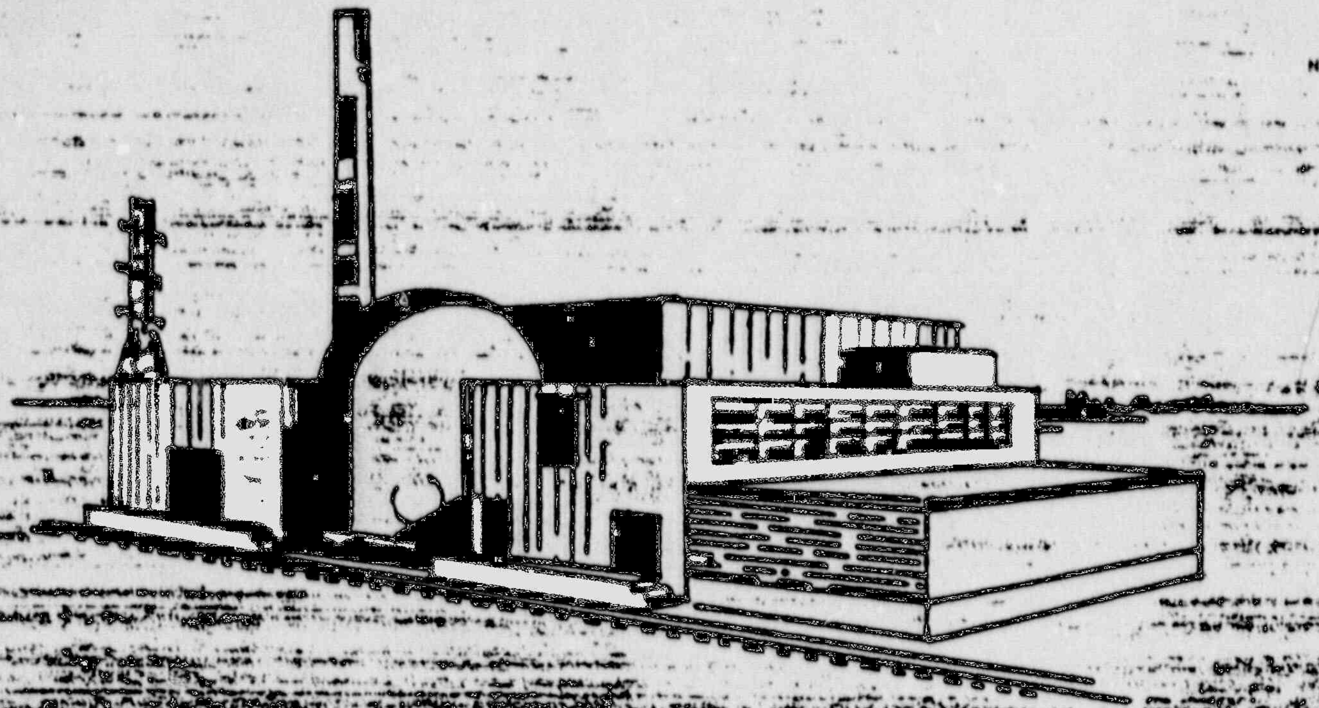
NSP 6603



Pathfinder Atomic Power Plant

INCORPORATING INTEGRAL NUCLEAR SUPERHEAT

SIOUX FALLS, SOUTH DAKOTA

REPORT TO UNITED STATES ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

LICENSE NO. DPR-11

SIX - MONTH PERIOD REPORT NO. 1

PLANT OPERATING EXPERIENCE

MAY 19, 1966 TO NOV. 19, 1966

NORTHERN STATES POWER COMPANY

PATHFINDER ATOMIC POWER PLANT

SIX MONTH REPORT

Initial Steam Flow Testing
through 40% Power Testing

Prepared By:

Northern States Power Company

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I. Introduction

Northern States Power Company's provisional operating license No. DPR-11 for the Pathfinder Atomic Power Plant requires that:

> "Within 30 days after the completion of six months of operation of the reactor (calculated from the date of completion of Phase II of the Power Operation Test Program), and at the end of each six-month period thereafter Northern States shall submit a written report to the Commission which summarizes the following:
>
> (a) Total number of hours of operation and total energy generated by the reactor;
>
> (b) Number of shutdowns of the reactor with a brief explanation of the cause of each shutdown;
>
> (c) Operating experience including levels of radioactivity in principal systems; routine releases, discharges, and shipments of radioactive materials; a description of tests performed in the reactor; and the results of any test analyses completed during the period including results of tests required by the Technical Specifications; a summary of experiments conducted; number of malfunctions in the control and safety systems with brief explanations of each; and a discussion of data obtained relating to superheater operation;
>
> (d) Principal maintenance performed and replacements made in the reactor and associated systems including a report on various tests performed on components of the reactor and associated systems;
>
> (e) A description of the leak tests performed pursuant to the Technical Specifications and the results of such tests including a description of any necessary corrective measures taken to meet the requirements of the Technical Specifications for assuring the specified containment leak tightness;
>
> (f) Significant changes made in operating procedures and in plant organization;
>
> (g) Radiation levels recorded at both on-site and off-site monitoring stations."

Section 50.59, 10 CFR 50, permits the holder of a license authorizing operation of a nuclear reactor to make changes in the facility, changes in procedures, and conduct tests or experiments, provided that the change, test or experiment does not involve a change in the technical specifications or an unreviewed safety question. The licensee is required to report such changes to the Commission.

This report is intended to summarize plant and reactor operations for the six month period commencing with the completion of Phase II testing, thereby satisfying the reporting requirements of the Provisional Operating License DPR-11 and Section 50.59 of 10 CFR 50.

II. General Summary

A. Introduction

The Pathfinder Plant achieved initial criticality on a fifteen element slab core on March 24, 1964. The full core loading, boiler and superheater, was completed in a deliberate manner and zero power testing on the 2.2 w/o reference core was completed late in the Fall of 1965. At that time in the testing program, additional excess reactivity was gained by loading thirty-two 3.2 w/o boiler elements in place of 2.2 w/o elements. Then, the reference core for power operation included 64 2.2 w/o elements, 32 3.2 w/o elements, and 40 poison shims in the boiler core and 409 highly enriched elements in the superheater core. Additional zero power, Phase I, tests were completed on this core early in 1966. The results of the Phase I testing were reported in the Phase I Report - NSP 6601 previously submitted to the AEC.

The Nuclear Instrumentation Calibration Test, the Superheater Radiative Cooling Test, and the Flooded Nuclear Warmup Test were completed at reactor power levels less than 8 MWT. These tests were completed in May of 1966 and the results were reported in the Phase II Report - NSP 6602 previously submitted to the AEC.

Phase III testing began with reactor testing up to 8 MWT with steam flow conditions. The test procedures used to complete the Phase III testing and to reach full power in a safe manner are:

1. Test 277.2A "Initial Steam Operations to 8 MWT."

2. Test 278.1A "Initial Reactor Operations to 40 MWT and Turbine Startup."

3. Test 278.2A "Reactor and Turbine-Generator Operation 20% to 100% Power (190 MWT)."

This report summarizes the reactor and plant testing beginning with the initial steam flow tests through the 40% (76 MWT) power testing phase.

B. Summary of Pathfinder Startup Procedure

To fully appreciate the startup and testing phases of the Pathfinder reactor, a short summary of the Pathfinder Startup Procedure is in order:

1. <u>Nuclear Warmup</u>

The flooded reactor vessel is heated up to approximately 400°F at nuclear power levels not exceeding 6 MWT. During the warmup, a minimum reverse flow of 150,000 lbs/hr is required through the superheater to prevent local boiling. The reactor is pressurized to keep the superheater water temperature at least 2[illegible]°F subcooled.

2. <u>Draining the Superheater</u>

The superheater is drained with the reactor in the shutdown condition. A normal boiler water level is established with the reactor at approximately 420°F with a saturation pressure of approximately 300 psig.

3. <u>Reactor Critical - Raise Power to 200 KWT</u>

The maximum allowable core power without steam flow with the superheater drained is 200 KWT. A power-to-flow safety system prevents a higher power level without steam flow.

4. <u>Establish 12,000 lbs/hr Bypass Steam Flow - Raise the Reactor Power to 2 MWT</u>

At 200 KWT, the steam flow is increased prior to raising the power to a level of 2 MWT.

5. <u>Establish 25,000 lbs/hr Bypass Steam Flow - Raise the Reactor Power to 6 MWT</u>

At 2 MWT, the steam flow is increased prior to raising the power to a corresponding level of 6 MWT.

6. <u>Establish 70,000 lbs/hr Steam Flow - Raise the Reactor Power to 17 MWT</u>

At 6 MWT, the steam flow is increased prior to raising the power to a corresponding level of 17 MWT.

Note: The power-to-flow safety system requires that the steam flows are established prior to switching the Intermediate Nuclear Channel (INC) ranges for a power rise. The power to flow ratio is limited to just slightly greater than one during the power rise to 17 MWT. If a range change occurs prior to establishing the required increase in steam flow, a scram will occur. Also, after a range change is made, if the steam flow drops below the steam flow setpoint, a scram will occur.

7. Open the Main Steam Isolation Valve

At a steam flow of approximately 70,000 lbs/hr, the main steam line low flow meters are on scale. The MSIV is opened and the power-to-flow protection system relies on a signal from the main steam line low flow meters.

8. Raise the Reactor Power to 40 MWT and Pressurize to 540 psig

The steam flow is raised to 90,000 lbs/hr to clear the flow interlock scheme and the reactor power is raised and maintained at a slightly greater than equilibrium condition to heat up the reactor (and pressurize) at a rate of approximately 1°F/minute. When the pressure reaches 540 psig, the pressure control system is put on automatic.

9. Turbine Startup at 20% Power

The turbine is put on the line at 20% power. The steam line pressure may be automatically controlled by the inlet valves; however, during the present testing some of the steam flow is bypassed to the condenser through the dump valve. Under these conditions, the dump valve automatically controls the steam line pressure.

10. Power Escalation

After the turbine is on the line, the power escalation is a routine matter. The superheater protection is afforded by the steam temperature safety system and the steam line pressure safety system. Also, a steam flow less than 80,000 lbs/hr will initiate a scram. The control rod positions determine the maximum allowable steam temperature.

C. Summary of Reactor Testing - Initial Steam Flow through 40% Power

Beginning with the initial rise to 8 MWT and for each subsequent power increase of not more than 20%, the rod positions, the superheater fuel temperatures, and the outlet steam temperatures were predicted. The various operating parameters were closely watched and whenever they were any value other than expected, the testing was halted until the differences were understood. All differences noted during the rise to 40% power were not of any safety significance.

Several of the predicted rod heights were in error because of the lack of knowledge of the rod worths at power. The total rod worths and the reactivity capabilities of the core appear to be known very accurately.

The superheater fuel temperatures were predicted conservatively high. Based on heat balance results, the superheater power fraction was slightly less than expected, and correspondingly, the outlet steam temperatures were slightly less than expected.

The superheater outlet steam temperature response to the position of the superheater rods was different than expected but the total temperature rise during the rod withdrawal was as expected. During the boiler rod interchange at 40% power, the superheater temperature response was as predicted.

At each 20% power step, a series of tests were repeated. These tests include:

1. Fluid Dynamics Effects

   The Dynamics Tests (433) check the dynamics effects of changes in feedwater temperature, feedwater flow, reactor pressure, recirculation flow, and control rod motion. The tests are performed by establishing steady state conditions and then introducing known changes and recording the effects of these changes on other critical parameters.

   These tests were all completed with satisfactory results except for the recirculation flow tests. The recirculation flow tests include the tripping of running recirculation pumps. Because of the unexpected large backflow through a tripped pump, the dynamics responses were not accurately predicted. It was also determined that the increase in recirculation flow, caused by the closing of the discharge valve of a tripped pump, may exceed the technical specification limit of 455 gpm/sec. A pump restart at power was never completed because of the same tech spec limit. Although the pump trip tests have been discontinued, the recirculation flow tests involving discharge valve motion have continued.

   A safety system has been added to initiate a reactor scram whenever a recirculation pump accidentally trips or the pump flow drops below a minimum value.

2. Reactor Shutdown Tests

   Reactor shutdown tests (431) were completed at each power level to show that the reactor and plant systems would respond in a safe manner to runback, scram, isolation scram, load dump, and turbine trips. All tests were not necessarily performed at each 20% power level.

   Early in the testing program, all runback signals, with the exception of the short period on the Log N channel and the high power levels on the power channels, were converted to initiate a controlled shutdown. During a controlled shutdown, all rods are run in until they are bottomed; whereas, during a runback, the rods will run in only until the initiating signal is cleared. Until the effects of various operating rod positions can be more thoroughly evaluated, this action will remain in effect.

Load dump tests have been completed up to 50% power without resulting in a reactor scram.

The reactor and plant systems responded in a safe manner to all reactor shutdowns, load dump tests, and turbine trips. None of these initiated tests resulted in any superheater fuel temperature increases.

3. Stability Evaluation

Several superheater elements are instrumented with thermocouples. The information obtained by virtue of these thermocouples has been invaluable during the power escalation program. If work were to be conducted in the boiler core region, the holddown mechanism would have to be removed from the reactor vessel. The removal of the holddown mechanism would seriously endanger the reliability or workability of the superheater thermocouple. To insert the oscillator rod into the boiler core at 40% power, as required, was therefore undesirable. Because of this, NSP requested that the insertion of the oscillator rod be made optional after 100% power is achieved. The AEC approved the Technical Specification Change No. 12, which granted this option.

In lieu of the oscillator rod testing, a noise analysis program was initiated. Noise analysis data have been obtained at 20%, 40%, and 60% power levels. The noise analysis gives no indication of unstable tendencies indicating that a substantial margin exists between the 60% power level and any unstable higher power level.

4. Xenon Reactivity

The xenon follow tests performed at 40% and 60% power levels verify the calculational predications.

The effect of the xenon buildup after shutdown appears to decrease the superheater power fraction on subsequent startups.

5. Radiation Testing

Radiation level data have been obtained at each power level step. A radiation field around the purification system piping in the Fuel Handling Building required that the piping be shielded with a lead filled jacket. The radiation levels around the turbine have been increasing substantially with power. This operational inconvenience has not yet been resolved. All other measured radiation levels are in general agreement with predictions.

6. Water Level Calibration and Steam Dryer Efficiency

These tests have been completed at each 20% power level step. The water level measurements have been as predicted and the steam quality downstream of the steam dryers is below the maximum design value of .5%.

In summary, the behavior of the reactor at power up to 40% has been thoroughly tested and evaluated with no basic control or safety system problems encountered. There has been absolutely no indication of boiler or superheater fuel element failures.

D. Summary of Plant System Difficulties

The major plant equipment has operated in an acceptable manner during the startup program, but several difficulties have been encountered with some plant systems which have caused delay in the testing program. Although these problems may be reported in more detail in other sections of this report, a brief summary of each problem is stated.

1. Pressure Control System Hardware

The pressure control system is designed to control the steam line pressure automatically with either the turbine inlet valves or the dump valve to the condenser. Several problems have occurred with the hardware of the dump valve and its hydraulic positioning system; however, these problems have been resolved. The pneumatic positioning equipment for the inlet valves has been shown to be inadequate and automatic pressure control operations using the inlet valves has been discontinued except for very limited testing.

All turbine operations are now conducted with a low steam flow to the condenser with the dump valve automatically controlling the steam line pressure.

The plant was in a shutdown condition for approximately three weeks in November while a hazards analysis review was completed u the dual valve mode of pressure control operation. The review verified the previous hazards analysis; namely, that the high pressure protection and steam temperature protection would limit the transient fuel temperature to less than 1650°F. A "loss of steam flow" scram protection system was temporarily added for backup protection during the testing program. This system will initiate a scram if the steam flow suddenly drops by more than one-third of the existing flow.

Control system hardware for the inlet valves is on order and the system modifications will be completed at a later date.

2. Steam Flow Meters

There are three sets of steam flow meters on the bypass and main steam lines. Each set of flowmeters, which are required at different ranges of steam flow, have presented operational problems. The bypass steam flow meters were made operational by moving the condensate pots to a level in line with the top of the flow nozzle; however, all efforts to eliminate the problems with the main steam line flow meters have failed. The flow meters indicate the flow accurately enough but they are very sensitive and occasionally "bounce", particularly during reactor startups. The "bounce" in the indication is often enough to reach the low flow scram setpoints and a scram results. All other reactor parameters do not change when the "bounce" is detected, and in most instances, the "bounce" is seen only on a single flowmeter. This problem remains to be solved.

3. Main Steam Line Isolation Valve Leakage

The leakage history of the MSIV has compelled us to measure the valve leakage whenever practical. During the initial steam flow testing phases, the MSIV leakage was excessive on occasions when measured in the shutdown condition because of weld slag particles which apparently fell on the seat of the valve plug. The valve is located in a vertical section of the steam line. After the valve seat was cleaned, the leakage was a minimum detectable value.

More recent leakage measurements tend to show that the valve seating is proper and the leakage remains at a tolerable level. An additional Isolation Valve will be ordered and will be placed in the steam line if future leakage measurements show that the present valve is unacceptable.

4. Off-Gas System Hydrogen Concentrations

During initial steam flow operations, possible explosive hydrogen concentrations ($\sim$ 4%) were approached in the off-gas system because of non-effective recombiner operation. Off-gas system modifications to pre-heat the off-gas were effective in reducing the moisture content of the off-gas and the recombiner efficiency was increased. The recombiner problem has been eliminated but an off-gas system moisture problem still exists in the downstream off-gas system piping.

5. Faulty "Water in the Steam Line" Scram System

Several reactor scrams were caused by the magnetrol which transmits the "water in the steam line" signal. A reorientation of the instrument apparently eliminates a float vibration problem which previously existed.

6. Control Rod Drives

Maintenance was required on two control rod drives which developed poor dashpot action because of faulty dashpot piston rings. The piston rings were replaced in the two drives and operational testing is periodically conducted to detect any further failures of a similar nature.

One control rod drive occasionally exhibits a slow insertion time when gang lowered with other rods. The drive remains operational although the source of this difficulty has not been determined.

In general, the operation of the control rod drives has been very satisfactory considering their operational history and requirements.

A combination of the operational difficulties encountered during startups of the Pathfinder superheat reactor and equipment shortcomings during operation have resulted in a large number of reactor shutdowns. These shutdowns have been of continual concern to the management of Northern States Power Co. and Allis-Chalmers Co, and the safety committees for Pathfinder operations. As the testing program progresses, the operational shutdowns have become much less common and the equipment failures are being eliminated. See Appendix A.

E. Chronological History of Testing and Operating Data Summary

A chronological history of the testing schedule for the six month reporting period is listed:

May 20th	Started the initial steam operation to 8 MW Thermal. Completed the testing on May 26th.
May 25 and 26th	Safety Committee Meeting
May 26th	Plant shutdown for miscellaneous maintenance projects.
June 15th	Plant Startup. Started the initial reactor operation to 40 MWT (20% power).
June 26th	Plant shutdown for steam line valve leakage measurements. Preparations for turbine operations also planned.
July 14 and 15th	Safety Committee Meeting
July 22nd	Plant startup. Power testing at 20% was continued.
July 25th	Initial Turbine-Generator operation.
July 30th	Plant shutdown for miscellaneous maintenance.
August 6th	Plant startup. Power testing at 20% was completed on August 7th.
August 8th	Power testing at 40% begun.
August 30th	Plant shutdown for investigation of pressure control system problems, control rod drive maintenance, and reactor source replacement.
September 7 and 8th	Safety Committee Meeting
October 7th	Plant startup. Power testing at 40% continued.
October 27 and 28th	Safety Committee Meeting
November 8th	Plant shutdown. Hazards analysis of the pressure control system begun.
November 29th	Plant restart. Power testing at 40% to be completed and 60% power testing to be started.

The operating data for the six month period is:

	MWD	Hrs Critical
May 19 - June 30	146.07	250.64
July	81.35	97.40
August	996.26	463.50
September	0.00	0.00
October	799.58	364.00
November 19th	67.05	46.73
TOTAL	2090.31	1222.27

This is equivalent to approximately 300 MWD/T burnup for the boiler fuel.

III. Testing Results - Initial Steam Flow to 40% Power

The significant testing results which pertain to reactor physics, reactor dynamics, shutdown tests, and radiation surveys from the initial steam flow testing through the 40% power testing are reported in this section.

The power escalation testing at Pathfinder will be completed early in 1967 and an ACNP report will be prepared by Allis-Chalmers personnel. The information reported herein may be useful for extracting general information and conclusions; however, it must be recognized that information gained from the power escalation to 100% power may result in corrections to the preliminary reported data.

A. Reactor Physics Measurements

The following information has been extracted from A-C test reports submitted to the Pathfinder Safety Committee.

1. Initial Steam Flow Testing to 8 MWT (Test 277.2A)

For this test the superheater was drained. The control rod configuration was Groups I and II full in, Groups IV and V full out, and Group III controlling. The Group III critical height at 420 F was 24.3 inches.

Reactor power was increased to 6 MWT and after approximately 16 hours of operation the Group III critical height was 27.4 inches with a reactor temperature of 418 F. Correcting to 420 F the Group III height would be 27.6 inches. Thus, 3.3 inches of Group III movement or 0.4% Δk were required to bring the reactor to the above described conditions. For the 0.4% Δk, the reactivity balance related to Group III movement is estimated as follows:

TEST 277.2A REACTIVITY BALANCE

UO_2 temperature above 420F (0.024% Δk) = 0.2 in.

Xenon for 16 hours (0.15% Δk) = 1.3 in.

Therefore, voids required approximately 1.8 inches of Group III movement or 0.22% Δk. For the conditions of Test 277.2A the core average voids were reported as 2.5%. Thus, a void coefficient of 0.09% Δk per percent core average voids can be inferred from this data. Calculations predict a void coefficient of 0.1% Δk per percent of core average voids for these conditions.

2. Reactor Testing to 40 MWT (20% Power) (Test 278.1A)

The reference core for this test was clean with 64 2.2 w/o U-235 and 32 3.2 w/o U-235 boiler elements, 40 B-SS shims, and all superheater fuel elements. The boiler reference core loading and control rod identification are shown in Figure 1. The superheater loading is shown in Figure 2.

All measurements were done with a moderator temperature of approximately 420 F. For those measurements not at 420 F, critical rod heights are corrected to 420 F using the measured temperature coefficient reported in ACNP-65600. This coefficient at 420 F is -1.6×10^{-4} $\Delta k/^{o}F$ for the voided superheater.

a. Power Increase to 6 MWT

At zero power and a moderator temperature of 420 F, the control rod configuration was Groups I and II full in, Groups IV and V full out, and Group III controlling at 24.5 inches. Reactor power was increased to 6 MWT and after approximately 16 hours of operation the Group III critical height was 27.5 inches with a reactor temperature of 420 F. Thus, 3.0 inches of Group III movement or 0.36% Δk were required to bring the reactor to the above described conditions.

Test 277 results differ slightly from these. For 277 the reactivity loss for the same step was reported as 0.40% Δk as compared to the 0.36% Δk reported here. However, in view of the varied xenon history, this difference is within experimental accuracy.

At 6 MWT copper wires were exposed in each of three superheater ion chamber channels. These data were normalized to the average counts for each wire and are plotted on Figure 3. The ion chamber channel locations are shown on Figure 2 (L-9 is at the superheater center). All three wires show the same general axial shape with a peak to average of approximately 1.52. The calculated axial peak to average for the same reactor condition is 1.55.

b. Power Increase to 40 MWT

Power was increased stepwise to 18, 28, 33, and 40 MWT by withdrawing Group III keeping pressure approximately constant at 300 psia - 420 F coolant temperature. On Figure 4 are plotted the Group III heights versus core power. All values are corrected to 420 F using a temperature coefficient of -1.6×10^{-4} Δk/in and zero power measurements of Group III differential worth versus Group III height. The solid line on Figure 4 through the dots below all data points represents the estimated xenon free Group III heights. These estimates should be fairly accurate since the time at each power level represented by the low Group III points was only a few hours. For example four hours of operation at 20 and

40 MWT yield expected xenon worth of approximately 0.2 to 0.4% Δk. Thus an error of 25% in the xenon correction affects the Δk in voids by less than 10%.

Based on Figure 4 the following reactivity balance is given for the clean core at 6, 18, and 40 MWT and at 300 psia.

TEST 278.1A REACTIVITY BALANCE

6 MWT

Parameter	% Δk
UO_2 above 420 F	0.02
Voids	0.10
Total	0.12

Power coefficient (% Δk/MW) at 6 MWT = 1.9×10^{-2}

18 MWT

UO_2 above 420 F	0.07
Voids	0.30
Total	0.37

Power coefficient (% Δk/MW) at 18 MWT = 2.6×10^{-2}

40 MWT

UO_2 above 420 F	0.16
Voids	1.15
Total	1.31

Power coefficient (% Δk/MW) at 40 MWT = 3.5×10^{-2}

For 40 MWT and 300 psia, the expected void reactivity defect was approximately 2.0%. This is based on a boiler core average void of 14% of coolant volume. For a void defect of 1.15% Δk, these same calculations would require a core average void of 9.5% of coolant volume. Thus at 40 MWT and 300 psia, the voids are lower than expected and/or the void defect calculations yield higher values than measured.

c. Core Pressurization

Pressurization from 300 to 550 psia began with Group III at 42.3 inches and the core power at 40 MWT. Initially the Group III rods were fixed. At 430 psi the power level as indicated by nuclear instrumentation had decreased to approximately 33 MWT. Group III rods were then withdrawn to keep power indication constant to 550 psia.

On a second approach in pressurization nuclear instrumentation was kept constant by withdrawal of Group III rods. This required Group III movement from 40.8 inches at 300 psia to 46.7 inches at 550 psia.

It had been expected that with core pressurization, core power would decrease. This results since the temperature coefficient effect from 420 to 475 F inserts more negative reactivity than the positive reactivity inserted by the decrease in voids as the pressure is increased from 300 to 550 psia.

Group III withdrawal from 40.8 to 46.7 inches is worth approximately 0.37% Δk. Xenon addition during pressurization is estimated to be 0.15% Δk. Thus, the net reactivity loss due to increasing core temperature with a decrease in voids is estimated to be 0.2% Δk.

The heat balance taken at full pressure gave inconclusive results. Power levels from 34 to 40 MWT were calculated depending on which flow meter reading is accepted. During pressurization the water temperature increased from 420 to 475 F. This is expected to result in a decrease in neutron attenuation through the reflector by a factor of 1.39. However, during pressurization the Group III rods moved out and voids in the core decreased. Both of these effects reduce current indication for a fixed power level and taken together result in a 1.25 decrease in current. Thus, it is estimated that with constant current during pressurization, the core power level at full pressure was

$$40 \times \frac{1.25}{1.39} = 36 \text{ MWT}$$

d. In-Core Ion Chamber Data

In core ion chambers were monitored continuously during 278.1A. The output from these chambers was used to provide qualitative information on core response to power level changes. As examples of this, the N-1 string chamber currents in milliamps are listed for five power levels at 300 psia and one at 550 psia.

TEST 278.1A

N-1 ION CHAMBER RESPONSE

Chamber	Actual Power Level (MWT)					
	6	18	28	33	40	[illegible]*
7 (T)	.0154	.032	.043	.046	.0525	.068
[illegible] (M)	.0242	.052	.090	.105	.135	.135
9 (L)	.0218	.049	.088	.11	.132	.097
Avg	.0205	.044	.074	.087	.107	.100
Power (MWT)	---	13	30	34	41	37

*550 psia

	Normalized to Average at Each Level					
7	0.75	0.73	0.58	0.53	0.49	0.68
8	1.18	1.18	1.22	1.21	1.26	1.35
9	1.06	1.11	1.19	1.26	1.23	.97

The power levels across the top are from heat balance calculations. Those across the bottom use the percentage increase in average current from the N-1 string and the heat balance values at each level to estimate the power at the next higher level. These extrapolations agree with the heat balance values to within 10% with exception of the first step. These data also indicate that with pressurization, the power decreased from 40 to 37 MWT.

The letters T, M, L refer to chambers at approximately 54, 36, and 18 inches from the bottom of the fuel in channel N-1. The individual currents divided by the string average current are listed in the lower half of the table. As core power is increased, the power distribution shifts into the lower half of the core. However, when the pressure is increased power shifts back into the upper half of the core.

3. Reactor Testing to 76 MWT (40% Power) (Test 278.2A Partial)

Measured results for control rod positions, in-core ion chamber readings, and flux wire counts are given for power escalation to 90 MWT, for superheater rods (Group I) withdrawal, and for boiler rods interchange (Group I with IV and V). Results are also given for a xenon follow (Test 335) after a shutdown from 76 MWT.

The experimental results are used to calibrate control rods. The reactivity worths of xenon, voids, and Doppler are deduced and compared with calculational results. Measured power and void coefficients are estimated. Measured flux peaks and the flux peak trends with control rod motion are compared with calculated values. The measured superheater power fractions are compared with calculated values.

Predictions of core physics behavior from 40% to 100% power are reviewed. A recommendation is made for an abbreviated xenon test to be done at 60% power.

a. Summary of Operation and Experimental Results

1. Power Increase to 60 MWT

Core power was increased to 40 MWT at 1000 CDT on 8/5/66. At this time the Group III rods were at 42 inches and the reactor water temperature was 478°F. The power level was held at approximately 40 MW for 28 hours until 1400 on 8/6/66. During this period Group III rods were withdrawn to 56 inches to compensate for xenon buildup.

Cu wires were activated with Group III at 47 inches. In-core ion chamber currents were recorded as a function of Group III withdrawal. Of special interest is the IIC string located in superheater channel N-1. The power level in this channel is close to that in Z-1 (the hot channel). The average of the current reading for IIC's No. 7, 8, and 9 located at 54, 36, and 18 in respectively, from the bottom of the fuel is a measure of the superheater hot channel power and of the core power. Ratio of individual IIC readings to the string average indicate axial power shifts as rods are withdrawn and power is increased.

At 1030 CDT on 8/8/66 the core power level was 38 MWT with Group III at 51.6 in. By 2200 the core power level was 60 MWT with Group III at 73 inches.

2. Power Increase to 80 MWT and Withdrawal of Superheater Rods

At 0930 on 8/9/66 the core power level was 59 MWT with Group III at 73 inches. The superheater power fraction was 0.064. Superheater rods were withdrawn in eight steps until full out. At 0030 on 8/10/66 Group I was at 73 inches, the core power level was 86 MWT, and the superheater power fraction was 0.115.

Withdrawal of the Group I rods increased core power by 27 MWT including Xe effects. With Xe corrected out, the Group I rods are worth 30 MWT.

3. Boiler Control Rod Interchange

At 1100 on 8/15/66 the core power level was 76 MWT and the superheater power fraction was 0.118 with Group II at 14 inches. The reactor water temperature was 478 F and xenon was at equilibrium.

After several reactor shutdowns, the interchange between boiler control rod Group II and Groups IV and V began at 2230 on 8/16/66. The core power level, and rod group positions are summarized in Table II.

TABLE I

BOILER CONTROL ROD INTERCHANGE

Time	Power (MWT)	Rod Group Heights (in) II	Rod Group Heights (in) IV and V
2230	65	0	73
2315	60	0	60
2400	78	14	60
0030 (8/17)	64	14	53
0930	64	14	53
1030	72.5	16.5	53
1100	56	16.5	45
1145	69	22	45
1215	60	22	40
1300	81	27	40
1340	55	27	30
1430	80	36	30
1445	45	36	24
1530	73	56	24
1600	51	56	18
1845	63	73	18
1900	81	73	24

At the start of the interchange with Group II at zero, the superheater power fraction was 0.111. At the end with Groups IV and V at 18 in., the power fraction was 0.140.

4. Test 335 - Xenon Follow

Immediately after the rod interchange, the rods were returned to the normal startup configuration (Groups III, IV and V at 73 inches, Group II full in, and Group I controlling) with core power at 76 MWT. This power was held from approximately 2200 on August 17 to 0700 on August 18. Group I reached 73 inches by 0700. At this later time the core was shutdown. The core was then returned to critical at 0924 with Groups I and II full in, Group III controlling, and Groups IV and V at 73 inches. The reactor water was held at 450 ± 5F and criticality maintained until 1400 on August 20. Group III was controlling during this period of approximately 55 hours, and Group III differential worths were measured for approximately every inch of rod motion.

b. Analysis of Results

Results from Test 335 - Xenon Follow - are discussed and analyzed first. This is done since these results provide a differential and integral worth curve for Group III control rod positions from approximately 25 to 55 inches. Next the core reactivity balance to 80 percent power is presented. Then results from the IIC's and wire exposures are analyzed. Finally, expected rod positions are revised using results from 278.2A at 40 percent power.

