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MASTER

PATHFINDER ATOMIC POWER PLANT

DYNAMICS REACTIVITY TESTS (433)

Submitted to

U. S. ATOMIC ENERGY COMMISSION

NORTHERN STATES POWER COMPANY

and

CENTRAL UTILITIES ATOMIC POWER ASSOCIATES

by

ALLIS-CHALMERS MANUFACTURING COMPANY

ATOMIC ENERGY DIVISION

Bethesda, Maryland

ALLIS-CHALMERS

Ref: AEC Contract No. AT(11-1)-589

DISCLAIMER

ACNP-67515

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by J. T. Stone

Submitted to

U. S. ATOMIC ENERGY COMMISSION 602400
NORTHERN STATES POWER COMPANY
and
CENTRAL UTILITIES ATOMIC POWER ASSOCIATES 038400
by
ALLIS-CHALMERS MANUFACTURING COMPANY
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between
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under
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PATHFINDER ATOMIC POWER PLANT

DYNAMICS REACTIVITY TESTS (433)

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FOREWORD

One of a series of reports on research and development in connection with the design of the Pathfinder Atomic Power Plant, this particular report deals with the Dynamics Reactivity Tests (433). The Pathfinder Plant is located at a site near Sioux Falls, South Dakota and reached criticality early in 1964. Owners and operators of the plant will be Northern States Power Company of Minneapolis, Minnesota. Allis-Chalmers is performing the research, development and design, as well as being responsible for plant construction.

The U. S. Atomic Energy Commission, through Contract No. AT(11-1)-589 with Northern States Power Company, and Central Utilities Atomic Power Associates (CUAPA) are sponsors of the research and development program. The plant's reactor will be of the Controlled Recirculation Boiling Water type with Nuclear Superheater.

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TABLES

FIGURES

1. INTRODUCTION

During the early design phases of the Pathfinder Atomic Power Plant, a post-construction Research and Development program was planned in order that the operating characteristics of a boiling water reactor with an internal nuclear superheater could be adequately observed. This report summarizes the 433 series dynamic tests performed during the escalation to full power as part of the planned post-construction R&D program. The specific purposes of Test 433, Dynamics Reactivity Tests, are as follows:

(a) To evaluate the effects on the reactor of certain dynamic disturbances which are likely to occur during the operation of the plant.

(b) To demonstrate that all control systems (pressure, feedwater level, feedwater temperature) are adjusted to respond properly to the various disturbances likely to occur during plant operation.

(c) To verify that the reactor system response to the various planned disturbances is well within the limits of the reactor protection system.

(d) To obtain reactor stability information at various power levels in order to predict reactor stability at higher power levels.

(e) To determine the accuracy of prediction of certain system responses as indicated by the Pathfinder analog simulator.

Prior to each reactor system disturbance introduced during Test 433, estimates were made of the expected response and the adequacy of safety system trip points. Prior to each set of tests at a given power level, the entire test procedure was approved by the Pathfinder Reactor Operations and Safety Committees.

1.1 GENERAL TEST METHODS

Prior to performing the tests, the following general safety procedures were established.

(a) The magnitude and condition of all reactor system test disturbances were planned such that safety system limits would not be exceeded. That is, none of the tests planned in this series were expected to cause a reactor scram or runback. During all tests, the safety system had to be operable with at least the nominal set points specified in the Technical Specifications.

(b) For each type of disturbance, transients were first initiated in the direction which would cause superheater fuel temperatures to decrease. All transients were, where possible, first done slowly, then repeated in shorter time intervals so as to obtain the necessary dynamics information.

(c) During disturbances which result in increasing superheater temperatures, the limits placed on the disturbance were such as to limit fuel hot spot temperatures to those stated in answer 1-4, pp. 1-4.6 and 1-4.19 of the answers to AEC questions submitted in Amendments 12 and 13.[1] "The . . . temperature increases are a maximum of 150 F above steady-state for the operating . . . transients." (1300 F was taken to be the steady-state operating temperature. All transients in Test 433 are "operating transients.")

(d) Testing was repeated at progressive steps of the power escalation, as prescribed by the Technical Specifications, to demonstrate safe reactor behavior at each step in power level before advancing to the next level.

At the various steps of power escalation, reactor power was stabilized and initial values of pertinent variables were recorded. The transient response of these pertinent variables was continuously recorded when the reactor was then subjected to disturbances of the following system parameters:

(i) pressure set point
(ii) feedwater flow
(iii) Feedwater temperature
(iv) recirculation flow
(v) control rod position

After each disturbance, the reactor was returned to its initial condition by again changing the same variable, but in the opposite direction. The magnitudes of disturbance were initially small, and the tests were repeated with gradually increasing magnitudes until the desired size of disturbance was obtained. Each of the disturbances was introduced at 20, 40, 60, 80, and 90 percent of full power; and the results of each set of disturbances were analyzed prior to escalating to the next step in power. The test results were analyzed to determine whether the reactor responses to the disturbances were as predicted, and whether any evidence of system instability was indicated. The information derived from Test 433 is contained in the recorded responses, the sum of which provides indication of plant response to the various dynamic disturbances.

1.2 SUMMARY OF TEST PROCEDURE

The tests included controlled change of each property or mechanism by which changes in reactor power can be effected. A brief description of each type of test and the relevant procedures is given in the following subsections.

1.2.1 Steam Line Pressure Changes

To determine the effect of fairly rapid, but small, changes of reactor system pressure, the pressure control system, on auto control, was utilized to move either the turbine inlet or the dump valve, depending on which set of valves was controlling pressure at the time. Valve motion was achieved by fast movement of the pressure set point dial. This dial movement established a pressure error signal in the pressure control system, and the valve then moved after the error signal had been amplified and integrated by a conventional proportional plus reset controller.

1.2.2 Feedwater Flow Changes

The effects of varying core inlet subcooling were observed by variation of the feedwater flow rate while initially at a constant power level. The reactor liquid level control system was transferred to hand operation, and feedwater flow was reduced by varying amounts by manual reduction of the level set point. After about 2 min (before the low water level trip points were reached) the water level was restored by manual increase of the level set point.

1.2.3 Feedwater Temperature Changes

Another method of varying subcooling is to change feedwater temperature; the effect of such variation was observed during the feedwater temperature change tests. The feedwater temperature control system was placed on auto operation, and the feedwater temperature set point was raised or lowered as desired to observe the dynamic effects of this system on the reactor. The effect of transport lag time between the last feedwater heater (No. 14 heater) and the reactor core was also studied.

1.2.4 Recirculation Flow Changes

One of the original novel features of Pathfinder was the designed capability for varying the forced recirculation flow rate. Butterfly valves, located at the discharge of each of the three recirculation pumps, can be manually positioned from the control room. During the recirculation flow tests, the valves were usually first ganged partially closed, and a steady-state power level was established in this condition. After a heat balance was taken, the valves were ganged open (usually in steps), and the resulting transient was recorded. Special care was exercised during these tests to avoid tripping the power-to-recirculation flow scram circuits and causing unnecessary plant shutdown.

Trip and startup of two of the three recirculation pumps was originally planned as a part of the recirculation flow test. However, because of uncertainties in the flow acceleration resulting from backflow following the trip of a first pump, these planned tests were curtailed during the initial power escalation; and only a one-pump trip test was performed.

1.2.5 Control Rod Motion

Response of the reactor to control rod motion, particularly response of the in-core ion chambers, was the last of the disturbances studied. Addition of postive reactivity was limited to slow rates and small amounts by the restriction that individual rods move no more than 2 in. from a banked position. In some cases, the entire group of boiler rods (rods No. 1 through No. 16) were inserted 5 in., and system response was studied.

1.3 INSTRUMENTATION

1.3.1 Parameters Recorded

Equipment available for these tests included a two-channel Brush recorder and an eight-channel, Type R Dynograph, Offner recorder. The recorders themselves have frequency responses flat to 40 cps, whereas most of the hardware that produces the recorded signals has first-order time constants of about 0.5 sec. The parameters in the following list were available for recording on the strip-chart recorders:

(a) reactor power (out-of-core channel 5)

(b) reactor power (in-core ion chambers 1 to 9, of which three could be recorded simultaneously)

(c) reactor dome pressure - P_1 (transmitter No. 935)

(d) steam line pressure - P_2 (instrument No. 249B)

(e) feedwater flow rate - W_{fw} (P/E* on instrument No. 252A)

(f) feedwater temperature - T_{fw} (P/E* on instrument No. 252B)

(g) main steam flow - W_s (P/E* on instrument No. 253A)

(h) recirdulation flow from each recirculation loop - W_r

(i) superheater fuel temperature - T_f (instrumented superheater assemblies)

(j) main steam temperature at superheater exit - T_2 (instrument No. 249A)

(k) reactor water level (instrument No. 251A)

(l) turbine inlet valve position (position feedback pot on turbine cam shaft)

(m) dump valve command (output of Compudyne amplifier in pressure control system)

(n) dump valve position (feedback pot on the dump valve)

(o) bypass steam flow (bypass flow meter)

*P/E = special pneumatic to electric transducer.

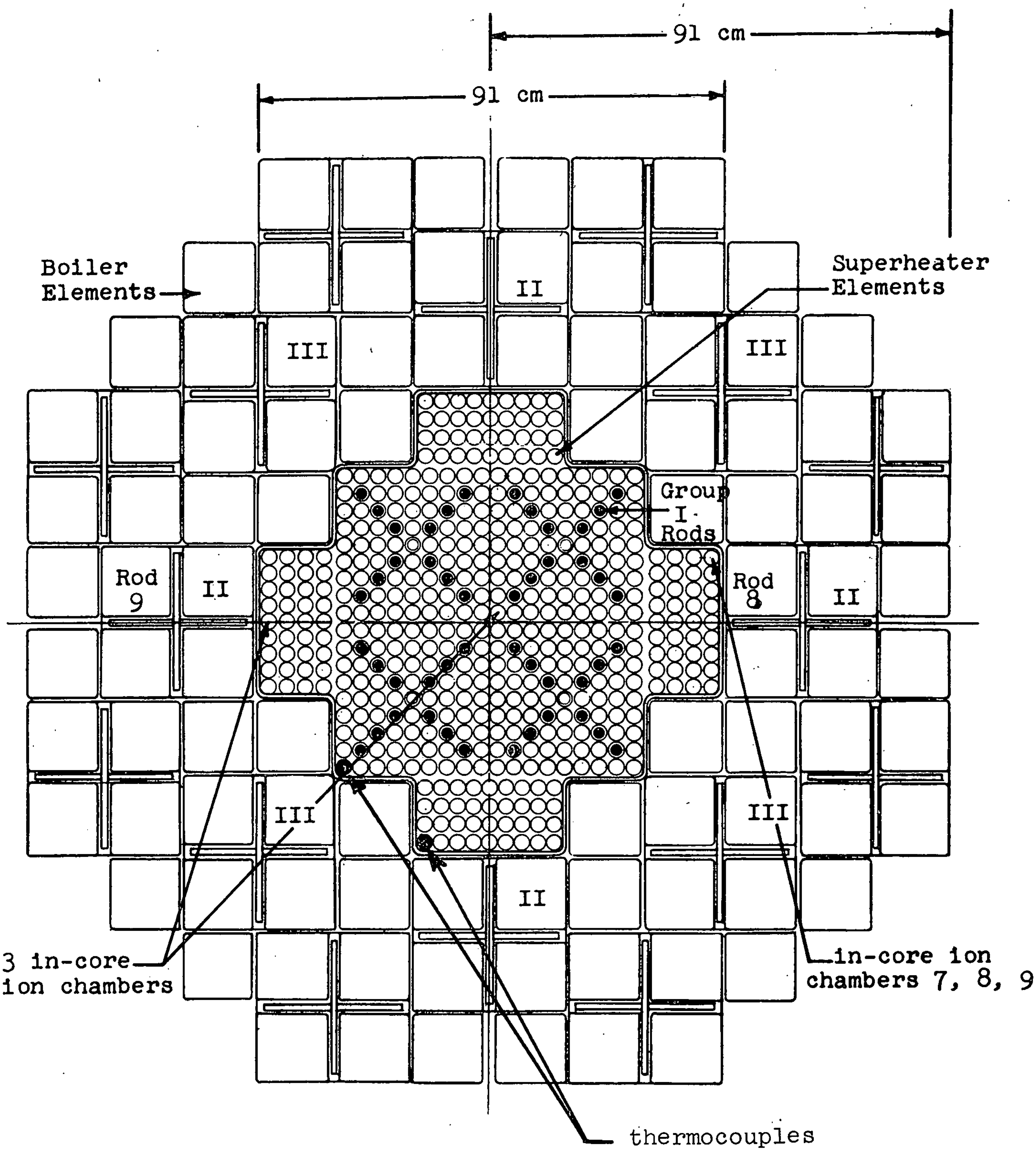
Each of these parameters could be used as inputs to the eight-channel Offner recorder or to the two-channel Brush recorder; the parameters selected for recording were varied according to requirements of the particular test. In order to prevent loss of signal and spurious plant safety action when cables were changed, buffer amplifiers (d-c chopper-stabilized Donner wide-band amplifiers) were used in some instances between the transducer and the recorder.

1.3.2 In-Core Ion Chambers

As part of the R&D program, a string of three ion chambers occupies each of the three superheater locations N-1, L-9, and S-5, in place of superheater fuel elements. In each string, the three ion chambers are axially positioned 18 in., 36 in., and 54 in. from the bottom of the core. The instrumentation limited simultaneous ion chamber reading to only three of the nine ion chambers, but a coaxial switching scheme provided an easy means of selecting any three of the nine chambers for observation and recording. Each string of three ion chambers is provided with a flux wire thimble which was periodically used to calibrate the ion chambers. Figure 1.1 shows the location of the ion-chambers in the core. Ion chambers 1, 2, and 3 are in position S-5 with chamber #1 located at the 54 in. core elevation; chamber #2 at 36 in. and chamber #3 at 18 in. Similarly, in position N-1, chamber #7 is at 54 in. core elevation; chamber #8 is at 36 in.; and chamber #9 is at 18 in.

1.3.3 Superheater Fuel Thermocouples

During most of this test program, the outputs of thermocouples O-10 and I-39 were recorded with the other reactor parameters. The thermocouple locations are shown on Fig. 1.1. Thermocouple O-10 is located 10 in. from the bottom of the core and is welded to an outer fuel tube in superheater fuel location A-18. Thermocouple I-39 is located 39 in. from the bottom of the core and is welded to an inner fuel tube in superheater fuel location E-9. These thermocouples were chosen for recording since they are closest to the calculated superheater "hot spot."



PATHFINDER CORE, PLAN VIEW

FIG. 1.1

2. RESULTS OF PRESSURE DISTURBANCES

The first test performed at each new level of power escalation was usually a steam line pressure disturbance. This test was easily and quickly accomplished. It tested both the pressure control system and the reactor response, and the results were predictable with a high degree of accuracy.

Table 2-1 lists the initial conditions for those pressure disturbances at each power level that were most descriptive of reactor system response.

The reactor system responses were generally as expected and as predicted by pre-test calculations. The actual responses are shown in Figs. 2.1 through 2.5. In these tests, the pressure control system was in the automatic mode, and the disturbance was introduced by making a manual change of the pressure set point and then allowing the system to respond. The reactor exhibited stable response to these changes in pressure control system set point at all power levels.

One of the expected results that occurred was that a given change in steam flow (lb/hr) resulted in slightly less severe power disturbances at the higher initial power levels; this occurs because a given lb/hr change of flow represents a smaller portion of the total flow at the higher flow rates. This is obvious upon comparing the power responses listed in Table 2-2. The effect of superheater control rods being inserted and affecting the power split between the boiler and superheater can be seen by comparing the change in bulk steam exit temperature at 38 Mwt (ΔT_2 = +8 F) and the change in this parameter at 169 Mwt (ΔT_2 = +13 F).

One of the goals of Test 433 was to observe reactor response independent of the pressure control system. For this purpose, the master control station of the pressure control system was placed in the manual mode of control, and the dump valve was moved stepwise in discrete amounts. Figure 2.6 shows the reactor system response to one of these sudden steam flow changes at an initial power of 30 percent (200,000 lb/hr steam flow). Results of other smaller sudden steam flow changes are shown in Table 2-3. One interesting result shown in this table is that the magnitude of the power response and the rate of change of power response both increase as the percentage of steam flow disturbance increases.

2.1 COMPARISON OF EXPERIMENTAL RESULTS AND CALCULATIONS

Another principal goal of Test 433 was to provide a means of comparing the experimental results with pre-test calculations. This "feedback" would then provide the basis for assessments of analytic models[2] and the results[3] obtained through their use. Figure 2.7 shows a comparison of calculated and measured power response and bulk steam temperature for a sudden steam flow reduction at 30 percent initial power. The calculated power response yields a higher overshoot (due most likely to a different void reactivity coefficient in the two cases), but the initial rate of power rise (for the first 2 sec of the transient) is identical for the measured and calculated cases. The calculated and measured cases are also compared in Table 2-3 where it can be seen that the calculated rate of pressure rise is about 50 percent higher than the measured vessel pressure rise. While instrumentation inaccuracies may account for part of the difference, the value of the denominator D in equation 5.6 of the calculational model[2] should probably be increased somewhat to compensate for this difference.

Table 2-4 is a summary of selected comparisons of pre-test calculations and the experimental results. This table shows that the pre-test calculations closely matched the measured test results. The differences in power overshoot at the 189 Mwt/169 Mwt comparison can be directly attributed to different values of reset in the pressure control system for the two cases. The calculated fuel temperature overshoot, ΔT_F, was always larger than the measured overshoot on thermocouple #O-10 because the calculated values of T_F were for the hottest spot in the superheater, as calculated with conservative hot spot factors. In addition, thermocouple #O-10 was not at the core hottest spot. In the 38 Mwt case the calculated $\Delta T2$ overshoot was larger than in the measured case because both superheater control rods and the Group II boiler control rods were actually inserted during the test, causing a low superheater power fraction.

TABLE 2-1

INITIAL PLANT CONDITIONS FOR PRESSURE SET POINT CHANGES

Reactor Power	38 Mwt (6/20/66)	76.0 Mwt (8/12/66)	109 Mwt (12/3/66)	142 Mwt (12/18/66)	169 Mwt (1/5/67)
Channel 5 Current	0.43×10^{-6} amp	1.04×10^{-6} amp	0.20×10^{-5} amp	0.30×10^{-5} amp	0.345×10^{-5} amp
Steam Flow	150,000 lb/hr	275,000 lb/hr	380,000 lb/hr	530,000 lb/hr	570,000 lb/hr
Feedwater Temperature	390 F	378 F	376 F	374 F	374 F
Total Recirculation Flow	62,000 gpm	59,000 gpm	57,400 gpm	55,500 gpm	56,000 gpm
Superheater Fuel Temp. t/c #O-10	675 F	690 F	752 F	765 F	765 F
Exit Steam Temperature	543 F	626 F	643 F	642 F	661 F
Reactor Pressure	542 psig	548 psig	553 psig	580 psig	587 psig
Dump Valve Position	13% open	~28% open	6.5% open	6% open	6% open
Inlet Valve Position	0% open	0% open	32.5% open	48% open	54% open
Pressure Control	Auto, on Dump Valve	Auto, on Dump Valve	Auto, on Dump Valve	Auto, on Dump Valve	Auto, on Dump Valve
Level Control	Manual	Auto, Single Element	Auto, Single Element	Auto, Single Element	Auto, Single Element
Feedwater Temperature Control	Manual	Manual	Auto	Auto	Auto

TABLE 2-2

SUMMARY OF REACTOR TRANSIENT RESPONSE TO PRESSURE SET POINT CHANGES

Reference Figure / Initial Reactor Power	Fig. 2.1 38 Mwt (6/20/66)	Fig. 2.2 76.0 Mwt (8/12/66)	Fig. 2.3 109 Mwt (12/3/66)	Fig. 2.4 142 Mwt (12/18/66)	Fig. 2.5 169 Mwt (1/5/67)
Disturbance	+5 psi	+3 psi in 3.5 sec	+5 psi in 3 sec	+5.5 in 3 sec	+5 psi in 5.5 sec
Max Δ Power on in-core #7, % of existing power	+13%	+5.6%	+6.5%	+6.2%	+5.1%
Max ΔT_F on t/c #O-10	+34 F	+22 F	+29 F	+30 F	+22 F
Max ΔW_s	--	-12,000 lb/hr	-10,000 lb/hr	-15,000 lb/hr	-14,000 lb/hr
Max ΔT_2	+8 F	+9 F	+11 F	+13 F	+13 F
Controlling Rod Group and Position	Group III at 49 in.	Group I at 53.8 in.	Group II at 27.5 in.	Group II at 38.8 in.	Group II at 54 in.

TABLE 2-3

SUMMARY OF MEASURED REACTOR RESPONSE TO STEP REDUCTIONS IN STEAM FLOW AT 30 PERCENT POWER (200,000 lb/hr STEAM FLOW)

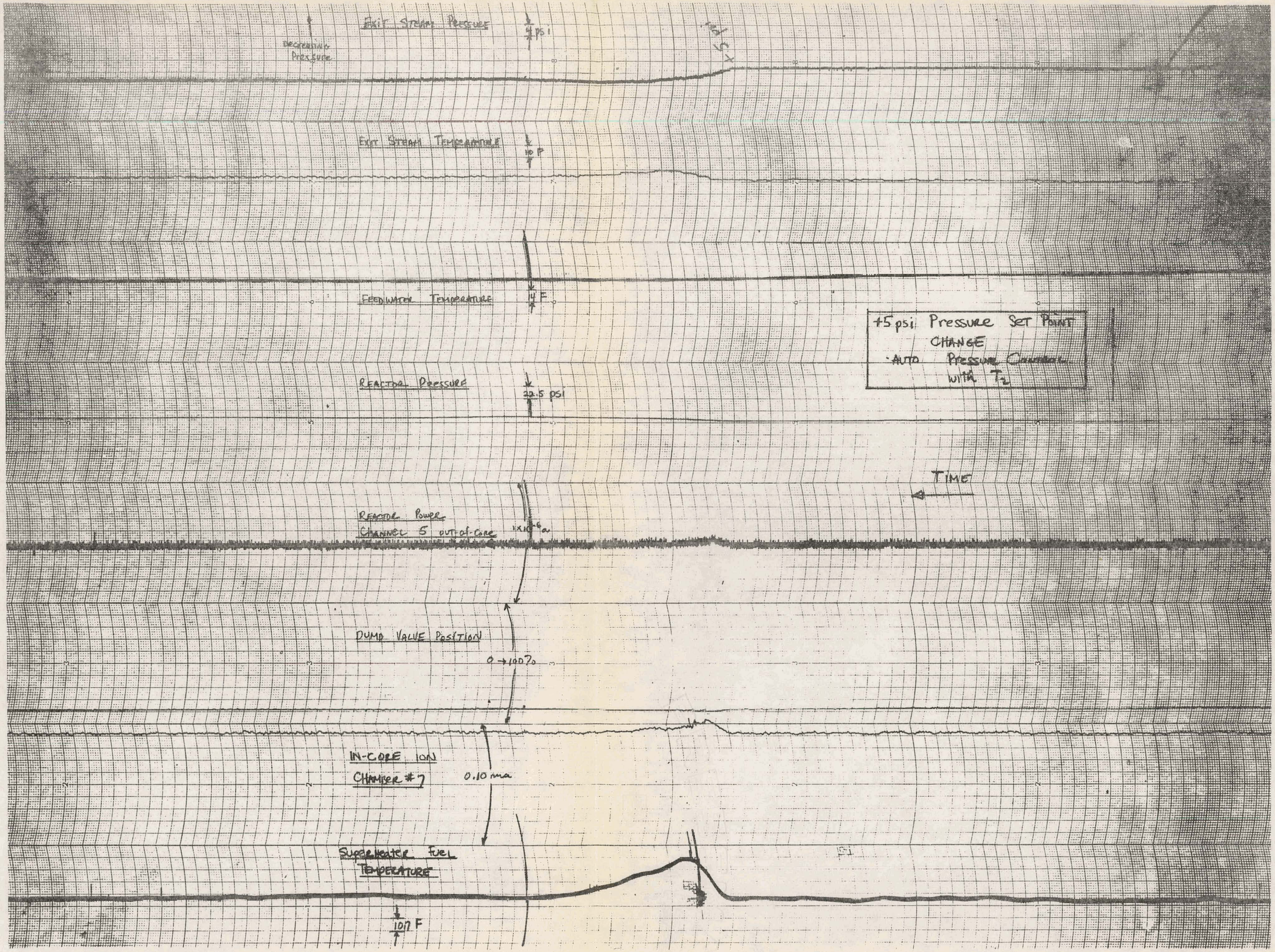
Manual Pressure Control

ΔW_s lb/hr	Max. Δn % of Initial Value	Max. $\dot{n}$ %/sec	Max. ΔT_F t/c #O-10	$\dot{P}_i$ psi/sec
-5,000	+5.2	+1.9	+30 F	0.078
-10,000 avg	+8.1	+2.4	+32 F	0.30
-14,000 avg	+9.6	+7.9	+50 F	0.30
-20,000 avg	+15.0	+9.4	+56 F	0.35
-35,000 avg	+18.9	+3.9	--	0.60
-20,000 calculated (Fig. 2.7)	+17.7	+9.4	--	0.53

TABLE 2-4

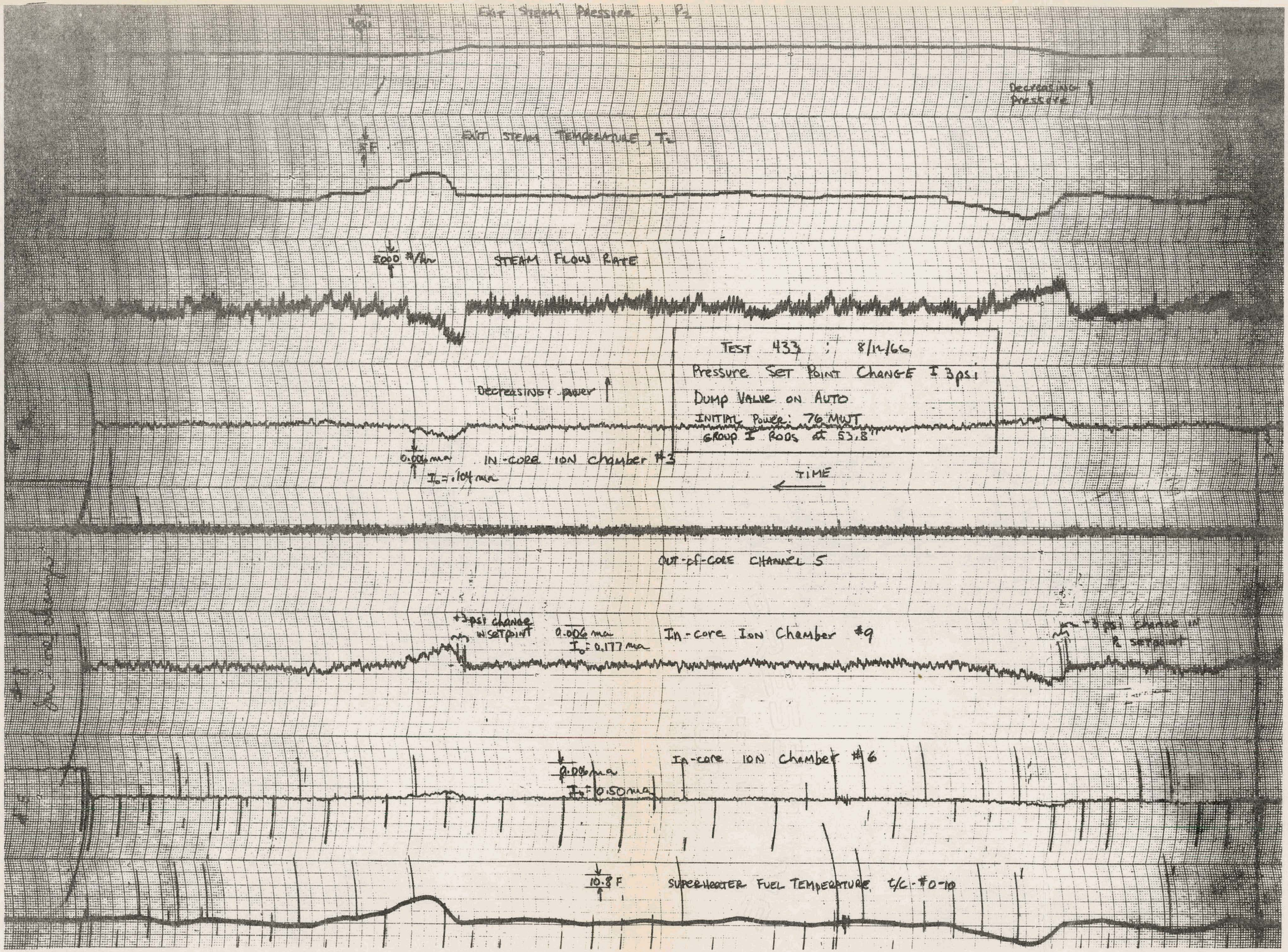
COMPARISON OF TRANSIENT PARAMETERS: CALCULATED PRE-TEST PREDICTIONS VS. EXPERIMENTAL RESULTS

	Disturbance	Initial Power Level	Max. Δ Power % of Initial Power	Max. ΔT_F Pre-Test (hot spot) Experiment (t/c #O-10)	Max. ΔT_2	Damping — Time for System to Return to Steady-State
Pre-Test Calculations	+5 psi in 3 sec	38 Mwt	+14%	+75 F	+25 F	75 sec
Experimental Results	+5 psi in 3 sec	38 Mwt	+13% on in-core #7	+34 F	+8 F	80 sec
Pre-Test Calculations	+5 psi in 3 sec	114 Mwt	+4%	+40 F	+13 F	55 sec
Experimental Results	+5 psi in 3 sec	109 Mwt	+6.5% on in-core #7	+29 F	+11 F	55 sec
Pre-Test Calculations	+5 psi in 3 sec	189 Mwt	+2.6%	+25 F	+9 F	50 sec
Experimental Results	+5 psi in 5.5 sec	169 Mwt	+5.1% on in-core #7	+22 F	+13 F	50 sec



