SITE PROBLEM

REPORT TRANSMITTAL

RANCHO SECO

**** CLEARED ****

JUL 13 1978

To: Change Control For Distribution
S. H. Klein - Quality Assurance
Central Engineering Files
R.W. WINKS - Task Engineer(s)
J.T. JANIS - Project Manager

File: 13-11-330
Contract No.: 620-00 11
SPR: 330
Title: RAPID COOLDOWN TRANSIENT
Date: MAY 8, 1978
Status Code: C

(DISTRIBUTION - SEE ATTACHED LIST)

Attached is one copy of Site Problem Report No. 330 which was processed on Contract 620-00 11. Future contracts have been reviewed for the potential of a similar problem. This problem is/~~is not~~ considered applicable to other contracts All operating.

REMARKS:

COMPLETED

David H. Ancheson
NUCLEAR SERVICE SUPPORT ENGINEER

CLEARED

1 of 27

7910040556

SITE PROBLEM REPORT — BABCOCK & WILCOX

CUSTOMER	ORIGINATOR / DATE	DOC. ID. CONT. NO.	SPR NO.	REV. NO.
SMUD	D. CULBERSON 5/16/78	13 - 620-0011	330	0

VENDOR	P.A. NO.	PART NO./TASK NO.	GROUP NO.	SEQ. NO.
BAILEY (BMCO)		22	001	001

TITLE (MAX 30 CHARACTERS)	PROBLEM CONTACT
RAPID COOLDOWN TRANSIENT	R. WINKS

PROBLEM IDENTIFICATION

DESCRIPTION OF PROBLEM: 3/20/78 - Plant operating at 70% power (reduced power level due to one main transformer being out of commission).

0425 - While attempting to replace a burned out indicating light on turbine header pressure selector switch, the operator dropped the bulb into the switch causing a short. This short resulted in a loss of power to a portion of NNI System and subsequent loss of many input signals to the control room, computer and ICS. The result was a Reactor trip and rapid cooldown and pressure transient on the primary plant. For details of event, see attached trip report by Bob Winks, March 29, 1978.

STATUS-ACTION TO DATE (INCLUDING PERSONS CONTACTED):

Transient analysis has been performed by B&W Engineering, and SMUD has been given approval to return to power operations. Detailed reports and other correspondence are attached for reference.

FURTHER ACTION RECOMMENDED BY SITE PERSONNEL: Maneuvering limits should be applied for this plant startup; increase surveillance of LPMS for at least one week; perform an operability check of on-line and redundant NNI instrumentation; establish a procedure for restoring NNI power in event of sustained power loss; operator training on loss of NNI power; surveillance of primary and secondary chemistry daily for at least one week.

NSS LOST CAPACITY

RESOLUTION

RESOLUTION:

See attachments. Our other customers will be notified of this transient, by Site Instruction, in the hopes of preventing similar occurrences at other sites. This SPR for information only.

PREPARED BY	DATE	APPROVED BY	DATE
DH Culberson	5/16/78	[illegible]	6/5/78
REVIEWED BY	DATE		
R. [illegible] for H. [illegible]	5/17/78	Joel G. [illegible]	6/2/78

COST CATEGORY	FIELD CHANGE REQ	F.C.A. NO.	SIGNIF. DEFICIENCY
☐ NORM ☐ OTHER	☐ YES ☒ NO	04- NA	☐ YES ☒ NO

COMPLETION

SITE COMPLETION REPORT:

INFORMATION ONLY

DEVIATIONS: ☐ NONE	SPR REV NO. ☐
DATE COMPLETED:	3/24/78
COMPLETED BY: DH Culberson	DATE: 5/16/78

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To: DISTRIBUTION	
From: R.W. WINKS - CONTROL ANALYSIS (EXT. 2864)	BDS 663.5
Cust.: SMUD	File No. or Ref. NSS 11/T3.4
Subj.: REPORT OF TRIP TO PLANT REGARDING REACTOR TRIP OF MARCH 20, 1978	Date: MARCH 29, 1978

This letter to cover one customer and one subject only.

On Monday, March 21, 1978, SMUD informed B&W of the circumstances of the reactor trip on the day before and requested permission to return to power. An immediate response was initiated and SMUD began to supply B&W plant data recorded during the transient. This reactor and turbine trip transient was more extreme than any others due to cooling the Reactor Coolant System 300 degrees Fahrenheit per hour. With preliminary plant transient data, personnel worked late Monday night in describing the pressures and temperatures of the primary and secondary systems. Late Tuesday afternoon it was decided that B&W would send Mr. Art Brown and the writer to SMUD to assist them in forming a complete description of this transient and validating the data being used in analyses.

We left Lynchburg at 7:15 p.m. and traveled all night to Sacramento, California. We arrived at their plant at 9:30 a.m. and met Mr. Norm Brock, SMUD Instrumentation and Controls supervisor. Art Brown was to delve into the SMUD design of the 24 vdc power distribution system for the nonnuclear instrumentation, and I was to help define the primary and secondary system pressures and temperatures during the entire transient.

We called John Castanes (B&W) from Sacramento at 12 a.m. prior to visiting SMUD to obtain the latest concerns and specific directions. When we arrived at the SMUD plant, we found that SMUD personnel had already determined which NNI channels were valid to the computer, the control room, and to the ICS. Also, Mr. Norm Brock provided us a copy of his notes of the interview with the control room operators following the transient.

I met Mr. Don Blanchley, SMUD, and he provided us a copy of the Post-Trip Review - a record of selected channels for 15 minutes after the reactor trip.

SMUD wanted to know the cause for the instantaneous reduction in Loop A and B feedwater flow. We analyzed the situation and concluded that the single hot leg temperature was below the range for the "BTu Limits" circuit,and it drove the feedwater demand signal to zero. With plant performance data in hand, we left to start work back at the motel (and to rest). We called Bruce Karrasch, Plant Integration, and John Castanes again before 5 p.m. (in Lynchburg).

Art Brown and I visited SMUD all day Thursday and continued in the development of the sequence of events and operations which was consistent with the recorded valid data. The twenty pages of the alarm typer printout was scrutinized to provide any extra data points. Details on auxiliary feedwater and turbine bypass valve system operation were also pursued. Another set of control room operator interview notes was provided to me by Don Blanchley. He also showed me the plant hourly log which contained valid data on primary and secondary pressures and temperatures at precisely 5 and 6 a.m. on the day of the reactor trip (~4:26 a.m.). We returned to the motel and I began to write a comprehensive report of the reactor trip transient.

On Friday, March 24, 1978 I arrived at the plant at 8 a.m. and gave my report to Norm Brock and Don Blanchley to review. A copy is included with this report as Attachment #1. Together, we could not resolve why Loop A auxiliary feedwater control valve should have a 100% open demand on it and the Loop B auxiliary feedwater control valve have a 0% open demand (and neither steam generator level was changing). Finally, we hypothesized a set of conditions and decided to go into the control room (plant was at hot shutdown conditions ~ 532 F) and test it.

