no stock

APDA - 308

110
8/20
Fast Reactor
PLR

MASTER

APDA

DEVELOPMENT OF THE PRIMARY SYSTEM FOR THE ENRICO FERMI ATOMIC POWER PLANT

United States Atomic Energy Commission
Contract No. AT (11-1) - 865
Project Agreement No. 15

N. T. Peters

June 1968

ATOMIC POWER
DEVELOPMENT ASSOCIATES, INC.

DISCLAIMER

DEVELOPMENT OF THE PRIMARY SYSTEM FOR THE ENRICO FERMI ATOMIC POWER PLANT

United States Atomic Energy Commission
Contract No. AT (11-1) - 865
Project Agreement No. 15

N. T. Peters

Approved E. C. Kovacic

E. C. Kovacic
Senior Project Engineer
APDA

Approved James G. Duffy

J. G. Duffy
Head, Engineering Division
APDA

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

June 1968

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

ABSTRACT

This report documents the course of development of the primary heat transport system of the Enrico Fermi reactor, starting with the early history of the objectives and organization of the Fermi Project and continuing through the evolution of the system design concepts to the final design. The construction, testing, and operation of the system with reactor power levels up to 100 Mwt are also discussed. An evaluation of the general performance and important features and characteristics of the system, including recommendations for use in the future design of new liquid-metal cooled fast breeder reactor power plants, is presented.

The interaction of the primary system design with the design of other features of the plant, such as the reactor containment building, is also documented. System components and auxiliary features are described only to the extent that they are relevant to the discussion of the design of the primary system as an integrated, functional unit; they, therefore, do not receive comprehensive coverage in this report.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

TABLE OF CONTENTS

	Page
LIST OF ILLUSTRATIONS	ix
LIST OF TABLES	xiii
I. PREFACE	1
II. INTRODUCTION	3
A. 1951 STUDY AGREEMENT AND REPORT	3
1. Study Agreement	3
2. 1951 Report	3
B. 1953 AND 1954 ORGANIZATION AND OBJECTIVES	4
1. Organization	4
2. Objectives of Project in 1953	4
3. Organization and Intermediate Goals in 1954	5
C. FORMATION OF ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. IN 1955	5
D. FORMATION OF POWER REACTOR DEVELOPMENT COMPANY	6
III. FACTORS AND PRELIMINARY STUDIES THAT INFLUENCED THE CONCEPTUAL DESIGN OF THE PRIMARY SYSTEM 1951 - 1955	7
A. POWER PLANT OUTPUT AND EFFICIENCY	7
B. SPECIAL DESIGN CONSIDERATIONS	7
C. CORE DESIGN	8
1. Change from Liquid to Solid Fuel	8
2. Specific Power and Power Density	8
3. Core Outlet Temperature	8
D. COOLANT SELECTION	8
E. INTERMEDIATE LOOP CONCEPT	10
F. PRIMARY SYSTEM DESIGN, 1953-1954	10
1. Design of Cast-Segment Core	10
2. Design Limitations to Heat Transfer with Cast-Segment Core	10
3. General Arrangement of Core and Blanket	12
4. Coolant Flow Arrangement	15
5. Primary System Components and Arrangements	15

TABLE OF CONTENTS (Continued)

Page

G. 1955 PRIMARY SYSTEM DESIGN 21
1. Significant Changes 21
2. Core Design 26
3. Reactor Design 27
4. Primary System Arrangement 27
5. Components 31
6. Primary System Materials 36
7. Heating and Insulation 36
8. Emergency Cooling Provisions 36
9. Primary System Control 37
10. Auxiliary Systems 37
H. PERFORMANCE SPECIFICATIONS 38

IV. DESCRIPTION OF THE FINAL DESIGN 39
A. SYSTEM PERFORMANCE AND OPERATING CONDITIONS. 39
B. CORE AND BLANKET DESIGN 41
C. HEAT TRANSPORT SYSTEM 41
D. COMPONENTS 47
1. Reactor Vessel 47
2. Intermediate Heat Exchangers 54
3. Primary Sodium Pumps 56
4. Valves. 62
5. Primary Piping System Arrangement 65
6. Component Supports 69
7. Secondary Containment 71
8. Insulation and Heating 71
9. Shielding 75
E. EMERGENCY COOLING SYSTEMS 75
1. Pony Motors 75
2. Reserve Sodium Supply System 77
F. DESIGN FOR LOSS-OF-COOLANT ACCIDENT 80
G. PRIMARY SYSTEM CONTROL 81
H. SERVICE SYSTEMS 84
1. Sodium Service 84
2. Inert Gas Service 84

V. EFFECT OF PRIMARY SYSTEM DESIGN ON THE CONTAINMENT BUILDING AND ASSOCIATED SYSTEMS 89
A. EFFECT OF PLANT LAYOUT ON REACTOR BUILDING DESIGN 89

TABLE OF CONTENTS (Continued)

	Page
1. Building Diameter	89
2. Building Geometry, Cylinder Versus Sphere	90
3. Building Height	90
4. Design Pressure of Reactor Building	90
5. Equipment Door in Building	91
B. BUILDING VENTILATING AND COOLING SYSTEMS	91
1. Function of Ventilating and Cooling Facilities	91
2. Lower Chamber Facilities	93
3. Upper Chamber Facilities	96
4. Atmospheres in Containment Building	97
C. DESIGN OF SHIELD SYSTEM	98
1. General Considerations	98
2. Primary Shield	100
3. Secondary Shield	102
4. Biological Shield	102
5. Sodium Pipe Shielding	102
6. Pump and IHX Shield Plugs	103
D. SECONDARY SYSTEM DESIGN	103
VI. FABRICATION, ERECTION, AND TESTING	105
A. ERECTION OF TEST FACILITY	105
1. Description of Test Facility	105
2. Test Site Selection	105
3. Erection Sequence Planning	106
B. MATERIALS SPECIFICATIONS	106
C. REACTOR VESSEL	108
D. PRIMARY SYSTEM PIPING	111
1. Testing Erected Sodium Piping System	115
2. Testing Secondary Containment	116
3. Pipe Shielding	116
4. Insulation and Heating	116
E. INTERMEDIATE HEAT EXCHANGERS	117
F. PRIMARY SODIUM PUMPS	117
G. VALVES	120
1. Check Valves	120
2. Throttle Valves	121
VII. OPERATION OF THE PRIMARY SYSTEM	125
A. NONNUCLEAR OPERATION	125
1. Reactor Components Test	125
2. Preparation for Nuclear Operation	137

TABLE OF CONTENTS (Continued)

			Page
	B.	NUCLEAR OPERATION	139
		1. Replacement of Check Valves	139
		2. Intermediate Heat Exchanger Performance	140
		3. Primary Pump Performance	142
		4. Primary System Carburization Program	142
		5. Summary of High-Power Operation, >1 Mwt	142
VIII.		EVALUATION	147
	A.	GENERAL PERFORMANCE	147
	B.	SYSTEM CONFIGURATION	147
	C.	SECONDARY CONTAINMENT	148
	D.	SYSTEM SUPPORT AND FLEXIBILITY	148
	E.	ACCESSIBILITY AND MAINTENANCE FEATURES	149
	F.	INSTRUMENTATION AND HEATERS	150
	G.	PRIMARY SODIUM SERVICE SYSTEM	150
	H.	INERT GAS SYSTEM	151
	I.	SYSTEM CONTROL	151
REFERENCES			153
APPENDIX:		MISCELLANEOUS STUDIES ASSOCIATED WITH THE PRIMARY SYSTEM	A.1
	1.	CYCLE STUDIES	A.1
	2.	STEAM GENERATOR STUDIES	A.1
	3.	SIMULATOR PROGRAMS	A.2
	4.	TRANSIENT STUDIES	A.2
	5.	DECAY HEAT STUDIES	A.4
	6.	EMERGENCY COOLING	A.6
	7.	NATURAL CIRCULATION STUDIES	A.6
	8.	CAPABILITY OF PONY MOTOR DESIGN	A.10
	9.	DECAY HEAT TEST, 1964	A.10
	10.	SETTING LOSS TESTS, 1966	A.11
	11.	STRESS ANALYSIS OF PRIMARY SYSTEM	A.11
	12.	STUDIES OF PIPING AND CONTAINMENT SUPPORT	A.12
	13.	SYSTEM PRESSURE DROP STUDIES	A.15
	14.	CAPABILITY OF HANDLING 150% OF 300-MWT FLOW	A.17
	15.	PLANT CONTROL STUDIES	A.19
	16.	REMOTE MAINTENANCE STUDY	A.19
	17.	CONCRETE POURING HAZARD STUDIES	A.19

LIST OF ILLUSTRATIONS

Figure		Page
1	View of Cast-Segment Fuel Element	11
2	Reactor Arrangement for Cast-Segment Core, a 1955 Concept	13
3	Alternate Arrangements of Sodium Flow Path Through Reactor Core, a 1953-54 Concept	14
4	Schematic of a 1953-54 Concept of Heat Transport and Steam Power Systems	16
5	Plan View and Cross Section of Reactor, Process Chamber, and Handling Device, a 1953-54 Concept	17
6	Proposed Arrangement for Nuclear Power Plant, a 1953-54 Concept	18
7	Design of Mechanical Handling System, a 1953-54 Concept	19
8	Plant Arrangement Incorporating Separate Fuel Processing Facility and Vertical Piping Layout, a 1953-54 Concept	20
9	Intermediate Heat Exchangers, a 1953-54 Concept	22
10	A 1953-54 Concept of a Plant Elevation	23
11	Primary System with Component Tanks, a 1955 Concept	24
12	Reactor Arrangement, a 1955 Concept	25
13	Perspective View of Reactor, a 1955 Concept	28
14	Reactor Vessel Elevation, a 1955 Concept	29
15	Schematic of 1955 Concept of Liquid Metal and Steam Power Systems	30
16	A 1955 Concept of an Alternate Plant Arrangement Using Horizontal Piping	32
17	Plan View of Plant Arrangement Using Horizontal Piping, a 1955 Concept	33

LIST OF ILLUSTRATIONS (Continued)

Figure		Page
18	Intermediate Heat Exchanger, A 1955 Concept	34
19	Sodium Pump Assembly, A 1955 Concept	35
20	Core Subassembly	42
21	Reactor Cross Section	43
22	Radial Blanket Subassembly	44
23	Flow Diagram of the Heat Transport System	45
24	Perspective View of Reactor	52
25	Reactor Vessel Elevation	53
26	Intermediate Heat Exchanger, Initial Design	55
27	Intermediate Heat Exchanger, Second Design	57
28	Intermediate Heat Exchanger, Final Design	58
29	Primary Sodium Pump, Reference Design	60
30	Primary Sodium Pump, Final Design	61
31	Average Full-Speed Characteristics of the Primary Sodium Pumps	63
32	Pressure Drops for the Primary Sodium Pumps	64
33	Throttle Valve	66
34	Elevation of Primary System	67
35	Plan View of Primary System	68
36	Plan View of Reactor Showing Vessel Supports	70
37	Schematic of Primary Sodium System Elevation	72
38	Typical Liquid-Metal Pipe Section Showing Secondary Containment	73

LIST OF ILLUSTRATIONS (Continued)

Figure		Page
39	Shielding on 30-Inch Primary Coolant Pipe	74
40	Reserve Sodium Supply System	78
41	Operating Instrumentation, Reserve Sodium Supply System	79
42	Schematic of Operating Control System	82
43	Tentative Schedule of Operating Conditions	83
44	Flow Diagram of Primary Sodium Service System	85
45	Primary Inert Gas and Recirculating Gas Systems	86
46	Inert Gas Flowsheet for Primary Sodium System	87
47	Reactor Building Ventilation	92
48	Plan View of Shield System	99
49	Primary Shield System	101
50	Erection Sequence Studies	107
51	Lower Section of Primary Shield Tank	109
52	Reactor Vessel in Place in Reactor Building	110
53	Primary Piping Assemblies	112
54	Plan View of Primary Piping	113
55	Elevation View of Primary Piping	114
56	Intermediate Heat Exchanger Tube Bundle	118
57	Sodium Pump Internals	119
58	Throttle Valve Assembly	122
59	Flow Characteristics of Throttle Valve	123

LIST OF ILLUSTRATIONS (Continued)

Figure		Page
60	Reactor Components Test Facility	126
61	Schedule of Major Construction, Modification and Test Operations	127
62	Sodium Flow Paths for Filling Primary System	129
63	Sodium Flow Path for Primary System Cleanup	130
64	Reactor Vessel Temperature During 1000 F Test	132
65	Primary System Characteristics Curve, No. 3 Throttle Valve Open	135
66	No. 1 Primary Pump Flow Decay from 700 rpm, 500 F Sodium	136
67	Check Valve Designs	138
68	Performance Curves of Old and New Check Valves	141
69	Fermi Operating History, 1966	143
70	Pump Flow Decay	145
A.1	Design Valves for the Decay Heating for Central Core Subassemblies	A.5
A.2	Recommended Design Valves for Decay Heating for Central Core Subassembly in Core A	A.7
A.3	Decay Heat per Subassembly per Mw Operation	A.8
A.4	Piping and Containment Support	A.13
A.5	Piping and Containment Support	A.14

LIST OF TABLES

Table		Page
1	Coolant Properties	9
2	Performance Specifications	38
3	Projected Plant Performance	40
4	Plant Design and Performance Data	46
5	Primary Sodium System Pressure Drops	48
6	Thermal Transient Design Conditions for Reactor Vessel	49
7	Thermal Transient Design Conditions for Intermediate Heat Exchangers	50
8	Thermal Transient Design Conditions for Primary Sodium Pump	51
9	Design Data, Intermediate Heat Exchanger	59
10	Performance Specifications at 430-Mwt Conditions	59
11	Design Data, Primary Sodium Pump	62
12	Design Data for Lower Chamber Cooling	94
13	Distribution of Upper Chamber Cooling Loads	96
14	Design Data for Upper Chamber Ventilation System	96
15	Final Cold Spring Gaps	115
16	Operating Characteristics of No. 1 Primary Pump and Calibration Data for 14-Inch and 6-Inch Flow Meters, Loop 1	134
17	Sodium Level Differential Between Reactor and Pump Tank	137
18	IHX Performance During Heat Balance Tests	140
19	History of Pump Operation	142

LIST OF TABLES (Continued)

Table		Page
A. 1	Natural Circulation Pressure Drops	A. 9
A. 2	System Pressure Drops	A. 16
A. 3	IHX Performance, 150% Flow	A. 18

I. PREFACE

A. SCOPE

This report is one of a number of reports being written under AEC Contract No. AT(11-1)-865, Project Agreement No. 15, and under the sponsorship of the Division of Reactor Development and Technology. The object of these reports is to make available to the designers and engineers of future reactors the experience that has been gained from the Enrico Fermi Fast Breeder Reactor Project and to provide guidance in the planning of research and development programs on fast reactor and sodium-cooled reactor technology. Each of the reports deals with a specific component, system, or technical aspect of the Fermi reactor project and is written in such a manner that the treatment of the material is sufficiently complete in itself that reference to other documents is not essential. Each of the reports includes a general description of the reactor plant, together with sufficient detailed description to provide an adequate understanding of the specific component, system, or technical aspect that is the subject of the report. The technical content of the reports generally is divided into three principal categories; namely, (1) a review of the evolution from the earliest concepts to the final plant design, (2) a summary of the experience including discussions of problem areas as well as meritorious performance and also of modifications, and (3) an evaluation of the system.

B. RELATIONSHIP TO OTHER PROJECTS

This project is related generally to the AEC's program for the development of fast reactor technology.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

II. INTRODUCTION

A. 1951 STUDY AGREEMENT AND REPORT

1. Study Agreement

In April 1951, The Dow Chemical Co. and The Detroit Edison Co. signed an agreement with the United States Atomic Energy Commission (USAEC) to undertake a study of the practicability of using nuclear power for industrial purposes, with the main objective of creating a large-scale reactor to produce power and fissionable materials as joint products.

The purpose and scope of the study as stated in the agreement was

- To determine the engineering feasibility of designing, constructing, and operating a materials and power producing reactor
- To examine the economic and technical aspects of building this reactor in the next few years
- To determine the research and development work needed, if any, before such a reactor project can be undertaken
- To offer recommendations in a report to the Commission concerning such a reactor project and industry's role in undertaking and carrying it out.

2. 1951 Report

In the first report of this Project, submitted to the AEC in December 1951,[1] seven requirements for a reactor were enumerated and discussed. These requirements were determined as the result of an analysis made to resolve the fundamental principles and criteria most likely to result in an economic nuclear reactor to produce electric power and by-product materials. It was stated that an economic power reactor should

- Be operated at high temperatures
- Be a breeder reactor
- Have maximum breeding gain
- Have fuel in mobile or liquid form
- Have fuel that will lend itself to rapid and low-cost processing
- Require minimum exclusion area
- Be capable of inherent self-control.

These points appeared to indicate that the most desirable ultimate reactor for power and/or by-products is a high-temperature liquid-metal cooled fast breeder reactor with the fuel in liquid form arranged for removal and replenishment in a rapid cycle.

B. 1953 AND 1954 ORGANIZATION AND OBJECTIVES

1. Organization

The Babcock & Wilcox Co. became an associate in the study and Nuclear Development Associates, Inc., was engaged to provide nuclear consulting services to the Project.

Soon after acceptance of the joint development program in April 1952 by the AEC, plans were made to organize the Project on a more effective basis which would be commensurate with the work required. Recognizing the magnitude of the effort and the need to proceed rapidly on as broad a front as possible, steps were taken to interest other industrial firms who would be willing to contribute to the project.

In September 1952, a request was made of the AEC to allow 11 companies to become associated with the project, which would be known as the Dow Chemical-Detroit Edison Nuclear Power Development Project. This association was approved by the AEC on October 16, 1952. On April 24, 1953, the AEC approved further enlargement of the Project by the addition of 12 companies, bringing the total to 26. Included were 18 privately owned electric utilities, 4 manufacturing industries, 1 chemical company, and 3 engineering and construction firms.

A Nuclear Power Development Department was organized in the General Offices of The Detroit Edison Co., at Detroit, Michigan, in which classified facilities were provided to carry out the development studies.

2. Objectives of Project in 1953

When the Atomic Energy Commission stated in 1952 that there would be no guaranteed government market for fissionable material, study groups were urged to direct their efforts towards an unsubsidized power reactor. Fissionable material had to carry its own weight in the open market as a reactor fuel. This event did not affect this project since very early in the study it was decided that this was the only basis on which private atomic power industry could exist.

The main objective of the Project was made clear in the first statement of a letter dated October 20, 1953, to AEC Chairman Strauss, signed by Mark E. Putnam and Walker L. Cisler:

"The main objective of the Dow Chemical-Detroit Edison Nuclear Power Project is the development of a new source of heat energy, that is, nuclear fuels, to compete commercially with conventional fuels. Our specific interests are the release and utilization of heat from the fission process for the economical production of electric power, and the production of a high-grade by-product fuel. Other by-products of the fission process, such as the fission products, would be utilized and marketed for the maximum use and value which can be developed for them."

3. Organization and Immediate Goals in 1954

On April 14, 1954, an agreement was reached with the AEC setting forth the conditions under which the Project was to carry on its program. The program, supported by an effective organization of 26 companies comprising Dow Chemical-Detroit Edison and Associates, included research development and engineering to obtain the data needed for technical and economic analysis of the reference design reactor.

The immediate goals of the Project were:

- To determine the engineering feasibility of the reference design by resolving certain major problems such as

  Developing a satisfactory fuel element

  Developing a rapid, economic fuel processing system

  Developing a plant design suitable for location in a populated area

- To obtain an approximate cost and economic analysis for a nuclear power plant utilizing a reactor of the reference design.

C. FORMATION OF ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. IN 1955

On March 10, 1955, a certificate of incorporation was filed to establish a nonprofit membership corporation known as Atomic Power Development Associates, Inc. The objective of the organization was to apply the knowledge and experience of its members and associates to develop atomic power into a commercially practicable means of electric power generation. The corporation consisted of a group of 25 electric power systems, 4 manufacturing enterprises, and 4 engineering organizations.

D. FORMATION OF POWER REACTOR DEVELOPMENT COMPANY

In August 1955, a nonprofit membership corporation consisting of 21 industrial and utility companies was established, Power Reactor Development Company, to build and operate the nuclear portion of the atomic power plant.

III. FACTORS AND PRELIMINARY STUDIES THAT INFLUENCED THE CONCEPTUAL DESIGN OF THE PRIMARY SYSTEM, 1951-1955

The developmental factors affecting the primary system concepts for the Enrico Fermi Project are outlined in detail in References 2 through 6; the following sections summarize these factors.

A. POWER PLANT OUTPUT AND EFFICIENCY

At the time of the conceptual design of the primary system, power plant efficiencies were in the 30 to 35% range. With the original decision[2] that the plant would have a net electric capacity of 150 Mw, the thermal output of the core was set at 500 Mw. A later decision[4] established the electric capacity at 100 Mw and the thermal output at 300 Mw.

B. SPECIAL DESIGN CONSIDERATIONS

Some of the more important factors influencing nuclear plant design were (1) the high specific performance of the reactor, (2) the fuel processing and fabricating system, (3) the radioactivity, and (4) reactor hazards control. With these factors in mind, studies of plant arrangement included at least the following features:

- Liquid sodium as the reactor coolant.
- Reactor core and breeder blanket divided into subassemblies suitable for remote handling and processing in an integrated processing system.
- Use of sodium-potassium (NaK) alloy as a nonradioactive intermediate heat transport fluid between the sodium and the boiler water to separate the radioactivity and the sodium-water reaction hazards. This also eliminates the possibility of accidentally introducing hydrogen into the reactor core which could induce a serious nuclear accident.
- Location of the reactor and the other radioactive systems partially or wholly below grade to facilitate the shielding and containment problems.
- Multiple coolant loops and heat exchangers to improve reliability.
- Separate compartments for each intermediate heat exchanger and steam generator to isolate the effects of a failure in any unit.

C. CORE DESIGN

1. Change from Liquid to Solid Fuel

As stated in Reference 1, the most desirable ultimate reactor included fuel in liquid form. From subsequent studies, however, it appeared to be a very long range developmental effort to build a liquid-fueled fast reactor since available fuel materials had melting points which were too high to permit liquid operation with the available container materials. A revised design formulated in June 1952, had fuel in solid rather than liquid form.

2. Specific Power and Power Density

One of the objectives of core design was to develop an arrangement of fuel and coolant which would permit the reactor to be operated at a power density which would provide the maximum power output from each kilogram of fuel. The cast-fuel concept, described in Reference 2 listed an approximate specific power of 1400 kw/kg.

3. Core Outlet Temperatures

To achieve the most favorable plant efficiencies, the highest possible steam temperatures were sought at the steam generator outlet. This fact spurred a constant drive to attain the highest possible core outlet temperature. Four important factors involved in achieving this goal were

- The maximum allowable fuel temperature
- Resistance to flow of heat between fuel and coolant
- Heat removal capacity of coolant flow
- Power density.

D. COOLANT SELECTION

The choice of a coolant for a fast neutron power reactor included a search for certain desirable properties: heat transfer, nuclear, chemical, and handling. Table 1 lists some of the properties of the best available coolants. It is readily seen that sodium, although not the most desirable in every respect, had the best thermal properties and no objectionable nuclear properties.

There was some experience with operation of a radioactive NaK system in Experimental Breeder Reactor I (EBR-I), that had been in service since 1951. For the Fermi Project, liquid sodium was selected for the primary system because experience in the field showed that, with reasonable precautions, sodium systems could be designed, constructed, and operated safely.

TABLE 1 - COOLANT PROPERTIES[2]

Property	Sodium	Sodium-Potassium 56% Na - 44% K	Bismuth	Mercury	Potassium
1. *Relative Pumping Power	1[a]	2.2	4.9	5.2	5.7
2. Thermal Conductivity, Btu/ft-hr-F	40.3	16.0	9.0	7.3	21.8
3. Boiling Temperature, F (at one atmosphere)	1621	1518	2691	675	1400
4. Heat Capacity, Btu/lb-F	0.31	0.25	0.04	0.03	0.18
5. Melting Temperature, F	208	66	520	-38	147
6. Half-lives of important radio-isotopes formed	Na-24, 15 hr Na-22, 3 yr	See Sodium and Potassium	Bi-210, 4.8 day Po-210,** 138 day	Hg-203, 43.5 day	K-42, 12.4 hr
7. Neutron capture cross section at 0.5 Mev, barns	6×10^{-6}, Na-22 0.001, Na-24	See Sodium and Potassium	0.006	0.180	0.006

* Relative pumping power based on removal of equal amounts of heat with equal temperature rises.
** Formed by decay of Bi-210.
a. Total pumping power for primary sodium system is 4000 kw (see Ref. 2).

NaK (56% sodium - 44% potassium) was the initial selection for the secondary coolant system because of its low freezing temperature, 66 F, even though its heat transfer properties are not quite as good as those of sodium.

E. INTERMEDIATE LOOP CONCEPT

Reference 4 listed the reasons for using a primary and separate secondary liquid-metal system as follows: to avoid the hazard of chemical reaction between water and radioactive sodium in case of an internal boiler leak; to eliminate the possibility of the steam and water system becoming radioactive; to eliminate the possibility of introducing hydrogen into the reactor core which, could induce a serious nuclear accident; and to ensure containment of radioactivity in the reactor building. A NaK-water reaction in the secondary system would not rupture the primary coolant system or violate the integrity of the building.

F. PRIMARY SYSTEM DESIGN, 1953-1954

1. Design of the Cast-Segment Core

The reference design core[2] was approximately cylindrical in shape and made up of cast segments of uranium-chomium eutectic alloy. Each segment was pierced longitudinally by a large number of small tubes through which the sodium coolant flowed. A cutaway sketch of this design is shown in Figure 1. Tube dimensions were 0.145-inch ID and 0.165-inch OD with an average tube pitch of 0.195 inch. The average cross-sectional area of the core reserved for coolant flow was 50%. Ligament thickness was a design limitation; it was desirable to design the fuel element with the smallest possible fuel thickness between the coolant channels. The state of casting development indicated that a ligament, or the minimum distance between adjacent tubes, could be as small as 0.030 inch.

With concern for the possibility of a temporary power transient, the reactor was designed to accommodate a steady overload of 30%. This safety margin was referred to as a control factor of 1.3. It was felt that the control factor could be predicted and applied more accurately as more information became available.

2. Design Limitations to Heat Transfer With Cast-Segment Core

There were three design limitations that affected the cast-segment core. The first was a maximum allowable fuel temperature of 1580 F, the melting point for the uranium-chomium eutectic alloy. Temperatures higher than this would result in local melting of the fuel alloy and release of fission product gases. The second limitation was the cladding temperature, which had to be low enough to resist thermal stresses and corrosion by the coolant. The third limitation was coolant velocity which had to be high enough to remove a large amount of power with permissible temperature differences

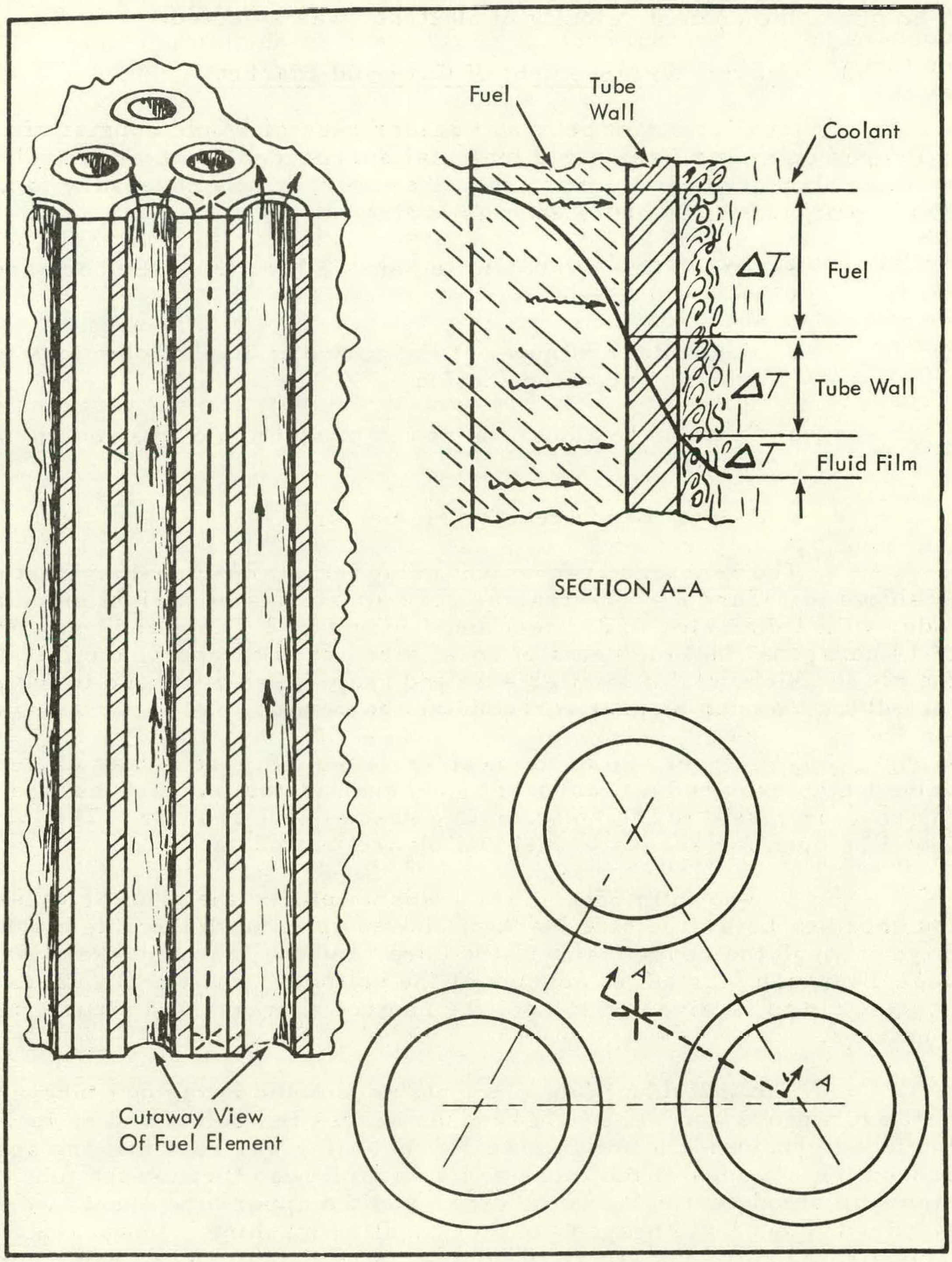


FIG. 1 VIEW OF CAST-SEGMENT FUEL ELEMENT , A 1953-54 CONCEPT

across the reactor and low enough not to require excessive pumping power. A compromise coolant velocity of 30 ft/sec was selected.

3. General Arrangement of Core and Blanket

Ideally, a fast neutron breeder reactor would consist of a spherical core containing fissionable material surrounded by an effectively infinite breeder blanket of fertile material. However, it was necessary to alter the ideal configuration to provide the following:

- Circulation of coolant to remove heat from the core and blanket at useful temperatures
- Mechanical removal of the core and blanket elements for processing and refabrication
- Controls to adjust the power production of the reactor as required
- Support of the reactor elements.

The reactor arrangement using fuel of 26-inch-long cast segments is shown in Figure 2. The reactor core approximated a right circular cylinder with a diameter of 2.4 feet and a height of 2.1 feet and was composed of 19 hexagonal fuel elements of equal size, each extending the full height of the core. Elements of similar size and shape were arranged to form a 24-inch-thick breeder blanket surrounding the core.

The upper end of the reactor vessel (Figure 2) was closed by a valve which provided a reasonably good seal against coolant leakage from the reactor vessel to the fuel handling space of the reactor. This valve would be open for access to fuel and blanket elements.

As shown in Figure 3A, coolant entered the reactor vessel through four nozzles below the side blanket, flowed up through the side blanket then down through the upper blanket, the core, and the lower blanket, leaving the vessel through four outlet nozzles at the bottom. Small percentages of the coolant had to be diverted to cool the control elements and structural members.

The individual fuel elements resembled hexagonal tube-and-shell heat exchangers and were 7-1/4 inches across the points and 30 inches long. Enriched uranium-chromium eutectic fuel alloy was cast into the space around the outsides of the tubes and sodium flowed through the tubes. The elements stood vertically in the core, and the upper tube sheets were provided with lugs that engaged the fuel handling machine. Low-carbon iron (Globeiron), stainless steel, titanium, and zirconium were being considered as container materials.

From a nuclear standpoint, it was felt desirable to have the core surrounded by a dense breeder blanket with a small coolant flow area to obtain minimum critical mass and maximum breeding gain. On the other hand, large tube diameter and low coolant velocity were considered desir-

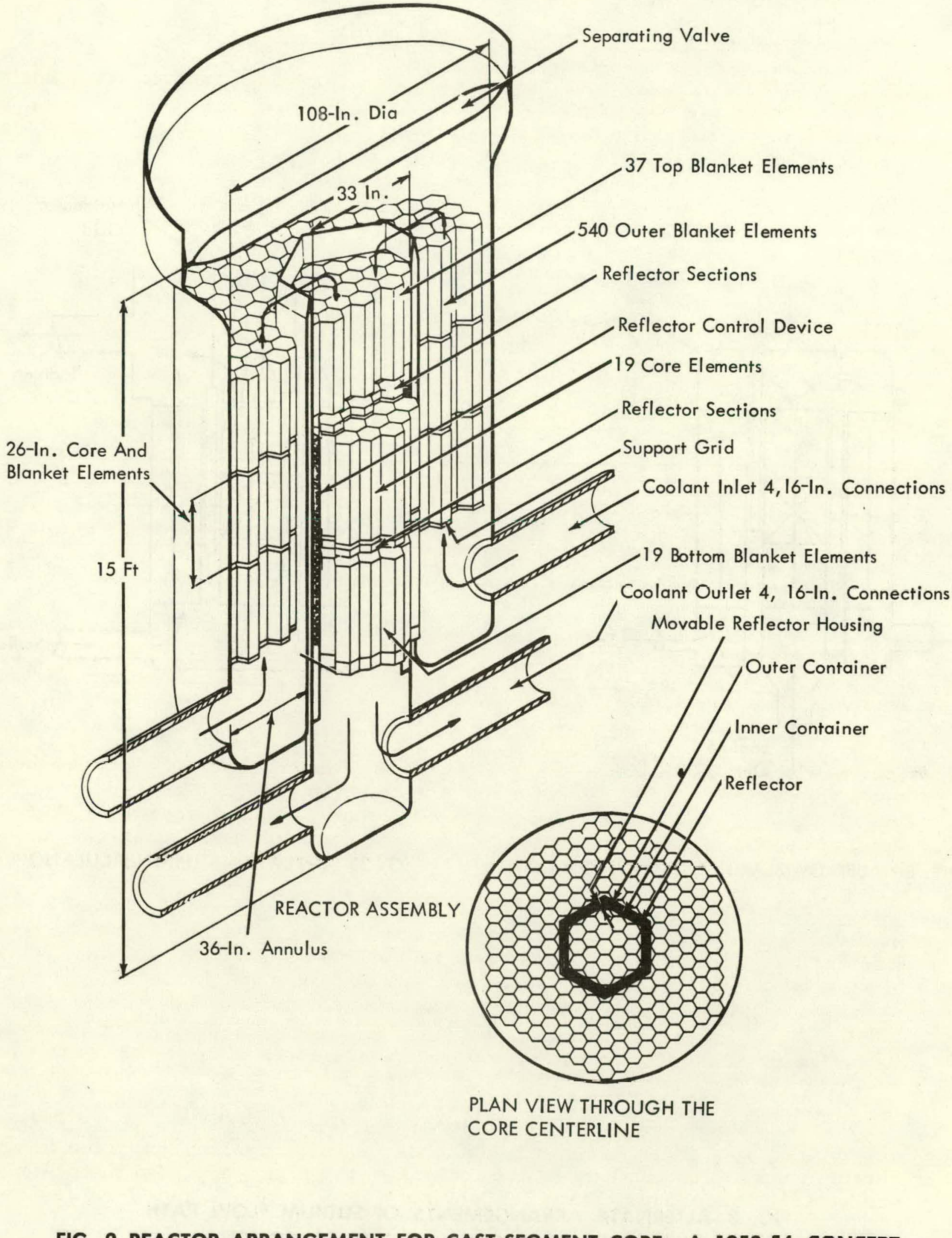


FIG. 2 REACTOR ARRANGEMENT FOR CAST-SEGMENT CORE, A 1953-54 CONCEPT

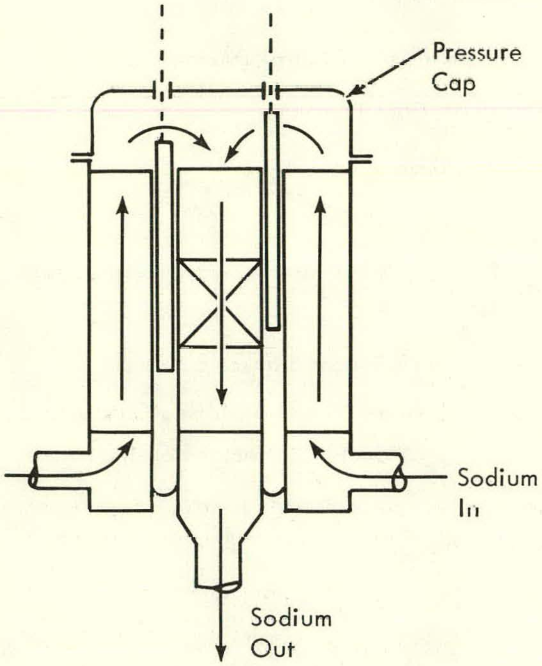


FIG. 3A UPFLOW BLANKET, DOWNFLOW CORE

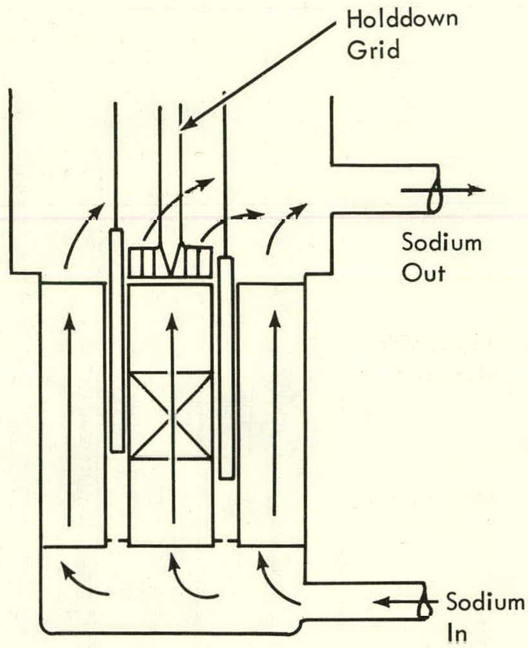


FIG. 3B UPFLOW-NATURAL CIRCULATION

FIG. 3 ALTERNATE ARRANGEMENTS OF SODIUM FLOW PATH THOUGH REACTOR CORE , A 1953-54 CONCEPT

able to keep the required pumping head to a minimum. It was also desirable to have uniform elements so that their positions in the blanket could be interchanged during their residence time in the reactor to optimize plutonium buildup in the blanket.

Based on these considerations, the breeder blanket elements were similar in size, shape, and construction to the fuel elements to facilitate handling and processing. The tubes were spaced to give the required flow area. The fertile material was uranium-chromium alloy; however, the uranium was either natural or depleted rather than enriched as in the core.

4. Coolant Flow Arrangement

The coolant flow arrangement was studied because it had considerable effect on all of the above variables. Numerous flow arrangements were conceived but the most practical and advantageous scheme was to put the entire coolant flow through the blanket before it passed to the core. Since uniform flow area was desirable, this arrangement would require the flow to be orificed to the outer rows of elements to supply the needed coolant to the inner row of elements.

Another flow arrangement which received serious consideration was that of passing the coolant through the core before it entered the blanket. This proposal had two advantages. First, the lower coolant temperature entering the reactor core would make it possible to obtain a higher coolant temperature rise through the core; this would result in a higher specific power in the core. Second, the lower heat density in the blanket would make it possible to gain a higher coolant outlet temperature without exceeding the permissible fuel hot-spot temperature. These advantages, however, appeared to be more than offset by the disadvantage of a more complicated reactor shell design and a more difficult handling problem for the elements. An additional baffle would be required at the top of the blanket which would have to be removed remotely each time the blanket elements were handled.

These flow arrangments were compared with the upflow pattern shown in Figure 3b. It was pointed out that the upflow pattern would require a holddown device for center elements, orificing of inlets to prevent temperature degradation, exposure of the handling devices to high temperatures, and operation of the control rods against sodium pressure.

5. Primary System Components and Arrangements

A diagram of the heat transport and steam power systems; a plan and elevation of the reactor, the process chamber, and fuel handling equipment; and an overall view of the reactor plant are shown in Figures 4, 5, and 6, respectively.

The decision was made to perform the fuel processing in a separate facility; Figure 7 illustrates the fuel handling system and Figure 8 the corresponding plant arrangment.

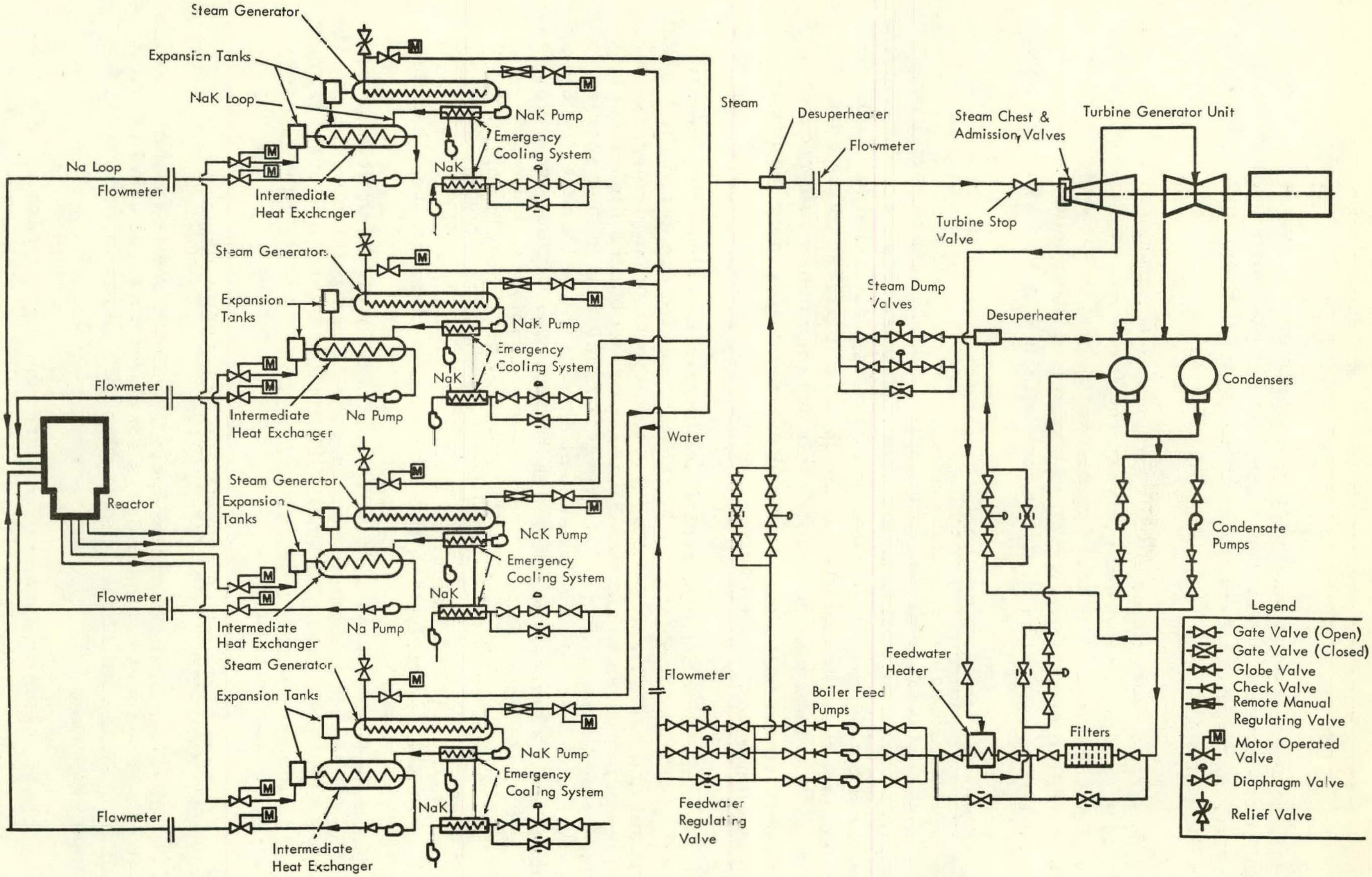


FIG. 4 SCHEMATIC OF A 1953-54 CONCEPT OF HEAT TRANSPORT AND STEAM POWER SYSTEMS

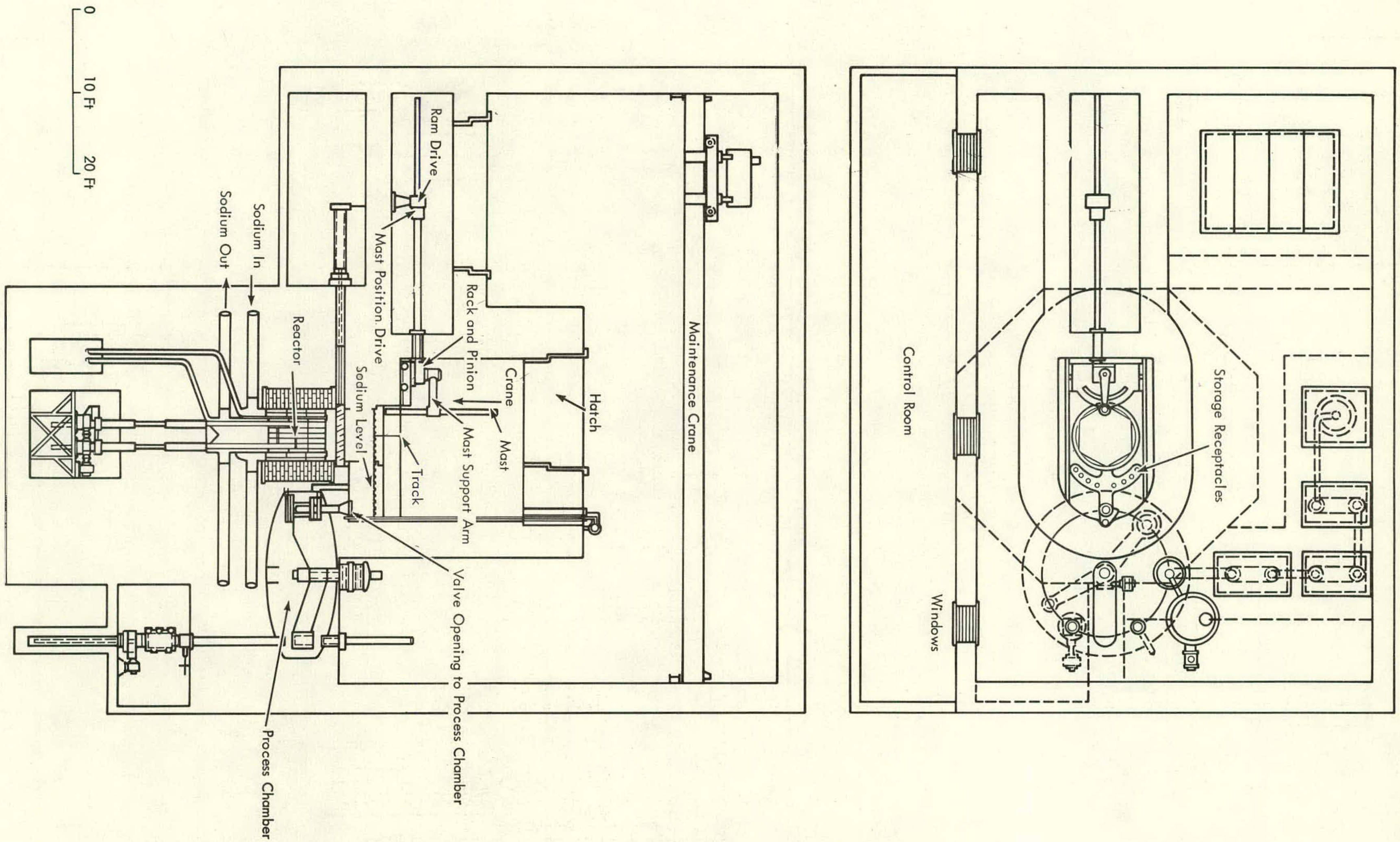


FIG. 5 PLAN VIEW AND CROSS SECTION OF REACTOR, PROCESS CHAMBER, AND HANDLING DEVICE, A 1953-54 CONCEPT

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

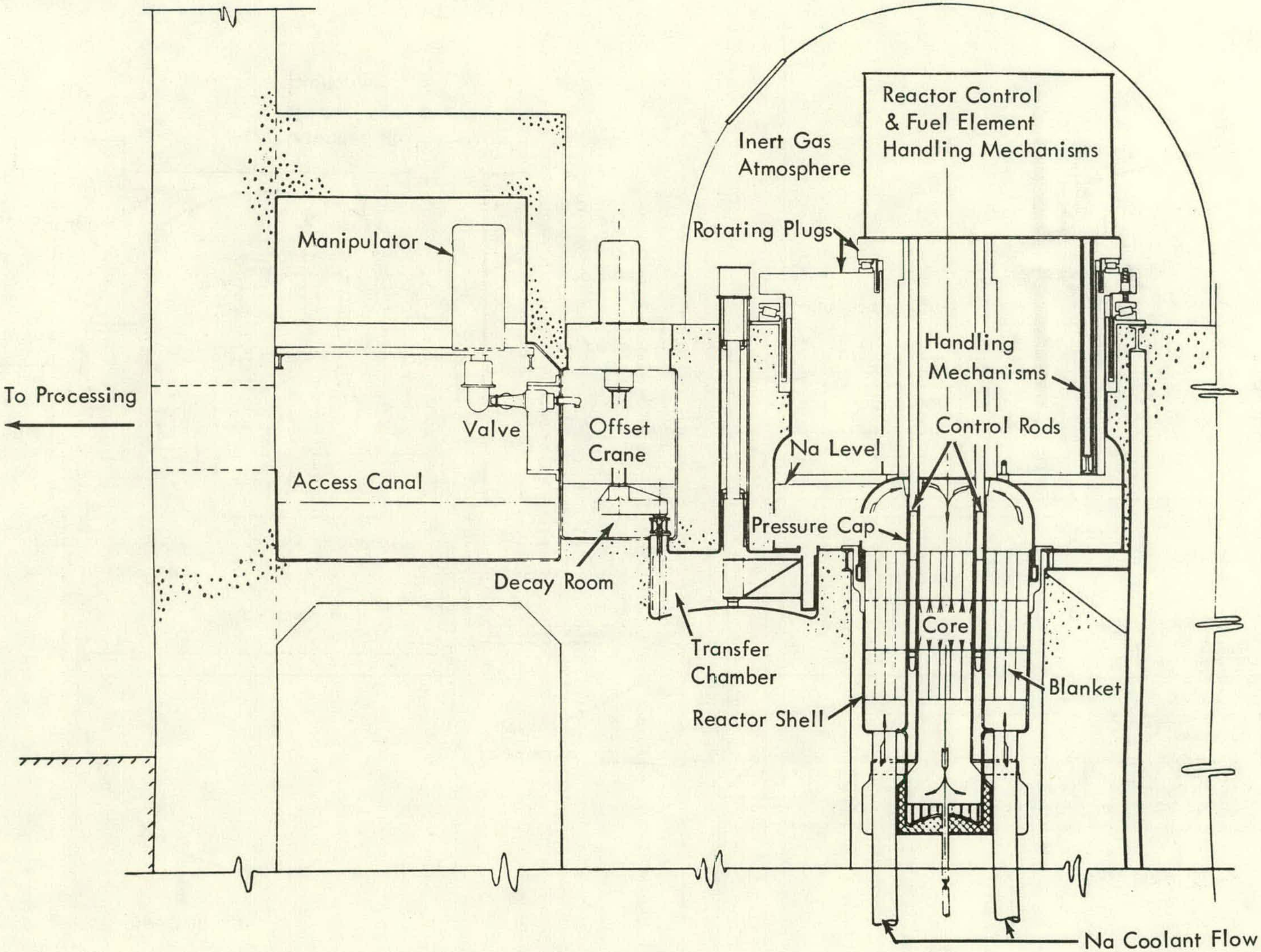


FIG. 7 DESIGN OF MECHANICAL HANDLING SYSTEM, A 1953-54 CONCEPT

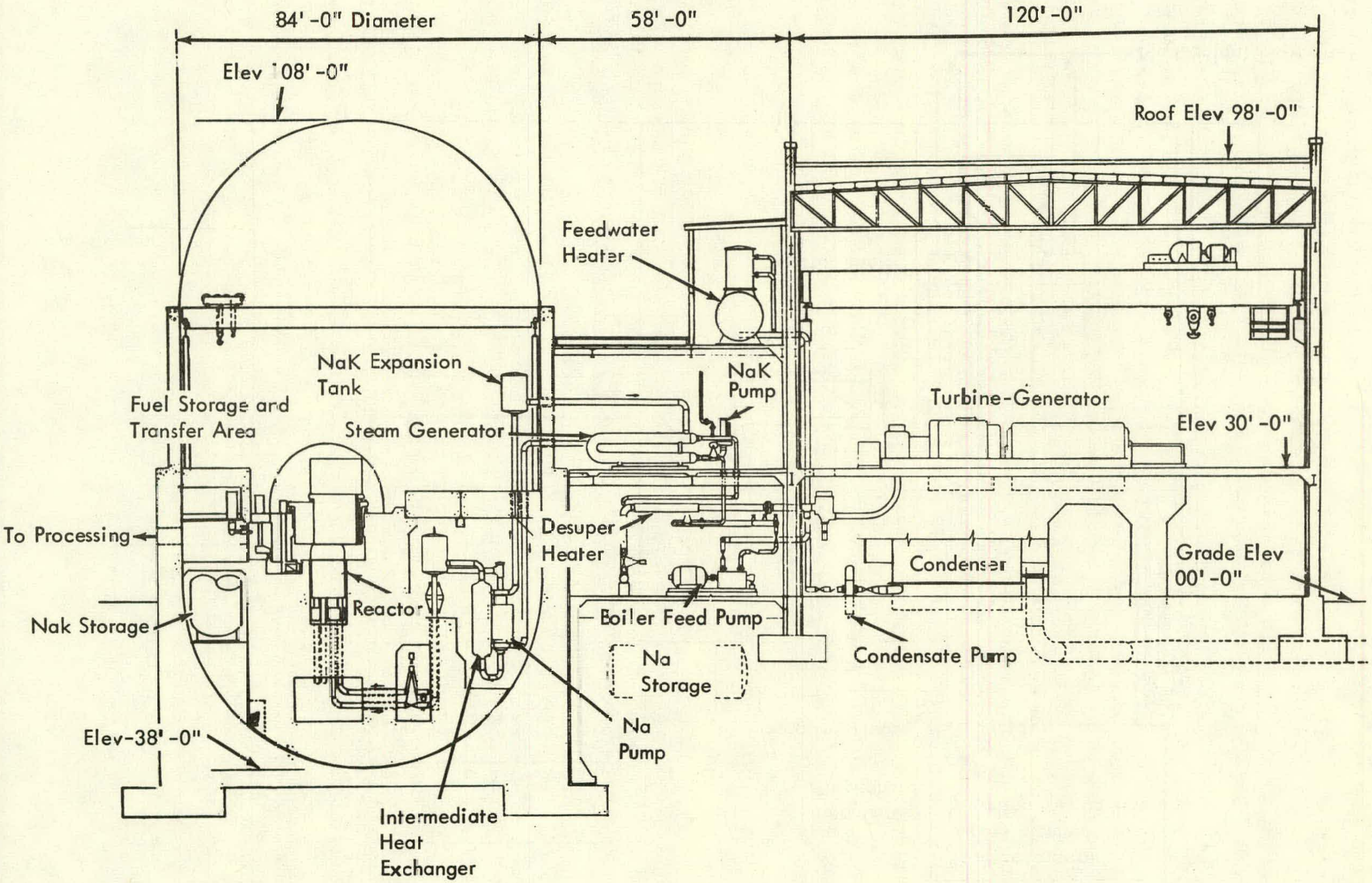


FIG. 8 PLANT ARRANGEMENT INCORPORATING SEPARATE FUEL PROCESSING FACILITY AND VERTICAL PIPING LAYOUT , A 1953-54 CONCEPT

The plant arrangement used a vertical shell-and-tube heat exchanger as shown in Figure 9. A vertical Einstein-Szilard pump was arranged so that the parts could be removed from above-floor using the building crane. An expansion tank was provided at the high point, the isolation valves were located in a shielded trench, and angle check valves were indicated. An alternate system arrangement is shown in Figure 10. Three features of this arrangement are significant: the spherical containment building, the U-type intermediate heat exchanger (IHX), the extensions on the isolation valves, and the use of diffusion cold traps.

