UNIVERSITY OF
ARIZONA LIBRARY
Documents Collection
NOV 7 1960

PM-1 NUCLEAR POWER PLANT PROGRAM
5TH QUARTERLY PROGRESS REPORT
MARCH 1, 1960 TO MAY 31, 1960

By
F. Hittman

July 5, 1960

Nuclear Division
Martin Company
Baltimore, Massachusetts

UNITED STATES ATOMIC ENERGY COMMISSION
Office of Technical Information

metadc303786

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

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

Printed in USA. Price \$3.00. Available from the Office of Technical Services, Department of Commerce, Washington 25, D. C.

MND-M-1816
PM-1 Nuclear Power Plant Program
5TH QUARTERLY PROGRESS REPORT
March 1, 1960 to May 31, 1960

Contract AT(30-1)-2345

5 July 1960

Prepared and Approved By:

F. Hittman
Project Manager

KEY TO PM-1 DRAWING

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

ABSTRACT

This report contains a description of the work accomplished during the fifth contract quarter (March 1, 1960 to May 31, 1960) of Contract AT (30-1)-2345 between The Martin Company and the USAEC.

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

The principal efforts during the fifth project quarter were the completion of the final design submittals and specifications for plant components.

Systems development work included structural testing of a full-scale test package, performance of a loading demonstration in a C-130A aircraft and completion of testing of the air-steam condenser model at Eglin Air Force Base.

Reactor development work included:

- (1) Initiation of the flexible zero-power test (PMZ-1) experimental program.
- (2) Fabrication and delivery of a PM-1 type fuel element for irradiation in the SM-1 and initiation of a loop irradiation test at the WTR.
- (3) Continuation of reactor flow and heat transfer tests.
- (4) Development work on rare earth control rods.
- (5) Testing and redesign of the prototype magnetic jack-type control rod actuator.

Core fabrication continued with the delivery of UO_2 ordered to date and the completion of the production runs of fuel elements for the zero-power test, PMZ-1.

For progress during the preceding period, see MND-M-1815.

FOREWORD

This is the fifth quarterly progress report submitted to the U. S. Atomic Energy Commission under Contract AT (30-1)-2345. It covers the work accomplished by The Martin Company on the PM-1 Project for the period from March 1, 1960 through May 31, 1960.

CONTENTS

	Page
Abstract	v
Foreword	vi
Contents	vii
Program Highlights	ix
Introduction	xi
PM-1 Nuclear Power Plant Design Summary	xii
A. Reactor Design Characteristics	xii
B. Systems Design	xv
C. Secondary System	xvii
D. Packaging	xxii
I. Task 1--Preliminary Design--System Development . .	I-1
A. Subtask 1.1--Package Development and Test	I-1
B. Subtask 1.2--Incore Instrumentation	I-2
C. Subtask 1.5--Instrumentation and Control Research and Development	I-9
D. Subtask 1.3--Shielding Measurements	I-9
E. Subtask 1.6--Secondary System Development . . .	I-10
II. Task 2--Preliminary Design--Reactor Development .	II-1
A. Subtask 2.1--Flexible Zero-Power Test	II-1
B. Subtask 2.2--Irradiation Test	II-59
C. Subtask 2.3--Reactor Flow Studies	II-59
D. Subtask 2.4--Heat Transfer Tests	II-78
E. Subtask 2.5--Actuator Program	II-96
III. Task 4--Final Design	III-1
IV. Task 5--Core Fabrication	IV-1
A. Subtask 5.1--Fabrication of the PMZ-1 Core . . .	IV-1
B. Subtask 5.2--Conversion of UF_6 to UO_2	IV-1
C. Subtask 5.3--Fuel Element Development	IV-1
D. Subtask 5.4--PM-1 Core Fabrication	IV-10
E. Subtask 5.5--Fuel Element Irradiation Program .	IV-10
V. Task 6--Fabrication of Dummy Core	V-1
A. Accomplishments this Quarter	V-1
B. Anticipated Accomplishments Next Quarter	V-1

CONTENTS (continued)

	Page
VI. Task 7--Fabrication and Assembly of Plant	VI-1
A. Controls and Instrumentation Work	
Accomplishments	VI-1
B. Primary System	VI-7
C. Secondary System	VI-11
VII. Task 8--Packaging	VII-1
A. Primary System	VII-1
B. Secondary System Packages	VII-1
VIII. Task 9	VIII-1
A. Subtask 9.4--Plant Rearrangement	VIII-1
IX. Task 11--Site Preparation and Installation	IX-1
A. Subtask 11.1--Site Preparation	IX-1
X. Task 12--Package Loading Demonstration	X-1
A. Loading Demonstration	X-1
XI. Task 13--Manuals	XI-1
A. Subtask 13.1--Maintenance Manual	XI-1
B. Subtask 13.2--Operating Manual	XI-1
C. Subtask 13.3--Training Manuals	XI-3
XII. Task 14--Training	XII-1
A. Subtask 14.1--Training Development	XII-1
XIII. Task 15--Project Services	XIII-1
A. Subtask 15.1--Film and Photographs	XIII-1
B. Subtask 15.2--Models	XIII-1
XIV. Task 16--Consulting	XIV-1
XV. Task 17--Reports	XV-1
A. Subtask 17.1--PM-1 Hazards Summary	
Request	XV-1
B. Subtask 17.2--Reports Other than Hazards	XV-1

PROGRAM HIGHLIGHTS

1. The package development and test program was completed and the loading demonstration of the test package, followed by a test flight in a C-130A aircraft, was successfully accomplished (Subtask 1.1 and Task 12).
2. The air-steam condenser tests were completed at the Eglin Air Force Base climatic chamber and the condenser model was shipped to The Martin Company for post-test inspection (Subtask 1.6).
3. The PMZ-1 zero power test program was initiated. Criticality was achieved March 17, 1960 (Subtask 2.1).
4. The major portion of the bundle orifice tests was completed. Component fabrication for the full-scale flow tests was initiated (Subtask 2.3).
5. The heat transfer test program continued satisfactorily. Test sections were run successfully under local boiling conditions and heat fluxes in excess of those anticipated in the PM-1 (Subtask 2.5).
6. The final design subsystem submissions for the PM-1 nuclear power plant were completed and delivered to the AEC (Task 4.0).
7. Fabrication of the PM-1-M fuel element for irradiation in the SM-1 was completed (Subtask 5.5).
8. Equipment vendor selection, plant fabrication and engineering liaison were continued satisfactorily (Task 7.0).
9. The PM-1 site foundation design was completed (Subtask 11.1).

INTRODUCTION

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

During the fifth quarter, Tasks 1, 2, 4, 5, 6, 7, 8, 9, 11, 12, 13, 14, 15, 16 and 17 were active. For convenience, work on Task 6 (dummy core) is reported with Subtask 2.3 (reactor flow studies). Also, all technical discussion regarding plant design and procurement of components is reported under Task 7 (fabrication and assembly of plant). Tasks 8 (packaging) and 9 (preshipment test) became active during this project quarter.

PM-1 NUCLEAR POWER PLANT DESIGN SUMMARY

A. REACTOR DESIGN CHARACTERISTICS

1. Overall Performance Data

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

2. Core Design Characteristics

Geometry, right circular cylinder (approximately)

Diameter, average (in.)	23.6	←
Active length (in.)	30	
Overall length of fuel tube (in.)	33-1/4	
Core structural material	Modified ASTM 304 and 347	←
Fuel element data, tubular, cermet type		
Outside diameter (in.)	0.500	
Inside diameter (in.)	0.416	

← Denotes change in plant parameter from previous submission.

Clad thickness (in.)	0.006	
Clad material	AISI Type 347 stainless steel, modified, 0.01 wt % Co maximum, 0.03 wt % Co plus Ta maximum	←
U-235 loading/tube, nominal, (gm)	39.4 ± 4%	
Averaged over core	39.4 ± 2%	
Number	732	
Meat composition, nominal (wt % UO ₂)	28	
Burnable poison element data, unclad, cylindrical, boron stainless steel, al- loy type		
Outside diameter (in.) varies to compensate for actual boron loadings obtained	0.500/0.475	
Basic poison material	ASTM Type 304 stainless steel, 0.01 wt % Co maximum, 0.03 wt % Co plus Ta maximum	
Boron loading (natural) in grams of B ¹⁰ per rod in stainless steel alloy	0.640 ± 2%	
Number	75	
Control element data, Y-shaped, cermet type		
Arm length--total overall from pickup ball center- line (in.)	38-3/8	
--active (in.)	30	

← Denotes change in plant parameter from previous submission.

Arm width--total (in.)	3-7/8
--active poison (in.)	3-1/2
Arm thickness (in.)	5/16
Clad thickness (in.)	0.030
Clad material	AISI Type 347 stainless steel, modified, 0.05 wt % Co maxi- mum, 0.15 wt % Co plus Ta maxi- mum
Poison element	Europium com- pound dispersed in stainless steel (equivalent to 30 wt % Eu_2O_3)
Number	6
*Nuclear worth of six control rods ($\% \Delta \rho$)	-31.8
*Nuclear worth of five control rods ($\% \Delta \rho$)	-16.5
*Nuclear worth of four rods, minimum worth ($\% \Delta \rho$)	-9.3
*Minimum shutdown control margin, approximately midlife, two of six rods stuck in operating condition ($\% \Delta \rho$)	0.4
*Average thermal core flux	
Initial ($n\phi$)	0.7×10^{13}
At 2 years ($n\phi$)	1.4×10^{13}
*Average temperature coefficients	
Overall, 65° to 463° F ($\Delta \rho$ /° F)	-1.2×10^{-4} (initial); -0.9×10^{-4} (at 2 years)

*Denotes new items added to plant parameters.

Operating temperature ($\Delta\rho$ /° F)	-2.1 x 10 ⁻⁴ (initial); -1.9 x 10 ⁻⁴ (at 2 years)
--	--

3. Core Heat Transfer Characteristics

Heat flux (Btu/ft²-hr)

Average	73,000
*Heat flux, maximum	290,000
Average coolant temperature (° F)	463

4. Reactor Hydraulic Characteristics

Coolant flow rate (gpm)	2125
-------------------------	------

B. SYSTEMS DESIGN

1. General Plant

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

2. Main Coolant System

Number of coolant loops	1
Coolant flow rate (gpm)	2125
Coolant system design pressure (psig)	1485

*Denotes new items added to plant parameters.

Coolant velocity in piping (main loop) (fps)	26
Coolant pipe size, main loop, nominal (in.)	
Schedule 80	6
System basic material	
Reactor pressure vessel	AISI 347
Piping	AISI 316
Remainder	AISI 304
Main coolant pumps	
Pumps, number (canned rotor type)	1
Steam generator	
Number of units	1
Design pressure, shell side (ap- proximately) (psi)	600
Type	Vertical with inte- gral steam drum and separators
Temperature, primary inlet, full power, approximately (° F)	479
Temperature, primary outlet, full power, approximately (° F)	447
Temperature, steam side outlet, full power (° F)	417
Access	Shell and tube side bolted
Tube material	Inconel

*Denotes new items added to plant parameters.

3. Pressurizing and Pressure Relief System

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

4. Coolant Purification and Sampling System

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

5. Primary Shield Water System

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

C. SECONDARY SYSTEM

1. General Plant

Steam flow, full power (lb/hr)	34,312
Steam flow, turbine, full power, straight condensing (lb/hr)	26,253

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

2. Turbine-Generator Set

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

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

3. Condenser System

Number of units (each 1/4 capacity)	4	←
*Number of units per C-130A flight	2	
Type	Direct air to steam	
Duty--heat rejected, full load, per unit (Btu/hr)	5.05×10^6	←
Design heat load per unit (Btu/hr)	5.2×10^6	←
Tubes	Horizontal, finned aluminum	
*Number of fans per unit	2	

4. Feedwater System

Deaerator	
Type	Atomizing
Feedwater design flow (lb/hr)	37,912

*Denotes new items added to plant parameters.

← Denotes change in plant parameter from previous submission.

Design pressure (psia)	50
Oxygen removal guarantee (cc/ liter remaining)	0.005
Storage (min)	5
Boiler feed pumps	
Number	2
Drivers	One steam driven, one electrical driven
Type	Vertical, centrifugal
Closed feedwater heaters	
Number	1
Type	Tube and shell, horizontal
*Number of passes, tube side	6
*Number of passes, shell side	1
*Tube material	Admiralty metal
*Design pressure, shell side (psig)	110
*Design pressure, tube side (psig)	585
*Heat transfer surface (ft ²)	220

5. Auxiliaries

Evaporator-reboiler

Capacity (lb/hr of 35 psia steam)	7500
Design pressure, shell side (psia)	65

*Denotes new items added to plant parameters.

*Design pressure, tube side (psia)	165
Makeup water temperature, minimum (° F)	40
Condensate return temperature (° F)	172
*Coil steam, operating pressure (psia)	115
Feedwater storage tank	
Capacity, approximately (gal)	1950
Turbine steam bypass system	
Type	Manual with de-superheater station
*Maximum bypass flow	5% of full load
*Desuperheated steam, maximum temperature (° F)	220
Auxiliary generator unit	
Type	High-speed diesel
Number	1
*Capacity (kw, at 6500-ft elevation)	150
Electrical characteristics	480 volts, 60 cps, 3 phase
Emergency power	
D-C power source	Batteries
Capacity at 8-hr discharge rate (amp-hr)	160
A-C power source	2-unit MG set

*Denotes new items added to plant parameters.

D. PACKAGING

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

	<u>Uncontained</u>	<u>Contained</u>
Primary loop packages including waste disposal system	6	8 ←
Secondary loop packages	9	9 ←
Decontamination package	1	1

← Denotes change in plant parameter from previous submission.

Fig. 1. Flow Diagram of Primary System

Fig. 2. Flow Diagram of Secondary System

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

Project Engineers--Subtask 1.3, R. Akin;
1.2, G. Zindler;
1.1, C. Fox

A. SUBTASK 1.1--PACKAGE DEVELOPMENT AND TEST

(A. Layman, R. Dugas)

During the fifth quarter, the planned objectives were to complete all design element tests, fabrication of test fixtures, hoisting tests, limit load impact tests, limit and ultimate snow load tests, the vibration survey, limit and ultimate drop tests, ultimate vertical inertial test and ultimate load impact tests. In addition, the loading demonstration was to be performed (see Task 12), using the test package for a portion of the quarter.

All of the major objectives pertaining to the development and testing of the PM-1 Test Package were completed during the quarter with the exception of the drop tests and ultimate impact and inertia tests which were deleted from the program for budgetary reasons. With respect to the impact and inertia tests, the limit load data were adequate for substantiation of design by extrapolation to ultimate load levels.

The element testing of the various tie-down fittings was successfully completed. The following is a partial list of those fittings with pertinent test data.

<u>Fitting</u>	<u>Yield Point (% DLL)</u>	<u>Ultimate Failure (% DUL)</u>	<u>Date Completed</u>
Hoist	120	122	2/22/60
"Merry-go-round"			
drag brace	110	98 ▲	2/23/60
Hoist	140	132	2/26/60
32° swivel	130	106	2/24/60
45° swivel	150	176	3/15/60
25 kip tie-down	150	166	3/16/60
Hoist base	105	100	3/28/60
U-bolt tie-down	100	135*	3/28/60
	100	130*	3/28/60

DLL--Design Limit Load

DUL--Design Ultimate Load

▲ Failure occurred in shearing of 5/16-inch retaining bolts. Changed to 3/8-inch bolts.

* Failure occurred in a standard aircraft tie-down fitting which connects to the U-bolt fitting.

The 2 g limit hoisting test and the package ultimate snow load test (45 psf) were completed, and both of these tests yielded strength and stiffness results which were very satisfactory and in close agreement with calculated data.

The longitudinal impact test was successfully completed up to 3 g input (limit design condition). Visual inspection revealed no evidence of yielded structure. Acceleration data is now being evaluated. The lateral impact test was successfully completed up to 1-1/2 g ultimate input on 4/8/60. Visual inspection revealed no evidence of yielded structure, and the test data is being evaluated at this time. At the 1-1/2 g lateral input, there was some yielding of the standard aircraft "D" ring hold-down straps on the test pallet. These straps are part of the C-130 aircraft cargo tie-down provisions. This yielding of the straps was expected at this impact load level and demonstrates that the package surpasses the aircraft with respect to lateral load capability.

The vibration survey of the test package was completed during the quarter. The total duration of the survey was two weeks. Vibration force inputs of 50 pounds at a frequency range of 5 to 350 cps were applied to the package through the test pallet, along the three principal axes, and transmissibility responses were recorded at the center of gravity of each major piece of equipment within the package.

Figures I-1 through I-6 illustrate and explain significant tests and structures.

Objectives for the coming quarter are to complete the test program report and to close out the subtask.

B. SUBTASK 1.2--INCORE INSTRUMENTATION

(G. F. Zindler)

1. Work Accomplished

The classified technical memorandum reporting this work was in the process of printing and distribution. The efforts under this subtask are now completed.

2. Anticipated Accomplishments Next Quarter

All efforts under the Incore Instrumentation Program have been completed and no further efforts are planned under the present PM-1 Program.

Fig. I-1. Lateral Impact Test Setup

Fig. I-2. Impact Test, Cable Arresting Gear

Fig. I-3. Impact Test, Package Suspension and Tie-Down

Fig. I-4. Element Test--Jack and Hoist Fitting

Fig. I-5. Snow Load Test to 45 lb/ft^2

Fig. I-6. Completed Test Package Being Hoisted by Crane

C. SUBTASK 1.5--INSTRUMENTATION AND CONTROL RESEARCH AND DEVELOPMENT

(G. Zindler)

1. Work Accomplished

A topical report was mailed during this quarter covering the R&D efforts under this subtask. The report is MND-M-1914, entitled "PM-1 Nuclear Power Plant Program Controls and Instrumentation Report," dated February 1960. This R&D study was carried out under a subcontract with the Stromberg-Carlson Corporation. The results of the study program have been incorporated into the final design drawings and specifications of the PM-1 Controls System.

2. Anticipated Accomplishments Next Quarter

All efforts under this subtask were completed during the quarter. No further efforts are planned.

D. SUBTASK 1.3-SHIELDING MEASUREMENTS

(R. J. Akin, D. Owings, E. Koprowski)

During this quarter, a detailed evaluation of the activation analysis was made. The results of this evaluation have been collected in a final report.

During the next quarter, the final report will be issued as MND-M-1917 and the subtask will be terminated.

The following conclusions were reached, based upon a detailed evaluation of the activation analysis from MITR and the analytical work done at The Martin Company:

- (1) The preliminary conclusions reported in MND-M-1815 were verified.
- (2) Relatively good correlation was attained between the experimental and analytical studies.
- (3) No foreseeable hazards from the activated earth are expected.