1. Test 335 - 40 Percent Power

Using period data taken during the xenon follow test, a Group III differential worth curve was constructed as shown on Figure 5. The dashed curve above 58 inches is estimated.

The integral of this curve is also shown on Figure 5.

Calculations were done to exactly match the power history just prior to the reactor shutdown from 76 MWT to perform the xenon test. The calculated worth of xenon is shown on Figure 6. Also on Figure 6 is shown the measured worth of xenon.

Since the calculated and measured xenon worth curves are in good agreement from 3 to 60 hours, the calculated worth was taken to be the measured value at shutdown. This assumption gives a Group III height of 50 inches at shutdown for zero power and 450 F. It is also the same height as at 17.8 hours after shutdown. This time agrees well with calculations which predict the xenon worth at 17 hours after shutdown to be the same as at shutdown. This point is called the xenon return time.

Table II summarizes results of this test.

TABLE II

XENON FOLLOW AT 76 MWT TEST SUMMARY

	Xenon Worth (% Δk) Calculated	Measured	Calculated for Equil. at 76 MWT
At Shutdown	2.03	2.03	2.28
Maximum	2.70	2.75	3.10

	Times (hrs) Calculated	Measured
Maximum	7.0	7.8
Return	17.0	17.8

Calculated results in Table II also yield the worth of Group II control rod from zero to 14 inches to be 0.25% Δk since at equilibrium xenon for 76 MWT the Group II position is 14 inches.

As an additional check of the xenon calculations and the Group III rod worth shown on Figure 5, calculations were done to follow the xenon buildup for 28 hours at 40 MWT. These calculations predict a xenon worth of 1.56%. This compares with a Group III worth of 1.60% Δk for movement from 42.0 to 56.2 inches --- in good agreement.

The xenon free Group III height at 450F is estimated to be 31.3 inches. A clean core Group III height at 450F is estimated to be 28.0 inches. This estimate uses a 420F clean core Group III height of 24.5 inches and a temperature coefficient of -1.55×10^{-4} $\Delta k/^{o}F$. The worth of Group III from 28.0 to 31.3 inches is 0.41% Δk. The calculated worths of samarium at shutdown and at 60 hours are 0.31% Δk and 0.34 Δk. Thus, 0.07% Δk is attributed to fuel burnup and long term fission product burnup. At this time the fuel exposure in the boiler core is estimated to be 762 MWD.

2. Reference Core Reactivity

Prior to Test 278.1A, prediction had been made of expected control rod positions for 20 percent power and equilibrium xenon. These positions were Groups I, III, IV and V control rods at 73 inches and Group II control rods full in. As a result of this test it appeared that the reactivity worth of voids was being over calculated and estimates were redone for Test 278.2A. From results at 40 percent power it appears that the newer predictions are in excellent agreement with measurements. However, this may be somewhat fortuitous.

A reactivity balance is done for 76 MWT. For this it is assumed that the calculated xenon worth and UO_2 heating reactivity worths are accurate.

76 MWT

<u>478 F Clean Group III at 31.7 in</u>

	<u>Measured</u>	<u>Calculated</u>
Voids and Doppler ($-1.95\%\ \Delta k$)	---	III at 50"
Equilibrium Xenon ($-2.28\%\ \Delta k$)	II at 14"	II at 11"

The reactivity difference between 11 and 14 inches on Group II is calculated to be worth $0.15\%\ \Delta k$. At the time this measurement was made it is estimated that $0.35\%\ \Delta k$ in fission products other than xenon had built up, whereas the calculations had assumed no fission product buildup. [illegible], the core reactivity loss of 76 MWT appears to [illegible]. 0. [illegible] Δk less than predicted.

The void worth to 76 MWT thus appears to be accurate. Of the $1.95\%\ \Delta k$ for voids and Doppler, $1.7\%\ \Delta k$ is in voids. This is based on 69 MWT in the boiler. At 550 psia and full flow, the average and exit voids are calculated to be 14.5 percent and 24 percent. These values yield a void coefficient of $0.12\%\ \Delta k/\%\ \alpha$ from zero to 76 MWT.

A power coefficient can also be determined from measured data by noting the power increase resulting from Group I and Group II withdrawal. Withdrawal of Group I control rods from 0 to 73 inches increased power by 30 MWT and Group II rod withdrawal from 0 to 14 inches increased power by 18 MWT. The deduced measured worth of these rods is $0.95\%\ \Delta k$. The calculated worth is $1.15\%\ \Delta k$. Thus, a power coefficient is estimated between 2.0 and $2.4 \times 10^{-4}\ \Delta k/MW$.

A reactivity balance for the core (just prior to escalation to 60 percent power) is given in Table III.

TABLE III

CORE REACTIVITY BALANCE

Condition	k
Cold Clean	1.098
478F	
Clean	1.083
Sm	1.079
Fuel Burnup	1.078
76 MWT	
Voids plus Doppler	1.058
Equilibrium Xenon	1.036

Critical with II at 16 inches

Based on results to date, predictions are made of the Group II height for escalation of core power to 100%. These are listed in Table IV. Expected positions are given for the equilibrium xenon and xenon free conditions. For the equilibrium position an uncertainty is included. The uncertainty is equivalent to approximately 30% of the Δk in xenon and voids beyond that at 76 MW.

TABLE IV

EXPECTED ROD HEIGHTS (IN.)

60% to 100% Power

MWT	Xenon Free	Equilibrium Xenon Min	Equilibrium Xenon Expected	Equilibrium Xenon Max
76	III at 56		II at 16	
114	III at 67	23	26	29
152	II at 0	27	32	38
190	II at 10	32	40	58

The difference between these expected positions and those shown in Test Procedure 278.2A is that the Group II integral worth curve has been adjusted to reflect the lower than calculated rod worth deduced from experiments to date for the lower half of the core. The total Group worth was not adjusted.

3. Power Distribution

Table V lists calculated and measured axial peak to average values for each wire exposure. The agreement is fair.

TABLE V

AXIAL FLUX PEAKS

Case	Meas.	Cal.
III at 47 in.		
N-1	1.74	1.58*
L-9	1.74	1.60*
II at 0 in.		
S-5	1.31	1.67
L-9	1.28	1.46
IV and V at 24 in.		
S-5	1.62	1.48**

*III at 43 in.
**IV and V at 16 in.

Calculations for other control rod configurations show that at low void levels as rods are withdrawn the superheater axial peak increases. So it is expected with III control rods at 47 in. and Groups IV and V at 24 in., the calculated peak values would be greater than those listed in Table VI.

Calculated values to compare with the IIC readings are not presented herein. However, the intent is to use the axial power shapes computed as a fraction of control rod withdrawal to obtain flux values at 18, 36, and 54, in. at locations N-1 and L-9.

The superheater power fraction increased from 0.064 to 0.115 as Group I control rods were withdrawn. Then during the rod interchange it increased to 0.140. These values compare with calculations of 0.096 to 0.116 and 0.150. Thus, with superheater rods in, the power fraction was substantially underestimated. With superheater rods out, the agreement is good.

B. Reactor Dynamics Testing (Test 433)

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 is not severe and is well within the limits of the reactor protection systems;

(3) to demonstrate that all control systems (pressure, feedwater level, 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 to predict the response at a higher power level; and

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

The following information has been extracted from A-C test reports submitted to the Pathfinder Safety Committee.

1. Reactor Testing at 38 MWT (20% Power)

a. Response to Pressure Setpoint Changes and Control Rod Movement

The following table shows a comparison between measured and calculated parameters for the pressure setpoint and control rod movement changes. These calculations were done assuming a 15% superheater-to-boiler power fraction and associated flux peaking, whereas the estimated power split at the time of the experiment was about 5%. The flux peaking at this condition has been shown to be equivalent to a power split of 7.5%. The power coefficient of reactivity for the calculations was about $2.3 \times 10^{-2}\% \, \Delta k/MW$ and was measured to be about $1.5 \times 10^{-2}\% \, \Delta k/MW$ at 30 MW and 540 psi.

The effect of a smaller power split would be to lower transient temperature peaks, where as the effect of a smaller power coefficient of reactivity would be to increase transient temperature peaks, for a given fixed disturbance. Thus, using the power peaking associated with the 7.5% power split a correction factor to be applied to the calculated fuel temperature results, is found to be

$$\frac{7.5}{15.0} \times \frac{2.3}{1.5} = 0.765.$$

The application of the power coefficient part of the correction factor directly to the fuel temperature is not strictly correct due to the non-linear character of the pressure control system; however, it is more realistic to include this term than to omit it.

The other important factor to be considered in evaluating these results is the nature of the disturbances. In the calculations all step disturbances were true steps whereas "step disturbances" during the test were actually ramp changes of about 2 seconds duration. (This was as fast as the operator could safely make exactly a +5 psi setpoint change.) These slower disturbances would have the effect of reducing transient power peaks and transient temperature peaks and could well account for the remaining "discrepancy" between measured and calculated results.

The response of the reactor system to control rod motion was less than anticipated since CRG III was moved a maximum of +0.9 inches, whereas the calculated results were based in a CRG II motion of 2.0 inches.

TRANSIENT PARAMETER VALUES

Initial Reactor Power: 38 MW; Pressure: 540 psi; superheater thermocouple No. 0-10 = 675F

Controlling with GP III rods; Bulk steam: 543F; feedwater temperature = 390F

Parameter Measured	Test Results	Calculated Results (see report text for conditions)
1.) Pressure Set Point Change +5 psi with T_2 control		
a. Max ΔT_F t/c No. 0-10	+34F	+75F
b. Max Δ power, incore No. 7	+13% of existing power	----
c. Max Δ power, out-core 5	+8.2% of existing power	+14% of existing power
d. Max ΔP_2	+5.5 psi	+5.5 psi
e. Max ΔT_2	+8F	+25F
2.) CRG III Motion, 0.9" out with T_2 control		
a. Max ΔT_F t/c 0-10	+15F	+42F for 2" of CRG II Motion
b. Max Δ power, in-core No. 7	+14.5% of existing power	----
c. Max Δ power, out-core 5	+14.5% of existing power	+37% of existing power for CRG II motion
d. Max ΔP_2	+1 psi	----
e. Max ΔT_2	+15F	+42F for CRG II motion

b. Response to Feedwater Flow Rate Change at 38 MWT

The feedwater flow rate was first reduced, causing reactor water level to fall, and was then increased to prevent a low level scram. The results are listed.

1. Disturbance: -105,000 lb/hr in 11.5 seconds
   MAX Power Change: -9% of existing power on in-core ion chamber No. 9
   MAX ΔT_F -28F on t/c No. 0-10
   MAX ΔP_2 -1.7 psi

2. Disturbance: +150,000 lb/hr in two separate disturbance of 5.5 seconds and 12.0 seconds each
   MAX Power Change: +14.6% of existing power on in-core chamber No. 9
   MAX ΔT_F: +45.5F on t/c No. 0-10
   MA[illegible]$_2$: +9F
   MAX [illegible]$_2$: +2 psi

3. Expected Results
   Disturbance: +124,000 lb/hr in 4 seconds
   MAX Power Change: +32.5% of existing power
   MAX ΔT_F: +125F

Comments:

In this run, the superheater rods were at 0" so that the power peaking associated with this condition is representative of a 7.5% superheater-boiler power split. Thus, in the second disturbance the peak superheater temperatures are considerably below the calculated peak disturbance. Also, since the actual power peak is well below the calculated power peak the effect of changing the feedwater flow rate apparently did not have as large an effect on subcooling as expected - or the subcooling-reactivity relationship is less than expected. As Test 433 progresses these effects will be analyzed and an attempt will be made to obtain quantitative relationships.

The reactor responses to this disturbance in the subcooling is damped and shows no tendency toward oscillatory behavior.

c. Response to Recirculation Flow Change at 34.5 MWT with the Dump Valve on Auto Control

Initial Plant Conditions

Reactor Power:	34.5 MWT (The flow reduction started at 45.0 MWT and ended at this power when ganged valves were at 60% open - all three pumps running.)
Reactor Pressure:	54[illegible] psia
Superheater Fuel Temperature:	692F on t/c No. 0-10

Steam Exit Temperature:	554 F
Steam Flow Rate:	124,500 lb/hr
Feedwater Temperature:	367 F
Recirculation Flow Rate:	48,000 gpm
Pressure Control System:	Auto on Dump Valve
Feedwater Temperature:	Hand
Reactor Water Level:	Auto

1. Disturbance:	Recirculation Valves ganged open from 60% to 100% (+14,000 gpm in 124 sec)
MAX Power Change:	+24.3% on in-core chamber No. 8
MAX ΔT_F:	+13 F on t/c No. 0-10
MAX ΔP_2:	+2 psi
MAX ΔT_2:	+6 F

2. Expected Results:	(calculated)
Disturbance:	+14,000 gpm in 130 seconds at 24% power
MAX Power Change:	+20%
MAX ΔT_F:	+18 F
MAX ΔP_2:	+2 psi
MAX ΔT_2:	+6 F

Comments:

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 same initial conditions. Analog computer results show that this rate of recirculation flow change (255 gpm/sec) is worth approximately 0.7 cents/second in excess reactivity rate. Again, no tendency towards instability is evident.

d. Test 433 at 38 MWT with Turbine Inlet Valves on Auto

The turbine inlet valves were in far from an optimized state; thus, the results of planned disturbances in this plant condition is not presented. When the inlet valve response is improved several of these tests will be performed to demonstrate satisfactory plant response with the turbine operating.

2. Reactor Testing at 76 MWT (without turbine) while on AUTO Dump Valve Control

a. Reactor Response to Control Rod Motion

The reactor response to withdrawal of the superheater control rods (Group I) from 47" to 49" was recorded. The chart showed very little change in any of the recorded variables because the superheater rods were worth very little reactivity in this position. Other rod maneuvers, especially gang lowering of the superheater rods several inches while at this power level, show a more marked response - but a response that is smooth and which shows no tendency toward reactor system instability.

b. <u>Reactor Response to Pressure Setpoint Changes</u>

This disturbance was repeated six times to yield information on the response of the various in-core ion chambers. The disturbance was ± 3 psi rather than the planned ± 5 psi because of a scram that occurred due to high exit steam temperature; this scram may have been due to operator error since the operator was "following" the exit steam temperature transient with the floating scram setpoint. The ± 5 psi transient would have been preferrable to analyze, but because of the safety restrictions placed on Test 433, the setpoint disturbance was lowered.

<u>Initial Plant Conditions</u>

Reactor Power:	76 MWT
Recirculation Flow:	60,000 gpm
Group I rods:	53.6 inches
Exit steam temperature:	626 F
Superheater Fuel Temp. on t/c No. 0-10	690 F (67.4 MV)
Steam Flow Rate:	275,000 lb/hr
Reactor Pressure:	548 psig
Pressure Control:	Dump Valve on Auto
Feedwater Temperature Control:	Manual
Reactor Level:	Auto

The transient results are listed.

1. <u>Disturbance</u>: -5 psi in 6 seconds
   - MAX Power Change: -10.4% of existing power on in-core No. 9
   - MAX ΔT_F: -34.6 F on t/c No. 0-10
   - MAX ΔT_2: -18 F
   - MAX ΔP_2: -6 psi
   - MAX ΔW_s: +20,000 lb/hr

2. <u>Disturbance</u>: -3 psi in 2.2 seconds
   - MAX Power Change: -5.4% on in-core chamber No. 9
   - MAX ΔT_F: -17.3 F on t/c No. 0-10
   - MAX ΔT_2: -10 F
   - MAX ΔW_s: +10,000 lb/hr

3. <u>Disturbance</u>: +3 psi in 2.8 seconds
   - MAX Power Change: +6.3% of existing power in in-core No. 9
   - MAX ΔT_F: +21.6 F on t/c No. 0-10
   - MAX ΔT_2: +9 F
   - MAX ΔW_s: -12,000 lb/hr
   - Damping: T_F reaches steady state in 55 seconds

4. <u>Expected Results</u>: (calculated on analog computer)

Disturbance:	+2.8 psi in 3.5 seconds at 42.1% power
MAX Power Change:	+3.1% of existing power
MAX ΔT_F:	+25 F
MAX ΔT_2:	+9 F
Damping:	T_F reaches steady state in 50 seconds

<u>Comments</u>:

The +5 psi disturbance results are shown in order to compare the 76 MWT results to the previously reported 38 MWT results. While the 38 MWT results were for a +5 psi change, a comparison shows that changes in t/c No. 0-10 response at 30 MW and 76 MW were identical. Power peaks were also quite comparable, but the change in exit steam temperature was only 8F at 38 MW, while ΔT_2 at 76 MW was 18F. This difference can be attributed to the fact that superheater Group I rods were inserted at 38 MW and were withdrawn to about 53 inches for the 76 MW runs. Overall, the responses at the two power levels are quite comparable, with no tendency of reactor system instability being in evidence.

The comparison of the +3 psi setpoint change to the analog computer calculation shows very favorable results with the superheater fuel temperature, exit steam temperature, and return of the system to a steady state giving the best comparisons.

c. <u>Response to Feedwater Flow Rate Changes</u>

1. <u>Disturbance</u>: -100,000 lb/hr in 4 seconds

MAX Power Change:	-6.8% of existing power on in-core ion chamber No. 9
MAX ΔT_F:	-19.5 F on t/c No. 0-10
MAX ΔT_2:	-8 F
MAX ΔP_2:	-2 psi

2. <u>Disturbance</u>: +125,000 lb/hr in 10 seconds

MAX Power Change:	+8.2% of existing power on in-core ion chamber No. 9
MAX ΔT_F:	+12 F on t/c No. 0-10
MAX ΔT_2:	+6 F
MAX ΔP_2:	+2 psi

3. <u>Expected Results</u>:

Disturbance:	+124,000 lb/hr in 4 seconds
MAX Power Change:	+12% of existing power
MAX ΔT_F:	+60 F

<u>Comments</u>:

In this run the transient response is less than expected, as was the case for the same disturbance at 38 MWT. While sufficient time and evidence has not been available for quantitative explanations of the differences, it is expected that lower measured response is due to operating with a lower value of subcooling than the calculations assumed. For example, when feedwater temperature is 390 F, subcooling is about 0.8 BTU/lb, whereas when feedwater temperature is 340 F, the subcooling is 3.6 BTU/lb. Thus, in the actual test condition only about 15 cents reactivity is tied up in subcooling (at 0.8 BUT/lb), whereas at 3.6 BTU/lb about 75 cents is tied up in subcooling. It was noted that the ratio $\frac{\text{Peak power at 40\%}}{\text{Peak power at 20\%}}$ measured is about the same as the ratio

$\frac{\text{Peak power at 40\%}}{\text{Peak power at 20\%}}$ calculated. This ratio is approximately 0.5.

d. Response to Feedwater Temperature Change

1. Disturbance: Feedwater temperature setpoint changed 43 lbs, equivalent to 16.8 F in 16 seconds

MAX Power Change:	+3.8% of critical power on in-core chamber No. 7
MAX ΔT_F:	+16.2 F
MAX ΔT_2:	+6 F
MAX ΔP_2:	+2.4 psi

2. Expected Results:

MAX Power Change:	+4.0% of existing power
MAX ΔT_F:	+18 F
MAX ΔT_2:	+6.5 F

Comments:

Although the test results and the calculated results compare favorably as far as disturbance magnitudes are concerned, the transient parameter shapes are quite different in the two cases. The calculated transients are very smooth and have no "peaks" or "troughs". Final steady-state values also compare favorably. The discrepancy between calculated and expected results undoubtedly is caused by the gross simulation of the feedwater temperature control system. A simple first order lag and clamping circuit were used to represent an obviously more complex system. The simulation was done like this in order to conserve computing hardware, and because the system dynamic response has been unknown.

e. Response to Raduced Recirculation Flow Change

This test was not as specified in the procedure for Test 433; the final ganged valve position was changed from 45% open to a final position of 60% 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% open appeared to be adequate, the procedure was changed accordingly. A heat balance at 100% valve open position yields 73.3 MWT and at 60% valve position, a heat balance gave 60.8 MWT.

1. <u>Disturbance</u>: +1874 gpm on Pump No. 13 in 20 seconds
   MAX Power Change: +15.9% of existing power on in-core chamber No. 7
   MAX ΔT_F: +34.6 F on t/c No. 0-10
   MAX ΔT_2: +14.5 F
   MAX ΔP_2: +3.2 psi

2. <u>Expected Results</u>:

   No direct comparison available

3. <u>Results at 38 MWT</u>
   <u>Disturbance</u>: +1781 gpm on Pump No. 12 in 14 seconds
   MAX Power Change: +[illegible].0% of existing power on in-core chamber No. 9
   MAX ΔT_F: +32.4 F on t/c No. 0-10
   MAX ΔT_2: +7 F

<u>Comments</u>:

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. <u>Reactor Testing at 76 MWT (with turbine) while on AUTO Dump Valve Control with the Inlet Valves on Hand</u>

During the week of November 28, 1966, the runs of Test 433 at 76 MWT with the dump valve automatically controlling pressure and the inlet valves on Hand were completed. These tests, called the "split flow" tests, were done at 76 MWT to provide a base with which future higher power test results can be compared. In all these cases the inlet valve position was held fixed by the mechanical load limit control.

Prior to these tests, analog computer studies were performed that compared system transient response when the dump valve was on auto with the split flow arrangement as was actually used during the testing. These computer studies predicted that changes in system parameters would be nearly the same whether the dump valve was on auto passing all of the generated steam or whether the split flow arrangement was used with the dump valve on auto and passing about 60,000 lb/hr. Disturbance studies were pressure setpoint changes, feedwater temperature changes, control rod motion, and recirculation flow changes.

This memo presents the initial conditions and transient conditions during the planned disturbances as well as comparisons of previous 76 MWT tests done with the dump valve on auto. The conclusions drawn as a result of these comparisons are those that were presented to the Operations Committee after the tests and prior to the 60% escalation step. These conclusions are:

1) The system response to the Test 433 disturbances is the same whether the dump valve handles all of the generated steam flow or whether the split flow arrangement is used.

2) No apparent safety hazard exists because of the split flow mode of operation.

4. Recirculation Pump Trip Tests

Thus far three attempts have been made to trip a recirculation pump with the reactor producing power. The first attempt on 10/18/66 ended in a scram due to high steam temperature approximately 21 seconds after the pump was tripped. In this case steam temperature was approximately 40F 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 held on 10/19/66 with the low steam temperature out of range runback bypassed and the high steam temperature scram set at +25 F above the initial steam temperature. A scram occurred because the high steam temperature setpoint was reached.

The third attempt to trip a recirculation pump was made on November 7, 1966, and was carried through to completion. Prior to the pump trip, recirculation flow was reduced by ganging all three discharge valves from 100% to 60% open. Initial conditions taken from a heat balance just prior to pump trip are listed in Table I.

TABLE I

Initial Conditions for One Pump Trip 11/7/66

Reactor Power:	58.1 MWT
Group I rods controlling at:	33.5 inches
Channel 5 current:	0.63×10^{-6}a
Channel 6 current:	0.58×10^{-6}a
Channel 7:	28%
Reactor Pressure:	543 psig
Feedwater Temperature:	380°F
Feedwater Flow Rate:	210,000 lb/hr
Steam Flow Rate:	210,000 lb/hr

Recirculation Flow Rates:	15,800, 17,000 17,000 gpm on Inst 168
Exit Steam Temperature:	580oF
Superheater Fuel Temperature T/C No. 0-10	727°F
Dump Valve Position:	18%
Pressure Control:	Auto on Dump Valve
Feedwater Temperature Control:	Auto
Level Control:	Auto on single element

Table 2 lists the important transient parameter values during the pump trip.

TABLE 2

Results of One Recirculation Pump Trip at 58.1 MWT

1. Disturbance: Tripped Pump No. 11

MAX Power Change:	-56% of initial power on in-core chamber No. 3
Final Steady-state power change:	74.9% of initial power with valves of two operating pumps at 60%
Final flow, valves at 6%, 60%, 60%:	38,000 gpm
Max Positive Rate of flow change:	+416 gpm/sec measured on pump No. 11 flow trace
Max changes in fuel temperature:	-152 F at 16 seconds and +115 F at 74 seconds
Max changes in steam line temperature:	-34 F at 16.5 seconds and +27 at 75 seconds
Max change in Steam Flow Rate:	-74,000 lb/hr at 51 seconds
Max change in Steam Line Pressure:	-14.8 psi

2. Calculated Results on Analog Computer, Case 2, Exp 66-27.

Disturbance:	Tripped one recirculation pump
Max Power Change:	-50% of initial power
Final steady-state power:	77% of initial power (with valves operating at 100%)
Final flow, valves at 6%, 100%, 100%:	40,000 gpm
Max positive rate of flow change:	+432 gpm/sec
Max change in steam line temperature thermocouple:	-130 F at 21 seconds +80 F at 65 seconds (superheater rods all out)
Feedwater control system:	Auto

Max change in hot spot fuel temperature:	-420F at 15 seconds +300F at 55 seconds
Pressure Control system:	Auto on Dump Valve Reset approx. 2.5 repeats/minute, No T_2 control

This test was successful in showing the reactor response to a pump trip at approximately 40% power. As predicted, the reactor safely returned to an equilibrium condition. While all the transient parameters did not correspond exactly with predicted magnitudes and time relationships, most parameter values were in excellent agreement with prediction.

It is interesting to compare the response of thermocouple No. 0-10 to the computed hot spot temperature. Since the thermocouple does not measure hot spot temperature a correction must be made to its values to obtain hot spot temperatures. Table 3 shows a comparison between calculated and measured superheater fuel temperatures.

TABLE 3

Comparison between Calculated and Measured Maximum Transient Temperatures during a One Pump Trip at 60 MWT

	Max Hot Spot Temperature change with uncertainties	Max Hot Spot Temperature with uncertainties	Max Expected Hot Spot without uncertainties
Calculated (analog)	+300F	1570F	-----
Measured, based on a +115F peak in t/c - No. 0-10	+265F	1275F	1080F

From Table 3 it can be seen that the changes in hot spot fuel temperature predicted by the analog calculations for the pump trip are quite close to those that must actually occur at the hot spot during the pump trip. The initial fuel hot spot temperature for the analog calculation was 1270F, whereas in the reactor the initial hot spot temperature (including uncertainties) was 1010F.

The results of shutdown recirculation flow tests indicated that recirculation pump trips and restarts may result in rate of flow increases exceeding the Technical Specification limit of 455 gpm/sec. During a pump trip, the reverse flow through the tripped pump is great enough so that as the pump discharge valve closes, the rate of flow increase may approach 455 gpm/sec. However, the rate of flow increase for a pump trip test can be controlled by the initial discharge valve position and pump trips with an initial valve position of 60% are definitely within the tech spec limits.

On November 3rd, the Operations Committee decided that a pump trip or a loss of indicated recirculation flow should result in a reactor scram. The necessary circuitry changes were completed prior to the recirculation pump trip test of November 7th. During this trip test, the discharge valve of the three pumps were closed to a 60% position, and the scrams, associated with the pump that was tripped, were bypassed. The tripped pump was not restarted at power.

Considerable concern was expressed about the boiler hydrodynamics and burnout margins for operations involving reduced recirculation flow rates during pump trip tests. This matter was thoroughly reviewed by A-C personnel and reported to the Safety Committee. From the A-C analysis memo 'Burnout Margin during Reduced Recirculation Flow during Test 433 Testing" dated December 9, 1966, from L L Kintner to R W Klecker the analysis is summarized.

"In summary, the boiler hydrodynamics and burnout margins have been adequately analyzed for all anticipated operations, including pump trips and operation at reduced flow. The analysis, based on test data, shows that the design calculations reported in the Technical Specifications (reproduced in Figure 7) are conservative. The amount of conservatism more than offsets the slightly reduced recirculation flow (57,500 gpm at 60% power) compared with the design full power recirculation flow (64,000 gpm) so that the full power burnout margin will be greater than 1.9. The power-to-flow scram circuit is set to account for power decalibration and backflow during pump trips so that Burnout Ratios will not exceed the values indicated in Figure 7."

Further recirculation pump trip tests and restart tests have been deferred until after 100% power testing is completed. It is anticipated that future test results will justify the removal of the scram protection on the recirculation pump trips and loss of indicated flow. The recirculation flow-to-power protection remains in effect and will not be removed.

5. <u>Pressure Control System Failure and Loss of Steam Flow Analyses</u>

The following information was extracted from an A-C report "Pressure Control System Failure and Loss of Steam Flow Analyses" dated November 21, 1966, prepared by J T Stone, D H Swanson, and L L Kintner.

"On November 8, 1966, Pathfinder was shutdown at the request of Allis-Chalmers.

During the October 26-28, 1966 Safety Committee meeting A-C presented an alternate method of continuing the power escalation program. This alternate was required because of difficulties experienced in control of the turbine inlet valves with automatic pressure control system. The alternate approach consisted of splitting the steam flow from the reactor with the majority of the flow passing to the turbine; the remainder being dumped directly to the condenser. The inlet turbine valves would be on "load limit" control; the dump valve would be on automatic pressure control. A-C had performed a technical specification review and a review of the action by the safety system with this mode of operation.

The Safety Committee requested that the accidents considered in the Safeguards report be re-reviewed from the standpoint of the alternate mode of operation.

Analysis of this mode of operation, and investigation of previous analyses of pressure control system failures showed that this particular split valve arrangement was not one of the initial conditions used in the analog computer studies. However, from the various accidents studied, the simultaneous valve closure accident (where the dump fails to respond when the inlet valve closes rapidly) was found to be most severe, in that superheater temperatures approached 1650F for about 2 seconds; the power level at which this occurred at 50%. This particular study led to decisions to add the following three safety actions to protect the reactor from damage in the unlikely event that this accident did occur.

1. Scram when the inlet valves are closed and the dump valve is closed less than x% (Presently x = 4.5%)

2. Scram upon high steam line pressure. (Present set point is 565 psig)

3. Scram upon high steam temperature. (Present set point is +25F above ambient steam temperature). This was the first time high T_2 scram was deemed necessary.

A reevaluation of this accident was made for the case in which the dump valve was partially open (above "closed" contact at 4.5% open) and was disabled because of system failure.

If during this condition another system failure or load dump caused the inlet valves to suddenly close thereby suddenly reducing steam flow, the superheater fuel temperature would rapidly rise until high T_2 or high P_2 scram terminated the excursion. The question which could not be directly documented by analog calculations was: Would the superheater fuel temperature (T_f) rise above 1650F during this excursion, no matter what the initial power was? Extrapolations showed that the fuel temperature would peak below the 1650F point. However, for lack of definitive calculational proof, the plant was shut down until the situation could be thoroughly studied. The results of the study showed that the extrapolations, based on earlier analog work, were correct in predicting fuel temperatures peaking less than 1650F for all power levels.

Based upon the results of analog computer runs and the observation of the reactor experimental data following small changes in steam flow rate, Allis-Chalmers concluded that there is no unreviewed safety question involved with either the present mode or alternate mode of operation of the pressure control system and that the consequences of a failure of the pressure control system are not as severe as reported in the Pathfinder Safeguards Report.

Since there was some uncertainty at the outset of our review concerning the consequences of failure of the pressure control system which could result in a partial loss of steam flow but yet not cause a scram because of simultaneously closure, it was decided to pursue an independent means of protecting the superheater from loss of flow. As a result, a low steam flow scram circuit is being added to provide back-up protection from 20 to 100% power. Since it is now believed that this additional safety protection is unnecessary for protection against pressure control system failures it may, if desired by NSP, be removed at a later time.

In summary, Allis-Chalmers concludes that the plant can be safely operated up to 100% power with the automatic pressure control in either mode of operation, i.e. with the dump valve on auto pressure control and the inlet valves on manual control or with both the dump valve and the inlet valves on auto control after system performance is demonstrated."

The A-C analysis was reviewed by the Operations Committee and accepted prior to the continuation of reactor testing on November 29th. The Safety Committee subsequently reviewed the analysis in December.

C. Reactor Shutdown Testing (Test 431)

The purpose of Test 431 is to determine the response of the reactor to runback, scram, turbine trip, and load dump at various levels during the Test 278 power escalation series.

Specifically, the test has four purposes:

(1) to observe, if superheater fuel thermocouples are available, if local hot spots may be generated in the superheater by unfavorable rod bank configurations which might occur due to runback;

(2) to confirm that all pertinent control systems (pressure, reactor level, etc.) are adjusted to respond properly and in a safe direction when any of the subject safety actions are imposed on the reactor and evaluate their performance;

(3) to observe reactor and turbine behavior as predicted in response to turbine trip and turbine load dump and evaluate their performance;

(4) to evaluate the accuracy of the Pathfinder Analog Simulator Results.