Previous experience with liquid-metal pumps was limited to pumps of low capacities and low heads. The types of pumps considered were mechanical centrifugal pumps, flat-bed electromagnetic pumps, and Einstein-Sziland reverse flow pumps. It was felt that mechanical pumps could be designed as ordinary centrifugal pumps; however, shaft seals presented a very difficult problem. Electromagnetic pumps had the big advantage of having no moving parts and hence could be a sealed unit; however, pump efficiency was expected to be low.

G. 1955 PRIMARY SYSTEM DESIGN

1. Significant Changes

In 1955, an entirely different primary system arrangement evolved as indicated in Figures 11 and 12. The most significant changes were:

- Subassembly length was increased threefold to include the upper and lower blanket in a single unit. The concept of a cast-segment core was changed to that of a pin or plate type.
- An upper sodium pool was included in the reactor vessel design for complete submergence of the subassemblies during fuel transfer to a transfer tank located at the same level as the lower vessel; this pool became a part of the hydraulic flow path.
- Pump design was changed from an Einstein-Szilard to a mechanical system type with integral check valves.
- The reactor, IHX's and pumps were to be tanks, interconnected by piping.
- Gravity flow was adopted from the reactor through the IHX to the pump tank.
- The reactor was housed in a secondary containment tank (primary shield tank).

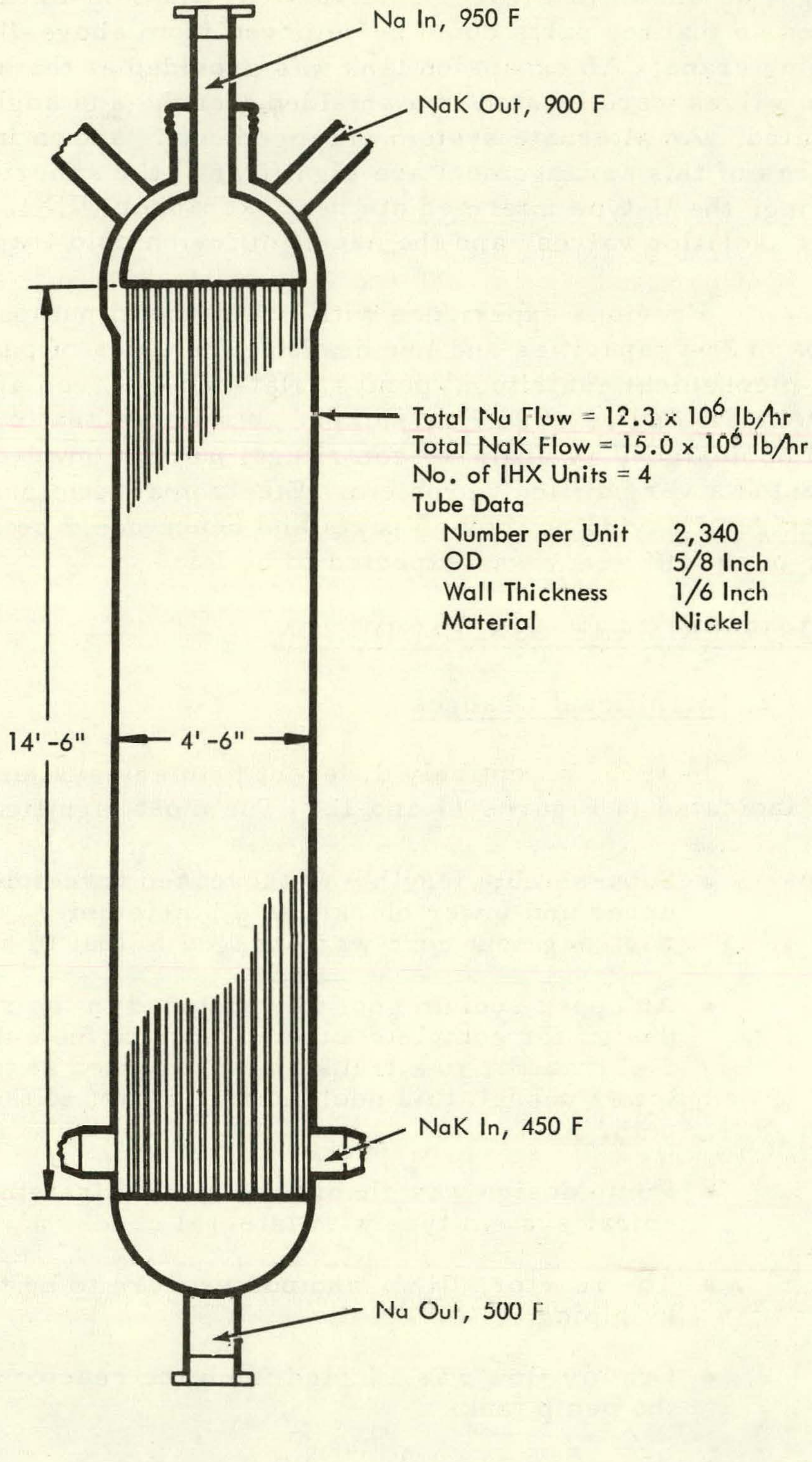


FIG. 9 INTERMEDIATE HEAT EXCHANGER, A 1953-54 CONCEPT

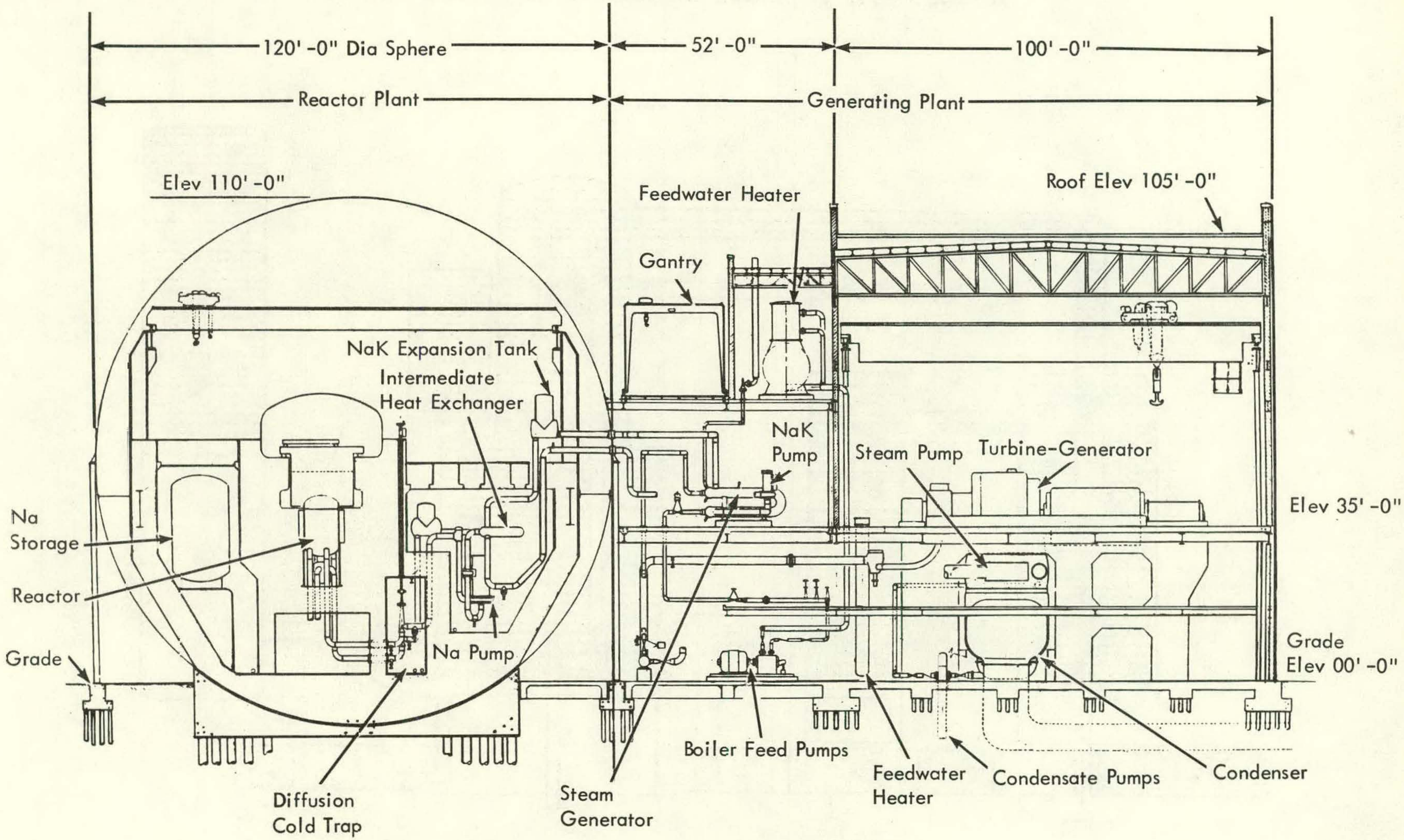


FIG. 10 A 1953-54 CONCEPT OF A PLANT ELEVATION

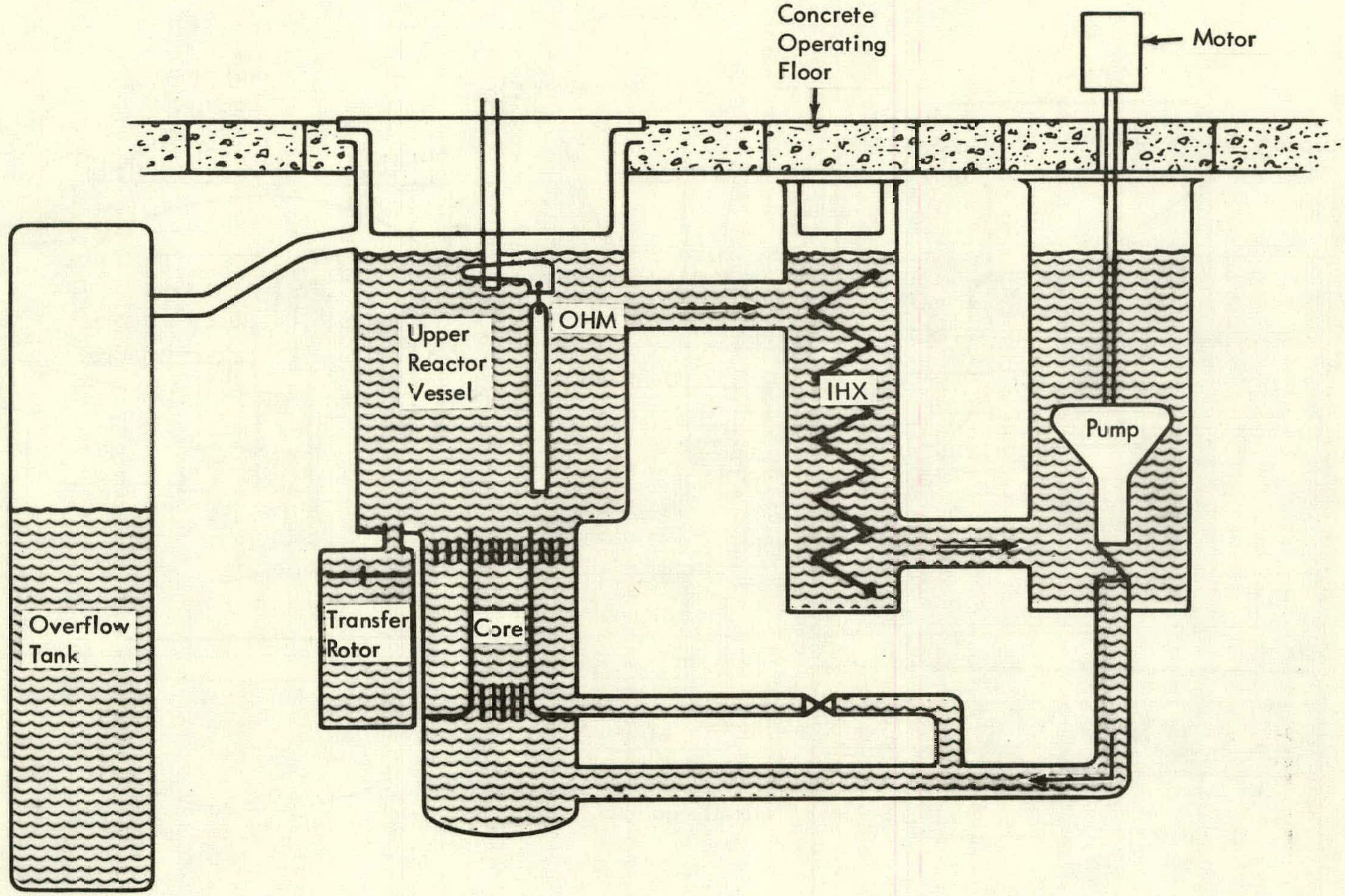


FIG. 11 PRIMARY SYSTEM WITH COMPONENT TANKS , A 1955 CONCEPT

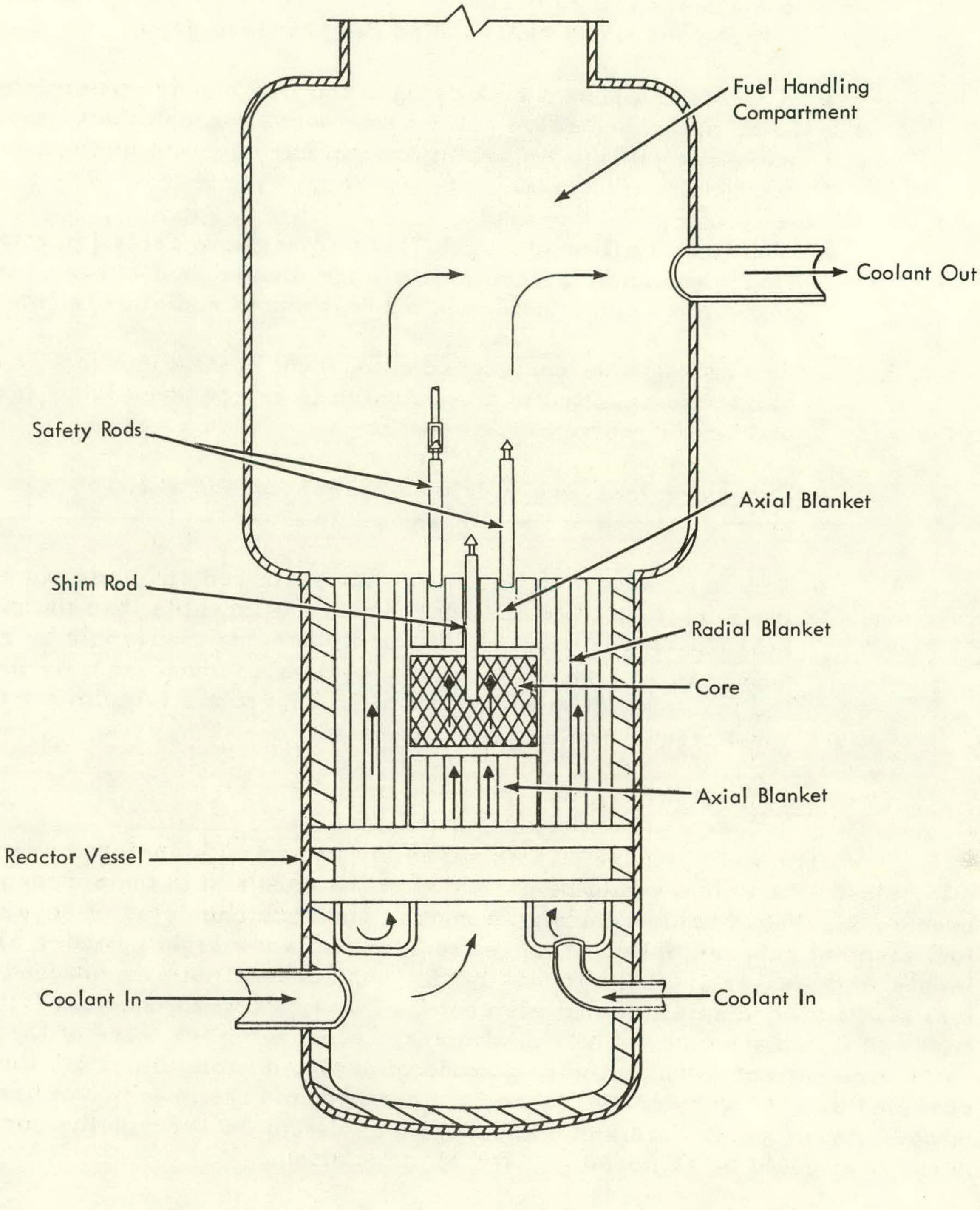


FIG. 12 REACTOR ARRANGEMENT, A 1955 CONCEPT

- A level was maintained in the reactor by use of an overflow line attached to an overflow tank. The difference in levels between the tanks represented the pressure drop.

- With the component tank design, the IHX bundle, pump internals, and check valve could be removed through the tops of the tanks without disturbing the primary system piping connections to the tanks.

- With gravity flow, all of the primary system seals for rotating shafts and handling equipment were located in a lower pressure cover gas region instead of a pressurized sodium region.

- An analysis that considered component costs and effects on plant size resulted in a conclusion to select three loops instead of four for the reference design.

- Expansion tanks were deleted because of the available gas spaces in each of the IHX and pump tanks.

- Stop valves were deleted from the main coolant lines since their integrity was considered more vulnerable than the piping, heat exchanger housing, and pump housing they would be required to isolate. The valves would also unnecessarily increase the pressure drop which had to be kept to a minimum for natural circulation emergency cooling.

2. Core Design

The reactor core was an assembly of partially enriched uranium alloy pins. Plutonium would be produced in the core and in the surrounding breeder blanket of depleted uranium rods. The core consisted of square fuel element subassemblies arranged to approximate a right cylinder 31 inches in diameter and 30.5 inches high. The subassemblies contained the fuel elements and axial blanket elements. Two types were considered: pin type and flat-plate type. The radial blanket subassemblies were of the same size but contained cylindrical rods of depleted uranium. Both the core and blanket were cooled by sodium pumped into the lower chamber and upward through both core and blanket. By using upflow through the core, decay heat could be removed by natural circulation.

The criterion for the pin-type fueled core was that a temperature of approximately 1220 F for the fuel alloy transformation temperature would not be exceeded during steady-state full-power operation. Another criterion was that the blanket would not exceed 1220 F at high-power operation. An alternate design using plate-type fuel elements under same conditions, required a maximum fuel temperature of 1020 F, approximately 200 F lower than the pin-type criterion. In both cases, the core would not be orificed; however, the radial blanket would be orificed to keep the coolant temperature rise in the blanket subassemblies approximately the same as that through the core.

3. Reactor Design

The reactor vessel (Figure 13) consisted of (1) a lower vessel which contained the inlet plenums for both core and blanket, the 14-inch and 4-inch inlet nozzles, and the fuel subassemblies; (2) an upper vessel which contained a mechanical holddown over the core subassemblies, an upper pool for handling subassemblies while submerged, and the 24-inch outlet nozzles; (3) a transfer rotor container used to transfer fuel from the upper vessel to the decay tank; and (4) the rotating shield plug and its associated mechanisms. Figure 14 shows details of the inlet plenum, primary shield tank containment penetrations, and the support structure.

4. Primary System Arrangement

The arrangement of the primary sodium system is shown in Figure 15. Sodium flowed by gravity from the free surface pool of the reactor upper chamber to the shell side of the intermediate heat exchanger and then to the pump tank. The volume above the sodium level in each tank was interconnected and filled with inert gas at a pressure slightly below atmospheric.

The pump delivered the sodium to the reactor in two lines in each of the three loops. The bulk of the flow was to the core, while a sidestream controlled by a throttle valve supplied the radial blanket section.

Since the three sodium loops had a common point in the reactor, a pump failure could result in flow reversal. For this reason, a check valve was used in the discharge line of each pump.

To prevent a major sodium leak in the inlet system piping external to the reactor from draining the reactor vessel due to siphon action, a siphon-breaker line was provided between each reactor inlet line and the reactor pool.

A removable filter was mounted in each pool outlet line to remove foreign material from the sodium system during initial cleaning and filling operations.

Diffusion cold traps (16 inches ID and 4 feet long) were located in each pump discharge line.

A primary shield tank was provided to contain the reactor vessel and adjacent piping. The principal purpose of the primary shield tank is to act as secondary containment in the event of a sodium leak from the reactor vessel or any of the piping.[4]

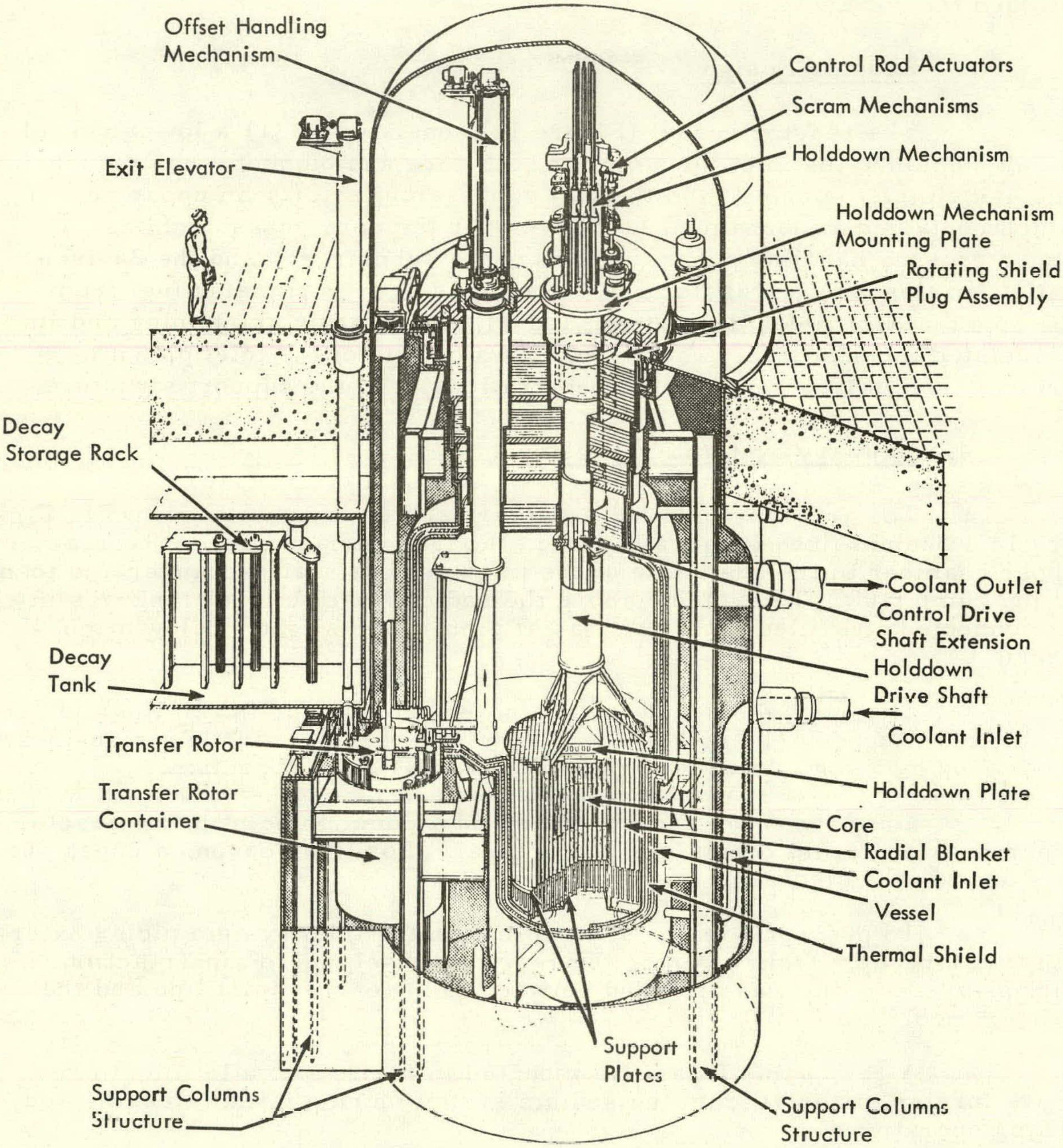


FIG. 13 PERSPECTIVE VIEW OF REACTOR, A 1955 CONCEPT

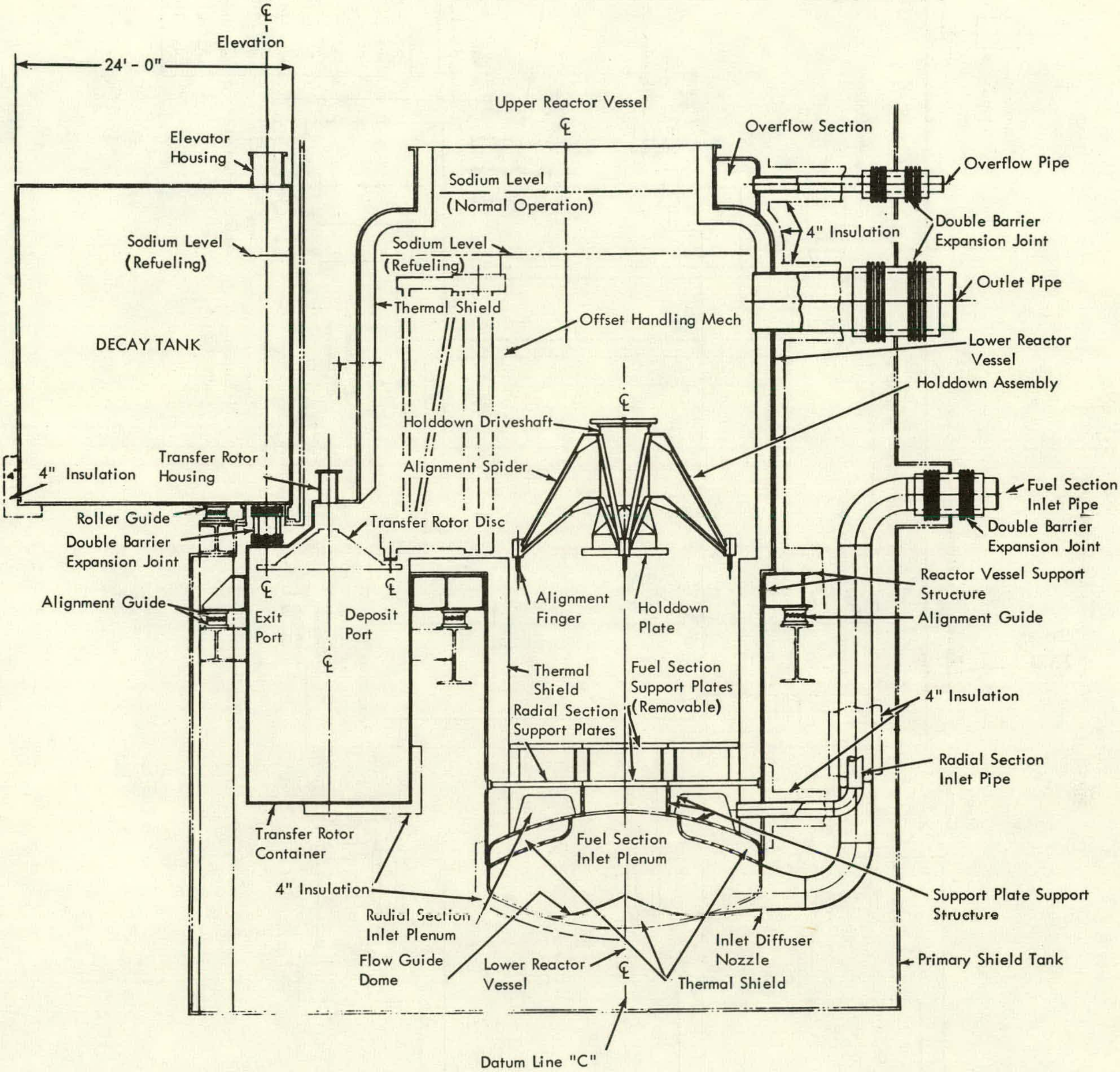


FIG. 14 REACTOR VESSEL ELEVATION ,A 1955 CONCEPT

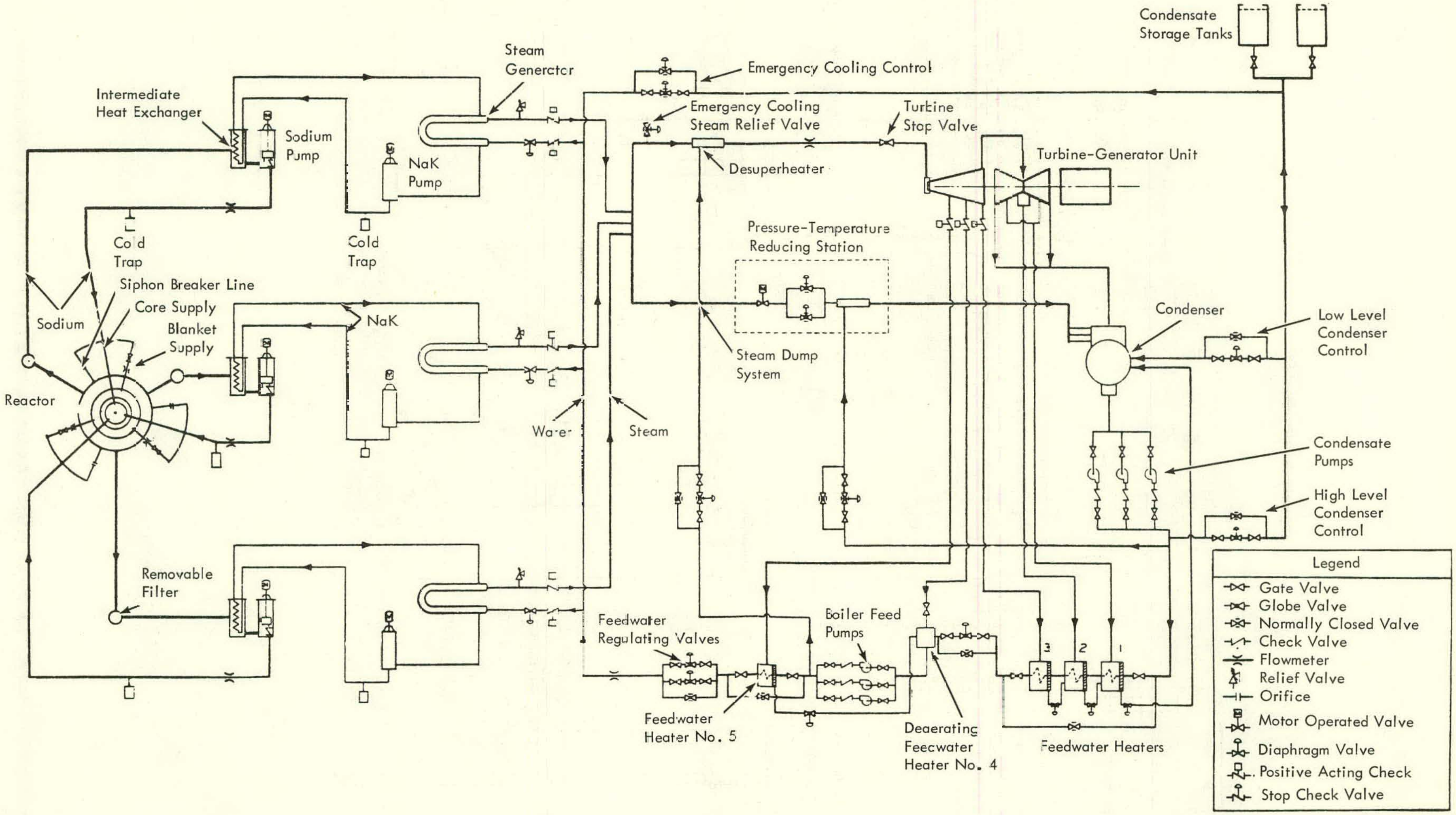


FIG. 15 SCHEMATIC OF 1955 CONCEPT LIQUID METAL AND STEAM POWER SYSTEMS

The arrangement of the systems and components is shown in Figures 16 and 17, indicating the following significant features:

- A cylinderical containment building having an elliptical bottom and hemispherical top that was later selected for the final design.

- Neutron shielding consisting of 30 inches of boron-containing material adjacent to the primary shield tank; direct horizontal piping penetrations were made through the shield.

- Expansion loops in the sodium systems were horizontal instead of vertical and pitched to achieve complete drainage of sodium.

- Siphon breakers were shown at the top of the vertical expansion loops in the pump discharge piping to prevent siphoning sodium out of the reactor in the event of a leak in the piping at the lower levels (see Figure 15).

- Secondary coolant piping is shown penetrating the shield plug of the IHX to the above-floor area.

5. Components

a. Intermediate Heat Exchanger

The intermediate heat exchanger, shown in Figure 18, was a shell- and tube-type unit with primary sodium on the shell side and secondary NaK on the tube side; it featured a removable tube bundle. The 4-foot dimension between the sodium level and reactor level datum line represented a loss of effective surface due to anticipated pressure drop.

The NaK piping penetrated the operating floor above the IHX by a circuitous path for shielding purposes. The bundle would be removed by severing the secondary lines above floor and unbolting the below-floor flanged joint with remotely operated tools.

b. Primary Sodium Pump

The primary pumps, Figure 19, were electric-motor driven, centrifugal, sump-type pumps with an inert gas shaft seal. The pump shaft, impeller, and casing were capable of being removed as a unit. The 6-foot dimension represented anticipated pressure drop from the reactor vessel to the pump inlet. The center discharge design was specified to faciliate removal and replacement operations. Ball-type check valves were preferred over the swing type because it was suspected that bearing problems might be incurred with the latter. The check valve was integrated with the pump disso that it could be removed with the pump.

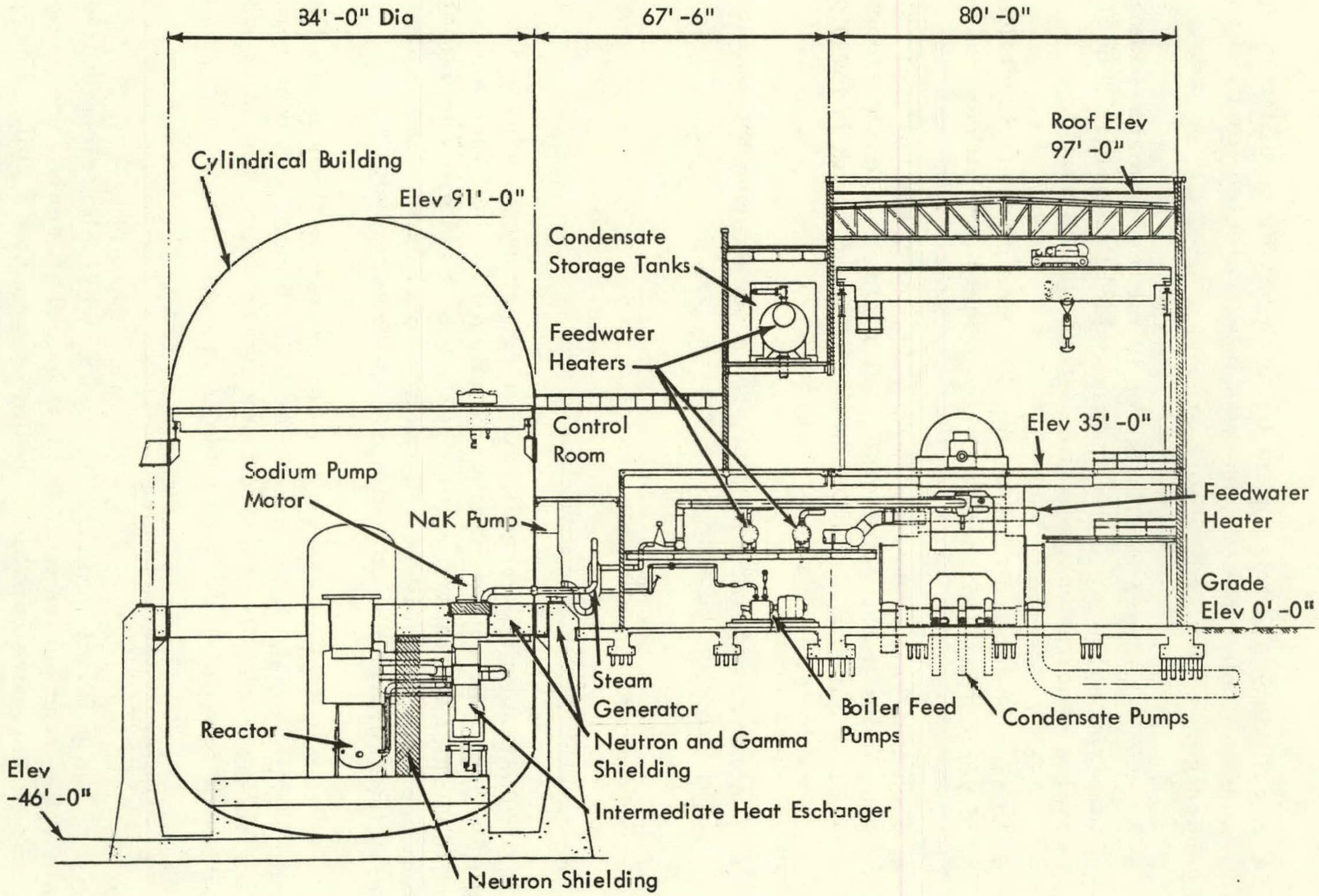


FIG. 16 A 1955 CONCEPT OF AN ALTERNATE PLANT ARRANGEMENT USING HORIZONTAL PIPING

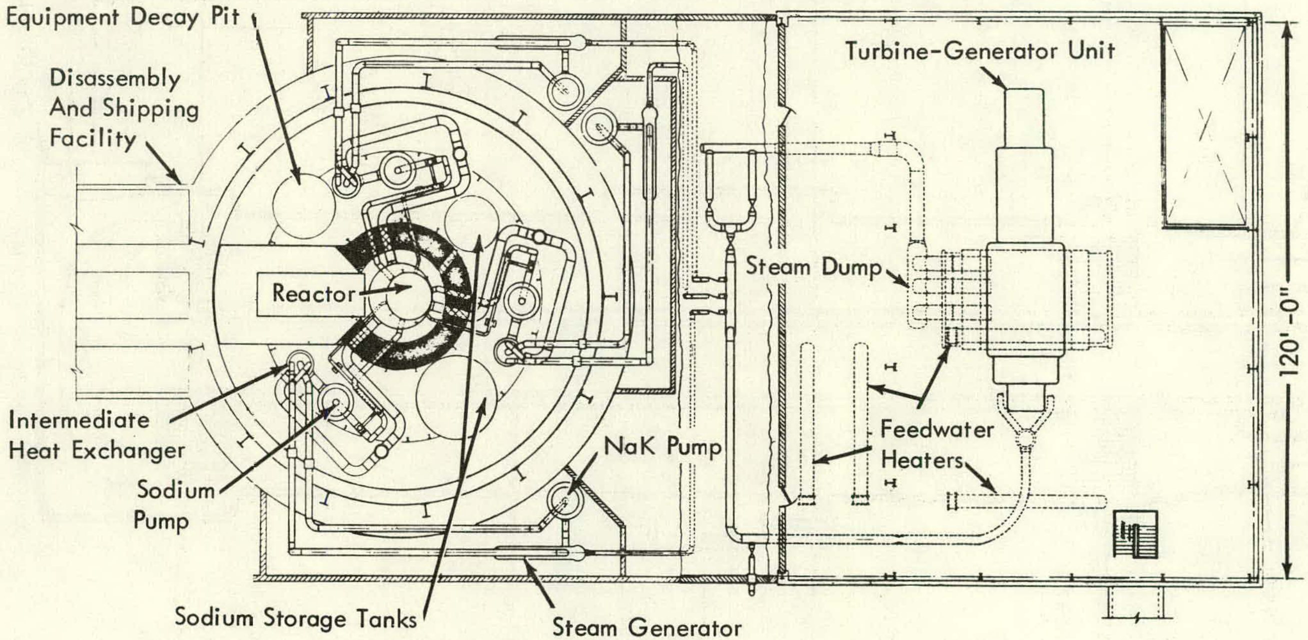


FIG.17 PLAN VIEW OF PLANT ARRANGEMENT USING HORIZONTAL PIPING LAYOUT, A 1955 CONCEPT

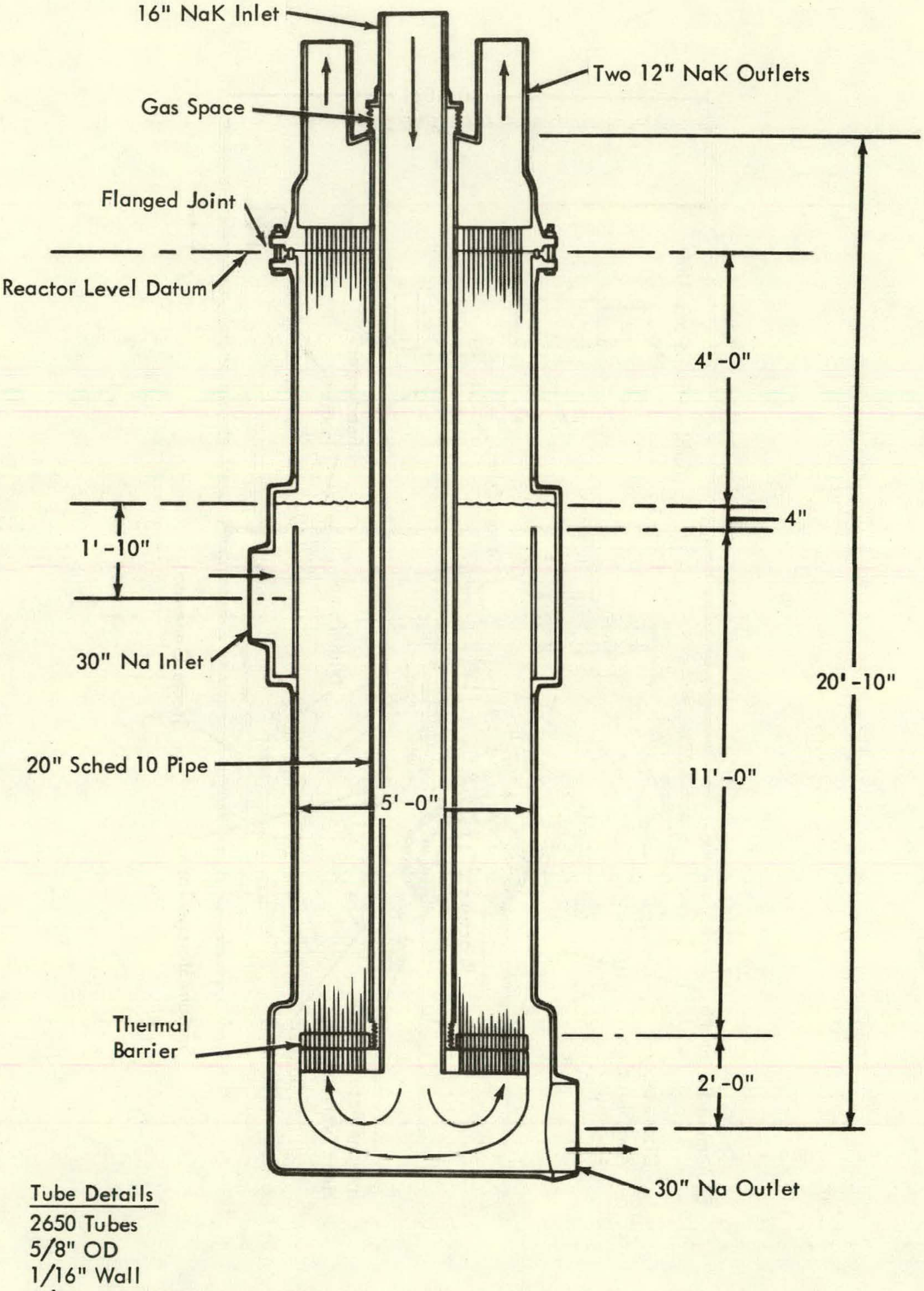


FIG. 18 INTERMEDIATE HEAT EXCHANGER, A 1955 CONCEPT

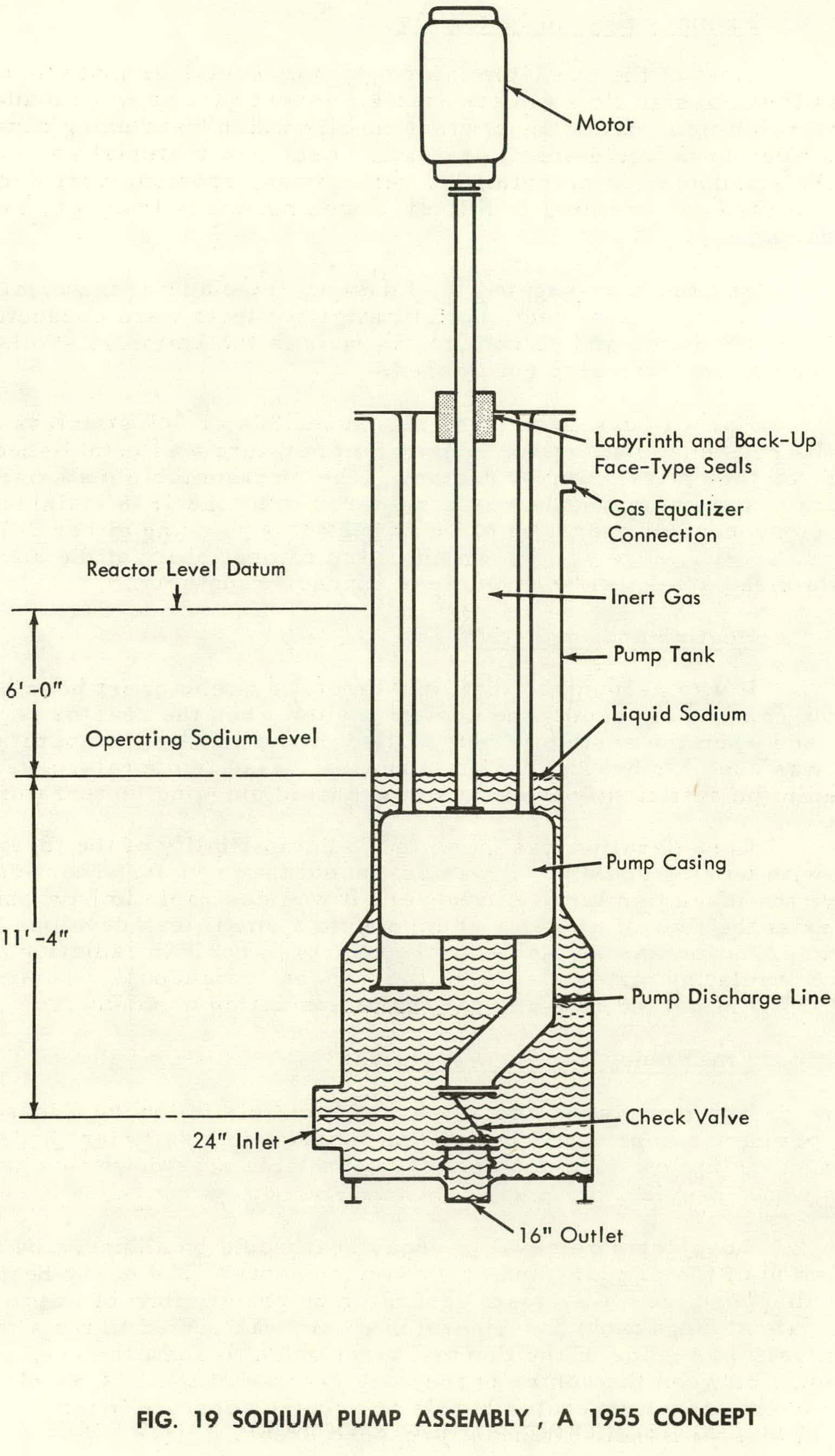


FIG. 19 SODIUM PUMP ASSEMBLY, A 1955 CONCEPT

6. Primary System Materials

Most of the available information on container materials was related to the austenitic stainless steels. Investigations were made to develop similar information about other metals which were being considered for installation in liquid-metal systems. Each new material was considered from the standpoint of intergranular penetration, erosion, corrosion by contaminants, self-welding or diffusion bonding, mass transfer, and radiation damage.

For the cast-segment fuel design, the cladding material for the reference was stainless steel, but investigative tests were conducted on Globeiron, titanium, and zirconium, as well as the stainless steels. The use of unclad fuel was also considered.

For all subsequent designs, Types 304 or 347 stainless steels were the reference materials. Design temperature was established at 1000 F for the entire primary system. The most suitable material for the primary system components was considered to be the 18-8 stainless steels, based on successful operation of liquid metal loops using either 347 or 304 stainless steel. More was known about the campatibility of the alkali metals with stainless steel than with any less expensive material.[2]

7. Heating and Insulation

It was recognized that liquid metal systems must have heating systems capable of keeping the coolant molten when the reactor is not operating, and when the system is being filled or drained. A temperature of 400 F was used for heating design purposes. This temperature was considered to be sufficiently in excess of expected plugging temperatures.

Consideration was given to the compatibility of the thermal insulation with leaking sodium. It was suspected that a violent reaction would remove the insulation locally; however, it was desirable to have the insulation resist the flow of molten sodium should a small leak develop. Very little information was available on the effects of nuclear radiation on the various insulating materials. The Liquid Metals Handbook,[7] published in July 1955, did not indicate any particular insulating problems.

8. Emergency Cooling Provisions

The removal of decay heat immediately following a loss of power to the primary pumps would be handled by providing sufficient hydraulic inertia in the upflow core coolant. The available coastdown flow from the pumps would provide this.

Long-term removal of decay heat would be achieved by thermal circulation of the primary and secondary coolants. The decay heat was finally disposed from the steam generator by gravity flow of water from the condensate storage tank; the generated steam was vented to the atmosphere. An analysis was made of the thermal circulation to show the needed elevation difference between the center of the core, center of the IHX bundle, and center of the steam generator bundle to achieve adequate lengths for the hot and cold legs with their attendant pressure drops.

In addition to natural circulation, it was planned to use battery power as a backup for pump requirements.

