MND-M-1815
REACTORS—POWER

PM-1 NUCLEAR POWER PLANT PROGRAM 4TH QUARTERLY PROGRESS REPORT [FOR] DECEMBER 1, 1959 TO FEBRUARY 29, 1960

By
E. H. Smith

April 5, 1960

Nuclear Division
Martin Company
Baltimore, Maryland

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PM-1 Nuclear Power Plant Program

4TH QUARTERLY PROGRESS REPORT

1 Dec 1959 to 29 Feb 1960

Contract AT(30-1)-2345
5 April 1960

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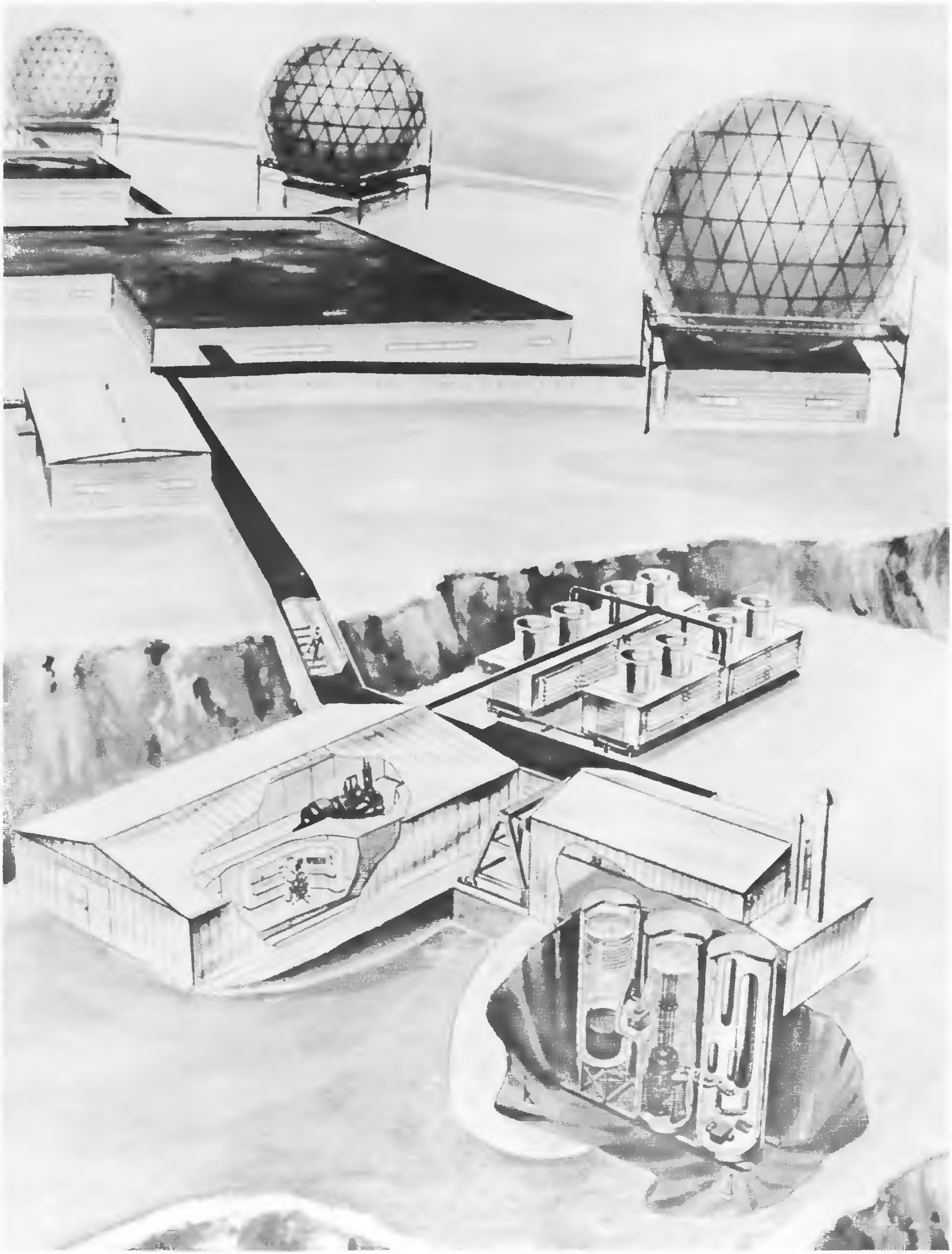
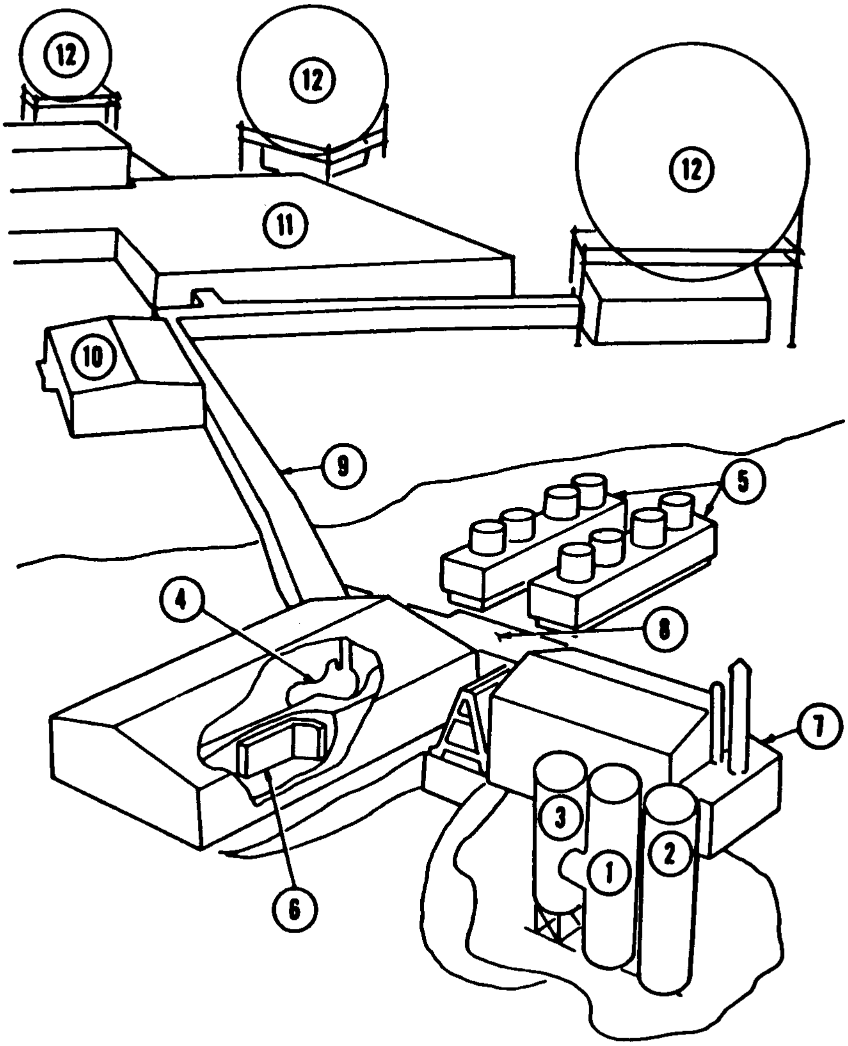


KEY TO PM-1 DRAWING

1. Reactor Tank
2. Steam Generator Tank
3. Spent Fuel Storage Tank
4. Steam Turbine and Electric Generator
5. Air Steam Condensers
6. Control Console
7. Shield Water Cooler
8. Decontamination and Water Chemistry Laboratory
9. Covered Walkway
10. Base Technical Supply Building
11. Base Operations Building
12. Radar Installations



ABSTRACT

This report contains a description of the work accomplished during the fourth contract quarter (1 December 1959 to 29 February 1960) of Contract AT(30-1)-2345 between The Martin Company and the USAEC.

The objective of the contract is the design, development, fabrication, installation and initial testing and operation of a prepackaged air-transportable pressurized water reactor nuclear power plant, the PM-1. The specified output is 1 Mwe and 7 million Btu/hr of heat. The plant is to be operational by March 1962.

The principal efforts during the fourth project quarter were the near-completion of the final design and preparation of specifications for plant components. The entire power plant has been divided, for final design purposes, into 37 subsystems. The status of work on each subsystem at the close of the period is reported. A revised summary of design parameters is given, with flow diagrams of the primary and secondary systems. The final design work will be concluded early in the next report period.

Systems development work included full-scale structural testing of a test package, preparation for a loading demonstration and testing of the air-steam condenser model at Eglin Air Force Base.

Reactor development work included:

(1) Final preparations for the flexible zero-power test (PMZ-1) program.

(2) Final preparations for the revised fuel element irradiation test program.

(3) Continuation of ractor flow tests. One-fourth-scale model flow tests were completed and preparations for bundle flow and full-scale tests were continued.

(4) Further work on the heat transfer test program, including fabrication, installation and testing of test section STTS-3 and fabrication of test sections STTS-4 and SETCH-2.

(5) Final design and fabrication of the prototype magnetic jack-type control rod actuators.

Core fabrication continued with delivery of additional UO_2 and production runs of fuel elements for the zero-power test, PMZ-1. Work continued on study of control rod materials, with emphasis on fabrication techniques and stabilization of rare earth oxides.

For the preceding period, see MND-M-1814.

FOREWORD

This is the fourth quarterly progress report submitted to the U.S. Atomic Energy Commission under Contract AT(30-1)-2345. It covers the work accomplished by The Martin Company on the PM-1 Project for the period from 1 December 1959 through 29 February 1960.

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PROGRAM HIGHLIGHTS

1. The test package was fabricated and tests initiated. The loading demonstration arrangements have been made for the last week of April. (Subtask 1.1)

2. The development effort on the reactor control system was completed. (Subtask 1.5)

3. The air-steam condenser was completed, shipped to Eglin Air Force Base and tested in the climatic chamber at temperatures as low as -65° F. (Subtask 1.6)

4. All hardware for the PMZ-1 zero power test was fabricated, the Hazards Summary Report was submitted to the AEC licensing branch, and fuel elements were produced in order to realize a scheduled criticality date of the last week of March. (Subtasks 2.1 and 5)

5. Plans and negotiations continued satisfactorily for fuel element irradiations. (Subtask 2.2)

6. 1/4-scale reactor model flow tests were completed. Heat transfer tests continued, and preparations were made for full-scale reactor flow tests. (Subtasks 2.3 and 2.4)

7. The majority of the plant final design subsystems were submitted to the AEC and completion of this effort is scheduled for the second week of March (Task 4.0)

8. Procurement of major plant components continued as scheduled. (Task 7.0)

9. A vendor was selected and work initiated for the PM-1 site foundations which are to be installed this summer. (Subtask 11.1)

10. The PM-1 Plant Hazards Summary Report was approved by the AEC. (Subtask 17.1)

PM-1 NUCLEAR POWER PLANT DESIGN SUMMARY

A. REACTOR DESIGN CHARACTERISTICS

1. Overall Performance Data

Pressurizer water, nominal operating pressure (psia)	1300
Design pressure for heat transfer analysis (psia)	1200
Design pressure for structural analysis (psia)	1485
Average core coolant temperature, nominal (°F)	463
Reactor thermal power, nominal (Mw)	9.37
Reactor thermal power, design (Mw)	10.31
Core life, nominal (Mw-yr)	18.74

2. Core Design Characteristics

Geometry, right circular cylinder (approximately)		
Diameter, average (in.)	23.6	←
Active length (in.)	30	
Overall length of fuel tube (in.)	33-1/4	
Core structural material	Modified ASTM 304 and 347	←
Fuel element data, tubular, cermet type		
Outside diameter (in.)	0.500	
Inside diameter (in.)	0.416	
Clad thickness (in.)	0.006	
Clad material	AISI Type 347 stainless steel, modified, 0.01 wt % Co maximum, 0.03 wt % Co plus Ta maximum	←

*Denotes new items added to major plant parameters.
←Denotes change in plant parameter from previous submission.

U-235 loading/tube (gm), nominal,	39.4 ± 4%
averaged over core	39.4 ± 2%
Number	732
Meat composition, wt % UO_2 (nominal)	28
Burnable poison element data, unclad, cylindrical, boron stainless steel, alloy type	
Outside diameter (in.) varies to compensate for actual boron loadings obtained	0.500/0.475
*Basic poison material	ASTM Type 304 stainless steel, 0.01 wt % Co maximum, 0.03 wt % Co plus Ta maximum
Boron loading (natural) in grams of B^{10} per rod in stainless steel alloy	0.640 ± 2%
Number	75
Control element data, Y-shaped, cermet type	
Arm length--total overall from pickup ball centerline (in.)	38-3/8
--active (in.)	30
Arm width--total (in.)	3-7/8
--active poison (in.)	3-1/2
Arm thickness (in.)	5/16
Clad thickness (in.)	0.030
Clad material	AISI Type 347 stainless steel, modified, 0.05 wt % Co, maximum; 0.15 wt % Co plus Ta, maximum.
Poison element	Europium compound dispersed in stainless steel (equivalent to 30 wt % Eu_2O_3)

Number	6
*Nuclear worth of six control rods % $\Delta\rho$	-31.8
*Nuclear worth of five control rods % $\Delta\rho$	-16.5
*Nuclear worth of four rods (minimum worth) % $\Delta\rho$	-9.3
*Minimum shutdown control margin (approximately mid-life, two of six rods stuck in operating condition) % $\Delta\rho$	0.4
*Average thermal core flux	
Initial, $n\theta$	0.7×10^{13}
At 2 years, $n\theta$	1.4×10^{13}
*Average temperature coefficients	
Overall (65° to 463° F) $\Delta\rho$/°F	1.2×10^{-4} (initial); 0.9×10^{-4} (at 2 years)
Operating temperature $\Delta\rho$/°F	2.1×10^{-4} (initial); 1.9×10^{-4} (at 2 years)

3. Core Heat Transfer Characteristics

Heat flux (Btu/ft^2-hr)	
Average	73,000
*Heat flux (Btu/ft^2-hr) (maximum)	290,000
Average coolant temperature (°F)	463

4. Reactor Hydraulic Characteristics

Coolant flow rate (gpm)	2125

B. SYSTEMS DESIGN

1. General Plant

Reactor power output, nominal (Mw)	9.37
Steam generator power output, nominal (Mw)	9.37
Steam pressure, full power, minimum (psia) (saturated)	300
*Steam pressure, zero power, maximum (psia)	485
Steam quality, full power, maximum	1/4% moisture

2. Main Coolant System

Number of coolant loops	1
Coolant flow rate (gpm)	2125
Coolant system design pressure (psig)	1485
Coolant velocity in piping (main loop) (fps)	26
Coolant pipe size, main loop (in.), nominal	
Schedule 80	6
System basic material	
Reactor pressure vessel	AISI 347
Piping	AISI 316
Remainder	AISI 304
Main coolant pumps	
Pumps, number (canned rotor type)	1
Steam generator	
Number of units	1
Design pressure (shell side) (approximately) (psi)	600

Type	Vertical with integral steam drum and separators
Temperature, primary inlet, full power (approximately) (°F)	479
Temperature, primary outlet, full power (approximately) (°F)	447
Temperature, steam side outlet, full power (°F)	417
Access	Shell and tube side bolted
Tube material	Inconel

3. Pressurizing and Pressure Relief System

Number of pressurizers	1
Type	Steam
Temperature, normal (°F)	577
Pressure, normal (psia)	1300
Pressure element (decreasing)	Water spray head
Pressure element (increasing)	Electric immersion heaters

4. Coolant Purification and Sampling System

Number of purification loops	1
Purification device	Ion exchange resin
Inlet temperature to ion exchanger (maximum) (°F)	120
Maintenance provisions	Recharge with fresh resin

5. Primary Shield Water System

Primary shield water cooler	Air blast type
Purification loop	Ion exchange resin
Maintenance provisions	Recharge with fresh resin

C. SECONDARY SYSTEM

1. General Plant

Steam flow, full power (lb/hr)	34,312
Steam flow, turbine, full power (straight condensing) (lb/hr)	26,253
Steam flow to evaporator-reboiler, full load (lb/hr)	7859
Steam pressure, full power, dry and saturated (psia)	300
Feedwater flow, full power (lb/hr)	34,512
Rated gross electrical output, 0.8 pf(kw)	1250
Net electrical output, 0.8 pf(kw)	1000
Line voltage	4160/2400
Cycles	60
Phases	3
Auxiliary equipment voltage	480
Process heat, 6815 lb/hr of 35 psia dry and saturated steam (Btu/hr)	7×10^6
Design elevation (ft)	6500
Auxiliary power (approximately) (kw)	135

2. Turbine- Generator Set

Type	Horizontal, single extraction turbine
Number of stages	5
Throttle pressure, full power (psia)	290
Extraction steam pressure, full load (psia)	90
Extraction steam flow, full load (lb/hr)	3224
Turbine steam exhaust conditions, full power	
Pressure (in. Hg abs)	9
Moisture (%)	12.2

Lube oil cooler, type	Air cooled
Turbine speed (rpm)	8050
Generator rating (kva)	1562.5
Generator rating, 0.8 pf (exclusive of excitation power) (kw)	1250
Generator type	Salient pole
Generator speed (rpm)	1200

3. Condenser System

Number of units	2
Type	Direct air to steam
Duty--heat rejected, full load, per unit (Btu/hr)	10.1×10^6
Design heat load per unit (Btu/hr)	10.4×10^6
Tubes	Horizontal, finned aluminum

4. Feedwater System

Deaerator	
Type	Atomizing
Feedwater design flow (lb/hr)	37,912
Design pressure (psia)	50
Oxygen removal guarantee (cc/liter remaining)	0.005
Storage (min)	5
Boiler feed pumps	
Number	2
Drivers	One steam driven, one electrical driven
Type	Vertical, centrifugal
Closed feedwater heaters	
Number	1
Type	Tube and shell, horizontal

5. Auxiliaries

Evaporator-reboiler	
Capacity (lb/hr of 35 psia steam)	7500
Design pressure (psia)	65
Make-up water temperature (minimum) (°F)	40
Condensate return temperature (°F)	172
Feedwater storage tank	
Capacity (approximately) (gal)	1950
Turbine steam bypass system	
Type	Manual with desuperheater station
*Maximum bypass flow	5% of full load
Auxiliary generator unit	
Type	Hi-speed diesel
Number	1
*Capacity (kw, at 6500-ft elevation)	150
Electrical characteristics	480 volts, 60 cps, 3 phase
Emergency power	
DC power source	Batteries
AC power source	2-unit MG set
Capacity at 8-hr discharge rate (amp-hr)	160

D. PACKAGING

Number of shipping packages in basic plant (exclusive of housing and site preparation):

	Uncontained	Contained
Primary loop packages including waste disposal system	5	7
Secondary loop packages	9	9
Decontamination package	1	1

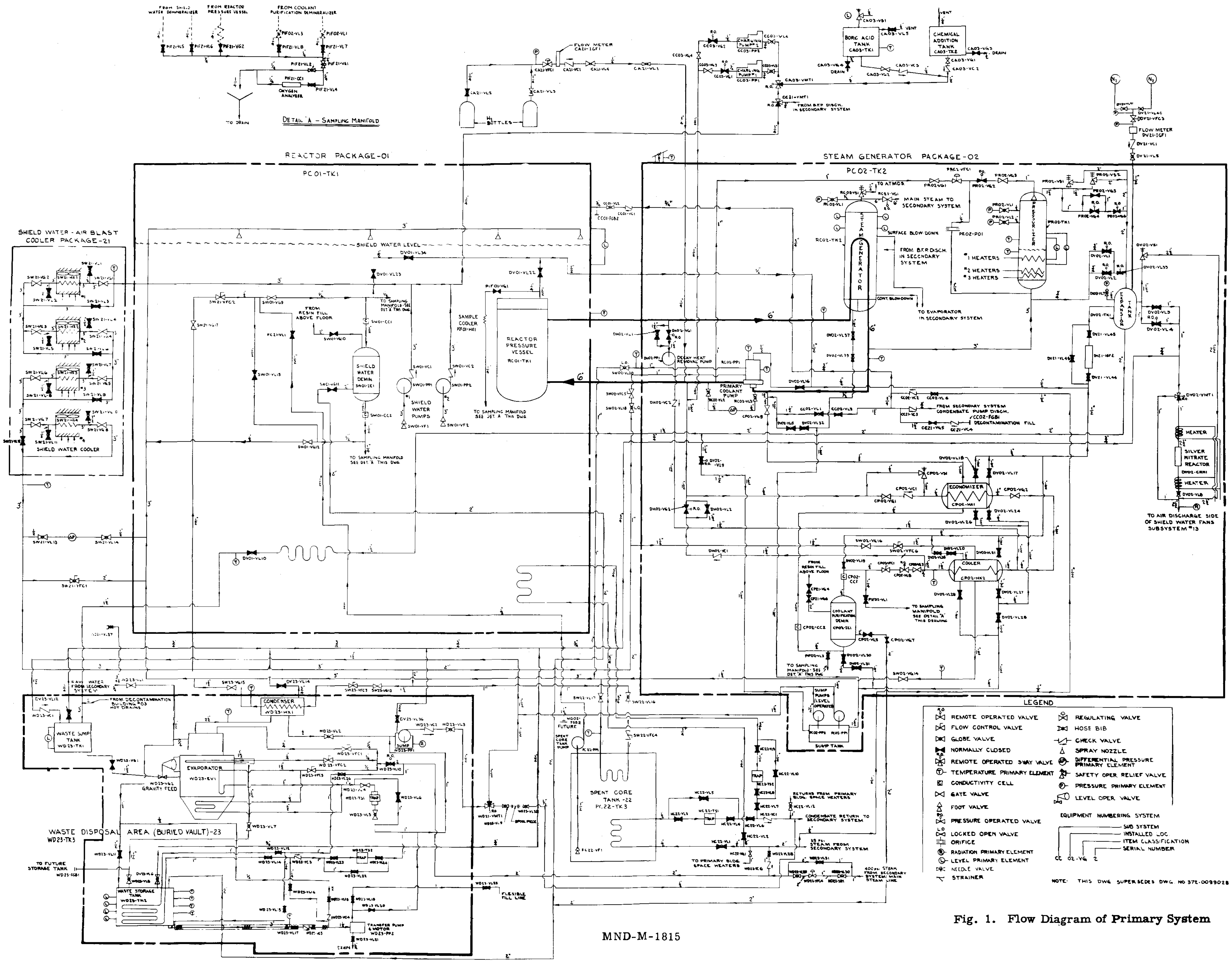

Fig. 1. Flow Diagram of Primary System

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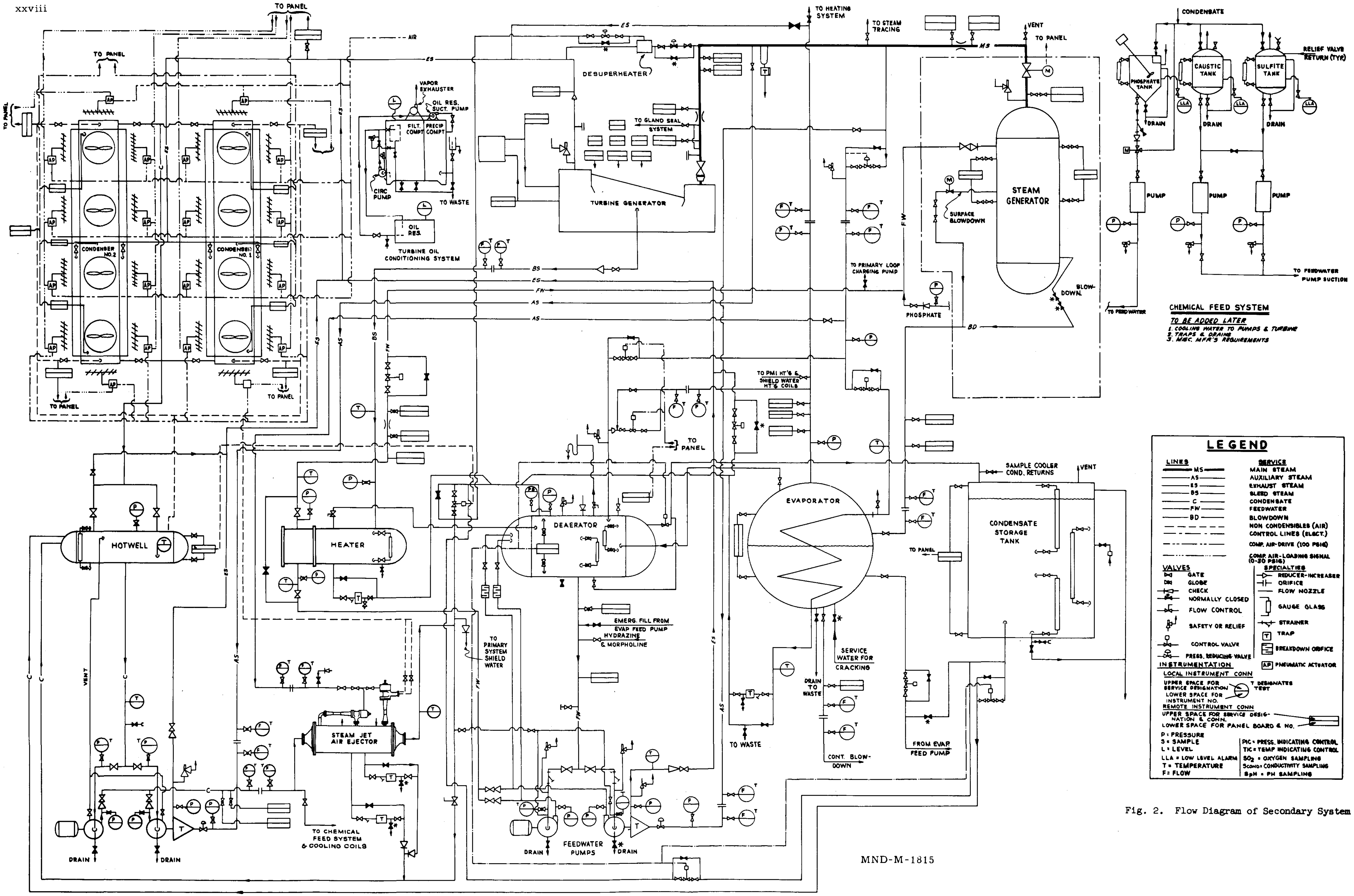


Fig. 2. Flow Diagram of Secondary System

I. TASK 1--PRELIMINARY DESIGN--SYSTEM DEVELOPMENT

Project Engineers--Subtasks 1.3, R. Akin; 1.2, 1.5, G. Zinoler; 1.1, 1.6, C. Fox

A. SUBTASK 1.1--PACKAGE DEVELOPMENT AND TEST

J. Cosby, A. Layman

During the fourth quarter, the planned objectives were to complete the design of test apparatus and to complete the fabrication of the Test Package. Fabrication of the test fixtures was scheduled to start. These were accomplished during the reporting period. In addition, the two loading and handling tests of the Test Package were successfully conducted to load limit conditions. The package test conditions were: (1) supported at both ends; and (2) supported at one end and the center.

A design element test was conducted to evaluate the use of lag screws versus bolts in wood for equipment tie-down in the Decontamination Package. Higher than anticipated values were obtained and the use of the lag screw seems quite feasible.

In the interest of economy, a minor rearrangement was made in schedule sequence. It was decided to install the insulated side panel, which simulates the Decontamination Package construction, after completion of the snow load test and to leave it in place for the remainder of the test program and the loading demonstration (Task 12). This simplifies handling and eliminates unnecessary repetition of tests to evaluate the strength capabilities of the insulated panel. The load-carrying truss structure in this panel is identical to that in the shipping package panel.

Figures I-1 through I-6 show the test package in fabrication and some of the tests performed on the test package during the quarter.

Test program objectives for the coming quarter are:

(1) To conduct scheduled tests.

(2) To complete fabrication of the test fixtures, including the impact test fixture for simulating impact shock loads.

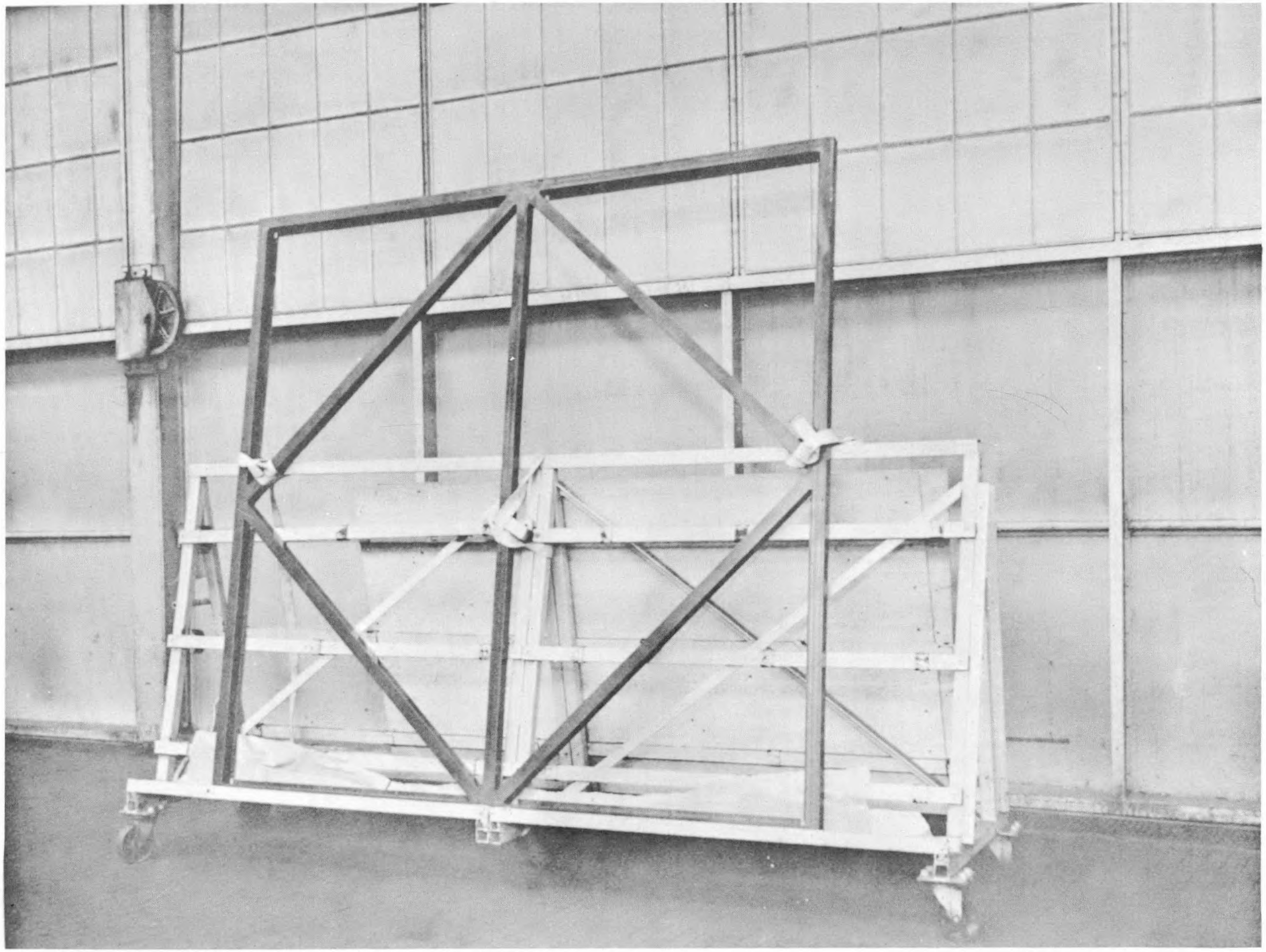


Fig. I-1. **PM-1** Truss Package--View of Completed End Truss

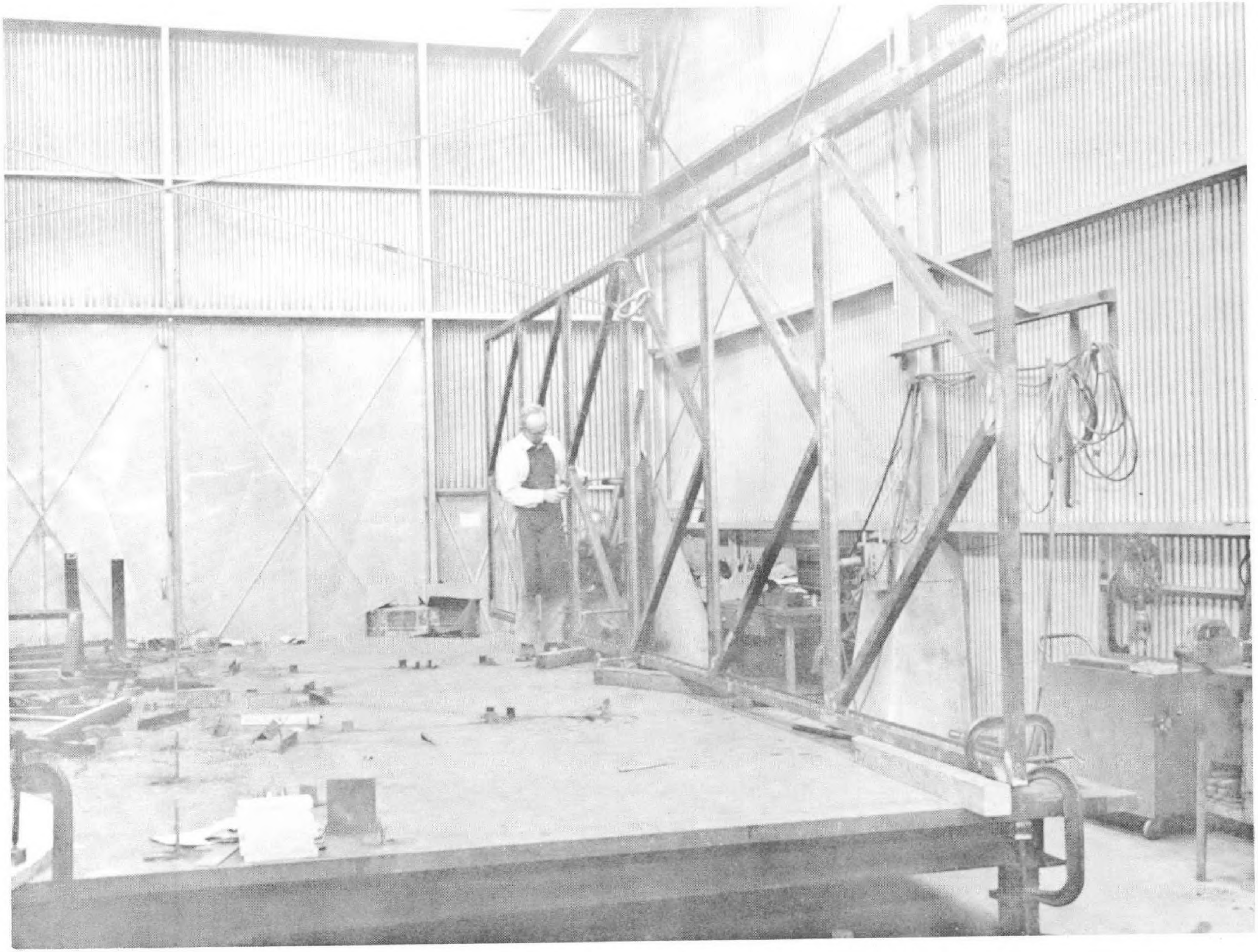


Fig. I-2. PM-1 Truss Package--View of Side Truss Assembly

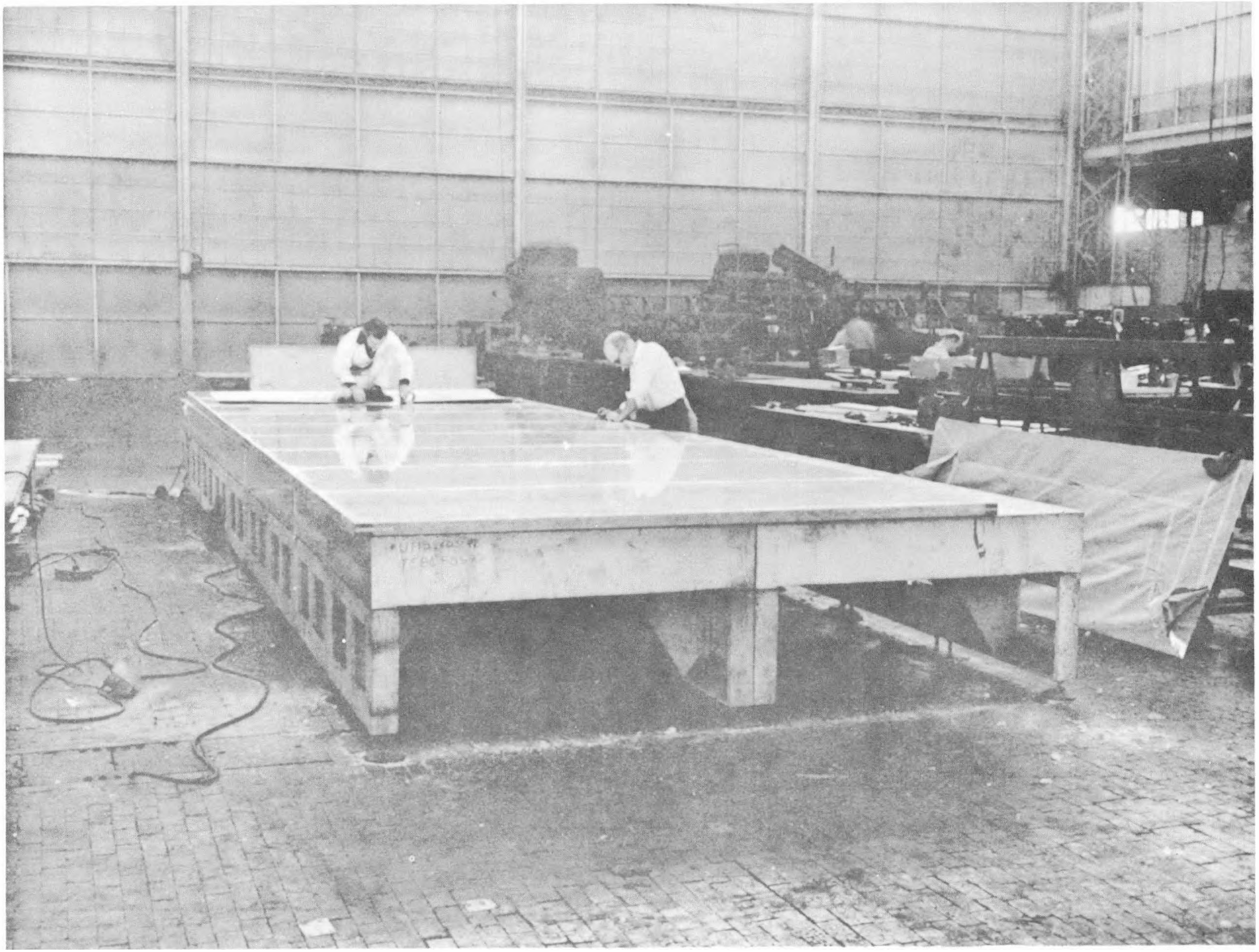


Fig. I-3. PM-1 Truss Package--View of Completed Roof Panel

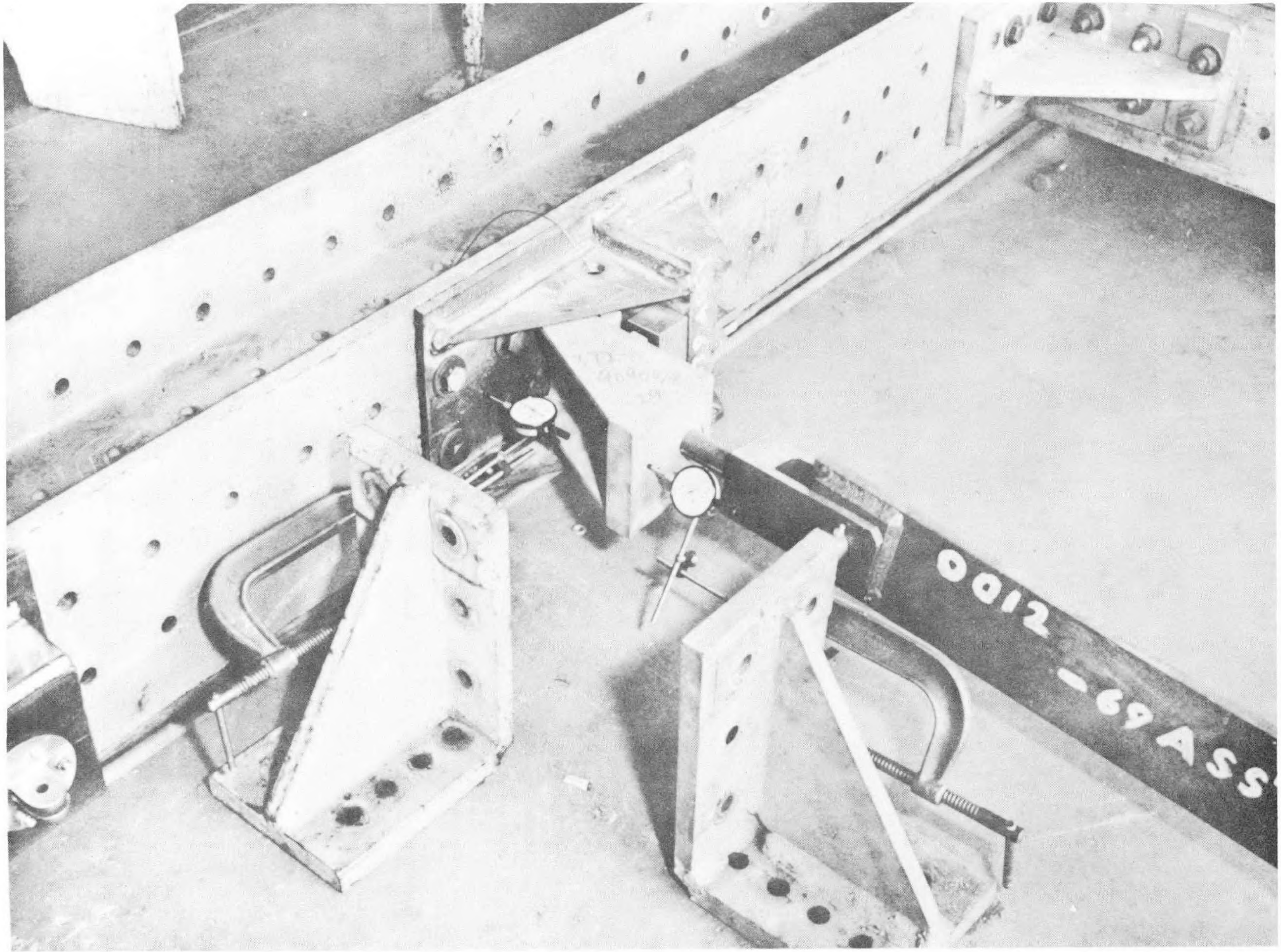


Fig. I-4. PM-1 Package Test Setup--Fitting, Jacking and Hoisting Adapter

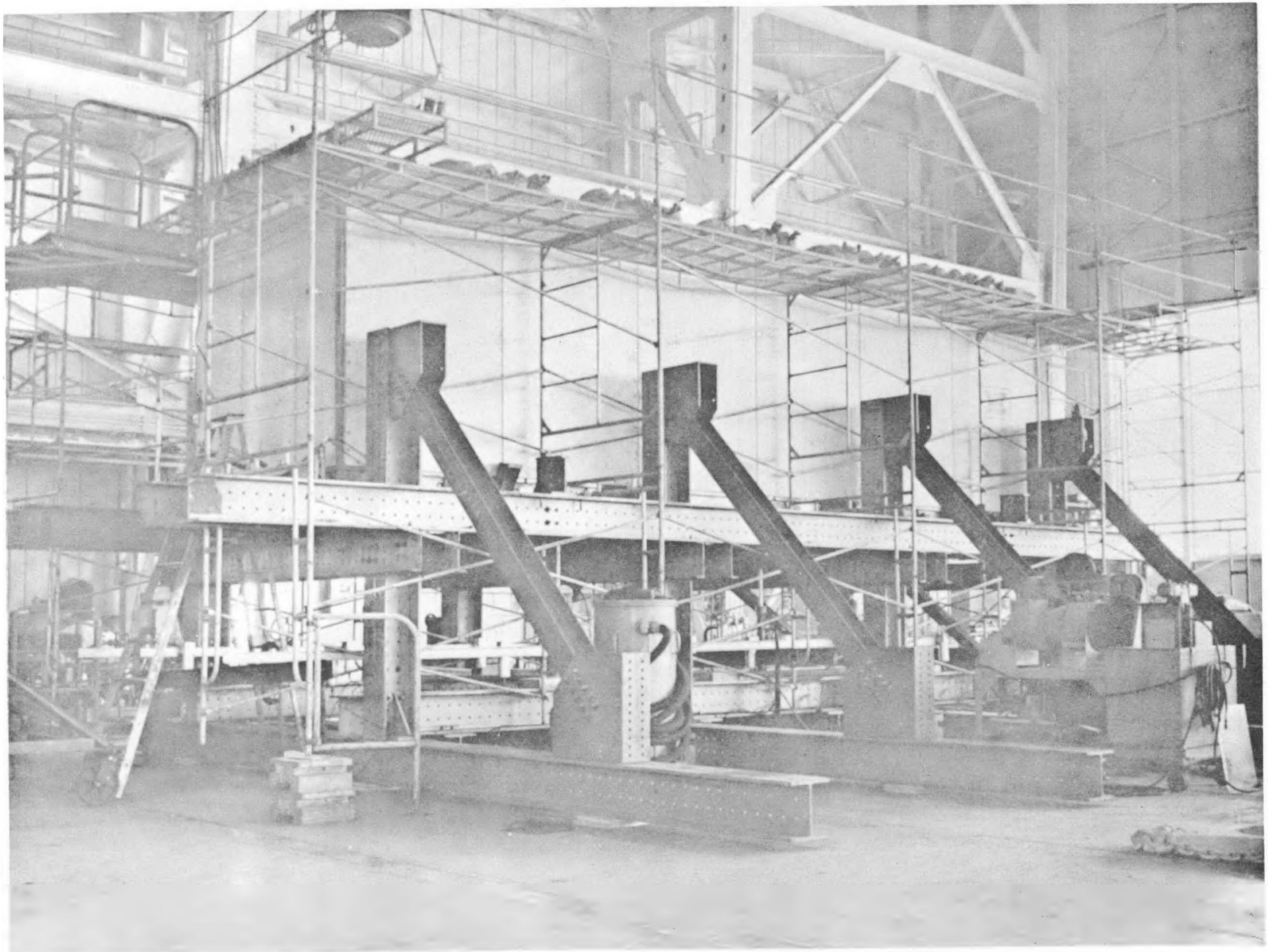


Fig. I-5. PM-1 Test Package End Support Condition--Overall View

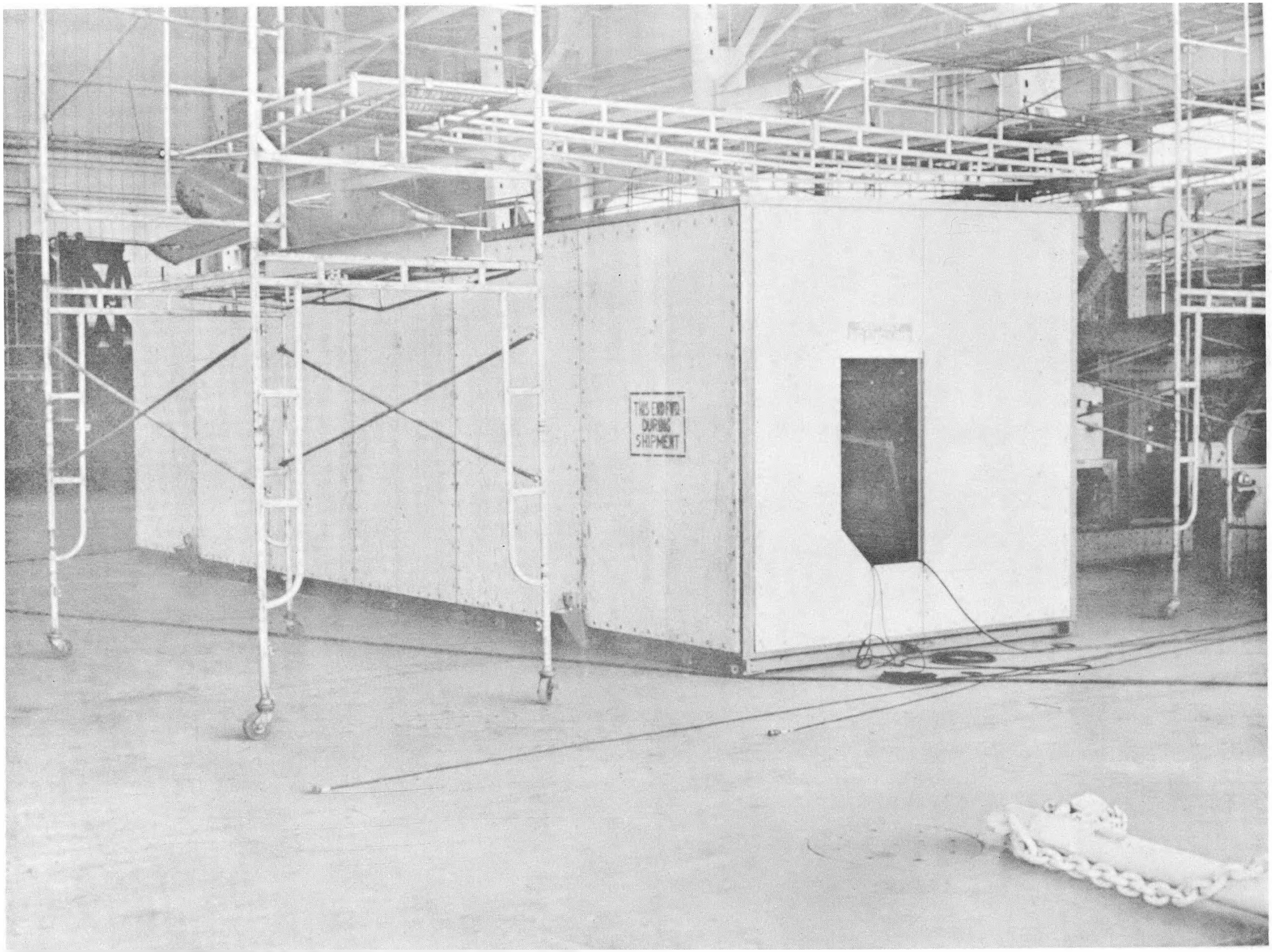


Fig. I-6. PM-1 Package in Preparation for Snow Load Test. View Shows Part of the Package Sling in the Right Foreground

B. SUBTASK 1.2--INCORE INSTRUMENTATION

G. F. Zindler

The objectives of Subtask 1.2 are to evaluate methods of measuring core parameters of interest.

The study was completed during this quarter and a classified technical memorandum was issued. The memorandum is entitled Incore Instrumentation Feasibility Study, MND-M-2297, Revision 1.

This completes all efforts under this subtask.

C. SUBTASK 1.3--SHIELDING MEASUREMENTS

During this quarter, activation analysis was performed on Borrow Area No. 2 and Borrow Area No. 4 (plant site) samples at the MITR. The activation analysis confirms the chemical data. Material from Borrow Area No. 2 will be used as the source of local shielding material.

During the next quarter, a detailed evaluation of the activation analysis will be made and a final report submitted.

Based on a preliminary evaluation of the activation analysis, the following general conclusions have been reached:

(1) Borrow Area No. 4 was found to build up activities about twice as great as those of Borrow No. 2 under the same irradiation conditions. This agrees with the preirradiation prediction.

(2) Decay rates were, in general, consistent with the chemical and spectrographic analysis of the samples.

(3) Borrow Area No. 2 decays considerably faster than Borrow Area No. 4.

D. SUBTASK 1.5--INSTRUMENTATION AND CONTROL

G. Zindler

This subtask had the objective of developing advanced instrumentation for application to the PM-1. The effort was subcontracted to the Stromberg-Carlson Corporation.

During this reporting period, all efforts under Subtask 1.5 were completed. A Topical Report entitled "PM-1 Power Plant Program Controls and Instrumentation Report," MND-M-1914, was prepared and issued. The report covers the entire program and presents recommendations and design changes to be applied to the PM-1 Instrumentation System.

This report completes the efforts under this subtask.

E. SUBTASK 1.6--SECONDARY SYSTEM DEVELOPMENT

The objective of this subtask is the development of components for the PM-1 Nuclear Power Plant Secondary System which are not commercially available.

Planned accomplishments during this period were:

(1) Completion of condenser model fabrication and installation at Eglin AFB Climatic Laboratory.,

(2) Initiation of condenser testing.

(3) Completion of certified outline drawings of the turbine-generator, switchgear and motor control center.

During this period, the following work was actually accomplished:

(1) The condenser model was completed, shipped to and installed at the Climatic Laboratory, Eglin AFB.

(2) Testing of the condenser model was accomplished and is currently near completion. The tests include shakedown, steady-state, and transient runs over ambient temperature ranges from +70 F to -65° F. At this point of development, the following test results affect the design of the prototype.

Design concepts which have proven satisfactory on the condenser model are as follows:

(a) Aluminum appears to be a most satisfactory construction material for the unit operating over the noted range of ambient temperatures.

(b) Induced draft fans have proven satisfactory.

(c) The separate noncondensable gas cooling section is required for proper operation of the system. A cooling section with more heat transfer area than the present model will be used on the prototype.

(d) Finned tubes in a horizontal configuration have proven satisfactory.

(e) The heat transfer area and auxiliary power requirement calculations have proven to be quite accurate.

Design concepts which are being modified as the result of the condenser model test are:

(a) The louver hardware and louver actuators will be installed inside the louvers. The louver hardware, louver blades and louver supports will be "beefed up."

(b) A two-pass U-tube, rather than the present single-pass straight tube, configuration will be used on the prototype. This will improve the design from several standpoints. First, the U-tubes lead to a natural distribution of steam more nearly optimum than is the case for straight tubes. This significantly reduces the need for inlet orificing. Second, the U-tubes may be replaced, whereas the original concept could not be retubed without major structural work. Third, the U-tube more readily accommodates thermal expansion differences due to nonuniform heating at startup.

(c) A steam distribution pipe appears desirable in the inlet headers to obtain better steam distribution to the lower tubes.

(d) A noncondensable collection pipe appears desirable in the exhaust headers to obtain better collection of noncondensables from the lower tubes.

(e) Both steam and electric tracing will be provided on all piping and valves containing condensate.

(f) A drain plug will be provided at the low point of the condenser.

In general, tests have verified the basic approach to direct air-steam condenser design including the use of pneumatic louver actuators, exhaust stacks and butterfly valves. Rail and truck transport of the test model to Eglin AFB clearly demonstrated the transportability of the unit, which suffered no damage whatsoever in handling. Improper support of the steam supply line to the test unit caused the unit to be lifted off the ground at one end by the inlet flange, but no damage was caused. The tests determined the operating procedures for the prototype unit. The areas needing improvement have been brought to light in the test program and are now under study. Figures I-7 through I-12 show the condenser model during fabrication and test.

(3) The certified outline drawings of the turbine-generator unit, switchgear and motor control center were received from westinghouse.

(4) During this quarter, the tests performed on the PM-1 air-cooled condenser model have been divided into three portions: shakedown, testing at ambients of 0° F and 70° F, and testing at ambients of -45° F and -65° F.

a. Shakedown

From January 13 through January 19, shakedown tests of the PM-1 air-cooled condenser model were run at Eglin AFB. The following items were noted at that time:

(1) Operation of the fans in both forced and induced draft was very smooth; vibration induced in tubes is virtually nil.

(2) The louver supports are subject to slightly excessive vibration when the louvers are in the fully closed position with all fans operating.

(3) Exhaust louvers were not installed on the model during this period; Westinghouse manufactured butterfly valves for the stacks rather than louvers in order to minimize exhaust pressure losses.

(4) There was excessive air leakage between the bottom of the tube bundles and the structure. The same is true around the sides of the air cooler section. This situation was remedied on the model by caulking with fiberglas insulation.

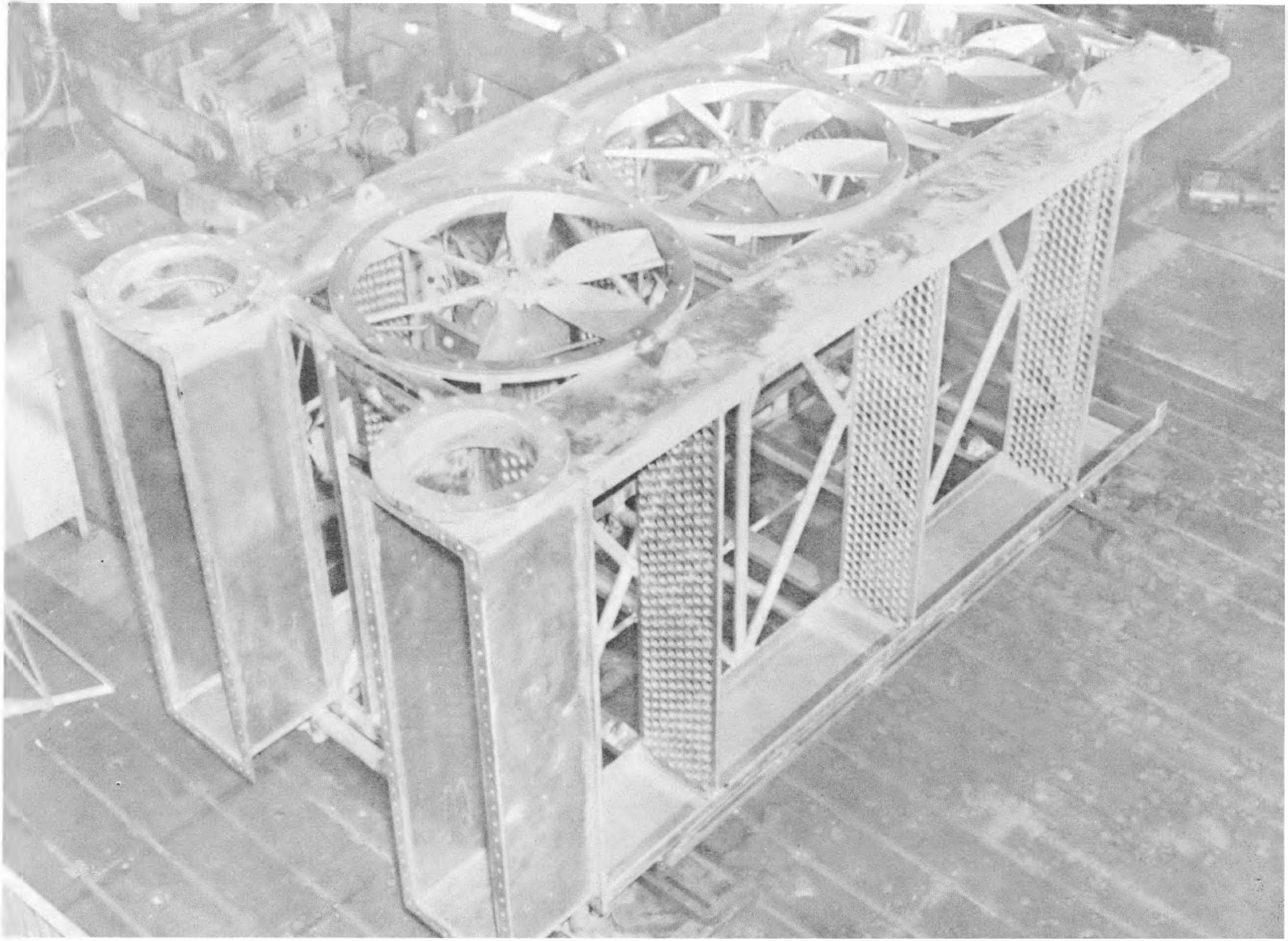


Fig. I-7. Condenser Model, Showing Assembled Condenser Model Frame, Tube Support Sheets and Header Boxes, with Fans Mounted at Top

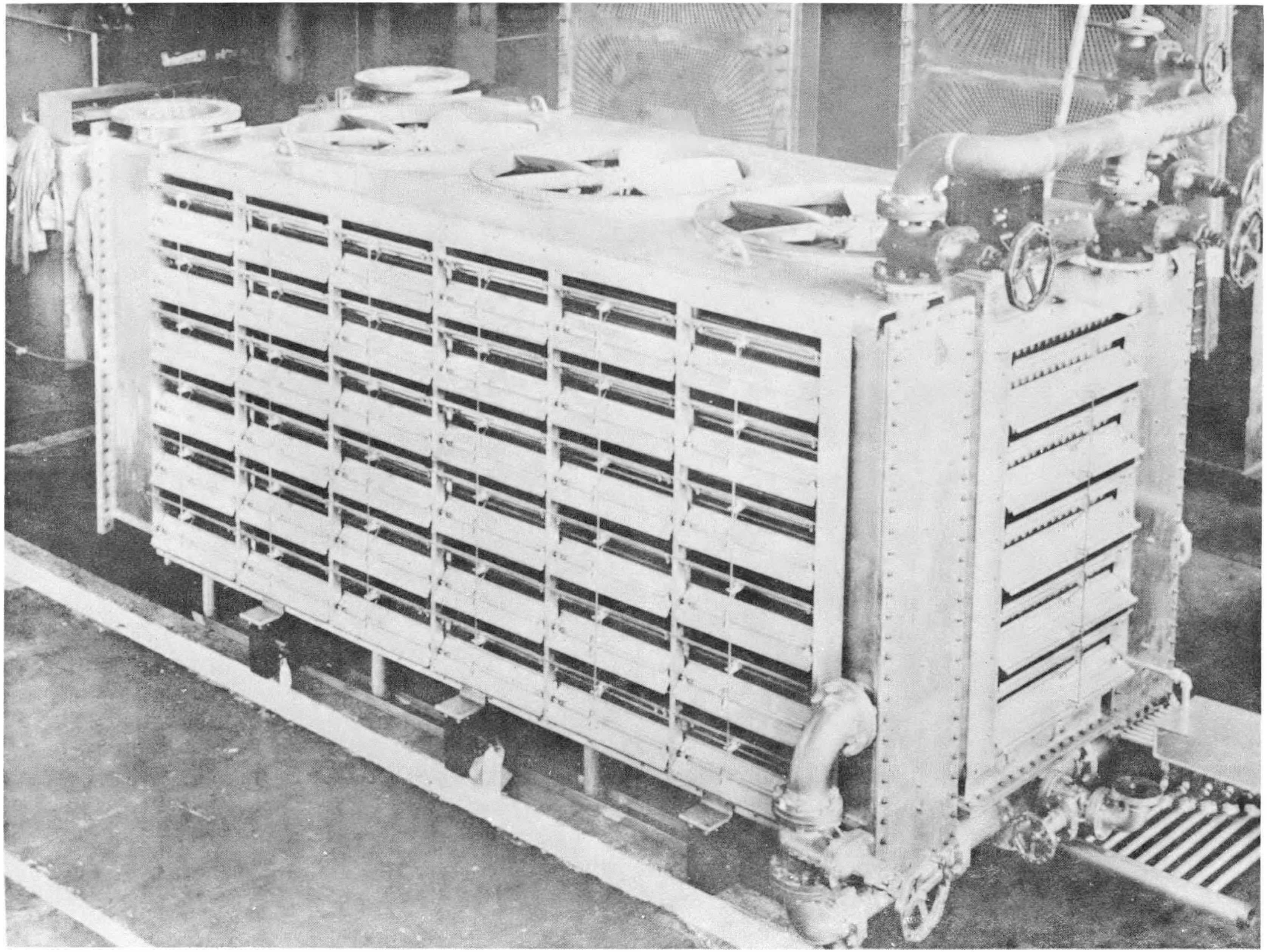


Fig. I-8. Completed Condenser Test Model, Showing Precooler End and Side Construction. Top Fan Exhaust Ducts have not been Added

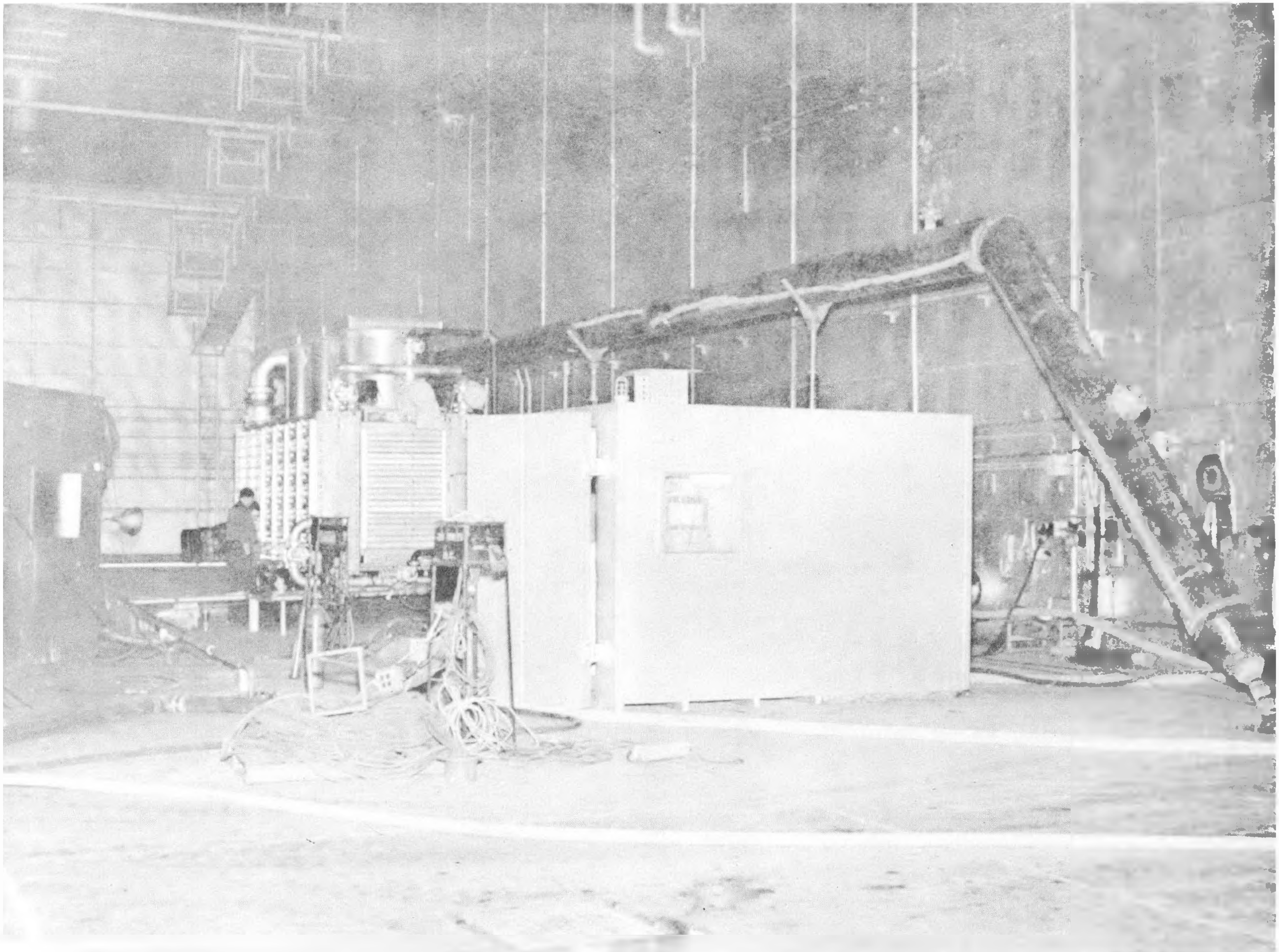


Fig. I-9. General View of Test Floor in Region of Condenser Model

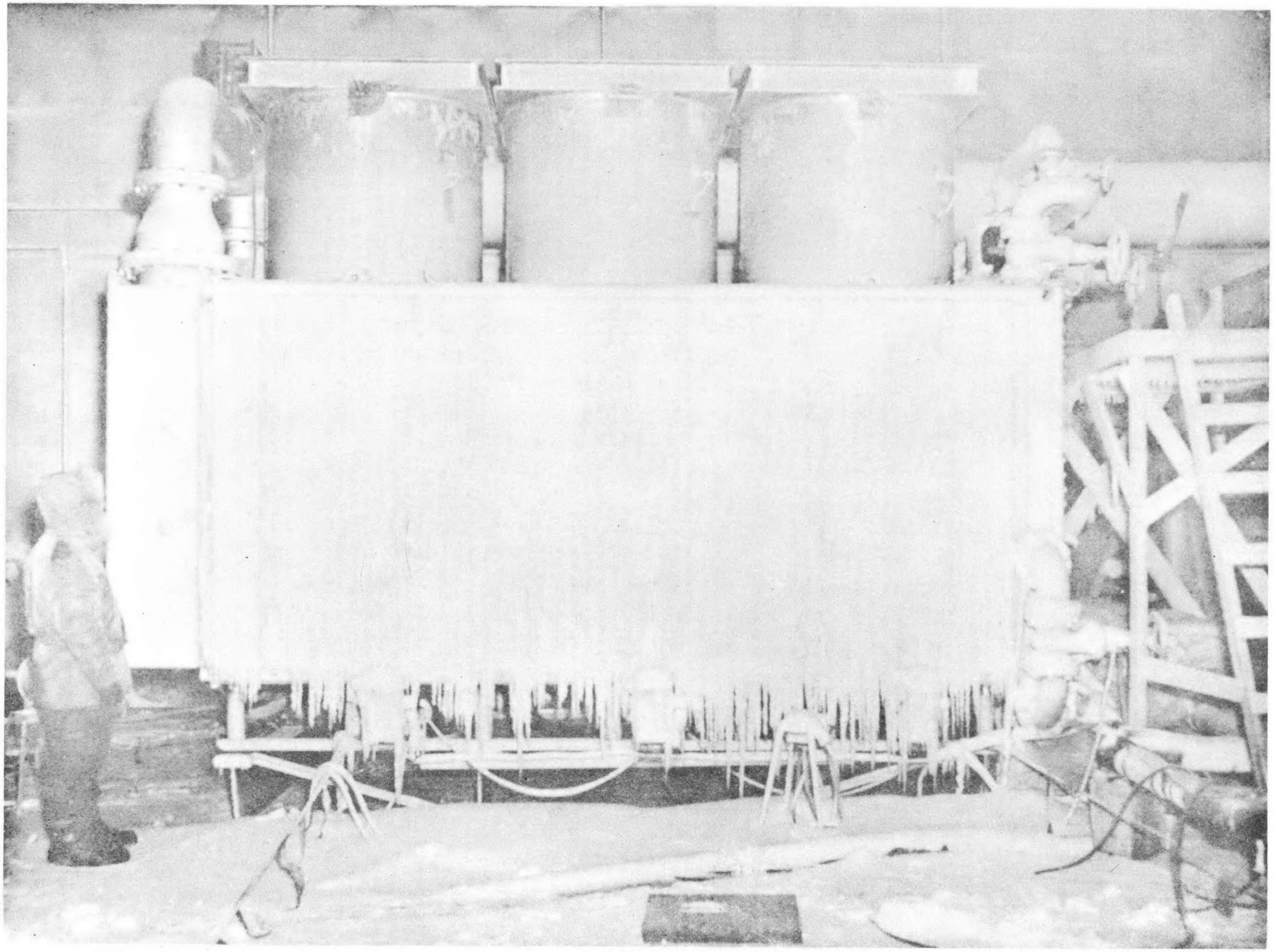


Fig. I-10. Inspecting Heavily Iced Condenser Test Model at -45° F

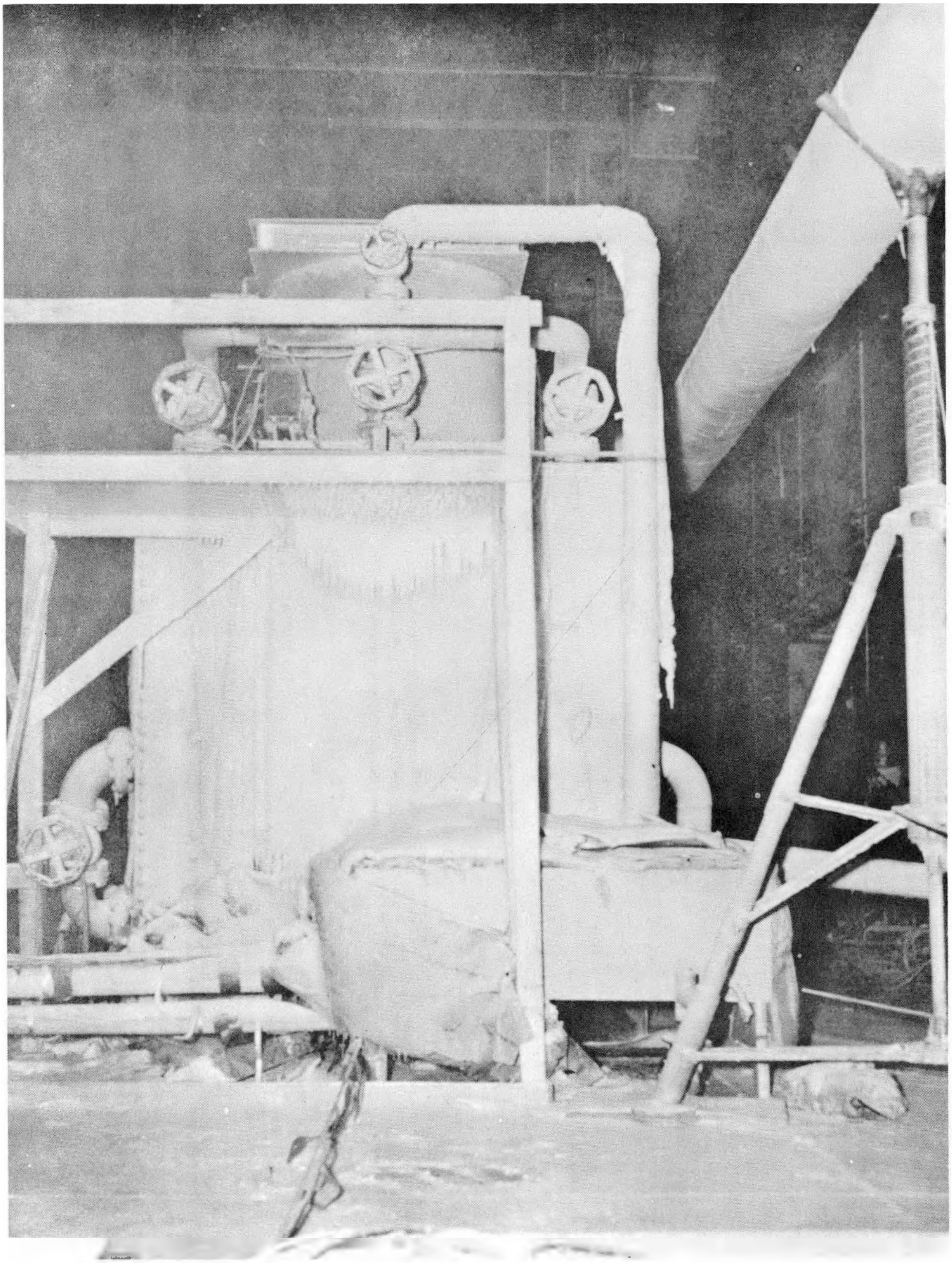


Fig. I-11. Heavily Iced Condenser Test Model at -45° F, Showing Precooler End (steam supply line at upper right)

Fig. I-12. Heavily Iced Condenser Test Model and Test Personnel at -45° F

(5) The louver drive linkage between the Baily servomotor and the louver is too delicate.

(6) Difficulty was experienced with both the condensate pump and the air ejector. In the case of the pump, minute leaks in the suction line resulted in pumping losses. The ejector malfunctioned as a result of moisture in the Eglin AFB steam supply and the steam pressure drop which accompanies increased steam flow. Indications are that the unit also malfunctions as a result of the hotwell not being vented to the condenser. These conditions were remedied on the model by installing a larger steam nozzle and properly venting the hotwell.

(7) The test rig appears to be relatively vacuum-tight at 70° F and at 150° F. The leakage rate is approximately 0.001 scfm, or the unit loses 1 inch Hg vacuum per hour at or above critical pressure ratio.

b. Tests at 70° F and 0° F ambient

Tests were performed at Eglin AFB with ambient temperatures of 70° F and 0° F. The tests were then discontinued for a short period to permit analysis of data. The conclusions made at that time are listed below.

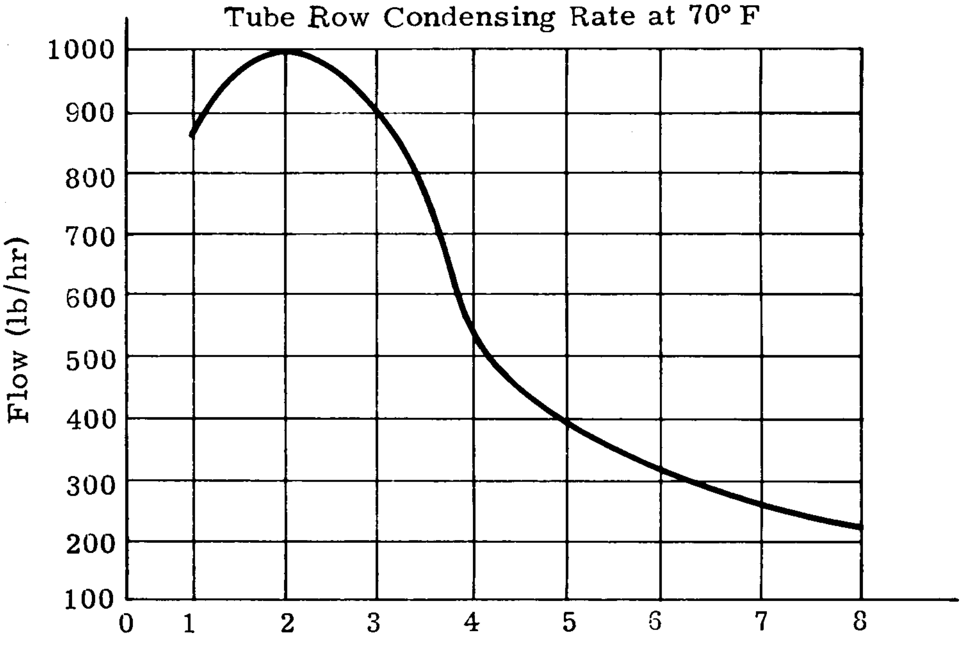
The areas of design upon which rework is warranted are steam distribution and cooling air distribution. Attempts were made to orifice the tube bundle at the inlet end with some success, but optimum orificing was not yet satisfactorily resolved. Additional work along these lines was postponed until March when the chamber will be returned to 70° F. The major work in this area will be installing air flow guides in the plenum chamber. (Currently, the air flow at the top of the bundle is approximately 20% greater than at the bottom.)

It was thought that perhaps the tubes on the outside of the bundle were becoming choked with condensate, thereby causing a maldistribution of steam. However, after increasing the tube slope from two inches to six inches, no change in steam distribution was observed. This indicated that the steam distribution is a function of condensing potential and air buildup, rather than condensate choking.

At 0° F, the unit was coated with a layer of ice approximately 1/4 inch thick. This was quickly removed by admitting steam to the unit and operating one fan in reverse.

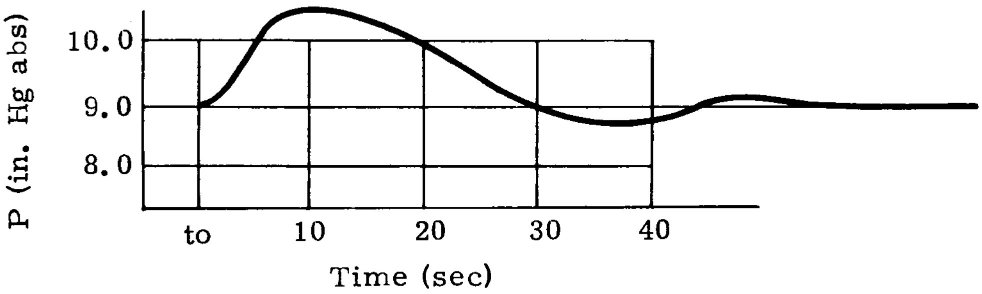
Curve I is representative of the condensing rates in each tube row at approximately full load.

Curve I
Tube Row Condensing Rate at 70° F

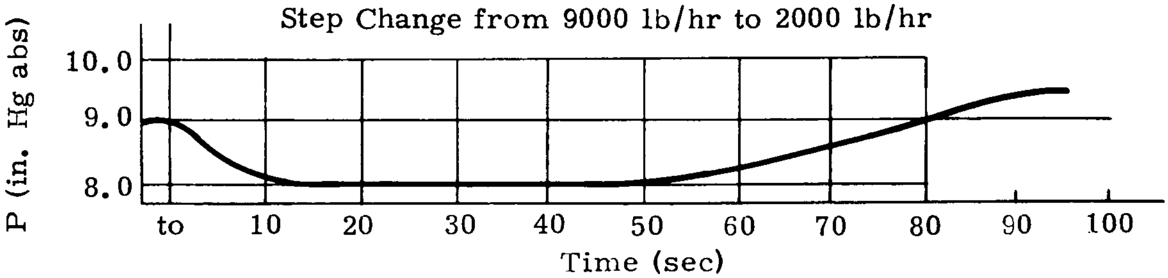


Transient maneuver tests indicate that the transient response of the unit is quite satisfactory, as indicated by Curves II and III, made at 70° F ambient.

Curve II
Step Change from 2300 lb/hr to 10, 300 lb/hr



Curve III
Step Change from 9000 lb/hr to 2000 lb/hr



The data in Table I-1 is typical of steady-state full load testing.