The following information has been extracted from A-C test reports submitted to the Pathfinder Safety Committee.

1. Reactor Testing at 40 MWT (20% Power)

The following actions have been tested in accordance with Test 431.

1. Reactor scram at 40 MW, 300 psig.
2. Raector isolation scram at 40 MW, 540 psig, manual pressure control.
3. Reactor Scram at 40 MW, 540 psig, automatic pressure control.

The following is a presentation of the test results:

a. Reactor Scram at 40 MW, 300 psig

A reactor scram was manually initiated from 40 MW, 300 psig with the reactor pressure, feedwater temperature, and water level control systems in HAND operation on June 21, 1966.

Reactor power fell to essentially zero in about one second and fuel and steam temperatures decreased.

The reactor system behaved as expected following the scram.

b. Reactor Isolation Scram at 40 MW, 540 psig

Reactor isolation scram was manually initiated from 40 MW, 540 psig with reactor pressure, feedwater temperature, and water level control systems in HAND operation on June 24, 1966. Several reactor parameters were recorded on the Offner recorder.

Reactor power as indicated by Channel 5 (out-of-core ion chamber) and in-core ion chambers No. 8 and No. 9 fell off in about one second. Reactor pressure and steam temperature are seen to decrease following scram. The oscillations in the exit steam pressure trace are the results of noise pickup rather than actual pressure oscillations.

Steam flow through the Main Steam Isolation Bypass Valve was recorded. Bypass steam flow was monitored to be zero preceding the isolation scram since nearly all of the steam was being passed by the Main Steam Isolation Valve. Following isolation scram, the MSIV begins to close and is fully closed in 17 seconds. Bypass flow increases while the MSIV is closing as is indicated by the recorded results. Indicated flow is constant during the initial 6 seconds of MSIV closure because the steam flow meter is not yet on scale and during the final 6 seconds of valve closure because the flow meter is saturated at full scale.

Following isolation scram, the bypass isolation valve also begins to close and is fully closed in 120 seconds. Indicated bypass steam flow comes on scale again after 36 seconds (full scale = 75,000 lbs/hr) and indicates zero flow after 58 seconds. If this indication is correct, then steam flow to the main condenser following isolation scram was terminated (or at least reduced to less than 10,000 lb/hr) after about one minute rather than the expected two minutes. This condition is not satisfactory and will be investigated at the earliest possible date. The investigation will be initially directed towards the valve operator which may be highly non-linear (causing the valve to go much more than half-closed in only half the total closing time).

(Note: Information presented to the Safety Committee on September 7, 1966, shows that the closing time of the bypass valve is satisfactory.)

c. Reactor Scram at 40 MW, 540 psig, AUTO Pressure Control

Reactor scram was initiated from 40 MW, 540 psig with reactor pressure and water level control in AUTO and feedwater temperature control in HAND operation on June 26, 1966. The scram was not manually initiated, but rather happened when the operator attempted to drain water from the main steam line.

The results of the test were recorded. As before, reactor power fell off in about one second and fuel temperatures decreased.

Although the reactor water level control system was on AUTO at the time of scram, this system had only single-element control (level) and would therefore be very slow acting. As a result, following a scram, the operator switches level control to HAND and the net effect of it being on AUTO is insignificant.

The pressure control system is left on AUTO following scram. Normal system operation following scram would be dump valve closure in response to decreasing pressure with closure limited at the 8% minimum stop. Dump valve position was recorded and indicates that the valve opened from 11% to 14%. There is no explanation for this behavior and it is concluded that the polarity of the valve position signal was reversed somehow and therefore, that the valve actually closed from 11% to 8%. This conclusion will be checked out during future testing.

2. <u>Reactor Testing at 76 MWT (40% Power)</u>

a. <u>Turbine Trip at 38 MWT</u>

On July 29, 1966, the turbine was tripped with reactor power at about 38 MWT and the dump valve slightly open (about 2%) and controlling pressure. Following the manual trip, the stop valves closed and the dump valve opened to a position sufficient to pass existing main steam flow. The dump valve then slowly closed attempting to control pressure until the minimum-valve-position limit was reached (about 9% indicated).

The turbine trip initiated scram as it should and reactor power fell off in less than one second. Superheater temperature and steam pressure also decreased following the turbine trip and resultant scram.

Proper switchgear action disconnected the generator from the NSP grid immediately after the turbine trip and the turbine-generator coasted down in a satisfactory manner.

b. <u>Generator Load Dump at 38 MWT</u>

On July 29, 1966, the generator load was dumped with reactor power at about 38 MWT and the dump valve slightly open (about 2%) and controlling pressure. Load dump is interlocked to:

(1) quickly close the turbine inlet valves to prevent turbine overspeed,

(2) quickly open the dump valve to a position sufficient to pass existing main steam flow, and

(3) after five seconds, re-open turbine inlet valves to a position sufficient to maintain house load.

The system responded satisfactorily to the load dump.

Offner traces illustrate the dump valve opening at the time of the load dump followed by partial closure five seconds later to compensate for re-opening of the turbine inlet valves. It may be noticed that exit steam pressure and temperature drop off somewhat. This results from the dump valve's greater flow capacity and is satisfactorily corrected by the control system in about 40 seconds.

c. Reactor Scram at 62 MWT

On August 8, 1966, the reactor was scrammed from 62 MWT with automatic pressure control and without turbine operation.

Reactor power fell off in less than one second and superheater temperature and steam pressure decreased. The pressure control system responded satisfactorily.

d. Plant Shutdown at 76 MWT

On August 12, 1966, the reactor was successfully shut down from 76 MWT by control rod run-in, with automatic pressure control and without turbine operation.

Reactor power, superheater temperature, and steam pressure decreased as the control rods were driven into the core. The pressure control system performed satisfactorily.

D. Radiation Survey Results (Test 332)

Test 332 - Radiation Monitoring, is a formal requirement of the testing program. Limited reporting is given here; however, Section V of this report reviews the Pathfinder Chemistry and Radiation Experience to 60% Power in detail.

1. Introduction

Extensive surveys of radiation levels in the Pathfinder plant are being made as a part of the startup test program. The survey areas include the reactor building operating floor, mezzanine floor, and plug floor, and the operating, mezzanine, and basement floors of the fuel building and turbine building. Radiation levels in the nuclear instrumentation ports are also recorded.

2. Results

Representative results are given in Table I and points of measurement are as shown on Figures 8, 9, and 10. The reported dose rates for 100% power (190 Mwt) are given for some points.

3. Conclusions

In several cases, observed dose rates exceed reported results when extrapolated from 20% power to full power. Some discrepancy can be expected because the complex geometry forces rough estimates to be used in many locations.

If dose rates become high enough to hinder operation, recommendations will be made on actions to reduce the levels at these points. It can be expected that each problem which might arise will require a solution pertinent to its own area. However, no operational problems are known to exist at present (40% power). The data to be taken at higher power levels will be analyzed and these extrapolations will again be compared with values reported in ACNP-62016.

TABLE I

OBSERVED DOSE RATES AT SELECTED POINTS

Points	ACNP-62016 Dose Rate 100% Power	Units	Run of 8/5/66 40 MWT	Run of 8/9/66 65 MWT	Run of 8/9/66 80 MWT
		Power Level	40	65	80
		Recirc Flow	62,000	66,000	60,000
		Steam Flow	139,000	210,000	265,000
		Reactor H_2O	475	479	478
		Steam Temp	510-520	560	610
ctor Building					
20			3.0	5.0	6.0
22		closed hatch	14βγ, 200n	17βγ, 250n	25-250
23	< 10 mr		6.0	9.0	4.0
24			3.0	5.0	5.0
25	65 mr	closed hatch	14βγ, 120n	17.-200	27-320
26			2.0	4.0	5.0
28		closed hatch	12βγ, 45n	14.-70	17-94
Handling Bldg					
1	0.6		.02	.08	.15
2			.05	.09	.15
4	.75		.02	.05	.05
5	.3		.02	.02	.05
bine Building					
2	11.6		1.2	2.0	6.0
5	1.80		.02	.05	.05
7	11.0		.02	.05	.02
9	32		.8	1.5	2.0
15			.05	.09	.2
21			.02	.35	.15

E. Two Phase Level Measurements and Steam Quality Measurements

1. Two Phase Level Measurements

Two-phase level measurements were made at 55 Mwt and 80 Mwt with reactor pressure at 540 psig. For the 80 Mwt power, measurements were made on both LL-3, which is 15 inches above normal level (1297'-5") and on LL-2, which is 33 inches above normal level. Comparison of measured and expected values are given below.

Run No.	4	5	
Reactor Power, Mwt	55	80	
Reactor Pressure, psig	540	540	
Indicated Level, LR-251, in	+ 1.0	-1.7 (LL-3)	+14.0 (LL-2)
Two-phase Level, inches (above 1297'-5")	+15	+15	+33
Difference between two-phase measurement and water column, inches	14	16.7	19
Exit Void Fraction	0.19	0.25	0.25
Expected difference between two-phase measurement and water column, inches	10.8	-15.6	13.2 - 21.0(LL-3) 17.8 - 25.6(LL-2)

Comparing the measured difference between the two-phase level and the indicated level with the expected difference in levels, for the lower power level (55 Mwt) the measured is closer to the upper limit. For the higher power level, the measured value is closer to the lower limit, as anticipated.

The high water level scram at full power (+4 inches, indicated) is based on the calculated upper limit of two-phase. The data of test 278.2A shows this scram limit is still valid. If the data at higher powers approaches closer to the lower limit as expected, it may be possible to raise the high water level scram.

2. Steam Quality Measurements

Initial steam quality measurements made since the start of power testing indicated some moisture content both upstream and downstream of the dryers, even when the reactor was shut-down. It was concluded that heat losses in the sample piping were causing erroneous readings. The piping was modified (shortened) and insulated and the steam flow sample was increased. These modifications resulted in low power measurements of about 0.6 to 0.8% moisture. It was concluded that this is the best which can be obtained with the long sample tube required. Normally sample tubes for conventional application are very short. The indicated readings will be corrected by -0.6% to obtain the true moisture content.

Data obtained during test 278.2A is reproduced below:

Power Level Mwt	Pressure psig	Moisture %			
		Upstream		Downstream	
		Measured	Corrected	Measured	Corrected
0	292	0.872	--	0.637	--
7	400	0.605	0	0.605	0
56	532	0.785	0.18	0.810	0.205
80	549	0.870	0.265	0.765	0.160

Downstream moisture contents are less than design (0.5%).

IV. Superheater Performance

The testing results at Pathfinder are of particular interest because the reactor core contains an integral superheater. This report section reviews the significant thermal performance aspects of the superheater from the initial steam flow testing through the 40% power testing. The information has been extracted from A-C test reports submitted to the Pathfinder Safety Committee.

A. Initial Steam Flow Testing to 8 MWT

1. Superheater Thermal Performance

During the first power increase, on May 20, 1966, steam flow was established at 10,000 lbs/hr and power was increased to 2 Mw indicated (1.4×10^{-8} amps on Channel 5). These conditions were held long enough to evaluate superheater thermocouples. The superheater reaches equilibrium temperature in a few minutes. A comparison of measured temperatures with expected (most probable) temperatures for these conditions is shown in Figure 11. The expected temperatures are calculated assuming turbulent flow in all channels. Reynolds number is about 1500 for 10,000 lbs/hr steam flow. Measured temperatures were lower than expected at the steam outlet end (bottom of the core).

Next, steam flow was increased to approximately 28,000 lbs/hr and reactor power was raised to 8 Mw (5.6×10^{-8} amps on Channel 5) with "holds" at 4 Mw and 6 Mw to evaluate superheater temperatures on the three thermocouples recorded continuously in the control room. (Steam outlet (1-00) and outer fuel at 64 inches from the top (0-10) in element A-18 and inner fuel at 56 inches from the top (1-21) in element E-9). At each power level the measured temperature was less than expected. The thermocouples responded smoothly and as expected to changes in steam flow and power. As steam flow was increased, temperatures decreased and when power was increased, temperatures increased. A few minutes was required to reach equilibrium superheater temperatures. There was no tendency for temperature overshoot, i.e., for superheater temperature to exceed the equilibrium value and then return to its equilibrium value.

The next day (May 21, 1966) power was raised to 8 Mw in three steps: 10,000 lbs/hr steam flow and 2 Mw, 28,000 lbs/hr steam flow and 6 Mw and then raised to 8 Mw. This time reactor power was kept at about 8 Mw (5.6×10^{-8} amps on Channel 5) for about 10 minutes, which was long enough to obtain thermocouple data from the multipoint recorders in the containment building. The reactor was not quite at equilibrium since reactor water temperature was decreasing. Again, measured temperatures near the steam outlet end were lower than expected.

During the afternoon of May 21, another run at 8 Mw was made which reproduced the results of the previous 8 Mw run. The reactor was not quite at equilibrium since the reactor water temperature was decreasing. For the next run, the minimum steam flow scram was reduced to 21,500 lbs/hr, and the maximum power scram increased by 8% and minumum reactor water temperature was limited to 400 F.

On May 23, 1966, another run was made at 6×10^{-8} amps on Channel 5 (8 Mw indicated) and at about 23,000 lbs/hr steam flow. Equilibrium reactor conditions were achieved and a heat balance was made. The heat balance gave a reactor power of 6.0 Mw. Measured superheater temperatures are compared with expected in Figure 12. Temperatures in elements A-18 and Z-1 are close to expected, while temperatures in E-9 and W-9 are slightly higher than expected (most probable) temperatures. All temperatures are considerably below the maximum temperatures (including uncertainties). A comparison of measured thermocouple temperatures with the range of expected temperatures is given in the following table for the three thermocouples recorded in the control room.

COMPARISON OF MEASURED AND CALCULATED (NOTE 1)

SUPERHEATER THERMOCOUPLES

Thermocouple Designation	Fuel Element Designation	Calculated (Note 2) Most Probable	Calculated (Note 2) Maximum	Measured
1-00	A-18	585	762	560
0-10	A-18	563	680	560
1-21	E-9	556	693	560

NOTE 1 Comparison made for 6 Mwt, 23,300 lbs/hr steam flow, Group III control rods at 27 inches and reactor water temperature = 420 F.

NOTE 2 Most probable from Figure 12.

The most accurate comparison of expected and measured temperatures is that for steam outlet thermocouples, since no correction is needed for thermocouple temperature relative to fuel temperature. Thermocouples within the lower 10 inches of fuel length have a small correction and those at 35 and 53 inches from the top have the largest correction. The calculated thermocouple temperatures assume the largest value of thermal resistance between the fuel and thermocouple. The steam outlet temperatures indicate that power may be slightly lower than expected in Element A-18 and slightly higher than expected in Element W-9.

The high measured value of 600 F on I-21 of Element E-9 compared to a most probable value of 556 F may be due to a lower thermal resistance between the fuel plate and thermocouple than that used in the calculation. However, it is still considerably below the maximum value of 693 F.

It is concluded from these measurements that:

1. Turbulent flow exists in fuel channels at a steam flow as low as 10,000 lbs/hr. In Figure 11, measured temperatures are compared with expected temperatures based on turbulent flow; the margin between expected and measured is about the same as for flow rates twice as high where turbulence is assured (Reynolds number is greater than 3,000).

2. Superheater temperatures in the instrumented fuel elements are closely predicted using most probable values of flux distribution and quantities used in the temperature calculation.

3. The most probable peak temperature in the superheater at 6 Mw power, 23,300 lbs/hr steam flow, 420 F reactor water temperature with Group III control rods at 27 inches is 687 F. The maximum peak temperature including uncertainties is 915 F.

3. Superheater Power Fraction

The fraction of reactor power generated in the superheater is deduced from the superheater outlet steam temperature. Superheater outlet steam temperature is measured on several instruments. The most reliable measurements gave a superheater outlet steam temperature ranging from 480 to 490 F at 6 Mw equilibrium conditions. This corresponds to a superheater power fraction of .05 to .06, including a correction for heat loss to the moderator. The calculated heat loss to environment from the steam pipe is small (less than 1 F at 25,000 lbs/hr steam flow). The predicted value for control rods Group III at 27 inches based on physics calculations is 0.067.

The lower measured power fraction compared to predicted may be due to larger heat loss to the moderator than calculated or lower power generated in the superheater. At higher power levels, heat loss to the moderator is less significant in deducing power fraction from superheater outlet temperature.

B. Reactor Testing to 40 MWT (20% Power)

1. Summary

Superheater temperatures were close to most probable calculated temperatures and considerably less than maximum calculated temperatures (including uncertainties) throughout Test 278.1A. No unexplained anomalous thermocouple behavior was observed. A total of 23 thermocouples (12 with continuous leads and 11 with connections) are still functioning.

The peak measured superheater temperature was 716 F for equilibrium conditions of 40 Mw and 540 psig. The corresponding calculated peak superheater temperature is 733F (most probable) and 945F (including uncertainties). The maximum measured steam outlet temperature was 545F which corresponds to a superheater power fraction of 0.057. The calculated power fraction is 0.078.

After reactor shutdown, steam flow of 10,000 lbs/hr was maintained for 10 minutes and then cut off. Superheater fuel temperaturas decreased rapidly to approximately saturated steam temperature during the steam flow period and then increased after steam flow was cut off. The maximum superheater temperature increase following steam flow cutoff was about 80F following an isolation scram after about 9 hours operation at 40 Mwt. Peak temperatures occurred about 30 minutes after shutdown.

2. Superheater Temperatures during Power Operation

Superheater temperatures during equilibrium runs at 18 Mwt at 300 psig and for 40 Mwt at 540 psig are given in Table 2. The calculated peak superheater temperature for these equilibrium conditions is also given in Table 2.

The local temperature of the superheater relative to core inlet steam temperature is a significant measure of the superheater performance. This quantity is determined by flux distributions, steam flow distribution, steam physical properties, and geometry of the fuel elements. During the test, measured and calculated temperature excess above steam inlet for each working thermocouple was used to evaluate superheater performance.

In Table 2, the ratio of measured to calculated temperature excess above inlet steam temperature are reproduced for each thermocouple. The calculated temperature of each thermocouple is the most probable temperature for the measured power, steam flow and control rod position.

The measured temperature excess is close to the most probable calculated temperature excess. The majority of measured temperatures are less than the most probable calculated temperatures. The ratio of measured-to-calculated temperature excess is largest (1.21 at 28 Mwt at 300 psig) for the inner fuel tube thermocouple, 43 inches from the top in element E-9. This is still considerably below the allowable margin included in the design by use of hot channel factors. The margin due to hot channel factors,

Reactor Power, Mwt	17.8	40.0
Reactor Pressure, psia	325	570
Reactor Water Temperature, °F	425	480
GP III Control Rod Position, in.	28.8	47.4
Reactor Steam Flow, lbs/hr	66,000	161,000
Calculated Peak Superheater Temperature (Note 1)		
a. Nominal	647	733
b. Maximum	835	945
c. Ratio: $\frac{\text{Maximum - Inlet}}{\text{Nominal - Inlet}}$	1.85	1.83

Local Superheater Temperature (Note 2)

Fuel Element Number	Distance from Top, In.	Thermocouple Welded to	T_m	T_c	$\frac{T_m - T_{in}}{T_c - T_{in}}$	T_m	T_c	$\frac{T_m - T_{in}}{T_c - T_{in}}$
A-18	35	Poison Tube	483	488	0.92	557	560	0.96
	35	Inner Fuel	502	506	0.95	593	587	1.06
	35	Outer Fuel	488	504	0.80	561	595	0.71
	64	Outer Fuel	608	603	1.03	683	685	0.99
	68	Outer Fuel	589	604	0.92	668	685	0.92
		Outlet Steam	602	601	1.01	669	692	0.89
Z-1	35	Inner Fuel	515	506	1.11	597	587	1.09
	66	Poison Tube	605	603	1.01	662	693	0.86
	70	Inner Fuel	632	624	1.04	682	713	0.87
E-9	17	Poison Tube	448	448	1.00	495	500	0.75
	17	Inner Fuel	457	461	0.89	508	513	0.85
	17	Outer Fuel	457	464	0.82	508	515	0.80
	35	Inner Fuel	527	520	1.07	592	590	1.02
	35	Outer Fuel	500	520	0.79	567	590	0.79
	53	Poison Tube	572	563	1.07	631	645	0.92
	53	Inner Fuel	610	586	1.15	698	670	1.15
W-9	35	Inner Fuel	529	520	1.09	592	590	1.02
	35	Outer Fuel	511	520	0.91	580	590	0.91
	68	Outer Fuel	615	604	1.06	669	690	0.90
	70	Inner Fuel	658	624	1.17	716	717	1.00
		Outlet Steam	615	603	1.06	671	690	0.91
K-10	53	Poison Tube	475	474	1.02	507	525	0.60
	53	Inner Fuel	475	482	0.88	524	535	0.80
	53	Outer Fuel	475	479	0.93	528	535	0.87

NOTE (1) Nominal is the most probable value; maximum includes uncertainties in neutron flux and temperature calculation

(2) T_m is measured temperature; T_c is calculated (most probable); T_{in} is inlet steam temperature = Reactor Water Temperature

expressed as a ratio of temperature excess above inlet coolant is about 1.8. (See Table 2, calculated peak temperature). Scram setpoints on power and flow during step increases and on steam outlet temperature are based on temperatures calculated using hot channel factors.

As expected, the maximum fuel temperatures occur within the bottom 12 inches of the fuel element. This is due to the large steam temperature rise to the hot spot relative to the temperature difference between fuel plate (on T/C) and bulk steam. For the thermocouple having the highest measured temperature at 40 Mw and 540 psig, (716° F for inner fuel T/C at 70 inches in element W-9) the calculated temperatures differences are:

Bulk steam at T/C minus Inlet - 225°F

T/C minus Bulk steam at T/C - 12° F

Fuel Plate minus Bulk steam at T/C - 34° F

Since steam temperature rise is most significant for the maximum fuel temperatures (near the outlet end of the fuel) the quantities which are of most significance are those which affect the heat generation over the length of the fuel element or coolant flow in the fuel element. Local conditions (axial flux, local hot channel factors and thermocouple correction factor) are less significant on maximum temperature.

Temperatures in the upper half of the core are affected more by local conditions. For the 40 Mwt and 540 psig run, calculated temperature differences for the inner fuel thermocouples in element E-9 at 17 inches and 35 inches are:

Distance from Top of Core, In.	17	35
Bulk Steam at T/C minus Inlet Steam	22	97
T/C minus Bulk Steam	11	29
Fuel Plate minus Bulk Steam	31	83

Therefore, for temperatures in the upper core region, local flux and thermocouple correction factors are more significant in the ratio of calculated-to-measured temperature excess above inlet steam temperature. However, the temperatures of major interest are the maximum temperatures which occur near the bottom of the fuel, and these are not significantly affected by local conditions.

There are a total of twelve high-heat-generation elements in the reactor, four of which contain thermocouples. These are located at the corners of the superheater-boiler interface.

The flux wire irradiation at 6 Mwt showed that an element midway between the corner elements (S-5) has an average flux over the length of 0.81 times that of a corner element (N-1). Four instrumented elements out of twelve total hot elements is a significant number.

The following conclusions are made based on steady-state temperature measurements in Test 278.1A.

1. Hot channel factors used to calculate maximum superheater temperatures are conservative. The largest ratio of measured-to-most probable calculated temperature excess (above inlet coolant) is 1.21 whereas the hot channel factors give a ratio of 1.8. For the majority of thermocouples, the measured temperature excess above inlet coolant is less than the most probable temperature excess.

2. No significant discrepancy in calculated flux in the hot fuel elements is apparent. Temperatures in symmetrical elements on opposite sides of the core are about the same, indicating no significant flux tilt. Calculated temperatures based on calculated flux distributions agree well with measured temperatures.

Three superheater temperatures were recorded in the control room during Test 278.1A. Thermocouples responded rapidly and as expected with changes in power or steam flow.

Figure 13 is typical of succeeding power increases from 6 - 16 Mwt - a smooth decrease in temperature as flow is increased and a smooth increase in temperature with no over-shoot as power is increased. The heat balance at the 16 Mw nominal power level gave a thermal power of 17.8 Mw compared to an expected value of 16.7 Mwt. Therefore, the maximum height of Group III control rods was limited to 36 inches instead of 50 inches to account for the higher than expected power. Subsequently, when it became apparent that xenon would require higher rod positions, the minimum water temperature was raised from 390 F to 405 F to account for the higher power and the control rod limit was increased to 50 inches. The scram power limit of 115% of range and the minimum steam flow limit of 56,000 lbs/hr were retained.

Figure 14 shows the first successful attempt of valve interchange (Bypass valve to MSIV) and power increase from 16 Mwt to 30 Mwt. Temperatures decrease as flow is increased above 80,000 lbs/hr and then increases as power is increased.

After equilibrium was achieved at 300 psig and 30 Mwt, the steam outlet scram setpoint was calculated, based on the measured outlet steam temperature. The expected outlet steam temperature was 510 F and the corresponding scram setpoint was 586 F. The measured outlet steam temperature was 470 F. Therefore, the scram setpoint was reduced to 559 F (Reduction is 2/3 of the difference between expected and measured outlet steam temperature). Control rod position was 30 inches.

Figure 15 shows the fuel temperature during a power increase from 35 - 40 Mw. Steam flow is increased simultaneously with power so that superheater temperatures change only a small amount. The steam outlet temperature scram setpoint based on the measured outlet temperature at 40 Mw, 300 psig and control rods at 40 inches is 560 F.

Figure 16 shows the pressurization step. Good control of steam flow and power results in only a slight variation in superheater temperature. The gradual increase in temperature is due to increasing reactor pressure. It was not necessary to reset the outlet steam scram setpoint during pressurization. The steam outlet temperature at 540 psig was 540 F and the scram setpoint for 285 psig based on measured data was 560 F. After equilibrium conditions were achieved at 540 psig, the scram setpoint for 500 psig minimum pressure was calculated to be 586 F.

It may be seen in Figure 17 that when the reactor was shutdown by a scram superheater temperatures decreased rapidly during scram and decreased slowly until steam flow was cut off. After steam flow was cut off, the outlet steam T/C (1-00), decreased to reactor water temperature; the fuel thermocouples (0-10 and 1-39) increase and reach a maximum about 1/2 hour after shutdown. The maximum temperature increase of fuel thermocouples during Test 278.1A was about 80 F.

3. Superheater Power Fraction

The superheater power fraction based on measured steam outlet temperature is compared with the calculated power fraction in Table 3. The power fraction was expected to increase as control rod Group III was raised. The measured power fraction remained fairly constant for control rod positions between 27 inches and 50 inches.

The superheater outlet temperature is most accurately measured by the thermocouples in the main steam line which are used in the steam outlet temperature scram. However, the minimum temperature which can be measured on these thermocouples is 500 F. Steam outlet temperatures from the resistance thermometers in the main steam line were used for Runs 1, 2, 3, and 4. The measured temperatures were increased by 20 F in computing power fraction, since comparison with the thermocouples for Runs 5 and 6 showed that the resistance thermometers read 20 F less than the thermocouples.

The power fraction is calculated from enthalpies of outlet steam saturated vapor and feedwater. The heat loss and energy required to heat up the seal water are almost completely off set by the pump power and may be neglected for powers above 18 Mwt.

TABLE

SUPERHEATER POWER FRACTION

Run No.	1	2	3	4	5	6
Reactor Power, Mwt	6	17.8	27.9	40.2	40.0	39.5
Reactor Pressure, psia	317	325	290	310	570	562
Reactor Water Temperature, F	422	425	414	420	480	476
Steam Outlet Temperature, F (Note 1)	500	510	490	500	540	545
Steam Flow, lbs/hr	25,000	66,000	105,000	150,000	161,000	150,000
Feedwater Temperature, F	372	362	388	376	394	400
Group III Control Rod Position, in	27.3	28.8	30	34.9	47.4	50.1
Fraction of Power in Superheater (Note 2)						
a. Measured	.057	.061	.056	.059	.055	.057
b. Calculated	.067	.067	.068	.070	.076	.078
c. Ratio: Measured / Calculated	0.85	0.91	0.82	0.84	0.72	0.73

Note 1: Runs 1 through 4 are measured values from resistance thermometers plus 20 F. Resistance thermometers read 20 F less than main steam line thermocouples. Runs 5 and 6 are from main steam thermocouples.

Note 2: Measured Power Fraction = $\frac{\text{Enthalpy of Outlet Steam - Sat Vapor Enthalpy}}{\text{Enthalpy of Outlet Steam - Feedwater Enthalpy}}$

4. Shutdown Cooling

The isolation scram on June 24, 1966 was analyzed to compare measured temperatures after shutdown with calculated temperatures. The reactor has been operated for 12.4 hours at 18 Mwt and 9.33 hours at 40 Mwt. The emergency condenser was cut off 6 minutes after shutdown. The reactor water temperature dropped from 476 F to 444 F in 1 minute and to 431 F in the next two minutes. At the time of steam flow cut off (6 minutes) the reactor water temperature was about 425 F. Peak superheater temperatures after steam flow cut off were reached about 30 minutes after shutdown.

The operating power - time history resulted in a calculated decay power at 30 minutes after shutdown of 0.358 Mwt. The measured temperatures show reasonably good agreement with the most probable calculated temperatures. As expected the peak temperature occurred near the core center. The maximum temperature was 513 F compared to a calculated maximum temperature of 570 F.

The decay heat curves used in the above calculation will be used in Test 278.2A. The data from Test 278.1A shows that peak superheater temperatures for no steam flow are adequately predicted using these decay heat curves for the limited power time history achieved in Test 278.1A. Temperatures after shutdown will continue to be monitored during Test 278.2A and compared with calculated values.

C. Reactor Testing to 76 MWT (40% Power)

1. Summary

Superheater peak temperatures during the superheater rod withdrawal, control rod interchange and power escalation from 40 Mwt to 80 Mwt were close to the most probable calculated value and considerably less than the maximum which includes design uncertainties. The maximum measured temperature was 850°F, which occurred with CRG II fully withdrawn and IV and V at their minimum position (18 inches). This corresponds to a most probable peak fuel temperature of 907°F and an upper limit, based on thermocouple data, of 1130°F.

There are a total of 22 superheater thermocouples still functioning (12 with continuous leads and 10 with connectors) compared with 23 at the end of Test 278.1A. Thermocouple 1-39 (W-9) has failed. No unexplained anomalous thermocouple behavior was observed.

Superheater outlet steam temperature was 680°F with superheater rods (CRG I) and CRG II withdrawn and CRG IV and V low in the core. This corresponds to a superheater power fraction of 14% compared with expected of 14.6%. Reactor power was 64 Mwt and pressure was 540 psig.

The maximum measured superheater temperature following shutdown was 640°F in the middle of the core, compared with a calculated most probable thermocouple temperature of 680°F and calculated decay heat of 0.73 Mwt. Measurements of superheater outlet steam temperature and reactor water temperature may be useful as an alternative to the present procedure of calculating residual decay heat for normal shutdown procedures.

Two-phase level measurements at 80 Mwt indicate that the present high level scram for full power operation (+4 inches, indicated) will be adequate and that it may be possible to raise the scram limit.

Steam quality measurements at 80 Mwt show that moisture of steam downstream from the dryer is about 0.2% at 80 Mwt. The design value is 0.5%.

2. Superheater Temperatures during Power Operation

Data of 278.1A was used to evaluate the upper limit on superheater fuel temperature, compared with the calculated upper limit based on design hot channel factors. It was concluded that a factor of 1.52 applied to the most probable value of ($T_{max} - T_{in}$) would give the upper limit of superheater temperature compared with a design factor of 1.80, for the power level and control rod configurations of Test 278.1A. Thus for a peak superheater temperature of 1270°F, using design hot channel factors, the upper limit based on 278.1A results is 1045°F.

The data obtained during Test 278.2A confirm the conclusions of Test 278.1A; i.e., that the design hot channel factors are too conservative.

During the power escalation to 80 MWT, and superheater rod withdrawal, the maximum ratio of

$$\frac{T_m - T_{in}}{T_c - T_{in}} = \frac{\text{(measured temperature - inlet steam temperature)}}{\text{(calculated temperature - inlet steam temperature)}}$$

was 1.45 and it occurred in a central element (K-10), where heat flux was low and small errors in measured temperature would influence the factor significantly. The highest measured temperatures which are most significant, are in the lower 12 inches of the core boundary elements. For these thermocouples, the highest value of the ratio was 1.18.