When the NNI power was lost, both startup level indications to the ICS became zero volt signals. Also, when the 24 vdc powered buffer amplifiers lost power, they decayed either up or down but not together, so Loop A signal dropped to -10 vdc in about six minutes while Loop B signal increased to +10 vdc and maintained the 0% demand on the Loop B auxiliary feedwater control valve. That caused the "A" steam generator to start filling before "B". The actual filling of the OTSG's was accomplished by the SFAS system using parallel valves which were motor-operated and wide open.

Before leaving SMUD for Lynchburg, I made schematics of the auxiliary feedwater system, the turbine bypass valve control system,and the main feedwater flow control system. I prepared the appropriate sequence of operations of each of the systems, and it is shown as Attachment #2 of this report.

On Friday I learned that SMUD was able to hold the primary system parameters very nearly constant over a 9-minute period following the reactor trip by a wide-open primary system safety relief valve. As RC pressure dropped it reseated, and the additional decrease in RC pressure was due to H.P. injection and cooling by the Loop A steam generator.

Representatives from NRC, San Francisco office, visited SMUD and asked quite a few questions of the operators. They also received a description of the transient from SMUD. By Friday, March 24, 1978 B&W and NRC had reviewed the transient sufficiently to give SMUD permission to return to power. By Monday morning the SMUD plant was operating at its temporary maximum limit of 72%.

Robert Winks
R.W. Winks

RWW/dmb

ATTACHMENT #1

EVALUATION OF REACTOR TRIP
TRANSIENT AT RANCHO
SECO ON MARCH 20, 1978

by

Robert Winks, Art Brown
Babcock & Wilcox Co
Lynchburg, Virginia

March 24, 1978

Introduction:

The NSSS designed by B&W allows a specific number of upset transients over the design life of the plant and each customer maintains an accumulative record of each kind of transient experienced by their plant. On March 20, 1978 SMUD subjected the Rancho Seco plant to the most severe reactor trip transient yet experienced at any of the operating B&W plants. The primary and secondary loops were cooled at an excessive rate of 300 degrees Fahrenheit in one hour and the concerns by B&W included the following:

1) excessive stresses in the reactor vessel
2) excessive stresses in the steam generator tubes
3) Single phase fluid conditions in the primary loop for adequate reactor core cooling
4) proper operation and effect on the RC pumps.

As recorded data could be obtained, both SMUD and B&W personnel evaluated the limited available plant parameter data during the reactor trip transient and resolved the above concerns.

Summary:

The reactor trip transient was initiated by a dropped, burned out lamp bulb causing a short in one of the power supplies for non-nuclear instrumentation. The loss of the particular power supply caused a major portion of the measured plant parameters to indicate improperly. The ~~following~~ plant response following a reactor trip was not the usual transient in that the ICS operation was affected by the erroneous input signals and the Control Room Operators had extreme difficulty in determining the status of the plant and operating systems.

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From the transient data supplied by SMUD, B&W was able to ascertain that the last two concerns (satisfactory reactor coolant conditions and proper RC pump operation) were not problems by the morning of March 21, 1978. SMUD transmitted more extensive data which permitted estimates to be made for steam generator and reactor vessel pressure and temperature profiles during the reactor trip transient. The B&W Stress Analysis Group at Mount Vernon, Indiana, determined that the accidental filling of both steam generators with cold water which excessively cooled down both the primary and secondary loops did not cause any major problem with stresses in the reactor vessel or the steam generators. This conclusion was submitted by B&W to SMUD on March 23, 1978.

The severity of the lamp bulb shorting out a power supply is due to a change made by SMUD to the power distribution system design for the non-nuclear instrumentation. Agreement has been reached between SMUD and B&W for an improved redundant power supply design which can cope with the loss of one power supply without affecting the input signals to the Control Room, the ICS, and the plant computer.

Two B&W personnel arrived on site March 23, 1978 to assist SMUD in determining the sequence of events leading to the reactor trip and the resulting behavior of the plant and ICS, and on the validity of the performance data that was recorded. The objectives were accomplished and an interim report was prepared for SMUD on Friday, March 24, 1978.

Recommendations:

1. SMUD should modify the present NNI power distribution system design to a more complete redundant design (agreed upon by SMUD and B&W)

2. SMUD should determine a list of necessary plant parameters for shutting down the plant and ensure that it will be valid and available to the Control Room Operators under unusual or accident conditions. (With SFAS initiated, operators need to know the changing conditions of the plant.)

3. SMUD should increase the usage of the Post Trip Review feature of the plant computer by obtaining the stored plant parameter data and restarting it manually as required during a particular transient.

4. B&W should review operation and controllability of SFAS when initiated for "operational" type transients rather than accident conditions.

SEQUENCE OF EVENTS

The following description is an abbreviated sequence of events leading to and following an automatic trip of the reactor at Rancho Seco on March 20, 1978.

Plant was steady at 70% power level and four RC pumps running.
Operator decided to replace burned out indication light on turbine header pressure selector switch. He dropped small lamp bulb into switch causing a short.

A 24 vdc power supply was current limited and opened two circuit breakers and there was no backup power supply for this portion of the Non-Nuclear Instrumentation System.

Many input signals to the control room, the plant computer, and the ICS indicated either mid scale or zero, or otherwise improperly.

The loop A and B BTU Limits developed a zero feedwater limit signal and shut off feed water flow to both steam generators.

As steam generator levels, dropped, the primary side stored more energy and RC pressure increased to the high RC pressure trip setpoint of 2355 psig tripping the reactor. The reactor trip condition tripped the turbine.

Within 100 seconds after tripping the reactor both main feed water pumps were on a minimum speed of 2200 rpm. Both steam generators were at low level and decreasing, and steam pressure was beginning to decrease.

The auxiliary feedwater pump (steam driven) was automatically started due to low feedwater pump discharge pressure.

Operator noticed that Loop A aux. feedwater control valve had a 100% open demand signal while the Loop B valve had a 0% open demand.

Loop A Steam Generator had higher level than Loop B steam Generator (Full Range levels were only valid indications)

Finally, operator manipulated to gain control of the Loop A feedwater pump and increased speed up to 3500 rpm.

Due to position of feedwater valves, feedwater only entered into A steam generator. Increasing level indication corresponds to rapid cooldown of RC system. B Generator remained "dry".

When RC pressure decreased below 1600 psig initiating SFAS, the electric motor driven auxiliary feedwater pump was started, the motor-operated aux. feedwater valves opened, and feedwater entered both steam generators.

Both steam generators were filled with cold water before the operator stopped feedwater flow to the two generators.

Power was restored to the faulted NNI system and the input signals to the control room, computer and ICS were once again normal. SFAS was bypassed by Operator.

ANALYSIS OF TRANSIENT DATA

Plant Performance data during the reactor trip transient was obtained from the following sources:

1) plant computer Post Trip Review - 15 minutes of valid and invalid channels

2) Alarm typer printout - operating during period of NNI power outage

3) Selected Control Room recorder traces - those known to be valid

4) plant computer hourly log printouts - particularly two records during the power outage period

5) Estimates of pressure and temperature profiles during transient where insufficient data was available.

Figure 1 shows RC pressure and both a hot leg and cold leg temperature for the interval of time (13 minutes) between reactor trip and initiation of SFAS. Data was obtained from the Post Trip Review printout. At nine minutes increased cooldown of the RC system appears to be a result of the operator adding feedwater to the loop A steam generator.

