SITE PROBLEM

REPORT TRANSMITTAL

THREE MILE ISLAND-1

**** CLEARED ****

FILE: 13-5-322

TO: _____ For Information

Central Engineering Files

C. C. Plunkett - Contract Admin.

S. H. Klein - Quality Assurance

R.J. Shepherd - Task Engineer

R.A. Gayers - Project Manager

CONTRACT NO: 620-00 05

SPR 322 REV. 1

TITLE Low Press. Level Following Reactor Trip

DATE: 6-25-76

The attached, cleared SPR is submitted for your information.

TO:

_____ E. L. Logan - FLORIDA _____

_____ L. C. Rogers - MET.ED. _____

_____ R. J. Baker - TOLEDO _____

_____ B. I. Day - Intl. Support _____

_____ P. E. Perrone - OFR _____

_____ J. L. Donnell - OFR _____

L.M. Kolonay

Attached is one copy of Site Problem Report No. 322 which was processed on Contract 620-00 05. Future contracts have been reviewed for the potential of a similar problem. This problem ~~is~~/is not considered applicable to other contracts oc.

REMARKS: _____

cc: G. M. Jacks - Plant Integration

This SPR has been reviewed IAW NPG-1707-01

Chris C. Lockard

NUCLEAR SERVICE SUPPORT ENGINEER

CLEARED

7910040 530

SITE PROBLEM REPORT — BABCOCK & WILCOX

CUSTOMER MET ED		CONTRACT NO. 620-0003	SPR NO. 322	REV. NO. 1
VENDOR BMCo	P.O. NO.	TASK NO. 21	GROUP NO. 01	SEQ. NO. 01
SITE ENGINEER S. P. MAINGI		REQ'D. RESOL. DATE	REQ'D. COMP. DATE	

TITLE
LOW PRESSURIZER LEVEL FOLLOWING REACTOR TRIP

DESCRIPTION OF PROBLEM

SEE ATTACHED SHEET

STATUS - ACTION TO DATE INCLUDING PERSONS CONTACTED

L. M. Kolony of Engineering informed.

FURTHER ACTION RECOMMENDED BY SITE PERSONNEL

Review and implement Logic modification such that reactor trip should block calibrating integral RC 9.12, to achieve a slower cooldown rate of RC System and a more acceptable pressurizer level.

This problem should be reviewed for all other sites.

SITE ENGINEER	DATE	SIGNATURE	DATE
S P Maingi	10/14/74	L C Rogers	10/14/74

RESOLUTION

SEE COMPLETION REPORT

APPROVED BY	SIGNATURE	DATE
N.S. SUPPORT ENGINEER	Chris C. Lockard	6-23-76
TASK ENGINEER/NS UNIT MGR	L. Kolony / Kindalt	6/23/76
OP. PLANT MGR.	[illegible]	6/24/76
PROJECT MANAGER/SERVICE MGR.	R Howers	6/23/76

COST CATEGORY ☐ NORM ☐ C ☐ D ☐ C ☐ ☐ VENDOR CLAIM

AUTH. CHARGE NO ☐ FIELD CHANGE REQ FC NO.

SITE COMPLETION REPORT

FURTHER INVESTIGATION SHOWED THAT OPERATOR PUT FW CONTROL IN MANUAL. THE PROBLEM RESULTED FROM OPERATOR ACTION INSTEAD OF AUTOMATIC ICS ACTION.
RECOMMEND SPR BE CLOSED. (R. [illegible] (Kohn)) BMCo 6/23/76

☐ RECOMMENDED STDS. CHANGE

DEVIATIONS ☐ NONE ☐ SEE SPR/REV. NO. ______

DATE COMPLETED | SIGNED BY

S.O.M. CONSTR. REP. APPROVAL R Howers DATE 6/23/76

FINAL DISTRIBUTION
PROJECT MANAGER
S.O.M./CONST. REP.
QA DOC FILE
CENT. ENGR
FILE 12M.2

ATTACHMENT
SPR 322 Rev. 1

DESCRIPTION OF PROBLEM:

Following a reactor trip the pressurizer level goes as low as 40 inches. SPR 322 Rev. 0 pointed out the main reason that pressurizer level goes as low, because the trapped steam pressure in turbine header decays slowly, resulting in turbine bypass remaining open for time longer than is necessary.

Further examination of the reactor trip data revealed that immediately following the reactor trip the feedwater demand actually reduced to 20% instead of reducing to 5% (cross limit value as neutron power following the trip is zero).

This additional feed to steam generators contributes to the excessive cooling of the RC syrtem and hence lower pressurizer level.

See attached BWCo report on the subject.

PRODUCT SYSTEM / FILE NO. / CMD USE ONLY

DATE OF PROBLEM: 9/4/74
UNIT NO. OF PROBLEM: 1501
UNIT SERIAL NO.: 290523
NOMENCLATURE: X
DATE CODE: X

FROM: R. S. Hand
EMP NO: 81239
ACCT NO: 779
MAIL STA NO: FO for 352
Middletown PA 17057
SERIAL NO. AND/OR PART NO.: X

PRT
ANY ONLY) AND TYPE OF SYSTEM/PRODUCT

FIELD ITR INTERVENTION

References: My FSR of 9/23/74

On 9/4/74 Nat. Ed. Plant Superintendent , Jack Herbein stated that "On a Reactor Trip the pressurizer level should not drop as lov as it does. (to approximately 40 inches) Analysis of the 8/13/74 Generator breaker trip test reveals the cross limits from neutron power error to feedwater flow control did Not perform as expected. The neutron power error cross limit should have reduced the feedwater demand immediately after reactor trip to approximately 5%. The neutron power error cross limit actually reduced the feedwater flow demand to approximately 20%. This additional feed to the steam generators contributed to the excessive rate of cooling of the RC system and the resultant drop in pressurizer level.

The reason for the above undesirable performance is as follows. On a reactor trip the CRD system transfers to manual. When the CRD is in manual the neutron power error is applied to the T_{ave} calibrating integral.

FOR FIELD USE
DAYS SERVICE: 8
SERVICE $: 8
EXPENSES $:
MATERIAL $: 8
SEO NO.: P72-236

ALLOCATION
- ☐ PRODUCT
- ☐ PRODUCT APPLICATION
- ☒ SYSTEM
- ☐ SYSTEM APPLICATION
- ☐ WARRANTY
- ☐ OTHER

CORRECTIVE ACTION RETURN TO
- ☒ NO
- ☐ YES ON _____ DATE
- RM NO

FAILURE OCCURRED
- ☐ ON RECEIPT
- ☐ IN SERVICE (WARRANTY)
- ☒ IN SERVICE (NON-WARRANTY)
- >2 YRS 1ST TIME IN SERVICE

POSSIBLE CAUSE FOR PROBLEM
- ☐ 50 FAULTY MATERIAL
- ☐ 60 FAULTY MFG/FABRICATING
- ☐ 70 FAULTY DESIGN
- ☐ 80 PERFORMANCE DEFICIENCY
- ☐ 04 FAULTY PACKAGING
- ☐ 05 COMPONENT FAILURE
- ☐ 06 WEAROUT
- ☐ 07 IMPROPER APPLICATION
- ☐ 08 OPERATING ENVIRONMENT
- ☐ 09 INSUFFICIENT INSTRUCTIONS (DOCUMENTATION)

CUSTOMER ATTITUDE:
- ☒ MAJOR CONCERN
- ☐ CONCERNED
- ☐ UNCONCERNED

EFFECT ON SYSTEM (BRIEFLY): Pressurizer level less than 50 inches cuts off heaters RC pressure can not be controlled.

DATE SOLUTION REQUIRED: 10/14/74 OR ☐ INFO. ONLY

SIGNATURE: [signature] DATE: 9/27/74

REPORT OF INVESTIGATION & CORRECTIVE ACTION (IF FIELD IF APPLICABLE)

OPERATING CONDITIONS: NA
AMBIENT TEMP: ____ °F
ATMOSPHERE: ☐ CLEAN ☐ AVERAGE ☐ DIRTY
HUMIDITY ☐ HI ☐ LO ☐ AVG.

TIME REQUIRED TO: NA
REPAIR ____ TROUBLESHOOT ____
RECALIBRATE ____

FAILURE DETAILS: NA
BMCO PART NO.
DESCRIBE (DIODE, CAP, TRANSISTOR, ETC.)
CIRCUIT SYMBOL (R1, R1, Q1)
MFG. OF PART (IF KNOWN)

HOW PART FAILED:
- ☐ SHORT ☐ OPEN
- ☐ MECH. DAMAGE
- ☐ ADJUSTMENT
- ☐ DIRTY ☐ UNKNOWN
- ☐ OTHER (DESCRIBE)

(QA USE ONLY)

PROBLEM TYPE	FAILURE CAUSE CODE	PRN SYSTEM	AGENCY DUE	ACTION TAKEN BY

COPIES:
- QUALITY ASSURANCE
- PROD. LIABILITY
- PROD. PLANNING
- NPO
- FPO
- CONTRACT OPER.
- WARRANTY REPAIR
- CORPORATION ENGR.
- FIELD SERVICES
- ORDER CENTER
- COMPONENT ENGR.

COPIES

☐ PRELIMINARY ANS
☐ FINAL SOLUTION

SIGNATURE DATE

FOLLOW-UP ON CORRECTIVE ACTION

QTY	NAME	PART NO.	COMMENTS

APPROVAL DATE

DISP. OF R.M.
DATE REC'D
DATE RETURNED

HOURS SPENT SOLUTION

DEPT	MAN	HRS

FORM 505-276

ATTACHMENT

PROBLEM: (Cont'd)

At this time the neutron error is very large and rapidly causes the calibrating integral to travel to its limit. Reference DWG D553732 and D556175E. When the T_{ave} calibrating integral travels to its limit the effective neutron power error to the cross limit is reduced and therefore results in inadequate reduction in feedwater flow.

RECOMMENDATIONS:

On DWG D556175 T1 (Calibrating Integral Operation), Logic Type ... should be added between RC D3.1 and RC D4.1. This Logic Block to read "... REACTOR TRIPPED? (By Diamond)". If YES, Block Calibrating Integral RC3.14, If NO, Proceed to RC D4.1.

DESIRED ACTION:

Please contact B&W to see if they would like to implement this ICS improvement.

APPLIES TO OTHER NSS CONTRACTS: YES

... APPROVAL | DATE | CENT. ENGR FILE 13M 2

To NSS-5 SPR 322 File (Pittman)

On 5-12-75 Discussed this with Bob Winks. Bob feels that this particular problem is part of an ongoing evaluation underway at TMI ~~and~~ to maintain the plant on line after a Turbine Trip. He says he's committed to making a final recommendation for the 205 NSS which would be issued for NSS-7, 14, 6 etc once TMI completes additional testing later this year.

7/[illegible]/75 ECR - 174 At BMCo Leo Kolonay

2-2-76 322 rev. 0 closed this date. 322 rev. 1 was generated based on [illegible] information → [illegible] in manual during transient. Don [illegible] will review rev. 1 and [illegible] as to whether or not he thinks this SPR should be closed. ([illegible])

(2695)
3-16-75 TALK TO STAN MAINGI TO SEE IF HE LOOKED INTO THIS - WAS THE PLANT IN MANUAL? SHOULD SPR BE CLEARED? ([illegible])

-?1091-2 (9-73)

SITE PROBLEM REPORT

BABCOCK & WILCOX

CUSTOMER MET ED	CONTRACT NO. 620-0003	SPR NO. 322	REV. NO. 1	
VENDOR B&W	P.O. NO.	TASK NO. 21	GROUP NO. 01	SEQ. NO. 01
SITE ENGINEER S. P. MAINGI	REQ'D. RESOL. DATE	REQ'D. COMP. DATE		

TITLE
LOW PRESSURIZER LEVEL FOLLOWING REACTOR TRIP

DESCRIPTION OF PROBLEM

SEE ATTACHED SHEET

STATUS - ACTION TO DATE INCLUDING PERSONS CONTACTED

L. M. Kolony of Engineering informed.

FURTHER ACTION RECOMMENDED BY SITE PERSONNEL

Review and implement Logic modification such that reactor trip should block calibrating integral RC 9.12, to achieve a slower cooldown rate of RC System and a more acceptable pressurizer level.

This problem should be reviewed for all other sites.

[illegible]	10/4/74	[illegible] SITE REP. SIGNATURE	10/4/74

RESOLUTION

APPROVED BY	SIGNATURE	DATE
N.S. SUPPORT ENGINEER		
TASK ENGINEER		
PROJECT MANAGER		

COST CATEGORY ☐ NORM ☐ C ☐ D ☐ G ☐ L ☐ VENDOR CLAIM

AUTH CHARGE NO | ☐ FIELD CHANGE REQ | FC NO

COMPLETION NOTIFICATION

SITE COMPLETION REPORT

☐ RECOMMENDED STDS. CHANGE

FINAL DISTRIBUTION

PROJECT MANAGER
S.O.M. CONST. REP.
QA DOC. FILE
CENT. ENGR. FILE 12N.2

DEVIATIONS ☐ NONE ☐ SEE SPR REV. NO. _____

DATE COMPLETED | SIGNED BY

S.O.M. CONSTR. REP. APPROVAL | DATE

ATTACHMENT
SPR 322 Rev. 1

DESCRIPTION OF PROBLEM:

Following a reactor trip the pressurizer level goes as low as 40 inches. SPR 322 Rev. 0 pointed out the main reason that pressurizer level goes as low, because the trapped steam pressure in turbine header decays slowly, resulting in turbine bypass remaining open for time longer than is necessary.

Further examination of the reactor trip data revealed that immediately following the reactor trip the feedwater demand actually reduced to 20% instead of reducing to 5% (cross limit value as neution power following the trip is zero).

This additional feed to steam generators contributes to the excessive cooling of the RC system and hence lower pressurizer level.

See attached B&Co report on the subject.

PROBLEM REPORT

Field	Entry
Customer & Plant	Met Ed SMI #1
Problem	POSSIBLE ICS IMPROVEMENT
Date of Problem	9/4/74
B&W S.O. No.	150L
Cust. Order No.	290628
Serial No.	X
Date Code	X
From	R. S. Rand
Emp. No.	81839
Mail Sta.	PO Box 352, Middletown, PA 17057
Serial No. and/or Part No.	X

Problem:

Reference: My PR#7 of 9/23/74

On 9/4/74 Met. Ed. Plant Superintendent, Jack Herbein stated that "On a Reactor Trip the pressurizer level should not drop as low as it does. (to approximately 40 inches) Analysis of the 8/13/74 Generator breaker trip test reveals the cross limits from neutron power error to feedwater flow control did Not perform as expected. The neutron power error cross limit should have reduced the feedwater demand immediately after reactor trip to approximately 5%. The neutron power error cross limit actually reduced the feedwater flow demand to approximately 20%. This additional feed to the steam generators contributed to the excessive rate of cooling of the RC system and the resultant drop in pressurizer level.

