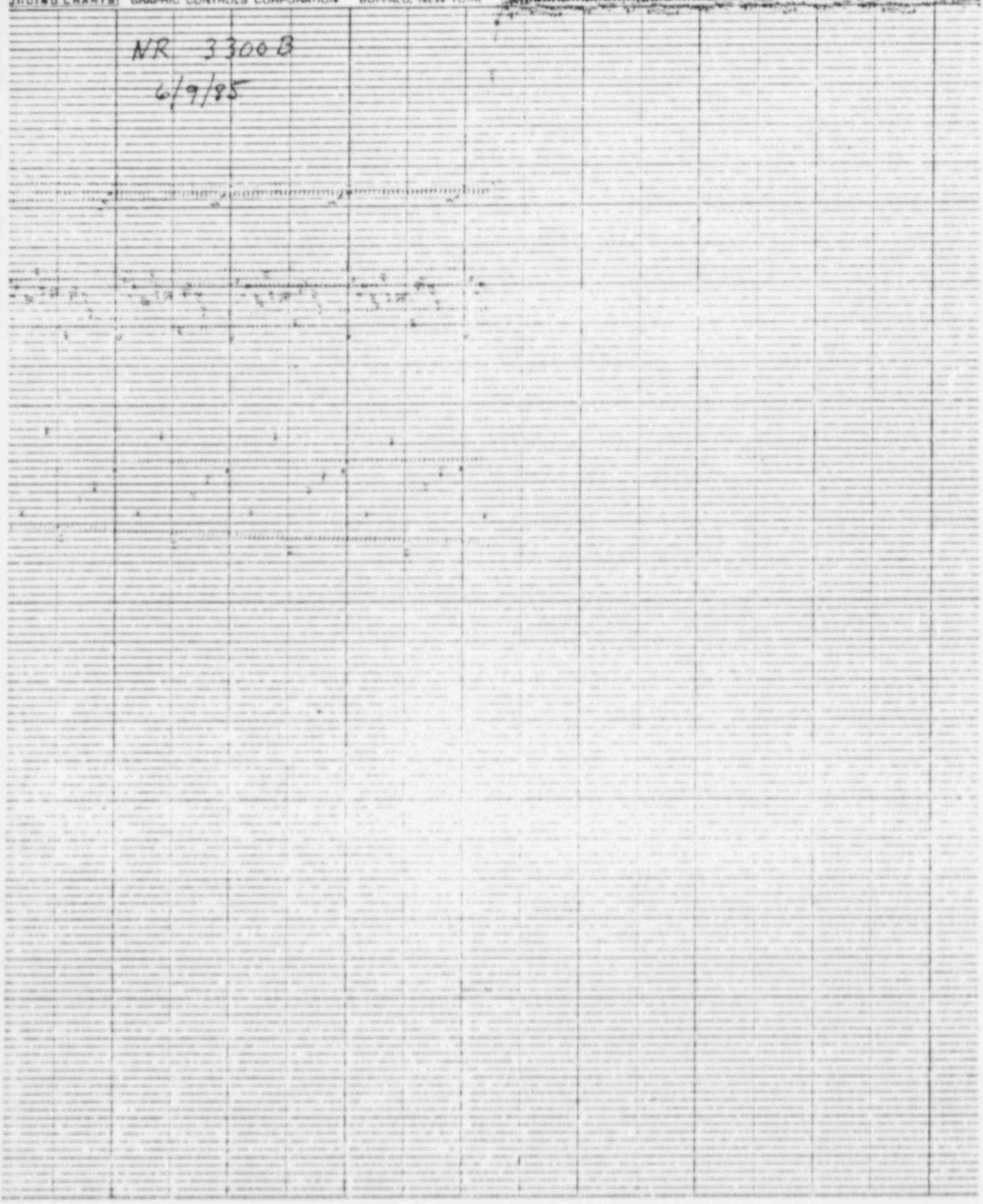
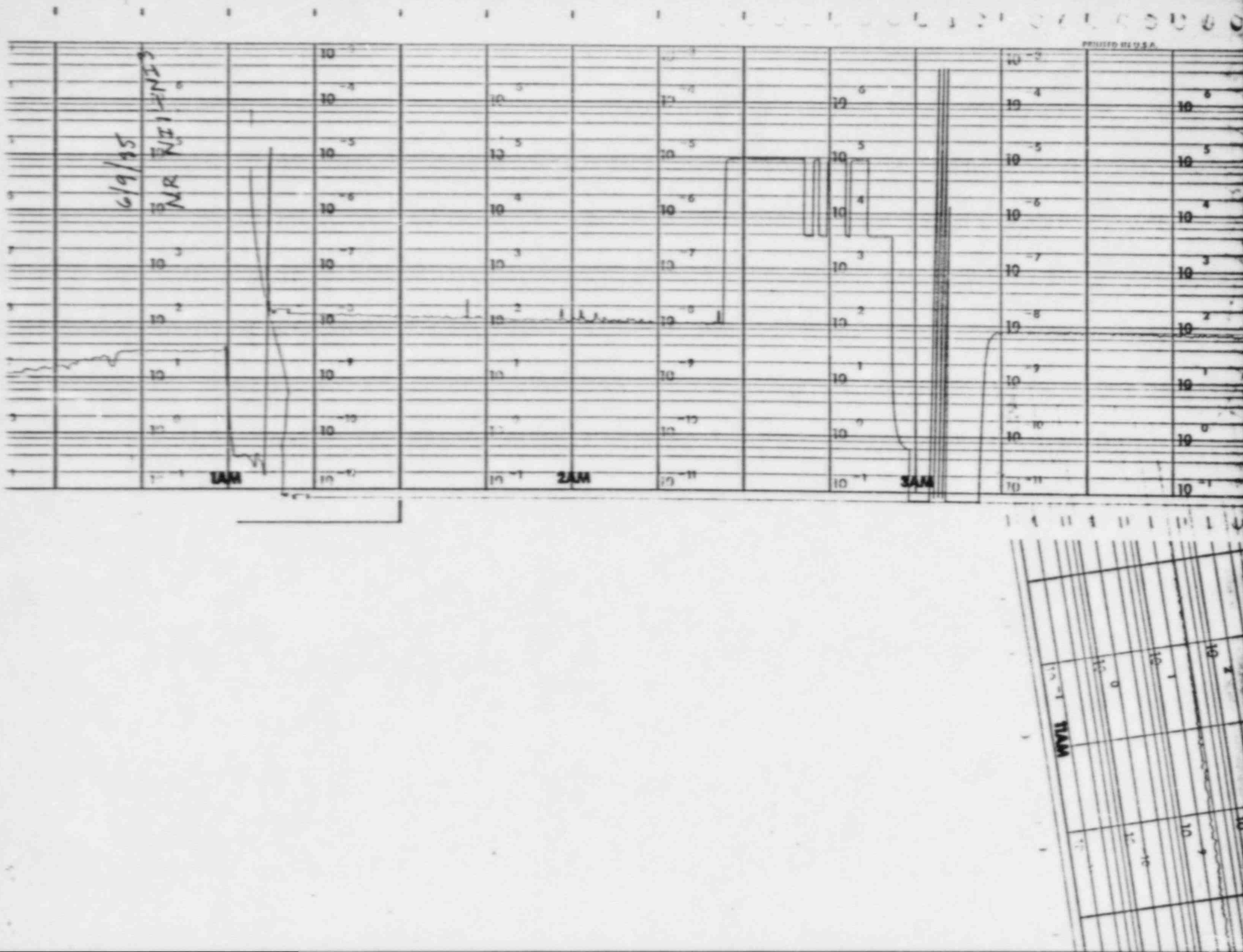
2 AM
4 AM
6 AM
8 AM
PRINTED IN U.S.A.

