Report No. ACNP-63034

PATHFINDER ATOMIC POWER PLANT

TECHNICAL PROGRESS REPORT

JULY 1963 – SEPTEMBER 1963

Submitted to

U. S. ATOMIC ENERGY COMMISSION
NORTHERN STATES POWER COMPANY
and
CENTRAL UTILITIES ATOMIC POWER ASSOCIATES

by

ALLIS-CHALMERS MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
Milwaukee 1, Wisconsin

Facsimile Price $ 4.60
Microfilm Price $ 1.64

Available from the
Office of Technical Services
Department of Commerce
Washington 25, D. C.

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

DISCLAIMER

ACNP-63034

PATHFINDER ATOMIC POWER PLANT

TECHNICAL PROGRESS REPORT

JULY 1963 – SEPTEMBER 1963

Submitted to

U. S. ATOMIC ENERGY COMMISSION
NORTHERN STATES POWER COMPANY
and
CENTRAL UTILITIES ATOMIC POWER ASSOCIATES

by

ALLIS-CHALMERS MANUFACTURING COMPANY

Under
Agreement dated 2nd Day of May 1957, as Amended
between
Allis-Chalmers Mfg. Co. & Northern States Power Co.
under
AEC Contract No. AT(11-1)-589

December 30, 1963

Classification – UNCLASSIFIED

Approved: C. B. Graham

C. B. Graham
General Manager's Staff

Approved: Hibbert Hill

Hibbert Hill
Vice President-Engineering

ALLIS-CHALMERS MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
WASHINGTON, D. C. 20011

NORTHERN STATES POWER COMPANY
15 SOUTH FIFTH STREET
MINNEAPOLIS 2, MINNESOTA

ACNP-63034

PATHFINDER ATOMIC POWER PLANT

TECHNICAL PROGRESS REPORT

JULY 1963 - SEPTEMBER 1963

DISTRIBUTION

USAEC, Chicago Operations Office -- 9800 South Cass Avenue, Argonne, Illinois	5
USAEC, Division of Reactor Development -- Washington 25, D.C.	8
USAEC, OTIE -- Oak Ridge, Tennessee ... OFFSET MASTERS PLUS	20
Northern States Power Company and CUAPA	26
Allis-Chalmers Manufacturing Company	39
TOTAL	98

CONTENTS

Page

Distribution 1

List of Illustrations 4

Foreword 5

Design Data 6

PART A - PRE-CONSTRUCTION RESEARCH AND DEVELOPMENT

1.0 Fuel Element Research and Development

1.5 Nuclear Handling Tools 8

1.5.1 Design of Underwater Cutting Facility 8

3.0 Nuclear Analysis

3.1 Reactor Physics (Statics) 15

3.1.1 Pathfinder Xenon Transient Studies 15

PART B - POST CONSTRUCTION RESEARCH AND DEVELOPMENT

1.0 Initial Stability and Performance Tests ... Post Construction R & D 31

1.0.1 Dynamics 31

4.0 Mechanical Tests ... Post Construction R & D 32

4.0.1 Pre-Critical Testing 32

4.0.1.1 Introduction 32

4.0.1.2 Reactor Internals 32

4.0.1.3 Stud Cover Seal 35

4.0.1.4 System Tests 35

6.0 Physics ... Post Construction R & D 36

6.0.1 Introduction 36

6.0.2 Rod Drop Calculations 36

CONTENTS

Page

8.0 Core Performance Tests ... Post Construction R & D 44

8.0.1 Instrumented Superheater Fuel Elements 44

8.0.1.1 Purpose of Elements 44

8.0.1.2 Instrumentation Design and Test Conclusions.. 44

ILLUSTRATIONS

Fig. No.	Title	Page
1.1	Underwater Saw Facility - Plan View	11
1.2	Underwater Saw Facility - Vertical View	12
1.3	Photo of Assembled Underwater Saw Unit	13
3.1	Xenon Buildup During Power Operation	22
3.2	Xenon Buildup - Per Cent Power Step After 60 Hr of Full Power	23
3.3	Xenon Buildup - Per Cent Power Step After 60 Hr of 75% Power	24
3.4	Xenon Buildup - Per Cent Power Step After 60 Hr of 50% Power	25
3.5	Xenon Concentration After Shutdown	26
3.6	Xenon Analysis of Power Operation After Shutdown of Four Hr	27
3.7	Xenon Analysis of Power Operation After Shutdown of Eight Hr	28
3.8	Pathfinder Reactivity Loss as a Function of Reactor Power Level	29

POST CONSTRUCTION

6.0.1	"Boiler Only" Rod Drop Curves	40
6.0.2	"Cold Boiler Plus Superheater" Rod Drop Curves	41
6.0.3	Linear Plot of "Cold Boiler Plus Superheater" Rod Drop Curves	42
6.0.4	Schulz Constants - Cold Boiler Plus Superheater	43
8.0.1	Instrumented Superheater Fuel Element Assembly	46

FOREWORD

This is one of a series of reports covering technical progress on the research and development program being performed in connection with the design of the Pathfinder Atomic Power Plant. This plant will be located at a site near Sioux Falls, South Dakota and is scheduled for operation in 1964. Owners and operators of the plant will be the Northern States Power Company of Minneapolis, Minnesota.

The U. S. Atomic Energy Commission, through Contract No. AT(11-1)-589 with Northern States Power Company, and Central Utilities Atomic Power Associates* (CUAPA), are sponsors of the research and development program.

Allis-Chalmers Manufacturing Company of Milwaukee, Wisconsin, under contract with Northern States Power Company, is performing the research, development, and design; and will construct the plant including the reactor, which is designated the Controlled Recirculation Boiling Reactor (CRBR) with Nuclear Superheater. Pioneer Service and Engineering Company of Chicago, Illinois is providing the architect-engineer services to Allis-Chalmers. Portions of the R & D program, particularly in connection with fuel development, have been subcontracted by Allis-Chalmers.

---

*CUAPA MEMBER COMPANIES:

Interstate Power Company
Iowa Power and Light Company
Iowa Southern Utilities Company
Madison Gas and Electric Company
Northern States Power Company
Northwestern Public Service Co.
Otter-Tail Power Company
St. Joseph Light & Power Co.
Western Power and Gas Company
Wisconsin Public Service Corp.

DESIGN DATA

CRBR WITH NUCLEAR SUPERHEATER

Plant

Power, boiler region	157,200 kw
Power, superheater region	42,400 kw
Steam flow at rated power	616,125 lbs/hr
Total core power	199,600 kw
Gross electrical capability	66,000 kw
Net electrical output	62,000 kw
Net efficiency	31.0 per cent
Steam outlet pressure (reactor)	535 psig
Reactor operating pressure	600 psig
Temperature, boiler region	489 F
Outlet temperature, superheater region	825 F
Gross heat rate	10,199 Btu/kw-hr
Reactor building size	50 ft dia x 120 ft

Reactor

Vessel size (over-all)	11 ft 6 in o.d. 36 ft 1 in
Total core dimensions	6 ft x 6 ft
Dimensions of superheater region	6 ft x 30 in
Fuel, boiler (Zr-2 clad)	approx. 2.2 per cent enriched UO_2
Fuel, superheater (S.S. clad)	approx. 93 per cent enriched UO_2
Fuel, loading, boiler (U-235)	145.6 kg
Fuel, loading, superheater (U-235)	42 kg
Power density (boiler core coolant)	87 kw/liter
Average heat flux, boiler region	128,000 Btu/hr-ft^2
Average heat flux, superheater region	77,800 Btu/hr-ft^2
Maximum heat flux, boiler region	462,000 Btu/hr-ft^2
Maximum heat flux, superheater region	245,000 Btu/hr-ft^2
Recirculation rate	65,000 gpm
Recirculation pump power	823 kw
Neutron flux	approx. 5×10^{13} n/cm^{2}sec

I. FUEL ELEMENT RESEARCH AND DEVELOPMENT

1.5 NUCLEAR HANDLING TOOLS

The objective of this project is to perform conceptual engineering and experimental work as required for development of simplified but reliable fuel handling tools and special tools for repairing or replacing reactor parts.

1.5.1 DESIGN OF UNDERWATER CUTTING FACILITY

Removal and shipping of boiler and superheater control rods after use in an operating reactor present a problem because of their irradiation and long axial dimension. Special tools have been devised to handle the removal, but casks for shipping irradiated materials are not long enough to accommodate the complete control rod. These rods (of welded construction) must therefore be cut to fit the shipping casks. Since they will be radioactive, they must be cut under water at the site. Thus an underwater cutting facility was designed for this purpose.

Various cutting techniques were studied, such as an underwater cutting torch (which proved unworkable), before deciding to use a saw. After soliciting bids from saw manufacturers, it was concluded (in view of the high bids, including one bid of $40,000) that a small standard pipe-cutting hack saw would be purchased and modified for use at the Pathfinder site. The modified saw will operate in any position and will permit cutting control rods while they are in a vertical position. The unit is light in weight since the main body is constructed of aluminum. The cutting motion is supplied by an air motor which can be adapted for underwater use. The saw blade is fed by a hand crank.

The basic cutting facility concept involved mounting the saw on a platform. This platform is positioned on the top of a stand which holds the control rods and collects the chips during the sawing operation. A hydraulically-operated clamp is also mounted on the saw platform. The platform will permit the saw to be withdrawn from the pool for easy maintenance and storage.

In this design, the saw is fed by turning a long floating handle. A flexible shaft transmits the motion from a vertical head on the top of the platform to the horizontal feed screw on the jaw. The clamp design utilizes a rack and pinion drive for the two opposing jaws. This was necessary because of the limited space between the control rod storage racks. The hydraulic cylinder, operated by a hand pump, lends itself very well to remote operation.

Actual construction of this modified hack saw unit presented several problems. Protection of components against corrosion was one problem area. Some of the saw parts proved difficult to chrome plate satisfactorily and were nickel plated instead. For example, the teeth on the racks and pinions of the clamp drive mechanism were chrome plated; inspection, however, showed plating build-up on the tooth edges and a lack of plating in the "valleys." This caused rough operation and permitted corrosion. The racks and pinions were therefore machined out of stainless steel. Other components of the hack saw were also replaced by stainless steel parts.

Scaling of the air motor was found necessary in order to eliminate air bubbles which impaired underwater vision. This was accomplished by remachining the mating surfaces of the exhaust manifold, and replacing gaskets. All other joints were sealed with gasket-forming material. Shaft bearings were replaced with sealed bearings, and additional seals were added to the output shaft. Upon completion of these modifications, the motor was found to be satisfactorily sealed.

Initial testing of the saw was carried out in air. Both the cruciform section of the control rod and the thin wall tubing of the superheater inner insulating tubes were clamped in the jaws and cut with the saw.

Saw blades with teeth of 6-pitch and 10-pitch were available. Both were tested. The 6-pitch blade was too course for the thin wall tubing, but the 10-pitch blade worked out well. Since the 10-pitch saw blade also cut the cruciform satisfactorily, it was decided to use this blade for all operations.

During cutting of the tubing, it was observed that the clamp jaws did not hold the tubing securely. After serrating the clamping surface of the jaws, further checks showed the revised clamping surface to be satisfactory for both the tubing and the cruciform.

Following these modifications, the saw was set up with the stand and used to cut a dummy superheater control rod. A section of the boiler control rod extension was also cut. In both cases, the saw performed as required. It was subsequently shipped to the Pathfinder site.

Figures 1.1 and 1.2 illustrate the design and assembly of this underwater saw facility. Figure 1.3 is a photograph of the assembled unit.