The reason for the above undesirable performance is as follows. On a reactor trip the CRD system transfers to manual. When the CRD is in manual the neutron power error is applied to the T_{ave} calibrating integral.

Customer Attitude: ☒ Major Concern ☐ Concerned ☐ Unconcerned

Effect on System (Briefly): Pressurizer level less than 80 inches cuts off heaters RC pressure can not be controlled.

☒ Cont'd on separate sheet

Date Solution Required: 10/14/74 or ☐ Info. Only

Signature: [signature] Date: 9/27/74

For Field Use: Days Service 0; Service $0; Expenses $0; Material $0; S.F.O. No. F72-236

Allocation: ☐ Product ☐ Product Application ☒ System ☐ System Application ☐ Warranty ☐ Other

Defective Part Returned: ☒ No ☐ Yes On ____ Date ____ RM No. ____

Failure Occurred: ☐ On Receipt ☐ In Service (Warranty) ☒ In Service (Non Warranty) >2 yrs Est. Time in Service

Possible Cause for Problem:
- 00 ☐ Faulty Material
- 01 ☐ Faulty Manufacturing
- 02 ☒ Faulty Design
- 03 ☐ Performance Deficiency
- 04 ☐ Faulty Packaging
- 05 ☐ Component Failure
- 06 ☐ Wearout
- 07 ☐ Improper Application
- 08 ☐ Operating Environment
- 09 ☐ Insufficient Instructions (Documentation)

Report of Investigation & Corrective Action (by Field if Applicable):

Operating Conditions: NA
Ambient Temp ____ °F
Atmosphere ☐ Clean ☐ Average ☐ Dirty
Humidity ☐ Hi ☐ Lo ☐ Avg

Time Required to: NA
Repair ____ Troubleshoot ____
Recalibrate ____

Failure Details: NA
BMCO Part No. ____
Describe (Diode, Cap., Transistor, etc.) ____
Circuit Symbol (C1, R1, Q1) ____
Mfg. of Part (If Known) ____

How Part Failed: ☐ Short ☐ Open ☐ Mech. Damage ☐ Adjustment ☐ Dirty ☐ Unknown ☐ Other (Describe)

(QA Use Only)

Problem Type	Failure Cause Code	P/A System	Answer Due	Action Taken By

Copies: Quality Assurance; Prod. Liability; Prod. Planning; NPD; FPO; Contract Oper.; Warranty Repair; Co-ordination Engr.; Comm. Services; Order Center; Component Engr.

Copies: ____

☐ Preliminary Ans. ☐ Final Solution — Signature ____ Date ____

Approval ____ Date ____

Follow Up on Corrective Action: ____

Disp. of R.M. ____
Date Rec'd ____
Date Returned ____

Hours Spent Solution

Dept.	Man	Hrs.

Qty	Name	Part No.	Comments

Form SDS 274

ATTACHMENT

PROBLEM: (Cont'd)

At this time the neutron error is very large and rapidly causes the calibrating integral to travel to its limit. Reference DWG D553732 and D556175H. When the T_{ave} calibrating integral travels to its limit the effective neutron power error to the cross limit is reduced and therefore results in inadequate reduction in feedwater flow.

RECOMMENDATIONS:

On DWG D556175 T1 (Calibrating Integral Operation), Logic block should be added between RC D3.1 and RC D4.1. This logic block to read "IS REACTOR TRIPPED? (By Diamond)". If YES, Block calibrating Integral RC9.12; If NO, Proceed to RC D4.1.

DESIRED ACTION:

Please contact B&W to see if they would like to implement this ICS improvement.

APPLIES TO OTHER MSS CONTRACTS: YES

APR 16 1975

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To
C. A. Creacy, Project Management

From
R. W. Winks, Control Analysis, EXT 2544 RWW

BOS 663.8

Cust.
Duke Power Company

File No. or Ref.

Subj.
A Feedwater Pump Speed "Kicker" Circuit for Oconee Unit #1 (and #2 and #3)

Date
April 16, 1975

This letter to cover one customer and one subject only

The following information on a modified Feedwater Pump Speed Control Circuit, which has been implemented at the TMI-1 Plant, is being forwarded at this time to allow incorporation prior to any additional load rejection tests scheduled for Units 1, 2, or 3. B&W recommends this as a temporary change to your ICS which will be followed up by a field change package.

The purpose of the Pump Speed "Kicker" Circuit is to enable the feedwater flow control system to continue to deliver a high level of flow to each Steam Generator, even though the steam outlet pressure has suddenly increased approximately 150 psi. The modification originated and implemented at TMI-1 improved the overall plant performance by directly increasing feedwater pump speed more rapidly than could be accomplished by the combined action of the feedwater control valve and pump speed controllers in the ICS.

The modification to the present Unit #1 ICS is the following:

> Route an additional wire from the turbine header pressure error signal (IC/B on Drawing Unit #1) to two summers which are shown on Drawing D8032313F, The particular summers are FW28.8B(6-1-3) and FW28.12B(6-2-1) and the new signal is to be connected to each summer. In the new line also install a diode and a 100 k ohm resister in a path to ground. Refer to the attached schematic.

The turbine header pressure control error signal has the following effective range:

+ 10 Volts = 300 psi above setpoint.
+ 0.5 Volts = 15 psi above setpoint.
Any signal below 0.5 Volts will be blocked by the diode in the new line.

The gain for the turbine header pressure error signal on each summer will have to be calculated and set by Duke Power Company. The calculation used at TMI-1 is given below:

1. Assume pump discharge pressure varies with the square of pump speed when suction pressure is constant.

2. For two feedwater pump operation find the pump speed (S_0) when the plant is at 100% power.

3. This speed corresponds to a turbine header pressure of 885 psig.

4. For a turbine header pressure of approximately 1050 psi the required pump speed would have to be: $(S_1) = (S_0)\sqrt{1050 \div 885}$. The increase in pump speed $(S_1 - S_0)$ represents a fraction of the total range of feedwater pump speed (2700 rpm to 5400 rpm at TM-1) and the output voltage of each summer should be adjusted to cause an increase in feedwater pump speed to (S_1) when the turbine header pressure error signal corresponding to 1050 psi occurs on each summer.

The following numerical calculation is only an example:

Assume (S_0) is 4500 rpm at 100% power. Then (S_1) is 4500 x 1.089 = 4900 rpm and $(S_1 - S_0)$ = 400 rpm. Assume total speed range of feedwater pumps is 2700 to 5400 rpm or the total increase in pump speed is 2700 rpm. The gain at each summer is that which will cause a pump speed of 4900 rpm when the error in turbine header pressure is 165 psig.

If you have any questions, I would be happy to discuss them with you.

RNW:lr

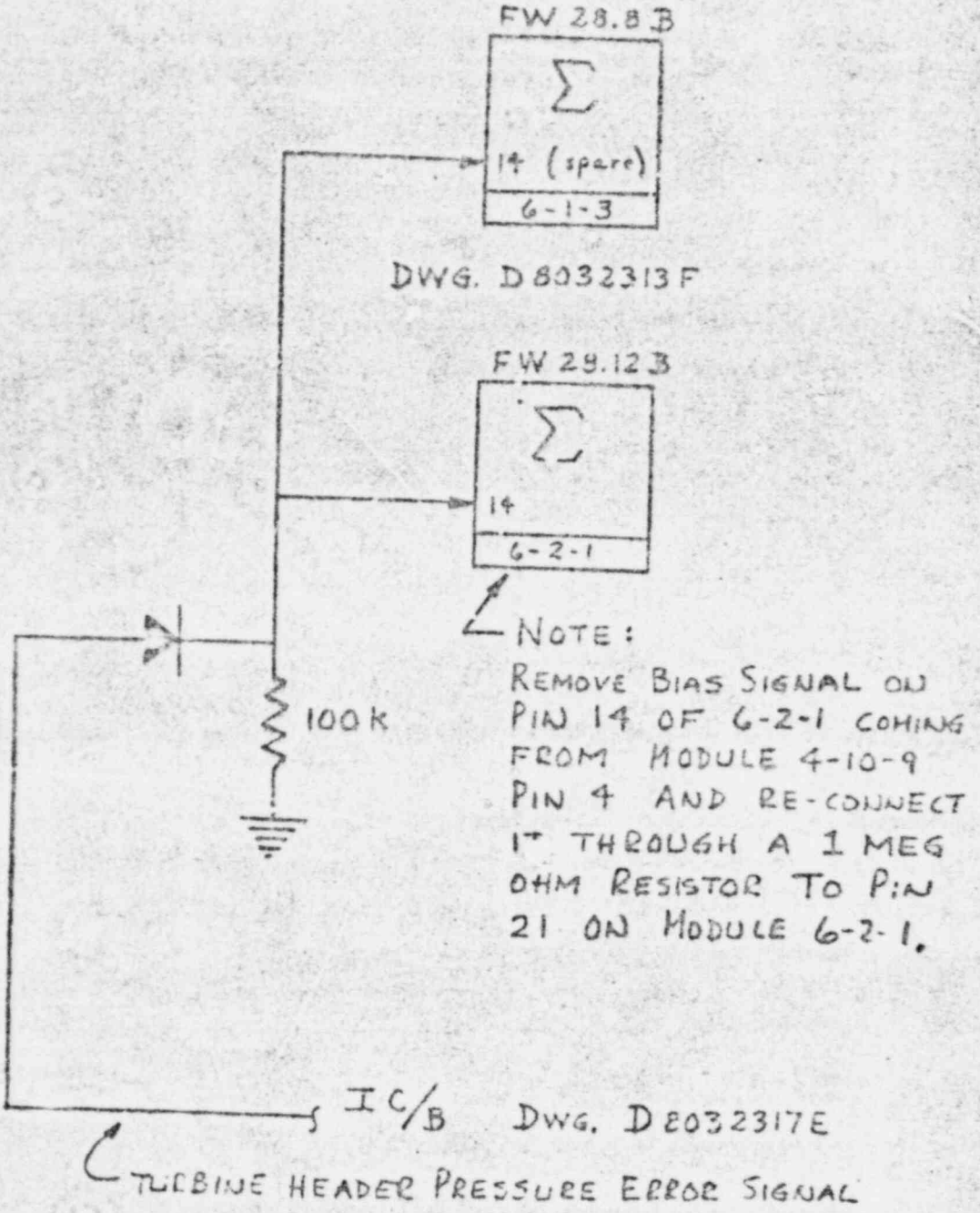
cc: B. A. Karrasch
A. W. Brown
W. Van Scoter
E. F. Ryan
W. E. Wilson
J. T. Janis
J. J. Galan
R. S. Rand (EM Co)

Q/A The information contained in this memo has been checked for applicability and completeness.

Signature R D Randall

Date 4/4/75

FEEDWATER PUMP SPEED "KICKER" MODIFICATION TO THE ICS



PROBLEM REPORT

Date: 9-4-74 | 1500 | 290522

R. S. [illegible] | 51239 | 779 | Middletown, PA 170[illegible]

TMI #1

ICS SYSTEM INSTRUMENTS (721)

During a B&W presentation to the customer September 4, 1974, Mr. J. G. [illegible], Plant Superintendent stated that there are two areas of operation that require improvement.

1. On a Reactor trip the pressurizer level should not fall as low as it does. (To approximately 40 inches)
2. On a Turbine trip from 100% power the reactor should not trip.

The following are suggestions based on field observations and study of various parameters from the data available to date at Met. Ed. TMI #1. Attached please find revised logic and schematic drawings to show the following suggestions and also charts and table 1 from generator trip test 8/13/74.

PROBLEM 1:

It appears that after a reactor trip the turbine header pressure is considerably greater than Steam Generator pressure, probably due to the check valve action of MSV-1A, 1B, 1C, 1D which are installed in the Main

EFFECT ON SYSTEM (BRIEFLY): 40" pressurizer level cuts off heaters, If "locked in pressure" stayed high (due to feed pump trip) R.C. System would be further reduced. [illegible]

CUSTOMER ATTITUDE: ☒ [illegible] CONCERN ☐ CONCERNED ☐ UNCONCERNED

DATE SERVICE REQUEST: 10/1/74 ☐ INFO ONLY

☒ CONT'D ON SEPARATE SHEET

SIGNATURE / DATE: [illegible] 9/4/74

[illegible] 9/4/74

ACTION / INVESTIGATION & CORRECTIVE ACTION (USE FIELD IF APPLICABLE)

ALLOCATION: ☐ PRODUCT ☐ PRODUCT APPLICATION ☐ SYSTEM ☒ SYSTEM APPLICATION ☐ WARRANTY ☐ OTHER

DEFECTIVE PART RETURNED: ☒ NO ☐ YES

PROBLEM:

Steam lines between the instrument connections for Turbine Header pressure and Steam Generator pressure.

Attached please find charts recorded during a generator trip test conducted 8-13-74 which resulted in a reactor trip. From these charts and table 1 it can be noted that while there was flow through the line there was a pressure drop i.e., SG pressure is 30 psi higher than THP, immediately after the trip with no pressure drop due to flow the pressures equalized. For the next 3.5 min. the SG pressure is less than THP. Also note the waves in the SG pressure recorded during the first minute are not present in the THPressure. These conditions seem to be a good indication that the check valves did close.

The fact that the Turbine Header pressure remains higher for a time holds the Turbine Bypass Valves open longer than necessary. (AFTER The Safety Valves have closed)

It is believed that this area could be improved by having the turbine bypass valves modulate to control Steam Generator pressure when the reactor is tripped instead of turbine header pressure. Getting these valves closed sooner after a trip should keep the Steam generator pressure from dropping so low which would keep the steam generator temperature higher and therefore TAVE. This should then keep the pressurizer level from dropping so low because it would not have to make up as much volume in the RC system.

