MND-M-1813

REACTORS—POWER

PM-1 NUCLEAR POWER PLANT PROGRAM
2ND QUARTERLY PROGRESS REPORT
1 JUNE TO 31 AUGUST 1959

By
J. S. Sieg
E. H. Smith

October 5, 1959

Nuclear Division
Martin Company
Baltimore, Maryland

UNITED STATES ATOMIC ENERGY COMMISSION
Technical Information Service

This report has been reproduced directly from the best available copy.

Printed in USA. Price $4.00. Available from the Office of Technical Services, Department of Commerce, Washington 25, D. C.

AEC Technical Information Service Extension
Oak Ridge Tennessee

MND-M-1813

PM-1 Nuclear Power Plant Program

2ND QUARTERLY PROGRESS REPORT

1 June to 31 August 1959

Contract AT(30-1)-2345
5 October 1959

Prepared By:

J. S. Sieg

J. S. Sieg

E. H. Smith

E. H. Smith

Approved By:

F. Hittman

F. Hittman
Project Manager

ABSTRACT

This report contains a description of the work accomplished during the second contract quarter (1 June to 31 August 1959) 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 second project quarter, were completion of the plant parametric study and preparation of the preliminary design. A summary of design parameters is given.

Systems development work included study and selection of packages for full-scale testing, a survey of in-core instrumentation techniques, control and instrumentation development, and development of components for the steam generator, condenser, and turbine generator, which are not commercially available.

Reactor development work included completion of the parametric zero-power experiments and preparations for a flexible zero-power test program, a revision of plans for irradiation testing PM-1 fuel elements, initiation of a reactor flow test program, outlining of a heat transfer test program, completion of the seven-tube test section (SETCH-1) tests, and evaluation of control rod actuators leading to specification of a magnetic jack-type control od drive similar to that reported in ANL-5768.

Completion of the preliminary design led to initiation of the final design effort, which will be the principal activity during the next two project quarters. Preparations for core fabrication included procurement of core cladding material for the zero-power test core, arrangement with a subcontractor to convert UF_6 to UO_2 and to commence delivery of the oxide during the next quarter, development of fuel element fabrication and ultrasonic testing techniques, and study of control rod materials, UO_2 recovery techniques, and boron analysis methods. Preliminary work on site preparation was pursued with receipt of USAEC approval for a location on the eastern slope of Warren Peak at Sundance, Wyoming. A survey of this site is underway. A preliminary Hazards Summary Report is also in preparation. For the preceding period, see MND-M-1812.

Blank Page

FOREWORD

This is the second Quarterly Progress Report submitted to the US 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 June through 31 August 1959.

CONTENTS

CONTENTS (continued)

CONTENTS (continued)

PROGRAM HIGHLIGHTS

1. It appears that the problems resulting from operation of nuclear instrumentation in close proximity to a radar transmitter will not be great (Task 1).
2. Inconel was selected for use on the primary side of the steam generator (Task 1).
3. The scope of the irradiation testing program was reduced (Task 2).
4. The decision to utilize a magnetic jack actuator was made and a vendor selected (Task 2).
5. The Parametric Study was completed (Task 3).
6. The Preliminary Design was completed (Task 3).
7. The decision to limit placement of burnable poison to discrete lumps in the core was made (Task 3).
8. Fuel element development work continued successfully (Task 5).
9. The site location recommended by The Martin Company was approved (Task 11).

INTRODUCTION

This is the second of 12 Quarterly Progress Reports required by Contract AT(30-1)-2345 between The Martin Company and the USAEC.

During the second quarter, Tasks 1, 2, 3, 4, 5, 11, 14, 15, 16, and 17 were active. Task 3 terminated at the end of the quarter; Tasks 6 and 7 are scheduled to commence during the next quarter.

The Plant Design Summary appears in this report for the first time. Subsequent summaries will include an identification of those parameters changed during the period of the report.

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)	9.35
Reactor thermal power, design (mw)	10.47
Core life, nominal (mw yr)	18.7

2. Core Design Characteristics

Geometry, right circular cylinder (approx)	
Diameter, average (in.)	23
Active length (in.)	30
Overall length (in.)	33-1/8
Core structural material	Stainless steel
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 348 stainless steel Co and Ta controlled
Pitch, triangular (in.)	0.665
Number of tubes	862
Number of fuel tubes	725 to 750
U-235 inventory (kg)	26.7
U-235 burnup (kg)	9.0
Meat composition (UO_2 - w/o)	25

3. Core Heat Transfer Characteristics

Heat flux (Btu/ft^2 - hr)	
Average	70,000
Average coolant temperature (°F)	463

4. Reactor Hydraulic Characteristics

Coolant flow rate (gpm)	1900

B. SYSTEMS DESIGN

1. General Plant

Reactor power output, nominal (mw)	9.35
Steam generator power output, nominal (mw)	9.35
Steam pressure, full power, minimum (psia) (saturated)	300
Steam quality, full power	1/4% moisture

2. Main Coolant System

Number of coolant loops	
Coolant flow rate (gpm)	1900
Coolant system design pressure (psia)	1485
Coolant velocity in piping, main loop (ft/sec)	23
Coolant pipe size, main loop, inches nominal, schedule 80	6
System basic material	
Reactor pressure vessel	AISI 347
Piping	AISI 316
Remainder	AISI 304
Main coolant pump (mechanical seal type) (number)	1
Steam generators	
Number of units	1
Design pressure (approx) (psi)	600
Type	Vertical with integral steam drum and separators
Temperature primary inlet, full power (approx) (° F)	481
Temperature primary outlet, full power (approx) (° F)	445
Temperature steam side outlet, full power (°F)	417
Access	Shell and tube side, bolted port

3. Pressurizing and Pressure Relief System

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

4. Coolant Purification and Sampling System

Number of purification loops	1
Purification device	Ion exchange resin(s)
Inlet temperature to ion exchanger (max) (° F)	140
Maintenance provisions	Cartridge type 1-yr life

5. Primary Shield Water System

Primary shield water cooler	Air blast type
Purification loop	Ion exchange resin(s)
Maintenance provisions	Cartridge type 1-yr life

C. SECONDARY SYSTEM

1. General Plant

Steam flow at full power (approx) (lb/hr)	35,000
Steam conditions at full power, 300 psia, dry and saturated (°F)	417
Feedwater flow at full power (approx) (lb/hr)	35,500
Rated gross electrical output (kw at 0.8 pf)	1250
Net electrical output (kw at 0.8 pf)	1000
Line voltage (4 wire wye)	4160/2400
Cycles	60
Phase	3
Auxiliary equipment voltage	480
Process heat 6609 lb/hr of 35-psia dry and saturated steam (Btu/hr)	7×10^6

Design elevation (ft)	6500
Auxiliary power (approx) (kw)	135

2. Turbine Generator Set

Type	Single extraction turbine
Throttle flow, full power (approx) (lb/hr)	26,600
Throttle pressure (psia)	290
Turbine steam exhaust conditions, full power	
Pressure (inches Hg ab)	9
Moisture (%)	12.2
Lube oil cooler	Air cooled
Turbine speed (approx) (rpm)	7500
Generator rating (kva)	1562.5
Generator rating (kw at 0.8 pf)	1250
Generator type	Salient pole
Generator speed (rpm)	1200

3. Condenser System

Number of units	2
Type	Direct air-to-steam
Duty (Btu/hr rejected) (approx per unit)	10.1×10^6 each

4. Feedwater System

Deaerator	
Type	Atomizing
Feedwater design flow (approx) (lb/hr)	40,000
Design pressure (psia)	50
Oxygen removal guarantee (cc/1 remaining)	0.005
Storage (minutes)	5
Boiler feed pumps	
Number	2
Drivers	One steam driven One electrically 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 (°F)	40
Condensate return temperature (°F)	180
Feedwater storage tank	
Capacity (approx) (gal)	2750
Turbine steam bypass system	
Type	Manual
Auxiliary generator unit	
Type	High-speed diesel
Number	1
Capacity (kw)	200
Electrical characteristics	480 volts, 60 cps, 3ϕ
Emergency power	
DC power source	Batteries
AC power source	3 unit MG set

D. PACKAGING

Type	
Primary system	Tank
Secondary system	Arctic, integral housing
Number	
Primary system (including housing)	
Non-contained	4
Contained	6
Secondary system (including housing)	9

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

Project Engineer--Subtasks 1.3: R. Akin, 1.1, 1.2, 1.5, 1.6: C. Fox

The objective of this task is to provide for the performance of investigations which are prerequisite to system design.

A. SUBTASK 1.1--PACKAGE DEVELOPMENT AND TEST

J. Cosby

During the second quarter, it was planned to select two packages for full-scale testing and to begin the final package test specifications, the detailed test schedule, and the final test package design. This was accomplished. These efforts will be completed next quarter and fabrication of test packages will begin.

During the quarter, a literature survey of standard package test specifications was completed. A study was then made of the overall environment of the PM-1 packages and a preliminary test specification was written. Basically, the test program will be as follows:

(1) Structural element tests to ultimate conditions.

(2) Static, handling, and load tests of the Heat Transfer Apparatus Package (B-2) to limit conditions:

  (1) Hoisting tests

  (2) Supporting skid at center and one end

  (3) Supporting skid at ends

  (4) Testing fittings

  (5) Testing slings and tow cables.

(3) Impact drop tests of package (B-2) to limit conditions:

  (1) Uni-axial impact tests

  (2) Edge drop tests

  (3) Corner drop tests.

(4) Environmental testing of packae B-1 (Switch Gear Package):

  (1) Heat loss test

  (2) Seal and hardware tests.

(5) Shelter, assembly, and load tests.

  (1) Erection trials

  (2) Wind load tests

  (3) Snow load test

  (4) Seal and hardware tests.

(6) Static, handling, and load tests of package B-2 to ultimate conditions (simulated aircraft crash-landing loads).

(7) Heat and fire tests:

  (1) Temperature tests of specimens

  (2) Fire tests of a complete panel.

(8) Vibration test on completed Controls Package (B-3) in the "as-shipped" conditions.

(9) Package loading demonstration (Task 12):

  (1) Loading of package B-1 or B-2 into C-130A aircraft from C-2 trailer

  (2) Unloading of package onto C-2 trailer and onto ground, without use of crane

  (3) C-130A flight checkout with 30,000-lb package

  (4) Crane handling of 30,000-lb package.

A study was made of the various shelter packages in order to select the most critical single package to be subjected to the static and dynamic test program. For this purpose the Heat Transfer Apparatus Package was chosen since its gross weight is close to the maximum allowed, and the mounting of the various pieces of equipment will permit the maximum test of design considerations. This package will be built with equipment supports and with simulated equipment.

The selection of the Heat Transfer Apparatus Package as the critical test package automatically establishes the Switch Gear Package as the mate for the shelter assembly tests. The two packages will be erected, together with the end sections and flooring, to evaluate erection procedures, seal performance, and wind and snow loads. Erection of the two packages will provide a complete check of secondary system building erection methods and will provide the detail necessary to write the erection procedure manual.

The Switch Gear Package will be built without simulated equipment. The rigid base, wall panels, and roof panel will be built to provide a fully enclosed package for environmental testing at -65° F to determine basic heat transfer data, seal and hardware performance, and any effects of temperature gradients through the package walls. One end of this package will have a door in it to test the Decontamination Package door design with respect to seals and hardware.

During the quarter, a sample of the proposed polystyrene panel insulation was vacuum tested to simulate exposure to low atmospheric pressure in unpressurized aircraft at high altitude (50,000 feet). No damage occurred.

B. SUBTASK 1.2--INCORE INSTRUMENTATION

G. Zindler, D. Talbot

The objectives of this subtask are to determine methods of measuring fuel element temperature, coolant channel pressure, local neutron flux, and various other operating core parameters and to incorporate certain of these methods into the design of the PM-1 reactor.

During the second project quarter, instrumentation techniques developed elsewhere were reviewed and a means for introducing these instruments into the PM-1 core was included in the preliminary design of the plant.

During the next quarter the feasibility study of PM-1 incore instrumentation will be completed.

1. Temperature Measurements

According to the literature, the greatest incore instrumentation success to date has been in the area of temperature measurement. The following thermocouple was used in the S 1W and appears generally suitable for use with the PM-1.

Supplier: Thermo Electric Company

Type: chromal-alumel

Lead wire diameter: 0.010 in.

Sheath OD: 0.066 in.

Sheath thickness: 0.007 in.

Sheath material: 304 SS

Insulation: zirconium oxide

Measurement accuracy: ± 2° F

Life expectancy: 70% of units operable at the end of one year.

The following types of problems have plagued many experimenters.

Thermocouple shorting.- In many cases this problem has been traced to the cermet insulation utilized. When the thermocouple lead is cut, prior to brazing a stainless steel bead on the tip, moisture is absorbed by the $Mg0_2$ insulation, resulting eventually in a short. When $Zr0_3$ is used, moisture is not attracted and the problem is resolved.

Lead failure.- Cracking of thermocouple lead material due to the techniques utilized in bending the leads has caused many thermocouples to fail. Indiscriminate lead-bending caused cold working in the material; since the stresses (probably due to chloride stress corrosion) were not relieved, the leads cracked.

Well problems.- Leakage of water into the thermocouple wells has resulted in boiling of the water and erroneous readings. Improved sealing techniques have minimized these problems. In some cases, during thermocouple calibration, the different thermal expansion rates of the thermocouple and fuel element materials caused the thermocouple material to yield--resulting in inadequate bottoming of the thermocouple in its well. This problem has been alleviated by filling the wells with graphite.

2. Pressure Measurements

Pressure measurements have been reported by one experimenter who utilized a static tap at both ends of a flow channel. When local boiling occurred, results were erratic. Both static and dynamic sensing heads were utilized later--the channels were monitored at locations beyond the fuel meat to eliminate heating effects. Hydraulic noise, caused by the primary pumps, appeared in the pressure signals.

No reliable pressure signals have been received to date from a boiling channel.

3. Neutron Flux Measurements

Very few incore flux measurements have been made. Those to date have, for the most part, relied on miniature ion chambers. Although those developed at Bettis and KAPL were reported to have failed very shortly after being placed in a full-power core, the chamber developed at the GE West Lynn Plant is reported to be successful. Their chamber is about 1-1/2 inches long and 3/8 inch in diameter, costing about $1,500.

4. Instrument Bundle

A removable center fuel bundle has been incorporated into the preliminary PM-1 core and pressure vessel design. This design feature permits the insertion and replacement of instrumented assemblies.

C. SUBTASK 1.3--SHIELDING MEASUREMENT

The objective of this subtask is the analysis of potential site shielding materials. No work was planned or performed during the second project quarter.

During the next quarter the earth borings obtained from the site (Task 11) will be analysed chemically and spectroscopically. Activiation and attenuation analyses will be performed on these and other possible fill materials.

D. SUBTASK 1.5--INSTRUMENTATION AND CONTROL

G. Zindler

The objective of this subtask is to develop an advanced, highly reliable, and easily operable and maintainable instrumentation and control system. This effort is subcontracted to Stromberg-Carlson Corp.; they are performing research and development work and will assemble the instrumentation specified by The Martin Company.

The major work of this reporting period involved:

(1) Period circuit development

(2) Study of fault monitoring and self-checking methods

(3) A radar interference study.

During the next project quarter, it is anticipated that:

(1) Stromberg-Carlson will complete its research and development efforts.

(2) Martin will complete the design of all plant controls and instrumentation; an internal review of the design will be made.

1. Period Circuit Development

Four methods of measuring the reactor period have been studied and compared. These are:

(1) The RC differentiator integrator network (RCDI) presently used in Stromberg-Carson equipment.

(2) The Stromberg-Carlson integrating amplifier feedback system (SCIAF).

(3) The Martin time delay system (MTD).

(4) An RC differentiating network followed by a nonlinear filter.

The performance of the RCDI is at least as good as that of methods (2) and (3) above. On the basis of reliability and simplicity, the RCDI is superior to these methods. The performance of the RCDI has been further improved, without appreciably slowing its response time, by adding a nonlinear filter. The result of this, method (4), remains less complex than methods (2) or (3). Development of the optimum nonlinear filter for use with the RC differentiating network will be continued.

2. Fault Monitoring and Self-Checking

Preliminary design efforts in this area have disclosed the need for two separate methods of indicating and locating a fault. Comparison techniques are employed between adjacent operating channels which are not continuously in use. Continuous testing is employed for power range channels and/or those channels or units associated with a scram.

After a fault has been noted (by either method) a manual procedure must be followed to locate it. The manual operations involved do not require any great skill in instrument repair.

Since these techniques offer the maximum fault monitoring and self-checking consistent with simplicity and reliability of operation, fabrication and demonstration of these systems will be supported.

3. Radar Interference Study

This study was made to determine whether adverse effects on the primary loop instrumentation and control equipment would result from close proximity to a high energy, high frequency, radar transmitter. The study consisted of laboratory evaluation followed by field testing of representative circuitry. A Keithley 410 micro-microammeter and certain input modules of the Stromberg-Carlson equipment were considered to be most susceptible to radar interference.

Both laboratory and field tests showed electromagnetic radiation to have little or no effect on the test units. During the first part of the field test, small and somewhat erratic readings were observed. These were believed to be due to failure to provide sufficient warm-up time for the micro-microammeter, although only further testing can prove this. It was concluded that very little electromagnetic energy penetrates the RG149/U shielded cable.

It is recommended:

(1) That special attention be given to the terminals of the RG149/U cable.

(2) That connections be made with connectors constructed for microwave equipment.

(3) That high-quality dielectrics be used when necessary.

(4) That terminal circuits be enclosed and shielded against high frequency electromagnetic energy.

Circuits enclosed in standard module cases will be mounted on the output end of the RG149/U cable. These cases are adequately shielded for low frequency radiation, but microwaves could easily pass through slots and small crevices.

It is recommended, as a general precaution, that all equipment be located in a building with double continuous copper screening enclosing the inside; or in a metal building enclosed by a single continuous copper screen.

E. SUBTASK 1.6--SECONDARY SYSTEM DEVELOPMENT

W. Koch, L. Hassell, R. Groscup

The objective of this subtask is to develop those components required for the PM-1 Nuclear Power Plant which are not commercially available.

During this quarter, the following work was accomplished:

(1) Control analysis work was essentially completed.

(2) Steam generator design conditions were determined.

(3) Condenser model design, preliminary test procedure, test schedule, and test site selection were completed.

(4) Turbine generator design was continued.

During the next quarter:

(1) The final steam generator design and specifications will be completed.

(2) The final condenser test model test procedure will be completed.

(3) Final condenser model installation and test procedures will be coordinated with Eglin AFB personnel.

(4) Work will continue on condenser model fabrication.

(5) Control analysis work will be completed.

(6) Secondary system design development efforts will continue, with emphasis placed on the condenser and turbine generator.

(7) Procurement type specifications for all elements of the secondary system will be completed.

1. Control Analysis

The ability of the plant to respond to an instantaneous 30% load change is a function of the control system and of the inter-related transient responses of the reactor, the steam generator, and the turbine generator.

Analog data were obtained to define the response of the plant. In general, this study indicated that the dip in steam pressure to be expected at the steam generator outlet and turbine generator inlet as a result of the application of a load transient is slight. Westinghouse Corp. has indicated that turbine control and performance will not be adversely affected.

A mechanical-hydraulic turbine governor of Westinghouse design, the basic component of which is a centrifugal throttle control, is being specified for the PM-1. The response of the mechanical-hydraulic governor is shown in Fig. I-1. Curve A shows the response of a 5% basic regulation governor with no reset or compensation action. Curve B shows the response of a 5% basic regulation governor with full compensation. Curve C shows the response of a 5% basic regulation governor with full compensation plus an external load-sensing device. It was concluded that transient response is adequate when the turbine is equipped with the Westinghouse mechanical-hydraulic turbine governor (Curve A).

2. Steam Generator Design

The basic steam generator design will be discussed as part of Subtask 3.2.

The relative merits of stainless steel and inconel as tube materials have been investigated under this subtask. Pressurized water steam generators now in service use stainless steel tubes. Experience with these generators, including the results of various corrosion tests, shows that a chloride stress corrosion problem exists. This problem does not occur when inconel tubes are used.

The major problems associated with the use of inconel tubes concern the making of tube-to-tube sheet welds. Westinghouse, under a contract with the AEC and the General Electric Company for Knolls Atomic Power Laboratory, has investigated and reportedly resolved this welding problem. On August 28, 1959 their inconel tube-to-tube sheet welding procedure (using 1/2-inch OD tubes) was qualified for the Navy.

To gain added insurance against chloride stress corrosion, inconel has been selected for the tube material; an inconel-clad (primary side) tube sheet will also be used. The remaining primary side wetted surfaces will be clad with stainless steel. The steam generator shell will be fabricated from commercial quality carbon steel. Handholes are provided in the primary hemisphere to allow access for inspection and for tube plugging. Handholes are also provided in the upper portion of the shell for inspection purposes and for maintenance of the steam moisture separating equipment.

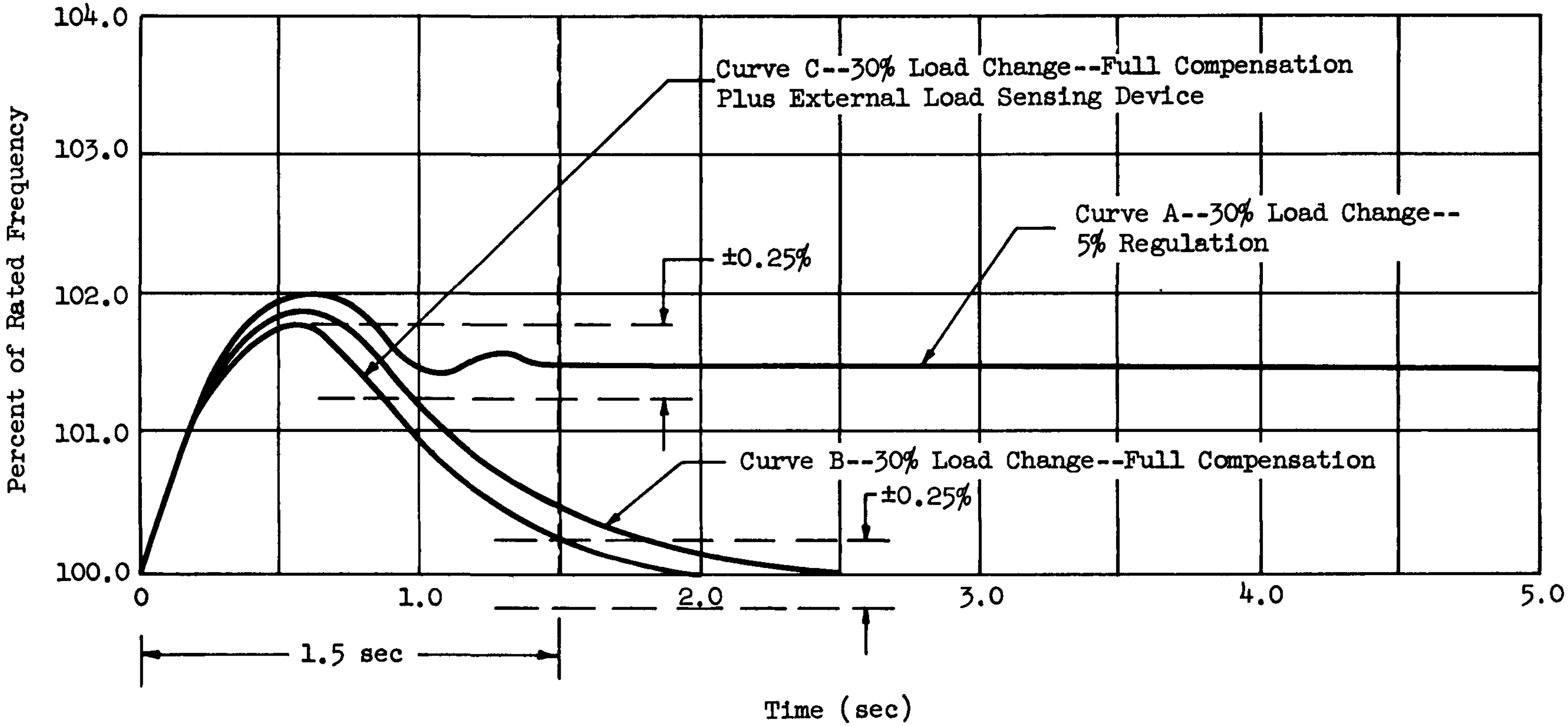


Fig. I-1. Mechanical-Hydraulic Governor Response

The response of the steam generator to the reference maneuver of instantaneous load change of 30% of plant capacity has been investigated on an analog computer. The steam generator general equations were derived by taking a mass and heat balance with constants determined from the latest design values of area, volume, level, etc., supplied by Westinghouse. While the equations were, of necessity, kept fairly simple, it is felt that they included sufficient detail to assure the accuracy of the results.

This steam generator model was then combined with primary system, turbine, and feedwater controller models. Control of the primary system was achieved using only the negative temperature coefficient (-1.3×10^{-4} K/° F). An isochronous governor was assumed to provide steam turbine control.

The reference maneuver was performed, both up and down, over two ranges: 125 to 425 kw and 950 to 1250 kw. The results of these maneuvers are shown in Figs. I-2 and I-3. The maneuver of primary interest is the load change from 950 to 1250 kw, since it results in the greatest dip in steam pressure. From Fig. I-3, it can be seen that this change is 8.5° F or about 30 psi. The dip in temperature below the new steady-state value at 1250 kw is approximately 1.5° F or 6 to 7 psi.

It has been concluded that the steam generator transient response to the reference maneuver is adequate.

3. Condenser System

Efforts undertaken during the parametric study to evaluate the types of condenser systems applicable to the PM-1 plant were completed (MND-M-1812). Based upon these studies, the advantages of a direct steam-to-air heat exchanger were judged sufficient to warrant building and testing a full-scale mode. The test objective is to determine condenser performance characteristics under normal and arctic environmental conditions. Preliminary test outlines were developed.

A direct steam-to-air condenser model was designed and is being built. It is a replica of one of the two proposed PM-1 condenser units (see Fig. I-4), except that it is approximately one-half as long (the model is 16 feet instead of 30 feet long). There are also three induced-draft fans instead of four. Both ends of the structure are enclosed; the after end houses the vertical finned-tube air cooler section while the forward end is blanked off to eliminate air flow. The total heat transfer surface is 26,000 square feet instead of the approximately 32,000 square feet that would be contained in each one of the two proposed PM-1 condensers. Other features are similar to those specified for the plant during preliminary design (see Subtask 3.2).

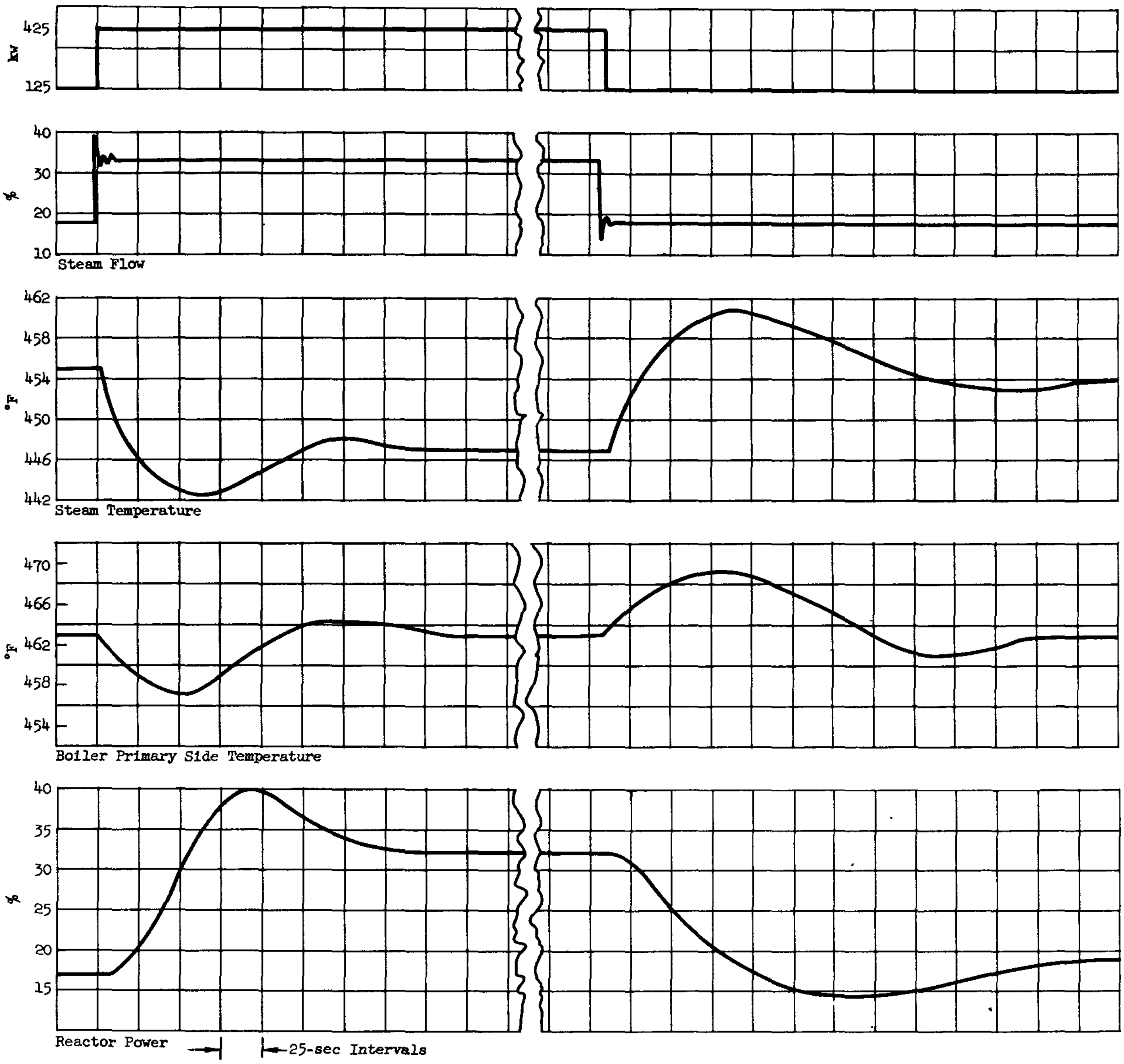


Fig. I-2. Load Change (125 to 425 to 125 kw); No Heating; 12% Auxiliary Steam Load; SG Secondary Water 1500 lb at 15% Load

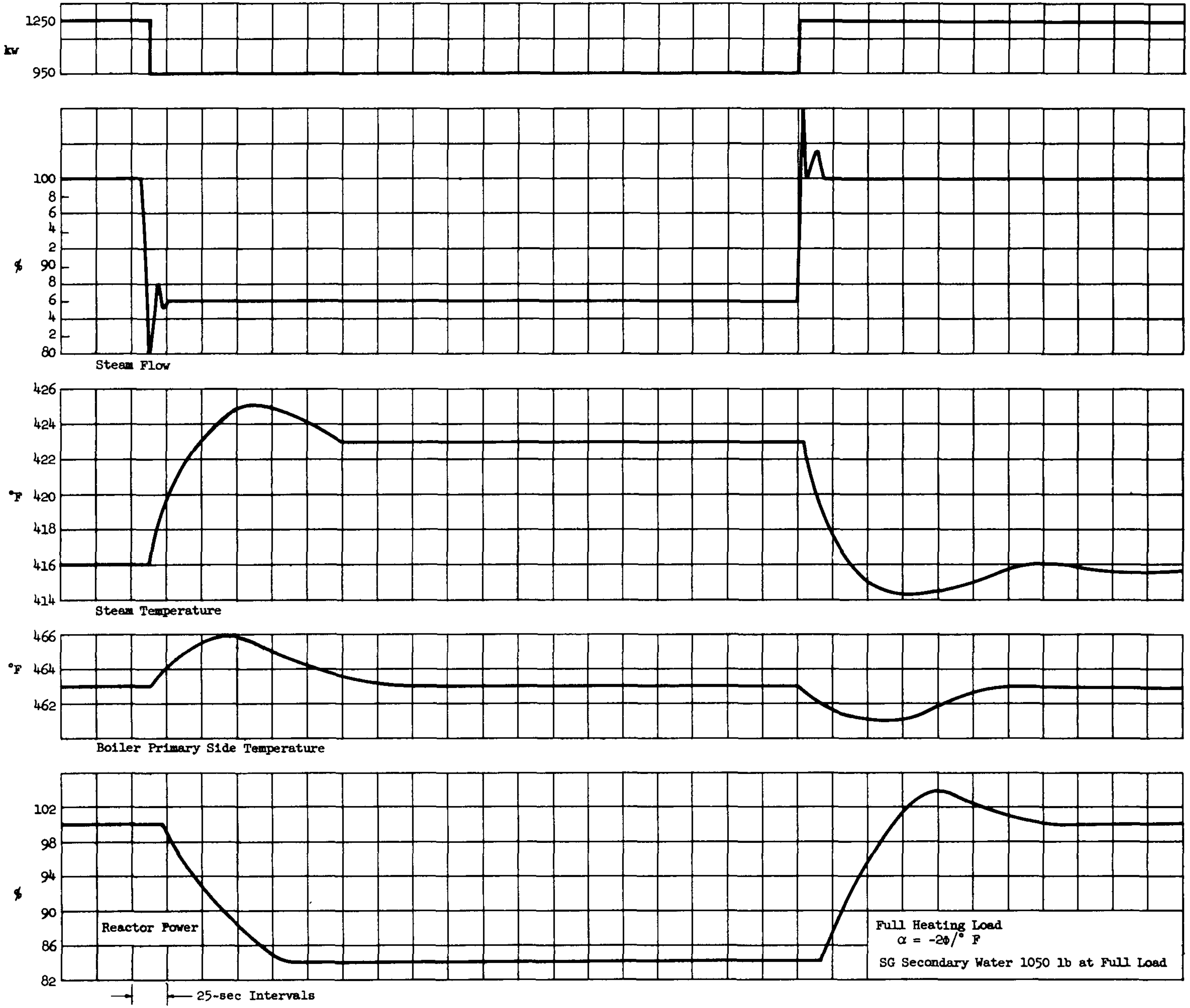


Fig. I-3. Load Change (1250 to 950 to 1250 kw)

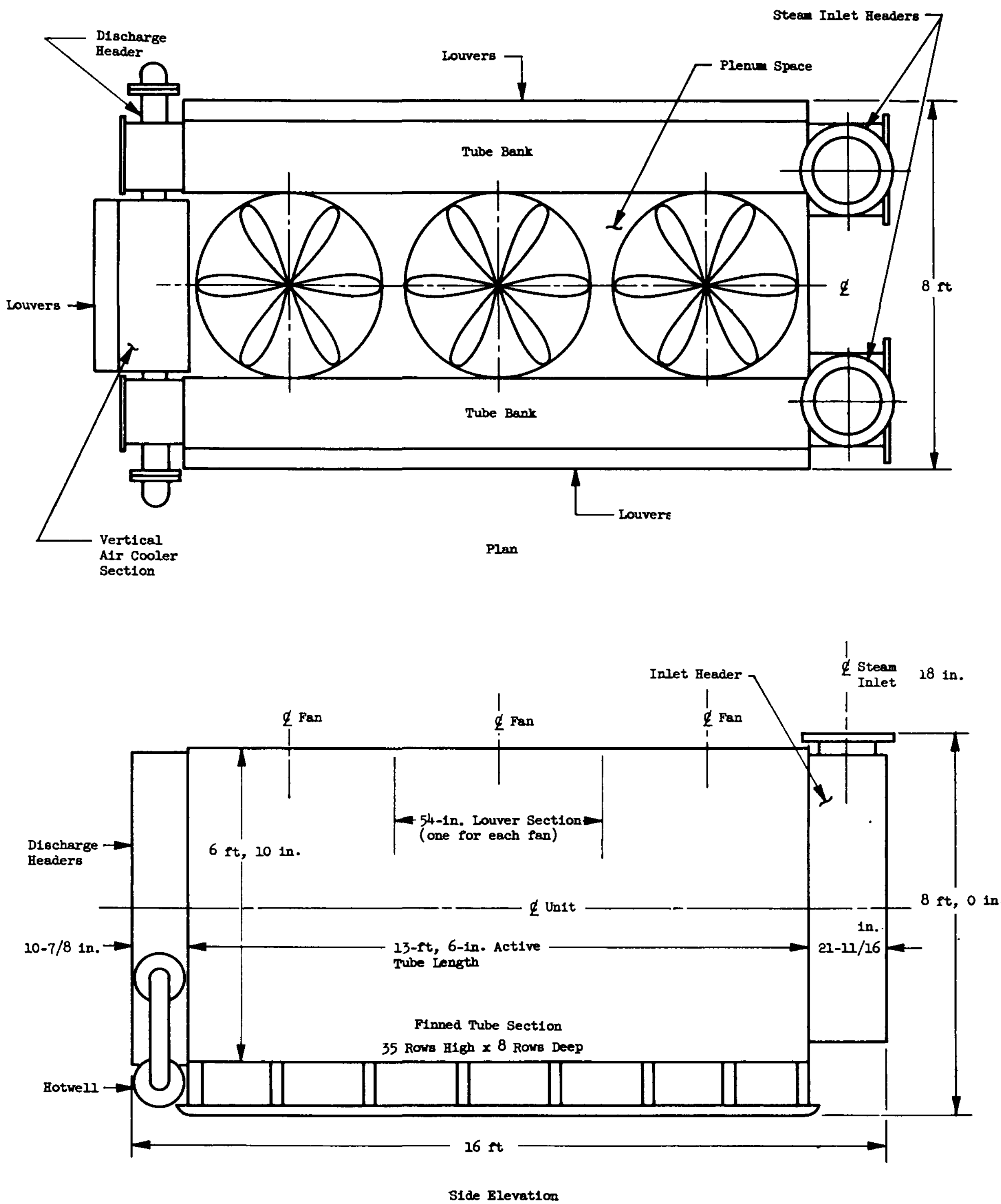


Fig. I-4. PM-1 Air Cooled Condenser Model

Operation differs somewhat from the preliminary design condenser. Steam enters two headers, one for each tube bank, at the forward end of the unit, then flows inside the tubes where it is condensed; the condensate then drains from the slightly sloped tubes into the discharge headers at the after end of the unit and into the hotwell. Noncondensable gases are drawn from the discharge headers and passed through the vertical finned-tube air cooler section where the remaining steam is condensed; the condensate flows to the hotwell and the noncondensables are drawn off by the steam jet ejector. The model will, therefore, operate in the same manner as an oversized half section of one full-size condenser.

The condenser model is to be tested in the main climatic chamber at Eglin AFB from 4 January through 11 March 1960 at the temperatures shown on the following test schedule:

1960 Test Schedule--Main Chamber

Test Cycle	
4 Jan through 15 Jan	Installation period and checkout
16 Jan through 17 Jan	Soaking at +70° F
18 Jan through 22 Jan	Testing at +70° F
23 Jan through 24 Jan	Soaking at 0° F
25 Jan through 29 Jan	Testing at 0° F
30 Jan through 31 Jan	Soaking at -25° F
1 Feb through 5 Feb	Testing at -25° F
6 Feb through 7 Feb	Soaking at -45° F
8 Feb through 19 Feb	Testing at -45° F
20 Feb through 21 Feb	Soaking at -65° F
22 Feb through 4 Mar	Testing at -65° F
5 Mar through 6 Mar	Soaking at +70° F
7 Mar through 11 Mar	Testing at +70° F

The preliminary outline for the condenser test at Eglin AFB has been prepared and is presented below. This test is divided into two parts: one part at 0° F and above and the other part at below 0° F. A drawing of the condenser test control system is shown in Fig. I-5. Table I-1 is a key to the control panel.

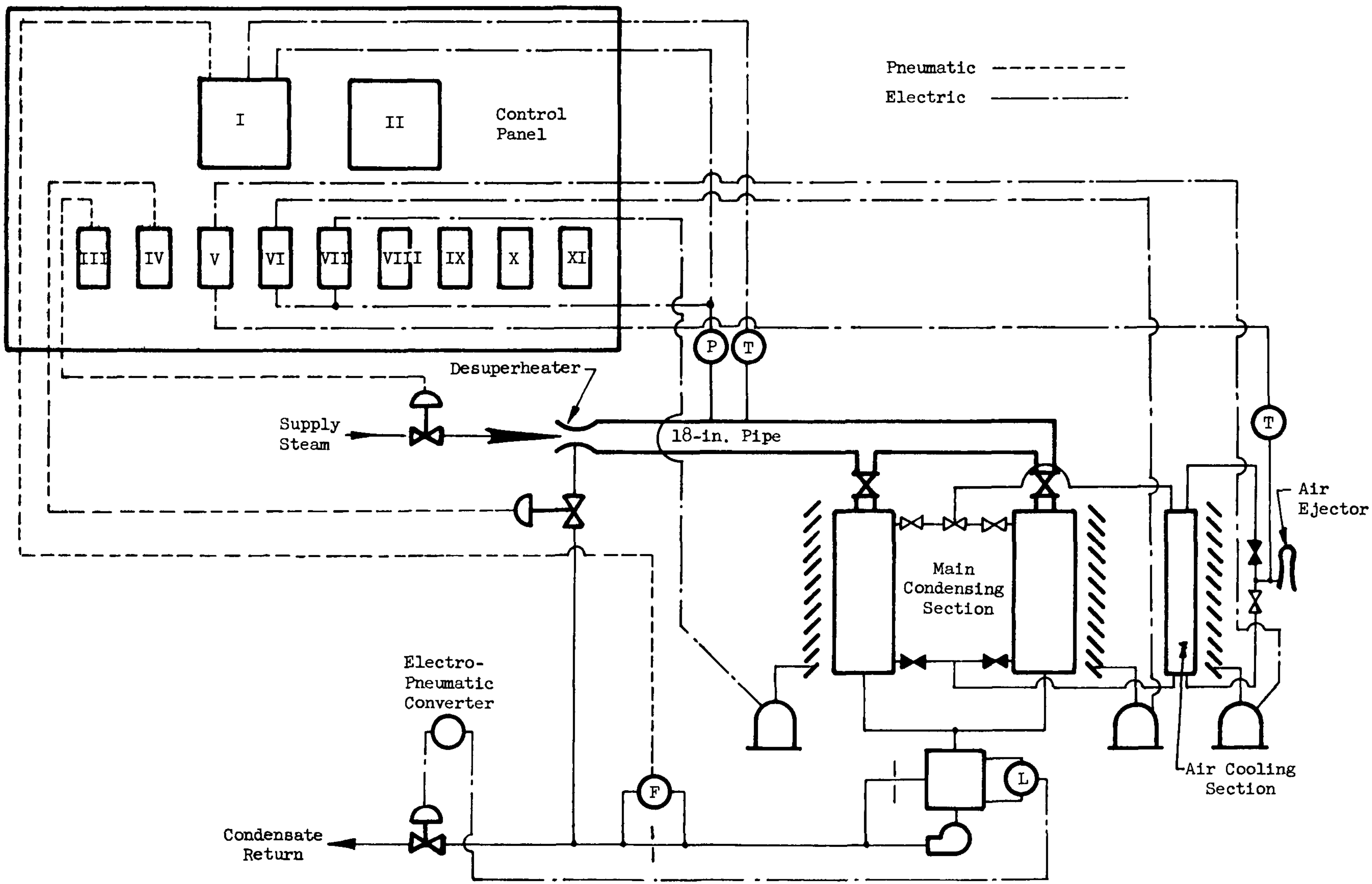


Fig. I-5. PM-1 Condenser Test Control System

TABLE I-1

Control Panel Key

I Circular chart recorder.

- (1) Condensate flow.
- (2) Condenser pressure.
  - (1) Controls main damper position.
- (3) Inlet steam temperature.
  - (1) Controls desuperheating water valve.

II 16-point temperature recorder.
A number of thermocouples will be located at points of interest.

III Manual loading for main steam valve.

IV Manual-auto selector station for desuperheater water valve with temperature set-point control.

V Manual-auto selector station for air-cooling section damper, controlled by air off-take temperature.

VI and VII Manual-auto selector stations for main damper positioners, controlled by inlet steam pressure.

VIII Condensate pump start switch.

IX and X Fan start switches and power indicators.

The preliminary outlines of the above- and below-zero °F condenser tests were prepared. See Tables I-2 and I-3.

TABLE I-2

Preliminary Outline for PM-1 Condenser Test at 0° F and Above

1. Determine air handling capacity of fans and their power requirements. Observe the distribution of the airflow. For the system, plot cfm versus kw with % louver opening as a parameter.

2. With maximum air flow, test over a range of steam flows and determine an overall heat transfer coefficient, "U", versus steam flow.

3. With a constant steam flow (less than maximum capacity), test over a range of air flows by adjusting dampers determining overall "U" versus air flow.

4. Determine the condensing capability of the unit with fans stopped and dampers open, 1/2 open, and closed.

5. Measure condensing pressure as a function of steam flow and back pressure.

6. Observe distribution of steam to the various rows by observing the rate of rise of condensate in the compartmented outlet chamber.

7. Measure condensate subecooling as a function of load, back pressure, ambient temperature., etc.

8. Determine the best flow direction in the air off-take section and its effect on condensate sub-cooling, overall "U", etc.

9. Determine the effect of increased tube slope on tube drainage and on removal of noncondensable gases.

10. Run transient tests with constant air flow and note rate of system response.

11. Run transients with automatic control of dampers to determine their ability to hold a constant back pressure.

12. Vary air leakage to determine the effect on overall heat transfer coefficient.

13. Vary air distribution along tube length to determine the effect on overall "U".

14. Determine the amount of dissolved O_2 in condensate.

TABLE I-3

Preliminary Outline for PM-1 Condenser Test at Below 0° F

1. Conduct startup and shutdown tests at lowest available ambient temperature.

2. Test at 15% of maximum design flow at lowest attainable ambient temperature.

3. Determine the ability of the controls to function at low ambient temperature.

4. Ice up dampers and use reverse air flow to free them if necessary.

Performance characteristics can then be extrapolated from the previously discussed (Table I-2) tests to low-temperature conditions.

4. Turbine Generator Design

Development efforts on turbine generator design, leading to the preliminary design discussed in Subtask 3.2, were chiefly concerned with selection of the oversize generator to assure power quality and selection of the satellite-faced steam admission valves to reduce moisture erosion.

Present turbine generator design indicates that the complete unit, less auxiliaries such as the oil coolers, etc., can be shipped in a package of less than 30,000 pounds.

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

Project Engineer--Subtasks 2.1, 2.2, 2.3, 2.4: J. O'Brien
Project Engineer--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. Rosenthal, R. Magladry, C. Cyl-Champlin, E. 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. Experimental support for the preliminary design is also included in that experimental data on specific items, such as lumped burnable poison worth, the worth of various types and geometrical configurations of rods, and flux distributions are required to verify analytical techniques and design approaches.

During this quarter, activities on the flexible zero-power test were:

(1) The conduct of an experiment to determine the effect of critical experiment (CE) control rods in small cores.

(2) The preparation of a statement of work for the flexible zero-power test (FZPT) and the preliminary outline of the experimental program.

(3) Completion of the analysis of the parametric zero-power experiments (PPM-1) reported in the last Quarterly Progress Report.

(4) Continuation of effort on the PM-1 FZPT and power plant excursion analyses.

During the next quarter, the following is expected to be accomplished:

(1) The design of flexible zero-power test components will be completed.

(2) The procurement and fabrication of FZPT components will be initiated.

(3) A detailed experimental program will be completed for final review in the FZPT Hazards Summary Report.

(4) The rough draft of the FZPT Hazards Summary Report will be completed and prepared for printing.

(5) Analysis of all FZPT experiments requiring pre-experiment evaluation will be initiated.

(6) The excursion analysis will be completed.

1. CE Control Rod Perturbation Evaluation

The question arose as to whether critical experiment (CE) control rods introduce perturbations that either invalidate experimental data or make it difficult to interpret. These CE control rods had been used in the zero-power test program performed under AEC Contract AT(30-3)-277 and reported in MND-MPR-1646. They were also used in the parametric zero-power test program (PPM-1) reported in the 1st PM-1 Quarterly Report. To resolve the question, an experiment was performed to study the magnitude and extent of the perturbation. Within the limits of experimental error, this demonstrated that the influence of the control rod in the core studied was no longer noticeable at distances greater than 5 inches, or approximately 60°, from the inserted control rod.

Figure II-1 illustrates the experimental critical core (consisting of 651 fuel elements of 0.375-inch OD) and the locations in which the measurements were made.

The technique employed for this series of measurements was to evaluate clusters of 5 fuel tubes in various circumferential positions in a representative section of the core. The resultant data are summarized in Table II-1.

TABLE II-1

Circumferential Fuel Worth

Circumferential Distance from Center of CE Rod (degrees)	Circumferential Distance from Center of CE Rod (inches)	Reactivity Worth* ($\% \Delta k/k$)
- 25	-2.23	0.004
- 10	-0.89	0.036

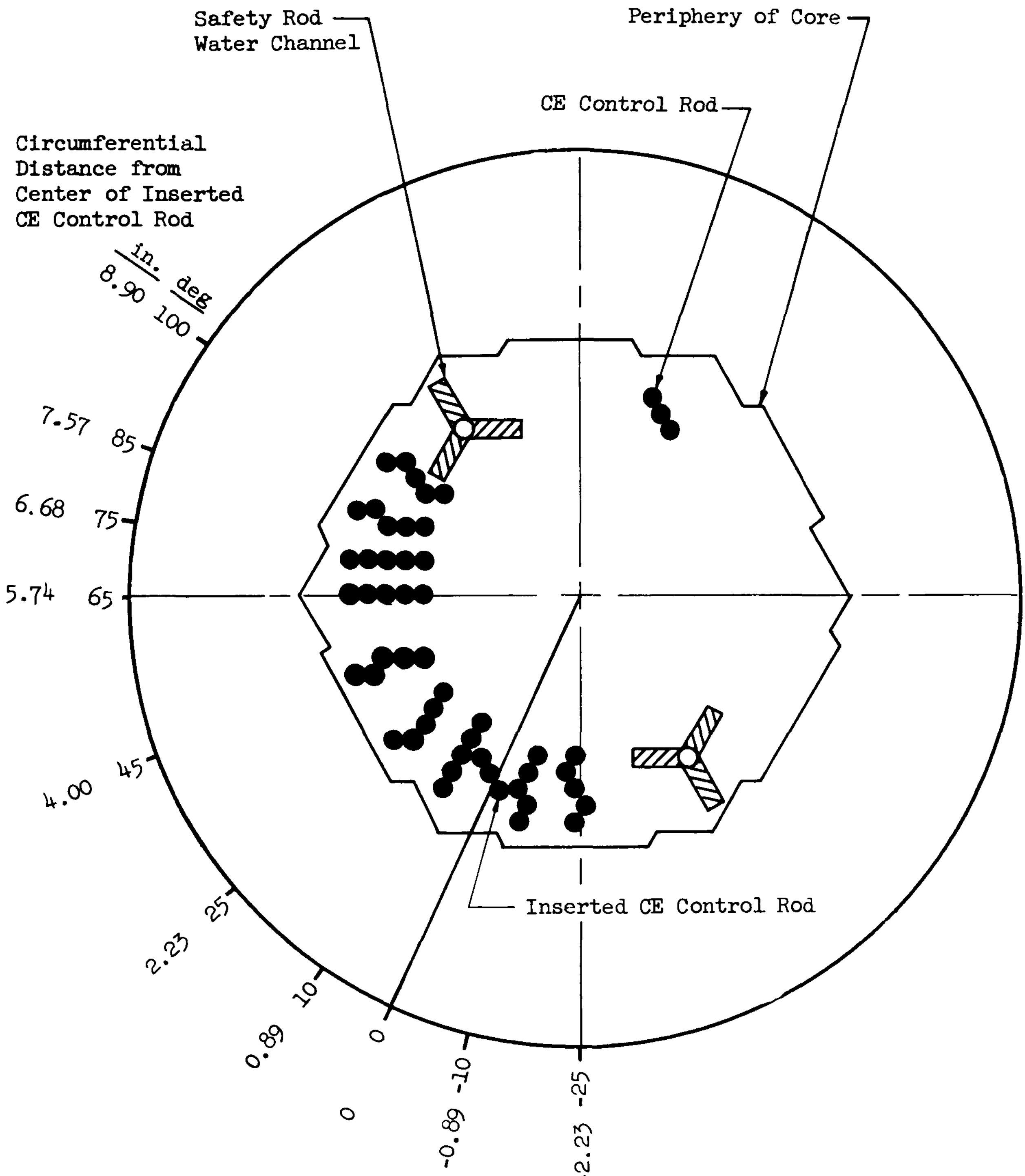


Fig. II-1. Experimental Critical Core (PPM-1 Core No. 6)

TABLE II-1 (continued)

Circumferential Distance from Center of CE Rod (degrees)	(inches)	Reactivity Worth* (% k/k)
10	0.89	0.056
25	2.23	0.021
45	4.00	0.024
65	5.79	0.013
75	6.68	0.002
85	7.57	0.023
100	8.90	0.050

*Worth of five tube clusters as shown in Fig. II-1.

Figure II-2 shows how the measured reactivity worth of each fuel tube cluster varies as a function of distance from the CE control rod in the chosen experimental section of the core. In the range from 25 to 85°, a mean worth of about 0.017% Δ k/k was anticipated. The worths of the clusters of fuel tubes from 25 to 100°, inclusive, are generally of the relative magnitudes that would be expected. The worth of the cluster near the safety rod water channel was greater than the others due to the peaking of the flux in the channel. However, the measurements conducted near the inserted CE control rod (+10° and -10°) exhibited an unexplained increase rather than the anticipated decrease in magnitude. The +10° measurement was repeated in order to confirm the previous data. A similar value was obtained (within the limits of error of the first measurement).

The resultant data have been interpreted as follows:

(1) The present type of CE control system can be used for future PM-1 zero-power test experiment measurements.

(2) The perturbations caused by the presence of the CE control rod are a localized effect (if the CE rod is small).

(3) Beyond a distance of approximately 5 inches, the effect of the CE control rod is negligible.

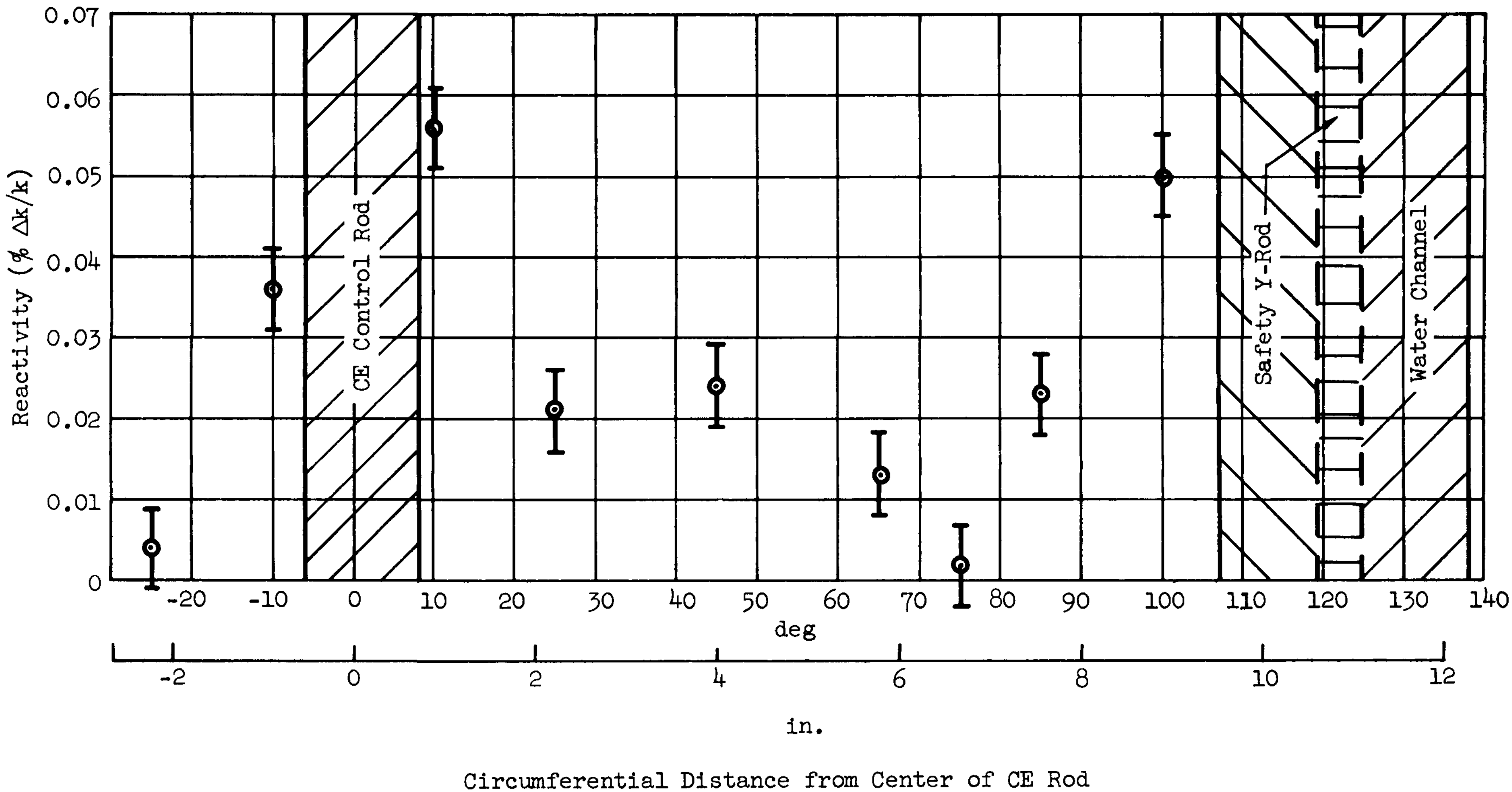


Fig. II-2. Reactivity Worth of Five Fuel-Tube Clusters as a Function of Distance from Inserted CE Control Rod

(4) The design of a new CE control system should not leave water voids in the experimental core; the control rods should either be provided with followers or should be inserted within the fuel tubes.

(5) The safety rod water channels create perturbations which appear to be as large in magnitude as those caused by the CE control rod.

2. Flexible Zero-Power Test--Experiment Outline

Preliminary planning of the FZPT experimental program was completed. Experiments and analyses to provide data in support of the final design of PM-1 and its evaluation have been outlined as follows:

Reactivity studies.-

(1) Determination of critical core configuration with no lumped burnable poison.

(2) Determination of the critical six-rod bank position with no lumped burnable poison and a full-sized core.

(3) Determination of the critical six-rod bank position with lumped burnable poison and a full-sized core.

(4) Determination of total core reactivity with no lumped burnable poison.

(5) Determination of total core reactivity with lumped burnable poison.

(6) Determination of radial reactivity worth of each core material.

(7) Determination of temperature coefficient of reactivity for (1), (2), (3), (4) and (5) through ~ 150° F.

(8) Determination of void reactivity effects:

*(1) Radial and axial void worth for several void fractions.

*(2) Void coefficient of reactivity for uniformly distributed voids for several void fractions (analytical).

---

*Either (1) or (2) will be performed.

(3) Void coefficient of reactivity for predicted local-boiling void distribution, if possible.

Lumped burnable poison studies.- Determinations of:

(1) Radial reactivity worth of lumped poisons in a full-sized core.

(2) Lumped poison self-shielding factor as a function of reactor operating time, i.e., poison concentration in the lumps.

(3) Effect of poison particle size on behavior of poison lumps.

(4) Reactivity worth of alternate poison distribution.

(5) Self-shielding as a function of lumped poison burnup (analytical).

Control rod studies.- Incremental and total rod worth of:

(1) Six rods

(2) Five rods

(3) Four rods, with two rods stuck in their operating positions

(4) One rod

(5) Different rod patterns, if required.

Flux and/or power measurements.- Determinations of:

(1) Radial and axial distributions for major core configurations

(2) Fine flux distribution for:

  (1) Lumped poison locations

  (2) Control rod channels

  (3) Fuel element cell

(3) Percent thermal fissions.

Thermal shield and pressure vessel flux measurements.-

(1) Fast neutron flux above 1 mev

(2) Gamma flux.

Miscellaneous studies. -

(1) Startup sensor locations

(2) Shipping configuration safety study

(3) Epicadmium worth of alternate control materials.

3. Parametric Zero-Power Test Analysis

During this quarter, analysis of the parametric zero-power tests (PPM-1) was completed. All experimental data and some of the analytical results were reported in the previous Quarterly Progress Report.* Analytical results for the control rod studies, lumped poison studies and fast flux studies are given below. Abstracts of the IBM-704 codes used in the analysis will be submitted later in a technical memorandum.

a. Control rod studies

Effect of slab rod width on reactivity worth (PPM-1, Core II). One assumption implicit in the determination of the average flux spectrum in the absorber (to be used in calculating few group constants for the control material) is that the absorber may be approximated by a semi-infinite slab. In finite slabs, end effects may cause the actual flux spectrum to differ from that calculated on the basis of a semi-infinite slab. In order to determine whether this effect is present in slabs in the range of interest, the relative worths of three slabs of different widths were evaluated, both experimentally and analytically, in the center of a critical core. Using a critical core minimized the requirement for control rods within the core which, in turn, minimized flux perturbations in the core and simplified the analysis.

A comparison of relative worths of the slab rods for varying width is given in Table II-2. The boron density is the density of natural boron, as B_4C, in the slab. Aluminum was used as the diluent.

---

*MND-M-1812, First Quarterly Progress Report, PM-1, 6 July 1959.

TABLE II-2

Relative Worths of Slab Rods for Varying Width

Slab Width (in.)	Slab Thickness (in.)	Boron Density (gm/cc)	Relative Worth Experimental	Relative Worth Analytical
2.5	0.205	0.747	1.0	1.0
3.0	0.204	0.747	1.17	1.14
3.5	0.214	0.747	1.29	1.31

The above results show good agreement between the analytical and experimental results. This indicates that the semi-infinite slab approximation is satisfactory.

Reactivity versus slab rod thickness (PPM-1, Core II).- The effect of varying the rod thickness was investigated in order to determine its increase in worth for increased thicknesses in the range of interest.

Four slabs, 3 inches in width and 0.166 inch, 0.204 inch, 0.252 inch, and 0.287 inch in thickness, were evaluated. The rod compositions and a comparison of the relative worths of the rods are given in Table II-3.

TABLE II-3

Relative Worths of Slab Rods of Varying Thickness

Slab Width (in.)	Slab Thickness (in.)	Boron Density (gm/cc)	Relative Worth Experimental	Relative Worth Analytical
3.0	0.166	0.747	1.00	1.00
3.0	0.204	0.747	1.05	1.06
3.0	0.252	0.781	1.09	1.13
3.0	0.287	0.781	1.13	1.17

Increasing the absorber thickness obviously increases the rod worth. However, since all four rods were black to thermal neutrons, i. e., $e^{-\Sigma_a x} < 0.002$, the increase in worth is due to the increase in epithermal absorption. The increase in worth is ≈ 1% per 10-mil increase in thickness.

The deviation between the experimental and analytical worths is small. Since the average flux within the absorber is dependent upon the thickness of the material, and since this, in turn, is reflected in the few-group flux-weighted cross sections, the probable source of error is in the calculation of the average epithermal fluxes in the absorber material.

Lumped burnable poison studies (PPM-1, Core V).- The lumped burnable poison study was undertaken in order to provide data useful in the evaluation of the analytical technique for calculating the worth and usefulness of lumped burnable poisons in the PM-1. Two of the important nuclear properties of the lumped poisons required for the analysis are the reactivity worth and the time-dependent self-shielding factor, g(t). Since g(t) is essentially equal to g(N), i.e., a function of poison concentration, the time-dependent self-shielding factor can be obtained by evaluating a poison lump with different concentrations of the poison. A knowledge of the reactivity worth of the lump as a function of poison concentration allows calculation of g(t) using perturbation theory.

Specifically, the objective of this study was to obtain both experimental and analytical radial reactivity worth for three different poison concentrations in a rod configuration and one poison concentration in a tubular configuration.

In order to assure the proper choice of poison concentrations, the self-shielding factor was initially calculated as a function of poison concentration. This was done using diffusion theory which, although not precisely correct, gives the correct relative shape of the curve. From these results, poison concentrations of 0.05, 0.1, and 0.4 gm/cc were selected.