E. SUBTASK 1.6--SECONDARY SYSTEM DEVELOPMENT

(C. Fox, L. Hassel)

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

Planned accomplishments during this period were:

- (1) Completion of the condenser test at Eglin AFB, Florida.
- (2) Completion of the final condenser test report.
- (3) Completion of 90% of the condenser design.

During this period, the following work was actually accomplished:

- (1) The condenser test was completed and the condenser model was shipped to The Martin Company, Baltimore, for post-test inspection.
- (2) The condenser test report was completed and is currently being reproduced for distribution to the AEC.
- (3) The certified design of the switchgear was reviewed and returned to the vendor for incorporation of comments.
- (4) The motor control center design was held for the condenser fan motor selection.
- (5) Design of the condensers is approximately 30% complete. This item is behind schedule as a result of a two-week strike at the vendor's plant.

The following progress is anticipated in the next quarter:

- (1) The final design of the condenser will be completed.
- (2) The final design of the switchgear will be completed.
- (3) The final design of the motor control center will be completed.

II. TASK 2--PRELIMINARY DESIGN--

REACTOR DEVELOPMENT

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

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

A. SUBTASK 2.1--FLEXIBLE ZERO-POWER TEST (H. B. Rosenthal, E. A. Scicchitano)

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

The work planned for this project quarter included:

- (1) Completion of installation and checkout of zero-power test hardware.
- (2) Initiation of critical experiments on the design PM-1 core.
- (3) Completion of pre-experiment analysis of all major cores to be studied.

The experimental work accomplished during this quarter included:

- (1) Completion of system installation and checkout.
- (2) Fabrication of all components required for most of the experimental program.
- (3) Amendment of Facility License CX-7 by the AEC on March 8, 1960, authorizing performance of the PMZ-1 program.
- (4) Determination of cold, clean PMZ-1 critical mass.
- (5) Determination of cold, poisoned PMZ-1 critical mass.
- (6) Determination of critical 6-Y-rod bank position in Core 27.
- (7) Reactivity evaluation of Core 27.

- (8) Evaluation of 6-Y-rod bank in Core 27.
- (9) Determination of reactivity difference between Cores 27 and 0.
- (10) Reactivity evaluation of Core 52.
- (11) Temperature coefficient for Core 52 with Y-rods fully withdrawn.
- (12) Flux, power and gamma mapping of Core 52 with Y-rods fully withdrawn.
- (13) Subcritical evaluation of 3-, 4-, 5- and 6-Y-rod banks in Core 52.

The analytical work accomplished during this quarter included:

- (1) Determination of reactivity as a function of stable reactor period.
- (2) Critical configuration studies.
- (3) Total core reactivity studies for:
 - (a) PM-1 design core.
 - (b) Core 27--PM-1 design core with PMZ-1 lumped burnable poison rods No. 1 (0.275 wt % natural boron) replacing the design rods.
 - (c) Core 52--PM-1 design core with PMZ-1 lumped burnable poison rods No. 2 (0.525 wt % natural boron) replacing the design rods.
 - (d) Core 84--PM-1 design core with PMZ-1 lumped burnable poison rods No. 3 (0.845 wt % natural boron) replacing the design rods.
 - (e) Core 0--PM-1 design core with stainless steel rods replacing the design rods.
 - (f) Core FE--PM-1 design core with fuel elements replacing the design rods.
- (4) Relative worth versus insertion for 3- and 6-rod banks.

- (5) Critical rod bank position studies for the cores in (3) above.
- (6) Temperature coefficient studies for the critical configuration and full-size cores.
- (7) Power density distribution studies for various positions of the 6-rod bank.

Post-experiment analysis and evaluation of the completed experiments were initiated.

During the next quarter, it is planned that experimental studies will be performed to determine:

- (1) Total reactivities of Cores 0, 52, 84 and FE.
- (2) Critical 6-Y-rod bank position for Cores 0, 52, 84 and FE.
- (3) Temperature coefficients for Cores 0, 52, 84 and FE.
- (4) Power map of Core 52 with critical 6-Y-rod bank.
- (5) Evaluation of 3-, 4-, 5- and 6-Y-rod banks in Cores 52 and FE.
- (6) Effect of lumped poisons on Y-rod worth.
- (7) "Stuck rod" evaluation.

During the next quarter, post-experiment analytical studies will be completed on all cores experimentally studied during this quarter.

1. General

The order of presentation of the work accomplished under Subtask 2.2 is as follows:

- (1) All experimental work done during the quarter.
- (2) All analytical work, including pre- and post-experiment analysis.

During the current and next quarters, six different cores will be studied. For convenience, the following nomenclature has been adopted.

- (1) Critical configuration--Those cores containing only fuel elements (or fuel elements and lumped poisons) in all locations, including those normally occupied by Y-rods and Y-rod guides.
- (2) Core 0--That core containing 732 fuel elements and 75 stainless steel lumped poison rods which have no boron.
- (3) Core 27--That core containing 732 fuel elements and 75 lumped poison rods which have 0.27% boron by weight.
- (4) Core 52--That core containing 732 fuel elements and 75 lumped poison rods which have 0.52% boron by weight.
- (5) Core 84--That core containing 732 fuel elements and 75 lumped poison rods which have 0.84% boron by weight.
- (6) Core FE--That core containing 807 fuel elements and no lumped poison rods.

2. Flexible Zero Power Test--Experimental Studies

Critical configuration studies. A minimum-size core was assembled with fuel tubes in all types of locations (Y-rod slots, lumped poison and stainless steel positions). The loading was accomplished by standard multiplication techniques requiring 5 fuel loadings totaling 127 fuel tubes with 5011 grams of U-235 to achieve criticality. Figure II-1 shows the loading steps and weights. Figure II-2 is a photograph of this core. This core was found to have a positive temperature coefficient, at least between 10° to 20° C, of approximately $+4 \times 10^{-5} \Delta K/K/^{\circ}C$.

Several configurations of lumped poison were studied for data to check cell correction calculations. Figure II-3 shows the first distribution which has 19 lumped poison rods containing 0.27 wt % boron and required 174 fuel tubes (6875.9 grams of U-235) to achieve criticality with a small excess. (Multiplication curves indicate approximately 173 tubes for criticality with no excess.) The final distribution (Fig. II-3) shows that 12 of the 19 boron steel rods are located at the periphery of the core--not the most desirable distribution for an analytical check. Therefore, a new distribution, which is essentially a shift to the right of one tube, was studied. The final distribution contained 17 boron steel rods and indicated that 172 fuel tubes would be required for criticality (see Fig. II-4). Still another distribution, shown in Fig. II-5, requires 165 tubes (6520.55 grams of U-235) to attain criticality with 14 lumped poison rods. The last configuration, shown in Fig. II-6, contained 13 lumped poison rods and 156 fuel tubes (6163.45 grams of U-235) but was not critical.

<u>Loading</u>	<u>Tubes (No.)</u>	<u>Total Tubes (No.)</u>	<u>Weight U-235 (gm)</u>	<u>Total Weight U-235 (gm)</u>
1	31	31	1227.08	1227.08
2	31	62	1227.22	2454.30
3	31	93	1218.25	3672.55
4	18	111	705.89	4378.44
5	16	127	632.64	5011.08

Mean diameter, core OD - 8.08 in.

Active height - 30.00 in.

Fig. II-1. PMZ-1 Loading Steps and Weights

Fig. II-2. Critical PMZ-1 Core Configuration

Tubes
(No.)

174

Total Weight
(gm)

6875.9

0.27 wt % Boron Tubes
(No.)

19

Fig. II-3. PMZ-1 Loading Steps and Weights

Tubes
(No.)
172

Total Weight
(gm)
6797.0

0.27 wt % Boron Tubes
(No.)
17

Fig. II-4. PMZ-1 Loading Steps and Weights

Tubes
(No.)

165

Total Weight
(gm)

6520.55

0.27 wt % Boron Tubes
(No.)

14

Fig. II-5. PMZ-1 Loading Steps and Weights

<u>Tubes (No.)</u>	<u>Total Weight (gm)</u>	<u>0.27 wt % Boron Tubes (No.)</u>
156	6163.45	13

Fig. II-6. PMZ-1 Loading Steps and Weights

Critical 6-Y-rod bank position in Core 27. Core 27 (see Fig. II-7) was built up in 7 steps employing multiplication data obtained on water fill, Y-rod and CE rod in and out. Figure II-7a shows the loading steps taken, while Fig. II-8 shows the 6-Y-rod bank positions as functions of effective core diameter. The critical 6-Y-rod bank position was 7.42 inches withdrawn.

Table II-1 gives the pertinent data for the fully loaded core.

TABLE II-1
Core 27 Specifications

Number of fuel tubes	732
Weight of U-235*	28,794.7 gm
Number of 0.27 wt % lumped poison rods	75
Number of dummy fuel tubes	19
Number of stainless steel rods	3
6-Y-rod bank position at criticality	7.42 in. withdrawn

*Actual fuel tube U-235 weights as supplied by accountability.

Total reactivity and 6-Y-rod bank evaluation in Core 27. The total reactivity of Core 27 was obtained by incremental insertion of 0.385-inch x 0.0625-inch x 36.75-inch boron-in-polyethylene strips containing 5.09 wt % natural boron. At each incremental insertion, a reactivity evaluation was made of 10 strips from the same locations. Figure II-9 shows the results of these evaluations at various loadings of boron plastic. Applying the reactivity worths per strip to the total number of strips added in that increment yields the reactivity change made. Figure II-10 shows the accumulated total reactivity versus number of strips. Figure II-11 shows the 6-rod bank position as a function of boron strip loading. Cross plotting of Figs. II-10 and II-11 will yield Fig. II-12 (bank position versus reactivity). The results of these measurements indicate the excess reactivity to be $13.2 \pm 0.5\%$ $\Delta K/K$, of which 12.85% is controlled by the boron plastic and 0.350% by the CE rods.

- ⊙ Dummy fuel element
- 0.27 wt % boron steel
- ◐ Stainless steel
- Fuel tubes

Fig. II-7. Core 27 Configuration

<u>Loading</u>	<u>Tubes (No.)</u>	<u>Total Tubes (No.)</u>	<u>Weight U-235 (gm)</u>	<u>Total U-235 (gm)</u>
1	105	105	4148.55	4148.55
2	99	204	3916.52	8065.07
3	156	360	6096.60	14161.67
4	60	420	2378.68	16540.35
5	90	510	3540.33	20080.68
6	108	618	4265.87	24346.55
7	114	732	4448.38	28794.73

Fig. II-8. Effective Core Radius as a Function of 6-Rod Bank Position for Core 27

Fig. II-9. Boron Evaluation Reactivity per Strip as a Function of Core Loading for Core 27

Fig. II-10. Reactivity Versus Boron Strip Loading--Core 27

Fig. II-11. Six-Rod Bank Position Versus Number of Boron Strips--Core 27

Fig. II-12. Reactivity Versus Six-Rod Bank Position--Core 27

A cross check of this method was made at the fully poisoned condition by making radial evaluations of the boron strips. The resulting reactivity was determined to be $13.2 \pm 0.5\% \Delta K/K$, of which 12.81% was controlled by the boron plastic strips. The two independent methods check out within experimental error. Table II-2 summarizes these results.

TABLE II-2
Integrated Boron Evaluation--Core 27

<u>Ring No.</u>	<u>% $\Delta K/K$/strip</u>	<u>Number of Strips in Ring</u>	<u>% $\Delta K/K$/ring</u>
0	0.0339	11	0.373
1	0.0445	23	1.024
2	0.0388	28	1.086
3	0.0380	24	0.912
4	0.0287	51	1.464
5	0.0276	93	2.567
6	0.0209	90	1.881
7	0.0149	162	2.414
8	0.0118	<u>92</u>	<u>1.086</u>
Total excess controlled by boron plastic		574	12.807
Excess controlled by CE rods			<u>0.350</u>
Total excess reactivity			13.157 % $\Delta K/K$

Figure II-13 shows the location of the boron plastic strips, while Table II-3 gives the pertinent boron inventory.

- Stainless steel dummy fuel tubes
- ▨ Stainless steel rods
- 0.27 wt % boron steel
- ⊗ Fuel tubes not containing boron strips
- Fuel tubes that contain boron strips

Fig. II-13. Boron Strip Locations--Core 27

TABLE II-3

Distributed Boron Inventory for Core 27

Number of 5.09 wt % boron plastic strips	574
Total weight (36.75 in.)	6792 gm
Active core weight (30.0 in.)	5545 gm
Weight boron active core	282.2 gm
Reactivity controlled by 574 strips	12% $\Delta K/K$
Reactivity in CE rods	0.35% $\Delta K/K$

Facility power calibration. The facility power calibration experiment was performed to calibrate the reactor instrumentation and to determine the radiation levels around the facility as a function of reactor power. The experiment was carried out with one reactor run of 40 minutes. The experiment consisted of three parts: (1) a measurement of the relative power density distribution throughout the reactor core, (2) a comparison of the power density at one particular point in the core with a known standard and (3) a health physics survey of the building to check gamma radiation levels.

The first part was accomplished by picking nine fuel tubes which were uniform in U-235 weight within $\pm 1\%$ and which had never been irradiated at high power levels. These were inserted in the core along a rough radial line in Section I (see Fig. II-7) from near the core center to the core periphery. After the irradiation for 40 minutes at a uniform power level, the tubes were removed from the core and the amount of fission product activity (which is proportional to the power density) in each was measured as a function of axial position with a set of scintillation detectors. Preset count techniques were used to eliminate any necessity for decay time corrections. Graphical integration of the measured axial distributions was carried out to determine the average power density in each fuel tube measured. These data were plotted as a function of distance of the tube from the centerline of the core to enable the power density in the other fuel tubes in the core to be estimated. The average power density in the whole core was then computed by use of this graph and compared to the measured power density at that point in the core where the absolute power level had been determined by the second part of the experiment.

In the above-mentioned second part of the experiment, the absolute power level at one point in the core was determined by irradiating a uranium foil in the core at that point and comparing the activity induced in it with that induced in an identical foil irradiated in the known thermal neutron flux of a paraffin sigma pile for the same time interval. The absolute power density in watts per gram of U-235 generated in the foil in the sigma pile was computed from the U-235 thermal cross section and the thermal neutron flux. Comparing the observed count rates in the foil from the sigma pile and the foil from the core gave the absolute power density at that one point in the reactor core and, hence, by means of the calculations mentioned in the previous paragraph, the average power density in the whole core. This number multiplied by the total U-235 weight in the core gave the total core power. The result obtained for the reactor run was a power level of 1.9 watts. Comparison with the readings of the reactor console instruments indicates that sufficient range is available on them with the neutron and gamma sensing detectors at their present locations for any foreseeable measurements.

The health physics survey part of the experiment showed no measurable amount of gamma radiation in the control room or work room and a maximum of 3.5 mr/hr on the outside of the building at a power level of 1.9 watts.

Reactivity difference between Core 27 and Core 0. The determination of the reactivity difference between Core 27 and Core 0 was accomplished by replacing the 75 lumped poison rods of Core 27 with stainless steel rods and compensating for the increased reactivity by inserting boron plastic strips. Core 27 contained 574 boron plastic strips (5.09 wt % boron), compensating for the removal of the 6 Y-rods with an excess of 0.35% $\Delta K/K$ controlled by the CE rods.

Due to the presence of the existing boron plastic loading, only 74 locations were available for additional strips. An additional 290 strips were required to remove the Y-rods and were added to existing locations, making double strips. At each boron plastic insertion, a reactivity evaluation was taken at 10 sample locations. Figure II-14 shows the 5-rod bank position as a function of the number of boron strips added to the system beyond what was initially there at the end of the Core 27 reactivity experiment. Figure II-15 shows the accumulated total reactivity as a function of boron strip loading, indicating a total reactivity change of $6.3 \pm 0.7\% \Delta K/K$. Figure II-16, which is derived from Figs. II-14 and II-15, shows the 5-rod bank versus reactivity. Table II-4 shows the boron plastic inventories for the stainless steel core with the Y-rods fully withdrawn. All other materials are the same as reported in Table II-1.

Fig. II-14. Five-Rod Bank Position Versus Boron Strip Loading--Core Zero

Fig. II-15. Reactivity Versus Boron Strip Loading--Core Zero

Fig. II-16. Five-Rod Bank Position Versus Reactivity--Core Zero

TABLE II-4
Distributed Boron Inventory for Core 0

<u>Inventory of Strips Added Beyond Experiment 3</u>	
Number of 5.09 wt % boron strips added	364
Total weight of strips added (gm)	4,125
Total weight of strips (active core--30 in.) (gm)	3,516
Total weight of boron (active core--30 in.) (gm)	179
<u>Total Boron Strip Inventory</u>	
Number of 5.09 wt % boron strips	938
Total weight of strips (gm)	10,773
Total weight of strips (active core--30 in.) (gm)	9,061
Total weight of boron (active core--30 in.) (gm)	461

Total reactivity of Core 52. To measure the total reactivity of Core 52, the 75 stainless steel rods from Core 0 were replaced with 75 lumped poison rods containing 0.52 wt % boron. Criticality was obtained by boron strip removal. Table II-5 gives the pertinent data for this core with the Y-rods fully withdrawn.

TABLE II-5
Core 52 Specifications

Number of fuel tubes	732
Weight of U-235 (gm)	28,794.95
Number of 0.52 wt % lumped poison rods	75
Number of stainless steel rods	3

Number of dummy fuel tubes	43
Number of 5.09 wt % boron strips	450
Total weight of strips	5100
Total weight of strips (active core--30 in.) (gm)	4347
Total weight of boron (active core--30 in.) (gm)	221
Reactivity controlled by CE rods	0.53% $\Delta K/K$

A radial evaluation of the boron strips, similar to that reported previously, was performed, indicating a total excess of $\pm 0.5\% \Delta K/K$. The results are tabulated in Table II-6.

TABLE II-6
Integrated Boron Evaluation--Core 52

<u>Ring No.</u>	<u>% $\Delta K/K$/strip</u>	<u>Number of Strips in Ring</u>	<u>% $\Delta K/K$/ring</u>
0	0.0309	9	0.278
1	0.0357	18	0.643
2	0.0350	22	0.770
3	0.0340	19	0.646
4	0.0257	40	1.028
5	0.0236	72	1.699
6	0.0176	72	1.267
7	0.0140	126	1.764
8	0.0101	72	0.727
Total excess controlled by boron plastic		450	8.822
Excess controlled by CE rods			0.53
Total excess reactivity			9.352% $\Delta K/K$

Figure II-17 is a plot of the radial strip evaluations made in Cores 27 and 52. The rings do not represent true radial distances but are presented this way to show the consistency of data obtained by this method. The evaluations made near the center of the core (Rings 0 and 1) are subject to large statistical error since these regions contain so few pieces and are in a region of high reactivity worth.