PROBLEM 2:

On August 13 the generator breaker trip from 100% power transient was tested. The Reactor tripped on high RC pressure four seconds after the generator trip. After analysis of Reactimeter data it was observed that the system came very close to NOT tripping the reactor. We had previously sustained two turbine trips from 70% power without tripping the reactor. (The reactor did trip on the second turbine trip but not until the reactor had run all the way back to 20% power and this trip was due to operator error.) The most important item to be worked on to prevent reactor trip would be to reduce the maximum steam generator pressure immediately after the generator trip. This might be achieved by lowering the popping pressure on the last two banks of safety valves. In addition, the Emergency Relief Valve control could be modified to utilize the emergency relief valves (atmospheric dump valves) as additional steam relief capability. Presently the BTU limits as specified by B&W call for immediate reduction of feedwater flow demand to approximately 60% due to the rise in steam generator pressure from 910 psig to 1070 psig on a turbine trip. The elimination or delay of the BTU limits in this situation would help to maintain the required feedwater flow which would in turn tend to prevent overheating the primary coolant and to prevent the Reactor tripping on high RC pressure.

The attached logic and schematic drawings reflect a suggested way to utilize the atmospheric relief valves to open for overpressure at any time, whether the turbine bypass valves are available or not.

It must be considered that this change will substantially increase the use of these atmospheric relief valves and might increase their maintenance requirements.

DESIRED ACTION:

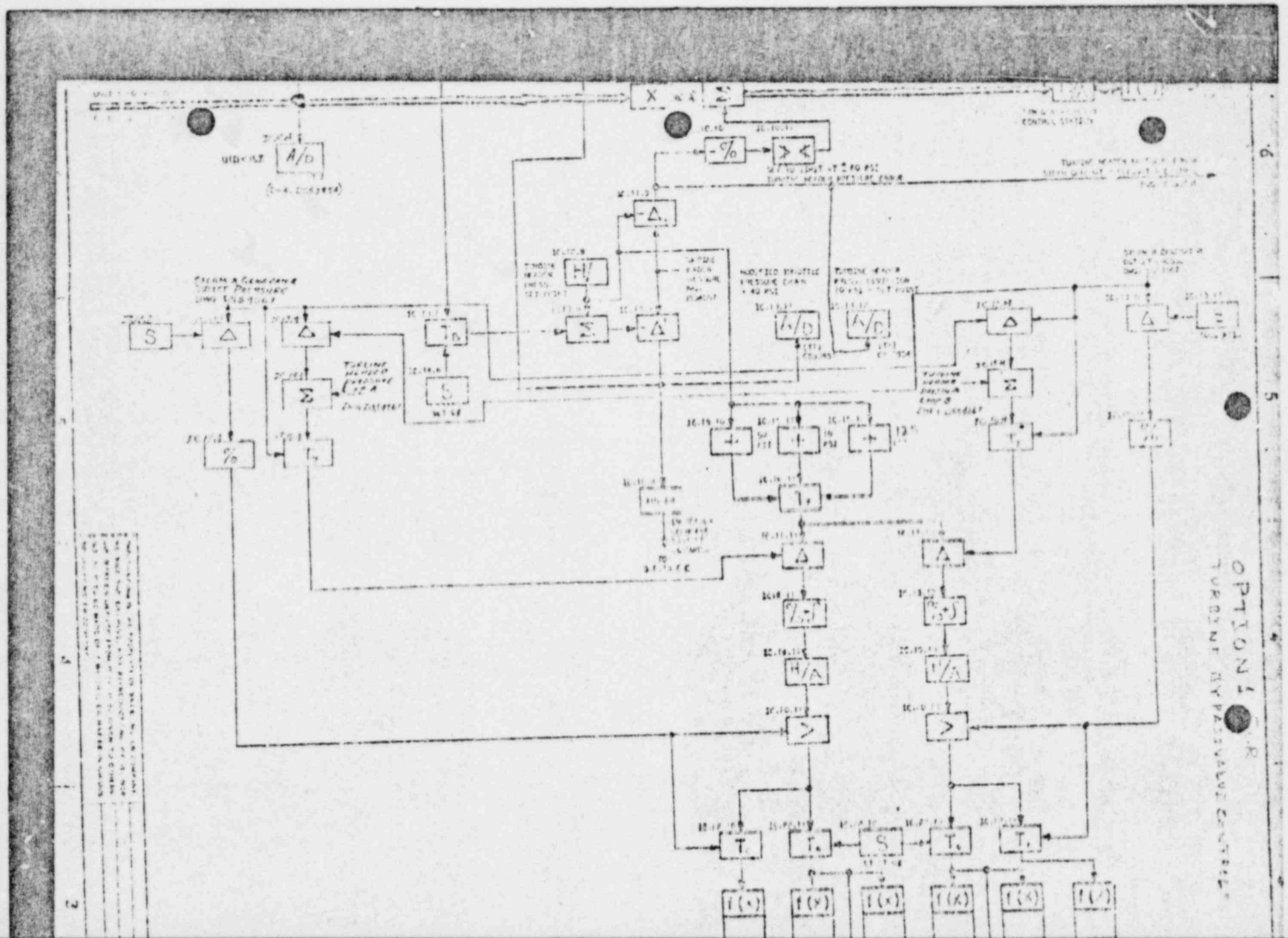
Contact B&W to see if they would like to implement either of these suggestions.

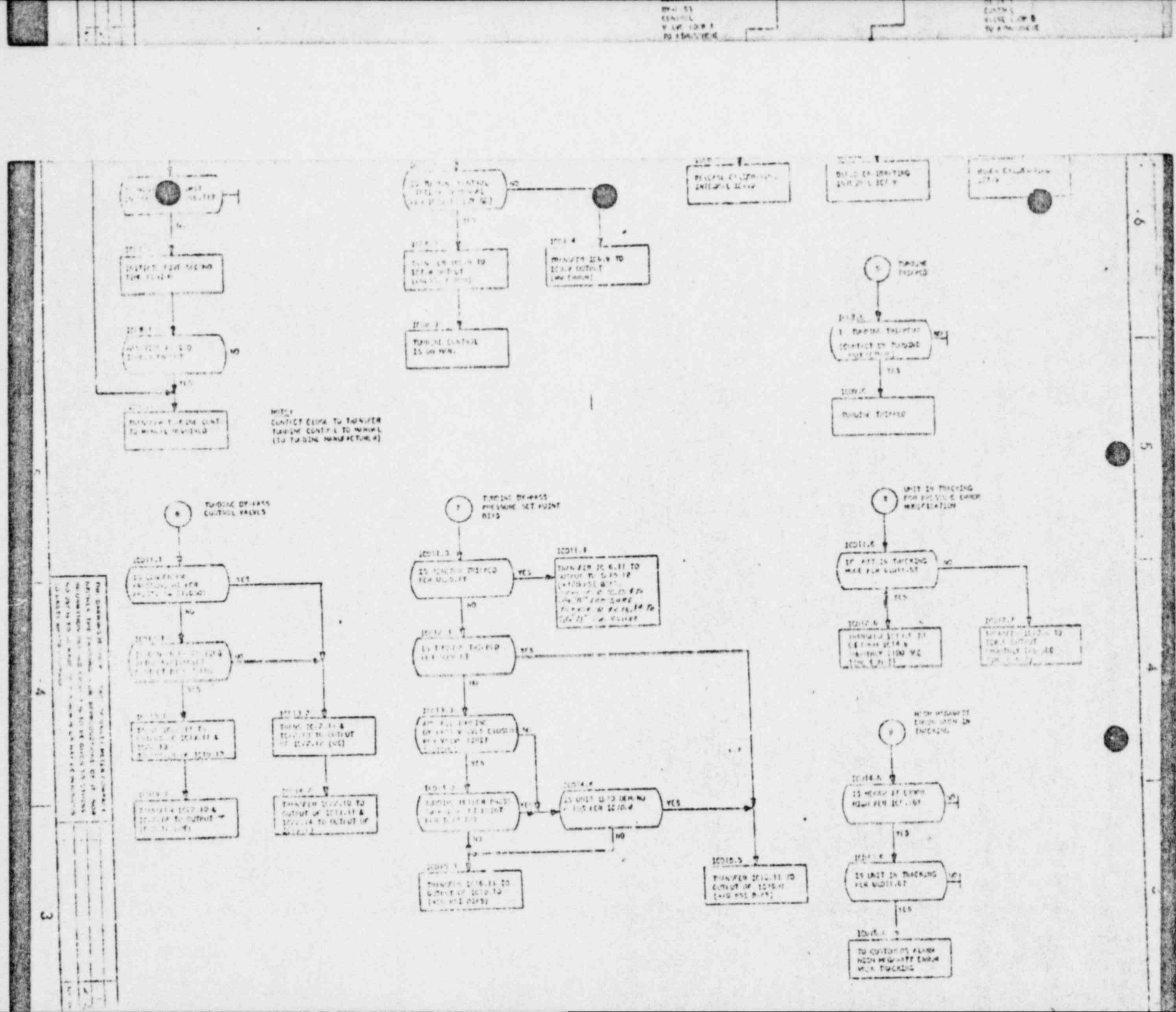
APPLIES TO OTHER NSS CONTRACTS: YES (ITEM 2)