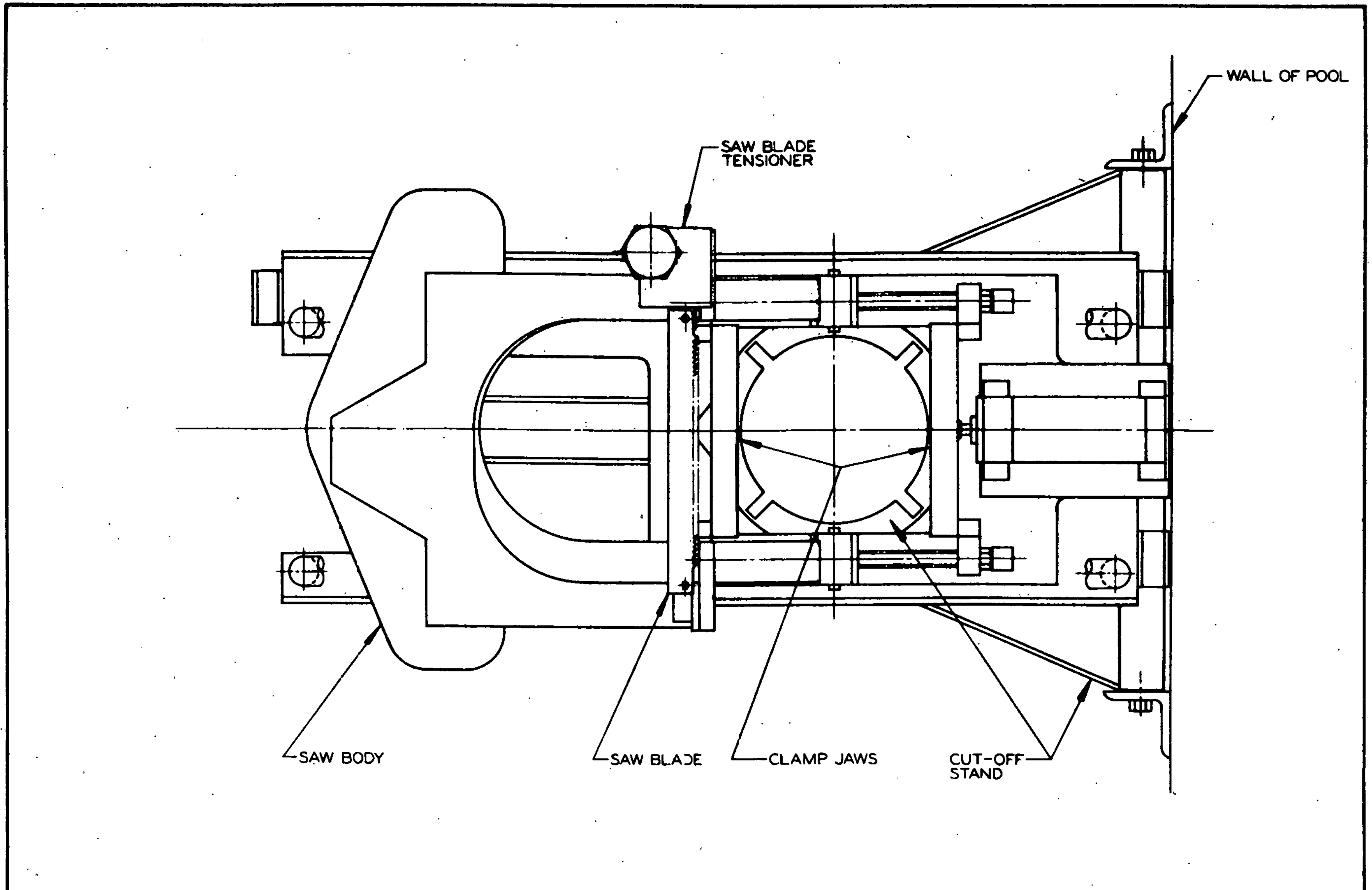


Fig. 1.1...Underwater Saw Facility - Plan View (43-202-795)

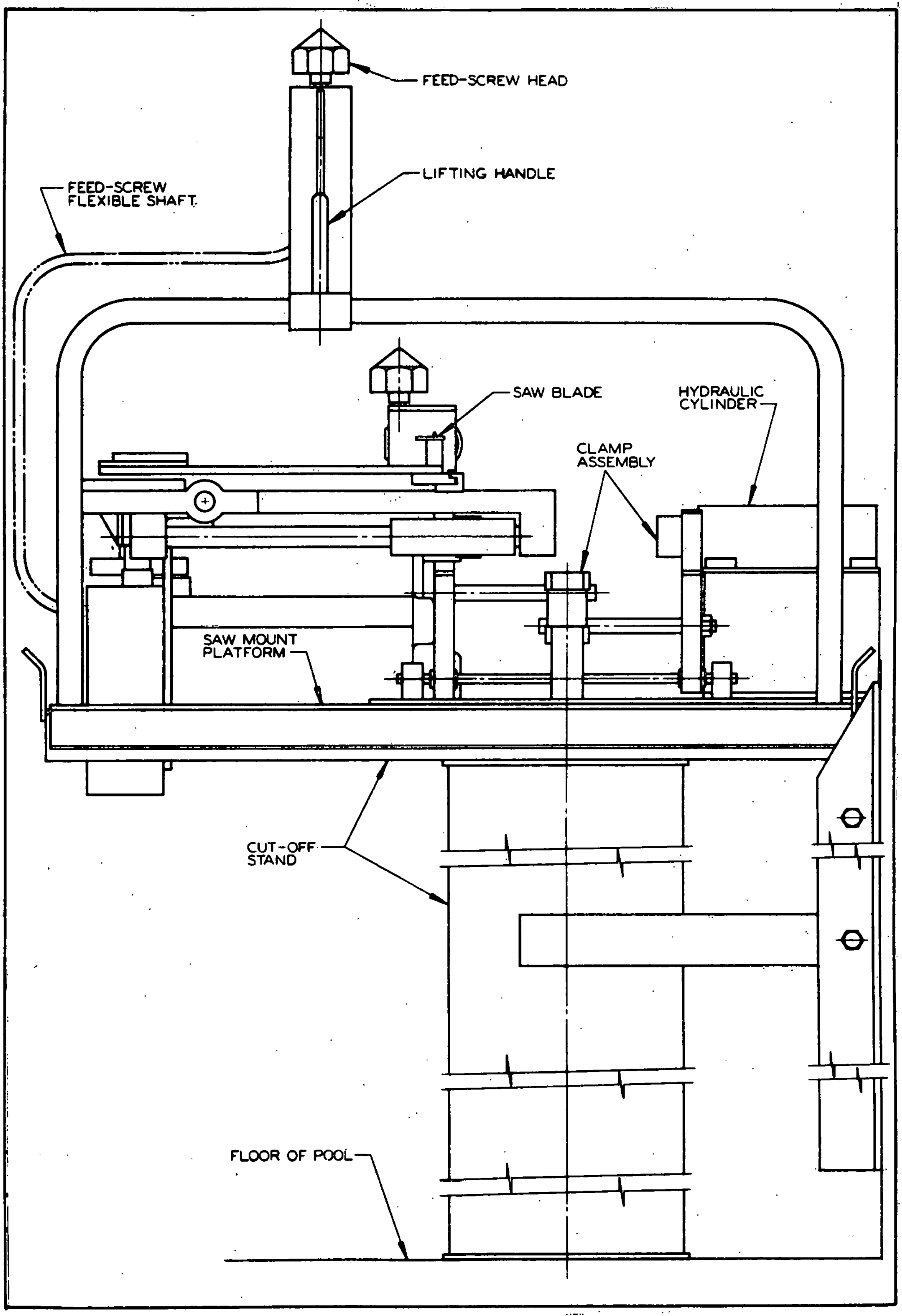


Fig. 1.2...Underwater Saw Facility - Vertical View (43-202-800)

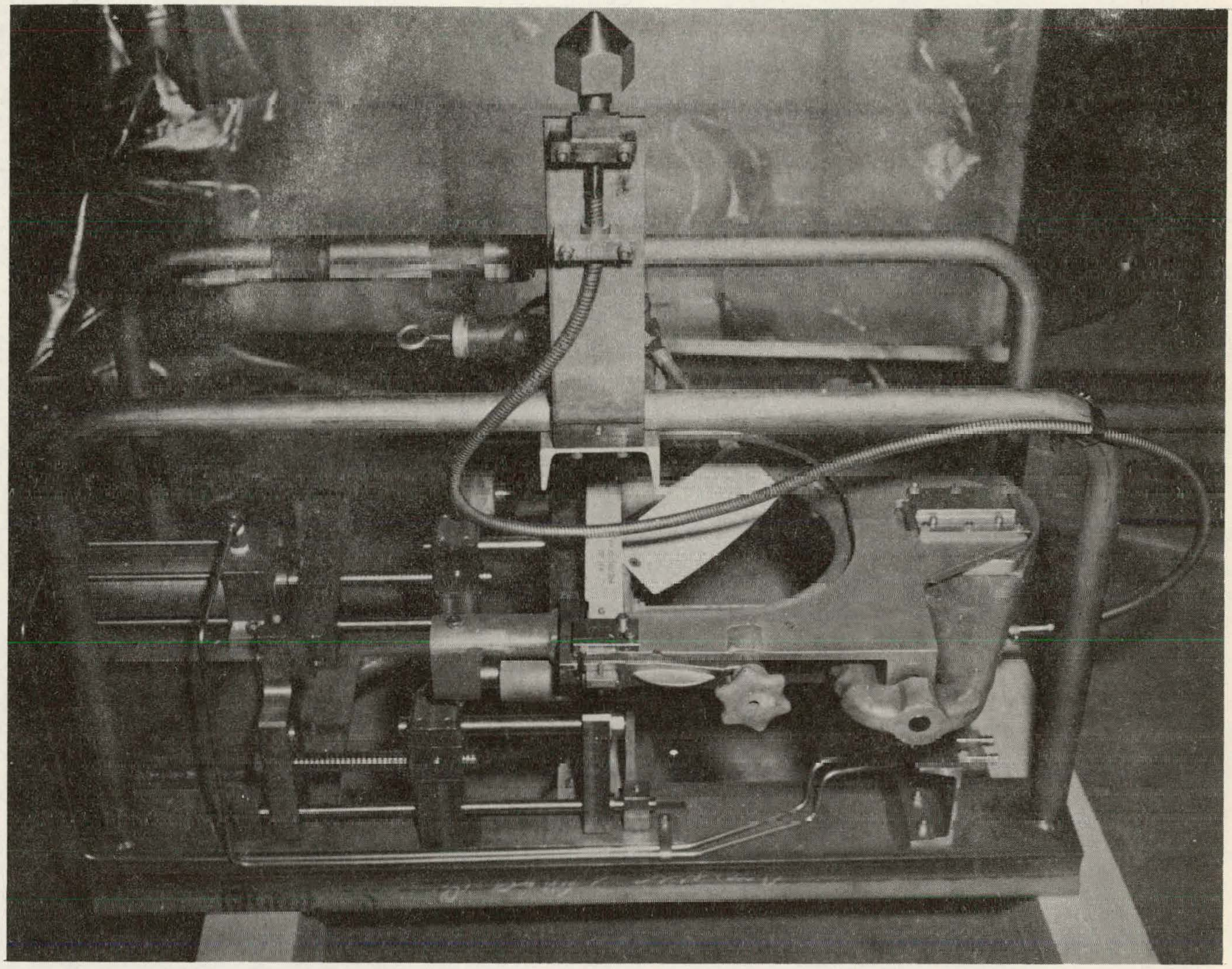


Fig. 1.3...Photo of Assembled Underwater Saw Unit

3. NUCLEAR ANALYSIS

3.1 REACTOR PHYSICS (STATICS)

The objective of this project is to perform physics calculations such as computer programming and operation, and to determine the critical mass, neutron and gamma flux and power distribution, enrichment, coefficients of reactivity, control rod effectiveness, and conversion ratio with respect to a Pathfinder core with an integral high enrichment superheater region. An additional objective is to determine shielding requirements for the Pathfinder plant.

3.1.1 PATHFINDER XENON TRANSIENT STUDIES

Because of the high microscopic absorption cross section σ_a (2200 m/sec) = 2.7×10^6 barns, the presence of Xe-135 in the reactor has a significant effect on the core reactivity. Xenon is produced in an operating reactor by direct fission product formation and by radioactive decay from other fission products. The primary contribution to the xenon production is from the decay of radioactive iodine. Approximately 0.061 atoms/fission of I-135 are formed, compared to 0.003 atoms/fission of Xe-135. Therefore, during reactor operation with changes in reactor power, the production of xenon remains somewhat constant until the iodine concentration changes to its new equilibrium level while the removal of xenon varies with the flux. In addition, the xenon concentration increases significantly after shutdown because the I-135 fission product continues to decay to form Xe-135 while the only removal of Xe-135 is from radioactive decay. The half-life of iodine is shorter than that of xenon, resulting in a xenon-concentration increase.

Of primary concern in reactor operation is the amount of reactivity controlled by the equilibrium xenon poison and by time-dependent changes to that poison; particularly at the end of core life, when large increases in the poison concentration could result in shutting down the reactor. However, in a boiling water reactor the reactivity lost due to the poison increase can be compensated for by the reactivity gain due to reduced void content and fuel temperature at lower power operation.

The performance of the Pathfinder reactor during these transient conditions has been analyzed.

