MND-M-1812

REACTORS — POWER

PM-1 NUCLEAR POWER PLANT PROGRAM QUARTERLY PROGRESS REPORT NO. 1 FOR MARCH 9 TO MAY 31, 1959

By
J. S. Sieg

July 6, 1959

Nuclear Division
Martin Company
Baltimore, Maryland

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PM-1 Nuclear Power Plant Program

1ST QUARTERLY PROGRESS REPORT

9 March to 31 May 1959

MND-M-1812

UC-81

Contract AT (30-1)-2345

6 July 1959

Prepared By:

J. S. Sieg

J. S. Sieg

Approved By:

F. Hittman

F. Hittman
Project Manager

MARTIN
NUCLEAR DIVISION

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ABSTRACT

This report contains a description of the work accomplished during the first contract quarter (9 March to 31 May 1959) of Contract AT (30-1)-2345 between The Martin Company and the USAEC.

The contract calls for the performance of design and development, final design, fabrication, installation, and initial testing and operation of a factory pre-packaged air-transportable pressurized water nuclear power plant. The plant is to produce 1000 kw of net electrical power and 7,000,000 Btu/hr of heat suitable for central heating. It is scheduled to be operational, at a government site, by 9 March 1962.

The principal effort of the quarter was directed toward parametric study of reactor, system, and configuration design variables. An experimental program was initiated to confirm analytical results and, in the case of fuel elements, to perform prefabrication studies.

Although the parametric studies are not scheduled to be completed until the second quarter--when they will provide the basis for preliminary design--numerous results of individual studies are included in this report. The conclusions drawn from these studies will be presented in the second quarterly progress report.

FOREWORD

This is the first 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 9 March through 31 May 1959.

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PROGRAM HIGHLIGHTS

1. A contract was signed with the Stromberg-Carlson Company for the performance of research and development work leading to the design and procurement of a solid state nuclear instrumentation system (Task 1).

2. A contract was signed with the Westinghouse Electric Corporation for the performance of research and development work leading to the procurement of a packaged secondary system (Tasks 1 and 3).

3. The use of an air-steam condenser was found to be feasible and desirable (Task 1). A testing program for such a condenser is being considered.

4. The parametric study portion of the experimental zero power test program was completed (Task 2).

5. The parametric study neared completion (Task 3).

6. Fabrication of 1/2 in. OD fuel tubes, containing as high as 30 wt % UO_2 in the meat, was found to be feasible (Task 5).

7. A location for the PM-1 plant within the Sundance Air Force Station was recommended (Task 11).

8. Site data were collected in support of the preparation of the preliminary Hazards Summary Report (Task 17).

INTRODUCTION

This is the first of 12 quarterly progress reports required by Contract AT (30-1)-2345 between The Martin Company and the USAEC.

The basic objectives of the PM-1 program are to design, to fabricate, to install, and to test a factory pre-packaged pressurized water power plant which meets the following general criteria:

(1) The plant shall be air-transportable by packages.

(2) It shall consist of a minimum number of packages.

(3) Since installation is contemplated in frigid climates, expeditious package placement and interconnection will be required; site installation procedures shall therefore be as simple as possible.

(4) Plant reliability shall be high; maintenance requirements shall be minimized.

(5) The core shall be designed to deliver full power for two years.

(6) The net plant output shall consist of 1000 kw of high quality electrical power and 7,000,000 Btu/hr of space heat.

(7) The plant shall be designed for operation with a minimum sized crew.

(8) A major portion of the plant must be easily relocated.

The program was divided into the following tasks:

(1) Preliminary Design--System Development

(2) Preliminary Design--Reactor Development

(3) Preliminary Design--Study, Selection, and Specification

(4) Final Design

(5) Core Fabrication

(6) Dummy Core Fabrication

(7) Fabrication and Assembly of Plant

(8) Packaging

(9) Preshipment Test

(10) At-Site Testing

(11) Site Preparation

(12) Package Loading Demonstration

(13) Manuals

(14) Training

(15) Project Services

(16) Consulting

(17) Reports

During the first quarter, Tasks 1, 2, 3, 5, 11, 14, 16, and 17 were active; in addition to the preceding, Tasks 4 and 15 are scheduled to become active during the second quarter.

Progress will be reported by subtasks. The Project Engineers responsible for the various subtasks are identified. The scientists and engineers who have made significant contributions are identified by subtask areas.

Subsequent reports will contain a summary of design parameters.

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

Project Engineer--Subtasks 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 1.2--INCORE INSTRUMENTATION

G. Zindler

Core instrumentation is required if temperature, flow, flux distribution and fuel burnup data, not readily obtainable from detectors located outside the reactor, are to be obtained. Such information is desirable to provide for the safe and efficient operation and understanding of the core, and to provide data upon which future design improvement may be based.

Major subtask effort, during the first quarter, was devoted to compilation of general incore instrumentation data. Information compilation will be completed during the second quarter, and preliminary design will be accomplished.

A literature survey was completed and available documents were obtained and reviewed.

A feasibility study was started to determine whether techniques which have been applied by others, and about which operating data have been collected, may be used to advantage on the PM-1 Project. Attempts to obtain core instrumentation data from the Naval Reactor Program have been unsuccessful to date.

Discussions were held with Alco Products, Incorporated, (Alco) concerning their feasibility studies of APPR-1B incore instrumentation. Their efforts covered design analysis of a system to measure incore flow and temperature. Major problems encountered included the routing of signal cables and tubes through small flow channels without changing or disrupting flow paths, and swelling of the mineral (magnesium oxide) signal cable insulation when wetted. Swelling results in cracking of the metal cable sheath thereby allowing the reactor coolant to attack the signal-carrying wires. Although the use of other types of insulation is being considered, no specific solution to this problem is known.

With completion of the feasibility study in the next quarter, a preliminary design will be started on an instrumented subassembly for the PM-1 core. This design will take advantage of information gained during the feasibility study and will allow assessment of the problems of installing detectors and signal lines in the space gained by removing several tubular fuel elements.

B. SUBTASK 1.5--INSTRUMENTATION

G. Zindler

Efforts under Subtask 1.5 are directed toward development of control system designs and specific control components for the PM-1 plant. Emphasis is being placed on improved reliability and greater simplicity of systems. In the primary system, development of new component designs is included; in the secondary system, efforts are concentrated on selection of appropriate controls from existing equipment.

1. Primary System Controls and Instrumentation

This effort is subcontracted in total to the Stromberg-Carlson Company, Rochester, New York. Their research and development effort is directed toward increasing reliability and simplifying PM-1 control and operation.

Efforts during this reporting period included preparation of a statement of work for the contract and negotiations as to the level of effort and cost. Following these, on 19 May 1959, approval was given the subcontractor to proceed. His areas of investigation are as follows:

(1) Reliability analysis--This analysis is based upon a complete and integrated control system as it could be delivered without development. It does not take into account any of the modifications resulting from the development efforts under way at this time. Reliability analysis of these developments will be covered in final design efforts.

(2) Fault monitoring and self-checking--This effort is directed toward improving nuclear power plant fault monitoring and self-checking. Increased reliability through more expeditious diagnosis of failure and repair and maintenance of equipment is expected to result.

(3) Control console configuration--A human engineering approach is being taken in specifying size, shape and arrangement of the PM-1 control console. Reduction of the manpower required to safely operate the power plant is of major importance.

(4) Control room design criteria--A design criteria is being prepared, again with a human engineering approach, to provide efficient working conditions.

(5) Automatic startup controls--Effort is being directed toward providing automatic startup control for the PM-1. This would cover the range from cold shutdown to a power level sufficient to bring a primary loop to near design temperature and pressure--although little or no heat is delivered to the secondary loop. Studies of system behavior during automatic startup are discussed under Task 3.1.

(6) Reactor period instrument development--Because existing period measurement instruments have been plagued by the poor signal-to-noise-ratio found in reactor startup instrumentation, a development effort is underway to provide a superior period measuring system.

2. Secondary Loop Controls and Instrumentation

The secondary loop steam generator control and instrumentation study, by Westinghouse Electric Corporation, has been directed toward the use of proven components suitable for a plant of this size. It has been assumed, from experience with other operating pressurized water plants, that no feedback from the electrical generator output to the primary control system will be necessary. (This assumption is being investigated.) Using these ground rules, a preliminary control diagram for the secondary loop steam generator system has been developed and is shown in Fig. I-1. A preliminary list of equipment associated with the secondary system was developed. Table I-1 shows the direct readings required, Table I-2 the recorders required, Table I-3 the annunciators required and Table I-4 shows the remotely operated controls.

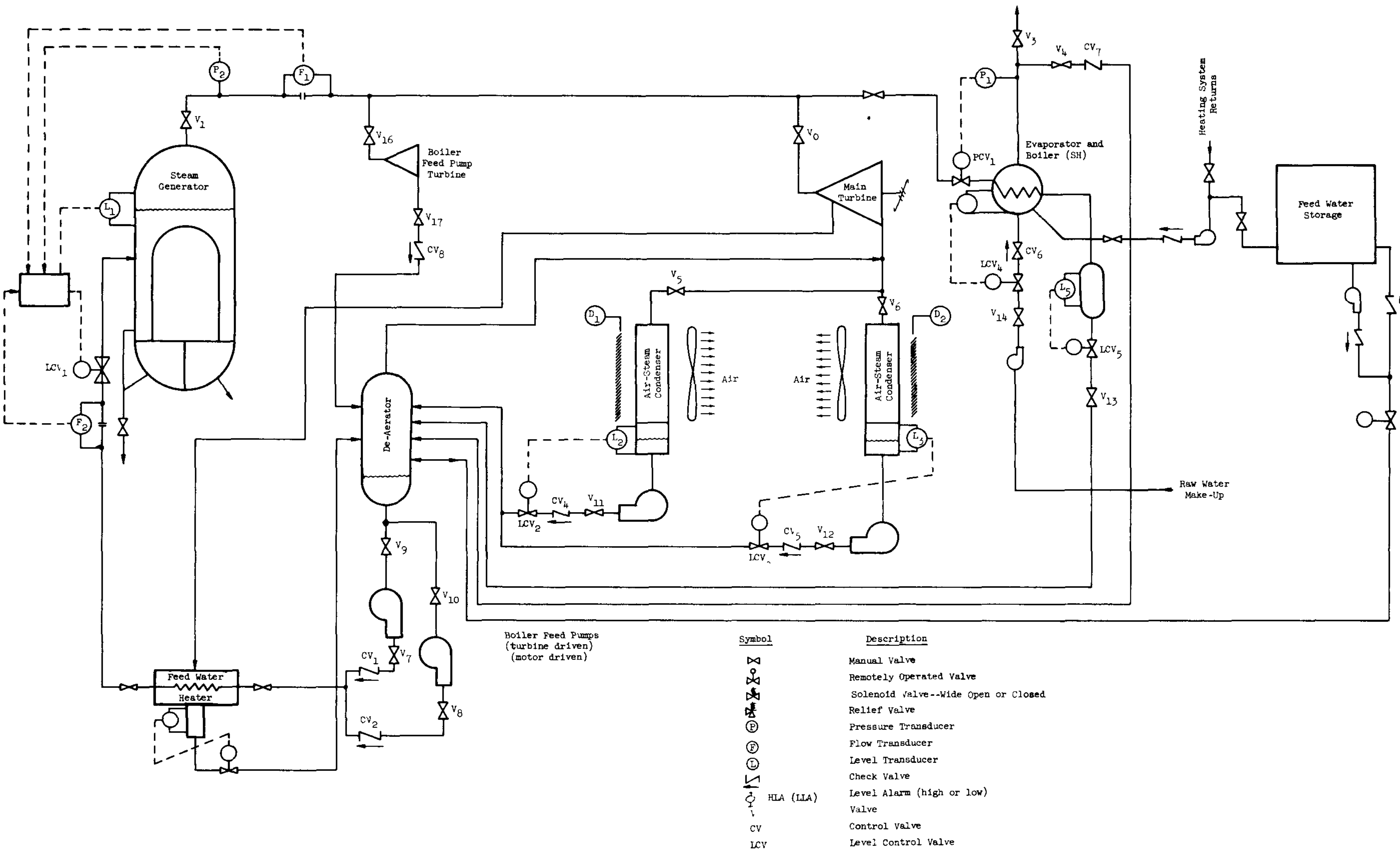


Fig. I-1. Preliminary PM-1 Control Diagram

TABLE I-1

Control Room Direct Readings

Steam flow from steam generator	Evaporator pressure shell side
Steam pressure	Evaporator pressure tube side
Steam generator water level	Evaporator water level
Condenser pressure	Process steam flow
Voltage	Feed storage tank level
Amperage	Feedwater storage tank temperature
Power output	Condenser hot well level (2)
Turbine inlet pressure	Condenser temperature (4)
Turbine extraction pressure	Deaerator pressure
Turbine lubricating oil temperature	Deaerator level
Turbine lubricating oil pressure	Feedwater temperature
Turbine bearing temperature	Turbine speed
Generator bearing temperature	Feedwater flow
Generator starter temperature (2)	Miscellaneous (6)

TABLE I-2
Recorders

Kilowatt hours	Steam generator water level
Steam flow from steam generator	Condenser pressure
Steam pressure at steam generator	Feedwater temperature
Feedwater flow	

TABLE I-3
Annunciators

Steam generator water level--low	Feedwater storage tank level--low
Steam pressure--low	Feedwater temperature--low
Condenser pressure--high	Condenser temperature--low
Voltage--high or low	Condenser temperature--high
Turbine lubricating oil temperature--high	Deaerator water level--low
Turbine lubricating oil pressure--low	Feedwater pump failure
Turbine bearing temperature--high	Condensate pump failure
Generator bearing temperature--high	Condenser fan failure
Generator station temperature--high	Turbine overspeed
Evaporator water level--low	Miscellaneous (4)

TABLE I-4

Remotely Operated Controls

Turbine speed control	Service water pump
Synchronization panel	Turbine lubricating oil cooler fans (2)
Boiler feed pump	Battery charger
Condensate pumps (4)	Voltage control
Condenser fans (8)	Vacuum pump (2)
Feedwater transfer pump	Evaporator feed pump (2)
	Miscellaneous (10)

The degree of automation that can be obtained within the scope of the PM-1 requirements and objectives is being investigated. Under normal operation, modern steam-electric systems of this size are fairly "automatic". A very important factor, when considering the desirability of providing automation, is the determination of how many operators can be replaced. With the control scheme discussed, it is considered that one secondary loop operator per shift will be sufficient for normal operation. One qualified machinist, an electrician, and a janitor-handy man will be required as general backup for one shift per day. Further reduction in operator requirements does not appear feasible.

A program for developing a mathematical model of the steam generator and secondary loop for analog computer study is underway. This will check the present control philosophy and will allow study of the effect of various factors on power quality.

Normally, pneumatic controls would be used on steam electric systems of this size. But, because of the low temperature operating requirements, other types of controls are being evaluated. An electric three-element steam generator control that has been used in some naval applications is available; it will be studied for PM-1 application.

Electrically and hydraulically operated motor valves of integral design are being studied. At present, the latter appears easier to control and more suited to our plant. This study will be completed during the preliminary design phase of the contract.

The use of "in-line" automatic water sampling devices, such as those manufactured by the Cambridge and the Swarthout Companies for pH and oxygen content determination, are being investigated. Also, the use of portable "plug-in" instrumentation during the PM-1 startup and six-month test period is being analyzed. Both of these studies will be completed during the preliminary design phase.

C. SUBTASK 1.6--SECONDARY SYSTEM DEVELOPMENT

W. Koch

Work under this subtask was subcontracted to the Westinghouse Electric Corporation and was directed toward solving power quality problems and accumulating and evaluating data on direct air-steam and air-glycol-steam condensers.

During the next quarter a specific approach to solving the power quality problem will be selected, a condenser system will be selected, and development work will be initiated, if necessary, on equipment specified during the preliminary design.

1. Power Quality

The power quality problem is three-fold; the meeting of voltage fluctuation criteria, the meeting of deviation factor and harmonic content criteria, and the meeting of frequency regulation criteria.

Three methods are being investigated to meet the voltage quality objective that fluctuations be limited to ±0.5% during steady state, and ±2% between 10 and 120% of rated load when the plant is subjected to an instantaneous load change of 30% of rated capacity at 0.80 power factor.

The first method studied, apparently the most feasible, is to use a generator having a fast acting, static excitation, voltage regulator system and enough iron in its construction to lower its transient reactance. Three different types of generators were considered:

(1) A 1200-rpm salient pole type

(2) A 1800-rpm salient pole type

(3) A 3600-rpm non-salient pole type.

Each of these types was considered at ratings of full and one-half of required plant capacity. The studies on Systems 2 and 3 have not been completed, but preliminary information makes them appear less desirable than System 1.

Table I-5 shows the results of the study on the 1200-rpm salient pole machines rated at 650 and 1250 kw.

TABLE I-5

1200 rpm Salient Pole Generator

	650 kw		1250 kw	
Transient reactance--%	25	8	25	8
Temperature rise--°C	70	50	70	50
Applied load--kw	150	150	300	300
Voltage dip--%	6	2	6	2
Generator weight--lb (without exciter)	6350	10,400	9400	18,000

It can readily be seen that a considerable weight penalty is paid when the transient reactances of the generator are decreased to the required 8%, but it is felt that the generator weights can be reduced during the design development effort, and that this approach can be utilized successfully. As mentioned in the turbine generator discussion under Subtask 3.1, the oversized 1250-kw generator (18,000 lb) can be designed into a turbine generator set weighing less than the shipping weight limit of 30,000 lb.

The second method being investigated utilizes an electric load dump device to impose a load ramp transient, instead of a load step transient, on the generator. If the ramp transient can be applied properly, a standard generator could be used and the voltage kept within the desired limits. Although this investigation has not been completed, the approach does not appear feasible. This is mainly due to problems associated with controlling the switching of the compensating loads to and from the line. These problems are exaggerated by our desire to avoid actuating the PM-1 nuclear control system by load-anticipating devices.

The third method utilizes a synchronous condenser to increase the power factor at the generator to near unity, thereby eliminating the effect of generator reactance. This study has not been completed, but it appears that switching problems will limit its feasibility.

The use of the oversized generator and fast acting static excitation voltage regulator system requires no elaborate control system and/or feedback from the load as is the case with the electric resistance load dump device system, the synchronous condenser system, or systems involving load change anticipating devices at the load.

Meeting the harmonic content criteria cannot presently be guaranteed through design, because it is load dependent, and detailed load data are not available. The no-load deviation factor of the generators under consideration will not exceed 2%, the rms of all harmonics will be less than 2% and no single harmonic will exceed 1.5%. The harmonics that may exceed 0.75% are the 3rd, 5th, 7th, and 9th. Since the 3rd and 9th will be zero in the line-to-line voltage, any excesses of these harmonics over 0.75% would appear to be of little import. Either the 5th or 7th harmonic may be reduced at the expense of the others. The remaining harmonic can be reduced by a filter if necessary, but only after the generator has been completely designed.

The power quality objective concerning frequency states that the plant shall be inherently capable of limiting frequency fluctuations to within ±2% between 10 and 120% of rated load, when subjected to an instantaneous load change of 30% of rated capacity at 0.80 power factor; that recovery time from the initiation of the transient to steady state condition shall not exceed 1.5 sec; and that, under steady state conditions, the plant shall limit frequency fluctuations to ±0.25%. A complete investigation is being made of these criteria, but it appears that they can be met without difficulty by using an isochronous governor. This type governor is being used successfully on such turbine-driven equipment as paper dryers, nylon spinning machines, etc., where speed must be regulated to within ±0.1%.

2. Condenser System Study

The use of a direct air-steam condenser system may have weight and equipment simplification advantages. To ascertain if such is the case, Gibbs and Hill, Incorporated conducted an independent analysis of condenser systems in support of the Westinghouse Electric Corporation subcontract study.

The Gibbs and Hill report covered tubular surface condensers utilizing ethylene glycol or direct air coolers, and a direct contact condenser. In the latter system, the steam is condensed by the injection of water into the condenser; the injected water having been cooled in an auxiliary air cooled condenser. All systems were considered for ambient temperatures of - 70 to 70° F.

The results of the Gibbs and Hill study may be summarized as follows:

(1) Ethylene-glycol system--The major advantage of the ethylene glycol system is the elimination of freezeup problems permitting a simpler control system and less insulation.

The disadvantages include the requirement for glycol storage facilities, special seals for use with ethylene glycol, additional circulating pump power, higher system weight, and more packages.

(2) Direct air-steam system--The direct air-steam condenser system consists of a finned tubular air blast condenser; waste heat is rejected to the atmosphere.

Three air-steam condenser systems have been studied. These include the installation at the National Reactor Test Station, the Eilson Air Force Base installation, and the power trains built for the USSR during World War II.

SL-1.- Data regarding this system were obtained from the Modine Company and Mr. A. Smaardyke, Argonne National Laboratory.

The system consists of five finned tubular coil sections fabricated from 6061 aluminum, ASTM-GS-11A, with the coils so connected that the air passes over each coil in series. The steam flows in four parallel paths downward through the last four coils, the condensate draining to a bottom header. The first coil, a precooler, is connected to the same header. The blower-type fan is driven through a variable speed fluid drive. A ducting and damper system mixes fresh and recirculated air in a manner to preclude condenser freezeup. The equipment has operated smoothly with inlet fresh air temperatures as low as -24° F. The entire system is located indoors in a space provided with emergency space heaters which may be used to help prevent freezeup.

Unfortunately, this system has never been able to meet its design requirements. Figure I-2 shows actual and calculated performance curves. The manufacturer believes that half of its reduced capacity is due to faults in piping configuration and the other half to lack of sufficient heat exchange surface. The operators have been unable to balance the steam flow in the last four sections with respect to heat transfer surface. An analysis of the system indicates that the mass velocity of steam through the various sections is not uniform. This may be due to variation of the mean temperature difference from section to section.

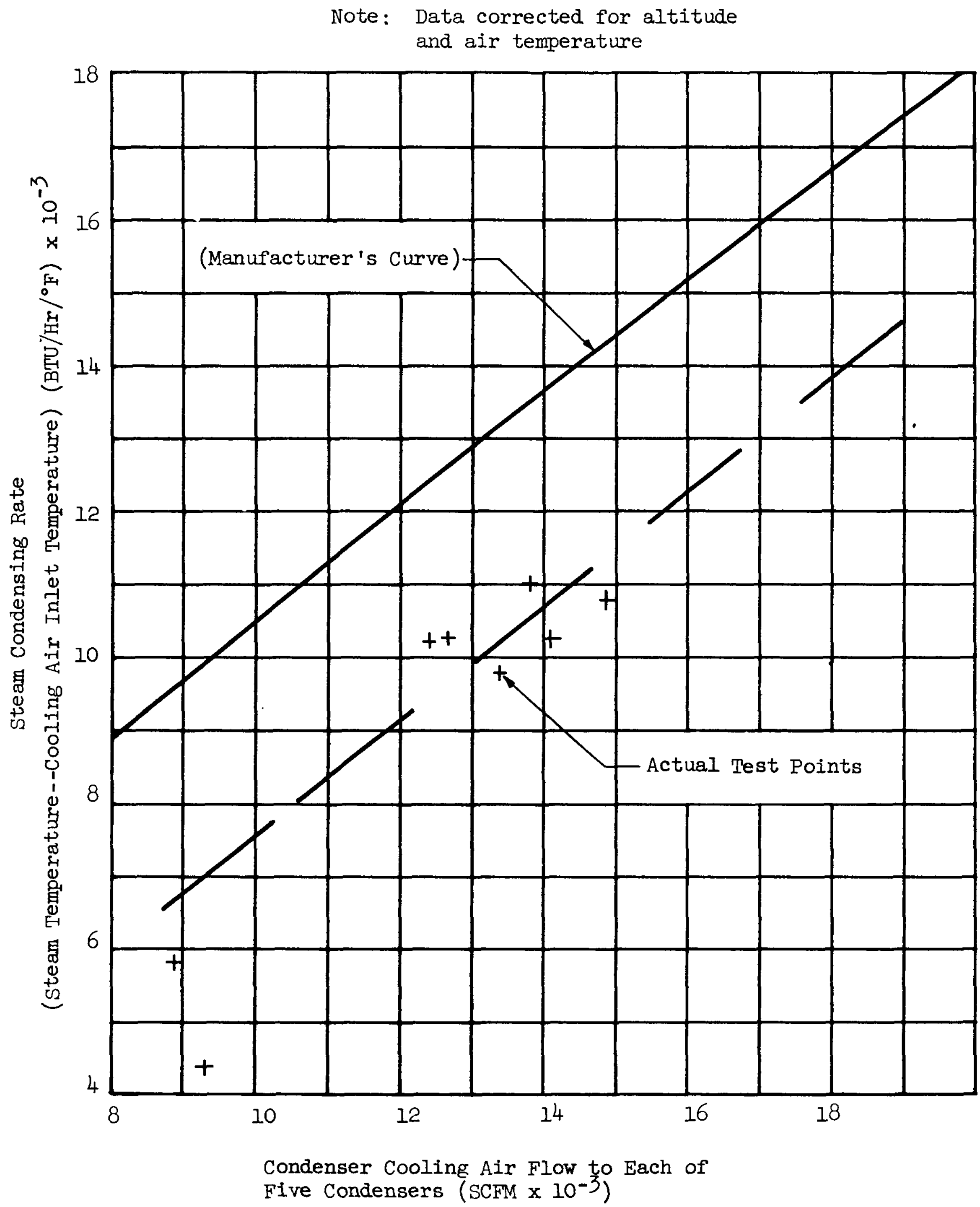


Fig. I-2. SL-1 Condenser Performance

Eilson Air Force Base, Alaska.- Data for this installation were obtained from Office of the Corps of Engineers, Seattle, Washington; Headquarters, U.S. Air Force, Washington, D.C.; and the Marley Company.

An air cooled steam condenser was installed outdoors at the Eilson Air Force Base generating plant. Design conditions covered a temperature range of - 60 to + 90° F. Due to freezeup trouble, and resulting tube ruptures, a shed was later erected around the condenser, and the slope of the condenser tubes was increased to facilitate drainage. For a brief period after these modifications were made, the unit operated satisfactorily. However, trouble subsequently developed because inlet air temperature control had not been provided, and unsatisfactory operation was experienced due to the wide range of air temperatures. The installation was changed to an indoor water cooled surface condenser. Personnel intimately associated with this project apparently have no basic prejudice against the use of steam air condensers for cold weather service, provided such equipment is carefully designed.

Power trains for the USSR.- Data for these installations were obtained from an article by R. A. Bowman, November 1945, ASME Transactions, and from the Westinghouse Electric Corporation.

During the early part of World War II, Westinghouse designed and built railroad car-mounted generating plants using air cooled steam condensers. Each unit had a net capacity of 5000 kw. Unfortunately, information on actual operating experience was never received from the Russians.

The condensers for each plant consisted of 16 sections, eight of which were mounted on single railroad flat cars. Each section had 42.5 sq ft of face area (air side). Induced draft fans pulled air across the tubes and discharged it through a Venturi-type stack.

Steam was exhausted from the turbine into an insulated steam duct located below the condenser tubes. The steam then flowed upward into the first eight rows of tubes. Any uncondensed steam or noncondensables passed through an inner and an outer vent chamber and thence downward through the last two rows of tubes which connected to an air ejector. These last two rows of tubes acted as an air cooler. Condensate from the air cooler section drained into the steam duct through a water trap. Condensate in the first eight rows of tubes flowed back to the steam duct. This counter flow of condensate and steam tended to heat the condensate and prevent freezeup. The exhaust from each of the first eight rows of tubes was orificed thus balancing the heat transfer through each section.

One major point which could be troublesome during extremely cold weather is that air, which is heavier than steam, must be forced up to the air ejector. It would seem more feasible to take the noncondensables from the bottom to take advantage of the density factor.

Table I-6 shows the shop tests performed on the USSR power train condensers.

TABLE I-6

USSR Power Train Condenser Test Results

Test Number	1	2	3
Ambient, °F	94.25	86.8	70.3
Barometer, in. Hg	28.95	28.93	28.86
Air temperature from condenser, °F	202.2	184.2	171.3
Steam temperature, °F	225.4	217.2	202.4
Steam condensed, lb/hr	79,860	78,640	78,140
Air pressure drop, in. of water	2.4	2.53	2.64
Heat load, Btu/hr x 10^6	75.33	74.36	75.83
Heat transfer rate, Btu/hr/ft^2/°F	13.4	11.7	12.0

3. Direct Contact Water-Cooled System

Figure I-3 shows the direct contact water cooled condenser system. The basic arrangement utilizes a low level jet or direct contact condenser. The mixture of condensate and condensing water is removed from the hotwell by the condensing water pump. A portion of the water is sent back to the cycle and the remainder is recirculated through an air blast water cooler. The cycle is completely sealed and all water is of feed water quality. This type system has been used to a limited extent in Europe.

The main disadvantage of this system, as compared to an air-steam system, is that a direct contact vertical condenser is required which would be approximately 4 ft in diameter and 11 ft high. Also, a water circulating pump is required, increasing the auxiliary power requirements. This system does have some advantages as compared to the

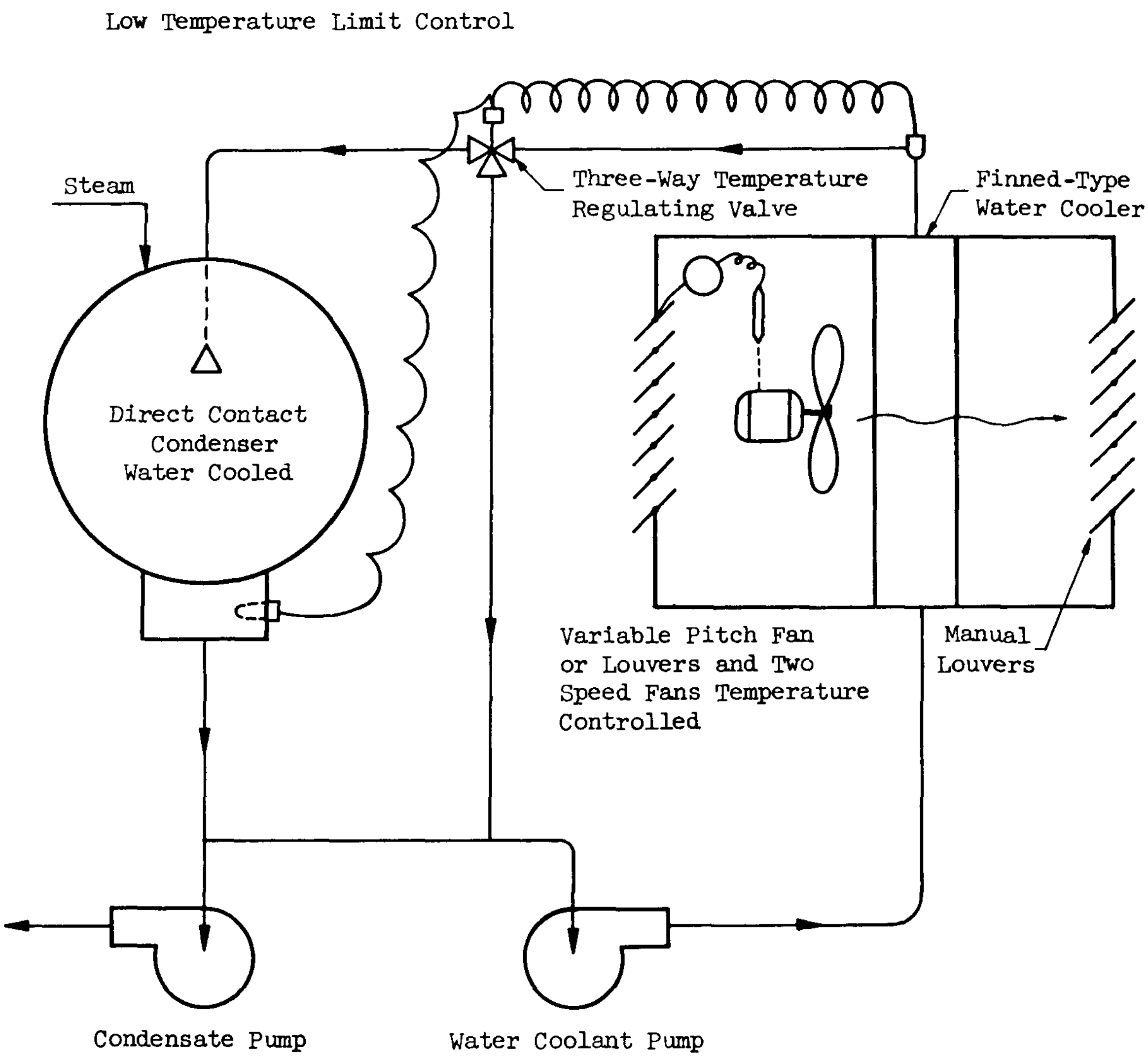


Fig. I-3. Direct Contact Water-Cooled Condenser

air-steam system in that elimination of noncondensables in the cooler reduces the air blast area approximately 20%, considerable experience with water cooled surface condensers is available, and the control system is somewhat simpler.

4. Gibbs and Hill Comments

The Gibbs and Hill study made the following comments concerning the design of air-steam and water-cooled condenser systems:

(1) All lines, duct work, and equipment must be insulated and weather-proofed with thicknesses calculated to permit gravity drainage before freezeup could occur in the event of plant shutdown.

(2) Steam lines not having a continuous flow, liquid lines, air lines, and gas lines must be steam traced with a double run of piping (one on each side of the line). If this type of freeze protection is not used, electric resistance heating cable should be used for freezeup protection.

(3) Heat transfer apparatus located outdoors should be mounted either vertically or at a sharp slope to permit quick gravity drainage.

(4) Pumps and other equipment should be located indoors whenever possible.

(5) The manufacturers of lubricating oils should be contacted to obtain a recommendation for oils suitable for use at these low temperatures.

(6) It is advisable that equipment and piping be installed in an enclosure having electric space heating for use during plant shutdown.

(7) Careful consideration must be given to low temperature metallurgy in the design, fabrication, and operation of these systems.

Based on presently available information, the USSR snow train experience, general past experience in condenser and heat exchange equipment, and the Gibbs and Hill studies, Westinghouse recommended that the direct air-steam type condenser system be used.

At present the design of a test model is being completed. The model will be one-half of a full size condenser section using horizontal aluminum finned tubes, variable pitch fans, louvers, and all auxiliary equipment required for the actual condenser system. Possible locations where environmental tests may be performed, such as Mt Washington, Aberdeen Proving Grounds and Eglin AFB, are being investigated.

Some items considered will be:

(1) Freezeup problems at various loads

(2) Startup and shutdown procedures

(3) Flow distribution of air

(4) Flow distribution of steam

(5) Controls

(6) Methods of removing noncondensables

(7) Housing or insulation requirements.

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

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

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

A. SUBTASK 2.1--FLEXIBLE ZERO POWER TEST

H. B. Rosenthal E. A. Scicchitano

The objective of the flexible zero power test is to provide experimental data to support the final core design of the PM-1 nuclear power plant. 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/or various design approaches.

During the quarter, a parametric zero power test program (PPM-1) was undertaken to obtain experimental data in support of the parametric core analysis work being done under Subtask 3.1. The experimental program was designed to provide information on:

(1) Reactivity effect of varying the radial location of Y-rods.

(2) Reactivity effect of varying slab-rod width.

(3) Reactivity effect of varying slab-rod thickness.

(4) Reactivity of step and cruciform rods.

(5) Temperature coefficient.

(6) Reactivity effects of lumped burnable poison.

(7) Distribution of fast flux in the core and pressure vessel.

Fuel elements fabricated under AEC Contract AT(30-3)-277 were utilized in the experiments. The experimental portion of the program was completed during this quarter.

The results of analytical calculations of rod worth, temperature coefficients, and the self-shielding factors for various configurations of lumped burnable poisons were compared with experimental results.

1. Reactivity Effect of Varying the Radial Location of Y-Rods (Core I)

Previous studies have shown that control rod worth calculations using the two dimensional code "PDQ" for single and multiple banks of Y-shaped rods were in good agreement ($\approx \pm 10\%$) with experimental results. In these studies, the radial location of the eccentric ring of rods was constant at ≈ 5 in. Since PM-1 control studies include rod location optimization, it is desirable to compare analytical and experimental results showing the effect of varying the radial location of eccentric rods.

Comparison of experimental and analytical results indicate that the present method of calculating rod worth utilizing PDQ is satisfactory for the preliminary design studies. However, for rod design studies, the conservative $\pm 10\%$ error will be assumed.

A summary of the characteristics of all cores is given in Table II-1.

Experimental study.- To determine the shadowing effects of control rods at various radial distances, the peripheral Y-rods were evaluated at four radial distances from the central axis of the core. To minimize the effort necessary for core changes while still maintaining sufficient shadowing effects, the positions of only three of the six peripheral Y-rods were varied.

The evaluation of the worth of a four-rod bank (center and three peripheral rods) was made in PPM-1, Core I, as shown in Fig. II-1. After appropriate cadmium adjustments, criticality was achieved with the central Y-rod partially inserted in the core.

The radial locations of Y-rods, 1, 3 and 6 were nominally 5, 5-1/2, 6, and 6-1/2 in. from the center of the core as shown in Fig. II-1. At each point, all three rods were at the same radial distance. For each change in rod location, rod guides and fuel tubes were relocated to their proper positions. The actual radial distances differ somewhat from the nominal, since each step was one fuel cell (0.485 in.) instead of 1/2 in. In addition, the radial arm of each rod was not on a true radius, but was displaced one tube. The actual distances are listed in Table II-2. Throughout these measurements the central Y-rod remained in its inserted position.

Measurements were made using subcritical techniques. These techniques may be generally described as follows:

TABLE II-1

Composition of Cores Used in Parametric Zero Power Test

Core I

Fuel tubes - 1694 at 11.87 gm = 20.108 kg U-235
No baffle
Thermal shield ID - 22.75 in.
Pressure vessel ID - 30.00 in.
Total reactivity (including reactivity poisoned out with cadmium) - 12.9% Δ K/K at 20° C
Operating core reactivity - 1.08% Δ K/K at 20° C

CE rods** - 3
Safety rods - 2
Y-rods - 4

Core II

Fuel tubes - 637 at 11.87 gm = 7.56 kg U-235
Equivalent radius* - 6.6 in. (based on 0.2037 sq in. unit cell)
No baffle
Thermal shield - 22.75 in.
Pressure vessel - 30.00 in.

CE rods - 2

Safety rods - 2
Total core reactivity - 0.63% Δ K/K at 20° C

Core III

Fuel tubes - 1782 at 11.87 gm = 21.16 kg U-235
Space occupied by test hole - 61 tubes
Equivalent radius - 11.0 in. (based on 0.2037 sq in. unit cell)
No baffle
Thermal shield ID - 22.75 in.
Operating core reactivity - 0.52% Δ K/K at 20° C

Pressure vessel ID - 30.00 in.
CE rods - 3

Safety rods - 2
Total reactivity (including reactivity poisoned out with Cd) 13.5% Δ K/K at 20° C

Core IV

Fuel tubes - 643 at 11.87 gm = 7.63 kg U-235
Equivalent radius - 6.6 in. (based on 0.2037 sq in. unit cell
No baffle
Thermal shield ID - 22.75 in.

Pressure vessel ID - 30.00 in.

CE rods - 3
Safety rods - 2
Total core reactivity - 0.79% Δ K/K at 20° C

Core V

Fuel tubes - 1723 at 11.87 gm = 20.45 kg U-235
Equivalent radius - 10.7 in. (based on 0.2037 sq in. unit cell)
No baffle

Thermal shield ID - 22.75 in.
Pressure vessel - 30.00 in.
Operating core reactivity - 0.77% Δ K/K at 20° C

CE rods - 3

Safety Rods - 2
Total reactivity (including reactivity poisoned out with Cd) 13.3% Δ K/K at 20° C

* The radius of a cylinder of equal circular area as the experimental core.

**Critical experiment control rods - used for fine control and measurement. Total worth of three rods does not exceed 2% Δ K/K

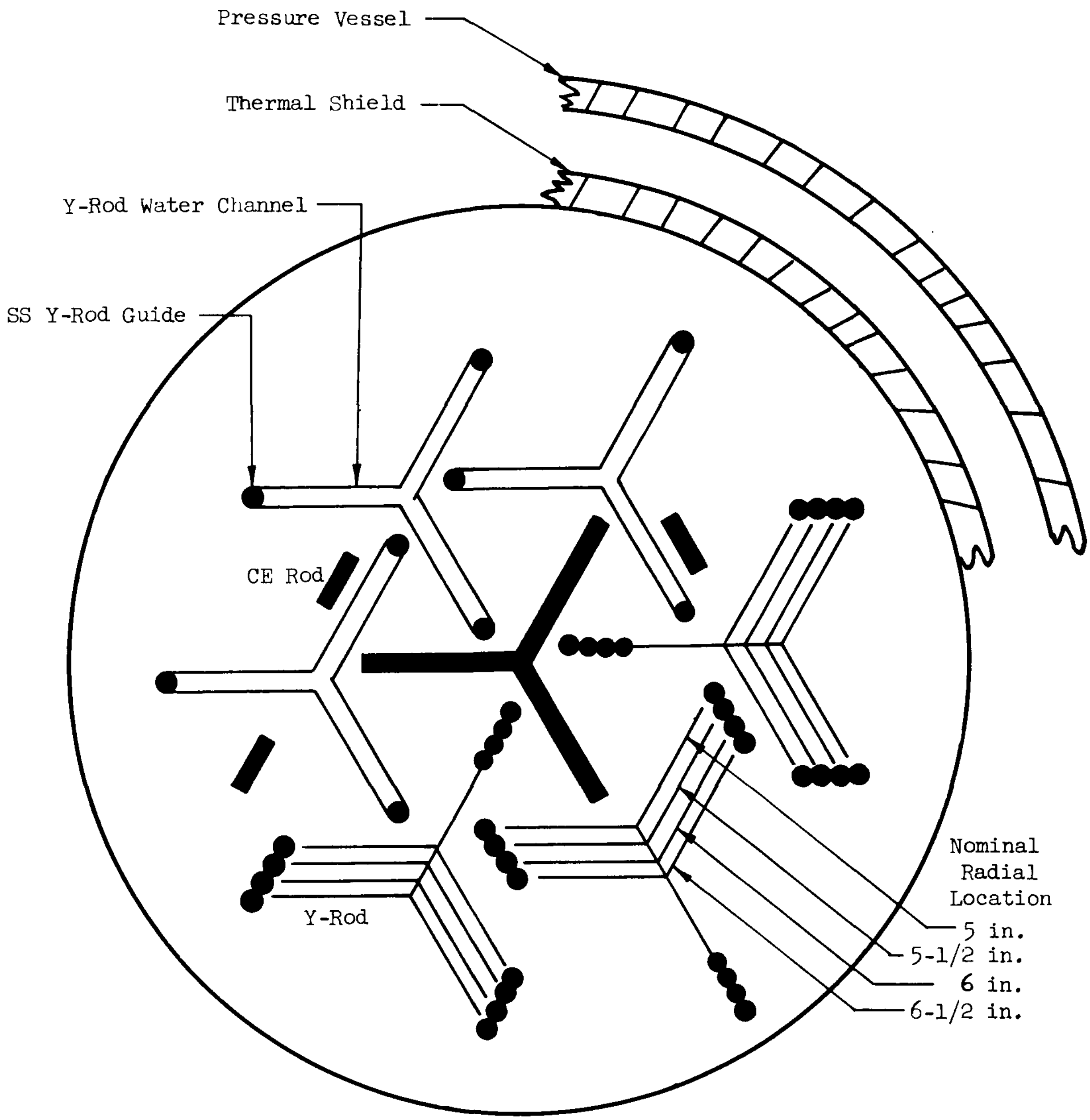


Fig. II-1. Core I Showing Y-Rods in Various Radial Locations

(1) A source term is obtained by bringing the reactor subcritical using a critical experiment (CE) rod of known worth.

(2) The source term is related to counting rates by Eq (1).

$$S_o = \frac{C\rho}{\rho - 1} \qquad (1)$$

S_o = source term

C = measured count rate

ρ = measured reactivity

(3) Measured reactivity is determined.

(4) The reactivity value of the rod bank can now be determined.

(A complete description of the subcritical technique is given in Appendix A of MND-MPR-1646.)

Table II-2 shows the reactivity worth of the Y-rods as a function of radial location. The effect of varying the radial location of the Y-rod water channels was determined from a comparison among the measurements with the Y-rods fully withdrawn. Rod worths show the values of the rod bank in the various locations. Total worths show the combination of rod worth and water channel worth. Movement of the three peripheral rods from 5.0 to 5.5 in. does not produce a significant effect since considerable shadowing is still taking place. Thus, movement to 5.5 in. can be accomplished with a negligible reduction in the worth of the four-rod bank. In moving the bank to 6.0 and 6.5 in., shadowing becomes less important but the worth of the bank decreases due to its location in a less important, i.e., less reactive, portion of the core.

TABLE II-2

Reactivity as a Function of Radial Location of Y-Rods

Radial Distance (in.) Nominal	True	Total Worth (% Δ K/K)	Water Channel Worth (%Δ K/K)	Four-Rod Bank Worth (% ΔK/K) Subcritically	Critically
5	5.110	-11.97	0	-11.97	--
5-1/2	5.594	-11.89	+0.02	-11.91	--
6	6.077	-10.16	+0.07	-10.23	--
6-1/2	6.561	- 9.64	+0.20	- 9.84	-10.48

To verify the absolute reactivity value obtained subcritically, sufficient cadmium was removed to bring the reactor critical with the four Y-rods (including the central rod) fully inserted at the 6-1/2-in. radial location. The worth of the four Y-rod band was determined to be 10.48% Δ K/K.

Analytical study.- An IBM-704 machine code, COFWAC (Calculation of Flux Weighted Absorber Constants) was written to simplify absorber constant computation, for use in three-group diffusion theory codes.

COFWAC is a Fortran II (FORmula TRANslation) coded, compatible program which calculates flux-weighted, few-group constants for those extraneous regions within a reactor core which may be approximated by a semi-infinite slab. This includes such core components as flow baffles and control rods. Functionally, COFWAC consists of a linkage of the codes which were used previously to obtain similar constants.

The output consists of the multigroup constants and fluxes for both the absorber and core materials, the calculated multigroup flux ratios or cell corrections, flux-weighted three-group constants for PDQ and CURE (XY, Rθ, and RZ) and appropriate input data. In addition, any one of the aforementioned sets of three-group constants may be punched on cards in the format required for direct input to CURE or PDQ.

Checkout of the parts of the code needed was completed. Checkout of the remaining portions of the code will be done later. (COFWAC was used in conjunction with the analysis of the Y-rod bank experiment as well as the other experiments given in the following sections.)

2. Reactivity Versus Rod Geometry

To fully evaluate the adequacy of the techniques utilized to calculate control rod worth, four experiments were designed to evaluate "simple" rod geometries in clean cores, i.e., cores in which major perturbations do not exist. In utilizing a clean core, the measurements which are taken, as well as the calculations and analyses which follow, are greatly simplified.

Experimental study.-

Reactivity versus slab rod width (Core II)--For this study the core shown in Fig. II-2 was assembled. All cadmium was removed from the fuel elements of Core I, all prototype Y-rods were removed and the core rebuilt to the reduced size shown. The safety rods were completely withdrawn during the experiments, with fine control and experimental reactivity provided by the CE control rods. All measurements were made with the slab rods inserted in the same test slot using subcritical techniques as explained previously.

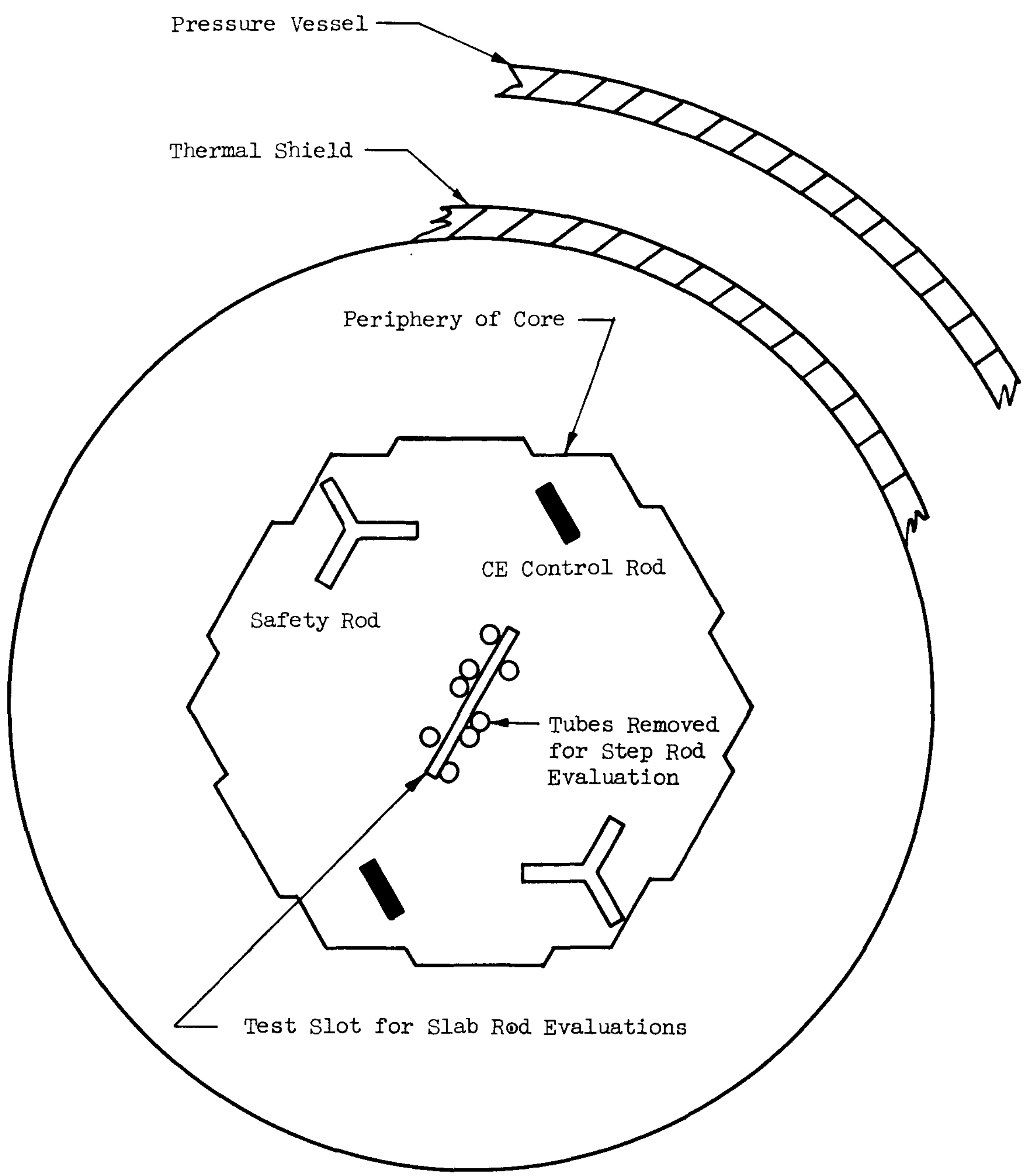


Fig. II-2. Core II Used for Slab and Step Rod Configuration

The slab rods were made of a mixture of B_4C and aluminum, encased in a 1/4 in. wide stainless steel picture frame, clad with 0.025 in. stainless steel. All slabs had a nominal thickness of 0.200 in. The B_4C was diluted with aluminum to a nominal natural boron density of 0.75 gm/cc.

The reactivity values for the various rods are shown graphically in Fig. II-3. As a check, the worth of the 3 in. x 0.200 in. slab was also determined critically.

The data indicate the expected increase in rod worth as the width of the rod increases. Departure from the linear was anticipated since the rod occupies space in the less reactive portions of the core as the rod width is increased.

Reactivity versus slab rod thickness (Core II)--In evaluating the reactivity effect of slab rods with various thicknesses, PPM-1 Core II (Fig. II-2) was again utilized. The slab rods were inserted in the same test slot shown in Fig. II-2. Subcritical techniques were used.

All rods had a nominal width of 3 in. Average boron density in the rods was 0.76 gm/cc.

The experimental results are shown in Fig. II-4. It is apparent that complete "blackness" in the rods has not been reached in terms of rod thickness for a fixed concentration of boron.

Step rod evaluation--The analytical technique utilized in evaluating the worth of Y-rods is done in X-Y geometry. Therefore, it is necessary to approximate the arms of the rod with a step model. The adequacy of this step approximation requires examination from an experimental point of view.

A step rod corresponding to the approximation of a Y-rod arm 3 in. wide and 0.200 in. thick was fabricated and an experimental comparison of the two rods was made. Sketches of each rod are shown in Fig. II-5. To accommodate the step rod, eight additional fuel tubes were removed from the test slot shown in Fig. II-2. Both the step rod and the slab rod were evaluated in this larger slot.

The reactivity values obtained are presented in Table II-3.

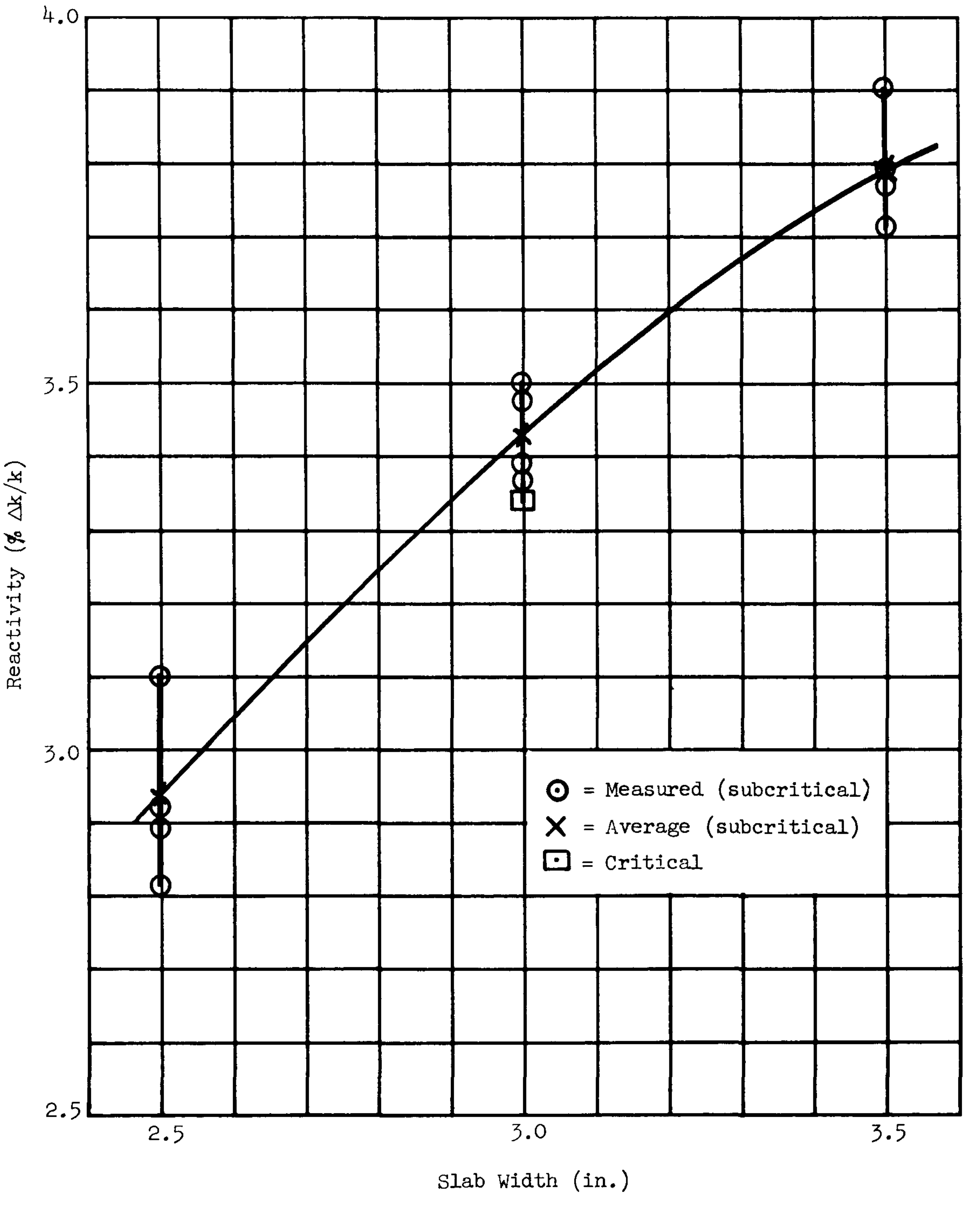


Fig. II-3. Reactivity as a Function of Slab Rod Width --Core II

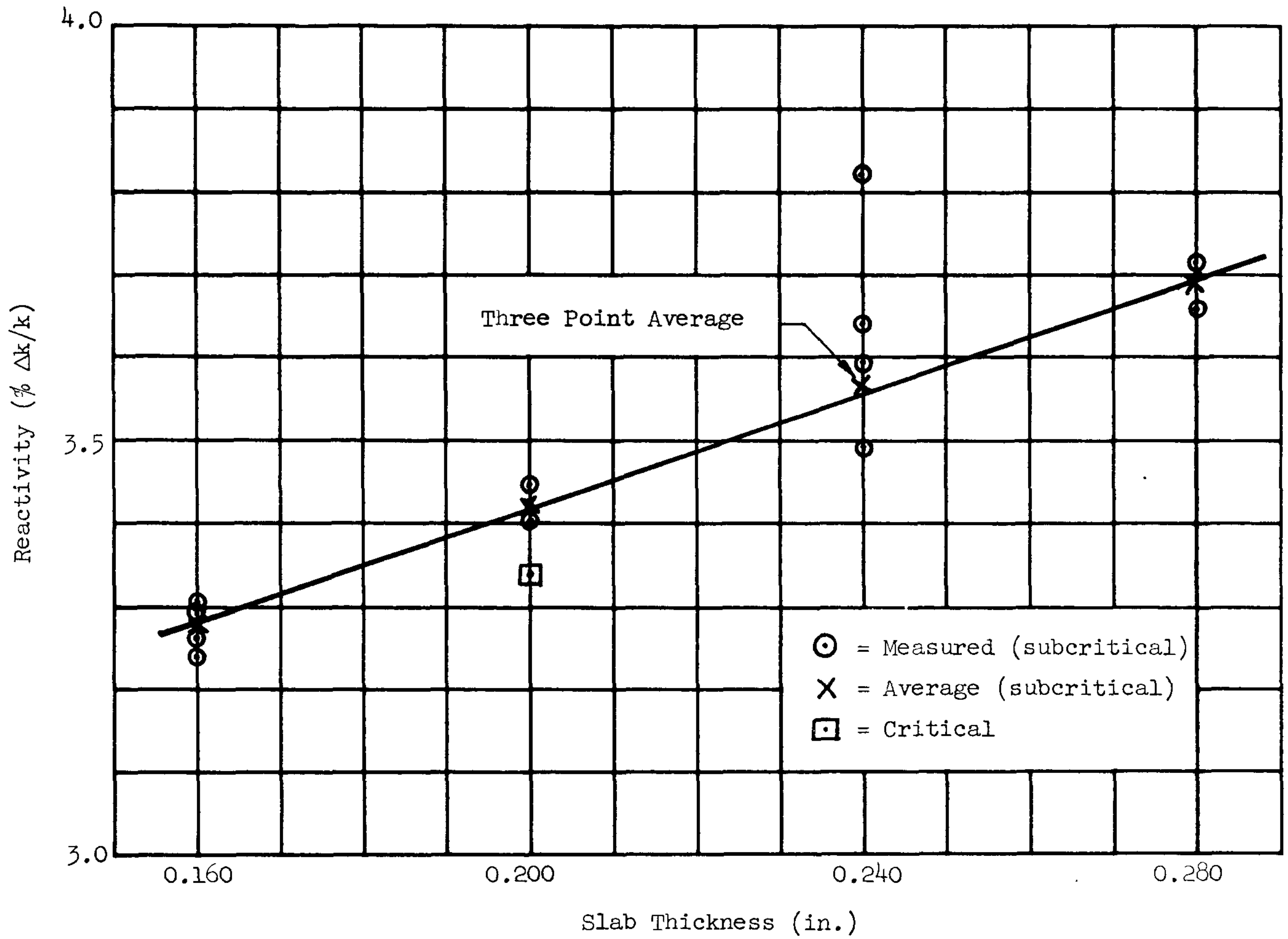


Fig. II-4. Reactivity as a Function of Slab Rod Thickness--Core II

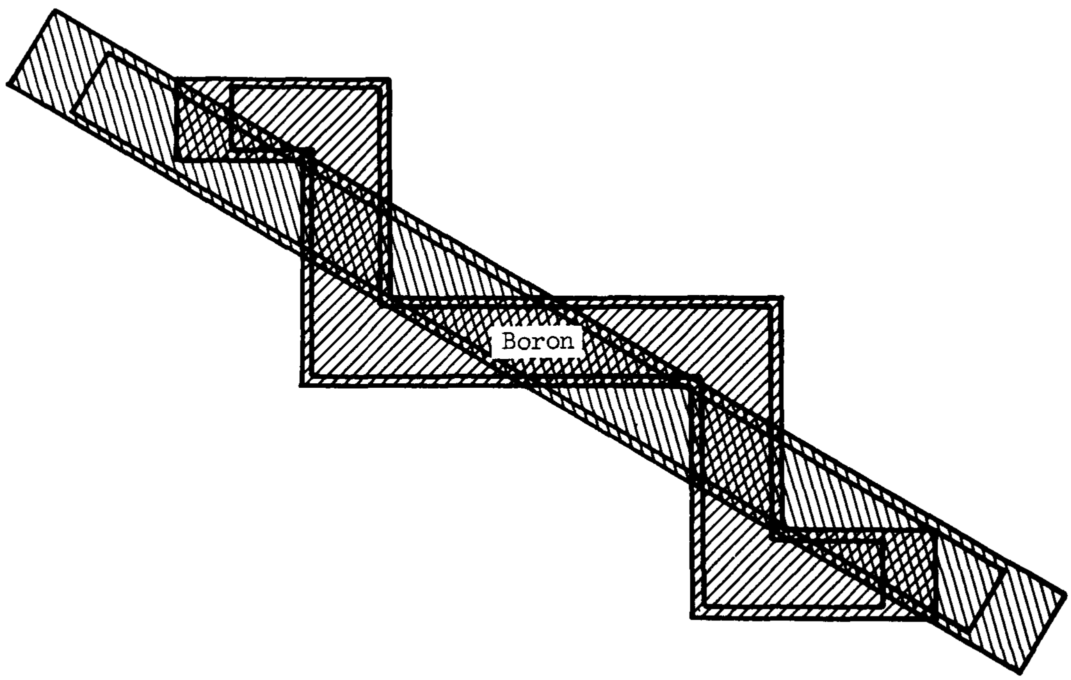


Fig. II-5. Step Rod with Slab Rod Superimposed

TABLE II-3

Step Rod Evaluation

Rod	Slab Thickness (in.)	Boron Density (gm/cc)	Boron Weight (gm)	Reactivity (% Δ K/K) Subcritical	Critical
3 x 0.200	0.204 ± 0.0015	0.7468	175.0	3.42	3.34
Step	0.207 ± 0.0017	0.7476	229.5	3.66	--

Comparing the two reactivity values shows that the reactivity worth of the step rod is 7% greater than the 3 x 0.200 slab. Since boron density for the step and slab rods are the same, the measured reactivity difference is due to the difference in geometry and control area.

Reactivity versus cruciform rod width (Core III)--In the event that parametric and preliminary design dictate placement of fuel elements in a square rather than a triangular pitch, the Y-rods would be replaced with cruciform rods. Therefore, an experimental evaluation of the effect of cruciform rod arm width was undertaken.

To produce data with minimum influence from edge effects, a core was built up with a uniform distribution of cadmium in the form of a cadmium-tin mixture rolled into tubular form and inserted inside the fuel tubes. This core had a test hole in the center in which the cruciform rods were studied. The cruciform arms were made of a mixture of B_4C and aluminum, encased in a 1/4 in. wide stainless steel picture frame, clad with 0.025 in. stainless steel. All arms had a nominal thickness of 0.200 in. The B_4C was diluted with aluminum to a nominal natural boron density of 0.75 gm/cc.

Table II-4 presents the boron dimensions and concentrations for the various cruciform arms. The reactivity values for the various cruciform rods are presented in Table II-5. Worth determinations were made subcritically with a critical check. For convenience, measurements were made in two existing instrument channels.

TABLE II-4

Summary on Cruciform Rods

Boron Width (in.)	Arm	Thickness (in.)	Boron Density (gm/cc)	Boron (gm)
2-1/2	1	0.210 ± 0.0043	0.7475	138.3
	2	0.205 ± 0.0030	0.7484	140.0
	3	0.204 ± 0.0015	0.7504	138.8
	4	0.210 ± 0.0023	0.7470	139.7
3	1	0.204 ± 0.0024	0.7448	172.8
	2	0.200 ± 0.0047	0.7438	169.0
	3	0.218 ± 0.0057	0.7419	176.6
	4	0.196 ± 0.0047	0.7484	170.6
3-1/2	1	0.204 ± 0.0040	0.7432	189.5
	2	0.200 ± 0.0064	0.7478	192.9
	3	0.205 ± 0.0036	0.7491	198.5
	4	0.211 ± 0.0021	0.7427	202.0

TABLE II-5

Reactivity Worth of Cruciform Rods

Arm Width (in.)	Reactivity (% Δ K/K) Subcritical Channel 1	Subcritical Channel 2	Average	Critical
2-1/2	4.94	5.7	5.35	5.34
3	6.03	7.02	6.53	--
3-1/2	6.84	7.97	7.40	--

The effect on reactivity of cruciform arm width is essentially the same as that previously noted with slab rods. The anticipated increase in reactivity with arm width exhibits a tendency toward saturation at large values of arm width.

Analytical study.- Analytical studies which represent additional fundamental control rod studies for support of preliminary design studies are in progress. Compilation of all of the input data for the "PDQ" analyses were completed. Three-group constants for the absorber regions were calculated using the absorber constant program, COFWAC, described.

3. Temperature Coefficient Measurements

A question raised by previous temperature coefficient measurements was whether air bubbles, generated through water degassing during heating, created voids--thereby resulting in incorrect reactivity values. A primary objective of this experiment was investigation and elimination of the effect of this variable. The following approach was utilized:

(1) Determination of reactivity values while heating the water, maintenance of the water at 70° C for at least 30 min, then determining the reactivity values while cooling the water. A difference in the determinations would indicate the presence of voids.

(2) If the results of the first approach indicate the absence of bubbles, adding a wetting agent to the water and determining reactivity values while cooling the water.

A comparison of experimental and analytically determined temperature coefficients was a secondary objective.

Experimental study.-

Temperature effect--Temperature coefficient studies were performed on PPM-1, Core IV, which was essentially the same as Core II except that a CE rod was inserted in the center test hole with the gaps on either side filled with fuel elements. Measurements were made by establishing criticality, increasing the temperature of the water approximately 10° C with steam heaters, maintaining the temperature until equilibrium was reached, and re-establishing criticality. The reactivity effect of temperature by cooling was determined by reversing the process.

During the initial run, measurements were made as the temperature was increased from 16.66 to 63.44° C. In the final run the temperature was increased from 45.0 to 69.8° C.

To maintain the temperature at 70° C, it was necessary to apply steam so that the temperature was cycled over a range of approximately ± 1° C. The temperature was maintained at about 70° C for 38 min during the final run.

The temperature was decreased by running cold water through the steam lines. Figure II-6 shows a plot of the data obtained. It should be noted that all reactivity values have been normalized to 70° C.

The temperature coefficients of reactivity for PPM-1, Core IV, obtained by differentiating the curves in Fig. II-6, are summarized in Table II-6. The total reactivity difference between 20 and 70° C was -0.325% Δ K/K by heating and -0.316% Δ K/K by cooling.