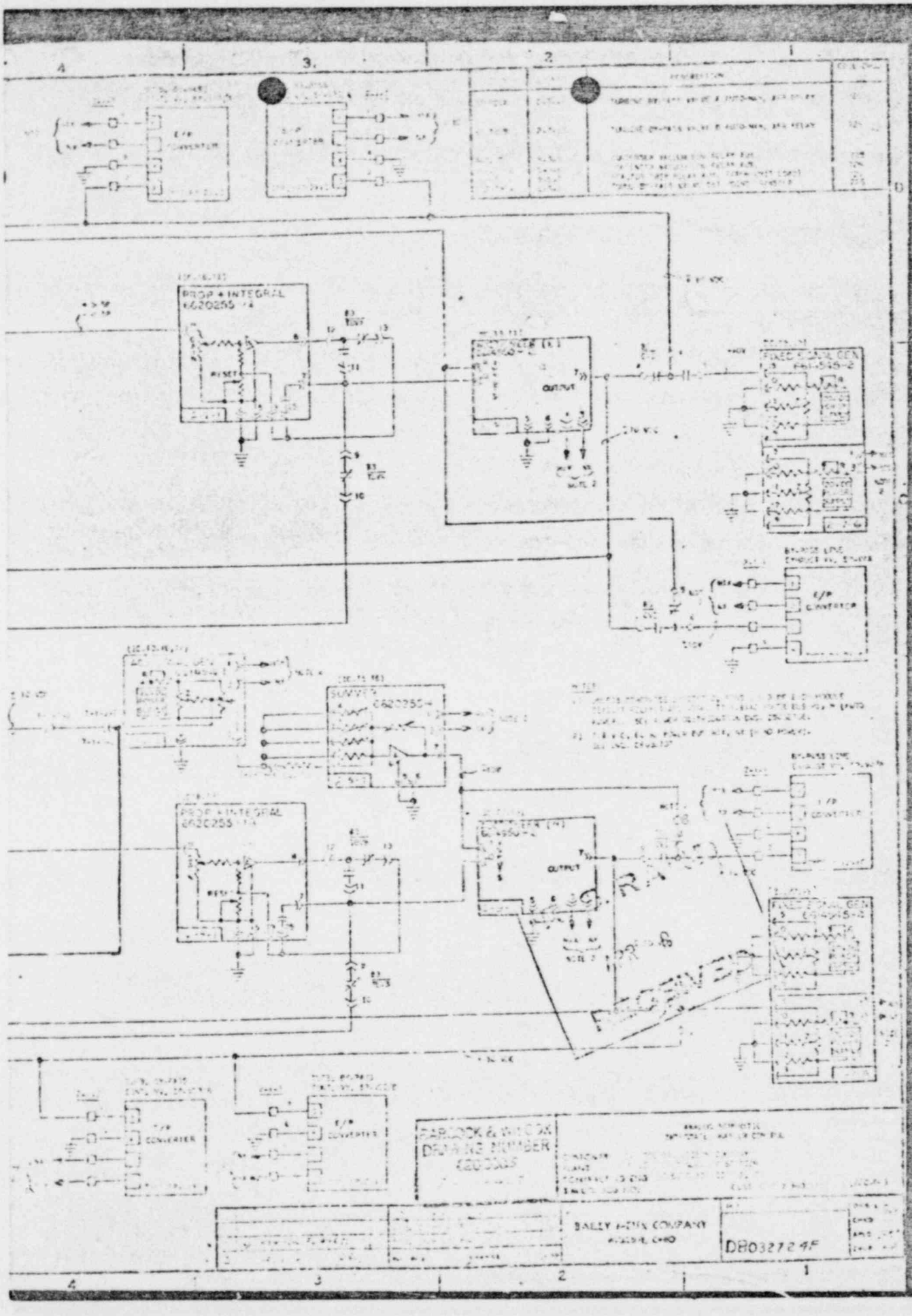
TABLE 1

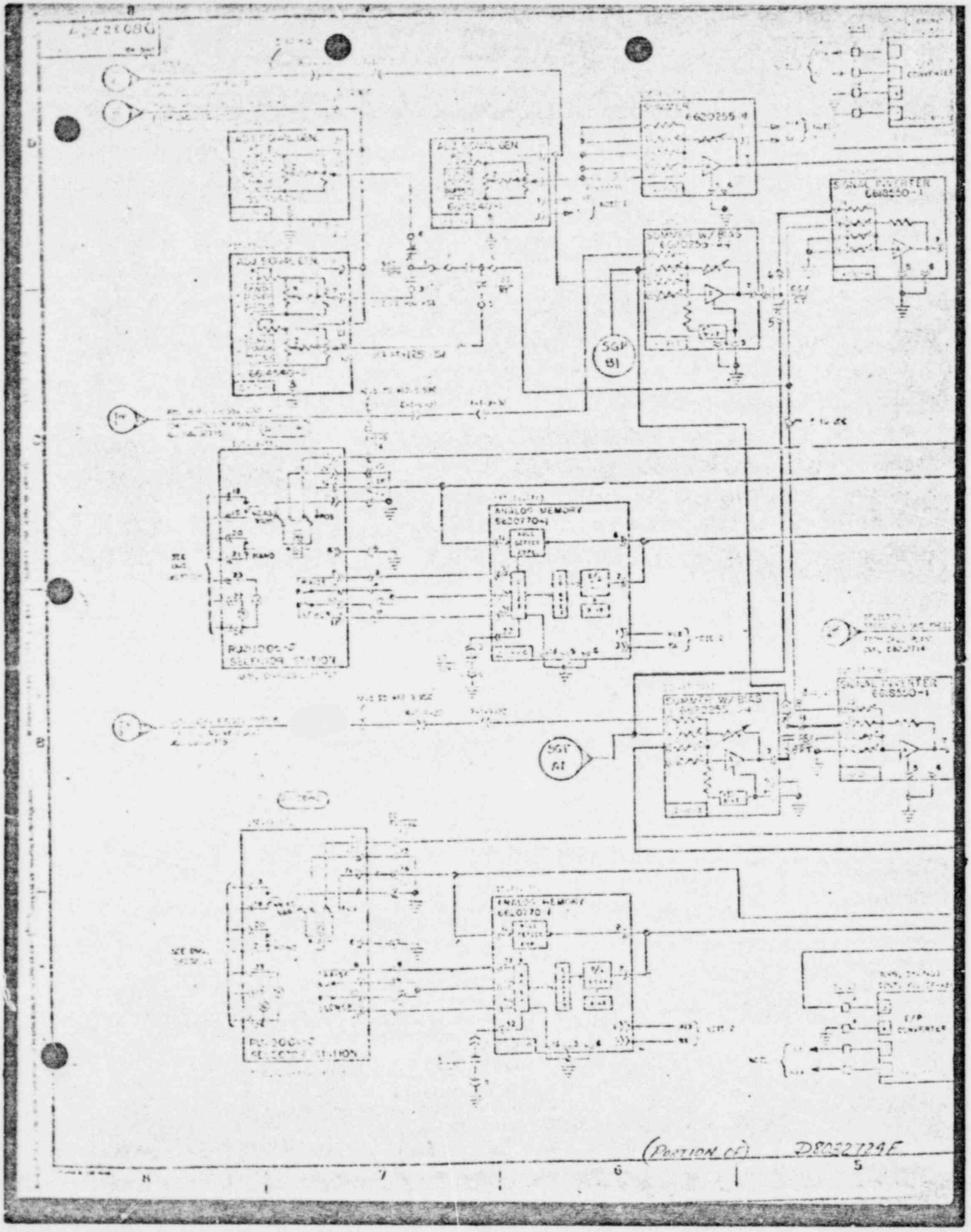
Steam Generator pressure and Turbine Header Pressure recorded during generator trip test 8/13/74 at TMI Unit 1.

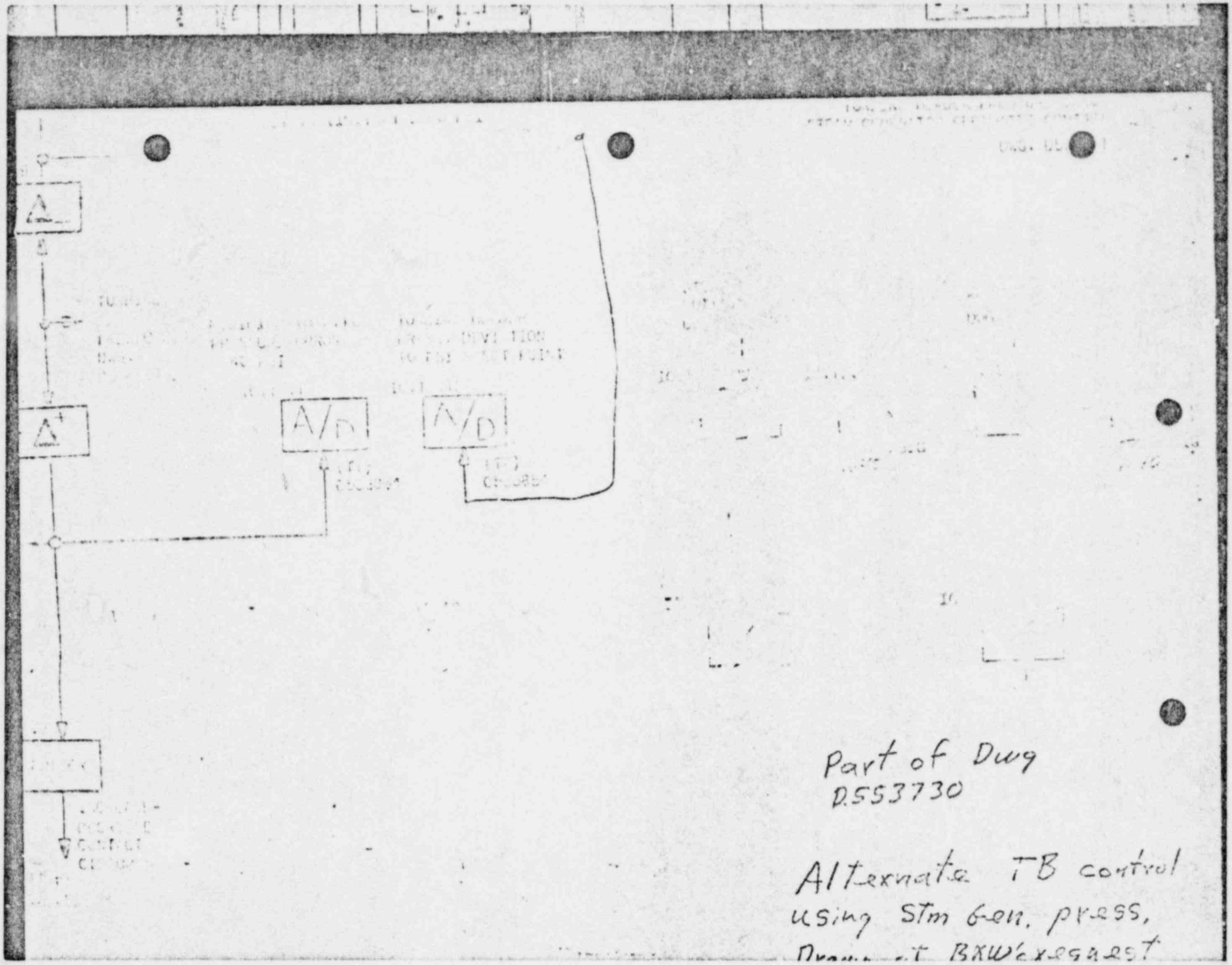
	SG Press.	TH Press.	THP-SG Press.
Before the trip	912 PSI	882 PSI	-30 PSI
peak	1080	1080	0
12 sec.	1068	1074	+6 PSI
24 "	1032	1059	+27
36 "	1008	1046	+38
48 "	984	1038	+54
1 min./60 "	966	1032	+66
12 "	948	1021	+73
24 "	930	1017	+87
36 "	926	1008	+82
48 "	936	1004	+68
2 min./60 "	962	997	+35
12 "	984	994	+10
24 "	986	987	+1
36 "	1002	996	+6
48 "	1008	1005	-3
3 min./60 "	1014	1011	-3
12 "	1032	1020	-12
24 "	1032	1023	-11
36 "	1020	1020	0
48 "	1020	1020	0
4 min./60 "	1020	1020	0