9. Primary System Control[11]

Two types of plant control systems were considered: one was based on a constant coolant flow with a temperature increase through the reactor proportional to the power; the second was based on a variable coolant flow and a constant temperature.

With emphasis on reactor control, since the plant was a base load plant, the constant flow control system was selected because of better nuclear stability and reduced thermal shocks during scrams, especially from low power levels.

Concurrent with development work on reactor control and instrumentation, a reactor dynamics simulator study was initiated. This included a general study of the problem of simulating the operation of the complete power plant.

For startup, the secondary NaK system would be brought up in temperature concurrently with the primary system. Primary and secondary system flows would be held constant as reactor power was increased.

For normal operation, the coolant systems would be maintained at constant flow while the coolant ΔT would be varied over a load range of approximately 20 to 100%. A means could be provided, if necessary, to vary coolant rates if reactor conditions dictated this requirement.

10. Auxilary Systems

In order to provide a means of analysis of the primary sodium for the presence of oxides, a relatively simple plugging indicator was proposed. A small stream of sodium would be bypassed through an orifice while the temperature was lowered. This plugging temperature would be maintained well below the operating temperature by the removal of oxides in the cold trap.[2] Preliminary purification would be accomplished by distillation or filtration. Entrained gases would be removed from the high point of each loop.

Helium was initially chosen as the cover gas for the primary system and was also considered for use in the ventilating systems. A receiving and storage station for the cover gas included purification and sampling facilities.

A vent gas system was connected to the primary cover gas system to monitor and dispose of gaseous fission products.[4] The equipment of this system included a vapor trap and storage tank and a bypass line to the stack, where gases would be diluted with building ventilation air.

H. PERFORMANCE SPECIFICATIONS

Table 2 lists the design specifications of the 1953-1954 design described in Reference 2 and the component tank and gravity flow design described in Reference 6.

TABLE 2 - PERFORMANCE SPECIFICATIONS

	Reference 2	Reference 6
Plant		
Gross electric capacity, Mw	-	100
Net electric capacity, Mw	150	90
Reactor		
Reactor power, Mwt	500	300
Average heat flux, Btu/hr-sq ft	1,300,000	765,000
Heat flux, maximum to average	1.4	1.43
Power density, kw/cu ft	50,000	23,500
Specific power, kw/kg U-235	1400	612
Core diameter, inches	26	30.5
Core length, inches	29	30.5
Coolant volume, %	50	53
Reactor sodium flow rate, gpm	29,000	30,000
Maximum fuel temperature, F	1330	1220
Primary System		
No. of Loops	4	3
Reactor outlet temperature, F	950	800
IHX outlet temperature, F	500	550
Flow per loop, gpm	7200	10,000
Pump capacity, gpm	7800	11,000
Pump dynamic head, psi	160	110
Secondary System		
Steam generator inlet temperature, F	900	750
Steam generator outlet temperature, F	450	500
Flow per loop, gpm	9000	12,500
Steam temperature, F	800	730
Steam pressure, psi	1200	600
Steam flow, lbs/hr x 10^6	1.38	1.03

IV. DESCRIPTION OF THE FINAL DESIGN

The information given in this section is basically a description of the final design of the primary system and its components. The Appendix describes studies covering heat cycles, thermal transients, decay heat, emergency cooling, etc. that were involved in designing the system.

A. SYSTEM PERFORMANCE AND OPERATING CONDITIONS

By the end of 1955, the concept of the primary system had been finalized, evolving into a reactor with a sodium-free surface, gravity flow of sodium through the IHX to the pump, component tank-type system, and three primary coolant loops. The reactor power level was established at 300 Mw thermal and the plant generating capacity was rated at 100 Mw electrical.[8,9]

At this time, nuclear fuel technology was considered to be sufficiently advanced to produce a core for 300 Mwt plant operation; however, it was anticipated that the design of future cores using more advanced technology would result in a core that would permit plant operation at its maximum capability.

The ultimate capability of the plant then was considered to be 430 Mw, the thermal power required for a 150-Mw turbine-generator. One of the features of the component tank design was that the fixed portions of the system, i.e., the reactor vessel, component tanks, and piping, could be designed for the ultimate capacity, while the removable internals of the system components could be designed for lower power operation. These internals consisted of the intermediate heat exchanger (IHX) bundles and the pump internals, including the check valves; the blanket flow throttle valves, although not categorized as internals, are removable in the same fashion.

In the case of the IHX bundle, the surface initially provided was sufficient for 430-Mwt operation. The primary pump was designed for 110% of 300-Mwt flow at 1000 F; however, the performance curve shows there would be no difficulty in attaining 120% flow at its normal operating temperature of 600 F and at a slightly reduced head. The other plant components, including the secondary system and the turbine-generator were sized for 430-Mwt conditions.

In 1958, Table 3, in which the plant performance based on 430-Mwt operation was projected for a 10-year period, was published. In the final design of Core A subassemblies, the reactor power level was reduced to 200 Mwt to limit coolant flow and the resultant forces and internal pressure loads within the subassembly and thereby alleviate anticipated design and fabrication problems. Thus the first high-power operating license was obtained for a 200-Mwt level with the reactor core fueled with the initial subassembly loading (Core A). The reactor vessel inlet plenum was designed for an operating pressure of 110 psig; however, a pressure limitation of 65 psi was placed on Core A because of the structural characteristics of the subassembly wrapper can. With this reduced allowable pump discharge head, the flow

TABLE 3 - PROJECTED PLANT PERFORMANCE

Year	Reactor Power, Mwt	Primary Sodium Temperatures, F		Sodium Flow, lb/hr	Secondary Sodium Temperatures, F		Feedwater Temp, F	Steam Flow, lb/hr	Steam Temp, F	Steam Press., psia	Estimated Electric Output, Mwe	Heat Rate, BTU/Kw-hr
		Reactor Inlet	Reactor Outlet		IHX Inlet	IHX Outlet						
Test Period	---	550	800	13.2×10^6	---	---	---	---	---	---		
1	300	550	800	13.2×10^6	500	750	400	1,023,000	742	600	102	9995
2	300	550	800	13.2×10^6	500	750	400	1,023,000	742	600	102	9995
3	350	600	850	15.5×10^6	540	790	410	1,185,000	783	720	124	9609
4	350	600	850	15.5×10^6	540	790	410	1,185,000	783	720	124	9609
5	400	600	875	16.1×10^6	532	807	420	1,374,000	790	830	144	9467
6	430	600	900	15.9×10^6	520	820	430	1,509,000	780	900	155	9470
7	430	600	900	15.9×10^6	520	820	430	1,509,000	780	900	155	9470
8	430	600	900	15.9×10^6	520	820	430	1,509,000	780	900	155	9470
9	430	600	900	15.9×10^6	520	820	430	1,509,000	780	900	155	9470
10	430	600	900	15.9×10^6	520	820	430	1,509,000	780	900	155	9470

becomes 191 gpm through each subassembly. For 430-Mwt operation, only the fuel subassemblies would have to be changed, and the core pressure drop would have to be compatible with pump developed head performance at 12,000 gpm (120% flow).

The projected plan then included operating the reactor at 200 Mwt in a power demonstration program, preceded by a period of low-power operation for nuclear and component testing. Subsequent to the 200-Mwt program, an 18-month fuel irradiation program would be conducted at 110 Mwt based on a maximum reactor fuel temperature of 800 F to prolong fuel life. Operation at this level would result in a reactor outlet temperature of 565 F and steam conditions that would result in approximately 30 Mwe.

B. CORE AND BLANKET DESIGN

During the years 1955 to 1959, developmental work on the core and blanket was concerned with (1) the selection of the final design of fuel and blanket subassemblies, including such factors as support, configuration, diameter, and cladding thickness; (2) sodium flow rates; and (3) core and blanket dimensions.

The reference fuel subassembly design[8] was a pin- or plate-type element supported either by elliptical lenticular wires or honeycomb grids. Hydraulic tests, conducted in water, resulted in the selection of the pin-type, honeycomb-grid-supported subassembly, shown in Figure 20, in March 1960. Although the plant components were designed for 300 Mwt and an ultimate maximum of 430 Mwt, flow limitations were set to moderate the pressure drop for which the Core A fuel subassembly would have to be designed. This dictated a maximum reactor power of 200 Mwt for the Core A fuel loading.

The arrangement and identification of the reactor positions are shown in Figure 21. The subassemblies, supported in a double support plate structure in the reactor vessel, are removable, and to a certain extent interchangeable. The fuel core region is about 31 inches high and 31 inches in diameter and is radially surrounded by breeder blanket subassemblies, shown in Figure 22.

A mechanical restraint against uplift and radial movement of subassemblies is provided by the holddown mechanism over the 149 lattice positions located above the high-pressure plenum, through which $\sim$90% of the primary sodium flow passes. Over the outer radial blanket positions above the low-pressure plenum, the pressure drop force acting on the blanket subassemblies is less than their weight; therefore, no holddown is required.

C. HEAT TRANSPORT SYSTEM

The flow diagram for the primary system and the entire plant is shown in Figure 23. Table 4 lists the plant design and performance data for Core A at 200 Mwt as well as 300-Mwt and 430-Mwt operation.

The reference system design[8] was changed from previous designs to allow the use of sodium in both the primary and secondary systems after a comparison was made. This included a review of heating requirements,

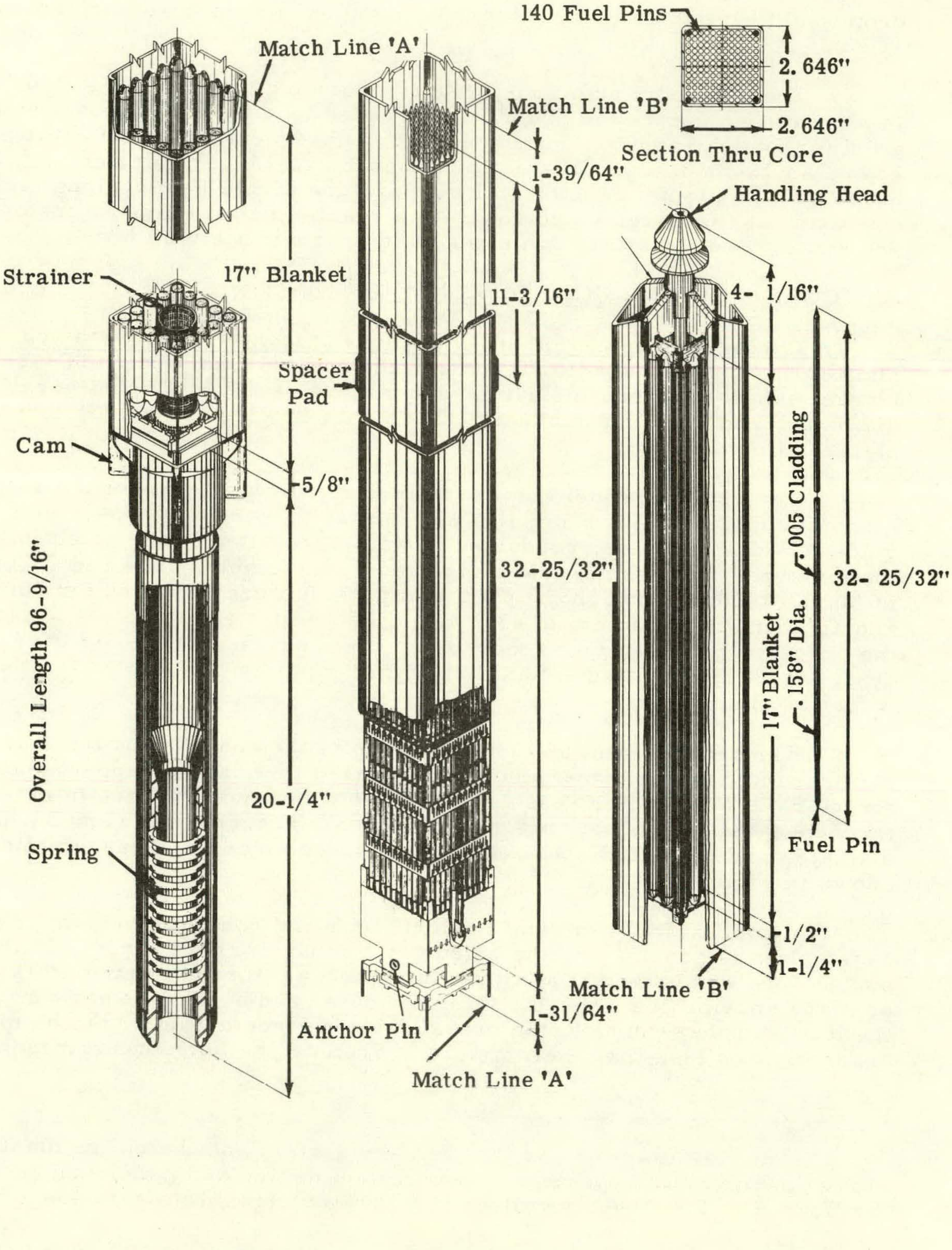


FIG. 20 CORE SUBASSEMBLY

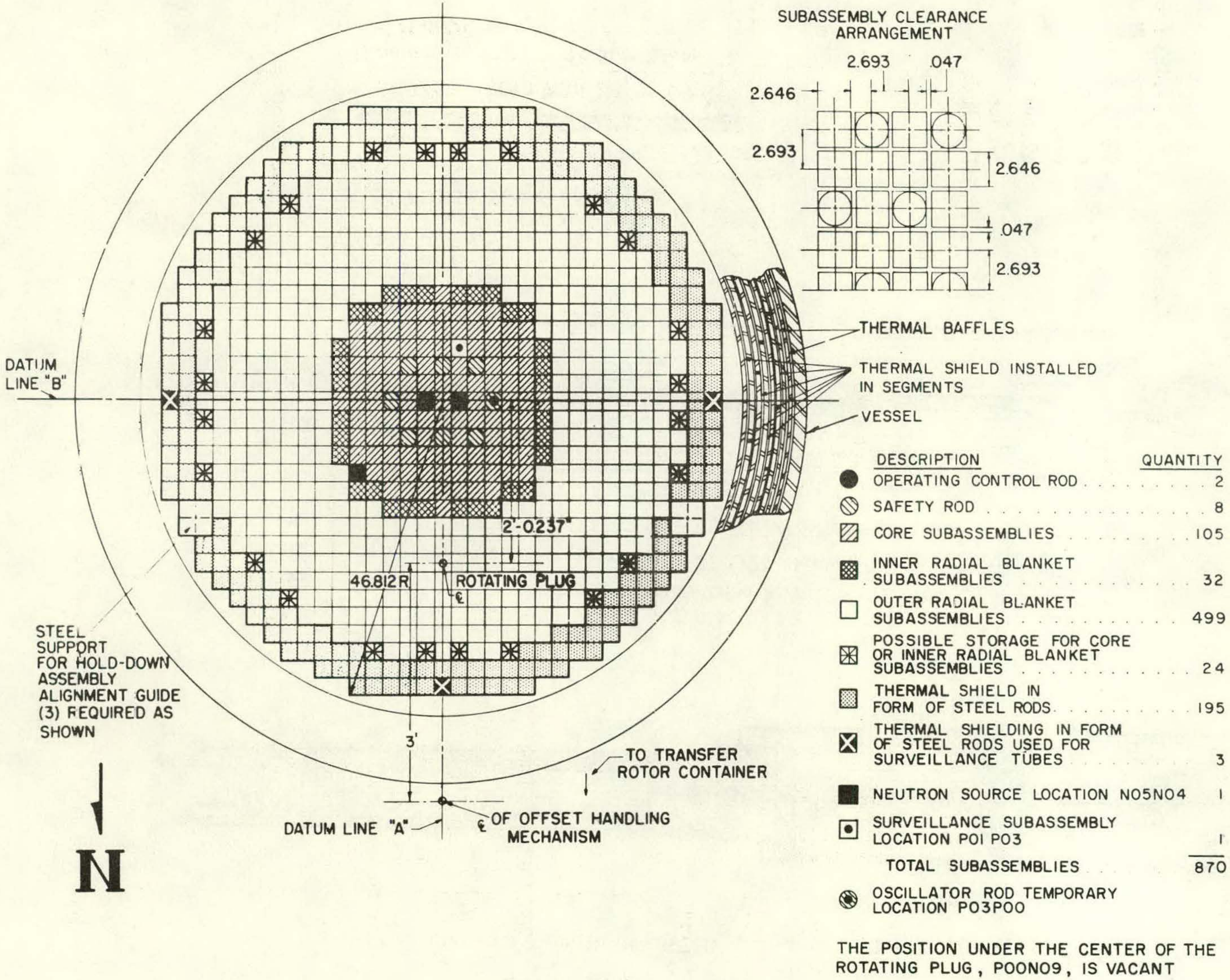


FIG.21 REACTOR CROSS SECTION

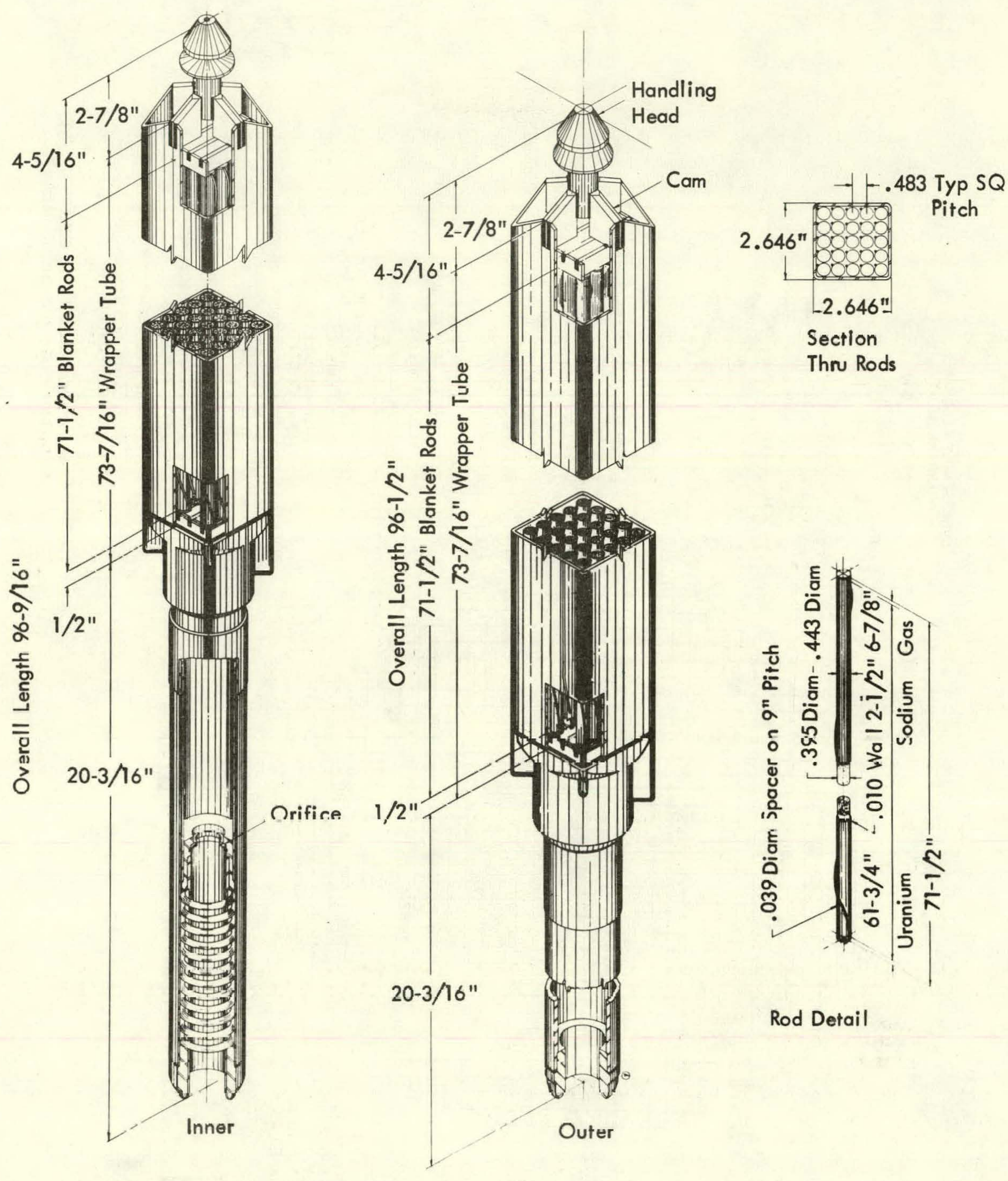


FIG. 22 RADIAL BLANKET SUBASSEMBLY

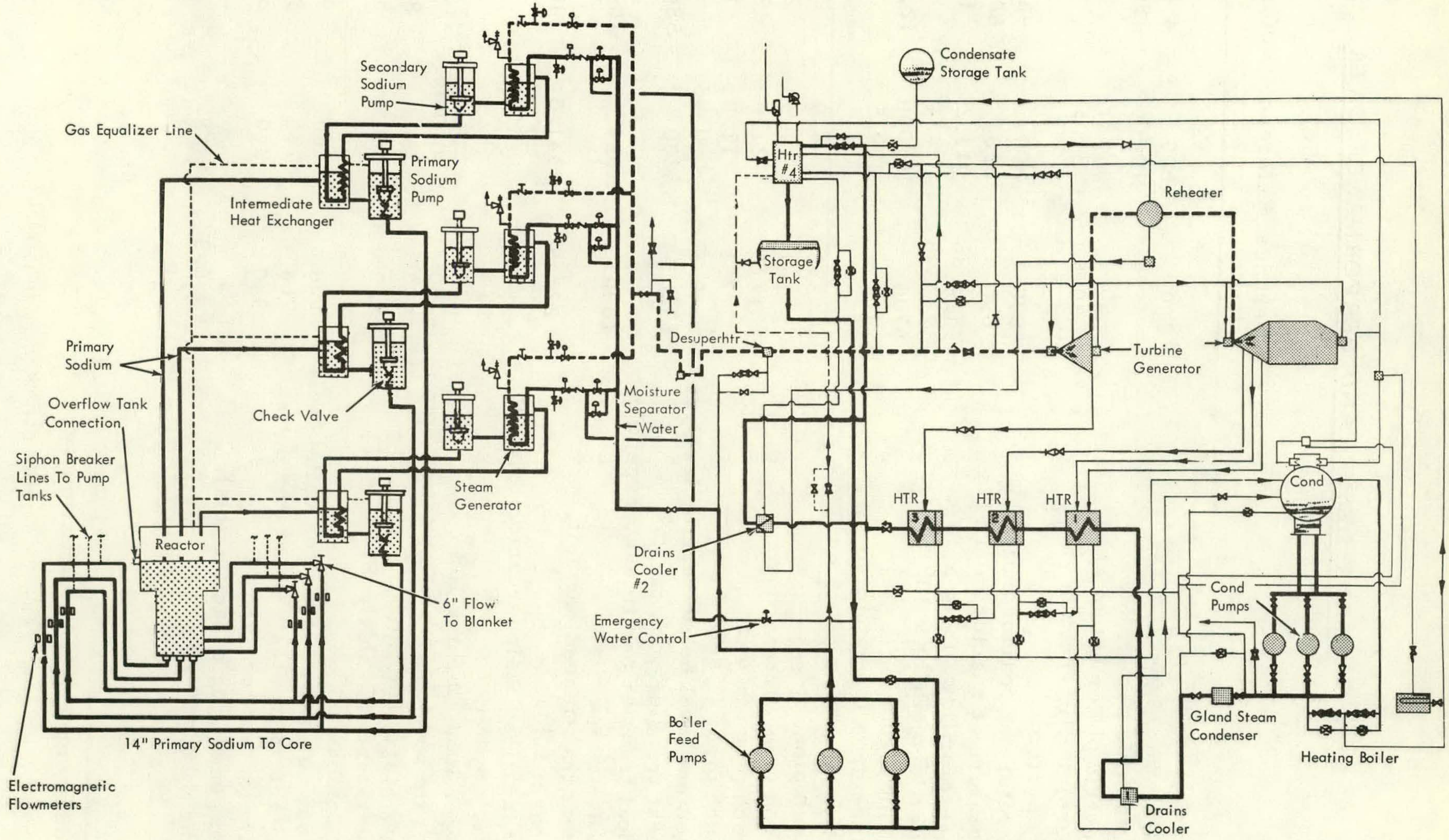


FIG. 23 FLOW DIAGRAM OF THE HEAT TRANSPORT SYSTEM

TABLE 4 - PLANT DESIGN AND PERFORMANCE DATA

	Reference 10	Reference 9	Reference 9*
Plant Performance			
Reactor power, Mwt	200	300	430
Gross electric power, Mwe	65.9	104	150
Net electric power output, Mw	60.9	94	-
Net thermal efficiency, %	30.5	31.3	-
Primary System			
Temperature at reactor outlet, F	800	800	900
Temperature at reactor inlet, F	550	550	600
Total reactor flow, lb/hr x 10^6	8.86	13.2	15.87
Flow per loop, lb/hr x 10^6	2.95	4.4	5.3
Pump flow, gpm	6700	10,000	12,000
Pressure drop in system, ft	126	299.5	-
Velocity			
30-inch pipe, ft/sec	3.3	4.9	6.1
16-inch pipe, ft/sec	11.7	17.5	21.2
14-inch pipe, ft/sec	14.0	21.0	28.1
6-inch pipe, ft/sec	7.3	11.0	13.0
IHX sodium level below reactor level, ft	0.45	0.98	1.37
IHX heat transfer surface, ft^2	6320	6200	5840
IHX heat transfer coefficient, expected, Btu/hr-ft^2 - F	1090	1059	1048
IHX heat transferred per unit, Btu/hr x 10^6	227.5	341.3	489
Pump tank level below reactor level, ft	1.26	2.75	4.09
Secondary System			
Flow per loop, lb/hr x 10^6	2.95	4.4	5.3
Flow rate per loop, at pump temp, gpm	6700	10,000	12,000
Temperature IHX outlet, F	767	750	820
Temperature IHX inlet, F	517	500	520
Steam Conditions			
Pressure at steam generator outlet, psia	600	600	900
Temperature at steam generator outlet, F	764	740	780
Feedwater temperature, F	340	340	380
Steam generating capacity, total, lb/hr x 10^5	6.34	9.6	14.3

* Includes unpublished calculations.

heat transfer properties, costs and operating expenses, reactions with water, fire hazards, duplication of handling facilities, and nuclear properties. Specifically, sodium was selected as the coolant for the secondary system because it eliminated the need for two independant handling systems and the problem of mixing sodium and NaK should the secondary system leak into the primary system. Also, sodium is less expensive than NaK.

Pressure drop data for the primary system based on 200-Mwt plant operation are shown in Table 5; and Tables 6, 7, and 8 describe the thermal transient conditions that were specified for design of the primary system components such as the reactor vessel, the intermediate heat exchanger, and the primary pump.

D. COMPONENTS

1. Reactor Vessel

The final reference design for the reactor vessel represented a considerable refinement of details of vessel support, core support, and provisions for transferring and storing fuel elements, as shown in Figures 24 and 25.

The lower reactor vessel is a 114-inch-diameter cylinder with a 2 to 1 dished elliptical bottom head. The wall is 1-1/2 inches thick in the plenum region and 2 inches thick above. The lower portion of the vessel has three 14-inch-diameter inlet nozzles to the high-pressure core inlet plenum for core coolant flow, and three 6-inch-diameter inlet nozzles in a separate low-pressure blanket inlet plenum for outer radial blanket and thermal shield coolant flow. The two support plates, located in the lower vessel, are 2 inches thick and are spaced 14 inches apart by ribs welded to the plates. The holes in both support plates are spaced in a square array on 2.693-inch centers. The lower support plates have permanent orifice holders installed. Removable orifices with hole diameters varying from 0.25 inch minimum to 1.20 inches maximum are installed in the outer radial blanket area above the low-pressure plenum. No orifices are installed in the core or inner radial blanket positions above the high-pressure plenum. To prevent the insertion of a core subassembly in an orificed blanket position, the diameter of the holes in the radial blanket support plates is smaller than that of the core support plates.

The transition section connecting the lower reactor vessel, the upper reactor vessel, and the transfer rotor container consists of a flat head with vertical ribs and a deck plate. Brackets for the reactor vessel support are welded under the transition section in line with the ribs. Eight flexplate support columns, 2 inches thick and 7 feet long, are used to support the vessel from the transition deck and allow free thermal expansion while holding the upper centerline of the reactor vessel in a fixed position.

TABLE 5 - PRIMARY SODIUM SYSTEM PRESSURE DROPS [10]

	Per Cent of Total Sodium Flow in Section	Calculated Pressure Drop, Feet of Head
Check valve and 16-inch piping to tee	100	6.4
30-inch piping	100	0.6
IHX, including nozzles,	100	0.9
		7.9
14-inch piping	87	4.0
High-pressure reactor plenum	87	5.7
Core and axial blanket	80.2	106.0
Holddown mechanism	87	2.4
		118.1
6-inch piping and throttle valve, adjusted lift	13	109.6
Low-pressure reactor plenum and radial blanket	13	8.5
		118.1
Total system pressure drop for design flow condition of 8.86×10^6 lb/hr		126.0

TABLE 6 - THERMAL TRANSIENT DESIGN CONDITIONS FOR REACTOR VESSEL[10]

Origin of Transient	Region Directly Affected	Temperature at Start of Transient, F	Temperature at End of Transient, F	Maximum Rate of Change of Temperature, F/sec	Duration of Transient at Maximum Rate, sec
Reactor scrams, all cooling circuits continue to operate.	Holddown plate	900	600	-300	1
	Holddown assembly, including stabilizer arms	900	600	-30	10
	Outlet nozzle region	900	600	-15.8	19
	Internal surface of upper vessel shielding	900	600	-12	25
One secondary sodium pump stops, interlock fails, primary pump continues to operate, other circuits are operating.	Radial blanket inlet nozzle and plenum	600	700	+3.3	30
	Core inlet nozzle region	600	900	+10	30
	Core inlet plenum	600	750	+5	30
All pumps operating, feedwater cold slug.	Radial blanket inlet nozzles and plenum	600	550	-10	5
	Core inlet nozzle region	600	450	-21.4	7
	Core inlet plenum	600	550	-10	5

TABLE 7 - THERMAL TRANSIENT DESIGN CONDITIONS FOR INTERMEDIATE HEAT EXCHANGER[10]

Origin of Transient	Region Directly Affected	Temperature at Start of Transient, F	Temperature at End of Transient, F	Maximum Rate of Change of Temperature F/sec	Duration of Transient of Maximum Rate, sec	Total Duration of Transient (Includes time at maximum rate), sec
Reactor scrams, all cooling circuits continue to operate.	Primary sodium inlet	950	600	-30	6	120
Arbitrary reactivity insertion, all circuits operating.	Primary sodium inlet	900	1200	+10	10	120
	Primary sodium outlet	600	900	+ 5	20	120
One secondary sodium pump stops, interlock fails, primary sodium pump in same circuit continues to operate, other circuits operating.	Primary sodium	600	950	+30	6	15
All pumps in one circuit stop, check valve fails to close, flow reverses in IHX, other circuits operating.	Primary sodium inlet	950	600	-30	8	20
All pumps operating, feedwater cold slug	Secondary sodium inlet	550	450	- 2	10	120
All pumps operating, loss of feedwater	Secondary sodium inlet	550	900	+30	6	15
One primary sodium pump fails, interlocks fail, the other pumps in the same circuit continue to operate, the check valve closes, other sodium circuits operate.	Secondary sodium outlet	900	550	-70	2	10

TABLE 8 - THERMAL TRANSIENT DESIGN CONDITONS FOR PRIMARY SODIUM PUMP [10]

Origin of Transient	Region Directly Affected	Temperature at Start of Transient, F	Temperature at End of Transient, F	Maximum Rate of Change of Temperature F/sec	Duration of Transient at Maximum Rate, sec	Total Duration of Transient (Includes time at maximum rate), sec
Arbitrary reactivity insertion, all circuits operating	Primary sodium inlet outlet, and pump impeller	550	1100	+10	10	120
One secondary pump stops and interlock fails, primary pump in same circuit continues to operate, other circuits operating	Primary sodium pump tank, inlet and outlet nozzle pump impeller, and check valve.	550	950	+30	6	15

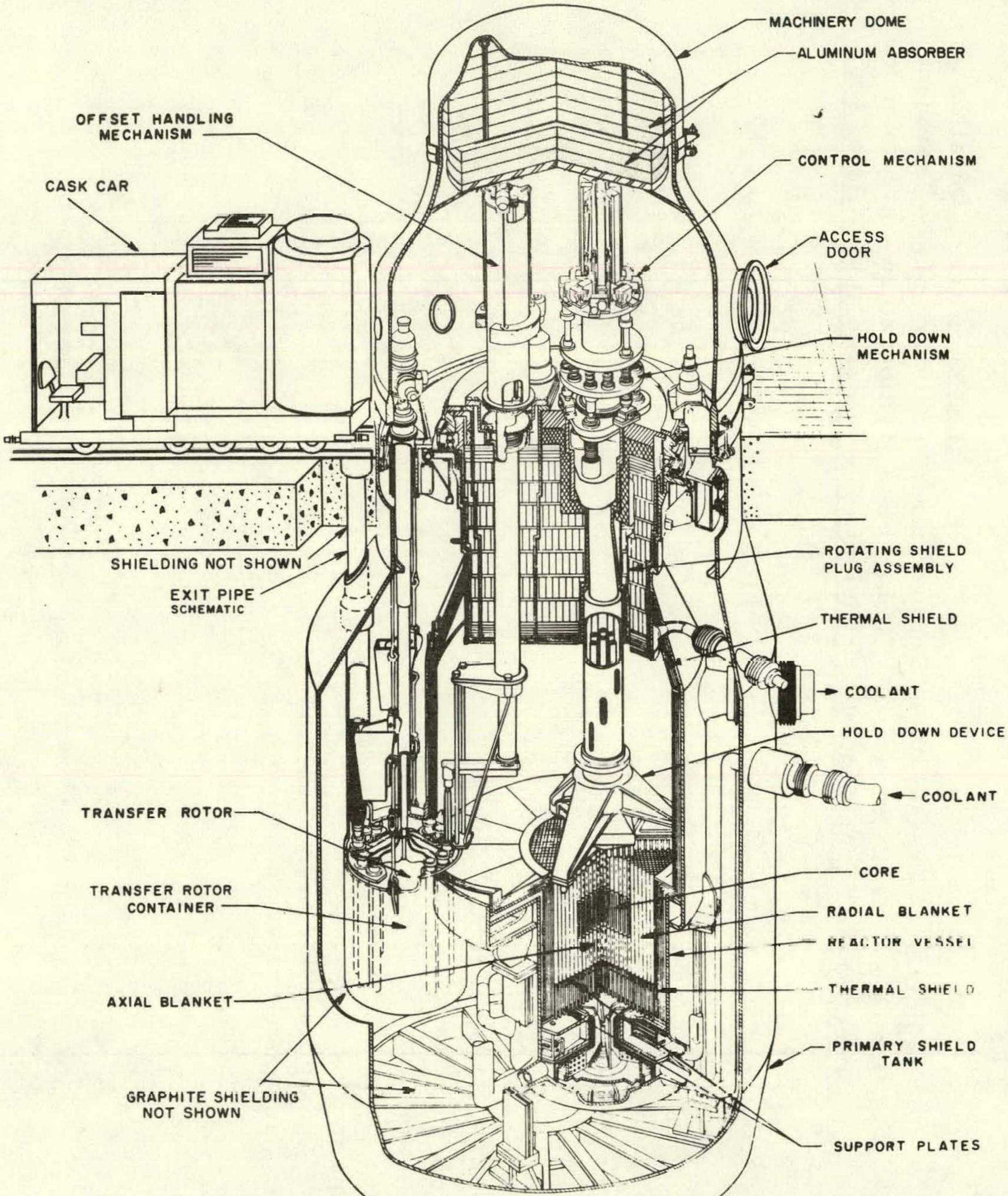


FIG. 24 PERSPECTIVE VIEW OF REACTOR

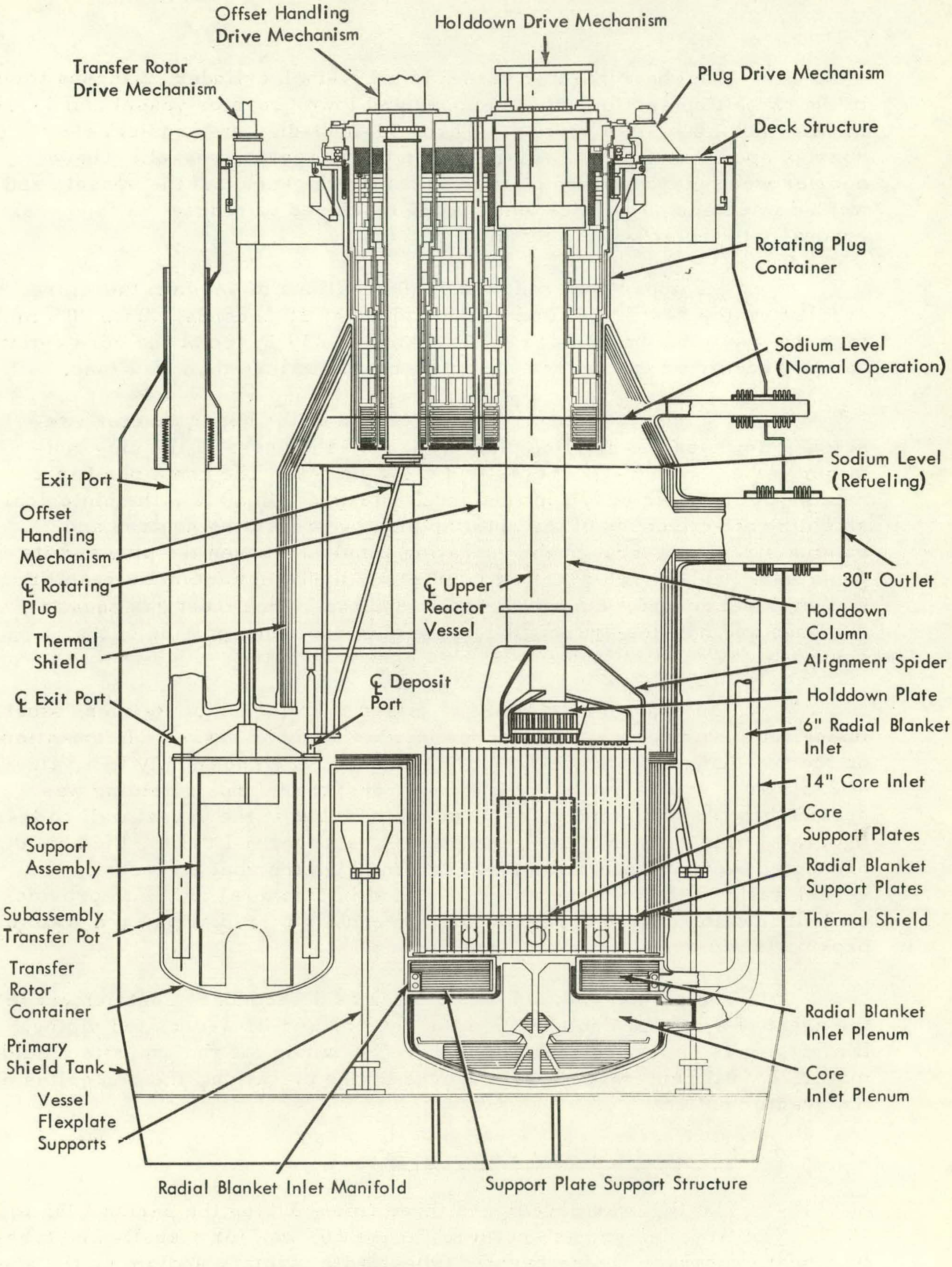


FIG. 25 REACTOR VESSEL ELEVATION

The upper reactor vessel is a 194-inch cylinder, 2 inches thick in the transition section between upper and lower reactor vessel and 1-1/2 inches thick above this section. Three 30-inch-diameter sodium outlet nozzles are located in the upper section of the reactor vessel. These nozzles were pierced from plate of the same thickness as the vessel, and welded to extension nozzles which were machined to reduce the thickness gradually to the 3/8-inch thickness of the pipe.

The upper pool contains 15,000 gallons of sodium; therefore, at full flow the residence time of the sodium is 30 seconds. With 70% mixing, the maximum transient rate of change of 250 F/sec at the core outlet can be reduced at and beyond the outlet nozzle to less than 15 F/sec.

The plug container is an extension of the upper reactor vessel which houses the rotating plug. The wall thickness of the plug container varies from 1-1/2 inches to 1-3/32 inches. The container has a maximum diameter of 115 inches and is stepped to maintain the biological shielding effectiveness of the rotating plug by preventing neutron and gamma streaming through the operating annulus between the plug and its container. The 6-inch overflow nozzle and pipe which maintain reactor sodium level constant during operation and the 2-inch inert gas equalizing lines and nozzles are located in the plug container portion of the vessel.

The radiation shielding consists of 11 inches of stainless steel inside the lower vessel and 5 inches inside the upper vessel. Information on the irradiation resistance of stainless steel was known only to a value of 2.5×10^{21} equivalent of 0.1 Mev neutrons; therefore, shielding was provided to limit the total integrated neutron flux at the vessel wall to less than this value over the life of the plant. As a thermal baffle, the shielding prevented thermal transients from causing high thermal stresses in the vessel wall. The shielding was subdivided into several plates to provide coolant flow to remove heat generated internally by radiation, and also to provide thin members for stress purposes.

The primary shield tank serves as a secondary containment in the event of a loss of coolant from the reactor and its associated piping. It also serves as a regulated atmosphere container for the graphite shielding material. All penetrations were moved to one elevation: the centerline of the reactor outlet.

2. Intermediate Heat Exchangers

The IHX was redesigned three times during the period 1955 to 1959. The first design, as shown in Figure 26, was for a shell- and tube-type heat exchanger, using bayonet tubes, with primary sodium on the shell side and secondary sodium in the tubes. This design was not adopted be-

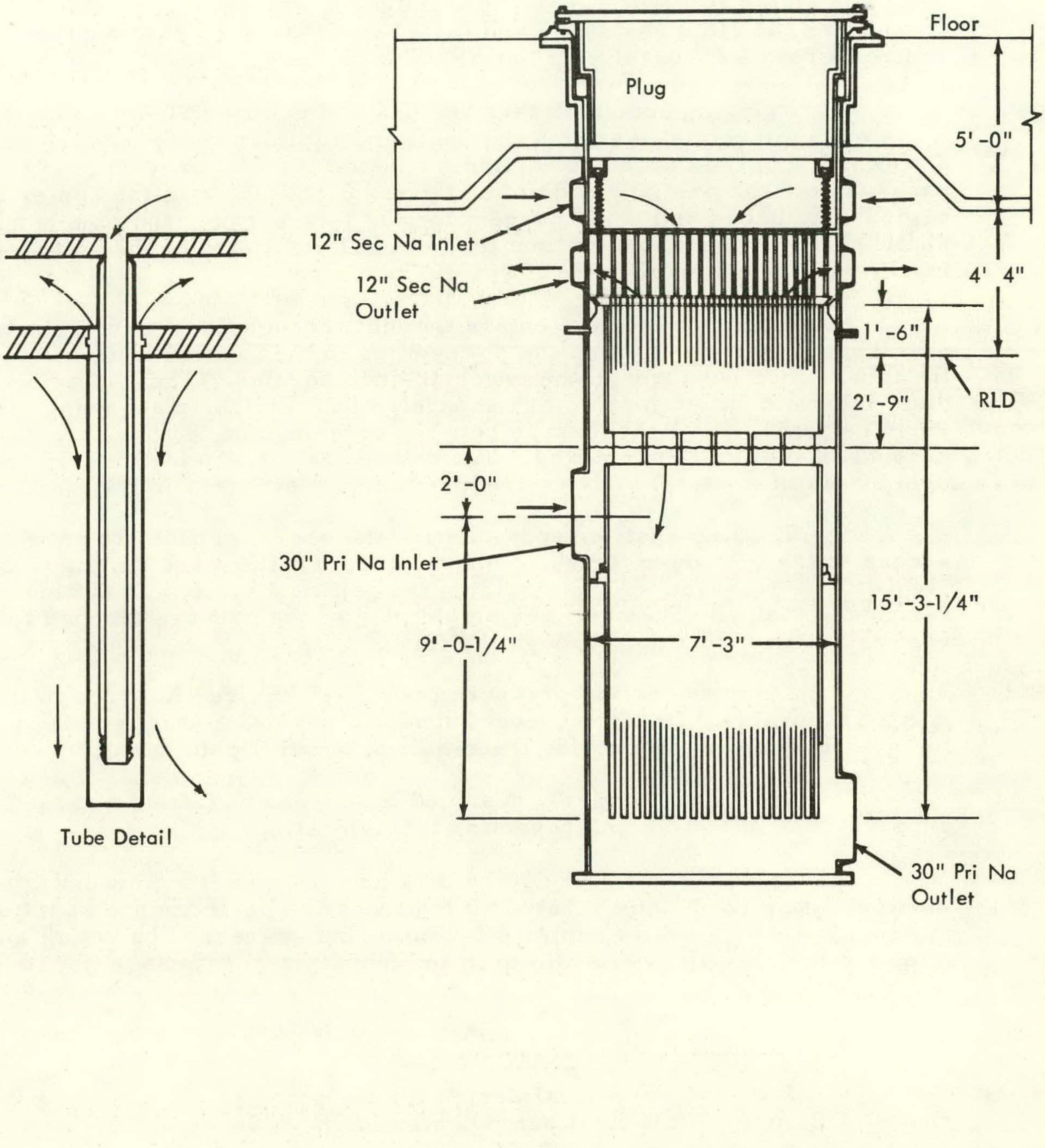


FIG. 26 INTERMEDIATE HEAT EXCHANGER, INITIAL DESIGN

cause of a lack of operating experience. The design using a central downcomer and bottom tube sheet was selected as the reference design. In this arrangement, Figure 27, the secondary connections penetrated the shield plug; they had to be cut to remove the bundle.

The final arrangement of the IHX, a counterflow, shell- and tube-type unit with primary sodium on the shell side and secondary sodium in the tubes, is shown in Figure 28. The three units were designed; fabricated from Type 304 stainless steel; and tested in accordance with Section VIII, Unfired Pressure Vessels, of the ASME Boiler and Pressure Vessel Code.

Primary sodium enters the IHX at the upper 30-inch pipe and flows circumferentially around the shell and radially inward before passing downward. Three concentric cylinders inside the unit provide four flow passages for the primary sodium and prevent crossflow in the center bundle region. A shroud and seals prevent bypass flow between the tube bundle and shell. The primary sodium then flows radially outward at the bottom of the bundle and leaves through the lower 30-inch pipe.

Secondary sodium enters the unit through the upper 12-inch nozzles, flows down through the downcomer to the floating head, up through the tubes, then out through the lower 12-inch nozzles. The secondary sodium inlet and outlet nozzles are separated by a divider plate using piston rings as seals. A slight leakage between entering and leaving primary and secondary sodium is permitted. The downcomer is shielded inside and outside by stainless steel plates to reduce thermal stresses in the downcomer.

The IHX shell extends through the operating floor to permit access to the top cover for removing the tube bundle. The units are designed to be removed without draining the primary system or cutting the secondary piping. The shell and shield plug of the unit are stepped to prevent radiation streaming through the floor.

A low shell-side pressure drop was required in order to maintain a reasonable differential level between the reactor and primary pump for gravity flow and to provide the required NPSH for the pump.

The use of properly designed impingement baffles and tube supports was effective in preventing tube vibration.

Table 9 shows the design data specified for the unit under 430-Mwt operating conditions. Table 10 indicates the performance specifications at 430-Mwt conditions as supplied by the manufacturer. The actual and effective tube lengths were left up to the manufacturer and are not covered in the above tables.

3. Primary Sodium Pumps

The 1956 conceptual design for the primary sodium pump [8] is shown in Figure 29; the final pump design is shown in Figure 30.[9,10]

The pump was designed for a flow of 4.84×10^6 lb/hr, which corresponds to a 10% excess over the 300-Mwt rated flow. At the maximum design temperature of 1000 F, this corresponds to 11,800 gpm. Table 11 presents the design data for the pump.

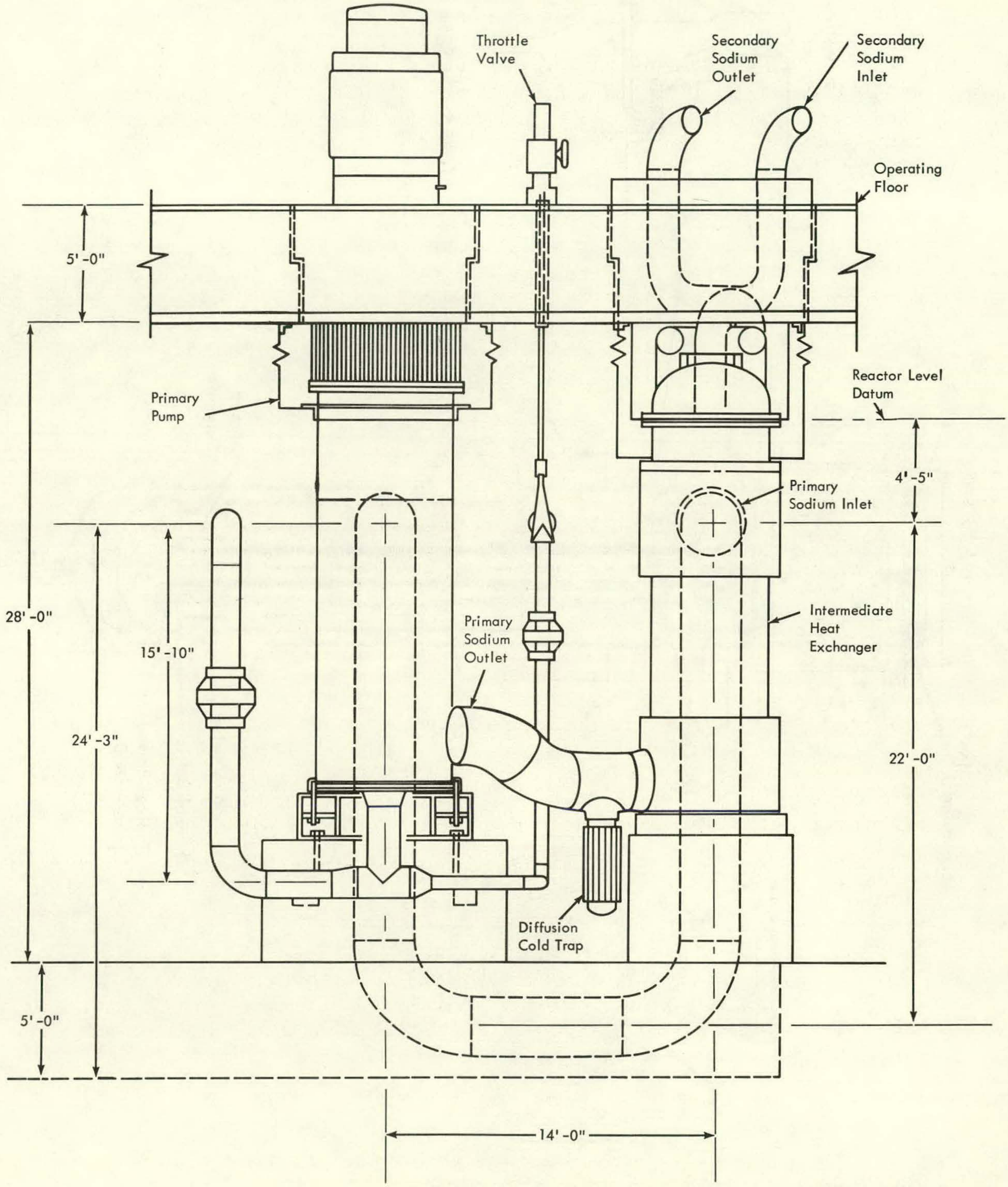


FIG. 27 INTERMEDIATE HEAT EXCHANGER, SECOND DESIGN

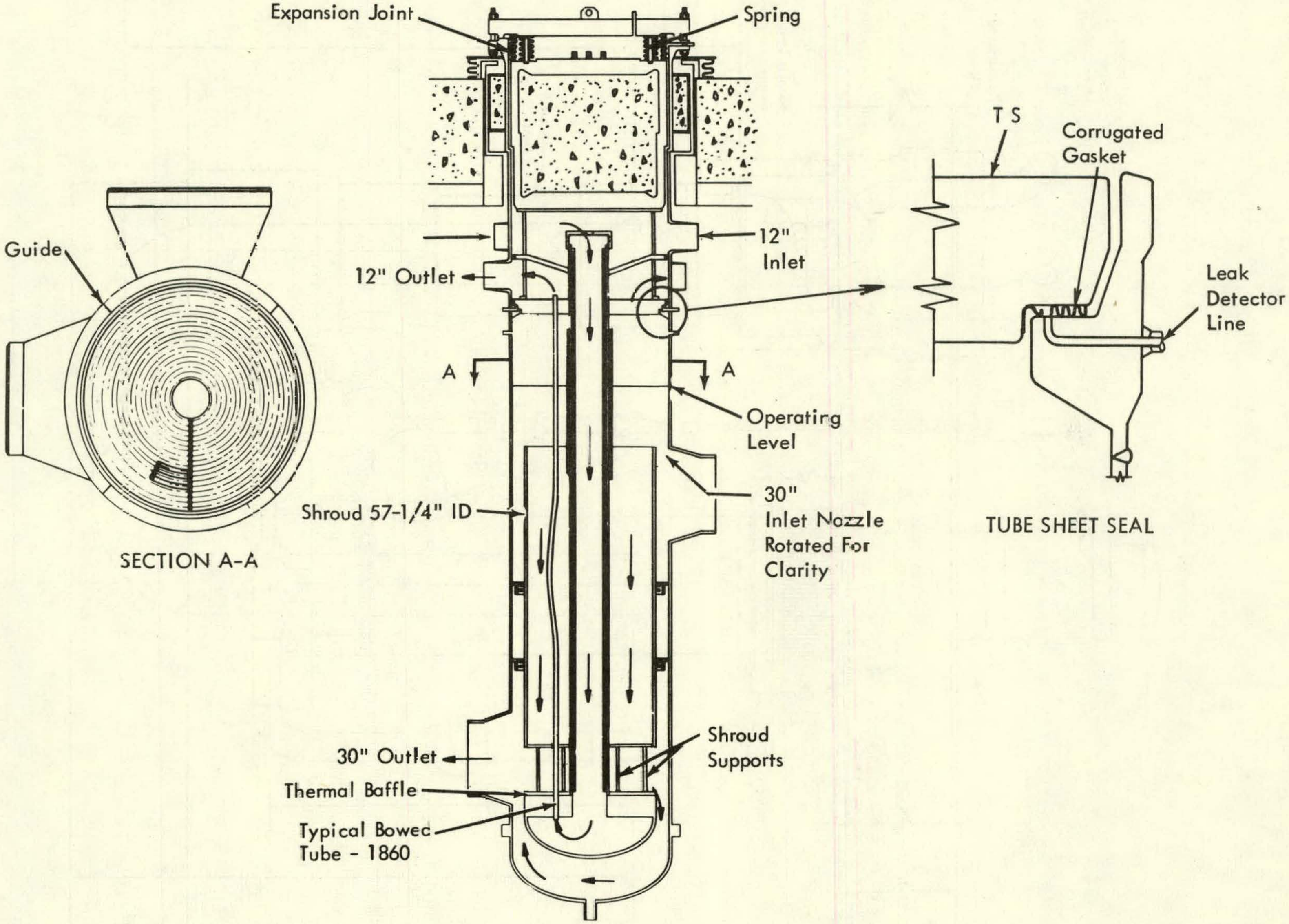


FIG. 28 INTERMEDIATE HEAT EXCHANGER, FINAL DESIGN

TABLE 9 - DESIGN DATA, INTERMEDIATE HEAT EXCHANGER, 430 Mwt CONDITIONS

Heat Transferred per Unit, Btu/hr	489×10^6
Heat Transfer Surface, ft^2	5840
Overall Heat Transfer Coefficient, Btu/hr-ft^2-F	1048
Shell-Side Fluid	Na
Flow rate, lb/hr	5.3×10^6
Temperature in, F	900
Temperature out, F	600
Pressure loss, ft of Na	2.9
Fluid in Tubes	Na
Flow in tubes, lb/hr	5.3×10^6
Temperature in, F	520
Temperature out, F	820
Pressure loss, ft of Na	12.8
Material, tubes and shell	Type 304 SS
Number of Tubes	1860
Tube Size	7/8-in. OD x 0.049-in. wall

TABLE 10 - PERFORMANCE SPECIFICATIONS AT 430-Mwt CONDITIONS*

	Shell Side	Tube Side
Flow, lb/hr	5,320,000	5,290,000
Gravity, lb/ft^3	53.3	54
Viscosity	.277	.3
Temperature, in, F	900	520
Temperature, out, F	600	820
Operating Pressure	-0.5 water gauge	60 psi
Velocity, ft/sec	3.37	4.56
Pressure Drop	2.95 ft Na	4.7 psi
Heat Exchanged, Btu/hr	489,000,000	
MTD, corrected, F	80	
Transfer Rate, Service, Btu/hr/ft^2/F	978	
Transfer Rate, Clean, Btu/hr/ft^2/F	1,200	
Effective Surface, ft^2	6,250	

* Alco Products Co., Inc. Exchanger Specification Sheet #16, dated 8/30/60.