Reactivity worth of the equilibrium xenon concentration had been previously determined using two-dimensional (RZ) diffusion theory calculations. For these calculations, the spatial variation of the equilibrium xenon concentration was computed using flux and power distributions obtained from a clean, hot, voided core with criticality achieved by a uniform poison in the boiler core. From these calculations, the equilibrium xenon was predicted to be worth -3.0 per cent $\Delta k/k$.

The spatial variation of the xenon concentration as described above would be different if criticality had been achieved by insertion of some of the outer eight boiler control rods, as will be the case during normal operation. Using perturbation analysis, the worth of equilibrium xenon concentration for this control configuration relative to a homogeneous poison control was determined to be only slightly dependent upon the control rod configuration. Because of this, the reactor performance during transient xenon conditions has been analyzed using a point reactor system normalized to the 3.0 per cent $\Delta k/k$ value.

To do this, a computer program was written for the IBM-1620 which solves the time-dependent xenon and iodine isotopic concentration equations as a function of reactor flux and power. Using the average reactor flux and power, these concentrations were calculated for reactor conditions as a function of time and previous reactor operating history. The average equilibrium xenon concentration at full power was calculated and the poison worth set equal to the previously calculated value of -3.0 per cent $\Delta k/k$. Changes to the concentration were then calculated, using the program and the poison worth normalized to the full power value.

A transient analysis was made on the reactor for the following conditions:

1. Normal reactor startup with no xenon or iodine present initially. Reactor is held at a constant power until equilibrium is obtained. This was done for four different power levels; 25%, 50%, 75%, and 100% full power.
2. Changes to reactor power level during operation. Reactor is at equilibrium condition prior to change of power level. Both increases and decreases to power operation were considered.
3. Time dependence of xenon poison after reactor shutdown from equilibrium condition for four power levels mentioned in item 1, above.
4. Startup procedure using low power operation from shutdown reactor condition. Reactor is shut down, and is brought to power with xenon poison increasing or at the peak value.

The following figures show the transient behavior for Pathfinder for the conditions discussed above.

Figure No.	Title
3.1	Xenon poison buildup to equilibrium during power operation.
3.2	Xenon poison behavior after decrease in reactor power from full power operation.
3.3	Xenon poison behavior after power change from 75 per cent power operation.
3.4	Xenon poison behavior after power change from 50 per cent power operation.
3.5	Xenon poison buildup after reactor shutdown.
3.6 & 3.7	Xenon poison behavior during reactor startup from peak poison condition.
3.8	Reactivity changes resulting from power operation in Pathfinder. (Include void and doppler effect.)

Several conclusions can be drawn from the results shown in Fig. 3.1 through 3.8. The time to reach the equilibrium poison condition will depend to some extent upon the power level of operation. Approximately 90 per cent of the equilibrium concentration is reached in 25 hrs at full power while it takes about 40 hrs at 25 per cent of reactor power, (see Fig. 3.1).

If the power level is reduced (such as shown in Figure 3.2), the xenon poison worth rises for the first four to eight hours and then drops to the new equilibrium value. Likewise, if the power level is raised, (such as shown in Fig. 3.3 and 3.4), the xenon poison worth drops for the first few hours and then rises to the new equilibrium value.

Near the end of core lifetime, if the increased value of the xenon poison after a power level decrease was greater than the reactivity gain (due to decreased power operation), the reactor would continue to shut down. To evaluate this mode of operation, Fig. 3.8 is also shown. Note that the reactivity gained in going from full to 75 per cent power is 0.45 per cent $\Delta k/k$, while the xenon poison worth only increases 0.29 per cent $\Delta k/k$. For a boiling water reactor, such as Pathfinder, the reactivity gain due to decreased power level operation more than compensates for the increased xenon poison value.

Immediately following reactor shutdown from an equilibrium condition, the xenon poison value rises for the first six to eight hours and then decreases toward the initial condition. The magnitude of this increased poison worth is strongly dependent upon the power level of operation from which the shutdown occurs (see Fig. 3.5). Within 24 hours after shutdown, the level drops below the equilibrium concentration.

It may be desirable to start the reactor up while the xenon poison level is near the maximum. Near the end of core life it would be impossible to override the peak xenon poison at full power, but the reactor could be operated at low power until the xenon is burned out and full power achieved within a reasonably short time. (See Figures 3.6 and 3.7). The operation represented by Figure 3.6 and Figure 3.7 is broken into three periods: operation to obtain the equilibrium poison condition at full power...reactor shutdown time, 4 hours for Figure 3.6 and 8 hours for Figure 3.7...and partial power operation for an additional 30 hours. In addition, the reactivity gain due to reduced power operation is given on the right side of both of these illustrations (Figs. 3.6 and 3.7). Reactivity is shown as the power level for which operation is possible if there is essentially zero excess reactivity at full power, and thus represents the end of life condition.

If the reactor is started up four hours after shutdown, between 25 per cent and 40 per cent of full power is possible. (See Fig. 3.6.) If it is operated at 25 per cent power, the power level can be raised to 50 per cent in 6 hours...and in another 9 hours or less, the full power level can be achieved. In both cases, full power can be reached in approximately 14 hours.

If the reactor is started up at the peak poison value, approximately 30 per cent of full power operation is possible 8 hours after shutdown. Fig. 3.7 shows the behavior if the reactor is operated at 25 per cent power and 40 per cent power. In both cases, the power level can be raised to 50 per cent approximately 3 hours after starting up...and in an additional 6 hours, full power can be achieved.

All of these calculations are subject to the limitation that they do not include the effect of spatial variations. Not only do they neglect the time-dependent spatial variation of the xenon concentration, but they neglect the spatial variation of the worth of various core regions.

An attempt to evaluate this effect was made. The spatial variation of the peak xenon concentration was computed using a two-dimensional PDQ-RZ calculation with partial insertion of the outer control rods. The peak xenon concentration and the equilibrium samarium concentration were then inserted into a clean PDQ-RZ calculation and the poison worth evaluated. This calculation yielded the poison reactivity worth as -5.6 per cent $\Delta k/k$. If equilibrium samarium is worth 0.8 per cent $\Delta k/k$, then the peak xenon concentration is worth -4.8 per cent $\Delta k/k$ after shutdown from full power operation. This excellent agreement has to be somewhat fortuitous. However, the one-point model results shown in Figures 3.1 through 3.7 should be adequate to describe the general reactor performance for the xenon transient behavior of the Pathfinder reactor.

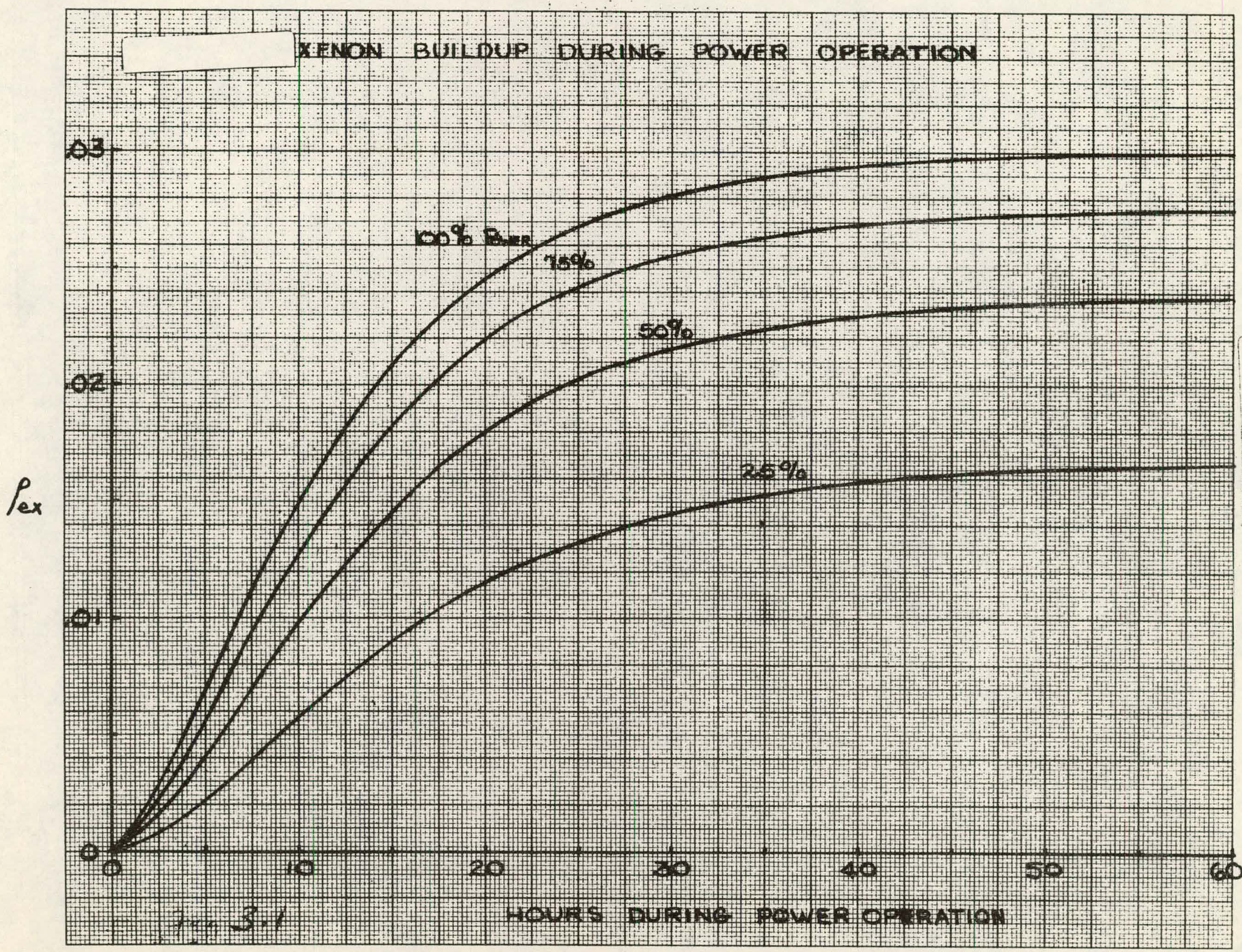


Fig. 3.1...Xenon Buildup During Power Operation

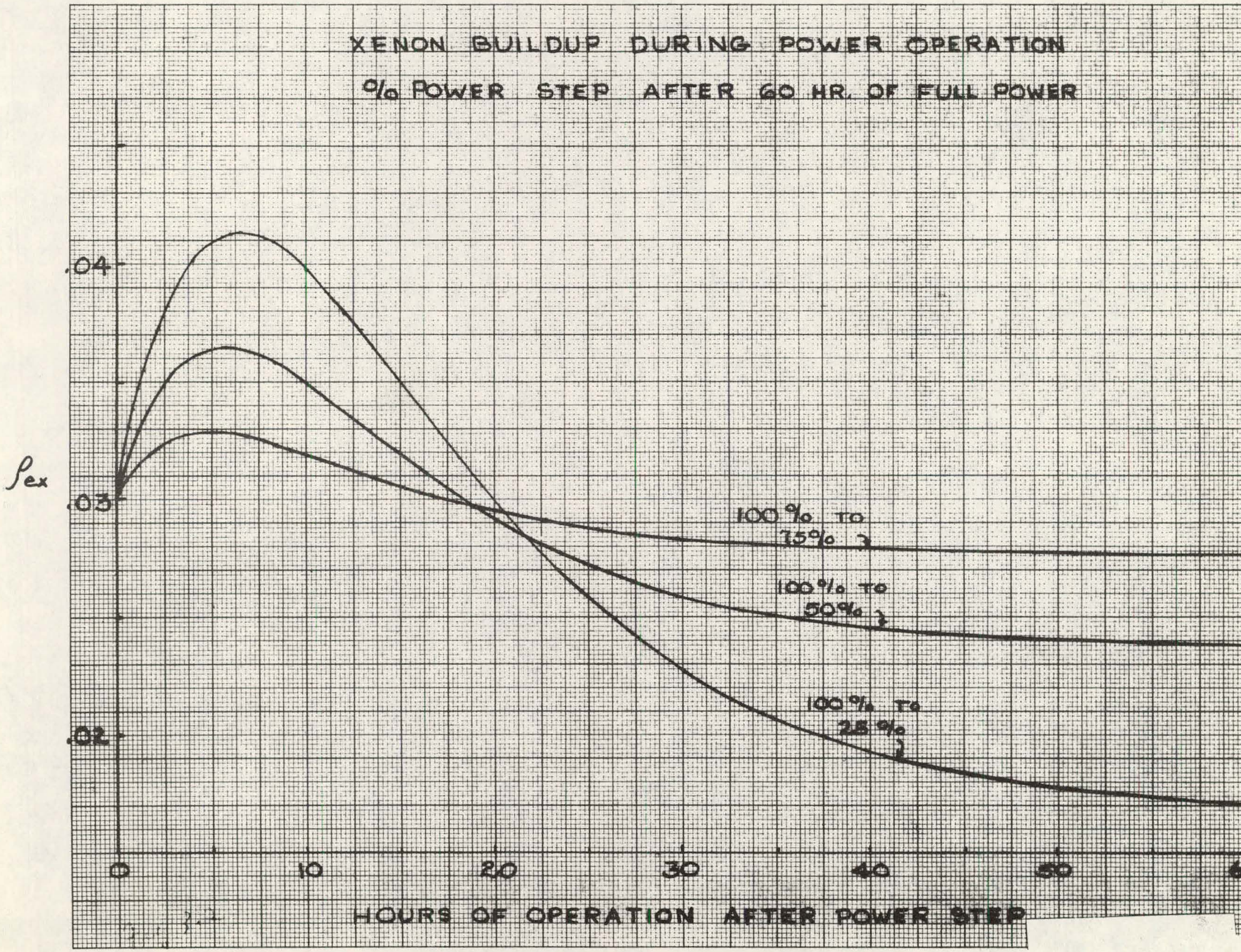


Fig. 3.2...Xenon Buildup During Power Operation, Per Cent Power Step After 60 Hr of Full Power