TABLE II-6

Temperature Coefficient of Reactivity

Temperature (°C)	Curve	Temperature Coefficient (% Δ K/K/°C)
20	Heating	-0.002
	Cooling	-0.002
70	Heating	-0.009
	Cooling	-0.008

Void effect--The second question to be resolved by this experiment was whether the reactivity versus temperature curves obtained were valid, or whether the voids created by degassing of the water invalidated the results.

An attempt to see whether air bubbles remained on the surface of the fuel tubes during heating, using a telephoto lens and a TV camera, was unsuccessful.

A second approach, direct observation of bubble size and distribution over a portion of the surfaces of several tubes under varying conditions of temperature, was next utilized. Based upon evaluation of data developed using this approach, it was concluded that:

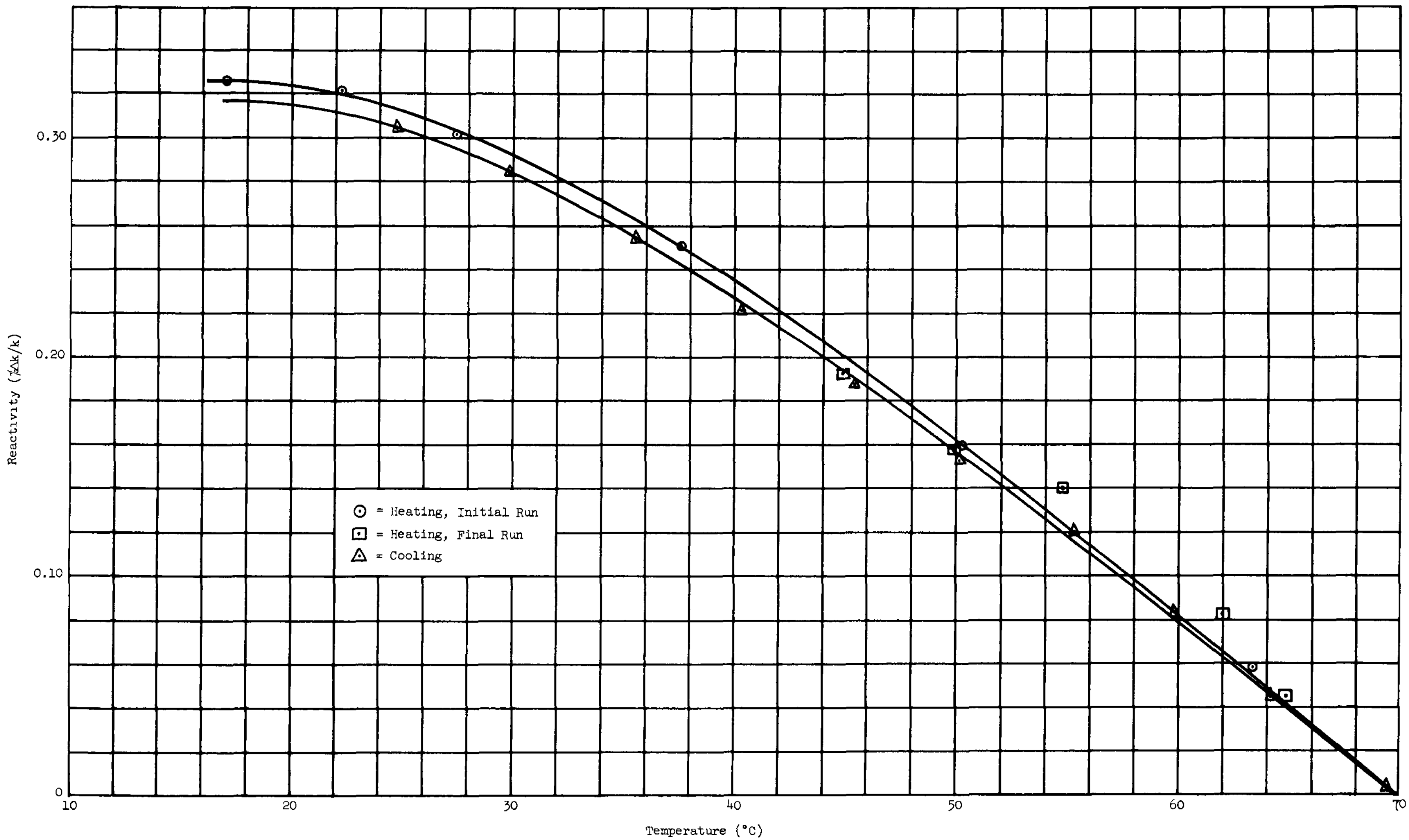


Fig. II-6. Reactivity as a Function of Temperature--Core IV

(1) The reactivity of the voids is small, not exceeding 3% of the measured value of the temperature effect.

(2) The maximum amount of gas void retained on the fuel elements occurs at approximately 53° C. This value is equivalent, based on the assumptions used, to a uniform film thickness of 0.0008 in. on both the inside and outside of tubes.

(3) A heat-up rate of 0.2° C/min gives less gas retention than a rate of 0.5° C/min (0.8×10^{-3} compared to 1.95×10^{-3} volume/sq in. surface maximum).

(4) The rate of resaturation of water with air using a limited free surface area (0.044 sq ft/gal water) is very low.

(5) Adding wetting agents and thus reducing surface tension decreases the amount of gas retention. Twelve parts per million of sodium lauryl sulfate had little effect, but 50 ppm essentially eliminated all gas.

(6) An increase in wetting agent concentration decreases the water temperature at which maximum bubble formation occurs as well as decreasing the amount of gas retention.

Analytical study.- The reactivity of the core was calculated for different temperatures from 68 to 180° F (20 to 82.2° C). Both nuclear and density temperature coefficients 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 between 68 and 180° F.

The effect of voids on the reactivity variation with temperature was also investigated. Analytical results are shown in Fig. II-7.

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

Correlation.- The relative shapes of the analytical and experimental reactivity versus temperature curves show good agreement. The calculated decrease in reactivity for a temperature increase from 20 to 70° C is 0.0046. The experimental decrease for this temperature increase was 0.0033. The difference of 0.0013 between the experimental and analytical reactivity changes is well within the total of experimental and calculational error.

Although the presence of voids results in a decrease in core reactivity, the temperature coefficient remains approximately constant for a constant void. Additional void studies will be performed in evaluating local boiling conditions in the preliminary design analysis.

4. Evaluation of Lumped Burnable Poisons (Core V)

Lumping of the burnable poison in the form of rods or tubes in the PM-1 core presents a method for controlling the reactivity-lifetime excursion (described qualitatively in the reactor studies of Subtask 3.1). This degree of control is dependent upon the time variance of the self-shielding factor for the lump as well as the effect of the other time dependent variables, i.e., fuel burnup, fuel thermal disadvantage factor, poison buildup, etc. The objectives of this study are to determine experimentally the effect of lumping the poison material at zero time, i.e., in a cold and clean reactor, and to compare the experimental data with analytically determined reactivity effects of lumping.

Experimental study.- The reactivity worth of lumped burnable poison was evaluated by obtaining radial worths of rods and tubes containing various concentrations of boron. Measurements were made in PPM-1 Core V along the radial traverse shown in Fig. II-8. In the case of the three outer points on the radial traverse, the reactivity value of some of the samples was so low that it was necessary to measure the value of two corresponding samples simultaneously. In these cases, samples were located in positions on symmetrical radii as indicated in Fig. II-8.

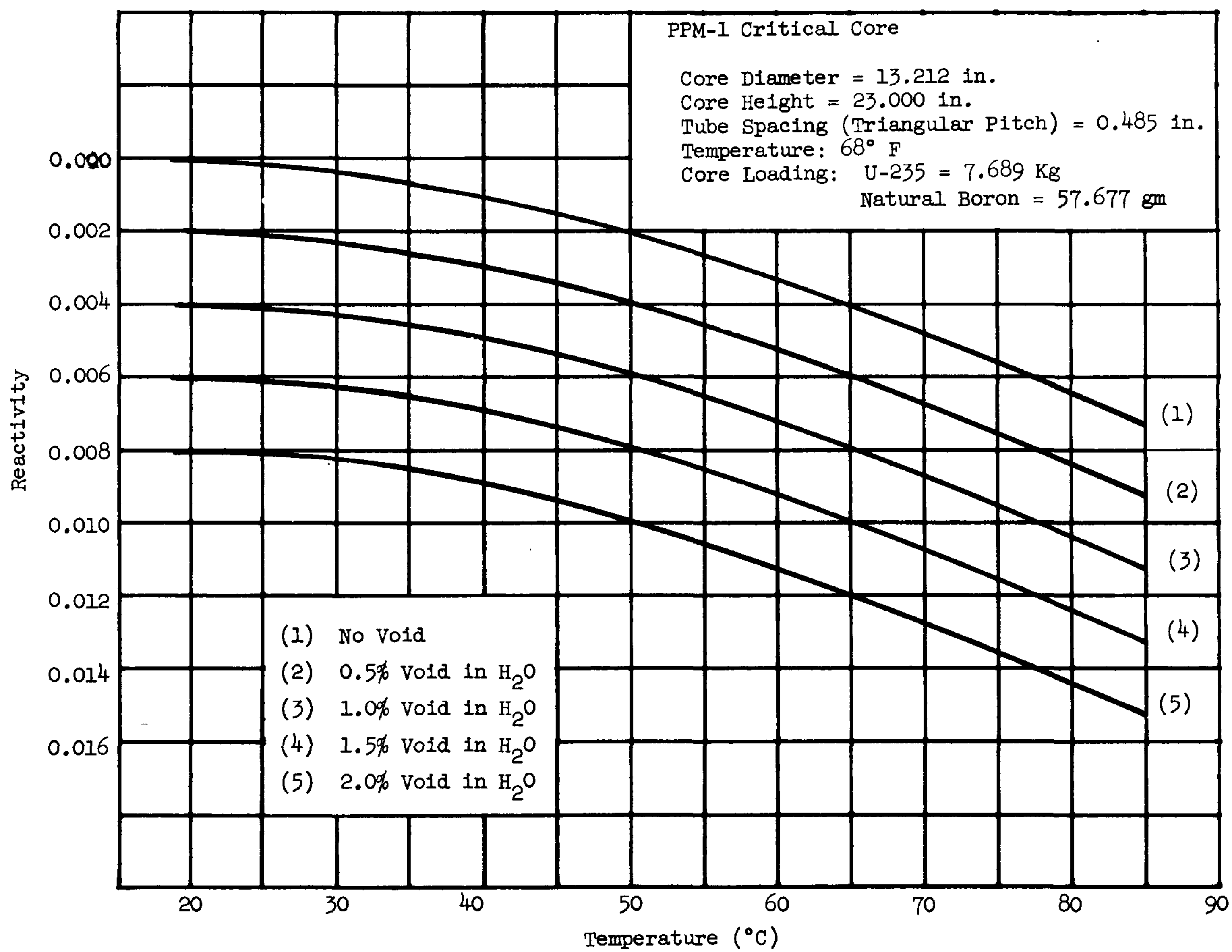
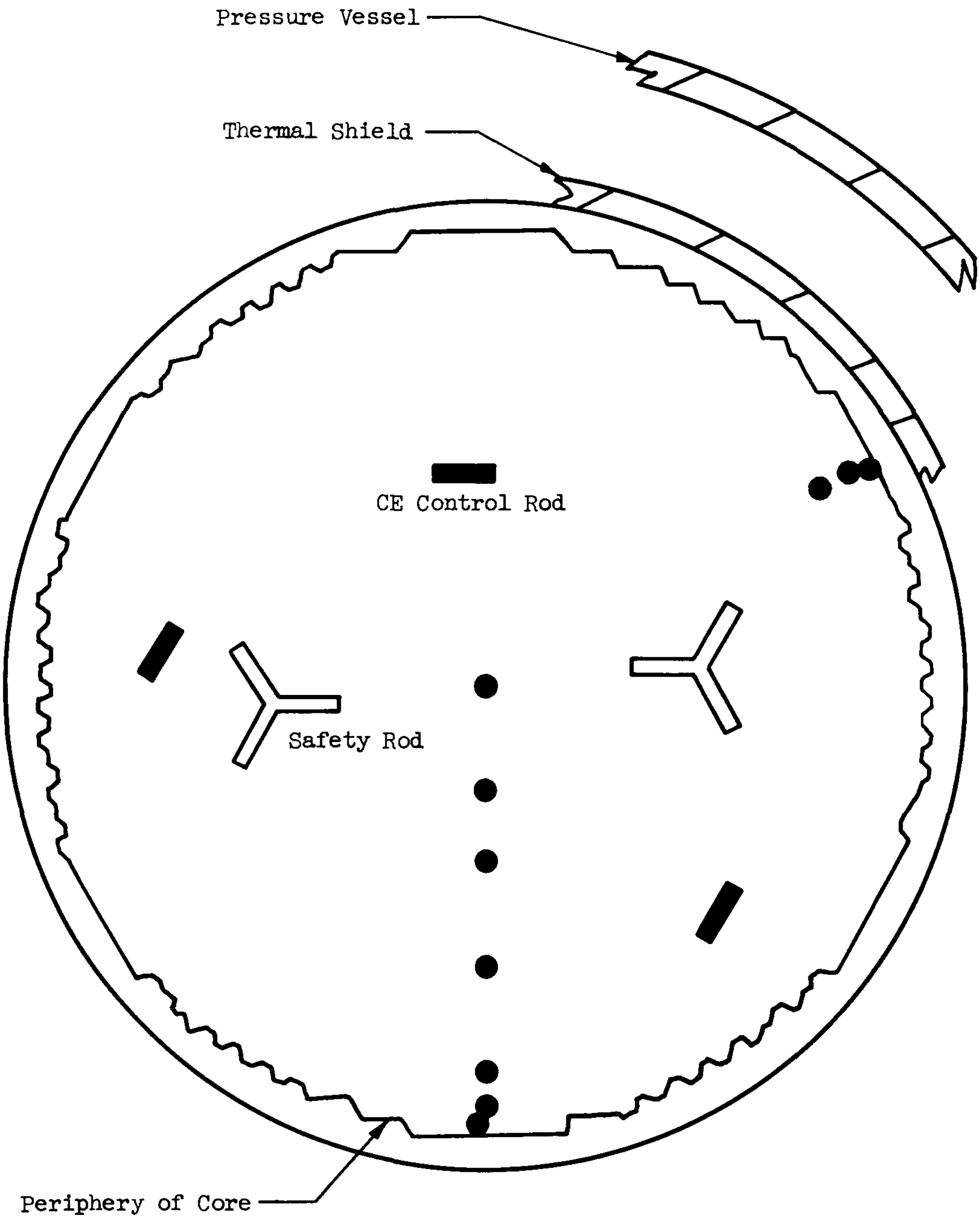


Fig. II-7. Variation of Reactivity with Temperature (Calculated)



● = Additional Points Used When Reactivity was too Low for Good Measurement with One Rod

● = Location of Points Along Radial Traverse for Lumped Poison Evaluation

Fig. II-8. Core V Used for Lumped Poison Evaluation

Two types of samples were studied: rods of a B_4C and aluminum mixture, and tubes of the same mixture--all clad in stainless steel. The nominal boron concentrations of the rods were 0.4, 0.1, and 0.05 gm/cc. In the tube, the boron concentration was 0.1 gm/cc.

Table II-7 presents the boron concentrations for the various samples studied.

TABLE II-7

Boron Concentrations in Lumped Burnable Poison Specimens

Sample No.	Boron Density (gm/cc)	Boron Weight (gm)
Rod 1	0.4033	19.12
2	0.1025	4.86
3	0.1032	4.89
4	0.0508	2.42
5	0.0508	2.41
Tube 2	0.1019	1.46
4	0.1024	1.47

In the instances where only one sample was evaluated, Rods 1, 2, 4, and Tube 4 were used. As the reactivity values decreased, the additional samples, Rods 3, 5 and Tube 2, were located along the alternate radius in conjunction with their corresponding samples.

The reactivity values per unit samples are presented in Table II-8.

TABLE II-8

Radial Reactivity Worths of Lumped Burnable Poisons (% Δ K/K)

Radial Location (cm)	0	6.40	10.67	17.07	23.47	25.61	26.67
Sample (nominal gm/cc)							
0.4 rod	0.427	0.392	0.304	0.191	0.076	0.059	0.043
0.1 rod	0.297	0.271	0.212	0.136	0.053*	0.041*	0.036*
0.05 rod	0.217	0.198	0.154	0.111	0.041*	0.035*	0.033*
0.1 tube	0.174	0.169	0.133	0.087	0.031*	0.031*	0.024*

*These values are based on a measurement using two samples.

Figure II-9 presents the same data in terms of reactivity per gram of boron. In order to check the validity of some of the reactivity worths, a few selected points were rerun. The variation in reactivity worths did not exceed 0.004% Δ K/K.

The curves in Fig. II-9 do not show an increase in reactivity near the periphery of the core. This is probably due to the presence of the thermal shield and the fact that the water channel between the outer fuel tube and the thermal shield is small. The equivalent radius of the core is 27.08 cm. The actual radial distance of the outer edge of the outer tube is 27.15 cm. Thus the water channel between this tube and the thermal shield is only 1.74 cm.

The data are presented as reactivity per gram of boron as a function of boron concentration in Fig. II-10. These curves are based on the data for poison lumped in rods. Examination of Fig. II-9 shows the difference in the self-shielding factor for equal concentrations of boron in rods and tubes. It is evident that the reactivity controlled per gram in the tube is greater than in the rod because of the ineffectiveness of the boron close to the center of the rod.

Analytical study.- Two of the important nuclear properties of the lumped poisons required for analysis are the reactivity worth and the time dependent self-shielding factor, g(t), of the lumped poison. Since g(t) (where g = $\bar{\phi}$ poison/$\bar{\phi}$ unperturbed) is essentially equal to g(N), i.e., a function of poison concentration, a knowledge of reactivity worth allows this function to be calculated using perturbation theory.

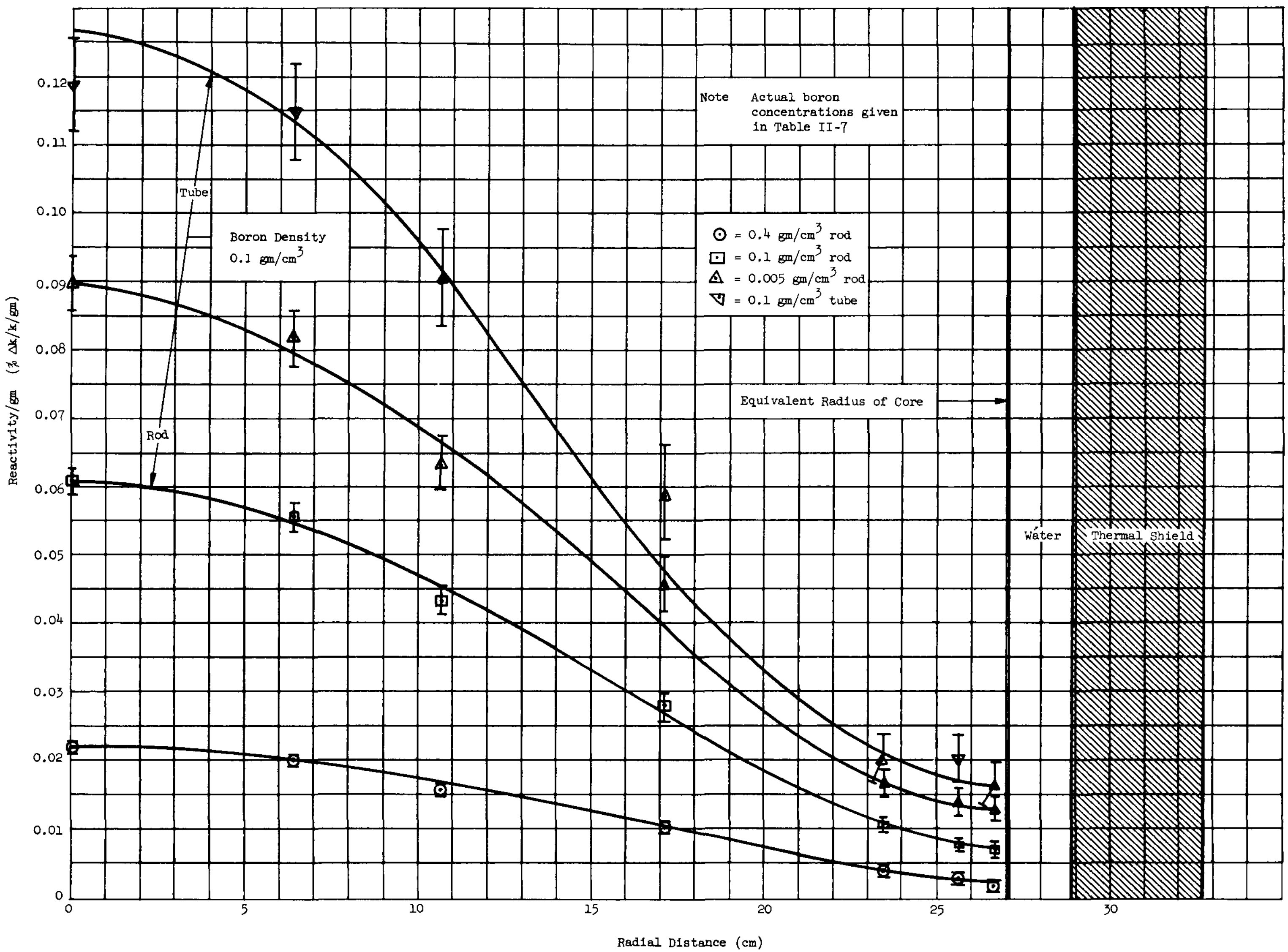


Fig. II-9. Reactivity per Gram of Lumped Poison as a Function of Radial Location--PPM-1, Core V

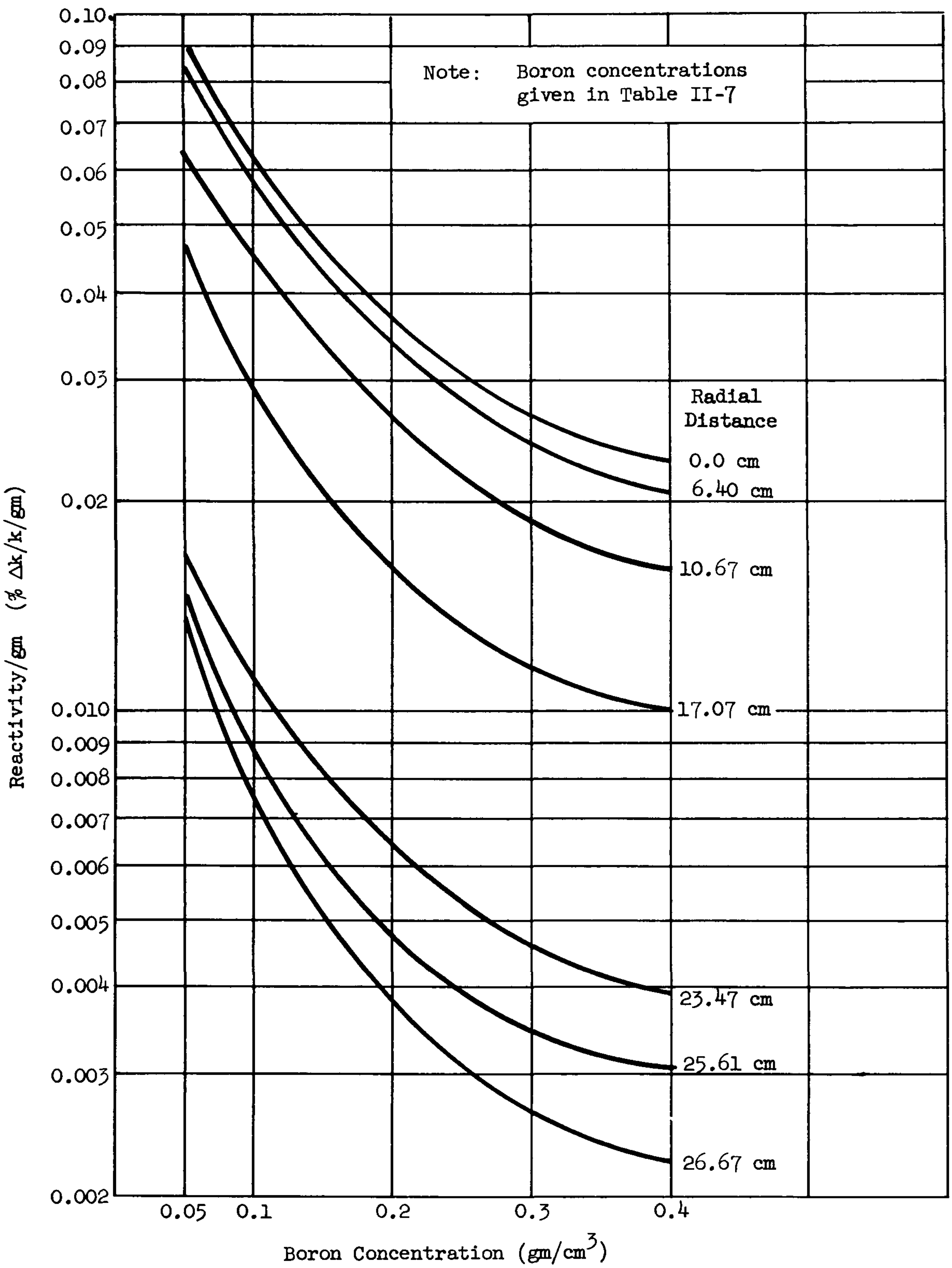


Fig. II-10. Reactivity per Gram of Lumped Poison Rods as a Function of Boron Concentration, Core V

In order to assure the proper choice of poison concentrations for experiment, the self-shielding factor as a function of poison concentration was calculated. This was done using diffusion theory which, although not precisely correct, would give the relative shape of the curve. From these results, shown in Fig. II-11, poison concentrations of 0.05, 0.1 and 0.4 gm/cc were selected for experimentation. A direct comparison of Figs. II-10 and II-11 can be made in which the shapes of the curves may be compared. Experimental points normalized to the self-shielding factor corresponding to a boron density of 0.05 gm/cc are shown in Fig. II-11 for a radial location of 23.47 cm. To correlate the experimental and analytical results, it was necessary to assume that the self-shielding factor (Fig. II-11) is directly proportional to the reactivity per gram controlled by the lumped poison (Fig. II-10) irrespective of the concentration of poison material. To a first approximation, this assumption is true, but becomes less applicable as the concentration of poison increases. The agreement between experiment and analysis is, however, quite good.

5. Fast Flux Measurements (Core V)

To implement shielding calculations of fast neutron flux outside of the core and in the pressure vessel, the fast neutron flux was measured in Core V, adjacent to the thermal shield, and adjacent to the pressure vessel.

The flux measurements were performed using the (n,p) threshold reaction in phosphorus which gives an activation product of Si-31 with a 2.62-hr half-life. The effective threshold for this reaction is about 2.5 mev. Foils were made by moistening reagent-grade dibasic ammonium phosphate powder and tamping it into place in 2-in. diameter plastic petri dishes to a depth of about 1/8 in. This, when allowed to dry, formed a compact stable mass. No covering over the phosphorus compound was employed except during exposure in the reactor, at which time they were enclosed in waterproof plastic bags.

The distribution of fast flux was measured along a radial line through the core at its midplane, from the core center outwards through the thermal shield, and along another radius at the same height adjacent to the pressure vessel.

The results of these measurements are plotted as a single radial distribution of Fig. II-12, although the three sets of points represent different measurements in three separate reactor runs. The curve was drawn through the data points so as to give the best visual fit with the slopes of the curve within the pressure vessel and within the thermal shield drawn so as to be equal. In cases where one of the three data points in a set was far different from the other two, it was discarded.

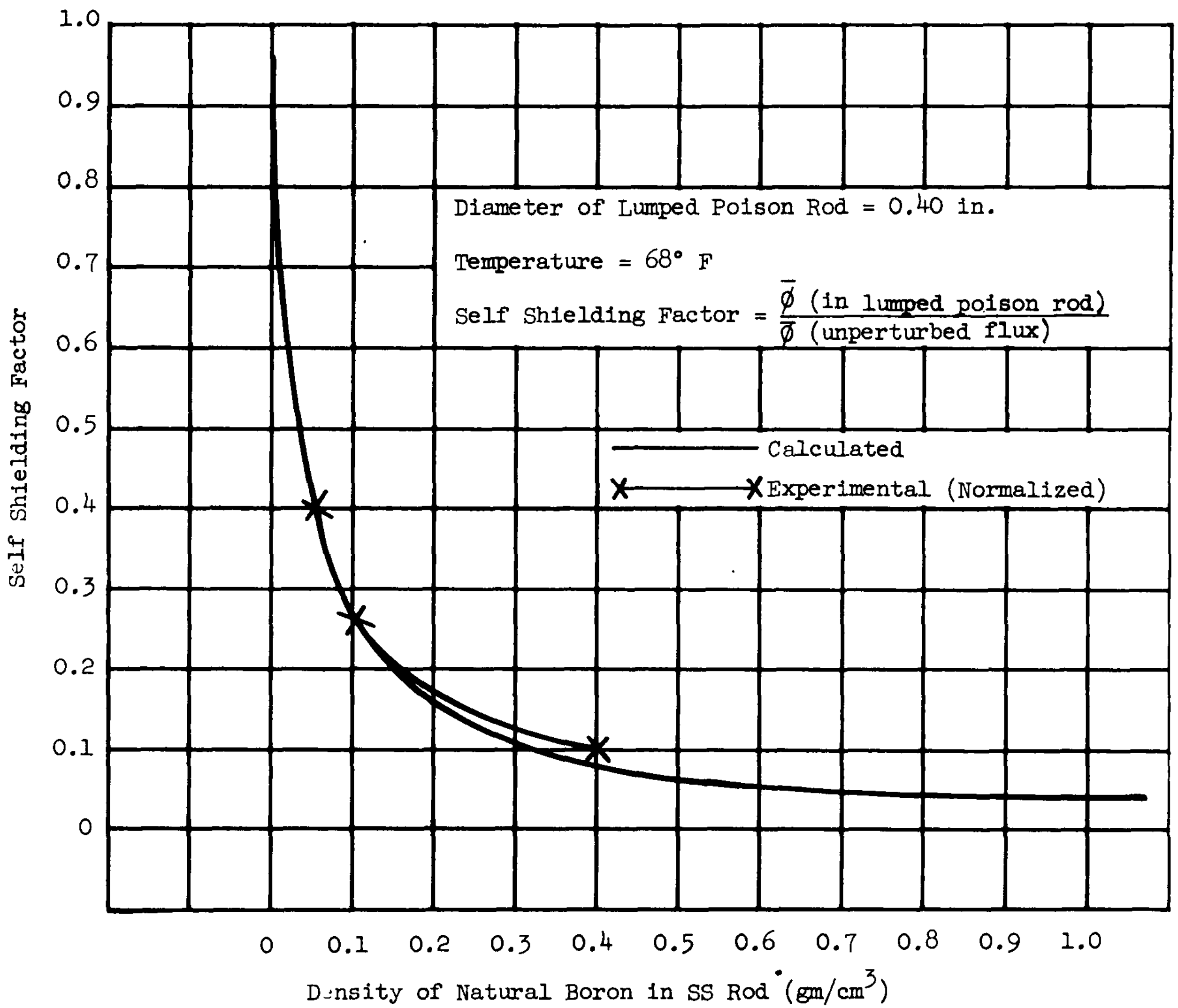


Fig. II-11. Self-Shielding Factor vs Density of Natural Boron in Lumped Poison Rod

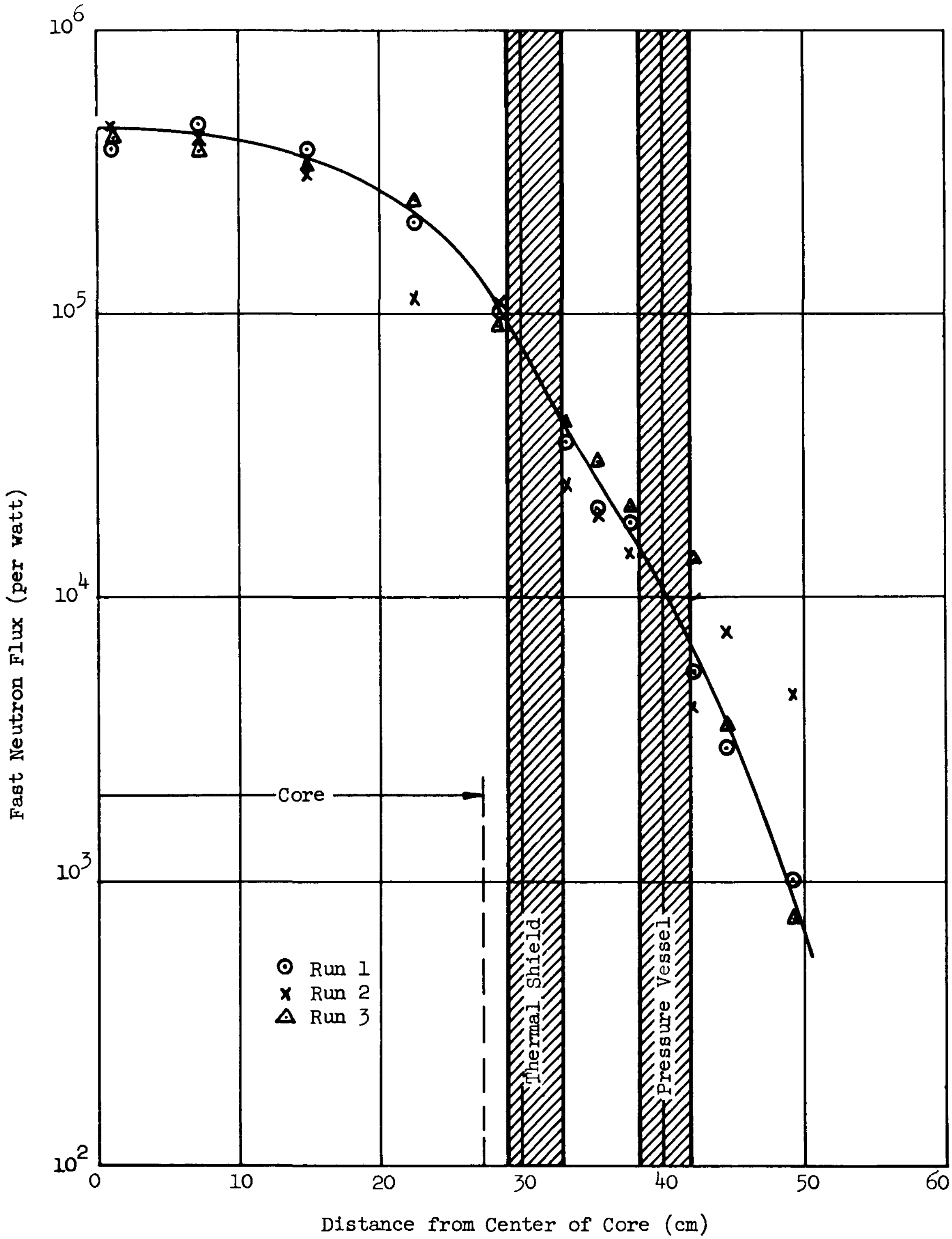


Fig. II-12. Fast Flux Distribution 29.2 cm from Bottom of Active Core

It is estimated that the absolute value of the scale of the neutron flux is accurate to ± 20%. Sources of error arise mainly in the statistics of the measurement of counting rates. It is estimated that an individual data point outside the pressure vessel has an uncertainty of ± 30%. Inside the core the uncertainty reduces to about ± 10%.

Analytical evaluation of the data is not complete.

6. Excursion Analysis

The excursion analysis for the PM-1 flexible zero power experiment was started. Methods of solution to the kinetic equations were investigated. The most promising were those presented in UN 58/629 "Some Topics in Reactor Kinetics" by E. Richdard Cohen. The theory of this method was reviewed and the extension of this method, as mentioned by Cohen, to include the precursor equations by metods similar to Runge-Kutta procedures, was investigated. An IBM-704 machine program for solution of the equations has been coded in Fortran. Checkout of these equations with available data fr ı ERDL excursion analysis data will be done next.

7. Summary

The parametric zero power test program was initiated and the experimental portion completed during this quarter.

The reactivity of three eccentric Y-rods with a central Y-rod in place varied from 11.97% Δ K/K at 5 in. to 9.84 Δ K/K at 6.5 in.

The reactivity worths of slab rods 0.200 in. thick varied from 2.93% Δ K/K for a 2.5-in. wide slab to 3.79% Δ K/K for a 3.5-in. wide slab.

Slab rods three in. wide had worths varying from 3.28% Δ K/K for a 0.16-in. thick slab to 3.70% Δ K/K for a slab 0.28-in. thick.

Comparison of a step rod, duplicating the analytical model with a slab, showed the worth to be 7% greater for the step rod. Cruciform rod worths varied from 5.3% Δ K/K for a rod with 2.5-in. arms to 7.40% Δ K/K for 3.5-in. arms.

The temperature coefficient of a 6.6-in. radius core was measured to be -0.002% Δ K/K at 20° C and -0.008% Δ K/K at 70° C. The temperature effect from 20 to 70° C was found to be 0.315% Δ K/K. Agreement with the analytically determined temperature effect was good.

Lumped burnable poisons were evaluated in a core with a 10.7-in. radius. Good agreement with analytical self-shielding factors was found.

Fast flux measurements were made at the midplane of the core as 4.5×10^5 n/cm^2 sec watt above 2.5 mev at the center and as 8.6×10^2 n/cm^2 sec watt above 2.5 mev 3 in. outside the pressure vessel.

The excursion analysis for the PM-1 flexible zero power test was initiated.

8. Anticipated Work for Next Quarter

Analytical evaluation of all experiments completed during the parametric zero power test will be completed.

Planning and initial design for the flexible zero power experiment will be initiated.

Compilation of the Hazards Summary Report for the flexible zero power test will be initiated.

The excursion analysis for the flexible zero power test will be programmed completely and initial check runs made.

B. SUBTASK 2.2--IRRADIATION TEST

J. B. Zorn

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.

In general, the best test a fuel element can undergo is a service test under actual conditions, i.e., in a prototype reactor made up of prototype elements. In the case of the PM-1, such a test is not suitable because of schedule requirements. It becomes necessary, therefore, to devise methods of testing the fuel elsewhere as rapidly as possible. This can be done in a reactor in which the neutron flux has been increased to a larger than normal value. Acceleration of burnup is thereby obtained, but several consequences result immediately:

(1) Since power in the irradiated element is directly proportional to the neutron flux, the power generation rate is increased.

(2) A much larger quantity of heat is generated and removal of the heat from the sample element is made more difficult.

(3) Temperatures become more difficult to maintain at nominal values as well as reasonably constant values.

(4) Stresses due to the existence of thermal gradients are increased.

Some specific considerations applicable to an irradiation program are listed.

1. Fuel Element Specimen Size

It is necessary to adhere as closely as possible to the actual physical dimensions of the prototype fuel element in the samples to be tested. Tube diameters, wall thickness, cladding thickness, and the physical size of the cermet core constituents can easily be maintained. However, because of the heat removal problem as mentioned, it may be necessary to reduce the sample length. It is inadvisable to do this for the following reasons:

(1) Full size tests are more reliable. Results are better interpreted if size and/or scale factors do not affect the test.

(2) In the particular case of PM-1 fuel elements, higher L/D ratios are structurally less stable and should, therefore, be tested.

(3) Longer fuel element samples involve greater masses of fissionable material in one element. Therefore, larger amounts of fission gas are potentially available for channeling or diffusion into preferred areas.

2. Heat Removal from the Fuel Specimen

Under service conditions, heat is lost through both the inside and outside surfaces of the PM-1 tubular fuel element. Under this condition, a certain temperature profile across the core of the element is maintained. It is essential that the temperature profile as well as the maximum meat centerline temperature be maintained as close to the actual case as possible. Thermal differentials and the resulting internal stresses can have considerable influence on the structural and metallurgical integrity of the specimen. Severe stress gradients could cause gross distortion in the element with possible unbonding of the clad and core.

3. Temperature and Coolant Flow

Maintenance of a reasonable maximum core centerline temperature (for PM-1 elements, it should not exceed approximately 750° F) is imperative in order that the sample be given a representative test. The maintenance of the maximum core centerline temperature throughout the test (in this case a 100° F total variation is permissible) is also a requirement although it is not as strict as the other one given.

4. Specimen Monitoring

Insofar as possible, as much instrumentation as is practicable should be included to monitor:

(1) Flux and burnup

(2) Temperature

(3) Fission product buildup

(4) Flow and pressure

(5) Specimen failure.

With these general objectives and considerations in mind, the following was accomplished during the quarter.

A proposal for the irradiation testing of PM-1 fuel tubes in the Vallecitos Boiling Water Reactor (VBWR) was received from General Electric. Since it would take considerably longer than desirable to obtain the necessary U-235 burnup rates in the VBWR, and since the plant will be shut down for a period of six months, starting about October 1959, this reactor was excluded from further consideration as a possible facility for irradiation testing under this program.

Several visits took place between Battelle Memorial Institute (BMI) and Martin personnel; these resulted in a proposal from BMI for capsule irradiation of PM-1 fuel specimens in the MTR or ETR. The proposal covered design and fabrication of capsules, irradiation to 36 and 72 at. % of U-235 burnup, and final post-irradiation testing. The program submitted by BMI can be summarized as follows:

(1) Capsules inherently restrict heat dissipation from the central regions of tubular fuel elements; excessive temperatures will result if specimen lengths are too long. An investigation into the limits imposed by capsule design revealed that the maximum active fuel element length that could be tolerated is 10 in.

(2) It was proposed that 16 specimens (5 and 10 in. long) be irradiated. A total of 11 capsules would be utilized for irradiating the samples to burnups of 36 and 72 at. %.

(3) Approximately 4-1/2 mo would be required from initiation of the program to design and fabricate the first of the required 11 capsules. All capsules should be completed in approximately 6 mo.

(4) Irradiation time for 36 and 72 at. % burnup will be 4 and 12 calendar months, respectively.

(5) Results on specimens irradiated to 36 at. % burnup should be available about 13 mo after program go-ahead while preliminary results on the 72 at. % burnup specimens should be available after approximately 18 mo. These time schedules presuppose availability of space in the MTR and/or ETR so that all specimens can be inserted simultaneously.

Negotiations with the Westinghouse Corporation, Pittsburgh, Pennsylvania, on irradiation testing in the WTR, resulted in proposals for both capsule and in-pile loop programs.

In-pile loop testing.- The original proposal from Westinghouse on in-pile loop testing indicated that 13 full length PM-1 fuel tubes could be irradiated in the WTR central thimble with little or no modification to the facility. Further investigation indicated that accelerated power generation in the elements, due to high neutron flux levels, would limit this number to approximately eight specimens. In this case, the heat removal capacity of the loop would have to be increased by at least 500 kw. A more recent review of the problem revealed that four elements can be irradiated in the loop if flow restrictions are employed to obtain coolant velocities of from 15 to 20 fps with no increase in heat exchanger capacity.

It is estimated that the WTR should be at full power (30 mw) by 1 September 1959, and loop space will be available for the PM-1 experiment at this time. The approximate perturbed, thermal neutron flux will be 1 x 10^{14} nv and burnups of 36 and 72 at. % of U-235 can be achieved in 4-1/2 and 12 calendar months, respectively.

Excluding the cost of neutrons, the overall cost of irradiation and post-irradiation testing is somewhat less for the loop test than for the capsule test, although cost per specimen for loop testing is higher.

Capsule testing.- The proposal received from Westinghouse for capsule irradiations included design and fabrication of instrumented capsules, irradiation in the WTR, and post-irradiation examination. In this case, 14 specimens (four 10 in. long and ten 5 in. long) are to be irradiated to burnups of 36 to 72 at. %. Irradiation periods were estimated to be 4-1/2 and 12 calendar months, respectively. Since BMI would design and fabricate the capsules under subcontract with Westinghouse, the same time schedule would apply as discussed previously.

AEC Forms 320, requesting capsule irradiation in the MTR and/or ETR and loop irradiation in the WTR were prepared and submitted to the AEC for approval.

A detailed technical comparison between loop and capsule tests was prepared with the final conclusion favoring loop tests on full length PM-1 fuel elements. A summary is given in Table II-9.

A survey was made of the different fuel element parameters being considered and the various combinations which would statistically provide the most information were selected for irradiation testing. A representative (though not final) list is given in Table II-10 for capsule specimens.

The Materials Test Reactor coolant flow characteristics and the capsule design utilized during the previous Martin irradiation test on miniaturized (3 in. long) PM-1 fuel elements were investigated to determine whether insulating specimens to maintain desired temperatures during irradiation is possible. By this means the more costly and hazardous NaK capsules could be eliminated. With 10 mils of insulation on the

TABLE II-9

Irradiation Testing--Comparison of Loop and Capsule Tests

	Loop	Capsule
Heat removal from both inside and outside of tubes	yes	no
Irradiation of full length fuel elements	yes	no
Continual monitoring of heat transfer medium	yes	no
Detection of element failure	yes	no
Temperature control of specimen	yes	yes, but not after 50% burnup
Production of large mechanical stresses	no	yes
Production of large thermal gradients	no	yes
Simulation of actual temperature profiles	yes	no
Duplication of absolute temperature values	doubtful	very doubtful
Number of elements per physical unit	4	1
Number of parameters varied	6	6
Estimated total cost per specimen (exclusive of neutron charges)	\$16,310 max	\$13,700
Information available (estimated)		
36% burnup	Jan 1960	May 1960
72% burnup	Oct 1960	Jan 1961
Complete	Jan 1961	Mar 1961

TABLE II-10

Capsule Irradiation Specimens

Specimen No.	Tube Diameter OD (in.)	Meat Thickness (mils)	UO_2 in Meat (wt %)	Poison in Meat (wt %)	Active Length (in.)	Burnup (at. %)
136	0.5	30	30	0.22 B as B_4C	5	36
172	0.5	30	30	0.22 B as B_4C	5	72
236	0.5	30	25	0.22 B as B_4C	5	36
272	0.5	30	25	0.22 B as B_4C	5	72
336	0.5	30	25	0.22 B as ZrB_2	5	36
372	0.5	30	25	0.22 B as ZrB_2	5	72
436	0.5	30	25	None	5	36
472	0.5	30	25	None	5	72
572	0.375	30	30	0.22 B as B_4C	5	72
672	0.375	30	25	0.22 B as B_4C	5	72
772	0.5	30	30	0.22 B as B_4C	10	72
872	0.5	30	25	0.22 B as B_4C	10	72
972	0.5	30	30	0.22 B as ZrB_2	10	72
1072	0.5	30	25	0.22 B as ZrB_2	10	72
100	0.5	30	None	3.0 B as B_4C	Max	100
200	0.5	30	None	1.0 B as B_4C	Max	100

outside of the element and coolant flow on the inside and outside, an average element temperature of 650° F can be maintained in a bare element irradiation with a range of insulation thermal conductivity of from 0.2 to 2.5 Btu/hr ft °F. For 10 mils of insulation on both the inside and outside of the fuel element, the variation in thermal conductivity required is from 0.1 to 1.15 Btu/hr ft °F. To date, a combination of solid and/or gaseous insulators which will adequately cover either of these ranges has not been found.

Heat transfer calculations based on MTR conditions indicated that a capsule containing a 10-in. long PM-1 fuel sample in an approximate neutron flux of 10^{14} nv would not cause local boiling of the coolant water. The capsule wall temperature was found to have a maximum value of about 223°F while a wall temperature of 308 to 319°F is required to initiate nucleate boiling.

5. Anticipated Results for Next Quarter

A recommendation on the type of test to be conducted will be submitted to AEC.

Design and preparation for irradiation testing will be initiated.

C. SUBTASK 2.4--HEAT TRANSFER TEST

M. P. Norin

The PM-1 heat transfer program is designed to obtain heat transfer correlations of heat transfer coefficients, tube wall temperature, bulk temperature, and pressure drop and flow effects in both non-boiling and nucleate boiling ranges. Experimental data obtained during the second quarter and correlated with the parametric study program will aid in determining whether or not the PM-1 core can be operated under local boiling conditions.

The experimental program is required because accepted heat transfer design procedures rely on semiempirical correlations, many of which are only accurate to within ± 25%. As a result, these inaccuracies must be incorporated into the design of the reactor as anticipated hot channel factors with a resultant increase in plant capital and operating costs, and a decrease in plant efficiency.

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 first quarter a general heat transfer program was outlined and evaluated.

The test program requires that several full-scale simulated fuel elements be arranged to duplicate flow rates, pressure drops, and other hydraulic characteristics encountered in the PM-1 core. Internal heat generation, through use of a 300-kw controlled d-c power supply, will provide heat fluxes throughout the range of interest.

The experimental program will make use of The Martin Company heat transfer loop and the following three individual test sections:

(1) An existing (0.375 OD) seven-tube unit, designated as SETCH-1 and fabricated during a previous program, will be operated to compare such non-boiling and local boiling heat transfer characteristics as inlet and outlet temperature, wall temperature, and bulk water temperature. Approximately 120 individual runs are scheduled. Test conditions in the PM-1 range of interest span inlet velocities of up to 6 ft/sec, temperatures ranging from 470 to 580°F, pressures of from 1200 to 2000 psi, and a heat flux range of from 0.05 to 0.45 x 10^6 Btu/hr ft^2.

(2) A local boiling pressure drop correlation will be required for the program. This cannot be obtained on the seven-tube section, in which only flow inside the tubes is measured. A single tube test section, fabricated and tested during the shakedown of the heat transfer loop, will be modified to provide the required information. The test section, designated STTS-2, will be operated under the same conditions as SETCH-1.

(3) Based upon information obtained in the experimental programs of SETCH-1 and STTS-2 and results of the parametric studies, a PM-1 test section will be designed and tested. This multiple or single tube unit will duplicate the final selected PM-1 core conditions of geometry, flows, pressures, temperature, heat flux, and other hydraulic dimensions. This unit will be used to evaluate the PM-1 final heat transfer design.

1. Work Completed

Fabrication of SETCH-1 has been completed. Calibration of the center tube thermocouples was made and data obtained on the experimental friction factor of the tubes as a function of Reynolds' number.

Welding of test section mating piping into the Martin heat transfer loop was completed. The test section was installed in the loop in prepatation for instrumentation hook-up. The test section is shown in Fig. II-13.

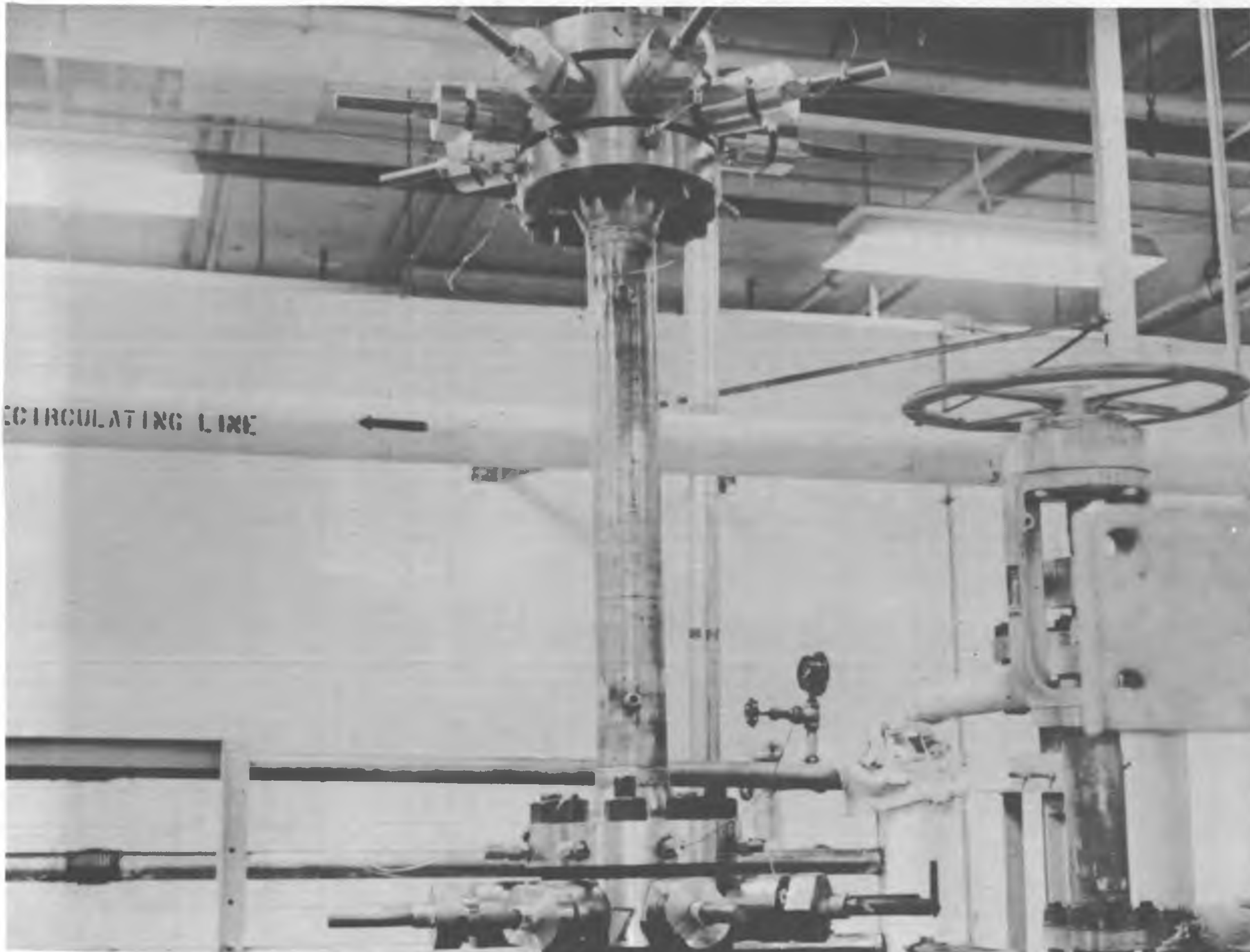


Fig. II-13. Seven-Tube PM-1 Test Section

Design of the single tube test section (STTS-2) was initiated and completed. Dimensions of this test section are such that it is suitable for installation within the present heat transfer loop. This section consists of a single stainless steel tube of 0.375 in. OD and 0.315 in. ID and an effective length of 23 in. Taps for obtaining pressure drops within the tube are located on 4 in. centers along the effective length of the tube. Allowances for the mounting of thermocouple fittings are made on 2 in. centers. Smooth flow at the entrance and exit of the test section is established through the use of long flow transition pieces. Electrical bus connections are available at each end. Structural integrity is maintained during testing through the use of backup plates. Fabrication was started during this quarter and is scheduled for completion during June 1959.

2. Anticipated Work for Next Quarter

During the next quarter testing of SETCH-1 and STTS-2 will be completed. Planning and design of final PM-1 heat transfer test will be initiated.

D. SUBTASK 2.5--ACTUATOR PROGRAM

J. Sieg R. Manoll W. Dallam

During the first quarter, the actuator program work was concentrated in two main areas: seal studies, and preparation for selection of an actuator vendor, during the second quarter.

1. Seal Studies

The objective of the seal studies was to determine the suitability of seal systems for use with the PM-1 actuators. Information will also be applicable to other sealing requirements, such as pump shafts, valve stems, etc.

The current status of seal technology was surveyed by:

(1) Extending to qualified seal vendors invitations to bid on a seal system meeting the following criteria:

For rotary shaft seals:

Shaft diameter: 1/2 to 2 in. diameter (5/8 to 3/4 in.) preferred.

Temperature: 500 to 650°F.

Pressure: 1000 to 2500 psi.

Lubrication: High purity water.

Leakage: 70 cc/hr max.

Operating speeds: 1 rpm, min to 1 rps, max.

For the linear (reciprocating shaft) seal:

Shaft diameter: 1/2 to 2 in. diameter (5/8 to 3/4 in.) preferred.

Temperature: 500 to 650°F.

Pressure: 1000 to 2500 psi.

Lubrication: High purity water.

Leakage: 70/cc/hr max.

Operating speeds: 1 fps, min to 8 fps, max.

(2) Evaluating the proposals received. All the manufacturers contacted except three, Koppers, Cartriseal and Hydrodyne, declined to propose a seal for our service.

(3) Mating proposed systems to an existing actuator design.

Three basic types of seals were studied:

(1) The floating ring seal as shown in Fig. II-14.

(2) The split ring type as shown in Fig. II-15.

(3) The bellows-operated pressure plate type as shown in Figs. II-16 and II-17.

The data pertaining to the seals was evaluated (see Table II-11) on the basis of reliability, wear rate, leakage, life expectancy, maintenance, and ease and cost of replacement. The floating ring seal or labyrinth is a constant leak rate type of seal. External pressure slightly higher than the pressure of the medium being sealed is provided at the seal gland, with this flow being diverted both inward and outward. The outward flow (leak rate) remains constant over a long period of time and is not affected greatly by reasonable wear. This type of seal is relatively large, requires external piping, and is quite costly. On the other hand, seals of this type have been used very successfully for a number of years.

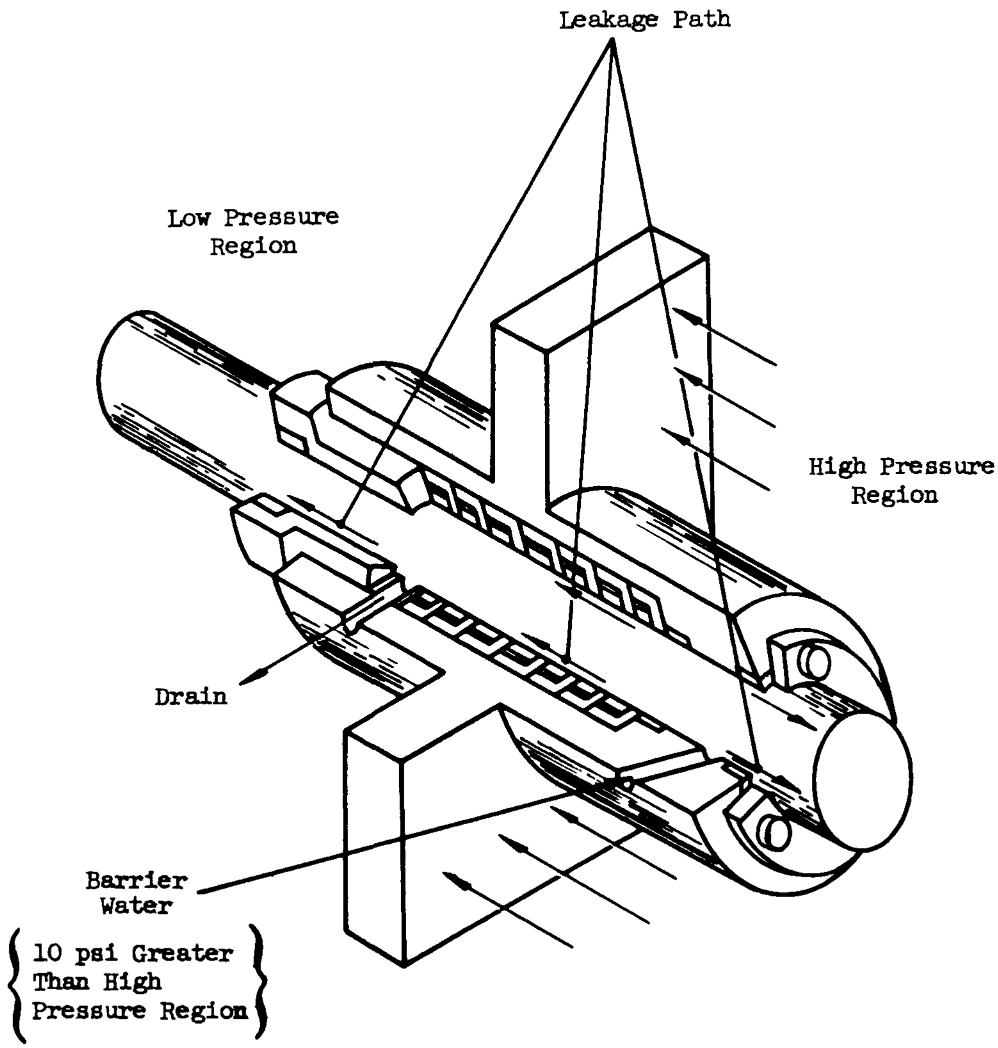


Fig. II-14. Koppers Company Floating Ring Seal

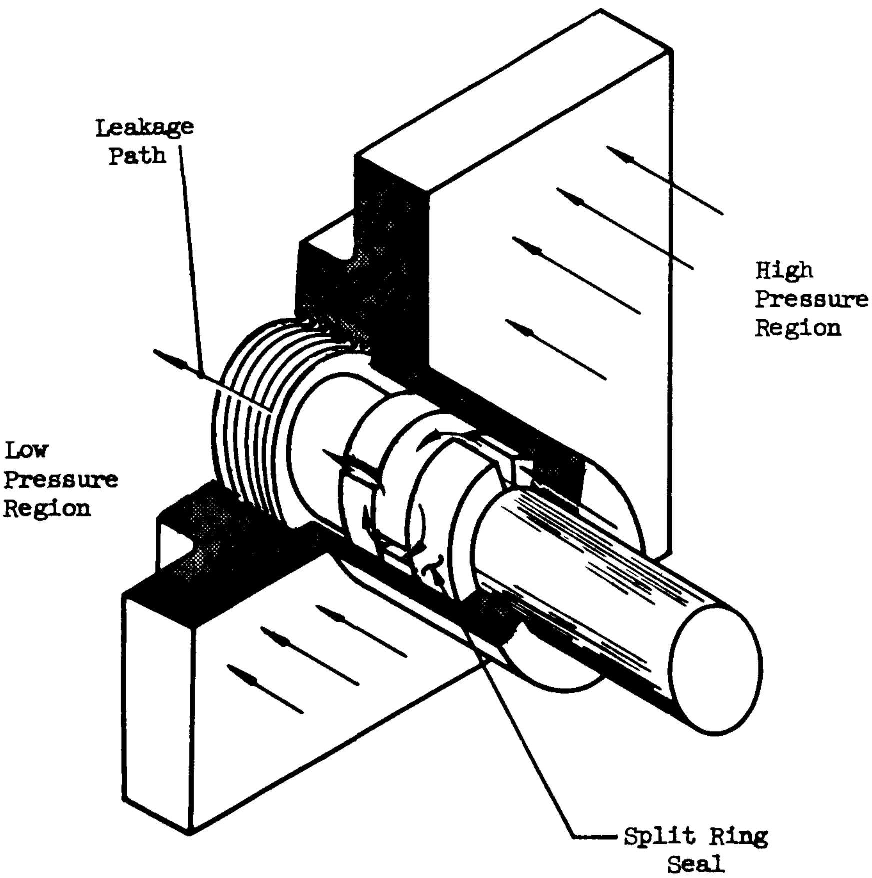


Fig. II-15. Koppers Company Split Ring Seal

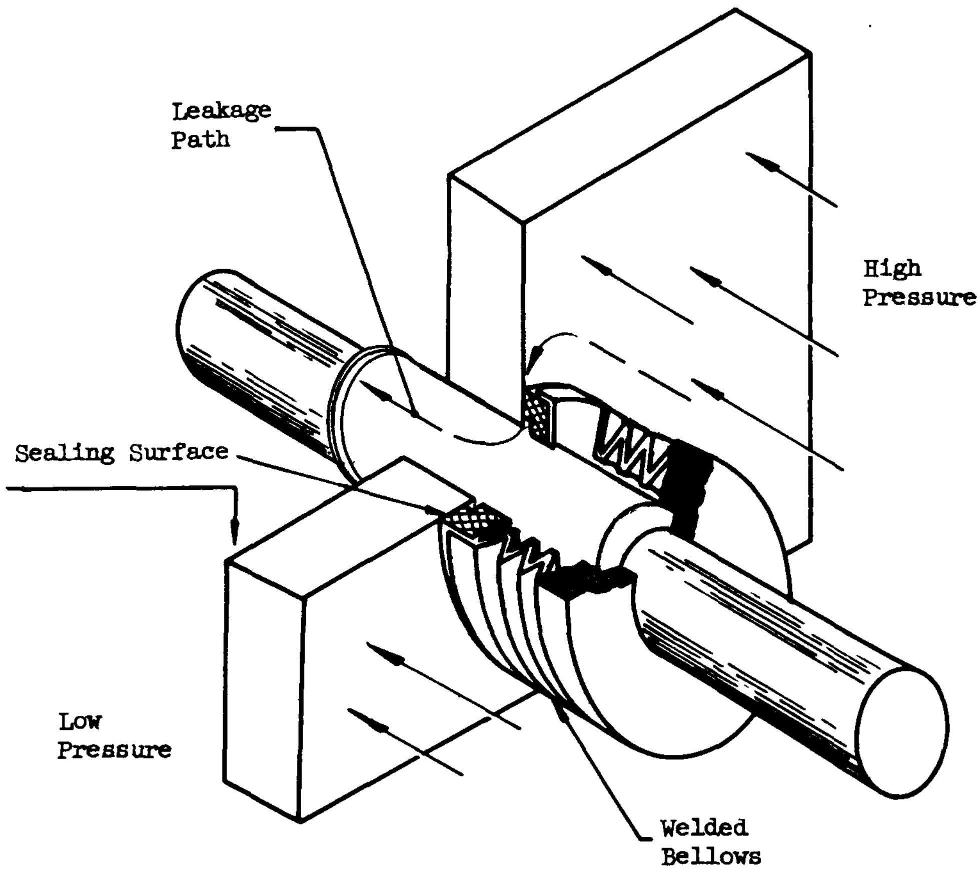


Fig. II-16. Cartriseal Corporation Bellows Seal

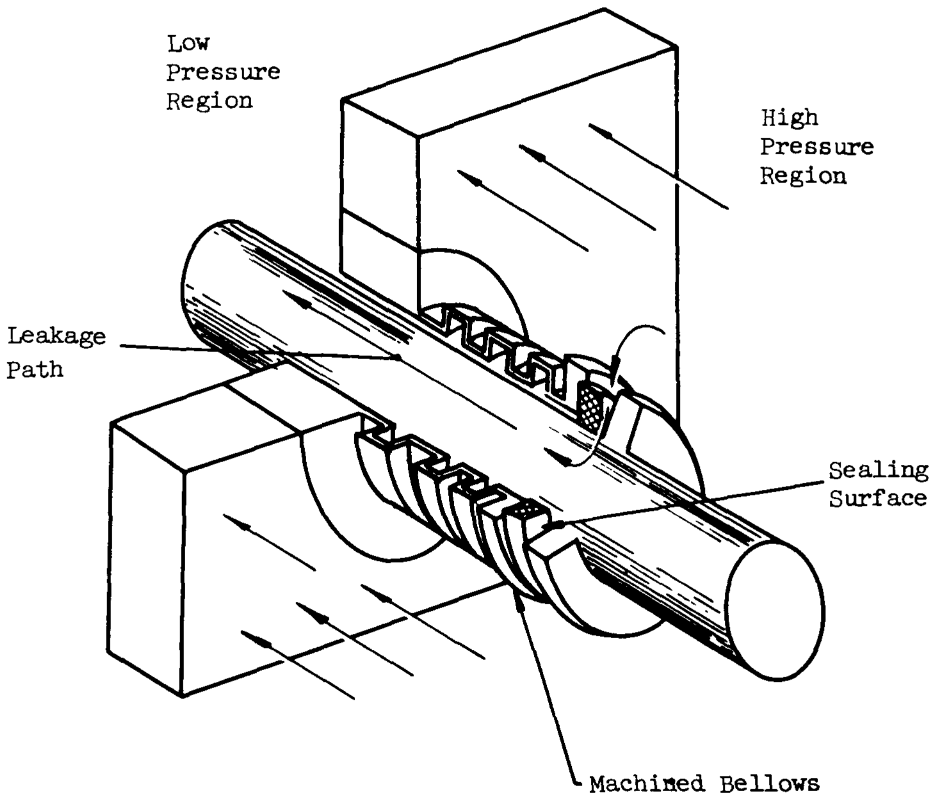


Fig. II-17. Hydrodyne Corporation Bellows Seal

TABLE II-11

Seal Data Evaluation

Manufacturers Number	Seal Type	Rotary	Linear Recip-rocating	Leak Rate Static (cc/hr)	Wear Rate	Max Temp (°F)	Max Press (psi)	In Service	Seals	2-yr Life	Cooling Water Required	Fab Time (wk)	Cost/Unit ($)	Remarks
Koppers No. 300017	Floating Ring (Bushing)	Yes	Yes	1 gal/hr	Very low over 2 yr Period	650	2500	Yes	HP Water	Yes (1)	Yes (2)	Not Stated	800 Approx	Not adaptable for remote removal due to external piping.
Koppers No. 700052	Welded Bellows	Yes	No	30	Very low over 2 yr Period	650	2500	No	-	Yes (3) (8)	No	Not Stated	1800	This is an unproved seal and is still in the development stage.
Koppers No.	Split Ring	Yes	Yes	70	Very low over 2 yr Period	650	2500	Yes	Oil	Yes (4)	No	4 to 6	26 Each (5) 3 Required	This seal shows promise but requires testing.
Hydrodyne No. 10255	Machined Bellows	Yes	No	30	Very low over 2 yr Period	650	2500	Yes	No HP Water Exp	Yes (6)	No	6	203	This seal shows promise but requires testing.
Cartriseal No. 3246	Welded Bellows	Yes	No	30	Very low over 2 yr Period	650	2500	Yes	No HP Water or Nuclear Exp	Yes (7) (8)	No	10	167.50	Metallurgical problems have to be overcome. (8)
Cartriseal No. 3240	Spring Loaded	No	Yes	30 to 50	Very low over 2 yr Period	650	2500	No	No HP Water or Nuclear Exp	Yes (7)	No	10	85	Metallurgical problems have to be overcome. (9)

(1) This seal has been used in high purity water service and test data are available from ANL.
(2) External piping required for water inlet at a higher pressure than the reactor vessel.
(3) Fatigue life of bellows unknown and wear rate of sealing surface uncertain.
(4) Wear rate unknown using high purity water as a lubricant.
(5) Cost does not include seal housing included on other quotes.
(6) Wear rate of sealing ring in HP water unknown.
(7) Materials of construction unsuitable for radiation level and HP water service.
(8) Crevice corrosion is a factor at the welded seam of the bellows.
(9) This company shows no experience in the field of nuclear problems.