A measurement was made of the change in reactivity when 24 dummy fuel tubes were added at the core periphery in the locations required for the PM-1 design as shown in Fig. II-18. The results of this addition show a 0.09% $\Delta K/K$ decrease in core reactivity.

Figure II-19 is a photograph of Core 52, the PMZ-1 design core.

Temperature coefficient for Core 52 with Y-rods fully withdrawn.
The temperature coefficient of reactivity for Core 52 with Y-rods fully withdrawn was measured, using both a heating and a cooling cycle. The constituents of the core were those specified in Table II-5.

The heating curve was run by heating the water in the reactor tank with steam heaters and compensating for the decrease in reactivity by withdrawing a calibrated CE control rod. Temperature plateaus were established approximately every 5 degrees from 21° to 70° C and a criticality was established at each temperature equilibrium. The difference in the calibrated CE control rod position (based on reactivity evaluation during the cooling curve as described below) at criticality for each of the temperature equilibriums is the measure of reactivity change due to temperature change (see Fig. II-20).

The cooling curve was run by cooling the water in the reactor tank with cold water running through the steam heaters. The water was previously heated in the outside storage tank to above 80° C. As the water in the reactor tank cooled, temperature equilibriums were established approximately every five degrees and criticalities were determined by inserting the calibrated CE control rod. Reactivities were evaluated between various criticality positions and were used as a rod calibration. This reflects the change in rod worth as a function of temperature. The cooling curve (based on this rod calibration) was normalized to the heating curve at 21° C by increasing all reactivity values of the cooling curve by 0.028% $\Delta K/K$.

Temperature coefficients were obtained by differentiating both the heating and cooling curves and plotting reactivity as a function of temperature (see Fig. II-21). Both curves indicate a continuous negative temperature coefficient from 21° to 70° C. The temperature coefficients are summarized in Table II-7.

Fig. II-17. Reactivity per Strip Versus Ring Number (0.51 wt % lumped poison)

● Stainless steel fuel tubes (dummy)

Fig. II-18. Dummy Tube Location--Core-Reflector Interface

Fig. II-19. Core 52--PM Design Core

Fig. II-20. Core 52 Reactivity Versus Temperature

Fig. II-21. Core 452 Temperature Coefficient Curves

TABLE II-7
Temperature Coefficient of Reactivity for Core 52

<u>Temperature (°C)</u>	<u>Curve</u>	<u>Temperature Coefficient (% $\Delta K/K/^\circ C$)</u>
21	Heating	-0.001
	Cooling	-0.002
70	Heating	-0.013
	Cooling	-0.013

The temperature coefficient was shown by both the heating and cooling curves to be negative from 21° to 70°C. The total reactivity difference between 21° and 70° C was -0.320% $\Delta K/K$ by heating and -0.343% $\Delta K/K$ by cooling.

Evaluation of the temperature coefficient required calibration of the CE control system. One of the CE control rods was calibrated from full in to full out at 21° C and again partially at 70° C with good agreement (see Fig. II-22). The rod calibration taken by period evaluation as the system was being cooled reflects the change in rod worth as a function of temperature. The curve was normalized to the rod calibration at 21° C by taking the values at 1.66 inches to be the same and subtracting cumulative reactivities. A reactivity difference of 0.06% at 16.48 inches was noted between the two curves. One source of error could be the presence of the boron strips in the core.

Subcritical evaluation of 3-, 4-, 5- and 6-Y-rod banks in Core 52.
An attempt was made to obtain reactivity worth versus insertion curves for 3-, 4-, 5- and 6-Y-rod banks in Core 52 by subcritical techniques. The data demonstrated that this technique does not lend itself to dealing with as large reactivities or asymmetrical configurations as were being studied.

No attempt was made to refine the technique since the banks will be evaluated in future experiments by critical techniques.

Flux, power and gamma mapping of Core 52 with Y-rod fully withdrawn. Measurements in this experiment included:

Thermal flux

Epicadmium flux

Fast (above 2 mev) flux

Fig. II-22. Core 52 Rod Calibrations

Power density distribution

Gamma dosage

Percent of fissions below cadmium cutoff.

The measurements were completed during this quarter. During the next quarter, the data will be reduced and reported.

3. Flexible Zero-Power Test Analytical Studies

Reactivity versus stable reactor period. Reactivity versus stable reactor period was calculated for stable reactor periods from 1 to 500 seconds. The relationship between reactivity and asymptotic period, shown graphically in Fig. II-23, is given by the equation:¹

$$\rho = \frac{\ell}{T K_{\text{eff}}} + \sum_{i=1}^m \frac{\beta_i}{1 + \lambda_i T} = \frac{\ell}{\ell \omega + 1} + \frac{1}{\ell \omega + 1} \sum_{i=1}^m \frac{\omega \beta_i}{\omega + \lambda_i}$$

where

$$\rho = \frac{K_{\text{eff}}^{-1}}{K_{\text{eff}}}$$

ℓ = mean prompt neutron lifetime

T = stable reactor period

β_i = delayed neutron fraction per i^{th} delay group

λ_i = decay constant per i^{th} delay group

m = number of delay groups (six)

$\omega = 1/T$.

The values for the delay constants and yields of delayed neutron precursors for thermal fission of U-235 are given in Table II-8. The delayed neutron fraction for thermal fissions is equal to 0.0064 ± 0.0003 .

Fig. II-23. Reactivity Versus Stable Reactor Period

TABLE II-8
Delayed Neutron Yield from Thermal Fission in U-235²

Delayed Neutron Group Index	Decay Constant λ_i	Delayed Neutron Fission Per Group β_i
1	0.0124	2.112×10^{-4}
2	0.0305	1.4016×10^{-3}
3	0.111	1.2544×10^{-3}
4	0.301	2.528×10^{-3}
5	1.13	7.36×10^{-4}
6	3.00	2.688×10^{-4}

The mean prompt neutron lifetime ℓ was calculated using the expression:

$$\ell = \frac{-\Delta \rho}{V \Delta \Sigma_a}$$

where

$$\Delta \rho = \text{change in reactivity} \left(\rho = \frac{K_{\text{eff}} - 1}{K_{\text{eff}}} \right)$$

addition of a small amount of $1/v$ absorber (boron was used) to the core of interest

$\Delta \Sigma_a$ = the change in thermal macroscopic absorption cross section due to the addition of the absorber

V = average velocity of thermal neutrons = $1.3 \times 10^4 \sqrt{T_K}$ =
 2.225×10^5 cm/sec at 68° F.

Prompt neutron lifetimes were calculated for several different PMZ-1 cores of interest. The variation in ℓ from 1.3×10^{-5} second to 1.7×10^{-5} second for the different cores had a negligible effect on reactivity (i.e., 3×10^{-7} for 20-second periods and 3×10^{-8} for 200-second periods). The value of $\ell = 1.3 \times 10^{-5}$ second was used in calculating reactivity versus period since this was the calculated value for the PM-1 design core.

Keepin's data in Table II-8 is the most recent delayed neutron fraction data available. Reactivity for any period (T) calculated using this data is ~5% lower than that calculated from previous data³. However, from the nature and precision of the more recent experiments, the data in Table II-8 supersedes the previous results.

Although the reactivity versus period relation shown in Fig. II-23 represents the best available at this time, additional studies are considering dynamic multiplication factors and their relationship to the kinetic equations. This work is based on studies by Gross at ORNL.

Critical configuration studies. The effective multiplication factor (K_{eff}) calculations for determining the critical core configuration were made using the IBM-709 machine multigroup diffusion code Program C-3. The reflected core was treated as an equivalent bare cylinder and the distribution of the core materials was treated as homogeneous. A thermal disadvantage factor was used to account for heterogeneity. Epithermal disadvantage factors were evaluated but were found to have a negligible effect on the results. Reflector savings were calculated using the consistent K_{eff} method, from a combination of the multigroup criticality Program C-3 and Program F-3, a one-dimensional, multiregion, multigroup diffusion code. The calculations were performed using both two- and three-group models. A sufficient number of calculations were performed to obtain curves of K_{eff} versus number of fuel elements. From these curves, the critical configuration was determined.

The calculated critical core configuration, using the method described above, contained 142 fuel elements (each fuel element containing 39.5 grams of U-235) from the three-group calculation (three-group constants and three-group buckling) and 129 fuel elements from the two-group (two-group constants and two-group buckling) calculation.

The experimental critical core, as shown in Fig. II-1, contained 127 fuel elements.

The core multiplication factor for this core was recalculated using the same methods as described above. The resultant multiplications were 0.97 and 1.0 for the three- and two-group models, respectively, as indicated by the K_{eff} versus number of fuel element curves.

Evaluation of the MPR zero-power test critical core data (MND-MPR-1646) had indicated that the calculated effective multiplication factor would be approximately 1% high for the three-group calculation and approximately 2.4% high for the two-group calculation. This would indicate an even higher calculated critical mass, i.e., 147 and 141 fuel elements versus the 142 and 129 given above.

The main reason for the disagreement between the calculated and experimental critical mass is due to the inability, under diffusion theory, to calculate the leakage from the system correctly. (The fact that the two-group calculation correctly predicted the critical mass is not significant.) Although, in general, diffusion theory calculations are usually 1% to 3% high, such calculations for small cores have been found to allow for too much leakage. This is especially true for the critical configuration investigated in which the critical core radius was 4.04 inches and the height-to-diameter ratio equal to 3.72.

One method to compensate for this is to replace the leakage term

$$DB^2 = \frac{B^2}{3 \Sigma TR} \quad \text{by} \quad \frac{B}{\text{TAN}^{-1} \frac{B}{\Sigma TR}} - \Sigma_{TR}$$

This was done for the three-group, one-dimensional axial calculation. The resulting K_{eff} was 1.01.

Calculations of the critical cores containing lumped burnable poisons gave results similar to those above, i.e., K_{eff} between 0.97 and 1.0. However, the additional number of fuel elements required to compensate for the lumped poisons was correctly predicted.

Total core reactivity studies. Pre-experiment analyses of the full-size core reactivity experiments were completed for the following cores (see Figs. II-24 and II-25):

- (1) PM-1 design core.

Mean Diameters

Equivalent core diameter	22.48 in.
Outer edge of outer fuel element	22.80 in.
Shroud ID	23.100 in.
Shroud OD	23.600 in.
First thermal shield ID	24.275 in.
First thermal shield OD	27.478 in.
Second thermal shield ID	28.150 in.
Second thermal shield OD	34.950 in.
Pressure vessel ID	35.625 in.
Pressure vessel OD	40.000 in.
Active core height	30.000 in.

- Fuel element
- ⊗ Lumped poison rod
- Stainless steel tube
- Source tube location
- Stainless steel rod

Number of fuel elements	732
Number of lumped poison rods	75
Fuel element spacing (except across split line)	0.665 in.
Core loading U-235	28.255 kg
Number of control rods	6

Fig. II-24. Design Core Configuration (top view)

Region	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>
○ Fuel element	12	6	126	156	180	252
● Lumped poison rods	3	0	24	30	18	0
⦿ Stainless steel rods (center bundle structure)	3	0	0	0	0	0
⊙ Stainless steel tubes	0	0	0	18	0	0
Control rod guides	0	6	0	0	12	0
⊕ Source	1	0	0	0	0	0

Core	Lumped Poison Rods			Fuel Element Loading
A	PM-1 design	0.462 wt % natural boron		39.5 gm U-235
27	PMZ-1 rod 1	0.275 wt % natural boron		39.337 gm U-235
52	PMZ-1 rod 2	0.525 wt % natural boron		39.337 gm U-235
84	PMZ-1 rod 3	0.845 wt % natural boron		39.337 gm U-235
Zero		100 % stainless steel		39.5 gm U-235
FE	Lumped poison rods replaced by fuel elements			39.5 gm U-235

Fig. II-25. Core Regional Material Data

- (2) Core 27--0.27 wt % natural boron lumped poison rods replacing the design burnable poison rods.
- (3) Core 52--0.52 wt % natural boron lumped poison rods replacing the design burnable poison rods.
- (4) Core 84--0.84 wt % natural boron lumped poison rods replacing the design burnable poison rods.
- (5) Core zero--stainless steel rods replacing the design burnable poison rods.
- (6) Core FE--fuel elements replacing the design burnable poison rods.

The effective multiplication factor, K_{eff} , was calculated using IBM-709 machine one-dimensional, multiregion, three-group diffusion code F-3 as described in previous quarterly progress reports. The core was divided into six radial regions as shown in Fig. II-25. The distribution of core materials within a region was considered to be homogeneous. A thermal cell correction was used to account for heterogeneity. Three-group constants for the different regions of the core and reflectors were obtained from the multigroup diffusion code C-3.

In the PMZ-1 studies, both two- and three-group calculations were performed for some critical and full-size cores. The K_{eff} from the two- and three-group models converged with increasing core diameter. Two- and three-group K_{eff} calculations were performed for the design core and Cores 27, zero and FE. The difference in K_{eff} between the two- and three-group calculation, in all cases, was less than 0.005. The rest of the analysis was done using three-group calculations. This corresponds to the PM-1 design calculations in which three-group theory was used to obtain more accurate burnup (and reactivity) results.

Total core reactivities are given in Table II-9. Core descriptions are given in Figs. II-24 and II-25.

Experiments for evaluating total core reactivity for the different cores are in progress and will be completed during the next quarter. From results obtained to date, a comparison between analytical and experimental results shows very good agreement. The calculated K_{eff} is (at maximum) approximately 1.5% high. The calculated changes in reactivity between Cores zero and 27 and between 27 and 52 are within the experimental error.

TABLE II-9

Core (see Figs.II-24 and II-25 and text above for description)	Total Core Reactivity (% ρ)		Difference in Core Reactivities, ($\Delta\rho$)	
	Analytical	Experimental	Analytical	Experimental
PM-1 design core	11.8			
27	14.6	13.2 ± 0.5		
52	10.9	9.44	6.4 (Core zero - Core 27)	6.3 ± 0.7
84	7.2			
Zero	21.0	--		
FE	25.3	--	3.7 (Core 27 - Core 52)	3.76

Rod bank worth versus insertion. Worth versus insertion for the three- and six-rod banks was determined in the PM-1 design analysis studies. The results are shown in Fig. II-26. Also included in Fig. II-26 are experimental points for part of the six-rod bank curve.

The relative effectiveness of the rod banks as a function of the axial position in the core was calculated using a "window shade" model technique. First, a concentration of poison in the core that resulted in a decreased K_{eff} equal to that resulting from the full insertion of the rod bank was calculated. This concentration of poison was then included with the core materials in the regions of rod bank insertion in a multi-region, one-dimensional, three-group calculation. By varying the rodged region height, a series of core reactivities was obtained from which the worth versus insertion was calculated.

Part of the six-rod bank curve has been evaluated experimentally. The experimental data, normalized to 22.6 inches insertion, are shown in Fig. II-26. The agreement between the analytical and experimental points is very good.

Critical rod bank studies. The pre-experiment analysis studies to determine the critical six-rod bank position for cores of interest were completed.

The critical rod bank positions were calculated from results of core reactivity, rod bank worth and worth-versus-insertion studies. The ratio of total core reactivity to rod bank worth is the fraction of the rod bank worth that must be inserted. The distance that the rod bank is inserted is then read directly from Fig. II-26. The calculated critical bank positions are given in Table II-10.

TABLE II-10
Calculated Critical Six-Rod Bank Positions

<u>Core Description</u> (see Reactivity Studies, Figs. II-3 and II-4)	<u>Critical Rod Bank Position</u> (inches inserted)
PM-1 design core	20.3
Core zero	24.2
Core 27	21.6
Core 52	19.8
Core 84	17.2

Fig. II-26. Relative Rod Bank Worth Versus Insertion

The experimentally determined critical rod bank position for Core 27 was 22.58 inches. The agreement of the analytical value (21.6 inches) within one inch of the critical bank position is considered very good. The difference is believed due to an overestimate of the total bank worth. The calculated critical rod bank positions for the other cores given in Table II-10 will, therefore, probably also be less than the experimental value.

Temperature coefficient studies. Preliminary studies to determine the reactivity of the core as a function of temperature for several of the PMZ-1 experimental cores were completed. The cores investigated included: (1) the clean (all fuel elements), critical core configuration; (2) a clean (all fuel elements), core (temperature coefficient as a function of core diameter); and (3) the PM-1 design core.

The results indicate a positive temperature coefficient for the clean core configuration over part of the temperature range, a positive temperature coefficient (over part of the range) which decreases with increasing core diameter, and a negative temperature coefficient for the PM-1 design core, for all concentrations of boron in the lumped poison rods.

The effective multiplication factor, K_{eff} , was calculated using the multigroup diffusion code C-3 and the multiregion diffusion code F-3.

The change in reactivity with temperature was assumed to be due to:

- (1) The change in microscopic thermal cross sections with temperature where the thermal cross sections were Maxwellian-averaged cross sections.
- (2) The change in thermal heterogeneity factor as a function of temperature.
- (3) The change in reflector savings resulting from a change in buckling with temperature.
- (4) The change in density of water with temperature. Core materials other than water were assumed to have a constant density in the temperature range of interest.

For a PM-1 type core, the effect of (1) and (2) is negative, the effect of (3) is positive and the effect of (4) is usually negative. For some cores in which the effect of a decrease in absorption, resulting from the decrease in density of water, is more important than the decrease in moderator (the density effect), (4) can be positive. For the PMZ-1 cores

investigated, the most significant temperature effects were: (1) the positive effect due to the change in buckling and (2) the negative effect due to the change in density of water. The buckling effect is more important for a small core where a small change in dimension is relatively large.

a. Clean, critical configuration (see Fig. II-2)

The changes in K_{eff} with temperature for the clean, critical core are given in Table II-11.

TABLE II-11

Change in K_{eff} with Temperature for the Clean, Critical Core

Temperature (°F)	ΔK_{eff} ($K_{\text{eff}}(T) - K_{\text{eff}}(68^\circ \text{ F})$)	Temperature (°F)	K_{eff}
68	00	150	+0.01344
70	+0.00018	175	+0.01139
75	+0.00136	200	+0.00844
80	+0.00246	250	-0.00007
120	+0.01004	300	-0.01151

As seen from the data in Table II-11, there is a positive change in K_{eff} with increasing temperature up to $\approx 150^\circ \text{ F}$. Above 150° F , an increase in temperature results in a decrease in core reactivity and the rate of decrease increases with increasing temperature.

