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RANCHO SECO
MARCH 20, 1978

JUL 31 1978

186

MEMORANDUM FOR: Paul S. Check, Chief, Reactor Safety Branch, DOR

FROM: Richard Lobel, Reactor Safety Branch, DOR

SUBJECT: SUMMARY OF MEETING HELD AT RANCHO SECO NUCLEAR POWER PLANT ON JUNE 10, 1978 TO DISCUSS A RECENT COOLDOWN EVENT

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On June 20, 1978 I attended a meeting at the Rancho Seco nuclear power plant to discuss an abnormal event which occurred on March 20, 1978. This event resulted in a cooldown of the reactor vessel in excess of that allowed by the Rancho Seco Technical Specifications, specifically, Section 3.1.2.

An attendance list is given in Attachment 1. An agenda is given in Attachment 2. Attachment 3 is a disucssion of the cooldown event and a summary of the meeting. Attachment 4 is a SMUD internal report on the transient with proposed actions to prevent the reoccurrence of such an event.

Attachment 5 contains my conclusions reached from studying this event.

Richard Lobel

Richard Lobel
Reactor Safety Branch
Division of Operating Reactors

Enclosure:
As stated

cc: G. Zwetzig
M. Chiramal
D. Tondi
T. Marsh
R. Woods
P. Norian

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ATTACHMENT 1

LIST OF ATTENDEES

MEETING ON RANCHO SECO COOLDOWN TRANSIENT

SMUD

Ron Colombo
S. I. Anderson
Lloyd Stephenson
John D. Dunn
Norm Brock
Bob Dieterich
Ron Rodriguez
Donald C. Blachly
John V. McColligan
Pierre Oubre

NRC

Phil Johnson-IE:V
M. Chiramal
John Anderson-Oak Ridge Nat'l. Lab
Richard Lobel
Dom Tondi
Gerald B. Zwetzig
Bob Dodds

B&W

Joel T. Janis

ATTACHMENT 2

AGENDA

March 1978 Reactor Transient

(1) A brief description of the Integrated Control System as installed at Rancho Seco, presented by Norm Brock, Senior Instrument & Control Engineer.

(2) General description of sequence of events prior to, during and following cooldown and a summary of the valid and nonvalid instrument indications available to Operators during the transient, presented by Don Blachly, Associate Mechanical Engineer.

(3) Review of electrical drawings pertinent to event, presented by John Dunn, Supervising Electrical Engineer.

(4) Corrective action (implemented and planned) including a discussion of adequacy, presented by John Dunn, Supervising Electrical Engineer.

(5) Reasons for providing SFAS automatic start of auxiliary feedpumps and consequences if SFAS automatic start is deleted, presented by Stan Anderson, Associate Nuclear Engineer.

(6) "Brainstorm" session on other initiating events which could cause significant cooldown transients.

ATTACHMENT 3

MEETING SUMMARY

The meeting began with a description by Norm Brock of SMUD of the Integrated Control System (ICS) used at Rancho Seco. The Integrated Control System coordinates the reactor, steam generator feedwater control and the turbine under all operating conditions. The ICS maintains the required balance between the reactor, the steam generator and the turbine for coordinated control operation under all conditions. **The ICS is capable of following the load demand over the load range of 5% to 100%** rated power. The ICS consists of five subsystems as shown in Figure 1 attached - Unit Load Demand, Integrated Master, Reactor Control, Steam Generator/Feedwater Control and Turbine Control.

There are four modes of operation of the ICS. These are: Start-up, low load control (5% to 15%), full load range (15% to 100%) and **contingency operation (runback or limit load on certain malfunctions)**.