TABLE I-1

Typical Data--Steady State, Full Load

Test No.	$T_{ambient}$ (°F)	ϕ_{Steam} (lb/hr)	P_{Steam} (in. Hg abs.)	Manometers (in. Hg) LI	LD	RI	RD	AOT
8a	65	11,400	11.0	10.85	10.70	10.95	10.65	10.60
8b	65	10,000	9.0	8.89	8.85	9.05	8.75	8.70

Test No.	Fan Air Temp. (°F) 1	2	3	T_{Steam} Inlet (°F)	$U\left(\frac{Btu}{hr\text{-}ft^2\text{-}°F}\right)$	θ (° F)	ϕ_{Air} (lb/hr)	Q (Btu/hr)
8a	157	156	140	169	10.1	43.9	5.59×10^5	11.5×10^6
8b	150	149	135	170	9.95	39.5	5.35×10^5	10.2×10^6

Abbreviations:

LI--Left Steam Inlet Chamber
RI--Right Steam Inlet Chamber
LD--Left Condensate Discharge Chamber
RD--Right Condensate Discharge Chamber
AOT--Air Offtake Section--Top Header

Fan power consumption: No. 1--16 kw, No. 2--16 kw, No. 3--17.6 kw

U and ϕ are based on an average of the discharge air temperatures.

When corrected to 9 inches Hg abs steam and 70° F inlet air, the calculated transfer rates yield a condenser model heat rejection of 9.8×10^6 Btu/hr.

c. Testing at ambient of -45° F and -65° F

During the period from 9 February to 19 February 1960, tests were performed at Eglin AFB at an ambient temperature of -45° F.

(1) The condenser was sprayed with water to form an ice coating of approximately 1-1/2 inches thick. To deice the unit, steam was fed into the unit, the butterfly valves were closed and one fan was operated in reverse. Several problems became apparent during this test, namely:

(a) The heavy ice tended to "hang up" on the external louver hardware.

(b) Several of the condenser tubes near the bottom of the tube bank froze. This was apparently caused by the fact that the single-pass tubes were orificed for airflow from the outside of the condenser in. Consequently, with reversed airflow, the inside tubes were starved and did not receive enough steam to keep them warm.

(2) During condenser startup and operation at low loads, two problem areas became apparent, namely:

(a) Differential tube expansion. This problem is a result of a nonuniform tube bundle temperature which reached a difference as high as 125° F. A number of tubes buckled, bowed or pulled away from their seals in the headers during the fast startup runs. While these conditions were more severe than would be permitted in normal plant operation, the design basis for differential tube temperature is being raised to 220° F.

(b) Tube freezing. This problem appears to be the result of a combination of factors; principally, improper removal of noncondensables from the lower tubes, maldistritubion of steam in the bundle and airflow control.

Freezing occurred in the lower tubes under conditions where prolonged operation at subfreezing tube temperature was permitted. Temperatures as low as -6° F were permitted for 15 minutes with no damage, but extended operation under these conditions would prevent proper operation of the unit. In one instance, the condensate line froze up, causing the unit to fill with condensate and freeze a number of tubes. Fortunately, a plug could be removed to drain the unit. Instrumentation on the prototype units will indicate condensate temperature in the discharge header and will signal any stoppage in the condensate drain line by noting a drop in condensate temperature.

During the period from 29 February to 4 March 1960, at an ambient temperature of -65° F, a four-hour, steady-state test run was performed.

Also, during this period, the steam distribution and noncondensable collection pipes for the inlet and exhaust headers, respectively, were prepared for installation prior to 70° F testing.

(3) During this quarter, the design of the switchgear was completed and the certified outline drawings received for review. The PM-1 switchgear now includes an antimotoring relay for the protection of the diesel-generator unit.

The design of the motor control center was also completed and the certified outline drawings received for review. This review will be delayed until the final electrical requirements of the plant are known. At that time, any necessary changes will be made and the drawings corrected.

(4) The final certified drawing of the turbine-generator unit was completed and received for review.

The following progress is anticipated in the next quarter:

(1) The condenser test at Eglin AFB, Florida, will be completed.

(2) The final condenser test report will be completed.

(3) The final design of the condenser will be 90% completed.

(4) Switchgear fabrication will start (Subtask 7.2).

II. TASK 2--PRELIMINARY DESIGN--REACTOR DEVELOPMENT

Project Engineer--Subtasks 2.1, 2.2, 2.3, 2.4: J. O'Brien
Subtask 2.5: R. Akin

The objective of this task is to provide for the performance of the necessary analytical and experimental investigations which are prerequisite to the PM-1 reactor design.

A. SUBTASK 2.1--FLEXIBLE ZERO-POWER TEST

H. B. Rosenthal, E. A. Scicchitano

The objective of the flexible zero-power test is to provide experimental data to support the final core design of the PM-1 Nuclear Power Plant.

The work planned for this project quarter included:

(1) Submission of the Hazards Summary Report to the AEC Licensing Branch.

(2) Completion of all component design.

(3) Completion of fabrication of all major components.

(4) 80% completion of installation of all major components.

(5) Pre-experiment analysis on all major experimental cores to be 75% completed.

The work accomplished during this project quarter included:

(1) The Hazards Summary Report was submitted to the AEC Licensing Branch (see MND-M-1854).

(2) Component and system design was completed.

(3) Fabrication of components was 90% completed.

(4) Installation of the system was 40% completed.

(5) A study was performed on the type of distributed poison to be used in the program.

(6) A study was completed on the structural integrity of the lower polypropylene grid under excursion conditions.

(7) Tests were performed (a) to assure the safe operation of finger-type safety rods, and (b) to check out the operation of the Teleflex prototype rod actuators.

(8) An alternate safety rod design was evaluated.

(9) Experiments involving homogeneous fuel elements and homogeneous poison elements were reviewed.

During the next quarter, it is planned that:

(1) The system will be completely installed and checked out.

(2) The reactor will be brought to criticality and the experimental program commenced.

(3) Pre-experiment analysis of all major cores will be completed.

1. Lumped Burnable Poison Studies

A series of lumped burnable poison studies will be performed in the PMZ-1 program to evaluate the reactivity worth and self-shielding factor of lumped poisons of different concentrations. Studies to determine poison concentrations for the lumped poison rods and nuclear design of the special components for these experiments were completed.

Based on nonuniform burnup core lifetime studies described in the Task 4 portion of this report, a reference design concentration of 0.04 gram of natural boron per cubic centimeter of rod was established.

For evaluation of the physics calculations of lumped poison worth and overall self-shielding factor in the core, two additional concentrations plus a zero-concentration stainless steel rod were specified. These concentrations were based on results of the time-dependent self shielding (which is actually a function of concentration) studies as shown in Fig. II-1. The concentrations of poisons were so chosen that, within the manufacturing tolerances and measurable reactivity differences, results will be obtained to evaluate a curve similar to Fig. II-1. The concentrations thus specified are as follows:

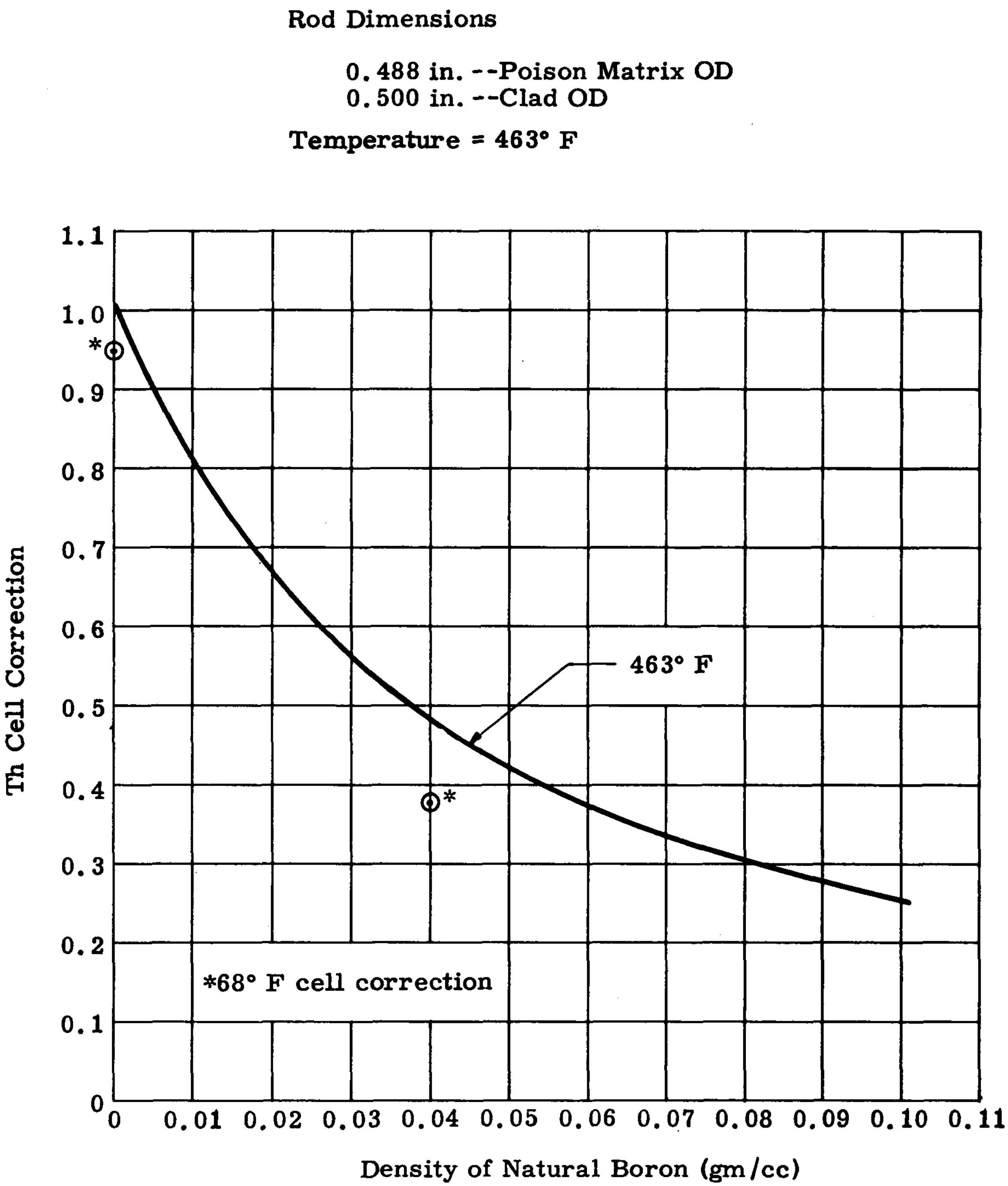


Fig. II-1. Lumped Poison Rod Self-Shielding Studies--Cell Correction Versus Boron Density

Natural Boron Concentration (gm/cc)	Boron (wt %)
0	--
0.02	0.253
0.04	0.510
0.06	0.769

Detailed self-shielding evaluation will be made from results of experiments in which the worth of lumped poison versus homogeneously distributed poison will be investigated. These measurements will be made in a special test bundle located in the center of the core and in an eccentric location in the core. Design specifications for the test bundle have been completed.

2. Prototype Rods

The PM-1 prototype rods will be mocked up exactly for the PMZ-1 experiments. Studies to determine the final design of the rods were completed. The poison section of each of the Y-shaped rods is 0.25 inch x 3.50 inches x 32.0 inches. The control poison material is Eu_2O_3 in stainless steel and is clad with 1/32-inch stainless steel. The Eu_2O_3 concentration is 30 wt % in Eu_2O_3-SS. This amounts to 1.90 grams of Eu_2O_3 per cubic centimeter or 1.64 grams of Eu per cubic centimeter.

3. Critical Experiment Control Studies

a. Homogeneous poison

Reactivity studies to determine homogeneous poison requirements for the experiments in which the prototype rods will be withdrawn above the critical bank position were completed. Several schemes were investigated:

(1) Soluble poison.

(2) Cadmium tin strips.

(3) Boron finger rods.

(4) Boron tapes.

(5) Boron-polyethylene strips and rods.

Results indicated that systems 1, 4 and 5 were the most desirable for providing the required homogeneous control.

b. Critical experiment and safety rod design studies

The preliminary designs of the critical experiment (CE) and safety rods were described in the third quarterly progress report (MND-M-1814). The design considered was a hexagonal bank of fingers (small rods, as shown in Fig. II-2), each finger of which fits inside a fuel element when inserted in the core. The poison (natural boron) section is approximately 0.32 inch OD. During the fourth quarter, a detailed evaluation of the rod design shown in Fig. II-2 was completed. The rod bank reactivity worths were calculated using the two-dimensional, three-group diffusion code, CURE. The calculated worth of the rod banks A and B located in the center of the core are -1.4 and -3.8% $\Delta\rho$, respectively. These values were corrected for a 23-inch length (the length in the PMZ-1 core) and an eccentric core location (≈ 6 inches) based on experimental worth versus insertion and radial worth data given in MND-MPR-1646 (Figs. VIII-13 and IV-3). The corrected worths for the CE and safety rods are -0.6 and -1.7% $\Delta\rho$, respectively.

An alternate safety rod design study was also completed. Results of the rod worth calculations for three rod designs, Y, and were -1.2, -1.4 and 2.1%, respectively.

4. Hazards Report

The PMZ-1 Hazards Summary Report, MND-M-1854, was submitted to the AEC Licensing Branch on 23 December 1959, with the request that Facility Operating License be amended by 1 March 1960. The Licensing Branch has been reviewing the hazards report during the current quarter.

5. Component Fabrication

During the current quarter, 90% of the component fabrication was completed. Items yet to be completed include:

CE control rods

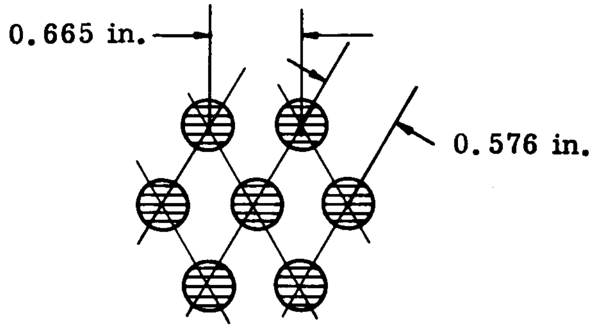
Safety rods

Rod extensions

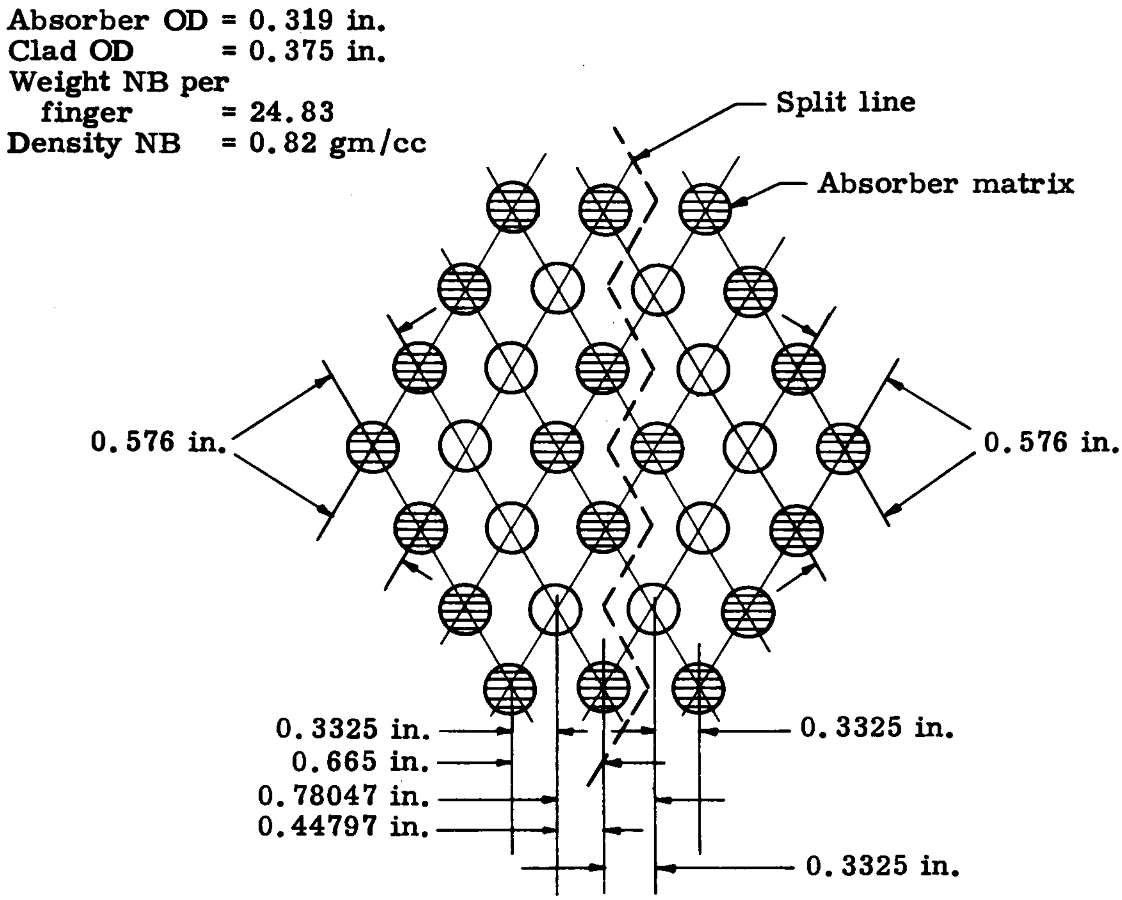
Y-rods

Grids.

The PMZ-1 top assembly is shown in Fig. II-3.



A. CE Rod Design



B. Safety Rod Design

Fig. II-2. CE and Safety Rod Design

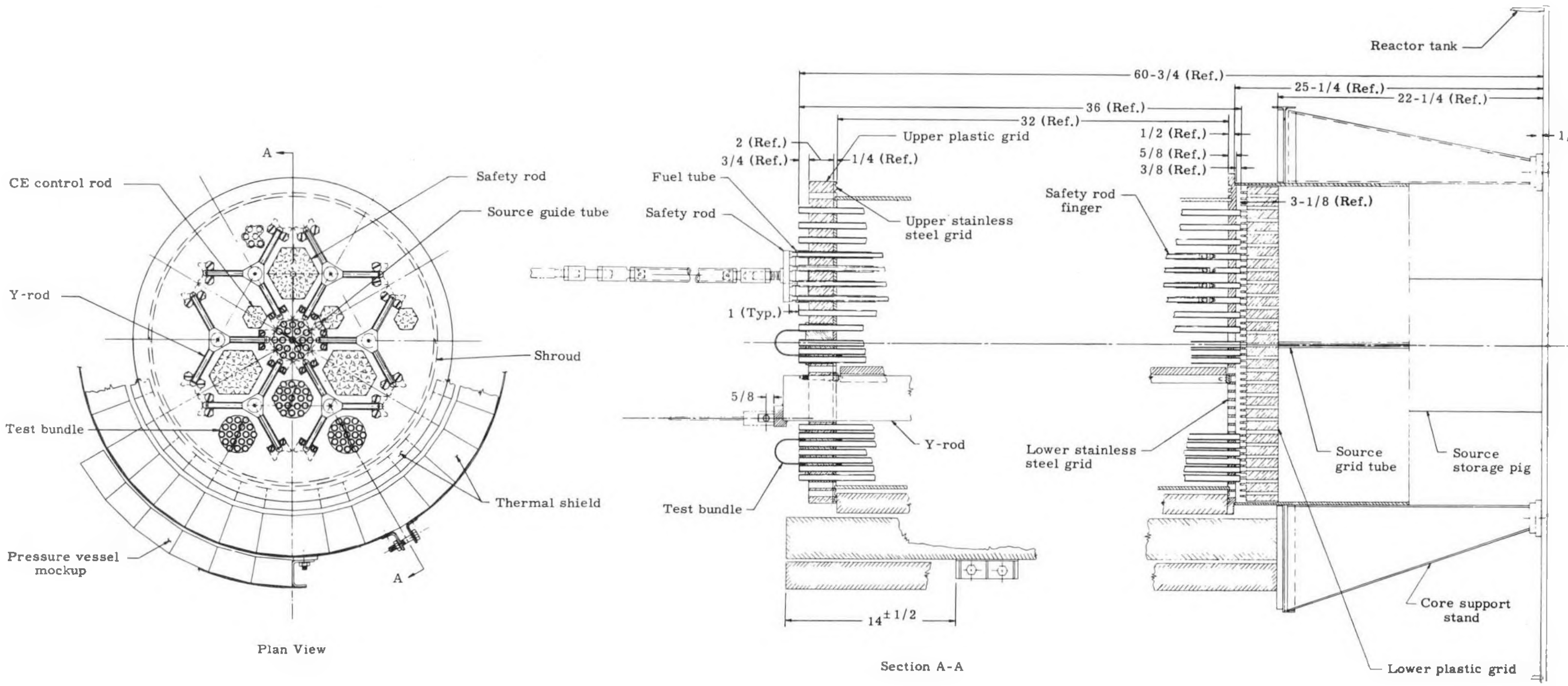


Fig. II-3. Core Assembly and Installation PMZ-1 Zero Power Test

6. Installation

During the current quarter, all extraneous components from the AEC Contract AT(30-1)-277 zero-power test were removed from the system and appropriate modifications made to the facility for PMZ-1. Forty percent of the installation was completed. Items completed include:

Relocation of reactor tank standpipe

Raising of actuator support structure

Modification of actuator support structure

Modification of steam heaters

Relocation of source storage pig

Location of core support stand

Modification of instrumentation standpipe

Overhaul of instrumentation system.

7. Distributed Poisons

During this quarter, a study was conducted to establish the most appropriate distributed poison for the PMZ-1 experiments. Five possibilities were considered:

(1) Boron-impregnated tapes.

(2) Borated polyethylene.

(3) Soluble boron.

(4) Boron steel.

(5) Cadmium-tin strips.

Cadmium-tin was ruled out because of its effect on the neutron spectrum and the cost and time involved in fabrication. Cost, time, lack of flexibility and excessive moderator displacement made boron steel undesirable.

The advantages to be gained by using soluble poisons are the speed and ease of changing poison loading and the uniformity of core poisoning. The latter effect is quite important in the PMZ-1 program, since approximately 1/7th of the core cannot be poisoned by other means due to the use of fuel tubes as guides for safety and control rod fingers.

Personnel from other laboratories where soluble poisons had been employed were contacted to learn of their experience. Two major drawbacks to the method were mentioned: (1) the difficulties encountered in determining boron concentration, and (2) cleanup of the system after its use. The boron content uncertainties can be minimized by a good chemical analysis along with careful control of weighed boron additions to known volumes of solution. Rinsing the system several times with tap water after using soluble poisons reduces the content of boron to 1 to 2 ppm and reproduces initial reactivities to between 10^{-3} and 10^{-4} Δk. Based on these discussions, a test was set up to determine the boron content of rinse water after a test bundle had been allowed to stand in a boric acid solution. The results to date indicate that, in the PMZ-1 program, approximately 0.1 gram of boron will be left in the system after using a saturated boric acid solution following five rinses. After two rinses, the system yielded less than 1 ppm boron and 98% of the residual boron content due to droplet formation had been removed.

The use of soluble boron poison for the PMZ-1 program would require changes in the reactor water system. A separate poison tank and fill line using the same electrical circuitry with a selector switch for fresh or poisoned water system would be required. When poisoned water experiments are to be performed, all fresh water systems must be deactivated by taking such steps as dumping the fresh water from the storage tank and disconnecting reactor tank panel heaters.

Changes would also have to be made in the experimental program, since all clean water experiments would be performed first and then all poisoned water experiments. Since the poisoning is accomplished more conveniently as an irreversible process, experimental procedures would be changed so that reactivity evaluation of all possible Y-rod configurations would be taken at each poison concentration. Because the panel heaters would be disconnected, temperature coefficients with distributed poison in the core would not be performed. However, other temperature coefficients would be substituted for those affected.

The problems associated with soluble boron are sufficiently complex that it was felt that it should not be used in the early phases of the program, but the experimental advantages are important enough that it should be considered for use later in the program.

Boron-impregnated tapes and borated plastic present the best alternatives initially. Although the tapes offer more flexibility, the lower cost of the borated polyethylene led to its being selected as the distributed poison that would be used in the early phases of the PMZ-1 experimental program.

8. Polypropylene Grid

The adequacy of the lower polypropylene grid was studied during the report quarter. The problem of whether the lower polypropylene grid would maintain its structural integrity during the nuclear excursion associated with the maximum credible incident was investigated.

There were two facets to the problem: (1) effects of heat transferred into the grid and (2) effects of pressure buildup. Calculations were performed which demonstrated that the heat transferred to the grid will not raise the temperature above 100° C. It was experimentally determined that, at 100° C, it did not soften nor show any evidence of deformation. Conservative calculations demonstrate that the grid can withstand pressures of at least 30 psig. The maximum pressures exerted during the maximum credible incident would be less than 8 psig.

It was concluded that the lower polypropylene grid is adequate as designed.

9. Safety Rod and Control Rod Tests

a. Finger-type safety rods and scram actuators

Before installation in the PMZ-1 core, the finger-type safety rods were tested to study the mechanical feasibility and safety of this type rod. Factors that were studied included alignment and binding, rod insertion time, tube damage and effective dashpot operation. In addition, one of the three safety actuators which will be checked out to assure good operating condition was tested during this quarter.

The test equipment, designed to mock-up the PMZ-1 structure, consisted of:

(1) One safety rod actuator connected to a cluster of 30 poison rods, or fingers, by means of a 4-foot extension rod.

(2) A 12-inch diameter water containment vessel in which aluminum grids hold thirty 1/2-inch diameter stainless steel tubes in a hex-shaped pattern.

(3) An 11-foot aluminum support structure upon which the safety rod actuator rests, and upon which a slidewire is mounted.

(4) A brush oscillograph and d-c amplifier which receives its input from the slidewire across a 6-volt battery.

(5) A Sorenson ac-dc voltage regulator, a control board, a cycling device consisting of a small synchronous motor and 5 microswitches, and the associated wiring.

The following requirements were to be verified before the safety rod is installed in the PMZ-1 core:

(1) Rod drop time for 2/3 rod insertion shall be 250 milliseconds or less.

(2) Magnet current shall not exceed 250 milliamperes.

(3) Air pressure for cocking should not exceed 90 psig.

(4) Dashpot operation must be consistent and adequate for safely decelerating the moving mass.

(5) The finger-type safety rods must operate through 1000 cycles without failure.

Three orifices of different sizes (0, 0.052 and 0.104 inch in diameter) were placed in the air exhaust from the actuator. By means of an oscillograph, the rod cocking and drop times were measured for a piston diameter of 3.060 inches and for each orifice. Using an orifice of 0.052 inch, the average rod drop time for 2/3 insertion was 257 milliseconds; the average signal time plus residual magnetism time was 46 milliseconds; and the average time for full insertion was 395 milliseconds. The size of the orifice does not change the time for 2/3 insertion, but it does affect the dashpot operation. It was found that an orifice of 0.052 inch in diameter resulted in the most effective dash.

To measure wear on the inside of the tubes, ink was poured down the inside of six tubes in various locations in the cluster and the cycling device started. After 500 cycles, the tubes were inspected and showed little sign of wear or scratching.

There was some difficulty in initial alignment of the fingers due to crookedness and bowing of the finger rods and the extension rod, but after 1000 cycles, there was no evidence of binding, swelling or physical damage to the actuator or finger rods tested. The minimum air pressure for cocking was 85 psig.

b. Teleflex control actuators

Upon delivery of six control actuators by the Teleflex Corporation, a pre-installation test was conducted to confirm the established specifications for the actuators. These tests determined safety and reliability features of the control actuators before installation in the PMZ-1 core. During this quarter, all of the six actuators have been tested. One of these was found to be defective. It was repaired by the Teleflex Corporation and now satisfies specifications.

The test equipment consisted of:

(1) A three-phase, motor-controlled cable and reel system manufactured by Teleflex Corporation (six units).

(2) A control board and associated wiring..

(3) A d-c power supply (Kepco) for four "Nixie" indicator lights.

(4) A 48-pound weight and 2 deflection gages to measure deflections of 0.001 inch.

The pre-installation tests of the control actuators were performed in order to confirm the following specifications:

(1) The actuator will perform as specified with a load of about 40 pounds.

(2) Total rod travel will be 30 inches.

(3) Position will be accurately indicated to ± 0.01 inch.

(4) Rod withdrawal and insertion rates for the drive mechanism will be between 5 and 7 inches per minute.

(5) Limit switches and indicating lights will be operable.

(6) The actuator will be stable (no drift) with no current applied to the motor drive.

With the 48-pound weight attached, the limit switches were set so that the weight would travel 30 inches. The weight was put at 00.00 inch, raised to 30.00 inches and then returned to 00.00 inch, using a stopwatch to find the rate of withdrawal (5.66 in./min) and rate of insertion (6.5 in./min). To find the accuracy of the position indication, the weight was put at an arbitrary point and the numbers indicated by the "Nixie" lights were recorded. The deflection gages were set and the weight lifted. When the weight came back to the same position, as shown by the "Nixie" lights, the deflection gage indicated the difference in deflection from before the movement. Reproducibility at a point was also checked by this method. Backlash, overtravel, and drift were measured and found to be negligible. The first control actuator tested was found to be binding and could not lift the 48-pound weight. The unit was repaired by the Teleflex Corporation. Thereafter, no further binding of any of the actuators occurred.

The results of the tests on all six actuators were as follows:

(1) The actuators performed as specified with a load of 48 pounds.

(2) The total rod travel was limited to 30.00 inches by the limit switches.

(3) The position was accurately indicated to ± 0.01 inch.

(4) Rod insertion rate was 6.5 in./min, while rod withdrawal rate was 5.66 in./min.

(5) Limit switches and indicating lights operated satisfactorily.

(6) The drift for a 12-hour period, with no current to the motor drive, was negligible.

(7) Backlash and overtravel were negligible.

10. Revision of Experimental Program

A review of the experimental program outlined in the last quarterly report was made. As a result, the experiments were revised as follows:

Experiment Number New	Experiment Number Old	Title
1	(1)	Critical Core Geometry. This experiment will include a "clean" core and a core containing lumped poisons. Since the finger-type safety rods operate satisfactorily, the experiment will be "clean" with no rod water channels to produce extraneous perturbations.
2	(3)	Six-rod bank position for criticality for the full-size core without lumped poison.

Experiment Number New	Experiment Number Old	Title
3	(7)	Facility power level calibration.
4	(4)	Temperature coefficient for full-size core with no lumped burnable poison.
5	(5)	Critical six-rod bank position for the design core.
6	(6)	Temperature coefficient for design core with the six-rod bank in the critical position.
7	(8)	Power density distribution of the design core with six-rod bank inserted to the critical position.
8	(10)	Power density distribution of the design core with the three-rod bank partially inserted.
9	(11)	Total core reactivity--design core.
10	(12)	Temperature coefficient for design core with all prototype rods fully withdrawn.
11	(13)	Reactivity evaluations of 3-, 4-, 5- and 6-rod banks.
12	(16)	Total core reactivity--full-size core without lumped burnable poison.
13	(17)	Temperature coefficient for full-size core with no lumped burnable poison and all prototype rods fully withdrawn.
14	(14)	Flux and power mapping of the design core with all prototype rods fully withdrawn.
15	(15)	Fast neutron and gamma measurements for the design core with all prototype rods fully withdrawn.
16	(19)	Evaluation of reactivity worth of lumped burnable poison of different concentrations.
17	(20)	Self-shielding factor evaluation of the lumped poisons of different concentrations.

Experiment Number New	Old	Title
18	--	Self-shielding factor evaluation of the tubular fuel element. (This experiment has always been included in thinking and discussions of the program, but it has not previously been designated explicitly.)
19	(22)	Epicadmium worth evaluation of rare earth materials.
20	(31)	Startup sensor location.
21	(32)	Core reactivity for the design core with the lumped burnable poisons redistributed.
22	(33)	Power measurements for the design core with redistributed lumped burnable poisons.
23	(34)	Determination of shipping container multiplication.

Experiments not Previously Included

24	--	Effect of stainless steel tubes on Y-rod worth. The current PM-1 design calls for stainless steel dummy fuel elements at the intersections of the Y-rod arms. This experiment will compare the worth of the Y-rods with the stainless steel tubes and lumped burnable poison rods in this location. Measurements will be made by calibrating a portion of a Y-rod, or Y-rod bank, for both conditions. Calibrations will be made either by period evaluation of the rod or by distributed poison exchange, depending on the magnitude of the effect being measured. Measurements will be made on the design core with the six-rod bank in the critical position and on the design core with the four-rod bank fully inserted.
25	--	Effect on Y-rod worth of fuel and poison distribution in the central bundle. It is the intent of this experiment to study the effect on the worth of the four-Y-rod bank of varying the distribution of fuel and lumped burnable poison in the central bundle. Two distributions besides the design core will be studied. Measurements will be made using the techniques described under Experiment 24.

11. Homogeneous Fuel Elements

Experiment 18 above was proposed to evaluate the self-shielding factor of the fuel element. Calculations indicate that, for a cell containing 19 fuel tubes, the maximum reactivity difference between the cell with fuel tubes and a cell with the fuel, steel and water homogeneously distributed will be 0.08% $\Delta k/k$. Experimental error and manufacturing tolerances combine to make the measurement of such a small difference questionable. Since the fabrication of a homogeneous fuel element is both time-consuming and costly, it was decided that the experimental results which could be obtained would not warrant expending the required effort. Furthermore, the desired information can be inferred from data obtained in other experiments. As a result, this experiment was eliminated from the program.

12. Homogeneous Poison Elements

A homogeneous poison element is being designed to measure the self-shielding factor for the lumped poison. This will consist of a can which will enclose approximately 19 fuel elements. Measurements will compare the reactivity worths of lumped poison rods containing various boron concentrations with the worths of the same quantities of boron homogeneously distributed in soluble form in the moderator inside the can.

B. SUBTASK 2.2--IRRADIATION TEST

J. B. Zorn, C. Smith

The objective of the irradiation program is to subject the PM-1 fuel element to burnup of fuel in an environment which simulates, as nearly as possible, the conditions of temperature, heat flux, coolant subcooling, coolant temperature rise and heat removal to be experienced during operation of the PM-1 Nuclear Power Plant.

Accomplishments during this quarter included:

(1) Completion of final design and analysis of the PM-1-M fuel element for insertion in the SM-1.

(2) Receipt of SM-1 end boxes, springs and side plates for the PM-1-M fuel element from Sylcor.

(3) Receipt of bids for the ETR-MTR bare element irradiations from BMI and GE Vallecitos.

(4) Selection of WTR for the in-pile pressurized water loop irradiation program and initiation of subcontract negotiations.

During the next quarter, it is anticipated that:

(1) Fabrication of the PM-1-M fuel element will be completed.

(2) Fabrication of all bare elements will be completed.

(3) ETR-MTR irradiations will be initiated.

(4) Work at the WTR will be initiated.

1. MTR-ETR Irradiation Program

The statement of work for irradiating two standard PM-1 fuel tubes to burnups of approximately 70 to 90 a/o U-235 in the MTR or ETR was revised. This statement covers pre-irradiation inspection of the elements, irradiation of the bare elements in contact with the reactor process water, and subsequent post-irradiation testing. It was forwarded to Battelle Memorial Institute and GE Vallecitos as an invitation for their bids.

2. WTR Loop Irradiation Program

A trip was made to the Westinghouse Testing Reactor, Pittsburgh, Pennsylvania to renew negotiations on an in-pile loop irradiation of PM-1 core components. As a result of this meeting, the following decisions were made:

The peripheral 3-inch diameter in-pile loop in the WTR is more suitable for irradiation of PM-1 elements than the central thimble because of the general increase in flux levels expected when the reactor power is escalated to 60 Mw on 14 March 1960.

Complete background information on the PM-1 elements, including a summary of all destructive and nondestructive test results, will be furnished to the WTR Safeguards Committee to facilitate approval of the program. Among the tests considered most important in a pre-irradiation examination are:

(1) Thermal cycling.

(2) Metallographic examination.

(3) Porosity.

(4) Ultrasonic inspection.

(5) Radiography.

(6) Deformation.

Also, all previous irradiation test results will be reviewed.

A conceptual design for the holder for irradiation specimens will be supplied to Westinghouse, primarily to indicate the manner in which the specimens should be positioned in the loop. Final design and fabrication of the holder will be Westinghouse's responsibility.

A statement of work covering pre-irradiation inspection, irradiation and post-irradiation testing of four full-length PM-1 fuel tubes and one burnable poison rod will be furnished to Westinghouse as an invitation for their bid.

Subsequent to the above meeting, three different holders for PM-1 elements in the WTR reactor loop (3-inch diameter) were designed. Two of the designs will accommodate four elements and will allow a center-to-center spacing between elements which is approximately equal to that specified for the standard PM-1 core. The third design is for five elements; however, the center-to-center spacing between the fuel tubes and poison rod has been reduced to accommodate this number. The closer packing of elements in this manner will be investigated further to see if any deleterious effects could possibly result.

The statement of work was prepared and sent to Westinghouse as planned. After several conferences with WTR personnel, a proposal was received. Contractual negotiations are currently in progress.

3. SM-1 Irradiation Program

Under this program, a simulated SM-1 fuel element (containing PM-1 core components) will be inserted into the SM-1 reactor for irradiation in an environment resembling that of the PM-1 reactor. In this test, the simulated assembly will replace an SM-1 fuel element as an actual operating component in the SM-1 core.

Initial studies for the adaptation of PM-1 elements to a standard SM-1 fuel element enclosure resulted in two holders or retaining jig designs. Both holders employed 14 PM-1 fuel tubes, 2 burnable poison rods and 5 SM-1 fuel plates since it was found that this combination most closely approximates the SM-1 fuel element loading and reactivity while still maintaining a desirable configuration. The first holder design comprised two multiorificed grid plates which were drilled to accommodate 0.500-inch OD fuel tubes and poison rods. The second design consisted of a grid of diagonal struts (screen type) to reduce the overall pressure drop, permit greater coolant flow area for the outside of the tubes and preclude streaming or channeling of the coolant which could result in dead spaces at the coolant inlet end of the bundle. These designs were later discarded in favor of a design in which the PM-1 elements were swaged at each end to reduced diameters of 0.349-inch OD and 0.265-inch ID. In this case, the holders consist of multi-orificed grid plates, one at each end of the bundle, but the reduced swaged diameters of the tubes permit the required extra flow area to the outside of the tubes. This design is completely compatible with pressure drop characteristics of the SM-1 core. An additional factor favoring selection of this design is that such swaged tubes are identical to the tubes which will be employed in the PM-1 core. This replacement bundle was designed on the basis of thorough thermal, hydraulic and nuclear analysis in order to simulate as closely as possible a standard SM-1 plate-type fuel element and thereby obviate any inconsistencies in SM-1 operation.

These analyses, together with a complete description of the modified PM-1 elements and a final assembly design, were compiled in report form (Report No. MND-M-2321) and submitted to the Atomic Energy Commission, Division of Reactor Development, ARM. This report was prepared to support and supplement an SM-1 Test Request Form titled "PM-1 Fuel Element Radiation Stability Test," dated 30 September 1959, and submitted to the New York Operations Office of the AEC.

The remainder of this subtask section comprises substantial excerpts from this report.

4. Description of the PM-1-M Fuel Element

In order to avoid confusion in the nomenclature used for fuel units, the following designations have been established.

(1) PM-1 fuel tube or "fuel tube"--a tube containing a cermet composed of UO_2 and stainless steel, clad with stainless steel.

(2) SM-1 fuel plate or "fuel plate"--a flat plate containing a cermet composed of UO_2, stainless steel and B_4C, clad with stainless steel.

(3) Burnable poison rod or "poison rod"--a cylindrical rod composed of a natural boron-stainless steel alloy.

(4) SM-1 fuel element--a standard SM-1 (APPR-1) fuel unit composed of end boxes, dead side plates and 18 SM-1 fuel plates.

(5) PM-1-M fuel element--a fuel element (PM-1-Modified) containing PM-1 fuel tubes, burnable poison rods, and SM-1 fuel plates, enclosed in the SM-1 fuel element envelope.

a. Fuel tubes

Each fuel tube is a composite tube in which the meat or fueled section is 30 mils thick and consists of UO_2 (highly enriched) homogeneously dispersed in a stainless steel matrix. Dead ends (1-1/8-inches long) of stainless steel are incorporated at the ends of the meat. The complete tubular core is then clad on both the inner and outer surfaces with stainless steel (AISI Type 347 stainless steel, modified to 0.072 wt % Co and 0.098 wt % Ta) tubing, 6 mils thick. The hole assembly is heat treated to achieve a sound metallurgical bond between clad and core. Each tube will have an active fueled length of 22 inches and an overall length of 24-1/4 inches. The inside and outside diameters of the fueled section of the tubes are 0.416 inch and 0.500 inch, respectively. However, both dead ends are swaged to the smaller dimensions of 0.349-inch OD and 0.265-inch ID during fabrication (see Fig. II-3a).

The composition of the active core is given below:

Composition of the Core of the Fuel Tube

Constituents	Total Weight (gm)	Weight Percent
302-B stainless steel (304-SS, with high Si)	88.4	71.4
UO_2 (highly enriched)	35.5	28.6
U-235 (as UO_2)	28.9	23.3

b. Burnable poison rods

Each burnable poison element is comprised of a single unclad rod containing 0.510 wt % natural boron distributed homogeneously as an alloy of AISI Type 347 stainless steel.

The rod has an overall maximum diameter of 0.500 inch. The active material is 22-3/8 inches long, but each end is threaded to accommodate a dead stainless steel end piece which has been machined to simulate the swaged end of a fuel tube. The total length including the end pieces is 24-1/4 inches. Each rod contains a total weight of 2.7 grams of natural boron.

c. PM-1-M fuel element assembly

Fourteen fuel tubes and two burnable poison rods will be symmetrically positioned between five equally spaced SM-1 fuel plates in an overall configuration having the identical external dimensions of the standard SM-1 fuel element. The Martin Company will furnish all fuel tubes to be irradiated, procure all equipment and fixtures (including fuel plates) necessary for the fabrication of this bundle and subsequently undertake its manufacture. Dead side plates, fuel plates, end box fixtures, etc., have been obtained from Sylvania-Corning Nuclear Corporation from their production run for SM-1, Core 2. In conformance with the standard

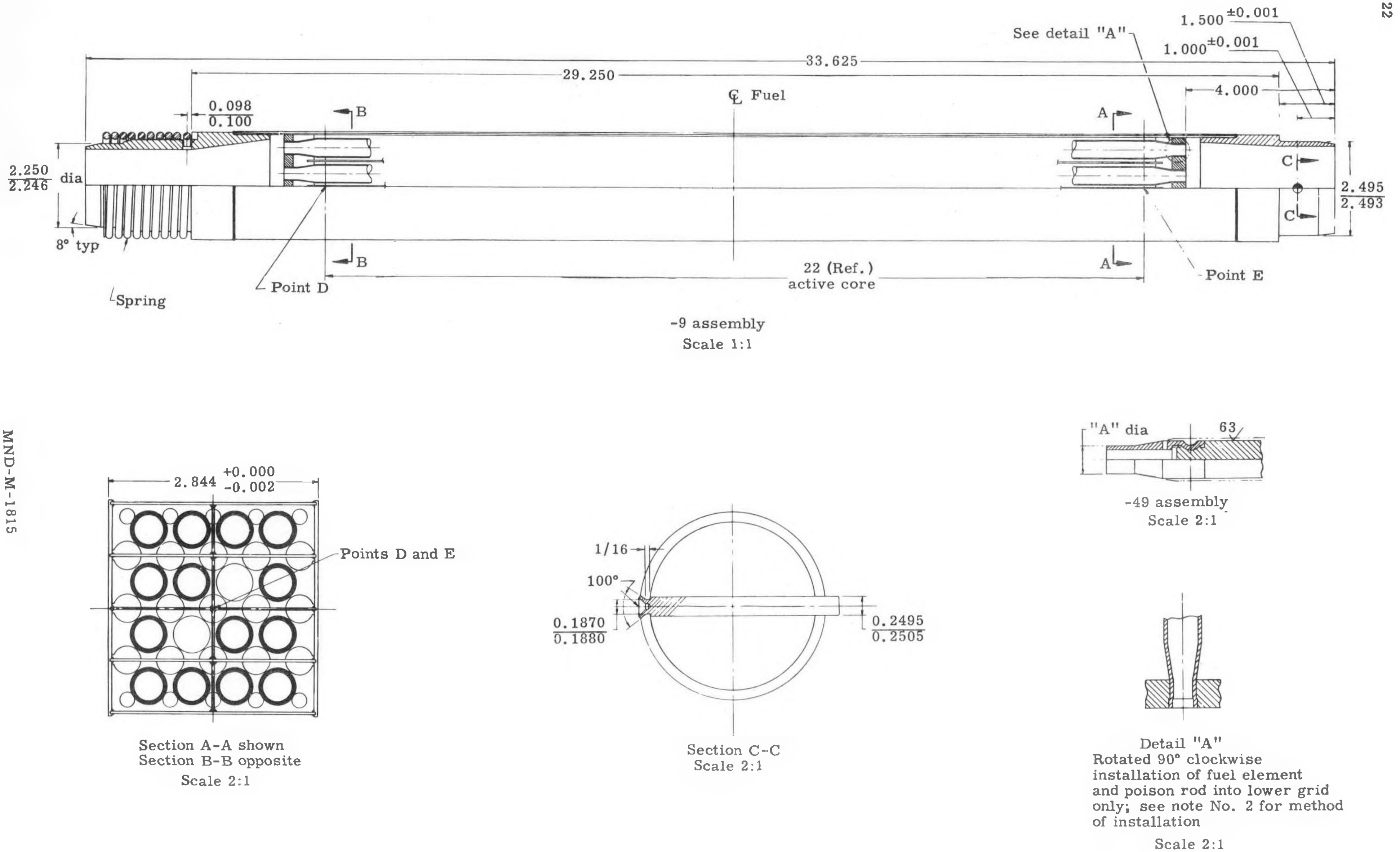


Fig. II-3a. PM-1-M Special Fuel Element (Irradiation Test, SM-1)

SM-1 fuel element design, the PM-1-M fuel element will be in a box-type configuration with the sides of the box consisting of 2 dead side plates and 2 fuel plates. Enclosed in this box are 14 PM-1 fuel tubes, 2 burnable poison rods and 3 equally spaced fuel plates. The SM-1 fuel plates are brazed into machined slots in the dead side plates, but the PM-1 components are held in place by perforated grid plates which are welded to each end of the box-like enclosure. Holes are drilled in these grid plates to accommodate the swaged ends of the fuel tubes and poison rods and to permit coolant flow through the tubes. The fuel tubes and poison rods are fixed in place in the grid at the coolant inlet end of the fuel element by expanding the tube ends into the grid holes. Tube holes in the grid plate at the outlet end are machined oversize to obtain a slide fit with the elements, thereby allowing for thermal expansion. Additional holes (fifteen 0.380-inch diameter and ten 0.200-inch diameter holes) are drilled in both the top and bottom grids to permit adequate coolant flow on the outside of the tubes. All external dimensions and end fixtures are identical to the present SM-1 fuel element so this modified element should be mechanically compatible with all aspects of the SM-1 core. The complete bundle contains approximately 547.6 grams of U-235 (including five SM-1 fuel plates) and 5.4 grams of natural boron. This compares favorably with a standard SM-1 fuel element which contains 516.9 grams of U-235 and 2.65 grams of natural boron. It is shown in the later section on Nuclear Analysis that the expected increase in reactivity due to a larger amount of U-235 in the PM-1 experimental element is more than balanced by the larger amounts of boron and stainless steel present.

5. Hydraulic and Thermal Analysis

a. Hydraulic analysis

The final design of the PM-1-M fuel element containing PM-1 components calls for two retaining grid plates at each end to hold the PM-1 fuel tubes and poison rods in place. Each of these grid plates have fifteen 0.380-inch diameter holes and ten 0.200-inch diameter holes which comprise the outside flow area for the tubes. Coolant flow to the inside of the 14 fuel tubes occurs through the swaged fuel tube ends which have an inside diameter of 0.265 inch.

The average coolant velocity in the SM-1 core is given as 4.12 fps (Ref. 1) which is corrected by multiplying by 0.906 (Ref. 2) to obtain an average velocity of 3.73 fps in the confined or internal channels. Since the flow area in the SM-1 fuel element assembly is 0.0456 ft^2 (Ref. 1), the average flow rate to the inside of the fuel element is 76.4 gpm. For various locations in the SM-1 core, the following V/V_{avg} are given for orifice plate No. 3 (Ref. 3).

TABLE II-1

V/V_{avg} in the SM-1 Core

Location Identification	V/V_{avg}	Q (gpm)
33, 35, 53, 55	1.347	102.9
62, 22, 26, 66	0.922	70.4
73, 75, 51, 57 31, 37, 13, 15	0.752	57.5

Pertinent flow areas for the PM-1-M fuel element are as follows:

TABLE II-2

Flow Areas for PM-1-M Fuel Element
(see Fig. II-4)

1. At top and bottom grids:

   (a) Total flow area to inside of tubes (14 x 1.265 in.) = $0.005354\ ft^2$

   (b) Total flow area to outside of tubes (15 - 0.380 in. dia; 10 - 0.200-in. dia) = $0.01399\ ft^2$

2. At element midplane:

   (a) Inside flow area

      i. Total = $0.01321\ ft^2$

      ii. Area "A" = $0.000944\ ft^2$

   (b) Outside flow area

      i. Total = $0.02857\ ft^2$

      ii. Area "A" = $0.00132\ ft^2$

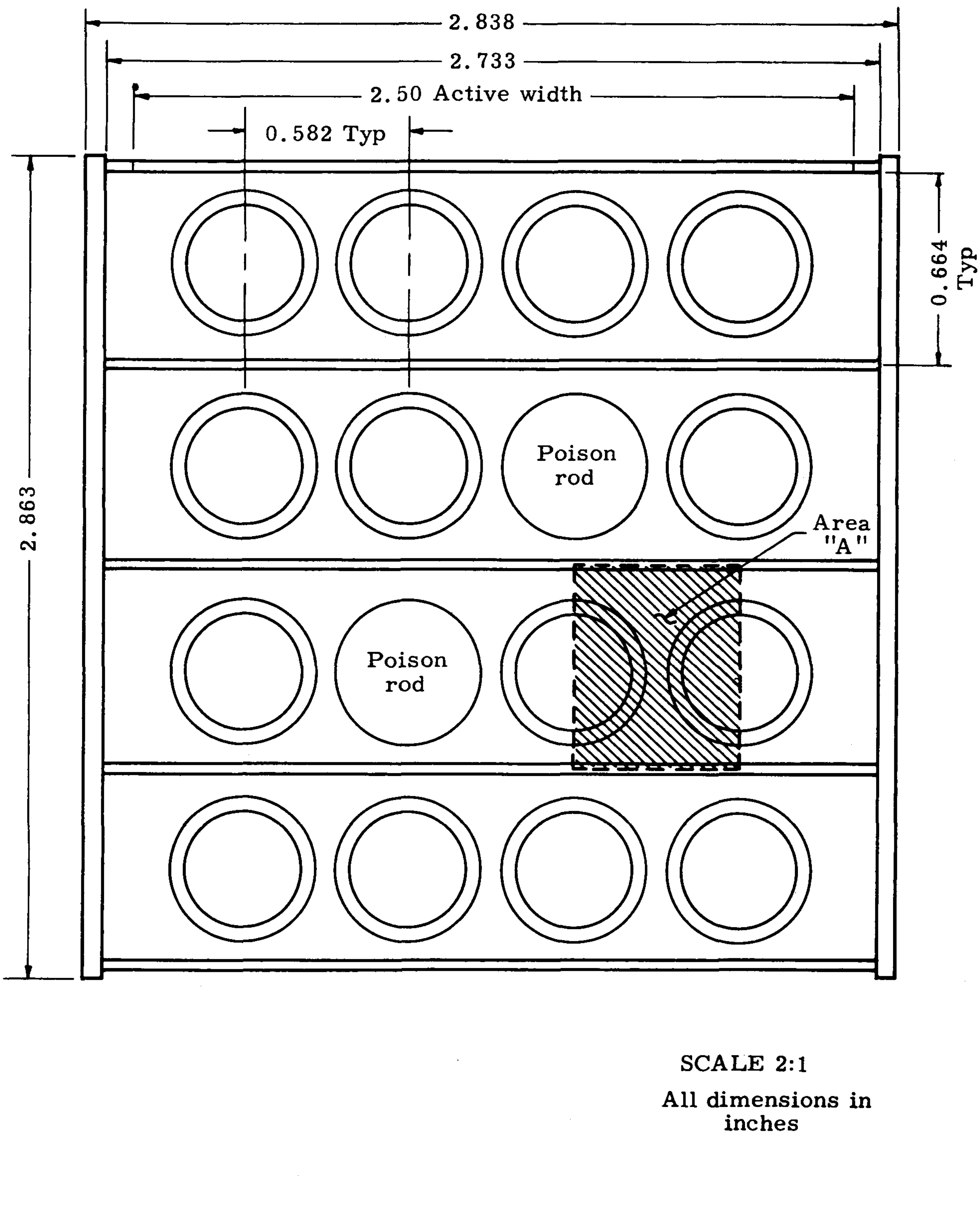


Fig. II-4. Fuel Element Assembly Mid-Plane

Utilizing the flow areas presented in Table II-2, the pressure drops experienced by flow on the inside and outside of the tubes (contraction, expansion and friction losses) were calculated and summed into ΔP_I and P_O terms, respectively. Employing the relationship:

$$V \text{ (velocity in fps)} = \frac{Q \text{ (flow rate in cfs)}}{A \text{ (area in ft}^2\text{)}},$$

the pressure drop through the PM-1 fuel tube bundle (inside the tubes) was found to be:

$$727\ Q_I^{\ 2} + 44.8\ Q_I^{\ 1.75} = \Delta P_I \text{ (ft of } H_2O) \qquad (1)$$

and the pressure drop through the outside flow channel of the modified assembly (outside the tubes) was found to be:

$$104.2\ Q_O^{\ 2} + 18.88\ Q_O^{\ 1.75} = \Delta P_O \text{ (ft of } H_2O), \qquad (2)$$

where

Q_I = inside flow rate (cfs)

Q_O = outside flow rate (cfs).

Since the pressure drop across the length of the bundle is the same for both the inside and outside channels ($\Delta P_I = \Delta P_O$), expressions (1) and (2) were equated. Subsequent application of the relationship:

$$Q_O \text{ (outside flow rate in cfs)} + Q_I \text{ (inside flow rate in cfs)} =$$

$$Q_T \text{ (total flow rate in cfs)} = 0.1415 \text{ cfs}$$

results in a flow split of 29% of the total flow passing through the inside of the fuel tubes and 71% of the total flow in the outside flow channel. Adding the other head losses of the PM-1-M fuel element (losses from the fuel tube ends to the mouth of the header, but excluding the standard SM-1 orifice plate), the total head loss across the assembly was found to be:

$$58.0\ Q_T^{2} + 10.36\ Q_T^{1.75} = \Delta P_{TB}\ (\text{ft of } H_2O). \qquad (3)$$

NOTE: 1. This equation does not include pressure drop across the SM-1 orifice plate.

2. TB = tube bundle.

The head loss in the SM-1 fuel assembly is given as follows:

Flow (gpm)	ΔP_B (ft of H_2O)
100	1.25
200	4.70

These values were then plotted on log paper to obtain a curve of pressure drop across the 18-plate SM-1 bundle (ΔP_{PB}) versus the flow rate (Q). It should be noted that this ΔP_{PB} does not include the pressure drop across the standard SM-1 flow-adjusting orifice plate (ΔP_{OR}). By employing this curve with the flow rates specific to the various assembly locations, listed in Table II-1, it was possible to determine the pressure drop across the fuel plate assembly (ΔP_{PB}) for each assembly location. Knowing that the total pressure drop across the core (ΔP_T) from plenum to plenum is 4.75 feet of H_2O (Ref. 1), the pressure drop across the orifice plate (ΔP_{OR}) is then readily determined for each location; this, in turn, enabled calculation of the corresponding orifice plate coefficients. Knowledge of the orifice plate coefficients coupled with the use of Equation (3) permitted determination of the resultant flow rate through the PM-1-M fuel element assembly for the various SM-1 core locations. Results are given in Table II-3.

TABLE II-3

Flow Rate Through the PM-1-M Fuel Element in the SM-1

Location Number	Head Loss (ft of H_2O) ΔP_{TB} (bundle)	ΔP_{OR} (orifice plate)	Flow (gpm)
35	2.90	1.85	79.3
26	1.70	3.00	61.1
57	1.30	3.45	53.0

For purposes of comparison, Table II-4 covers pressure drop and flow data for a stationary SM-1 fuel element in the various locations listed in Table II-3.

TABLE II-4

Flow Rate for a Stationary SM-1 Fuel Element

Location Number	Head Loss (ft of H_2O) ΔP_{PB} (bundle)	ΔP_{OR} (orifice plate)	Flow (gpm)
35	1.60	3.15	103.0
26	0.77	3.98	70.5
57	0.52	4.23	57.5

b. Thermal analysis

Calculations to determine surface temperatures of the tubes and plates were based on the considerations given below.

(1) The integrated average heat generation per fuel element assembly for various locations in the SM-1 core was established in accordance with the following conditions:

(a) Rod bank at 6.3 inches.

(b) Heat generation in the channels of the shim rods (Ref. 4) and center rod was neglected.

(c) The reactor power is 10 Mw.

(d) Reference 5, Fig. 58, was used to determine the integrated average heat generation per fuel element assembly as follows:

$$q_r = \frac{3413 \times 10^4}{\Sigma Q_r} \quad Q_r = 818{,}100\ Q_r \text{ Btu/hr},$$

where

ΣQ_r = summation over all the fuel elements and the two safety rods of the normalized integrated activity per fuel element assembly as shown in Fig. 58 of Ref. 5.

Q_r = integrated average activity of fuel element assembly at any "r" location.

(2) The axial distribution of the heat generation for a given fuel element assembly was determined by the relative activity curves in Figs. 54 and 55 of Ref. 5 for various fuel element assembly locations and for 6.3-inch rod location.

$$q_r(z) = 818{,}100 \; P_r(z) \;\; \text{Btu/hr},$$

where

$q_r(z)$ = heat generation, Btu/hr, per PM-1-M fuel element assembly at the "r" location in the SM-1 core and at an axial distance "z" above the bottom of the core.

$P_r(z)$ = relative activity at axial distance "z" and "r" location as per Figs. 54 and 55 of Ref. 5.

(3) The heat flux at any axial position of an assembly is directly proportional to the fuel loading and thickness. Since the PM-1 fuel tube has a loading per unit surface area which is 75% greater than the SM-1 plate, the heat flux is divided as follows:

$$(q/A)_p = \frac{1}{A_p + 1.75\,A_t}\, q_r(z), \text{ plates}$$

$$(q/A)_t = \frac{1.75}{A_p + 1.75\,A_t}\, q_r(z), \text{ tubes},$$

where

A_p = total heat transfer surface area, plates.

A_t = total heat transfer surface area, tubes.

(4) The surface temperatures were investigated along channel "A," as shown in Fig. II-4, which yields the highest heat generation to flow rate ratio in the PM-1-M fuel element assembly. Formulas used to calculate the surface temperatures are summarized as follows:

$(T_t)_i = 430 + \Delta\theta_i + (\Delta T_t)_i$, tube, inside wall temperature

$(T_t)_o = 430 + \Delta\theta_o + (\Delta T_t)_o$, tube, outside wall temperature

$T_p \quad = 430 + \Delta\theta_o + \Delta T_p$, plate, wall temperature

$$\Delta\theta_i = \int^z q_i(z)\, dz / (A_F)_i\, V_i\, \rho\, c_p$$

$$\Delta\theta_o = \int^z q_o(z)\, dz / (A_F)_o\, V_o\, \rho\, c_p$$

$$(\Delta T_t)_i = (q/A_H)_t \quad \frac{1}{h_i}$$

$$(\Delta T_t)_o = (q/A_H)_t \quad \frac{1}{h_o}$$

$$\Delta T_p = (q/A_H)_p \quad \frac{1}{h_o}$$

$$\int_0^z q_i(z)\, dz = \frac{z}{L} (A_H)_{ti} \quad (\overline{q/A})_t$$

$$\int_0^z q_o(z)\, dz = \frac{z}{L} \quad (A_H)_{to} \; (\overline{q/A})_t + (A_H)_p \; (\overline{q/A})_p$$

$(q/A)_t$ and $(q/A)_p$ is the mean heat flux from 0 to z.

The results of the thermodynamic study are plotted in Figs. II-5, II-6 and II-7 for various fuel element assembly locations. From these curves, it is seen that the maximum surface temperature decrease is proportional to the distance of the fuel element assembly from the center of the core.

The above figures show that the maximum difference between the inside and outside surface temperatures of the fuel tube is approximately 18° F. This difference is conservative since more heat would conduct to the cooler surface yielding approximately equal temperatures.

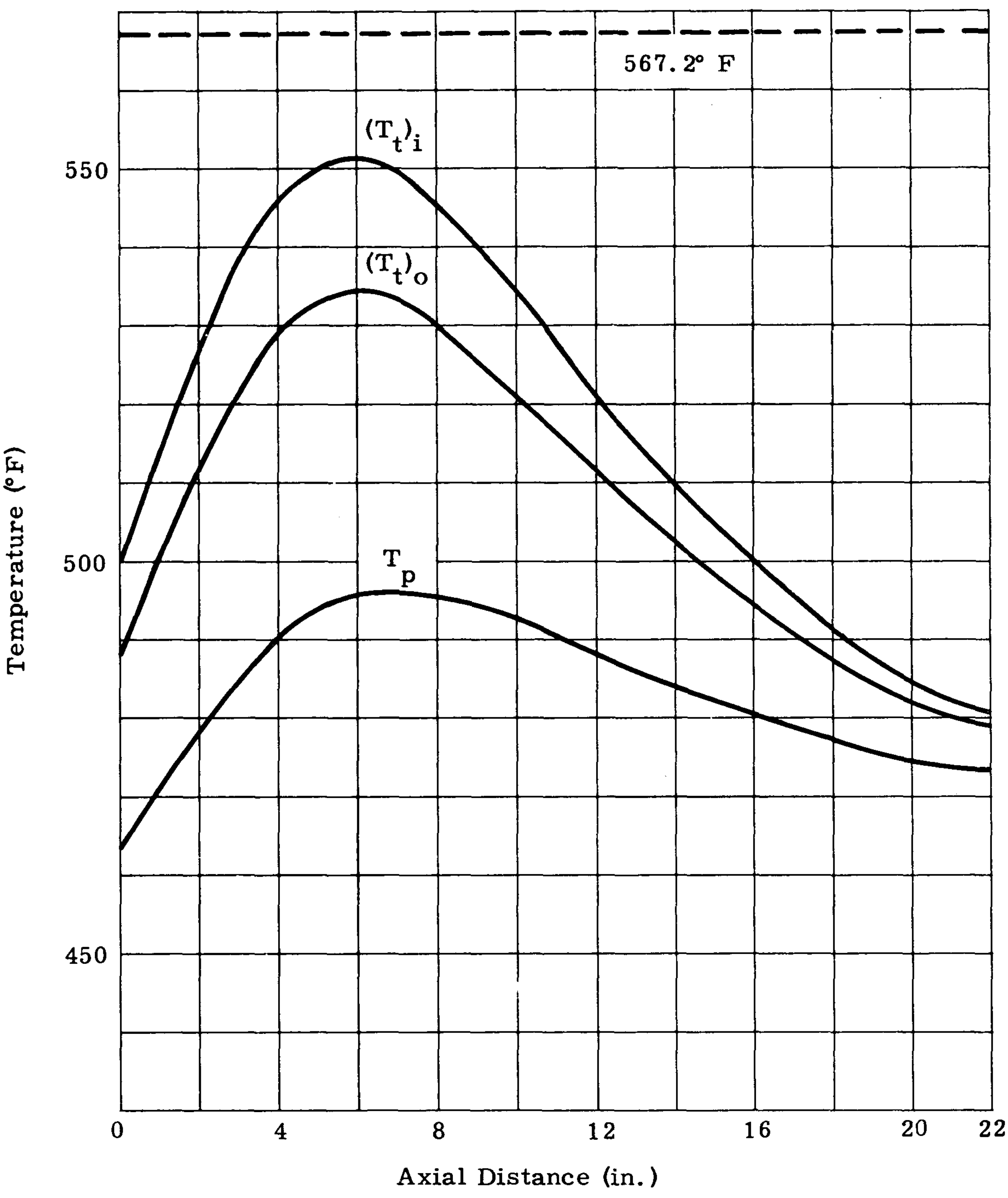


Fig. II-5. Surface Temperatures--Fuel Element Assembly No. 35

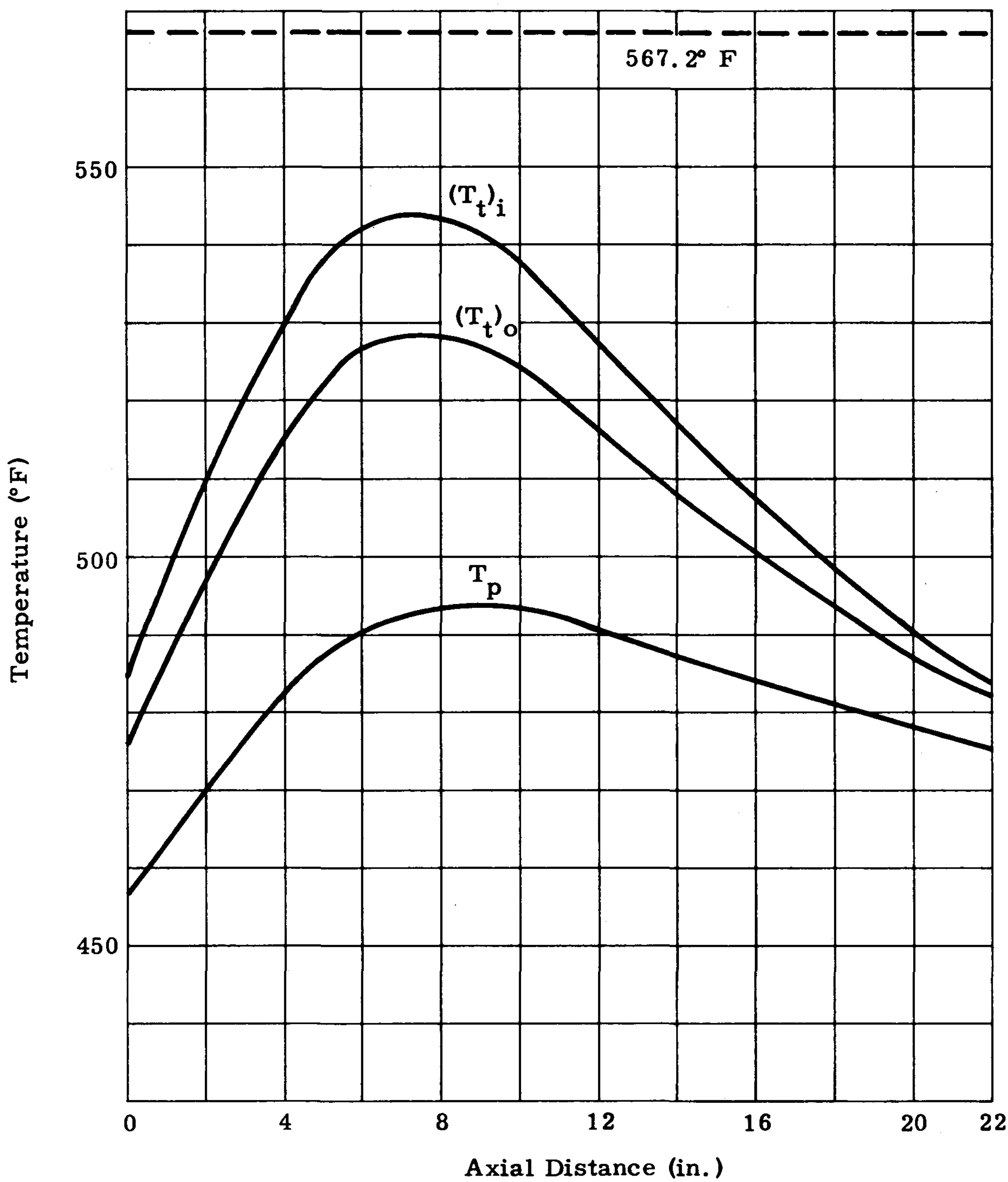


Fig. II-6. Surface Temperatures--Fuel Element Assembly No. 26

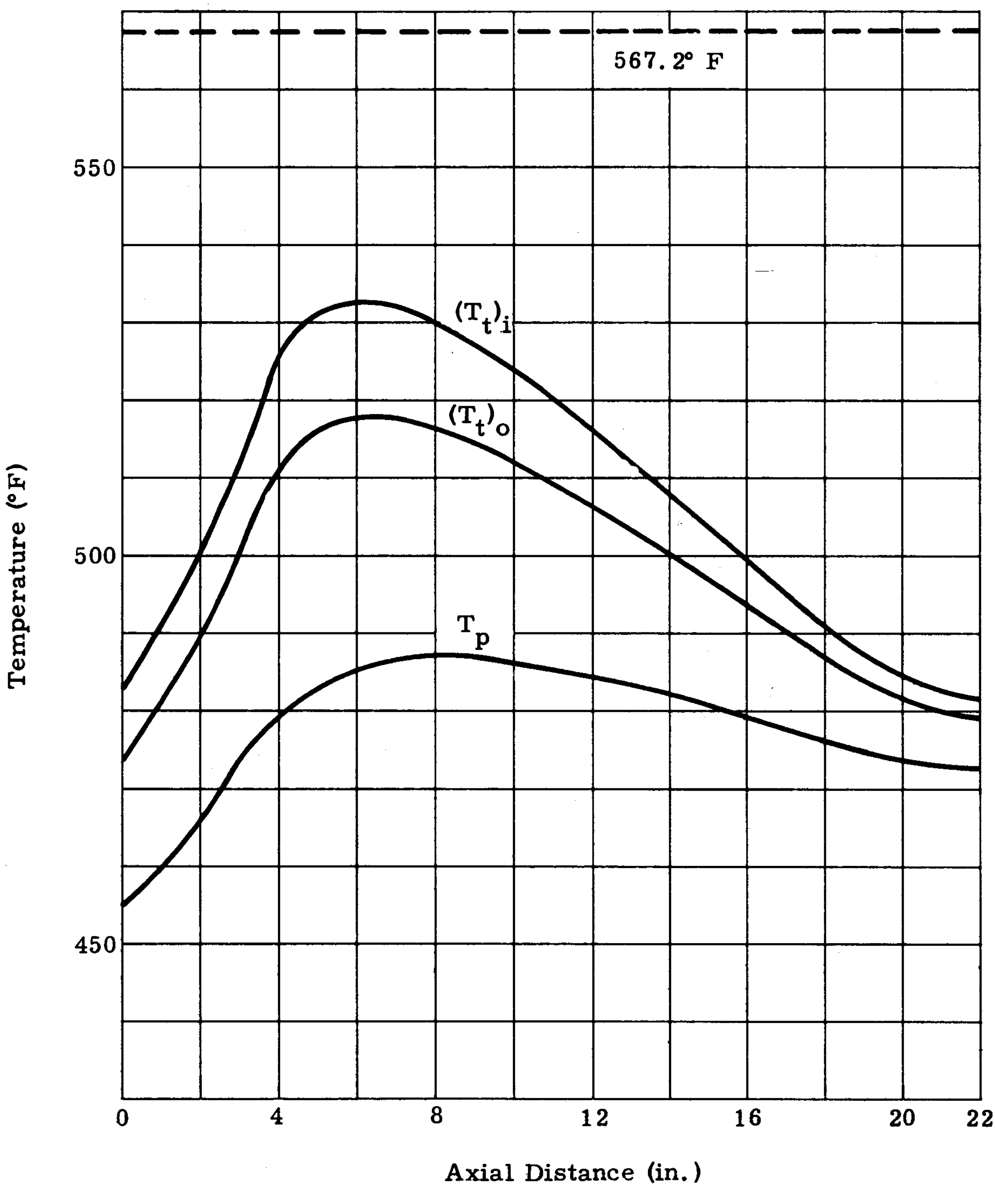


Fig. II-7. Surface Temperatures--Fuel Element Assembly No. 57

A summary of pertinent data on the PM-1-M fuel element is given in Table II-5.

TABLE II-5

Pertinent PM-1-M Fuel Element Data

A. General	
1. Number of SM-1 fuel plates	5
2. Number of PM-1 fuel tubes	14
3. Number of poison rods	2
B. Flow areas (ft^2)	
1. At top and bottom grids:	
a. Total inside flow area	0.005354
b. Total outside flow area (15--0.380-inch dia and 10--0.200-inch dia)	0.01399
2. At element midplane:	
a. Inside flow area	
(1) Total	0.01321
(2) Area "A"	0.000944
b. Outside flow area	
(1) Total	0.02857
(2) Area "A"	0.00132
C. Heat transfer areas (ft^2)	
1. Inside heat transfer area:	
a. Total	2.795
b. Area "A"	0.1997
2. Outside heat transfer area:	
a. Total	7.180
b. Area "A"	0.4178
(1) Tube	0.2400
(2) Plate	0.1778

c. Nuclear analysis

A one-group calculation of the reactivity change occurring at beginning of life upon substituting a PM-1-M fuel element into the SM-1 was made for several element locations employing the material worths and cell corrections from APAE-27 and the material specifications and cell corrections for PM-1 tubes and poison rods.

The following material worths (Ref. 6) were computed and later used witn the assumption that the SM-1 fuel cell correction applied to them.

	Element Location						
	34	35	25 or 23	26	14	57	16
U-235 cents/gm	+0.215	+0.264	+0.354	+0.116	+0.116	+0.124	+0.107
B-10 cents/gm			-67.0				
Stainless steel cents/gm			-0.0100				
H_2O displacement cents/gm of stainless steel			-0.0069				

The fuel plate cell correction was found to be 0.881 (Ref. 7). Normalization is to average moderator flux. (APAE-27 pp. 41 to 46 and Appendix.)

Table II-6 shows the composition of a normal SM-1 element and the PM-1-M element employed in these calculations.

TABLE II-6

Composition of Comparative Fuel Elements

Cell Corrections		SM-1 Element	PM-1-M Element
Fuel region	UO_2 SS B	0.881	0.881 (no boron)
Poison rods	SS B		0.483

TABLE II-6 (continued)

Cell Corrections	SM-1 Element	PM-1-M Element
Material:		
U-235 (gm)	515.13	547.59 (0.134 in plates)
B-10 (gm)	0.4829	1.148 (1.014 in rods)
SS in fuel (gm)	1796.0	2364
SS in clad (gm)	1281.4	1291
SS in poison rods		1051.75

From Table II-6, the substitution of the PM-1-M fuel element implies:

(1) An addition of 32.5 grams of U-235 at a cell correction of 0.881.

(2) Addition of 1.0141 grams of B-10 at a cell correction of 0.483 and the removal of 0.349 grams of B-10 at a cell correction of 0.881. This results in an addition of 0.207 grams of B-10 at a cell correction of 0.881.

(3) Addition of 568 grams of stainless steel at a cell correction of 0.881.

(4) Addition of 1051.75 grams of stainless steel at a cell correction of 0.483. This results in the addition of 576 grams of stainless steel at a cell correction of 0.881 with the water displacement corresponding to 1051.75 grams of stainless steel.

Combining the above material additions with their worths gives the results shown in Table II-7. (The material worth table was filled in using the ratio of U-235 worths times the unknown worth.)

TABLE II-7

Reactivity Differences Between the SM-1 and PM-1-M Fuel Elements

Element Location	34	35	25 or 23	26	14	57	16
$= \frac{\Delta k}{k}$	-0.0011	-0.00136	-0.00182	-0.00060	-0.00060	-0.00064	-0.00055

No correction was made for the tube center-to-center spacing of 0.665 inch in the normal PM-1 fuel tubes as compared to the value of 0.582 inch in the PM-1-M fuel element in employing the quoted PM-1 cell corrections. This fact would increase the worth of the added materials, tending to make the reactivity slightly more negative. The error in this approximation will tend to cancel the error of opposite sign due to the one-group approximation.

It is concluded that, at the beginning of life, the reactivity difference between the SM-1 and PM-1-M fuel elements is negligible.

CONCLUSIONS

Results of the hydraulic analysis indicate that the pressure drop characteristics of the PM-1-M fuel element are compatible with the SM-1 core and will not short-circuit coolant flow. At the standard SM-1 plenum-to-plenum pressure drop of 4.75 feet of water, the coolant flow through the PM-1 assembly is somewhat less than that through an SM-1 element in the same core location (compare Tables II-3 and II-4). However, thermal analysis of the bundle at different core locations indicated that this lower flow can be tolerated without fear of excessive temperatures

As shown in Figs. II-2, II-3 and II-4, maximum surface temperatures of the fueled components decrease in proportion to the distance of the fuel element assembly from core center. Therefore, in order to maintain relatively low surface temperatures, the PM-1-M fuel element should be placed in either Position 26 or Position 57. In both cases, the maximum fuel element surface temperature occurs on the inside of a fuel tube, being about 24° F below coolant saturation temperature at 1200 psi in Position 26 and about 35° F below saturation in Position 57. Considering the likelihood of an unpredictable thermal flux peak in Position 57 which could lead to local boiling in the element, as well as the higher fuel depletion rate in Position 26, it is recommended that the PM-1-M fuel element be placed in Position 26 or its mirror location in the SM-1 core lattice. Here, no danger exists from excessive temperatures.

Calculations of reactivity changes occurring as a result of substituting an SM-1 fuel element with the PM-1 experimental bundle indicate a negligible effect. In fact, the reactivity change in all core locations considered was slightly negative. Although the PM-1 bundle contains an excess of U-235 (approximately 30 grams), it was found that the poisoning effect of the increased amounts of boron and stainless steel more than compensated for the expected increase in reactivity. As seen from Table II-7, the magnitude of the reactivity change is so small that it should have no appreciable influence on SM-1 operation.

As a final summation, it can be stated that the simulated SM-1 fuel element to be supplied by Martin for irradiation in the SM-1 is completely compatible with the SM-1 core.

REFERENCES

1. Telephone Call: W. Richards (Alco) and J. Beam (Martin) on 1 February 1960.

2. APAE Memo 108, "Coolant Flow Tailoring Program of the APPR-1 Core Employing a Full Scale Model of the Reactor Vessel," page 52.

3. APAE Memo 108, page 18, Table I.

4. APAE Memo 160.

5. APAE 8, "Army Package Power Reactor Zero Power Experiment (ZPE-1)," pages 108, 109 and 112.

6. APAE 27, pages 61 to 63.

7. APAE 27, pages 41 to 46 and Appendix A.

C. SUBTASK 2.3--REACTOR FLOW STUDIES

J. Starr, M. P. Norin, W. J. Taylor, J. Sevier

The objective of PM-1 Reactor Flow Studies is to evaluate and optimize the hydraulic design of the reactor. The work is being conducted through three tests: two serve to give preliminary information for use in reactor design and in design of the third test; the third test makes use of a full-scale flow model.

Work planned for this quarter included:

(1) Completion of the simplified flow model (1/4 scale) tests.

(2) Completion of bundle orifice test component fabrication.

(3) Completion of loop modifications necessary for the bundle orifice tests.

(4) Design of experimental program for the bundle orifice tests.

(5) Initiation of 5000-gpm loop modifications for full-scale tests.

(6) Completion of full-scale test pressure vessel.

(7) Completion of design of the full-scale test components.

(8) Completion of full-scale flow test instrumentation layout.

Actual accomplishments of this quarter were:

(1) Tests of the revised configuration of the simplified 1/4-scale flow model were completed.

(2) Fabrication of bundle orifice test components was completed.

(3) Loop modifications for the bundle orifice test were completed.

(4) The experimental test procedure for the bundle orifice test was issued.

(5) The complete instrumentation for the full-scale flow test was specified.

During the next quarter, it is expected that:

(1) Bundle orifice tests will be completed.

(2) Modifications to the 5000-gpm flow loop will be completed for the full-scale flow test.

(3) Component fabrication for the full-scale flow test will be 50% completed.

1. One-Fourth-Scale Flow Model Test No. 2

An experimental study of the revised configuration of the prototype reactor was completed using the simplified 1/4-scale flow model. The objective of the work was to determine, for design purposes, the gross coolant flow pattern of the prototype reactor as influenced by the inlet water box orifice plate and the outlet configuration.

The results of the tests are summarized below.

(1) Flow distribution in the inlet water box was found to be relatively independent of velocity and Reynolds number in the range tested which included the prototype design velocity.

(2) A pressure recovery from the kinetic energy of the inlet flow was observed. This caused a minimum flow from the water box at the inlet side of the vessel and a maximum flow at the opposite side. The flow variation $\frac{Q_{max}}{Q_{min}} - 1$ was about 42%.

(3) The flow was reduced to 36% in the flow annulus formed by the thermal shields. This reduction is not as great as that obtained in the initial configuration.

(4) In general, the changes in the water box configuration coupled with the enlargement of the water box orifices produced more severe distortions of flow through the water box orifice plate and the flow annulus formed by the thermal shields.

(5) The overall pressure drop measured from inlet to outlet was 24.2 feet of head for a prototype flow of 2125 gpm as compared to the predicted value of 27.6 feet. The orifice diameter in the water box orifice plate, however, was increased by 11% over the value used in the predictions.

In the flow model, the internal geometry of the water box was duplicated in 1/4 scale. All other flow channel dimensions were approximately 1/4 size. The flow channels between the thermal shields were approximated by a single annulus to allow sufficient space for instrumentation sized according to good design practice. That is, the instrumentation was small with respect to the dimensions of the annulus. For simplicity, an orifice plate simulated the reactor core. The orifices were sized to approximate the core pressure drop at prototype velocity. The upper shroud was duplicated and contained discharge orifices equivalent to PM-1 design. A dual outlet was provided. One outlet simulated that of the prototype. The other outlet was located at the top of the model and was symmetrical about the vertical centerline of the vessel. Valving was provided so that either outlet could be used during testing. When the outlet through the top was used, the exit flow restriction did not disturb the flow distribution upstream. Thus, with this configuration, nonuniform distributions may be traced directly to the inlet configuration. The side outlet permits the determination of any nonuniform distributions traced to the PM-1 exit configuration.