The nuclear and density effects were calculated separately. These results are given in Table II-12.

TABLE II-12

Nuclear and Density Temperature Effects on K_{eff} for the
Clean, Critical Core over a Temperature Range of 68° F to 150° F

Temperature (°F)	<u>Effect of Change in</u>		
	<u>Cell Corrections and Cross Sections</u>	<u>Reflector Savings</u>	<u>Water Density</u>
68	00	00	00
70	- 0.00003	+ 0.00061	- 0.00043
75	- 0.00011	+ 0.00227	- 0.00083
80	- 0.00019	+ 0.00392	- 0.00130
120	- 0.00082	+ 0.01702	- 0.00625
150	- 0.00130	+ 0.02660	- 0.01208

As seen from the data in Table II-12, the main reason for the positive effect over the range from 68° to 150° F is the increase in K_{eff} due to the change in buckling with temperature.

The temperature coefficient for this core was not evaluated experimentally. However, experimental observations did indicate a positive temperature coefficient at least over the range of 50° to 68° F.

b. Effect of core diameter

The change in multiplication with temperature for a core containing only fuel elements as a function of core diameter is given in Table II-13.

TABLE II-13

Change in K_{eff} with Temperature as a Function of Core Diameter

<u>Radius (cm)</u>	<u>68° F</u>	<u>150° F</u>	<u>300° F</u>
8.46	00	+ 0.0142	--
14.60	00	+ 0.0071	--
20.74	00	+ 0.0027	- 0.0305
28.25	00	+ 0.0002	- 0.0263

The results indicate a small positive temperature coefficient over part of the temperature range. The positive temperature effect on reactivity decreases with increasing core diameter and increasing temperature. As noted previously, the positive change due to the temperature-dependent buckling becomes less important for larger cores. Since the PM-1 core is considerably different from this core, no attempt was made to verify this analysis experimentally.

c. PM-1 design core

The reactivity of the PM-1 core as a function of temperature from 68° to 473° F for clean and operating conditions as a function of core burnup was given in the fourth quarterly progress report (MND-M-1815, Fig. III-21, and pages III-93 and III-95).

Results of pre-experiment analysis studies for PMZ-1 design size cores are given in Table II-14. Basically, these cores are the PM-1 design core with the design lumped poison rods replaced by rods containing different concentrations of boron (including zero concentration, i.e., all stainless steel rods). For these studies, the nonuniform distribution of core materials was taken into account by dividing the core into six radial regions and calculating the K_{eff} using the multiregion, three-group diffusion code F-3.

TABLE II-14
Reactivity as a Function of Temperature

<u>Temperature (°F)</u>	<u>PM-1 Design</u>	<u>Core 27*</u>	<u>Core Zero</u>	<u>Core FE</u>
68	0.1183	0.1450	0.2135	0.2533
70	0.1181	0.1449	0.2134	0.2533
80	0.1177	0.1445	0.2133	0.2531
120	0.1156	0.1428	0.2124	0.2523
170	0.1120	0.1397	0.2105	0.2504

* 0.284 wt % instead of 0.275 wt % boron in the lumped poison rod was used in this study.

The calculated temperature coefficient for all of the design size cores was negative.

The temperature coefficient studies for experimental PMZ-1 core were completed. A preliminary evaluation indicates good agreement within the total experimental and analytical error.

Fast fission fraction. The epithermal and epicadmium fission fractions for Core 52 (see Figs. II-24 and II-25) were calculated separately for each of the six regions, assuming a homogeneous core.

The epithermal and epicadmium fission fractions were calculated as follows:

The epithermal fission fraction (i.e., fraction of fissions above 0.322 ev) is

$$\frac{\Sigma_f \phi_{\text{epithermal}}}{\Sigma_f \phi_{\text{total}}} = \frac{\int_{u=0}^{u=19.554} \Sigma_f(u) \phi(u) du}{\int_{u=0}^{u=19.554} \Sigma_f(u) \phi(u) du + \Sigma_{f_{th}} \phi_{th}}$$

The epicadmium fission fraction (i.e., fraction of fissions above 0.4 ev) is

$$\frac{\Sigma_f \phi_{\text{epicadmium}}}{\Sigma_f \phi_{\text{total}}} = \frac{\int_{u=0}^{u=17.034} \Sigma_f(u) \phi(u) du}{\int_{u=0}^{u=19.554} \Sigma_f(u) \phi(u) du + \Sigma_{f_{th}} \phi_{th}}$$

Results obtained were:

<u>Region</u>	<u>Epithermal Fission Fraction</u>	<u>Epicadmium Fission Fraction</u>
1	0.40	0.20
2	0.20	0.08
3	0.41	0.21
4	0.35	0.17
5	0.38	0.19
6	0.42	0.21
Volume weighted average	0.38	0.19
Core 52--homogeneous distribution	0.38	0.19

Experimental subcadmium fission fractions will be obtained in the flux experiments to be conducted during the next quarter.

Power density distributions. Power density distributions for the PM-I design core, including the effect of rod insertion, were evaluated. From these studies, axial peaking, axially integrated radial power distribution and "peak" power fuel elements were determined. Sufficient data was obtained to permit analysis of the gross three-dimensional relative power distribution in the core. Gross power density distributions obtained in these studies included:

- (1) One-dimensional, "window shade" model axial power density distribution with six rods partially inserted for different insertions of the rod bank (Fig. II-27).
- (2) Two-dimensional, radial power density distribution for the core without control rods (Fig. II-28).
- (3) Two-dimensional, radial power density distribution for the core with six rods fully inserted (Fig. II-29).

The one-dimensional, "window shade" model axial power densities were obtained from the three-group, one-dimensional, multiregion diffusion theory code F-3. The two-dimensional, radial power density distributions for the core were obtained using the three-group, two-dimensional diffusion theory code PDQ.

Fig. II-27. Relative Fission Density Versus Distance from Bottom of Core for Different Rod Bank Insertions

Decimal point locates position of value given
Power relative to radial average axially integrated power

Fig. II-28. Radial Power Density with Control Rods Fully Withdrawn

The axial peak power density relative to the axial average power density in the core for different six-rod bank insertions are given in Table II-15.

TABLE II-15
Axial Peak-to-Average Power Versus Six-Rod Bank Insertion*

<u>Rod Bank Insertion (in.)</u>	<u>Relative Insertion</u>	<u>Peak-to-Average Power</u>	<u>Location of Peak (in. from top of core)</u>
0	0	1.39	15.3
6.0	0.2	1.56	17.1
10.5	0.35	1.79	19.5
15.0	0.5	2.12	21.6
19.5	0.65	2.54	23.7
24.0	0.8	2.72	25.8
30.0	1.0	1.39	15.3

* At PM-1 operating conditions, the axial peak to averages are $\approx 5\%$ lower.

The axially integrated power densities relative to the radial average axially integrated power density in the core for representative "peak" locations are given in Table II-16. The approximate fuel element locations are shown in Fig. II-30.*

*For these studies, a fuel element was placed in location No. 1 to show relative power peaking near the control rod channel. In the PM-1 design the fuel element is replaced by a stainless steel tube.

NOTE:

For this study, a fuel element was located in location 1.
In the design core configuration this fuel element is
replaced with a stainless steel tube.

Fig. II-30. Location of Fuel Elements for Power Distribution Studies

TABLE II-16

Relative Radial Power Distribution Versus Six-Rod Bank Insertion

Fuel Element Location (see Fig. II-30)	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	Control Rod Bank Insertion (%)
	2.4	2.1	2.1	1.2	1.4	0
	2.3	2.0	2.0	1.3	1.5	19.7
	2.1	1.9	1.9	1.3	1.6	45.9
	1.8	1.7	1.7	1.3	1.7	65.6
	1.6	1.6	1.7	1.3	1.7	72.2
	0.6	1.0	1.2	1.5	2.2	100.0

The power density at any location in the core relative to the average in the core is approximately equal to the product of the relative radial and axial power densities. From the data given in Figs. II-27, II-28 and II-29, a three-dimensional gross distribution of the power density in the core can be obtained for any insertion of the six-rod bank. For example, the power distribution in the core (relative to the average in the core) for 20% insertion of the six-rod bank for six representative fuel elements is given in Table II-17.

TABLE II-17

Axial Power Distribution Relative to Average Core Power
for 20% Rod Bank Insertion

Axial Location Along Fuel Element (cm from top of core)	Fuel Element Location (see Fig. II-30)					
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>
0	0.07	0.11	0.13	0.16	0.24	0.12
10	0.23	0.39	0.47	0.58	0.86	0.43
15.24	0.33	0.55	0.66	0.82	1.21	0.60
20	2.14	1.87	1.87	1.07	1.25	0.62
30	3.17	2.77	2.77	1.58	1.85	0.92
40	3.70	3.23	3.23	1.85	2.16	1.08
50	3.60	3.15	3.15	1.80	2.10	1.05
60	2.93	2.56	2.56	1.46	1.71	0.85
70	1.80	1.58	1.58	0.90	1.05	0.52
76.2	1.75	1.53	1.53	0.88	1.02	0.51

The flux and power distribution experiments will be completed during the next quarter.

REFERENCES

1. Glasstone, "Principles of Nuclear Reactor Engineering," Equations 4.17.2 and 4.35.1.
2. "Reactor Physics Constants," ANL-5800, Table 1-13, Data by Keepin, Wimelt and Ziegler of LASL.
3. Progress in Nuclear Energy, Vol. 1, Physics and Mathematics-- McGraw-Hill, 1956, Chapter 7, Delayed Neutrons, J. Keepin.

B. SUBTASK 2.2--IRRADIATION TEST

The work accomplished during this quarter on Subtask 2.2 is reported under Task 5, Subtask 5.5, Fuel Element Irradiation Program, and will be reported thereunder in the future.

C. SUBTASK 2.3--REACTOR FLOW STUDIES

I. Starr, W. Taylor, K. Dufrane

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

Work planned for this quarter included:

- (1) Completion of bundle orifice tests.
- (2) Completion of modifications of the 5000-gpm flow loop for the full-scale flow test.
- (3) Fifty percent completion of component fabrication for the full-scale flow test.

Actual accomplishments of this quarter were:

- (1) The major portion of the bundle orifice tests (the flow ratio determination) was completed.
- (2) Modifications of the 5000-gpm flow loop for the full-scale flow test were 75% completed.

- (3) Component fabrication for the full-scale flow test was 20% completed.
- (4) Instrumentation fabrication technique for full-scale core pressure taps was developed.

During the next quarter, it is expected that:

- (1) Modifications of the 5000-gpm flow loop will be completed for the full-scale test.
- (2) Component fabrication for the full-scale flow test will be completed.
- (3) A test program for the full-scale flow test will be issued.
- (4) Instrumented tubes for the full-scale flow test will be calibrated.
- (5) The 5000-gpm flow loop will be acid cleaned and an operational checkout will be performed.
- (6) Full-scale flow test instrumentation will be installed.
- (7) Flow blockage tests (the second portion of the bundle orifice tests) will be completed.

1. Bundle Orifice Flow Test (Flow Ratio Determination)

Description of test rig. Water flow tests run at relatively low pressures and temperatures, using bundles of tubes to simulate the PM-1 reactor core configuration, yielded results which determine the ratio of flow inside the tubes to that outside the tubes. Three bundle sizes were tested in which the ratio of housing wall surface to number of tubes was varied. Results indicated the effect of wall drag on the flow ratio. Flow ratios determined in this test effort, ranging from 0.996 to 1.119, were higher than the design value of 0.923.

The test loop consisted of a 200-gpm centrifugal pump, discharging into 2-inch diameter piping. The test section was inserted in a vertical pipe run with flow upward. Total loop flow was measured in the return vertical pipe, using an orifice meter. This orifice was compared in the loop with two other flowmeters: a potter meter and a volumetric meter employing the wobble plate principle. At low flow rates, the orifice plate was calibrated by weighing the discharge. The pipe containing the orifice plate discharged into an open 50-gallon suction tank. A pump bypass was provided. Control of flow was effected through a combination of settings of throttle and bypass valves.

A test section consisted of a group of tubes held between two orifice plates and surrounded by a hexagonal housing. Three test sections were used containing 19, 37 and 61 tubes, respectively. Inlet and outlet transition pipes were provided for each test section to connect to the 2-inch diameter loop pipe. The inlet transition pipe (a diffuser) for the 61-tube bundle was sized to obtain a recommended gradual increase in area along its length. Thus, the transition pipes for the 37-tube and 19-tube bundles also fell well within the recommended value.

The following is a list of bundle test specifications:

(1) Tube spacing--triangular pitch	0.665 in.
(2) OD of tube	0.500 in.
(3) ID of tube	0.416 in.
(4) Inlet orifice diameter of tube	0.267 in.
(5) Outlet diameter of tube	0.332 in.
(6) Length of tube	32-5/8 in.
(7) Thickness of inlet orifice plate	5/8 in.
(8) Thickness of outlet orifice plate	1/2 in.
(9) Diameter of holes in inlet orifice plate for flow between tubes	0.200 in.
(10) Diameter of holes in outlet orifice plate for flow between tubes	0.250 in.
(11) Maximum operating temperature	150° F
(12) Maximum operating pressure	50 psig

Test section instrumentation. Test section instrumentation consisted primarily of static pressure taps located along the length of selected tubes for the measurement of the pressure drops associated with flow in the tubes. Each instrumented tube was one of a group of tubes which were similar with respect to location in the bundle. The flow rate measured in an instrumented tube determined the flow rate in a group. Since every tube in each bundle was part of a group in which there was one instrumented tube, the total flow through all tubes was determined. A measurement of total loop flow made it possible to determine the total flow outside the tubes by subtracting the total flow through the tubes.

Thus, the ratio of total flow through the tubes to total flow between the tubes was determined. Figures II-31, II-32 and II-33 show the locations of instrumented tubes in each test section. Although the specific elevation of static taps varies with tube location due to staggering, the distance between taps on each tube is 20 inches. The taps were staggered to reduce flow blockage at any one elevation. Instrumentation was also provided for the measurement of the overall head loss across each bundle and the system pressure. These static taps were located in the transition sections three inches above and below the upper and lower grids, respectively.

Pressure tap calibration. Each instrumented tube was individually calibrated to establish the relationship between head loss and flow rate in the PM-1 velocity range. The flow rate was measured by weighing the discharge over a known time and the head loss was measured on a differential manometer. Figure II-34 shows a schematic diagram of the tube calibration test setup.

Figure II-35 shows a typical plot of calibration data. This plot is a straight line. The slope of this line gives a value for α in the equation $h = KV^\alpha$. K is also determined from this graph. The relationship between the friction factor and Reynolds number is determined by substituting f and K in the following equation:

$$f = \frac{c D^{3-\alpha} K}{v^{2-\alpha} R_e^{2-\alpha}}$$

where

f = friction factor

$$f = \frac{2g D \Delta h}{v^2 l}$$

v = kinematic viscosity in ft^2/sec

c = a constant for all cases.

This relationship is plotted over a Reynolds number range from 4×10^3 to 2.5×10^4 .

Figure II-36 gives the friction factor versus Reynolds number relationship which is obtained from Fig. II-35.

*Instrumented tube

Fig. II-31. PM-1 Bundle Orifice Test--19-Tube Bundle

*Instrumented tube

Fig. II-32. PM-1 Bundle Orifice Test--37-Tube Bundle

*Instrumented tube

Fig. II-33. PM-1 Bundle Orifice Test--61-Tube Bundle

Fig. II-34. Schematic Piping Diagram of Tube Calibration Test Setup

Fig. II-35. PM-1 Bundle Orifice Test Calibration--Tube 19

Fig. II-36. PM-1 Bundle Orifice Test Calibration--Tube 19

Test results. At predetermined loop flow rates and temperatures, selected to produce desired velocities and Reynolds numbers in the test section, differential heads measured in instrumented tubes were recorded. The overall test section head loss and the loop temperature were also recorded. These measurements were made in each of the three test sections. Flow ratios calculated from these data were compared to determine the effect of housing wall drag, changes in velocity and changes in kinematic viscosity.

The test data were reduced in the manner described below.

Corrected manometer deflections were converted to head loss in feet of water at loop temperature. Reynolds number, friction factor and velocity were calculated using a trial and error method of solution in conjunction with the tube calibrations. The velocity in a tube was converted to gpm. This gpm was multiplied by the number of tubes located in similar positions in the bundle. With all tubes accounted for, the total inside flow was found by adding the flows associated with each group of tubes. The total loop gpm was found from a calibration curve. The inside-to-outside flow ratio was calculated.

Three bundle sizes were tested to determine the effect of wall drag on flow ratio. The ratio of wall surface to the number of tubes in the bundle decreases with increasing bundle size. It follows then that the percent outside flow will increase with increasing bundle size and the inside-to-outside flow ratio will decrease. The following tabulation (Table II-18) of average flow ratios for all acceptable test runs for each bundle shows this to be the case.

TABLE II-18
Average Flow Ratio for Each Bundle

<u>Number of Tubes in Bundle</u>	<u>Average Inside-to-Outside Flow Ratio</u>
19	1.090
37	1.043
61	1.004

These average flow ratios are plotted against bundle size in Fig. II-37. All test data lie within the maximum and minimum limits shown.

Fig. II-37. PM-1 Bundle Orifice Test

Figure II-38 shows the flow variation from tube to tube for the three bundles. Velocities in the instrumented tubes are divided by the average velocity of inside flow and plotted against radial displacement from the center of the bundle. The distortion from a flat profile which is observed in the 37- and 61-tube bundles is probably a result of a greater angle in the entrance diffuser. The diffuser was sized using the maximum permissible angle for the 61-tube bundle. The maximum flow variation, $(V_{\max}/\bar{V} - 1)$, is 11.6%.

Analysis of test data. The flow passages in the PM-1 upper and lower grid plates were designed to obtain a division of flow (inside to outside of fuel element) so that the coolant temperatures at the outlet of the hottest element are equal inside and outside the tube. Since the design of the grid plates was based on head loss coefficients for enlargements and contractions occurring in a well-defined flow passage, the bundle orifice tests were performed to verify the applicability of these coefficients.

The following conclusions were made as a result of analysis of the test data:

- (1) The flow division measured experimentally was nearly equal to that desired.
- (2) No changes in the PM-1 core final design are warranted.

The data seemed to indicate a slight trend toward a higher proportion of the flow inside the tubes as the flow rate per element increased. The magnitude of this effect is so small, however, that it was difficult to separate it from the slight scatter in the data.