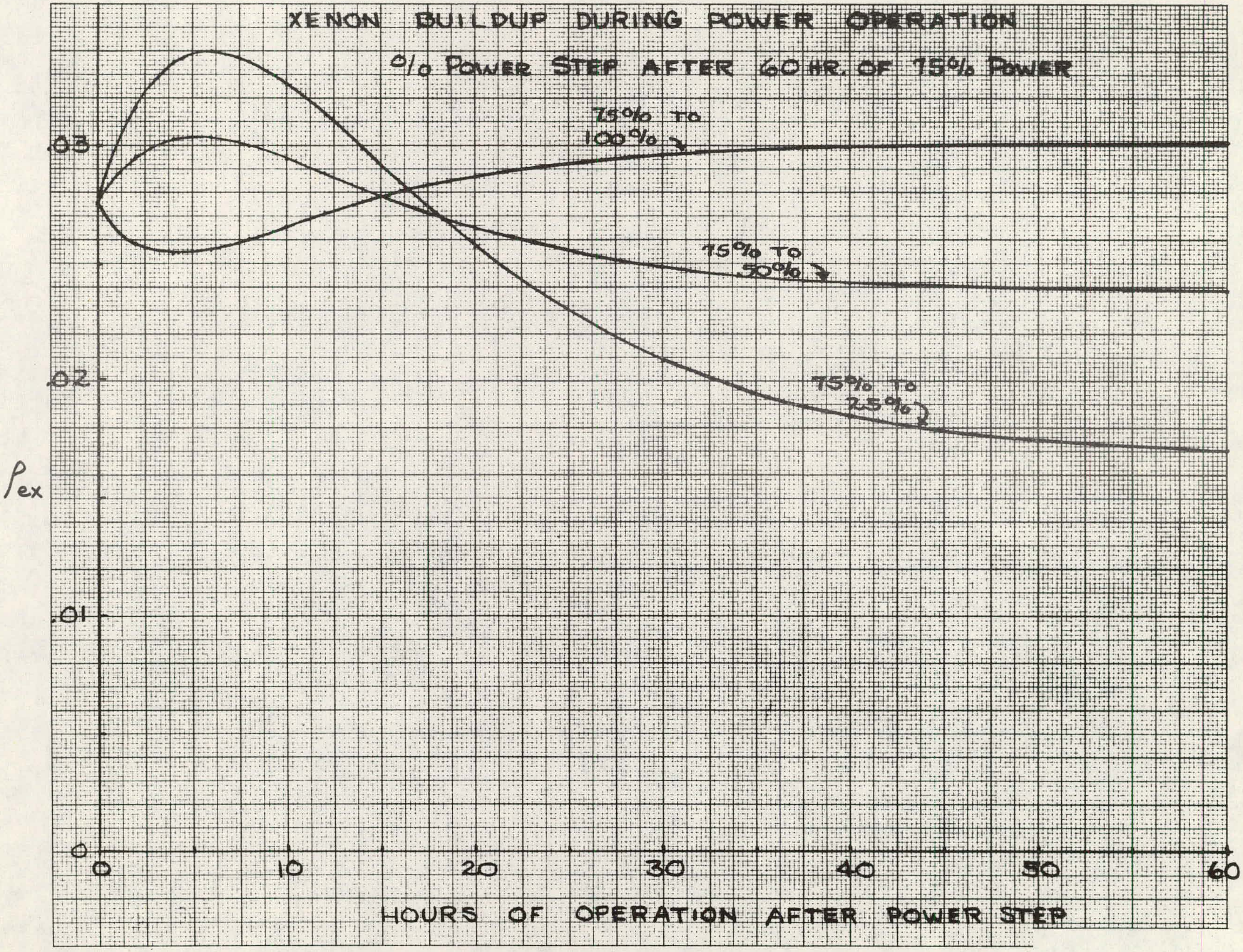


Fig. 3.3...Xenon Buildup During Power Operation, Per Cent Power Step After 60 Hr of 75% Power

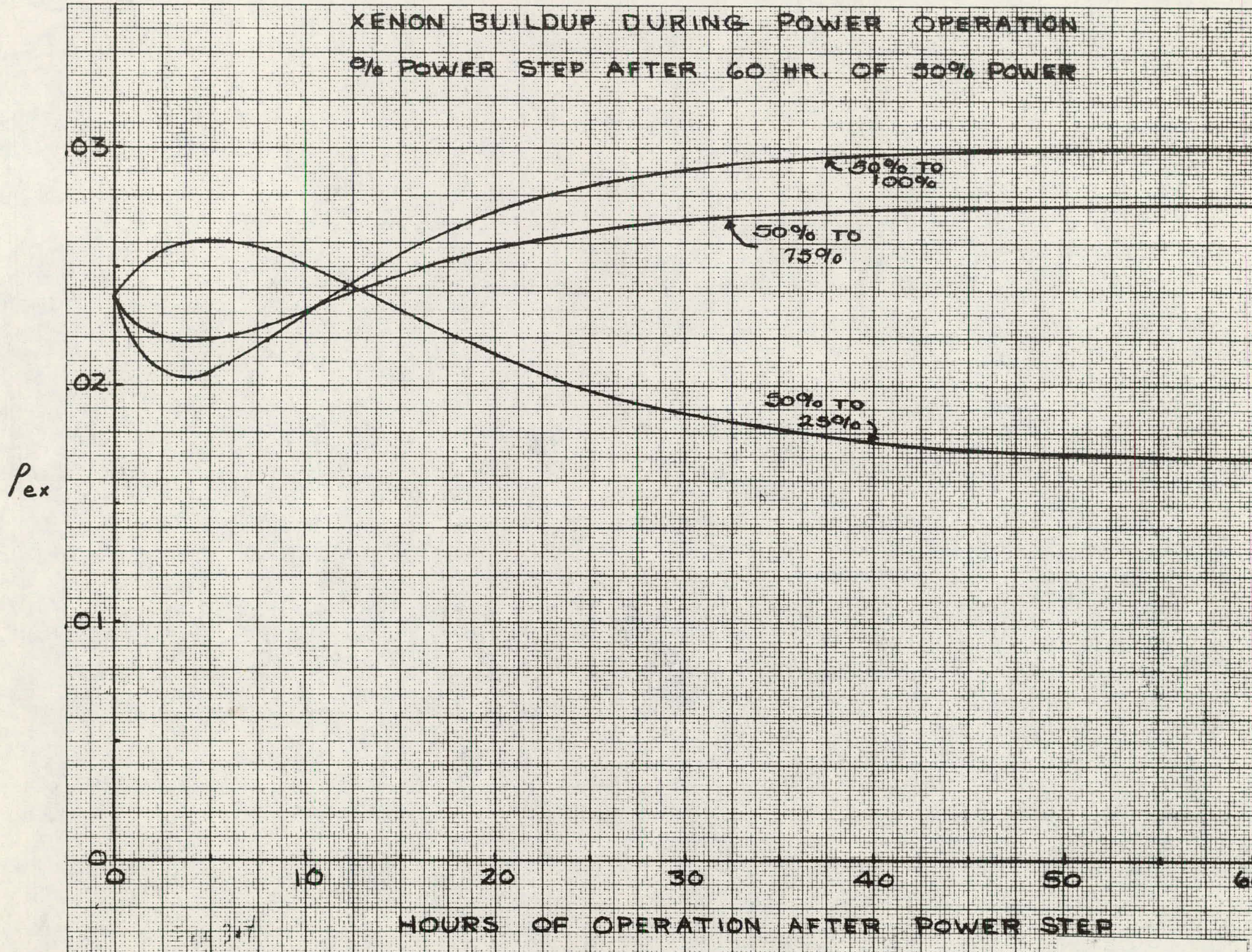


Fig. 3.4...Xenon Buildup During Power Operation, Per Cent Power Step After 60 Hr of 50% Power