During the control rod interchange the highest value of $\frac{T_m - T_{in}}{T_c - T_{in}}$ was 1.48 and it occurred in element Z-1 at the middle of the core. For thermocouples in the lower 12 inches of the core, where highest temperatures occur, the maximum ratio is 1.05. The average of all thermocouples for each run is below 1.0.

The superheater temperatures varied periodically when the reactor was on automatic pressure control. The frequency was about 0.5 cycles per minute and the maximum amplitude on the hottest thermocouple Element A-18 T/C 0 - 10 was about ± 25°F.

The superheater temperatures during the power increase from 40 MWT to about 60 MWT did not change appreciably. Maximum measured temperatures increased from 700°F to 740°F when Control Rod Group III was moved from 52 inches to 73 inches.

During the superheater rod withdrawal the measured temperature increased from 714°F to 780°F as control rods were withdrawn from 0 to 73 inches, due primarily to the increase in superheater power fraction. The largest changes in superheater temperature were, as expected, on the central element (1-21 on Element K-10) which increased from 521°F to 628°F.

During the control rod interchange withdrawing Group II rods increased peak superheater temperatures, while inserting Groups IV and V did not change peak superheater temperatures significantly. The highest measured temperature was 850°F and it occurred in Element W-9 on thermocouple 1-04 when Group II was fully withdrawn with IV and V at 18 inches. For this condition, the most probable peak fuel surface temperature is 907°F and the upper limit is 1130°F, using the factor of 1.52 based on experimental results. The maximum peak temperature using the design hot channel factor (1.70) is 1208°F.

It is concluded from the evaluation of superheater temperatures during Test 278.2A that the design hot channel factors used to predict peak superheater temperatures are too conservative. For design maximum temperatures of 1270°F, the data indicates that the upper limit is 1045°F and for maximum design temperatures of 1450°F, the upper limit based on the data is 1300°F. Data will be further evaluated during power escalation to 100% power.

3. <u>Superheater Outlet Steam Temperature</u>

Superheater outlet steam temperature increased significantly as Group I rods were withdrawn, and a lesser amount with the rod interchange. Superheater outlet temperature is directly related to superheater power fraction.

From Figure 18, it is seen that with Group III rods withdrawn and prior to the superheater rod withdrawal, the outlet temperature is about 40°F lower than calculated. With superheater rods withdrawn, the outlet temperature was close to calculated. The shape of the outlet temperature versus rod position curves is S-shaped, whereas the predicted curve is linear. For the first 20 inches of rod withdrawal of Group I and Group III, outlet temperature is not appreciably affected.

When Group II rods were fully withdrawn, the steam outlet temperature was again about 40°F below calculated. Insertion of CRG IV and V has a small effect on superheater outlet temperature, for motions between 24 inches and 73 inches. The largest effect with Group IV and V are with Group II fully withdrawn and IV and V low in the core.

During the rod interchange, the maximum power fraction in the superheater was 14%. The expected power fraction was 16%, including an uncertainty factor of 1.10. Thus the most probable calculated value of power fraction is 14.5%, which agrees well with the measured value.

4. Shutdown Cooling

The procedure for shutdown cooling in Test 278.2A, using calculated residual decay heat, proved satisfactory. Graphs of power versus time are maintained on a daily basis and steam flow cut-off times are selected from the curves in Figure 278.2A.7.9 of Test Procedura 278.2A. When the total thermal power approached the limit specified in the procedure (20,000,000 Kwhr), the procedure was modified to permit operation up to 200,000,000 Kwhr. The modification consists of adding 0.24 Mw to the residual decay heat. The 0.24 Mwt is the residual decay heat at one week after shutdown from 1000 hours of operation at 200 Mwt.

The response of reactor water temperature and steam outlet temperature following a scram was reviewed to determine their usefulness as an alternate or supplementary procedure to calculating residual decay heat. Both measurements appear to be of sufficient magnitude to be useful. The rate of water temperature decrease during the constant steam flow interval (10,000 lbs/hr) is about 12°F in 10 minutes and is linear over a 10 minute period. Steam outlet temperature response is more rapid than water temperature, as expected. For higher decay heat levels, steam temperature and rate of water temperature increase are expected to be more useful.

The maximum measured temperature following a scram was 640°F on thermocouple 1-39 in element Z-1. This occurred 38 minutes after a scram from 76 Mwt. (11:33, 8/16/66). Steam flow was cut off 17 minutes after the scram. The calculated decay heat at 38 minutes after the shutdown was 0.73 Mwt and the calculated nominal temperature of this thermocouple was 680°F. Superheater fuel temperatures are only about 20°F higher than the thermocouple temperature.

V. Pathfinder Chemistry and Radiation Experience to 60% of Full Power

1. Chemistry Analysis

It was originally assumed that the primary system limits stated in the Tech Specs would require a daily laboratory analysis for each variable listed. These variables are conductivity, pH, chlorides, iodine, and boron. This requirement has been relaxed through discussions with the USAEC and the program now depends on inline instruments and lab analyses to monitor the system. For example, we depend on inline instruments for conductivity and chloride values. These values are checked periodically in the lab. The gross activity in the primary system is determined daily and therefore a gross iodine number is not determined every day. The gross iodine spectrum is used for fuel surveillance.

The chloride ion content in the primary system has been near the minimum detectable limit of 12 ppb until recently. There have been two occasions following startups during which the chloride ion content has risen to a maximum during the first 24 hours and then dropped off. There was a high of 90 ppb on October 26 and a resample one hour later was 64 ppb. The next day it was down to 12 ppb. Recently the chloride ion content has been varying between 12 and 30 ppb.

The pH of the reactor water and the feedwater has been very stable. The feedwater pH is 7 to 7.5, and the reactor water pH is 8.3 to 8.8. The original limit set by A-C Co. for the reactor water pH was 8.4. This limit was discussed during an Operations Committee meeting and the A-C Co. chemist was contacted as a result. It was decided that the 8.4 limit was actually a feedwater limit and should not be held as a reactor water limit.

The gross β activity in the primary system has been increasing steadily and the present activity is in the 10^{-2} uc/cc range. This activity level has caused some concern. It was first assumed that the extended shutdowns without purification flow were causing the high activity but recently it has been found that the purification flow does not remove the activity when there is no other flow in the primary system. Spectrum analysis indicates that the activity is activated corrosion products and the isotopes are the components of the steels. Recently a trace amount of zirconium has been detectable.

The crud levels in the primary system have been a problem to analyze. The extended period of time required to filter large volume samples has not been generally available because approximately 18,000 cc of primary system water must be filtered to get a weighable quantity of particulate matter. The filter paper usually reads 100 mr/hr or more after the filtration and from .02 to .1 ppm of filterable crud has been found.

The oxygen levels have been very steady. The feedwater contains .06 ppm and the reactor water contains .17 ppm.

The boron content in the primary system has never approached the maximum limit of .5 ppm.

Attempts to measure the silica content in the primary system have been unsuccessful because it is very difficult to get the apparatus clean enough to permit sample concentration with consistent results.

The inline conductivity instruments are checked daily by lab analyses. The feedwater has been consistently below 1 umho/cm. The reactor water conductivity increases after startups and then settles to a steady value within the first 24 hours. The maximums have been as high as 4.5 umho/cm and 3.5 umho/cm.

The isotopes in the primary system have been followed by gamma spectroscopy. Fresh primary system samples contain N^{13}, Cu^{64}, Cr^{51}, Mn^{56}, Zn^{65}, and Fe^{59}-Co^{60} in identifiable quantities. Older samples, on the order of ten days, contain 99% Zn^{65} with identifiable quantities of Cr^{51}, Co^{60}-Fe^{59}, and Zr^{95}-Nb^{95} in equilibrium.

2. Radiation Measurements

Most of the complete plant radiation surveys are being done under the Test 332 format and the predicted radiation levels are identified by areas in the ACNP-62016 report. The surveys are done to monitor the buildup at specific pieces of equipment. The results of all surveys are filed with NSP-site, A-C Co-site, and A-C Co-Washington.

The first detectable radiation problem was the high radiation levels in the basement of the fuel handling building around the purification line. The level reached 340 mr/hr on August 13 with the reactor at 76 MWT and 47,000 lbs/hr purification flow. A sleeve was built around the line and the volume between the line and the sleeve was filled with lead shot. The next survey was made on August 20 with 72 MWT and 30,000 lbs/hr purification flow. The radiation level on the outside of the lead shot shield was 4 mr/hr.

The recirculation pump vaults have been locked and the hatch covers shielded since 30 MWT operation. Personnel are not allowed in the vaults when the reactor is operating. The last nautron survey that had readings within the range of the survey instrument was made on August 13 with the reactor at 76 MWT. At that time measurements of 1.3 to 1.5 R/hr of neutrons were made. The 115 MWT survey gave indication that the field in the vaults was well beyond the instrument maximum of 1.5 R/hr of neutrons. The gamma field just below the hatch was 250 mr/hr in the highest vault. The plug floor in general was 4 to 10 mr/hr with approximately 2-1/2 feet of concrete block over the hatches. No other radiation problems have been found in the reactor building.

The lower floor of the hot side of the turbine building has been a controlled area for some time. The highest readings have been around the air ejector, No. 14 heater, and the main steam line. The highest reading around the air ejector was 1.5 R/hr at 115 MWT. The bottom of No. 14 heater was 1.1 R/hr. This heater was 250 mr/hr at the mezzanine level. The main steam line was 600 mr/hr at the elbow just after the expansion bellows. The basement of the stage heater area was generally at 100 mr/hr or slightly higher. These areas have shown increases roughly proportional to steam flow with turbine operation.

The radiation levels on the turbine building operation floor have come up with the power increase from 40 to 60%. The readings taken at 17 MWE on October 20 around the turbine were 13 to 21 mr/hr. The readings taken at 25 MWE were as high as 115 mr/hr and the 2.5 mr/hr isodose line was approximately 12 feet from the turbine. The hallway at the entrance to the turbine building was .2 to .5 mr/hr. This increased the background on the hand and foot counter from 150 to 400 cpm and required the relocation of the instrument.

The turbine building and reactor building exhaust duct monitors are both reading high during operation. At 60% power both of these monitors were reading approximately 1-1/2 decades above the sampled activity. This effect has been increasing with increasing power operation and it is suspected that the turbine building exhaust duct monitor is seeing shine from the turbine and the reactor building exhaust duct monitor is seeing shine from the steam chase. This shine effect has also been seen on the fuel handling building duct monitor during high purification flow rates.

The shine in the count room has increased the background on the instruments by 50%. This is not a health problem but is mentioned here to point out that a new background must be determined at each power change in order to do low level counting.

3. Off-Site Monitoring

The environment was continually monitored with film badges and by samples of the biota. There were no increases in radiation levels in the environment during this period which can be attributed to plant operation.

There was an increase in film darkening during the latter part of the summer. This was considered to be a temperature effect. The film holders were modified to shade the film packets and allow ventilation. The modification appears to have corrected the problem. The USAEC supplied thermo luminescent dosimeters to be installed at the monitoring stations. These dosimeters were posted during November and the results were compared with the film results. The film monitors will be checked again in the future with TLD's supplied by the Commission.

4. Activity Releases

Month	Liquid		Gaseous	
	Total (μc)	After Dilution(μc)	Total Gas (μc)	Total Particles(μc)
June	493.3	1.43×10^{-8}	1.49×10^{8}	[illegible]$.12 \times 10^{3}$
July	2,533.9	8.68×10^{-8}	1.95×10^{8}	5.58×10^{3}
August	10,804.3	4.15×10^{-8}	6.17×10^{8}	1.44×10^{4}
September	2,683.3	1.03×10^{-7}	Shutdown	Shutdown
October	19,899.7	1.55×10^{-7}	7.48×10^{8}	9.49×10^{3}
November	4,712.4	7.81×10^{-8}	1.07×10^{8}	5.18×10^{3}

No solid waste shipments from the site during this six-month period.

5. System Radiation Levels

Reactor water

7.4×10^{-2} uc/cc gross β

1.5×10^{-5} uc/cc gross iodine

Main steam and feedwater

8×10^{-4} uc/cc gross β

6. Health Physics

There have been no health physics problems in the area of personnel monitoring. The highest exposure in any two week period was 210 mrem. During this period we were working on the SbBe sources and inspecting a reactor head control rod drive nozzle. There were 21 exposures in all during this two week period of September 9 to September 30. There have been no significant "surprise" exposures.

We are in the process of whole body counting the Pathfinder personnel. The intent is to discontinue bioassay as a routine body burden surveillance program. The preliminary results indicate Zn-65 in some of the clothing but no internal deposition.

7. Miscellaneous

Two trial runs with the sample boards were made at 60% power. These runs were made to verify that the boards worked and to determine the radiation levels which would exist. Two items were apparent. The

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crud levels in the reactor water are such that an extended run would be very difficult. The filter loads and flow adjustments are required at least every 1/2 hour. During a 2 hour test run the filter reaches 50 mr/hr. The radiation levels were higher on the resin columns than the filters and the reactor water cation column reached 420 mr/hr during this two hour run.

The liquid waste discharge scheme will probably never be used as it was originally intended. We have not been able to keep the volume chamber for the discharge monitor clean and at a low background. An extra flanged opening and a steam line have been installed for cleaning purposes, but discharge by the information from the inline monitor would require almost daily steam cleaning of the monitor. At present all batches are held up until a lab analysis can be made.

The off-gas system problems have not been resolved. The first absolute filter gets wet and blows out during off-gas surges and we have not been able to keep the pressure differential equipment in service across this filter. It is uncertain when the filter gets wet but it is assumed that the filter blows during off-gas surges when first pulling condenser vacuum or following the transfer of auxiliaries after scrams. Under these conditions, the second absolute filter, downstream of the off-gas hold-up tanks, maintains the integrity of the system. This filter has never been blown.

A leak in the off-gas system was found at the off-gas compressor. At high off-gas flows the gas passes the piston and blows out of the crank case opening. A better system of venting the compressor is being studied at present.

The airborne activity release from the fuel handling building sumps has not been solved. The floor drains are kept covered to prevent the release.

The primary system tritium levels have been between 1×10^{-3} and 3×10^{-3} uc/cc. This is somewhat above the expected levels but are still not above the 3×10^{-3} uc/cc waste release limit. The tritium analyses are run by Isotopes, Inc.

VI. Safety System Operations

Throughout the testing program associated with the escalation to 100% power, several safety system modifications have been completed. These modifications (50.59 review items) are submitted to the Pathfinder Operations Committee as "Items for Safety Review" and the Operations Committee actions are reviewed by NSP management personnel and the Safety Committee. Safety system modifications which are not within the scope of the Operating License DPR-11, Section 50.59 of 10 CFR 50, or the Technical Specifications, require submittals to the AEC, generally in the form of Technical Specification changes.

This section of the report summarizes the results of the required safety system checks and reviews the safety system changes made during the six month operating period.

A. Routine Safety System Checks

All safety system checks required by the Technical Specifications have been performed as required and include the following:

1. Boron Injection System

Testing is required every two months of power operation; however, operational tests were performed every two months regardless of reactor operation. No maintenance has been required on this system since its pre-operational checkout. Periodic tests on this system were completed on May 31, July 26, September 22, and November 10, 1966.

2. Process Instrumentation Systems

Testing requirements for these systems are that components which supply input signals shall be tested each time the reactor system is depressurized if more than 90 days have elapsed since the previous test. Testing philosophy to date has been to initiate a calibration signal into the system, if possible, and observe the system actions through to the final action device. All of these systems operated satisfactorily and no maintenance was required during the report period. Systems included in this category and the dates tested are:

a. Reactor Scram Modes

Testing completed June 15 and September 26, 1966

b. Reactor Building Isolation Scram

Testing completed June 10 and October 1, 1966

c. Reactor Isolation Scram

Testing completed June 10 and October 1, 1966

d. Runback

Testing completed June 14 and August 10, 1966

e. Control System Interlocks

Testing completed June 3 and September 3, 1966

Safety systems, in addition to the above, which are periodically tested using the testing criteria given for the Process Instrumentation include the following:

a. Reactor Building Spray Water Valve

Testing performed on May 27 and September 10, 1966. No maintenance has been required on this system.

b. Condenser Emergency Fill Valve

Testing performed on May 26 and September 3, 1966. No maintenance was required on this system during the report period.

B. Pressure Control System

The reactor pressure control system has been a major source of maintenance and operational problems. These problems include: frequent hydraulic oil leaks and piping failures, sluggish operation of the dump valve due to dirt in the oil system, inadequate cooling of the hydraulic system, loss of prime on the idle hydraulic pump, excessive vibration of the hydraulic unit, erratic response when rate action is used in the control system, electrical noise pickup, drift of signals due to temperature and power supply frequency effects, non-linearity of turbine inlet valve position feedback, difficulty in obtaining proper adjustment of control units, bumps on transfer from auto to hand and vice versa, deadband and time lag in the inlet valve controls causing hunting of the system when the inlet valves are on auto, "normal" maintenance and replacement of failed modules.

The majority of the above items have been brought under control by changes in operating procedures or equipment modifications. Modifications to the hydraulic system are described in section 6 of this report. Signal drift due to frequent variations in vital bus power should be eliminated in the near future with the installation of a static inverter.

The major remaining problem is the unsatisfactory response of the inlet valves on automatic. The system is presently being redesigned. The inlet valve pneumatic control drive will be replaced with an electric drive. Turbine first stage pressure will be used as a feedback signal to the drive and to the dump valve in place of the present inlet valve position feedback. The drive will receive a pulse type input signal which should allow stable control in spite of the deadband and time lag in the valve mechanism.

The majority of plant operation and testing has been performed with the dump valve controlling reactor pressure. During turbine operation the inlet valves are manually positioned using the load limit control. This "split flow" mode of operation has presented no difficulty. Response of the system during Tests 433 (feedwater temperature and flow changes, pressure setpoint changes, power changes, etc.) and 431 (scram, load dump, runback, etc.) has been satisfactory.

C. Change of Runback Signals to Controlled Shutdown

The controlled shutdown (CSD) contact chain was modified to change all runback signals except those listed below to controlled shutdowns (i.e. the control rods will be fully inserted even if the initiating signal clears.) The signals that will remain runbacks are:

1. LRBN - 110% (110% level indications on the power channels initiate runback signals into 2 of 4 logic. 115% level indications initiate scram signals into 2 of 4 logic.

2. K206 - Channel 4 (Log N) Monitor

3. K202 - Channel 4 (Log N) Short Period.

This change was made to avoid the possibility of adverse flux shapes due to rod positions resulting from runback (the control rods will stop when the initiating signal clears.) until these rod patterns can be evaluated under controlled conditions.

Refer to Figure 19 for a schematic description of the change.

D. Recirculation Pump Trip and Low Recirculation Loop Flow Scram Circuitry

As mentioned in a previous section (III B) of this report, operational testing associated with the recirculation pump trip tests resulted in the addition of scram action on the loss of any one of the three recirculation pumps. Since it may be postulated that a loss of recirculation flow may occur without tripping the pump breaker, a low recirculation loop flow scram was also added. The addition of these scram actions is considered to be temporary and further testing results, or technical specification changes, may support the removal of these scram actions.

The recirculation flow-to-power scram protection scheme is not affected by these system additions.

Refer to Figures 20 and 21 for a schematic description of scram system additions. Note that the manual bypass key circuit for the recirculation flow-to-power scram contacts also bypasses the added low recirculation flow scram contacts.

E. Loss of Steam Flow Protection

As previously described in Section III B of this report, a "Loss of Steam Flow" scram protection system has been added to serve as backup protection against possible improper pressure control system operation. The circuitry will scram the reactor on sudden large decreases in steam flow and is presently set to initiate a scram if the steam flow drops by more than one third of the existing flow at any time. The circuitry compares the existing steam flow signal with a delayed and attenuated steam flow signal; if the latter equals or exceeds the former, a scram will occur.

Since the circuitry may put out an unnecessary scram signal at low steam flows, a manual bypass key switch is provided to bypass both (redundant) circuits simultaneously for startup. The existing power to steam flow key switch (No. 324) will provide the ability to bypass one side at a time for maintenance or testing.

See Figure 22 for a schematic description of how the safety system additions have been accomplished.

F. Emergency Condenser Operation

Several changes have been completed on the control systems associated with the emergency condenser operation.

1. The limit switches on the emergency condenser outlet valves, CV-1078 and CV-1578, have been rewired to activate follower relays 1078x and 1578x which cause a scram (redundant) when the valves are open. Contacts of the follower relays are located in the redundant process scram chains. The relays also initiate an annunciator and valve "open" light.

2. The pushbutton (PB56 "OPEN") which opens valves CV-1078 and CV-1578 has been disabled and replaced with additional contacts on the isolation scram pushbutton (PB77).

3. The emergency condenser control circuitry has also been changed to initiate the emergency condenser if an isolation scram is received while in the flood mode.

4. Switching of the emergency condenser action from a low steam flow setpoint of 40,000 lb/hr to a low steam flow setpoint of 19,500 lb/hr was formerly a function of the "closed" contacts on the MSIV CV-1033. This transfer is now a function of the "open" contacts on the MSBV CV-1034. The setpoint transfer will now occur when the bypass valve is moved from the fully opened position. (See Sections VI-H-2 and VI-H-3).

5. On occasion, we have performed maintenance on steam flow meters which supply signals into the power-to-flow protection circuitry. The key bypass (No. 324) which enables us to bypass the contacts in the power-to-flow scheme does not bypass the low steam flow contacts in the emergency condenser circuitry. An additional key bypass (No. 331) has been added to perform this required function.

6. The location of the Isolation Scram Reset Button has been changed to the Vertical Panel B to allow for the installation of the key bypass (No. 331) described above.

   See Figure 23.

G. Modifications to the Startup Channels

The startup channel circuitry modifications were completed to accomplish basically two functions; manual resetting of the low count rate trip and automatic bypass of the short period scram when the HV is turned off when operating in the overlap range. (Startup channel and intermediate channel overlap.)

The modifications do accomplish the following purposes:

1. Manual resetting of the low count rate trip will be required.

2. If the HV of any channel is turned off when the reactor power level is below the overlap range a scram signal is automatically put into the 2 of 3 short period logic.

3. If the HV is turned when the power is above the overlap range,

   a. The scram signal into the 2 of 3 short period logic is automatically bypassed.

   b. A high count rate signal is automatically put into the 2 of 3 logic associated with the overlap interlocks.

4. Inserting a test signal via the trip test switch will remove the scram bypass and will insert a scram signal into the 2 of 3 logic.

5. If the function switch is turned off of the OPERATE position,

   a. A scram signal is automatically put into the 2 of 3 short period logic, and

   b. A high rate signal is automatically put into the 2 of 3 logic associated with the overlap interlocks, and

   c. A low count rate signal is automatically inserted into the 2 of 3 logic associated with the rod withdrawal prohibit circuitry.

H. Additional Changes to the Safety System Circuitry

1. Automatic Closure of the MSIV on Low Reactor Pressure

To prevent excessive cooldown rates following a scram from powers greater than 80 MWT while on manual pressure control, a circuit has been added to automatically shut the MSIV when either low pressure scram switch is operated. On startups, this circuit will be defeated until the low pressure scram bypass resets on increasing pressure. If a scram occurs before the low pressure scram is reset, manual action is required to close the MSIV.

2. Prevention of Closing the MSBV when the MSIV is Open

As described above (5), the MSIV will automatically close when the reactor pressure drops below the low pressure scram setpoint. To prevent the complete loss of steam flow when this occurs, an interlock has been added to prevent the closing of the bypass valve when the MSIV is not closed. After the MSIV is closed, it will be necessary to push the reset button to position the bypass valve. On an isolation scram, the MSBV will begin closing when the MSIV begins to close.

3. Power Actuated Safety Valve Operation

The power actuated safety valve was formerly closed or prevented from opening when the MSIV was fully closed. This function will now be performed by the fully open limit switch of the MSBV. Furthermore, a one minute time delay has been added to this action after the MSBV begins to close. See Figure 24.

4. Automatic Reactor Water Level Control

The reactor level controls are designed to operate as a "3 - element" system in which a signal proportional to the difference between feedwater flow and steam plus purification flow is used to bias the level setpoint. Under optimum conditions, the feedwater and steam flow meters will come on scale at about 10% of full range and will not give fully reliable indication below 20% of full range. Therefore, to operate on auto level control below 20% flow it is necessary to revert to single element (level signal is the only input) control.

A toggle switch has been added to the circuitry to allow the selection of either single element or three element control. The selector switch is located on console C.

5. Steam Line Draining Circuitry

It is necessary to drain the condensate from the main steam line while operating with the Bypass Valve (CV-1034) open. The existing control system did not permit this because the CV-1034 "open" contacts prevented the energizing of the drain valve (CV-1040) solenoid. A relocation of the CV-1034 "open" contacts allowed the draining of the steam line while maintaining the origin function of the contacts.

I. Technical Specification Change

Technical Specification Change Requests No. 12 and No. 13, dated August 16, 1966, and August 18, 1966, requested permission for

(1) reactor cooling rates in excess of 200°F/hr during the power escalation program up to full power, and

(2) use of the reactor noise analysis technique in lieu of the oscillator rod technique to evaluate the stability of the Pathfinder reactor.

These changes had been requested to allow certain tests which could result in rapid cooldown during the startup program and to minimize disturbances of the superheater thermocouple leads which would occur as a result of the installation of the oscillator rod into the boiler core.

These change requests were designated as Proposed Change No. 12 and approved by the Division of Reactor Licensing. The Technical Specifications were changed as follows:

1. Change Section 8.0 to read:

"During testing of the emergency condenser and during the start-up program, and with the superheater drained, the controlled cooling rate of the reactor may exceed 200°F per hour under the following conditions:

(a) A temperature reduction not to exceed 50°F may occur at higher rates during any half-hour period in which the overall temperature variation does not exceed 100°F, or

(b) During performance testing of the emergency condenser cooling at higher rates may occur during a single 20 minute period occurring within any given hour. The cooling shall not exceed 80°F during this 20 minute period, and the number of such tests to be performed under this temporary authorization shall not exceed five."

2. Change Section 7.5.3.6 to read:

"7.5.3.6 Stability Evaluation

Transfer function measurements shall be made at zero power (less than 5 MW), and subsequently may be made at higher power levels, to evaluate reactor stability. An oscillator rod, if used, shall be calibrated at several angular positions to assure that peak-to-peak worth does not exceed 10 cents in the configuration used for transfer function measurements. The range of investigation may extend from 0.01 to about 12 cycles per second. Evaluation of reactor stability shall be made at initial power escalation steps (as specified in 7.5.3 above), before proceeding to the higher power steps. Such evaluation and extrapolation of stability shall be based on analysis of reactor power noise (noise transfer function) or analysis of transfer function measurements made with the oscillator rod mechanism."

J. Non-Compliance Item Concerning the Use of a Safety System Bypass

An item of non-compliance concerning the use of a safety system bypass was cited on the Form AEC-592 dated 8-22-66. The NSP reply to this citation stated:

To minimize the possibility that safety circuits are inadvertently rendered inoperative in the future, we are instituting the following two actions:

1. A refresher course on the Technical Specifications section relating to the use of bypasses is being conducted for Pathfinder shift supervisors and members of the plant supervisory staff to assure that responsible plant personnel are familiar with the intent and specific limitations of the sections.

2. In addition to Operations Committee approval of the general methods and intended use of bypasses, it shall henceforth be required that shift supervisors receive written authority from the Operation Supervisor for each specific use of the bypass. Such written authorization shall reference the related Technical Specification authority, the purpose of the bypass, and shall clearly define the limitations of the use of the bypass with respect to reactor conditions.

The two actions stated above have been fully complied with.

VII. Major System and Equipment Performance

A. Primary System

In June of the report period the Main Steam Line was cleaned in an attempt to eliminate the collection of corrosion scale deposits on the seats of the Main Steam Isolation Valve, the Main Steam Bypass Valve, Dump Valve, and the Turbine Inlet Stop Valves. The above valves were disassembled, cleaned, and their seating surfaces refinished at this time. It was necessary to replace the seat in the Main Steam Bypass valve; the new seat was seal welded to the valve body.

A crack developed in the bellows of the safety valve discharge line expansion joint in September. The crack was temporarily repaired by welding and a new bellows installed in November. Additional pipe supports were installed to prevent a re-occurrence of this failure. A cracked seal bellows on No. 12 Reactor Safety Valve was also replaced in November of this period.

The lower shaft seal bushings on No. 12 and 13 Reactor Recirculation Pumps were replaced in July. These bushings had become worn to the point where excessive seal water flow was required.

B. MSIV and MSBIV Leakage

The MSIV and MSBIV have caused problems in maintaining the containment leakage rate within acceptable limits. Prior to this reporting period, the valve leakage rates were acceptable. The valves had been apart, cleaned and retested in April 1966. Foreign material (believed to be welding slag) was found on the seating surfaces of the MSIV and the MSBIV seating surfaces were damaged. New parts were ordered for the MSBIV. The combined leakage at this time was 8.6 SCFH at 80 psig and 8 psig on the reactor building. This leakage was acceptable.

Test 277.2A was completed on May 25, 1966, and the steam valves were closed after a 16 hour period of 25,000 lb/hr steam flow through the MSBIV. The valves did not leak in the "as closed" condition as determined by leakage testing. The valves were operated a number of times, retested and found to leak. The MSIV was cleaned and retested and the leakage reduced to an acceptable value. A new plug and seat were installed in the MSBIV. A leakage test on the MSBIV determined that the valve was now leak tight. Concurrently, the secondary steam isolation valves (a parallel combination of 10 valves including the steam dump valve, turbine stop valves, and the steam supply to the high pressure feedwater heater) were tested and repaired so that the steam line leakage would be acceptable even if the MSIV did leak excessively.

Reactor testing continued following test procedure 278.1A. During this test, the MSIV was opened for the first time during steam flow conditions, and at flows of up to 150,000 lb/hr at 477°F and 540 psig. The valves were leak tested in the "as closed" condition on

June 25, 1966. The combined leakage was excessive, and the valves were retested after a number of operations, and the leakage remained excessive. The secondary steam valves were retested and turbine stop valves were "lapped in" to reduce the valve leakage to zero. At this time, the steam line leakage would be acceptable even if the MSIV and the MSBIV did not close to isolate the containment. The MSBIV was repaired by welding the seat to the body of the valve, and lapping the seating surfaces. These repairs reduced the valve leakage to zero. The MSIV seating surfaces were cleaned and retested. Leakage tests now showed the valve leakage rates to be acceptable.

The second phase of 278.1A was completed on July 30, 1966, and the valves were tested in the "as closed" condition. No leakage could be detected. The valves were cycled and the MSIV was found to leak excessively. The MSIV was repaired by cleaning the seating surfaces and retested. The combined leakage after repair was acceptable during leakage testing on August 2, 1966.

The valves were tested on September 1, 1966, after steaming operations were concluded on August 31, 1966. The leakage was zero with the valves in the "as closed" position. The MSIV remained closed until September 28, 1966, when the valves were again tested. The MSIV leakage was unacceptable and the valve was repaired by cleaning. The valve was retested on September 30 and the leakage was acceptable.

Reactor operation was resumed and operations continued to November 1966. The MSIV and MSBIV leakage was measured on November 19, 1966, and found to be acceptable. The valves were tested in the "as closed" condition and were not opened.

The MSIV and MSBIV leakage problem appears to diminish as reactor operation continues. A number of problems have been found that contribute to the MSIV leakage. Steam line travel resulting from building pressure caused distortion in the MSIV body allowing leakage. This distortion has been minimized by limiting the steam line travel to only the necessary amount. Foreign material was found on the MSIV seating surfaces and appeared to have cut and damaged the MSBIV. The steam line has been cleaned as much as possible by wire brushing and flushing. The foreign material that escaped the cleaning appears to have been "swept" out during steaming operation. The leakage monitoring program will continue to assure that containment leakage rate limits are met.

C. <u>Control Rod Drive Maintenance</u>

The control rod drive (CRD) operations have been very satisfactory during the last six months despite the operational requirements for the reactor testing phase. All control rod drives have responded properly to the safety system or control system demands on every occasion.

Three situations of concern did occur during this six month period.

1. Unusual Occurrence Summary

This summary concerns the lifting of CRD No. 10 from the vessel head when the reactor vessel was pressurized to establish purification flow.

On September 2, 1966, all control rods were unlatched in preparation for the removal of Control Rod Drive No. 3 from the vessel head. CRD No. 3 was being removed for an examination of the dashpot.