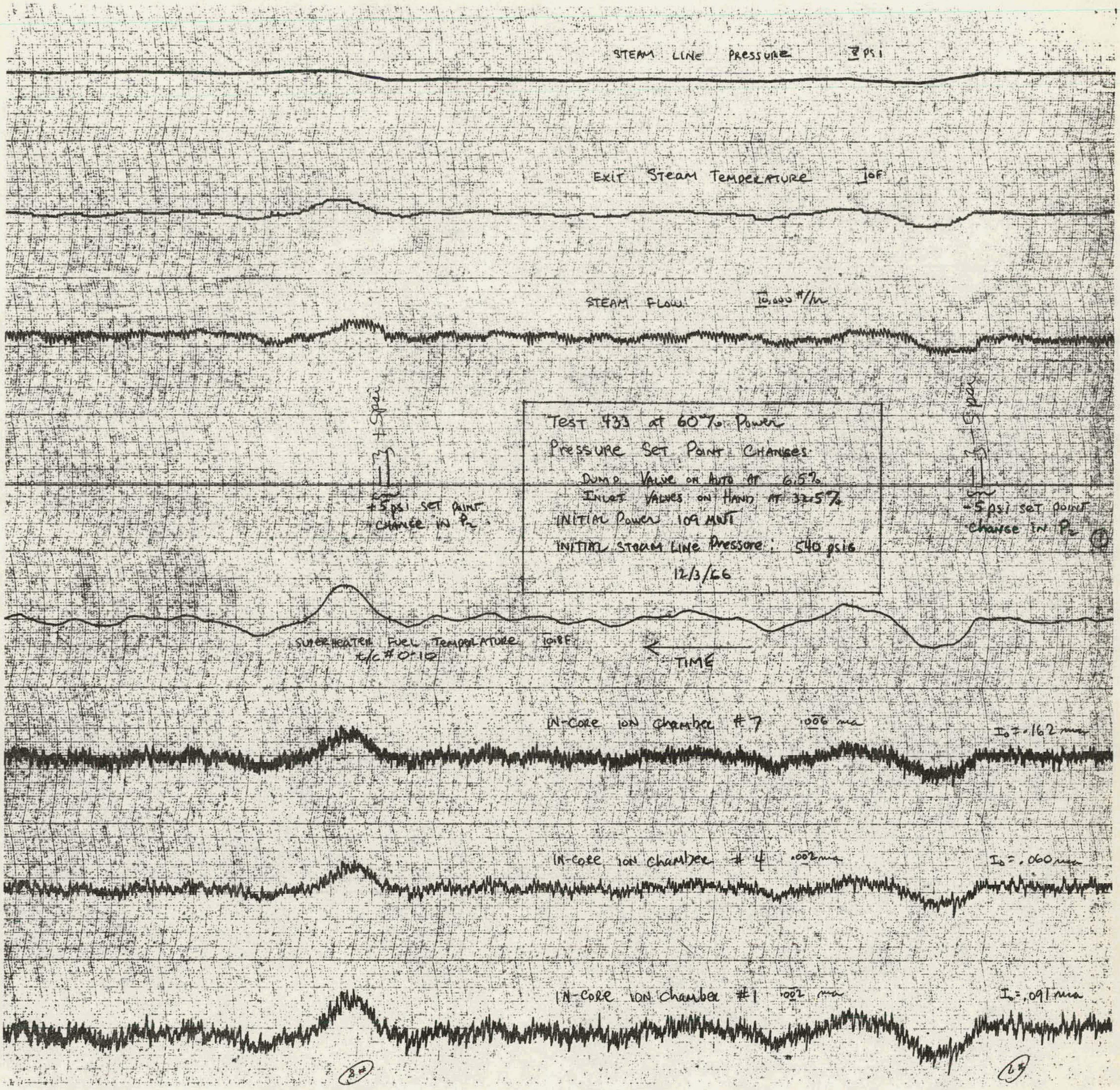
PRESSURE SET POINT CHANGE: 38 Mwt

FIG. 2.1



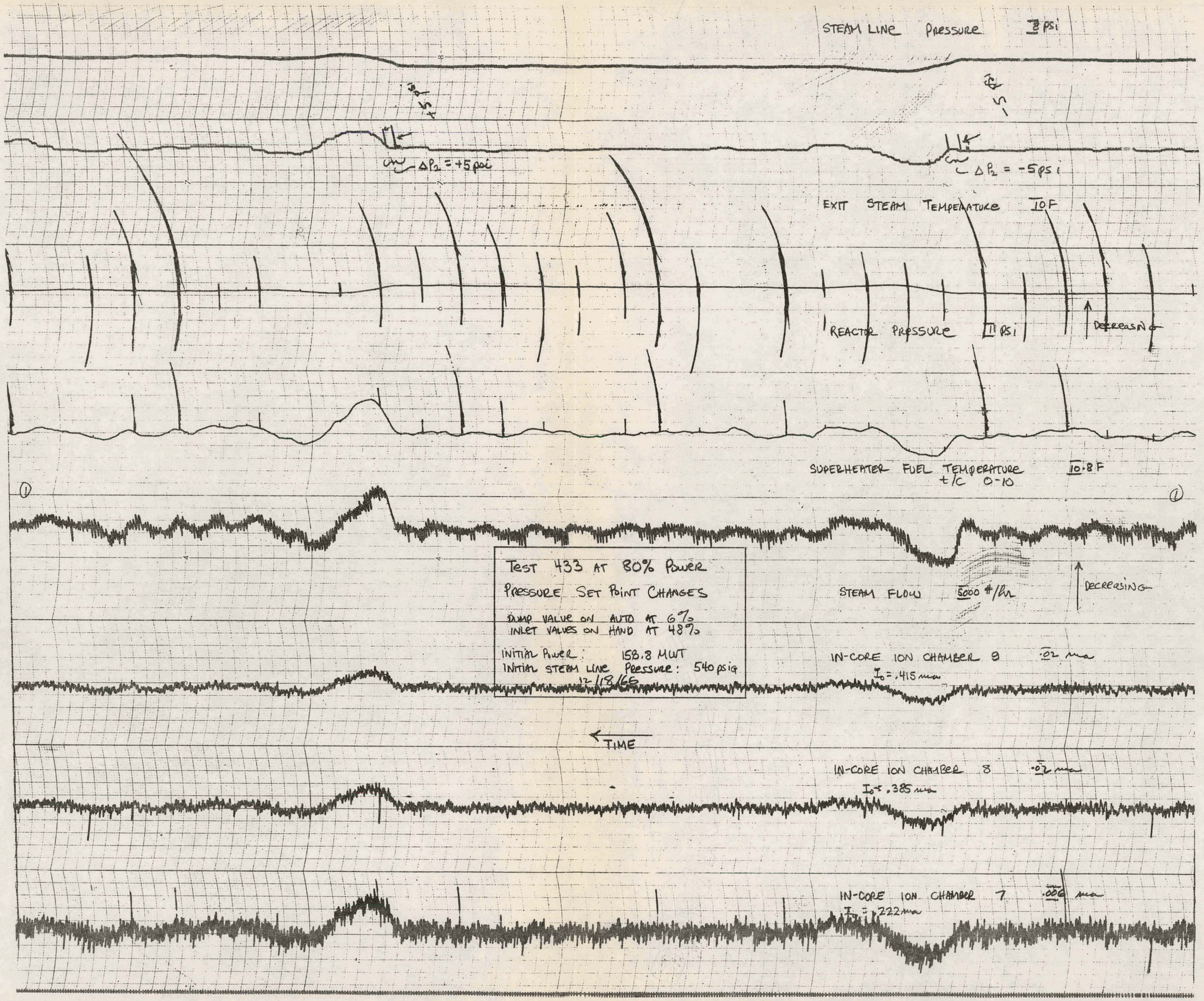
PRESSURE SET POINT CHANGE: 76 Mwt

FIG. 2.2



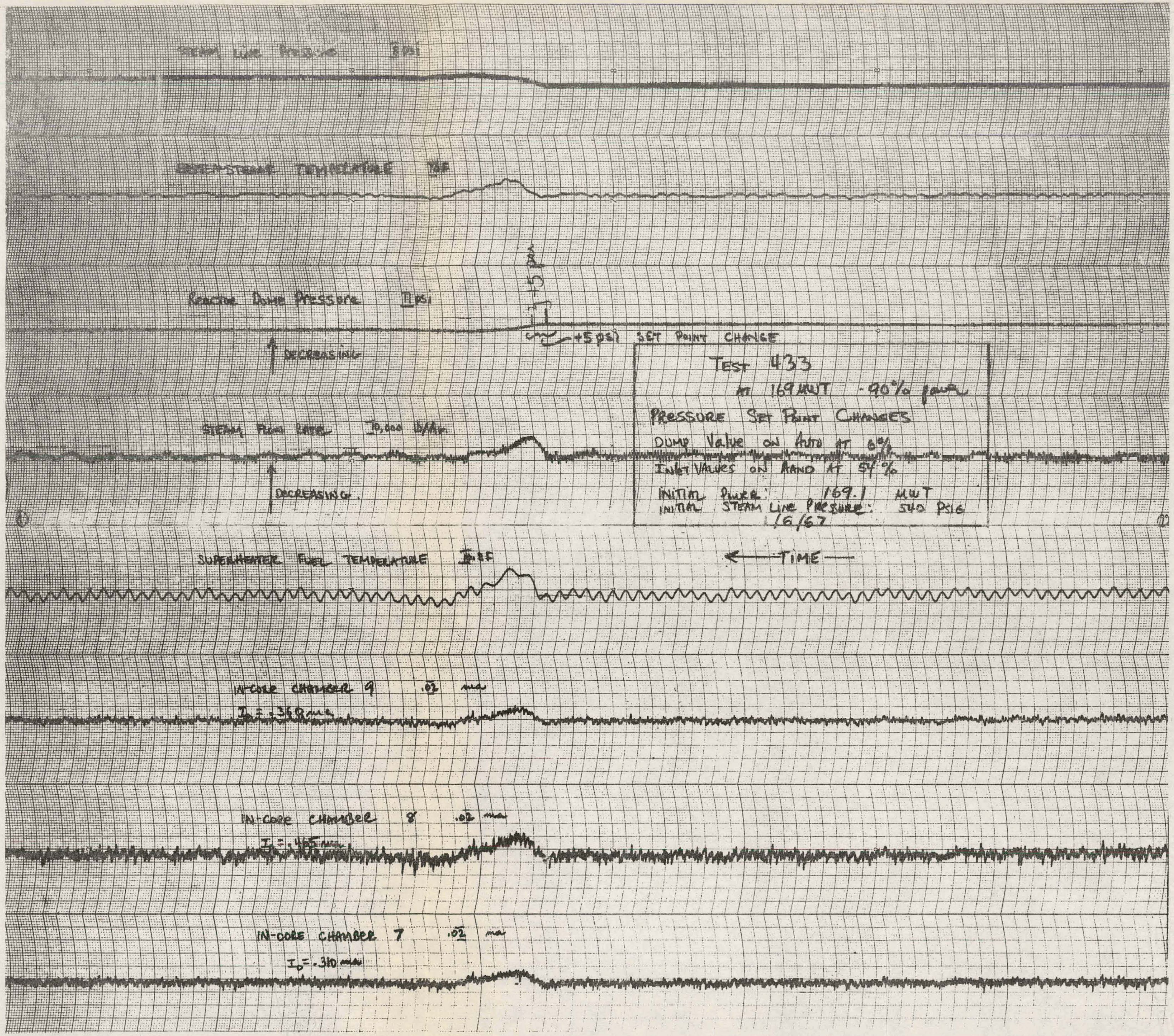
PRESSURE SET POINT CHANGE: 109 Mwt

FIG. 2.3



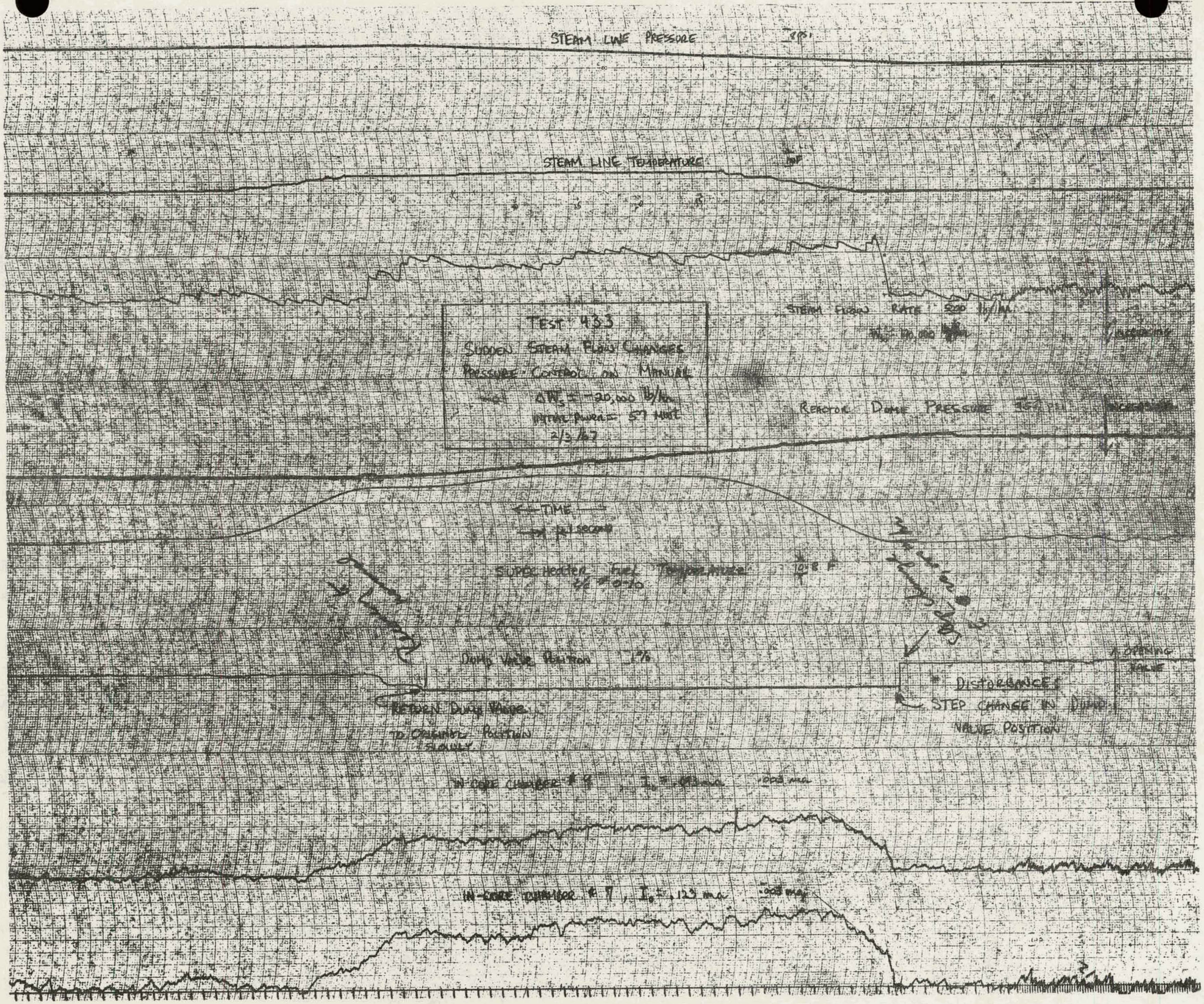
PRESSURE SET POINT CHANGE: 144 Mwt

FIG. 2.4



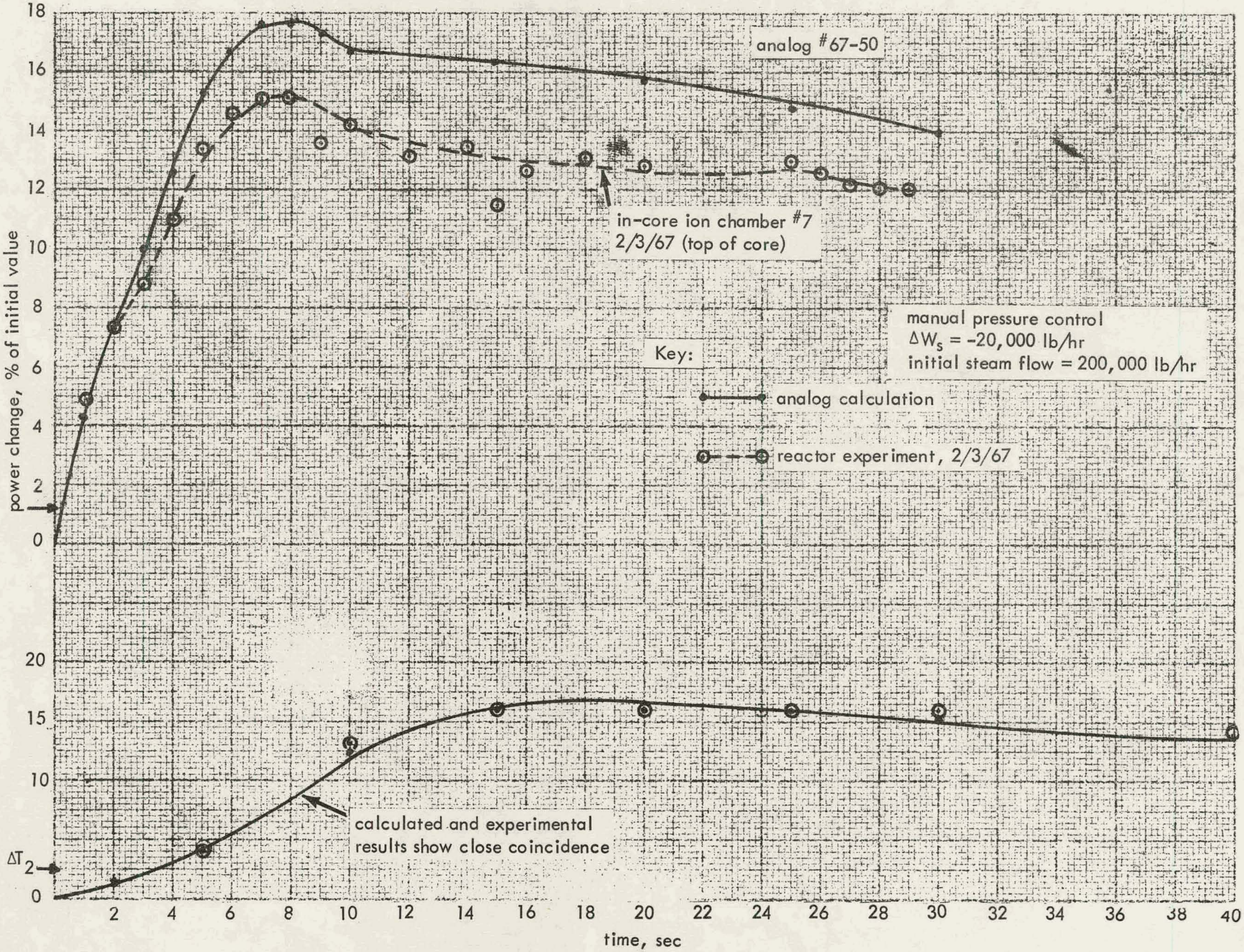
PRESSURE SET POINT CHANGE: 169 Mwt

FIG. 2.5



STEP CHANGE IN STEAM FLOW: 30% POWER

FIG. 2.6



COMPARISON OF ANALOG PREDICTION AND EXPERIMENT
FOR A STEP REDUCTION IN STEAM FLOW

FIG. 2.7

3. RECIRCULATION FLOW CHANGES

Changing recirculation flow rate in a boiling water reactor is one of the means available for varying reactivity and reactor power. In Pathfinder the flow can be varied by means of variable-position butterfly valves located in the external recirculation piping at the pump discharge. The actual plant layout is presented in an earlier document.(4) These valves are usually full open so that full pump output from all three recirculation pumps is available to the boiling core. The valves may be moved singly or in a ganged fashion to control recirculation flow. The response of the reactor power and other pertinent system parameters to the recirculation flow changes is presented in this section.

Table 3-1 lists the initial conditions for specific recirculation flow disturbances chosen as most descriptive of reactor system response. Figures 3.1 to 3.5 are the actual responses to these disturbances. In all these figures, except Fig. 3.5, the pressure control system was in the automatic mode of control.

3.1 RESPONSE AT 20 PERCENT NOMINAL POWER

Figure 3.1 shows the reactor system response to recirculation flow changes while nominally at 38 Mwt. Prior to the conditions shown on this chart, recirculation flow was slowly reduced by ganging all three butterfly valves from the 100 percent open to the 60 percent open position. The valves had also been opened slowly from the 60 percent to the 100 percent position before this particular transient was run. While at the reduced flow rate of 48,000 gpm, a heat balance was taken which showed that reactor power was 34.5 Mwt. This power level lies on a line in Fig. 433.8.9 of Appendix A, drawn from the origin of this figure to the 22.5 percent power - 62,000 gpm recirculation flow point (the point from which flow reduction started). This single point seems to indicate that, at higher powers, the power-recirculation flow ratio will be approximately 1.0 when flow is reduced from the 100 percent power - 100 percent recirculation flow point.

The transient on Fig. 3.1 shows that, as flow was increased by ganging open all three butterfly valves from 60 percent to 100 percent open in 124 sec, power, recirculation flow, and superheater fuel temperatures gradually increased with very little overshoot, and that they then settled to a final steady state with no attendant oscillatory behavior.

The comparison between calculated and experimental results is excellent; this is so, in part, because the calculated results were run on the analog computer after the experimental data were obtained in order to establish the proper initial conditions for the calculations. Analog computer results show that this rate of recirculation flow change (255 gpm/sec) is worth approximately 0.7¢/sec in reactivity insertion rate at this recirculation flow rate.

3.2 RESPONSE AT 40 PERCENT NOMINAL POWER

Figure 3.2 shows the reactor system response to a ganged motion of the discharge valves beginning at higher initial power level. The ganged valve opening in this case is only from 60 percent to 65 percent open. Note that this test was not as specified in the procedure for Test 433; the final ganged valve position was changed from 45 percent open to a final position of 60 percent open because it became apparent that the power-to-recirculation-flow scram would be reached during the transient. Since it was not necessary to observe another scram during the course of this test, and since system response information to 60 percent open appeared to be adequate, the procedure was changed accordingly. A heat balance at the 100 percent valve open position yielded 73.3 Mwt; at the 60 percent valve open position, a heat balance gave 60.8 Mwt.

Plotting these points on Fig. 433.8.11 (Appendix A) shows the low flow point to lie on a straight line between the 0 and 100 percent flow points; this same result was obtained at 38 Mwt, and it points to a power-flow ratio at higher powers as predicted.

Comparing the results at 38 Mwt and 76 Mwt, it appears that the same flow disturbance causes larger power changes at the higher power level. This is an expected effect, since more reactivity should be tied up in voids at the higher power, which will have the effect of causing larger power changes.

3.3 RESPONSE AT 60 PERCENT NOMINAL POWER

Figure 3.3 shows the effect of recirculation flow changes while at approximately 114 Mwt.

Starting at 100 percent flow and 114 Mwt, the recirculation flow was reduced by gang closing the discharge valves until the No. 11 valve was 76 percent open. Discharge valves on Pumps 12 and 13 were further closed individually to 67 and 71 percent open respectively; at this point the flow was 52,000 gpm and power was 104.9 Mwt. Group II control rods were controlling at 36 in.

For the measured disturbance, the valves were ganged open about 5 percent from these positions.

The results are similar to those obtained at 76 Mwt both during the transient and in terms of power changes after the transient. Very close comparisons between the two cases are not possible because initial valve positions were not the same, and a 5 percent movement in each case results in a different gpm change.

Comparison of Fig. 3.3 with ganged valve movement at other power levels shows about the same type of transient response. This chart shows about the same effects as those seen at 76 Mwt. There seems to be a slight upward shift in axial power shape when flow is increased, and there seems to be an even change of flow radially across the core (judging from the nearly identical response of Chambers 3 and 9 when the valves are ganged open).

Comparing the various responses to this disturbance for a number of initial powers, no trend or evidence of any instability is noticed when recirculation flow is changed.

3.4 RESPONSE AT 72 PERCENT NOMINAL POWER

Figure 3.4 shows the results of a recirculation flow change at the nominal 137 Mwt point. Starting at 137 Mwt (Group II rods at 37 in.), flow was slowly reduced by ganging the discharge valves closed and then making the flow approximately equal in all loops by individually trimming the flows in each loop. At this point, power was 117.5 Mwt and flow was 49,700 gpm. The valve movement was a ganged movement to a 5 percent further open position.

3.5 RESPONSE WITHOUT PRESSURE CONTROL

Figure 3.5 shows the reactor system response at 38 Mwt to a recirculation flow change while the pressure control system was on manual. This disturbance was a large one, and it shows that even under a transient of this nature the reactor is very stable. In addition, a comparison of this transient with the transient in Fig. 3.1 shows the important role the pressure control system plays in plant operation.

This disturbance began with the plant at the identical initial conditions as shown in Fig. 3.1, except that the pressure control system was on manual. Initial power was 34.5 Mwt and initial recirculation flow was 48,000 gpm.

These results show the inherent stability of the reactor at this power level. The transient was ended, prior to complete opening of the butterfly valves, by manual opening of the dump valve in order to prevent reactor steam line pressure from rising to the scram trip point. Figure 3.5 clearly shows the smooth response of reactor power to recirculation flow changes. Flow is not recorded on this figure, but the flow response is identical to that on Fig. 3.1; valve opening positions are marked on Offner Channel #2 (which is in-core ion chamber current #9). Comparing in-core and out-of-core chamber responses, it can be seen that a substantial difference in indication exists between the two. This is some evidence of the decalibration effect that exists when recirculation flow is changed.

3.6 OBSERVATIONS OF POWER DISTRIBUTION

In other recirculation flow tests (charts not shown in this report), results show that the core exit is most greatly affected during the flow change (max Δ powers are: in-core chambers #7 = +9.5 percent; #8 = +8.7 percent; #9 = +6.1 percent) and that in the final steady state the core exit power density shows the largest increase (final steady-state Δ powers are: in-core chambers #7 = +6.1 percent; #8 = +4.7 percent; #9 = +3.8 percent). This indicates a slight upward shift in axial power shape.

Other identical changes in flow were done with a radial set of ion chambers recorded. In-core chambers #3, #6, #9 (each 18 in. from the bottom of the core) all indicated peak powers of about +8.5 percent during the transient, indicating that the radial distribution of flow is apparently unchanged during the transient.

TABLE 3-1

INITIAL PLANT CONDITIONS FOR RECIRCULATION FLOW CHANGES

Reactor Power	34.5 Mwt (8/5/66)	60.8 Mwt (10/20/66)	104.9 Mwt (12/4/66)	117.5 Mwt (12/20/66)	34.5 Mwt (8/5/66)
Channel 5 Current	----	0.60×10^{-6} amp	0.155×10^{-5} amp	0.205×10^{-5} amp	----
Steam Flow	124,500 lb/hr	215,000 lb/hr	365,000 lb/hr	400,000 lb/hr	124,500 lb/hr
Feedwater Temp.	367 F	387 F	376 F	374 F	367 F
Total Recirculation Flow	48,000 gpm	47,500 gpm	52,000 gpm	49,700 gpm	48,000 gpm
Superheater Fuel Temp. t/c #O-10	692 F	751 F	748 F	740 F	692 F
Exit Steam Temp.	554 F	625 F	640 F	655 F	554 F
Reactor Pressure	540 psig	540 psig	551 psig	565 psig	540 psig
Dump Valve Position	----	18%	8.5%	7%	----
Inlet Valve Position	0%	0%	26%	41%	----
Pressure Control	Auto on Dump Valve	Auto on Dump Valve	Auto on Dump Valve	Auto on Dump Valve	Manual
Level Control	Auto	Auto	Auto	Auto	Auto
Feedwater Temp. Control	Manual	Auto	Auto	Auto	Manual

TABLE 3-2

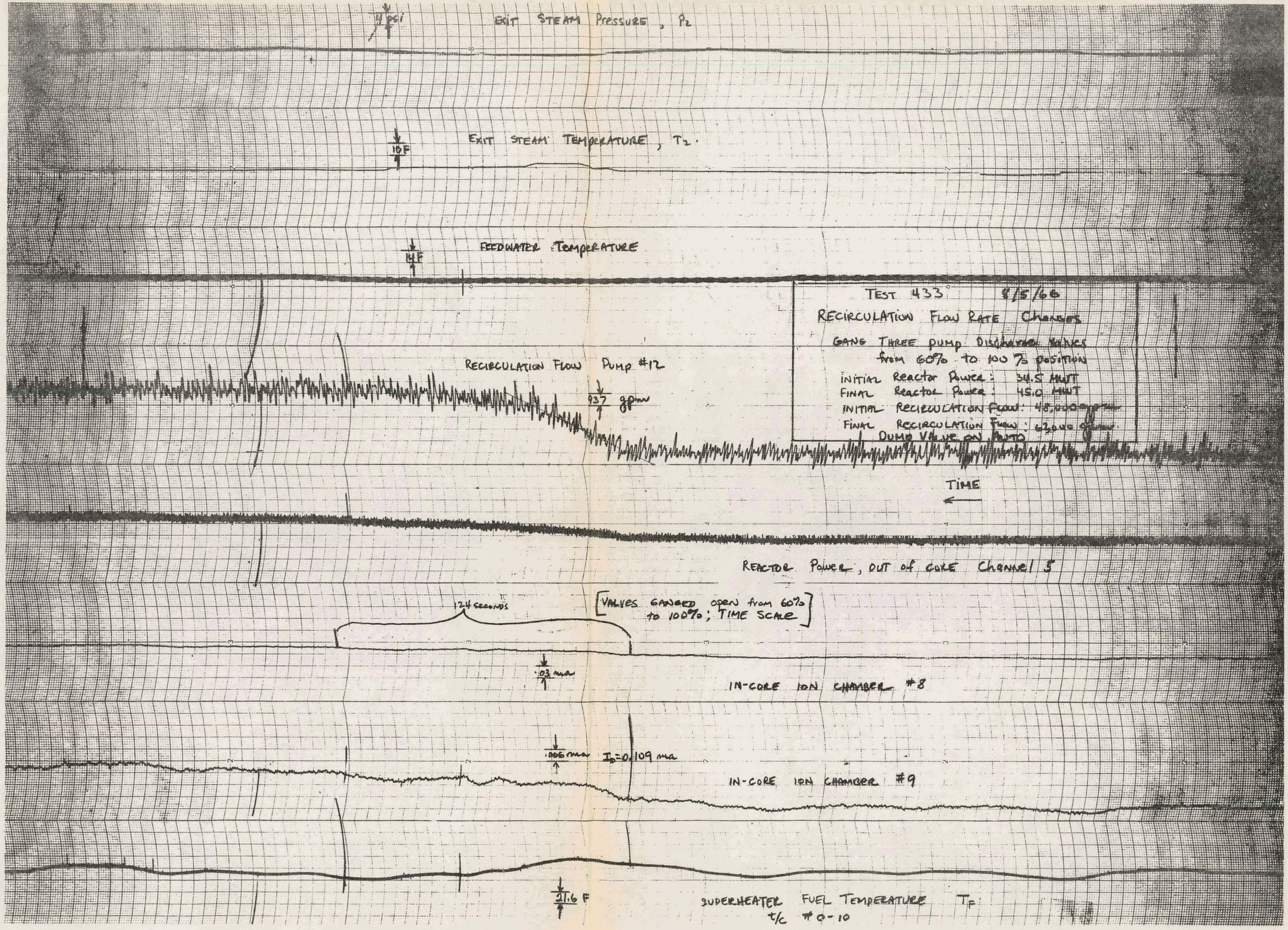
SUMMARY OF REACTOR SYSTEM RESPONSES TO RECIRCULATION FLOW CHANGES

Reference Figure	Fig. 3.1	Fig. 3.2	Fig. 3.3	Fig. 3.4	Fig. 3.5
Initial Reactor Power	34.5 Mwt (8/5/66)	60.8 Mwt (10/20/66)	104.9 Mwt (12/4/66)	117.5 Mwt (12/20/66)	34.5 Mwt 8/5/66 Manual P_2 Control
Disturbance	+14,000 gpm 124 sec	+4780 gpm in 25 sec	+2100 gpm in 17 sec	+2800 gpm in 25 sec	+13,000 gpm in 100 sec
Max Δ Power on in-core #7, % of existing power	+24.3% (in-core #8)	+15.9%	+7.5%	+9.4%	+35.3% (in-core #8)
Max Δ T_F on t/c #O-10	+13 F	+34.6 F	+18 F	+18 F	+97 F
Max Δ W_s	----	----	+11,000 lb/hr	+30,000 lb/hr	----
Max Δ T_2	+6 F	+14.5 F	+10 F	+8 F	+23 F
Controlling Rod Group and Position	Group III at 48.7 in.	Group II at 13 in.	Group II at 35.7 in.	Group II at 37 in.	Group III at 48.7 in.

TABLE 3-3

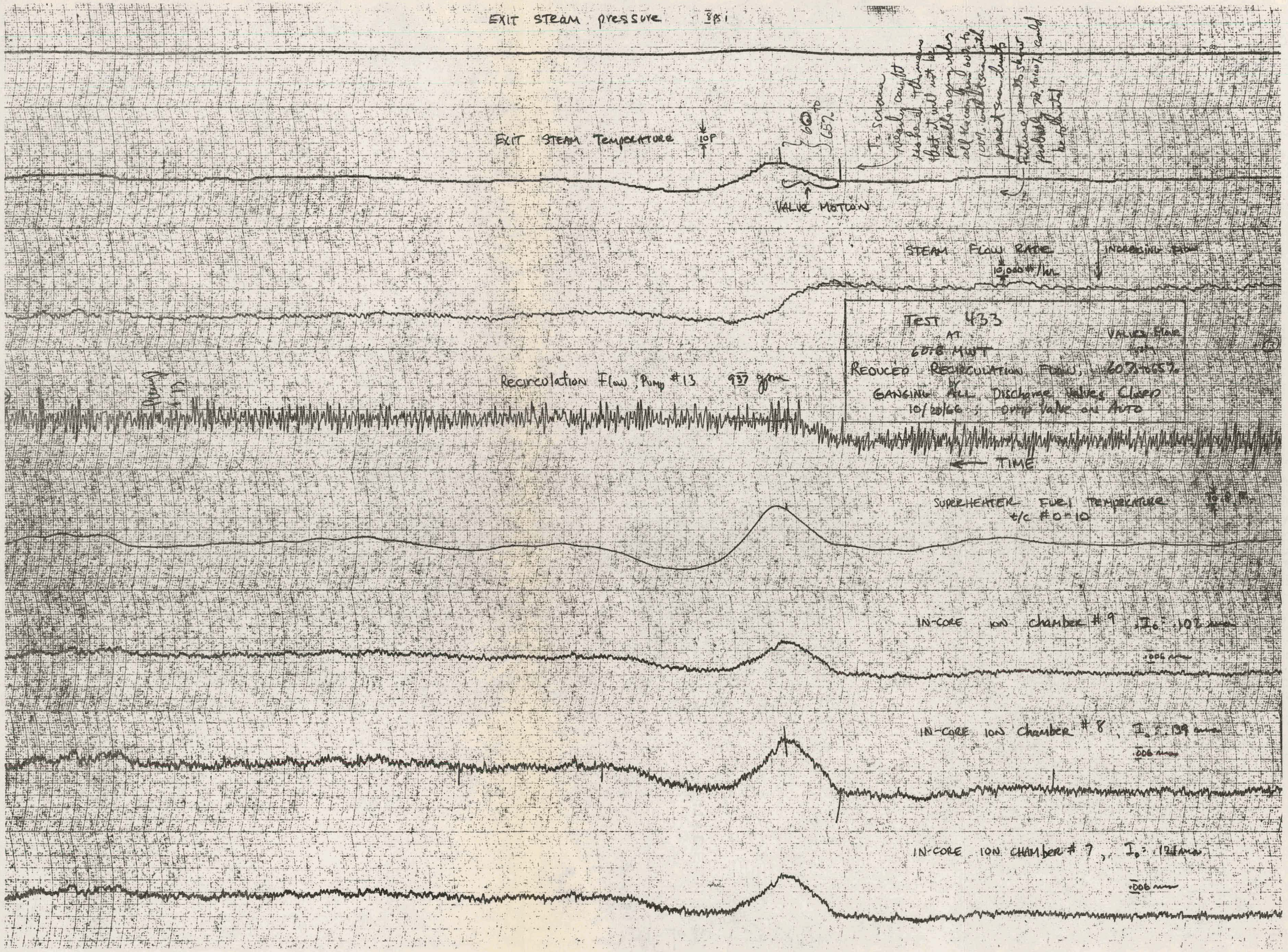
COMPARISON OF TRANSIENT PARAMETERS -
CALCULATED PREDICTIONS AND EXPERIMENTAL RESULTS

	Disturbance	Initial Power Level	Max Δ Power % of Initial Power	Max ΔT_F -Calculated (hot spot) -Experiment (t/c #O-10)	Max ΔT_2	Max ΔP_2
Calculated Predictions	+14,000 gpm in 130 sec	45 Mwt	+20%	+18 F	+6 F	+2 psi
Experimental Results	+14,000 gpm in 124 sec	34.5 Mwt	+24.3% on in-core Chamber #8	+13 F	+6 F	+2 psi