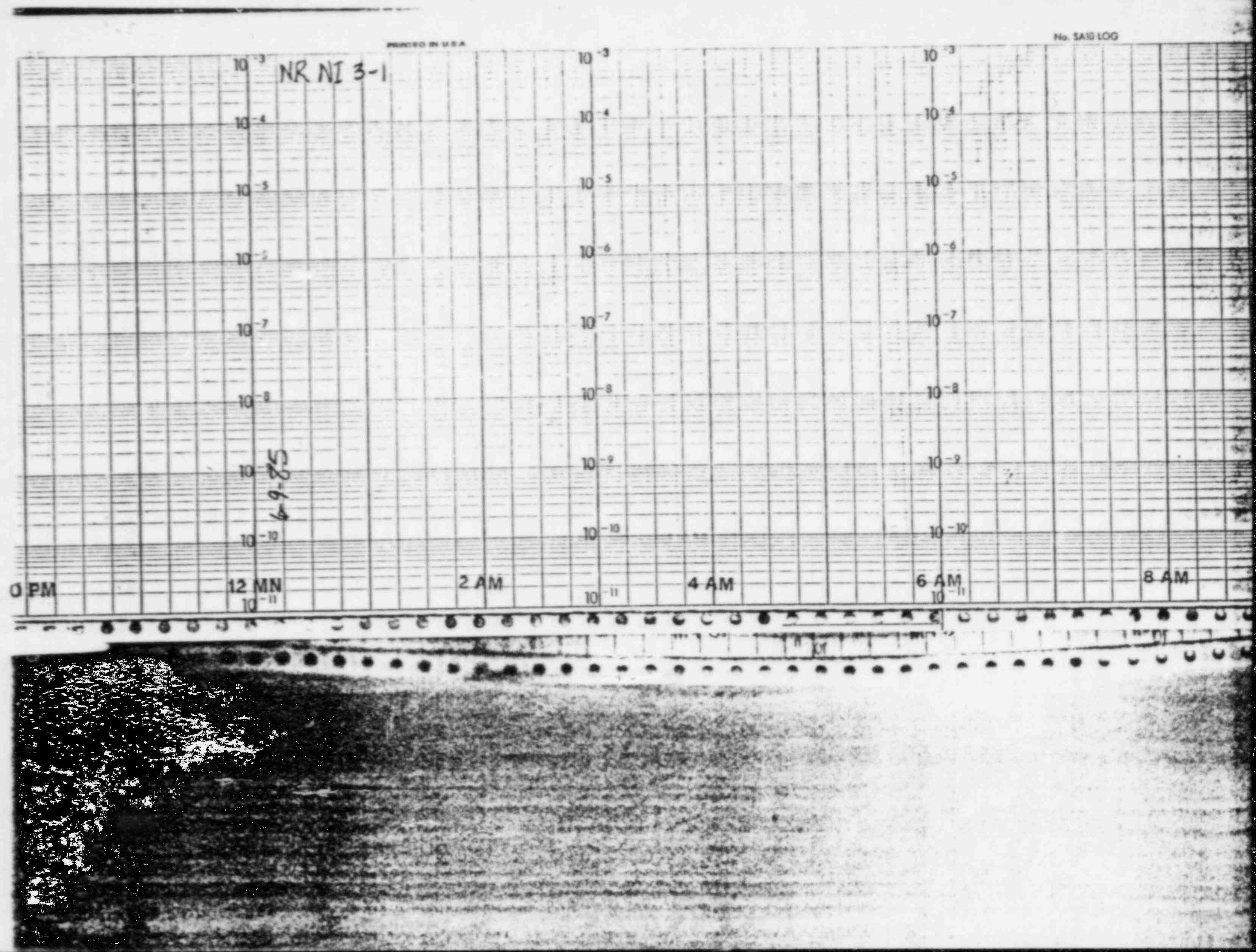

LRS RC14
6/9/85
320
280
240
200
160
120
80
40
0
12 MN
2 AM
4 AM
6 AM
8

RECORDING CHARTS GRAPHIC CONTROLS CORPORATION BUFFALO, NEW YORK




1AM
2AM
3AM
11AM
6/9/85
NR NI...