The main concern on RC temperature-induced stresses centered on the 300 F decrease starting at the nine minute mark and continuing until RC temperature reached a low value of about 285F (or lower) in about one hour.

ANALYSIS CONTINUED:

Figure 2 describes the Loop B Steam Generator during the initial 15 minutes of the transient. The data was obtained from the Post Trip Review printout and is limited to only the B steam Generator. Since this generator was boiled dry longer than the A generator, one would suppose that the curves in Figure 2 are more extreme than occurred in the loop A steam generator.

Figure 3 shows a comparison of the levels as measured by the full range level instruments on both the A and B steam generators. At nine minutes the operator was successful in adding feedwater into the A steam generator. ~~through the main feedwater nozzles~~ The status of valves in the Loop B prevented the operator from adding water to the B generator.

Figure 4 displays the primary system pressure and "averaged" pressurizer level indications during the transient before the SFAS was initiated. The primary system was not in trouble during the first 10 minutes of this transient. When the increased cooldown was initiated, then it appear that it was destined to initiate the SFAS.

An Estimate of the change in RC temperature over a 3 hour period is shown in Figure 5. Two data sources were the Post Trip Review and the control room cold leg temperature recorder chart. At 25 minutes after the reactor trip RC temperature was below 520 F and the operator shutdown one RC pump. That is consistent with part of the data obtained from the 5 AM hourly log printout. The actual change in RC ~~pressure~~ temperature is probably less severe than shown as an estimate in Figure 5.

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The change in steam pressure and temperature in one of the two steam generators for approximately 90 minutes following the reactor trip is displayed in Figure 6. The Post Trip Review printout was used to describe the two parameters for the initial 15 minutes and the alarm typer printout and the plant computer hourly log printout were used to support the estimated profiles for steam pressure and temperature. This data is similar to that used by the B&W Stress Analysis Group to evaluate the concern for overstressing the tubes in the steam generator. No steam pressure and temperature data exists for the Loop A steam generator and it is possible to conjecture that the higher level indication and the quicker filling rate would lead to less extreme values of steam pressure and temperature for the Loop A steam generator.

Figure 7 exhibits the approximate change in level for both steam generators versus time after the reactor trip. Post trip Review data describes the initial 15 minutes very well. The only data available after that is the alarm typer stating the time each generator becomes full.

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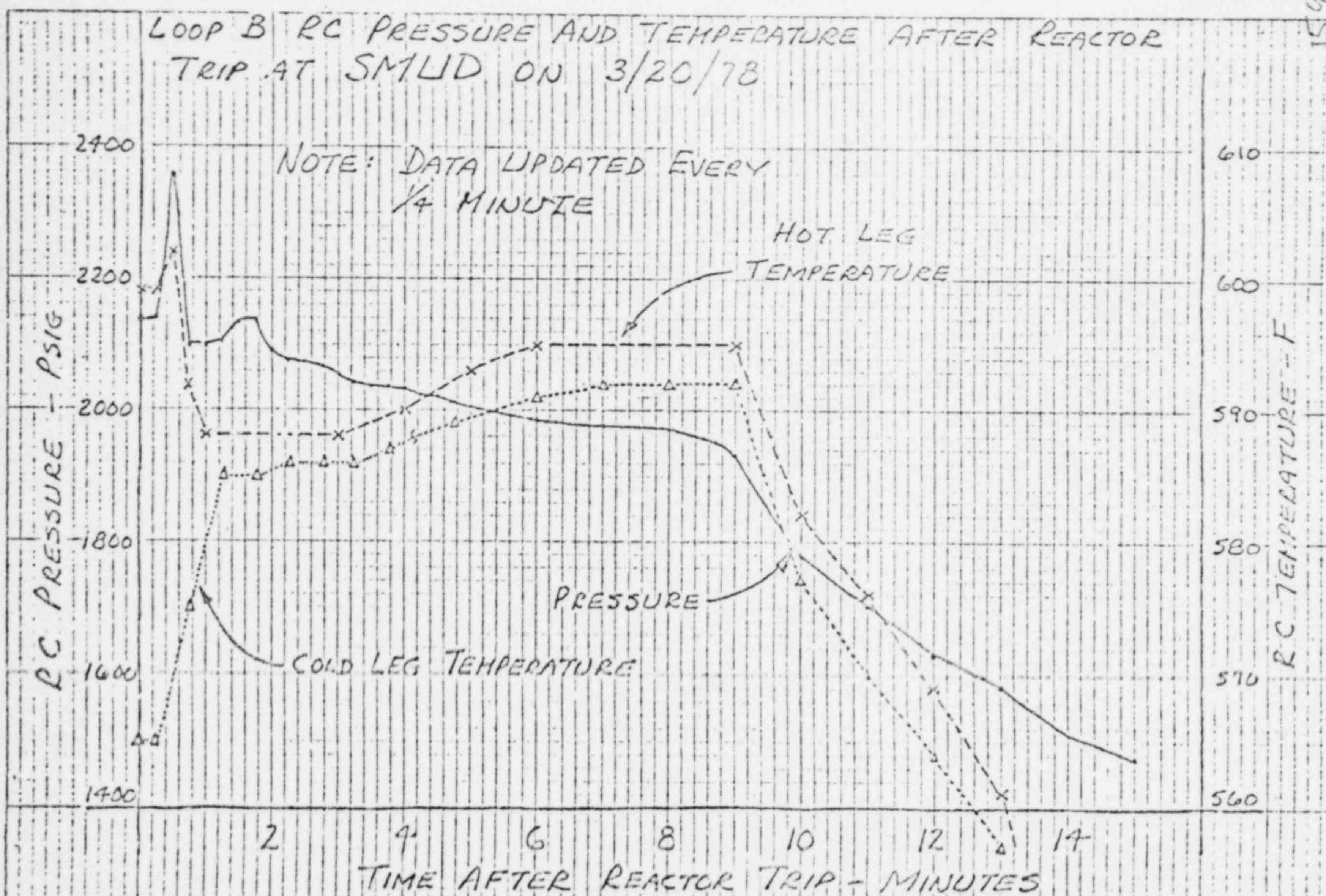

LOOP B RC PRESSURE AND TEMPERATURE AFTER REACTOR TRIP AT SMUD ON 3/20/78
NOTE: DATA UPDATED EVERY 1/4 MINUTE
HOT LEG TEMPERATURE
PRESSURE
COLD LEG TEMPERATURE
RC PRESSURE - PSIG
2400
2200
2000
1800
1600
1400
RC TEMPERATURE - F
610
600
590
580
570
560
2
4
6
8
10
12
14
TIME AFTER REACTOR TRIP - MINUTES

FIG 1.

