Indian Point Unit No. 2
Results of Test Program Following
Modifications to Steam Generator Internals

August 30, 1974

References: 1) Con Edison Letter (W. J. Cahill) to USAEC (J. F. O'Leary) Dated March 12, 1974

2) Con Edison Letter (W. J. Cahill) to USAEC (J. F. O'Leary) Dated January 14, 1974

Introduction

The reports transmitted with References (1) and (2) detailed the sequence of events following the incident of November 13, 1973 which resulted in the failure of the 18" feedwater pipe to Steam Generator No. 22 of Indian Point Unit No. 2. They also discussed the system and equipment modifications that were made to minimize the possibility of recurrence of damaging waterhammer shocks in the feedwater system, and the test programs that were carried out, or planned, to confirm the effectiveness of the changes.

The purpose of this report is to summarize the results of the "Phase III Tests". These tests were briefly described in the report forwarded with Reference (1). They were performed during the period March 16 to April 18, 1974 and the favorable results confirm the effectiveness of the modifications made to the feedwater system piping, feedwater regulating valves and steam generator internals.

Description and Results of Phase III Test Program

The Phase III test program consisted of several tests with the plant in the subcritical and critical condition, as follows:

1. Steam Generator Tests

   To confirm the leak tightness of the welds made during modifications to the steam generator feedwater rings (Figure 1 and 2), hydraulic leak tests of the modified rings were performed with the reactor in the cold shutdown condition. The feedwater rings were filled to overflowing using a motor-driven auxiliary feedwater pump, and the welds were visually inspected for leakage. The results were acceptable for all four steam generators. During these tests, there was a small continuous leakage around the thermal sleeve annulus in the feedwater nozzle, as would be expected from the design of the sleeve.

2. Subcritical Tests

   With the plant in the hot shutdown condition, individual tests were performed on each of the four (4) feedwater systems under simulated conditions, similar to those under which waterhammer shocks were sometimes observed prior to the modifications. The steam generator feedwater ring was initially covered with water to a level well above the tops of the inverted J-tubes. Water level in the steam generator was then lowered by means of blowdown to an extent which insured that the level was below the

bottom of the feedwater ring and the horizontal feedwater pipe entering the steam generator. During the period when water level was being lowered, auxiliary feedwater flow was maintained to the generator to simulate conditions existing after a turbine or reactor trip. When water level reached approximately 6% of narrow range scale, blowdown and auxiliary feedwater flow were stopped for approximately 15 seconds. A motor-driven auxiliary feedwater pump was then restarted with the auxiliary feedwater regulating valves at their normal (50%) preset opening. When water level in the steam generator began to rise perceptibly, the flow was stopped. After 5 minutes, auxiliary feedwater flow was reinitiated until the water level again began to rise. At this point, the flow to the steam generator was stopped once more. Following an interruption lasting 10 minutes, the auxiliary feedwater flow was reinitiated once again and the steam generator water level was raised until the feedwater ring was recovered. The auxiliary feedwater flow rate was increased gradually as the feedwater ring was recovered, up to maximum auxiliary feedwater flow rate.

During each of the tests, plant instrumentation and the special instrumentation system were monitored. This special instrumentation system, as described in Appendix I of the report forwarded with Reference (1), consisted of strain gages, accelerometers, pressure transducers, thermocouples

and scratch pads mounted at appropriate locations on the feedwater piping. In addition, observers were stationed inside and outside of reactor containment to determine if any noise or pipe motion occurred during the tests. The instrumentation data and the observers both confirmed that no waterhammer shocks were experienced during these tests. The tests were run on March 20 and 21, 1974.

3. Power Tests

Following successful completion of tests with the reactor subcritical on March 21, 1974, the reactor was brought to a power level of approximately 7% with feedwater under manual control. This condition simulated that which existed on November 13, 1973, just prior to the incident which resulted in the feedwater pipe failure. Level in the steam generators was lowered until a low-low level reactor trip was initiated. Following this trip, the main feedwater pump was manually tripped and the auxiliary boiler feed pumps were then manually started. There was no indication of any waterhammer shock either on instrumentation readout or by reports from observers stationed inside and outside containment. Level data and feedwater pipe temperature data for Steam Generators 21 and 22 are included for comparison to the curves transmitted in Report No. 2 dated March 12, 1974 (see Figures 3, 4, 5 and 6).

On March 22, 1974, Unit No. 2 was brought to 35% reactor power, with the feedwater system in automatic control. A plant trip was initiated again by allowing level in a steam generator to drop below its low-low point and then subsequently tripping the main boiler feed pumps and starting the auxiliary boiler feed pumps. This simulated the plant trip which occurred on January 29, 1974 when a waterhammer shock was experienced in the feedwater piping to Steam Generator No. 21 (Reference 1 Attachment, Page 3). During the March 22, 1974 test, no waterhammer shock was observed on instrumentation or by observers outside of containment. Steam Generator 21 and 22 data for this trip are plotted in Figures 7 & 8 and Figures 9 & 10, respectively.

On April 18, 1974, the unit was being brought to full reactor power in preparation for scheduled physics tests at 100% power. Special test instrumentation was being monitored as the plant approached 100% power in preparation for a trip, and to observe any potential waterhammer shocks. The plant tripped unexpectedly at approximately 100% reactor power. There was no indication of any waterhammer shock either on special test instrumentation or from reports of observers outside containment. This test constituted the 100% trip test as described in Page 16 of the attachment to Reference 1, and the special test program was therefore considered to be successfully completed. The data for this test appear as Figures 11 and 12 for Steam Generator 21 and Figures 13 and 14 for Steam Generator 22.

Conclusions

The test program that followed the modifications made to the steam generator internals (so-called Phase III testing) was undertaken to assure that waterhammer shocks were not likely to recur. To determine the effectiveness of the modifications in precluding waterhammer shocks, plant conditions that preceeded and probably caused or contributed to shocks experienced in the past were duplicated in this program.

A dropping steam generator water level and the subsequent draining and recovering of the feed ring was recognized as a principal cause of the hydraulic shock experience. The alterations that were made were designed to limit draining of the feed ring despite a drop in steam generator level below the ring.

A second factor that appeared to have contributed to the creation of the waterhammer shocks was the flow rate of relatively cold auxiliary boiler feed water into the steam generator feed ring. Two hydraulic shocks were produced during the Phase II test program which was run at various auxiliary feed flow rates while the reactor was maintained in a subcritical condition. As part of the Phase III test program, many of the plant conditions that were present during the Phase II program were reproduced. No waterhammer shocks were experienced, however, and the data obtained indicated that the modifications that were

made had corrected the mechanism that had caused the hydraulic shocks during the Phase II testing program.

The final part of the Phase III testing program was a series of three reactor trips with the reactor at various power levels. These tests reproduced the conditions of the Phase I test program and the approximate plant conditions under which waterhammer shocks were experienced on November 13, 1973 and January 29, 1974. Again, the data from the installed special instrumentation and from personnel observation verified that hydraulic shocks did not occur.

It was therefore concluded that there was reasonable assurance that Indian Point Unit No. 2 could be operated at all power levels up to 100% power without the risk of damaging waterhammer shocks.