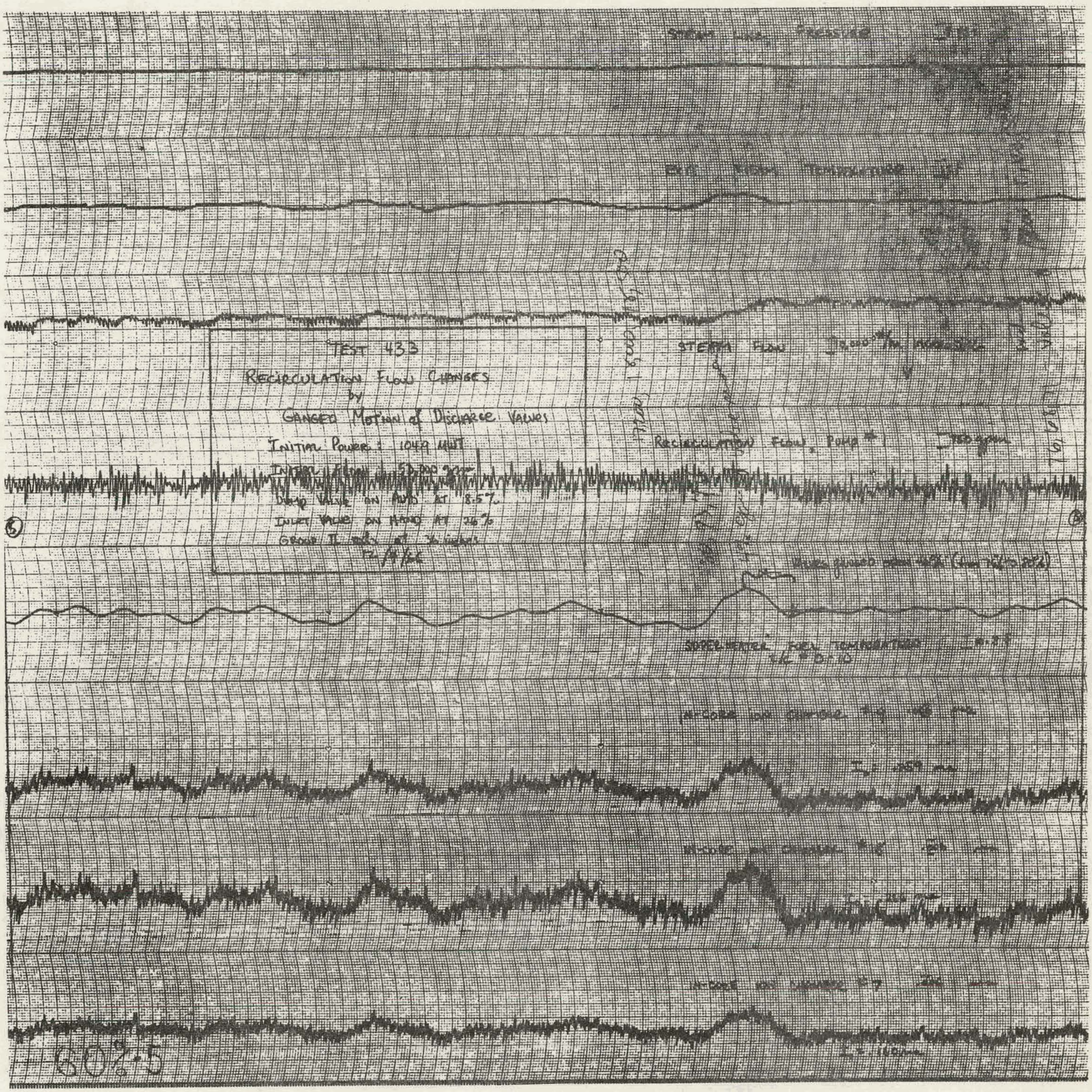
RECIRCULATION FLOW RATE CHANGE: 34.5 Mwt

FIG. 3.1



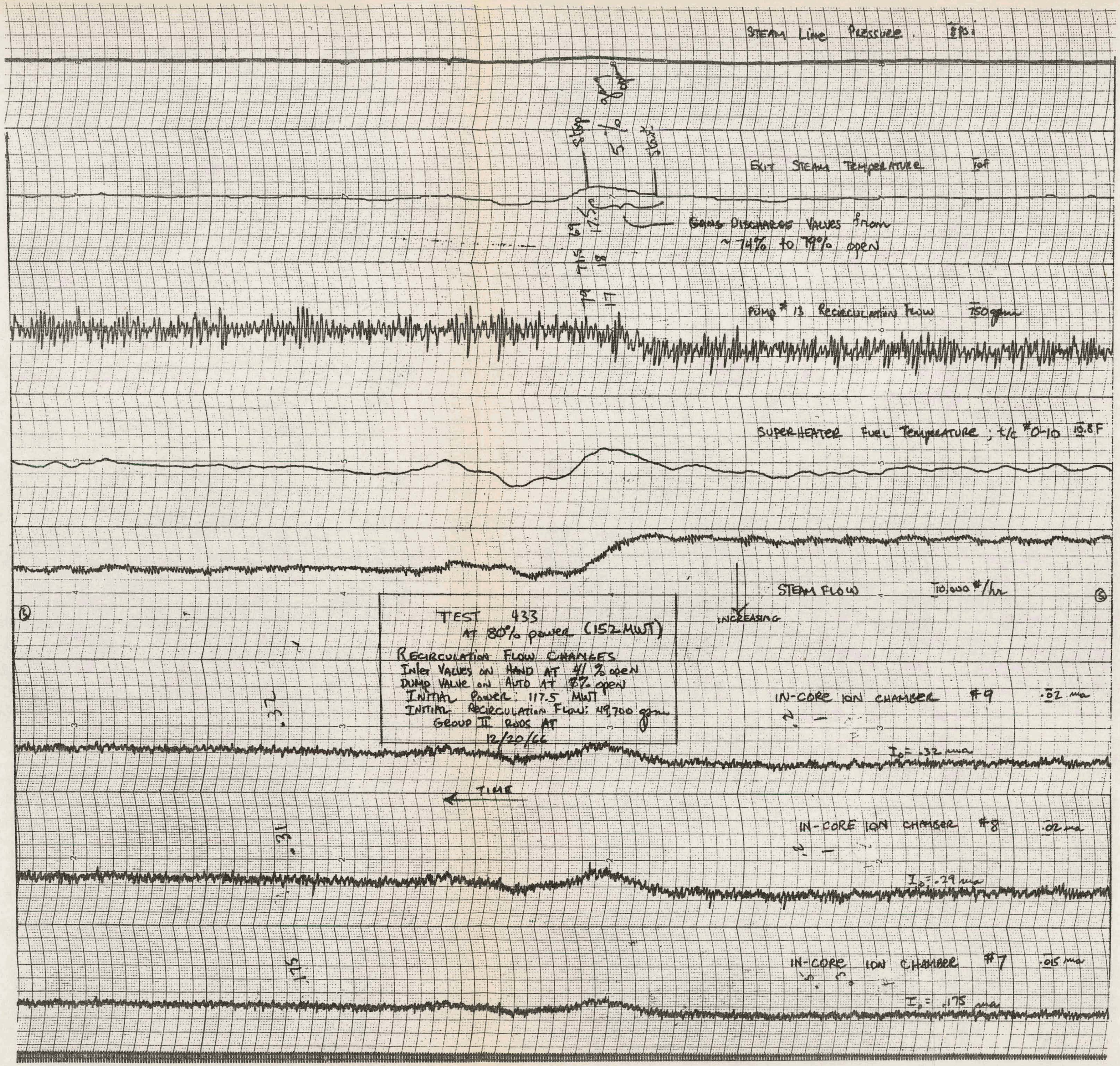
RECIRCULATION FLOW RATE CHANGE: 60.8 Mwt

FIG. 3.2



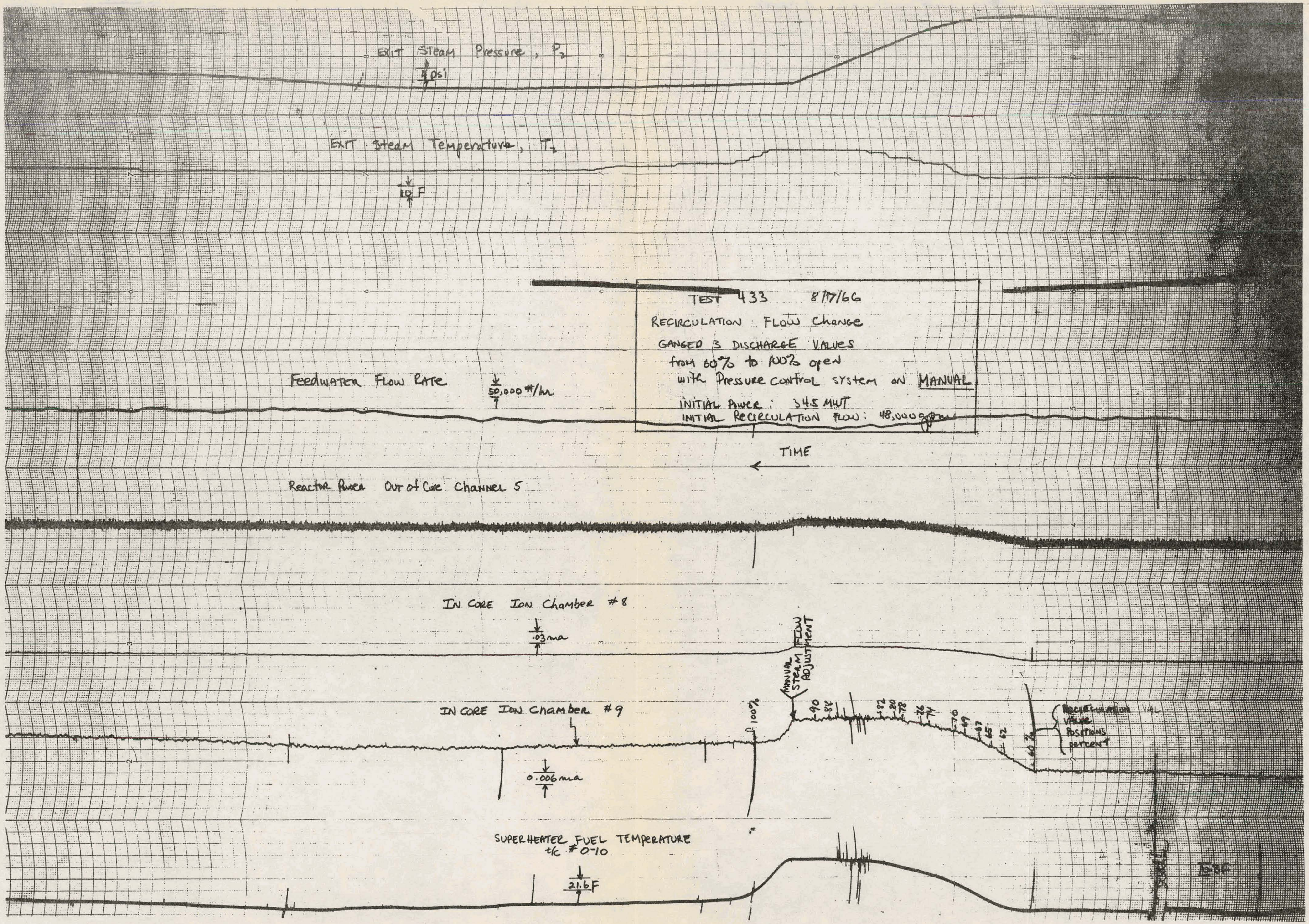
RECIRCULATION FLOW RATE CHANGE: 104.9 Mwt

FIG. 3.3



RECIRCULATION FLOW RATE CHANGE: 117.5 Mwt

FIG. 3.4



RECIRCULATION FLOW RATE CHANGE,
MANUAL PRESSURE CONTROL: 34.5 Mwt

FIG. 3.5

4. RECIRCULATION PUMP TRIP TESTS

One of the original design objectives for Pathfinder was to be able to operate the reactor at reduced powers with only two and possibly only one recirculation pump operating. Further, it was hoped that accidental tripping of one and possibly two recirculation pumps could be tolerated by the reactor and could be accomplished without reactor shutdown. The pump trip tests were designed to test the reactor during these transients.

Prior to each test, a reactor heat balance was taken and the instrumentation was prepared for the transient. The pumps were switched off by manual removal of electrical power from the pump motor.

4.1 RESULTS OF ONE-PUMP TRIPS

Three attempts were made during Test 433 to trip a recirculation pump while the reactor was producing power. The first attempt ended in a scram due to high steam temperature approximately 21 sec after the pump was tripped. In this case, bulk exit steam temperature was approximately 40 F below the initial value at the time of scram; but the reactor operator failed to keep steam temperature within range, and a high T_2 scram resulted.

The second recirculation pump trip was performed with the low steam temperature out-of-range runback bypassed and with the high steam temperature scram set at +25 F above the initial steam temperature. However, high steam line temperature (T_2) scrammed the reactor at 58 sec after the pump was tripped. The power response was not as calculated in pre-test calculations because the transient calculations did not account for backflow through the tripped pump. Just after a pump is tripped, its discharge valve is wide open, and the shutdown loop partially short-circuits the core of coolant flow. As time progresses, the discharge valve on the tripped pump automatically begins to close, reducing the effect of the core short-circuit and increasing core flow rate, which has a positive reactivity effect. Thus, the experimental results yielded a larger power dip after the pump trip than had been expected, and a higher power overshoot followed the power dip.

The third attempt to trip a recirculation pump was carried through to completion, not being interrupted by a scram. The results of this pump trip test are presented in this report. Figure 4.1 shows the response of various reactor parameters during this pump trip. Prior to this pump trip, recirculation flow was reduced by closing all three discharge valves from 100 percent to 60 percent open in order to reduce the backflow that would occur through the tripped pump. A heat balance was then taken, and the initial conditions for the one pump trip were as listed in Table 4-1.

The results shown in Fig. 4.1 confirm that the reactor can easily withstand this pump trip transient. However, there was indication that the net flow acceleration occurring in the core during termination of the backflow might exceed technical specification limits in the case of valves that were initially fully open. This consideration, together with the general caution exercised in operation of Pathfinder, led to the decision to install temporary scrams on loss of flow in any pump, and to proceed in the power escalation without further pump trip tests at this time.

4.2 ANALYSIS OF RESULTS

Figure 4.2 shows the comparison of the reactor power response for this pump trip test and for a prior test in which pump discharge valves were initially 100 percent open. This comparison shows the effect of backflow subsequent to a pump trip.

Figure 4.3 shows analog computer results (from which the data in Table 4-2 was extracted). Many other analog studies with different flow disturbances, including the backflow phenomenon, were also performed. The comparison between the successful pump trip and the analog computer study shown in Fig. 4.2 is the best comparison obtained between measured and calculated pump trips. In order to more closely reproduce the test data, only two modifications were made to the analog simulator. First, the actual flow trace from the #11 tripped pump, including the indicated backflow on this trace, was used as the input disturbance for the computer. Second, the pressure control system settings on the computer were adjusted to approximate settings actually used during the test. Prior analog computer runs had not used backflow and, most importantly, had used a much larger value of reset than was actually used on the pressure control system during the test. Also, it was later determined that the T_2 bias was improperly set during the test; this was simulated by removal of the small T_2 bias from the computer model.

Close similarity is noted between the analog and actual test results for power and fuel temperature, both in transient and final steady-state values. The transient performance of the computer could be made to represent the test results by further adjustments to the flow disturbance and to the control system, but available computer time limited further studies in this regard. The test results indicate a slightly higher and larger power rise between 60 and 80 sec; this may have been partially caused by a third phenomenon, but one that has second-order effects. This third phenomenon is an overabundance of feedwater being supplied to the primary system because the level control system was on automatic level control alone. When the pump was tripped, the water level was observed to fall. In turn, this caused the feedwater control system to supply more feedwater than was actually necessary, since vessel water inventory had not changed. This "extra" feedwater would cause subcooling to increase, which would have the effect of increasing power and fuel temperatures until the water level tended to return to the former equilibrium value.

The second rise in power on the analog simulator (between 90 and 160 sec) was undoubtedly due to the manner in which the control system dynamics were simulated.

One of the important goals of Test 433 was to determine how measured fuel temperature transients compared with calculated transient fuel temperatures. Table 4-3 shows a comparison of calculated and measured temperatures for the pump trip shown in Fig. 4.1. Note that calculated temperatures are hot spot superheater temperatures (initial hot spot temperature = 1270 F), and that the measured hot spot temperatures are based on temperatures recorded for thermocouple #O-10. The +115 F transient peak temperature actually measured on thermocouple #O-10, when corrected for the fact that this thermocouple is not located exactly at the hot spot, and when heat transfer uncertainties are accounted for, yields a hot spot transient temperature rise of +265 F. This compares favorably with the calculated +300 F rise for the hot spot transient temperature.

Figure 4.4 is a plot of reactor power versus recirculation flow. Flow was decreased to 38,000 gpm in several steps, as indicated on the figure; and a heat balance was taken when each steady-state was obtained.

TABLE 4-1

INITIAL PLANT CONDITIONS FOR ONE PUMP TRIP TEST

Reactor Power	58.1 Mwt
Group I Rods Controlling at	33.5 in.
Channel 5 Current	0.63×10^{-6} amp
Steam Flow	210,000 lb/hr
Feedwater Temperature	380 F
Total Recirculation Flow	49,800 gpm
Superheater Fuel Temperature t/c #O-10	727 F
Exit Steam Temperature	580 F
Reactor Pressure	543 psig
Dump Valve Position	18% open
Inlet Valve Position	0% open
Pressure Control	Auto, on dump valve
Level Control	Auto, single element
Feedwater Temperature Control	Auto

TABLE 4-2

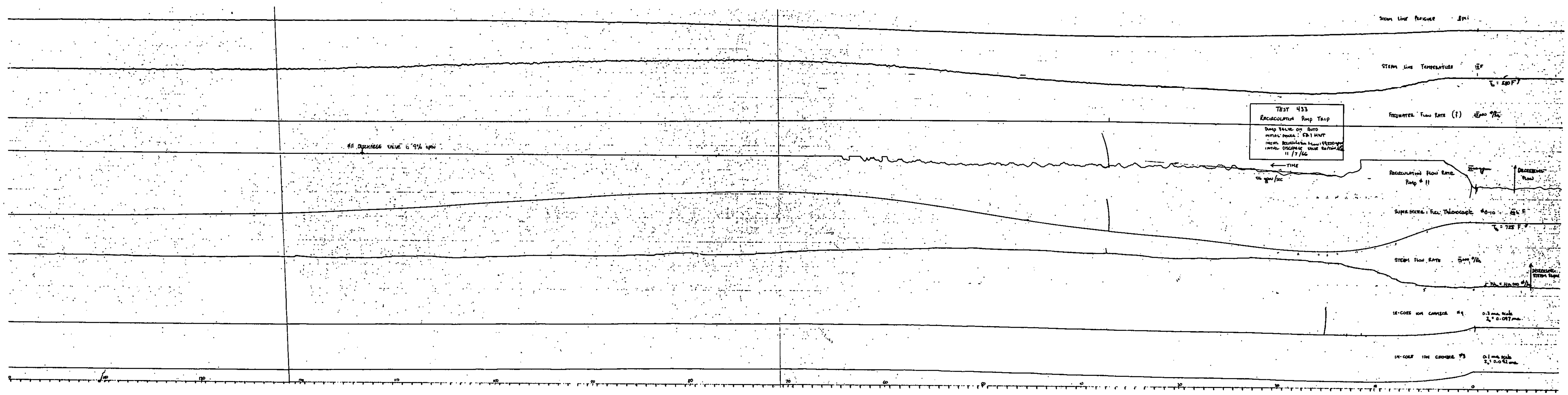
REACTOR RESPONSE TO ONE PUMP TRIP TEST AND COMPARISON WITH CALCULATION

	Test Results	Calculations, Analog Case 2,66-27
Initial Power	58.1 Mwt	58.1 Mwt
Disturbance	Tripped Pump #11	Tripped one recirculation pump
Max. Power Change	-56% of initial power on in-core Chamber #3	-50% of initial power
Final Steady State Power	74.9% of initial power with valves of two operating pumps at 60%	77% of initial power with valves of operating pumps assumed to be 100% open
Final Flow, Valves at 6%, 60%, 60%	38,000 gpm	40,000 gpm, valves at 6%, 100%, 100%
Max. positive rate of change of flow	+416 gpm/sec	+432 gpm/sec
Max. Changes in Fuel Temperature	-152 F at 16 sec -115 F at 74 sec measured on t/c #O-10	-420 F at 15 sec -300 F at 55 sec (calculated hot spot temperatures)
Max. Changes in Steam Line Temperature	-34 F at 16.5 sec and +27 F at 75 sec	-130 F at 21 sec +80 F at 65 sec (superheater rods assumed all out)
Pressure Control System	Auto on dump valve reset ≈ 2.5 repeats/min	Auto on dump valve reset = 2.5 repeats/min

TABLE 4-3

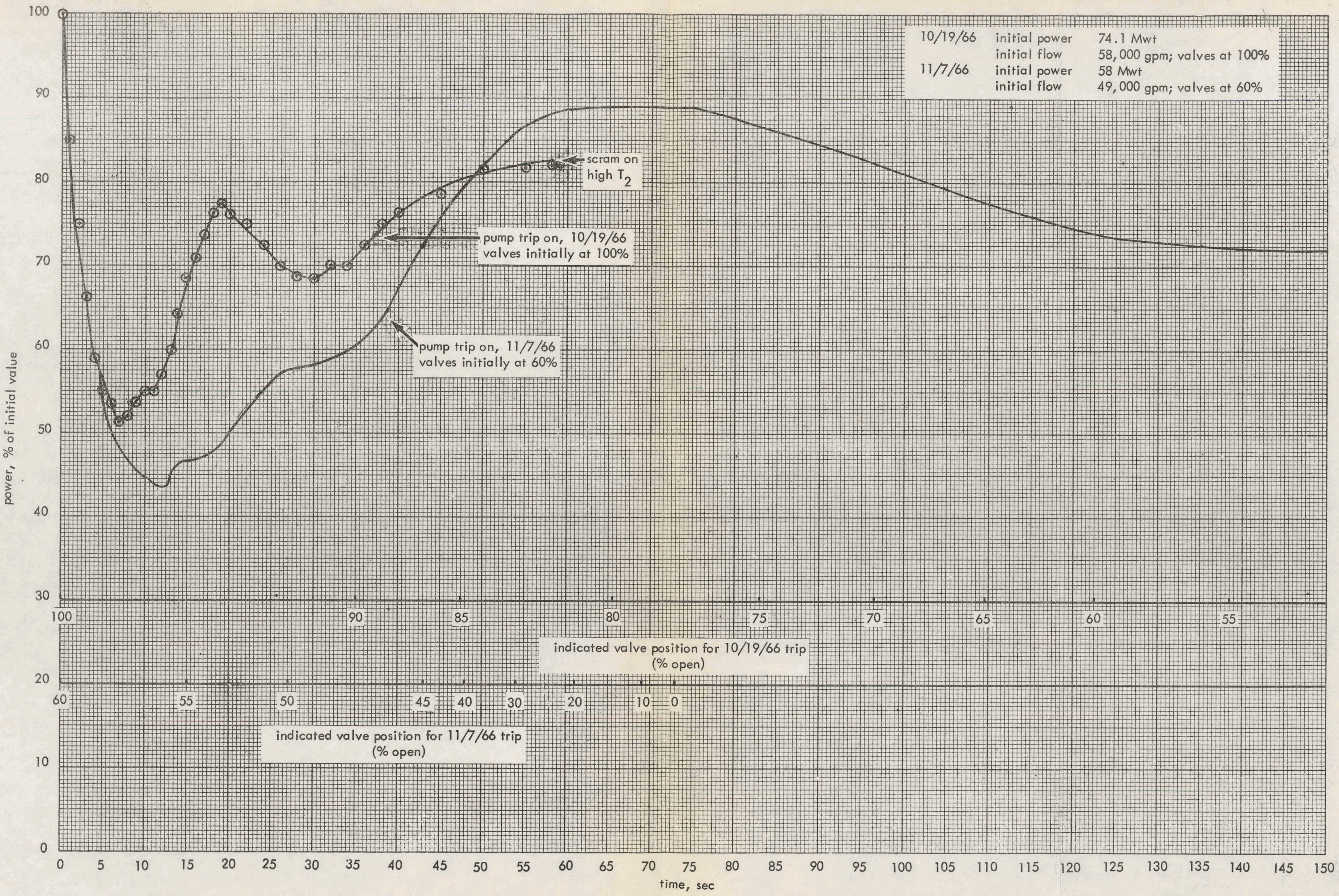
COMPARISON BETWEEN CALCULATED AND MEASURED MAXIMUM TRANSIENT TEMPERATURES DURING A ONE PUMP TRIP AT ≈60 MWT

	Max Hot Spot Temperature Change With Uncertainties ΔT_F	Max Transient Hot Spot Temperature With Uncertainties $T_{Fo} + \Delta T_F$	Max Expected Hot Spot Without Uncertainties $T_{Fo} + \Delta T_F$
Calculated (analog)	+300 F	1570 F	--
Measured, based on a +115 F peak on t/c #O-10	+265 F	1275 F	1080 F



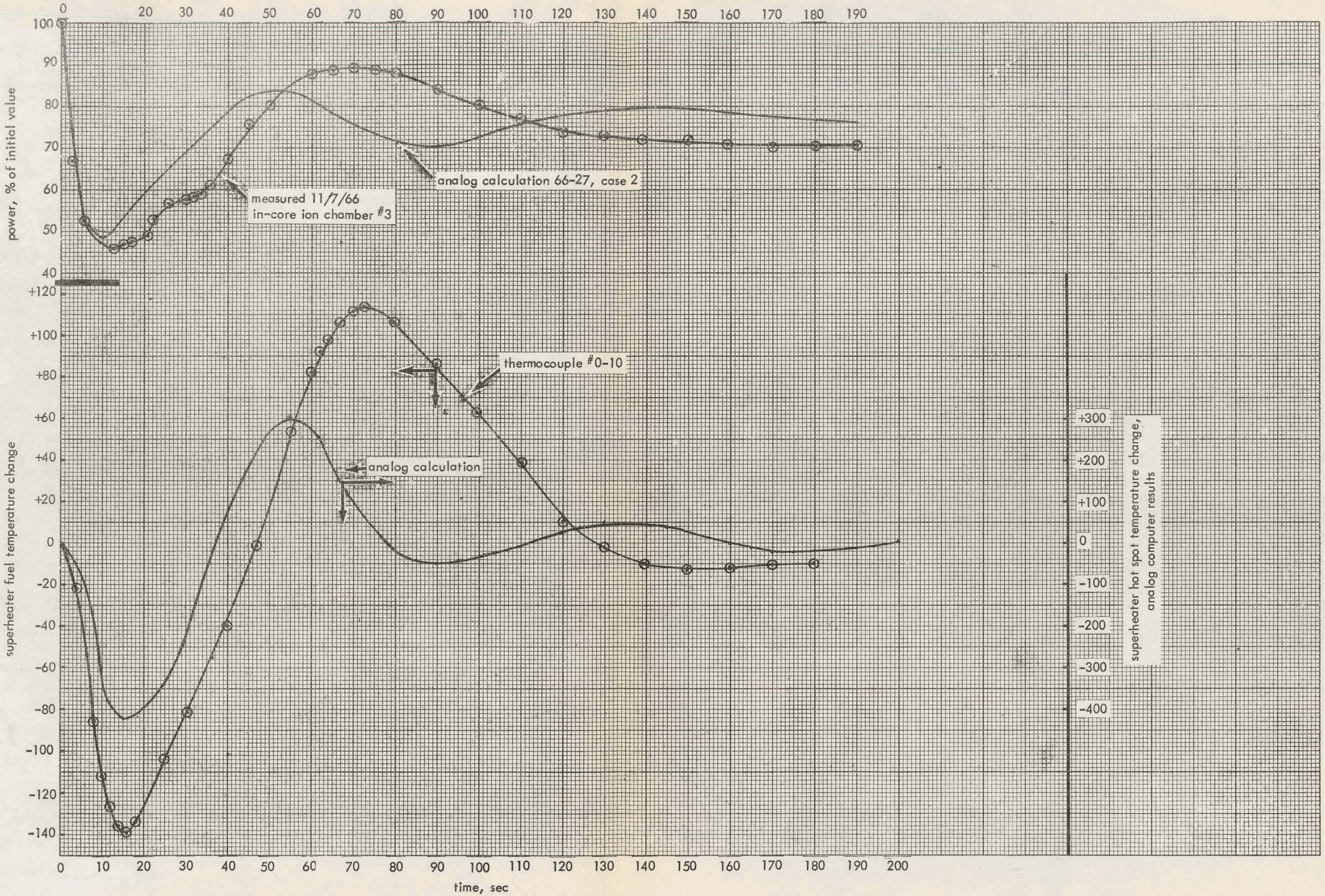
RECIRCULATION PUMP TRIP: 58 Mwt

FIG. 4.1



PUMP TRIP COMPARISONS: REACTOR POWER VS. TIME

FIG. 4.2



COMPARISON OF CALCULATED AND MEASURED
PUMP TRIP RESPONSE

FIG. 4.3

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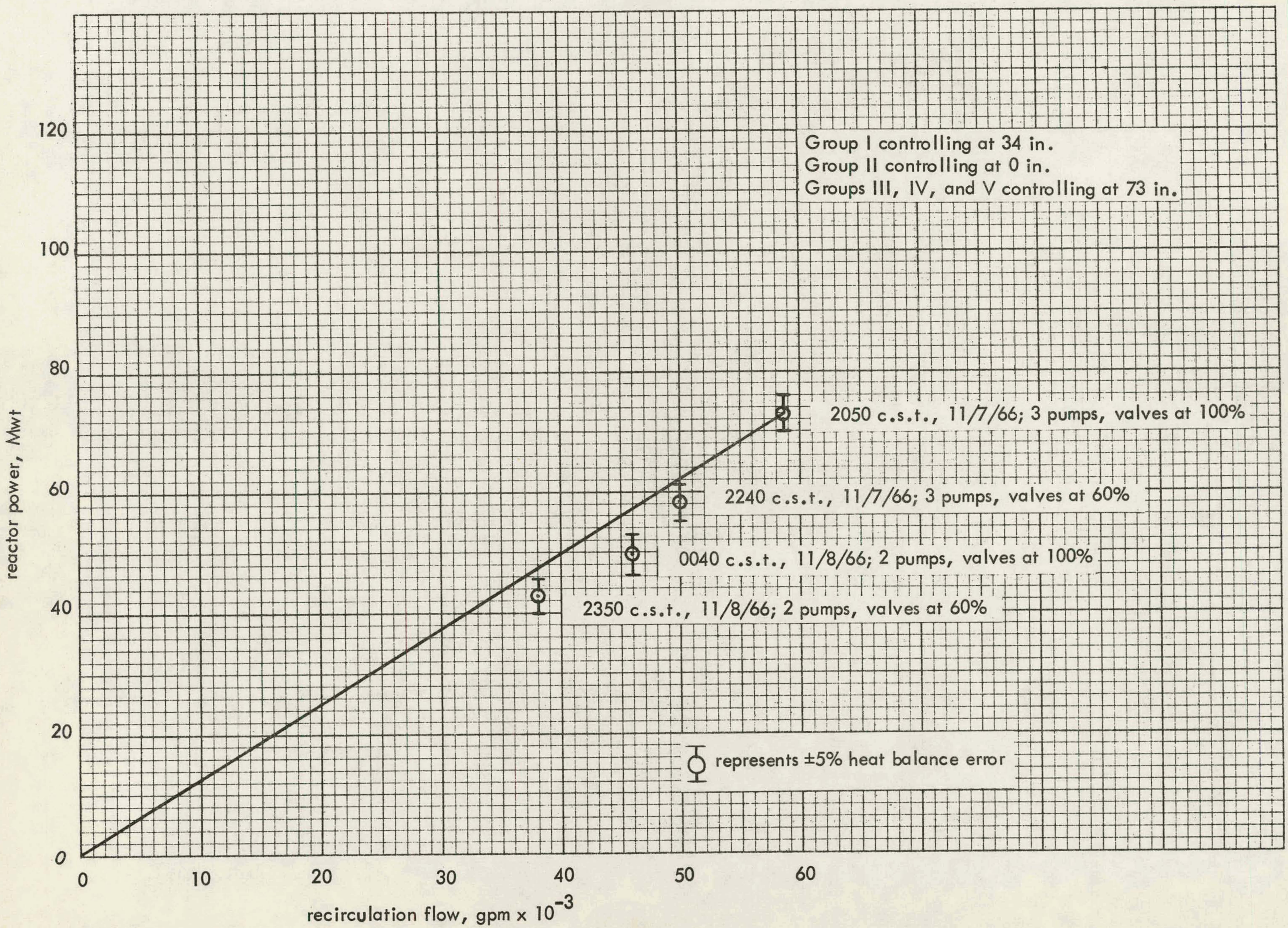

REACTOR POWER VS. RECIRCULATION FLOW AT 76 Mwt

FIG. 4.4

5. RESPONSE TO FEEDWATER FLOW CHANGES

In order to measure the effects of varying core inlet subcooling, both the feedwater flow rate and feedwater temperature were changed at various reactor initial conditions. This section of the report deals with feedwater flow rate changes.

The flow changes were accomplished by placing the feedwater system on manual and then throttling the feedwater pump discharge valves. The actual reactor system responses are shown in Figs. 5.1 through 5.5. In these figures, the pressure control system was in automatic mode at all power levels. The reactor response was stable after the disturbance was introduced.

Each of the disturbances discussed in this section shows feedwater flow rate initially decreasing, causing reactor water level to fall. Prior to reaching the low water level runback trip, the feedwater flow rate was manually increased in order to restore water level to its initial value.

Table 5-1 lists the initial conditions for the disturbances described in this report, and Table 5-2 summarizes the parameter transient responses to the feedwater flow changes.

In all the cases studied, at each power level step, the actual reactor response to the feedwater flow changes was about half of pre-test calculated responses for disturbances of the same magnitude. Possible reasons for this discrepancy include operating at higher initial values of feedwater heating than the calculations assumed. For example, the calculations assumed a 340 F feedwater temperature, while the actual tests were performed with feedwater temperature near 380 F. This higher value of feedwater temperature means less subcooling and consequently a smaller change in reactivity as the core inlet temperature is varied. Other explanations of the discrepancy include the possibility of a magnitude error in the reactivity-feedwater flow rate relationship in the calculational model.

It was noted, however, that the measured reactor system responses decreased as power increased, for a fixed magnitude of the disturbance. (See Table 5-2.) This effect is due to the fact that the 100,000 lb/hr change in feedwater flow becomes a much smaller part of the initial feedwater flow as power is increased.