NR/ NI2 NI4 6/9/85


PRINTED IN U.S.A.
No. SA10 LOG
NR NI 3-1
6-9-85
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
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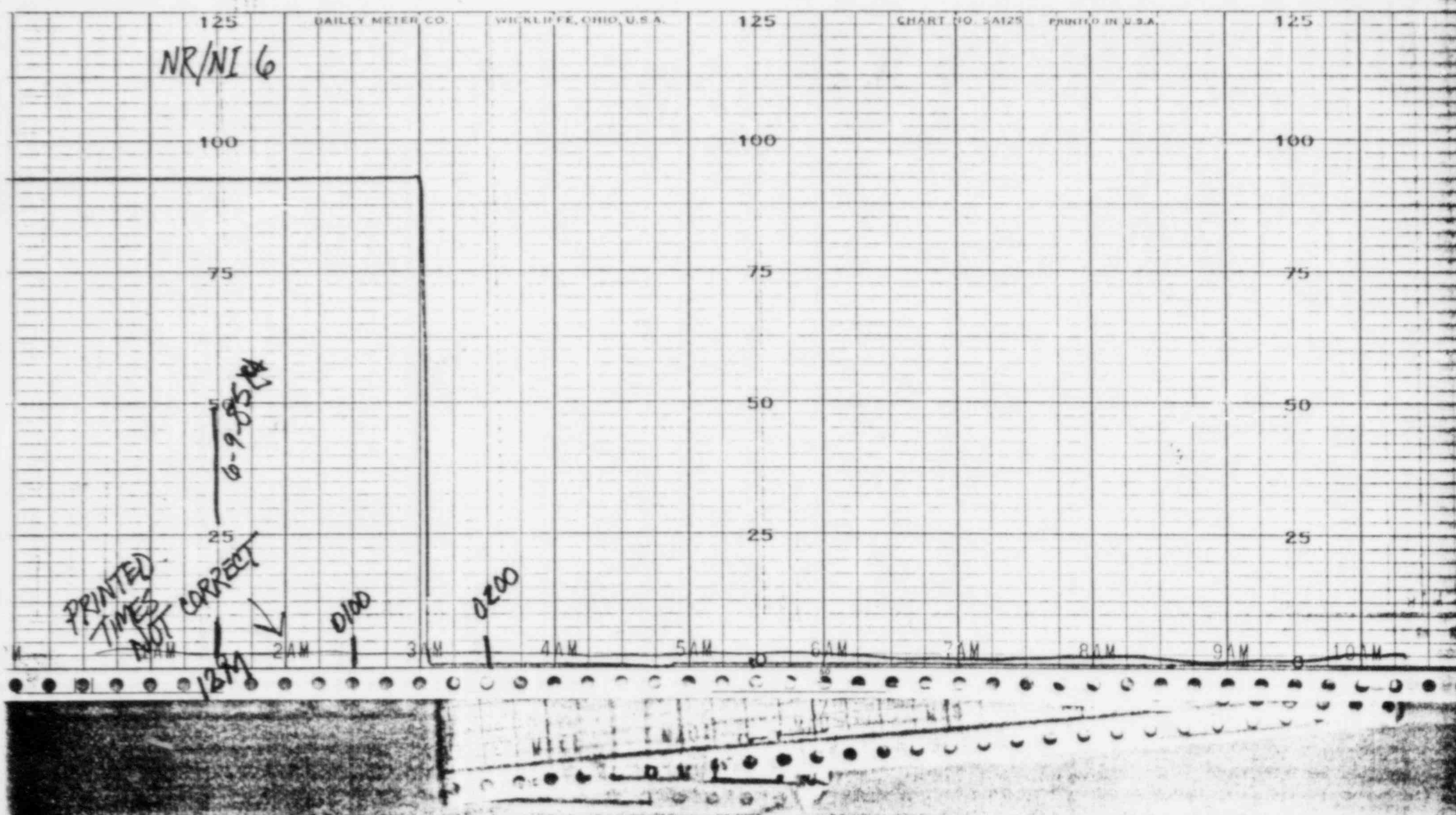
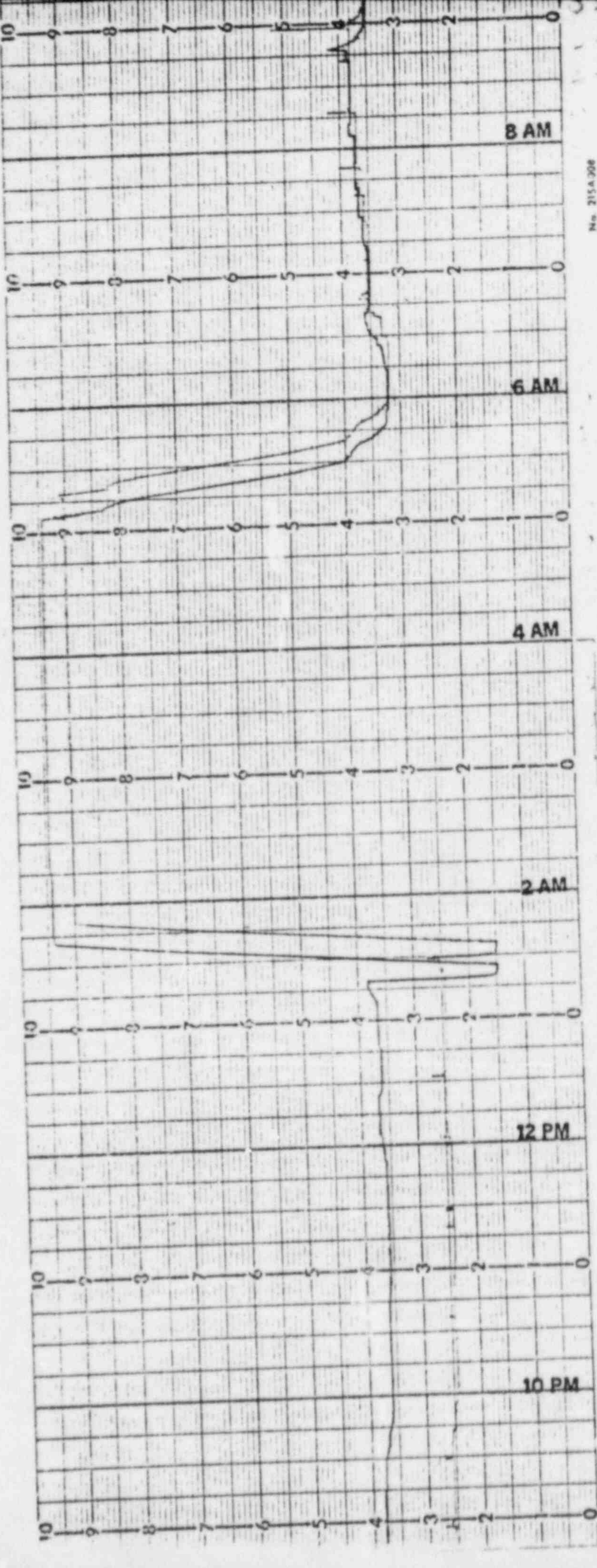
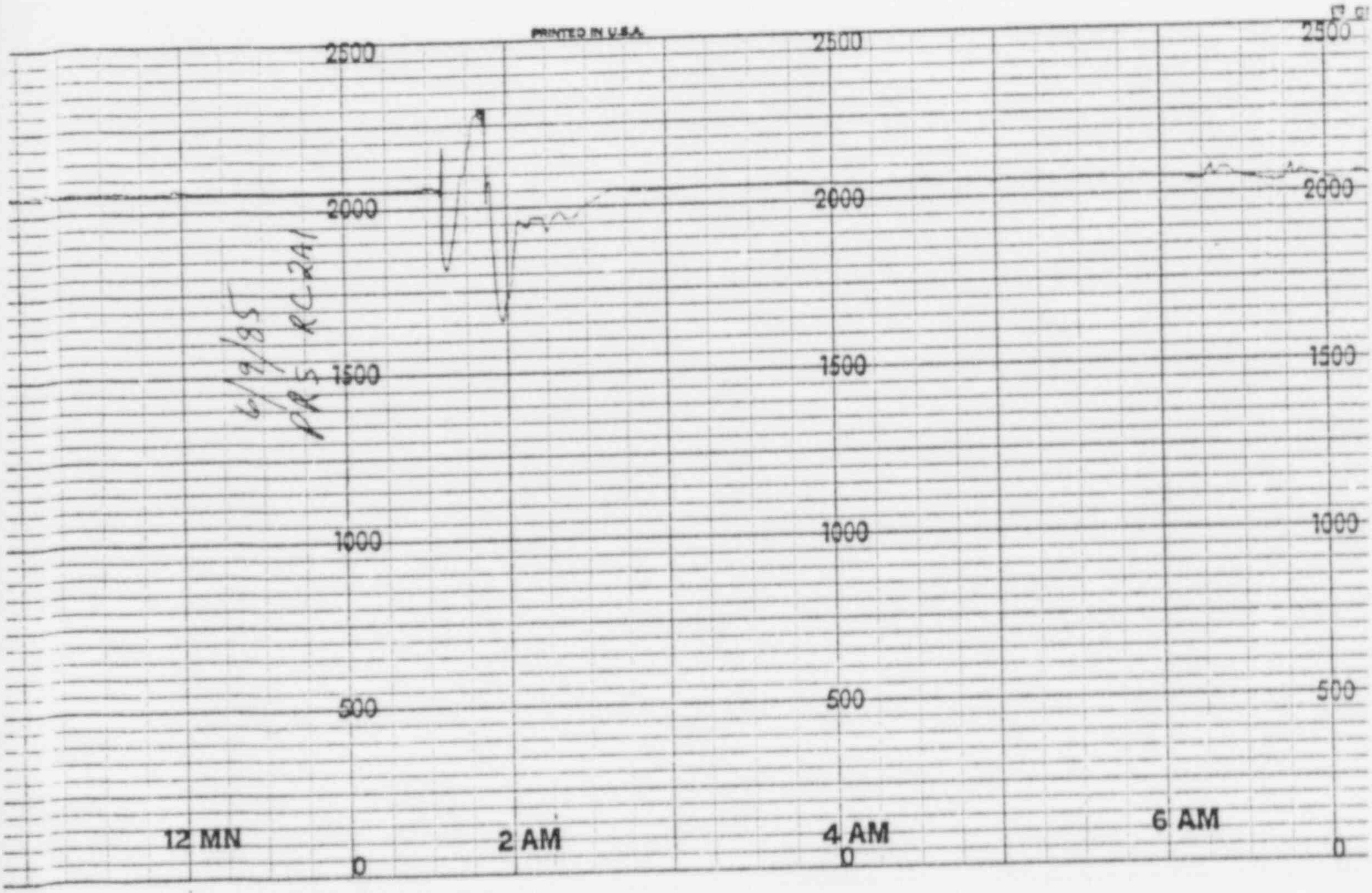
10^{-11}
0 PM
12 MN
2 AM
4 AM
6 AM
8 AM