Figure II-8 shows a sectional view of the model. This drawing is approximately 1/4 actual model size. The upper and lower outside sections and the upper shroud are made of plexiglass to allow visual observation during testing. Figures II-9, II-10 and II-11 show the model at various stages of assembly. Figure II-12 shows the model in test.

The major changes incorporated in the revised 1/4-scale test model were as follows:

(1) The water box was changed from a truncated annulus to one with a horizontal upper surface having a slight rise at the inlet. The flow area is constant throughout 66% of the water box (120 degrees each side of the point diametrically opposite the inlet).

(2) The axis of the outlet nozzle was located at a 60-degree displacement from the axis of the inlet nozzle. The elevation of the outlet nozzle is one diameter greater than that of the inlet nozzle. The original configuration had the outlet nozzle at 180 degrees from the inlet and at the same elevation as the inlet.

(3) The water box orifices were enlarged from 0.125 inch in diameter to 0.139 inch in diameter.

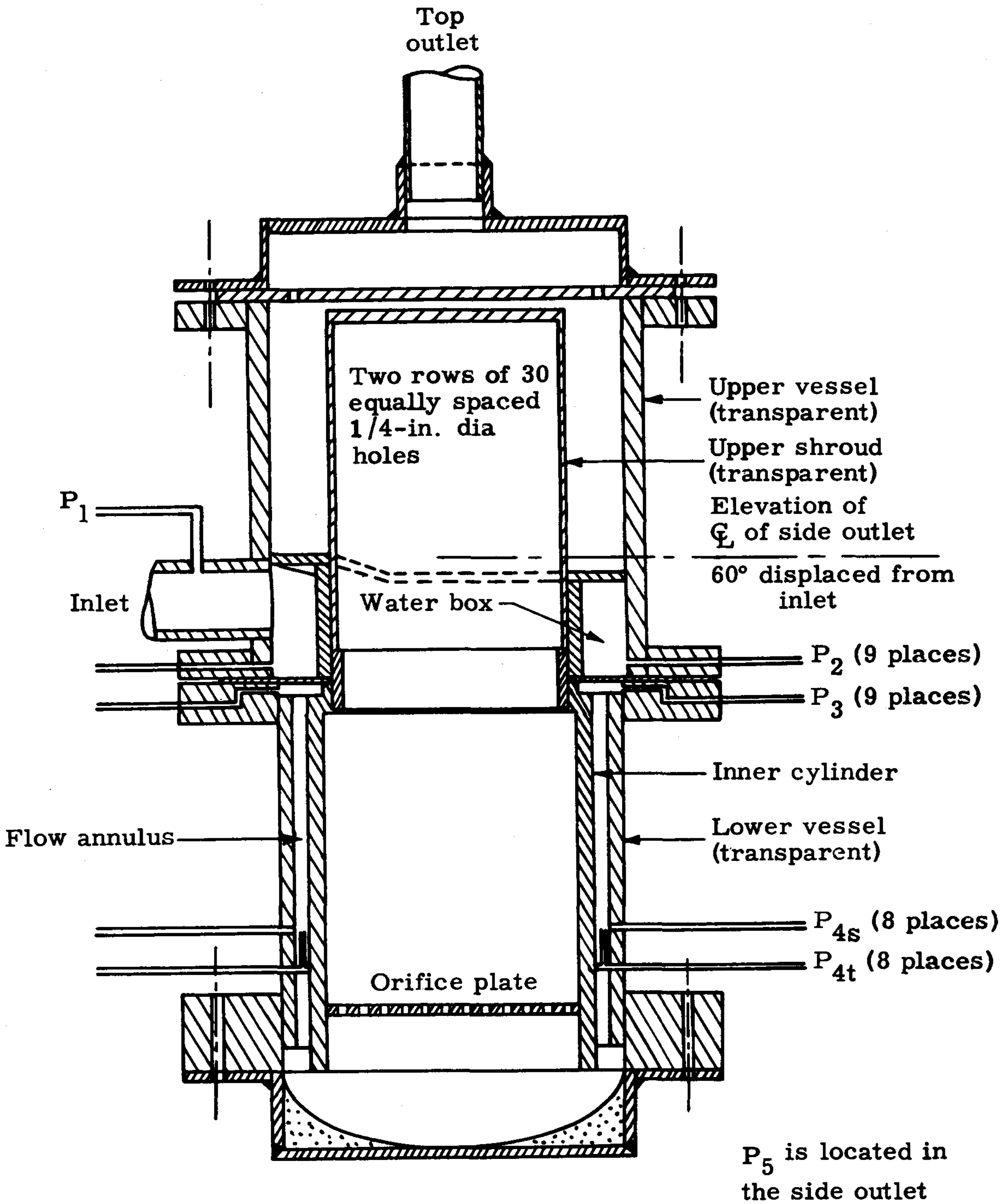


Fig. II-8. 1/4-Scale PM-1 Flow Model Revised Configuration

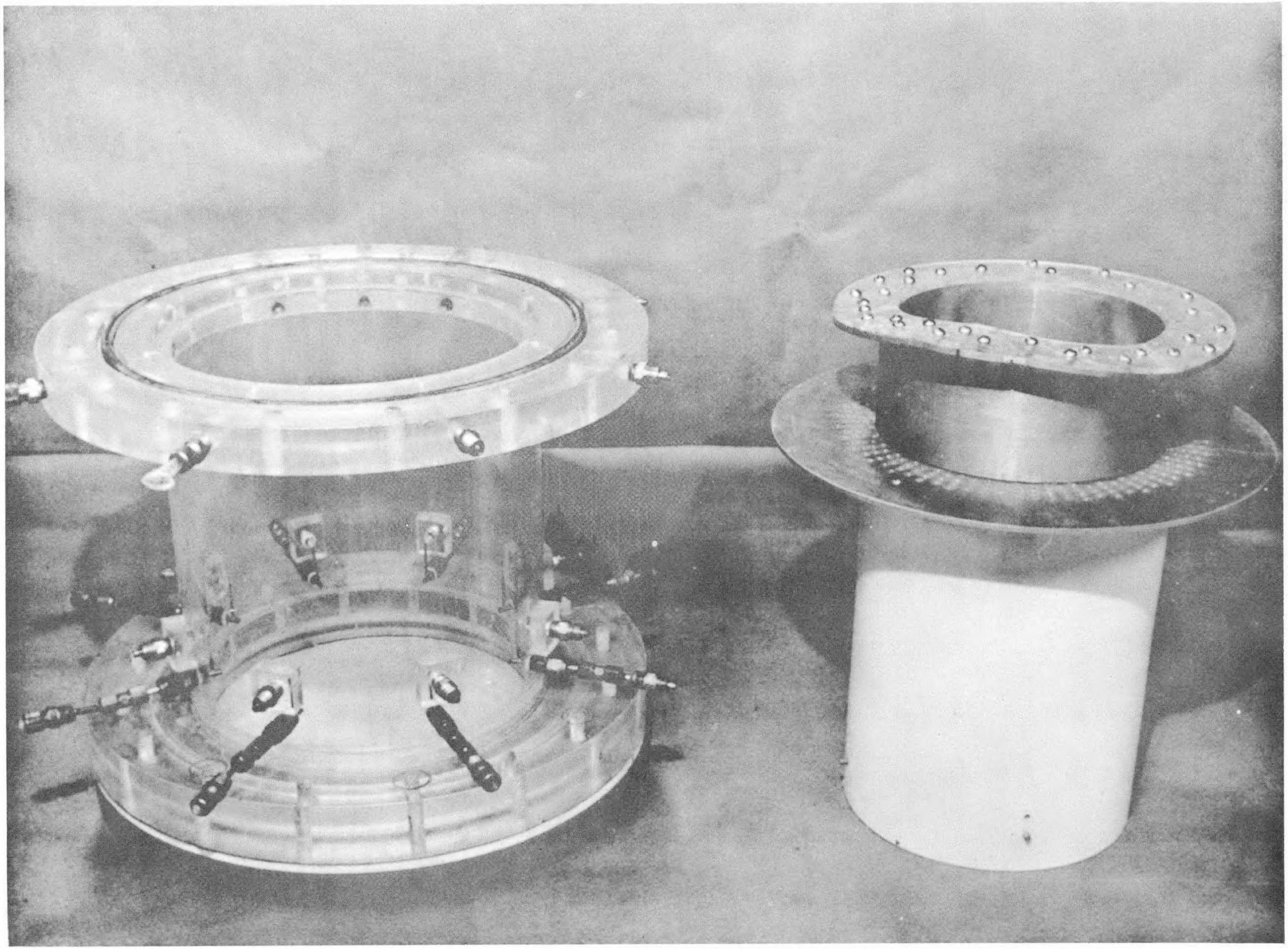


Fig. II-9. Lower Vessel and Internals (1/4-Scale PM-1 Flow Test No. 2)

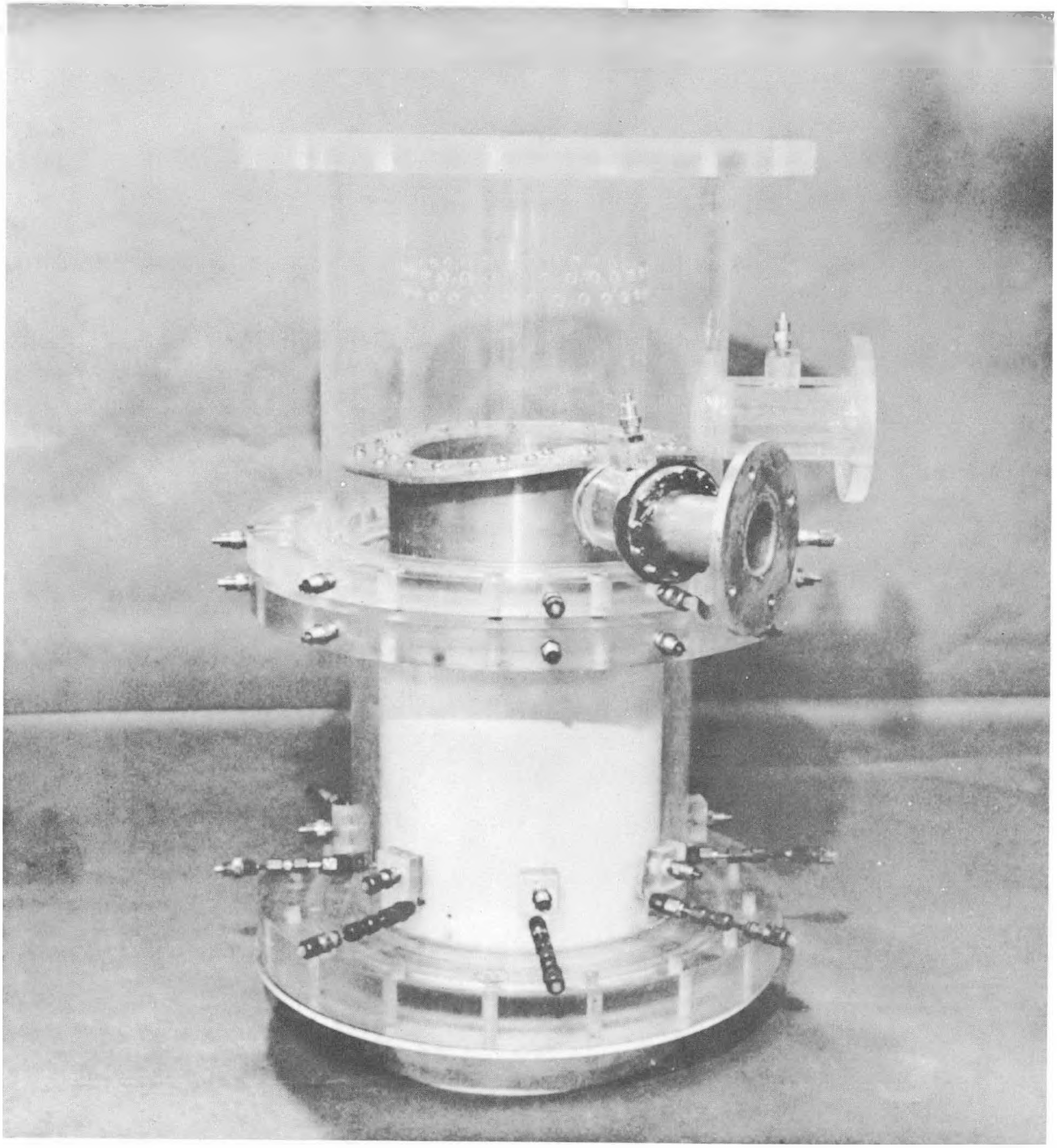


Fig. II-10. Assembly Showing Upper Vessel Forming Water Box (1/4-Scale PM-1 Flow Test No. 2)

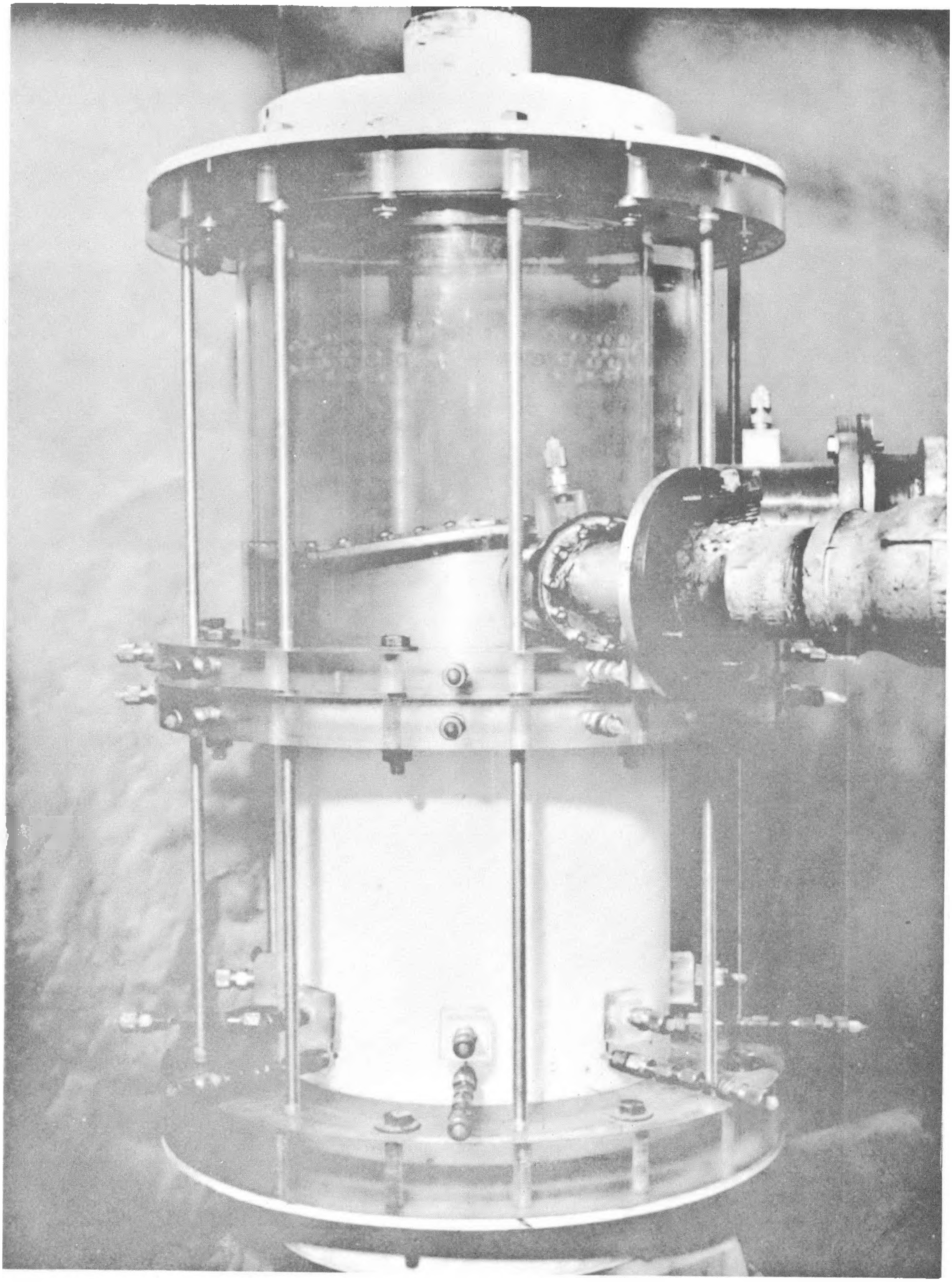


Fig. II-11. Assembled PM-1 Scale Test Vessel (1/4-Scale PM-1 Flow Test No. 2)

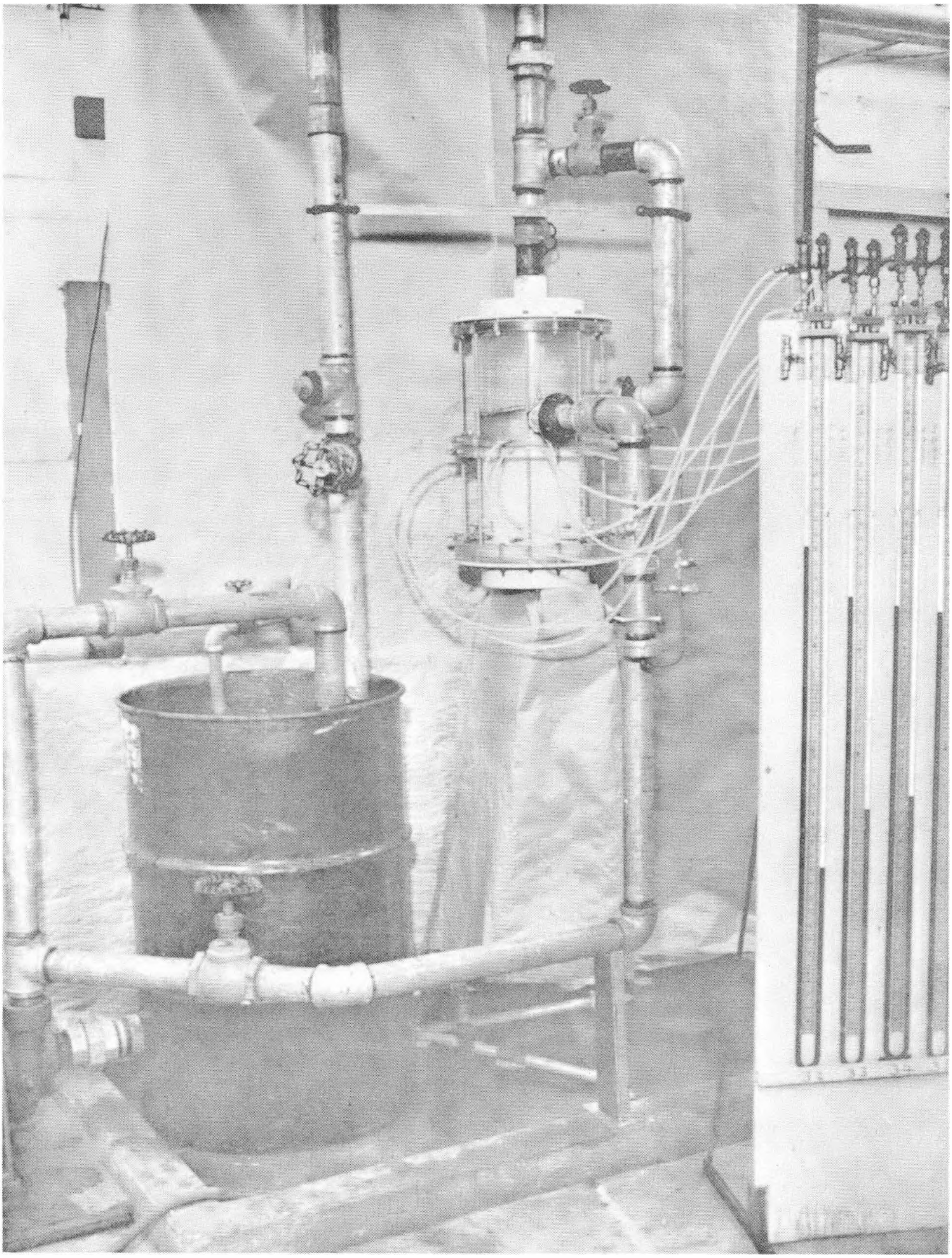


Fig. II-12. Installed Flow Model (1/4-Scale PM-1 Flow Test No. 2)

(4) A 30-degree inlet elbow similar to the PM-1 piping design was included.

The following specifications apply to the model:

(1)	Operating temperature	100° F maximum
(2)	Operating pressure	30 psig maximum
(3)	Inlet and exit pipe diameter	1-1/2 inches
(4)	Internal vessel diameter at water box	9 inches
(5)	Width of flow annulus between simulated thermal shields	0.374 inch
(6)	Maximum flow rate	160 gpm
(7)	Water box orifice plate	
	Quantity of holes	252
	Size of holes	0.139-inch diameter
(8)	Orifice plate simulating core	
	Quantity of holes	73
	Size of holes	0.310-inch diameter
(9)	Upper shroud	
	Quantity of holes	60
	Size of holes	0.250-inch diameter

Instrumentation used in the 1/4-scale model included: flange taps across the water box orifice plate located at 9 positions around the vessel; velocity head probes which read total and static pressures in the lower annulus below the water box at 8 positions around the vessel; and static pressure taps in the inlet and outlet nozzles which indicate the overall vessel head loss.

The flow test was performed in a cold water low pressure test loop which consists of a 55-gallon open suction tank, a centrifugal pump with a rating of 200 gpm at 155 feet, 2-inch diameter Schedule 40 loop piping, a test section, a flow orifice, a loop bypass and necessary control and bleed valves.

Loop instrumentation consisted primarily of an orifice plate which measured total flow rate. The orifice plate was fabricated in the laboratory and was sized to obtain an orifice diameter-to-pipe diameter ratio of 0.700.

A plot of the pressure drop across the water box orifice plate versus angular displacement around the vessel is given in Fig. II-13. The non-symmetry of the curves about Position A is explained in that a 30-degree elbow is located close to the inlet to the water box and directs the water toward Taps B, C and D. A high velocity component parallel to the orifice plate may account for readings at "B" being less than those at "A". An estimate of the maximum flow variation through the orifice plate may be made. Assume that the following relationship holds:

$$Q \propto (\Delta h)^{1/2}$$

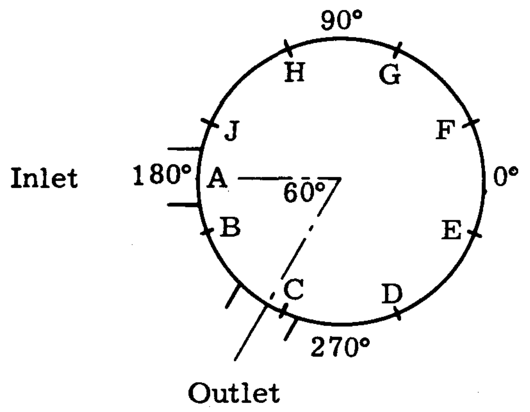
$$\frac{Q_G}{Q_B} = \left(\frac{\Delta h_G}{\Delta h_B}\right)^{1/2}$$

Then, for a model flow rate of 133 gpm, which conforms to the inlet flow velocity of the prototype,

$$\frac{Q_G}{Q_B} = \left(\frac{5.11}{2.53}\right)^{1/2} = 1.421$$

or, 42% more flow is passed at Position G as compared to B. This reasoning neglects any differences in flow coefficients which may exist at the respective locations.

An overall flow coefficient for the multihole orifice plate may be calculated from the data presented in Fig. II-13.



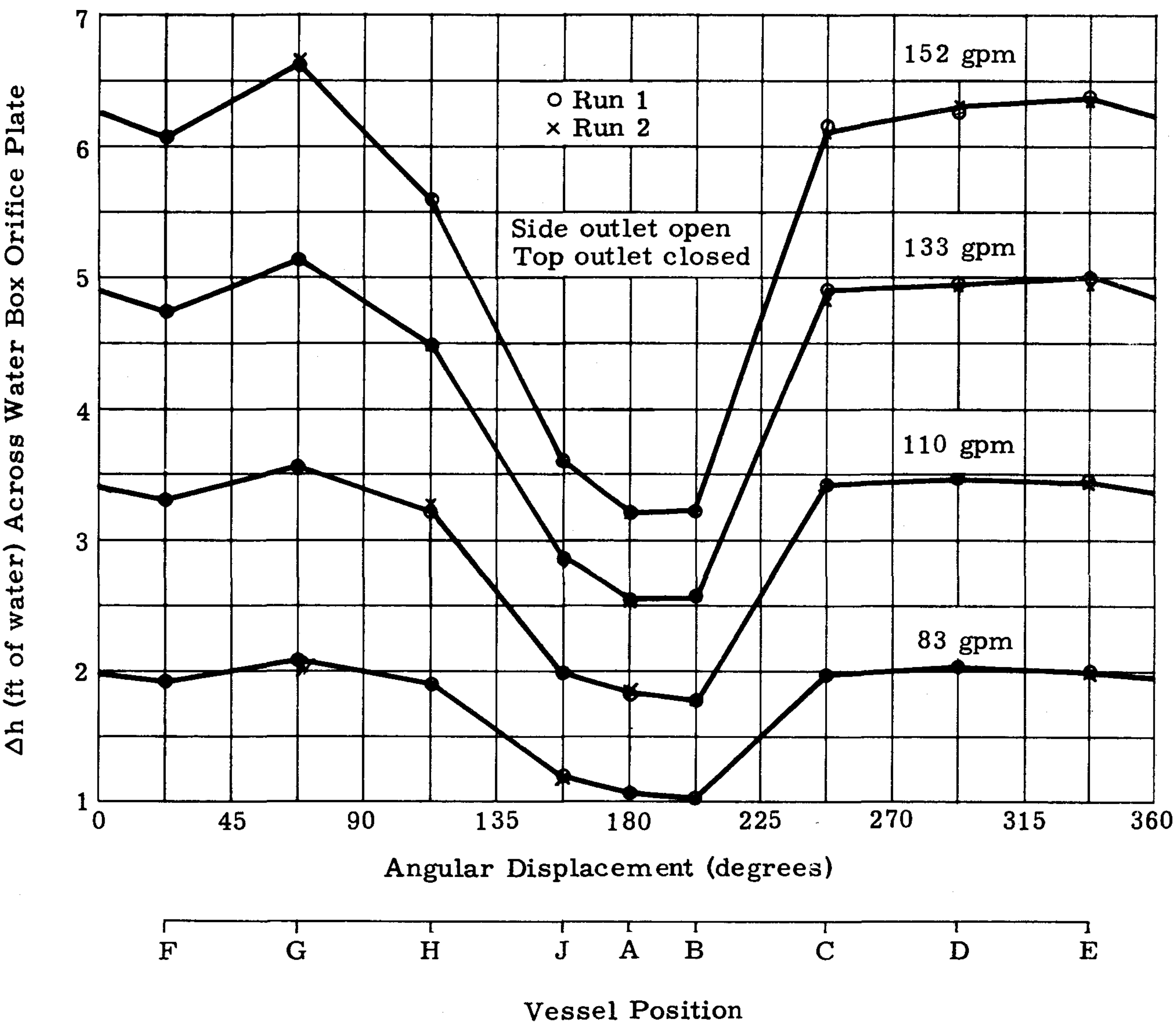


Fig. II-13. Head Loss Across Water Box Orifice Plate at Various Flow Rates (1/4-Scale PM-1 Flow Test No. 2)

The equation,

$$Q = KA\ (2g\Delta\bar{h})^{1/2},$$

may be solved for K (the flow coefficient) if $\Delta\bar{h}$ is known for a particular value of Q. $\Delta\bar{h}$ is the mean differential head across the orifice plate, obtained by integrating the curves of Fig. II-13. Position A is the total flow area of 252 orifices of 0.139-inch diameter. Values for the flow coefficient are given in the table below for each test flow rate. The flow coefficients are plotted in Fig. II-14.

TABLE II-8

Overall Flow Coefficients--Water Box Orifice Plate

Flow Rate, Q (gpm)	Mean Head, $\Delta\bar{h}$ (ft)	Flow Coefficient (K)
83	1.76	0.653
110	3.01	0.662
133	4.32	0.669
152	5.45	0.680

Cross plots of Fig. II-13 are presented in Fig. II-15. These curves do not deviate greatly from the relationship,

$$Q \propto h^{1/2}$$

as would be expected for an orifice.

The curves in Fig. II-13 are normalized in Fig. II-16 to allow a comparison of distribution at different flow rates; that is, normalized distributions of differential head across the water box orifice plate are shown for the 4 test flow rates. These were obtained by calculating the ratio of Δh to the mean differential head $\Delta\bar{h}$. The plot indicates that the distribution does not change appreciably with flow rate in the range tested. The greatest change takes place at Position G. Water is deflected at the inlet by the inclined upper surface of the water box and appears to impinge upon the orifice plate in the vicinity of Position G. This was observed visually when dye and air were injected into the inlet flow stream. A reduction or increase in inlet velocity could change this flow characteristic and thus affect the pressure drop at Position G to a greater extent than that at other positions.

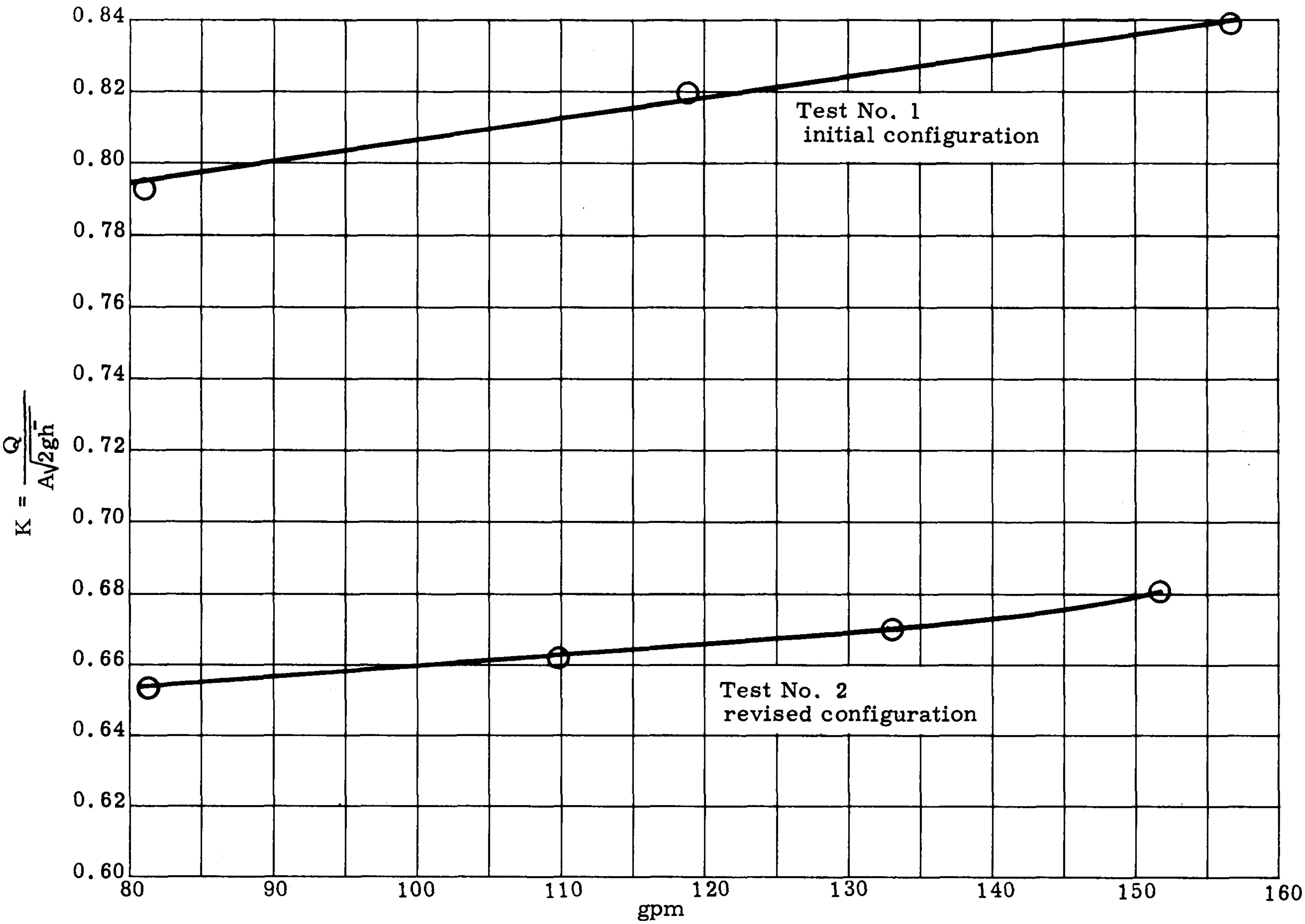


Fig. II-14. Overall Flow Coefficients for Tests 1 and 2 Versus Flow Rate (1/4-Scale PM-1 Flow Test)

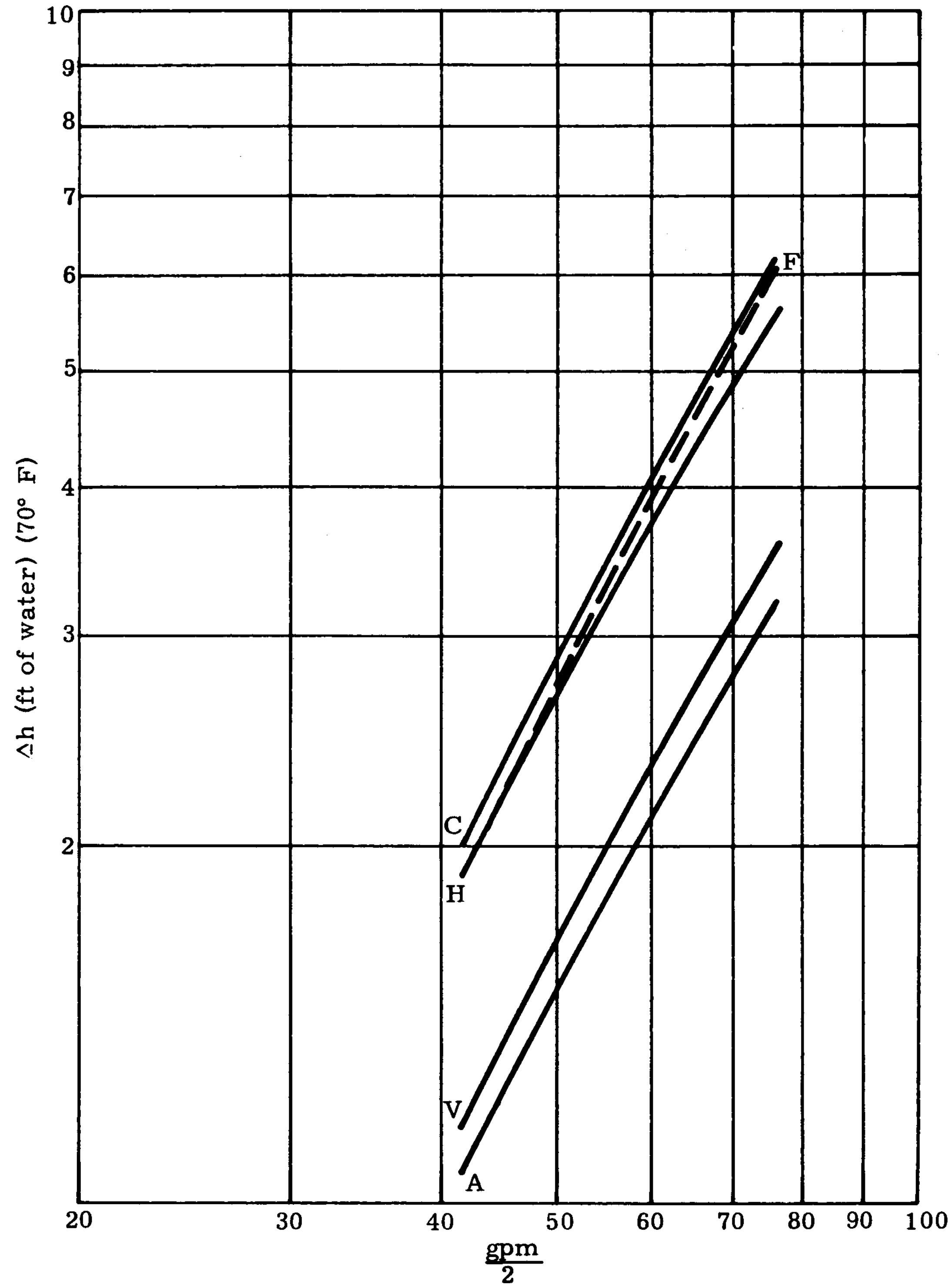
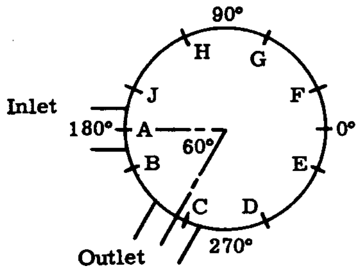


Fig. II-15. Pressure Drop Across Water Box Versus Total Flow for Different Vessel Positions 1/4-Scale (PM-1 Flow Test No. 2)



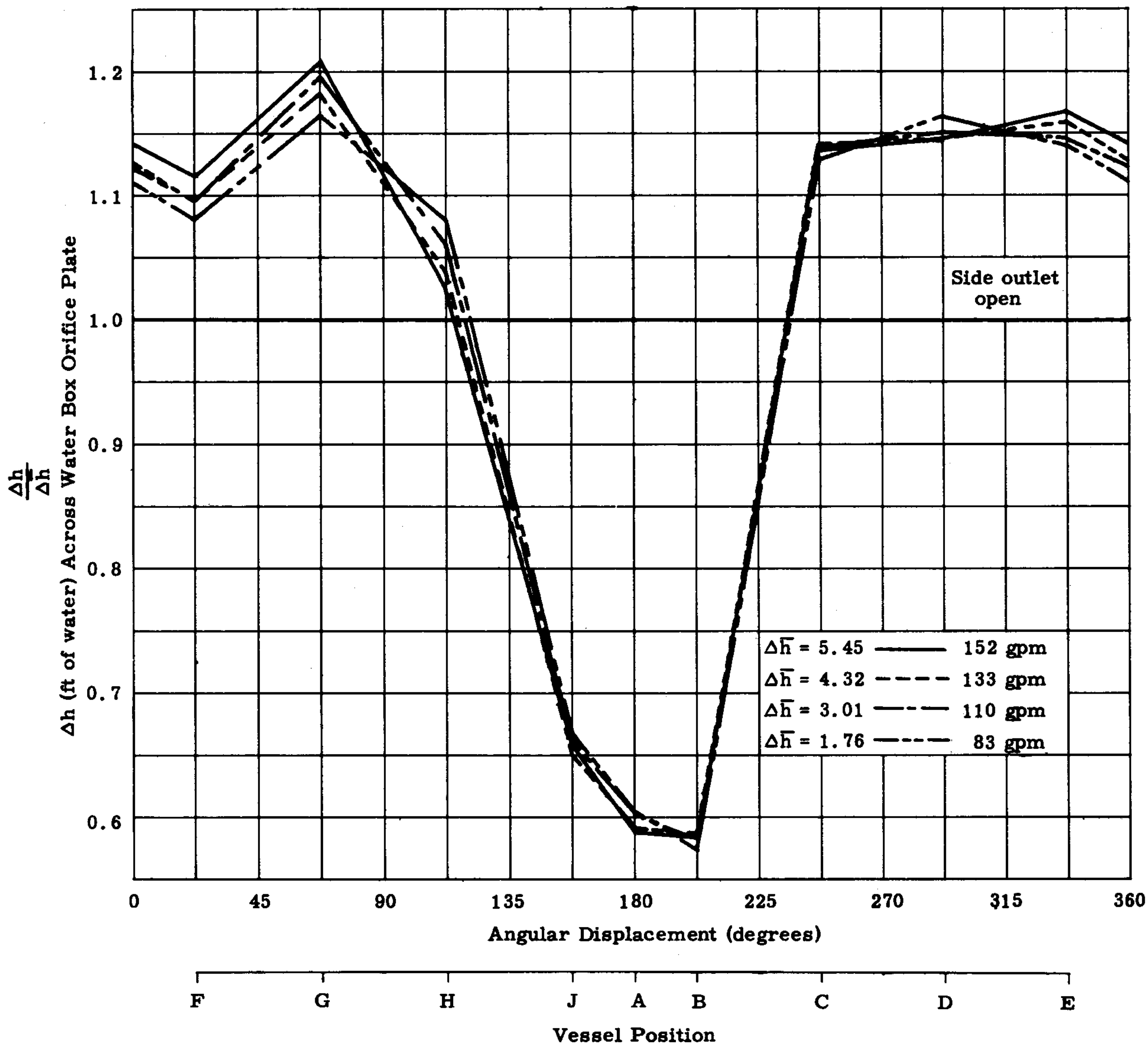


Fig. II-16. Normalized Head Loss at Various Flow Rates (1/4-Scale PM-1 Flow Test No. 2)

Figure II-17 serves to show the extent of pressure recovery in the water box. Plotted values are the pressure differentials between tap AP_2 and each other P_2 tap at a flow rate of 133 gpm. The inlet velocity head at this flow rate is

$$h = \frac{V^2}{2g}$$

$$V = \frac{Q}{A} = 24.2 \text{ ft/sec}$$

$$h = 9.1 \text{ ft.}$$

Figure II-17 indicates a maximum recovery of 2.5 feet. It is significant to note that the pressure is essentially constant from Position D to Position G and the cross-sectional area of the water box is also constant in this area. Slightly higher readings observed at Positions C, D and E as compared to those observed at H, G and F are to be expected since the elbow at the inlet directs the flow toward Positions C, D and E.

Figures II-18 and II-19 show plots of velocity near the exit of the flow annulus versus vessel position for 4 test flow rates with top outlet and side outlet, respectively. A comparison of these 2 graphs clearly indicates that the effect of the side outlet is not observed at this upstream position in that all measurements repeat within reasonable experimental deviations. Both the skewness about the inlet centerline and the flow distribution caused by the water box configuration with the 30-degree elbow inlet can be observed in these velocity measurements. That is, the form of the distribution observed through measurements of differential head across the water box orifice plate is also observed near the exit of the thermal shield annulus. For a model flow rate of 136 gpm, which nearly matches the inlet flow velocity of the prototype (less than 3% difference), an estimate of the maximum flow variation may be made.

$$\frac{V_E}{V_B} = \frac{5.55}{4.10} = 1.357$$

or, 36% more flow is passed at Position E as compared to B. The velocities plotted are average velocities along a radial profile at each position.

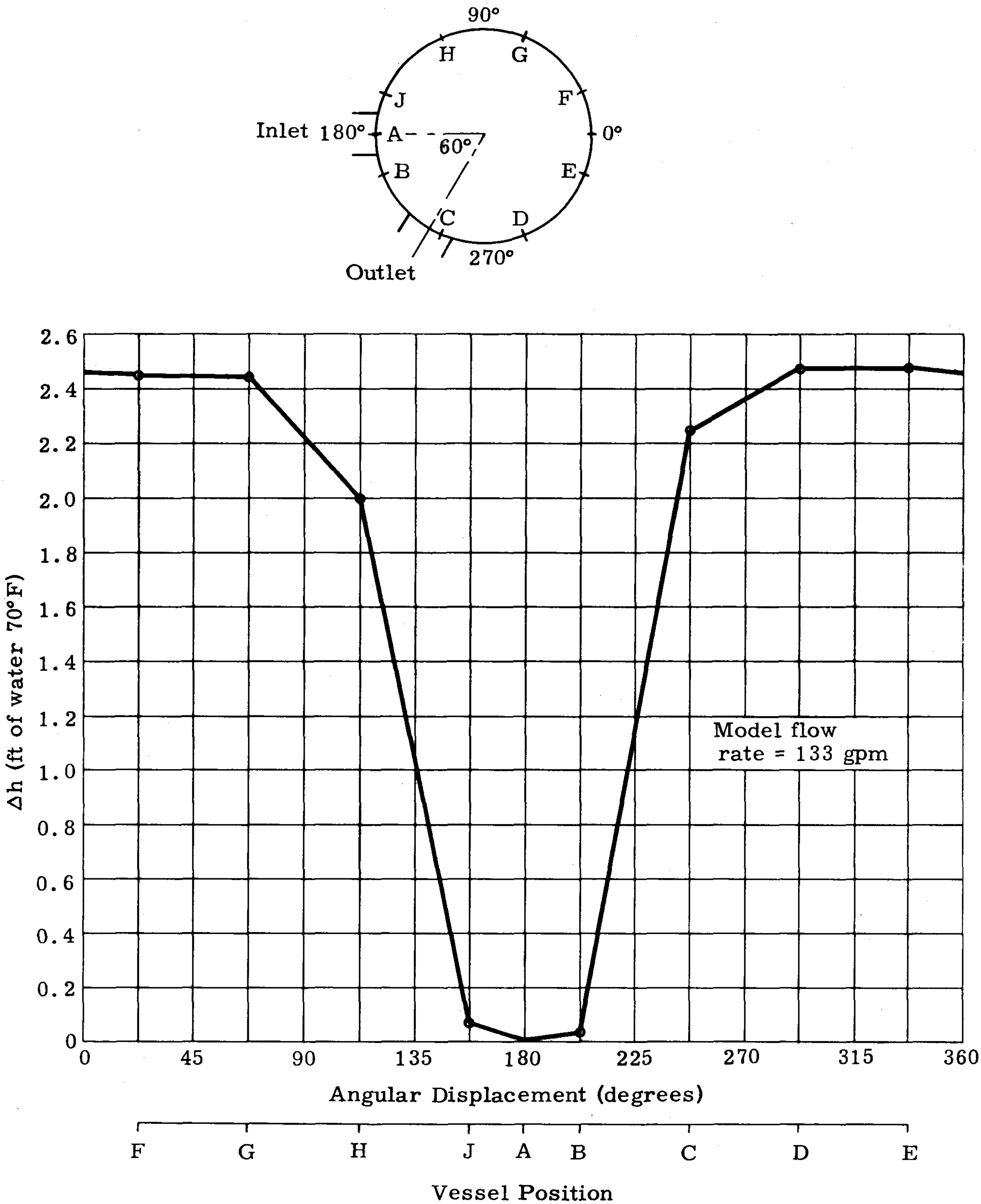
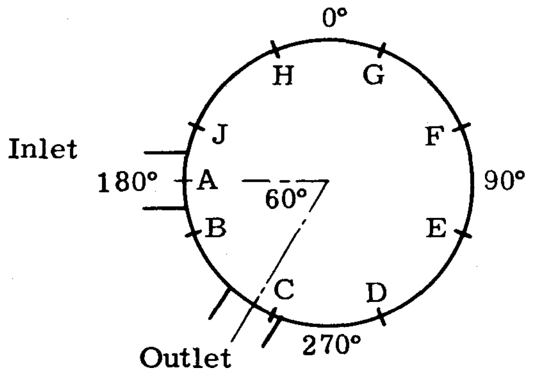


Fig. II-17. Pressure Recovery in Water Box Relative to Position "A" Versus Vessel Position (1/4-Scale PM-1 Flow Test No. 2)



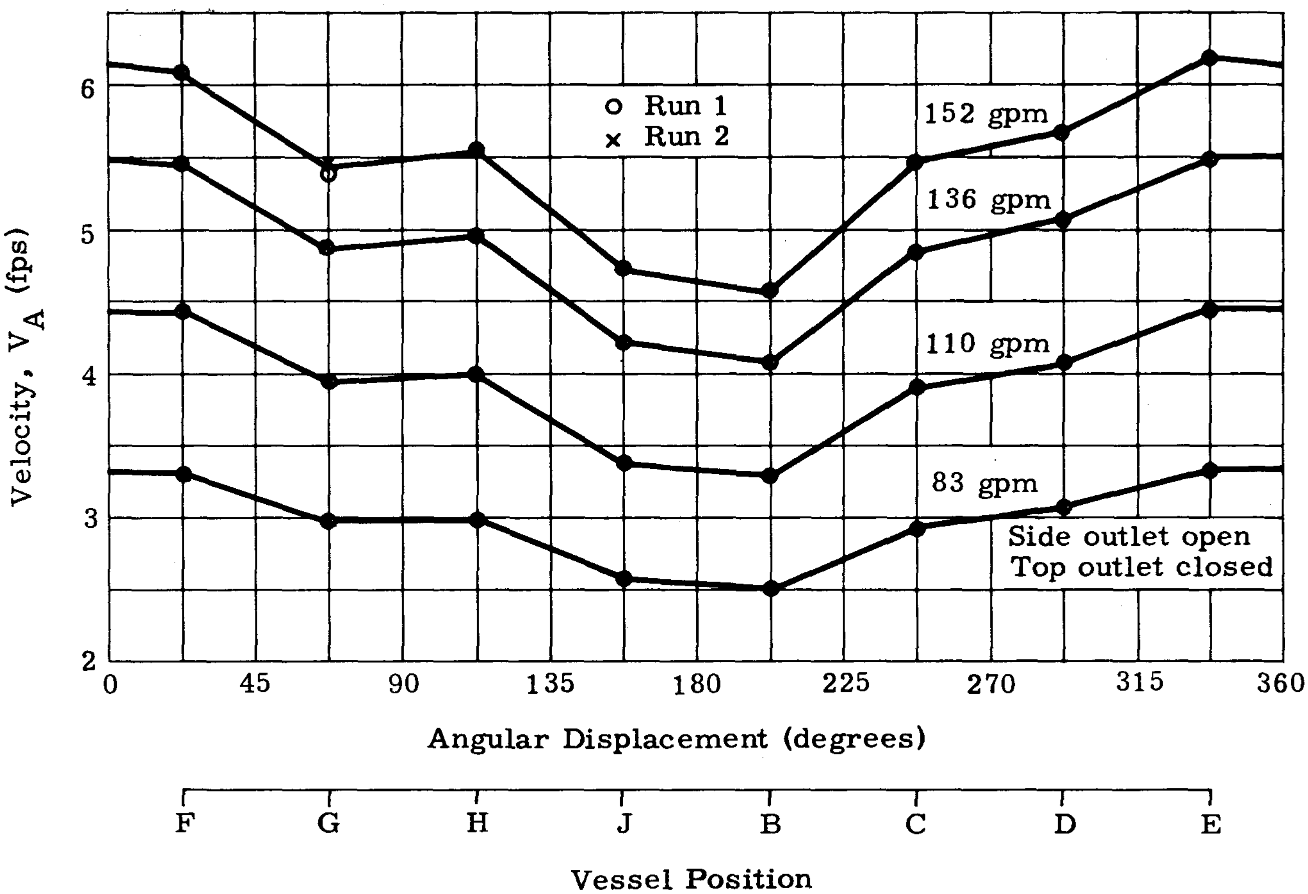


Fig. II-18. Velocities in Lower Portion ofthe Flow Annulus at Various Flow Rates (1/4-Scale PM-1 Flow Test No. 2)

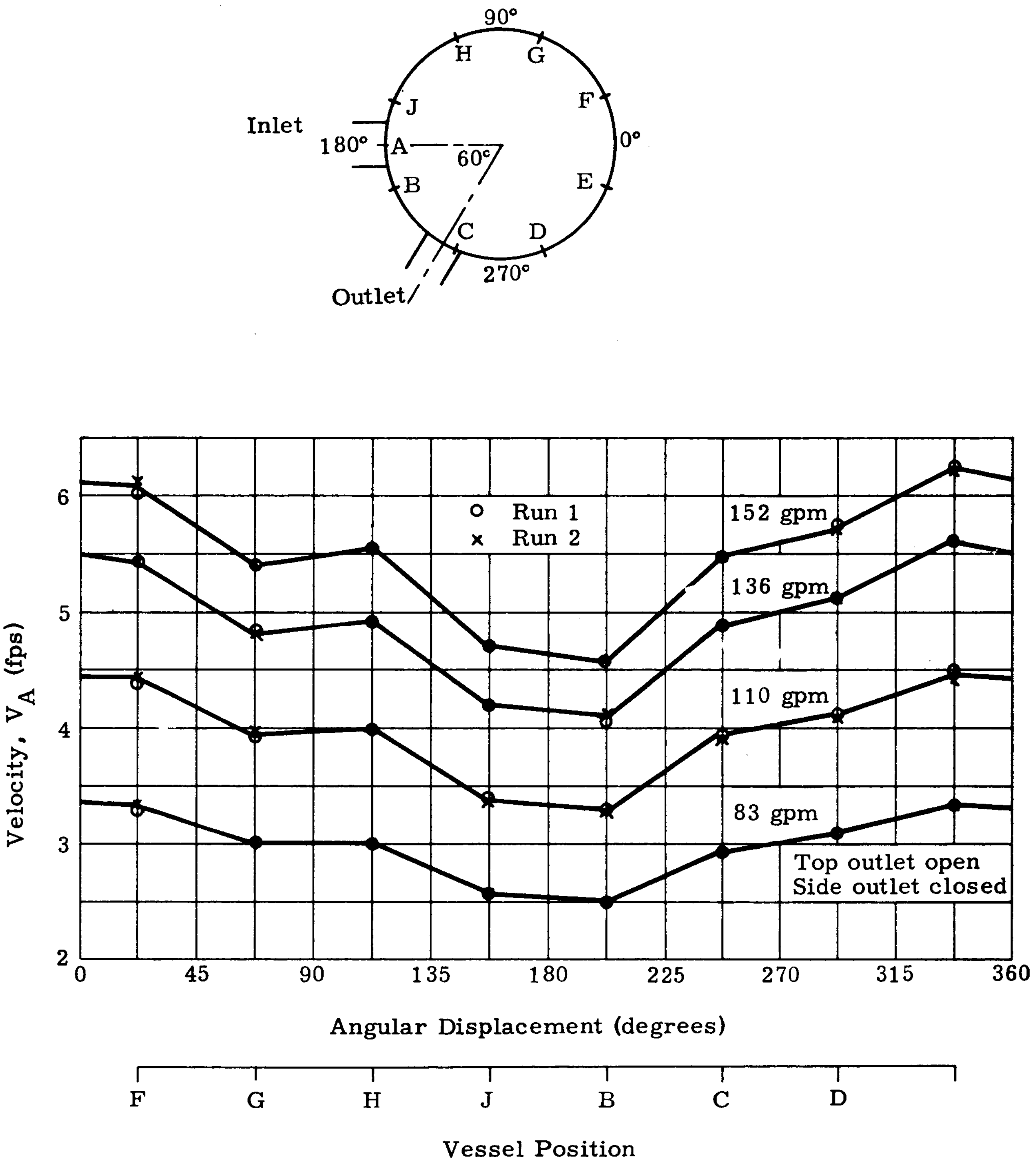


Fig. II-19. Velocities in Lower Portion of the Flow Annulus at Various Flow Rates (1/4-Scale PM-1 Flow Test No. 2)

Observations of dye and air injected into the inlet flow stream to show flow patterns indicated the following:

(1) The inclined top surface of the water box directs flow at an angle to impinge on the orifice plate.

(2) Interference of flow through each half of the water box occurs between Positions E and F.

(3) Swirling was not observed in the thermal shield flow annulus.

(4) Flow from the core region impinges upon the inside top surface of the upper shroud before passing through the shroud orifices.

(5) The fluid impinges on the inside wall of the upper vessel as it discharges from the upper shroud orifices. The impingement was quite evident during visual observation.

The following conclusions were derived from the results of 1/4-Scale Test No. 2:

(1) A pressure recovery observed in the water box downstream of the inlet caused a greater flow throughout the water box orifice plate in the section opposite the inlet.

(2) The minimum flow through the water box orifice plate occurred at vessel Position B which is 22-1/2% to the right side of the inlet. The effect of the inlet on pressure drop across the orifice plate was observed through approximately 70 degrees on each side of the inlet centerline.

(3) The variations in velocity in the flow annulus below the water box were nearly as great as those observed across the water box orifice plate. The flow variation $\left(\frac{Q_{max}}{Q_{min}} - 1\right)$ across the water box was 42%, and in the flow annulus, $\frac{V_{max}}{V_{min}} - 1 = 36\%$ at prototype velocity. This result indicates poor flow distribution as compared to Test No. 1 where a variation of 25% across the water box was reduced to 11% in the flow annulus.

(4) Any effect of the outlet nozzle location on flow distribution in the area instrumented in this test was not detectable.

(5) The overall pressure drop compared favorably with the predicted value for PM-1--being within 14%.

(6) Flow distribution in the inlet water box was relatively independent of velocity and Reynolds number in the range tested, which included the prototype design velocity. This gives further assurance that the results are applicable to the prototype.

(7) In general, the changes in the water box configuration produced more severe distortions of flow through the water box orifice plate and the flow annulus formed by the thermal shields.

2. One-Fourth-Scale Flow Model Test No. 3

The 1/4-Scale Flow Test No. 3 consisted of velocity and head loss measurements in the 1/4-scale flow model with a revised water box orifice plate. The revised plate contained 252 holes of 0.100-inch diameter in the same configuration as in 1/4-Scale Flow Test No. 2. In 1/4-Scale Flow Test No. 2, the orifice diameter was 0.139 inch. The object of this effort was to obtain data upon which final design orifice sizes could be based. The testing procedure followed was essentially the same as that used in 1/4-Scale Flow Test No. 2. The results of this test are presented in the attached tabulation charts and graphs.

Figure II-20 is a plot of head loss across the water box orifice plate versus angular displacement for various flow rates including 133 gpm (133 gpm gives prototype velocities in the model). The head loss is considerably higher than that measured in Test No. 2 (approximately 9 feet greater). Figure II-21 shows the ratio of the head loss at a particular location, Δh, to the mean head loss, $\overline{\Delta h}$, for a model flow rate of 133 gpm. The square root of $\frac{\Delta h}{\overline{\Delta h}}$ is plotted to indicate flow variation. The maximum deviation in the head loss from the mean value (13.32 feet H_2O) was -15.9%. This is less than the value obtained in Test No. 2 (-38.7%).

Figure II-22 shows a plot of velocity near the exit of the flow annulus versus vessel position for 4 test flow rates. The difference observed in Runs 1 and 2 occurred during testing and was not the

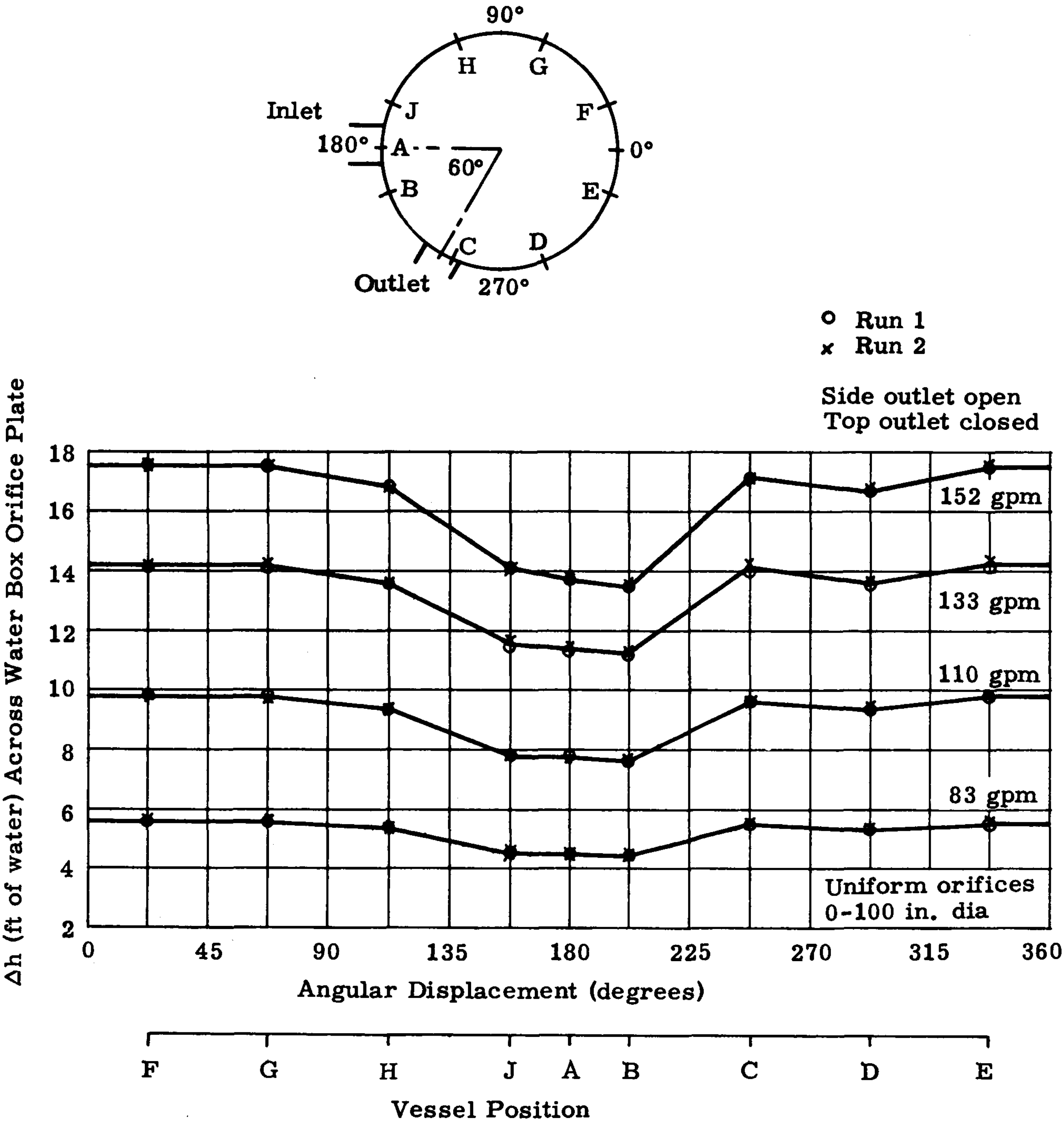


Fig. II-20. Head Loss Across Water Box Orifice Plate at Various Flow Rates (1/4-Scale PM-1 Flow Test No. 3)

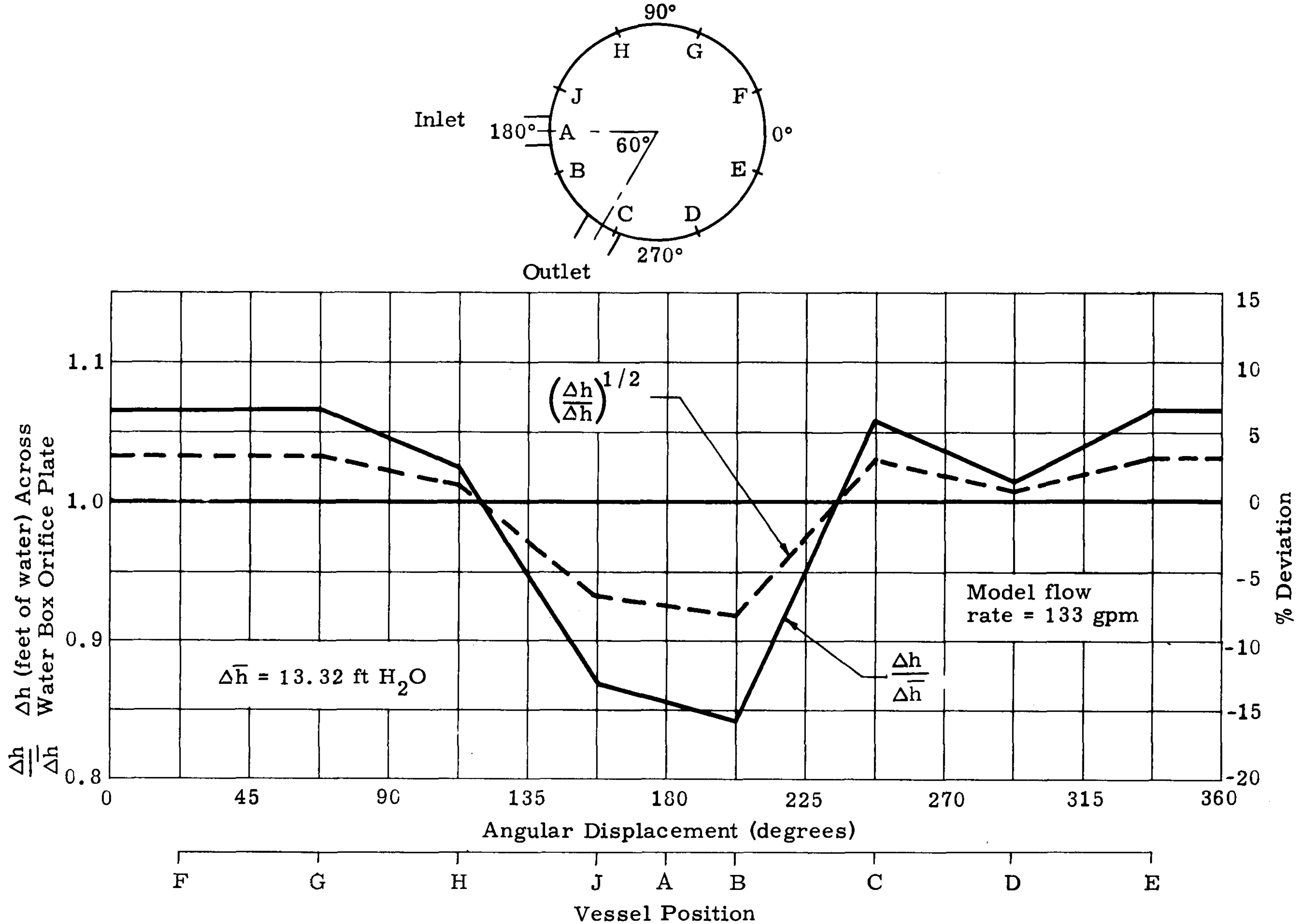
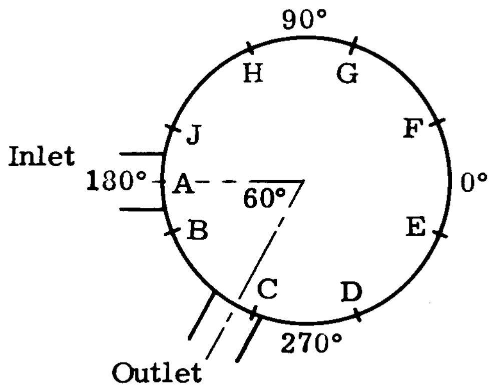


Fig. II-21. Ratio of Head Loss at a Particular Location to Mean Head Loss (1/4-Scale PM-1 Flow Test No. 3)



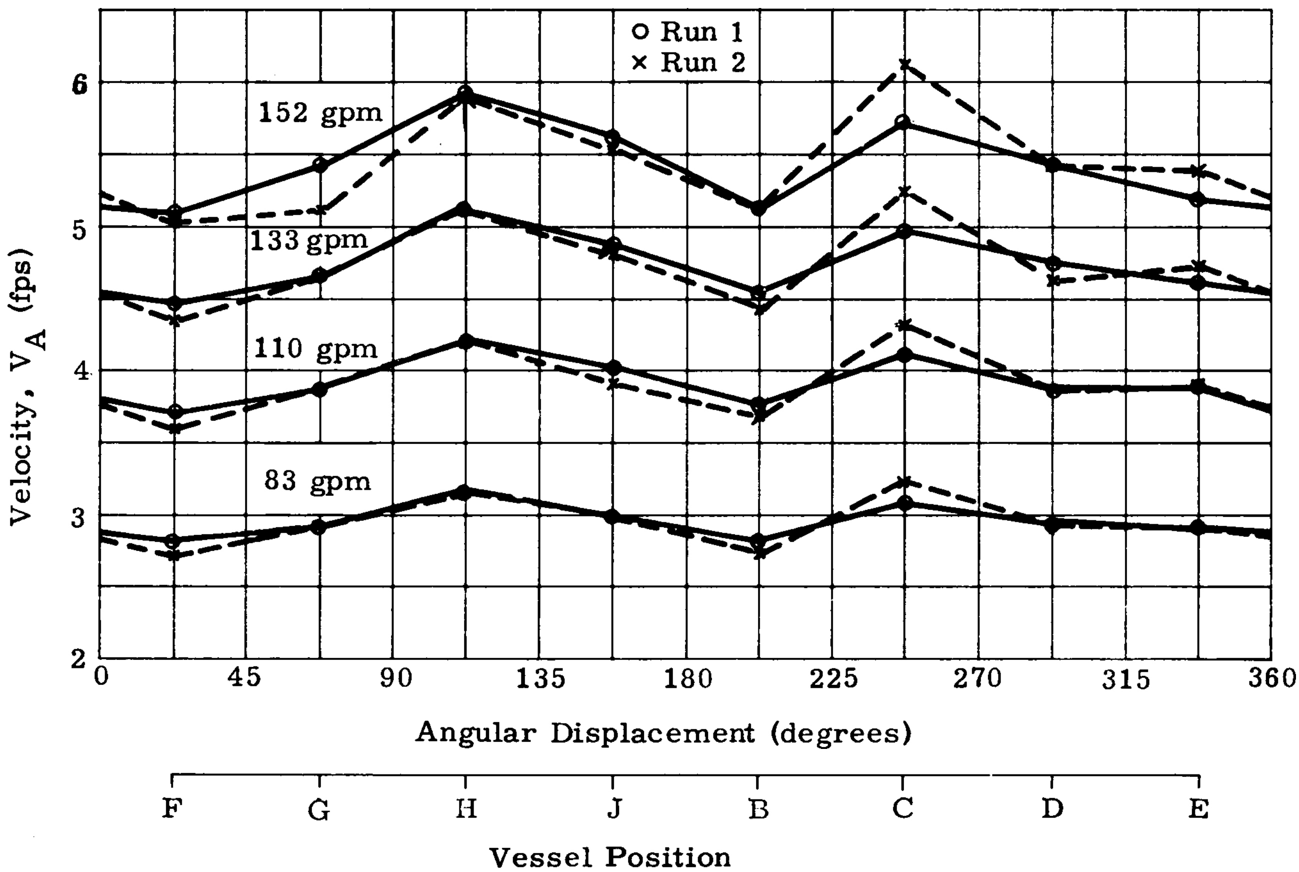


Fig. II-22. Velocities in Lower Position of the Flow Annulus at Various Flow Rates (1/4-Scale PM-1 Flow Test No. 3) result of the operators' making changes in flow. An unstable flow apparently produces this effect. Actual observations of this instability were the following:

(1) The desired loop flow rate was set and manometer readings were observed to be steady.

(2) After approximately 15 minutes, some manometer deflections changed and once again became steady at a new value. As one deflection increased, another would decrease although the manometer systems were not externally interconnected.

(3) A second shift in readings was observed after another 15 to 30 minutes elapsed.

(4) Loop flow rate could be changed without affecting a particular distribution.

Figure II-23 shows a normalized curve (ratio of velocity to mean velocity) for 133 gpm. A comparison between this curve and Fig. II-24 (similar values for Test No. 2) indicates a reduction in flow variation.

At 133 gpm, which provides velocities in the model similar to those of the prototype at 2125 gpm, the vessel head loss is 32.3 feet. This is an increase of 8.1 feet over that observed in Test No. 2.

Table II-9 shows a comparison of results obtained in Test No. 2 and Test No. 3. Test No. 2 was performed with an orifice diameter of 0.139 inch in the water box orifice plate. Figure II-20 shows the relationship between average head loss across the water box orifice plate versus orifice diameter.

3. Reduction and Analysis of One-Fourth-Scale Flow Model Data

During this quarter, the testing of the 1/4-scale model was completed and the data reduced and analyzed. The tests were timed to be performed before the completion of final design so that changes in the vessel design could be made if the results of the tests indicated that any were necessary. The tests were run in 3 separate parts. Test 1 included testing with the inlet water box incorporated in the preliminary design of the PM-1 reactor. The water box orifice plate contained 252 holes, 0.125 inch in diameter. Tests 2 and 3 were performed using the final inlet water box design. However, orifice plate diameters were modified to 0.139 inch for test 2 and 0.100 inch for test 3.

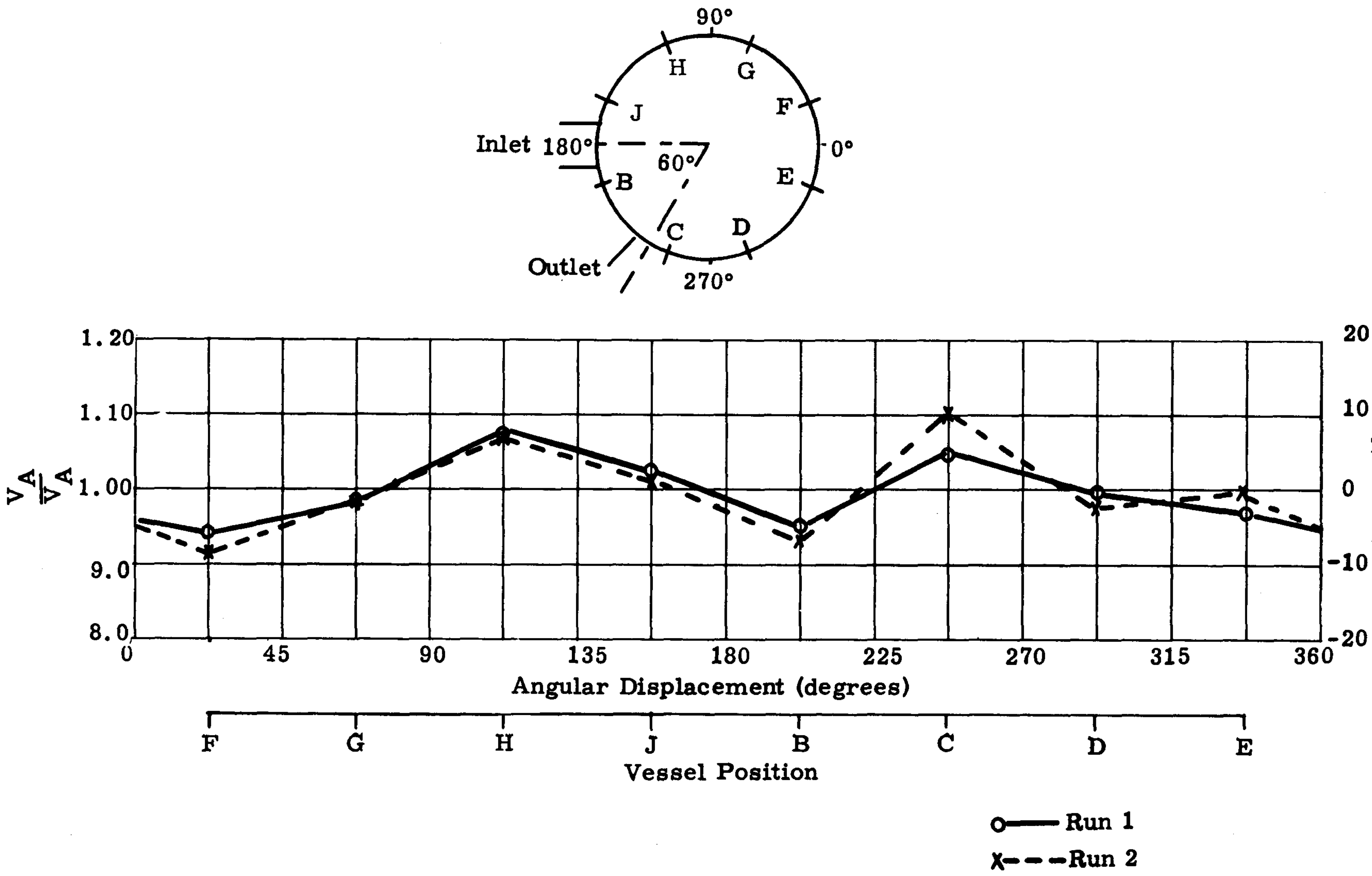


Fig. II-23. Velocities in Lower Portion of the Annulus--Normalized Curve (1/4-Scale PM-1 Flow Test No. 3)

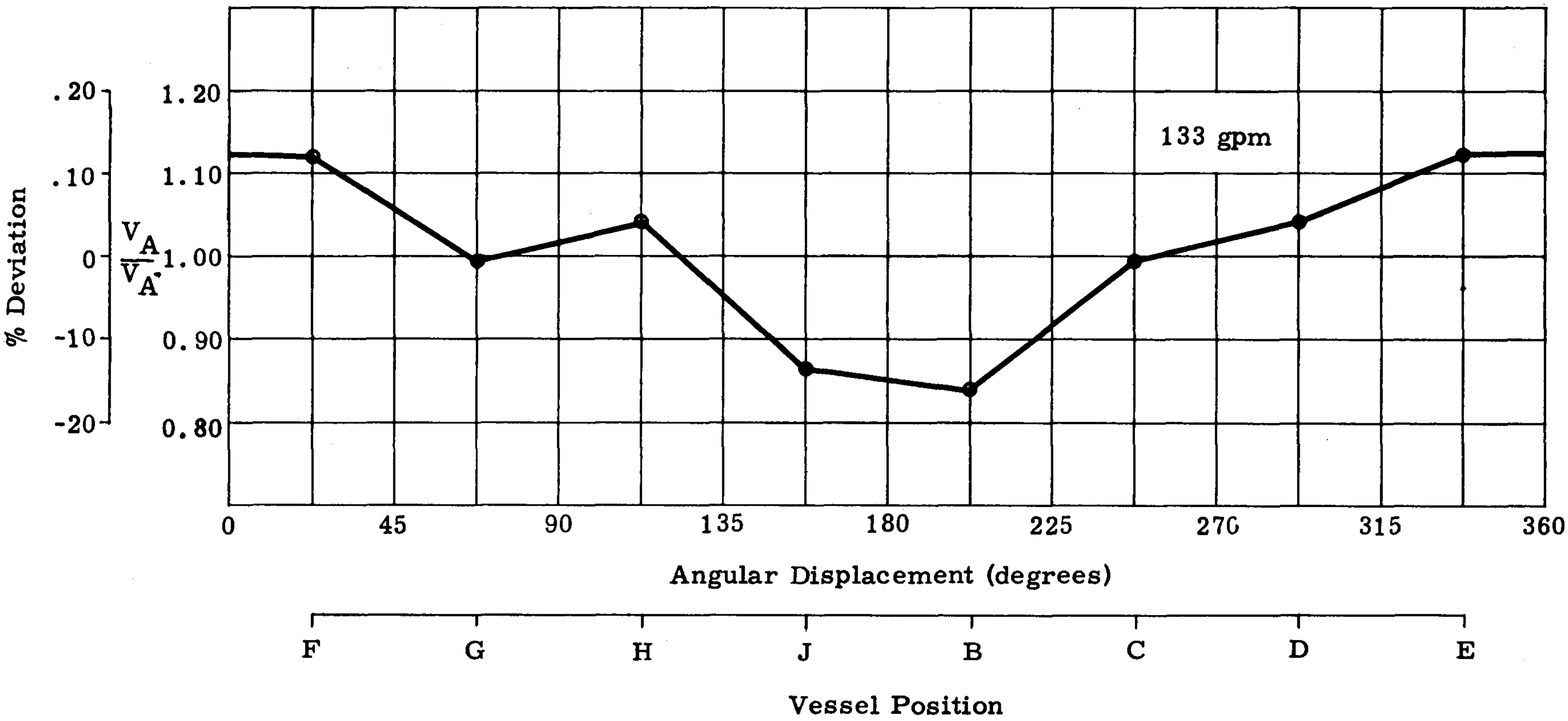


Fig. II-24. Velocities in Lower Portion of the Flow Annulus-- Normalized Curve (1/4-Scale PM-1 Flow Test No. 2 for Comparison with Results of Test No. 3)

TABLE II-9

Comparison of Results of Tests Nos. 2 and 3

Test No. 2, 0.139-inch orifice diameter

Test No. 3, 0.100-inch orifice diameter

Loop Flow Rate 133 gpm	Head Loss Across Water Box Orifice Plate		Pressure Recovery in the Water Box		Velocities in the Lower Flow Annulus	
	Test No. 2	Test No. 3	Test No. 2	Test No. 3	Test No. 2	Test No. 3
Maximum value	5.11 ft	14.2 ft	2.47 ft	2.52 ft	5.37 ft/sec	5.25 ft/sec
Minimum value	2.53 ft	11.2 ft	0.00 ft	-0.11 ft	3.99 ft/sec	4.35 ft/sec
Average value	4.32 ft	13.32 ft	--	--	4.76 ft/sec	4.76 ft/sec
Maximum positive deviation from mean	18.5%	6.6%	--	--	12.5%	10.1%
Maximum negative deviation from mean	38.7%	15.9%	--	--	16.5%	8.2%
Average deviation from mean	18.7%	7.4%	--	--	7.9%	4.5%

Analysis of the data obtained from these tests yielded the following conclusions:

(1) Flow down through the water box orifice plate was more evenly distributed around the periphery of the vessel in Test No. 1 than in Test No. 2 (see MND-M-1814).

(2) Decreasing the size of the orifice plate holes from the 0.139-inch diameter of Test 2 to the 0.100-inch diameter of Test No. 3 considerably improved the flow distribution; however, the head loss across the orifice plate at the PM-1 flow was prohibitive.

(3) The decrease in orifice hole diameter for Test No. 3 resulted in a more symmetrical flow pattern through the orifice plate with respect to the reactor inlet.

(4) For each water box, a characteristic curve of head loss versus local velocity through the orifice plate could be plotted. All the data fell on or very near the corresponding curve, independent of orifice plate hole diameter, loop flow rate or location on the vessel periphery for each test. Thus, it is concluded that water box configuration has the only influencing effect on the relationship between these two parameters. Head loss varied with local velocity to the 1.82 power in Test 1 and to the 1/90 power in Tests 2 and 3.

(5) Flow symmetry near the bottom of the thermal shield annulus with respect to points on the vessel periphery equal angular distances from the inlet was good.

(6) Although the flow distribution across the water box orifice plate was not uniform, the effects of the thermal shield annulus and entrance to the core inlet plenum worked to substantially improve this condition at the plenum entrance. Analysis of the test data has shown that the pressure distribution at the plenum entrance is relatively even. Maximum variation from the mean pressure in the core inlet plenum was 0.3 foot in Test 1 and 0.15 foot in Tests 2 and 3.