In the full load range the basic requirement of the ICS is to **match the generated MW with the MW demand**. The ICS does this by coordinating the flow of steam to the turbine and the rate of steam production. The flow of steam is controlled by the turbine throttle valves. The rate of steam generation is controlled by varying the total amount of feedwater and reactor power and also maintaining a proper ratio between feedwater flow and reactor power so that the proper steam conditions exist. The feedwater flow is controlled by the feedwater valves and pumps and the reactor power is controlled by the control rods. The primary flow in the steam generator is constant for all load points so that temperature of the reactor outlet fluid ($T_h$) is a measure of the Btu's available. The reactor inlet temperature (Tc) compared against the outlet temperature would produce an index of the amount of energy transferred to the feedwater. With a fixed primary flow and a given $T_h$ for any load, Tc is affected by the feedwater flow. If more feedwater flows through the steam generator at any given load the Tc will tend to be lower. The values of $T_h$ and Tc vary with load and the average of these two temperatures (Tave) is used as an index of the balance between feedwater flow and the heat available in the primary fluid or reactor power. Tave is controlled at a constant value from 15% to 100% reactor power. There are a number of variables that determine the amount of energy available from the steam generator. These are: reactor coolant flow, reactor outlet temperature, feedwater temperature, steam generator pressure and feedwater flow. If an attempt is made to remove more energy from the steam generator than there is available, a reduction in final steam temperature would occur. To prevent this from happening the demand for each loop feedwater flow is limited by the conditions in that loop. The effect of the limiting values are expressed by:

Btu limit = ($T_L$ + Fwt, + Psg - 200) Rcf,

where $T_L$ = Reactor outlet temperature limit
Fwt = Feedwater temperature limit
Psg [illegible]erator pressure limit
Rcf = Reactor Coolant Flow limit

The feedwater control subsystem (See Figure 2) is designed to maintain a total feedwater flow equal to the feedwater flow demand. Feedwater flow is controlled through the use of feedwater control valves while the pressure difference across these valves is maintained constant through the use of the turbine driven feed pumps. The pressure drop across the feedwater valves is monitored and compared to an adjustable set point and the error signal controls the pumps. In order to anticipate the pressure difference change due to a load change a feed forward signal from feedwater flow demand is added to the pressure difference signal.

The feedwater flow to each steam generator is compared to the loop feedwater demand and the difference, feedwater error, establishes the position of the feedwater valves. In order to provide a uniform sensitivity over the full range of flow control the two feedwater valves (main and start-up) are used.

To prevent the feedwater flow control from reducing the level in the steam generator below a predetermined minimum, a low range level meter output is compared to an adjustable set point and the difference is auctioneered against the feedwater flow error. If the feedwater flow error tries to reduce feedwater to a point where the level will drop below its minimum value, the level error will take control of the feedwater valves. A similar control is imposed at high loads where the level can rise to a maximum value.

If main feedwater pumps fail the minimum steam generator level is maintained by the emergency feedwater pump through the emergency feedwater valve.

The sequence of events during the cooldown transient was next described by Don Blachly of SMUD.

The reactor was initially operating at 72% of rated power at a pressure of 2155 psi. The plant was generating 680 Mw(e). The average reactor coolant temperature was 582 F and the inlet (cold leg) temperature was 520 F . 565 The electromatic relief valve was out of service. (Because the valve had been leaking, it was gagged so that it could not open). One HPI pump was in operation as a make-up pump which is the normal configuration at Rancho Seco.

The plant has two hot legs and four cold legs with two once-through steam generators (OTSG's). The ICS monitors steam pressure from either loop A or loop B. The selection is made manually. Before and during this transient, loop B was selected. The pressurizer is attached to the B loop.

At 4:25 a.m. an operator attempted to change a light bulb in a control panel causing a short to ground which caused the loss of a majority of the indication and control parameters.

Upon loss of one of the two power supplies (designated NNI-Y) the hot leg temperature indication went to zero. This made the term in the first parentheses of the Btu limit equation less than zero and therefore set the feedwater demand equal to zero. This resulted in essentially a termination of main feedwater flow to both steam generators by running both feedwater pumps down to minimum speed and shutting the main and startup feedwater valves. Upon loss of main feedwater flow (as measured by feedwater pump discharge pressure) the auxilliary feedwater pumps started. However, due to loss of function of the ICS, the auxilliary feedwater valves did not open. This was because the steam generator water level indications to the ICS were also rendered incorrect. The indicated water level in the A steam generator began drifting downward while the indicated water level in the B steam generator began drifting upward. The auxilliary feedwater valves did not open until the indicated water level in the A steam generator reached the low level setpoint (20 inches). Thus, there was no main or auxilliary feedwater flow to either steam generator at this time.

This loss of feedwater flow caused the reactor coolant temperature to increase, causing an insurge to the pressurizer which caused the reactor coolant system pressure to rise sharply.

One of the two pressure safety valves opened below its setpoint of 2500 psi providing some pressure relief. At 2355 psi the reactor was tripped by a high pressure trip.* The peak pressure reached was 2425 psi. It is assumed that the safety valve was alternating open-to-closed-to-open (simmering) during the remainder of the transient.