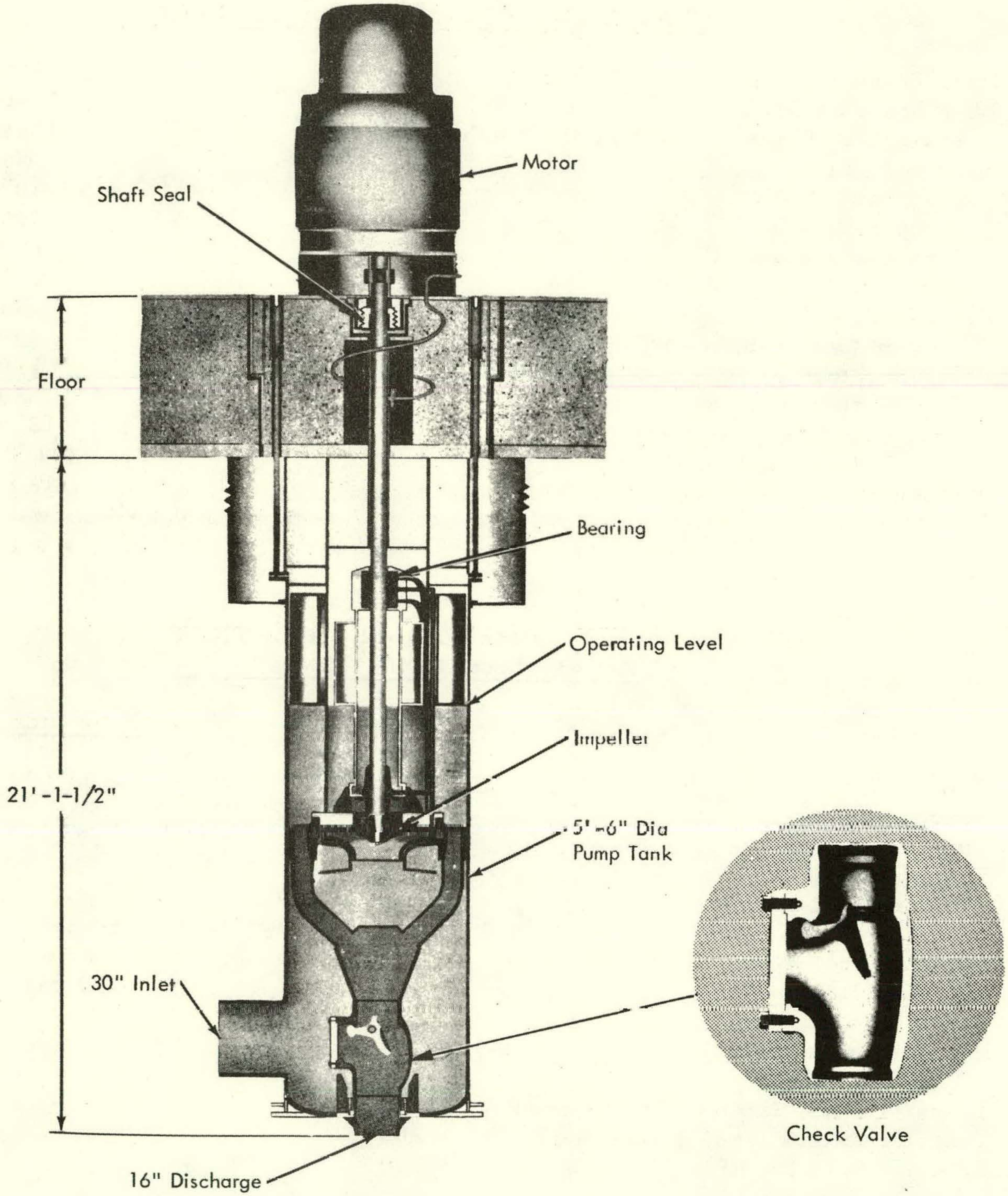


FIG. 29 PRIMARY SODIUM PUMP, REFERENCE DESIGN

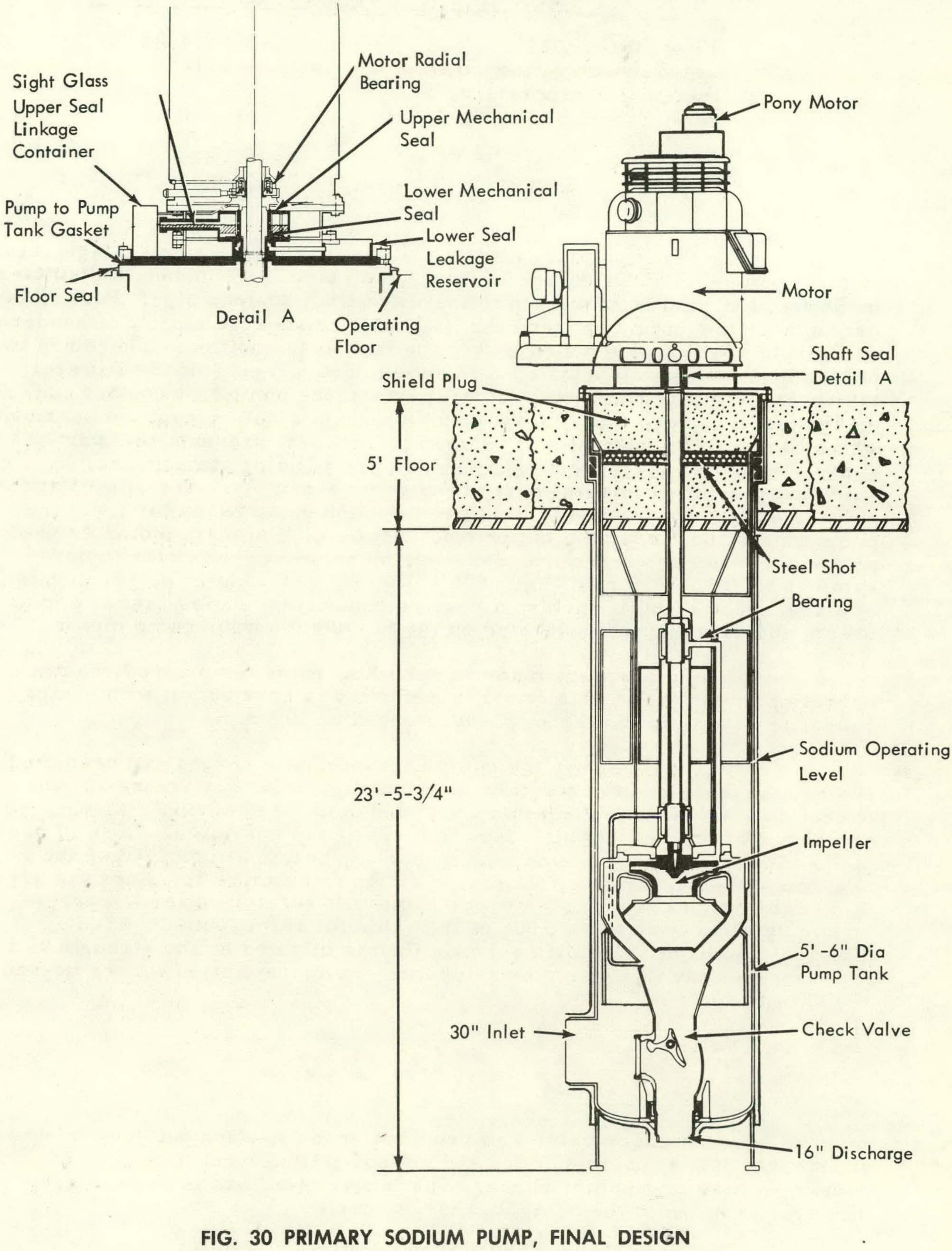


FIG. 30 PRIMARY SODIUM PUMP, FINAL DESIGN

TABLE 11 - DESIGN DATA, PRIMARY SODIUM PUMP

Flow, lb/hr x 10^6	4.84
Total Dynamic Head, psi	110
Pumping Temperature, F	
Minimum	300
Maximum	1,000
Speed, rpm	875
Capcity, gpm (@ 1000 F)	11,800

The three primary pumps are vertical-shaft, single-stage, centrifugal pumps mounted in tanks. Each pump tank is 64 inches in diameter by 26 feet long and is connected to the IHX with a 30-inch pipe. Suction is drawn from the pump tank, and the five volute discharge pipes are headered to a single 16-inch pipe which leaves the tank at the bottom. There are two hydrostatic bearings lubricated with sodium and a mechanical shaft seal lubricated with fluorocarbon oil. All parts of the pump that contact sodium or sodium vapor are made of Type 304 stainless steel, except the bearings which have a Colmonoy inlay. The pump shaft seal prevents the inert gas above the sodium level from escaping into the building atmosphere. A shielding plug is an integral part of the pump assembly. The slip fit at the discharge pipe makes it possible to remove the pump assembly from the pump tank without draining the primary system. The main motor drives are liquid-rheostat controlled, 3-phase, 60 cycle, wound-rotor motors rated at 1000 hp, 900 rpm, and 4800 volts. A pony motor drive, consisting of a gear-head electric motor, a housing containing driving gears, and an overriding cam clutch, is located on the top of each main pump motor.

Provisions were made to vary flow rates for planned reactor operating levels (see Table No. 4). Refueling is carried out with pumps operating at a minimum recommended speed of 300 rpm.

Figure 31 shows the pump characteristic curves and predicted system pressure drop curves are given in Figure 32. In Figure 32, the operating point (A) and the minimum speed point used during refueling (B) are shown on the curve which depicts three-pump operation. Loss of one pump will cause the two remaining pumps to operate at point (C) of the two-pump operation curve. Operator action is required to return the system to the original value of flow per pump (D); reactor power is correspondingly reduced to two-thirds of the value for three-pump operation. The upper limit of head for a specific flow is dictated by the strength of a fuel subassembly wrapper tube as shown. Pump cavitation occurs beyond 15,000 gpm.

4. Valves

a. Check Valves

A check valve was provided at the 16-inch outlet nozzle as an integral part of each primary sodium pump to prevent back flow of sodium in case of a pump failure. The initial valve was of the balanced disc type, welded to the pump discharge nozzle.

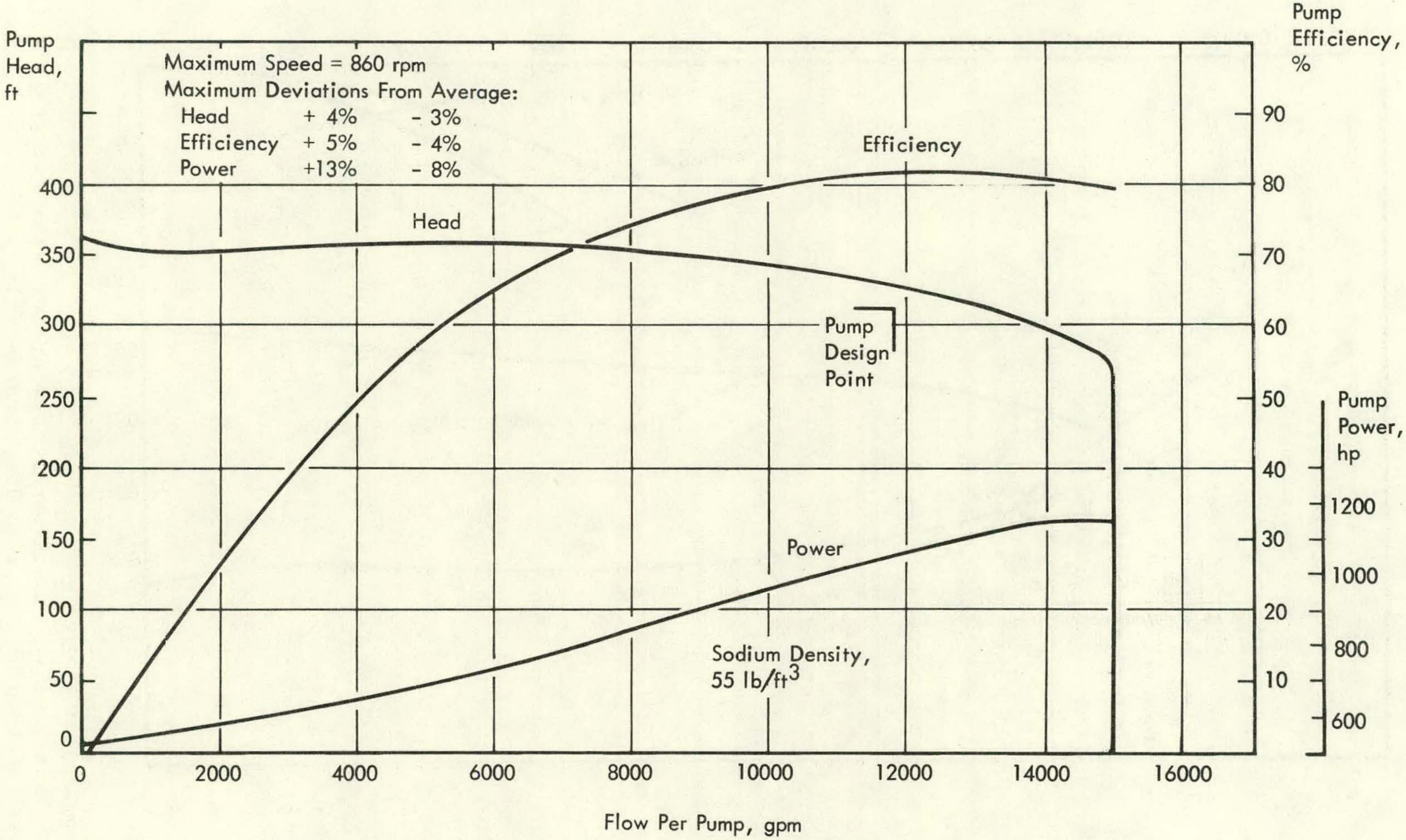


FIG. 31 AVERAGE FULL-SPEED CHARACTERISTICS OF THE PRIMARY SODIUM PUMPS

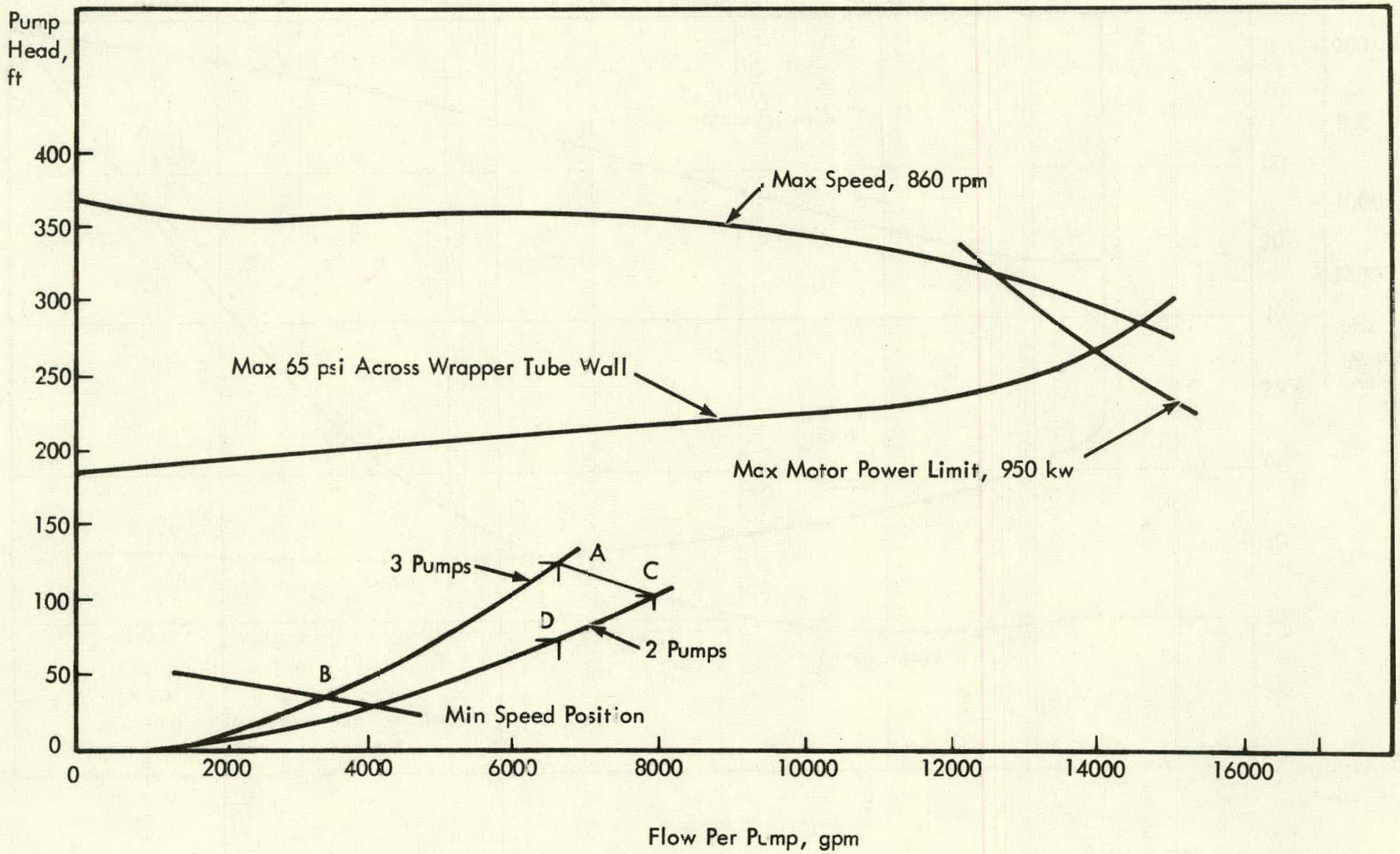


FIG. 32 PRESSURE DROPS FOR THE PRIMARY SODIUM PUMPS

During preoperational testing of one primary loop, sodium hammer and pipe movement were noted under certain conditions. As a result, the original check valves were replaced by improved tilting disc-type check valves. The valve body was connected by a bolted flange. The valves, fabricated of Type 304 stainless steel, feature an enlarged body casting, spring loaded hinge pins, and a dashpot to control closing. All metal-to-metal contact surfaces are provided with Stellite facings. To avoid an accumulation of sodium impurities, channels are provided for sodium circulation along contacting surfaces (see Section VII, A and B for details).

b. Throttle Valves

The final design of the blanket throttle valve is shown in Figure 33. This valve is designed so that the entire mechanism can be withdrawn through a plug in the operating floor without draining the primary system. This valve controls the amount of coolant flow to the radial blanket over a range of 5 to 20% of the total flow. A restriction in the valve prevents complete closure which would cause overheating of the blanket material.

A leak detector was provided between the primary and secondary bellows, and the flange was seal-welded for leaktightness. The floor plug was supported on hangers and was used as a relief plug for the below-floor atmosphere.

5. Primary Piping System Arrangement

In the arrangement of the primary system loops (Figures 34 and 35), the primary pumps are the anchor points in the system. The primary pumps were located as close to the reactor as possible, since the distance between anchors determined the amount of flexibility required. However, between these two anchors space was required for the radiation shielding. Direct streaming from the 30-inch lines penetrating the shield had to be prevented. The expansion loop was shifted to a vertical plane approximately perpendicular to the line between the anchors to keep this distance to a minimum. The bottom run of the loop passed through a trench under the secondary shield wall to make use of available space in the building.

Despite the 22-foot vertical length of the 30-inch expansion loop and the innate flexibility of thin-walled piping and elbows, the moments on the reactor nozzle were in the range of 50,000 ft-lb during the cold erected condition. The moment on the reactor nozzle was the criterion for flexibility requirements, whereas piping stress was never a problem. The piping stress under the most adverse conditions was 12,000 psi, the allowable stress range being 25,500 psi.

Close-coupling of the pump and IHX tanks with a roller support on the IHX tank proved to be practical for horizontal movement. However, for vertical movement it was necessary to have a minimal differential growth between the skirt supports on the two tanks. This proved to be a constant concern because of the inflexibility of the short section of 30-inch pipe join-

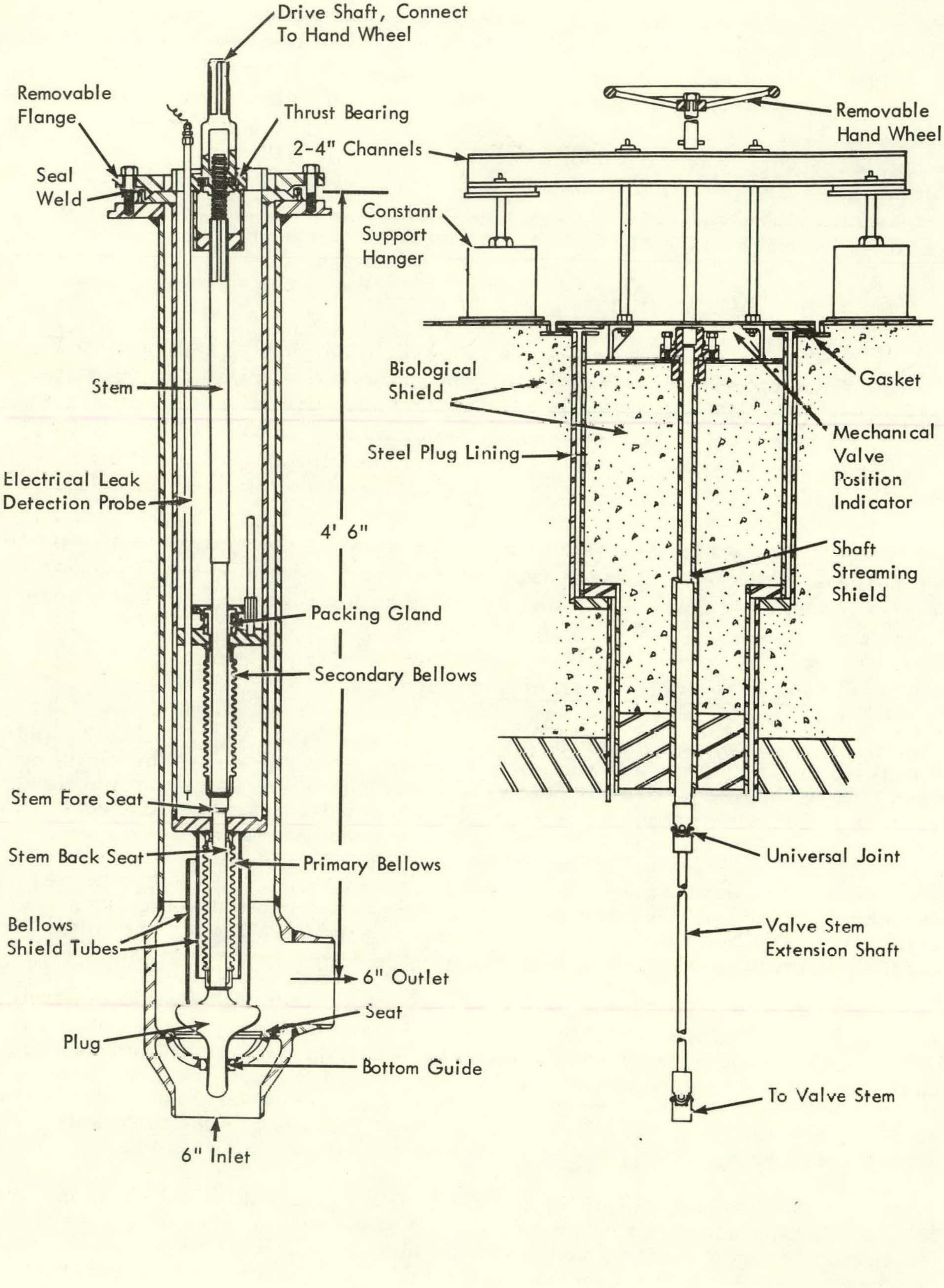


FIG. 33 THROTTLE VALVE

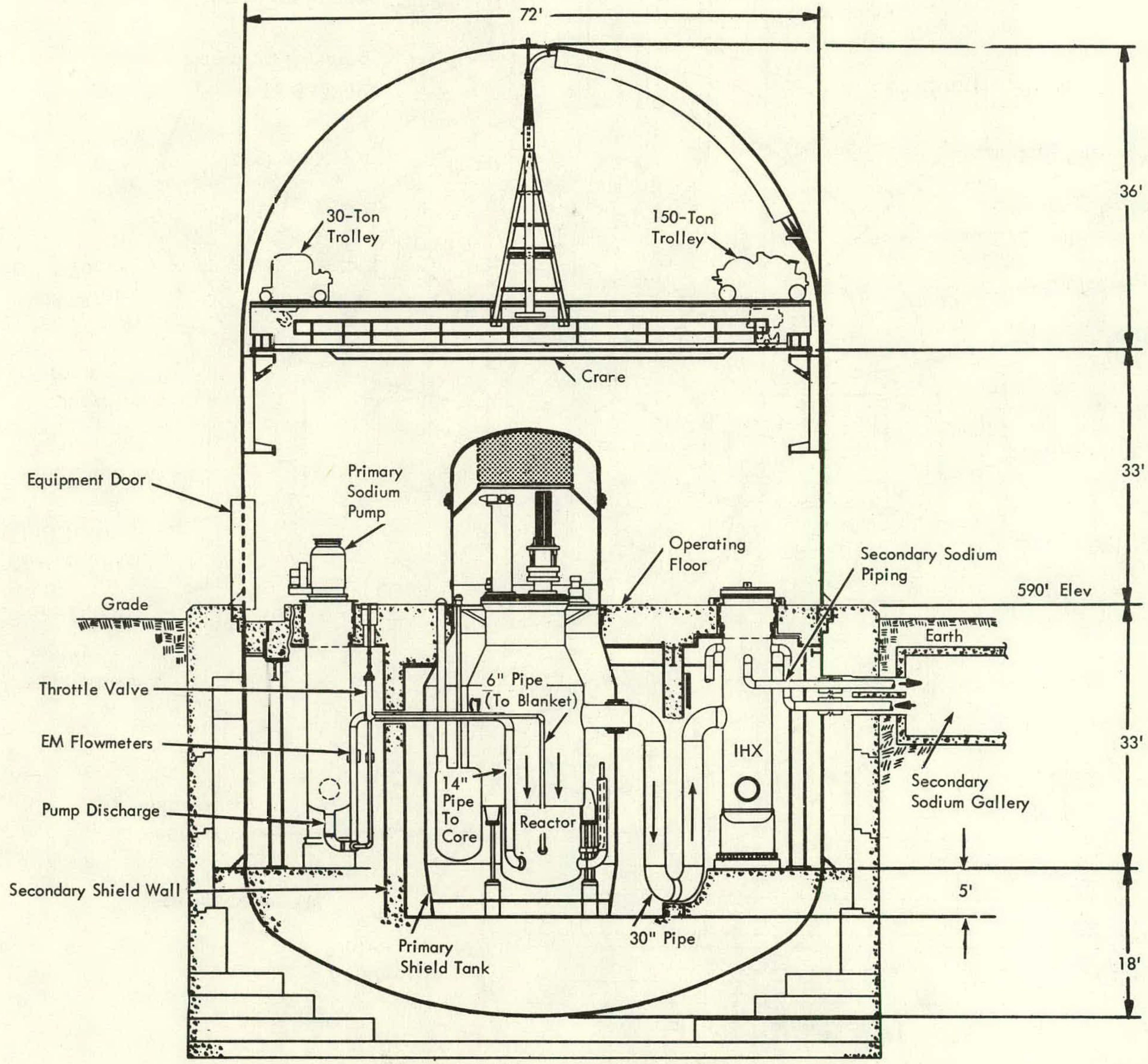


FIG. 34 ELEVATION OF PRIMARY SYSTEM

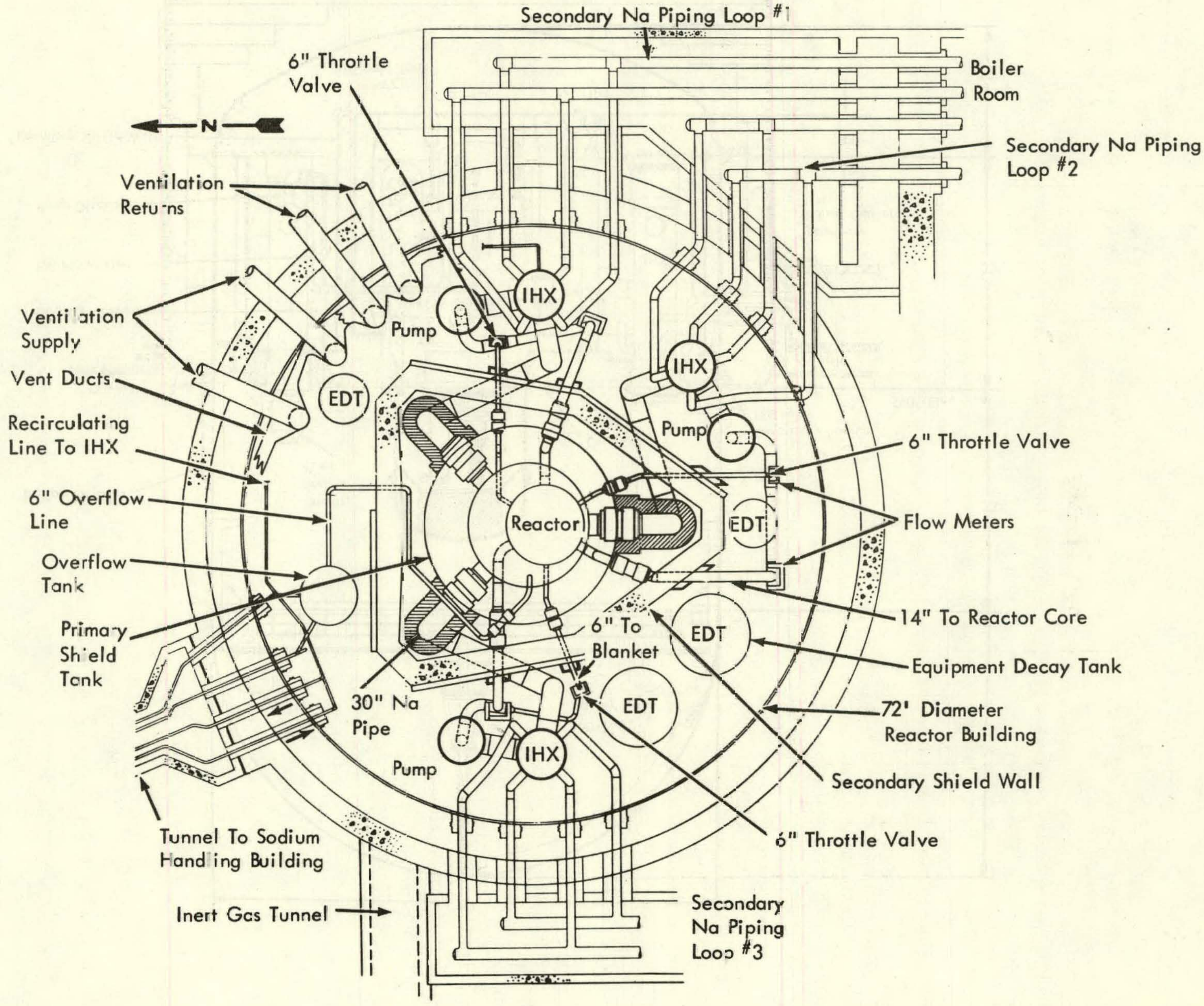


FIG. 35 PLAN VIEW OF PRIMARY SYSTEM

ing the two tanks. An allowable differential growth of 15 mils between the two tank skirts was established by stress calculations, and several thermocouples were located on each pump and IHX skirt to monitor the temperatures used in computing the differential growth. At the time of installation and the first heatup, insulation was added and removed until the growth was equalized. By control of the heaters, this criterion was met under operating conditions.

6. Component Supports

a. Reactor Vessel Supports

The unsymmetrical configuration of the reactor vessel presented a unique support problem. Close alignment was required between the upper and lower sections of the vessel for fuel handling purposes, which called for a support structure that would not distort under operating or refueling conditions. Normal dead loads consisted of the rotating plug and its associated mechanisms, the vessel and its contents, and the transfer rotor. In addition to the normal dead loads, there were live loads which consisted of forces and moments imposed by nine primary piping connections, as well as the thermal and nuclear heating effects. The 6-inch and 14-inch piping inside the primary shield tank was supported at the top of each riser by a spring hanger attached to the wall of the PST.

With adoption of cask car unloading through a transfer rotor and an exit port, a design using flexplate support columns was selected. The eight support plates extend upward from the steel web structure of the primary shield tank base to the midpoint of the vessel or transistion deck. The 2-inch-thick, 7-foot-high plates are positioned radially around the lower reactor vessel as indicated in Figure 36. Use of a plate-type column allowed radial thermal expansion of the vessel at the top of the support column. It was soon recognized that because of unsymetrical storage of irradiated fuel elements in the reactor, uneven temperatures could be produced in the flex legs.

In a field design revision, five of the eight flexplates supporting the reactor vessel were changed to use a mounting of Belleville springs. This change was made to prevent overstressing of the vessel deck structure due to unequal growth of these support columns during operation.

b. Pump and IHX Supports

Support of the primary system components was dependent on the reactor support. It was decided that only the reactor vessel and the sodium pump would be anchored members; the IHX could be a floating member to reduce flexibility requirements of the piping system. Vertical supports were located at the same general elevation to eliminate differential vertical movements at the tank nozzles.

The proximity of both pump nozzles to the bottom of the pump tank determined the logical support location for the pump. A cylindrical skirt of the same diameter as the pump tank was selected as support for the vertical tank. The skirt was approximately 5 feet high and was fastened by anchor bolts through a bottom flange into a concrete base.

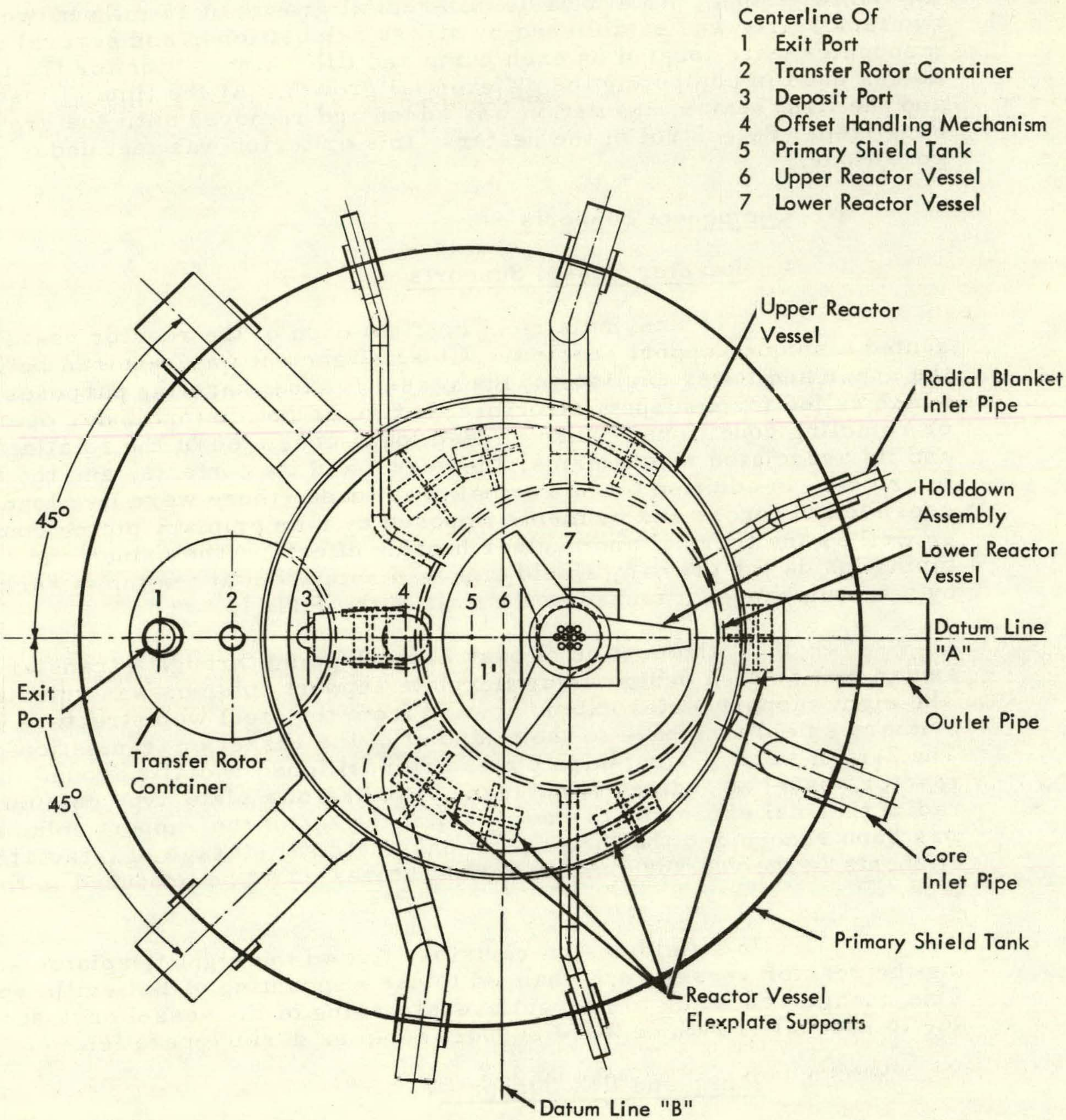


FIG. 36 PLAN VIEW OF REACTOR SHOWING VESSEL SUPPORTS

With the vertical IHX design, consideration was given to the use of constant spring supports; however, the excessive weight and the minimal vertical movement of the vessel led to adoption of the solid vertical support. To compensate for the maximum of 1 inch of travel between the pump and IHX, unidirectional rollers and guides were used on the solid support.

c. Piping Support with Secondary Containment

Use of secondary containment enveloping the primary piping posed the problem of a major hanger load change under emergency conditions. Such a change would occur when a primary system leak occurred and the 36-inch-diameter secondary piping containment support would have to change from the empty to the filled condition. Even with automatic load sensing equipment that would increase the hanger tension, damage to equipment nozzles could occur. The solution to the problem was to use common hangers for both piping and containment as shown in Figure 37 and in Appendix Figure A. 5.

7. Secondary Containment

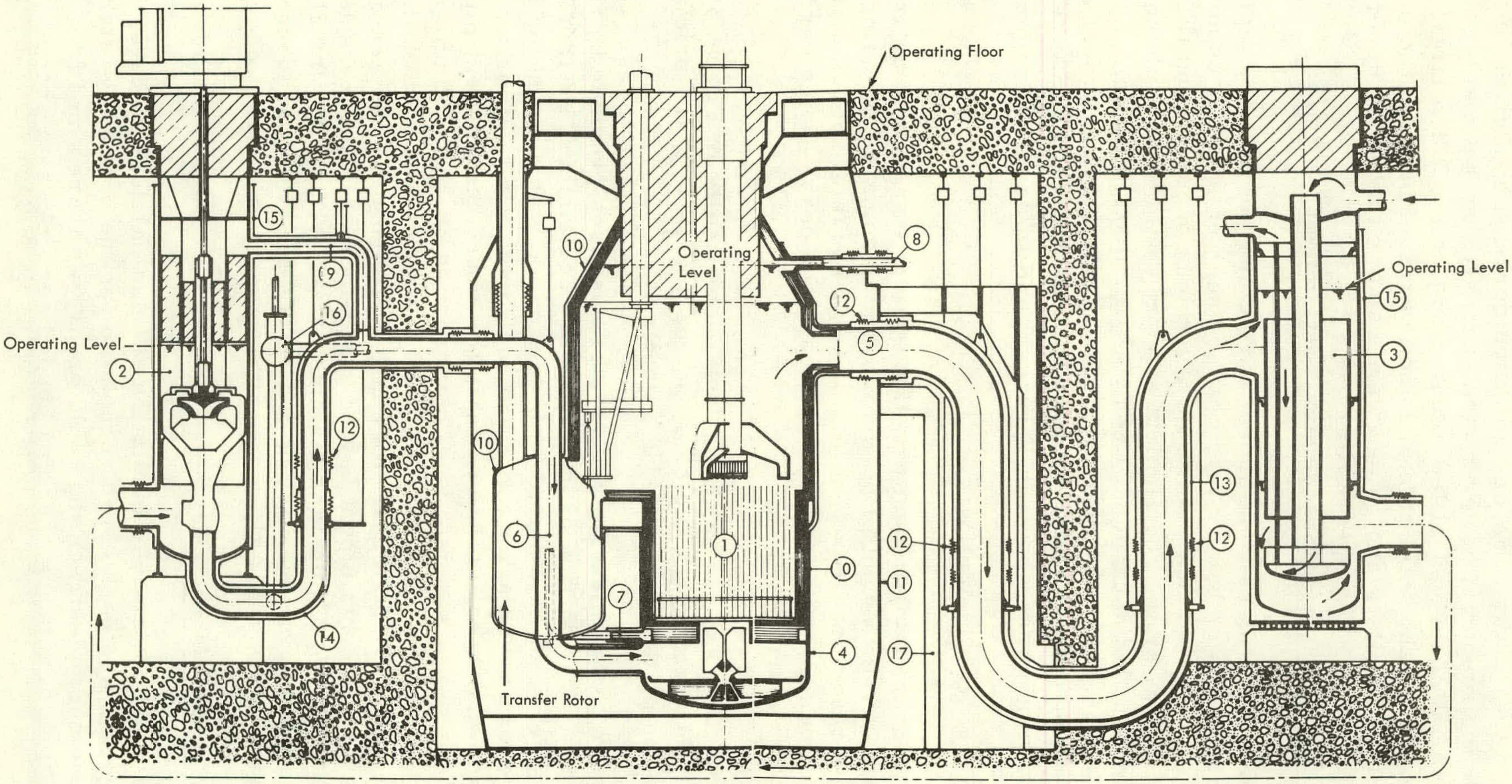
The secondary containment protects against loss of sodium from the primary system. The concept of secondary piping containment using induction heating is shown in Figure 38. An annular space of approximately 2 inches is left between the main sodium piping and the heavy-gauge carbon steel secondary containment pipe.

The secondary containment, shown schematically in Figure 39, is a welded, leaktight system that encloses the reactor vessel, the primary system piping, the pump tanks, and the IHX shells.

Secondary containment for the reactor vessel is provided by the primary shield tank. That portion of the secondary containment system outside the primary shield tank is isolated from the primary shield tank by bellows seals and is open above the maximum coolant level. In this portion of the system, 36-inch-diameter, 5/16-inch-wall secondary containment pipe fabricated of chromium-molybdenum steel encloses the 30-inch sodium piping. Carbon steel containment having a 3/8-inch wall thickness with a clearance annulus of approximately 2-1/2 inches is installed around each IHX and each pump tank. Carbon steel containment having a 3/8-inch wall thickness is also used on the pump discharge piping, and stainless steel bellows are installed at the flowmeters. The containment was applied in halves with subsequent welding of the two longitudinal seams; all such welds were dye penetrant checked.

8. Insulation and Heating

Conventional power-type insulation such as calcium silicate, fiber felt, and 85% magnesia was considered acceptable for the primary system. Only in areas where neutron streaming could occur was there a need for a high-performance-type insulation, such as bonded aerogel, as a space saving measure.



1 Reactor Core
2 Primary Sodium Pump
3 Intermediate Heat Exchanger
4 Reactor Vessel
5 Na Outlet, 30-inch diameter
6 Na Inlet, Core, 14-inch diameter
7 Na Inlet, Blanket, 6-inch diameter
8 Overflow Line
9 Siphon Breaker
10 Leak Detector Sheath
11 Primary Shield Tank
12 Bellows
13 Secondary Containment of Outlet Piping
14 Secondary Containment of Inlet Piping
15 Secondary Containment of Pumps and Heat Exchangers
16 Throttle Valve
17 Neutron Shielding

FIG. 37 SCHEMATIC OF PRIMARY SODIUM SYSTEM ELEVATION

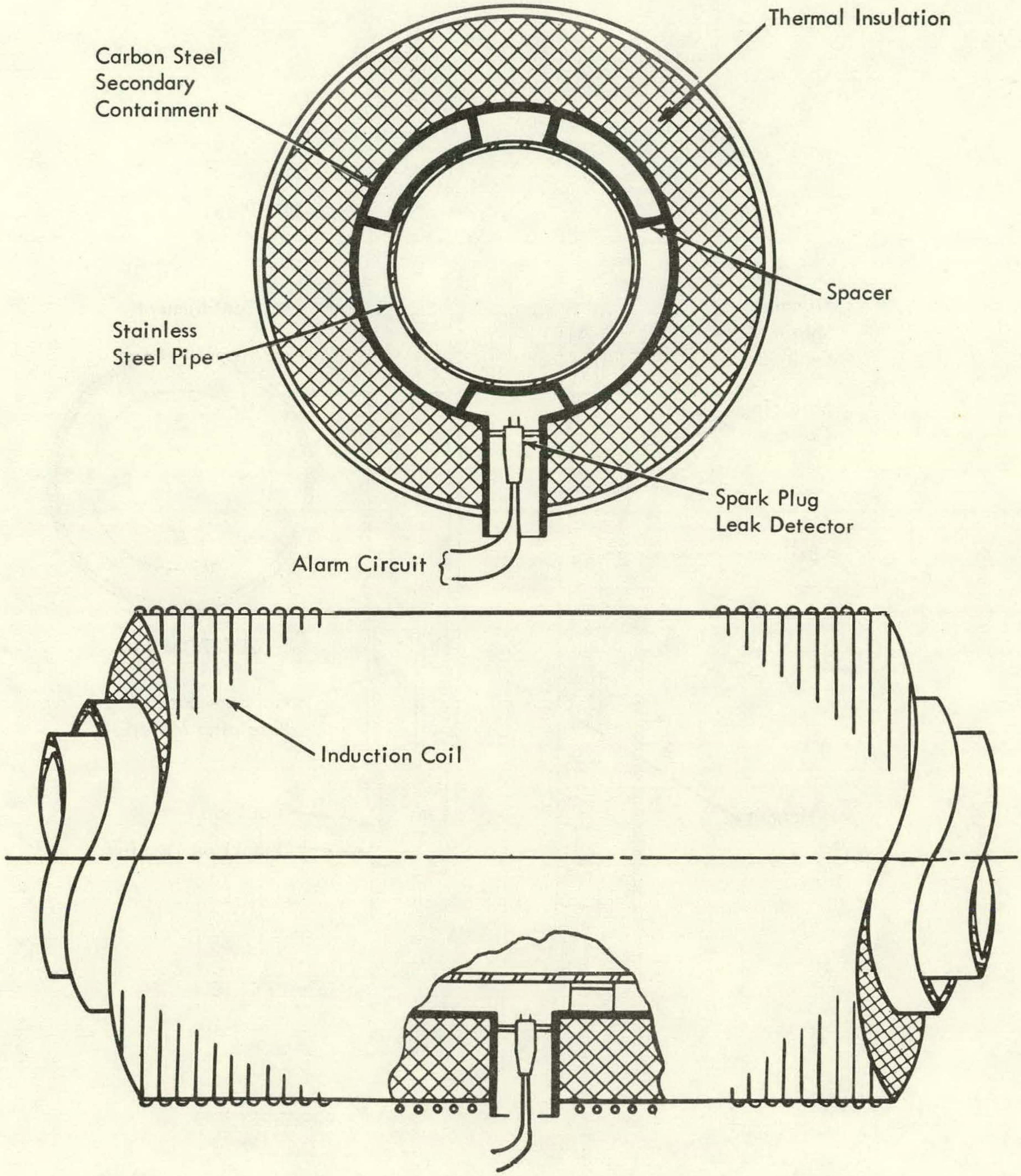


FIG. 38 TYPICAL LIQUID-METAL PIPE SECTION SHOWING SECONDARY CONTAINMENT

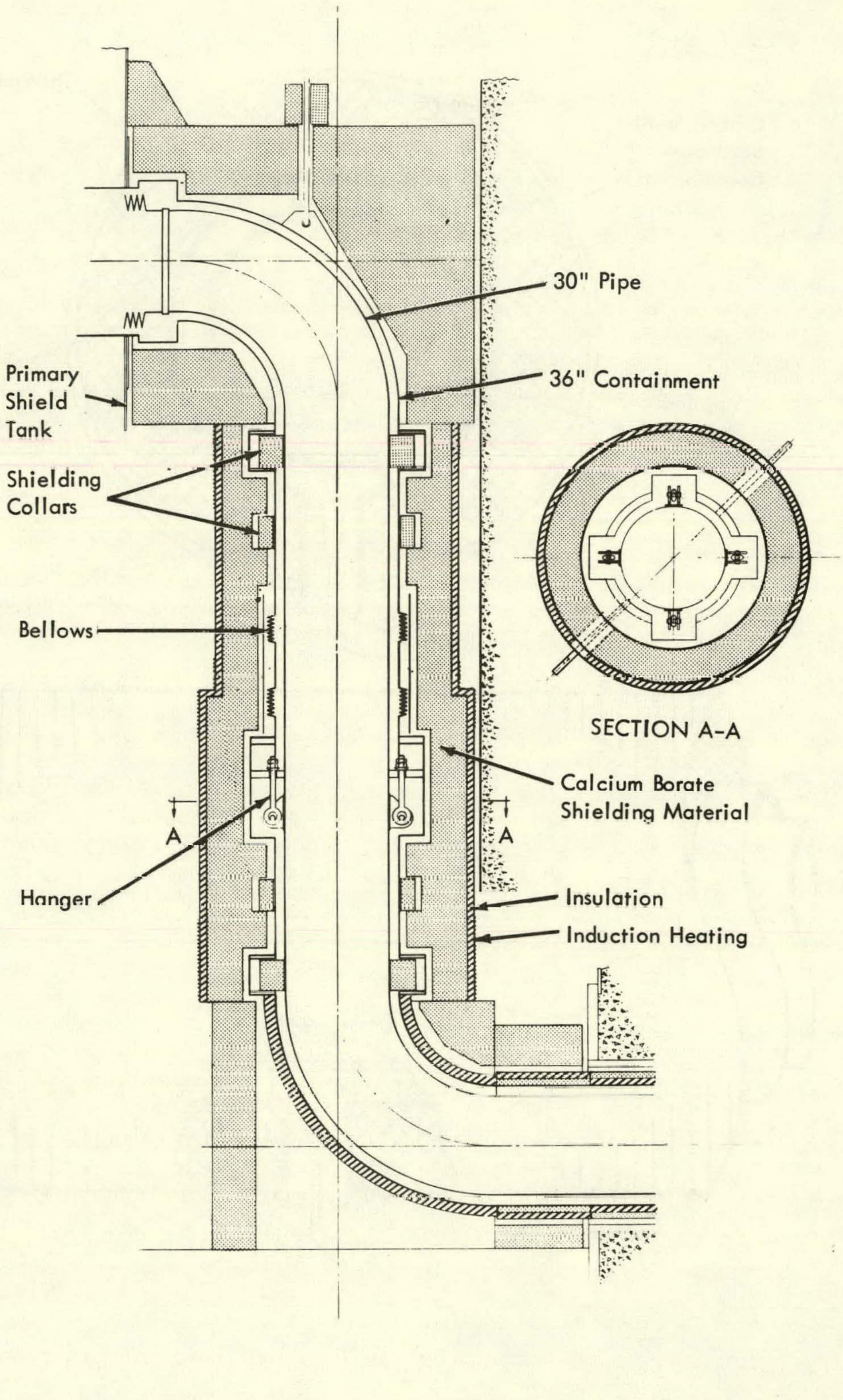


FIG. 39 SHIELDING ON 30-INCH PRIMARY COOLANT PIPE

The large diameter and nonuniform configuration of the reactor vessel made it impossible to use induction heating. The tubular-type electrical resistance heaters selected have a total capacity of about 1200 kw. They are designed to heat the reactor vessel to 400 F in 72 hours prior to filling with sodium. Since these heaters are buried in graphite and inaccessible for replacement, safety and reliability was a design requirement. The heaters were mounted on leak detector sheaths to prevent contact with the reactor vessel wall. They were designed to operate at approximately 2-1/2 times the normal operating condition of 98 volts. The heaters were radiographed and given electrical tests to ensure the highest quality possible. One hundred percent spare heater capacity was installed to provide for possible loss of heaters. In the rest of the primary system, induction heaters were selected. Though some concern was expressed at the reported low efficiency of these heaters, ease of replacement was a deciding factor in their selection.