4. Bundle-Orifice Test Program

The major objective of the experimental effort is to determine by laboratory investigation the fuel tube inlet orifice diameter which regulates the flow distribution so that 52% of the total flow passes through the outside of the tubes. Design of the test bundles is shown in Fig. II-25. The

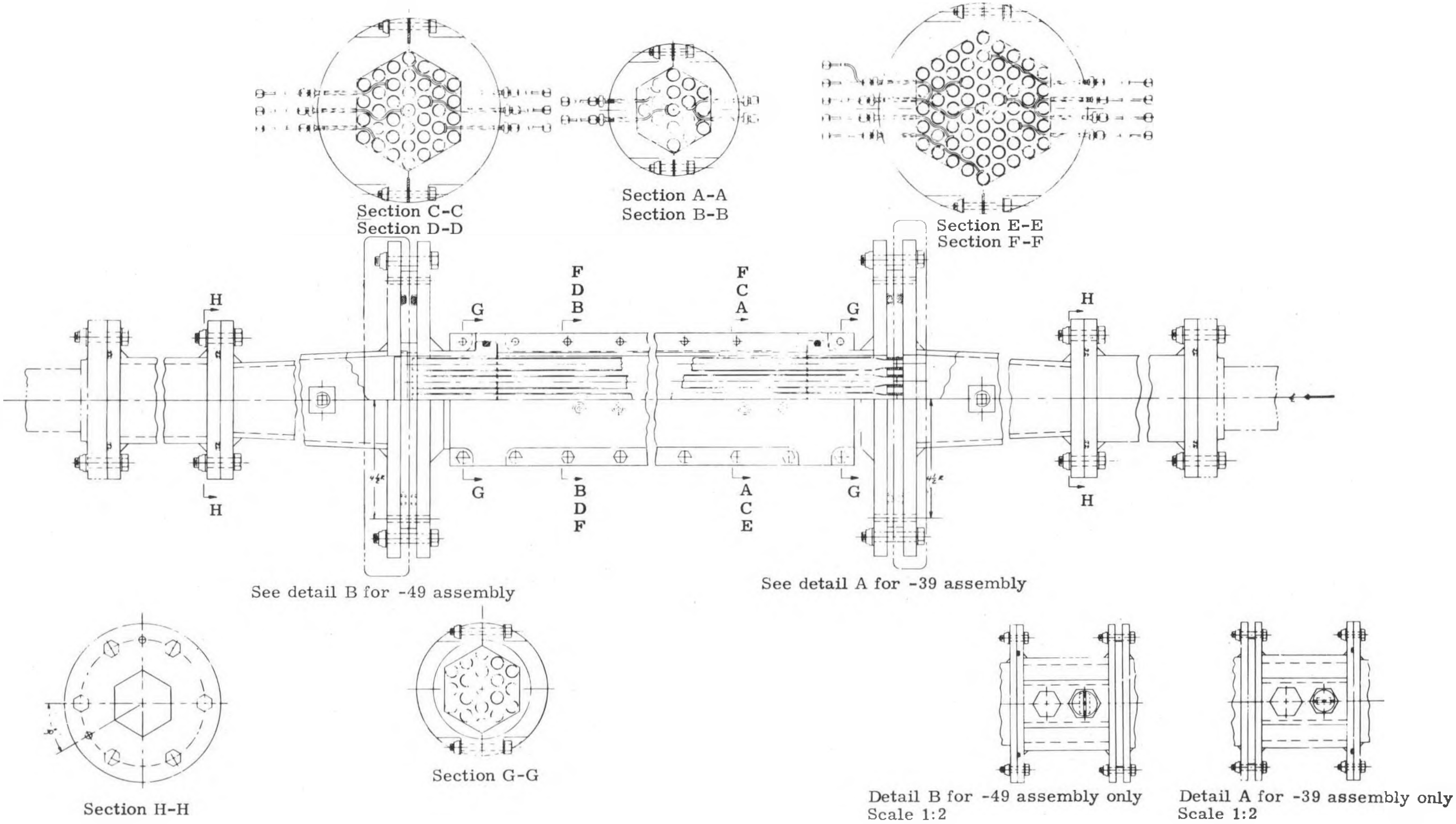


Fig. II-25. Bundle Assembly PM-1 Flow Test influence of various numbers of solid poison rods on this distribution will be determined. Also, the dependence of this distribution on Reynolds number through a range limited by the capabilities of the testing facility will be determined.

The test loop consists of a 200-gpm centrifugal pump discharging into 2-inch diameter piping. The test section is inserted in a vertical pipe run with upward flow. Total loop flow is measured in the return vertical pipe using an orifice meter. This pipe discharges into a 50-gallon suction tank. A pump bypass is provided. Control of flow is effected using a combination setting of throttle and bypass valves.

A test section consists of a group of tubes held between 2 orifice plates and surrounded by a hexagonal housing. Three test sections are to be used in the program containing 19, 37 and 61 tubes, respectively. Inlet and outlet transition pipes are provided for each test section to connect to the 2-inch diameter loop pipe.

Loop instrumentation consists of an orifice meter to measure total loop flow and a thermometer to measure loop water temperature.

Test section instrumentation consists primarily of static pressure taps located along the length of selected tubes for the measurement of the pressure drops associated with flow in the tubes. Each instrumented tube is one of a group of tubes which are similar with respect to location in the bundle. Assuming absence of nonsymmetrical flow distortions, each similar tube will pass the same flow. The flow rate measured in an instrumented tube would determine the flow rate in a group. Since every tube in each bundle is part of a group in which there is one instrumented tube, the total flow through all tubes may be determined. A measurement of total loop flow makes it possible to determine the total flow between the tubes by subtracting the total flow through the tubes. Thus, the flow ratio of total flow through the tubes to total flow between the tubes is determined. Instrumentation is also provided for the measurement of the overall head loss across the bundle.

Figures II-26 and II-27 show a typical set of bundle test components and an installed unit, respectively. The general method of testing will be as follows.

At predetermined loop flow rates, selected to produce desired flow velocities in the test section, differential pressures measured in instrumented tubes will be recorded. The overall test section pressure drop will also be recorded. These measurements will be made in each of the three test sections. Data taken at the same average flow velocity will be compared for each test section to determine the effect of the housing wall on flow ratio. A flow ratio applicable to the PM-1 core will be determined from these data. This ratio will be compared to the desired ratio and an adjustment of the inlet orifice diameter will be made to match these ratios as necessary.

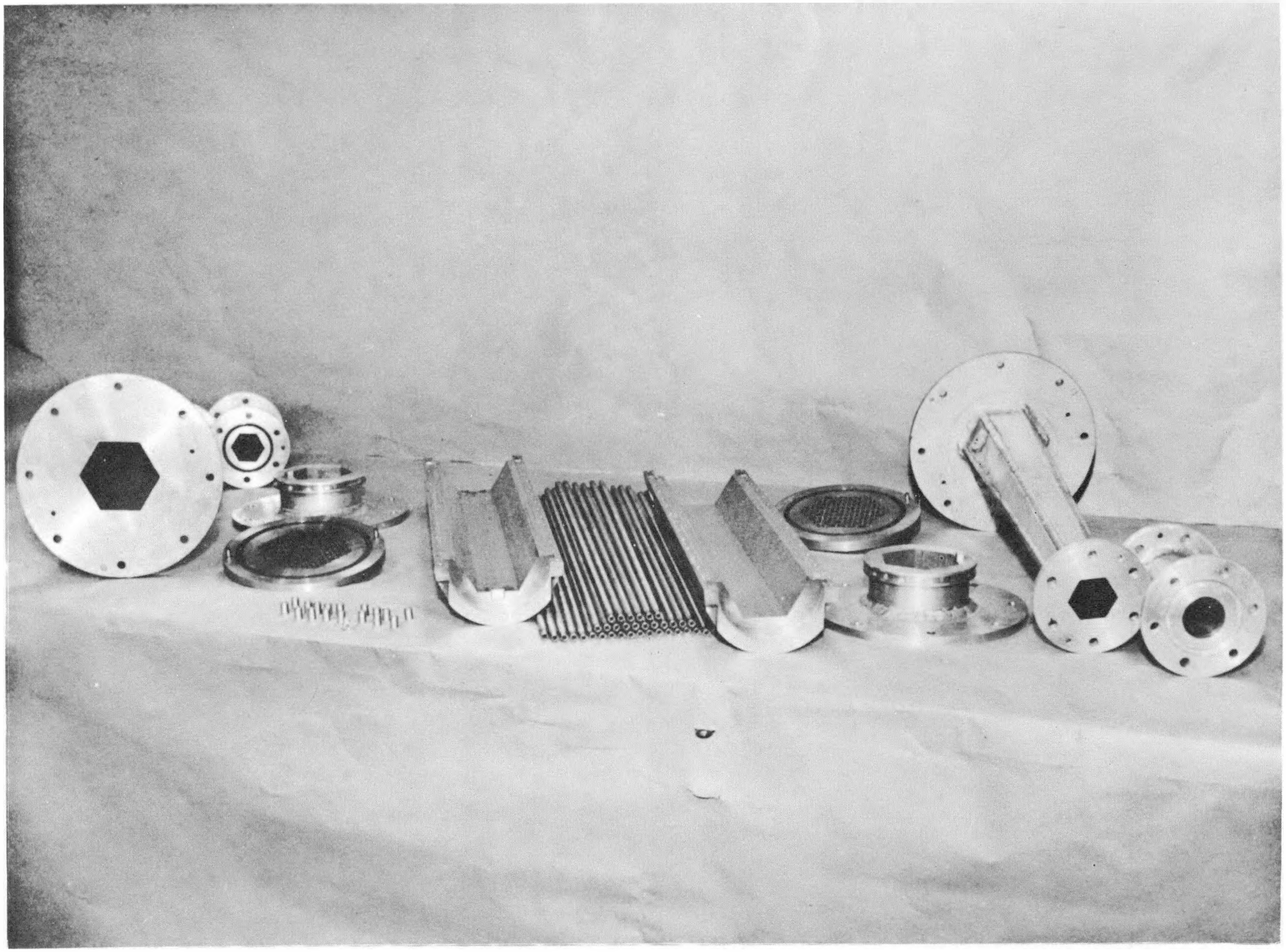


Fig. II-26. Components for 37-Tube Bundle Assembly (PM-1 Bundle Orifice Test)

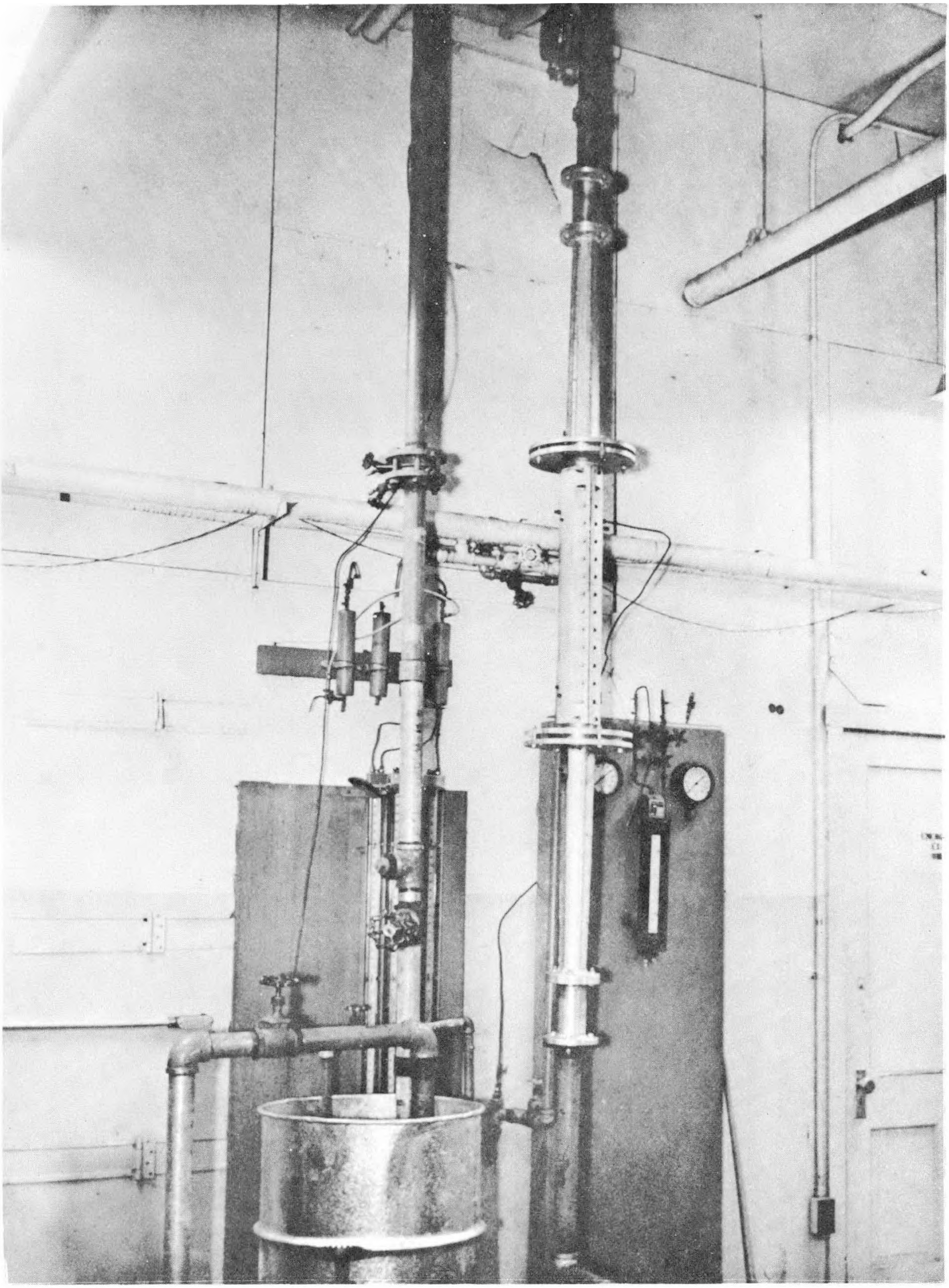


Fig. II-27. Installed 19-Tube Bundle Assembly (PM-1 Bundle Orifice Test)

The results of flow ratio determinations will be presented as a tabulation of flow ratio for each flow condition.

A test will be run to determine the effect burnable poison rods have on the flow ratio in the core. Most likely, the 61-tube bundle will be rerun with sufficient rods to duplicate geometry and the ratio of burnable poison rods to tube locations established in the core. The amount of data required in this test (number of flow rates, conditions, etc.,) will be determined from results of the tests previously described.

5. Full-Scale Reactor Flow Model

This portion of Subtask 2.3 describes the technical work being done under what is administratively Task 6 (dummy core). The reporting is arranged in this fashion to preserve continuity.

Design of the full-scale reactor flow model pressure vessel and head were completed. These components are being fabricated. Design of test components has been made so that all core components are duplicates of those in the PM-1 core. Thus, actual PM-1 fabrication and assembly procedures as well as special tooling will be used for the full-scale flow test.

The required instrumentation of the above test has been established as follows:

a. Thermal shields

Two static pressure taps will be located 20 inches apart at 4 equally spaced locations around the periphery of the reactor vessel.

Sixteen retractable pitot-static tubes will be located below, but as close as practical, to the thermal shields to measure velocity at various points on the periphery of the reactor vessel.

b. Upper skirt

Four pitot-static tubes will be located so that flow through the upper skirt orifices can be determined at four equally spaced locations around the periphery of the skirt.

c. Dome-shaped head

One pitot-static tube will be so placed in the center bundle hold-down tube that flow rate which cools the vessel head can be determined.

d. Control rods

Two static taps will be located 20 inches apart on one blade of a selected control rod to measure static pressure drop adjacent to the blade.

An impact tube will be located in one blade and a tube with a static tap will be located in the other blade of the two remaining blades of the control rod. These will be interconnected so that flow in a control rod channel can be determined for different rod positions.

e. Head on upper skirt

One impact tube will be passed through the head on the skirt to measure total water pressure on the head. This will be interconnected with a tube having static taps located above the head (fairly stagnant zone) so that pressure difference across the head may be determined.

f. Core

Inside tubes. Every tube in the core through which water flows will be grouped into one of 25 groups based on similar anticipated flow characteristics. One selected tube in each group will be instrumented with 2 pressure taps, 26 inches apart, so that static pressure drop over this length can be measured. The whole core will be capable of rotation, thereby covering all tube locations with a minimum amount of instrumentation.

Outside tubes. Flow areas outside the tubes will be grouped in the same manner as flow areas inside the tubes. Static pressure taps will be located in three adjacent tubes at different heights. These taps will be interconnected and properly valved so that static pressure drop along both the length of insertion of a partially inserted control rod and the remaining length of the control rod channel can be determined for the instrumented flow area.

One tube in the core close to a control rod channel will be capable of rotation and contain two static pressure taps. This tube will be used for determination of the direction of cross flow to a control rod channel. It will extend through the vessel head. Static taps will be located in selected dummy elements near the outskirts of the core to determine flow in that area. Two taps, 20 inches apart, will be used.

D. SUBTASK 2.4--HEAT TRANSFER TESTS

J. J. Jicha, M. P. Norin, S. Frank, C. Eicheldinger

The objectives of this subtask are to obtain experimental data to support refined local boiling thermal and hydraulic design of the PM-1 core. The program is designed around 3 test sections, which are described below:

(1) STTS-3, Single-Tube Test Section--A single-tube design with flow inside the tube only. This test section is instrumented to obtain local boiling pressure drop and heat transfer data inside of tubes.

(2) STTS-4, Single-Tube Test Section--A single tube contained within a housing so as to provide coolant flow inside and outside of the tube. This unit will be used for local boiling burnout studies.

(3) SETCH-2, Seven-Tube Test Section--This unit will have coolant flow outside the tubes only and will be instrumented to obtain pressure drop and heat transfer data outside of tubes.

Accomplishments during this quarter were:

(1) STTS-3: Loop modification, test section fabrication and instrumentation installation were completed. The test section was installed and operated.

(2) STTS-4: Test section fabrication and test program design were completed.

(3) SETCH-2: Test section fabrication and test program design were completed.

The anticipated accomplishments for the next quarter are:

(1) STTS-3: Testing will be completed.

(2) STTS-3: Thermal and hydraulic data analysis will be initiated.

(3) SETCH-2: Loop modification and installation will be completed.

(4) SETCH-2: Testing will be completed.

1. STTS-3 Single-Tube Test Program

The work accomplished on this program can be subdivided into the following phases: loop modification, test section fabrication, test section installation, instrumentation installation and testing.

Loop modifications were confined to two areas: the in-line preheater and loop piping immediately upstream and downstream of the test section. A saturable reactor proportional control unit was installed to afford better control of the in-line preheaters and, therefore, better control of the test section inlet temperature. Subsequent checkout of this unit was successful. However, steps taken to obtain better flow distribution at low flow rates by means of baffles in the line heater, were not successful and the line heater containment vessel failed due to overheating. This containment vessel and its connecting pipes were replaced with a configuration design to eliminate this difficulty. The second area in which the loop was modified was the substitution of stainless steel tubing for 3-inch pipe. The necessary tubing connections were made for both the horizontal and vertical test section installations during the report period.

Test section fabrication was completed during this report period. The test section was hydrostatically tested to 2100 psi at 200° F and no leaks were observed. Electrical resistances of the test section tube and collar junctions were checked and approximately 30 kw of electrical power applied with no adverse effects. The burnout detector was calibrated and checked out for the STTS-3 test element. Figure II-28 shows the STTS-3 element mounted for the isothermal horizontal flow calibration tests. The 12 pressure taps and the manometer valving assembly are shown in Fig. II-28. The 6 high pressure inclined manometers employed to measure 5 incremental and 1 overall pressure drop were wall mounted as shown in Fig. II-29 and connected to the manometer valving assembly.

Fifteen wall thermocouples and two bulk coolant thermocouples were calibrated and installed on the test section as shown in Fig. II-30. A close-up of the wall thermocouple installation (also the pressure taps) and the components involved is shown in Fig. II-31. Mica tape was used as the electrical insulation between the thermocouple and current-carrying test section. The thermocouple junction is protected from the spring steel retaining clip by glass fiber tape.

Figure II-32 shows the test section completely insulated and ready for the isothermal horizontal flow tests. These flow tests were completed during the report period with the realization of the following information:

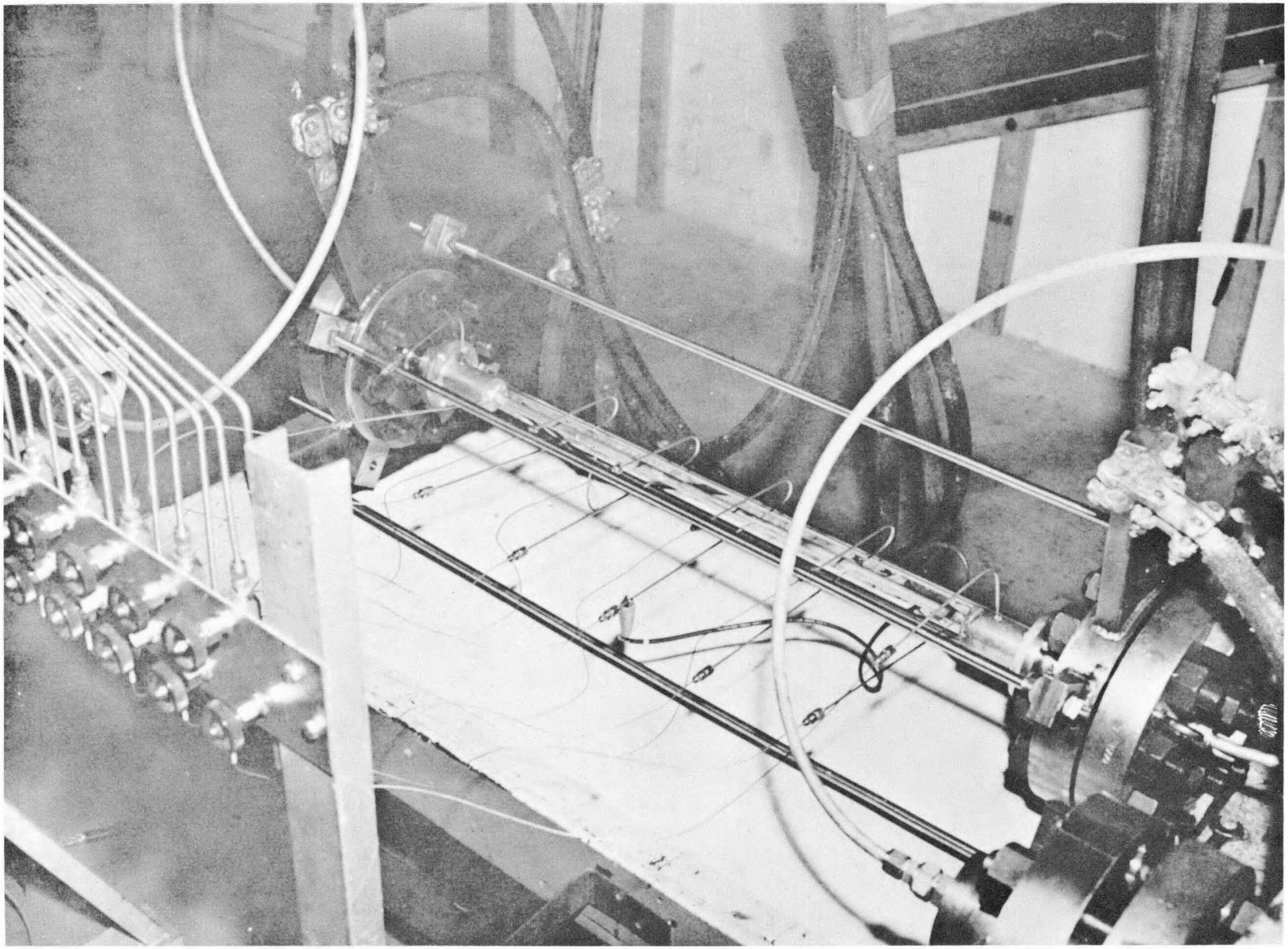


Fig. II-28. STTS-3 Pressure Tap and Manometer Valve Assembly

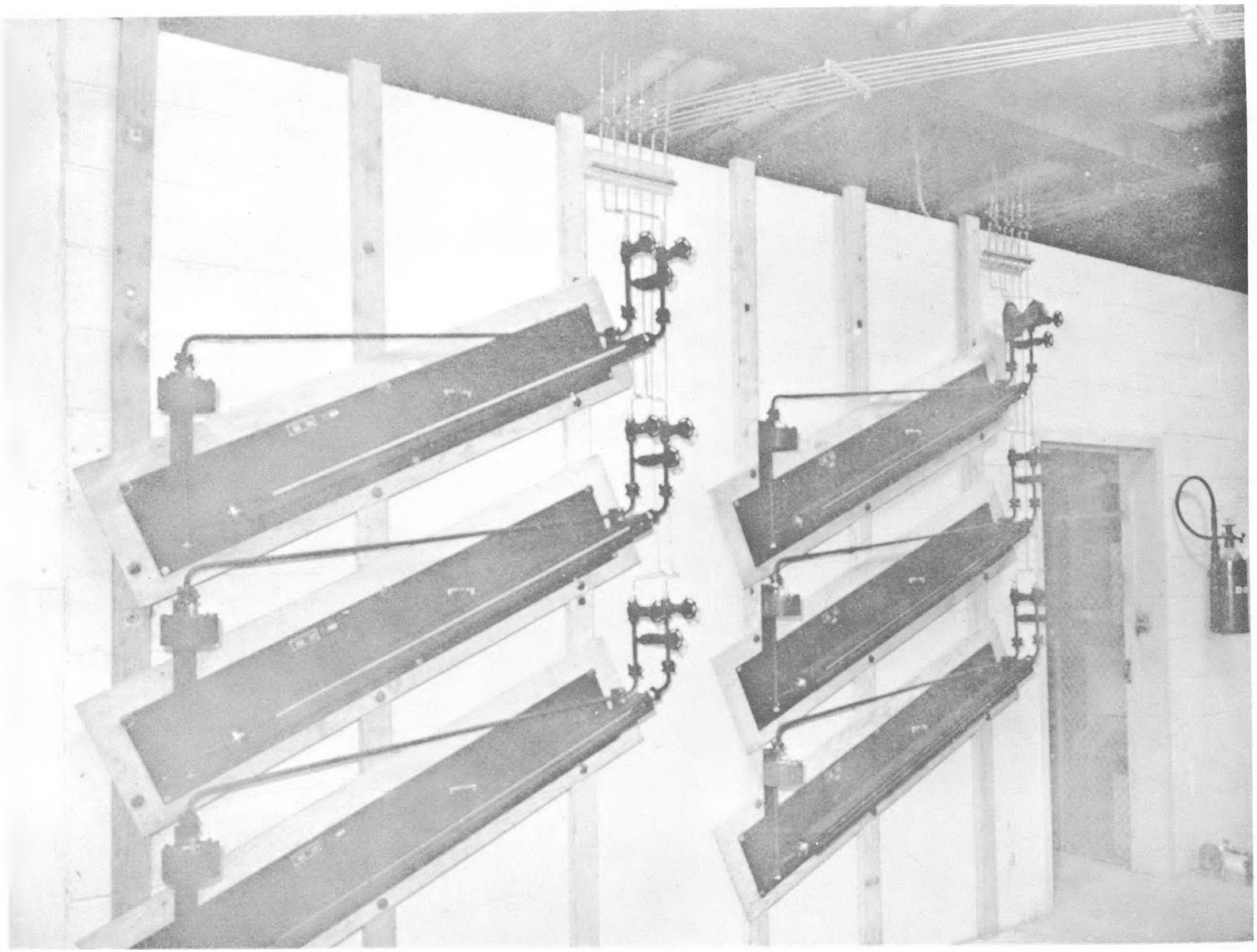


Fig. II-29. Six High Pressure Inclined Well-Type Manometers Employed in the STTS-3 Program

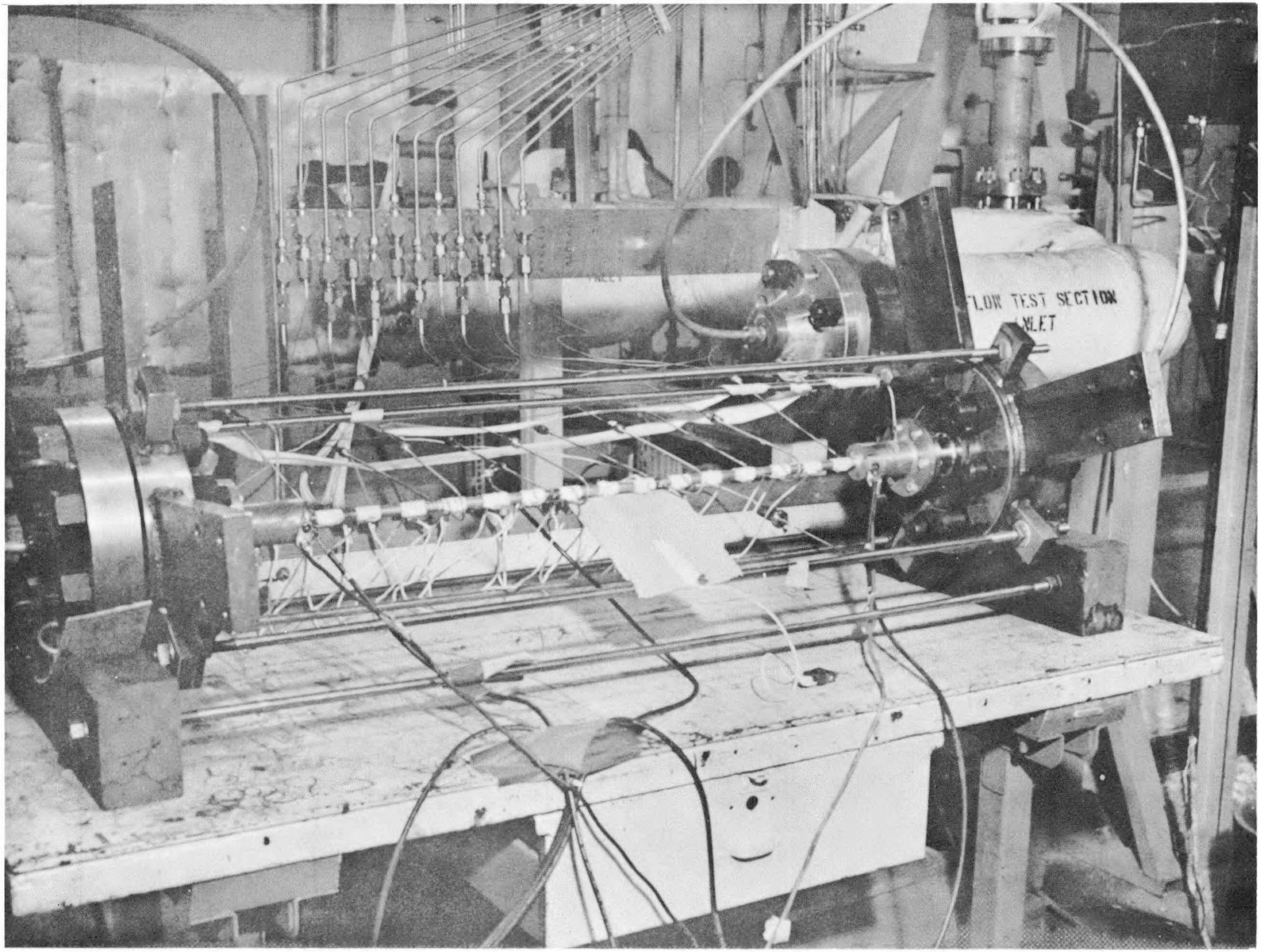


Fig. II-30. STTS-3 Wall Thermocouples Installed

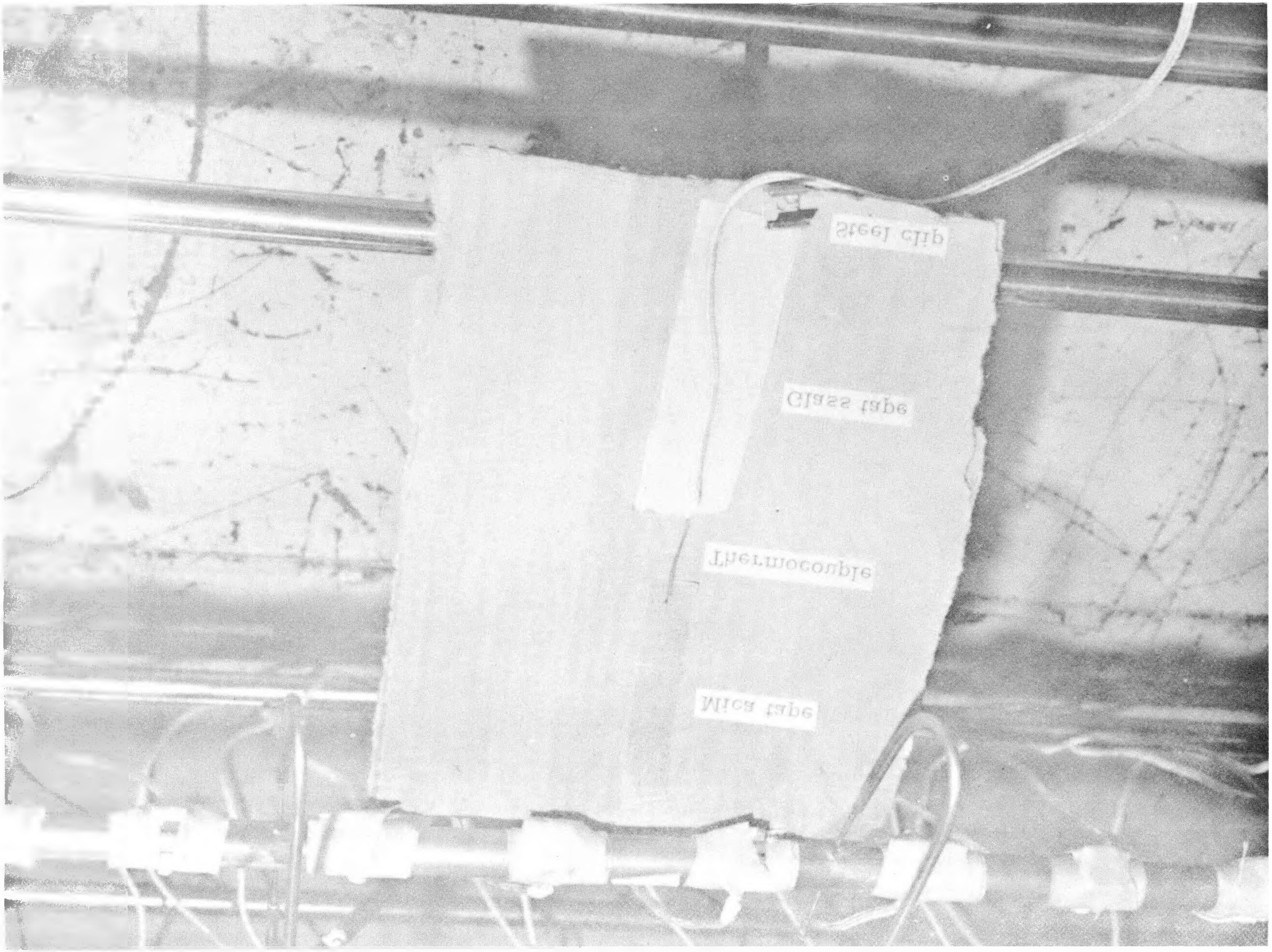


Fig. II-31. Close-Up of Thermocouple and Pressure Tap Installation

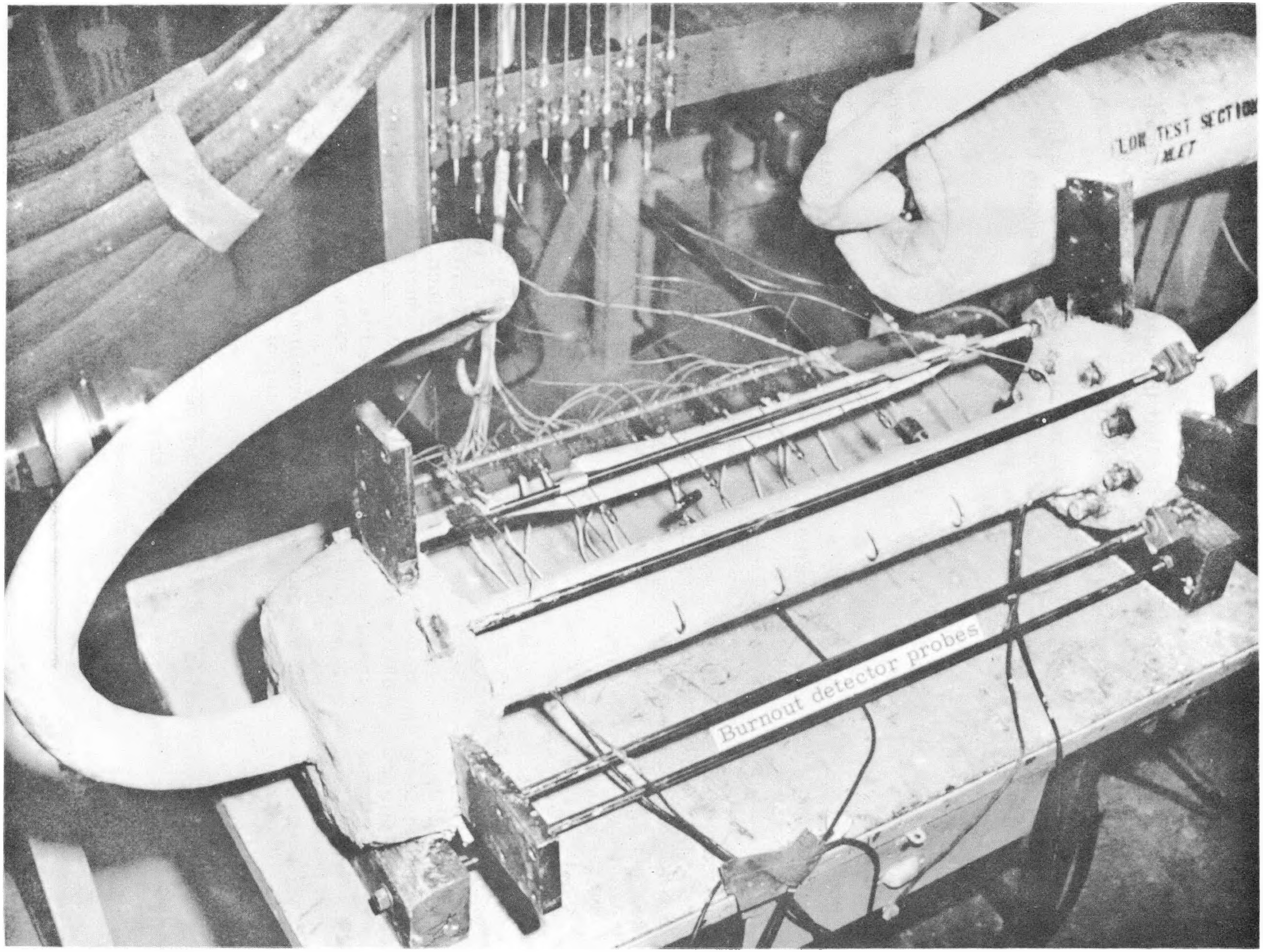


Fig. II-32. SSTS-3 Completely Installed for Horizontal Pressure Drop Test

(1) Friction factor data indicated that the test section is hydraulically acceptable, i.e., it compared favorably with smooth tube data presented in the literature. *

(2) One of the 12 pressure taps is not acceptable but this tap can be bypassed and a manifold system employed to obtain the pressure drop increment in question.

2. STTS-4 Burnout Test Program

The components for the single-tube burnout test section were fabricated during the report period. Brazing of these components is presently scheduled for early next quarter.

The objective of the STTS-4 Experimental Program is to obtain local boiling burnout data at the parameters of interest for the PM-1 reactor. This program is to be conducted with a single-tube test section having coolant flow both inside and outside the tube. Burnout data are desired where the coolant is subcooled in order that the burnout safety margin at hot spots can be determined. In this type of burnout (local boiling), the overall bulk coolant temperature is not appreciably affected by hot spots and the local subcooling is maintained. A program conducted at various subcooling allows simulation of the conditions at various locations in the reactor core. In order to obtain the subcoolings desired in a test section where the heat flux is uniform, a reduced length test section is required. Thus, the length was reduced from the 30 inches of the PM-1 fuel element to 15 inches. A second dimensional change was dictated by the power requirements for local boiling burnout. The test section wall thickness was sized to provide an electrical resistance which would draw the full capacity of the 80-volt, 3750-ampere power supply. A wall thickness of 0.0205 inch was selected. This is slightly less than 1/2 of the PM-1 fuel element wall thickness of 0.042 inch. Burnout data, therefore, should be conservative due to the reduced heat capacity of the test section relative to that of the actual fuel element. The tube inside diameter was made the same as that of the fuel element. The outer annular flow path was sized to produce the same coolant velocity outside the tube as that which exists in the PM-1 core. The test section design enables the full utilization of the available 296 kw. The reduced length test section will allow investigation of local boiling burnouts with up to 80° F subcooling.

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*See Fig. II-33 which is a plot of head loss per inch of test section length versus flow. The range of Reynolds number depicted hre is 3.9×10^4 to 1.56×10^5 which includes the range of interest in the local boiling tests.

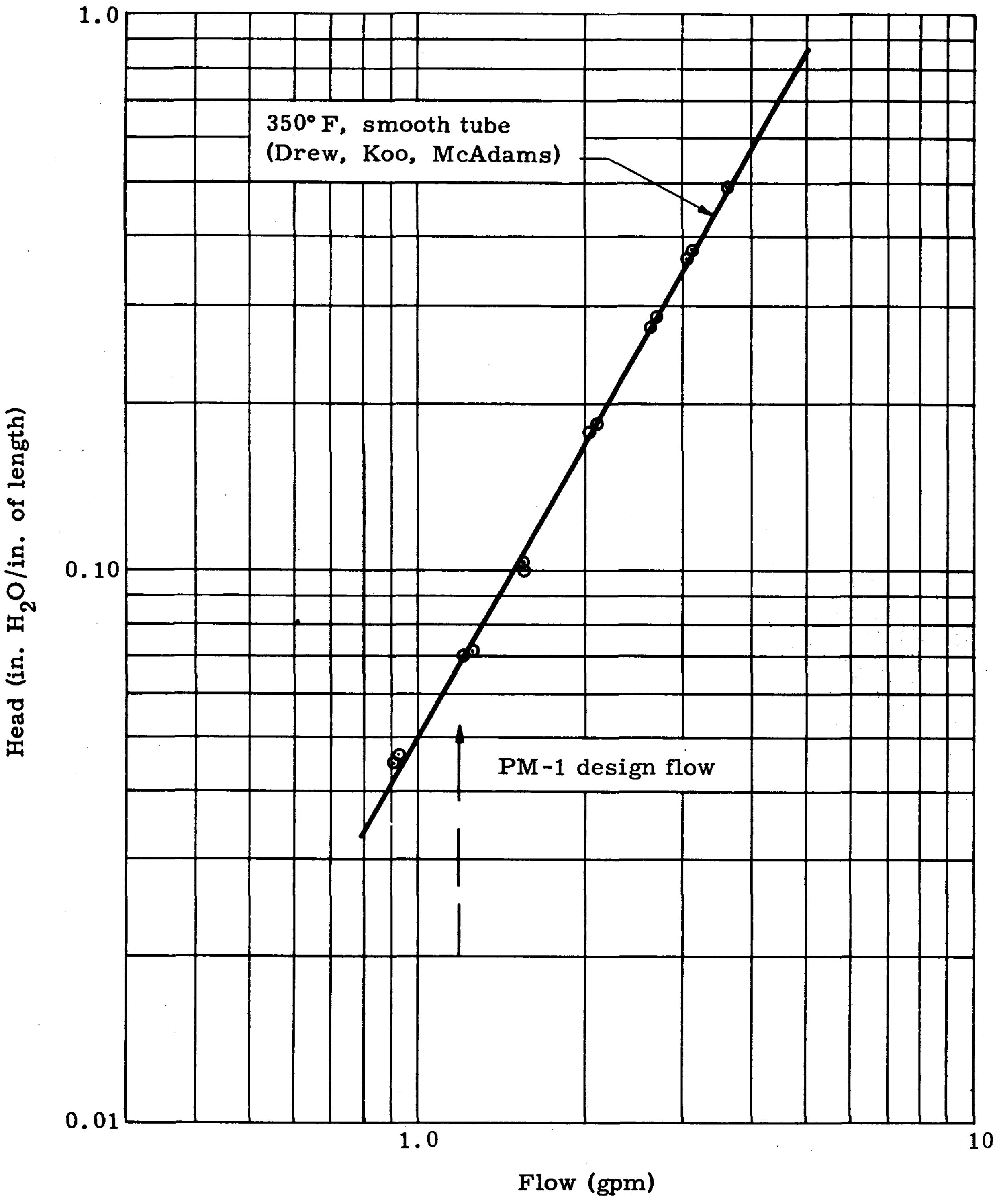

Fig. II-33. STTS-3 Isothermal Pressure Drop Test

The basis of the test section design, that of attaining maximum heat flux and large subcoolings, has resulted in significant test section dimensional changes as noted above. Reducing the length should not appreciably affect the burnout results if the results as reported in the literature can be substantiated. WAPD (Ref. 1) presents a correlation that includes an L/D term (length-to-diameter ratio). The WAPD correlation indicates that burnout heat flux is increased by about 2% for the length reduction in STTS-4. Bernath (Ref. 2) shows evidence that, for test section lengths greater than 8 inches, burnout data are essentially independent of length, while for lengths shorter than 8 inches, higher heat fluxes are required for burnout. On the other hand, the reduced thickness, i.e., reduced heat capacity, of the test section could result in errors up to 10% (Ref. 2). It should be noted, however, that the heat capacity effects diminish rapidly with increasing thickness and may be nonexistent at the STTS-4 thickness. The critical thickness in this respect is reported (Ref. 2) as 0.04 inch. Thus, the STTS-4 test section could give burnout at somewhat lower heat flux than the actual fuel element. This possible error, however, would give results that are conservative.

A schematic of STTS-4 is shown in Fig. II-34. The instrumentation that has been directly tied into the test section has been limited to ensure a minimum of disturbance to the physical model in the burnout area. Thus, there are only three voltage probes attached to the test section tube. These probes will be employed to give the voltage drop across the test section and to compare the electrical resistances of the upstream and downstream halves of the test section via the burnout detector. There are no wall temperatures recorded during this test, since the temperature of primary concern is at the burnout point and this cannot be realized without affecting burnout conditions. Bulk coolant temperature, however, will be recorded at four points (Fig. II-34):

(1) Mixed mean inlet temperature, T_1.

(2) Mixed mean outlet temperature, T_4.

(3) Effluent temperature in annular flow path, T_5.

(4) Effluent temperature inside the tube, T_2.

Coolant flow rate will be measured in the main flow stream and in the flow path for the annular region of the test section as shown in Fig. II-35. Also illustrated are the flow control valves for both annular and inside tube flow.

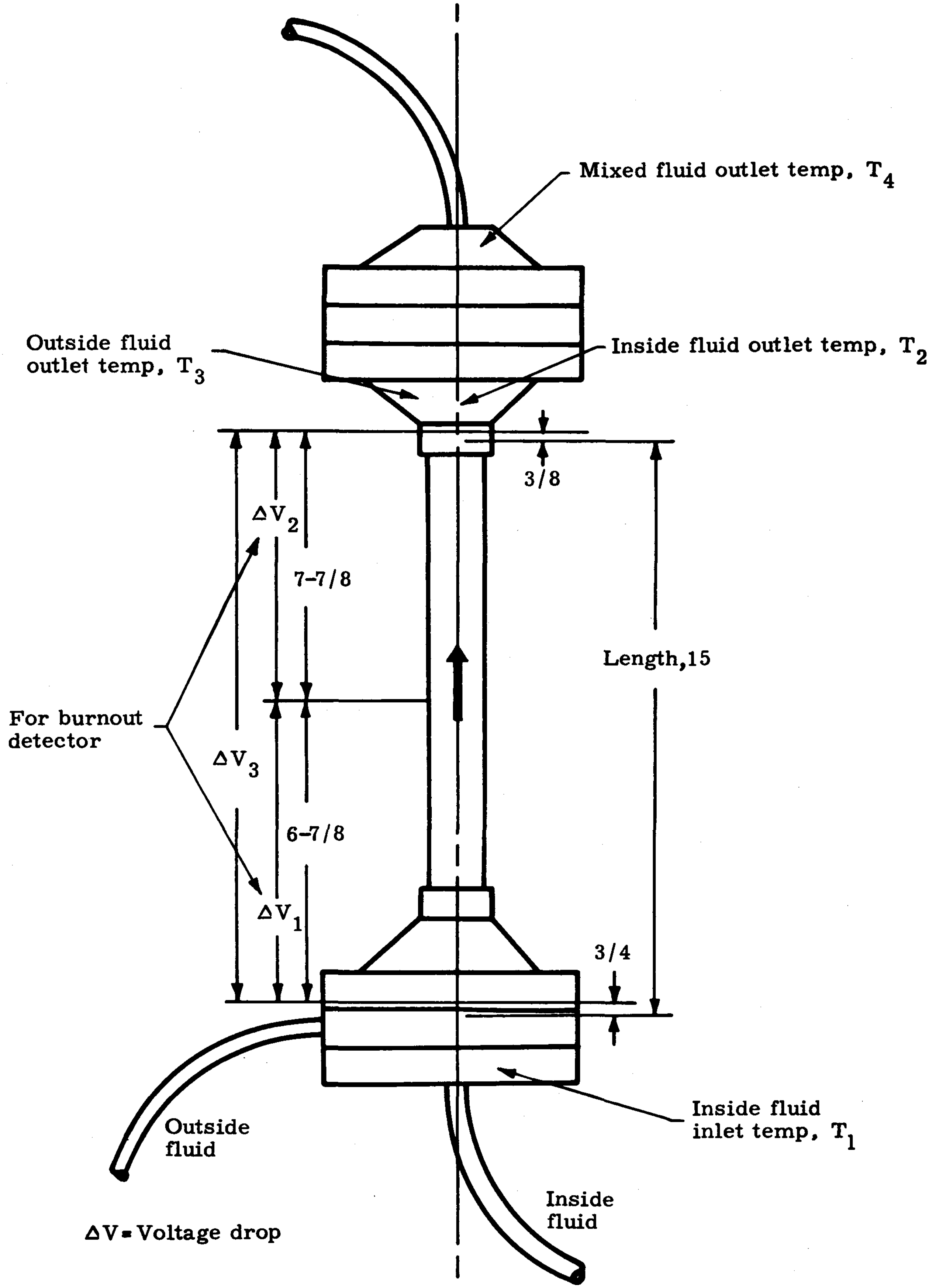

Fig. II-34. STTS-4 Locations of Instrumentation

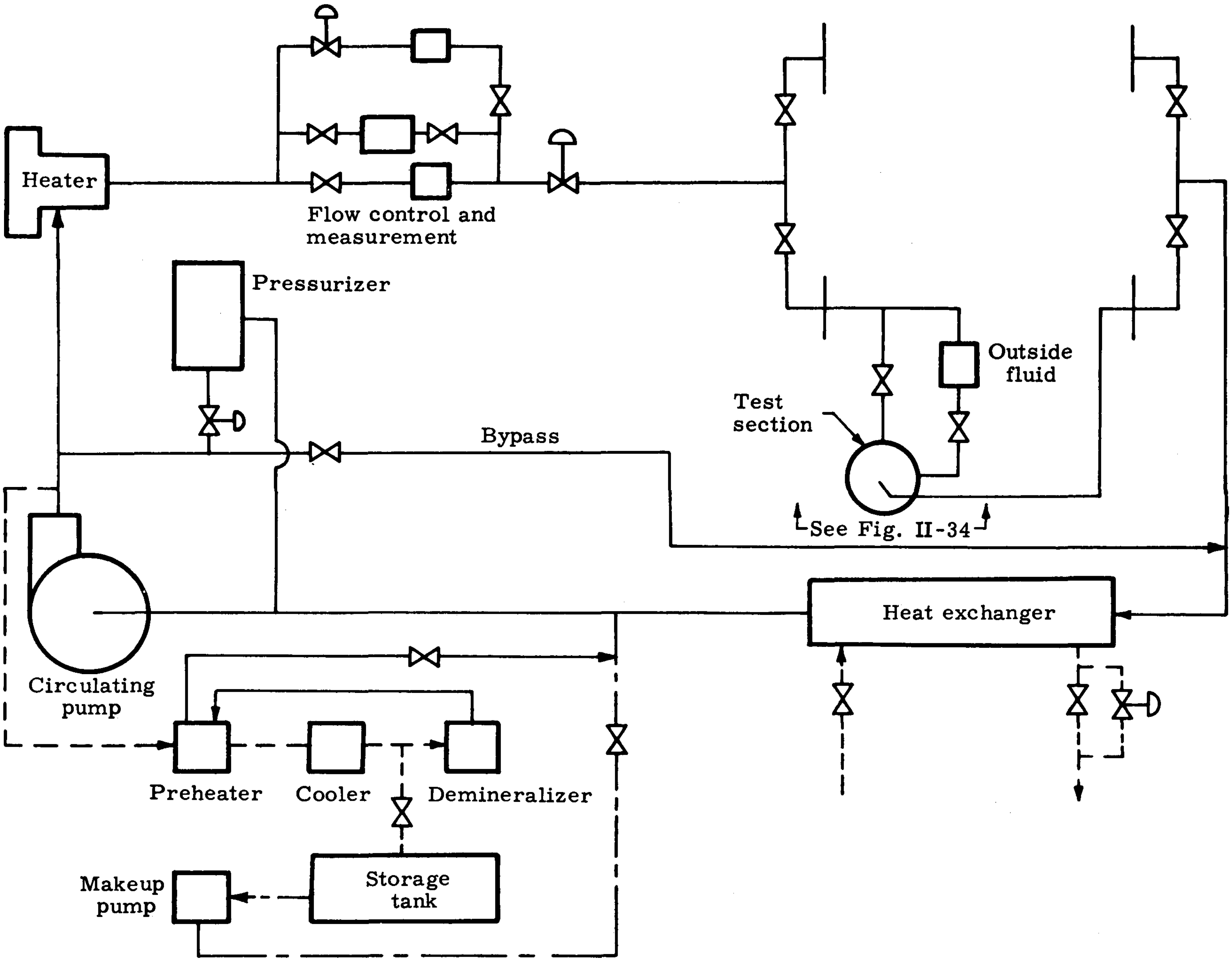

Fig. II-35. Heat Transfer Loop

Conservation of the test sections will be accomplished by employing a burnout detector. The burnout detector is a device designed to detect small changes in the electrical resistance of the test section. This small resistance change occurs at incipient burnout when a localized area starts to overheat due to the initiation of partial film boiling. A continuous comparison is made between the resistances of the upstream and downstream, i.e., burnout, halves of the test section. At burnout, the Wheatstone bridge circuit of the detector is unbalanced, activating the trigger unit. The trigger unit sends a signal to a high speed d-c circuit breaker which cuts off the power to the test section.

a. Determination of test parameters

The local boiling test parameters were determined by employing the Jens and Lottes correlation (Ref. 3):

$$\left(\frac{q/A}{10^6}\right)_{BO} = c \left(\frac{G}{10^6}\right)^m (\Delta T_{sub})^x$$

where

c = pressure dependent constant

G = mass flow rate, lb/hr-ft^2

T_{sub} = subcooling (saturation temperature--bulk temperature), °F

m = pressure dependent constant

x = constant = 0.22

$(q/A)_{BO}$ = burnout heat flux, Btu/hr-ft^2.

Since burnout occurs at the outlet for uniformly heated test sections, various outlet subcoolings were investigated and the burnout heat flux determined. With the burnout heat flux so determined, it was possible to calculate the coolant inlet temperature for various flow rates. An important assumption that was made in the above calculation was that the total integrated heat flow to the coolant flowing inside the tube was equal to that flowing outside the tube. This assumption is quite good if large portions of both heat transfer surfaces are undergoing local boiling. During local boiling, the heat transfer surface is essentially isothermal and the quantity of heat transferred is dependent on the heat flux and the pressure. This mode of heat transfer is independent of coolant velocity. At the high heat fluxes experienced prior to burnout, the above situation is adequately approximated. The results, based on the above discussion, are shown in Figs. II-36 through II-40 and are considered again below.

b. Test procedure

Burnout conditions which are caused by the initiation of metastable (i.e., partial) film boiling can be approached by 4 methods, each of which, in effect, reduces the bulk coolant subcooling. Subcooling is defined as the difference between the local saturation temperature and the local bulk coolant temperature. The methods of approaching burnout are:

(1) Increase the power (heat flux) to the test section at a given flow rate, pressure and inlet temperature, thereby increasing the bulk coolant outlet temperature.

(2) Decrease the coolant flow rate at a given power level, inlet temperature and system pressure, resulting also in an increased bulk coolant outlet temperature.

(3) Increase the inlet bulk coolant temperature at a given power level and coolant flow rate by a combination of increased preheater power and decreased heat rejection at the heat exchanger.

(4) Decrease the system pressure at a given power level, coolant flow rate and coolant inlet temperature, thereby reducing the saturation temperature and, subsequently, the subcooling.

The first method enumerated, (1), of increased power to the test section, is used more widely than the others because of its direct approach. This method has certain inherent disadvantages such as:

(1) Burnout can occur while the power is being increased, resulting in an uncertainty in the actual burnout heat flux.

(2) The attainment of premature burnout as a result of too rapid an increase in power.

Obviously, these disadvantages can be readily eliminated if care is taken with a power control system capable of small incremental power advances. The power control system as presently constituted cannot accomplish the required power control, but such means will be avail-

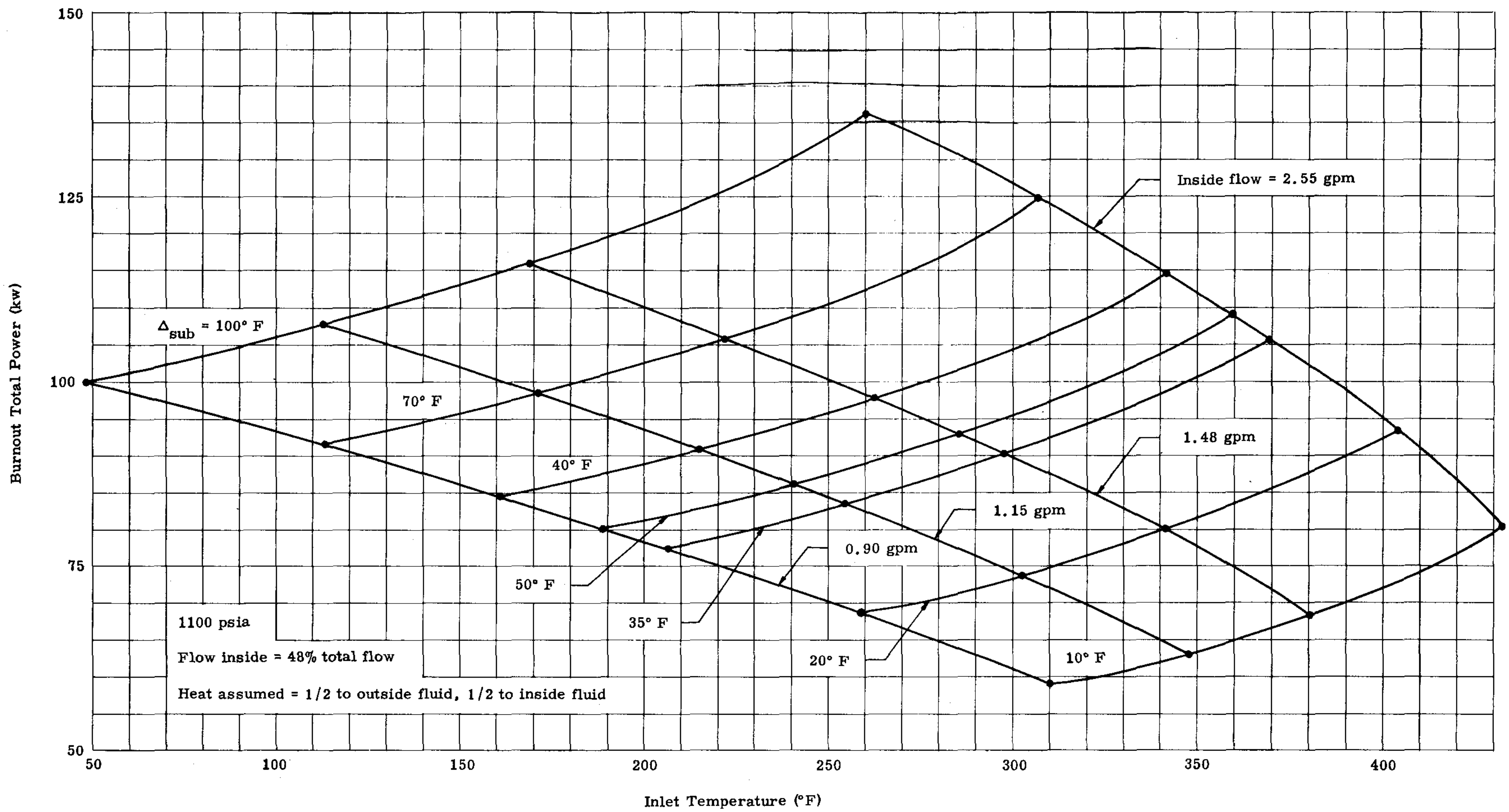

Fig. II-36. STTS-4 Test Parameters

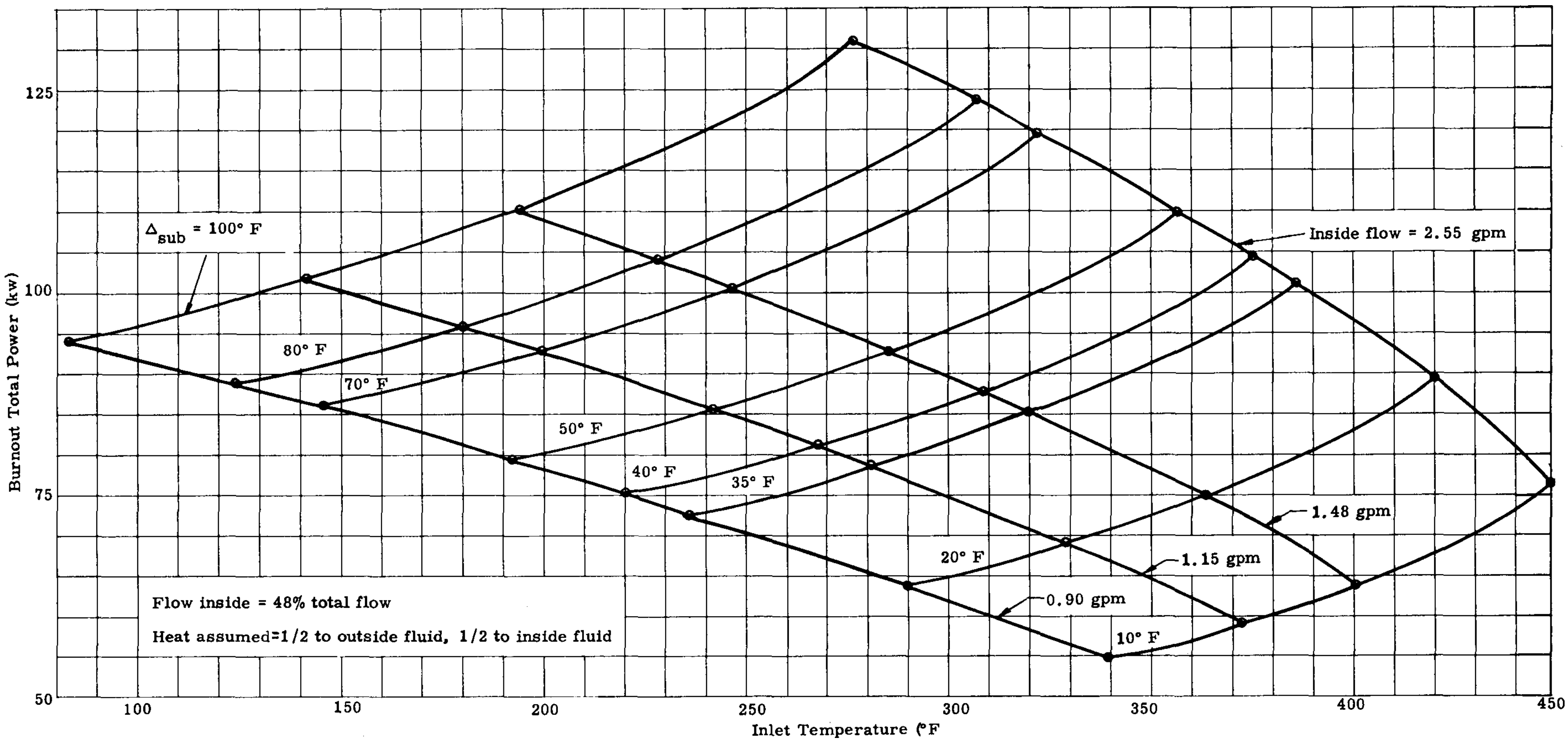

Fig. II-37. STTS-4 Test Parameters--1200 psia

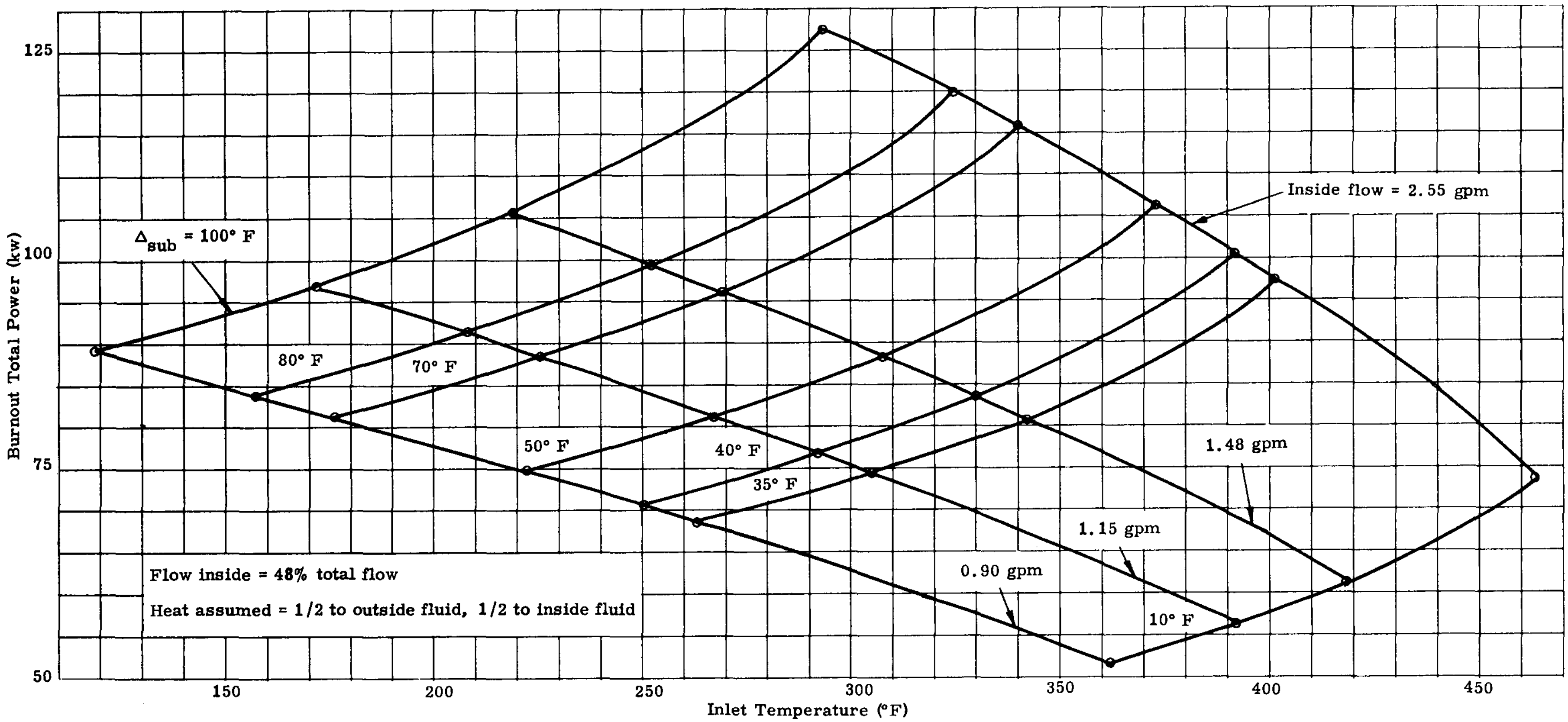

Fig. II-38. STTS-4 Test Parameters--1300 psia

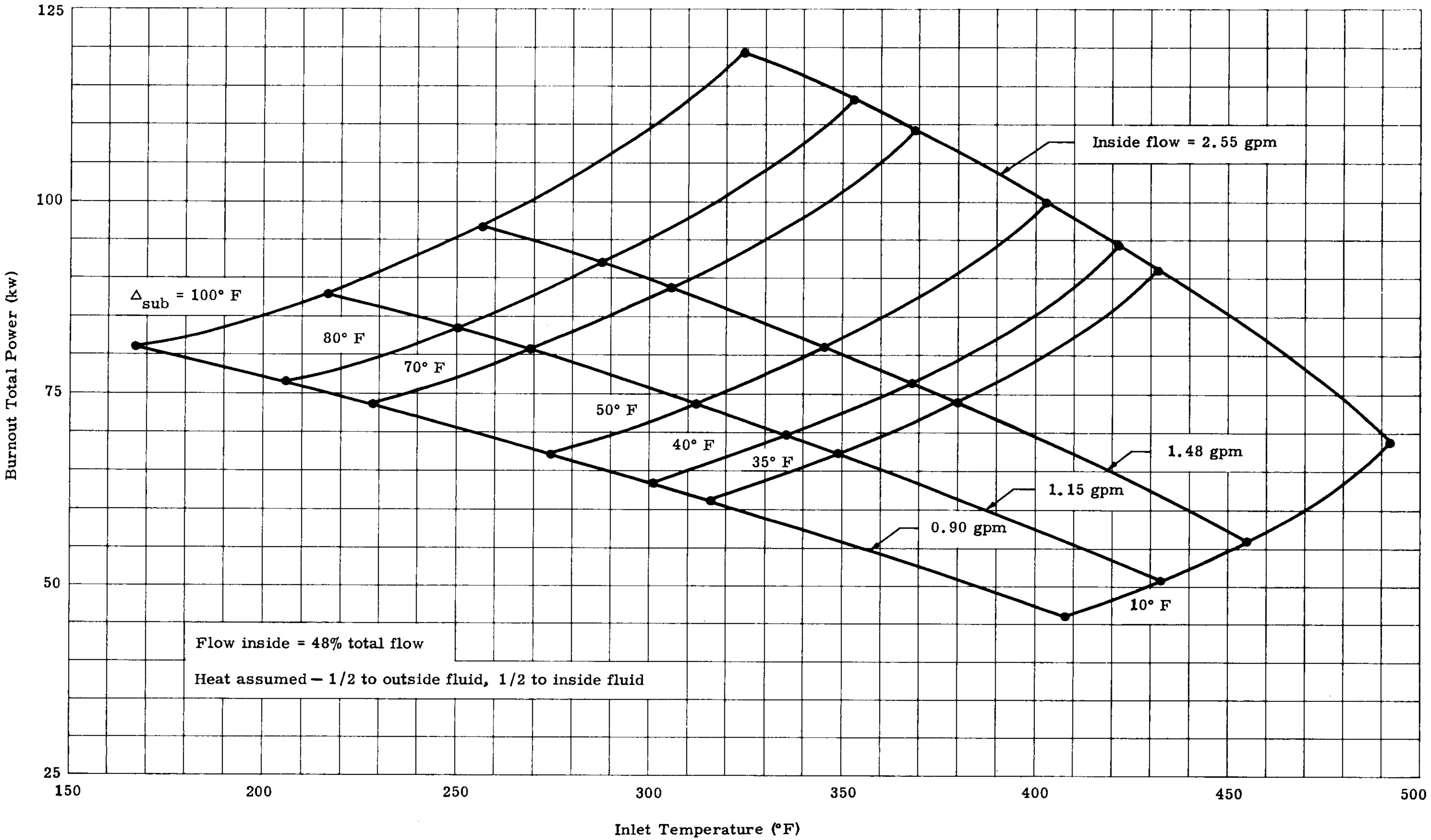

Fig. II-39. STTS-4 Test Parameters--1500 psia

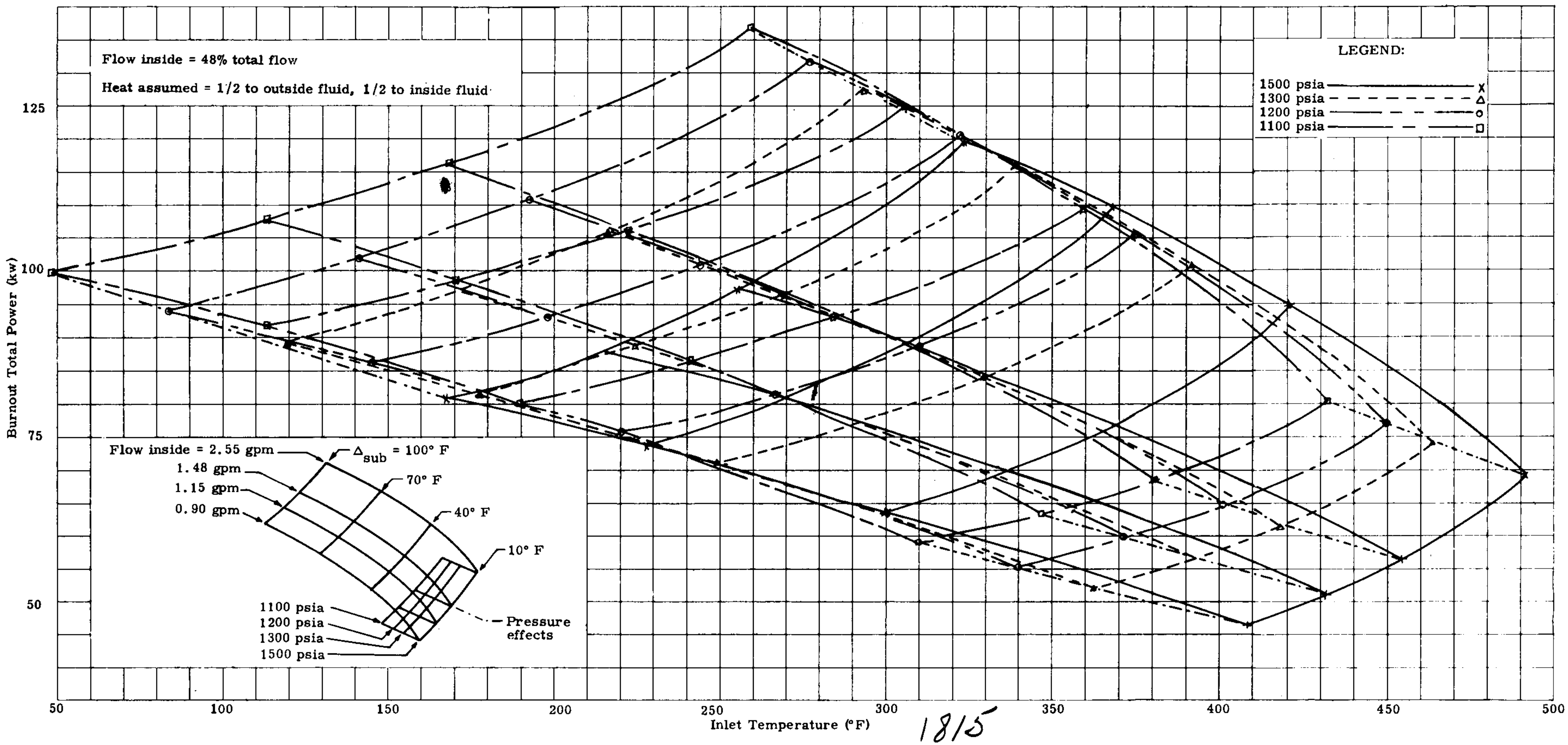

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Fig. II-40. STTS-4 Test Parameters

able for the STTS-4 program. In any event, the second method of approaching burnout will be employed to ascertain the existence of any effect of the power advance technique. Thus, the power will be set at a high flow rate and then the flow rate decreased slowly until burnout is realized. The second method will be employed periodically (approximately every 25 runs) to check Method (1).

Neither Method (3) nor (4) will be attempted during this program due to the absence of the required control systems.

c. Test program

Figures II-36 through II-39 show the results for one set of the originally proposed test parameters which were as follows:

Pressure: 1100, 1200, 1300, 1500 psia

Total coolant flow rate: 1.88, 2.40, 3.08, 5.31 gpm

Flow ratio, $\frac{\text{inside flow}}{\text{outside flow}}$: 0.5, 0.75, 0.923, 1.25

Outlet subcooling: 10, 40, 70, 100° F.

Figures II-36 through II-39 are plots of total power to the test section at burnout versus bulk coolant inlet temperature at various pressures and flow ratios with both outlet subcooling, ΔT_{sub}, and flow rate as parameters. These plots give a graphic indication of the various approaches to burnout at a given pressure. Inspection of the curves reveals that the required inlet temperatures for 100° F subcooling are often very low. However, the heat transfer loop can provide a minimum inlet temperature of about 120° F. This limit necessitated a reconsideration of the proposed parameters. It was beyond the capability of practical testing to obtain data at about 12% of the proposed sets of parameters. It was found that local boiling burnout at 100° F outlet subcooling was not feasible at a flow ratio of 0.5. (The flow ratio is defined as the ratio of the flow rate inside the tube to that of the outside flow path.) This flow ratio is of some interest, since the PM-1 core in the area of the control rods will experience a similar flow ratio. Some runs will be made, therefore, at this ratio, but at subcoolings ranging from 10 to 80° F.

Figure II-40 is a composite plot of Figs. II-36 through II-39. It shows the effect of pressure on burnout power along with outlet subcooling and coolant flow rate. The true value of these plots is that they enable the history of any set of test parameters to be followed up to the burnout point.

The tests comprise the following phases:

(1) Physical Burnout Tests--Physical burnout of 3 test section tubes will be accomplished to ensure a positive calibration of the burnout detector. These runs will be carried out at the PM-1 design flow rate of 2.40 gpm per cell, pressure of 1300 psia, inside-to-outside flow ratio of 0.923 and a bulk coolant outlet subcooling of 35° F. The burnout flux herein obtained will be compared to those obtained at the same conditions to be run in Part (2) of this program. Burnout will be achieved in this case, as well as throughout the bulk of the program, by increasing the power at a constant flow rate, pressure and inlet coolant temperature. The burnout detector will be connected to the test section, but the "trigger unit" of this device will not be connected to the high speed d-c circuit breaker. The visual burnout indication will be used to determine the power at which the detector would have tripped the d-c breaker. Thus, the confidence in the indicated burnout data can be evaluated.

(2) Indicated Burnout Tests (burnout detector)--The normal burnout runs, i.e., those that will be conducted with the burnout detector in operation, will be terminated at electrically defined burnout points. The parameters to be investigated are:

Pressure: 1100, 1200, 1300, 1500 psia

Flow ratio: 0.75, 0.923, 1.25 (1.88 to 5.31 gpm)

Outlet subcooling: 10, 20, 35, 50, 80° F

Total number of runs: 192.

(3) Periodic Burnout Check Test--Burnout detector check runs will be made after every 25 normal burnout runs. Two such check runs will be made; one by increasing the power at a constant flow rate, pressure and inlet temperature and the second by setting a power level at a constant pressure and inlet temperature and then decreasing the flow rate until burnout is realized. Since there are 192 normal burnout runs, a total of 16 check runs will be performed at PM-1 design flow, pressure and 35° F subcooling.

(4) Special Burnout Tests--Special burnout conditions will be investigated if time permits. Eight extra runs will be carried out to obtain additional 20° F subcooling burnouts. These runs are necessitated by correlation requirements and will be conducted at the following conditions:

Pressure: 1200 psia

Total coolant flow rate: 1.875, 2.40, 3.08, 5.31 gpm

Flow ratio: 0.923, 1.25

Outlet subcooling: 20° F.

In addition, 4 runs will be made to investigate burnout in the area applicable to the PM-1 core control rods at the following conditions:

Pressure: 1300 psia

Total coolant flow rate: 3.45 gpm

Flow ratio: 0.5

Outlet subcooling: 10, 35, 50, 80° F.

3. SETCH-2 Seven-Tube Test Section Program

Fabrication of SETCH-2 reached the 80% completion point during this quarter. This unit will be ready to follow STTS-3 into the heat transfer loop.

This test is the complement of the single-tube test, STTS-3, and is designed to obtain local boiling pressure drop and heat transfer data outside of simulated PM-1 tubes. This program, in conjunction with the STTS-3 program, gives the complete local boiling heat transfer and pressure drop characteristics of a PM-1 fuel element.

Four pressures(ranging from 1100 to 1500 psi) and 3 inlet temperatures (445, 465 and 485° F) will be investigated. Six coolant flow rates, ranging from 10.90 to 30.88 gpm, give the corresponding outside velocities for those inside velocities investigated in the STTS-3 program.

Heat loss and nonboiling pressure drop tests are included. The nonboiling pressure drop tests are to serve as a reference for the local boiling tests. Isothermal empty shell pressure drop tests (Test A) will be employed to factor out the contribution of the nonboiling test section shell during local boiling runs. A number of forced convection runs will be made to determine the resistances between the thermocouples and the outside surfaces of the tube walls. Forced convection runs will be made periodically during the program to determine any change in resistance with time and to check instrumentation.

A total of 215 runs are outlined in the program.

The locations of the instrumentation in the test section are shown in Fig. II-41. Figure II-42 shows the wall thermocouples inside the 7 tubes. The center tube has 16 thermocouples at regular intervals along the entire tube. The 6 and 12 o'clock tubes have only 1 thermocouple at the outlet end but also contain the voltage probes. The remaining tubes contain 1 inlet and 1 outlet thermocouple each.