Shortly after trip, the reactor operator initiated flow from a second HPI pump (the makeup pump was still running). This pump uses the Borated Water Storage Tank as a water source.

*The backup trip would have been high hot leg temperature at 619 F. As seen from Figure 3 the hot leg temperature was never above approximately 602 F.

The reactor coolant system pressure then dropped due to opening of the main steam safety valves in response to increased pressure in the steam generators due to loss of feedwater. This provided some cooling to the primary side. The pressure then remained fairly constant for approximately 7 minutes. Both steam generators dried out at approximately 1 minute after initiation of the event. The steam generator level signals are shown in Figure 3.

During this time the reactor operator was trying to determine what instrumentation was available to provide him with reliable indications of the plant's condition. He chose pressurizer level and pressurizer pressure (at Rancho Seco there are no valves between the core and the pressurizer so the pressure transducers are mounted on the pressurizer). At approximately 7 minutes, the water level indication in the A steam generator fell below the low level set point (See Figure 4). This caused the ICS to send a demand signal to the main feedwater pump control valves to provide water to the A steam generator. These valves opened. However, the main feedwater pump was at low flow due to the false hot leg temperature indication and could not respond to the new demand (since the hot leg signal is given priority in the ICS over the level signal). The ICS then called upon the auxilliary feedwater valves to supply water to the A steam generator. The auxilliary feedwater pumps were already operating at this time due to the loss of main feedwater flow signal. Thus, flow from the auxilliary feedwater pumps was sent to the A steam generator.

The reactor operator observed that there was a 100% demand from the A steam generator but 0% demand from the B steam generator. He therefore removed the trip signal from the main feedwater pumps, leaving the main feedwater control valves in automatic. Operation of the main feedwater pumps caused the closure of the auxilliary feedwater valves. The reactor operator was then manually controlling the main feedwater flow to the A steam generator.

Due to the cooldown from the feedwater flow to the A steam generator, the reactor pressure fell to the set point for actuation of the Safety Features Actuation System (SFAS). SFAS consists of two HPI pumps and the auxilliary feedwater pumps. The two HPI pumps and the auxilliary feedwater pumps were now delivering 100% flow (the third HPI pump was still delivering makeup flow). The water source for the HPI pumps was the Borated Water Storage Tank and the water source for the auxilliary feedwater pumps was the Condensate Storage Tank.

Soon after SFAS initiation, the operator took over manual operation of the HPI flow and reduced it from the full flow called for by SFAS. The steam driven auxilliary feedwater pump continued operation, supplying water to both steam generators after SFAS actuation. The motor driven auxilliary feedwater pump was removed (to shed electrical load upon SFAS initiation).

The reactor coolant system pressure and temperature were decreasing at this point. The pressure reached a minimum of 1490 psi.

After approximately one hour, the NNI-Y power supply was restored and the operator became aware that the reactor coolant temperature was only 35 F and he was therefore in violation of Section 3.1.2 of the Technical Specifications.

The operator then took the following actions to bring the reactor into compliance with the Technical Specifications:

(1) spraying the pressurizer to reduce the reactor coolant system pressure,
(2) keeping three reactor coolant pumps operating (operating procedure calls for only three pumps in operation when the coolant temperature is less than 520°F).
(3) shutting off the auxilliary feedwater heaters, and
(4) draining the steam generators.

After this description of the transient John Dunn, Supervising Electrical Engineer, presented a review of the electrical drawings pertinent to the event. I was not involved in this discussion.

After a lunch break John Dunn presented an oral summary of corrective actions, both implemented and planned. These actions are discussed in an internal SUMD document which is Attachemnt 4 of this meeting report.

Among the actions proposed by SMUD is the preparation of a precedure which tells the operator what instrumentation he can rely upon if he loses the Non-Nuclear Instrumentation power supplies.

Stan Anderson of SMUD gave reasons for providing automatic start of auxilliary feedwater pumps along with the HPI actuation safety signal. There are three accidents in which credit is taken for the auxilliary feedwater actuation. These are the Steam Line Break, and the Large Break and Small Break LOCAs.