The optimum flow division based on the thermal design of the core is 47.8% of the flow inside the elements. The 61-tube bundle, considered to be the best representation of the PM-1 core, gave 49.7% of the flow inside of the elements at the PM-1 flow rate. This results in an outside outlet coolant temperature of only 2.7° F in excess of the optimum value.

Data for the three tests are summarized in Table II-19.

Fig. II-38. PM-1 Bundle Orifice Test--Flow Variation Within Bundles

TABLE II-19
Summary of Bundle Orifice Flow Test Data

<u>Bundle (No. of Tubes)</u>	<u>Test Run</u>	<u>(gpm/tube)</u>	<u>Temperature (° F)</u>	<u>Inside Flow (%)</u>	<u>Outside Flow (%)</u>
19	9	2.25	76	51.9	48.1
19	11	3.50	75	51.9	48.1
19	12	4.38	77	51.9	48.1
19	13	4.38	78	52.0	48.0
19	14	3.50	78	52.0	48.0
19	15	2.25	79	52.8	47.2
19	16	2.25	147	52.5	47.5
37	1	2.28	77	50.7	49.3
37	4	2.29	74.5	50.7	49.3
37	6	3.26	77	51.2	48.8
37	7	4.66	77	51.7	48.3
37	8	4.66	76	51.6	48.4
37	10	2.31	150	50.6	49.4
37	11	3.26	72	50.9	49.1
61	1	2.31	79	49.1	51.9
61	2	2.31	70	50.2	49.8
61	3	3.10	70	50.2	49.8

2. Full-Scale Flow Test

General. The following was accomplished under the full-scale flow loop modification effort:

- (1) The pump suction tank was delivered and installed.

- (2) The pressure vessel was delivered. It was temporarily installed in the loop to facilitate the alignment of the loop piping.
- (3) The test platform was modified.
- (4) The vessel support stand was delivered and installed.
- (5) The major loop pipes and fittings were cut and fitted in place.

All design drawings were reviewed for modifications (manufacturing and tolerances) made in the basic PM-1 core design. Basically, the two designs are identical except as follows:

- (1) Slight modifications were necessary to route and lead out the required instrumentation. However, blockage or cutouts required for the instrumentation is at a minimum to minimize the effect on flow patterns.
- (2) Aluminum was substituted for stainless steel, where feasible, to reduce material and manufacturing costs.
- (3) Tolerances were relaxed in the pressure vessel design to reduce overall cost. To eliminate any adverse effect on the thermoshield flow passages, the method of core alignment was altered slightly to better suit the tolerance modification. Also, as a design simplification, a straight cylindrical vessel was utilized for the reactor flow test with internal aluminum fairing added to reproduce the reduced diameter at the throat of the PM-1 pressure vessel.

The completed drawings were stress checked and released for fabrication.

Instrumentation. During this quarter, the instrumentation scheme outlined in the fourth quarterly progress report was reviewed. The 16 retractable pitot-static tubes to be located below the thermal shields will not be used since the reactor internals prevent satisfactory installation. The present scheme is:

- (1) Outside annulus--Two static pressure taps located 20 inches apart at 4 equally spaced locations around the vessel periphery.
- (2) Middle annulus--Two static pressure taps located 20 inches apart at 18 locations around the vessel periphery. Twelve of these locations will be in the two quadrants adjacent to the inlet pipe since this is the area where the greatest variation in flow distribution is anticipated.

- (3) Inside annulus--Two static pressure taps located 20 inches apart at 2 locations on the periphery of the core shroud. Rotation of the core will permit these taps to move, giving readings for at least 6 locations.

Predicted test results. Results predicted for the gross flow test have been classified by reactor components and are as follows:

- (1) Inlet water box--The static pressure just below the water box is expected to be uniform around the vessel periphery. Numerically, the head loss across the water box orifice plate at any point is expected to be 2.4 feet plus the difference in static pressure between the point in question and the reactor inlet.
- (2) Thermal shields--Static (frictional) pressure drop between taps should be approximately 0.4 feet. A uniform velocity distribution is anticipated at the bottom of the thermal shields.
- (3) Upper skirt orifices--A small variation in flow distribution is expected around the periphery of the skirt due to the location of the outlet. However, this should not be extended back to the core. The average head loss across the skirt is expected to be 3.0 feet.
- (4) Center bundle hold-down tube--Five percent of the total loop flow (105 gpm) is expected to be directed up through the center bundle hold-down tube in order to cool the dome-shaped vessel head.
- (5) Control rod channels--The core orifice plate is designed so that there will be little or no cross flow into the control rod channels with the control rods fully inserted. With the rods in this position, it is expected that the flow rate in each control rod channel will be 210 gpm.

The flow rate in the control rod channels with the rods withdrawn depends on the amount of cross flow into these channels. The magnitude of this is very difficult to predict analytically. The PM-1 core hydraulic analysis was based on data obtained from the MPR Gross Flow Test. The predicted flow rate in each control rod channel with the rods withdrawn is 385 gpm.

- (6) Head on upper skirt--The differential pressure across the head on the upper skirt is expected to be 1.5 feet.

- (7) Core--Total flow rate inside the tubes should be 800 gpm with 0.15 foot of frictional pressure drop between taps. Total flow outside the tubes (including flow adjacent to the control rods) should be 1325 gpm with 0.056 foot of frictional pressure drop between taps. Of this amount, 208 gpm is expected to occur at the periphery of the core and 180 gpm between the pie-shaped bundles.

Pretest work. In preparation for the full-scale flow test, the following items were accomplished:

- (1) A numbering system has been devised and every tube in the core has been designated.
- (2) All the core instrumentation has been defined in terms of typical flow passages inside and outside the tubes.
- (3) All instrumentation external to the core has been designated.
- (4) An outline which establishes the method of data reduction has been written.
- (5) Static taps were fabricated to prove out a suggested welding technique. The pressure tap lines are fusion welded to the inside surface of a 1/16-inch thick boss prior to attaching the boss to the fuel tube. The pressure tap holes will be final-drilled from the inside of the fuel tube toward the boss to eliminate burrs. This method proved to be highly satisfactory in the bundle orifice test.

3. Flow Blockage Tests

Tests to determine the amount of fuel tube flow reduction caused by core support members were initiated during this quarter. This effort consists of placing rectangular bars upstream and downstream of the orifice plate and upper grid of the 61-tube test bundle previously used in the bundle orifice test. Only the center tube of this bundle is instrumented. Head loss measurements are made in the calibrated center tube before and after the insertion of the bar and indicate the extent of flow reduction. The parameters varied in these tests are velocity, bar size, and spacing between the bar and the orifice plate or upper grid. Figure II-39 shows the flow blockage chamber and the 61-tube bundle installed in the flow loop. Results of these tests will be reported in the next quarterly report.

Fig. II-39. Flow Blockage Chamber and 61-Tube Bundle in Flow Loop

D. SUBTASK 2.4--HEAT TRANSFER TESTS
J. J. Jicha, M. P. Norin, S. Frank, C. Eicheldinger

The objective of Subtask 2.4 is to obtain experimental data to support refined local boiling thermal and hydraulic design of the PM-1 core. The program is designed around three test sections, which are described below:

- (1) STTS-3, single-tube test section--A single tube design with flow inside the tube only. This test section is instrumented to obtain local boiling pressure drop and heat transfer data inside tube.
- (2) STTS-4, single-tube test section--A single tube contained within a housing so as to provide coolant flow inside and outside of the tube. This unit is employed in local boiling burn-out studies.
- (3) SETCH-2, seven-tube test section--This unit has flow outside the tubes only, and is instrumented to obtain pressure drop and heat transfer data outside tubes.

Work planned for this quarter included:

- (1) STTS-3--Testing to be completed.
- (2) STTS-3--Thermal and hydraulic data analysis to be initiated.
- (3) SETCH-2--Loop modification and installation to be completed.
- (4) SETCH-2--Testing to be completed.

Accomplishments during this quarter were:

- (1) STTS-3--Local boiling heat transfer and pressure drop test program completed and final report initiated.
- (2) STTS-4--Local boiling burnout tests were initiated.
- (3) SETCH-2--Loop modifications completed.
- (4) SETCH-2--Isothermal empty shell tests were completed and the seven-tube subassembly reworked.

The anticipated accomplishments for the next quarter are:

- (1) STTS-3--Final report will be completed.
- (2) STTS-4--Burnout test program will be completed and the final report initiated.
- (3) SETCH-2--Local boiling heat transfer and pressure drop tests will be initiated.

1. STTS-3 Test Program

The local boiling heat transfer and pressure drop tests were terminated during this report period. The proposed test runs at 1500 psia were eliminated in favor of special high heat flux runs and the time conservation realized by reduction from 203 to 147 runs. These high heat flux runs at PM-1 flow conditions represent the most critical conditions that could be experienced in the PM-1 core and are typified by the following.

<u>Pressure (psia)</u>	<u>Coolant Inlet Temperature (°F)</u>	<u>Flow Rate (gpm)</u>	<u>Heat Flux (Btu/hr-ft²)</u>
1305	364	1.099	3.05×10^5
1300	335	1.101	3.69×10^5
1290	280	1.096	4.54×10^5

These conditions were realized with no adverse effects on the test section. The general range of test parameters successfully investigated was:

- (1) Flow rate: 0.90 to 2.55 gpm.
- (2) Heat flux: 1.20×10^5 to 5.55×10^5 .
- (3) Inlet temperature: 420° to 450° F.
- (4) Test section pressure: 1100, 1200 and 1300 psia.

Post-test calibrations were completed for the bulk coolant and wall thermocouples and for both Pottermeters employed during the test. These calibrations resulted in refinements in the data that were reduced during the report period.

2. STTS-4 Test Program

STTS-4 was hydrostatically leak-tested and installed in the loop during the report period. Figure II-40 shows STTS-4 installed in the loop with instrumentation for control valves and loop tubing connected. A front view of the fully insulated and test-ready assembly is shown in Fig. II-41. While STTS-4 was being installed, the 300-kw selenium rectifier power supply was modified. It is presently wired to give 80 volts and 3750 amperes, thereby making the maximum power of the facility available for the burnout runs.

The present test concept of employing a burnout detector to cut off the test section power at incipient burnout has given rise to leakage problems. It was found that, after sudden interruption of power, excessive leakage occurred at the upper flange assembly because of the differential contraction of the assembly components. It was necessary to connect the primary coolant pump directly into the burnout detector triggering circuit to ensure the loss of coolant flow at the time of indicated burnout. This eliminated the leakage problem due to thermal contractions. Rewiring of the burnout detector cutoff point from the d-c to the a-c side of the rectifier was necessitated to incorporate the above feature. Other action taken to ensure the integrity of the unit and reduce "downtime" consisted of serration of the upper assembly flange and the employment of copper gaskets in the electrode flanges.

A total of 26 burnout runs was conducted during the report period. These initial results indicated that the Jens and Lottes burnout correlation employed in the test program design is conservative. Burnout heat fluxes were higher than those predicted and, therefore, burnout occurred at lower outlet subcoolings than anticipated. Thus, the program was revised to investigate the maximum burnout subcooling possible with STTS-4 test elements.

Four test sections were physically burned out due to malfunctions of the burnout detector, operator error and inadequate electrical contact at the inlet electrode flange. In order to alleviate the problems with the burnout detector, a regulated power supply, a large programmed burnout signal, and a more readily adjusted helipot for the precision potentiometer in the burnout detector triggering unit have been employed. Design changes in the burnout detector (voltage) probes and the test section brazed junctions have been made to improve the electrical contact and mechanical strength of the test section assembly.

Fig. II-40. STTS-4 Burnout Test Section--Loop Mounted

Fig. II-41. STTS-4 Test Ready

Figure II-42 is a picture of the STTS-4 test section tube after a physical burnout. Scaling indicated here has been reduced by abandoning the sodium phosphate water treatment. Instead of operating at a 9.5 pH, demineralized water with a resistivity of 8×10^5 ohm-cm is being employed. A closeup of the stainless steel tube and nickel extension after physical burnout is shown in Fig. II-43.

3. SETCH-2 Test Program

The SETCH-2 test assembly was disassembled for the isothermal forced connection empty shell tests. The empty shell was installed in the loop, hydrostatically leak tested and the isothermal pressure drop tests completed. Pressure drop data realized indicated an effect of the right angle flow entrance. This effect, however, is not expected to be significant in tests with the seven-tube subassembly installed.

The complete SETCH-2 test section, i.e., seven-tube subassembly plus the shell, failed to pass the inloop hydrostatic test. The unit was returned to Manufacturing for rework, and STTS-4 installed. SETCH-2 is presently being stored in the heat transfer loop area, and the test program will be initiated upon the termination of STTS-4 tests.

4. STTS-3 Data Reduction

From the new data, various heat transfer and pressure drop parameters were calculated so that the information obtained in the laboratory might be analyzed and meaningful conclusions drawn.

Heat transfer data. The millivolt readings representing the 15 measured outside wall temperatures and the two coolant inlet and two coolant outlet temperatures were recorded. The conversion from millivolts to degrees Fahrenheit was made using the standard table for chromel-alumel thermocouples. A calibration correction was then made, using an equation of calibration data obtained in preliminary tests with a standard thermocouple. These preliminary calibrations were made at temperatures of 375°, 400° and 425° F, and over a range of 450° to 690° F in 15° F increments.

The power and volumetric flow rate were obtained directly from the recorded data. The absolute system pressure was found next from the gauge pressure recorded in the laboratory. Since the flow measuring device was located between the heaters and the entrance to the test section, the proper coolant density for use in computing the mass flow rate is one determined at the inlet temperature. The temperature obtained from the coolant thermocouple in the inlet transition section was used since it gave a better result on the test section heat balance. The mass flow rate and the mass velocity were then calculated.

Fig. II-42. Physical Burnout of STTS-4

Fig. II-43. Close-up of Physical Burnout

Initially, the difference between the measured outside temperature and the inside wall temperature was calculated by the Wilson method. However, erroneous results were obtained and a different method was then employed.

The results of a number of runs with no power applied to the test section showed the magnitude of the heat loss to be within the accuracy of the instrumentation. The heat loss was therefore considered negligible. Of the four coolant temperature measurements made, the pair which gave the best heat balance was that in the transition sections. However, even this combination indicated in many cases that slightly more (within 5%) heat was removed by the coolant than was applied to the test section. Thus, the most accurate estimate of the heat removed by the coolant was taken to be the power applied to the test section as determined by the electrical measurements of current and voltage.

The diffusion equation for heat conduction in a tube in the radial direction only, with insulated outer surface and a uniform interval source, was solved and used to calculate the inside wall temperature from the measured outside surface temperature, the applied power and an average value of the metal thermal conductivity. The conductivity was determined by assuming a linear relationship between the two values given by McAdams for Type 347 stainless steel.

To find the overall coolant temperature rise ($\Delta\theta$), the applied power and the mass flow rate were used, together with the mean specific heat. To find the coolant temperature opposite each thermocouple along the tube, the value of $\Delta\theta$ was divided linearly and added to the inlet coolant temperature. The outlet coolant temperature might have been used as a base for these calculations, but the two inlet coolant thermocouples generally showed better agreement with each other than did the two outlet coolant thermocouples; therefore, the inlet rather than the outlet temperature was used.

After the local coolant temperature was determined, the temperature differences across the film, film coefficients, local subcoolings and wall superheats were calculated.

Pressure drop data. The corrected manometer reading for each increment was recorded, along with the axial distance from the tube inlet to the beginning of the corresponding increment. The manometer readings were corrected using the zero readings.

Since the scale of the inclined, well-type manometer is so calibrated as to account for both the angle of inclination and the small deflection of the fluid in the well, the reading is equivalent to the total deflection of a U-tube manometer. The densities of the manometer fluids were found at room temperature with the use of the curves of specific gravity versus temperature supplied by the manufacturer. The 1.2 specific gravity oil was used in the manometer which measured the incremental pressure drops, while the heavier fluid was used in the instrument measuring the drop over the whole length of tube. This latter measurement was corrected and is used to check the sum of the incremental drops. However, the taps across which the total measurement was made span a larger interval than that between the first and last incremental taps, the ratio of the intervals being 1.162. The use of this ratio, then, only provides an approximate comparison between the incremental and overall drop, since a linear extrapolation is not strictly correct unless the pressure gradient is uniform.

The coolant temperature opposite each incremental pressure tap was calculated in the same manner as described previously. The coolant densities were then found at the same locations by using the local coolant temperature and the system pressure.

In order to obtain the friction pressure drop from the measured data pressure drop, it is necessary to subtract the drop due to elevation and acceleration. Thus, the acceleration and elevation drops were evaluated and, together with the static drop found from the manometer readings, were used to calculate the friction drop. The Darcy friction factor was then determined from the pressure loss due to friction. It should be noted that this value of friction represents an average for the increment.

The subcooling and Reynolds number for each increment were found next.

The isothermal friction factor was then determined by substituting the Reynolds number into an empirical equation found by means of a least squares fit of data obtained in preliminary isothermal runs.

The Stanton number was then calculated using a film coefficient at the midpoint of the pressure drop increment. This was found by interpolation of the heat transfer data. Finally, the heat transfer factor was computed using a Prandtl number which also represented an average for the increment.

5. STTS-3 Data Analysis

In analyzing the heat transfer data, the inside wall temperature was plotted against distance from the tube inlet (see Fig. II-44) from data of ten runs chosen at random from the total of 96 local boiling runs. These graphs indicated that the wall temperature does (as has been found by previous investigations) remain constant over the length of tube in local boiling. Therefore, in plotting wall superheat versus heat flux, only one value for each run was used; this value was the arithmetic mean of all readings in local boiling. Rather than plot wall temperature versus distance for each run, the point of incipient local boiling was determined by inspection of the tabulated data. All values of wall superheat after this point were used to compute the mean.

As can be seen from the plots of wall superheat versus heat flux (Figs. II-45, II-46 and II-47), the values obtained in this test are higher than predicted by the Jens and Lottes equation. The difference could quite conceivably be due to the temperature drop across the boiler scale which is known to have formed on the inside of the tube. The inverse trend of wall superheat with pressure is present. At this time, there is no explanation for the considerably greater amount of scatter in the data at 1300 psia.

In considering the results of the pressure drop measurements, the ratio of f/f_{iso} has been used, rather than the value of f alone, in an attempt to eliminate the effect of scale. As was mentioned above, f , the isothermal friction factor, was determined experimentally for this test section, and it should be added that the determining runs were made after some 25 runs with power applied, so that at least some scale had formed at the time of the isothermal test. For this reason, the results were somewhat higher than smooth-tube data. The question naturally arises as to whether all the scale was formed initially (prior to the isothermal runs) or whether it was formed gradually over the duration of the entire test program. In attempting to answer this, several plots of f/f_{iso} versus subcooling were made, where the data were grouped according to date. No trend with time was discernible. However, subsequent analysis showed f/f_{iso} to correlate better with parameters other than subcooling, and further investigation of the scale problem is necessary.