9. Shielding

Special shielding was required around the 30-inch lines, as shown in Figure 39, for a distance of about 35 feet from the nozzle because of the neutron leakage from the reactor nozzles and their associated annuli. These shields prevented neutrons from activating the heat exchanger compartment either by leakage through the wall or through the pipe annuli. Two materials were considered: a borated diatomaceaous earth (aggregate concrete containing 2 w/o boron) and a calcium borate material containing 10 w/o boron.

The neutron shielding material was erected as a free-standing column of calcium borate sheets bolted together to form thicknesses from 9 to 20 inches. The same material was used in the shielding collars and in the wall penetrations.

E. EMERGENCY COOLING SYSTEMS

1. Pony Motors

The primary sodium pumps remain in operating during all scrams except those that are caused by multicircuit shutdown* or by loss of electric power to the primary pump motors. Thus, following many scrams, the decay heat from the reactor would be removed by the normal full sodium flow provided by the primary pumps. Analysis has indicated, however, that if the scram was caused directly or indirectly by failure of electric power to the primary sodium pumps, the maximum temperature in the fuel might rise to a level which would damage the fuel subassemblies. Therefore, a means of removing decay heat from the reactor following a complete loss of 4800-volt a-c auxiliary power has been provided by installing pony motors on the primary pump drive motors (pony motors are also installed on the secondary sodium pumps). The pony motor on the circuit 1 primary pump motor is supplied by the d-c bus, while on the corresponding pumps of circuits 2 and 3 the electric supply is the 120-volt a-c essential bus.

---

* Refer to Section IV-G, Primary System Control

Following complete loss of electric power to the main primary pump motors, a scram will be initiated by at least one of three signals, based on either the primary sodium temperature increase, the reduction in flow, or the neutron flux response. After the scram, flow decay, together with pony motor action and natural circulation effects, is capable of removing maximum decay heat without damage to the fuel subassemblies. For a scram from a steady-state heat generation of 200 Mwt, the maximum sodium temperature in the hottest channel of the Core A fuel is calculated to be 885 F if three pony motors are operating and 912 F if two are operating, while the normal maximum temperature is 981 F. Sodium temperatures decrease continually and slowly after passing through the maximum. The above temperatures are based on the primary sodium pumps driven by pony motors delivering 2000 gpm total for two-pump operation and 2160 gpm total for three-pump operation.

Each pony motor drive includes speed reduction gears and an overriding ball clutch. The output gears, driven by the pony motors, rotate at 70 rpm and 85 rpm for the primary and secondary pumps, respectively. When the main pump motors coast down to these speeds, the overriding clutches engage and the pony motors begin to drive the main motors.

The pony motors start in the following three ways: manually by a switch on the control console, automatically on the opening of the main motor breaker, and automatically from single-circuit shutdown circuitry. Six situations initiate single-circuit shutdown: feedwater demand exceeding actual feedwater flow, feedwater flow exceeding feedwater demand, high sodium temperature leaving steam generator outlet, low sodium temperature leaving steam generator outlet, primary sodium flow less than 25%, and secondary sodium flow less than 60%. To prevent feeding more sodium to the reactor, the pony motors on any given circuit will not start if the rupture disc of the steam generator in that circuit has ruptured. In the case of loss of 4800-volt auxiliary power, the first initiating signal would be the secondary sodium flow decreasing to 60% in about 2 seconds.

Dual voltage starting is employed on the a-c motors to limit inrush currents and applied torque if the pumps are started from rest with the pony motors. When the starting current is reduced to nearly running current and after one second elapses, full voltage is applied to the motors. The primary pony motor always starts first. After full voltage is applied to the primary pony motor, the secondary pony motor is started with its dual voltage starter. The delayed starting of the secondary pony motor is to reduce inrush on the bus.

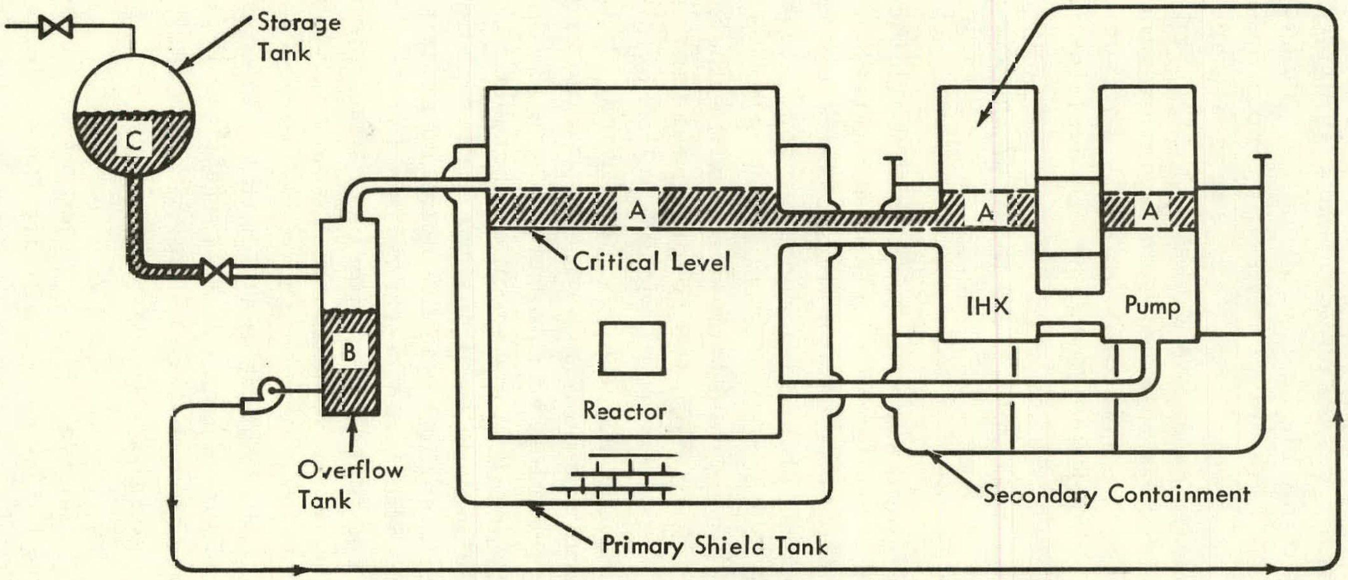
The d-c pony motors are started similarly to the a-c pony motors. Resistance starting is employed to limit the current during acceleration.

2. Reserve Sodium Supply System

If a leak occurred in the reactor vessel or in the piping inside the primary shield tank, the shield tank would serve as secondary containment; however, if the leak were sufficiently severe, the sodium level in the primary shield tank and in the reactor vessel could potentially drop to a point below the critical level (the outlet nozzle or 576.50 feet) needed to ensure natural circulation of sodium in the primary system. To bring the level up to the critical level, sodium is available from three sources (Figure 40) (a) excess sodium in the reactor, pumps, and IHX tanks above the critical level at the time of the leak; (b) sodium available from the overflow tank; and (c) a volume of sodium stored in one of the storage tanks. The sodium from the first source is already in the system and that from the overflow tank can be transferred into the system without difficulty. The sodium in the storage tank, the means of transfer, and the instrumentation make up the reserve sodium supply system.

The reserve sodium system operates through the use of signals actuated by the level indicators in the reactor, pump tanks, and overflow tank as shown in Figure 41. Alarms and annunciators in the control room warn the operator of low or high levels, and emergency levels energize either valves or the overflow pump. If a leak occurs in a primary system component within the primary shield tank, there will either be a leak detector alarm or a low level alarm from the reactor vessel or the pump tanks. The operator can, at this time, take corrective action by manually starting the overflow pump and starting a shut-down procedure. If the level continues to drop, the overflow pump will be automatically started and the sodium in the overflow tank will be transferred into the system. If the system is undergoing a cold-trapping operation at the time, there is no difficulty since the pump is already operating and the lines are already filled. If the overflow tank volume is insufficient to bring the primary system level up to the critical level, additional sodium can be transferred from the storage tank into the overflow tank. The instrumentation to initiate this transfer consists of a low level signal in the overflow tank which opens a valve in the emergency makeup line from storage tank No. 1. The storage tank is pressurized with argon over the sodium surface and both the tank and the line are kept heated so that the sodium is immediately available if needed. As a backup for instrumentation malfunction in the overflow tank, a low sodium level in the reactor will open the valve from the storage tank.

The volume of reserve sodium, when added to the sodium already in the primary system, is large enough to flood the system. The gas spaces in the reactor, pumps, IHX's, and overflow tank would accommodate 1076 cubic feet of sodium before flooding the seals at the top of the system; this volume is less than the 1442 cubic feet of sodium in the storage tank. Accordingly, a high level signal in the overflow tank is provided to close the storage tank valve. As a further precaution, high level signals to close the



Void Volume in Primary Shield Tank To 576 ft 6 in. = 2370 cu ft

Volume, cu ft

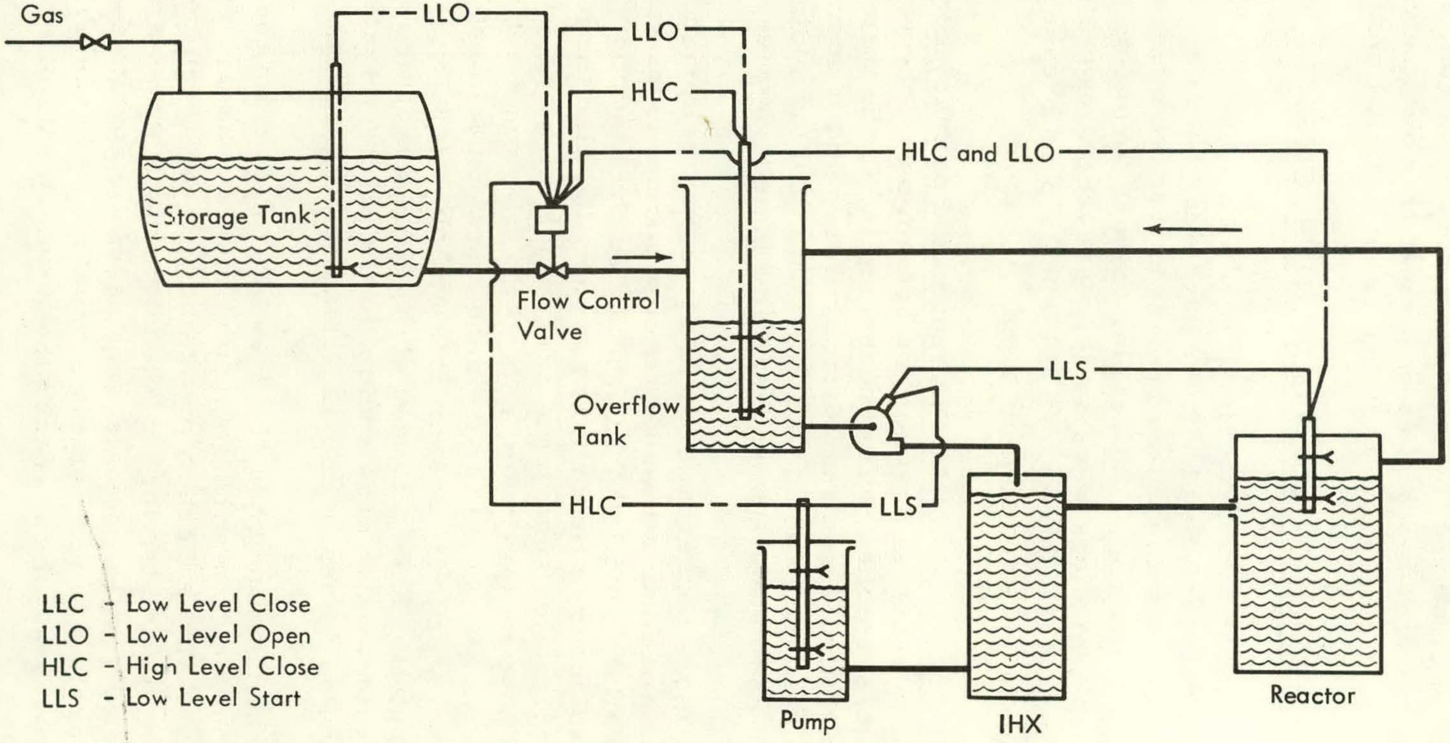
	A	B	A+B	C = 2370-(A+B)
300 Mwt - 576 ft 6 in.	572	359	931	2370-931 = 1439 cu ft
430 Mwt - 576 ft 6 in.	551	377	928	2370-928 = 1442 cu ft

For Storage Tank Volume Use 1442 cu ft

FIG. 40 RESERVE SODIUM SUPPLY SYSTEM

1. A low level in the reactor or in a pump tank will transfer the sodium into the system from the overflow tank.

2. If this volume is insufficient, the low level in the overflow tank will open the valve from the storage tank.



LLC - Low Level Close
LLO - Low Level Open
HLC - High Level Close
LLS - Low Level Start

FIG. 41 OPERATING INSTRUMENTATION, RESERVE SODIUM SUPPLY SYSTEM

valve are also installed in the gas spaces in the reactor and in one of the pump tanks. In each case alarms are also provided. In the event that the valve fails to close and the level indicators show continued rise in sodium level, the operator has sufficient time to transfer the reserve sodium to the No. 2 or No. 3 storage tanks which are normally empty.

F. DESIGN FOR LOSS-OF-COOLANT ACCIDENT

The most serious accident would be a loss of coolant from the core area of the reactor vessel. A less serious, but also intolerable accident, would be that condition where coolant is lost from the system and the sodium level drops below the outlet nozzle, thereby throttling natural circulation flow. Four provisions were made to prevent a loss of coolant accident: high penetrations of the primary shield tank, siphon breakers, secondary containment, and the reserve sodium supply system.

The loss-of-coolant accident in the primary containment was initially conceived to result in draining the sodium out of the reactor core and exposing the subassemblies. This reasoning led to the installation of siphon breakers on the 6-inch and 14-inch risers, and it dominated the design of the vessels to the extent that high rather than low piping connections were stipulated and bottom drains were eliminated wherever possible.

A more conservative loss-of-coolant criterion was later established where the minimum allowable reactor sodium level was the centerline of the 30-inch outlet, which was recognized as the minimum level for natural circulation to occur in the primary system and also the level required to keep the subassemblies submerged during fuel unloading. In addition, below this level the heat removal capability of the IHX would not be available. This criterion was the guide for the secondary containment system which will accommodate a primary system leak and not allow the level to drop below the reactor outlet. This criterion also guided the design of the reserve sodium supply system which adds sodium to the primary system in the event of a leak in the equipment within the primary shield tank.

The consequences of a loss of coolant from the primary system were considered also from the standpoint of a sodium-air reaction, and thus the below-floor atmosphere is maintained as an oxygen-depleted nitrogen atmosphere. Also, in striving for absolute leaktightness and integrity during the plant lifetime, all materials were carefully specified and nondestructive testing was applied to the materials or fabricated assemblies. This quality assurance program far exceeded the requirements of available codes and standards.

G. PRIMARY SYSTEM CONTROL[11]

The design of the Fermi control system is predicated on three basic conditions: (1) power changes will be initiated at the reactor, (2) sodium flows in the primary and secondary loops will be constant from startup to 100% of specified power levels, and (3) the normal rate of change of power will be preset in the control system. The system programs the reactor outlet temperature from a temperature set point and maintains a programmed reactor outlet temperature regardless of the action of other variables, such as actual reactor power or sodium flows. With constant sodium flows and a scheduled reactor inlet temperature, reactor power is a function of the reactor outlet temperature. The reactor outlet temperature signal is used to position the regulating rod.

The major components of the reactor control system are shown in Figure 42, and the associated tentative temperature versus power is shown in Figure 43 for 300 Mwt. The system consists of three channels: a temperature-error rate-of-change of power demand channel, a neutron flux or actual rate-of-change channel, and a regulating rod velocity demand channel. A given power level setting is represented by the temperature demand signal, which is compared with the actual reactor outlet temperature signal. The difference between the temperature demand signal and the actual temperature signal is converted to a rate-of-change of power demand. A limiter provides positive protection against excessive loading rates. The lag unit prevents a sudden change in the rod velocity demand channel if a step change is made in the rate-of-change of power demand channel.

The neutron flux or actual rate of change channel is driven by the larger of two auctioneered signals from uncompensated ionization chambers. After amplification to a level compatible with the control system, the signal is differentiated in the rate unit. The output of the rate unit is the actual rate of change signal.

The regulating rod velocity demand channel compares the neutron rate of change signal with the temperature demand signal, and its output is the error signal. Because of reactor nonlinearity with power level, the error signal is divided by the neutron flux signal in the servo divider to provide a constant control loop gain at all power levels. The output of the servo divider is the regulating rod velocity demand signal, which is one input to the control servo; the second input is actual rod velocity. The two inputs are combined, and the resulting error signal is used to supply a variable voltage to the regulating rod drive motor.

A manual-automatic transfer station is provided so that the operator can remove the regulating rod from automatic control. This enables the operator to drive the regulating rod in or out of the reactor core at a fixed rate.

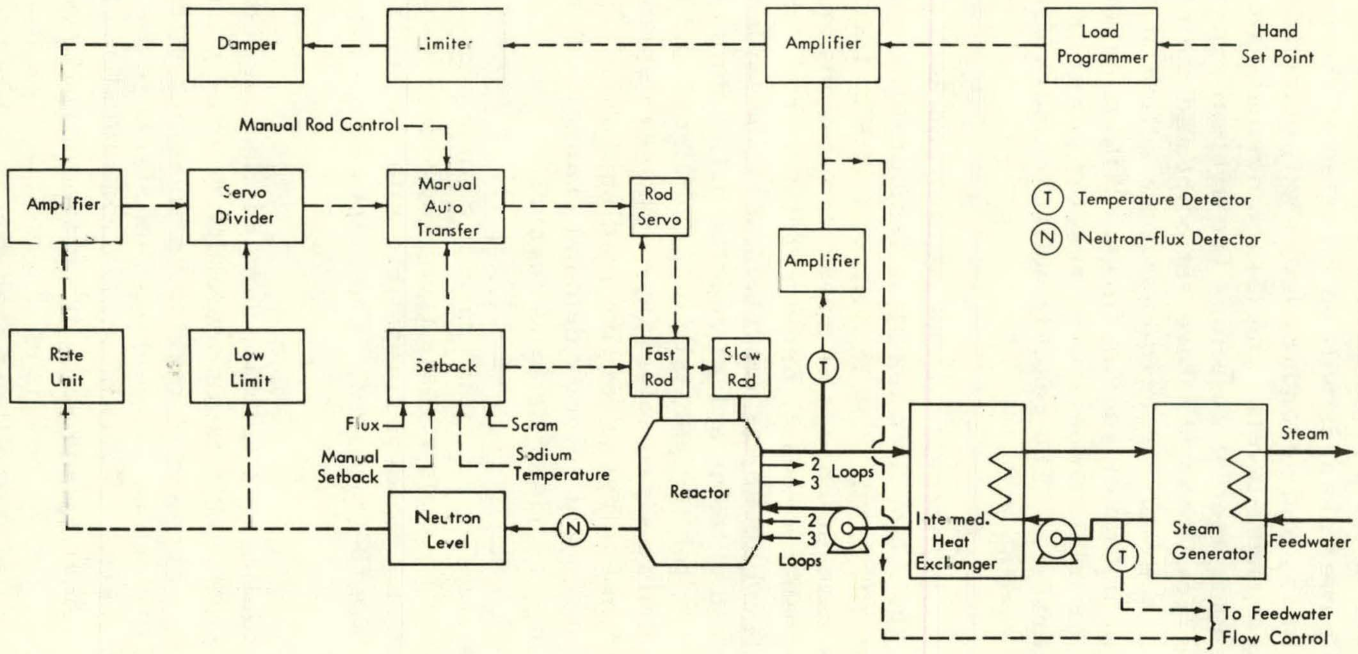


FIG. 42 SCHEMATIC OF OPERATING CONTROL SYSTEM

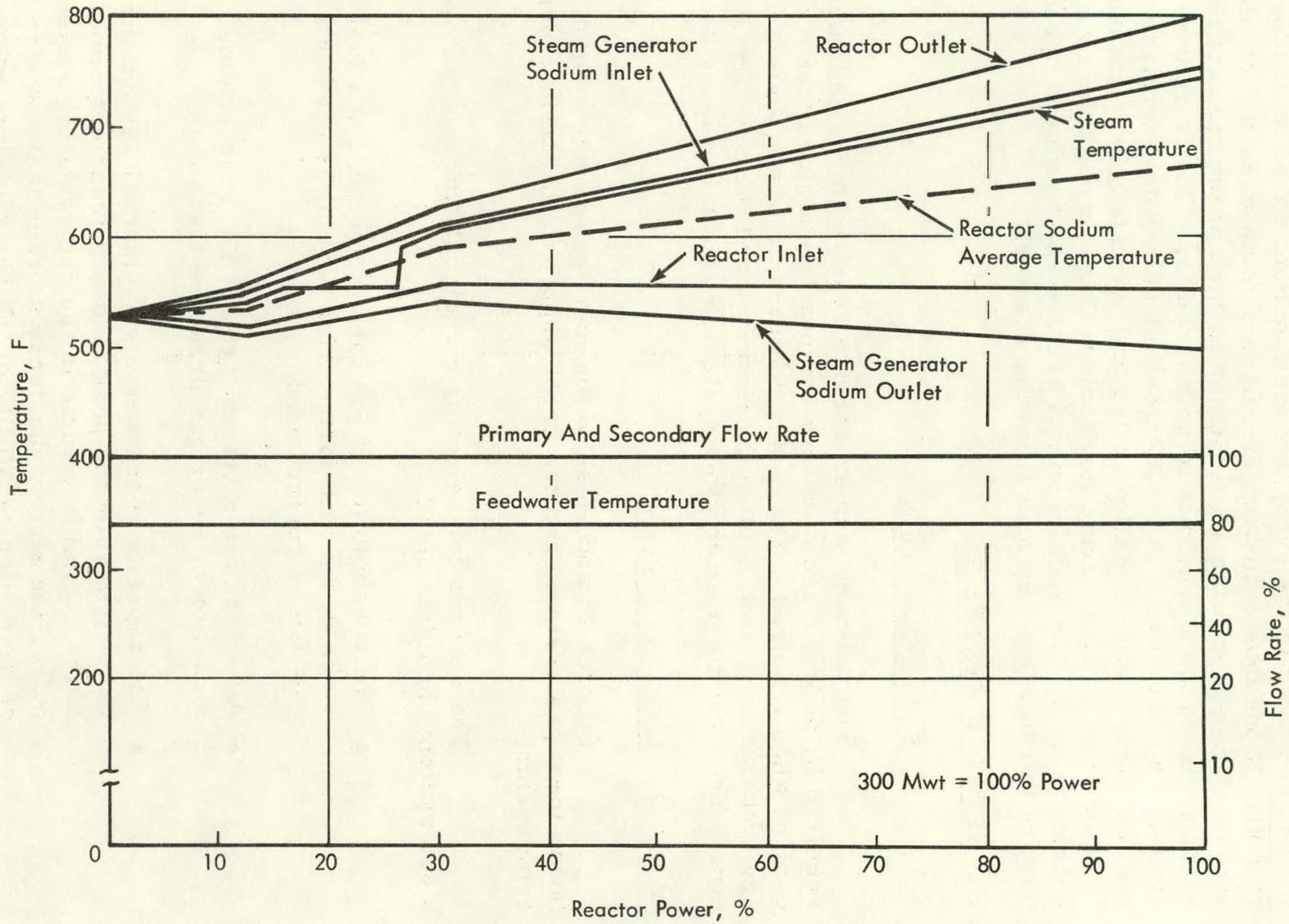


FIG. 43 TENTATIVE SCHEDULE OF OPERATING CONDITIONS

The Fermi plant is designed to operate with two of three primary and secondary heat transport circuits and two of three feedwater and steam circuits. This feature allows the plant to operate at 67% of normal power without shutdown in the event that malfunctions cause a shutdown of one circuit. If one heat transport circuit is shut down as the result of a single-circuit signal, a similar signal in one or both of the circuits that are operating will cause a shutdown of both circuits. This is referred to as a multicircuit shutdown. A multicircuit shutdown de-energizes the main sodium pump motors on all primary and secondary heat-transport systems. Reactor scram results from low sodium flows or negative rate of change of reactor power. The sodium pumps are driven at reduced speeds by pony motors to remove decay heat from the reactor.

H. SERVICE SYSTEMS

1. Sodium Service

The primary sodium service system stores and purifies the sodium received by tank car and monitors and purifies a side stream of the primary coolant when required. The major components of the system are three storage tanks, a cold trap, a plugging indicator and the primary system overflow tank. The flow diagram for this system is shown in Figure 44.

2. Inert Gas Service

Figures 45 and 46 show the inert gas supply system, the recirculating inert gas system, and the argon supply to the various primary system components.

The reasons for selecting a recirculating system over a feed-and-bleed system should be noted:

- The consumption of gas for cooling, e. g., a fuel subassembly stuck in the exit port, or for purging is greatly reduced if the gas can be recirculated.

- A recirculating system provided a large volume for the accommodation of cover gas pressure transients.

- The capacity of the waste gas disposal system did not have to be designed for the peak load of a transient demand because of the available volume in the recirculating system. Furthermore, the size of the decay storage tanks was reduced because of the design capability of the recirculating system to handle fission products.

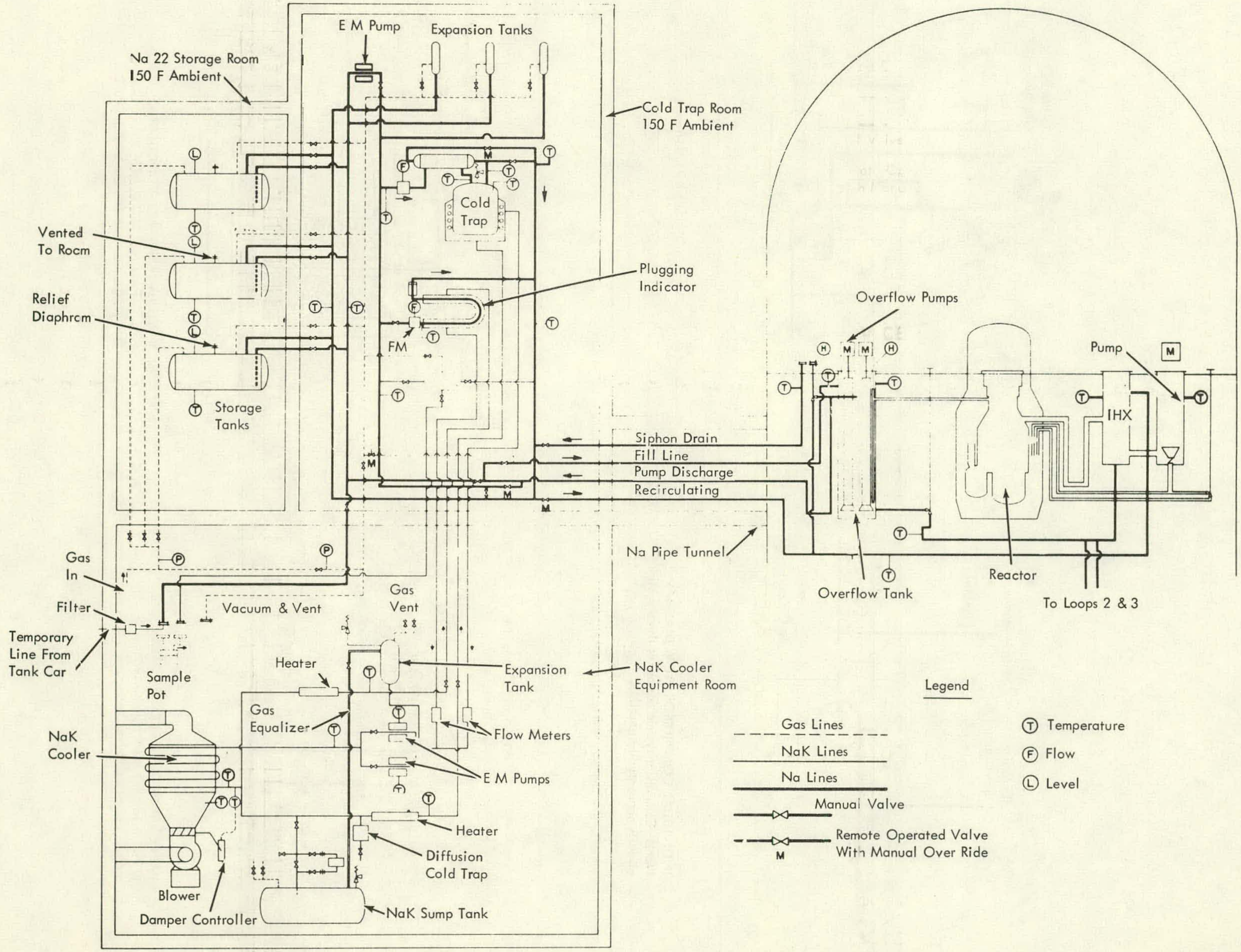
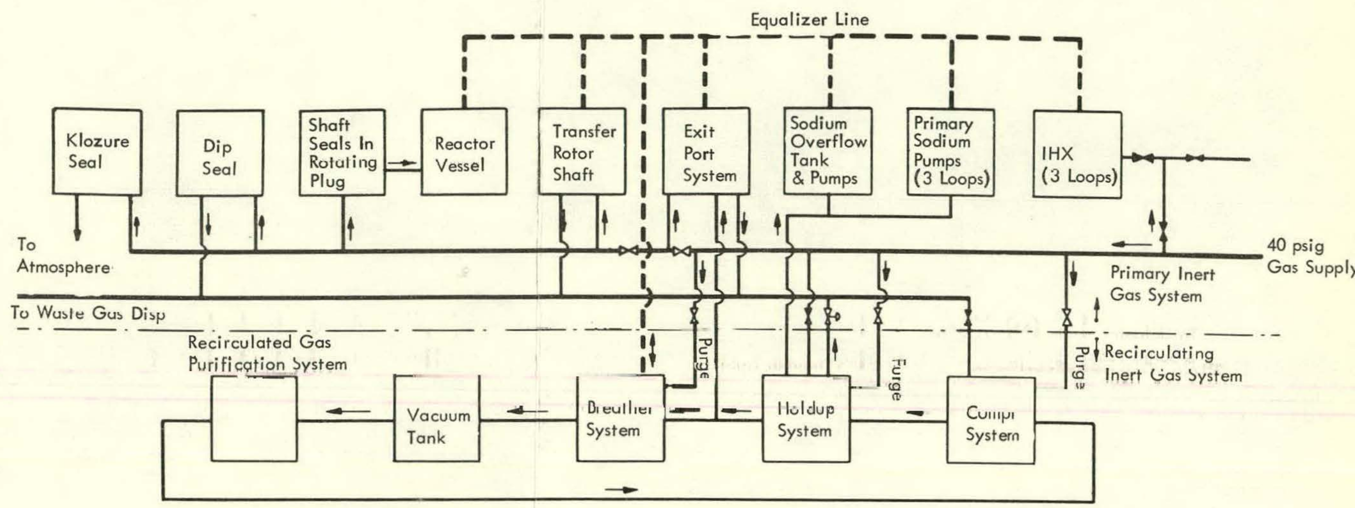
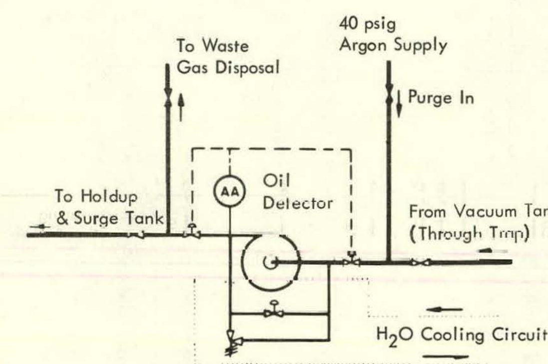


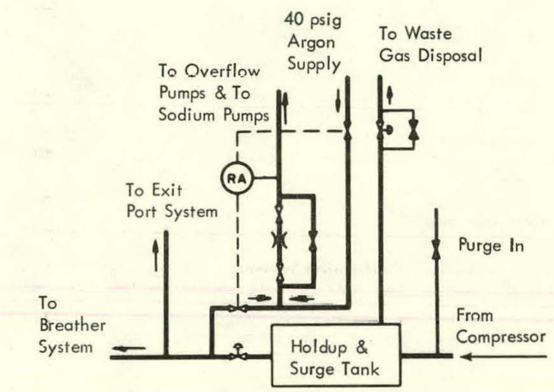
FIG.44 FLOW DIAGRAM OF PRIMARY SODIUM SERVICE SYSTEM



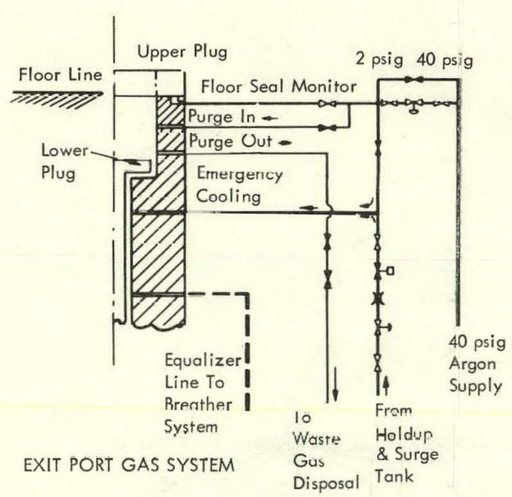
BLOCK DIAGRAM OF PRIMARY INERT GAS SYSTEM AND OF RECIRCULATING INERT GAS SYSTEM



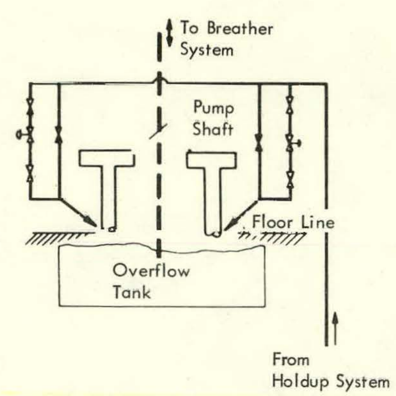
INDIVIDUAL COMPRESSOR FLOW DIAGRAM
Note: Compressor system comprises three such circuits connected in parallel.



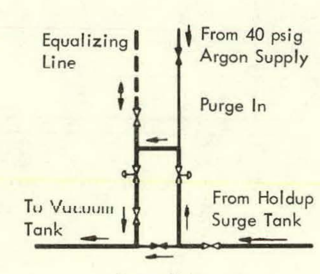
HOLD UP SYSTEM



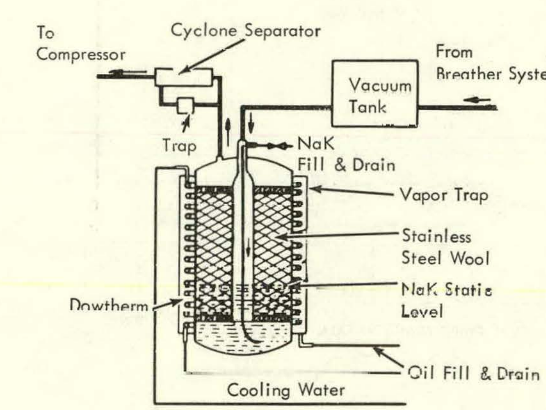
EXIT PORT GAS SYSTEM



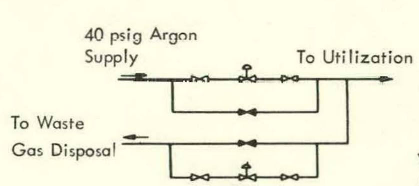
OVERFLOW PUMP AND TANK



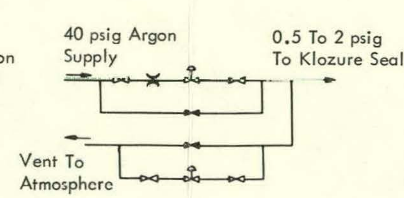
VALVE ARRANGEMENT OF BREATHER
Note: Breather system has two such circuits operating in parallel



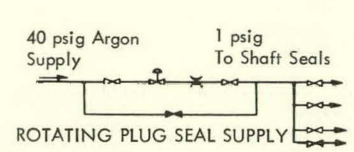
VACUUM TANK AND RECIRCULATED GAS CLEANING FLOWSHEET SHOWING SCHEMATIC OF VAPOR TRAP



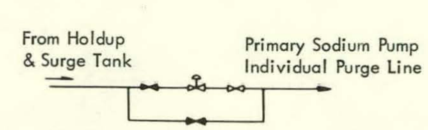
FLOWSHEET FOR NaK DIP SEAL AND TRANSFER ROTOR SHAFT
Note: Transfer rotor shaft has connection to equalizing line.



FLOWSHEET FOR KLOZURE SEAL



ROTATING PLUG SEAL SUPPLY



PRIMARY SODIUM PUMP PURGE SUPPLY

FIG. 45 PRIMARY INERT GAS AND RECIRCULATING GAS SYSTEMS

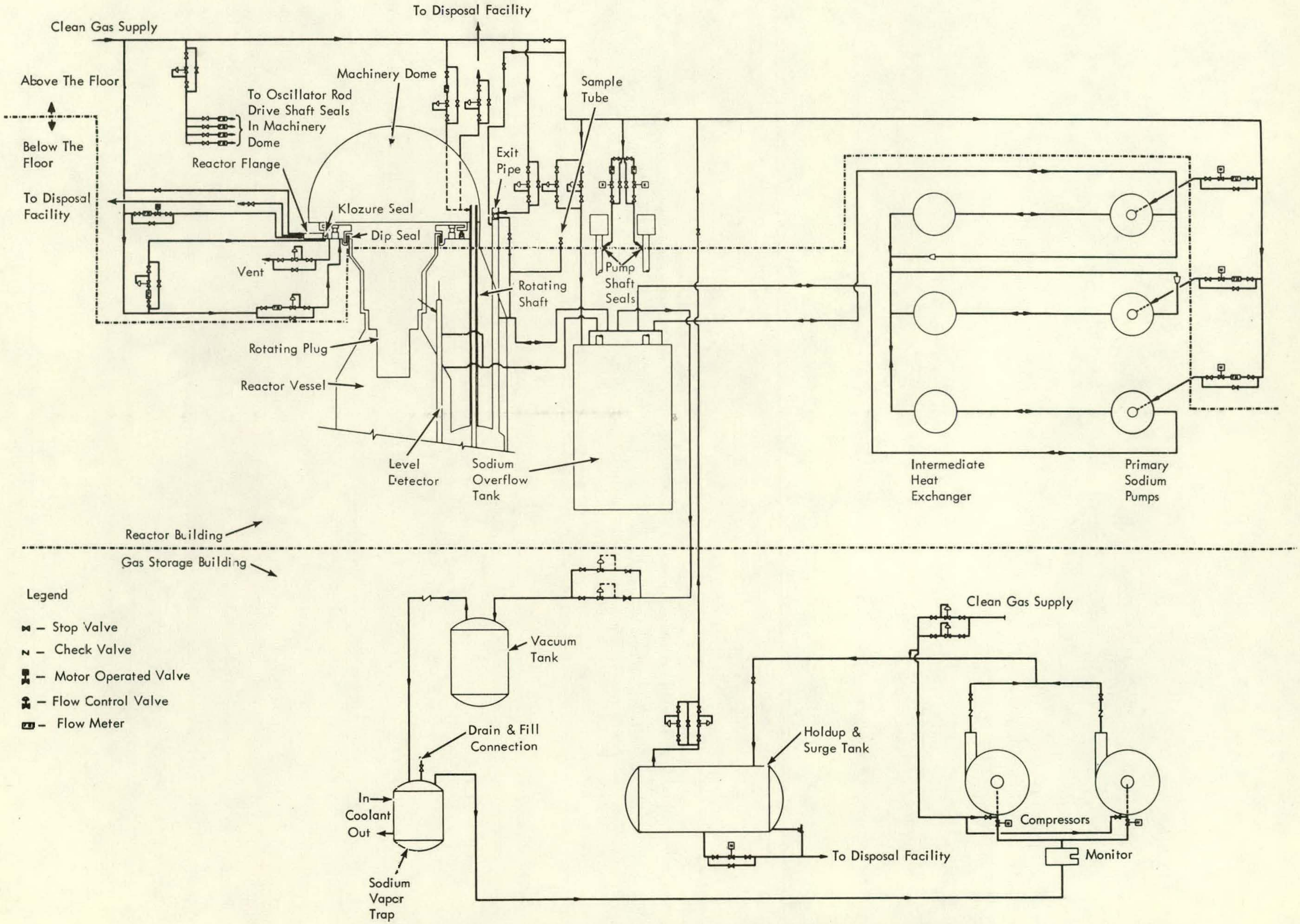


FIG. 46 INERT GAS FLOWSHEET FOR PRIMARY SODIUM SYSTEM

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

V. EFFECT OF PRIMARY SYSTEM DESIGN ON THE CONTAINMENT BUILDING AND ASSOCIATED SYSTEMS

A. EFFECT OF PLANT LAYOUT ON REACTOR BUILDING DESIGN

1. Building Diameter

The amount of space required for the layout of the primary system had a direct effect on the diameter of the steel containment building designed to house the system. A radial increase in the layout of the primary system conponents was economically evaluated in terms of added circumferential concrete external to the building below grade, increased crane span in the building, added wall thickness of the containment building according to code requirements, the necessity to stress relieve at wall thicknesses above 1.25 inches, and added foundation requirements. The layout of the primary system was also evaluated in terms of flexibility design requirements which dictated that the components be located as close as possible to the reactor (the primary anchor) to form a close-coupled arrangement. These considerations contributed to the selection of a 72-foot-diameter cylindrical building and a piping layout with vertical loops instead of horizontal loops as shown in Figure 34.

With the incentive to maintain a close-coupled layout, another design objective was to establish the minimum space between the system components and building wall to achieve a minimum-optimized building diameter. The minimum distance between the building wall and the extreme position of the crane hooks was 5 feet 2 inches for the 20-ton crane and 7 feet 3 inches for the 150-ton crane. Therefore, the components and the decay tanks had to be within this circle. Room was also needed for erection of the equipment and for electrical and instrument trays as well as auxiliary piping. It was subsequently determined that space was also required for steel structural columns that were erected for support of the concrete operating floor. This space requirement added to the layout diameter determined the diameter of the reactor building.

The final layout of the Fermi primary system had a pump and IHX centerline radius of 27.5 feet, with a maximum component radius of 3.5 feet; thus all components were contained within a radius of 31 feet. Crane coverage for the large crane would call for the addition of the 27.5-foot-centerline radius and the 7.25-foot-minimum crane distance, for a minimum building radius of 34.75 feet. The final building radius of 36 feet was selected. This allowed 5 feet for the clearance space between the outer edge of components and the wall.

2. Building Geometry, Cylinder Versus Sphere

The spherical and cylindrical types of containment buildings were evaluated for enclosure of the primary system. The cylindrical shape was selected because it had the lowest atmospheric volume, it was the most adaptable to the vertical tank and vertical loop positions selected for the primary system, and the crane could be supported from the building walls. A spherical building would have been suitable for a primary piping system using horizontal loops as shown in Figure 10. The two geometries were economically competitive, however, a complete evaluation was not made of the concrete shielding and crane support structure inside the spherical building since a vertical tank and loop layout had been selected.

3. Building Height

The steel containment vessel is basically cylindrical, with a hemispherical head and a 2:1 hemispherical bottom. The length of the cylinder was determined by the height of the primary system components above and below the 5-foot-thick operating floor that served as a biological radiation shield.

Below this floor, the height of the reactor and its primary shield tank extended from the center concrete elevation 552 feet to the top of the rotating plug at elevation 590 feet, established as the top of the operating floor. The pump and IHX components were shorter, extending from the outer concrete elevation 557 feet to the operating floor at 590 feet. These elevations established the length of the building cylinder below the floor elevation at 33 feet.

Above the floor, the cylinder length between elevation 590 feet and the intersection with the hemispherical head was determined by the height required for withdrawing, transferring, and reinserting an IHX tube bundle or the primary pump internals from the primary system tanks to the decay tanks. This height was established at 33 feet based on a few feet minimum clearance between the bottom of the pump internals (longest component) and the operating floor when the pump was hanging on the crane. This height could not accommodate the longer reactor internals such as a control rod; however, it was felt that handling of such components should be infrequent and that special handling arrangements could be made. Adequate headroom for withdrawal of such components is available in the center of the hemispherical building head.

4. Design Pressure of Reactor Building

In sodium-cooled reactor plants, the reactor building is isolated in the event that sodium is lost from the system and exposed to the air atmosphere. The resulting atmospheric pressure is recognized as the design pressure of the building based on sodium burning in the isolated air atmosphere. When all of the oxygen is consumed, the burning would cease. The amount of air available to burn, therefore, was a critical factor in the design pressure of the building. With a building of larger diameter or greater height, the design pressure would be greater and would affect all aspects of the building design. The atmospheric and vessel wall temperature would also be affected by extended burning.

In consideration of this accident, there was an incentive to keep the atmospheric volume of the building at a minimum. The Fermi building design is based on a hypothetical accident pressure of 32 psig.[11]

5. Equipment Door in Building

The flanged and bolted equipment door, 11 feet 3 inches by 13 feet 6 inches was sized for the fuel-handling cask car. This size was also adequate for handling the pump and IHX internals in a horizontal position. During installation and erection of the primary system, a temporary opening was made in the reactor building to handle the 14-foot-diameter reactor vessel.

B. BUILDING VENTILATING AND COOLING SYSTEMS

1. Function of Ventilating and Cooling Facilities

The ventilating and cooling facilities in the reactor building are designed to meet the requirements of the primary sodium system components housed inthe building, to facilitate maintenance, to ensure containment, and to eliminate undesirable thermal gradients in equipment.

The reactor building is divided into sealed compartments, each with unique contamination problems and special access restrictions. Slight pressure differentials are maintained between these several areas to direct unavoidable leakage of contaminated atmospheres in preferred directions. The principal compartments in the reactor building are the upper chamber and the lower chamber as illustrated in Figure 47. The upper chamber is served by a regulated forced-air sweep which prevents the accumulation of contaminated gases in the work area and by air conditioning units installed in the reactor building dome. The lower chamber, which is not normally accessible to personnel, is filled with an oxygen-depleted atmosphere to prevent a fire in the event of a sodium spill with resulting distribution of Na-24 reaction products.

Closed system forced circulation is provided in the lower chamber to dissipate thermal and nuclear heat loads which would otherwise build up high temperatures and degrade the concrete shielding around the reactor vessel. Internal duct work is installed to distribute cooling flow to prevent hot spots and undesirable thermal movement in equipment and equipment supports.

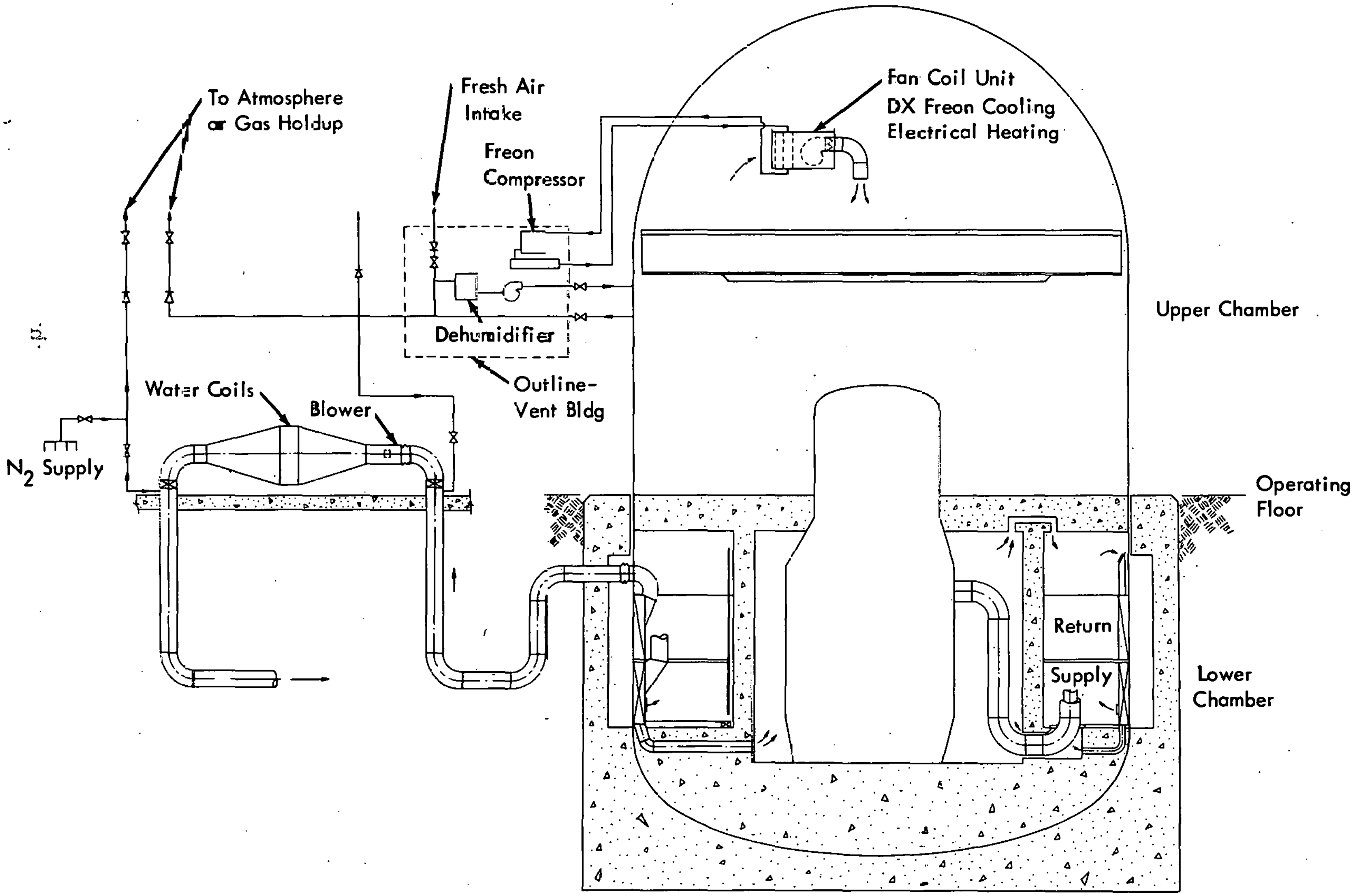


FIG. 47 REACTOR BUILDING VENTILATION

Design of the upper chamber ventilation and air conditioning system is only indirectly influenced by the requirements of the primary sodium system to ensure containment in the event of a major operating incident and to prevent the escape of activity above the maximum allowable level at site boundaries.

High integrity slam-shut safety valves are installed in all service lines penetrating the reactor building, including the upper chamber vent lines. Isolation is initiated by a combination of a high pressure signal in the lower chamber and a high radiation signal from either the upper or lower chamber. For this purpose, radiation monitors are installed in both the upper chamber vent exhaust line and the lower chamber recirculating exhaust line.