On September 9, 1966, the reactor was being prepared for water purification flow. The shutdown pump was unavailable, so it was necessary to pressurize the reactor to approximately 50 - 100 psig with the condensate pump to establish the purification flow. A spare CRD had been installed on the head in place of CRD No. 3. When the reactor was pressurized, CRD No. 10 was raised approximately three feet from the mounting nozzle, as observed by two shift personnel. The reactor was immediately depressurized and CRD No. 10, still in its mounting nozzle, slipped downward and came to rest in a slightly tipped position approximately two feet above the normal position. The drive was lifted from this position and inspected for damage. There was no visible serious damage, so the drive was replaced on the vessel head and the quick disconnect tightened. An unusual occurrence investigation was started immediately.

The quick disconnect coupling for CRD No. 10 had been decoupled when CRD No. 3 was removed from the head. Shift personnel had anticipated that CRD No. 10 would also be removed at some later date and had deliberately disconnected the coupling. This event was not logged and the Shift Supervisor on duty was only informed of the action taken with CRD No. 3. The only personnel aware of the condition of CRD No. 10 were the two men involved with the actual decoupling operation.

Control Rod Drive No. 10 was removed from the reactor vessel on September 9th and examined on September 10th. Scratches were observed on the CRD bushing nozzle, but subsequent examinations with the drive disassembled showed that no harmful effects resulted from the incident. The drive was reassembled and replaced on the vessel head.

While the drive was off of the head, the head mounting nozzle for CRD No. 10 was examined using the boroscope. After an examination of the photographs taken with the aid of the boroscope, A-C requested that a visual examination of the mounting nozzle be made possible. While the vessel head was removed to allow insertion of SbBe sources, the head was raised to the surface of the pool and a visual examination of the nozzle was completed. The scratches on the nozzle are not serious and do not effect the integrity of the nozzle.

An Unusual Occurrence Report is being prepared.

Shortly after the incident occurred, A-C was contacted for advice concerning the CRD and mounting nozzle examinations. NSP personnel in Minneapolis were notified and after considerable discussion, we concluded that although this type of occurrence may have serious consequences, this particular incident did not represent a nuclear safety problem because all the control rods were unlatched. Therefore, it was decided that a written report to the AEC was not required.

On September 12, Mr John Flora was advised of the incident and our decision with respect to the AEC reporting.

2. Failures of Control Rod Drive Dashpot Ram Piston Rings

During the week of August 31st, all control rod drives were tested to determine if the CRD dashpots were giving proper dashpot action. Each control rod was scrammed from a twelve inch position and the position indicating pointer action was observed when the rod reached the zero inch position. If the CRD being tested has proper dashpot, the indicator pointer will slow down smoothly as the zero inch position is approached. If the dashpot is not acting properly or if the dashpot ram is stuck in the fully depressed position the control rod will bottom in the fuel element quad box and the indicator pointer will "oscillate" from the rebounding action of the selsyn indicator system.

Prior to the testing on August 31st, it was considered that CRD No. 10 was the CRD of greatest concern; however, the testing results showed that CRD No. 3 had less dashpot action. After viewing the control rod scram tests it was decided to remove CRD No. 3 from the vessel head and disassemble the drive to determine the cause of the weak dashpot action. The removal of CRD No. 10 was delayed until the maintenance was completed on CRD No. 3. (See above summary of Unusual Occurrence.)

Control Rod Drives No. 3 and No. 10 were discovered to have broken dashpot ram piston rings. The dashpot action is caused primarily by orifice effects and the dashpot ram piston ring does have an effect on the orificing action.

The circumstances of the piston ring malfunctions were reviewed and Allis-Chalmers performed an analysis to determine the possible effects on both the control rod drive and the control rod. Conclusions of the analysis, extracted from the A-C report, "Control Rod Dashpot Analysis for Malfunction of the Piston Ring" dated November 15, 1966 written to J F Haines by H C Gignilliat are:

a. The results of the analysis indicate that the maximum expected velocity of the control rod for the postulated loss of sealing is approximately 47 inches/second at impact energy for complete loss of dashpot action when the control rod strikes the bottom of the quad box at a velocity of 120 in/second.

b. It is recommended that the action of the drive be observed on the position indicator when the drives are scrammed from the 6 to 8 inch position during the normal backstop clutch check prior to each reactor startup. Malfunction of the dashpot piston ring can be determined during this check. Repeated subjection of the control rod and control rod drive to the forces resulting from a piston ring malfunction will not cause damage to either.

Dashpot actions on all control rod drives have been observed on several planned testing occasions. No further dashpot piston ring failures have been observed.

3. Slow Insertion Time (Run-In) of Control Rod Drive No. 5

Control rod drive No. 5 has been observed to have a slow insertion time compared to other CRDs in the same rod drive group. During control rod drive testing, this condition has been reproduced; however, the condition generally exists only if the test includes a gang lower of the control rod group, but a slow insertion time is not always observed on a gang lower operation. All other control rod drive functions are normal including the rod scram times.

Efforts have been made to determine the cause of the unusual behavior, but the cause has not yet been discovered. Testing of the CRD is continuing on a routine basis and until further problems become evident, the CRD is considered as operational.

D. Nuclear Instrumentation

The nuclear instrumentation has operated satisfactorily during this reporting period. Periodic checks were performed as required and a detailed calibration was completed. The instrumentation has been stable and has needed only minor repairs or adjustments. Startup channel noise was measured and found to be less than 1 cps on each channel.

The startup channel interlocks have been modified to provide some additional features. A manual reset of the low count rate trip was installed to eliminate relay noise. A scram bypass was installed that will bypass startup channel period scrams (from noise) when operating above the startup range. The modifications were installed consistent with amendment 27 criteria. See Section VI-G.

E. Off-Gas System Difficulties

During initial reactor testing at 6 MWT, hydrogen levels in the off-gas system would approach approximately 3% requiring controlled condenser leakage to limit the hydrogen concentrations. The failure of the recombiner catalyst to do its job was determined to be caused by lack of sufficient operating temperature resulting in excess moisture in the recombiner.

The recombiner catalyst was cleaned by the manufacturer prior to power testing above 6 MWT and the efficiency of the recombiner was estimated to be approximately 75%. During the required shutdown, the recombiner casing was insulated and a temporary steam heating coil was added to the inlet of the recombiner.

The elevated temperatures of the off-gas increased the effectiveness of the recombiner during 40 MWT testing, but controlled condenser leakage was still required during operations. As the reactor power level was increased above 40 MWT, the efficiency of the recombiner increased and the hydrogen problem no longer exists.

Downstream off-gas system moisture problems and off-gas flow surges have been creating difficulties with the first absolute filter. The large off-gas flow occurs when pulling condenser vacuum or following the transfer of auxiliaries after reactor shutdown.

The off-gas flow surges have also caused operational difficulties with the loop seals on the air ejector and the off-gas delay pipe. The after-condenser loop seal has been lengthened to lessen the chances of "blowing" and a valve has been installed in the delay pipe loop seal for throttling purposes.

Additional system changes included the installation of a permanent preheater in the off-gas line prior to the recombiner and the addition of temporary cooling coils in the off-gas delay pipe. Further system modifications are expected.

F. Radioactive Waste Handling

A metered raw water dilution line was installed (December). This line provides raw water for diluting radioactive waste discharge to the river. It has been necessary in the past to use treated cooling tower makeup water for dilution.

Blisters and cracks developed in the plastic lining of the reactor sump. The sump was therefore relined.

The stack gas sample lines were relocated and modified to provide a more representative sample and prevent sample plate out.

G. Dump Valve Hydraulic Supply System

Problems with the dump valve hydraulic supply system have included leaks and pipe failures, vibration, dirt in the system, and insufficient cooling. Revisions to the system include replacement of all sharp piping bends with flexible hose sections, and removal of a check valve between the unloader and the accumulator. Additional bracing and piping supports were added to reduce vibration. A centrifugal filter was installed to aid in maintaining oil cleanliness. The cooling problems were resolved by changing the oil cooler supply to well water.

H. Area Monitoring System

Modification of the area monitoring system is being planned to improve system reliability. Some of the major difficulties with the present system are: frequent diode and meter burnout, frequent horn failure, and problems with the contacting meters (such as breakage of the foil which serves as the electrical conductor to the pointer, and corrosion of the contacts causing unreliable operation). The system is being redesigned to provide simpler and more reliable operation.

I. Plant Makeup Water Systems

The Cooling Tower Makeup Pumps and the Domestic Well Pump were disassembled and overhauled. The lime sludge basins were cleaned and enlarged. Corrosion scale and sludge buildup in the cooling water system continues to be a problem. The water side of the turbine oil coolers and the air compressor heat exchangers were cleaned during this report period.

J. Summary

The preventive maintenance program was continued on all system components to provide reliable operation. Only the significant operational difficulties have been reported in this section and unless otherwise reported, all major systems and system equipment have operated satisfactory.

VIII. Containment Leakage Testing

Two "class C" containment leakage tests were performed during the report period. A Class "C" test is an individual valve leakage measurement of specific valves listed in the tech specs. A total of three "class C" tests have been completed since the last "class A" or containment full pressure integral leakage rate test. The "class C" tests during this reporting period were completed on June 14, 1966, and November 16, 1966. The tests were completed successfully and did not require any maintenance to meet the tech spec acceptance criteria.

Class C Measured Leakages	June	November
1. Vacuum Breaker at 78 psig	0 SCFH	0 SCFH
2. Sump Isolation Valve at 78	0	0
3. Heating System Return at 78	0	0
4. Vacuum Drain Tank Valves at 78	0	0
5. RCP Gland seal water return at 40	0	0
6. Fuel Transfer Valve at 78	2	3
7. Shield Pool cooling water return at 78	5	2.4
8. Ventilation valves (each) at 78	.5	.5
9. Ventilation cooling water return at 78	1	1

IX. Plant Organization

A. General

The plant technical and supervisory staff remained essentially as reported in the Phase II Report NSP-6602, with the exception of the resignation of J Y Lee, the plant chemist. The Plant Organization is shown on Figure 25.

B. Plant Personnel Changes

1. Mr J Y Lee, Chemist, resigned on June 30, 1966.
2. Mr J Funke was hired as a laborer and started work on July 11, 1966.
3. Mr Einar Swanson, who had resigned on May 1, (See Phase II Report), left the employ of NSP on July 15, 1966.
4. Mr Garry Neils, who had been promoted to Nuclear Plant Supervisory Engineer (See Phase II Report) transferred to NSP - Minneapolis on September 1st.
5. Mr Arne Hunstad, Engineer, joined the Results Crew on September 6, 1966. Mr Hunstad previously worked at the NSP Riverside power station.
6. Mr Orville Todd, Machinist, was transferred to NSP - Minneapolis on November 20th.

C. Operator Licensing

No additional operator licenses were obtained during this reporting period. The licensed reactor operator status is:

Senior Reactor Operators - 12

C E Larson, M H Clarity, M N Bjeldanes, R T McKaughan, W A Sparrow, R A Mielke, J B Brokaw, W E Anderson, S L Pearson, H Seibel, R D Emerson, L W Severson.

Reactor Operators - 8

D L Magill, D E Severson, F J Schober, D W Cragoe, L V Triebwasser, E W Kruse, M J Balk, R S Holthe.

Senior Reactor Operators available but not on the Plant Staff

A E Swanson, G H Neils

D. Summary

NSP management has thoroughly reviewed each of the above changes with respect to Pathfinder requirements and find that the organization, as it exists on December 1, provides ample continuity and technical qualifications necessary for continued safe plant operation.

APPENDIX A

Pathfinder Scram History - May 19, 1966 to November 19, 1966

Shutdown No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
1	1811 5-20-66	Runback - Low FW Temp; Scram - Emerg Cond Operation	The runback occurred when reducing power due to high H_2 content in off-gas.	6 MWT
2	0951 5-21-66	Scram - Emerg Cond Operation	Reducing power due to high H_2 content in off-gas. Emerg Cond was unknowingly cocked on one channel.	8 MWT
3	1942 5-21-66	Scram - Steam flow/ power	Dump valve was closed from 8% to 6% open	8 MWT
4	1731 5-23-66	Turned Sw. 159 to Flood	Test 277.2A - Planned Scram	8 MWT
5	1758 5-23-66	Scram - Simul Valve Closure	Dump valve drifted closed to reach interlock setpoint.	0
6	1952 5-23-66	Scram - Steam flow/ power	Power was being held constant with steam flow at 25,000 lbs/hr. "A" flowmeter seemed unstable and spiked downward.	8 MWT
7	0759 5-24-66	Scram - Water in main steam line	Faulty operation of main steam magnetrol	200 KWT
8	1345 5-24-66	Scram - Steam flow/ power	Auto level control was being checked out. Steam flow was too close to the trip point.	8 MWT

Shutdown No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
9	0250 5-26-66	Scram - Steam flow/ power	Trip occurred at an indicated 23,000 lbs/hr flow. Setpoint is supposed to be 19,500 lbs/hr.	8 MWT
10	0354 5-26-66	Scram - Steam flow/ power	Operator changed range on Ch 5 before proper steam flow was established.	2 MWT
11	2228 6-6-66	Scram - Short Period Ch 3	Master Control Sw. introduced noise into startup channels.	0
12	1748 6-15-66	Scram - High level Ch 5	Low hotwell level caused loss of seal flow to recirc pumps. Recirc pump trips introduced a noise spike into Ch 5.	0
13	2154 6-15-66	Scram - Ch 3 Short Period	Master control switch caused noise on startup channels.	0
14	2317 6-15-66	Scram - Ch 3 Short Period	Master control switch caused noise on startup channels.	0
15	1445 6-16-66	Scram - Steam flow/ power	Had just established steam flow. Flow on one channel dropped down to setpoint.	200 KWT
16	2004 6-16-66	Runback - Low FW Temp Manual Scram	Control to No. 14 htr leveltrol was in bleed position. Error on part of operator who completed pre-start check-list on this system. Manual scram required when outer rods ran in more than 5".	6 MWT

*down No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
17	1635 6-17-66	Runback - Low Reactor Water Level	When the steam flow was increased to ~ 60,000 lbs/hr, the water level varied enough to cause runback. (-11")	7 MWT
18	1913 6-17-66	Scram - Simultaneous Valve Closure	For no apparent reason, the dump valve drifted closed.	6 MWT
19	2346 6-17-66	Scram - Steam Flow/ Power Ratio	Low steam flow resulted due to loss of pressure and temp while raising power from 7 MWT to 17 MWT.	15 MWT
20	1633 6-18-66	Scram - Steam Flow/ Power Ratio	Erratic operation of flow meter when opening the MISV	17 MWT
21	2154 6-18-66	Scram - High Reactor Water Level	When the steam flow was increased, the water level varied enough to cause scram. (+4")	19 MWT
22	0224 6-20-66	Scram - Startup Channel Short Period	Noise on Ch No. 2 gave scram even though the HV was off. Ch No. 3 was in "DWELL".	16 MWT
23	1115 6-21-66	Scram - High Steam Temperature	Spurious signal gave scram when technician was working on the Offner recorder.	17 MWT
24	2210 6-21-66	Scram - Manual Trip	While pressurizing the reactor, a decrease in seal flow to the recirc pumps caused one pump to trip.	17 MWT

...own ...o.	Time and Date	Initiating Action	Explanation of Cause	Power Level
25	2113 6-22-66	Scram - Emergency Condenser Operation	The action was caused while working on main steam flow trans. 1575. Bypass key action not thoroughly understood.	17 MWT
26	1555 6-23-66	Scram - Opening of No. 12 Safety Valve	Initiated by opening of the No. 12 Safety Valve. (Indicated)	17 MWT
27	1750 6-24-66	Scram - Manual Trip - Isolation Scram	Manual trip of PB No. 77 for Isolation Scram - Test	40 MWT
28	0826 6-25-66	Runback - Low Feedwater Temperature	Low FW Temperature when starting to increase power after opening MSIV.	32 MWT
29	1118 6-25-66	Scram - Startup Channel Short Period	A scram occurred prior to criticality by a noise spike on Channel No. 2	0 MWT
30	0018 6-26-66	Scram - Steam Line High Pressure	Pressure control system transient when applying temperature anticipatory signal.	40 MWT
31	0355 6-26-66	Scram - Steam Flow/ Power Ratio	Lost steam flow signal when raising power to 7 MWT.	2 MWT
32	1630 6-26-66	Scram - Water in Steam Line	Normal steam line draining operation apparently vibrated magnetrol which gives the water in steam line signal.	40 MWT

down No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
33	0738 7-23-66	Scram - Startup Channel Short Period	A scram occurred prior to criticality when draining the steam line. Channel No. 3 short period spike.	0 MWT
34	1008 7-23-66	Scram - Channel No. 6 High Level	A scram occurred while subcritical from a noise spike on Channel No. 6.	0 MWT
35	1245 7-23-66	Scram - Emergency Condenser Operation	Emergency condenser came into operation at about 10,000 lbs/hr flow. Faulty steam flow transmitter gave false flow signal.	2 MWT
36	1502 7-23-66	Scram - Emergency Condenser Operation	At a low steam flow, emergency condenser operated due to faulty flow transmitter.	2 MWT
37	1353 7-24-66	Scram - High Steam Temperature	While raising power with pressure control on HAND, steam flow dropped giving high steam temperature.	40 MWT
38	2345 7-24-66	Scram - High Steam Line Pressure	While raising the reactor pressure to clear the low pressure reset, scram occurred at an indicated pressure of 545 psig. The high pressure scram was set too low.	42 MWT
39	0230 7-25-66	Scram - Steam Flow/ Power Ratio	Operator turned range select switch too high prior to initiating steam flow.	200 KWT
40	0113 7-27-66	Scram - Channel No. 6 High Level	Channel No. 6 noise spike was caused when releasing startup chennel detector withdrawal button.	20 KWT

down No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
41	0919 7-27-35	Scram - Startup Channel Short Period	Startup channel high voltage was turned off and interlock requirements were not understood by operators.	20 MWT
42	1720 7-27-66	Scram - Water in Steam Line	The Water in Steam Line magnetrol was apparently vibrated during routine operations.	40 MWT
43	2257 7-27-66	Scram - Emergency Condenser Operation	Erroneous steam flow signal caused by faulty steam flow transmitter caused the mergency condenser operation.	2 MWT
44	0403 7-28-66	Scram - Water in Steam Line (?)	While transferring the pressure control system to auto, the dump valve "chattered" and apparently vibrated the "Water in Steam Line" magnetrol.	33 MWT
45	1432 7-28-66	Scram - Emergency Condenser Operation	Faulty flow signal tripped the emergency condenser when the steam flow was ~ 12,000 lbs/hr.	2 MWT
46	2158 7-28-66	Scram - Emergency Condenser Operation	Faulty flow signal tripped the emergency condenser.	2 MW
47	0236 7-29-66	Scram - Steam Flow/ Power Ratio	The steam flow indication dropped unexpectedly to give power to flow scram.	17 MWT
48	2247 8-4-66	Scram - Steam Flow/ Power Ratio	Steam flow dropped to give power to flow scram.	28 MWT

S[illegible]down No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
49	0319 8-5-66	Scram - Startup Channel Short Period	Noise on the startup channels gave short period scram. High voltage on channels was off.	18 MWT
50	1350 8-6-66	Scram - Reactor Pressure Control	While working on the pressure control system, pressure signal was removed causing the dump valve to open to ~ 20% giving reactor low pressure.	40 MWT
51	0537 8-7-66	Scram - Reactor Pressure Low	During an attempted shutdown, the dump valve opened abruptly while closing the inlet valves resulting in a low reactor pressure.	40 MWT
52	0149 8-10-66	Scram - Turbine Building Vent High Radiation	The turbine building vent monitor setpoint was not set for increased power operation. Off-gas system leakage was the cause of the activity level.	86 MWT
53	1952 8-10-66	Scram - Reactor Pressure Low	A reactor startup was started with high reactor temperature and pressure. When reducing the reactor temperature for normal startup, a low pressure scram occurred.	200 KW
54	1603 8-11-66	Scram - High Steam Temperature	During a pressure control system setpoint change (test), a high steam setpoint was reached.	76 MWT
55	1949 8-11-66	Scram - Startup Channel Short Period	A noise spike on the startup channels caused a short period scram. The high voltage to the channels was off.	17 MWT
56	1431 8-12-66	Scram - Recirc Flow/ Power Ratio	During Test 433 testing, while reducing the recirc flow, a scram occurred.	59 MWT

Shutdown No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
57	1253 8-13-66	Scram - Stop Valve Trip	Operator mistakedly operated Main Bank No. I Deluge Valves.	78 MWT
58	1650 8-13-66	Scram - Steam Flow/ Power Ratio	The dump valve started to drift closed reducing the steam flow.	8 MWT
59	1330 8-15-66	Scram - Feedwater Flow/Steam Flow Ratio	Faulty flow meter indication gave false flow ratio.	80 MWT
60	1133 8-16-66	Scram - High Level Channel 8	The high level setpoint on Channel 8 was reached when bumping control rods.	75 MWT
61	1331 8-16-66	Scram - Startup Channel Short Period	While withdrawing the startup channel detectors, a noise spike caused a short period scram on the startup channels.	50 KWT
62	1659 8-15-66	Scram - High Steam Temperature	When reducing the steam flow to No. 14 heater, the steam temperature setpoint was reached.	40 MWT
63	0703 8-18-66	Scram - Turbine Building Vent High Radiation	Apparent spike on the instrument during routine plant operations.	85 MWT
64	1910 8-18-66	Scram - High Level Channel No. 5	Power increase after a rod bump increased the power level to the scram level.	50 KWT

wn o.	Time and Date	Initiating Action	Explanation of Cause	Power Level
65	0556 8-20-66	Scram - Water in Steam Line	While draining the steam line, the water in Steam line magnetrol was apparently vibrated.	0 MWT
66	0705 8-20-66	Scram - Water in Steam Line	The Water in Steam Line magnetrol was apparently vibrated during a draining operation.	0 MWT
67	1100 8-20-66	Scram - Steam Flow/ Power Ratio	The steam flow dropped to the setpoint while increasing power.	10 MWT
68	1140 8-20-66	Scram - Steam Flow/ Power Ratio (?)	While subcritical, routine operations caused a scram of unknown origin.	0 MWT
69	1326 8-20-66	Scram - Water in Steam Line	During a steam line draining operation, the magnetrol was apparently vibrated.	200 KWT
70	2048 8-20-66	Scram - High Steam Temperature	During a stop valve trip test, the dump valve started closing causing high steam temperature.	40 MWT
71	2315 8-20-66	Scram - Startup Channel Short Period	A noise signal on the startup channels gave a short period scram. The high voltage to the channels was off.	18 MWT
72	0338 8-21-66	Scram - Water in Steam Line	The water in steam line magnetrol apparently vibrated while draining the steam line.	200 KWT

'own No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
73	1516 8-22-66	Scram - Stop Valve Trip	While taking the turbine off the line, a momentary dip in the stop valve oil pressure caused a scram.	72 MWT
74	1838 8-25-66	Scram - Steam Flow/ Power Ratio	Apparently adequate steam flow was not established prior to a uua range change.	2 MW
75	0554 8-27-66	Scram - Simultaneous Valve Closure	The dump valve drifted closed during a rod withdrawal to criticality.	0 MWT
76	1755 8-27-66	Scram - Stop Valve Trip (?)	During a turbine startup, a scram was caused by a stop valve trip.	40 MWT
77	0029 8-28-66	Scram - Steam Flow/ Power Ratio	A power to flow scram occurred while opening the MSIV	17 MWT
78	1632 8-30-66	Scram - High Steam Temperature	Technicians working on the Offner recorder caused dump valve motion which resulted in scram.	50 MWT
79	0201 10-7-66	Scram - Emergency Condenser Operation	The emergency condenser operated when the MSBV was opened with less than 40,000 lbs/hr flow indicated on the main steam low flow meters.	8 MWT
80	0407 10-7-66	Scram - Emergency Condenser Operation	The emergency condenser operated when the MSBV was opened to approximately 92%. (See 79)	8 MWT

utdown No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
81	1541 10-7-66	Runback - Low Feedwater Temperature	A runback was caused by low feedwater temperature. The reactor was in a subcritical condition.	0 MWT
82	1915 10-7-66	Scram - Emergency Condenser Operation	A scram resulted when the MSBV was opened with no flow indicated on meter.	2 MWT
83	0243 10-8-66	Scram - Steam Flow/ Feedwater Flow Ratio	A feedwater flowmeter spiked giving steam flow to feedwater flow difference of > 200,000 lbs/hr	32 MWT
	0619 10-8-66	Scram - Steam Flow/ Feedwater Flow Ratio	Erratic behavior of feedwater flow meter gave a flow spike.	17 MWT
85	1259 10-8-66	Scram - Main Steam Radiation High	The monitor setpoint had not been reset for increased power operation following the extended shutdown.	30 MWT
86	1842 10-9-66	Scram - Turbine Building Vent High Radiation	During a plant shutdown, a spike (one point) on the monitor occurred during the transfer of plant equipment auxiliaries.	45 MWT
87	1725 10-12-66	Scram - Emergency Condenser Operation	With a steady operating condition at ~ 25,000 lbs/hr flow, the emergency condenser came into operation.	8 MWT
88	2218 10-12-66	Scram - High Steam Temperature	Apparently the dump valve position changed, decreasing steam flow and causing high steam temperature.	30 MWT

own o.	Time and Date	Initiating Action	Explanation of Cause	Power Level
9	0523 10-14-66	Scram - Loss of Air Pressure	When resetting the turbine stop valves, a Loss of Control Air scram resulted.	40 MWT
0	1650 10-14-66	Scram - Loss of Clutch Power	Station electrician removed a fuse which resulted in a loss of clutch power.	65 MWT
1	0544 10-16-66	Scram - High Steam Line Pressure	The inlet valves were partially closed to open the dump valve, resulting in a high pressure scram.	76 MWT
2	1911 10-16-66	Scram - Emergency Condenser Operation	A main steam low flow meter indication sagged after the MSBV was opened.	17 MWT
3	1103 10-18-66	Runback - High Steam Temp Setpoints	While tripping the stop valves, the dump valve opened causing an increase in steam flow and a decrease in steam temperature.	80 MWT
	1415 10-19-66	Scram - High Steam Temperature	During dynamics testing, a recirc pump was tripped and the operator could not "follow" the steam temperatures changes.	80 MWT
	1635 10-19-66	Scram - Steam Flow/ Power Ratio	While adjusting steam flow to No. 14 heater, the steam flow dropped enough to give a scram.	10 MWT
	2319 10-19-66	Scram - High Steam Temperature	A recirc pump was tripped and the steam temperature exceeded the maximum allowable setpoint.	76 MWT

ɔwn No.	Time and Date	Initiating Action	Explanation of Cause	Power Level
97	0334 10-20-66	Scram - Steam Flow/ Power Ratio	Steam flow dropped while opening the MSIV.	17 MWT
98	0605 10-20-66	Scram - High Steam Temperature	During pressurization, the steam flow to No. 14 heater was being reduced. This caused a high temperature condition.	30 MWT
99	1940 10-20-66	Scram - High Steam Temperature	The problems with the pressure control system were being analyzed. Testing resulted in a scram.	76 MWT
00	2345 10-23-66	Scram - High Steam Temperature	An unexplained shutdown occurred after more than two days of smooth continuous operation. A high steam temperature did occur.	62 MWT
01	0440 10-24-66	Scram - Steam Flow/ Power Ratio	Steam Flow drop after the MSIV was opened caused the scram.	8 MWT
02	0806 10-24-66	Scram - Steam Flow/ Power Ratio	An unexplained shutdown occurred when the power level and steam flow were well above the protection system limits.	45 MWT
03	2201 10-31-66	Scram - Turbine Building Vent High Radiation	While transferring turbine seal supplies the seal header relief valve lifted. Reducing valve not controlling properly.	40 MWT
04	1040 11-3-66	Scram - Steam Flow/ Power	Reactor at steady state for 9 hours. No apparent decrease in steam flow. Steam flows at 67,000 lbs/hr and 64,000 lbs/hr. Trip setpoint is supposed to be 56,000.	17 MWT

down .	Time and Date	Initiating Action	Explanation of Cause	Power Level
95	2323 11-6-66	Scram - Emergency Condenser in Service	While opening MSIV the main steam flow meters became erratic.	17 MWT
96	1453 11-7-66	Scram - Loss of Recirc Flow	Work being done on Offner Recorder. Scram caused by shorting leads from No. 13 Recirc Pump flow signal.	76 MWT
97	1601 11-8-66	Scram P.B. 97	Manual scram Test 433	48 MWT

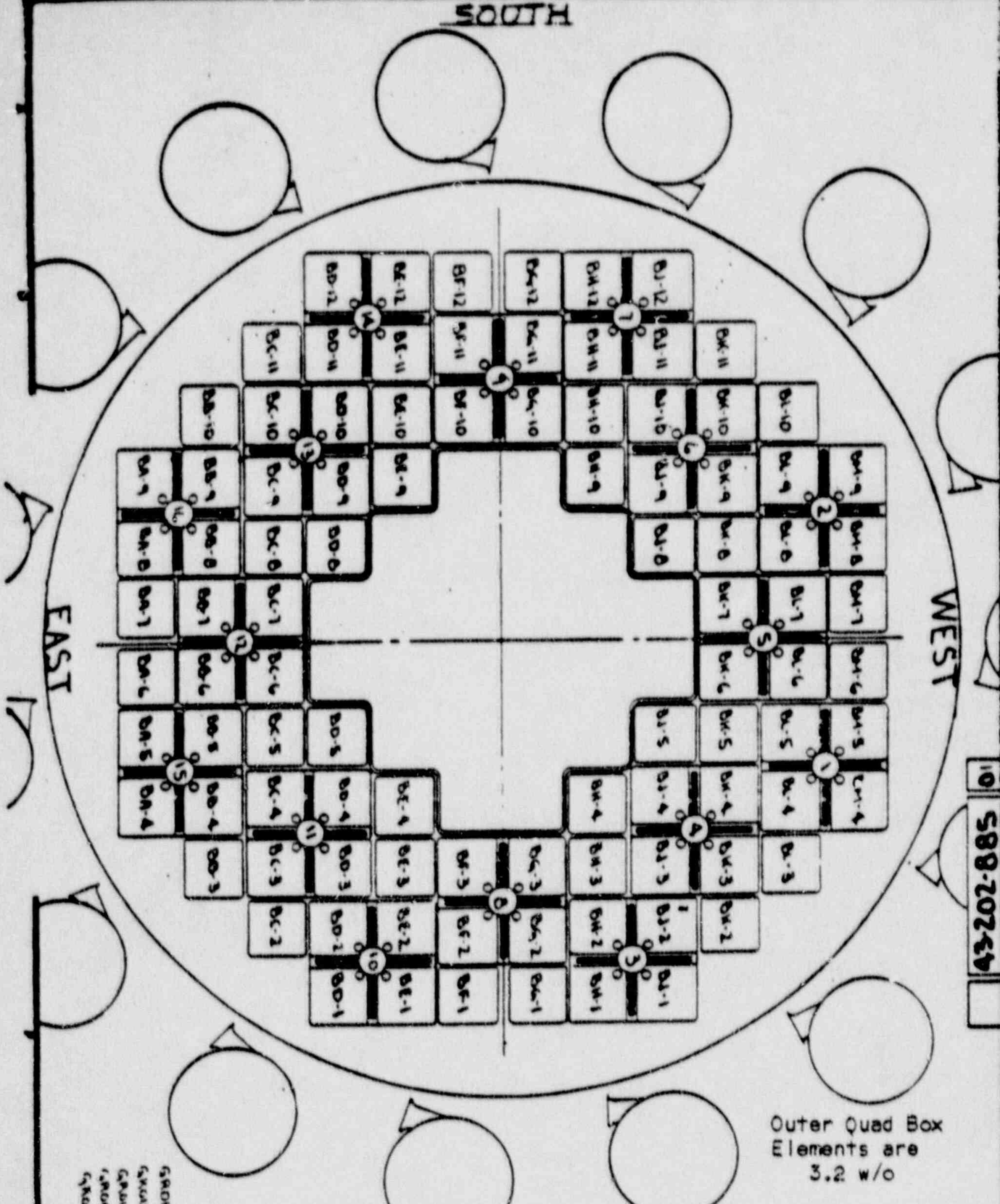


FIGURE 1

CONTROL RODS
GROUP I - 17, 18, 19, 20 (IN SUPERHEATER)
GROUP II - 5, 8, 9, 12
GROUP III - 4, 6, 11, 13
GROUP IV - 1, 7, 10, 16
GROUP V - 2, 3, 14, 15

01	10-4-63

UNLESS OTHERWISE NOTED DIMENSIONS ARE IN INCHES & MACHINING TOLERANCES ARE

— Confidential — Property of
ALLIS-CHALMERS MFG. CO., Milwaukee, Wis.