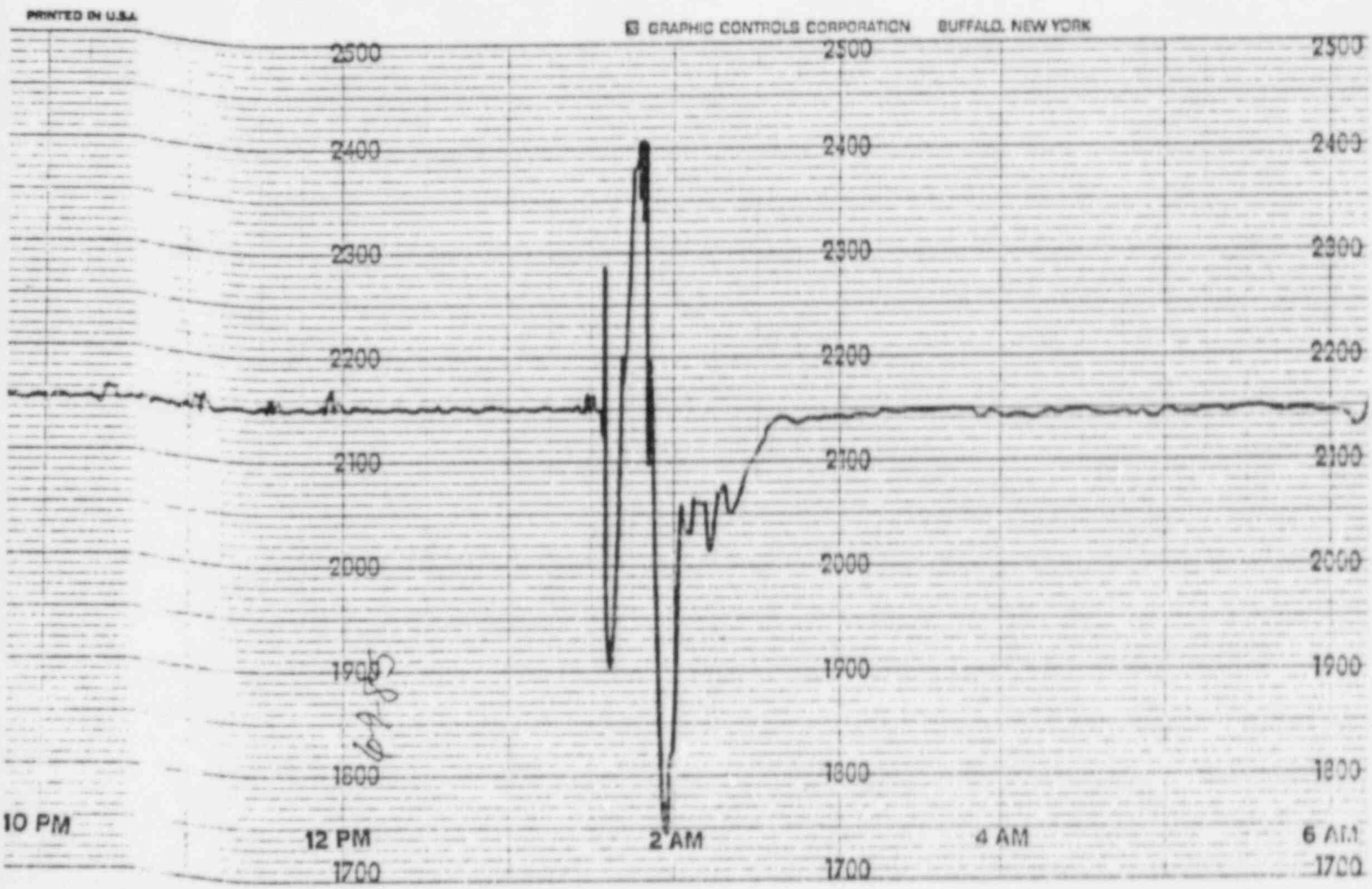
BAILEY METER CO.
WICKLIFFE, OHIO, U.S.A.
CHART NO. SA125
PRINTED IN U.S.A.
125
100
75
50
25
NR/NI 6
6-9-85
PRINTED TIMES NOT CORRECT
12M
0100
0200
2AM
3AM
4AM
5AM
6AM
7AM
8AM
9AM
10AM

PR 530/541
6/9/85




PRINTED IN U.S.A.
2500
2000
1500
1000
500
0
12 MN
2 AM
4 AM
6 AM
6/9/85
PR5 RC2A1

PRS RC2A2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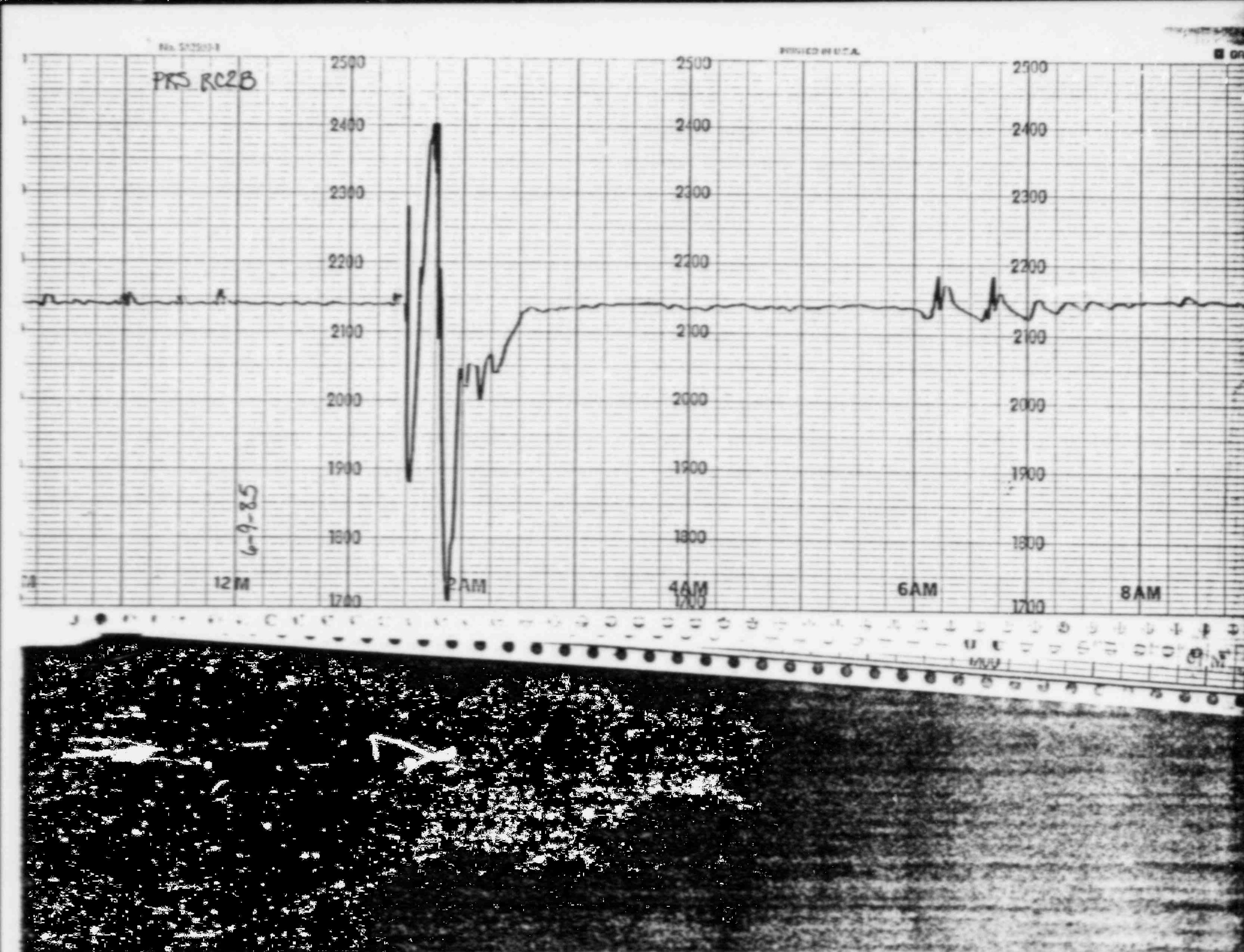
GRAPHIC CONTROLS CORPORATION BUFFALO, NEW YORK
2500
2400
2300
2200
2100
2000
1900
1800
1700
10 PM
12 PM
2 AM
4 AM
6 AM