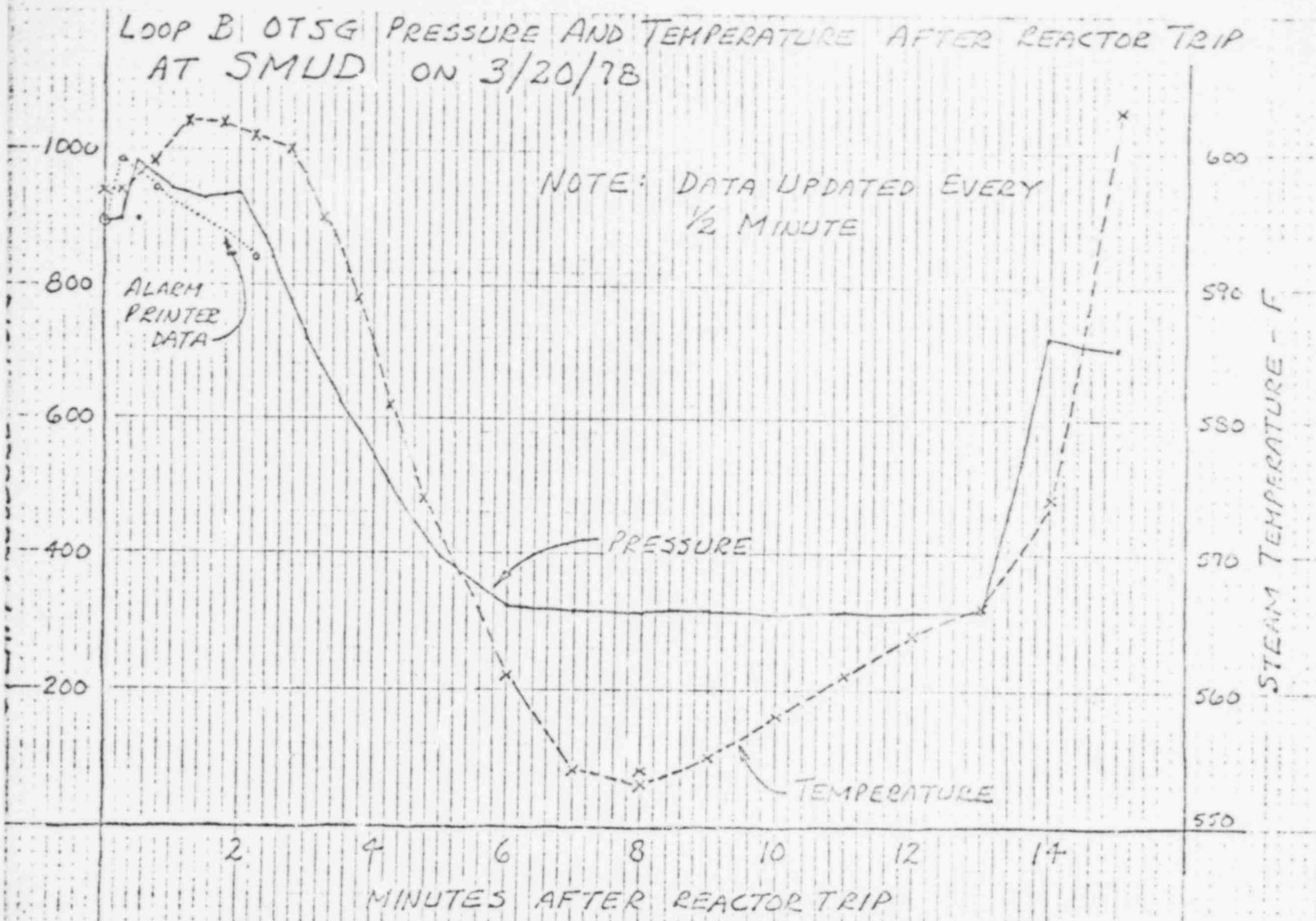
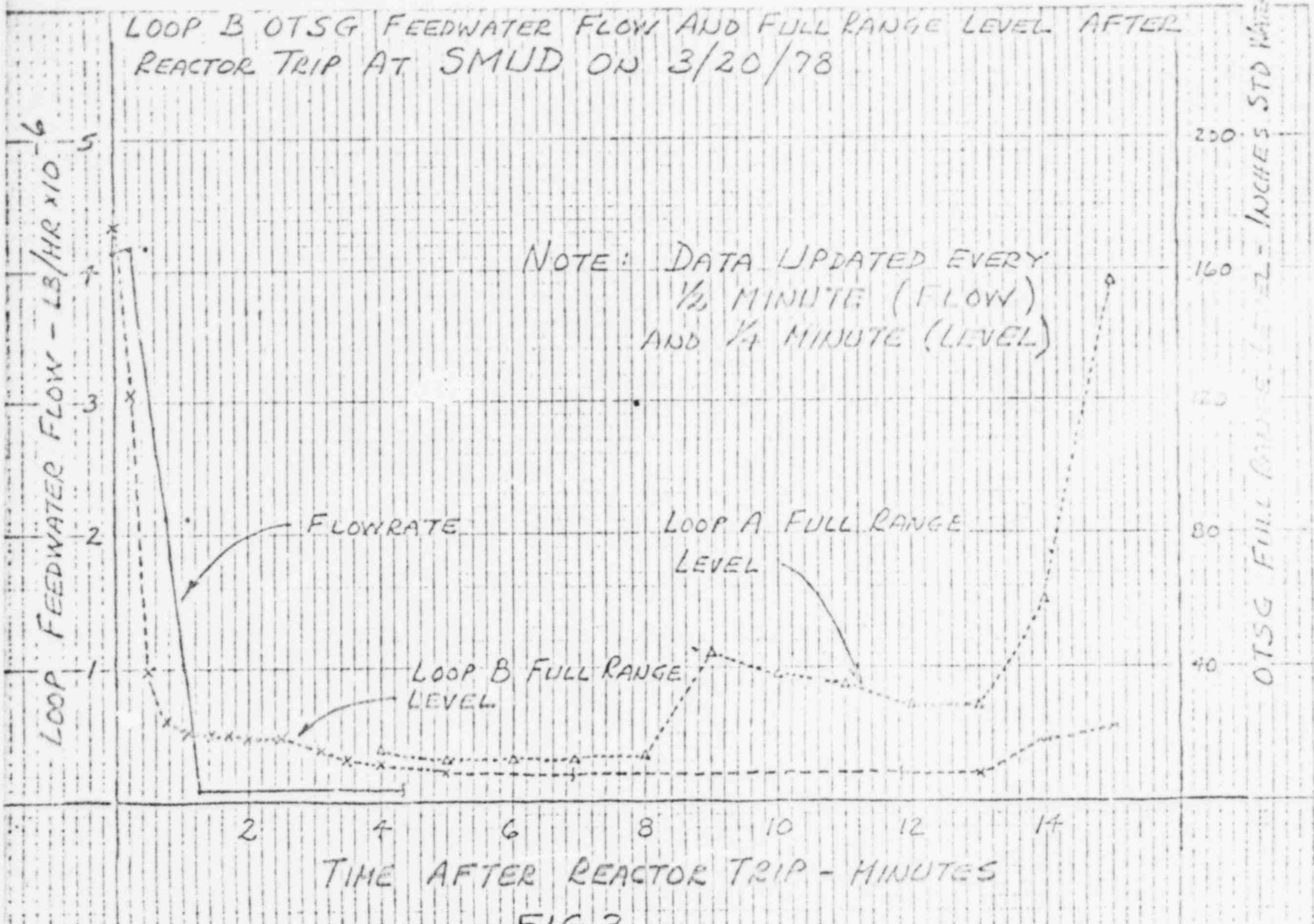