2. Lower Chamber Facilities

a. Inert Atmosphere

To suppress any fires resulting from a sodium spill, the lower chamber atmosphere is initially purged with nitrogen gas to 5% oxygen. This 80,000-cubic-foot space is then maintained at +2 inches water column (WC) above the pressure of the upper chamber which is maintained at barometric pressure (Section V-A.3b) to prevent in-leakage of air. Pressure regulation is accomplished by a nitrogen feed-and-bleed system that draws from a liquid nitrogen supply and discharges to the plant waste gas disposal. This arrangement results in a further reduction of the oxygen level (to about 2%) and a very dry atmosphere. As with other service lines penetrating the containment building, the nitrogen feed-and-bleed lines are automatically blocked by high integrity valves in the event of a building isolation signal.

b. Cooling Facilities

The lower chamber is divided by the secondary shield wall into an inner compartment housing the reactor vessel and primary shield tank (PST) and an outer compartment containing the primary and secondary sodium lines and the primary sodium operating components (pumps and IHX's). Concrete in this secondary shield wall and in the operating floor overhead is protected from dewatering by an airtight sheet metal encasement and by recirculating ventilation designed to maintain the concrete at 150 F in the wall and to 190 F in the operating floor.

One of the early schemes to reject heat from the lower chamber involved an assembly of cooling coils attached to the outside of the reactor building with high conductivity cement. This design used no building penetrations, but it was abandoned primarily because it provided no controlled circulation pattern and was therefore subject to hot spots. Another proposed design incorporated external recirculating cooling loops similar to those now installed, but with air-to-air cooling radiators. This arrangement required very large heat exchanger surfaces and very large high integrity containment housing. Freon cooling coils similar to the equipment presently installed in the upper chamber were also considered, but maintenance access problems and secondary hazards* appeared to outweigh the advantages. The formation of nitric acid (NO_2 fixation) due to activity in the nitrogen atmosphere of the lower chamber was investigated and was considered very unlikely.

The design data for the present cooling system is listed in Table 12. The system consists of twin 30-inch pipe loops external to the reactor building and constructed as an extension of high integrity containment. Each loop circulates 15,000 cfm through a vane axial fan and finned-tube water-cooled radiator. As an extension of the containment, these loops are not blocked off during building isolation, but continue to operate to remove decay heat after the reactor is scrammed. Radiation and moisture monitors are installed in the loops to initiate isolation of the cooling water lines in the event of a rupture of the radiator finned-tubes and discharge of water to the lower chamber.

TABLE 12 - DESIGN DATA FOR LOWER CHAMBER COOLING FACILITIES

Volume, cu ft	80,000
Maximum Permissable Temperature	
Secondary Shield Wall Concrete, F	150
Operating Floor Concrete, F	150
Constant Cooling Load @ 430 Mwt	
Nuclear, Btu/hr	454,000
Thermal, Btu/hr	806,000
Design Cooling Capacity, Btu/hr	1,260,000
Maximum Cooling Water Temperature, F	80
Circulation	
Flow, cu ft/min	30,000
Inlet Temperature, F	90
Outlet Temperature, F	130
Developed Fan Pressure, inches WC	15
System Pressure Drop, inches WC	9
Pressure Drop Across Cooling Radiators, inches WC	2 - 1/2

* Secondary hazards involved the formation of poisonous gas in the event of leakage (or total rupture) of freon into the lower chamber.

The requirements of the primary sodium system were not completely satisfied by the original duct work installed to route circulation in the lower chamber. In the basic distribution arrangement, the supply headers for the two cooling loops were manifolded into a single ring header running just inside the building wall at the floor level of the lower chamber with leads through the secondary shield wall, to the outside face of the secondary shield wall, to the bases of the primary pump tanks, and to the pipe trenches in the lower chamber floor. A parallel return ring header was hung just under the operating floor with intake grills appropriately spaced around the circumference of the building wall. Nonstreaming outlet ports were provided in the top of the secondary shield wall to discharge hot gas to the outer compartment for ultimate exhaust to the several intake grills in the return ring header.

After the initial sodium fill, in addition to the usual start-up problems with grills and dampers, there was concern over the design capacity of the cooling system to dispose of the output of the primary system heaters. It was also determined that no stray induction currents were present in the steel shielding plates and that no hysteresis heat was being generated in the steel. In modifying cooling flow distribution, the supply streams were deflected at the base of the pump tanks to avoid uneven thermal growth of the IHX and pump tank support skirts. Modifications also included a supply ring header around the base of the primary shield tank to distribute cooling equally at the elevation of the eight flex legs supporting the reactor vessel. Special supply ducts were run to the underside of the operating floor at the pump tank penetrations to eliminate hot spots. Additional exhaust grills were also installed to balance return flow.

In preoperational tests, it was found that the actual flow in each loop was 14,000 cfm instead of the design flow of 15,000 cfm. The loop pressure drop was 10 inches WC instead of the design value of 15 inches, and the pressure drop across the radiators was 4 inches instead of 2.5 inches. The design load for electrical heaters was 416 kw (1,500,000 Btu/hr). To maintain the system at isothermal conditions, an electrical load of 464 kw (1,640,000 Btu/hr) was measured. Cooling capacity of the system was checked by monitoring the return and supply air temperatures across the heat exchanger. The total heat removed was 1,233,000 Btu/hr, with one loop removing 637,500 Btu/hr and the other loop 595,500 Btu/hr.

Fans and coolers were procured based on the load of 1,260,000 Btu/hr shown in Table 12. The distribution of this load is shown in the first column of Table 13. When primary system arrangement changes were made, there was an increase in piping volumes and an increase in the gas line temperature. On this basis, the estimated load for the 430-Mwt condition was changed as shown in the second column of Table 13. For this condition, the capacity of the coolers would have to be increased.

3. Upper Chamber Facilities

a. Air Conditioning

The electric heating and freon cooling system installed in the dome of the upper chamber are sized according to the demands indicated in Table 14 to provide personnel comfort in the winter and to reject the heat from the primary sodium pump motors in the summer.

TABLE 13 - DISTRIBUTION OF UPPER CHAMBER COOLING LOAD, 430 Mwt CONDITIONS

	System Design, October, 1958	Re-estimate, January, 1960
Inside Secondary Shield Wall		
Nuclear, Btu/hr	135,000	135,000
Piping Setting Losses, Btu/hr	127,000	180,000
Reactor Setting Losses, Btu/hr	155,000	155,000
Outside Secondary Shield Wall		
Nuclear, Btu/hr	319,000	319,000
Setting Losses, Btu/hr	424,000	605,000
Sleeves, Btu/hr	100,000	100,000
Total, Btu/hr	1,260,000	1,494,000

TABLE 14 - DESIGN DATA FOR UPPER CHAMBER VENTILATION SYSTEM

Volume, cu ft	200,000
Maximum Cooling, B	825,500
Maximum Heating, Btu/hr	1,087,000
Outside Design Temperatures, F	10 to 100
Inside Temperatures, F	40 to 100
Design Relative Humidity	No Moisture
Air Flow, cu ft/min	150 to 200

Early designs considered control of humidity and the instalation of dehumidifying units. The need to exclude water from the containment building was recognized to the extent that condensate from the freon cooling coils is drained out of the building through a pressure sealed trap. A smaller (7-1/2 ton) freon cooling system services the machinery dome atmosphere and is set to maintain 80 F.

b. Isolation and Disposal of Contamination

Barometric pressure at the Fermi site ranges from 381 inches to 413 inches water column absolute (WCA). The upper chamber ventilation system was originally designed to maintain 414 inches WCA. Controls were later altered from absolute to barometric reference, and the upper chamber is now maintained at barometric pressure; this simplifies the operation of personnel airlocks and generally stabilizes the related building atmosphere control circuits. Clean air is supplied by a 200-cfm blower through a flow regulator valve and exhausts to a back-pressure regulator valve and a second 200 cfm blower; both systems act independently to avoid hunting.

A sealed machinery dome over the reactor plug area isolates all of the plug penetration seals, providing a line of defense against the minor hazards of plug seal failure by isolation of escaping radioactive cover gas. The machinery dome ventilation system is essentially a bypass slipstream (50 cfm) flowing parallel to the upper chamber ventilation feeding from the same forced draft supply and discharging to the same forced draft exhaust line.

The supply and exhaust lines contain quick-closing valves that close on high pressure or high radiation within the building. These valves are 6-inch, air-operated, globe body design with a closing time of less than 5 seconds from the detection of a concentration of radioactive material of 2 x 10^{-4} μCi/cc in the upper chamber.

4. Atmospheres in Containment Building

There are two basic purposes for controlling the atmosphere in the containment building: (1) exclusion of air and moisture and (2) the establishment of a preferred leak path through the various isolation barriers. Air is circulated above the operating floor through the upper chamber and the machinery dome and discharged to the waste gas stack. This permits personnel access and prevents the buildup of contaminants in the upper chamber atmosphere.

The lower chamber and primary shield tank are filled with depleted oxygen (98% nitrogen) to inhibit sodium fires and to protect the graphite shielding from chemical attack. Two isolated sections within the PST called the pan and the tub are packed with the original graphite shielding material which had objectionable offgassing characteristics. These spaces are serviced separately, at a slightly lower pressure, to prevent leakage into the relatively clean PST proper. The PST is maintained at a higher pressure to prevent in-leakage of lower chamber atmosphere which is more conducive to contamination. The primary cover gas is maintained well above barometric pressure to exclude air.

The operating pressures for these systems are arranged in the following cascade:

Space	Pressure	Atmosphere
Upper Chamber	Barometric	Circulating Air
Machinery Dome	Bar. - 1" WC	Circulating Air
Lower Chamber	Bar. +2" WC	95 to 98% N_2
PST	Bar. +10" WC	98% N_2
PST Pan	Bar. +7-1/2" WC	98% N_2
PST Tub	Bar. +7-1/2" WC	98% N_2
Cover Gas	Bar. +4" WC	Argon

In this way, leakage is prevented from the machinery dome to the upper chamber and from the upper chamber or the machinery dome to any area below the operating floor.

During the early stages of design, there was less confidence in the Klozure seal on the rotating plug and, in addition to a NaK-filled dip seal for backup, some consideration was given to a controlled argon atmosphere in the machinery dome. With this arrangement it would be possible to reverse pressure differential between the cover gas and the machinery dome if a leak should develop in the Klozure seal or some of the other plug seals, and the contaminated atmosphere of the cover gas spaces within the reactor vessel would be contained. The Klozure seal has performed so well that the NaK dip seal was never filled. The many objections to an argon atmosphere in the machinery dome (limited maintenance access, poor commutation, etc.) were avoided by installing the circulating air system with appropriate isolation mechanisms as stated in Section V-B.3.

C. DESIGN OF SHIELD SYSTEM

1. General Considerations

The overall shield system, shown in Figure 48, consisted of (1) a neutron and gamma-ray shield located directly outside the core and also contained within a vessel known as the primary shield tank, (2) a secondary neutron shield wall of concrete which divided the lower reactor building area into an inner reactor compartment and an outer heat exchanger compartment, and (3) a biological shield which completely surrounded the radiation area.

Neutron shielding was accomplished (1) by using a stainless steel layer to protect the reactor vessel against radiation damage due to the high energy components of the escape flux and (2) by using a good moderating material outside the vessel where there was no penalty attached to its use.

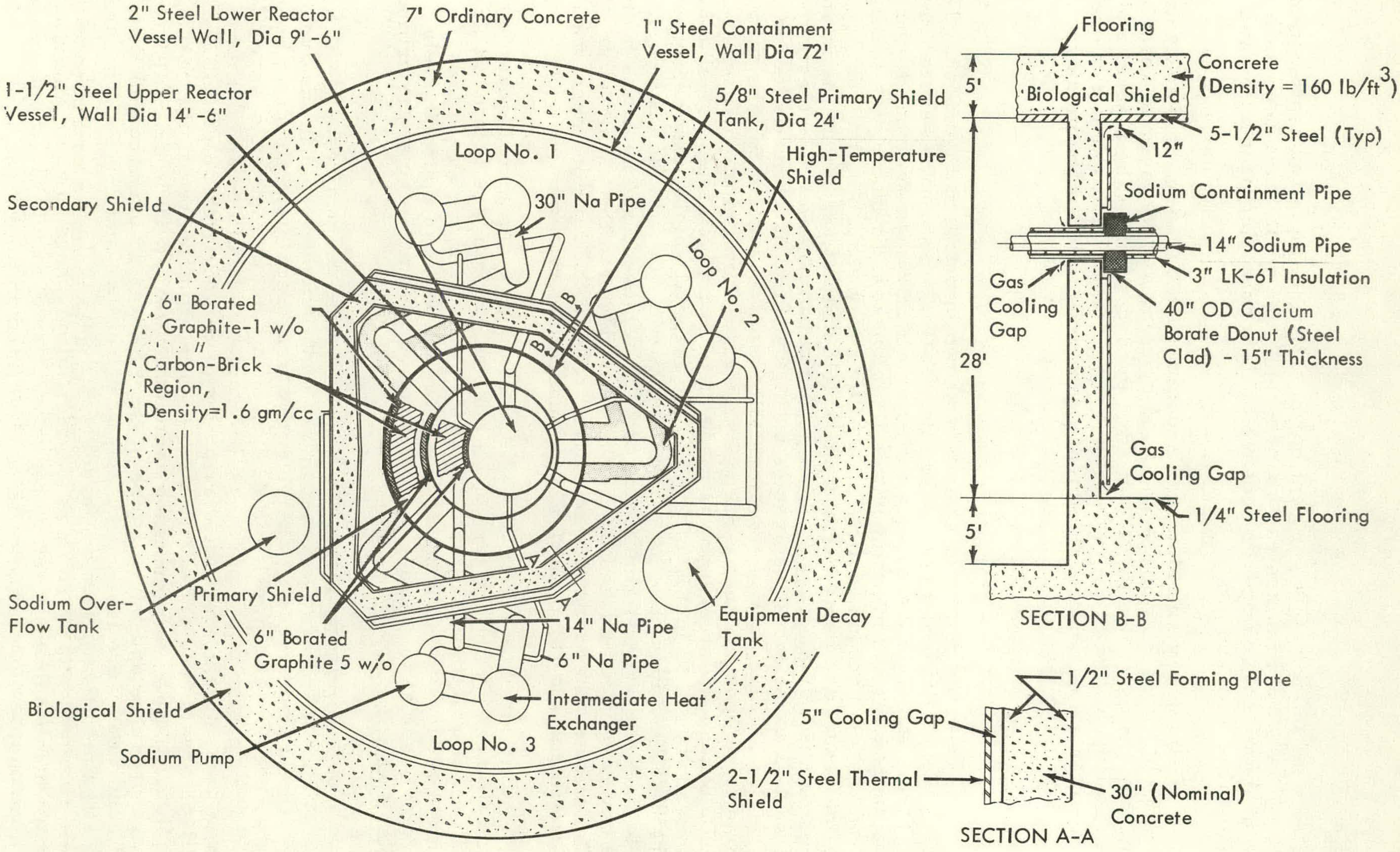


FIG. 48 PLAN VIEW OF SHIELD SYSTEM

Graphite, with a few weight per cent boron satisfied the latter requirement as well as being a strong neutron absorber. Water would have been an ideal shield but it was not considered because of its moderating properties and incompatibility with sodium.

Gamma ray shielding was accomplished by using the blanket uranium to shield the rays from the core and by using a steel layer to absorb a large part of the capture gamma radiation produced within the blanket and core. The escape flux of gammas from the primary shield was not of overriding importance because of the intense gamma activity of the primary coolant outside the shield.

2. Primary Shield

The primary shield system is shown in detail in Figure 49. The laminated thermal shield plates inside the reactor vessel have several functions. First, this shield acts as a reflector which return neutrons to the blanket, increasing the breeding within the blanket and reducing the leakage out of the vessel. Second, it acts as a gamma-ray absorbing medium, protecting the vessel and other parts of the shield against heating by the intense core and blanket gamma rays. Third, and most important, it protects the vessel walls against radiation damage by high-energy neutrons by its inelastic scattering ability. The laminations allow sodium coolant to pass through the shield to remove the heat generated in the steel by radiation absorption. The inner part of the steel layer is made up of dummy stainless steel subassemblies. In the 12 inches of steel in the thermal shield, fast neutrons of energies 1 Mev and above have an attenuation of 100. In the energy range of 0.5 to 1.0 Mev, a reduction factor of 20 is achieved, and neutrons in the 1 to 20 kev range are reduced by 2 or 3. In addition to the steady-state shielding effects, the thermal shield also protects the reactor vessel wall from the effects of a rapid thermal transient in the system.

The graphite shield consists of graphite and carbon blocks that need no cooling. Six-inch layers of 5% borated graphite are located at the reactor vessel wall and also inside the wall of the primary shield tank. This shield reduces the total neutron flux from a value of approximately 10^{13} n/cm^2 - sec at the vessel wall to about 10^8 n/cm^2 - sec at the primary shield tank wall.

In preliminary design the PST was to be an open vessel, since its chief function was to serve as a secondary containment vessel in the event of a leak in the reactor vessel. The objective was to maintain a sodium level above the reactor outlet level. (Note that open containment tanks were adopted in the final design of the pump and IHX tanks.) To afford an additional margin of protection against the dissipation of radioactive coolant to the below-floor area and to provide an oxygen-free atmosphere for the carbon and graphite, the PST was designed as a sealed vessel. The original internal design pressure and temperature of the PST was 25 psig and 800 F and normal conditions were less than 1 psig gas pressure and 250 F. The final rating, after modifications were made, was 8 psia to 7.38 psig for 800 F service. Nitrogen was selected as the inert atmosphere to avoid shielding problems which would be posed by the generation of argon-41. Rupture disks for both vacuum and pressure relief are mounted on the vessel.

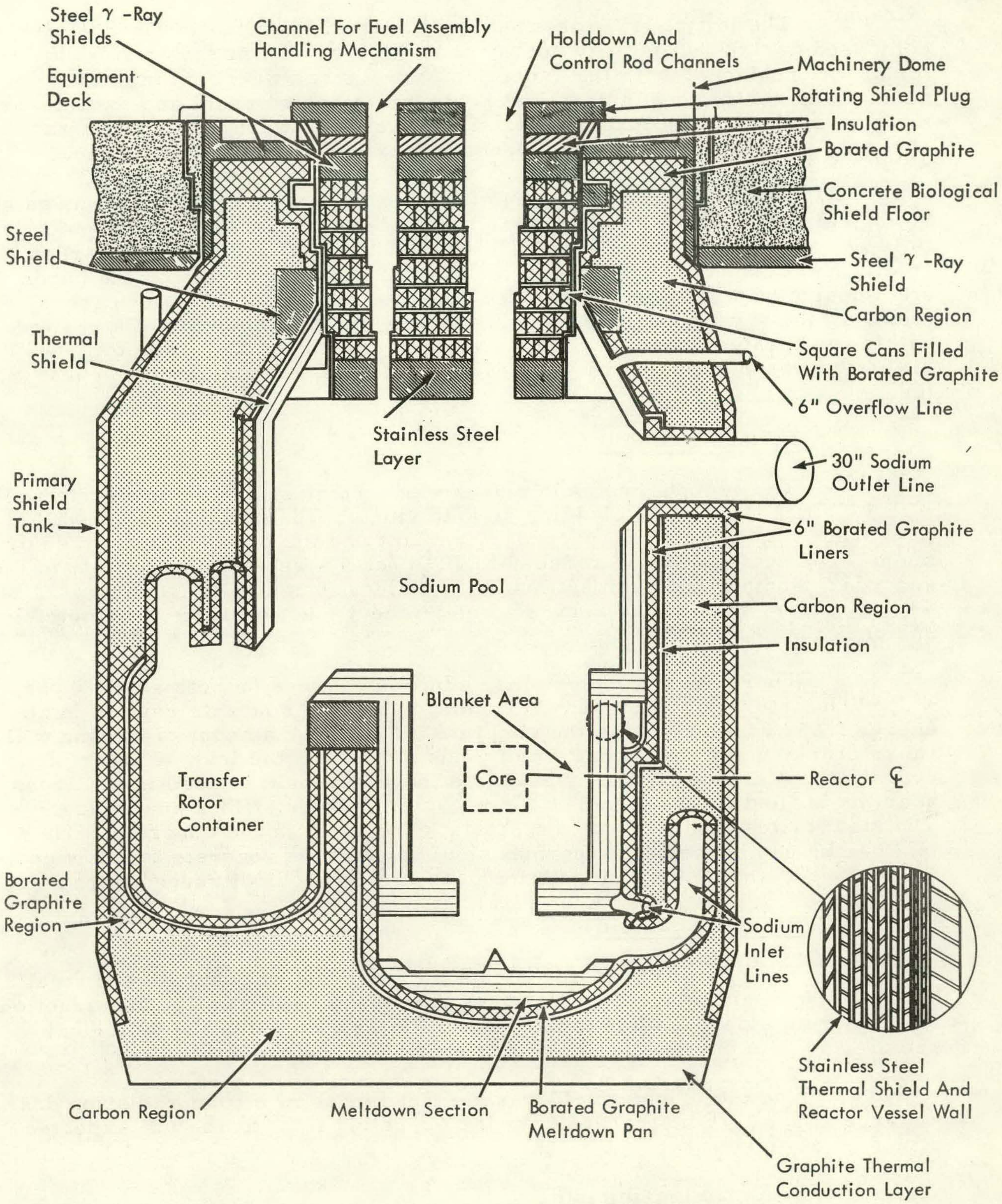


FIG. 49 PRIMARY SHIELD SYSTEM

The initial graphite shielding material installed in the PST decomposed during test facility operation at 1000 F. All of the original material was replaced, with the exception of two areas where maintenance access was restricted; the pan just below the reactor vessel and the tub, or bottom section, of the PST vessel. These volumes were sealed off with sheet metal partitioning and independent breather lines were installed.

The rotating plug on top of the reactor vessel also functions as a biological shield. The original design called for 18 inches of stainless steel, followed by 7 feet of boron steel (1% boron by weight) and 1 foot of carbon steel. However, to improve upon an excessive weight problem, the boron steel design was replaced by borated graphite with about 2 feet of carbon steel, 1 foot of which is at the top of the plug and the other foot distributed within the borated graphite to act as support layers for the graphite. The graphite is canned in square steel cans about 3 inches wide by 12 inches long.

3. Secondary Shield

The secondary shield consists of a nominal 30-inch concrete wall with 1/2-inch thick steel cladding on both sides. The purpose of the wall is to prevent activation of the secondary coolant and the primary system equipment. The neutron flux is reduced to 10^4 n/cm^2 - sec and the activity of the secondary sodium has been estimated to be less than 2 x 10^{-4} μCi/cm^3. The secondary shield also serves a containment function since it completely encloses the reactor system.

There are strong gamma radiation sources on both sides of the wall. The interaction between the radiation and the concrete results in an energy transfer which heats the concrete. Any large amount of heating will cause cracking and the formation of voids as well as the loss of water of hydration. A 2.5-inch-thick steel plate thermal shield is used in the areas near the largest Na-24 gamma sources, i.e., pumps, IHX's and piping. The steel plates limit the amount of heat generation in the concrete. There is a 5-inch gap between the thermal shielding and the concrete to allow cooling by forced and natural circulation of the below-floor nitrogen atmosphere.

4. Biological Shield

The operating floor consists of 5 to 7 feet of concrete and steel plates. Where there are penetrations, the plugs are of similar construction. The concrete shield outside the reactor building completes the biological shield and it is constructed as a continuation of the operating floor.

The biological shield was designed to allow a total radiation dose no larger than 0.75 mrem/hr above the operating floor at the full expected power level of 430 Mwt.

5. Sodium Pipe Shielding

The 30-inch-diameter reactor outlet piping would account for a large neutron leakage because of streaming through the sodium and in gaps around the pipes. A free-standing column of pressed calcium borate completely envelopes this pipe inside the secondary shield wall. The thickness varies from 9 to 20 inches, and shielding collars were located at the top and bottom of the column to prevent streaming.

6. Pump and IHX Shield Plugs

The bottom of the IHX shield plug is exposed to 800 F secondary sodium and the pump plug is exposed to the primary cover gas temperature of 300 F to 500 F. For these temperatures, the ordinary concrete of the operating floor would not be suitable; therefore, serpentine concrete has been used in the pump plug and dry serpentine has been used in the IHX. This material was tested in extensive soaking heat tests at 800 F for 1000 hours, and a water-loss rate of less than 0.5% per year was indicated.

D. SECONDARY SYSTEM DESIGN

With the radial location of the IHX's around the reactor in the round containment building, the question arose as to the routing of the secondary sodium piping. It was decided to direct this piping (two 12-inch supply and two 12-inch returns for each IHX) directly out of the building at the IHX locations. To keep the piping runs to a minimum, plant layouts were prepared with steam generator buildings located radially around the containment building. This arrangement had the advantage that any accident in a steam generator would be limited to one building and one loop. The disadvantage was the necessity for three separate buildings and crane facilities. It was also determined that for safety purposes, it would be preferable to have the containment building walls exposed to the ambient atmosphere for cooling during an accident. Therefore, the steam generators, pumps, and associated equipment were located in an in-line arrangement in a building adjacent to the containment building.

Chromium-molybdenum materials were selected for the secondary system, whereas Type 304 stainless steel was used in the IHX. Therefore, transition welds had to be made in the piping system located just outside the building penetrations.

A leak in the IHX between primary and secondary sodium was never considered a safety hazard. Detection of a leak into the primary system would be by level changes in the primary sodium overflow tank and in the steam generator.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

VI. FABRICATION, ERECTION, AND TESTING

A. ERECTION OF TEST FACILITY

1. Description of Test Facility

APDA had the responsibility for the research,development, and conceptual design of the reactor and primary system portion of the plant, as well as the conceptual design of the entire plant. To accomplish this work, the decision was made to build and operate, under nonnuclear conditions, enough of the reactor and primary coolant system to ensure that the design concepts and equipment for the reactor plant were acceptable for nuclear operation.

The test facility, furnished by APDA, included the reactor vessel, primary shield tank, rotating shield plug, fuel handling mechanisms, safety rod drives, primary sodium pump, piping, and instrumentation. These components, together with sodium purification equipment and temporary heater-cooler facilities, were assembled into a full-sized mechanical and hydraulic test facility consisting of the reactor and one primary coolant loop.

2. Test Site Selection

It had been planned originally that APDA would conduct a components test program at the Delray Power Plant of The Detroit Edison Company, with the subsequent removal of the components for use in a reactor power plant to be built at another location. Tentative plans were made for installation of test equipment in an existing building which had formerly housed a turbine-generator unit.

Subsequent studies indicated that it would be economically undesirable to utilize a temporary site. Further objections to this arrangement were presented by the fact that construction of the Fermi plant was approved, and the Lagoona Beach site had been made available.

The decision to incorporate the test facility as part of the final construction of the Fermi plant was made when it became evident that the extended time schedule for delivery of the major reactor system components would coincide with the completion of the Fermi reactor building. Thus, it was possible for the major system components to remain in place after testing.

3. Erection Sequence Planning

To facilitate construction by anticipating any interference problems that might be encountered or any special equipment that might be required, a one-twelfth scale model of the reactor building, all major components, shop-fabricated piping, structural members, and shielding was built. Each item was installed step by step in the model of the reactor building using the operable model crane and such handling devices as were found necessary. Plan and/or elevation photographs were made of each step of the sequence as shown in Figure 50. Each photograph was accompanied by descriptions in which any noncritical operations that occurred between photographs were noted and details of any special equipment required or interference problems encountered in the operation were shown.

As special problems arose, the necessary procedures and equipment were developed to circumvent or resolve the problem. Whenever equipment was incompatible with installation procedures, it was redesigned prior to delivery to the field. In its final form, the model erection sequence study resulted in a logical approach to completion of the plant. The model study made it possible to provide proper access for moving equipment into the building and ensured that equipment was assembled or installed in the proper sequence without costly dismantling of components already assembled or installed in error.

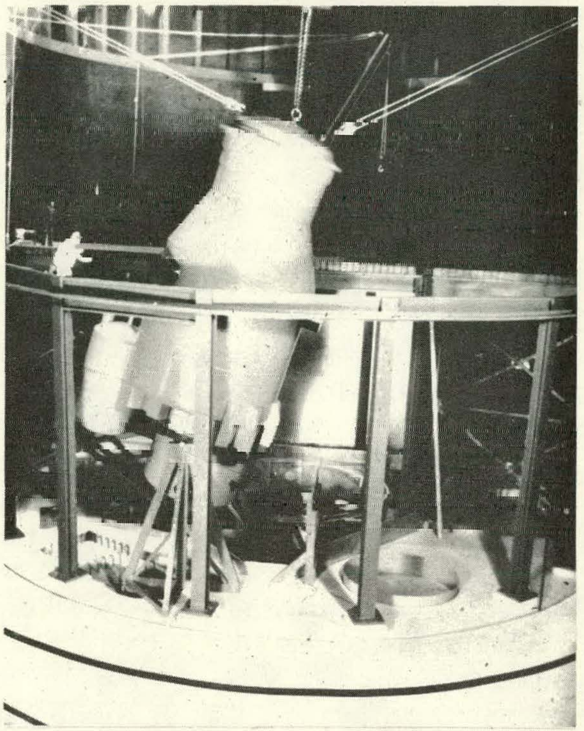
B. MATERIALS SPECIFICATIONS

In the original material specifications for reactor construction, the cobalt concentration in the stainless steel was restricted to a maximum of 0.2 %. The concern with the amount of cobalt relates to the long decay time (5-1/2-year half-life) associated with its isotope, cobalt 60, and its possible deposition in relatively cold areas of the system that would make maintenance difficult. The specifications were eventually relaxed by permitting the waiver of the cobalt limitation by written request subject to the specific use location in the reactor plant.

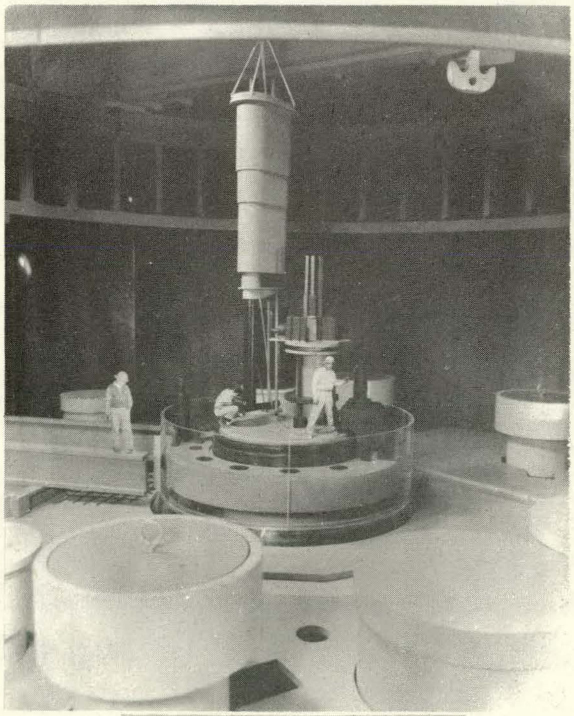
Originally, the APDA specifications required that the surface finish for all components in the primary system be 125 microinches RMS. This stipulation was based on the requirement for ultrasonic inspection since defects are easier to detect on a smooth surface by this method. The smoother surface also facilitated cleaning operations. However, manufacturers proposed that their commercial mill finish of 250 microinches RMS be accepted and, after considering all the factors, the specifications were so revised.

In the original component specifications, trichorethylene was selected as the degreasing and cleaning agent. However, its use was later prohibited because of the possibility that it could result in chloride stress corrosion in the stainless steel materials. Objections were also made by fabricators to

REACTOR VESSEL
INSTALLATION



REACTOR VESSEL
INSTALLATION



INSTALLATION OF OFFSET
HANDLING MECHANISM

STATUS AT START OF
TEST FACILITY OPERATION

FIG. 50 ERECTION SEQUENCE STUDIES

the use of this fluid in their shops due to its toxic properties. As a result, the cleaning specifications were revised and industrial cleaning materials such as water soluble detergents were used. The latter method provided adequate cleaning, and the original concern about residual water left in the primary system did not materialize.

C. REACTOR VESSEL

The principal material of construction was Type 304 stainless steel. The core and blanket support plates and the holddown plate assembly were fabricated of Type 347 stainless steel. The tops of the holddown mechanism support columns are of Inconel X. To ensure strength and ductility, the carbon range of the Type 304 material was limited to 0.06 minimum and 0.08 maximum. Ultrasonic tests were performed on all materials with few exceptions. Major strength welds and welds involving vessel containment integrity were inspected by radiography. All welds were also dye penetrant inspected.

To ensure that design conditions were met in fabrication, the reactor vessel underwent a shop hydrostatic test at room temperature. In addition, a vacuum hold test was performed, and all on-site piping welds were mass spectrometer leak tested in addition to the normal radiographic inspection.

The reactor vessel was shipped by barge from Chattanooga, Tennesee, to a point near Cincinatti, Ohio, where it was transferred to a special railroad car for shipment to the Fermi site near Monroe, Michigan. The special car was preceded by a "shadow car" run over the same route. The reactor building had been erected and tested, and a temporary opening had been cut in the western side of the building to accommodate the railroad car with the reactor shell on board. When the car was positioned in the center of the building, the crane lifted the reactor vessel and placed it on a trunnion for upending. The rail car was then moved out of the building, the tracks were removed, and the building opening sealed.

The lower 6 feet of the primary shield tank as shown in Figure 51, including the support web was in place to receive the reactor vessel. The placement of the reactor vessel on its supports in the primary shield tank is shown in Figure 52.

The flex legs were anchored to the reactor vessel at the bottom. Before attachment to the primary shield tank web, each flex leg was forced outward radially as a cold spring arrangement. As the reactor vessel was heated and filled with sodium the flex leg plates would become vertical due to radial thermal growth of the vessel.

The next step in the erection consisted of placing the internals in the vessel. These constituted the internal thermal shielding, support plates. and transfer rotor equipment. The next step was the installation of the holddown column and spider followed by installation of the rotating shield plug.

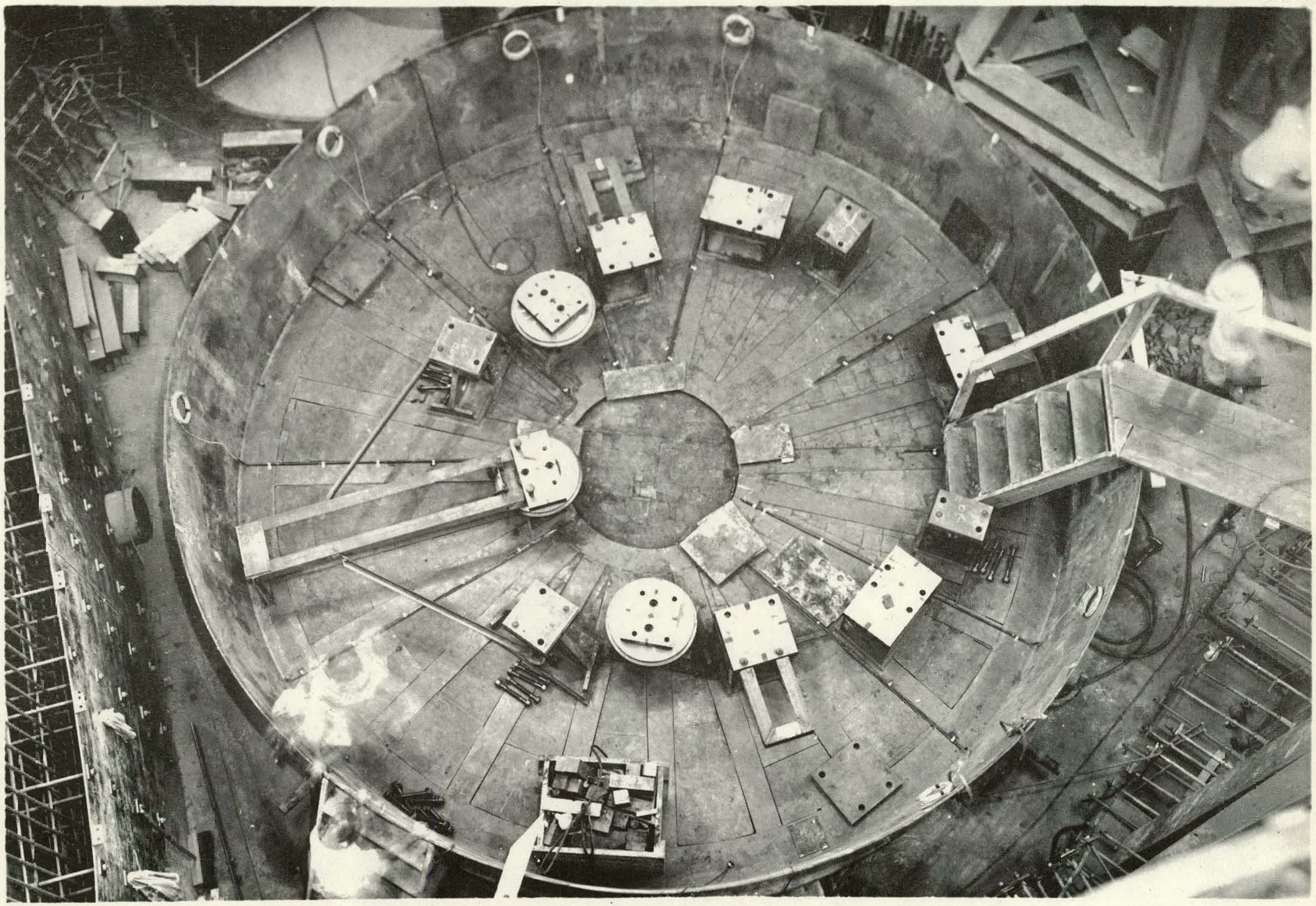


FIG. 51 LOWER SECTION OF PRIMARY SHIELD TANK

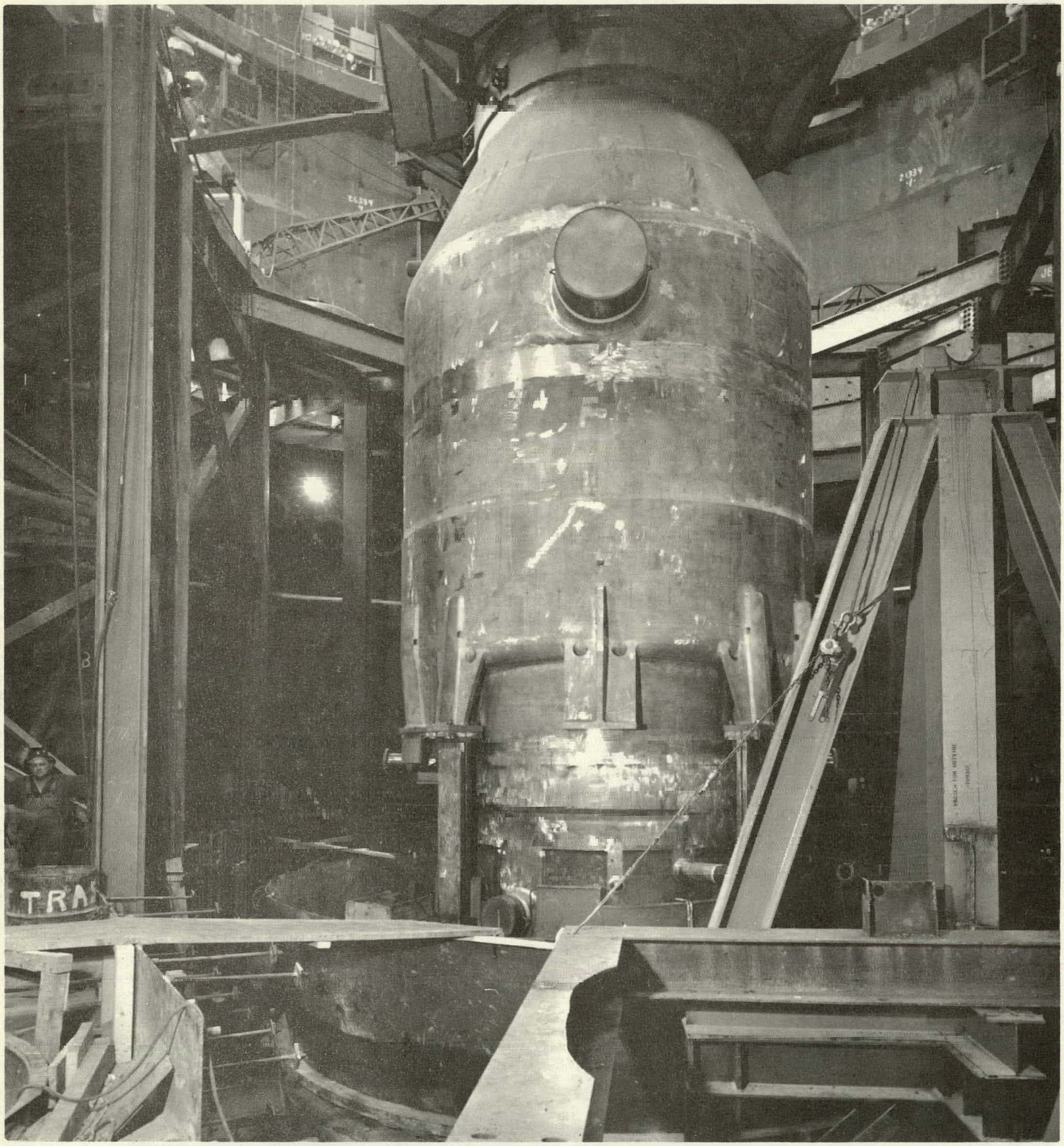


FIG. 52 REACTOR VESSEL IN PLACE IN REACTOR BUILDING

Shield plates and canned graphite blocks were installed within the rotating plug shell, and the dummy fuel subassemblies were installed in the reactor.

For test facility operations, dummy fuel subassemblies whose interior structure differed from that of actual subassemblies were installed in the lower reactor vessel. The dummy subassemblies retained features that would produce the same effect on flowing sodium as expected of the final plant subassemblies. Externally, the subassemblies were identical to actual fuel subassemblies, and they were inserted in the reactor in the normal lattice arrangement.

Following completion of the primary shield tank, the 30-inch, 14-inch, and 6-inch sodium pipes were run through prepared openings and welded to stubs on the reactor vessel. Bellows, previously installed on the pipes, were cold sprung and welded to the primary shield tank. The 30-inch U-loop assembly was placed in position before the secondary shield was poured. Both 30-inch visors were welded to the U-loop only after the IHX was in its final erected position. The cold-pull final weld was made at the IHX inlet. All 6-inch and 14-inch piping wall penetrations were made before the concrete was poured so that the wall sleeves could be positioned for the hot position. All welds were radiographed and dye-penetrant checked.

D. PRIMARY SYSTEM PIPING

The 30-inch, 16-inch, and 14-inch primary sodium piping is fusion welded Type 304 stainless steel fabricated of materials conforming to ASTM A-358 and ASTM A-240 3/8-inch plate. The 6-inch piping is Schedule 40 stainless steel fabricated of materials conforming to ASTM A-376.

All plate material was ultrasonically tested before being rolled into pipe cylinders and fittings. A No. 1 mill finish was given to the side of the plate which forms the inside of the pipe. All longitudinal and girth welds on pipes and fittings were given radiographic and dye penetrant inspections. Completed assemblies of both seamless and seam welded piping were given hydrostatic tests at the fabrication shop. The assemblies were then cut to cold-sprung length and given mass spectrometer tests. All welding attachments such as hanger lugs and instruments were welded to the piping in the shops. Cleaning was performed in accordance with rigid specifications, and the assemblies were sealed airtight and weathertight for shipment. The piping assemblies and layout are shown in Figures 53, 54, and 55.

During erection, the 30-inch piping was cold sprung 100% for 900 F and the pump discharge piping was cold sprung 100% for 600 F; Table 15 lists the final cold spring gaps for a typical loop. Field welds were given dye penetrant and radiographic tests and completed joints were given a local mass spectrometer leak test.

Carbon steel secondary containment was affixed to the straight sections of the primary piping before field assembly of the system, and sec-

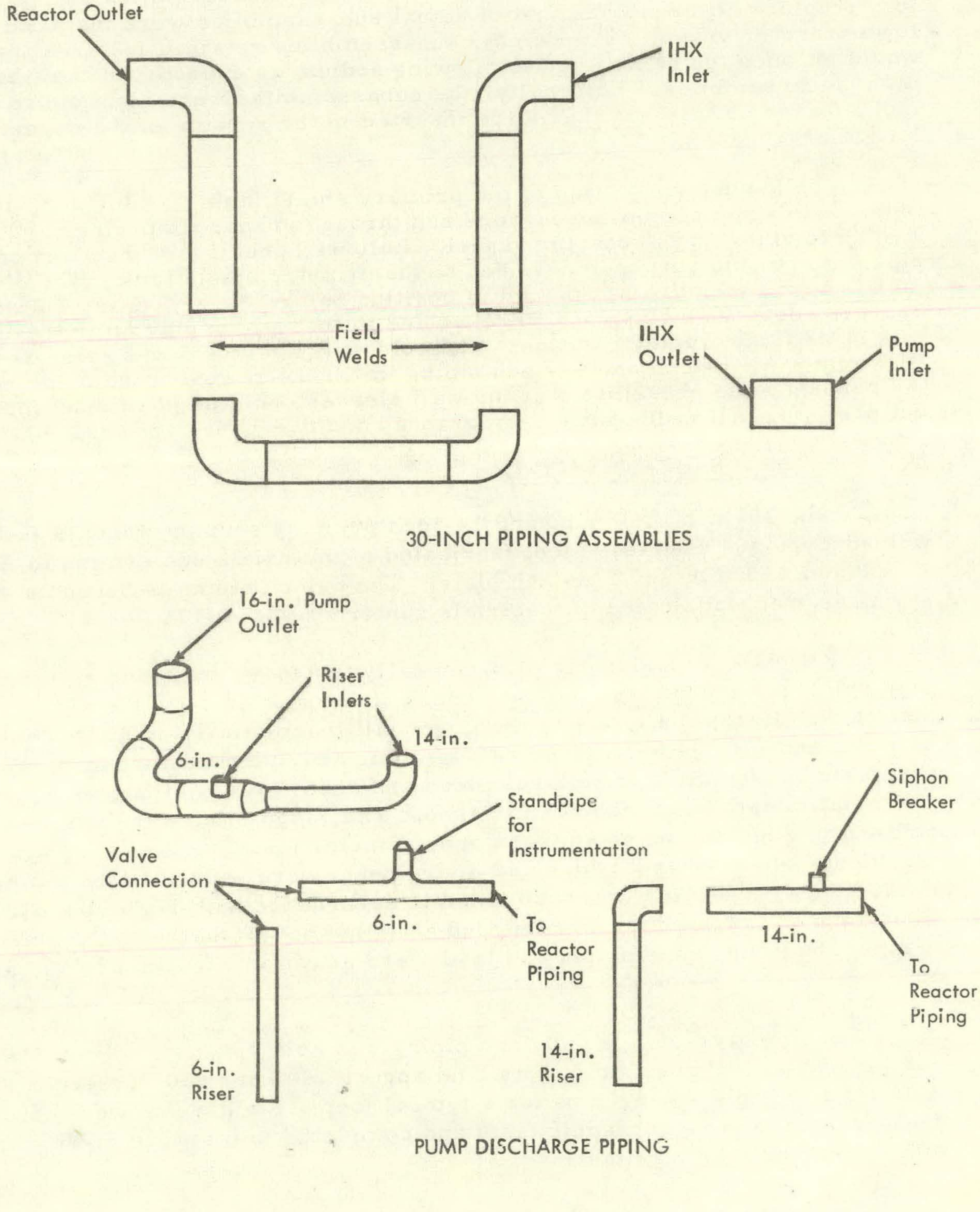


FIG. 53 PRIMARY PIPING ASSEMBLIES

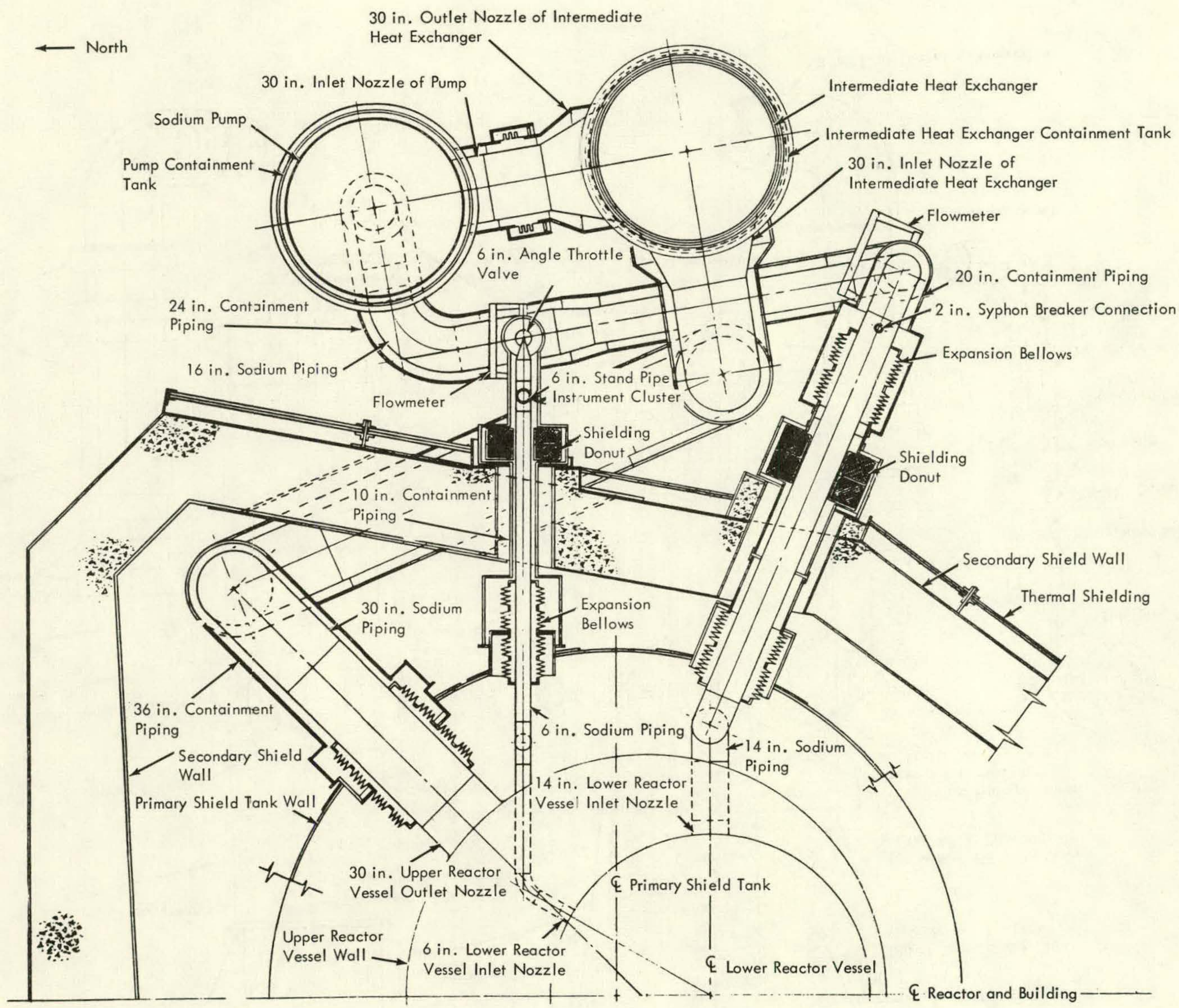


FIG. 54 PLAN VIEW OF PRIMARY PIPING

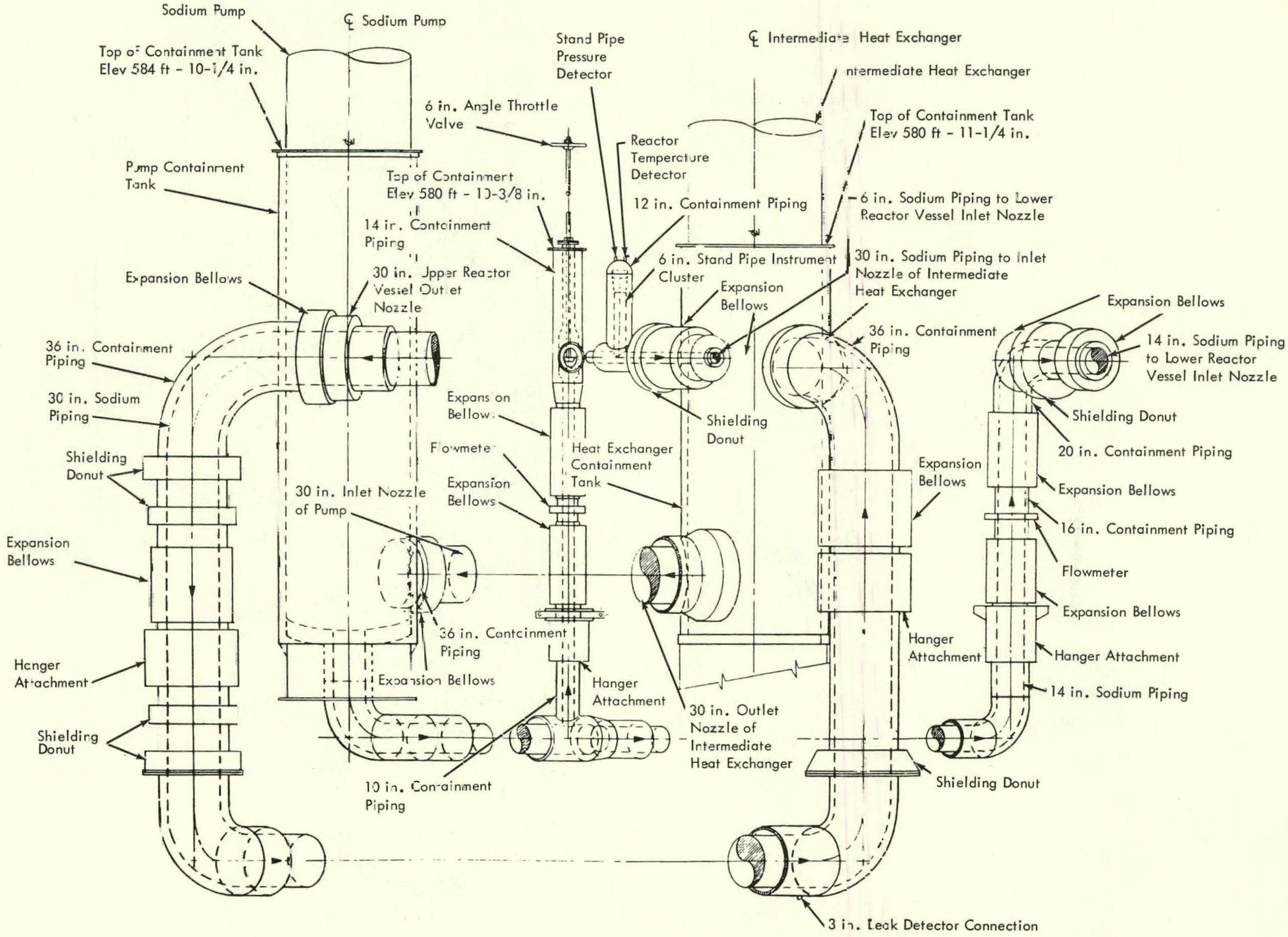


FIG. 55 ELEVATION VIEW OF PRIMARY PIPING

TABLE 15 - FINAL COLD SPRING GAPS

	Diameter of Pipe, Inches		
	30	14	6
Horizontal Linear X, in.	2-9/16	1-1/4	1-5/8
Vertical Y, in.	5/16	3/8	3/8
Horizontal Lateral Z, in.	1/16	1-1/8	9/16

ondary containment in the form of two clamshell sections, was put in place at the pipe bends after the primary piping assembly had been completed.