Indian Point Unit No. 2
Modifications to Steam Generator Feedwater Ring
(Typical)

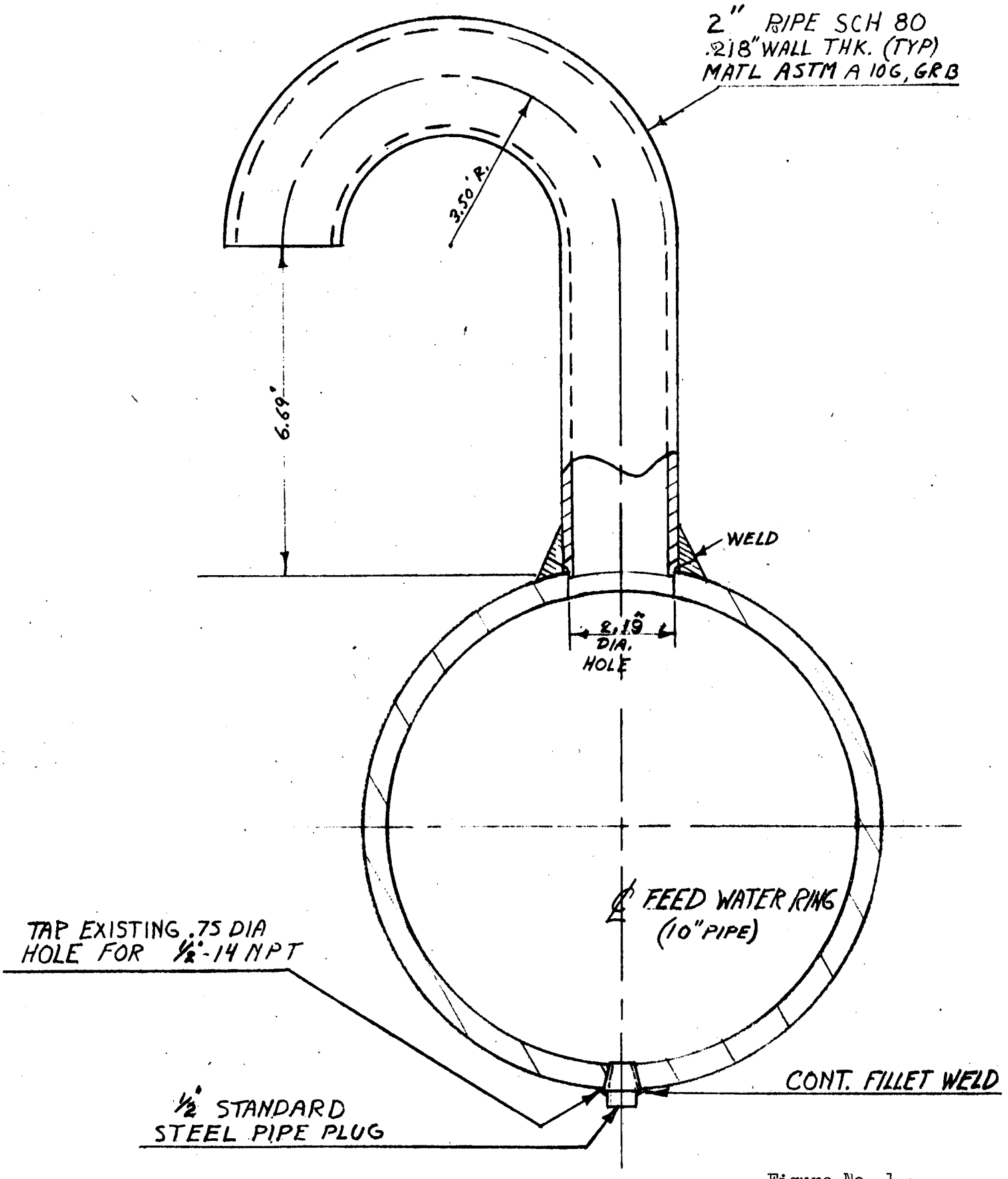


Figure No. 1

Indian Point Unit No. 2
Modification to Steam Generator Feedwater Ring
at 18" Tee (Typical for 2 Locations/Steam Generator)