The analytical radial worth of these rods was calculated using Cell Removal Theory (Program Delta). The weighting fluxes used to obtain the flux-weighted two-group constants were obtained from multigroup slowing down theory with each poison lump in a central position in the core. The actual flux weighting and two-group calculation was done using COFWAC.* The use of these constants for the central poison

---

* COFWAC - calculation of flux-weighted absorber constants, an IBM-704 code.

lump case yielded the perturbed normal fluxes required for Program Delta. For poison lumps in a noncentral position, the perturbed normal fluxes were obtained by depressing the unperturbed flux near the actual lump location to the same level as that obtained when the lump is at the core center. The necessary flux depression was determined by using Diffusion Theory (Program F). The unperturbed adjoint fluxes were obtained from an adjoint calculation for the core without rods.

The relative worth as a function of radial location was assumed identical for each lump, since, for the same rod, the worth at each point depends only upon the unperturbed flux at that point. Consequently, the radial distribution of worth was obtained analytically for only the rod configuration which contained 0.4 gm of boron per cm^3; the remaining rods and the tube were evaluated in the central position only. The worth of any of the compositions at any radial location was determined from the relative worth curve for the 0.4-inch rod.

The central location reactivity worths were also determined from a one-dimensional diffusion theory calculation.

The resulting radial rod lump worths and the corresponding experimental values are presented in Table II-4.

TABLE II-4

Radial Reactivity Worths of Lumped Poisons (Negative)

Radius (cm)		0	6.4	10.67	17.07	25.61
Boron Concentration in Lump (gm/cc)						
	A	0.427	0.392	0.304	0.191	0.059
0.4 Rod	B	0.315	0.301	0.258	0.130	0.039
	C	0.349				
	A	0.297	0.271	0.212	0.316	0.041
0.1 Rod	B	0.209	0.200	0.171	0.086	0.026
	C	0.215				

TABLE II-4 (continued)

Radius (cm)		0	6.4	10.67	17.07	25.61
Boron Concentration in Lump (gm/cc)						
	A	0.217	0.198	0.154	0.111	0.035
0.05 Rod	B	0.186	0.178	0.152	0.077	0.023
	C	0.162				
	A	0.174	0.169	0.133	0.087	0.031
0.1 Tube	B	0.160	0.153	0.131	0.066	0.020
	C	0.128				

A--Experimental value

B--Analytical value--Program Delta (Cell Removal Theory)

C--Analytical value--Program F (Diffusion Theory)

Using the methods described above, the calculated reactivity worths are 15 to 30% less than the experimental worths. Within the limitations of these studies, i.e., relatively small reactivity effect of one lump, particle self-shielding, boundary condition specifications, and diffusion theory, the comparison between experimental and analytical results is considered satisfactory. For the final design, however, more precise studies are in order. These are presently in progress and additional experimental studies have been planned.

Fast flux studies (PPM-1, Core V).- The use of two-group diffusion theory has been found to be adequate for obtaining gross flux distributions in the core. Comparison of analytical thermal flux distributions with experimental results* showed good agreement in both relative distributions and absolute values.

In order to determine the accuracy of the analytical method of calculating the fast (2.5 mev) flux, the results cited were supplemented by data obtained from this experiment.

---

*Rosenthal, H. B., and Sticchitano, E. A., "Nuclear Studies on the MPR Zero-Power Test Core," MND-MPR-1646, December 1958.

Radial fast flux distribution at the axial midplane through the core and radial reflector region (which contained two stainless steel thermal shields) were obtained both experimentally and analytically.

The analytical fast flux distribution was calculated using both the one-dimensional three-group diffusion code F-3 and the two-dimensional three-group diffusion code PDQ (in R-Z geometry).

To obtain the radial flux distribution at the axial midplane, the average radial distribution was multiplied by the ratio of the flux at the axial center to the average axial flux. This ratio, 1.32, was calculated from the output of a one-dimensional axial flux calculation.

The relative radial fast flux distributions showing both experimental and analytical results are given in Fig. II-3. The flux distributions are normalized to the PDQ flux at the center of the core.

The absolute fluxes at various locations are given in Table II-5. Both analytical and experimental fast fluxes are given. Analytical values of epithermal and thermal fluxes are also included: $_1$ represents the fast values, $_2$, the epithermal and $_3$, the thermal.

The relative flux distributions shown in Fig. II-3 show very good agreement between analytical and experimental results.

Agreement between the analytical absolute values of fluxes for all 3 groups and at all of the locations, calculated from PDQ and F-3 results, were very good. Except for the thermal flux at the inner surface of the pressure vessel, the deviation of the F-3 fluxes from the PDQ fluxes is less than 6%. The F-3 thermal flux at the pressure vessel is 14.7% higher than the PDQ value.

The experimental absolute fast fluxes obtained using a Po-Be calibration were lower than the analytical values by a factor of approximately 20. The experimental values were recalculated to eliminate the need for a Po-Be calibration by taking into account sample self-absorption and counter efficiency. The excitation function was weighed against the flux energy distribution at each of the locations to determine the flux-weighed reaction cross section (the neutron energy distribution at different locations was obtained from a multigroup, multiregion diffusion calculation--Program G-2). These studies are described in a memo from J. F. O'Brien.* The experimental flux values thus obtained are given in Column 5 of Table II-5. The ratios of analytical to experimental

*"PPM-1 Core V, Fast Neutron Flux Measurements," PM-1 Technical Memorandum, MND-M-1905, (JOB-4), September 1959.

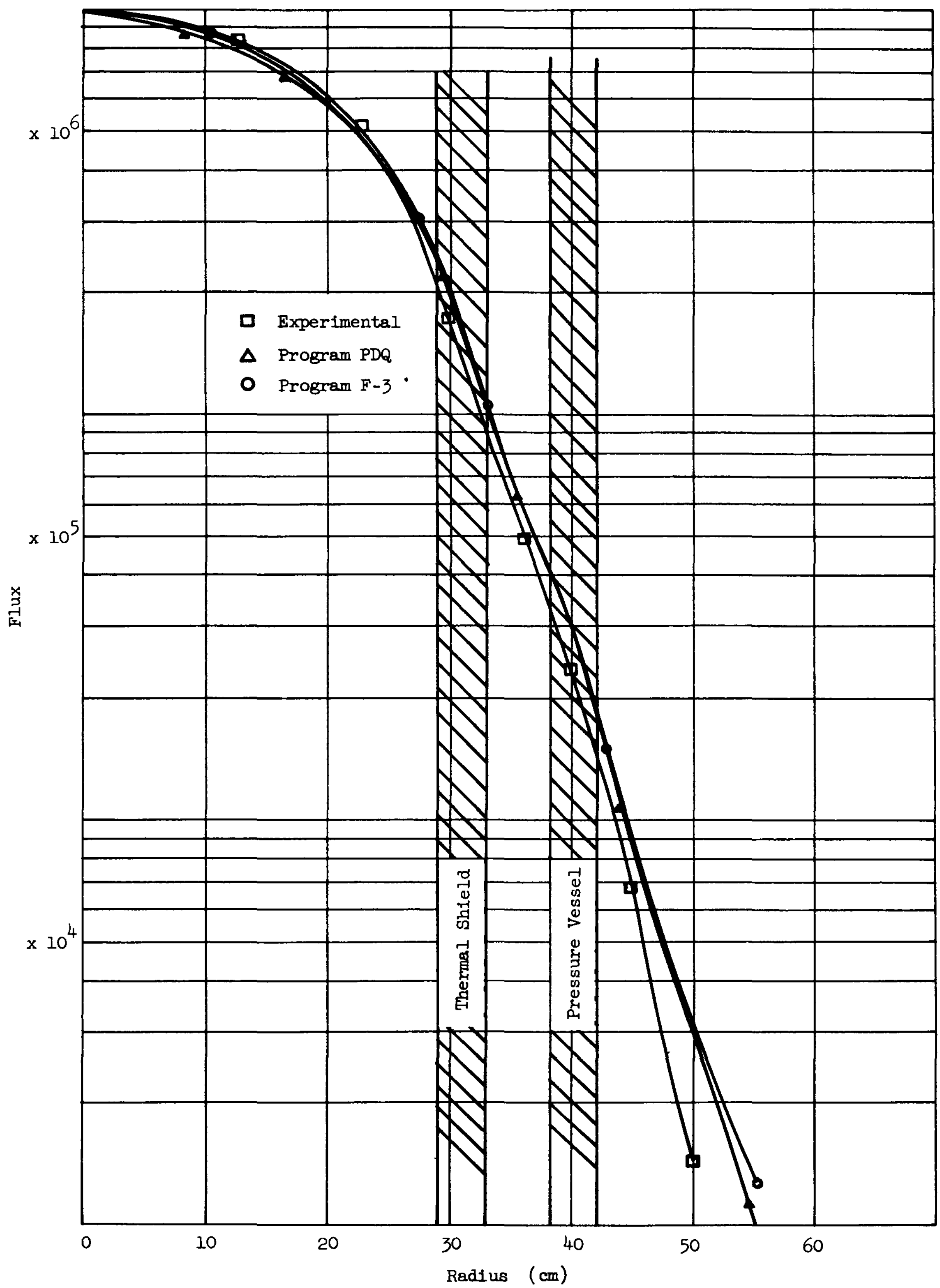


Fig. II-3. Radial Fast Flux Distribution in PPM-1, Core V

TABLE II-5

Absolute Fast Neutron Fluxes (neutrons/cm^2-sec/watt of power)

Location		Analytical PDQ	Analytical F-3	Deviation (%)	Experimental	$\phi_{1\,PDQ}/\phi_{Experimental}$
Center of core	ϕ_1	9.93×10^6	10.12×10^6	1.9	6.4×10^6 (4.5×10^5)*	1.56
	ϕ_2	1.68×10^7	1.68×10^7	0		
	ϕ_3	1.49×10^6	1.42×10^6	4.7		
Inner surface of thermal shield	ϕ_1	2.50×10^6	2.64×10^6	5.4	1.76×10^6 (9.6×10^4)	1.4
	ϕ_2	4.58×10^6	4.70×10^6	2.6		
	ϕ_3	8.85×10^5	9.40×10^5	6.0		
Outer surface of thermal shield	ϕ_1	1.02×10^6	1.107×10^6	5.2	6.2×10^5 (4.2×10^4)	1.65
	ϕ_2	3.22×10^6	3.31×10^6	3.0		
	ϕ_3	9.73×10^5	9.90×10^5	1.8		
Inner surface of pressure vessel	ϕ_1	4.45×10^5	4.60×10^5	3.3	3.2×10^4 (1.5×10^4)	13.4
	ϕ_2	1.31×10^6	1.34×10^6	2.4		
	ϕ_3	5.26×10^5	6.03×10^5	14.7		

*Values in parentheses were obtained using Po-Be neutrons for calibration of the detector.

fast flux (Column 6) are quite good. The analytical values were less than a factor of 1.7 higher (except for the value at the inner surface of the pressure vessel which was 13.4 times higher). The probable sources of these discrepancies are the neglect of the inelastic scattering (which was not included in the analytical calculation) and poor statistical accuracy in the counting of fast neutron detector foils placed at the inner surface of the pressure vessel.

4. Excursion Analysis

Kinetic equations.- The checkout of the IBM-704 machine program to solve the kinetic equations for the PM-1 flexible zero-power experiment excursion analysis has been completed. The analytical solution of the standard kinetic equations (1) and (2) used in this analysis was a direct extension of the methods presented in UN58/629**.

The kinetic equations*** may be expressed as follows:

$$\frac{dn}{dT} = \frac{k-1}{\ell}\, n - \frac{k\beta n}{\ell} + \sum_{i=1}^{6} \lambda_i C_i + S \qquad (1)$$

$$\frac{dC_i}{dT} = -\lambda_i C_i + \frac{k \beta_i n}{\ell} \qquad (2)$$

The analytical solution of these equations proceeds as follows.

Rearrange equations (1) and (2) and consider a time interval $t \quad t + h$ where t is some T.

Then,

$$\frac{dn}{d(t+\xi)} = \frac{dn}{d\xi} = \frac{1}{\ell}\left[(1-\beta)\, k(t+\xi) - 1\right] n(t+\xi)$$

$$+ \sum_{i=1}^{6} \lambda_i C_i(t+\xi) + S(t+\xi) \qquad (3)$$

**Cohen, E. Richard, "Some Topics in Reactor Kinetics," UN58/629.

***"Reactor Handbook, Physics," p 531.

where: $0 \leq \xi \leq h$

$$\frac{dCi}{d\xi} = -\lambda i\ Ci + \frac{\beta_i k(t+\xi)n(t+\xi)}{\ell} \tag{4}$$

and,

$$\frac{dn}{d\xi} = \frac{1}{\ell}\left[(1-\beta)k(t)-1\right]n(t+\xi)$$

$$+\frac{1-\beta}{\ell}\left[k(t+\xi)-k(t)\right]n(t+\xi)+\sum_{i=1}^{6}\lambda i\ Ci + S(t+\xi) \tag{5}$$

where:

β = $\sum_{i=1}^{6} \beta_i$

n = neutron density in the reactor

Ci = density of i'th-type delayed neutrons

S = source density, neutrons per unit volume

β_i = fraction of fission neutrons emitted by the i'th delayed neutron emitter

λ_i^{-1} = mean life of the i'th-type delayed neutron

k = multiplication constant

ℓ = lifetime

T = time variable

t = some specific time

h = time interval

ξ = time variable between $t \rightarrow t + h$.

Define:

$$R(n, \sum_{i=1}^{6} i\ Ci, t+\xi) \equiv (1-\beta)\left[k(t+\xi) - k(t) \quad n(t+\xi)\right]$$

$$+ \sum_{i=1}^{6} \quad i\ Ci + S \tag{6}$$

$$\alpha = \frac{1}{\ell}\left[(1-\beta)\,k(t) - 1\right] \tag{7}$$

$$R_i'(n, t+\xi) = \frac{\beta_i\, k(t+\xi)}{\ell}\ n(t+\xi) \qquad i = 1, 6 \tag{8}$$

$$\alpha_i' = -\lambda i \qquad i = 1, 6. \tag{9}$$

Thus, equations (5) and (4) may be written:

$$\frac{dn}{d\xi} = \alpha n + R(n, \sum_{i=1}^{6} \lambda i\ Ci, t+\xi) \tag{10}$$

$$\frac{d\,Ci}{d\,\xi} = \alpha_i'\ Ci + R_i'(n, t+\xi) \quad i = 1, 6 \tag{11}$$

The scheme used to solve equations (10) and (11) is related to the Runga-Kutta methods and can be considered a generalization of it.

Define:

$$\bar{C}m(x) \triangleq \int_0^1 \mu^{m-1}\ e^{x(1-\mu)}\ d\mu \tag{12}$$

Therefore,

$$\bar{C}_1(x) = \frac{1}{x}\left[e^x - 1\right] \tag{13}$$

$$\bar{C}_{m+1}(x) = \frac{1}{x}\left[m\ \bar{C}_m(x) - 1\right] \tag{14}$$

$$n_o = n(t) \tag{15}$$

$$Ci_o = Ci(t) \tag{16}$$

$$R_o = R(n_o, \sum_{i=1}^{6} \lambda i \; Ci_o, t) \tag{17}$$

$$Ri'_o = Ri'(n_o, t) \tag{18}$$

$$\delta_1 = \frac{h}{2} \bar{C}_1 \left(\frac{\alpha h}{2}\right) \left[R_o + \alpha n_o\right] \tag{19}$$

$$\delta i'_1 = \frac{h}{2} \bar{C}_1 \left(\frac{\alpha'_i h}{2}\right) \left[R'_{i\,o} + \alpha'_i \; Ci_o\right] \tag{20}$$

$$n_1 = n_o + \delta_1 \tag{21}$$

$$Ci_1 = Ci_o + \delta i'_1 \tag{22}$$

$$R_1 = R(n_1, \sum_{i=1}^{6} \lambda i \; Ci_1, t + \frac{h}{2}) \tag{23}$$

$$Ri_1 = Ri'(n_1, t + \frac{h}{2}) \tag{24}$$

$$\delta_2 = \frac{h}{2} \bar{C}_2 \left(\frac{\alpha h}{2}\right) \left[R_1 - R_o\right] \tag{25}$$

$$\delta i'_2 = \frac{h}{2} \bar{C}_2 \left(\frac{\alpha i' h}{2}\right) \left[Ri'_1 - Ri'_o\right] \tag{26}$$

$$n_2 = n_1 + \delta_2 \tag{27}$$

$$Ci_2 = Ci_1 + \delta i'_2 \tag{28}$$

$$R_2 = R(n_2, \sum_{i=1}^{6} \lambda i \; Ci_2, t + \frac{h}{2}) \tag{29}$$

$$Ri'_2 = Ri'(n_2, t + \frac{h}{2}) \tag{30}$$

$$\delta_3 = h\bar{C}_1(\alpha h)\left[R_o + \alpha n_o\right] + 2h\bar{C}_2(\alpha h)\left[R_2 - R_o\right] \tag{31}$$

$$\delta i_3' = h\bar{C}i(\alpha_i' h)\left[R_o + \alpha_i' n_o\right] + 2h\bar{C}_2(\alpha_i' h) \left[Ri_2' - Ri_o'\right] \tag{32}$$

$$n_3 = n_o + \delta_3 \tag{33}$$

$$Ci_3 = Ci_o + \delta i_3' \tag{34}$$

$$R_3 = R(n_3, \sum_{i=1}^{6} \lambda i\, Ci_3, t+h) \tag{35}$$

$$Ri_3 = Ri'(n_3, t+h) \tag{36}$$

$$\delta_4 = h\left\{\bar{C}_3(\alpha h) - \bar{C}_2(\alpha h)\right\}\left[R_o - 2R_2 + R_3\right] \tag{37}$$

$$\delta i_4' = h\left\{\bar{C}_3(\alpha_i' h) - C_2(\alpha h)\right\}\left[Ri_o' - 2Ri_2' + Ri_3'\right] \tag{38}$$

$$n_4 = n_3 + \delta_4 \tag{39}$$

$$Ci_4 = Ci_3 + \delta i_4' \tag{40}$$

$$R_4 = R(n_4, \sum_{i=1}^{6} \lambda i\, Ci_4, t+h) \tag{41}$$

$$Ri_4' = Ri'(n_4, t+h) \tag{42}$$

$$\delta_5 = h\left\{2\bar{C}_3(\alpha h) - \bar{C}_2(\alpha h)\right\}\left[R_4 - R_3\right] \tag{43}$$

$$\delta i_5' = h\left\{2\bar{C}_3(\alpha' h) - \bar{C}_2(\alpha' h)\right\}\left[Ri_4' - Ri_3'\right] \tag{44}$$

$$n(t+h) = n_4 + \delta_5 \tag{45}$$

$$Ci(t+h) = Ci_4 + \delta_5 . \qquad (46)$$

The evaluations of equations (15) through (46) are straightforward; however, the derivation of them is not so obvious. The derivation of these equations proceeds as follows:

(1) Compute n_1 and Ci_1 as first approximations to $n(t + \frac{h}{2})$ and $Ci(t + \frac{h}{2})$ assuming the R's constant.

(2) Compute n_2 and Ci_2 the second approximations assuming that R and Ri' vary linearly between R_o and R_1 over the time interval $t \rightarrow t + \frac{h}{2}$.

(3) Compute n_3 and Ci_3 the first approximations of $n(t + h)$ and $Ci(t + h)$ assuming that R and Ri' vary linearly between R_o and R_2 over the time interval $t \rightarrow t + h$.

(4) Compute n_4 and Ci_4 as the second approximation assuming that R and Ri' vary as a quadratic between R_o, R_2, and R_3 over the time interval $t \rightarrow t + h$.

(5) Compute a final value of n and C_i at t + h assuming that R and Ri' vary as a quadratic between R_o, R_2, and R_4.

The actual integration of equations (10) and (11) in each of the above steps was greatly simplified by the use of LaPlace transformations. These equations were programmed for the IBM-704 utilizing Fortran II. The only real problem encountered was in the evaluation of the $\bar{C}(x)$'s for small x's. For this reason, they were replaced with series of the form:

$$\bar{C}_3(x) = \left\{1 + \frac{x}{4}\left[1 + \frac{x}{5}(1 + \frac{x}{6} \ldots .)\right]\right\} /3 \qquad (47)$$

$$\bar{C}_2(x) = (1 + x\, C_3)/2 \qquad (48)$$

$$\bar{C}_1(x) = (1 + x\, \bar{C}_2) \qquad (49)$$

or:

$$\bar{C}_2(x) = \left\{1 + \frac{x}{3} \quad 1 + \frac{x}{4}(1 + \frac{x}{5} \ldots .)\right\} /2 \qquad (50)$$

$$\bar{C}_1(x) = 1 + x\,\bar{C}_2. \tag{51}$$

The results of this code were checked against the ordinary difference technique used in the NPFO* excursion analysis and gave excellent agreement (less than 0.3%) for time intervals as great as 2.5×10^{-3} seconds. This agreement was based upon an NPFO sample case with a 2% step in reactivity and a 2×10^{-5}-second time interval. The results of the preliminary excursion analysis further indicated good agreement with SPERT data.

Preliminary analysis.- The preliminary excursion analysis for the PM-1 core has been completed. Twelve cases have been evaluated by an IBM-704 machine program; 3 for step input and 3 for ramp input of reactivity, each set for both zero-power and operating-power initial conditions. The extent of the maximum credible and possible incidents, when selected for hazard consideration, may be interpolated from the excursion data produced.

The model used assumes that all the thermal effects which shut down an excursion of relevant magnitude can be lumped into two effects:

(1) Fuel tube temperature rise

(2) Steam (void) formation within the moderator.

In addition, it is assumed that:

(1) Heat transfer both through fuel and moderator is conductive.

(2) The thermal diffusivities of fuel and moderator are constant.

(3) Steam formation begins when the fuel tube surface temperature reaches the moderator saturation temperature.

(4) The rate at which steam forms is proportional to the heat current entering the moderator.

The machine program consists of a transient thermal subroutine embodying these assumptions, coupled to the PM-1 kinetics routine outlined above. The kinetics routine provides heat release rates for the thermal subroutine and it, in turn, feeds back reactivity loss to the kinetics routine.

---

*NPFO is the Nuclear Power Field Office.

Because of the phenomenological nature of both kinetic and thermal models, experimental transient information was used for transient reactivity coefficients, i.e., shutdown parameters. SPERT-I data obtained in testing of the reject SM-1-type core were used. The SM-1 core is somewhat similar to PM-1 as is shown in Table II-6.

TABLE II-6

General Comparison of PM-1 and SPERT-I, SM-1 Cores

	PM-1	SPERT-I, SM-1
Moderator		
Fraction	84.0 vol %	81.7 vol %
Fuel		
Fraction	16 vol %	18.3 vol %
Matrix thickness	0.030 in.	0.020 in.
Cladding thickness	0.006 in.	0.005 in.
UO_2	25.0 wt %	17.94 wt %
B_4C	0.4 wt %	0.18 wt %
Stainless steel	74.6 wt %	81.88 wt %

The comparison is made on a unit cell basis.

A parametric study was made in which 6 cases were run, varying the temperature and void reactivity coefficients. The program, with the shutdown parameters shown in Table II-7, produced calculated values in agreement within ±20% of the SPERT-I, SM-1 experimental values obtained for peak power, energy at peak, and fuel surface temperature rise. These calculations covered the range of the SPERT-I, SM-1 tests (in which saturation temperature was achieved).

TABLE II-7

Shutdown Parameters for PM-1 Excursion Analysis

		k_{eff}/t	k_{eff}/V
From SPERT-I, SM-1 transient experiments		$-3.95 \times 10^{-5}/°C$	-0.37/core void fraction
Steady-state analytic PM-1 (20° to 275° C)		$-2.05 \times 10^{-4}/°C$	-0.37/core void fraction
	Effective neutron lifetime		14 ms
	Effective neutron delay fraction		0.0075
	Zero-power case		
	Initial power		10^{-6} mw
PM-1	Initial temperature		20° C
	Saturation temperature		100° C
	Operating-power case		
	Initial power		9.35 mw
	Initial temperature		275° C
	Saturation temperature		338.7° C

The shutdown parameters shown in Table II-7 were then utilized in the PM-1 excursion calculations. The results of these calculations are shown in Table II-8.

Heat transfer equations.- An IBM-704 machine program for solution of the heat transfer equations to be used in conjunction with the kinetic equations has been coded. Checkout of these equations is approximately 95% complete. A brief code linking the two sets of equations has been written and checkout has begun. A report on the final code and the heat transfer equations will be written and incorporated in an appropriate report.

TABLE II-8

Preliminary PM-1 Excursion Data

Steps

Reactivity Input (k_{ex})	Asymptotic Period (msec)	Peak Power (Mw)	Reactivity at Peak (k_{ex})	Temperature at Peak* (° C)	Energy at $k_{eff} < 1$** (Mw sec)	Time to $k_{eff} < 1$ (sec)	Temperature* $k_{eff} < 1$ (° C)
				Zero Power			
0.010	18.7	2,029	0.0073	131.7	36.3	0.129	185.4
0.015	1.84	15,730	0.0074	296.4	90.8	0.0473	796.3
0.020	1.115	42,400	0.0086	463.7	40.2	0.0295	1044.8
				Operating Power			
0.010	13.7	2,169	0.0071	343.9	34.7	0.0425	531.6
0.015	1.77	10,107	0.0121	436.9	88.1	0.0176	986.9
0.020	1.03	41,922	0.0066	574.0	134.6	0.01147	1295.8

Ramps:

Reactivity Input (k_{ex}/sec)	Maximum Period (msec)	Peak Power (Mw)	Maximum Reactivity (k_{ex})	Temperature at Peak (° C)	Energy at $k_{eff} < 1$ (Mw sec)	Time to $k_{eff} < 1$ (sec)	Temperature $k_{eff} < 1$ (° C)
				Zero Power			
0.0050	431	556	0.0086	68.1	26.7	1.845	135.9
0.0075	317	937	0.0090	72.3	29.9	1.276	165.7
0.0100	111	1,309	0.0093	83.4	32.9	0.945	190.7
				Operating Power			
0.0050	1030	30.9	0.0032	340.8	21.1	1.173	340.9
0.0075	598	45.1	0.0045	341.0	21.3	0.972	341.7
0.0100	385	63.6	0.0054	341.1	21.5	0.818	342.6

*Temperatures are at center of fuel tube wall.

**The total power burst energy may be considered to be 1.1 times the energy release at $k_{eff} < 1$.

B. SUBTASK 2.2--IRRADIATION TEST

J. B. Zorn, A. Carnesale

The objective of the irradiation program is to subject the PM-1 fuel element to burnup 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.

The principal objectives during this period were:

(1) To complete the irradiation program and test specimen parameters for PM-1 elements and submit them to the USAEC.

(2) To fabricate test elements.

(3) To investigate analytically the thermal characteristics of these elements during irradiation in a pressurized water loop.

Although the above objectives were fulfilled, it was determined late in the quarter that it would be necessary to revise the irradiation program by limiting its scope.

During the next quarter, design work and other preparations for irradiation testing will be completed.

It was concluded during the previous quarter that the advantages of in-pile loop irradiation of full-length specimens far outweighed all alternate methods of testing; an investigation into the available facilities capable of supporting an in-pile loop program revealed that only the WTR (Westinghouse Test Reactor) could satisfy the general requirements and would be available.

Due to some confusion regarding power generation of the test specimens and subsequent heat removal capabilities of the WTR loop, a complete thermal analysis was performed by Martin to assure proper design of the experiment (see Table II-9). Hand calculations were verified by machine calculations on an IBM-704 in which the diffusion equation was solved for a tubular element with equal coolant flow on both the inside and outside of the tube (see Table II-10). The Dittus-Boelter equation was utilized to determine the heat transfer coefficient. Although the hand calculations indicated that a coolant velocity of 20 feet per second would be required for adequate heat removal, the machine calculations were based on both 15 and 20 feet per second to gain an insight into the temperature ranges encountered. A standard element 30 inches long (active length) x 0.500 inches OD x 0.416 inches ID was used in both cases.

Figure II-4 shows the expected temperature distribution on the inside surface of the fuel element in the axial direction during irradiation; Fig. II-5 illustrates the radial temperature profile at the hottest point in the tube.

TABLE II-9

Results of Hand Calculations--WTR Center In-Pile Loop

1. Total heat generated per element during irradiation . . . . 125 kw

2. Average heat flux for element during irradiation . . . . 7.12×10^5 Btu/hr-ft^2

3. Maximum heat flux for element during irradiation . . . . 8.76×10^5 Btu/hr-ft^2

4. Coolant inlet temperature . . . . 500° F

5. Coolant outlet temperature. . . . . 560° F

6. Coolant velocity . . . . 20 ft/sec

7. Burnout heat flux at 20 ft/sec . . . . 2.44×10^6 Btu/hr-ft^2

8. Fuel element surface temperature for worst possible case (inside of tube, midway between each end) . . . . 644° F

9. Temperature of fuel element at meat centerline (assuming no boiling) . . . . 741° F

10. Fuel element surface temperature required for local boiling . . . . 642° F

11. Conclusions:

    1. The WTR loop, having a heat removal capacity of 500 kw, can accommodate four full-length PM-1 fuel elements.

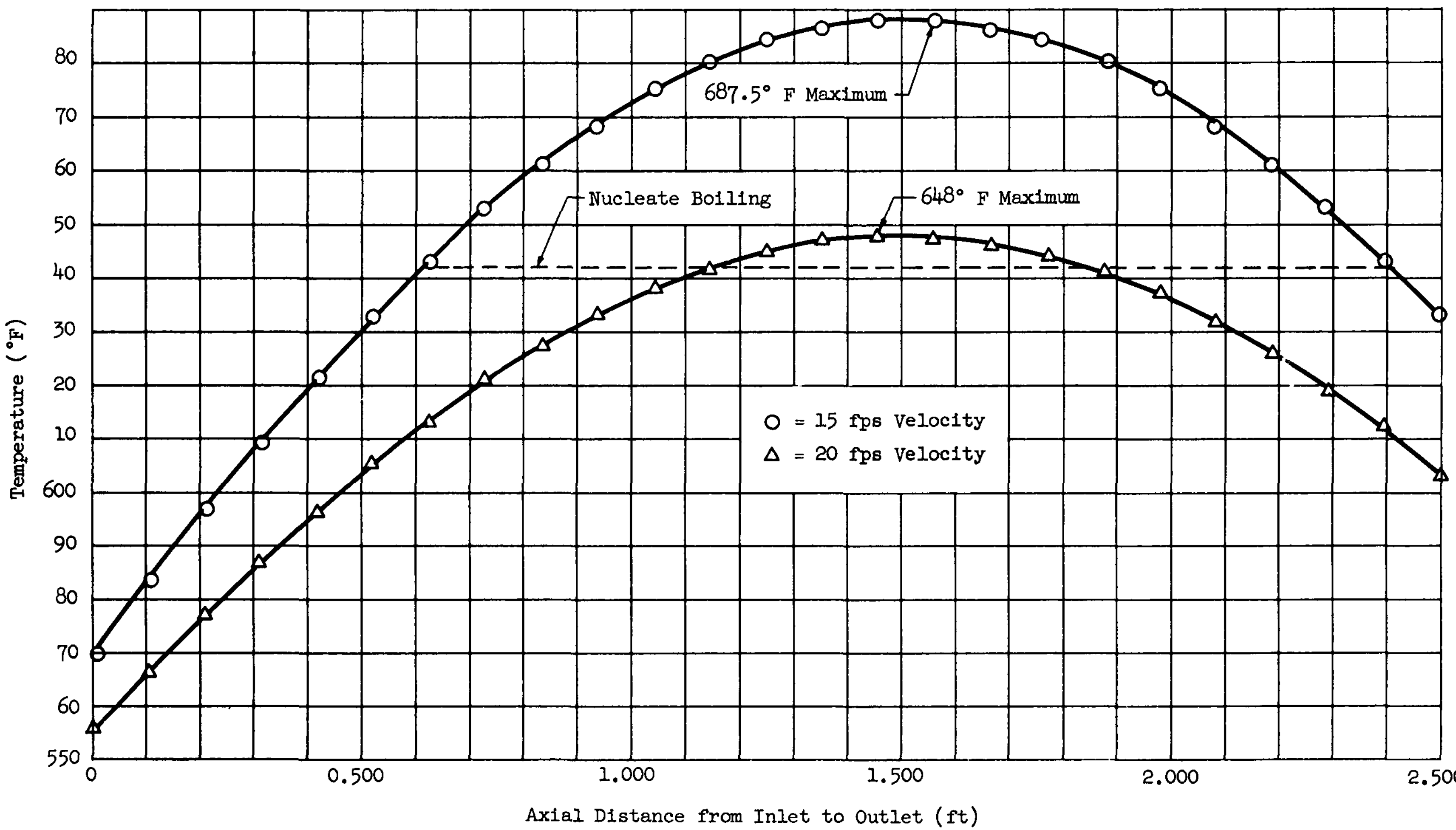


Fig. II-4. Axial Temperature Distribution of PM-1 Fuel Element (inside surface) During WTR Loop Testing

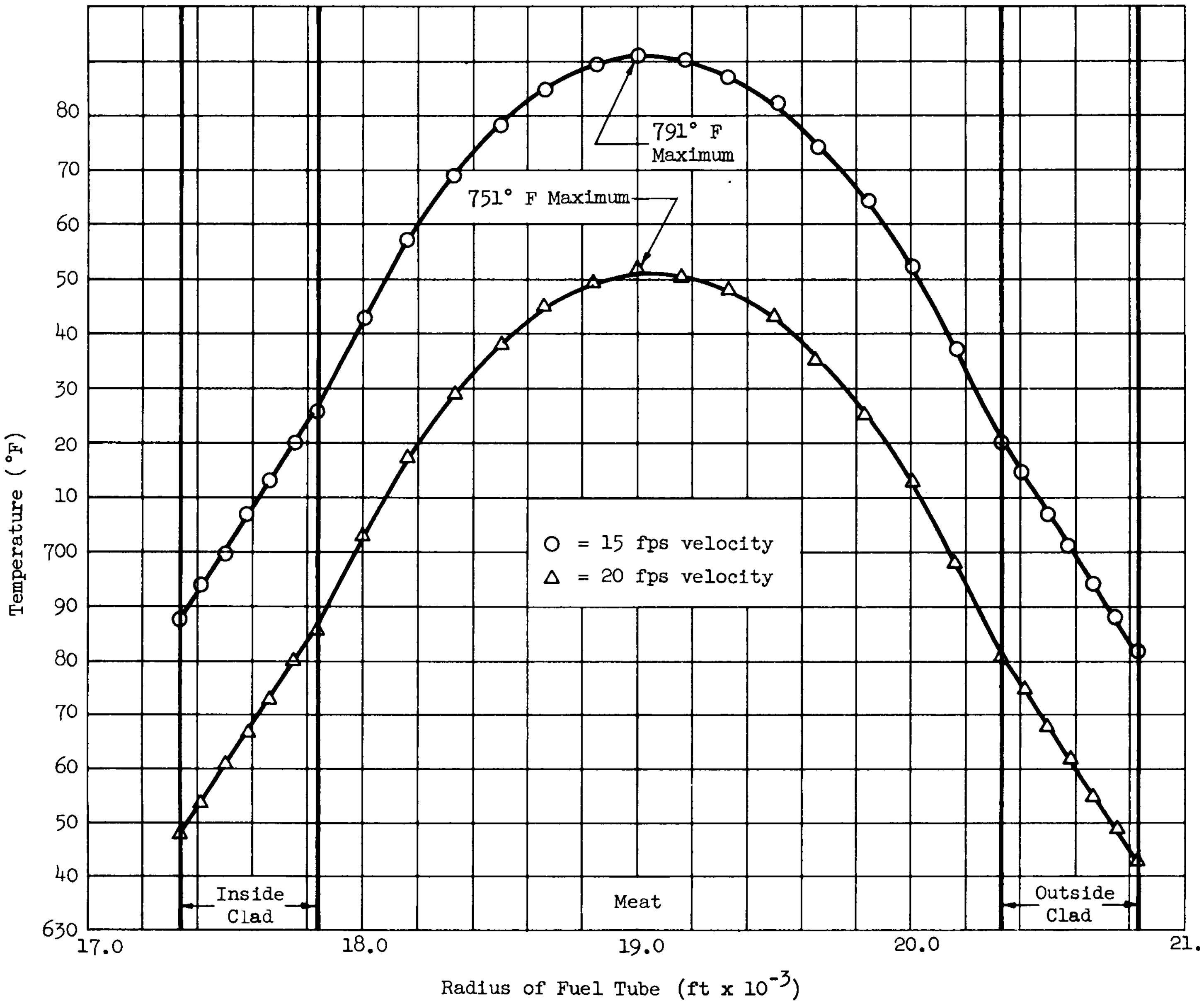


Fig. II-5. Radial Temperature Distribution at Hottest Point in Fuel Element During WTR Loop Testing (1.458 ft from inlet)

TABLE II-9 (continued)

2. A coolant velocity of 20 ft/sec should assure adequate heat removal.

3. A safety factor of about 2.8 exists between maximum heat flux attained during the irradiation and the burnout heat flux.

4. Local boiling may occur in the middle region of the fuel elements.

5. Maximum meat centerline temperature of the fuel element was calculated to be 741° F but local boiling will reduce the surface temperature to around 642° F and thereby cause a corresponding reduction of the meat temperature.

TABLE II-10

Results of Machine Calculations--WTR Center In-Pile Loop

1. For 20 ft/sec coolant velocity:

    1. Coolant inlet temperature . . . . 500° F

    2. Coolant outlet temperature (inside of tube) . . . . 551° F

    3. Maximum surface temperature of element (inside of tube at 0.6 of the length from the inlet end) . . . . 648° F

    4. Maximum temperature of element at meat centerline (assuming no local boiling) . . 751° F

2. For 15 ft/sec coolant velocity:

    1. Coolant inlet temperature . . . . 500° F

    2. Coolant outlet temperature (inside of tube) . . . . 568° F

3. Maximum surface temperature of element (inside of tube at 0.6 of the length from the inlet end) . . . . 688° F

4. Maximum temperature of element at meat centerline (assuming no local boiling) . . . 791° F

3. Conclusions:

1. If a coolant velocity of 20 ft/sec is maintained, only a small area at the center of the element will be in local boiling.

2. Meat centerline temperatures should be somewhat lower than indicated above, due to local boiling.

Test program and specimens.- After analyzing the fuel element parameters and the possible combinations most likely to give characteristics of interest, a program was developed to provide data to evaluate all concepts of interest. The proposed program will utilize fuel elements composed of varying amounts of stainless steel UO_2, and boron as burnable poisons. Some specimens were to be irradiated in an in-pile pressurized water loop, and others were to be irradiated directly in the test reactor coolant water, i.e., under test reactor operating conditions. In addition, lumped burnable poison specimens were to be irradiated in the test reactor coolant water.

Fabrication of all test elements was initiated during this quarter.

Fuel elements for in-pile loop testing.- Fuel element characteristics representative of those intended for loop testing are listed in Table II-11. It should be noted that these are only typical--exact data are presented later.

TABLE II-11

Typical Loop Irradiation Specimen Characteristics

1.	Active length	36 in.
2.	Outside diameter	0.500 in.
3.	Inside diameter	0.416 in.
4.	Meat thickness.	0.030 in.

TABLE II-11 (continued)

5. Clad material . . . . Modified 347 SS

6. Core matrix material . . . . 304 SS

7. Core composition:

   1. UO_2 . . . . 30 wt %

      U-235 . . . . 24.6 wt %

   2. B. . . . . 0.22 wt %

   3. SS . . . . 69.78 wt %

Since it appeared that only 4 full-length elements (36-inch active length) could be loop tested at any one time, it was planned to fabricate two of these elements in the form of composites consisting of 6 segments in one element and 7 in the other. This would permit testing of all the most promising combinations and variables. U-235 burnups of 36-, 52-, and 72-atom percent would be achieved during this program. To accomplish this, it would be necessary to remove one segmented fuel element after approximately 4 months of irradiation and to replace it with an identical segmented element for irradiation to 52% burnup. Those specimens remaining in-pile for the total irradiation period (approximately 12 months) would attain 72% burnup. Parameters of the 36% burnup samples are listed in Table II-12, and of the 72% samples, in Table II-13. As mentioned above, 6 segmented specimens identical to the 36% samples in Table II-12 would also undergo 52% burnup during the latter portion of the irradiation program.

TABLE II-12

36% Burnup Loop Specimens

Specimen No.	Tube Dia OD (in.)	Meat Thick. (mils)	Wt % UO_2 in Meat	Wt % Burnable Poison in Meat	Active Length (in.)
*L136	0.500	30	30	0.22 B as B_4C	5
L236	0.500	30	25	0.22 B as B_4C	5
L336	0.500	30	25	0.22 B as ZrB_2	5

TABLE II-12 (continued)

Specimen No.	Tube Dia OD (in.)	Meat Thick. (mils)	Wt % UO_2 in Meat	Wt % Burnable Poison in Meat	Active Length (in.)
L436	0.500	30	25	None	
L536	0.500	25	30	0.22 B as B-SS alloy	5
L636	0.500	30	25	0.11 B as B-SS alloy	5

TABLE II-13

72% Burnup Loop Specimens

Specimen No.	Tube Dia OD (in.)	Meat Thick. (mils)	Wt % UO_2 in Meat	Wt % Burnable Poison in Meat	Active Length (in.)
*L172	0.500	30	30	0.22 B as B_4C	4.25
L272	0.500	30	25	0.22 B as B_4C	4.25
L372	0.500	30	25	0.22 B as ZrB_2	4.25
L472	0.500	30	25	None	4.25
L572	0.500	25	30	0.22 B as B-SS alloy	4.25
L672	0.375	30	30	0.11 B as B-SS alloy	4.25
L772	0.375	30	25	0.22 B as B-SS alloy	4.25
LC72	0.500	30	30	0.11 B as B_4C	30
LA72	0.500	30	25	0.11 B as B-SS alloy	30

*Legend for sample identification:

L--designates "loop" specimens.
1,2,3,...--the digit following the "L" refers to the position of the segment in the overall test element--increasing in direction of coolant flow.
...36 or....72--the last two digits designate burnup.
LA72 and LC72--the full-length unsegmented elements in which "A" and "C" designate "Alloy" and "Carbide," respectively (the method of incorporating boron).

Fuel elements for irradiation directly in reactor coolant water.- The parameters of these elements are set forth in Table II-14. It should be noted that specimens B872, B972 and B1072 are MPR Zero-Power Test fuel elements which constitute a "shelf" item immediately available for testing. None of these elements would be segmented. Burnup has been specified as 72%. However, the elements could be removed from the test reactor at any time to follow progress of the irradiation.

TABLE II-14

Bare Fueled Specimens

Specimen No.	Tube Dia OD (in.)	Meat Thick. (mils)	Wt % UO_2 in Meat	Wt % Burnable Poison in Meat	Active Length (in.)
B872	0.375	18	23	0.11 B as B_4C	23
B972	0.375	18	23	0.11 B as B_4C	23
B1072	0.375	18	23	0.11 B as B_4C	23
B1172	0.500	30	30	0.22 B as B_4C	30
B1272	0.500	30	25	0.11 B as B_4C	30
B1372	0.500	30	25	0.11 B as B-SS alloy	30

Lumped burnable poison specimens.- Two burnable poison elements would be irradiated to approximately total burnup. These specimens would be 0.500 inch OD x 30 inches long (active length). Irradiation would be accomplished directly in the test reactor coolant water with the experiment designed so that the specimens can be periodically removed for inspection and then re-inserted inpile. At present the composition of these specimens has not been firmly determined.

Pre-irradiation testing.- All irradiation test specimens will undergo a thorough examination before testing. This examination is to serve two purposes: First, to compile data for comparison with post-irradiation test results, and second, to verify the acceptability of the specimens for irradiation. The following is the minimum effort planned for pre-irradiation inspection:

Nondestructive tests

(1) Porosity testing

(2) Ultrasonic testing

(3) Standard quality control testing to ensure adherence to length, thickness, etc. specifications.

Destructive tests--(to be performed on sample elements)

(1) Thermal shock test

(2) Peel test

(3) Intergranular corrosion tests

(4) Autoclave tests

(5) Metallography.

Inspection of SM-1 elements.- A trip was made to Oak Ridge National Laboratory to obtain information on reactor control materials. It was found that the APPR-1 control rods, recently removed from the reactor, had suffered cracking of the clad and some separation of the clad-to-core bond at the end of the element. These failures were attributed to a combination of all of the following devices:

(1) Helium gas pressure from transmutation of boron.

(2) Diffusion of boron into the clad to cause embrittlement and damage during irradiation.

(3) Excessive burnup and corresponding damage in the core areas adjacent to the clad.

Although no damage was discernible from visual inspection of the SM-1 fuel elements containing small amounts of boron as a burnable poison, some internal damage was anticipated. This prognosis is based on results obtained from irradiation testing of miniature specimens.

Alternate irradiation tests.- Toward the end of this quarter, it was determined that the irradiation program recommended could not proceed for several months until contractual arrangements between the AEC and the possessors of commercial testing reactors were settled. Therefore, an alternate irradiation test program will be proposed. Essentially, it consists of:

(1) Limited "bare" element irradiations.

(2) Insertion of a PM-1 fuel bundle into the SM-1.

(3) Insertion of PM-1 control rod material into the SM-1 for reactivity and burnup purposes.

It is anticipated that the program will suffer somewhat from the above restriction due to inability to obtain data at PM-1 operating conditions of temperature and pressure.

The situation will be somewhat alleviated if irradiations can be conducted in the SM-1. However, accelerated burnup will not be obtained in this case--with subsequent effect on the irradiation program time schedule.

C. SUBTASK 2.3--REACTOR FLOW STUDIES

M. P. Norin, W. J. Taylor, I. L. Gray

The objective of experimental PM-1 reactor flow studies is to evaluate and optimize the hydraulic design of the reactor. The work is to be conducted through 3 tests; 2 serve to give preliminary information for use in reactor design and in the design of the third test; the third test makes use of a full-scale flow model.

During the 2nd quarter, a test program was formulated and initiated. During the next quarter, it is expected that:

(1) The simplified flow model (1/4 scale) tests will be completed.

(2) The orifice-bundle tests will be completed.

(3) Design of the full-scale reactor flow test will be completed.

(4) Procurement of material and fabrication of components will be initiated.

Work was initiated with the formulation of a program for experimental work. The following 3 test setups are to be utilized in the program:

1. Orificed-Bundle

A full-size section of the core, containing up to 3 orifice plate designs, will be fabricated. The work performed using this section will provide information leading to the optimum orifice plate design; that is, the design which will give the required ratio of flow inside the tubes to that outside the tubes.

2. Simplified Flow Model

This will approximate the reactor vessel internal configuration at about 1/4 scale. This work is designed primarily for study of the effect of entrance nozzle and water box configurations on the flow pattern in the thermal shield region.

3. Complete Full-Scale Reactor Flow Model

The final experimental work will be performed with a full-size flow model of the prototype. This is ito provide a check of the hydraulic design and to provide information which cannot be obtained during the preliminary work with items (1) and (2).

A preliminary design of the bundle-type unit has been prepared. In these tests a solid boundary, formed by the containing vessel, is necessarily substituted for the fluid boundary present in the reactor. Thus, it is possible that wall effects may influence the data of the bundle tests. The design will be such that bundles of 19, 37, and possibly 61 tubes may be tested to properly account for wall effects. A representative number of tubes will be instrumented for the measurement of flow rate inside the tubes. Measurement of the total flow rate to the unit will allow computation of the flow outside the tubes.

Design of the scaled-down, simplified model has been initiated. The model will be instrumented for static pressure distribution around the periphery of the water box, pressure drop across the water box orifice plate, and flow distribution around the annular thermal shield region. Portions of the model will be made of transparent material and provisions will be made for the injection of dye to allow observation of the flow patterns in the regions of interest.

The flow loop to be used for the full-size model tests was reassembled in preparation for chemical cleaning of the components. Tubing of Type 347 stainless steel was ordered for the fabrication of simulated fuel elements for the preliminary and final test work.

D. SUBTASK 2.4--HEAT TRASFER TESTS

J. J. Jicha, M. P. Noren, C. Eicheldinger, I. L. Gray

The PM-1 heat transfer program is designed to obtain correlations of heat transfer coefficients, tube wall temperature, bulk temperature, and pressure drop and flow effects in both the nonboiling and the local-boiling ranges. Experimental data obtained during the second quarter, and correlated with the parametric study program, will aid in confirming that the PM-1 core can be operated under local-boiling conditions.

The overall objectives of the program are to obtain experimental data to support refined thermal and hydraulic design of the PM-1 core and to determine experimentally those quantities, such as burnout heat flux, which are difficult to calculate.

During the second quarter, the following objectives were fulfilled:

(1) The heat transfer test program was outlined

(2) SETCH-1 tests were completed.

The following is expected to be accomplished during the next quarter:

(1) Testing of STTS-2 will be completed

(2) STTS-3 design will be completed

(3) Work will be initiated on the fabrication of STTS-2.

Upon completion of the parametric study and the selection of design conditions for PM-1, the experimental program given in the first Quarterly Progress Report was revised to include three test sections based on the geometry of the PM-1 fuel elements. These are in addition to the seven-tube test section (SETCH-1) and the single-tube test section (STTS-2) which were on hand from a previous heat transfer program. The new test sections are:

(1) Single-tube test section (STTS-3)--This will be a single-tube test section similar in design to STTS-2 but with PM-1 fuel element dimensions. The test section will be instrumented to obtain local-boiling pressure drop and heat transfer data inside the tubes.

(2) Single-tube test section (STTS-4)--A single-tube test section will be fabricated which will accommodate coolant flow both inside and outside the tube. The unit will be used for burnout measurements.

(3) Seven-tube test section (SETCH-2)--The seven-tube test section will have coolant flow outside the tubes only. The test section will be instrumented to obtain pressure drop and heat transfer data outside the tubes.

The work of items (1) and (3) above are complementary, in that the two tests give heat transfer and pressure drop data for both the inside and the outside of the tubes. A breakdown of the work into two separate tests was dictated by experience with SETCH-1. It was found that the installation of instrumentation is greatly simplified if the coolant flow is along one surface only. Thus, the surface not in contact with fluid is readily available for instrumentation. The deviation from a true analog, due to removing heat from only one tube surface, can be compensated for more easily than can the problems of instrumentation installation.

Testing of SETCH-1.- In SETCH-1, thermocouples were installed at the outside surface of the central tube. This surface temperature is one of the parameters involved in the determination of local-boiling data. During preliminary operation, it was found that all of the thermocouples at the surface failed. The design of SETCH-1 was such that replacement of the thermocouples would require extensive rework of the unit. Since the dimensions of SETCH-1 are different from those required in a PM-1 test section, this rework was not deemed worthwhile. A curtailed experimental program of 28 runs was performed. Since the lack of surface temperature measurements precluded the accumulation of heat transfer data, the short program was performed primarily for the evaluation of the operational design of SETCH-1, the operating conditions being selected to conform to those of interest for PM-1. A photograph of SETCH-1 taken during testing is presented as Fig. II-6.

Testing of STTS-2.- Fabrication of STTS-2 was completed, thermocouples were calibrated and installed, and the unit was fitted into the heat transfer loop. The installation is shown in Fig. II-7. The loop was modified to include flow measurement and control components for loop operation at the relatively low flow rates required for PM-1. A contact heater, composed of resistance wire wrapped around the piping adjacent to the test section inlet and controlled by a variac, was installed to allow fine control of inlet temperature.

Design work.- STTS-3 design is being detailed. This test section is similar in design to that of STTS-2. Work was initiated on the design of STTS-4 and SETCH-2.

E. SUBTASK 2.5--ACTUATOR PROGRAM

J. S. Sieg, R. Manoll

During the second quarter, the objectives of the actuator program were:

(1) To evaluate actuator proposals.

(2) To determine the type of actuating system to be employed.

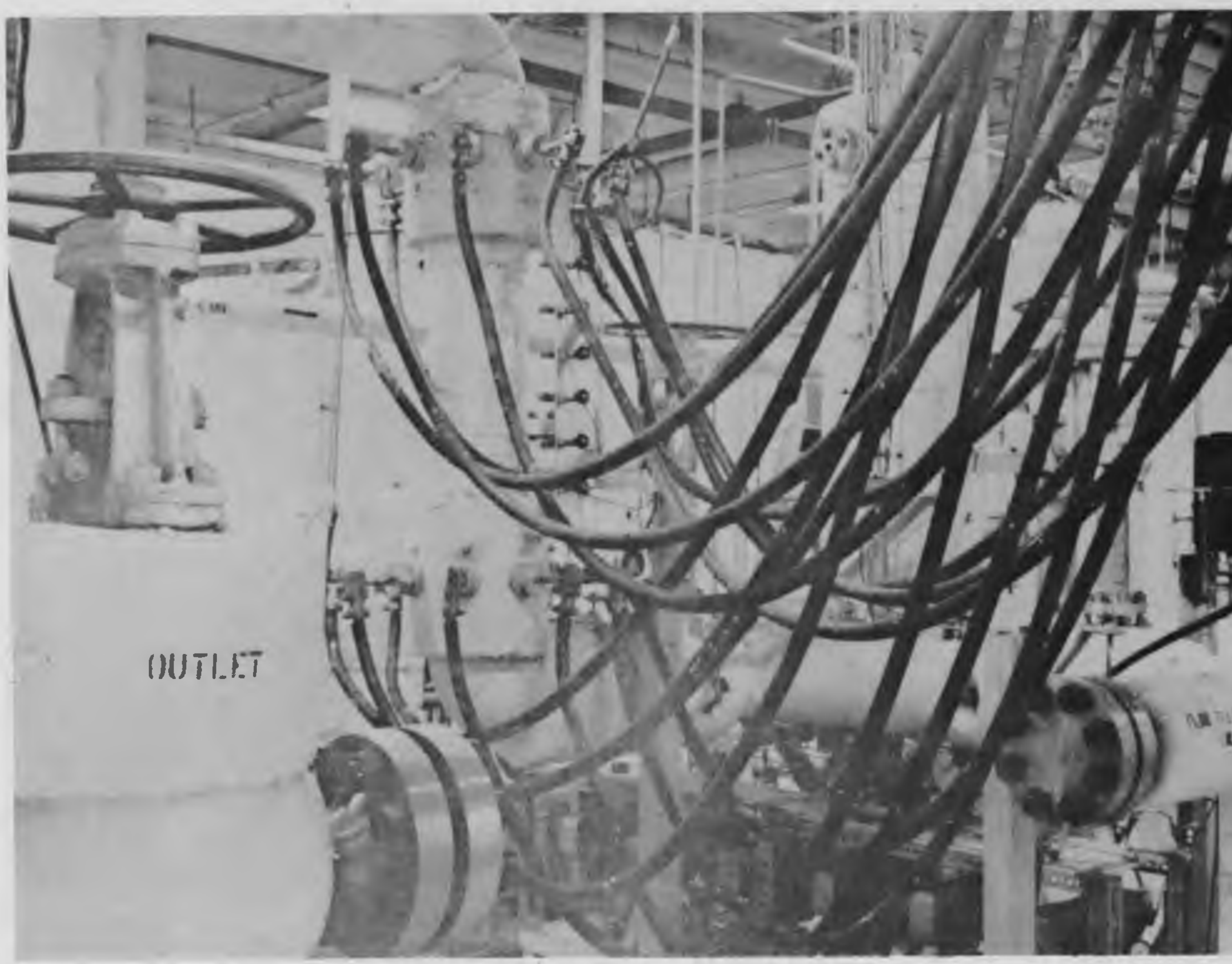


Fig. II-6. Setch-1 During Testing

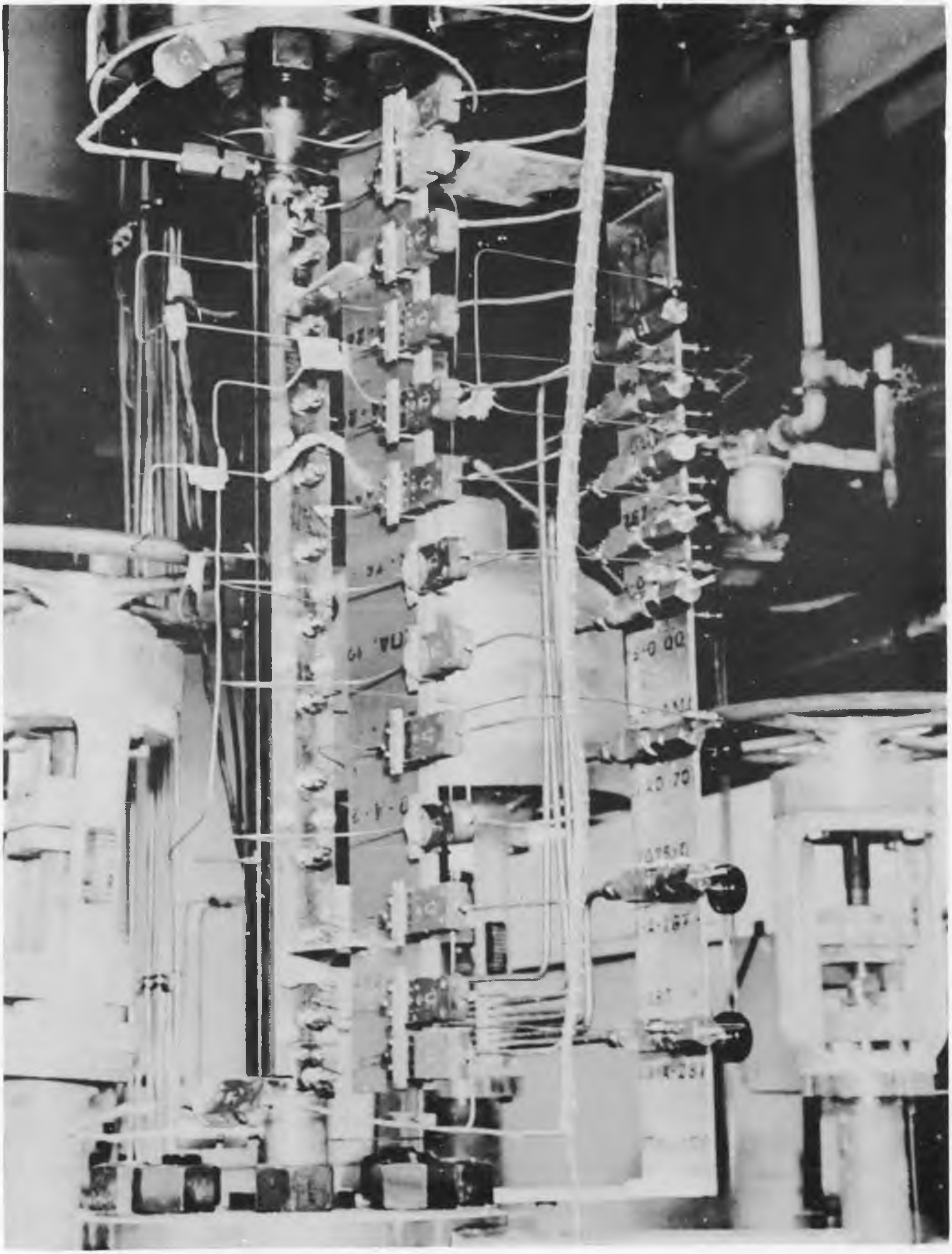


Fig. II-7. Installation of STTS-2

(3) To prepare a final specification to make immediate actuator procurement possible.

(4) To select an actuating system vendor.

During the quarter, various actuating systems were evaluated; the most promising one was selected for development; a final specification was prepared and submitted to the two vendors proposing the selected type, and an actuator system vendor was selected. In addition, a dummy segment of the core (including a dummy control rod) was designed. It will be used to determine the hydraulic characteristics of the actuator. The design was submitted to various vendors for price and delivery quotes.

The design of the actuator will continue through the next two reporting periods. Liaison will be maintained with the selected vendor.

1. Evaluation of Actuator Proposals

Seven designs were submitted by vendors. These were:

(1) Two rack-and-pinion designs--One employed a magnetic clutch to separate the motor from the rod drive mechanism; a constant leakage rate dynamic seal was used to isolate the high pressure water from the drive motor. The other used a canned motor to drive the rack directly; a scram spring was provided. Both configurations involved a vertical tube housing the rod and rack mechanisms, with the shaft of the drive motor placed at right angles to the tube. The first design utilized a servo driven by the pinion for position indication. The second utilized a coil stack transmitter activated by a plug attached to the rack.

(2) Two nut-and-lead screw designs--These employed a magnetically operated latch to separate the rod drive mechanism from the drive motor. One utilized a dynamic seal to isolate the high-pressure water from the drive motor; the other utilized a canned motor. The configurations involved a vertical housing for the lead screw, rotating nut, latch, gearing, bearings, position indicator, and motor.

(3) A magnetically locked nut-and-lead screw design--This is an in-line type of device in which the nut and its rotor extension were rotated by the rotating field generated by a stator located outside of the high-pressure housing. The nut was composed of several segments with each segment having a roller screw

section, spring loaded to remain in the open position, at one end and a rotor section at the opposite end. Scran was accomplished by collapsing the field of a magnet which held the nut segments in the closed, or locked, position, thereby permitting the segments to open and release the lead screw. No dynamic seals were employed with this type of actuator.

(4) Two magnetic jack designs--The magnetic jack type of actuator was selected because of its simplicity, potentially low cost, and because a major portion of it can be expeditiously replaced while the plant remains at operating pressure and temperature. In general, the other types could not compete with the magnetic jack in any of these areas.

2. Selection, Specification and Choice of Vendor

The actuator system to be installed in the PM-1 power plant will be designed, fabricated, and tested by the TAPCO Group of the Thompson Ramo Wooldridge Company in accordance with the requirements set forth in Martin Specification MN-7221.

3. Description of the Selected System

The main components of the PM-1 actuator system are:

(1) Six actuators

(2) A system power source

(3) System countrol and position-indicating equipment.

The operations of the actuator itself will be basically as described in ANL 5768* with two important exceptions: The actuator will be designed for ease of maintenance in that most failure-prone components can be replaced as a unit without primary loop depressurization, and a different position-indication transmitter will be utilized.

The pressure thimble which encloses the rod bundle deflected by magnetic flux, the movable armature, the holding structure, the scram spring, and the latch that locks the bundle to the control rod will be bolted to the pressure vessel head nozzles and will extend about seven feet above the pressure vessel head. A hollow cylinder containing the grip, lift, and hold coils, and the position-indicating transmitters will

*ANL 5768--Young, Joseph N., "Design and Performance Characteristics of Magnetic Jack-Type Control Rod Drive," December 1957.

be placed over and about the thimble and latched into position. This latch, which will be the only mechanical connection between the thimble and the cylinder containing most of the equipment prone to failure, will be designed for operation, in the event that equipment maintenance is required, from a platform located above the surface of the reactor shield water.

The position-indication transmitter associated with each actuator will utilize a differential transformer, the reluctance element of which is a plug attached to the rod bundle whose windings are positioned inside of the coil cylinder along the vertical axis of the actuator pressure thimble by a null-seeking servo motor driven by the transformer error signal. A synchro, which will also be driven by the servo motor, will transmit rod position signals to a receiver mounted on the plant control console.

The actuator system power sources will be mounted in drawers at the base of the control console and will serve two main functions: the conversion of plant power via silicon rectifiers into dc to be used to drive the coils, and the generation of appropriate signals to switch the d-c power to the various coils in the proper sequence for operation of the actuator.

Switching signals will be generated through a subsystem utilizing transistorized oscillators and flip-flops and variable RC circuits to activate or deactivate transistor switches.

The actuator system controls will be mounted on the control console and will allow considerable operating flexibility. The following options can, if desired, be granted the operator:

(1) Placing the actuator system under manual or automatic control.

(2) Scramming or setting back the system.

(3) Under manual control:

   (1) Holding the rod(s) steady, or extracting or inserting them at a 2- or 6-inch per minute rate.

   (2) Moving the rod(s) singly, as 2 banks of 3, or as a single bank of 6.

   (3) Moving the rod(s) through a single increment (0.080 inch) of motion rather than continually.

The position of each rod will be continuously indicated to within 0.05 inch of true position on one of the 6 dual-scale dials (4 to 5 inches in diameter) mounted on the control console.

Inasmuch as the selected design does not make use of a dynamic seal, the seal data collected during the first quarter will not be directly applicable to Subtask 2.5. It is, however, applicable to other areas, such as pump seal design, etc.

III. TASK 3--PRELIMINARY DESIGN STUDY, SELECTION AND SPECIFICATION

Project Engineers--Subtasks 3.1, 3.2, 3.3: R. Akin, C. Fox

This task covers preparation for and accomplishment of preliminary design.

A. SUBTASK 3.1--PARAMETRIC STUDY

During the quarter, the parametric study was completed. The methods, data, and results have been reported in MND-M-1852.