TABLE 5-1

INITIAL PLANT CONDITION FOR FEEDWATER FLOW RATE CHANGES

Reactor Power Date	39 Mwt 8/5/66	76 Mwt 10/19/66	114 Mwt 12/3/66	142 Mwt 12/18/66	169 Mwt 1/6/67
Channel 5 Current	0.52×10^{-6} amp	0.87×10^{-6} amp	0.185×10^{-5} amp	0.30×10^{-5} amp	0.34×10^{-5} amp
Steam Flow	150,000 lb/hr	260,000 lb/hr	375,000 lb/hr	530,000 lb/hr	582,500 lb/hr
Feedwater Temperature	375 F	371 F	378 F	374 F	370 F
Total Recirculation Flow	~60,000 gpm	59,500 gpm	57,500 gpm	55,500 gpm	54,700 gpm
Superheater Fuel Temp. t/c #O-10	692 F	756 F	747 F	765 F	765 F
Exit Steam Temperature		625 F	638 F	642 F	658 F
Reactor Pressure	535 psig	540 psig	552 psig	580 psig	584 psig
Dump Valve Position	-	22%	8%	6%	6%
Inlet Valve Position	-	0%	28.5%	48%	53%
Pressure Control	Auto on dump valve	Auto on dump valve	Auto on dump valve	Auto on dump valve	Auto on dump valve
Level Control	Manual	Manual	Auto	Auto	Auto
Feedwater Temperature Control	Manual	Manual	Auto	Auto	Auto

TABLE 5-2

SUMMARY OF REACTOR RESPONSE TO FEEDWATER FLOW CHANGES

Reference Figure / Initial Reactor Power	Fig. 5.1 39 Mwt (8/5/66)	Fig. 5.2 76 Mwt (10/19/66)	Fig. 5.3 114 Mwt (12/3/66)	Fig. 5.4 137 Mwt (12/18/66)	Fig. 5.5 169 Mwt (1/6/67)
Disturbance in W_{FW}	-105,000 lb/hr in 11 sec	-100,000 lb/hr in 4 sec	-137,500 lb/hr in 4 sec	-100,000 lb/hr in 5.5 sec	-90,000 lb/hr in 3 sec
Max ΔPower, % of Initial Power	-9% on in-core Chamber #9	-6.8% on in-core Chamber #9	-6.3% on in-core Chamber #7	-5.1% on in-core Chamber #7	-2.8% on in-core Chamber #9
Max ΔT_F on t/c #O-10	-28 F	-19.5 F	-17 F	-15 F	-8 F
Max ΔT_2	-4 F	-8 F	-8 F	-6 F	-4 F

NOTE: Time reference in "Disturbance" above refers to time taken to change feedwater system manual loader.

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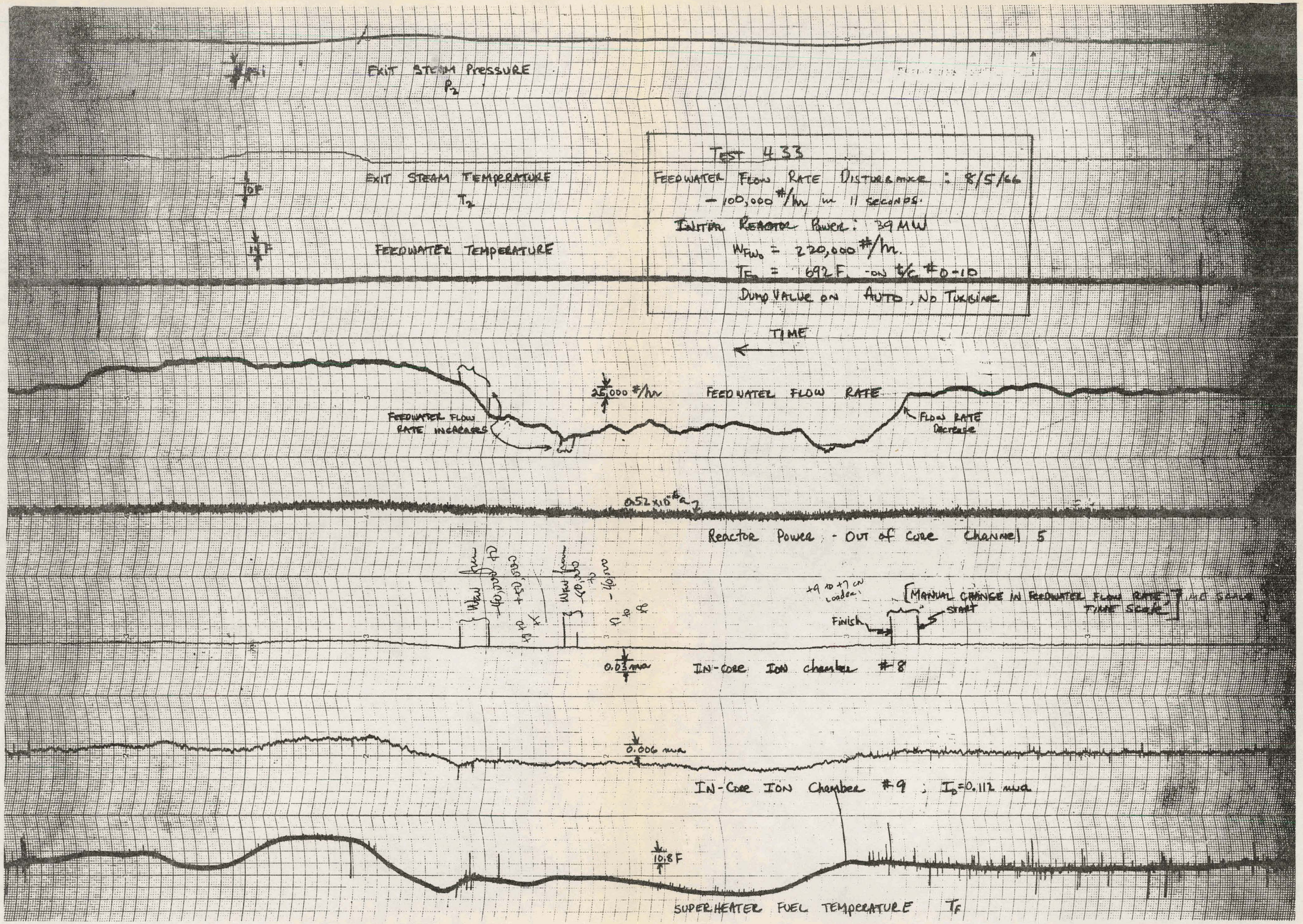
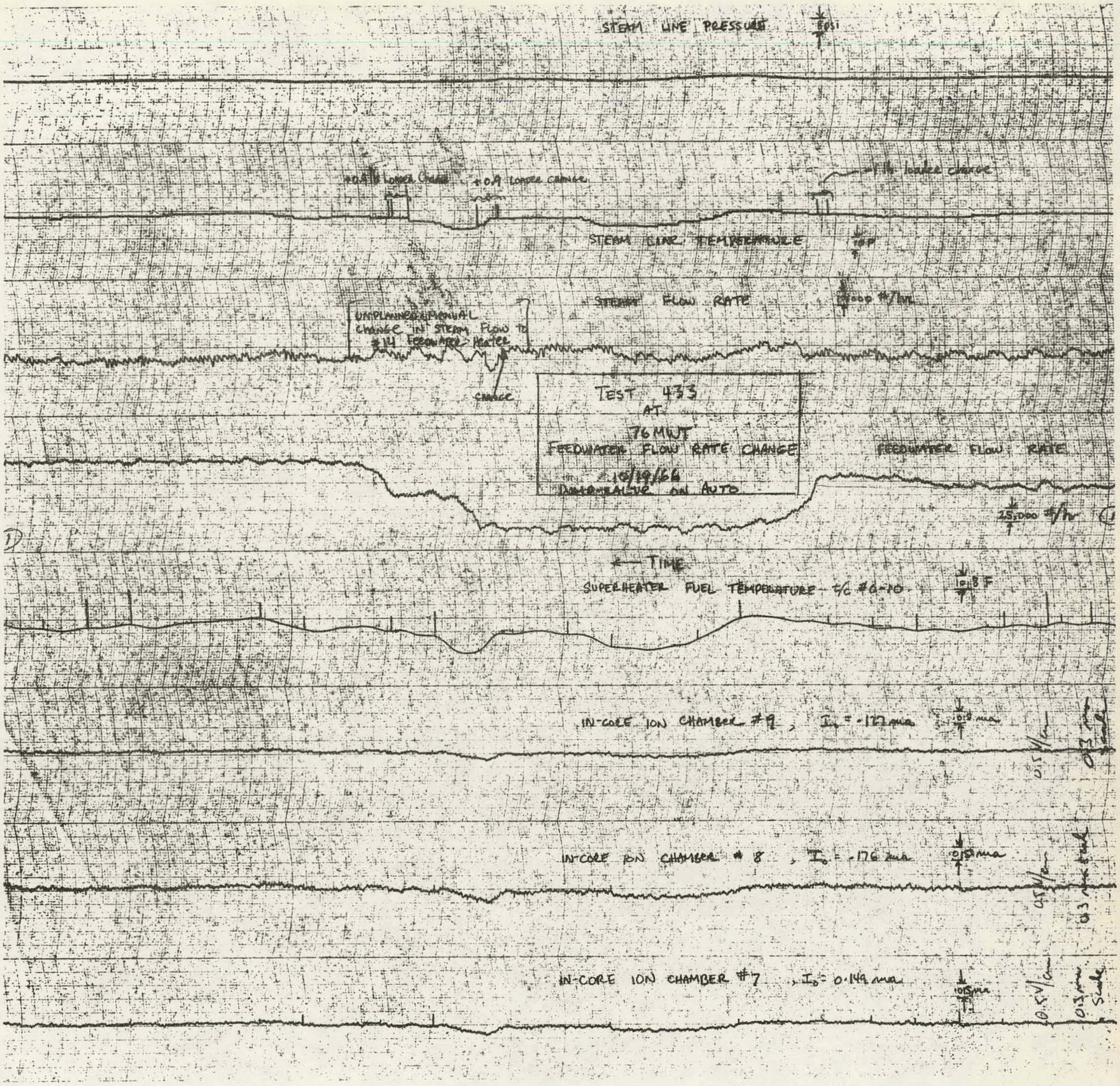


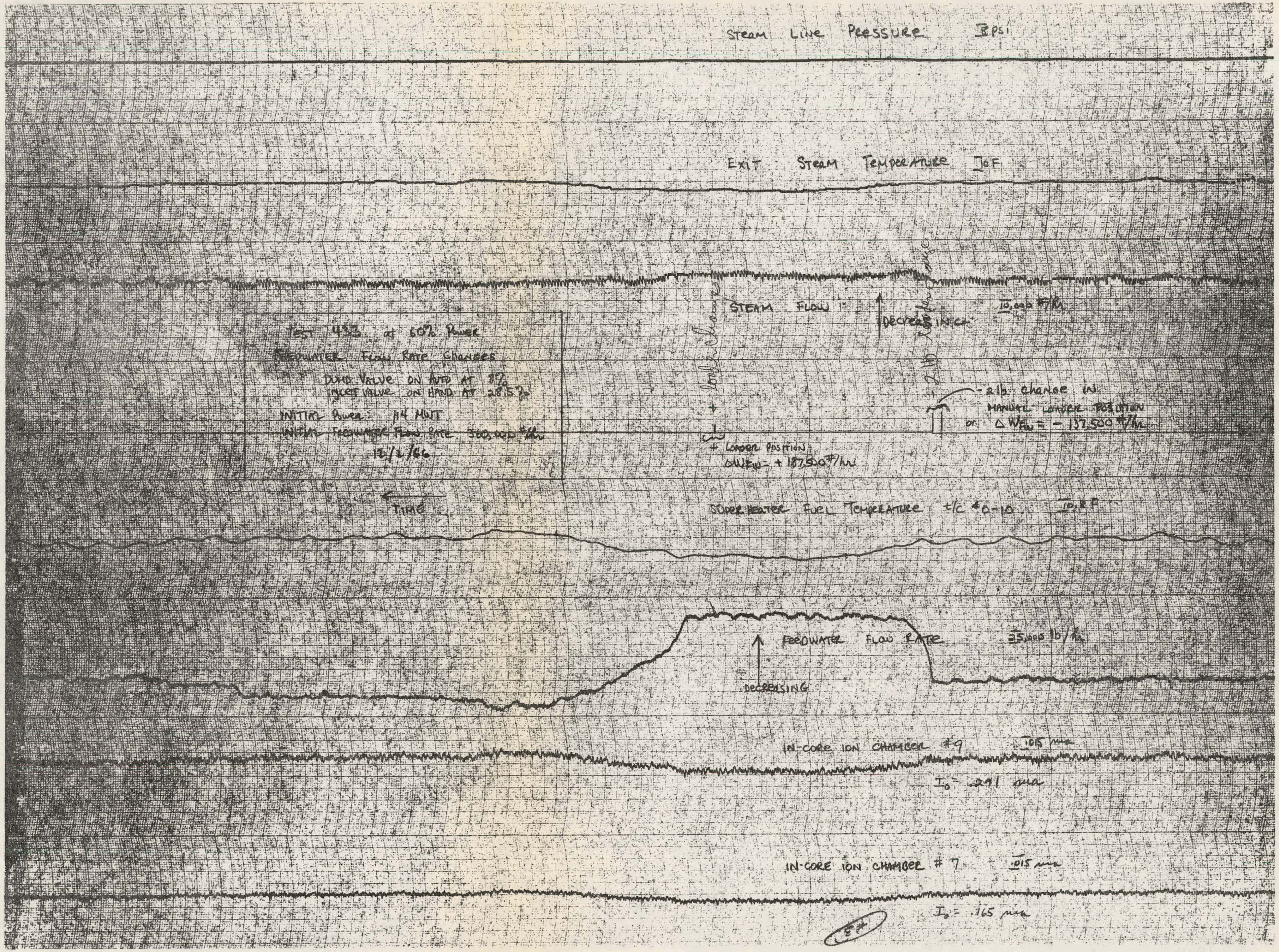
FEEDWATER FLOW RATE CHANGE: 39 Mwt

FIG. 5.1



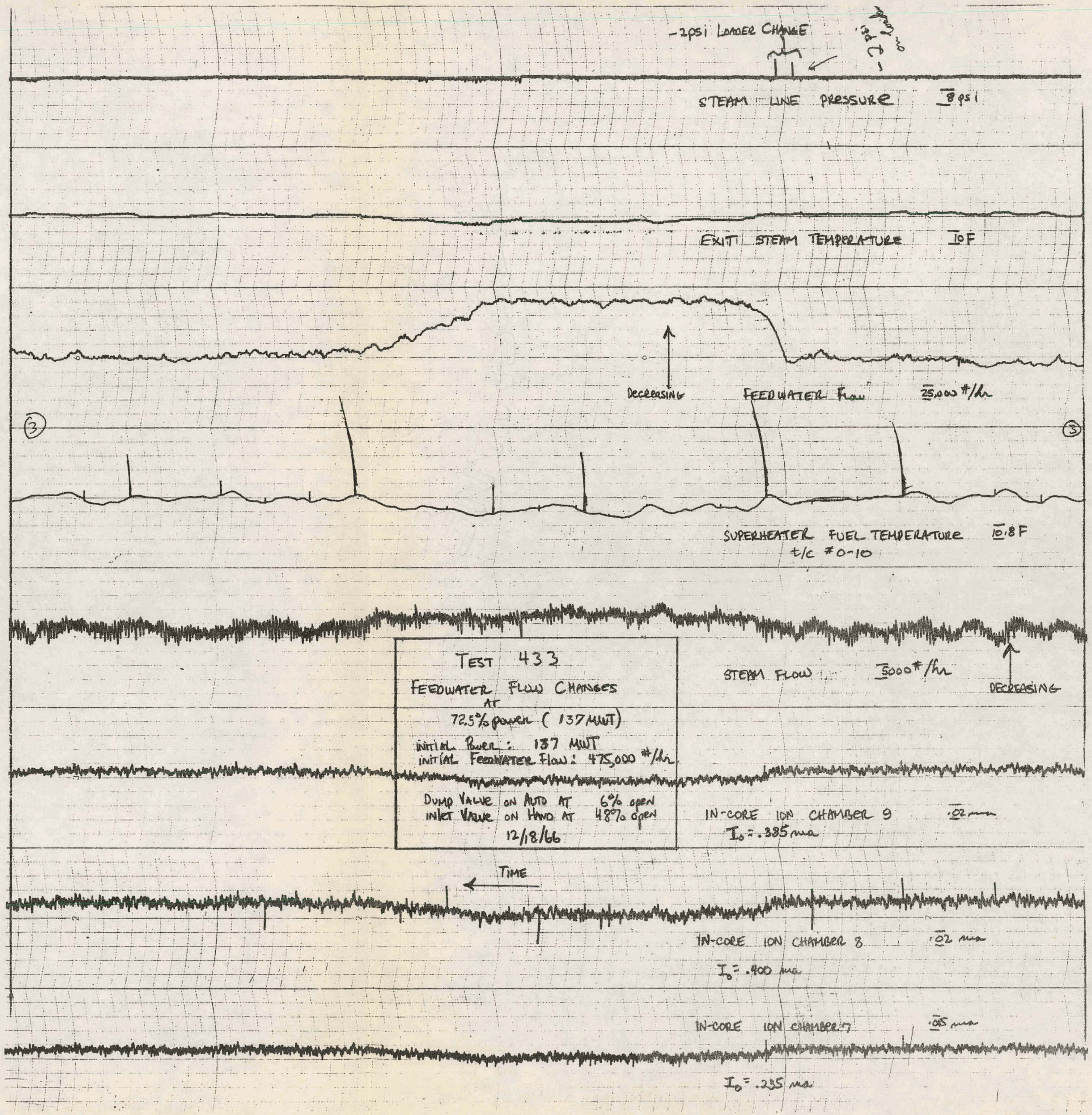
FEEDWATER FLOW RATE CHANGE: 76 Mwt

FIG. 5.2



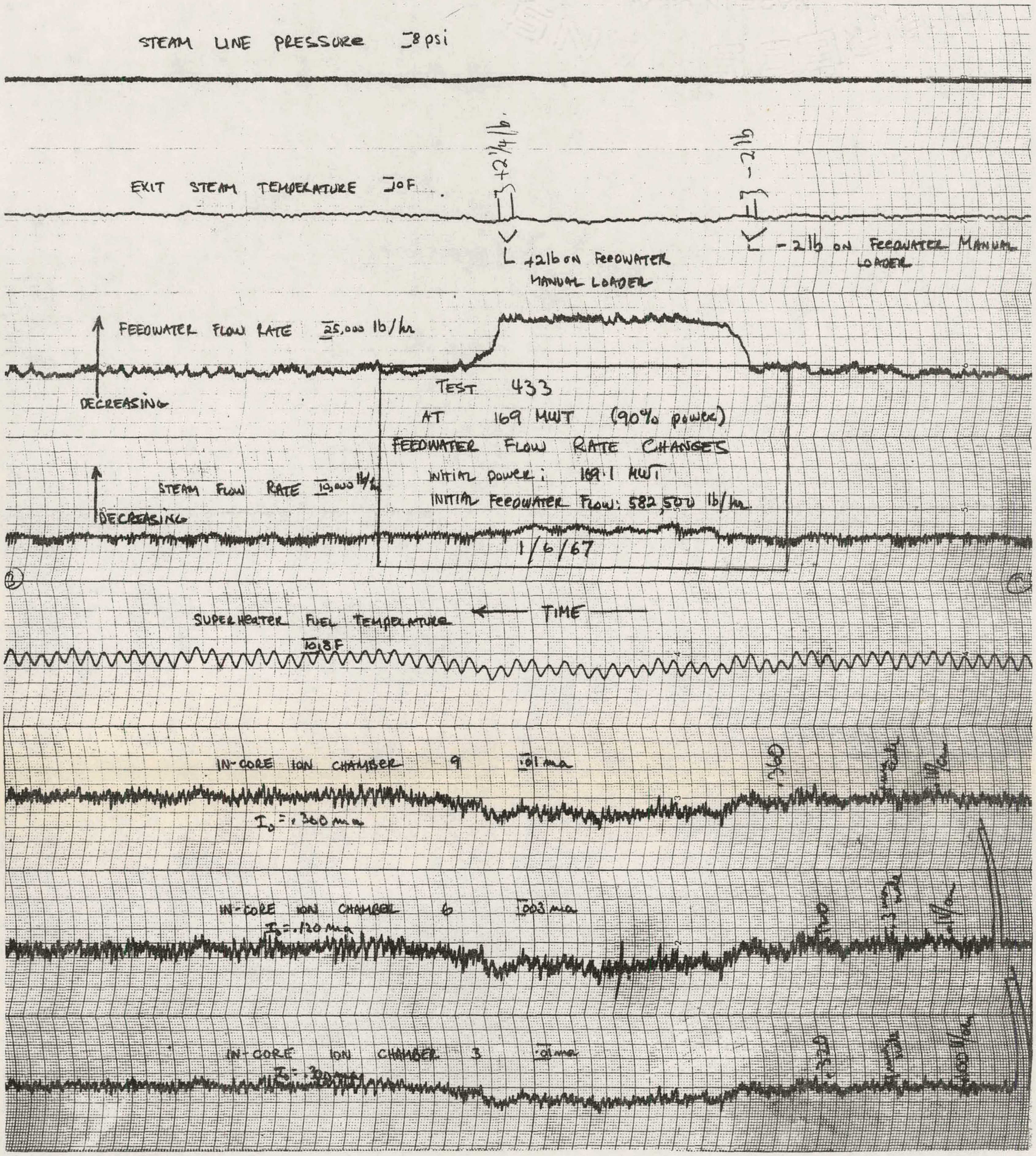
FEEDWATER FLOW RATE CHANGE: 114 Mwt

FIG. 5.3



FEEDWATER FLOW RATE CHANGE: 137 Mwt

FIG. 5.4



FEEDWATER FLOW RATE CHANGE: 169 Mwt

FIG. 5.5

6. RESPONSE TO FEEDWATER TEMPERATURE CHANGES

Another method of varying core inlet temperature and core subcooling is to vary the feedwater temperature. This step was performed at various power levels to determine the plant dynamic response to cold water insertion and to determine the effects of subcooling changes on reactor power and superheater fuel temperatures. Colder feedwater temperature causes two effects, both of which tend to increase the transient superheater fuel temperatures:

(a) At a given power level, the boiler portion of the core must deliver more heat in order to saturate the subcooled inlet water, leaving less heat available to boil the saturated water. This means that less coolant is available to the superheater;

(b) By quenching some of the core voids, and thus effectively raising the boiling boundary, reactor power is increased, causing superheater fuel temperatures to rise during the transient.

Table 6-1 lists the initial conditions for those feedwater temperature disturbances most descriptive of reactor system response during these transients. Table 6-2 summarizes the reactor transient response to these disturbances, and Figs. 6.1 through 6.4 are the actual response records.

The reactor system responses were generally as expected and as predicted by pre-test calculations, with one notable exception. The rise and fall in reactor power that occurs immediately after the feedwater system set point disturbance is caused by the closing and opening of the steam extraction valve that supplies main steam to the No. 14 feedwater heater. This detail was not simulated on the analog computer model; hence, the character of the calculated transient is different from the observed transient. The remainder of the transient, and the final steady-state values of power and steam temperature, are very close to predicted values. See Table 6-3 for a summary of calculated and predicted responses.

TABLE 6-1

INITIAL PLANT CONDITIONS FOR FEEDWATER TEMPERATURE CHANGES

Reactor Power	76 Mwt (10/19/66)	114 Mwt (12/3/66)	138 Mwt (12/18/66)	169 Mwt (1/6/67)
Channel 5 Current	0.88×10^{-6} amp	0.185×10^{-5} amp	0.30×10^{-5} amp	0.34×10^{-5} amp
Steam Flow	270,000 lb/hr	375,000 lb/hr	530,000 lb/hr	583,000 lb/hr
Feedwater Temp.	362 F	-	385 F	387 F
Total Recirculation Flow	59,500 gpm	57,900 gpm	55,500 gpm	54,700 gpm
Superheater Fuel Temp. t/c #O-10	775 F	752 F	765 F	765 F
Exit Steam Temperatures	635 F	644 F	642 F	658 F
Reactor Pressure	541 psig	551 psig	580 psig	584 psig
Dump Valve Position	23%	8%	6%	6%
Inlet Valve Position	0%	28.5%	48%	53%
Pressure Control	Auto on dump valve	Auto on dump valve	Auto on dump valve	Auto on dump valve
Level Control	Manual	Auto	Auto	Auto
Feedwater Temperature Control	Manual	Manual	Manual	Manual

TABLE 6-2

SUMMARY OF REACTOR TRANSIENT RESPONSE TO FEEDWATER TEMPERATURE CHANGES

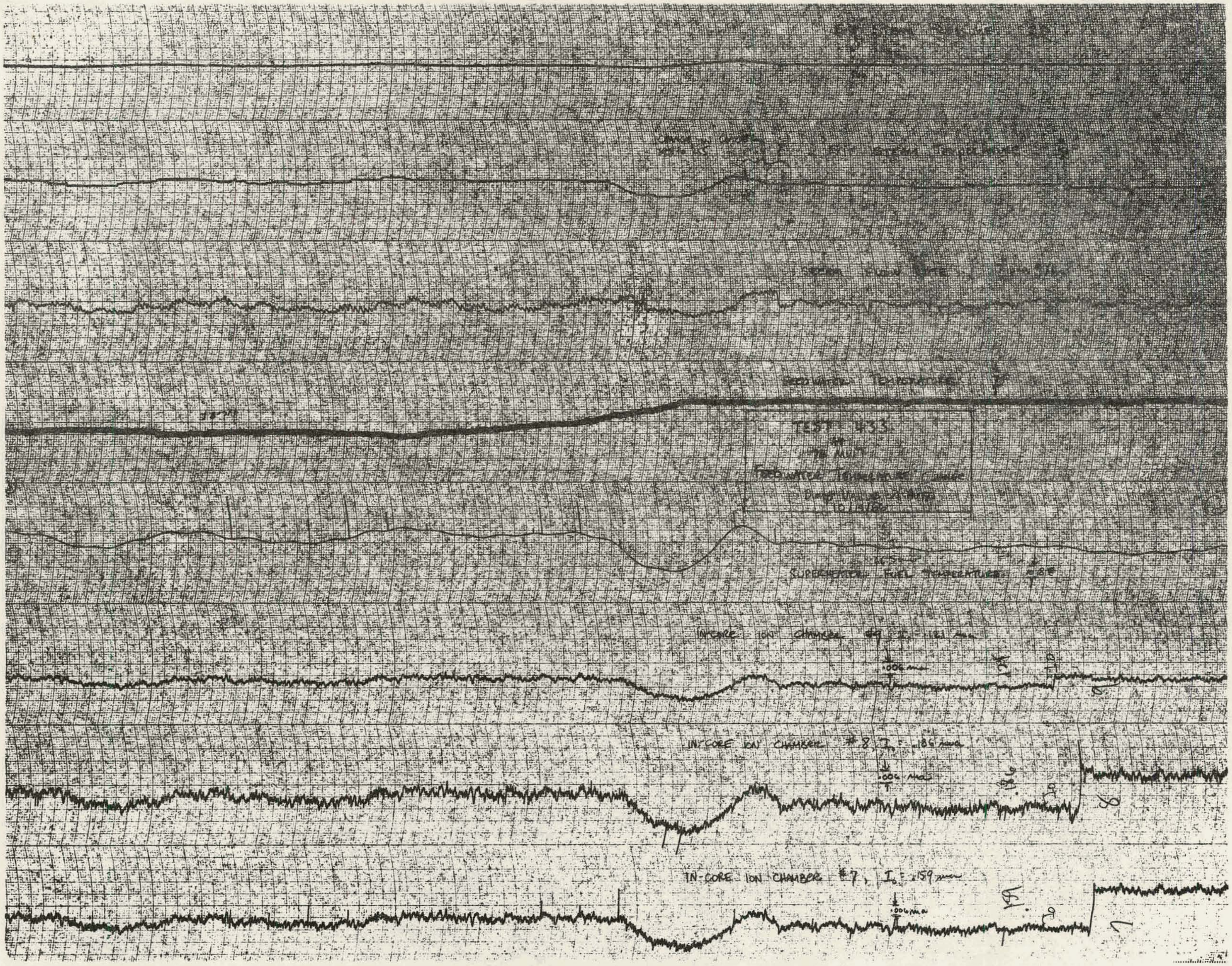
Reference Figure Initial Reactor Power	Fig. 6.1 76 Mwt (10/19/66)	Fig. 6.2 114 Mwt (12/3/66)	Fig. 6.3 138 Mwt (12/18/66)	Fig. 6.4 169 Mwt (1/6/67)
Disturbance* of T_{FW}	-16.8 F in 16 sec	-18 F in 7.5 sec	-15.4 F in 8 sec	-15 F in 5 sec
Max ΔPower, % of existing power	+4.2% on in-core #7	+4.5% on in-core #7	+3.9% on in-core #7	+3.9% on in-core #7
Max ΔT_F on t/c #O-10	+16 F	+13 F	+10 F	+11 F
Max ΔT_2	+6 F	+7 F	+7 F	+6 F
Final ΔT_2	+5 F	+5 F	+5 F	+5 F

*Time referred to in "Disturbance" above refers to time taken to change manual loader dial on feedwater temperature controller.