LOOP B OTSG PRESSURE AND TEMPERATURE AFTER REACTOR TRIP
AT SMUD ON 3/20/78
NOTE: DATA UPDATED EVERY ½ MINUTE
ALARM PRINTER DATA
PRESSURE
TEMPERATURE
1000
800
600
400
200
600
590
580
570
560
550
STEAM TEMPERATURE - F
2
4
6
8
10
12
14
MINUTES AFTER REACTOR TRIP

FIG 2


LOOP B OTSG FEEDWATER FLOW AND FULL RANGE LEVEL AFTER
REACTOR TRIP AT SMUD ON 3/20/78
NOTE: DATA UPDATED EVERY
½ MINUTE (FLOW)
AND ¼ MINUTE (LEVEL)
FLOWRATE
LOOP A FULL RANGE
LEVEL
LOOP B FULL RANGE
LEVEL
LOOP FEEDWATER FLOW - LB/HR x10^-6
OTSG FULL RANGE LEVEL - INCHES STD WATER
1
2
3
4
5
40
80
120
160
200
2
4
6
8
10
12
14
TIME AFTER REACTOR TRIP - MINUTES
FIG 3

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RC PRESSURE AND PRESSURIZER LEVEL AFTER REACTOR TRIP AT SMUD ON 3/20/78