A/D
OPTION 1
TURBINE BYPASS/HP CONTROL



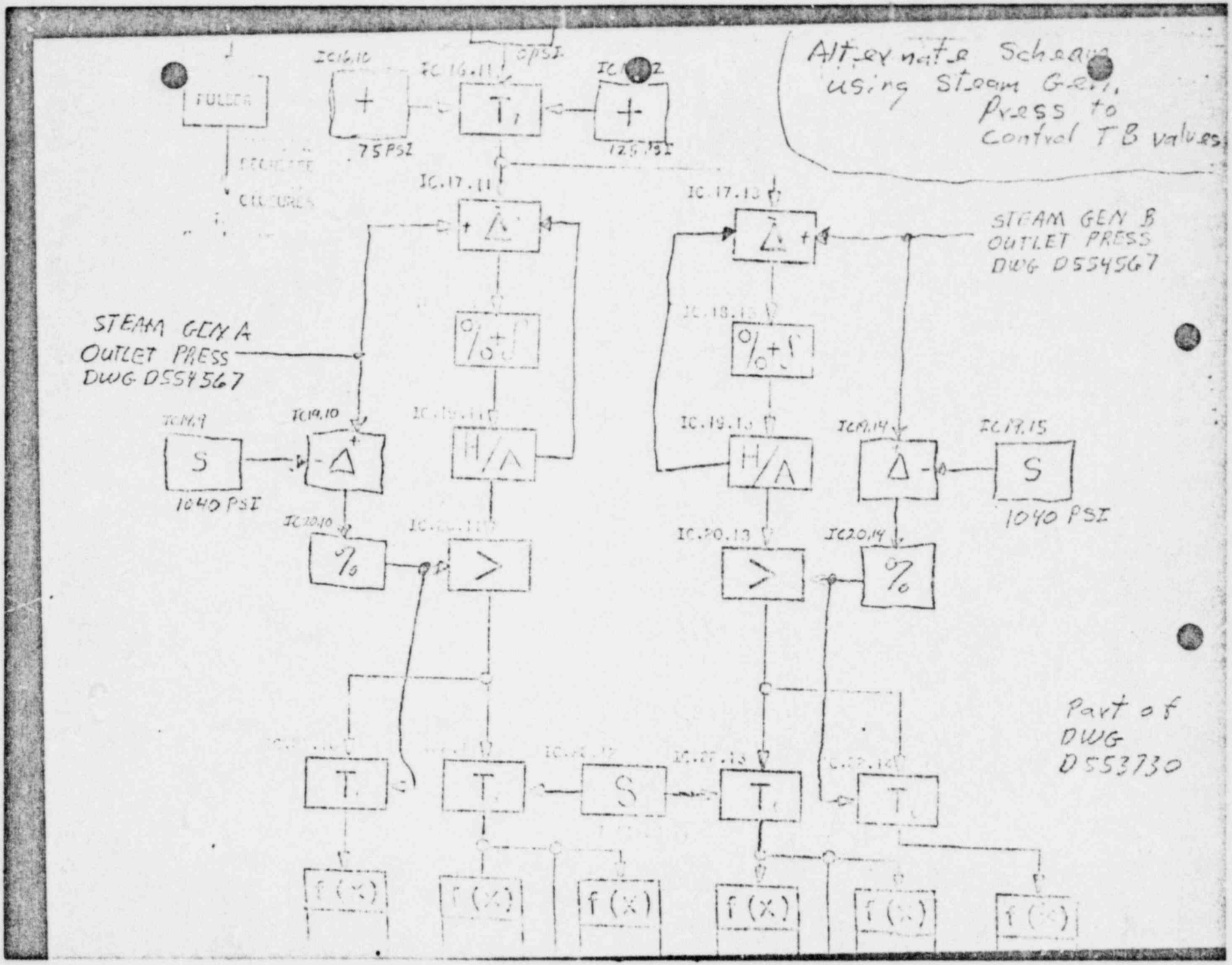

PROP + INTEGRAL
E/P CONVERTER
OUTPUT
RESET
SUMMER
BALLY MIDWAY COMPANY
DB0327249F

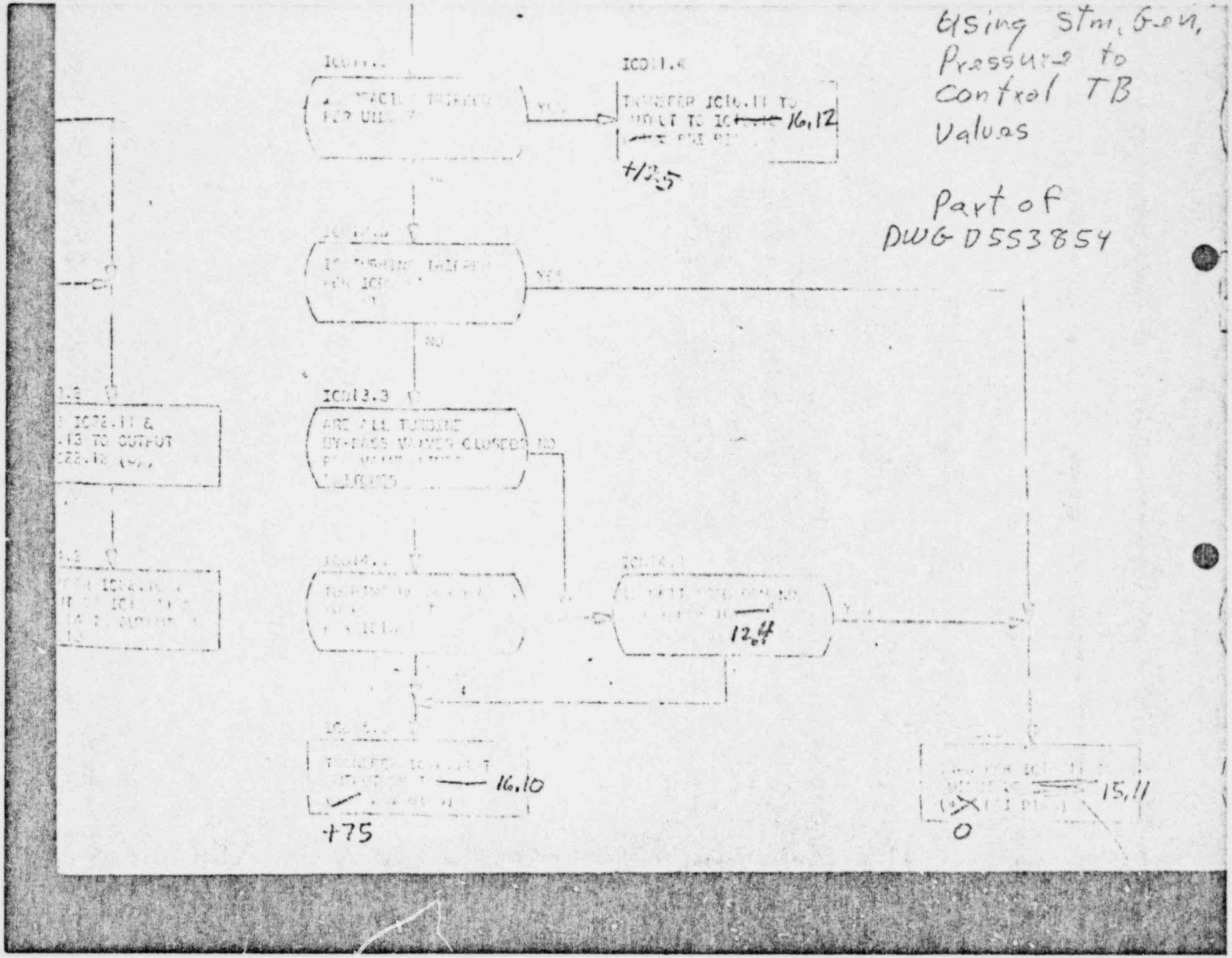

ANALOG MEMORY
SGP
B1
SGT
A1
(PORTION C1)
D8032724E



Part of Dwg
D553730

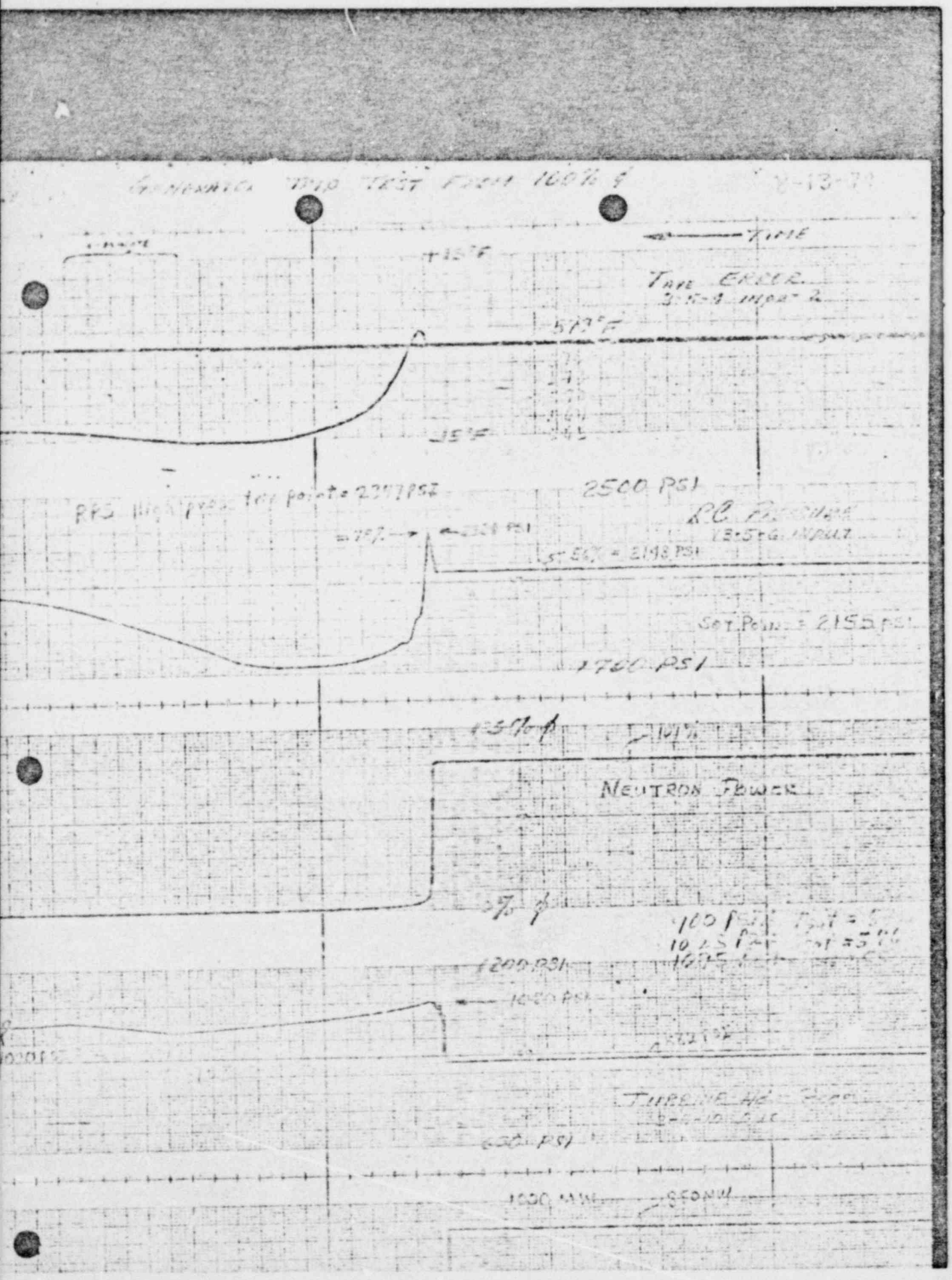
Alternate TB control
using Stm Gen. press,
Drawn at BKW's request

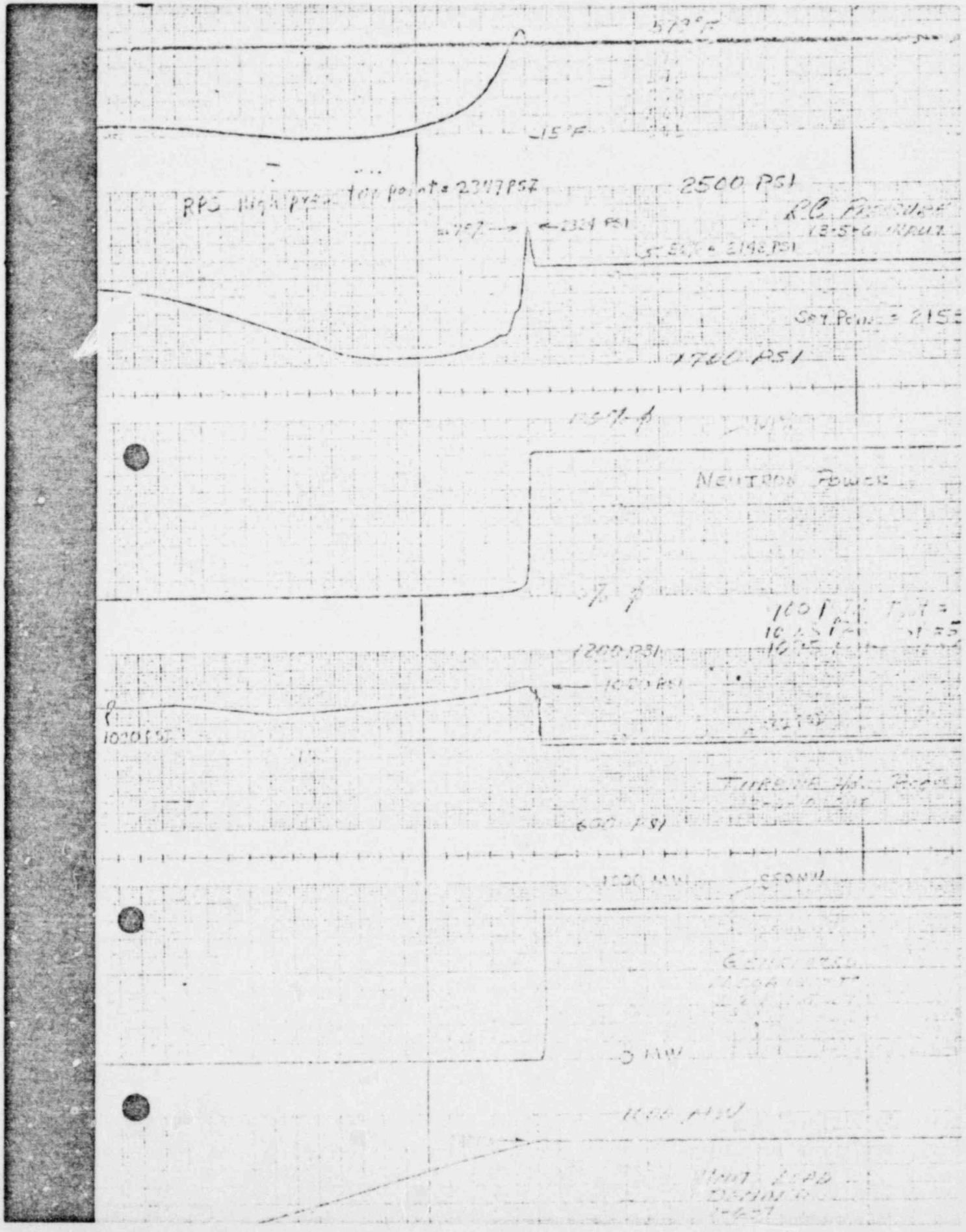

Alternate Scheme using Steam Gen. Press to control TB values
FULCRUM
IC16.10
+
75PSI
IC 16.11
T1
0PSI
IC
+
125 PSI
IC.17.11
IC.17.13
STEAM GEN A OUTLET PRESS DWG D554567
STEAM GEN B OUTLET PRESS DWG D554567
IC.18.13
IC.19.10
IC.19.13
IC19.14
IC17.15
S
1040 PSI
IC20.10
IC20.13
IC20.14
Part of DWG D553730
f(x)

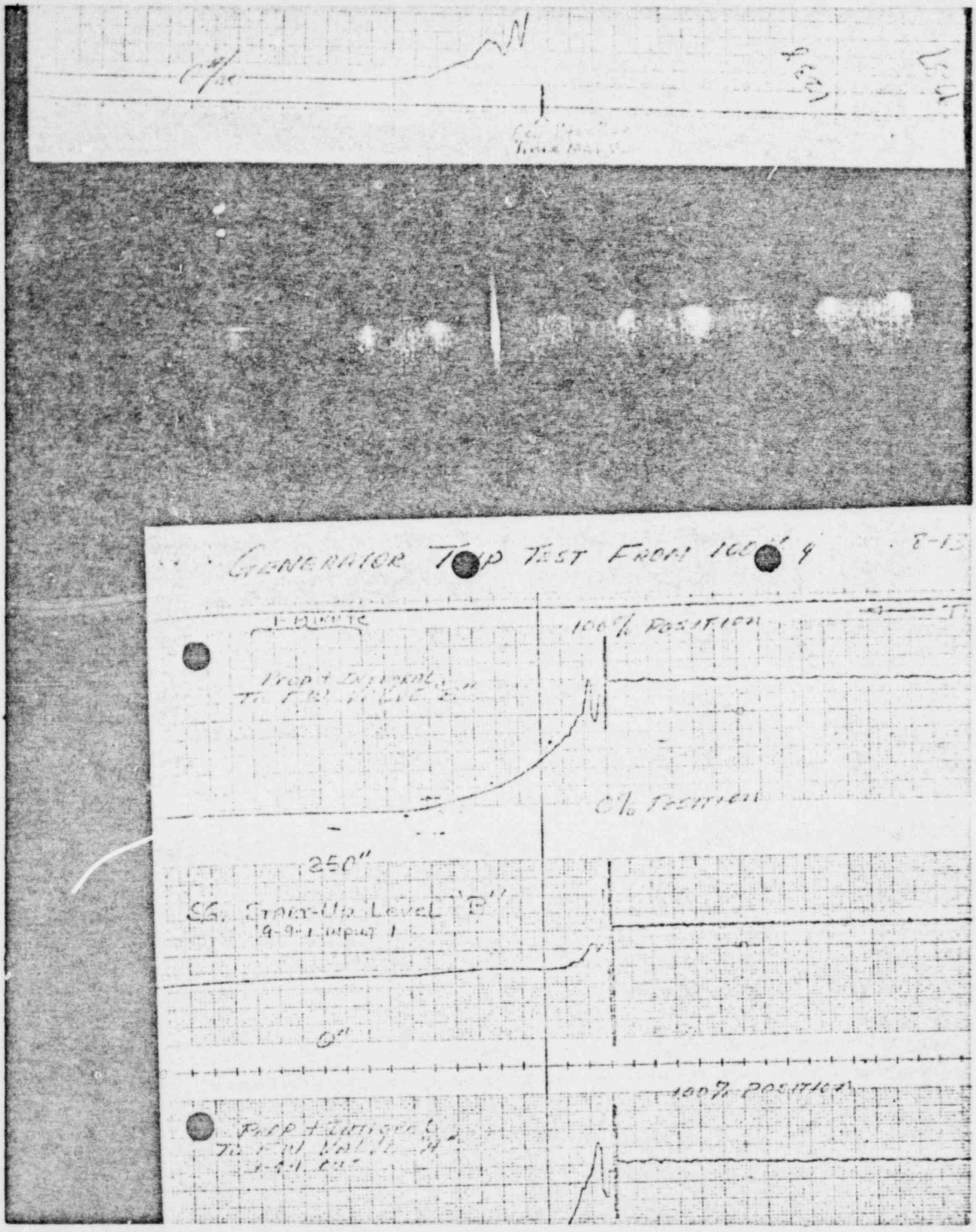


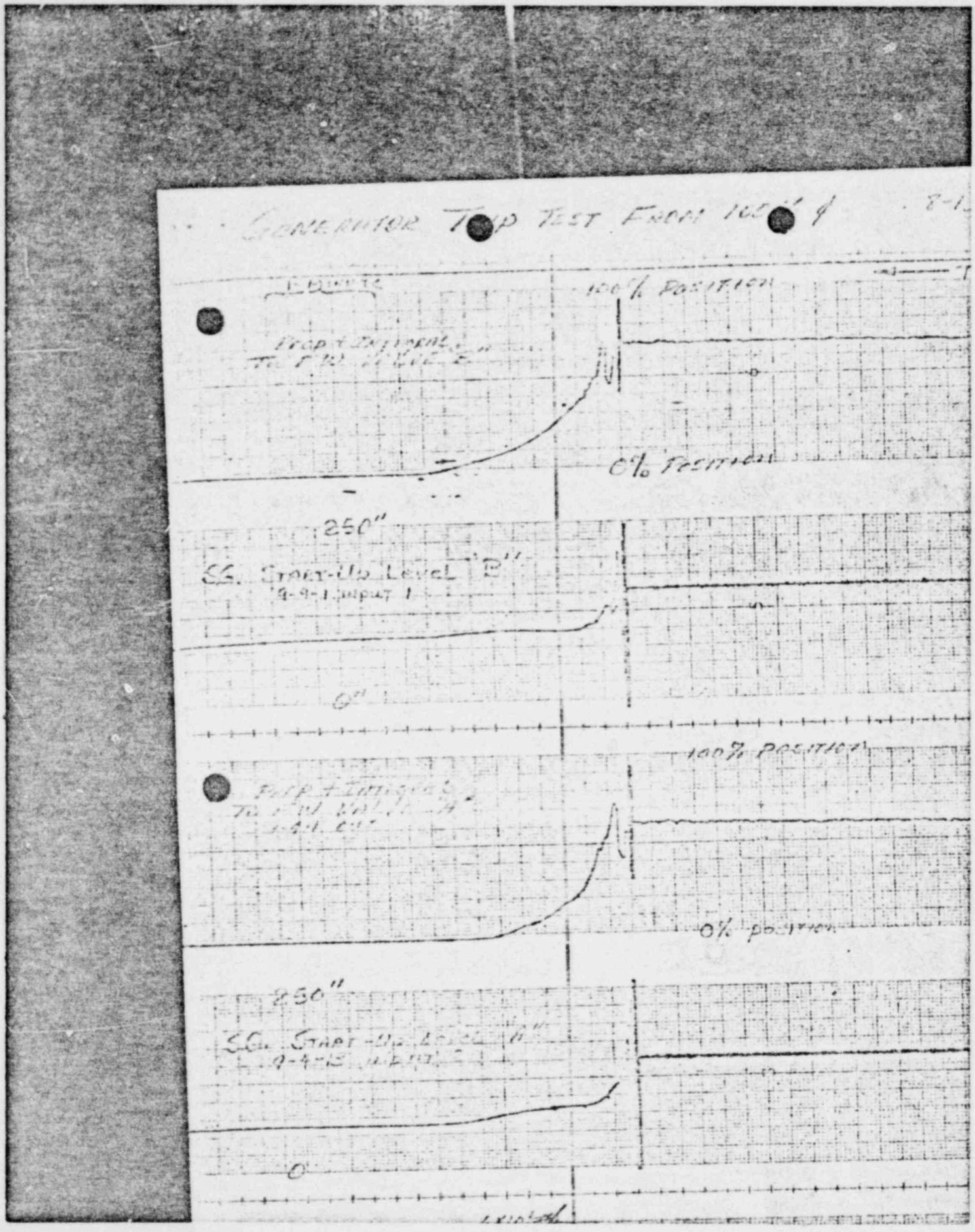
Using Stm. Gen. Pressure to control TB Values