Note: Leakage rates are calculated and not proven by lab tests.

The split ring seal is the type used in automotive engines (piston rings) and for other applications and requires very little space to install. This type of seal is relatively inexpensive. Its leakage should be low over a two-year period of operation, but may be expected to increase with continuing operation.

The bellows-type of seal which uses a bellows to apply pressure to a sealing disk has been used quite extensively with systems operating up to 5000 psi and handling various types of fluids. This type of seal has a relatively low initial leak rate, but leakage increases with wear. If a welded-type bellows is used, tests should be conducted to determine the effects of crevice corrosion at the base of the welds and determine the fatigue resistance of the metal in the weld area during cycling. The Hydrodyne bellows is machined from a solid piece of material. With this method of fabrication, crevice corrosion and discontinuities at welded joints are not factors of importance.

The seals studied may be catagorized in the following manner:

(1) The floating ring seal or labyrinth seal should be used where space is not a problem, and pressure and drain piping can be made readily, due to the high degree of reliability of the seal.

(2) The split ring seal and/or the machined bellows seal should be used where space is at a premium but replacement can be made rapidly. Tests would have to be conducted on the seals (as submitted) to determine seal reliability and life expectancy.

Decisions regarding the use of seals with actuators will be held in abeyance pending review of the actuator studies and establishment of a preliminary design. Some of the actuator manufacturers may have redesigned an existing seal for their required type of duty which is worthy of further consideration.

2. Preparation for Actuator Selection

The objective of the first phase of the actuator study was to accumulate sufficient actuator data to allow immediate actuator procurement when reactor preliminary design conditions are established.

The development of firm data to be used in selecting a type of actuator for procurement was begun. It was decided that the best way to obtain such data, in meaningful form, was to invite fixed price proposals for the supply of prototype and production equipment. A specification was prepared, bid invitations were extended to qualified vendors, evaluation criteria were established, personnel were mobilized for evaluation of design work, and technical liaison was initiated with vendors.

Information pertaining to the actuator study has not been formally submitted as yet by the various manufacturers. However, a review of typical actuators is continuing, based upon initial sketches provided by representatives of various firms.

Several methods of actuating the control and shim rods are presently employed in existing reactors in which the failure-prone components are removable without depressurizing the primary system. These fall into two basic categories:

(1) The type in which the mechanical components are placed within a pressure housing (restraining the primary system pressure) and are motivated by magnetic or hydraulic forces passing through the pressure housing. The electrical components are placed outside the pressure housing and the drive rod that propels the control rod is located within the pressure housing. In this type of actuator the leakage rate is zero and drag or friction forces that would result from the use of extremely close fitting parts are essentially eliminated.

(2) The type that employs a dynamic seal to isolate the elements that might readily fail from the primary system fluid. In this type of actuator some leakage is inherent and scram of the rod may have to take place through the seal.

Under the first type of actuator three basic actuators are being considered. These are:

(1) The magnetic jack; currently expected to be used with the Yankee and BR-3 (Belgian) reactors. This actuator is an in-line type of device in which the magnetic power bundle and the position indicating bundle are mounted outside the actuator pressure thimble. Replacement of the electrical parts without disturbing the internal components may quickly be made. This device operates from a square wave power supply and moves the control rod in steps or increments in either direction. Scram is accomplished by collapsing the field of the holding magnet permitting the rod to drop.

(2) The collapsible rotor ball screw drive presently used on nuclear submarines.

This actuator has the same accessiblity features as the magnetic jack and is an in-line type of device, but movement of the rod is accomplished by a ball screw and rotating nut.

The nut is composed of several segments, spring-loaded to remain in the open position, with each segment having a roller screw section at one end and magnetic material at the opposite end. The rotation of the nut is accomplished by energizing a rotating magnetic field (located on the outside of the pressure housing) which propels the nut. Scram is accomplished by collapsing the field of the nut-holding magnet permitting the segments to open and release the screw.

(3) The hydraulic piston--a concept which has not been built and tested. This actuator is an in line type device consisting of a piston in its pressure housing. Surrounding this housing is a spiral grooved cylinder acting as a valve. When lined up with a particular port in the inner housing this cylinder dictates the position of the piston which has pressure on each side balanced through the port. The outer spiraled cylinder (driven by a motor-operated gear through a seal) incorporates a spring to drive the cylinder to the down position for scram.

Several actuators employing a seal are under study; they employ either the rack and pinion, ball, or roller nut drives. These have been used extensively on heavy water, swimming pool, and boiling water reactors.

The rack and pinion type actuator employs a magnetic clutch for separation of the power source from the rod drive actuator tube for scram. The usual configuration of this type of drive is a vertical tube for the actuator rod and a horizontal mounting of the power package. This can present a problem in cases where compactness is of great importance.

The ball or roller nut screw-type actuators usually employ a contact magnet between the actuator rod and the power package. Collapse of this magnetic field results in a scram. This type of scram system may be arranged in-line for space savings.

The data pertaining to the various type actuators will be evaluated on the basis of reliability, wear rate, leakage, power consumption and requirements, life expectancy, maintenance, and ease and cost of replacement. Data pertaining to release and drop times and general operating characteristics will also be evaluated.

No conclusions have been reached on the actuators during the quarter due to the lack of final information requested from the manufacturers.

No decisions have been made relative to the selection of any specific type of actuator as yet; it is expected that a decision will be rendered during the second quarter, after receipt and evaluation of the vendors' final information and proposals.

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

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

Project Engineer--Subtask 3.3: C. Fox

This task covers preparation for and accomplishment of preliminary design.

A. SUBTASK 3.1--PARAMETRIC STUDY

The objective of the parametric studies is to examine the variables of plant design in sufficient detail to enable subsequent preliminary design of an optimized plant.

The parametric study may logically be subdivided into four major areas: primary loop, secondary system, instrumentation, and configuration. The major areas have been further subdivided as:

(1) Primary loop--R. Akin

Nuclear studies
Heat transfer studies
Shielding studies
System studies
Design studies

(2) Secondary system--C. Fox

General system studies
Steam generator studies
Turbine-generator studies
Miscellaneous studies

(3) Instrumentation--C. Fox

(4) Configuration--C. Fox

Packaging and housing
Containment

The parametric studies will be completed during the next quarter; the results of the study will form the basis for preliminary design.

1. Primary Loop

Nuclear studies.-

R. Hoffmeister E. Scicchitano F. Todt

Nuclear studies performed during the first quarter consisted of:

(1) Parametric core lifetime studies to provide data for establishing the preliminary core design.

(2) Studies of the worth of seven control rods under various conditions.

(3) Preliminary study of the feasibility of lumping burnable poison in the core.

The effect of eight independent variables on the initial K_{eff} and the life of uniformly loaded cores was studied. This made the selection and more detailed study of a series of cores meeting PM-1 criteria possible. Uniform burnup of both fuel and burnable poison was assumed, although an average nonuniform burnup effect was calculated and applied.

Evaluation consisted of arranging data to show fuel inventory versus number of fuel elements for cores giving two-year core life, to show the relative effects of the different variables on core life, and to show the variation of core life with deviation from the optimum levels of the different variables.

Detailed nonuniform burnup studies for several promising core designs were also completed. The K_{eff} was plotted against time for the core under operating (hot dirty) and down (cold clean) conditions; overall control requirements may be determined from these plots.

Control rod studies consisted of determining the worth of a seven-rod system as a function of the radial location of the eccentric ring of six rods, of the core diameter for fixed radial rod location, and of the core height for a fixed radial rod location.

A preliminary study to determine the feasibility and worth of incorporating lumped burnable poisons into the core was started. The variation of reactivity with time of cores incorporating homogeneously distributed poison and of cores incorporating lumped poisons of various initial concentrations are being determined.

During the next quarter, evaluation of the parametric core lifetime studies will be completed, nonuniform fuel loading will be investigated, and a preliminary core design will be established.

The number and size of control rods needed to meet the control requirements will be determined.

The preliminary lumped burnable poison study will be completed. Nonuniform loading of lumped poison will be investigated. A preliminary placement of lumped poison in the core will be determined.

Parametric core design studies.- The effects of eight independent design variables on core characteristics were evaluated to allow the range of subsequent investigation of each of the variables to be narrowed. The eight independent variables considered and the range investigated for each of the variables are shown in Table III-1. A synthetic design code was used which made it possible to plot complete curves based upon 81 calculations rather than the thousands of calculations required to determine five values of the dependent variable for each combination of independent variables. The validity of the synthetic design was confirmed at several points.

The parametric uniform loading core design studies resulted in the generation of a large quantity of data. Although presentation of all of the data is not made here, summaries of the results obtained, with sample representations of these data, are presented.

A complete series of graphs have been plotted to show the effect of the eight independent variables on the primary dependent variable, reactor lifetime, and the secondary dependent variables:

(1) Initial K_{eff} (a direct function of the primary variable)

(2) Fuel inventory

(3) Number of fuel elements.

A typical representation showing the effect of varying X_2, X_3, X_4, and X_6 on reactor lifetime is presented as Fig. III-1. In this particular case, X_1, X_5, X_7, and X_8 are held constant. Twenty-six other representations were prepared which indicated the effects on reactor lifetime of varying combinations of the other independent variables. To simplify reading the graph, only the extreme values of variable X_3 were plotted as surfaces; points for the middle surface are plotted but not connected.

TABLE III-1

Independent Variables Used in Parametric Core Design Studies

Symbol	Design Variable	Range	Synthetic Design Levels				
			-2	-1	0	+1	+2
X_1	Core height/diameter	0.85 - 1.45	0.85	1.0	1.15	1.30	1.45
X_2	Core diameter--in.	20 - 28	20	22.271	24.331	26.227	28
X_3	Fuel Tube OD--in.	0.31 - 0.50	0.310	0.3575	0.4050	0.43895	0.500
X_4	Fuel Matrix Thickness--in.	0.018 - 0.03	0.018	0.021	0.024	0.027	0.030
X_5	Temperature--° F	400 - 550	400	437.5	475	512.5	550
X_6	Tube Spacing/Tube OD	1.1 - 1.5	1.10	1.2	1.3	1.4	1.5
X_7	Grams of Boron/Grams U-235	0.0 - 0.013	0.0	0.00325	0.0065	0.00975	0.013
X_8	Weight % UO_2 in Matrix	23 - 30	0.23	0.2475	0.265	0.2825	0.30

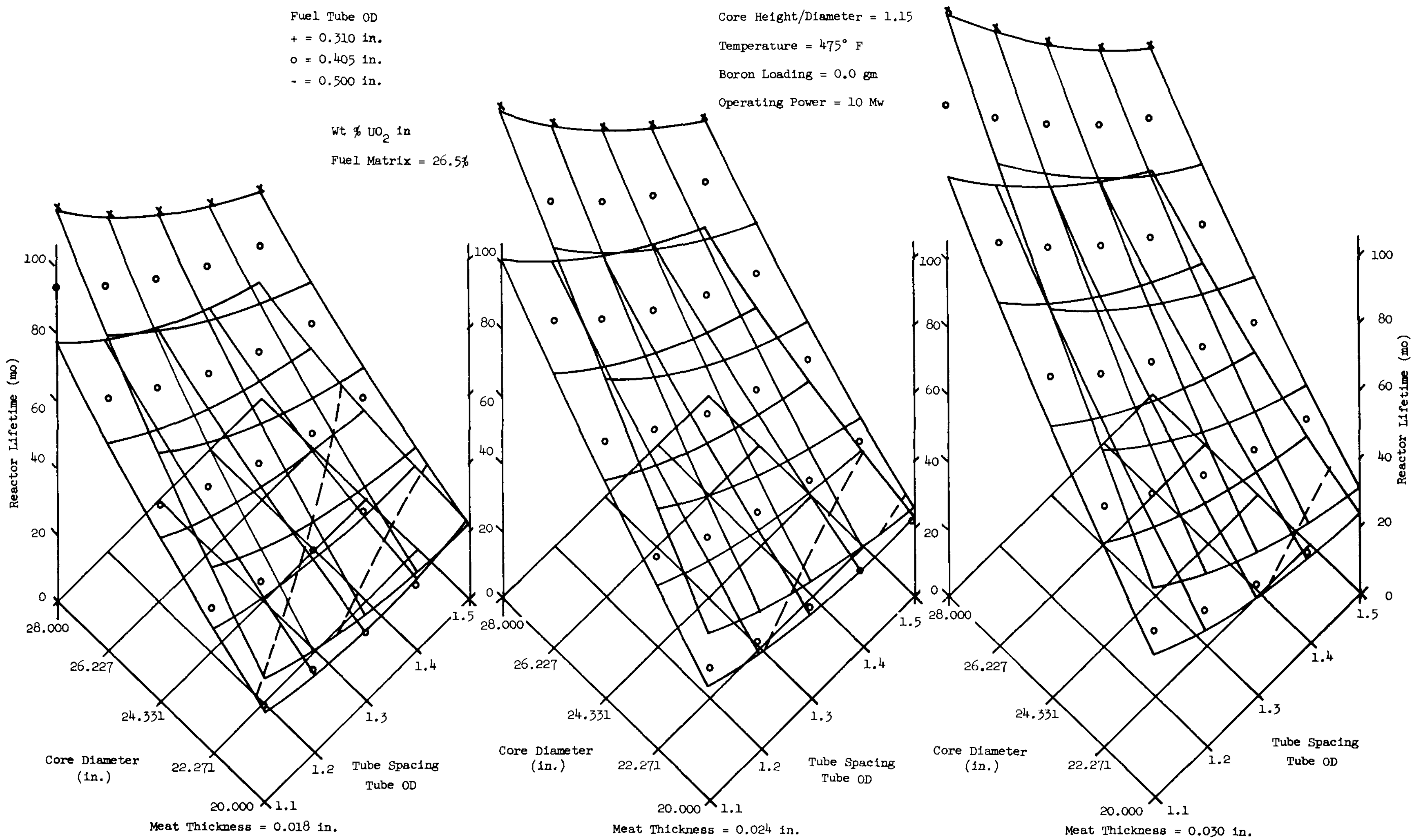


Fig. III-1. Reactor Lifetime--Uniform Core Loading

The dashed lines appearing on some of the surfaces of Fig. III-1 represent the intercept of a 24-mo reactor plant core life plane and the surface. It may be noted that the intercept is not precisely 24-mo (on the ordinate scale) above the surface of the base plane in any case; this is because certain effects were considered in plotting the intercepts of reactor plant core life which were not taken into account in the basic plot of reactor lifetime, namely:

(1) An across-the-board correction of 25% was applied to account conservatively for the effects of nonuniform burnup. (This is in addition to the correction applied in the computer code.) As a result of this, the 24-mo core life curve would be parallel to the base plane at a height corresponding to an ordinate reading of 30 mo.

(2) The effect of operating temperature on required reactor power was included by making allowance for the increase of plant thermal efficiency (Rankine Cycle) which accompanies an increase in operating temperature. Since the reactor operating power necessary to meet plant requirements at 475° F is somewhat less than the 10 mw power criteria used in the computer codes, a downward shift of the 24-mo core life curve from 30 to about 28 mo results.

A typical representation of the effect of varying X_2, X_3, X_4, and X_6 on U-235 inventory is presented as Fig. III-2. In this particular case, X_1, X_7, and X_8 are held constant. Eight other representations were prepared which indicate the effects on reactor lifetimes of varying combinations of the other independent variables. The elimination of variable X_5, temperature, explains the decrease in representations from 27 to 9.

The number of core fuel elements is a function of variables X_2, X_3, and X_6 only. A supplementary three-dimensional graph was prepared to allow one to determine quickly the number of tubes resulting from any combination of these parameters.

To simplify interpretation of the results in special cases, an alternate graphical presentation was compiled. These graphs, obtained by cross plotting information from the 27 representations similar to Fig. III-1, show the effect of each individual variable, with all other variables at either their mean or extreme levels, on either reactor lifetime or initial K_{eff}. Figure III-3, for example, shows reactor lifetime as a function of core diameter, fuel concentration, and tem-

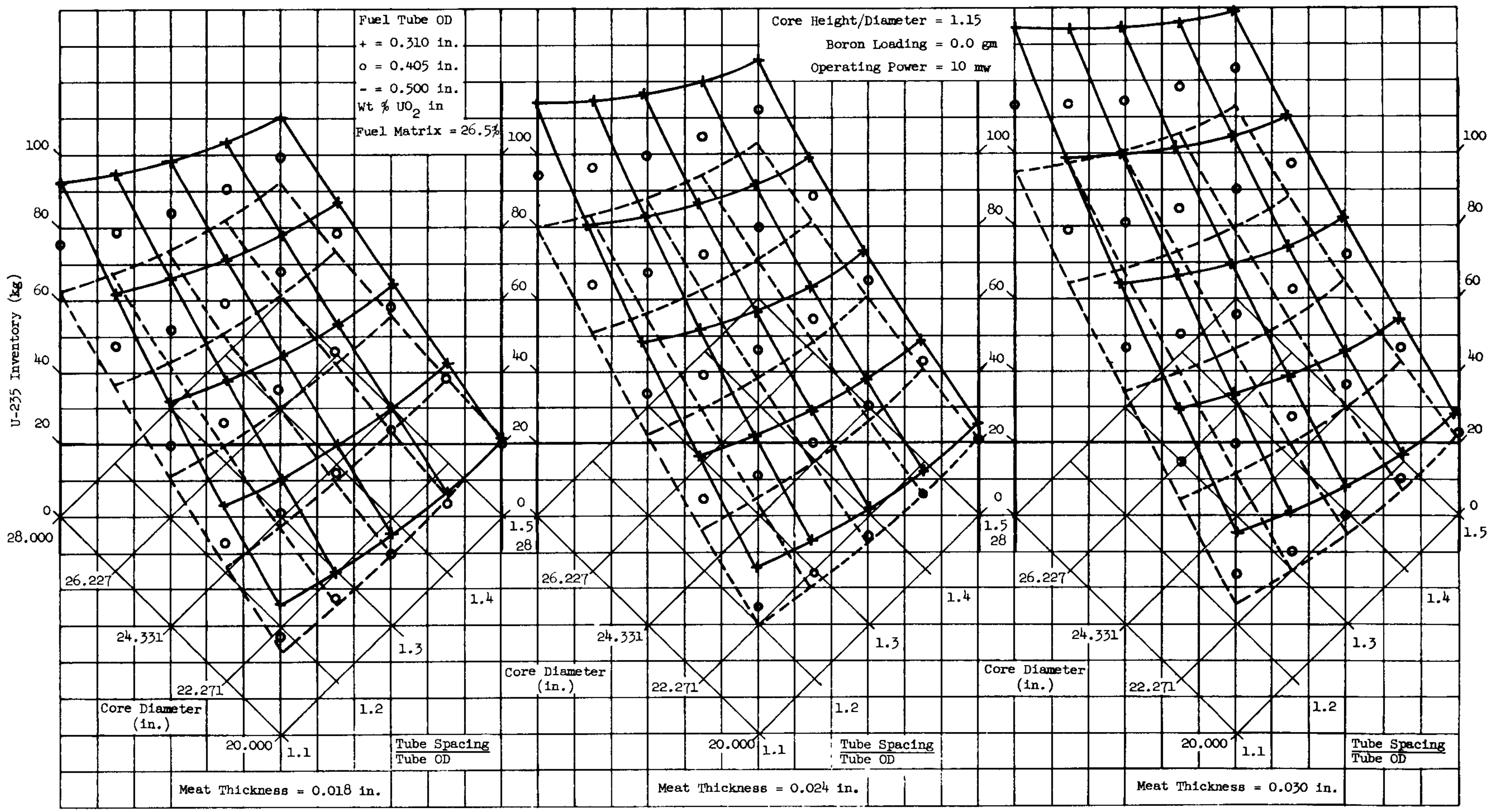


Fig. III-2. Fuel Inventory--Uniform Core Loading

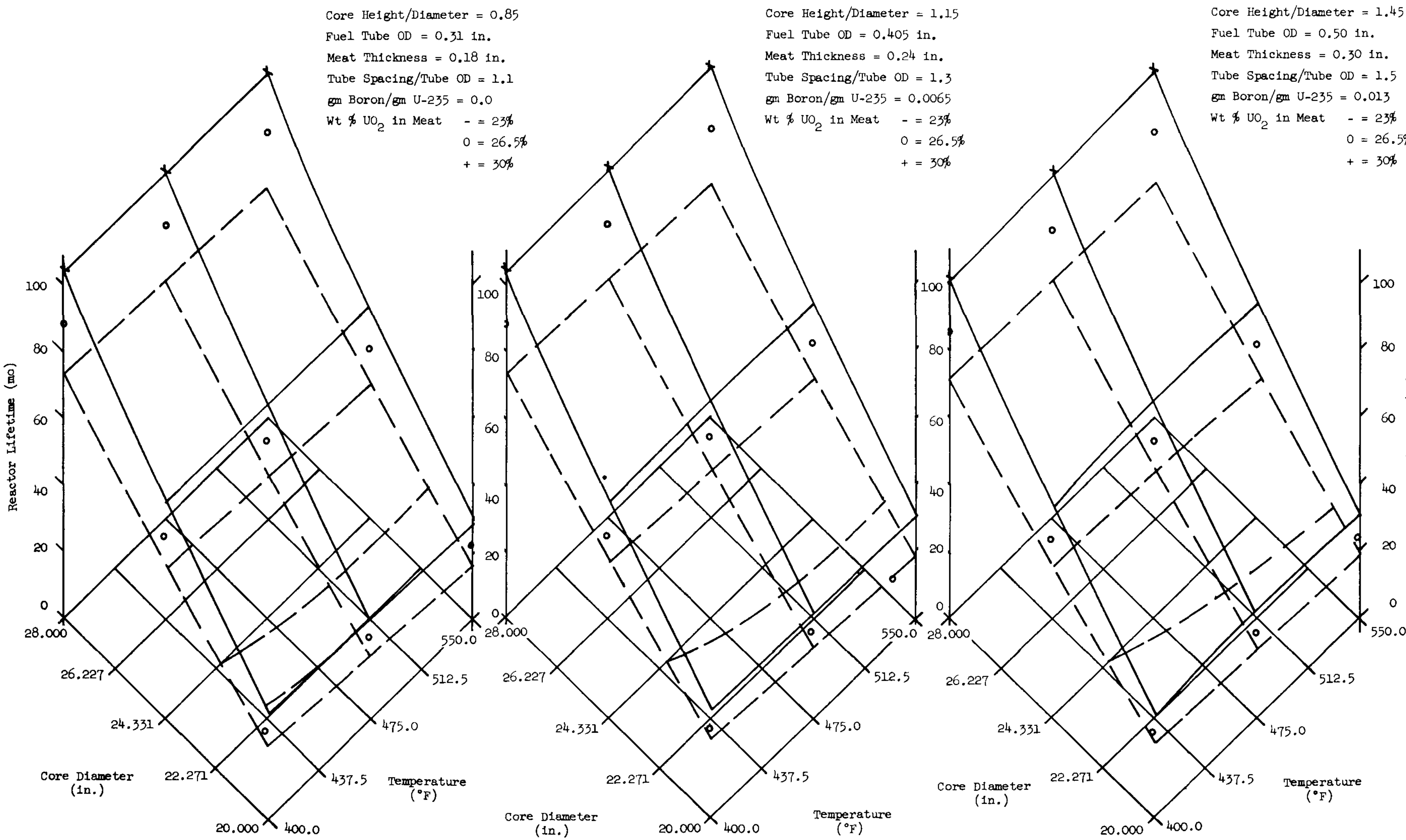


Fig. III-3. Reactor Lifetime--Uniform Core Loading perature: in graph 1, all other variables are held at the -2 level; in graph 2, all other variables are held at the 0 level; and in graph 3, all other variables are held at the +2 level.

Prior to evaluating the data obtained, it was necessary to establish the validity of the data generated. To this end, several studies of the accuracy of the synthetic design curve fit were completed. A check calculation of about 20 of the points in the -1 to +1 range indicated that the difference between calculated check points and plotted points was less than the error to be expected in reading the graphs. A series of 16 check point cases for combinations of extreme levels of several independent variables in the range of interest were run. Maximum error was $\approx$ 15%.

As mentioned previously, an average nonuniform burnup correction was applied to all cases. In order to economize on machine time, it was calculated only for the one core defined by middle-range data. The correction was checked by performing detailed nonuniform burnup studies for the three cases described in Table III-2. Curves similar to Fig. III-4 were obtained showing reactivity versus time for a hot dirty and a cold clean core. The core lives calculated in the uniform burnup study, in these particular cases, were found to be 10 to 20% less than were those calculated giving detailed consideration to nonuniform burnup.

TABLE III-2

Core Designs Evaluated in Detailed Nonuniform Burnup Studies

	No. 1	No. 2	No. 3
Core Height/Diameter	1.45	1.046	1.25
Core Diameter--in.	20	22	24
Fuel Tube OD--in.	0.50	0.50	0.50
Fuel Matrix Thickness--in.	0.030	0.030	0.030
Temperature--°F	475	475	475
Tube Spacing/Tube OD	1.3	1.3	1.3

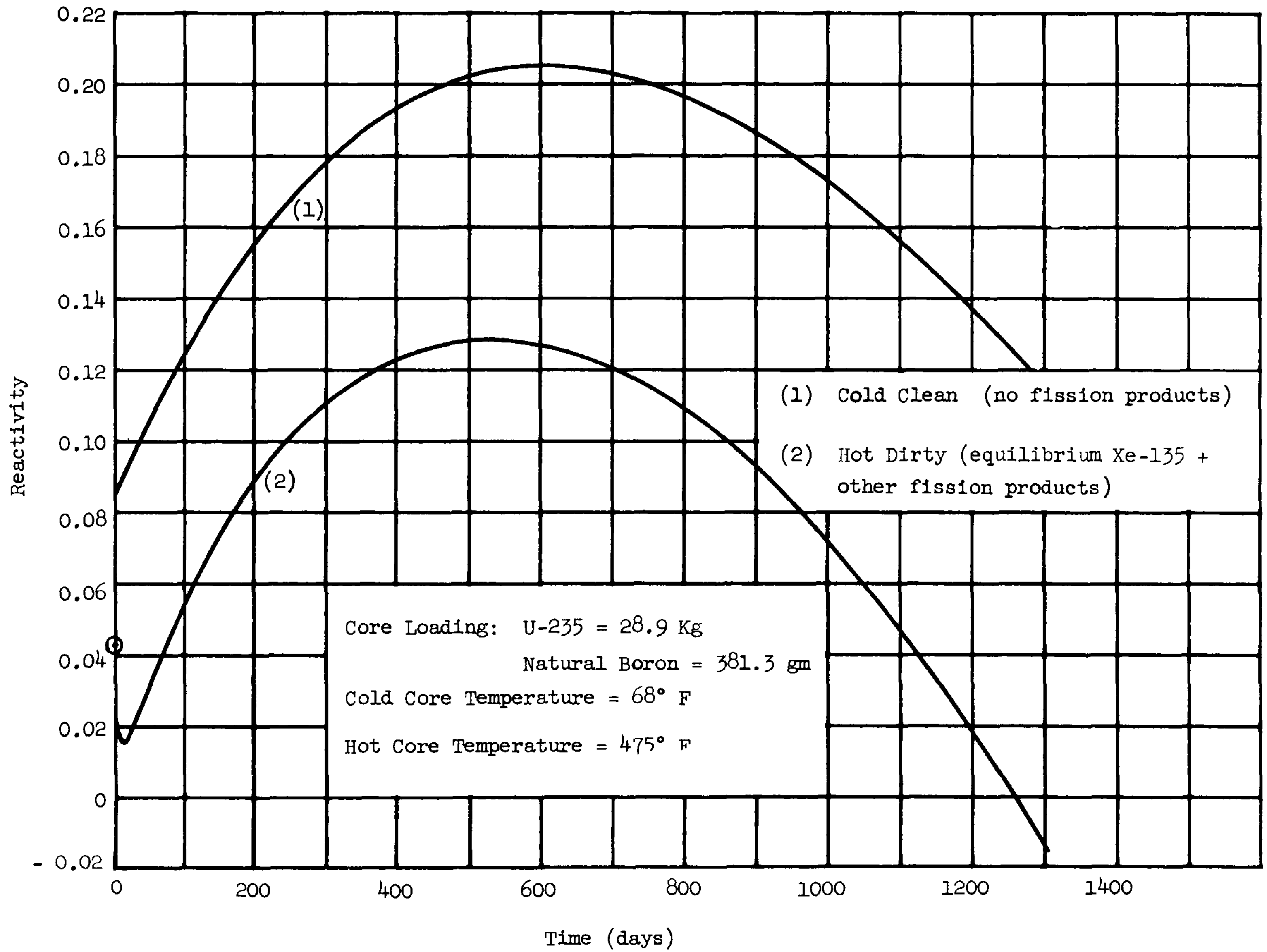


Fig. III-4. Reactivity vs Time, Nonuniform Burnup (Case 1, Table III-2)

TABLE III-2 (continued)

	No. 1	No. 2	No. 3
*Effective Tube Spacing/Tube OD	1.430	1.405	1.387
**Grams of Boron/Grams of U-235	0.0132	0.0136	0.0151
Weight % UO_2 in Matrix	28.25	28.25	23.00
Number of Fuel Tubes	709	889	1086
Fuel Inventory (U-235)--kg	28.9	28.8	36.7
Core Life (at 9.4 mw)--mo	42	42	52

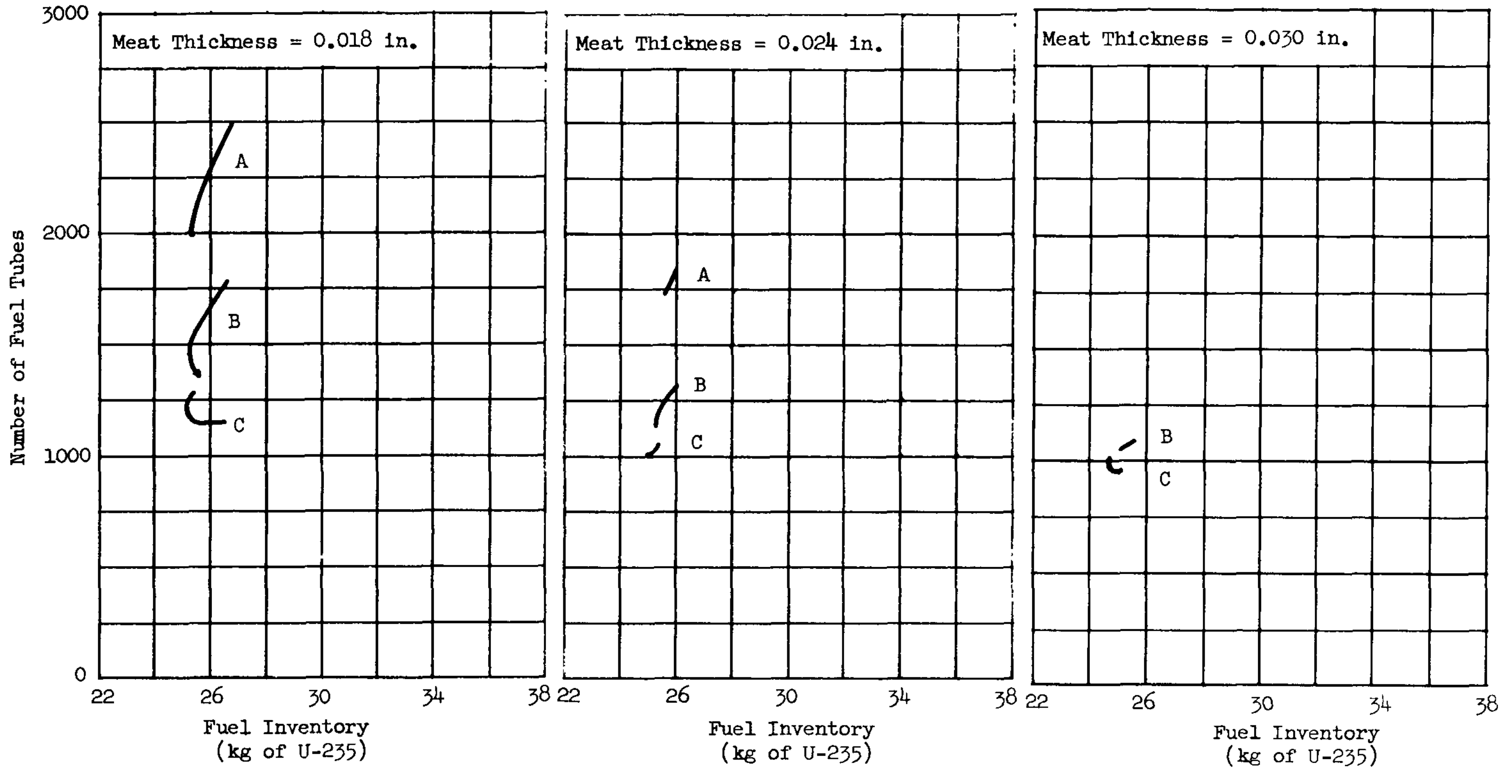
*Allowance was made for removal of 150 fuel elements to accommodate control rod channels.

**Amount of boron necessary to reduce initial K_{eff} with equilibrium Xe-135 to ~ 1.015.

The nuclear data obtained are considered to be reasonably good guides for the selection of a narrow range of parameters to be considered during preliminary design. Evaluations of studies completed to date include:

(1) A series of graphs, similar to Fig. III-5, plotting fuel inventory against the number of fuel elements contained in cores with a rated life of two years. The nonlinear shape of the curves results from the fact that both inventory and the number of tubes are functions of two variables: X_2 and X_6.

(2) A series of seven representations, similar to Fig. III-6, show the effect on core life, number of fuel elements, and fuel inventory of varying either X_1, X_2, X_3, X_4, X_5, X_6, or X_7 while holding the other variables constant. In most cases, three ranges of X_8 are also plotted. Points from which a



Core Height/Diameter = 1.15

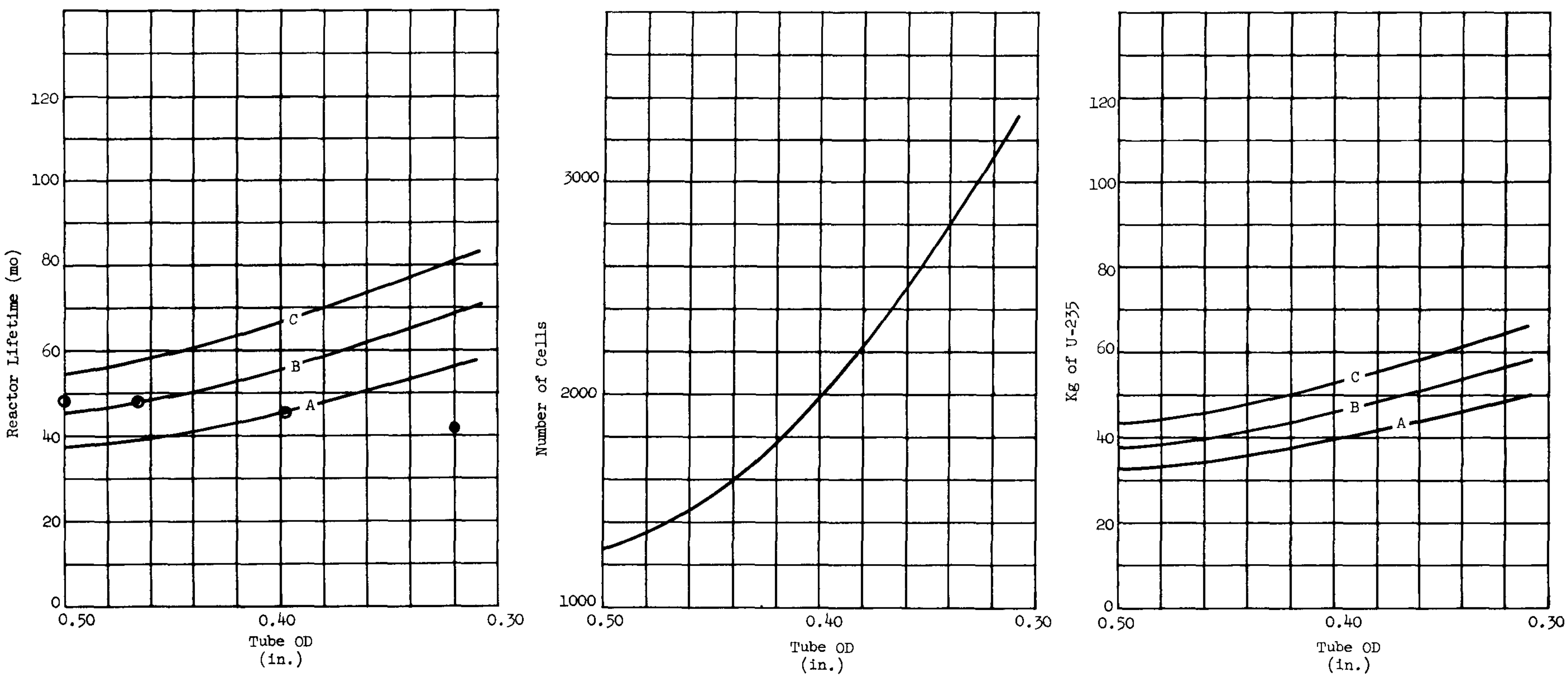
Tube OD = A-0.310, B-0.405, C-0.500 in.

Temperature = 475° F

gm Boron/gm U-235 = 0.0

Wt % UO_2 in Meat = 26.5%

Fig. III-5. Fuel Inventory vs Number of Fuel Tubes for 2-yr Core Life



Core Height/Diameter = 1.15

Core Diameter = 24.331 in.

Fuel Matrix Thickness = 0.024 in.

Temperature = 475° F

Tube Spacing/Tube OD = 1.3

GM Boron/gm U-235 = 0.0065

Wt % UO_2 in Matrix:

A = 23%, B = 26.5%, C = 30%

⊙ Fuel Inventory Constant = 40 kg of U-235

Fig. III-6. Fuel Tube OD vs Lifetime, Cells, and Fuel Inventory--Uniform Core Loading curve showing the variation of reactor lifetime as a function of the independent variable for constant fuel inventory can be plotted are superimposed on the reactor lifetime graph.

Control rod studies.- A parametric study to evaluate a seven-rod control system in cores of varying heights and diameters was completed.

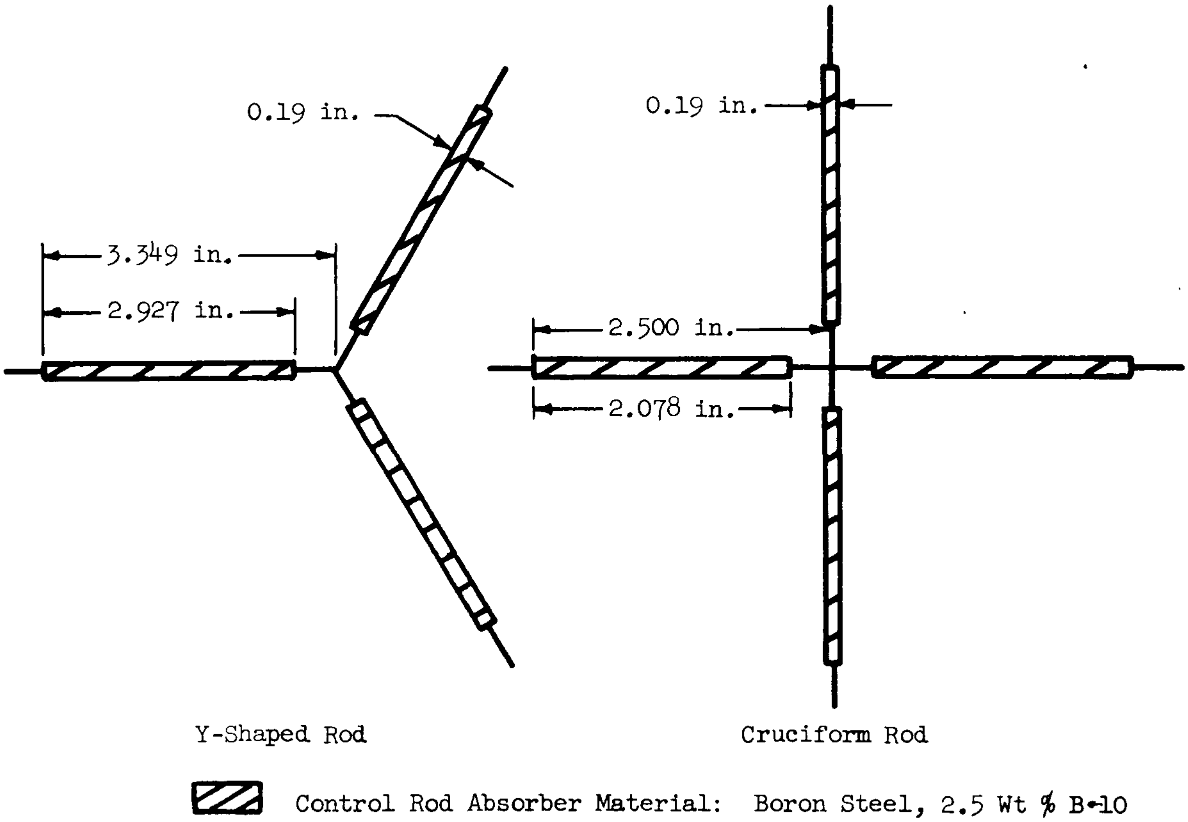
The results obtained, presented in Table III-3, and Figs. III-7 and III-8, give the worth of the seven-rod system as a function of: the radial location of the eccentric ring of six rods; increased arm width; core diameter for a fixed location of the rods; and core height for a fixed location of the rods. Additional studies will be performed, if necessary, as the core configuration becomes more definite.

TABLE III-3

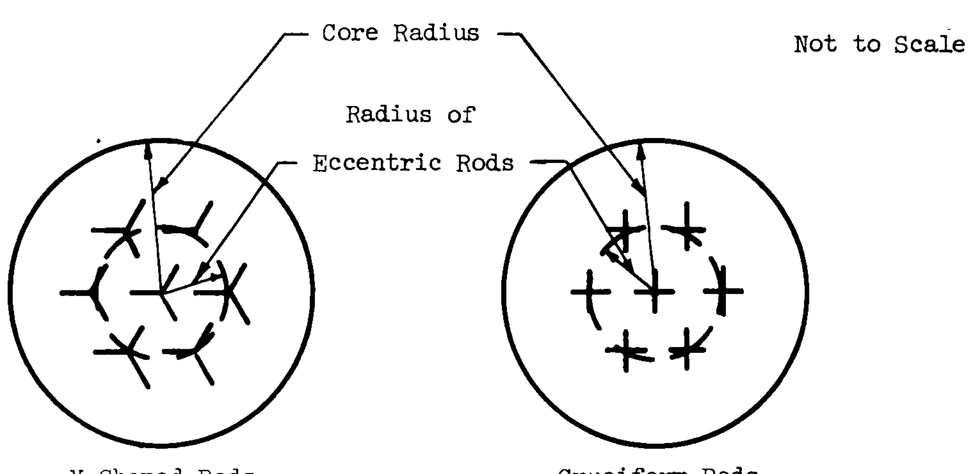
Results of Studying the Worth of a Seven-Rod Bank

Configuration	Core Height (in)	Core Diameter (in)	Radius of Eccentric Ring of Six Rods	Worth (%ΔK/K)
1. Y (Fig. III-7)	23	25	5.0	-15.3
	23	25	6.0	-17.9
	23	25	6.5	-18.9
	23	25	7.5	-17.7
2. Y (Fig. III-7) with rod width = 3.382 in. instead of 2.927 in.	23	25	5.82	-20.5
3. Y (Fig. III-7)	23	20	5.5	-29.6
	23	23	5.5	-20.9
	23	25	5.5	-15.3
4. Cruciform (Fig. III-7)	20	25	6.0	-17.1
	23	25	6.0	-16.9
	26	25	6.0	-16.7

According to Table III-3, and Fig. III-8, maximum rod bank worth occurs when the radial spacing is about 6.75 in.



Control Rod Configuration Showing Absorber Section



Control Rod Location

Fig. III-7. Rod Configuration and Location

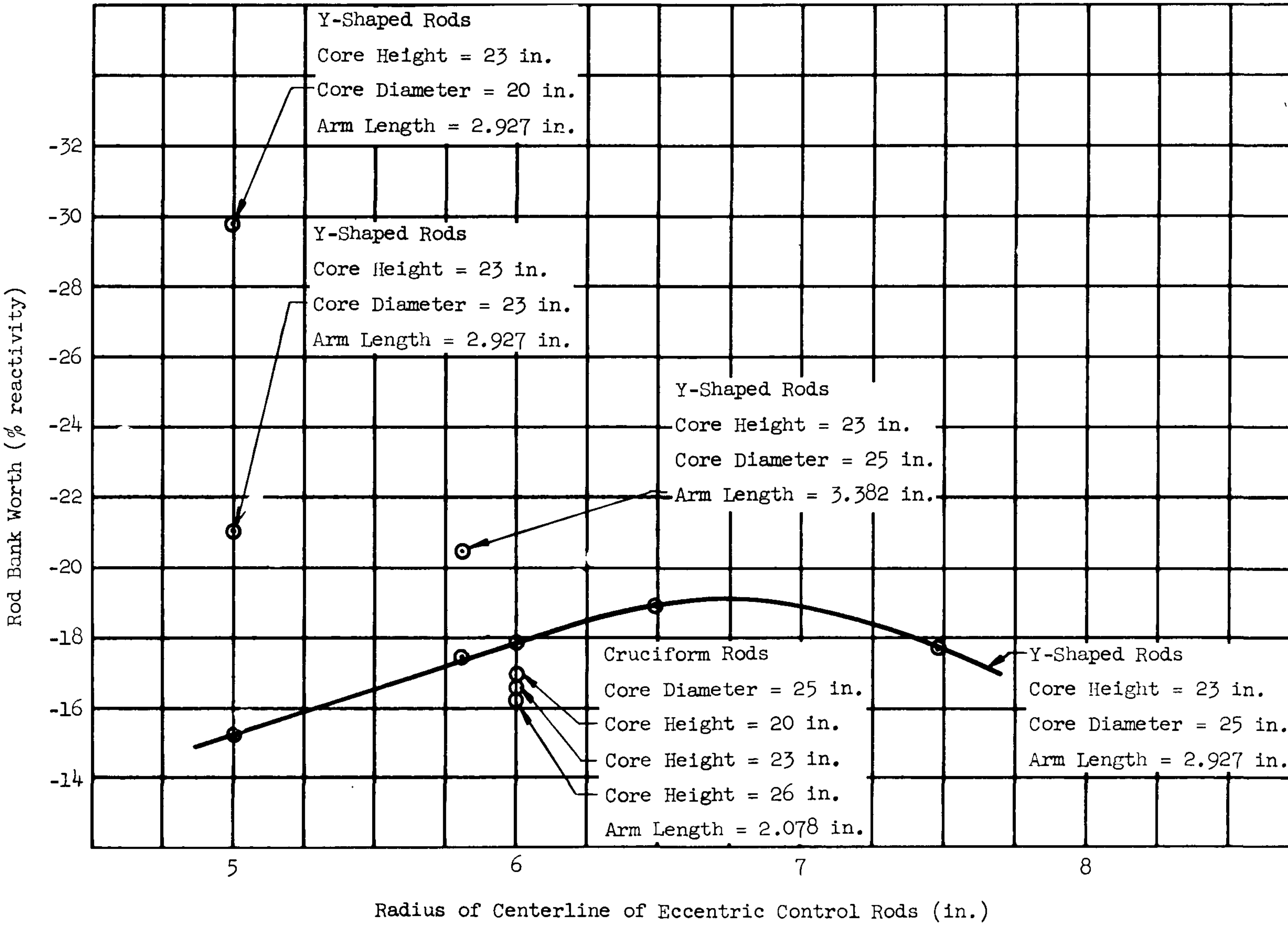


Fig. III-8. Results of Control Rod Studies for a Seven-Rod Bank

It may be seen from 1 and 2, that an increase of 0.455 in. or 15.5% in rod width, resulted in an increase of 3.0% $\Delta K/K$, or 17.2% in the rod bank reactivity worth. Thus, the advantages of increasing rod width are not too great.

Variation in core diameter, Item 3, Table III-3, had a significant effect on rod bank worth.

Core height, as was expected, had no significant effect on rod bank worth.

The relative worth of cruciform rods versus Y-rods is also indicated from case 2 of Items 1 and 4, Table III-3. The cruciform rods with arm widths of 2.078 in. were worth -16.9% as compared to the -17.9% worth of the Y-rods with arm widths of 2.927 in. Since the Y-rods contained 5.6% more absorbing area than the cruciform rods and the difference in rod worth was 5.9%, it appears that both shapes of rod are about equally effective.

Lumped burnable poisons.- The use of burnable poisons, such as boron, to reduce the magnitude of reactivity to be controlled at the beginning of core life is quite common. If reactor core life is relatively short, a homogeneous distribution of burnable poison does not cause excessive peaking of reactivity with time. For reactors having longer core lives, however, the addition of sufficient boron to reduce reactivity to approximately zero during initial operation may result in sufficient increase of reactivity as the poison burns out to pose a serious control problem at some time during core life. The self-shielding effect of lumping some or all of the poison may make it possible to load more poison into the core and still maintain the same minimum of excess K_{eff}. The increased loading could, in turn, result in a shift and a lowering of the plot of peak reactivity against time as hypothesized in Fig. III-9. If the effect on overall core life is negligible, or if the desirability of lowering peak reactivity outweighs the disadvantage of loading a higher fuel inventory, the use of lumped poisons can be highly beneficial.

Studies are in progress to determine the effective multiplication factor, K_{eff}, as a function of time at operating conditions for: a core with no burnable poison; a core with homogeneously distributed poison; a core with homogeneously distributed lumped poisons; and several cores utilizing different initial concentrations of lumped poisons. The initial poison concentrations for all cases are such that the initial K_{eff} is ≈ 1.015.

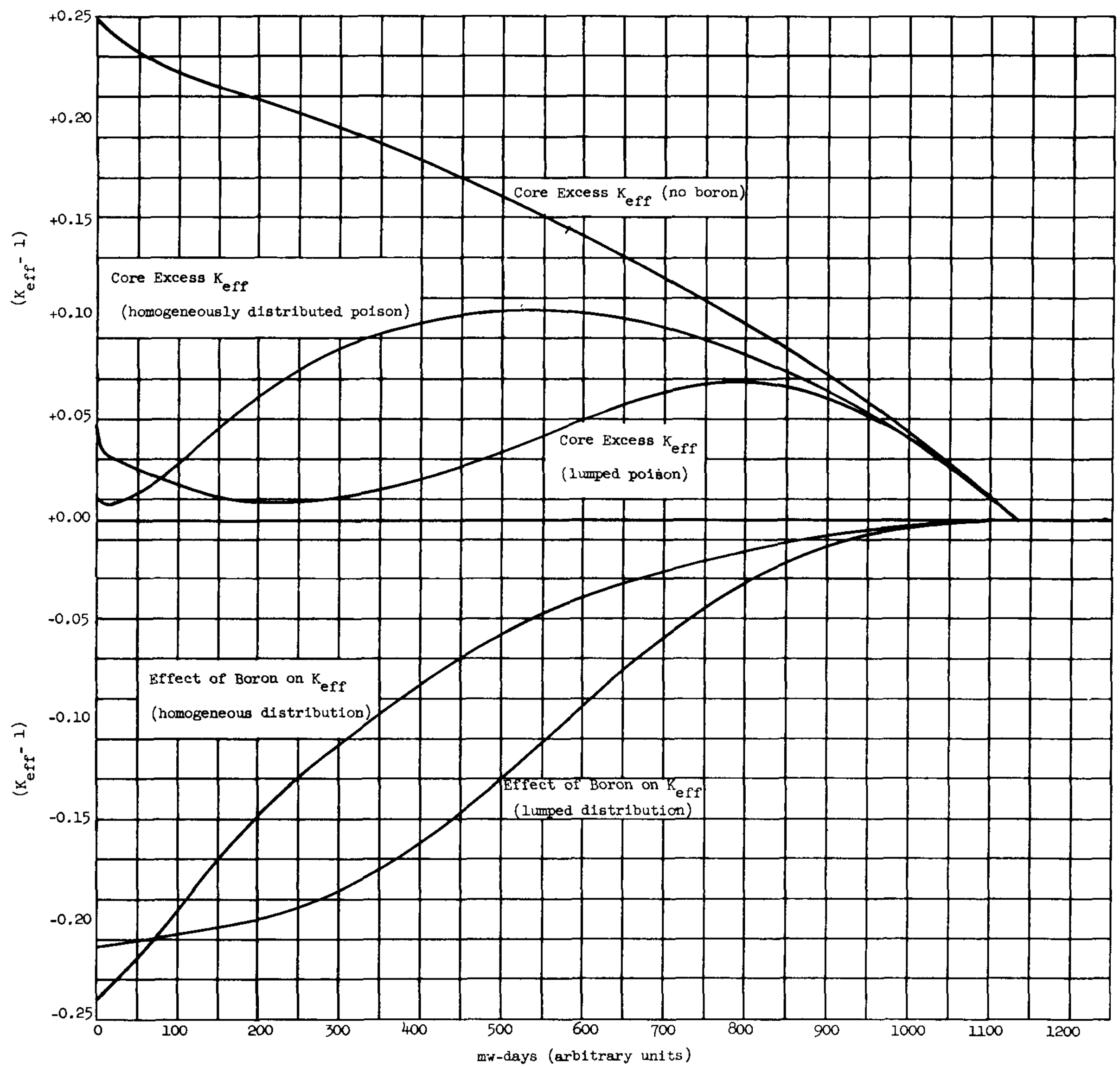


Fig. III-9. Possible Effect of Lumping of Burnable Poison in a Typical Core

The variation of K_{eff} with time will be determined by using the two-group equation:

$$K_{eff} = \frac{\dfrac{\nu\Sigma f_1}{\Sigma a_1}(1-P')}{(1 + L_1^2 B^2)(1 + L_2^2 B^2)} + \frac{\dfrac{\nu\Sigma f_2 P'}{\Sigma a_2}}{(1 + L_2^2 B^2)}$$

where: Σ_{a_1} = fast group macroscopic absorption cross-section

Σ_{a_2} = thermal group macroscopic absorption cross-section

Σ_{f_1} = fast group macroscopic fission cross-section

Σ_{f_2} = thermal group macroscopic fission cross-section

ν = number of neutrons per fission (2.46 for U-235)

$P' = \dfrac{\Sigma_{s_1}}{\Sigma_{s_1} + \Sigma_{a_1}}$ where Σ_{s_1} = fast group slowing down probability per unit path length

L_1^2 = fast group diffusion area

L_2^2 = thermal group diffusion area

B^2 = total geometric buckling

and: $\Sigma = \sum_{i=1}^{i=m} N^i \sigma^i$ with N^i = atomic density for material i

σ^i = microscopic cross-section for material i

The time dependent quantities are N^{U-235} in $\Sigma_f = N^{U-235} \sigma_f^{U-235}$, N^{U-235} and N^{B-10} in $\Sigma_a = \sum N^i \sigma_a^i$, and g(t) the thermal self-shielding factor. Other quantities are assumed to remain constant, but will be included later if the effects of spectral energy shifts are found to be significant. The time dependent atomic densities are being calculated from the equation.

$$N(t) = N(t-1) - N(t-1) \Delta \, (\sigma_{a_1} \phi_1 + g(t-1) \sigma_{a_2} \phi_2)$$

where: Δ = time interval, sec

ϕ = group flux

The time dependent self-shielding factor g(t) is essentially a function of atomic density, i.e., g(t) = g(N). This function is obtained by calculating the ratio of the flux in the poison to the unperturbed flux using multiregion one-dimensional diffusion theory. Several check points are available from experimental data obtained under Subtask 2.1 where the reactivity effects of lumped poisons were determined for three different concentrations of lumped poisons. These data will be related to self-shielding factors using perturbation theory.

Heat transfer studies.-

R. Baer | A. Carnesale | S. Frank

The heat transfer studies performed during the past quarter had three general objectives: first, to obtain design information on nucleate boiling and pressurized water cores; second, to prepare a detailed digital computer code pertaining to heat transfer characteristics of both boiling and nonboiling cores; and third, to investigate the effect of nucleate boiling on coolant flow distribution.

During the next quarter, heat transfer analysis required in support of the preliminary design will be performed. If the preliminary design utilizes a core having local boiling, a more detailed investigation of the heat transfer and hydrodynamic performances of nucleate boiling cores will be made.

Nonboiling pressurized water core study.- The pressure required to prevent boiling in a core is a function of water saturation temperature which, in turn, is a function of the maximum fuel element surface temperature occurring in the core. Therefore, an analytical study was made to determine the range of maximum surface temperature in a series of cores defined by various parameters.

Since reactor power varies only slightly for various secondary loop configurations, reactor power was held constant at 10 mw thermal throughout the study.

Core diameters of 22.5 and 25 in. were studied. The smaller of these diameters represents about the smallest feasible core size for a pressurized water reactor producing 10 mw. The larger core was selected to give a measure of the variation of power plant cost and weight with core diameter.

Heat transfer performance of the core improves as the ratio of the core length to core diameter increases. However, the critical mass of the core increases as the value of this ratio exceeds unity. A length-to-diameter ratio of 1.25 was selected as a compromise between these two conditions.

Fuel element inside diameters of 0.25 to 1.0 in. were covered in the study. This range is sufficient to include all feasible cores for the range of other parameters studied. The fuel element thickness was held constant at 0.030 in. for purposes of this study since the maximum fuel element surface temperature is essentially unaffected by element thickness.

Primary coolant flow rate was varied from 1400 to 2600 gpm. This represents the maximum range of flow rates believed to be feasible after giving due consideration to the pump costs, weights, and power requirements, defined by primary loop pressure drop--primarily a function of flow rate.

Since thermal performance of the core increases with heat transfer surface area per unit core volume, it would be desirable to minimize tube spacing. Fabrication considerations, however, limit the ratio of tube pitch to tube outside diameter to a minimum of 1.2; this ratio was held constant throughout the investigation.

Only two-pass cores were included in this study because previous investigations of one- and two-pass cores have demonstrated conclusively that they are thermodynamically superior. The major advantages of a two-pass system are inherent power flattening in each pass and increased fluid velocity for a given flow rate. For a fixed reactor power output, a reduction in primary loop flow rate of the order of 50% is attainable over a single-pass system.

All steady-state core thermal calculations were made using an IBM-704 digital computer. The assumptions used in the analysis are:

(1) Coolant flow channels are well defined.

(2) Coolant mixing may be neglected.

(3) Axial heat conduction in the fuel elements and coolant may be neglected.

(4) Thermal properties of the fuel element and coolant may be assumed to be constant.

(5) Perfect bonding exists between the meat and the cladding.

(6) Axial symmetry exists over the active length of the fuel element.

(7) A heat transfer film coefficient, representing the resistance to heat transfer between the fuel element surface and the coolant, may be predicted.

Since the nuclear and heat transfer studies were performed concurrently, reactor flux and power data had to be based on cores believed similar to the eventual PM-1 core. Data used are as follows:

(1) Peak-to-average flux ratio in first pass (including local flux perturbations) --2.24

(2) Peak-to-average flux ratio in second pass (including local flux perturbations) --1.87

(3) Fraction of total power produced in first pass --0.50

(4) Fraction of total power produced in second pass --0.50

Hot spot factors used in the analysis are presented in Table III-4.

The results of the study were plotted in terms of maximum element surface temperature minus the coolant inlet temperature. A sample set of these results is shown in Fig. III-10.

TABLE III-4

Hot Spot Factors

	Variation (%)						
	OB(1)	IB(2)	$F_{b_{OB}}$(3)	$F_{\varphi_{OB}}$(4)	$F_{b_{IB}}$	$F_{\varphi_{IB}}$	F_q(5)
Plenum Chamber Flow Variation	12	7	1.137	1.117	1.078	1.081	-
Velocity Variation due to Channel Dimension Uncertainty	1	1	1.040	1.024	1.024	1.036	-
Velocity Variation due to End Spacer Dimension Uncertainty	10	10	1.067	1.043	1.072	1.044	-
Variation in Meat Thickness	3	3	1.030	1.030	1.030	1.030	-
Fuel Concentration	2	2	1.020	1.020	1.020	1.020	-
Inability to Predict Heat Transfer Film Coefficient	20	20	-	1.294	-	1.300	-
Inability to Predict Neutron Flux Distribution	10	10	-	-	-	-	1.10
Uncertainty in Power Requirements	10	10	-	-	-	-	1.10
TOTAL			1.326	1.622	1.243	1.600	1.210

(1) Outside Baffle

(2) Inside Baffle

(3) Hot Spot Factor to Account for Variations in Bulk Coolant Temperature Rise

(4) Hot Spot Factor to Account for Variations in the Temperature Drop Across the Film

(5) Hot Spot Factor to Account for Variations in Power

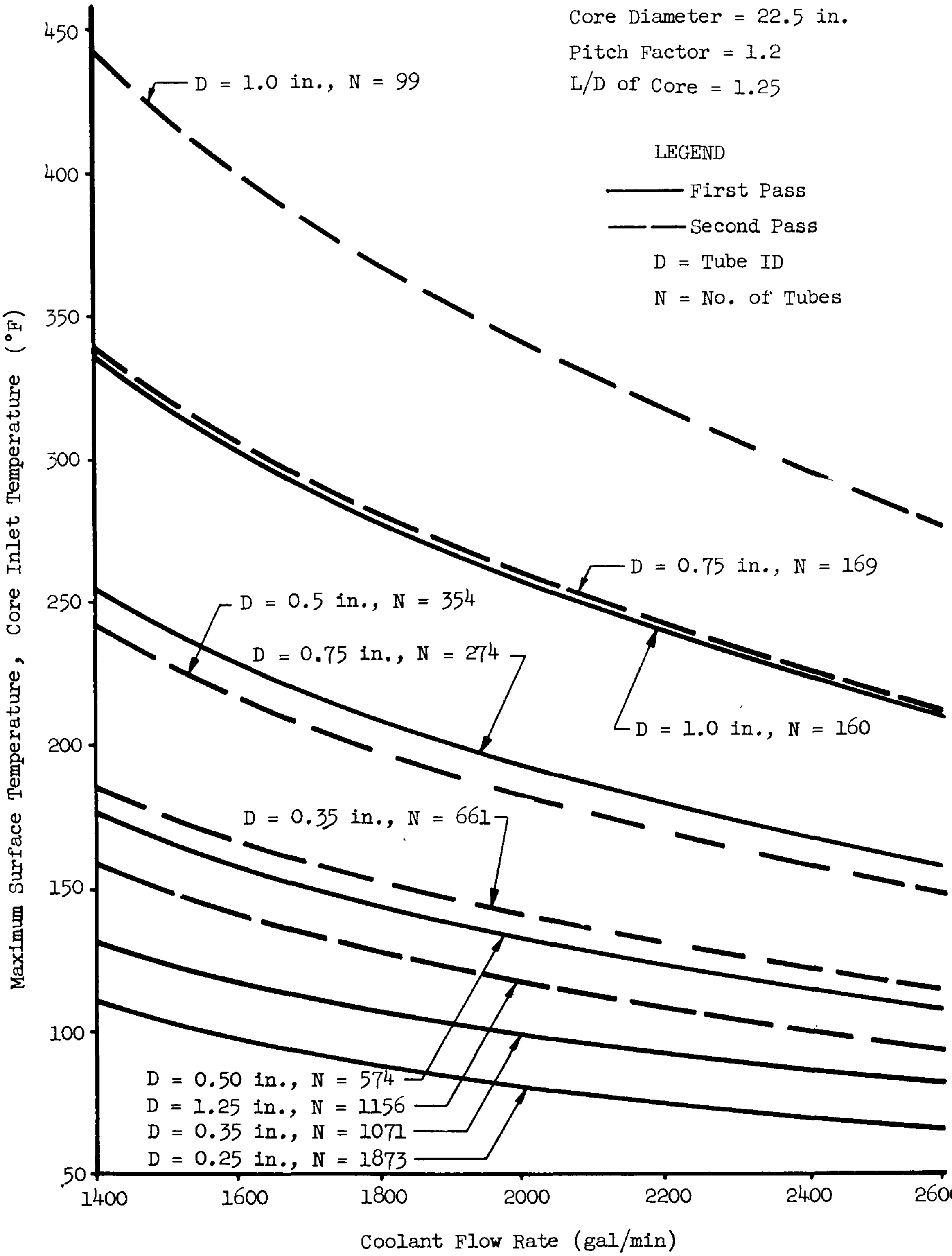


Fig. III-10. PM-1 Nonboiling Parametric Study

To use these curves:

(1) A pressure is specified, thereby determining the maximum allowable element surface temperature.

(2) A flow rate is specified, thereby determining, for a given power level, the temperature rise across the core.

(3) A mean temperature is specified, making calculation of the core inlet temperature possible.

The ordinate and abscissa may now be determined and the sizes and number of tubes in each pass may be found from the graph.

Local boiling cores--A study was performed to determine the maximum ratio of actual heat flux to burnout heat flux in the cores defined by the range of parameters.

The parameters studied were quite similar to those considered for pressurized water cores. The exceptions were that pressures within the design range of interest (900 to 2000 psia) were added as independent variables; core diameters as small as 20-in. were investigated, since smaller core diameters are feasible with nucleate boiling; and the range of the ratio of tube pitch to tube outside diameter was extended to 1.5.

Only single-pass cores were included in the study. Since possible bulk boiling in the first pass of a two-pass core would cause severe design problems and since the only advantage of a two-pass core is its nherent power flattening, the benefits gained from a two-pass design did not seem worthwhile.

An analytical code developed for the IBM-704 was employed in this study. The basic assumptions used in the analysis were:

(1) The radial peak-to-average power distribution ratio is two.

(2) The axial power distribution may be represented as a chopped cosine function with an extrapolated length of 1.3 times the active fuel element length.

(3) The maximum fraction of burnout heat flux occurs at a point 65% of the way up the core.