B. SUBTASK 3.2--PRELIMINARY DESIGN

During the second quarter, the preliminary design of the PM-1 power plant was accomplished. The results of the design will be submitted during the next quarter in the form of:

(1) A topical report

(2) A set of preliminary design drawings

(3) A book of outline specifications.

The main areas considered during preliminary design were:

(1) The primary system

(2) The secondary system

(3) Controls and instrumentation

(4) Packaging and housing

(5) Reliability.

Each will be discussed in turn.

In general, the areas of preliminary design discussed here will be extended in support of final design during the next quarter.

1. The Primary System

Primary system considerations were further subdivided into nuclear analysis studies, heat transfer and system analysis studies, shielding analysis studies, and design studies.

a. Nuclear analysis studies

E. Scicchitano, R. Hoffmeister, F. Todt

Non-uniform loading.- A preliminary study to determine the feasibility of loading the PM-1 core non-uniformly was completed. It was found that although non-uniform loading could increase core life, reduce the maximum fuel burnup, and result in constant power distribution in the core with time, inherent limitations, such as the allowable ranges of the different variables, manufacturing costs for making several different tube sizes and loadings, and complexity and flow problems resulting from variable spacing, caused it to be of questionable value in the first core.

Fuel loading.- The initial fuel loading was determined based upon a heat production rate of 9.35 megawatts, a core life of 18.70 megawatt years, a cold-core temperature of 68° F, a hot-core temperature of 463° F, the use of water at 1300 psia as moderator and coolant, and the following design characteristics:

Geometry, right circular cylinder (approximately)	
Diameter	$\approx$ 23.0 in.
Height (active)	30.0 in.
Fuel element OD	0.50 in.
Fuel element ID	0.416 in.
Fuel matrix thickness	0.030 in.
Clad thickness	0.006 in.
Number of fuel elements	725
Pitch, triangular	0.665 in.

Results of the studies based upon the above criteria are shown in Figs. III-1, III-2, and III-3. Figure III-1 shows the effect on core life of varying the w/o loading of UO_2 (non-uniform burnup), Fig. III-2 shows the comparative effects on core life of assuming non-uniform as opposed to uniform burnup, and Fig. III-3 shows the effects of introducing poison in various forms.

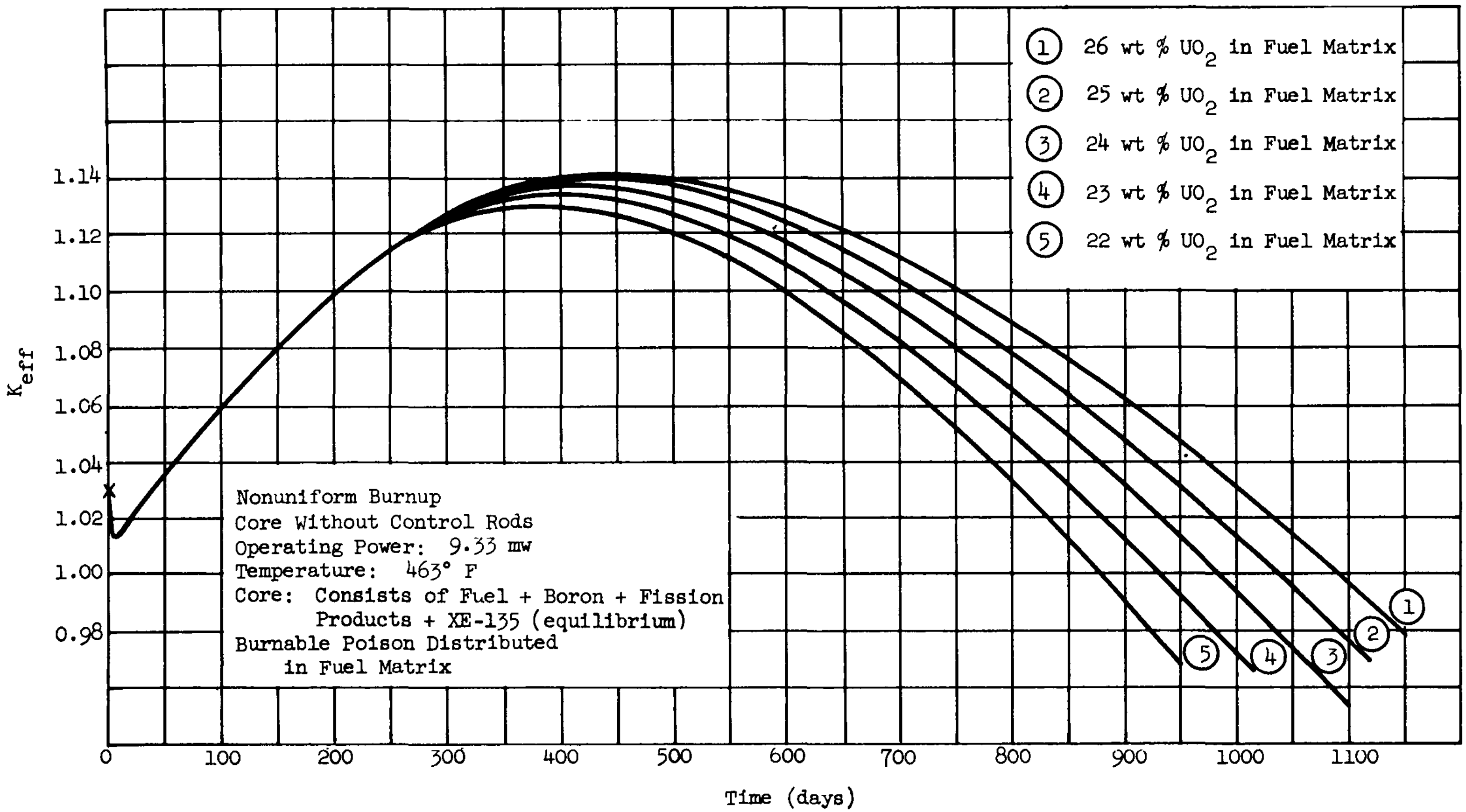


Fig. III-1. Effect of UO_2 Wt % in Fuel Matrix on K_{eff} with Time

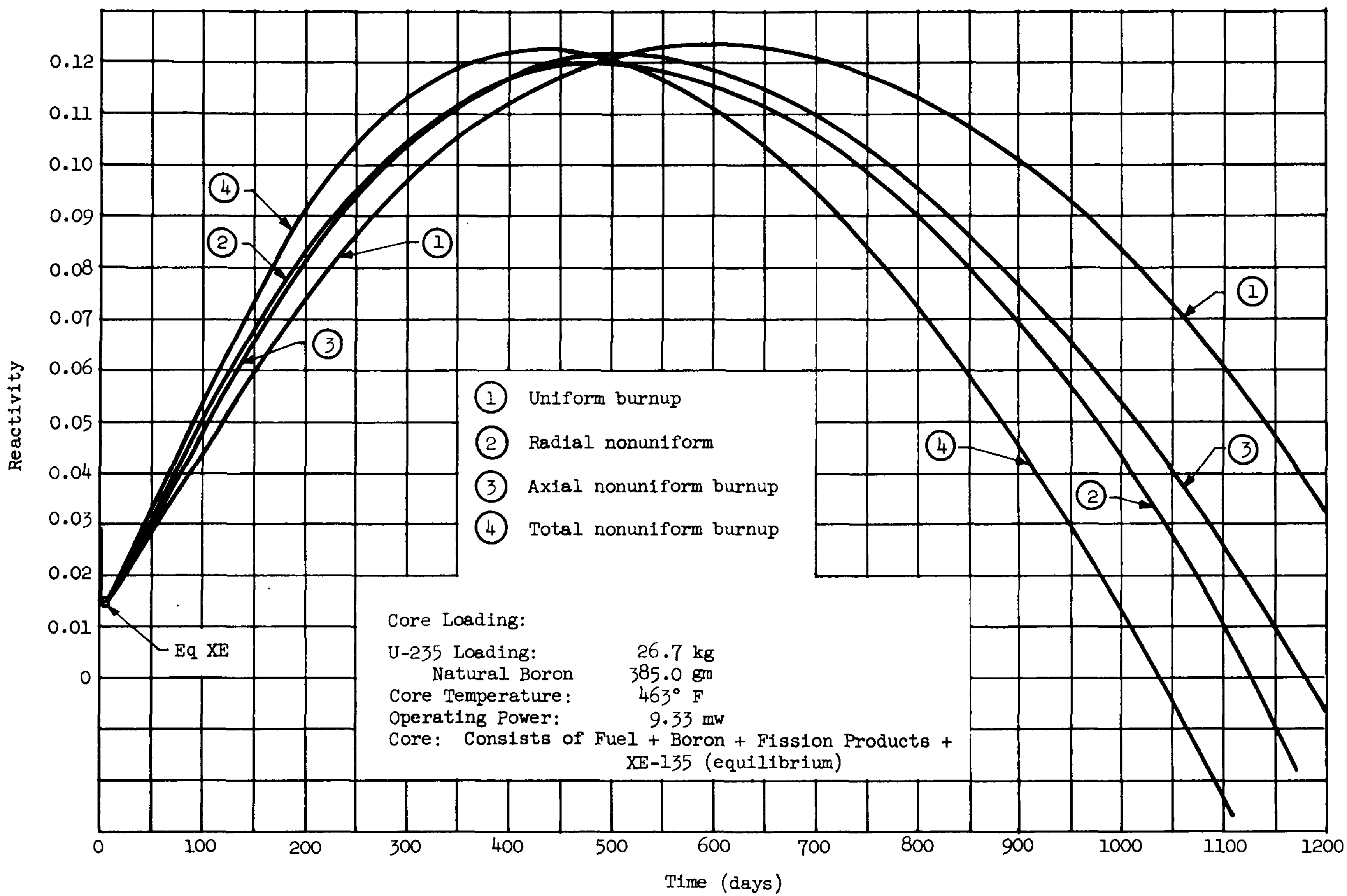


Fig. III-2. Reactivity versus Time--Effect of Nonuniform Burnup

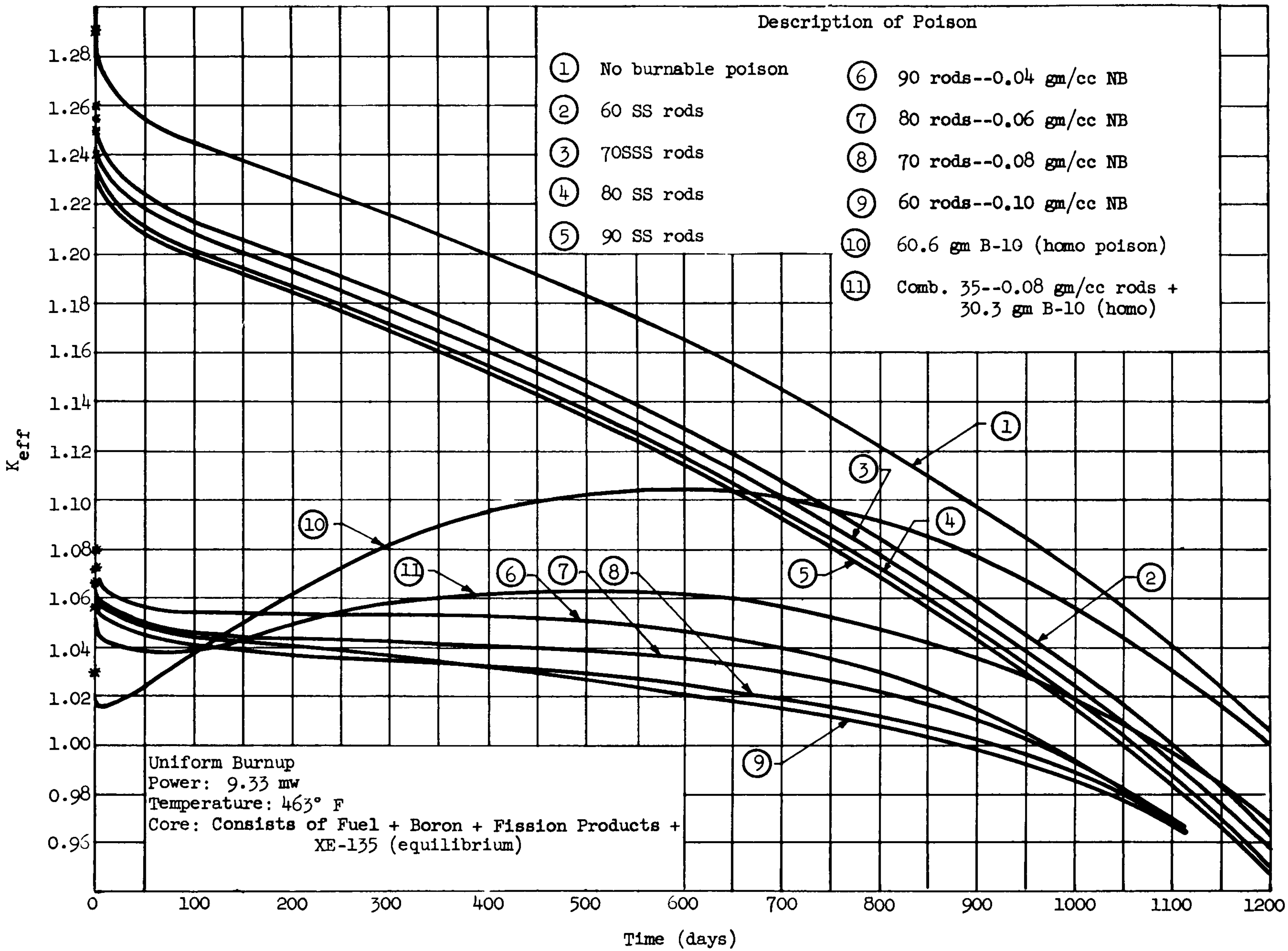


Fig. III-3. Burnable Poison Study--K_{eff} versus Time

Analysis of the curves resulted in the decision to:

(1) Select a 25 w/o loading.

(2) Utilize lumped burnable poison in order to make use of the selected 6-Y-rod control system.

The U-235 inventory then became 26.7 kg and anticipated burnup, 9.0 kg. Core life exceeds two years.

In view of the many problems inherent with the introduction of burnable poison into the fuel elements, and the ease of introducing lumped poisons in the PH-1 core, the decision was made to lump all burnable poison.

The non-uniform burnup calculations of Fig. III-1 were performed using a multigroup criticality code linked with a two-group, multiregion, one-dimensional code. The core was divided into 6 regions in both the radial and axial directions.

Rod design and worth.- Studies to determine preliminary design specifications for size and location of the control rods were completed. The rod system was designed to utilize the minimum number of control rods consistent with the core design.

Results of parametric studies and initial preliminary design studies indicated that the maximum reactivity that must be controlled would be sufficiently reduced by incorporating lumped burnable poisons in the core to allow a 6-rod system to meet the control requirements.

The rod poison used in the preliminary design studies was boron-stainless steel (2.5 wt % B-10). Even if boron is not to be used as absorber material, the results of these studies for determining size and location of the rods are applicable to other materials, since all of the absorbers considered will be thermally black.

The worth of 6 cruciform rods (see Fig. III-4A for rod configuration and location) was found to be $\approx -21\%\ \Delta\rho$. Since the decision to use a triangular pitch for the fuel elements was made for other than nuclear design reasons, cruciform rod studies were discontinued. Control design studies then proceeded using Y-shaped rods.

The worth of 6 Y rods as shown in Fig. III-4B (R = 6.5 inches) was $-23.6\%\ \Delta\rho$. The worth of 5 of the 6 rods was $-14.8\%\ \Delta\rho$; the worth of 4 adjacent rods was calculated to be $-8.5\%\ \Delta\rho$.

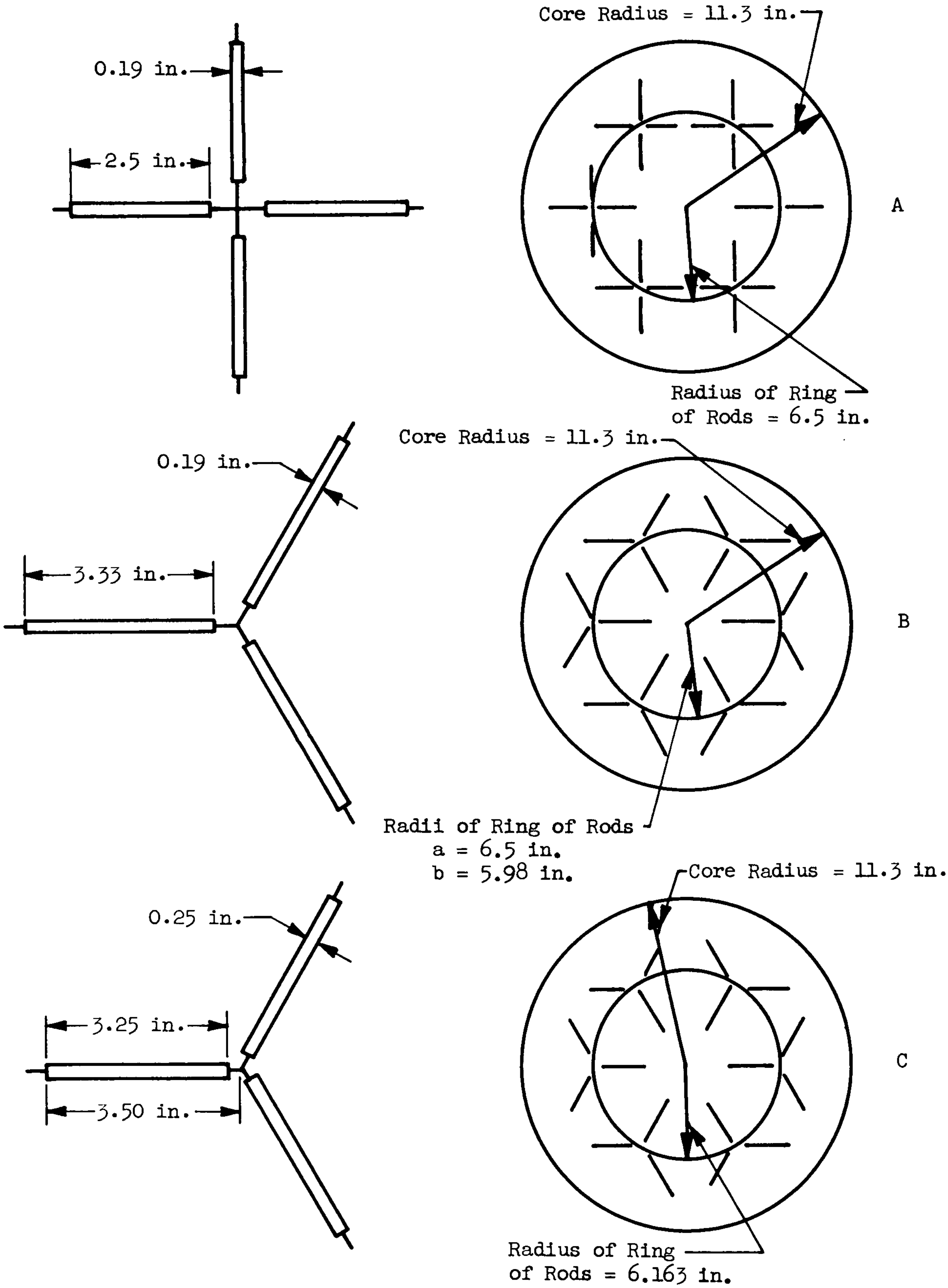


Fig. III-4. Rod Configuration and Location

The flux distribution for the core with the above 4 rods fully inserted indicated that the worth of these rods could be increased by placing them closer to the center of the core. The rods were relocated to a radius of 5.98 inches, and the worth of the 4-rod bank was found to increase to $\approx$ -9.4% $\Delta\rho$. The worths of the 5- and 6-rod banks were not recalculated; however, the flux distributions indicate that the worths of these rod banks will also increase.

The rod configuration and locations consistent with the preliminary design core, considering actual fuel element locations, rod guides, etc., are shown in Fig. III-4C. For these conditions, the 4-rod bank is worth $\approx$ -9.3% $\Delta\rho$. The worths of the 5- and 6-rod banks will be greater than or equal to the values given above.

Control rod worths were calculated using the two-dimensional, three-group, IBM-704 machine code PDQ in X-Y geometry*. The machine-calculated rod bank worths (i.e., the difference between core reactivities with and without rods) were corrected for the effect of the step approximation required in mapping 2 of the 3 arms of the Y-shaped rods in X-Y geometry. Comparison of the analytical results obtained using this technique with the experimental results of previous studies indicated that the analytical results obtained are good within ±10%. For the rod design studies described above, the calculated worth was assumed to overestimate the actual worth by $\approx$ 10%; the bank worths reported above are 10% below calculated values.

Control requirements.- Preliminary studies were performed to establish control requirements. Since overall requirements are strongly dependent on the burnable poison scheme used, this study evaluated gross control requirements for different schemes.

It was found that proper choice of a lumped poison system could significantly reduce the maximum amount of reactivity that must be controlled. In addition, reactivity peaking with time could either be reduced or completely eliminated.

It appears that the reactivity which will have to be controlled throughout the core life at operating conditions (i.e., 463° F, equilibrium xenon, plus other fission product poisons) is $\approx$ 3 to 6%.

The difference in reactivity between the hot-operating conditions and the cold, clean core condition (68° F and assuming that all of the fission products decay out of the system) was calculated for various times. This change in reactivity increased from $\approx$ 6.1% initially to $\approx$ 7.2% at midlife and to $\approx$ 8.3% at end-of-life, the increase in reactivity being primarily due to the increase in fission products with time.

---

*"Zero-Power Test - Engineering Report," MND-MPR-1646, December 1958.

The maximum amount of reactivity that must be controlled is, then, from Fig. III-3 (Curve 6) and the $\Delta \rho$ calculated above (operating condition to cold, clean), $\approx$ 12.2%.

As a matter of interest, the reactivity effect of equilibrium xenon for full-power operation and maximum xenon buildup after shutdown at initial startup are -1.4% and -0.23% (-1.63% total), respectively.

Control design criteria established for the PM-1 core are that the rods must be adequate to shut down and hold down the reactor at any time of the core life with: (1) one rod stuck in the full-out position, or (2) any two rods stuck in the operating position.

The rod bank worth for 5 of the 6 rods was calculated to be $\geq$ -14.8%. Since the maximum cold, clean core reactivity is $\approx$ 12.2%, the shutdown reactivity is $\approx$ -2.6% $\Delta \rho$, which satisfies the first criterion.

In order to evaluate the shutdown condition of any two rods stuck in the operating condition, consideration was given to both the cold, clean core reactivity and the hot-operating core reactivity. Presently, it is planned to use a 6-rod bank operation for control, although alternating 3-rod banks may be used near the end of core life to permit burnup of fuel and burnable poison in the extreme upper portion of the core.

For the present studies, the worth of 4 rods fully inserted and 2 rods partially inserted was calculated as follows:

$$\Delta \rho = \Delta \rho_4 + \frac{-\rho_{core}}{\Delta \rho_6} (\Delta \rho_6 - \Delta \rho_4)$$

where

$\Delta \rho_4$ = worth of 4 rods fully inserted

$\Delta \rho_6$ = worth of 6 rods fully inserted

ρ_{core} = reactivity of core at operating conditions

For example, for the core using lumped poisons (0.04 gm/cc) shown in Curve 6 of Fig. III-3, the minimum worth of the system is:

For initial conditions:

$$\Delta \rho = -9.3 + \frac{-(5.9)(-23.6 + 9.3)}{-23.6}$$

$$= -12.9\% \ \Delta \rho$$

For midlife

$$\Delta \rho = -9.3 + \frac{-(5.0)(-23.6 + 9.3)}{-23.6}$$

$$= -12.3\% \ \Delta \rho$$

For end-of-life

$$\Delta \rho = -9.3 + \frac{0(-23.6 + 9.3)}{-23.6}$$

$$= -9.3\% \ \Delta \rho$$

Since the control requirements for initial operation, midlife, and end-of-life are 12.0%, 12.2%, and 8.3%, respectively, sufficient control is available. Similar calculations, using a burnable poison loading corresponding to Curve 7 of Fig. III-3, show a greater shutdown safety margin.

Control rod material.- A preliminary study was performed to evaluate various materials for use as control rod poisons. The materials considered were boron, hafnium, cadmium-indium-silver, europium, and gadolinium-samarium. The specific rod material compositions that were studied are as follows:

(1) $Boron^{10}$ - stainless steel (2.5 wt % B^{10}).

(2) Hafnium (Hf metal).

(3) Cadmium-indium silver (5, 15, and 80 wt % respectively).

(4) Europium oxide - stainless steel (30 wt % Eu_2O_3).

(5) Gadolinium oxide - samarium oxide - stainless steel (15 wt % Gd_2O_3 and 15 wt % Sm_2O_3).

The rod absorber thickness for all cases was 0.25 inch (0.625 cm).

The nuclear characteristics of the control rod poison materials and results of burnup calculations described below are summarized in Table III-1.

The initial thermal and epithermal (0.0322 to 2.439 x 10^4 ev) macroscopic absorption cross sections for the different rods are given in columns 4 and 5. The epithermal cross sections were weighted by the average energy-dependent fluxes in the rod before being reduced to an epithermal constant, i.e.:

$$\Sigma_a = \frac{\int \Sigma_a (E) \, \phi (E) \, dE}{\int \phi (E) \, dE}$$

A measure of the "blackness" of a rod is obtained by calculating the fraction of incident flux transmitted through a thickness by the expression:

$$\frac{I}{I_o} = e^{-\Sigma_a x}$$

If $\Sigma_a x$ =	3	4
$e^{-\Sigma_a x}$ =	0.05	0.02
For x =	0.625 cm	0.625 cm
Σ_a must =	4.72	6.30

All of the rods are essentially black to thermal neutrons.

Two different methods were used for evaluating rod burnup. In both methods, however, only the thermal cross sections were considered. Essentially, the first method consisted of calculating the concentration of the absorbing isotope at time t from the expression:

TABLE III-1

1	2	3	4	5	6	7	8	9	10	11	12
				Initial Σ_a (cm^{-1})				K_{eff} = 1.04		K_{eff} = 1.08	
Rod	Important Isotope	$\sigma_{a_{th}}$ (Barns)	Epithermal (0.032 to 2.4 x 10^4 ev)	Thermal	$\bar{\phi}$ Poison / $\bar{\phi}$ Core	$e^{-\sigma_a g \phi t}$	$\Sigma_{a_{th}}$ at t = 2 yr	$\frac{N}{N_0}$	$\Sigma_{a_{th}}$ at 2 yr	$\frac{N}{N_0}$	$\Sigma_{a_{th}}$ at 2 yr
A	B-10	3,470	0.902	40.12	0.012	0.978	39.24	0.99	39.77	0.98	39.41
B	Hf-177	320	0.602	4.18	0.117	0.981	4.10	0.99	4.12	0.98	4.07
C	Cd-113	18,000	-	-	-	0.675	-	0.69	9.05	0.37	6.61
	In	165	-	-	-	0.996	-	0.99	11.47	0.97	11.45
	Ag-107	26	0.601	11.54	0.043	0.999	8.95	0.99	11.47	0.98	11.45
	Ag-109	71	-	-	-	0.998	-	0.99	11.48	0.99	11.47
D	Eu-151	7,800	0.874	24.80	0.020	0.924	22.94	0.97	24.02	0.94	23.22
E	Gd-155	60,600	-	-	-	0.889	-	0.77	105.08	0.53	80.49
	Gd-157	139,000	0.475	129.54	0.004	0.763	102.09	0.78	106.64	0.56	83.62
	Sm-149	57,200	-	-	-	0.894	-	0.76	123.95	0.52	118.24

$$\frac{N}{N_o} = e^{-\sigma_a g \phi t}$$

where

$\frac{N}{N_o}$ = fraction of concentration remaining after time t (column 7, Table III-1)

$g = \frac{\phi \text{ poison}}{\phi_{core}}$ = ratio of average flux in the poison to average core flux (colume 6, Table III-1)

σ_a = thermal microscopic absorption cross section

ϕ = average core thermal flux = 8.0×10^{12}

Then, the macroscopic thermal absorption cross sections were re-calculated for t = 2 years. These values are shown in column 8, Table III-1.

In the second method, the concentration of absorbing nuclei at time t = 2 years was calculated by subtracting from the initial concentration, one nucleus for each neutron absorbed. The number of excess neutrons that must be absorbed was calculated by multiplying the number of fissions required for 2 years of operation at full power by the excess number of neutrons that must be absorbed. Two cases were considered: (1) an average K_{eff} at operating condition of 1.04 (0.04 excess neutrons per fission) and (2) K_{eff} = 1.08. The ratios of final-to-initial concentrations of the absorbing isotopes are given in columns 9 and 11 of Table III-1. The new $\Sigma_{a_{th}}$ are given in columns 10 and 12. For rods "C" and "E," multiple values of σ_a are given. Each value represents the rod σ_a, making the assumption that all of the absorptions are due to that particular isotope.

Based on nuclear considerations described above, any of the rod compositions evaluated are suitable. The following simplifying assumptions were made in performing the calculations:

(1) No account was taken of high cross section daughter formation.

(2) Each isotope in a mixture was assumed to be the sole absorber.

(3) The complete rod system was used to absorb the neutrons.

Items (1) and (2) are conservative, since in some cases daughters do form and in mixtures each isotope is effective to some degree. Item (3) is not conservative, since only part of each rod will be inserted during operation. A more detailed analysis of the samarium-gadolinium system indicates that the effects of Item (3) are approximately counterbalanced by Items (1) and (2).

As indicated, this study was preliminary in nature. More detailed analyses considering epithermal absorptions, spatial burnup in the rod and iteration-type burnup will be made.

Temperature coefficient studies.- The reactivity of the core was calculated for different temperatures from 68 to 473° F. Both nuclear and density temperature effects were considered. Specifically, the change in reactivity with temperature was assumed to be due to:

(1) The change in microscopic thermal cross sections with temperature.

(2) The change in reflector savings resulting in a change in the buckling with temperature.

(3) The change in density of water with temperature. Core materials other than water were assumed to have a constant density in the temperature range studied.

The reactivity as a function of temperature is shown in Fig. III-5. The average temperature coefficient from 68 to 463° F is $-1.14 \times 10^{-4}\ \Delta\rho/°F$. At operating temperature, the temperature coefficient is $-2.84 \times 10^{-4}\ \Delta\rho/°F$.

Reactivity was calculated using a 3-group diffusion code, Program C-3. The thermal disadvantage factors, calculated using Program I-2 and reflector savings (both calculated as a function of temperature) were used to account for heterogeneity and reflector effects.

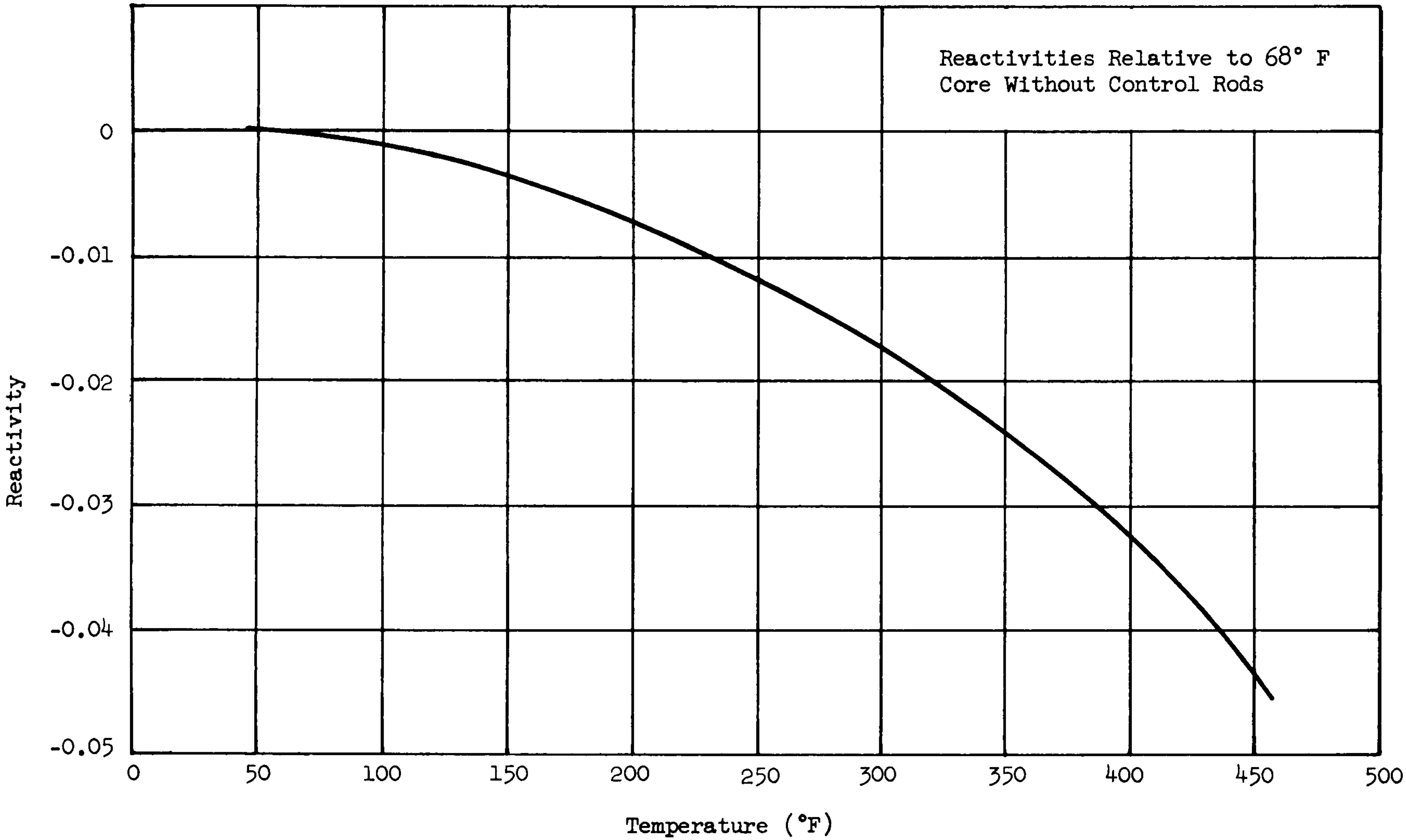


Fig. III-5. Reactivity as a Function of Temperature

Void coefficient studies.- The effect of voids on reactivity was calculated for 0.5, 1, 1.5, and 2% void in the moderator. For these studies, a uniform void distribution was assumed and the void formation was represented by a change in volume of the moderator. The reactivity (relative to 68° F and no void) as a function of temperature for different void fractions is shown in Fig. III-6.

Flux and power distributions.- Flux and power distributions obtained during the preliminary design studies include:

(1) Relative radial and axial power density distribution for the core without control rods as a function of time (Figs. III-7 and III-8).

(2) Relative radial and axial thermal fluxes for the core without control rods as a function of time (Figs. III-9 and III-10).

(3) Radial flux distribution for the core, with 4, 5 and 6 control rods fully inserted (Figs. III-11, III-12, and III-13).

(4) Radial flux distributions near the control rod water channels with the rods fully withdrawn.

Flux and power distributions for the core without control rods, as a function of time, were calculated using 2-group multiregion one-dimensional diffusion theory.

Radial flux distributions for the core with control rods inserted were calculated using the two-dimensional code "PDQ."

Radial flux distributions near the control rod water channels were also obtained using "PDQ." The case evaluated was for cruciform rod channels and are not presented here. However, results did indicate that there may be some peaking in the fuel element near the center of the rod channel. If further analysis indicates that the flux peaking is excessive, the fuel element will be replaced by a stainless steel rod (or a lumped poison rod, if this does not adversely affect the rod worth).

Experimental axial flux distributions with control rods partially inserted are available from previous studies. These will be evaluated later in detail.

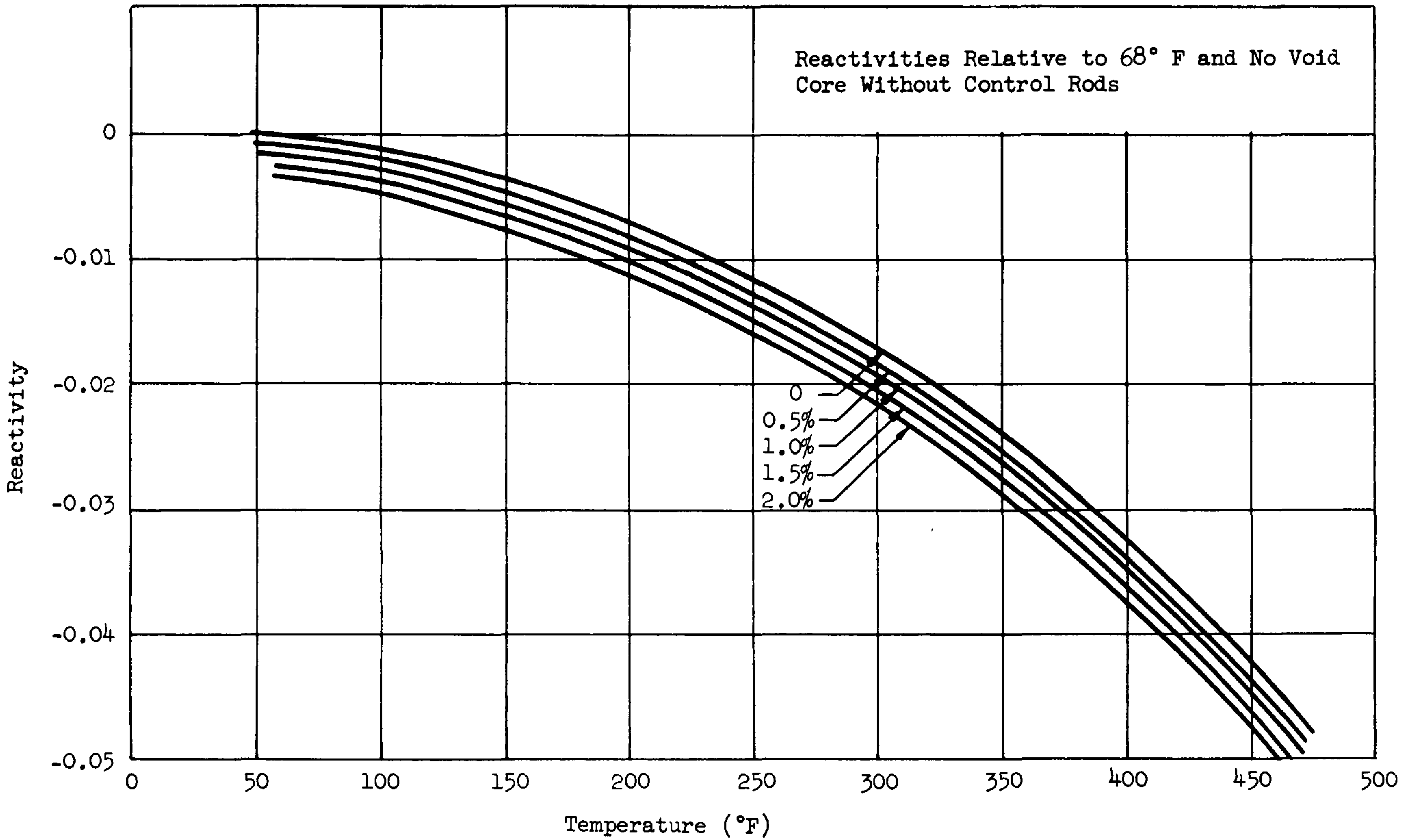


Fig. III-6. Effect of Voids on Reactivity

Power Density Distributions Relative to Average in Core
Core: Without Control Rods
Temperature: 463° F

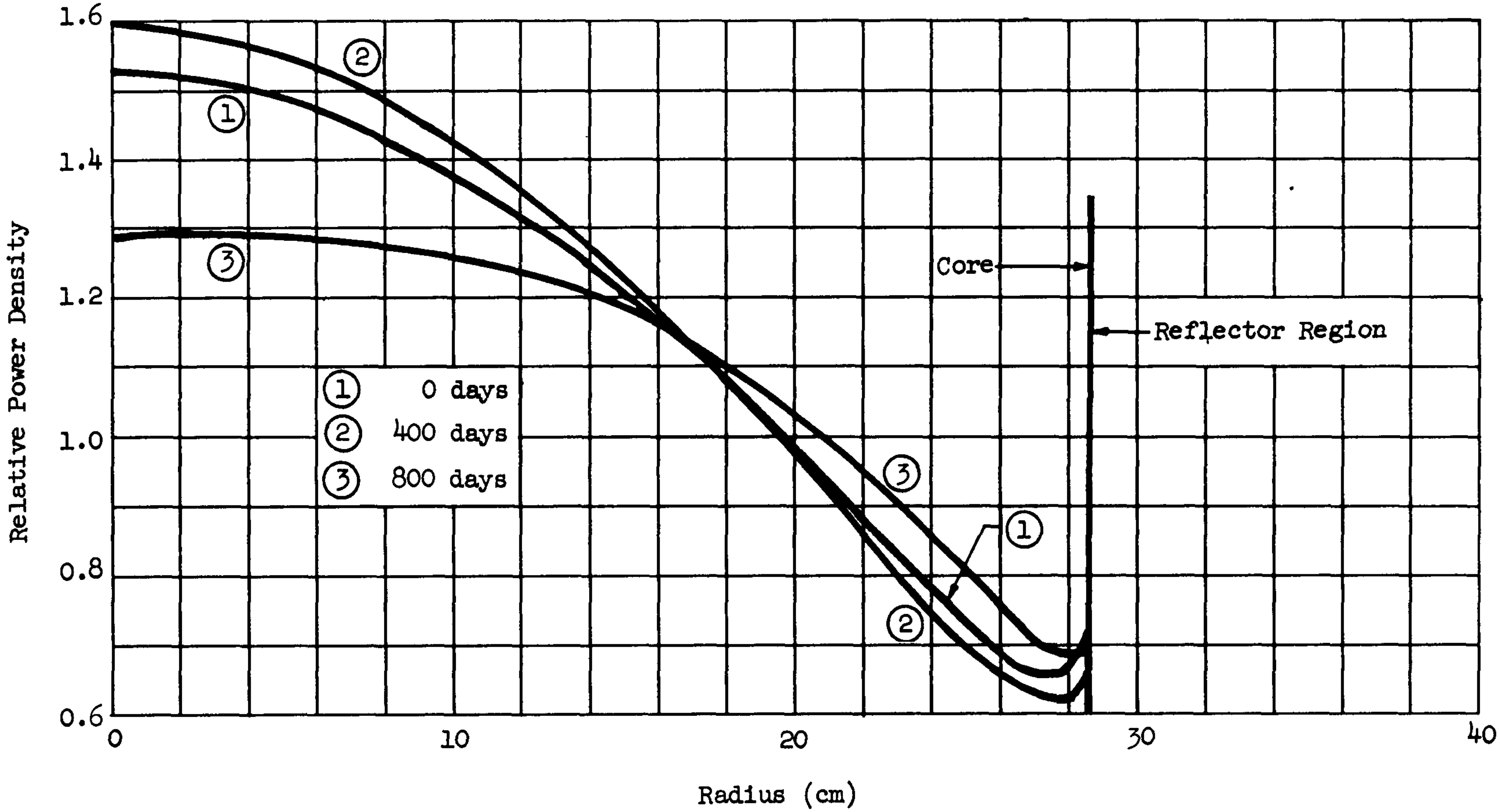


Fig. III-7. Relative Radial Power Density Distributions with Time

Power Density Distributions Relative to Average Power Density in Core
Core: Without Control Rods
Core Temperature: 463° F

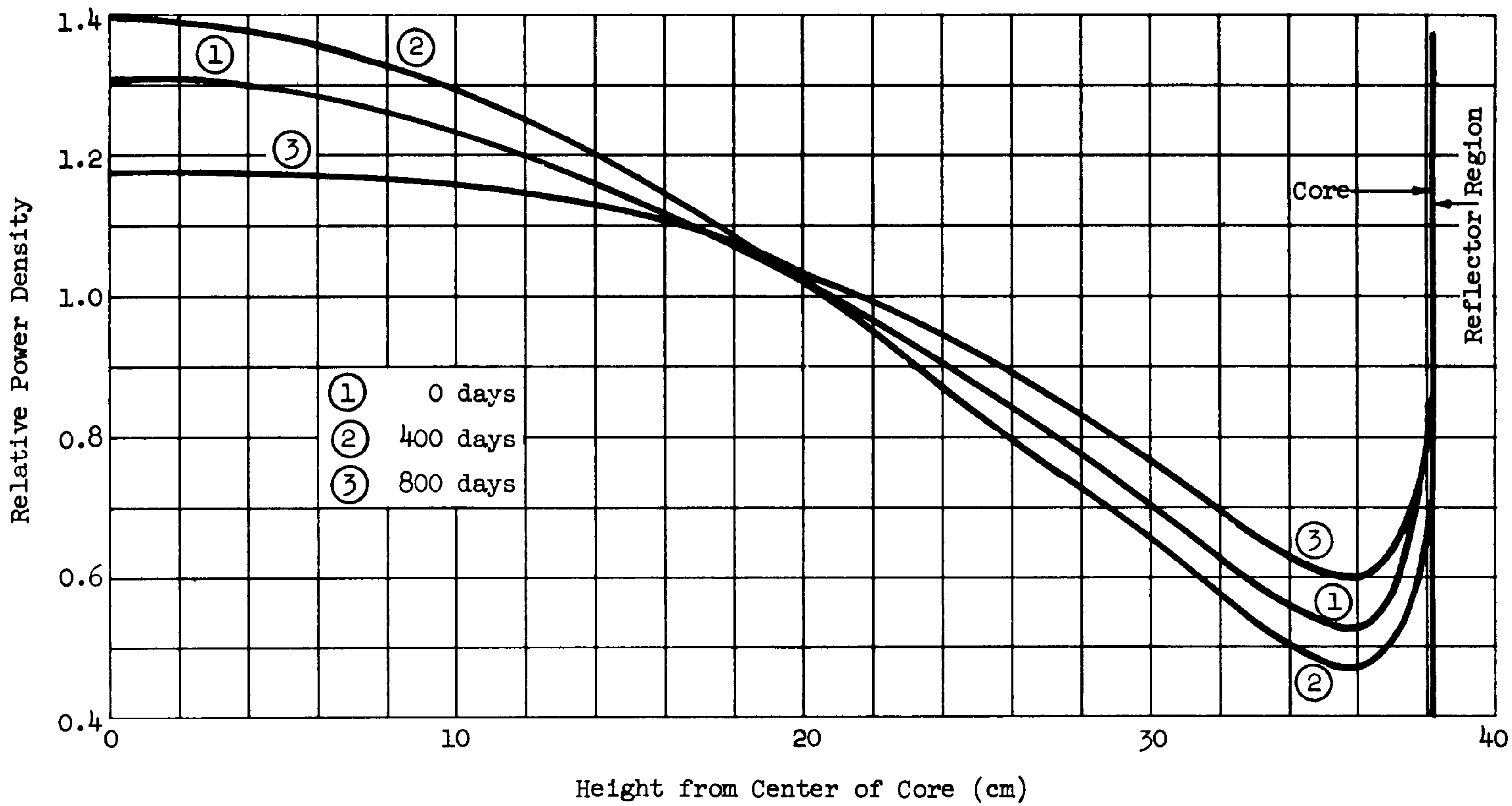


Fig. III-8. Relative Axial Power Density Distributions with Time

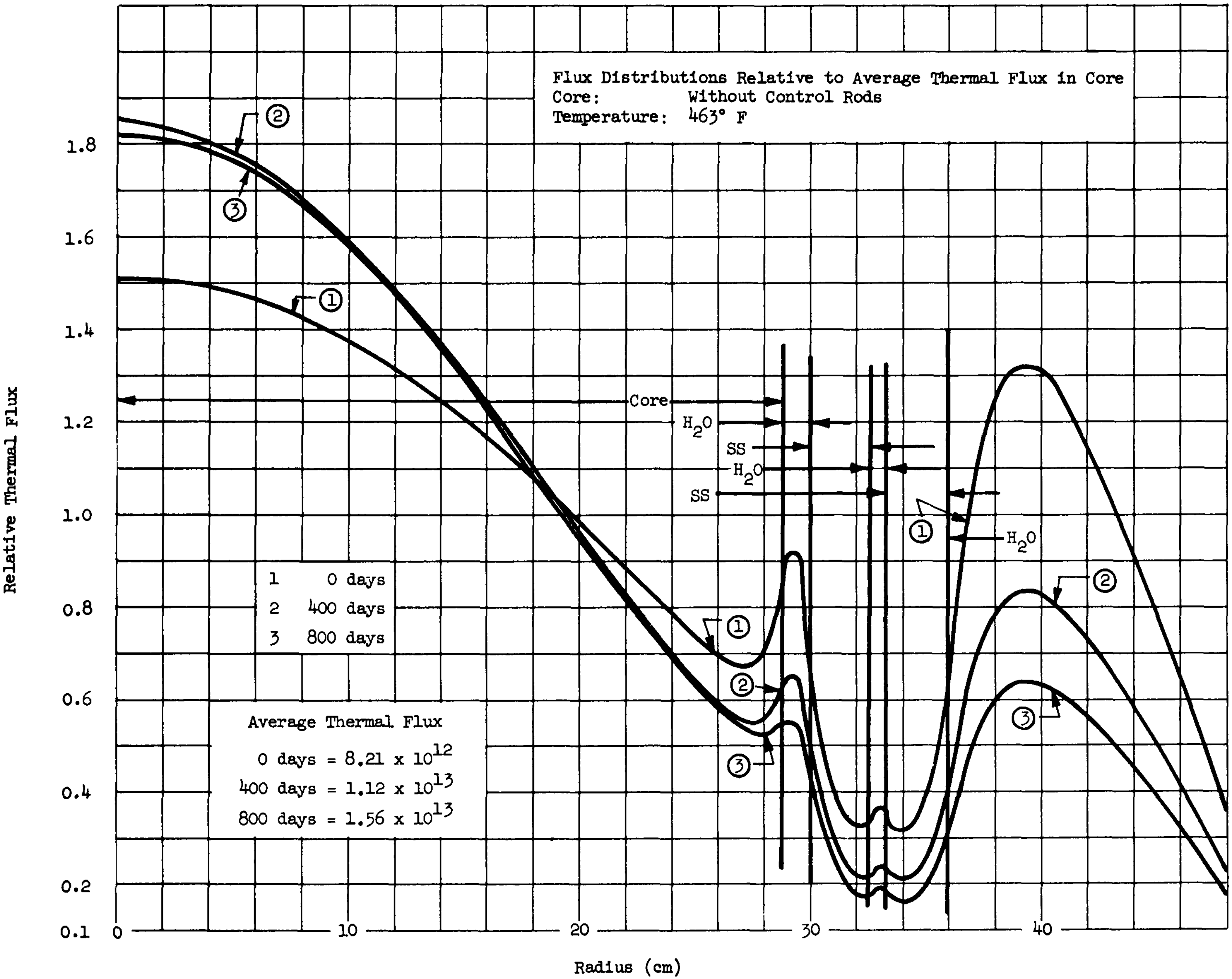


Fig. III-9. Relative Radial Thermal Flux Distributions with Time

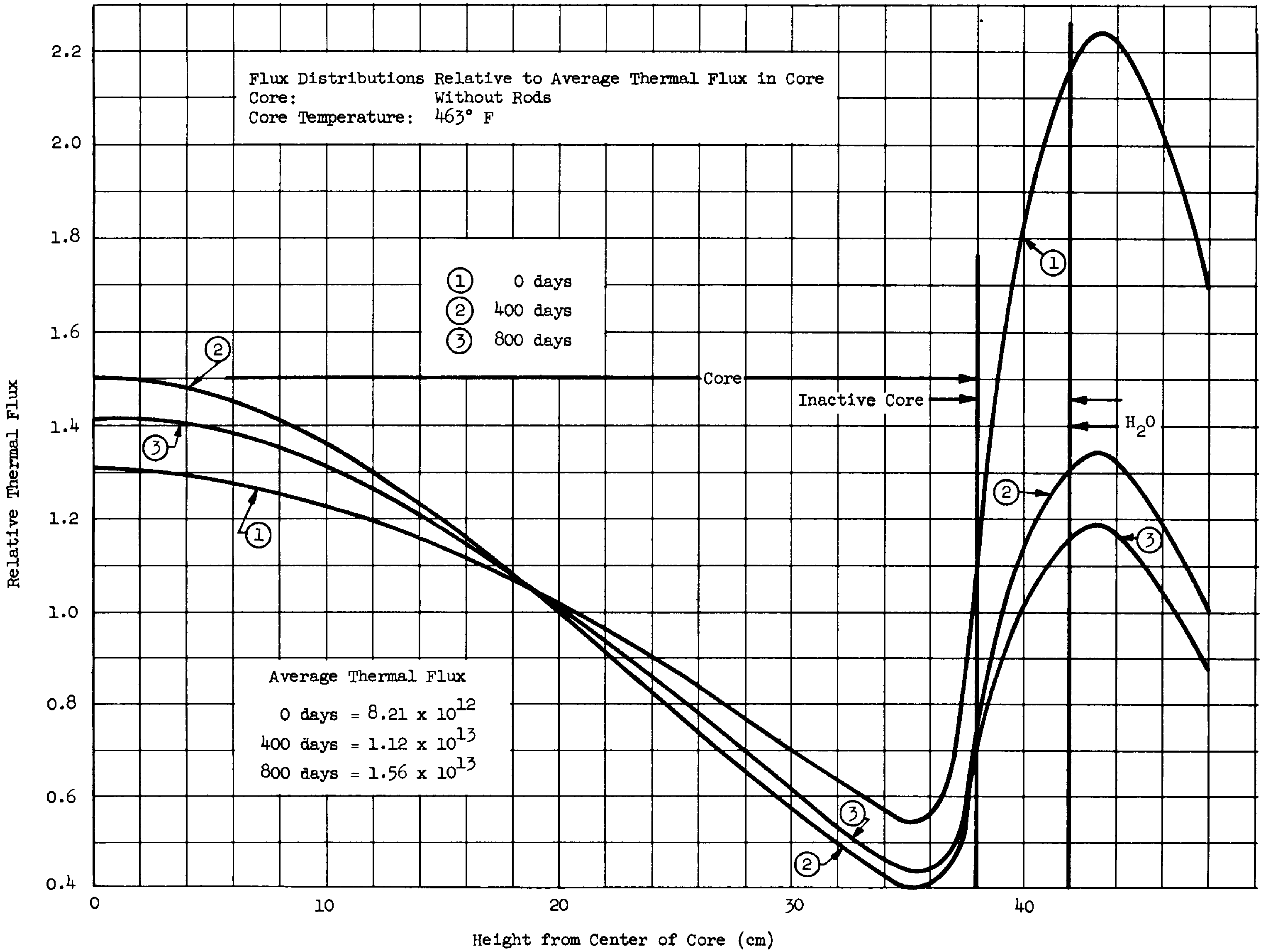


Fig. III-10. Relative Axial Thermal Flux Distributions with Time

NOTE:

Flux values relative to average thermal flux in core

Decimal points represent the location of the flux value

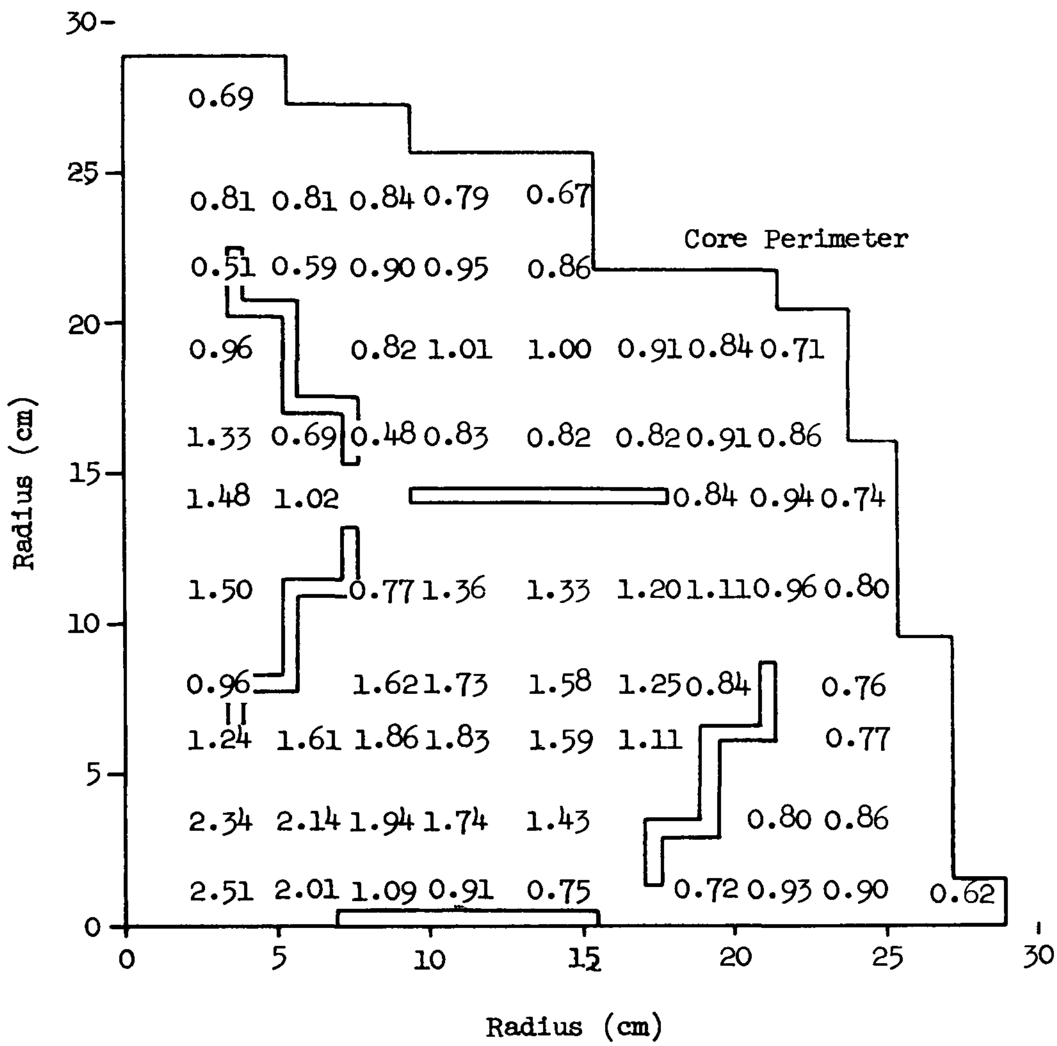


Fig. III-11. Thermal Radial Relative Flux Distribution with Six Control Rods Fully Inserted

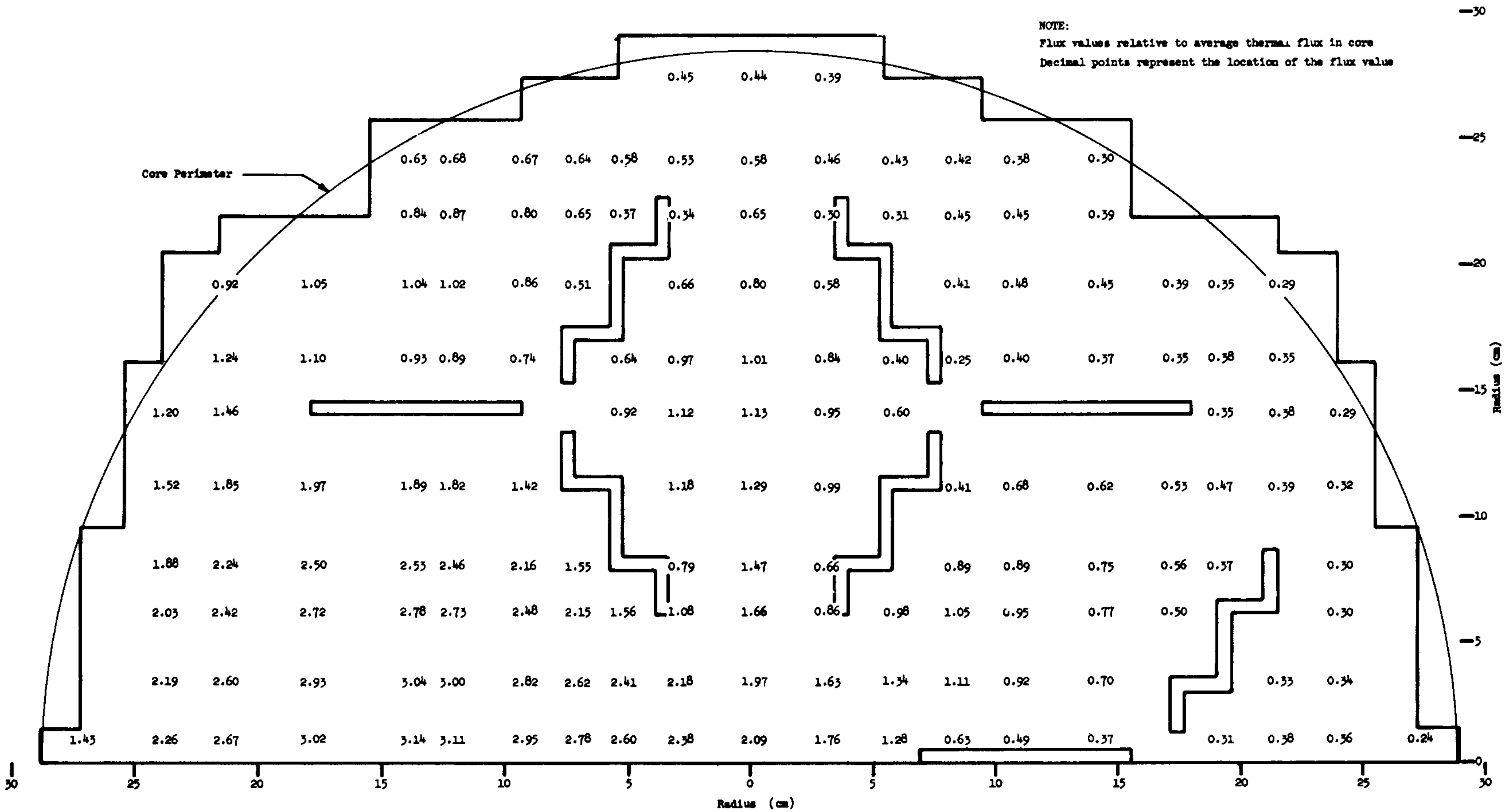


Fig. III-12. Thermal Radial Relative Flux Distribution with Five Control Rods Fully Inserted

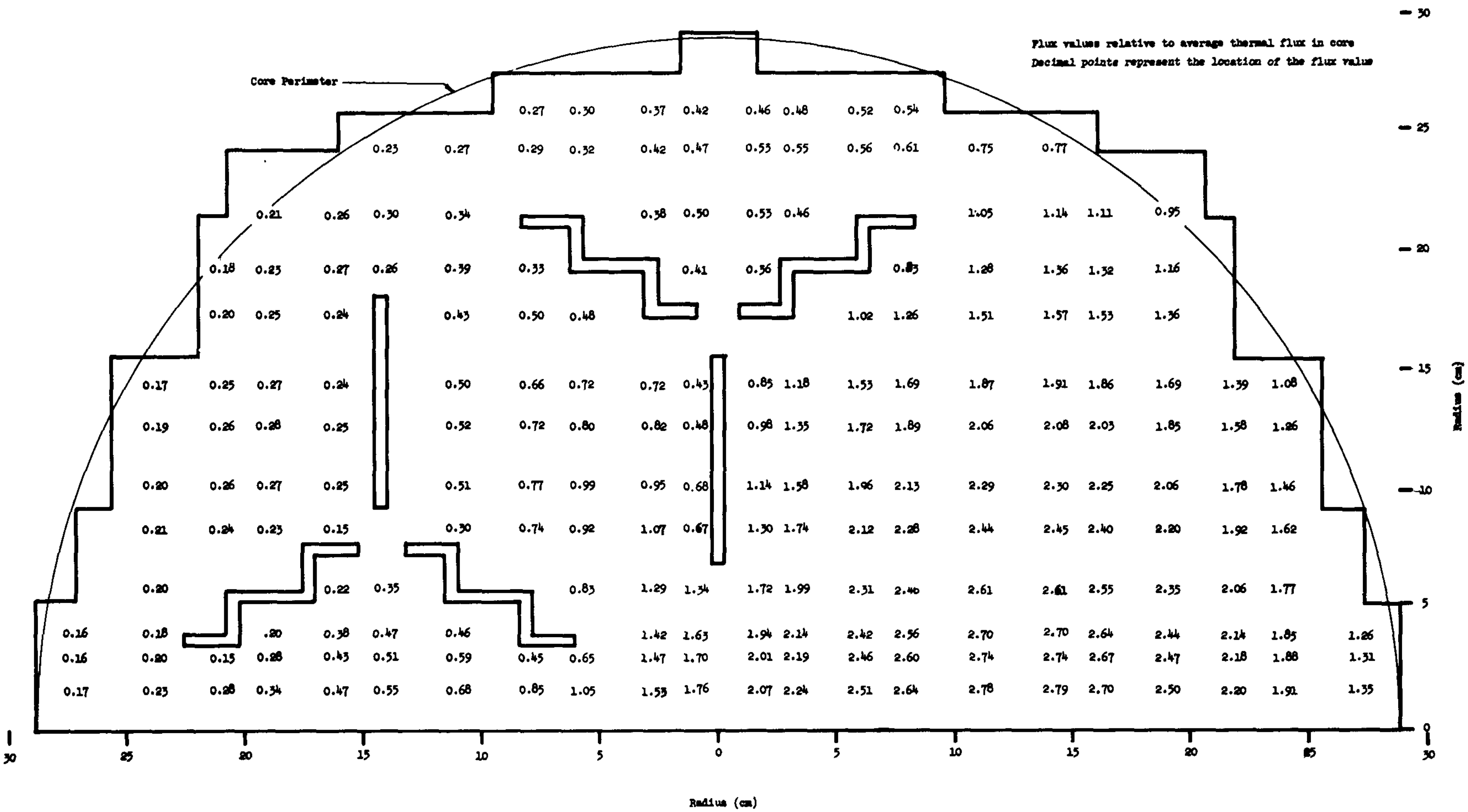


Fig. III-13. Thermal Radial Relative Flux Distribution with Four Control Rods Fully Inserted

2. Heat Transfer and System Analysis Studies

R. Baer, A. Carnesale, C. Smith, J. Beam

The studies performed during the past quarter satisfied the general objectives of checking the digital computer code for the thermal and hydraulic analysis of local boiling cores and of performing necessary heat transfer and system analyses in support of preliminary design.