The various tests proposed and the test parameters are given below.

a. Empty shell pressure drop

This test will be run with the test section in a horizontal position. The configuration will be similar to the SETCH-2 assembly except that the seven tubes will be omitted. This will be accomplished by removing from the complete assembly the brazed assembly which includes the tubes and orifice flange. A standard blind flange shall be put on the resulting open end. This test will yield pressure drop measurements of the shell housing which will not be heated during the local boiling runs. The tests will be performed in the Reynolds number range to be used in the test programs. The operating conditions are given in Table II-10.

TABLE II-10

SETCH-2 Operating Conditions for Empty Shell Pressure Drop

Run Number	Pressure (psia)	Inlet Temperature (°F)	Flow Rate (gpm)
1	100	150	2.30
2	100	175	2.30
3	100	175	10.80
4	100	150	10.80
5	100	150	19.30
6	100	175	19.30
7	100	175	27.80
8	100	150	27.80
9	100	150	36.30
10	100	175	36.30
11	100	175	44.50
12	100	150	44.50

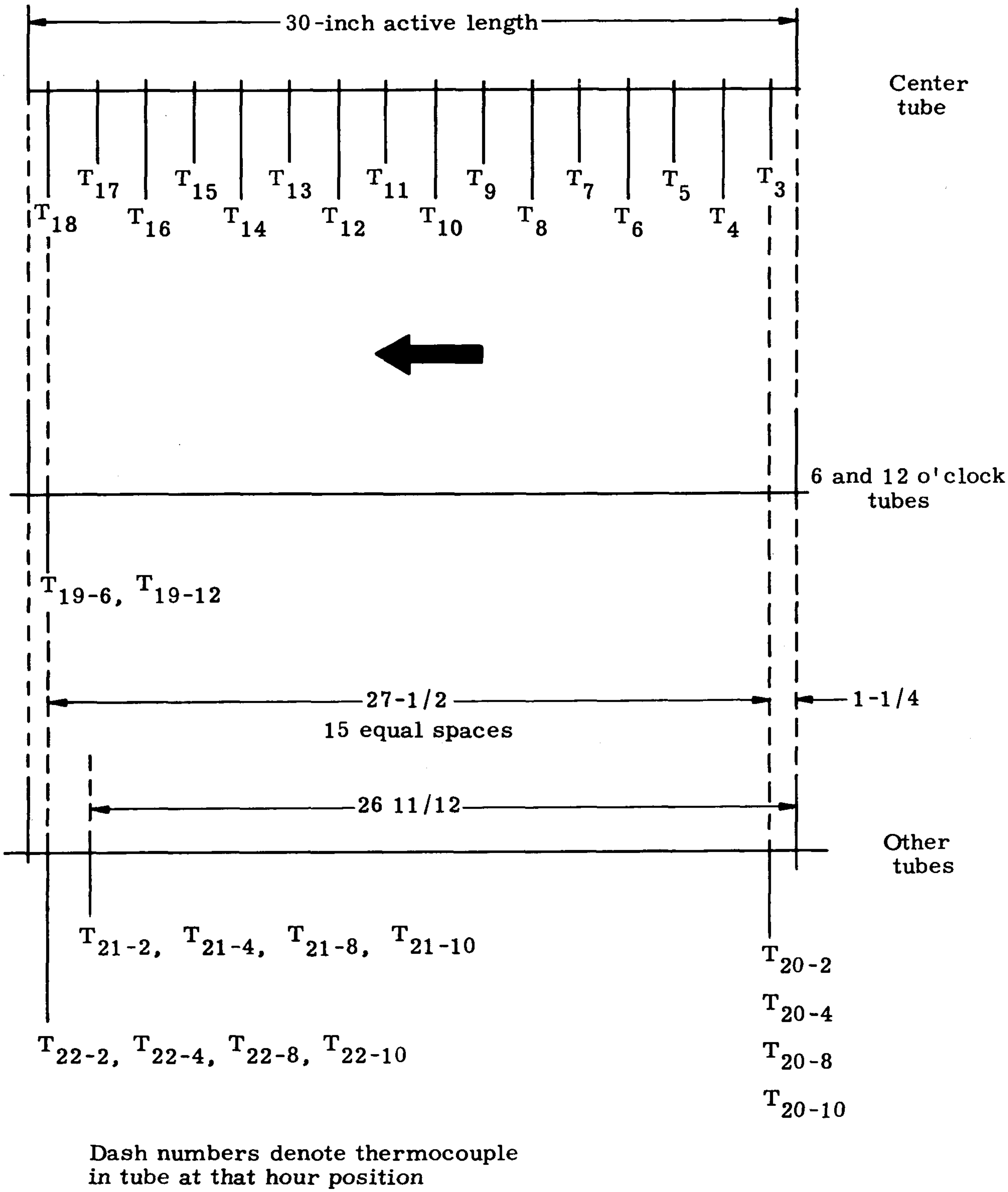


Fig. II-41. SETCH-2 Wall Thermocouple Locations

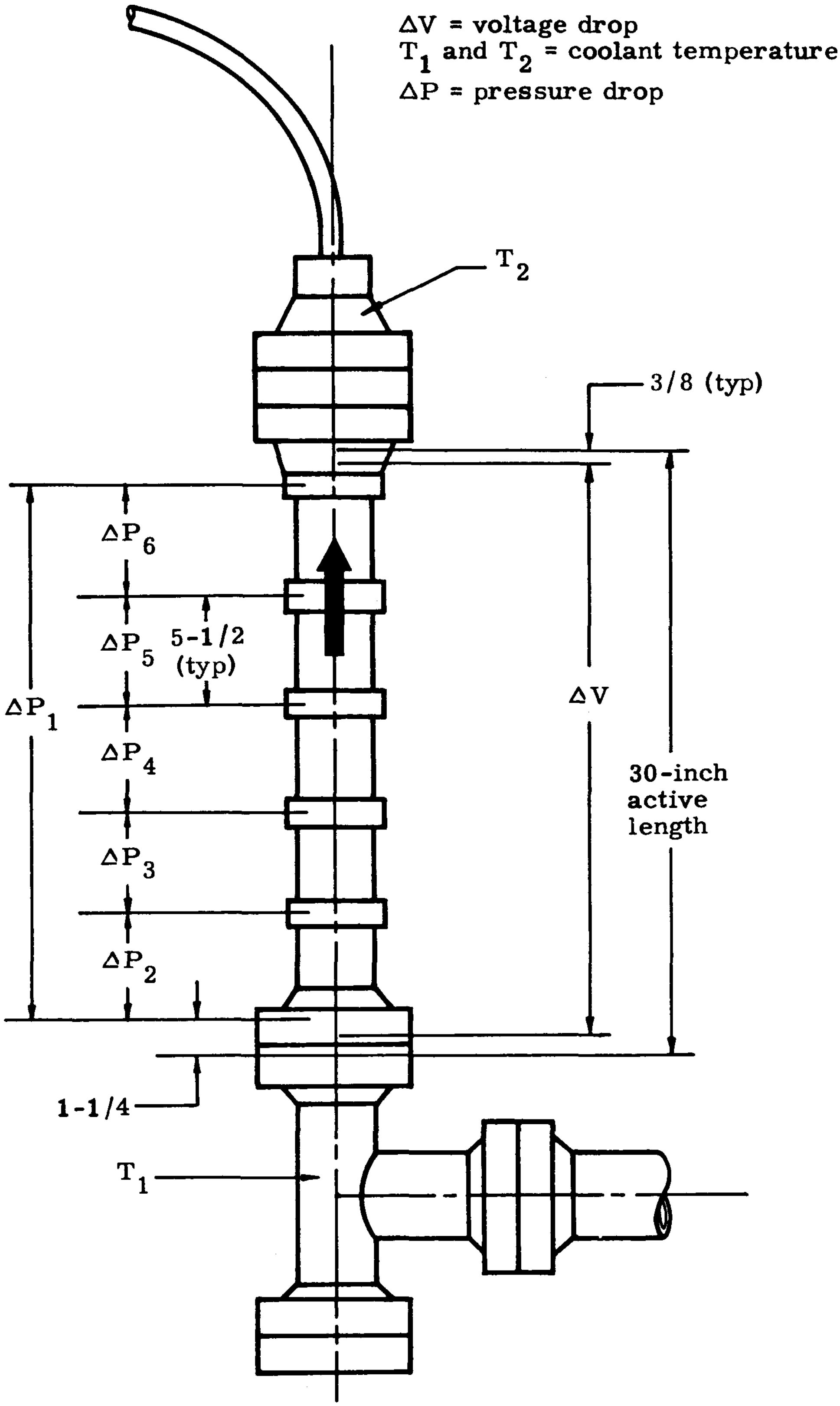


Fig. II-42. SETCH-2 Instrumentation Locations

b. Horizontal flow calibration

This test will be run with the complete test section installed in a horizontal position to afford a means of obtaining isothermal pressure drop data free of the elevation pressure drop component. The operating conditions are given in Table II-11.

TABLE II-11

Operating Conditions for Horizontal Flow Calibration

Run Number	Pressure (psia)	Inlet Temperature (°F)	Flow Rate (gpm)
13	1300	445	6.0
14	1300	465	6.0
15	1300	465	13.2
16	1300	465	13.2
17	1300	465	17.0
18	1300	445	17.0
19	1300	445	22.0
20	1300	465	22.0
21	1300	465	27.0
22	1300	445	27.0
23	1300	445	35.0
24	1300	465	35.0

c. In-place flow calibration

For this test, the test section will be installed in the vertical position. Runs 13 through 24 will be repeated as Runs 25 through 36. This will allow a comparison with the results of Test B for the effects of gravity.

d. Forced convection heat transfer

This test will yield overall thermal resistance data for the wall thermocouple measurements using the Wilson plot method. Operating conditions are given in Table II-12.

TABLE II-12

Operating Conditions for Forced Convection Heat Transfer Test--
System Pressure: 1300 psia, Coolant Inlet Temperature: 445° F

Run Number	Flow Rate (gpm)	Power Input (kw)
37	45.0	70.0
38	45.0	105.0
39	45.0	140.0
40	45.0	175.0
41	70.0	70.0
42	70.0	105.0
43	70.0	140.0
44	70.0	175.0
45	90.0	70.0
46	90.0	105.0
47	90.0	140.0
48	90.0	175.0
49	110.0	70.0
50	110.0	105.0
51	110.0	140.0
52	110.0	175.0

e. Local boiling pressure drop and heat transfer test

TABLE II-13

Group I

System pressure: 1300 psia
Coolant inlet temperature: 445° F

Run Number	Flow Rate (gpm)	Power Input (10^3 Btu/hr-ft^2)	(kw)
53	10.90	1.10	73.8
54	10.90	1.35	90.6
55	10.90	1.65	110.75
56	13.20	1.35	90.6

TABLE II-13 (continued)

Run Number	Flow Rate (gpm)	Power Input (10^3 Btu/hr-ft^2)	(kw)
57	13.20	1.65	110.75
58	13.20	2.00	134.25
59	14.52	1.35	90.6
60	14.52	1.65	110.75
61	14.52	2.00	134.25
62	17.92	1.65	110.75
63	17.92	2.00	134.25
64	17.92	2.70	181.2
65	21.79	2.00	134.25
66	21.79	2.70	181.2
67	21.79	3.30	221.5
68	30.88	2.70	181.2
69	30.88	3.30	221.5
70	30.88	4.20	281.8
71 Repeat Run No. 37			
72 Repeat Run No. 41			

Group II

System pressure: 1300 psia
Coolant inlet temperature: 465° F
Repeat Runs 53 through 72 and record as Runs 73 through 92.

Group III

System pressure: 1200 psia
Coolant inlet temperature: 465° F
Repeat Runs 53 through 72 and record as Runs 73 through 112.

TABLE II-13 (continued)

Group IV

System pressure: 1200 psia
Coolant inlet temperature: 445° F
Repeat Runs 53 through 72 and record as Runs 113 through 132.

Group V

System pressure: 1100 psia
Coolant inlet temperature: 445° F
Repeat Runs 53 through 72 and record as Runs 113 through 152.

Group VI

System pressure: 1100 psia
Coolant inlet temperature: 465° F
Repeat Runs 53 through 72 and record as Runs 153 through 172.

Group VII

System pressure: 1500 psia
Coolant inlet temperature: 465° F
Repeat Runs 53 through 72 and record as Runs 173 through 192.

Group VIII

System pressure: 1500 psia
Coolant inlet temperature: 485° F
Repeat Runs 53 through 72 and record as Runs 193 through 212.

f. High heat flux tests

The following runs simulate special PM-1 conditions of local boiling.

TABLE II-14

High Heat Flux Tests

Run Number	Inlet Temperature (°F)	Power Input (10^5 Btu/hr-ft^2)	Power Input (kw)
213	429	2.763	185.3
214	410	3.315	222.4
215	375	4.330	290.4

REFERENCES

(1) DeBortoli, R.A., et. al., "Forced Convection Heat Transfer Burnout Studies for Water in Rectangular Channels and Round Tubes at Pressure Above 500 PSIA," WAPD-188.

(2) Bernath, L., "A Theory of Local Boiling Burnout and Its Application to Existing Data," Third National Heat Treansfer Conference, ASME AIChE, August 1959.

(3) Jens, W.H., and Lottes, P.A., "Analysis of Heat Transfer Burnout, Pressure Drop and Density Data for High Pressure Water," ANL-4627, May 1951.

E. SUBTASK 2.5--ACTUATOR PROGRAM

J. Sieg, R. Manoll

During the fourth quarter, The Martin Company continued to monitor and control the development of the prototype magnetic jack actuator system by the TAPCO group of The Thomson Ramo Wooldridge Company.

TAPCO was scheduled to complete prototype actuator system design and fabrication and to initiate system testing during the quarter.

Design and fabricate was completed in all but two areas:

(1) Tooling.

(2) Insulating the scram spring housing to limit the loss of heat into the shield water.

Subsystem testing has been completed; system testing will begin early in March with the hydrostatic testing of the actuator pressure shell. Reliability and life tests are expected to be initiated by mid-March, after the termination of hydrostatic testing.

Considerable scheduled disruption was caused by the unanticipated failure of two successive armature housing stock forgings to pass ultrasonic inspection. The forging is extremely critical in that the 0.10-inch thick pressure shell about the actuators is fabricated from this component.

After the existence of the problem was recognized, a recovery schedule was prepared and is being followed.

It presently appears that the time delay caused by forging failure will result in:

(1) Initiating reliability and life testing immediately after hydrostatic testing. The performance of humidity, shock and vibration testing will be postponed until after the reliability and life tests have been run.

(2) Delaying the initiation of the system reliability and life tests by approximately 2 weeks.

(3) Placing the order for production system forgings and spares earlier than was originally scheduled.

The overall system test program is expected to be completed at the end of the next quarter, as was originally scheduled.

1. Design

(1) System shop drawings were submitted to The Martin Company and to the government.

(2) The system design analysis was submitted to The Martin Company and the government.

(3) Preliminary tooling designs were submitted to The Martin Company; redesign, based on Martin Company comments, is in process.

(4) The preliminary design of an insulating scheme for the scram spring housing is due The Martin Company on 15 March 1960.

2. Fabrication

(1) Figure II-43 shows the fabricated armature housing and flange, fixed and movable armatures, belleville springs and armature fittings.

(2) Figure II-44 shows the assembled actuator pressure shell prior to hydrostatic testing.

(3) Figure II-45 shows the assembled drive coil stack.

(4) Figure II-46 shows the assembled position indicator differential transformer, lead screw, drive motor and synchro transmitter.

Fig. II-43. Fabricated Armature Housing and Flange, Fixed and Movable Armatures, Belleville Springs and Armature Fittings

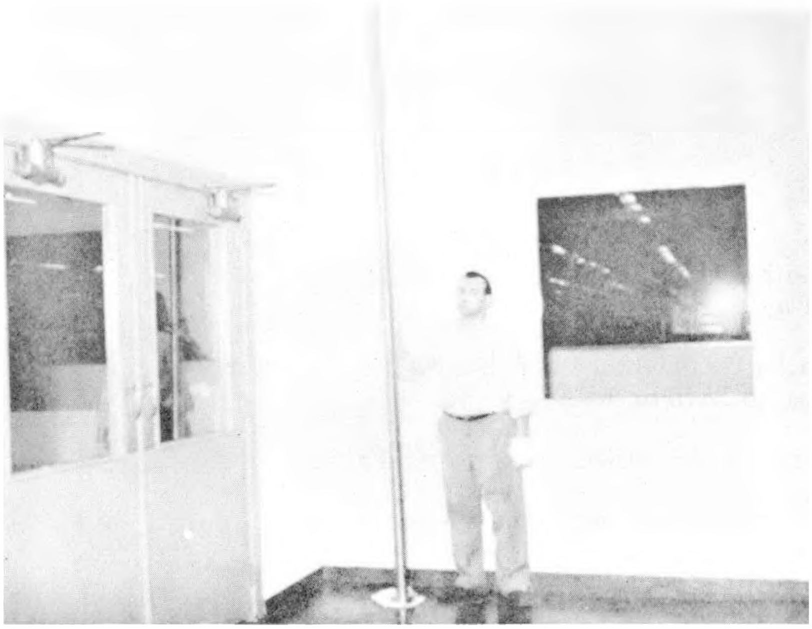


Fig. II-44. Assembled Actuator Pressure Shell Prior to Hydrostatic Testing

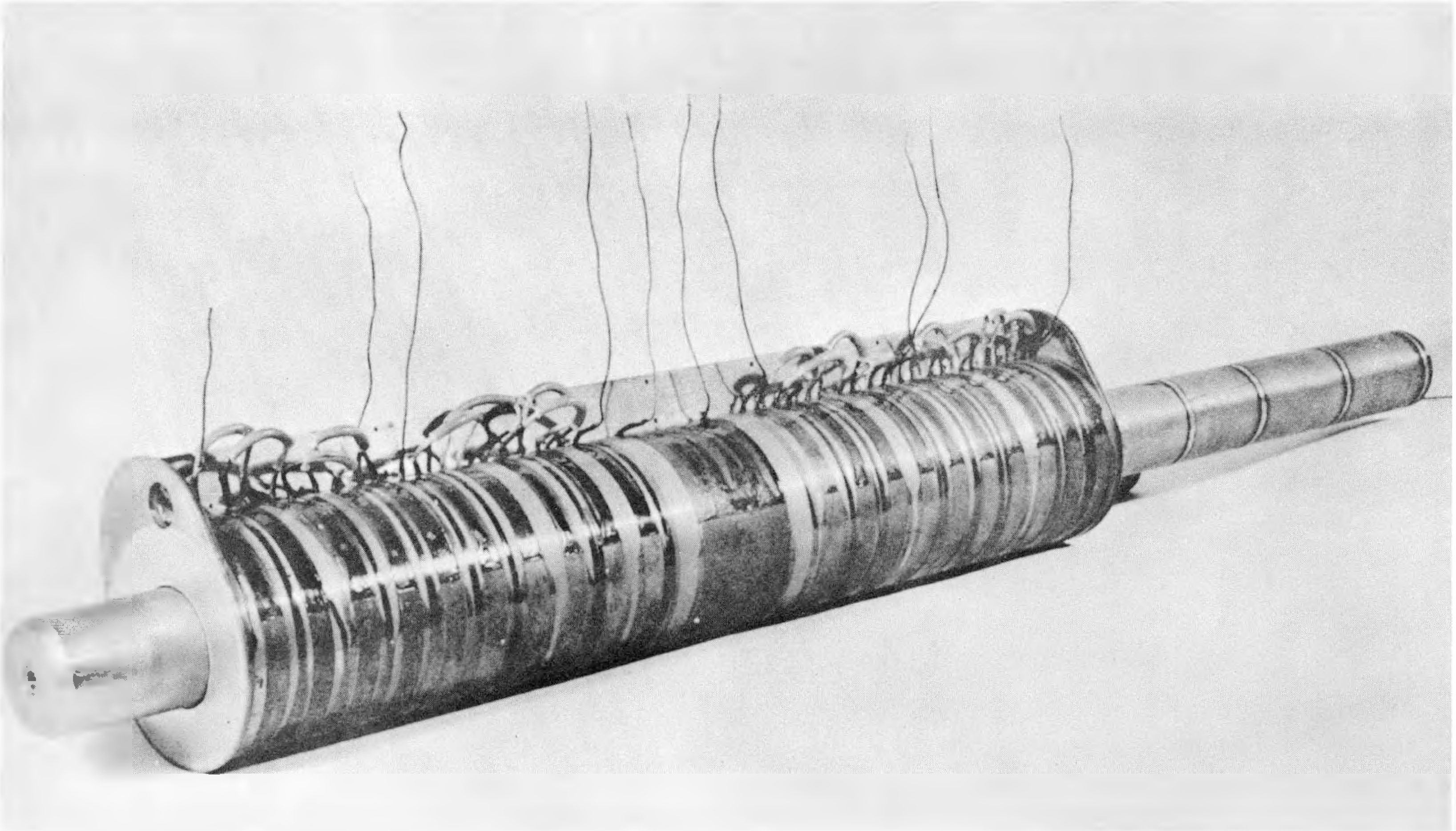


Fig. II-45. Assembled Drive Coil Stack

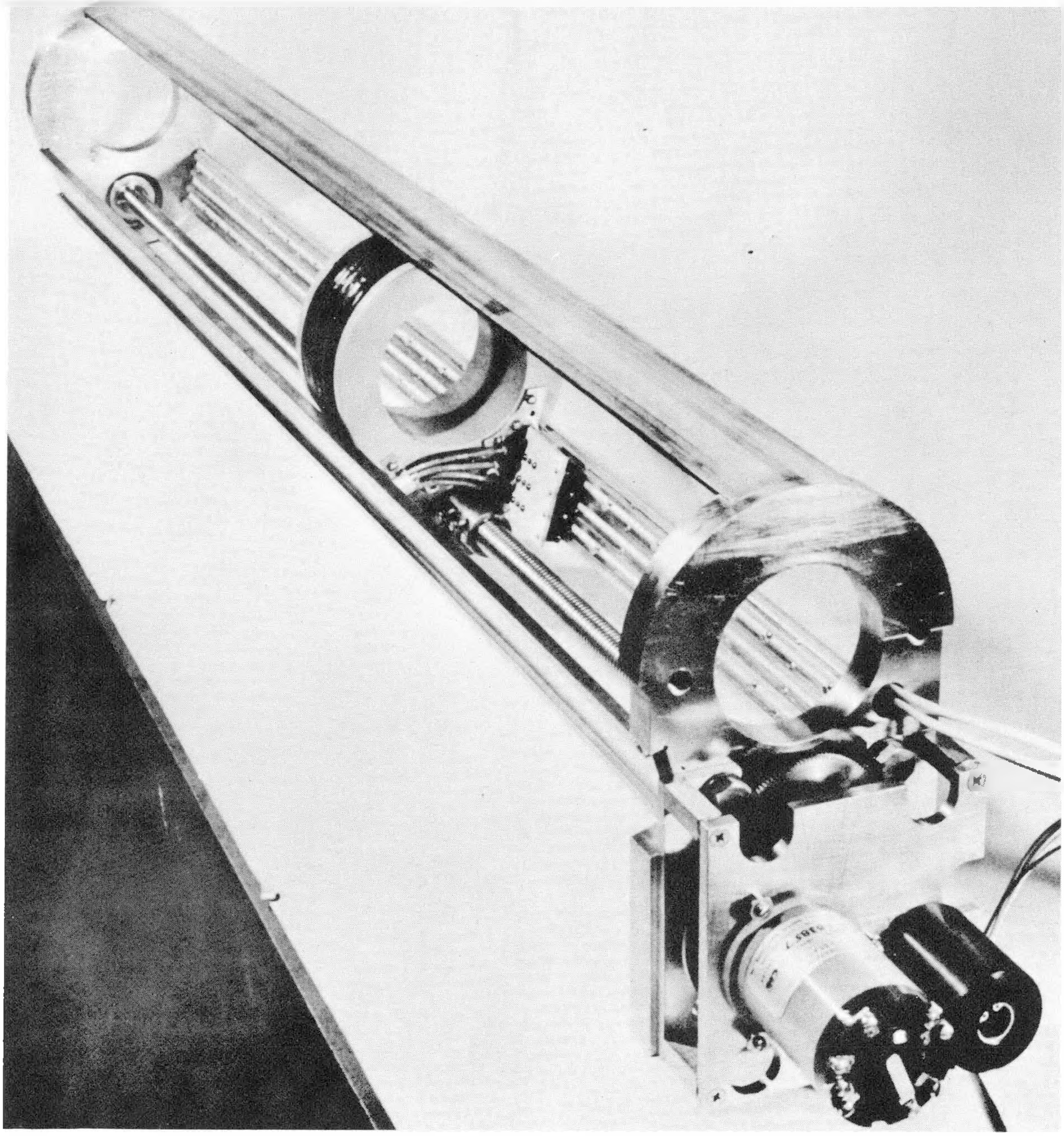


Fig. II-46. Assembled Position Indicator Differential Transformer, Lead Screw, Drive Motor and Synchro Transmitter

III. TASK 4--FINAL DESIGN

Project Engineers--R. Akin, C. Fox, G. Zindler

The objective of this task are to prepare and analyze the final design of the PM-1 Nuclear Power Plant. This was the major effort during the third project quarter and will continue into the fourth quarter. The task is scheduled to be completed in March 1960.

It should be noted that subcontractor efforts contributing to the final design are reported under other tasks mainly for administrative reasons:

Westinghouse Electric Corporation--Secondary System Development, Subtask 1.6

Gibbs and Hill, In.c--Consulting, Task 16.

The organization of this task into 37 subsystems for purposes of final design submission was explained in the preceding Quarterly Progress Report, MND-M-1814. This procedure is continued and progress of work is described by subsystem. A few changes in the names and contents of certain subsystems are noted where appropriate. More general discussions of nuclear analysis, shielding analysis, secondary system and miscellaneous matters follow.

By the end of the quarter, final design submittals had been made for Subsystems 1, 2, 4 through 14, 18 through 22, 24, 27, 28, 30 and 35. Partial submittals were made on Subsystems 17, 25 and 29. The remaining subsystems and any corrections desired by the AEC were scheduled for submittal early in the next project quarter.

For convenience, some procurement and technical liaison events which cross the administrative boundaries of Task 7 (fabrication and assembly of plant) are narrated in this section under the appropriate subsystems.

A. SUBSYSTEM 1--PLANT INSTRUMENTATION AND CONTROL

In this particular case, a more general review is presented under the heading of a single subsystem. The objective is to give a synoptic review of the control and instrumentation system. The review continues in Subsystems 2, 4 and 5.

A brief description of some analog computer studies precedes the review.

1. Computer Studies

a. Loss of coolant flow

An analog study of PM-1 plant scram due to loss of coolant flow was performed. The study investigated the effect on the reactor hot channel outlet temperature of the loss of flow scram set point and the time constant of the instrumentation that determines the actual or relative value of flow. All runs were initiated with the reactor at full power--the severest initial condition.

The differential equation which describes the reactor coolant flow rate under transient conditions was programmed on the analog computer.

Before the flow coastdown in a loop with a heat source could be determined, a model of the heat source was developed. When the reactor is scrammed, there are two sources of heat: heat from fission decay and heat from fission product decay. The heat from fission decay, shown as Curve A in Fig. III-1, was taken from the two-delay group analog model of the reactor kinetics. The fission product decay curve is shown as Curve B. The sum of these 2 curves is Curve C, and Curve D is the analog representation of Curve C.

The reactor coolant flow rate as a function of time is shown in Fig. III-2. Curve A results from neglecting the natural convection term. The flow coastdown considering natural convection is shown in Curve B. A series of runs was performed to determine the hot channel outlet temperature as a function of flow scram set point and flow instrumentation time constant. The procedure followed was to simulate loss of power to the coolant pump, vary the flow scram set point and delay the signal to the scram relay to account for instrumentation lag. The results of these runs are shown in Fig. III-3.

2. Control and Instrumentation Specifications

The following review presents abstracts of the requirement for controls and instrumentation for the PM-1.

Plant Instrumentation and Controls (Subsystem 1)

a. Primary instrumentation and fluid sampling

(1) Reactor temperatures

Reactor coolant outlet temperature. There is one reactor outlet temperature instrument set that measures the coolant temperature leaving the reactor over the range of 40 to 600° F. The set consists of a detector, amplifiers or transmitters, alarm bistable, scram bistables and meters.

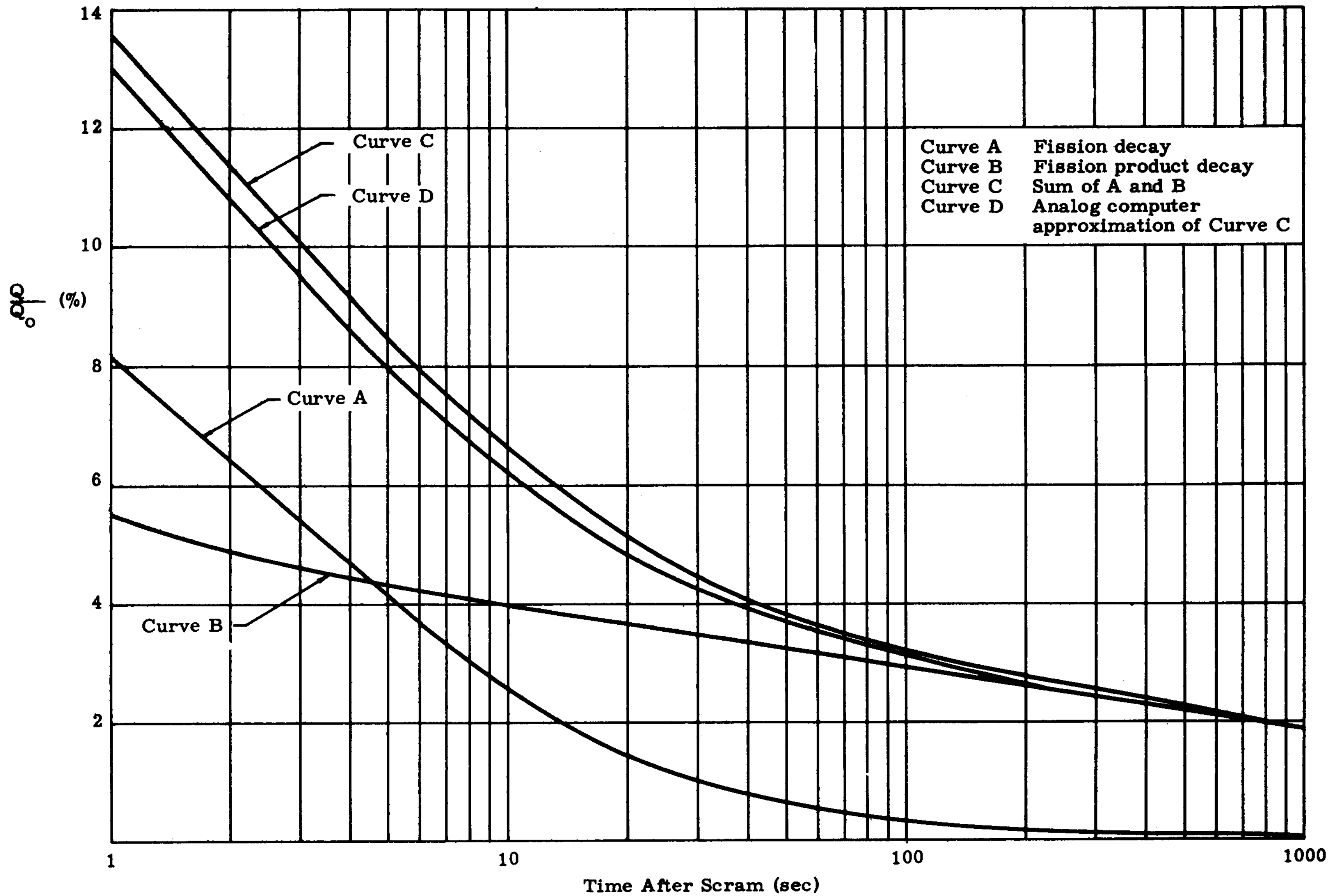


Fig. III-1. PM-1 Reactor Afterheat

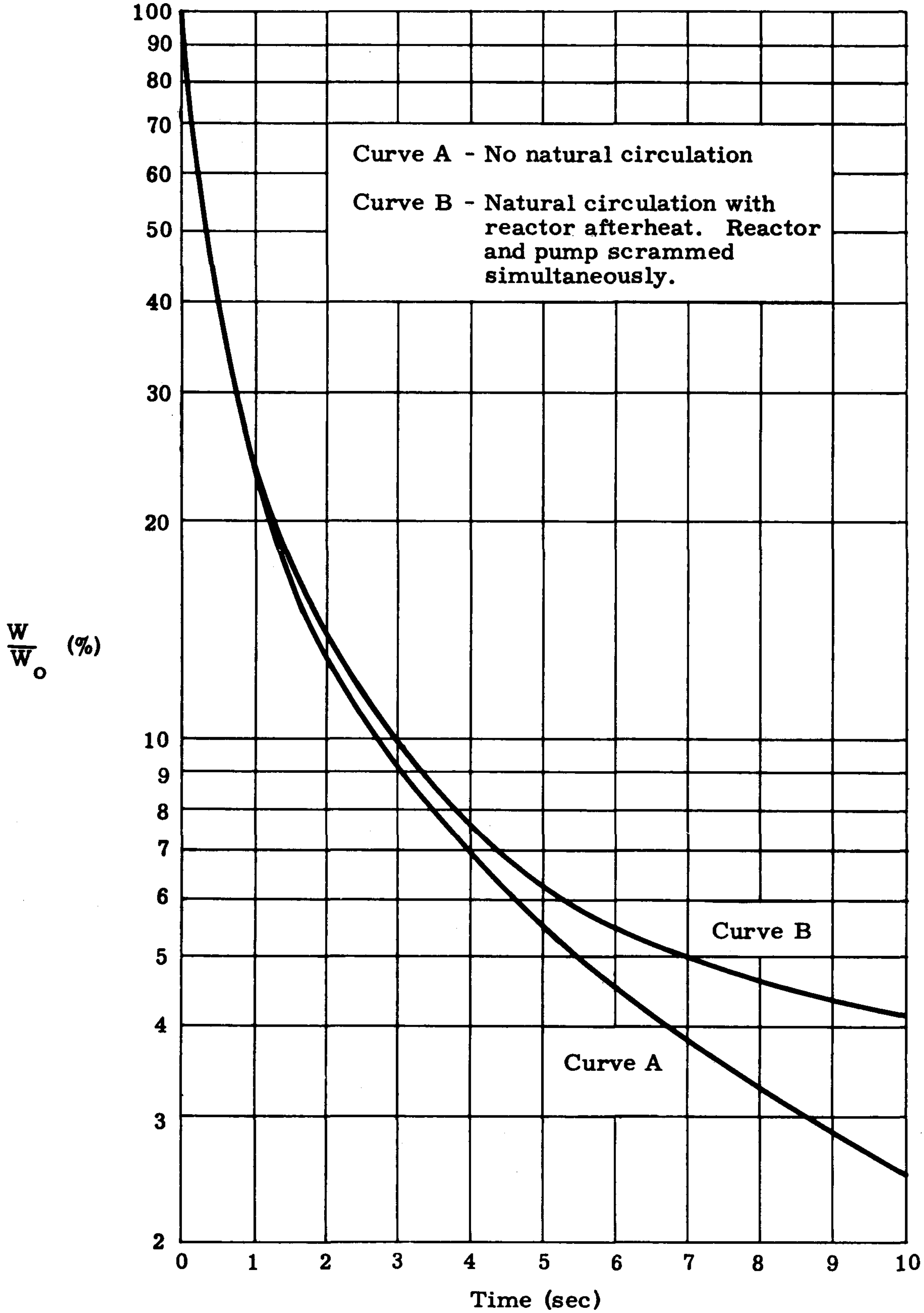


Fig. III-2. PM-1 Flow Coastdown

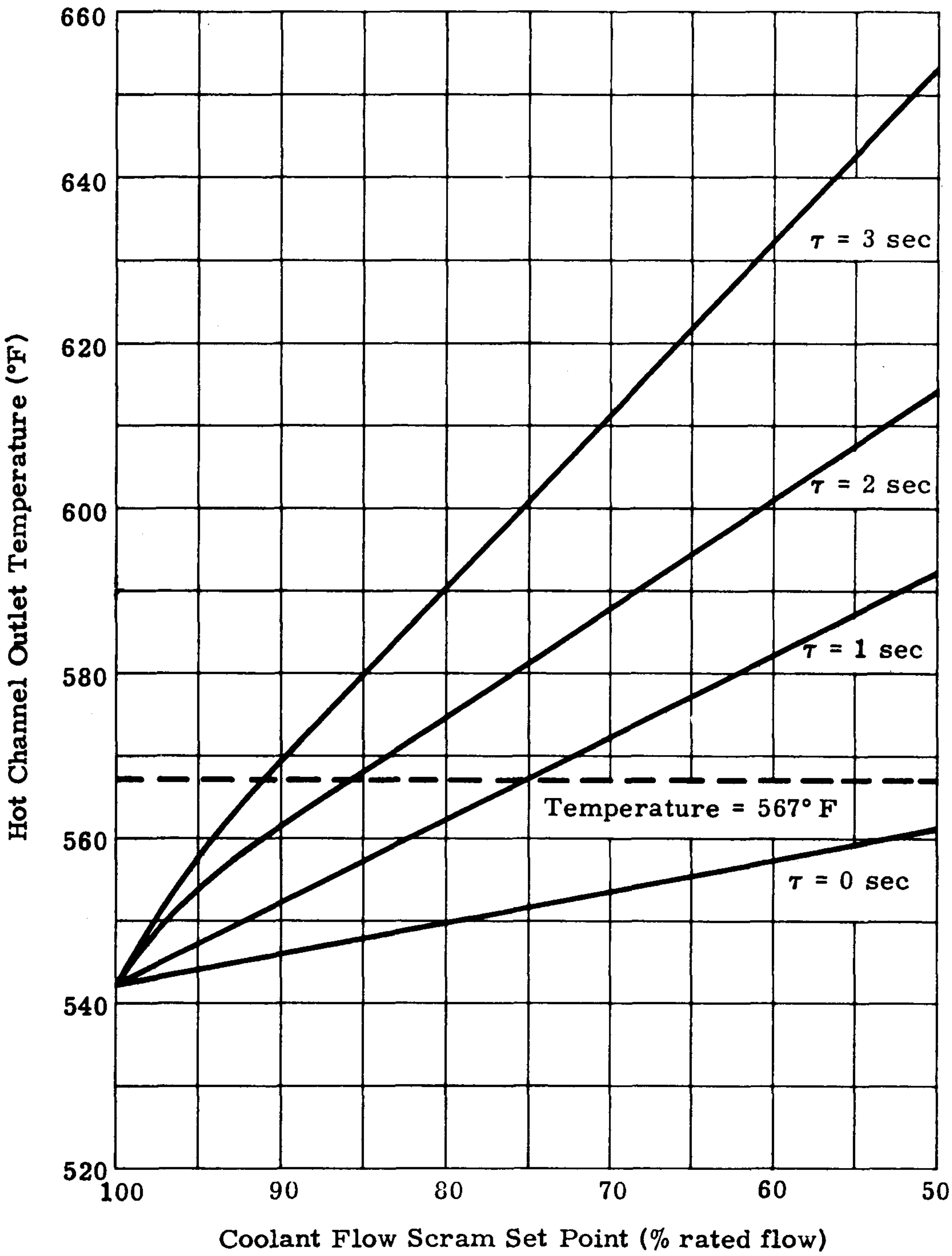


Fig. III-3. Maximum Hot Channel Outlet Temperature as a Function of Coolant Flow Scram Set Point and Instrumentation Time Constant (τ)

Reactor coolant inlet temperature. There is one reactor inlet temperature instrument set which measures the coolant temperature entering the reactor over the range of 40 to 600° F. The set consists of a detector, transmitters and meters.

Average reactor coolant temperature (T_{ave}). There is one average reactor coolant temperature instrument set which computes the average temperature by summing the outputs of the inlet and outlet expanded scale amplifiers and dividing by 2. This temperature is also recorded on the outlet temperature meter.

(2) Primary system pressurizer

Pressurizer pressure. There is one instrument set that measures pressurizer pressure over the range of 0 to 1600 psia. The set contains 2 detectors, 2 amplifiers, a control switch, alarm bistable, 3 control bistables, 2 scram bistables, an indicating meter, a recorder, a deviation alarm, a heater and spray valve bistables.

Pressurizer water level. There is one instrument set which measures the water level in the pressurizer over a 34.5-inch range. This set consists of 2 detector amplifiers, 2 high-level alarms, 2 low level alarms, 2 heater cutoff alarms, a switch and an indicator.

Level calibration unit. There will be 2 instrument sets used for remote calibration of the pressurizer level instrument sets. Each set consists of 2 detectors. A 4-position switch will be used to supply a common amplifier and indicator.

(3) Expansion tank pressure.

There is one instrument set that measures expansion tank pressures over the range of 0 to 600 psia. The set contains a detector, amplifier, alarm bistable and 2 meters.

(4) Primary pump Δp

Reactor coolant flow (pump Δp). There is one differential pressure instrument set that measures the Δp across the coolant pump over the range of 0 to 35 psi. The set contains a detector, scram bistable and meter.

(5) Primary pump temperatures

Reactor coolant pump temperatures. There shall be 3 temperature instrument sets that measure pump bearing temperature and stator winding temperature and the pump coolant temperature over the range of 50 to 500° F. The instrument sets shall be composed of detector, amplifier, alarm bistable, switch and a common meter.

(6) Primary system leakage detection

There will be 4 leak detectors, one located at each flanged fitting in the 6-inch primary piping. Each leak detector will be equipped with a level probe and transmitter to give a positive indication of system leakage. The instrument set will be composed of 4 level probes, 4 transmitters, 1 selector switch and 1 alarm unit.

(7) Shield water temperature

Shield water temperature, air blast cooler outlet. There will will be one instrument set that measures shield water temperature as it leaves the air blast cooler. The temperature measurement is to be made over the range of 40 to 140° F. The set consists of a detector, amplifiers, alarm bistable and meter. A control amplifier will be used for controlling the flow of outside air to the coolers. This controller will position an air recirculation door and thus control the air blast cooler. This door regulates the amount of cool outside air that is brought into the air blast cooler package and the amount of recirculated air. The door will be positioned in proportion to the difference between the shield water outlet temperature and a reference temperature of 106° F. When the outlet temperature equals the reference temperature, the recirculation door will be closed and only outside air used for cooling. As the outlet temperature drops, the recirculation door will open and allow recirculation of air. The temperature regulation is to be adjustable.

(8) Shield water level

Shield water tank water level. One instrument set will be provided to measure the water level in the shield water tank over the range of 0 to 18 inches. The set will be composed of a detector, duplex alarm bistable and meter.

(9) Shield water flow

Pressure drop across shield water-cooled components. There is to be one instrument set that measures the pressure drop across the paralleled components cooled by shield water. The range will be 0 to 30 psi. The instrument set will be composed of a detector, interlock bistable, alarm and control bistable, and a meter.

(10) Shield water component temperatures

Shield water temperature, outlet of cooled components. There will be one temperature instrument set that measures the shield water temperature at the outlet of 3 of the cooled components over the range of 40 to 140° F. The instrument set consists of 3 detectors with transmitters, a switch, amplifier and meter.

(11) Cooler outlet temperature (high pressure demineralizer inlet temperature)

Cooler outlet temperature. There will be one instrument set that measures the temperature of the water leaving the demineralizer cooler over the range of 40 to 140° F. The set consists of a detector, amplifier, alarm bistable and meter.

(12) Sump tank level

Sump tank level. There is to be one instrument set that measures the liquid level in the sump tank over the range of 0 to 60 inches. The set will consist of a detector, alarm bistable and meter.

(13) Waste storage tank level

Waste storage tank level. There is one instrument set that measures the liquid level in the waste storage tank. The set consists of 4 detectors and transmitters and a meter.

(14) Primary system water analysis

A system for measuring:

(a) The conductivity of the inlet and outlet of both the high and low pressure demineralizers.

(b) The pH at the outlet of the low pressure demineralizer.

(c) An O_2 cell with portable measuring device will be provided.

The system consists of in-line sample cells and a multipoint recorder.

b. Secondary instrumentation and fluid sampling

(1) Feedwater control equipment

One 3-element feedwater control system will be furnished. The system will measure main steam flow, feedwater flow and steam generator level, and will position the feedwater flow regulating valve so as to maintain the feedwater flow steam flow balance. The system will be arranged to operate with a lagging feedwater flow when under automatic operation.

Under remote manual operation, the feed lag provision will be cut out. The system will be arranged to integrate with time the amount which the control is off the set point and to readjust to correct for same (integral-error control). Provision will be made for programming of the water level with load.

(2) Deaerator level control equipment

One system of controls will be furnished for maintaining deaerator level. The equipment provided will measure the level of water in the deaerator and transmit a signal which will position various valves and thereby control the water level.

The controls will function as follows:

(a) Condensate pumped from the condenser hotwell will enter the deaerator through the condenser hotwell level control valve. Should the deaerator level rise to 2 inches above normal, the deaerator bypass valve will start to open and the condensate will discharge to the condensate storage tank. This valve will be wide open at a level 4 inches above normal.

(b) If the water should continue to rise to 6 inches above the normal level, a deaerator overflow valve will open to allow the water to overflow to the condensate storage tank. This valve will close when the level returns to 4 inches above normal.

(c) Should the deaerator level fall to 6 inches below normal, a valve located in the line leading from the condensate storage tank to the condenser hotwell will open to supply water to the condenser hotwell. This valve will close when the deaerator level rises to within 2 inches of the deaerator normal water level.

(3) Condenser hotwell level control equipment

One system of controls will be furnished for maintaining condenser hotwell level. The equipment provided will measure the level of water in the condenser hotwell and position a valve in the condensate pump discharge line. The system will control the level to within 2 inches above or below this level.

The control will function to open the condensate discharge valve with rising level and to close the valve with falling level.

(4) Condenser louvers controls

One system of controls will be furnished for the 2 air-cooled steam condensers.

The control system shall function as follows:

(a) The fans will be started manually and stopped by means of panel-mounted control switches.

(b) The air cooler coil louvers will be modulated individually, automatically, through a temperature-sensing element set at 150° F and measuring the temperature in the air offtake line to the air ejector. Each louver will have a manual-automatic selector station.

(c) All other louvers will be controlled by the turbine steam exhaust line (back-pressure) pressure set at 9 inches Hg absolute. Each condenser will have one manual-automatic selector station for the east side louvers and one for the west side louvers. The louvers will be controlled so that they will open sequentially in both the manual or the automatic position. The sequence will be: Louver 1, then 2, for the west louvers and Louver 3, then 4, for the east louvers. In the fully automatic position, louvers 1 and 3 open together and then 2 and 4 open together; with both condensers in operation, all louvers modulate in the sequence indicated. Louver modulation (not 2-position) is required. Local manual control will be provided.

(d) All operators will have position feedback to indicate actual position of louver operators.

(5) Condensate storage tank level equipment

The equipment required will consist of one level measuring device and transmitter for signaling high and low tank level and for level indication.

(6) Evaporator pressure control equipment

One system of controls will be furnished for maintaining evaporator pressure. The equipment provided will maintain evaporator vapor pressure within 2 psi of the set pressure of 35 psia.

The system will operate so that a signal from the pressure transmitter on the evaporator vapor line will act to position the evaporator steam control valve.

(7) Deaerator pressure control equipment

One system of controls will be furnished for maintaining deaerator pressure. The equipment provided will maintain deaerator pressure within 1/2 psi of the set pressure of 12 psig (23.4 psia).

The normal source of steam for pegging deaerator pressure is that used for plant make-up and deaeration which comes from the evaporator at 35 psia. In the event of loss of this steam source, a backup pegging line is provided from the main steam line. This steam line operates at a pressure of 125 psia. The primary heat source to the deaerator is from the evaporator coil drain.

The system will operate so that a signal from the pressure transmitter on the deaerator acts to position the normal pegging steam line valve set to maintain the deaerator at 12 psig. Should the pressure rise to 12.5 psig, the evaporator coil drain valve will open, relieving the pressure and allowing this drain to flow to the exhaust steam line.

(8) Desuperheating system

A system will be provided for desuperheating the steam produced by reactor decay heat when the turbine throttle is tripped. This system is required for protection of the aluminum condenser tubes which must not be subjected to a temperature in excess of 220° F. The system will be designed for a steam flow of 1700 pounds per hour of 480-psia saturated steam discharging directly into an air-cooled condenser, normally maintained at 9 inches Hg absolute. A mechanical (flooding) type desuperheater will be used.

Water for desuperheating will be available from the condensate pump discharge at a pressure of about 40 psig and a temperature of 157° F.

The steam valve will be arranged for remote manual actuation only. The water valve will open in combination with the steam valve, but will be controlled from a temperature-sensing element located ahead of the condenser in the pipe line. An alarm will be provided to signal the high condensing temperature of 215° F.

(9) Feedwater pump antiflash protection

A system of controls will be furnished for the feedwater pump antiflash protection.

This system will supply condensate to the feedwater pump suction in the event of a drop of deaerator pressure to 4 psig. This will be accomplished through the use of a pressure switch controlling a solenoid valve (on-off). This switch will be mounted on the deaerator.

(10) Conductivity instrumentation

One system of instrumentation for measuring conductivity of water and steam samples will be furnished. A continuous record of the electrical conductivity of samples of the following will be provided:

Location	Operating Conditions
(a) Steam leaving the steam generator	290 to 480 psia, sat.
(b) Vapor leaving the evaporator	35 psia, sat.
(c) Steam generator blowdown	290 to 480 psia, sat.
(d) Evaporator shell blowdown	35 psia, sat.

(11) pH instrumentation

One system of instrumentation for measuring pH of water and steam samples will be furnished. A continuous record of pH samples of the following will be provided:

Point No.	Location	Operating Conditions
1	Evaporator vapor	35 psia, sat.
2	Steam generator blowdown	290 to 480 psia, sat.
3	Condensate leaving the hotwell	60 psia, 168° F

(12) Backup steam generator water level

There will be one set of steam generator level backup instrumentation. This set will be identical to that used for the 3-element feedwater controller. This set will consist of a pressure-compensated level transmitter with power supply and amplifier. The outputs of the transmitter will be routed through a switch and to eiher the controller or an indicator.

(13) Throttle steam system

There will be one throttle steam set of instrumentation to measure and record flow and to indicate throttle steam pressure. The measurement will be pressure-compensated. The steam flow signal will be transmitted to a panel-mounted recorder. The instrument will have a flow measuring mechanism of the square root type to permit recording on a uniformly graduated chart over a range of 0 to 40,000 pounds per hour.

The pressure measuring equipment will consist of a pressure transmitter and indicator. Range of measurement shall be 0 to 500 psia.

(14) Temperature scanning system

This system will consist of 23 iron-constantan thermocouple assemblies and protecting tubes with the appropriate selector switches, alarms and indicators.

Locations:

Point No.	Location
1	Ambient air entering condenser
2	Air leaving condenser east
3	Air leaving condenser west
4	Air leaving condenser air cooling section north (East Condenser)*
5	Air leaving condenser air cooling section south (East Condenser)
6	Air leaving condenser air cooling section north (West Condenser)
7	Air leaving condenser air cooling section south (West Condenser)
8	Extraction steam
9	Main steam
10	Turbine exhaust
11	Spare
12	Condensate leaving air ejector
13	Feedwater leaving deaerator
14	Feedwater leaving heater
15	Oil entering oil cooler
16	Oil leaving oil cooler
17	Oil leaving Bearing Position No. 1
18	Oil leaving Bearing Position No. 2
19	Oil leaving Bearing Position No. 3
20	Oil leaving Bearing Position No. 4

Point No.	Location
21	Oil leaving Bearing Position No. 5
22	Oil leaving Bearing Position No. 6
23	Oil leaving Bearing Position No. 7

*Compass directions refer to the plant layout planned for the Sundance site.

Points 16 through 23 inclusive shall be equipped with one adjustable common alarm contact to "make" on high temperature.

(15) Annunciator system

This system will supply the necessary visual and audible indications to warn the operator of off-normal plant operating conditions and to aid in locating the fault. The system will consist of 72 annunciator points. One common alarm horn will sound whenever any of the preset values for annunciation is exceeded.

Two sets of annunciators will be arranged in 6 rows with 6 annunciators per row. Each annunciated point will have a white translucent backlighted nameplate and white indicator light. Each annunciator, with nameplate, will be 1 inch high and 1-1/2 inches wide. The annunciators will be plug-in modules so that a unit can be replaced simply by withdrawing it and inserting another.

When an annunciated condition occurs, the light will flash and the alarm horn will sound. An acknowledge button, when pressed, will silence the horn and convert the flashing light to a steady glowing light. The system will reset manually when the process condition(s) annunciated return to normal. A panel light test button will be provided for testing all lights simultaneously.

The nameplates and points are listed below.

(16) Intercommunication system

This system will provide for intercommunication within the PM-1 power plant. The system will consist of 1 master station located on the control room console and 5 substations located in the following general areas:

(a) Primary building

(b) Decontamination building

(c) Heat transfer area

(d) Maintenance area

(e) Office area

The substations can "talk back" to each other while being overheard by the master station. The master station will be capable of communication with any number of substations at the same time. The substations can contact the master station by activating a "call" button circuit. An "all stations" key will be provided to call all areas at once.

B. SUBSYSTEM 2--NUCLEAR INSTRUMENTATION SYSTEM

The nuclear instrumentation system will consist of 2 source range channels, 2 intermediate channels and 3 power range channels. The system is designed to provide thermal neutron flux level and reactor period information over a range from 0.25 to 2.5×10^{10} nv in 3 overlapping ranges. There will be, as a minimum, 1 decade of sensitivity between the source and intermediate ranges and between the intermediate and power ranges.

The nuclear instrumentation system will perform the following functions.

<u>Source range.</u> Provide source range flux and period information over a range of flux levels from 0.25 to 2.5×10^4 nv. Two separate channels will be provided for this measurement. Indication will be provided locally on front-panel mounted meters from each channel. In addition, recorders and selector switches will be provided to permit recording the output of either channel. The source range channels designed for this application incorporate a preamplifier as a precautionary measure against any possible effects of radio or radar interference etc. The 2 source range channels will provide output for the following functions:

(1) A safety signal to scram circuit when the source range channels indicate a positive period less than a preset value.

(2) A signal to the rod withdrawal and annunciator system when the source range channels indicate a positive period less than a preset value.

(3) A signal to the rod withdrawal interlock channel when the source range channels indicate a log count rate less than a preset value.

Intermediate range. Provide intermediate range flux and period information over the range of flux values from 2.5×10^3 to 2.5×10^{10} nv. Two separate channels will be provided for this measurement. Indication will be provided locally on front panel-mounted meters from each instrument system. In addition, recorders and selector switches will be provided to permit recording the outputs of either channel. The two intermediate range channels will provide outputs for the following functions:

(1) A signal to the safety system when the intermediate range channels indicate a positive period of less than a preset value.

(2) A signal to the rod withdrawal and annunciator system when the intermediate range channels indicate a positive period of less than a preset value.

(3) A means of automatically removing the source range detector high voltage and prohibiting safety system inputs from the source range channels when the intermediate flux level exceeds a preset value.

Power range. Three separate power range channels will provide linear power information over a minimum range of 2 decades. The corresponding flux range will be 2.5×10^8 to 2.5×10^{10} nv. The 3 channels will have individual panel-mounted indicators. In addition, a recorder and selector switch will be provided to permit recording the output of any channel. The 3 power range channels will provide outputs for the following functions:

(1) A signal to the safety system whenever any 2 of the 3 channels indicate a flux level above a preset limit.

(2) A signal to prohibit period scram from the intermediate channel.

(3) A signal to the annunciator system whenever any 1 of the 3 power channels indicate a flux level above a preset limit.

Fault Location Unit. This unit will provide a rapid means of locating a faulty component or module within an instrumentation channel of the nuclear instrumentation system. The unit will be self-contained and portable.

The input signal(s) will be introduced to the Fault Location Unit through a single cable which will connect to the individual instrument channel or to an instrument drawer.

The output of the Fault Location Unit will be displayed on a meter on the panel of the unit.

Operating instructions shall be provided with the unit in appropriate form to permit rapid operation of the unit by semiskilled personnel.

C. SUBSYSTEM 3--REACTOR ROD CONTROL SUBSYSTEM

See Subtask 2.5, The Actuator Program.

D. SUBSYSTEM 4--REACTOR SAFETY SYSTEM

The reactor safety system will consist of the necessary logic elements and an output mixer driver to perform the following safety actions:

Reactor "scram." Upon receipt of input signals from other measurement instrumentation channels of the overall PM-1 controls and instrumentation system, the reactor safety system will transmit "scram" signals to initiate the following actions:

(1) Automatic rapid insert of all reactor control rods. This will be accomplished by removing an output signal of +10 volts (10 ma) normally directed to the reactor rod control system. Under scram conditions, this output signal from the reactor safety system shall change to 0.

(2) Deny power to the primary coolant pump. This will be accomplished by the reactor safety system opening a pair of normally closed relay contacts, thereby causing the primary coolant pump motor starter to shut down the pump.

(3) Deny power to the main steam stop valve actuator. This will be accomplished by the opening of a pair of normally closed relay contacts in the reactor safety system, causing the main steam stop valve actuator to close the valve.

Reactor "hold." Upon receipt of signals from other measurement instrumentation of the overall PM-1 controls and instrumentation system, the reactor safety system will transmit a "hold" signal to the reactor rod control system. The hold signal will prevent control rod withdrawal and, therefore, prevent reactor startup.

The reactor safety system will initiate "scram action" as a result of the following input signals:

(1) A short, positive period signal from both source range channels unless the channels are locked out by the intermediate range interlock.

(2) A short, positive period signal from both intermediate range channels unless the channel period trip is locked out by the power range period lockout circuit.

(3) A high flux level in excess of a preset value on 2 of the 3 power range channels.

(4) A high reactor outlet temperature signal from both the expanded and full range measurement channels.

(5) A low primary coolant flow signal from either the differential pressure measuring channel across the primary coolant pump or an abnormal pump current signal unless these signals are bypassed for startup.

(6) A low primary pressurizer pressure signal from both the full and expanded range measurements unless these signals are bypassed for startup.

(7) Manual scram signal from a switch on the control console.

The reactor safety system will initiate "hold action" as a result of the following input signals:

(1) A low count rate signal from both source range channels.

(2) A low water level signal from the steam-generator level channel.

E. SUBSYSTEM 5--RADIATION MONITORING SYSTEM

The radiation monitoring system is designed to provide the necessary equipment, other than normal survey equipment, dosimeters and film badges for adequate indication of the radiation levels within the PM-1 plant. This equipment is required to help protect personnel from excessive radiation and to aid in indicating proper operation of the plant.

The system will have the following functions:

(1) To monitor the steam-generator water radioactivity to determine possible leakages of primary fluid into the secondary system.

(2) To monitor the water leaving the waste disposal system condenser to determine if the activity is low enough to be placed in the secondary storage tank or has to be recirculated through the waste disposal system.

(3) To monitor the radioactive gases going to the shield water cooler exhaust.

(4) To provide a portable, continuous indication for beta-gamma particulate activity.

(5) To monitor selected areas for gross gamma activity.

(6) To obtain air samples for integrated particulate activity.

(7) To provide counting equipment for the analysis of prepared samples.

The activity measured by the area monitors, fluid monitors and off-gas monitor are indicated, alarmed and recorded in the control room. Each channel has a built-in check source and an adjustable alarm set point. Both an audible and visual alarm actuate on high radiation level. The audible alarm is silenced by the operator while the indicating light remains energized until the condition causing the alarm has been corrected.

The continuous air monitor will be a self-contained, moving filter paper-type instrument. It is capable of being moved to plant areas where particulate radiation is expected and will indicate, alarm and record the activity measured.

The air samplers will be positioned at various points throughout the plant area to obtain an integrated particulate activity. The sample obtained will be counted by the lab counting equipment.

The lab counting equipment will consist of the required detectors, indicators, recorders and miscellaneous equipment to measure the activity of prepared samples. Gamma spectrometry can be performed to aid in the identification of various radioactive isotopes.

1. Anticipated Accomplishments (Subsystems 1, 2, 4 and 5)

Final vendor selection, contract award and instrumentation liaison are the only remaining efforts.

F. SUBSYSTEM 6--REACTOR COOLANT SYSTEM

1. Thermal and Hydraulic Supporting Analysis

C. Smith, S. Frank, R. Baer, A. Carnesale

The supporting analyses performed during the fourth project quarter were concerned with 3 main areas:

(1) Review of the reactor coolant flow rate.

(2) Thermal and hydraulic analysis of reactor internals.

(3) Design of the inlet water box orifice plate.

a. Reactor coolant flow rate

As reported in the Third Quarterly Progress Report (MND-M-1814), the core previously was composed of 725 active fuel elements, 18 dummy tubes and 77 burnable poison rods. The core design has since been revised and now has the following composition:

732 active fuel elements

21 dummy rods

75 burnable poison rods

828 elements (total)

The change from 725 to 732 active fuel elements reduces the flow requirement per element by 1.1%.

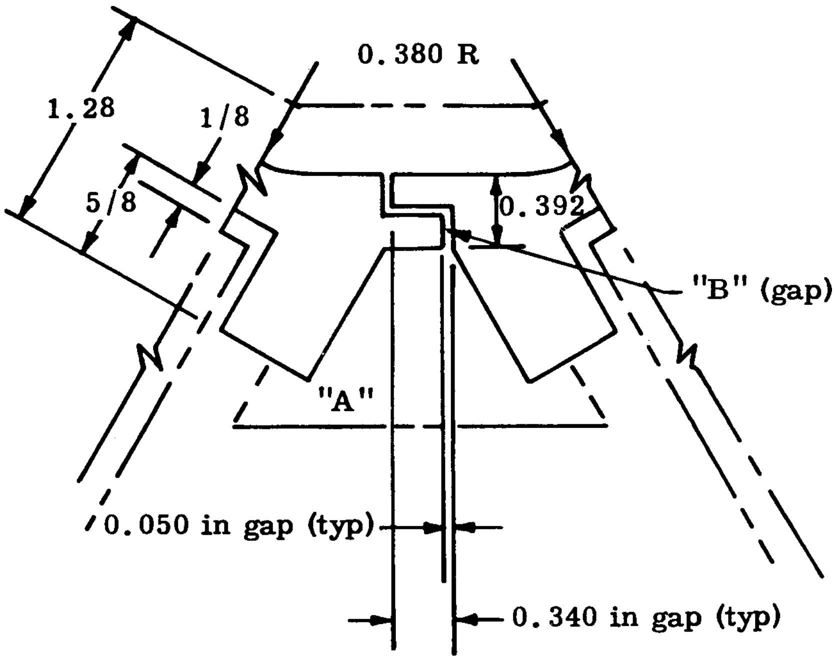
However, manufacturing limitations have required that the tolerance of fuel concentration in a fuel element be increased from ±2% to ±4%. This change necessitates a linear increase in the flow requirement per element by 1.96% which, when combined with the previous correction, results in a net increase in the flow rate requirement from 2.185 to 2.205 gpm per element.

Several changes in the flow areas in the core outside the fuel elements were also made. The control rod guides were redesigned and are shown in Fig. III-4. The baffle, on the periphery of the core, was replaced by twenty four 1/2-inch diameter rods. Employing the flow requirement of 2.205 gpm per fuel element with 52% of the flow outside the elements (828 in all) and the above quoted data, the required flow outside the elements was calculated to be 1283.3 gpm.

The required flow rate inside the elements was calculated to be 775.7 gpm. This gives a total required flow rate of 2059 gpm.

The required flow rate is that which is necessary to prevent bulk boiling at the outlet of the hottest fuel element concurrent with the following:

Sections 6 and 7



Section 5

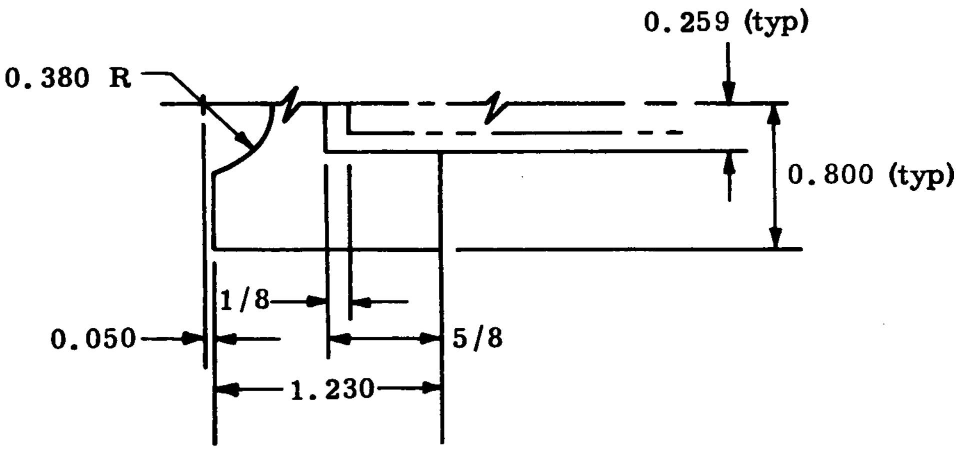


Fig. III-4. Control Rod Guides

(1) All hot channel factors acting adversely.

(2) Reactor power at 120% of design.

(3) System pressure at 1200 psia.

Since the reactor coolant system pump is designed to yield a flow rate of 2125 gpm, no change in the pump specification is required.

b. Thermal and hydraulic analyses of reactor internals

During the past quarter, thermal and hydraulic analyses were performed concerning the following areas:

(1) Bottom head of pressure vessel.

(2) Top head of pressure vessel.

(3) Control rods.

(4) Inlet nozzle of pressure vessel.

Bottom head. The variation of volumetric heat generation in the bottom head was caluclated. Since this varied considerably with location along the head, the possibility of a thermal stress problem was investigated.

It was found that temperature variations in this area were principally a function of the variation of film coefficient. A conservative estimate of the temperature variations was made and it was found that the thermal stresses were within the limits allowed by the ASME unfired vessel code.

Top head. A method of cooling the top head was developed and analyzed.

The coolant leaving the core divides into 3 flow paths, namely:

(1) Up the control rod guide tubes to the plenum above the upper skirt assembly.

(2) Up the center bundle hold-down tube into the plenum above the upper skirt assembly.

(3) Through the holes in the upper skirt into the outlet water box.

The coolant flowing in paths (1) and (2) combine in the plenum. This fluid then flows through the restriction between the upper skirt assembly and the pressure vessel and combines with the fluid flowing in path (3) above.

The flow area between the center bundle hold-down tube and the plenum was sized to be equal to that within the tube. This results in a low flow resistance in flow path (2). Equations relating the following parameters were then written:

(1) Head loss in path (1) as a function of flow rate in path (1).

(2) Head loss in path (2) as a function of flow rate in path (2).

(3) Head loss in path (3) as a function of flow rate in path (3).

(4) Head loss across the restriction between the upper skirt assembly and pressure vessel as a function of the total flow rate in paths (1) and (2).

(5) The total flow rates in paths (1), (2) and (3) equals the total system flow rate.

(6) The head loss in Equations (1) or (2) above are equal.

(7) The sum of the head loss in Equations (1) or (2) and (4) equals the head loss in Equation (3).

This results in seven equations with seven unknowns. These were solved and it was found that 5% of the total system flow goes up the center bundle hold-down tube. This is more than adequate to cool the top head.

Control rods. The variation of heat generation in the control rod blades was calculated. Conservative estimates of the resultant temperature distribution in the blades were made. These are currently being analyzed to determine the magnitude of the thermal stress and bowing.

Inlet nozzle. The temperature distribution in the reactor vessel inlet nozzle was determined in order that the thermal stresses could be calculated.

The conservative assumption of unidirectional heat flow was made. An iterative procedure was employed to determine the direction and temperature distribution along flux lines based on equal maximum temperatures occurring along the insulated boundary with all the heat conducted in only one direction. The resultant temperature distribution is shown in Fig. III-5. The results are accurate in the region of the pressure vessel wall where the constant temperature lines are parallel. In the other regions, the results should be conservative.

c. Design of water box orifice plate

The tests performed on the 1/4-scale mode of the PM-1 reactor vessel and internals showed that the flow distribution in the thermal

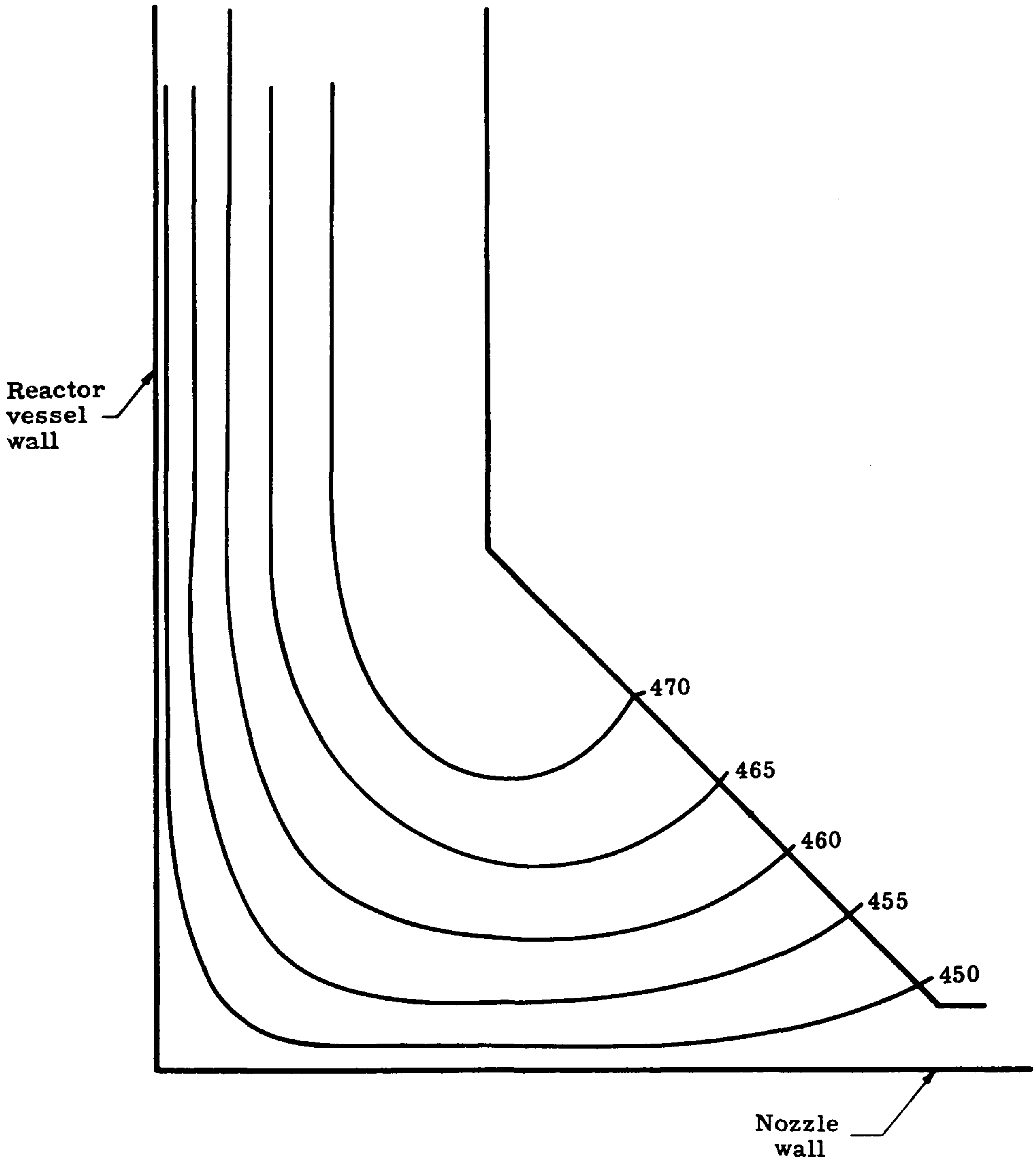


Fig. III-5. Reactor Vessel Inlet Nozzle Temperature Distribution shield region was rather nonuniform. Further testing with a water box orifice plate having a greater hydraulic resistance showed that the distribution was improved considerably. However, the system head loss was increased significantly. Therefore, it was decided to design a water box orifice plate having a hydraulic resistance which varied with peripheral location.

It was decided that the easiest configuration to manufacture would result if:

(1) A constant hole size were used.

(2) The radial location of the holes were kept constant.

(3) The variable hydraulic resistance were obtained by varying the angular spacing between holes.

TABLE III-1

Tabulation of Angles

Row Number	Angle from Centerline of Inlet Pipe (degrees of arc)
0	0
1	3.5
2	7.0
3	10.5
4	14.0
5	17.5
6	21.0
7	24.5
8	28.2
9	31.9
10	35.7
11	39.7
12	43.8
13	48.1
14	52.4
15	56.9
16	61.5
17	66.2
18	70.9
19	75.6
20	80.4
21	85.2

TABLE III-1 (continued)

Row Number	Angle from Centerline of Inlet Pipe (degrees of arc)
22	90.0
23	94.9
24	99.8
25	104.8
26	109.9
27	115
28	120
29	125
30	130
31	135
32	140
33	145
34	150
35	155
36	160
37	165
38	170
39	175
40	180

As a result of this, it was decided to use 4 holes, each 1/2 inch in diameter on each radial line, with the angular spacing between radial lines to be determined by analysis (see Fig. III-6).

The final design of the water box orifice plate is based on the following information obtained from the 1/4-scale tests:

(1) The angular distribution of static pressure in the water box remained essentially constant, with 2 orifice plates having greatly different hydraulic resistance.

(2) The head loss across the orifice plate was a function only of the water box configuration and the velocity through the plate. It was not influenced by peripheral location, hole size or total flow rate.

(3) The angular distribution of flow corresponds very closely with that which would be predicted based on the information contained in items (1) and (2) above.

The head loss across the water box orifice plate below the inlet was set equal to that obtained in the 1/4-scale test run between 18 December 1959 and 11 January 1960. Therefore, it is anticipated that the reactor head loss will be equal to that obtained in this test, namely, 24.2 feet. This is 3.5 feet less than that previously calculated.

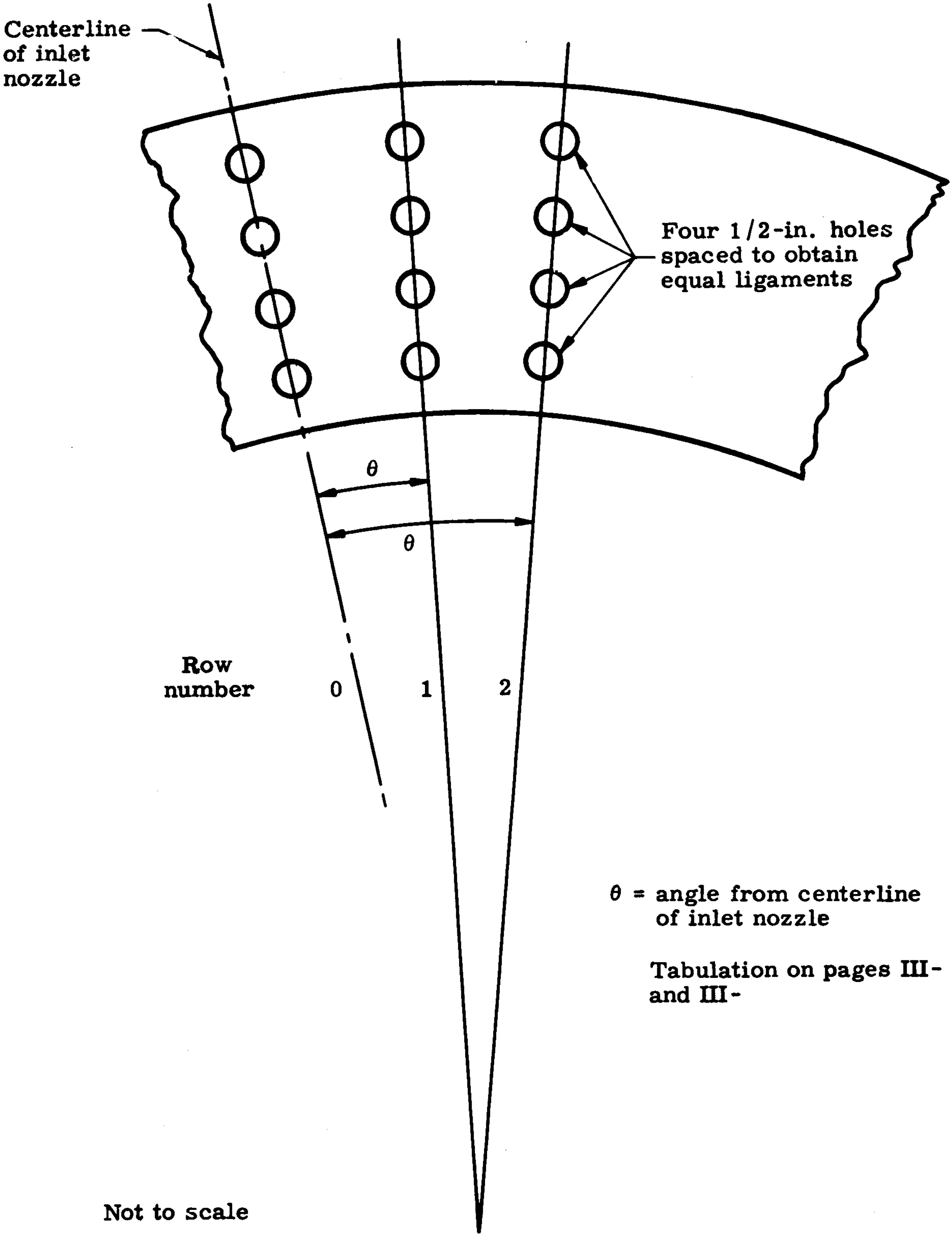


Fig. III-6. Orifice Plate (top view)

Using the above information, the angular spacing between rows of holes was calculated so as to result in equal peripheral flow distribution and static pressure below the orifice plate.

The final configuration will consist of 80 rows, located symmetrically about the inlet.

2. Supporting System Analysis Studies

J. Beam, R. Baer

All the system parameters had been established prior to the past quarter. The only additional analysis performed during the fourth quarter was concerned with the method of sampling the reactor coolant.

Normally, the condition of the reactor coolant system water will be monitored by samples taken from the high pressure demineralizer influent and effluent. However, during startup, there will be no flow through the purification system since the demineralizer would remove the hydrazine used for oxygen control. Therefore, some method of removing and cooling a sample taken directly from the reactor coolant system is required in order to monitor the condition of the water during startup.

Three different combinations of sampling point location and sample cooler appeared feasible. The method chosen uses a vent in the control rod actuator and a cooler in the shield water.

3. Design Studies

During the fourth quarter, it was anticipated that the following would be accomplished:

(1) Core design, drawings and specifications would be completed.

(2) Vendor's proposals would be evaluated.

(3) Contracts would be let for reactor coolant system components (i.e., pressure vessel, primary circulating pump and steam generator).

(4) The stress analysis for the pressure vessel would be completed.

(5) The stress analysis for the reactor core would be completed.

(6) Submittal of subsystem design and supporting data to the AEC for approval.

The next quarter efforts will be concerned with:

(1) Completing the detailed drawings of the core and submitting them for approval.

(2) Completing the control rod specifications and submitting them for approval.

(3) Completing the technical memo of the reactor reference core design for final design submittal.

The work accomplished during the quarter was:

a. System (M. Rosenberg)

The subsystem description was prepared and approved by the AEC.

b. Reactor pressure vessel (J. Goeller)

Procurement data and vendor proposals. Technical proposals accompanied by fixed price bids were received and evaluated with the resultant selection of a contractor for the detailed design, fabrication, testing and delivery of the reactor pressure vessel.

Flange redesign. The internal flange design, utilizing T-bolts, was modified to an external flange utilizing conventional-type bolts in slotted holes (see Fig. III-7). The transition section between the flange hub and basic shell is a truncated cone. The change was made to allow for quicker bolt removal and ease the problem of removing broken bolts. The thinner wall sections also result in lower thermal stresses.

Head change. The flat head with welded actuator nozzles was changed to a hemispherical head with a flat upper surface for mounting the actuators. This change was necessitated because tolerances on perpendicularity of welded nozzles could not be guaranteed by the vessel vendor. Also, the hemispherical head results in a sizeable weight reduction.

A fatigue analysis of the hemispherical head indicated no fatigue problems for startup and shutdown rates of 100° F/hr. The analysis indicated, however, that the head must be insulated in the area of the actuators.