TABLE 6-3

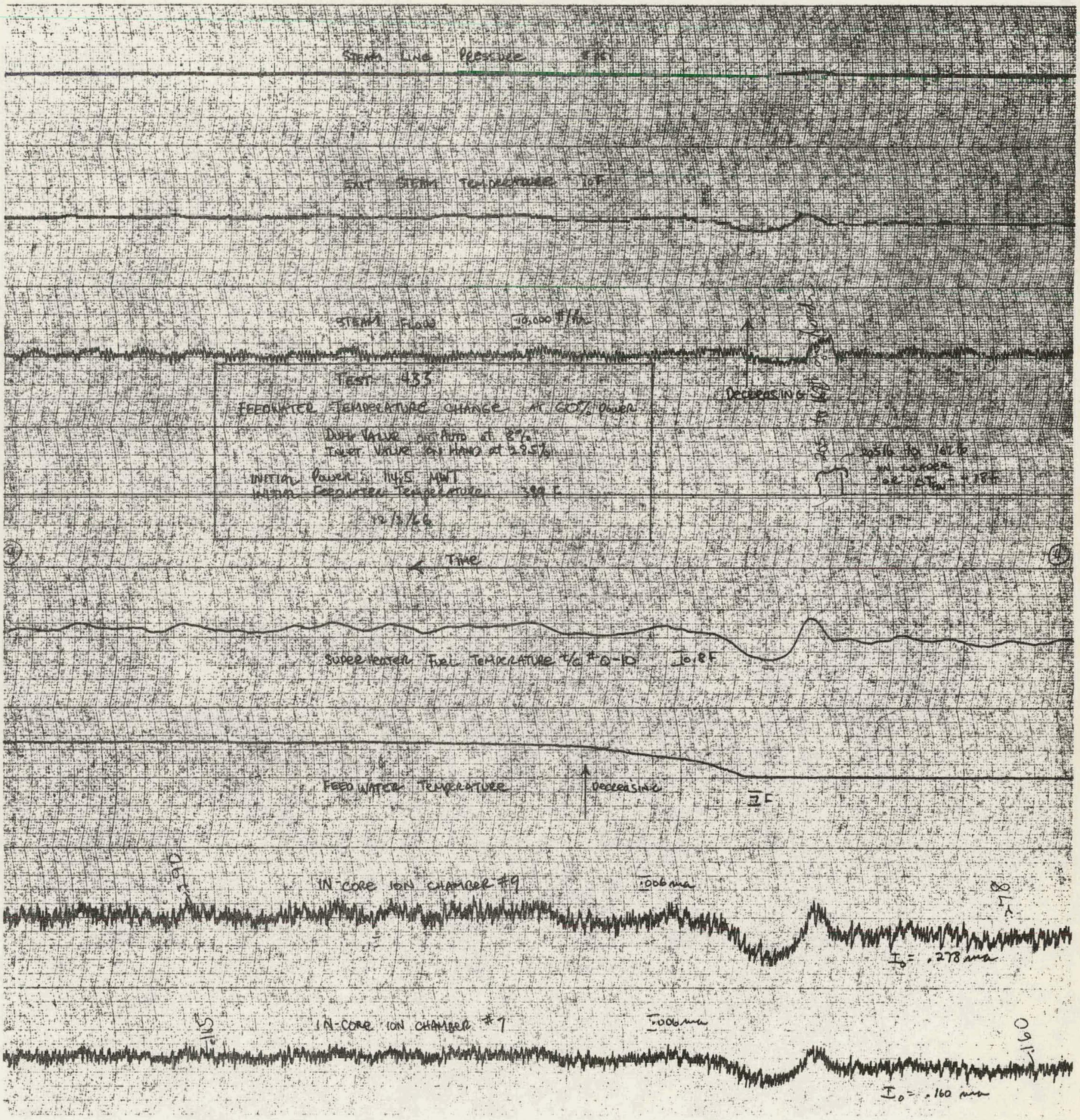
COMPARISON OF PARAMETERS: CALCULATED PRE-TEST PREDICTIONS VS. EXPERIMENTAL RESULTS

	Disturbance	Initial Power Level	Max Δ Power % of Initial Power	Final Δ Power % of Initial Power	Max ΔT_F Pre-Test (hot spot) Experiment (t/c #O-10)	Max ΔT_2	Final ΔT_2
Pre-Test Calculations	ΔT_{FW} - 15 F step	76 Mwt	+4%	-	+18 F	+6.5 F	-
Experimental Results	ΔT_{FW} = -16.8 F in 16 sec	76 Mwt	+3.8% on in-core #7	+3% on in-core #7	+16.2 F	+6 F	+3 F
Pre-Test Calculations	ΔT_{FW} = -15 F in 22 sec	114 Mwt	+3.3%	+3.3%	+17 F	+6 F	+6 F
Experimental Results	ΔT_{FW} = -18 F in 7.5 sec	114 Mwt	+4.5% on in-core #7	+3.7% on in-core #7	+13 F	+7 F	+5 F



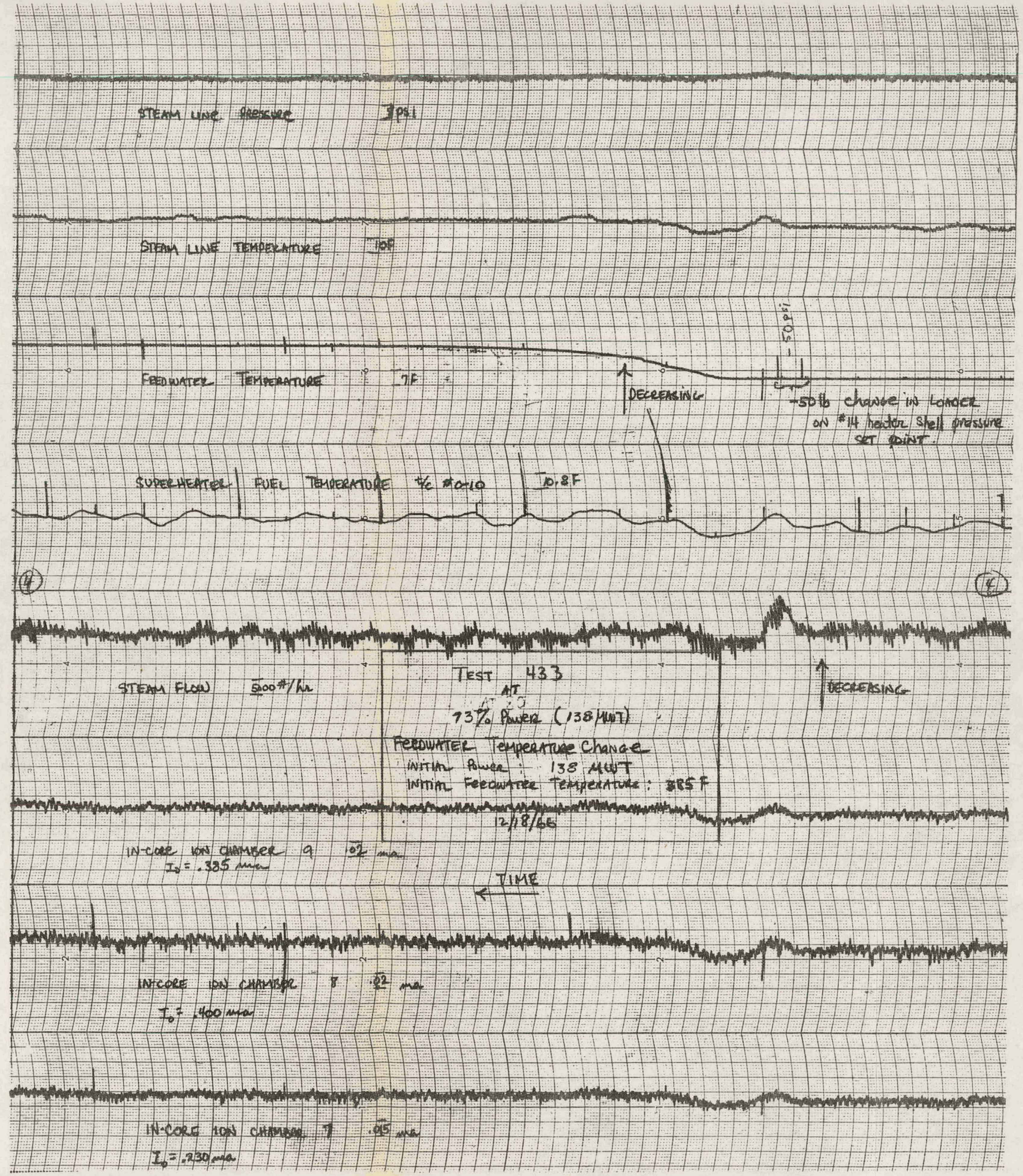
FEEDWATER TEMPERATURE CHANGE: 76 Mwt

FIG. 6.1



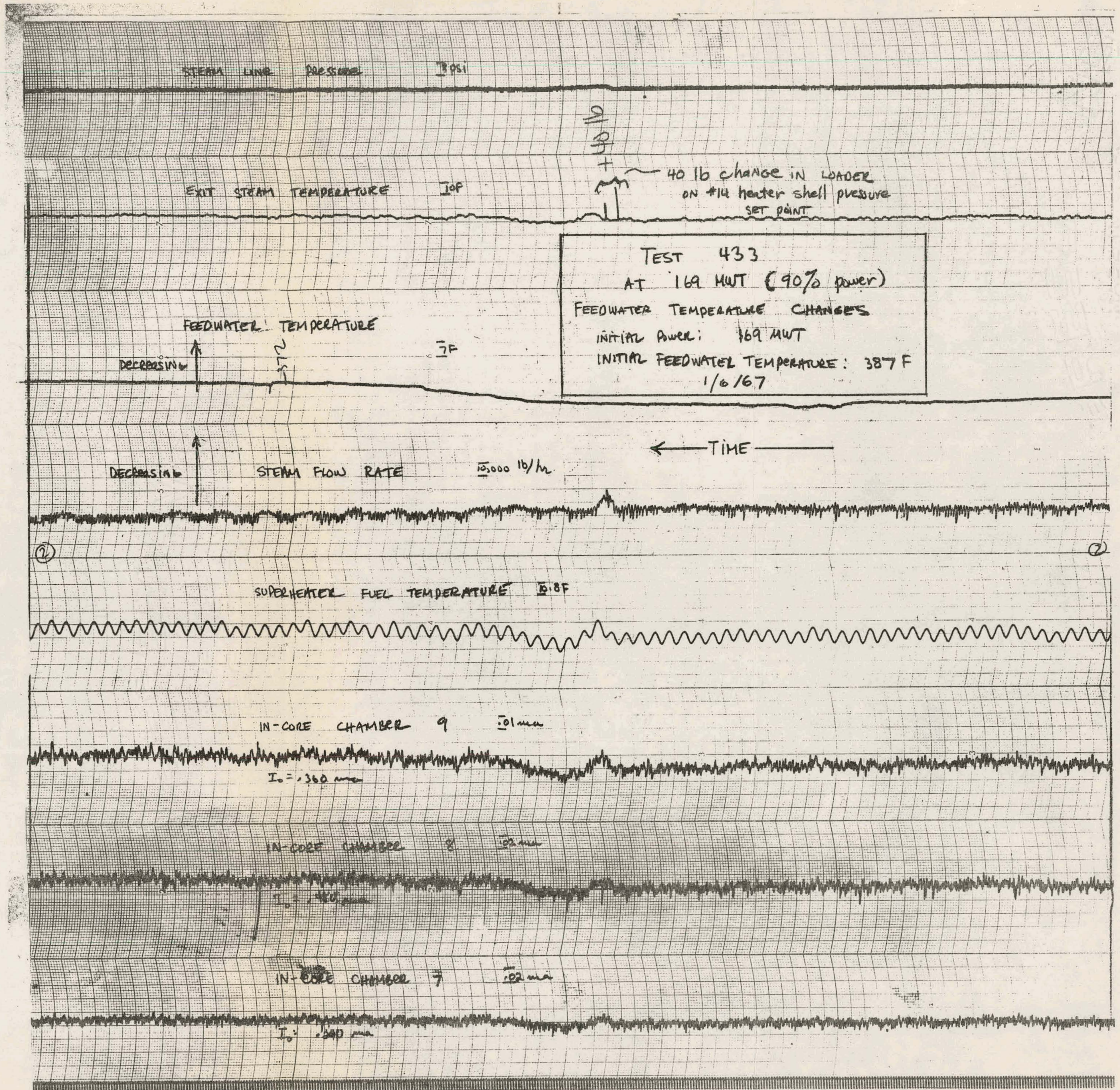
FEEDWATER TEMPERATURE CHANGE: 114.5 Mwt

FIG. 6.2



FEEDWATER TEMPERATURE CHANGE: 138 Mwt

FIG. 6.3



FEEDWATER TEMPERATURE CHANGE: 153.8 Mwt

FIG. 6.4

7. RESULTS OF CONTROL ROD MOVEMENTS

The last of the disturbances studied in Test 433 was the reactor system response to control rod movement. The rod movement was limited to no more than 2 in. for individual rods and 5 in. for a group of rods in order to prevent power peaking and resultant superheater temperature hot spots. Because of this restriction, the reactor system responses to these disturbances were minimal; in-core ion chamber responses were the responses that yielded the most information.

Table 7-1 lists the initial conditions for those control rod disturbances most descriptive of reactor system response. Figures 7.1 through 7.4 show the actual responses to these movements.

Table 7-2 summarizes the transient responses of in-core Chambers 7, 8, and 9 to 2-in. withdrawal of a single control rod. The top entries in each position in Table 7-2 are the maximum transient local powers achieved during rod motion in terms of percent of the initial power. The lower values in the table are the final power levels arrived at following the rod motion, also expressed in terms of percent of initial power.

Examination of the results in Table 7-2 shows both radial and axial flux tilts, as expected when one rod is moved slightly. Withdrawal movement of rods that are positioned low in the core causes power density to increase in the lower part of the core and to decrease in the upper core. This is due to the fact that the additional voids formed low in the core (where the local power rises) move up the coolant channel and cause a decrease in moderator density in the upper core, thus reducing local power density in the upper core. In a sense, this is an uncoupling of two sections of the core. However, when rods are moved more, or when they are moved from a higher initial position in the core, the flux tilting is not as noticeable; and the various sections of the core appear to be better coupled.

Because the calculational model used for Pathfinder dynamic studies was a space-independent model, no direct calculational comparisons with the experimental results are available.

TABLE 7-1

INITIAL PLANT CONDITIONS FOR CONTROL ROD MOVEMENTS

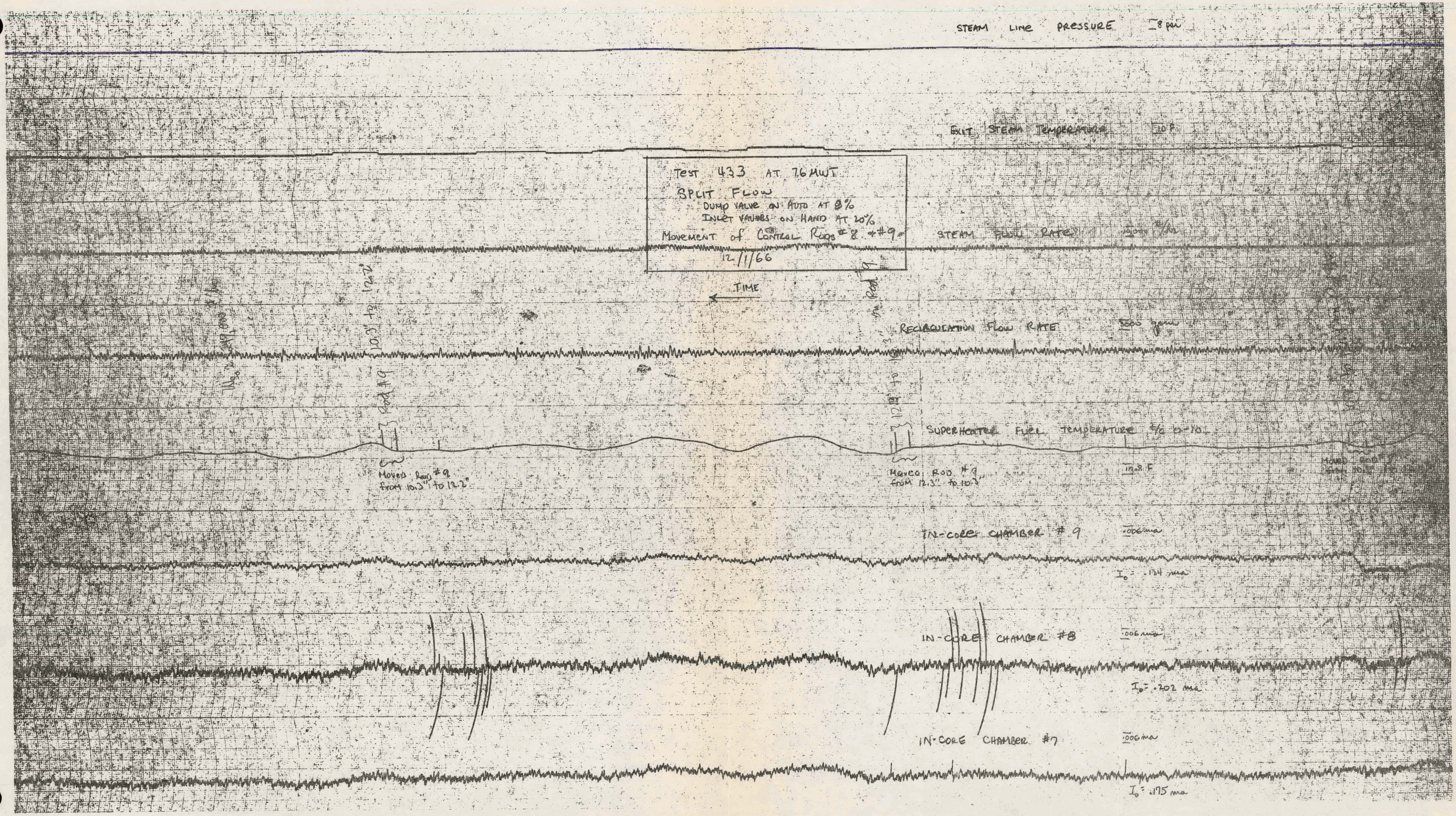
Reference Figure Reactor Power	Fig. 7.1 71 Mwt (12/1/66)	Fig. 7.2 138 Mwt (12/18/66)	Fig. 7.3 169 Mwt (1/6/67)	Fig. 7.4 50 Mwt (1/24/67)
Channel 5 Current	1.02×10^{-6} amp	0.3×10^{-5} amp	0.34×10^{-5} amp	-
Steam Flow	252,500 lb/hr	530,000 lb/hr	582,500 lb/hr	180,000 lb/hr
Feedwater Temperature	378 F	374 F	370 F	-
Total Recirculation Flow	59,000 gpm	55,500 gpm	54,700 gpm	-
Superheater Fuel Temp. t/c #O-10	751 F	765 F	765 F	-
Exit Steam Temperature	633 F	642 F	658 F	-
Reactor Pressure	-	580 psig	584 psig	-
Dump Valve Position	9%	6%	6%	-
Inlet Valve Position	20%	48%	53%	-
Pressure Control	Auto on dump valve	Auto on dump valve	Auto on dump valve	Auto on inlet valve
Level Control	Auto	Auto	Auto	Auto
Feedwater Temperature Control	Auto	Auto	Auto	Auto

TABLE 7-2

SUMMARY OF IN-CORE ION CHAMBER RESPONSES TO MOVEMENT OF SINGLE CONTROL RODS

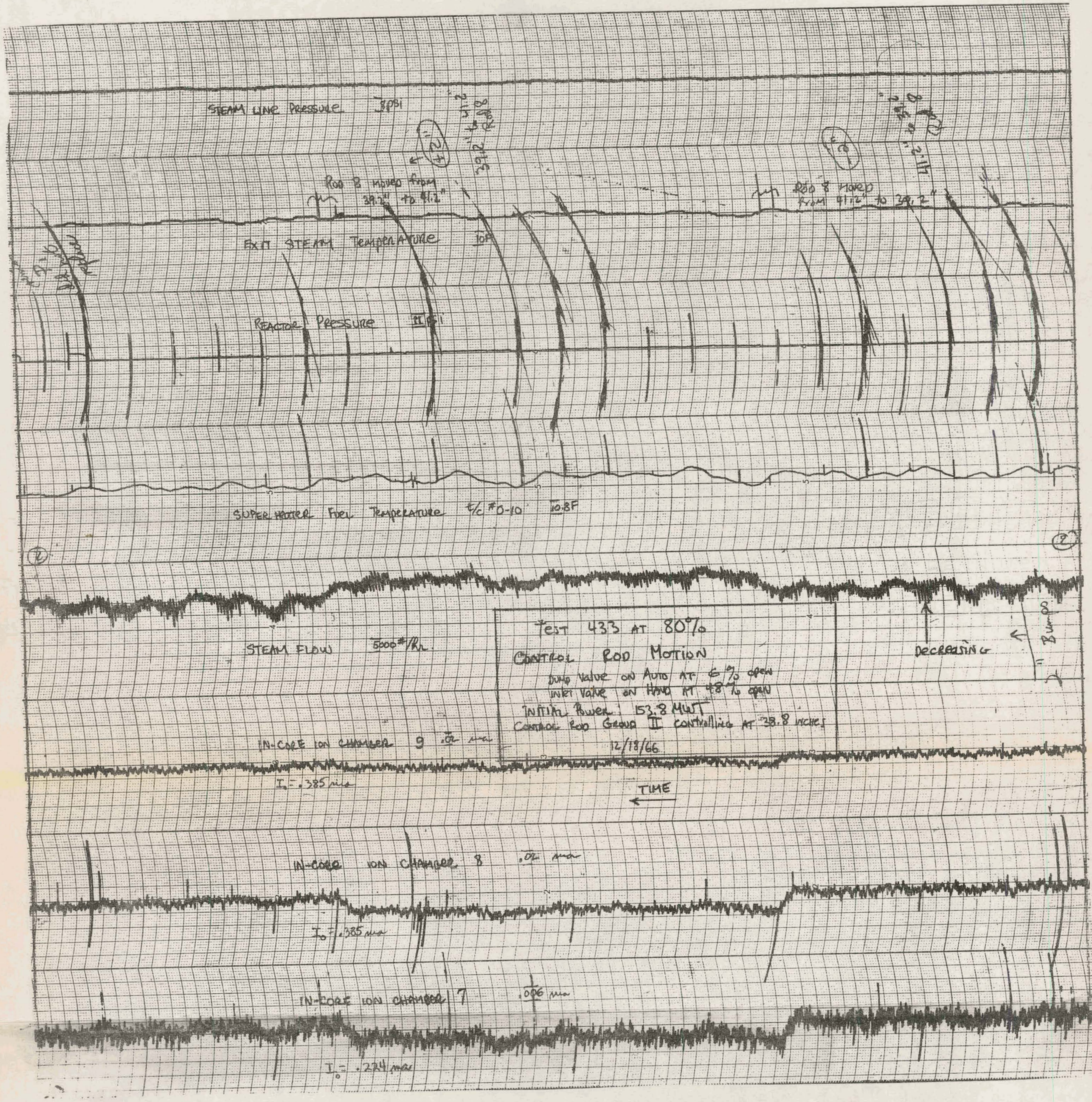
Initial Reactor Power	Control Rod Movement	Max Δ Power In-Core 7	Max Δ Power In-Core 8	Max Δ Power In-Core 9
71 Mwt (Fig. 7.1)	Rod 8 moved from 10.2 in. to 12.4 in.	+0% Final: -1.3%	+2.0% Final: +0%	+8% Final: +6.2%
71 Mwt (Fig. 7.1)	Rod 9 moved from 10.3 in. to 12.2 in.	+1.0% Final: -0.8%	+1.8% Final: -0.6%	+2.2% Final: 0%
109.1 Mwt	Rod 8 moved from 27.2 in. to 29.2 in.	+2.7% Final: +0.5%	+7.2% Final: +4%	+8.1% Final: +5%
138 Mwt (Fig. 7.2)	Rod 8 moved from 39.2 in. to 41.2 in.	+3.2% Final: +2.3%	+6.2% Final: +5.2%	+1.5% Final: +1.0%

NOTE: In-core Chambers 7, 8, 9 are next to rod 8 and are axially positioned 54 in., 36 in., and 18 in. from the bottom of the core respectively. Rod 9 is symmetrically opposite rod 8, across the superheater. See Fig. 1.1.



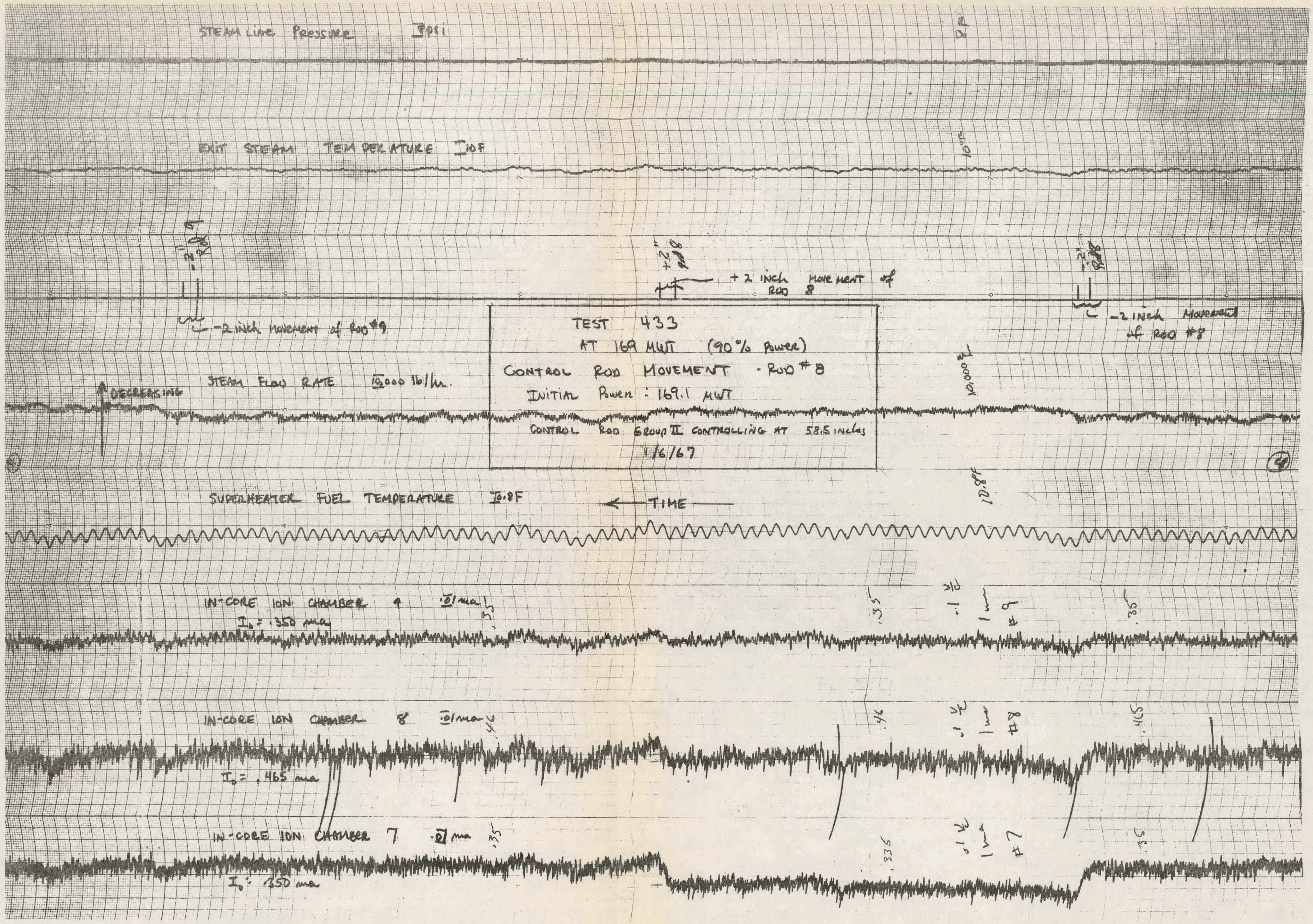
CONTROL ROD MOVEMENT: 76 Mwt

FIG. 7.1



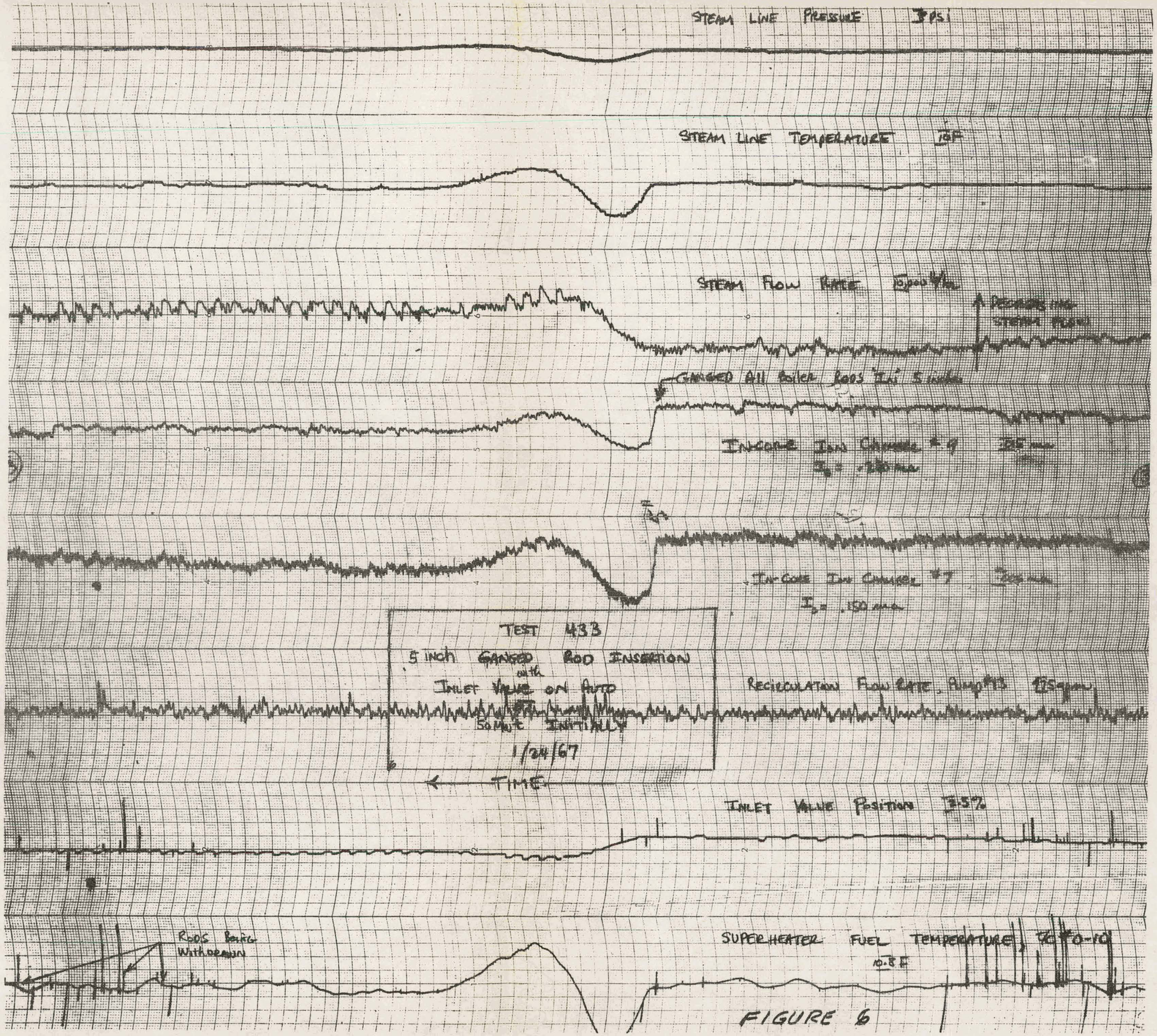
CONTROL ROD MOVEMENT: 138 Mwt

FIG. 7.2



CONTROL ROD MOVEMENT: 169 Mwt

FIG. 7.3



CONTROL ROD MOVEMENT, INLET VALVES
ON AUTOMATIC: 50 Mwt

FIG. 7.4

8. REFERENCES

(1) Amendments 12 and 13 to Pathfinder Licensing Application, Docket 50-130; Answers to AEC Letter of 11/26/62.

(2) ACNP-62001 - "Pathfinder Analog Simulator," January 1962.

(3) ACNP-63033 - "Analog Simulator Results," March 1964.

(4) ACNP-5905 - "Pathfinder Atomic Power Plant Safeguards Report."

APPENDIX

TEST PROCEDURE 433

Fluid Dynamics Reactivity Effects

PATHFINDER ATOMIC POWER PLANT

REVISION 8

Dated November 25, 1966

to

TEST PROCEDURE 433

TITLE: Fluid Dynamics Reactivity Effects

Approved for Allis-Chalmers /S/ D. A. Lampe
D. A. Lampe
Assistant Project Manager
Pathfinder Project

Date: 11-25-66

TEST PROCEDURE APPROVAL SHEET

TEST 433

ALLIS-CHALMERS

Prepared by	/S/ J. T. Stone Responsible Engineer	Date	6/7/66
Approved by	/S/ J. T. Stone Section Head	Date	6/7/66
Reviewed by	/S/ J. P. Kelly Site Operations Engineer	Date	6/7/66
Approved by	/S/ R. W. Thiel Site Operations Manager	Date	6/7/66
Approved	/S/ D. A. Lampe Assistant Project Manager	Date	6/7/66

NORTHERN STATES POWER

Approved by	Operations Committee	Date	May 4, 1966
Reviewed by	Safety Committee	Date	May 26, 1966

Approved
June 3, 1966

CONTENTS

433. FLUID DYNAMICS REACTIVITY EFFECTS

433.1 Purpose

The purpose of Test 433 is:

(1) *to evaluate the effects on the reactor of certain fluid dynamic disturbances* which are likely to occur during the operation of the plant;

(2) to verify that the transient response to the disturbances listed herein is not severe and is well within the limits of the reactor protection systems;

(3) to demonstrate that all control systems (pressure, feedwater flow, feedwater temperature, etc.) are adjusted to respond properly to the various disturbances imposed in this test;

(4) to obtain reactor stability information at various power levels and

a) predict the response at a higher power level, and

b) correlate this information with the results of Test 432; and

(5) to determine the accuracy of certain system responses as indicated by the Pathfinder analog simulator model.

The purposes of changing recirculation flow in Test 433 are as follows:

(1) To observe and verify that flow variation is a legitimate and safe means of changing reactor power at a variety of initial power levels.

(2) To construct an operational control map that will relate control rod motion, recirculation flow changes and reactor power level.

(3) To verify that it is possible to go from three to one pump operation--and then back to three pump operation while the reactor is at power.

(4) To determine whether it will be feasible to raise power from 70% to 100% without control rod motion so as to circumvent the problems of power channel de-calibration caused by control rod motion.

433.1a Ground Rules

(1) All reactor or system disturbances are such as to remain below, or within, safety system limits. That is, none of the tests planned in this series will cause reactor scram or runback. During all tests, the safety system will be operable with at least the nominal set points specified in the Tech Specs.

(2) In the safety analyses presented in this writeup, nominal values were assumed for the system parameters such as fuel time constants and reactivity coefficients. However, the calculations were done for the hottest superheater element so that the equilibrium hot spot temperature is seen to be 1270 F for rated reactor conditions.

(3) The transients are to be first initiated in a direction so as to cause superheater fuel temperatures to decrease. All transients will be done first slowly, and then will be repeated in a shorter time interval so as to obtain the necessary dynamics information.

(4) When superheater hot spot temperature increases, the limits placed on the transient are taken to be those stated in the "Answers to DRL Questions," Answer 1-4, pp 1-4.6 and 1-4.19. "The . . . temperature increases are a maximum of 150 F above steady-state for the operating and 350 F for the accident transients." (1300 F is taken to be the steady-state operating temperature. All transients performed in Test 433 are "operating transients.")

(5) During the test, a selected person shall be designated by the reactor supervisory engineer to watch parameter changes and to determine whether "maximum permissible values" are being approached and whether appropriate safety action is necessary.

(6) If maximum permissible parameter changes are reached, the test will be halted until the Operations Committee has reviewed the situation.

433.2 <u>General Test Method</u>

The reactor will be stabilized at various power levels and initial values of pertinent variables recorded. The reactor will then be subjected to disturbances of the following system variables and the resulting transient responses will be continuously recorded:

(1) feedwater temperature,
(2) feedwater flow,
(3) pressure set point,
(4) recirculation flow, and
(5) control rod motion.

The reactor will then be returned to its initial condition by changing the same system variable in the opposite direction. Again, the resulting transients will be recorded. The information derived from Test 433 will be contained in these recorded responses, the total of which will provide an adequate evaluation of the entire plant response to various fluid dynamic disturbances.

433.2.1 Personnel and Administrative Requirements

The organization of the personnel required to perform these experiments and their respective duties is described in the Pathfinder Program and Organization for Preoperational and Nuclear Testing (ACNP 6112-Rev. 2). Differences of opinion among these persons which cannot be otherwise resolved, will be referred to higher management. Any opinion which requires reactor shutdown will override all others until the issue is settled.

Personnel will not be permitted in the reactor building during positive periods when the reactor is in the non-boiling condition.

The NSP Operations Supervisor has overall operations responsibility. All Pathfinder Technical Specifications requirements shall be maintained. Administrative control of operations shall be specified to limit operations in a conservative manner to minimize incidence of inadvertent automatic safety actions.

Changes in these procedures will be permitted as long as they are not contrary to the scope or intent of the General Test Method or Technical Specifications and do not introduce an unreviewed operating or safety question. Such changes must be approved by the NSP Operations Supervisor and the A-C Operations Engineer. Details of change and reasons therefore shall be recorded in both the reactor log and on master copy of this procedure, with signature of approving parties. A change will be referred to Operations Committee for approval, upon request of any member. In the event that a condition is discovered which is significantly beyond the expected limits of the test, the matter will be referred to the Safety Committee.

Radiation protection standards and radiation procedures as described in E.1 and E.2 of the Operations Manual shall be adhered to.

The NSP Cognizant Engineer shall be responsible for insuring that all the data required by the procedure is recorded on appropriate logs and for the preparation of these logs.

Operations shall be conducted in accordance with the applicable sections of the Operations Manual, unless specifically stated in the test procedure. Any changes in routine, integrated plant operation shall be given widest possible dissemination to all operating personnel.

Routine water chemistry samples shall be taken in accordance with the NSP operating requirements as established by the Radiation & Chemical Engineer. Corrective actions shall be taken to ensure that the limits as specified in the Technical Specifications are not reached.