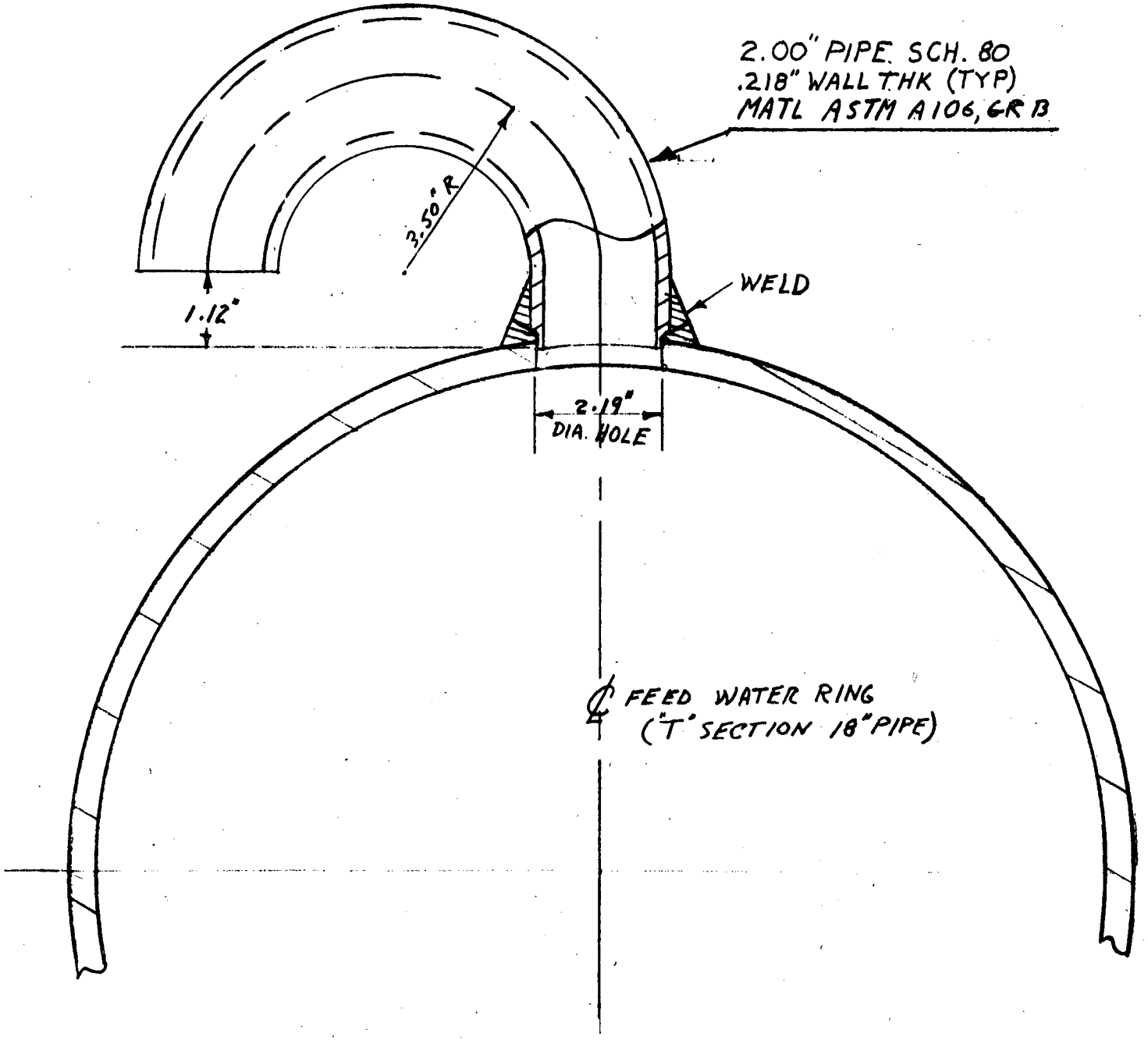


Figure No. 2

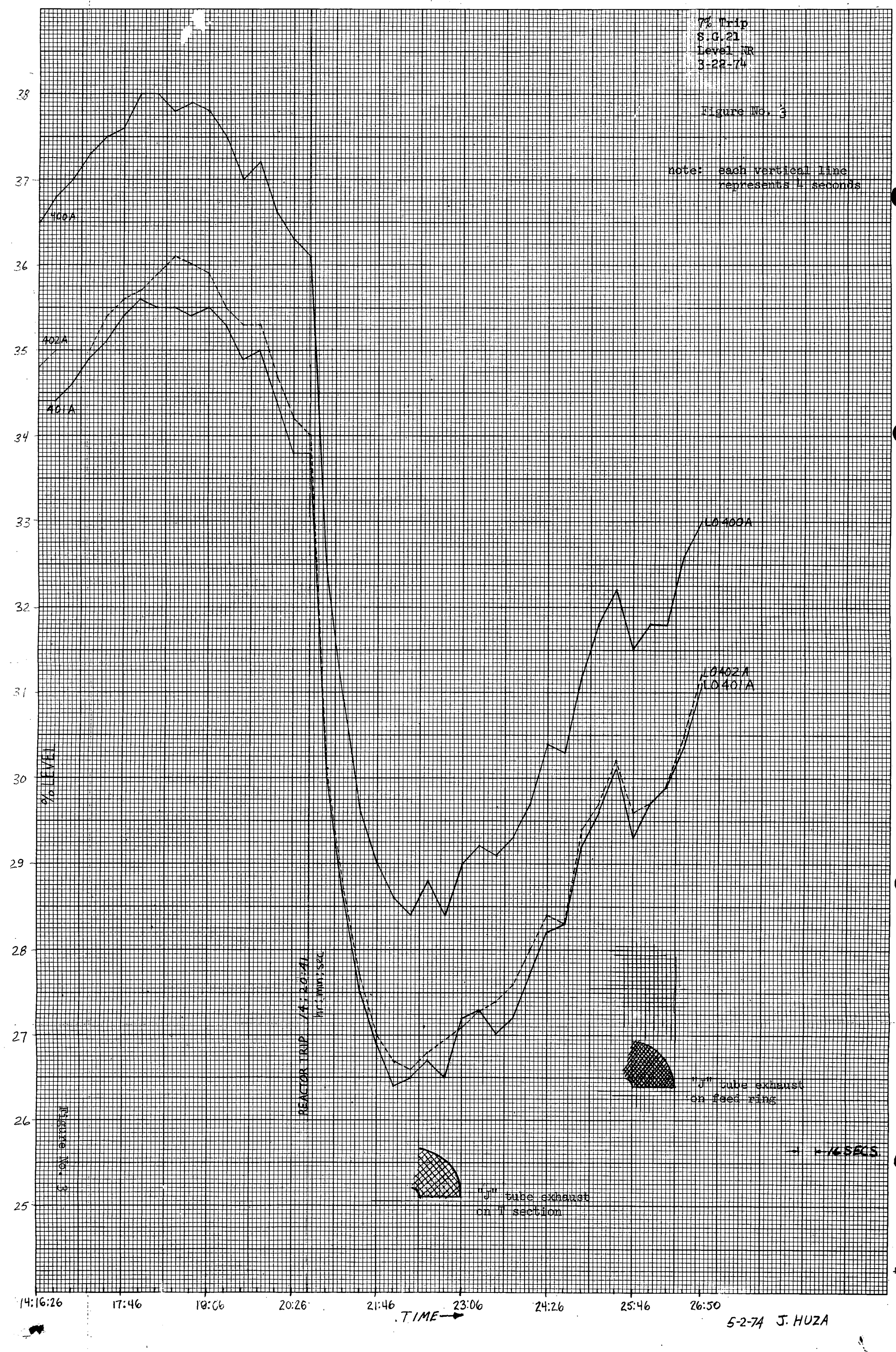

7% Trip
S.G.21
Level NR
3-22-74
Figure No. 3
note: each vertical line represents 4 seconds
400A
402A
401A
LO400A
LO402A
LO401A
% LEVEL
REACTOR TRIP 14:20:41
Hr:Min:Sec
"J" tube exhaust on feed ring
"J" tube exhaust on T section
16SECS
38
37
36
35
34
33
32
31
30
29
28
27
26
25
14:16:26
17:46
19:06
20:26
21:46
23:06
24:26
25:46
26:50
TIME
5-2-74 J. HUZA