FIG 4

46 0702

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RC COLD LEG TEMPERATURE AFTER REACTOR TRIP AT SMUD ON 3/20/78

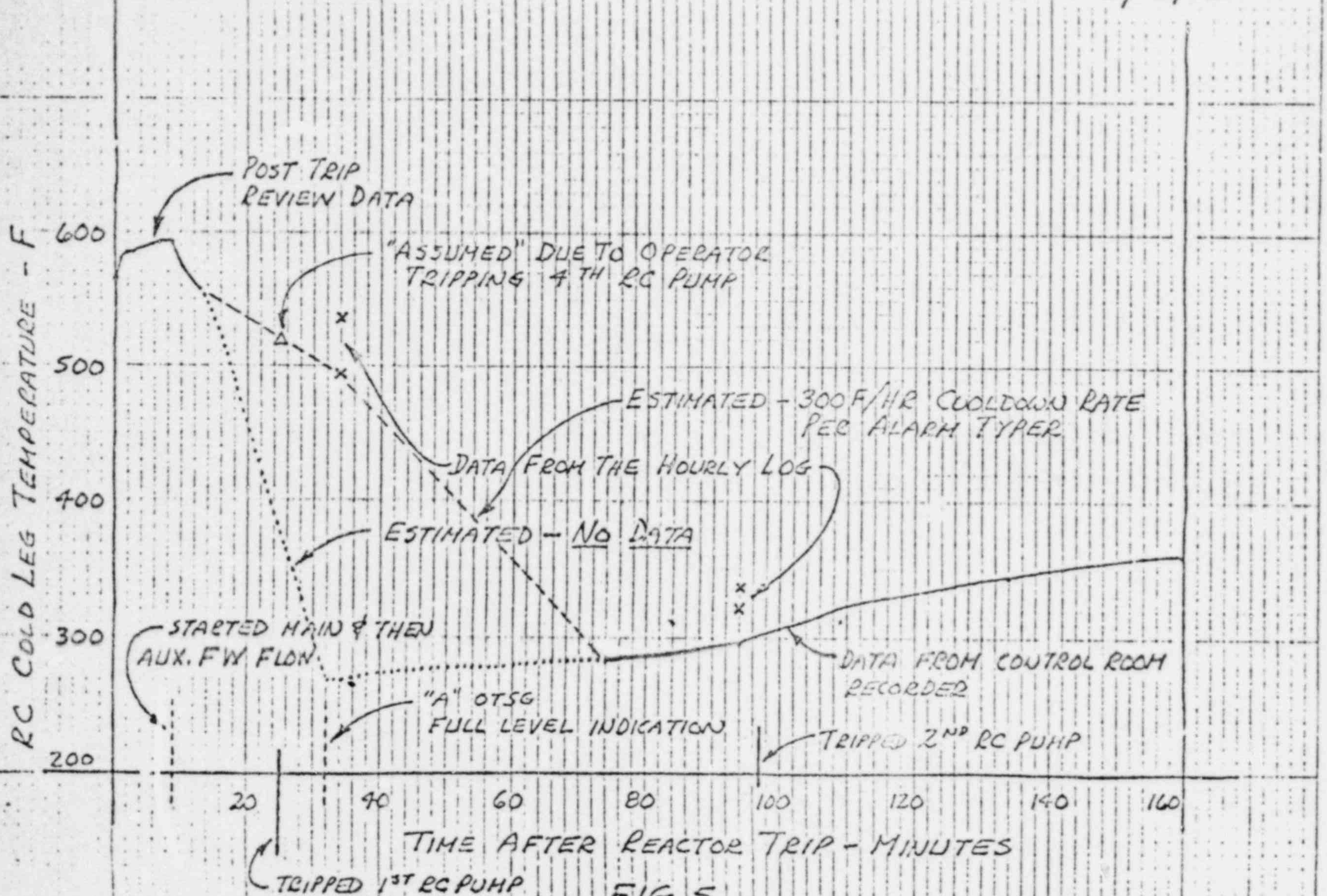

FIG 5

ESTIMATED "B" OTSG STEAM PRESSURE AND TEMPERATURE PROFILES AFTER REACTOR TRIP AT SMUD ON 3/20/78

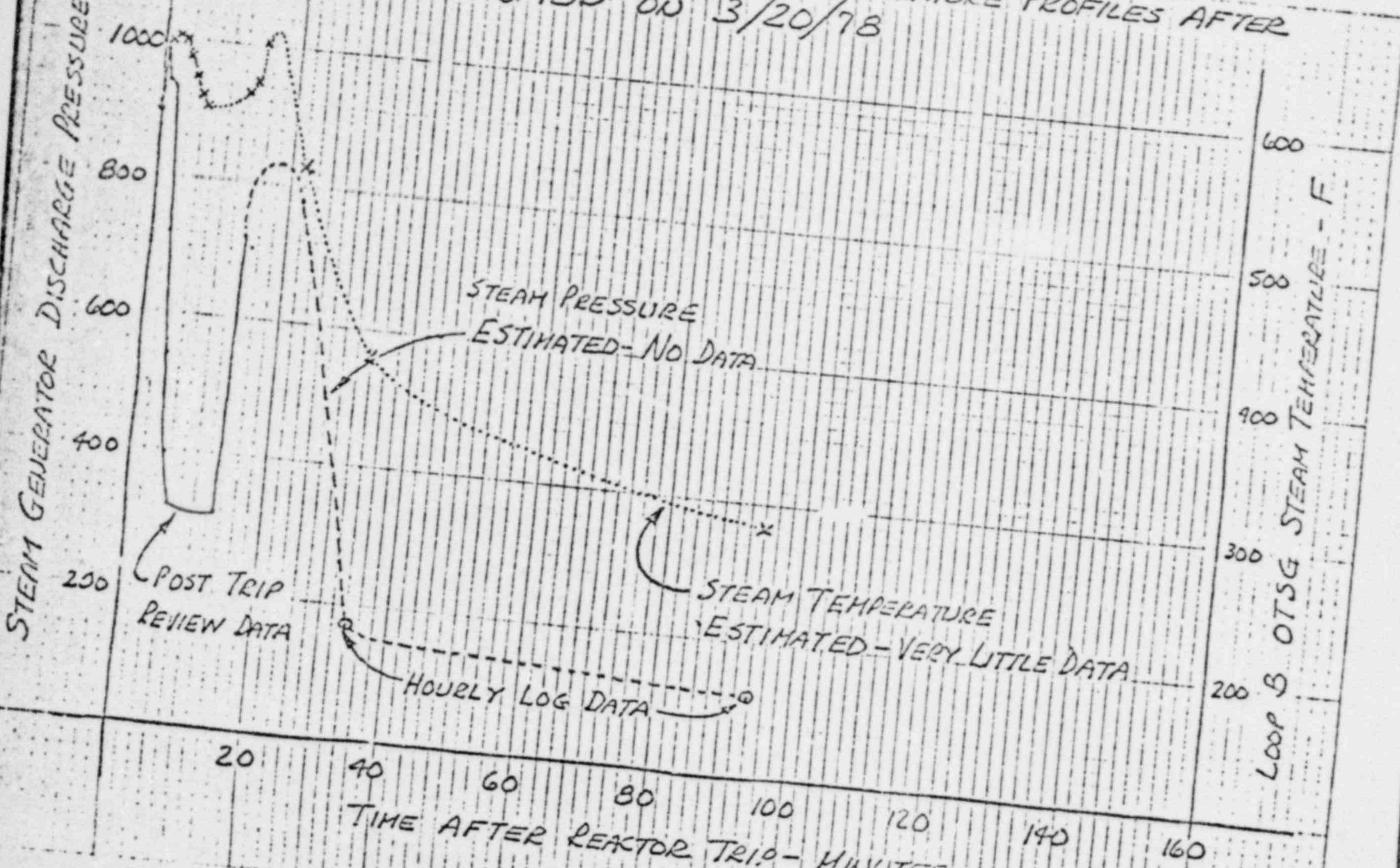

FIG 6

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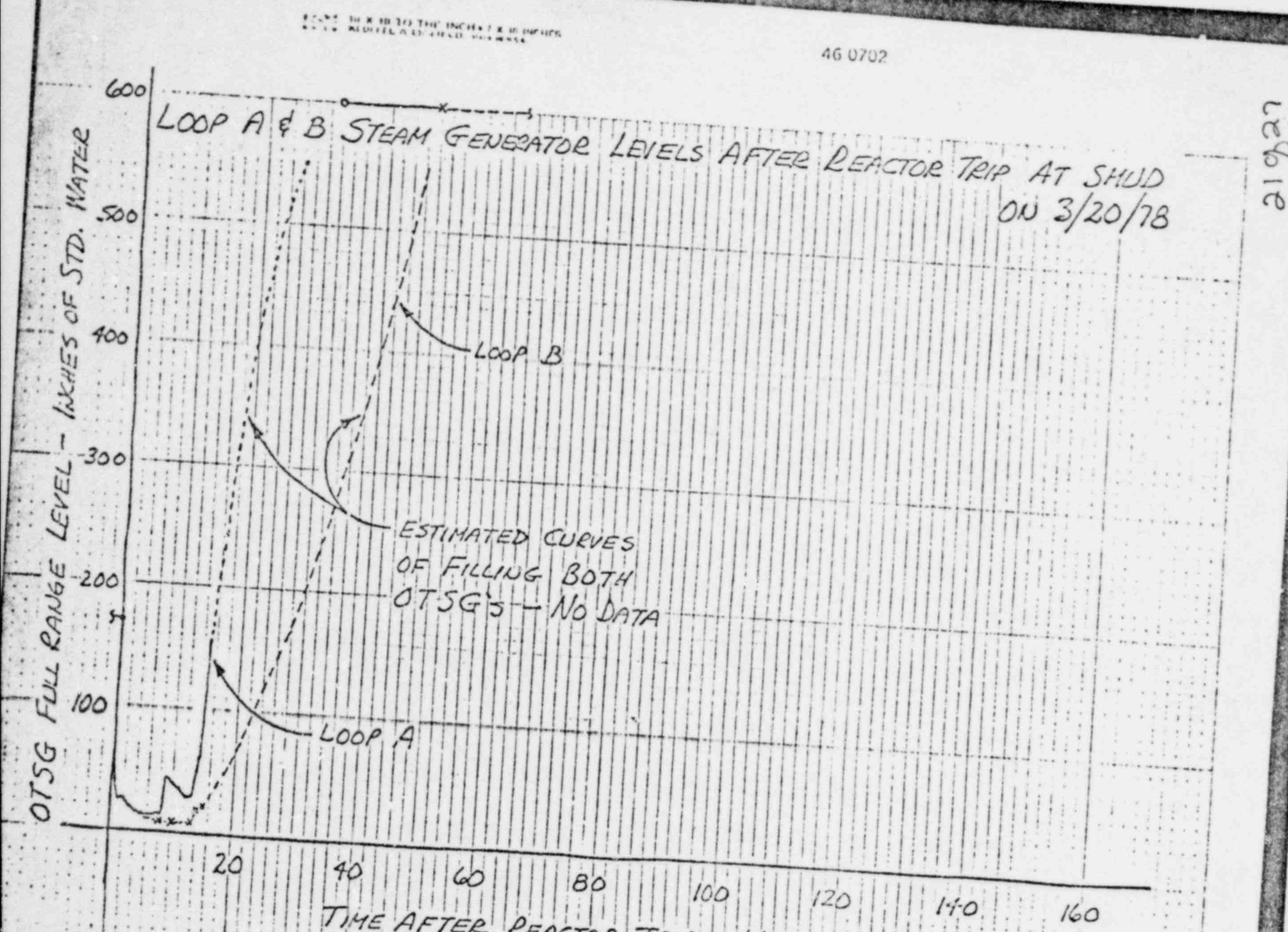


46 0702
LOOP A & B STEAM GENERATOR LEVELS AFTER REACTOR TRIP AT SMUD
ON 3/20/78
OTSG FULL RANGE LEVEL - INCHES OF STD. WATER
600
500
400
300
200
100
LOOP B
ESTIMATED CURVES
OF FILLING BOTH
OTSG'S - NO DATA
LOOP A
20
40
60
80
100
120
140
160
TIME AFTER REACTOR TRIP - MINUTES

FIG 7

21927

ATTACHMENT #2

Suggested Operation of the Auxiliary Feedwater System During the Reactor Trip Transient

1. The turbine driven (and the motor driven) Aux. feedwater pumps were started about 2 minutes into the transient when the main feedwater pump discharge pressures were less than 800 psig.

2. The Loop A and B Startup levels were suddenly improper to the ICS but the buffered signal to Each Aux. Feedwater Valve decayed very slowly and neither Aux. Feedwater valve opened. After 6 ½ minutes the Loop A "Startup level" dropped below the setpoint and the Loop A Aux Feedwater Valve had a 100% open demand signal. By this time the Loop B signal had increased toward full scale and thus generated a 0% open demand signal.