433.2.2 Summary of Test Procedure

The initial and final values of certain plant variables will be recorded on data sheets for each test and the transient responses of these variables will be continuously recorded on strip chart recorders. These variables are:

(1) power level from channels 4, 5, and 7;
(2) recirculation water temperature;
(3) main steam temperature (from thermocouples);
(4) main steam temperature (from resistance thermometer);
(5) superheater exit steam pressure;
(6) reactor water level;
(7) feedwater flow;
(8) feedwater temperature;
(9) main steam flow;
(10) superheater fuel temperature;
(11) reactor dome pressure;
(12) recirculation flow; and
(13) all control rod positions (not recorded continuously).

The tests which will be conducted are as follows:

433.2.2.1 Feedwater Temperature

The feedwater temperature set point will be increased in three 15 F steps, decreased in one 45 F step, and then further decreased in one 15 F step to 325 F; it will then be returned to the initial set point. All reactor systems will be allowed to stabilize between set point changes. The initial (reference) temperature of 340 F may be changed due to reactivity and superheater temperature considerations. The mode of control of feedwater temperature will be varied from MANUAL to AUTO at 20% power to determine:

a) the effect of transfer of the operational mode on the reactor system, and

b) the ability of a control mode to cope with plant disturbance.

433.2.2.2 Feedwater Flow

Reactor level control will be transferred to HAND operation and feedwater flow decreased by percentages up to 40% of the existing flow at 20% and 40% of full power and by 20% of full flow at 60%, 80%, and 100% power levels. The decrease will be accomplished utilizing the feedwater bypass valve at 20%, 40%, and 60% power and the main feedwater valve at 80% and 100% power. After reactor systems have stabilized, feedwater flow will be increased to the initial flow and reactor level control returned to AUTO operation. The initial reactor water level will be restored by slowly increasing the level set point. The feedwater flow reductions will be changed somewhat as dictated during the course of the experiment.

433.2.2.3 Recirculation Flow

A rather extensive test will be conducted to determine the effect of recirculation flow on reactivity. The test will include:

(a) decreasing and increasing recirculation flow by gang operation of the discharge valves;

(b) recirculation pump tripout - one and two pump tripout;

(c) recirculation pump startup - one pump to two pump operation and two pump to three pump operation.

433.2.2.4 Superheater Exit Steam Pressure

The P_2 set point will be decreased 5 psi, reactor systems allowed to stabilize, and then increased 5 psi. At 60%, 80% and 100% power, an additional change of ± 10 psi will be made. The magnitudes of the pressure steps may be altered as dictated by the initial responses to 5 and 10 psi steps.

433.2.2.5 Control Rod Positioning

Control rod withdrawal and insertion will be performed at 20, 40, 60, 80 and 100% power and flow and also at reduced recirculation flow (at rated power-to-flow ratio) to obtain:

(a) transient response to control rod motion; and

(b) steady-state correlations between control rod reactivity, recirculation flow, and reactor power.

433.2.2.6 Testing Sequence

The following chart indicates the power levels at which tests will be conducted and the variables which will be changed at each power level. The variables include feedwater temperature (T_{FW}) and flow (W_{FW}), superheater exit steam pressure (P_2), recirculation flow (W_{RW}), and control rod position (CRP).

POWER LEVEL (Mw)

	Without Osc.			Without Osc.			
Mwt	38 Mw	76 Mw	95 Mw	114 Mw	133 Mw	152 Mw	190 Mw
T_{FW}	X	X		X	X	X	X
W_{FW}	X	X		X	X	X	X
P_2	X	X		X	X	X	X
W_{RW}	*X	X		X	X	X	X
CRP	X	X		X	X	X	X

433.3 Applicable Theory

433.3.1 Mechanism of Reactivity Insertion

The principal mechanism of reactivity insertion for all the tests covered by Test 433 (except control rod maneuvers) is the reactor's void coefficient of reactivity. The mechanism is such that increasing the void fraction results in a negative reactivity insertion.

The core average quality ($\overline{X}$) and reactivity in voids are functions of core exit quality (X_e) and inlet subcooling (Δh), such that:

$$\overline{X} = C_1 X_e - C_2 \Delta h \tag{1}$$

$$k_v = - C_3 X_e + C_4 \Delta h \tag{2}$$

where $C_1 = \int_0^1 \varphi_1' (z)\, Q(z)\, dz = 0.537$

$\varphi_1' (z)$ = local axial void reactivity importance function

$Q(z)$ = axial power distribution

and $\int_0^1 \varphi_1' (z)\, dz = 1.0$

C_2 = is graphically evaluated from axial void distributions to be $0.375/h_{fg}$

$C_3 = \frac{\partial k_v}{\partial X_e}$ = \$34.4/unit X_e at rated conditions

$C_4 = \partial kv/\partial(\Delta h)$ = - \$0.02/Btu/lb.

Furthermore, the exit quality can be written simply in terms of the boiler heat transfer rate ($\dot{Q}_h$), coolant flow through active core (W_W), and the subcooling (Δh) as follows:

$$X_E \cong \frac{Q_h/W_W - \Delta h}{h_{fg}} \tag{3}$$

Therefore, an increase in exit quality is a negative reactivity effect and causes power to decrease. An increase in subcooling is a positive reactivity effect and causes power to increase.

The subcooling is determined primarily by feedwater enthalpy and flow and recirculation flow such that:

$$\Delta h \simeq \frac{W_{FW} (h_f - h_{FW})}{W_{RW}} \quad (4)$$

Therefore, the test procedure (433.6 - Step 3) in which feedwater temperature is increased and the test procedure (433.6 - Step 6) in which feedwater flow is decreased, will both cause a decrease in subcooling resulting in a negative reactivity effect. However, both of these effects will be noticed by the reactor in a relatively long time interval (time lag greater than 6 sec), especially the temperature effect, due to feedwater passage through the feedwater line plus the recirculation loops before entering the core.

An increase in recirculation flow decreases the average void fraction in the core and hence is a positive reactivity effect. This effect is noticed by the reactor in a relatively short time since an almost immediate reduction in average core void content will result from an increase in recirculation flow. Similarly, recirculation flow reductions tend to decrease reactor power quickly.

It is seen that while exit quality is itself a function of subcooling, the negative reactivity in voids:

(a) increases with increasing exit quality, causing reactor power to decrease;

and

(b) decreases with increasing subcooling, causing reactor power to increase.

Another variable which will be changed is exit steam pressure. Decreasing exit steam pressure will cause reactor dome pressure (which determines the system's boiling point) to decrease also, resulting in saturated water flashing to steam. The void content of the core will thus be increased resulting in a negative reactivity insertion with decreasing pressure.

Control rod positioning implies both withdrawal and insertion of control rods at various initial power-to-flow ratios. Withdrawal of control rods at power is similar yet distinctly different than increasing recirculation flow. Reactor power responds in a similar manner to flow changes and control rod positioning since an incremental flow change and rod position change are each "worth" a certain power change. However, while recirculation flow changes tend to cause equilibrium reactivity-in-voids (k_V) to stay constant, control rod positioning causes reactivity-in-voids to change an amount almost equal to that in the rods.

433.3.2 Reactor Response to Reactivity Insertion

When the reactor is critical, it will respond to a given amount of reactivity insertion in the following manner:

(1) Initially (for less than 2 sec), reactor power will follow the response predicted by the neutron kinetics model without feedback effects affecting the response (the zero power response).

(2) Reactor power will then begin to deviate from the "zero power" response, due to negative reactivity feedback from voids and fuel temperature.

(3) Reactor power will stabilize at some new power level when the original reactivity insertion is cancelled by the negative reactivity feedback.

The power feedback loops operate primarily through the reactor fuel temperature and the void fraction of the core. A change in fuel temperature causes an insertion of reactivity (Δk) equal to $(\Delta k/\Delta T) \times (\Delta T)$ where $\Delta k/\Delta T$ is the temperature coefficient of reactivity and is negative for Pathfinder. However, while operating at power, the temperature effect is quite small in comparison to the effect of changing the void fraction of the core. Changing the void fraction causes a reactivity insertion due to the change in moderator density (moderator-to-fuel ratio) which affects the leakage and absorption rates of neutrons in the core. Therefore, the reactor's inherent ability to cancel reactivity insertions is essentially proportional to the product of void coefficient of reactivity ($\Delta k/\Delta\bar{\alpha}$), which is negative, and the power-to-flow ratio (n/W_{RW}).

Feedback effects are derived primarily from the boiler zone on Core I, since void and Doppler effects from the high enriched superheater zone are negligible. The void feedback effects in the upper and lower parts of the boiler are similar, although the water-to-fuel ratio is greater in the upper than in the lower region. The Pathfinder infinite multiplication constant has been computed as a function of core exit void fraction; taking into account the proper power distribution for each void fraction. Even for low power levels and corresponding low void fractions, the calculations predict a negative void coefficient for all operating conditions. The values of k_{eff} for both the upper and lower boiler lattices decrease continuously as the upper and lower void fractions increase, respectively.

433.4 Prerequisites

433.4.1 Special Equipment

(1) The two-channel Brush recorder in the pile oscillator instrumentation will be calibrated and connected to record reactor dome pressure (P_1) and superheater exit steam pressure (P_2) after the transfer function test (432). Electrical signals are available from retransmitter #1500 (P_1) and from recorder #249B (P_2).

(2) The Offner eight-channel strip chart recorder (Type R Dynograph) is calibrated and connected to record:

a) power level from channel 5 (#250);

b) reactor dome pressure - P_1 (transmitter #1500 under the control room);

c) feedwater flow - W_{FW} (P/E* on #252A);

*P/E = pneumatic to electrical transducer.

d) feedwater temperature - T_{FW} (P/E on #252B);

e) main steam flow - W_S (P/E on #253A);

f) superheater fuel temperature - T_f (available from recorder in the control room);

g) individual loop recirculation flow; and

h) in-core ion chambers.

(3) The Testing Terminal Box (see A-C Dwg. #43-401-509) is connected to the eight variables listed in (2) and also:

a) recirculation water temperature (#248D);

b) superheater exit steam pressure (#249B);

c) main steam temperature (#249A and resistance thermometer); and

d) two additional superheater fuel thermocouples.

(4) The superheater fuel thermocouples, if available, are connected to the two multipoint recorders and the three highest reading thermocouples are connected to the three strip chart recorders in the control room.

(5) Permanent plant recorders are calibrated and connected to read:

a) power level from channels 4, 5, and 7 (#262, #250, and #254);

b) recirculation water temperature (#248D);

c) superheater exit steam temperature (#249A);

d) superheater exit steam pressure (#249B);

e) water level (#251A);

f) feedwater flow (#252A);

g) feedwater temperature (#252B); and

h) main steam flow (#253A).

NOTE: Variables recorded on Offner recorder will be alternated during testing.

433.4.2 Reactor Conditions

Before each test, the reactor is stabilized at the prescribed power level.

433.4.3 System Conditions

(1) REACTOR BLOWDOWN VALVE Selector Switch 24 on HAND during all testing.

(2) FEEDWATER VALVES TRANSFER Selector Switch RMC-172 on MAIN-AUTO during all testing.

(3) FEEDWATER VALVE Selector Switch RMC-27 on AUTO except during tests which require changing feedwater flow manually.

(4) MAIN STM TO #14 HTR Selector Switch 148 on HAND for the initial 20% tests and on AUTO subsequent to these tests.

(5) REACTOR LEVEL Transfer Valve Selector Switch 112 on NARROW during all testing.

(6) MASTER PRESSURE CONTROL Selector Switch 129 and MAIN STEAM DUMP VALVE Selector Station 30 on AUTO during all testing.

(7) Before each test disturbance, all systems will have attained a steady-state condition.

(8) All valves, components, and systems involved in Test 433 must have been directly tested and found to be in satisfactory condition. In addition, the following control systems must be optimized prior to running this test:

a) pressure control, and
b) reactor level control

433.4.4 Safety System Conditions

(1) Normal pre-startup instrumentation checks are completed.

(2) The nuclear instrumentation safety system will be operating with 2 of 4 coincidence logic at all power levels involved in Test 433.

(3) The Log N channel will be key bypassed, since boiling noise is expected to cause period trips.

(4) No bypasses will be operated in the power range channels #5, #6, #7, and #8 during the conduct of Test 433.

(5) Instrumentation channels 1 through 8 are ON and are operational, in accordance with Tech Spec requirements.

(6) All set points for control systems and safety system will be in accordance with the applicable 278 series test, appropriate to the power level at which Test 433 is being conducted. No change in set points will be made from the 278 series unless it is specifically requested as a separate change to Test 433.

433.5 Hazards and Precautions

Table 1 is a summary of the most severe transients expected during Test 433. Note that the values listed in this table show the highest expected superheater temperatures. Also listed on Table 1 are maximum permissible indicated values of several imporant parameters. If either of the indicated parameters start to exceed these maximum permissible values, that particular test is to be halted for Operations Committee review.

The following transients will be considered in this section: (Refer to Table 1 for safety actions and expected max transient conditions.)

(1) change in feedwater temperature;
(2) change in feedwater flow;
(3) change in superheater exit steam pressure set point;
(4) recirculation flow changes; and
(5) control rod positioning.

433.5.1 Change in Feedwater Temperature

The reactivity effect due to a change in feedwater temperature was described in Sec. 433.3. Simulator studies of reactor response to changes in feedwater temperature were conducted and are given in Figs. 433.8.1, 433.8.2, and 433.8.3. The first two of these, which involve increasing feedwater temperature, indicate decreasing power and superheater fuel temperature as can be expected. Figure 433.8.3 indicates response of reactor power and superheater fuel hot spot temperature to decreasing feedwater temperature 15° (from 340 F to 325 F).

During normal operation and during this test, a runback is initiated if feedwater temperature drops below 320 F.

433.5.1.1 Change in Feedwater Temperature Set Point

A change in feedwater temperature set point introduces an error signal into the temperature control system which operates a valve controlling steam to the high pressure feedwater heater #14.

433.5.1.2 Change in Mode of Operation of Feedwater Temperature Control System

A change in the mode of operation of the feedwater temperature control system can lead to a change in feedwater temperature. The magnitude of change will depend on the ability of the instrumentation to affect a smooth transfer (from HAND to AUTO).

433.5.2 Change in Feedwater Flow

Changes in feedwater flow will be accomplished manually, the maximum change at 20% and 40% power being 40% of the existing flow. Figure 433.8.4 shows the analog simulator results for fixed changes in feedwater flow of 20% of full power (rather than 20% of existing flow) at initial power levels of 100%, 50% and 20%.

The magnitudes of the transients for identical feedwater changes are seen to be functions of the initial power level, becoming smaller as initial power level is increased. Since Test 433 specifies proportionately reduced changes in feedwater flow at lower power levels, % power, % steam flow, and superheater temperature deviations from initial values should be approximately equal at each power level.

Safety actions available in the event of excessive changes in feedwater flow are:

(1) high power scrams on channels 5 and 6 (also channels 7 and 8 at 100% power);

(2) low-water-level scram;

(3) high-water-level scram; and

(4) high superheater exit steam temperature scram.

433.5.3 Change in Superheater Exit Steam Pressure Set Point

The P_2 set point will first be changed 5 psi at various power levels between 20% and 100% of full power. Subsequently, set point changes of 10 psi are planned. If the response to these changes causes the system response to approach a safety system set point, the set point change of P_2 will be adjusted appropriately. Reactor response to changing the P_2 set point in a direction so as to increase superheater temperatures as indicated by the analog simulator is given in Figs. 433.8.5 and 433.8.6. The magnitude of the power and temperature response is seen to be approximately inversely proportional to the initial power level. That is, the change in reactor power and superheater hot spot temperature is approximately twice as great at 50% power than at 100%

power for the same change in pressure set point. Therefore, the most severe transients during this test should occur at the 20% power level. Simulator results at this power level are well within safe limits, as specified in Sec. 433.1a, #4. (See Fig. 433.8.7.)

Safety actions available are a high-steam-temperature scram, high-steam-pressure scram (565 psig), low-steam-flow-to-power ratio scram.

433.5.4 Recirculation Flow Changes

Recirculation flow transients at power will be performed in the following manner:

(1) discharge butterfly valve motion, ganged;
(2) pump tripout, singly and in pairs; and
(3) pump startup, singly only.

Reactor response to all of these maneuvers has been analyzed with the analog simulator. In order to prudently perform this test, the flow transients will commence with butterfly valve maneuvers and progress to the larger, more rapid pump tripout transients. Recirculation flow can thus be increased by (1) opening discharge valves of operating loops, and (2) starting up pumps individually.

433.5.4.1 Discharge Valve Maneuvers

The initial recirculation flow changes at power will be obtained by closing and opening the discharge butterfly valves.

The fastest flow response is achieved by ganged operation of the (3) valves in the 45 to 100% power open range. Corresponding flow accelerations are calculated to be less than 455 gpm/sec and corresponding reactivity insertion rates are calculated to be less than 8¢/sec, regardless of reactor operating conditions.

433.5.4.2 Recirculation Pump Tripout

A review of pertinent recirculation pump and discharge valve operating procedure following pump tripout is first given. Following a pump tripout, the discharge valve is interlocked to move to the 6% open position regardless of initial valve position. The 6% valve opening is sufficient to allow enough bypass flow to prevent formation of a cold leg when other loops are operating, but it is small enough to keep the bypass flow below 1000 gpm/loop. Thus, following a two-pump tripout, the two discharge valves will move automatically to 6% open while the third loop continues to operate. If the operating loop discharge valve is 100% open, the flow through the core (W_{RW}) should be about 28,000 gpm.

The fastest and largest change in recirculation flow involved in Test 433 is accomplished by simultaneously tripping two recirculation pumps while at 100% power. A power and recirculation operating diagram which indicates acceptable operating regions for one, two, and three pump operation is given in Fig. 433.8.9. With discharge valves initially 100% open, power and recirculation flow can be expected to stabilize at about 40% of the initial values after a two-pump tripout (for a power-to-flow ratio of one). Power decreases with recirculation flow because of the initial rapid void volume increase in the core and is expected to decrease nearly proportionally.

Analog simulator results for a two-pump tripout (refer to Fig. 433.8.10) agree fairly well with the final values of power and recirculation flow indicated in the operation diagram for a power-to-flow ratio of one.

At no time during Test 433 should operating conditions exist which would cause pump cavitation; but it is possible that valve cavitation may be experienced as a result of tripping two recirculation pumps. However, the effect of cavitation during such a short period of time should be negligible.

The boiler power-flow protection system will initiate a scram should power exceed the allowable power at the existing recirculation flow by 30 Mw. The allowable power and burnout margin vs. flow are shown in Fig. 433.8.11. Also, a scram is initiated if superheater exit steam temperature should become excessive after the pumps are tripped. This safety trip is not expected to operate since the superheater temperatures are expected to continuously decrease after a pump trip. Should the third recirculation pump be accidentally tripped, controlled plant shutdown is automatically initiated.

433.5.4.3 Recirculation Pump Startup

A recirculation pump motor can be started only if its discharge valve is closed and neither of the other two valves are opening or closing. Furthermore, a separate interlock prevents more than one pump from being started simultaneously. When a pump motor is started, the discharge valve in that loop opens automatically to 45% open.

The maximum reactivity insertion rates coincident with pump startup will occur for one to two pump operation. The maximum flow rate-of-increase expected for one to two pump operation is 163 gpm/sec which is calculated to be worth about 7¢/sec, regardless of power level. The expected maximum for two to three pump operation is 135 gpm/sec which is calculated to be about 2.5¢/sec.

Reactor response as indicated by the analog simulator is given in Figs. 433.8.12 and 433.8.13 for one to two pump and two to three pump operation. The results in both cases indicate that reactor power follows recirculation flow closely, while moderate deviations occur in the other variables.

The same safety actions listed for pump tripout will apply for pump startup test. Figures 433.8.14 and 433.8.15 show power vs. recirculation flow and the flow protection scheme respectively.

433.5.5 Control Rod Positioning

Control rods will be withdrawn and inserted at various power levels and power-to-flow ratios. Reactivity addition rates will normally be quite low (less than 10¢/sec), since interlocks allow only one rod to be withdrawn at a time. Negative reactivity addition can be more rapid since rods can be gang inserted. Individual rods can be moved from banked positions no more than 2 in. Associated reactivity rates, reactivity insertions and expected transient ranges are given in Table 1. (See Fig. 433.8.8.)

433.6 Detailed Test Procedure

NOTE: Recirculation flow to be 100% of rated flow for all tests except where specified otherwise.

STEP 1: Reactor power at P_o per Test 278.

STEP 2: 2.1 Check to assure that all recorders mentioned in 433.4.1 are ON and operating.

2.2 When all variables have achieved a steady-state condition, read and record on the appropriate data sheet the following:

(1) power level from channels 4, 5, or 6 and 7 - P_4, P_5, and P_7, (#118, #120 and #122);

(2) recirculation water temperature - T_{RW} (#248D);

(3) superheater exit steam temperature - T_2 (#3A);

(4) superheater exit steam pressure - P_2 (#3B and #3C);

(5) reactor water level - L_W (#4F);

(6) feedwater flow - W_{FW} (#4D);

(7) feedwater temperature - T_{FW} (#4E);

(8) main steam flow - W_s (#4B);

(9) superheater fuel temperature - T_f (on Offner recorder);

(10) reactor dome pressure - P_1 (#261);

(11) recirculation flow - W_{RW}; and

(12) all control rod positions - CRP (#271).

STEP 3: 3.1 Check to assure prerequisites in 433.4 have been complied with.

3.2 Set chart speed on Offner Recorder to 1 mm/sec.

3.3 Record data as in Step 2.

3.4 Switch MAIN STM TO #14 HTR Selector Switch 148 to AUTO. Allow all variables to stabilize. RMC 148 to remain in AUTO position through Step 10.

3.5 Record data as in Step 2.

3.6 Restabilize reactor at P_o if necessary.

3.7 Increase feedwater temperature set point 45 F in three 15 F steps at MAIN STM to #14 HTR Selector Station 148. Allow all variables to stabilize between each 15° step.

3.8 Decrease feedwater temperature set point 45° in one step at Selector Station 148. Allow reactor to stabilize. Then decrease feedwater temperature set point 15° in one step. Allow reactor to stabilize.

3.9 Increase feedwater temperature set point 15 F at Selector Station 148 to return to 340 F.

3.10 Record data as in Step 2.

NOTE: All parameter changes will be done slowly and in step changes in accordance with the Ground Rules (433.1a) prior to the test transient.

NOTE: These changes in temperature set point may be modified if the present reference temperature of 340 F is changed.

STEP 4: 4.1 Remove the charts from the recorders listed in 433.4.1.

4.2 Check to see that the following information has been recorded on each chart for each recorded system variable:

(1) identification (e.g., P_1, W_s);

(2) time and date of test and initials of cognizant engineer;

(3) scale factor;

(4) chart speed;

(5) the starting points of the test and the initial values; and

(6) power level.

STEP 5: Stabilize reactor power at P_o.

STEP 6: 6.1 Check to assure that prerequisites in 433.4 have been complied with.

6.2 Set chart speed on the Offner recorder to 1 mm/sec.

6.3 Record data as in Step 2.

6.4 Set the feedwater control system to manual, and watching the main feedwater flow meter, at $P_o = 38$ Mwt decrease feedwater flow by 40% of the existing value. Record data and return flow rate to original value. At an initial power of 76 Mwt decrease feedwater flow by 40% of the existing value, record data, and return flow rate to the initial value. At initial powers of 114, 152 and 190 Mwt, decrease feedwater flow by 20% of full power value, record data and return flow rate to the initial value.

NOTE: It is estimated that about 3.5 min at 50,000 lb/hr can be tolerated (operating from 0 in. indicated initial level) before the low level trip point is reached. Note other limits placed on this test -- described in Table 1.

6.5 Immediately after reactor systems (except water level) have stabilized, slowly return feedwater flow to the initial flow.

6.6 Record data as in Step 2.

6.7 Remove recorder charts as in Step 4.

6.8 Balance the MAIN FEEDWATER VALVE Selector Station (RMC-27) by adjusting the SET POINT control knob so that the TRANSFER pressure (Gage C) is equal to the CONTROL pressure (Gage D).

6.9 Turn the Selector Station (27) transfer switch to AUTO and return water level to the initial level.

STEP 7: Stabilize reactor power at P_o.

STEP 8: 8.1 Check to assure prerequisites in 433.4 have been complied with.

8.2 Set chart speed on the Offner recorder to 1 mm/sec.

8.3 Record data as in Step 2.

8.4 Decrease superheater exit steam pressure from existing pressure by adjusting pressure set point knob at Selector Station 129 for a 5 psi decrease.

8.5 After all reactor systems have stabilized, increase pressure set point 5 psi.

8.6 Record data as in Step 2.

8.7 Remove recorder charts as described in Step 4.

STEP 9: Stabilize reactor power at P_o.

STEP 10: 10.1 With the reactor at equilibrium at 38 Mw, the estimated rod height of the controlling group (Group II) will be as specified in Test 278.2A procedure.

10.2 With the pressure control system in AUTO, raise Group II 2 in.

10.3 When the reactor has stabilized, record data as in Step 2.

10.4 After the data is recorded, return Group II to the initial position for P_o.

10.5 Shift the pressure control system to MANUAL.

10.6 Raise Group II 2 in. without changing the position of the Regulating Dump Valve.

10.7 When the reactor stabilizes under this new equilibrium position, record data as in Step 2.

10.8 After the data is recorded, adjust the Regulating Dump Valve to obtain the equilibrium conditions attained in Step 10.2 when the pressure control was in AUTOMATIC.

10.9 Record data as in Step 2.

10.10 After the data is recorded, return Group II to the initial position for P_o without changing the position of the Regulating Dump Valve.

10.11 Record data as in Step 2.

10.12 Return the pressure control system to AUTOMATIC.

STEP 11: 11.1 Check to assure prerequisites in 433.4 have been complied with.

11.2 Set chart speed on the Offner recorder to the desired speed.

11.3 Record data as in Step 2. Initial power should be 38 Mwt and initial recirculation flow 100%.

11.4 Close recirculation pump discharge valves to 60% open by GANG operation. Power will fall to about 28.5 Mwt. Take a heat balance.

11.5 Open discharge valves to 100% open by GANG operation.

11.6 Record data as in Step 2.

NOTE: All reactor systems to stabilize between each step and indicate step number on all recorder charts. Level control, and feedwater temperature control systems should be on AUTO. Pressure control system is on AUTO on dump valve control only, dump valve handling all the steam flow.

STEP 12: With the reactor stabilized at 76 Mwt per Test 278, perform each of the following tests:

(1) Change in feedwater temperature set point - Step 3.
(2) Change in feedwater flow rate - Step 6.
(3) Change in superheater exit steam pressure set point - Step 8.
(4) Change in control rod position - Step 10.

STEP 13: Recirculation Flow Changes

(1) Prior to operation below 100% recirculation flow, reset the power-flow set points.

Based upon the 40% power results, new set points for the Recirc. Flow to Power scram will be made for the higher power level steps (i.e., 60%, 80%, 100%).

(2) Whenever the procedure calls for "RESET POWER-TO-RECIRCULATION FLOW SCRAM SETTING" the following steps will be performed:

a. Calculate the actual power-to-flow for the present scram setting, using the specified heat balance data and specified minimum recirculation flow.

b. If actual power de-calibration is greater than prescribed value reset power-flow scram circuit value.

(3) Whenever the procedure calls for "RESET STEAM OUTLET TEMPERATURE SCRAM SETTING," the steam outlet temperature will be reset in accordance with the following criteria:

$$\Delta T_{out} = (T_{out} - T_{in}) \left[\frac{(T_F - T_{in} + \Delta T_F)}{(T_F - T_{in})\,F} - 1 \right]$$

ΔT_{out} = scram setting above equilibrium T_{out}, F

T_{out} = equilibrium outlet steam temperature, F

T_{in} = equilibrium inlet steam temperature, F

T_F = equilibrium fuel thermocouple temperature, F

ΔT_F = permissible increase in fuel thermocouple temperature above equilibrium value

F = 1.20 for CRG III between 45 in. and 73 in. and for CRG I between 0 and 73 in.

F = 1.10 for CRG II between 0 and 73 in.

NOTE: Equilibrium refers to conditions immediately preceding the transient. The value of ΔT_F is +75 F for initial runs. Higher values of ΔT_F up to +100 F must be approved by the Operations Committee.

13.1 Obtain heat balance for equilibrium conditions at 76 Mwt and 100% recirculation flow rate. Reset Steam Outlet Temp Scram Setting.

13.2 Start up the "down" pump and open discharge valve to 100% open position. When equilibrium is reached, power should again be at 76 Mwt and flow at 100%.

13.3 At initial values of 76 Mwt and 100% recirculation flow, drop out one pump. When flow has stabilized, gang discharge valves on remaining two pumps to 45% open. Power should fall to about 22%. Take a heat balance. Gang valves back to 100% open.

13.4 When power and flow have stabilized (at about 30% power and 50,000 gpm) drop out the second pump. Power should stabilize at about 20% and flow at about 27,500 gpm. Take a heat balance.

13.5 Move the one valve to 20,000 gpm indicated flow and take a heat balance. Return the valve to 100% open.

13.6 Start up the second pump and move its discharge valve to 100% open.

13.7 Start up the third pump and move its discharge valve to 100% open. Power should now be at 40% and recirculation flow at 100%.

13.8 Drop out two pumps simultaneously. Record power and flow transients.

13.9 Start up the second and third pumps and move their valves to the 100% open position.

13.10 From initial valves of 40% power and 100% flow, drop out three pumps simultaneously. Controlled shutdown should be initiated; flow should drop to about 8300 gpm when the discharge valves are 100% open.

13.11 Gang close all three recirculation flow valves slowly in steps to 60% open. Obtain heat balance at 60% valve position. Reset Steam Outlet Temp. Scram Setting. Gang open all three valves slowly in steps to 100%.

STEP 14: 14.1 Repeat Steps 12 and 13 when the dump valve is open to about 10% and is on AUTO and the inlet valves are handling the remainder of the flow on load limit control.

STEP 15: This step is to be performed only if Test 432 is deferred after the initial escalation of power to 100%.

Stabilize the reactor power at 76 Mwt per Test 278 and at 100% recirculation flow. Pressure control system is on AUTO.

(1) Move one Group V rod in 2 in. into the core and immediately withdraw the same rod 2 in. to the former equilibrium position. This rod motion should be accomplished so as to generate a "triangular reactivity shape."

(2) Observe and record the response of reactor power on either channel 5 or 6 and on three selected in-core ion chambers. Also record, superheater fuel temperature, reactor steam flow, reactor vessel pressure, and exit steam temperature.

(3) It is expected that reactor power will respond to this rod motion in a highly damped fashion. Determine the damping factor, if slight oscillatory response is observed, or the time for reactor power to return to 90% of equilibrium value. Plot either the damping factor or "time to return to 90% of equilibrium" versus power. As power increases slight decreases in damping may be observed.

(4) Repeat the rod motion in step (1) in a periodic fashion for 10 complete reactivity cycles. This will result in a periodic disturbance with a period of about 4 sec. Observe the damping of the system and record all variables.

NOTE: This qualitative stability determination is not intended to supplant the quantitative data that will be determined with the pile oscillator. This step should, however, produce results that will enable the reactor operators to escalate power.

STEP 16: Stabilize the reactor power at 114 Mwt per Test 278 for each of the following tests:

(1) change in feedwater temperature set point - Step 3

(2) change in feedwater flow - Step 6

(3) change in superheater exit steam pressure set point - Step 8

(4) change in control rod position - Step 8

(5) recirculation flow changes - Step 13

(6) qualitative stability performance - Step 15

STEP 17: Stabilize the reactor power at 152 Mwt per Test 278 for each of the following tests:

(1) change in feedwater temperature set point - Step 3

(2) change in feedwater flow - Step 6

(3) change in superheater exit steam pressure set point - Step 8

(4) change in control rod position - Step 8

(5) recirculation flow changes - Step 13

(6) qualitative stability performance - Step 15

STEP 18: Stabilize the reactor at 190 Mwt per Test 278 for each of the following tests:

(1) change in feedwater temperature set point - Step 3

(2) change in feedwater flow - Step 6

(3) change in superheater exit steam pressure set point - Step 8

(4) change in control rod position - Step 8

(5) recirculation flow changes - Step 13

(6) qualitative stability performance - Step 15

433.7 Report

The report shall include the following:

(1) description of any deviation from the test procedure;
(2) description of any unusual conditions encountered during the test;
(3) all recorder chart paper resulting from the test; and
(4) a complete set of data taken during the test.

MATERIAL AFFECTED BY REVISION EIGHT

November 25, 1966

Summary

This eighth revision to the Test 433 Procedure concerns performance of the test with the "split flow" arrangement of the turbine inlet valves and the dump valve. Because of prior problems with the turbine inlet valves operating on AUTO, the mode of permissible operations will be as follows until new inlet valve actuator equipment is installed. The dump valve will be opened to about 10% and will automatically control pressure, while the inlet valves will be placed on manual, i.e., on "load-limit" control.

This revision to the test procedure also defers performance of the pump trip tests to a later date, in order to speed up the testing schedule and permit earlier escalation to higher powers. The recirculation flow tests will consist of flow reduction by ganged operation of the pump discharge valves.

Analog Computer Results

A recent series of analog computer studies was completed to show whether any major differences could be expected in test results with the "split flow" mode of pressure control, compared to pressure control by either inlet or dump valves alone. Because of the nature of the simulation, as long as the dump valve is open far enough to handle changes in pressure produced by the planned disturbances of Test 433, no difference is seen in system response between the "split-flow" arrangement and control with the dump valve handling all the flow. The simulation does not take into account any non-linearities in dump valve characteristics that may exist at very low flows.

The results of these studies run at powers between 40% and 100% show that for all disturbances planned in Revision 8 of Test 433 that the dump valve can handle all expected pressure perturbations if it is on AUTO control and is opened to 10%. (Only the pump trip test at 40% initial power produced a perturbation too large for the dump valve to handle when on auto at 10%. On this basis, a pump trip test would not be recommended when the "split flow" mode of control is used.)

Chart A shows comparisons of the results of various disturbances on dump valve control and split flow control with the dump valve at 10% open. These cases are for 40% power; other studies were for 60%, 80% and 100% power and show results similar to those listed in the chart.

Based on these studies it is concluded that approximately the same results should be obtained for Test 433 disturbances, whether the dump valve is handling all of the existing flow or is open only 10%.