Figure No. 3

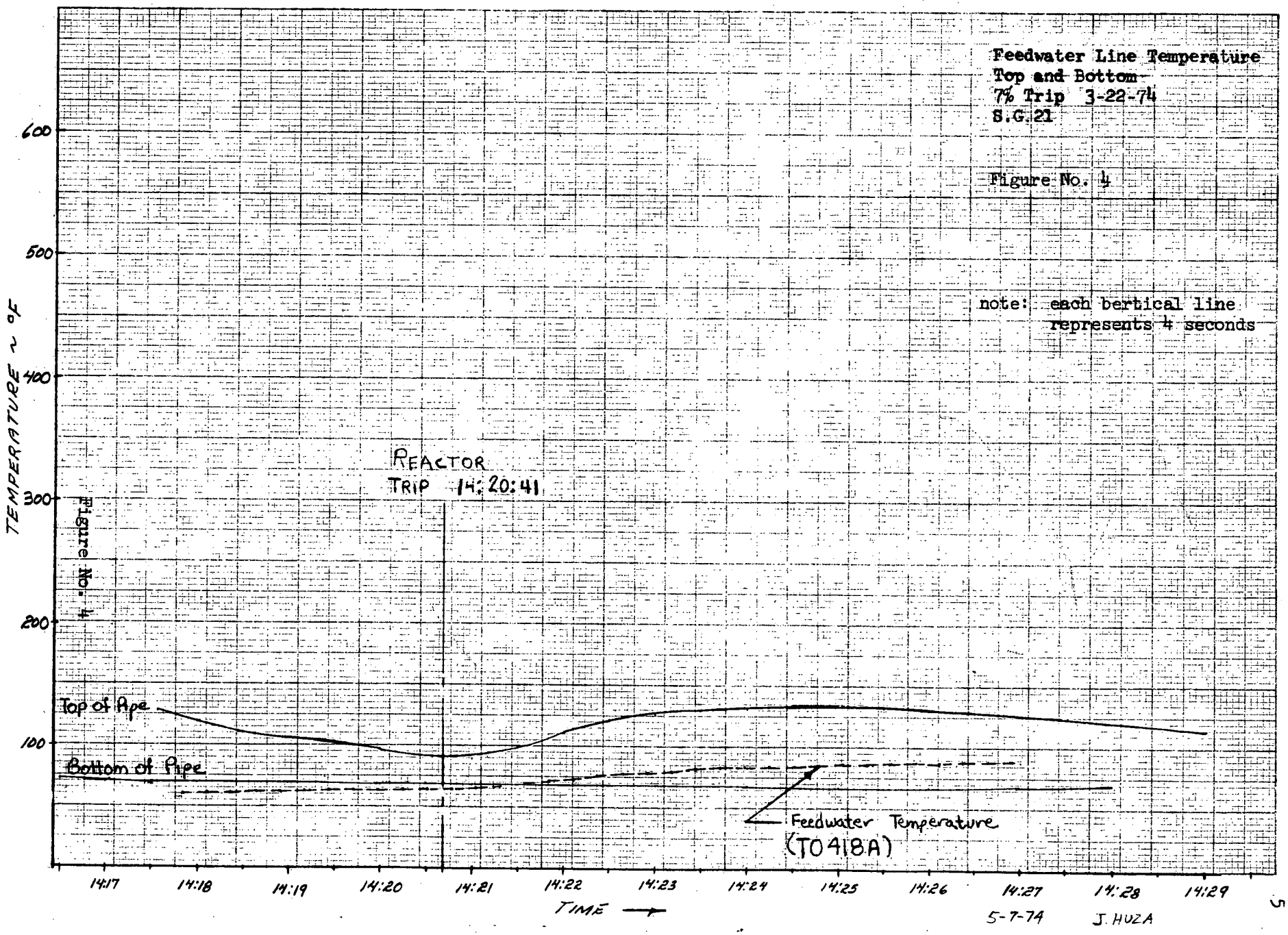
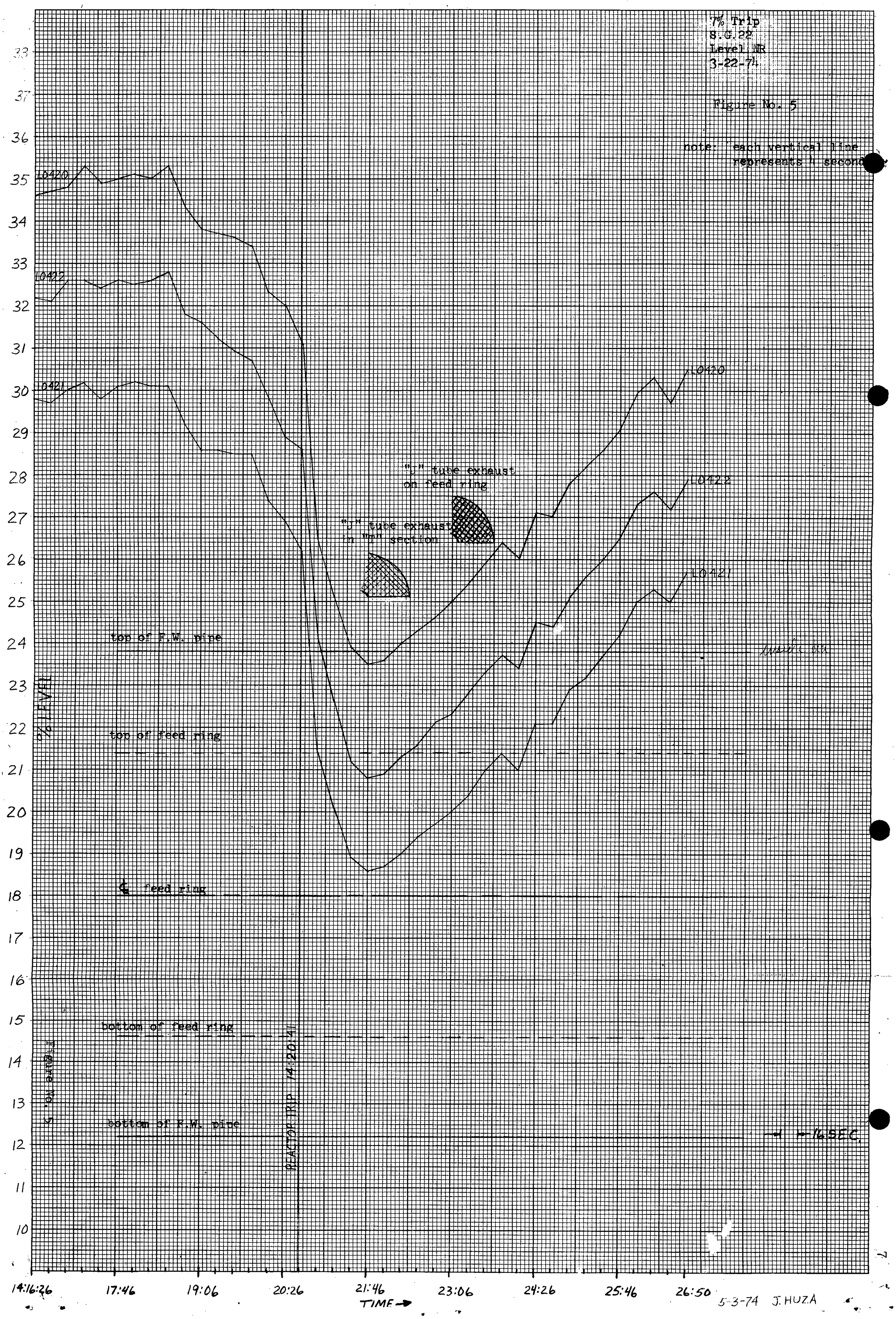
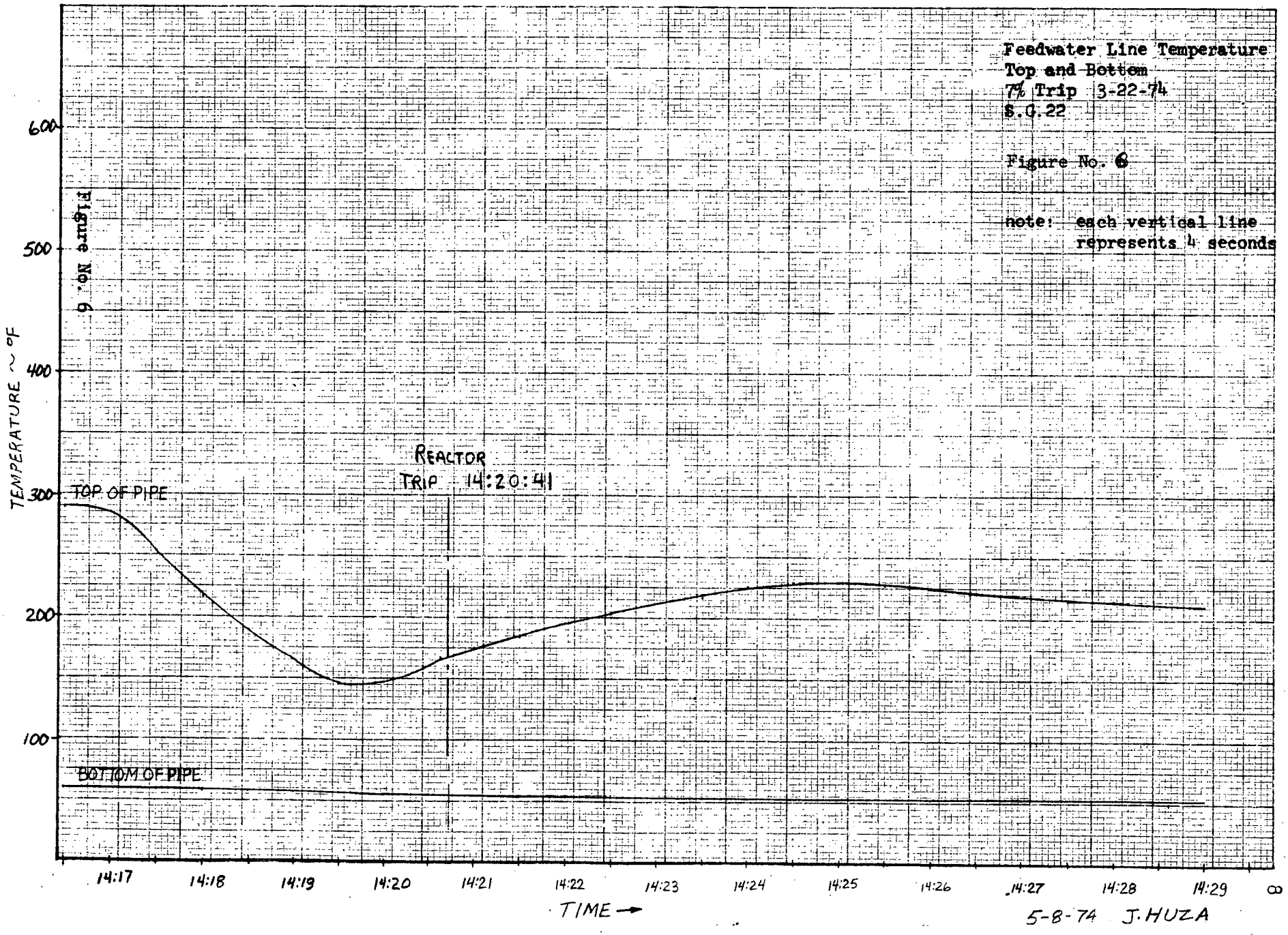


Figure No. 4


7% Trip
S.G. 22
Level NR
3-22-74
Figure No. 5
note: each vertical line represents 1 second
L0420
L0422
L0421
"J" tube exhaust on feed ring
"J" tube exhaust in "T" section
top of F.W. pipe
top of feed ring
℄ feed ring
bottom of feed ring
bottom of F.W. pipe
REACTOR TRIP 14:20:41
16 SEC.
% LEVEL
TIME →
33
37
36
35
34
33
32
31
30
29
28
27
26
25
24
23
22
21
20
19
18
17
16
15
14
13
12
11
10
14:16:26
17:46
19:06
20:26
21:46
23:06
24:26
25:46
26:50
5-3-74 J. HUZA


Feedwater Line Temperature
Top and Bottom
7% Trip 3-22-74
S.G.22
Figure No. 6
note: each vertical line represents 4 seconds
Figure No. 6
TEMPERATURE ~ °F
600
500
400
300
200
100
TOP OF PIPE
BOTTOM OF PIPE
REACTOR
TRIP 14:20:41
14:17
14:18
14:19
14:20
14:21
14:22
14:23
14:24
14:25
14:26
14:27
14:28
14:29
TIME →
5-8-74 J. HUZA