PRS RC2B
6-9-85
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2100
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1700
12M
2AM
4AM
6AM
8AM
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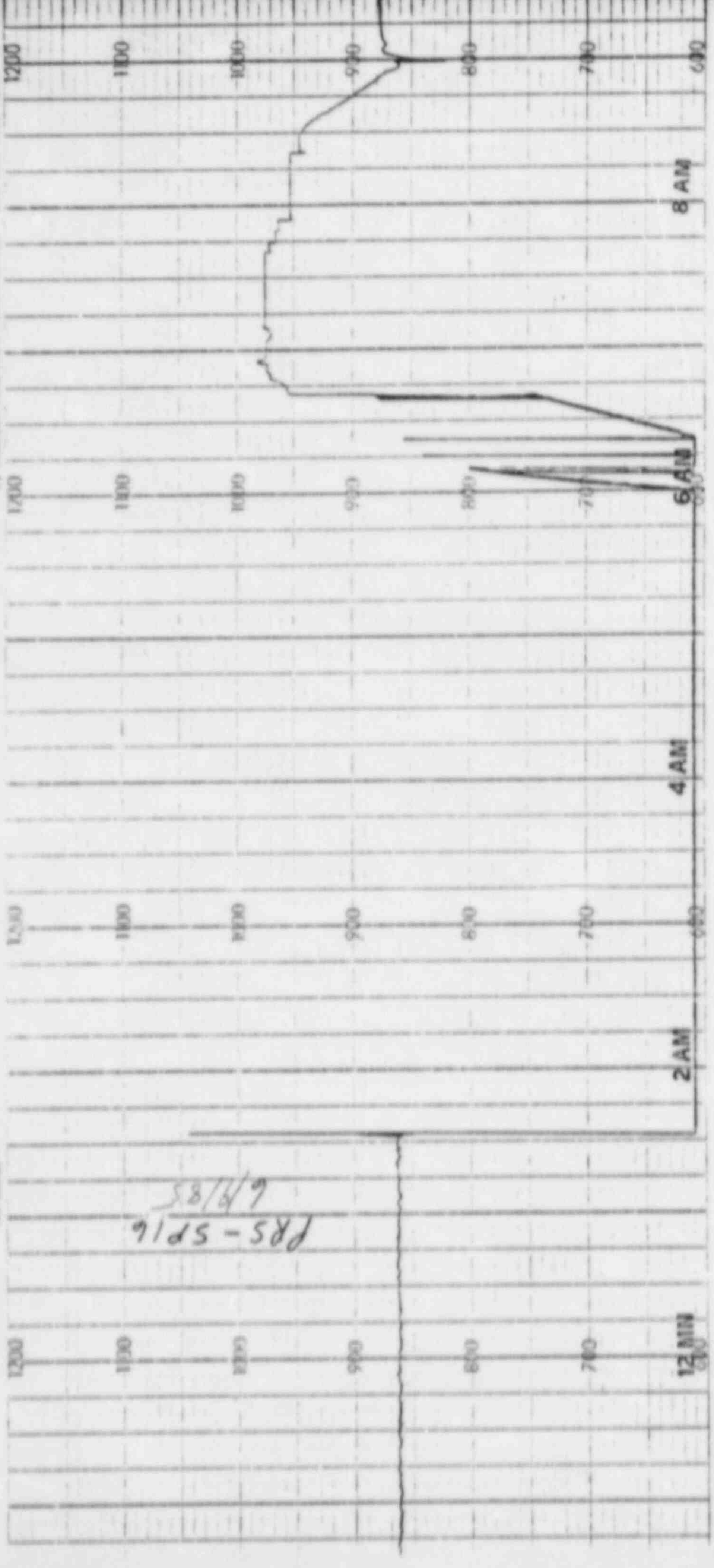
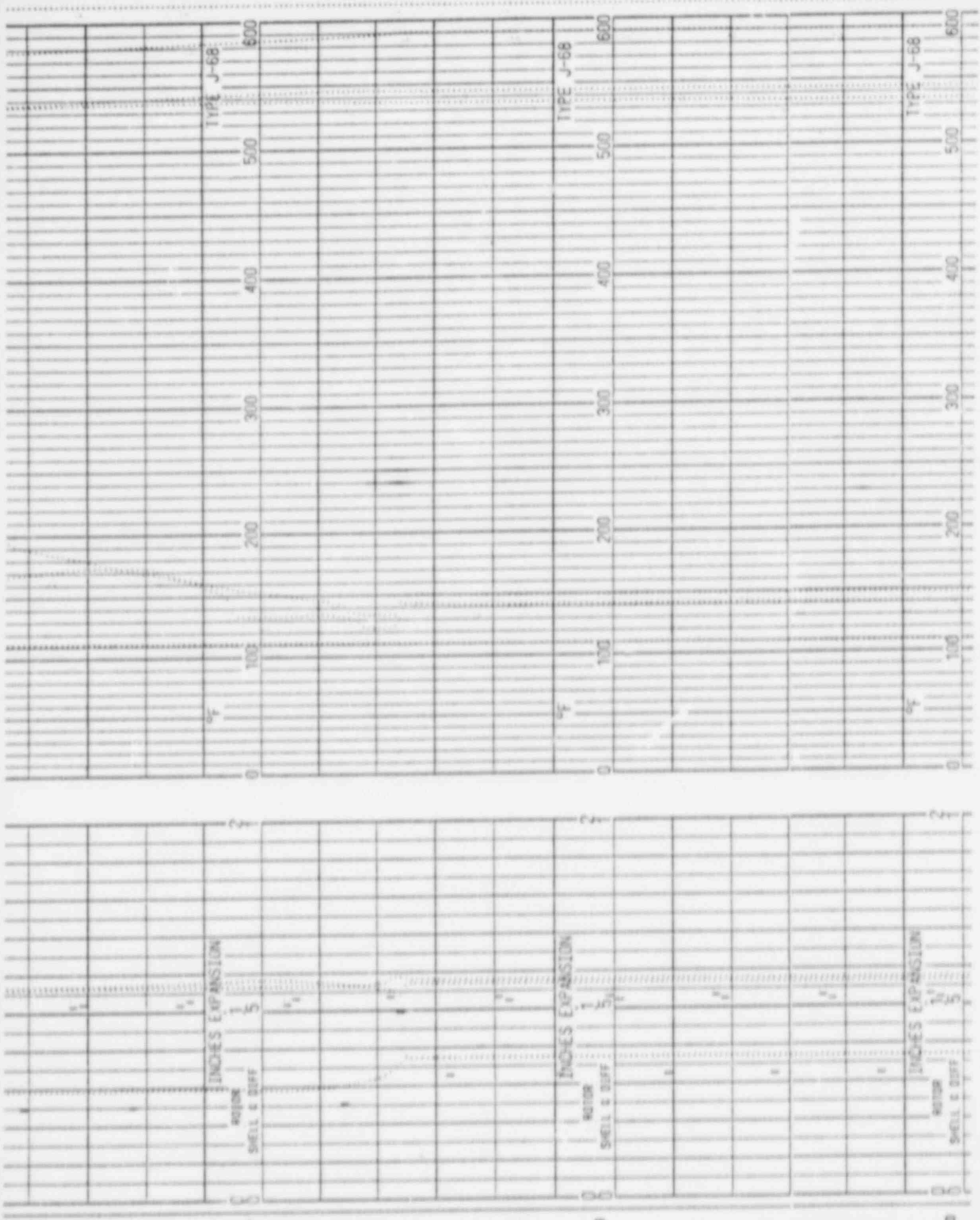
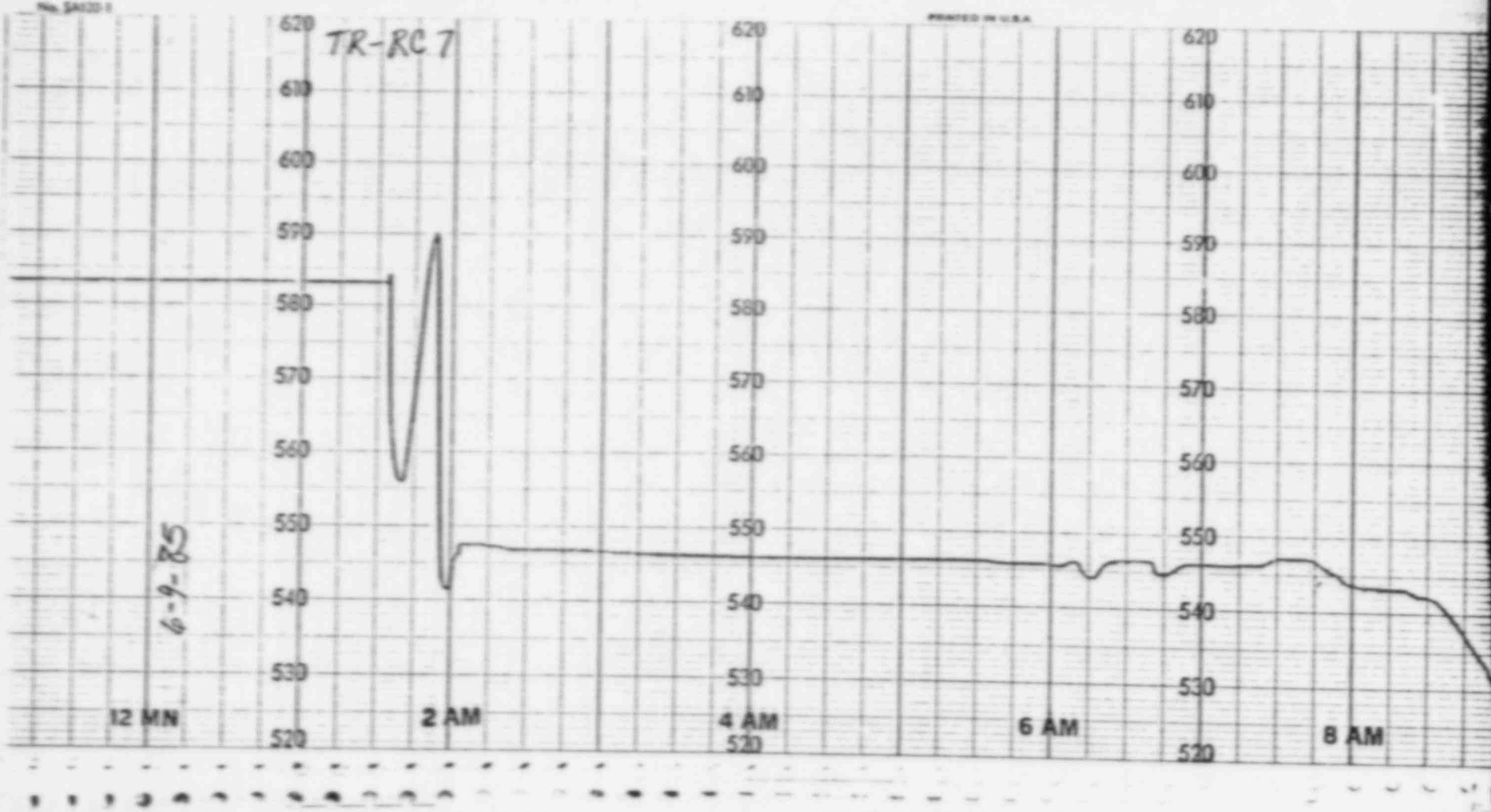