3043

DWN JW 10-4-63

CKD HEN 10-9-63

PATHFINDER BOILER CORE LOADING

SHEET 1 OF 1

43-202-885

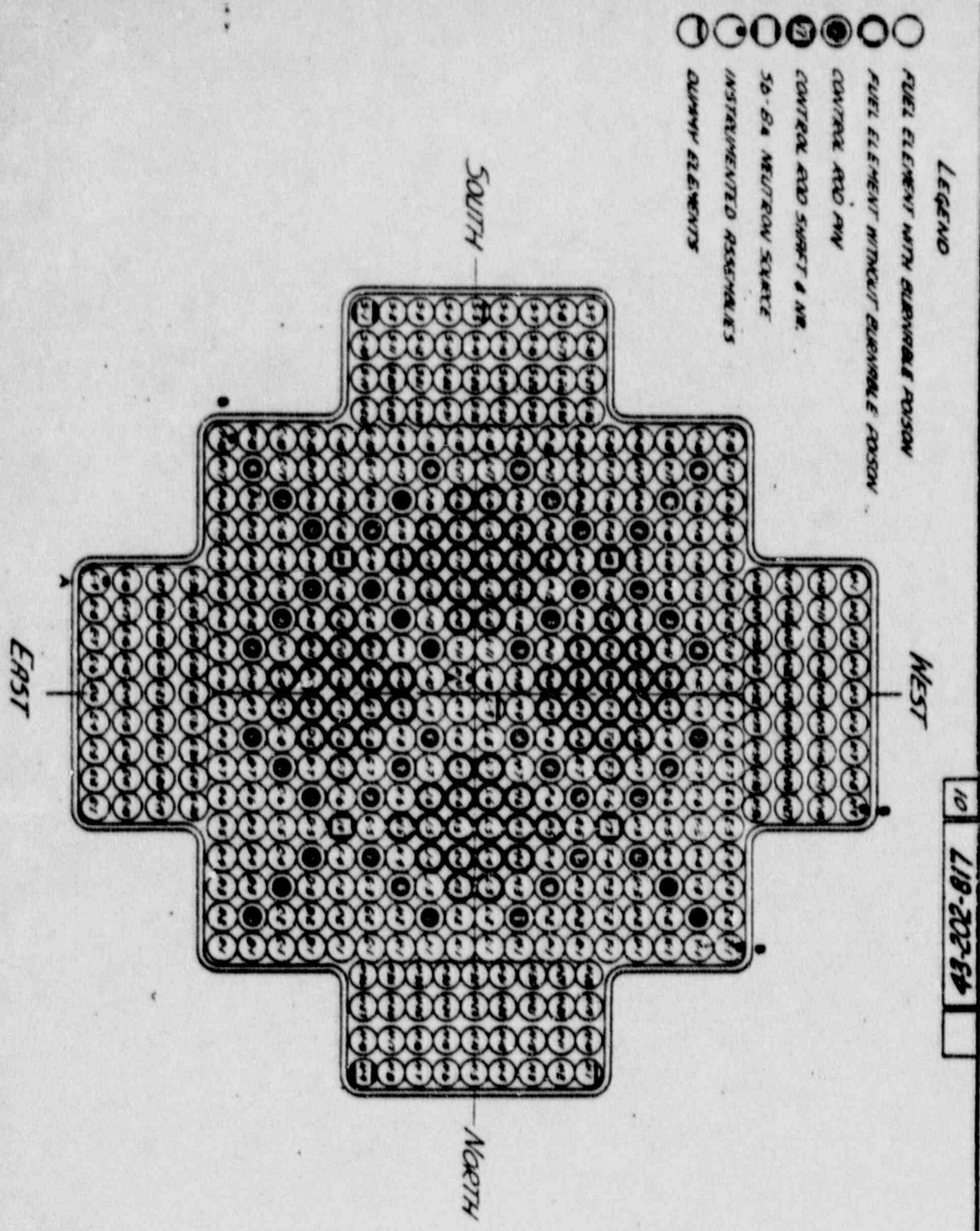


FIGURE 2

01 5-6-63
D.J FN

UNLESS OTHERWISE NOTED DIMENSIONS ARE IN INCHES & MACHINING TOLERANCES ARE: ANGLES ±

DIMENSIONS	UP TO 6 INCL.	OVER 6 TO 24 INCL.	OVER 24
FRACTIONAL	±	±	±
DECIMAL	±	±	±

— Confidential — Property of ALLIS-CHALMERS MFG. CO., Milwaukee, Wis.

3043 DEPT.

DRN 4-15-63 JAMES
CH'D 5-6-63 FN
TR'D
APP'D

PHASE III
SUPERHEATER
LOADING

SIMILAR TO: 43-202-174 SCALE 3

SHEET 1
OF 1

43-202-817

A-C MFG. CO. MILWAUKEE, WIS.

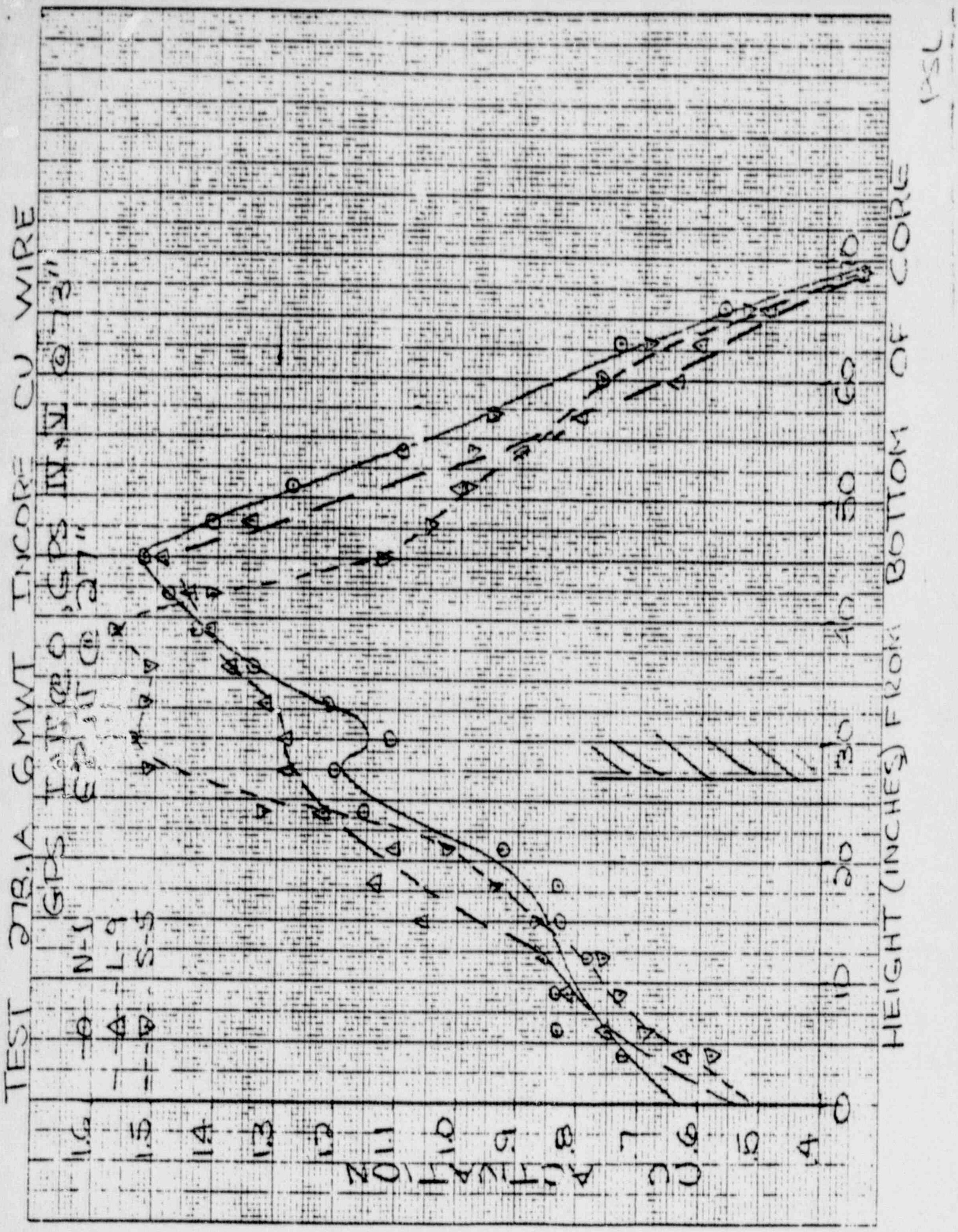

TEST 27B.1A 6 MWT INCORE CU WIRE
GPS I @ 0 GPS IV + V @ 73"
GPS II @ 37"
N-1
L-9
S-5
CU ACTIVATION
HEIGHT (INCHES) FROM BOTTOM OF CORE
0
10
20
30
40
50
60
70
.4
.5
.6
.7
.8
.9
1.0
1.1
1.2
1.3
1.4
1.5
1.6

FIGURE 3

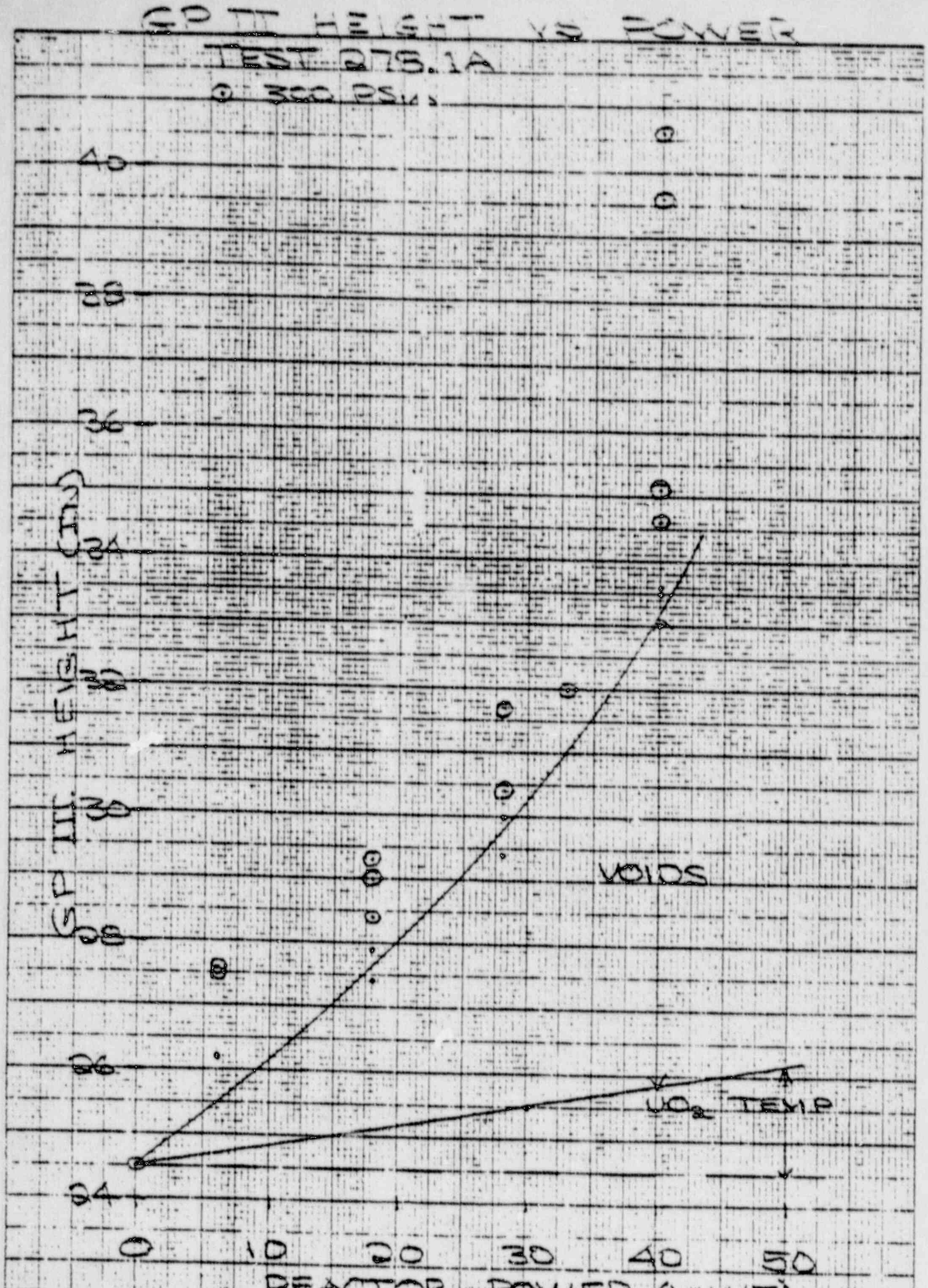

GP III HEIGHT VS POWER
TEST 275.1A
⊙ 300 PSIA
40
38
36
34
32
30
28
26
24
GP III. HEIGHT (IN)
VOIDS
UO2 TEMP
0
10
20
30
40
50
REACTOR POWER (MWT)

FIGURE 4

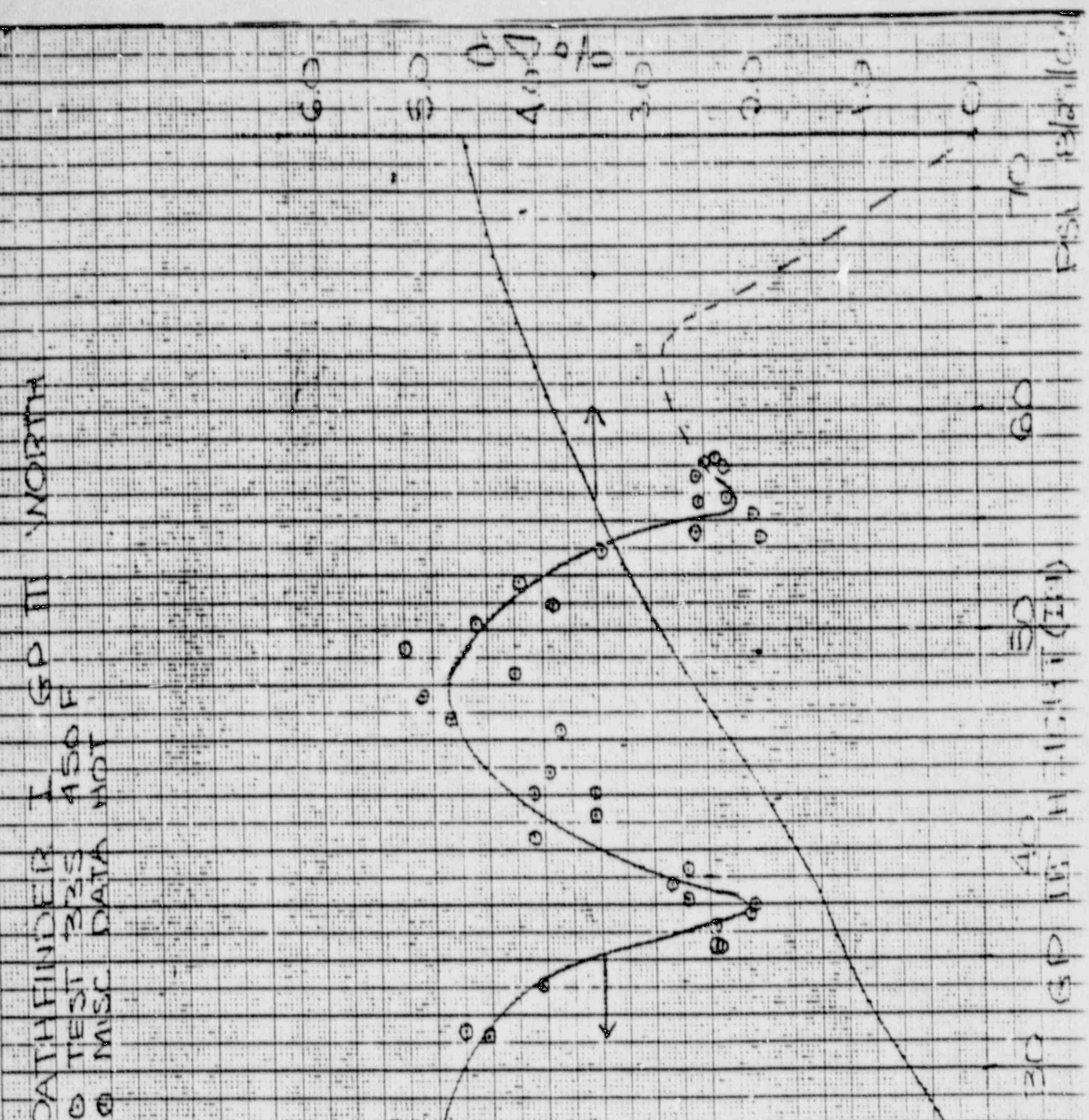
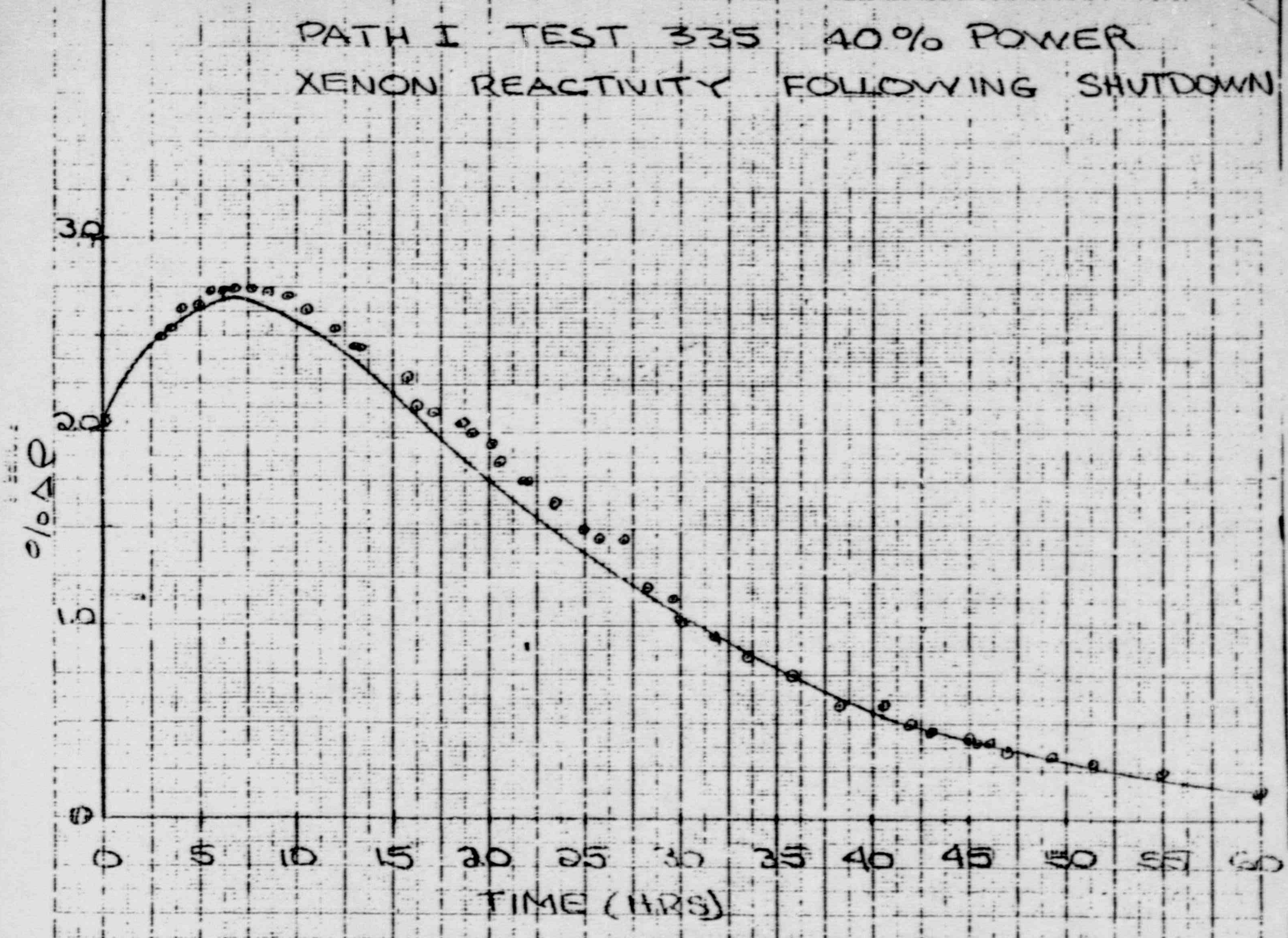

PATHFINDER I GP III WORTH
TEST 335 450 F
MISC DATA HOT
% Δk/k
GP HEIGHT (IN)
% Δk
30
40
50
60
.03
.05
.07
.09
.11
.13
.15
1.0
2.0
3.0
4.0
5.0
6.0

FIGURE 5


PATH I TEST 335 40% POWER
XENON REACTIVITY FOLLOWING SHUTDOWN
3.0
2.0
1.0
0
% Δρ
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME (HRS)

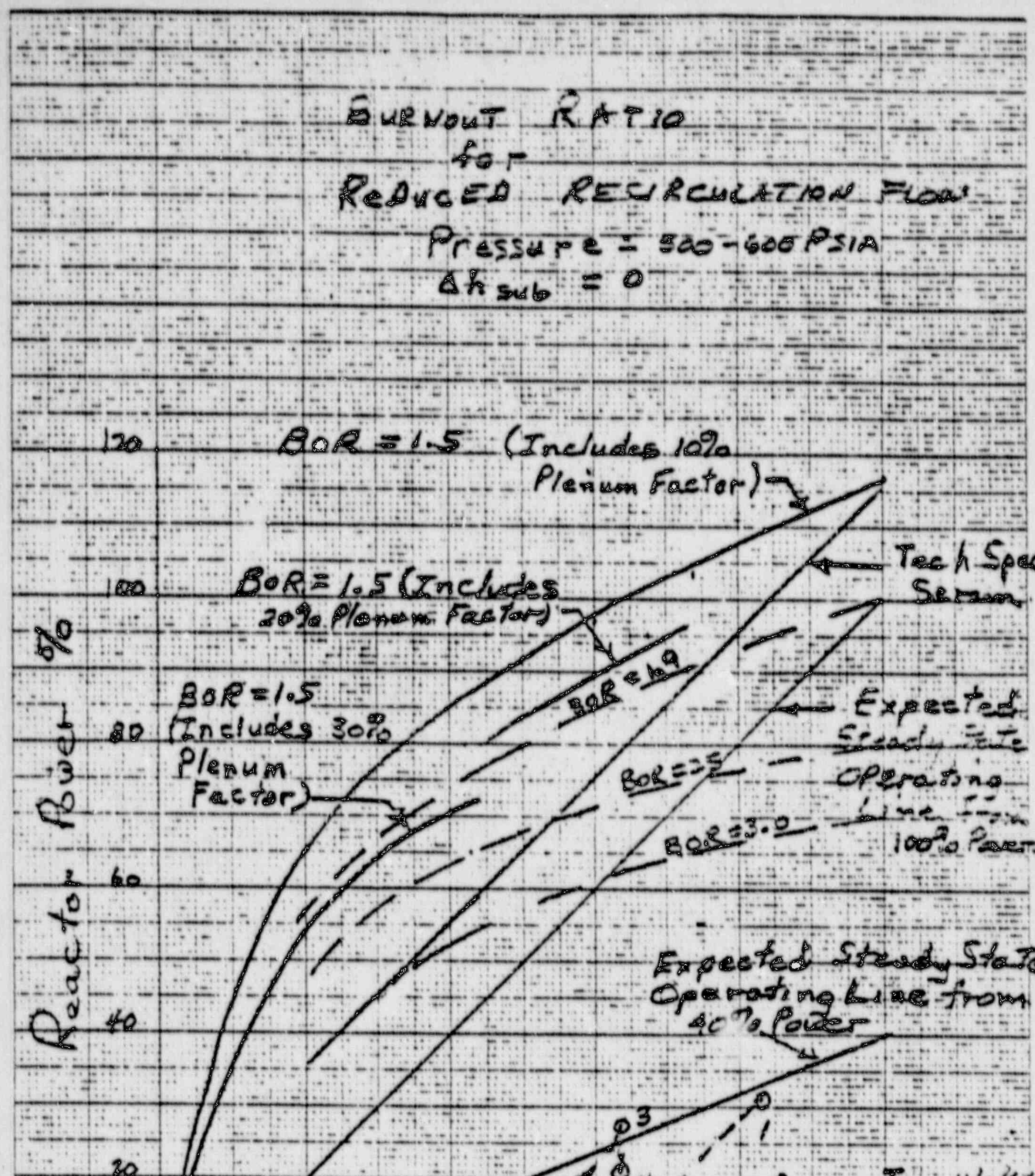
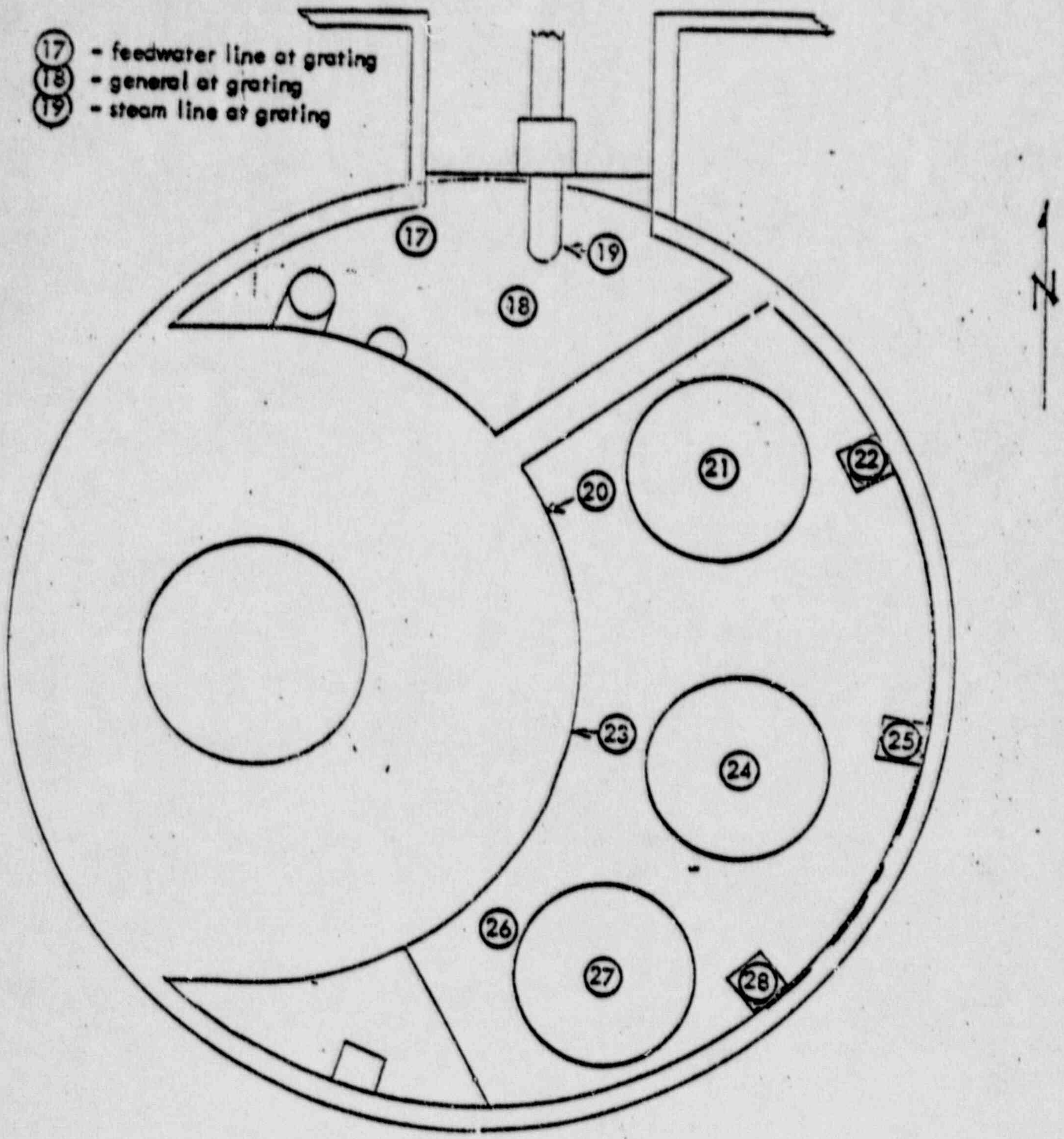
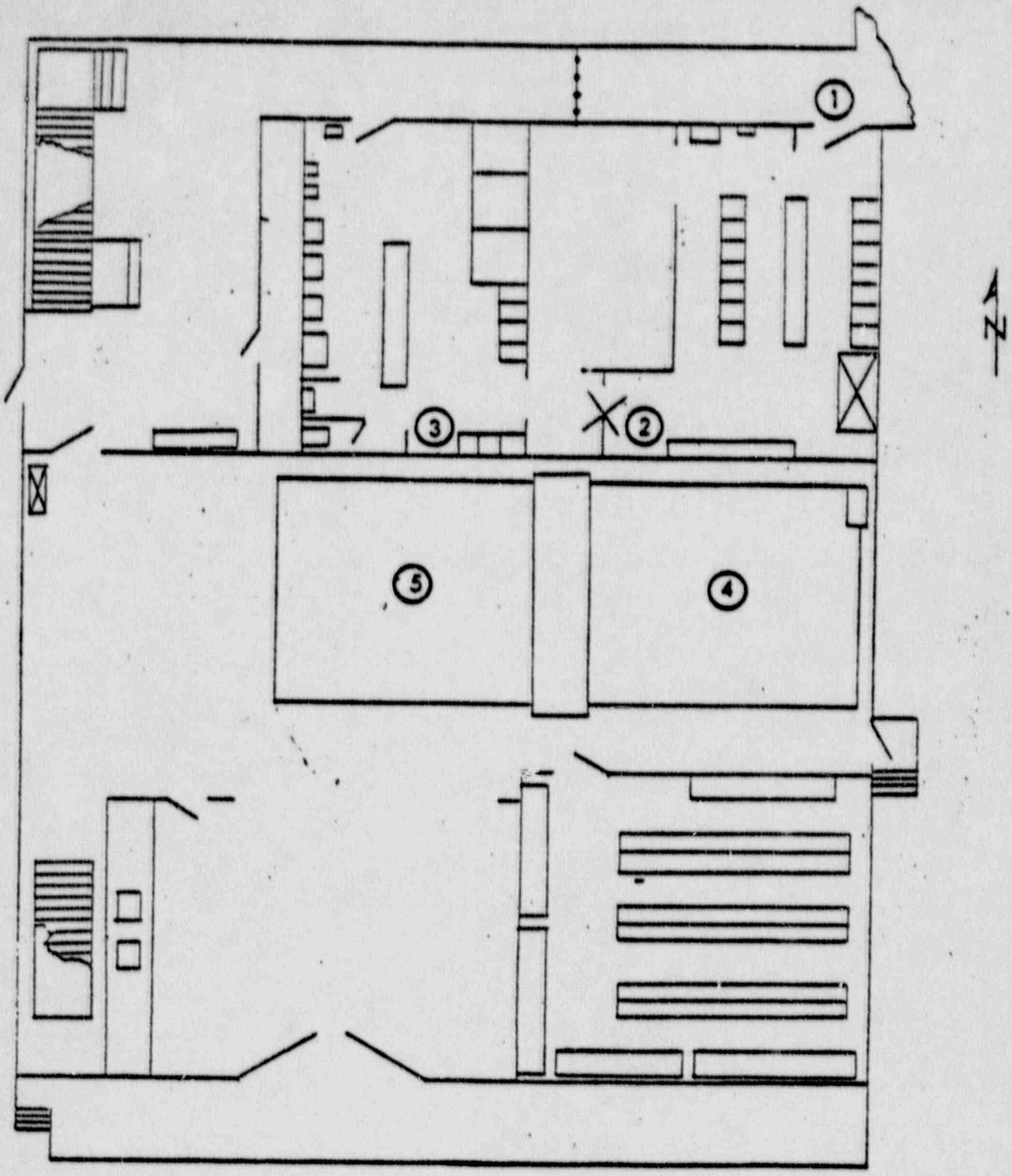