Digital computer code.- The steady-state thermal and hydraulic analysis code (BUBBLES-1) was successfully checked out for nonboiling and local-boiling analysis of tubular fuel elements.

Code checkout was made difficult by its extreme length and complexity. In order to comply with the tabular limitations of the FORTRAN coding system, the program had to be written and assembled in three parts. The resultant machine language programs were then hand-linked. This complicated the tracing of errors during normal "debugging." The code was considered checked out for a particular type of analysis when substitution of the machine solutions into the previously derived analytical equations satisfied these equations. The code was checked for the thermal and hydraulic analysis of tubular fuel elements operating in nonboiling and local boiling prior to using it in preliminary design investigations. It was used to determine the axial and radial temperature distributions of the fuel element, as well as the axial distributions of coolant temperature, pressure, velocity, enthalpy, heat flux, saturation temperature, and film coefficient.

Heat transfer analysis--summary.- The parametric study of the PM-1 Power Plant established the following reactor criteria:

(1) Local boiling will be permitted during steady-state operation

(2) Primary loop operating pressure will be 1300 psia

(3) Primary loop mean temperature will be 463° F

(4) Primary loop flow rate will be 1900 gpm

(5) Reactor power will be approximately 9.35 Mw.

The following areas were investigated in support of preliminary design:

(1) Reactor power required

(2) Flow rate per element

(3) Proportion of flow inside and outside of the elements

(4) Orifice requirements

(5) Core temperature distributions

(6) Stability

(7) Control rod coolant flow requirements

(8) Control rod shroud requirements

(9) Total core flow rate

(10) Hot channel factors

(11) Reactor pressure drop

(12) After-heat dissipation

(13) Earth shield temperature distribution

(14) Spent core cask heat removal system.

As a result of the analyses performed in support of the preliminary design work, the following conclusions were reached:

(1) No local boiling occurs in a fuel element generating the average amount of power and receiving the average amount of coolant flow.

(2) A fuel element generating more than the average amount of heat receives slightly more than the average amount of flow.

(3) The coolant flow rate of 1900 gpm, specified as a result of the parametric study, is slightly more than that required to establish the desired thermal margin of safety to prevent bulk boiling.

(4) The use of shrouds or baffles to isolate flow in the control rod passage is not warranted economically and would add significantly to the complexity of the core.

(5) The primary loop arrangement provides sufficient natural convection to adequately handle after-heat resulting from the decay of fission products.

Reactor power requirement.- The heat losses in the primary loop were calculated and are tabulated below:

(1)	Pipe insulation (1.5 inches)	4.2 kw
(2)	Steam generator insulation (1.5 inches) (primary loop)	0.7 kw
(3)	Reactor vessel insulation (1.5 inches)	4.0 kw
(4)	Pump seal leakage (2 gph at 32° F make-up)	1.7 kw
(5)	Demineralizer cooler	30.0 kw
	Total	40.6 kw

The energy input at the shaft of the primary pump of 38 kw essentially equals the primary loop heat losses. Thus, the required reactor power is that which need be delivered to the steam generator (9.35 Mw).

Flow rate per element.- The mean temperature, flow rate, and reactor power specified by the results of the parametric study establish a reactor inlet temperature of 445° F. The arbitrary, conservative, criterion used in establishing the flow rate per element was that a 50% power transient shall be required to cause bulk boiling at the outlet of the hottest channel (calculations assume a system pressure of 100 psi less than the normal operating pressure, and a hot channel generating twice the average power). A flow rate of 2.27 gpm per element at the core inlet was found, using BUBBLES-1, to satisfy these criteria.

Proportion of flow inside and outside of the elements.- The thermal margin of safety regarding the prevention of bulk boiling is a function of how far the coolant temperature is below saturation temperature at the exit of the hot channel. The most favorable situation also occurs when the coolant exit temperatures from the inside and the outside of the hottest element are equal.

BUBBLES-1 was used to determine the required rates of flow inside and outside of the elements. Various values of the ratio of flow inside of the element to the total element flow were used as inputs to the code. The outlet coolant temperature both inside and outside of the element were plotted against this ratio. The results are shown in Fig. III-14 for triangular pitches of 0.650 and 0.695 inch; a value of 0.480 was established for preliminary design.

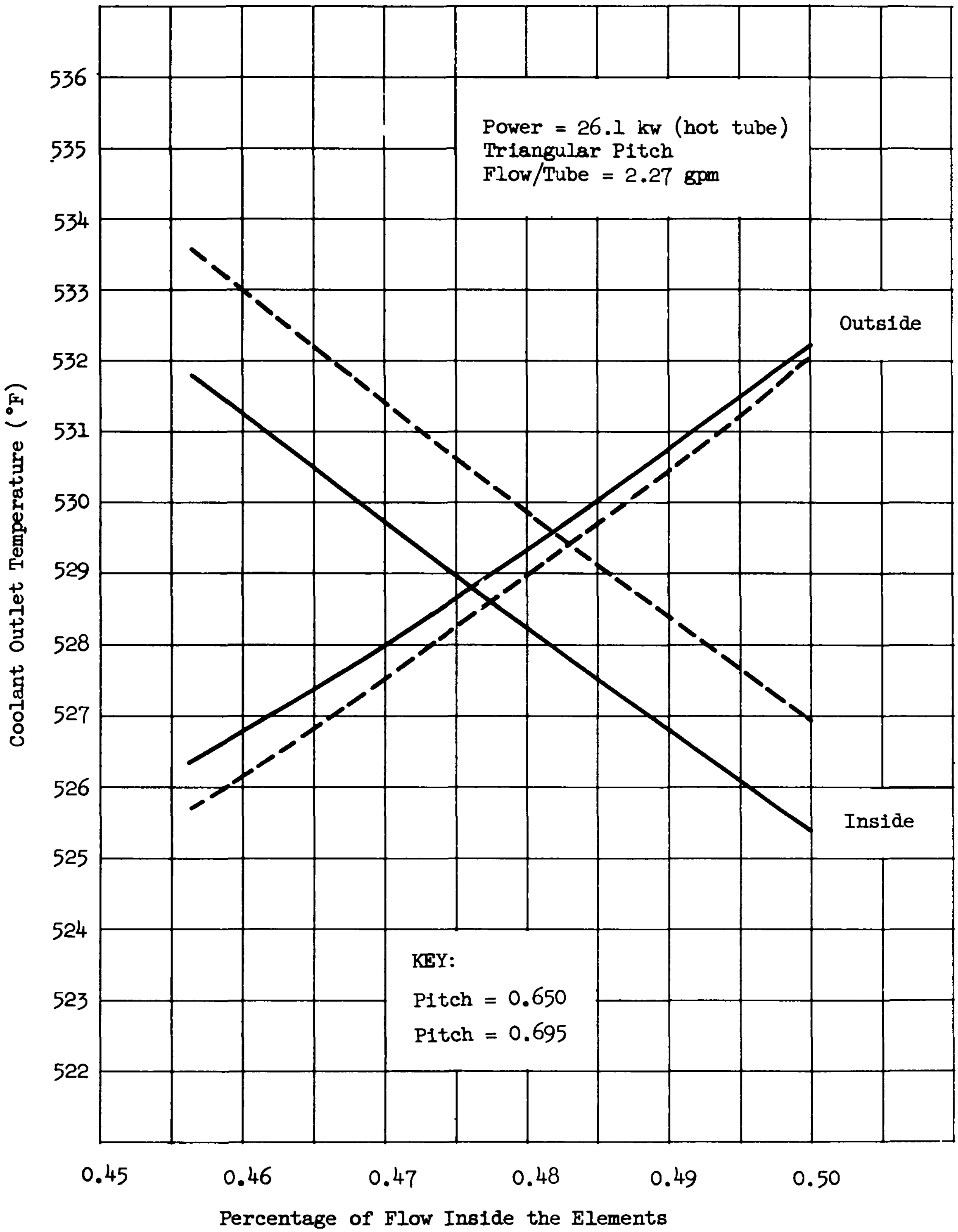


Fig. III-14. Coolant Outlet Temperature versus Inside to Total Flow Ratio

Orifice requirements.- The PM-1 core is orificed at the core inlet to obtain the proper distribution of flow between the inside and the outside of the fuel elements, and to provide high enough flow resistance to make the flow rate relatively insensitive to those changes in the heat transfer characteristics of the fuel elements which could result from different modes of operation.

The lower fuel element dead end is swaged to a smaller diameter and fitted into holes provided in the lower grid plate. The reduced diameter of the element forms the orifice which controls the flow rate inside of the element.

Additional holes are drilled in the lower plate to provide coolant flow outside of the elements. Standard pressure drop correlations were employed to determine the drops across the inlet and exit constrictions, and along the fuel element length. It was found that 0.200-inch orifices are small enough to cause a pressure drop sufficient to make the flow rate relatively insensitive to changes in heat transfer conditions along the outside of the fuel elements, yet large enough so that manufacturing tolerances and reasonable amounts of erosion and corrosion will not significantly alter their hydraulic characteristics.

The total head loss experienced by the flow inside the elements must be the same as that outside of the elements since the coolant in each flow path has a common inlet and outlet. An element orifice ID of 0.265 inch, after swaging, was calculated to be necessary if the required head loss at the entrance of the fuel element was to be provided.

Core temperature distributions.- All temperature distributions in the core were obtained using the BUBBLES-1 code. Slight alterations of the fuel element dimensions were made at the end of preliminary design. However, these variations will not significantly affect the element thermal and hydraulic behavior set forth below.

Temperature distributions in and along various elements are shown in Figs. III-15, III-16, III-17, and III-18. It was shown that local boiling does not occur in a fuel element producing an average amount of power and receiving an average flow rate. The axial variation of the surface and coolant temperatures of such an element are shown in Fig. III-15. The temperature distributions through the tube wall at various axial locations are shown in Fig. III-16. Although present data indicates that no fuel element produces as much as twice the average power, for conservatism, such a hypothetical element was analyzed--about two thirds of the element would be in local boiling. The maximum fraction of burnout heat flux is 21% as determined by the Jens and Lottes correlation of burnout heat flux-to-flow rate, pressure, and degrees of subcooling. The axial variations of surface and coolant temperatures in such an element are shown in Fig. III-17. The temperature distributions through the tube wall at various axial locations are shown in Fig. III-18.

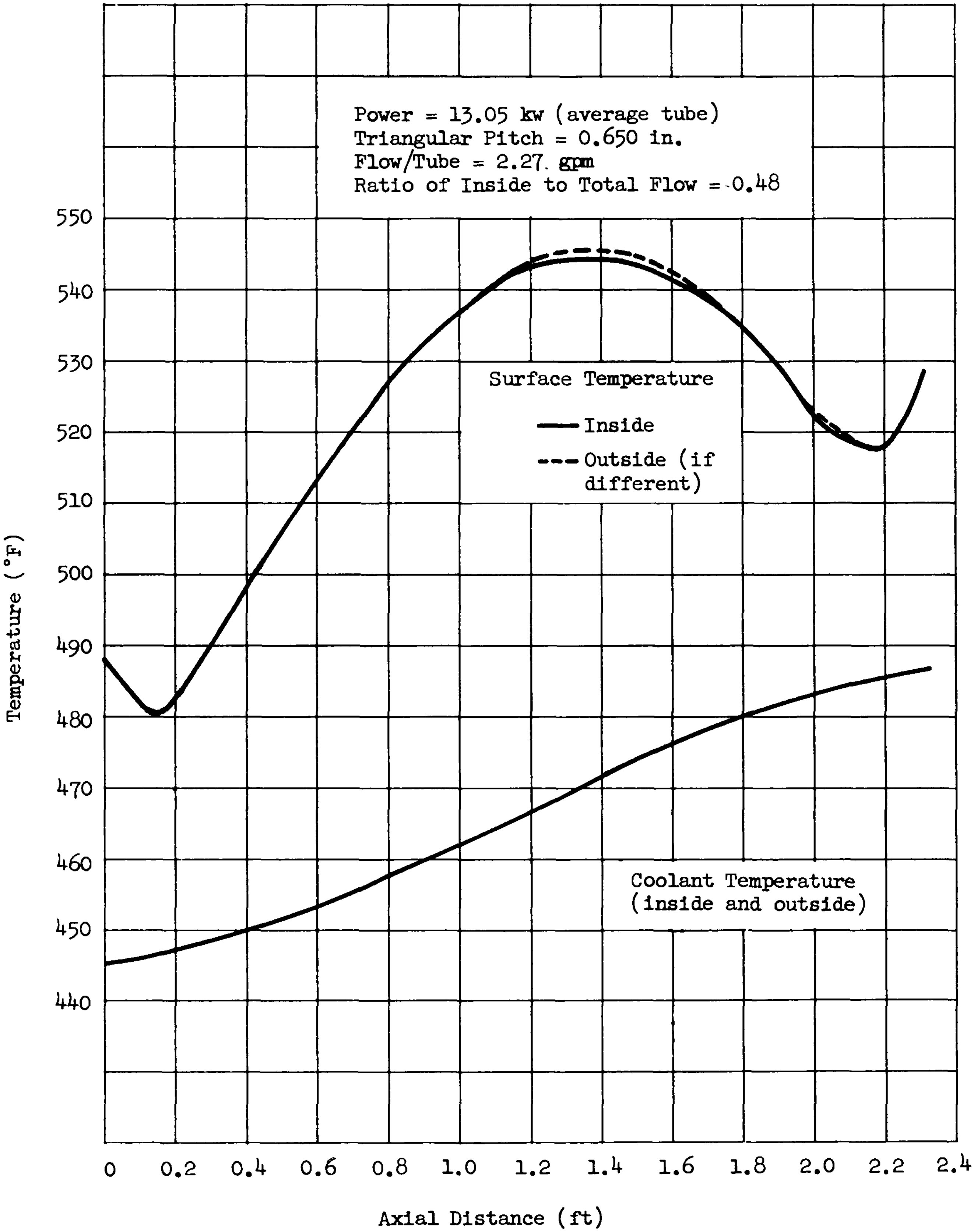


Fig. III-15. Fuel Tube Surface Temperatures and Bulk Coolant Temperatures

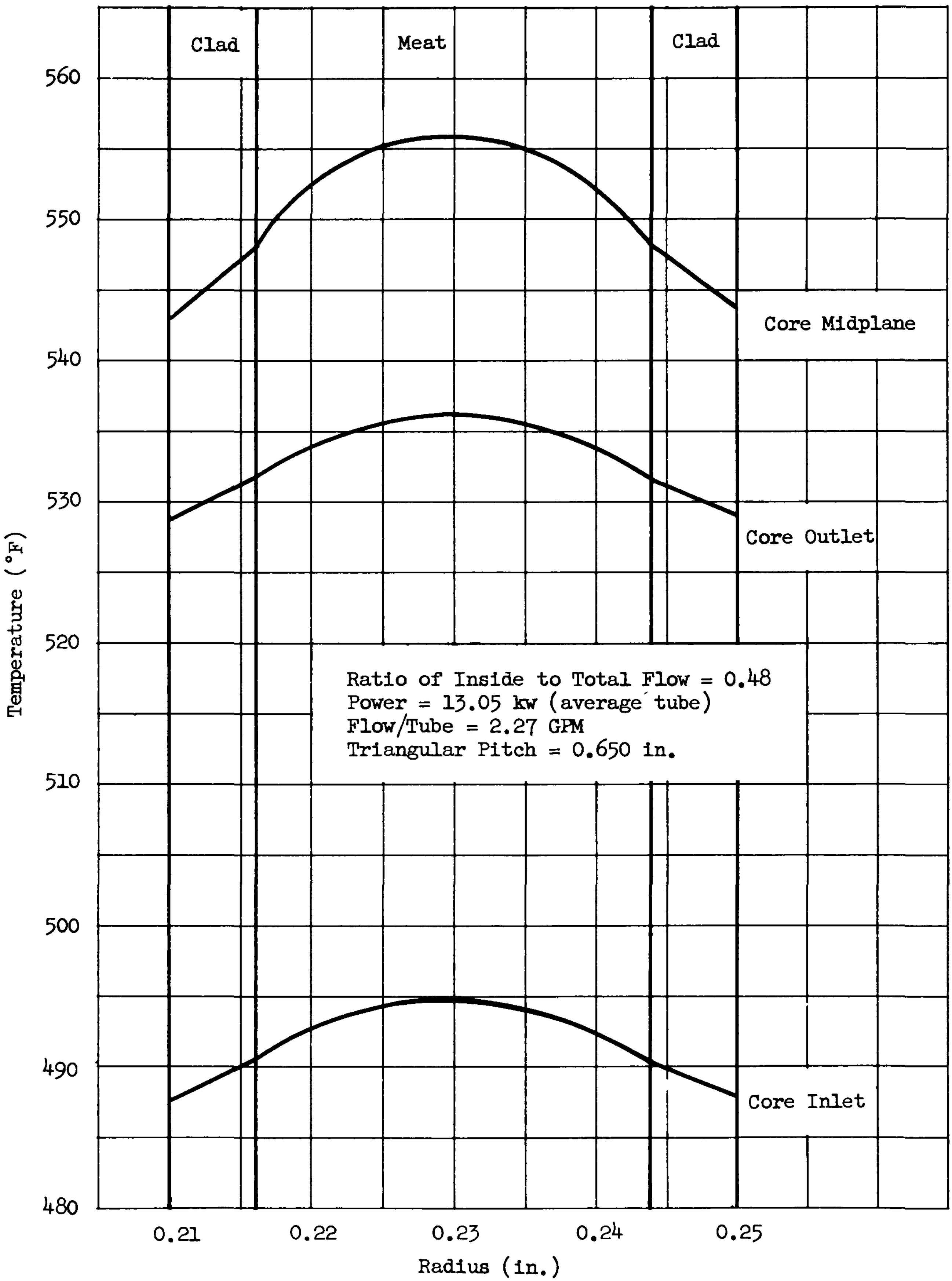


Fig. III-16. Radial Fuel Tube Temperatures

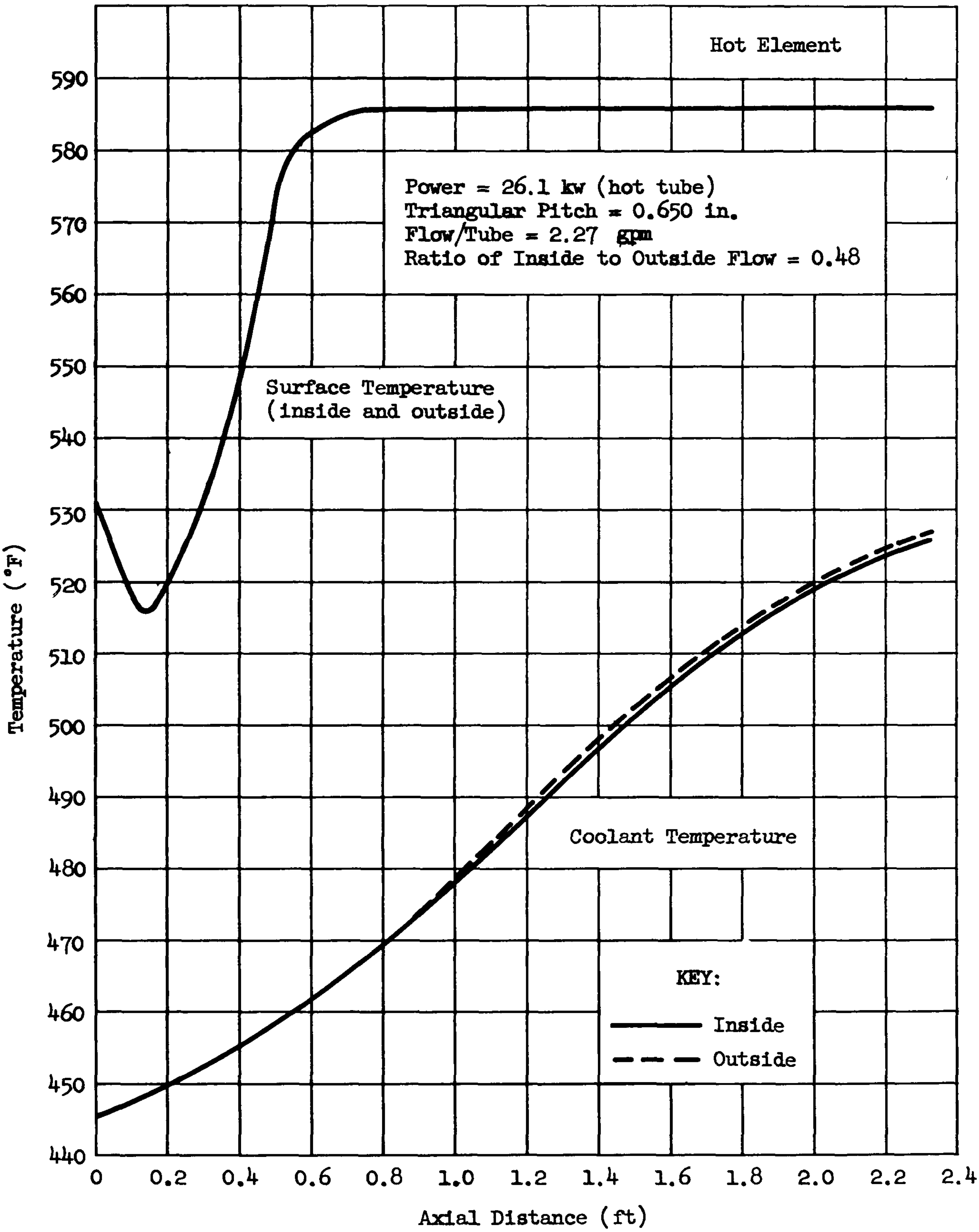


Fig. III-17. Fuel Tube Surface Temperatures and Bulk Coolant Temperatures

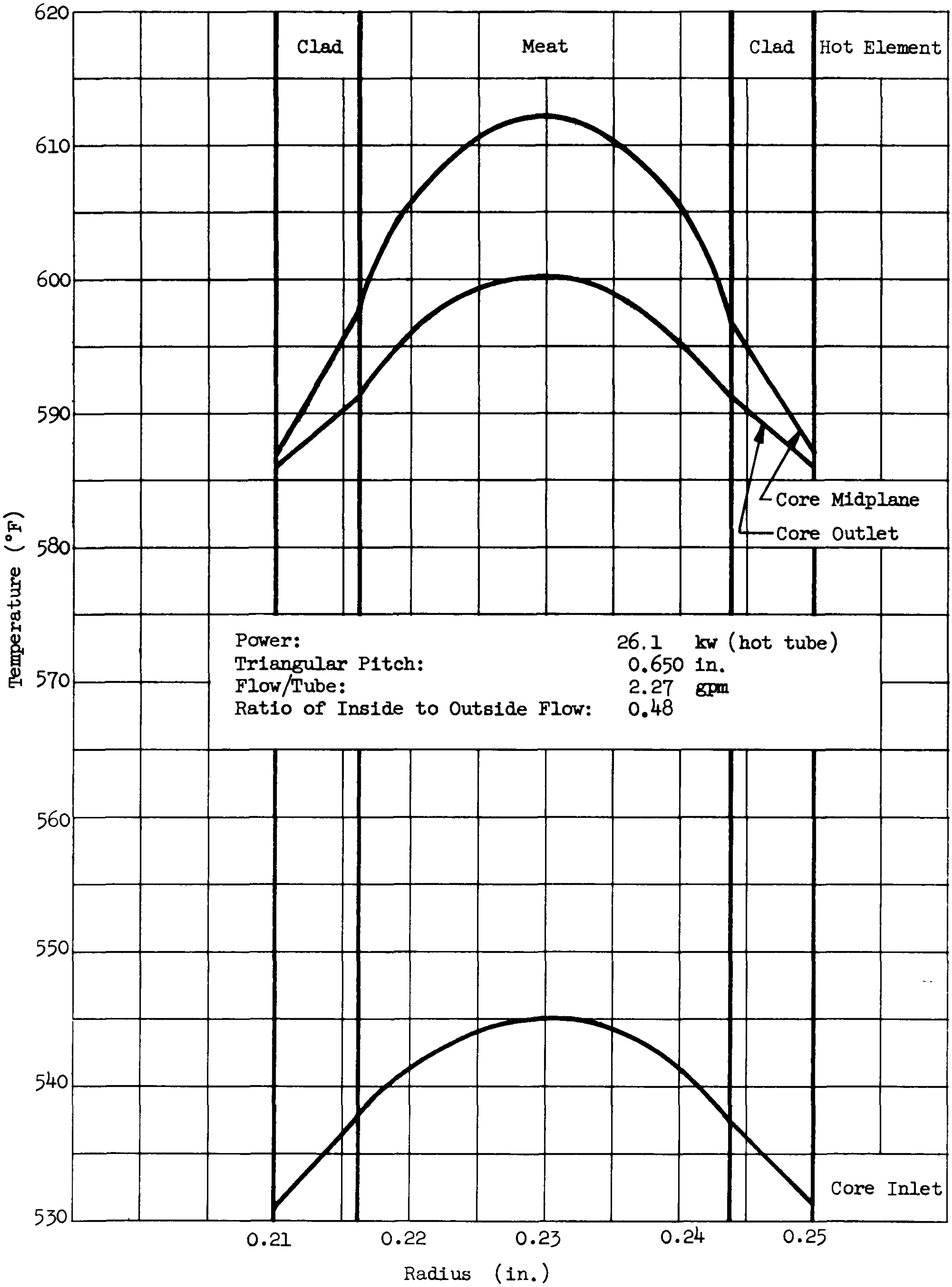


Fig. III-18. Radial Fuel Tube Temperatures

Stability.- Although the increased frictional pressure drop associated with local boiling heat transfer tends to decrease the flow rate in the hotter elements of the core, the higher-than-average heat flux necessary to cause local boiling also increases the bulk temperature of the coolant in these channels; this reduces the average coolant density. This latter effect results in a lower elevation head loss at the hotter elements; this tends to increase the coolant flow rate.

The fuel element dimensions employed in this analysis also differed slightly from those selected at the end of preliminary design, the conclusions reached, however, are still applicable.

The analysis was performed using BUBBLES-1 and temperature-dependent coolant properties. The flow-increasing effect of the density change was found to be more significant than the flow-decreasing effect of the local-boiling friction factor for a tube producing twice the average power. In order for the pressure drop across this hot tube to be equal to that of an average tube (which must be the case), the hot-tube flow rate must be 2.29 gpm as opposed to 2.27 gpm through the average tube; this is shown in Fig. III-19.

This analysis indicates that increased local boiling in the PM-1 core is self-correcting from a fluid flow standpoint; that is, the hydraulics of the situation are quite stable, even ignoring the density effects of void formation.

Rod coolant flow requirements.- The passages for the control rods are formed by the removal of fuel elements from the core. The coolant in these passages flows axially upward and cools both the control rod blades and a portion of the fuel elements adjacent to the passage.

The approach used in the core design was to avoid, as much as possible, any pressure gradient which would result in radial flow of the coolant across the elements. This can best be accomplished by providing coolant flow adjacent to the control rods so that the pressure drop per unit length along these channels is identical with that outside of the elements. The holes in the bottom grid plate which supply this flow may be sized so that the desired flow is obtained. When a control rod is extracted, some radial flow in the direction of the vacant channel will occur. The extraction of the rod, however, results in a local flux peaking. The power produced by the fuel elements adjacent to the control rod channel will then increase, and the increased flow will aid in cooling these elements.

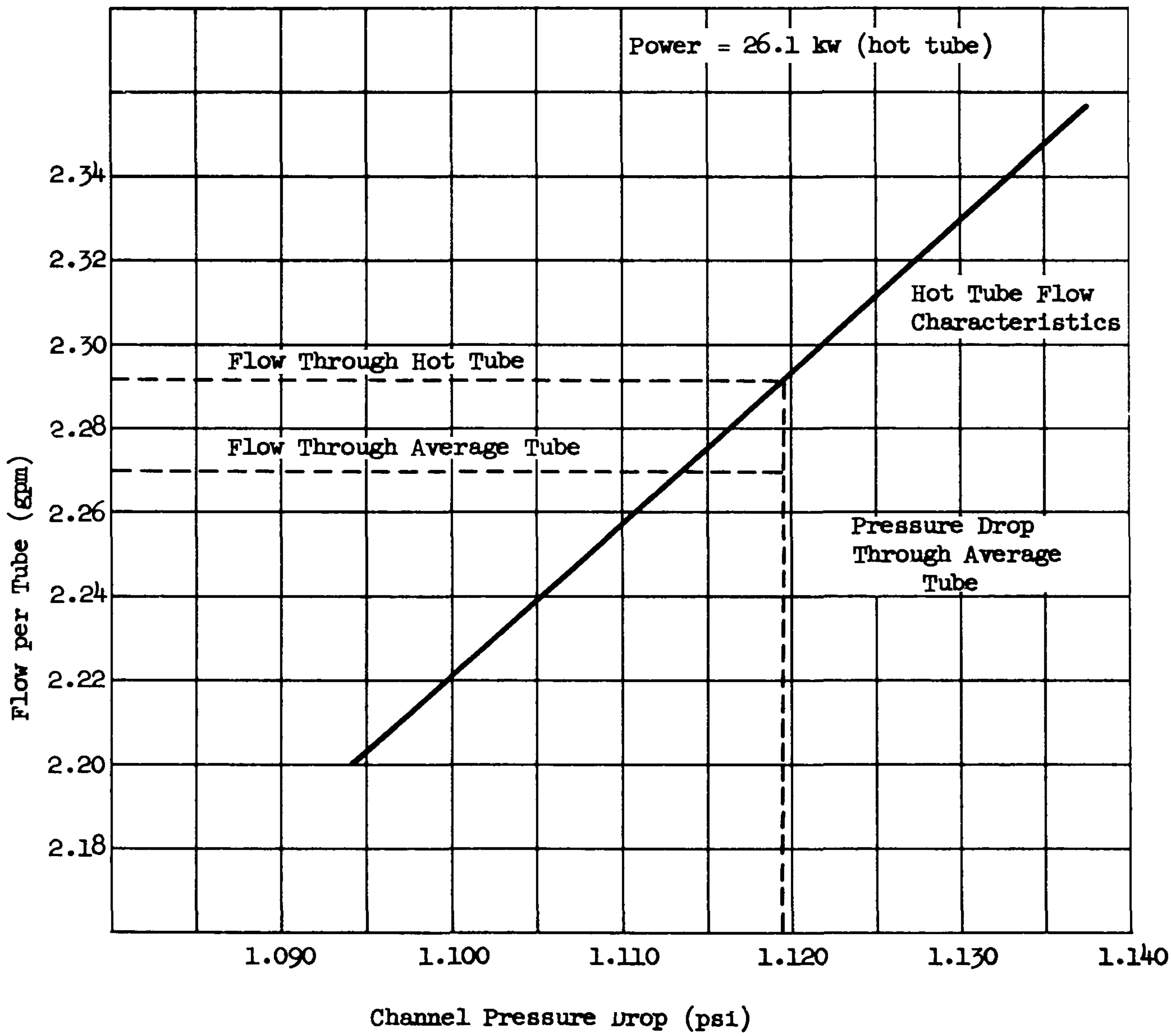


Fig. III-19. Hot Fuel Tube Flow Characteristics

With the rods fully inserted, a total of 212 gpm flows through the 6 control rod passages. (This calculation was based upon a triangular pitch of 0.665 inch and a control rod blade width of 0.312 inch.)

Control rod shroud requirements.- In general, the amount of coolant flow in the control rod passages is more than is needed for the removal of heat from the control rod and the adjacent fuel elements. The possibility of using a shroud or baffle to limit flow in the control rod passage to the amount necessary for heat removal was investigated. It was concluded that, for the control rod blade thickness (5/16 inch) currently contemplated, the use of shrouds or baffles is not warranted.

The analysis consisted of comparing the resultant cost of the primary pump with and without the use of shrouds around the control rods. The difference in pump cost is shown for various control rod blade thicknesses in Fig. III-20.

Total core flow rate.- In order to prevent radial flow in the core, the flow rate along the burnable poison rods was made equal to that outside of the fuel element even though a relatively small amount of heat is generated in these rods. The 100 burnable poison rods require a total flow of 118 gpm.

The flow in the control rod passage will be 212 gpm. In addition to cooling the control rod blades, this flow also removes heat from the outside of half of the 180 elements which are adjacent to the control rod passages. The remainder of the fuel elements require a flow rate of 1540 gpm.

The total coolant flow rate required is, therefore, 1870 gpm. The 1900 gpm specified by the parametric study will provide a slightly greater margin concerning bulk boiling.

Hot-channel factors.- In order to ensure that bulk boiling does not take place at any point in the core, the effect of known uncertainties in the analytical techniques and empirical correlations employed, and the effect of variations in design dimensions due to manufacturing tolerances must be considered. The hot-channel factors associated with these uncertainties are all assumed to be acting in a detrimental fashion on a single element; thus, the design flow rate will be sufficient to satisfy the worst possible conditions which may exist in the core.

In a core in which the occurrence of local boiling is accepted, the limiting thermal criteria are:

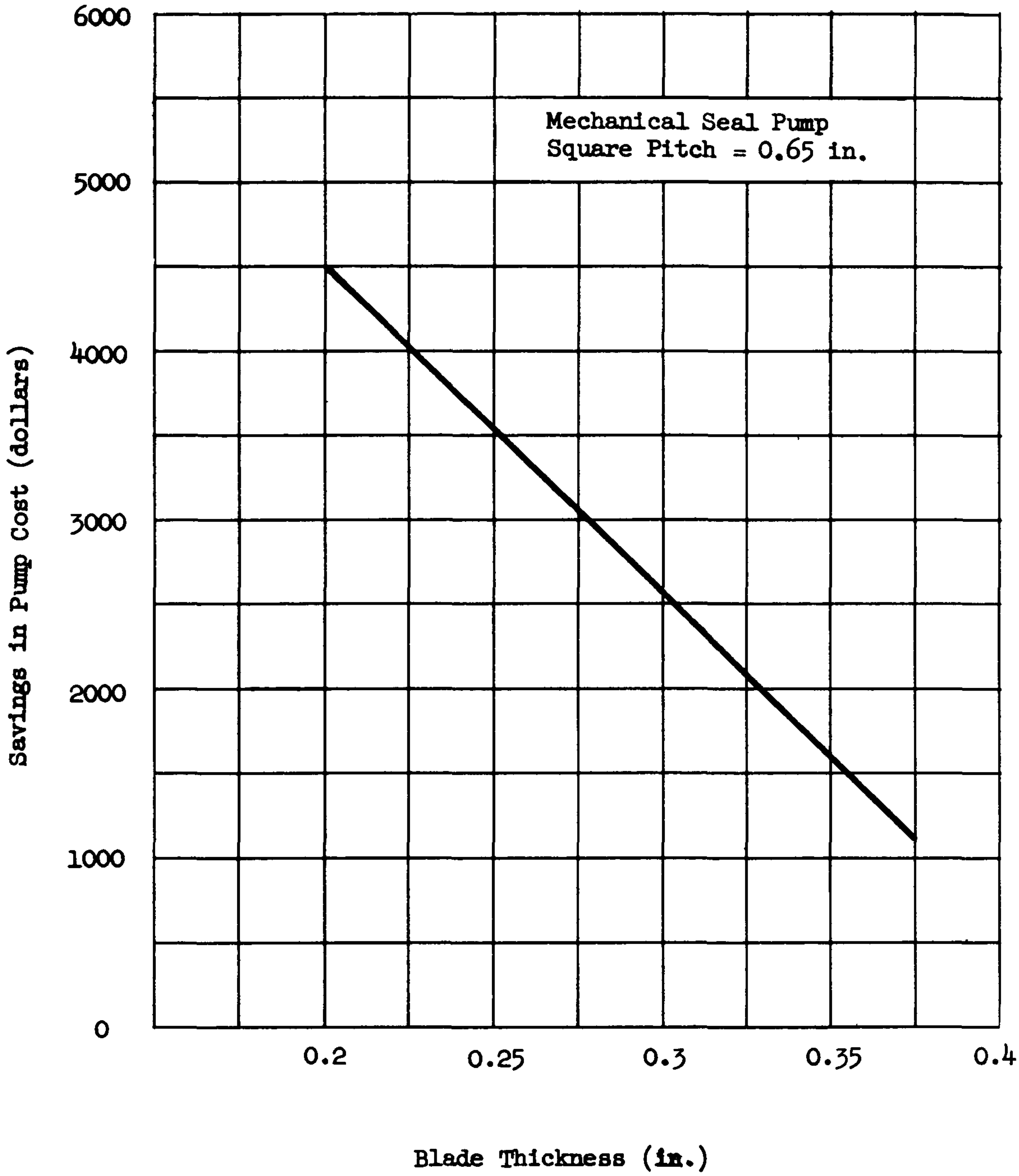


Fig. III-20. Variation of Pump Cost with Control Rod Thickness

(1) The bulk coolant temperature may not exceed the saturation temperature.

(2) At no point may the burnout heat flux be exceeded.

The surface temperature of the fuel element, which is generally the limiting thermal criterion in nonboiling cores (in order to prevent local boiling), need not be considered.

The hot-channel factors may be considered in two parts: F_q, the hot-channel factor to account for uncertainties in neutron flux determinations and power measurements, and F_b, the hot-channel factor contributing to the bulk coolant temperature rise. The maximum possible exit coolant temperature is then determined by:

$$\theta_{\substack{\text{exit}\\ \text{max}}} = \theta_{\text{inlet}} + \beta F_q F_b \left(\theta_{\substack{\text{exit}\\ \text{hot tube}}} - \theta_{\text{inlet}} \right)$$

where:

β = peak-to-average ratio of radial power distribution.

Based upon known manufacturing tolerances and estimated uncertainties in the analytical techniques, the following hot-channel factors were determined:

Hot-Channel Factors

	Contribution to F_b	Contribution to F_q
Uncertainty in neutron flux		1.10
Uncertainty in power level		1.12
Variation in meat thickness	1.03	
Variation in fuel concentration	1.02	
Plenum chamber flow variation	1.07	
Flow variations due to channel and orifice dimension variation		
Inside tubes	1.011	
Outside tubes	1.022	

	Contribution to F_b	Contribution to F_q
Resultant factors		
F_b inside	1.137	F_q = 1.232
F_b outside	1.149	

It was found that, by substituting the above factors and expressions for β into the equation for exit temperature, the limiting thermal criteria set forth previously would not be exceeded.

Reactor pressure drop.- The head loss in the reactor from inlet to outlet was calculated by summing the consecutive losses in the coolant flow path. These losses were calculated using standard pressure-drop equations and are tabulated below.

Reactor Head Loss

	(feet)
Reactor inlet pipe	8.6
Inlet water box	0.3
Water box orifice	3.0
Thermal shield	0.7
Entrance to bottom plenum	Negligible
Core	0.5
Outlet of upper plenum	3.0
Outlet of water box	0.3
Reactor outlet pipe	4.6
Total	21.1

After-heat dissipation.- The arrangement of the primary loop must be such that sufficient natural convection is present to adequately remove the heat released due to fission product decay in the event that electrical power is not available.

An equation was derived which relates flow rate to available pump power. The results are plotted in Fig. III-21 for the case where primary loop head losses are assumed to vary as the flow rate squared and for the case where they are assumed to vary as the flow rate to the 1.8 power.

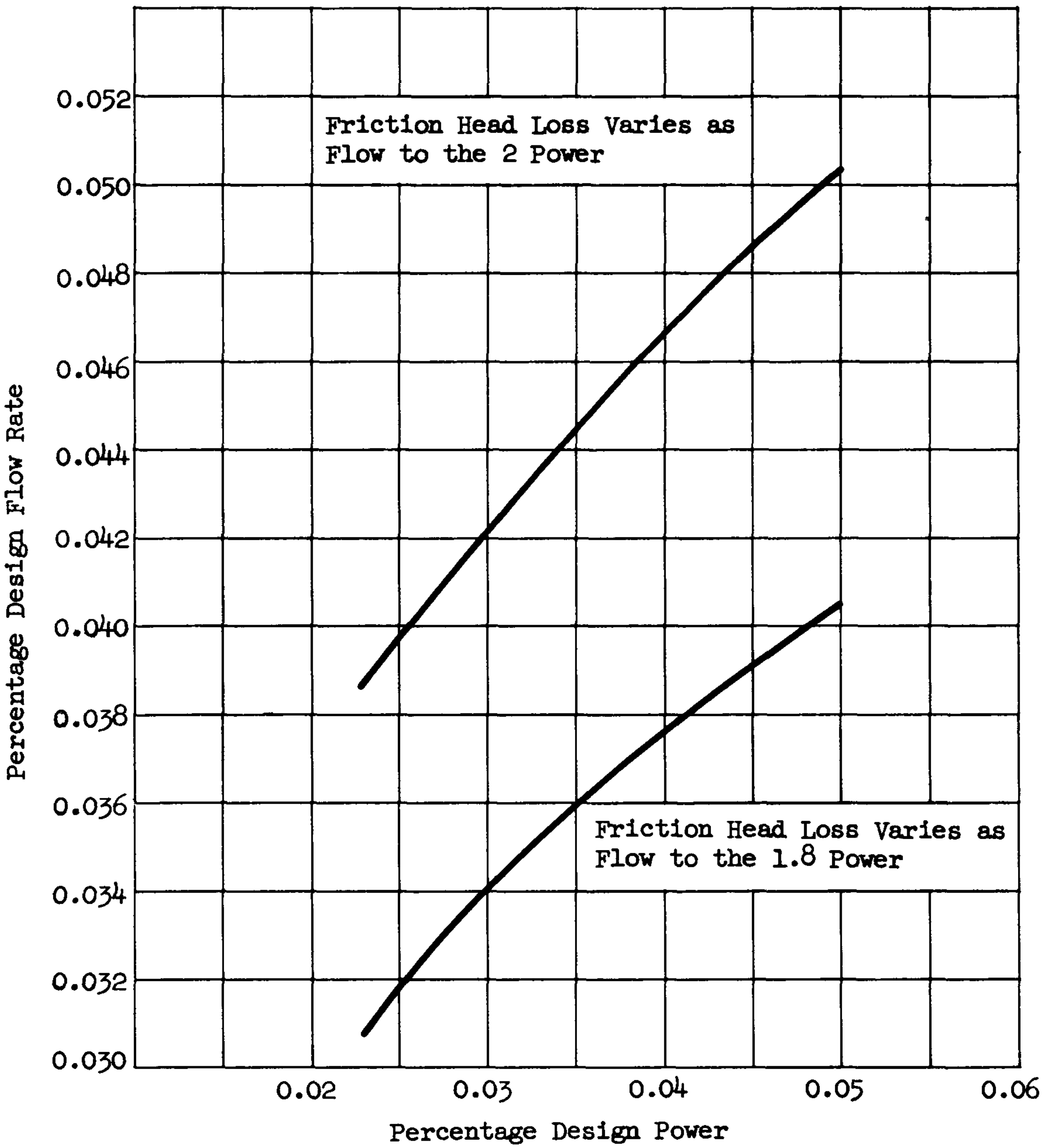


Fig. III-21. Percentage Flow Rate (natural convection) versus Percent of Power

Immediately after a scram, the heat released by the decay of fission products is 5% of the original core power. Figure III-21 shows that, for the case where flow resistance varies as the square of the flow rate, the percentage of design flow rate is always equal to or greater than the percentage of design power generated in the core. In other words, there is sufficient natural convection to lower primary loop temperature immediately after the cessation of fission heat. For the case where flow resistance varies as the flow rate to the 1.8 power, the percentage of design flow rate exceeds the percentage of design power generation after power production drops below 3.6% of design value. Since after-heat power drops below this value after about 7 seconds, natural convection will be sufficient to decrease temperatures in the primary loop shortly after shutdown.

The variation of the reactor outlet temperature during the first seven seconds after shutdown was investigated using an analog computer. In this investigation, the flow rate was decreased to 3% of design value as a step function. The heat transfer coefficient in the steam generator was lowered to account for this flow rate in the primary side and for pool-boiling in the secondary side. The reactor scram was delayed 0.5 seconds to account for delays in the rod movement. The power, produced in the core after the scram, was made to approach 5% of design power asymptotically so as to simulate heat released by fission product decay. The analog plot of the coolant temperature leaving the reactor is shown in Fig. III-22.

Actually, the results shown in Fig. III-22 are conservative since the flow rate in the primary loop coasts down rather slowly. The percentage of design flow rate in the primary loop for a short time after a loss of power in the primary pump is shown in Fig. III-23.

Earth shield temperature distributions.- In order to obtain sufficient room in the package for the placement of a spent core during refueling, the reactor is located close to one side of the package. This results in a high heat generation rate in the earth adjacent to that side of the package.

The package containing the reactor utilizes an air-filled double liner which insulates the earth, to some extent, from the shield water inside of the package. The problem was treated as one of conduction in the radial direction only (a one-dimensional solution of the diffusion equation); hence, the results are somewhat conservative.

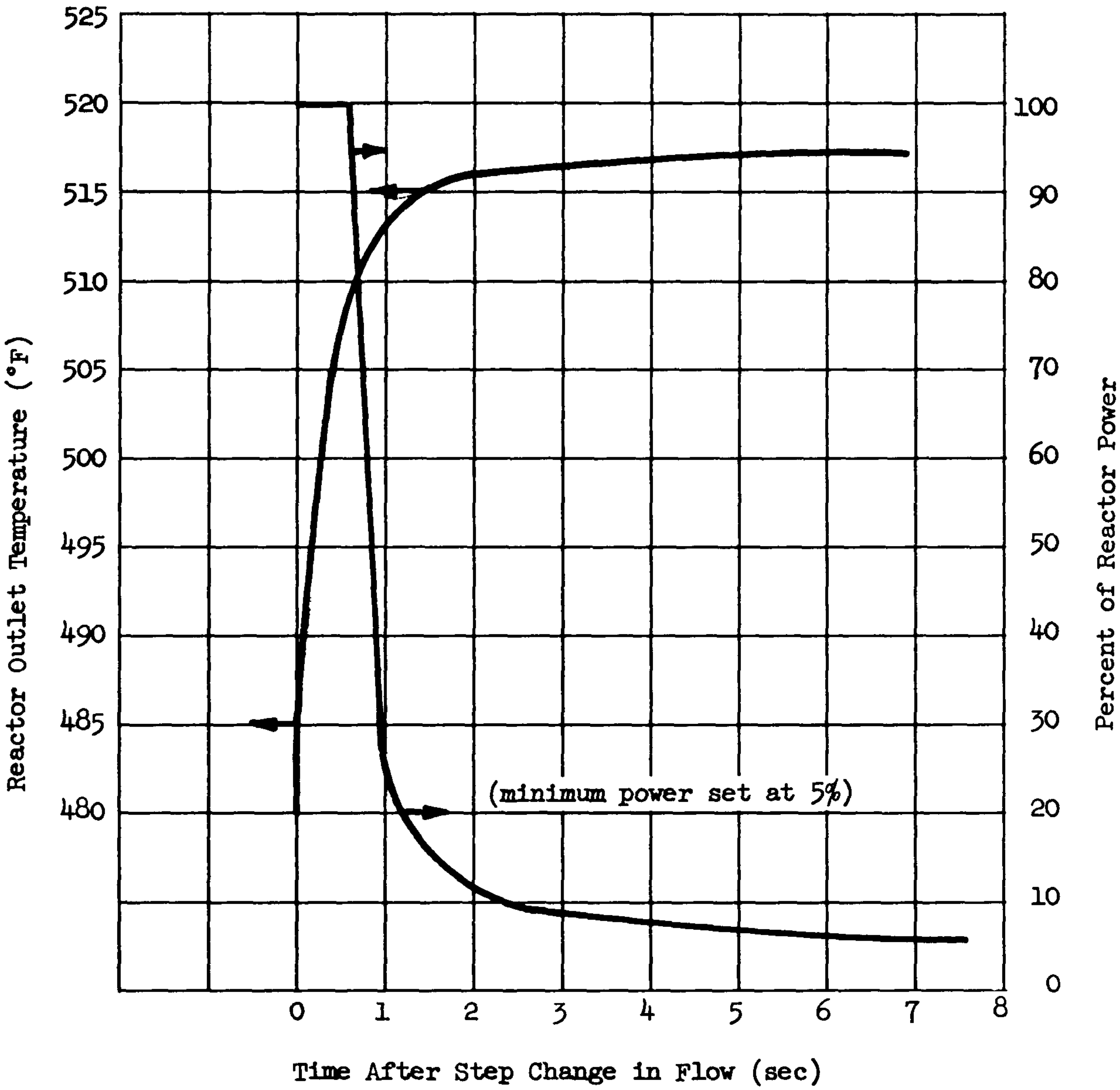


Fig. III-22. Reactor Power and Outlet Temperature for a Step Change in Flow (100% to 3%)

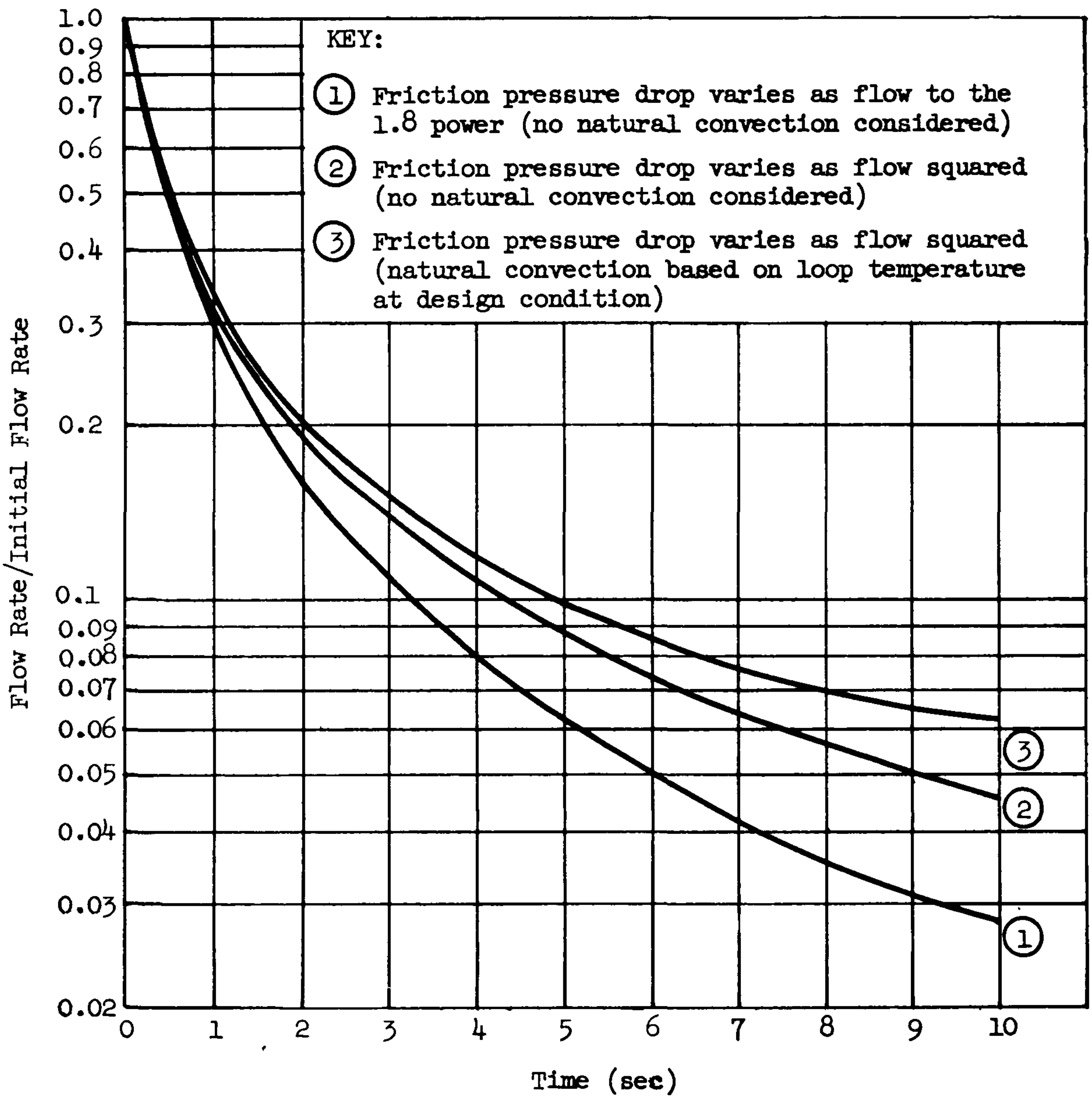


Fig. III-23. Flow Coastdown in Primary Loop

It was found that shielding in addition to the shield water is required to maintain reasonable temperatures in the earth. With a heat source term based on the addition of 2 inches of lead between the reactor vessel and the package wall, the temperature distribution in the earth was calculated for two ambient temperatures extremes. One, of 100° F, represented the maximum summer temperature; the other, of -60° F, represented the minimum winter temperature. The results are shown in Fig. III-24.

Spent core cask heat removal system.- After the spent core is removed from the reactor package, it will be placed in a shielded cask and cooled at the plant site, using a separate section of the shield water cooler, for a period of ninety days prior to shipment to the reprocessing plant.

The shipping cask will be equipped with a heat removal system which will serve as a standby cooler during the initial cooling period and will remove all after-heat during shipment. An air-cooled reflux condenser was selected because of the simplicity and reliability attainable. The core will be immersed in water, which will be allowed to boil. The vapor will be condensed in tubes and returned to the water surrounding the core. The condenser tubes will be designed to maintain the vapor at slightly less than atmospheric pressure.

System analysis summary.- Analyses of primary systems were performed during the quarter in support of preliminary design. Systems analyzed were:

(1) A heat balance and flow diagram

(2) The primary coolant pump

(3) The pressurizer

(4) Waste disposal

(5) Pressure relief valves

(6) After-heat removal

(7) Pressure relief system

(8) Demineralizers

(9) Charging system

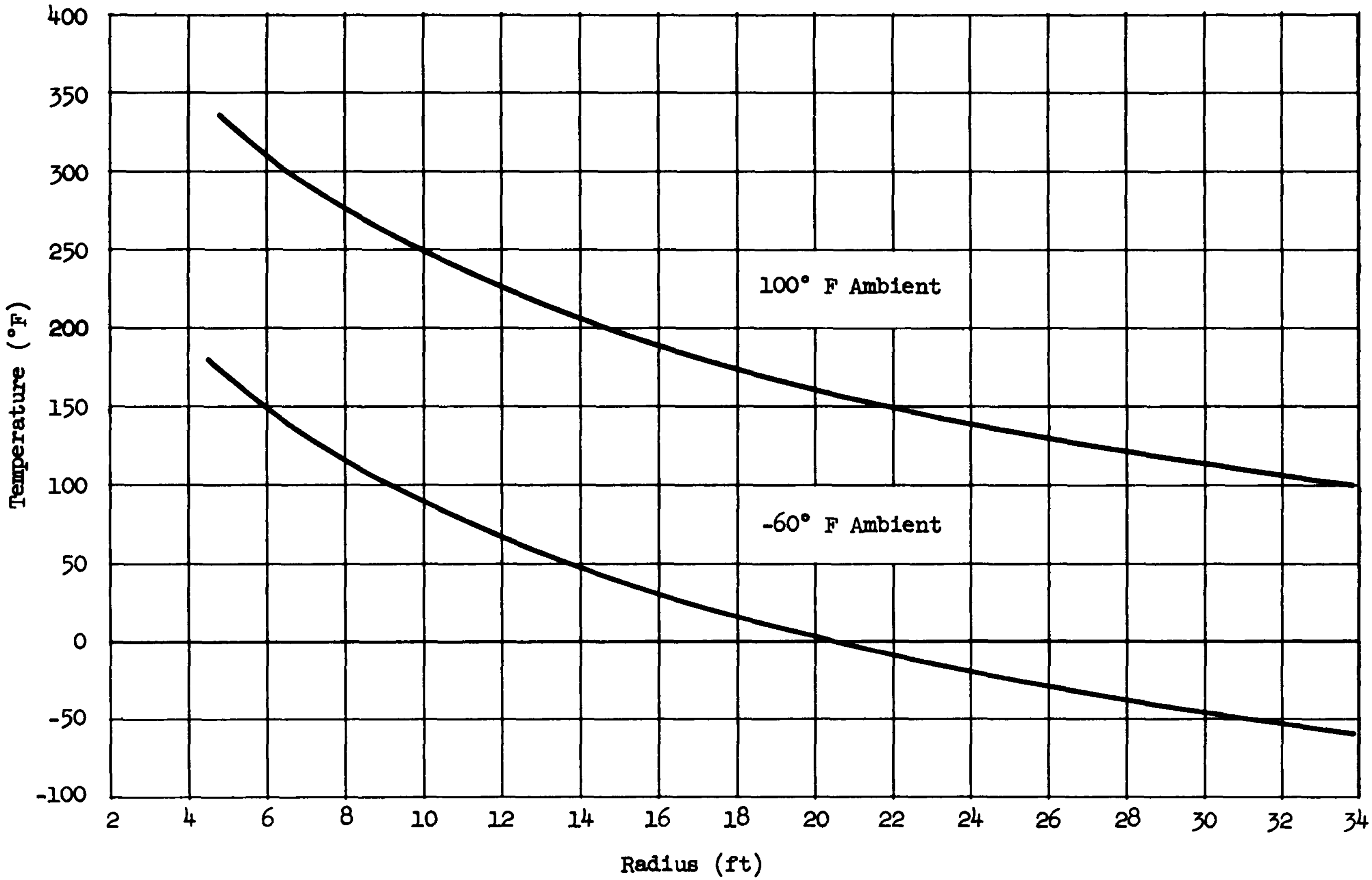


Fig. III-24. Temperature Distribution in Earth with Shielding (radial conduction only)

(10) Gas removal system

(11) Blowdown system

(12) Shield water system

(13) Startup heat requirements

(14) Auxiliary power requirements

(15) Plant containment

(16) Chemical addition and fill.

Primary system heat balance and flow diagram.- A primary system heat balance was developed and is presented in Fig. III-25; the flow diagram is presented in Fig. III-26. The primary loop is drained from the bottom of the reactor vessel and other low points in the system to the sump tank or the shield water, depending on the water condition at the time of draining. The system is filled by a fill pump located in the secondary system. The filling water enters at the bottom of the reactor. Vents at all high points in the system are opened to the atmosphere.

Most of the other equipment shown in the flow diagram is discussed under the respective systems.

The modes of failure, operation, and location of remote and self-actuated valves were dictated by reliability and safety considerations. Normally, more than two failures are required to cause an immediate or eventual shutdown of the primary system.

The primary coolant pump.- A centrifugal-type pump is used to circulate coolant through the primary loop. It must be capable of overcoming flow resistance produced by coolant flow through the reactor, the main loop piping, and the steam generator.

The main piping flow resistance was calculated based upon a flow requirement of 1900 gpm, 43 feet of 6-inch schedule 80 piping, coolant velocity of 23.4 ft/sec, 8 long-radius 90-degree elbows, and 2 long-radius 45-degree elbows. The total piping resistance was calculated, by the method presented in the Hydraulic Pipe Friction Manual, to be 26.1 feet of fluid. Westinghouse Electric Corporation advised us that the head loss in the steam generator would be approximately 40.6 feet of fluid with a 1900-gpm flow rate.