CHART A

COMPARISON OF RESPONSES WHEN DUMP VALVE CONTROL HANDLES ALL THE FLOW AND WHEN ON 10% DUMP VALVE CONTROL AT 40% POWER

Condition	Disturbance	MAX Δn % of full power	MAX ΔT_F °F	MAX ΔP_2 psi
All flow through dump valve	ΔW_{FW} = +20.8 lb/sec in 2 sec	+6.4%	+75 F	+7
10% of full flow through dump valve, 30% flow through fixed inlet valves	ΔW_{FW} = +20.8 lb/sec in 2 sec	+6.8%	+74 F	+7
Flow through dump valve	control rod motion +10¢/sec for 5 sec	+3.5%	+30 F	+3
10% flow through dump, 30% through inlet	control rod motion +10¢/sec for 5 sec	+3.5%	+30 F	+3
All flow through dump valve	ΔT_{FW} = −15 F	+1.4%	+18 F	+1.6
10% flow through dump valve, 30% through inlet valve	ΔT_{FW} = −15 F	+1.6%	+19 F	+1.7

TABLE 1

EXPECTED TRANSIENT VALUES AND SAFETY ACTIONS FOR 433 AT 38 MW

Parameter Changed	Initial Value	Amount Changed	Initial Power	Initial Recirc. Flow	Calc. Initial S.H. Hot Spot	Calc. Peak Transient Hot Spot	Max. Calc. Value of Hot t/c on A-18 #O-10 or #O-06	Relevant Safety Action	Safety Action Set Pt.	Max. Exp. Value Safety Action Parameter	Exp. Max Δk Rate	Exp. Total Δk	Exp. Peak Nominal t/c Reading on #O-10 or #O-06	Max. Permissible Indicated Parameter Values
T_{FW}	340 F	-15 F	38 Mw	100%	1180 F	1202 F	1182 F	high power scram	55 Mw	40 Mw	0.02¢/sec		897 F	Power: P_o ± 6 Mw S.H. t/c: #O-10 T_o + 66 F
								low T_{FW} CSD	320 F	325 F				
W_{FW}	123,000 lb/hr	+49,300 lb/hr	38 Mw	100%	1180 F	1232 F	1212 F	high power scram	55 Mw	42.4 Mw	1.5¢/sec		927 F	Power: P_o ± 12 Mw S.H. t/c: #O-10 T_o + 150 F
								high water level scram	+4 in.	+3 in.				
P_2	557 psia	+5 psi	38 Mw	100%	1180 F	1255 F	1235 F	high power scram	55 Mw	42.8 Mw (+5 psi)	2.5¢/sec		950 F	Power: P_o ± 12 Mw S.H. t/c: #O-10 T_o + 150 F } +5 psi change
								high stm. pressure low stm. flow scram	580 psia 80,000 lb/hr					
CRP	Group II controlg. at 12 in. Groups I, III, IV, V out	+2 in. per rod	38 Mw	100%	1180 F	1222 F (for 3¢/sec 4 rods moved in Group II each 2 in.)	1202 F	high power scram	55 Mw	52 Mw	3¢/sec per rod max.	17¢ per rod max. 57¢ per Group II max.	917 F	Power: P_o ± 12 Mw S.H. t/c: #O-10 T_o + 126 F
W_v *	48,000 gpm	+20,000 gpm ganged valves	34 Mw	80% (48,000 gpm)	1180 F			high power scram	55 Mw		8¢/sec			
								high power-to-flow scram	68 Mw					

NOTE: (1) In all cases, P_o = 38 ± 2 Mw.

(2) If oscillations occur after any disturbance the magnitude of the oscillation must be less than the max. permissible power indicated in the last column of this table. The oscillations must also be damped, i.e., the power must converge to a steady state within the magnitude limits.

TABLE 2

TRANSIENT VALUES AND SAFETY ACTIONS FOR TEST 433 AT 76 MW AND 114 MW; MOST SEVERE CASES

Parameter Changed	Initial Value	Maximum Amount Changed	Initial Power	Calc. Init. S.H. Hot Spot Design Values Compounded H.S. Factor (NOTE 6)	Init. S.H. Hot Spot Test Values (NOTE 6)	Calc. Max. Transient ΔT_F	Estimated Peak Transient Hot Spot Based on Test Results	Exp. Peak Nominal Temp. t/c #O-10	Range of ΔT_2 Peak Transient Temperature	Max. Permis. Indicated Values
								(Initial) (Peak) To		
T_{FW}	340 F 340 F	-15 F -15 F	76 Mw 114 Mw	1244 F 1286 F	1113 F 1128 F	20 F 16 F	1133 F 1144 F	(770) 790 F (777) 793 F	5 - 7 F 4 - 5 F	Power: $P_o \pm 12$ Mw S.H. t/c #O-10: $T_o + 75$ F
W_{FW}	246,000 lb/hr 369,000 lb/hr	+98,300 lb/hr +123,000 lb/hr	76 Mw 114 Mw	1244 F 1286 F	1113 F 1128 F	70 F 45 F	1183 F 1173 F	(770) 840 F (777) 822 F	18 - 23 F 11 - 15 F	Power: $P_o \pm 20$ Mw S.H. t/c #O-10: $T_o + 75$ F
P_2	555 psi 555 psi	+5 psi +5 psi	76 Mw 114 Mw	1244 F 1286 F	1113 F 1128 F	60 F 50 F	1173 F 1178 F	(770) 830 F (777) 827 F	15 - 20 F 13 - 17 F	Power: $P_o \pm 12$ Mw S.H. t/c #O-10: $T_o + 75$ F
CRP	rod program at time of test	group movement +2 in. +2 in.	76 Mw 114 Mw	1244 F 1286 F	1113 F 1128 F	50 F 55 F	1163 F 1183 F	(770) 820 F (777) 832 F	13 - 17 F 14 - 18 F	Power: $P_o \pm 12$ Mw S.H. t/c #O-10: $T_o + 75$ F
W_v	48,000 gpm 48,000 gpm	gang 3 valves open from 45% to 100% gang 3 valves open from 45% to 100%	≈50 Mw ≈80 Mw	1244 F 1286 F	1113 F 1128 F	75 F 75 F	1188 F 1203 F	(770) 845 F (777) 852 F	19 - 25 F 19 - 25 F	Power: $P_o \pm 12$ Mw S.H. t/c #O-10: $T_o + 75$ F

NOTES:

1. Table 2 lists the results of disturbances planned during Test 433 at 76 Mw and 114 Mw that will lead to the highest transient superheater fuel temperatures. As indicated in the ground rules on pages A-1 and A-2 of Test Procedure 433, and as discussed during Pathfinder Safety Committee meetings, the planned reactor system perturbations will be done first slowly and in smaller steps than indicated in Table 2. The disturbances performed in Test 433 were chosen to be representative of actual operational maneuvers and were deemed necessary to verify the correct and safe operation of the entire reactor plant. The ultimate "full" disturbance magnitudes listed on Table 2 and in Test 433 were chosen such as to lead to superheated fuel transient temperatures that were reasonable and safe and yet the disturbances were of a magnitude such as to yield reliable and understandable information about the reactor system operation.

2. The temperatures listed in the fifth column from the left, "Calculated Initial Superheater Hot Spot Temperatures Design Values with Compounded Hot Spot Factors," are based on expected Group II rod positions at 40% and 60% power, with equilibrium xenon in the core.

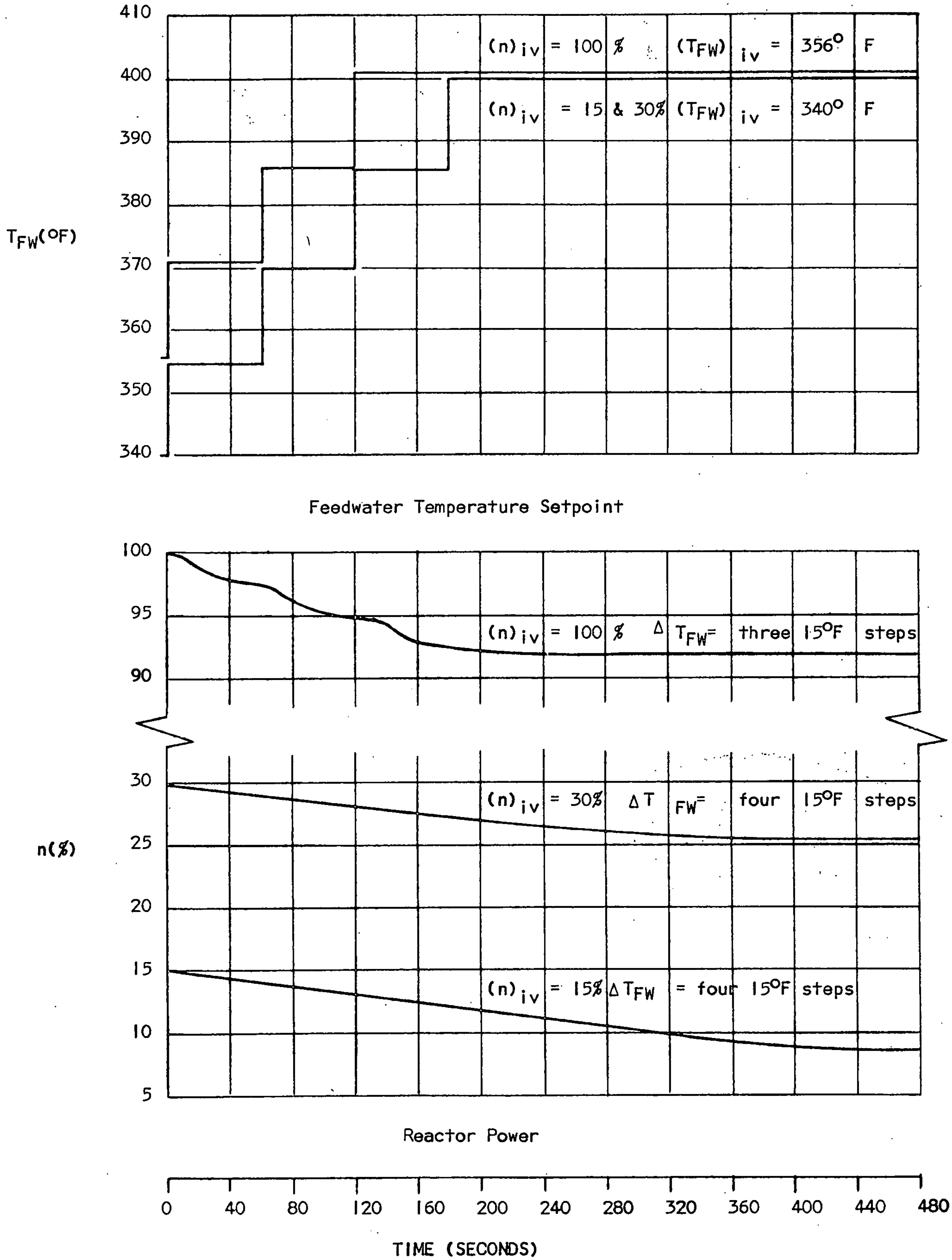
3. The second column from the right lists the range of bulk exit steam temperature expected for these tests. The lower temperature was obtained from the relation between fuel temperature transients (ΔT_F) and bulk steam temperature transients (ΔT_2) (a factor of 4) observed at 20% power tests. The higher temperature was obtained from the calculated relation between ΔT_F and ΔT_2 (a factor of 3). Since all fuel temperature transients were conservatively calculated it is believed that all disturbances, with the exception of the recirculation flow perturbations, can be performed with the steam temperature scram settings as listed in the next paragraph (#4).

4. The high steam temperature scram set points are to be set according to the following philosophy: For the first run of each transient, the high steam temperature scram set point will be set no higher than 15 F above the existing steady state steam exit temperature. If this setting causes a scram or is likely to cause a scram on a subsequent run when the planned full perturbation is made to get meaningful experimental data, the scram set point may be raised to as high as 25 F above the existing steady state steam exit temperature on approval from the Operations Committee.

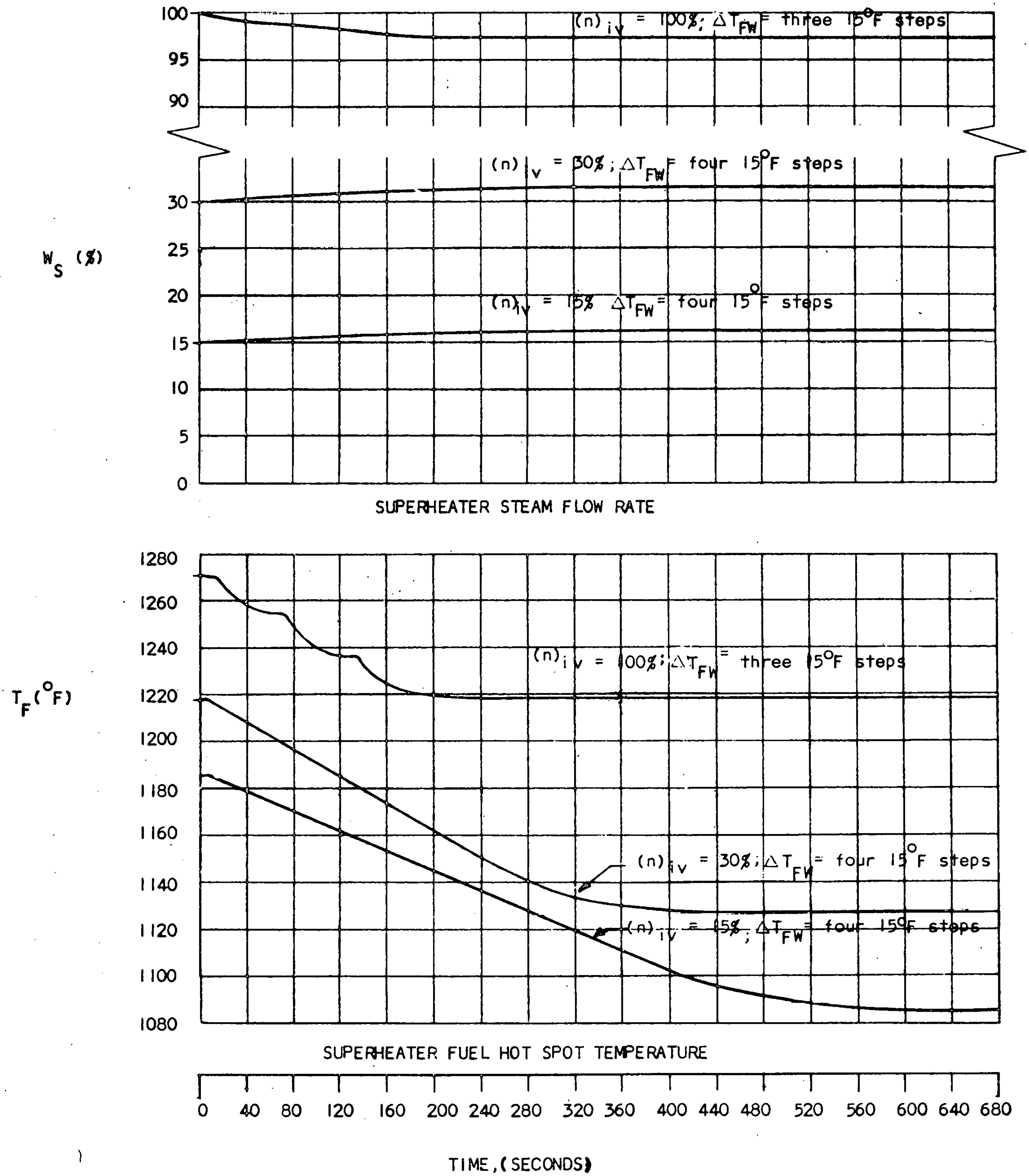
5. If it can be determined that it is permissible to escalate to full power prior to Test 432 - Transfer Function Test with the Pile Oscillator - the following statement regarding system stability is an operating limit: If any tendency toward divergence occurs, or if the magnitude of the oscillations is significantly greater than anticipated from previous analog computer studies, then Test 433 will be stopped, the phenomenon will be investigated and a review by the Operations Committee will be made before the test is continued.

   Rod perturbations which are to be used as a qualitative check on system stability until the pile oscillator data can yield quantitative stability results are described on pages A-21 and A-22 of this test procedure.

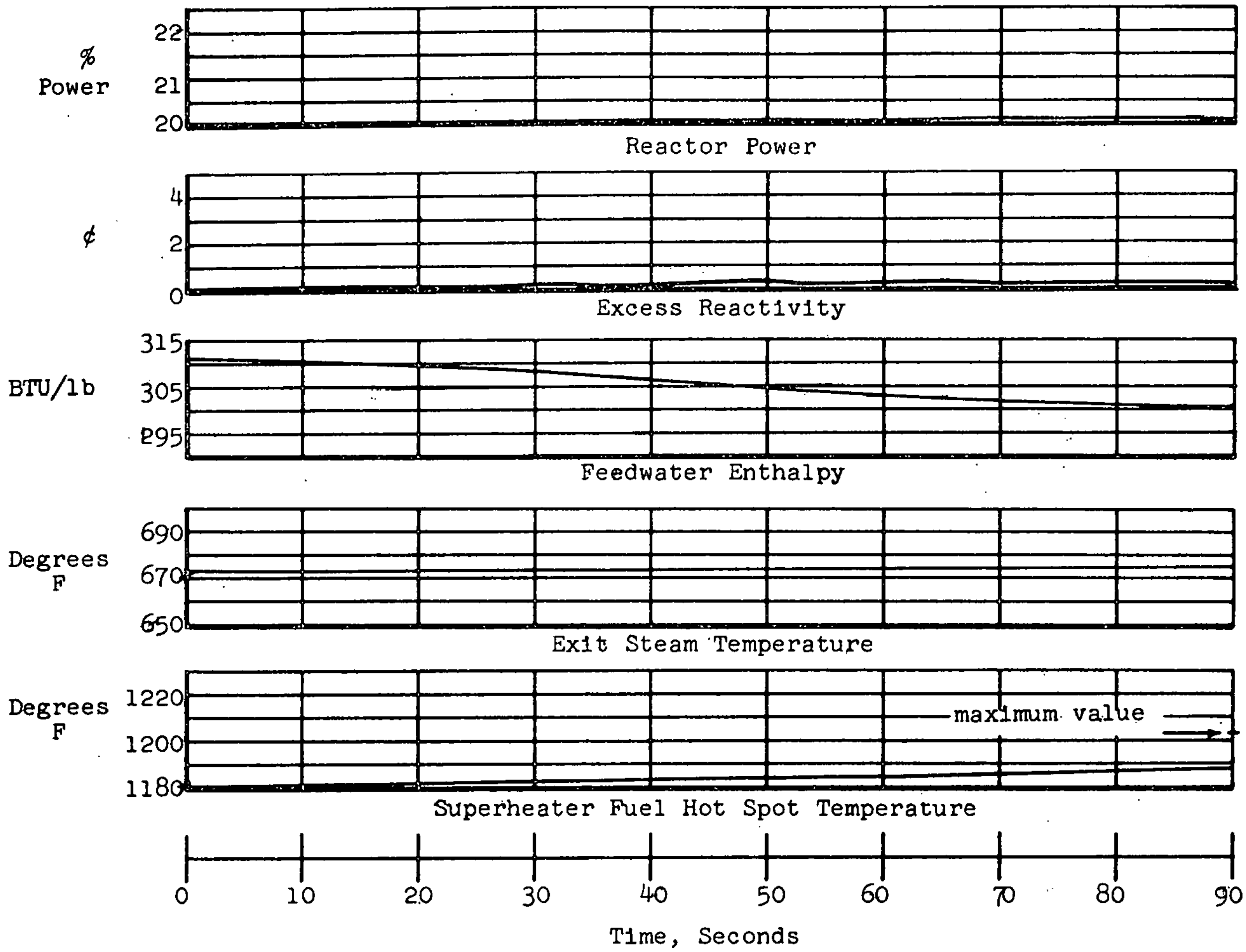
6. See memorandum dated July 21, 1966, L. L. Kintner to J. T. Stone. Subject: Maximum Equilibrium Superheater Fuel Temperatures for Test 433.



MANUAL CHANGE OF FEEDWATER TEMPERATURE SETPOINT FIG. 433.8.1

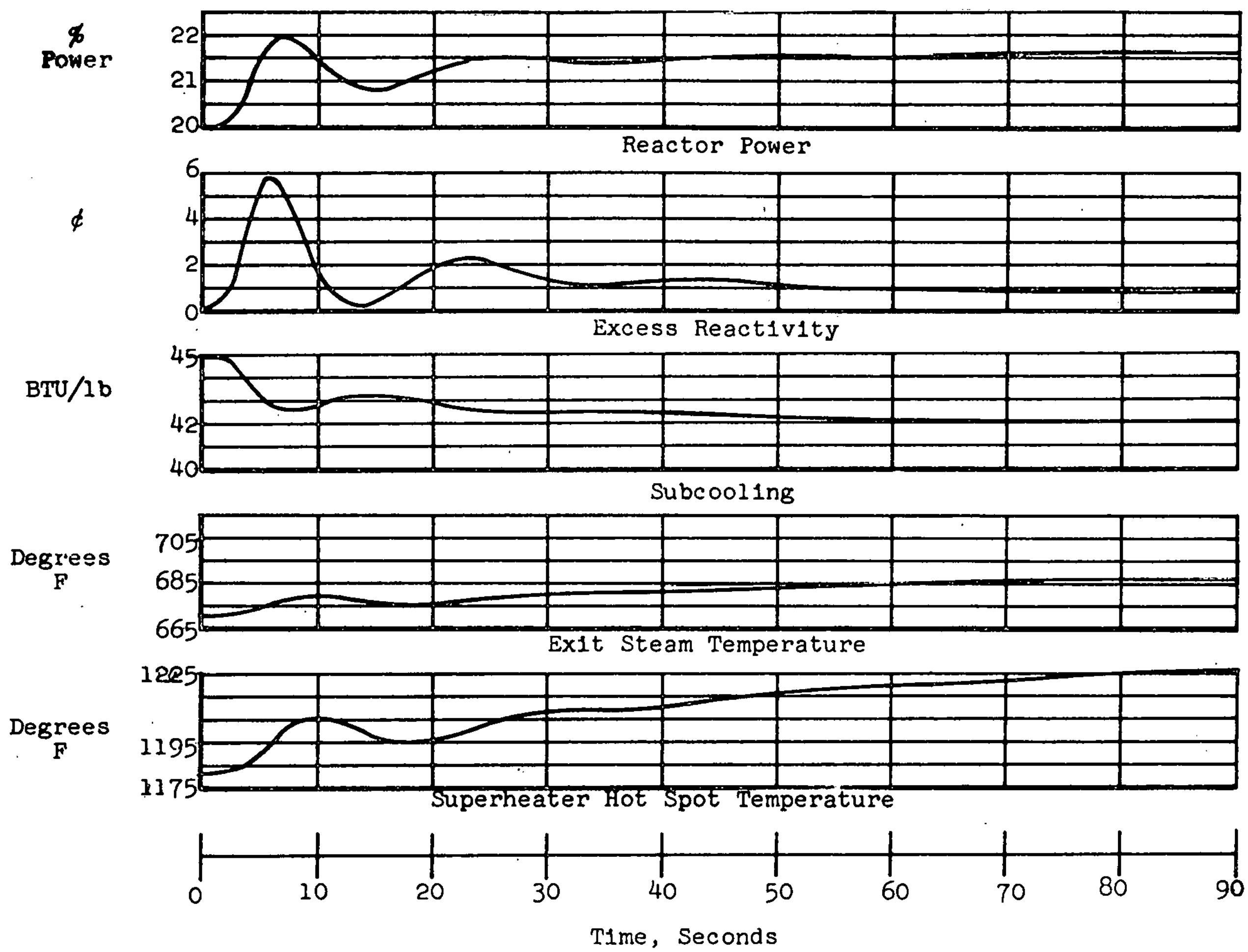


MANUAL CHANGE OF FEEDWATER TEMPERATURE SETPOINT FIG. 433.8.2



Initial Power = 20%   Initial Recirculation Flow = 100%

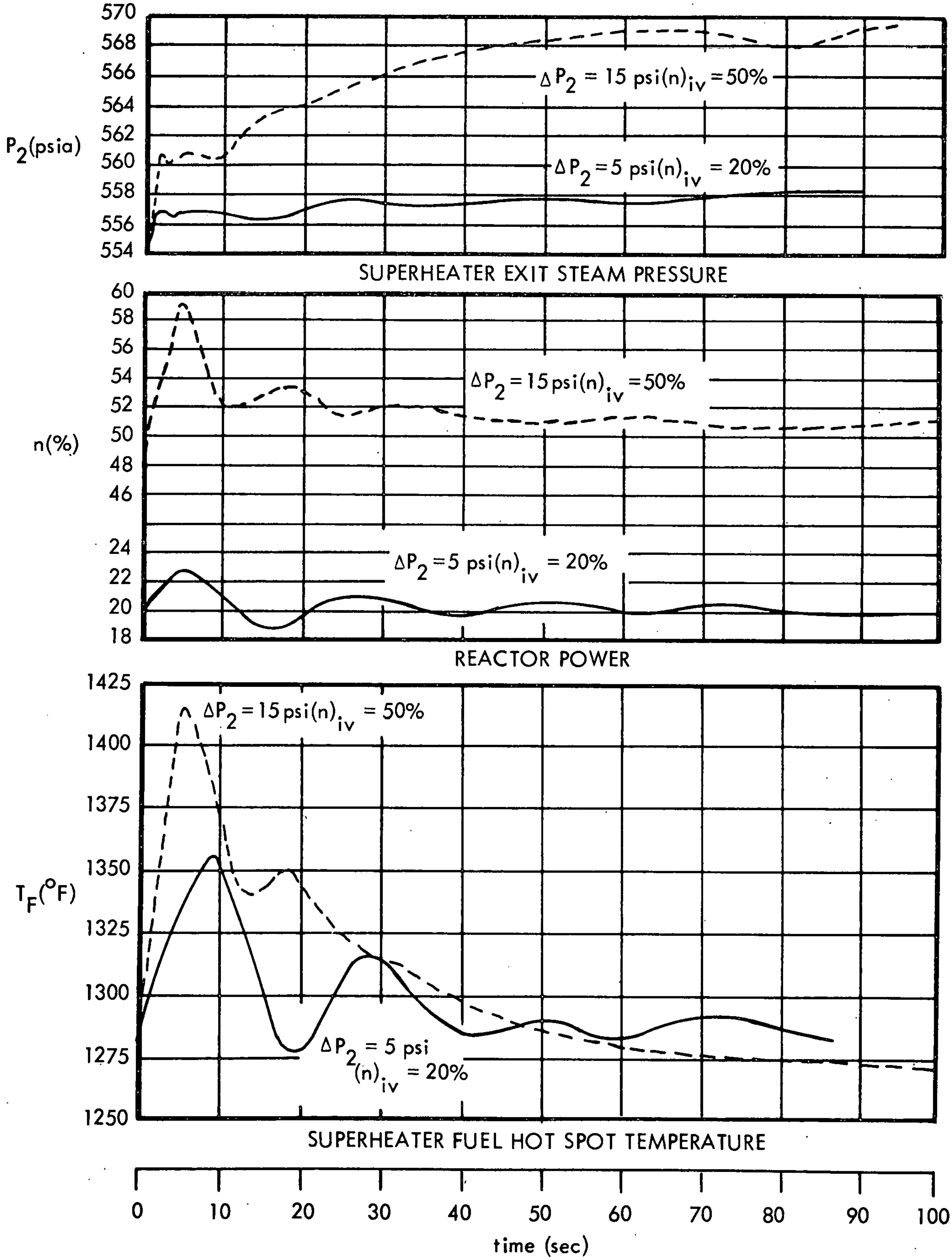
CHANGE IN FEEDWATER TEMPERATURE FROM 340 F TO 325 F   FIG. 433.8.3



40% of Existing Value ΔW_{fw} - +49,300 lb/hr
Initial Power = 20%   Initial Recirculation Flow = 100%