The pressure drop data at 1300 psia did not compare well with the Westinghouse equation for f/f_{iso} ,⁽¹⁾ although there was some correlation with subcooling. The same plots were repeated for 1100 and 1300 psia with groupings on heat flux (q''), i.e., with heat flux as a parameter,

Fig. II-44. Inside Wall Temperature Versus Axial Distance

Fig. II-45. Wall Superheat Versus Heat Flux (1100 psia)

Fig. II-46. Wall Superheat Versus Heat Flux (1200 psia)

Fig. II-47. Wall Superheat Versus Heat Flux (1300 psia)

but no dependency was observed; also with $q''/q''_0 - 1$ * as a parameter with the same negative result. A plot of f/f_{iso} versus $q''/q''_0 - 1$, with pressure as parameter, was made and compared with Robde's correlation of the Buchberg data⁽²⁾. The approximate inverse predicted trend with pressure could be detected and there was correlation with $q''/q''_0 - 1$, but as in the comparison with the Westinghouse equation, the STTS-3 values of f/f_{iso} were higher. The discrepancy could be the effect of scale.

Use of a Reynolds-type analogy between heat transfer and momentum exchange in the local boiling region was suggested in a paper by Sobersky and Mulligan⁽³⁾. This type of analogy is usually presented with the Stanton number as a linear function of the friction factor, and sometimes includes a dependency on the Prandtl number⁽⁴⁾. A further dependency of the relationship on Reynolds number has also been presented in connection with a fluidized bed⁽⁵⁾.

The STTS-3 data at pressures of 1100 and 1200 psia have successfully been correlated in the form

$$f/f_{iso} = a + bj + cRe,$$

where j is the product of the Stanton number and the Prandtl number to the $2/3$ power, and a, b and c are constants determined by a least squares regression. The 1100-psi data and correlation are shown in Fig. II-48. The correlation coefficients were 95 and 92%. The constants in each case were different. The data for 1300 psia has not yet been fitted. Data of J. B. Reynolds⁽⁶⁾ was correlated in the same form equation with correlation coefficients better than 85%. Since the data of Reynolds were for pressures below 100 psia and the constants were widely different from those in the STTS-3 equations, a dependency on pressure is indicated.

6. Preliminary Analysis of STTS-4 Data

The data from the first few test runs made with the STTS-4 test section were briefly analyzed and compared with the correlation used in the design and analysis of the PM-1 core. It was found that the design correlation (Jens and Lottes) was quite conservative at low values of subcooling. That is, burnout occurred at considerably higher heat fluxes than anticipated. However, at higher values of subcooling, this correlation appears to accurately predict the burnout heat flux.

* Where q''_0 is the nonboiling heat flux which the same film drop and the Dittus-Boelter coefficient would yield.

Fig. II-48. Local Boiling Friction Factor Correlation at 1100 psia

REFERENCES

1. Green, "Revised Pressure Drop Recommendations for Pressurized Water Reactor Design," WAPD-TH-326, July 1957.
2. Jens and Lottes, "Recent Developments in Boiling Research," Journal of the American Society of Naval Engineers, No. 67, 1955.
3. Sobersky and Mulligan, "On the Relationship Between Fluid Friction and Heat Transfer in Nucleate Boiling," Journal of the American Rocket Society, January 1955.
4. Van Karman, "The Analogy Between Fluid Friction and Heat Transfer," ASME Transactions, November 1939.
5. Othinc, "Fluidization," Reinhold Publishing Corporation, New York, 1956.
6. Reynolds, J.B., "Local Boiling Pressure Drops," USAEC, ANL-5178, March 1959.

E. SUBTASK 2.5--ACTUATOR PROGRAM (J. Sieg)

During the fifth quarter, the prototype actuator vendor, the TAPCO group of the Thompson Ramo Wooldridge Company, was scheduled to test the prototype unit as required by system specifications.

System fabrication was completed early in March, and hydrostatic tests were conducted. Although the system withstood hydrostatic test pressure without damage, leakage exceeded that allowable by specification. Rather than conduct helium tests, which would certainly be unsuccessful, the unit was released for operation (reliability and life) testing.

It was then found that the net lifting force was inadequate. It was theorized that the cause of failure was the development of an excessive side thrust, which pressed the movable armature against the interior wall of the pressure thimble with sufficient force to reduce net lift to an unacceptable level.

Four suggested methods of alleviating the problem were:

- (1) Incorporating low friction devices, such as ball or graphitar bearings, in the region between the armature and the pressure thimble.
- (2) Placing a nonmagnetic steel wear sleeve between the armature and the pressure thimble, which would, by maintaining a gap in the magnetic circuit, reduce the side thrust. (This approach may also be expected to reduce the grip force developed.)
- (3) Redesigning the armature and coils to obtain lower flux densities in the air gap.
- (4) Increasing the capacity of the lift circuit in order to overcome the final drag force.

Each method was investigated, and it was found that no single approach would solve the problem.

The use of graphitar or ball bearings was generally undesirable, in that additional complexity would be added in a region that is extremely difficult to maintain.

The capacity of the lift circuit was increased by adding additional turns to the lift coil and by incorporating 1010 steel, which allows more magnetic flux to be passed through the circuit with the same magnetic activation in the body of the armature. Figure II-49 shows the lift capacity attained in a test rig with the various indicated circuit modifications. For ease of testing, no bundle segments were inserted inside the armature. Figure II-50 shows the effect on lift capacity of including bundle segments, and also of activating the grip coil. The curve for the bundle assembly with 10 amperes of grip current is most meaningful. Minimum system lift requirements are 168 pounds. It should be noted that graphitar bearings were included in this experimental setup.

An experimental armature, which included a wear sleeve and redesign of the armature position of the lift and grip circuits, was fabricated and tested in May. Testing confirmed that the effects of side thrust were essentially eliminated and indicated that actuator redesign, as follows, should solve the operating problems to date:

- (1) Design of grip coils to hold a 500-pound load at a coefficient of friction of 0.25.
- (2) Design of hold coils to hold a 250-pound load at a coefficient of friction of 0.25.
- (3) Addition of a pull-down coil to assure smartness of cycling and to make it unnecessary to rely upon gravity to pull down the armature during cycling.

The greater capacity of the grip over the hold coils is required to control the effect of the inertia of the central rod bundle during armature acceleration and deceleration.

The above redesign was accomplished by placing 10 redesigned grip coils in the space that would formerly have been required for 12 coils, by reducing the number of hold coils to 4 and placing them in the space that would formerly have been required for 5 coils, and by adding the pull-down coil in the space that would formerly have been taken up by 3 coils. Additional ampere-turns were added to all coils through the use of smaller wire.

The redesigned actuator is expected to be tested in mid-June.

Fig. II-49. PM-1 Control Rod Actuator Lift Capability

Fig. II-50. PM-1 Control Rod Actuator Lift Capability

III. TASK 4--FINAL DESIGN

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

The objectives of this task are to prepare and analyze the final design of the PM-1 nuclear power plant. This was the major effort during the third and fourth project quarters and was completed during the fifth project quarter.

The completion of submittals of the PM-1 final design (by subsystem) to the AEC for approval was accomplished during the quarter. By March 9, 1960 all subsystems except Subsystem 16--"Radioactive Waste Disposal System" had been submitted as scheduled. The Radioactive Waste Disposal System design was submitted on May 16, 1960.

Since the submittals to the AEC, comments have been received, reviewed and incorporated as required into the PM-1 final design drawings and specifications.

Since all further design work is of a hardware revision and engineering liaison type, further efforts are, and will continue to be, reported under Task 7.0--Fabrication and Assembly of Plant, and Task 8.0--Packaging.

IV. TASK 5--CORE FABRICATION

Project Engineer--Subtask 5.1, 5.2, 5.3, 5.4, 5.5--J. F. O'Brien

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

A. SUBTASK 5.1--FABRICATION OF THE PMZ-1 CORE

S. Furman, J. Neal, B. Sprissler

During this quarter, fabrication of the PMZ-1 fuel elements was completed. A total of 850 fuel tubes is in use in Task 2, Subtask 2.1.

Fabrication of a can to contain boron in solution for the PMZ-1 tests was initiated during this quarter and will be completed early next quarter.

B. SUBTASK 5.2--CONVERSION OF UF_6 TO UO_2

W. Thompson

Delivery of 102.5 kilograms of UO_2 was completed during this quarter. An additional 15 kilograms of UO_2 was ordered from the Mallinckrodt Chemical Company for use in PM-1 spare fuel elements.

C. SUBTASK 5.3--FUEL ELEMENT DEVELOPMENT

W. Precht, B. Sprissler, J. Waltman

The general objectives of this task are to determine the limits of control rod and fuel element fabrication techniques and to develop those mechanical aspects of core fabrication which are applicable to the PM-1 core.

During this quarter, the program on rare earth stabilization was completed and programs on control rod development and mechanical core development were initiated. These programs will continue through the next quarter.

1. Stabilization of Eu_2O_3

A major concern with the increased ceramic additions in the cermet cores for control blades is the effect of those additions on the fabrication characteristics of the blades. Sample control blades were fabricated by the standard picture frame technique to determine the effect of higher additions of the ceramic. Two of these blades contained cores consisting of 1:2 mol ratio $\text{Eu}_2\text{O}_3 + 2 \text{TiO}_2$ in Type 302B stainless steel, while two others had cores consisting of 1:1 mol ratio $\text{Eu}_2\text{O}_3 + \text{TiO}_2$ in Type 302B stainless steel.

Various particle sizes of the ceramic blend were used in both types of the 1:2 and 1:1 mol ratios. One core of each type had a particle size of -170 +200 mesh while the other cores, one of each type, utilized a particle size of -200 +325 mesh size. In each case, the europium content was maintained equivalent to that found in a 30 wt % Eu_2O_3 -stainless steel cermet. Prior to assembly, the sintered and coined cores (approximately 80% theoretical density) were flame sprayed with a 0.004- to 0.005-inch layer of stainless steel. The surfaces sprayed were those which contacted the top and bottom clads. The purpose of spray cladding the cores was to promote bonding between core and clad in the early stages of rolling. The assemblies were heated to 1200° C under vacuum and rolled. The rolling reductions averaged 10% per pass until an overall 3.5 to 1 reduction was achieved.

Gammagraphs of the as-rolled blades showed good "fill-in" of the core with no obvious defects. Metallographic examination indicated no cracking within the core and the existence of a good metallurgical bond, particularly at the core-clad interface where the flame-sprayed stainless steel had been applied. Metallographic examination shows little difference in the effect of original particle size of oxide in the core. Figures IV-1 and IV-2 show the microstructure of the core-clad picture frame interfaces. Figure IV-3 shows the core-clad interface only. The zone shown at the core-clad interface in Fig. IV-3 is the flame spray coating on the core.

Figures IV-4 and IV-5 show the same sample control blades before and after water autoclave testing at 640° F and 2200 psi for 144 hours. The control blades contained intentional defects in the clad so that the core area would be exposed. In addition, the test pieces were cut at each end to again expose the core section to the high temperature water.

Dimensional examinations were made, with extra care given to the exposed core areas. After testing, examination showed that no measurable dimensional changes took place during the test. Chemical analysis of the water from each autoclave after testing showed less than 1 ppm of europium from the blades containing the blend of 1:2 mol ratio of europium oxide to titanium oxide in Type 302B stainless steel and 7 ppm from the blades with cores containing the blend of 1:1 mol ratio of europium oxide to titanium oxide in Type 302B stainless steel.

NOTE: 3.5 to 1 Reduction in Thickness
37 wt % Eu_2O_3 + 2TiO_2 --Type 302B Stainless Steel Matrix

Fig. IV-1. Transverse Core Clad Picture Frame Interface

NOTE: 3.5 to 1 Reduction in Thickness
37 wt % Eu_2O_3 + 2TiO_2 --Type 302B Stainless Steel Matrix

Fig. IV-2. Transverse Core Clad Picture Frame Interface

NOTE: 3.5 to 1 Reduction in Thickness
37 wt % Eu_2O_3 + TiO_2 --Type 302B Stainless Steel Matrix

Fig. IV-3. Longitudinal Clad Core Interface Showing Flame Sprayed Zone

NOTE: Upper Samples Contain Cores of 42.6 wt % $\text{Eu}_2\text{O}_3 + 2\text{TiO}_2$.
Lower Samples Contain Cores of 37 wt % $\text{Eu}_2\text{O}_3 + \text{TiO}_2$.

Fig. IV-4. Sections of Control Blades Before Autoclave Test

NOTE: Upper Samples Contain Cores of 42.6 wt % $\text{Eu}_2\text{O}_3 + 2\text{TlO}_2$.
Lower Samples Contain Cores of 37 wt % $\text{Eu}_2\text{O}_3 + \text{TlO}_2$.

Fig. IV-5. Sections of Control Blades After Autoclave Test

A second autoclave test was run on the same samples duplicating the conditions of the first test, i.e., 640° F at 2200 psi for 140 hours. Dimensional examinations were again made before and after testing, and no measurable changes were noted. The chemical analysis of the water from each autoclave showed less than 1 ppm of europium from either sample.

2. PM-1 Control Rod Development

Two compacts were fabricated, consisting of 32.8 grams of Eu_2O_3 + TiO_2 and 77.5 grams of 304 stainless steel powder. These compacts were made for the purpose of taking gamma-radiographs to check the dispersion of europium titanate in the matrix of stainless steel. Due to the high density, the gamma rays would not penetrate. A deliberately segregated compact was made with the ceramic concentrated in one end of the compact. The radiograph indicated the density gradient, but particle distribution was not visible.

It was then decided to substitute samarium oxide with titanate in test compacts. This substitution was made to conserve the limited supply of europium oxide. Eight compacts were pressed at 30 tsi and encapsulated in a Type 304 stainless steel picture frame and cladding material. The sandwich was evacuated while cold and sealed off.

Hot rolling was done according to the following schedule:

- (1) Preheat for 1 hour at 2200° F in H_2 .
- (2) Hot roll, 0.020 inch per reduction for 10 minutes and reheat between passes.

The evacuation tube ruptured at the 11th pass. No bonding was evident, and the cladding separated from the frame.

In an attempt to salvage the specimen, it was furnace-cleaned, reassembled and evacuated according to the following schedule:

- (1) Preheat for 1 hour at 2200° F in H_2 .
- (2) Hot roll, 0.040 inch per reduction for 10 minutes and reheat between 4 passes.
- (3) Hot roll, 0.050 inch per reduction for 10 minutes and reheat between 8 passes.

The specimen again ruptured. A total of 14 passes was made, but the cladding was spalled off and the compacts ruptured laterally. The second assembly was made with the following hot rolling schedule:

- (1) Preheat for 1 hour at 2200° F in H₂.
- (2) Hot roll, 0.080 inch reduction for 10 minutes and reheat.
- (3) Hot roll, 0.060 inch reduction for 10 minutes and reheat between 11 passes.

This specimen showed evidence of one blister. Radiography indicated good filling of cavity and no cracks.

3. Tube Configuration for Tube-to-Grid Assembly

Investigation of the tube and grid configuration required to meet the inside diameter size and the proper fit to meet a 500-pound pull requirement was conducted.

It was found that wrought tubing and fabricated fuel elements responded differently to the swaging operation. The wall thickening of the wrought tubing was less than that previously experienced while using the laminated tubing. This variance required that swaging dies of different dimensions, one for the gross flow tubes (dummy tubing) and another for PM-1 tubes, be ordered. These dies have been ordered.

Test runs to check the inside diameter of the rolled tubes were conducted using both wrought and luminal tubing. It was found that the same variance in the hole size was required. Samples were made up using both straight holes and undercut holes in the receiving blocks. It was found that the undercut is definitely beneficial insofar as the amount of pullout force is concerned, undercut blocks requiring at least 150 pounds of additional pull to remove tubes.

The use of a torque wrench has increased the uniformity of the test results. Minor variances in the fit of the tube in the block or in the wall thickness of the tube have a gross effect on the pullout force required. Process controls must be extremely stringent to overcome this variance.

4. Control Rod Guide Rail Fabrication

Because of possible fabrication problems associated with the control rod guide rails, a sample rail was fabricated for evaluation. This part, containing the cutouts as required for nuclear considerations, met the design tolerances without difficulty. As a result, several manufacturing changes and simplifications have been proposed for the remaining production parts.

D. SUBTASK 5.4--PM-1 CORE FABRICATION

S. Furman

During this quarter, tests conducted on the PM-1 fuel element cladding material indicated that the eddy current test was not sensitive enough to detect the 0.7-mil, 1-inch long notch required as a defect standard. It was demonstrated that immersion ultrasonic testing was sensitive enough and, therefore, this technique will be used. The cladding material should be delivered early next quarter.

An order was placed for 66 pounds of Eu_2O_3 for fabrication of the PM-1 control rods and one spare control rod. Delivery of this material will be completed next quarter.

Fabrication of the PM-1 core master assembly tool and the peripheral bundle master assembly tool was initiated and will be completed next quarter.

Fuel element storage boxes were ordered and will be delivered early next quarter. Advance information has been submitted to several vendors for bids on special Type 304 and 347 stainless steel melts for fabrication of core components prior to placement of orders early next quarter.

E. SUBTASK 5.5--FUEL ELEMENT IRRADIATION PROGRAM

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

Accomplishments during this quarter included the completion and delivery of the PM-1-M fuel element to the SM-1 at Fort Belvoir, Virginia.

During the next quarter, it is anticipated that the PM-1-M fuel element will be inserted in the SM-1, and irradiation will be initiated.

1. SM-1 Irradiation Program

B. Sprissler, J. Neal

The first dummy bundle of the PM-1-M test program was assembled. The dummy tubes were expanded into one grid and allowed to float in the

other. It was found that, in swaging the tubes to the smaller diameter required for this assembly, even wrought tubing cracked. Some tubes cracked during swaging and others during the expansion operation. This was overcome by anneals between reductions and before expansion. The assembly operations, although tedious and time-consuming, went smoothly.

It was established on this bundle that stop-off was required to keep the braze material from flowing out over the core area of the plates. The braze itself flowed well. Examination showed that braze material flowed completely around the edges of the plates. The only problem area was around the grid section. One section between the grooves in the side plates was void in the first test assembly made. This was probably due to poor application. All other areas showed excellent melting and brazing. The plates remained flat and the dimensions of the brazed assembly were close to those of assembly before brazing. It was discovered during assembly that the grid into which the tubes had been rolled had grown. This was hand filed to fit the assembly and it was felt that this fit was responsible for the dimensional errors in one end of the assembly.