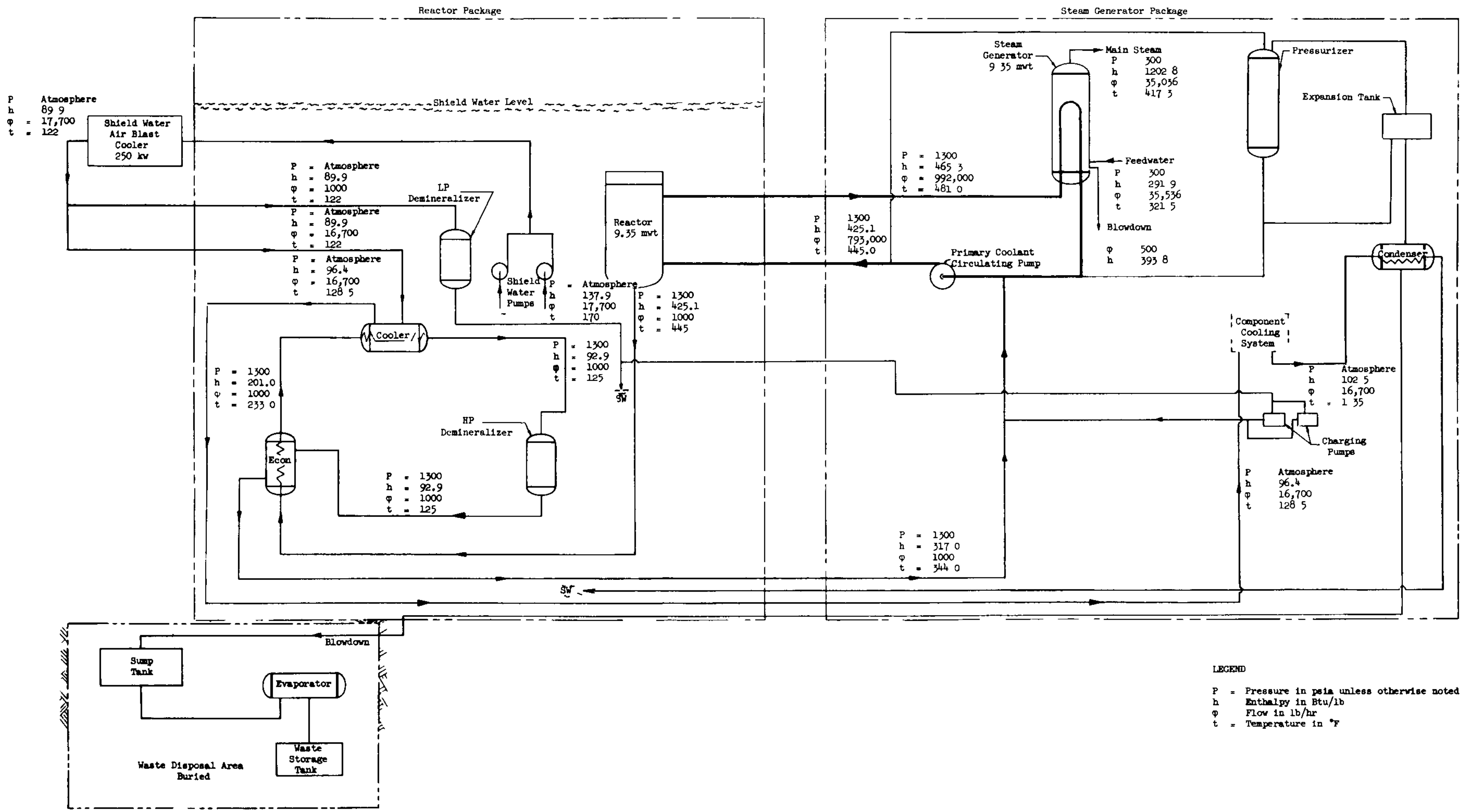


Fig. III-25. Heat Balance--Primary System--PM-1 Power Plant

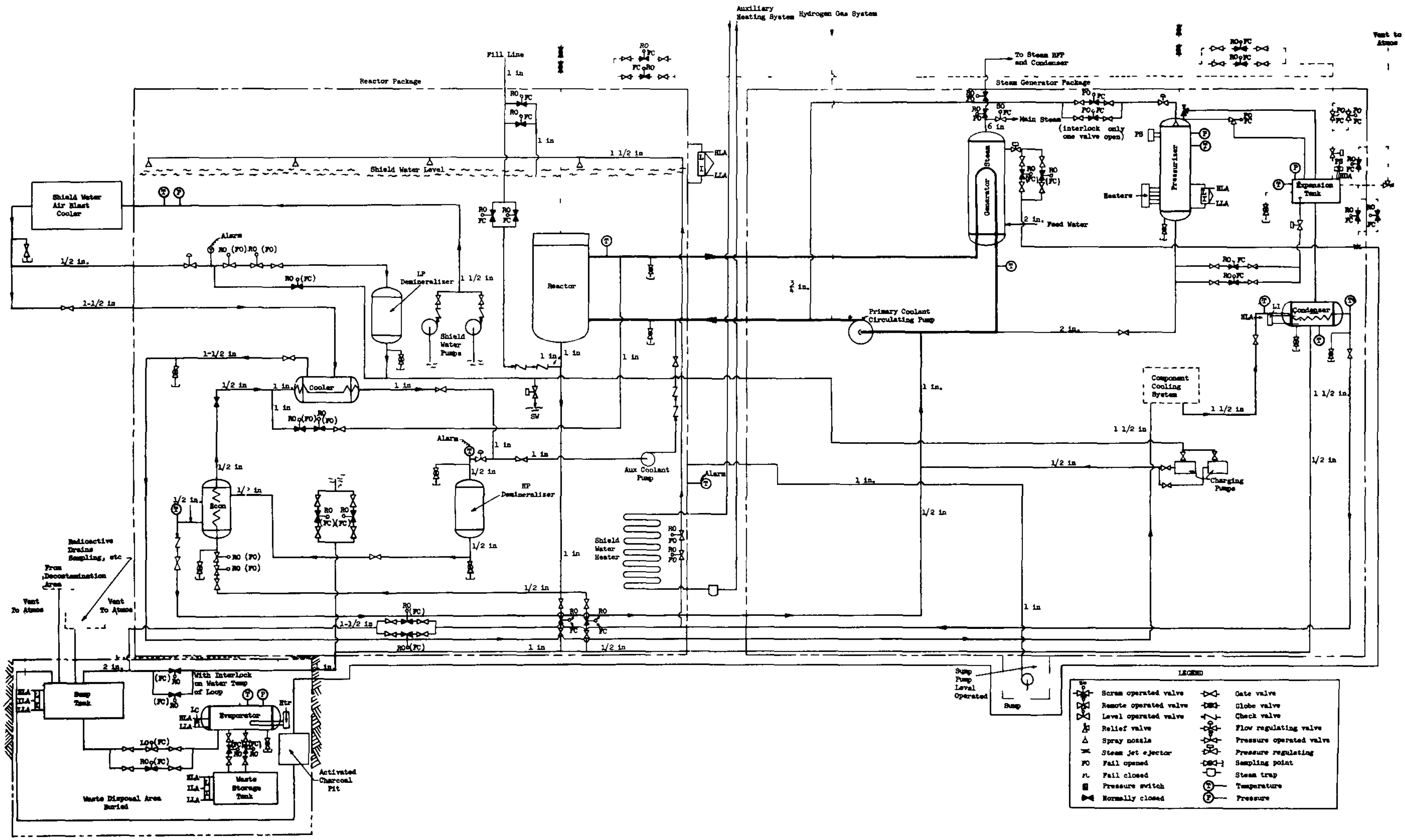


Fig. III-26. Piping Diagram--Primary System--PM-1 Power Plant

The reactor head loss, as reported previously, was calculated to be 21.0 feet. The total head for the primary system, after allowing an additional 10% for aging, uncertainties in calculation, etc., is about 96.5 feet.

The required pumping power, based upon a 0.825 specific gravity and a 60% overall pump efficiency, was calculated to be 49.2 kilowatts.

A listing of the information needed from vendors to thoroughly evaluate proposals to be submitted by them was made, as was a list of centrifugal-type pump suppliers for nuclear applications. Copies of the pump specifications will be sent to appropriate suppliers.

The pressurizer.- The pressurizer serves to maintain primary system pressure within limits above and below the 1300-psia normal operating pressure in spite of instantaneous positive or negative full load transients imposed through the steam generator--the full load transients imposed on the steam generator being due to abrupt changes of electrical load on the power plant.

General pressurizer design criteria are that the design pressure is not to be exceeded during an insurge, and that the system pressure shall be sufficient to prevent bulk boiling any place in the core during an outsurge.

The assumptions (in most cases conservative) used in the pressurizer analysis were:

(1) Heaters and spray nozzles are inoperative during transients

(2) Steam expansion and compression is isentropic

(3) Pressurizer water is saturated.

The transient found by analog studies to produce the most severe expansion of the primary system water volume occurs when:

(1) Heating load is zero.

(2) Plant electrical load is instantaneously decreased from 1250-kw gross to the 125-kw plant auxiliary load.

As a result of this transient, the primary water volume may be expected to expand 1.3 ft^3 in 92 seconds.

The transient producing the most severe contraction of the primary system water volume occurs when the instantaneous load change is the reverse of that defined above. As a result of this transient, the primary water volume may be expected to contract 1.25 ft^3 in 37 seconds.

The pressurizer volume defined by the parametric study was 26 ft^3. A layout of the pressurizer allowing for maximum positive and negative surges, prevention against heater exposure, and water level variation was made; it was found that a steam volume of 13.7 ft^3 and a water volume of 12.3 ft^3 was required.

With the pressurizer sized in the above manner, the primary system pressure remains within limits of 1470 and 1172 psia while undergoing the instantaneous negative or positive full load transients described above. These limits are very conservative since no advantage has been taken of mixing on an insurge or flashing of water on an outsurge. Although a system pressure of 1172 psia is lower than the thermodynamic design pressure of 1200 psia, this is considered acceptable since the maximum reactor power is not attained during this transient.

The transient producing the greatest reactor power accompanies the load pickup described above with the exception that the plant is supplying the full heating load in its initial state. In this case, the water volume in the main coolant loop contracts 0.9 ft^3 in 35 seconds and the primary system pressure decreases to 1210 psia--which is above the thermodynamic design pressure of 1200 psia.

The total installed pressurizer heater capacity is estimated to be 48 kw. It is expected that further detailed studies will reduce the installed capacity and volume considerably.

Waste disposal.- The philosophy adopted for the disposal of the PM-1 radioactive wastes is concentration and storage as liquid or solids. Dilution and dispersion is not feasible from the viewpoint of total activity and is not consistent with the basic plant location criteria, since dilution water may or may not be available at a given site. Dilution in air is not feasible due to the relatively large activity to be dispersed.

Equipment is provided to handle both liquid and gaseous wastes.

The liquid waste disposal system equipment consists of a sump tank, evaporator, waste storage tank, and the interconnecting piping. The sump tank serves to collect and retain all wastes. The evaporator, which is a high decontamination factor type, concentrates the wastes fed to it. The concentrated wastes resulting from evaporation are sent to the waste storage tank for an undetermined period of time.

The main sources of influent to the liquid waste disposal system are:

(1) Blowdown from the primary loop

(2) By-products from the decontamination room

(3) Shield water wastes

(4) Decontamination cycle products.

The gaseous waste disposal system is a charcoal bed located in the waste disposal area. The bed serves to remove radioactive gases by adsorption on charcoal.

The main sources of influent to the gaseous waste disposal system are:

(1) Gases vented from the primary system

(2) Fission gas release.

Pressure relief valves.- The reactor pressure vessel, the pressurizer, and the main coolant piping loop are protected from overpressure by safety relief valves.

The ASME Boiler and Pressure Vessel Committee has issued a special ruling on interpreting Section VIII of the Code with respect to safety requirements for pressurized water reactor vessels. To conform with the special ruling and Section VIII of the Code, the following conditions were selected for the design of the reactor coolant pressure relief valves:

(1) The maximum allowable coolant pressure shall not exceed the design pressure by more than 10%.

(2) A 3% overpressure over the initial set pressure will suffice to give the valve its full lift for maximum capacity.

(3) Two totally-enclosed, pop-type safety valves will be provided.

(4) The discharge capacity is 100% of the actual capacity.

(5) A set pressure tolerance of plus or minus 3% is used.

(6) The 2 valves are located on the pressurizer vessel to permit relief of steam from the vapor section.

(7) A back pressure of 50 psia is exerted on the safety valves.

(8) There are no valves between the reactor vessel and the pressurizer which could isolate either vessel.

The maximum coolant pressure of 1470 psia, as determined by analog studies reported previously, is caused by expansions of the primary system water volume.

The set points of the 2 safety valves are 1550 psia and 1600 psia.

After-heat removal.- The after-heat removal system employs a pump and cooler. It is used when after-heat removal by natural convection at the steam generator is impossible, due to the opening of the primary loop.

The cooler, which normally cools the blow-down entering the high-pressure demineralizer, is sized to remove 93.5 kwt (1% of full power) of core after-heat. The after-heat removal pump takes suction at the reactor outlet and pumps 10 gpm (4870 lb/hr) of primary coolant water through the cooler and back in the primary loop at the reactor inlet. Water enters the cooler at 205° F and leaves at 140° F.

Pressure relief system.- The pressure relief system includes the pressurizer steam relief valves, an expansion tank, and a condenser. Steam released from the pressurizer is condensed and drained to the waste disposal system if found to be contaminated. Non-condensibles are held in the expansion tank and pumped to the activated charcoal bed by a gas blower, if found to be radioactive.

Two analog runs were made to determine the rate at which steam should be released from the pressurizer. On the first run, a step change from 100% load to 0% load was made at the steam generator. All coolant flow was stopped immediately except for a 3% natural convection flow. No control rod movement was permitted and the reactor power was decreased by a conservative negative temperature coefficient to 5% power. After an initial expansion of 0.6 ft^3 in 12 seconds, the maximum expansion rate in the loop was found to be 2 cfm. No peaking occurred during this run and the expansion rate leveled off to 0.4 cfm after a total expansion of 1 ft^3 in 25 seconds.

The second run was similar to the first run with the exception that the step load change was from 76% to 12% at the steam generator and the primary coolant pump remained on the line at 1900 gpm. After an initial expansion of 0.6 ft^3 in 10 seconds, the maximum expansion rate in the loop was found to be 3 cfm. A peak expansion occurred at 1.3 ft^3 after 42 seconds.

Assuming that the spray nozzles in the pressurizer are inoperative, that the pressurizer does not handle any of the loop expansion, and the pressure relief condenser does not start to condense until the expansion tank has accumulated steam at 50 psia, equivalent to 0.6 ft^3 of loop expansion, the maximum steam release rate, determined by the second analog run, was found to be 3 cfm.

If the entire 9.35 Mwt of reactor power is used initially to heat the coolant at 1300 psia and 463° F, an initial expansion rate of 9.2 cfm is attained. The reactor negative temperature coefficient reduces the expansion rate to less than 4.7 cfm very rapidly. The force volume of the pressurizer and the capacity of the safety valves are such that the system pressure never exceeds 1650 psia.

The pressure relief valves (2) are each sized to relieve 2.35 cfm of steam at 1500 psia and 14° F superheat. Design of the valves is in accordance with a special ruling (Case No. 1271N) by the ASME Boiler and Pressure Vessel Committee interpreting Section VIII of the Unfired Pressure Vessel Code with respect to safety requirements for pressurized water reactor vessels.

Demineralizers.- Mixed-bed demineralizers are used to economically produce high purity water and to maintain control over coolant pH. A mixed-bed demineralizer is desirable since, with an influent having a relatively low solid content, it produces higher quality water than a multi-bed system. This type of unit is also effective in removing total dissolved solids in that it acts as a filter bed. The major function of the cation resin, is removal of cations found in the compounds of the primary system corrosion products. The anion resin serves mainly to control the coolant pH. Regeneration was not found to be feasible.

The most suitable resin commercially available is Amberlite XE-209. XE-209 may be used in the high-pressure (primary loop) and low-pressure (shield water) demineralizers. XE-209 which is composed of 1-1/2 parts type XE-77 cation resin and 1 part type XE-78 anion resin produces an exchange basis of approximately 2-1/2-to-1 cations over anions. The recommended operating temperature for this resin is 120° F.

A study was made of three demineralizer designs which were initially deemed to be feasible. The types considered were: (1) cartridge replaceable, (2) throw-away, and (3) regenerative.

Results of studies made on the feasible concepts showed only the cartridge replaceable and the throw-away type to be practical.

Resin lives of 6, 12, and 24 months were initially considered for both the low- and high-pressure demineralizers. Consultation with vendors revealed that designing for a resin life of more than 12 months is not feasible. The recommended life for the high- and low-pressure units is 12 months.

A comparison of the cartridge type and throw-away type based on 12-month resin life is tabulated below:

Low-Pressure Demineralizer

	Cartridge Type	Throw-Away Type
Total weight, lb	200	150
Total cost, dollars	1200	800
Cartridge cost, dollars	250	--
Shell size	9-inch OD x 48-inch overall length	

High-Pressure Demineralizer

	Cartridge Type	Throw-Away Type
Total weight, lb	1000	--
Total cost, dollars	5100	4500
Cartridge cost, dollars	300	--
Shell size	17-inch OD x 84-inch overall length	

The cartridge-type demineralizer using Amberlite XE-209 resin was recommended for both the high- and the low-pressure demineralizers.

Charging system.- A reciprocating, high-head pump is used to charge the primary system with make-up water from the shield water system.

A flow of 1 gpm was selected in order to prevent thermal stress problems. The charging pump takes its suction from the shield water system at the outlet of the low-pressure demineralizer, thus providing a positive head at the suction. The discharge from the pump mixes with the blowdown flow (2 gpm) entering the loop between the pump suction and the steam generator outlet. The temperature difference between the charging flow and blowdown flow is 210° F. After mixing, the temperature of the fluid is 277° F which is 186° F below the loop mean temperature.

Gas removal system.- Since radioactive gases such as xenon and krypton can escape into the primary loop, vents at the high points of the system open into the expansion tank where the activity level of the vented gases can be measured. If the gases are found to be contaminated, the expansion tank is vented and then evacuated to a pressure of 2 psia, while the contaminated gases are sent to an activated charcoal bed.

The expansion tank, which is designed for the pressure relief system can hold 0.044 lb-mol of gas at a partial gas pressure of 25 psia. This is considered to be the maximum gas pressure allowable, since the expansion tank must hold 2.0 lb of 30° superheat steam at a partial pressure of 50 psia without lifting the expansion tank relief valves.

A pressure-reducing valve in the vent line ahead of the expansion tank closes the vent line when the pressure in the tank is 25 psia, and a pressure switch sounds an alarm to alert the plant.

Blowdown system.- In order to limit the concentration of impurities in the primary coolant water, the coolant water is continuously recirculated at a rate of 2 gpm at 125° F to a high-pressure demineralizer. Since a resin bed melts at a temperature in the neighborhood of 140° F, the recirculated primary coolant must be cooled before entering the demineralizer. Cooling is accomplished by passing the recirculated coolant through an economizer and cooler before entering the demineralizer.

The economizer decreases the temperature of the recirculated coolant 212° F. The cooler further drops the temperature 108° F to 125° F. An economizer is employed in the blowdown system for two reasons:

(1) To increase the temperature of recirculated coolant entering the loop from 125° F to 344° F.

(2) To decrease the heat load (65.5 kwt) on the demineralizer cooler and also the shield water air blast cooler which ultimately must reject the heat load of the cooler.

The demineralizer cooler has a terminal temperature difference of only 3° F. This low temperature difference is due to the fact that the cooler is designed to reject the reactor core after-heat and, therefore is over-designed for use as a demineralizer cooler.

Whenever the temperature of the blowdown flow entering the high-pressure demineralizer reaches a temperature of 130° F, and alarm is sounded and the blowdown system is shut down. This precaution is taken to prevent possible melting of the demineralizer bed.

Shield water system.- The shield water system has several objectives:

(1) To provide water for shielding use during operation and maintenance.

(2) To continuously purify the shield water.

(3) To provide cooling for the pressure relief system.

(4) To cool the shield water which is heated from the following sources:

(1)	Gamma heating	145 kwt
(2)	Spent core heat removal	43 kwt
(3)	Demineralizer cooler	32 kwt
(4)	Component cooling	30 kwt
		250 kwt

A shield water level is maintained in the reactor package for shielding purposes. The shield water circulating pump removes water from the bottom of the tank and pumps the water through the shield water system, which includes:

(1) Shield water air blast cooler

(2) Low-pressure demineralizer

(3) Demineralizer cooler

(4) Component cooling system

(5) Pressure relief condenser.

The shield water air blast cooler removes 250 kwt from the shield water with 100° F coolant air available. Water enters the shield water cooler at 170° F at a rate of 36.3 gpm and leaves at 122° F. After leaving the air blast cooler, the shield water flow is divided by sending 2 gpm to the low-pressure demineralizer and 34.3 gpm to the demineralizer cooler. If the shield water temperature leaving the air blast cooler climbs to 133° F, a high-temperature alarm is sounded and the low-pressure demineralizer is shut off by remotely-operated valves.

The shield water removes 32 kwt in the demineralizer cooler, raising its temperature from 122° F to 128.5° F. After leaving the demineralizer cooler, the shield water is available for cooling the primary coolant pump and the actuator. Capacity is available to cool additional components.

Leaving the component cooling system at 135° F, the shield waters enters the pressure relief condenser where it is continuously available as a heat sink in the event that the pressure should increase. From the condenser, the shield water enters the reactor package through spray nozzles located at the periphery of the tank just above the shieldwater level. Should the shield water be contaminated by leakage, etc., it is diverted to the sump tank in the waste disposal system.

During a prolonged shutdown, the shield water air blast coolers would be shut down and drained to prevent freeze-up. Steam-heating coils, located in the shield water tank, will receive steam from the auxiliary heating system and will maintain the shield water temperature above feeezing.

Startup heat requirements.- Heat must be added to the pressurizer and primary loop in order to bring the system up to operating pressure and temperature. The reactor core may be used to bring the primary loop up to operating temperature but electrical heaters must be used to bring the pressurizer to operating pressure. Calculations showed that 180 kw-hr are required to bring the pressurizer up to pressure. The heater capacity in the pressurizer will, therefore, determine the start-up time required to pressurize the system. The rate at which the loop is brought up to temperature will be determined by thermal stress considerations.

Auxiliary power requirements.- Auxiliary power is required in the primary system for the following items:

(1)	Primary circulating pump	45.0 kw
(2)	Pressurizer heaters	54.0 kw

(3)	Charging pump (1 gpm at 1300 psi)	1.2 kw
(4)	Shield water circulating pump (36.3 gpm, 176 feet of head)	2.4 kw
(5)	Shield water cooler fans (48,600 cfm, 0.75 inches of head)	6.7 kw
(6)	Containment sump pump (10 gpm, 100 feet of head)	0.3 kw
(7)	After-heat removal pump (10 gpm, 83 feet of head)	0.3 kw
		109.9 kw

The power requirement of the pressurizer heaters is expected to decrease during final design. The power requirement of the pumps was based on what are believed to be more than ample pumping heads. The primary circulating pump, which is one of the largest users of auxiliary power, cannot be accurately sized at present.

Plant containment.- The plant containment, if required, consists of three interconnected T-1 steel tanks. The primary function of the containment is to retain primary system fluids subjected to the energy released during an excursion.

The estimated amount of energy which may be released from the primary system is 1,912,400 Btu. The sources of this energy are:

(1)	Excursion energy	300,000 Btu
(2)	Primary loop energy, 56 ft^3 of water at 463° F	1,270,000 Btu
(3)	Pressurizer energy	
	(1) 14.5 ft^3 of 1300 psia D and S steam	48,400 Btu
	(2) 11.5 ft^3 of saturated water	294,000 Btu

The containment volume, 4100 ft^3, was assumed to contain, initially, 200 pounds of air at 170° F and 11.6 psia. After the excursion energy is released, assuming a constant internal energy process, the equilibrium temperature becomes 338° F. Neglecting shock effect, the total pressure becomes 118.0 psig--the steam partial pressure is 115 psia and the air partial pressure is 14.6 psia.

The capacity of the upper reactor containment vessel connection to remove vapor during an excursion was investigated. It was assumed that a complete shear of one of the main primary loop pipes occurred and that the steam partial pressure in the top of the reactor containment vessel approached its design maximum (158 psia). Under these conditions, 562 pounds of steam would fill the 450 ft^3 volume. Flow through the tank interconnection would be approximately 6850 lb/sec. Since the flow from the primary system after removal of the first 562 pounds would only approximate 1550 lb/sec, it was concluded that no problems would result.

In order to minimize humidity problems during operation, a vapor seal will be placed over the shield water tank. During shutdown maintenance, fresh air will be circulated at 50 cfm to maintain acceptable working conditions inside of the container.

Chemical addition and fill system.- The primary system is filled from a distilled water storage tank in the secondary system. During initial startup, when there is no gamma flux, a hydrazine solution is charged into the primary loop to control the oxygen content; during operation, hydrogen gas is injected into the primary system from 2200-psia bottles.

Provisions have been made to fill the primary system with decontamination and rinse solutions if decontamination should be necessary. A stainless steel mixing tank and low-head stainless steel pump are provided to fill the system with hydrazine or decontamination and rinse solutions. The mixing tank has a capacity of 325 gallons, 1/2 the total primary loop volume.

In the event of the loss of the primary coolant flow and the failure of the control actuators to scram the reactor, boron will be injected into the system to ensure shutdown. Analog runs show that the negative temperature coefficient will bring the reactor down to subcritical (20% power in 10 seconds and 5% power in 60 seconds) and that the relief valves will pop (at 1500 psia) approximately a minute after the incident even if the spray does not function and no heat is withdrawn by the steam generator. The boron injection system need not, therefore, be complicated by a requirement for rapid action. In order to shut down the reactor core, 100 grams of Boron 10 must be injected into the reactor core volume of 7.85 ft^3. This calls for injecting a total of approximately 2 pounds of Boron 10, or 11 pounds of natural boron into the loop. The chemistry of the boron injection solution and the mode of injection have not been determined at this time. Chemistry will be based upon the solubility and decomposition properties of the selected boron compound in the environment of the primary loop during and after injection. The mode of injection (i.e., gas pressure or pump) will be determined by the quantity of the solution involved, the time requirement for complete injection, and the safety requirement.

3. Shielding Analysis Studies

W. D. Owings, E. L. Divita, E. F. Koprowski

Shielding analysis studies involved determination of primary system shielding, spent fuel handling, primary loop coolant activation, and reactor vessel and heating problems.

Primary system shielding.- During this quarter, preliminary shield designs were developed for contained and uncontained versions of the PM-1 reactor. The primary consideration in shield design was minimization of air-transported shield materials by using materials locally available at the Sundance site. This was accomplished by surrounding the vertical primary system packages to a height of approximately 22 feet above their bases with earth. (Water was not suggested for exterior shielding use due to the inadequate supply at the Sundance site, climatic considerations, additional plant equipment required, and shield water tank maintenance problems.) The major disadvantage of the earth shield is possible neutron activation. An evaluation of earth activation over extended periods of reactor operation will be performed during final design to determine whether additional shielding will be required adjacent to the reactor package in the vicinity of the reactor vessel. In other respects, the earth mound shield is considered superior to a water shield or combination water-earth shield.

The use of interior shield water, which extends to a height of 13 feet above the reactor pressure vessel, was found to be consistent with the design requirements for limited personnel access to the housing area during full-power operation and for personnel access during refueling operations. A water shield offers the following advantages over other shielding materials:

(1) Excellent neutron shield; neutron fluxes in the housing area being reduced to neglible quantities.

(2) Limited amounts of water are available locally, thus reducing the amount of air-transported shield materials required.

(3) Shielding such as lead (gamma) or organic plastics (neutron) would tend to overcrowd the package, creating additional inspection or maintenance problems.

(4) Water allows good visibility for maintenance, inspection, and refueling operations.

The vertical steam generator incorporated in the present design was determined to be superior from the shielding aspect since the secondary water and the top head of the generator provide self-shielding overhead (in the housing area) from radiation originating in the primary coolant water. Local shielding for the primary piping may be necessary in the steam generator package.

A description of the shielding evolved to date for the contained and uncontained versions with dose rates in pertinent areas and a discussion of computational methods follows.

Total dose rates in the contained version at any point within the shelter above the containment will not exceed 200 milliroentgens per hour during full-power operation. Based on a maximum of 230 milliroentgens per week, this will permit personnel access in the shelter for approximately one hour per man. For the uncontained model, dose rates in the shelter area above the primary tanks will not exceed 350 milliroentgens per hour; the maximum dose being at the shelter floor level on the reactor vessel axis. A summary of total dose rates from neutrons and gammas during full-power operation at the various points of interest depicted in Fig. III-27 (contained version) and Fig. III-28 (uncontained version) are given in Tables III-2 and III-3.

Total dose rates during operation from neutron and gamma radiation originating in the core, thermal shields, and reactor vessel were computed by the equations and methods described in MND-MPR-1581 for dose rates along the core radial centerline and axis. Significant radiation sources during operation are the reactor vessel and internal components, and the primary loop coolant. Since the reactor vessel and thermal shield configuration have not been definitely ascertained, neutron and gamma source strengths estimated for the PM-1 reference design (MND-1558) were used for computation of dose rates from these sources. Revised estimates were obtained for source strengths of intrinsic activity of the primary coolant in the steam generator, pumps, and primary loop piping based on the contained version system design.

After reactor shutdown, shielding for personnel in the vicinity of the steam generator is provided by the primary shield water in the reactor package and the earth fill of the uncontained version and shield water in the package interconnect of the contained version. Lead shielding placed in the reactor package between the reactor vessel and steam generator package of the contained version was considered. It was found that this lead shield could be eliminated by the inclusion of shield water in the previously dry package interconnect. For steam generator maintenance operations, approximately 8 hours are required for primary system cooldown and drainage. Maximum dose rates in the steam generator package,

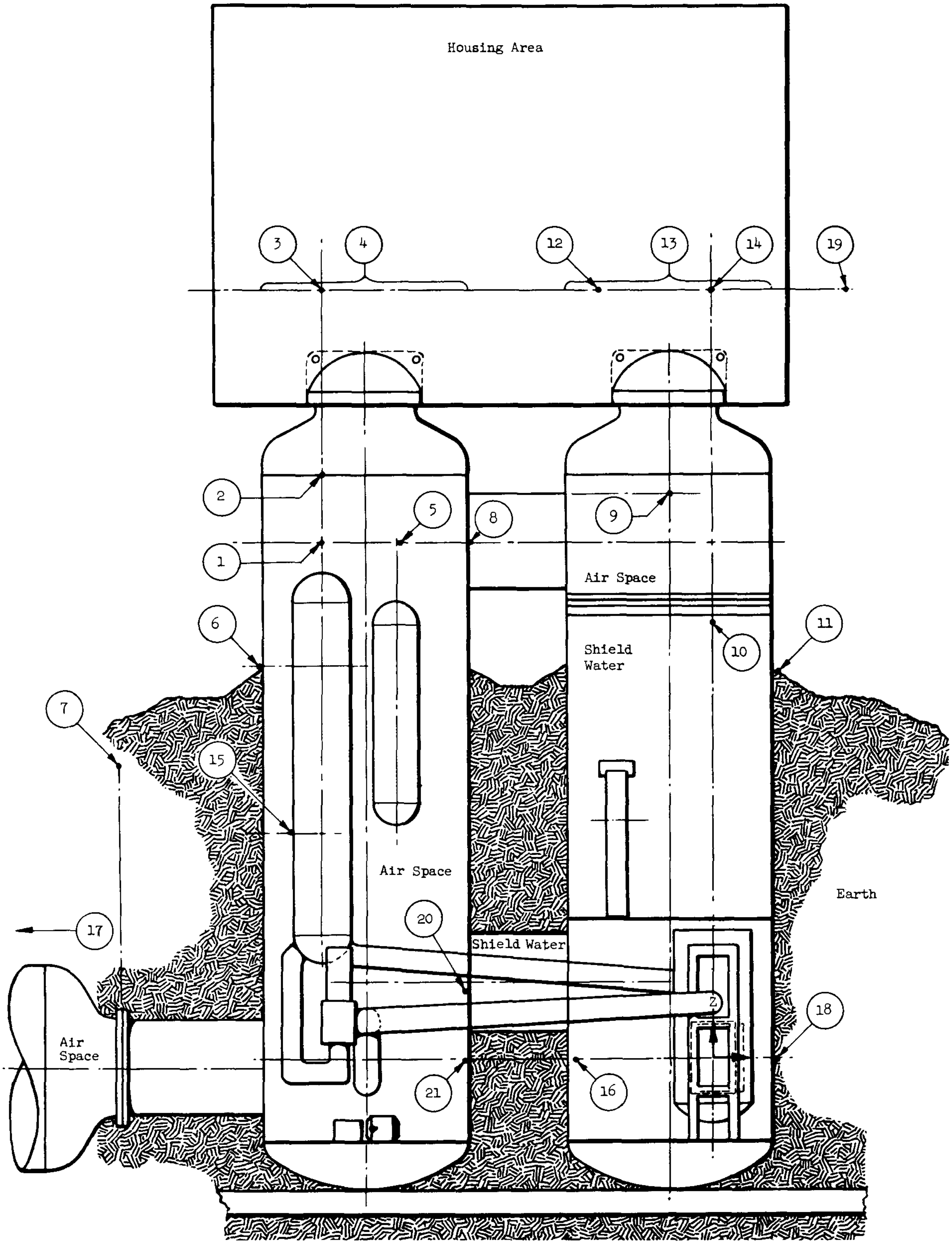


Fig. III-27. Dose Points--PM-1 Power Plant (contained)

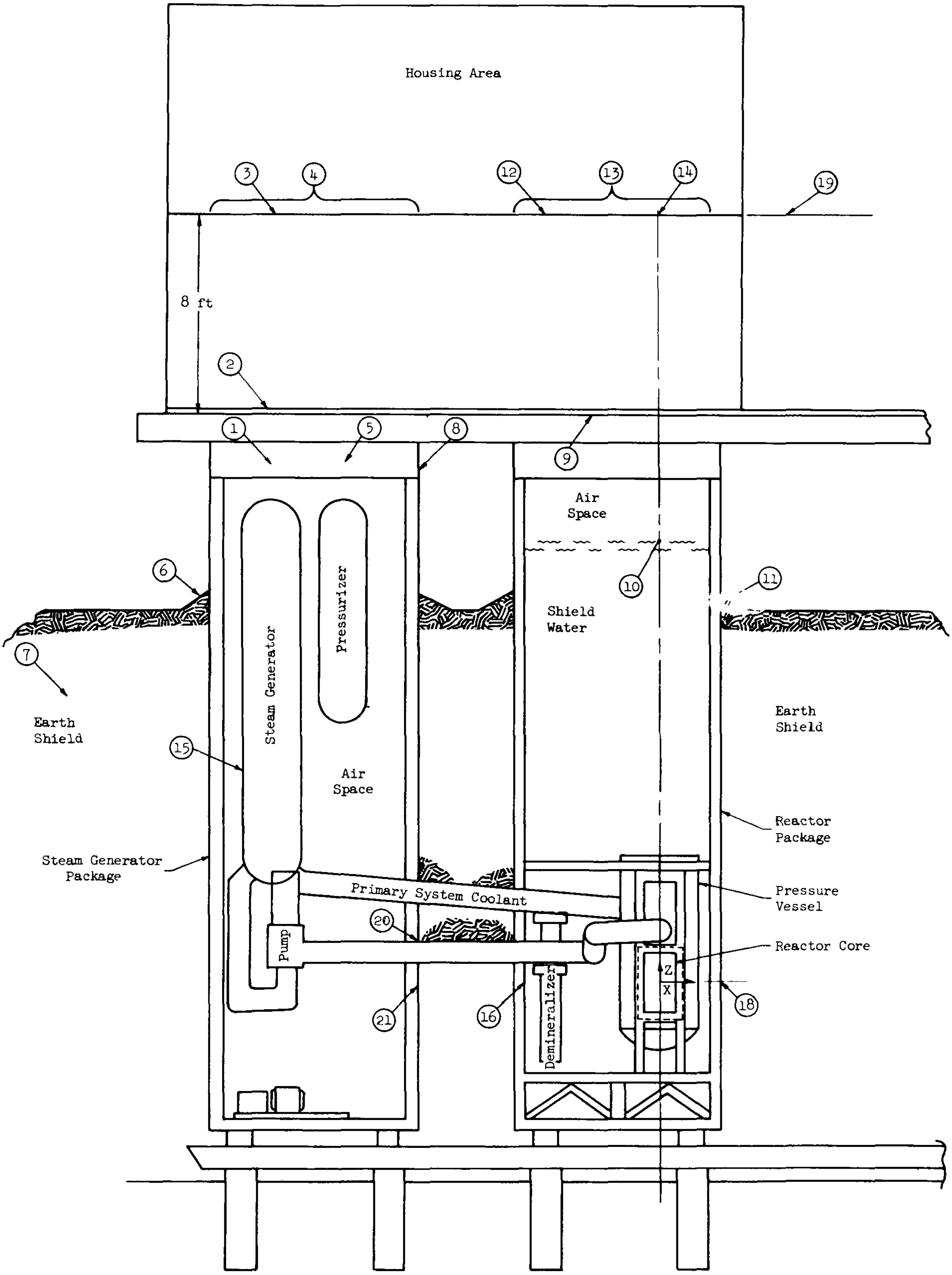


Fig. III-28. Dose Points--PM-1 Power Plant (uncontained)

TABLE III-2

Full-Power Operation Dose Rates--Contained Version

Dose * Point	Coordinates ** X (ft)	Z (ft)	Dose Rate (milliroentgens per hour) Steam Generator and Primary Piping	Reactor Vessel	Total
1	-16.35	21.65	200	-	200
2	-16.35	24.45	140	-	140
3	-16.35	32.2	26	-	26
4	-10.1 to -18.75	32.2	26	-	26
5	-13.0	21.65	70	-	70
6	-18.75	16.4	1.8×10^3	-	1.8×10^3
7	-24.75	12.25	2	-	2
8	-10.1	21.65	130	-	130
9	-1.8	23.7	150	174	324
10	0	18.35	-	300	300
11	2.65	16.1	-	1.23×10^3	1.23×10^3
12	-4.7	32.2	30	-	30
13	-6.1 to 2.55	32.2	-	90	90
14	0	32.2	-	90	90
15	-17.6	9.5	2.2×10^4	-	2.2×10^4
16	-5.6	0	-	5.0×10^6	5.0×10^6
17	-54.5	0.7	-	5.4×10^1	5.4×10^1
18	2.5	0	-	2.5×10^9	2.5×10^9
19	5.65	32.2	-	305	305

* Dose points are as shown on Fig. III-27.

** Coordinates of dose points are relative to the center of the core.

TABLE III-3

Full-Power Operation Dose Rates--Uncontained Version

	Coordinates **		Dose Rate (milliroentgens per hour)		
Dose * Point	X (ft)	Z (ft)	Steam Generator and Primary Piping	Reactor Vessel	Total
1	-16.35	21.65	200	-	200
2	-16.35	24.45	140	-	140
3	-16.35	32.2	26	-	26
4	-10.1 to -18.75	32.2	26	-	26
5	-13.0	21.65	70	-	70
6	-18.75	16.4	1.8×10^3	-	1.8×10^3
7	-24.75	12.25	2	-	2
8	-10.1	21.65	130	-	130
9	-1.8	23.7	150	174	324
10	0	18.35	-	300	300
11	2.65	16.1	-	1.23×10^3	1.23×10^3
12	-4.7	32.2	30	-	30
13	-6.1 to 2.55	32.2	-	90	90
14	0	32.2	-	90	90
15	-17.6	9.5	2.2×10^4	-	2.2×10^4
16	-5.6	0	-	5.0×10^6	5.0×10^6
18	2.5	0	-	2.5×10^9	2.5×10^9
19	5.65	32.2	-	305	305

* Dose points are as shown on Fig. III-28.
** Coordinates of dose points are relative to the center of the core.

8 hours after shutdown, from fission products in the core and activated fuel cladding, thermal shields, and reactor vessel will be less than 10 milliroentgens per hour for both the contained and uncontained version. Although the effect of gammas streaming through the primary piping to the steam generator package have not been computed specifically, it is anticipated that dose rates from this effect will be negligible. Tables III-4 and III-5 list dose rates at specific points of interest 1.67 minutes and 8 hours after reactor shutdown for the contained and uncontained version.

TABLE III-4

After-Shutdown Dose Rates--Contained Version

	Coordinate**		Dose Rate (milliroentgens per hour)	
Dose* Point	X (feet)	Z (feet)	1.67 Minutes After Shutdown	8 Hours After Shutdown
20	-10.1	2.85	187.0	6.0
21	-10.1	0.0	3.5	0.1

* Dose points are as shown in Fig. I-1.
** Coordinates of dose points are relative to the center of the core.

TABLE III-5

After-Shutdown Dose Rates--Uncontained Version

	Coordinate**		Dose Rate (milliroentgens per hour)	
Dose* Point	X (feet)	Z (feet)	1.67 Minutes After Shutdown	8 Hours After Shutdown
20	-10.1	1.7	187.0	6.0
21	-10.1	0.0	3.5	0.1

* Dose points are as shown in Fig. I-2.
** Coordinates of dose points are relative to the center of the core.

Primary coolant will be drained into a remote storage tank during steam generator maintenance, thus eliminating circulating corrosion product activity as a source of radiation to maintenance personnel. It is not expected that deposited activity in the steam generator and primary piping will be a major problem during maintenance operations.

After extended periods of reactor operation it may, however, be desirable to decontaminate the primary system. Methods of decontamination are presently being considered.

After-shutdown dose rates at the surface of the shield water will not exceed 2 milliroentgens per hour when all primary system components are in their respective normal-operating positions. Normal maintenance operation and refueling in the reactor package will be performed remotely from the surface of the shield water.

Preliminary calculations of gamma heating in the earth shield at the nearest point to the reactor vessel along the core radial centerline indicates excessive temperature (1800° F) in the earth. A 2-inch lead shield, or its equivalent in concrete, placed between the reactor vessel and earth in the vicinity of the core will, as reported earlier, significantly reduce maximum earth temperature.

Beyond a 6-foot perimeter about the primary system packages, the major source of radiation is air-scattering of gammas originating in the reactor vessel. The primary loop shielding and shield water above the core will sufficiently reduce the neutron flux in the air above the reactor so that neutron scattering is negligible. In order to determine whether excessive dose rates occur in the vicinity of the primary systems packages, the air-scattered gamma dose rate was evaluated at a point 35 feet from, and in the same horizontal plane as, the center of the core. The total air-scattered dose rate during 10-megawatt operation at the above-mentioned dose point will be 5×10^{-4} milliroentgens per hour. A similar calculation was performed at a proposed control room site located at a horizontal distance of 42 feet from the core axis and 28 feet above the reactor core. Air-scattered dose rates at this point were computed to be less than 2×10^{-3} milliroentgens per hour. The results of these calculations indicate that the 13 feet of shield water above the reactor vessel head efficiently reduces operating dose rates to negligible amounts beyond a 10- to 15-foot radius from the primary packages.

Spent fuel handling.- A spent fuel removal procedure was established which involves remote handling of the fuel within the shield water approximately 8 hours after reactor shutdown. With the pressure vessel head removed, the dose rate from fission product activity was evaluated at the surface of the shield water on the core axis and found to be less than 2 milliroentgens per hour. Loading of the fuel core or a portion of the core into a storage cask will necessitate raising it from the reactor vessel to a height such that the top of the active portion of the fuel tubes are within 65 inches of the shield water surface. For loading of bundles into storage casks, the core will first be raised from its operating position to a rack above the reactor vessel. The relative positions of the reactor vessel, storage cask, and various cores and bundles in the contained and uncontained versions are shown in Figs. III-29 and III-30.

Dose rates on the axis of the core and fuel bundle have been computed for the core and bundle as a function of water thickness above the source at 8 hours after shutdown. From plots of dose rate given in Fig. III-31, dose rates at the surface of the shield water for core and bundle in various positions may be determined. A bundle consists of approximately 1/6 of the entire core. In using Fig. III-31 to determine dose rates at the surface of the shield water, dose rates from the core and from a bundle removed to another position must be summed to determine total dose rate. The assumption that the core with bundle removed is still a full core will give only a slightly conservative estimate of the dose rate at the surface of the water. Other activated components of the primary system in the reactor package such as the pressure vessel, thermal shield, primary piping, and high-pressure demineralizer were thought not to contribute significantly to the shield water surface dose rate 8 hours after shutdown. A summary of the shield water surface dose rates for the configurations in Figs. III-29 and III-30 for various core and bundle positions is given in Table III-5.

TABLE III-5

Maximum Dose Rates at the Surface of the Shield Water 8 hours After Shutdown

Configuration	Dose Rate (mr/hr)
(1) Core in operating position, reactor vessel head removed	2
(2) Core on rack above reactor vessel	56
(3) Entire core raised to highest position during cask loading operation	8×10^4
(4) Core on rack above reactor vessel, bundle raised to highest position during cask loading	2.2×10^4

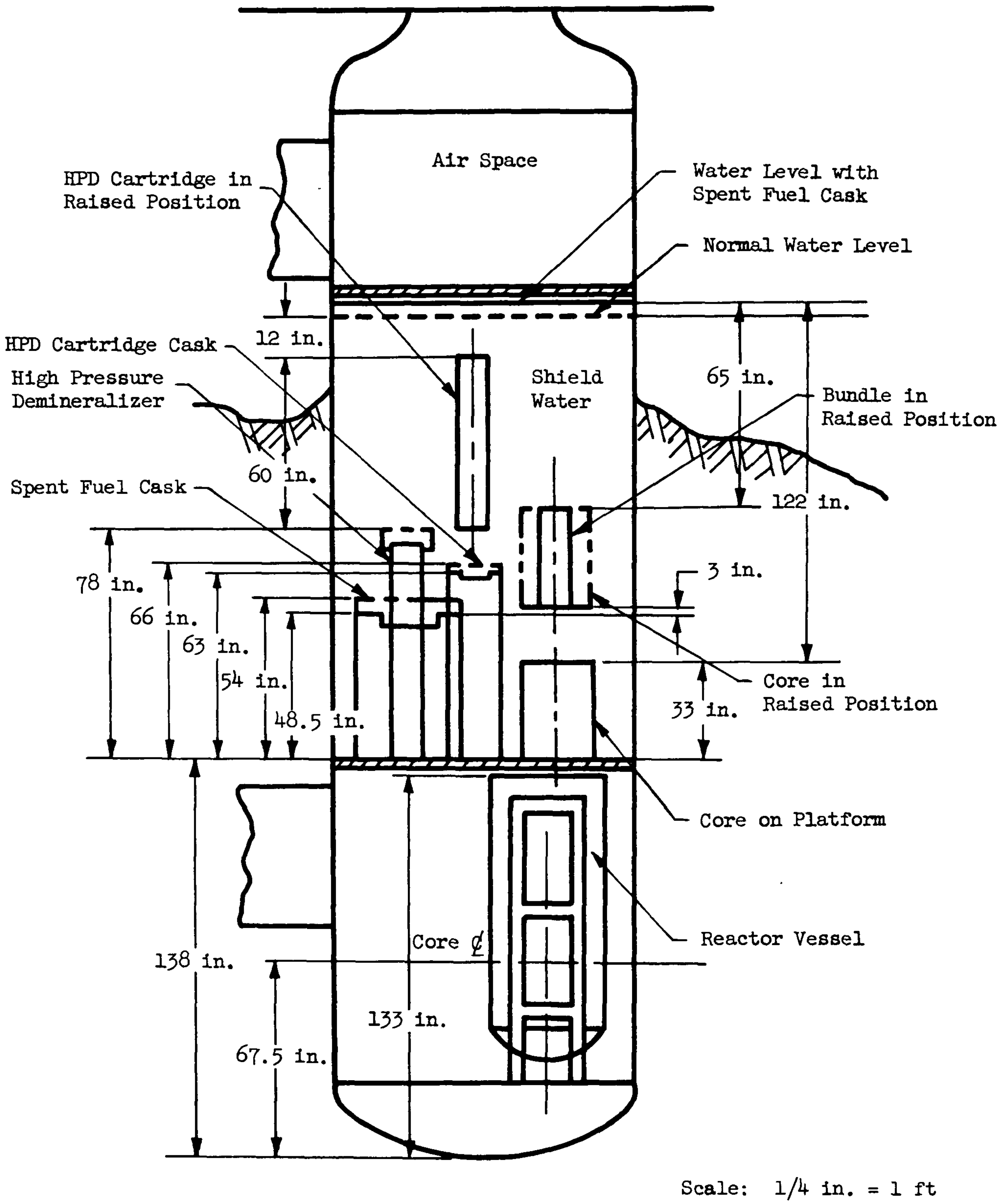


Fig. III-29. Reactor Package Showing Spent Fuel Cask (contained version)

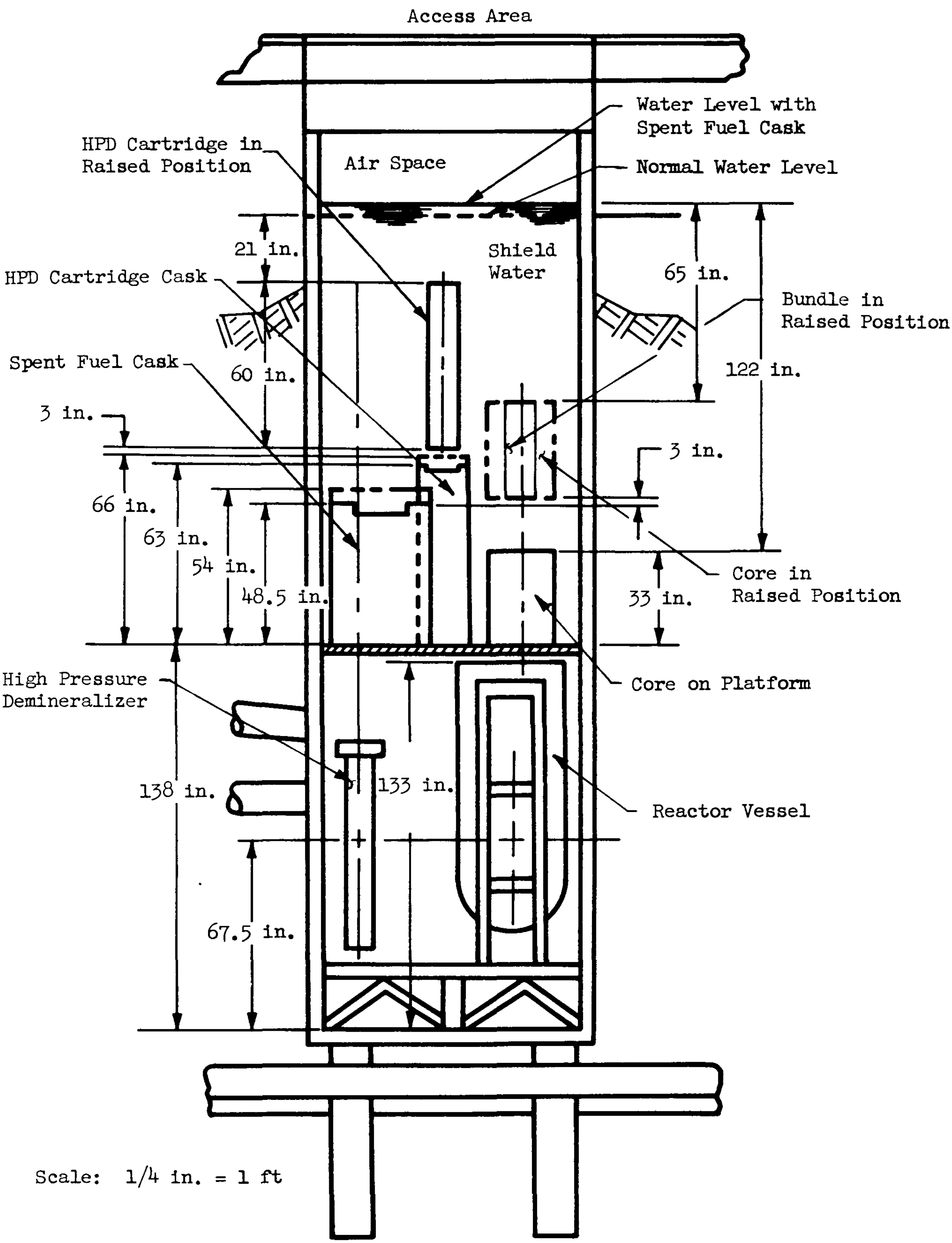


Fig. III-30. Reactor Package Showing Spent Fuel Cask (uncontained version)

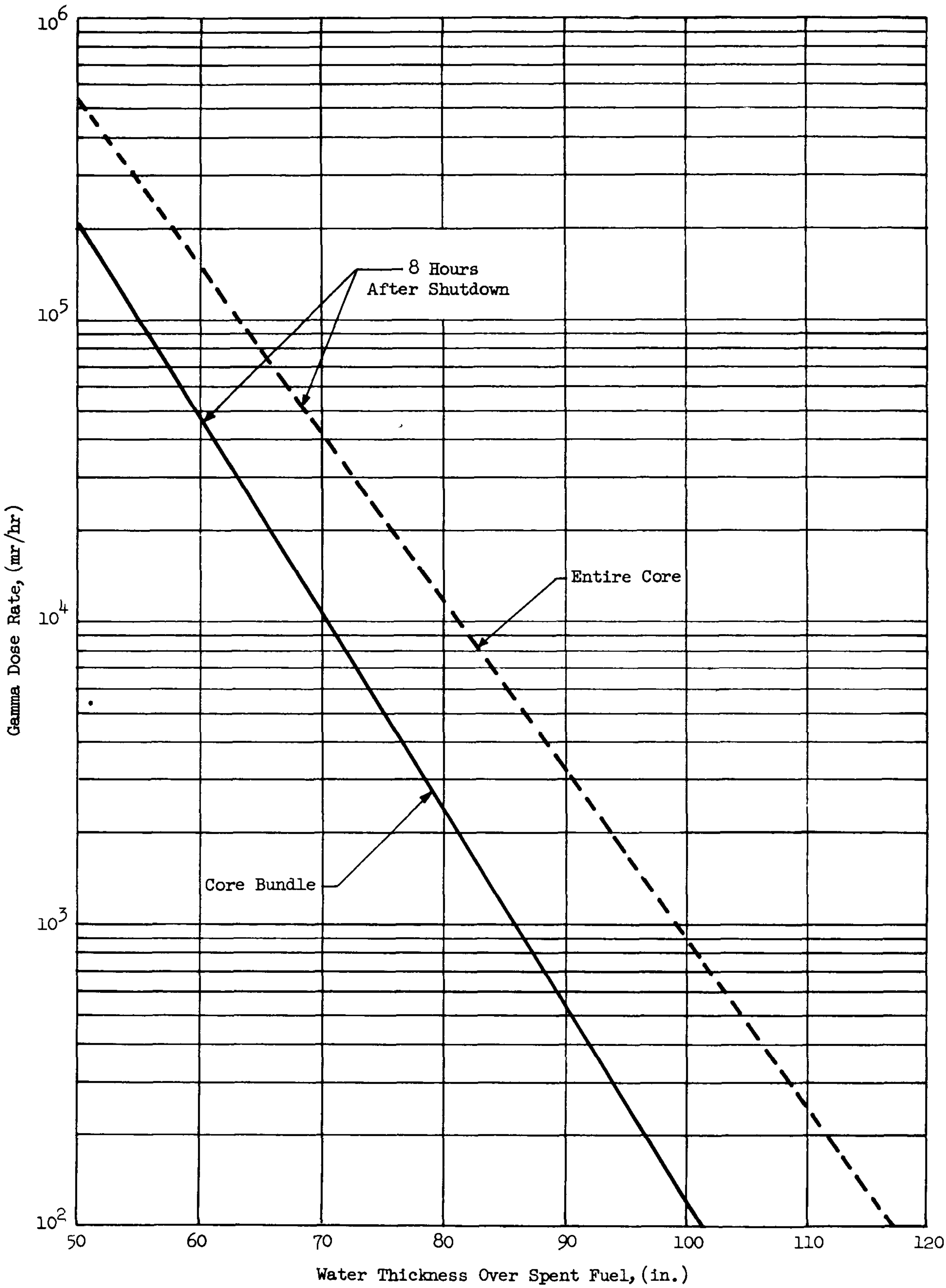


Fig. III-31. After-Shutdown Dose Rates in Water on the Axis of the Spent Core

TABLE III-5 (continued)

Configuration	Dose Rate (mr/hr)
(5) Core on rack, bundle in uncapped storage cask	56
(6) Entire core in uncapped storage cask	200

Excessive dose rates are seen to occur when raising the core or bundle to the height required for loading into storage casks. An estimated 5 inches of lead over the full core in the raised position was found to decrease the shield water surface dose rate from 8×10^4 to 100 milliroentgens per hour. The corresponding lead thickness for a raised bundle is 4 inches. In the uncontained version, the shield water level could also be further raised during the transfer of the bundle to the cask, thereby reducing the dose rate.

Lead cask designs were established for shipment of spent fuel after a 90-day cooling period. Two casks, one for the entire core (Cask A) and one for a single fuel bundle (Cask B), were designed to conform with ICC shipping regulations which require:

(1) Radiation not greater than 200 milliroentgens per hour any place on the outside of the carrier.

(2) Radiation not greater than 10 milliroentgens per hour at one-meter distance from the carrier.

(3) Radiation not greater than 11-1/2 milliroentgens for any 24-hour period at 15-foot distance from the carrier.

A third cask (Cask C), which will accommodate the entire core, was designed to minimize shipping weight. It will require remote handling for moving and loading at the time of fuel shipment for reprocessing. Preliminary designs of core and bundle casks are shown in Figs. III-32 and III-33. Maximum surface dose rates, lead thickness, and cask weights are listed in Table III-6.

Lead cask thicknesses were obtained from computation of dose rates in a cylindrical lead shield around the core and bundle. Dose rates through the lead from the dry core and bundle for 8 hours and for 90 days after shutdown are shown in Fig. III-34.

TABLE III-6

Spent Fuel Shipping and Storage Casks

	Overall Diameter (in.)	Overall Height (in.)	Time After Shutdown	Lead Thickness (in.)	Cask Weight (lb)	Dose Rates Surface (mr/hr)	1 Meter (mr/hr)	5 Meters (mr/hr)	10 Meters (mr/hr)
Cask A Entire core	A = 43.2	B = 54.2	8 hr	T = 9.1	26,000	2,500	314	24.5	6.8
			90 days	9.1	26,000	65	8.2	0.6	0.2
Cask B Core bundle	33.0	53.5	8 hr	8.75	16,000	2,350	183	12.5	3.4
			90 days	8.75	16,000	65	5.2	0.3	0.1
Cask C Entire core	A = 39.0	B = 50.2	8 hr	T = 7.0	18,000	45,000	4930	366	100
			90 days	7.0	18,000	1,250	137	10.2	2.9

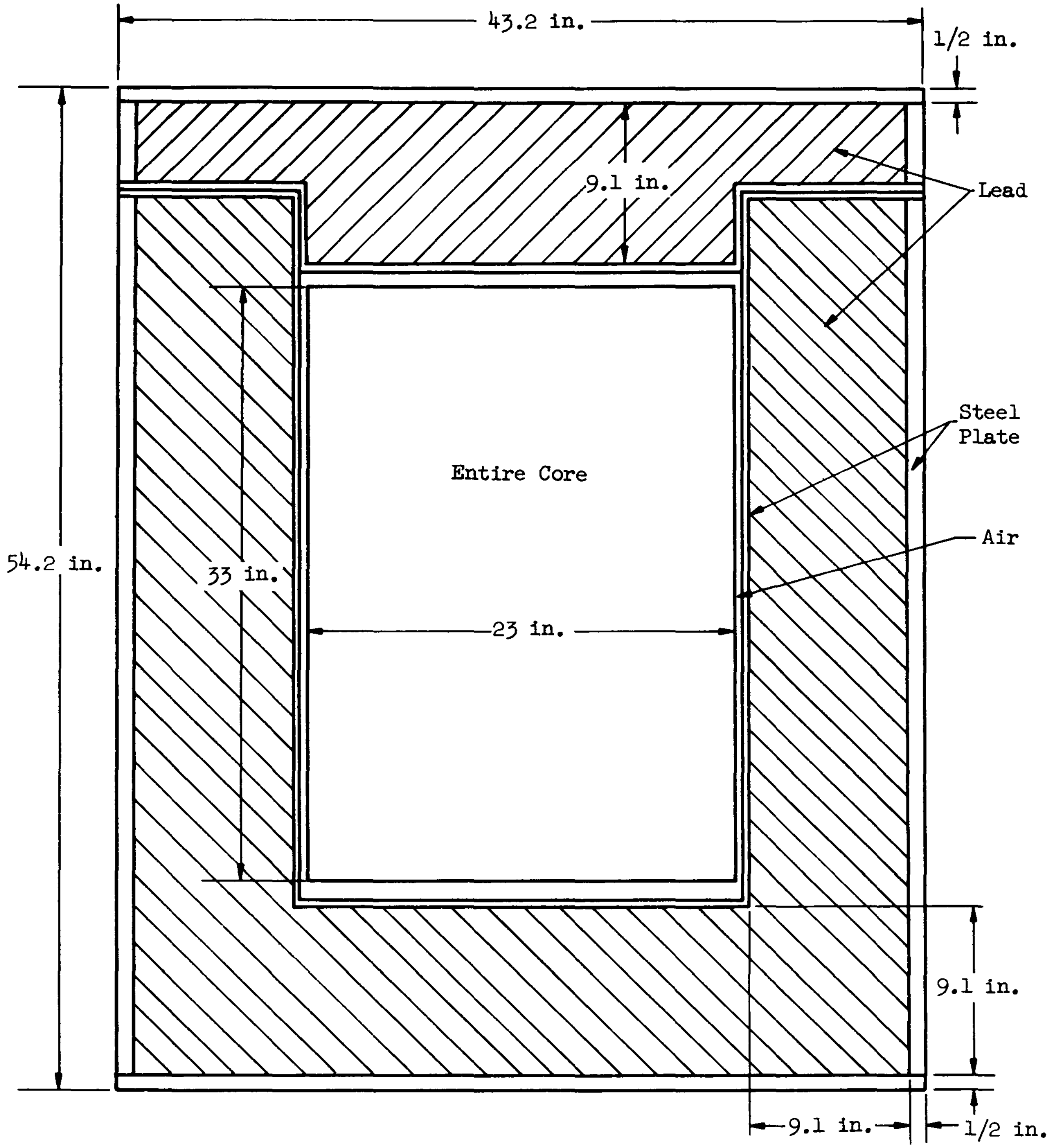


Fig. III-32. Spent Core Shipping Cask

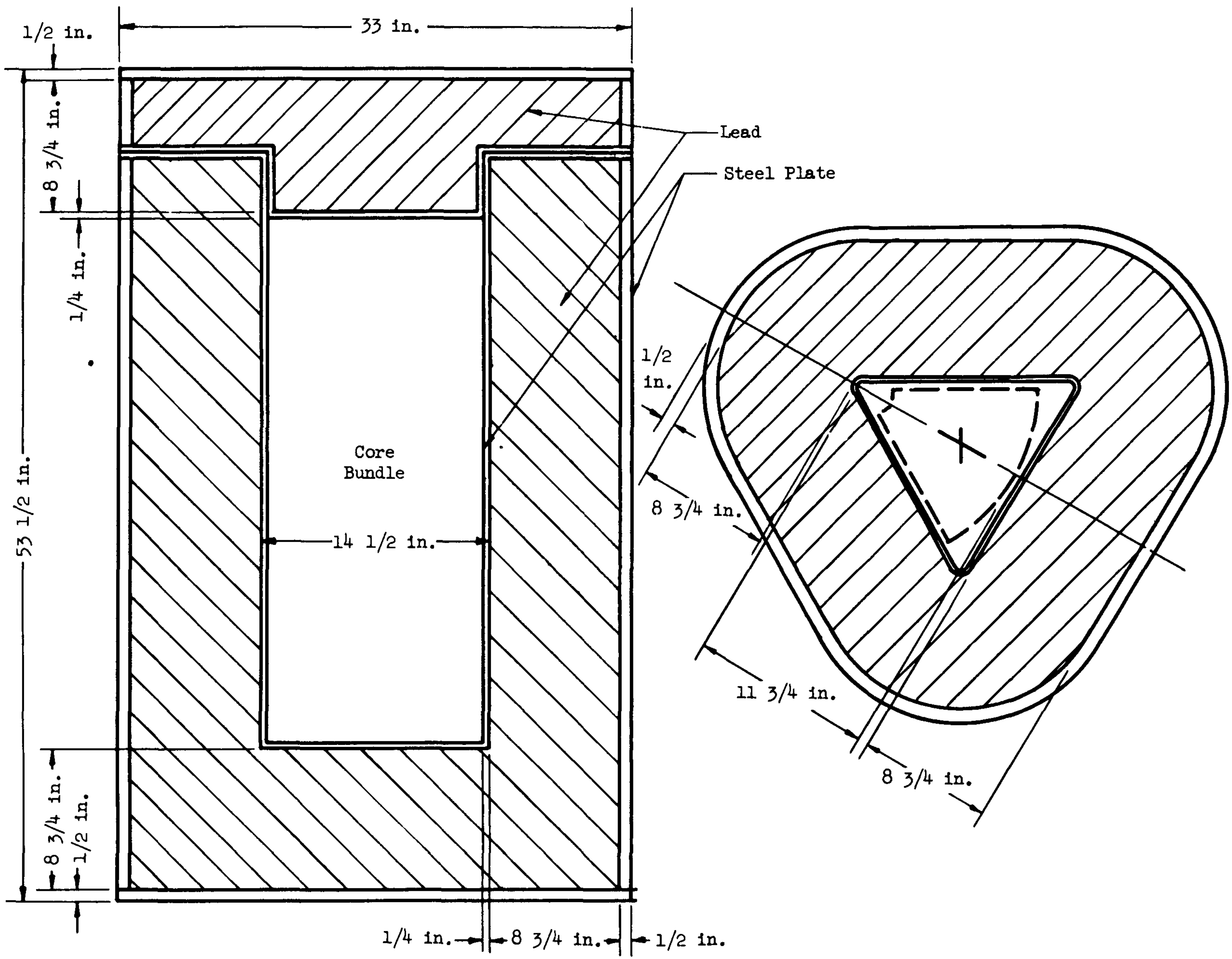


Fig. III-33. Spent Fuel Bundle Shipping Cask

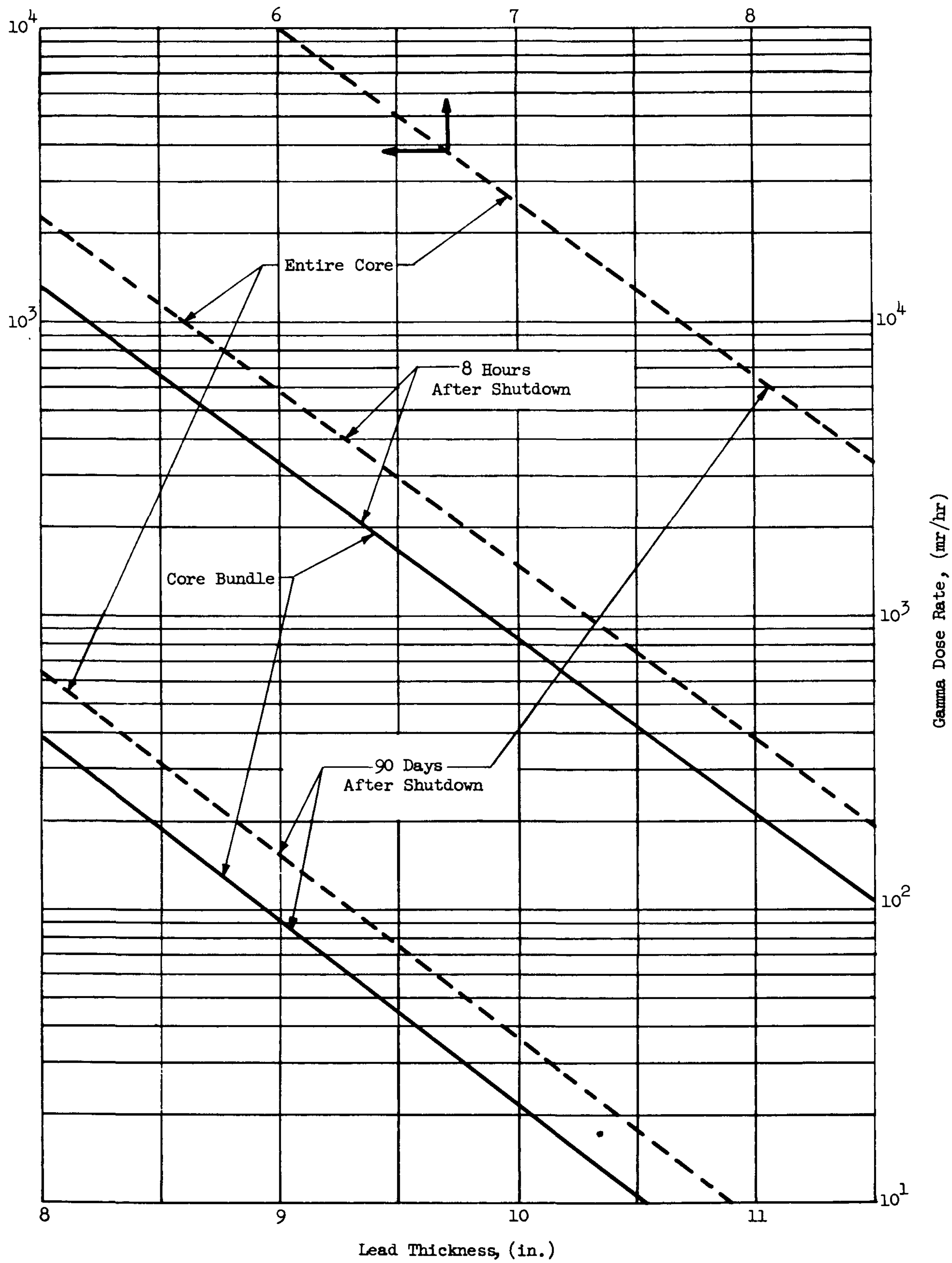


Fig. III-34. Radial Dose Rates in Lead from Dry Spent Core and Bundle

For computation of dose rates, fission products were assumed to be uniformly distributed throughout the core. In reality, a higher concentration of fission products near the center of the core is predicted by non-uniform burnout studies. Thus, computed dose rates through lead and water are somewhat conservative. Gamma source strengths were computed from the data of Perkins and King, which lists 123 major gamma- and beta-emitting fission products and daughter products with associated half-lives, fission yields, and beta and gamma energy release. Volume source strengths for 7 gamma energy groups after 2-year operation at 10 megawatts, and for several after-shutdown times, were obtained by use of the IBM-704 code for computation of fission product activity that was described in MND-1721, "A Program for the Computation of Fission Product Activity."