BURNOUT RATIO
for
REDUCED RECIRCULATION FLOW
Pressure = 500-600 PSIA
$\Delta h_{sub} = 0$
BOR=1.5 (Includes 10% Plenum Factor)
BOR=1.5 (Includes 20% Plenum Factor)
BOR=1.5 (Includes 30% Plenum Factor)
BOR=1.9
BOR=2.5
BOR=3.0
Tech Spec. Scram
Expected Steady State Operating Line from 100% Power
Expected Steady State Operating Line from 40% Power
Pump Trip 11/7/66
① Initial Power
② Min. Power
③ Max Power
④ Final Power
Reactor Power %
Recirculation Flow %
0
20
40
60
80
100
120

FIGURE 7



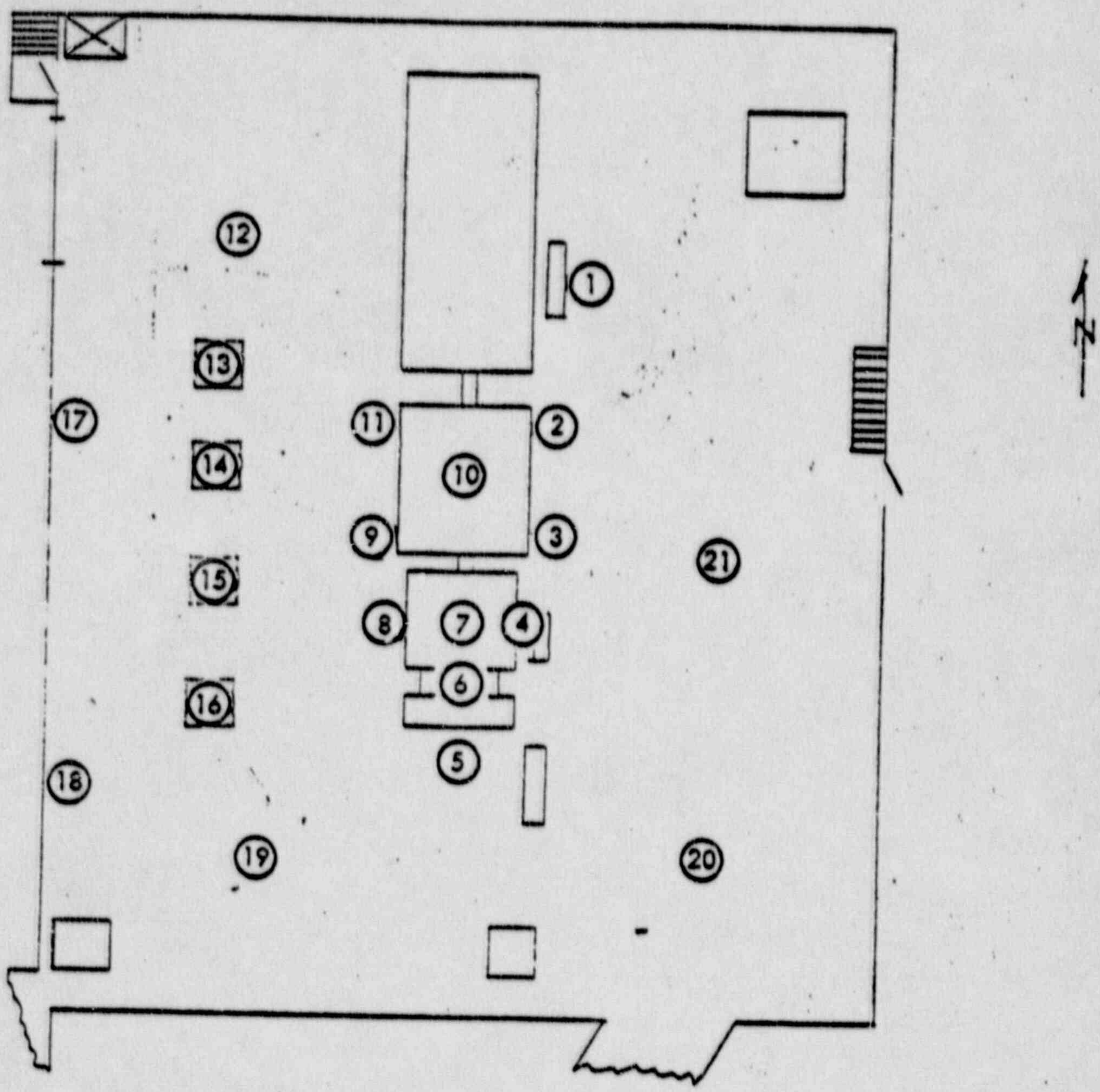
REACTOR BUILDING - PLUG FLOOR

FIGURE 8



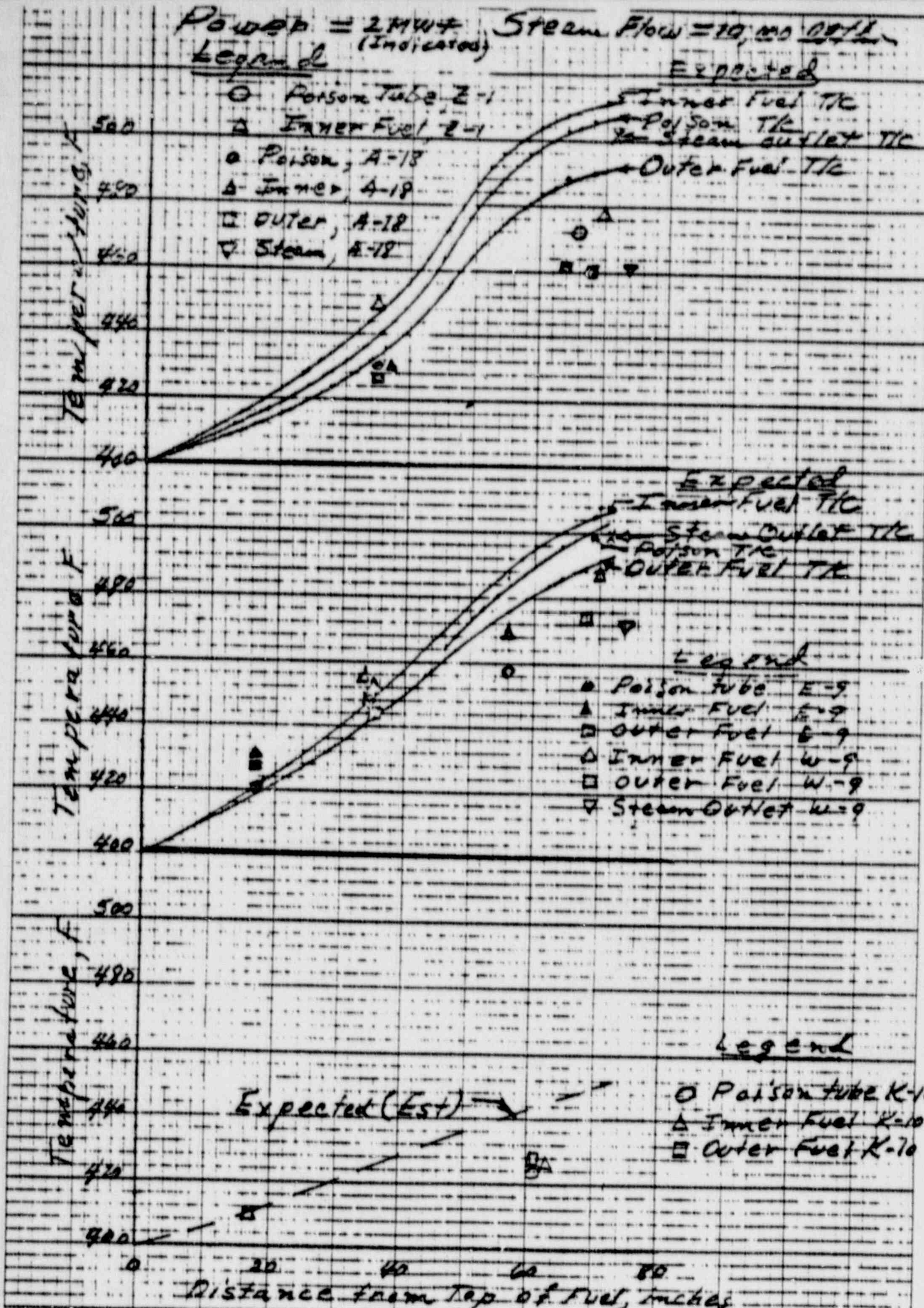
FUEL BUILDING - OPERATING FLOOR

FIGURE 9



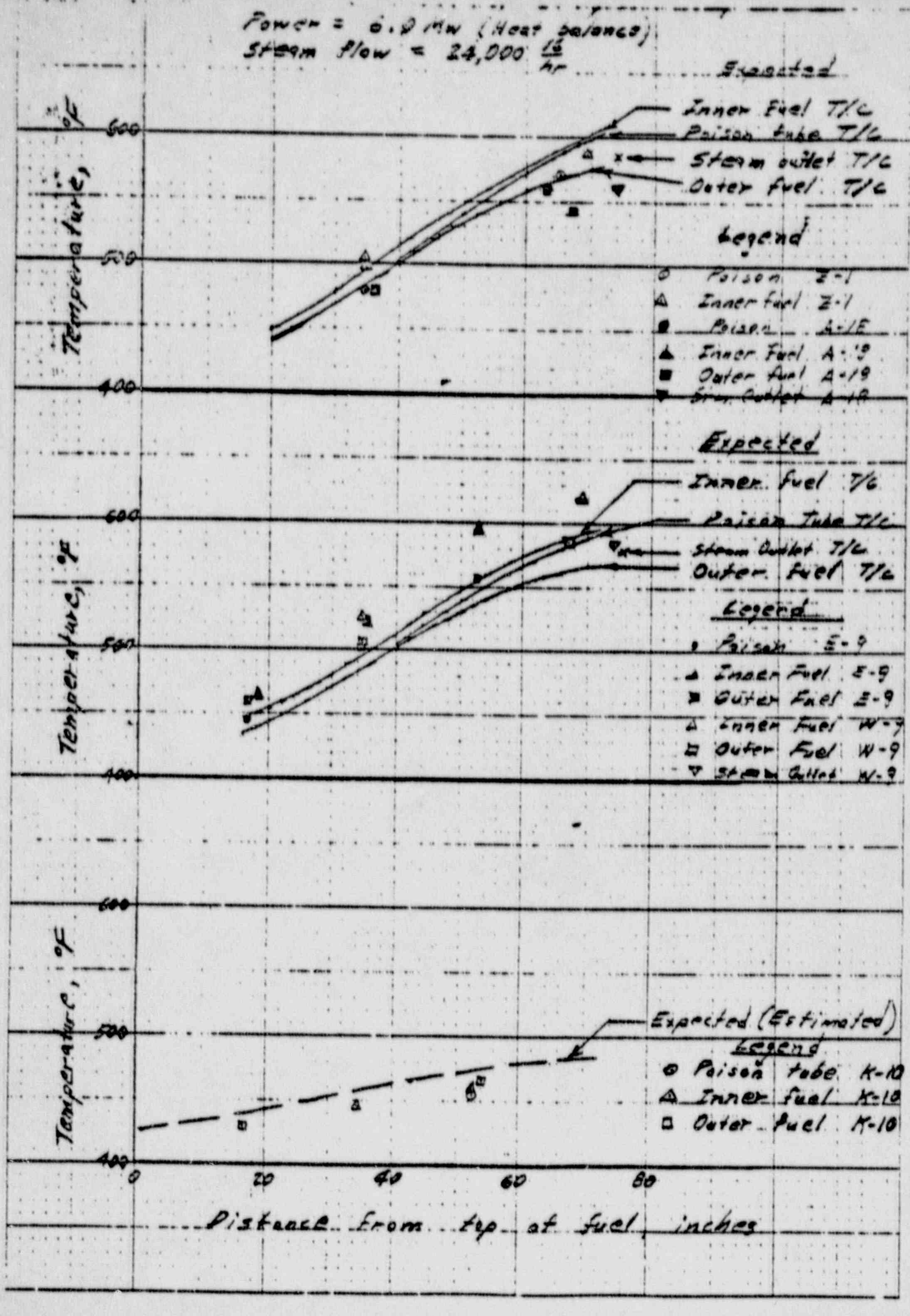
TURBINE BUILDING - OPERATING FLOOR

FIGURE 10



Power = 2MWt (Indicated)
Steam Flow = 10,000
Legend
Poison Tube E-1
Inner Fuel E-1
Poison, A-18
Inner, A-18
Outer, A-18
Steam, A-18
Expected
Inner Fuel T/C
Poison T/C
Steam outlet T/C
Outer Fuel T/C
Temperature, F
400
420
440
460
480
500
Expected
Inner Fuel T/C
Steam Outlet T/C
Poison T/C
Outer Fuel T/C
Legend
Poison tube E-9
Inner Fuel E-9
Outer Fuel E-9
Inner Fuel W-9
Outer Fuel W-9
Steam Outlet W-9
Legend
Poison tube K-10
Inner Fuel K-10
Outer Fuel K-10
Expected (Est)
0
20
40
60
80
Distance from Top of Fuel, inches

FIGURE 11


Power = 6.9 Mw (Heat balance)
Steam flow = 24,000 lb/hr
Expected
Inner Fuel T/C
Poison tube T/C
Steam outlet T/C
Outer fuel T/C
Legend
Poison Z-1
Inner fuel Z-1
Poison A-1E
Inner Fuel A-19
Outer fuel A-19
Stm. Outlet A-19
Expected
Inner fuel T/C
Poison Tube T/C
Steam Outlet T/C
Outer fuel T/C
Legend
Poison E-9
Inner Fuel E-9
Outer Fuel E-9
Inner Fuel W-9
Outer Fuel W-9
Steam Outlet W-9
Expected (Estimated)
Legend
Poison tube K-10
Inner fuel K-10
Outer fuel K-10
Temperature, °F
600
500
400
0
20
40
60
80
Distance from top of fuel, inches

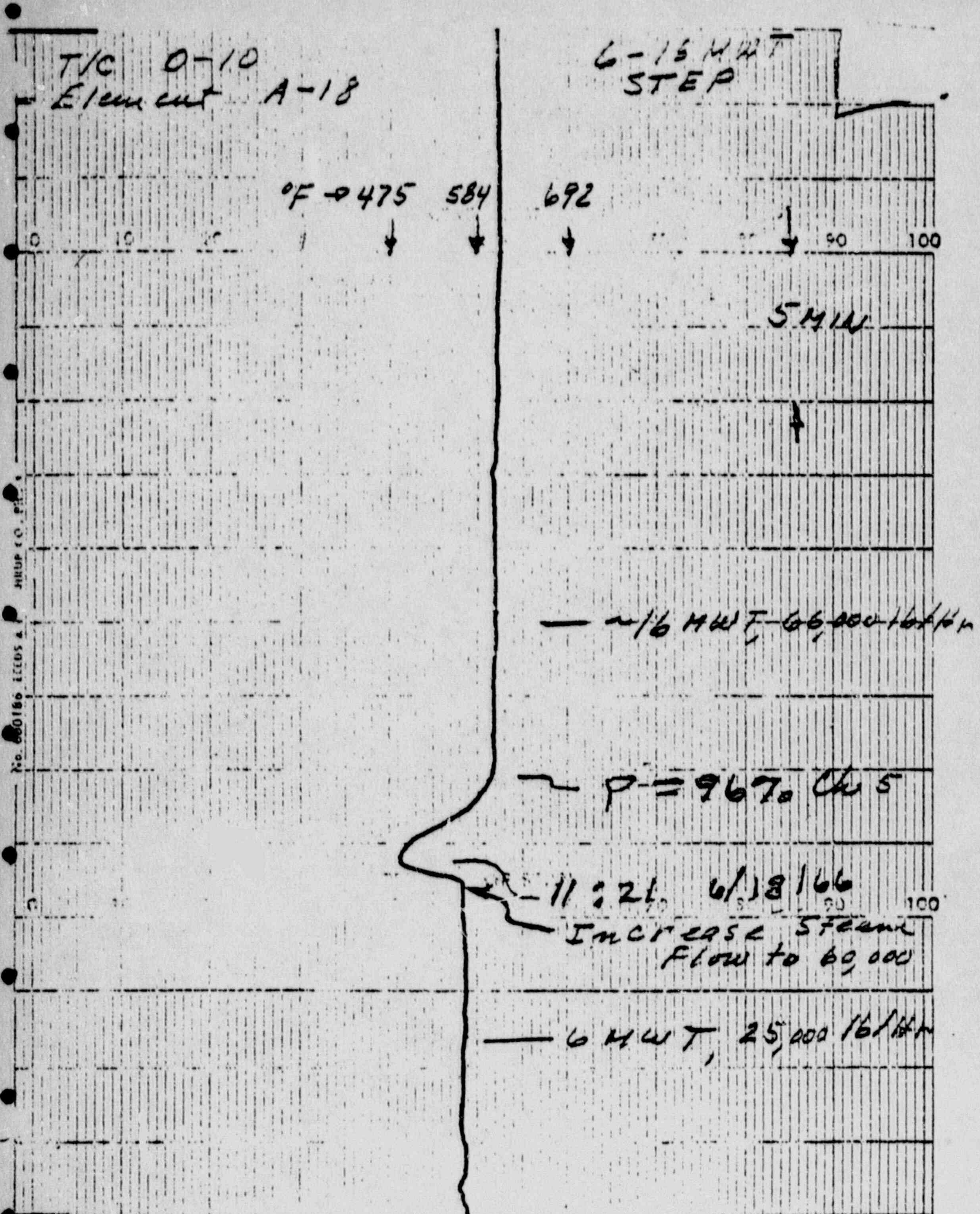

T/C 0-10
Element A-18
6-15 MWT
STEP
°F → 475
584
692
0
10
20
90
100
5 MIN
~16 MWT 60,000 lb/hr
P = 967. Ch 5
11:21 6/18/66
Increase Steam
Flow to 60,000
6 MWT, 25,000 lb/Hr

FIGURE 17

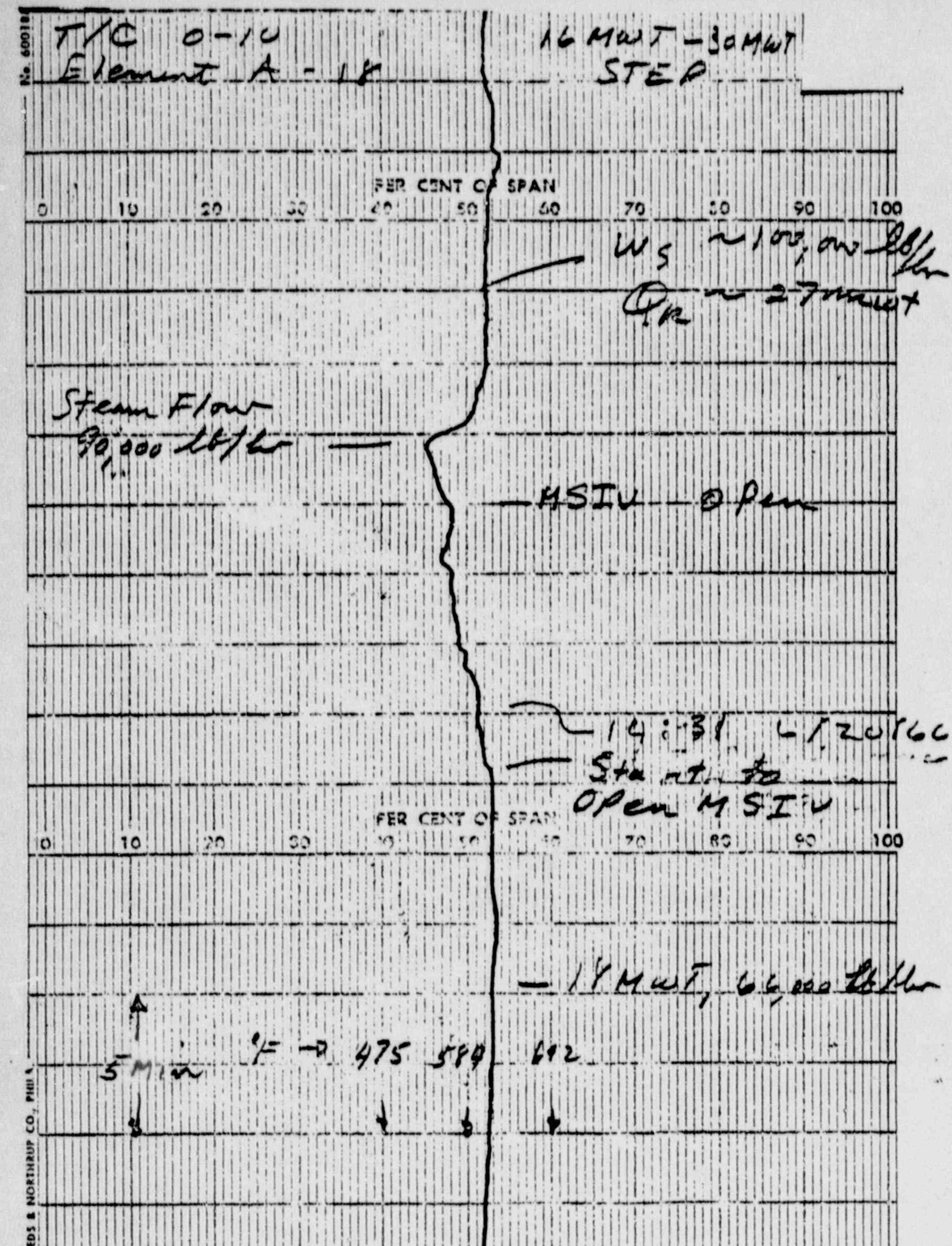

No. 60018
T/C 0-10
Element A-18
16 MWT – 30 MWT
STEP
PER CENT OF SPAN
0 10 20 30 40 50 60 70 80 90 100
Ws ~100,000 lb/hr
Qr ~ 27 MWt
Steam Flow
90,000 lb/hr
MSIV Open
Start to
Open MSIV
PER CENT OF SPAN
10 10 20 30 40 50 60 70 80 90 100
18 MWT, 66,000 lb/hr
5 Min
°F → 475 580 612
LEEDS & NORTHRUP CO., PHILA.

FIGURE 14

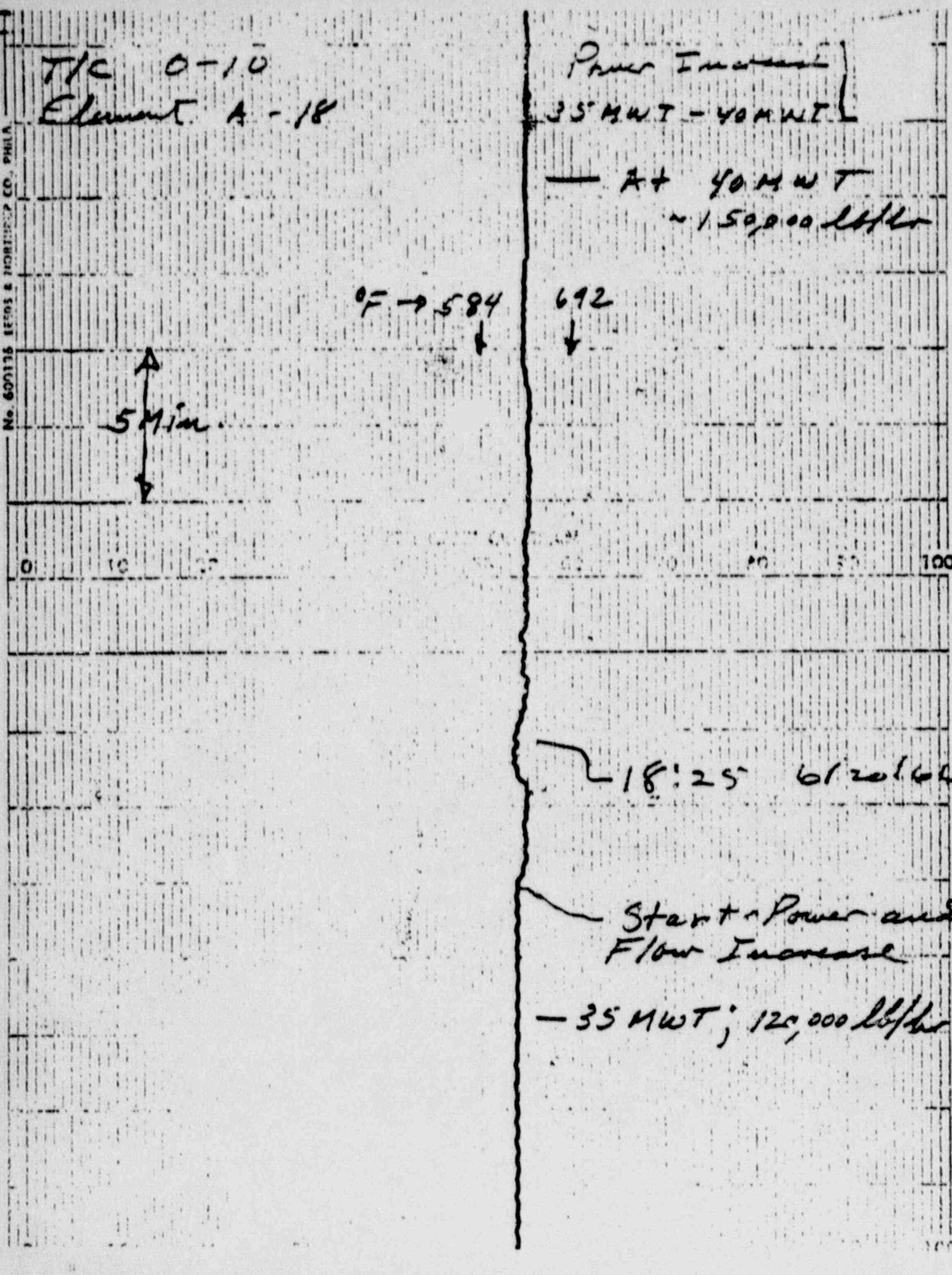

T/C 0-10
Element A-18
Power Increase
35 MWT - 40 MWT
At 40 MWT
~150,000 lb/hr
°F → 584
692
5 Min.
0
10
100
18:25 6/20/64
Start - Power and Flow Increase
35 MWT; 120,000 lb/hr
No. 600115 LEEDS & NORTHRUP CO. PHILA.

FIGURE 15

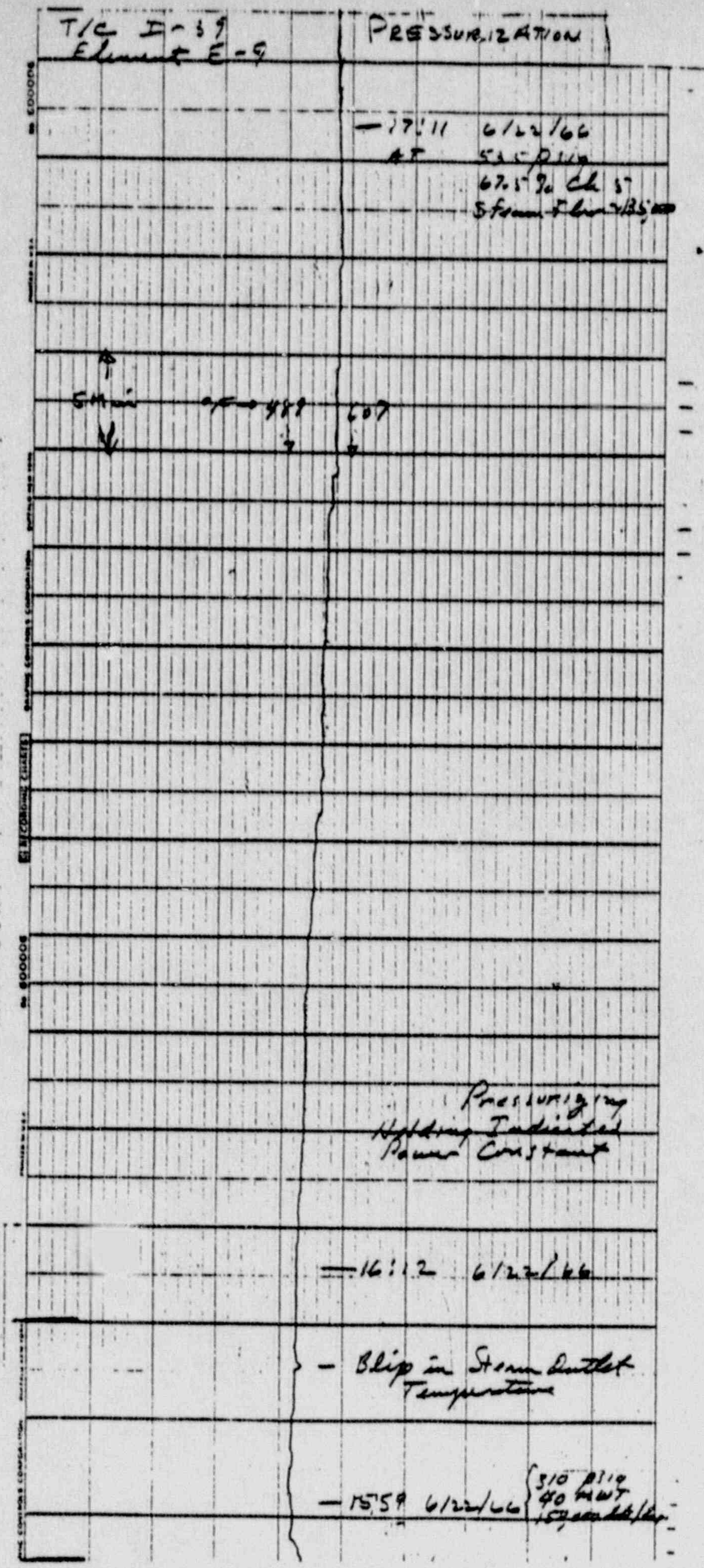

T/C I-39
Element E-9
PRESSURIZATION
17:11 6/22/66
5 Min
489
607
Pressurizing
Holding Indicated
Power Constant
16:12 6/22/66
Blip in Steam Outlet
Temperature
15:59 6/22/66
310 psig
40 MWT

FIGURE 16

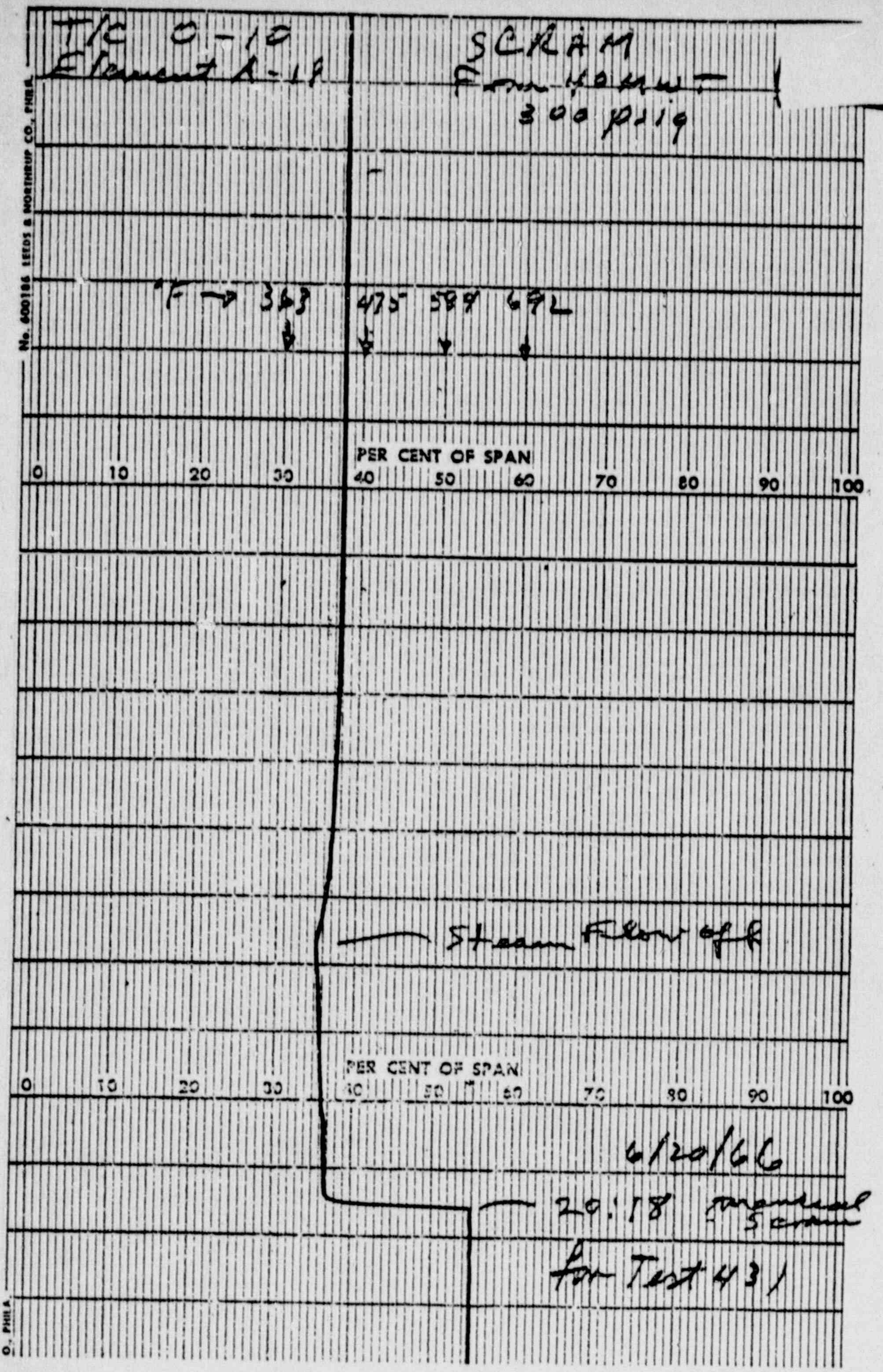

T/C 0-10
Element A-17
SCRAM
From 40 MW
300 psig
°F → 363 475 589 692
PER CENT OF SPAN
0 10 20 30 40 50 60 70 80 90 100
Steam Flow off
6/20/66
20:18 manual scram
for Test 431
No. 600186 LEEDS & NORTHRUP CO., PHILA.