It was apparent also that the tubes in the first dummy assembly took a set during the brazing operation. The second dummy bundle was assembled and brazed in the vertical position to overcome this discrepancy. Some difficulty was encountered during assembly because the shorter plates were thicker than those used previously. These were reworked to permit assembly of the unit.

Details for the first actual assembly using fuel tubes were prepared. The results of this second assembly dictated that the assembly should be made in two steps:

- (1) Assemble all components except lower grid and tubes and braze into place.
- (2) Assemble tubes into lower grid, insert into brazed assembly and complete bundles.

This technique was used in the assembly of the actual PM-1-M bundle shown in Figs. IV-6 and IV-7. Figure IV-6 is an overall view of the bundle and Fig. IV-7 is a view directly into the top of the element. The fuel tubes at this end are allowed to float freely in the upper grid. The end box spring is located in the foreground, three fuel plates are visible and the two lumped poison rod dead ends are seen in the center four-tube cluster in the upper right and lower left corners.

A dummy PM-1-M fuel element which has been sectioned is shown in Fig. IV-8.

Fig. IV-6. Overall View--PM-1-M Fuel Element

Fig. IV-7. Top View--PM-1-M Fuel Element

Fig. IV-8. Sectioned Dummy PM-1-M Fuel Element

Upon completion, the PM-1-M fuel element was compared with a standard SM-1 fuel element in Cell 2 of the Critical Facility. The reactivity difference between the two was found to be $0.386\% \Delta K/K$, with the PM-1-M element being less reactive.

2. Inpile Pressurized Water Loop Irradiation Tests (WTR Tests)

R. J. Akin

During this quarter, this subtask was initiated. The thermal, nuclear, hydraulic and mechanical design of the test bundle was started. Critical experiments were begun.

During the next quarter, the design of the test bundle will be completed, the critical experiments will be finished and the sample will be installed in the WTR reactor.

This task was set up to obtain additional inpile data on PM-1 fuel and burnable poison elements operating at PM-1 temperature conditions.

The test sample, as presently contemplated, will consist of four PM-1 fuel elements and one PM-1 burnable poison element. The five will be irradiated in the center loop in the WTR until a peak burnup of 90% in the fuel elements is achieved.

A complete hydraulic, thermal and mechanical design of the test bundle was completed. However, since it was held unsatisfactory by the WTR internal hazards review committee, a complete reanalysis must be done. Work is in progress on this reanalysis. Critical experiments using PM-1 fuel elements supplied by The Martin Company have begun. The results of these experiments indicate that 90% burnup can be achieved in the center loop in approximately 19 months.

A hydraulic test was also planned based on the rejected design. This also must be redone since a new design is being contemplated.

V. TASK 6--FABRICATION OF DUMMY CORE

Project Engineer--J. F. O'Brien

A. ACCOMPLISHMENTS THIS QUARTER

- (1) Delivery of material for fabrication of the dummy core was 90% completed during this quarter.
- (2) Fabrication of the component assemblies was 20% completed.

B. ANTICIPATED ACCOMPLISHMENTS NEXT QUARTER

Fabrication of the dummy core will be completed.

VI. TASK 7--FABRICATION AND ASSEMBLY OF PLANT

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

The overall objective of this task is the fabrication and assembly of the PM-1. The engineering liaison and engineering changes required during the fabrication period are also performed under this task.

A. CONTROLS AND INSTRUMENTATION WORK ACCOMPLISHED

Project Engineers--G. Zindler, J. Sluss, R. Wilder

All PM-1 instrumentation specifications were completed, reviewed and issued during the quarter.

Purchase orders were issued to the Foxboro Company to supply the plant instrumentation and control console specified in MN-7601 with the exception of Part IV, Nuclear Instrumentation which is to be supplied by Stromberg-Carlson. The Radiation Monitoring System (MN-7602) has been ordered from the Tracerlab Company.

1. Process Instrumentation

General. Approximately 1/3 of the vendor certified drawings of process instrumentation have been received. The additional drawings are scheduled for delivery by June 15, 1960. Modifications in the process instrumentation have been completed since the last report period. These modifications affect certain of the temperature measurements throughout the plant, including the air-steam condenser controls. A temperature scan system has been incorporated to replace certain of the original temperature instrumentation and to extend the temperature measurement coverage of the air-steam condenser. This system consists of a 72-point monitor with continuous automatic scanning and alarm of 56 points with readout on demand. Three points are provided with readout on demand but no alarm. The remaining 13 points are spares. The readout measurements in the scan system are as follows:

<u>Point No.</u>	<u>Location</u>
1 to 9	Tubing on condenser section 1
10 to 18	Tubing on condenser section 2
19 to 27	Tubing on condenser section 3

<u>Point No.</u>	<u>Location</u>
28 to 36	Tubing on condenser section 4
37 to 43	Turbine bearing oil
44 to 49	Main generator static winding
50 to 53	Lube oil cooler inlet and outlet
54	Radioactive waste evaporator fluid
55	Cooling water outlet (primary pump)
56	High pressure demineralizer inlet
47 to 65	Spares
66	Cooling water outlet, RDWS (Radioactive Waste Disposal System) condenser
67	Cooling water outlet spent core tank
68	Cooling water outlet high pressure demineralized cooler
69 to 72	Spares

Points 1 through 36 have a common, set low alarm limit; points 1 through 9, 10 through 18, 19 through 27 and 28 through 36 may be manually disabled to prevent the constant alarming of an idle condenser section. Points 37 through 65 have individually set alarm limits. Points 66 through 72 are not scanned or alarmed and are read out on demand only. The system graphically displays off-normal conditions on individual annunciator lights and a common audible alarm.

2. Condenser Control

This system has been modified and the control scheme and instrumentation is as follows.

Pressure Control. The condenser pressure is varied by modulating the louver position or the fan speed. The fans are operated manually from the control console. One of the two fans in each condenser section has a two-speed motor and both fans may be operated either forward or reverse. The butterfly valves located in the fan discharge duct are controlled either manually or automatically to "open" or "close" from the control console. The condenser louvers are controlled either manu-

ally or automatically from the control console. The louvers are set to open sequentially in either control mode. The control signal for louver position is turbine exhaust pressure, which is set at 9 in. Hg absolute. Four auto-manual control stations are provided, one for each condenser section.

Air Cooler Section Control. The air cooler louvers are modulated individually either in the automatic or manual mode. The control signal is from a ΔP cell that measures the pressure difference between the condenser discharge header and the air cooler header. The louvers are positioned to maintain a constant ΔP or flow rate of noncondensables and steam.

Equipment. Certified drawings of the secondary system control valves have been received. These valves are pneumatically operated through an electro-pneumatic transducer and are so designed that the required equipment to operate the valve is mounted integrally on the valve upper works. This equipment includes an air filter-regulator, an electro-pneumatic transducer and an air lock valve that functions to lock the valve in operating position in case of instrument air failure. Also included is an air motor to the valve, and an air relay to regulate the drive motor air supply. Design of the desuperheater and feedwater flow nozzles are under way and certified drawings are scheduled for June 15, 1960.

Control Console. The preliminary control console layout has been completed by the Foxboro Company and has been forwarded to The Martin Company for review. The functional console layout is shown in Fig. VI-1.

A full-scale mockup of the console has been built and will be used in determining the optimum layout of switches and meters. Fabrication drawings and wiring diagrams for the console have been started. This work is being started slightly ahead of schedule to ensure meeting the December 1, 1960 delivery date.

3. Nuclear Instrumentation

General. Work was initiated during the quarter with Stromberg-Carlson to obtain a final nuclear instrumentation system. Module design was completed and released to manufacturing except for the fault location system, which will be released in the next quarter.

Equipment. Westinghouse neutron detectors have been selected. These detectors can be obtained in waterproof cans, permitting operation while submerged in the shield water tank. The exact locations of the detectors around the reactor have been established as well as a mechanism for positioning the startup detectors. The intermediate and power range

Totals

94 switches
 57 meters
 10 recorders
 21 control stations
 101 indicating lights

A-A

Fig. VI-1. PM-1 Control Console Layout

detectors will be in fixed locations and, if required, neutron shielding will be inserted between the detector housing and the waterproof can. Certified outline drawings have been received for the drawers and for the preamplifiers. A preliminary arrangement was made for six drawers to be mounted in one section of the console and the seventh (scram and logic) drawer to be mounted remotely in the console. The drawers consist of two startup range drawers, two intermediate range drawers, one power range drawer with two meters, a test drawer with the third power meter, and the scram and logic drawer.

A list and color selection of the pilot lights for the nuclear instrumentation was transmitted to Stromberg-Carlson.

4. Radiation Monitoring

General. Specifications MN-7602, describing the radiation monitoring system for the PM-1, was completed, approved and sent to prospective vendors for proposals.

Following proposal evaluations, Tracerlab was selected and a purchase contract let for the equipment.

Equipment. The equipment to be supplied consists of a ten-channel area monitoring system, six air sample stations, two water monitors and the associated laboratory monitoring equipment for counting samples and for limited gamma spectrometry analysis.

Tracerlab has begun preliminary design and layout work for the equipment and is working on the certified dimensions associated with mounting the equipment in the console and the mounting of detectors in the lines or in the various assigned areas.

A preliminary layout of the equipment in the control console has been completed. The equipment will be housed in one rack which will include a multipoint recorder, the area monitoring system, water monitors and amplifiers, and associated alarms and indicators.

5. Control and Instrumentation Accessories

General. Accessory instrumentation not included in MN-7601 has been specified in PM-1 data sheets. The following data sheets have been prepared:

(1) Primary system

- (a) Datum column for use with the pressurizer level set--Preliminary vendor information has been received and arrangement compatibility is being determined.

- (b) Shield tank level detector--A mechanical-electrical method of determining shield tank level is required due to arrangement difficulties encountered in conventional differential pressure transmitters. The vendor has been selected and the purchase contract placed.
- (c) Expansion tank level alarm switch--This switch is mounted on the expansion tank and functions to annunciate the water level in the expansion tank. The vendor has been selected and the purchase contract placed.

(2) Secondary system

- (a) Local mounted pressure gauges--Proposals from several vendors have been received and evaluated. Additional information has been requested.
- (b) Local mounted thermometer--Same as item (a) above.
- (c) Float switch for level alarm--Two switches are being furnished to alarm high level in the drain lines from the feedwater heater and evaporator coil. The vendor has been selected and the purchase orders placed.
- (d) Absolute pressure gauge--This gauge is used to measure turbine exhaust pressure. It will be a mercury well-type manometer and will be mounted on the control console. The vendor has been selected and the purchase order placed.
- (e) Pressure switch--This switch will provide antirush protection for the feedwater pumps and will be mounted on the deaerator. Its function is to open a valve bypassing the deaerator on low deaerator pressure.

6. Miscellaneous

A data sheet was prepared for the instrumentation maintenance shop. This area is housed on the control room skid on the end opposite the control console. The shop is so arranged that one side will constitute a work bench, the end and opposite side shall be used to house maintenance and test equipment in drawers and on shelves behind sliding doors. The work bench will be used for normal repair and maintenance of all plant instrumentation, including the pneumatic equipment provided for the main steam condenser and control valves.

A data sheet was issued to obtain a work bench to provide the required work area, in the decontamination package, for laboratory equipment. This includes the previously mentioned sample charger, well-type scintillators,

a scanning-type, single-channel analyzer, count rate meter, decade scaler and a strip chart recorder. This bench shall be of the "knockdown" type, so it can be shipped disassembled and assembled at the site.

Anticipated accomplishments next quarter.

- (1) Continued liaison with the Foxboro Company.
 - (a) Obtain all certified drawings.
 - (b) Complete console layout.
- (2) Continued liaison with Stromberg-Carlson Company.
 - (a) Obtain all certified drawings.
 - (b) Complete fault location design.
- (3) Continued liaison with the Tracerlab Company.
 - (a) Obtain all certified drawings.
 - (b) Release to manufacturing all long lead items.
- (4) Issue purchase orders for all miscellaneous equipment handled on data sheets.
- (5) Issue purchase orders on special test and maintenance equipment.

B. PRIMARY SYSTEM

Project Engineers--R. Akin, M. Rosenberg, K. Dufrane, J. Goeller
P. Medina, J. Todd, W. Dallam, H. Clark

During this quarter, the following items were scheduled and accomplished:

- (1) Incorporation of final design changes as a result of final design submittal to the AEC.
- (2) Completion of design of Radioactive Waste Disposal System (Subsystem 16).
- (3) Initiation of procurement on long lead items.

- (4) Selection of vendors for most primary loop equipment and establishment of liaison with the vendor.
- (5) Engineering revisions where necessary to reflect vendors' equipment requirements, including equipment rearrangement where necessary to meet the shipping criteria.

During the next quarter, the following items will be accomplished:

- (1) Procurement of all remaining plant equipment and components.
- (2) Maintaining present liaison and establishing liaison with new vendors.
- (3) Incorporation of vendor data into the plant design where necessary.

The final design of the primary system was completed and submitted to the AEC for approval. Subsystems 3, 6 to 17 and the core design were submitted. All of these subsystems have been described in previous progress reports under Task 4.0, except Subsystem 16, Radioactive Waste Disposal System.

A modular liquid waste disposal system was designed which enclosed the entire system within a single shipping and installation container. The system is shown in Fig. VI-2. The 1600-gallon tank is integral with the container. The 500-gallon storage tank is mounted in the bottom of the container below two feet of earth shielding. The evaporator and condenser have been redesigned to fit into the revised space limitations. The performance characteristics are unchanged from those reported before. The tank fits into the spent fuel storage tank for shipping purposes.

The following items have been placed for fabrication during this quarter:

- (1) Pressurizer
- (2) Coolant charging pumps
- (3) Expansion tank
- (4) Purification demineralizer
- (5) Shield water demineralizer
- (6) Shield water cooler
- (7) Shield water pumps

Fig. VI-2. Radioactive Waste Disposal System Arrangement

- (8) Decay heat removal pump
- (9) Steam generator container vent fan
- (10) Gantry crane
- (11) Steel superstructure
- (12) Steam generator support
- (13) Refueling dolly
- (14) Containers (Tanks 1, 2 and 3)
- (15) Waste disposal container
- (16) Waste storage tank
- (17) Waste disposal evaporator
- (18) Waste disposal condenser
- (19) Waste disposal sump pump
- (20) Portable demineralizer
- (21) Silver nitrate reactor
- (22) Purification economizer
- (23) Purification cooler
- (24) Spent fuel tank pump
- (25) Primary system piping
- (26) Core refueling tools
- (27) Core shipping canisters

The remaining procurement items in the primary system are awaiting bid information from vendors before they can be placed.

Engineering has been completed and fabrication initiated on the primary coolant pump and the reactor pressure vessel. Engineering is virtually complete on the steam generator, but fabrication has not started to date.

The pressure vessel design has been modified by eliminating the separate flux suppressor and replacing it with a shielding basket which will be filled at the site with lead shot. The design and fabrication of this basket is in progress.

An overall alignment study, considering all tolerances and fits between the reactor core, pressure vessel and control drive mechanisms, was completed. Although final confirmation of the control drive mechanism's design tolerances is still required, the following tentative results were obtained:

- (1) Adequate clearance is maintained between the control rod and supporting structure under the worst possible tolerance buildup.
- (2) Mismatch between the control rod and the control rod drive mechanism's pickup arm during refueling is well within acceptable limits.
- (3) Maximum possible bending in the mechanism's pickup arm decreases from 1/16 inch at full insertion to zero as the rod is withdrawn. Because of the flexibility of the arm, this amount of mismatch has a relatively minor effect on the operation of the mechanism.

C. SECONDARY SYSTEM

Project Engineers--C. Fox, L. Hassell, R. Groscup, J. McNeil,
A. Layman, R. Dugas

During the fifth program quarter, it was planned that vendors be selected for all secondary system equipment, piping, buildings, electrical material and maintenance equipment.

The following scheduled efforts were accomplished:

- (1) Evaluation of vendor proposals and designation of all major contractors and most vendors for minor items.
- (2) Maintaining of technical liaison with present subcontractors.
- (3) Initiation of fabrication by subcontractors.
- (4) Revisions, where necessary, to reflect vendors' equipment certified drawings.
- (5) Preparation of secondary system engineering requirements for site work scheduled in 1960.

- (6) Continuation of weights and center-of-gravity determinations for plant equipment to complete the shipping configurations.
- (7) Maintaining of liaison with Gibbs and Hill steam-electric system efforts.

The efforts planned for the next quarter will be concerned with:

- (1) Procurement of all remaining plant equipment and components.
- (2) Maintaining present liaison and establishing liaison with new contractors.
- (3) Continuation of fabrication, with delivery of several items such as the diesel generator, primary and secondary buildings, de-aerator, chemical injection equipment, lube oil conditioner and maintenance equipment.
- (4) Incorporation of vendor data into the plant design, where necessary.
- (5) Completion of the plant bill of materials and cable schedules.
- (6) Completion of the analysis of the piping interconnects between the primary and secondary systems, incorporating all associated vendor data.
- (7) Completion of the remaining package layouts, incorporating vendor data and showing equipment installations and package assemblies.

The subsystem descriptions were completed and submitted to the AEC for final approval for the Cooling Water System (Subsystem 23), Water Treatment System (Subsystem 29, A, B and C), Fire Protection System (Subsystem 31), Primary Building (Subsystem 34), Maintenance Items and Tools (Subsystem 37), and for the Plant Interconnecting Walkways, PM-1 Shipping Packages, the PM-1 Package Contents Summary and the Criteria for Maintenance. In addition, change notices incorporating minor corrections and additions were submitted for:

Subsystem

- 18 Main Steam System
- 19 Main T-G Unit

Subsystem

21	Feedwater System
22	Extraction Steam and Heater Drain System
24	Main Transformer and Distribution System
25	Station Service System
27	Plant d-c System
28	Emergency Power System
30	Condensate Makeup System
33	Building Heating, Air Conditioning and Ventilating System
35	Secondary System Building

Technical proposals accompanied by fixed price bids were received and evaluated, with the resultant selection of contractors for the following items, for which engineering drawings were reviewed and comments made.

- (1) Oil Cooler
- (2) Condensate Return Unit
- (3) Diesel Generator
- (4) Decontamination Laboratory Furniture
- (5) Lavatory-Shower
- (6) Chemical Treatment Equipment
- (7) Primary Building
- (8) Secondary Building
- (9) 12T Volt Battery
- (10) Lube Oil Conditioner

Bid evaluations continued for the maintenance tools and welding equipment.

Technical liaison was maintained with all secondary system manufacturers during the quarter. Engineering designs and data were received, reviewed internally and comments made.