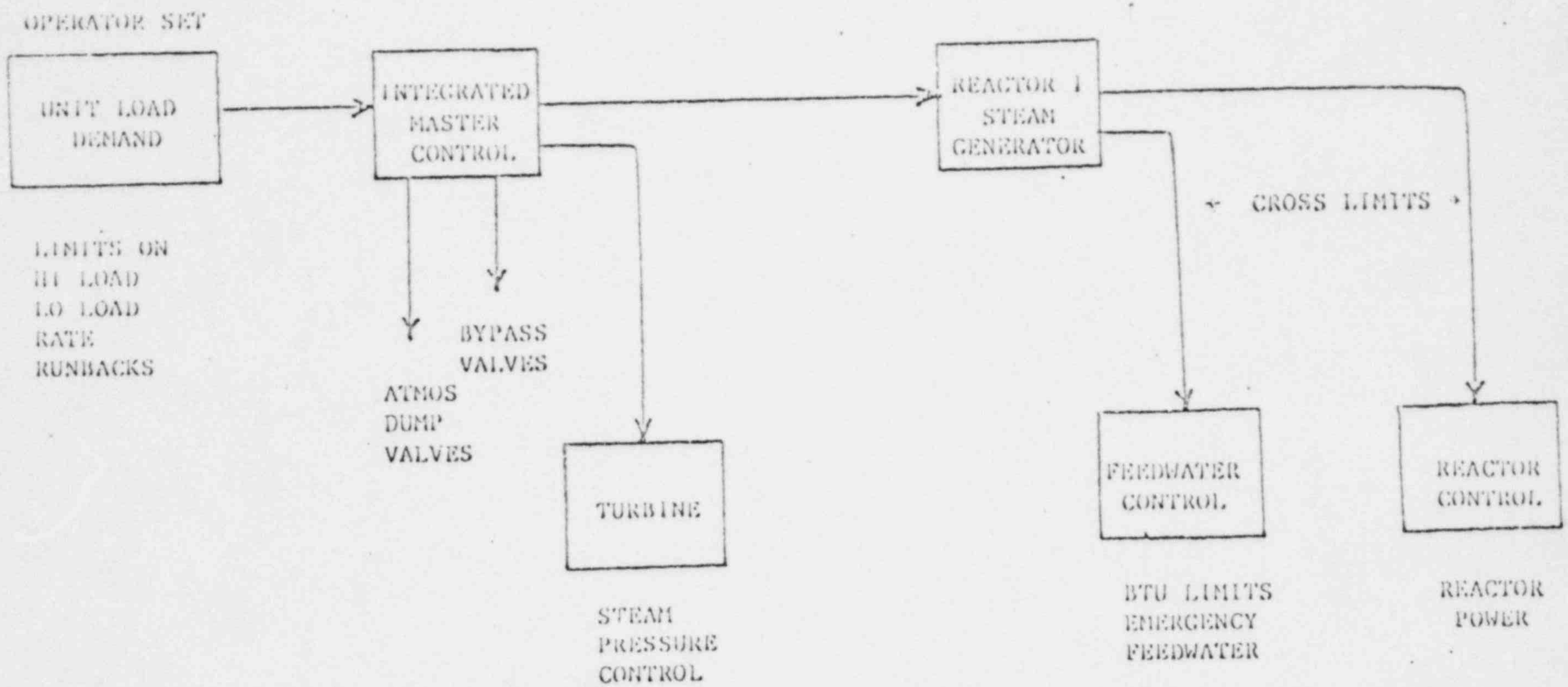
SMUD is asking Babcock and Wilcox (B and W) to evaluate the necessity for automatic actuation of the auxilliary feedwater after HPI actuation. The effect on the Large Break and Small Break LOCAs is thought to be small. The B&W evaluation is expected within two weeks of the meeting date.

NRC raised the question of preservation of shutdown margin during the cooldown transient. SMUD presented calculations which showed that a 1% shutdown margin was maintained during the transient.

The final item on the agenda was a discussion of other possible mechanisms for causing a severe cooldown transient. Depressurization due to a faulty electromatic relief valve or safety valve was the only possibility discussed.

The meeting adjourned at approximately 4:30 p.m. after which the NRC representatives were given a tour of the control room to get a better idea of the location of the equipment available to the operator during the transient. This included the RPS panels which were in a separate room adjacent to the control room.