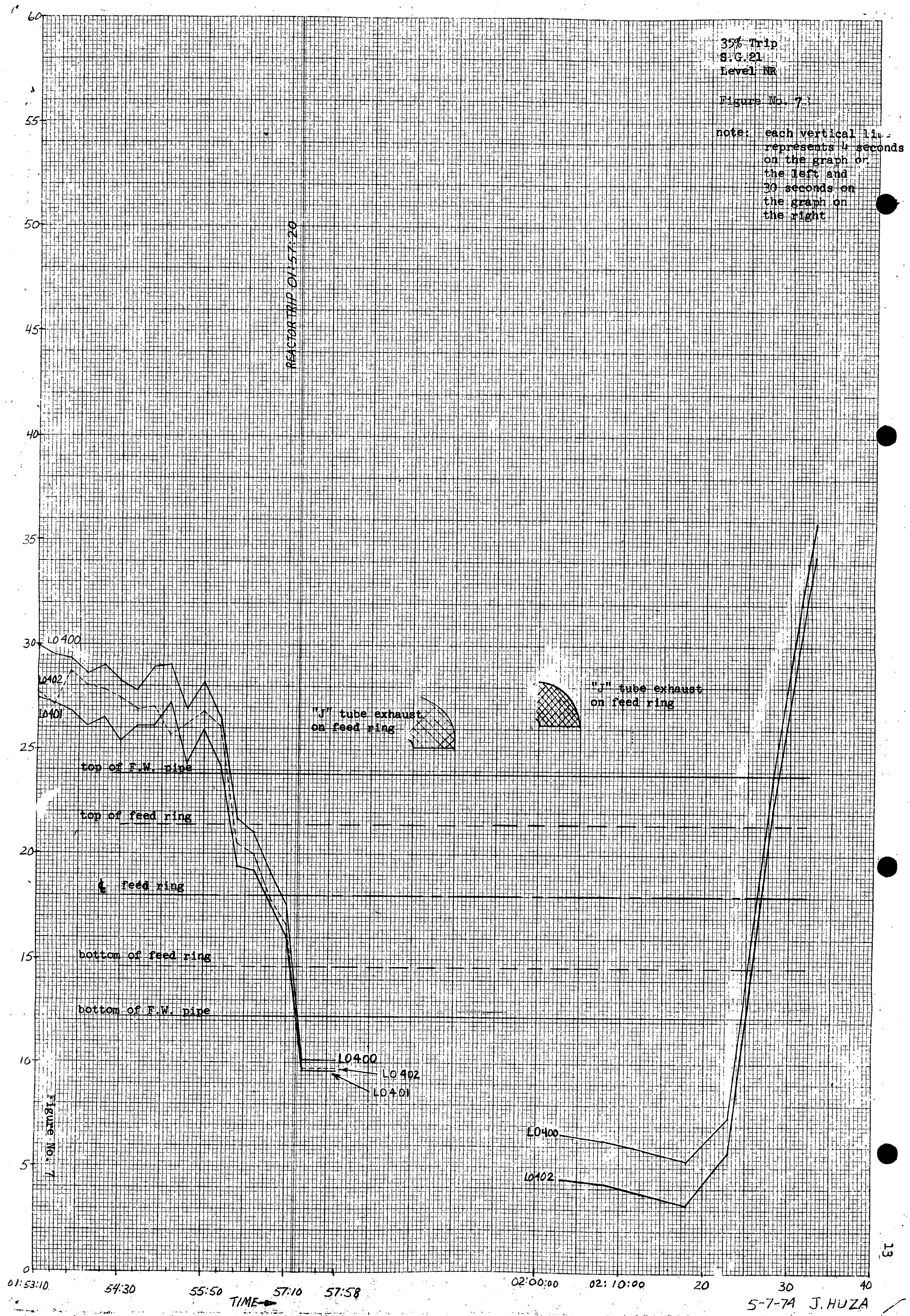


Figure No. 7

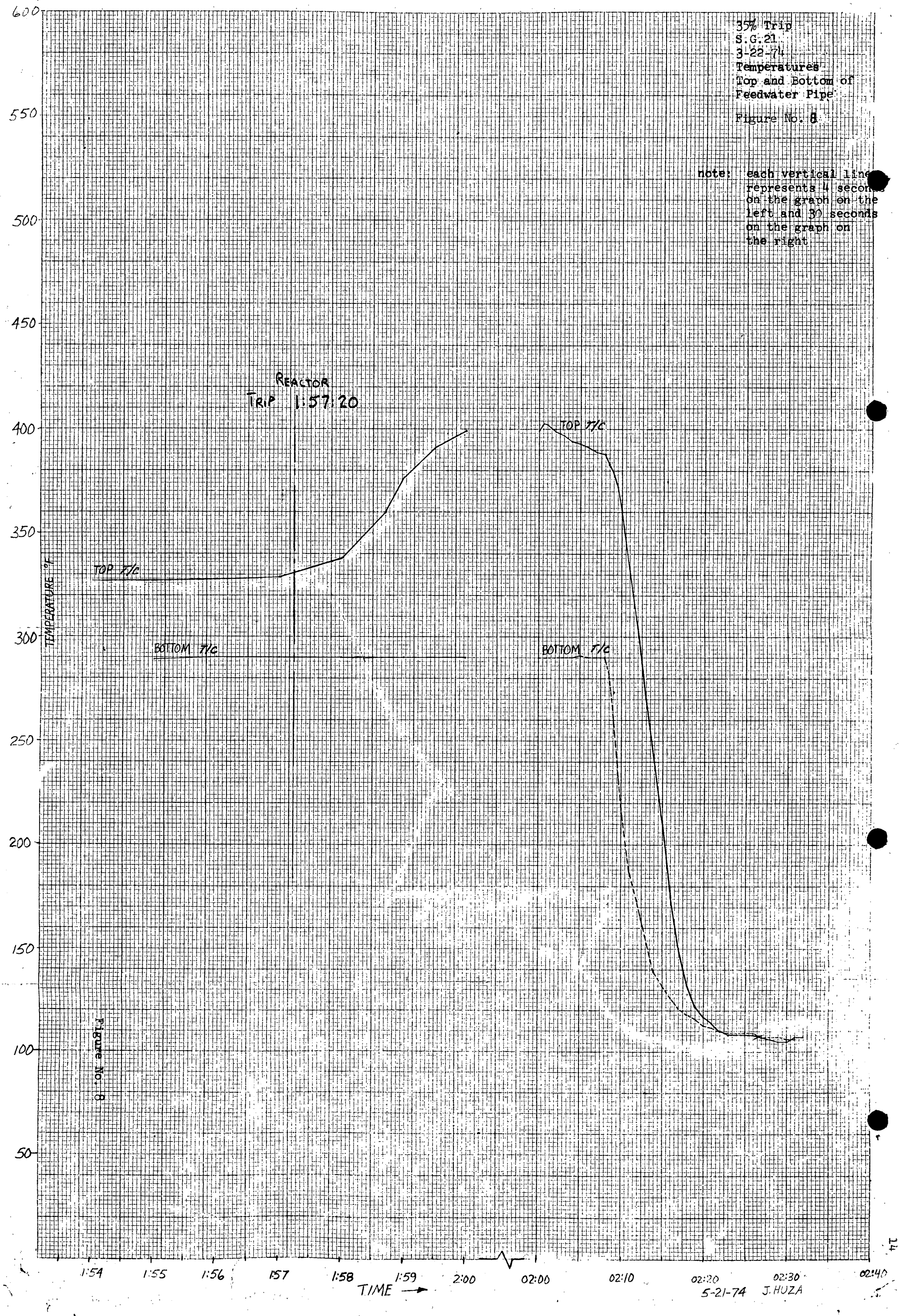


Figure No. 8

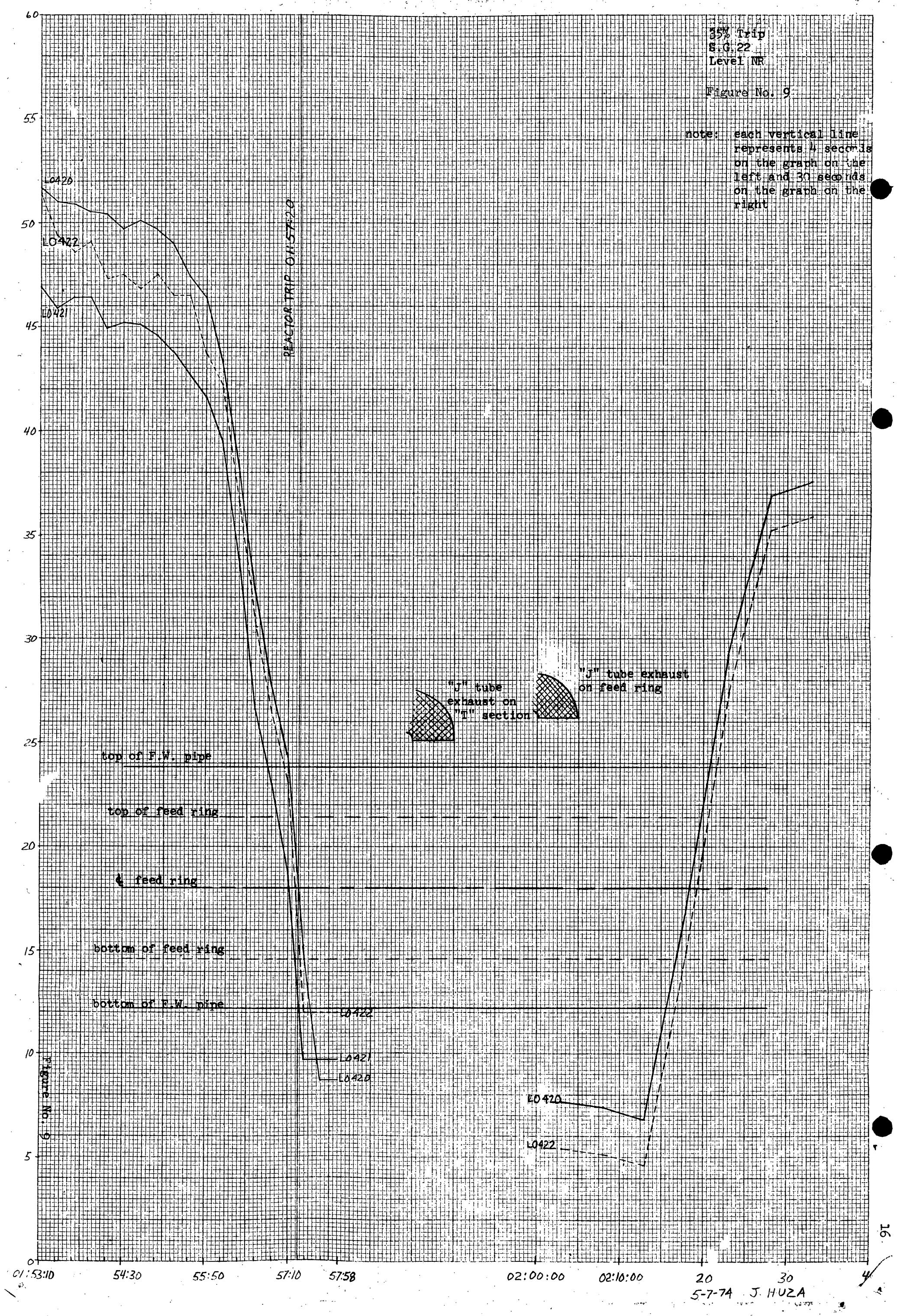


Figure No. 9

47 1323

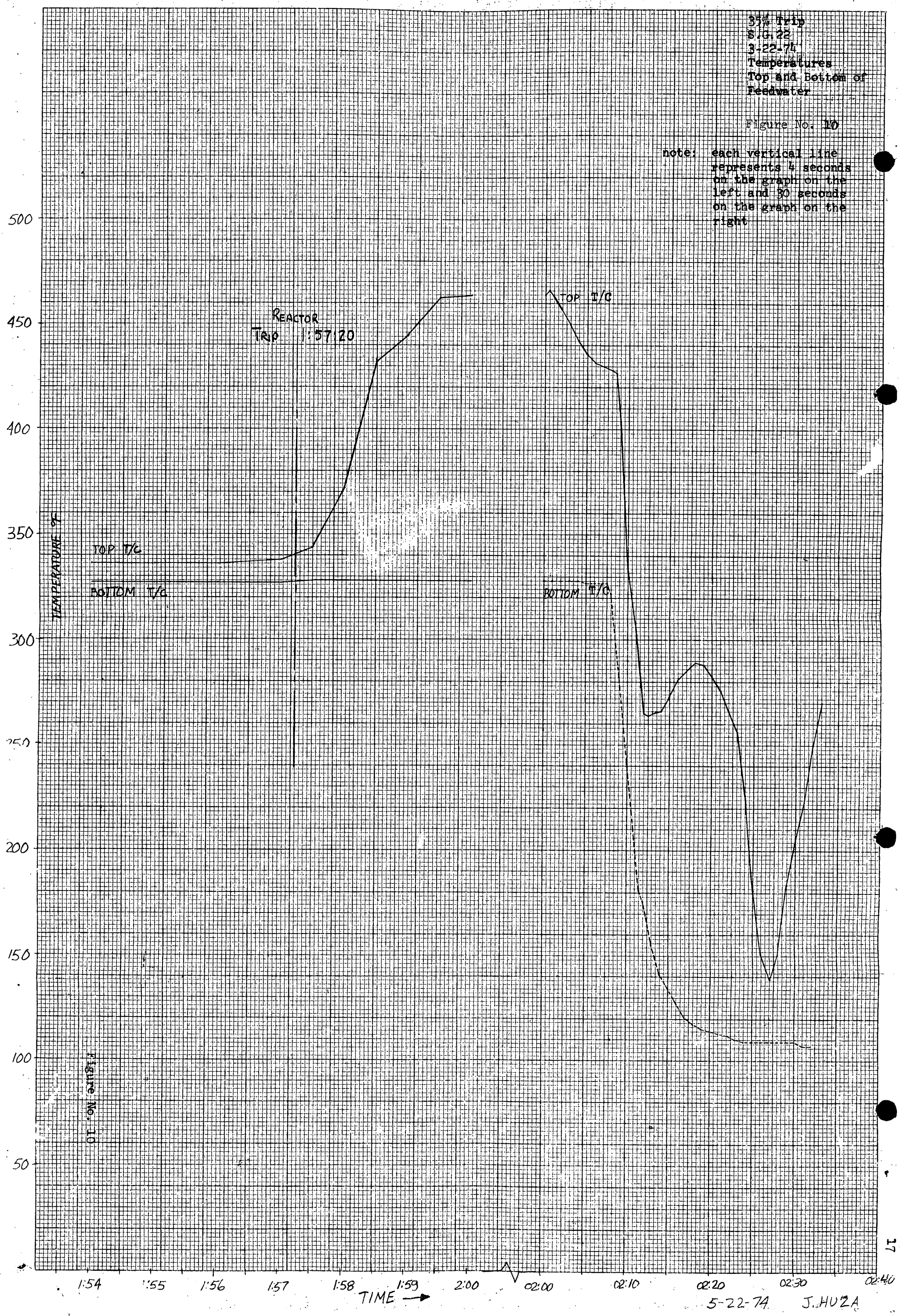


Figure No. 10