(4) The velocities inside and outside of the tubes are equal.

(5) The heat fluxes inside and outside of the tubes are equal.

(6) A 50% power increase will cause incipient bulk boiling at the exit of the hot tube.

(7) Twenty-five percent of the coolant flows through vacant control rod channels.

(8) Coolant flow channels are well defined.

(9) Coolant mixing may be neglected.

(10) Axial heat conduction may be neglected.

(11) Thermal properties may be assumed constant.

This code was used to determine the inlet, exit, and mean coolant temperatures; the point at which local boiling begins; and the burnout heat flux.

It was found that, for the range of parameters investigated, the highest percent of burnout heat flux obtained was 29%. In most cases the percentage was between 5 and 15%. Since burnout heat flux is the only essential geometry-dependent thermal performance parameter, it appears that the thermal performances of the local boiling cores will be acceptable (in the regions of interest) regardless of core configuration. In other words, the mean coolant temperature attainable is a function of the coolant flow rate and the saturation pressure and is not restricted by fuel tube heat transfer considerations. These general relationships are shown graphically in Fig. III-11.

Nucleate boiling code--A thermal and hydraulic analysis of plate, rod, or tubular fuel elements operating under nonboiling, local boiling, or bulk boiling conditions has been programmed for the IBM-704 (or 709). The assumptions inherent in this analysis are similar to those of the nonboiling code with the exception that coolant properties may be temperature dependent. The axial and radial temperature distributions of the fuel element are determined, as are the axial distributions of coolant temperature, pressure, velocity, quality, void fraction, and enthalpy. The axial distributions of heat flux, local boiling, heat flux, burnout heat flux, saturation temperature, and film coefficient are also determined.

At this time, the code (BUBBLES-1) is in the process of being checked out.

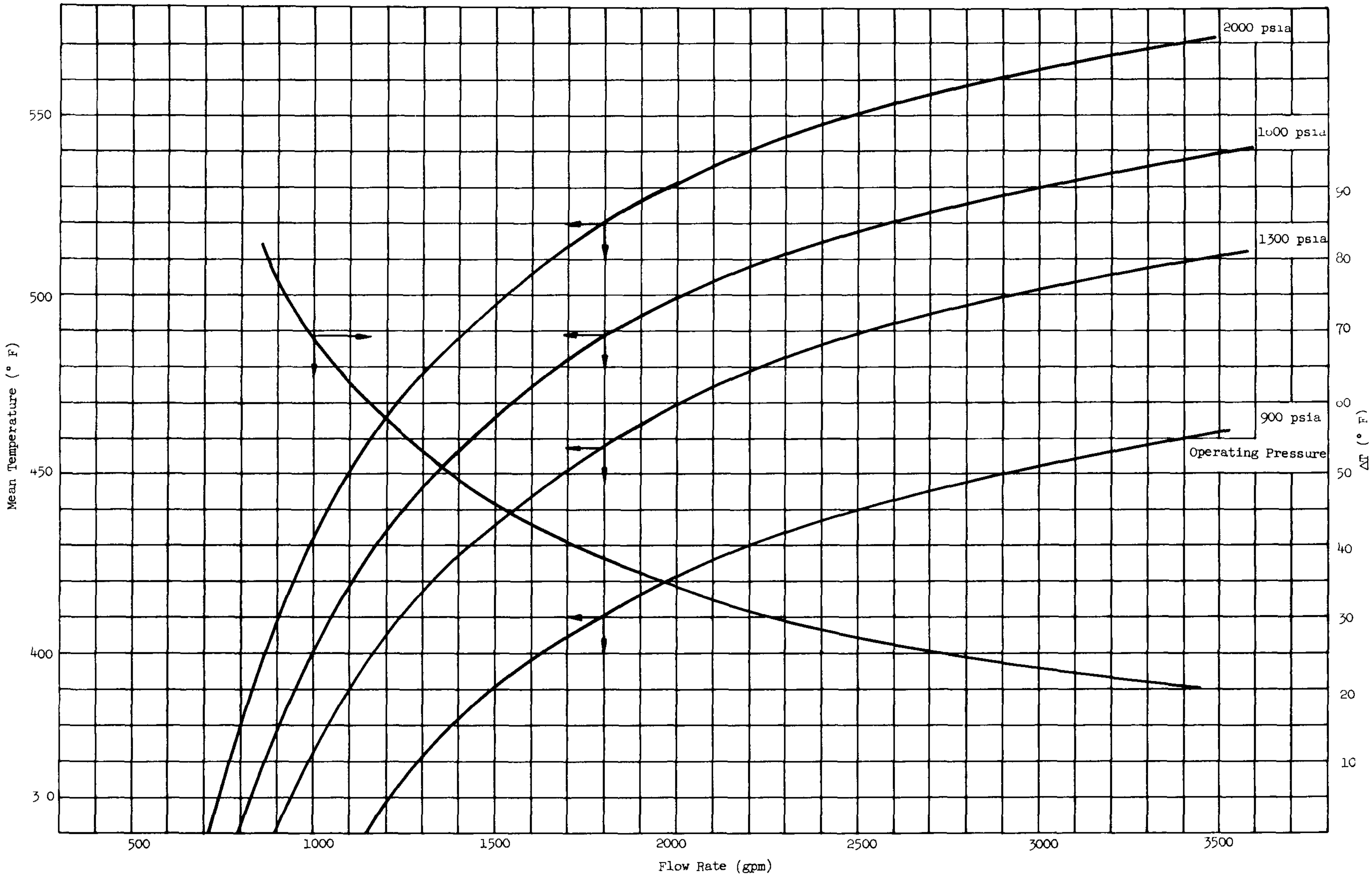


Fig. III-11. PM-1 Nucleate Boiling Core--9.5 mw Thermal Reactor Power

Flow distribution with nucleate boiling--An analysis was started to determine the effect of local boiling on flow distribution within the core. Preliminary results were obtained concerning the maximum variations in reactor coolant flow.

The extremes in the possible flow pattern outside of the elements would occur when the resistance to flow normal to the elements is either zero or infinite.

When the resistance is zero, calculations show a reduction in velocity of 23% in the channel having local boiling. Preliminary calculations indicate that, for this decrease in velocity, the burnout (point at which transition from nucleate to film boiling occurs) heat flux is lowered by 14% at the location where the ratio of heat flux to burnout heat flux is a maximum.

When the resistance is infinite, the coolant is restricted to a single flow path (similar to the coolant flow inside of the elements) and orificing may be used to control flow. The orifice pressure drop is a function of flow rate only, and not of the heat transfer characteristics of the fuel elements. Hence, orifices will tend to stabilize the flow. For example, if the orifice pressure drop is set at four times the friction pressure drop which would occur without boiling, local boiling would result in only a 6% decrease in coolant velocity and a negligible drop in burnout heat flux.

It was concluded, therefore, that major changes in flow distribution can be avoided by the proper choice of core design, flow rates, and temperature levels.

Shielding studies.-

E. Devita D. Owings

The relatively high fluxes present in a compact 10-mw core produce intense sources of neutron and gamma radiation within the primary system. The objectives of this phase of Subtask 3.1 were to determine:

(1) Methods of providing adequate primary and secondary shielding.

(2) The effect of varying pressure vessel and thermal shield diameter on pressure vessel gamma heating.

(3) The extent of primary loop coolant activation.

The objectives of the shielding study were to determine methods of providing primary and secondary biological shielding, to estimate shield requirements, and to examine specific shielding problems encountered during primary system relocation.

A preliminary investigation of several general shield configurations provided a background for analysis of the specific designs presented. Placement of the primary packages (s) was studied, as was the utilization of various types of air transportable or locally available shield materials. Several designs were evolved which featured placement of the reactor and steam generator packages above, below, and partially below the ground surface. These configurations utilized water, lead, iron, plastic, and earth shielding. The more interesting concepts were incorporated into the specific designs of Figs. III-12 to III-15. It should be noted that the various shielding configurations have been selected, in many cases, to define extreme dose rates and do not represent a suggested configuration or method of operation. An example of this is the use of four feet of shielding water during operation as shown in Configuration 5 (Fig. III-15).

During the next quarter, these objectives will be met, and the determinations will form the basis for preliminary design.

The following provides a description of the configurations.

(1) The following is applicable to all configurations:

(1) Air scattered neutron and gamma radiation external to the packages is negligible.

(2) The radiation from the primary coolant within the piping and the pump will cause high local dose rates, approximately equal to those from the steam generator.

(3) Unless otherwise noted, dose rates were obtained under operating conditions.

(4) Shield water tanks and lead shielding may be designed to be completely air transportable.

(5) Shield optimization was not attempted due to present uncertainties concerning radiation source strengths. Considerable weight reduction may be realized when core, reactor vessel, and steam generator geometries are better defined.

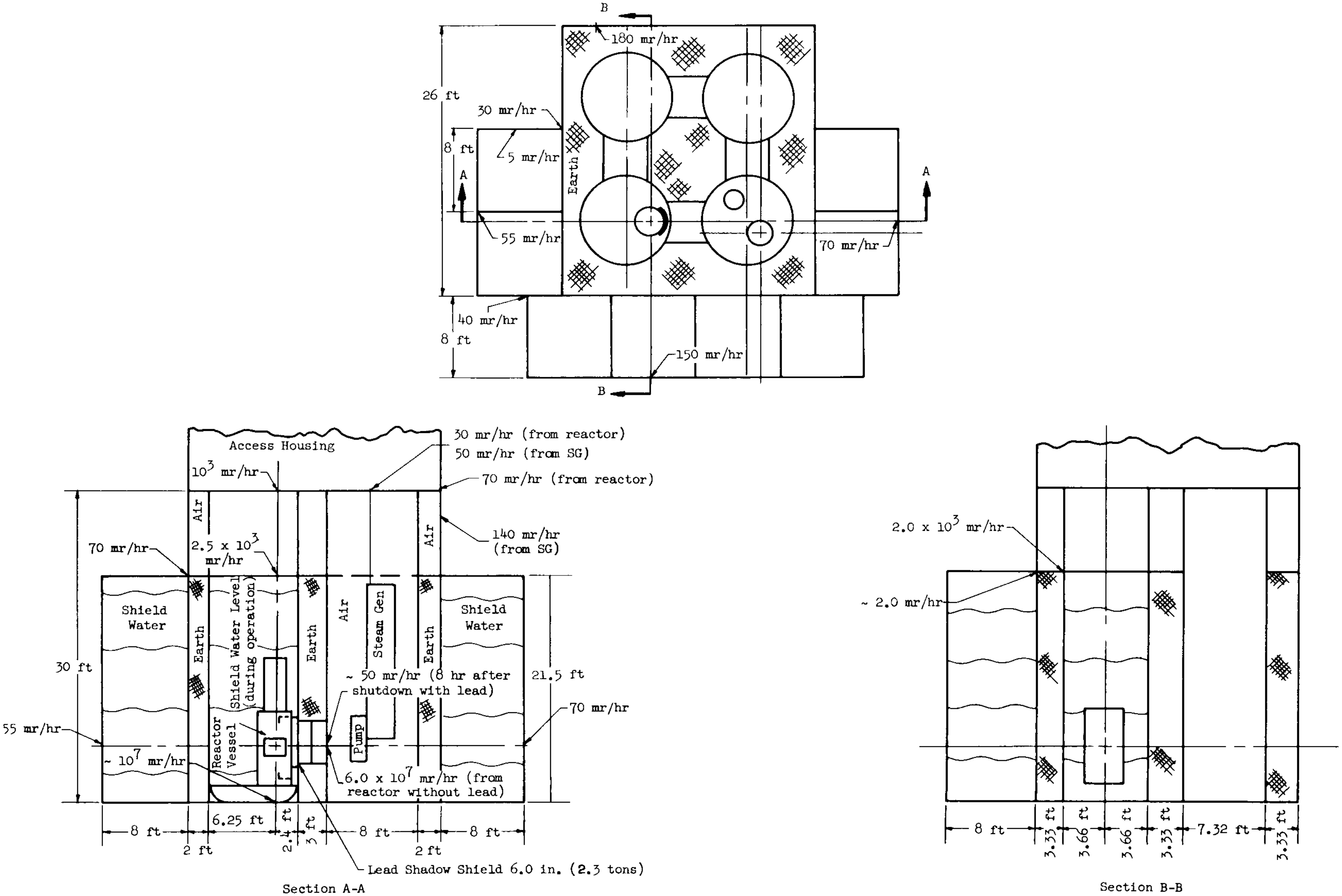


Fig. III-12. Preliminary Shield Evaluation--Configuration 1

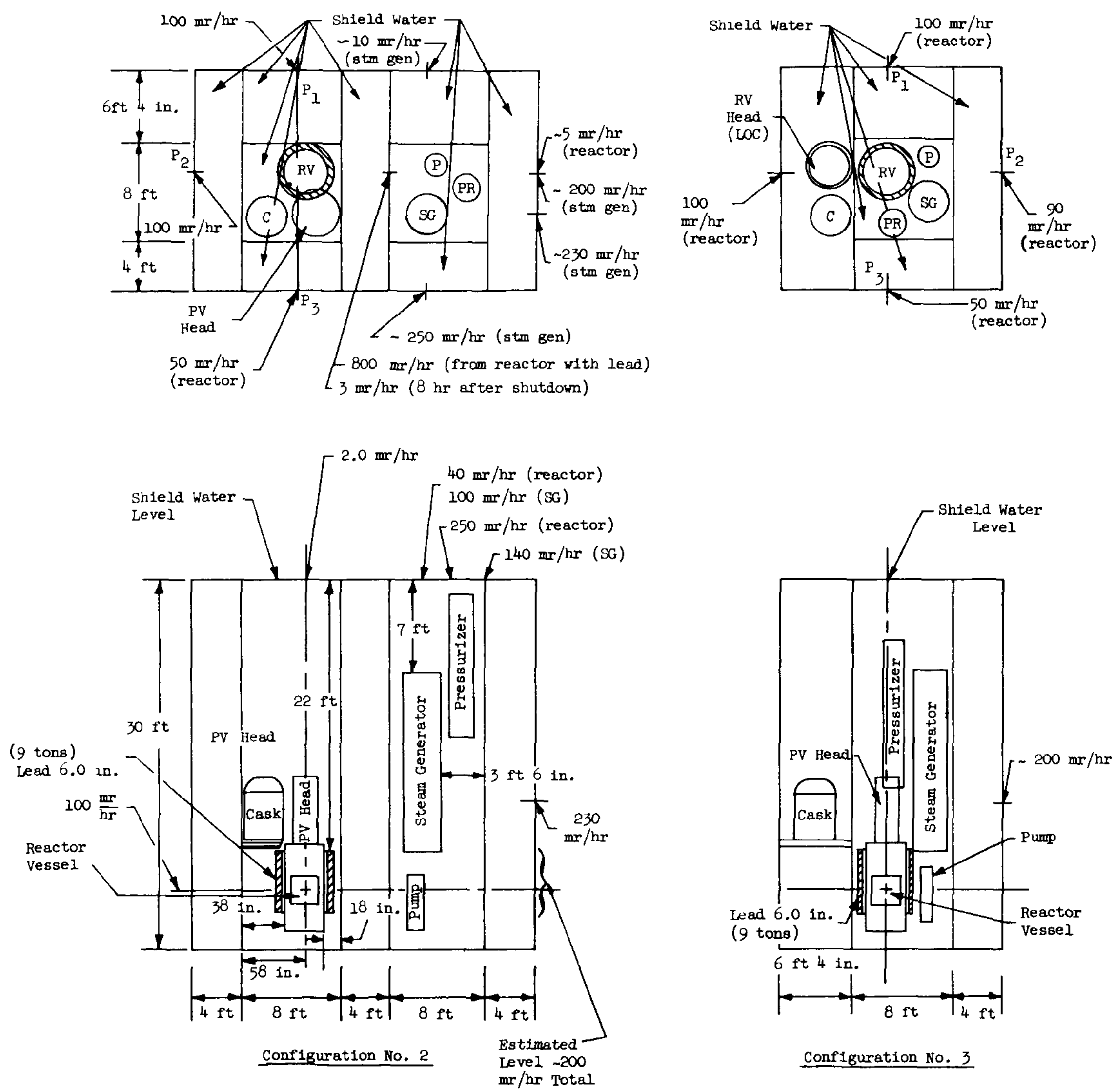
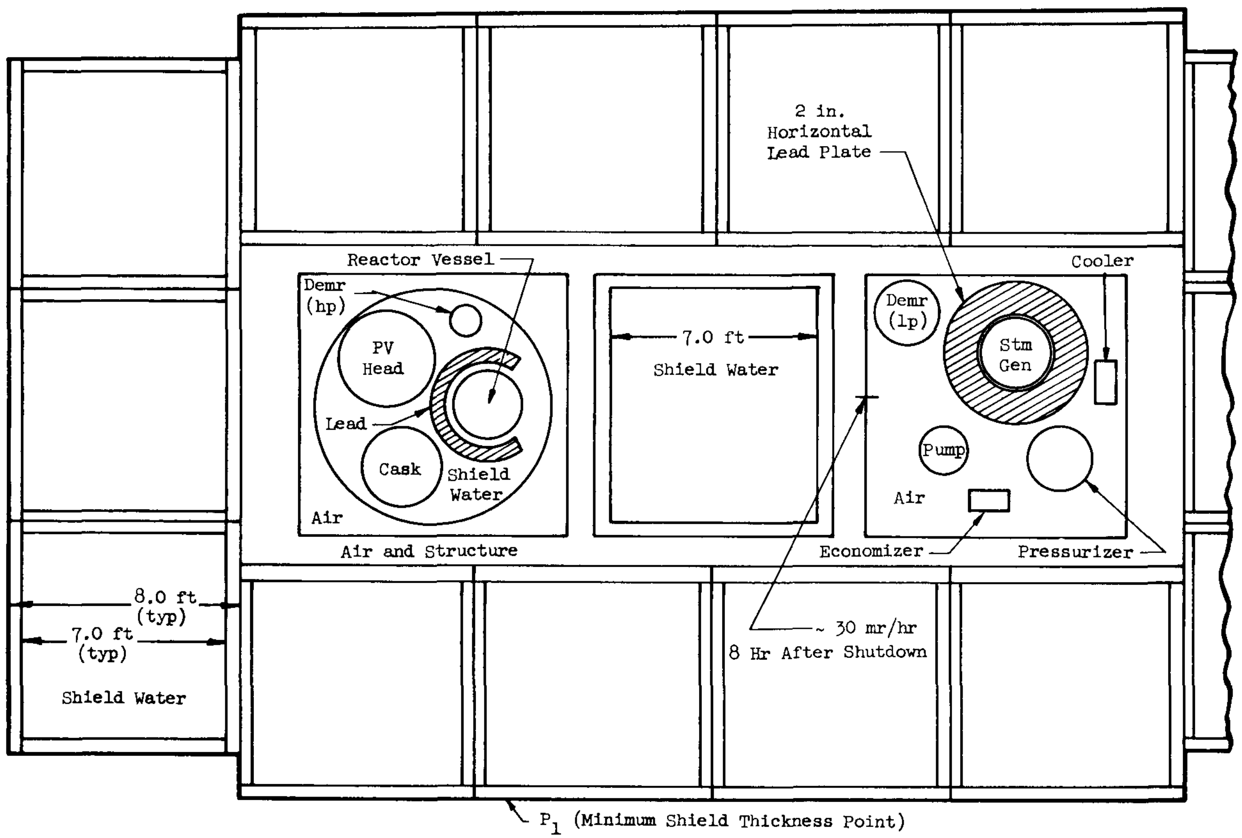


Fig. III-13. Preliminary Shield Evaluation--Configurations 2 and 3



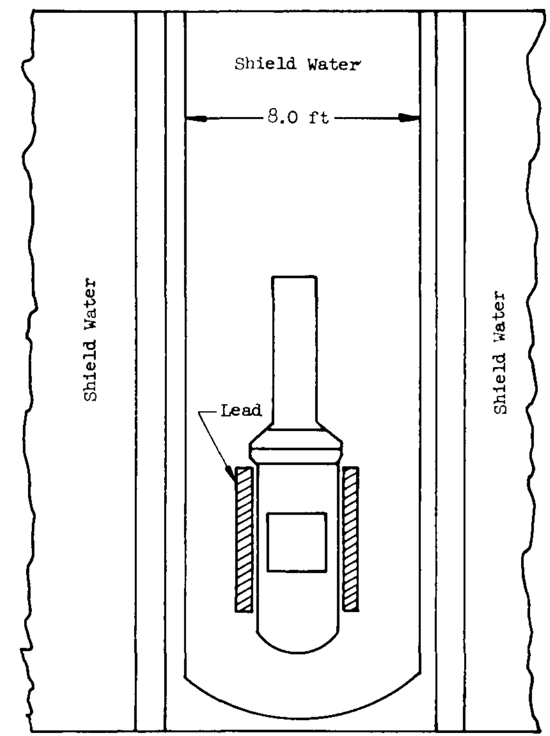


Fig. III-14. Shield Evaluation--Design Configuration No. 4

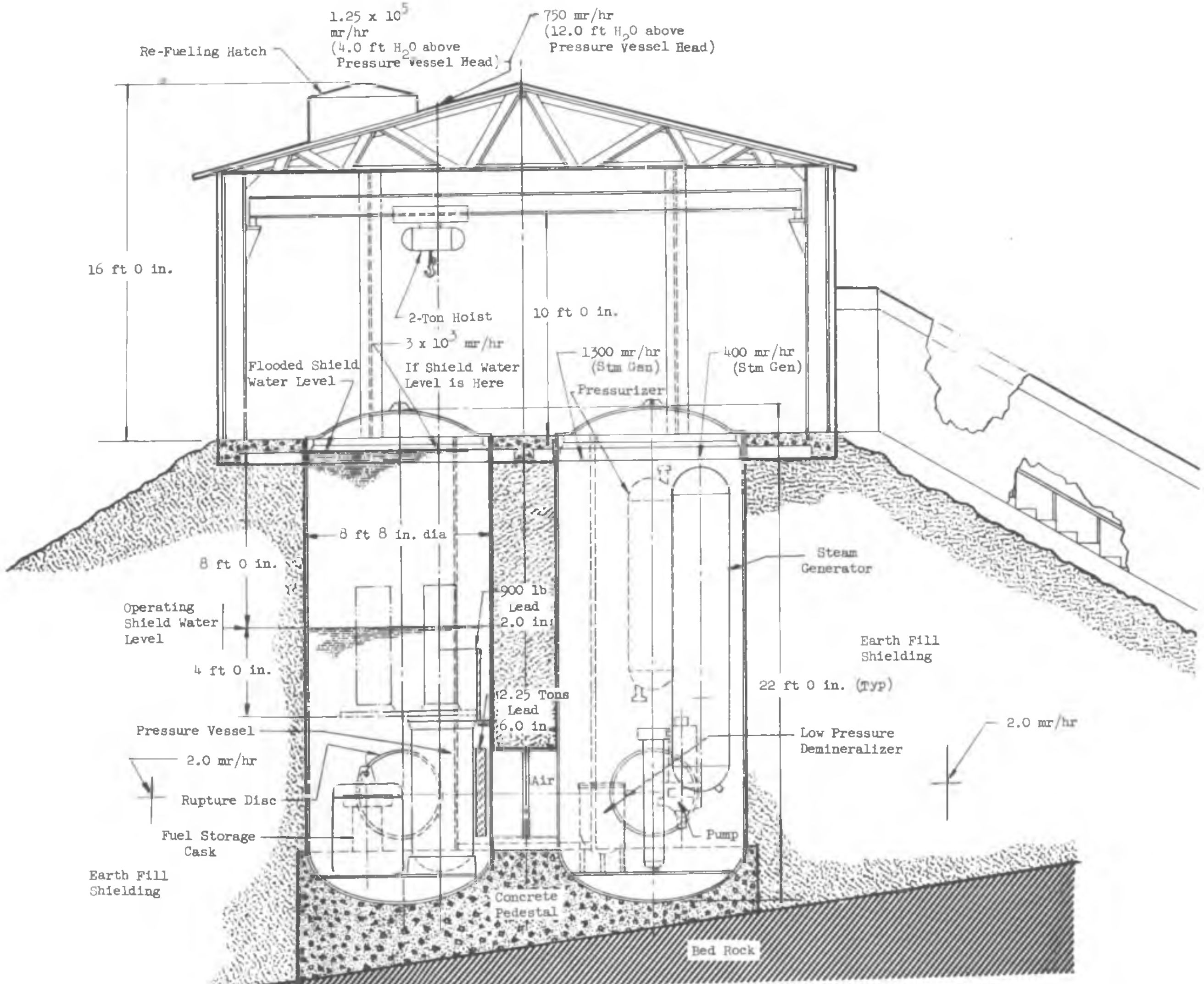


Fig. III-15. Preliminary Shield Evaluation--Design Configuration No. 5

(2) Figure III-12, Configuration 1

(1) The estimated volume of earth is 10,000 cu ft.

(2) The shield water tanks could be eliminated by using a 6-in. by 5 ft cylindrical lead shield around the reactor vessel, and 1.5 ft of shield water (or equivalent plastic) placed externally and adjacent to the square earth container.

(3) The shield water tanks could be replaced by contained earth in 5 ft x 8 ft x 21.5 ft tanks. The estimated additional volume of earth is 15,500 cu ft.

(3) Figure III-13, Configurations 2 and 3

(1) A 2-in. lead shell placed around the activated region of the steam generator would reduce the containment surface dose rates stemming from the steam generator by a factor of 10.

(2) The shield water tanks around the reactor package could be replaced at the following points by contained earth in tanks of the following thicknesses:

P_1	80 in.
P_2	52 in.
P_3	52 in.

With the earth shielding, only a 3-in. cylindrical lead shield around the reactor vessel is required. The gamma dose rates at the outer surface of the containment are less than 100 mr/hr. The fast neutron dose rates at the outer surface of the containment are less than 25 mr/hr. The shield water tanks adjacent to the steam generator package of Configuration 2, and the refueling tank adjacent to the reactor package of Configuration 3, would be retained.

(3) The estimated fast neutron flux in the region of the primary coolant pump in Configuration 3 is 5×10^9 neutrons/cm^2 sec.

(4) Maximum dose rates on the containment surfaces are of the order of 10^5 mr/hr, if lead reactor shielding is eliminated.

(4) Figure III-14, Configuration 4

(1) A 2-in. lead plate over the steam generator would allow access during operation to regions above the steam generator.

(2) Thicknesses of lead shielding placed around the reactor vessel and corresponding dose rates estimated at P_1 are as follows:

7 in.	2 mr/hr
6 in.	10 mr/hr
4 in.	100 mr/hr

The weight of the 6-in. lead shield would be approximately 9 tons.

(5) Figure III-15, Configuration 5

(1) The 2-in. lead shield is 36 in. high and shields personnel in the housing area above the containment after shutdown; the 6-in. lead shield is 58 in. high and shields maintenance personnel in the steam generator package after shutdown.

(2) The maximum dose rate 8 hr after shutdown at any point within the housing area is estimated to be less than 50 mr/hr.

(3) Earth shielding for this configuration has an estimated volume of 50,000 cu ft.

(4) A 2-in. lead shield placed around the active region of the steam generator would reduce the containment surface dose rate from the steam generator by a factor of 10.

Shield size may be reduced by forbidding or limiting access to certain areas adjacent to the primary packages during reactor operation. For the most part, high radiation levels are present only in those areas requiring little or no personnel access. An exception to this occurs in the single package design of Configuration 3, Fig. III-14, where excessive dose rates from fission products and the activated pressure vessel occur after shutdown in the vicinity of the steam generator. The lead shielding required to reduce these dose rates to tolerable values is excessive. The primary loop pump is also located in a region of fast neutron flux high enough to damage certain types of motor insulation. As a result of this, the single package concept is considered unacceptable from a shielding standpoint. The following were true for all designs other than the single package:

(1) Access is possible to the steam generator and coolant pump eight hours after shutdown.

(2) Adequate shielding against radiation eminating from the core, the pressure vessel, and activated impurities in the primary loop piping is provided during operation.

(3) Neutron activation of nearby earth or subterranean water is prevented.

(4) Sufficient shielding is provided to allow safe refueling.

Preliminary data for shielding a vertical steam generator were developed. Primary coolant activation was assumed to be the same as in the PM-1 reference design. Shielding requirements along the radial centerline were found to be approximately the same as in the horizontal version. However, appreciably lower dose rates occur along the axial centerline because of the presence of additional material, shielding effects of the secondary water, and due to the large steam void area above the radioactive portion of the steam generator. The vertical design is, therefore, considered more acceptable from a shielding standpoint than the horizontal design.

A variety of shields were evaluated for the active regions of the pump and the primary coolant piping. It was found that, in more cases, the coolant piping will be so positioned that a minimum of local shielding is required.

Computed dose rates and shield thickness are correct within a factor of 10. Uncertainty stems mainly from one or both of the following calculational techniques.

(1) A single average energy and source strength was used to estimate the dose rates in various media.

(2) A dose rate ratio approximation was used to obtain the attenuation through several different layers of shield media. For example, the dose rate beyond a combination of lead

thickness, t_1, followed by water of thickness, t_w, is obtained by multiplying the dose rate in pure water at a distance $t_1 + t_w$ by the ratio of the dose rate at t_1 in pure lead to the dose rate at t_w in pure water.

The use of infinite media buildup factors in finite regions and the fact that source strengths in general were assumed constant over the source and to be of maximum strength, results in conservative estimates.

Gamma heating of the pressure vessel.- Previous efforts to determine the effects of thermal stresses created by gamma heating and the effects of integrated neutron flux on the reactor pressure vessel have demonstrated that variations of water gap and of thermal shield thickness have a considerable bearing on such effects. During the first quarter, studies of reactor vessel and thermal shield spacing and thickness were begun.

In view of the wide variation of design parameters being considered, it was considered most meaningful to present gamma heating rate and fast neutron flux data for various thermal shield thicknesses and pressure vessel inner diameters. Heating rates within the reactor vessel are a maximum along the core radial centerline and may be expressed as

$$Q(x) = Q_o e^{-\beta x} \qquad (1)$$

where: $Q(x)$ = gamma heating rate (Btu/in.^{3} hr) occurring in the reactor pressure vessel (along the extended core radial centerline) at a distance x from the vessel inner surface

Q_o = heating rate (Btu/in.^{3} hr) at the vessel inner surface

β = attenuation coefficient (in.$^{-1}$)

Graphs of Q_o and β for gaps, measured along the core radial centerline between the core outer surface and pressure vessel inner surface, of from 4.0 to 28 in., and for thermal shield thicknesses to 1.5 in. are given in Figs. III-16 to III-19. Integrated fast neutron flux at the inner surface of the pressure vessel after a year of operation at 10 mw may be determined approximately from Fig. III-20.

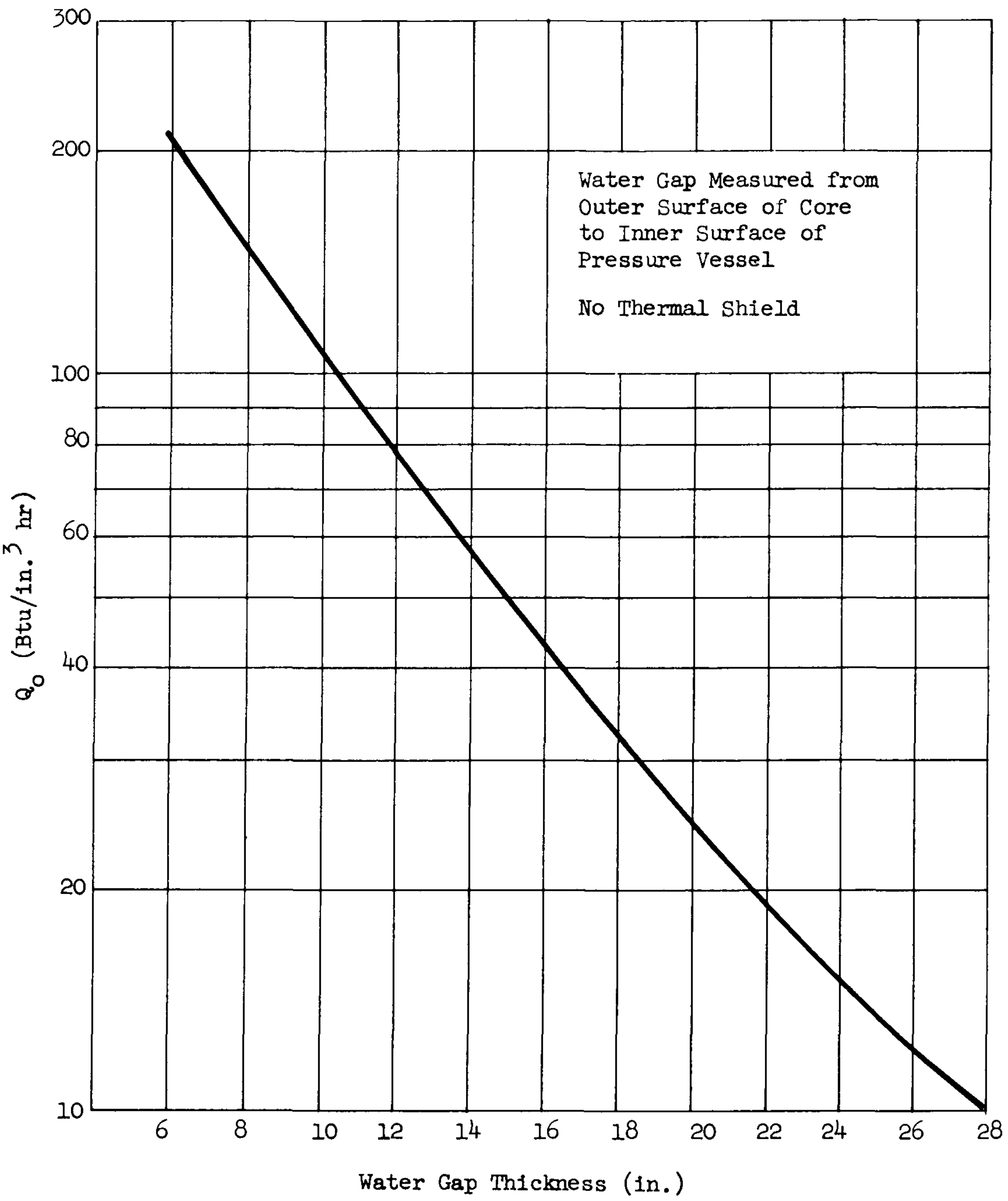


Fig. III-16. Pressure Vessel Gamma Heating

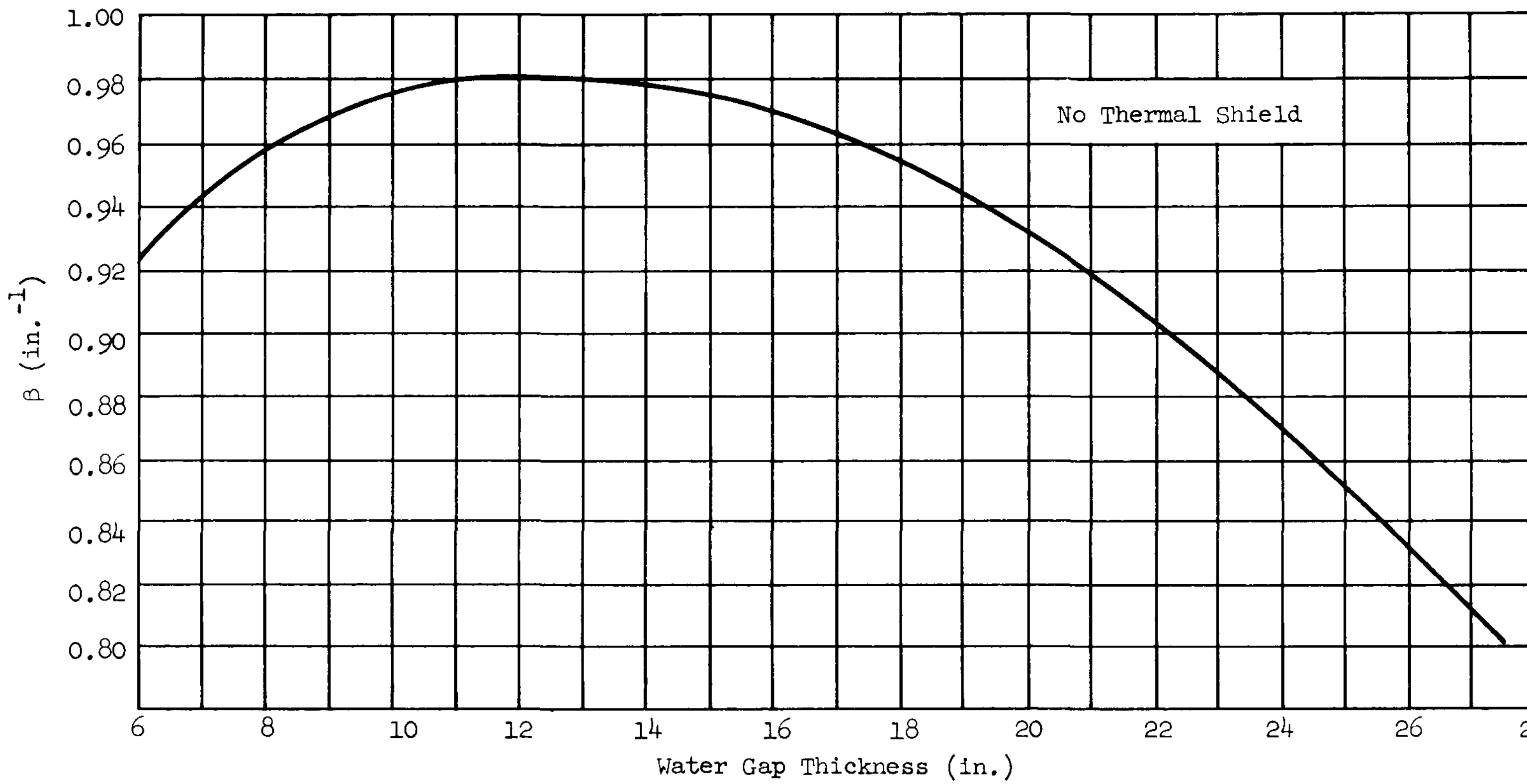


Fig. III-17. Pressure Vessel Gamma Heating

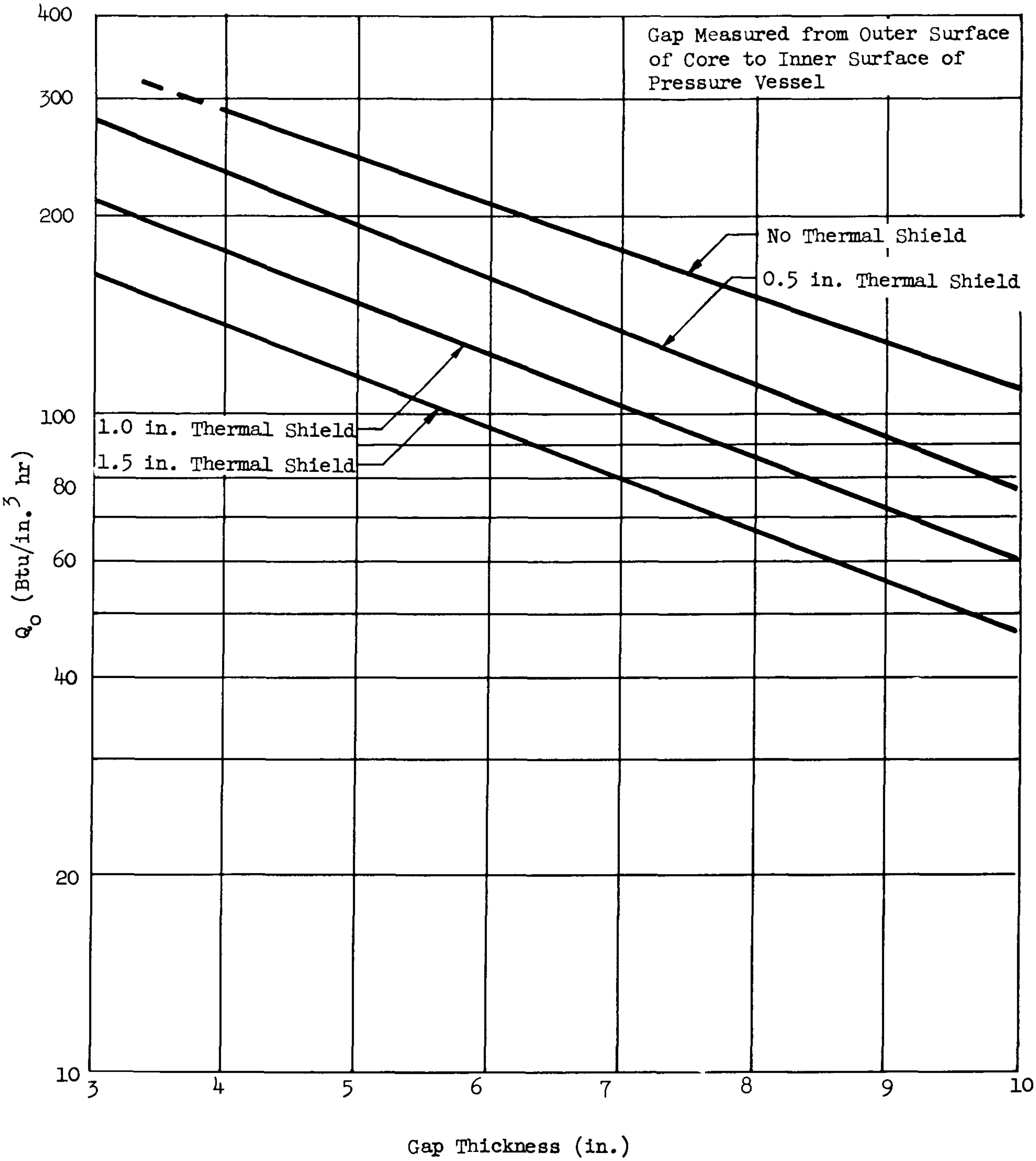


Fig. III-18. Pressure Vessel Gamma Heating--Q_o Versus Gap Thickness

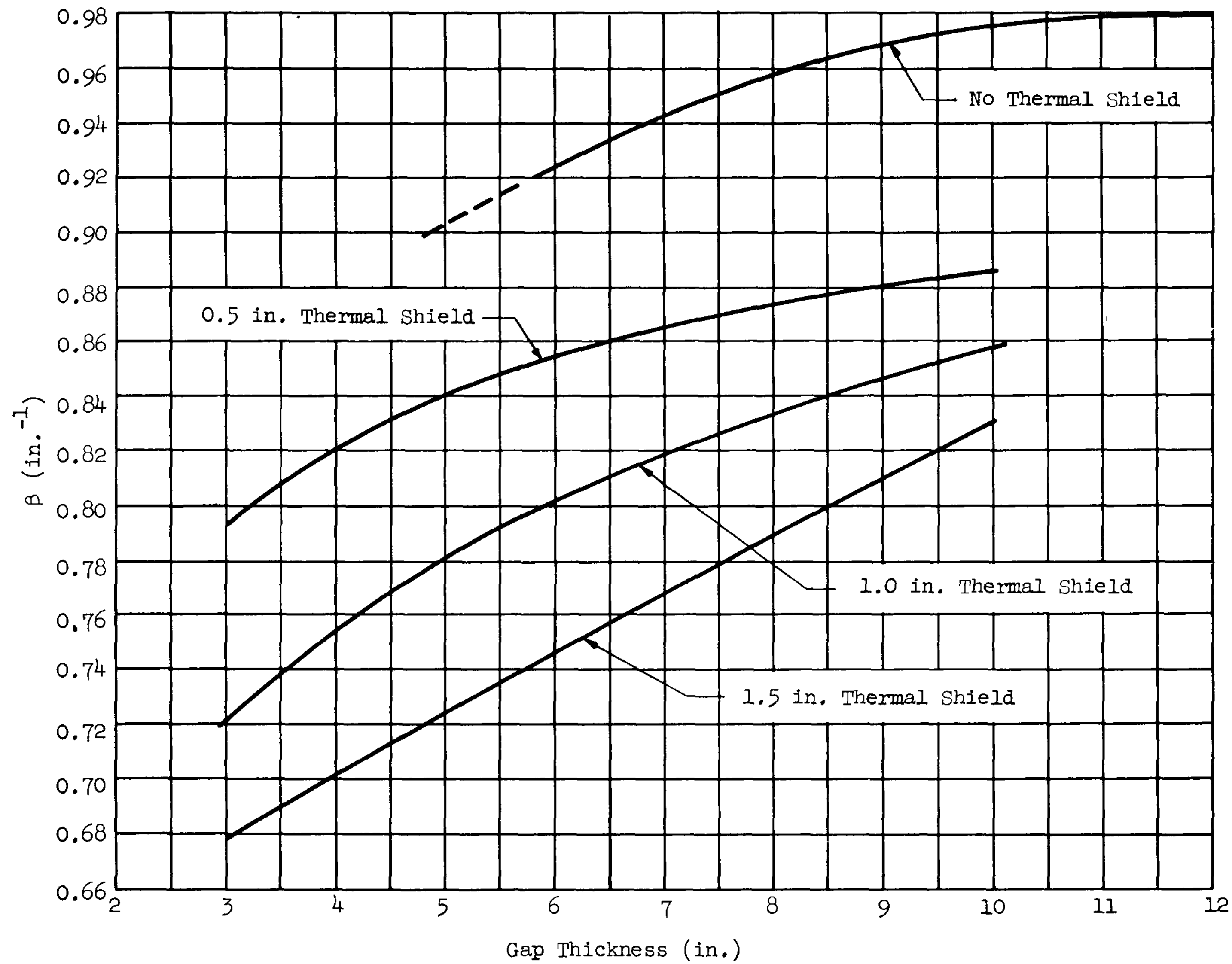


Fig. III-19. Pressure Vessel Gamma Heating

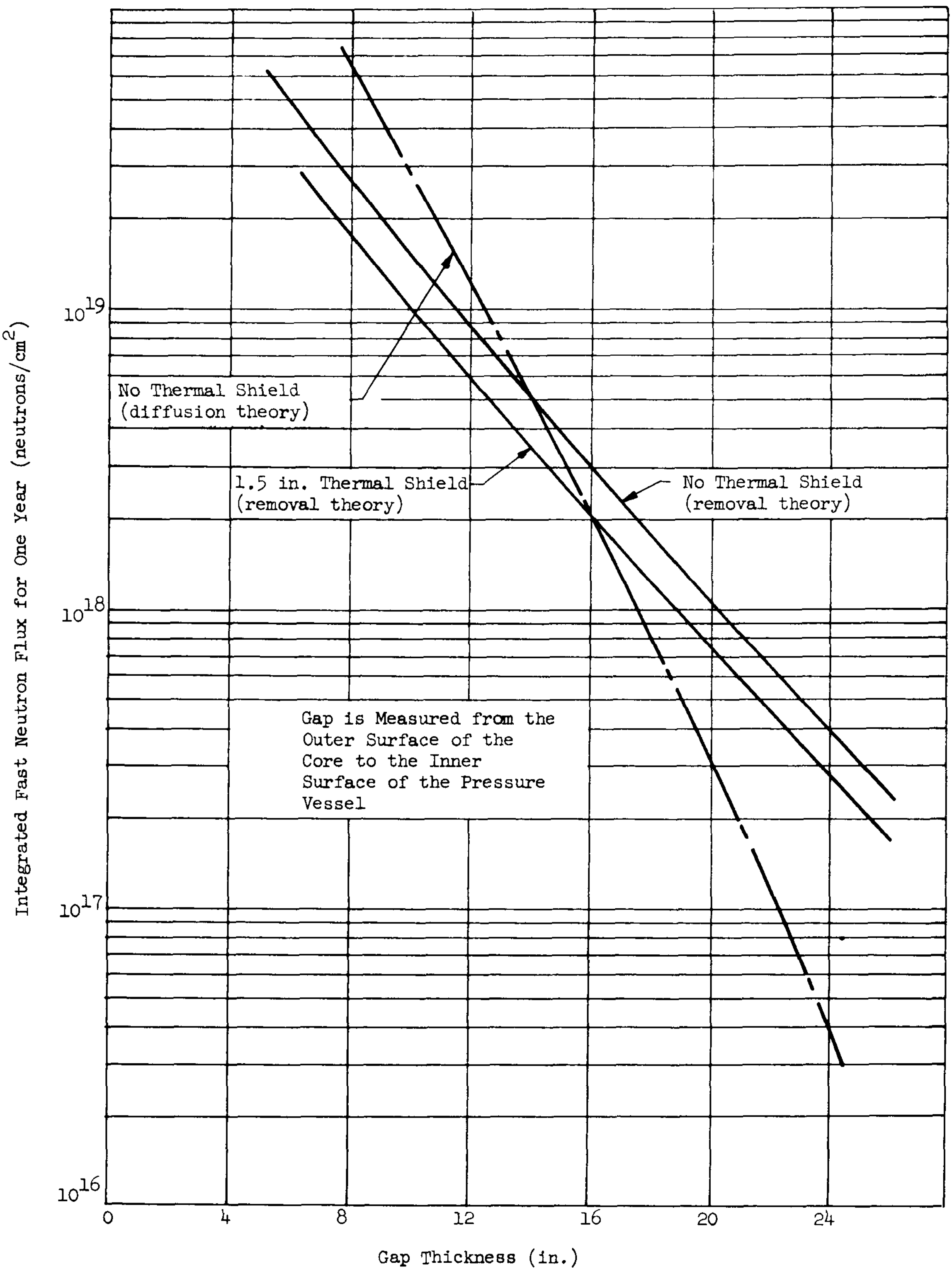


Fig. III-20. Integrated Fast Neutron Flux at the Inner Surface of Pressure Vessel for One Year Operation at 10 mw

Sources of gamma radiation which may contribute significantly to the total heating rate are prompt fission gammas, capture gammas, fission product decay gammas, and inelastic scattering gammas originating in the core; and capture gammas and inelastic scattering gammas originating in the primary water, shield water, thermal shield, and pressure vessel. Source strengths of prompt fission gammas, fission product gammas, and capture gammas were estimated from general reactor data. In estimating gammas from neutron inelastic scattering, all stainless and low alloy steel components were assumed to be iron with an inelastic cross-section of 0.1035 cm^{-1}. All gammas from this source were further assumed to have an energy of 1.0 mev.

Elastic scattering of fast neutrons in the pressure vessel imparts heat to the scattering nuclei. This contribution to the total heating is given by

$$H(x) = 1.47 \times 10^{-13} \phi_f(x) \quad (\text{Btu/in.}^3 \text{ hr}) \tag{2}$$

where $\phi_f(x)$ = fast neutron flux (neutron/cm^3 sec), x being measured radically outward from the pressure vessel inner surface. This approximation is based on fast neutron heating theory developed in TID-7004.

The core was assumed to have operated continuously for one year at a constant power of 10 mw. The primary water was assumed to be at 510° F, 2000 psi. A lower operating temperature will result in increased attenuation of core and thermal shield gammas accompanied by an increase of water capture gamma source strengths. Both effects on the total heating rate are small and tend to cancel.

The thermal shield was assumed to be centrally located in the gap between the core outer surface and inner wall of the pressure vessel.

Heating rates from the various components of the system were computed separately and summed to give a total heating rate for each configuration analyzed. Where applicable, multi-energy group calculations over the gamma spectrum were obtained. The core was assumed to have uniform source strength and an equivalent line source was used to compute heating rates in the pressure vessel. The thermal shield, pressure vessel, and water regions were treated as infinite slab sources. Previously developed line and slab source codes were used to compute the heating rates. A single buildup factor over the total number of mean free paths from source to detector point was used.

Since the system is composed entirely of steel and water, the infinite medium energy absorption buildup factor for a point isotropic source in water was used, except where use of the corresponding buildup factor for iron is obviously more correct.

In general, the gamma heating rates presented represent a conservative estimate. Choice of the buildup factor for water when attenuating through alternate layers of steel and water will give conservative values for the gamma flux. Minor variations in the primary loop design pressure and core dimensions will not have a significant effect on the heating rates. Placement of the thermal shield within the reactor vessel will affect the thermal stress within the vessel wall, a centrally placed shield giving an average value. Placing the thermal shield close to the core will produce a small increase in pressure vessel thermal stress while placement close to the vessel will result in a small decrease. Considerations other than those of gamma heating will probably determine the placement of the thermal shield.

It should be noted that the use of diffusion theory results in a different prediction of vessel integrated flux (Fig. III-20) than that obtained from the use of removal theory. Although removal theory is considered to be the more appropriate method of computation, the results should be treated as having order-of-magnitude accuracy.

Circulating activity of the primary loop coolant and fission product activity.- Activated material circulating through the primary loop of a pressurized water reactor system represents a potential hazard in the event of loop rupture. The three major sources of activated material are coolant activation, coolant impurity activation, and fission product release. Circulating activity in the PM-1 primary water has been calculated. The constants used in computation were:

Average thermal neutron flux in core region- 1.7×10^{13} neutron/cm^2 sec

Average fast neutron flux in core region--10^{14} neutron/cm^2 sec

Ratio of time of primary water in region of high neutron flux to total circulating time--0.0915

Volume of primary loop--2.152×10^6 cc

Total surface area of primary loop--1.733×10^6 cm^2

Stainless steel corrosion release rate (assumed constant)
--1.93×10^{-11} gm/cm^2 sec

Flow rate to demineralizer--1.26×10^2 cc/sec.

Saturation values of circulating and deposited activity were computed using methods set forth in TID-7004. Specific activities investigated are those given in Table III-5. Sodium is initially present as an impurity in the primary water; all other activities originate in the stainless steel. Cobalt-60m and Tantalum-182m were assumed to decay entirely to Co-60 and Ta-182 and, since both metastable activities are short lived, activities for the ground state were computed using summed cross-sections. Removal and deposition rates by primary loop components other than the demineralizers must be inferred from measurements taken in reactor systems. Values used are as stated in TID-7004 for the Naval Reactor Testing Facility, Arco, Idaho. The loop activity is directly proportional to the thermal flux, which increases with reactor operating time. For conservatism, values of the fast and thermal flux at the end of core life were used. The demineralizer was assumed 100% efficient, i.e., all circulating activity entering the demineralizer was assumed removed from the system.

The induced gamma emitting 0-16 (produced by the 0-16 (n,p) N-16 fast neutron reaction) and the neutron-emitting N-17 (produced by the 0-17 (n,p) N-17 fast neutron reaction) are the dominant sources of radiation during reactor operation. These activities are very short-lived and decay to negligible quantities a few minutes after removal of the fast neutron flux. Equilibrium values of these activities were computed by standard equations given in TID-7004. Specific activities averaged over the loop during reactor operation and at shutdown are given in Table III-6.

Saturation activity of various nuclides generated by activation of stainless steel and water impurities is given in Table III-6; as is the residual activity after various shutdown times. The totals shown represent more than 80% of the total non-fission product circulating activity.

Fission products may be introduced into the coolant by microscopic defects in fuel elements and fuel element failure. Buildup and decay of total fission product activity in the core as a function of reactor operating time and time after shutdown for the reference system is as given in Tables III-7 and III-8. The methods and data of Perkins and King were used for computation.

TABLE III-5

Corrosion Products and Impurity Activities in Primary Loop Water

Nuclide	Method of Production	Half-Life	Activation Cross-Section (barns)	Decay Constant (sec^{-1})	f_n (a)	f_s (b)
Na-24	Na-23(n,γ)Na-24	15.0 hr	0.54	1.28×10^{-5}	1.0	3×10^{-7}
Cr-51	Cr-50(n,γ)Cr-51	27.8 d	13.5	2.89×10^{-7}	0.0449	0.19
Mn-54	Fe-54(n,p)Mn-54 (c)	310 d	0.46	2.59×10^{-8}	0.0581	0.649
Mn-56	Mn-55(n,γ)Mn-56	2.58 hr	13.3	7.46×10^{-5}	1.0	0.0200
Fe-55	Fe-54(n,γ)Fe-55	2.96 yr	2.5	7.42×10^{-9}	0.0581	0.649
Fe-59	Fe-58(n,γ)Fe-59	46 d	0.98	1.74×10^{-7}	0.0034	0.649
Co-58	Ni-58(n,p)Co-58 (c)	72 d	0.31	1.11×10^{-7}	0.6776	0.12
Co-60	Co-59(n,γ)Co-60	5.28 yr	20.0	4.16×10^{-9}	1.0	0.0004
Co-60m (d)	Co-59(n,γ)Co-60m	10.4 min	16.0	1.11×10^{-3}	1.0	0.0004
Ta-182	Ta-181(n,γ)Ta-182	111 d	19.0	7.23×10^{-8}	1.0	0.001
Ta-182m (e)	Ta-181(n,γ)Ta-182	16.4 min	0.03	7.04×10^{-4}	1.0	0.001

(a) f_n is the natural abundance of the target nuclide

(b) f_s is the abundance by weight of the target nuclide in the system material

(c) Neutron-proton (n-p) reactions are fast neutron reaction

(d) 99.7% of 10.4 min Co-60 decays to 5.28-yr Co-60 (isomeric transition)

(e) ~95% of 16.4 min Ta-182 decays to 111-d Ta-182 (isomeric transition)

TABLE III-6

Primary Loop Circulating Activity

(Curies)

Nuclide \ Decay Time (sec)	0	10^2	10^3	10^4	10^5	10^6	10^7	10^8	10^9
Na-24	6.5×10^{-1}	6.49×10^{-1}	6.42×10^{-1}	5.72×10^{-1}	1.80×10^{-1}	2.72×10^{-6}	-	-	-
Cr-51	1.3×10^{-1}	1.3×10^{-1}	1.3×10^{-1}	1.3×10^{-1}	1.26×10^{-1}	9.74×10^{-2}	7.28×10^{-3}	-	-
Mn-54	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}	1.07×10^{-2}	8.49×10^{-3}	8.25×10^{-4}	-
Mn-56	1.55×10^{0}	1.54×10^{0}	1.44×10^{0}	7.35×10^{-1}	8.99×10^{-4}	-	-	-	-
Fe-55	2.9×10^{-3}	2.9×10^{-3}	2.9×10^{-3}	2.9×10^{-3}	2.9×10^{-3}	2.87×10^{-3}	2.69×10^{-3}	1.38×10^{-3}	1.68×10^{-6}
Fe-59	1.4×10^{-3}	1.4×10^{-3}	1.4×10^{-3}	1.4×10^{-3}	1.38×10^{-3}	1.17×10^{-3}	2.37×10^{-4}	-	-
Co-58	6.2×10^{-2}	6.2×10^{-2}	6.2×10^{-2}	6.19×10^{-2}	6.13×10^{-2}	5.55×10^{-2}	2.03×10^{-2}	8.99×10^{-7}	-
Co-60m	1.9×10^{-2}	1.7×10^{-2}	6.26×10^{-3}	-	-	-	-	-	-
Co-60	2.3×10^{-4}	2.3×10^{-4}	2.3×10^{-4}	2.3×10^{-4}	2.3×10^{-4}	2.29×10^{-4}	2.20×10^{-4}	1.51×10^{-4}	3.59×10^{-6}
Ta-182m	3.2×10^{-5}	2.98×10^{-5}	1.58×10^{-5}	2.88×10^{-8}	-	-	-	-	-
Ta-182	1.7×10^{-3}	1.7×10^{-3}	1.7×10^{-3}	1.67×10^{-3}	1.66×10^{-3}	1.55×10^{-3}	8.10×10^{-4}	1.21×10^{-6}	-
N-16	6.68×10^{2}	5.36×10^{-2}	-	-	-	-	-	-	-
N-17	1.01×10^{1}	-	-	-	-	-	-	-	-
Total	6.8×10^{2}	2.47×10^{0}	2.30×10^{0}	1.52×10^{0}	3.85×10^{-1}	1.69×10^{-1}	4.00×10^{-2}	2.36×10^{-3}	5.27×10^{-6}

The primary loop attains 98% of saturation activity after approximately 700 hr of continuous operation.

TABLE III-7

Buildup of Fission Product Activity

Reactor Operating Time	Activity (curies)
0	0
1 hr	1.24×10^{7}
10 hr	1.89×10^{7}
100 hr	2.43×10^{7}
1000 hr	2.70×10^{7}
2 yr	2.78×10^{7}

TABLE III-8

Decay of Fission Product Activity

Time after Shutdown	Activity (curies)
0	2.78×10^{7}
8 hr	1.14×10^{7}
1 day	9.19×10^{6}
30 days	3.24×10^{6}
60 days	2.20×10^{6}
90 days	1.64×10^{6}
120 days	1.28×10^{6}
1 yr	3.78×10^{5}

Although it is recognized that circulating loop activity varies with the primary system design and geometry, it is interesting to note that the isotopic activity levels computed for the PM-1 system were found to be within an order of magnitude of the average values of the measured isotopic activities in the APPR-1 primary loop.

System studies.-

J. Beam C. Smith

The integration of the primary and secondary loops for nucleate boiling and non-boiling cores was started at the end of the quarter. In preparation for this integration, the costs and weights of various components of the primary loop were determined and either plotted or tabulated.

During the second quarter, primary loop integration will be completed and will form the basis for preliminary design.

First quarter analyses were as follows:

Core fabrication--Information was obtained on core fabrication costs based on past experience attained under Contract AT(30-3)-277. The data were extrapolated for different size tubes and meat thicknesses.

Primary pump--Information on canned rotor, mechanical seal, and controlled leakage pumps was obtained. Evaluation of this information indicates that mechanical seal pumps cost about 60% as much, are approximately 1.5 ft longer, are about 1000 lb heavier, and have an efficiency about 10% higher than comparable canned rotor pumps.

Reactor vessel--The weight of the reactor vessel was calculated for various values of vessel internal diameter, core length, and system pressure. The cost was computed on the basis of an estimated cost of $2.80/lb; the results are shown in Fig. III-21. Actuators and actuator housings are not included.

Pressurizer--A design study was conducted to determine a reasonable ratio of primary system design to operating pressure. As this ratio increases, the pressurizer volume decreases but the pressurizer shell thickness increases. A value of 1.15 was chosen as being a good compromise between these two conditions.

The assumptions used in the analysis were:

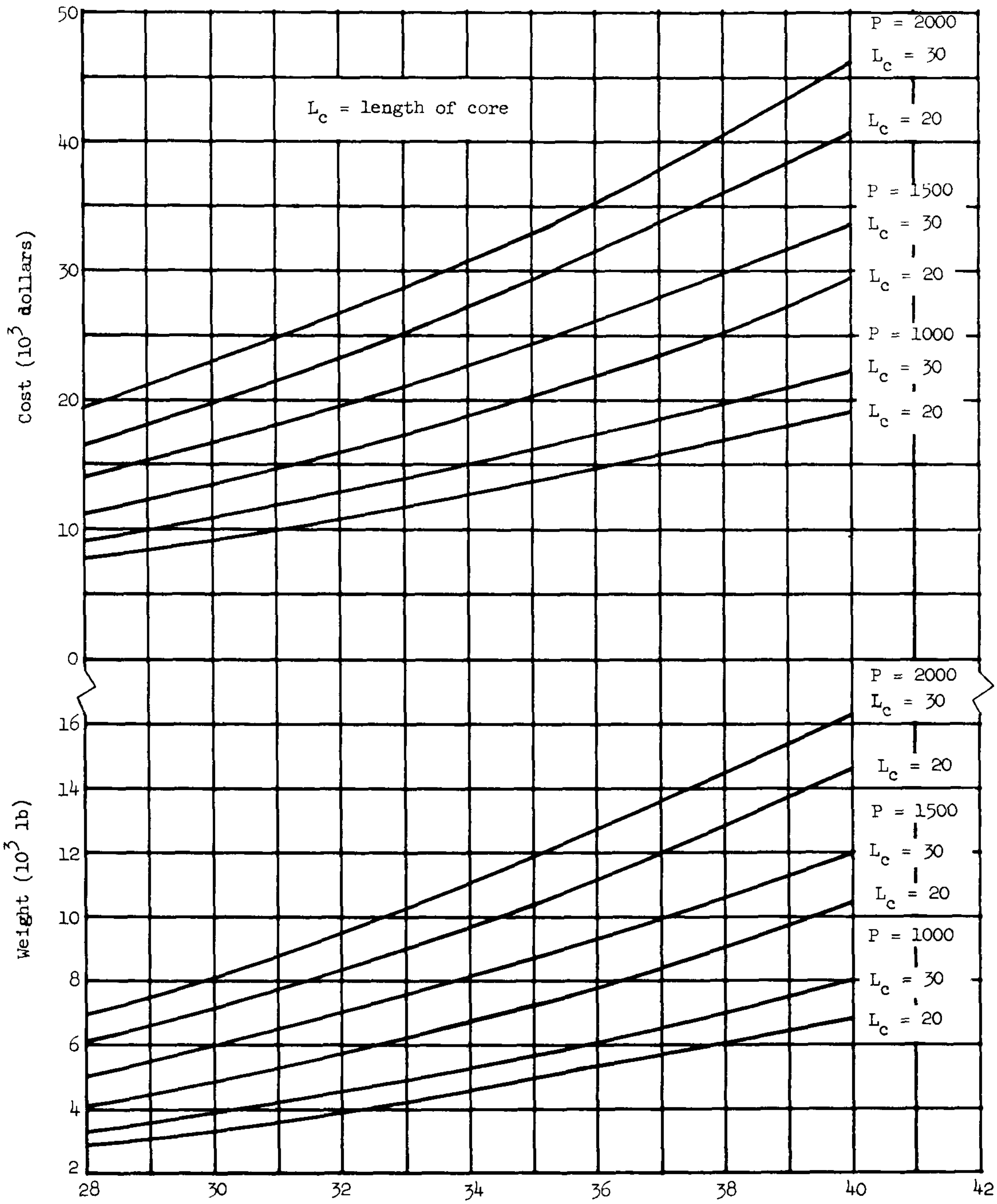


Fig. III-21. Reactor Vessel Weight and Cost Variation

(1) Heaters and spray nozzles were inoperative during transients.

(2) Steam expansion and compression is isentropic.

(3) Liquid expansion is isentropic.

(4) The pressurizer is half-full of water; the balance of its volume is filled with steam.

(5) Pressurizer water is saturated.

(6) Loop temperature is 100° F below the saturation temperature in the pressurizer.

With these assumptions it was found that for a fixed pressurizer volume and a given primary loop temperature variation, the magnitude of the pressure drop associated with an outsurge was approximately 1/6 of the magnitude of the pressure rise associated with an insurge.

The variation of pressurizer shell weight with system pressure was found assuming: a 15° F mean loop temperature variation; a loop volume of 75 cu ft; and a maximum insurge pressure of 15% above the operating pressure. The cost of the pressurizer shell was calculated on the basis of an estimated price of $2.70/lb. The cost and weight of the pressurizer shell is shown in Fig. III-22. Multiplier factors (to be applied to pressurizer shell weights and volumes) were determined for other loop temperatures and volumes. These are shown along with pressurizer volume in Fig. III-23.