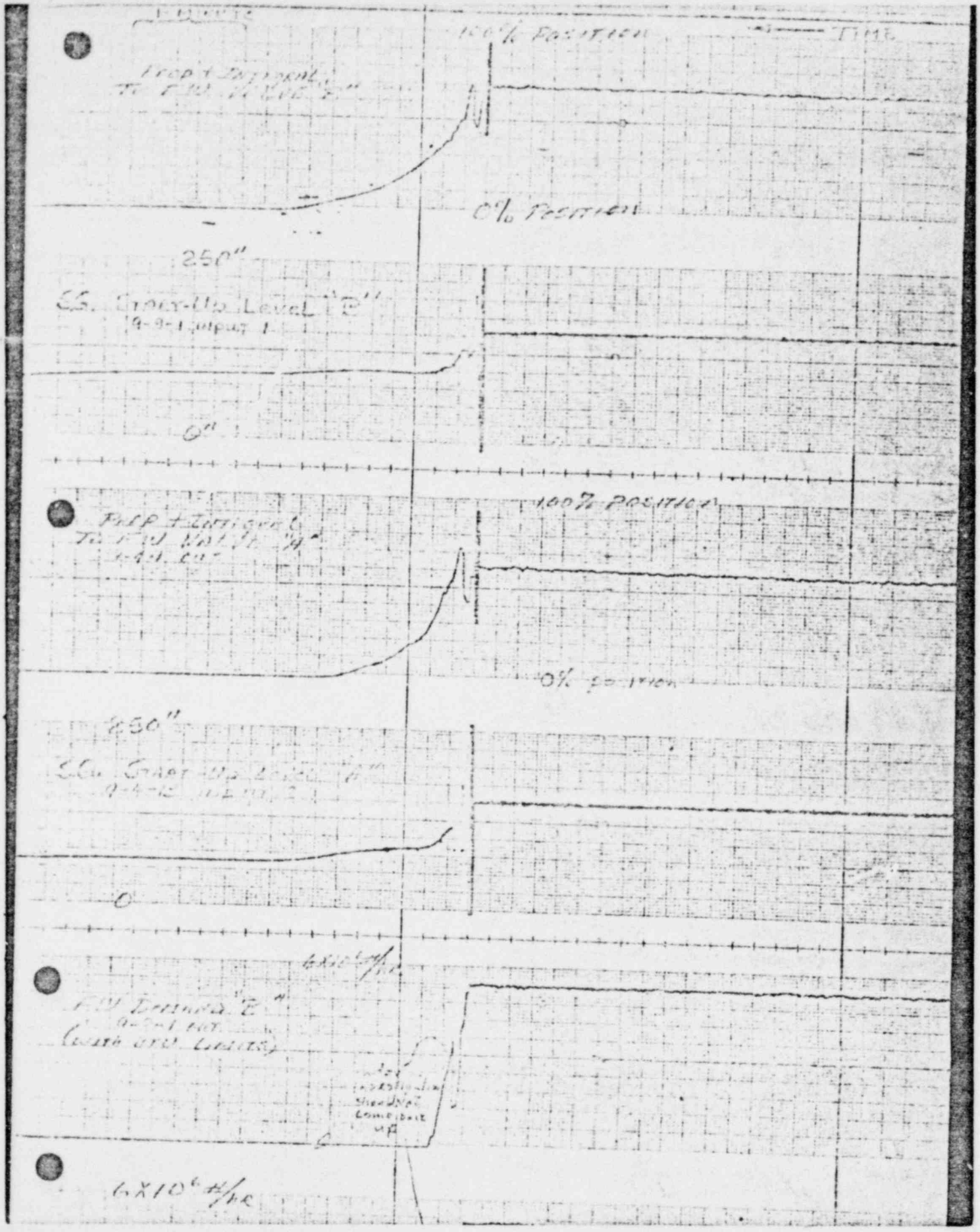
Part of DWG D553854

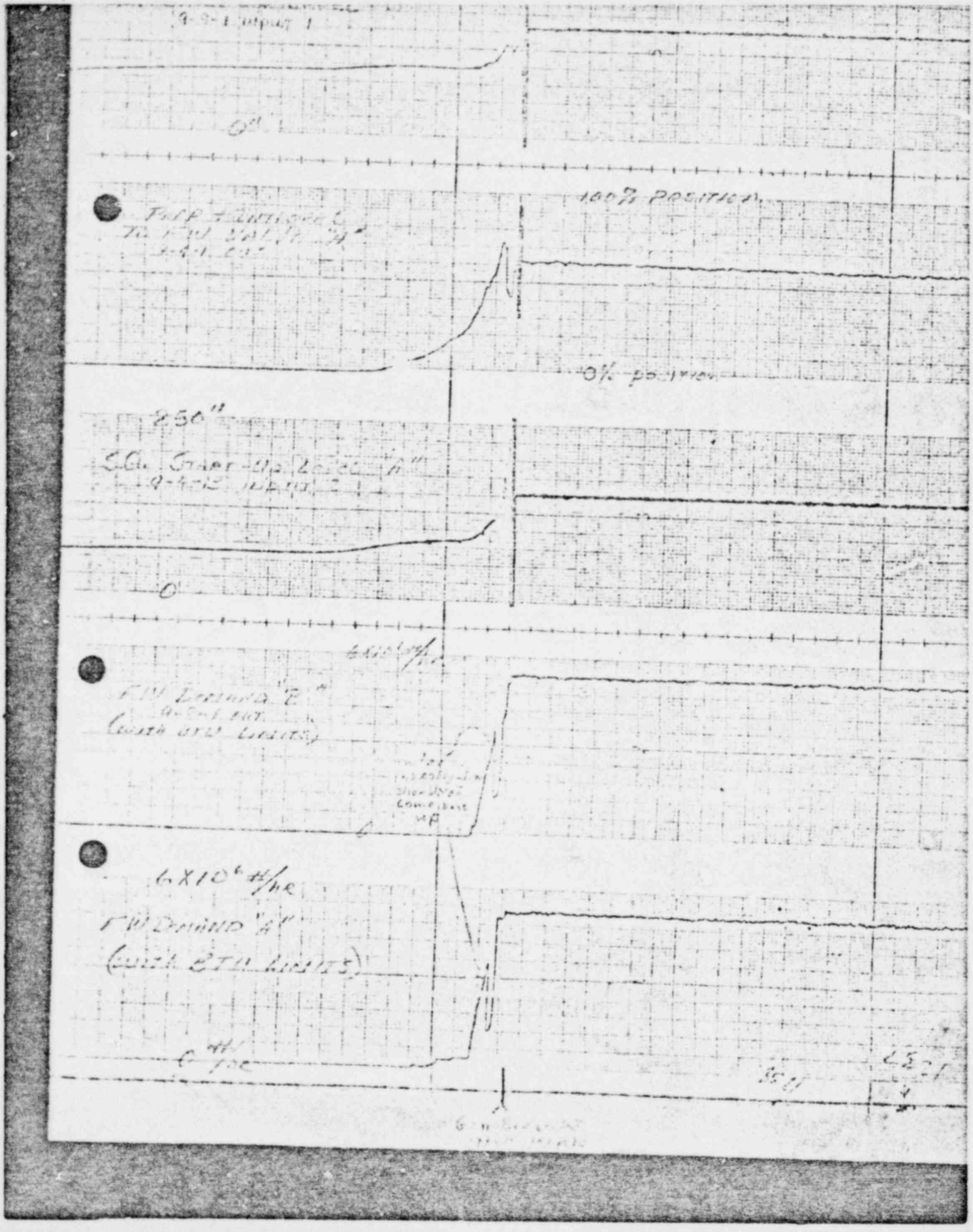

TIME
TAVE ERROR
2500 PSI
NEUTRON POWER
TURBINE MW

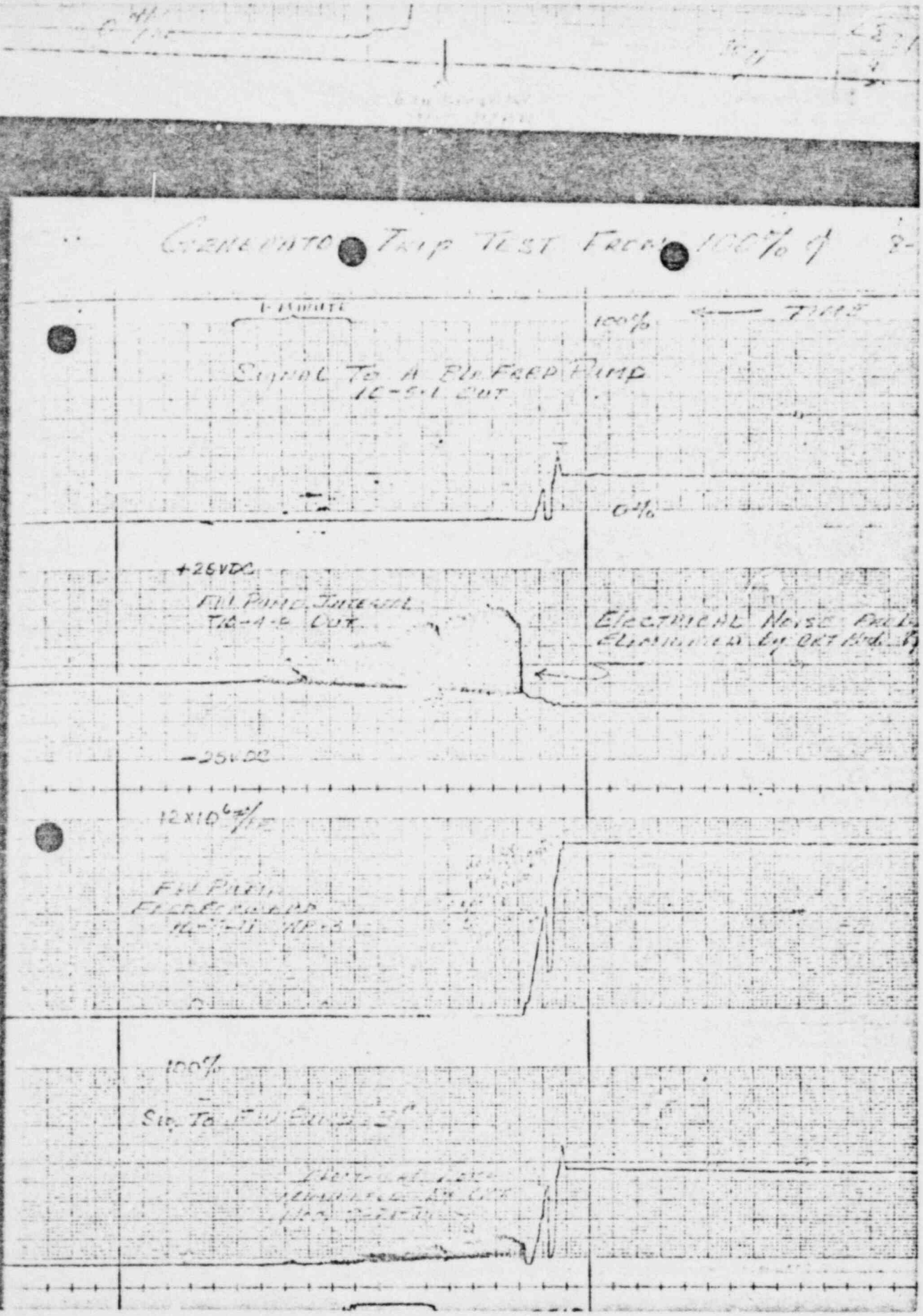

RPS High Pressure Trip point = 2317 PSI
2500 PSI
RC Pressure
-324 PSI
Set Point = 2155
1760 PSI
NEUTRON POWER
1200 PSI
1020 PSI
600 PSI
5 MW