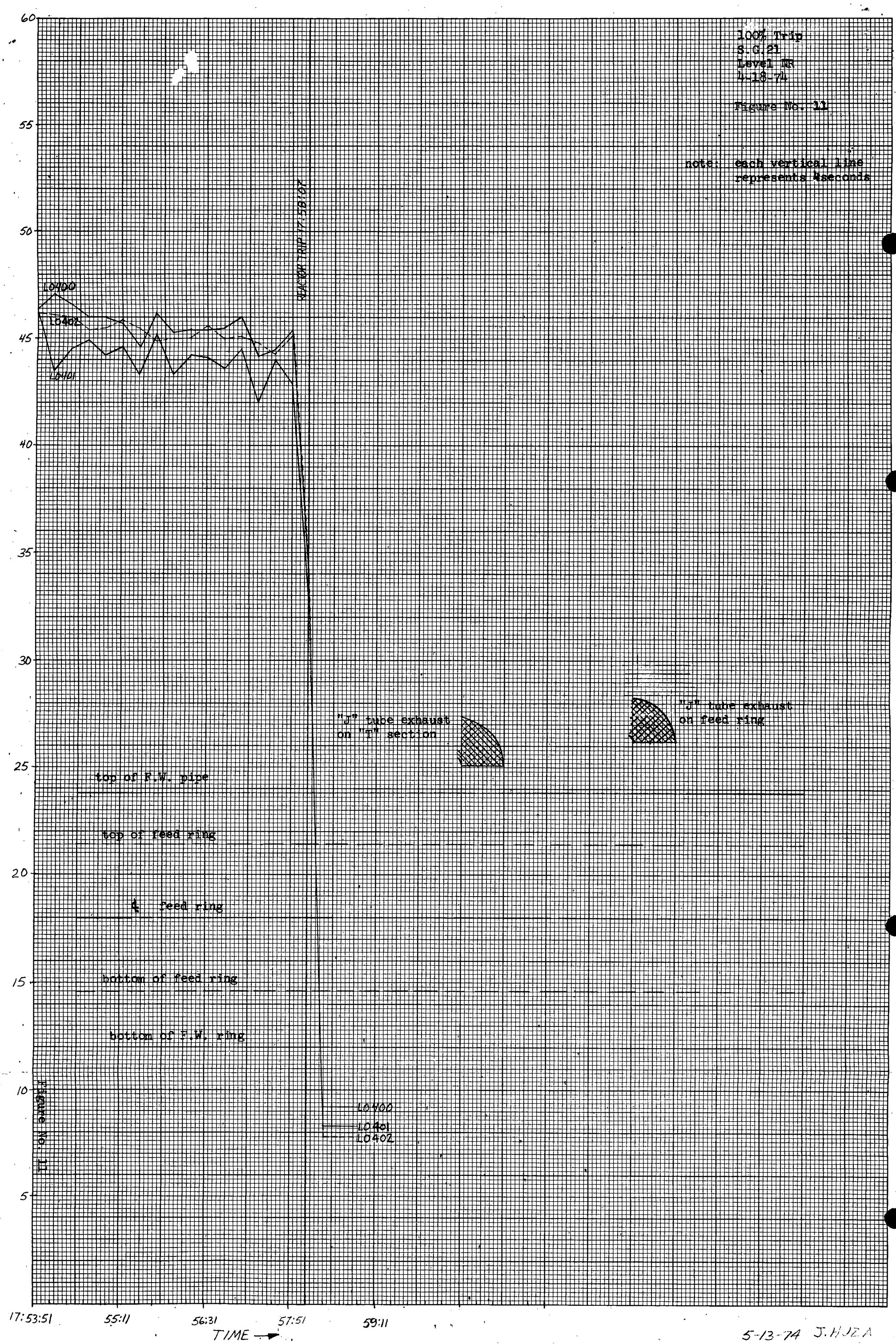
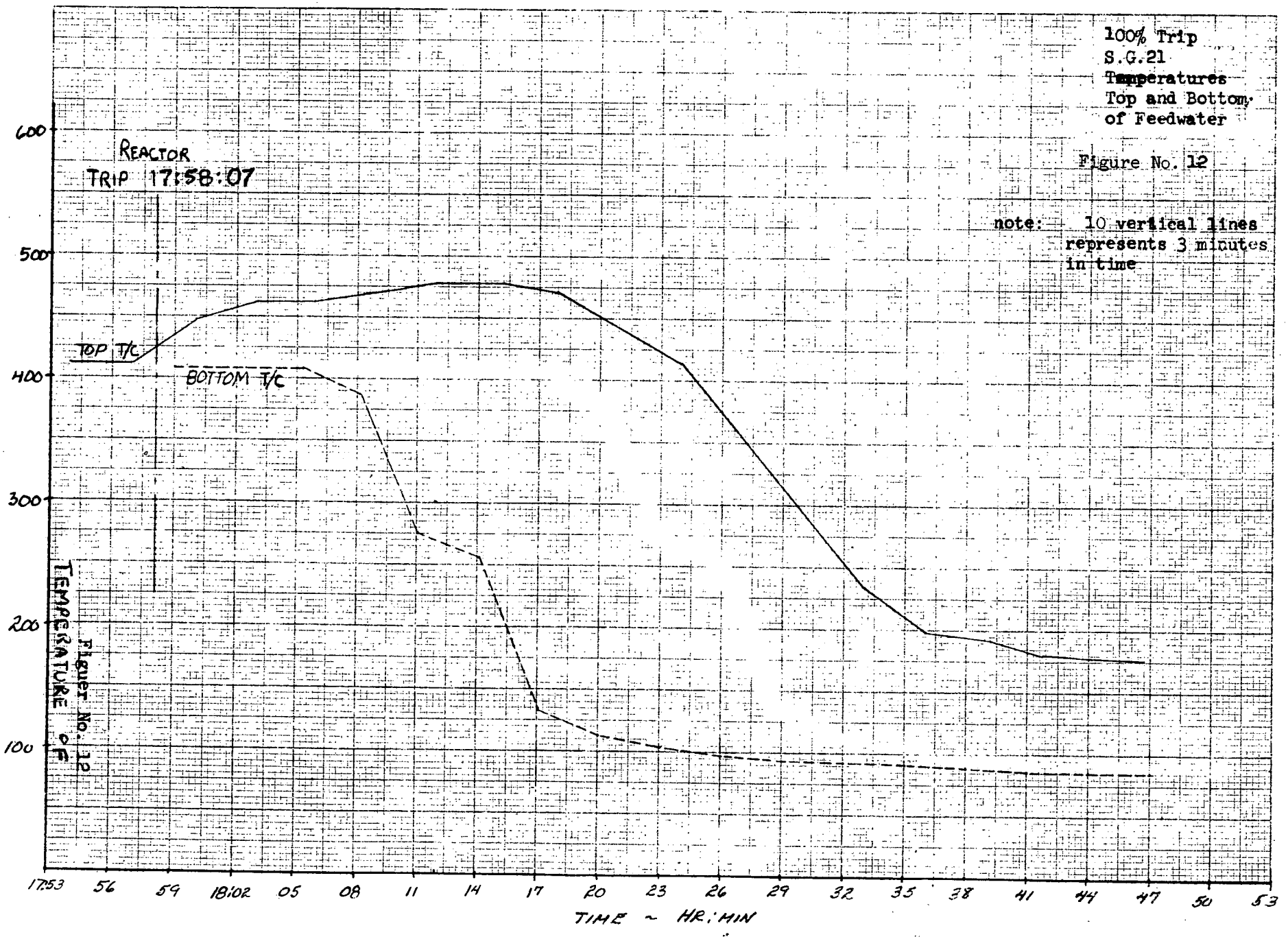

100% Trip
S.G.21
Level IR
4-18-74
Figure No. 11
note: each vertical line represents 4seconds
60
55
50
45
40
35
30
25
20
15
10
5
L0400
L0402
L0401
REACTOR TRIP 17:58:07
"J" tube exhaust on "T" section
"J" tube exhaust on feed ring
top of F.W. pipe
top of feed ring
feed ring
bottom of feed ring
bottom of F.W. ring
L0400
L0401
L0402
17:53:51
55:11
56:31
57:51
59:11
TIME
5-13-74 J.HUZA