Primary loop piping--The weight and cost of the primary loop main coolant piping was calculated based upon the following assumptions:

(1) The total length of piping is 50 ft

(2) There are 12 long-sweep elbows

(3) The maximum fluid velocity is 30 fps

(4) The ratio of design pressure to operating pressure is 1.15

(5) Seamless pipe is used

(6) Allowable pipe stress is 15,000 psi

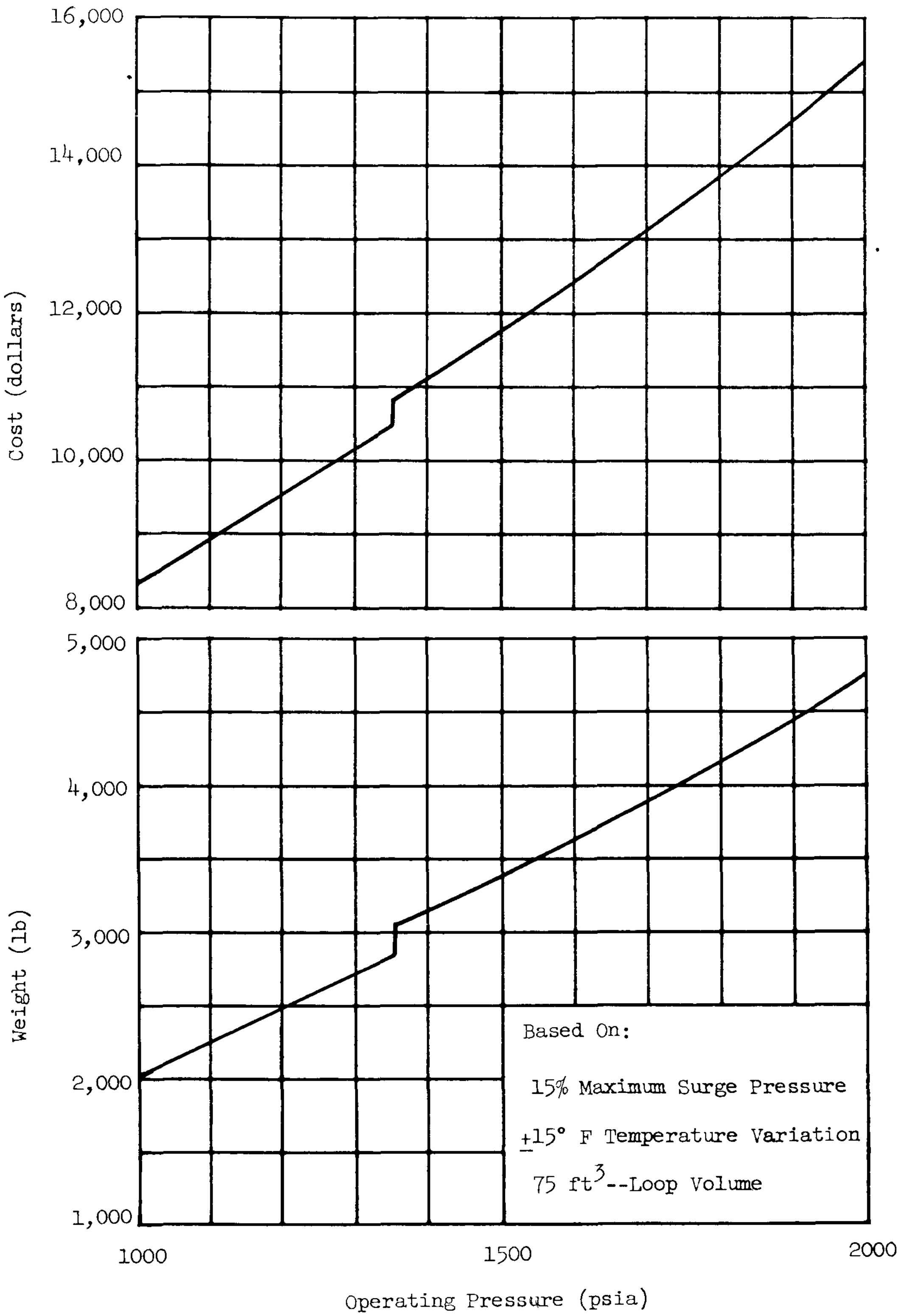


Fig. III-22. Pressurizer Weight and Cost vs Operating Pressure

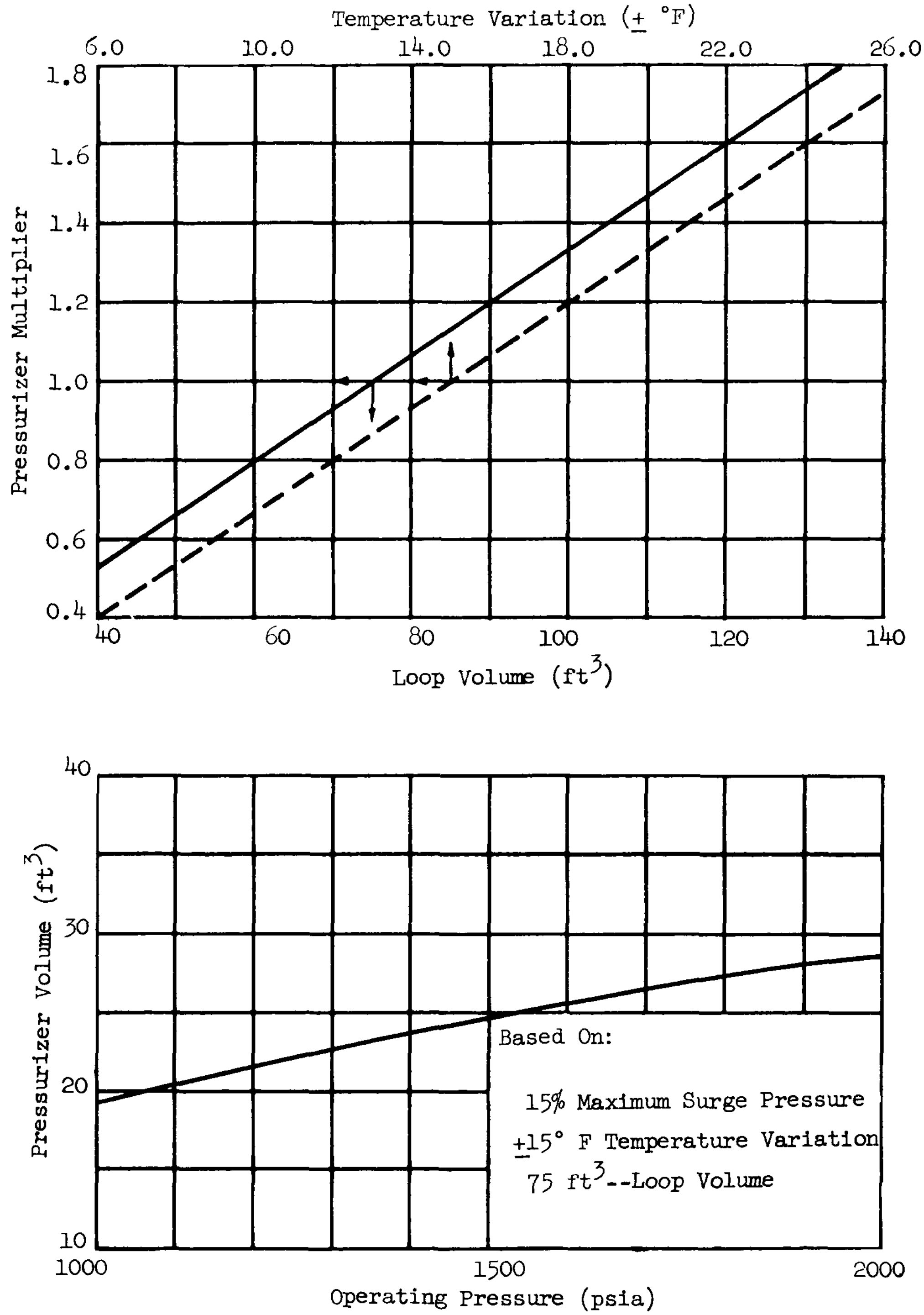


Fig. III-23. Pressurizer Size vs Operating Pressure, Loop Volume and Temperature Variation

(7) The corrosion allowance is 0.065 in. over the life of the plant

(8) There are five flanged connections.

The cost and weight of the main primary coolant piping was plotted versus system operating pressure for 5-, 6-, and 8-in. pipe of schedules 80 to 160 inclusive. These plots are shown in Fig. III-24.

Auxiliary system--A study was made of the weights and cost of the auxiliary systems. Included in the study were the primary coolant purification, the primary coolant blowdown, the shield water, the heat ejection from the steam pressurizer blow-off, and the storage and purification of liquid waste systems. Cost and weight data for stainless steel piping, fittings, and valves ranging in size from 1/4 to 2 in. were obtained from vendors. The variation of system cost and weight with pressure was determined and is shown in Fig. III-25.

A brief study of the cost and weight of non-boiling and nucleate boiling primary systems for fixed values of core diameter, fuel element geometry, and mean temperature was prepared. System pressure was taken as the independent variable and the required flow rate was determined. The primary pump and steam generator were then sized and their costs and weights determined.

The results are shown in Tables III-9 and III-10. The costs and weights listed represent only those components of the system which vary significantly with the variables considered in this study. These were:

(1) Primary loop pump

(2) Reactor vessel

(3) Pressurizer

(4) Primary loop piping

(5) Auxiliary systems

(6) Steam generator.

The integration of the primary and secondary loops for nucleate boiling and non-boiling cores was started at the end of the first quarter. Upon completion of the two studies, comparisons will be made of the cases representing the lowest capital investment in each type of core.

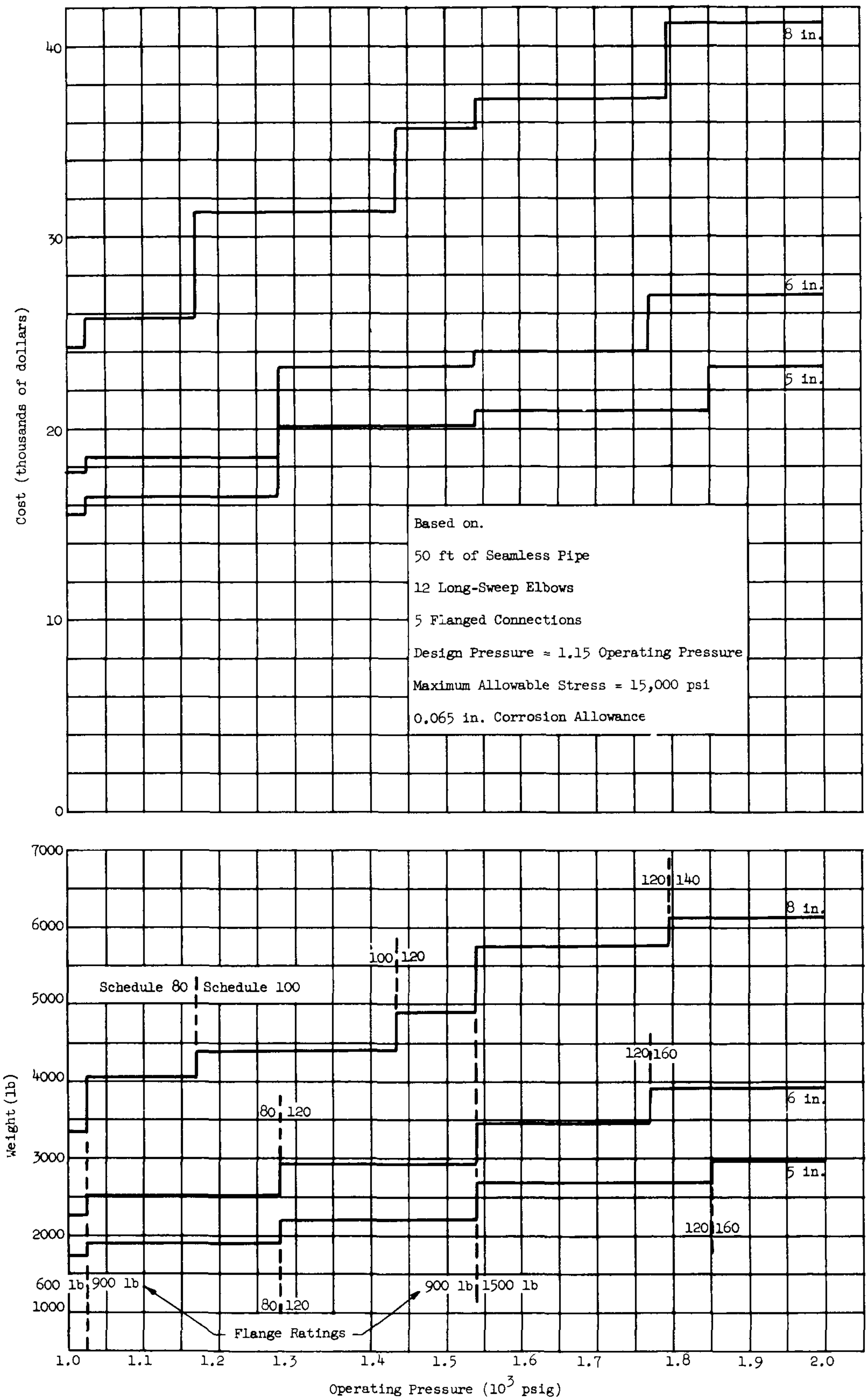


Fig. III-24 Primary Coolant Piping Weight and Cost vs Operating Pressure

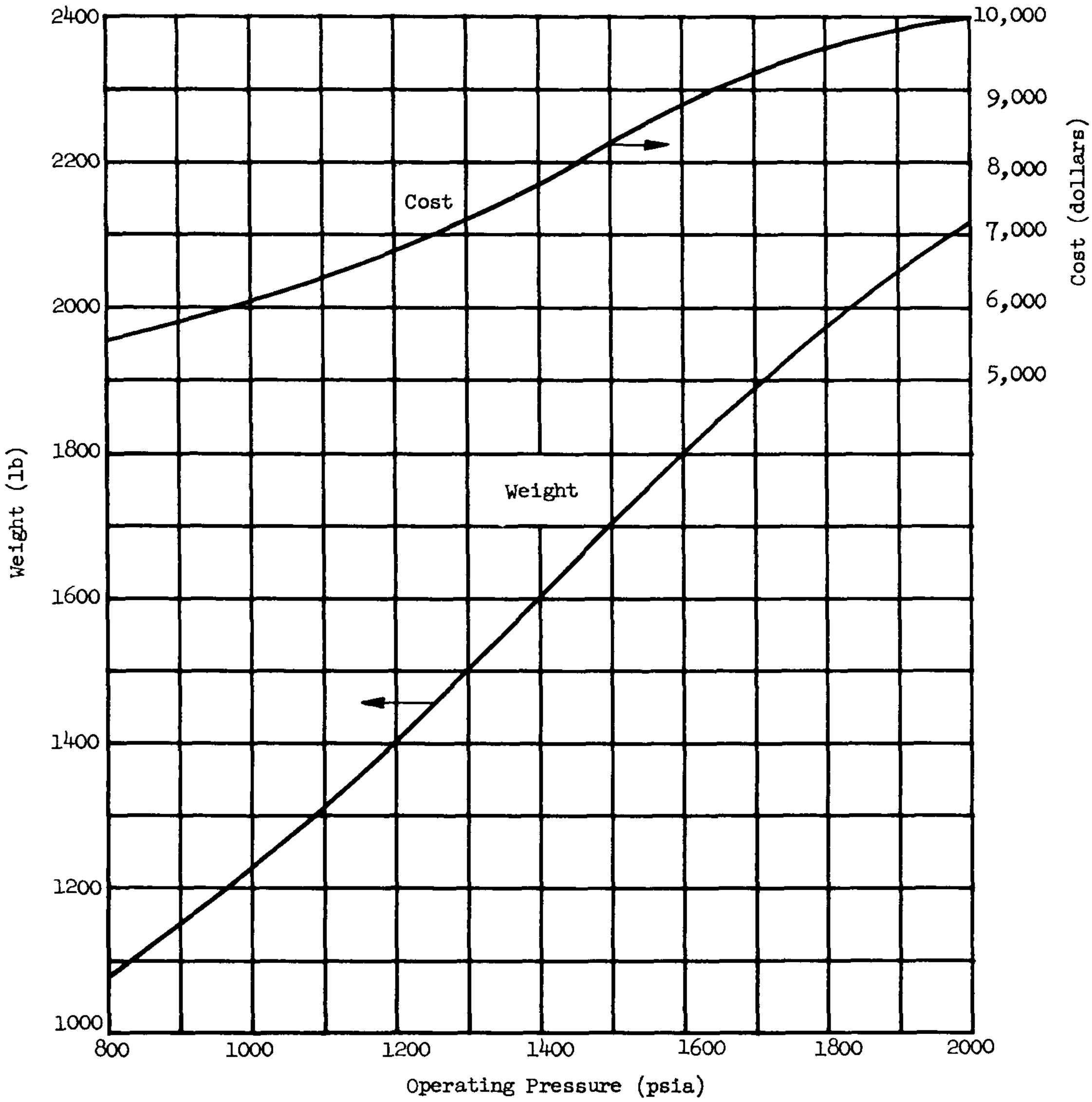


Fig. III-25. Primary Auxiliary Systems Weight and Cost vs Operating Pressure

TABLE III-9

Nucleate and Nonboiling Primary Loop Comparison Study

Case No.	Primary Characteristics							Secondary Characteristics		System Integration Auxiliary Power (kw)	Equipment			Cost			
	Pressure P_p (psia)	Flow Rate F_p (gpm)	Average Temperature $\bar{T}$ (° F)	Temperature Difference Reactor ΔT_p (° F)	Core Diameter D_c (in.)	Fuel Elements Outer	Fuel Elements Inner	Pressure P_s (psia)	SG LMTD	Pri Pump	Primary (lb)	Steam Gen (lb)	Total (lb)	Pri Equip ($)	Core Fab ($)	Steam Gen ($)	Total ($)
			Nucleate Boiling														
A-1	1100	2840	475	24.0	20	780-- Total 0.5 in. OD		300	57.0	107.5	13,000	9,100	22,100	129,090	43,000	73,500	245,590
A-2	1300	2130	475	32.5	20	780-- Total 0.5 in. OD		300	56.0	75.5	12,845	9,340	22,185	109,425	43,000	74,500	226,925
A-3	1600	1620	475	42.5	20	780-- Total 0.5 in. OD		300	55.0	35.9	16,657	9,600	26,257	100,400	43,000	75,100	218,500
A-4	1100	2840	475	24.0	24	1120-- Total 0.5 in. OD		300	57.0	107.5	13,000	9,100	22,100	129,090	62,500	73,500	265,090
A-5	1300	2130	475	32.5	24	1120-- Total 0.5 in. OD		300	56.0	75.5	12,845	9,340	22,185	109,425	62,500	74,500	246,425
A-6	1600	1620	475	42.5	24	1120-- Total 0.5 in. OD		300	55.0	35.9	16,657	9,600	26,257	100,400	62,500	75,100	238,000
			Nonboiling														
K-1	1300	1710	475	40.5	24	765/ 0.5 OD	1470/0.295 in. OD	300	55.0	45.6	12,950	9,600	22,550	98,025	111,000	75,200	284,225
K-2	1600	1240	475	56.0	24	765/ 0.5 OD	1970/0.255 in. OD	300	53.0	28.8	15,087	10,100	25,187	95,100	136,000	76,400	307,500
K-3	1300	2110	475	32.5	24	765/ 0.5 OD	885/0.380 in. OD	300	56.0	81.4	12,945	9,390	22,335	116,625	83,800	74,400	274,825
K-4	1600	1525	475	45.5	24	765/ 0.5 OD	885/0.380 in. OD	300	54.5	33.5	16,332	9,730	26,062	99,600	83,800	75,400	258,800

TABLE III-10

Summary of Table III-9

Case No.	Primary Pressure P_p	Weight (lb)	Cost ($)	Pumping Power (kw)
A-1	1100	22,100	245,590	107.5
A-4	1100	22,100	265,090	107.5
A-2	1300	22,185	226,925	75.5
A-5	1300	22,185	246,425	75.5
K-1	1300	22,550	284,225	45.6
K-3	1300	22,335	274,825	81.4
A-3	1600	26,257	218,500	35.9
A-6	1600	26,257	238,000	35.9
K-2	1600	25,187	307,500	28.8
K-4	1600	26,062	258,800	33.5

Design studies.-

H. Brainard H. Clark J. Goeller J. Todd

The objective of this study is the definition of several primary loop arrangements.

The following basic ground rules were established for the parametric study of primary loop design and arrangement.

(1) The minimum number of packages will be used. Dimensions were limited to 8 ft 8 in. wide by 8 ft 8 in. high by 30 ft long.

(2) Total package weight, including equipment, will not exceed 30,000 lb during shipment.

(3) The primary loop will be designed to eliminate the need for field welding.

(4) No concrete will be employed.

(5) The package will either be installed above ground or will require minimum excavation. (This minimizes the amount of required site preparation.)

(6) Packages must be adaptable to relocation.

(7) Personnel must be afforded access to the steam generator while the primary loop is at operating pressure, and access for refueling the reactor eight hours after shutdown.

Several arrangements of primary loop equipment were investigated. Typical of these were:

(1) Arrangement of the primary loop into two flat-sided packages of square cross-section, Fig. III-26. This arrangement would be used with a water biological shield.

(2) Arrangement of the primary loop into a single square package containing both the steam generator and the reactor, Fig. III-27. A water biological shield would be used.

(3) Arrangement of the primary loop into two cylindrical packages, Fig. III-28. (Since the inlet pipe elevation depends on the heat transfer system used, two variations are shown.) Biological shielding is provided by backfilling.

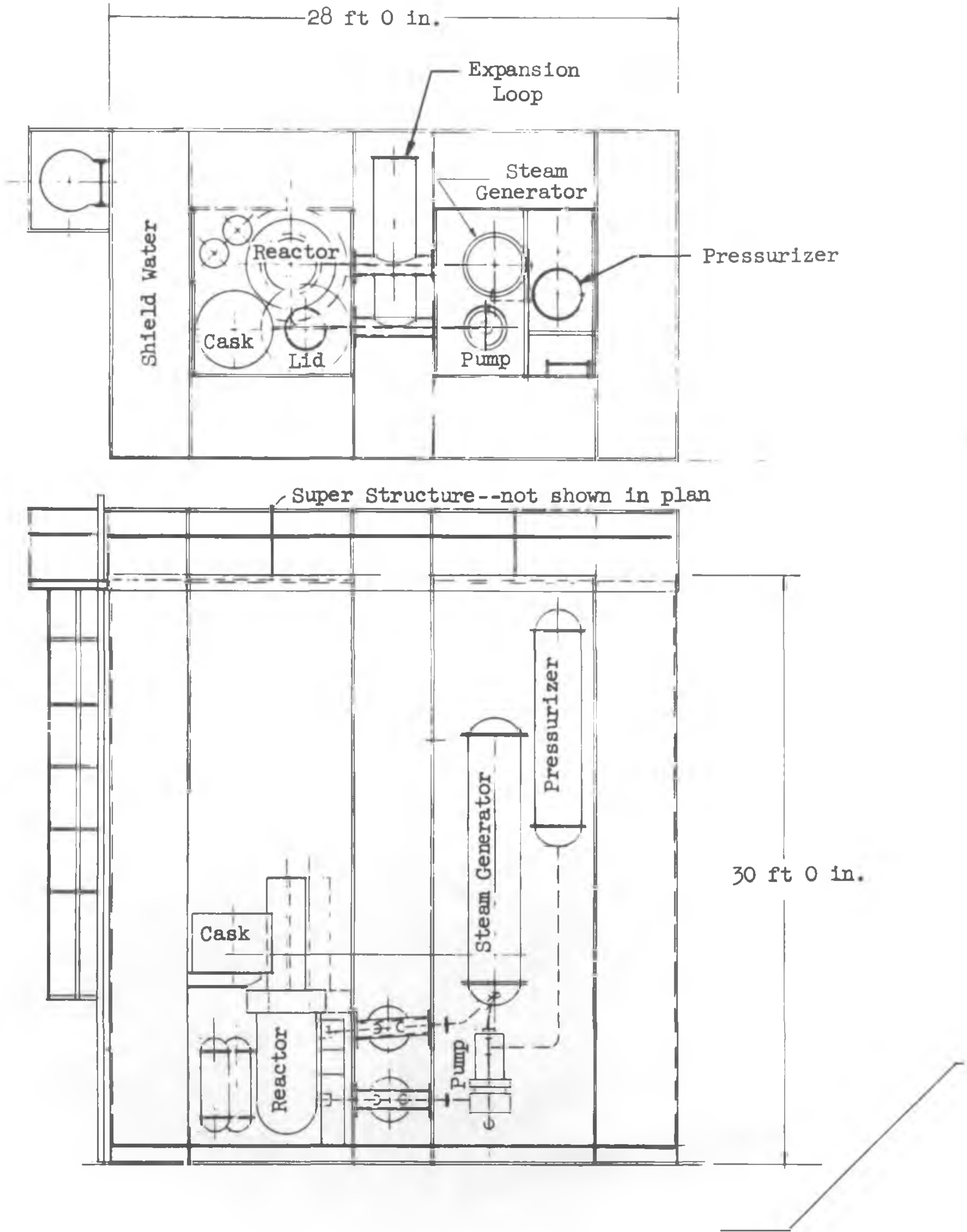


Fig. III-26. Primary Loop Arrangement--Two Square Packages

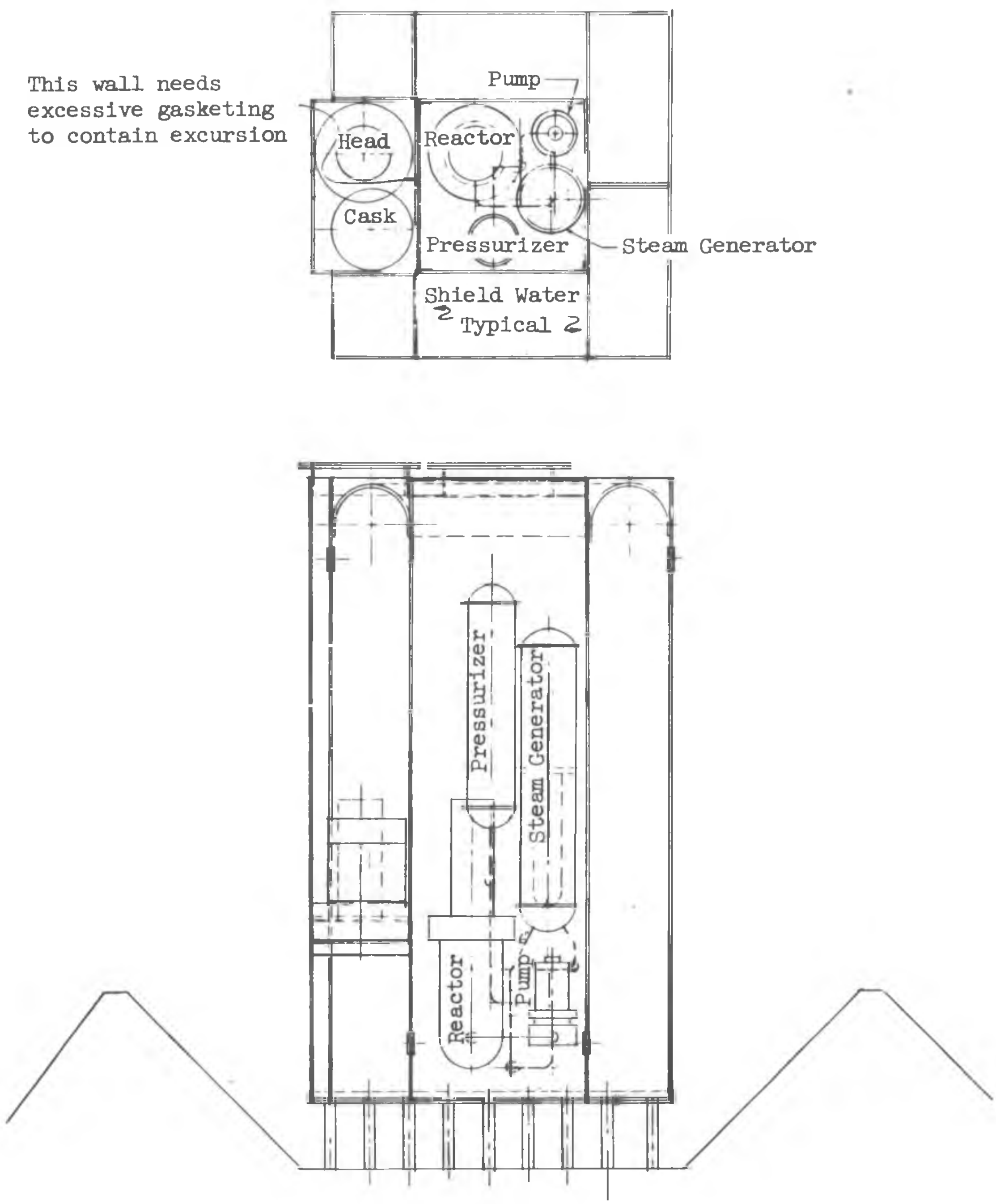


Fig. III-27. Primary Loop Arrangement--One Square Package

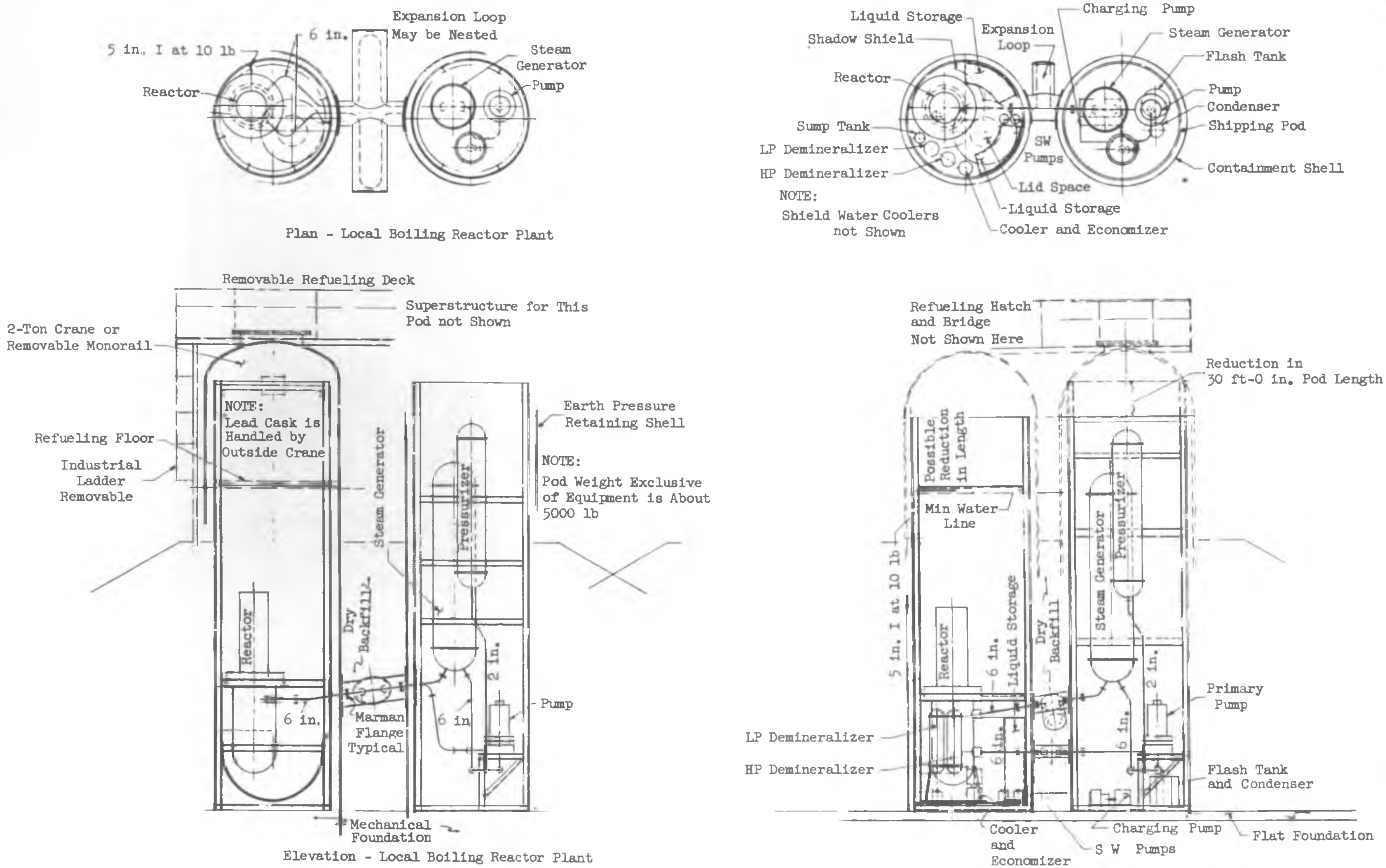


Fig. III-28. Circular Tank Complex--Dry Backfill

The advantages of the first method mentioned are:

(1) Maximum space is available for equipment placement and subsequent maintenance.

(2) Maximum space is available for refueling operations.

(3) No excavation is required; placement is above ground.

(4) Minimum erection and site preparation effort is required.

Its disadvantages are:

(1) Elaborate structural reinforcement is necessary to guarantee structural integrity of flat sided tanks against hydrostatic pressure. Although required structural reinforcement can be reduced by using a sandwich-type construction paneling, it is relatively expensive and may not maintain its strength when subjected to radiation.

(2) The water supply at the proposed site is insufficient to fill the shield water tanks in a reasonable time.

(3) The primary loop complex would exceed four airplane loads.

The second design mentioned was examined to determine whether so compact a unit was feasible. It was found not to be feasible because:

(1) Little maintenance room is available.

(2) Radiation levels would be too high in the vicinity of the steam generator (even after shutdown) unless a great deal of shielding was provided.

(3) The neutron radiation level at the pump would be excessive during operation.

(4) Placement of a removable panel between the package and the refueling area present many problems.

The cylindrical tank arrangement appears to be satisfactory. Use of the earth retaining shell is optional; this arrangement has not been completely evaluated as yet.

Aluminum was seriously considered for reducing overall package weight, although welding reduces its allowable working stress considerably with a resultant increase in required material thickness and weight.

The following conclusions were reached after considering the various arrangements:

(1) The concept of using water as the single biological shield material was abandoned. Site backfill is often desirable since earth mounds can be easily erected and disassembled, thereby avoiding the additional cost of fabricating and shipping shield water tanks.

(2) Water must be available for personnel shielding during refueling operations as well as to remove spent core afterheat.

(3) Arrangement of the primary system is to be such that a minimum amount of modification is required whether the system is contained or non-contained.

(4) The package containing the reactor vessel must be vertical to provide room for refueling operations and to maintain approximately 9.7 ft of water above the reactor core for protection of maintenance personnel during these operations.

(5) The additional packages containing the steam generator, pressurizer, and associated components may be horizontal or vertical.

In general, all assumptions, for purposes of stress analysis, were conservative.

Layouts and basic design data to optimize the primary loop equipment arrangement from the standpoint of piping, shielding, and maintenance, are major problems to be considered during the second quarter.

2. Secondary System Studies

W. Koch | L. Hassel

During the first quarter the main effort associated with the secondary loop portion of Subtask 3.1 was devoted to coordinating the Westinghouse Electric Corporation performance of the following:

(1) General secondary system studies

(2) Steam generator studies

(3) Turbine-generator studies

(4) Miscellaneous studies.

These studies will be completed in the next quarter. They will form the basis for preliminary design.

General studies.- The general secondary system studies included:

(1) Isolation of system equipment and promising ranges of input and exhaust pressures through preparation and analysis of representation heat balances.

(2) Analysis of the effects of varying system back pressure and the boiler feed pump prime mover upon a fixed pressure system.

(3) Analysis of the effect of back pressure on the required number of condenser packages.

(4) Analysis of the feasibility of utilizing a single condenser package.

Preliminary investigation of possible PM-1 primary loops and steam generators indicated that the most promising range of secondary loop steam pressures was from 200 to 600 psia. This was based on probable primary coolant temperatures and the desire to minimize steam generator size and weight. Since considerable work had already been done on the ethylene-glycol type condenser system, and since direct air-steam systems appeared to offer significant advantages, it was decided to concentrate on direct air-steam condenser systems. Past studies had indicated that, when the condenser heat must be rejected by air cooling, the size, weight, and electrical power requirements make the range of 6 to 11-1/2 in. Hg abs turbine back pressure attractive.

Both the two-turbine generator system (without extraction) and the single-turbine generator system both with and without extraction) were investigated within the steam pressure and turbine back pressure ranges stated. A total of 250 ekw was allowed for primary and secondary loops (125 kw each) auxiliary power, requiring that the gross output of the plant be 1250 ekw to attain a net output of 1000 ekw (plus 7×10^6 Btu/hr of process heat in the form of 30 psia steam).

Figures III-29, III-30, and III-31 show some of the representative heat balances. Figure III-29 shows the heat balance for a 1250-kw turbine, zero extraction, cycle operating at 500 psia steam pressure and 9-in. Hg abs back pressure. Figure III-30 shows the heat balance for a similar unit but with extraction.

Figure III-31 shows a two-turbine (625 kw each) cycle, with no extraction, operating at 200 psia steam and 11-1/2-in. Hg abs back pressure.

The results of the heat balance studies for the three systems are plotted in Figs. III-32, III-33 and III-34 as steam inlet pressure versus the steam generator thermal output required to generate required power at three turbine exhaust back-pressures.

As expected, the most efficient cycle is the system using 1250-kw turbine generator set with extraction to a closed feed water heater. The two-turbine cycle without extraction is the least efficient.

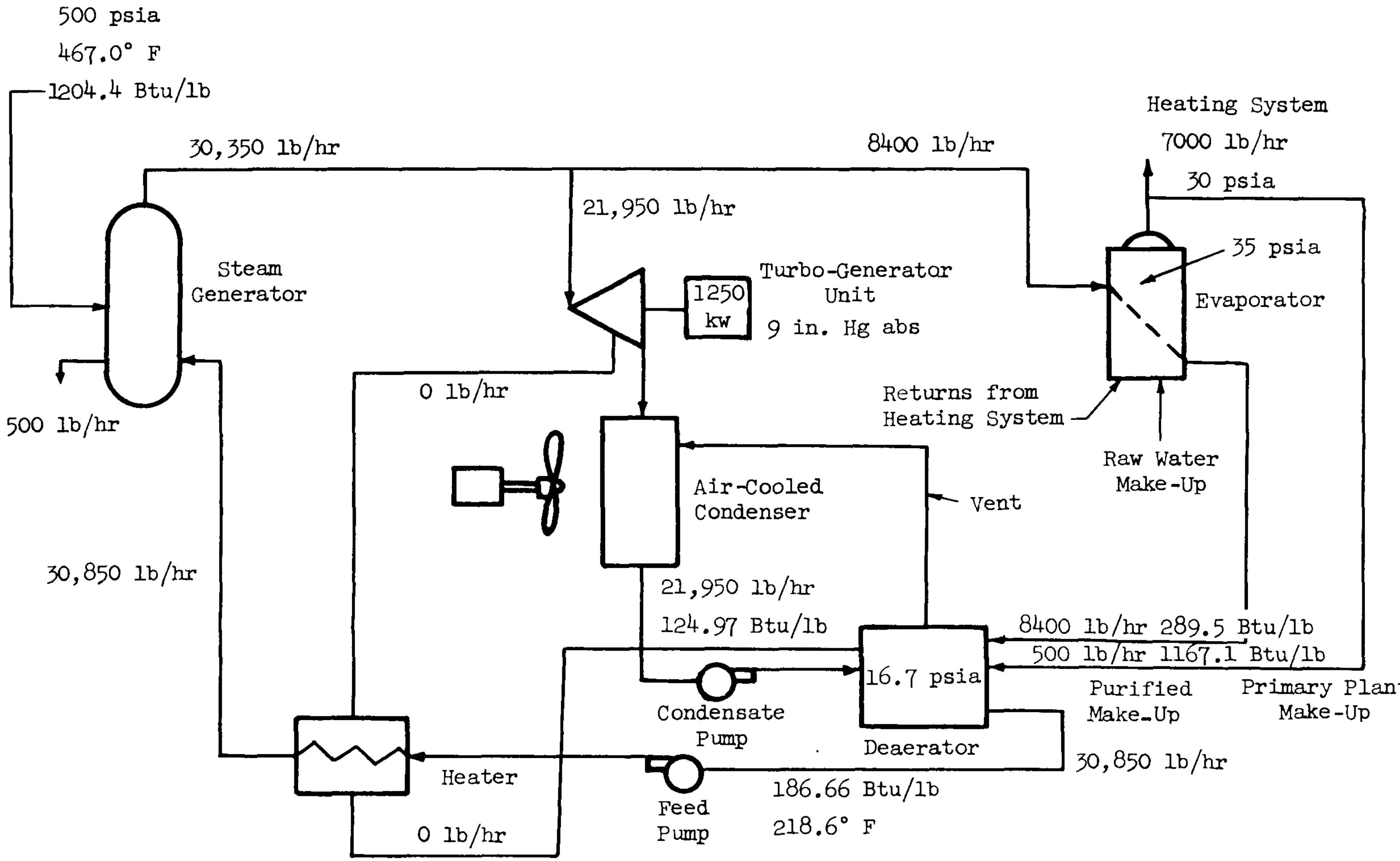


Fig. III-29. Geared Turbine-Generator Basic Heat Balance--Zero Extraction

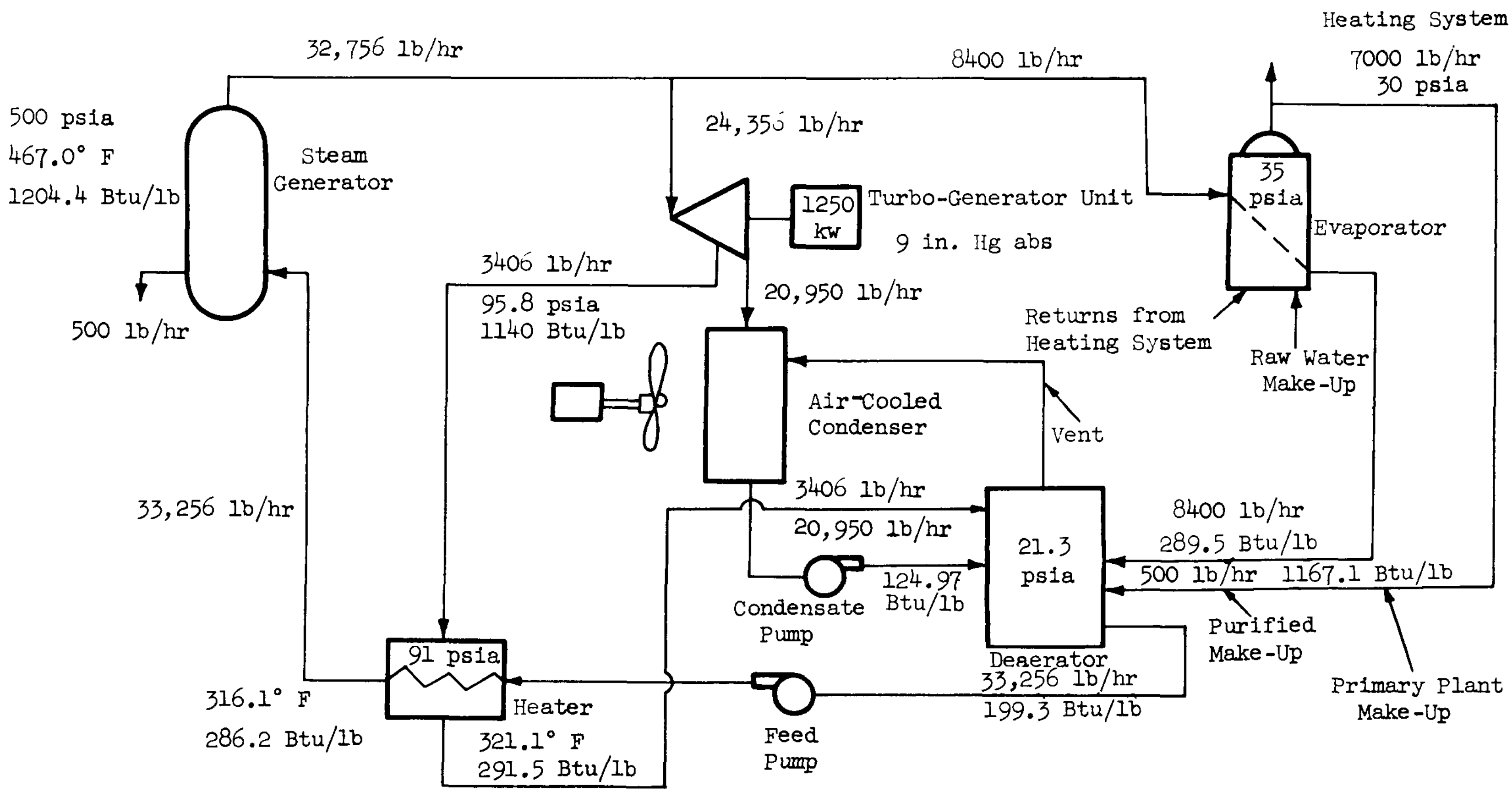


Fig. III-30. Geared Turbine Generator Basic Heat Balance Extraction Unit

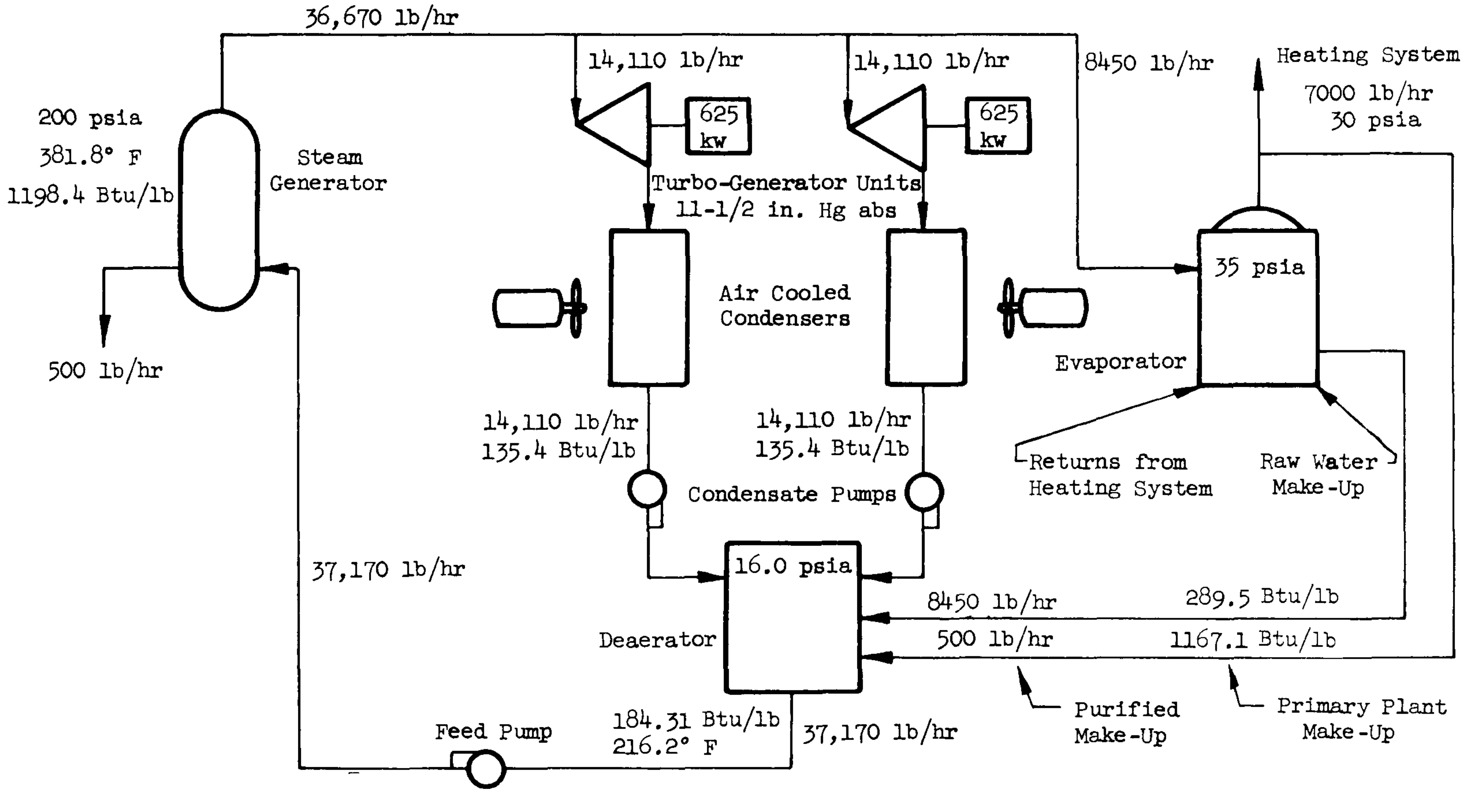


Fig. III-31. Geared Turbine-Generator Basic Heat Balance

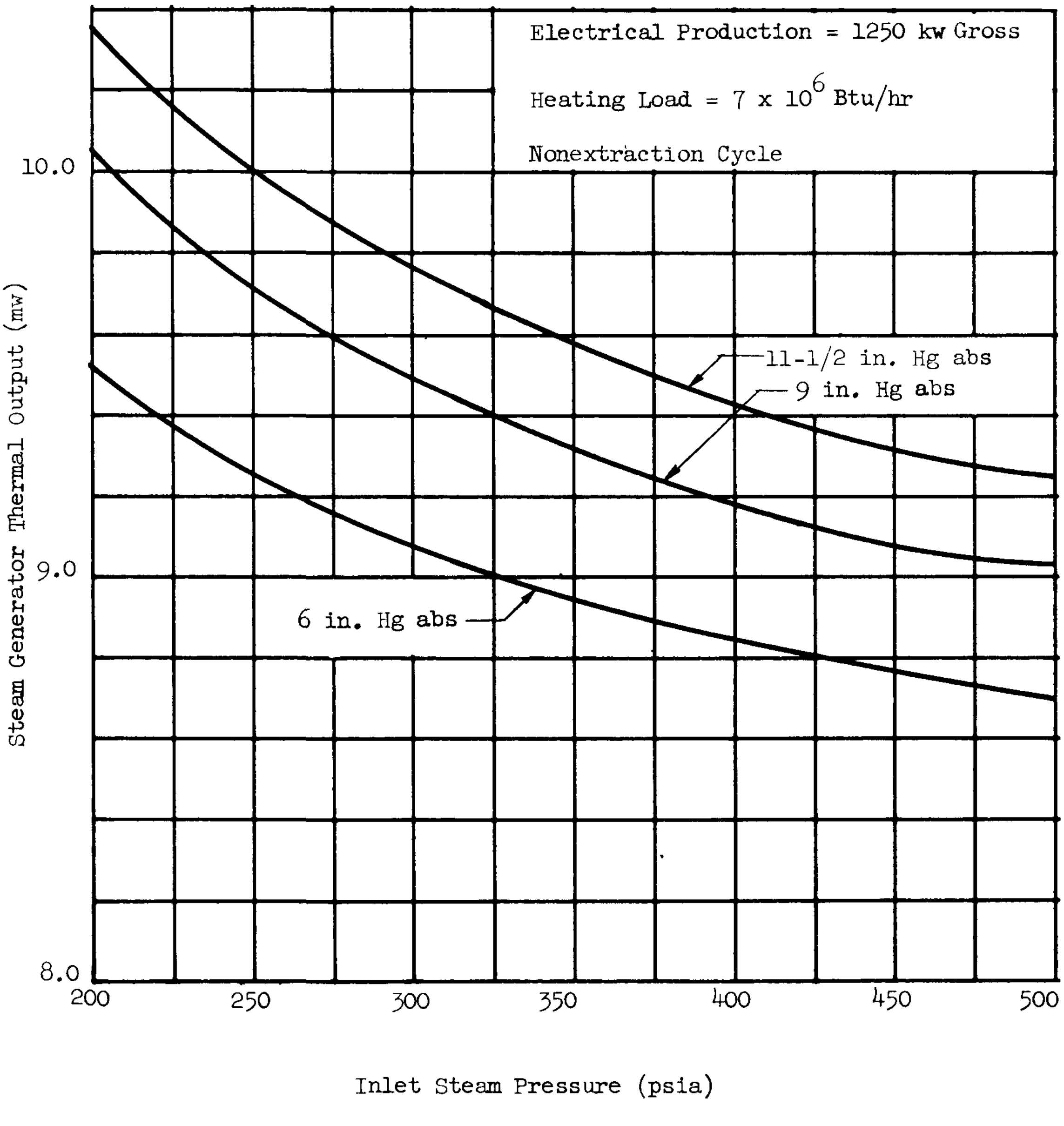


Fig. III-32. Steam Generator Thermal Output vs Inlet Steam Pressure-- 1250 kw Geared Turbine-Generator

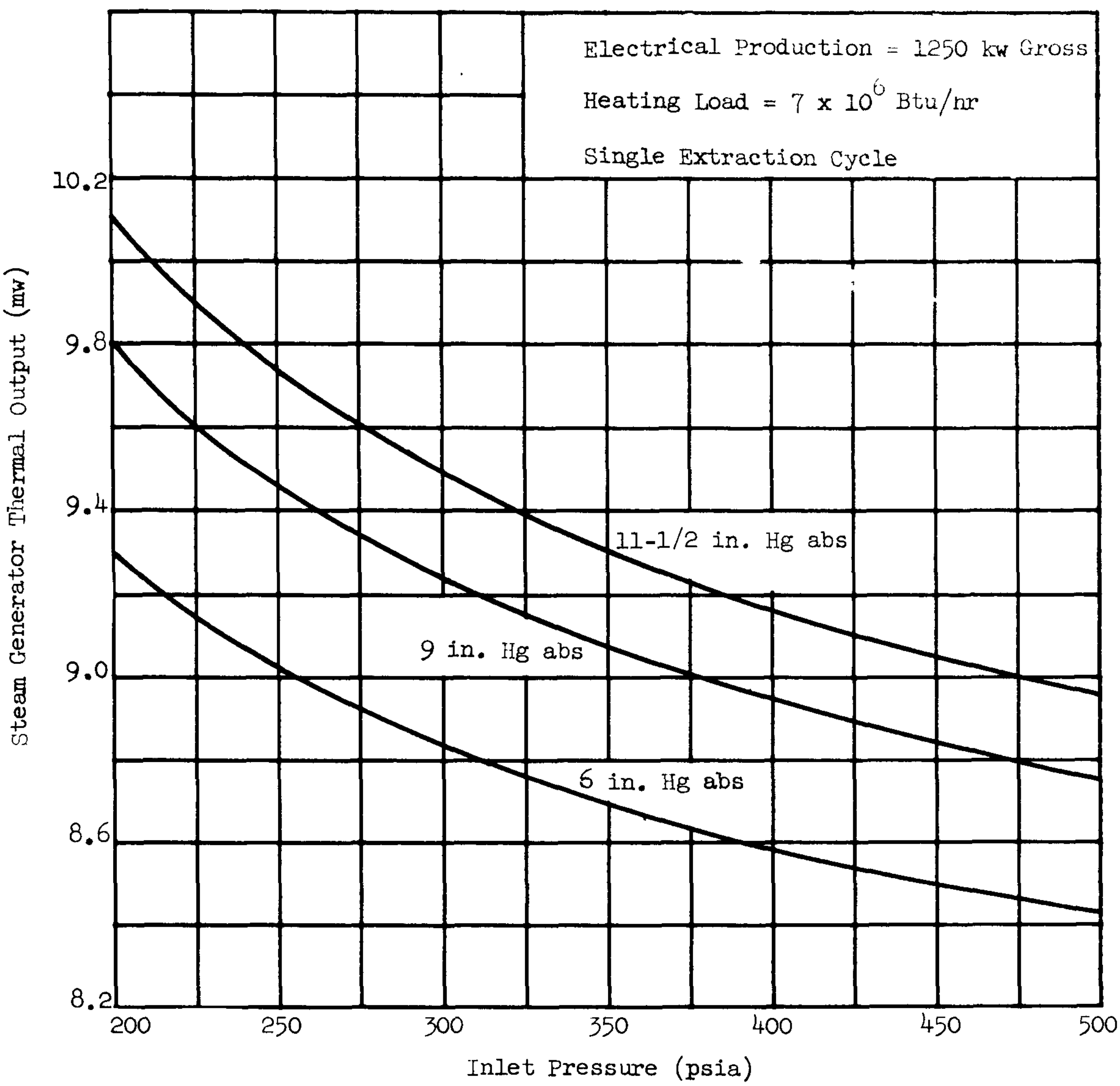


Fig. III-33. Steam Generator Thermal Output vs Inlet Steam Pressure--
1250 kw Geared Steam Turbine-Generator

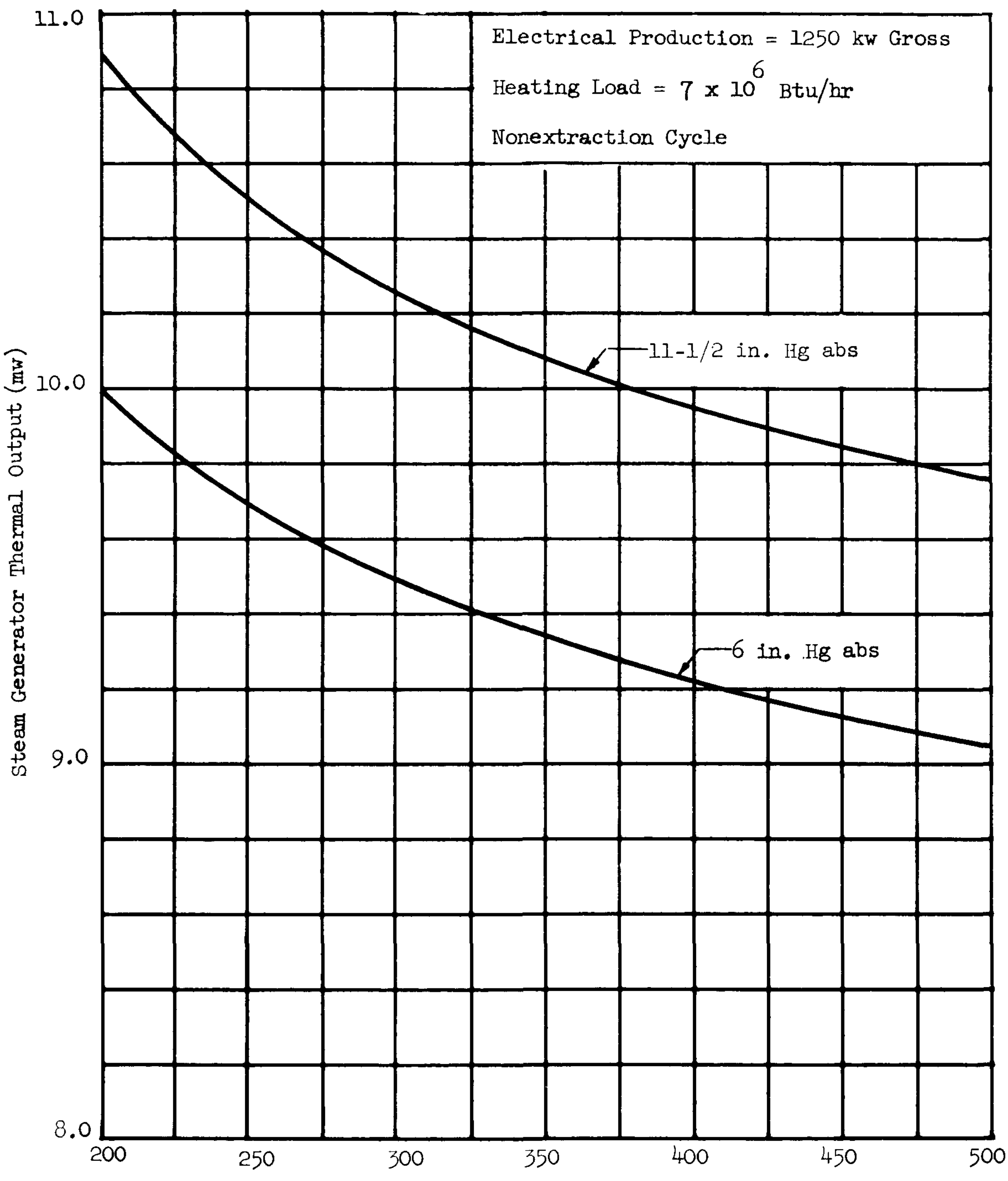


Fig. III-34. Steam Generator Thermal Output vs Inlet Steam Pressure--625 kw Geared Steam Turbine Generator

In addition to the thermal output requirements developed in this study, several cost and packaging aspects of the various cycles were noted. It was learned that with a turbine inlet steam pressure below about 275 psia and secondary loop auxiliary power limited to 125 kw, three instead of two air-steam condenser packages are required due to the higher heat rejection required and limitations on fan power. Table III-11 gives the number of packages and estimated cost variation for the cycles studied. The cost figures refer to variations from a fixed price quote on a two-turbine-generator system producing 1300 ekw gross, 1000 ekw net, and operating with 400 psia steam and 11-1/2 in. Hg abs turbine back pressure.

Analysis of Table III-11 indicated that the most interesting cycle, in terms of cost and number of packages, utilizes a single turbine operating with 275 psia, or higher, steam. It is advantageous, due to primary loop and steam generator considerations, to keep steam pressure as low as possible. It was, therefore, determined to study in greater detail cycles operating with 280 psia steam. Again, 6, 9 and 11-1/2 in. Hg abs turbine back pressures were used, with the gross output remaining at 1250 ekw; a comparison of electrically driven and steam turbine driven boiler feed pumps was included. Figures III-35 and III-36 show representative heat balances for this study. Table III-12 presents a summary of these heat balance studies.

The next area of interest was the determination of the minimum turbine back pressure that could be used without requiring more than two condenser packages; both the ethylene-glycol type condenser system and the direct air-steam system were considered. First, the required system auxiliary power for each back pressure was calculated, excluding the fan power. These power requirements are tabulated in Tables III-13, III-14 and III-15. Then the fan power required for a two-condenser arrangement was plotted against turbine back pressure as shown in Fig. III-37. The fan power requirements for the various systems were then included in Tables III-13 to III-15, together with the total auxiliary power requirement for each of these systems. It will be noted that in all the ethylene-glycol condenser systems and in the direct air-steam condenser system operating at 6 in. Hg abs back pressure and using a motor driven boiler feed pump, more than 125 kw of auxiliary power is required. Thus, 7 out of the 12 systems studied required increased auxiliary power if two-condenser packages were to be used. Figure III-38 illustrates this fact with a series of curves showing the cycles studied. The intersection of any cycle curve with the zero ordinate defines the minimum turbine back pressure that can be used with a two-condenser package (allowing 125 kw auxiliary power for the secondary loop). Figure III-39 shows the results of a similar study done with systems at 500 psia turbine throttle steam.