Comments were received from the AEC on final design work previously submitted. Some revisions were already made to reflect these comments and the remaining changes are being incorporated into the final drawings and subsystem writeups.

Revisions continued, when necessary, in the incorporation of vendor certified data. Equipment bolt-down data was included in the package pallet drawings for approximately 80% of the secondary system equipment. An improved shipping package tie-down fitting design was completed. Liaison with Gibbs and Hill continued, incorporating detailed vendor information into final drawings.

Engineering drawings were prepared for vendor bidding, establishing the requirements for the lighting, heating and ventilating installation contractors. These drawings, together with the foundation requirements as delineated by PUMO and the primary and secondary buildings, set forth the entire 1960 site installation effort.

Weights and center of gravity of all electrical equipment, except the motor control center, were obtained to complete the shipping configuration.

The diesel generator specification was revised to include field rheostat, transfer switch, voltmeter and ammeter for mounting on the diesel generator. This change was initiated to facilitate the use of the diesel generator during the installation of the PM-1 at Sundance. Drawings of the mounting skid, wiring diagram and the outline configuration of the diesel generator were received. These were reviewed, comments made and returned to the supplier as approved.

In the decontamination package, the washer-dryer platform and pan drain were eliminated to conserve space and take full advantage of the area around the decontamination pad, especially during the use of the portable crane. Louvers were eliminated in the door separating the decontamination space from the main part of the package and detailed drawings of steam and drain piping and vent penetrations were initiated.

Most of the maintenance equipment was ordered; including all equipment that is to be permanently mounted on the maintenance package floor. Also added to the maintenance package during the quarter was a hotwell for use with the auxiliary boiler and a second air compressor with its receiver, dryer and filter. Maintenance tools that are not to be permanently mounted on the floor of the package will be crated for shipment.

VII. TASK 8--PACKAGING

Project Engineers--C. Fox, R. Akin

The overall objective of this task is to prepare the PM-1 packages for shipment. This includes fabrication of the PM-1 package skids and frames, assembly of these packages and preparation of the plant for shipment.

A. PRIMARY SYSTEM

The work on the primary system plant containers has been described under Task 7.0 and centered on the detailed design and fabrication of the primary loop container tanks during the last quarter.

B. SECONDARY SYSTEM PACKAGES

During this report period, efforts were initiated for fabrication of secondary system packages and for liaison between engineering, manufacturing and vendors. As noted under Task 7.0, the decontamination building drawings were submitted to the Commission for approval. After completion of design modifications reflecting vendor certified equipment data (i.e., the steel pallet stringers were located within the package, and the drawings were released), all basic steel design was completed, and basic fabrications were ordered. The insulated floor stringers were located in the decontamination package, and the drawings will be available during the next quarter.

Final design drawings of miscellaneous shipping packages were submitted to the AEC for approval during the quarter.

During May of the reporting period, at the request of the Commission, an alternate design of the heat transfer apparatus package was initiated, completed and submitted for approval. This design closely simulates the design of the 30-foot packages in that truss sides and a sandwich roof are used. The drawings previously submitted to the AEC used removable low beam sides and no roof (a tarpaulin was to be used for weather coverage).

Equipment support stringers were located in the controls package pallet. For the switchgear and maintenance package pallets, this effort was started and will continue into the next quarter.

The first packages to be completed for equipment installation will be the controls package, the switchgear package and the decontamination package.

During the next quarter, all secondary system packages are to be fabricated and made available for initiation of equipment installation efforts. The controls and switchgear packages will be shipped to subcontractors for initial equipment installation.

VIII. TASK 9

A. SUBTASK 9.4--PLANT REARRANGEMENT

1. Work Accomplished

During this reporting period, the plant rearrangement subtask was initiated. An area of 15,000 square feet of floor space was allotted in the engineering test building for PM-1 assembly.

An arrangement plan was prepared covering the location of storage cribs, machine tools, clean areas, etc., required during assembly.

2. Anticipated Accomplishments Next Quarter

During the next quarter, the various facilities will be installed. These include:

- (1) Machine tools
- (2) Cleaning tanks
- (3) Welding booths
- (4) Workbenches
- (5) Electrical wiring benches
- (6) Steel grillage under plant assembly area
- (7) Scaffolding around primary system assembly
- (8) Material and tools storage crib
- (9) Utilities--water, electricity, phones
- (10) Engineering office.

IX. TASK 11--SITE PREPARATION AND INSTALLATION

Project Engineer--G. Zindler

The objectives of this task are to prepare the site for the orderly installation of the PM-1 and to install and interconnect the packages into an operable nuclear power plant.

A. SUBTASK 11.1--SITE PREPARATION

(W. Thompson, R. Manoll)

1. Work Accomplished

During this reporting period, the design of the PM-1 foundations and grading was completed. This was accomplished under a subcontract to Porter, Urquhart, McCreary and O'Brien of Newark, New Jersey. In addition, drawings and specifications were completed by The Martin Company covering the plant components to be installed during the 1960 construction season. These include:

- (1) Primary system superstructure
- (2) Primary building
- (3) Secondary building
- (4) Building heating and ventilating system
- (5) Building lighting system
- (6) Primary building floor
- (7) Gantry crane
- (8) Shield water cooler housing.

Orders were placed for the equipment listed above, and deliveries were scheduled for the latter part of July and early August to meet the foundation construction schedule.

Invitations to bid were sent out to six firms active in the Sundance area for the site construction work. Bid opening and award of contract were scheduled for 3 June 1960 and the week following.

During this period, two trips were made by Martin representatives, in the company of AEC officials, to the Omaha District Corps of Engineers and Air Force Installations Representative Offices to coordinate the design and construction of the PM-1 with the Sundance Air Force Stations Operations area.

2. Anticipated Accomplishments Next Quarter

A construction contract will be awarded and construction will be initiated at the PM-1 site during the first part of July. It is expected that foundations will be completed by the first week in August. The erection of the buildings and installation of the other equipment will be completed by early September.

X. TASK 12--PACKAGE LOADING DEMONSTRATION

Project Engineer--C. Fox

The objective of this task is the demonstration of loading and unloading a typical PM-1 nuclear power plant package in a C-130A aircraft.

A. LOADING DEMONSTRATION
(A. Layman)

The objectives during this quarter were to prepare for and to conduct a demonstration showing how a typical PM-1 package, 30,000 pounds in weight, 30 feet in length, 8 feet 8 inches in height and 8 feet 8 inches in width, could be loaded onto a C-130A aircraft and flown. The objectives were successfully met in all respects.

Early in the quarter, tests were conducted to determine the coefficient of friction of wood shoring sliding on wood shoring with grease lubricant. A sliding friction of 0.14 was indicated. In conjunction with the demonstration, several drawings were prepared, covering details of the operation. These were:

- (1) The shoring arrangement within the C-130A aircraft (Fig. X-1).
- (2) The 30-foot test package tie-down arrangement within the C-130A aircraft (Fig. X-2).
- (3) The package tie-down plan and shoring requirements on the trailer (Fig. X-3).

The Air Force assigned a Load Master, Staff Sgt. John Kimes, to the Loading Demonstration. Sgt. Kimes made a preliminary visit to Martin-Nuclear in March, at which time he reviewed the plans.

Arrangements were made with the Air Force for the loan of a 40-foot C-2 trailer which was delivered early in April. The only work required on the trailer was the laying of a deck of shoring. This consisted of 2 x 10-inch planking laid crosswise, thus extending the width of the trailer to 8 feet 10 inches as well as supplying a skidding surface. Guide rails were nailed parallel to and 8-1/2 inches in from each side to ensure proper alignment of the package while being moved.

Arrangements were made with a local rigging firm for use of a 20-ton capacity mobile crane with a 45-foot boom.

Fig. X-1. Shoring Arrangement Within the C-130 Aircraft

Fig. X-2. Thirty-foot Test Package Tie-down Arrangement

Fig. X-3. Package Tie-down Plan and Shoring Requirements on the Trailer

Final preparation of the package included painting and stenciling of tow and tie-down fitting data.

By arrangement with the AEC, the public demonstration was extended to two days instead of one as originally planned.

On Monday, 25 April, the C-130A aircraft arrived at the Martin airport from Seward AFB, Tennessee. On Tuesday morning, the package was transported from the Engineering Test Building to the Martin Airport. The trailer, with package aboard, was backed partially under the empennage of the C-130A. It was found that minor adjustments were required in the airplane landing gear struts and the cargo door uplock, after which the trailer and package were moved all the way forward to the edge of the loading ramp. The trailer was then moved away from the airplane and the package was crane hoisted from the trailer to greased wood shoring laid on the ground. A tow cable was attached to the package and a dynamometer was rigged in the cable linkage. The starting pull was approximately 8000 pounds and, after starting, dropped to less than 4000 pounds.

On Wednesday, 27 April, additional check operations were conducted, including moving the package from the trailer into the airplane, partially tying down the package within the airplane, and moving the package out of the airplane back onto the trailer. Our first attempt was not successful due to high friction loads caused by grit the runners had picked up the day before. The package was removed from the trailer, its runners cleaned off, and sheet aluminum facing was added to the upper surface of the shoring, both on the trailer and on the airplane. With these improvements the package moved smoothly in and out of the airplane. The airplane's own winch was used to pull the package into the plane, and a Martin D-8 tractor was used to withdraw the package. However, given sufficient length of cable, the package could easily have been off-loaded with the airplane winch.

Check operations included practice in moving the package from trailer to timber grillage and, by the use of jacks and sequential withdrawal of grillage tiers, lowering the package down to ground level.

On Thursday, 28 April, and Friday, 29 April, formal demonstrations of the airplane loading and unloading and handling by crane, jacks and timber grillage were conducted.

Of particular significance, on Thursday, 28 April, after the package was loaded onto the airplane and properly tied down, the airplane made a high speed taxi run, accelerating from 0 to 85 knots in 5 seconds--followed by an equally rapid deceleration. No displacement of the package occurred.

On Friday, 29 April, after the package was loaded onto the airplane at the Martin Airport and properly tied down, the airplane made a demonstration flight with the package. Here again, no displacement of the package occurred.

The loading demonstration was completed during the quarter, and no further efforts will be scheduled.

Fig. X-4 shows the test package being drawn into the C-130A aircraft.

Fig. X-5 shows the package being tied down to the aircraft.

Fig. X-6 shows the aircraft taking off with the loaded package.

Fig. X-7 shows the package being withdrawn from the aircraft.

Fig. X-4. PM-1 Test Package Being Loaded into C-130A Aircraft

Fig. X-5. PM-1 Test Package Tied Down in C-130A Aircraft

Fig. X-6. C-130A Aircraft Taking Off at Martin Company
Airport Carrying PM-1 Test Package

Fig. X-7. PM-1 Test Package Being Withdrawn from C-130A Aircraft

XI. TASK 13--MANUALS

Project Engineers--C. Fox, F. McGinty

The objective of this task is the preparation of operating, maintenance and training publications required for the PM-1 nuclear power plant.

A. SUBTASK 13.1--MAINTENANCE MANUAL (R. Groscup)

During the quarter, effort was initiated on the preparation of the Plant Maintenance Manual. The manual format was prepared with two basic approaches outlined for internal review. Also completed were a cataloguing system for collating vendor maintenance data on all major components in the plant and an overall schedule establishing vendor submittal dates on maintenance requirements. Meetings were held with members of the Franklin Company to discuss their approach on rewriting of the SM-1 Maintenance Manual.

During the next quarter, vendor information will be tabulated and reviewed for content, the final outline of the manual will be completed and submitted, and preparation of the manual will begin.

B. SUBTASK 13.2--OPERATING MANUAL (O.R. Millman, F. Holliday)

During the quarter, it was planned to continue the efforts on the Plant Operating Manual form and contents as set forth in the 4th Quarterly Progress Report. A letter outlining the procedures as to (1) table of contents, (2) chapter outline and (3) sample chapter writeup were submitted to the AEC for review and were approved.

The chart entitled "PM-1 Operating Manual Status" (Table XI-1) gives a graphic picture of what has been completed in this quarter. The darkened bars denote work accomplished to date. The diagonal hatching represents anticipated accomplishments for the next quarter. This bar chart will be repeated each quarter to denote progress.

TABLE XI-1
PM-1 Operating Manual Status

Chapter	First Draft			Second Draft			First Issue to AEC
	Started	Circulated	Reviewed	Revised	Circulated	Reviewed	
1. General Description of Plant							
2. Nuclear Instrument System							
3. Reactor Rod Control System							
4. Reactor Safety System							
5. Reactor Coolant System							
6. Pressurizer and Pressure Relief System							
7. Coolant Purification System							
8. Coolant Charging System							
9. Coolant Discharge and Vent System							
10. Coolant Chemical Addition System							
11. Shield Water System							
12. Decay Heat Revival System							
13. Radioactive Waste Disposal System							
14. Reactor Plant Containers Heating and Cooling System							
15. Radiation Monitoring System							
16. Main and Auxiliary Steam System							
17. Main Turbine and Generator Unit							
18. Main Condenser and Condensate System							
19. Feedwater System							
20. Extraction Steam and Heater Drain System							
21. Turbine Exhaust System							
22. Secondary Sampling Cooling Water System							
23. Water Treating System							
24. Condensate Mockup System							
25. Instrument Air System							
26. Main Station Transformer and Distribution System							
27. Station Service System							
28. Plant D-C System							
29. Lighting and D-C Emergency Light System							
30. Emergency Power System							
31. Fire Protection Equipment							
32. Building Heating, Ventilating and Air Compressing System							
33. Primary and Secondary Plant Fill and Drain Procedure							
34. Startup Operation							
35. Running Operation							
36. Shutdown Operation							
37. Emergency Operation							
38. Check Lists							
39. Glossary of Terms, Abbreviations and Symbols							

KEY:

———— Accomplished

//////// Planned for Next Quarter

C. SUBTASK 13.2--TRAINING MANUALS (F. McGinty)

The objective of this subtask is to prepare a manual designed for instructor and training administrator use. It will contain a syllabus of instruction, training chart, course outlines and detailed lesson outlines.

The work planned for this reporting period was limited to the preparation of portions of the manual's introductory material.

The planned work was accomplished.

During the next quarter, work will be initiated on developing detailed lesson outline materials for the instrumentation portion of the course.

XII. TASK 14--TRAINING

Project Engineer--F. C. McGinty

The objectives of this task are to develop and implement a program to train military personnel to supervise and conduct operations and maintenance of the PM-1 nuclear power plant.

A. SUBTASK 14.1--TRAINING DEVELOPMENT

The work planned for this quarter included:

- (1) Preparation of material for the instructor development course.
- (2) Revision of the Training Plan to reflect the USAEC comments submitted last quarter with conditional approval of the Training Plan.

During the next quarter, work is planned to be accomplished in:

- (1) Instructor development course lesson plan preparation.
- (2) Coordination of the Training requirements of the Technical Manuals with the technical writers.

A Plant Information Course Outline revision was submitted to the USAEC and approval was granted.

The conditionally approved Training Plan was revised to reflect USAEC review comments. The plan was resubmitted and approved.

An Instructor Development Program Course outline draft was prepared and lesson planning efforts were initiated.

XIII. TASK 15--PROJECT SERVICES

Project Engineer--C. Fox

A. SUBTASK 15.1--FILM AND PHOTOGRAPHS

The following planned efforts were accomplished during the quarter:

- (1) Film coverage was obtained on the PM-1 loading demonstration, package tests and core zero-power testing.
- (2) Progress photographs were obtained of the PM-1 package tests, the loading demonstration, the primary system superstructure, the fuel element bundle flow tests and fuel element work. Several of the more interesting photographs are presented with their respective subtasks.

B. SUBTASK 15.2--MODELS

In accordance with the planned objective during the fifth program quarter, the first model was delivered on 7 March 1960 for display and presentation purposes. It was returned to The Martin Company, shortly after, for minor repairs and incorporation of plant design modifications. At this time, it was packaged into its carrying case and delivered on 28 March 1960.

Construction efforts continued during the quarter on the remainder of the display models with work directed principally toward completion of model bases and on the fabrication of details. Assembly of details was started in the primary system.

Using the first model as the standard for future models, a new work plan and completion schedule was established. Construction of the remaining models will contain the following details, not originally required in the model specifications:

- (1) Construction of modules of the plant which can be removed and handled.
- (2) Installation of lighting effects throughout the entire display.
- (3) Incorporation of final plant design features.

Efforts will continue during the coming period of the remaining display models, with a scheduled completion date of 5 August 1960 for the second model, and completion of the remainder in sequence to an end date of 16 September 1960.

Several views of the first model are shown in Figs. XIII-1 through XIII-3.

Fig. XIII-1. PM-1 Model

Fig. XIII-2. PM-1 Model

Fig. XIII-3. PM-1 Model

XIV. TASK 16--CONSULTING

The purpose of this task is to secure technical consulting services as required for the PM-1 project.

Gibbs and Hill, Incorporated, provided support in the following areas during this quarter:

- (1) Review of the switchgear, motor control center and turbine-generator work performed by Westinghouse.
- (2) Review of the condenser model test.
- (3) Preparation of appendices for the piping and insulation specifications, which included instructions to the vendor for preparing bids for the secondary system.

The work accomplished by Gibbs and Hill in the PM-1 steam-electric system area during this quarter was:

- (1) Continued incorporation of vendor data into the final design of the secondary steam-electric system.

During the next quarter, the efforts of Gibbs and Hill will be to:

- (1) Continue final design of the secondary steam-electric system through incorporation of vendor data.
- (2) Provide consulting support as required.

XV. TASK 17--REPORTS

The objective of this task is to accomplish the timely preparation of reports required by the USAEC.

A. SUBTASK 17.1--PM-1 HAZARDS SUMMARY REPORT

1. Work Accomplished

During this quarter, no activities were planned on the PM-1 Hazards Summary Report.

2. Anticipated Accomplishments Next Quarter

During the next quarter, preparation of an amendment presenting the details of the final design of the PM-1 Plant will be initiated. In particular, any design changes occurring during the final design will be called out to up-date the existing hazards report, since the existing hazards report was based on the preliminary design.

B. SUBTASK 17.2--REPORTS OTHER THAN HAZARDS

This subtask includes all reports submitted to the USAEC except those on Hazards.

During the fifth project quarter:

- (1) The Fourth Quarterly Progress Report was completed and delivered to the AEC (MND-M-1815).
- (2) Subsystem submittals for the final design were essentially completed except for revision necessary due to AEC comments.

During the next project quarter, the revisions to the final design submittals will be completed, the Fifth Quarterly Progress Report will be prepared and distributed and a topical report on Subtasks 1.1 and 1.3 prepared and distributed.