3. At 13 minutes after the reactor trip the SFAS was initiated by RC pressure dropping below 1600 psig. It opened both Loop A and B motor operated valves, See Figure 8, and water flowed to both Steam Generators

4. "A" steam generator filled up in 19 minutes but the "B" steam generator filled up in 35 minutes. Reason why is unknown.

Sequence of Operation of Main Feedwater System During Reactor Trip

1. Upon loss of NNI power supply, selected hot leg temperature signal to ICS dropped to midscale — zero volts or 570 °F. Therefore each loop BTU limits demanded 0% feedwater flow.

2. Main feedwater valves, then Startup valves then Block valves went closed.

3. Loop B Block valve stopped midway due to circuit breaker tripping. Valve was then manually closed.

4. At 9 minutes after reactor trip water was supplied to Loop A steam generator either by:
   (a) the Auxiliary feedwater valve opened automatically (pump was running);

   (b) Operator manually increased Loop A main feed water pump speed from 2200 rpm to 3500 rpm

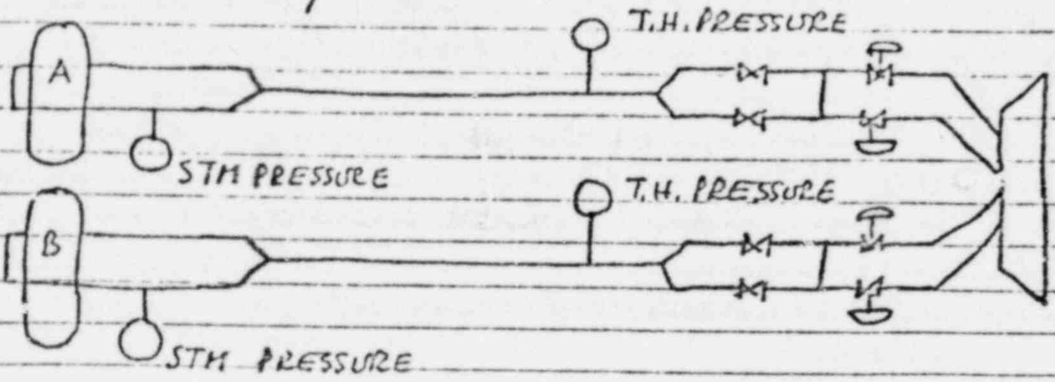
5. All main and auxiliary feedwater valves on Loop B were closed and no water entered the "B" steam generator until the SFAS opened the motor-operated auxiliary feedwater valve at 13 minutes.

Operation of Turbine Bypass System During Reactor Trip

Facts: Steam flow thru the TB Valves was noticeably shorter than other Reactor trips.

When main circuit breakers are opened each loop TB Valve is controlled by the loop Turbine header pressure.

SMUD Steam system schematic:



Loop A T.H. Pressure signal to ICS was Valid.
Loop B T.H. Pressure signal to ICS was bad.

SMUD has 25% steam relief capacity to Condenser.

SMUD has 15% steam relief capacity to atmosphere also.

Loop B steam pressure was measured as high as 984 psig by Post Trip Review.