For attenuation calculations, the IBM-704 cylindrical volume source code was used. This code computes gamma flux from cylindrical or cylindrical annulus sources through concentric cylindrical shield regions by integrating over the volume of the source.

Gamma and beta energy release from fission products were evaluated and found to produce significant heating. After extended core operation, approximately 5% of the total operating power is from fission product decay. Table III-7 presents beta, gamma, and total energy release from fission product decay at various times after shutdown for the PM-1 core after 2 years of full-power operation. Because of the relatively short range of beta particles, essentially all of the beta energy will be released within the fuel elements. Gamma energy release will be in the core and surrounding absorbing media. When the fuel is placed in any of the storage and shipping casks described above, essentially all of the gamma energy will be absorbed by the fuel elements and cask.

The heating of the refueling cask remains to be evaluated in order to determine the total cooling requirements for spent core and cask. The heat released 8 hours after shutdown from a full core is approximately 61 kw. After 90 days, the heat released is approximately 7.15 kw.

The refueling scheme was established as follows:

(1) The core is removed from the open vessel and placed on the rack.

(2) The upper portion of the core structure is removed.

(3) The lower portion of the core, including the control rods, is transferred into the storage cask. During this operation, the head of the cask is kept between the operating floor and the core to reduce the dose rate at the operating floor.

TABLE III-7

PM-1 Fission Product Activity

Total Fission Product Activity After Two Years Operation at 10 Megawatts Thermal

Time After Shutdown	Total Activity (dis/sec)	Total Activity (curies)	Energy Release γ (mev/sec)	Energy Release β (mev/sec)	Energy Release Total ($\gamma + \beta$) (mev/sec)
2 minutes	1.10×10^{18}	2.97×10^{7}	8.20×10^{17}	8.00×10^{17}	1.62×10^{18}
30 minutes	7.80×10^{17}	2.11×10^{7}	4.70×10^{17}	4.30×10^{17}	9.00×10^{17}
1 hour	6.90×10^{17}	1.86×10^{7}	3.85×10^{17}	3.50×10^{17}	7.35×10^{17}
4 hours	5.30×10^{17}	1.43×10^{7}	2.50×10^{17}	2.20×10^{17}	4.70×10^{17}
8 hours	4.55×10^{17}	1.23×10^{7}	2.00×10^{17}	1.70×10^{17}	3.70×10^{17}
1 day	3.60×10^{17}	9.73×10^{6}	1.50×10^{17}	1.13×10^{17}	2.63×10^{17}
30 days	1.23×10^{17}	3.32×10^{6}	4.60×10^{16}	4.20×10^{16}	8.80×10^{16}
90 days	6.95×10^{16}	1.88×10^{6}	1.95×10^{16}	2.58×10^{16}	4.53×10^{16}
6 months	4.20×10^{16}	1.13×10^{6}	8.60×10^{15}	1.75×10^{16}	2.61×10^{16}
1 year	2.22×10^{16}	6.00×10^{5}	3.00×10^{15}	1.03×10^{16}	1.33×10^{16}

(4) The head of the cask is secured and the cask is lifted until its top is just above the surface of the shield water.

(5) The auxiliary cooler is placed on the top of this cask. This is an airblast-type cooler which is sized to reject more than 61 kw of heat with a circulating fan operating.

(6) The cask is transferred to the storage area in the primary housing. Coolant lines which connect to the shield water cooler are connected to the cask to provide cooling. The integral cooler is then used for backup.

(7) After 90 days, the cask may be shipped. The heat produced is such that the integral cooler can dissipate it without the use of auxiliary power.

(8) The single cask containing the full core will be shipped as one plane load. The weight of the cask is limited to approximately 22,000 pounds since the weight of the cask plus core, water, auxiliaries, and skids must not exceed 30,000 pounds. This results in a cask approximately 8 inches thick. The approximate radiation levels are:

Surface	~450.0 mr/hr
1 meter away	~35.0 mr/hr
5 meters away	~2.5 mr/hr
10 meters away	~0.7 mr/hr

after 90 days. These values are in excess of ICC regulations but are reasonable for military transport. The alternative is to make more plane trips, provide more casks, or increase the cooling period. The dose rates in the crew area of the C-130 aircraft will be within tolerance during the entire flight.

The problem of transporting a full core and the actual final design of the cask to remove heat and provide adequate shielding remains. Sufficient work has already been done to indicate that such a cask can be designed. The problem of transporting a full core has been evaluated. There appear to be attainable solutions to the problem of maintaining the system subcritical. The exact AEC requirements that must be met will be established before final design is accomplished.

Primary loop coolant activation.- The major radiation source in the primary loop is the intrinsic activity of the loop coolant. In regions of high neutron flux in and surrounding the reactor core, the fast neutron reactions O^{16} (n,p) N^{16} and O^{17} (n,p) N^{17} produce radioactive N^{16} and N^{17} with half-lives of 7.35 seconds and 4.14 seconds respectively. Decay of N^{16} yields a 6.13 and a 7.1 mev gamma in the ratio of 12.5 to 1 for 82% of the disintegrations. Decay of N^{17} yields one neutron per disintegration; the decay neutron spectrum is peaked at 1.0 mev. In a closed cycle such as the PM-1 primary loop, saturation values for these activities are reached shortly after reactor startup; the number of disintegrations at any one point in the primary loop will be approximately constant for a given operating power. Saturation specific activity at any point in the primary loop was computed using methods set forth in TID-7004.

Neutron and gamma volume source strengths at various points within the primary loop with units of mev/cm^3-sec were determined to be as follows:

	Neutrons	Gammas
Pressure vessel outlet nozzle	7.5×10^2	4.8×10^7
Inlet to steam generator	6.4×10^2	3.3×10^7
Outlet to steam generator	4.3×10^2	2.6×10^7

For computation of dose rates from the steam generator, the portion containing the primary water, tubes, and secondary water surrounding the U-tubes was assumed to be a homogeneous mixture. Using the appropriate weight fractions of stainless steel, primary water, and secondary water, gamma absorption coefficients were computed by standard methods given by Goldstein in "The Attenuation of Gamma Rays and Neutrons in Reactor Shields." The steam generator shell and the secondary water above the U-tubes contain no significant source of neutrons and gammas and were considered as shielding for attenuation calculations. The cylindrical equivalent line source solution (TID-7004) was used for the computation of dose rates. Dose rates from neutrons are negligibly low. Gamma dose rates at various points of interest were given in Tables III-2 and III-3.

For computation of dose rates from primary piping, the cylindrical equivalent line source solution was used. The coolant N^{16} gamma dose rate at the surface of the reactor vessel outlet primary piping during full-power operation is 80.0 roentgens per hour. The surface dose rate is proportional to the decay factor $e^{-\lambda t}$ and will decrease by a factor of 0.64 at the inlet to the pressure vessel giving a dose rate of 51 roentgens per hour at this point.

Because of the relatively short half-lives of N^{16} and N^{17}, this activity decays rapidly after shutdown of the reactor and is down to negligibly low levels a few minutes after shutdown.

Reactor vessel gamma heating.- In a continuation of the effort to optimize reactor vessel and thermal shield configurations, reactor vessel gamma heating rates were determined for 6 configurations of stainless steel thermal shields and reactor vessels. Radiation heating rates through the reactor vessel wall along the core radial centerline were determined as follows, assuming full power (10 megawatt thermal) operation:

Total Gap (in.)	Thermal Shield Thickness (in.)	Pressure Vessel Thickness (in.)
6.0	3.0	2.25
6.0	3.5	2.25
6.0	4.0	2.25
9.0	1.5	2.25
9.0	2.0	2.25
9.0	2.5	2.25

The total gap listed in the above table is defined as the radial distance between the surface of the core and the inner surface of the reactor vessel. A single thermal shield was assumed for each configuration with an inner radius of 11.55 inches. The placement of the thermal shield adjacent to the core will, in general, result in higher heating rates than the same thermal shield placed adjacent to the reactor vessel. To be conservative, the above thermal shields are placed as close as possible to the core so that computed heating rates are a maximum for the given thermal shield thickness.

Heating rates through the reactor vessel wall in the form of a single exponential representation may be determined from Figs. III-35 through III-38 which show Q_o (heating rate at the inner surface of the reactor vessel at the core radial centerline in Btu/in.^{3}-hr) and β (attenuation coefficient along the radius through the reactor vessel wall in inches^{-1}).

$$Q(x) = Q_o e^{-\beta x} \qquad \text{Btu/in.}^3\text{-hr}$$

In the above equation, Q(x) is the radial heating rate and x is the radial distance (inches) through the reactor vessel measured from the vessel inner surface. Data is presented for the 6- and 9-inch total gaps. For intermediate gaps, a first approximation may be obtained by cross plotting and extrapolation of this data. The reactor core was assumed to have been operated continuously at 10 megawatts for 2 years. At this time, thermal neutrons in the core and surrounding media will be a maximum with resulting maximum gamma production and heating rates. Other pertinent core data are as follows:

PM-1 Core

Mean radius	11.3 in.
Height of active fuel region	30 in.
Operating temperature	463° F
Operating pressure	1300 psi

Composition

Material	Percent (by volume)
UO_2	1.5
Stainless steel (AISI 348)	15.0
Pressurized water (ρ = 0.82 gm/cm^3)	83.5

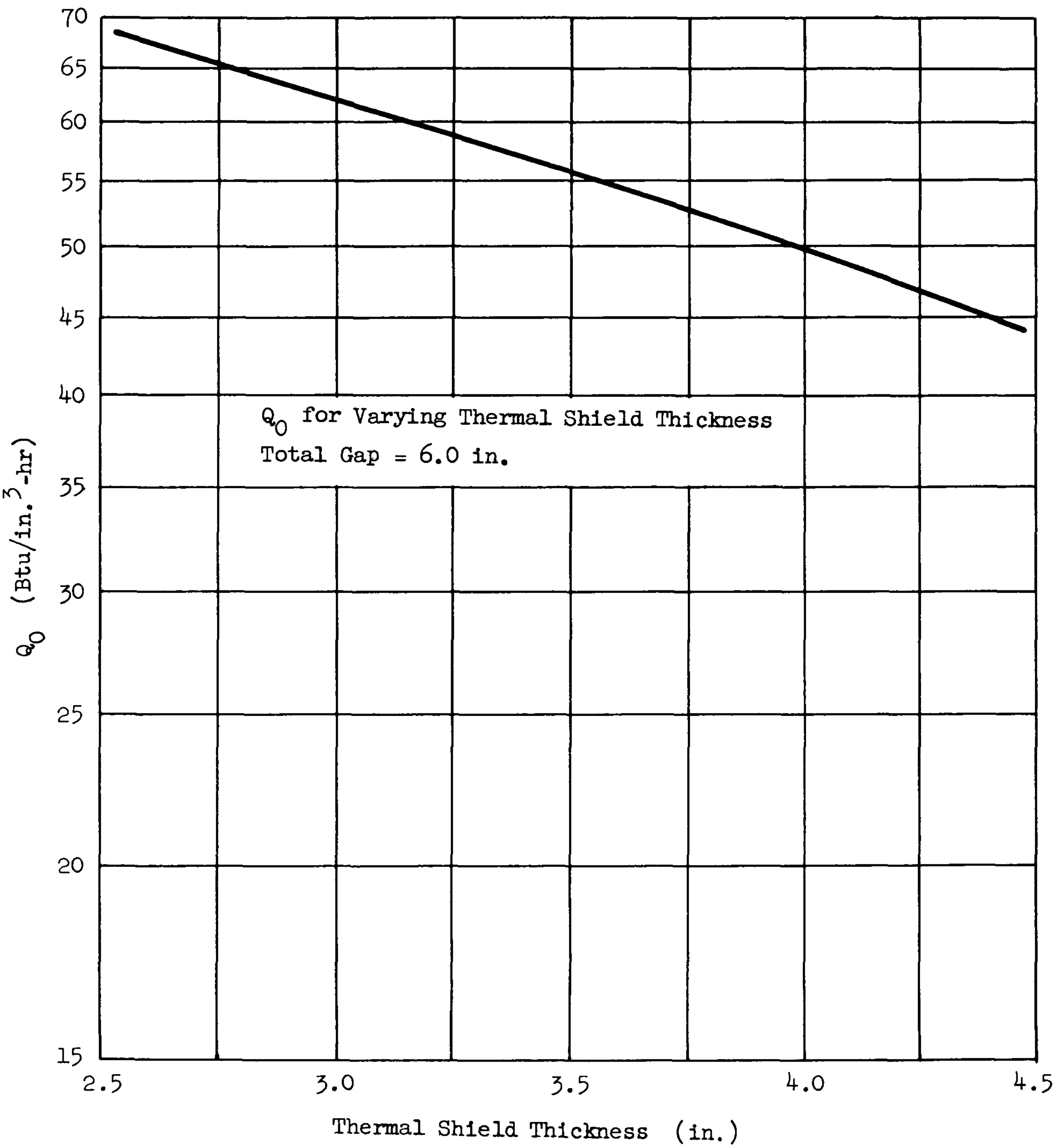


Fig. III-35. Heating Rate at Reactor Vessel ID as a Function of Thermal Shield Thickness

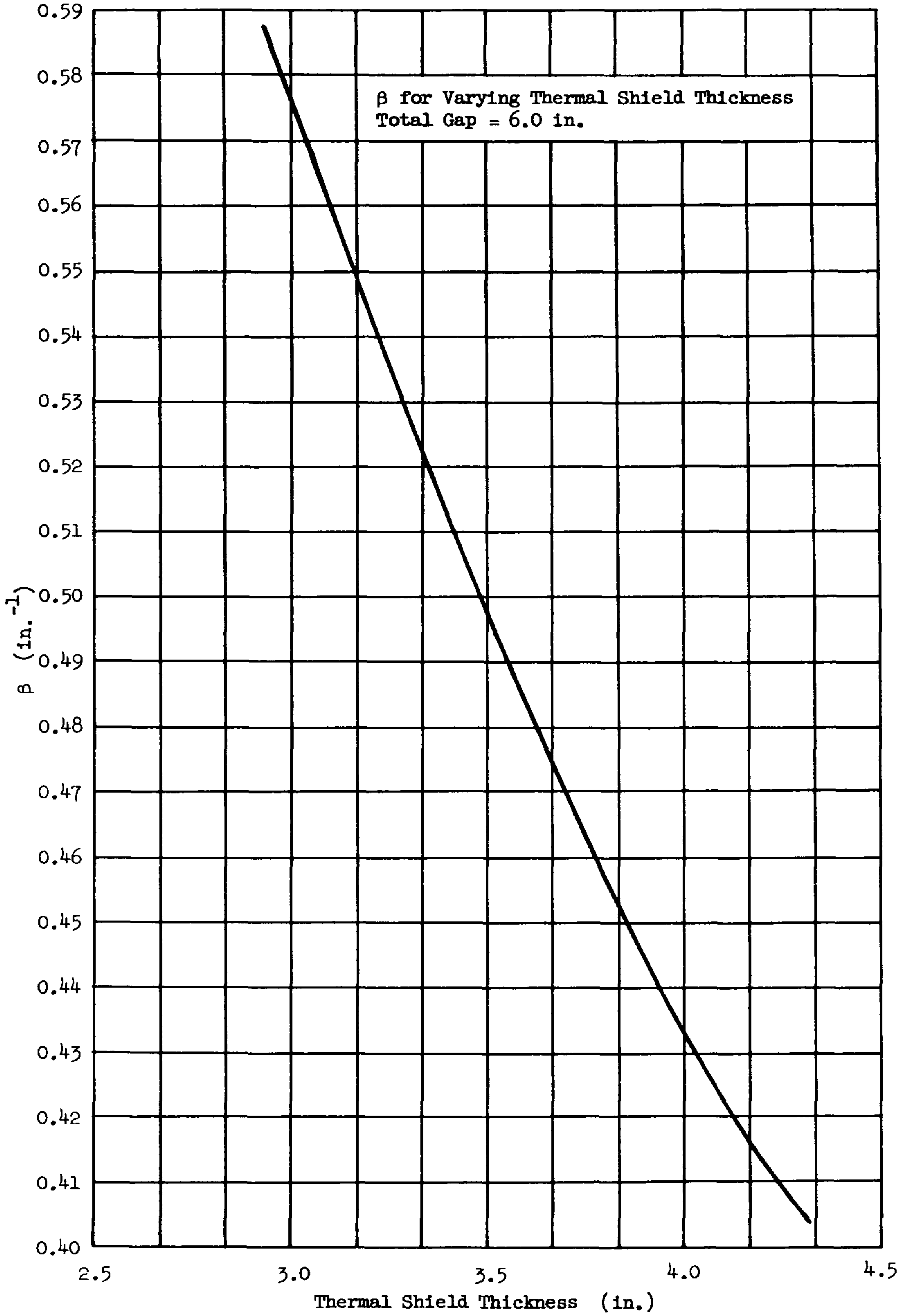


Fig. III-36. Vessel Attenuation Coefficient as a Function of Thermal Shield Thickness

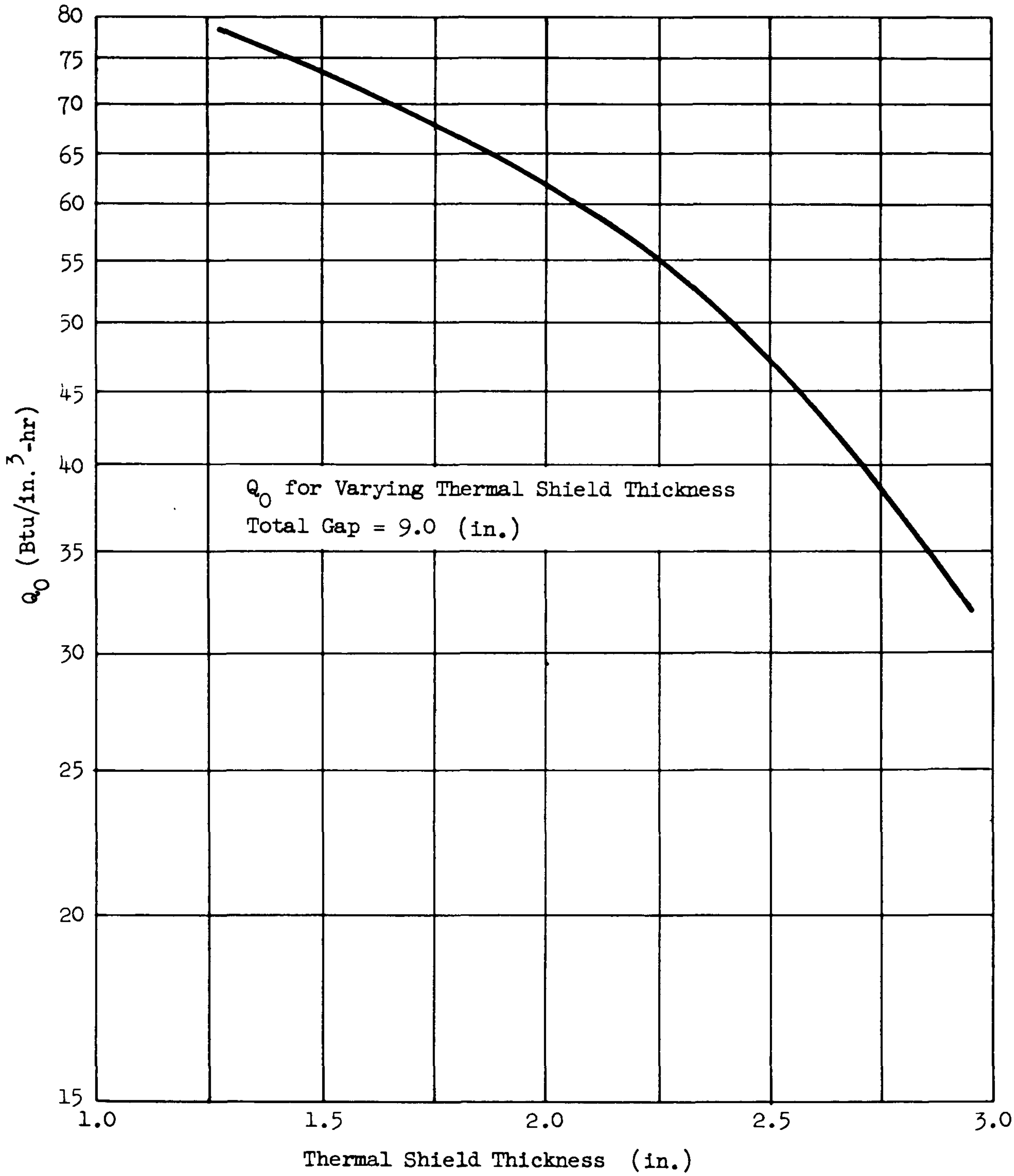


Fig. III-37. Heating Rate at Reactor Vessel ID as a Function of Thermal Shield Thickness

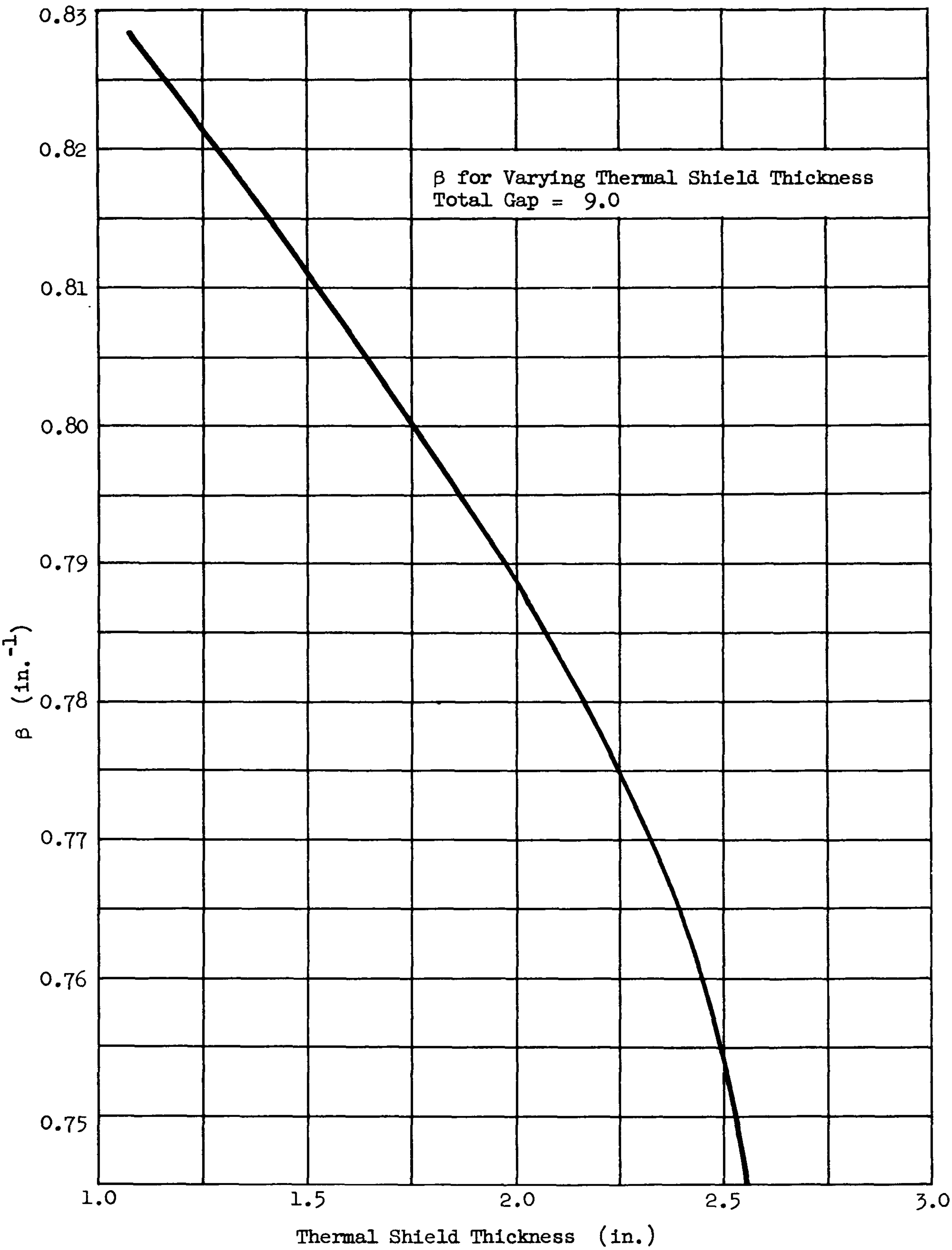


Fig. III-38. Vessel Attenuation Coefficient as a Function of Thermal Shield Thickness

Revised estimates of core gamma source strengths after 2 years of operation and of gamma absorption coefficients were computed using the above parameters. Prompt fission gamma activity was estimated from the experimentally measured spectrum presented in TID-7004. Fission product activity determined from the data of Perkins and King (Nuclear Science and Engineering, 3, 726 to 746 (1958)) by use of the IBM-704 fission product activity code (MND-1721) for 7 energy groups was scaled to give a total energy release of 2.28×10^{11} mev/watt-second as per the methods discussed in TID-7004. The scaling up was performed to account for very short-lived activities not accounted for by the Perkins and King data. Capture gammas created in the stainless steel fuel cladding were estimated from the data of Deloume (APEX 407). Gammas from induced activity in the fuel cladding were estimated from data in Table 3.7 of TID-7004. Source strengths of inelastic scattering gammas for stainless steel were computed from the following:

Material	Σ_{in} (cm^{-1})	E_o (mev)
Chromium	0.112	1.4
Iron	0.104	0.9
Nickel	0.073	1.45

A single gamma per inelastic scatter event was assumed released at the above listed energy (E_o). The total core gamma spectrum was grouped into 6 line energies with volume source strengths as follows:

Energy Group	Source Strength per Unit Volume: Gammas/cm^3-sec	mev/cm^3-sec	$\bar{E}$ (mev)	E_o (mev)
I	6.58×10^{12}	3.12×10^{12}	0.47	0.5
II	9.80×10^{12}	9.77×10^{12}	1.00	1.0
III	5.47×10^{12}	1.04×10^{13}	1.90	2.0
IV	5.94×10^{11}	1.78×10^{12}	2.99	3.0
V	5.16×10^{11}	2.52×10^{12}	4.88	5.0
VI	3.46×10^{11}	2.75×10^{12}	7.97	8.0

E_o is the assumed energy for gamma attenuation calculations. Gamma absorption coefficients were computed by the standard methods of weighting elemental data as presented in Goldstein.

Gamma attenuation calculations were performed by use of the IBM-704 line source and slab source codes. Methods of computation and geometrical assumptions used are essentially the same as those described in MND-M-1812 (1st PM-1 Quarterly), pages III-43 and III-44. Exceptions are as noted in the above text.

Estimates of the fast and thermal neutron flux in the reactor core and surrounding media were obtained by the use of GE IBM-704 Programs C-3 and F-3. A two-group analysis was used. Generally, neutron fluxes computed by Program F-3 are correct to within 20% in fuel regions. Greater error may occur in metal regions exterior to the core, due to the exclusion of the effect of inelastic scatter in the basic cross section data used. Presently, an attempt is being made to modify cross section data to include this effect. Modified cross sections and diffusion constants will be used to obtain three-group neutron fluxes in the reactor, core and surrounding media. If possible, results will be compared to experimental data to determine accuracy of the computed fluxes. Future gamma heating calculations will be based on fluxes computed with the modified cross sections.

The computed gamma heating rates are conservative due to the following assumptions:

(1) A single buildup factor over the total number of mean free paths from source to dose point was used in gamma attenuation computations. When two or more shield media were involved, the single buildup factor chosen was that of the material which gives the highest value of the gamma flux.

(2) Source strengths for inelastic scatter gammas and thermal neutron capture gammas in the reactor vessel and thermal shields were computed assuming the thermal neutron flux along the core radial centerline, which is the maximum value of the thermal neutron flux in the component. The thermal shields and pressure vessel were further assumed to be infinite slabs having an axially constant source distribution equivalent to the radial centerline values.

Improved methods would involve use of two-dimensional neutron flux codes and the inclusion of axial source distribution and finite cylindrical annulus geometry for gamma sources exterior to the reactor core and methods of estimating gamma buildup boundary effects. Present methods of analysis will be used for the current preliminary studies, since improved methods would involve excessive amounts of IBM-704 machine time.

As previously mentioned, no attempt has been made toward optimum positioning of thermal shields in the configurations of this analysis. Minimization of water gap thickness and displacement of thermal shields to a position adjacent to the reactor vessel would result in lower heating rates and correspondingly lower thermal stress. Future effort will be directed toward optimization of the amount of thermal shield required, and of the positioning of the thermal shield.

4. Core Design Studies

K. Dufrane, S. Kershaw

Core design studies involved a detailed determination of:

(1) A minimum core geometry consistent with the selected fuel element size, thermal and nuclear requirements.

(2) A refueling concept which minimizes the reactor downtime for both the initial and reloading operations.

(3) A method of accurately aligning the control rods with the actuators.

(4) Means of eliminating thermal stresses between members by combining hold-down with free expansion.

A reference core design was established.

Reference design description.- The overall core design is illustrated in Fig. III-39. The basic core configuration, both fuel and supporting structure, consists of 6 peripheral and one center fuel bundle, alignment spiders, a hold-down spring, and necessary flow baffles and supporting skirt. With the exception of the center bundle, individual positioning and support is provided each fuel bundle through the alignment spiders and the upper skirt. With this arrangement, the core may be handled and refueled either as a complete assembly or by single bundles. The center fuel bundle, supported from the pressure vessel head, was provided to make possible future incorporation of in-core instrumentation.

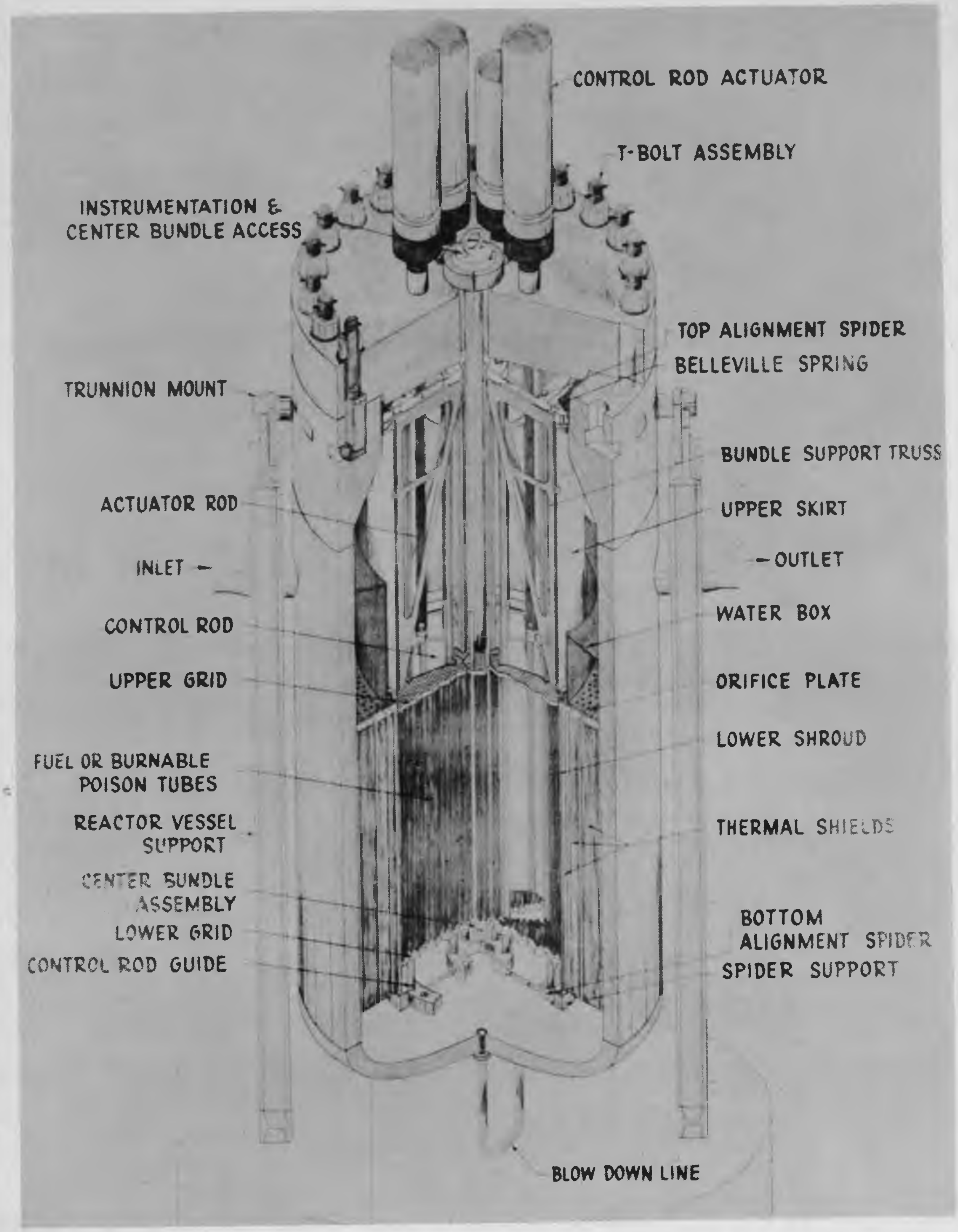


Fig. III-39. Reactor Pressure Vessel and Core--PM-1 Power Plant

From both a cost and design complexity standpoint, it is desirable to control the core with a minimum number of control rods--previously determined by nuclear considerations to be 6. It also appeared advantageous to subdivide the core into sections to facilitate manufacturing, handling and shipping. Since difficult alignment problems could be avoided by containing each control rod and its guides within a given segment, it was then necessary to subdivide the core into a minimum of 6 bundles.

These core subdivisions have a major effect on the geometrical arrangement of the fuel elements in the active core region. Two possible arrays are readily apparent: square and triangular. The square array has the advantage of reducing the number of orifice holes required to supply flow external to the elements, which reduces the design complexity of the inlet orifice plate. However, the square array, which also dictated the use of cruciform control rods, prohibited the core from being subdivided into identical bundles (exclusive of the central bundle). Because of this, triangular spacing utilizing Y-shaped control rods was selected for the reference core pattern. The manufacturing advantages gained by having the core assembled from bundles of a single type more than offset the orificing disadvantage. In addition, the triangular array provided a more compact core pattern which slightly reduced the required core diameter.

A second possible control rod configuration, adaptable to the triangular pattern, was briefly considered. This consisted of 3 cylindrical poison tubes (each replacing 7 fuel elements) ganged together and positioned by a single control drive mechanism. Lack of sufficient nuclear data on its control effectiveness made further investigation unfeasible.

The reference design provides support to the fuel region of each bundle through the control rod guides. These guides are, in effect, hung from the upper support truss. The advantage of this system may be summarized as follows:

(1) The amount of structural material contained within the active core region is limited to that already required to guide the control rods.

(2) Accurate alignment and support is provided at both the top and bottom of the control rod stroke.

(3) Stresses due to differential thermal expansions during temperature transients are completely eliminated in the fuel bundles, support skirt, and shrouds by allowing free expansion.

(4) The basic pressure vessel design is not affected nor in any way compromised by the support mechanisms.

Alternate methods of core support, both from the bundle midpoint and its lower portion, were evaluated. In general, a portion of the advantages mentioned above could be realized with each. However, none could duplicate all the advantages gained with the top support or offer additional advantages. Several decisive disadvantages were, in fact, noted.

Bottom support schemes increased the hold-down problem by one or a combination of the following:

(1) Hold-down through the control rod guides is undesirable as it puts these relatively long columns under compression loading.

(2) Increasing the guide cross section to eliminate the above objection or adding separate hold-down tubes displaces fuel in the active core region. This increases the core diameter and also complicates nuclear analysis.

(3) Hold-down external to the core (i.e., through the lower shroud) is difficult to achieve while maintaining the capability of removing individual fuel bundles. The methods examined tended to increase the refueling complexity as well as the minimum pressure vessel throat diameter.

Other general design disadvantages included:

(1) The addition of a heavy, load-bearing, support plate at the bottom increases cost, since a support plate is also required at the pressure vessel midpoint to complete the water box and to provide flow orificing for the thermal shields. The lower plate is difficult and complex to incorporate due to a combination of weld fabrication and volume limitations.

(2) A low point of support increases both the tolerance and thermal displacements that must be accommodated by Belleville or other types of hold-down springs.

Core support at the midpoint has similar hold-down disadvantages in addition to the following:

(1) The need for an additional alignment surface at the midpoint (in addition to the top and bottom), thereby increasing overall cost.

(2) Midpoint support must be sufficiently above the upper tube grid to minimize the effect of the structural cross bars on the core flow patterns. This lies in the plane of the inlet water box, thereby increasing its complexity.

Limitations and anticipated problems.- The following summarizes the general areas where additional detailed studies are required:

(1) Ability to replace the lower alignment spider for greater flexibility in future designs.

(2) Evaluation of the several manufacturing methods for securing the fuel tube to the lower grid.

(3) Determination of the overall alignment requirements for tolerance studies and for detailed evaluation of the present design and alignment methods.

(4) Evaluation of methods for separating control rod guides from the active core, thereby minimizing the shipping volume requirements of the spent fuel bundles.

(5) Detailed design analysis of the upper support truss.

Design features.- In considering the several designs discussed, the overall design features of the proposed reference design are briefly summarized.

(1) With the proposed design, each of the peripheral bundles is a complete self-contained unit with its control rod and guides, orifice plates and complete supporting structure. Each is identical and completely interchangeable with the obvious cost advantage for single-unit manufacturing and of later mass production. The overall logistic problem is greatly simplified by minimizing the type and number of required spares. In addition, in future reactor field installations where the in-core instrumentation would be eliminated, the peripheral bundles can be easily extended to include the central area. Thus, only one type of fuel bundle will be required.

(2) The unit-bundle approach offers a good deal of versatility in the refueling and assembly operations. With the single-bundle design, both the refueling equipment and operational techniques are simplified to the greatest possible extent. Another feature of the proposed refueling process is that a single bundle, including its control rod, may be easily replaced if the occasion arises.

(3) Because both spiders and the core shroud are completely removable, maximum flexibility is built into the reactor design for the possible incorporation of future overall system changes.

(4) In supporting the core bundle from the top, the weight of the core places each control rod guide into stable tension loading. In addition to allowing free thermal expansion of all parts, the overall design concept is simplified (including refueling, handling, storage, and the pressure vessel design).

5. Reactor Pressure Vessel Design Studies

H. Brainard, J. Goeller

Preliminary design of the reactor pressure vessel and its components, exclusive of the core, was completed with the exception of the closure seal configuration, the thermal insulation and the thermal shields. These items will be designed under Task 4 during the subsequent quarter.

The main efforts during the next quarter will involve:

(1) Final stress analysis of the reactor vessel and its external and internal components.

(2) The writing of final specifications.

(3) Preparation of layout and detail drawings.

(4) Contacting vendors and obtaining bids.

(5) Determining the final weight of the complete reactor vessel.

Preliminary vessel design.- The vessel shape was established as basically cylindrical with a 2-to-1 ellipsoidal bottom and a flat circular head. Other vessel shapes were considered. In particular, spherical shapes were investigated but found unfeasible because girth diameters exceeded those allowed by packaging requirements, and because the internal core and thermal shield supports were relatively complicated. A flat head was chosen because the heavy reinforcement necessary for control rod penetrations made it impossible to effect reductions in weight or complexity by using an ellipsoidal or a hemispherical head.

The reactor vessel dimensions were tentatively determined to be as follows:

Vessel OD	40 in.	(Determined by core and thermal shield requirements)
Wall thickness	2-5/16 in.	(ASME Boiler and Pressure Vessel Code)
Overall height (insulation not included)	99 in.	
Head thickness	7 in. at flange--7-1/2 in. at vessel	(ASME Boiler and Pressure Vessel Code)
Material	AISI Type 347 stainless steel	(Result of radiation damage considerations)

Layouts made early in the quarter (see Fig. III-39) indicated that an internal flange must be used at the vessel main closure, since the head bolt circle diameter has to be held to a minimum to provide the space necessary for refueling operations.

A T-bolt fastener design was developed to replace the steel bolts that would ordinarily be required with an internal flange design (see Fig. III-40). The T-bolt design will be investigated further during the next report period. A detailed stress analysis will be made, manufacturing processes will be investigated, and a test program will be specified.

Primary loop arrangements completed during the reporting period showed that the expansion loops, originally employed to accommodate thermal expansion of the primary piping, congested the working area at the reactor vessel to an unacceptable degree. It was decided to trunnion-mount the pressure vessel on a double A-frame support. This will allow the vessel to swing slightly when the piping expands. Inlet and outlet nozzles were fixed diametrically opposite to one another and in the same horizontal plane to minimize torque resulting from vessel movement. In order to mount the nozzles in the same plane, the water box was tapered from 18 inches at the inlet nozzle to 6 inches at the outlet nozzle.

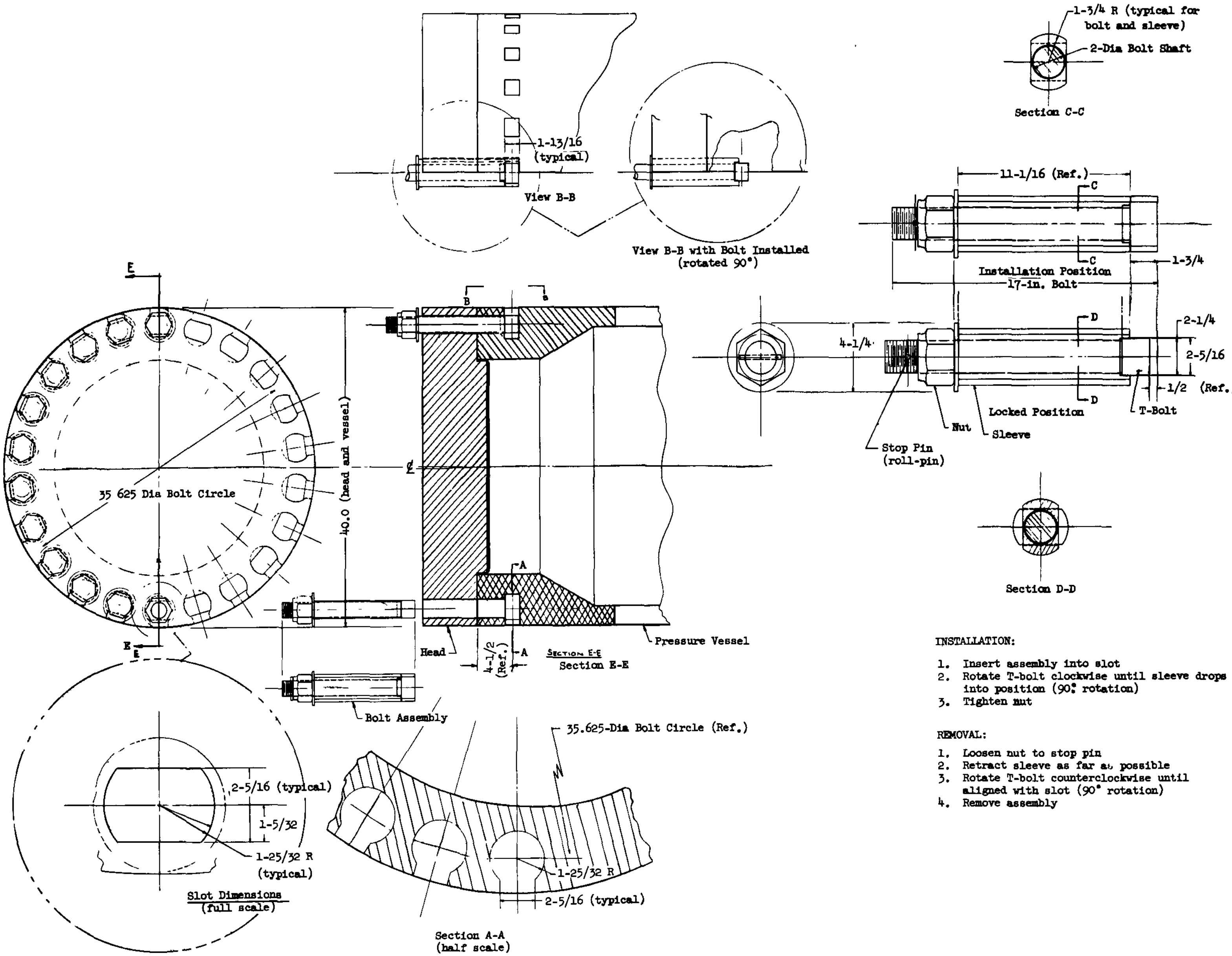


Fig. III-40. Pressure Vessel Bolting Scheme--PM-1 Power Plant

Radiation damage.- A radiation damage study of various types of steels was completed; the data will be incorporated in Martin-Nuclear Report No. MND-M-1901, "Investigation of Reactor Pressure Vessel Materials." As a result of this study, AISI Type 347 (high alloy austenitic steel) was chosen for the pressure vessel material. Materials having a body-centered cubic lattice structure normally exhibit a rise in their nil ductility transition temperature after significant neutron irradiation. Type 347 stainless steel has a face-centered cubic-lattice structure, and, consequently, exhibits no nil ductility transition temperature. It is believed by most authorities that Type 347 stainless steel will be least affected by neutron irradiation.

Vessel seal.- A number of seal manufacturers were consulted while gathering data for an evaluation of possible reactor vessel closure seals. The type of seal to be utilized has not yet been determined, but the use of a split O-ring type (trade name--"Hi Ceal") appears likely. This type of seal requires that only a minimum preload be imposed on the head bolt. Since it is fabricated from stainless steel, it has enough rigidity to allow remote underwater installation with relative ease.

Stress analysis.- A preliminary stress analysis of the pressure vessel and its components was performed; the vessel was found to be satisfactory for the following conditions:

(1) Mechanical loadings

- (1) Internal design pressure of 1500 psi.
- (2) External loads and torques imposed on the vessel nozzles by the piping.
- (3) Reaction loads at trunnions due to weight of vessel and horizontal piping thrusts.

(2) Thermal loadings

- (1) Temperature of 600° F.
- (2) Internal gamma heating based on data obtained from shielding studies.
- (3) Transient temperature conditions during startup (based on heat transfer analog studies).

Preliminary stress analysis of the reactor vessel was completed under the provisions of Section VIII of the ASME Pressure Vessel Code. Analysis of areas not covered by the code were considered, using methods suggested in "The Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components" issued by the Bureau of Ships, U.S. Navy.

Structures and supports were analyzed using standard AISC procedures.

Piping stresses and end reactions at the reactor vessel nozzles were determined under the provisions of the "Code for Pressure Piping" and "Tube Turn" analysis methods (Elastic Center Method) which are accurate to within 10%.

The "coded" techniques listed in the preceding paragraph all carry a safety factor of 4 in allowable stresses. U.S. Navy BuShips procedures, while less stringent concerning allowable stress values, are more rigorous in derivation and more detailed with regard to cyclic and radiation heating stresses.

6. Primary Loop Design Studies

P. Mon, J. Todd, R. Manoll, J. Goeller

Primary loop design studies involved the preparation of preliminary studies and designs for the contained and noncontained configurations of the primary loop, the establishment of a method for installing and erecting the primary loop packages and related support structure; and the establishment of the major contents of the packages.

During the next quarter (under Task 4), final refueling, erection, primary piping and superstructure layouts will be made for the contained and noncontained plants; assembly drawings of the entire primary loop system for both versions will be begun; detail drawings of the support base, grating and the erection trunnions will be made; drawings of the various transfer casks will be started; and structural drawing of the superstructure will be initiated.

Primary loop configuration.- Two basic containment configurations were analyzed during the first portion of the preliminary design period. Both configurations involved multiple tank containment using three 8-foot, 8-inch dia x 30 foot long tanks to obtain the necessary containment volume. One configuration used 3 vertical tanks mounted on a prepared base (see Fig. III-41). The other configuration consisted of two vertical tanks mounted on a prepared base with the third tank mounted in a horizontal position on top of the vertical tanks (see Fig. III-42).

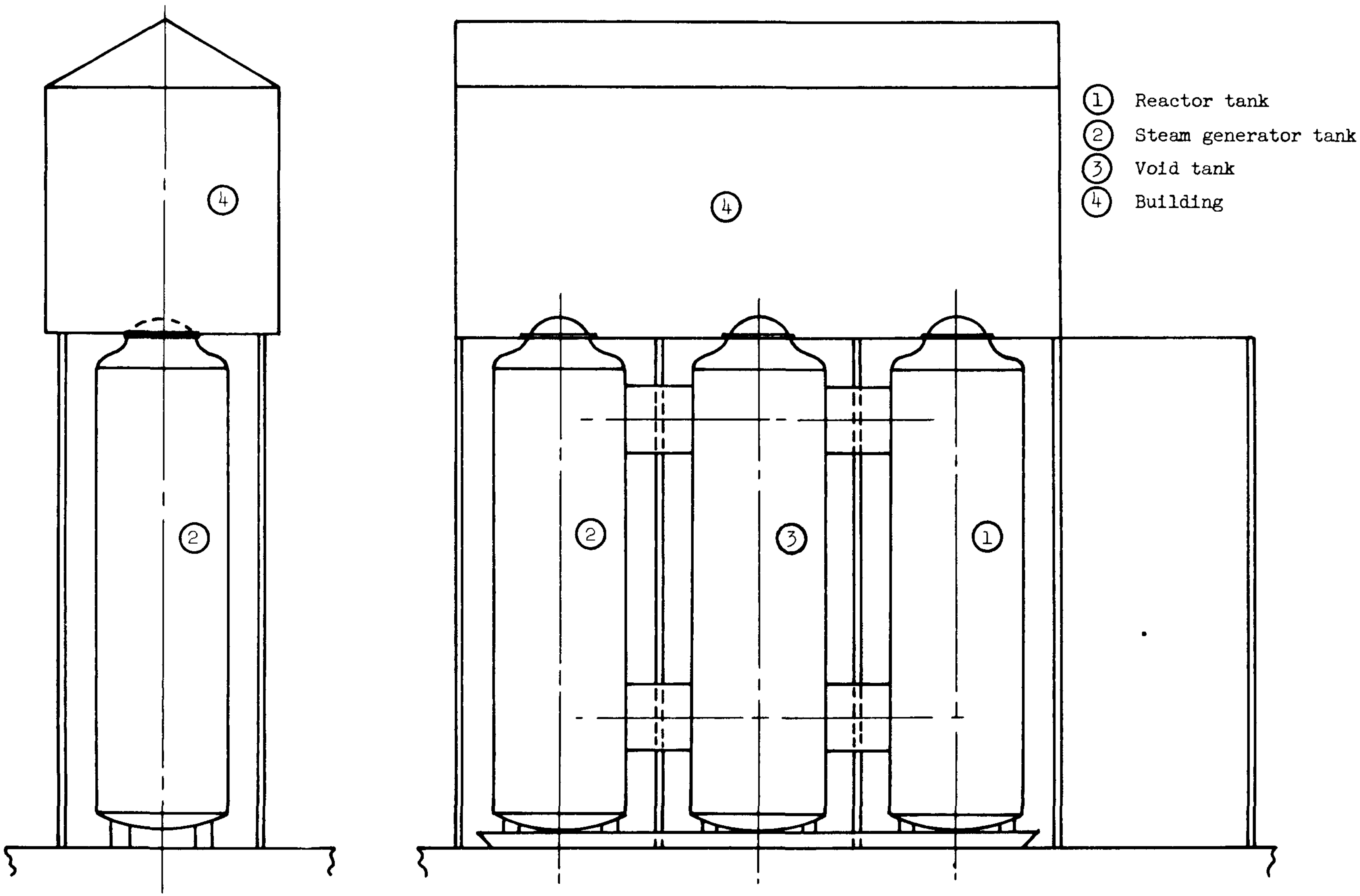


Fig. III-41. Primary Loop Configuration (three vertical tanks)

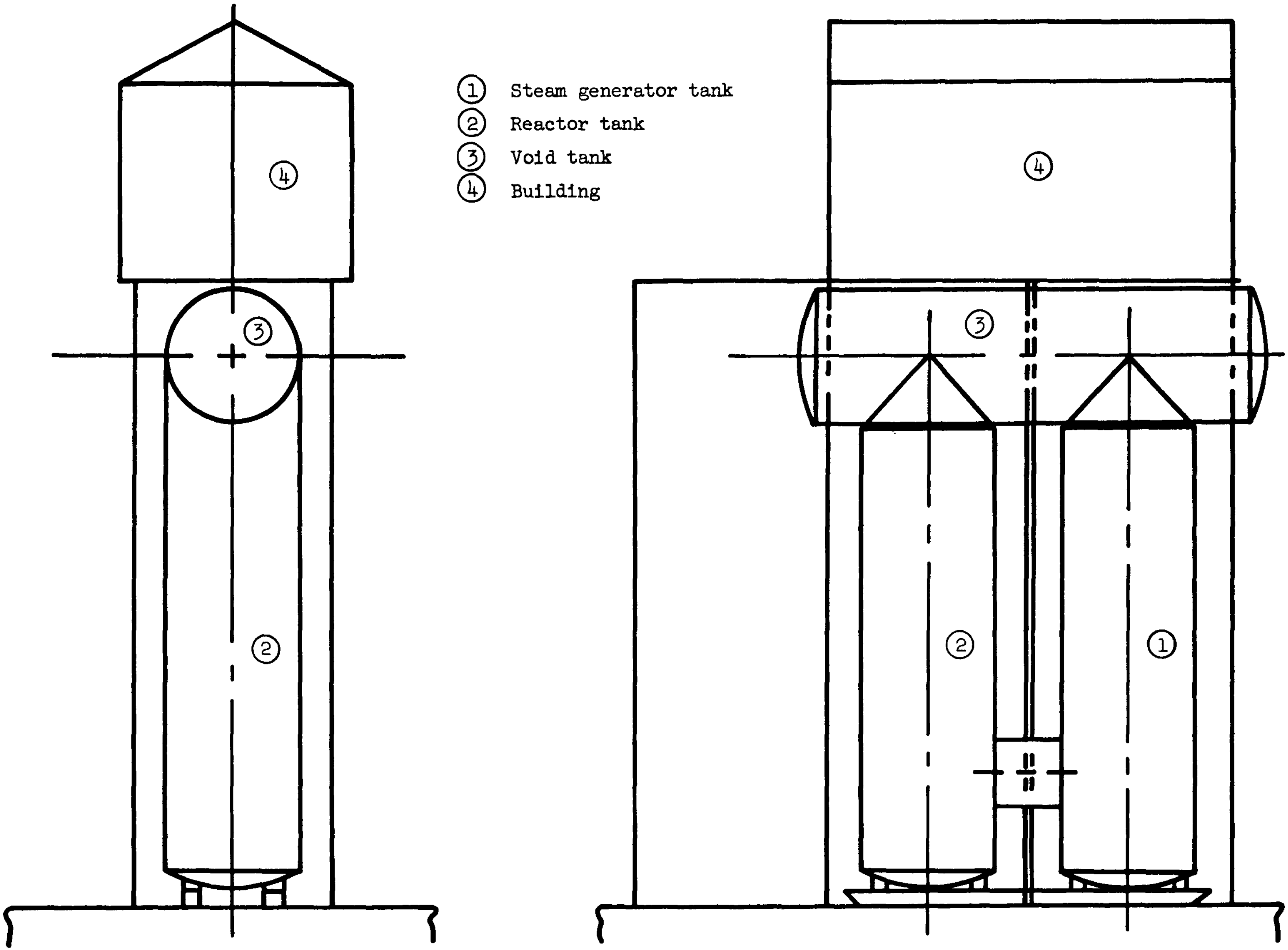


Fig. III-42. Primary Loop Configuration (two vertical tanks, one horizontal elevated tank)

The first version, 3 vertical tanks, has the following advantages:

(1) No deviations from the ASME Code are required.

(2) Primary pipe expansion loops are contained in a centrally located containment tank where they are readily accessible.

(3) Relocation procedures are a reversal of the erection procedures--a minimum of replacement equipment is required.

Disadvantages of this configuration include:

(1) The necessity for shipping the reactor pressure vessel separately in order to meet the 30,000-pound limitation for air transportation.

(2) The building size and structure support required are not minimized.

(3) Use of expansion loops is undesirable because

  (1) The length of pipe and elbows required increases the total head losses in the main coolant plant.

  (2) The complex arrangement of the piping increases the difficulties in assembly and installation of the main loop.

  (3) Usable space inside the tanks is reduced considerably.

The second configuration, two vertical tanks with one horizontal tank atop them, has one basic advantage: The full 8-foot, 8-inch dia. is used to connect the vertical tanks, allowing its 59.4-ft^2 connecting area to be utilized during an excursion instead of a restrictive connection of 12.5 ft^2.

Disadvantages of this version are:

(1) Special flange tolerances are required to assure alignment within 1/4 inch between the vertical tanks.

(2) The vertical tanks are fixed at the upper and lower ends, creating stresses due to thermal expansion.

(3) More welding work at the site is necessary. Relocation would involve more difficulties than in the first version because of added weldments and because the horizontal tank would interfere with equipment removal.

(4) The ASME Code does not provide for intersection of two tanks of the same diameter, so that additional development and test work would be necessary.

(5) Expansion loops are housed in the interconnect between the tanks where they are covered by backfill and, therefore, inaccessible after plant startup.

(6) As in the first version, building size and weight are not minimized.

The two versions are alike in most particulars of erection procedure and equipment. The shipping packages and equipment arrangements are the same except for the primary piping expansion loops. Earth backfill is used to the same level (15 feet above the center of the reactor core).

Comparison of the two versions indicated areas of possible improvement in each and led to a third version (Fig. III-43). This configuration utilizes 2 vertical tanks with the third tank horizontal at the ground level. The advantages obtained by this arrangement are:

(1) Reduction in required size of building and support structure.

(2) Reduction in the amount of field welding work required (by using one bolted flange joint at the interconnect between the horizontal void tank and the steam generator tank).

(3) Reduction in erection time since the void volume tank can be pulled into place using roller conveyors and lowered into position without special rigging equipment. Also, since there is only one joint between the void tank and the steam generator tank, alignment is less critical.

(4) The expansion loop is eliminated. Instead, the reactor pressure vessel is mounted on a trunnion. This reduces the number of bends required and shortens the overall piping run.

(5) Full utilization of the void tank as a shipping container reduces the number of aircraft loads required for shipment.

This version is believed to be distinctly superior to either of the other contained versions.

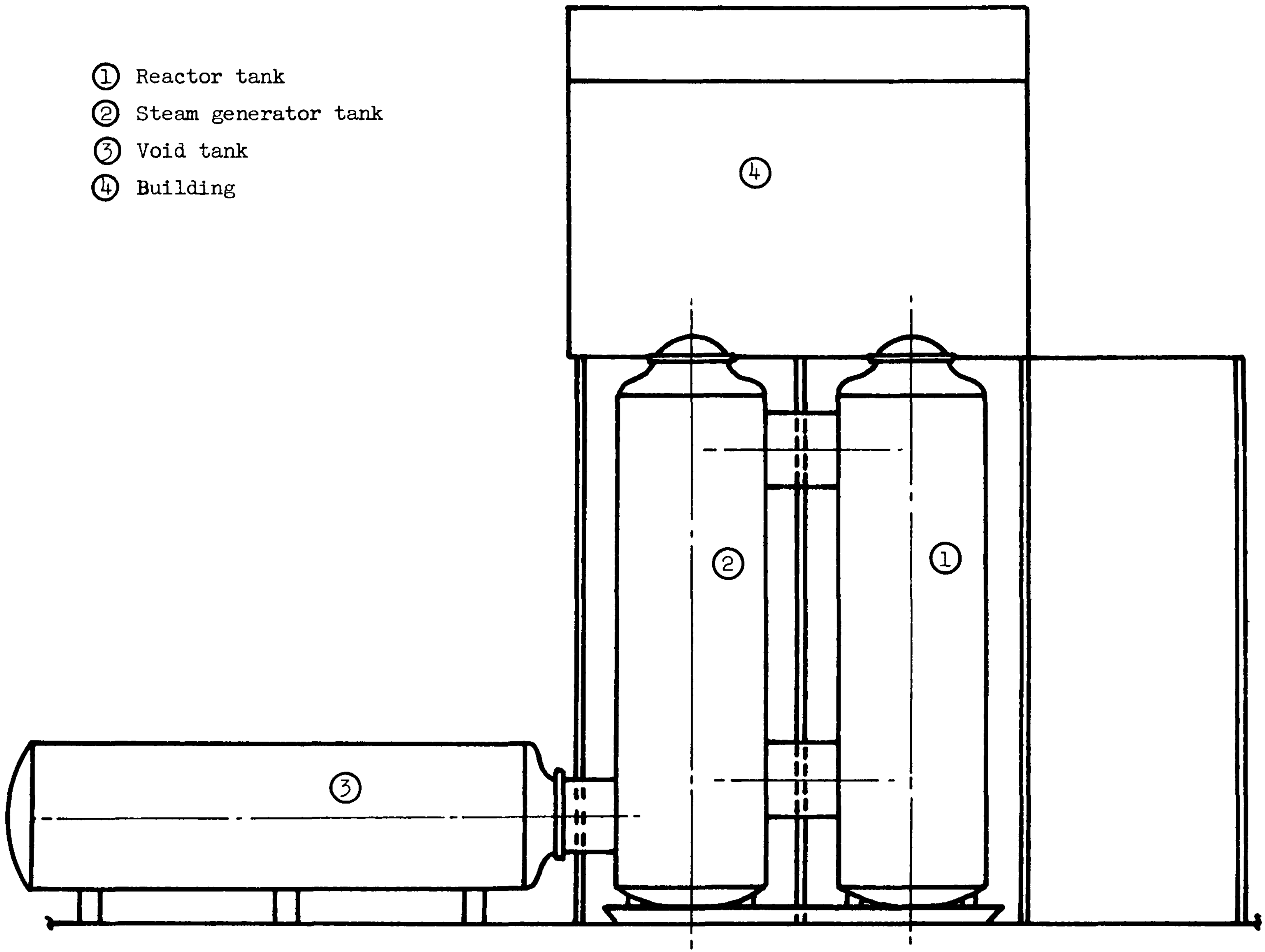


Fig. III-43. Primary Loop Configuration (two vertical tanks, one horizontal tank at ground level)

If this third version is compared to a noncontained configuration, only a few differences appear. The lower connection between the two equipment packages is one large expansion joint for the contained version but uses two joints, one for each pipe-run, in the noncontained version. Although the single large interconnect is closer to ASME practice than using two vessel penetrations in close proximity to each other, the double interconnect is desirable in the noncontained case because it is lighter and more easily assembled in the field. Since only shield water and earth backfill static pressures are encountered, this is acceptable. The steam generator package is in the form of a separate unit which is slid into the second containment vessel; in the noncontained version,this equipment can be integrally mounted in a thinner vessel without exceeding the 30,000-lb weight limit. The non-contained version has no third unit to correspond to the void containment vessel. In the contained version, the reactor pressure vessel is shipped in a separate package minus the upper head but including all interior components. The containment unit for the steam generator, minus its upper head, is used for shipping miscellaneous components. The void tank is also used for miscellaneous items but travels with its heads in place.