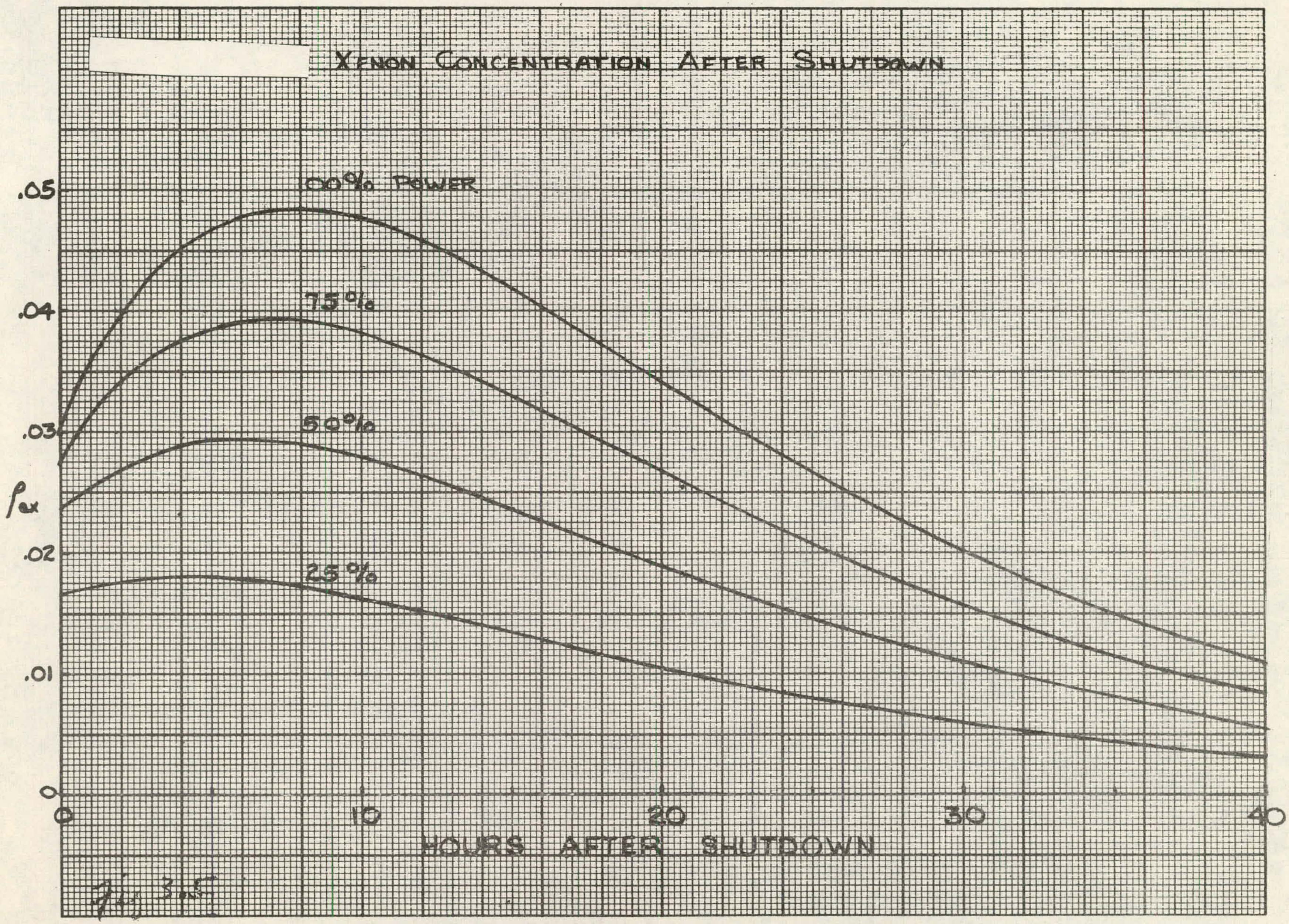


Fig. 3.5...Xenon Concentration After Shutdown

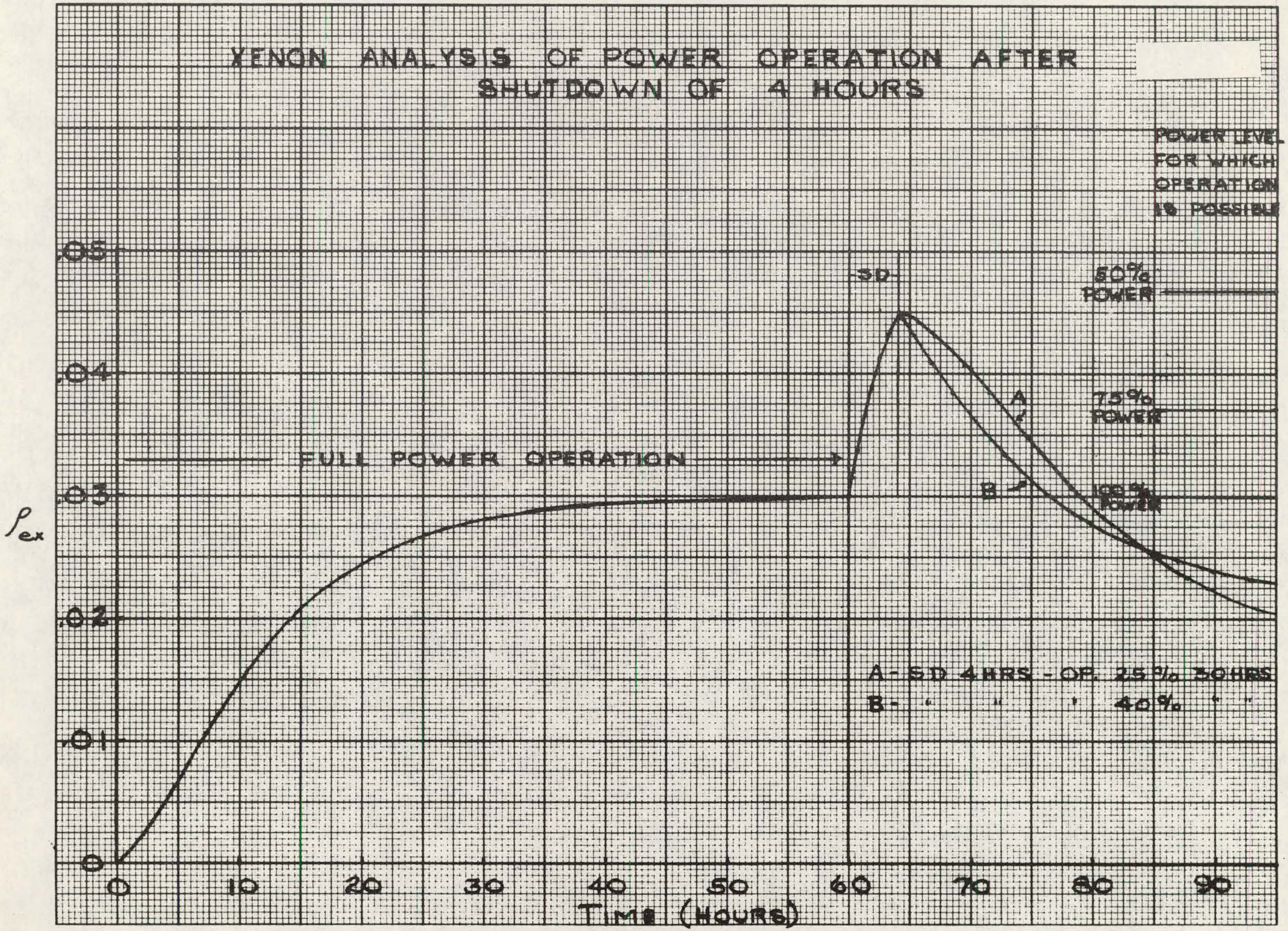


Fig. 3.6...Xenon Analysis of Power Operation After Shutdown of Four Hours

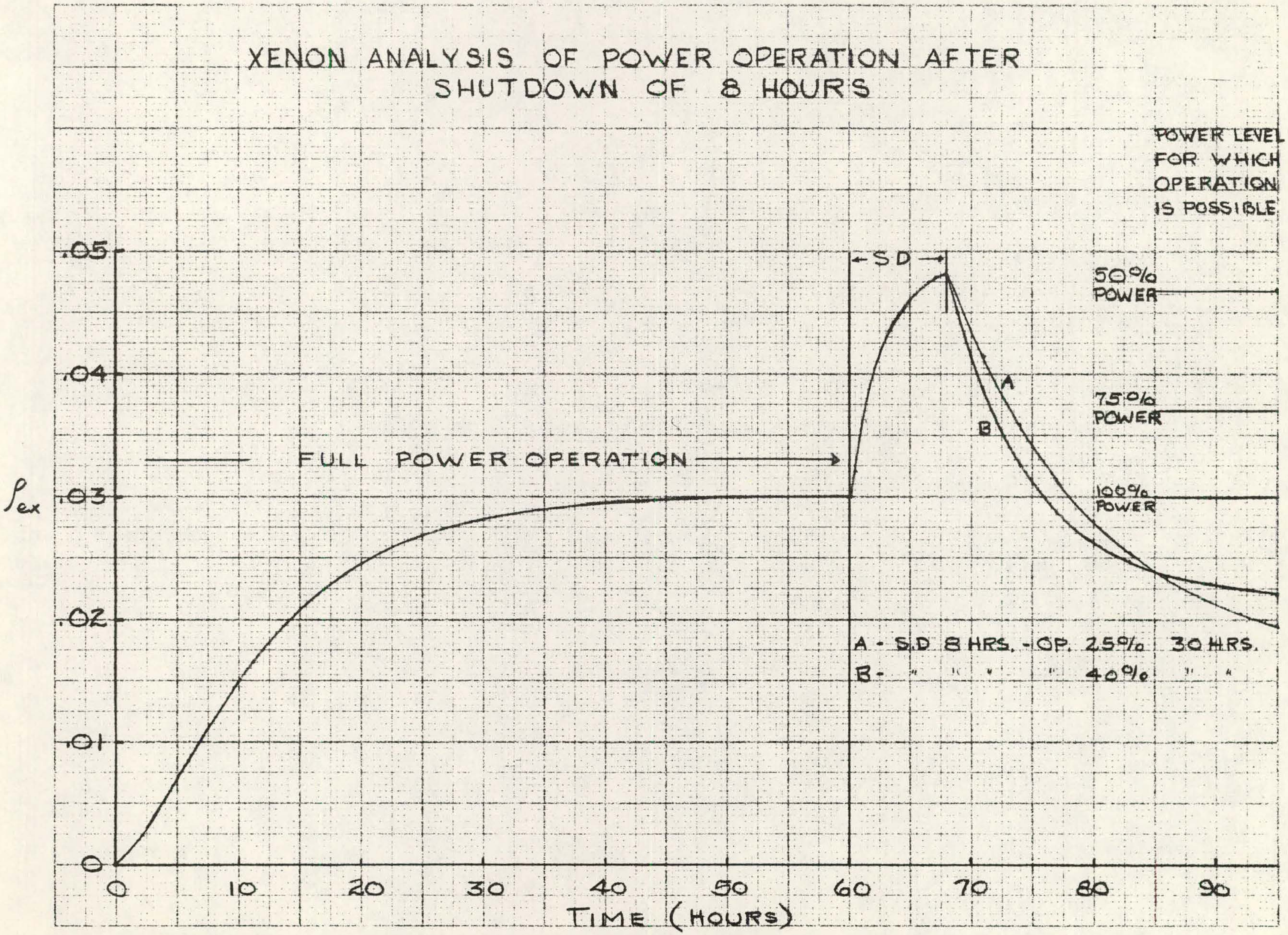


Fig. 3.7...Xenon Analysis of Power Operation After Shutdown of Eight Hours

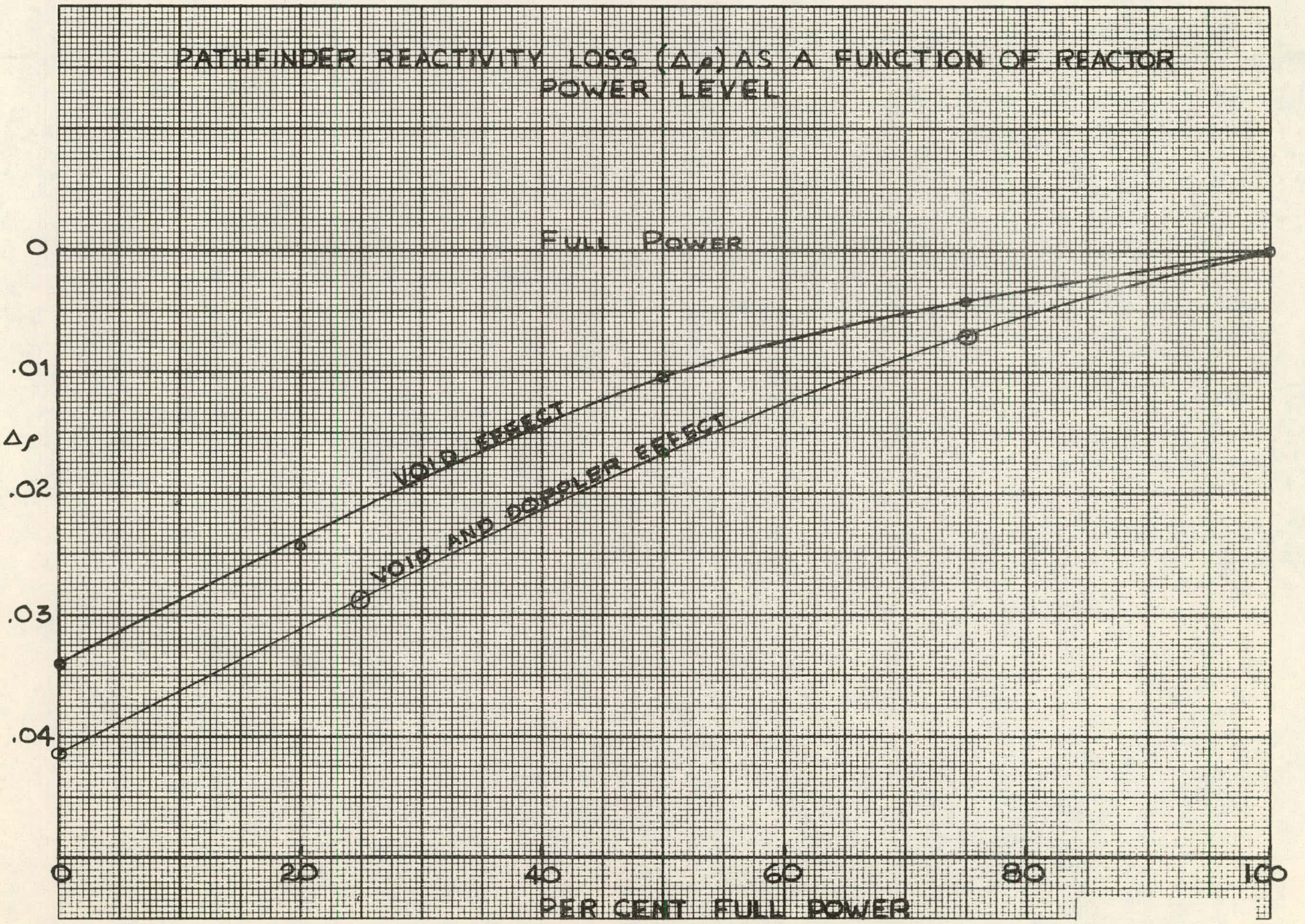


Fig. 3.8...Reactivity Loss as a Function of Reactor Power Level

PART B

POST CONSTRUCTION R&D

1.0 INITIAL STABILITY AND PERFORMANCE TESTS ... POST CONSTRUCTION R & D

The objectives of this project are to design and fabricate a special oscillator rod and drive mechanism together with suitable instrumentation and recording equipment to measure and record the resulting variations in neutron flux, and to conduct oscillator tests to verify dynamic performance calculations and to determine experimentally the stability of the reactor system. In addition, certain measurements of power, flows, pressures, and temperatures will be made to determine the transient response of the reactor system. The data will be analyzed to determine certain dynamic parameters.