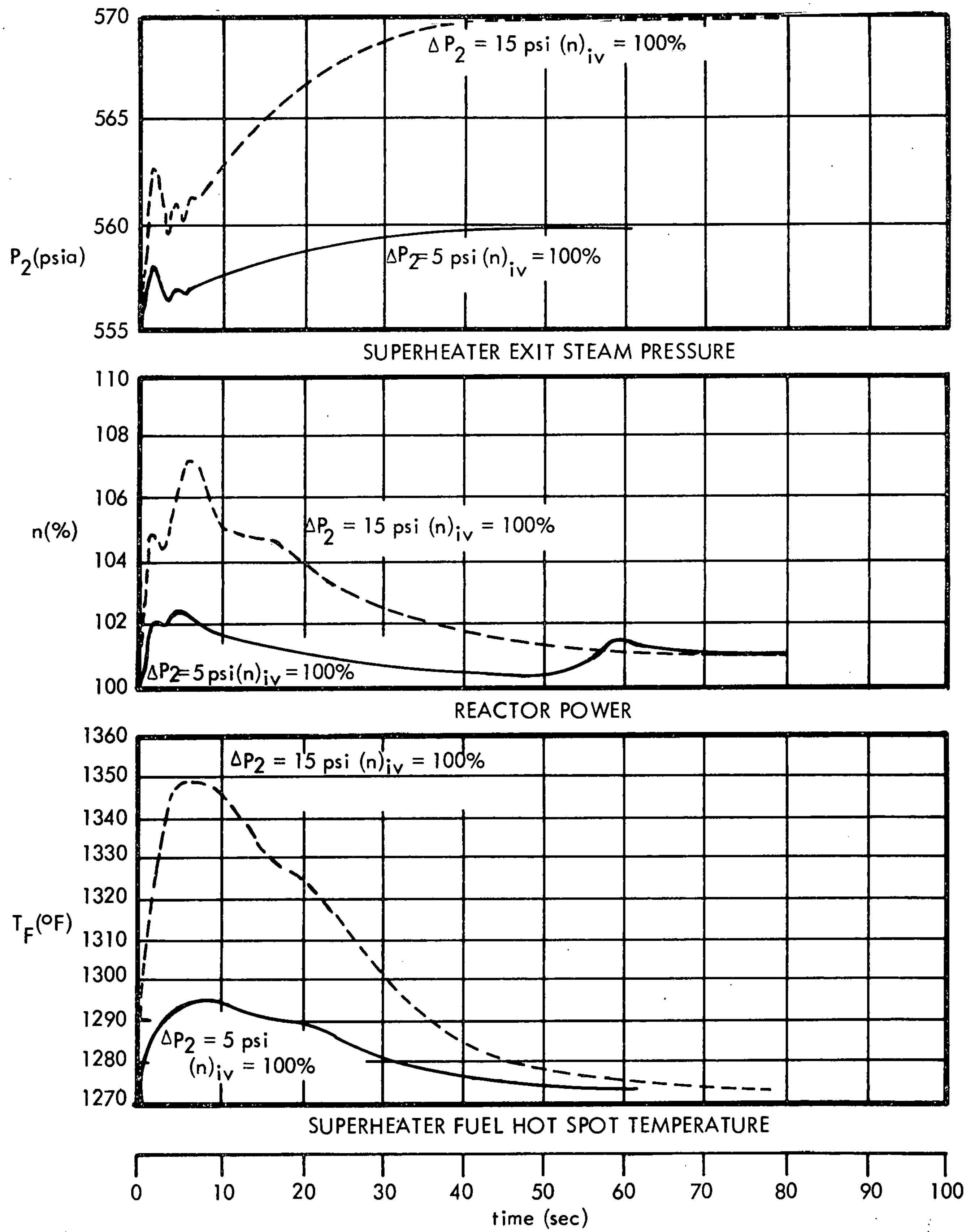
CHANGE IN FEEDWATER FLOW

FIG. 433.8.4

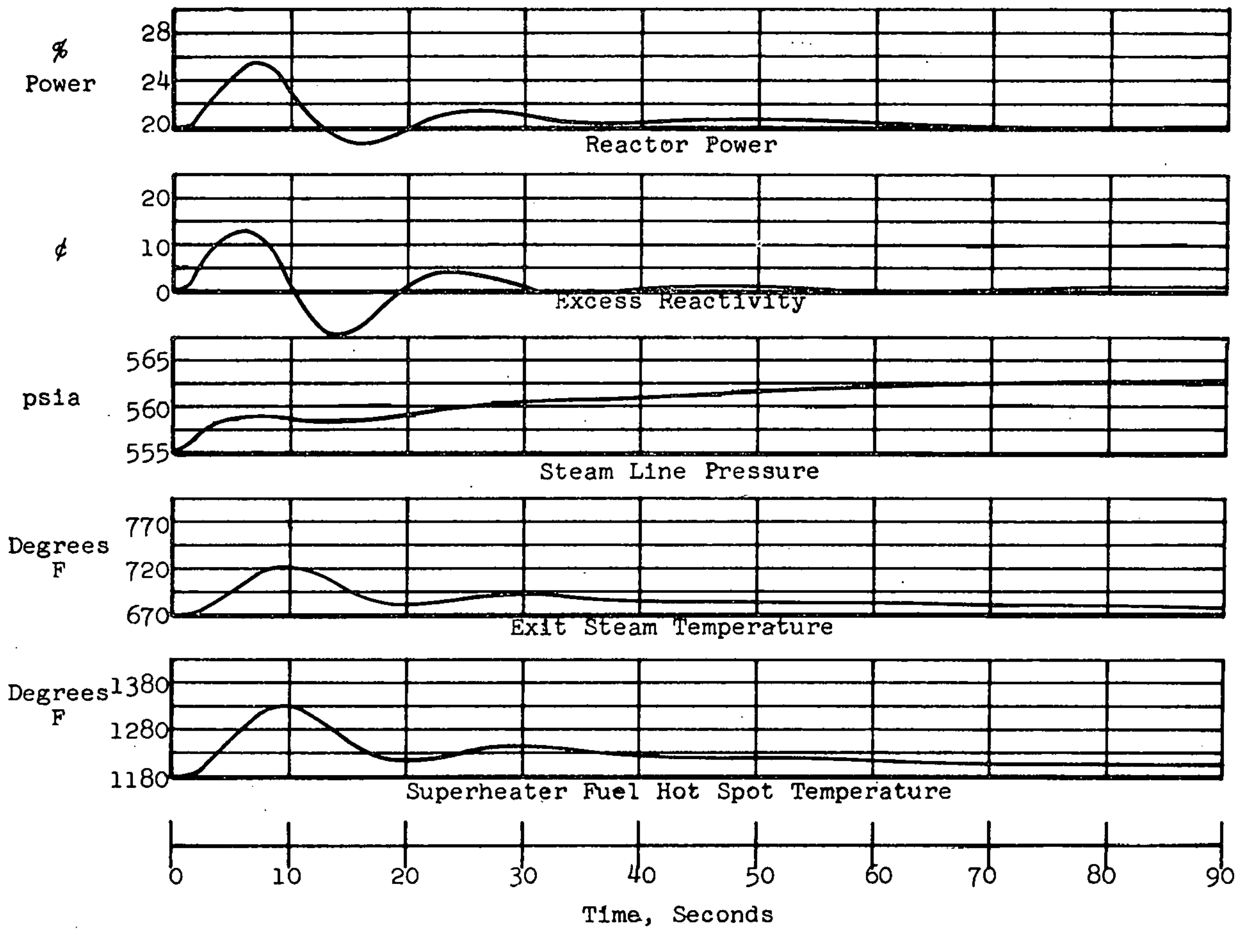


REACTOR RESPONSE TO CHANGE IN P_2 SET POINT

FIG. 433.8.5



REACTOR RESPONSE TO CHANGE IN P_2 SET POINT FIG. 433.8.6

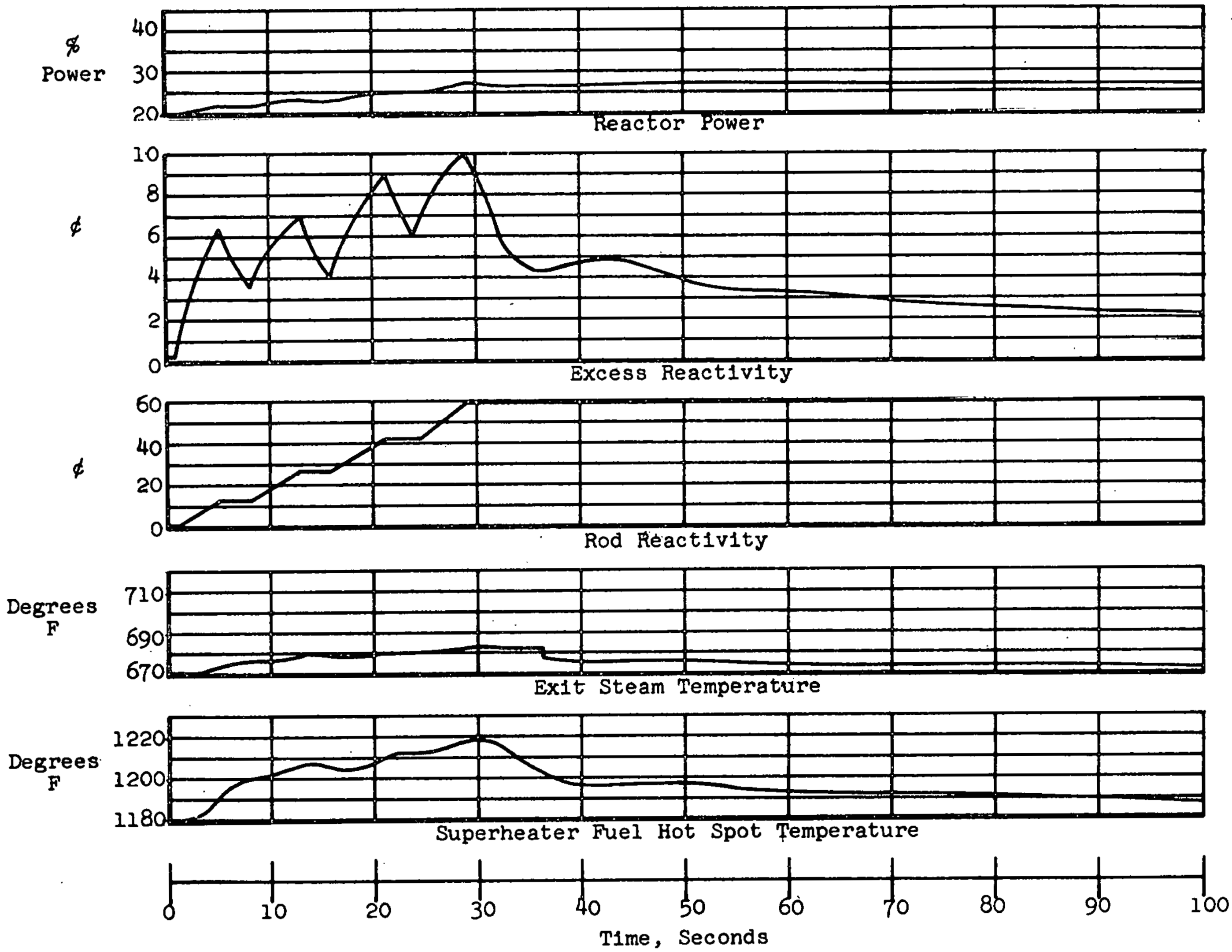


$\Delta P_2 = +10$ psi

Initial Power = 20% Initial Recirculation Flow = 100%

CHANGE IN P_2 SET POINT

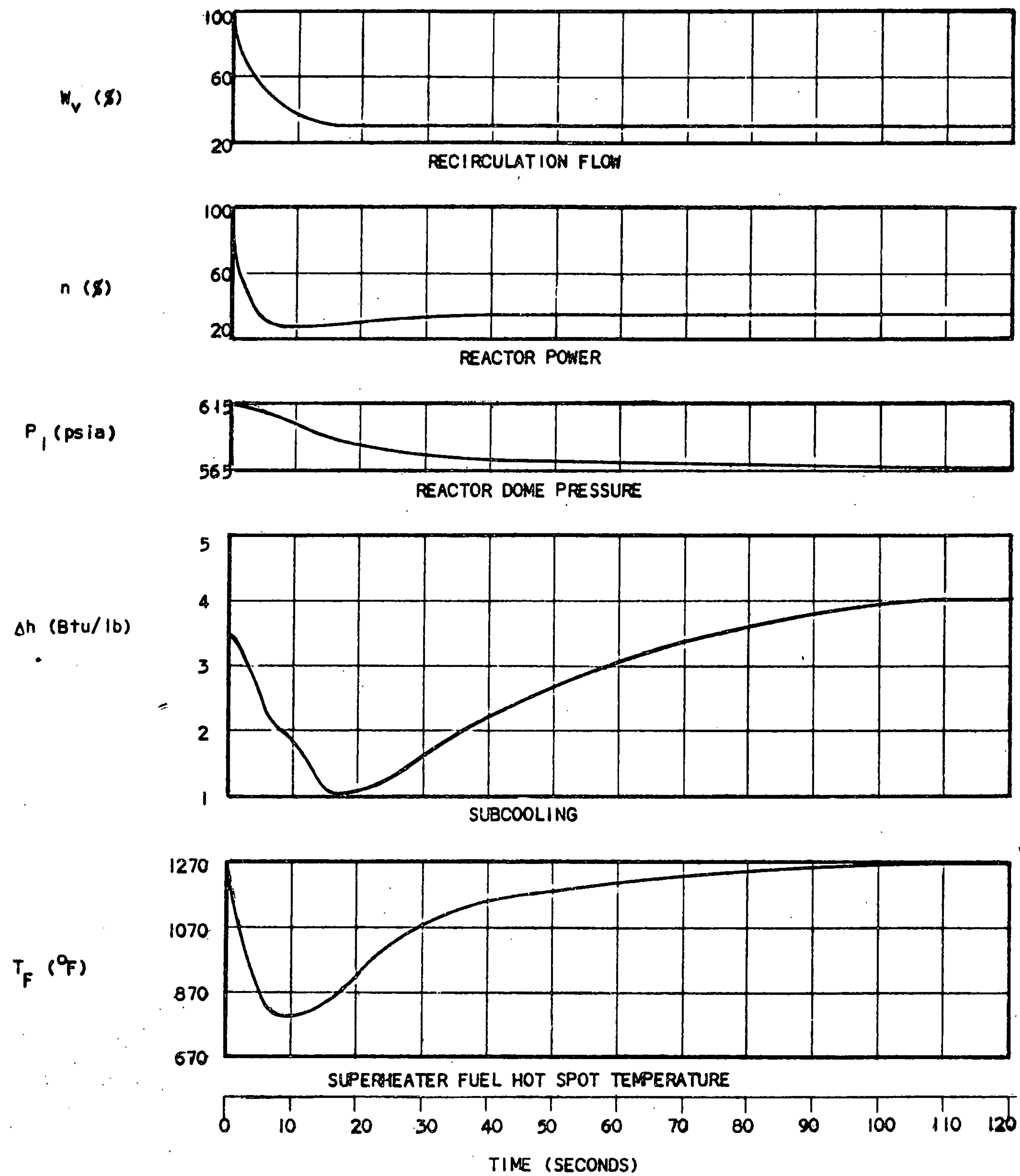
FIG. 433.8.7



Movement of Each Rod in Group II of 2 in.
Δk Rate Per Rod Is 3¢/sec

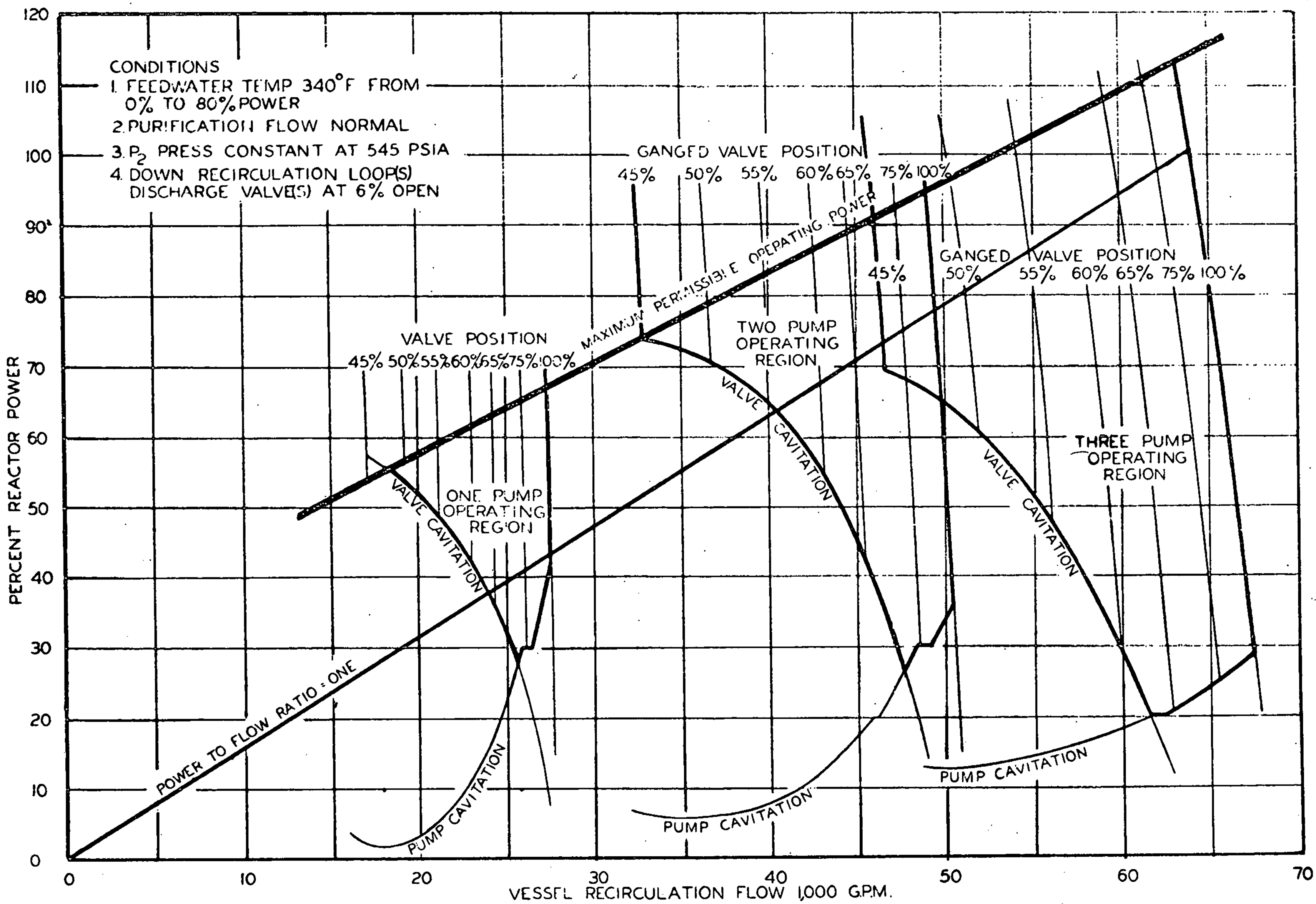
CONTROL ROD WITHDRAWN

FIG. 433.8.8



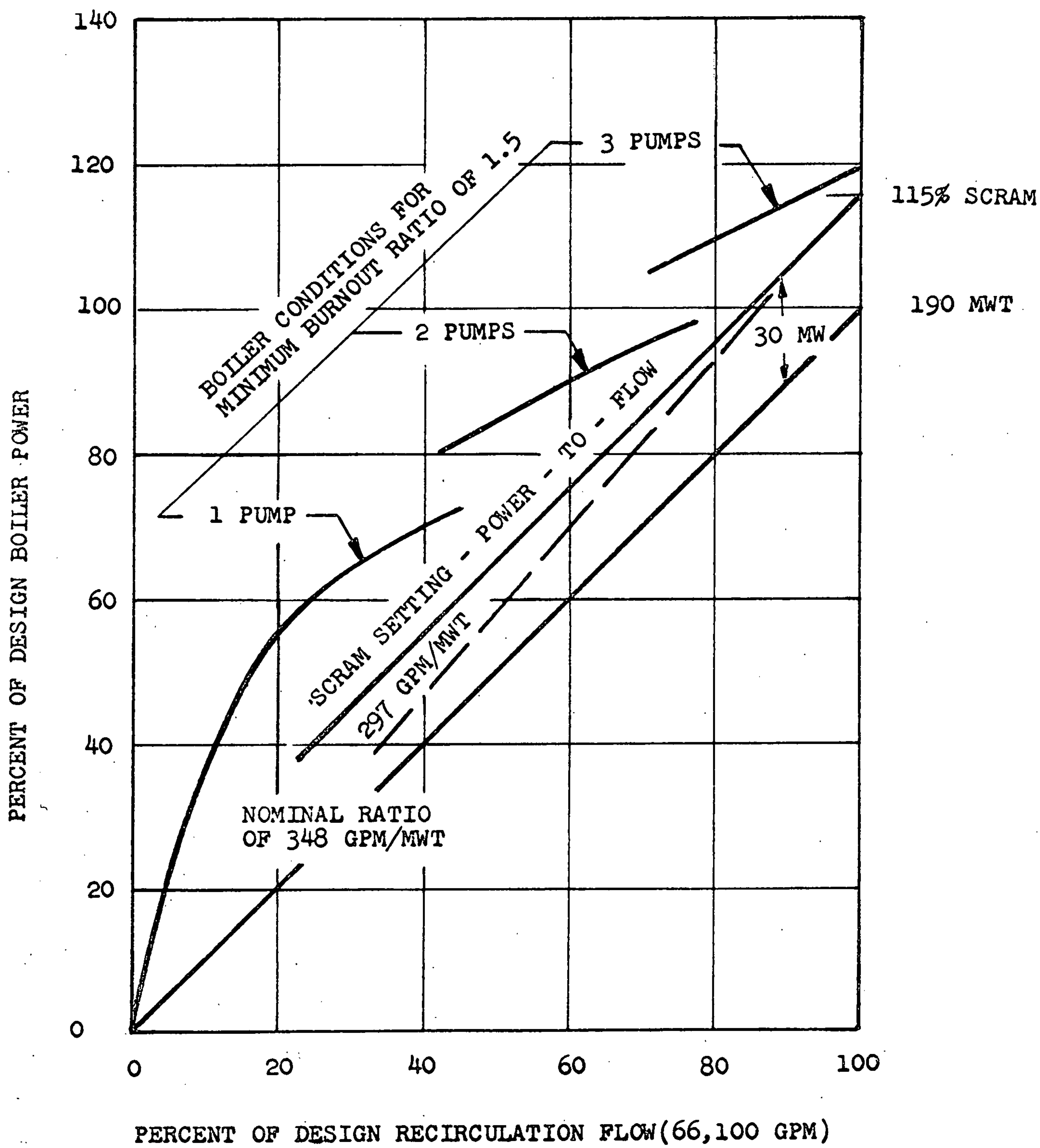
PATHFINDER RECIRCULATION-PUMP TRIPOUT; TWO PUMP TRIPOUT

FIG. 433.8.10



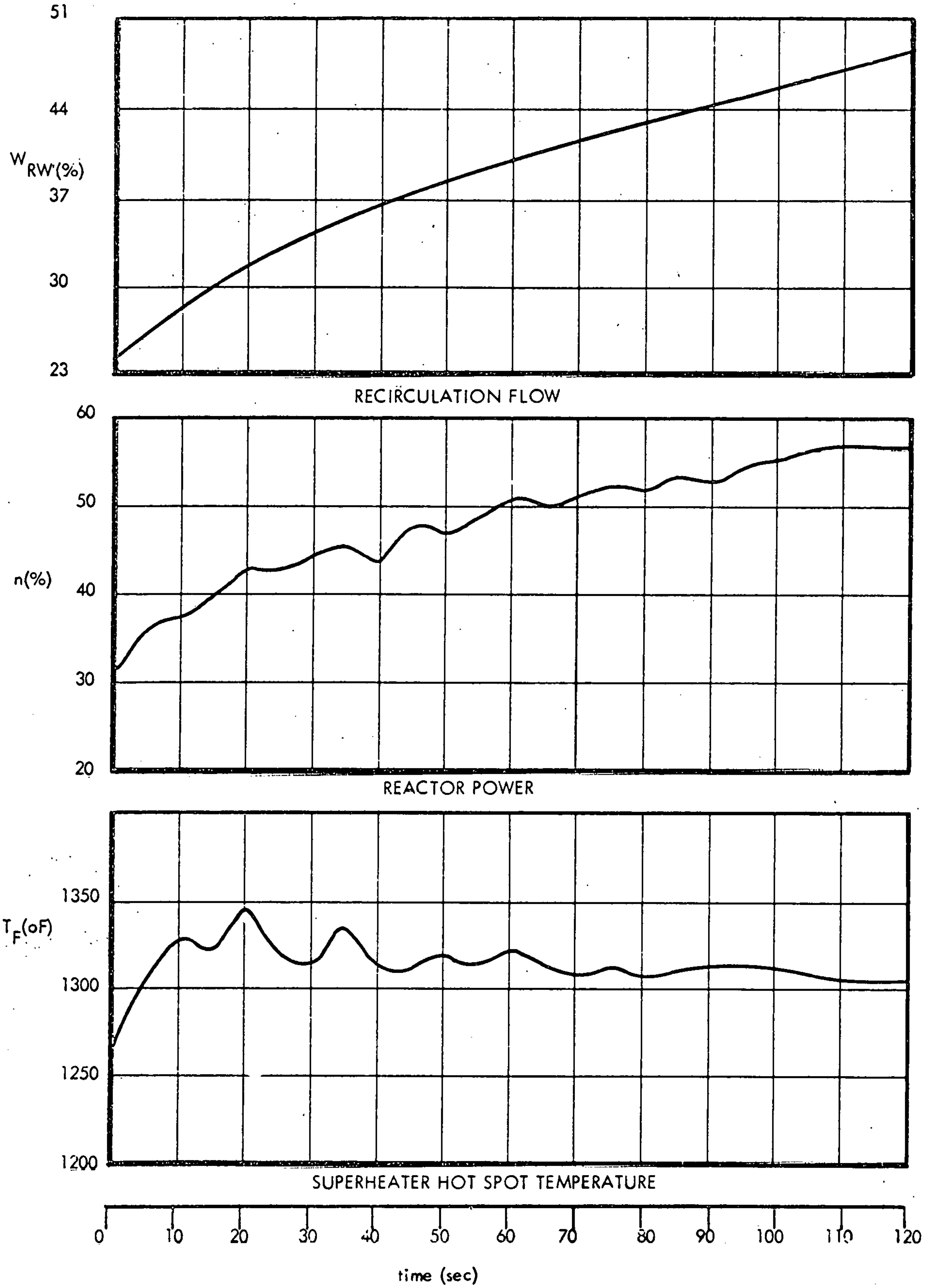
PATHFINDER ATOMIC POWER PLANT REACTOR POWER AND RECIRCULATING OPERATING DIAGRAM

FIG. 433.8.9



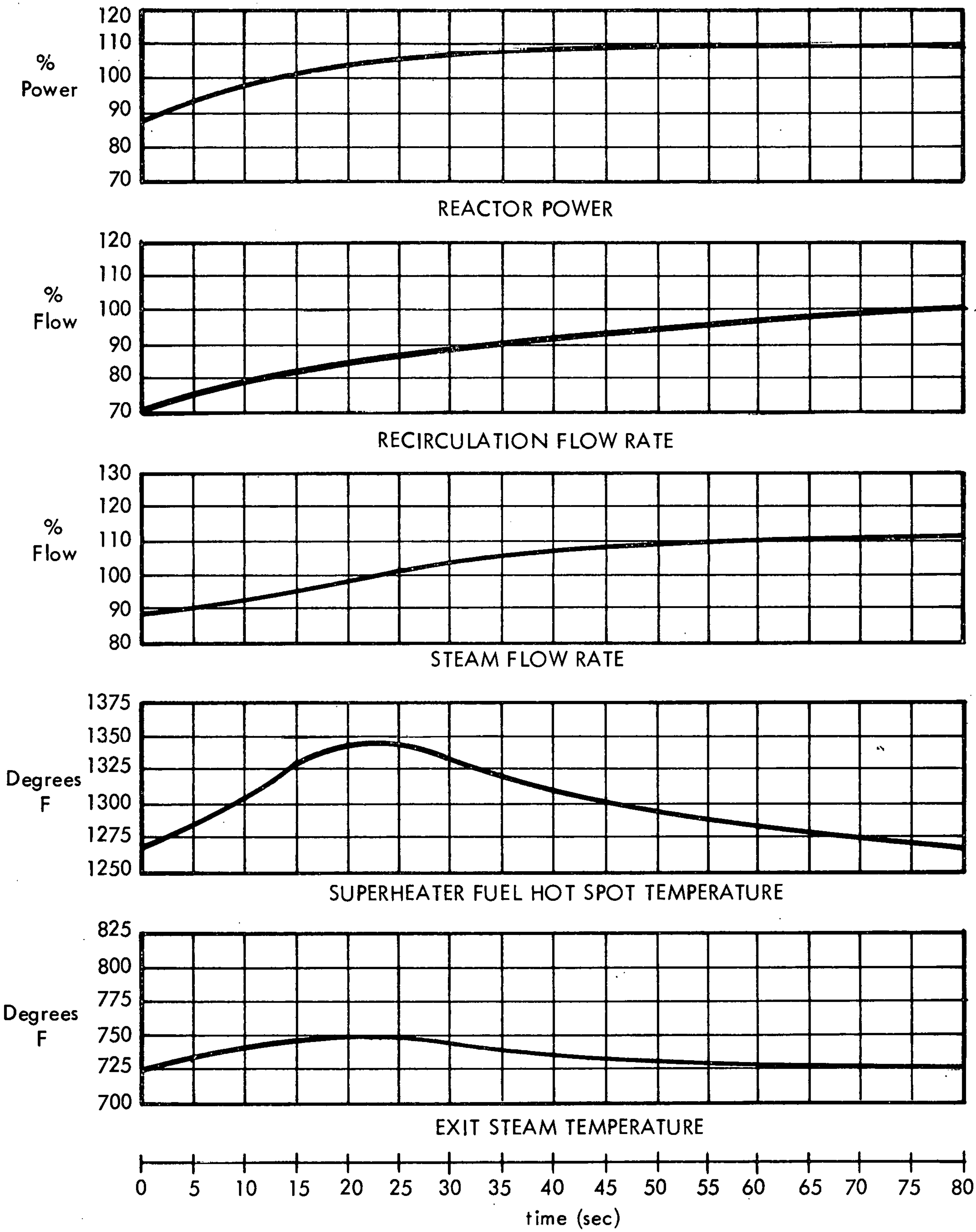
SCRAM SETTING FOR POWER-TO-RECIRCULATION FLOW CHANNEL
RELATIVE TO DESIGN APPROACH TO BURNOUT

FIG. 433.8.11



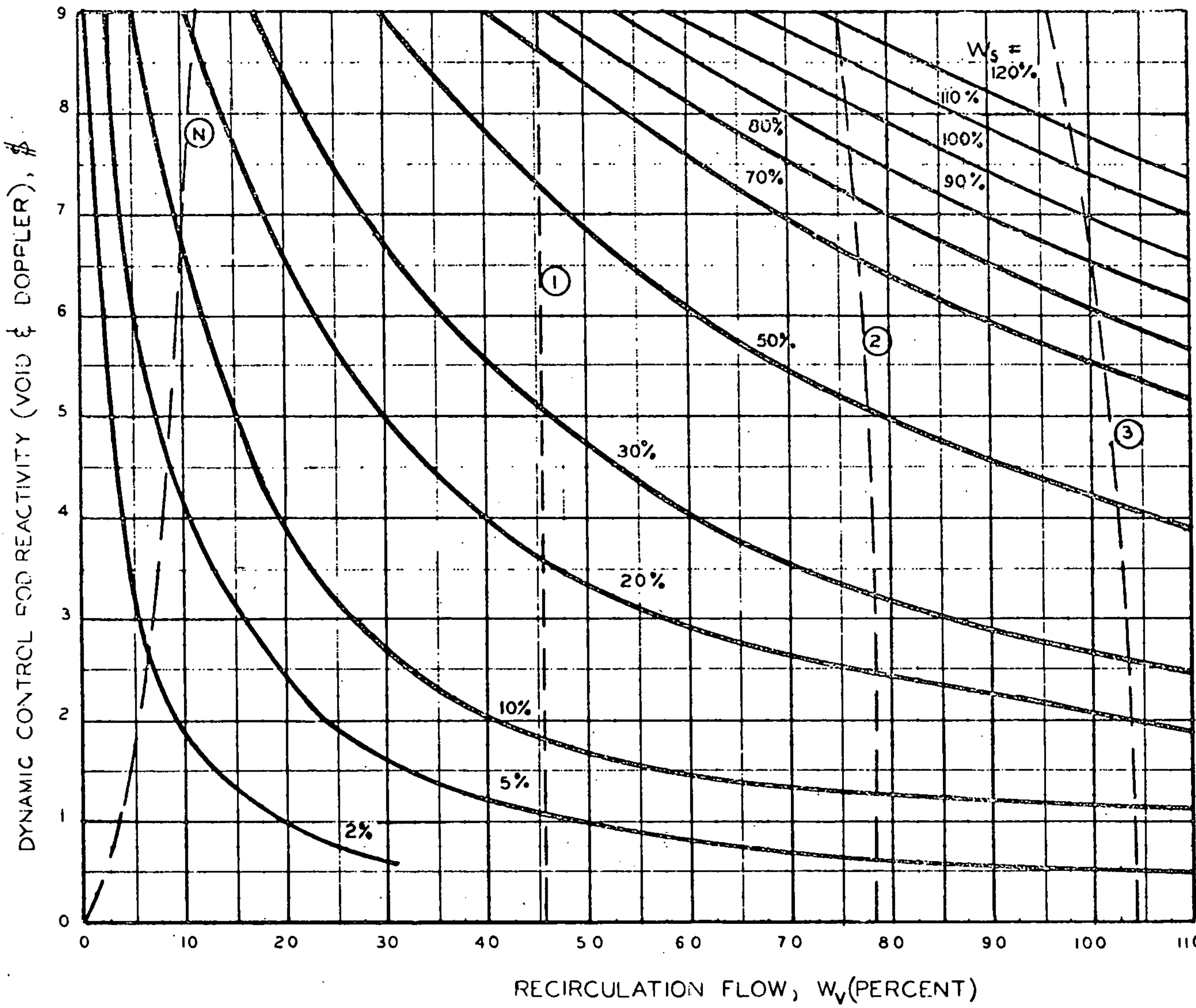
INCREASE IN RECIRCULATION FLOW RATE - ONE PUMP TO TWO PUMP OPERATION $(n)_{iv}$: 30.5%

FIG. 433.8.12



REACTIVITY INCREASE FROM GANGED OPERATION OF
THREE RECIRCULATION PUMP DISCHARGE VALVES
INITIAL POWER = 88% INITIAL RECIRCULATION FLOW = 71.5%

FIG. 433.8.13



ASSUMPTIONS:

1. SUPERHEATER EXIT PRESSURE = 545 PSIA (CONSTANT)
2. RATED SUPERHEATER PRESSURE DROP = 70 PSI
3. FEEDWATER TEMP CONSTANT AT 340°F BELOW 80% W_S
4. PURIFICATION FLOW PROP TO STEAM FLOW (100% W_P = 11.1 #/SEC.)
5. PUMP SEAL FLOW = 2.3 #/SEC
6. REACTOR WATER LEVEL CONSTANT
7. β, EFFECTIVE DELAYED NEUTRON FRACTION = .007
8. MODERATOR TEMPERATURE COEFFICIENT IS NEGLIGIBLE

CHARACTERISTICS:

(N) NATURAL CIRCULATION, VALVES WIDE OPEN
(1) ONE PUMP FORCED CIRC.
(2) TWO PUMP FORCED CIRC
(3) THREE PUMP FORCED CIRC.

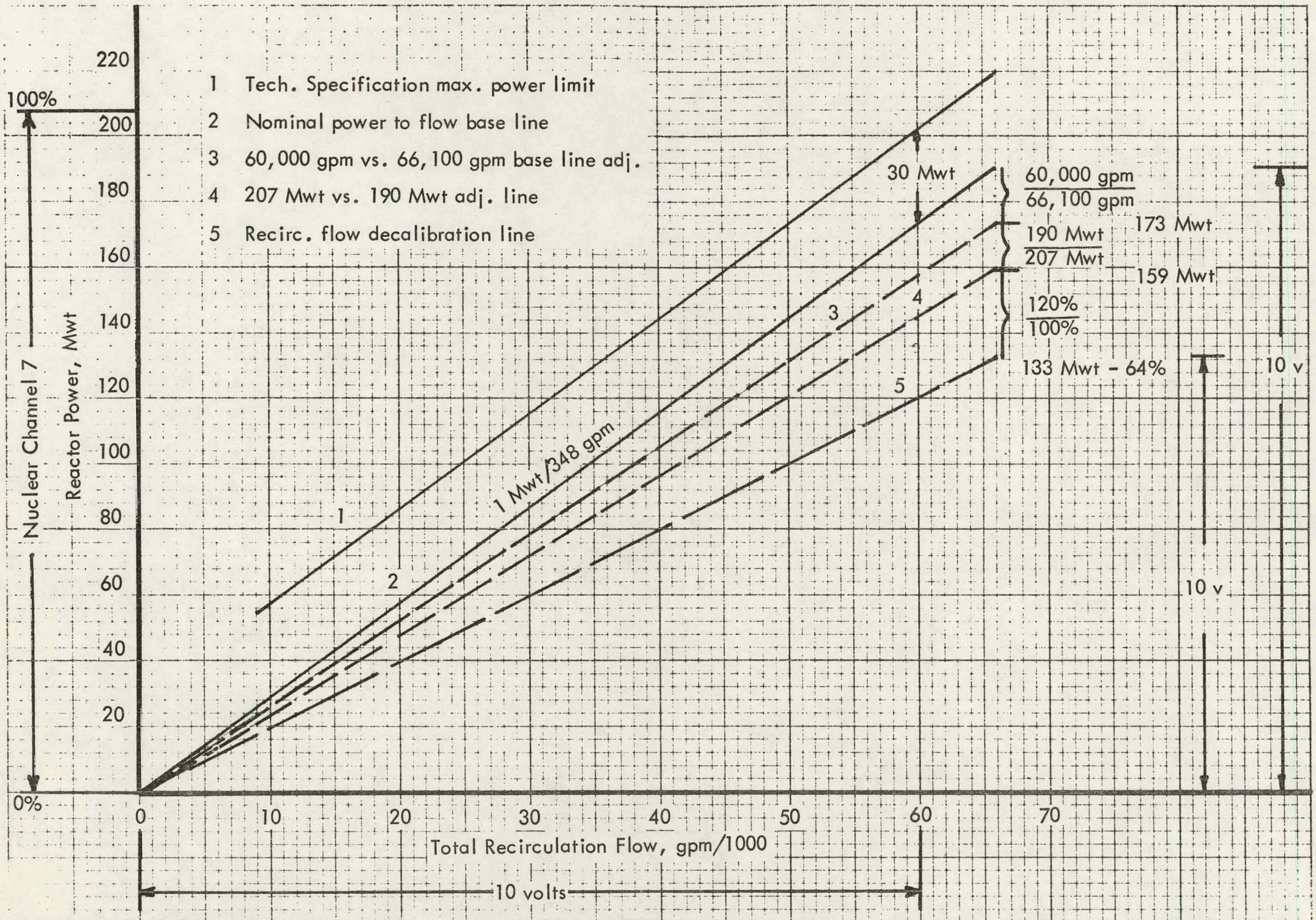
NOTE: RUNNING PUMP HAS DISCHARGE VALVE WIDE OPEN ($77\frac{1}{2}°$), DOWN PUMP HAS DISCHARGE VALVE AT 5°

RATED VALUES

100% W_S = 171.1 #/SEC.
100% W_V = 7250 #/SEC.
100% η = 1.806×10^5 BTU/SEC.

PATHFINDER BOILER OPERATION CONTROL MAP

FIG. 433.8.14



PATHFINDER ATOMIC POWER PLANT REACTOR POWER
TO RECIRCULATION FLOW SCRAM

FIG. 433.8.15