SCHEMATIC OF AUXILIARY FEEDWATER SYSTEM AT SMUD

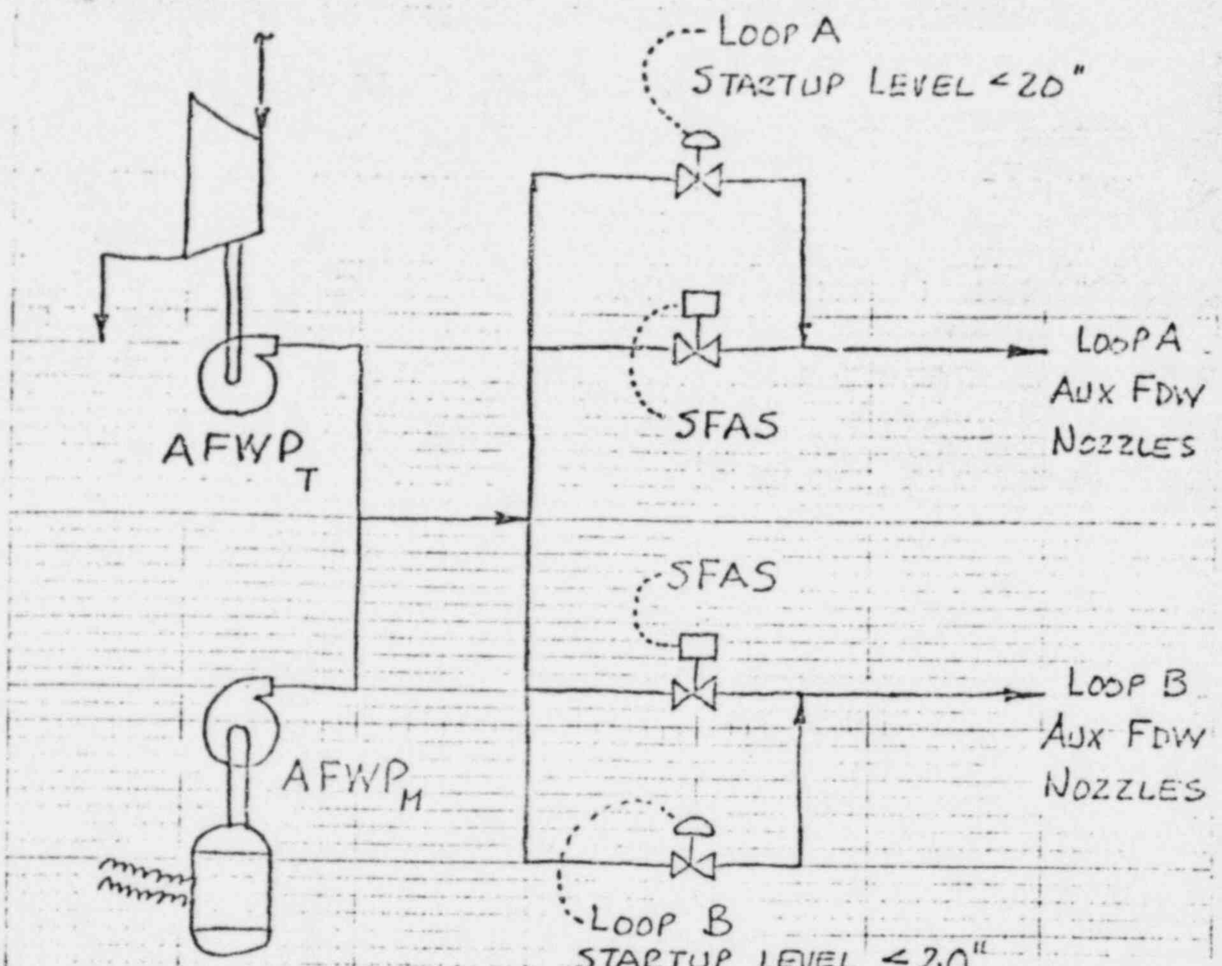


FIG 8

SCHEMATIC OF MAIN FEEDWATER SYSTEM AT SMUD

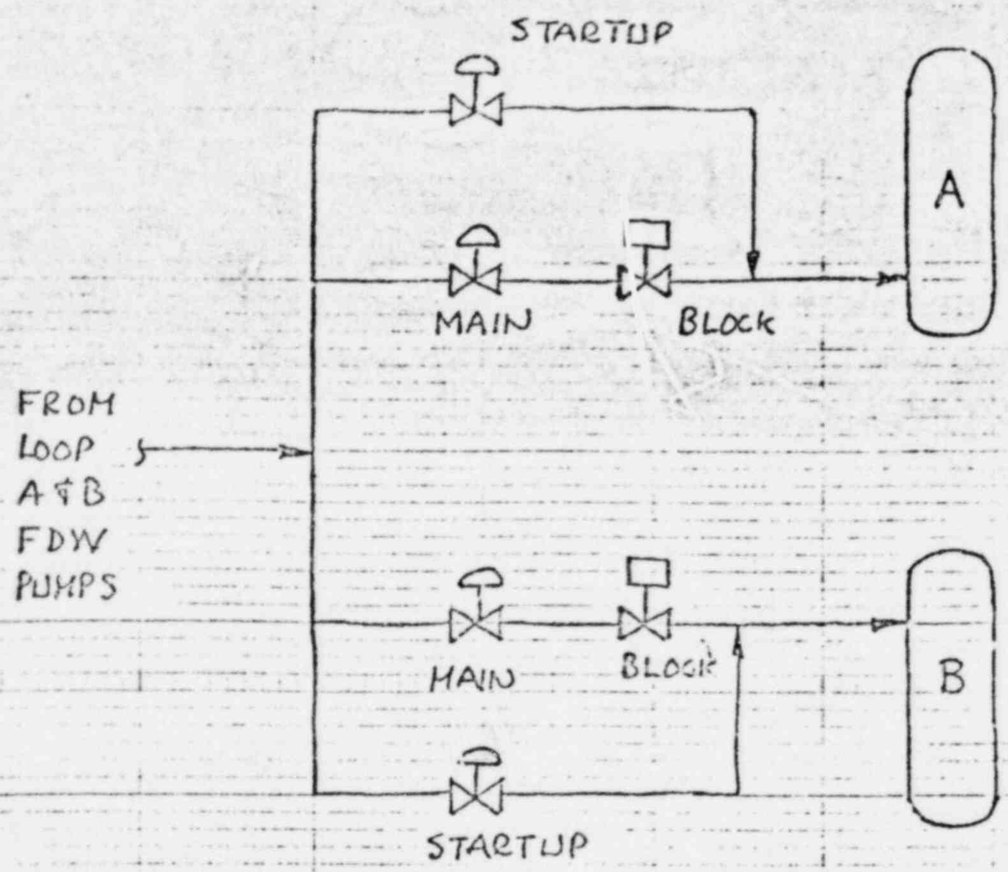
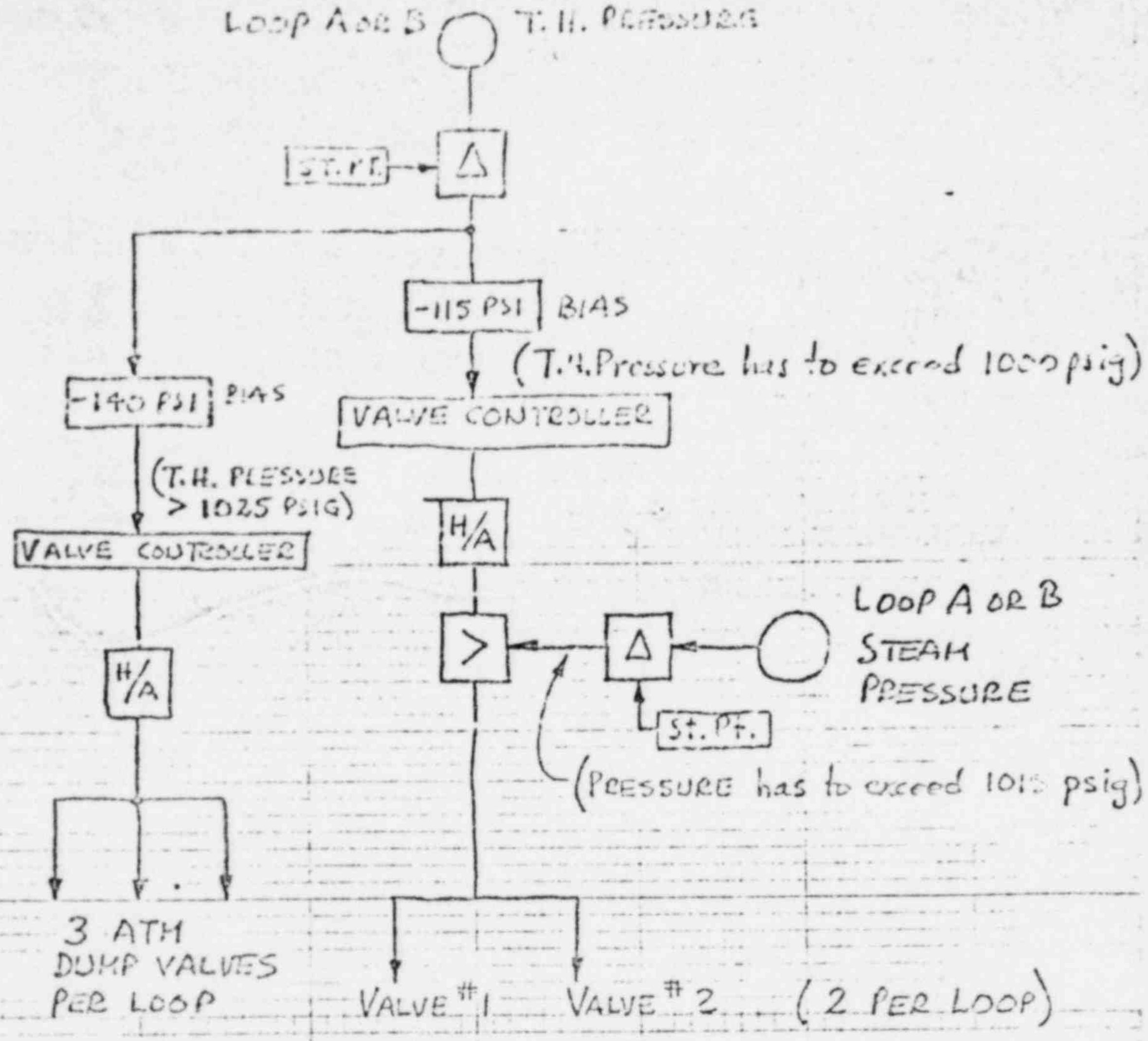


FIG 9

SCHEMATIC OF TURBINE BYPASS VALVE CONTROL SYSTEM AT SMUD



NOTE: ONLY 2 T.B. VALVES COULD HAVE OPENED BRIEFLY (LOOP A) DURING REACTOR TRIP TRANSIENT

FIG 10