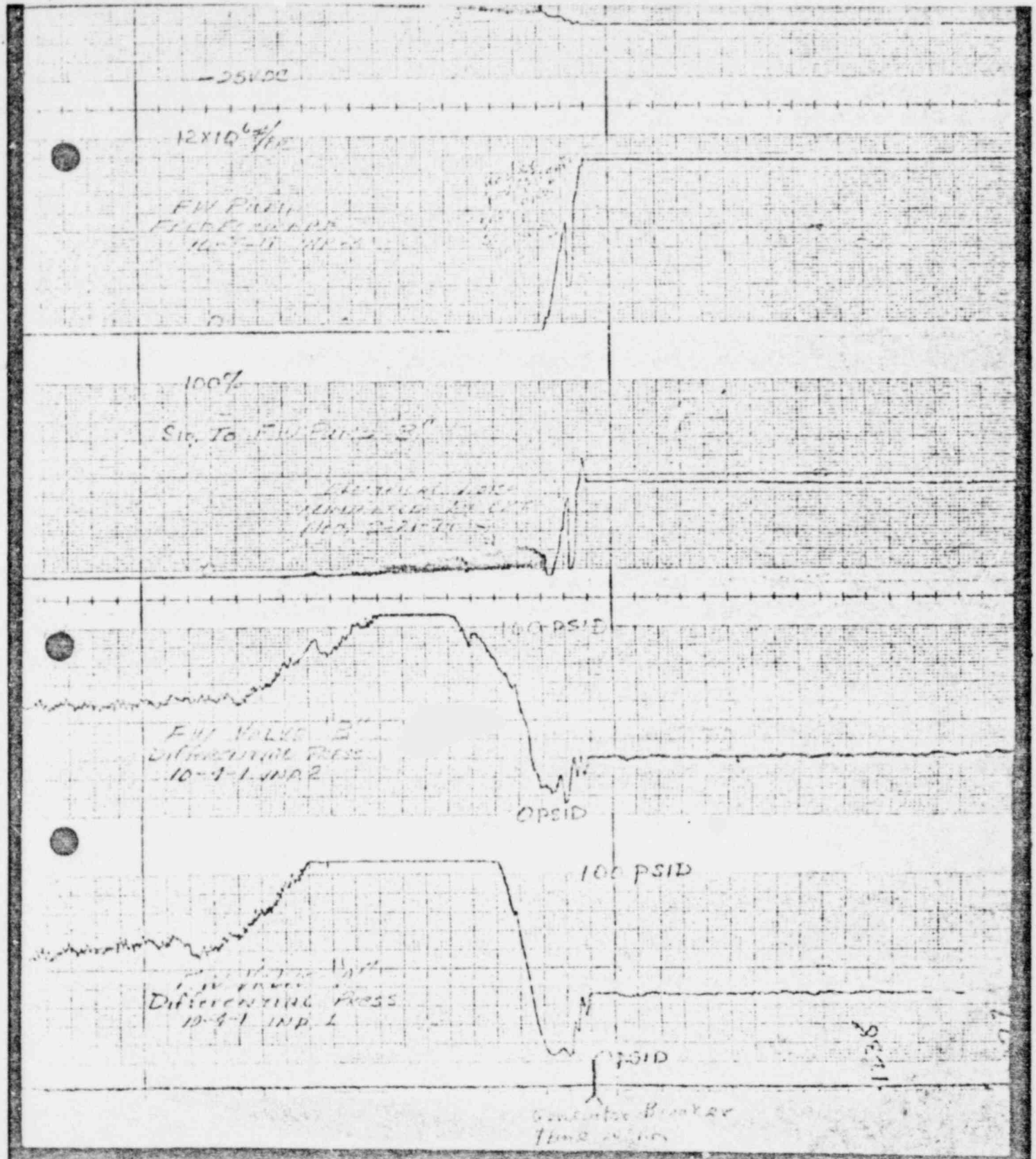

GENERATOR TRIP TEST FROM 100% 9
100% POSITION
0% POSITION
250"
0"
100% POSITION


GENERATOR TRIP TEST FROM 100% ϕ
100% POSITION
0% POSITION
250"
0"
100% POSITION
0% POSITION
250"
0"

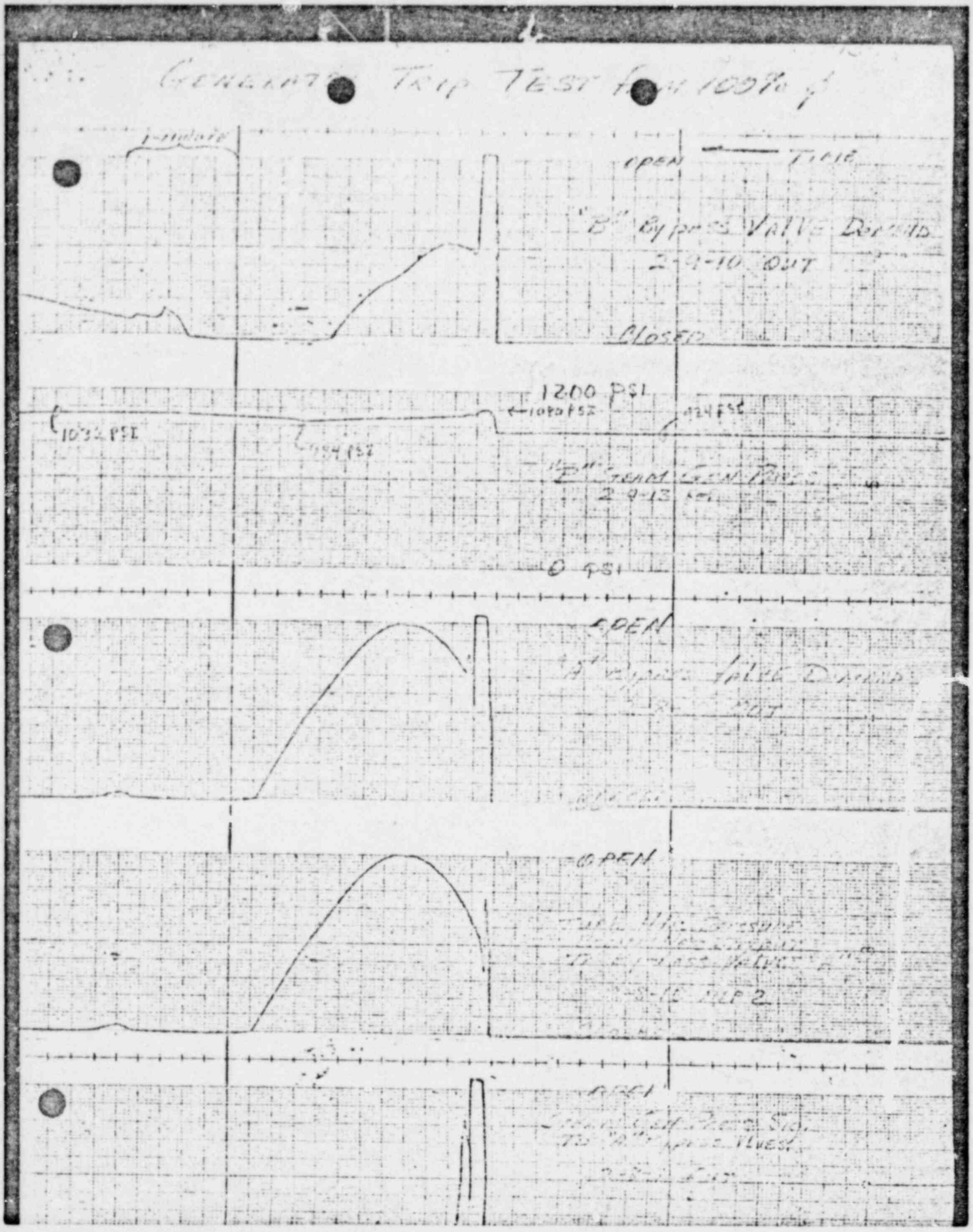

100% POSITION
TIME
0% POSITION
250"
0"
100% POSITION
0% POSITION
650"
0
6X10⁶ #/hr

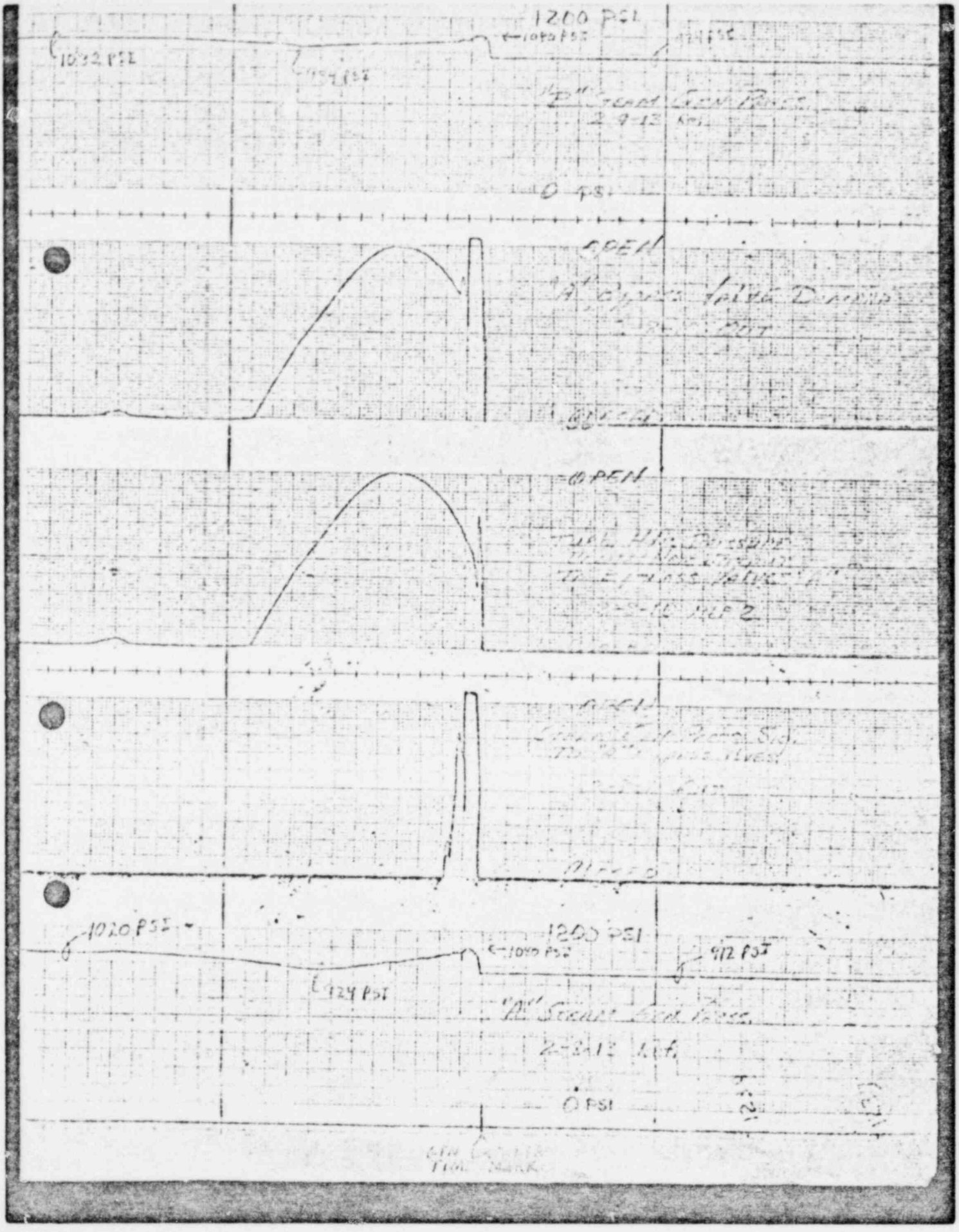

100% position
0% position
6X10⁶ #/hr


Trip Test From 100% of
1 minute
100%
Trip
Signal To A Fw Feed Pump
12-5-1 Out
0%
+25VDC
Fw Pump Turbine
Trip-4-6 Out
Electrical Noise
-25VDC
12x10⁶#/hr
Fw Flow
100%



GENERATOR TRIP TEST from 100% f


GENERATOR TRIP TEST
1-minute
OPEN
TIME
"D" Bypass Valve Dump
2-9-10 OUT
CLOSED
1200 PSI
1050 PSI
1032 PSI
1024 PSI
1054 PSI
"E" Trip Valve
2-9-13 OUT
0 PSI
OPEN
"F" Bypass Valve Dump
OUT
OPEN
2-10 SLP 2
OPEN


1200 PSI
1032 PSI
124 PSI
OPEN
0 PSI
OPEN
1020 PSI
1200 PSI
912 PSI
124 PSI
0 PSI

(3)

SITE PROBLEM

REPORT TRANSMITTAL

**** CLEARED ****

FILE: 12M2

TO: _____ For Information

Central Engineering Files

C. C. Plunkett - Contract Admin.

S. H. Klein - Quality Assurance

B. J. SHEPHERD - Task Engineer

R. A. GAUERS - Project Manager

CONTRACT NO: 620-00 05

SPR 322 REV. 0

TITLE ICS PERFORMANCE FOLLOWING GENERATOR-REACTOR TRIP

DATE: 3-12-76

The attached, cleared SPR is submitted for your information.

TO: _____ E. L. Logan - FLORIDA _____

_____ L. C. Rogers - MET.ED. _____ R. W. WINKS

_____ R. J. Baker - TOLEDO _____ L. M. KOLONAY

_____ B. L. Day - Intl. Support _____

_____ P. E. Perrone - OFR _____

_____ J. L. Donnell - OFR _____

Attached is one copy of Site Problem Report No. 322 which was processed on Contract 620-00 05. Future contracts have been reviewed for the potential of a similar problem. This problem is/~~is not~~ considered applicable to other contracts 2 → 14.

REMARKS: _____

cc: G. M. Jacks - Plant Integration

This SPR has been reviewed IAW NPG-1707-01

Chris C. Lochard

NUCLEAR SERVICE SUPPORT ENGINEER

CLEARED

NS-21091-2 (9-73)

SITE PROBLEM REPORT

BABCOCK & WILCOX

CUSTOMER MET ED	CONTRACT NO. 620-0005	SPR NO. 322	REV. NO. 0
VENDOR BWCo	P.O. NO.	TASK NO. 21	GROUP NO. 01	SEQ. NO. 01
SITE ENGINEER S. P. MAINGI	REQ'D. RESOL. DATE	REQ'D. COMP. DATE	

TITLE
ICS PERFORMANCE FOLLOWING GENERATOR-REACTOR TRIP

DESCRIPTION OF PROBLEM
#1 Following a Reactor Trip the pressurizer level goes as low as 40 inches.
#2 The Reactor Trips on high RC pressure within few seconds, following a Generator/Turbine trip at 100% power.
Per customer these situations are unacceptable. See BWCo. problem report attached.

STATUS - ACTION TO DATE INCLUDING PERSONS CONTACTED
R. Winks of Control Analysis and L. H. Kolony of Engineering are made aware of the problem.

FURTHER ACTION RECOMMENDED BY SITE PERSONNEL #1 Issue field change covering Turbine Bypass Valves control signals following the reactor trip should be from OTSG's.

#2 To review recommendations in BWCo problem report, specially elimination or relaxation of BTU Limits and lowering settings on last two banks of safety valves.