Seal design. The present seal configuration, as proposed by the pressure vessel vendor, consists of two metallic O-rings. A double seal is provided so that possible leakage through the inner seal during transients can be collected and measured.

c. Steam generator (M. Rosenberg)

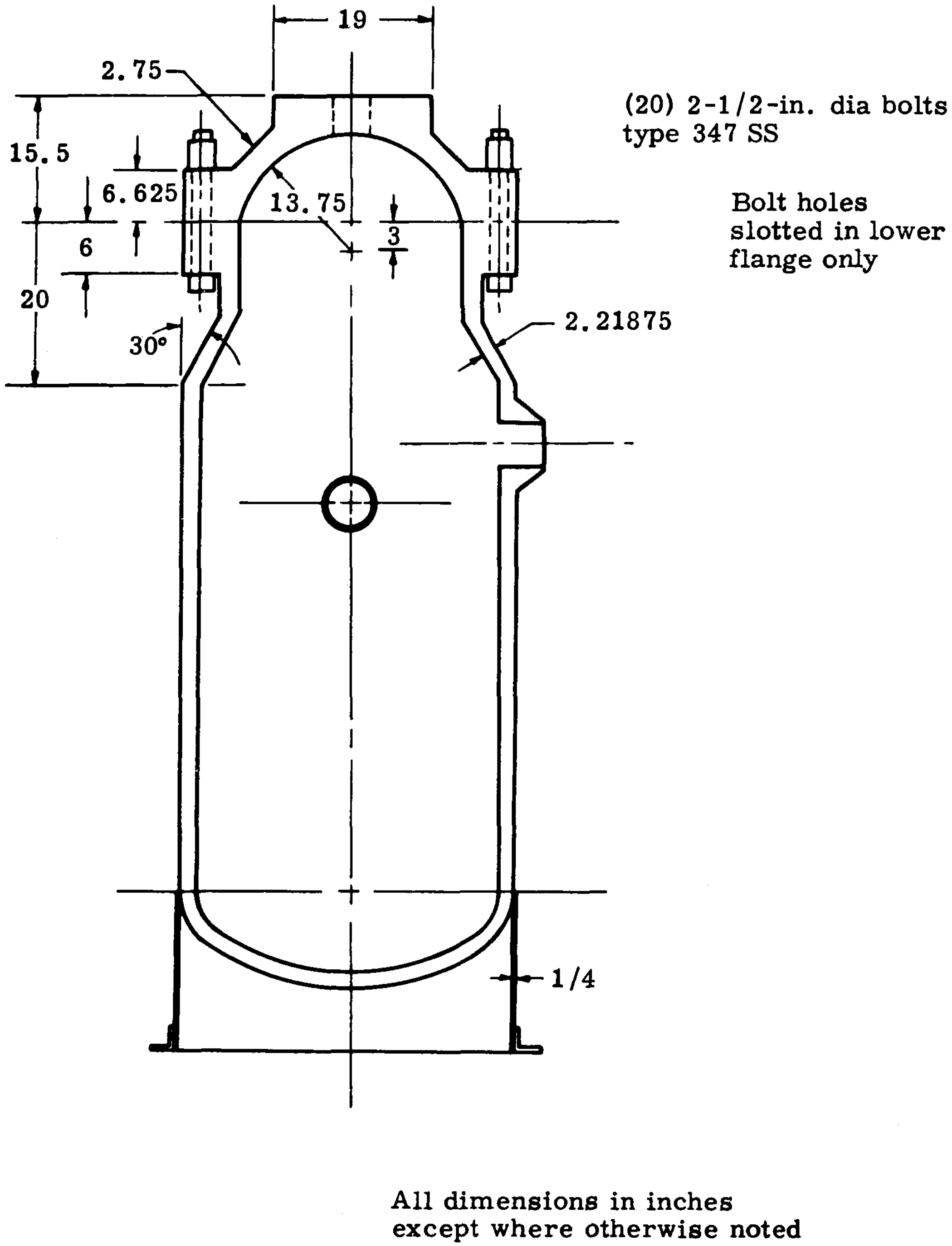


Fig. III-7. Reactor Pressure Vessel

The specification for the steam generator was revised prior to its submittal for approval. The outline drawing of the steam generator was also completed and submitted for AEC approval.

The vendor to supply the steam generator was selected.

d. Primary coolant pump (M. Rosenberg)

Technical proposals with fixed price bids were received for the primary coolant pump from various pump manufacturers specializing in the canned motor design. The pump specification was submitted to the AEC and approved. The vendor proposals were evaluated and a vendor was selected to design, fabricate, test and deliver the primary coolant pump.

e. Piping (J. Goeller)

The stress analysis of the primary loop piping for restrained thermal expansion was completed. The analysis was done by the Elastic Center Method and the Principle of Virtual Work.

The maximum computed thermal stress was found to be approximately 10,000 psi. The loads on the steam generator and primary coolant pump nozzles are tentatively within tolerable limits. The computed values were forwarded to the vendors affected for evaluation.

f. Core (K. Dufrane)

All design problem areas were resolved and detailing of the core components neared completion. In addition, the fuel element and burnable poison element specifications were completed and the reactor core specification rewritten to incorporate both AEC and Martin Nuclear Division comments. A stress review of the complete core was made and the detailed stress analysis report completed.

Incorporation of the domed pressure vessel head (as discussed under the pressure vessel section) required several modifications to the basic core design. In order to eliminate any possibility of steam pocket formation along the head (which would prevent proper cooling), it was necessary to direct a portion of the primary coolant to the domed area. This was accomplished by ducting flow through the center bundle hold-down tube to the top of the domed section. Necessary orificing was placed at the top of the peripheral bundle guide tubes to maximize the center bundle tube flow and to reduce the velocity along the control rod drive mechanism's latch bundle. The flat head was previously used to supply the restraining load to the core hold-down spring. To maintain this function, it was necessary to incorporate a plate in the domed head along its bottom edge.

The proposed control rod design incorporated 3 poison plates, suitably clad and welded together along the central "Y" axis. The lower wear pads will be made of 17-4pH material and brazed to the poison plates during a final annealing operation. The upper guide ring and control rod pickup ball will be made as a unit (17-4pH material) and mechanically fastened to the control rod hub. The actual control rod design will not be completed until the present manufacturing-development effort on the poison plates is further advanced. More information will then be available pertaining to the physical and mechanical properties of the rods and their relative effect on the proposed manufacturing methods.

Unclad burnable poison elements, containing approximately 0.5 wt % natural boron, will be used. The poisoned rod extends only over the active core region, with unpoisoned sections being mechanically fastened at each end. These ends will be identical to the fuel element ends, allowing complete interchangeability at any time prior to bundle assembly.

As a design simplification, the contoured flow baffles used to fill in the water gaps at the core periphery were eliminated. Blockage of the larger gap is maintained by the substitution of dummy steel rods. These are attached to the lower grid and float in the upper grid in a manner identical to the fuel elements. The secondary function of the flow baffles (i.e., to supply guidance during individual bundle replacement) was delegated to the control rod guides after the incorporation of a relatively minor change.

Several methods have been investigated for reducing the diameter of the fuel element dead ends and attaching the elements to the lower grid. Diameter reduction by drawing through a die has been selected with an expanding tool used to mechanically lock the fuel tube to the lower grid. Preliminary tests using wrought tubes were very successful, with loadings of 700 to 1000 pounds required to break this connection. Manufacturing development is now continuing with actual fuel tubes to further investigate the process variables.

A design similar to the lower grid has been selected for the upper grid. The tubes will freely fit through holes in this plate (to allow for free thermal expansion) with interstitial holes provided for the flow outside of the tubes. The upper end of the fuel tube will be reduced slightly in diameter to provide the proper balance of internal and external flow area. Type 304 stainless steel material has been selected for the upper grid. This is quite compatible with the fuel element material and, with the clearances provided, will not allow sufficient (if any) corrosion products to build up and restrain the thermal expansion motion.

The primary source (polonium-beryllium) will be located in a tube at the central location of the center bundle. Just prior to loading the reactor, the source will be inserted through and then locked into the center bundle upper grid. The source will be allowed free thermal expansion into the lower grid.

G. SUBSYSTEM 7--PRESSURIZER AND PRESSURE RELIEF SUBSYSTEM

1. Supporting System Analysis

J. Beam, L. Hassell, R. Baer

The system analysis effort during the past quarter was concerned with 5 major areas. These were:

(1) Pressurizer size.

(2) Spray rate.

(3) Relief valves.

(4) Effect of a malfunctioning relief valve.

a. Pressurizer size

Numerous studies were made on the analog computer to determine the magnitude of the reactor coolant loop volume changes produced by various transients. It was found that the greatest effect on system volume changes resulted from power changes rather than from loss of reactor coolant flow. The transients which result in the largest volume surges into and out of the pressurizer were found to be caused by step changes from 78 to 18% of design steam flow and the reverse excursion. These correspond to summer operation where the 18% steam flow corresponds to the auxiliary electrical load and the 78% steam flow corresponds to the full electrical load. Using a negative temperature coefficient of -2.58×10^{-4} $\Delta k/°F$ (10% less than the expected value), the maximum insurge and outsurge volumes were found to be 0.75 and 0.55 cubic feet, respectively.

The calculated steam volume is 9.79 cubic feet. Analysis of pressurizer performance with the 9.79-cubic foot steam volume shows that it is capable of maintaining the pressurizer pressure at 1197 psia. It should be noted that this pressure is greater than the lower allowable limit of 1185 psia, providing some safety margin.

The required water volume is the summation of the following:

(1) Bottom hemispherical head and heater volume.

(2) Instrument error volume.

(3) Manual control volume.

(4) Heater exposure volume.

(5) Negative surge volume.

(6) Flash water volume.

The total water volume required (not including allowance for volume occupied by heaters) is 4.74 cubic feet, by summation of allowances. The total pressurizer volume is 14.53 cubic feet.

b. Pressurizer spray rate

The effect of spray rate upon pressure reduction time can be seen in Fig. III-8. The higher spray rate (2000 pounds per hour) offers rapid pressure reduction, but the potential to overshoot the 1350-psia lower pressure is greater. The lower spray rate (250 pounds per hour) requires a longer pressure reduction time, but the overshoot of the higher 1420-psia pressure during the initial stage of spray injection makes it undesirable from the operational point of view. A spray rate of 750 pounds per hour was selected to effect a pressure reduction in 33 seconds while dispensing with some undesirable features of the lower and higher rates. The maximum pressurd available for spray injection is the head of 86.5 feet developed by the primary coolant pump.

c. Relief valves

During the past quarter, the required relief valve size was determined based on the final pressurizer size and a realistic negative temperature coefficient. The calculations for the sizing of the pressure relief valves were based on the following plant parameters and transient maneuver, which is the most severe that the plant may undergo.

Initial reactor power = 9.37

Closing of steam generator steam valve in 10 seconds

Primary pump on

Reactor temperature coefficient = $-2.6 \times 10^{-4} \quad \frac{\Delta k}{k\text{-}^\circ F}$

No spray

Loop volume = 56 ft^3 (excluding pressurizer)

Pressurizer:

Minimum normal operating steam volume = 9.20 ft^3

Maximum normal operating steam volume = 5.33 ft^3

Maximum normal operating steam pressure = 1295 psia.

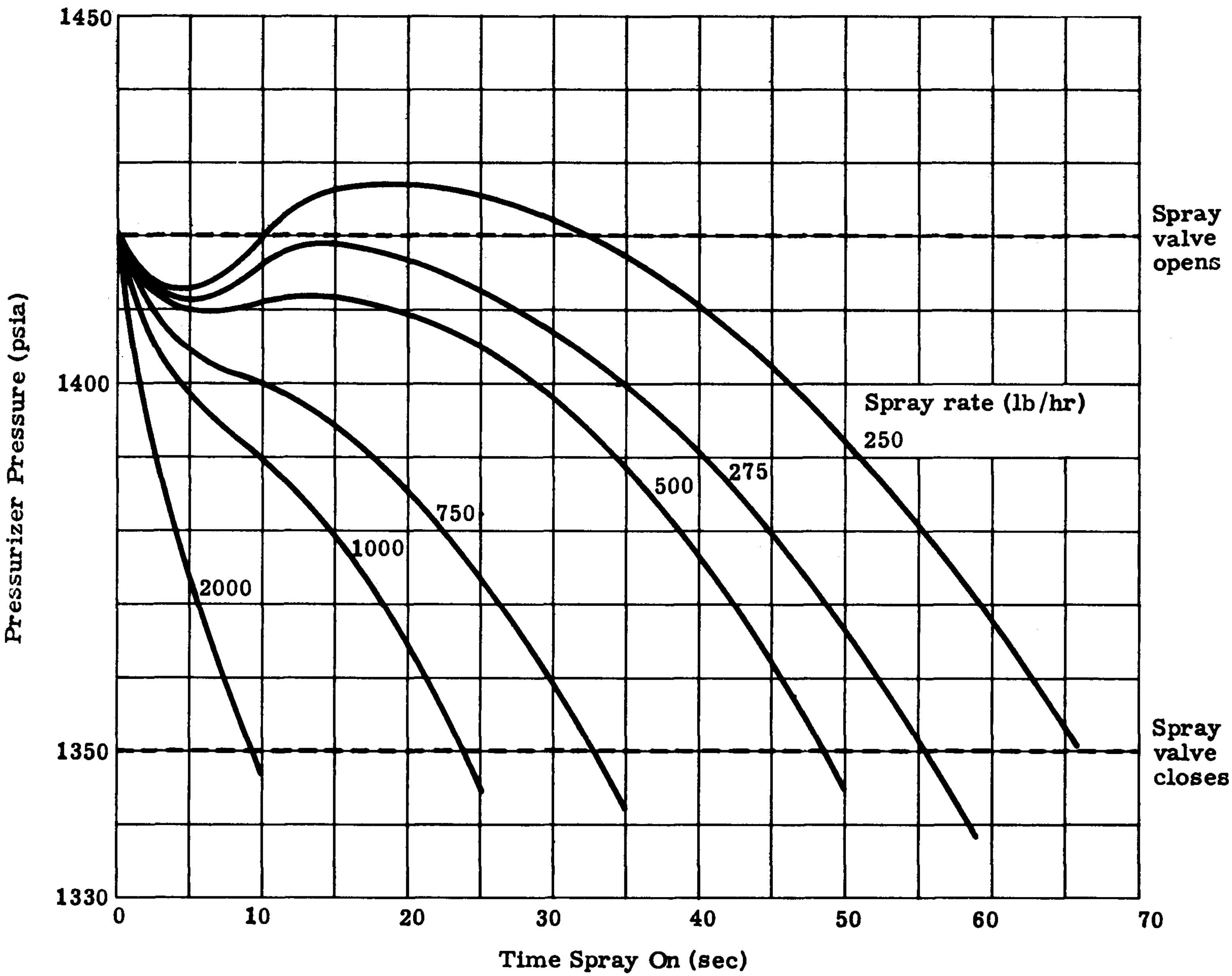


Fig. III-8. Pressurizer Spray Time Versus Pressure (Parameter- Spray Rate)

The compression of the pressurizer steam volume by the primary water insurge resulting from the noted maneuver is estimated to be isentropic; this is a conservative estimation since condensation on the vessel walls and water face is neglected.

The theoretical flow area required in the relief valve was found to be 0.0255 square inches.

d. Effect of a malfunctioning relief valve

In the event that a relief valve on the pressurizer is lifted, there is a possibility that it might stick in the open position or not reseat properly. In that event, the reactor coolant system would experience a loss of coolant resulting in decreasing loop pressure and temperature. The magnitude and rate at which this occurs was investigated.

The first aspect of the problem analyzed was the maximum amount of coolant which might be lost from the reactor coolant system in the event that a relief valve remained open. The initial energy contained in the reactor coolant system and pressurizer was equated to the energy contained in an equal mass of water and steam at atmospheric pressure. The solution indicated that there would be 35.8 cubic feet remaining in the reactor coolant system. This is 3.0 cubic feet more than required to keep the inlet and outlet nozzles of the reactor flooded. Hence, core cooling could be accomplished.

Steam released through the pressurizer relief valves discharges into an expansion tank. This tank is sized to accommodate the mass of steam required to lower the system pressure to 1300 psia. Since the rate of steam release from the pressurizer relief valves is much greater than the condensing rate in the condenser if the pressurizer relief valve remains open, then the relief valve on the expansion tank would also open. Therefore, the rate of change of pressure in the reactor coolant system will depend on whether or not the relief valve on the expansion tank continues to operate properly or sticks in the open position.

Both of these possibilities were investigated. The analyses assumed a pressurizer relief valve having a throat area of 0.110 square inches since this was the smallest applicable relief valve commerically available.

The case of the expansion tank relief valve sticking open was treated first. This relief valve will permit steam flow at a somewhat faster rate than the pressurizer relief valve, thereby eliminating any possibility of restricting the steam discharge rate from the pressurizer. In the event that both valves stick, the pressurizer relief valve essentially discharges directly to the atmosphere. The rate of decrease of pressure, temperature and system volume is shown in Fig. III-9.

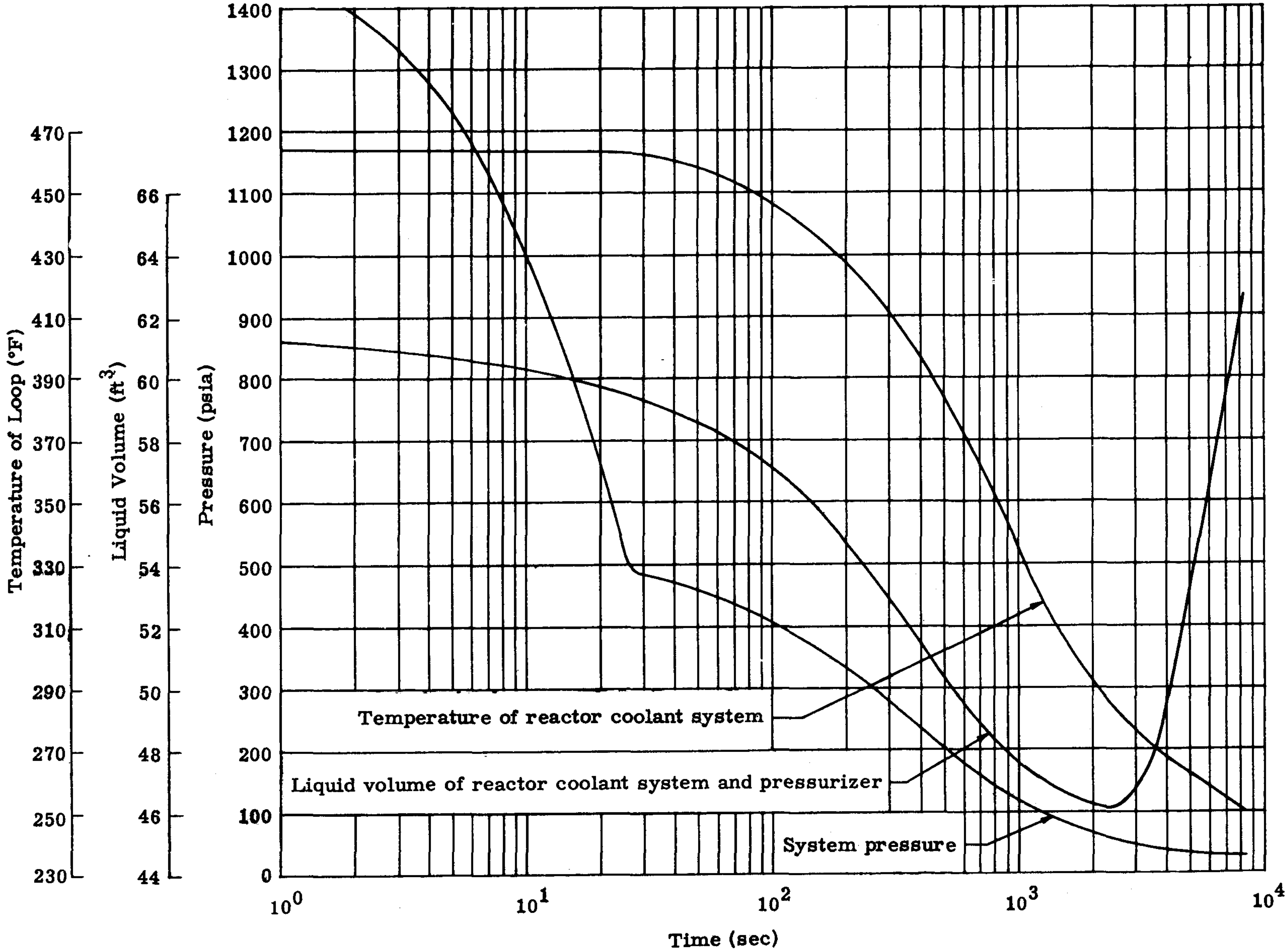


Fig. III-9. Pressurizer and Expansion Tank Relief Valves Both Stuck Open

The case of the expansion tank relief valve operating properly was then treated. In this case, the pressure in the expansion tank quickly reaches 575 psia and remains at this value until the condensing rate exceeds the rate of energy release from the pressurizer.

A stuck pressurizer relief valve would result in a reactor scram due to low pressure. This, in turn, would cut the power to the reactor coolant system pump. The temperature in the hot leg of the reactor coolant system was investigated for this situation. The variation of temperature with time showed that the maximum temperature which occurs is 480° F, corresponding to a saturation pressure of 566 psia.

Therefore, with a pressure of 575 psia maintained in the expansion tank, no gross flashing occurs in the reactor coolant system. Some flashing might occur at local hot spots, but any vapor formed would condense since it must pass through the piping, steam generator and pressurizer before leaving the system. The net effect of this local flashing would be only to increase the mixing rate within the reactor coolant system.

Once the pressure in the pressurizer decreases to 575 psia, the mass decrease in the pressurizer and reactor coolant system is limited by the condensing rate in the coolant discharge and vent system condenser. This is only 0.1 lb/sec at a pressure of 575 psia, and this amount of fluid can be replaced with a single charging pump operating. The energy removed from the system by this means is less than that added by the core afterheat. Therefore, further temperature decrease in the reactor coolant system will be dictated by the heat transferred through the steam generator, using the normal cooldown procedure.

2. Design Studies

R. Moore, M. Rosenberg

During the fourth quarter, completion of design and preparation of the specifications and drawings for final design submittal were the subsystem objectives. In conjunction with this effort, specifications and drawings were scheduled for release to prospective vendors. During the next quarter, upon AEC approval, the vendor bids will be evaluated, recommendations for contract negotiations will be made, and liaison and follow-up will continue.

Both the pressurizer assembly drawing and the pressurizer specification were completed and submitted to the AEC for approval. The drawing and specification were also distributed to suppliers. Requests have been made for preliminary bids on the complete pressurizer assembly.

A study of thermal shock possibilities in the pressurizer and relief subsystem was made. As a result of this study, a bypass line with orifice is now provided to establish a continuous flow of fluid in the pressurizer spray line. In addition, effort has been made to arrive at a satisfactory heater and heater well design for the pressurizer.

The final subsystem description for the pressurizer and pressure relief subsystem was completed and submitted for customer approval.

H. SUBSYSTEM 8--COOLANT CHARGING SYSTEM

R. Moore, M. Rosengerg

The planned objectives for the fourth quarter included completion of the final design of the subsystem, preparation of specifications for the system components, preparation of a subsystem description and requests for vendor bids on subsystem components.

The final subsystem design was established during the quarter and a subsystem description was prepared and submitted for AEC approval. Specifications were prepared for the components involved in this subsystem and were released to vendors with invitations to bid.

I. SUBSYSTEM 9--COOLANT DISCHARGE AND VENT SUBSYSTEM

1. Supporting Systems Analysis

L. Hassell, S. Zwickler

During the past quarter, the analysis effort was concerned with 2 major areas, namely:

(1) Size of expansion tank.

(2) Design of condenser.

Expansion tank size. The water insurge into the pressurizer, resulting from the most severe transient maneuver, would produce a pressure of approximately 1575 psia and enthalpy at 1195 Btu per pound provided that the pressurizer safety valves did not open. To return the primary loop to its normal pressure of 1300 psia requires that the pressurizer discharge approximately 5 pounds of the steam in its dome. The size of the expansion tank versus the operating pressure is shown in Fig. III-10. The design point of 500 psia is chosen as producing a reasonably sized expansion tank.

Condenser design. After a consideration of several schemes, the plan of submerging a coil in the shield water appears to be the most feasible manner of rejecting the heat transferred to the expansion tank in the form of steam from the pressurizer safety valves.

The steam, initially at a temperature of 467° F, flows from the expansion tank into the coils of the condenser where it is condensed by

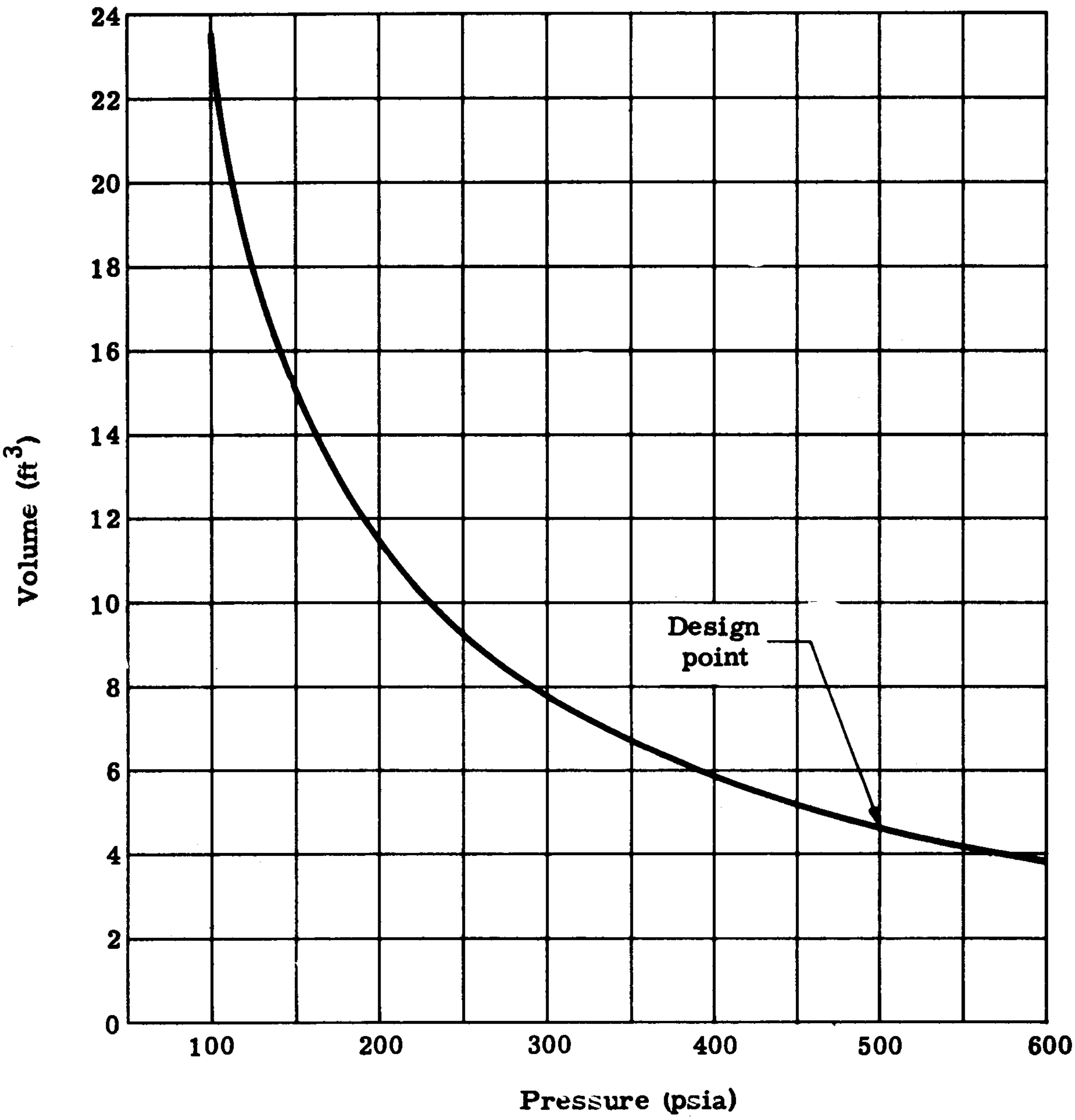


Fig. III-10. PM-1 Subsystem No. 9 Coolant Discharge and Vent System Expansion Tank Volume Versus Initial Pressure transferring heat through the tube wall to the shield water, the temperature of which is maintained at 125° F. High heat transfer coefficients are achieved over the major portion of heat rejection due to high condensing coefficients inside and high boiling coefficients outside the tube.

The overall heat transfer coefficient was based on the following assumptions:

(1) Tube material is Type 304 stainless steel with conductivity of 9.4 Btu-ft/hr-ft^2-° F.

(2) Inside scale factor of 3000 Btu/hr-ft^2-° F.

(3) Outside scale factor of 2500 Btu/hr-ft^2-° F.

It was found that, for a condenser having a surface area of 1.0 square foot, approximately 8-1/2 minutes are required to condense the steam.

The condenser will consist of 12 linear feet of 1-1/2-inch OD 16BWG tubing. This has a surface area of 4.7 square feet, resulting in a condensing time of less than 2 minutes.

2. Design Studies

R. Moore, M. Rosenberg

During the fourth quarter, the planned objectives were the completion of the subsystem design, completion of specifications and submittal of data for approval. Also, vendor contacts were to be established and requests made for bids on system components.

During the next quarter, objectives will be evaluation of vendor bids, recommendations for component procurement and liaison and followup for the procurement, fabrication, testing and delivery of the subsystem components.

The design of the subsystem has been completed. Specifications on all subsystem components have been completed and released to vendors for bidding purposes. All subsystem data was submitted for AEC approval.

The expansion tank was established as an all-welded, cylindrical vessel with a volume of 4.6 cubic feet, designed for 600 psia and 500° F.

The silver nitrate reactor in the off-gas system is made of 2-1/2-inch Schedule 40 pipe, approximately 22 inches in length, with a 10-inch central section containing 1/4-inch Berl saddles.

The condenser requirements were found to be less critical than originally expected. This component now consists of a 6-foot-high coil of 1-1/2-inch pipe located in the shield water and fabricated as a pipe detail.

J. SUBSYSTEM 10--COOLANT PURIFICATION SUBSYSTEM

1. Supporting Systems Analysis

C. Smith

The values of all the basic parameters of this system were established prior to the past quarter. During the past quarter, one revision to the basic flow diagram was made. This was the elimination of the bypass line around the demineralizer and the associated valving. The bypass line was originally provided to allow a small backflow through the demineralizer to loosen the resin prior to flushing. However, it was found that the same result could be accomplished merely by opening the valve of the sampling point at the inlet of the demineralizer while the remote-operated valve at the demineralizer inlet is in the closed position.

2. Design Studies

R. Wolfe

During the fourth quarter, a final design of components, preparation of specifications for these components and completion of a subsystem description were the planned objectives. In addition, the specifications were to be submitted for approval and released to vendors with an invitation to bid.

During the next quarter, effort will be directed toward evaluation of bids, selection of a contractor, and liaison and followup in the procurement, fabrication, testing and delivery of the subsystem components.

The thre components of this subsystem, the economizer, the cooler and the demineralizer, were sized in accordance with a detailed analysis. Specification data sheets were prepared and forwarded to a number of vendors specializing in each type of component with a solicitation to bid.

A subsystem description was prepared and submitted for AEC approval.

K. SUBSYSTEM 11--COOLANT CHEMICAL ADDITION SYSTEM

R. Wolfe

The planned objectives of the fourth quarter were a study of the subsystem requirements, preparation of a final flow diagram, preparation of component specifications, completion of the subsystem description and submittal of subsystem effort for approval.

The next quarter's efforts for this subsystem will be the evaluation of bids, contractor selection, and liaison and followup of the procurement, fabrication, testing and delivery of the components.

Specifications were prepared for the procurement of the components. These specifications were forwarded to vendors with requests for bids. A subsystem description was prepared and the complete subsystem design was submitted for AEC approval.

L. SUBSYSTEM 12--DECAY HEAT REMOVAL SUBSYSTEM

1. Supporting Systems Analysis

C. Smith, R. Baer

The values of all the parameters of this subsystem had been established prior to this past quarter. The additional analysis performed this quarter is concerned with the temperatures occurring during one mode of operation not previously investigated.

The decay heat removal system pump takes suction from the hot leg of the reactor coolant system, pumps this fluid through the demineralizer cooler and then back into the cold leg of the system. The reactor coolant and decay heat removal systems are so designed that the steam generator can be drained and maintenance performed on the reactor coolant system pump while the decay heat removal system is operating. In that event, all the flow from the decay heat removal system will flow through the reactor and the reactor inlet and outlet temperatures will be equal to the demineralizer cooler outlet and inlet temperatures, respectively, namely, 107° F and 140.2° F.

The question has arisen as to the flow rates and temperatures in the 2 systems which will occur in the event the decay heat removal system is operated while the steam generator is not drained. This was analyzed based on the following conservative assumptions:

(1) No heat is transferred from the reactor coolant system fluid in the steam generator.

(2) The system head loss varies from the value calculated for design flow rate to the 1.8 power.

As a result of the study, the following conclusions were made:

(1) There is sufficient natural convection in the reactor coolant system to prevent reverse flow through the steam generator.

(2) The reactor outlet temperature is determined by the heat transfer rate in the demineralizer cooler and, therefore, remains at 170° F.

2. Design Studies

M. Rosenberg

The planned objectives during the fourth quarter were to complete a subsystem design, prepare component specifications, complete a subsystem description, to submit the system design for approval and to contact vendors of components.

For the next quarter, the objectives will be the evaluation of vendor bids, recommendation of equipment contractor and liaison and followup for the procurement, fabrication, testing and delivery of the subsystem components.

The design of the subsystem was completed and a specification was prepared for the decay heat removal pump required for subsystem operation. A subsystem description was preapred. The system design and data were submitted for AEC approval. The component specification was released to various vendors with an invitation to bid.

M. SUBSYSTEM 13--SHIELD WATER SYSTEM

M. Rosenberg

The fourth quarter objectives for this subsystem included preparation of a subsystem description, preparation of component specifications for release to various vendors with invitations to bid and submittal of subsystem design for approval.

Objectives for the next quarter will be the evaluation of vendor bids, selection of a contractor, liaison and followup of the procurement, fabrication, testing and delivery of the components required by the subsystem.

During the development of a conceptual drawing for the Shield Water Air Blast Cooler, it became apparent that optimum design would require the unit to be furnished as a complete package by one vendor. This would expedite the design, fabrication, delivery and erection of a complex component and assign the responsibility for reliability and overall performance of the entire unit to one contractor. The specification for this component was written to cover this design concept. Specifications were prepared for other components in the subsystem and released to vendors with invitations to bid. The Air Blast Cooler package specification was also submitted for AEC approval. A subsystem description was prepared and the system design was submitted for approval.

N. SUBSYSTEM 14--REACTOR PLANT HEATING AND COOLING SYSTEM

1. Supporting System Analysis

R. Baer, J. Kudrick

During the past quarter, the analysis effort was concerned with two major areas:

(1) Heating system in the shield water and spent core tanks.

(2) Review of steam generator heating problems.

Heating System in the Shield Water and Spent Core Tanks. The two tanks will be heated by steam coils submerged in the water. In order to determine the required coil sizes, the following analyses were performed.

Heat Loss Calculations. All surfaces of the tanks, with the exception of the tops, were in contact with the earth.

The calculation of the het flux through the top of the tanks was based on a 1-inch thickness of commercial insulation, an ambient temperature of -65° F outside and a 70° F temperature inside the tanks. A heat flux of 48.7 Btu/hr-ft^2 was determined.

The calculation of heat flux into the earth surrounding the tank was treated as het transfer into a semi-infinite solid. The heat flux after 1 hour was calculated to be 130 Btu/hr-ft^2.

The heat fluxes were multiplied by the associated areas to obtain the heat losses of the 2 tanks. Ten percent was then added to give design heat loads of 71,000 and 127,000 Btu/hr for the spent core and shield water tanks, respectively.

Heater Coil Design. Both 100-psia and 35-psia steam are available for heating purposes. The higher pressure was selected so that there would be no difficulty in returning the condensate from near the bottom of the tanks to the condensate heater.

Since it is desirable to keep the heater coils as simple as possible, it was decided to have the heater merely an immersed extension of the steam supply line, provided sufficient heat transfer area could be obtained. The pressure drop in the steam line was limited to approximately 2 psia for 100 feet of straight pipe. The standard pipe size giving the closest to desired pressure drop was selected. This resulted in a 1/2-inch pipe for the spent core tank and a 3/4-inch line for the shield water tank.

The required heat transfer area was then calculated considering a boiling coefficient outside and a condensing coefficient inside the heater. In addition, scale having equivalent film coefficients of 3000 and 2500 inside and outside the heater was assumed.

The required length of heater was then calculated to be 6 feet and 8.6 feet in the spent core and shield water tanks, respectively.

2. Design Studies

J. Todd

The objectives for the quarter were to establish the heating and/or cooling requirements for the primary system container tanks, complete design and analytical supporting data, prepare component specification data sheets and submit the subsystem design for approval.

During the next quarter, vendor contact will be established and component equipment will be selected. Equipment mounting requirements will be closely coordinated with the container subcontractor for integration into the container design.

The shield water heating system requirements within the reactor containment tank and spent fuel storage tank were established. Ventilatin requirements for the steam generator container tank were finalized.

Due to the anticipated high local dose rates within the steam generator tank during operation (i.e., 200 r/hr), shielding requirements within the primary building above the steam generator required the installatpon of a water-filled cover over this container tank. To accept this, a separate exhaust duct had to be installed in the container for exhausting the 125° F air to the eave of the building. From there, it is directed to atmosphere by the building roof ventilator fan.

To ensure proper cooling of the equipment within the steam generator container tank environment, the cooling air requirements were increased from 650 to 1000 cfm. Duct losses were estimated as 1.50 inches H_2O static pressure. A thermometer and manually operated, variable vane-type damper were incorporated to control container temperatures. Aluminum alloy ductwork is to be used to maintain a relatively lightweight, noncorrosive and easily installed assembly.

O. SUBSYSTEM 15--CORE CASK COOLING SYSTEM

1. Supporting System Analysis

C. Smith

During the past quarter, the analysis effort was concerned with adequate het removal in the spent core tank. Both steady-state and transient operation were investigated.

Steady-State Operation. After the spent core has been removed from the reactor vessel, it is transferred and stored in the spent fuel tank where the het geneation rate and radioactivity of the spent core is allowed to decay before shipping. A cooler, which uses shield water as a coolant, is submerged in the spent fuel tank to remove the het. The cooler is sized to remove the het generated by the spent core 1 day after shutdown (144,500 Btu/hr), based on normal shield water temperature and a spent core tank water temperature of 135° F.

Transient Operation. In the transient analysis, both cases of the waste condenser operating and not operating were treated.

a. Waste condenser not operating

The earliest time at which the spent core can be removed and transferred to the spent fuel tank is an estimated 12 hours after shutdown. This is based on an 8-hour allowance before removing the reactor vessel head and 4 hours for transfer of the spent core. In the 8-hour period before the reactor vessel head is removed, the shield water is cooled to below 100° F. The temperature in the spent fuel tank will be 105° F (normal shield water cooler outlet temperature). A transient analysis was made to determine the temperature transients in the spent fuel tank from 12 hours to 24 hours after shutdown with the above initial temperature conditions. The shield water air coolers can remove 94 kw(t) (spent core after 8 hours after shutdown plus 34 kw(t)) with a water outlet temperature of 90° F and an air temperature of 70° F. With the shield water temperature entering the submerged spent fuel tank cooler constant at 90° F, the spent fuel tank water temperature was found to increase from 105° F to 120° F during the 24 hours after shutdown. This is acceptable, since it is less than the steady-state design temperature of 135° F.

b. Waste condenser operating

The worst temperature transients occur in the spent fuel tank when the waste condenser is in operation. The heat load on the shield water air coolers can be as high as 153 kw(t) (waste condenser 100 kw(t),s spent core plus primary coolant pump 53 kw(t)). The shield water air coolers can remove 153 kw(t) with a water outlet temperature of 100° F (shield water temperature, 117° F). With the shield water temperature entering the submerged spent fuel tank cooler constant at 100° F, the spent fuel tank temperature was found to increase from 105° F to 129° F during the 24 hours after shutdown. This is still below the steady-state design temperature.

2. Design Studies

R. Wolfe

The objectives during this quarter were to establish a system design and prepare the proper data sheets and specifications necessary for the procurement of the system components. In addition, the arrangement of the system equipment in the package form was to be accomplished.

A schematic drawing of the fuel cask cooling system is shown in Fig. III-11.

During the next quarter, the subsystem design and supporting data will be submitted for approval, evaluation of the various equipment vendors for the system components will be accomplished and procurement of the required system equipment will be initiated.

A system design was established, the system equipment arrangement drawing was made and a system description was prepared. In addition, consideration was given to the transfer and design of fuel casks and the removal of radioactivity and purification of the fuel storage tank water.

Component specification data sheets were prepared for the following system items:

(1) Spent fuel tank pump.

(2) Fuel cask (spent fuel).

(3) Fuel cask (failed fuel).

It is anticipated that the spent fuel from the PM-1 Plant will be processed at the Idaho Chemical Processing Plant at Scoville, Idaho. Therefore, the fuel casks will be designed to meet their requirements for the handling of irradiated fuel shipments. It is also anticipated that the spent fuel will be stored for approximately 1 year at the PM-1 site before shipment.

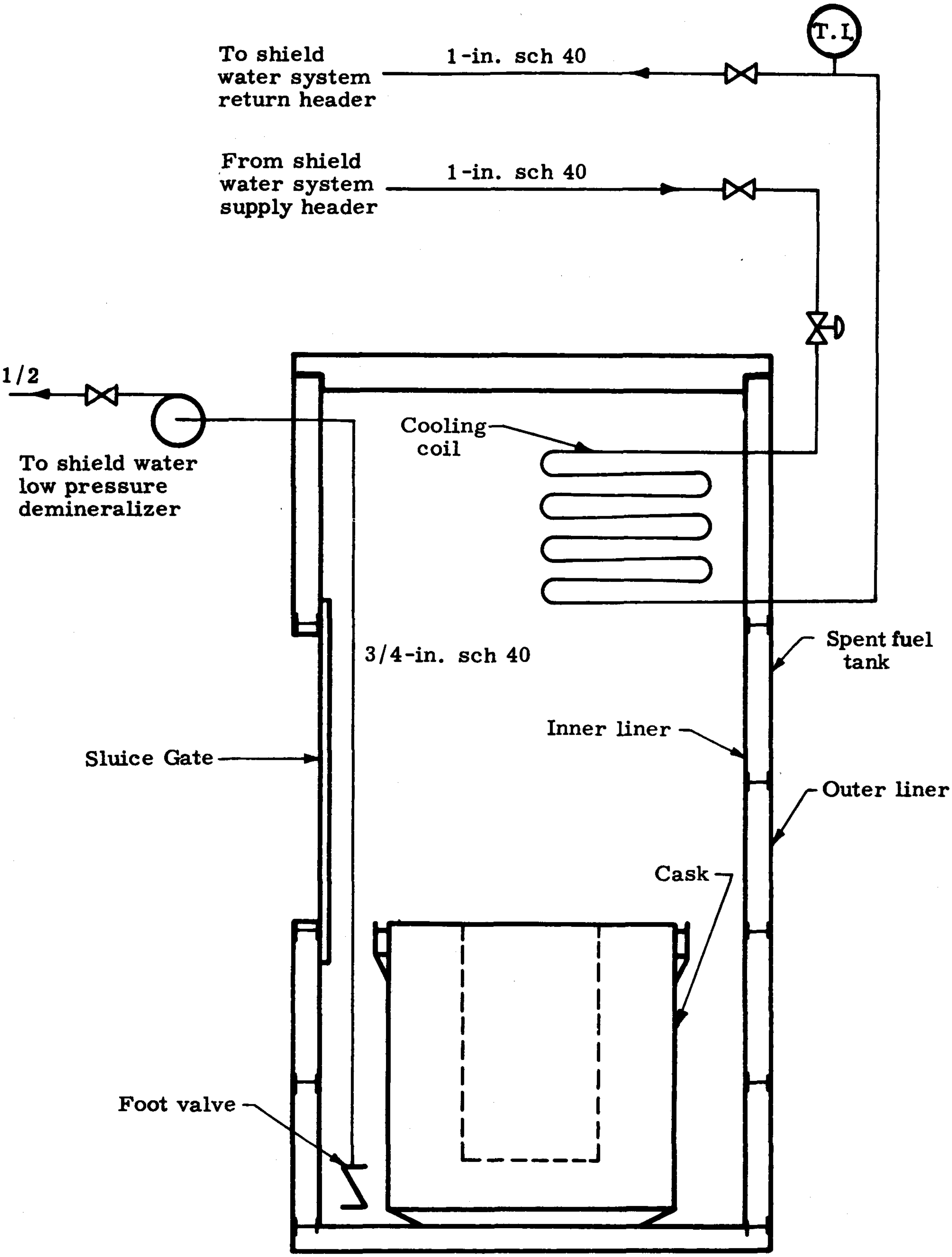


Fig. III-11. Core Cask Cooling System

P. SUBSYSTEM 16--RADIOACTIVE WASTE DISPOSAL SYSTEM

O. Millman

The objectives for the quarter were to complete configuration drawings, prepare data sheets and specifications for equipment, complete system descriptions and establish vendor contact with equipment manufacturers.

A schematic flow diagram of the waste disposal subsystem is shown in Fig. III-12.

System component design criteria were completed during the quarter. The waste sump tank capacity was established at 1600 gallons. This will enable the Turco cleaning process to be utilized in an effective manner.

The waste evaporator was sized for the available system steam (100 psig) and to yield an evaporation rate of 33 to 35 gph. This rate, though lower than originally contemplated, will be adequate for handling decontaminant wastes during a plant shutdown period. The time cycle is to be lengthened 12-1/2 to 15%. Design of the unit has been modified from a conventional evaporator design to include hopper heating coils, steam tracer discharge lines and various other accessories for simplifying operation.

The canned sump pump (as included in earlier designs) was eliminated in favor of gravity flow, from sump tank to evaporator, regulated by a float controlled valve. The design includes the condenser as an integral part of the evaporator assembly mounted in a vertical position for best usage of space. The condensate return pump receiver is positioned directly below the condenser. Discharge from the steam trap and transfer pump flush water will be returned to this small condensate receiver where a radiation detector indicated its degree of radioactivity. If within safe limits, the fluid is returned to the secondary system storage tank and, if not, it is pumped back to the sump tank for re-evaporation.

A steam heating coil was included in the final waste storage tank design to liquefy the waste solids within the tank if necessary. Subsequently, these wastes are then discharged by means of the transfer pump to smaller, transportable shielded waste casks for removal from the site. The pump suction line is heated by means of a steam tracer to assure proper flow from the tank. Water flushing connectins provide means for removing solids from within the pump upon com-

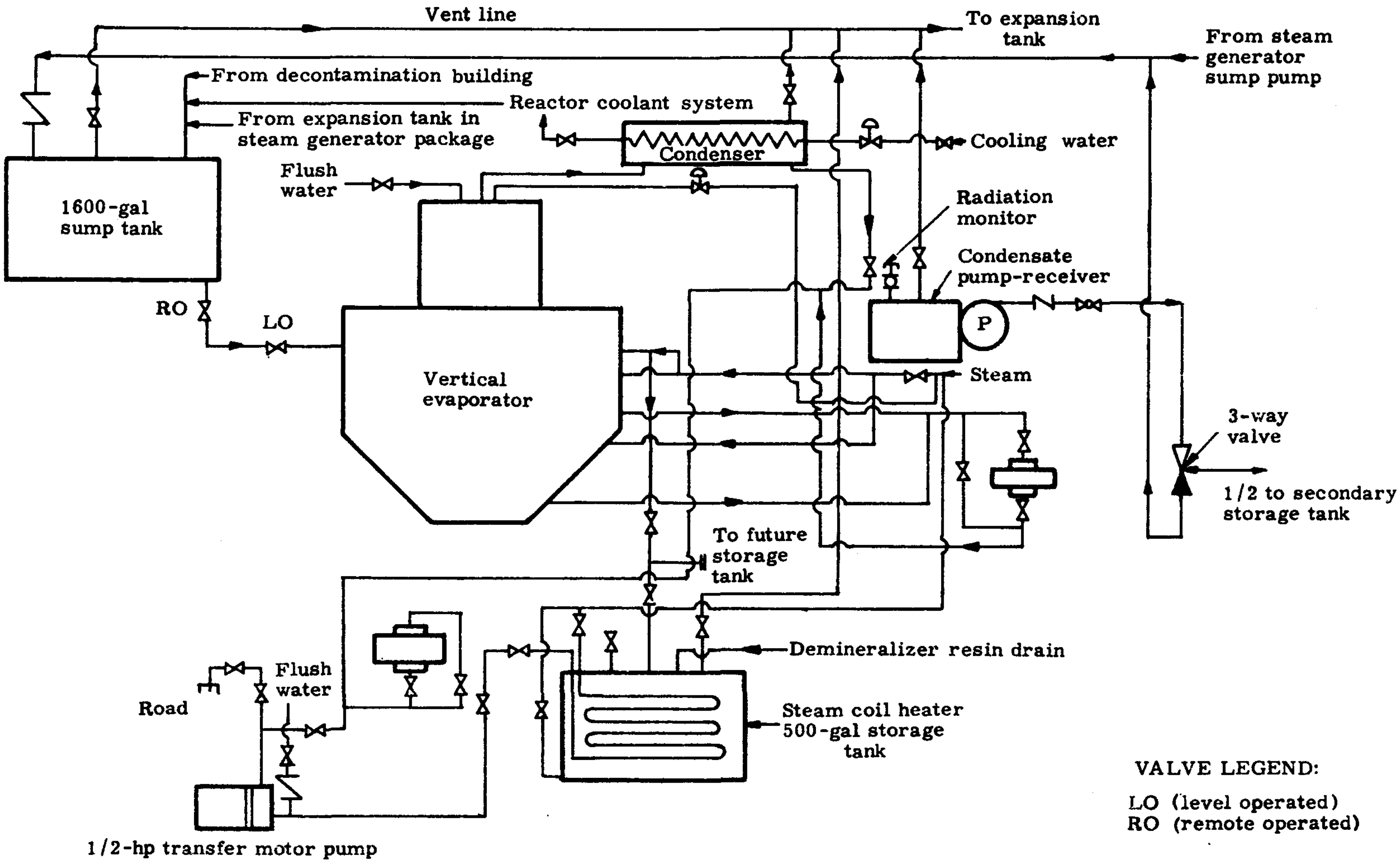


Fig. III-12. Radioactive Waste Disposal System (not to scale)

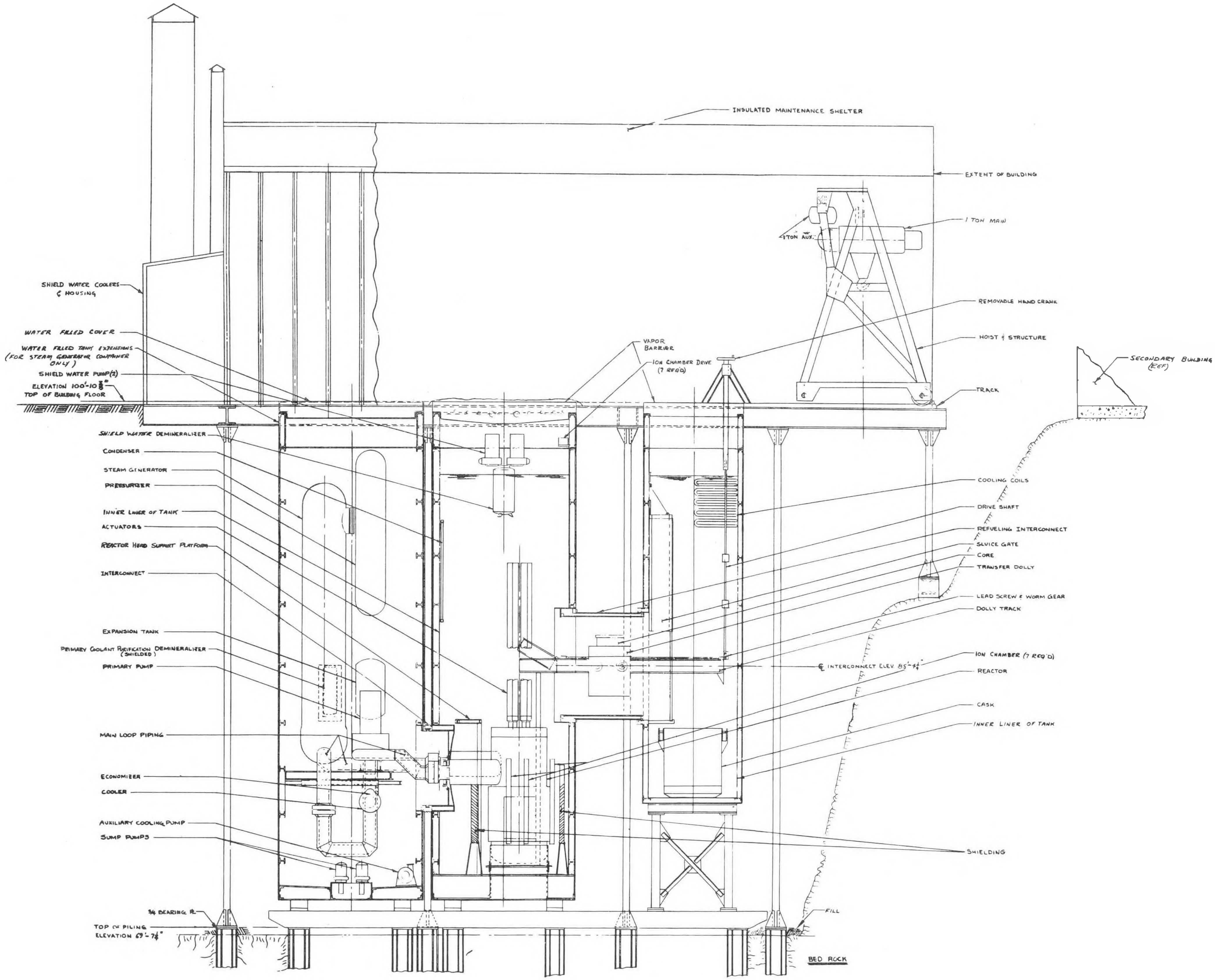


Fig. III-13. Primary System Arrangement

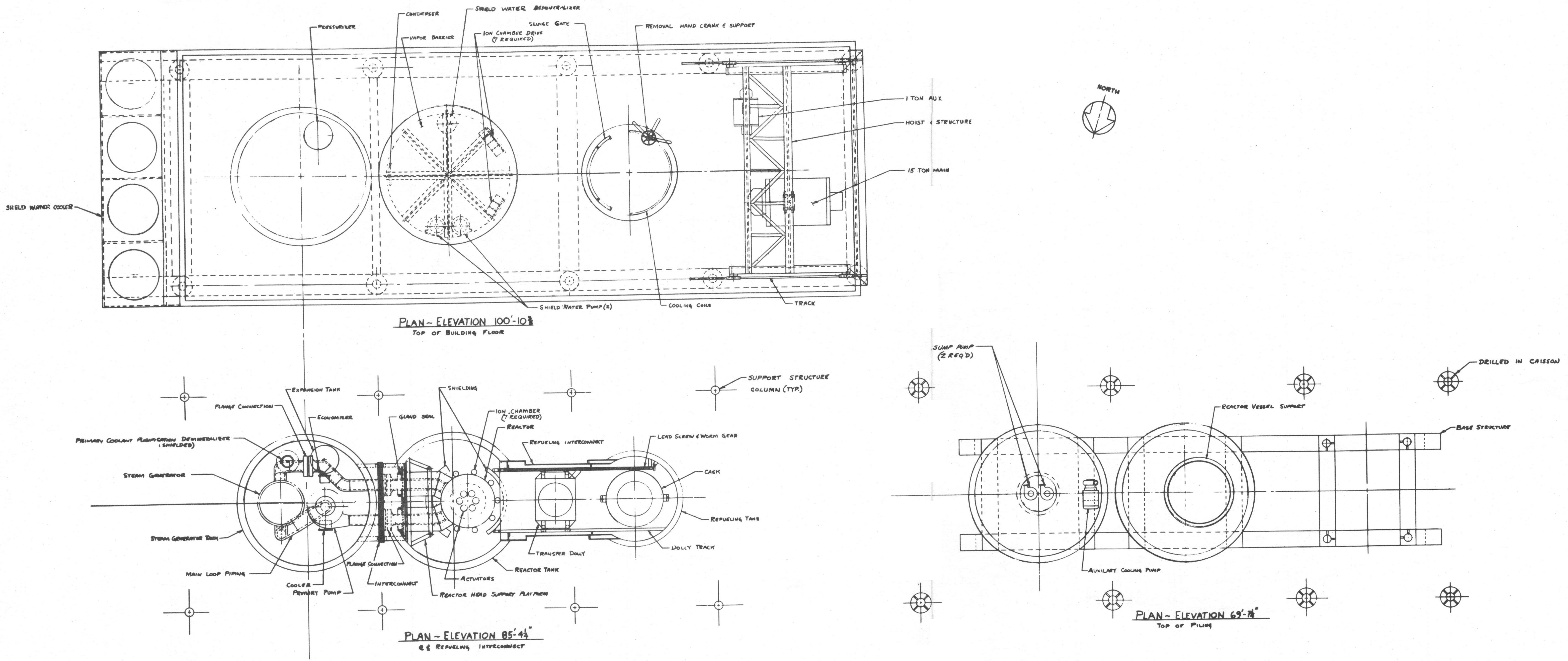


Fig. III-13. Primary System Arrangement (continued) pletion of the pumping sequence. Data sheet specifications were prepared for all pieces of equipment in the system. The evaporator specification was completed. Vendor contact was established. Final configuration drawings were prepared, all equipment was sized and foundation criteria was determined for the architect-engineer.

The objectives for the next quarter are completion of the modular package concept for the waste disposal system, submittal of the system design and supporting data for customer approval and the final selection of equipment making up the entire system. It should be noted that initial studies made into this concept have made feasible an improved method for integration of waste disposal system components. Design efforts are continuing along this approach and will be completed during the next quarter.

Q. SUBSYSTEM 17--PLANT CONTAINER

R. Manoll, J. Todd, W. Dallam, P. Mon

The plant container subsystem consists of the items previously mentioned in the Third Quarterly Report (MND-M-1814) plus:

(1) The vapor barrier for the spent core storage tank.

(2) The combination cover and shielding water container for Tank No. 2 (the container housing the steam generator).

(3) Erection trunnions for Tanks 1, 2 and 3.

The plant container subsystem arrangement is shown in Fig. III-13.

The objectives during the past quarter were to prepare final configuration drawings and specifications for the components, complete the stress reports for use in the system description and analysis, submit the subsystem design and supporting data for approval, establish vendor contact for the plant containers, superstructure and the gantry crane, and select a subcontractor for the plant containers. During the next quarter, vendor contacts will be maintained, changes will be incorporated into the final design to reflect vendors' certified drawings and vendor bids will be evaluated, recommendations made and procurement initiated for the remainder of the system components.

During the quarter, the work was accomplished under 7 categories which are:

(1) The primary system container.

(2) The steel structure supporting the primary system building and gantry crane.

(3) The gantry crane.

(4) The steam generator supports.

(5) The refueling dolly.

(6) Handling tools for the core and the basic refueling procedure.

(7) Erection equipment and procedure for the primary system plant.

The primary system container was expanded to include necessary shielding provisions at the upper portion of the steam generator container and erection trunnions for the spent fuel storage tank. The shielding is accomplished by providing a water-filled cover over the steam generator tank (Tank No. 2) and a water-containing extension for the top of the tank. Also, the spent fuel storage tank was enlarged to accept a transfer cask meeting Idaho Chemical Processing Plant regulations.

The steel superstructure now incorporates a wood skid track for the spent core transfer cask and supports for fresh core storage vaults in the loading bay. The structure has been designed to accept a longer primary system building and provide adequate strength for erecting the primary system containers with the gantry crane. The gantry crane is sized to lift the spent fuel shipping cask as well as the container system tanks and their contents during erection. A 15-ton hoist is used to hoist the primary loop tanks and other heavy components while a 1-ton auxiliary hoist handles the core and other light components having to be serviced.

Stress analysis was prepared for the steel superstructure and the steam generator support. The drawings prepared for these items reflect the calculations.

The charging pumps were removed from the steam generator tank and placed in the decontamination building to reduce the overall weight of the steam generator tank package during shipping and reduce the piping runs.

The subsystem description and supporting data were submitted for AEC approval.

The basic refueling procedure was established during the quarter to coordinate the core handling tools, reactor vessel head tools and actuator tools for continuity of refueling operation and storage space requirements.

R. SUBSYSTEM 18--MAIN AND AUXILIARY STEAM SYSTEM

R. Groscup

During the quarter, the following work was planned and accomplished:

(1) The piping requirements for this subsystem were established.

(2) The subsystem writeup and supporting data was submitted for approval.

During the next quarter, the following work is planned:

(1) Complete piping configuration layouts and establish a complete valve and specialty list.

(2) Determine shipping requirements for the system piping and locations of the site interconnections.

A revision in secondary system flow diagram was received during the quarter from the Gibbs and Hill Co.; however, pipe sizes and flows remained the same in this subsystem.

All piping runs between the primary and secondary systems will be run over the decontamination package in a heated piping duct.

S. SUBSYSTEM 19--MAIN TURBINE AND GENERATOR UNIT

R. Groscup

During the quarter, the following work was planned and accomplished:

(1) The subsystem writeup and supporting data was completed and submitted for approval.

(2) Vendor bids were evaluated and a manufacturer selected for the air-cooled lube oil cooler and the lube oil conditioner.

(3) A revised certified outline drawing of the turbine-generator unit was received from Westinghouse.

(4) The air-cooled oil cooler certified drawing was received, reviewed and the recommended installation changes made.

(5) The certified drawing of the lube oil conditioner was received, reviewed and the recommended installation changes made.

During the next quarter, the following efforts are planned:

(1) Complete drawings showing the turbine, lube cooler and oil conditioner foundation supports at the site.

(2) Complete drawings showing interconnecting piping and wiring between the turbine generator and its auxiliaries.

(3) Complete the integration of the turbine generator and its auxiliaries in the overall plant.

(4) Vendor liason will be conducted under Subtask 7.2.

The following design modifications were made to the turbine-generator unit during the quarter:

(1) Well-type oil thermometers were replaced with bearing cap thermocouples.

(2) Steam and oil pressure gauges were mounted on the turbine skid for visual checking during startup and operation.

(3) The turbine-generator mounting skid design was modified to comply with the conditions set forth in MN-7052, the shipping specification.

(4) Flow and temperature conditions in the lube oil air cooler were modified for improved efficiency.

T. SUBSYSTEM 20--MAIN CONDENSER AND CONDENSATE SYSTEM

L. Hassell

During this quarter, the following planned work was accomplished:

(1) The condenser model test was 90% completed at Eglin AFB.

(2) The 2-pod condenser and condensate system layout was completed except for minor dimensional changes required by vendors' certified drawings.

(3) The condensate pumps and hotwell specifications were completed, submitted to the AEC for approval and sent out to vendors for bids.

(4) The subsystem writeup and supporting data was developed and submitted to the AEC for approval.

During the next quarter, the following work is planned:

(1) The final design of the PM-1 condensers as modified by the condenser model test results will be 90% completed.

(2) The condenser site layout based on the revised design will be completed.

(3) The condenser and air-ejector specifications will be completed and submitted to the AEC for approval.

During this quarter, the condenser model test was 90% completed at Eglin AFB as discussed in Subtask 1.6. The layout and specifications for the condenser, air ejector, condensate pumps and hotwell were received from Gibbs and Hill as discussed in Task 16. Vendor liaison will be conducted under Subtask 7.2.

U. SUBSYSTEM 21--FEEDWATER SYSTEM

L. Hassell

During this quarter, the following planned work was accomplished:

(1) The deaerator, feedwater pumps and feedwater heater specifications were completed, submitted to the AEC for approval and sent to vendors for bidding.

(2) The feedwater system layout was completed except for minor dimensional changes required by vendors certified drawings.

(3) The subsystem writeup and supporting data was developed and submitted to the AEC for approval.

During the next quarter:

(1) The feedwater system layout drawings will be completed with the use of vendor certified drawings.

(2) The deaerator, feedwater pumps and feedwater heater will be ordered and vendor liaison will be established under Subtask 7.2.

During this quarter, the specifications for the deaerator, feedwater heater and feedwater pumps were complete with the incorporation of review changes.

The layout of this equipment was completed by Gibbs and Hill except for minor dimensional changes due to information from vendors' certified drawings (see Task 16).

V. SUBSYSTEM 22--EXTRACTION STEAM AND HEATER DRAIN SYSTEM

L. Hassell

During this quarter, the following planned work was accomplished:

(1) The layout of the system was completed except for minor dimensional changes required by the receipt of certified equipment drawings from vendors.

(2) The subsystem description and supporting data were developed and submitted to the AEC for approval.

During the next quarter:

(1) The layout of this subsystem will be completed.

(2) Vendor liaison will be conducted under Subtask 7.2.

During this quarter, the subsystem layout was completed by Gibbs and Hill (see Task 16) except for minor dimensional changes which will be required due to vendors' certified drawings.

W. SUBSYSTEM 23--COOLING WATER SYSTEM

W. Koch

During this quarter, the following planned work was accomplished:

(1) The subsystem description and supporting data were developed for submittal to the AEC for approval.

(2) The cooling water requirements of the PM-1 plant equipment were established.

During the next quarter:

The layout of the cooling water system will be completed.

The secondary system components such as the turbine lube oil cooler, condensate pump turbine drive bearings, feedwater pump turbine drive bearings and condenser, which are normally water cooled, are designed for air cooling in the PM-1 plant.

The only cooling water required from an external source for the PM-1 plant is for the oxygen sample cooler. This sample is only taken several times per day and the cooling water used for this operation is salvaged by dumping into the ondensate collection tank (Subsystem 33).

Cooling water is also required for the in-line water analyses sample coolers on the steam generator blowdown, evaporator vapor and evaporator blowdown. For these coolers, 157° F condensate is used as the cooling water and is dumped to the condensate storage tank after passing through the cooling coils.

X. SUBSYSTEM 24--MAIN TRANSFORMER AND DISTRIBUTION SYSTEM

J. McNeil

The main transformer and distribution system consists primarily of switchgear to control the major electric power sources and loads. The switchgear contains circuit breakers and auxiliary equipment for control of the turbine-generator, diesel-generator, tie for 2-way feed to a separate diesel-generator plant, substation, main transformer and motor control center. The main transformer reduces the main bus voltage for utilization by motors and heaters of plant auxiliaries and lighting systems.

The objectives for the system during the quarter were:

(1) Completion of switchgear and power center transformer specification.

(2) Preparation of the subsystem description and supporting data and submission for approval.

(3) Completion of switchgear layout.

(4) Completion of electrical one-line diagram.

The efforts accomplished this quarter in the completion of the subsystem final design were the following:

(1) The switchgear and power center transformer specification was completed.

(2) Wiring diagrams and layout drawings of the switchgear and transformer were received and reviewed internally.

(3) The subsystem writeup was completed and submitted for approval.

(4) An antimotoring relay for protection of the diesel-generator was included in the 480-volt switchgear.

(5) The portion of the one-line diagram covering the switchgear was completed.

(6) A 120-volt power system was designed for feeding heaters, fans and other fractional horsepower motors from circuits in the 120/208-volt panel.

Planned objectives during the next quarter are as follows:

(1) The final weight and center of gravity data will be obtained for the shipping configuration.

(2) Contractor liaison will be maintained under Task 7.2.

Y. SUBSYSTEM 25--STATION SERVICE SYSTEM

J. McNeil

The station service system controls and distributes electric power to plant auxiliaries. Motor starters and heater contactors are mounted in a motor control center. Manual switches or automatic control contacts in a remote console yield signals to control the contactors in the motor control center. A small number of plant auxiliaries are equipped with local controls and starters which are powered from an auxiliary distribution system.

The objectives for the system during the quarter were:

(1) Completion of the motor control center specification.

(2) Preparation of the subsystem description and supporting data and submittal for approval.

(3) Completion of the motor control center layout.

(4) Completion of the electrical one-line diagram.

The objectives completed during the quarter in support of the subsystem final design were:

(1) The specification of the motor control center was completed. A few primary loop requirements must be finalized to determine the exact quantity and rating of the starters.

(2) The subsystem writeup was completed and submitted for approval.

(3) The motor control center layout was received and drawings for mounting in the switchgear package completed.

(4) The electrical one-line diagram was brought up to date.

The planned objectives for the next quarter are as follows:

(1) The electrical one-line diagram will be completed.

(2) Weight and center of gravity data will be obtained for the shipping configuration.

(3) Contractor liaison will be maintained under Subtask 7.2.

Z. SUBSYSTEM 26--LIGHTING SYSTEM INCLUDING EMERGENCY LIGHTING

J. McNeil

The lighting system provides illumination through all plant areas, indoor and outdoor, when energized from the turbine-generator, diesel-generator or site power supply, or from batteries when all a-c power is down. Also included in this system are the 115-volt, 1-phase convenience outlets which are fed from the lighting panel.

The planned objectives during the quarter in support of final design were:

(1) Completion of lighting calculations.

(2) Completion of the lighting layout.

(3) Selection of lighting circuits.

(4) Preparation of the subsystem description and supporting data and submittal for approval.

Those efforts completed toward the subsystem final design during the quarter were:

(1) Lighting levels and fixture types were selected.

(2) The lighting and convenience outlet layout was completed, based on suggested mounting heights.

(3) Circuits were selected from the lighting panel.

(4) System description and supporting data writeup were completed and submitted for approval.

The planned objectives during the next quarter are:

(1) Revisions to layouts will be made to reflect manufacturers' certified equipment drawings.

(2) Vendors will be selected and liaison established under Subtask 7.2.

A.A. SUBSYSTEM 27--PLANT D-C SYSTEM

J. McNeil

The plant d-c system furnishes a source of direct current and distributes it to the diesel-generator starter, switchgear, solenoid valves, vital power supply and other loads requiring high reliability of power supply.

Those efforts planned for the fourth quarter included:

(1) Completion of the station battery specification.

(2) Preparation of the subsystem description and supporting data and submittal for approval.

(3) Commencement of subsystem direct current circuit design.

Accomplishments in support of the subsystem objectives for the quarter were:

(1) The station battery specification was completed and a manufacturer selected.

(2) The system description and supporting data were completed and submitted for approval.

(3) Design of the d-c distribution system was initiated.

Objectives during the next quarter for this subsystem will be:

(1) The d-c distribution system will be completed and submitted for approval.

(2) Contractor liaison will be maintained under Subtask 7.2.

B.B. SUBSYSTEM 28--EMERGENCY POWER SYSTEM

J. McNeil

The emergency power system encompasses 2 sources. The first is an uninterrupted power supply and the second is a power supply feeding loads which can stand a short interruption. The uninterrupted supply serves loads from a vital bus. An a-c system or m-g set feeds this vital bus. This system supplies power to nuclear and other vital instruments. The second emergency source is the 200-kw (sea level rating) diesel-generator. This unit has a 150-kw capacity at a 6500-foot elevation, sufficient to start up the nuclear plant while furnishing lights, compressed air and other services.

Objectives planned for the subsystem during the quarter were:

(1) Completion of the diesel-generator specification and submittal for approval.

(2) Completion of the d-c bus and a-c vital bus equipment specification and submittal for approval.

(3) Preparation of the subsystem description and supporting data and submittal for approval.

(4) Requests for vendor bid proposals on system components.

The efforts completed during the quarter in support of the system final design were:

(1) The diesel-generator specification was completed and submitted for approval. Invitations to vendors for bidding purposes were made and the received bids evaluated. A vendor was selected.

(2) The d-c bus and a-c vital bus equipment specification was completed and submitted for approval. Contractor bids were received and evaluated and a vendor selected. The unit is a d-c motor receiving power from a static charger to drive an alternator. This alternator is paralleled with the motor control center bus to feed the vital a-c loads. When the motor control center bus becomes de-energized, the d-c motor continues to run, powered by the battery, and the alternator continues to supply the vital a-c bus. A contactor opens to isolate the alternator so that it will not try to feed the motor control center. A duplicate static charger is supplied to ensure reliability.

(3) The subsystem description and supporting data were completed and submitted for approval.

(4) A reverse power relay was added to the distribution system to protect the diesel-generator unit from motoring.

(5) The a-c vital power circuits were completed.

Objectives during the next quarter for the subsystem are:

(1) Wiring diagrams will be completed to indicate connections and terminal block numbers and will be submitted for approval.

(2) Contractor liaison will be maintained under Subtask 7.2.

C.C. SUBSYSTEM 29--WATER TREATMENT SYSTEMS

R. Groscup

A chemical water treatment system is provided to control the feedwater chemistry in the secondary system. This feeding of chemicals (phosphate, sulfite, caustic) into the feedwater will control water conditions (pH, dissolved oxygen and solids).

During the quarter, the following planned efforts were accomplished in support of the subsystem final design:

(1) The chemical treatment equipment specification was revised and resubmitted for approval.

(2) Invitations were presented to manufacturers for bid proposals on system components, bids were evaluated and a vendor selected.

(3) The system description and supporting data were submitted for approval.

For the next quarter, the review of certified drawings from the selected manufacturer will start under Subtask 7.2 and revisions will be made to reflect contractor's certified equipment drawings.

This subsystem, as shown in Fig. 2, "Secondary Flow Diagram" in the Third Quarterly Progress Report, has been changed. The system at that time supplied morpholine and hydrazine to the secondary system. Laboratory tests were performed on aluminum of the type to be used in the PM-1 condenser by Westinghouse Corporation. This test indicated excessive corrosive effects on aluminum by the ammonia gases formed by decomposition of hydrazine in the feedwater. Although the test conditions were more severe than anticipated in the PM-1 plant, it was decided to use sulfite and caustic in the treatment of secondary system feedwater.

The physical location of this chemical feed equipment was changed during the quarter from the decontamination package to an area adjacent to the heat transfer apparatus package. The equipment will be shipped and mounted on its own base.

D.D. SUBSYSTEM 30--CONDENSATE MAKE-UP SYSTEM

L. Hassell

During this quarter, the following planned work was accomplished:

(1) The evaporator-reboiler specification was completed, submitted to AEC for approval and sent to vendors for bidding.

(2) The subsystem description and supporting data were developed and submitted to the AEC for approval.

(3) The layout of the evaporator-reboiler in the heat transfer apparatus package was completed by Gibbs and Hill (see Task 16) except for minor dimensional changes dependent upon vendor's certified drawing.

(4) The vendor was selected for the evaporator-reboiler.

During the next quarter:

(1) The evaporator-reboiler fabrication will start.

(2) The subsystem layout will be completed.

(3) Vendor liaison will be conducted under Subtask 7.2.

During this quarter, the major effort in this subsystem was the preparation of the final evaporator-reboiler specification and the subsystem writeup. Also, Gibbs and Hill completed the layout of the evaporator-reboiler in the heat transfer apparatus package except for minor dimensional changes dependent upon vendor's certified drawings.

E.E. SUBSYSTEM 31--FIRE PROTECTION SYSTEM

W. Koch, H. Clark

During this quarter, the following planned work was accomplished:

(1) The fire protection system requirements, equipment requirements and layout were completed.

(2) The subsystem writeup and supporting data were completed for submittal to the AEC for approval.

During the next quarter:

(1) Data sheets for the equipment in this subsystem will be written and sent out to vendors for bids.

The purpose of the PM-1 Fire Protection System is to provide the necessary and proper protection for plant personnel and equipment against fire and associated hazards.

The following Underwriters' Laboratory approved equipment will be furnished for the PM-1 Plant:

(1) Four dry chemical portable fire extinguishers--10-pound capacity, nitrogen filled.

(2) Thirteen carbon dioxide portable fire extinguishers--10-pound capacity.

(3) One wheeled portable fire extinguisher, dry chemical type--75-pound capacity.

(4) One stretcher with metal container, blanket and waterproof sheet.

(5) Two gas masks.

(6) One first aid kit.

F.F. SUBSYSTEM 32--TURBINE EXHAUST SYSTEM

NOTE: Subsystem 32 was formerly entitled "Auxiliary Steam System." The Auxiliary Steam System has now been absorbed into Subsystem 33--Building Heating, Air Conditioning and Ventilation.

During this quarter, the following planned work was accomplished:

(1) The subsystem writeup and supporting data were completed for submittal to the AEC for approval.

(2) The layout of this subsystem was completed except for minor dimensional changes required due to vendors' certified drawings.

During the next quarter:

(1) The layout of this subsystem will be completed.

This subsystem consists of:

(1) The main turbine exhaust line from the turbine exhaust flange to the inlet flange of the condensers.

(2) The condensate pump turbine drive exhaust from the exhaust flange to the main exhaust line.

(3) The boiler feed pump turbine drive exhaust flange to the main exhaust line.

(4) The desuperheater station.

Layout of this equipment has been completed by Gibbs and Hill (see Task 16) except for minor dimensional changes due to vendors' certified drawing information.

G.G. SUBSYSTEM 33--BUILDING HEATING, AIR CONDITIONING AND VENTILATING SYSTEMS

During the quarter, the following planned work was accomplished:

(1) The subsystem description and supporting data were developed for submittal to the AEC for approval.

(2) The system design was completed except for detailed layout.

(3) The equipment data sheets were prepared and sent to vendors for bids.

During the next quarter:

(1) The layout will be completed.

(2) Vendors will be selected and liaison will be established under Subtask 7.2.

During this quarter, a detailed analysis of the PM-1 heating, air conditioning and ventilating systems was performed, utilizing the latest building and equipment data. These calculations were based upon the Heating, Ventilating and Air Conditioning Guide, 29th edition.

Tables III-2 through III-7 show the system design data summary for this subsystem.

TABLE III-2

Primary Building

Primary building overall "U" for walls and roof, Btu/hr-ft^2-°F	0.21
Heating system design temperature, °F	-55
Elevation (design basis), ft	sea level
Inside building temperature, °F	60
Heat loss, walls and roof, Btu/hr	57,500
Heat loss, floor (U = 0.2 earth temperature 40° F), Btu/hr	2,600
Number of air changes per hour due to infiltration	1
Heat load due to infiltration, Btu/hr	20,760
Heat load due to two air changes per hr, Btu/hr	41,520
Total Btu/hr	122,380

TABLE III-3

Secondary Building

Secondary building overall "U" for walls and roof, Btu/hr-ft^2-°F	0.125
Heating system design temperature, °F	-35
Elevation (design basis), ft	6500
Inside building temperature, F	70
Wall and ceiling heat loss, Btu/hr	65,000
Air changes per hour due to infiltration	1

TABLE III-3 (continued)

Heat load due to infiltration, Btu/hr	45,500
Floor heat loss (including edge loss), Btu/hr	17,500
Heat load for two air changes per hr, Btu/hr	91,000
Total Btu/hr	219,000

TABLE III-4

Decontamination Building

Decontamination building overall "U" for walls, roof and floor, Btu/hr-ft-°F.	0.1
Heating design temperature, °F	-55
Elevation (design basis), ft	sea level
Inside building temperature, °F	70
Heat loss, walls, roof and floor, Btu/hr	14,300
Number of air changes per hour due to infiltration	1
Heat load due to infiltration, Btu/hr	4300
Heat load for 6 air changes per hour, Btu/hr	25,800
Total Btu/hr	44,400

TABLE III-5

Control Room

The control room walls, ceiling and floor area were included in the secondary building calculations. Since this is an enclosed area, a separate heating system is provided. Control room overall "U" for outside walls only (Arctic basis), Btu/hr-ft^2-°F	0.10
Heating design temperature, °F	-55
Elevation (design basis), ft	sea level
Inside design temperature, °F	70
Heat loss through wall, Btu/hr	5000
Heat loss through floor (U = 0.10, T = -55), Btu/hr	4000
Air changes per hour due to infiltration	1
Heat loss due to infiltration, Btu/hr	6520
Heat load for 110 cfm (2-plus air changes per hour) air at -35° F, Btu/hr	10,000
Total Btu/hr	25,520

TABLE III-6

Control Room Air Conditioning Load

Inside design temperature, °F	70
Secondary building temperature, °F	90
Outside temperature, °F	80
Outside wall "U," Btu/hr-ft^2-°F	0.125
Inside wall "U," Btu/hr-ft^2-°F	0.35
No. of people in control room	2
Metabolic rate per person, Btu/hr	850

TABLE III-6 (continued)

Heat gain through outside wall, Btu/hr	500
Heat gain through inside walls and ceiling, Btu/hr	5600
Heat gain from personnel, Btu/hr	1700
Heat gain from console and lights (est max), Btu/hr	24,000 (7 kw)
Total heat gain Btu/hr	31,800

TABLE III-7

Component List and Specifications

Two 60,000-Btu/hr projection-type unit heaters--Primary Building

Four 60,000-Btu/hr horizontal-type unit heaters--Secondary Building

One 15,000-Btu/hr horizontal-type unit heaters--Decontamination Building

One 30,000-Btu/hr horizontal-type unit heaters--Decontamination Building

One 3-ton air conditioner with 40,000-Btu/hr heating coils--Control Room

Two 22.5 gpm at 50-psig discharge pressure evaporator feed pumps

One 1×10^6 Btu/hr (30 hp) oil-fired auxiliary boiler operating at 115-psia pressure.

NOTE: This boiler will provide sufficient steam to furnish space heating steam plus 200-lb/hr steam to the air ejector (for afterheat removal operation) and 384 lb/hr to the waste evaporator simultaneously.

H.H. SUBSYSTEM 34--PRIMARY BUILDING

R. Stuenes, A. Layman

The major objective during the quarter, within this subsystem, was the completion of the detailed primary building size, type and functional requirements. This included establishing the design criteria for the building and initiating Specification MN-7055 and the subsystem description. This was accomplished and vendor contacts were established.

Objectives for the next quarter are completion of specifications for bid requests on the building, evaluation of these bids and final selection of a vendor. Liaison with the selected vendor will continue under Subtask 7.2 for integration of foundation and structural requirements. Also, the completed subsystem description and the specification will be submitted for approval.

Design objectives determined for the building during the quarter were:

(1) The building will be relocatable.

(2) The size and construction will be adequate to furnish a suitable protective shelter for the primary system equipment and maintenance personnel at an Arctic site as well as the site near Sundance.

(3) The building will be of the prefabricated panel type.

(4) The building length will be increased to cover the loading dock area.

During the quarter, the first draft of the Primary System Building Specification (MN-7055) was prepared for internal review. The design criteria established for the primary building are:

(1) Minimum clear internal dimensions are to be 41 feet, 8 inches long, 15 feet, 4 inches wide, and 13 feet, 6 inches high. The maximum external length and width will be 42 feet, 5 inches and 16 feet, 3 inches, respectively. There is no specific limitation on the maximum height, but roof slope is to be 3 in 12.

(2) The building must withstand 30-pounds per square foot snow load and 100-mph wind load.

(3) The average "U" factor of the exterior panels must not be greater than 0.22 Btu/hr-ft^2-°F.

(4) All components will be bolted or screwed together for complete final assembly of the building (no field welding).

(5) Materials used will be noncombustible.

(6) Exterior finishes are to provide a maintenance-free surface for a period of 20 years.

I.I. SUBSYSTEM 35--SECONDARY BUILDING

R. Stuenes, A. Layman

It was planned during the quarter to select the secondary system building requirements and vendor. This included establishing of the building design objectives and criteria, completion of the Building Specification (MN-7054), submittal of the system description for approval, solicitation of bids and designation of a building contractor for a building with a steel frame and prefabricated panels.