The general characteristics of the noncontained version are two vertical tanks, 8 feet, 8 inches in dia. x 27 feet, 3 inches long, to provide space for primary plant equipment and shield water. This design allows access during operation and makes it possible to accommodate a vertical steam generator and adequate shield water. Since the noncontained plant does not have to withstand excursion pressures, flat-bottomed tanks with less severe structural requirements can be used and the shipping weight is reduced enough to permit integral mounting of the reactor pressure vessel. The steam generator package was originally designed as the inner package of the contained version with an extra corrugated shield to strengthen the skin of the package to withstand loads imposed by the earth backfill. Since the corrugated shield imposes disadvantages by requiring extra shipping space and complicates the handling of the packages, a new steam generator package has been designed. This provides integral mounting of the equipment and a skin strong enough to bear the earth backfill loads; the full 8-foot, 8-inch dia is utilized to gain maximum space within the package. However, since the shipping weight is marginal, the pressurizer must be shipped separately. It is included in a sled package containing other components.

Details of the noncontained primary plant are shown in Fig. III-44, the method of erection in Fig. III-45 and the primary system packaging breakdown in Fig. III-46.

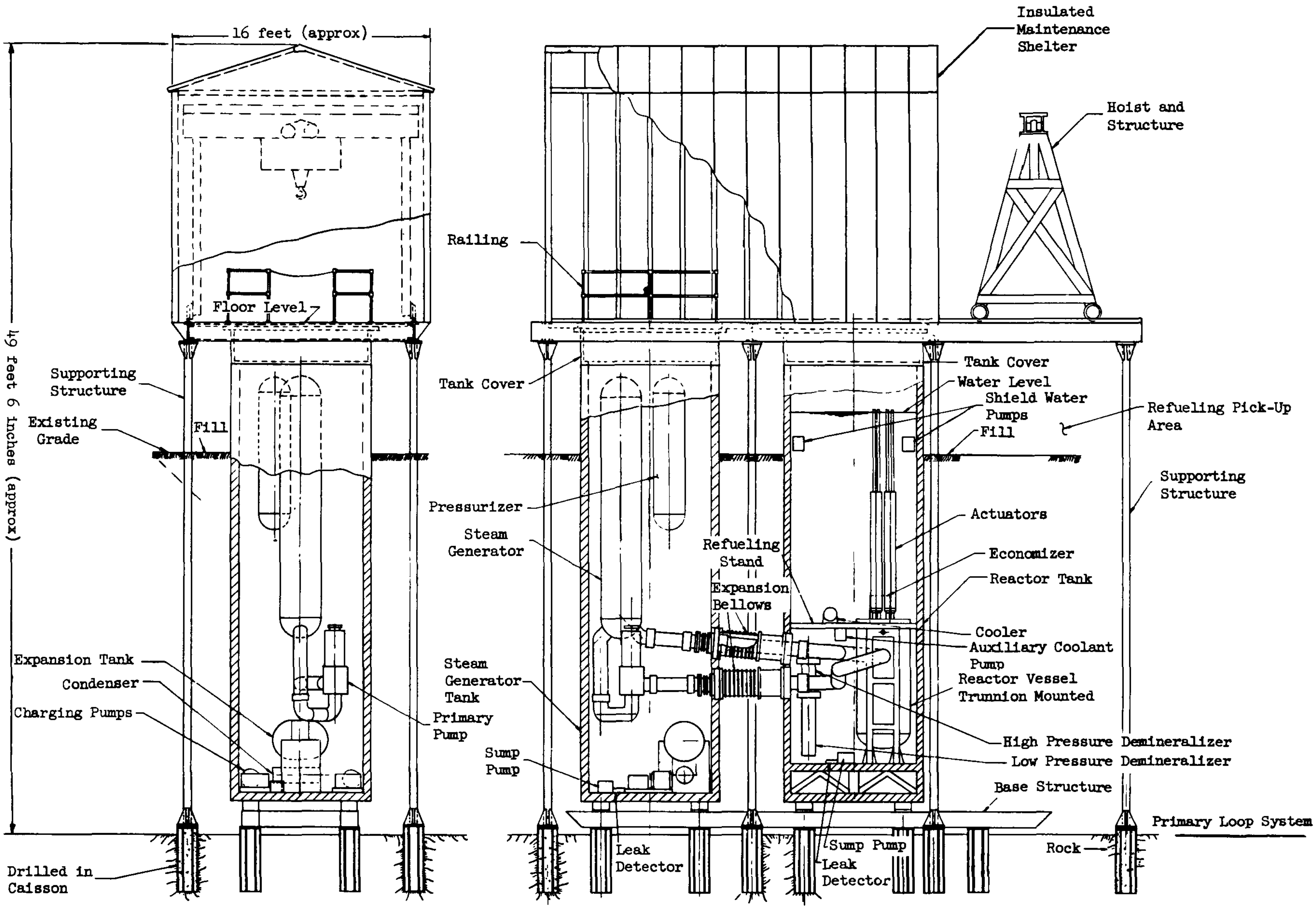


Fig. III-44. PM-1 Nuclear Power Plant Primary System Layout

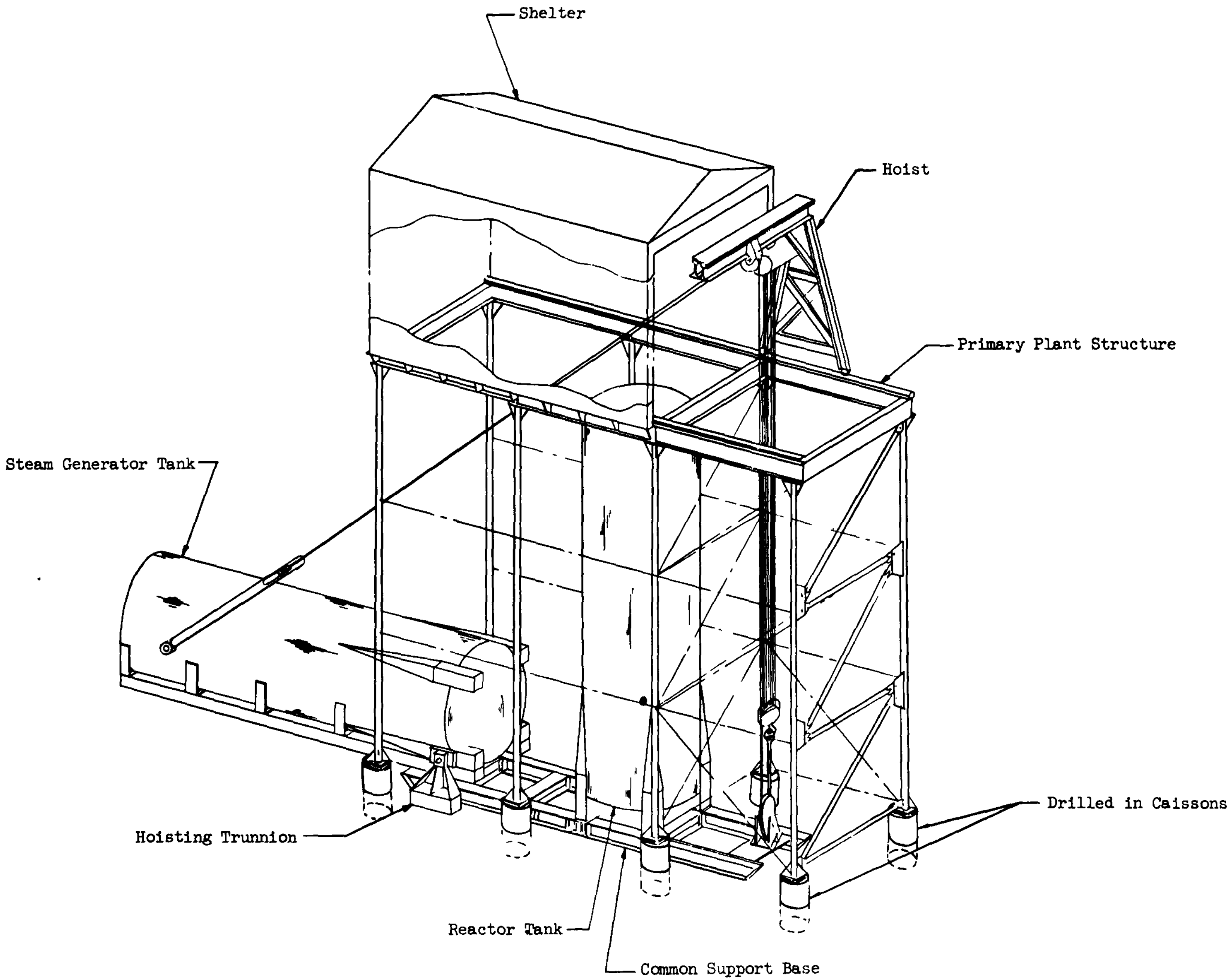
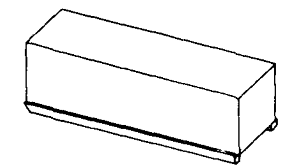
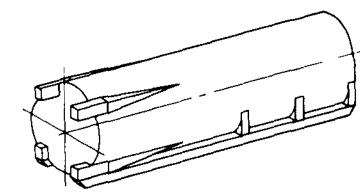


Fig. III-45. Primary System Erection--PM-1 Power Plant (uncontained)

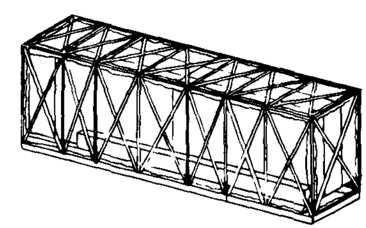
Package No.	Description	Contents	Gross Weight (estimated pounds)
A-1	Reactor tank: Non-contained, vertically mounted right circular cylinder 8 ft, 8 in. diameter x 27 ft, 3 in. double wall and floor construction consisting of ring frames, longerons, skin. Material:stainless steel Supplied with skids	Tank structure and secondary structure Reactor vessel (less head) Low pressure demineralizer High pressure demineralizer Cooler Economizer 2 Shield water pumps Sump pump Auxiliary coolant pump Piping	Shipping weight 26,000 Operational weight 90,000
A-2	Steam generator tank: Non-contained, vertically mounted right circular cylinder 8 ft, 8 in. diameter x 27 ft, 3 in. construction consisting of ring frames, longerons, skin. Material:stainless steel Supplied with skids	Tank structure and secondary structure Steam generator Primary coolant pump volute Condenser Expansion tank 2 Charging pumps Piping	Shipping weight 25,000 Operational weight 28,300
A-3	Equipment skid: Non-contained, flat bottom. Skid base with side and end trusses covered with a tarpaulin or plywood. Material:structural steel and plywood Skid used for shipping only	Air blast cooler Actuator package Pressurizer Tank covers and piping Reactor head Hoist package Primary pump package Support structure Skid	Shipping weight 27,000
A-4	Common base skid: Non-contained, flat bottom. Skid base with side and end trusses covered with a tarpaulin or plywood. Material: structural steel and plywood Skid used for shipping only	Support structure Hoist structure Miscellaneous equipment (tools, railing, hardware) Piping Skid Common support frame Maintenance shelter packages	Shipping weight 23,000



Package A-3



Package A-1 and A-2



Package A-4

Fig. III-46. Package Data--Primary System--PM-1 Power Plant (uncontained)

Details of the contained primary plant are shown in Fig. III-47, the method of erection in Fig. III-48 and the primary system packaging breakdown in Fig. III-49.

7. The Secondary System

W. Koch, L. Hassel, R. Groscup

The parameters of the PM-1 secondary loop were fixed and preliminary design equipment requirements were established based on the data developed during the parametric studies.

The basic work described in this subsection was performed by the Westinghouse Electric Corporation under subcontract to The Martin Company.

Heat balance and flow diagram.- A heat balance and flow diagram of the secondary system was developed and is shown in Fig. III-50.

System equipment.- System equipment was selected and is described in the following paragraphs.

The secondary system performs a dual function in converting the thermal energy received from the steam generator into high-quality electrical power and low-pressure steam for space heating, the specific productions being 1000 net kwe and 7 x 10^6 Btu/hr of space heat. The prime equipment in the system includes a steam generator, a turbine-generator unit, an air-cooled steam condenser system, steam- and electrically-driven boiler feed pumps, a combination evaporator-space heat reboiler, a steam jet ejector with after-condenser, a deaerator, a closed feedwater heater, electrical switchgear, emergency power source, and diesel auxiliary power unit. The functioning of the loop may be visualized by reference to Fig. III-50.

In general, standard commercial equipment is used. The components which have been given special attention to ensure that the PM-1 plant requirements are met include the steam generator, turbine-generator unit and steam condenser system. A description of these components follows.

Steam generator description.- The steam generator is of the vertical U-tube, integral drum type with primary coolant in the tubes and secondary water and steam in the shell. The reference design is a cylindrical shell with hemispherical ends, 16 feet, 7 inches high and 30 inches in outside diameter. The heat transfer surface is composed of 420 U-shaped tubes which have a 1/2-inch OD, 18 BWG* wall; the tubes represent a heat transfer area of 1080 ft^2 based on the outside surface. Based upon primary side

*BWG refers to Birmingham Wire Gauge

LEGEND:

1. Reactor
2. Cask location
3. Steam generator
4. Pressurizer
5. Pump
6. Condenser tank
7. Charging pumps
8. Self-equalizing stainless steel expansion joint--single type
9. Erection and positioning base frame
10. Shield water pump
11. Cooler
12. Economizer
13. Low-pressure demineralizer
14. High-pressure demineralizer
15. Auxiliary coolant pump
16. Self-equalizing stainless steel expansion joint--double type
17. Sump pump
18. Shield water leak detector

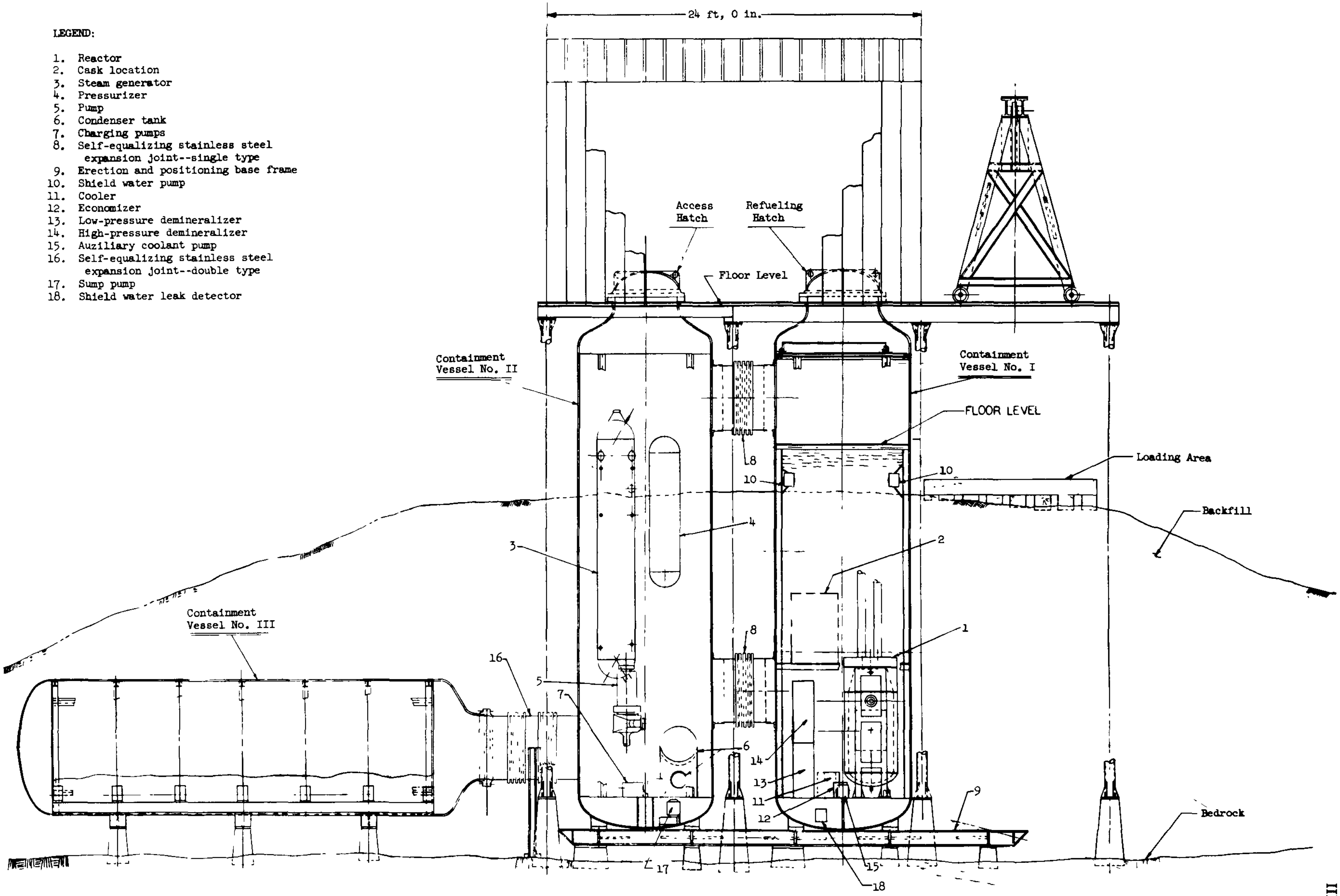


Fig. III-47. Primary System Layout--PM-1 Power Plant (contained)

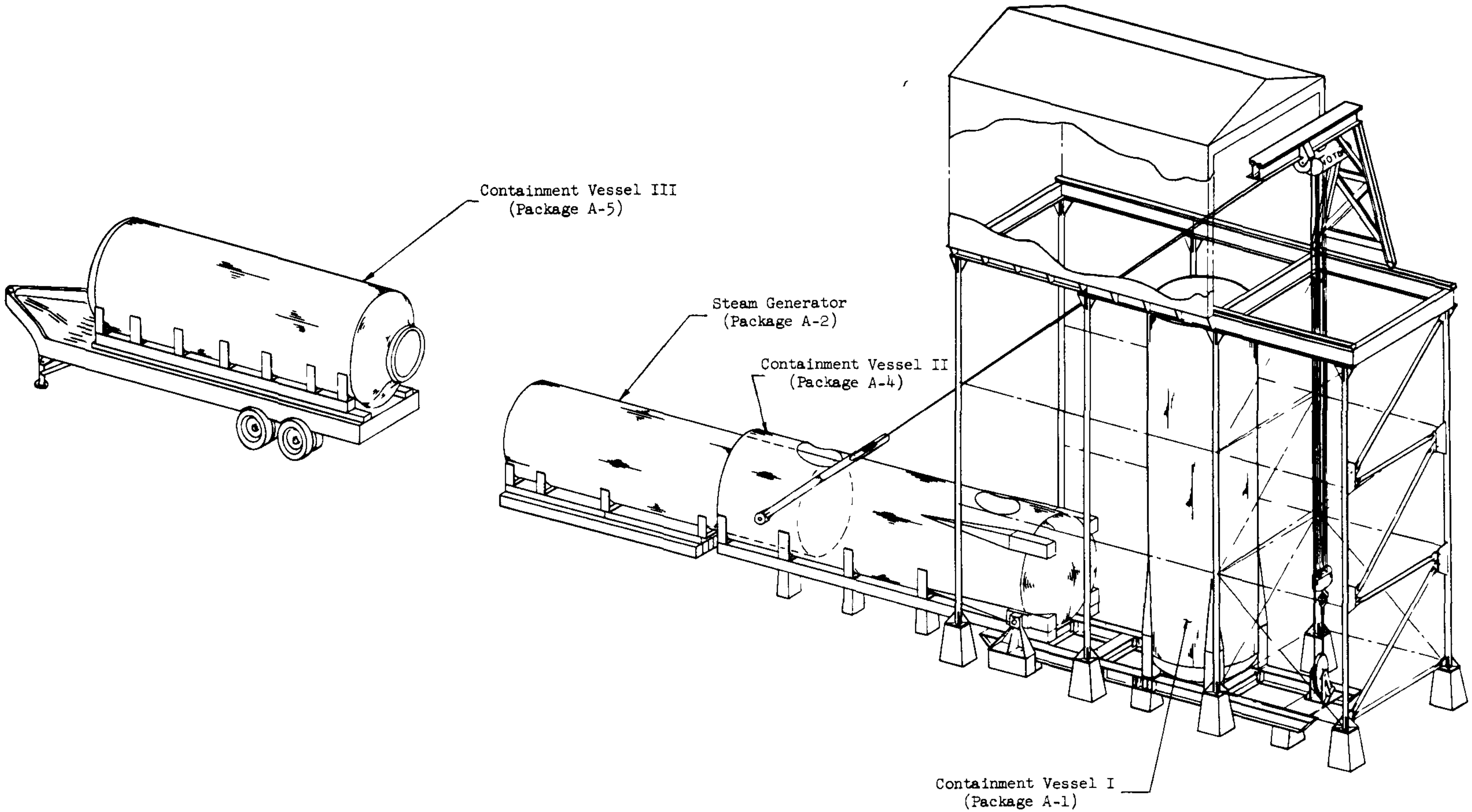
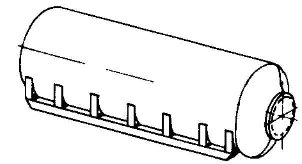
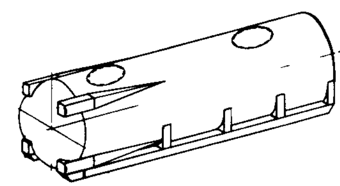


Fig. III-48. Primary System Erection--PM-1 Power Plant (contained)

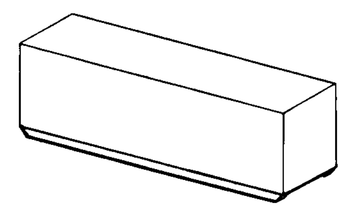
Package No.	Description	Type of Package Envelope Dimensions	Contents	Gross Weight (estimated pounds)
A-1	Reactor Containment Vessel I	Containment tank with skids (less head) 8 ft, 8 in. x 8 ft, 8 in. x 30 ft, 0 in.	Reactor mount and trunnion fittings Low power demineralizer plus valves and piping Primary loop pipe plus insulation and mounts Economizer plus piping Cooler plus piping Cask mount frame Auxiliary coolant pump plus piping 2 Shield water pumps plus piping Chocks and shoring for shipment High power demineralizer plus valves and piping	27,340
A-2	Steam Generator package	Cylindrical shipping container with temporary skids 7 ft, 6 in. x 7 ft, 9 in. x 27 ft, 3 in.	Steam generator and mounts Pressurizer and mounts Primary coolant pump volute Primary loop pipe, insulation and mounts Condenser tank and piping 2 Charing pumps and piping Sump pump Chocks and shoring for shipment Expansion tank	28,470
A-3	Reactor Package	Rectangular box frame with cover 8 ft, 6 in. x 8 ft, 8 in. x 30 ft, 0 in.	Reactor pressure vessel Reactor pressure vessel head Air blast cooler (crated) 6 Actuators (crated) Primary coolant pump--motor and impeller (canned) Ionization counters (crated) Chocks--cradles and shoring for shipment	29,960
A-4	Containment Vessel II	Containment tank with skids less head 8 ft, 8 in. x 8 ft, 8 in. x 30 ft, 0 in.	Aluminum support structure Bridge crane structure and sheave mount Ten ton electric hoist (crated) One hand geared trolley (crated)	28,790
A-5	Containment Vessel III	Containment tank with skids 8 ft, 8 in. x 8 ft, 8 in. x 30 ft, 0 in.	Armco S-2 building (or equivalent) (crated) Building floor 1 Lower sheave block Maintenance, erection and special tools (crated) Leveling jacks	28,800
A-6	Miscellaneous Skid Package	Rectangular box frame with cover 8 ft, 6 in. x 8 ft, 8 in. x 30 ft, 0 in.	2 Containment tank heads with access hatches 3 Containment vessel Expansion joint Interconnects Chocks and cradles for shipment 2 Hoisting trunnion Common support fram	25,160



Package A-5



Package A-4



Package A-3 and A-6

Fig. III-49 Package Data--Primary System--PM-1 Power Plant (contained)

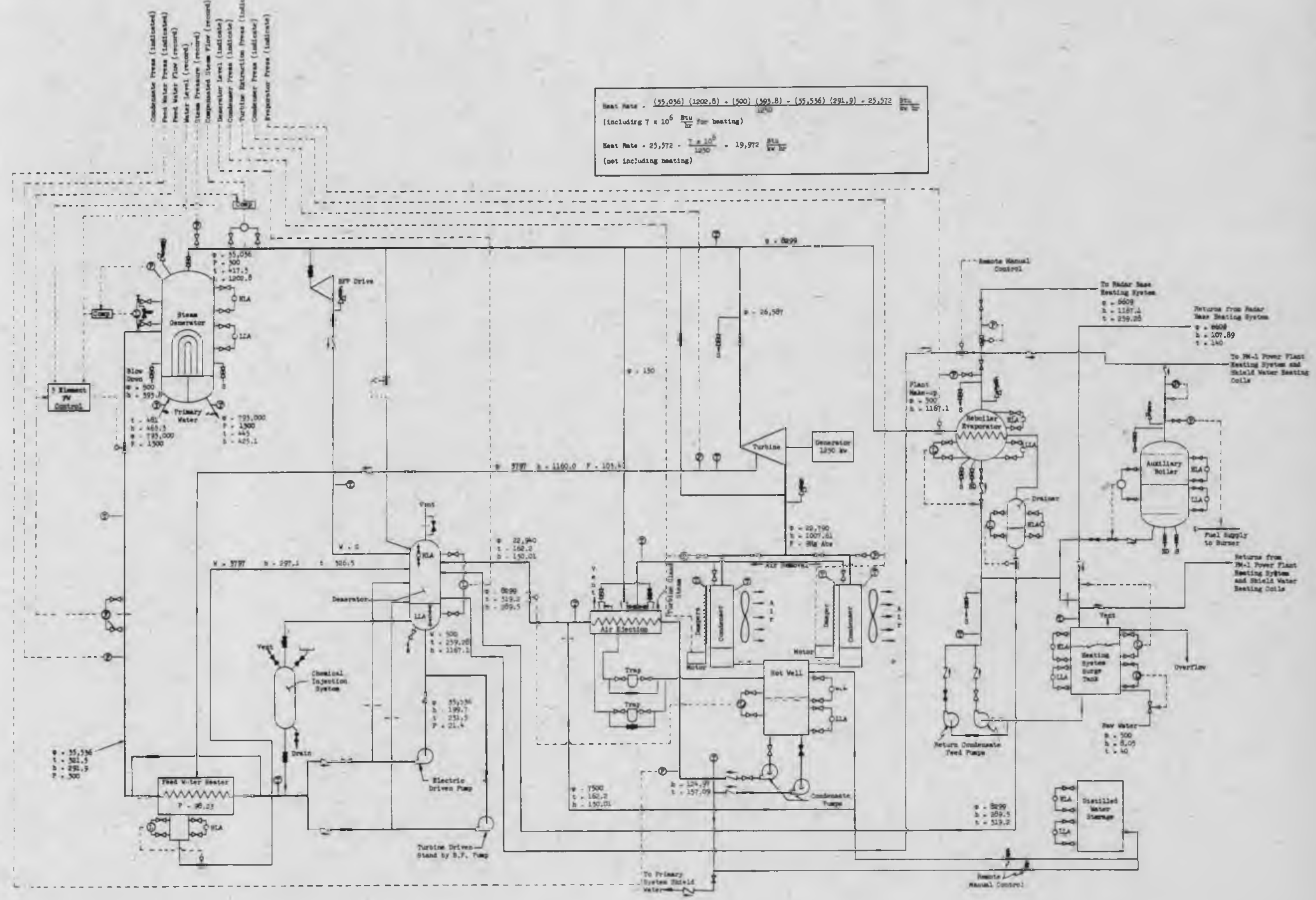


Fig. III-50. Heat Balance, Flow Diagram, Instrumentation Schematic--Secondary System--PM-1 Power Plant design conditions of 600° F and 1500 psi, and secondary side design conditions of 486° F and 600 psi, the estimated dry weight of the unit is 10,600 pounds.

When operating at capacity, the unit transfers 32 x 10^6 Btu/hr, producing 35,036 lb/hr of dry and saturated steam at 300 psia from feedwater at 321.5° F. Primary coolant conditions are: mean temperature of 463° F, flow of 1900 gpm, and operating pressure of 1300 psia.

During operation, primary coolant flows through a nozzle into the inlet side of a he hemispherical plenum at the bottom of the steam generator, up through the tubes, down into the discharge side of the plenum and out through the exit nozzle. Secondary feedwater is admitted to the shell through a feedwater nozzle located about 3/4 of the way up. The secondary flow is down through an annular section between the shell wall and the tube bundle, then up through the bundle where the secondary water is converted into steam. The steam flows upward through an impeller-shaped initial moisture separator, then upward and radially outward through a chevron-type second-stage separator, and finally through the third-stage centrifix separator and out through the steam nozzle.

The steam pressure in the generator will vary almost linearly with load, from 300 psia at full load to 481 psia at no load; temperatures will correspond to saturation at the given pressure.

For economic reasons indicated in the parametric study evaluation, the steam generator LMTD has been maintained at 43.5° F.

Turbine-generator.- The turbine-generator unit is a single-package module consisting of a turbine, single reduction gear, and generator, all integrally mounted on a single bedplate. The unit is rated at 1250 kwe continuous duty and has no overload capability in the turbine.

When operating at full capacity, the turbine receives 300-psia dry and saturated steam, extracts at 103.4 psia and exhausts at 9 inches Hg abs. The steam turbine is a high-speed (approximately 7500 rpm) impulse-type machine. When subjected to an instantaneous load change of 300 kwe at an 0.80 power factor, the turbine speed is controlled to within ±2% between 10 and 70% of rated net load with a load increase, and 40 and 100% with a net load decrease. Recovery time from initiation of the transient to steady-state conditions is approximately 1.5 seconds; the limitation on steady-state frequency fluctuations is ±0.25%. The speed controller is a fast-acting centrifugal governor which actuates a hydraulic servo system opening or closing a series of 5 steam admission valves.

When operating at capacity, the exhaust moisture content is 12.2%. Steps taken to minimize moisture erosion damage include using stellite-faced steam admission valves, high chrome content stainless steel material in the steam path and in portions of the casing subject to moisture attack, and limiting the maximum rim speed to approximately 800 ft/sec.

The reduction gear is a commercial design, producing a single speed reduction through a double-helical, side-offset type pinion and gear. Space and weight savings result from the combination of the turbine steam end bearing and support with the pinion bearing, and the combination of the generator forward bearing and support with the after gear bearing. The bearings of the entire turbine-generator unit are lubricated by oil under pressure, as are the gear teeth.

The generator is a salient pole machine designed to produce 1250 kwe net at 80% power factor and 1563 kwe net at 100% power factor. Output is 2400/4160 volts, 60-cycle, 3-phase, 4-wire; both ends of the windings are brought out for differential protection. Since the neutral will be solidly grounded, the windings will be braced to withstand a full line-to-ground fault. The generator is air cooled with an integral ventilation system. A static exciter is used with a coordinated static voltage regulating system. The exciter requires 33 kva of 480-volt, 3-phase, 60-cycle power which is drawn from the station power bus. The exciter and voltage regulator combination are designed to control the generator field so that the required power quality is achieved.

Condenser system.- The air-cooled condenser system consists of two identical finned-tube heat exchanger modules. Each unit is a box shaped, all welded, aluminum structure, 8 feet, 8 inches square and approximately 30 feet long. The sides are formed from finned tubes, several rows deep; the bottom of the structure is enclosed and the top serves as a mounting for 4 induced-draft fans. The total heat transfer surface of the condenser system is approximately 60,000 ft^2 in the form of finned aluminum tubes, the nominal dimensions of which are:

Root diameter	1.00 in.
Wall thickness	0.063 in.
Outside diameter	1.88 in.
Mean fin thickness	0.15 in.
Area ratio, $\frac{\text{outside}}{\text{inside}}$	16.26 in.

Operation of the condenser (see Fig. III-51) is as follows: Steam from the turbine exhaust enters a header located at condenser mid-length, then flows in either direction inside the finned tubes, where it is condensed and drains by gravity to the hotwells at the ends of the unit. Air flow is induced transversely across the tubes into the plenum space in the center of the structure and out through the exhaust fans. Air flow is controlled by automatically operated louvers which are positioned by a condenser pressure-sensing device. Fan power is monitored by manual control.

At design capacity, the condensers transfer 20.2×10^6 Btu/hr when supplied with exhaust steam at 9 inches Hg abs, ambient air temperature of 70° F and ambient measure that of a 6500-foot elevation. The electrical consumption of the fans under these conditions is 105 kwe.

Switch gear.- The switch gear is of commercial design and provides control and protection to the electric generator, the lines, and the station power transformer. The gear includes differential relays, overcurrent relays with voltage restraint and instantaneous attachments, and ground and anti-motoring relays for generator protection. Tie-line and transformer protection is by overcurrent and ground relays. Synchronizing equipment is provided for synchronization of the main generator with the auxiliary diesel generator or the tie line.

General heat transfer apparatus.- Included in this category are the deaerator, evaporator-reboiler, feedwater heater, and jet ejector-after condenser.

The deaerator is a commercial thermal atomizing-type unit of 40,000 lb/hr capacity, which also serves as an open heater. It receives make-up steam and drains, from the evaporator-reboiler, exhaust steam from the steam-driven boiler feed pump, drains from the feedwater heater, and condensate. The unit deaerates to an oxygen content of 0.005 cc/1, and has a feedwater storage capacity of 5 minutes at full load.

The evaporator-reboiler unit is a commercial, thermal descaling, tube-in-shell evaporator which produces 500 lb/hr of make-up steam and 7×10^6 Btu/hr of space heat in the form of 35 psia dry and saturated steam. Operation of the unit is by established power plant procedures.

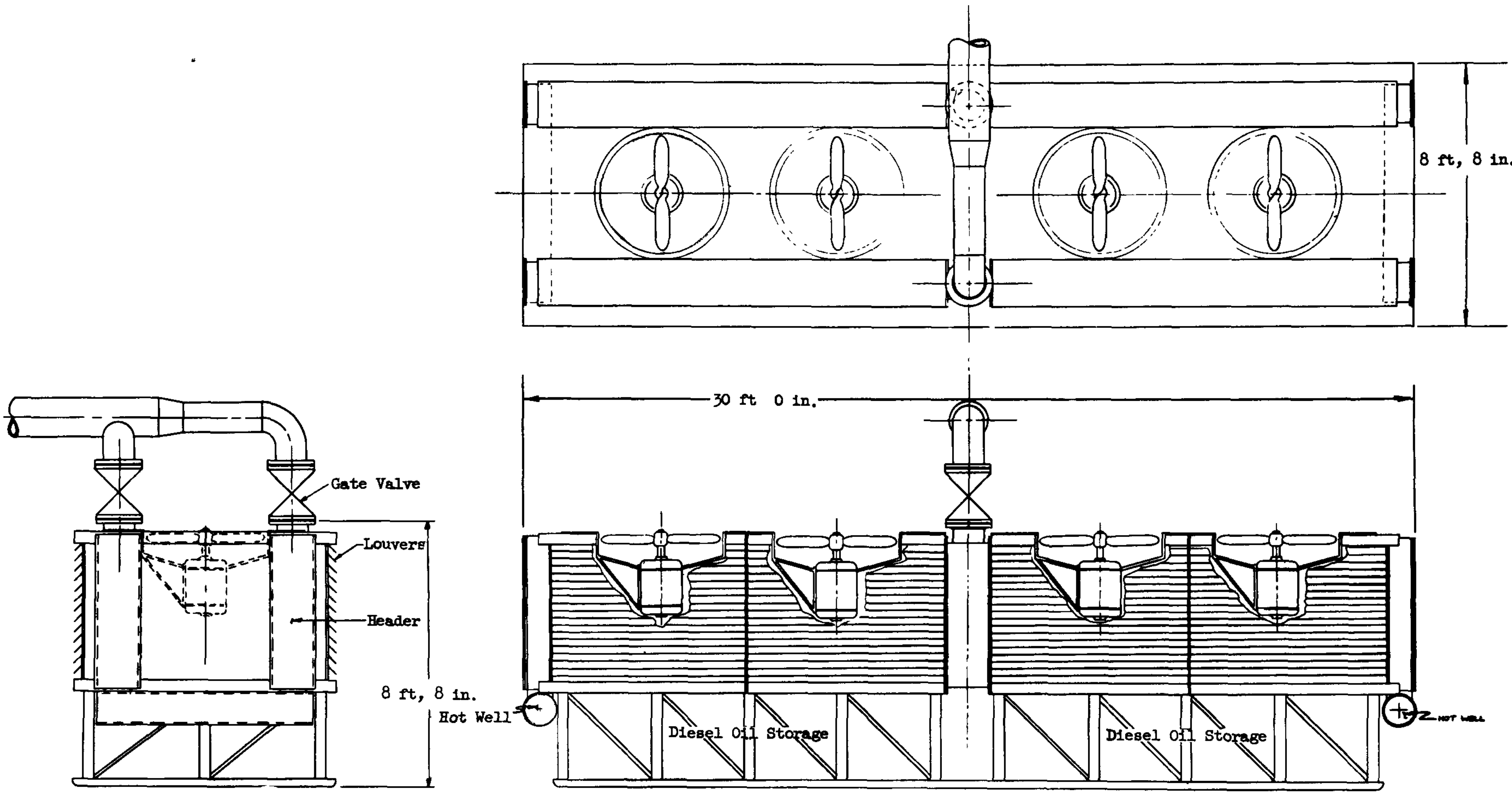


Fig. III-51. Condenser--PM-1 Power Plant

The feedwater heater is a commercial tube-in-shell heat exchanger which transfers heat from the turbine extraction steam to feedwater upon discharge from the boiler feed pump. Operation of the unit is automatic since control is exercised by level control valves. Design of the heater is according to the ASME Code, as are the designs of all pressure vessels in the secondary loop.

The steam jet ejector removes noncondensable gases from the condenser. The heat from the ejector is salvaged in the after condenser heat exchanger which also serves as the turbine gland seal steam condenser. The entire unit is of standard commercial manufacture and has a capacity of 6 cfm; two identical units are provided, one operating and one stand-by.

Auxiliary power supply.- Auxiliary power for starting and shutting down the plant is provided in the form of a 200-kw, high-speed, diesel-driven generator. The unit is capable of continuous duty and satisfies the need for emergency electrical power when the nuclear plant is down.

To ensure a source of uninterrupted power for instrumentation and control, a series of batteries and a motor-generator unit is provided.

Miscellaneous system equipment.- Two centrifugal vertical shaft boiler feed pumps are included in the loop; one electrically driven as the main unit and a full-capacity steam turbine-driven unit as a standby. Heat rejected from the standby unit is salvaged in the deaerator.

To prevent freezing of water within the system in the event of plant shutdown during the cold season, an auxiliary oil-fired,low-pressure steam boiler is supplied. The unit will operate on the same fuel provided for the auxiliary diesel generator and will be sized for an ambient temperature of -55° F.

3. Controls and Instrumentation

J. Henry, R. Caw, R. Wilder

The controls and instrumentation effort in support of preliminary design involved establishing a nuclear instrumentation layout, a reactor control scheme, the operating conditions which would result in scram or alarms, and primary and secondary system process controls and instrumentation.

All areas will be reported in detail in the preliminary design report. Inasmuch as the process controls and instrumentation are essentially those of a conventional small power plant, their description will not be repeated here. Discussions of the other areas are given below.

Nuclear instrumentation.- The instrumentation is completely transistorized and is housed in miniature plug-in assemblies. Reactor power levels will be measured from source, through intermediate, to power range using seven channels. Figure III-52 presents a simplified block diagram of the individual nuclear instrumentation channels. Their operation is generally as follows.

Channel I and II--source range instrumentation.- BF_3 proportional counters are used in each of the 2 source range channels. They furnish a signal in the form of a series of pulses to a linear amplifier. The amplified pulses pass into a discriminator which eliminates those pulses stemming from either noise or gamma radiation. The discriminator output passes to a count rate amplifier whose output is proportional to the logarithm of the count rate. Counts per second over a range of 1 cps to 1×10^5 cps are indicated on a logarithmic scale corresponding to neutron fluxes of from 0.25 nv to approximately 2.5×10^4 nv. In addition, the count rate amplifier output passes through a differentiating circuit to a period amplifier. Period is indicated over a range of -30 to to +3 seconds.

The period signal is used for high startup rate protection in the source range channel. If either instrumentation channel indicates the presence of a positive period of less than 15 seconds, further control rod extraction is prevented. If both channels indicate a period of less than 10 seconds, the reactor is scrammed.

Channels III and IV--intermediate range instrumentation.- Gamma compensated ion chambers are used in each of the intermediate channels. These chambers furnish a current signal to a log N amplifier to indicate the percentages of full reactor power attained over a range of from 2.5×10^{-5}% to 10%. This corresponds to a neutron flux of 2.5×10^3 nv to 1×10^9 nv. Period indications are similar to those associated with channels I and II, including the 15-second hold and 10-second scram functions.

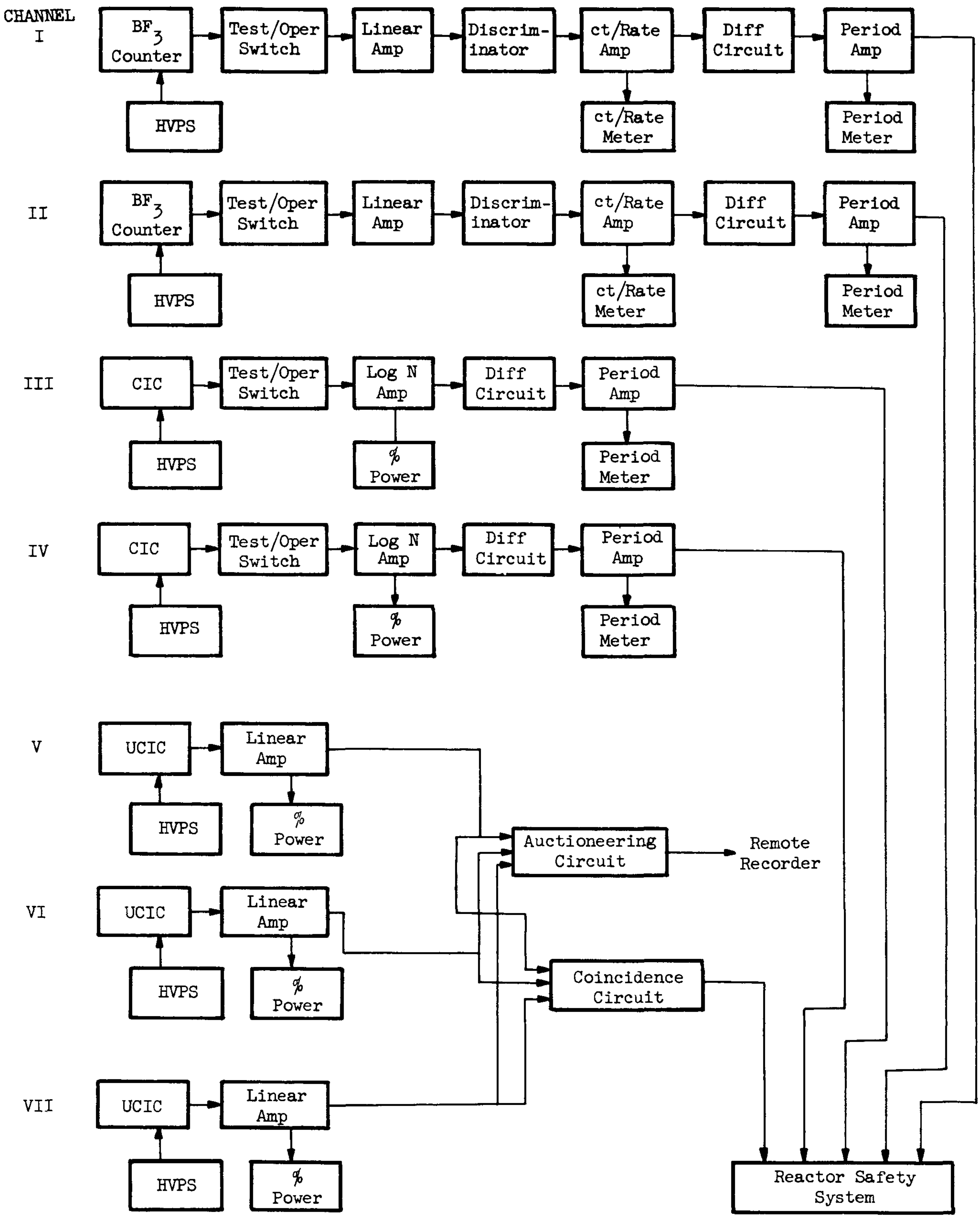


Fig. III-52. Nuclear Instrumentation Block Diagram

Channels V, VI and VII--power range instrumentation.- Uncompensated ion chambers are used in each of these power range channels. These chambers furnish a current signal to a linear amplifier to indicate reactor power over a range of from 1 to 150% of full power, which corresponds to a neutron flux of from 1×10^8 nv to 1.5×10^{10} nv. The three linear power signals are fed to individual trip circuits and to a coincidence circuit. The individual trip circuits operate annunciators when the indicated flux exceeds 120 to 130% of that associated with full power. When two of the three inputs to the coincidence circuit indicate flux levels in excess of 120 to 130% of full power, a scram is initiated.

Reactor control scheme.- Two methods of reactor control have been considered: automatic and manual. The strong negative temperature coefficient makes T_{av} control possible with either method of operation.

Automatic control.- The automatic control system allows the operator to establish a period to which the reactor will be held during power changes. The system will also hold a constant heating rate during the startup operation. During normal operation, the system will correct for fuel burnup and for poison generation.

It should be noted that the use of automatic control does not extend to the time during which the turbine generator is being placed on the line, nor does automatic control make it possible to reduce the size of the operating crew.

Manual control.- If manual control (operator control of the actuators) is utilized, additional operator functions are required during startups and for occasional adjustment of the control rods to accommodate burnup and poison buildup.

Because of the inherent stability of the PM-1 reactor, including the expectation that most load-changing maneuvers will be accommodated by the negative temperature coefficient (even under automatic control the negative temperature coefficient was expected to take over prior to exceeding the automatic control ΔT deadband), very little additional operator effort will be required under full manual control.

Although the automatic control method was retained through preliminary design, it will be closely scrutinized and possibly eliminated prior to the termination of final design.

<u>Operating conditions resulting in scrams or alarms.</u>- Table III-8 lists the abnormal system conditions which result in scrams or alarms. When both an alarm and a scram are indicated, the alarm set-point precedes the scram set-point by a margin sufficient to enable the operator to take corrective measures before shutdown occurs. During startup, the low-pressure scram signal is bypassed.

The use of set-back (i.e., automatic insertion of rod rather than rod drop) for certain abnormal conditions will be evaluated during final design when the system behavior has been more thoroughly determined.

TABLE III-8

Operating Conditions Resulting in Reactor Scram or Alarm

Item	Operating Conditions	Alarm	Scram
1	Fast period--Channel I	x	
2	Fast period--Channel II	x	
3	Fast period--Channels I and II	x	x
4	Fast period--Channel III	x	
5	Fast period--Channel IV	x	
6	Fast period--Channels III and IV	x	x
7	High neutron flux--Channel V	x	
8	High neutron flux--Channel VI	x	
9	High neutron flux--Channel VII	x	
10	High neutron flux--any 2 out of 3 (7-8-9)	x	x
11	Activation of manual scram		x
12	Reactor outlet temperature (high)	x	x
13	Reactor coolant flow (low)	x	x
14	Pressurizer pressure (high)	x	
15	Pressurizer pressure (low)	x	x
16	Pressurizer level (high)	x	
17	Pressurizer level (low)	x	x
18	Primary coolant pump cooling temperature (high)	x	
19	Shield water tank level (low)	x	

4. Packaging and Housing

A. Layman J. Reilly

The preliminary design of shipping packages was accomplished based on the information obtained from subtask 1.1 (Package Development and Test). The discussion of the preliminary design is incorporated below.

Secondary loop layout.- The PM-1 secondary loop layout is shown in Fig. III-53. This arrangement allows access to the equipment and a maintenance area in the central section of the building. The main electrical equipment, including the switchgear, motor control center, batteries, and three-unit M-G set is grouped in the switchgear package. The heat transfer apparatus and pumps, including the deaerator, evaporator, feedwater heater, condensate pumps, and evaporator feed pumps are grouped in the heat transfer apparatus package. The turbine-generator set has been located in the central area between the switchgear and heat transfer apparatus. The maintenance equipment, including air compressor, drill press, grinder, small tools, welding equipment, crane (portable), shaper, lathe, men's room, auxiliary steam boiler, and diesel generator is grouped in the maintenance package.

Figure III-54 shows the shipping layout of the secondary loop equipment in the various packages. This arrangement accounts for overall weight considerations, center of gravity, and providing for the maximum number of pieces of equipment to be mounted in the same position for both shipment and operation at the site.

Figure III-55 tabulates equipment and shipping weights for the various secondary system packages.

Packaging and housing.- In the first quarterly report, MND-M-1812, a description was given of the basic plan for a combination package shelter and of the reasons for selecting this plan. This report defined the maximum size and weight of the package and its center of gravity limitations to allow shipment within a C-130 aircraft as well as the size, weight, and center of gravity limitations of the equipment to be carried within the packages. No change has occurred in these basic items.

During this quarter, technical consultations were held with personnel in the Air Force Office of Civil Engineering, with Stewart Air Force Base personnel, and with Wright-Patterson Air Force Base personnel. These consultations concerned the feasibility of the package-shelter concept and the loading characteristics of C-130 aircraft. The structural requirements of the packages were studied; it appears that the handling and transportation loadings will be the principal design factors with the insulating requirements of an arctic shelter a strong secondary consideration.

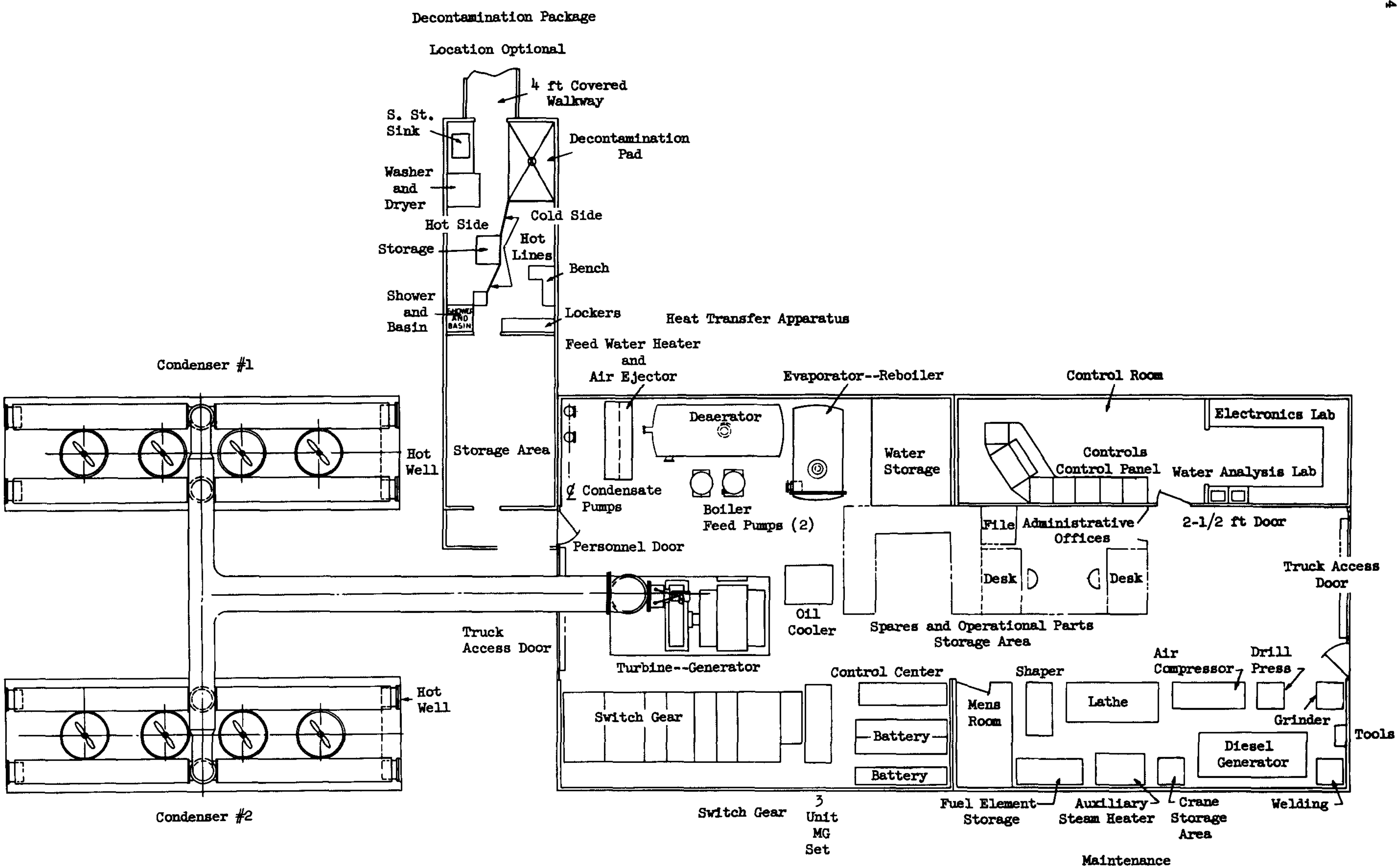
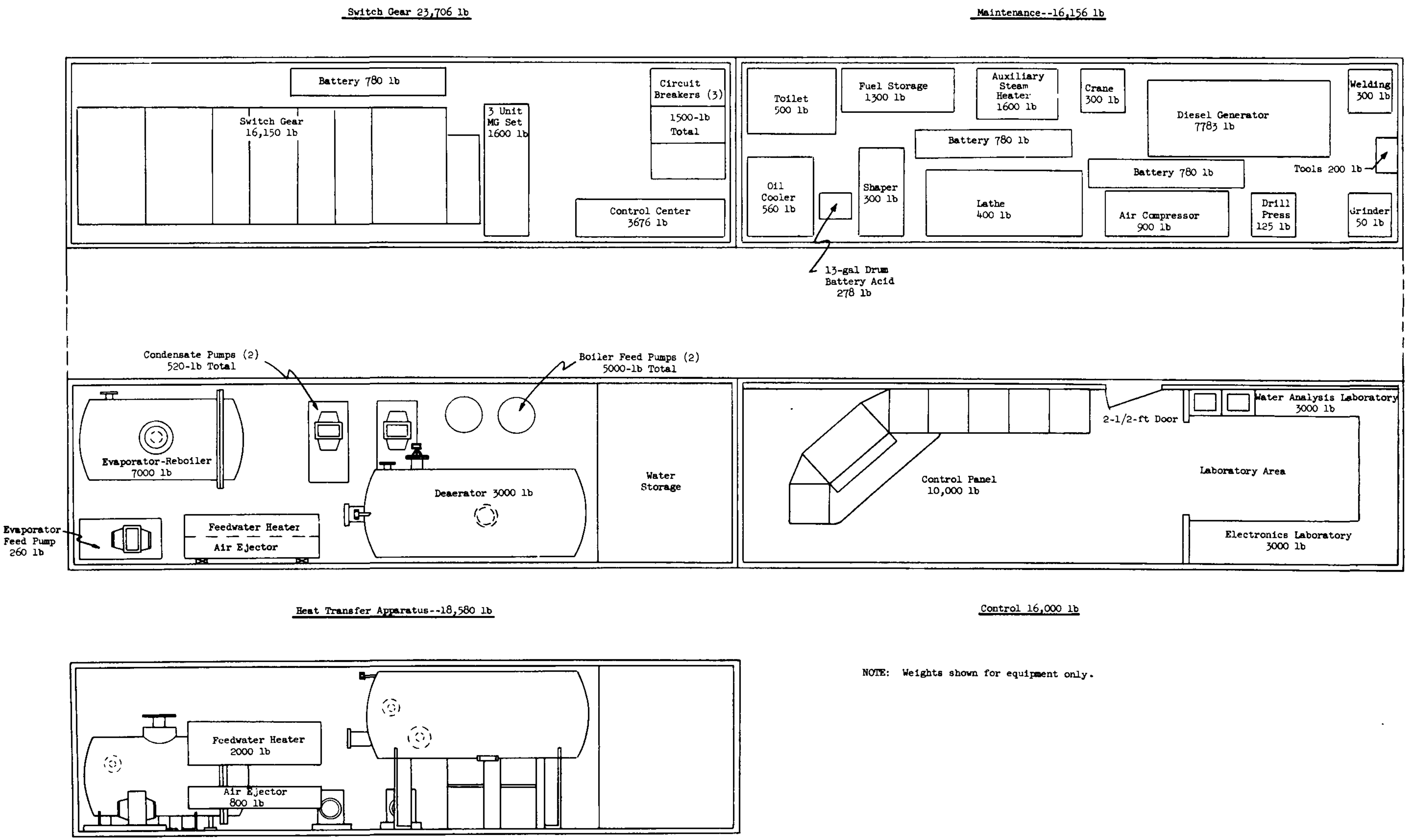


Fig. III-53. PM-1 Nuclear Power Plant Secondary System Layout



NOTE: Weights shown for equipment only.

Fig. III-54. Shipping Package Layout (secondary system)--PM-1 Power Plant

Package No.	Description	Type of Package Envelope Dimensions	Contents	Gross Weight (estimated pounds)
B-1	Arctic shelter Secondary building	Switch gear 8 ft, 6 in. x 8 ft, 8 in. x 30 ft	Switch gear Battery Motor generator set Control center Circuit breakers	29,366
B-2	Arctic shelter Secondary building	Heat transfer apparatus 8 ft, 6 in. x 8 ft, 8 in. x 30 ft	Evaporator-reboiler Evaporator feed pump Feedwater heater Air ejector Condensate pumps (2) Deaerator Boiler feed pumps (2) Water storage	24,240
B-3	Arctic shelter Secondary building	Controls 8 ft, 6 in. x 8 ft, 8 in. x 30 ft	Control panel Water analysis laboratory Electronics laboratory	26,000
B-4	Arctic shelter Secondary building	Maintenance and auxiliary power 8 ft, 6 in. x 8 ft, 8 in. x 30 ft	Tools Oil Cooler Fresh fuel storage Auxiliary system heater Shaper and lathe Diesel generator Air compressor Drill press Welding unit Grinder Men's room Batteries (2) Drum for battery acid	21,816
B-5	Arctic shelter	Decontamination building 8 ft, 6 in. x 8 ft, 8 in. x 30 ft	Washer and dryer Bench and lockers Basin and shower Hot drain Sink Storage area Miscellaneous equipment Shipped in storage area	Approximately 20,000
B-6	Turbine generator	Skid mounted only 5 ft, 8-1/2 in. x 5 ft, 6 in. x 13 ft, 3-1/2 in.	Turbine generator	29,700
B-7	Condenser No. 1	Self-contained 8 ft, 8 in. x 8 ft, 8 in. x 30 ft	Condenser No. 1	30,000
B-8	Condenser No. 2	Self-contained 8 ft, 8 in. x 8 ft, 8 in. x 30 ft	Condenser No. 2	30,000
B-9	Miscellaneous structure piping and flooring	Open truss 8 ft, 6 in. x 8 ft, 8 in. x 30 ft	Secondary building floor, end panels, piping, cables, etc.	Approximately 20,000

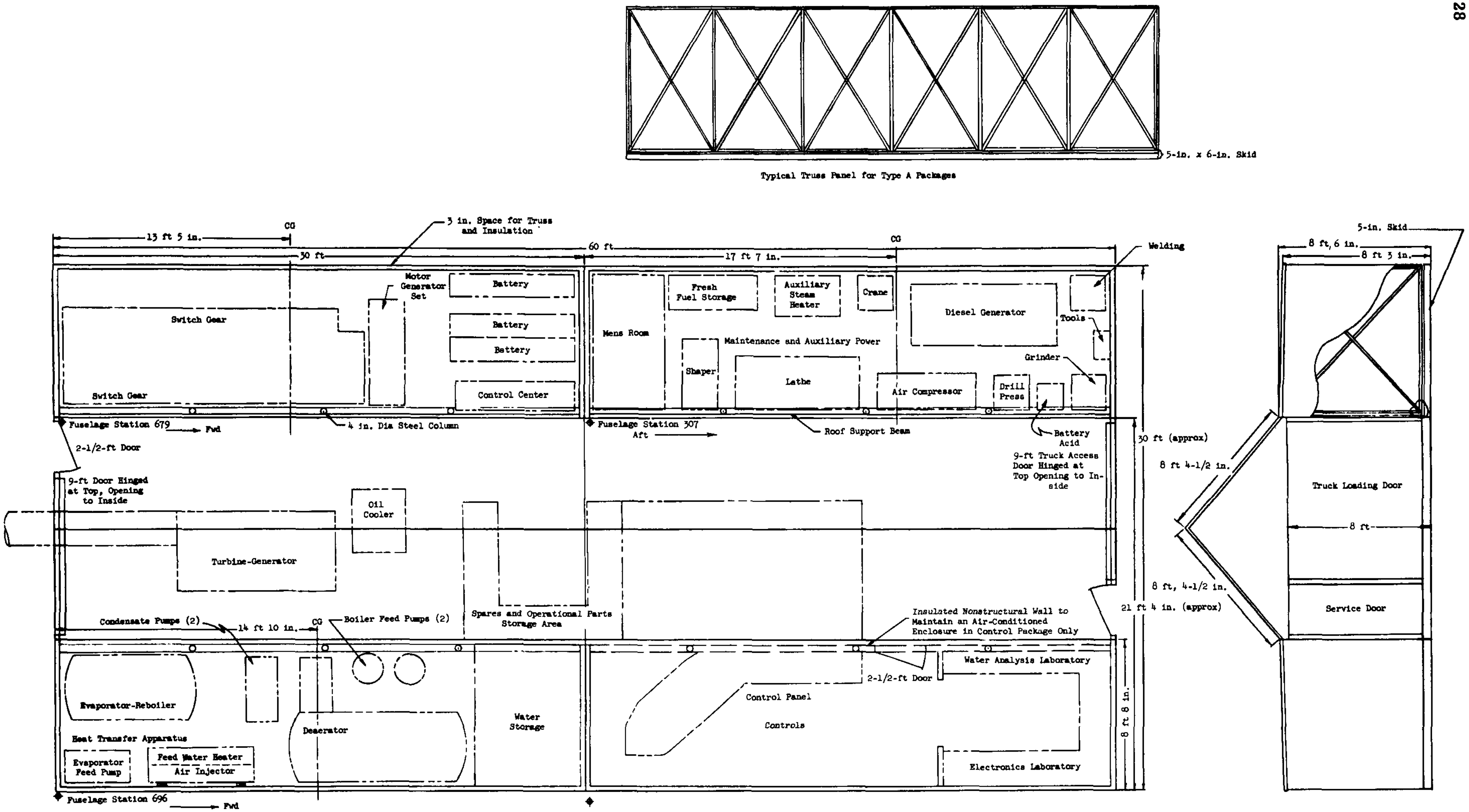
Fig. III-55. Package Data--Secondary System--PM-1 Power Plant

It was determined that the walls, roof, ends, and floor of each package should each be a single panel because single panels are more satisfactory as load-carrying members, will greatly reduce sealing problems, will be much lighter, and will be more economical. Each panel will be bonded sandwich structure utilizing 0.012-inch stainless steel sheathing on each face and a preformed polystyrene filler (density of 1.3 to 2.0 lb/ft^3). These panels will be edged with Douglas Fir frames (all wood used will be treated with a fire-retardant material). The side panels, end panels, and floor panel will be 3 inches thick. The roof will be 2 to 3 inches thick. Imbedded within the side and end panels will be a tubular steel truss, designed to provide building rigidity at the site as well as to support the handling loads encountered in transportation of the packages. The floor will be imbedded with Douglas Fir joists to which the equipment within the package will be bolted. The joists, in turn, will be bolted to steel structural members in the skid base. The side wall trusses will also bolt directly to the steel skid base. The skid base will consist of a welded steel channel grid supported by steel skids extending along the sides for the full length of the package. There will be no "thru metal" from the outer to the inner face of any of the insulated panels. This precaution will prevent condensation and/or the formation of ice balls inside of the package during periods of extremely low outside temperature. The stainless steel sheathing on the inner panel faces will provide a vapor barrier. The stainless steel sheathing on the exterior will provide a leakproof surface.