S. P. Maingi	9/25/74	L. C. Rogers	9/30/74

RESOLUTION 1) FC-174
2) Letter, L.C. Rogers to J.G. Herbein dated 5-27-75 (attached)

APPROVED BY	SIGNATURE	DATE
N S SUPPORT ENGINEER	Miles VanBlaricom	2-2-76
TASK ENGINEER		2/2/76
OTS manager		2/17/76
PROJECT MANAGER		2/6/76

COST CATEGORY ☐ NORM ☐ C ☐ D ☐ G ☐ L ☐ VENDOR CLAIM

AUTH CHARGE NO | ☐ FIELD CHANGE REQ | FC NO

COMPLETION

SITE COMPLETION REPORT

SEE ATTACHMENT

☐ RECOMMENDED STDS. CHANGE

DEVIATIONS ☐ NONE ☐ SEE SPR REV NO. _____

DATE COMPLETED 1/19/76 | SIGNED BY S. P. Maingi

S.O.M. CONSTR. REP. APPROVAL L. C. Rogers | DATE 2/19/76

FINAL DISTRIBUTION
PROJECT MANAGER
S.O.M. CONST. REP.
QA DOC. FILE
CENT. ENGR
FILE 12N 2

SITE COMPLETION REPORT:

1. Field change 174, changing Turbine Bypass Valve control to CPSI Pressure rather than Turbine header pressure has been implemented.

2. New relaxed BTU limits have been incorporated.

3. Relief settings on the two main safety valves have been reduced from 1092 PSIG to 1050 PSIG.

All these steps will help run the reactor back to 15% power on a Turbine/Generator Trip, so the SPR is being closed.

B.W. Wilcox

Babcock & Wilcox

Power Generation Group

P.O. Box [illegible], Lynchburg, Va. 24505

Telephone (804) 384-5111

May 27, 1975

REM-I-104

Mr. J. G. Herbein
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Revised BTU Limits for TMI-I
Reference: REM-I-62, L. C. Rogers to J. G. Herbein dated May, 1975

Dear Mr. Herbein:

B&W Engineering has recently completed work on the revised curves for the BTU Limit circuit for TMI-1 in preparation for plant operation and testing associated with the proposed turbine trip or load rejection test at rated power. Attachments include these curves and an appropriate table.

An analysis of plant operational characteristics during the January 23, 1975, power runback transient has led to a revision of the curves comprising the BTU Limit circuit. Figure 1 (attached) shows the revised curves which can be incorporated into the ICS at the first opportunity. Comparison with the curves of Figure 2 which are currently in use reveals that the steam pressure limit is significantly expanded, whereas the T hot curve is more limiting at lower temperatures and power levels. The feedwater temperature curve was changed to properly represent the effect of feedwater temperature on feedwater flow and steam superheat. Also, reactor coolant flow has been changed slightly. These revised curves are to be utilized for all plant operation from now on and are not only associated with the proposed plant runback tests.

The revised BTU Limit curves were tested on the B&W Old Forest Road PWR simulator for several major transients and the TMI-1 type plant with the new BTU Limits performed very well. When compared with the old curves, the new limits provided greater operating margin at full power.

The development of the revised curves was accomplished by using the B&W certified steam generator computer code and determining the limits for the four parameters which comprise the BTU Limits to exceed or maintain 35°F superheat. After incorporating the curves into the simulator ICS, the following operational transients were incorporated:

(a) ramping power up and down with both 3 and 4 RC pumps operating

(b) tripping 1 RC pump at 75% power level

(c) tripping the turbine at 100% power level

All of these transients were performed successfully without a reactor trip. In addition, transients in which the feedwater flow had to be limited were performed and the control of the feedwater flow by the new BTU Limits was excellent. Sufficient testing of the new BTU Limit curves has occurred and fewer operational problems should develop at TMI-1 since the BTU Limits are less restrictive than the curves presently in use.

The accompanying table presents the specific information which defines each of the four curves in the BTU Limit circuit.

Additional information with regard to lowering the setpoint for pressurizer electromagnetic relief valve, is included and has been reviewed by the Control Analysis Group of Babcock & Wilcox.

> The effect of lowering the setpoint of the pressurizer electromatic relief valve from 2255 to 2205 psig has been analyzed and will flow an additional 56 lbs of steam to the Quench Tank during a very severe transient.

If you have any further questions, please contact me.

Very truly yours,

L. C. Rogers
Resident Engineer Manager

LCR/SFA/ser

cc: J. J. Colitz
J. D. Phinney
K. F. Schmidt
D. B. Tulodieski
R. W. Winks
R. S. Rand

TABLE I

Revised ETU Limits for TMI-1

Steam Generator Pressure

Pressure psig	Feedwater Limit - %
Equal to and less than 1000	106
Equal to and more than 1125	50

RC Flow (Temperature compensated - each loop)

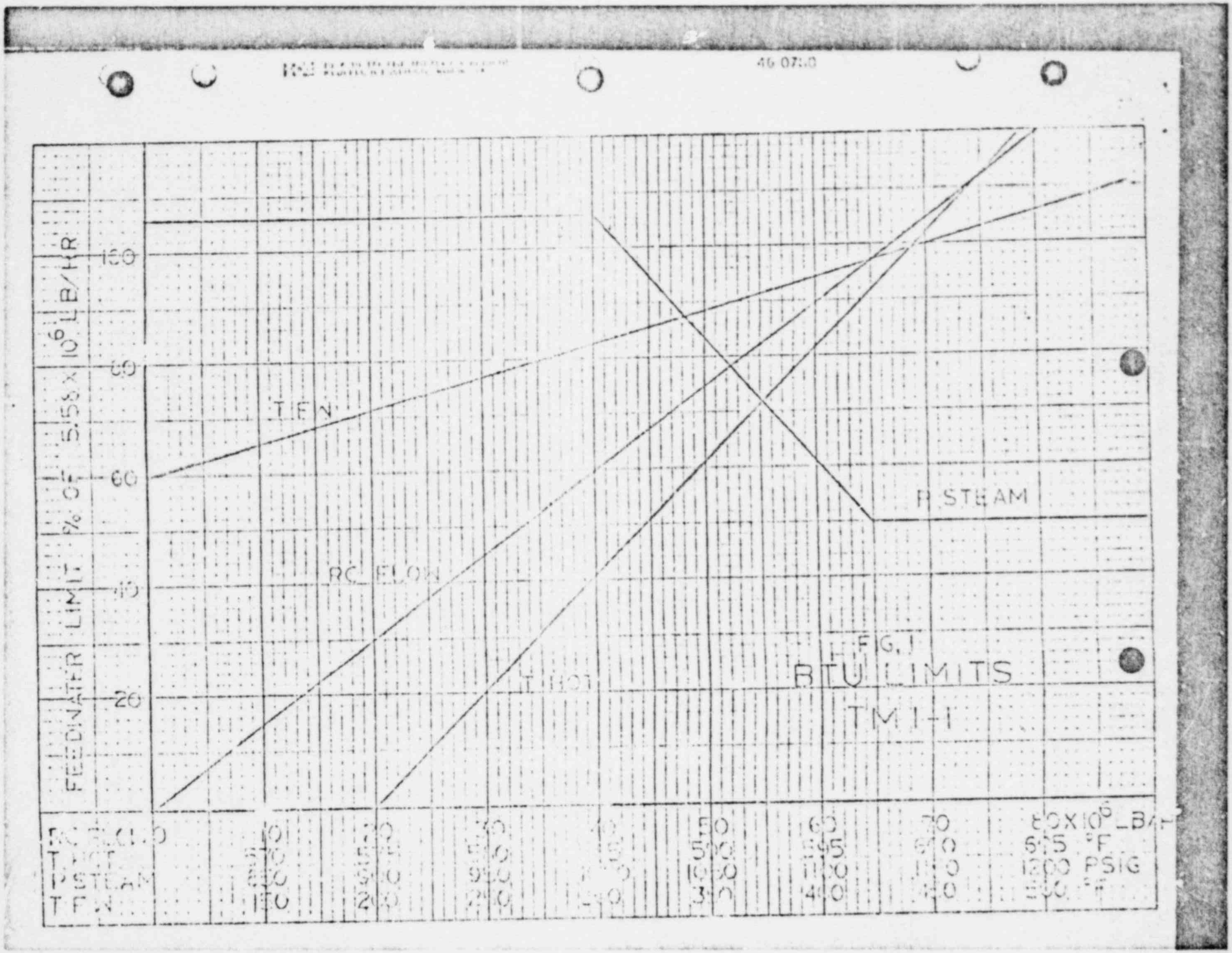
Flow Rate -10^6 lb/hr	Feedwater Limit - %
0	0
80	120

Reactor Outlet Temperature (OTSG Inlet Temperature)

Temperature, F	Feedwater Limit - %
575	0
604	118

Feedwater Temperature

Temperature, F	Feedwater Limit - %
100	60
500	105


FEEDWATER LIMIT % OF 5.53 x 10⁶ LB/HR
100
80
60
40
20
T FEED
RC FLOW
R STEAM
BTU LIMITS
TMI-1
T FEED
T STEAM
60 X 10⁶ LB/HR
515 °F
1800 PSIG
520 °F

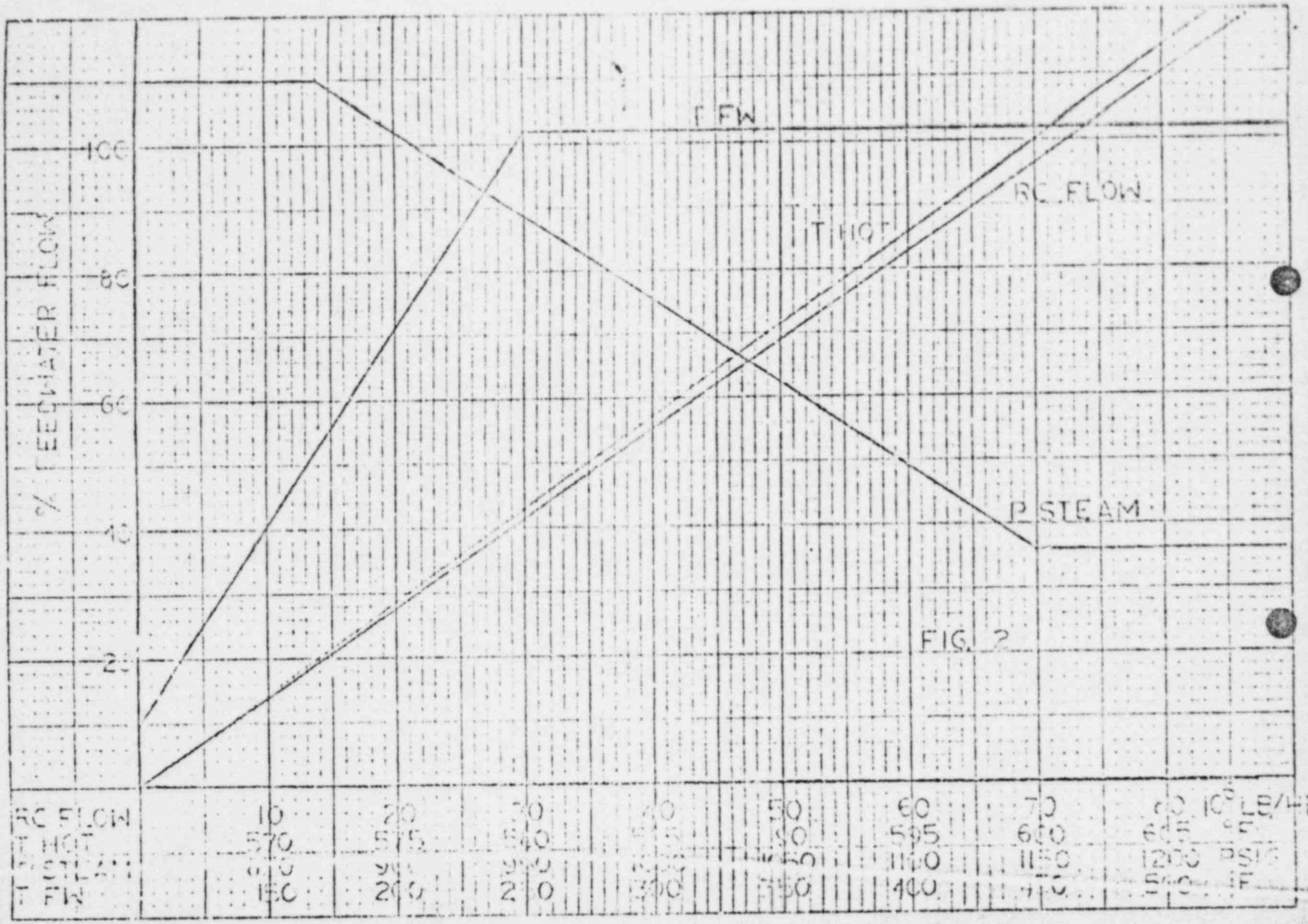

T FW
RC FLOW
T HOT
P STEAM
% FEEDWATER FLOW
100
80
60
40
20
RC FLOW 10 20 30 40 50 60 70 80 10⁶ LB/HR
T HOT 570 575 580 585 590 595 600 605 °F
P STEAM 1000 1100 1200 1300 1400 1500 1600 1700 PSIG
T FW 180 200 220 240 260 280 300 320 °F

FIG. 2

(2)

SITE PROBLEM REPORT

TRANSMITTAL FOR ACTION

FEB 26 1976

RECEIVED

TO: L Rogers For Action

For Action

TO: RW Winks For Information

R Coyers For Information

For Information

For Information

Contract: 620-00 05

SPR Number: 322 rev 0

Title: I & C performance following Hem-Ex trip

Date of Transmittal: 2-17-76

Reply Required By:

Action Requested: L Rogers is requested to complete SPR 322-0 when necessary actions is completed

Reply and Return This Transmittal to: Mike Vanderlinden
Nuclear Service Support Engineer

Reply: SPR was signed QA 1/19/76 and returned to OPR. Am again closing the SPR now that the original finally was returned to the site QA

This problem is/is not considered as applicable to other contracts: NSS-

(Signed)

cc: C. C. Plunkett - Contract Administration
S. H. Klein - NPG Quality Assurance
B. L. Day - Intl. Support
R. J. Baker - Toledo
L. C. Rogers - Met Ed
E. L. Logan - Florida
P. E. Perrone - OPR