Objectives for the next quarter will be maintaining liaison with the contractor and coordinating the certified drawings under Subtask 7.2.

Design objectives determined during the quarter for the building were:

(1) The size and construction will be based on supplying a suitable protective shelter for the secondary system equipment and operating personnel at the site near Sundance.

(2) The building will be a commercial unit, utilizing a steel frame and prefabricated panels.

The Secondary System Building Specification (MN-7054) was prepared and submitted for approval during the quarter. The design criteria established for the building are:

(1) Minimum clear internal dimensions are to be 38 feet wide, 62 feet long and 9-1/2 feet high.

(2) The building must withstand 30-pounds per square foot snow load and 100-mph wind load.

(3) The average "U" factor of the exterior panels must be greater than 0.125 Btu/hr-ft^2-°F (0.120 for the roof).

(4) All components must be bolted or screwed together for complete final assembly of the building, with maximum prefabrication.

(5) Materials used are to be noncombustible.

(6) Exterior finishes are to provide a maintenance-free surface for a period of 20 years.

In addition to the specification, the subsystem description was submitted for approval.

Vendors were contacted for bid proposals on the building. The bids were evaluated and a contractor was selected.

Detailed drawings of the secondary system building will be forwarded to The Martin Company for review during the next quarter.

J.J. SUBSYSTEM 36--DECONTAMINATION BUILDING AND EQUIPMENT

R. Groscup, A. Layman

The purpose of the decontamination building is to provide a weather-tight enclosure for the decontamination equipment and to serve as a link in the enclosed walkway between the primary and secondary buildings. The majority of the equipment items contained within the decontamination building will be bolted to the package floor at The Martin Company plant prior to shipment to the Sundance site and the building itself will serve as its own shipping package.

Objectives for the fourth quarter in support of final design of this subsystem were established as follows:

(1) Completion of the decontamination building design, including details of the walls, floor, roof, equipment mounting positions, etc.

(2) Determination of specific equipment and locations within the building and preparation of specification data sheets for initiation of procurement.

(3) Vendor contacts to be established, requests made for equipment bid proposals, bid evaluations and selection of the equipment vendor(s).

(4) Preparation of the subsystem description and submittal for approval.

During the quarter, design of the building proper was started and by the end of the reporting period, detailing and checking of design drawings neared completion. The building external dimensions were established as 30 feet in length, 8 feet, 6-5/8 inches maximum in height and 8 feet, 8 inches in width. Building shipping weight will not exceed 30,000 pounds. Construction of the building can best be described as sandwich panels for the walls, roof and ends, mounted on a rigid shipping pallet. Structural truss members within the side panels give additional strength and rigidity during shipment. Fireproof or fire retardant construction is used wherever possible. The sandwich panels have been designed for maximum thermal properties and ruggedness during handling and later use. Roof structure is 2 inches thick while the side panels are 3 inches thick. Stainless steel sheet is used for the outer panel cover and aluminum sheet covers the inside of the side panels. The pallet is of welded steel, using conventional rolled shapes of structural steel. The roof panel uses stainless steel skins on both sides with a polystyrene filler (2.5 lb/ft^3).

Resting on the pallet to form the floor is a 3-inch-thick sandwich insulation panel sheathed with aluminum alloy. Imbedded within this floor are wood joists for bolting down the equipment carried within the package. Separate lag bolts will attach the floor to the pallet. Through metal will be avoided in all areas of the roof, walls and floor except at the end panels which will always lead into a covered walkway area.

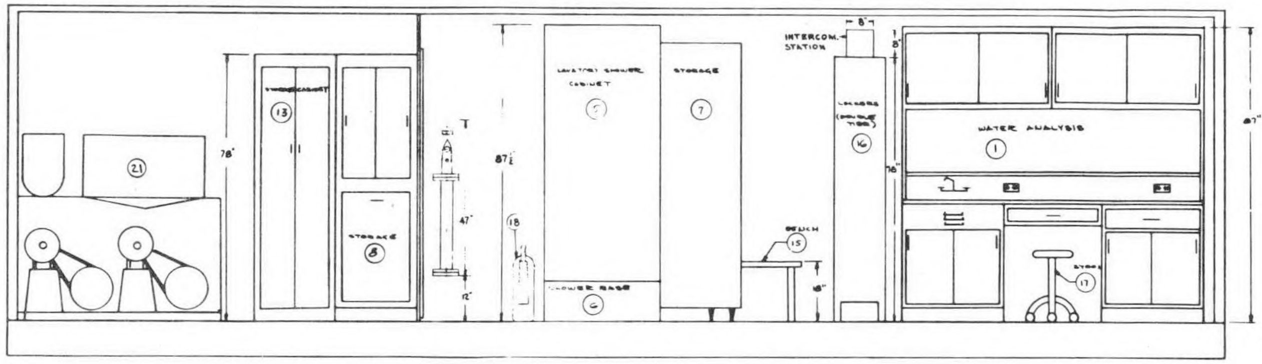
The basic design of the package was completed during the quarter. However, final locations of the pallet stringers and floor joists must await receipt of vendor certified data relative to exact mounting positions of the equipment carried within the package. This will be completed in the next quarter.

The laboratory furniture and cabinets within the package were selected during the quarter after a survey of the equipment available from various manufacturers. An equipment layout of the package floor plan was completed and sufficient aisle space was allowed through the package to move maintenance equipment from the secondary system to the primary system (see Fig. III-14).

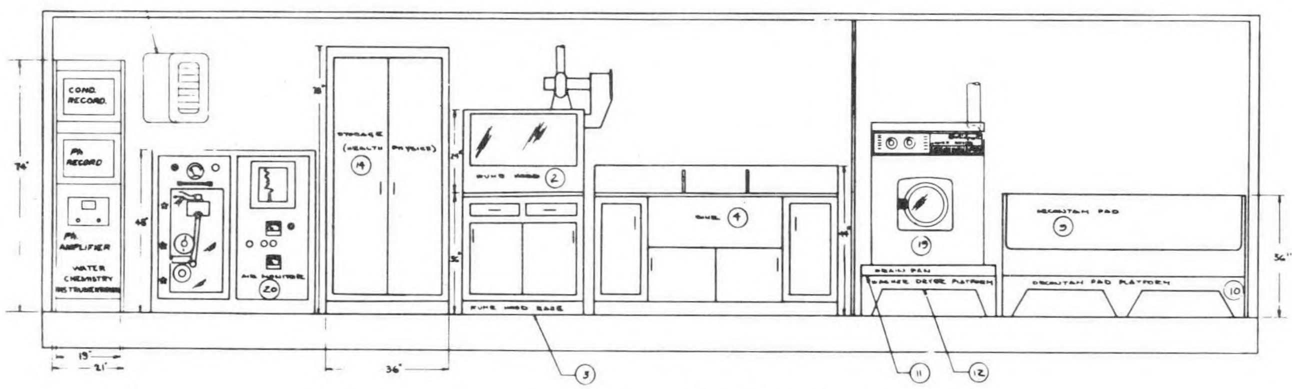
The subsystem equipment writeup and supporting data were submitted for approval. This writeup covered the decontamination package equipment only and was submitted as Subsystem 36A (Subsystem 36B will be submitted next quarter for approval of building structure).

As originally proposed, only a shower stall was considered for the decontamination package for personnel use after working in a "hot" area. No toilet facilities were to be provided in this area. An investigation indicated that a combination lavatory-shower was available and one was selected. This compact unit is a combination of wash bowl, shower and toilet. With this unit in the decontamination building, the lavatory in the maintenance package was eliminated.

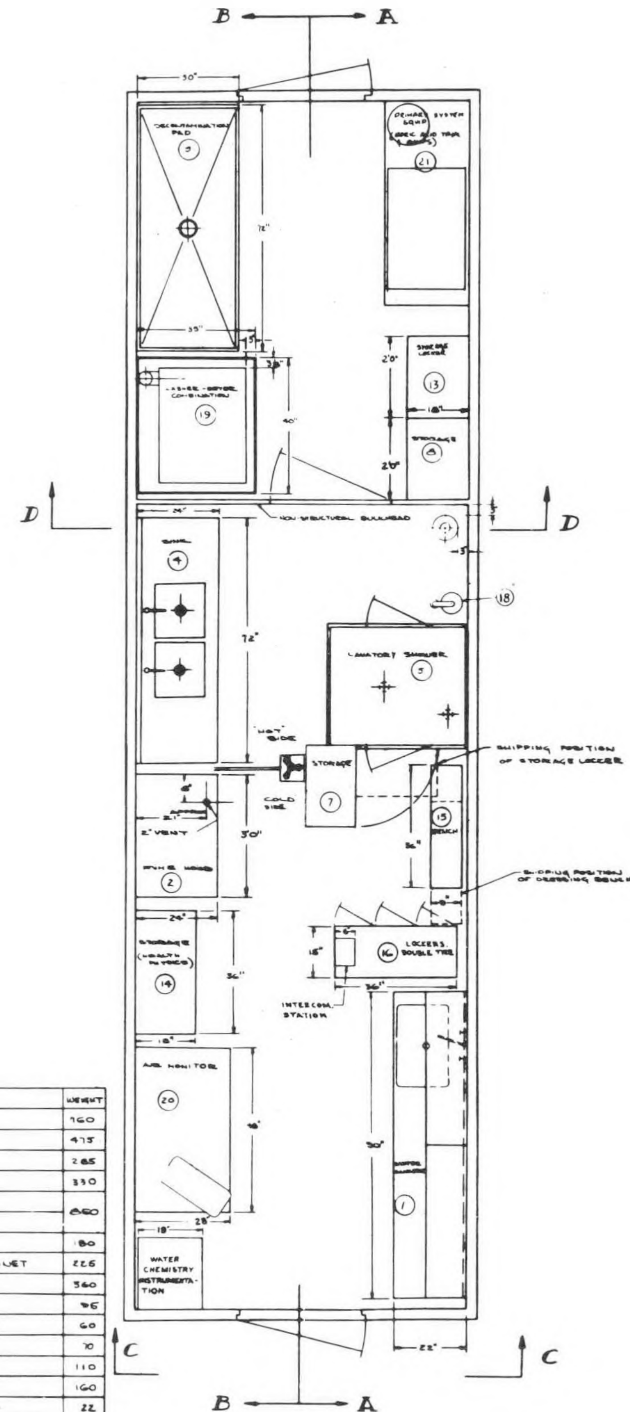
Further study of equipment sizes during the quarter indicated the desirability for removal of the chemical treatment equipment from the package to provide space for the constant air monitor and a cabinet to house the water analysis recording equipment.



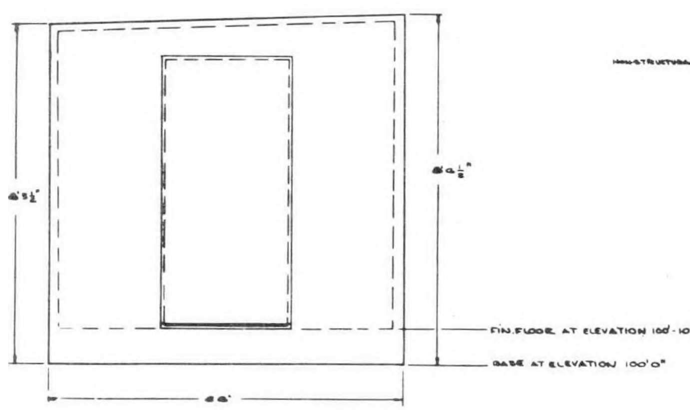
A-A ROTATED 90° COUNTER-CLOCKWISE



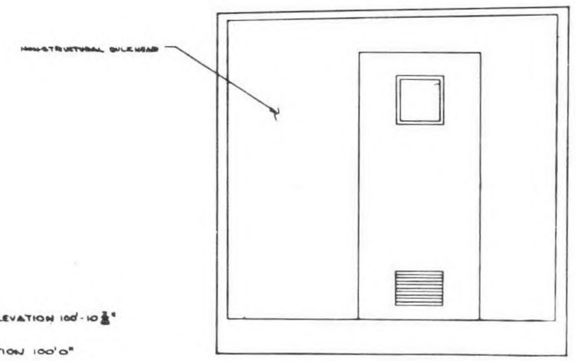
B - B ROTATED 90° CLOCKWISE



ITEM	UNIT NAME	WEIGHT
1	WATER ANALYSIS BENCH	760
2	FUME HOOD	475
3	FUME HOOD BASE	285
4	DOUBLE TUB SINK	330
5	LAVATORY SHOWER CABINET	860
6	SHOWER BASE	
7	STORAGE CABINET	180
8	CONTAM. CLOTHING STORAGE CABINET	226
9	DECONTAMINATION PAD	560
10	DECONTAM PAD PLATFORM	95
11	PAN DRAIN FOR WASHER-DRYER	60
12	WASHER-DRYER PLATFORM	70
13	STORAGE CABINET	110
14	STORAGE CABINET	160
15	DRESSING BENCH	22
16	DOUBLE TIER LOCKERS	85
17	STOOL	20
18	FIRE EXTINGUISHER	50
19	COMBINATION WASHER-DRYER	330
20	AIR MONITOR	300
21	CHARGING PUMPS & TANKS	1250



C - C



HEATER AND SINK OMITTED FOR CLARITY

D - D

Fig. III-14. Equipment Layout--Operating Positions, Decontamination Package

A water heater was added to the decontamination package to provide hot water for the laboratory sinks and showers.

Effort planned for the next quarter will be the completion of package design and detailing, submittal of subsystem writeup and data for approval, initiation of procurement of system equipment, revisions to reflect manufacturers' certified drawings for installation and shipping, preparation of detailed piping drawings for connection of the system equipment. Contractor liaison will be maintained under Subtask 7.2.

K.K. SUBSYSTEM 37--MAINTENANCE ITEMS AND TOOLS

R. Groscup

During the quarter, the following planned objectives in support of the system final design were accomplished:

(1) The preparation of maintenance tool lists by categories; i.e., machine shop, electrical, welding, general maintenance, card instrument tools and equipment.

(2) Establishment of a maintenance philosophy to determine the overall requirements for both the primary and secondary systems.

(3) Vendor solicitation for bids and estimates of the shipping packages and locations required for the small tools.

(4) Initiation of the equipment layout within the maintenance package, including sizes and weights of permanent maintenance shop equipment.

(5) Submittal of the subsystem description and supporting data for approval.

During the next quarter, efforts planned within this subsystem include the following:

(1) The completion of the maintenance package layout drawing.

(2) The completion of the shipping configuration drawing to indicate the crated tools shipped in this package.

(3) The initiation of equipment procurement for this subsystem and the maintaining of vendor liaison under Subtask 7.2.

During the quarter, design changes within the system were of a limited nature. The rest room was relocated from this package to the decontamination package, as explained under Subsystem 36. Also, two 10-hp instrument air compressors, a dryer and filter are now located in this package.

L.L. TASK 4--FINAL DESIGN SUPPORTING STUDIES

1. Nuclear Analysis Studies

E. Scicchitano, R. Hoffmeister, E. O'Farrell, P. Gilmore

Nuclear analysis performed during the fourth quarter consisted of a continuation of the design analysis studies and the completion of the final design specification studies. These included:

(1) Core lifetime studies.

(2) Lumped burnable poison studies.

(3) Control studies.

(4) Temperature coefficient studies.

(5) Xenon buildup studies.

Nonuniform burnup studies to determine the effective multiplication as a function of core life for cold clean and operating core conditions were completed. The effect of fuel and burnable poison loading specification tolerances on core life and control requirements was obtained. Studies to determine lumped poison rod specifications were completed. The control rod material specifications were established. The worths of 3-, 4-, 5- and 6-rod banks were determined. The variation of total thermal macroscopic absorption cross section of europium with time was evaluated. Worth versus insertion for the 3- and 6-rod banks was calculated.

The temperature dependence of reactivity as a function of core life was evaluated. The buildup of equilibrium xenon and maximum buildup after shutdown were evaluated for 0, 400 and 800 days. Intercell flux distribution was also determined.

During the next quarter, the design analysis studies will be completed.

2. Core Specifications and Nuclear Analysis Design Characteristics

The PM-1 final core design was established and the nuclear characteristics evaluated. The core nuclear design characteristics are summarized in Table III-8.

A top view of the final design core showing the basic core configuration, control rod locations, lumped poison rod locations, shroud, thermal shields and pressure vessel is shown in Fig. III-15.

TABLE III-8

Nuclear Analysis Design Characteristics

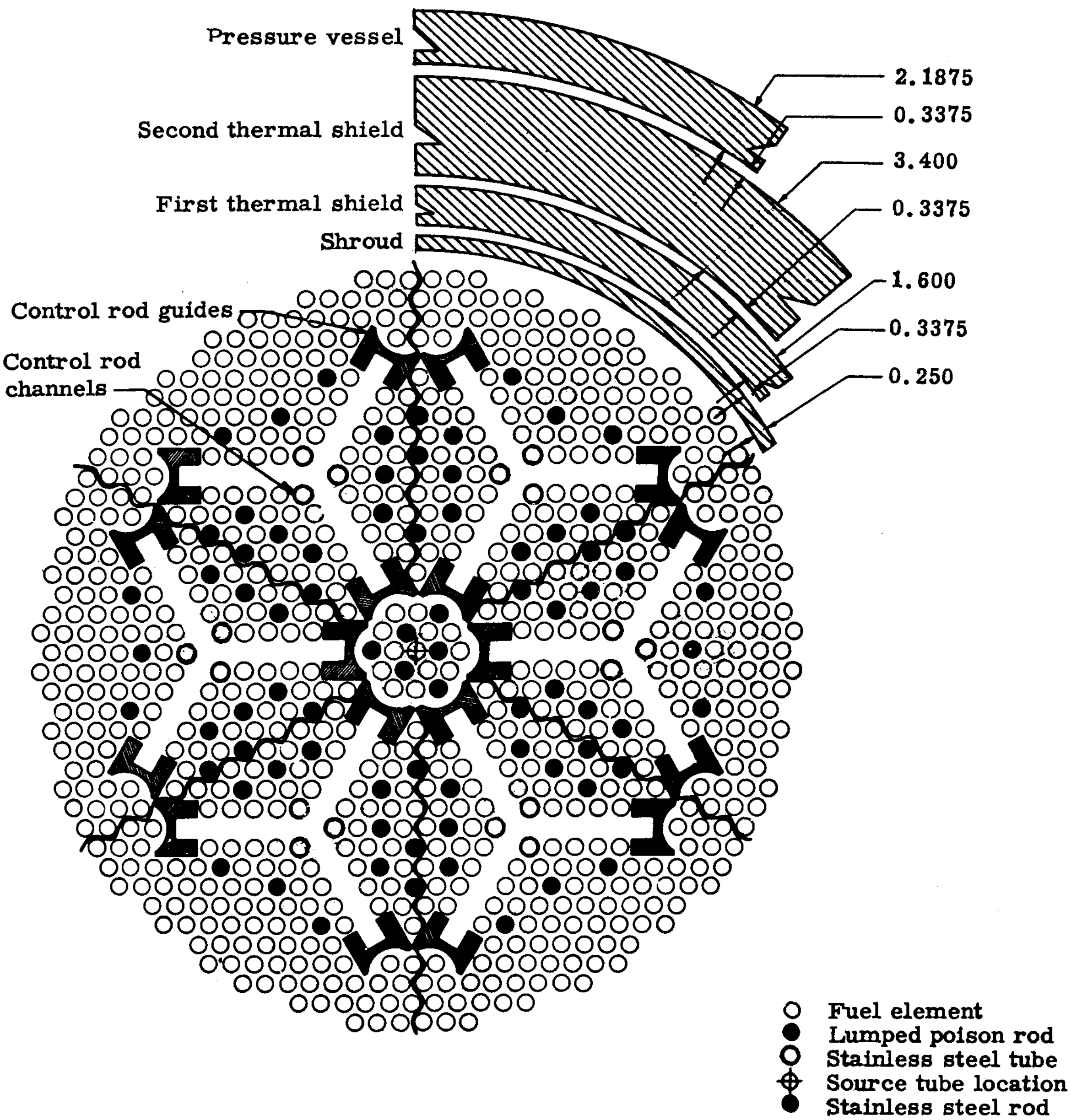
Overall Performance Data	
Reactor thermal power output	9.37 Mw
Core life requirements	18.74 Mw-yr
Design core life	19.2 Mw-yr
Average core coolant temperature	463° F
Core Design Characteristics	
Geometry, right circular cylinder (approximately)	
Diameter	23 in.
Equivalent core diameter	22.48 in.
Height (active)	30.0 in.
Moderator, coolant and reflector	H_2O at 1300 psia
U-235 inventory	28.255 kg
U-235 burnup (18.74 Mw-yr)	9.0 kg
Fuel Element Data	
Tubular, cermet type	
Pitch, triangular	0.665 in.

Outside diameter	0.500 in.
Inside diameter	0.416 in.
Clad thickness	0.006 in.
Clad material	Modified AISI 347 SS
Meat thickness	0.030 in.
Meat composition	
U-235	38.6 gm
UO_2	26.12 wt%*
SS (AISI 304)	73.88 wt%*
Length of fuel element (active)	30.0 in.
Number of fuel elements	732

Control System Characteristics

Control Rods	
Number of Y-shaped rods	6
Composition	Eu_2O_3 in SS (30 wt% Eu_2O_3)
Length (absorber)	32.0 in.
Blade width (absorber)	3.5 in.
Blade thickness (absorber)	0.25 in.
Worth of 6 rods	-31.8% $\Delta\rho$
Worth of 5 rods	-16.5% $\Delta\rho$

*Based on theoretical densities of material; actual densities will be determined later pending packing fraction data.



Mean Diameters

Equivalent core diameter	22.48 in.
Outer edge of outer fuel element	22.80 in.
Shroud ID	23.100 in.
Shroud OD	23.600 in.
First thermal shield ID	24.275 in.
First thermal shield OD	27.478 in.
Second thermal shield ID	28.150 in.
Second thermal shield OD	34.950 in.
Pressure vessel ID	35.625 in.
Pressure vessel OD	40.000 in.

Active Core Height	30.000 in.

Number of fuel elements	732
Number of lumped poison rods	75
Fuel element spacing (except across split line)	0.665 in.
Core loading U-235	28.255 kg
Number of control rods	6

Fig. III-15. PM-1 Core Configuration (Top View)

Worth of 4 rods (minimum worth)	-9.3% $\Delta\rho$
Worth of 3 rods (every other rod)	-10.8% $\Delta\rho$
Lumped Poison Rods	
Number	75
OD	0.50 in.
Material	Boron steel
Composition	0.04 gm of natural boron per cc

Average Core Neutron Fluxes	Initial	2 Years
Fast flux (2.5×10^4 - 1×10^7 ev)	5.1×10^{13}	5.5×10^{13} nv
Epithermal flux (0.056 - 2.5×10^4 ev)	3.3×10^{13}	3.6×10^{13} nv
Thermal flux (0.056 ev)	0.7×10^{13}	1.4×10^{13} nv
Average Temperature Coefficients		
Overall (68° F to 463° F)	1.2×10^{-4}	0.9×10^{-4} $\Delta\rho$/°F
Operating temperature	2.1×10^{-4}	1.9×10^{-4} $\Delta\rho$/°F

3. Core Life Studies

Nonuniform burnup studies to determine the reactivity as a function of core life for the final design core and the effect of specification tolerances on the fuel ($^{+4\%}_{-0}$) and burnable poison (±10%) loadings were completed.

Four separate cases were evaluated for these studies:

(1) A core containing the design fuel and poison loading.

*Based on theoretical densities of material; actual densities will be determined later pending packing fraction data.

(2) A core containing +4% initial fuel loading per fuel element.

(3) A core containing +10% initial boron loading per lumped poison rods.

(4) A core containing +4% fuel per fuel element and -10% boron loading per poison rod.

The effective multiplication factor, k_{eff}, as a function of time for the above four cases, is shown in Fig. III-16.

The corresponding cold clean k_{eff} versus time curves for Cases 1, 2 and 4 are shown in Fig. III-17.

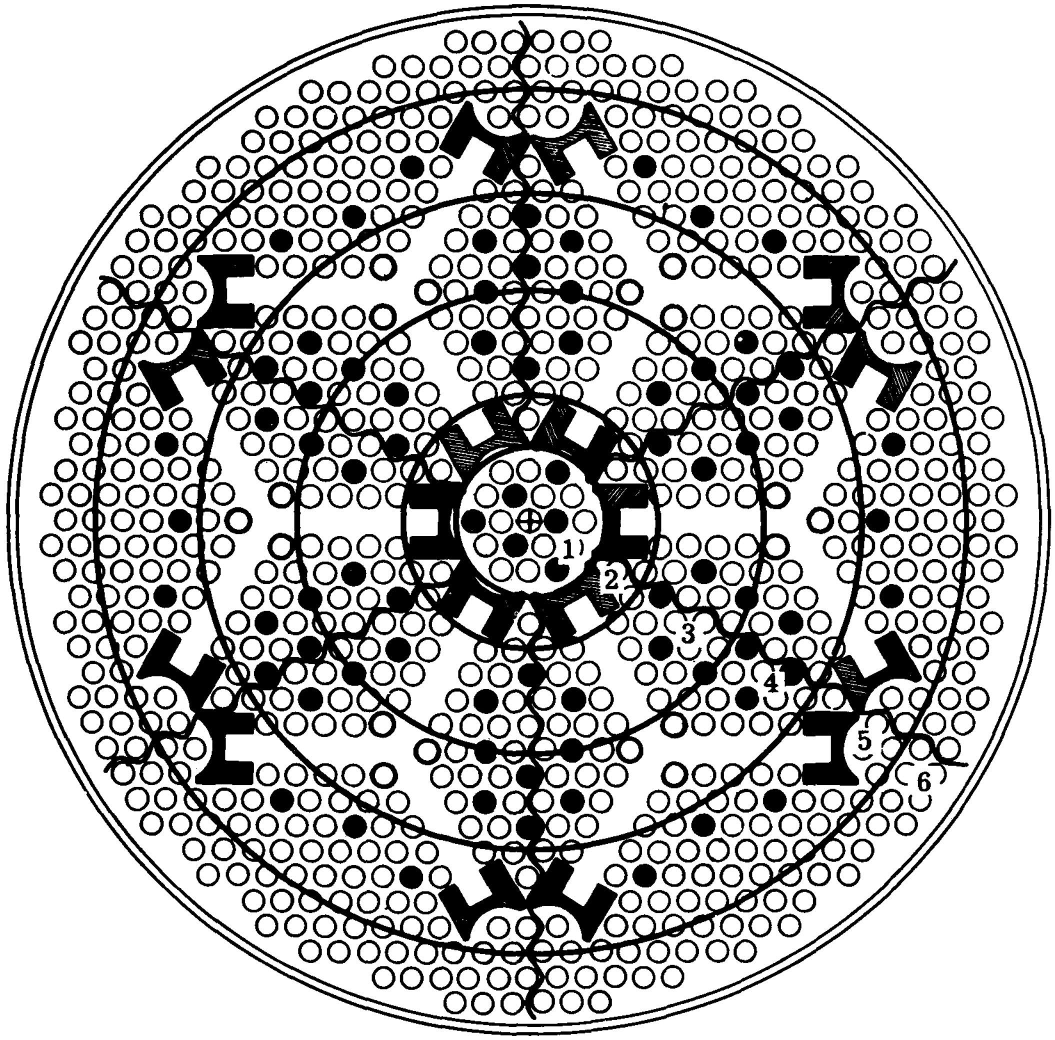
k_{eff} versus core life was calculated considering spatial nonuniform burnup of fuel and burnable poison and nonuniform buildup of fission product poisons. This was done using the three-group, one-dimensional, miltiregion burnup code as described in the Third Quarterly Progress Report (MND-M-1814). The core was divided into 6 regions in both the radial and axial directions. The initial material concentrations in the axial direction are the same in each region. A top view of the core, showing the 6 radial regions considered, plus a table summarizing the regional data are shown in Fig. III-18.

The core life for the final design core (Fig. III-15) as shown by Curve 1 in Fig. III-16 is 765 days at 9.35 megawatts. The effect of +4% initial fuel loading is an increase of ≈ 75 days in core life. The increase in k_{eff} at operating conditions varies from 0.010 to 0.016. The effect of +10% in initial boron loading in the lumped poison rods is a decrease of ≈ 15 days in core life. This is due to the additional boron which has not burned out resulting from the increased initial loading. Case 3 (+10% poison loading) represents the minimum k_{eff} versus time and core life conditions considering both the fuel and poison loading tolerances. A core containing +4% fuel and -10% poison represents the highest k_{eff} versus time and core life. In this case, there is an increase in core life due to both the increase in fuel and decrease in boron. From Curves 1, 3 and 4 of Fig. III-16, it is seen that the increase in core life is due primarily to the increased fuel loading. The increase in k_{eff} for Case 4 above Case 1 is ≈ 0.019.

4. Lumped Burnable Poison Studies

Final design lumped burnable poison studies consisted of analyses to determine the size and poison concentration specifications optimized within the limits of the overall core design to give minimum control requirements and maximum core life.

The results of studies which showed the relative effect on core life and control requirements for lumped poison rod systems of different sizes and poison concentrations (given in the Third Quarterly Progress Report) were evaluated. Based on this evaluation, a boron concentration of 0.04 gm/cc and a rod OD of 0.50 inch (includes an 0.007-inch stainless steel clad) were established. Nonuniform burnup studies for this system were performed. The results are shown in Fig. III-16 and III-17 which were described above (core lifetime studies).



Core description

Region	1	2	3	4	5	6
○ Fuel element	12	6	126	156	180	252
● Lumped poison rods	3	0	24	30	18	0
● Stainless steel rods (center bundle structure)	3	0	0	0	0	0
○ Stainless steel tubes	0	0	0	18	0	0
Control rod guides	0	6	0	0	12	0
⊕ Source	1	0	0	0	0	0

Fig. III-18. Nonuniform Burnup Studies--Core Description

Recent mechanical design studies have indicated that the lumped poison rods need not be clad. The effect of removing the clad and still maintaining the same rod OD will be a slight decrease in core life. From results of the evaluation studies described above, this decrease is $\approx$ 15 to 20 days. If desired, the decrease in core life can be compensated for by a decrease in boron concentration or by reducing the rod OD. Because of the flexibility of the poison system with regard to number, size and concentration, additional detailed studies were not performed at this time. These studies, when made, will include an evaluation of the results of the Flexible Zero Power Test which includes an evaluation of 3 different poison concentrations.

5. Rod Design and Worth

Final design of the control rods was completed during the quarter. The control rod poison material is Eu_2O_3 in stainless steel and is clad with 1/32-inch stainless steel. The poison section of each arm of the rods was 0.25 inch x 3.50 inches x 30.0 inches. The length of the poison section has been changed to 32.0 inches. The final rod design is shown in Fig. III-19.

Studies to establish the europium concentration specifications for the rod design described above were completed. The worths of 4-, 5- and 6-rod banks for rods with 15 and 30 wt % Eu_2O_3 in SS were determined. These results were as follows:

	Rod Bank Worth (% Reactivity)	
Rod Bank	15 wt % Eu_2O_3	30 wt % Eu_2O_3
6 rods	26.7	31.8
5 rods	14.9	16.5
4 rods (adjacent)	8.6	9.3
3 rods (every other rod)		10.8

The increase in concentration of Eu_2O_3 from 15 to 30 wt % results in a change of $\approx$ 8 to 19% in rod bank worth. This increased worth is due primarily to the increase in epithermal absorptions and was considered significant. Based on these results and a preliminary evaluation of the control requirements and stuck rod conditions, a europium oxide concentration of 30 wt % was established. The worth of a 3-rod bank for this design concentration is also given in the table above.

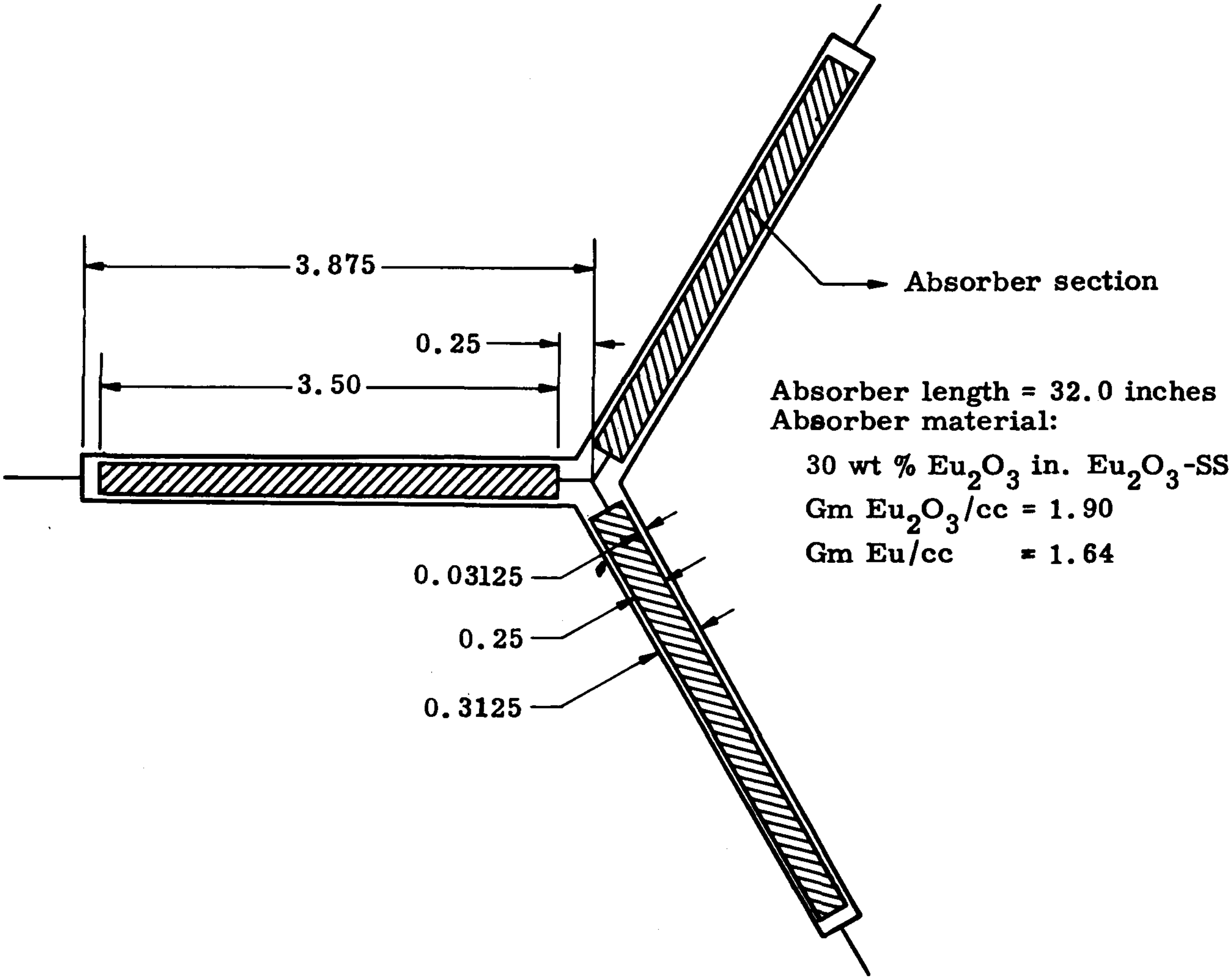


Fig. III-19. Rod Configuration

Control rod worth was calculated using the 2-dimensional, 3-group diffusion code, PDQ, as previously described in the quarterly progress reports.

6. Control Requirements

Design analysis studies to establish control requirements for the final design core were completed. These studies included the calculation of the following:

(1) Cold, clean core reactivity as a function of time.

(2) Hot operating condition core reactivity as a function of time.

(3) Effect of fuel and poison loading specifications on control requirements.

A detailed evaluation of stuck rod conditions in terms of rod worth and control requirements is almost completed. These results will be obtained in the next quarter.

7. Worth Versus Insertion

Studies to determine the worth versus insertion for the 3- and 6-rod banks were completed. Results are shown in Fig. III-20.

The relative effectiveness of the rod banks as a function of the axial position in the core was calculated using a "window shade" model technique. First, a concentration of poison in the core that resulted in a decreased k_{eff} equal to that resulting from the full insertion of the rod banks was calculated. This concentration of poison was then included with the core materials in the regions of rod bank insertion in a multiregion, 1-dimensional, 3-group calculation. By varying the rodded region height, a series of core reactivities was obtained from which the worth versus insertion was calculated.

8. Temperature Coefficient Studies

The reactivity of the core as a function of temperature from 68° F to 473° F was calculated for several times in core life. Nuclear, density and burnup effects were considered.

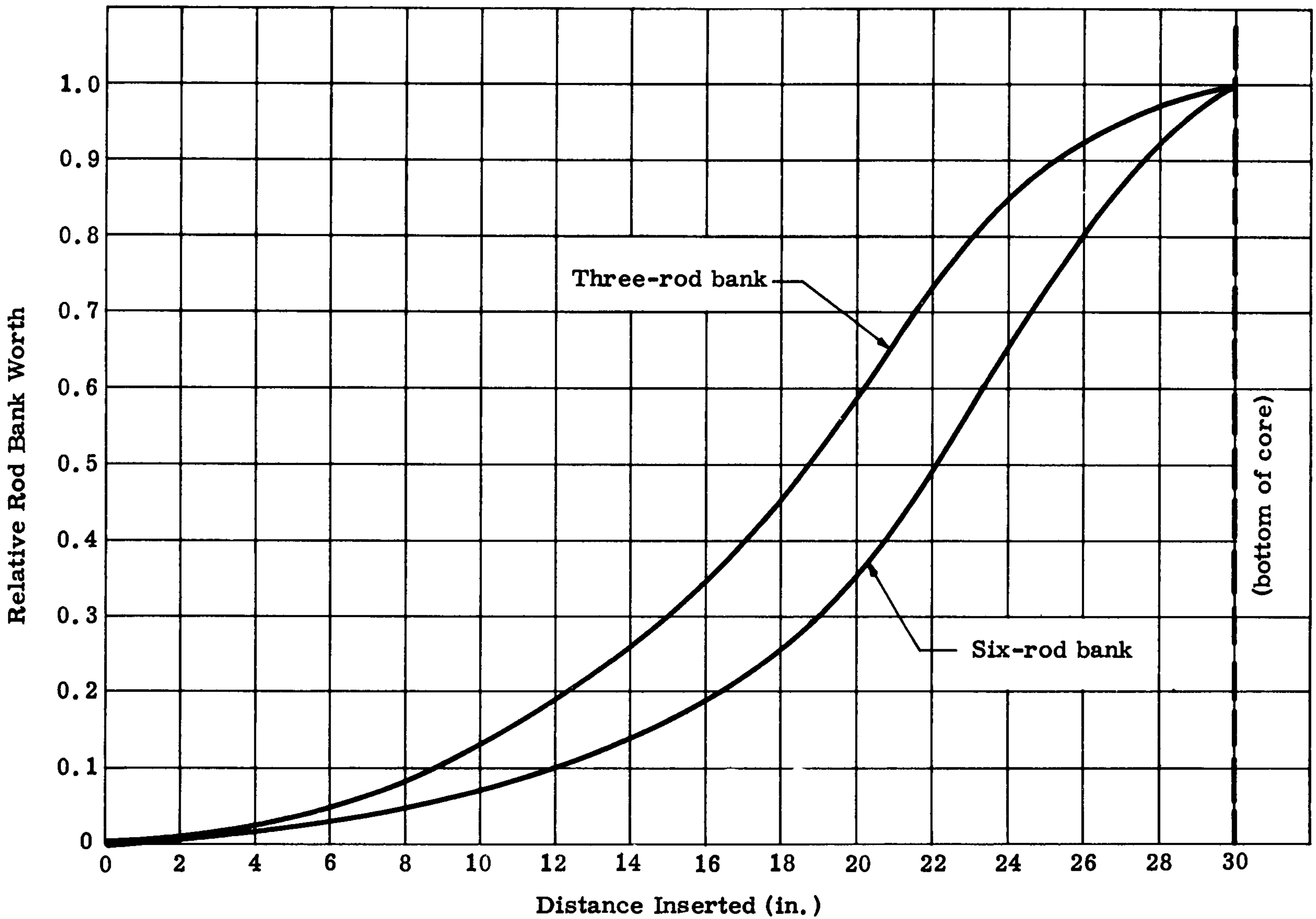


Fig. III-20. Relative Rod Bank Worth Versus Insertion

The variation of reactivity with temperature is shown in Fig. III-21 for the different core conditions investigated. Curve 1 is the reactivity versus temperature for initial clean startup. Curves 2 and 3 are reactivity versus temperature for the cold, clean to hot, clean core conditions (i.e., assuming decay of all fission products) for 400 and 800 days in core life. Curves 4 and 5 represent the decrease in reactivity with increase in core temperature from the cold, dirty to hot, dirty core conditions. The values of reactivity for the different temperature and core conditions shown on Fig. III-21 are relative since these studies were performed assuming a uniform distribution of core materials at the different times in core life.

The average temperature coefficients from 68° F to 463° F and the temperature coefficients at operating temperature (463° F) for the different core conditions are as follows:

Core Condition	Overall Temp. Coeff. ($-\Delta\rho/°F$)	Operating Temp. Coeff. ($-\Delta\rho/°F$)
0 days (clean)	1.22×10^{-4}	2.07×10^{-4}
400 days (clean)	1.01×10^{-4}	1.99×10^{-4}
800 days (clean	0.892×10^{-4}	1.87×10^{-4}
400 days (dirty)	1.06×10^{-4}	2.05×10^{-4}
800 days (dirty)	0.967×10^{-4}	1.98×10^{-4}

The overall temperature coefficient decreases from 1.2×10^{-4} to 0.97×10^{-4} $\Delta\rho$ /°F over the 2-year core life. The operating temperature coefficient decreases from 2.1×10^{-4} to 1.98×10^{-4} $\Delta\rho$ /°F over the 2-year period.

9. Xenon Studies

a. Equilibrium xenon

The equilibrium xenon concentration was calculated at each time interval for the core life studies. The concentrations for full-power operation of 9.35 mw at 0, 400 and 800 days are tabulated below. The decreases in reactivity resulting from the xenon are also given.

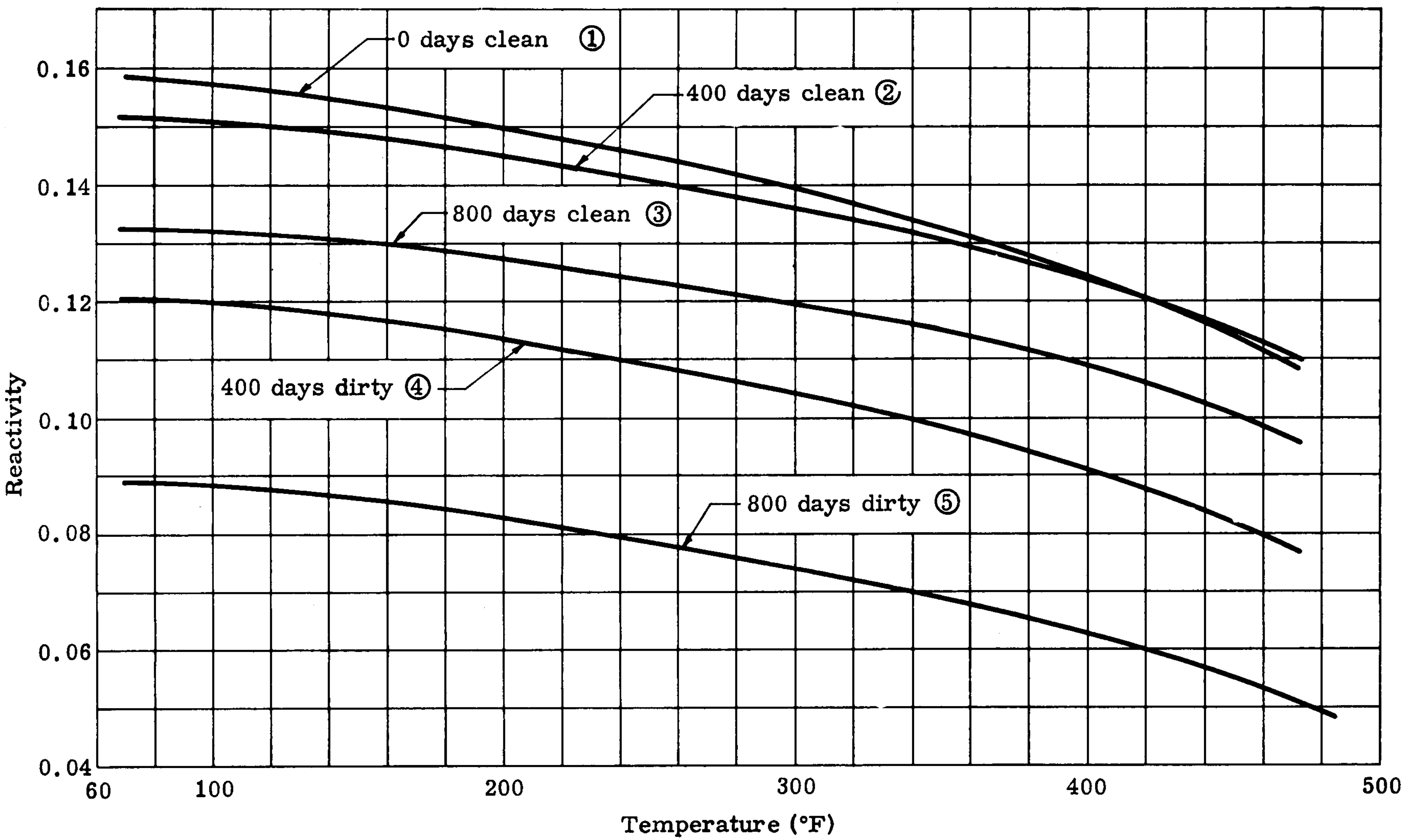


Fig. III-21. Reactivity Versus Temperature--PM-1 Core (see Fig. III-15)

Time (days at full power)	X_o (atoms/cc)	$\Delta\rho$
0	2.33×10^{15}	-0.0146
400	2.11×10^{15}	-0.0165
800	1.83×10^{15}	-0.0188

The equilibrium concentration decreases slightly with time due to a change in the core energy spectrum. However, the reactivity effect increases slightly.

b. Xenon buildup

The buildup at initial startup at full power of 9.35 mw or startup from zero concentration after 400 and 800 days of operation is shown in Fig. III-22. As seen in the curves in Fig. III-22, the zenon concentration increases to an equilibrium value in approximately 50 hours.

c. Maximum xenon

The xenon concentration as a function of time after shutdown is shown in Fig. III-23 for shutdown after 0, 400 and 800 days of operation at full power. As seen in Fig. III-23, the xenon concentration increases for several hours before reaching a maximum value, after which it decreases. The time of peak xenon concentration is approximately 5 hours after shutdown.

The reactivity associated with the additional xenon buildup (above the equilibrium concentration) increases from 0.002 to 0.003 to 0.005 for 0, 400 and 800 days of operation. From these results and the reactivity-versus-time curves (Fig. III-16), it is seen that enough reactivity is available for startup from maximum xenon.

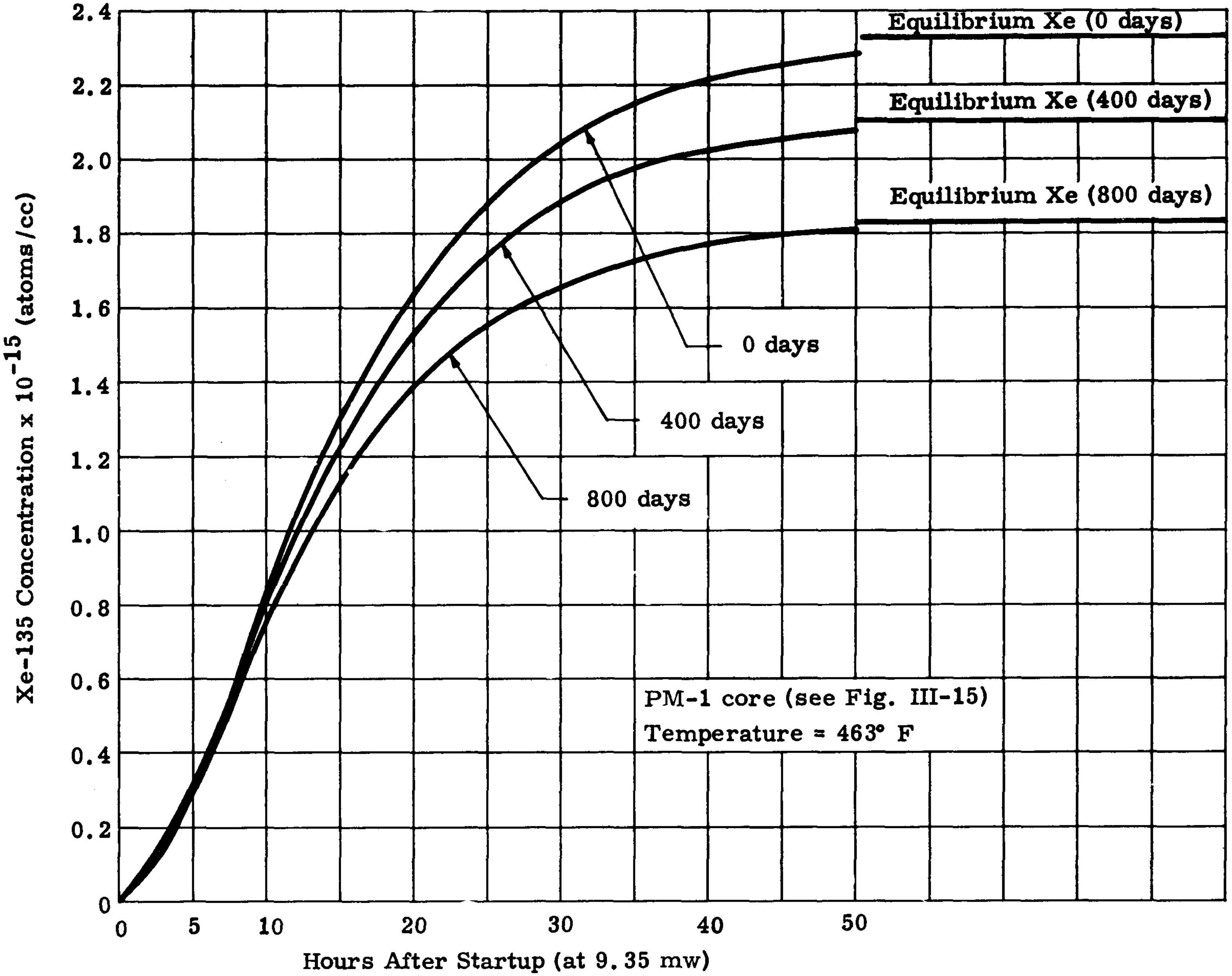


Fig. III-22. Xenon Concentration Buildup After Startup

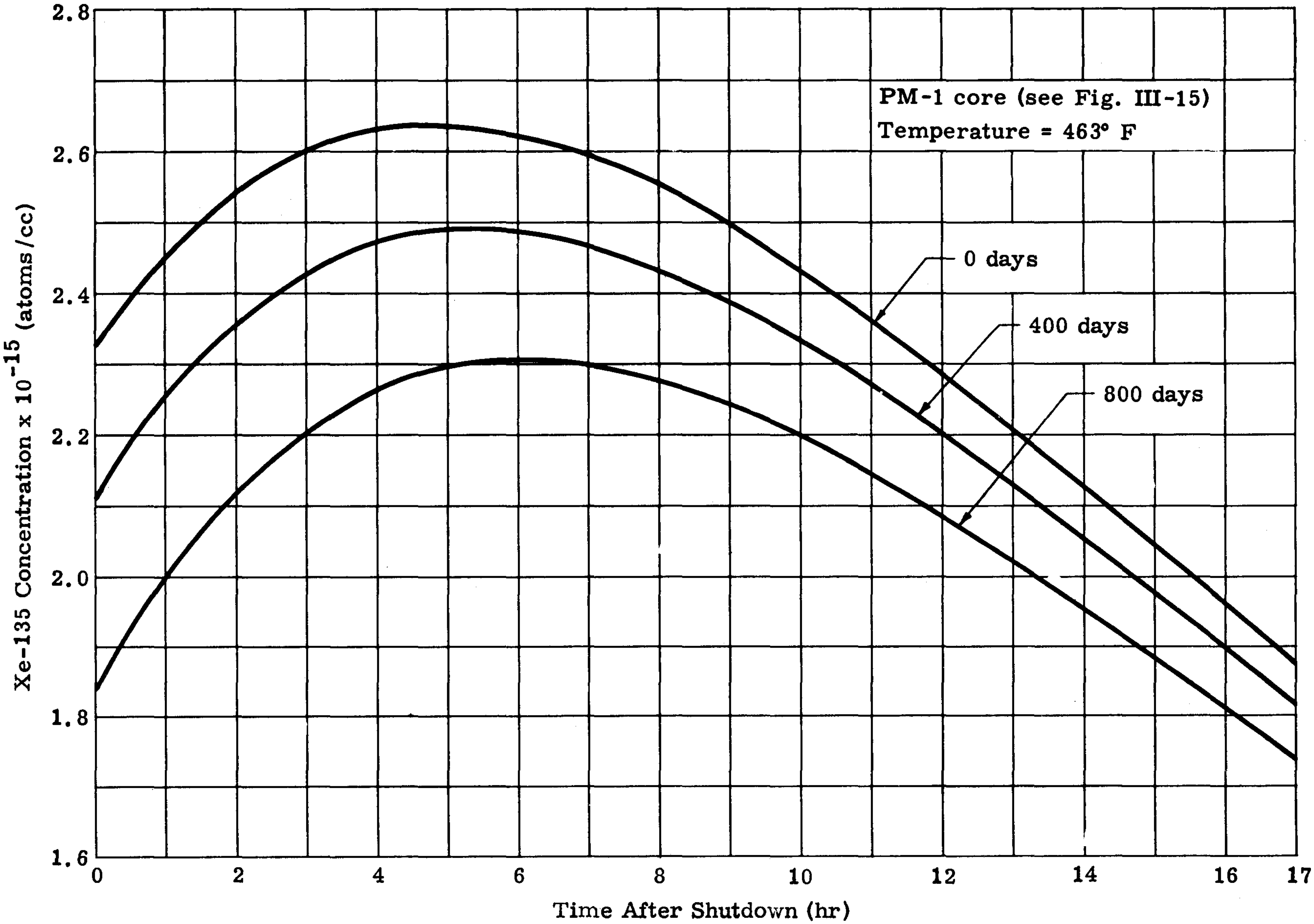


Fig. III-23. Xenon Concentration Buildup After Shutdown

M. M. SHIELDING STUDIES

D. Owings, E. Koprowski, E. Divita

Shielding analysis performed during the fourth quarter consisted of:

(1) Determination of radiation heating rates in the reactor vessel base, flange and inlet nozzle regions.

(2) Determination of radiation heating rates in earth adjacent to the reactor package.

(3) Package interconnect shielding.

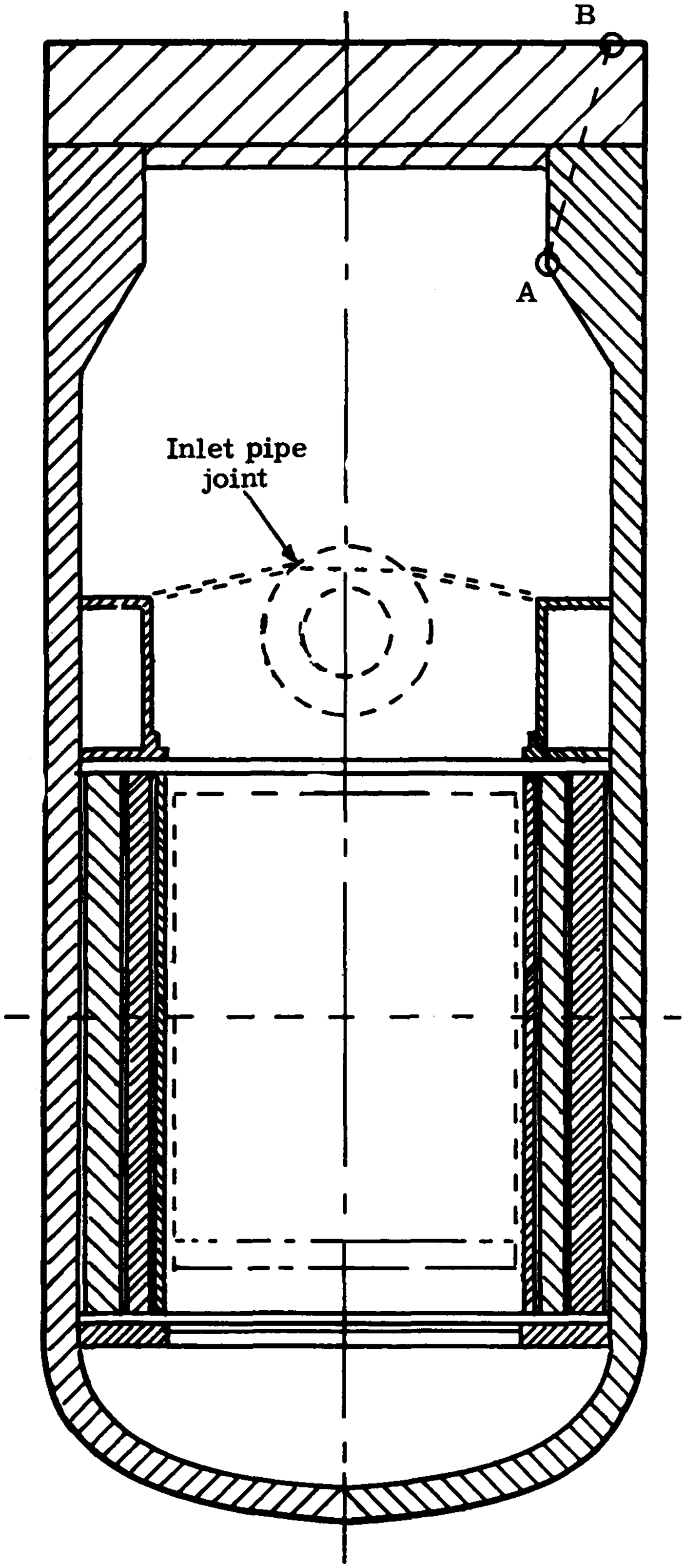
(4) Biological dose rates during full power operation.

(5) Argon activation of air within the maintenance shelter.

1. Radiation Heating Rates Within the Reactor Vessel

Heating rates within the reactor vessel base, inlet joint and flange region have been determined. The pressure vessel and thermal shield geometry assumed for these calculations is shown in Fig. III-24. The thermal shield configuration along the core radial centerline is not to scale in Fig. III-24 and it has been modified as follows:

Region	Inner Radius (in.)	Outer Radius (in.)	Thickness (in.)
Core shroud	11.55	11.80	0.25
Pressurized water	11.80	12.1375	0.3375
Thermal shield	12.1375	13.7375	1.60
Pressurized water	13.7375	14.075	0.3375
Thermal shield	14.075	17.475	3.40
Pressurized water	17.475	17.8125	0.3375
Pressure vessel	17.8125	20.000	2.1875
Insulation	20.0	21.0	1.0
Flux suppressor	21.0	22.0	1.0



Scale: 1/12 = 1

Fig. III-24. Reactor Vessel and Thermal Shield Configuration Assumed for Computation of Radiation Heating Rates

This configuration is based on a heat transfer analysis of the previously reported (Third Quarterly Progress Report, MND-M-1814) radiation heating rates through a similar thermal shield configuration.

Total radiation heating rates in the reactor vessel base through the vessel were computed at points shown in Fig. III-25. Assuming exponential attenuation through the vessel, gamma heating rates between indicated points are shown in Fig III-26. Figure III-27 is a cross plot indicating total heat release rate along the inner and outer surfaces of the vessel where indicated distances are measured along the inner and outer surfaces shown on Fig. III-25. Contribution from neutrons and gammas originating in the core and sources exterior to the core are indicated.

Heating rates in the pressure vessel inlet joint region at points indicated on Fig. III-28 are as follows:

Point	Heating Rate ($Btu/in.^3$-hr)
A	26.2
B	1.24
C	14.0
D	2.16

Conservatively, radiation heating rates may be assumed exponential horizontally or vertically between the above points. The relatively high heating rate at Point D is due to gammas originating in the core. At this point, the thermal shields do not significantly shield the core.

Total heating rates in the flange region at Points A and B of Fig. III-24 are 1.71×10^{0} and 5.72×10^{-4} $Btu/in.^3$-hr, respectively. Assuming exponential attenuation, the heating rate between these two points is given by:

$$Q(r) = 1.71\ e^{-0.509\ r}$$

where

$Q(r)$ = the heating rate ($Btu/in.^3$-hr) measured along line r drawn between Points A and B (r = 0 at Point A and is in inches).

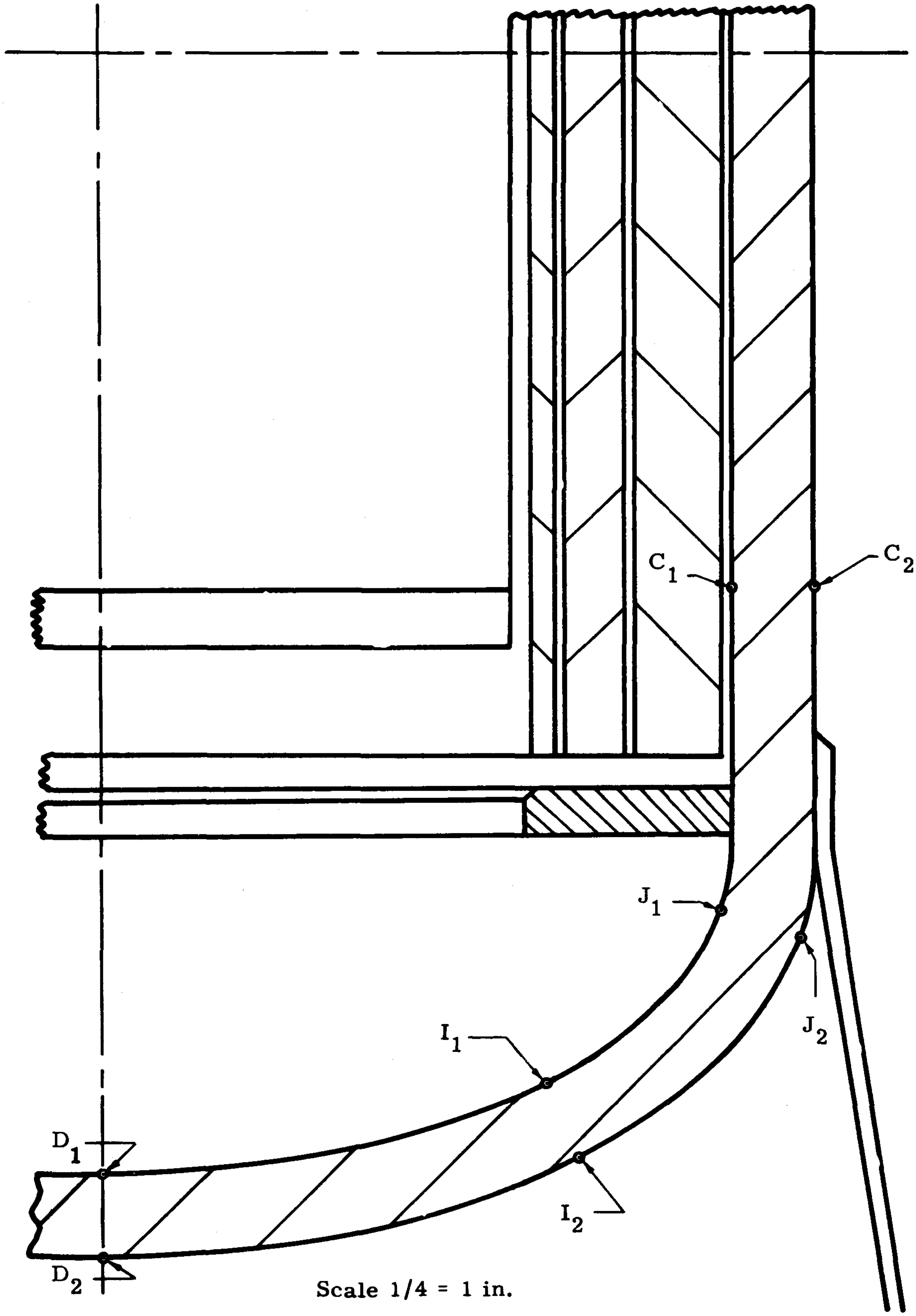


Fig. III-25. Cross Section of PM-1 Reactor Vessel Base

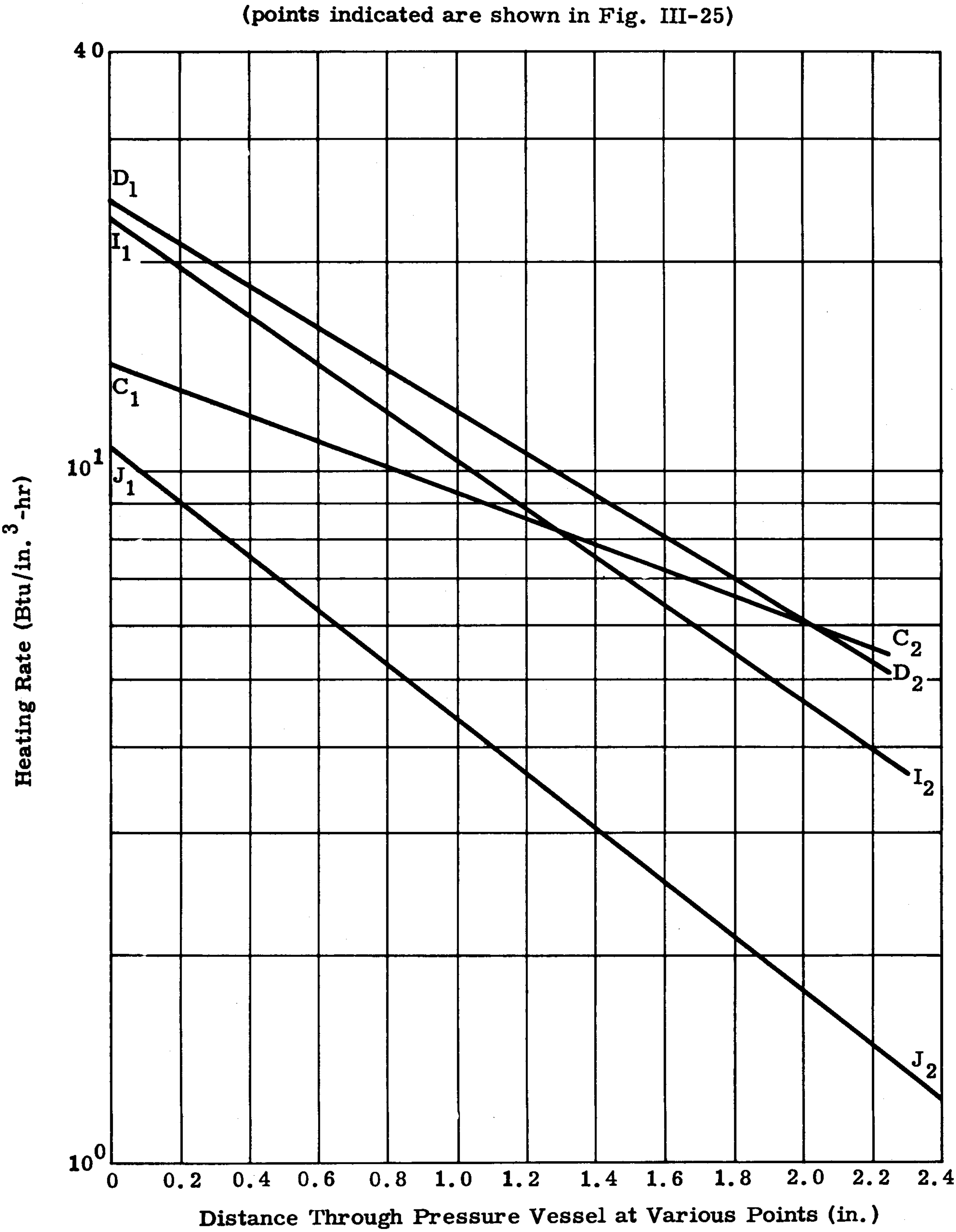
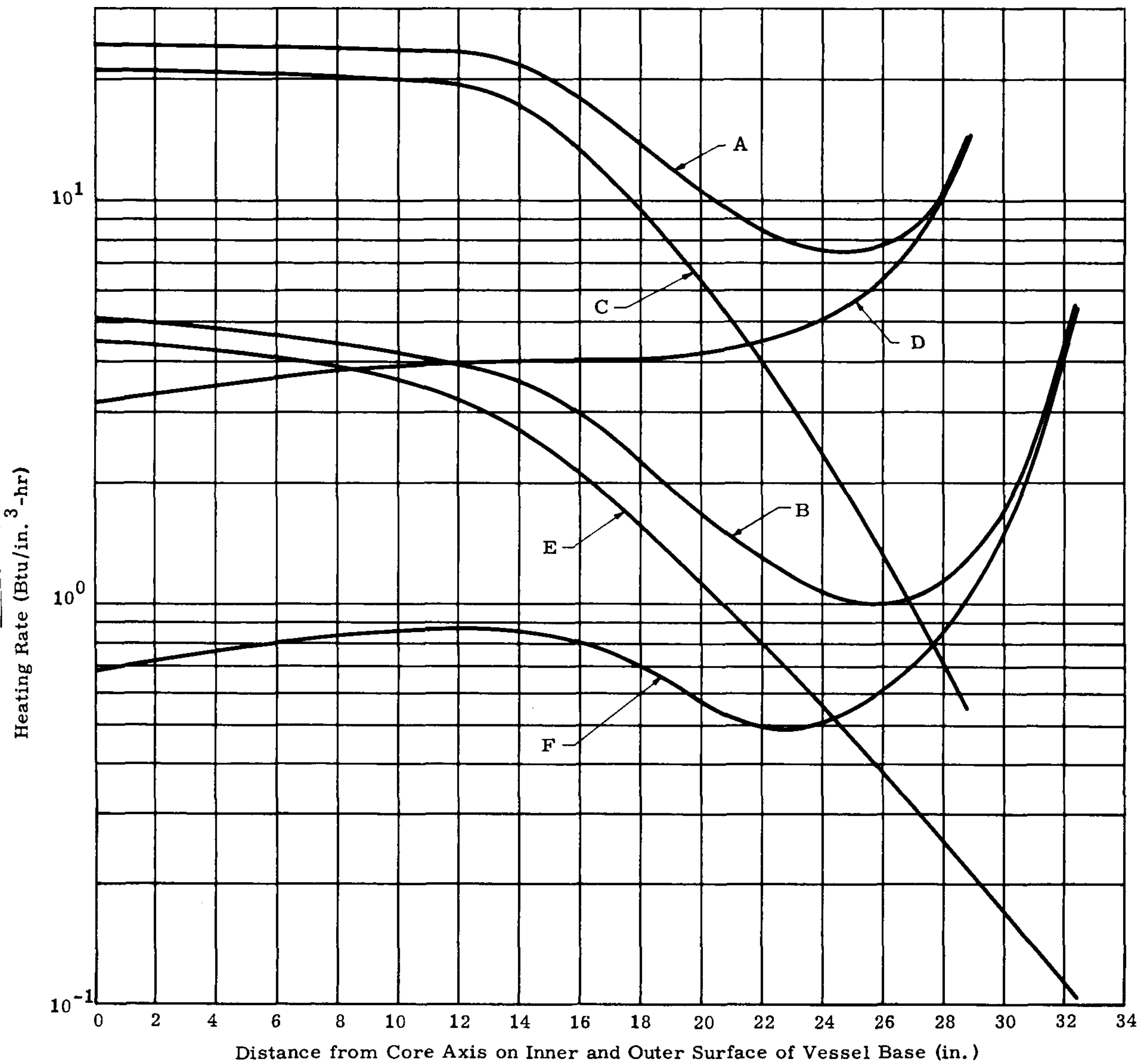


Fig. III-26. Gamma Heating Rates Through Reactor Vessel-- 10-mw Operation



LEGEND:

Curve A Total heating rate--inner surface

Curve B Total heating rate--outer surface

Curve C Core gammas--inner surface

Curve D Thermal neutron capture γ's from region exterior to core--inner surface

Curve E Core gammas--outer surface

Curve F Thermal neutron capture γ's regions exterior to core--outer surface

Fig. III-27. Gamma Heating Rates on Inner and Outer Surface of Reactor Vessel Base--10 mw Operation

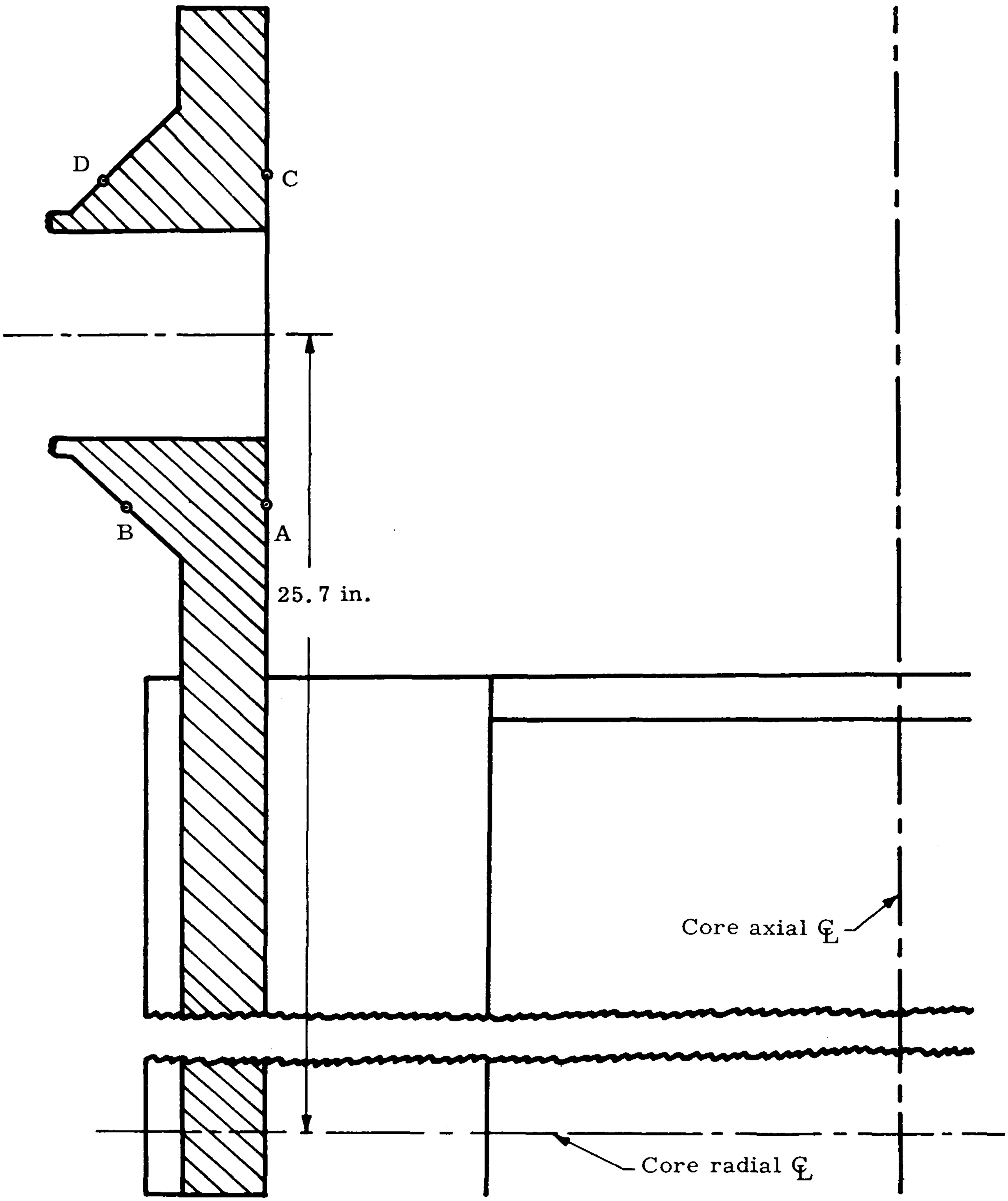


Fig. III-28. PM-1 Reactor Vessel Input Joint Region

All heating rates given above should represent conservative estimates of the maximum values attained over the useful core life.

2. Radiation Heating Rates Within the Earth Adjacent to the Reactor Package

Preliminary calculations of gamma heating in the earth shield indicated excessive maximum temperature (1800° F) occurring in the earth along the core radial centerline. Subsequent calculations have been made to determine the gamma heating rates in the earth, assuming additional shielding in the form of lead shot placed to the side and below the reactor vessel. For radiation attenuation calculations, it was assumed that a lead shot shield region is homogeneous and 65% lead by volume (lead density = 708 lb/ft^3). A lead shot shield was designed so that the maximum earth temperature during full-power operation will be approximately 370° F. Shield dimensions are indicated in Figs. III-29 and III-30. The shield height of 2.7 feet above and below the core radial centerline assumes a 7.0-inch water gap between the outer surface of the flux suppressor and shield. The alternate shield position assumes a 1.0-inch gap between the flux suppressor and shield. The shield will extend through an arc of 240 degrees around the reactor vessel as indicated on Fig. III-30. The radial shield thicknesses indicated in Fig. III-30 are such that heat generation rates along a horizontal plane through the center of the core at the surface of the earth are uniform.

Radiation heating rates in a plane through the axis of the core and along Line C-C of Fig. III-30 are given in Fig. III-31. Heating rates in earth below the reactor package, assuming a 3.0-inch thickness of lead shot shielding below the reactor vessel, are given in Figs. III-32 and III-33. Elimination of the shield below the reactor will result in earth temperatures below the reactor package not exceeding 370° F.

Earth shielding requirements for the 120° F section, not indicated on Fig. III-30, are more than adequately met by the shadow shield placed in this region to shield the package interconnect region after shutdown.

Temperature rise within the lead shot shield due to radiation heat generation in the shield region is negligible. Deterioration of lead shot by corrosion over extended periods will be negligible.