TABLE III-11

Secondary Loop Packages and Estimated Cost Variation

No. Turbine-Generator Sets	One Full-Capacity				Two Half-Capacity			
System Steam Pressure (psia)	200 to 275		275 to 500		200 to 275		275 to 500	
Turbine Cost Variation ($)	-131,000		-131,000		0		0	
No. of Condensers (air-to-steam type)	3		2		3		2	
Condenser Cost Variation ($)	+65,000		0		+65,000		0	
Extraction System Cost ($)	0	+ 7000	0	+ 7000	not considered	not considered	not considered	not considered
Packages Required (No.)	7	7	6	6	8	8	7	7
Package Cost Variation ($)	0	0	-5000	-5000	+5000	+5000	0	0
Total Cost Variation ($)	-66,000	-59,000	-136,000	-129,000	+70,000	+70,000	0	0

Area of greatest interest

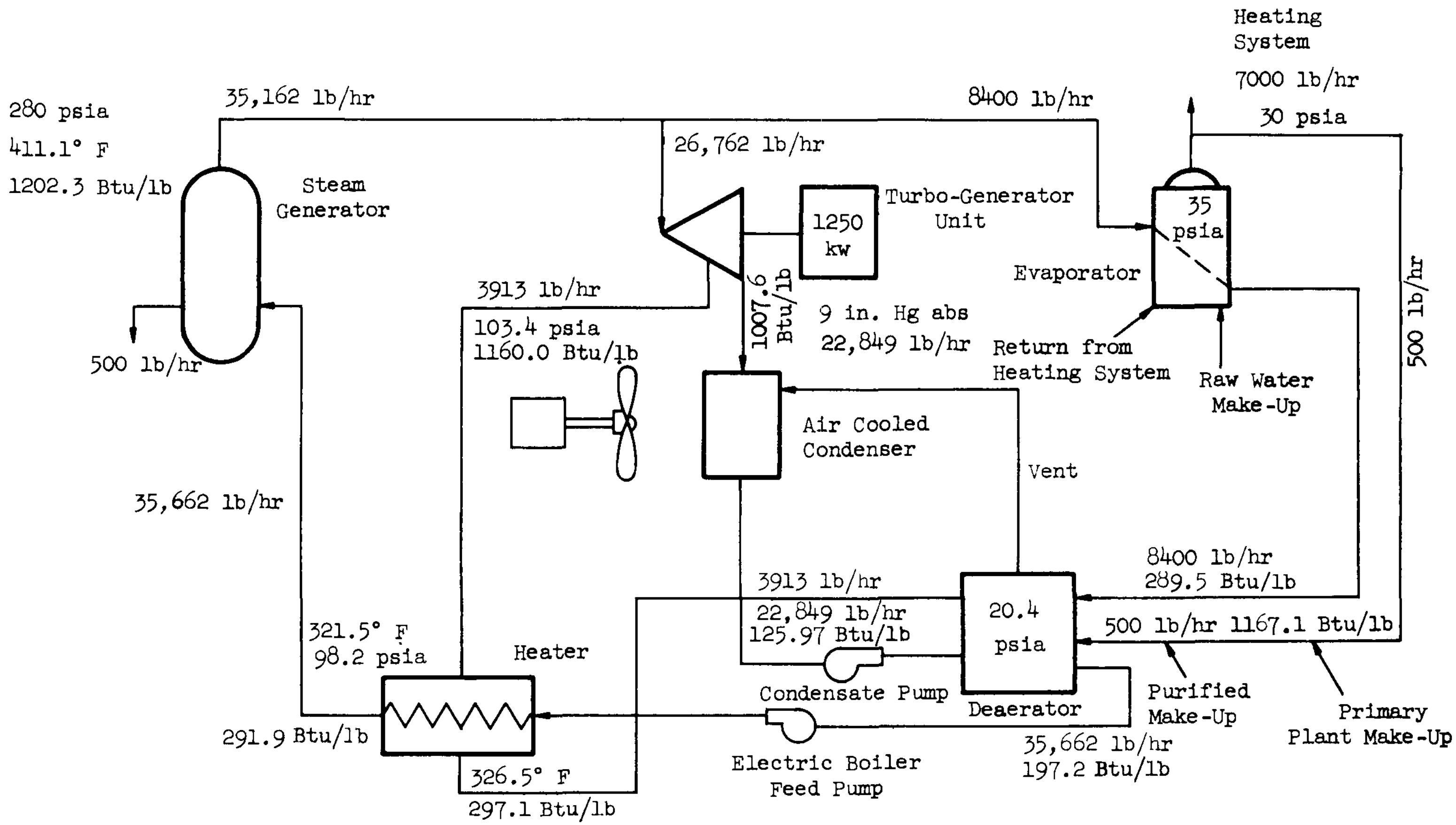


Fig. III-35. Geared Turbine Generator Extraction Unit--280 psia, Dry and Saturated, 9-in. Hg Abs, Electric-Driven Boiler Feed Pump

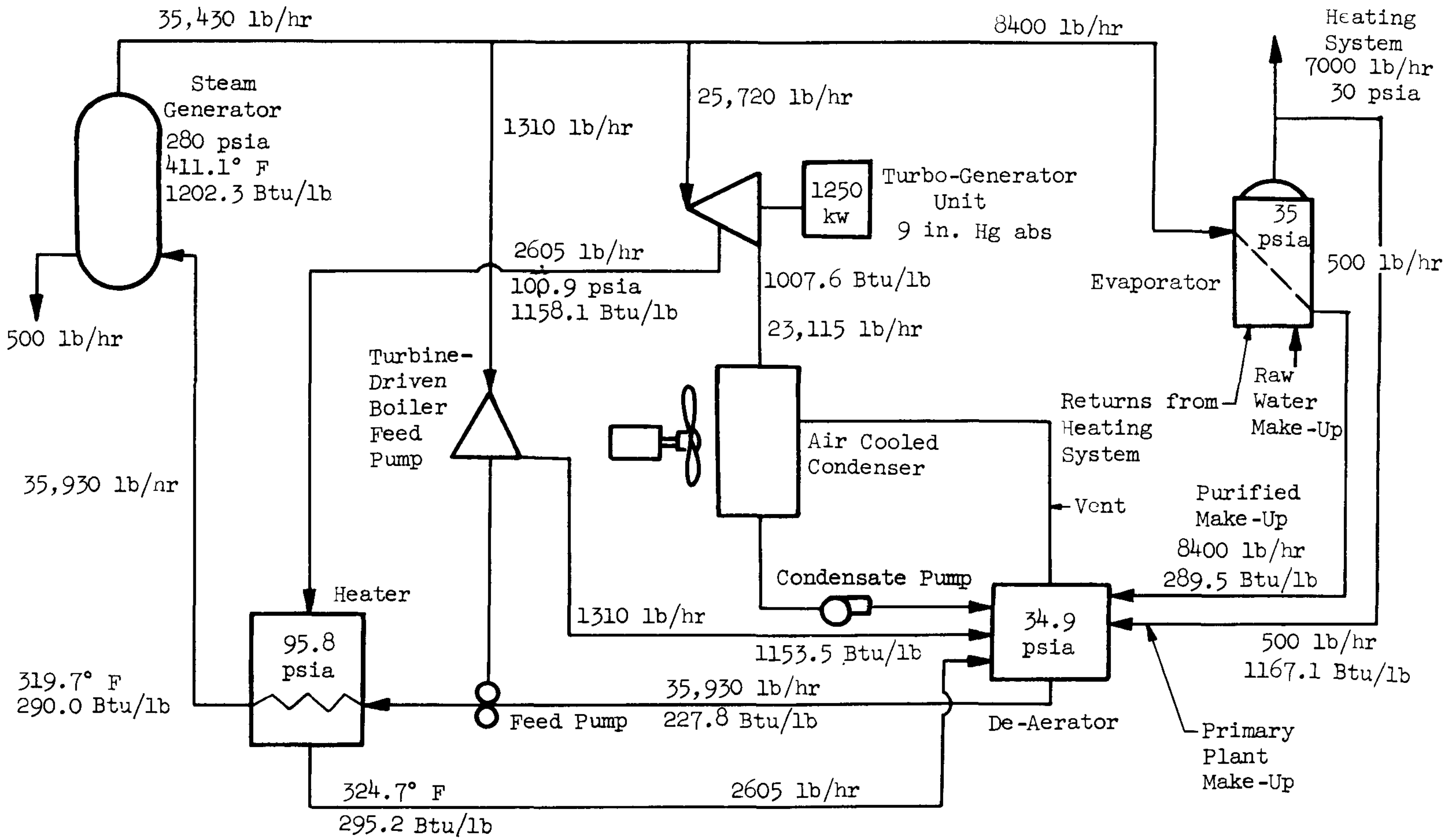


Fig. III-36. Geared Turbine Generator Extraction Unit--280 psia, Dry and Saturated, 9-in. Hg Abs, Steam Driven Boiler Feed Pump

TABLE III-12

Heat Balance Summary for 1250 Kw Plant

Steam Pressure (psia)	Condition Pressure (in. Hg abs)	Steam Generator Flow (lb/hr)	Turbine Flow (lb/hr)	Heat Rejected Btu/hr x 10^{-6}	Reactor Power (mw)	BFP
280	6	33,853	25,453	18.8	8.98	Elec
280	9	35,162	26,762	20.2	9.4	Elec
280	11.5	35,785	27,385	20.9	9.6	Elec
280	6	33,698	24,085	19.0	9.05	Turb
280	9	35,430	25,720	20.4	9.48	Turb
280	11.5	36,462	26,692	21.2	9.72	Turb

TABLE III-13

Auxiliary Power Required for a

280-psia 1250-kw Plant

with 6 in. Hg abs Back Pressure

Fan Power kw[a]	Motor Description	Auxiliary Power Required							
		Air-Cooled Condenser				Glycol-Cooled Condenser			
		Boiler Feed Pump Motor-Driven		Boiler Feed Pump Turbine-Driven		Boiler Feed Pump Motor-Driven		Boiler Feed Pump Turbine-Driven	
		kw[a]	Instld (HP)	kw[a]	Instld (HP)	kw[a]	Instld (HP)	kw[a]	Instld (HP)
	Condensate Pump	0.88	1.0	0.88	1.0	0.88	1.0	0.88	1.0
	Air Pumps	13.2	2-7.5	13.2	2-7.5	13.2	2-7.5	13.2	2-7.5
	Boiler Feed Pump[b]	21.1	30	--	--	21.1	30	--	--
	Evaporator Feed Pump	0.80	1.0	0.80	1.0	0.80	1.0	0.80	1.0
	Heating, Lighting, Aux, etc.	10.0	--	10.0	--	10.0	--	10.0	--
	Glycol Circulating Pump	--	--	--	--	59.0	75.0	59.0	75.0
	Total	46.0	47.0	24.9	17.0	105.0	122.0	83.9	92.0
Available[c]		79.0	--	100.1	--	20.0	--	41.1	--
Required		100.0	--	100.0	--	150.0	--	150.0	--
Excess		--	--	0.1	--	--	--	--	--
Deficiency		21.0	--	--	--	130.0	--	110.0	--

[a] Based on Motor Efficiency 85%

[b] Based on Pump Efficiency 46%

[c] Based on 125 kw Available for Station Auxiliary

TABLE III-14

Auxiliary Power Required for a
280-psia 1250-kw Plant
with 9 in. Hg abs Back Pressure

Fan Power kw[a]	Motor Description	Auxiliary Power Required							
		Air-Cooled Condenser				Glycol-Cooled Condenser			
		Boiler Feed Pump				Boiler Feed Pump			
		Motor -Driven		Turbine -Driven		Motor -Driven		Turbine -Driven	
		kw[a]	Instld (HP)	kw[a]	Instld (HP)	kw[a]	Instld (HP)	kw[a]	Instld (HP)
	Condensate Pump	0.88	1.0	0.88	1.0	0.88	1.0	0.88	1.0
	Air Pumps	13.2	2-7.5	13.2	2-7.5	13 .2	2-7.5	13.2	2-7.5
	Boiler Feed Pump[b]	21.9	30.0	--	--	21.9	30.0	--	--
	Evaporator Feed Pump	0.80	1.0	0.80	1.0	0.80	1.0	0.80	1.0
	Heating, Lighting, Aux, etc.	10.0	--	10.0	--	10.0	--	10.0	--
	Glycol Circulating Pump	--	--	--	--	59.0	75.0	59.0	75.0
	Total	46.8	47.0	24.9	17.0	105.8	122.0	83.9	92.0
Available[c]		78.2	--	100.1	--	19.2	--	41.1	--
Required		71.0	--	71.0	--	88.0	--	88.0	--
Excess		7.2	--	29.1	--	--	--	--	--
Deficiency		--	--	--	--	68.8	--	46.9	

[a]Based on Motor Efficiency 85%

[b]Based on Pump Efficiency 46%

[c]Based on 125 kw Available for Station Auxiliary

TABLE III-15

Auxiliary Power Required for a
280-psia 1250-kw Plant
with 11.5 in. Hg abs Back Pressure

Fan Power kw[a]	Motor Description	Auxiliary Power Required							
		Air-Cooled Condenser				Glycol-Cooled Condenser			
		Boiler Feed Pump Motor-Driven		Boiler Feed Pump Turbine-Driven		Boiler Feed Pump Motor-Driven		Boiler Feed Pump Turbine-Driven	
		kw[a]	Instld (HP)	kw[a]	Instld (HP)	kw[a]	Instld (HP)	kw[a]	Instld (HP)
	Condensate Pump	0.88	1.0	0.88	1.0	0.88	1.0	0.88	1.0
	Air Pumps	13.2	2-7.5	13.2	2-7.5	13.2	2-7.5	13.2	2-7.5
	Boiler Feed Pump[b]	22.2	30.0	--	--	22.2	30.0	--	--
	Evaporator Feed Pump	0.80	1.0	0.80	1.0	0.80	1.0	0.80	1.0
	Heating, Lighting, Aux, etc.	10.0	--	10.0	--	10.0	--	10.0	--
	Glycol Circulating Pump	--	--	--	--	59.0	75.0	59.0	75.0
	Total	47.1	47.0	24.9	17.0	106.1	122.0	83.9	92.0
Available[c]		77.9	--	100.1	--	18.9	--	41.1	--
Required		61.5	--	71.5	--	75.0	--	75.0	--
Excess		16.4	--	38.6	--	--	--	--	--
Deficiency		--	--	--	--	56.1	--	33.9	--

[a]Based on Motor Efficiency 85%

[b]Based on Pump Efficiency 46%

[c]Based on 125 kw Available for Station Auxiliary

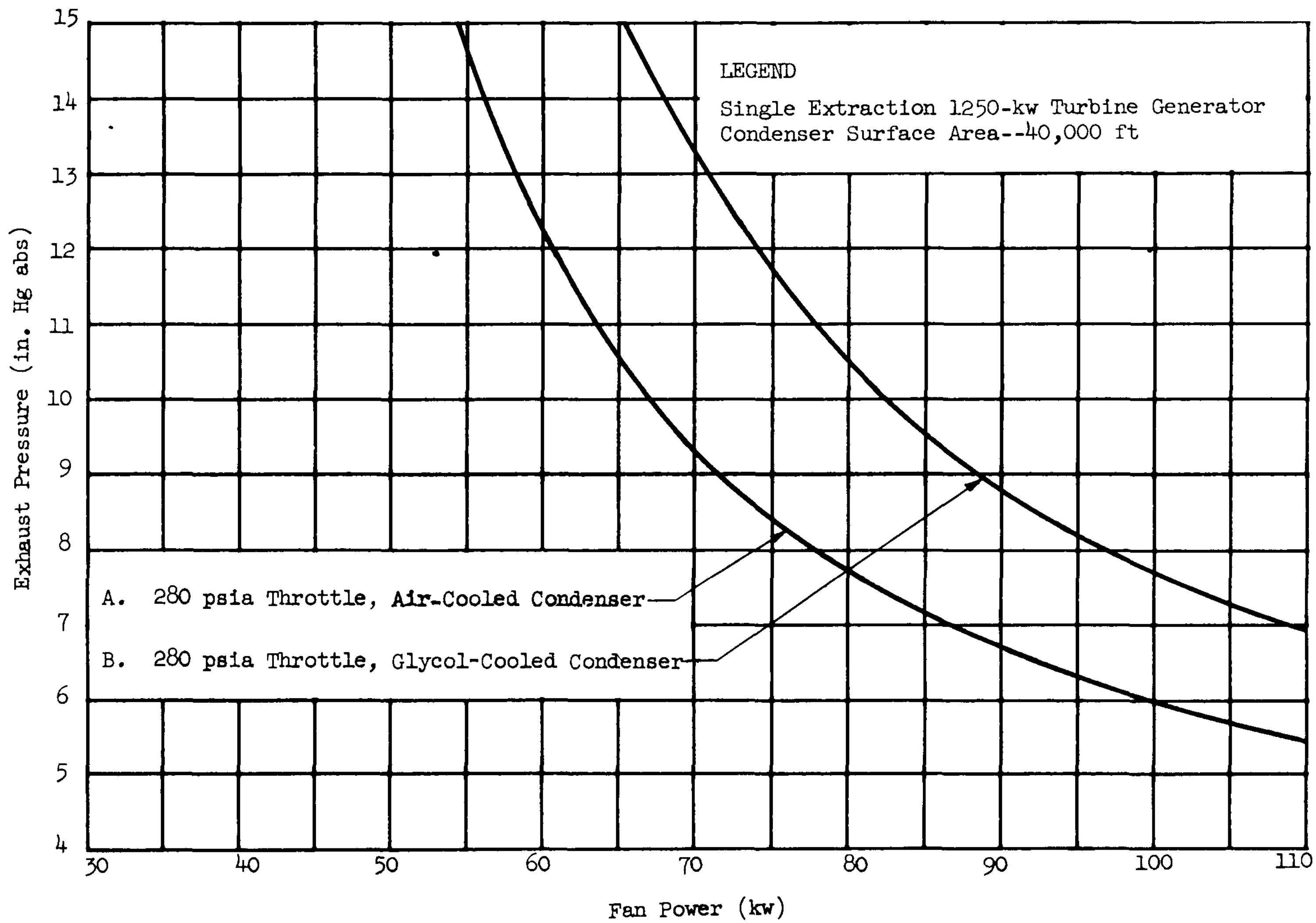


Fig. III-37. Effect of Exhaust Pressure on Fan Power for Air and Ethylene Glycol-Cooled Condensers

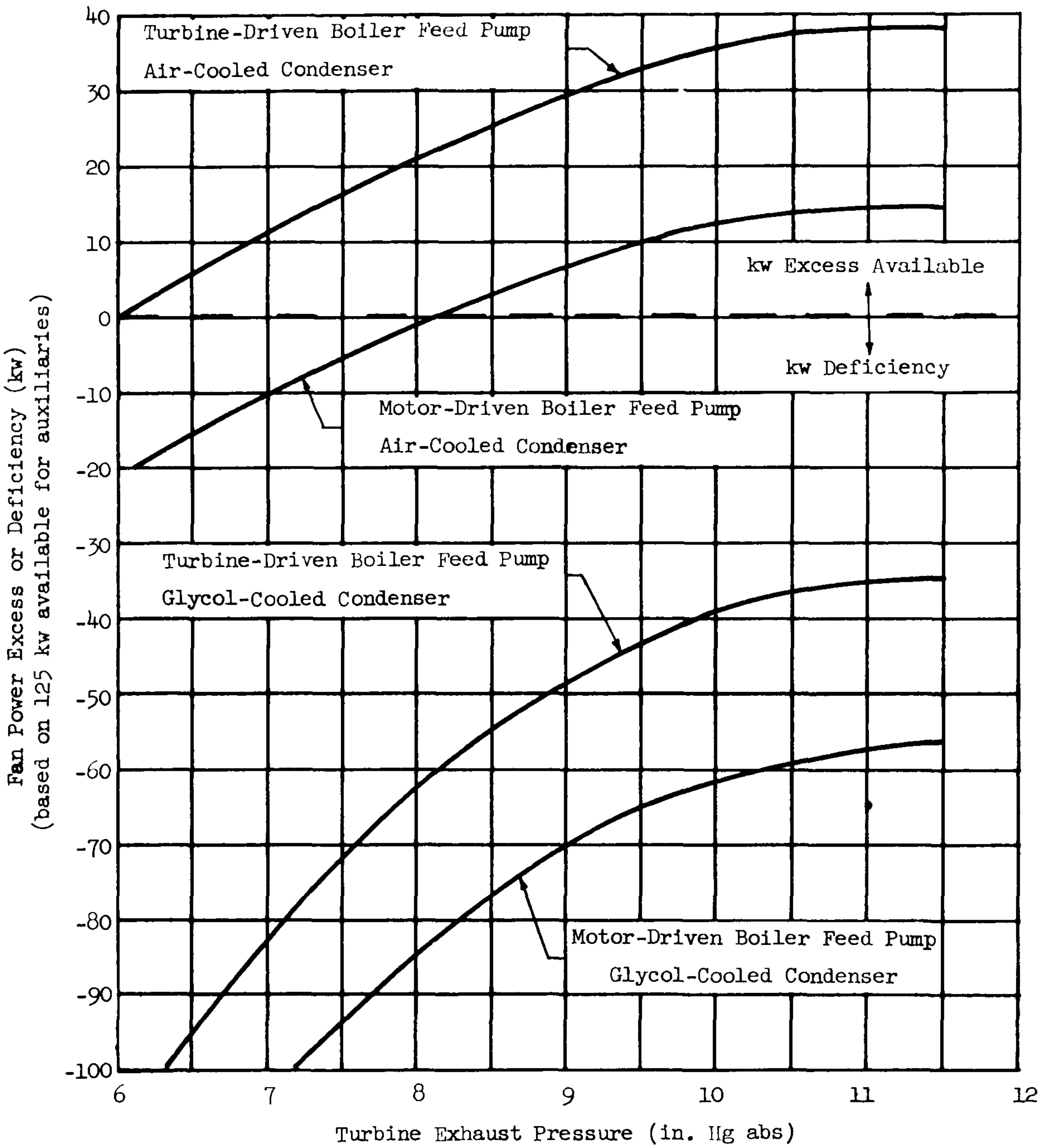


Fig. III-38. Fan Power Requirements--Single Extraction, 1250 kw Plant, 280 psia Turbine Throttle Pressure

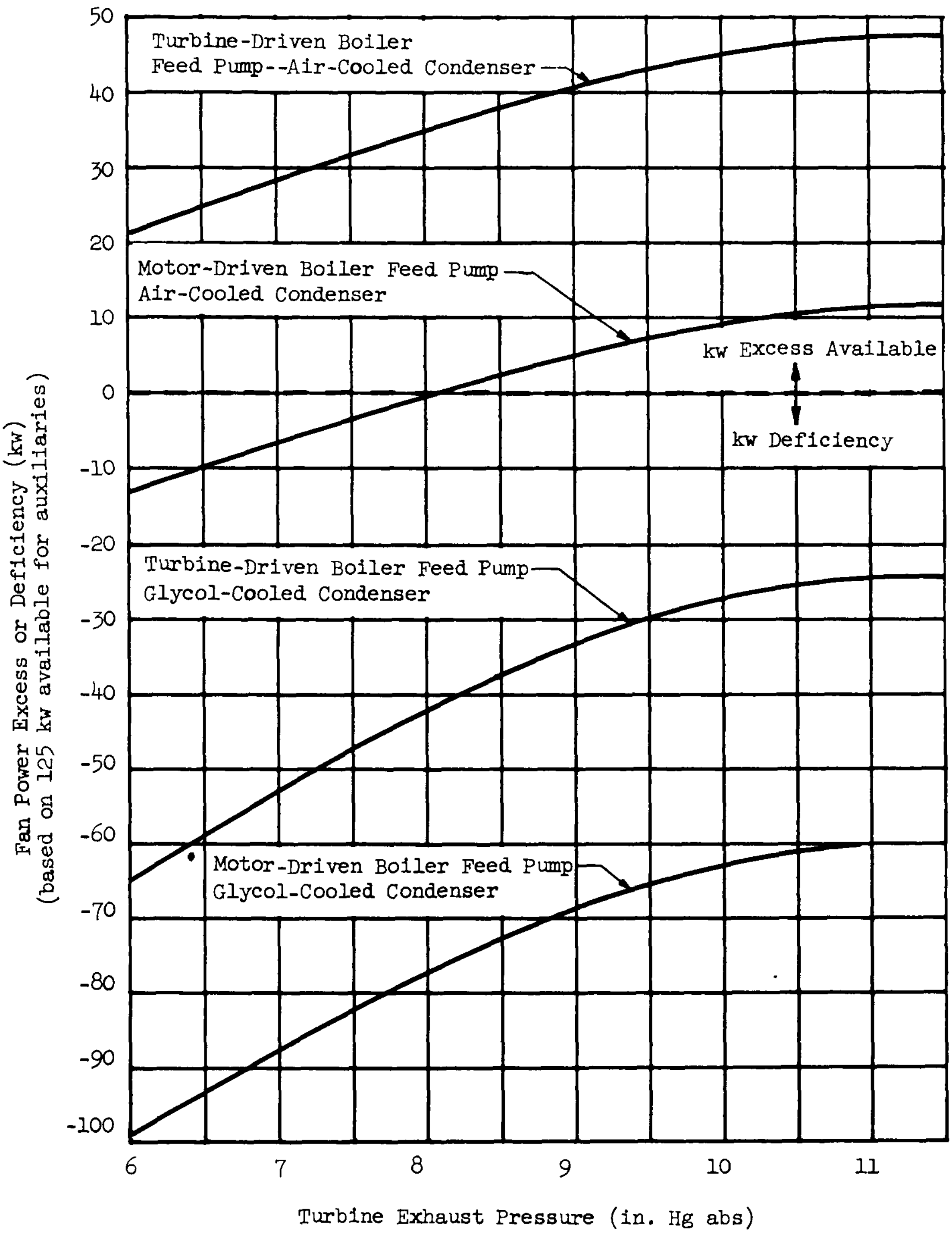


Fig. III-39. Fan Power Requirements--Single Extraction, 1250 kw Plant, 500 psia Turbine Throttle Pressure

Another study of the use of a single air-steam condenser package with a 150-kw auxiliary power secondary loop is nearing completion. It appears that, in order to maintain a condenser LMTD great enough to limit fan power to within auxiliary power limitations, turbine back pressures of between 15 and 20 psia will be required. These same preliminary results indicate that the primary loop thermal requirements will be increased by from 20 to 30%. A possible alternative is the use of a novel condenser design in which some 50,000 sq ft of surface area is housed in a single package; size, weight, and fan power requirements are being investigated for such a condenser.

Steam generator studies.- The parametric study of steam generators was completed. This study was to evaluate the effect of steam pressure, primary loop flow, log mean temperature difference, and tube material on cost, size, weight, and primary side pressure drop.

Preliminary primary loop layouts indicated that the horizontal type of steam generator offered no advantage. The cost and weight of a horizontal unit are significantly greater than for vertical units. Table III-16 shows a comparison between a horizontal and vertical steam generator suitable for the PM-1 plant.

TABLE III-16

Type	Horizontal	Vertical
TMW output	10	10
Design pressure--primary side (psi)	2,300	2,300
Design pressure--secondary side (psi)	1,125	1,125
Coolant flow (lb/hr)	982,000	982,000
Steam pressure at full load (psig)	400	400
Weight (dry) (lb)	20,000	11,000
Estimated cost ($)	125,000	75,000

For these reasons, studies were limited to vertical steam generators capable of functioning within the range of operating conditions considered during the primary and secondary loop studies. To facilitate this work, the following conditions were held constant:

Thermal output	9.5 mw--middle of range
Shell and channel ID	28-5/8 in.--minimum for maintenance access
Tube OD	1/2 in.--best over full range
Design pressure primary side	2300 psi--conservative basis
Number of tubes	420--use available shell ID
Access to primary and secondary side (2 each)	bolted-type hand holes
Inlet and outlet nozzle velocity--primary side	20 fps--upper limit
Type steam drum	integral
Moisture separators	3-stage integral
Material	304 SS (tube side) carbon steel (shell side)

Table III-17 presents the results of this study.

Stainless steel steam generators have been considered to date. The reliability of this type of steam generator is high, as demonstrated by the fact that no tube leaks have been experienced with the many steam generators supplied to date. Also, the bolted hand holes permit removal, replacement, and sealing on the primary side in 4 hr and on the secondary side in 1-1/2 hr.

The use of Inconel tube and tube sheet construction is also being considered. A study of water quality requirements and Inconel fabrication methods is underway.

Present experience with three-stage moisture separators indicates that the moisture content at the steam drum outlet under steady state conditions is less than 1/4%*. Tests also show that with a transient of from 15 to 100 % in 3 sec, the moisture content of the steam remains below 1/2 %*.

*R. L. Cort and C. C. Peake, "Performance Tests of a Vertical Steam Generator for Nuclear Power Plants," ASME Paper No. 58-PWR-1, Westinghouse Electric Corporation.

TABLE III-17

PM-1 Steam Generator Study

Steam Pressure	304 SS Tubes--Estimated Price ($)	Inconel Tubes--Estimated Price ($)	Primary Flow (gpm)	LMTD (°F)	Shell Design Pressure (psi)	Surface Area (sq ft)	Pressure Drop (psi)	Overall Length (ft)	Weight (lb)
250 psia 401 °F	86,500	95,200	1000	30	500	1840	6.9	23.5	13,300
	73,700	81,100		65	700	780	5.0	13.9	9,400
	69,100	76,000		100	900	475	4.4	11.1	8,000
	84,200	92,600	1700	30	500	1630	12.1	21.6	12,400
	72,700	80,000		65	700	700	7.7	13.1	9,000
	68,500	75,400		100	900	430	6.4	10.7	7,800
	81,900	90,000	2500	30	500	1520	20.2	20.6	10,900
	72,100	79,400		65	700	660	12.0	12.8	8,800
	68,100	75,000		100	900	410	9.7	10.5	7,700
375 psia 438 °F	87,500	96,300	1000	30	700	1840	6.9	23.2	14,500
	74,000	81,500		65	900	780	5.0	13.5	9,700
	70,500	77,600		100	1200	520	4.7	11.2	9,200
	73,000	80,300	1700	65	900	675	14.6	19.1	11,500
	85,000	93,500		30	700	1630	12.1	21.3	13,500
	72,700	80,000		65	900	700	7.7	12.8	9,200
	69,700	76,700		100	1200	470	7.0	10.7	8,900
	96,400	106,000	2500	20	600	2540	30.2	29.6	16,900
	80,200	88,300		30	700	1520	20.5	20.3	12,900
	72,100	79,400		65	900	660	12.4	12.5	9,000
	69,500	76,500		100	1200	460	11.3	10.6	8,800
500 psia 467 °F	87,900	96,700	1000	30	800	1840	6.9	23.0	14,900
	75,000	82,500		65	1200	800	5.2	13.4	10,900
	71,700	79,000		100	1500	550	5.1	11.2	10,500
	85,300	94,000	1700	30	800	1630	12.1	21.0	13,700
	73,600	81,000		65	1200	720	8.3	12.7	10,400
	70,500	77,600		100	1500	500	8.2	10.7	9,900
	83,800	92,100	2500	30	800	1520	20.2	20.0	13,200
	73,000	80,400		65	1200	700	13.5	12.5	10,300
	70,100	77,200		100	1500	480	12.6	10.5	9,700

Turbine generator studies.- The basic ground rules for turbine generator design and weight evaluation were that a gross output of 1250 ekw would be provided and that 300 psia throttle pressure and 9 in. Hg abs back pressure would be utilizied. All turbines are designed to withstand zero load steam conditions at the steam chest. With these parameters established, four different turbine-generator sets were investigated:

(1) A 1250-kw machine using an 8000 to 9000 rpm turbine and a 1200 rpm salient pole generator. With this system, a 6% voltage dip accompanies the application of a 300-kw step load transient.

(2) A 1250-kw machine similar to Item 1, but with an oversized generator yielding a 2% voltage dip with application of a 300-kw step load transient.

(3) A 625-kw machine, designed to the conditions of Item 1.

(4) A 625-kw machine, designed to the conditions of Item 2.

The results of this study are given in Table III-18.

TABLE III-18

Estimated Turbine-Generator Weights

1250-kw Capacity Machine

	System (1) (lb)	System (2) (lb)
Generator	9,400	18,000
Turbine	4,736	4,736
Gear	3,000	3,000
Oil cooler	320	320
Bedplate	2,160	2,160
Miscellaneous	658	658
Total	20,274 lb	28,874 lb

TABLE III-18 (continued)

625-kw Capacity Machine

	System (3) (lb)	System (4) (lb)
Generator	6,350	10,400
Turbine	4,280	4,280
Gear	1,250	1,250
Oil cooler	405	405
Bedplate	1,520	1,500
Miscellaneous	2,110	2,110
Total	15,915 lb	19,945 lb

These weights are the result of preliminary investigations; further reduction may result after detailed design. The largest machine is approximately 5 ft by 5 ft by 16 ft long.

Other weight reduction methods being studied include reducing the generator weight by using a 3600-rpm nonsalient pole machine, meeting the weight limitation on the 2% voltage fluctuation through attachments external to the electric generator (see Subtask 1.6), and using a planetary reduction gear rather than the standard gear and pinion normally used. Although the planetary gear for the 1250-kw machine is estimated to weigh 1700 lb, the entire difference between it and the 3000-lb standard gear cannot be realized due to a required change in the turbine-bearing bracket design. A comparatively small amount of operating data is available; it is, however, worthy of note that the N. S. Savannah will use a planetary gear with its 3500-kw generator. The use of lightweight materials in the bedplate and other parts of the machine is being studied.

It should be noted that the weights given in Table III-18 do not include the exciter. Static-type exciters presently used by the Navy (450 volts) can readily be adapted to this application and can be installed in the switch gear area. This not only eases the problem of generator weight, but also helps to achieve the desired power quality due to faster response time than those attainable with a rotating exciter.

Effects of number of turbine-generators--A single turbine-generator system offers sufficient advantage over a parallel system to warrant its choice, even if it is found that aircraft loading requirements necessitate shipping the generator separately. The assembly of the machine at the site would require only one or two extra days, and this is more than compensated for by the simplified piping and electrical connections. Some of the advantages to be realized by using a single machine are as follows:

(1) Cost savings of approximately $130,000 are anticipated.

(2) Overall system weight is less.

(3) Greater full-load reliability is believed attainable since there are fewer parts to fail, and the failure of one half-capacity machine under full load would probably result in failure of the second.

(4) Fewer piping and wiring connections need be accommodated.

(5) Operation is simpler.

(6) Switchgear, cable, conduits, and controls are simple and less costly.

(7) Efficiency is higher.

Steam pressure effects on weight--Recent investigations indicate that throttle pressure variations of 200 to 500 psia, and exhaust pressure variations of 6 to 11-1/2 in. Hg abs, are accompanied by less than a 5% variation in secondary system weight. Figures III-40 and III-41 show variations of inlet size with inlet pressure for 625- and 1250-kw turbines respectively. Figures III-42 and III-43 show variations of exhaust size with inlet steam pressure for 625- and 1250-kw turbines.

Moisture effects--Figure III-44 shows the variation of inlet pressure with percent moisture in the exhaust steam for various 1250-kw turbine back pressures. No erosion problem is anticipated since 12 to 13% chrome steel will be used in all metal parts in the steam path and the rim velocities will be approximately 800 fps. Considerable experience has been accumulated within the last five years for turbines operating under these moisture conditions. The turbine of a similar power plant has been in service for 5 yr; erosion has not been noted on any of the chrome steel parts, although there has been erosion of the carbon steel parts located in the steam path. Another turbine has been in operation for 2 yr under conditions of 11 to 12% exhaust moisture. Again, no problem of erosion has been experienced on the chrome steel turbine parts. The Shippingport turbine has a 1200-fps rim velocity with chrome steel parts in the steam path. To date, this installation has been free of erosion problems.

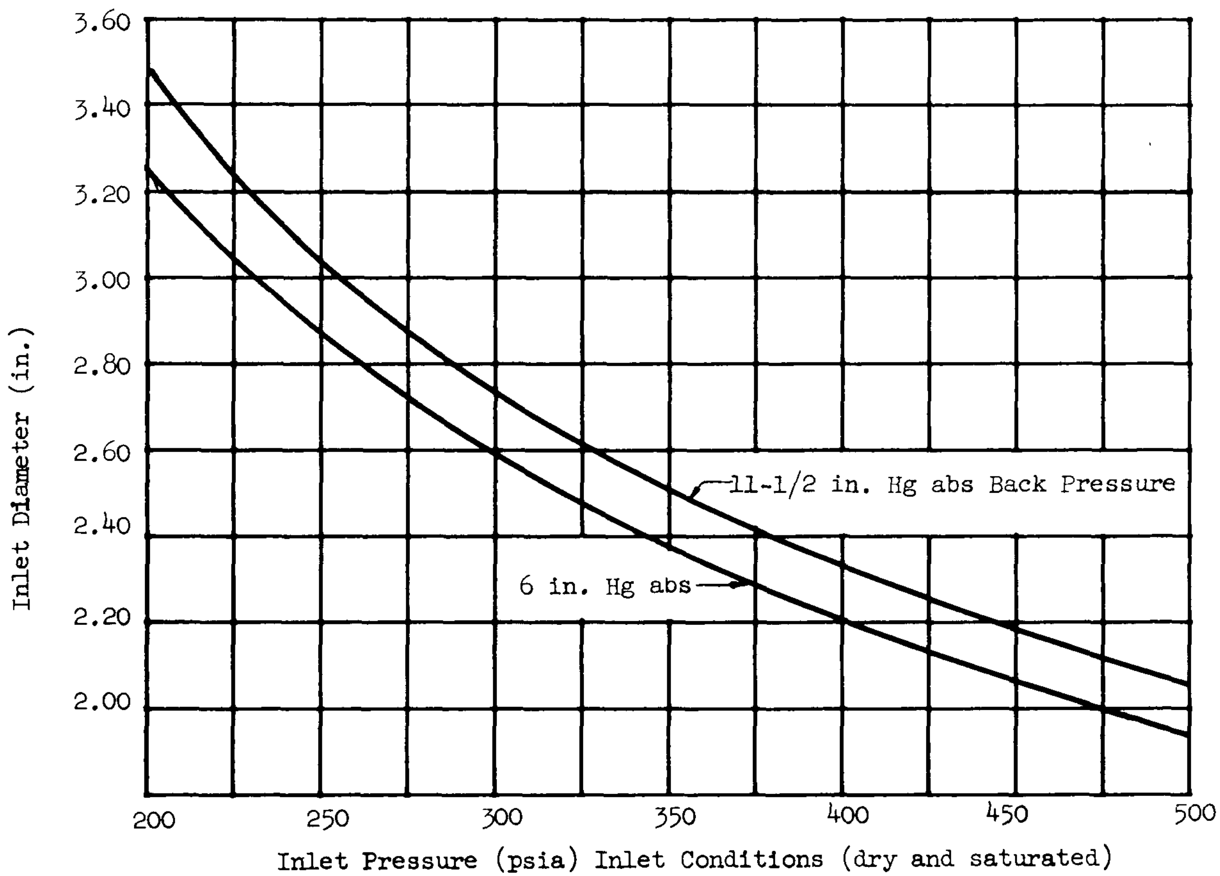


Fig. III-40. Inlet Diameter vs Inlet Steam Pressure (625 kw Geared Steam Turbine-Generator Unit)

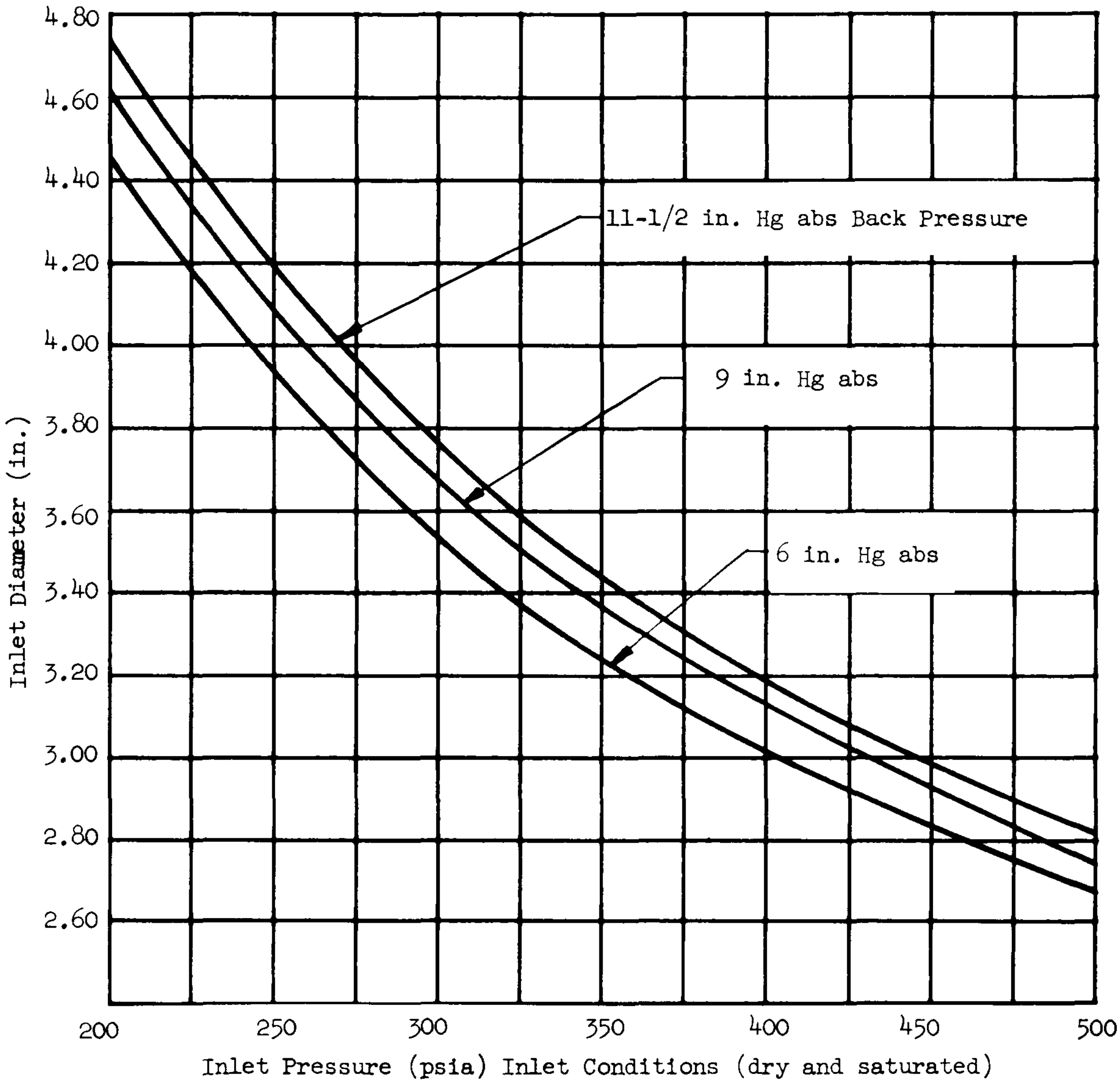


Fig. III-41. Inlet Diameter vs Inlet Steam Pressure (1250 kw Geared Steam Turbine-Generator)

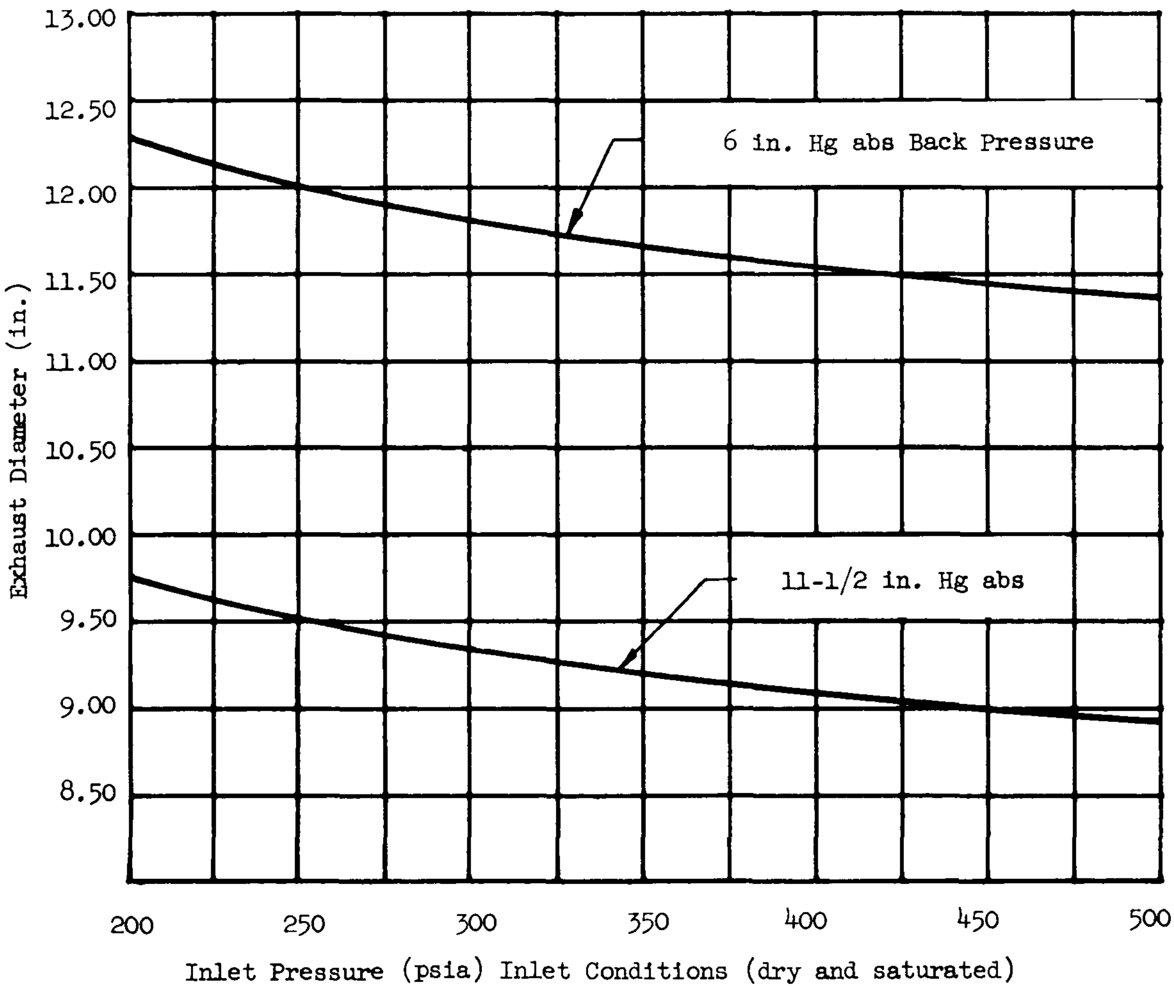


Fig. III-42. Exhaust Diameter vs Inlet Steam Pressure (625 kw Geared Steam Turbine Generator Unit)

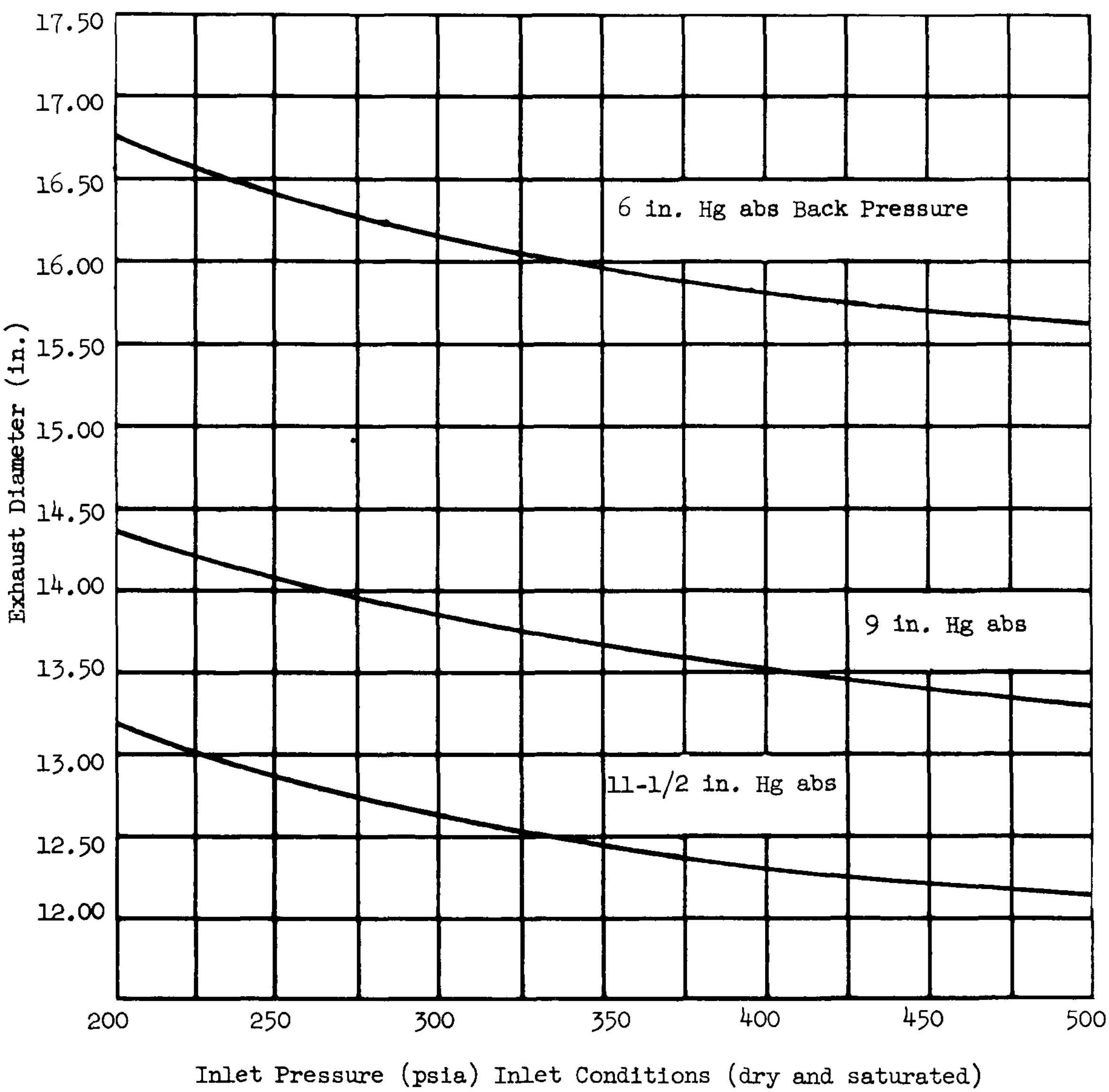


Fig. III-43. Exhaust Diameter vs Inlet Steam Pressure--1250 kw Geared Steam Turbine Generator Unit

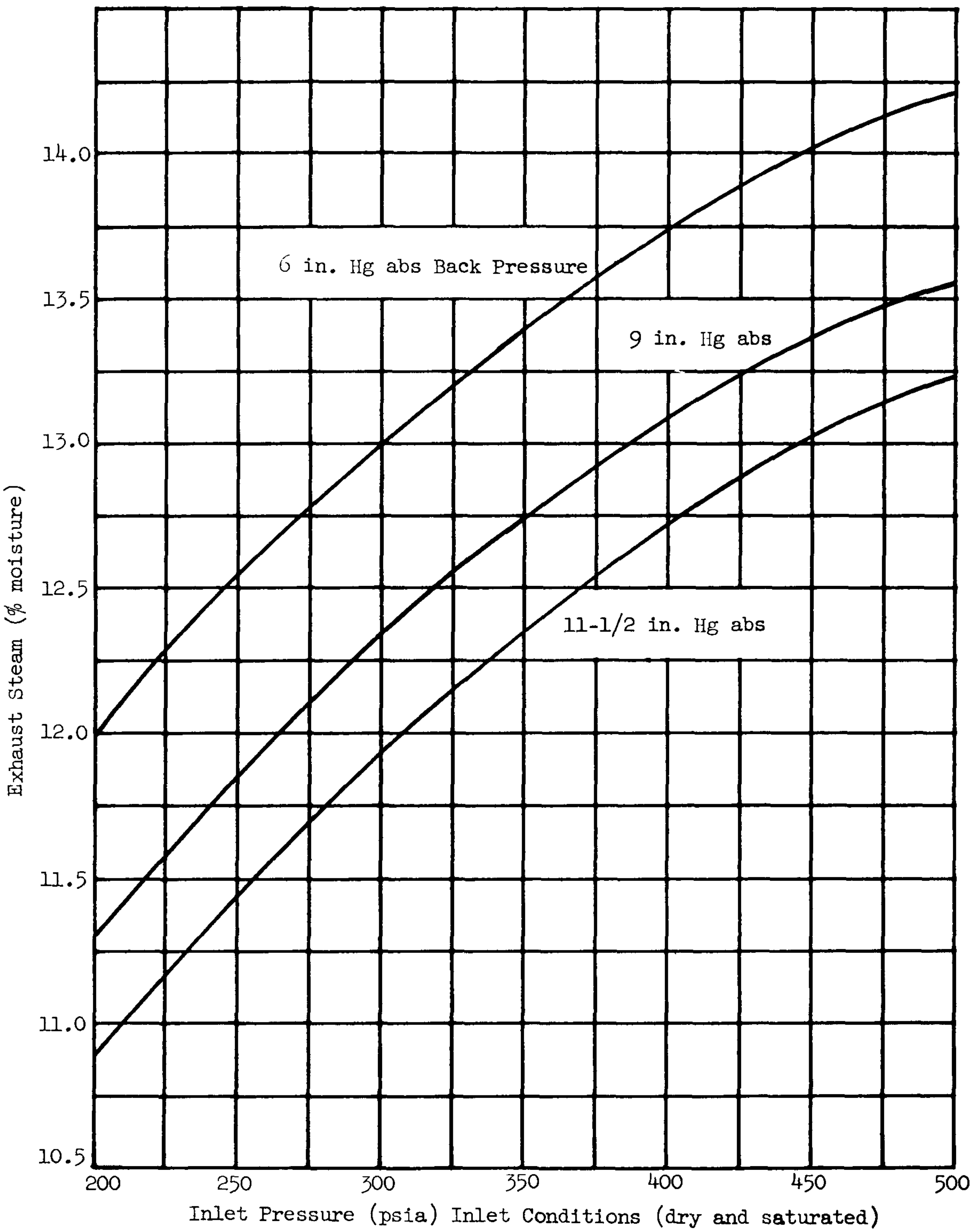


Fig. III-44. Inlet Steam Pressure vs Percent Moisture in Exhaust Steam (1250 kw Turbine Generator)

Elevation effects--The design conditions given in the contract for the PM-1 plant call for 760 mm Hg pressure (sea level). The plant installation at the Sundance site will be at an elevation of approximately 6500 ft and, therefore, electric generator performance will suffer due to poorer generator cooling system performance. Generator capacity drops by 1% for every 330 ft above 3300 ft elevation or 9.6% for this application. Gross generation will, therefore, be reduced by 120 kw, or from 1250 to 1130 kw. A study of the effect of altitude on condenser performance is underway and will be completed by the end of the parametric study; a similar drop in capacity is anticipated.

Foundation study--The turbine generator foundation and mounting requirements are being investigated; this study will also be completed by the end of the parametric study period.

Miscellaneous studies.-

Switch gear and motor control center--At present it appears that 35 units (including primary loop equipment) are required in the PM-1 motor control center. An investigation of both steel and aluminum construction shows that a significant weight saving can be achieved.

TABLE III-19

Comparison of Steel and Aluminum Construction in Motor Control Center

Type of Mounting	Front Mounted Sections	Front and Rear Mounted Sections
Weight per section with steel fabrication (less internals) (lb)	450	595
Weight per section with aluminum fabrication (less internals) (lb)	165	210
Number required	7	3 plus one 5-unit section
Total steel structure weight (lb)	3150	2235
Total aluminum structure weight (lb)	1155	795
Total weight savings with use of aluminum structure (lb)	1995	1440

The type of mounting (front or front and rear) will be determined by the final switchgear-control center package layout.

The question of whether 480 or 4160 volt generators yield a more economical, lightweight system is being investigated. Present indications are that the 4160-v system is better. Although the switchgear for the 4160-v system costs $12,000 more and weighs more than that for the 480-v system, this apparent disadvantage is balanced by the additional cost and weight of the 480- to 4160-v, 1250-kva, step-up, load transformer required. The 4160-v system, on the other hand, requires the provision of a 300-kva, 4160 to 480-v, step-down transformer for the generation of plant auxiliary power. The latter transformer can be located in a cubicle in the switchgear package. Weight, in the size range studied, is not appreciably affected by voltage.

Auxiliary power supply equipment--Auxiliary power requirements for plant startup and shutdown will be resolved during the preliminary design. Auxiliary power plants from 100- to 250-kw capacity, including both high speed diesel and gas turbine-generator systems are being investigated. The required fuel storage tanks, starting equipment, controls, cooling system, and lubrication system are being considered.

Lead-acid and nickel-cadmium battery emergency power supply systems are being compared with regard to cost, weight, and maintenance requirements. The battery system is being sized at 6-kw output for a 1/2-hr period.

Piping and wiring connections--Various types of quick connect-disconnect pipe couplings such as those manufactured by the Barco, Victaulic, and Dresser Companies are being investigated. No particular problem is anticipated in this area, due to the relatively low pressures and temperatures involved. Various types of wiring are also being investigated and again no problems are anticipated with 480-v or less. Present indications are that no suitable quick connector is available for 5-kv insulated cable.

Heat exchange and auxiliary equipment--Several methods of reducing heat exchanger (evaporator, deaerator, reboiler, and closed feed-water heater) weight and size are being investigated. These include the use of lightweight materials, and the use of a single heat exchanger to serve as evaporator for makeup water and as the process heat re-boiler.

A vacuum pump system is being evaluated to replace the steam-jet air ejectors. The purpose of this substitution would be to decrease installation time, to simplify piping and controls, and to simplify operation during startup and shutdown. The reliability and maintenance of the rotary-type vacuum pump will be evaluated.

The use of motors with aluminum frames, to reduce overall plant weight, is being evaluated. Motors from 1 to 20 hp are now available.

Plant arrangement and service requirements--A component and package arrangement study is in process. It appears that six packages (including the central control room package) will be required for the secondary loop equipment. In conjunction with the study, an investigation is being made of the heating, ventilating, lighting, sewerage, storage area, and maintenance area requirements of the secondary loop.

A complete list of required maintenance tools is being developed and will be completed during preliminary design.

Reliability study--A reliability study of the secondary loop is underway. The basic data of this investigation were reported in "Forced Outage Rates of High Pressure Steam Turbines and Boilers," by the Joint AIEE Subcommittee, Report No. 57-145, June 1957, which covers 880 steam electric plants of 10,000 ekw capacity or less. Present data for the PM-1 system indicates that less than seven unscheduled and six scheduled downtime days per year can be expected from this system. The first phase of this reliability study will be completed during the parametric study period.

3. Instrumentation

G. Zindler R. Caw E. Gasser

During the first quarter, automatic reactor startup and system warmup control systems were studied with the assistance of an analog computer. The objectives of this study were to determine the instrumentation requirements of such a system and to evaluate the system behavior as a function of the input control parameters. This work also supports the subcontract efforts under Task 1.5.

An analog program was developed that allowed study of reactor variables over about 12 decades of operation, of primary loop heating over a temperature range of 0-600°F, and of various rod control circuits used to initiate startup.

Analog runs were made to study the effects of such variables as rod speed, temperature coefficient of reactivity, water heating rate, and various set-points for period control.

Sufficient data for a complete presentation have not yet been obtained. Preliminary analog results show good operation of the system with the following control system equipment:

(1) A period control circuit that governs rod withdrawal until a power level sufficient to cause heating of primary loop water is reached. A minimum period of 13 sec appears adequate for safe operation with the constant speed rod actuators simulated in this study. To help eliminate excessive rod "jogging" due to period noise and the fact that the period changes rapidly near criticality, it is recommended that a dead interval of 30 or more sec be incorporated, so that rod withdrawal may be halted after 13 sec and initiated 40 to 45 sec later.

(2) During the approach to criticality, the period withdrawal circuits are monitored by a circuit which measures rate of temperature increase. This circuit will assume rod control when heating takes place, and will be in control from that point on.

(3) For additional safety and/or flexibility, a power level control circuit may easily be incorporated into this system for hot-startup applications.

The conclusions reached at this stage are that automatic control is feasible and should prove quite flexible, in that any desired period, temperature rate, or power demand signals can easily be accommodated.

Due to analog limitations, the full system including control dead-bands and such time-varying constants as temperature and heat transfer coefficients, could not be treated in this first study. Without these inputs, exact data, such as rod position for various startup conditions, cannot be predicted.

A major uncertainty is the difference between the simulated and the final PM-1 system design. However, barring large changes in system requirements, the overall system response is not expected to change appreciably.

Some typical system studies are shown in Figs. III-45 to III-47. Figure III-45 shows a case where startup is limited to a 13-sec minimum period, while in Figs. II-46 and III-47 it is limited to 40-sec periods; the programmed water heating rate is different in each case.

In addition to this work, efforts began for development of a suitable transient simulation of the total plant. This program will include power range automatic control, steam generator feedwater and steam flow control, reactor and steam generator heat transfer characteristics, and reactor kinetics. Load transients and plant stability will be studied in detail under this program during the next quarter in support of preliminary design.

4. Configuration

Packaging and housing.-

A. Lyman J. Reilly

Work on packaging and housing during the quarter was directed toward the evaluation of package concepts and the establishment of requirements and limitations for the equipment to be shipped or operated inside the packages. Areas of package design and shipment which affect equipment were studied and a design guide for equipment arrangement and mounting was prepared. Trips were made to several US Army Corps of Engineers establishments to obtain information on arctic shelters. Emphasis placed on avoiding metallic conduction paths through shelter walls and on the necessity for providing good vapor barriers in these walls make likely the selection of construction designs tailored for arctic use.

Evaluation of package concepts--The basic package shelter concept was established through evaluation of the following five possible methods for providing equipment shelters and packaging:

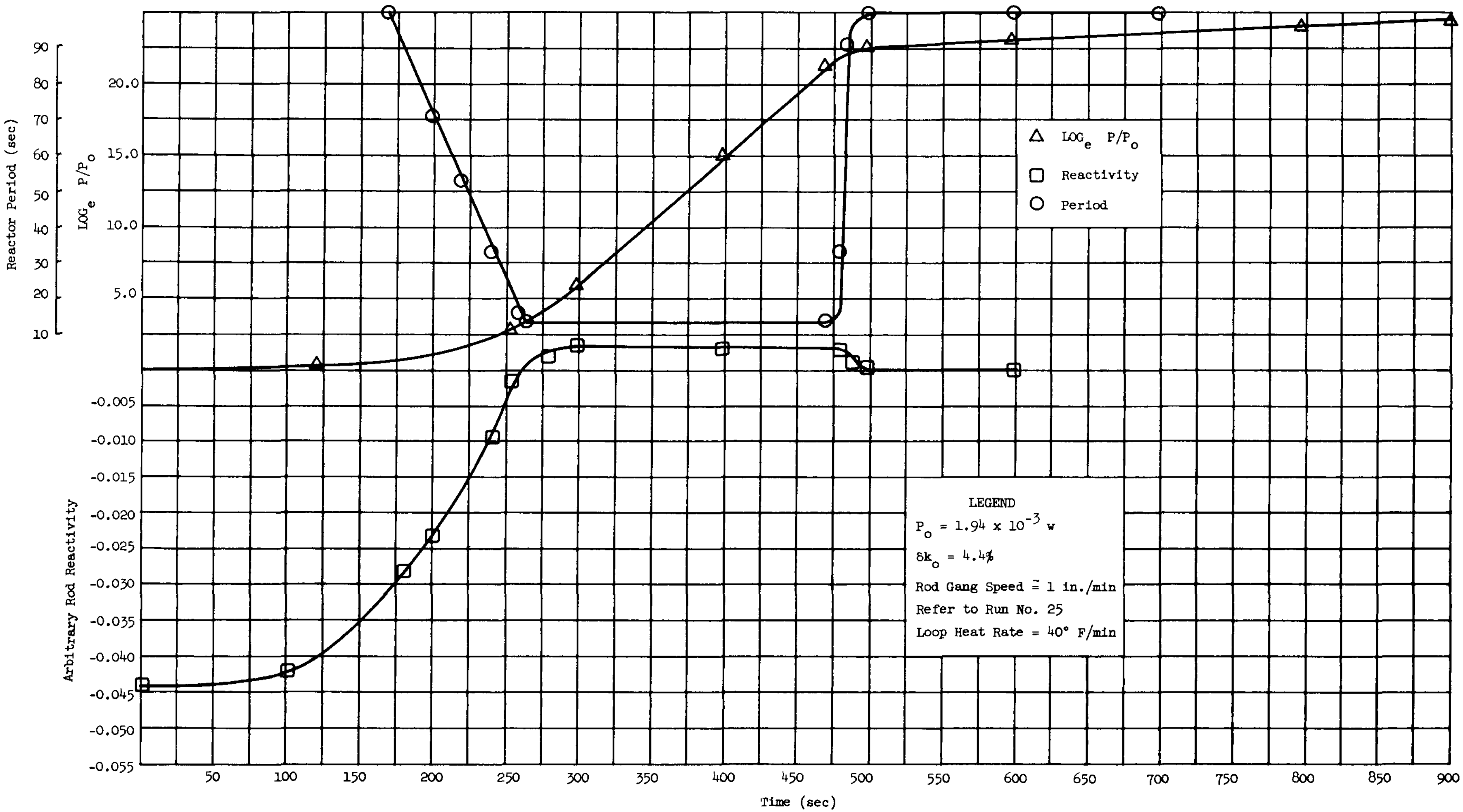


Fig. III-45. Reactor Startup Studies

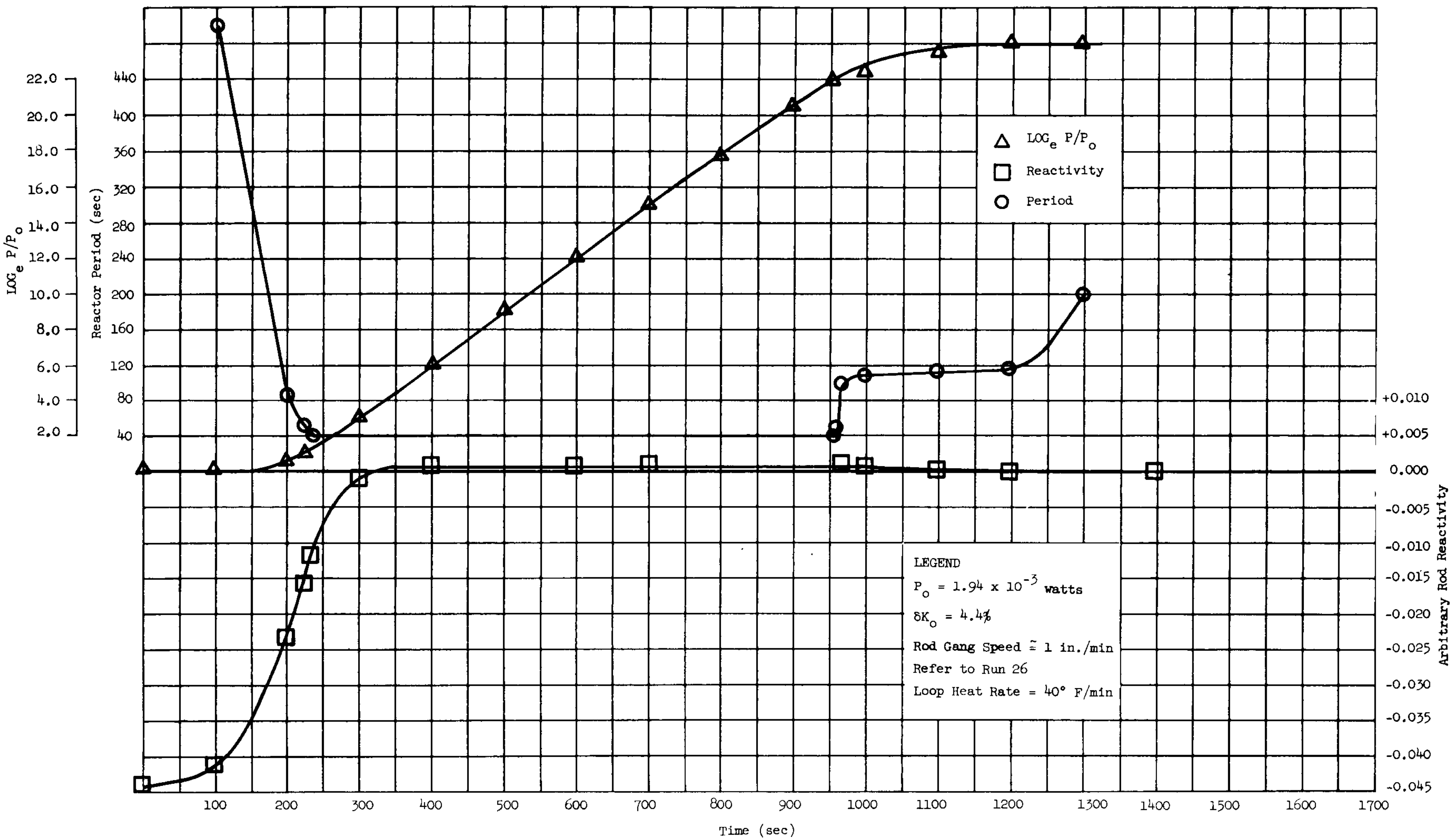


Fig. III-46. Reactor Startup Studies

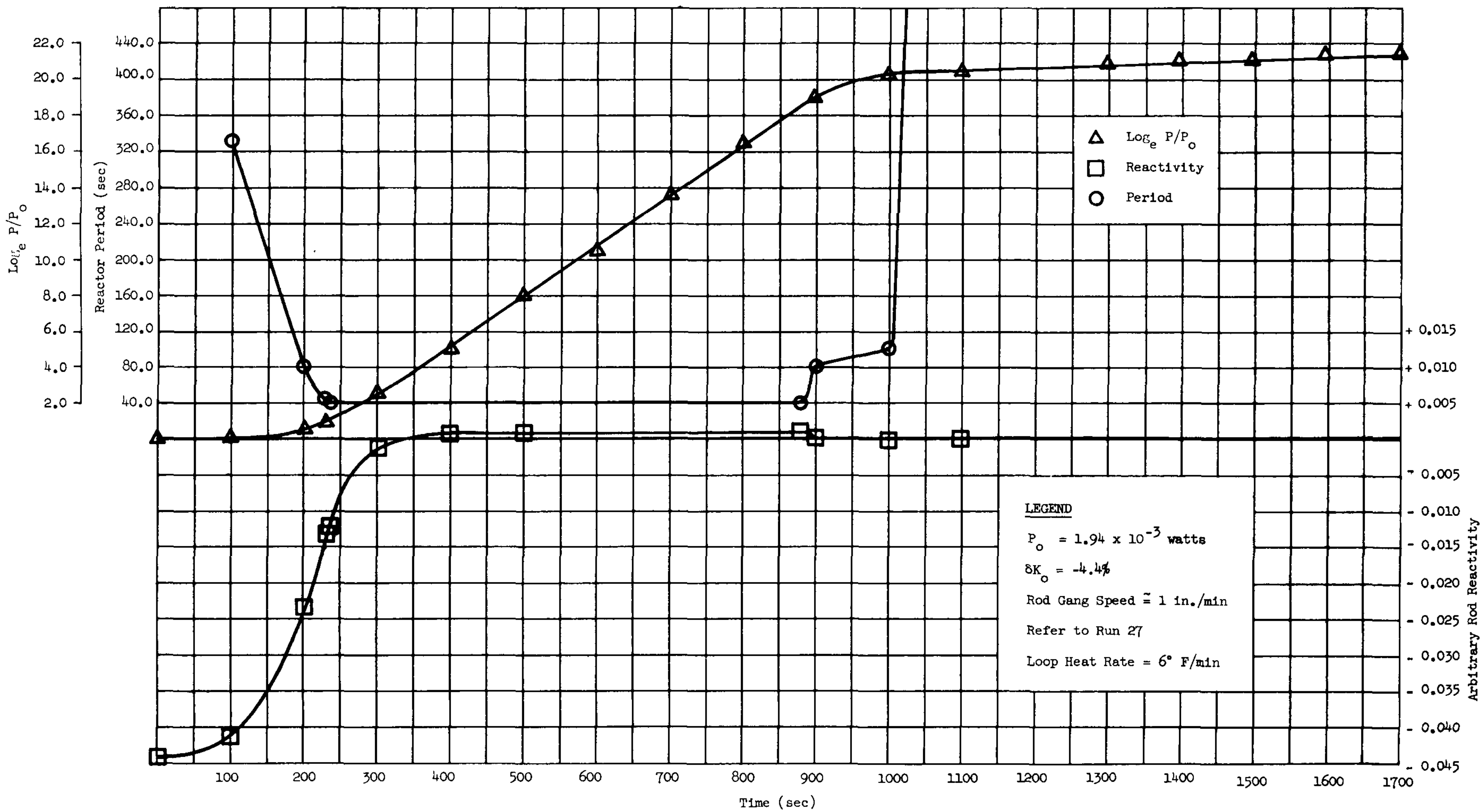


Fig. III-47. Reactor Startup Studies

(1) Use of a standard prefabricated building shipped in separate packages and erected on the site.

(2) Use of a specially developed arctic-type prefabricated building, constructed of structural sandwich panels, shipped in separate packages, and erected on the site.

(3) Use of a building constructed from the sides of the shipping packages.

(4) Use of a building formed by arranging and interconnecting the shipping packages, removing or folding out some of the panels.

(5) Use of each shipping package as the shelter building for the equipment contained therein.

Each of these methods has the advantages and disadvantages summarized in Table III-20.

It is evident from Table III-20 that the use of shipping packages which may be interconnected to form a shelter offers more advantages than the other methods considered. This concept was established as the one upon which the equipment design requirements and limitations are predicated. It is actually a modification of the shipping package-housing concept (Item 5, Table III-20), that provides the necessary space for access to equipment. This concept offers the best possibility of minimizing the number of packages and easing plant installation and relocation.