12 MN
2 AM
4 AM
6 AM
8 AM
600
700
800
900
1000
1100
1200
PRS-5P16
6/?/85



TSR-2508 6/9/85


TR-RC 7
6-9-85
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610
600
590
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520
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2 AM
4 AM
6 AM
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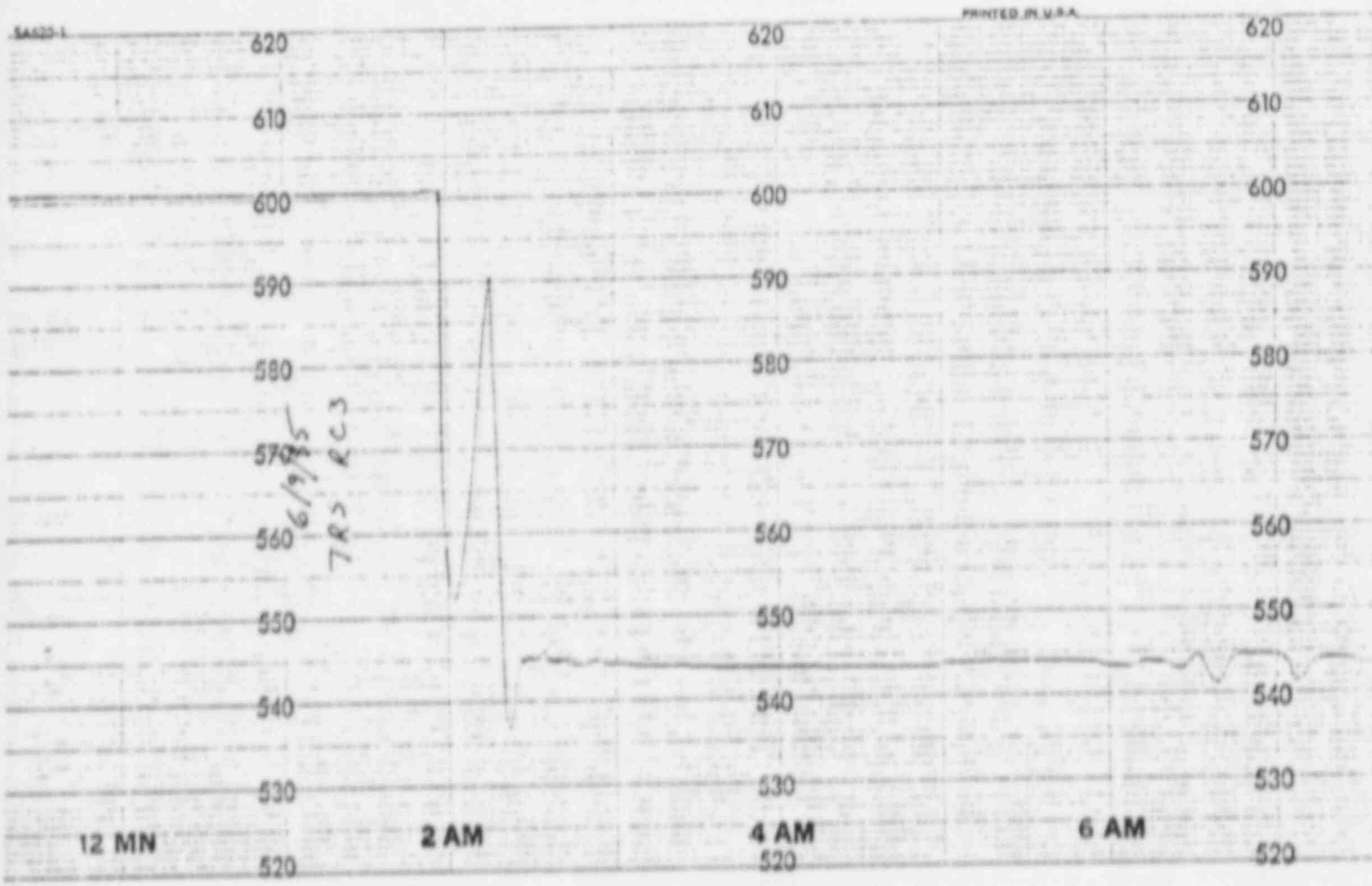
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12 MN
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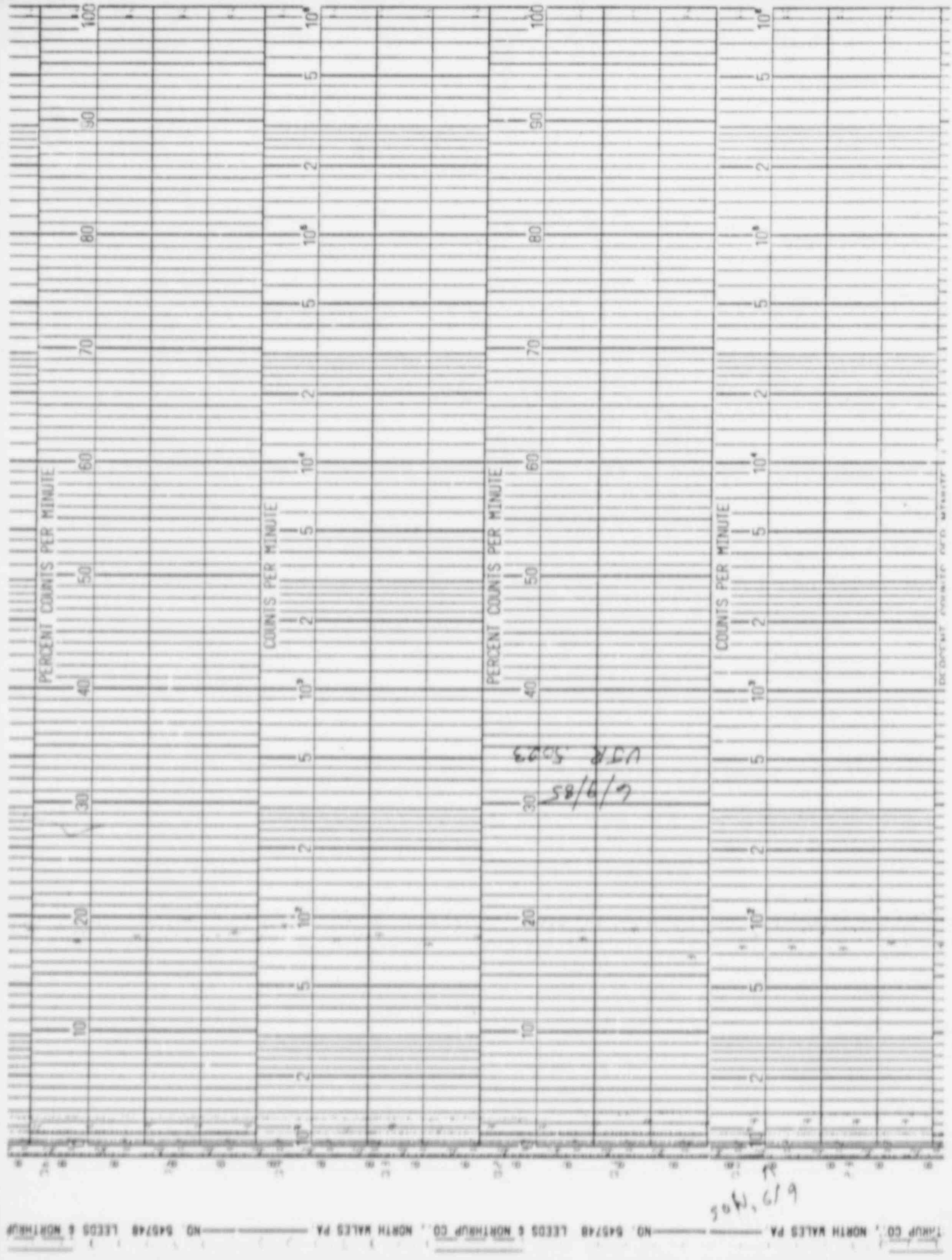