FIGURE 1
ICS




INPUTS:

STEAM PRESSURE
GENERATED Mw
FEEDWATER TEMP
FEEDWATER FLOW
STEAM GENERATOR LEVELS
RC FLOW (RPS)
REACTOR POWER (RPS)
RC TEMPERATURES, Tc, Th (TAVE)
FEEDWATER PUMP OUTLET PRESS

OUTPUTS FOR CONTROL:

TURBINE VALVES
FEEDWATER VALVES
BYPASS VALVES
ATMOS DUMP VALVES
FEEDWATER PUMPS
CONTROL RODS
EMERGENCY FW VALVES

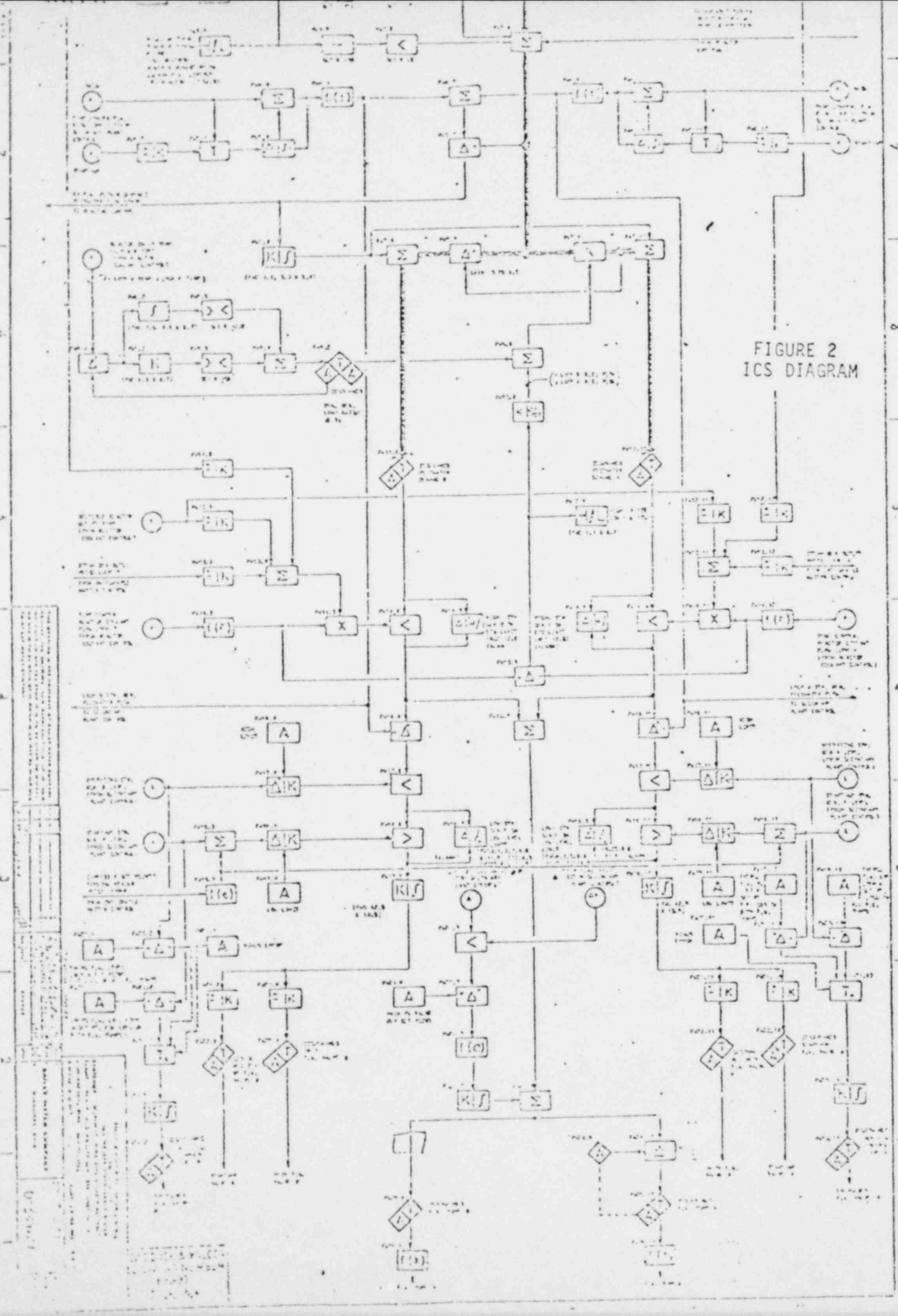


FIGURE 2
ICS DIAGRAM

FIGURE 3

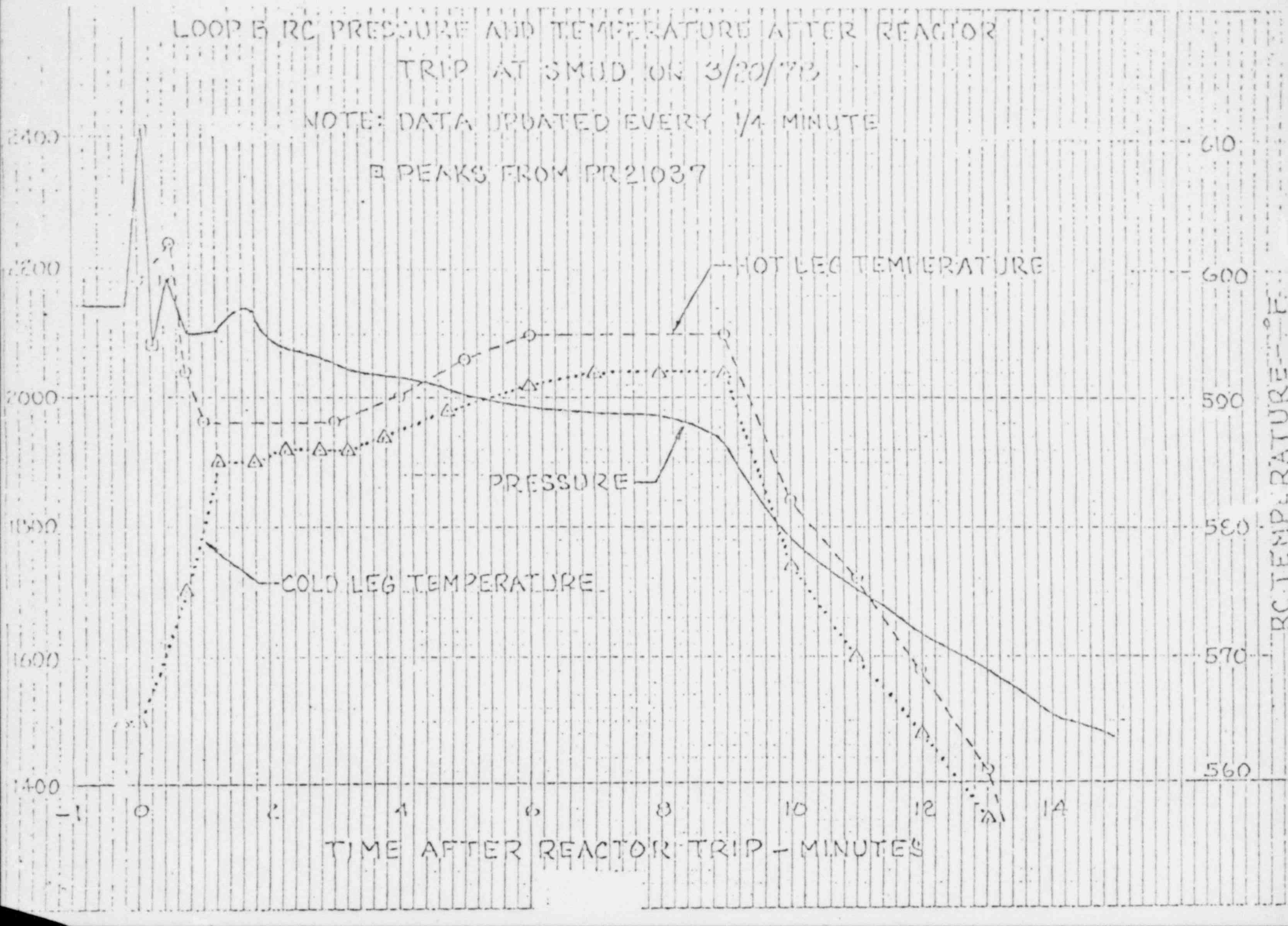
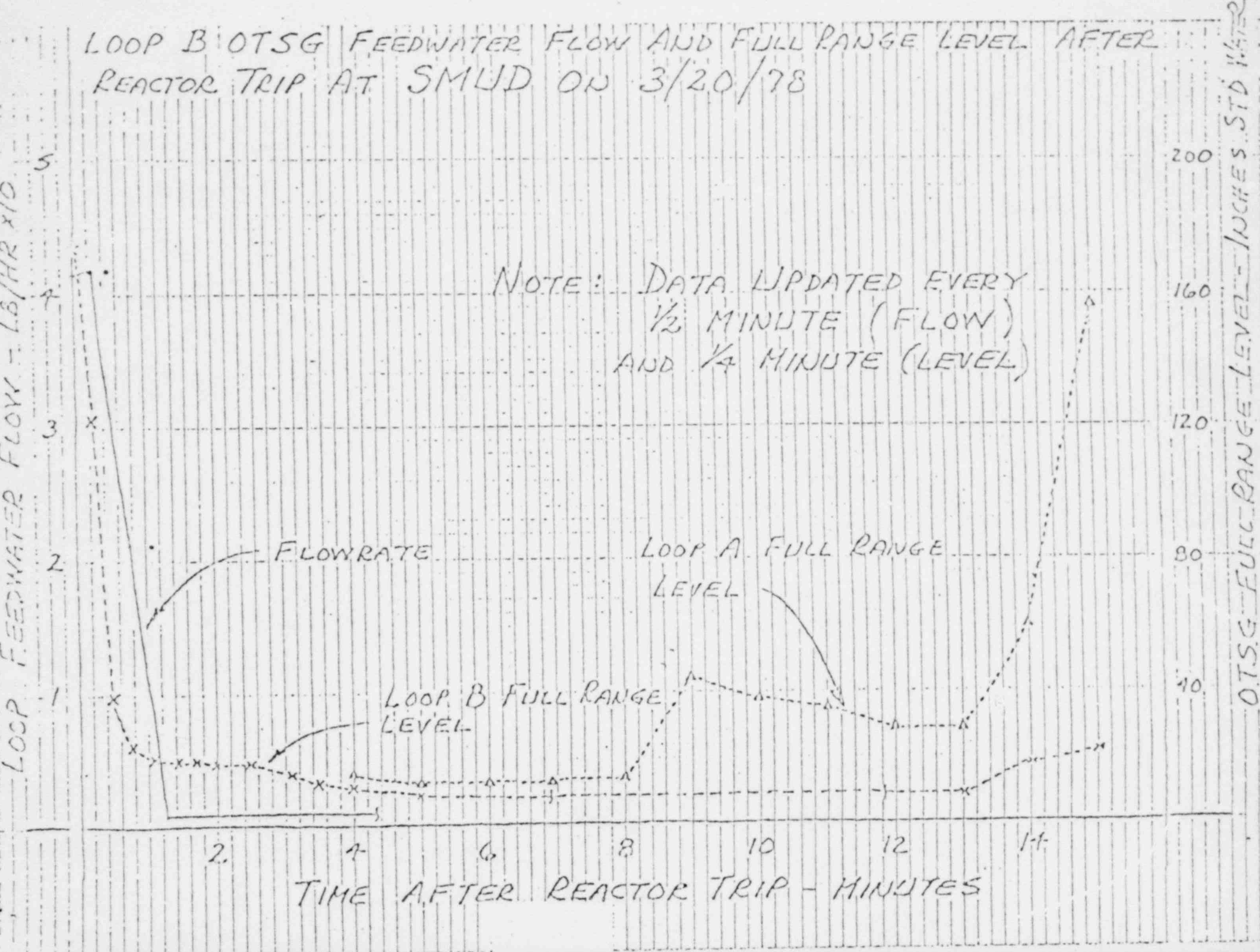

LOOP B RC PRESSURE AND TEMPERATURE AFTER REACTOR
TRIP AT SMUD ON 3/20/78
NOTE: DATA UPDATED EVERY 1/4 MINUTE
B PEAKS FROM PR21037
HOT LEG TEMPERATURE
PRESSURE
COLD LEG TEMPERATURE
2400
2200
2000
1800
1600
1400
610
600
590
580
570
560
RC TEMPERATURE - °F
-1
0
2
4
6
8
10
12
14
TIME AFTER REACTOR TRIP - MINUTES

FIGURE 4



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
1
2
3
4
5
OTSG FULL RANGE LEVEL - INCHES STD H2O
40
80
120
160
200
2
4
6
8
10
12
14
TIME AFTER REACTOR TRIP - MINUTES