Size limitations for the equipment were established by using a 3-in. thick panel (as discussed on the following pages) and deducting it from the maximum 8 ft 8 in. package width limitation for C-130 aircraft, leaving an 8 ft 2 in. width dimension for equipment. Although the height limitation in the C-130 is also 8 ft 8 in., after the panel roof thickness, flooring and shipping skid height are deducted, the height remaining for equipment is 7 ft 8 in. maximum (not allowing for a sloping shelter roof).

Maximum length has been arbitrarily set at 30 ft to permit adequate room for tie-downs and to allow a greater center of gravity range. It should be noted that up to 34 ft in length is possible although the cargo compartment is 40 ft long. The center of gravity range in a 34-ft package would be limited to 4-ft in a C-130A aircraft. It would be further restricted if the requirement is established that the packages be capable of shipment in either C-130 "A" or "B" aircraft, since the ranges in both aircraft do not coincide, see Fig. III-48. Assuming a maximum forward position of the 30-ft package at Station 283 and maximum aft position at Station 698, the maximum forward center of gravity location would be 2 ft forward of the package centerline and the maximum aft center-of-gravity location would be 6 ft aft of the package centerline. Thus, an 8-ft range in center-of-gravity is permissible with a 30-ft package using a C-130A.

TABLE III-20

Equipment Shelter Concepts

Method	Advantage	Disadvantage
1. Standard Prefabricated Building	Low initial cost Little R and D expense Readily available	Not a true arctic building, difficult to insulate and vapor seal Requires more packages than methods 3, 4 and 5 Heavier conventional packing crate required Assembled with many small parts, nuts, bolts, plates, etc. Long erection and relocation time, high erection costs
2. Special Prefabricated Building, Sandwich Panels	Panels less apt to be damaged than methods 3, 4 and 5 True arctic building Less maintenance than in method 1 Building can be set up before equipment arrives	More costly than method 1 Requires more packages than 3, 4 and 5. Additional packaging material required for shipping Longer erection time than 4 and 5
3. Shelter Building Made of Shipping Package Sides	May not require additional packages Less maintenance than method 1 True arctic building	Building cannot be set up until equipment arrives; then must be erected before equipment can be installed Panels more subject to damage then 1 and 2 Sizing panels would be a problem

TABLE III-20 (continued)

Method	Advantage	Disadvantage
4. Package Interconnection to Form a Shelter Building	No pre-erection required before arrival of equipment Does not require additional shipping packages Shorter setup time Shorter relocation time True arctic shelter Less maintenance than method 1 More systems can be built in Lighter weight packages than conventional packing crates	Less working space and access to equipment than 1 and 2 Requires development program More costly than method 1 Panels subject to damage (but easily replaced)
5. Shipping Package is Shelter Building	Minimum site erection time Minimum relocation time True arctic building	Work area in the shelter would be too confining and equipment inaccessable; no further consideration will be given this design

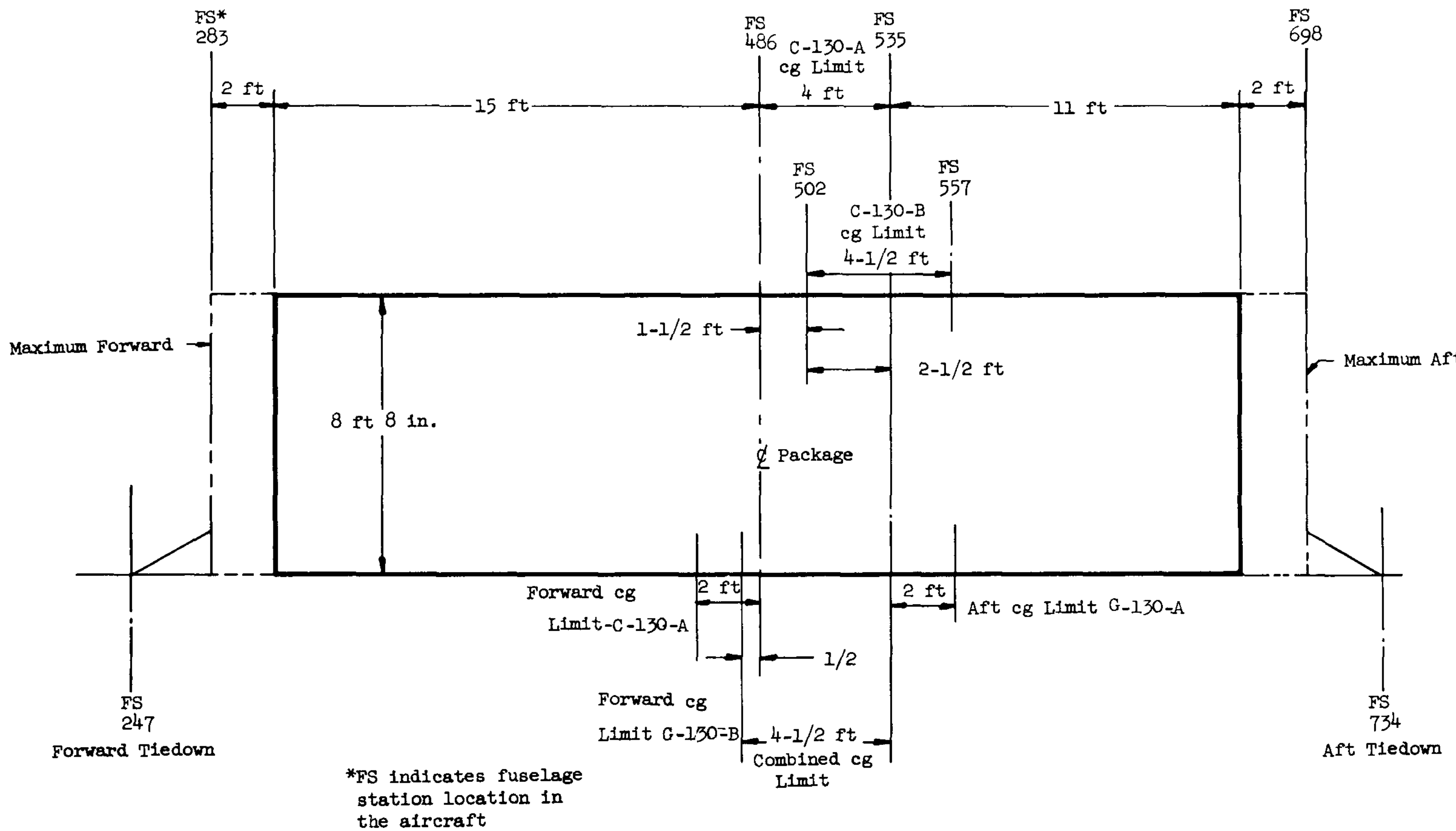


Fig. III-48. Center-of-Gravity Data--PM-1 Package

Shorter packages would permit shifting the center-of-gravity an additional amount that would be equal to the difference between the 30-ft and the shorter package.

Weight limitations of 30,000 lb were established for the complete packaged equipment and at 25,000 lb for the equipment in the package. Weight of the complete package was determined by mission analysis for the C-130A aircraft at a range of 1,000 mi, considering the fuel required for full gross weight at takeoff, no cargo on return, and no mission refueling. (Ref: Flight Handbook TO 1C-130 A-9, Lockheed Aircraft Company.)

The 30,000 lb weight and 1000 mi range combination cannot be exceeded in the C-130A aircraft. In the event any one package should exceed this weight, the C-130B aircraft might be considered if absolutely necessary. The B aircraft has a 10,000 lb higher gross weight. While most of this weight is accounted for in basic aircraft weight and fuel requirements, 3000 lb additional cargo can be carried. (Ref: Flight Handbook TO 1C-130B-1C, Lockheed Aircraft Company.) An additional runway length of 200 ft would be required for empty takeoff of the B aircraft at the site.

Equipment weight limitations were determined by deducting the package structure and panel weights from the 30,000-lb capacity of the aircraft. Package weight can be greatly affected by the type of material selected for use in the panels: a 3-in. foam core panel, 3 ft 9 in. by 8 ft, with fiberglass faces would weigh 50 lb; a 3-in. balsa core panel with stainless steel faces would weigh 88 lb; and plywood panels similar to those being constructed for the ice cap program would weigh approximately 115 lb each. Although limitations on the equipment were established on the basis of balsa core panels, the type of panel to be used has not been determined. Since the lighter panels are more costly, the least expensive panel compatible with package equipment weight will be selected.

Heat loss calculations were made for 2- and 3-in. balsa core panels at -55°F outside temperature. The results show that heat loss through the walls (excluding losses at joints and through doors) represents 25 to 30% of the estimated heat that must be dissipated by the equipment. Panel thickness can, therefore, be determined primarily by strength rather than by insulating properties.

Snow load requirement--Snow loading considerations do not appear to be critical in establishing panel thickness. Calculations show that a 2-in. panel can sustain the required 30 lb/sq ft snow load with only minor deflection (0.535 in. deflection over 8 ft 8 in.length).

Wind and shipping loads--The most critical condition is anticipated from wind, shipping, and handling loads. Little work was done in these areas during this quarter but these conditions will be evaluated next quarter.

Package bending during shipment may become a serious problem. Insufficient room is available in the aircraft to permit designing a skid or base of sufficient depth to provide minimum deflection and bending. A shallow beam will deflect when the package is being hoisted or skidded over uneven surfaces, and this bending can induce severe loads in the equipment. To provide a rigid structure, the package sides may have to be employed as a deep beam; if this is required, the panels would have to be designed as a continuous shear member.

Aircraft floor load limitations of 1000 lb/linear ft of cargo compartment prohibit concentration of loads inside the package when using a flexible platform. Therefore, proper distribution of the load over the aircraft floor again requires that deep beam construction be utilized. Similar structural requirements apply to the shipping of spent fuel casks.

Shock and vibration requirements--The PM-1 equipment will be designed for shipment by aircraft, truck, ship, or rail, and for handling methods normally used with equipment of comparable value and size. The packaging, equipment mountings, and the equipment itself will meet "limit" and "ultimate" conditions which are defined as follows:

> Limit conditions - the most severe shock and vibration conditions that can be imposed during shipment without damage to the equipment. These conditions will be set sufficiently high to ensure damage-free shipment on all common carriers.
>
> Ultimate conditions - the most severe shock and vibration conditions that can be imposed during shipment without causing major structural failures in the equipment. These conditions will be set sufficiently high to ensure that equipment will not tear loose and endanger personnel under accidental or emergency conditions such as crash landings of aircraft. The equipment need not be operable after being subjected to "ultimate" conditions.

The loads imposed by shock and vibration must be carefully defined as to point of input, attenuation through structures, and limits imposed on the power plant equipment. Figure III-49 is a

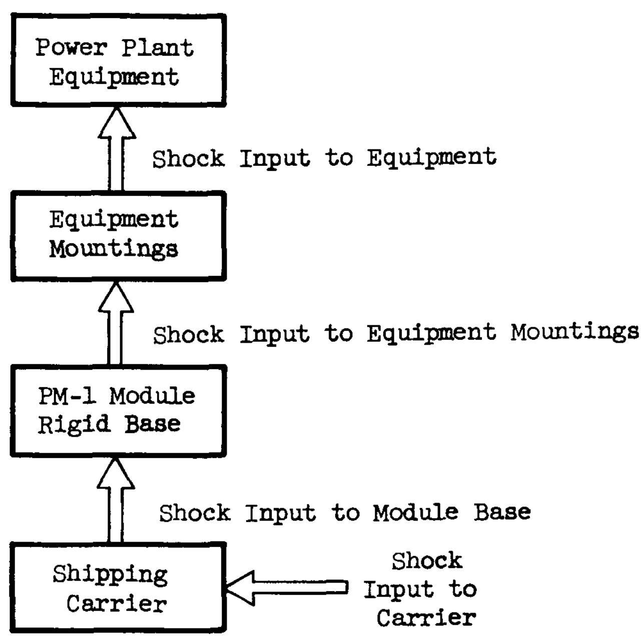


Fig. III-49. PM-1 Package Shipping Shock Loadings schematic representation of a single PM-1 module in shipment. The shock input to the carrier may be defined in terms of g loading in a given direction for a given duration of time. The carrier structure itself attenuates the input and applies a somewhat less severe input to the rigid base of the module. Similarly, the rigid base and the equipment mountings further attenuate the shock. The equipment itself must then withstand the conditions as applied after the series of attenuation steps.

The shock and vibration inputs to the modules during shipping and handling will be of random amplitudes and durations. The very short duration inputs are well attenuated, but the longer duration or cyclic inputs reach the equipment with little reduction in amplitude (g loading value). For trucks and aircraft, typical g loads imposed on equipment through the carrier, package, and equipment mountings are readily available. Data applicable to truck transport has been obtained from tests performed at Aberdeen Proving Grounds. Data applicable to air transport has been obtained from the Air Force Air Transportability Specifications MIL-A-8421-A. While some differences in package construction will exist between the PM-1 modules and the packages tested at Aberdeen and by the Air Force, it is considered reasonable to use the measured equipment loadings as a basis for preliminary design of PM-1 equipment. Special problems may then be resolved during final design. Thus the following conditions have been established as the "limit" and "ultimate" shock conditions applicable to PM-1 equipment for shipment by air and truck.

<u>Air transport - C-130</u>

1. Limit	3.0 g longitudinal and vertical 1.5 g lateral 0.10 sec or longer
2. Ultimate	8.0 g forward 4.5 g down 0.10 sec or longer

<u>Truck transport</u>

1. Limit	2.0 g vertical 1.0 g longitudinal and lateral 0.005 to 0.250 sec
2. Ultimate	3.0 g vertical 1.5 g longitudinal and lateral 0.055 to 0.250 sec

Since vibration inputs are of very short duration and are cyclical, attenuation through structures is primarily a matter of avoiding resonant frequencies. The available data in this case deals with inputs into the package base, and the following limit and ultimate conditions have been established for PM-1 modules:

Air transport	3 g	2-300 cps
Truck transport	3 g	1-200 cps

The shock and vibration conditions encountered in rail and ship transport and in package handling are not as well documented as for truck and air transport. Additional information is being sought and Martin has developed a limited amount of g loading data applicable to rail shipment. It is considered reasonable, however, to utilize special bracing and shock isolation as required to ensure that rail and ship loadings transmitted into the equipment will not exceed those encountered in air and truck transport. For this reason, the shock conditions stated for air and truck transport will be applied for all handling and shipping evaluations during preliminary design of PM-1 equipment. In addition, it is considered likely that railroad humping at 10 mph must be accommodated within limit conditions, and at 14 mph within ultimate conditions.

Limit conditions in handling should include a drop of 6 in. onto concrete, and a 9-in. drop should be feasible within ultimate conditions. Additional data and analysis will be required to translate these events into shock requirements. It must be remembered that the PM-1 modules will be quite large and heavy in comparison with packages described in existing shipment specifications. Handling of 30,000-lb modules will require caution and is not likely to lead to rapid motion except in an accidental drop. The swing and drop tests required for smaller packages are not considered applicable. The final shipping specifications will, however, cover all conditions which might reasonably be expected in shipment by any type of carrier.

Containment.-

J. Goeller P. Moll

The major areas of effort involved the determination and enumeration of potential accidents, a preliminary evaluation of these accidents, a survey of the present applicable methods and concepts of containment, and the conceptual development of 12 types of containment.

A survey of literature was made to determine the various concepts of containment in use on reactors similar to the PM-1 and on reactors in general. These concepts include absorption of incident energy via either tanks of water or trays of water within a vessel, conventional pressure vessel containment, blow-out into supplementary tanks, and limited (leakage) containment. Other methods, such as plastic containment shells, expanding containment, and underground containment were also investigated in a cursory manner.

A report detailing the investigation is in preparation.

Containment requirements were established during the quarter in preparation of a containment design guide specification.

For purposes of parametric design studies, a maximum credible incident was defined. This incident permitted a pressure-volume relationship to be developed for the containment calculations. The incident was defined as being the release of 88 cu ft of primary loop water with a total contained energy of 2,094,000 Btu, including 300,000 Btu from a nuclear excursion and 60,000 Btu from afterheat. Figure III-50 presents the resulting pressure-volume curve for expansion into an air-filled volume. These data will be revised after selection of the thermodynamic design parameters of the plant. The assumed incident was based on a 2000-psi primary system pressure to assure conservative results in the preliminary studies.

Numerous methods of containing primary loop incidents were considered and investigated during the quarter. Twelve configurations were developed to the point of making scaled sketches and determining design pressures, approximate weights, number of airplane loads involved, and shielding requirements. Four of the multiple tank configurations (see Figs. III-51 to III-54) will be evaluated in further study. Tables III-21 and III-22 present concepts considered and denote the concepts that are to be studied further.

The multiple tank configurations were chosen for further development during the next quarter because they lend themselves to conversion from contained to uncontained systems by either using or eliminating additional volume tanks. The multiple tank method is also adaptable to either above or below ground installations as required by the site. It is quite possible that field welds may be eliminated in this type configuration.

To keep the number of tanks required for containment to a minimum, the design pressures become high. The tanks then become heavy and do not allow for shipping equipment within them. To overcome this difficulty, the equipment, in some cases, may be mounted on skids to be shipped separately.

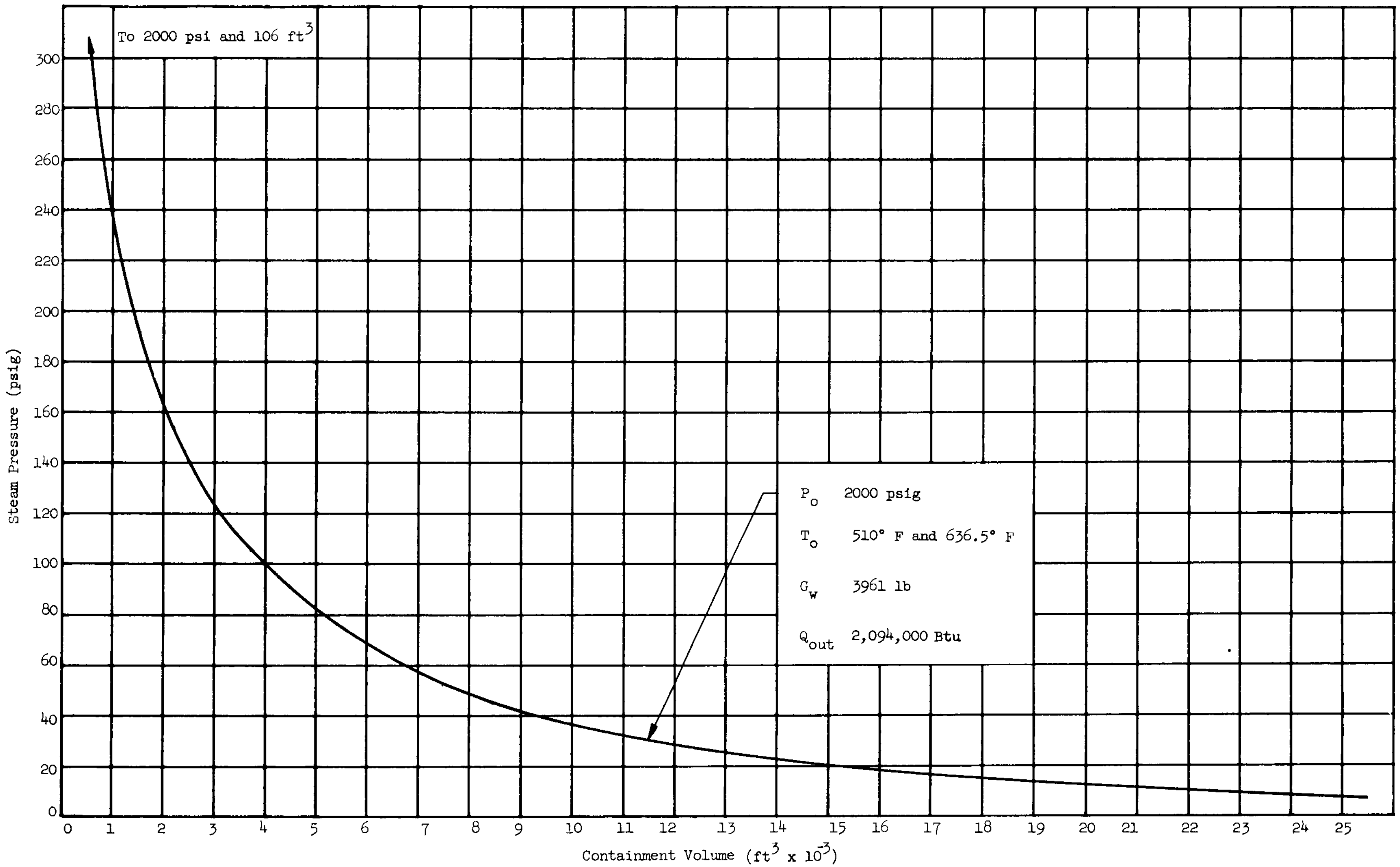
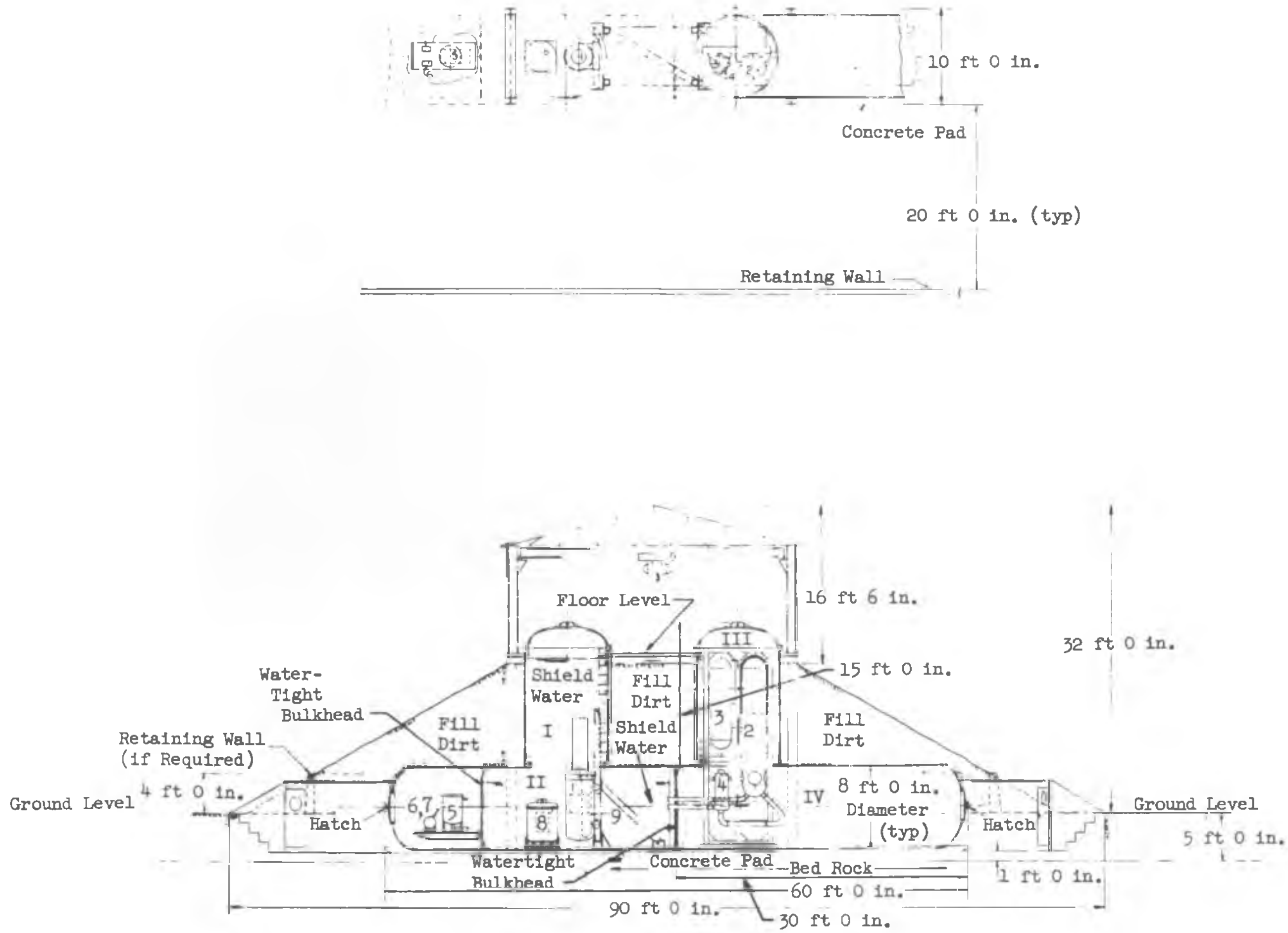


Fig. III-50. Pressure-Volume Curve for Typical Primary Loop Rupture



LEGEND

1. Reactor
2. Steam Generator
3. Pressurizer
4. Coolant Pump
5. LP Demineralizer
6, 7. Economizer and Cooler
8. Cask
9. HP Demineralizer

Containment: (4 Tanks)

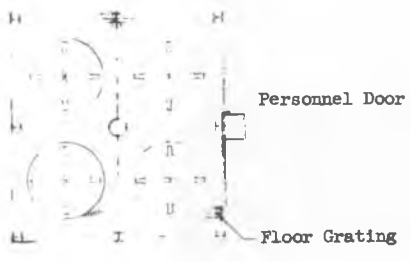
Gross Volume --2,620 cu ft
Net Volume --2,250 cu ft
Tank Pressure ~ 150 psig Design
Tank Pressure ~ 300 psig Burst
Skin Thickness ~ $\frac{1}{2}$ in.

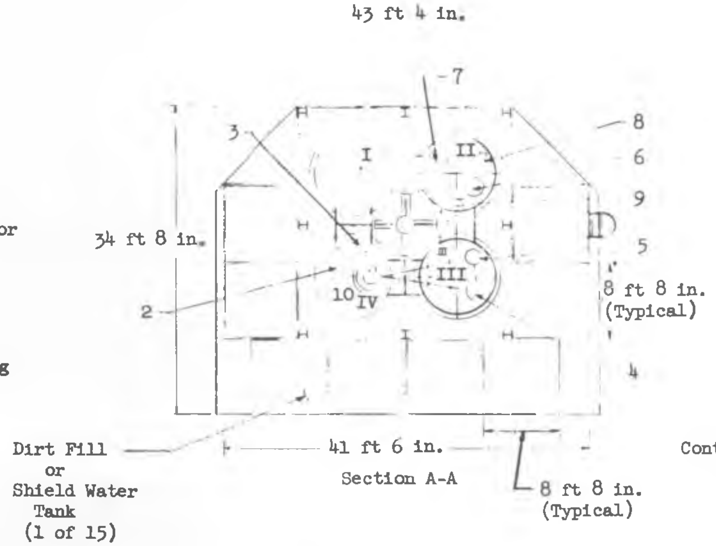
Approx Tank Weights (Steel)

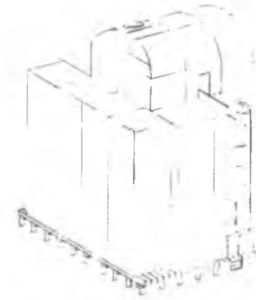
I ~ 14,000 lb
II ~ 28,000 lb
III ~ 14,000 lb
IV ~ 28,000 lb

No. Cargo Flights Required--7

Fig. III-51. Nonconventional Multiple Tank Containment--Concept 1





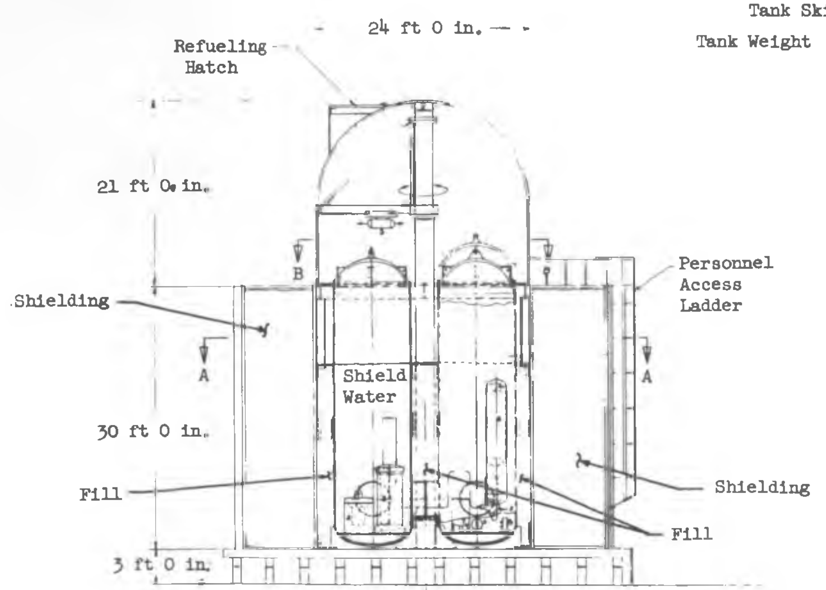


Containment:

Gross Volume	~ 5,160 cu ft (with Shield Water Level Shown)
Net Volume	~ 4,790 cu ft (with Shield Water Level Shown)
Tank Pressure	~ 82 psig (Design)
Tank Pressure	~ 164 psig (Burst)
Tank Skin Thick	~ 5/16 in.

Tank Weight

I	~ 17,000 lb
II	~ 23,000 lb
III	~ 28,000 lb (Less Pump)
IV	~ 25,000 lb (Less Reactor Head and Cask)



LEGEND:

1. Reactor
2. Cask
3. HP Demineralizer
4. Steam Generator
5. Coolant Pump
6. Pressurizer
7. LP Demineralizer
8. Economizer
9. Cooler
10. Reactor Head Storage Area

Fig. III-52. Conventional Multiple Tank Containment with Shield Water Tanks--Concept 1

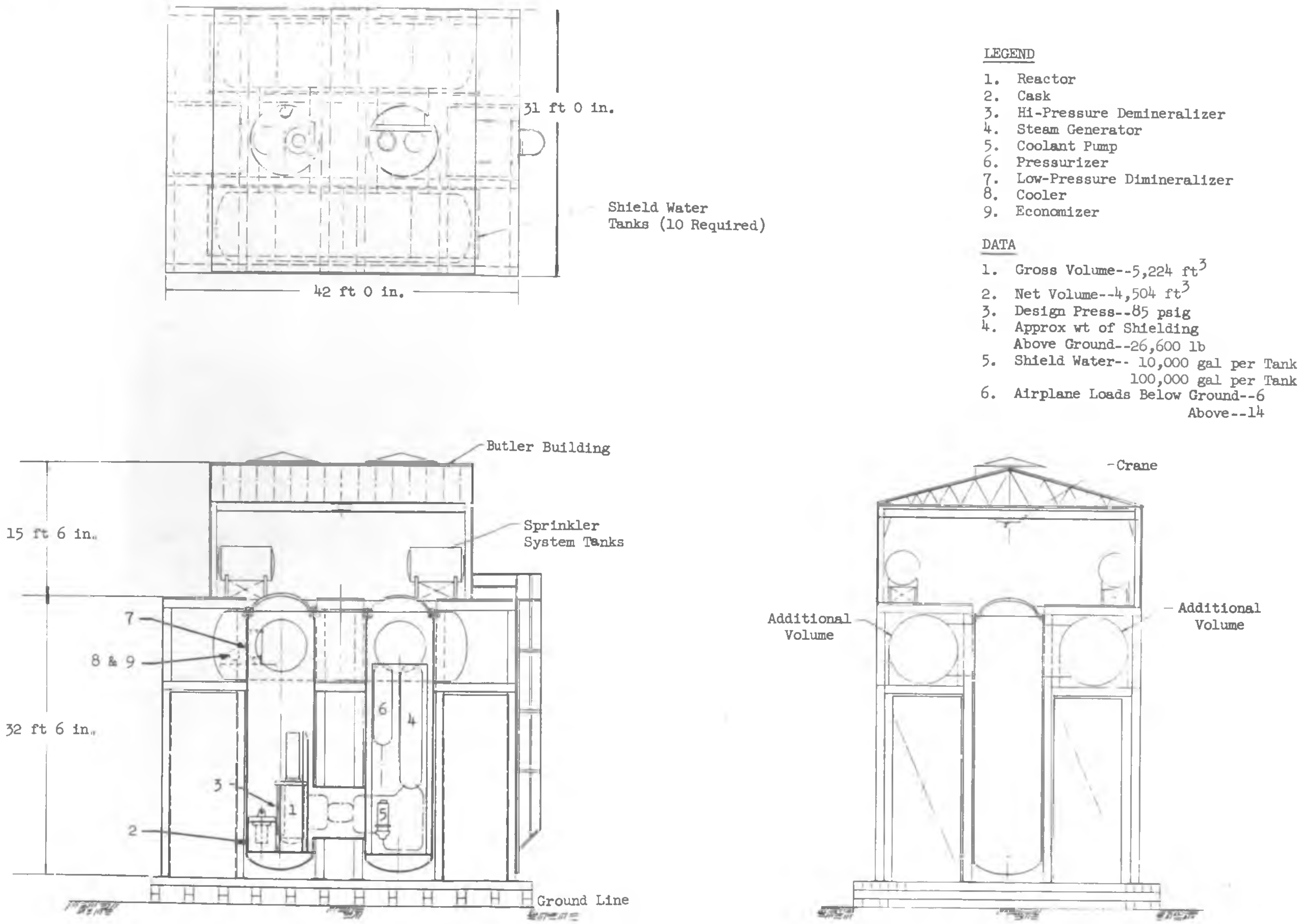
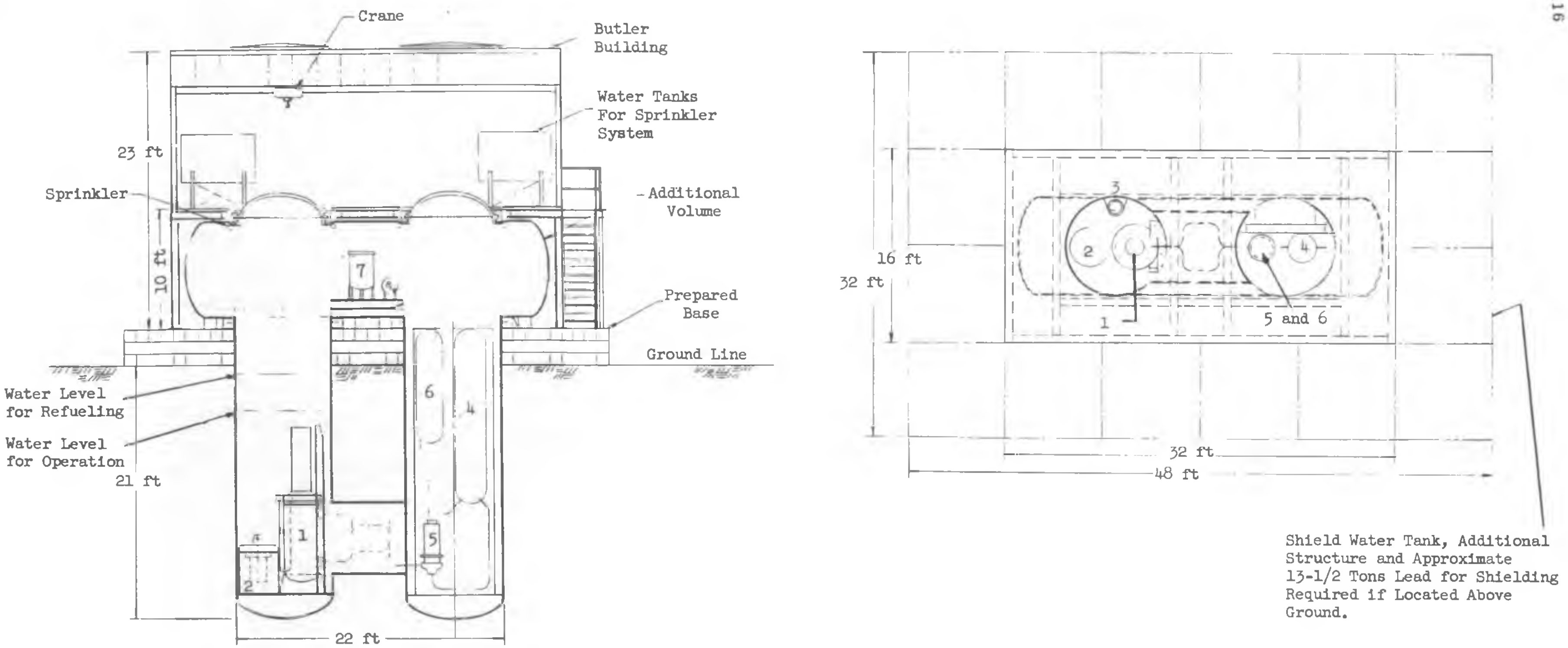


Fig. III-53. Nonconventional Multiple Tank Containment--Concept 2



LEGEND:

1. Reactor
2. Cask
3. Hi-Pressure Demineralizer
4. Steam Generator
5. Coolant Pump
6. Pressurizer
7. Low Pressure Demineralizer
8. Cooler
9. Economizer

DATA

1. Gross Volume - 3,020 ft^2
2. Net Volume - 2,300 ft^2
3. Design Pressure 150 psig
4. Approximate Weight of Shielding Above Ground Inst 26,600 lb
5. Shield Water - 11,970 gal per tank 191,520 gal Total.

Fig. III-54. Non-Conventional Multiple Tank Containment--Concept 3

TABLE III-21

PM-1 Containment Concepts

Conventional

No.	Description	Method of Containing Incident	Design Pressure (psig)	Vessels (No.)	Remarks	Further Investigation
1.	Multiple tanks 4, 30 ft long (Fig. III-52)	Expansion through small diameter interconnections into multiple pressure vessels	82	4 vessels	No field welding Earth back-fill or shield water tanks	Flange joints Materials Equipment arrangement Use of concentric shells
2.	Multiple tanks 4, 22 ft long (Fig. III-57)	Expansion through small diameter interconnections into multiple pressure vessels	115	4 vessels 1 shield water supply tank	No field welding Earth back-fill or shield water tanks	None. Not enough maintenance room
3.	Large volume welded vessel, 34-ft dia	Expansion into large volume	35	1 vessel 2 equipment packages	Requires field welding Requires concrete Earth back-fill Time-consuming installation and assembly	None

TABLE III-22

PM-1 Containment Concepts

Nonconventional

No.	Description	Method of Containing Incident	Design Pressure (psig)	Vessels (No.)	Remarks	Further Investigation
1.	Multiple tanks 2 Horiz - 30 ft long 2 Vertical - 15 ft long (Fig. III-51)	Expansion through full diameter connections and bulkheads with pressure diaphragms into multiple pressure vessels	150	4 vessels	Skid mounted equipment Field assembly not too difficult Pressure diaphragms required Large flanges a problem	Tank interconnections Materials Equipment arrangement Use of inner shells
2.	Multiple tanks -- 4 30 ft long (Fig. III-53)	Expansion through full diameter connections into multiple high pressure vessels	85	4 vessels	Equipment skid mounted Field assembly problems	Tank interconnections Materials Equipment arrangement Use of concentric shells
3.	Multiple tanks 2 25 ft long 1 30 ft long (Fig. III-54)	Expansion through full diameter connections into multiple pressure vessels	150	3 vessels	Equipment skid mounted Large flanges a problem	Tank interconnections Materials Equipment arrangement Use of concentric shells

TABLE III-22 (continued)

PM-1 Containment Concepts

Nonconventional

No.	Description	Method of Containing Incident	Design Pressure (psig)	Vessels (No.)	Remarks	Further Investigation
4.	Multiple tanks-- 2 Horiz - 30 ft long 26 ft long 1 Vertical - 15 ft long (Fig. III-56)	Expansion through full diameter connections and bulkheads with pressure diaphragms into multiple pressure vessels	200	3 vessels	Skid mounted equipment Field assembly not too difficult Pressure diaphragms required Requires horizontal steam generator Large flanges a problem Excessive foundation requirements	None
5.	Large volume--bolted flange vessel 26 ft dia (Fig. III-55)	Expansion into large volume	35	1 vessel 2 equipment packages	Sealing problems Time consuming - installation and assembly Fabricating problems	None
6.	Large volume vessel bolted flanges-- expanding dome 30 ft dia	Expansion into large variable volume	10	1 vessel 2 equipment packages	Sealing problems Time consuming - installation and assembly Fabricating problems	None - No weight savings More complex

TABLE III-22 (continued)

PM-1 Containment Concepts

Nonconventional

No.	Description	Method of Containing Incident	Design Pressure (psig)	Vessels (No.)	Remarks	Further Investigation
7.	Multiple tanks - directed incident 2 22 ft long 1 30 ft long	Incident directed through use of pressure and pressure relief valves	235	1 vessel 2 equipment packages	Reliability of equipment Possible loop release at corroded points rather than thru relief system	Task 1.4
8.	Multiple tanks - energy absorption 2 20 ft long	Absorption of energy by use of pebbles and high pressure vessels	160	2 vessels	Requires development and tests Installation time for pebble installation Refueling - must remove pebbles Activation of pebbles	Task 1.4
9.	Multiple tanks - energy absorption 2 20 ft long	Absorption of energy through use of aluminum wire mesh and high pressure vessels	160	2 vessels	Requires development and tests To refuel - must remove wire mesh Activation problem	Task 1.4

If the additional shielding required for above ground installation is shipped with the plant, this approximately doubles the number of airplane loads needed for the primary loop and containment. One alternative would be the use of shield water tanks for above ground installations. This, however, becomes bulky, even though tanks made up of panels for assembly at the site would reduce this problem. To ensure water tight integrity, flexible liners would be required inside the tanks. Although the concept of shield water tanks appears to be unduly complicated as compared with the use of cribbing and local site materials, vendors contacted have agreed that the concept is probably feasible, and additional data are being developed.

Suitable site materials should be available at all but ice cap locations, where timber cribbing and packed snow or a separation of approximately 50 ft through unpacked snow can be used.

The multiple tank concept requires additional study in several areas. First, it presents a problem in joining the tanks together. Conventional bolted flanges consume allowable shipping space and are very heavy. Preliminary analysis, using the ASME Code, indicates that a single flange thickness of approximately 7 in. is required for a full diameter seal. It should be noted that flange stiffness and proper gasket sealing, not strength, are the primary requisites in sealing a tank of this size (8 ft 8 in. OD) and, therefore, that the ASME Code design may not prove to be an adequate design guide. Flange design and the use of seal welds will be further evaluated during the next quarter. The Conoseal Division of the Marman Company has been contacted regarding the use of clamp-type joints in sizes ranging from 40 to 96 in. in diameter and at pressures from 65 to 150 psig; results are not encouraging for sizes above 3 ft.

Three containment concepts which are being dropped are of the large-volume single-vessel type. These concepts were rejected because they require either extensive field welding or involvement with difficult bolted joints. In addition, it is considered that the time to assemble these vessels and install equipment would be prohibitive. At arctic sites, plant installation if a large containment vessel is to be utilized, containment vessel construction would be required a year before the plant installation.

Three of the rejected containment versions are of the multiple tank-type. The two energy absorption concepts and the directed incident concept will be considered further under Task 1.4 because they require development and tests to prove their feasibility. The desirability of proceeding with experiments will be reviewed with the Commission in August prior to initiation of tests.

Three versions (see Figs. III-55 to III-57) were rejected because they do not provide enough room for maintenance access, and are otherwise similar to versions which do provide access.

B. SUBTASK 3.2 - PRELIMINARY DESIGN

Subtask 3.2 work was limited, in the first quarter, to the establishment of an overall reliability program and to some preliminary design of instrumentation. Plant preliminary design will be accomplished during the second quarter.

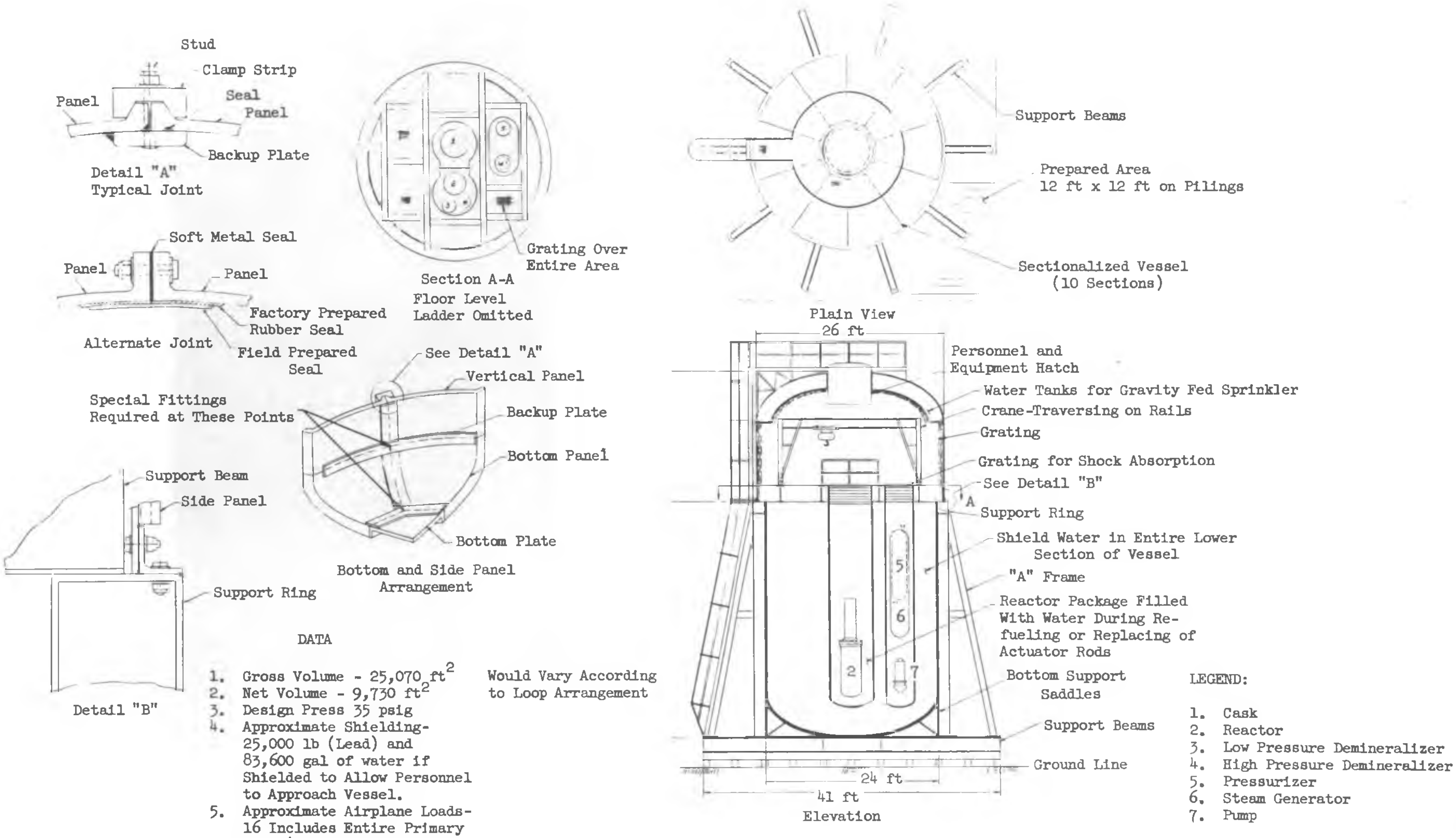
The reliability program will be followed through all phases of the PM-1 Project, including design, manufacture, test, and field operation. A file of reference material was accumulated which includes pertinent handbooks, specifications, etc.

The contract sets reliability criteria by specifying a maximum outage of 21 days/yr, including a scheduled annual downtime of 6 days. Fifteen days, therefore, are left for total unscheduled downtime. An initial division of this time was arbitrarily made between the three major systems of the plant (primary, secondary, and control) by allocating 1/3 of the unscheduled downtime to each. The scheduled annual downtime is then added, yielding a total allowable downtime, for each major system, of 11 days. This figure serves as the target for each system; re-evaluation may be made as more system characteristics are determined.

On the basis of the reliability assignments described, reliability requirements have been included in the contract statements of work for the Westinghouse and Stromberg Carlson subcontracts.

A preliminary analysis was made of required control rod actuator reliability prior to issuing specifications for bid. The requirement that reliability of the actuator group be not less than 0.99905 was tentatively established. This requires, in turn, that individual actuator reliability be not less than 0.99986, or that the mean time between actuator failures be of the order of 7000 hr, if 3 hr are lost per actuator failure.

The first reliability report, to be issued next quarter, will present a basic approach to reliability and will establish consistent definitions, methods, and techniques. It should be noted that reliability data in the nuclear industry, particularly failure information regarding primary loop pumps, actuators, etc., is notoriously lacking.



DATA

1. Gross Volume - 25,070 ft^2
2. Net Volume - 9,730 ft^2 — Would Vary According to Loop Arrangement
3. Design Press 35 psig
4. Approximate Shielding-25,000 lb (Lead) and 83,600 gal of water if Shielded to Allow Personnel to Approach Vessel.
5. Approximate Airplane Loads-16 Includes Entire Primary Loop-(Equipment Shipped on Separate Skids).

LEGEND:

1. Cask
2. Reactor
3. Low Pressure Demineralizer
4. High Pressure Demineralizer
5. Pressurizer
6. Steam Generator
7. Pump

Fig. III-55. Non-Conventional Single Tank Containment--Concept 5

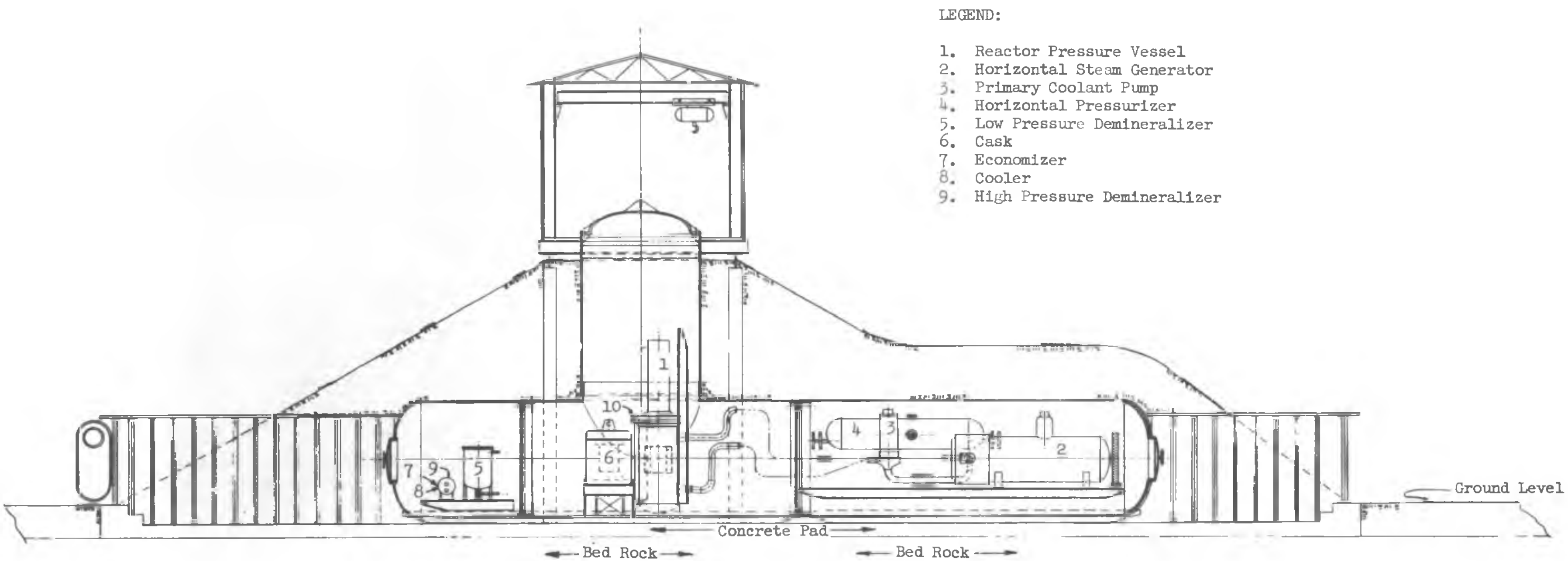
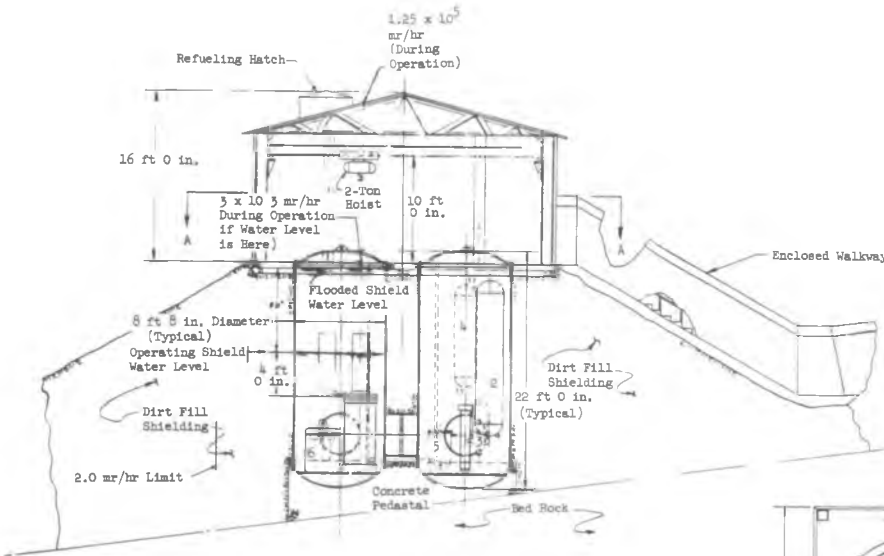
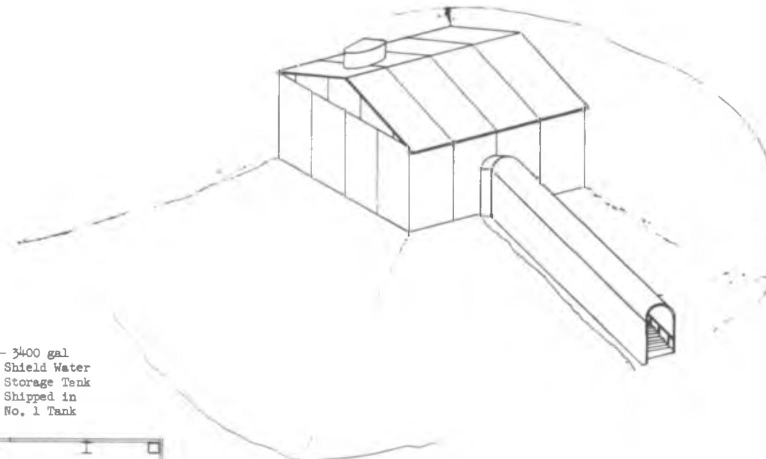
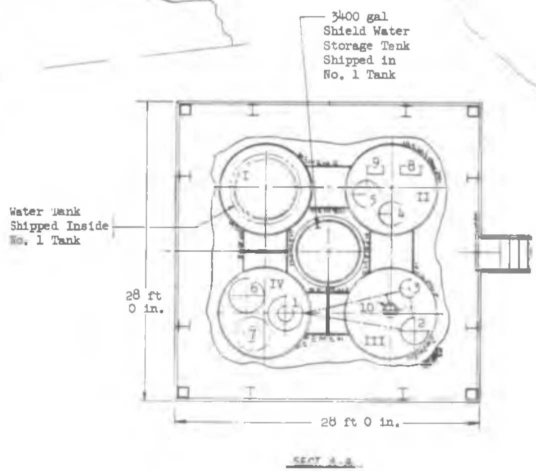


Fig. III-56. Non-Conventional Multiple Tank Containment--Concept 4







LEGEND:

1. Reactor Pressure Vessel
2. Vertical Steam Generator
3. Primary Coolant Pump
4. Pressurizer
5. Low Pressure Demineralizer
6. Cask
7. Reactor Head Storage Area (Refueling)
8. Economizer
9. Cooler
10. High Pressure Demineralizer

Fig. III-57. Conventional Multiple Tank Containment--Concept 2

1. Instrumentation, Preliminary Design

While the parametric study is being made of the overall power plant, a preliminary design covering the instrumentation and controls is under way. This phase of the work is broken down into several important systems, namely, nuclear, primary plant, auxiliary primary plant, reactor safety, health physics, intercommunication, and analysis instrumentation. Each requires system engineering to assure that it is wholly integrated within itself and with the others. System descriptions are being written which cover the design requirements. The results of the parametric study are to be factored into these and from the combined data a set of final equipment specifications will be written.

Brief system descriptions.- Nuclear instrumentation measures the reactor power level from source through intermediate to the power range and employs seven channels. The range of measurement is from 0.25 to 1.5×10^{10} nv.

Two identical source range channels, employing BF_3 proportional counters, furnish signal pulses to a count rate amplifier. The counter's range of one to 10^5 counts/sec corresponds to a measured neutron flux of 0.25 to 2.5×10^4nv. The output of the count rate amplifier also feeds a period amplifier to indicate reactor period over a range of -30 to +3 sec.

Two identical intermediate range channels employ gamma-compensated ion chambers. Measured neutron fluxes of 2.5×10^3 to 10^9nv yield a current which is applied to a log n amplifier. The amplifier output is the measure of reactor powers of from one four millionth to one tenth of full power. The output of the intermediate range amplifiers also drives period amplifiers to indicate period over a range of -30 to +3 sec. The period signals from both the source and the intermediate range instruments are used for startup protection of the reactor.

Three identical power range channels employ uncompensated ion chambers to furnish a current signal to linear amplifiers. Reactor power is indicated over a range of from 1 to 150% of full power. This corresponds to a measured neutron flux of 1×10^8 to 1.5×10^{10} nv. The linear power signals are fed to bistable amplifiers and to a coincidence circuit which will scram the reactor when two out of the three channels indicate an overpower condition.

Primary plant instrumentation.- The primary plant instrumentation measures the basic plant parameters and operating conditions required for safe and efficient operation of the reactor plant. The following parameters are measured:

Reactor inlet temperature	-	50-550°F
Reactor outlet temperature	-	50-550°F
Reactor average temperature	-	450-550°F
Reactor coolant flow	-	0-1 x 10^6 lb/hr
Pressurizer pressure	-	0-2500 psig
Pressurizer level	-	range to be determined
Pressurizer temperature	-	50-700°F

The signals obtained are used for indication and control. The reactor outlet signal temperature may operate a high temperature alarm or a high temperature scram; it is fed into a reactor average temperature computer. The reactor inlet temperature is indicated and provides a signal to the average temperature computer. The average temperature information generated permits the operator to determine the differential temperature across the reactor. The reactor coolant flow is determined by taking the differential pressure across the coolant pump or steam generator and the signal is used to provide a scram upon loss of flow. Pressurizer pressure is indicated and activates high and low pressure alarms, a low pressure scram, the pressurizer heater, and the spray valve control. Pressurizer level is indicated and activates high and low level alarms, and the pressurizer heater cutoff in the event the level drops below the heater wells. Pressurizer temperature is indicated.

Auxiliary primary plant instrumentation.- The auxiliary primary plant instrumentation measures and provides control signals for those components which indirectly affect the operation of the reactor plant. The following are measured:

(1) Shield water

(2) Cooler temperature

(3) Shield water level

(4) Coolant waste tank level.

The shield water coolers cool the water used for reactor shielding and for cooling the primary coolant pump. When the temperature of the water is above 110° F, the temperature control signal actuates the coolers to keep the water temperature below 110° F.

The shield water level is measured and controlled to prevent excessively low or high levels from occurring. The radioactive gas storage and coolant waste tanks provide for storage of radioactive wastes. Level is measured on both tanks to permit the operator to maintain safe levels.

Reactor safety system.- The reactor safety system utilizes signals from the nuclear and primary plant instrumentation systems to scram this reactor, thereby providing protection for the power plant against abnormal or unsafe operating conditions that could cause damage to equipment and personnel. The system employes a scram circuit which receives as inputs the following signals:

(1) High neutron flux

(2) High coolant temperature

(3) Low coolant pressure

(4) Low coolant flow

(5) Fast period during startup.

Any one or more of these signals will interrupt the current to the rod drive latch solenoids thereby releasing the control rods and causing a scram. Manual scram is also provided, as are interlocks to assure that the plant is brought to power safely.

Health physics instrumentation for the protection of personnel and equipment against radiation damage, analysis instrumentation which measures the purity of the primary coolant and determines the effectiveness of the purification system by measuring conductivity, pH, dissolved oxygen and dissolved hydrogen, and communicating equipment for rapid verbal transmission of information from one station to another are also provided.

C. SUBTASK 3.3 - PREPARATION OF SPECIFICATIONS AND COMPONENT AND FACILITY TEST LISTS

H. Clark J. Millette

During the first quarter, the following general work was done in preparation for writing the bulk of the preliminary specifications during the next quarter:

(1) Specification requirements were extracted from the contract and used as a basis for the formation of the specification plan and schedule. This plan and schedule is being used as a guide in the preparation of specifications for materials, processes, systems, and component procurement, and is integrated into the overall project plan.

(2) Preparation of a specification format, along with instructions, for the preparation of outline, final, and procurement specifications began.

(3) Indexes of the required codes, standards, and practices were prepared. These indexes cover approximately 6000 documents. Copies of these indexes are being reviewed to determine whether they are completely applicable, or if exceptions will be requested. A master list of applicable "Codes, Standards and Practices" will be used during succeeding quarters in designing and procuring equipment.

Preparation of preliminary specifications as well as component and facility test lists will begin in the next quarter.

IV. TASK 5--CORE FABRICATION

Project Engineer--Subtask 5.1, 5.2, 5.3: J. F. O'Brien

The overall objectives of Task 5 are to develop and fabricate the fuel elements required for the PM-1 Flexible Zero Power Test and the final PM-1 core.

A. SUBTASK 5.1--FABRICATION OF CORE

In view of the imminence of a steel strike, it was determined that cladding material specifications should be prepared and cladding material ordered during the first quarter. To this end, specifications were prepared and submitted to vendors for bid.

A major effort, associated with the preparation of material specifications, concerns impurity limitations. The desirability of reducing the amount of cobalt present in the reactor core has been demonstrated in the operation of the APPR-1. It is believed that the effects of cobalt activation and corrosion with subsequent crud deposition throughout the system would be minimized by reducing the cobalt content to 0.005 wt % or less.

Previous tubular elements have been fabricated from a modified Type 347 stainless steel, which contains columbium and some tantalum,. an impurity also subject to activation. The specific activitity of the tantalum is greater than that of cobalt, and may pose an equal or greater operating problem. A comparison of the characteristics of the two elements is given in Table IV-I.

TABLE IV-1

Nuclear Characteristics of Cobalt and Tantalum

Parent Material	σ act* (barns)	Isotope Produced	Half-Life	Decay Particles and Energy (Mev)
Co-59	36	Co-60	5.2 yr	$\beta(1.56, 0.31), \gamma(1.17, 1.33)$
Ta-181	19	Ta-182	122 d	$\beta(0.53), \gamma(0.07, 1.22)$

*At neutron energy corresponding to 2200 m/sec.

The energies of emitted gamma rays can be seen to be equivalent. The higher specific activity of tantalum, is due to its short half-life. If it is assumed that equal weight percentages of cobalt and tantalum are present at the beginning of core life, calcuation of the ratio of the tantalum activity to cobalt activity results in the figures given in Table IV-2.

TABLE IV-2

Activation Characteristics of Cobalt and Tantalum

Reactor Operating Time	Ratio, R, of Ta activity to Co activity	Decay Time for R to = 1
37 d	~ 4.5	~ 110 d
2 yr	~ 1.3	~ 40 d

The 37 day point is the time at which the ratio of the activities is greatest; 2 yr is the PM-1 core design lifetime. If approximately equal activities are to be attained, the tantalum content should be from 1.3 to 4.5 times less than the cobalt content. Roughly equal activities are desired, since little would be gained by attempting to continually decrease one activity if problems are defined by the other.

The relative activities which have been calculated are based on activation in the core with no system crud deposition taken into account. The removal of Ta-182 by neutron absorption to produce Ta-183 has not been taken into account. Recent unpublished MTR data indicate that Ta-182 has an activation cross-section of 17,000 ± 2000 barns measured with pile neutrons. Tantalum-183 is most recently reported to decay with the emission of a maximum energy 0.35 Mev gamma and some beta particles. The data, however, are quite sketchy. Using the quoted cross-section of 17,000 barns reduces the activation ration in Table IV-2 by about a factor of two for a thermal neutron flux of 10^{13} nv.

In practice, reduction of the tantalum content is difficult, since a minimum amount of columbium, having a constant proportion of tantalum present as an impurity, must be added to the steel to act as a stabilizing agent. This agent prevents carbide precipitation by fixing the specified minimum amount of carbon, 0.05 to 0.08%, present in the metal. The carbon is needed to maintain the properties of the steel. Reduction of the tantalum becomes, therefore, a problem of what purity of columbium can be attained in actual practice, and high purity columbium thus becomes extremely desirable. The specified chemical composition of the cladding material was established with these facts in mind and is given in Table IV-3.

TABLE IV-3

Specified Chemical Composition of PM-1 Fuel Element Cladding Material

	(%)
Carbon, max	0.05--0.08
Manganese, max	2.00
Phosphorous, max	0.045
Sulphur, max	0.030
Silicon, max	1.00
Nickel	9.00--13.00
Chromium	17.00--20.00
Columbium + Tantalum	10 times C min., 1.0 max
Tantalum, max	0.007
Cobalt, max	0.005
Iron	Balance

After establishing the procurement specification, Allegheny Ludlum, Davidson Steel, Universal Cyclops, Crucible Steel, and Allvac Metals were contacted. Bids were received from four of these companies on material fabricated to the chemical composition given in Table IV-3. Specifications and delivery could be met only by Allvac Metals Company, Monroe, North Carolina, who have guaranteed delivery to Babcock and Wilcox (B&W) by 19 June 1959 for tube blank piercing. Tube blanks will then be prepared by B&W with completion prior to the date of their anticipated strike (31 July 1959).

If all schedules which have been presented to date are held, the material delivery will be made in time to proceed with tube drawing after the strike in time to meet core fabrication schedules. If the schedule dates given above are not met, backup material (0.04% Co and 0.076% Ta) is available in the tube roller's (Superior Tube) stock. No formal request for this material has been made.

Use of backup material would not affect the nuclear or corrosion characteristics of the core, but would aggravate the problem of activation.

During the next quarter specifications for cold drawing of pierced tube blanks will be prepared and put out for bid. Tube drawing will then be initiated with delivery of all drawn tubing to be completed in October of 1959.

B. SUBTASK 5.2--CONVERSION OF UF_6 TO UO_2

This task will become active in the next quarter.

C. SUBTASK 5.3--FUEL ELEMENT DEVELOPMENT

B. Sprissler J. Kane J. Neace D. Grabenstein

During the quarter, investigation of the fabrication of tubular fuel elements of various tube diameters and material compositions was undertaken. Cermets containing up to 30 wt % UO_2 were fabricated as were elements of up to 0.5 in. OD with 30-mil meat thickness and containing 23 wt % UO_2. Testing techniques were also investigated. Studies were initiated to determine the temperature dependence of boron loss, using various boron compounds and boron alloys. Definite relationships of boron loss to temperature and initial boron content were noted. Several boron analysis techniques were investigated and a procedure for recovering UO_2 from a sintered cermet strip was established.

1. Fuel Element Fabrication

The prime objective for the past quarter has been to establish parameters for the fabrication of tubes containing a cermet core with up to 30 wt % UO_2, a 0.030 in. meat thickness, and diameter up to 0.500 in.

The parameters established and under investigation have been:

(1) Effect of UO_2 and B_4C powder concentrations on green strip rolling.

(2) Effect of roll spacing on thickness of green strip and UO_2 losses during rolling.

(3) Effect of simulation of UO_2 with stainless steel in powder rolling.

(4) Effect of roll parameters, i.e., width, flow, etc., on density of rolled strip.