1.0.1 DYNAMICS

Test procedures for pile-oscillator tests and the fluid dynamic reactivity tests were written and analyzed during this quarter. Each test procedure has been carefully scrutinized so that the reactor safety system will be closely monitoring all important parameters throughout the tests.

A test signal junction box was built to provide a convenient input for the multi-channel recorder that will be used during the tests. Inputs that will be brought to this box and recorded are: any three of the nine in-core ion chambers...feedwater temperature...recirculation flow rate... reactor pressure...reactor steam flow...recirculation water temperature ...superheater exit temperature...three of the superheater fuel thermocouples. Any six of these signals can be simultaneously recorded on a six-channel Offner recorder.

4.0 MECHANICAL TESTS ... POST CONSTRUCTION R & D

Objective of this project is to check the reactor and associated systems for proper operation under simulated operating conditions in order to insure that the plant is operational prior to introduction of fuel into the reactor.

4.0.1 PRE-CRITICAL TESTING

4.0.1.1 Introduction

Pre-critical tests have been grouped according to the necessary reactor conditions for the tests (see Table 4.1), starting with open vessel tests and progressing to tests with the reactor brought to temperature and pressure by the auxiliary startup heater. The pressure drop through the core, required for operation of the recirculation pumps, was obtained by means of an orifice plate placed over the boiler grid plate.

4.0.1.2 Reactor Internals

With the reactor vessel open, the reactor internals were checked for vibration with recirculation pumps in operation. No vibration of any consequence was noted. The water surface was checked to see if there was significant gradient between the water level in the center of the core and that in the downcomer. The water surface was found to be quite flat. The relative velocity of water through the grid plate was measured at a representative number of nozzles. This is being analyzed and correlated with the air test results reported in ACNP-5920 (Pathfinder Topical Report).

Table 4.1 - SEQUENTIAL TEST SCHEDULE

Pre-critical Tests...Condition 1

Test 111 - Liquid Level for Cold Critical

Test 114 - Boiler Core Flow Distribution

Test 112 - Water Level Profile

Test 113 - Check of Reactor Internals

Pre-critical Tests...Condition 2

Test 116 - Alignment of Ion Chamber Guide

Test 121 - Hydrostatic Test of Control Rod Drives

Test 231 - Cold Pressure Control

  Test 122 - Stud Cover Seal Check

Pre-critical Tests...Condition 3

Test 232 - Hot Pressure Control

  Test 131 - Check of Hot Pressure Seals

  Test 132 - Check of Vessel Movement

  Test 133 - Vessel Heating and Cooling Rate - (Heating)

  Test 135 - Recirculation Pump Movement

  Test 224 - Butterfly Valve Cavitation

  Test 134 - Vibration of Recirculation Piping

  Test 222 - Recirculation Flow Rate

Test 251 - Emergency Condenser Cooling - Flooded

Test 215 - Superheater Cooling Procedure

Test 233 - Flash Tank Nozzle Capacity

  Test 234 - Maximum Permissible Blowdown Rate

  Test 235 - Flash Tank Moisture Carryover

Pre-critical Tests...Condition 3 (continued)

Test 236 - Normal Cooler Performance

Test 237 - Purification System Temperature Control

Test 223 - Recirculation Pump Cavitation

Test 238 - Superheater Draining Procedure

Test 252 - Emergency Condenser Cooling - Voided

Test 261 - Feedwater Temperature Control

Test 275 - Steam Dump Valve Calibration

Test 136 - Liquid Level Calibration

Test 239 - Shutdown Procedure Check

Test 133 - Vessel Heating and Cooling Rate - (Cooling)

Pre-critical Tests...Condition 4

Test 151 - Operation of Control Rod Drives - Cold

Test 151 - Operation of Control Rod Drives - Hot

4.0.1.3 Stud Cover Seal

The reactor vessel was closed after the previous test was completed. It was then hydrostatically tested, and the stud cover seal was checked for self sealing while the shield pool was being flooded. The seal was accomplished without any difficulty.

4.0.1.4 System Tests

The pressure control system was checked, first with the reactor cold and then with the reactor hot. The system controlled the reactor pressure as expected.

The recirculation system was checked for proper operation with the reactor at temperatures of up to 420 F, and the pump and butterfly valve were checked for cavitation. The data is being analyzed; however, no severe cavitation was detected.

The purification system and emergency cooling system were operated at simulated operating conditions. The results are being analyzed but appear to be satisfactory.

Additional tests are still to be performed prior to the first fuel loading. They will be reported in the next quarter, together with the results of tests from this quarter.

6.0 PHYSICS ... POST CONSTRUCTION R & D

An initial startup program for Pathfinder has been specified. Test procedure specification and pre-testing analysis are currently in progress. In subsequent quarters, the results of the pre-testing analysis, experiments, and final analysis will be covered in Pathfinder Quarterly Reports. In this current report, the principal physics tests, their purpose, and the experimental procedure are briefly described.

6.0.1 INTRODUCTION

As a part of the Pathfinder Post Construction R & D Program, control rod drop experiments will be performed to determine the reactor shutdown margin. Rod drops will be performed as a function of core fuel and poison loading and moderator temperature. These core conditions are briefly described in Section 6.0 of ACNP-63027, Pathfinder Quarterly Report, April-June 1963.

6.0.2 ROD DROP CALCULATIONS

A survey of literature on rod drop techniques was completed and calculations were performed to provide two sets of rod drop curves for the Pathfinder reactor. The two cases calculated were for: a) boiler only...b) cold boiler plus superheater cores. These curves were calculated using the RE 126 code for the IBM 704 used by Toppel,[1] and an IBM 1620 code to perform the necessary integrations. The 1620 code also computed the Schulz constant for the cases investigated. The delayed neutron parameters were obtained using the MUFT 5 code.

Effective delayed neutrons fractions for six groups were obtained by correcting the data of Keepin[2] for the presence of U-238 and the

(1) B.J. Toppel, "Sources of Error in Reactivity Determination by Means of Asymptotic Period Measurements", N.S. and E.; 5, pp. 88-98, 1959.

(2) G.R. Keepin, "Neutron Data for Reactor Kinetics", Nucleonics, Vol. 20, No. 8, pp. 150-151, 1962.

appropriate delayed neutron importance applicable to the cores considered. The delayed neutron importances were developed on a basis of leakage considerations. These calculations were performed for four Pathfinder cores: boiler only...cold boiler plus superheater...slab...and, hot boiler plus superheater. The delayed fractions for the six groups for the boiler only and cold boiler plus superheater are shown in Table 6.0.1. The total delayed neutron fractions for all cores are shown in Table 6.0.2.

Table 6.0.1 EFFECTIVE DELAYED NEUTRON FRACTIONS

Boiler Only

Group	Bi
1	0.000270
2	0.00156
3	0.00141
4	0.00307
5	0.00104
6	0.000230

Cold Boiler Plus Superheater

Group	Bi
1	0.000258
2	0.00149
3	0.00134
4	0.00293
5	0.000990
6	0.000220

TABLE 6.0.2 EFFECTIVE DELAYED NEUTRON FRACTIONS

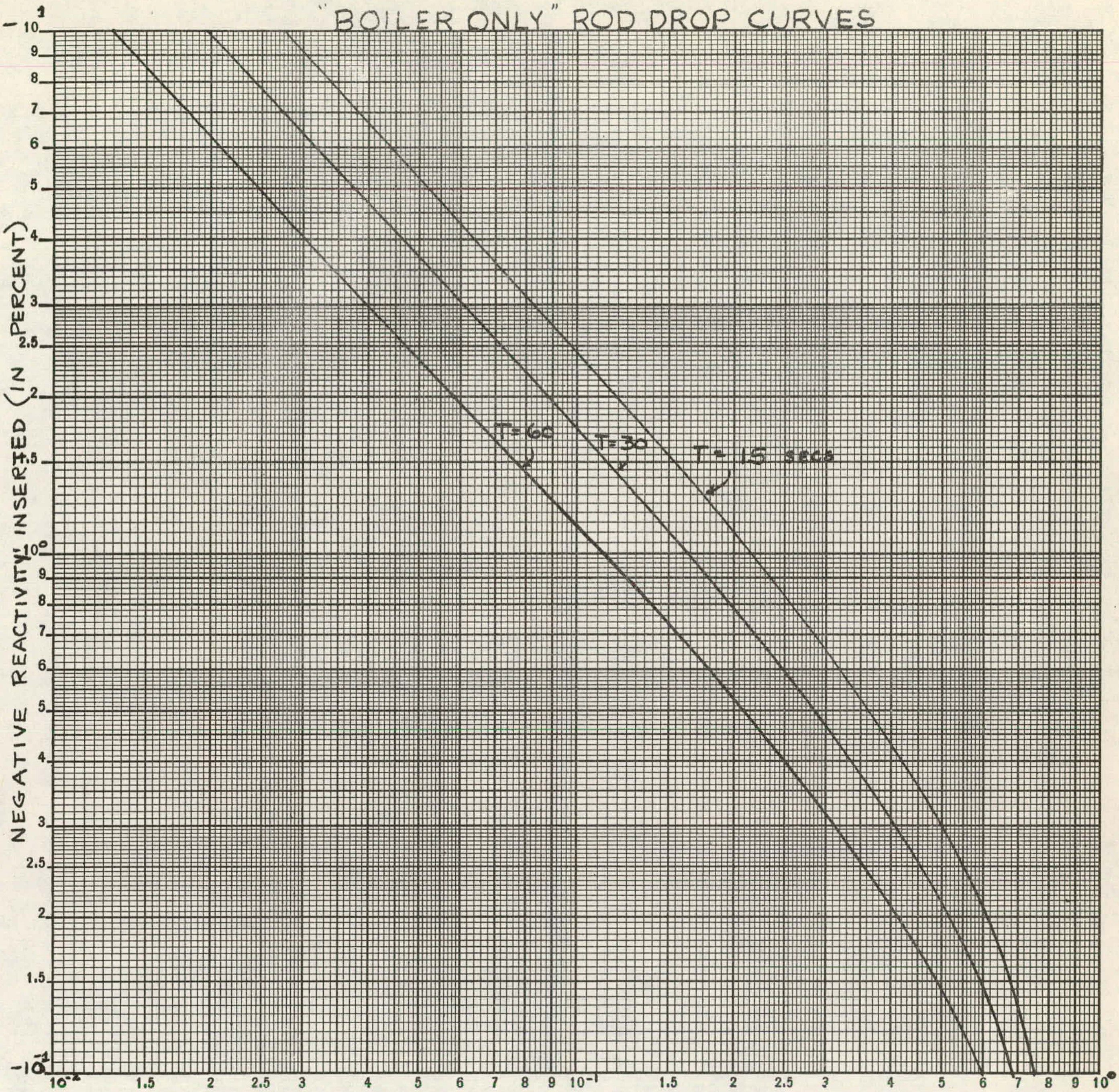
Core	B
Boiler Only	0.0076
Cold Boiler Plus Superheater	0.0072
Slab	0.0082
Hot Boiler Plus Superheater	0.0071

Utilizing the data of Table 6.0.1 and delayed neutron decay constants for U-235, the neutron densities after step reactivity insertions were computed for the boiler only and the cold boiler plus superheater cores. The 1620 code was then used to form the averages of these densities from a time of 0.9 seconds after the step insertion (anticipated Pathfinder rod insertion time) to 15, 30, and 60 seconds after. These calculations are shown in Figures 6.0.1 and 6.0.2. These curves can be directly used with scaler integrations of chamber pulses to obtain the reactivity inserted. Since actual rod insertions are not step functions of reactivity, some inaccuracies will exist for small values of rho.

Fig. 6.0.3 shows the data of Fig. 6.0.2 on a linear scale. It can be seen that the value of the reactivity inserted becomes more sensitive to the scaler accuracy as larger values of negative reactivity are approached. Scaler integrations at Pathfinder will start from a high neutron level at which no counting losses or core heating effects are expected to exist.

Some of the Schulz constants computed by the 1620 code for the cold boiler plus superheater core are shown in Fig. 6.0.4. The functions appear to be double valued if rho becomes smaller than about 3 per cent $\Delta k/k$.