Since both piping and containment are supported by a joint suspension system, both piping and containment had to be erected simultaneously. Expansion joints were provided in the secondary containment piping at points above and below the hanger attachments in the vertical risers. For shipment and erection they were blocked to prevent excessive expansion.

Shielding collars, made up of canned calcium borate, were fabricated in half-sections and welded together on the piping before the erection of containment was completed. Close attention was paid to the erected gap clearance between the collar and the primary pipe since the collar had to be free to move under all operating conditions, yet it had to be close enough to prevent neutron streaming. A similar collar was attached to the containment.

The 30-inch risers were shipped to the site with 36-inch-diameter chromium-molybdenum containment and bellows attached to the straight sections. The top containment elbows were installed in the field by welding together two halves that were joined at the inner and outer seams. The large 30-inch U-loop was shipped to the site with its containment pipe and two elbows already in place.

At each point in the containment system, a clearance and gap study was made to determine clearances that were necessary to accommodate relative movement between stainless steel piping, chromium-molybdenum containment, and the stationary column of calcium borate. The clearance had to be kept to a minimum, the collars had to be kept centered to prevent streaming, and there could not be any interference at any of the projected plant operating temperatures.

1. Testing Erected Sodium Piping System

There were no system pneumatic tests applied to the primary piping in the field. This procedure was in accordance with the original plan of conducting extensive testing of piping subassemblies in the shop thereby leaving only field welds to be tested at the time of erection. In the case of Fermi, there were no valves available for isolating the piping from the rest of the primary system which contained vessels with a design pressure of 50 psig. System testing on sodium systems is also impractical from the test fluid compatibility standpoint. For this reason, the inner surface of the piping subassemblies was prepared for sodium service at the time of shipment from the fabricator's shop; this surface was protected against the weather during shipment.

The entire system was tested pneumatically with argon during Test Facility operation. A complete vacuum and pressure check was made before sodium fill.

2. Testing Secondary Containment

As described previously, the secondary containment system was open to the below-floor nitrogen atmosphere at about the reactor datum level. To test the leaktightness of the containment, it was necessary to seal the openings which included the annuli between the pump and IHX tanks and the containment, some hanger rod penetrations, and joints in the gas piping. The system was pressurized to 10 psig and, after some difficulty in preventing leakage from the temporary seals, it was determined that the secondary containment system met the design criteria for leaktightness.

3. Pipe Shielding

The calcium borate shielding column, which was designed to envelope the 30-inch riser inside the secondary shield wall, consisted of a ceramic material which was supplied in sheets 1-inch thick. The self-supported column was to be erected so that piping and containment were centered in this column at normal operating conditions. Therefore, the outboard gap between the containment and piping was approximately 2 inches, whereas the inboard gap was approximately 1 inch. The thickness was specified at 20 inches at the upper elbow and 9 inches at the column. Indentations had to be made in the internal surface, and vertical gaps were specified for each. The sheets were bolted and cemented to provide the thickness. Also a sheet metal liner was installed along the inside surface to prevent any particles from falling down into the gaps and causing interferences. Additional structural support was added below the upper elbow by attaching to the secondary shield wall.

4. Insulation and Heating

Approximately 2 inches of low-temperature insulation was added to the 9 inches of calcium borate; none was added at the 20-inch sections. On the 30-inch lower section, a bonded Aenogel-type insulation was applied. This consisted of a layer of high-temperature insulation covered by a layer of low-temperature insulation.

3

The insulated surfaces were wrapped with induction heating wiring according to a specified number of turns per running foot; the wiring was then intermittently stapled to prevent shifting.

E. INTERMEDIATE HEAT EXCHANGERS

The intermediate heat exchanger was constructed of Type 304 stainless steel. Wherever possible all plates, tubes, forgings, and castings were subjected to ultrasonic tests before fabrication as a check against internal defects. All welded joints were radiographed and dye penetrant checked.

After the tubes had been cut to length and bent prior to being installed in the tube sheet, each tube was given a mass spectrometer test. Mass spectrometer leak tests were also performed after heat-welding the tubes to the tube sheet, after fabrication of the shell, and after any repair of a leaking weld.

A view of the completed tube bundle is shown in Figure 56. Both the tubes and the tank were hydrostatically tested, and the tanks were mass spectrometer tested. A primary flow test was conducted on a mock-up of the tube bundle using one half of the unit. As a result of these tests, flow baffles were added to the unit. The IHX tanks were shipped to the site and installed before test facility operation, and the tube bundles were shipped later and installed in the sodium system while it was filled. It was at this point that a decision was made to install an IHX drain system as a permanent feature of the primary system. The pump and IHX tanks were installed before the IHX inlet and pump outlet piping was attached. This procedure simplified establishing the proper elevations and erecting the roller assembly below the IHX tank. The short 30-inch pipe between the pump and IHX tanks was connected after the tanks were in place.

F. PRIMARY SODIUM PUMPS

Delays were experienced in fabricating the primary sodium pumps due to revisions of the primary system layout elevations and a decision to raise the top flange to the floor level. The check valves were sent to the pump manufacturer's shop for attachment to the pump internals. The completed pump tank and internals are shown in the manufacturer's shop in Figure 57. A late revision was the addition of a bell housing over the impeller housing to prevent gas-gulping at the inlet.

The materials were ultrasonically tested, welds were radiographed and liquid penetrant inspected and the pump tank was subjected to a hydrostatic test and a mass spectrometer leak test. The pump internals were tested for performance by water testing throughout the entire range of capacity. The pump characteristic curve shown in Figure 31 shows the equivalent sodium performance based on water tests. Difficulties were

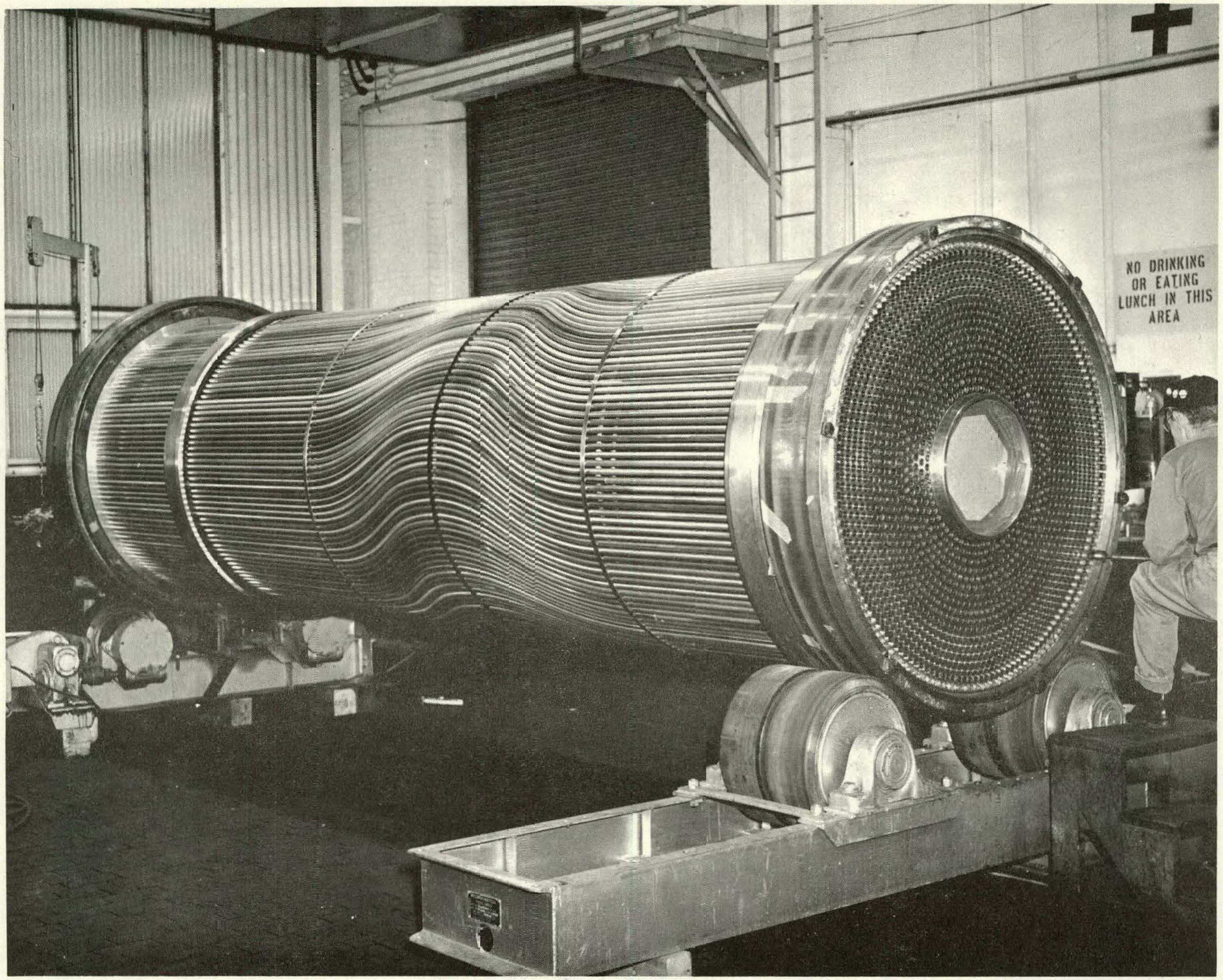


FIG. 56 INTERMEDIATE HEAT EXCHANGER TUBE BUNDLE

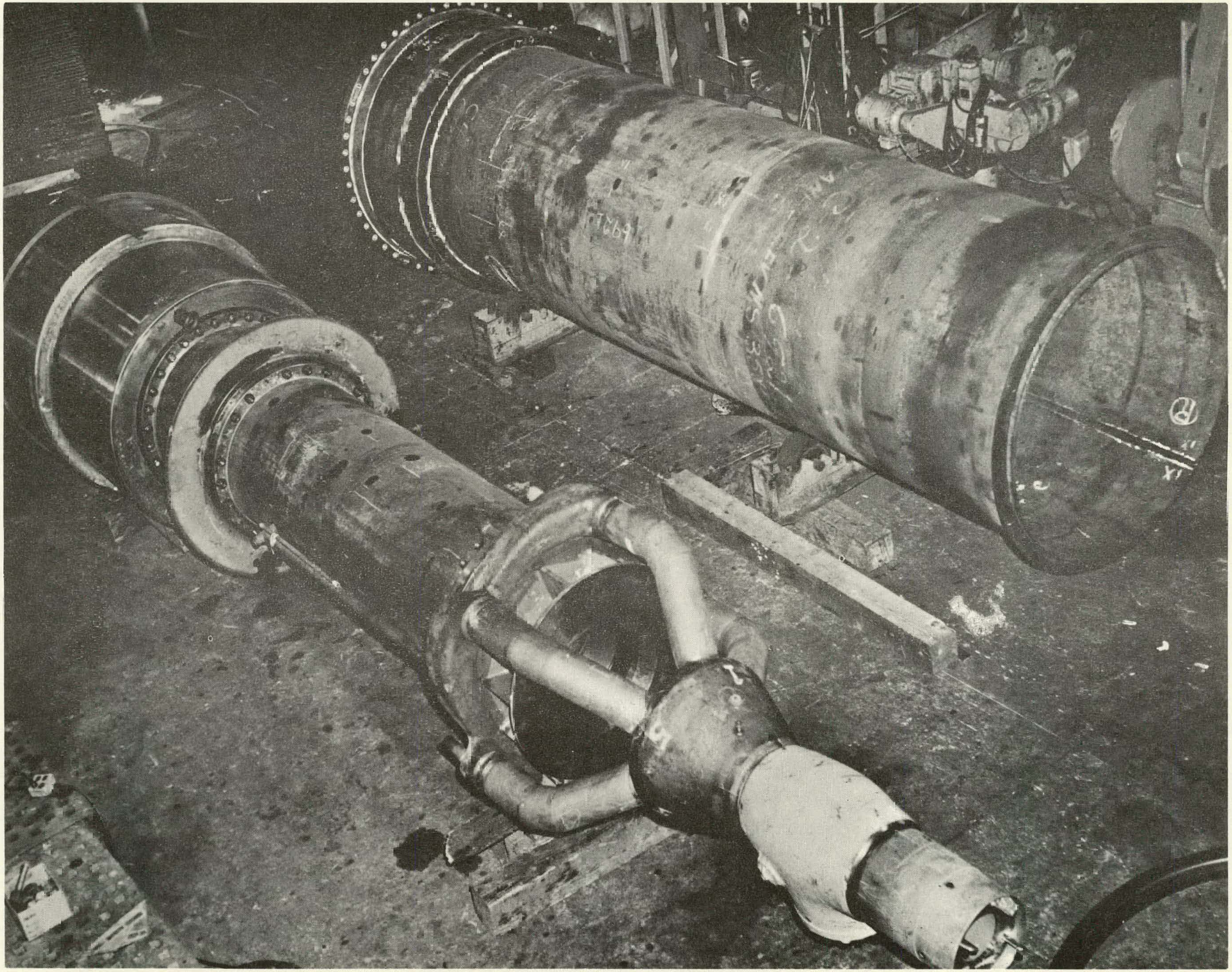


FIG. 57 SODIUM PUMP INTERNALS

experienced with the water testing program, and the pump bearing journals had to be remachined.

The pump tank was shipped to the site for installation with the other component tanks of the primary system. The pump internals were shipped later. At the site, final adjustments were made to the bottom discharge seal, the shield plug was filled with serpentine, and the gas space tanks were bolted into place. While fabrication of the first pump was in progress, orders were placed for the other two pumps. The only difference in design was a modified spring for the shaft seals.

G. VALVES

1. Check Valves

The pump body was cast of Type 304 stainless steel. Sample test bars were poured at the same time. These bars were subjected to a series of heat treatments identical to those proposed for the pump casting. Subsequent tensile tests on these bars showed an ultimate strength higher than that required by the ASTM and a yield point lower than that required to meet ASTM specifications. In view of the fact that the valve body would be subjected to stresses much lower than the observed yield point in the test bars, the initial check valve casting was accepted.

The valve body casting was completely radiographed and all imperfections ground out and repaired. Following the repairs, a dye penetrant examination was made.

Handling lugs used during machining of the valve body were welded in place before and machined off after heat treating. The machined valve body was heat treated at 1960 F to 2010 F.

The valve seat was Stellited after preheating at 800 F to 1200 F, then stabilized at 1850 F for 1 hour and cooled in a forced-air stream. Following this, the valve body was reradiographed 100%.

The hydrostatic test was conducted at 450 psig for 1.5 hours, and there was no leakage. Flow pressure drop tests were conducted in a 10-inch water loop with 8-foot lengths of 16-inch pipe on both sides of the valve. The angle of the disc was varied during these tests and flow was reversed as well as forward. A flow of 66,000 lbs/hr was achieved during those tests, and the predicted pressure drop at design flow was extrapolated.

After completion of the tests, the valve was thoroughly cleaned, dried, and placed in a sealed container in a nitrogen atmosphere under an initial pressure of 3 psig. The valve was shipped to the pump manufacturer for attachment to the pump discharge line.

When sodium hammer problems were experienced with the original check valves and the primary system arrangement, modified check valves were designed, fabricated, and installed by withdrawing each of the pump internals.

2. Throttle Valves

The throttle valve used to control flow through the outer radial blanket elements was fabricated of Type 304 stainless steel. The valve body was machined from a solid forging; the valve body extension was fabricated of Type 304 stainless steel pipe andwelded to the lower forging. Inlet and outlet nozzles were made of 6-inch schedule 40, Type 304 stainless steel pipe. The bellows were triple ply, Type 321 stainless steel and conformed to ASTM specification A 375-55T. The valve plug, as well as the valve seat, was faced with Stellite. Figure 58 shows major valve subassemblies and the completed assembly.

All plate, pipe, tubing, and forgings were ultrasonically tested before fabrication. In addition, the valve body forging was given a macroetch test to ensure quality and determine grain directions. The forging and all welds were subjected to radiographic and dye penetrant examinations.

To prevent contamination of the final valve and to provide production guides, flow tests were performed on a prototype valve. The valve performance characteristics are given in Figure 59.

The valve was thoroughly cleaned and given a hydrostatic pressure test using distilled water in a stainless steel pipe system. Following this test, the valve was dried and given a mass spectrometer leak test.

For shipment, a desiccant was placed inside the valve and the ports sealed. The entire valve was sealed in waterproof paper with a desiccant inside; the assembly was boxed to prevent shipping damage.

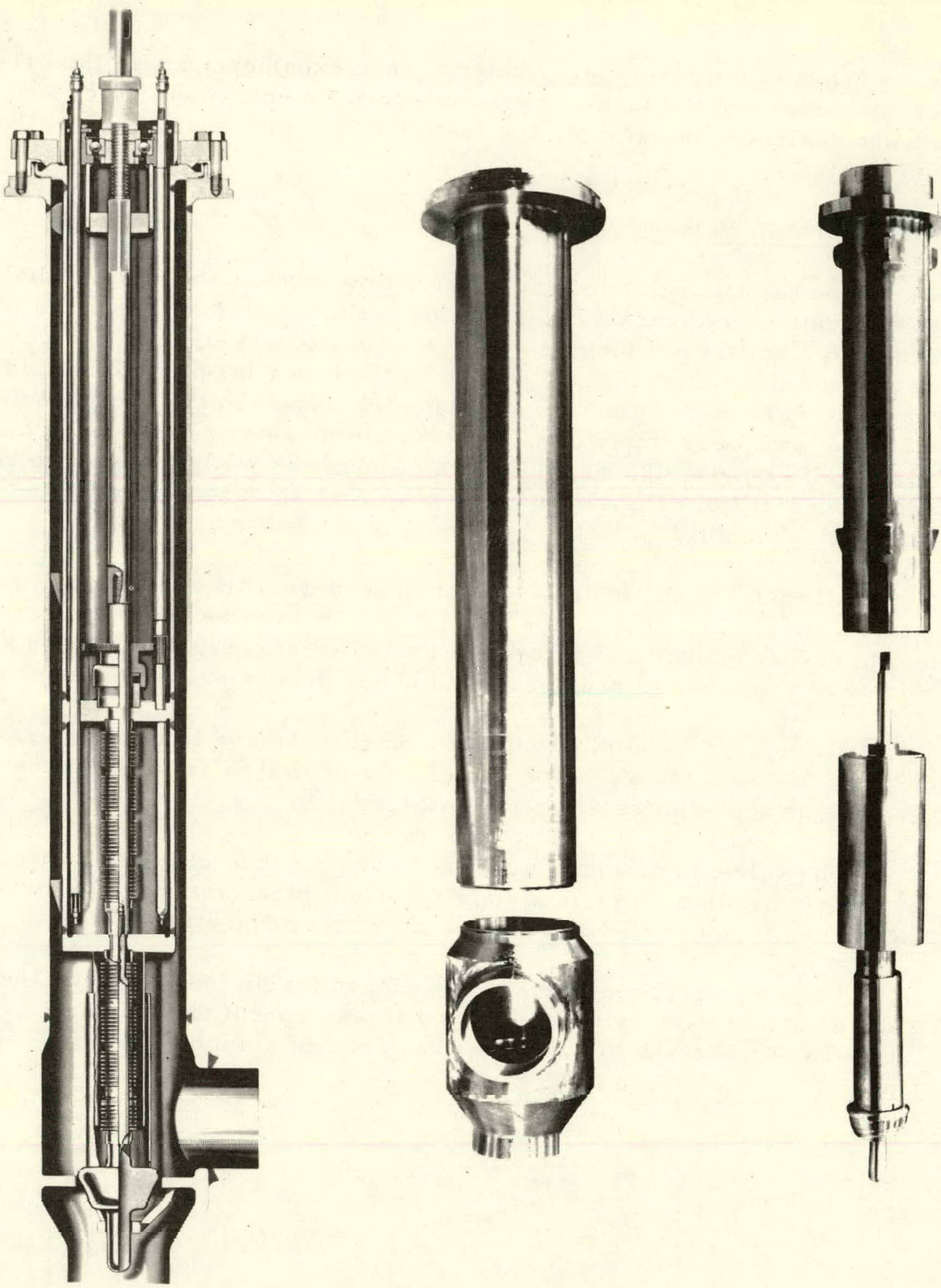


FIG. 58 THROTTLE VALVE ASSEMBLY

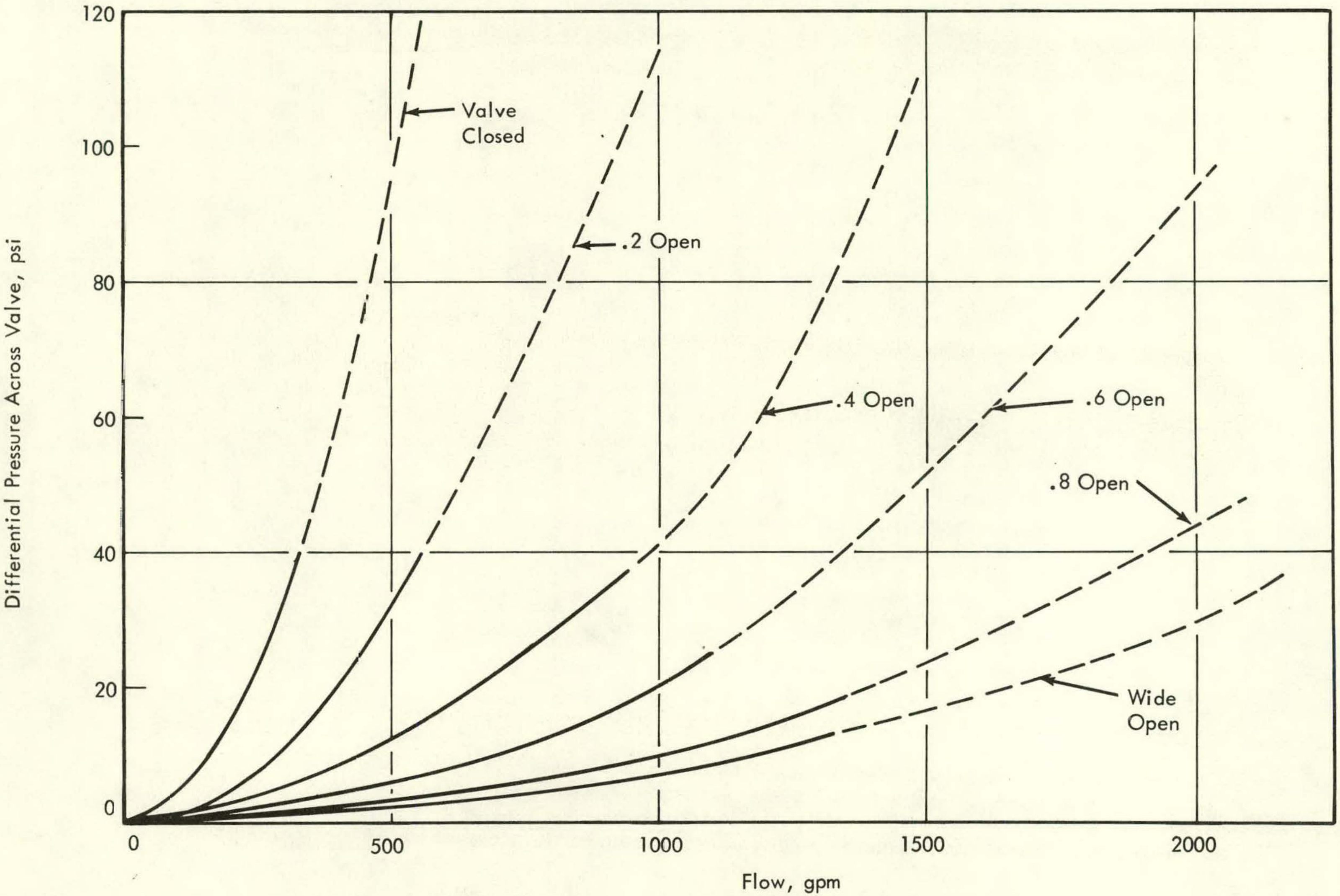


FIG. 59 FLOW CHARACTERISTICS OF THROTTLE VALVE

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

VII. OPERATION OF THE PRIMARY SYSTEM

A. NONNUCLEAR OPERATION

1. Reactor Components Test[12]

a. Test Facility

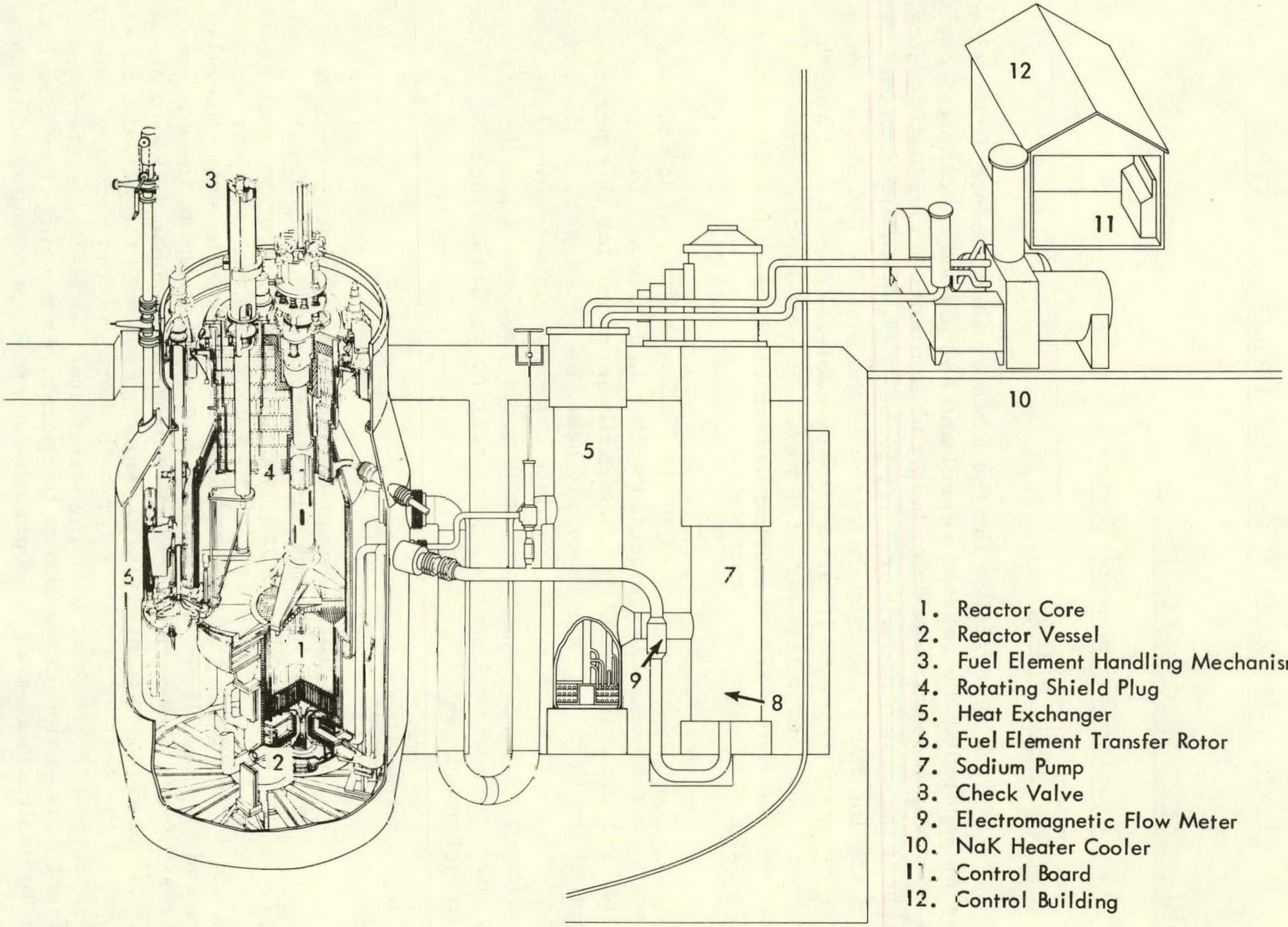
The test facility for the Reactor Components Test, shown in Figure 60, included the reactor vessel and associated mechanical components, the No. 1 primary sodium loop and equipment, and instrumentation throughout the vessel and all primary loops. The primary sodium service system and the inert gas system serviced the test facility and so were considered as parts of the facility. Primary coolant loops No. 2 and No. 3, although not part of the test facility, were subjected to the same test conditions because they were connected hydraulically with the No. 1 loop in the reactor vessel.

In addition, a temporary NaK test loop, which included heating and cooling facilities, was connected with No. 1 primary loop in the intermediate heat exchanger (IHX), temporarily modified for this purpose. The function of the NaK loop was to supply heat, as necessary, to the primary loop and the reactor vessel to simulate expected plant temperature conditions. Inasmuch as the Reactor Components Test was a nonnuclear operation, there was no heat source in the reactor vessel as there would be during actual plant operation.

b. Testing

The Reactor Components Test extended from July 20, 1959, to June 1, 1961 at which time the APDA portion of the plant was turned over to PRDC (see Fig. 61). Test work was carried out under various environmental conditions. The first testing was done in air at ambient temperature. This ambient test permitted observations not obtainable under the other conditions and also facilitated any necessary modifications to the equipment. Next, testing was conducted in air at temperatures up to 500 F to observe the effects of elevated temperature. Concurrent with the hot-air phase, the adequacy of the heating facilities for both inert gas and primary sodium service systems in the reactor building was tested. Finally, systems and equipment were tested in sodium at temperatures ranging from 500 F to 1000 F to duplicate as nearly as possible normal operating conditions.

Hot-Air and Hot-Gas Tests - Hot-air tests were carried out as two separate and distinct operations. During the first operation, a hot air atmosphere was created in the vessel and primary system to check the adequacy of all pipe and equipment heating facilities in the reactor



1. Reactor Core
2. Reactor Vessel
3. Fuel Element Handling Mechanism
4. Rotating Shield Plug
5. Heat Exchanger
6. Fuel Element Transfer Rotor
7. Sodium Pump
8. Check Valve
9. Electromagnetic Flow Meter
10. NaK Heater Cooler
11. Control Board
12. Control Building

FIG. 60 REACTOR COMPONENTS TEST FACILITY

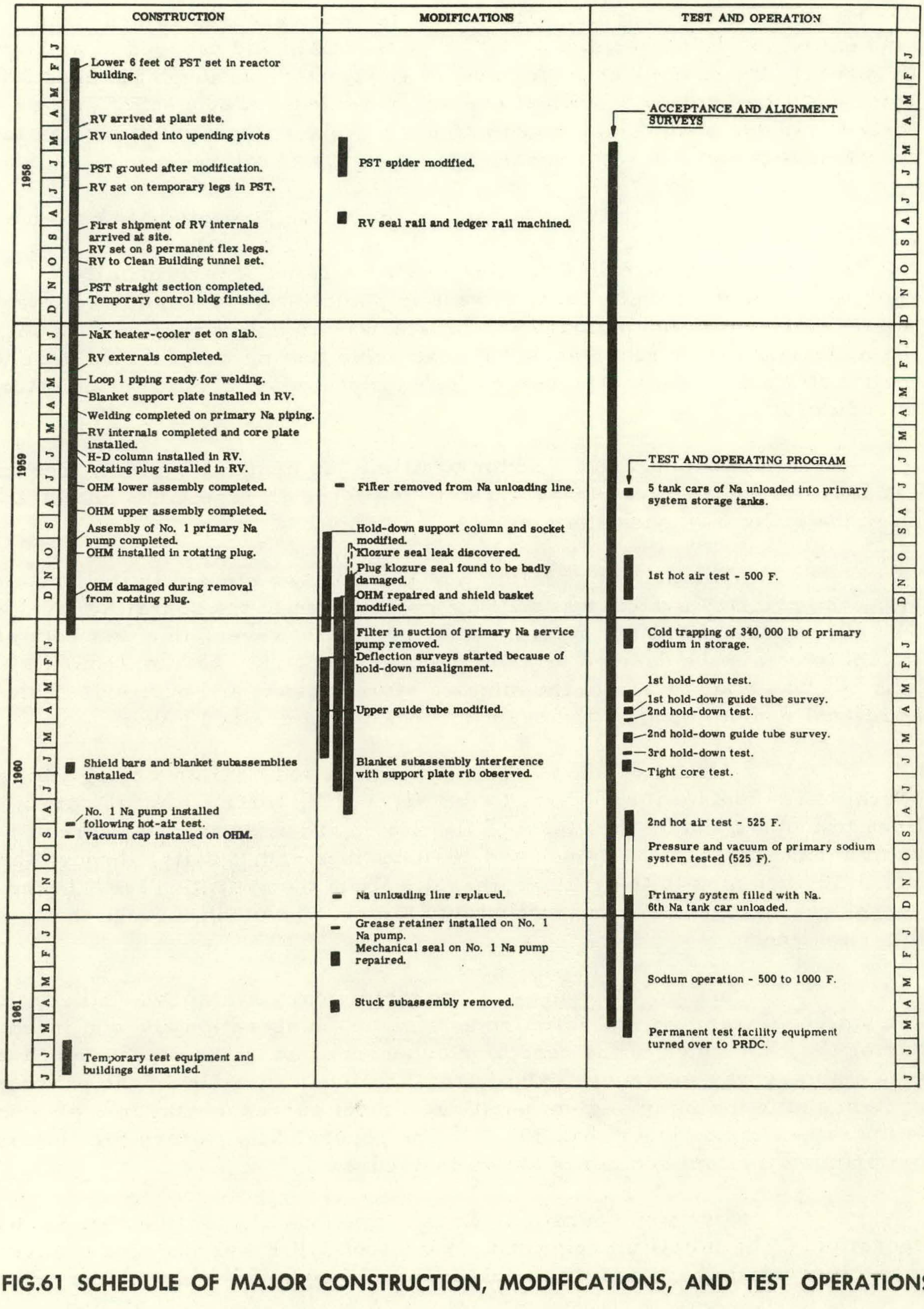


FIG.61 SCHEDULE OF MAJOR CONSTRUCTION, MODIFICATIONS, AND TEST OPERATIONS

building and to monitor the allowable temperature differentials at the various locations in the vessels. The vessel temperature was raised from ambient to 350 F isothermal, to 500 F isothermal, and reduced to ambient. In general, the equipment performed as anticipated, although the need for some additional minor modifications was revealed. A 2000-cfm fan was installed in one of the pump tanks and later replaced by a 7000-cfm fan to assist in equalizing reactor vessel and piping temperatures.

During the second heat-up operation, a heated atmosphere was provided for testing and preparing the system for sodium fill. For this test, the pumps and IHX bundles for loops 2 and 3 were installed, dummy core and blanket elements were in place, and the annulus between the reactor vessel and primary shield tank was completely filled with shielding material. After reaching 500 F, extensive testing was carried out and the reactor vessel temperature was lowered stepwise to 400 F preparatory to sodium fill.

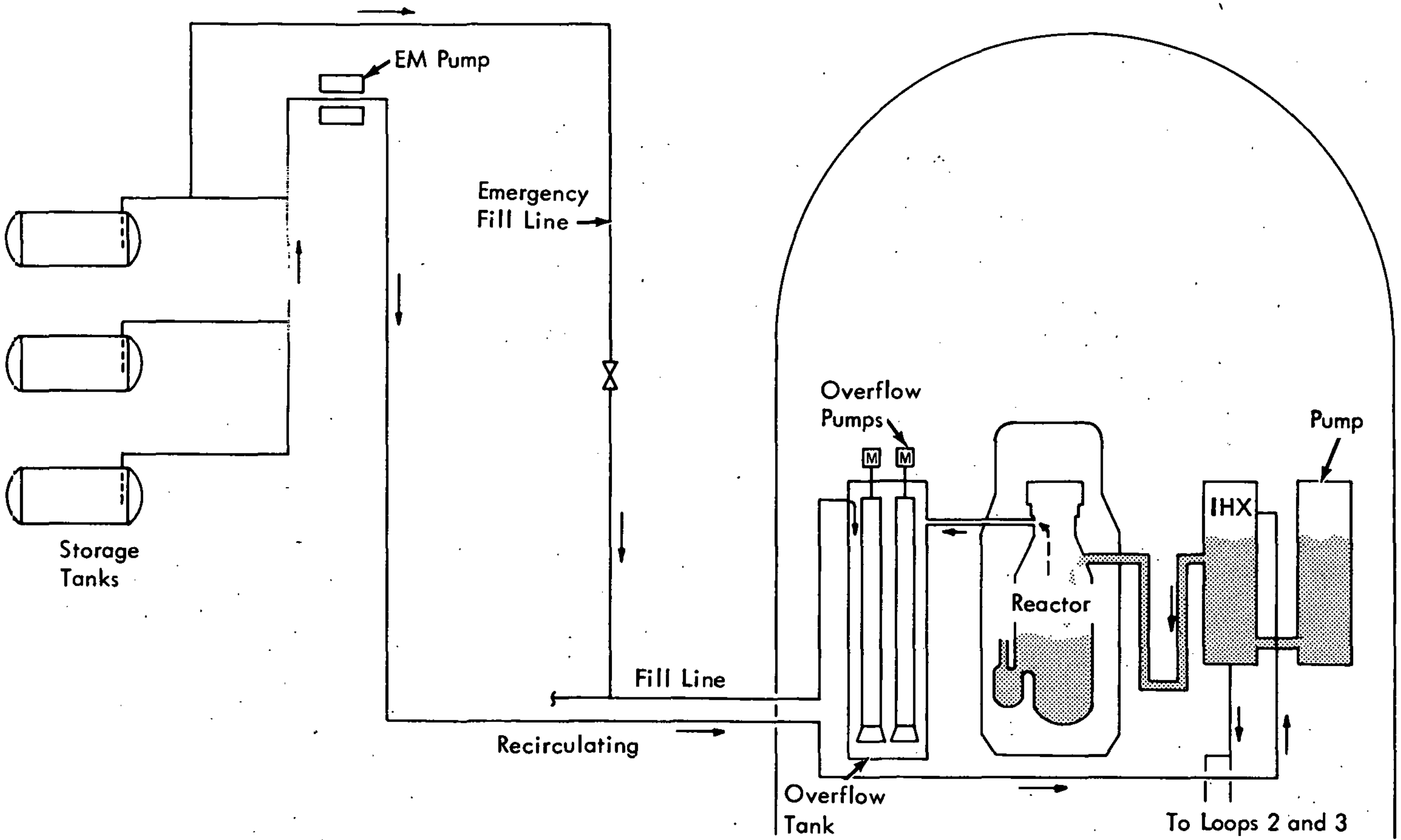
Sodium Fill - Prior to filling the primary system, five tank car loads of sodium were transferred to the three storage tanks and cleaned up in the sodium service system.

The primary system was filled in two steps. In the first step, the primary system was filled under vacuum to the operating level with 345,000 pounds of sodium over a 9-hour period; however, this was not sufficient to obtain the desired level in the overflow tank. Sodium from a sixth tank car was transferred to the emptied storage tanks and purified. This additional sodium brought the overflow tank to the desired level.

Figure 62 shows the main path of sodium flow during filling operations. Sodium flowed first to the No. 1 IHX, through the IHX drain lines to loops 2 and 3, then through IHX No. 2 and 3, through all three pump tanks, through the 6-inch and 14-inch lines and, finally, through the No. 1 30-inch pipe to the reactor vessel. When the operating level in the vessel was reached, sodium spilled into the overflow tank through the overflow line.

Sodium Cleanup - After the primary system was filled with sodium, cold trapping was initiated as a clean-up operation and continued during the remainder of the reactor components test. The objective of the cold trapping was to remove oxides from the inner surfaces of the primary system and to maintain sodium purity at a level corresponding to a plugging temperature of no higher than 300 F. The general flow pattern for cleaning the primary system sodium is shown in Figure 63.

Clean-up operations were continuous at both 500 F and 1000 F operation. The initial plugging temperature of 320 F was reduced to less than 250 F within 24 hours after cold trapping began. During the period



Note: Filled Primary System Nov. 30 and Dec. 1, 1960

FIG. 62 SODIUM FLOW PATHS FOR FILLING PRIMARY SYSTEM

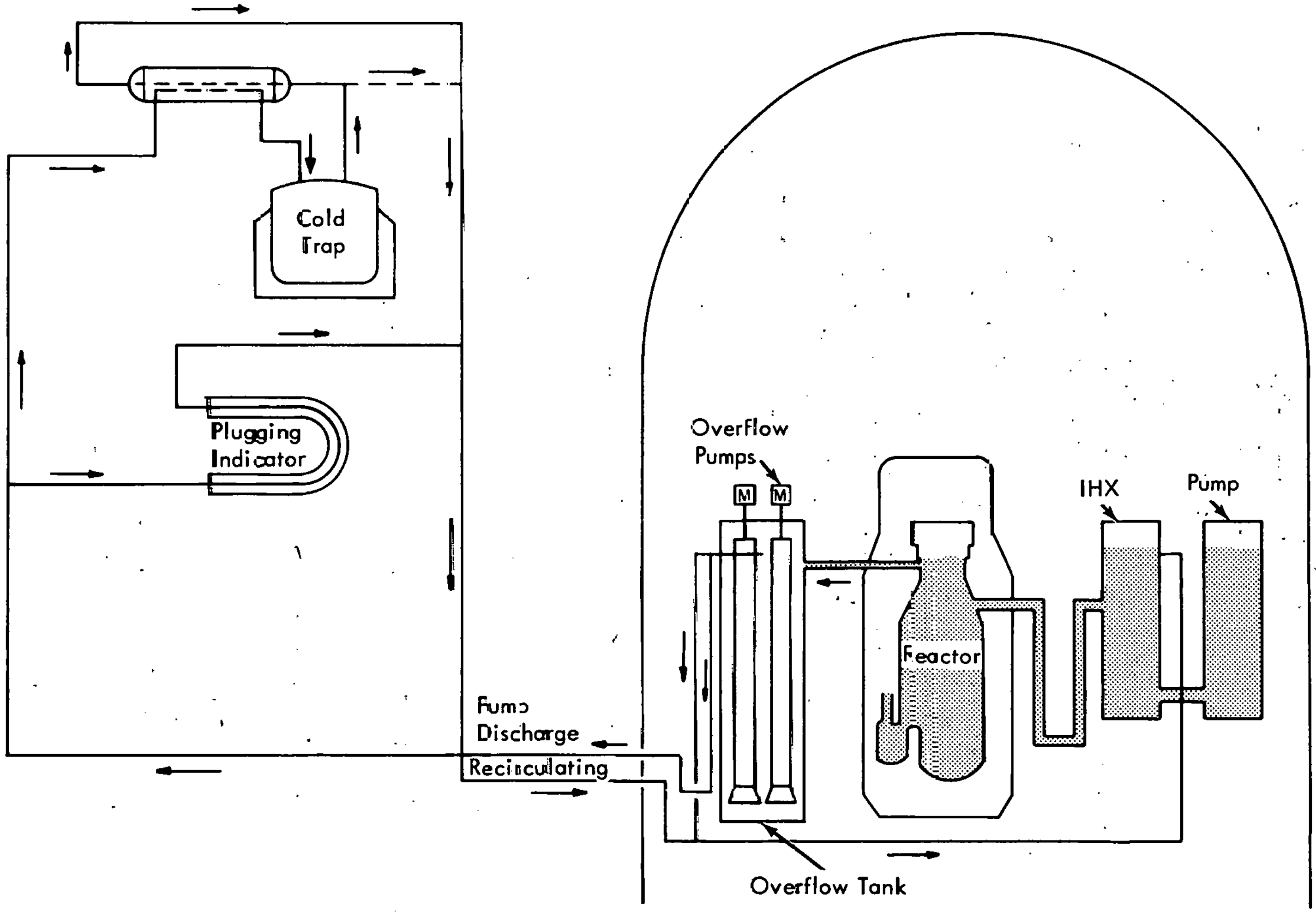


FIG. 63 SODIUM FLOW PATH FOR PRIMARY SYSTEM CLEANUP

when impurities could be observed on the surface of the sodium in the reactor vessel, cold trapping was continuous. When the surface appeared to be relatively free of impurities, at the 500 F isothermal test stage, cold trapping was changed from a continuous operation to intermittent operation.

Operation at 1000 F - The sodium-filled vessel and primary system was heated to 1000 F to determine any modifications necessary before the system became radioactive. It was also felt that any stress relieving of the reactor and fuel handling mechanisms that might occur at 1000 F would preclude further stress relieving during nuclear operation. Thus, subsequent settings of operating mechanisms for matching with subassemblies should remain valid for an extended period.

For the 1000 F tests, the No. 1 primary sodium pump and temporary NaK loop were the principal heat sources. Electric heating facilities were used only to maintain ΔT's at an acceptable value. There were cooling facilities in the NaK loop. To conserve system heat, it was necessary to discontinue operation of the cold trap and plugging indicator for three days while the NaK heater-cooler was being repaired. Resumption of plugging indicator operations indicated that the plugging temperature had risen to a point in excess of 550 F.

Figure 64 shows the system temperature during the 1000 F test period measured at the reactor support plate.

Outgassing of Plug Graphite - During operations at 1000 F, the NaK heater-cooler failed. Subsequent analysis showed 15% carbon dioxide in the cover gas and free carbon and sodium carbonate in the sodium coolant. Continuous operation of the cold trap combined with argon cover gas purging produced a more acceptable plugging temperature and reduced the carbon dioxide content in the cover gas, respectively.

From the various test data, the following conclusions were drawn:

a. During the hot air test, a considerable quantity of gases that had been absorbed by the graphite in the early test stages was liberated.

b. The evacuation operation prior to sodium filling pulled out and dispersed virtually all residual absorbed gases.

c. During sodium operation at 500 F isothermal, practically no carbon monoxide or carbon dioxide was evolved.

d. During the 1000 F tests, when the temperatures exceeded 700 F, the graphite in the plug could have liberated carbon monoxide and carbon dioxide in significant quantities.

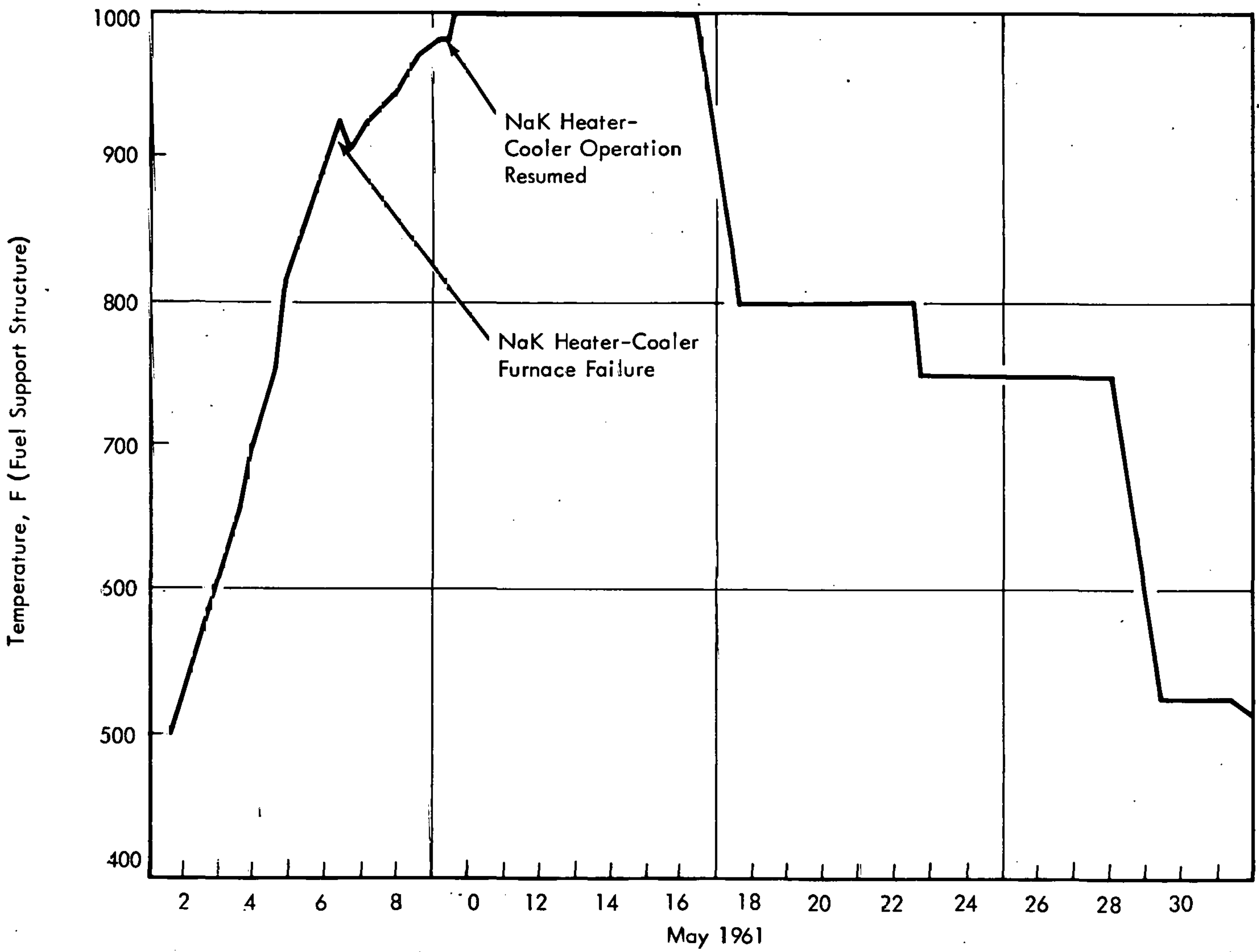


FIG. 64 REACTOR VESSEL TEMPERATURE DURING 1000 F TEST

e. The gases then combined with the sodium in the vessel to contribute to the formation of free carbon and sodium carbonate.[13]

After investigating the situation, it was concluded that future operating conditions would not lead to excessive offgassing and no futher corrective action was necessary with respect to the graphite in the shield plug.

Pump and Throttle Valve Tests - The No. 1 primary sodium pump was tested in sodium at temperatures up to 1000 F. During these tests, dummy subassemblies occupied the reactor core and blanket positions. Tests were also conducted at 800 F and 750 F to determine the hydraulic performance of the pump when operating singly in the primary loop and to produce calibration data for the 14-inch and 6-inch flowmeters in the primary sodium loop.

Table 16 is a summary of results obtained for all the pump test runs at 500 F. The developed head, pump speeds, and power values were plotted on the already established characteristic curves for the No. 1 pump as shown in Figure 65. These curves determined the triangles of errors shown, and in turn, the head-flow characteristic curve for the primary system with dummy subassemblies in the core and blanket. The actual pump discharge flow rate was then obtained from this system characteristic curve. Data from experimental operation determined the curve for 860 rpm, while the curves for all other speeds were established according to similarity laws.

Total pump operating time during the Reactor Components Test was 2035 hours, with the majority of time at 600 and 700 rpm.

Level differentials between the reactor and pump tank were measured and are shown in Table 17.

The pump was also tested for vibration, and shaft torque was frequently measured. In general, mechanical operation of the pump was very satisfactory.

Flow decay characteristics were measured at various temperatures by tripping the breaker at 700 rpm. A typical curve is shown in Figure 66.