FIGURE 17

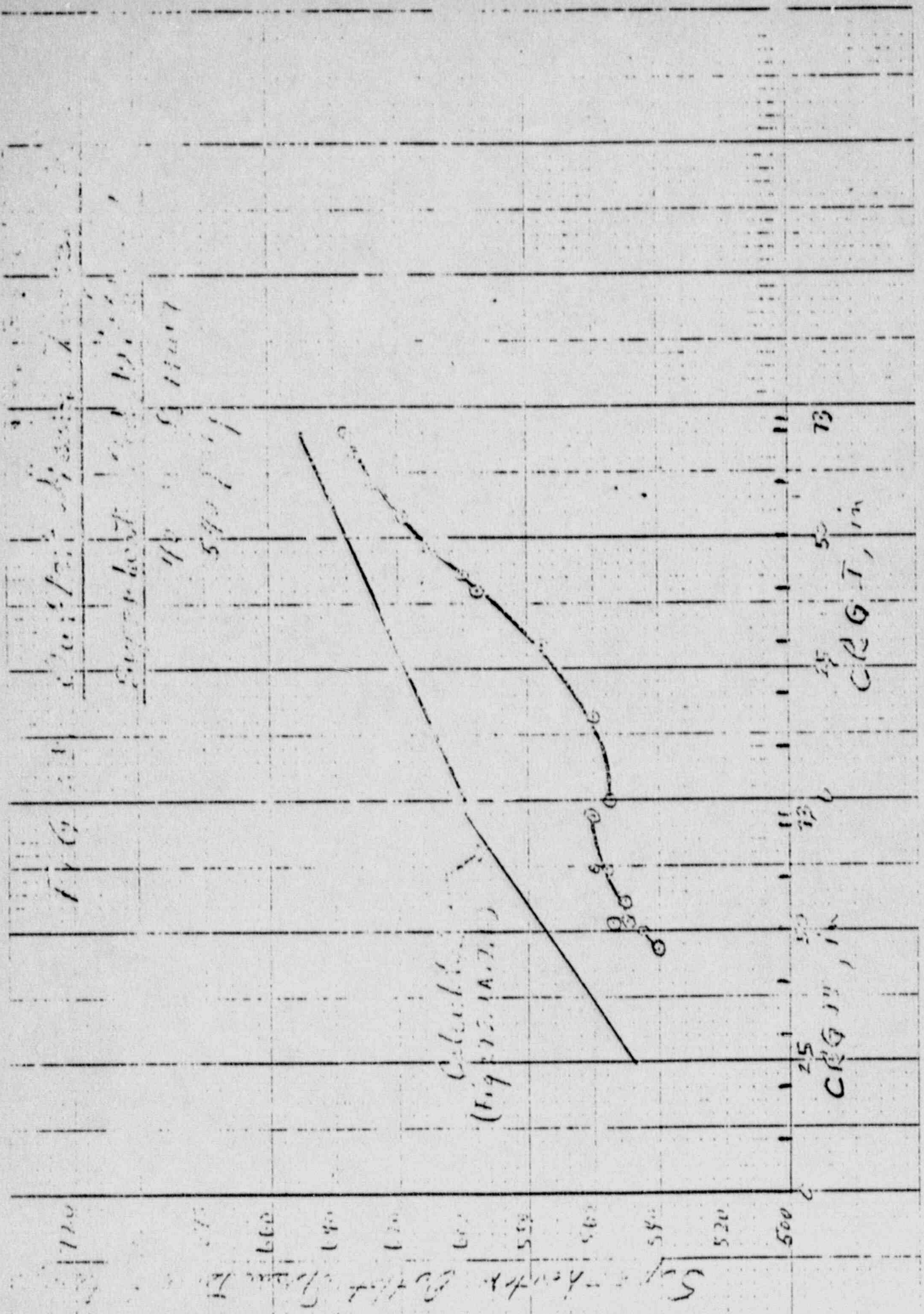
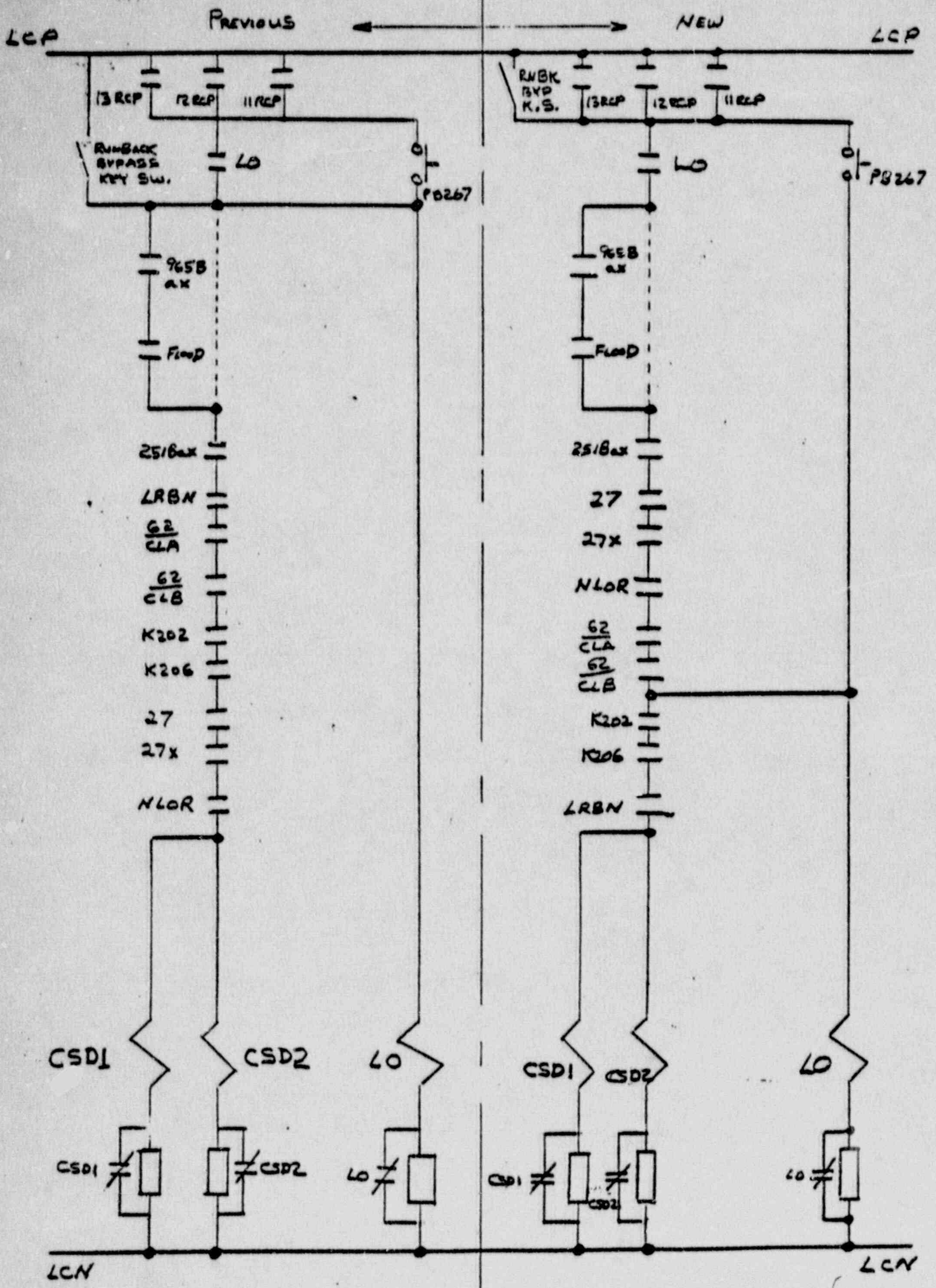


FIGURE 18

RUNBACK CIRCUITRY MODIFICATION



Derived from DE 84767 Sh.2

FIGURE 19

Process Scram Circuitry Additions

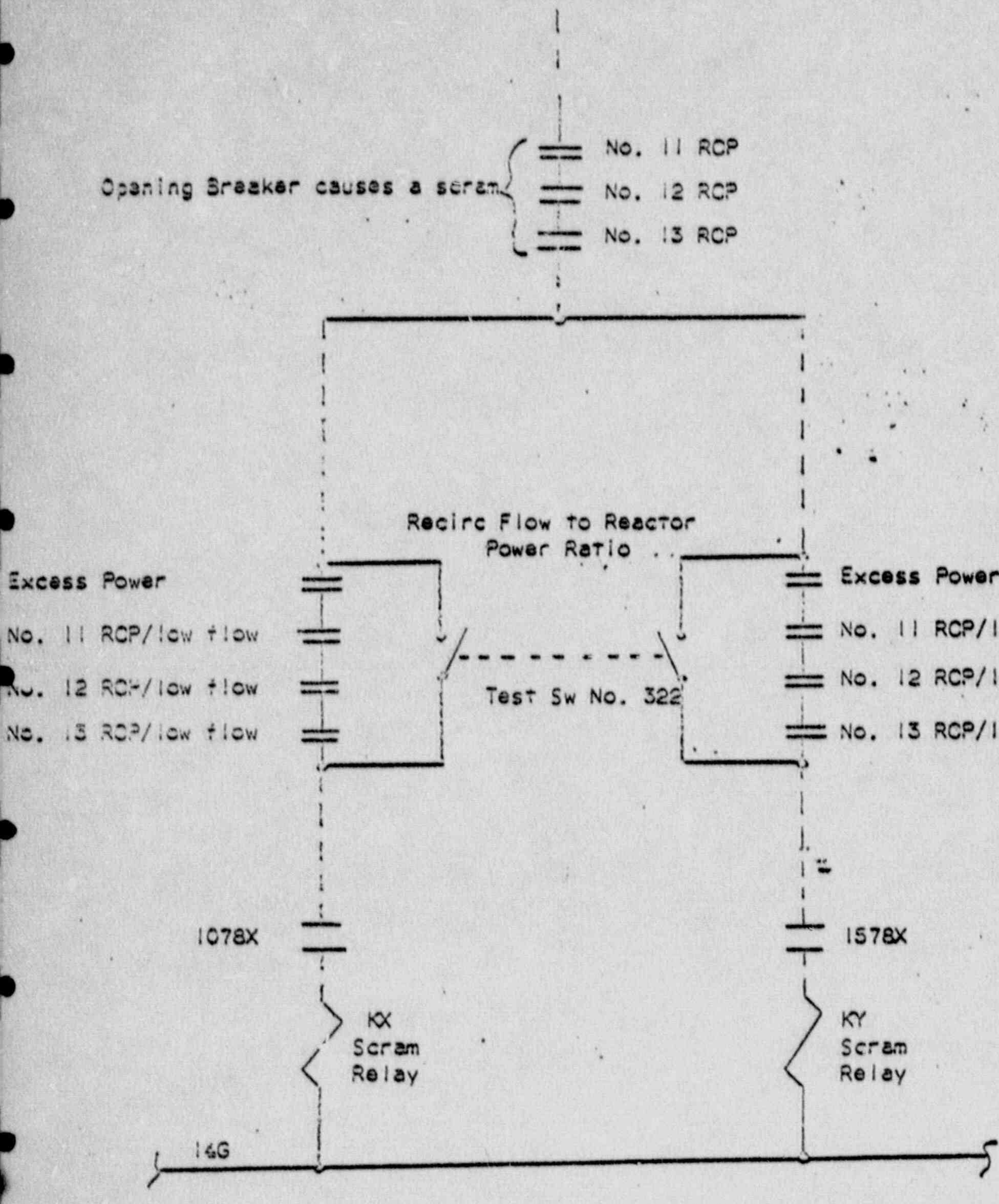


FIGURE 20

Recirculation Flow "Low Flow" Scram Circuitry

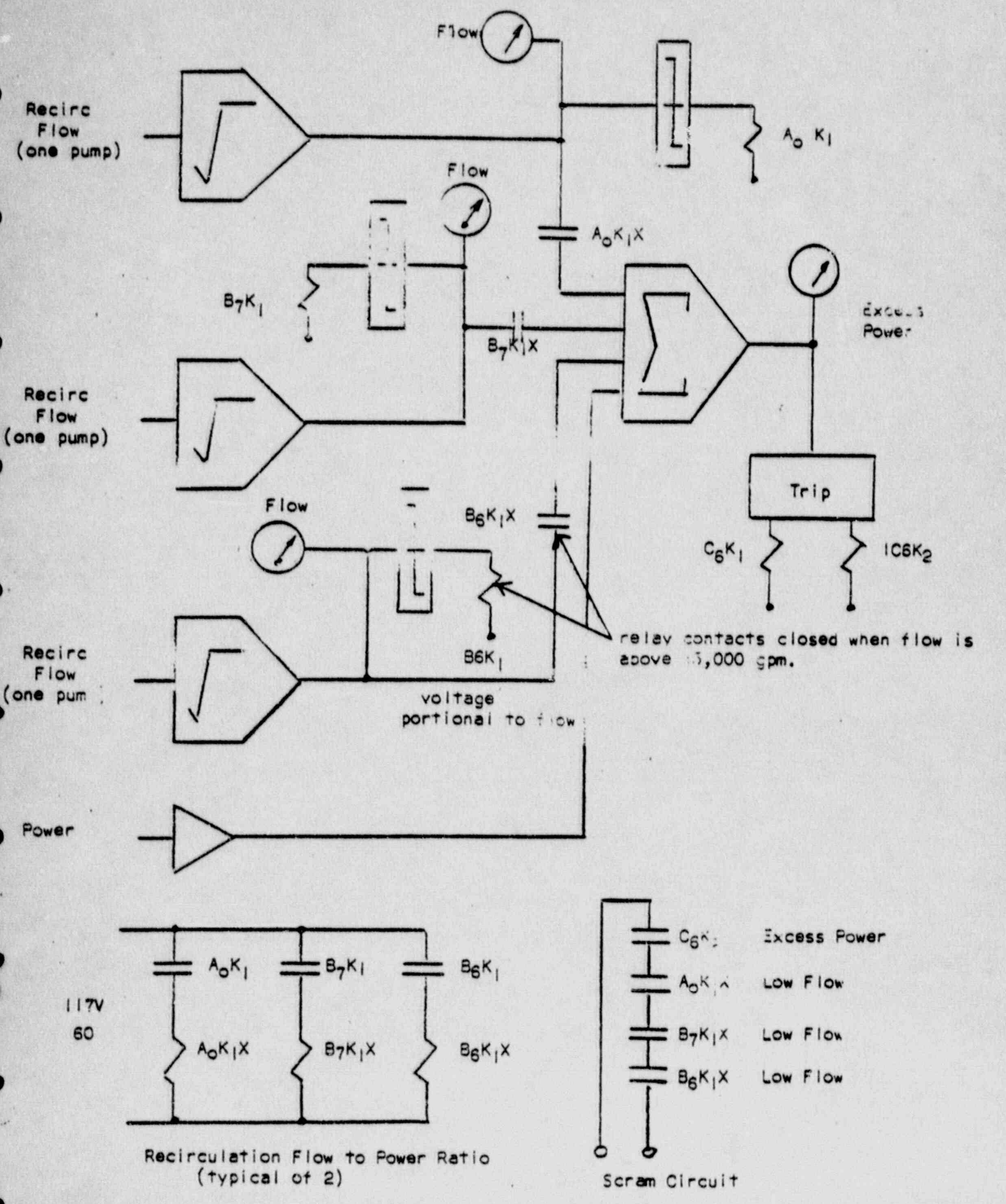
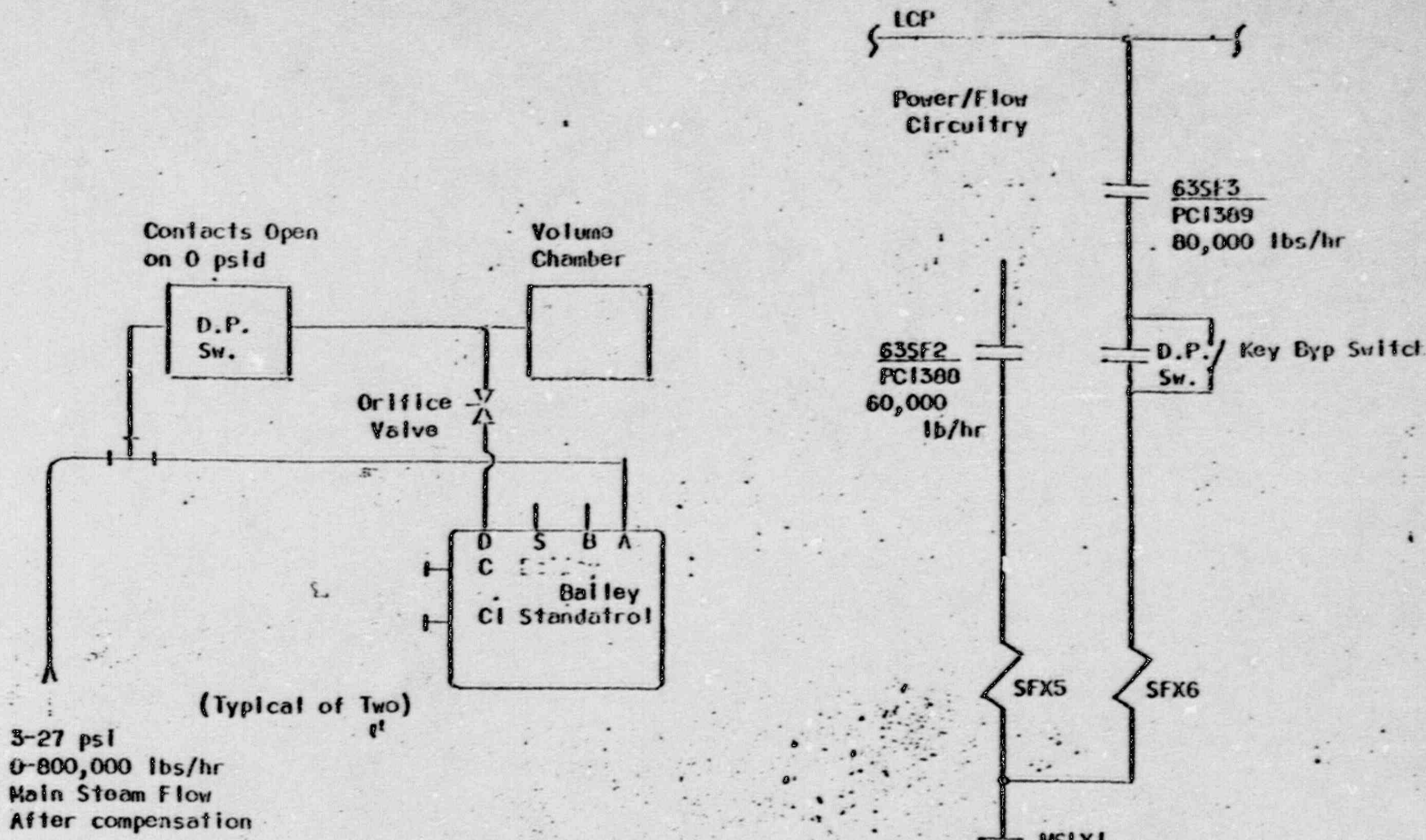


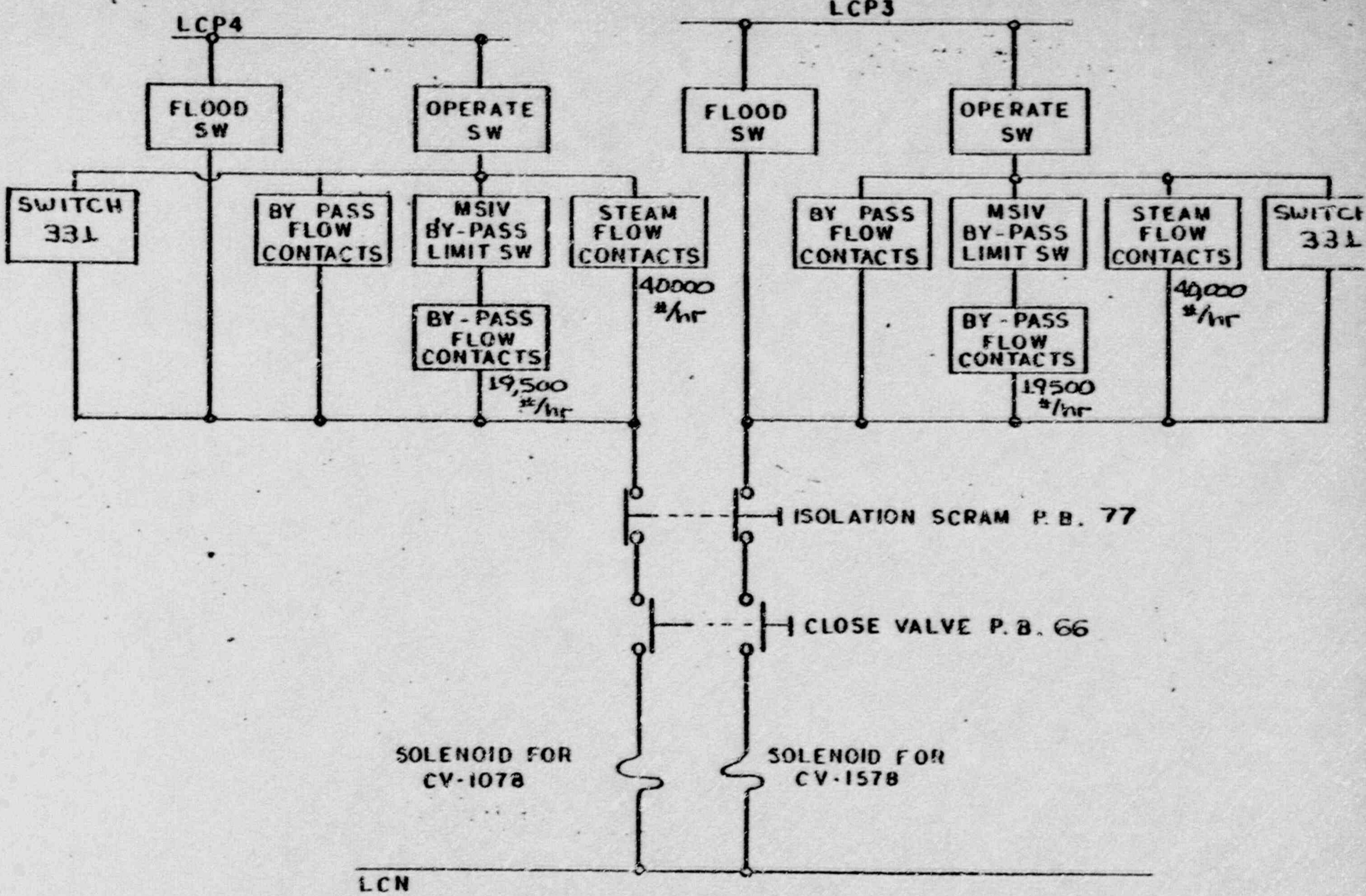
FIGURE 21

Loss of Steam Flo[illegible] [illegible]ction Circuitry



Loss of Steam Flow Protection Circuitry

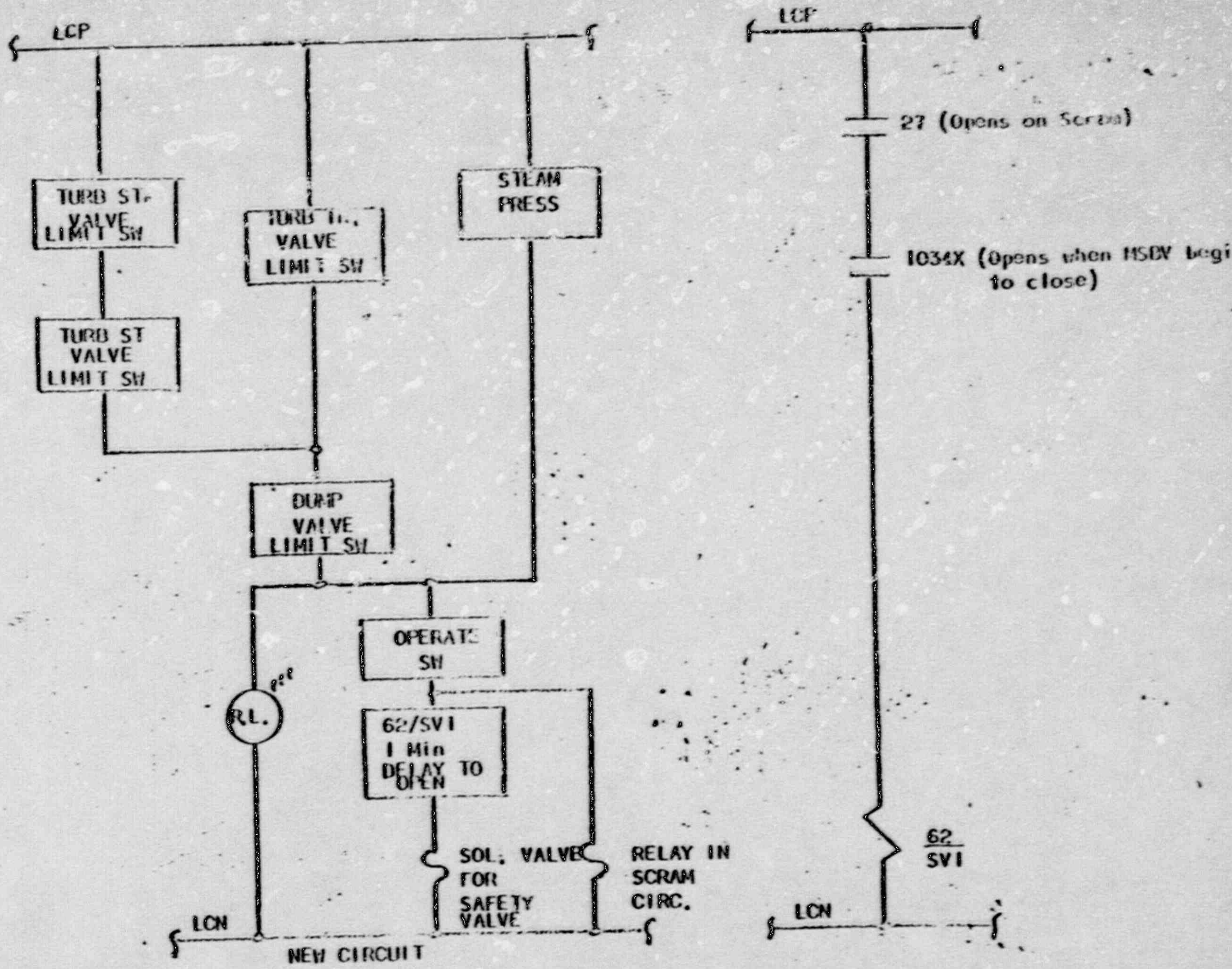
EMERGENCY CONDENSER OUTLET VALVES REVISED CIRCUITRY



SHUTDOWN CONDENSATE OUTLET VALVES CV 1078 & CV 1578 REVISED CIRCUITRY

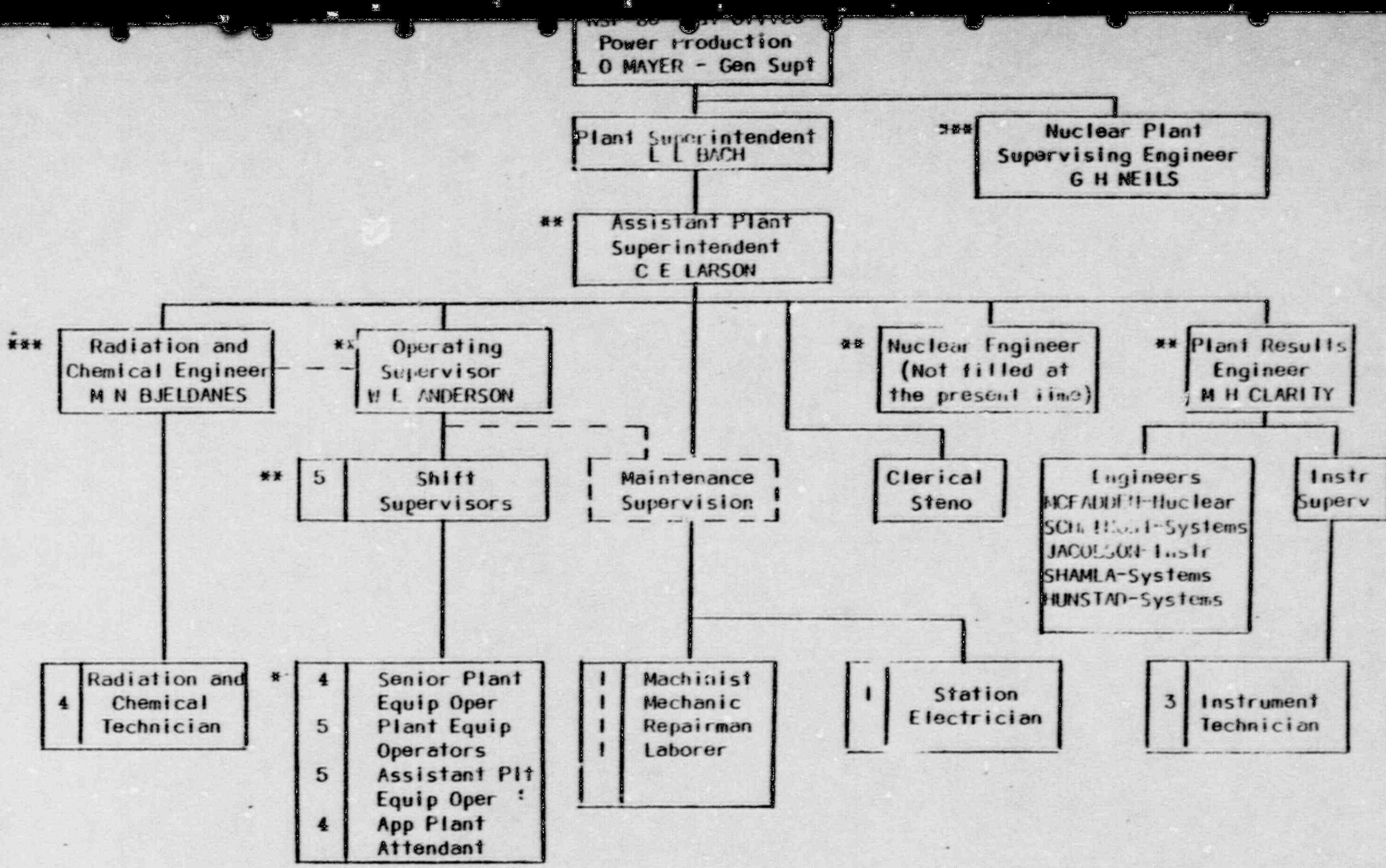
FIGURE 23

FIGURE 24



Power Actuated Safety Valve Operation

FIGURE 25



* AEC Licensed Operator positions
** AEC Licensed Senior Operator positions
*** AEC Licensed Senior Operator

Northern States Power Company
Pathfinder Atomic Power Plant
Organization Diagram
December 1, 1966