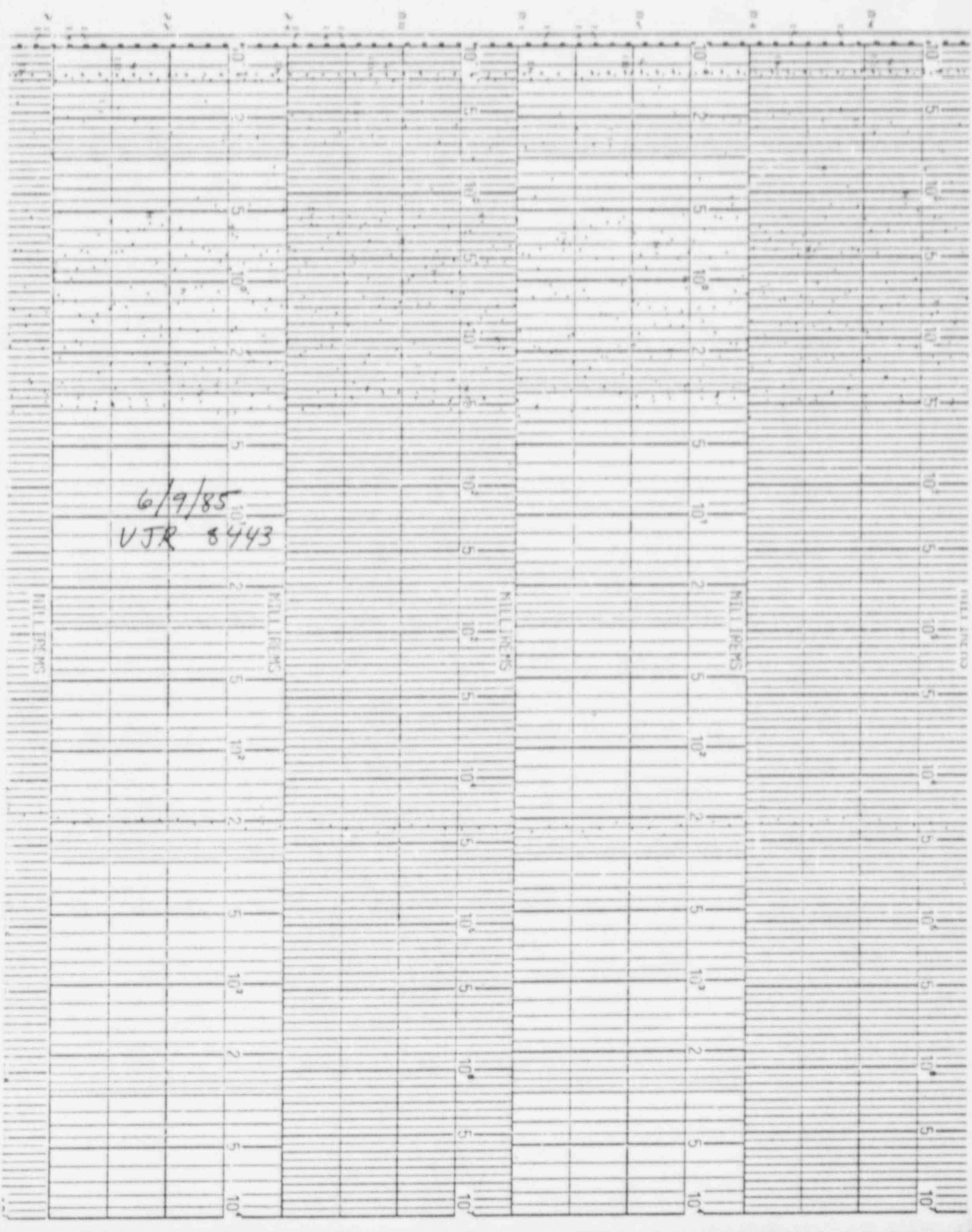
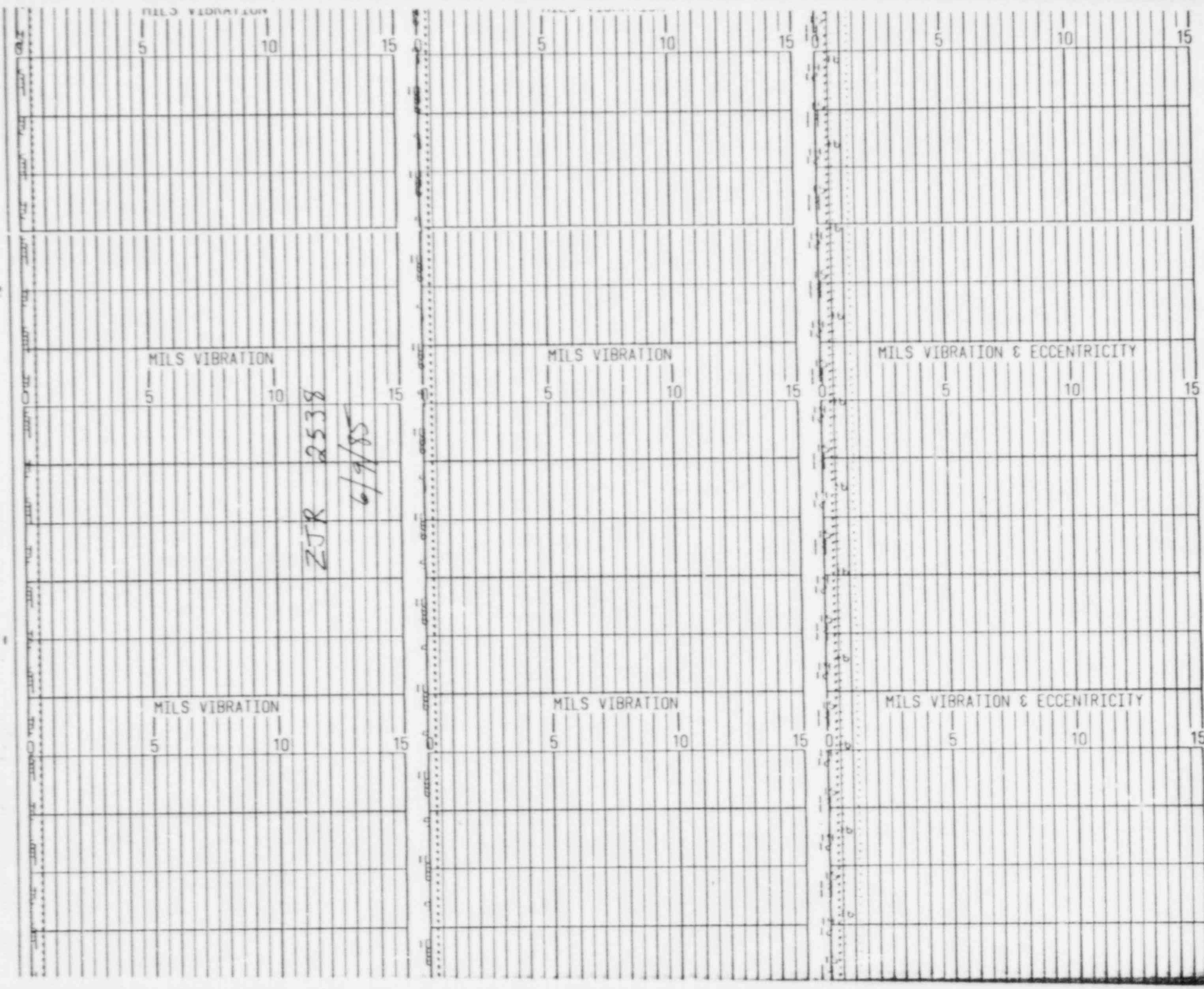


PERCENT COUNTS PER MINUTE
COUNTS PER MINUTE
PERCENT COUNTS PER MINUTE
COUNTS PER MINUTE
6/9/85
VSR 5023
soil, 6/9
NO. 545748 LEEDS & NORTHRUP CO., NORTH WALES PA

NO. 545747 LEEDS & NORTHRUP CO., NORTH WALES PA

NO. 545747 LEEDS & NORTHRUP CO., NORTH WALES PA




MILS VIBRATION
MILS VIBRATION
MILS VIBRATION & ECCENTRICITY
MILS VIBRATION
MILS VIBRATION
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ZJR 2538
6/9/85