3. Package Interconnect Shielding

The primary design consideration of shielding placed within and around the reactor package-steam generator package interconnect is accessibility to the region by maintenance personnel approximately 8

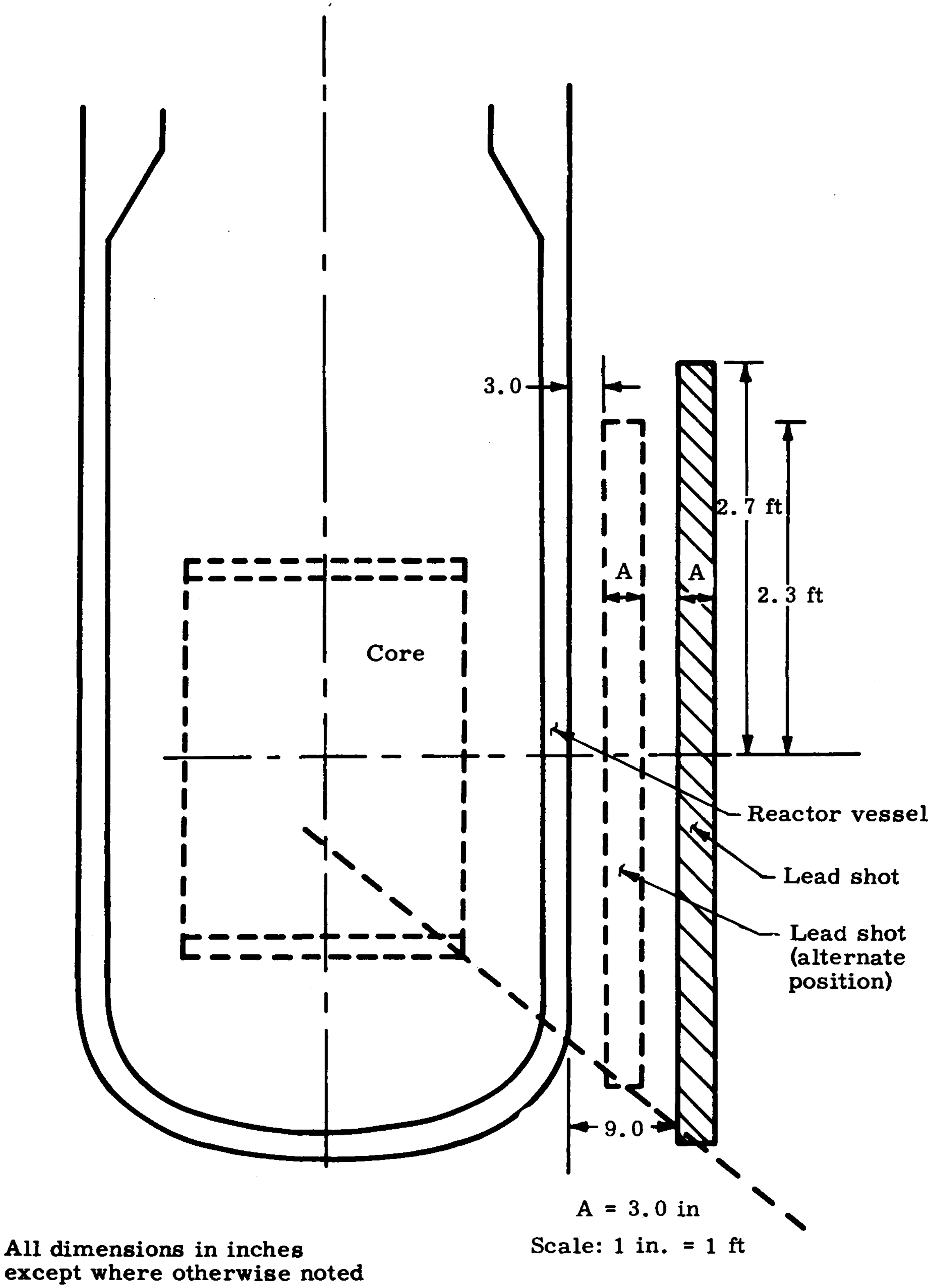


Fig. III-29. Lead Shot Shield for Reduction of Radiation Heating in Earth

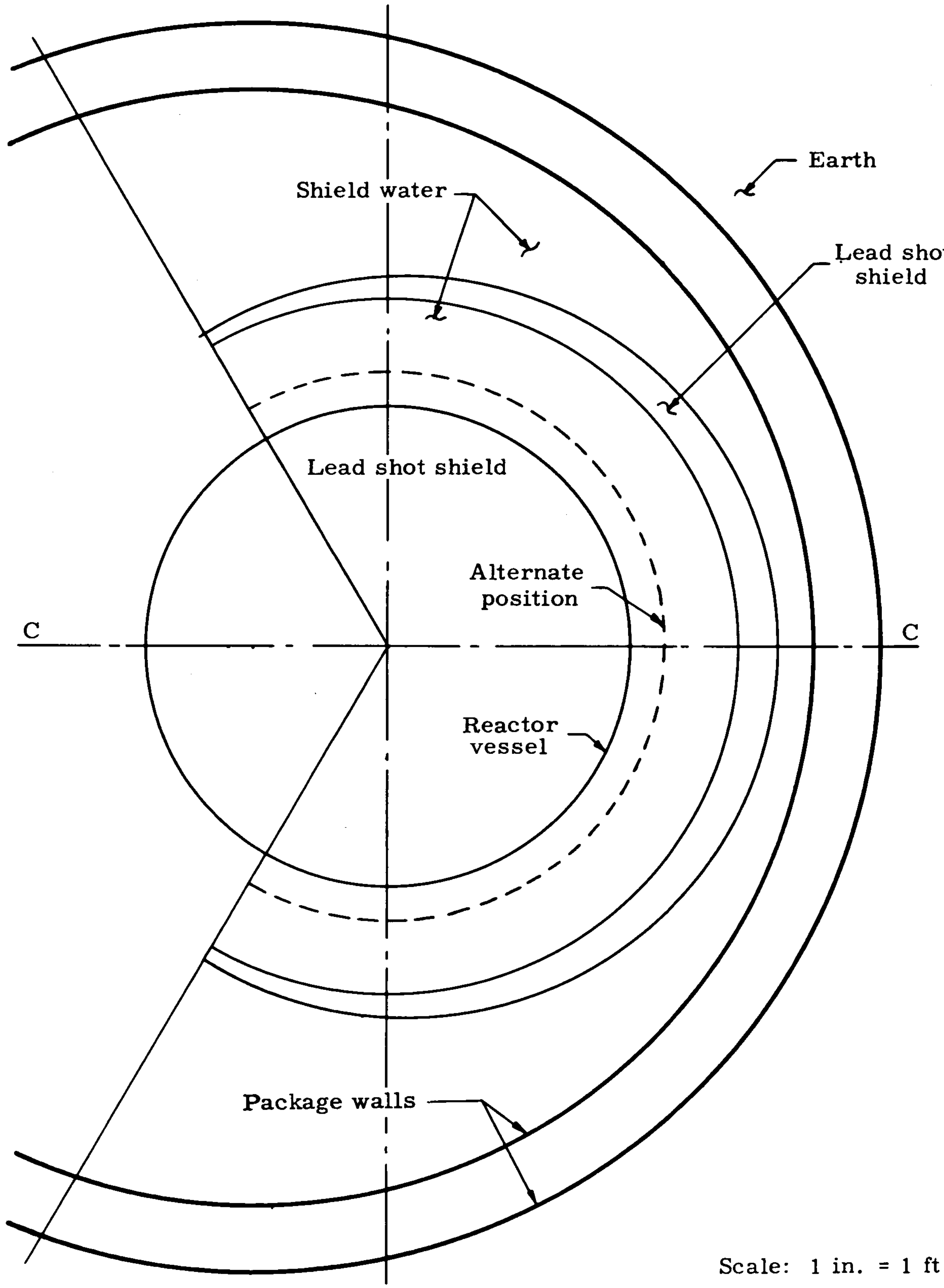


Fig. III-30. Lead Shot Shield for Reduction of Radiation Heating in Earth

Shield water and 3.0-in. lead shot shield

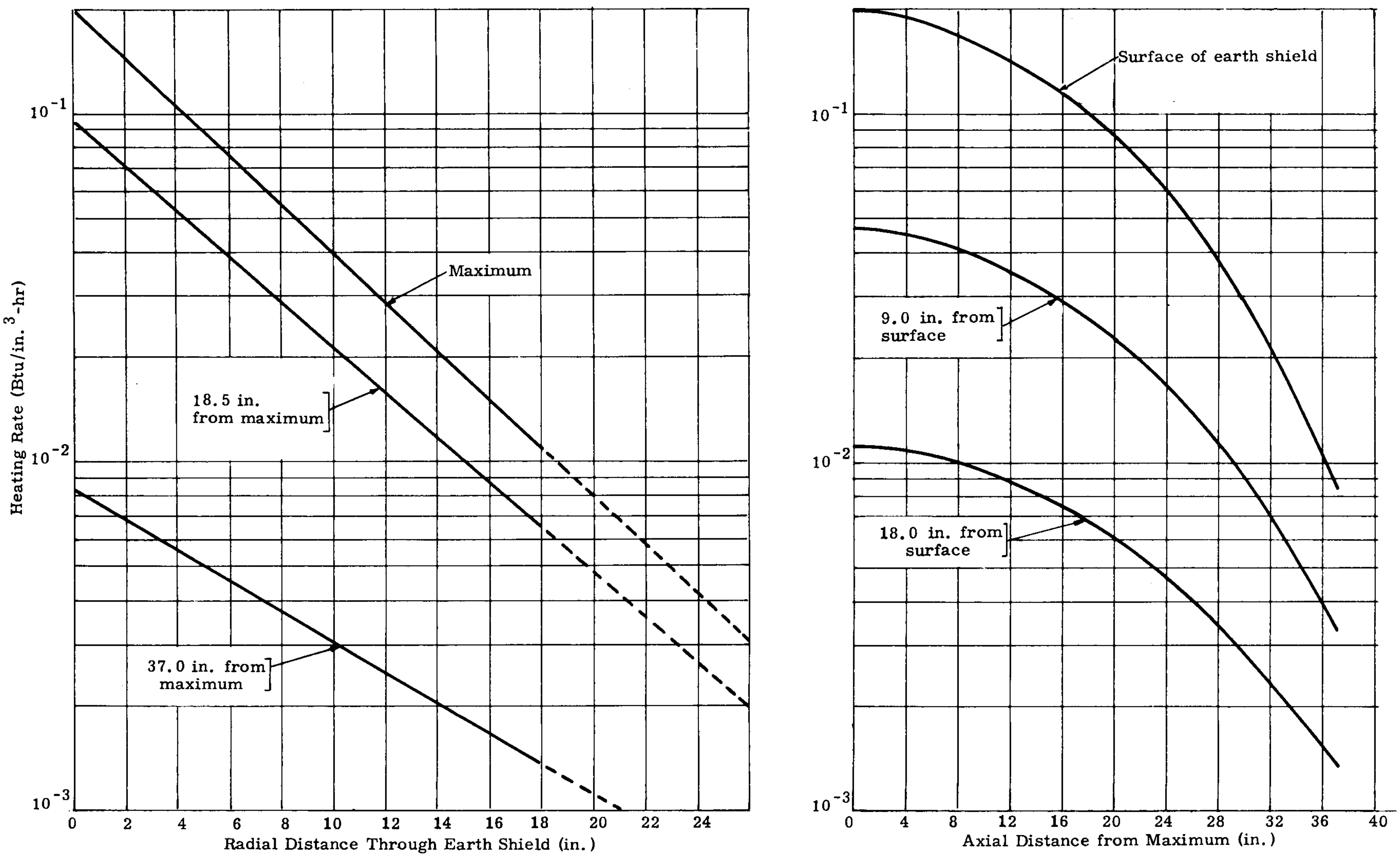


Fig. III-31. Radiation Heating Rates in Earth Adjacent to Reactor Package--10 mw Operation

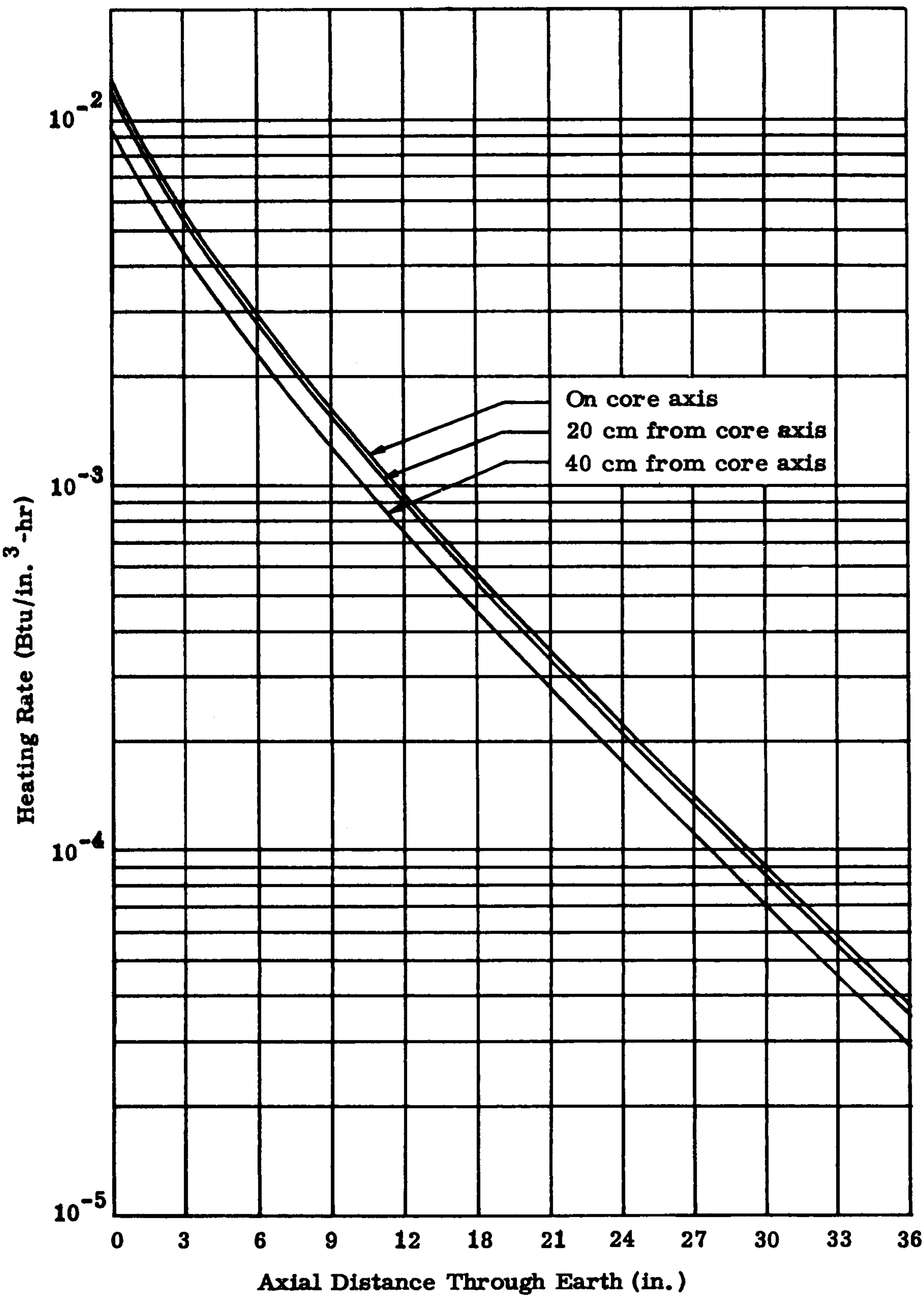


Fig. III-32. Radiation Heating Rates in Earth Below Reactor Package--10-mw Operation, 3.0-in. Lead Shot Shield

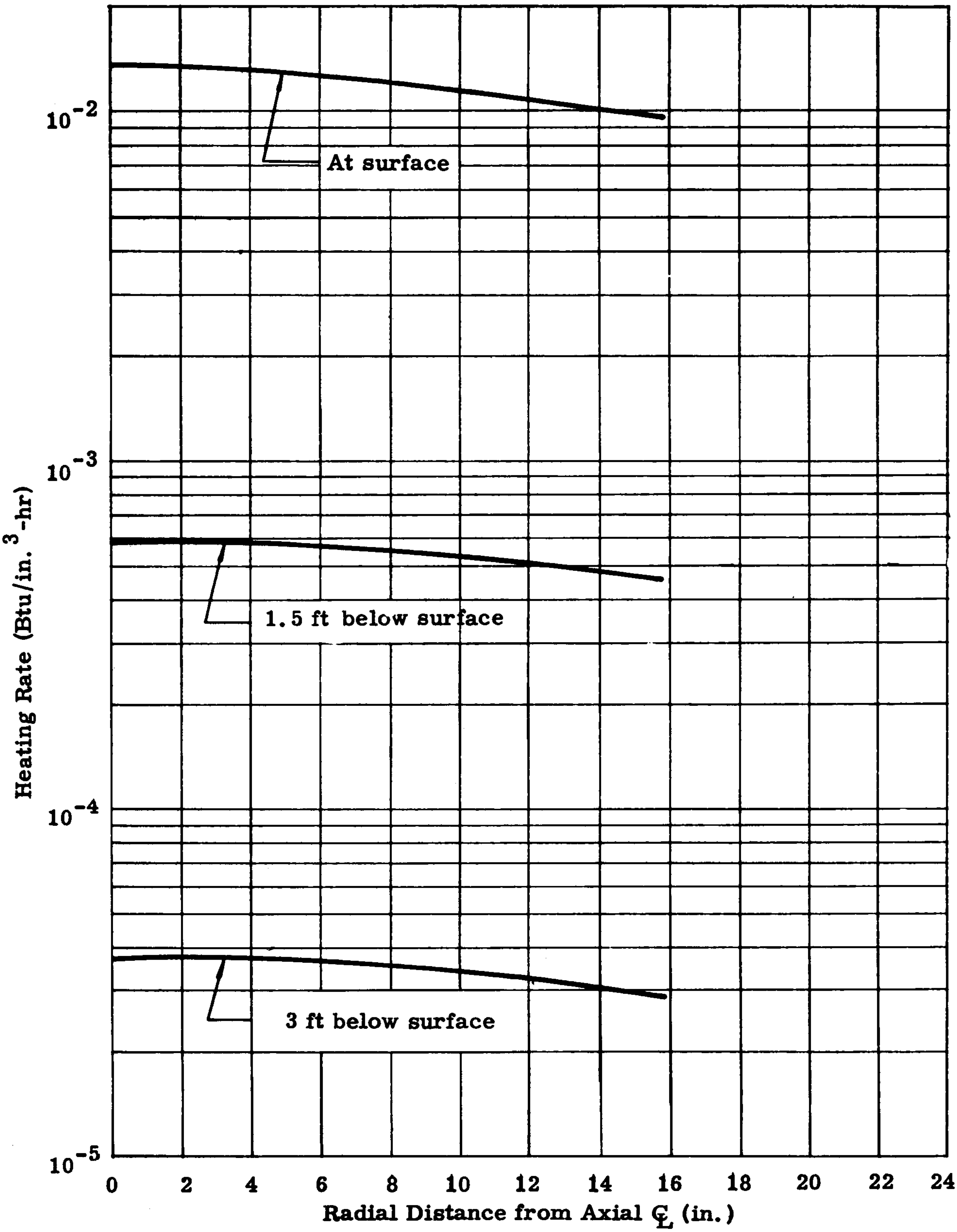


Fig. III-33. Radiation Heating Rates in Earth Below Reactor Package--10-mw Operation, 3.0-in. Lead Shot Shield hours after reactor shutdown at any time during the 20-year life of the system. Significant sources of radiation in this region after reactor shutdown are:

(1) Fission product activity in the reactor core.

(2) Neutron activated thermal shields and reactor vessel.

(3) Primary loop corrosion product and impurity activity deposited on walls of system.

(4) Neutron activated components of the interconnect region.

Shielding of radiation originating in the core and vicinity (Sources (1) and (2) above) is accomplished by placing a shadow shield between the reactor vessel and package interconnect. Assuming a solid lead shield (lead density = 708 lb/ft^3), minimum shield thicknesses are shown on Figs. III-34 and III-35. An alternative shield of lead shot may be used. Assuming the lead shot region to be 65% lead by volume, required thickness will be 1.5 times the lead thickness (measured in horizontal planes along radii of the core). The shield has been extended below the core to provide additional reduction of gamma flux and allow access after shutdown in the steam generator package.

To determine the effects of radiation heating in the lead shielding, it was assumed that the 5-inch thick lead shield was placed 3 inches from the reactor vessel surface. The radiation heating rate in the lead shield along the core radial centerline during full-power operation was then determined to be

$$Q(x) = 33.5\ \epsilon^{-1.755x} + 0.80\ e^{-0.263x}\ (\mathrm{Btu/in.^3\text{-}hr})$$

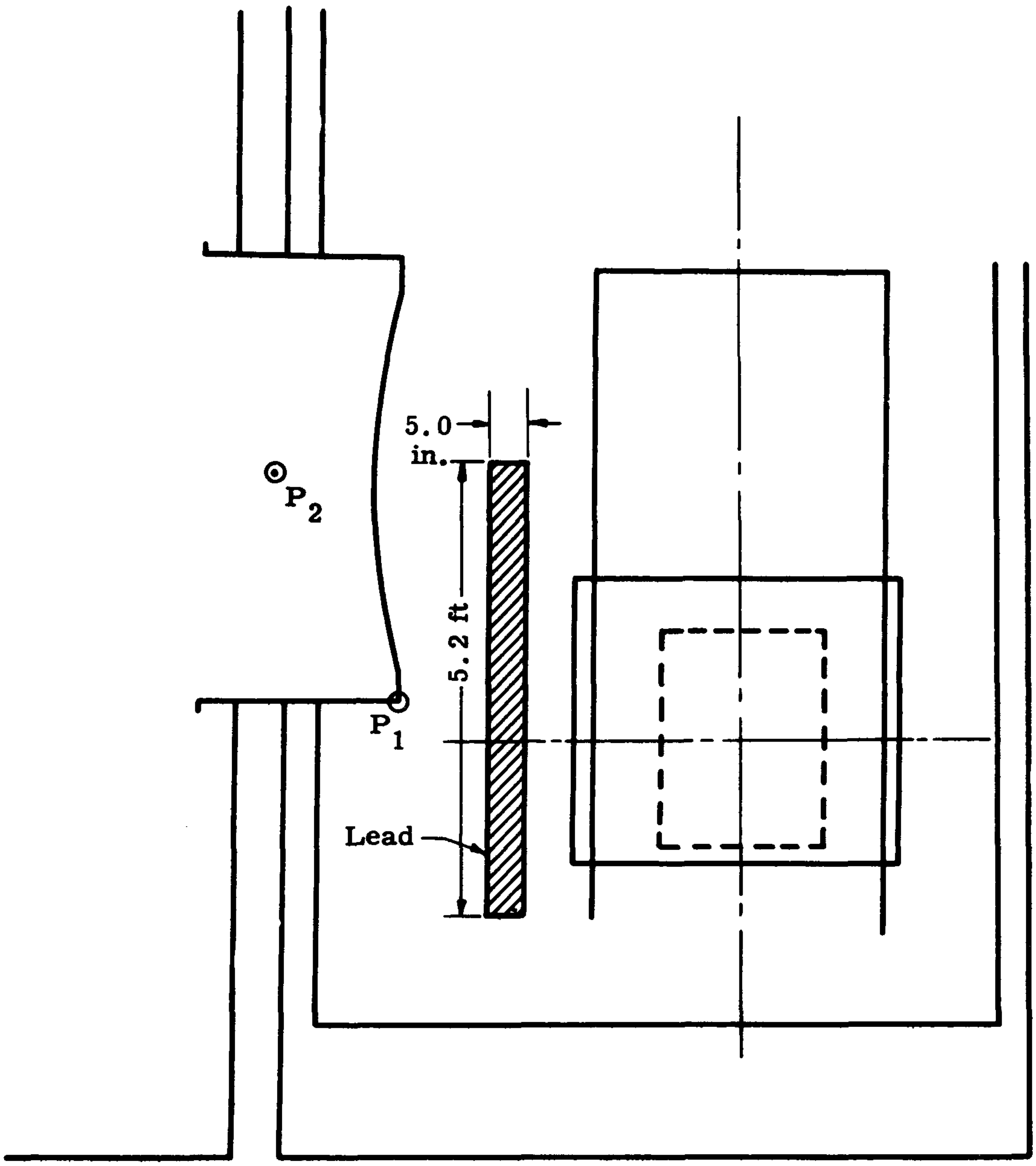
where x (in inches) is measured from the inner surface of the shield along the core radial centerline. This source of heat will effect a temperature rise of less than 5 degrees within the lead.

Estimates of the biological dose rates from fission products and activated stainless steel in the vicinity of the core are as follows:

At P1 (Fig. III-34)

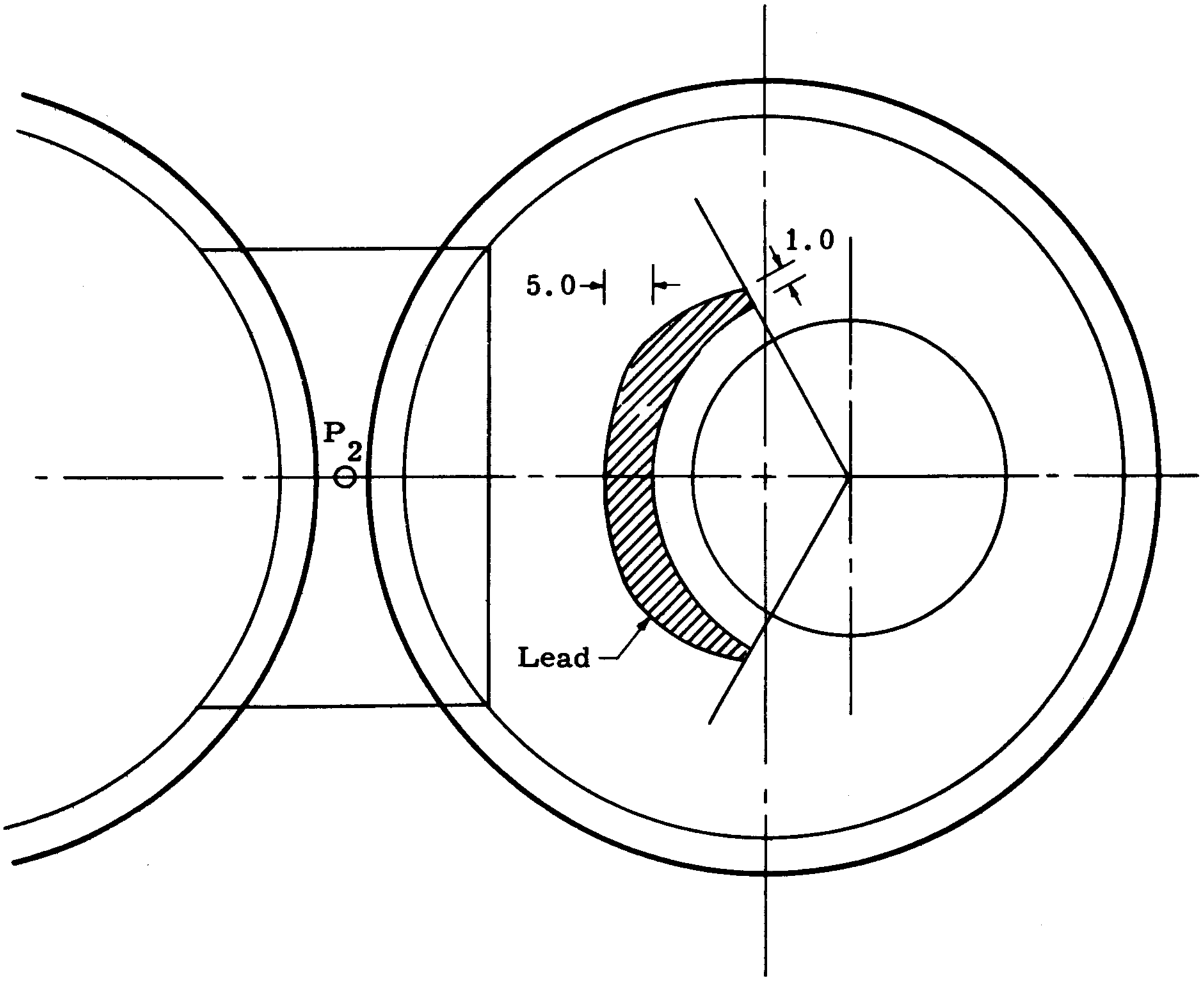
(1) Eight hours after shutdown

Fission product activity	70 mr/hr
Activated stainless steel	670 mr/hr



Scale: 1/2 in. = 1 ft

Fig. III-34. Interconnect Shield



Scale 1/2 in. = 1 ft

All dimensions in inches
except where otherwise noted

Fig. III-35. Interconnect Shield

(2) Nineteen hours after shutdown

Fission product activity	50 mr/hr
Activated stainless steel	180 mr/hr

At P2 (Fig. III-34)

Eight hours after shutdown

Fission product activity	<2 mr/hr
Activated stainless steel	7 mr/hr

The high dose rate at P1, 8 hours after shutdown, is due mainly to Mn-56 activity in the steel. This activity is relatively short-lived (2.58-hour half-life) and will decay to negligible quantities after a 20-hour shutdown period.

Attenuation of neutron radiation originating in the core by media between the core and interconnect was found to be insufficient. Without additional neutron shielding, the following problems are encountered:

(1) Thermal neutron activation of the interconnect structure produces excessive dose rates in the interconnect region after shutdown. Dose rates 8 hours after shutdown in this region will be approximately 5.r/hr.

(2) Structure scattering of fast neutrons in the steam generator package will produce a fast neutron dose rate of approximately 3.6 rem/hr at floor level above the steam generator package.

(3) Excessive neutron damage may occur in system components located in the steam generator package adjacent to the interconnect.

These problems may, for all practical purposes, be eliminated by the placement of additional neutron shielding in the interconnect region.

Thermal neutron flux in the interconnect walls adjacent to the shield water is reduced by the addition of boron carbide placed in a suitable form on interconnect wall surfaces exposed to shield water. The B_4C, suitably enclosed, will be of sufficient thickness to reduce the thermal flux by at least a factor of 10^4. The required thickness assuming a density of 1.90 gm/cm^3 is 0.072 inch. Burnout of boron for the B_4C shield over the 20-year life of the system is negligible.

Several materials were considered for placement in the interconnect to reduce neutron flux in the interconnect and steam generator package. The material selected for this purpose is borated polyethylene. Major advantages of the material are:

(1) Light weight (density approximately 0.96 gm/cm^3).

(2) Does not require "canning."

(3) Good attenuator of neutrons of all energies.

(4) Thermal neutron activation negligible.

Borated polyethylene blocks will be cut to shape so that essentially all the air space in the interconnect will be filled with this material. Placement of individual pieces will be such that neutron streaming is minimized.

Polyethylene containing 2% by weight natural boron will effect the following:

(1) Reduction of netron flux at the steam generator surface of the polyethylene to 4×10^3 neutrons/cm^2-sec (fast) and 1.7×10^2 neutrons/cm^2-sec (thermal). This reduces the scattered neutron dose rate at floor level above the steam generator package to less than 100 mrem/hr during full-power operation.

(2) Dose rate from stainless steel components of the interconnect in the center of this region with all polyethylene shielding removed will be less than 50 mr/hr, 8 hours after reactor shutdown.

Thermalization of neutrons by the primary coolant within the primary piping will cause activation of the primary piping. Polyethylene shielding will have little effect on thermal flux in this region. The effects of thermal neutron activation in the piping on dose rates in this region are difficult to estimate. However, since thermalization occurs within the pipe, most of the activity will be found near the inner surface of the pipe and will be attenuated by the relatively thick piping. This activation should not significantly affect biological dose rates in this area. Similarly, corrosion product activity plated on the walls of the primary piping in this region must be attenuated through the walls of the pipe and will have little effect on dose rates.

Other neutron shielding materials considered for use within the interconnect region were:

Material	Major Disadvantage
Lithium hydride	Must be sealed in watertight cans (reaction with water)
Borated water	Tanks would be difficult to remove for interconnect maintenance
Borated parrafin	Low melting point (118 to 133° F)

Temperature rise within the interconnect shield due to radiation heating would be negligible. Boron burnout over the 20-year life is negligible.

With inclusion of the above described shielding, biological dose rates in the center of the interconnect 8 hours after reactor shutdown are estimated at 60 mr/hr.

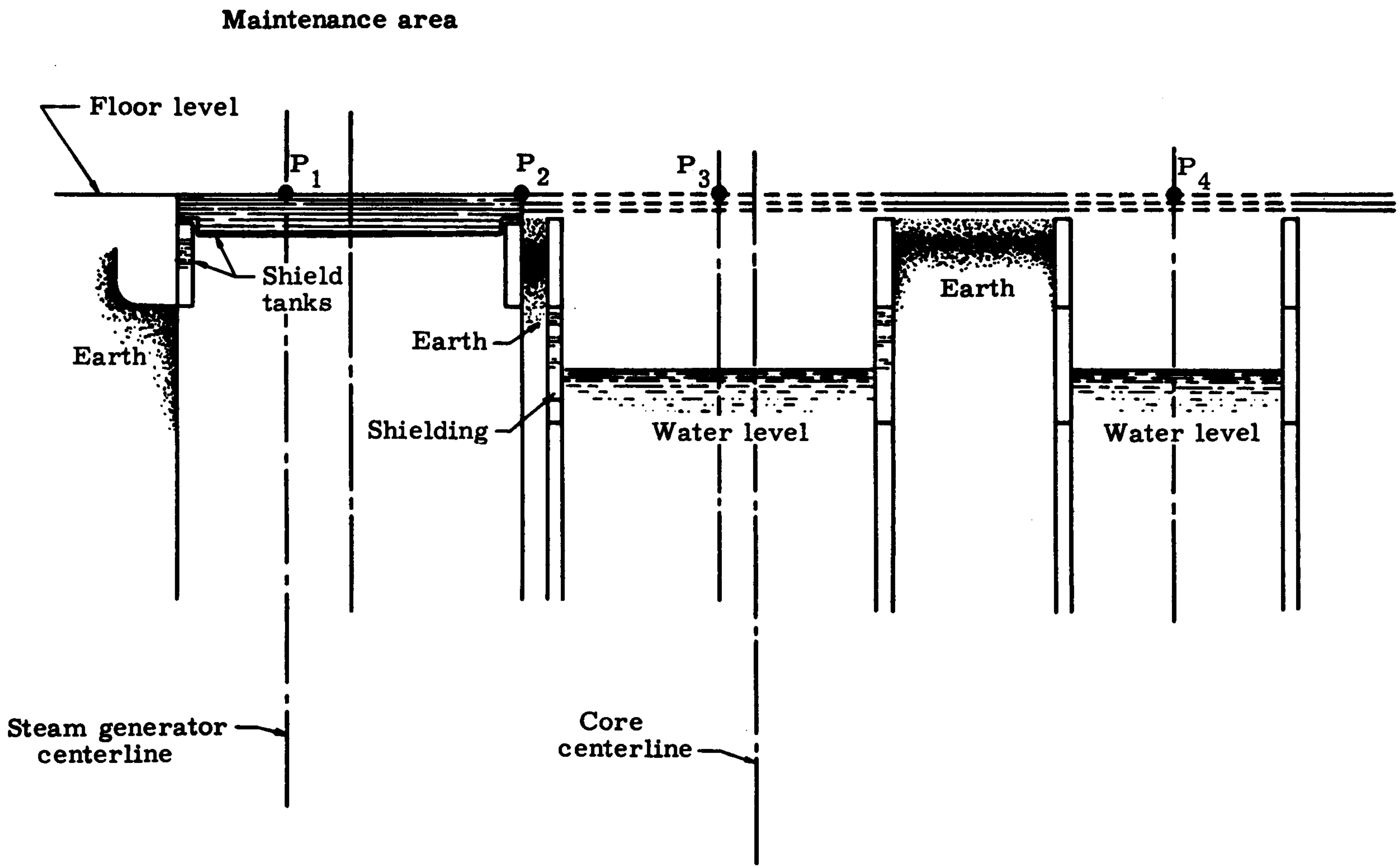
4. Biological Dose Rates During Full-Power Operation

During full-power operation, limited personnel access will be required in the building above the reactor and steam generator packages. Maximum dose rates in this area occur at floor level directly above the steam generator. The dominant source of radiation above the steam generator package is primary coolant intrinsic activity. Primary gamma radiation from the reactor core and gamma radiation originating from thermal neutron capture and neutron inelastic scatter is significantly reduced by the shield water. Neutron flux above the reactor package is negligible. Water within the spent fuel storage tank is of sufficient height to reduce fission product gamma actively to less than 2 mr/hr.

Dose rates during 10-Mw operation at points indicated on Fig. III-36 are as follows:

P-1 (asssuming no shield above the steam generator package)

Gamma radiation		r/hr
Steam generator package		1.76
Reactor package		0.02
Scattered neutrons		0.10
	Total (approx)	1.9



Scale: 1/4 in. = 1 ft

Fig. III-36. Primary System Tank and Maintenance Area

P-2 (assuming no shielding above steam generator package)

Gamma		mr/hr
Steam generator package		576
Reactor package		25
Scattered neutrons		100
	Total (approx)	700

P-3

Gamma		mr/hr
Steam generator package		133
Reactor package		33
	Total (approx)	170

P-4

Scattered Gamma radiation	20

The additional shield placed near the top of and adjacent to the shield water is to reduce scattered neutron streaming between the inner and outer walls of this package.

Provision is made for additional shielding above the steam generator package by the inclusion of the tanks indicated in Fig. III-36. These tanks may be filled with materials locally available, such as water or earth. Assuming that there is water in the 2 tanks, dose rates at point P-1 during operation is as follows:

Gamma	850 mr/hr
Neutron	10 mr/hr
Total	860 mr/hr

Contribution from radiation originating in the steam generator package to other dose points would be reduced proportionately.

Dose rates at floor level, two feet beyond the periphery of the steam generator package and reactor package (excluding the region between the packages), will be less than 200 mr/hr.

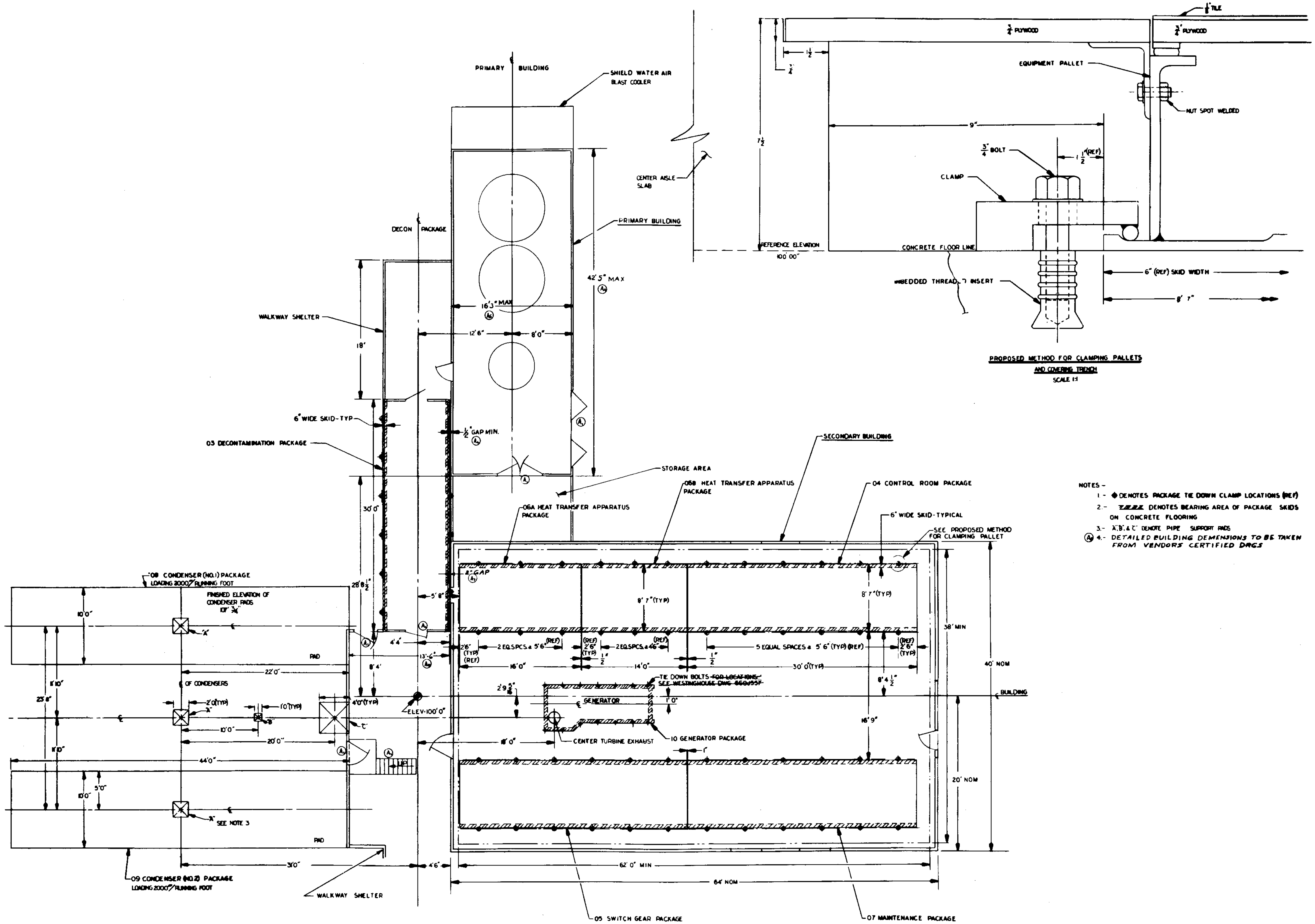

PROPOSED METHOD FOR CLAMPING PALLETS AND COVERING TRENCH
SCALE 1:1
NOTES -
1 - ◆ DENOTES PACKAGE TIE DOWN CLAMP LOCATIONS (REF)
2 - DENOTES BEARING AREA OF PACKAGE SKIDS ON CONCRETE FLOORING
3 - 'A', 'B', & 'C' DENOTE PIPE SUPPORT PADS
4 - DETAILED BUILDING DIMENSIONS TO BE TAKEN FROM VENDORS CERTIFIED DWGS
PRIMARY BUILDING
SHIELD WATER AIR BLAST COOLER
SECONDARY BUILDING
04 CONTROL ROOM PACKAGE
05 SWITCH GEAR PACKAGE
07 MAINTENANCE PACKAGE
10 GENERATOR PACKAGE
03 DECONTAMINATION PACKAGE
08 CONDENSER (NO.1) PACKAGE
09 CONDENSER (NO.2) PACKAGE
WALKWAY SHELTER
STORAGE AREA

Fig. III-37.

Total dose rates in the decontamination building during operation will be less than 2 mr/hr.

5. Argon Activation of Air Within Maintenance Shelter

To determine thermal neutron-induced Argon-41 activity in the air being circulated through the steam generator package, 2 extreme conditions were examined:

Closed loop circulation. It was assumed that air circulation through the steam generator package formed a closed loop with the maintenance building above the packages. Based on a change rate of 644 ft^3/minute, air will remain in regions of high thermal neutron flux approximately 1.26 minutes. The total loop time is 11.47 minutes. Saturated Argon-41 activity for this case will be 3.9×10^{-14} curies/mi of air. Maximum permissible concentration in air for continuous exposure is 5×10^{-13} curies/mi.

Open loop-one pass circulation. For this case, it was assumed that air is released after one pass through the region of high thermal neutron flux. During continuous full power operation, Argon-41 activity will be released to the atmosphere at the rate of 7.0×10^{-5} curies/day which is acceptable.

N.N. SECONDARY SYSTEM STUDIES

W. Koch, R. Groscup, A. Layman, L. Hassell

1. General PM-1 Secondary System Layout

During the quarter, the following secondary system work was planned and accomplished:

(1) Completion of the PM-1 arrangement for the Sundance site (see Fig. III-37).

(2) Revision of equipment layout to incorporate data from known vendors.

(3) Determination of detailed package contents and weights to the extent required for resolving number and type of packages.

(4) Determination of the feasibility of package tie-down off-center in the C-130 aircraft to provide a walkway past the packages into the aft section of the aircraft.

(5) Design of the spent fuel shipping cask skid.

Figure III-37 shows the basic arrangement for the Sundance site. This is essentially as shown in Fig. 4.58 in the PM-1 Third Quarterly Progress Report, but includes detailed dimensions and denotes the reference point for construction. Reference elevation has been established as 100 feet, 0 inch, at the height of the finished concrete pad under the decontamination package. The reference point has been selected at the intersection of the centerlines of the decontamination package and the secondary system building.

The heat transfer apparatus package (06) has been split into two shipping units (06-A and 06-B) due to excessive weight. It was considered desirable to maintain the compact equipment arrangement and the as-shipped interconnections to the maximum practical extent. Examination of the package equipment arrangement indicated an optimum split at 16 feet/14 feet. The total length of the package will still be 30 feet in the site configuration. The anticipated total weight of each section is below 30,000 pounds, allowing shipment of some additional cargo in each aircraft flight.

It will be noted from Fig. III-37 that the interconnecting walkways have been shown in outline form. The walkway section from the primary building to the decontamination package is 18 feet long, providing space for the operating location of the gas bottles for the primary system, for a moderate storage area of about 80 square feet and for a steam hose connection and several small fire extinguishers. The walkway between the secondary building and the decontamination package has been extended in width to the retaining wall at the elevated condenser pad. This simplifies construction and provides addiional space for miscellaneous storage. Enclosed stairs will be provided in this area for access to the elevated condenser pad.

It should be noted that the selected arrangement is that considered most appropriate for the Sundance site and that the basic design allows for a great deal of flexibility in site arrangement with relatively little change in interconnections.

The secondary system packages and estimated weights are summarized in Table III-9. It will be noted that all packages are brought to a total weight of 30,000 pounds by the inclusion of additional items for shipment only. In some cases, these additional items are shipped within a package

TABLE III-9

PM-1 Package Shipping Weight Summary--Secondary System

Package No.	Package Title	Structure Weight	Operational Equipment	Operational Interconnects	Shipping Only items	Package Weight
03	Decontamination	8696	7,820	784	12,700	30,000
04	Control room	7604	10,140	1935	10,321	30,000
05	Switchgear	8059	18,749	3192		30,000
06A	Heat transfer apparatus	5702	12,277	7361	4,660*	30,000
06B	Heat transfer apparatus	3964	9,250	6588	10,198*	30,000
07	Maintenance	7884	17,828	844	3,444	30,000
08	Air condenser No. 1		28,000	--	2,000*	30,000
09	Air condenser No. 2		28,000	--	1,700*	29,700
10	Turbine-generator		31,800			30,000
14	Miscellaneous equipment	8000	--	--	22,000	30,000

*Separate palletized load

and in other cases it is planned to use a small separate shipping pallet. It will also be noted from the table that the reported weight for the complete turbine-generator unit is 31,800 pounds. This includes besides the turbine-generator such items as the auxiliary oil pump, a local gage board, turbine lagging, throttle valve and control mechanism. Space has been reserved with Package 06-A for this possible overage, however the actual unit weight will not be known until fabrication is complete. The stated weight is considered to be conservative and does not include a recent reduction in base plate size.

2. Secondary System Package Design

A. Layman, R. Dugas, L. Noyes

During the quarter, it was planned to review the package design criteria, costs of several design methods and the effects of the abandonment of integral housing features for the Sundance installation. The final design of the secondary system equipment packages and miscellaneous shipping packages was then to be largely completed. This was accomplished.

The secondary system equipment will generally be shipped to the operating site mounted in its operating configuration on steel pallets. There will be three 30-foot shipping pallets carrying the switchgear, controls and maintenance equipment, respectively. Two other pallets will be provided, each approximately 15 feet in length. These will carry elements of the heat transfer apparatus in their operating configuration. In addition, several 30-foot shipping packages will be used for shipment of interconnections and miscellaneous equipment.

Fabrication of the 30-foot pallets will be of welded steel construction since this was found to be more economical than bolted construction. The steel pallets will be covered with a plywood floor (the equipment will be bolted directly to the steel pallet). Attached to the sides and ends of the pallets will be load carrying steel tube "K" trusses sheathed with 1/2-inch plywood as a weather covering. The packages will then be roofed with a 2-inch-thick sandwich panel with stainless steel faces and polystyrene core. It should be noted that the structure closely resembles the decontamination building discussed under Subsystem No. 36. The structural design is thereby maintained compatible with the integral shelter type of package in accordance with the PM-1 design concept of light weight and maximum structural integrity.

Each of the smaller heat transfer apparatus packages will weigh, when fully loaded, less than 30,000 pounds. The pallets will be of welded steel construction, covered with a plywood floor, with the equipment bolted directly to the steel pallet. Instead of high truss sides and a roof, these shorter pallets will have relatively shallow beam sides and ends and will be covered with a weather-proof tarpaulin during shipment to the Sundance site. After delivery and installation at the site within the secondary building, the side trusses and/or beams, roofs, tarpaulins, etc. will be removed from all packages and stored. The equipment will remain bolted to the pallets, ready for interconnection.

The spent fuel cask skid was also designed during the quarter. This skid consists of a relatively shallow pallet approximately 12 feet in length by 12 inches in height. The spent fuel cask is bolted directly to the pallet. The pallet weight is approximately 3000 pounds.

Earlier plans to use roller conveyors under the packages when loading on and off the C-130 aircraft have been revised in favor of the use of lubricated wood shoring. Roller conveyors, unless prohibitively large and heavy, are not rugged enough to do the job required to handle large, heavy shipping packages.

Designs were completed on a short-coupled tie-down device for use within the plane along the sides of the packages. There is insufficient room for use of conventional MB-1 devices within the C-130 aircraft. Package tie-down devices were designed to accommodate anticipated airplane flexural displacements. Measured deflection data from the static testing of these devices presently in progress is expected to substantiate the designs.

Recommended stiffness parameters, from the dynamic analyses of the heat transfer package, were used in the design of all such packages. It is thus expected that undesirable resonances and load amplifications will be avoided.

Requirements for readily loading and unloading the shipping packages from an airplane were evaluated during the quarter and it was determined that the ideal trailer would be approximately 42 inches high, 32 feet long and 9 feet wide, with a 15-ton capacity. Standard trailers are approximately 8 feet wide but do not meet our height and length requirements. The Air Force 40-foot C-2 trailer is the best suited to our requirements. If transportation of the packages from The Martin Company to the site is done by trailer only, no specific height is required and a standard commercial flat bed trailer can be readily converted to meet our requirements on width.

For calculating requirements for the site access road, typical tractor and trailer data was obtained from the Davidson Transfer Company, Baltimore, Maryland. This data was used for computing turn radii and trailer off-tracking for different length trailers. A plot of turn radius versus off-track was prepared for the range of anticipated trailer lengths.

Next quarter, the objectives for this effort will be the completion of the detailed package weight and number tabulation, with revisions where necessary to reflect vendors' equipment certified drawings, and initiation of pallet and package fabrication. Further study effort in support of the loading demonstration will be carried on under Task 12.

3. Interconnected Walkways

A. Layman

Interconnecting covered walkways adjacent to the primary and secondary buildings will be designed and built by the supplier of those buildings to ensure compatibility. No effort was scheduled for the quarter towards completion of this task.

Effort during the coming quarter will be to complete the necessary specifications, establish vendor contact and designate a contractor.

IV. TASK 5--CORE FABRICATION

Project Engineer--Subtask 5.1, 5.2, 5.3--J. F. O'Brien

The overall objectives of Task 5 are to develop and fabricate the fuel elements for the PM-1 Flexible Zero Power Test and the final PM-1 Core.

A. SUBTASK 5.1--FABRICATION OF PMZ-1 CORE

S. Furman, J. Neal, B. Sprissler

During this quarter, fabrication of the PMZ-1 fuel elements continued.

Fabrication of all PMZ-1 components will be completed during the next quarter.

1. Fuel Element Fabrication

a. Powder rolling

On completion of the cleanup after the preproduction run of natural uranium elements, production of enriched elements was begun. The first strip of enriched material rolled in a completely different manner from previous natural uranium blends. The edges were found to be irregular and the strip thickness uneven. The flow rate and apparent density results were only slightly different from the natural material. Eliminating all variables in the powder, it was concluded that the only significant change was in the condition of the roll surfaces. During the cleanup operation, the rolls had been washed with acetone, which leaves a film on the surface. Further preparation of the roll surface by rolling stainless steel powder removed this film and subsequent blends rolled normally. It has been observed that allowing the mill to remain idle for prolonged periods also affects the surface condition. It has been established that, if the mill is idle for more than three consecutive days, the surfaces have to be prepared by rolling stainless steel powder through the machine. Uniformity between the blends rolled indicates that this procedure is adequate for reproducibility.

A trip was made to Mallinckrodt, Hematite Plant. The purpose of the trip was to resolve the discrepancy in the physical characteristic test results between the vendor and Martin Nuclear Quality Control. The particular areas in question were the flow rate and sieve analysis results. It was found that the flow rate equipment used by the vendor was not standard. On checking the sieve equipment used, there was no obvious defect in equipment or method. To date, there is no agreement between the two sieve analyses. However, the powder has been and is being used successfully.

During the rolling of the first few blends, it was noted that the length of the green strip was such that it was difficult to obtain the four lengths required for the fabrication of eight elements. The batch size was therefore increased from 2000 to 2100 grams. The increase in batch size added only slightly to the amount of scrap green strip.

b. Core tube fabrication

On checking the loading of the tubes fabricated, it was noted that the tubes were generally low in the range or below the minimum loading. An effort was made to increase the loading by:

(1) Reducing the chemical cleaning losses.

(2) Increasing the thickness of the core strip.

To decrease the losses in chemical cleaning, the time in the acid dip was reduced to 1/3 the original. The effect of this was very slight. The leeching step was omitted entirely; use of the Oakite soak and scrubbing during the drag-out step proved quite satisfactory. The losses were reduced from approximately 3 grams to the range of 3/4 gram. Preliminary ultrasonic inspection showed no deleterious effect on the bonding.

The thickness of the flat strip was increased by 0.0005 inch. This created a problem in controlling the length. The amount gained by the increase was sufficient for loading requirements. The combination of the decreased losses and increased thickness resulted in a need to shorten the flat strip.

c. Tubular element fabrication

Forming contamination. Fabrication of enriched tubular elements was started with special attention being given to the shearing of the single width strip. During the preproduction run, it was noted that the butt seam in the dead-end material was predominant. Metallography verified the ultrasonic results which indicated that the butt joint was unbonded and had opened in instances (Fig. IV-1 shows a typical sheared edge seam). It is felt that these conditions were largely due to the uneven shearing. At the end of five lots, it was decided that another approach to the problem was required. In an effort to eliminate or minimize the butt seam condition, the following variations were employed:

(1) Machined flat edges--normal size.

(2) Machined flat edges--0.010 inch excess width.

(3) Machined angled edges, 10 degrees--normal size.

(4) Machined angled edges, 10 degrees--0.010 inch excess width.

(5) Premachined angled edges, 15 degrees--normal width.

Of these variations, the angled configuration gave the best results (Fig. IV-2), with an area for improvement towards the outer clad. To improve this area, the angle was increased to 15 degrees. To facilitate fabrication, it was decided to use premachined dead ends. This involved a change in the process in that the cermet strip has to be scarfed and sheared to single width before attaching to the dead end. Metallographic and ultrasonic results showed no difference between the premachined dead ends and those that were machined after welding. At this point, the dead end material was also changed from the commercial grade 321 stainless steel stock to 4-inch wide strip material for better dimensional stability. The new dead end material is Type 347 stainless steel. Some change in metallography was noted in that the areas of small grain size next to the interface (Fig. IV-3), found in the old material, disappeared and a more uniform grain size through the cross section (Fig. IV-4). Figure IV-5 is a typical core seam regardless of the type of dead end used.

d. Production of tubular elements

As the production of strip and tubes was increased, problem areas appeared during the forming operation. A large percentage of the strips were found to be cracking during the first stage of forming. This was attributed to a change in conditions in the furnace. A longer time at temperature during the anneal cycle alleviated this problem.

One of the first problems was achieving the thickness tolerance specified for the final rolling operation. After this tolerance was emphasized and the operators gained more experience in the operation, this problem disappeared.

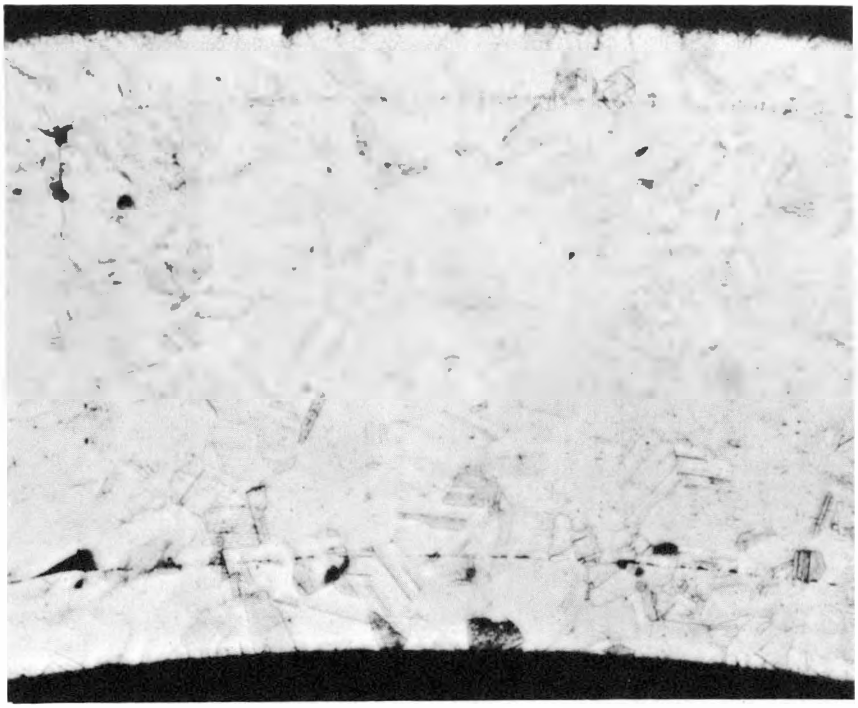


Fig. IV-1. Typical Sheared Butt Seam

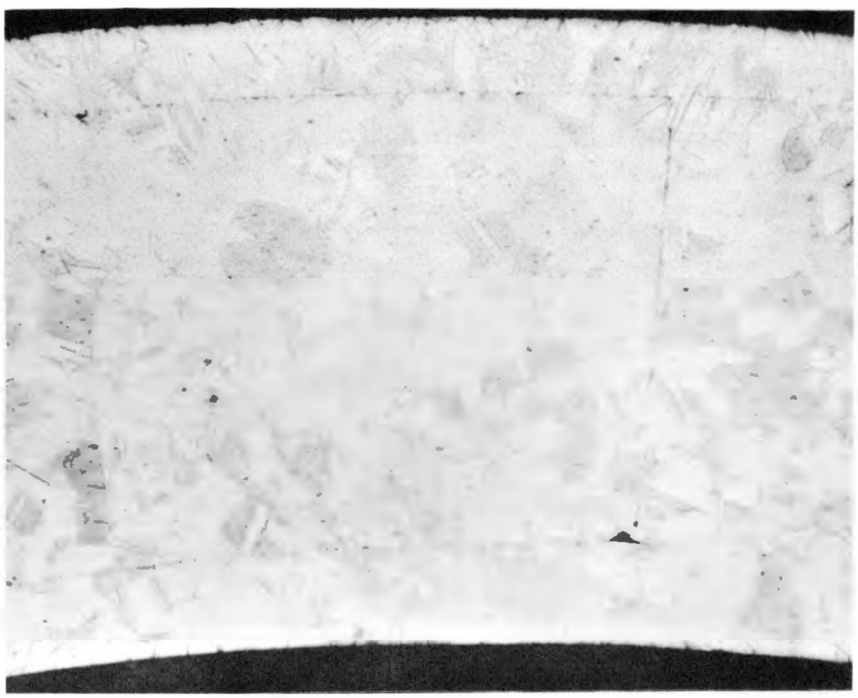


Fig. IV-2. Typical Machined Butt Steam 10-Degree Angle

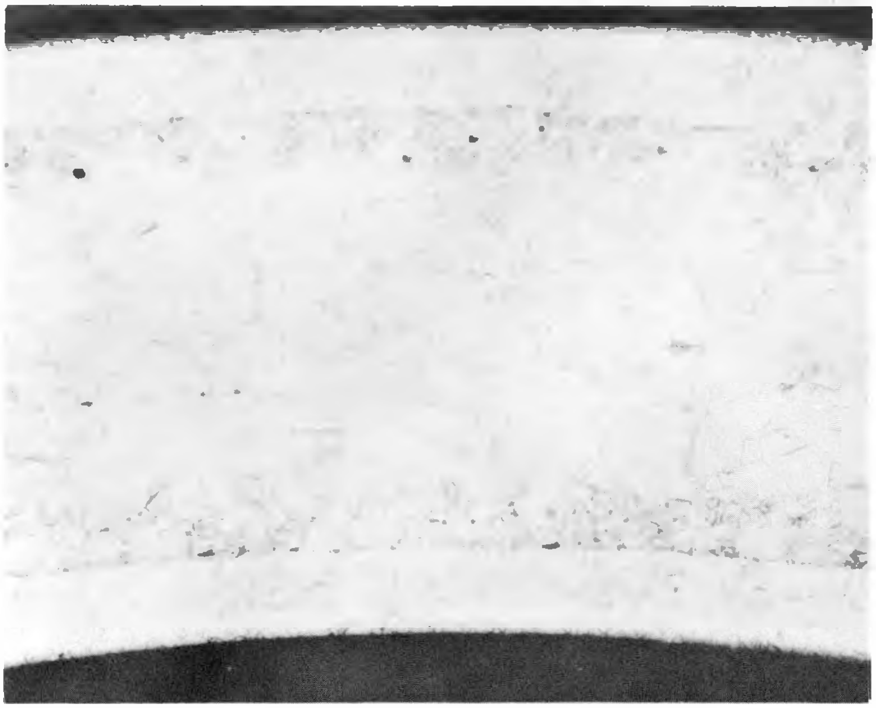


Fig. IV-3. Fine Grain Size Adjacent to Interface

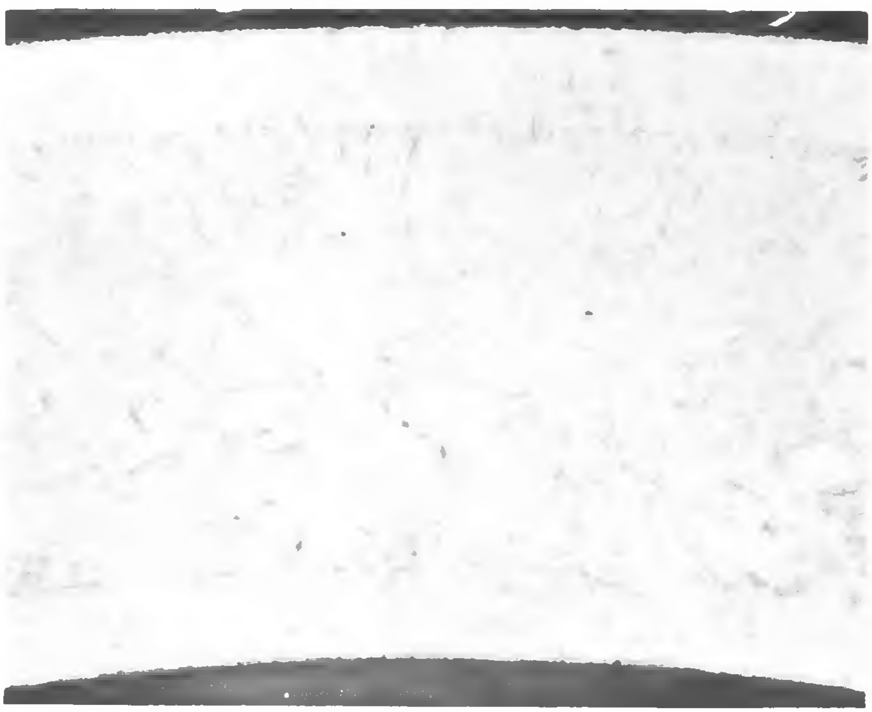


Fig. IV-4. Uniform Grain Size Through Cross Section

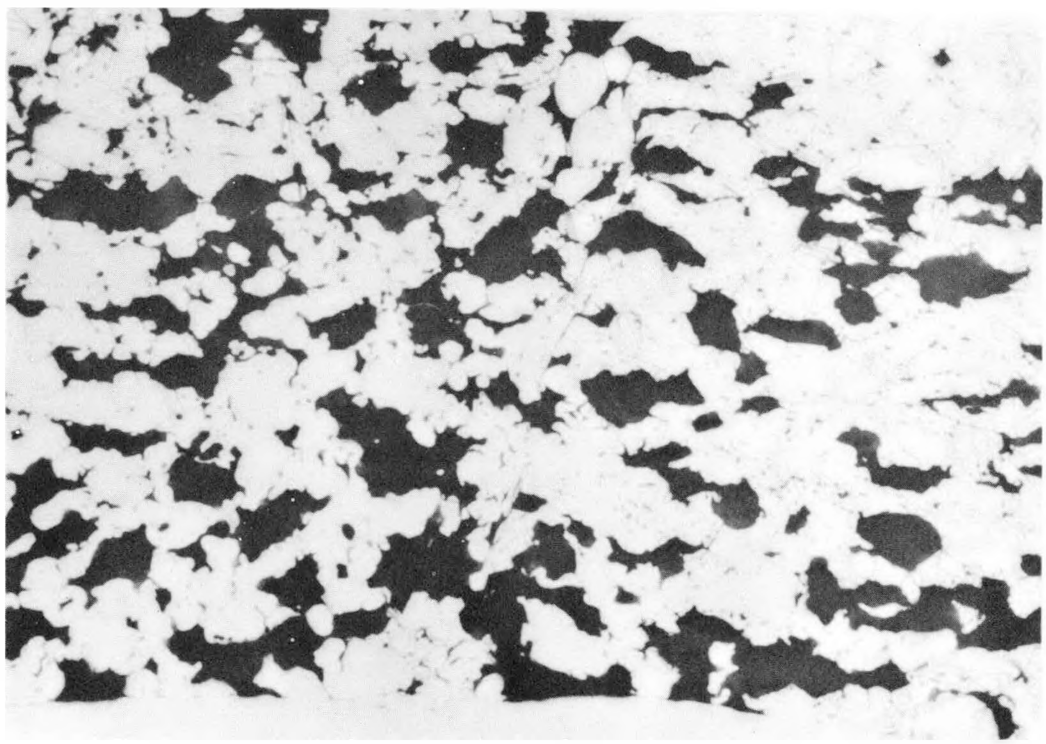


Fig. IV-5. Typical Core Butt Seam, Independent of Dead-End Configuration

After the production of some 30 lots of tubes, there was difficulty in meeting the length tolerance on the final drawing operation. Investigation showed that this was due to a thickness variation during the rolling operation. For the thickness value, a set of eight readings were averaged on the double width strip. This average fell within the tolerance. Upon checking the thickness of the single width strip, a larger value was found. The thickness readings taken on the double width strip did not indicate that the strip was crowned. At the present time, a deep throat sheet metal micrometer is being used which enables the operator to take readings at the thickest portion of the strip. It is believed that this change will alleviate the problem in drawing while changing the loading only slightly.

2. PMZ-1 Prototype Control Rods and Other Components

Approximately 15 kilograms of EU_2O_3 was delivered as Government-Furnished Equipment. Inspection after receipt of the material indicated slight discrepancies from AEC specifications. None of these was considered serious enough to warrant rejection of the material.

Investigation of the best method of fabricating PMZ-1 control rods indicated that the most economical subsequent recovery of EU_2O_3 could be made if it was mixed with iron powder of the same particle size. Magnetic separation followed by flotation, as required, was found to affect the most complete separation. Fabrication of the rods was initiated using prefabricated picture frames with welded enclosures to form a long slender box into which the proper mixtures of EU_2O_3 and iron powder could be packed. The assembly is then made watertight by welding.

Three heats of boron stainless steel alloy, to be used as lumped burnable poison material, was ordered from the Metallurgical Products Division of the General Electric Company. Preliminary analyses of the boron content yielded the following values:

Heat Number	Natural Boron (wt %)
U-059	0.284
U-062	0.846
U-063	0.523

It became necessary to remelt the first heat of the U-062 series since the boron content was too low. A remelt of the first heat of the U-063 series was required because of a hot short condition found during the forging of the material into a sound bar. Forging temperature was reduced to overcome this problem and further heats have been forged satisfactorily.

B. SUBTASK 5.2--CONVERSION OF UF_6 TO UO_2

W. Thompson

Delivery of 70 kilograms of UO_2 was completed during this quarter. The quantity on hand is more than sufficient to complete fabrication of PMZ-1 and irradiation test components.

C. SUBTASK 5.3--FUEL ELEMENT DEVELOPMENT

W. Precht, L. Frank

The general objectives of this task are to determine the limits of control rod and fuel element fabrication techniques and to determine the technique refinements which can be made. During this quarter, engineering assistance was given during the fabrication of the PMZ-1 fuel elements and the rare earth stabilization program was continued. During the next quarter, a program for control rod development will be initiated.

1. Stabilization of Eu_2O_3

The initial work to stabilize rare earth oxides utilized Nd_2O_3 and La_2O_3 to simulate Eu_2O_3. Table IV-1 shows the results through 67 hours of water reflux testing. The samples which showed promise were then subjected to 150 hours of autoclave testing at 570° F, 1200 psi. The results of this test are shown in Table IV-2. Similar compounds and mixtures were then prepared using Eu_2O_3. In all tests, the mixtures have been hand-blended wet, using acetone, then dried, pressed and sintered at 1500 to 1550° C in air. Table IV-3 shows the results of europium oxide mixtures prepared and tested as indicated. Although the additions of either Fe_2O_3 or SnO_2 show promise, the europium titanate form shows less overall weight change plus highest

TABLE IV-1

Effect of Stabilizing Additions to Rare Earth Oxides

Material (mole %)	Initial Weight (gm)	Weight After 120 hr in Air (gm)	Weight After 48 hr Distilled H_2O (RT) (gm)	Weight After 67 hr Water Reflux (gm)
Nd_2O_3-33 SnO_2-67	2.3193	2.3193	2.3194	2.3194
Nd_2O_3-33 ZrO_2-67	2.4863	2.4866	2.5147	Decomposed
Nd_2O_3-33 UO_2-67	4.4863	4.4857	4.5197	4.5300
Nd_2O_3-33 CeO_2-67	3.2173	3.2173	3.2276	Decomposed
Nd_2O_3-33 TiO_2-67	1.8290	1.8317	1.8291	1.8385
Nd_2O_3-25 MoO_3-75	Decomposed on firing			
Nd_2O_3-50 Al_2O_3-50	2.8580	2.8578	Decomposed	
Nd_2O_3-50 Fe_2O_3-50	2.7790	2.7788	2.7786	2.7762
Nd_2O_3-33 MnO_2-67	2.3491	2.3485	2.3474	2.3423
Nd_2O_3-50 Cr_2O_5-50	2.6250	2.6247	2.8566	Decomposed
Nd_2O_3-100	1.8458	1.8484	Decomposed	
Gd_2O_3-100	2.0040	2.0040	2.0038	Decomposed
La_2O_3-100	Decomposed after pressing			
La_2O_3-50 Fe_2O_3-50	1.8342	Not taken	1.8340	1.8330

TABLE IV-2

Effect of Stabilizing Additions to Rare Earth Oxides

Material	Weight Before (gm)	Weight After (gm)	Weight Loss After 150 Hours Autoclave (gm)
$Nd_2O_3 \cdot 2SnO_2$	2.3186	2.2727	0.0459
$Nd_2O_3 \cdot 2UO_2$	4.4865	4.4052	0.0813
$Nd_2O_3 \cdot 2\,TiO_2$	1.8275	1.7816	0.0459
$Nd_2O_3 \cdot Fe_2O_3$	2.7724	2.7239	0.0485
$Nd_2O_3 \cdot 2\,MnO_2$	2.3377	2.2868	0.0509
$La_2O_3 \cdot Fe_2O_3$	1.8338	1.7903	0.0435

TABLE IV-3

Effect of Stabilizer Additions to Eu_2O_3

Composition	Initial Weight Eu_2O_3 (wt %)	Initial Weight Admix (wt %)	Initial Weight (gm)	160-hr Dist H_2O	150-hr Boil H_2O	140-hr Autoclave 570° F at 1200 psi	Remarks	Change (wt %)
$Eu_2O_3 \cdot 2\,SnO_2$	54.2	45.8	0.3806	0.3845	-	-	0.0039% gain room temperature H_2O	1.02 gain
			0.4936	-	0.4936	-	Constant boiling H_2O	No change
			0.4046	-	-	0.4041	0.0005% autoclave loss	0.12 loss
$Eu_2O_3 \cdot 2\,UO_2$	39.5	60.5	0.8279	0.8442	-	-	0.0163% gain room temperature H_2O	1.97 gain
			0.9766	-	0.4080	-	0.5686% loss boiling H_2O	58 loss
			0.8773	-	-	0.8748	0.0025% loss autoclave	0.3 loss
$Eu_2O_3 \cdot 2\,TiO_2$	68.9	31.1	0.4925	0.4925	-	-	Constant room temperature H_2O	No change
			0.6041	-	0.6041	-	Constant boiling H_2O	No change
			0.9416	-	-	0.9412	0.0004% loss autoclave	0.04 loss
$Eu_2O_3 \cdot 2\,MnO_2$	67	33	0.3341	0.3331	-	-	0.0010% loss room temperature H_2O	0.3 loss
			0.5022	-	0.4988	-	0.0034% loss boiling H_2O	0.68 loss
			0.5066	-	-	0.5050	0.0016% loss autoclave	0.32 loss
$Eu_2O_3 \cdot Fe_2O_3$	69	31	0.5091	0.5093	-	-	0.0002% gain room temperature H_2O	0.04 gain
			0.6414	-	0.6408	-	0.0006% loss boiling H_2O	0.09 loss
			0.6388	-	-	0.6384	0.0004% loss autoclave	0.06 loss

comparable europium density. While samples were in process for the run shown in Table IV-3, some tests were made using high purity Eu_2O_3. Pellets of the as-received Eu_2O_3 were pressed and sintered at 1550 and 1760° C in air. All pellets lost their shape when subjected to water reflux tests and were reduced to powder after 1-1/2 hours of water refluxing.

Although these mixtures were stable when sintered in an air atmosphere, they must also be compatible with a stainless steel matrix and stable when fired in a reducing atmosphere. The data shown in Table IV-4 show the effect of sintering cermets of europium compounds in a hydrogen atmosphere with various stainless steel powders. The europium compounds were divided into two groups and subjected to two different heat treatments before incorporation into a cermet with stainless steel. One group was fired at 1550° C in air and the other was fired initially at 1550° C in air, then refired at 1250° C in H_2. All europium compounds were crushed and blended individually with each type of stainless steel. The cermet blends consisted of 45 wt % europium compound (corresponding to 30 wt % Eu_2O_3) plus 55 wt % stainless steel. The only pellets that retained their shape after water reflux testing were those containing the europium titanate. The conclusions that can be drawn from this test are:

(1) The compounds containing europium in a form other than the europium titanate were partially reduced during hydrogen firing.

(2) Those compounds fired only in air, prior to incorporation with a metal matrix, are somewhat protected by the metal matrix during subsequent hydrogen firing and water reflux testing.

(3) The europium titanate ($Eu_2O_3 \cdot 2\ TiO_2$) bearing cermets were the only samples to show a consistent volume decrease and to have a metallic luster after sintering in hydrogen.

(4) The europium titanate ($Eu_2O_3 \cdot 2\ TiO_2$) shows the most desirable overall properties since it is stable against hydration and also stable against reaction with silicon containing stainless steel.

TABLE IV-4

Properties of Europium Compounds--Stainless Steel Cermets *

Europium Materials	Type Stainless Steel Matrix	Sintered Density (gm/cc)	Change After Sinter (vol %)	Change After Water Reflux (wt %)	Reflux Time (hr)
Ceramic, Air-Fired, Then H_2 Fired					
$Eu_2O_3 \cdot 2\,TiO_2$	High silicon	5.22	-1.2	-0.035	Total 168
	Low silicon	-	-	-0.108	Total 168
	18-8	4.90	-0.7	-0.025	Total 168
$Eu_2O_3 + SnO_2$	High silicon	5.57	+8.1	-	Failed 96
	Low silicon	4.05	+43.7	-	Failed 12
	18-8	5.64	+2.3	-	Failed 24
$Eu_2O_3 + Fe_2O_3$	High silicon	5.24	+10.5	-	Failed 60
	Low silicon	5.36	+2.9	-	Failed 60
	18-8	5.21	+4.3	-	Failed 60
Ceramic, Air-Fired Only					
$Eu_2O_3 \cdot 2\,TiO_2$	High silicon	5.59	-7	-0.064	Total 168
	Low silicon	5.56	-6.5	-0.017	Total 168
	18-8	4.73	-12.3	-0.085	Total 168

* All cermets contain 45 wt% europium material and 55 wt % stainless steel and were sintered in hydrogen at 1250° C for two hours.

TABLE IV-4 (continued)

Europium Materials	Type Stainless Steel Matrix	Sintered Density (gm/cc)	Change After Sinter (vol %)	Change After Water Reflux (wt %)	Reflux Time (hr)
$Eu_2O_3 + SnO_2$	High silicon	5.08	+6.6	-0.475	Total 168
	Low silicon	5.84	-10	-	Failed 96
	18-8	5.41	-0.2	-	Failed 96
$Eu_2O_3 + Fe_2O_3$	High silicon	4.93	+5.4	-0.191	Total 168
	Low silicon	4.76	+7.3	-0.064	Total 168
	18-8	-	-	-	Failed 96

A similar cermet blend of europium titanate-stainless steel was autoclave tested at 670° F and 2200 psi for 140 hours. The weight loss was 0.06%. As a result of these tests, it is planned to fabricate control blades containing 42.5 wt % of europium titanate ($Eu_2O_3 \cdot 2\ TiO_2$) in stainless steel which will be equivalent to, in europium density, a 30 wt % Eu_2O_3, 70 wt % stainless steel mixture.

D. SUBTASK 5.4--PM-1 CORE FABRICATION

During the last quarter, fuel element cladding material for the PM-1 fuel elements was ordered. Preliminary analysis of the modified Type 347 stainless steel indicated a cobalt content of 0.005% and a tantalum content of 0.003%. This analysis meets the requirements of the cladding specification.

During the next quarter, the cladding material will be delivered and orders will be placed for the remaining core and fabrication materials and tools.

V. TASK 7--FABRICATION AND ASSEMBLY OF THE PLANT

A. SUBTASK 7.1

See under Task 4, Section III.

B. SUBTASK 7.2--SECONDARY SYSTEM

During this report period, no efforts were planned for the secondary system equipment. A number of vendors were selected, as reported under Task 4.2, and liaison with these, and others to be selected soon, will be conducted under Subtask 7.2 during the next quarter. It is planned that vendors be selected for all secondary system equipment, piping, buildings, electrical material and maintenance equipment during the next quarter.

VII. TASK 12--LOADING DEMONSTRATION

Project Engineers--A. Layman, C. Fox

The objectives for this task are those necessary to perform a loading and unloading demonstration of the PM-1 package aboard a C-130A aircraft.

Organizational planning was initiated during the quarter to make possible the successful completion of the loading demonstration. Martin manufacturing representatives, who specialize in this type of work, are assisting in these plans. 27th and 28th April were selected as the dates for the demonstration. The equipment will be set up and a "dry run" conducted by Martin personnel on the 27th with the official demonstration on the 28th. An Air Force loadmaster has been assigned and will be included in the planning. It is planned that he will come to Baltimore prior to the demonstration to participate in the detailed plans.

A C-130A aircraft has been scheduled for use during the demonstration and will arrive at The Martin Company airport on 26 April and depart on 29 April 1960. The demonstration will include loading and unloading of the C-130A and is planned to include flying the aircraft with the loaded test package in the shipping position.

Present planning for the demonstration requires use of an Air Force 40-foot C-2 trailer for transporting and loading the package.

During the next quarter, test planning will continue, a trailer will be readied and the demonstration will be conducted.

VIII. TASK 13--MANUALS

The objective of this task is the preparation of operating, maintenance and training publications required for the PM-1 nuclear power plant.

A. SUBTASK 13.1--OPERATING MANUAL PREPARATION

J. F. Holliday

During the quarter, it was planned to initiate efforts on the plant Operating Manual. This was accomplished. A table of contents, a chapter outline and a sample chapter writeup were prepared for internal review and comment on the manual form and contents. System description drafts were prepared for the Main and Auxiliary Steam System, the Main Condensate System, the Feedwater System, and the Extraction Steam and Heater Drain System.

During the next quarter the contents and the form of the manual will be established and will be submitted for approval. In addition, preliminary system description drafts and the manual draft flow diagrams and figures will be completed, and the preparation of the specific operating procedures will have been initiated. It is also planned to start efforts on the plant Assembly and Disassembly Manual and the Maintenance Manual.

B. SUBTASK 13.2--TRAINING MANUALS

F. McGinty

The objective of this subtask is to prepare a manual designed for instructor and training administrator use. It will contain a syllabus of instructions, training chart, course outlines and detailed lesson outlines.

The work planned for this reporting period was:

(1) To obtain approval of a Training Manual topical outline.

(2) To initiate preparation of introductory and nontechnical portions of the Training Manual.

The topical outline for the Training Manual was submitted to the USAEC for review. Approval was received and preparation of introductory material commenced.

This work will continue next quarter.

X. TASK 15--PROJECT SERVICES

Project Engineer--C. Fox

A. SUBTASK 15.1--FILM AND PHOTOGRAPHS

The following planned efforts were accomplished during the quarter.

(1) Film coverage was obtained on the PM-1 condenser model tests and the quarter-scale reactor flow model tests.

(2) Progress photographs were obtained of the PM-1 test package fabrication, test and test fixtures of the PM-1 condenser test model and of the PM-1 actuators.

(3) The outline of the project documentary film was completed and submitted for approval.

Film and progress photo coverage will be obtained as required next quarter.

Several of the more interesting progress photographs are presented with their respective development tasks.

B. SUBTASK 15.2--MODELS

In accordance with established specifications, actual fabrication was initiated for the five facility display models to represent the installed configuration.

The planned objective during the fourth quarter was to concentrate all efforts toward the completion of the first display model by 1 March 1960, if feasible, but no later than 12 March 1960.

The first model will be completed during the next quarter in accordance with this plan and complies with specifications previously submitted with the exception of the model size. This will not be in accordance with Item 3 in the specification, i.e., "Overall size of each assembled model shall be approximately 3 feet by 4 feet." The completed overall model size will now be approximately 4 feet by 4-1/2 feet.

Effort will continue during the coming period on the remaining display models to meet the schedule completion date of 10 June 1960 in lieu of April 1960. This revision in end dates is due to incorporation of design changes and to efforts to complete the first model early. This deviates from the more efficient method of making all parts and assemblies for all models simultaneously.

Work will continue during the next quarter on production of the remaining four models.

XI. TASK 16--CONSULTING

The purpose of this task is to secure technical services as required for the PM-1 Project.

During this quarter, the services of Mr. W.B. Willsey, a water chemistry consultant, were used to obtain:

(1) Recommended listing of laboratory equipment for water analysis.

(2) Recommended listing of procedures for manual water analyses.

(3) Recommendations and suggestions concerning the primary and secondary in-line water analysis system.

The Gibbs and Hill Company provided support in the following areas during this quarter:

(1) Review of the switchgear, motor control center and turbine-generator work performed by Westinghouse.

(2) Completion of the PM-1 Insulation Specification, MN-2011.

(3) Review of the primary system piping layout.

(4) Review of the condenser model test program.

(5) Review of the diesel generator specification.

(6) Estimate of operating supplies and storage area requirements for the PM-1 plant.

The work accomplished by Gibbs and Hill in the PM-1 steam-electric system area during this quarter was:

(1) Reviewed all secondary steam-electric system equipment proposals and recommended manufacturers.

(2) Reviewed and corrected, as required, all secondary system equipment certified drawings for return to the vendors.

(3) Submitted the final secondary steam-electric system heat balance, flow diagram, one-line electric diagram and miscellaneous wiring diagram for review and approval.

(4) Submitted the final standard and special tool list, test procedure and startup and shutdown procedures.

(5) Submitted the valve and specialty list, bill of materials, general mechanical layout drawings and general electrical layout drawings for information and review.

(6) Completed and submitted the secondary steam-electric system process instrumentation and control specification (MN 7601).

During the next quarter, the efforts of Gibbs and Hill will be to:

(1) Review the condenser design and test results.

(2) Continue final design of the secondary steam-electric system through incorporation of vendor data.

XII. TASK 17--REPORTS

The objective of this task is to accomplish the timely preparation of reports required by the USAEC.

A. SUBTASK 17.1--HAZARDS REPORTS

G. Zindler

During this reporting period, approval was received from the Government to continue the design of the PM-1 as defined in the report submitted earlier.

No specific efforts are planned under this subtask for the next quarter.

B. SUBTASK 17.2--REPORTS OTHER THAN HAZARDS

E.H. Smith

This subtask includes all reports submitted to the USAEC except those on hazards.

During the fourth project quarter:

(1) A classified technical memorandum on Incore Instrumentation was prepared and delivered to the AEC (MND-M-2297, Revision).

(2) A topical report on controls and instrumentation was prepared for distribution (MND-M-1914).

(3) The third quarterly progress report was completed and delivered to the AEC (MND-M-1814).

(4) Subsystem submittals for the final design were continued, substantially on schedule.

During the next project quarter, the final design submittals will be completed and a topical report on the final design analysis will be initiated. Any required amendments to the Preliminary Hazards Summary Report, reflecting pertinent design changes, will be prepared. The fourth quarterly progress report will be prepared and delivered.