(5) Effect of furnace cleaning of cladding material.

(6) Effect of sintering furnace temperature on cermet strip density.

(7) Formability of cermet strip with increasing UO_2 content.

(8) Fabrication of tubes of larger diameter.

As each step in the process is optimized, additional areas of interest will be evaluated.

The manufacturing equipment that was previously employed to produce a core of 0.375 in OD elements is being utilized. The only exception is in the cold rolling of the cermet strip. For this operation, a 3 in. by 5 in. horizontal Stanat rolling mill is being used. A Fenn rolling mill is available but has not been converted to the four-high combination and so could not be used as yet. Tube forming dies for forming the larger diameter tubular cores are required. All samples to date have been hand worked. An X-ray unit in the Non-Destructive Testing Laboratory provided an effective tool for checking density variation on the green strip.

Materials used were as follows:

Matrix powder--Type 302B stainless steel

Fuel--Hi-fired (natural U) UO_2 in the -200 +325 mesh fraction

Clads and mandrels--Type 347 stainless steel clad, mild steel mandrels

Burnable poison--B_4C powder

Preliminary work was completed on controlling density variations in the green strip. The feed hooper was adjusted and proper rolling mill settings were established by rolling as-received stainless steel powder compacts. The change-over to rolling cermet material containing UO_2 and B_4C did not show any effects that could not be explained by the differing flow characteristics of the material.

The green cermet strip had a very uniform appearance and the density variation was found to be no more than 5%. The edges of the green strip were trimmed by dressing with a piece of wire mesh. It was found that removing this low density material minimized cracking during subsequent operations. The trimming also removed a source of contamination during future handling.

The yields obtained after rolling one test run are given in Table IV-4.

TABLE IV-4

Yield and Loss During Rolling of Green Strip

Initial hopper charge	-	2100	gm
Product--usable strip	-	1308	gm
Reusable powder	-	758.5	gm
Scrap	-	52.0	gm
Powder loss	-	2.5%	
Product strip	-	63%	

The overage of 58.5 gm is due to the pick up of stainless steel after a previous dummy run used to condition the surfaces of the rolls.

To date, several batches of cermet have been rolled; these include 23 wt % UO_2, 25 wt % UO_2 and 30 wt % UO_2 cermets.

Sections of the 23 wt % strip were further heat treated, rolled and processed into tubular cores. After cold rolling and dead end attachment, the strips were sheared into the widths required for both 0.375 in. OD and 0.500 in. OD elements. The shearing was accomplished by one cut per side and the strip was uniform enough to form into a tube. The swaging operation produced a tubular core with a good longitudinal seam.

In processing the tubular elements further, variations of the previously established procedure were initiated to check the feasibility of simplifying the process. For example, furnace cleaning in a hydrogen atmosphere (instead of chemical cleaning) was used on the cladding material and the tubular element. Use of this technique would substantially reduce the time and effort involved in chemical cleaning of the clad material. The effect of reducing sintering furnace temperature was investigated, but results to date are inconclusive.

Several elements of 0.375 in. OD to 0.500 in. OD containing 23 wt % UO_2 with 25-mil meat thickness were fabricated. The bonding on several samples was not perfect, but this is considered to be due to the techniques used to expand the tubes away from the mandrel, and the difficulty encountered in removing the mandrel.

Fuel element samples of 0.375 in. and 0.500 in. OD are shown in Figs. IV-1 and IV-2. The composition of the meat in both cases is 23 wt % UO_2, 0.22 wt % B as B_4C, with the remainder Type 302B stainless steel. Meat thickness in both cases is 25 mils and cladding thickness 6 mils. Figure IV-1 shows a sintered cermet strip with attached dead ends, the rough formed tube after being swaged, and a finished element. Figure IV-2 shows a finished 0.5 in. OD element.

Overall results to date indicate that:

(1) Green strip containing 30 wt % UO_2 can be rolled.

(2) Process losses during the shearing operation can be reduced by controlling the green strip size and density.

(3) Larger diameter tubes can be fabricated.

(4) Furnace cleaning of the components before they are assembled is feasible, however, more work is required in this area.

During the next quarter investigation of the fabrication parameters listed will continue. A series of runs will be made to determine approximate batch sizes and UO_2 process losses. Fuel element fabrication runs will be initiated on irradiation test specimens.

2. Ultrasonic Testing

The objectives of the ultrasonic testing program are to improve upon existing testing techniques through the use of improved single and double-wall methods and the establishment of defect standards for ultrasonic testing of PM-1 elements.

0.025 in. Cermet Strip with Dead Ends

Strip Swaged to Tube

Clad and Bonded Fuel Element

Fig. IV-1. Fuel Element Sample

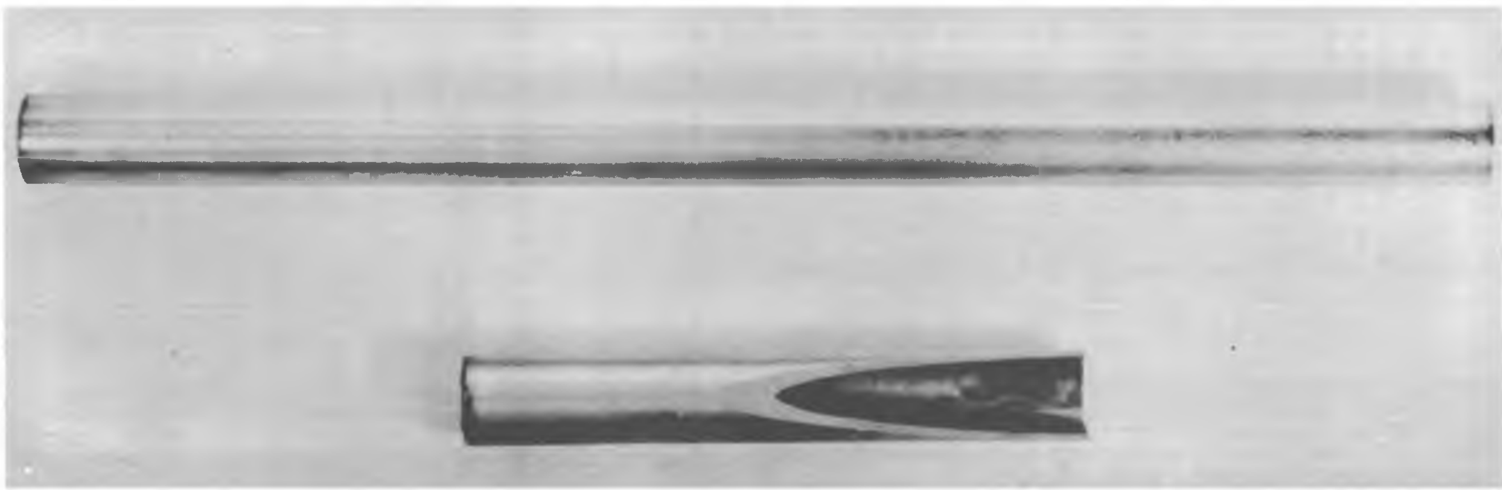


Fig. IV-2. Fuel Element Sample

In the first quarter, our immerscope was sent to Curtiss Wright for required repairs. This unit receives the signal from the crystal pickups and translates it into an electronic impulse proportional to the size and extent of the defect. The unit was repaired and returned later in the quarter. A long crystal was ordered to use in single-wall transmission measurements,but has not been received as yet. An electrostatic recorder, for use with the immerscope, was received and checked out. A substantial improvement in time response and in pinpointing defects over the previous method of defect recording was noted.

Investigation of single- and double-wall transmission technique improvement will be initiated in the next quarter. Sample fuel elements will be purposely defected and examined to establish defect criteria.

3. Control of Burnable Poison

During fabrication of fuel elements, some of the boron added to the cermet is lost and the boron content becomes uncertain. The purpose of these studies is to determine a method for adding boron that assures that the amount desired will remain in the fuel elements after fabrication.

The following parameters are being investigated:

(1) Effect of temperature on boron loss from B_4C-stainless steel specimens.

(2) Effect of temperature on boron loss from ZrB_2-stainless steel specimens.

(3) Effect of temperature on boron loss from a 0.38 B-stainless steel alloy.

(4) Effect of temperature on boron loss from a 0.07 B-10-stainless steel alloy.

(5) Effect of B_4C particle size on boron loss.

(6) Effect of precleaning stainless steel powders on boron loss.

(7) Effect of rate of gas flow on boron loss.

(8) Effect of specimen size on boron loss.

The raw materials used to date and their properties are given in Table IV-5.

TABLE V-5

Boron Containing Powders

Material	Source	Boron (Wet Chemistry)* (%)	Boron Theoretical (%)	Particle Size (mesh)
B_4C	The Norton Company	76.2	78.3	-325
				-230 +325
				-100 +230
ZrB_2	Cooper Metallurgical Association	18.8	19.2	-325
Type 302 B-Boron Alloy	Vanadium Alloy Steel Company	0.38	0.38	-100
Type 302-B^{10} Alloy	Vanadium Alloy Steel Company	Not Analyzed	0.06-0.08	-100
Type 302B-stainless steel	Vanadium Alloy Steel Company	0.02	0	-100

* Martin Analysis

Other raw materials, i.e., boron compounds and boron-stainless steel alloys, have been obtained which analyzed much lower than the stoichiometric boron content. These materials were not considered usable. Raw materials from other sources have been ordered and these materials will be tested if they analyze closer to stoichiometric than the ones being used.

The test specimens have a 3/4 in. diameter and are 0.100 in. thick. Material for each specimen is individually weighed. Weight measurements are made after compacting the powders at 30 tsi, and after sintering as a control to verify chemical analysis. Boron analysis is made on whole specimens to eliminate variations due to non-uniform boron distribution. The specimens were sintered at temperatures of 1000, 1050, 1100, and 1200°C. Figure IV-3 shows the range in the hot zone of the furnace and the position of the boat in the hot zone for the various temperatures. Dew points for the hydrogen atmosphere are -78°C entrance and a minimum of -60°C at exit.

Complete results have not been obtained on all materials investigated and some rechecks are being made. All results are, however, reported.

Figure IV-4 shows the variation in total weight loss with temperature for samples of B_4C and ZrB_2 in Type 302 B stainless steel. The connected points are the averages of three specimens at each temperature and the vertical lines represent the range of measurements found. It can be seen that the specimens generally lost weight, although those tested at 1050°C had a considerable weight spread and the ZrB_2 pellets gained weight in the runs at 1000 and 1100°C. Reruns are being made at 1050°C. It was anticipated that weight loss would increase with increasing temperature, since the tendency to drive off volatile materials is greater. Figure IV-5 shows the boron content versus temperature. Original boron content for the B_4C was 0.201% and for the ZrB_2, it was 0.205%. The test at 1050°C again shows a discrepancy.

The boron loss with B_4C appears to continually increase with temperature. It appears that use of the ZrB_2 would be more advantageous than B_4C, especially if a temperature of 1100°C could be used in sintering for the fuel element fabrication process. Further testing will be necessary to verify these findings.

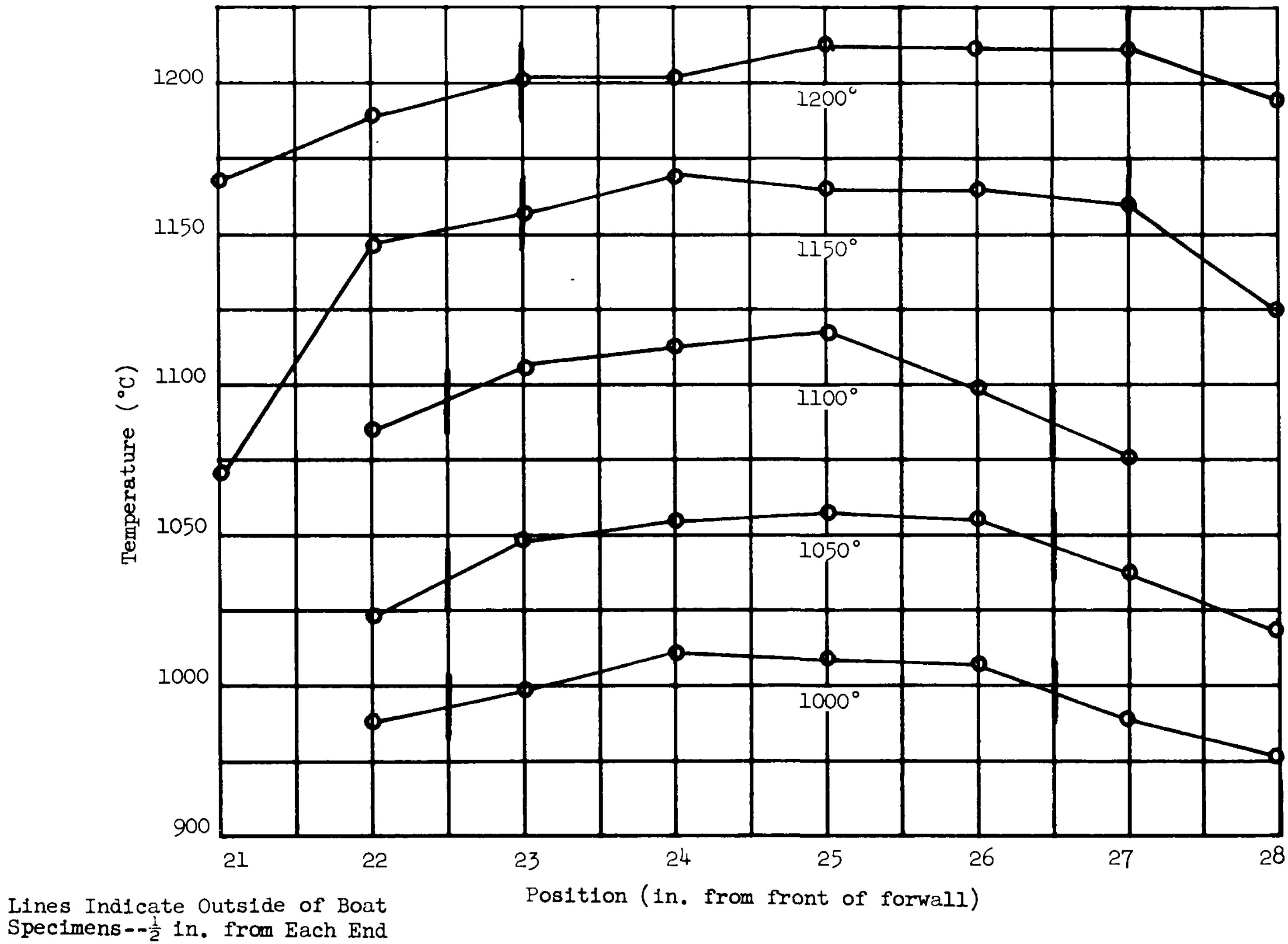


Fig. IV-3. Hot Zone Temperatures--Boron Sample Furnace

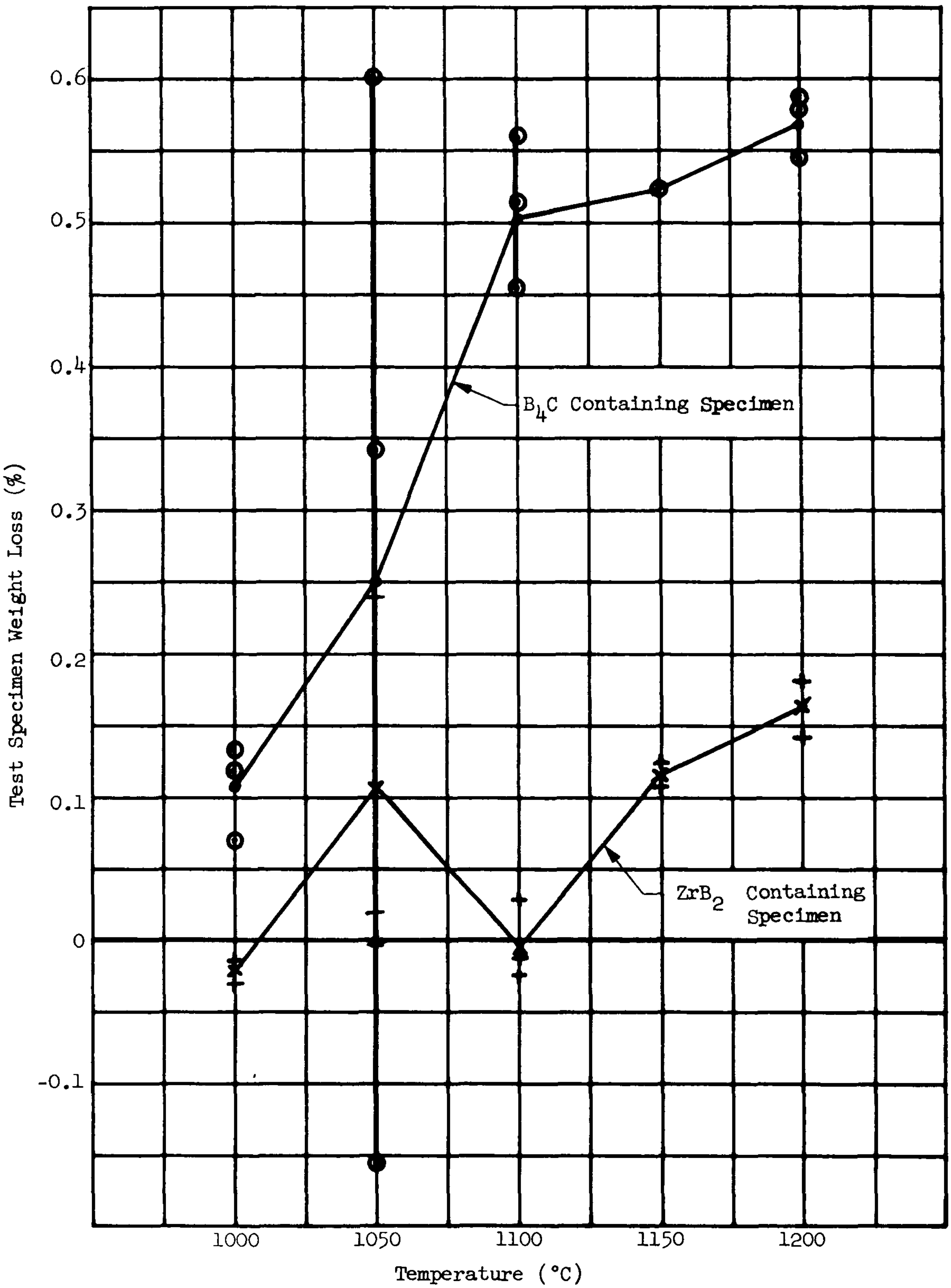


Fig. IV-4. Temperature vs Test Specimens Weight Loss (B_4C and ZrB_2 in Type 302B SS)

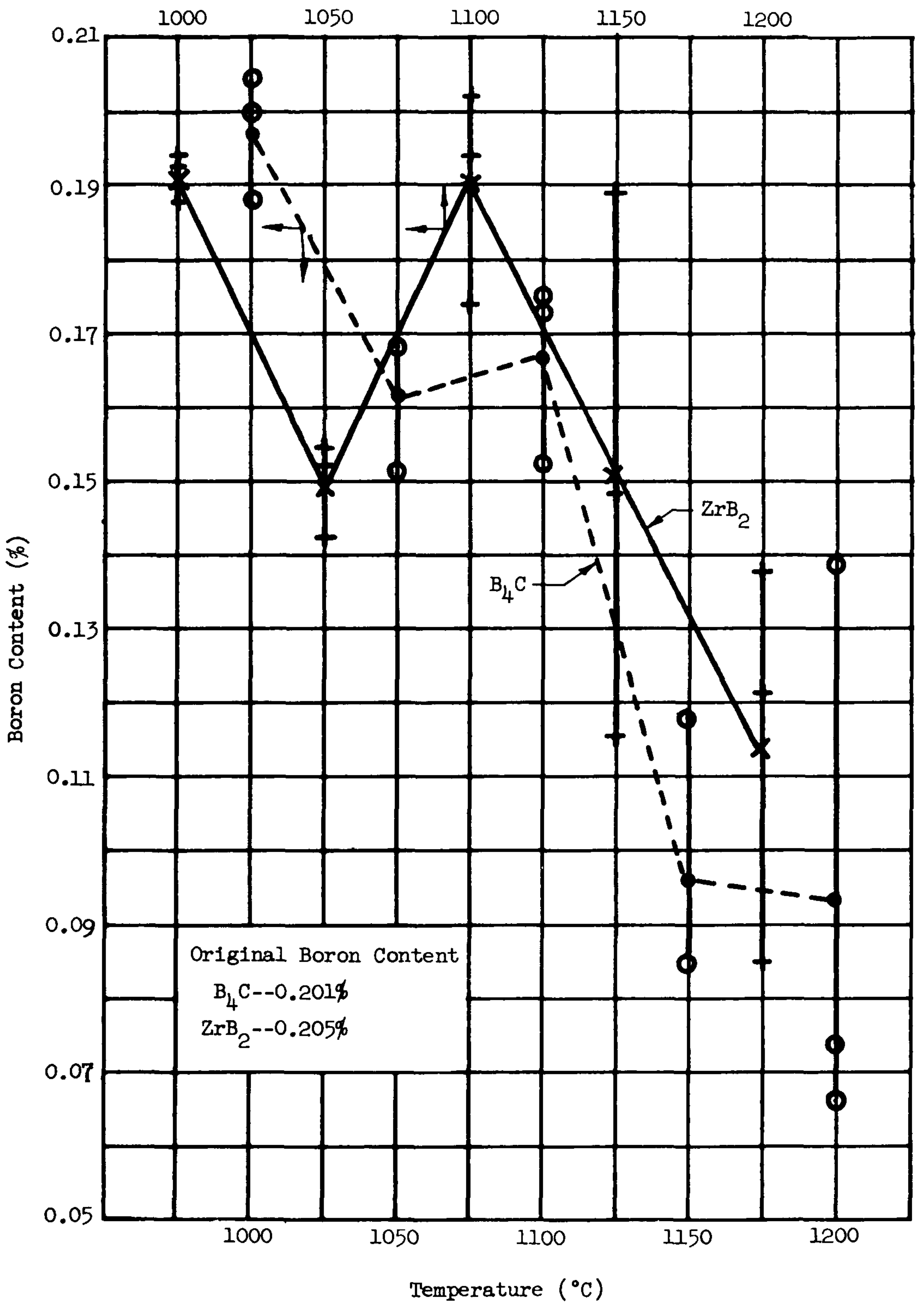


Fig. IV-5. Boron Content vs Temperature (B_4C and ZrB_2 in Type 302B SS)

In the study with the Type 302B stainless steel-0.38 boron alloy the boron content was varied by mixing the alloy with plain Type 302B stainless steel. Mixtures containing 25, 50, 75, and 100% of the Type 302B alloy were made; these contained 0.095, 0.190, 0.285 and 0.38% boron, respectively. Figure IV-6 shows the total sample weight loss obtained with these mixtures. The weight loss increases with temperature with all of the mixtures; however, at 1000°C, the weight loss is greatest for the 302B-0.38 B alloy, while at 1200°C the weight loss would probably be greatest for the Type 302B stainless steel. At this point no explanation can be given for this phenomenon. Table IV-6 shows the boron lost from various loadings at different temperatures. It can be seen that the boron loss is relatively independent of the amount of boron initially added, but that the loss is dependent upon temperature. Figure IV-7 shows plots of the losses from the various loadings against temperature. This plot shows that for an increase of 100°C the boron loss is doubled.

Tests on a 302B-0.07 B-10 stainless steel alloy were initiated. It is assumed that most, if not all, of the boron in this alloy will be in solid solution. Since it should be much more difficult to remove material in solid solution as opposed to material in a second phase, the boron loss from this alloy is expected to be much less.

Tests on the boron loss from B_4C of different particle sizes were also initiated. It is expected that the decrease in total surface area resulting from using the larger size particles will greatly decrease activity leading to losses.

Tests are also in process using precleaned stainless steel to determine if the oxide film on the stainless steel is a source of oxygen which promotes boron loss. The effects of hydrogen gas flow rates and specimen size or boron losses will be investigated after the tests discussed are completed.

In summation, ZrB_2 appears to be more stable with temperature than B_4C. Overall weight loss for plain Type 302B stainless steel appears to be greater than for 302B stainless steel-boron alloy at elevated temperatures.

It appears that the boron loss in boron-stainless steel alloy is relatively independent of the amount of boron added, but boron loss is still temperature-dependent.

Tests on all materials and alloys mentioned will be completed during the next quarter, as will tests on 302B-0.07 B-10 stainless steel alloy.

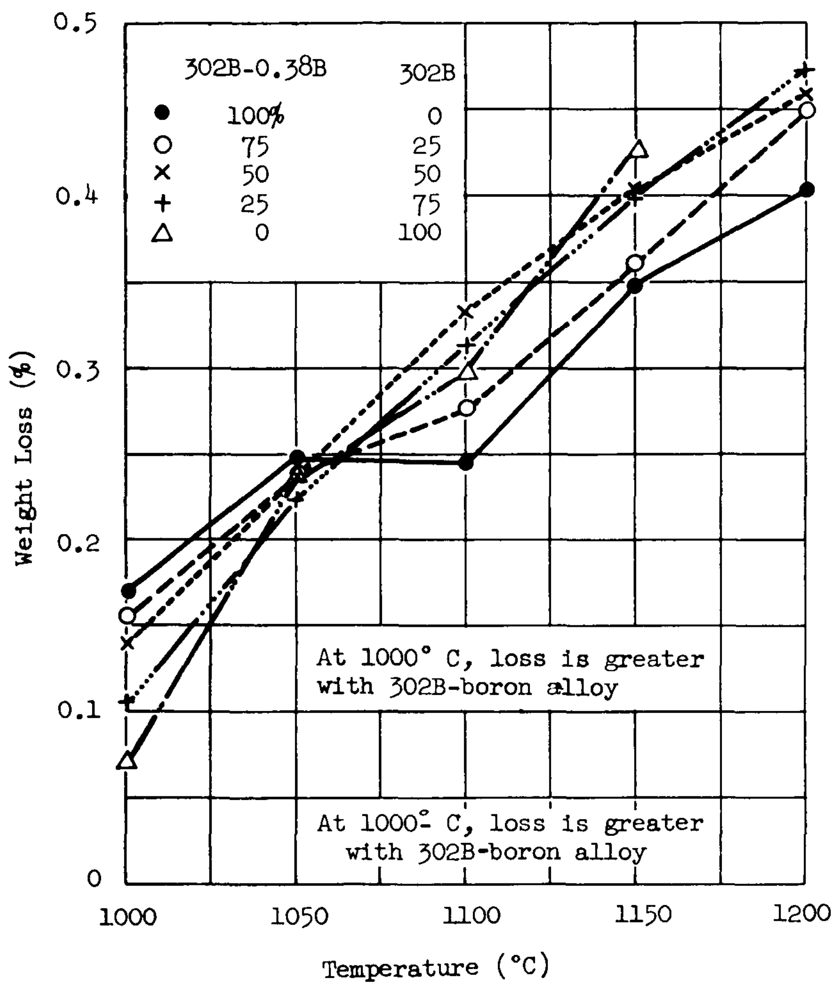


Fig. IV-6. Weight Loss vs Temperature (Type 302B Stainless Steel--0.38B)

TABLE IV-6

Boron Loss vs Temperature

302B Stainless Steel --0.38B Alloy

Temperature (°F)			1000°C	1050°C	1100°C	1150°C	1200°C
Ratio		B Added (wt %)	Losses (wt %)				
302B-0.38B	302B						
100	0	0.38	0.029	Analysis	0.050	0.067	0.098
75	25	0.285	0.020	not	0.047	0.077	0.095
50	50	0.190	0.020	completed	0.066	0.090	0.100
25	75	0.095	0.017		0.047	0.069	0.091
Average Loss			0.023		0.052	0.076	0.096

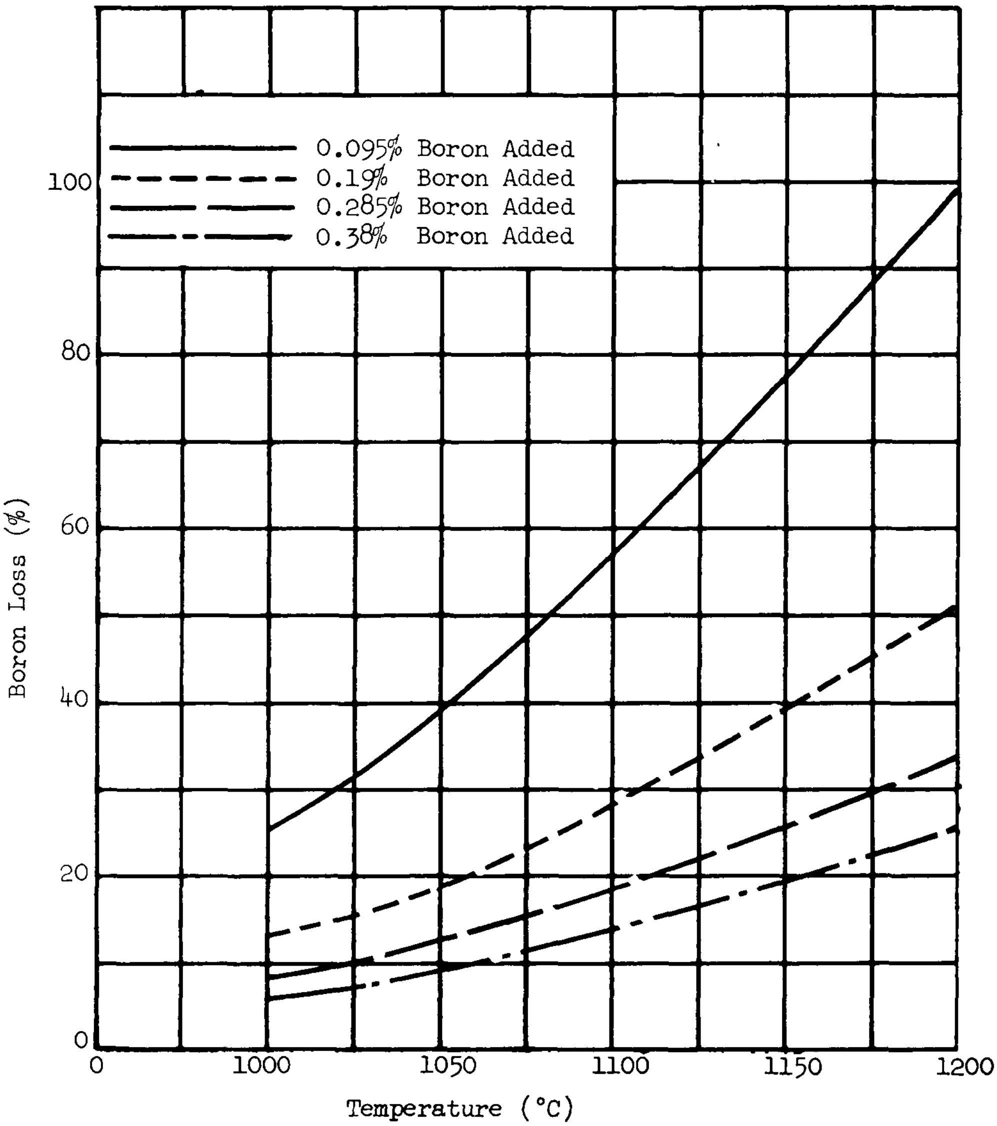


Fig. IV-7. Percent Boron Lost vs Temperature (SS-B alloy)

Tests of boron loss in B_4C as a function of particle size will continue, as will tests to determine the effect on boron loss of the oxide film on stainless steel.

4. Boron Chemical Analysis

The purposes of this study are threefold:

(1) To standardize and check for accuracy the two methods used here for boron chemical analysis.

(2) To furnish, without delay, backup work for boron loss studies.

(3) To make literature surveys of other methods now in use at other installations for possible use here.

Since there was an immediate need for reliable boron analysis of sintered pellets, the initial effort has been devoted entirely to standardizing the mercury cathode electrolysis and ion-exchange methods previously used. Special attention was given to the dissolution of the sample, since boron compounds are so volatile that boron is easily lost during this step. A combined acid-reflux and alkaline fusion dissolution was found effective for the various types of samples (ZrB_2-SS, B_4C-SS, and B-SS). When conditions were found which would hold the boron loss to a negligible quantity during the dissolution step, these conditions were adopted as standard and written into the procedure along with the subsequent steps of the analysis.

Boron samples analysed for backup work presently consist of three different types. These are powder blends, unsintered pellets, and sintered pellets of each of the folloing:

ZrB_2--302 B weighing 4.4 gm

B_4C--302 B weighing 4.4 gm

302 B (containing B)--302B weighing 2.0 gm

Twenty-one samples were prepared by mixing weighed quantities of ZrB_2, B_4C, or 302 B (containing B) powder with weighed quantities of Type 302 B stainless steel powder to form a 4.4-gm sample in the case of ZrB_2 and B_4C blends, or a 2.0 gm sample in the case of the Type 302 B blends. Each sample was completely dissolved and analyzed. The analyst was not aware of sample composition. The results are given in Table IV-7.

TABLE IV-7

Sample Analyses -- Boron-Containing Materials

Type of Sample	Theoretical Boron Added (%)	Boron Found (%)
100% 302B (containing B)-- 0% 302B	0.380	0.383
100% 302B (containing B)-- 0% 302B	0.380	0.383
100% 302B (containing B)-- 0% 302B	0.380	0.389
75% 302B (containing B)--25% 302B3	0.285	0.286
75% 302B (containing B)--25% 302B3	0.285	0.284
75% 302B (containing B)--25% 302B3	0.285	0.287
50% 302B (containing B)--50% 302B3	0.190	0.193
50% 302B (containing B)--50% 302B3	0.190	0.188
50% 302B (containing B)--50% 302B3	0.190	0.194
25% 302B (containing B)--75% 302B3	0.095	0.094
25% 302B (containing B)--75% 302B3	0.095	0.101
25% 302B (containing B)--75% 302B3	0.095	0.094
0% 302B (containing B)--100% 302B3	0.002	0.003
0% 302B (containing B)--100% 302B3	0.002	0.002
0% 302B (containing B)--100% 302B3	0.002	0.004
ZrB_2--302B	0.213	0.206
ZrB_2--302B	0.213	0.209
ZrB_2--302B	0.213	0.208
B_4C--302B	0.212	0.209
B_4C--302B	0.212	0.207
B_4C--302B	0.212	0.202

Surveys of the existing literature on boron analysis have been conducted but no reported method offers any advantage over the methods now used, either in accuracy of result or in time expended per sample. In the interest of uniformity of results and also because of the existing back-log of samples, no further experimentation in methods development has been planned until after the current series of bcron loss tests has been completed.

A method for the recovery of UO_2 from sintered green strip has been developed. Approximately 90% recovery of usable material is possible if slight stainless steel impurities can be tolerated.

During the next quarter, backup analyses for the boron program will continue.

Analysis of process sample elements will be initiated in support of the fabrication process development.

V. 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

P. Martin

This subtask was activated 13 mo earlier than was originally scheduled. The main objectives for the immediate effort under Subtask 11.1 were to evaluate the various locations about the radar station site which are adaptable to the PM-1 requirements and to make a firm site recommendation to the AEC. Once a PM-1 location has been approved, it is necessary to perform a detailed field survey and to obtain soil data to allow preparation of a plant foundation design; this work is expected to be accomplished during the next reporting period. The effort under this subtask to date has concerned the evaluation of possible PM-1 power plant locations in or near the Operations Area of the Sundance Air Force Station and a recommendation by The Martin Company as to its preferred location.

In evaluating the various suitable locations for the nuclear power plant, consideration was given to the general site environment in terms of population density, meteorology, geology, hydrology, and seismology. These factors were considered and will be discussed in greater detail later in this report (Subtask 17.1).

The Warren Peaks area, the site of the Sundance AFS Operations Area, consists of treeless peaks covered with wild grass and other vegetation. A US Forest Service lookout tower is located on the highest point. One to three feet of soil are present over bedrock. Essentially uniform drainage from the peaks occurs in all directions.

Hutchins Springs, approximately 3/4 mi northwest of the operations area, in the Lytle Creek watershed, will supply water.

Following a field survey trip and discussions with representatives of the Atomic Energy Commission, Air Force Installations, the Omaha District of the Corps of Engineers, and the Air Defense Command, several possible locations were made available for consideration.

These were screened and the three most suitable locations were selected for more detailed evaluation. Each was considered to the extent necessary to allow estimates to be made of the order of magnitude of work required to fully develop the location; to demonstrate that the power plant could be physically situated within the location without interfering with other proposed buildings or with the radar search beams; and to establish major advantages and disadvantages. For the purposes of these evaluations, a conventionally contained version of the PM-1 plant was assumed.

Preliminary plan and section drawings of each location were prepared in which the outside dimension of the primary loop containment vessel was indicated.

Each location evaluated had sufficient slope to allow normal plant drainage; this made it unnecessary to provide expensive tile and catch basins. Drainage was, therefore, a concern only from the nuclear hazards standpoint. The general isolation of the Warren Peaks area is extremely favorable, in this respect, since the problem reduces to consideration of the potential contamination of the operations area water supply at Hutchins Springs.

An important evaluation consideration was the relationship of the operations area to the location with regard to prevailing winds. It was assumed, due to the lack of specific data, that the prevailing winds would be westerly; i.e., within the southwest to northwest quadrant.

The advantage of an exclusion distance between the nuclear power plant and other facilities was considered with regard to a potential nuclear accident. Radiation levels may be expected to be high in the immediate vicinity of the power plant for a short time, in the event that a nuclear accident should occur. Considering the inverse square relationship between radiation intensity and distance, and the delay time required for any cloud to travel from the reactor to the operations area, an exclusion distance of the order of a few hundred feet significantly reduces the potential hazards to Operations Area personnel. On the other hand, during normal operation it will be desirable to combine, as much as possible, the nuclear power plant and the heating and emergency power plant operating crews. This can best be achieved if the distance between the two plants is held to a minimum. The capability of connecting the two power plants with an enclosed walkway was considered to be an important advantage.

Other features evaluated included:

(1) Possible plant integration into the Operations Area security system.

(2) Length of piping and electrical lines required to join the nuclear power plant, and the heating plant and emergency power building.

(3) Cost of PM-1 site preparation in terms of: excavation required, fill required, additional access roads required, space for convenient storage and subassembly of PM-1 plant components during erection, and possible relocation of operations area facilities as a result of PM-1 construction.

(4) Interference with the normal functions of the Operations Area during construction and normal operation of the PM-1 plant.

1. Location I

Location I (Fig. V-1) is within the operations area between the FPS-7 and FPS-26 radar towers 175 ft west-southwest (approximately 260°) from the present Forest Service lookout tower.

Estimated Distances from Various Structures in the Complex

Buildings or Towers	Primary Loop (ft)	Secondary Loop (ft)
FPS-7	97	12
FPS-26	17	32
Emergency Power Building	70	18
Operations Building	175	95

The finished floor elevation of the secondary loop is 6637 ft with the highest point of finished construction being 6657 ft.

Drainage would flow into Lytle Creek and onto the same watershed as Hutchins Springs, the Operations Area water source. About five miles down Lytle Creek is a ranch which also uses water from the creek for livestock. Lytle Creek flows northwest.

The prevailing westerly winds present an unfavorable wind condition.

The existing slope is 30%, but this will be filled and leveled in the construction of the radar site. The most representative test pit data shows 1 ft of topsoil; 2 ft of gravel, containing clay and large rock; 1.5 ft of weathered bedrock; and solid bedrock below.

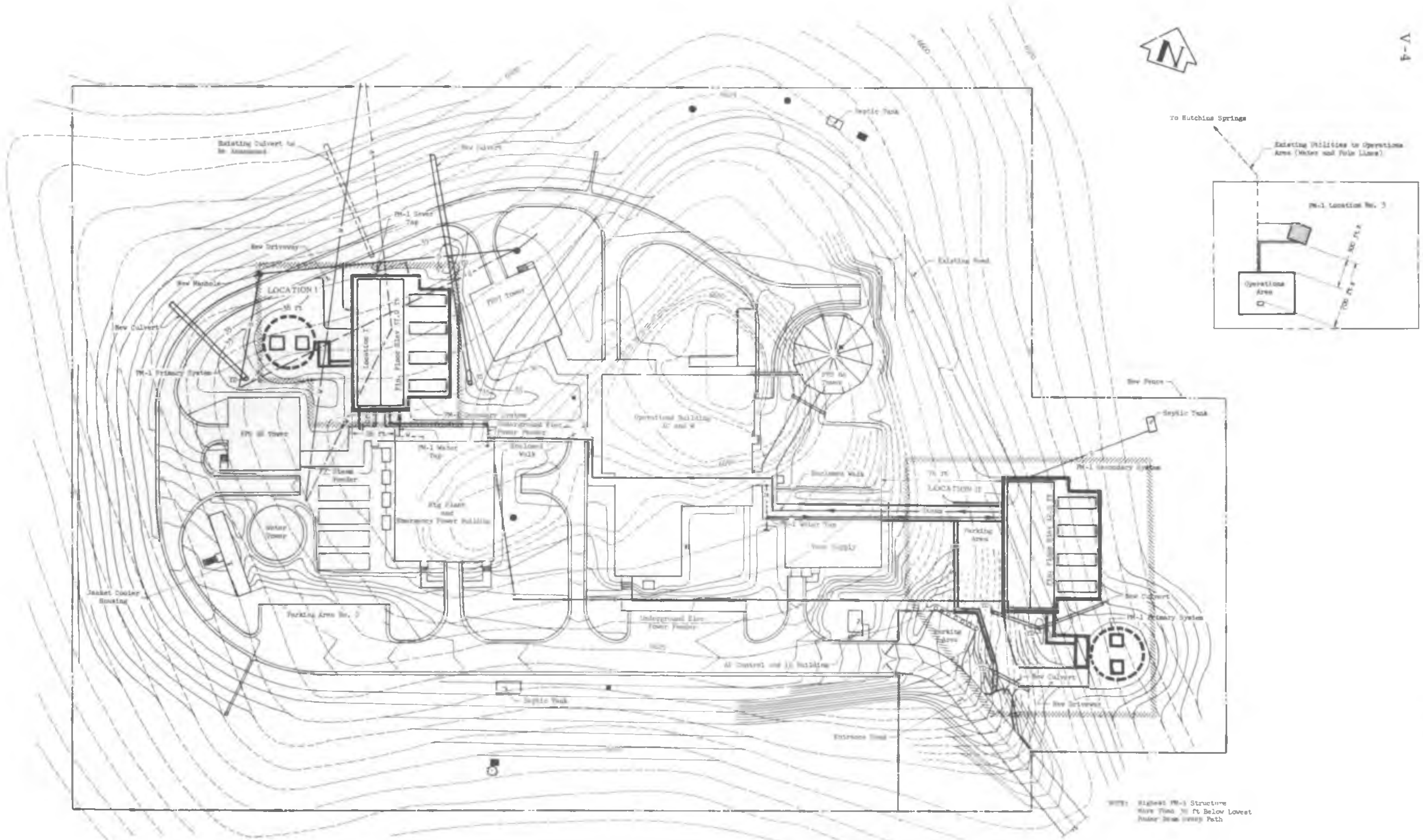


Fig. V-1. PM-1 Plant Locations--Sundance Air Force Station

Advantages.- The major advantages of this location would be:

(1) Short utility runs

(2) Minimum additional enclosed walks

(3) Minimum site development

(4) Minimum additional security requirements.

Disadvantages.- The major disadvantages of this location would be:

(1) No exclusion area

(2) Location upwind from most of Operations Area

(3) Drainage to Operations Area water source

(4) Jeopardy to existing structures during construction

(5) Possible interference with radar operation during construction

(6) Relocation of existing utility lines.

2. Location II

Location II (Fig. V-1) is located approximately due east and 180 ft from the closest point of construction of the present Forest Service lookout tower. This site straddles the location of the security fence on the east side of the hill and is 105 ft from FPS-6A radar tower, 75 ft from Technical Supply, and 155 ft from the Operations Building. The finished floor level elevation of the secondary system will be 6612 ft and the highest point of finished construction 6632 ft. The slope of the hillside is estimated to be 37.5%.

The drainage flows into Beaver Creek which flows east for a short distance and then turns abruptly northward. There are two springs (Davis and Cole Springs) with a flow of 10-20 gpm about 5 mi from the site on Beaver Creek, with water flow increasing a short distance downstream to 30 gpm. Numerous beaver dams exist along the upstream portions of Beaver Creek.

A typical soil profile for this site is shown in test pit data as 1 ft of topsoil, 3.5 ft of weathered bedrock, and solid bedrock below.

Advantages.- The major advantages of this location are:

(1) Some isolation from the Operations Area

(2) Possible use of terrain as natural shielding

(3) Essentially downwind from the Operations Area

(4) Can be connected to the Operations Area via an enclosed walkway.

Disadvantages.- The major disadvantages of this location are:

(1) Hillside construction

(2) Long utility runs

(3) Additional site development.

3. Location III

Location III (Fig. V-1) is located approximately 700 ft north-northeast of the present Forest Service lookout tower. This site is on the existing roadbed on the east side of a saddle. The finished floor level elevation of the secondary loop will be 6498 ft with the highest point of finished construction being 6518 ft. This saddle is fairly level on top and drops off sharply on each side. The east slope is estimated to be 52%.

The drainage is the same as Location II and flows into Beaver Creek. The only unfavorable wind direction is from the north-northeast with only slight effects from north and east winds.

No test pit data is available for this site. The nearest pit, some 300 ft away, shows a soil profile of 1 ft of topsoil, 4 ft of gravel containing clay and large rock, and solid bedrock below; a similar condition may be anticipated at Location III. The more level land at the saddle allows for the deposition of material washed from the upper slopes and, since runoff is slower, water penetrates much deeper, thereby causing greater decomposition and weathering of sub-surface materials. This area is somewhat protected from the wind by a grove of trees beginning just west of the saddle.

Advantages.- The major advantages of this location are:

(1) Exclusion from the Operations Area

(2) Minimum required excavation

(3) Favorable wind patterns.

Disadvantages.- The major disadvantages of this location are:

(1) Length of utility runs

(2) Requirement for separate septic tank

(3) Requirement for separate security fence and guard station

(4) Poor personnel access to and from the Operations Area.

4. Recommended Location

The major factors considered in evaluating the three proposed locations are summarized in Table V-1 together with the relative standing of each individual location. The major factors are listed in approximately descending order of importance.

Based upon this evaluation, The Martin Company recommended that Location II on the east slope of the Operations Area be selected for the PM-1 plant site. Location II is the only location that satisfactorily meets all of the major requirements for the location of the nuclear power plant.

TABLE V-1

Summary of Location Evaluations

Major Factors	Location I	Location II	Location III
Location development (length of lines, excavation, roads, septic tanks, etc.)	Good	Satisfactory	Satisfactory
Favorable wind direction	Poor	Good	Good
Proximity to heating plant and emergency power building	Good	Satisfactory	Poor
Exclusion distance	Poor	Satisfactory	Good
Drainage away from Hutchins Springs	Poor	Good	Good
Integration with existing security system	Good	Good	Poor
Noninterference with radar operations	Poor	Good	Good

VI. TASK 14 - TRAINING

Project Engineer Subtask 14.1: L. Burns

The objectives of this task are to develop and implement a program to train military personnel to supervise and conduct the operation and maintenance of the PM-1 nuclear power plant.

A. SUBTASK 14.1 - TRAINING PROGRAM DEVELOPMENT

F. McGinty

Subtask 14.1, Training Program Development, was initiated during this quarter to assure the proper integration of training into the entire PM-1 program at an early stage. Three general areas of work were started during this period; namely, training program scheduling, topical outlining of the Training Manual, and a personnel analysis.

The preparation of the preliminary overall training program schedule was completed (Fig. VI-1). It includes the key points in the development of the training program, as well as some of the key points of the PM-1 program that have a decided influence on the training program development.

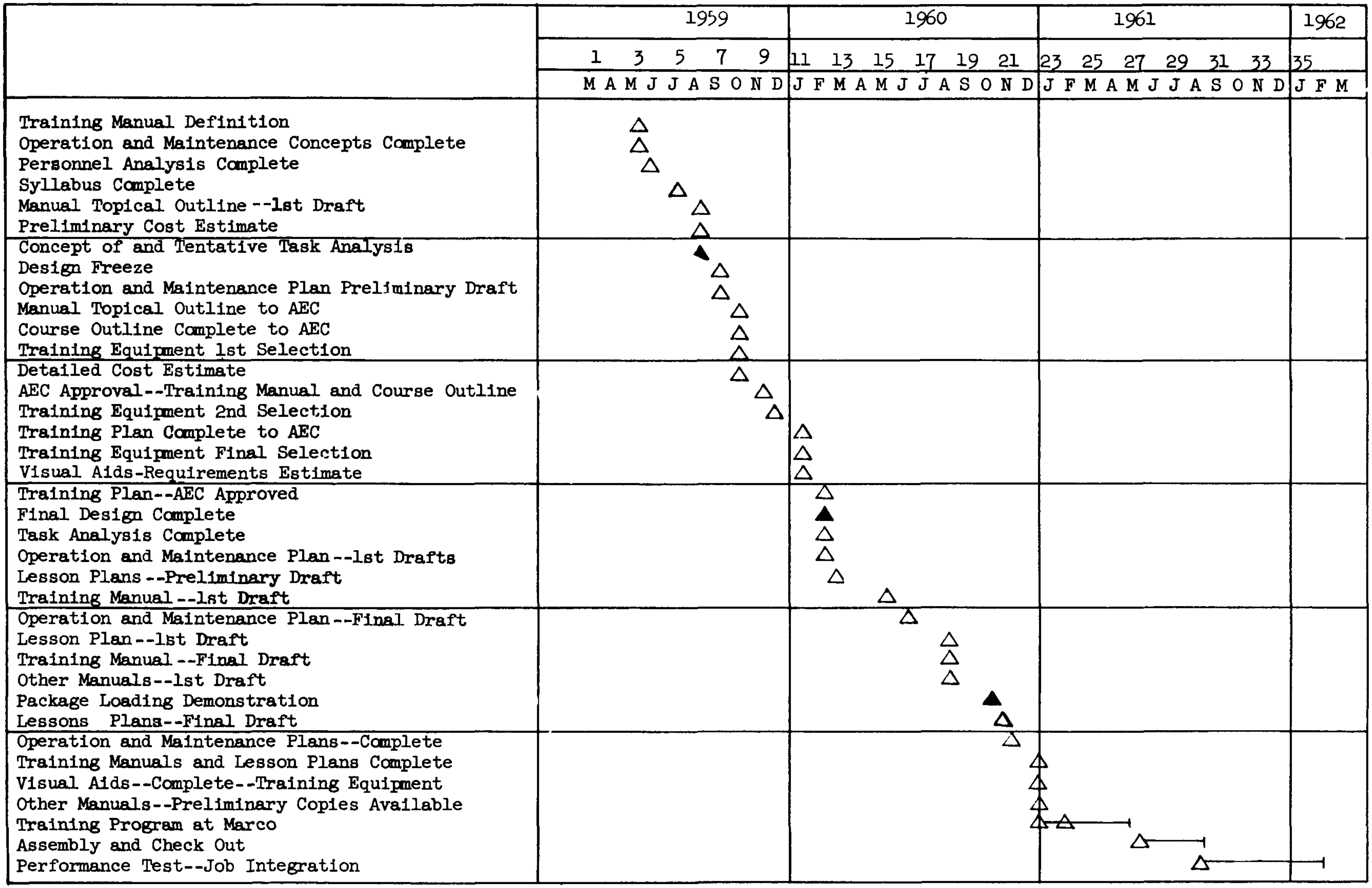
A concept of the Training Manual was developed, and a preliminary topical outline of its contents has been prepared. The Training Manual will include course outlines, course charts, lesson plans, student work sheets, evaluation tests, and training aid requirement listings. The lesson plans will not contain all of the detailed technical information on the PM-1 but merely outlines of the lessons that will be presented. The Technical Manuals, supplied by equipment vendors, will serve as the source of detail information. The reasons for this approach are:

(1) The student will learn to use and rely on the vendors' technical manuals during the training program.

(2) The vendor manuals will be kept current with revisions and changes in the equipment, whereas a training manual and training materials are not kept current and could, at some future data, present technical data or operating and maintenance instructions incompatible with subsequent equipment modifications.

(3) The vendor manuals are supplied with the equipment and will be available in their latest revisions to operating and maintenance personnel.

Preliminary Time Sequence Chart for Training Program PM-1



To define the program to be used in training the PM-1 operating crew, an analysis of their background skills, knowledge, and previous training is required. As all PM-1 operating and maintenance personnel will have completed the one year Fort Belvoir APPR-1 "Nuclear Power Plant Operators Course" (1958-1959), this training program is being studied in detail as representing the students' previous knowledge, training, and skill. This phase of the personnel analysis is 50% complete. The analysis will be completed with a detailed study of individual personal history folders and an evaluation of performance and development during the Fort Belvoir training program. No conclusions as to the detailed training requirement can be made until the completed personnel analysis can be compared to the operation and maintenance procedures for the PM-1.

The first PM-1 training meeting is scheduled to take place on 5 June 1959, at Fort Belvoir. Military, AEC, and Martin Company personnel will be participating. The purposes of the meeting are:

(1) To establish liaison between Martin and Fort Belvoir training personnel.

(2) To present the Martin approach to training.

(3) To present and discuss the concept of the Training Manual, and the preliminary topical outline of its contents.

(4) To discuss time phasing of students into the PM-1 training program.

(5) To familiarize Martin training personnel with the Fort Belvoir Nuclear Power Plant School facilities.

Other anticipated accomplishments for the next reporting period are:

(1) Writing of a syllabus of instruction for the PM-1 training program, containing course scope, course objectives, prerequisites, and tentative schedules.

(2) Based on the results of the June 5 meeting, preparing a second draft of the Topical Outline of the Training Manual.

(3) Completing studinent personnel analysis and initiating comparisons between it and the Task Analysis information.

(4) Continuing liaison with Fort Belvoir and AEC personnel on planning training. This will include the establishment of lesson plan and course outline formats, as well as work on the personnel analysis phase of the training program development.

VII. TASK 16 - CONSULTING

During the first quarter, the Gibbs and Hill Company was retained as a consultant in the general area of power plant design. They have assisted, and will continue to assist in evaluating the secondary system work being performed by the Westinghouse Electric Corporation.

Their specific contributions were as follows:

(1) Review of heat balances and steam generator data.

(2) Analysis of the ALPR steam-air condenser system performance.

(3) A comparative study of ethylene glycol, direct air-steam, and direct contact water cooled-water-air condenser systems.

Dr. T. J. Thompson of MIT was invited to visit The Martin Company for a one-day discussion of PM-1 hazards evaluation concepts.

During the next quarter the consulting efforts of Gibbs and Hill are expected to continue. Dr. Thompson will also be retained on an as-needed basis to provide assistance in the areas of core design and hazards analysis.

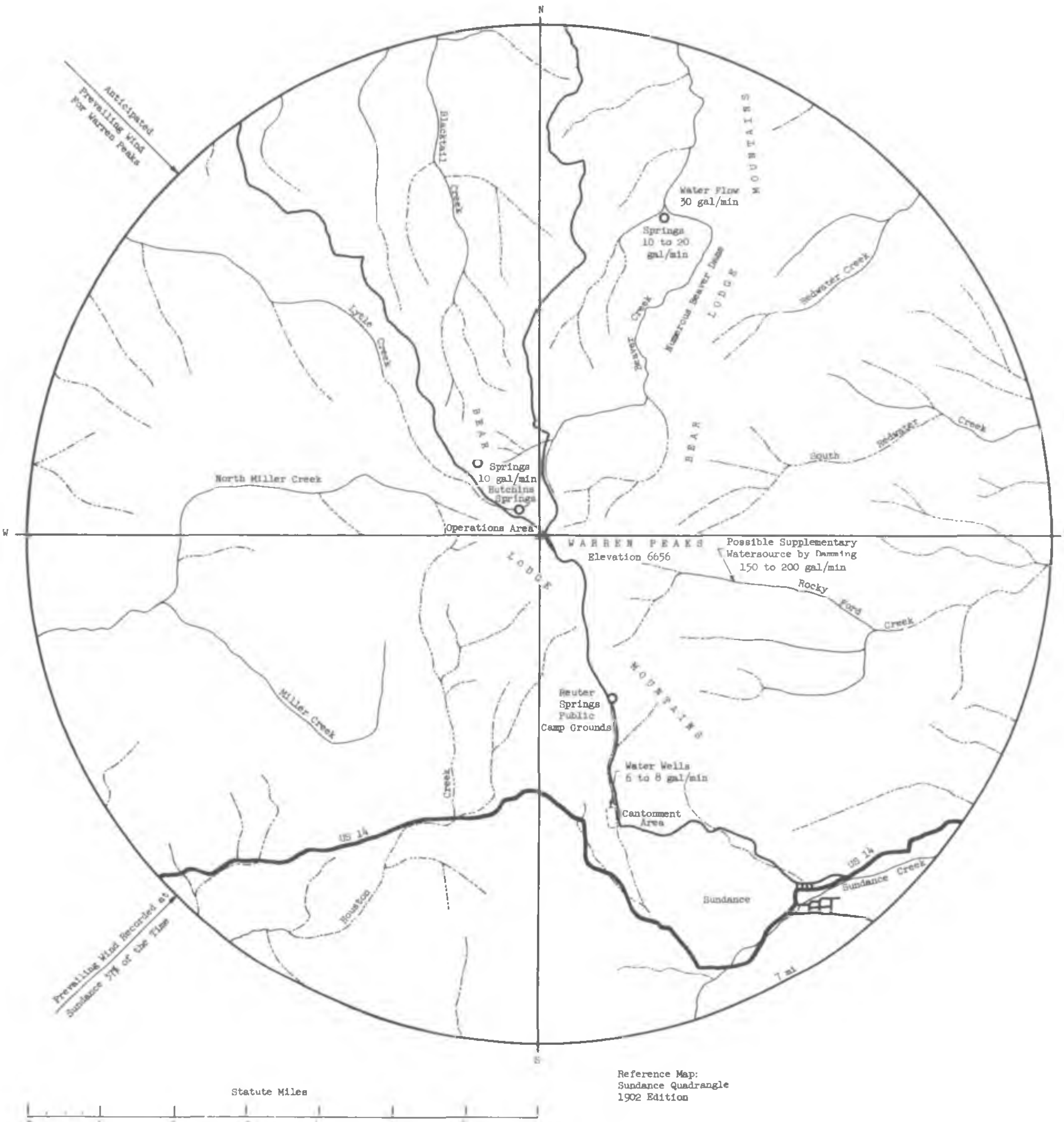


Fig. VIII-1. Drainage and Water Sources for PM-1 Site Area

National Forest. The PM-1 will be constructed to support the radar station and will be located within a few hundred feet of the radar operations area on Warren Peaks.

2. Population

Military.- The military personnel are concentrated in two areas which are significant to a hazards evaluation. The first is that of the radar station which will be manned by not more than 25 people at any one time. It is of primary importance in evaluating the consequences of a reactor accident because of its proximity to the PM-1 plant. An area of secondary importance is the Cantonment Area, located approximately 4 mi south of Warren Peaks, which will house 195 people, mainly military personnel and their families.

Civilian.- The town of Sundance is 6 mi southeast of Warren Peaks and represents the largest center of population in the area. Its population was 893 according to the 1950 census with recent local estimates ranging as high as 1000. Aside from this, population is confined to residents of scattered farm homes and ranches. About 50 residents are located within 7 mi of Warren Peaks. Other than these there are no significant population groups within a 15-mi radius of Warren Peaks.

In addition to permanent residents, the summer months attract transient people, such as campers, fishermen, prospectors, cattle grazers, etc., who utilize the resources of the national forest. Although it is impossible to estimate the exact number of transients in the area for any given period, their number is expected to be small.

3. Meteorology

General climatology.- The following climatic description is representative of the town of Sundance, which is almost 2000 ft lower in elevation than Warren Peaks and is somewhat sheltered from the prevailing winds.

The climate of Sundance is described as semi-arid with long cold winters and short hot summers. During the winter months more than 50% of possible sunshine is received; summer days are hot and nights are cool; the humidity is very low. There are few summer nights when the temperature remains above 60° F. During the warmest month, July, the average minimum temperature is in the mid 50's although temperatures above 90° are noted frequently. The winters are long and cold with January being the coldest month, however, the clear skies and dry air moderate the cold by permitting greater absorption of solar radiation. The cold weather comes from outbreaks of cold Canadian air moving southeastward from the Rockies, across the plains, to the Black

Hills. The initial onslaught of arctic air is usually accompanied by strong northerly or northwesterly winds with drifting snow. The coldest nights are associated with clear skies and very light winds.

More than half of the annual precipitation of 18.5 in. falls during the period from April through July. The three winter months constitute the period with the least moisture. Amounts of snowfall are quite large but the water content of the snow is usually low. During the spring months of March and April, precipitation often begins as rain mixed with snow which later turns into a heavy, wet snowfall; these snowstorms are frequently accompanied by strong winds and drifting. On the average, March has more snow than any other month. It is estimated that the average annual precipitation for Warren Peaks is 23 in. and the mean annual snowfall 110 in.

Freezing temperatures have occurred in the spring as late as June 25th and as early in the fall as August 28th. It must be realized that these dates will be considerably altered with the increased elevation and exposed position of Warren Peaks. Here, the last freezing temperature could conceivably occur on July 15th, and the first on July 16th. The average date of the last freezing temperature in the spring is May 26th and the first freezing temperature in the fall is September 18th.

Wind flow.- The only wind data available for the town of Sundance is old and questionable. The data (Fig. VIII-1) include direction only; wind speed was not available. Although the prevailing direction is from the southwest, it should be noted that Sundance is located in a valley between two mountain ranges which rise approximately 2000 ft above the surrounding terrain. This valley runs in a southwest-northeast direction and establishes a funneling effect. In all probability, this explains the reason why Sundance has a high percentage of southwesterlies in a region where prevailing winds are from the west and northwest. It is anticipated that at Warren Peaks winds will blow from southwest through northwest 80% of the time but that a higher portion of these winds will blow from west and northwest than was the case at Sundance.

As stated before, no reliable wind speed data are available for Warren Peaks. However, 28 days of scattered data was obtained from the Forest Service. These data were taken at the lookout tower on Warren Peaks during the period from 19 May through 9 July 1958, recorded at 1:00 PM. The average wind speed for this 28 days of record was approximately 15 mph with the lowest recorded at 5 mph and the highest 38 mph. Although no valid conclusions can be drawn from these data, they tend to indicate that wind velocities will generally be rather high and that calm conditions will seldom, if ever, exist.

Temperature.- The temperature recorded at the Forest Service lookout tower indicate that Warren Peaks is considerably colder than Sundance. The mean temperature at 1:00 PM for the 28 days of record was 59° F, with a high of 72° F on May 28th and a low of 28° F on July 9th. Although not directly applicable to Warren Peaks, the following table lists the monthly means and extremes of temperature for Sundance. No other temperature data are available.

TABLE VIII-1

Temperature Data--Sundance, Wyoming
45-Yr Record

Month	Maximum (° F)	Mean (° F)	Minimum (° F)
January	60	19	-39
February	66	21	-42
March	72	28	-24
April	82	40	-10
May	101	51	7
June	99	60	25
July	105	68	34
August	100	67	30
September	99	57	6
October	89	45	-17
November	87	31	-21
December	65	23	-32
Annual	105	43	-42

Precipitation.- The following table lists the means and extremes of precipitation for 48yr of record at the town of Sundance:

TABLE VIII-2

Precipitation Data--Sundance, Wyoming

	(in.)
Mean annual	18.5
Maximum year	27.81 (1922)
Minimum year	11.58 (1954)
Maximum mean monthly	3.3 (June)
Minimum mean monthly	0.8 (February)
Maximum amount in 1 mo	6.24 (June 1941)
Minimum amount in 1 mo	Trace inches (November 1917)
Maximum amount in a 24-hr period	2.73 (June 1940)
Mean annual number of days with 0.01 in. or more	86 days
Snowfall (40 yr of data)	
Mean annual snowfall	79
Maximum snowfall in 1 winter	168.7 (1916-17)
Maximum snowfall in 1 mo	53.5 (December 1916)

The precipitation and snowfall data listed for Sundance are not considered representative of Warren Peaks. Empirical estimates for Warren Peaks are:

	(in.)
Mean annual precipitation	23
Mean annual snowfall	110

4. Geology

The Bear Lodge Mountains were formed by an intrusive lenticular shaped body of molten igneous rock, known geologically as a laccolith, which forced itself between overlying sedimentary strata and the underlying mica schists. In so doing, the sedimentary rocks were uplifted into a dome-shaped structure forming the Bear Lodge Mountains. This geologic event was contemporary with and is considered a part of the Black Hills uplift (early Tertiary Era). Erosional processes have since removed the sedimentary rocks from the top and higher slopes of the dome, exposing the igneous core of the mountains. Sedimentary rocks are present only at lower elevations, around the flanks of the mountains.

Igneous monzonite-syenite porphyry bedrock underlies the entire Warren Peaks operational site. It has a yellowish, reddish, or greyish color at the surface and is greatly altered by decomposition and solution of some of its mineral constituents. Below the surface, progressively firmer, less altered rock is encountered. The weathered portion of the bedrock generally has a thickness of from two to three feet.

A thin mantle of soil, resulting from the decomposition of the parent igneous rock, covers the bedrock on Warren Peaks. It is usually a mixture of clay, quartz grains, mica and other materials. The thickness of this mantle is probably one foot or less on the more exposed and steeper slopes, and increases to five feet on the lesser slopes where greater deposition and less run-off occur.

5. Hydrology

Sub-surface water.- The central portion of Bear Lodge Mountains is probably not underlain by water-bearing formations. The outlook for obtaining water from underground sources in the immediate area (except from springs) is not good, although some possibilities do exist in neighboring valleys.

The water source for the operations area of the radar station will be from Hutchins Springs, which is about 3/4 mi northwest of Warren Peaks on the Lytle Creek water shed and has a flow of about 10 gpm. Table VIII-3 lists an analysis of the water from Hutchins Springs as prepared for the Omaha District of the Corps of Engineers.

TABLE VIII-3

Chemical Analysis of Water
Hutchins Springs, Warren Peaks, Wyoming
Date of Sample: 26 July 1958

Analysis	Parts Per Million
Suspended solids	0
Dissolved solids (residue at 103° C)	51
Alkalinity to phenolphthalein as $CaCO_3$	0
Alkalinity to methyl orange as $CaCO_3$ (total alkalinity)	4.3
Total hardness as $CaCO_3$	10.3
Calcium	1.4
Magnesium	1.7
Alkalies as sodium	3.3
Iron	0.1
Aluminum	0.0
Manganese	0.0
Sulfate	5
Chloride	1.4
Nitrate	0.3
Bicarbonate	5.5
Carbonate	0.0
Silica	15.4
Sodium	2
Potassium	1.3
pH	6.0

Another spring in the area, Reuter Spring, is about 2-1/2 mi south-southeast of Warren Peaks along the road from the Cantonment Area to the Operations Area. Other known springs in the immediate area are Davis and Cole Springs which are located about 5 mi north-northeast of Warren Peaks on Beaver Creek. These springs have a flow of 10 to 20 gpm.

Surface water.- The amount of flowing water in the Sundance area is not great; many of the draws, valleys and canyons are dry except when there is rain or melting snow. None of the streams has a continuous flow from head to mouth, and many appear at intervals in springs or bottom seeps which supply waters that flow for greater or lesser distances before evaporating or passing underground. Often the water in a valley consists of a series of pools.

Figure VIII-1 illustrates drainage and water sources for the PM-1 site area.

6. Seismology

There is no specific knowledge of earthquake conditions in this area. The Seismic Probability Map of the United States, contained in the Uniform Building Code issued by the Pacific Coast Building Officials Conference, designates the eastern third of Wyoming to be a "zero seismic zone." However, minor tremors have occurred in Cheyenne, also in the zero probability zone, every year or two. They have never been of damaging magnitude.

Design of the operational radar structures incorporated some seismic design criteria because of blasting and mining operations expected in the vicinity.