There will be 5 packages of the arctic shelter type, housing the heat transfer apparatus, the switchgear, the controls, the maintenance and auxiliary power area, and the decontamination area. The decontamination package will be a separate building at the site. The heat transfer, switchgear, controls, and maintenance packages will be combined at the site to form a single building approximately 30 feet x 60 feet, by placing two of the packages end-to-end on one side and the other two packages end-to-end on the other side; a space of approximately 13 feet will remain between the inner walls of each two-unit group (see Fig. III-56). These inner walls, which will be hinged along the top, will be unfolded-- swinging out and up approximately 130° so that their outer edges meet to form a peaked roof over the center portion of the building. Gable sections and doors will be added to each end of the middle portion, forming a complete building.

The buildings will house all of the secondary system except two air blast condensers which will not be housed--except for required plywood shipping protection. These condensers will be shipped without any additional external packaging and will be entirely self-contained.

The turbine-generator, which will be housed under the peak-roof portion of the larger secondary building, will also be shipped separately with adequate protective packaging.



NOTES:

1. All floor space not covered by equipment will be covered by a 1/2-in. plywood and 1/8-in. masonite walkway
2. All metallic components of the control package panels to be electrically connected and grounded to assure a shielded compartment
3. See Fig. III-54 for equipment weight and arrangement within packages as shipped
4. Fuselage station and direction arrows $\xrightarrow{\text{fwd}}$ indicate tie down positions in C-130 aircraft

Fig. III-56. Integral Housing--PM-1 Power Plant

The primary system will be covered with a portable panel building such as an Armco S-2 or equivalent. This will be shipped knocked down, within a trussed pallet package similar to the secondary packages, except that it will be uninsulated and, since it will be used at the site only for unheated storage, the walls and roof will be plywood.

Preliminary drawings for all of these packages have been completed. Currently, final design of the packages is in progress and will be completed in March of 1960.

5. Reliability

J. Stegemerten

A preliminary reliability analysis was performed to determine:

(1) The degree to which present state-of-the-art design concepts would fulfill the overall specified requirements.

(2) The degree of realism in our arbitrarily assigned subsystem down-times.

Valid data, dealing with failure rates of individual items of equipment, is rather sparse; hence, the preliminary analysis should be considered only as a qualitative indication of the degree of overall reliability. It serves to identify those equipment items and maintenance techniques warranting special attention during final design.

The following summarizes the conclusions reached, based on very limited practical experience data. The analysis indicates that:

(1) The contract specified in-commission rate (i.e., "up-time"), of 94.25% can be met by utilizing a large proportion of existing, available, and generally proven parts and techniques.

(2) Scheduled down-time will be examined critically during final design since a reduction would increase estimated plant reliability to a significant degree.

(3) Control rod actuators require special attention.

(4) An adequate preventive maintenance program is extremely important in meeting the plant reliability requirements.

Careful analysis of SM-1 maintenance records and available Navy data will be necessary during final design to ensure a realistic program.

The methods and techniques applied to the PM-1 reliability analysis were developed and reported in a reliability handbook to be used for reference thoughout the project. The basic concept is that the design engineer has primary responsibility for reliability in his particular area. The handbook provides an introduction to the subject and develops the basic methodology and standard terms used to describe and evaluate reliability. As specific techniques are developed, forms devised, and reference data gathered, they will be issued as addenda to the handbook.

Specific reliability requirements have been included in the preliminary outline specifications issued to date wherever applicable. When specific requirements are not known, the numerical requirement is left blank so that the most current figure can be inserted when the documents are issued to vendors for bids. The statement is worded so that vendors will be required to investigate operating history and failure data for the same type of equipment under similar conditions. The results are to be reported and summarized in the form of a predicted annual down-time for each component. This should result in an accumulation of reliability data useful in making an overall analysis and a reasonable prediction of down-time.

Reliability work will be continued during the next quarter with emphasis being placed on extending the scope of the reliability handbook and further review of overall plant reliability, giving due consideration to any additional component data received.

C. SUBTASK 3.3--PREPARATION OF SPECIFICATIONS AND COMPONENT AND FACILITY TEST LISTS

R. C. Groscup, W. Koch, G. Zindler, S. Zelubowski

This subtask is concerned with the preparation of outline specifications and lists of component and facility tests. The work planned and completed during the second project quarter comprised preparation of the outline specifications listed below to provide preliminary engineering data and serve as a basis for final procurement specifications, and the preparation of a list of component and facility tests which will be developed in detail and performed later as Tasks 9 and 10. This subtask is complete; all further specification and test efforts will be made as part of other Tasks (4, 9, and 10).

The following outline specifications were prepared during this quarter:

<u>PM-1 Nuclear Power Plant</u>

MN-7000	General Outline Specification for the PM-1 Nuclear Power Plant

<u>Primary System</u>

MN-7200	General Outline Specification for the Primary System of the PM-1 Nuclear Power Plant
MN-7890	Outline Specification for PM-1 Reactor Core
MN-7211	Outline Specification for PM-1 Reactor Pressure Vessel
MN-7221	Specification for Control Rod Actuating Systems PM-1 Reactor
MN-7361	Outline Specification for the PM-1 Reactor Pressurizer
MN-7311	Outline Specification for PM-1 Primary Coolant Pump
MN-7321	Outline Specification for Steam Generator

<u>Secondary System</u>

MN-7300	General Outline Specification for the Secondary System of the PM-1 Nuclear Power Plant
MN-7900	Outline Specification for Secondary System, Turbine Generator Unit

MN-8100	Outline Specification for Air Cooled Condenser
MN-7322	Outline Specification for the Evaporator-Reboiler for the PM-1 Nuclear Power Plant
MN-8711	Outline Specification for Switchgear and Power Center Transformer
MN-8630	Outline Specification for Auxiliary Power Plant for the PM-1 Nuclear Power Plant
MN-8130	Outline Specification for Deaerating Heater PM-1 Nuclear Power Plant
MN-8700	Outline Specification for Motor Control Center for the PM-1 Nuclear Power Plant
MN-7740	Outline Specification for the Closed Feedwater Heater for the PM-1 Nuclear Power Plant
MN-7750	Outline Specification for Boiler Feed Pump and Drivers for the PM-1 Nuclear Power Plant

Controls and Instrumentation

MN-7601	Outline Specification for Controls and Instrumentation for PM-1 Nuclear Power Plant

Packaging

MN-7052	Outline Specification for PM-1 Packaging, Shipping and Shelter

Miscellaneous and General

MN-2000	Outline Specification for Fasteners for the PM-1 Nuclear Power Plant
MN-2003	Outline Specification for Piping Requirements for the PM-1 Nuclear Power Plant

MN-2004	Outline Specification for Electrical Requirements for the PM-1 Nuclear Power Plant
MN-2005	Outline Specification for Welding Requirements for the PM-1 Nuclear Power Plant
MN-2001	Specification for Drafting Room Procedures for the PM-1 Nuclear Power Plant

Containment

MN-7251	Outline Specification for PM-1 Reactor Containment

Final specifications will be prepared as part of Task 4.

The preliminary list of component and facility tests is incorporated into the PM-1 Preliminary Design Technical Report as Section VIII. Preshipment tests, both at vendor's shops and at The Martin Company plant, and testing at site are included. The preshipment testing will be pursued as Task 9 and the at-site testing as Task 10, when the appropriate stages of progress are attained.

IV. TASK 4--FINAL DESIGN

Project Engineers--Subtasks 4.1, 4.2: R. Akin, C. Fox, G. Zindler

Project Engineer--Subtask 4.3: C Fox

This task covers the preparation and accomplishment of final design and the design analysis.

During the second quarter, it was planned to initiate Task 4. This was accomplished.

Final design was initiated for the secondary system under Task 4.2 including work on the steam-electric system, final package design, interconnections, maintenance equipment, and decontamination equipment.

Final design and specification for the primary system controls and instrumentation were initiated under Task 4.2.

Reliability evaluation efforts were initiated under Task 4.1 in support of secondary system design; reliability requirements were written into the statement of work for Gibbs and Hill. A technical memorandum on integral package and housing also was initiated under this subtask.

These efforts will continue during the next quarter.

V. TASK 5--CORE FABRICATION

Project Engineer--Subtasks 5.1, 5.2, 5.3: J. O'Brien

The overall objectives of Task 5 are to develop and fabricate the fuel elements required for the PM-1 Flexible Zero-Power Test and the final PM-1 core.

A. SUBTASK 5.1--FABRICATION OF CORE

R. Sipe

It was anticipated that material purchased from Allvac Metals would be processed and pierced for drawing, and that cladding fabrication would be initiated during the second quarter. Unfortunately, the Allvac material had to be rejected. Alternative materials were selected, however, and fabrication was initiated.

During the next quarter, "A" core (Zero-Power Test) cladding material will be delivered and fabrication of the "A" core will commence.

During the past quarter, three separate melts of low-colbalt, low-tantalum fuel element cladding material were made by the Allvac Metals Company, Monroe, North Carolina. The compositions of the melts are summarized in Table V-1. The first pour was rejected because of low columbium and because neither the cobalt nor tantalum analyses resulted in consistent values. Both lots of the second pour were rejected because of low chromium and apparently high colbalt. The third pour was rejected because of high cobalt. The third pour was carried through the first forging operation for tube blank piercing. The results of this operation were successful but, in later operations, excessive cracking was noted.

Because of the schedule requirement that core fabrication be initiated by November 1, 1959, and since no other material meeting the specification given in Table V-1 can be obtained until the conclusion of the steel strike, the following alternative procedures for core fabrication have been adopted:

(1) A small but comprehensive effort has been initiated with the General Electric Company, Metallurgical Products Department, Detroit, Michigan, to investigate the feasibility of fabrication of low-cobalt, low-tantalum material to the specification of Table V-1. Special low-cobalt, low-tantalum raw materials as well as standard raw materials will be employed.

TABLE V-1

Composition of Melts of Low-Cobalt, Low-Tantalum Cladding Material

Element	Spec Weight (%)	First Pour Weight (%)	Second Pour Weight (%) Lot 984	Second Pour Weight (%) Lot 985	Third Pour Weight (%) Lot 1094	Third Pour Weight (%) Lot 1095
C	0.05/0.08	0.051	0.51	0.51	0.062	0.056
Mn	2.00 max	0.10	--	--	0.10**	0.10**
Si	1.00 max	0.10	0.49	0.49	0.45	0.43
Cr	17.00/20.00	17.82,18.01	15.05,15.21	15.05,15.27	18.62	18.52
Ni	9.00/13.00	10.41,9.45	--	--	10.29	10.41
Co	0.005 max	0.003*	0.015*	0.015*	0.015	0.032
Ta	0.007 max	0.001*	--	--	0.001**	0.001**
Cb + Ta	10XC 1.0 max	0.27,0.25,0.34	0.53,0.53	0.49,0.50,0.52	0.64	0.67
S	0.03 max	0.009	0.008	0.009	0.004	0.003
P	0.045 max	0.018,0.017	--	--	0.005	0.006

*Analysis not certified

**Limit of detectability

(2) Off-the-shelf Type 348 pierced material ready for drawing into tubing has been purchased from the Superior Tube Company and will be drawn into tubing for delivery October 15, 1959. The composition of this material is given in Table V-2, and is referred to as the "A" core cladding material. It is to be noted that the cobalt and tantalum do not meet our original specification.

TABLE V-2

"A" Core Cladding Material
(Superior Tube Co.)

Element	Weight (%)
Co	0.076
Ta	0.043
Cb + Ta	0.753
Cr	18.29
Ni	11.65
Si	0.54
S	0.006
P	0.013
Mn	1.77

(3) The "A" core cladding material will be used to fabricate the fuel elements to be used for the PM-1 Flexible Zero-Power Test. The elements will be satisfactory for use in a power core except that the cobalt and tantalum content does not meet the required specification. The Flexible Zero-Power Test will, therefore, not be delayed due to a lack of suitable cladding material.

(4) Upon settlement of the steel strike, cladding material meeting our specification can be obtained and used in the fabrication of the "B" or PM-1 power core.

B. SUBTASK 5.2--CONVERSION OF UF_6 TO UO_2

G. H. Krug

During the quarter, this subtask was concerned with the selection of a subcontractor for the conversion of UF_6 to UO_2. During the next quarter, the material will be converted and shipped to The Martin Company.

Specifications for hi-fired UO_2 were prepared and submitted to the following vendors for conversion of UF_6 to UO_2.

Mallinckrodt Nuclear Corporation
St. Louis 7, Missouri

Spencer Chemical Company
Kansas City 5, Missouri

Nuclear Materials and Equipment Corporation
Apollo, Pennsylvania

Davison Chemical Company
Erwin, Tennessee

Based on a delivery price of $325 per kilogram and a guaranteed conversion loss of less than 1%, the Mallinckrodt Nuclear Corporation was chosen as subcontractor for conversion of 87.5 kg of UF_6 into UO_2. Delivery of the first 15-kg batch of UO_2 from Mallinckrodt to The Martin Company is scheduled to be made on October 15, 1959. Conversion and shipment of all UO_2 will be completed during the next quarter.

C. SUBTASK 5.3--FUEL ELEMENT DEVELOPMENT

B. Sprissler J. Kane J. Neace D. Grabenstein

The general objectives of this subtask are to determine the limits of control rod and tubular fuel element fabrication techniques and to determine what refinements of technique can be made.

During this quarter, efforts were devoted to:

(1) A further refining of 0.500-inch OD tubular fuel element fabrication techniques and study of methods of simplifying the fabrication process.

(2) Refining ultrasonic testing techniques.

(3) Studying the effects of various parameters on boron loss.

(4) Performing boron analysis work and developing UO_2 recovery techniques.

(5) Study of control rod materials.

During the next quarter:

(1) The fabrication process for PM-1 fuel elements will be established and a process specification will be written. Investigations of process improvements will continue.

(2) Ultrasonic testing procedures will be established for production elements.

(3) Boron loss tests will be completed and the boron loss study program will be terminated.

(4) Fabrication of control rods for irradiation testing will be initiated.

Equipment modifications.- In order to develop fabrication techniques, several equipment modifications were made during the quarter. The Fenn rolling mill was converted to the 4-high combination and was checked out. Ventilation around the unit was improved and changes were made to reduce contamination around it. The existing hopper was modified to produce a width of green strip that can provide two cores, and the rolling speed and flow through the hopper were changed. It was found that, by controlling the reduction between the first and second sintering operations, the density of the cermet strip was improved.

Irradiation test samples.- The stainless steel cladding material for the irradiation samples was received. Although it met the dimensional tolerances called for, the internal surfaces showed considerable ripping and chattering--the smaller diameter tubing being more seriously affected. The outer surface of the tubing has an abraded finish while the inner surface has an oxide film. Satisfactory results have been obtained with this material in both the furnace-and chemically-cleaned conditions.

Samples of irradiation test elements were fabricated using a cermet blend containing 302B stainless steel, uranium dioxide, and boron carbide. Samples have also been made using a blend of boron-stainless alloy and uranium dioxide.

UO_2 losses during fabrication.- A study was initiated to determine the elongation of a given thickness green strip in order to assure starting with the most economical length. Throughout this study, records will be maintained of the losses in the powder room and of the losses occurring during the shearing and forming operations. At present, UO_2 losses from the powder room to the sheared core are approximately 20%.

Mandrel studies.- The fabrication process studies on tubular elements included an effort to draw 3 concentric tubes using a floating mandrel. This method showed great promise when 3 wrought tubes were used but failed when the cermet core was introduced. The effort was then diverted to a restrained mandrel with results similar to those using the floating mandrel.

It was then found that by introducing a prebonding operation, the assemblies could be successfully drawn on a restrained mandrel. The important factor in this method is the design of the plug, particularly its bearing length and lubrication. A heavy oil (Quaker Cut No. 45) produced satisfactory results. Several segmented full-length elements have been fabricated employing the restrained mandrel. These were cut into 8-inch sections for further study on forming of the ends. Data and observations indicate that: (1) using a restrained mandrel is feasible although some development on plug design is required, and (2) length control must be accomplished between the first and second drawing operations since redrawing through a sizing die is impractical with a restrained mandrel.

2. Ultrasonic Testing

The ultrasonic test program was undertaken to refine present ultrasonic testing techniques in order to increase accuracy and to improve and speed up the testing procedures. The ultimate objective is to set up definitive standards for acceptance or rejection of elements. Since the PM-1 fuel element is longer than previously tested elements, present handling equipment had to be modified. A new and improved method of defect recording is in the process of being developed.

The modifications accomplished included the extension by 3 feet of the large ultrasonic tank used for double-wall transmission. The transport mechanism and accessory equipment were also modified. It is now possible to test fuel elements up to approximately 6 feet in length. The length of the small tank used for single-wall transmission is also being extended 3 feet. The internal crystal has been modified slightly so that fuel elements up to 34 inches in length can be tested in one operation. Only slight modification of the transport mechanism will be necessary.

Using a Brush Electrostatic Recorder in conjunction with a special triggering circuit, several Defect Contour Plot Recordings were made. The circuit is operated by a pulse approximately 4 millisconds wide from the video circuit of the Immerscope. When this pulse exceeds a preset threshold value, a 200-volt signal is sent to the pen of the electrostatic recorder. When the pulse is below the threshold value, a zero-volt signal is sent to the reocrder. Therefore, Whenever a defective area is scanned by the Immerscope, the recorder does not write. Pen flyback (Fig. V-1) was eliminated by placing a cam on the transport mechanism on the lead screw assembly which opens a microswitch, thereby eliminating the pen voltage during pen flyback, At present, a slight graininess, caused by the Immerscope itself, still persists. It is hoped that this may be eliminated shortly by raising the pulse generation rate of the Immerscope to approximately 1500 pulses per second.

The upper trace of Fig. V-2 shows the results of a double-wall transmission test on a standard defected fuel element. It can be seen that no information on the width of the defect causing the defect signal is given. The lower trace is the DCP recording of the same fuel element. Here, both the length and width of the defect can easily be seen. The actual fuel element tested is placed between the two traces.

Figure V-3 shows the resolution of the testing and recording technique. The defects shown are simulated unbonded areas 3/64 inch, 5/64 inch, and 1/16 inch in diameter.

Figure V-4 shows a partial trace of two enriched MPR fuel elements. It can be seen that the core seam on LF-12 is slightly open along the entire length of the element with larger unbonded areas seen along the seam. Element X-415 is seen to be unbonded along the entire length of the trace. Notice that small, closely spaced, unbonded areas can be easily resolved around the wall of the element.

When modification of the testing tanks is completed, fuel elements will be tested and metallographically sampled. From this, the size of the smallest defect that can be noted will be determined.

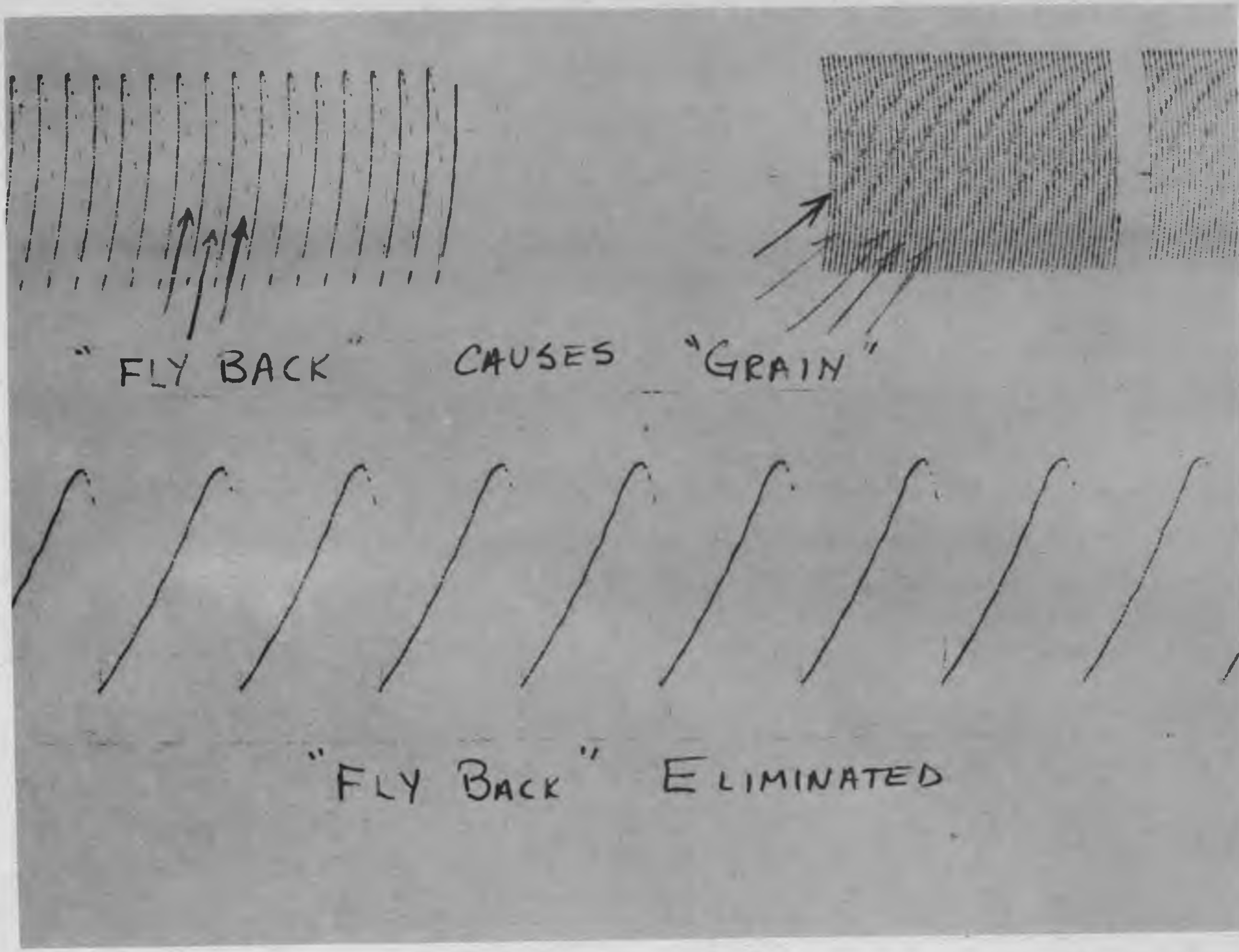


Fig. V-1. Pen Flyback

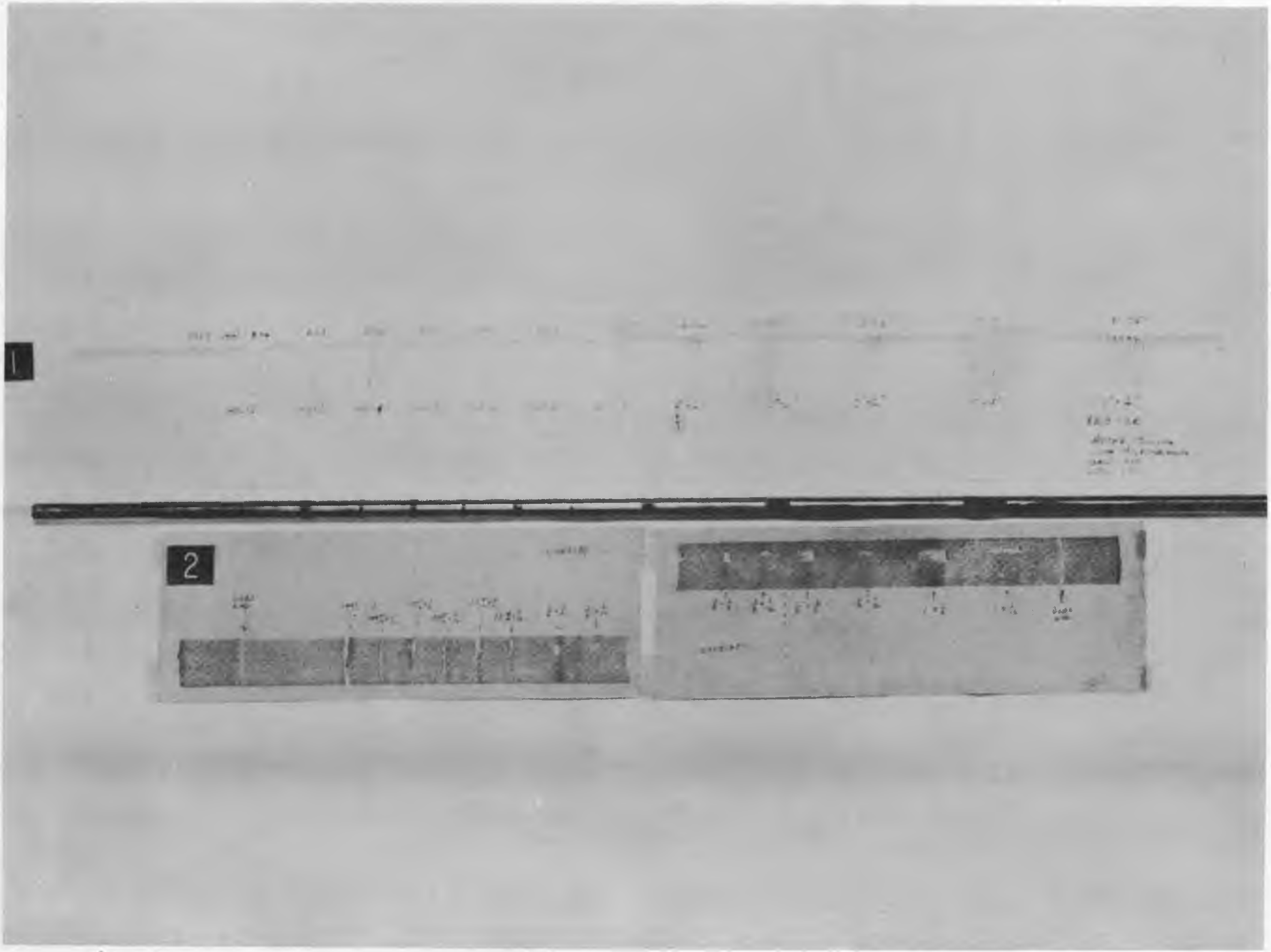


Fig. V-2. Results of Double Wall Transmission Test on a Standard Defected Fuel Element

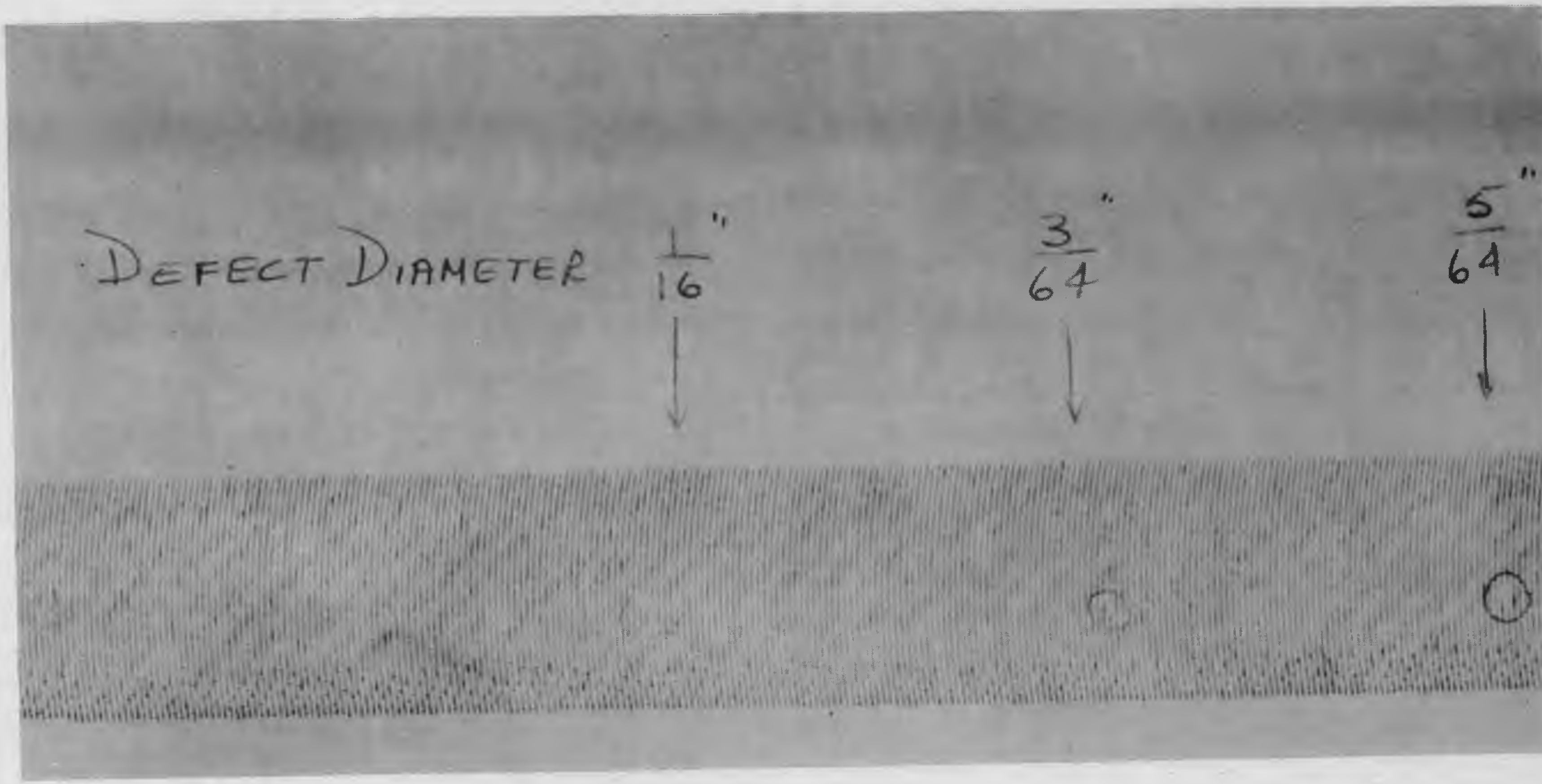


Fig. V-3. Resolution of Testing and Recording Technique

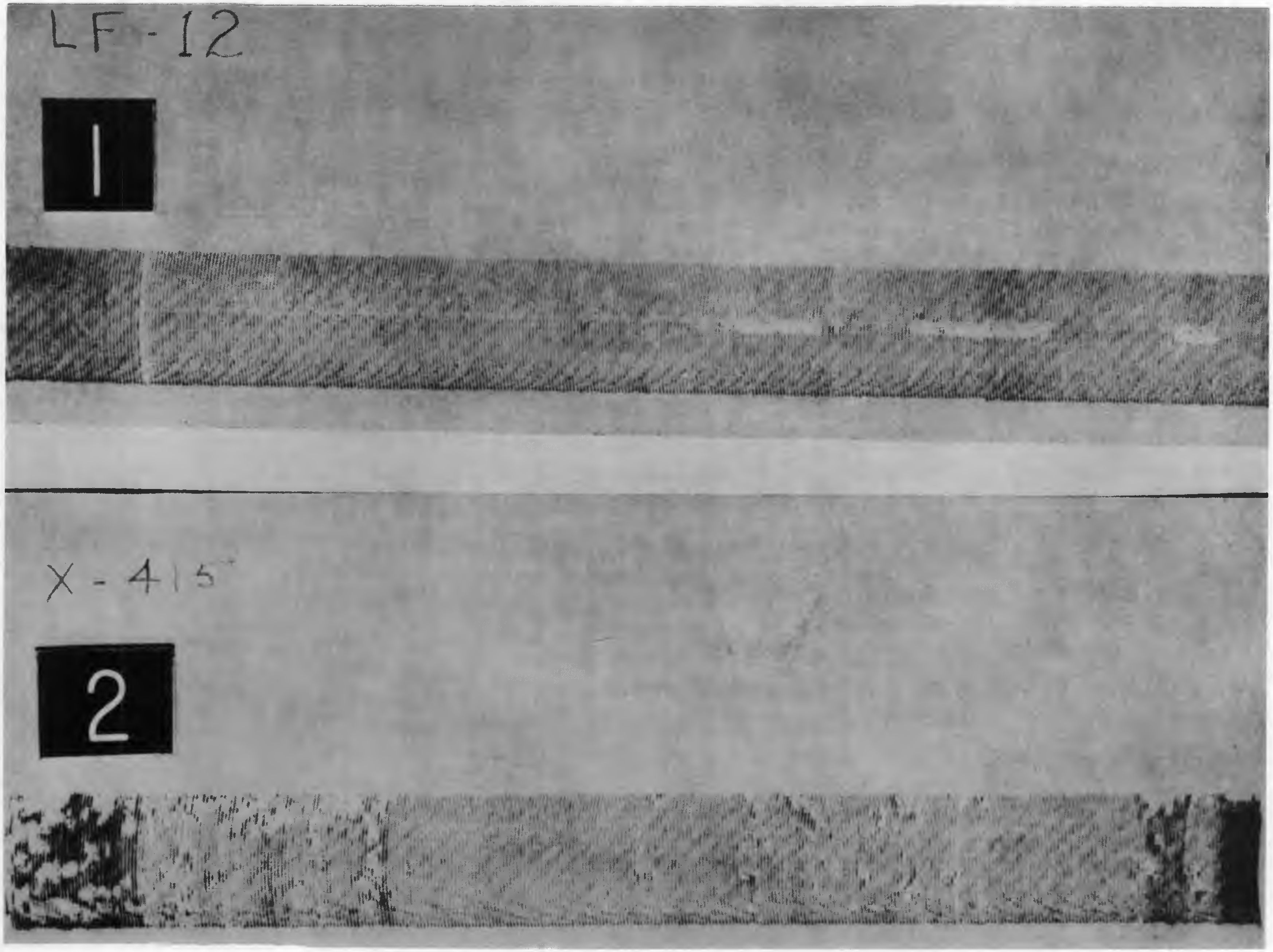


Fig. V-4. Partial Trace of Two Enriched MPR Fuel Elements

3. Control of Burnable Poison

In the first quarterly report, the parameters used in the study were outlined. The effect of temperature on the weight loss and boron loss of specimens containing B_4C and ZrB_2 was given and the effect of temperature on the weight loss and boron loss on an alloy of 302B stainless steel containing 0.38 w/o B was shown. It was found that the specimens containing B_4C and ZrB_2 did not show as consistent results as the boron-stainless steel alloy. The boron loss of the alloy was found to be independent of the amount added but dependent on temperature.

During this quarter, the following studies on boron loss were performed (all studies are not completed):

(1) Questionable data reported in the first quarterly report were rechecked and points not reported in the first quarterly report were determined.

(2) Effect of temperature on a 0.87 w/o B^{10}-stainless steel alloy.

(3) Effect of precleaning stainless steel powders.

(4) Effect of specimen size.

(5) Effect of B_4C particle size.

(6) Effect of UO_2.

(7) Effect of sintering time.

(8) Effect of furnace gas flow rate.

(9) Effect of silicon on steel.

First quarterly data.- In the first quarterly report the analysis for boron loss at 1050° C for the 0.38B-stainless steel alloy had not been completed. The average losses at all temperatures are shown in Table V-3.

TABLE V-3

Temperature vs Boron Loss (0.38 w/o B-Stainless Steel Alloy)

Temperature (° C)	Average Boron Loss (%)
1000	0.023
1050	0.028
1100	0.052
1150	0.076
1200	0.096

The loss at 1050° C falls, as expected, between the results at 1000 and 1100° C.

Retesting the specimens containing B_4C and ZrB_2 at 1050 and 1100°C confirms the hypothesis that the B_4C boron loss increases with increasing temperature, and that the ZrB_2 shows very little loss up to 1100° C but shows increasing loss with increasing temperature from this point. See Figs. V-5 and V-6.

It was hypothesized earlier that B^{10} added to stainless steel in an amount small enough to be in solid solution would be more difficult to remove compared to a material dispersed in a second phase. However, this hypothesis proved to be erroneous since essentially no boron was found at any temperature above 1000° C. Therefore, it can be assumed that either the boron was not in solid solution or diffusion takes place in either case. Data pertaining to effect of temperature on 0.87 w/o B^{10}-SS alloy have not been evaluated.

Precleaning stainless steel powder.- The specimens containing B_4C and ZrB_2 did show lower loss when the precleaned stainless steel rather than the normal stainless steel powder was used. Precleaning of stainless steel powders appeared to have very little effect on the boron loss of the 0.38 w/o B^{10} -stainless steel alloy; however, this test is being rerun. The results are given in Table V-4.

TABLE V-4

Comparison of the Effect of Normal and Precleaned Stainless Steel Powders on Boron Loss with ZrB_2 and B_4C

Identification	B_4C	ZrB_2	Alloy
Precleaned	0.084	0.030	0.086
Normal	0.124	0.070	0.078

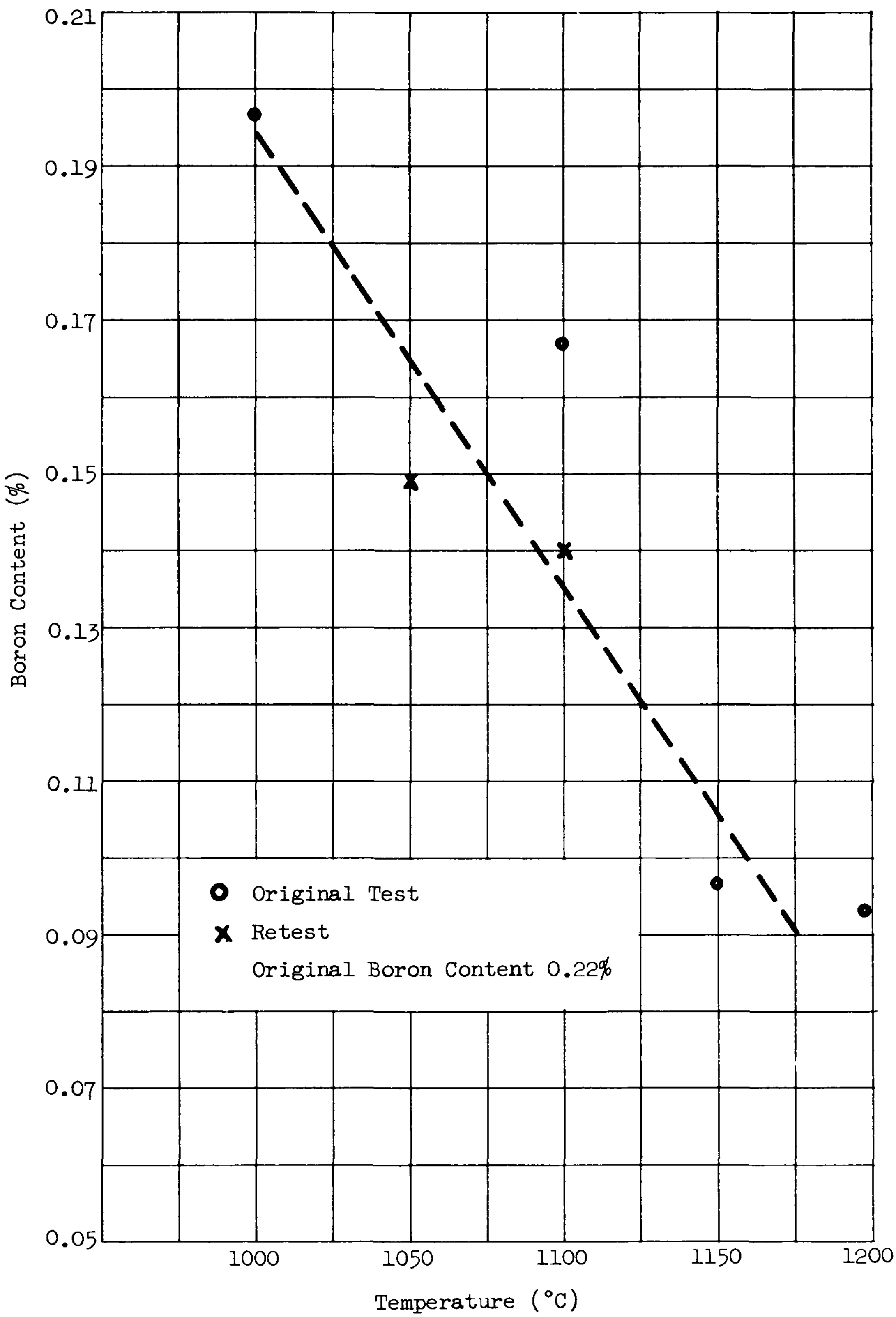


Fig. V-5. Boron Content After Test versus Temperature of Test (B_4C in 302B SS)

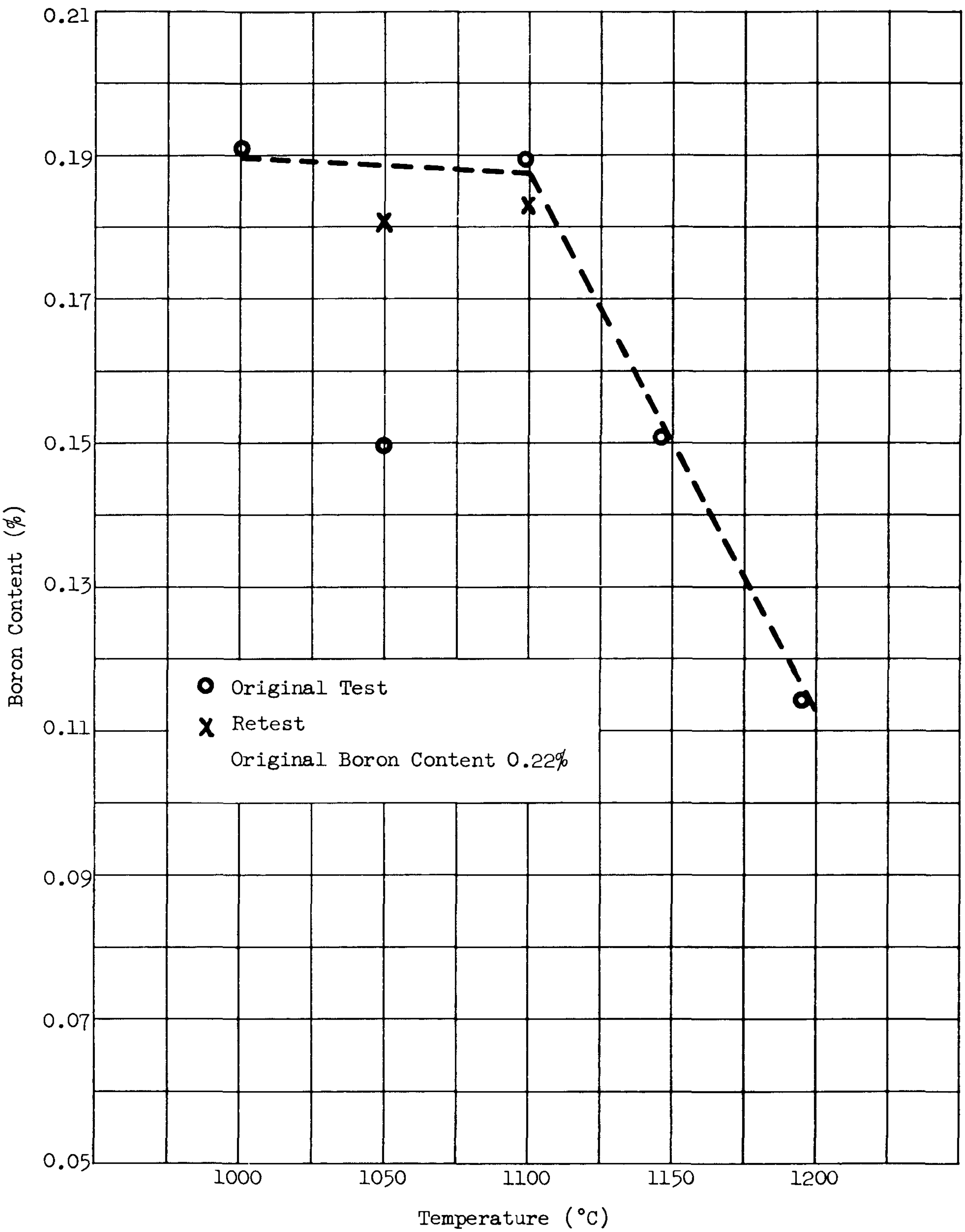


Fig. V-6. Boron Content After Test versus Temperature of Test (ZrB_2 in 302B SS)

All tests were made at 1150° C for 3 hours. The stainless steel powders were precleaned at 1100° C for one minute. At this time and temperature, sintering had taken place and any longer time would have sintered the powder to such an extent that it could not have been powdered.

Effect of specimen size.- It was suspected that the size of the specimens used in the test might have an effect on boron loss. A test was conducted using larger size specimens (1-3/8 inch diameter instead of 3/4 inch for the B_4C and ZrB_2 and 1/2 inch for the alloy). The tests were performed at 1150° C. It was found that the loss was less with the B_4C and ZrB_2 but that the difference was very small f r the alloy.

Effect of B_4C particle size.- The test on the effect of particle size of B_4C on boron loss was not conclusive. It appeared that the loss was slightly greater with the -325 mesh B_4C but considerably more testing would be required to completely establish the effect of particle size. Since no great difference was found, it does not warrant further investigation.

Effect of UO_2.- It was found that adding UO_2 had very little, if any, effect on boron loss at a test temperature of 1150° C and time of 3 hours. A comparison of the loss, with and without UO_2 in the test specimens, is shown in Table V-5.

TABLE V-5

Comparison of Boron Losses With and Without UO_2 in Specimen

Identification			
0 38B Alloy	302B	UO_2 Specimens	Control
100%	0	0.065	0.067
75%	25%	0.074	0.077
50%	50%	0.094	0.090
25%	75%	0.072	0.069
B_4C		0.102	0.105
ZrB_2		0.063	0.052

Effect of sintering time.- Since the boron loss might be dependent on the time that the specimens were sintered, a test was initiated to determine whether or not the loss was linear with time. The results obtained were conflicting in that the test specimens sintered for 3/4 of an hour lost more boron than those sintered for 1-1/2 hours. The results are given in Table V-6. The sintering temperature was 1150° C.

TABLE V-6

Effect of Sintering Time on Boron Loss

Identification		Boron Loss		
0.38B Alloy	302B	3/4 Hour	1-1/2 Hours	3 Hours
100%	0	0.054	0.047	0.066
75%	25%	0.065	0.041	0.077
50%	50%	0.060	0.061	0.090
25%	75%	--	0.035	0.069

Although an additional test in which 4 specimens were removed from the sintering furnace at intervals of 3/4, 1-1/2, 3, and 6 hours has been completed, results of the chemical analysis are not yet available.

Effect of furnace gas flow rate.- The effect of hydrogen flow on the boron loss has not been completely established. However, results at 20 cfh instead of the normal 10 cfh showed lower loss. Only the stainless steel-boron alloy was tested. The sintering temperature was 1150° C. Tests are in progress at flow rates of 5 cfh and 30 cfh.

Effect of silicon in steel.- Tests of a low-silicon stainless steel with B_4C and ZrB_2 at 1050° C and 1150° C sintering temperatures indicated that the boron loss from the B_4C was the same as with the 302B, high-silicon stainless steel. However, with the ZrB_2, the loss was less at 1150° C and there was virtually no loss at 1050° C. A retest was made at 1050° C which confirmed the results with ZrB_2. The results of the low-sil con stainless with ZrB_2 as compared to the 302B stainless steel are shown in Table V-7.

TABLE V-7

Comparison of Boron (ZrB_2) Weight Loss in Low- and High-Silicon Stainless Steel

	1050°C	1150°C
Low silicon	-0.011	0.004
Rerun (low silicon)	0.007	0.069
Regular	0.039	--

The high-silicon (302B) stainless steel is used to aid in powder rolling since irregular-shaped particles are formed by the addition of silicon. It may be desirable to study the addition of boron to a low-silicon stainless steel to see if the lack of silicon improves boron retention in this system.

Throughout the tests it has been found that adding boron as ZrB_2 and B_4C did not yield as consistent results as did adding it as boron-stainless steel alloy. Therefore, the alloy has been recommended as the means of introducing homogeneous burnable poison if it is employed in the elements. A preliminary study by our manufacturing personnel has indicated that the results found in these tests are applicable during manufacturing. Besides more consistent test results, the alloy has been recommended for the following reasons:

(1) A more uniform dispersion of boron should be obtained.

(2) Mixing problems should be eliminated.

(3) Radiation damage from boron should be low if present at all. KAPL reports that a stainless steel containing 1 a/o B^{10} shows very little radiation damage; our alloy contains less than 0.5 a/o B^{10}.

(4) The amount of boron can be controlled by mixing a stainless steel alloy, containing an excess of boron, with straight stainless steel.

(5) Problems of analyzing for boron control should be minimized since the sampling from a homogeneous mixture should be much more certain than from a dispersion of B_4C or ZrB_2.

4. Boron Chemical Analysis

The effort during this period was divided into two phases: (1) Analysis of the boron and uranium content of various specimens was performed using newly-developed analytical methods; and (2) development work on UO_2 recovery by a sulfuric-acid leach method was undertaken. The physical characteristics of the recovered material will be investigated prior to using it for fuel element fabircation.

The following samples were analyzed for boron and uranium as a support service to other activities:

Type of Sample	No. of Samples
B - SS	137
B_4C - SS - UO_2	44
ZrB_2- SS	40
B_4C - SS	30
B - SS - UO_2	27
ZrB_2 - SS - UO_2	4
ZrB_2	3

A sample of B-SS master alloy powder was obtained from ORNL and analyzed. The alloy was reported to contain 0.25 w/o boron as 92.74% B^{10} in Type 304SS. Analysis in the Martin Nuclear Division laboratory yielded a value of 0.264 w/o.

Approximately 1000 grams of clad SS fuel elements were obtained from the vault and the UO_2 content was recovered by the sulfuric-acid leach method. Chemical analysis of the recovered material showed a UO_2 content of 98.0% and a boron content of 0.04%. Some of the particles of UO_2 have fused together to f rm aggregates larger in particle size than the original starting material. The cause of this has not yet been determined.

5. Control Rod Studies

Studies were initiated to determine the most suitable control rod material for use in the PM-1. Materials being considered are:

Boron as cermet and stainless steel alloy
Europium
Silver-indium-cadmium
Gadolinium-samarium

Factors being evaluated are:

Fabrication of parent material and rods
Nuclear characteristics
Control rod burnup

Radiation stability
Cost

A program has been initiated to investigate the fabrication of control rods containing Eu_2O_3 and mixtures of Gd_2O_3 and Sm_2O_3.

VI. TASK 11--SITE PREPARATION AND INSTALLATION

Project Engineer, Subtask 11.1--G. Zindler

The objectives of this task are to prepare the site for the orderly installation of PM-1 packages and to install and interconnect the packages into an operable nuclear power plant.

A. SUBTASK 11.1--SITE PREPARATION

This subtask is concerned with site preparation prior to installation of the PM-1.

Approval was received from the USAEC to locate the PM-1 on the eastern slope of Warren Peak. This site was recommended by The Martin Company after evaluation of several possible locations (see MND-M-1812). Based upon this approval, action was taken to have the location surveyed in detail and to obtain core borings which will provide detailed information for exact power plant siting, for the design of the foundations, for grading, for roadways, etc.

The results of these field engineering surveys have not been received at this writing from the subcontractor, the firm of Porter, Urquhart, McCreary and O'Brien of San Francisco.

The report of the field engineers is expected to be delivered during the next project quarter for evaluation and study.

A current layout and elevations of the plant are shown in Fig. VI-1.

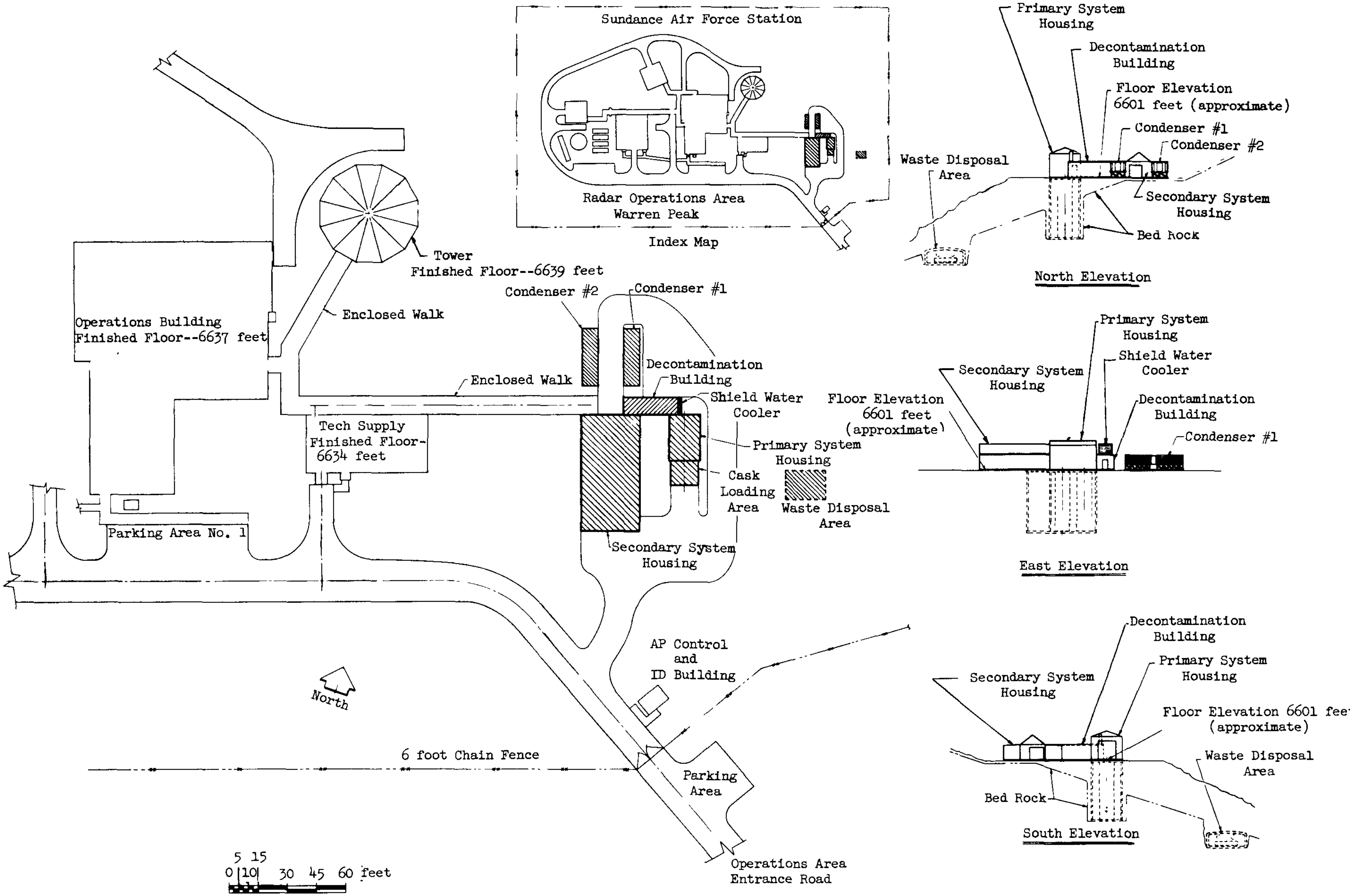


Fig. VI-1. PM-1 Nuclear Powerplant Site Layout and Elevations

VII. TASK 14--TRAINING

Project Engineer--F. McGinty

The objectives of this task are to develop and implement a program to train competent military personnel to supervise and conduct operation and maintenance on the PM-1 Nuclear Power Plant.

A. SUBTASK 14.1--TRAINING PROGRAM DEVELOPMENT

Efforts under this subtask were directed toward completing an analysis of student background, defining the Training Manual, and initiating work on the general training plans, and course scope.

During the next quarter it is expected that:

(1) Course outlines will be completed.

(2) The final draft of the Training Manual topical outline will be completed.

(3) Basic training equipment will be selected.

The first PM-1 training meeting was held at Fort Belvoir. Military, AEC and Martin Company representatives participated. Results of the meeting were:

(1) The establishment of January 16, 1961 as the tentative starting date for training.

(2) The decisions that:

   (1) Course outlines and lesson plans prepared by The Martin Company for PM-1 training will follow the format used by the Nuclear Power Training Branch at Ft. Belvoir.

   (2) The Training Manual will be for instructors and will contain information related to training such as: course outline, lesson plans, training equipment lists, visual aid lists, training charts, evaluation tests, etc. It will not contain detailed PM-1 technical data.

   (3) Student job assignments will be educational, constructive, and appropriate to rank and technical skill.

Investigation of student technical background (personnel analysis) composed the major effort in the development of the training program during this quarter. The sources of information used in this effort were:

(1) The program of instruction for the Nuclear Power Plant Operators Course (NPPOC):

  (1) Academic Phase

  (2) Specialty Training Phase

  (3) Operation Phase.

(2) Student Manuals used in specialty and operations training phases.

(3) Personal discussion with the Training Branch personnel in the four specialty areas (instrumentation, electrical, mechanical, and process control) and in operations.

(4) Review of maintenance records of the SM-1 reactor to determine the level of maintenance accomplished by on-site personnel.

Tentative conclusions from this study support our original hypothesis that the major portion of the formal training on PM-1 should consist of system and operation training, with a minimum allowance for training on major component functions. After the selection of specific PM-1 system components and equipment has been made, those items not adequately treated in the present scope of the NPPOC will receive detailed coverage during PM-1 training.

The first draft of the Training Plan is being prepared. It will contain:

(1) The concept of PM-1 training

(2) Course charts

(3) A syllabus of instruction

(4) Budget and manpower projections

(5) Facilities and materials requirements

(6) A training manual concept

(7) Program development--time sequence charts.

A draft of the syllabus of instruction for the training plan has been completed.

Work has been initiated on the development of the course outlines, the final draft of the training manual topical outline, and the selection of training equipment.

VIII. TASK 15--PROJECT SERVICES

Project Engineer, Subtasks 15.1, 15.2--C. Fox

The objectives of this task are the provision of documentary films and photographs, and the construction of models to support the PM-1 Project.

A. SUBTASK 15.1--PROJECT FILM AND PHOTOGRAPHS

This subtask was scheduled to become active during this quarter with the selection of personnel and the collection of site photographs.

During the next quarter, the outline of the project documentary film will be prepared and tested and project work will be filmed.

During this quarter, the Project Engineer was selected and lead men were assigned from the presentations section, including the film supervisor, a camera man and a script man. Film footage was obtained of the site at Sundance, Wyoming. A number of stills also were taken showing the site and the site survey activities (see Figs. VIII-1 through VIII-6).

The subtask budgetary allowance was reviewed and found adequate for a single documentary film. It is recommended, however, that a high footage-to-film ratio be maintained to enable production of any additional progress, training and topical films which may prove desirable.

A PM-1 photograph file was established to permit ready reference and reorder.

Fig. VIII-1. Core Drilling Rig and Drill Crew on Location at the Site. Part of Survey Team with Transit Shown at Right

Fig. VIII-2. Hutchins Spring, Located One-Half Mile NW of the Site, the Proposed Sole Water Source for Both Radar Base and PM-1

Fig. VIII-3. Approaching Warren Peak from the SE. Approximately One- Half Mile from the Site

Fig. VIII-4. A View of the Site from the Lower Road, Looking Approximately Due West

Fig. VIII-5. Sample Core Borings at the Site

Fig. VIII-6. A View of the Site from the Lookout Tower, Looking Approximately Due East

IX. TASK 16--CONSULTING

The purpose of this task is to secure expert technical advice as required for the PM-1 Project in the areas of power plant engineering and operation, reactor physics, and applied nuclear engineering.

The Gibbs and Hill Company provided consulting support in 3 main areas during this quarter:

(1) Review of secondary system work performed by Westinghouse.

(2) Review of outline specifications for secondary system components.

(3) Preparation of 3 outline specifications.

Complete review, evaluation, and recommendations were made on the secondary system heat balances, flow diagrams, control diagrams, and system layouts.

The 3 outline specifications prepared were:

(1) MN-2003 "Piping Requirements for PM-1"

(2) MN-2004 "Electrical Requirements for PM-1"

(3) MN-2005 "Welding Requirements for PM-1."

Nuclear Engineering consultations with Dr. Thompson of MIT continued.

During the next quarter, the consulting efforts of Gibbs and Hill will be continued. The scope of their efforts will be extended to include final engineering and design work on the entire PM-1 secondary steam-electric system. This will include final selection of equipment and preparation of final procurement specifications and detailed layout and arrangement of the secondary system equipment within the packages--all in conformance with the preliminary design already completed under subtask 3.2.

Consultations with Dr. Thompson will continue.

X. TASK 17--REPORTS

Project Engineer, Subtask 17.1--G. Zindler

The objective of this task is to accomplish the timely preparation of those reports required by the USAEC.

A. SUBTASK 17.1--HAZARDS SUMMARY REPORT

This subtask is concerned with the preparation and submittal of the Hazards Summary Report for the PM-1 Nuclear Power Plant.

The efforts under this task were applied in two major areas during the reporting period.

1. Site Background Survey

A team of Martin Company personnel visited the PM-1 site to gather information and samples for establishing existing site radiation levels. Samples taken included soil, vegetation and water. These were gathered from the immediate area of the PM-1 site and from outlying streams, etc. The samples gathered will be assayed for radiation levels and retained for future reference. During this trip, information was gathered concerning exact population distribution and estimated land utilization in the area during the summer months. This latter effort will support directly the preparation of the Preliminary Hazards Summary Report.

2. Preliminary Hazards Summary Report

During the later part of this quarter, the preparation of the Preliminary Hazards Summary Report was initiated in those areas other than site evaluation. (The site evaluation study was reported in the last quarter. See MND-M-1812.) The basic evaluation was augmented by the field trip mentioned above.

The Hazards Summary Report will provide an evaluation of potential hazards presented by the PM-1 as defined by preliminary design. The evaluation efforts now underway include:

(1) Determination of the maximum credible incident

(2) Effects of reactor excursions

(3) Determination of cloud dosages

(4) Effects of potential accidents.

The Preliminary Hazards Summary Report will be completed by 15 October 1959.

B. SUBTASK 17.2--REPORTS OTHER THAN HAZARDS

J. S. Sieg E. H. Smith

This subtask includes all reports submitted to the USAEC except those on hazards.

During the second project quarter:

(1) The PM-1 parametric study topical report (MND-M-1852) was prepared.

(2) The PM-1 preliminary design topical report was in preparation at the close of the quarter.

(3) The first Quarterly Progress Report on the PM-1 Nuclear Power Plant Program, (MND-M-1812) was prepared and delivered to USAEC.

During the next project quarter:

(1) The parametric study topical report will be delivered.

(2) The preliminary design topical report will be completed and delivered.

(3) The second quarterly progress report will be completed and delivered.