Figure No. 11



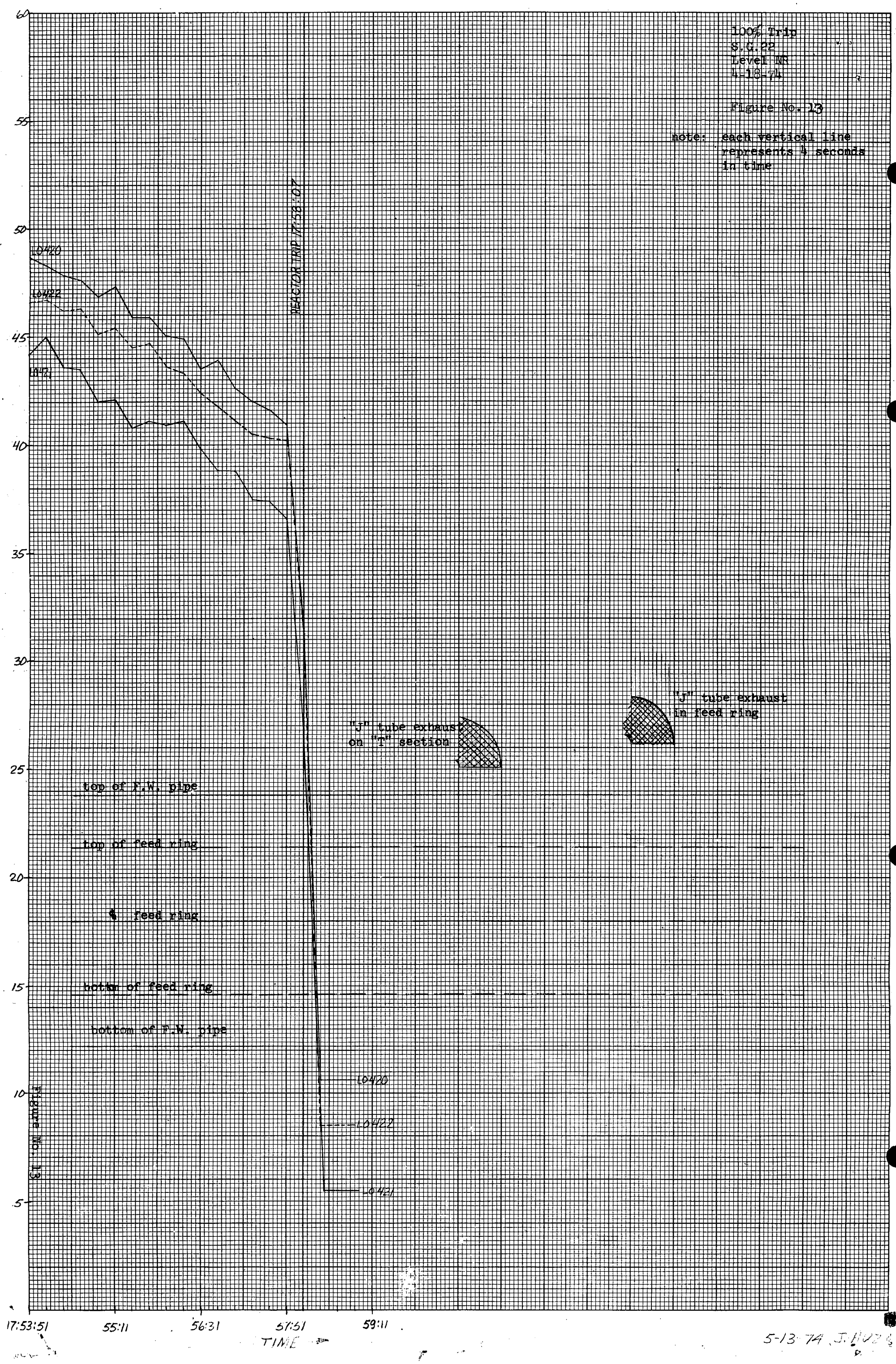

100% Trip
S.G. 22
Level WR
4-18-74
Figure No. 13
note: each vertical line represents 4 seconds in time
REACTOR TRIP 17:53:07
LO420
LO422
LO421
"J" tube exhaust on "T" section
"J" tube exhaust in feed ring
top of F.W. pipe
top of feed ring
feed ring
bottom of feed ring
bottom of F.W. pipe
60
55
50
45
40
35
30
25
20
15
10
5
17:53:51
55:11
56:31
57:51
59:11
TIME
5-13-74

Figure No. 13

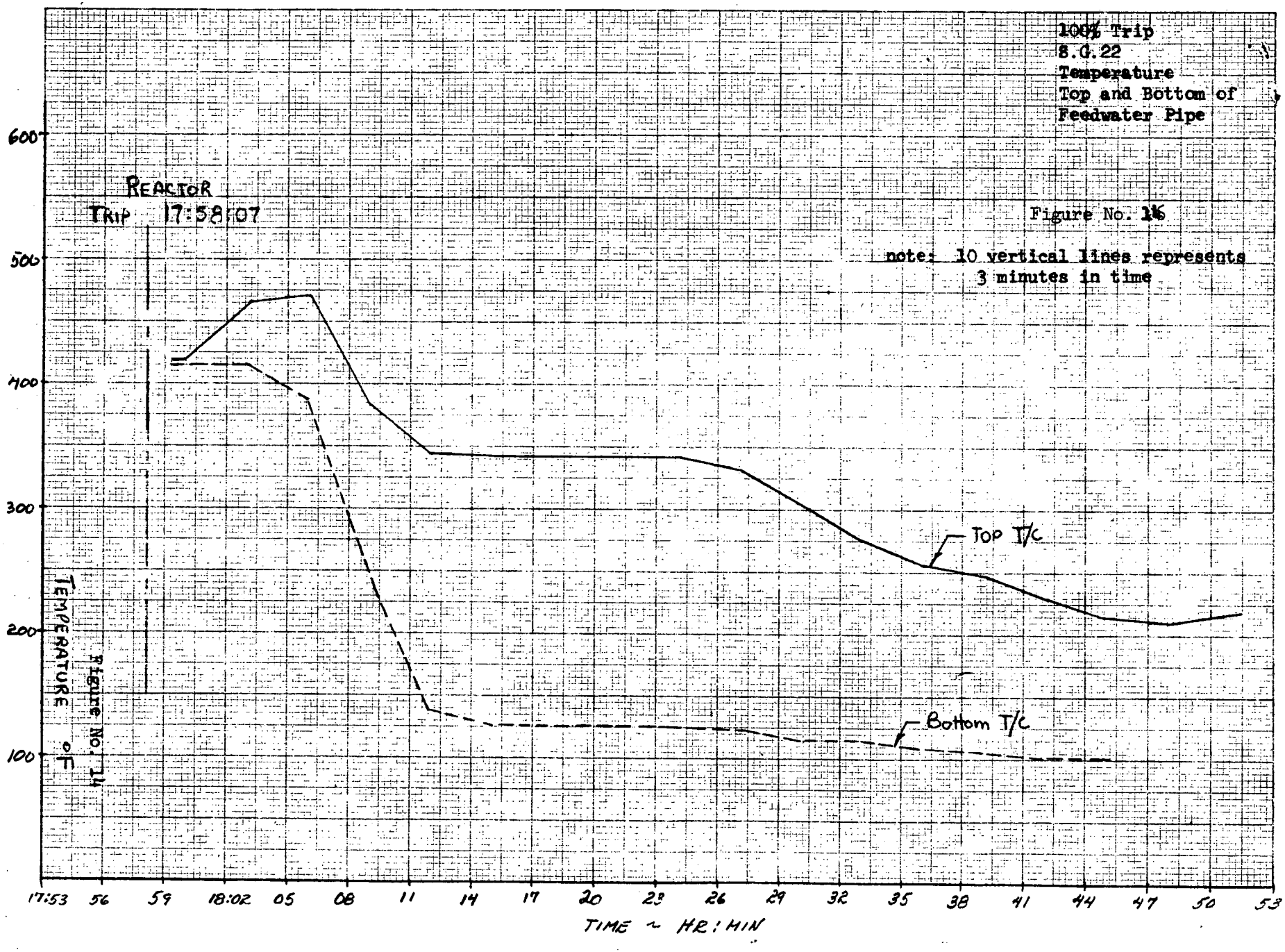


Figure No. 14