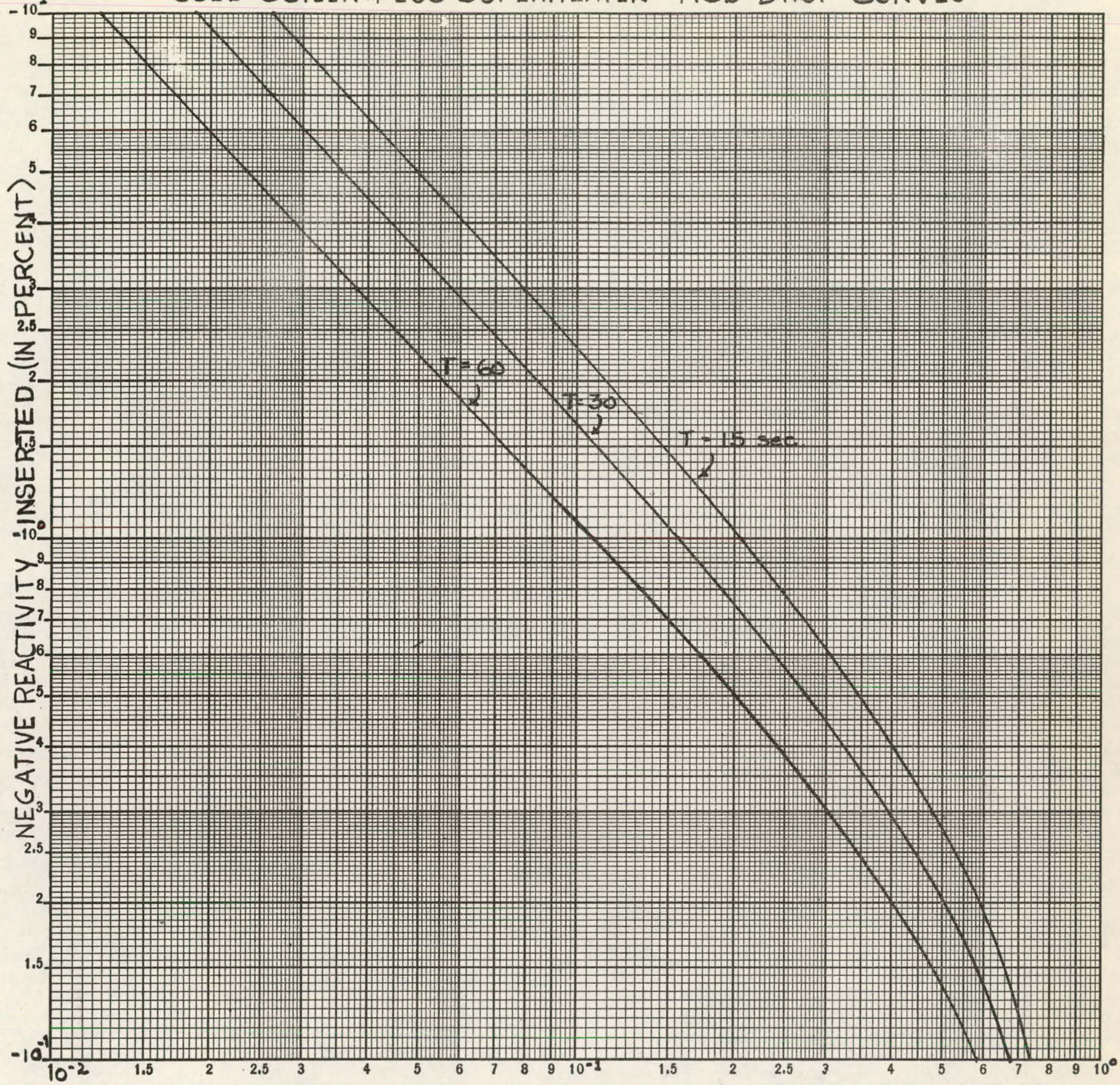
Rod drop measurements are to be performed on the Pathfinder core using chambers located in the vessel for the cold clean case. Measurements will also be obtained using the normal startup channels in the radial biological shield for the core in the cold clean condition...in the hot (420 F) clean condition...and in the hot (420 F) xenon poisoned condition. For drops of numbers of rods, the rods will be arranged in an all-in or all-out configuration as nearly as possible, to minimize the longitudinal perturbation of the core flux after rod insertion. These patterns will then be rotated in front of the detectors to obtain a determination of the radial sensitivity of the chamber decalibration.



$$\frac{\text{POST DROP INTEGRATED COUNTS}}{\text{PRE DROP INTEGRATED COUNTS}} = \frac{\int_{.9}^{T+.9} n_1(t)\,dt}{\int_0^T n_0(t)\,dt}$$

Fig. 6.0.1..."Boiler Only" Rod Drop Curves

"Cold Boiler plus Superheater" Rod Drop Curves



$$\frac{\text{POST DROP INTEGRATED COUNTS}}{\text{PRE DROP INTEGRATED COUNTS}} = \frac{\int_{.9}^{T+.9} n_1(t)\,dt}{\int_0^T n_0(t)\,dt}$$

Fig. 6.0.2..."Cold Boiler Plus Superheater"Rod Drop Curves

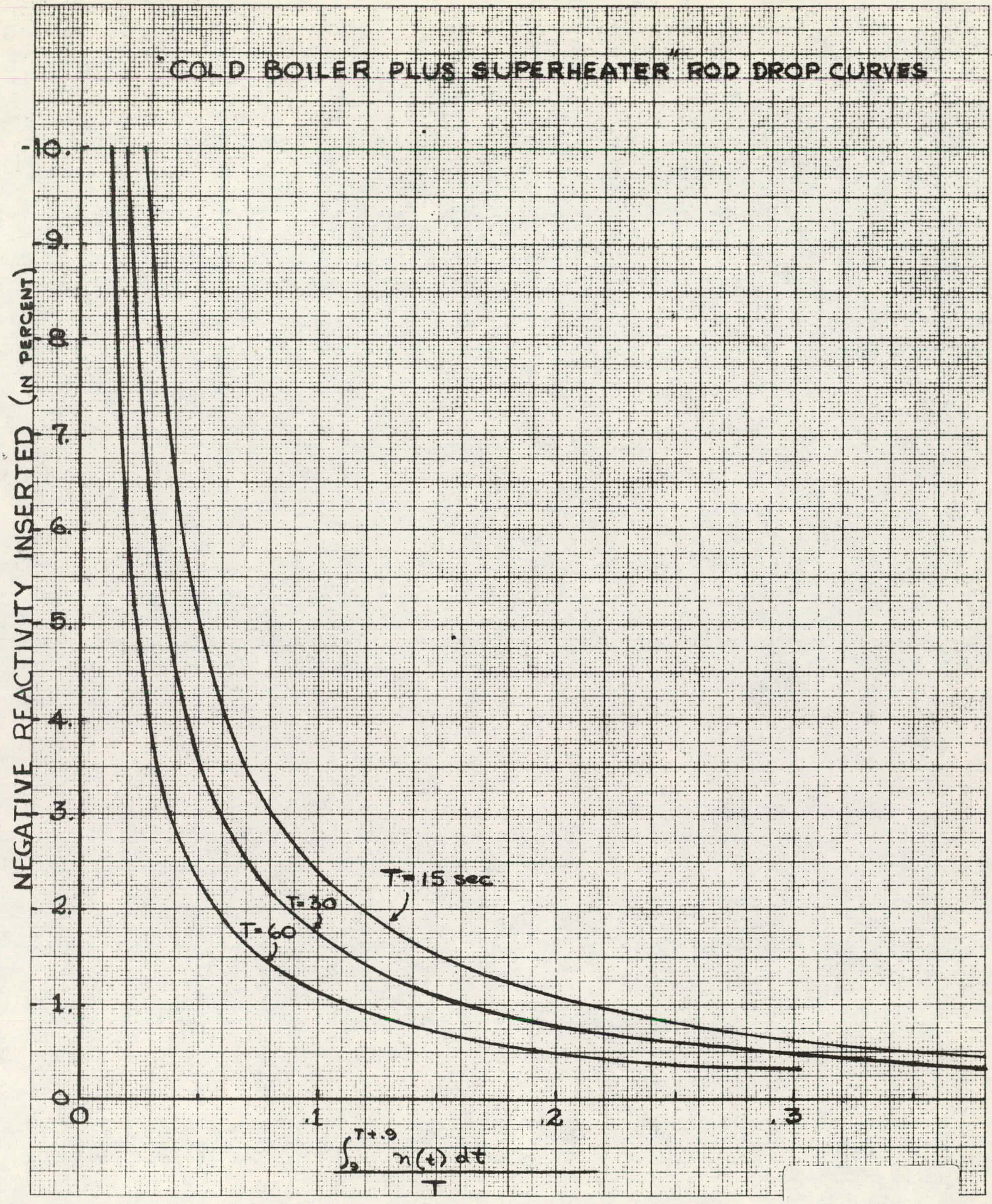


Fig. 6.0.3...Linear Plot of "Cold Boiler Plus Superheater" Rod Drop Curves

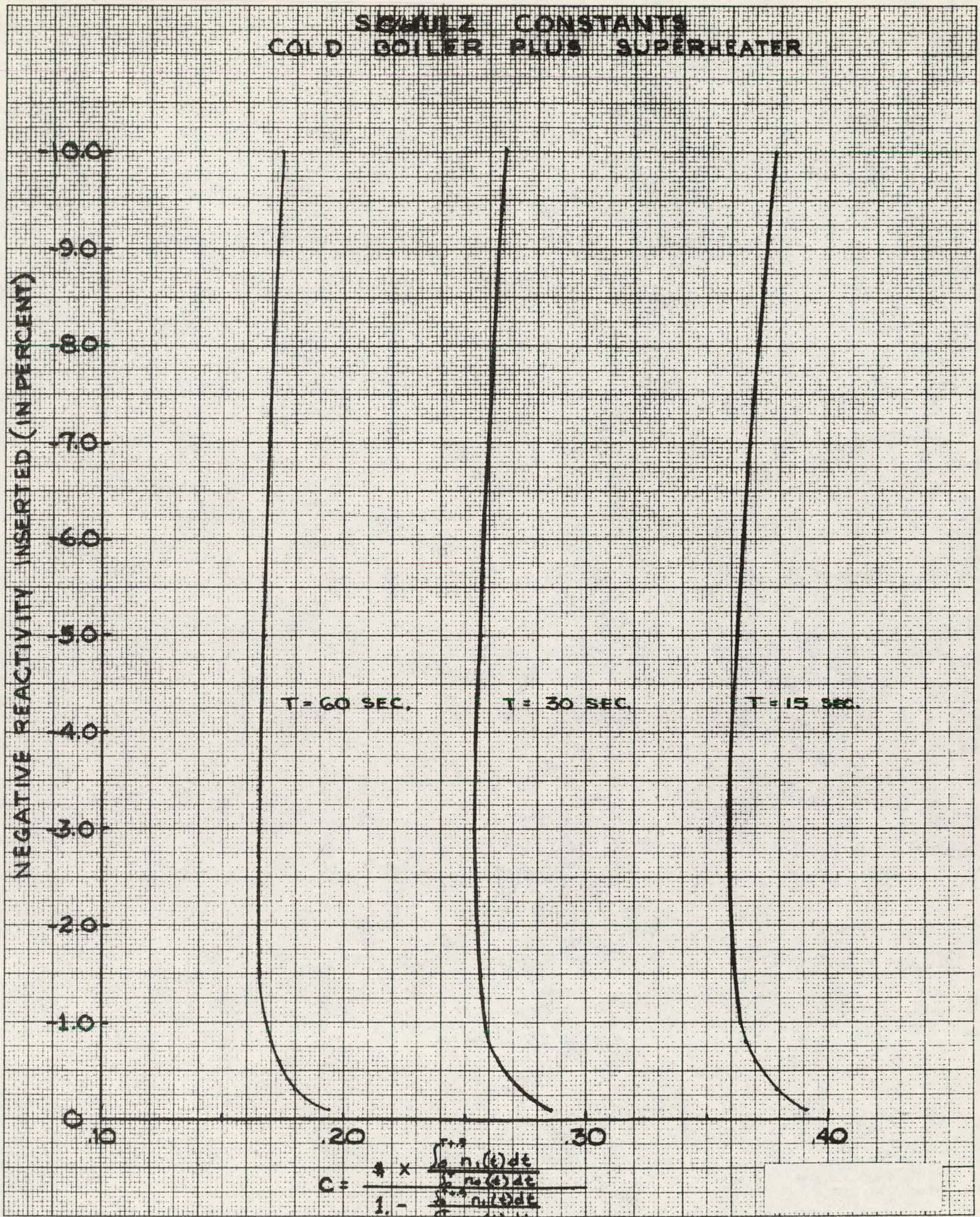


Fig. 6.0.4...Schulz Constants "Cold Boiler Plus Superheater"

8.0 CORE PERFORMANCE TESTS...POST CONSTRUCTION R & D

The objective of this project is to design the instrumented superheater fuel element assemblies. These assemblies, together with suitable recording equipment, will be used to record the temperature at various locations on the superheater fuel elements. Temperatures will be monitored during both low power (zero steam flow) and full power operations, and will be used to determine shutdown power conditions in the superheater and general superheater core performance during all reactor operating phases.

8.0.1 INSTRUMENTED SUPERHEATER FUEL ELEMENTS

8.0.1.1 Purpose of Elements

The instrumented superheater fuel elements have a twofold purpose:

1. They will be used to determine the ability of superheater fuel elements to cool themselves by thermal radiation and conduction at very low power and zero steam flow.

2. They will be used to obtain temperatures at various spots on the fuel assemblies, and also the exit steam temperature, during full power operation.

8.0.1.2 Instrumentation Design and Test Conclusions

The instrumented superheater fuel element assembly consists of fuel and poison tubes with thermocouples attached...a thermocouple connector pad ...thermocouple wires and a flexible wire rope to form the connection medium. Fig. 8.0.1 shows the instrumented assembly.

Initial assembly designs included ten thermocouple wires. Three wires were arranged around the periphery of each poison and fuel tube (3 assemblies), replacing the spacing wires. The tenth wire was intended to measure the outlet steam temperature.. This wire ran down the inside of the poison tube, with the poison pellets slotted to accommodate this wire.

Sealing this thermocouple wire into the center tube turned out to be prohibitively expensive and time consuming. The center thermocouple was therefore eliminated, and the top poison rod end fitting was simplified. Fig. 8.0.1 illustrates the current design.

As stated above, the thermocouples will replace the spacing wires in the element. One of the wires on the inner fuel tube will be extended to the spacer fin region and will be used to measure the outlet steam temperature. The hot junctions are located at various axial points so as to determine temperatures at those points.

As finally designed, the lifetime of the instrumented assemblies is expected to be sufficient to meet test objectives.

MK. 503 REQ.	MK. 502 REQ.	MK. 501 REQ.	IT.	DESCRIPTION	MATERIAL	PART NUMBER DRAWING	PART NUMBER MK.	WT.
/	/	/	36	THERMOCOUPLE WIRE		THIS	036	
/	—	—	37	THERMOCOUPLE WIRE		THIS	037	
/	—	—	38	THERMOCOUPLE WIRE		THIS	038	
/	—	—	39	THERMOCOUPLE WIRE		THIS	039	
/	—	—	40	THERMOCOUPLE WIRE		THIS	040	
/	—	—	41	THERMOCOUPLE WIRE		THIS	041	
/	/	/	42	EXTENSION TUBE-M/F ½ O.D.×#13(.095)WALL×78LG. TUBE	18-8 SSTL LOW CARBON	THIS	042	
1	1	1	43	END SPACER-M/F 5/8" DIA.×½" LG. BAR	316L SSTL LOW CARBON	THIS	043	
1	1	1	44	END CAP-M/F ½ DIA.×5½ LG. BAR	316L SSTL LOW CARBON	THIS	044	

MK. 503 REQ	MK. 502 REQ.	MK. 501 REQ'D	IT.	DESCRIPTION	MATERIAL	PART NUMBER DRAWING	PART NUMBER MK	WT
—	—	/	1	INSTRUMENTED SUPERHEATER FUEL ELEMENT ASSEMBLY		43-501-185	501	
~~1~~	~~1~~	~~1~~	~~2~~	~~½ O.D.× ... LG.~~	18-8 SSTL. LOW CARBON			
1	1	1	3	SUPERHEATER FUEL ELEMENT-43-500-881 MK 501 MODIFIED		THIS	003	
1	1	1	4	END FITTING-M/F 1¼ DIA.×5/8 LG. BAR	18-8 SSTL. LOW CARBON	THIS	004	
1	1	1	5	BRACKET-M/F ¼ TH'K×1¼×2 5/8 LG. BAR	18-8 SSTL. LOW CARBON	THIS	005	
1	1	1	6	GUIDE TUBE-M/F 1 DIA×3¼ LG BAR	18-8 SSTL. LOW CARBON	THIS	006	
~~1~~	~~1~~	~~1~~	~~7~~	~~SPACER-M/F 3/8 O.D.×#16GA(.065)×1 LG. TUBE~~	~~18-8 SSTL. LOW CARBON~~	~~THIS~~	~~007~~	
1	1	1	8	EXIT TUBE-M/F ½ OD×#16GA.(.065)×2¾ LG	18-8 SSTL. LOW CARBON	THIS	008	
1	1	1	9	LOCATOR RING-M/F 1⅛ DIA.×3/8 LG.	18-8 SSTL. LOW CARBON	THIS	009	
2	2	2	10	RING KEY-M/F 3/8 SQ×3/8 LG.	18-8 SSTL. LOW CARBON	THIS	010	
~~1~~	~~1~~	~~1~~	~~11~~	~~SNAP RING (WALDES 5100-50H)~~	ARMCO PH15-7 MO STN.STL	~~THIS~~	~~011~~	
6	6	6	12	MIDDLE FIN-M/F #19GA.(.044)×2×1¼ LG SHEET	18-8 SSTL. LOW CARBON	THIS	012	
YES	—	YES	13	LOWER THERMOCOUPLE ARRANGEMENT		43-401-392	501	
—	YES	—	14	LOWER THERMOCOUPLE ARRANGEMENT		43-401-392	502	
~~YES~~	~~YES~~	~~YES~~	~~15~~	~~ASSEMBLY PROCEDURE GUIDE~~		~~43-101-798~~	~~401~~	
—	YES	YES	16	LOCATION OF THERMOCOUPLE JUNCTION		43-401-477	401	
YES	—	—	17	LOCATION OF THERMOCOUPLE JUNCTION		43-401-477	402	
~~1~~	~~1~~	~~1~~	~~18~~	~~END CAP-M/F ½ DIA.×1½ LG. BAR~~	~~316L SSTL. LOW CARBON~~	~~THIS~~	~~018~~	
1	1	1	19	⅛ DIA (7×19)× LG WIRE ROPE ASS'Y-SA 155-4(¼-20 NC-2A TH'D) ONE END & SA 155-4(¼-20 NC-LH TH'D) OTHER END (MACWHYTE CAT'G 5801)	STN. STL ANNEALED	THIS	019	
—	/	—	20	INSTRUMENTED SUPERHEATER FUEL ELEMENT ASSEMBLY		43-501-185	502	
AS REQ'D	AS REQ'D	AS REQ'D	21	WELD WIRE (ASTM A371-55T CL. ER 308	STN. STL	THIS	021	
4	4	4	22	#5-40 UNC-3A×3/16 LG CUP PT. HDLESS HEX. SOC. SET SCREW	STN. STL TP-303	THIS	022	
YES	YES	YES	23	PRODUCT SPECIFICATION		43-101-819	401	
~~1~~	~~1~~	~~1~~	~~24~~	~~THERMOCOUPLE WIRE~~		~~THIS~~	~~024~~	
–	/	/	25	THERMOCOUPLE WIRE		THIS	025	
/	—	—	26	INSTRUMENTED SUPERHEATER FUEL ELEMENT ASSEMBLY		43-501-185	503	
3	3	3	27	CLIP-M/F #22 GA(.031)×¼×1½ LG. SHEET (INCONEL 600)	ASTM B 168 ANNEALLED	THIS	027	
2	2	2	28	CLIP-M/F #22 GA(.031)×¼×1 LG. SHEET (INCONEL 600)	ASTM B 168 ANNEALLED	THIS	028	
/	/	/	29	THERMOCOUPLE WIRE		THIS	029	
–	/	/	30	THERMOCOUPLE WIRE		THIS	030	
–	/	/	31	THERMOCOUPLE WIRE		THIS	031	
/	/	/	32	THERMOCOUPLE WIRE		THIS	032	
/	/	/	33	THERMOCOUPLE WIRE		THIS	033	
–	/	/	34	THERMOCOUPLE WIRE		THIS	034	
–	/	/	35	THERMOCOUPLE WIRE		THIS	035	

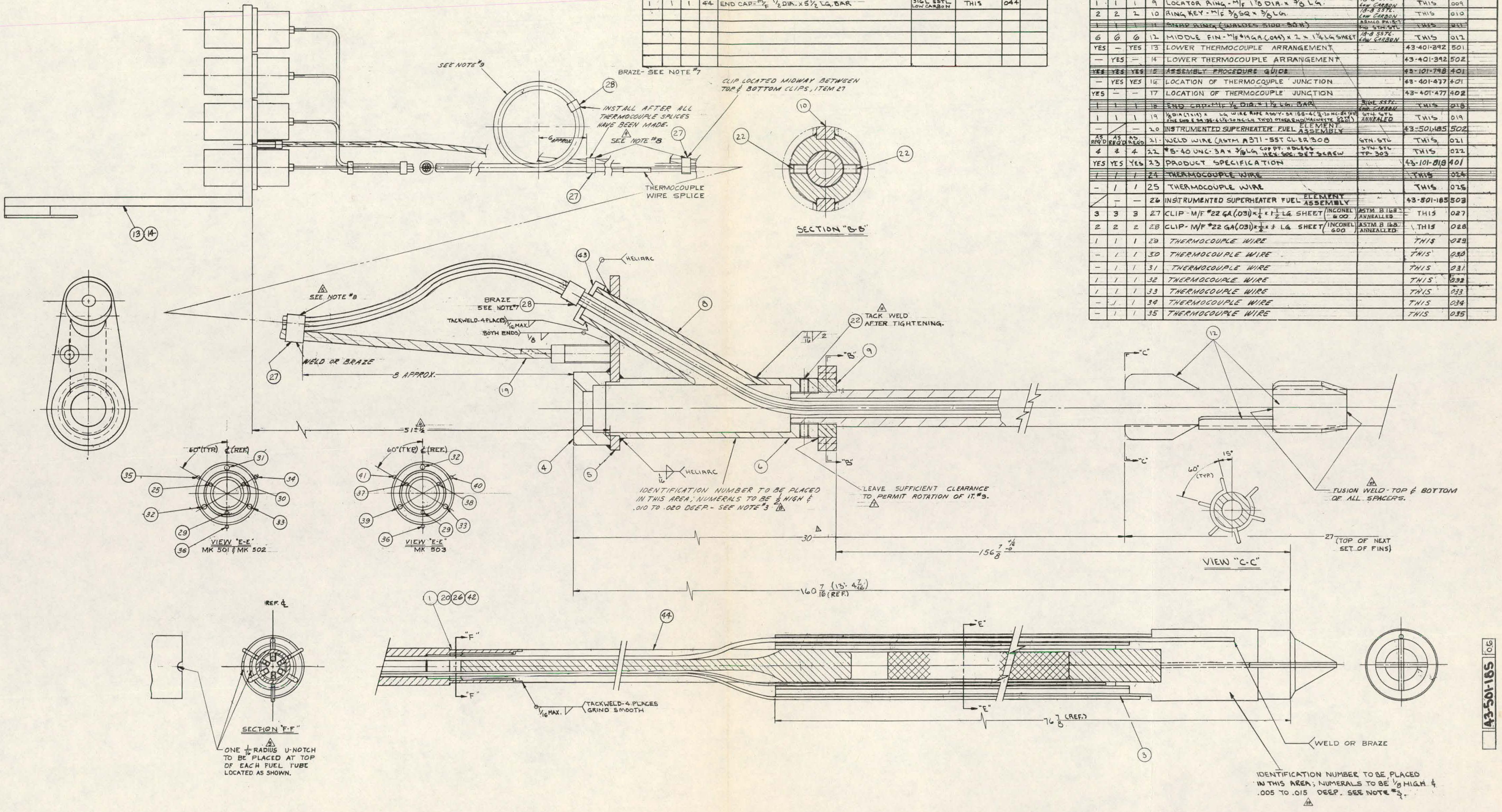


Fig. 8.0.1...Instrumented Superheater Fuel Element Assembly (43-501-185)