Calibration of Flowmeters - The flowmeter calibration work for the 14- and 6-inch flowmeters established a relationship between millivolt output from the flowmeter and the actual sodium flow obtained from pump test data. Actual flow through the 6-inch flowmeter was determined from the relationship:

$$Q = 31\sqrt{H}$$

TABLE 16 - OPERATING CHARACTERISTICS OF NO. 1 PRIMARY PUMP AND CALIBRATION DATA FOR 14-INCH AND 6-INCH FLOWMETERS, LOOP 1

Position of Loop 3 Throttle Valve	Pump Speed, Hand Tachometer, rpm	Head Developed by Pump, H, ft of Na	Power to Pump Motor, Kw	Pump Discharge Pressure, psig	Pump Discharge Flow, Q, gpm	* Flow Through Flowmeters: 6 Inch $Q_6 = 31\sqrt{H}$, gpm	14 Inch $Q_{14} = Q - Q_6$, gpm	Indicated Flows (Corrected for Zero Error): 6-Inch Pipe, gpm	14-Inch Pipe, gpm	** Millivolt Output from Flowmeters (From APDA Flow Recorder Calibrations of 1-12-60): 6-Inch Pipe, mv	14-Inch Pipe, mv
Open	863	306.5	955	117	13,250	528	12,722	637	8,041	9.07	11.81
Open	826	281.4	875	108	12,750	510	12,240	620	8,090	8.80	11.85
Open	802	265.4	820	102	12,400	493	11,907	602	7,860	8.60	11.52
Open	752	236.0	725	91	11,650	428	11,222	520	7,580	7.35	11.15
Closed	752	246.7	685	95	11,222	486		594	7,080		
Closed	750	243.8	693	94	10,550	482	10,070	593	7,240	8.44	10.60
Closed	704	215.3	595	83	9,850	485	9,365	650	6,650	7.82	9.80
Open	703	207.4	630	80	10,900			525	7,020		
Open	650	176.4	539	68	10,050			491	6,716		
Closed	650	188.9	512	73	9,000	425	8,575	518	6,289	7.38	9.23
Open	600	157.9	463	61	8,750			460	6,400		
Closed	600	162.8	440	63	8,150	394	7,756	480	6,060	6.83	8.90
Closed	550	139.3	370	54	7,550	366	7,184	442	5,730	6.30	8.40
Open	550	131.9	390	51	8,100			415	6,090		
Open	500	111.0	320	43	7,200			382	5,770		
Closed	508	116.0	318	45	6,850	334	6,516	404	5,368	5.75	7.88
Closed	465	95.1	264	37	6,400	302	6,098	368	5,053	5.23	7.40
Open	465	90.2	275	35	7,000			348	5,458		
Open	400	69.2	214	27	6,100			298	4,941		
Closed	402	71.5	205	28	5,750	260	5,490	312	4,570	4.42	6.70
Closed	384	66.4	190	26		252		308	4,425		
Open	362	61.3	181	24	5,650			276	4,650		
Closed	372	63.9	179	25	5,100	245	4,855	304	4,360	4.30	6.40

Test Period: January 25 through February 2, 1961

Sodium Temperature: 500 F
Throttle valves closed in loops 1 and 2

Note:
* Relationship $Q_6 = 31\sqrt{H}$ obtained from Loop 1 throttle valve test data

$Q_6 = 31\sqrt{H}$ applies when throttle valves, all loops, are closed.

** Millivolt output from flowmeters obtained from APDA flow recorder calibrations (mv vs indicated flow) (1-12-60). Assume mv output from flowmeter and mv input to recorder are the same.

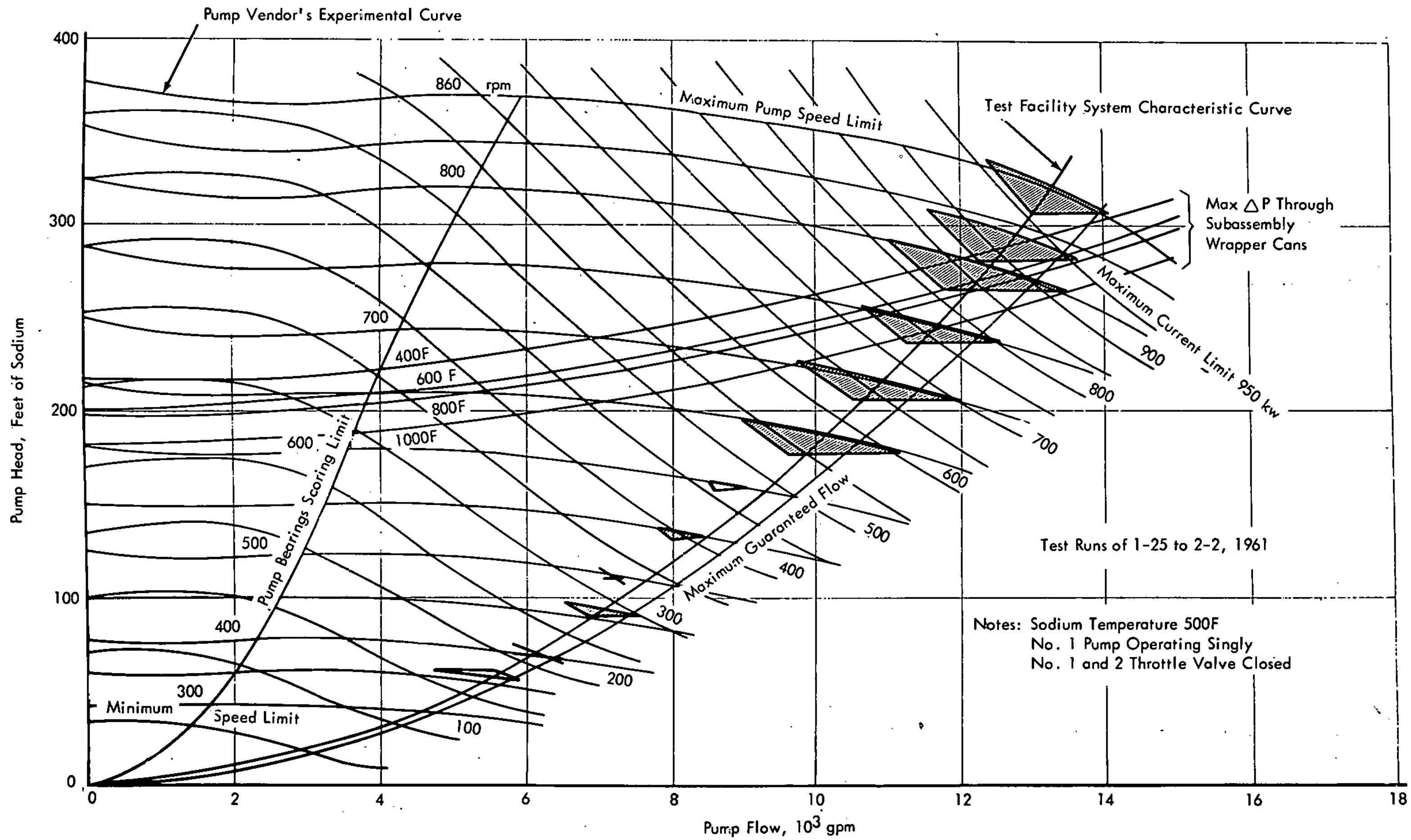


FIG.65 PRIMARY SYSTEM CHARACTERISTICS CURVE, NO. 3 THROTTLE VALVE OPEN

Ellis - I would like to use this; if there are any problems - let us know?

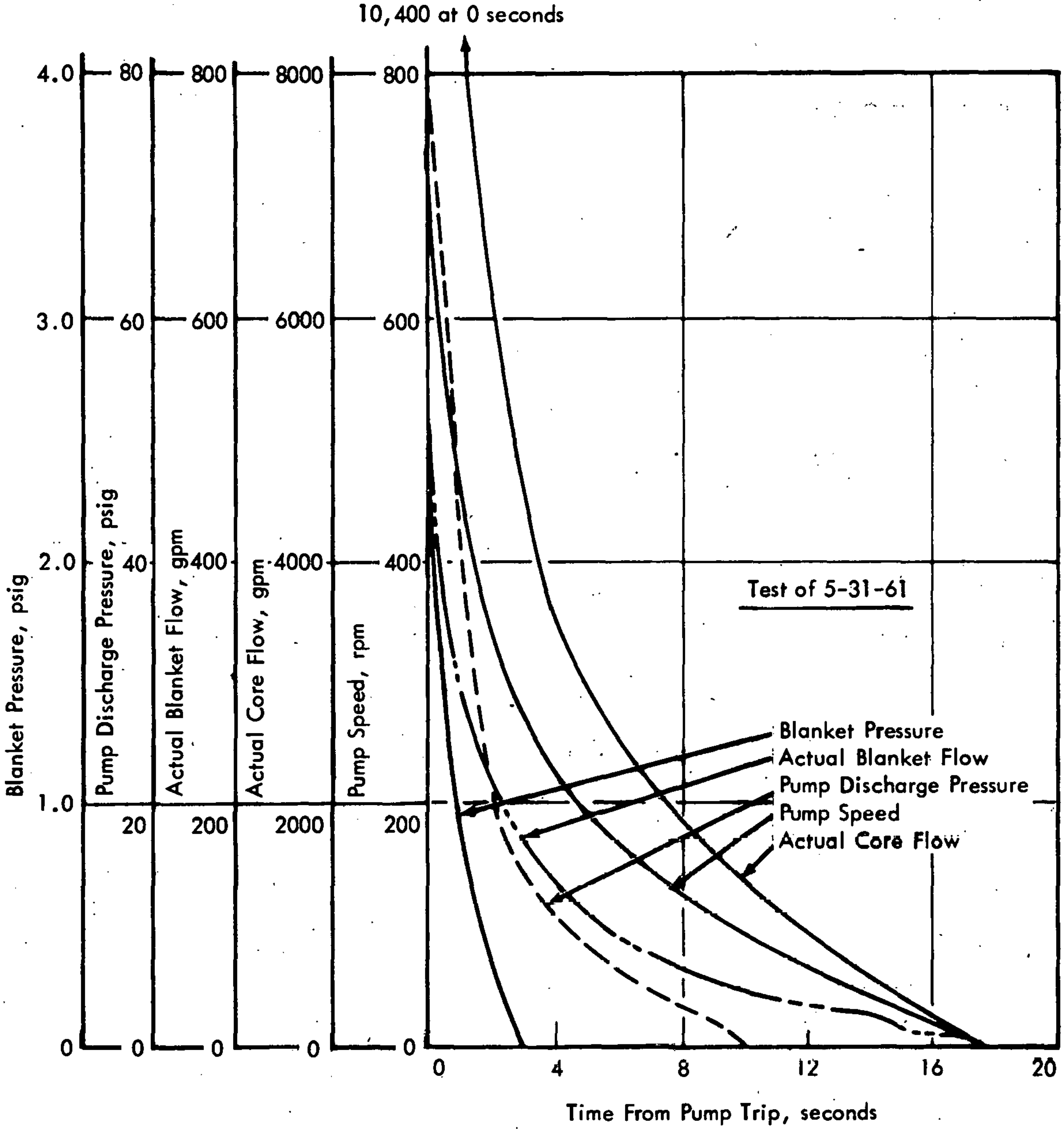


FIG. 66 NO. 1 PRIMARY PUMP FLOW DECAY FROM 700 RPM, 500 F SODIUM

TABLE 17 - SODIUM LEVEL DIFFERENTIAL BETWEEN REACTOR AND PUMP TANK*

Speed, rpm	Flow, gpm	Change in Level, ft of Na
860	13,250	2.22
800	12,400	2.17
700	10,900	1.57
600	8,750	1.13
500	7,200	0.93
400	6,100	0.77

* Throttle valves were closed in loops 1 and 2 and open in loop 3.

where H is the head developed by the pump. This formula was derived from throttle valve test data. The 14-inch flow was therefore the difference between total pump discharge flow as shown in Figure 65 and the corresponding 6-inch flow.

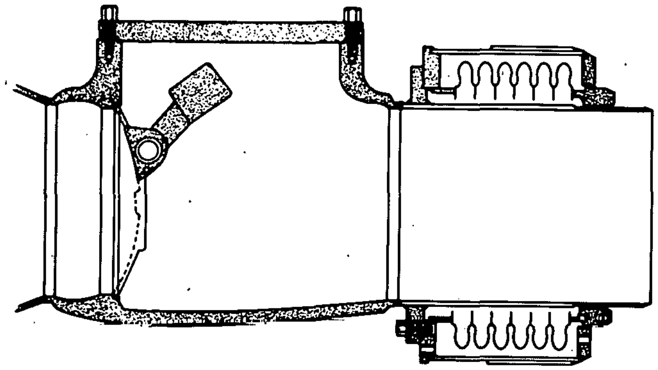
Deflections in the Primary Piping During Check Valve Closure - A sizeable deflection of the piping and an audible slam were observed and heard when the check valve was closed under certain conditions.* A special gage was designed and attached to the 14-inch piping at the 566-foot elevation, and a maximum vertical movement of 3/16 inch was measured. This led to the subsequent replacement of the check valves with an improved design valve as shown in Figure 67.

2. Preparation for Nuclear Operation

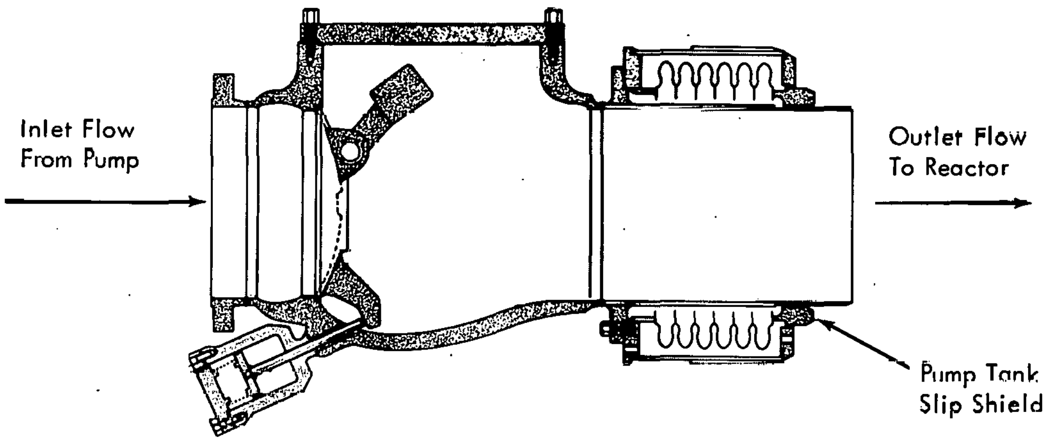
The temporary NaK heating coil in IHX tank No. 1 was replaced by the plant tube bundle while the system was filled with sodium. The need for a drain system that would drop the level in the IHX tanks became obvious. A drain system that could be frozen was installed at the bottom of each IHX. The entire header was terminated with one valve at the entrance to the overflow tank.

Extensive cleanup of the primary system sodium was conducted as a result of contamination with carbonaceous materials during operation of the test facility. This material emanated as offgasses from the canned graphite in the rotating plug during the 1000 F tests described on page 131. This clean-up operation included filtering, cold trapping, and hot trapping.

* Refer to Section VII-B for details.



OLD VALVE



NEW VALVE

FIG. 67 CHECK VALVE DESIGNS

B. NUCLEAR OPERATION

A provisional license was obtained to operate the reactor at one megawatt, and criticality was attained with 99 subassemblies in the core on August 23, 1963.

1. Replacement of Check Valves

During preoperational testing of the primary sodium system prior to initial criticality, visual inspection and measurements disclosed pipe movements of considerable magnitude, both when a single pump was started with the main motor and when a single pump was tripped with two or three pumps running in parallel. The check valves could be heard to slam shut during these operations.

The magnitude of the problem was not immediately determined, and to proceed with previously scheduled test operations without risking damage to the piping and equipment, two precautions were taken: the check valves were preclosed at reduced sodium flow before starting a main motor by operating the pumps at pony motor speed, and temporary interlocks were installed to trip all pumps simultaneously upon a single pump tripout.

After the preliminary observation of pipe deflection and audible indications of check valve slamming, field test work and calculations were undertaken to obtain sufficient data relative to the primary sodium flow rates and pipe deflections so that the surge pressures could be calculated and correlated with the measured pipe deflections. Each of the three primary sodium loops was equipped with a series of tranducers to measure pipe deflections during the pressure surges created by the valve closure.

In addition to the pipe deflections, the sodium pump speeds and flows were measured. The pump speeds were increased in 50 rpm increments up to as high as 550 rpm. The maximum deflection measured was 135 mils in loop No. 2.

After an intensive investigation of the effects of sodium hammer on the equipment nozzles, the decision was made to replace the original check valves. The replacement check valves had four design revisions, illustrated in Figure 67: a flange connection to discharge, a spring-loaded disc that overcame the closing inertia of the disc, a dash pot that slowed the closing of the disc in the final 12 degrees of closure, and an enlarged body design that allowed more backflow while the disc was closing. For natural circulation conditions, the disc remains open at 12 degrees.

Subsequent to initial criticality, each of the pump internals was withdrawn with remote maintenance removal equipment. The internals were disassembled, steam cleaned, and inspected. The check valves were replaced and the pumps were reinserted into the pump tanks. The results of tests conducted to show the improved performance of the new check valves are shown in Figure 68. As an added precaution, a restraint was added to the 6-inch and 14-inch piping at the reactor inlet of loop No. 3 to prevent excessive vertical movement of the piping. The restraint consists of a solid hanger attached to the wall of the PST and attached to the lower elbows to the 6-inch and 14-inch piping. Horizontal radial movement is permitted but vertical movement is restrained.

2. Intermediate Heat Exchanger Performance

In heat balance tests conducted in 1966, the reactor was operating at powers of 67 Mwt and 100 Mwt. Table 18 gives data on IHX performance.

TABLE 18 - IHX PERFORMANCE DURING HEAT BALANCE TESTS

	Heat Balance, 67 Mwt June 24		Heat Balance, 100 Mwt July 11		
	Loop 1	Loop 3	Loop 1	Loop 2	Loop 3
Primary Flow, lb/hr x 10^6	2.73	2.99	3.57	3.53	3.27
Temp in, F	671	675	655	656	657
Temp out, F	542.8	552.7	555.2	553.9	551.5
ΔT	128.2	122.3	100	102	106
Secondary Flow, lb/hr x 10^6	3.17	3.08	3.21	3.29	3.28
Temp in, F	519.5	516.9	518.5	516.2	516
Temp out, F	629.6	635.4	629.8	625.6	621.4
ΔT	110.1	118.5	111.3	109.4	105.4
Heat Exchanger, Mwt	31.91	33.37	32.67	32.94	31.65
Heat Exchanger, Btu/hr x 10^6	108.8	113.8	111.3	112.2	108
Effective Surface, ft^2	6620	6650	6550	6550	6550
Effective Tube Length, ft	15.5	15.6	15.35	15.35	15.35
Overall Heat Transfer Coefficient, Btu/hr-ft^2-F	527	453	562	471	462

The low heat transfer coefficient, 453 to 562 Btu/hr-ft^2-F, was approximately one-half the value predicted for full-power and full-flow conditions, (978 Btu/hr-ft^2-F at 143 Mwt conditions) for each IHX. There were no predictions by the manufacturer for performance at lower power levels.

Based on the manufacturer's performance predictions for full load, it was determined that his heat transfer coefficient was based on the full wetted area of the tubes. After the heat balance tests at 67 Mwt and 100 Mwt indicated a larger than anticipated ΔT across the units, APDA calculated the part load performance of the IHX based on wetted area. The calculated overall heat transfer coefficient for part load was 1300 Btu/hr-ft^2-F as compared to the considerably lower value given in Table 18.

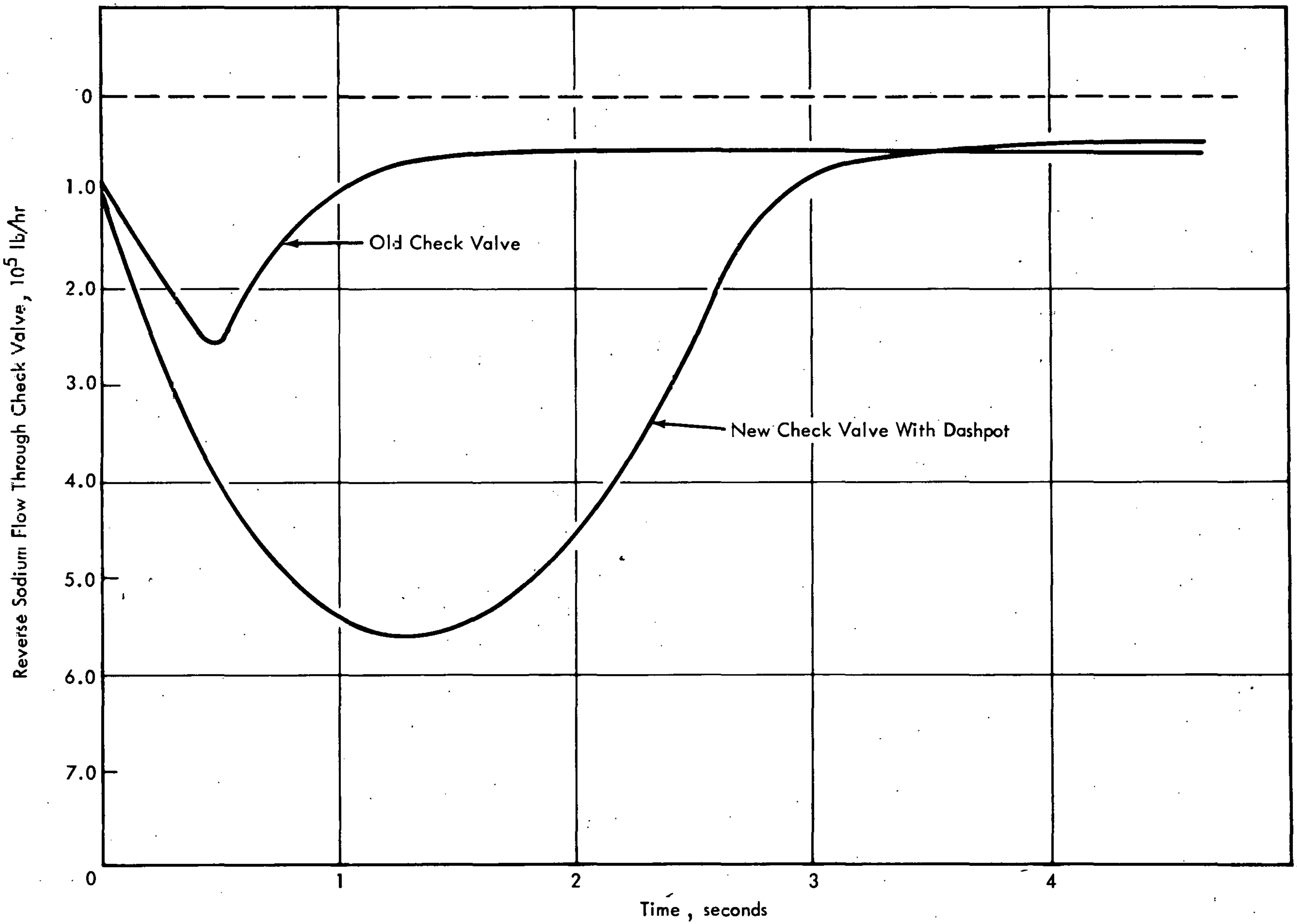


FIG. 68 PERFORMANCE CURVES OF OLD AND NEW CHECK VALVES

3. Summary of Primary Pump Operation

Table 19 presents the yearly operating hours for the primary pumps.

TABLE 19 - HISTORY OF PUMP OPERATION

	Time, Hours		
	Pump No. 1	Pump No. 2	Pump No. 3
Test Facility Operation, Jan - May 1961	2,035		
June 1961 to end of 1962	2,425	1,800	2,200
1963	3,128	4,739	4,680
1964	6,323	5,891	4,913
1965	7,683	7,347	6,868
1966	6,982	6,907	7,343
Total	28,576	26,684	26,004

4. Primary System Carburization Program

Test sections of material were subjected to 1200 F for periods of 100 hours in a facility attached to the primary sodium service system. In all tests the surface carbon content was well below the critical level of 0.5 w/o, and it was concluded that there was almost no carburization potential.

5. Summary of High-Power Operation, > 1 Mwt

On December 29, 1965, a series of tests was started and operation was scheduled at increasingly higher power levels intended to lead to eventual operation at 200 Mwt. A nuclear test program and a plant test program were coordinated for monitoring operation at the lower levels to predict operation at the higher levels. Figure 69 shows a record of reactor power for this period. Nuclear tests were conducted at 20-Mwt, 67-Mwt, and 100-Mwt levels.

Analysis of the nuclear tests conducted through 100 Mwt indicated a wide margin of stability for operation of the initial core loading at the design power of 200 Mwt. The transfer function measurements indicated no tendency whatever toward instability in either manual or automatic modes of control. Power coefficient measurements indicated a somewhat reduced value over that predicted, but it has been shown conclusively that the value obtained is well within the safety criteria.

In plant tests, particular attention was given to monitoring the cover gas and primary sodium impurities. Likewise, careful monitoring of the primary sodium systems was carried out. This included primary pumps, cover gas pressure control, and area ventilation and temperature

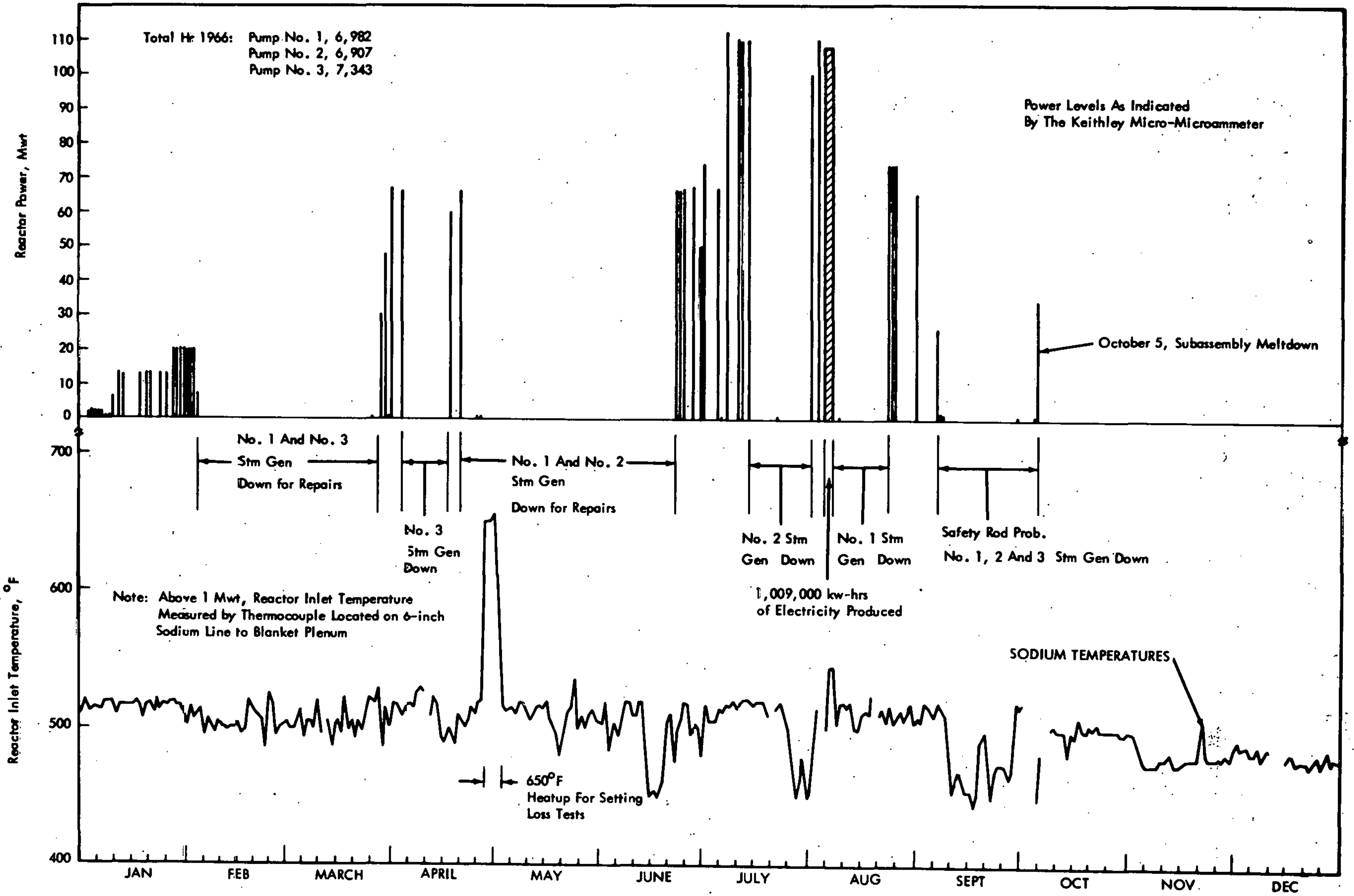


FIG. 69 FERMI OPERATING HISTORY, 1966

Any problems on this?

control. Satisfactory performance was obtained from primary system mechanisms such as the holddown device, offset fuel handling mechanism, oscillator rod assembly, and the control and safety rods as well as their drives. A comprehensive shield test program demonstrated the adequacy of all shields for operation with equilibrium levels of radioactivity throughout the primary system.

A number of flow decay tests were conducted in March 1966, to provide information for the 110-Mwt program hazards analysis. All tests were conducted with a primary system temperature of 500 F and a subcritical core loading of 101 core subassemblies. A total of fourteen tests was conducted, using flows of 2.57 or 3.07 x 10^6 lb/hr through the 14-inch lines with three loops operating, or using 3.7 to 3.8 x 10^6 lb/hr with two loops operating. Figure 70 shows the plot of flow in the 14-inch line when all three pumps were tripped simultaneously. The same chart shows data from another test when pony motor startup was defeated. In similar tests with two loop operation, the flows stabilized at pony motor flow after approximately 20 seconds.

Heat balance measurements were made at 67 Mwt and 100 Mwt, the results indicating that the performance of the intermediate heat exchangers was approximately 50% of that predicted by the manufacturer for 430-Mwt operation.

Steam generator maintenance, as well as instability problems associated with the units, resulted in frequent interruptions of reactor operation during the test program.

In spite of these problems which were revealed during plant operations up to 100 Mwt, most plant components and systems performed quite satisfactorily. On the weekend of August 5 through August 7, 1966, the reactor was operated for 53 hours at the 100-Mwt level with the turbine-generator in service, generating about 22 Mwe to The Detroit Edison Company's electrical system.

On October 5, 1966, while the reactor was being brought to power, high cover gas radioactivity and a loss of reactivity were encountered. As a result, the reactor was shut down from a power level of 34 Mwt. The indications were that fuel damage had occurred which was subsequently determined to have been the result of local fuel melting in two adjacent subassemblies. Subsequent investigations have indicated apparent flow blockage to the subassemblies by a zirconium sheet that had accidently become detached from a steel structure in the reactor inlet plenum.

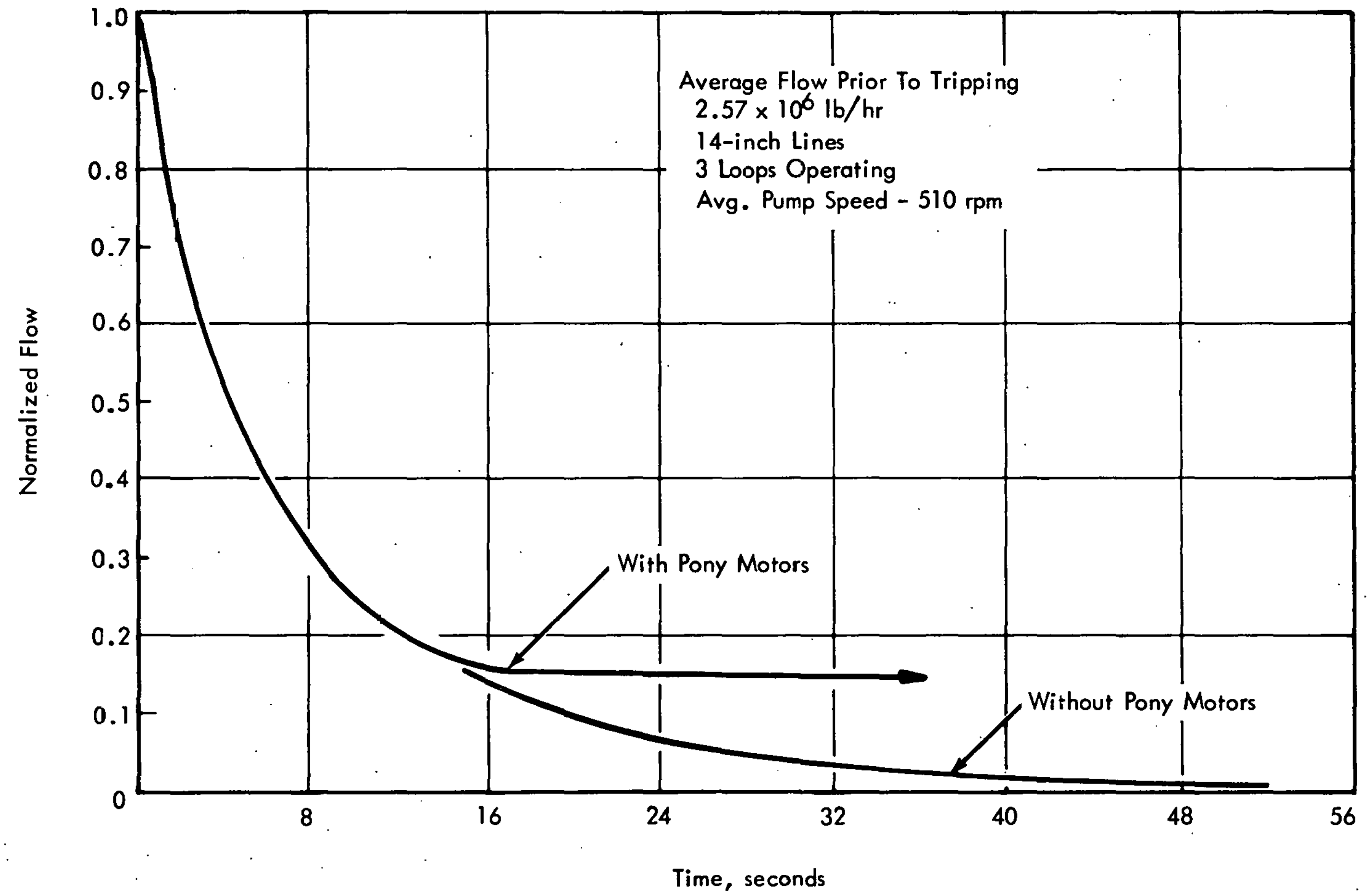


FIG.70 PUMP FLOW DECAY

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

VIII. EVALUATION

A. GENERAL PERFORMANCE

The performance of the primary system has been generally reliable and satisfactory; it provides a significant record of successful operation and durability of a large, piped sodium heat transport system in a nuclear reactor power plant. Since the system was filled with sodium in November 1960, the materials and components within the system have been exposed almost continuously to a sodium environment under various temperature and flow conditions. These conditions included operation of the Reactor Components Test Facility portion of the primary system at 1000 F isothermal for 7 days. It is noteworthy that prior to nuclear operation the sodium was drained from the reactor and part of the primary system to perform maintenance and repair work in the reactor vessel. Under this condition, some of the materials and components in the system were exposed to the argon cover gas environment for a period of several weeks until the system was again refilled with sodium. The monitoring and control of sodium purity has been satisfactorily accomplished and there has been essentially no evidence of deterioration of the materials and components which have been exposed to these environmental conditions for this long period of time.

The system has been subjected to a considerable number of startups and shutdowns with operation on both three loops and two loops over a wide range of flow rates and temperature levels. The system has experienced temperature differences as high as 125 F during operation of the reactor at power levels up to 100 Mwt. The mechanical and hydraulic performance of the primary system components has been generally excellent throughout this range of operating conditions.

B. SYSTEM CONFIGURATION

The three parallel loop arrangement of the Fermi primary system has been satisfactory, and it is doubtful that less than three loops would be desirable in a future sodium-cooled power plant. However, although it is not obvious that it would be an improvement, it would probably be worthwhile to consider a larger number of loops, such as four, with respect to questions of piping flexibility, optimization of system layout, and the effect of the inoperability of one loop on plant power factor.

The gravity return flow, nonpressurized system in the Fermi plant has been adequate and has minimized cover gas seal leakage problems. However, this approach to the design of the system also required use of large diameter pipes (30 inches) and intermediate heat exchangers with very low pressure drop characteristics on the primary (shell) side.

The large diameter of the pipes aggravates the piping flexibility and support design problem. The very low pressure drop in the intermediate heat exchangers has probably contributed to flow maldistribution. This may be the cause of the deficient heat transfer performance that these units have demonstrated at the present maximum operation of approximately 25% of rating. One important objective of this approach to the design of the system was to provide for decay heat removal by natural convection circulation after reactor shutdown. However, it was found that the effectiveness of this method of cooling might be marginal under certain conditions; therefore, pony motors with emergency electric power supply systems were added to the pump drive system to ensure adequate circulation for decay heat cooling in the event of emergency loss of power for the main drive motors of the pumps. Although this experience does not indicate that there is an obviously more desirable arrangement that could be used, it would be worthwhile when designing a new plant to study the possible advantages of pressurizing the system enough to allow the cold leg to be raised to a point that would ensure adequate natural convection circulation for emergency cooling under all conditions. Pressurization might also alleviate the need for large diameter pipes and a low primary side pressure drop in the intermediate heat exchangers. Location of the pumps in the hot leg, rather than in the cold leg as in the Fermi plant, or the use of a separate emergency cooling loop should also be considered as means of relieving these problems; such consideration should be carefully reviewed to assure that more formidable problems are not created. The use of a vented fuel may preclude the use of a pressurized system because of difficulty in designing seals against fission gas release.

C. SECONDARY CONTAINMENT

If a leak occurred in the reactor vessel or in that portion of the primary system located within the primary shield tank, it might be necessary to pump additional sodium from the reserve sodium supply into the system to keep the sodium level sufficiently high to continue system circulation. It was not possible to make the void volume within the primary shield tank small enough to avoid this situation. It is recommended that in future designs the primary shield should not be enclosed by the secondary containment if such enclosure creates the situation described.

The secondary containment system outside the primary shield tank is open to the below-floor nitrogen atmosphere at the intermediate heat exchanger and pump component tanks. With this arrangement it is not easy to seal the secondary containment system for purposes of leak testing, and it is especially difficult to recheck the system after the plant is in operation. In future designs, consideration should be given to studying the advantages versus the costs of completely sealing the secondary containment system.

D. SYSTEM SUPPORT AND FLEXIBILITY

The support legs for the Fermi reactor vessel are exposed to varying radiation heating and other thermal effects and, therefore, require constant careful monitoring of temperature conditions and vessel movement to avoid excessive stress conditions. Exposure of reactor vessel supports to such conditions should be avoided in future designs.

In each loop of the Fermi primary system, the IHX component tank stands close to the pump component tank and is connected to it by a large diameter (30 inch) stub pipe. This condition requires monitoring of temperatures of the tank supports to ensure that the tank nozzles are not overstressed due to differential vertical movement. This condition should be avoided if possible in future designs; instead, adequate flexibility should be provided.

Primary system pipe hangers should be made accessible for adjustment when required for unusual conditions such as draining the system. Lack of such accessibility in some cases has been an inconvenience in maintenance operations in the Fermi plant.

E. ACCESSIBILITY AND MAINTENANCE FEATURES

The component tank arrangement was designed to permit removal of components without draining the primary sodium, without cutting sodium pipe connections, or without having to go into the below-floor area which could be inaccessible for maintenance personnel due to the radioactivity level and the inert gas atmosphere. To date, the pumps were removed for replacement of the check valves, and the 6-inch throttle valve stems were removed for repair. As yet, the IHX bundles have not been removed because their lower performance has not affected the Fermi plant since the present plans do not require the full design capability of the units. However, the need for removal might be anticipated in future plants in view of the reduced heat transfer performance that might be encountered in such units.

The value of the removability principle has been demonstrated in the 8 years of Fermi operation, and it is recommended that this concept be considered in the design of future plants. It is further recommended that check valves be contained in a component tank that is separate from the pump tank, so that the pumps would not have to be pulled if maintenance was needed on the check valves.

An additional advantage of component tanks is that they provide points of entry into the system for maintenance purposes. An example of an unsatisfactory arrangement is the spider-mounted guide bushing in the throttle valves which prevents access to the system from this point. It is recommended that accessibility for internal inspection of all parts of the system be provided. The design should also facilitate, as much as possible, the repairability and replaceability of essentially any portion of the system. Repair of some portions of the Fermi primary system could be very difficult, especially those sections located within the primary shield tank. This should be an important consideration in the design of shielding, secondary containment, and other pertinent features of the system in future plants.

A sodium system should contain provisions for trapping and removing foreign articles from any portion of the system. Consideration should be given to locating traps that can be inspected and cleared of foreign material at several points in the system to protect major components, e.g., in the inlet lines to the intermediate heat exchangers, the pumps, the reactor vessel. The design should also provide means for draining the system with minimum hazard and danger of freeze-up during the process. The Fermi primary system design is not entirely satisfactory with respect to provisions for draining the system or for trapping and removing foreign material.

F. INSTRUMENTATION AND HEATERS

The electromagnetic flowmeters in the Fermi primary system were installed without prior calibration. The flowmeters were calibrated after installation by using pump flow characteristics; subsequently, the calibration was checked using data from system heat balance tests. Flowmeters should be calibrated before installation in future plants; furthermore, the systems should be designed to facilitate accurate recalibration at any time after installation.

In many cases, thermocouples and electric heaters in the Fermi primary system are essentially unreplaceable by any reasonable method after the system becomes radioactive. Thermocouples and resistance heaters have demonstrated a significant rate of failure (frequently due to mechanical causes) and should be either accessible for removal and replacement by direct contact methods or should be designed for remote removal and replacement.

Induction heating is used in many portions of the Fermi primary system and is operated at half the rated voltage to improve its reliability; however, some failures have occurred. One unforeseen consequence of use of this type of heating has been the increase in the heat load on the ventilating system for the under-floor nitrogen atmosphere because of stray heating of nearby structures and equipment. This effect should be anticipated in the design of future plants where induction heating is used.

In general, both resistance and induction heaters have performed satisfactorily.

G. PRIMARY SODIUM SERVICE SYSTEM

The reactor overflow line is the source of primary system sodium that is recirculated to the primary sodium service system. This overflow sodium is drawn from the overflow tank as needed for purification and checking in the service system components, such as the cold trap or plugging meter. It is then returned to the main flow stream of the primary system at the No. 1 intermediate heat exchanger tank in a condition satisfactory for use in a reactor power plant. The overflow connection on the reactor vessel was inadequate for sodium cleanup following outgassing of the shield plug graphite. This was due to the nozzle being made perpendicular to the conical wall of the reactor vessel (Figures 24 and 25); the upper surface of the nozzle thereby provided an inverted dam against surface-floating crud being discharged to the overflow tank.

The basic arrangement of the system, however, has apparently been adequate for operation of the Fermi plant; but it is recommended that in future designs the service system sodium should be drawn directly from the main flow stream to provide greater assurance of accurate sampling and fast response in monitoring the condition of the sodium.

The present plugging meter installation in the Fermi plant operates on a discontinuous cycle and, therefore, provides only intermittent data on plugging temperature. Replacement of this equipment with a continuous plugging meter is under consideration. Use of a continuous plugging meter is desirable to achieve continuous response and clear display of plugging temperature trends, and to minimize dependence on interpretation of the readout data by operating personnel. Some further developmental work must be done before a continuous readout plugging temperature meter can be calibrated with sufficient accuracy and reliability to be completely satisfactory for use in a reactor power plant.

H. INERT GAS SYSTEM

The argon-filled void spaces of the primary sodium components are interconnected by a pressure-equalizing network of piping to form a primary cover gas system. A feed-and-bleed recirculating system maintains a set pressure of 4 inches WC above barometric (see Fig. 45). This recirculating gas system has operated quite satisfactorily in providing surge capacity of cover gas for reactor transients. Recent estimates of reactor transients have indicated that this facility would not be recommended as currently designed for future plants, but that the present capacity of the Fermi clean argon supply and waste gas disposal will meet transient demands.

A suggested modification to the existing facility to reduce maintenance problems with the compressor would be to eliminate the recirculating gas compressors and vapor traps, to use the vacuum tank as a waste gas collection header, and to use the hold-up tank to provide surge capacity for the clean argon feed line.

I. SYSTEM CONTROL

The concept of the multicircuit shutdown mode of operation was based, in part, on consideration of a design feature of the primary sodium pumps installed in the primary system. Operation of the upper bearing of the pump is dependent on sodium lubrication which, in turn, is dependent on a minimum sodium level in the pump tank. The Fermi primary sodium design is such that should two primary loops shut down, the sodium levels in the primary system will unbalance, lowering the sodium level to the point where the upper bearing in the third primary pump will not be properly lubricated. Thus, it is essential that when two of the primary pumps shut down, the remaining pump is shut down in a reasonable amount of time to avoid possible damage to the bearing.

Because of this situation, the Fermi control system is currently set up such that if two of the sodium system loops shut down for any reason, the third loop is immediately shut down also. This will then lead to reactor scram when the reactor control system senses either loss of flow or negative dn/dt due to the negative temperature coefficient. Failure of the reactor control system to promptly respond to this condition could cause a serious temperature overshoot. Therefore, it is currently planned to modify the control system interlocks so that reactor scram will be initiated by shutdown of the second heat transport loop and shutdown of the third loop will not occur until reactor power has decayed to 2 Mwt or less. It is expected that under this revised system, the primary pump in the third loop will generally be shut down soon enough to avoid damage to the bearings. Future systems should be designed to minimize such interactions of mechanical conditions with control system functions and to eliminate the requirement to shut down any operable loop because of a pump bearing lubrication problem.

REFERENCES

1. "Study of Materials and Power Producing Nuclear Reactors," Attachments A, B, and C, Report to the United States Atomic Energy Commission by the Dow Chemical Company and The Detroit Edison Company, December, 1951.

2. "Information Report to the Project Companies of the Dow Chemical-Detroit Edison and Associates, Nuclear Power Development Project," Dow Chemical-Detroit Edison and Associates, TID-10077, December 1, 1953.

3. "Information Report by the Dow Chemical-Detroit Edison and Associates Atomic Power Development Project," Dow Chemical-Detroit Edison and Associates, DCDE-101, September 1, 1954.

4. Everett, J. L. and Jones, R. H. "Introduction to Power Plant and Test Facility," APDA-103, January 28, 1955.

5. "Unclassified Description of Proposed Developmental Fast Neutron Breeder Reactor," APDA-104, April 8, 1955.

6. "Description of Developmental Fast Breeder Power Reactor Plant," APDA-108, September 1, 1955.

7. Jackson, C. B., ed., Liquid Metals Handbook, Sodium-NaK Supplement, AEC Department of the Navy, 3rd Edition, July, 1955.

8. "Enrico Fermi Fast Breeder Reactor Plant," APDA-115, November 1, 1956.

9. "Enrico Fermi Atomic Power Plant," APDA-124, January, 1959.

10. "Technical Information and Hazards Summary Report, Enrico Fermi Atomic Power Plant," Power Reactor Development Company, Vols 1 through 9, AEC Docket No. F-16, 1964.

11. Yevick, J. G., ed., Fast Reactor Technology: Plant Design, The M.I.T. Press, Cambridge, Mass., 1966.

12. Costello, R. H., et al, "APDA Reactor Components Test," APDA-147, November, 1962.

13. Madsen, E. F., "Sodium Sampling in the Enrico Fermi Atomic Power Plant," APDA-306, to be published.

APPENDIX: MISCELLANEOUS STUDIES ASSOCIATED WITH THE PRIMARY SYSTEM

1. CYCLE STUDIES

The concept of a power plant using a nuclear reactor and a liquid metal as a heat source required cycle studies of the performance of the primary and secondary systems. The objective of the studies was to obtain the highest thermal efficiency possible which called for the maximum attainable temperatures. By investigating various sets of steam conditions for the steam generator outlet, the optimum reactor outlet temperature and the flows of the primary and secondary systems could be optimized. The cycle studies assisted in management decisions regarding plant size, turbine-generator size, plant efficiencies, etc.

On the reactor end, the differential between the core inlet and outlet was a limiting factor. On the steam end, the steam temperature had to be approximately 50 F lower than the secondary sodium temperature. A minimum of 40 F was used for the pinch-point* to prevent instabilities in the steam generator.

In the early studies, the core and axial blanket subassemblies were approximately one-third the length of the final design of the subassemblies; this allowed relatively high core ΔT's. For example, the final 500 Mwt design[1] showed steam conditions of 800 F and 1200 psia, with core inlet and outlet temperatures of 500 F and 950 F, respectively. When the power was reduced to 300 Mwt and the subassembly length was increased threefold, the core inlet and outlet temperatures were reduced to 800 F and 550 F, respectively. Steam conditions for the 300 Mwt plant design were reduced to 730 F and 600 psia.

2. STEAM GENERATOR STUDIES**

Experience on liquid-metal heated steam generators was limited when the Fermi Project was started. Extrapolating the design criteria on small-scale units to the dimensions required for the Fermi application indicated certain problems were involved that had a direct effect on the design temperatures and control of the primary system.

---

* The pinch-point is the temperature differential between the water saturation temperature and sodium temperature in the steam generator where the vaporization zone started.

** Anderson, R. H., Lindsey, P. S., and Ford, J. F., "Design, Fabrication, and Preliminary Operation of the Steam Generators for the Enrico Fermi Atomic Power Plant," APDA-307, to be published.

To assess the scope of these problems, a 1-Mwt, U-type steam generator was designed, fabricated, and tested in cooperation with The Babcock & Wilcox Company. Heat transfer performance was satisfactory when 800 F, 1200 psi steam was produced. However, stratification and bypass flow problems were encountered.

Another test was conducted with a bayonet tube steam generator to determine its operating characteristics. Results were not conclusive, and the testing program was terminated before all operating parameters could be investigated. Additional details can be found in referenced Steam Generator Report.

3. SIMULATOR PROGRAMS

Two simulator programs were conducted. One was conducted by Holley Carburetor Co., who developed an analog simulation of the entire reactor system, including the heat cycle and methods of operational control. The other program, conducted by Bendix Aviation Co., was directed toward a more precise simulation of the reactor core; however, the primary system was sufficiently represented to simulate situations relevant to the reactor safety system.

4. TRANSIENT STUDIES

Thermal Transients - Thermal transients in the primary system were considered even in the earliest conceptual design phases. In 1958, Franklin Institute was assigned to investigate the effects of the power plant system transients on the mechanical design of the reactor and its primary and secondary coolant loops.

In establishing the scope of the thermal transient study, it was determined that there were 13 transient-causing conditions that would have to be evaluated:

- Reactor scram with all pumps running (core temperature drop)
- Reactor scram with all pumps shut down simultaneously
- Reactor scram with two loops down, one loop continues operating
- Reactor scram due to control rod out until 1000 F with two loops down; one loop slowly to 60%

- Loss of power to one pair of coolant pumps and one loop
- Loss of power to one pair of coolant pumps with primary loop check valve failure
- Loss of all sodium pumps accompanied by simultaneous reactor scram
- Overspeed of boiler feedwater pumps to 150% rated capacity
- Overspeed of boiler feedwater pumps to 150% rated capacity with reactor scram
- Loss of feedwater flow with reactor scram; core outlet, 940 F; secondary sodium ΔT to zero in 10 seconds
- Loss of feedwater flow with reactor scram; core outlet, 940 F; secondary sodium ΔT to zero in 60 seconds
- Loss of feedwater flow with simultaneous reactor scram; secondary sodium ΔT to zero in 10 seconds
- Loss of feedwater flow with simultaneous reactor scram; secondary sodium ΔT to zero in 60 seconds

As a result of the Franklin Institute's study of the thermal transients, design changes and modifications were made in these transient conditions as the study progressed. The information was used in the specifications for primary system components (refer to Section IV).

Mixing Tests in Upper Sodium Pool of Reactor - Experiments were conducted by Franklin Institute on a quarter-scale model using the distribution of an electrolytic solution of sodium hydroxide as a means of determining the mixing characteristics of the upper sodium pool. Approximately 60% to 70% mixing was obtained without the use of vortex vanes. With a 70% mixing factor, the transients can be reduced at and beyond the outlet nozzle to less than 15 F/sec.

Sodium Level and Gas Pressure Variations Induced by Flow Transients - A study was conducted using the analog computer at Franklin Institute to simulate pump start-up and shut-down transients and pump response to forced oscillations. A large number of cases and variables were studied in which many assumptions were made to simplify the system geometry, including circuit resistances and system capacities. Equipment characteristics information included head and torque characteristics of the pumps, gas volumes, and performance of the gas equalizer lines.

The recorded data represented about 1250 curves. Certain cases were selected to obtain the following information:

- The NaK dip seal in the reactor rotating plug could be blown out if the pump speeds were increased at a certain rate of if the rotor of the motor was short circuited. However, the dip seal could not be drawn in because the gas space was common to the overflow tank by the 6-inch overflow line.

- In the IHX, level transients caused by an extreme condition of pump shut down from high speed could raise the sodium level to the tube sheet and flood the equalizer lines.

- During a reactor cover gas pressure increase, gulping will not occur in the pumps because of the small size of the equalizer lines and the large volume of gas in the pump tank during high flow conditions.

As a result of these studies, it was decided that the NaK dip seal would not be used for the reactor rotating plug and that a backup mechanical seal would be more effective. In addition, a baffle ring was added to the dry dip seal well. Another design change attributed to this study was the extension of the downcomer shield on the IHX to meet the bottom of the tube sheet.

5. DECAY HEAT STUDIES

An extensive study of the fission product decay heat resulting from fast neutron fissioning of uranium was made in 1958. The results of this study are shown in Figure A.1 for the central core subassemblies for the 300-Mwt and 430-Mwt conditions. A similar curve was prepared for the outer row core subassemblies, This study was the basis for specifying the reference design decay heat value per core subassembly of 8 kw at the time of removal from the reactor into the cask car. The evaluation was based on the initial core loading at 430 Mwt, 6% burnup, and 2 weeks decay.

The following scheme was used in deriving the recommended values:

Decay Time	Decay Heating Correlation
0 to 10 seconds	Assume a constant value of 0.065 watt of total decay power per watt of prior operating power
10 seconds to 1 hour	Use twice Way-Wigner
1 hour to 150 days	Dillon and Burris, ANL-5742

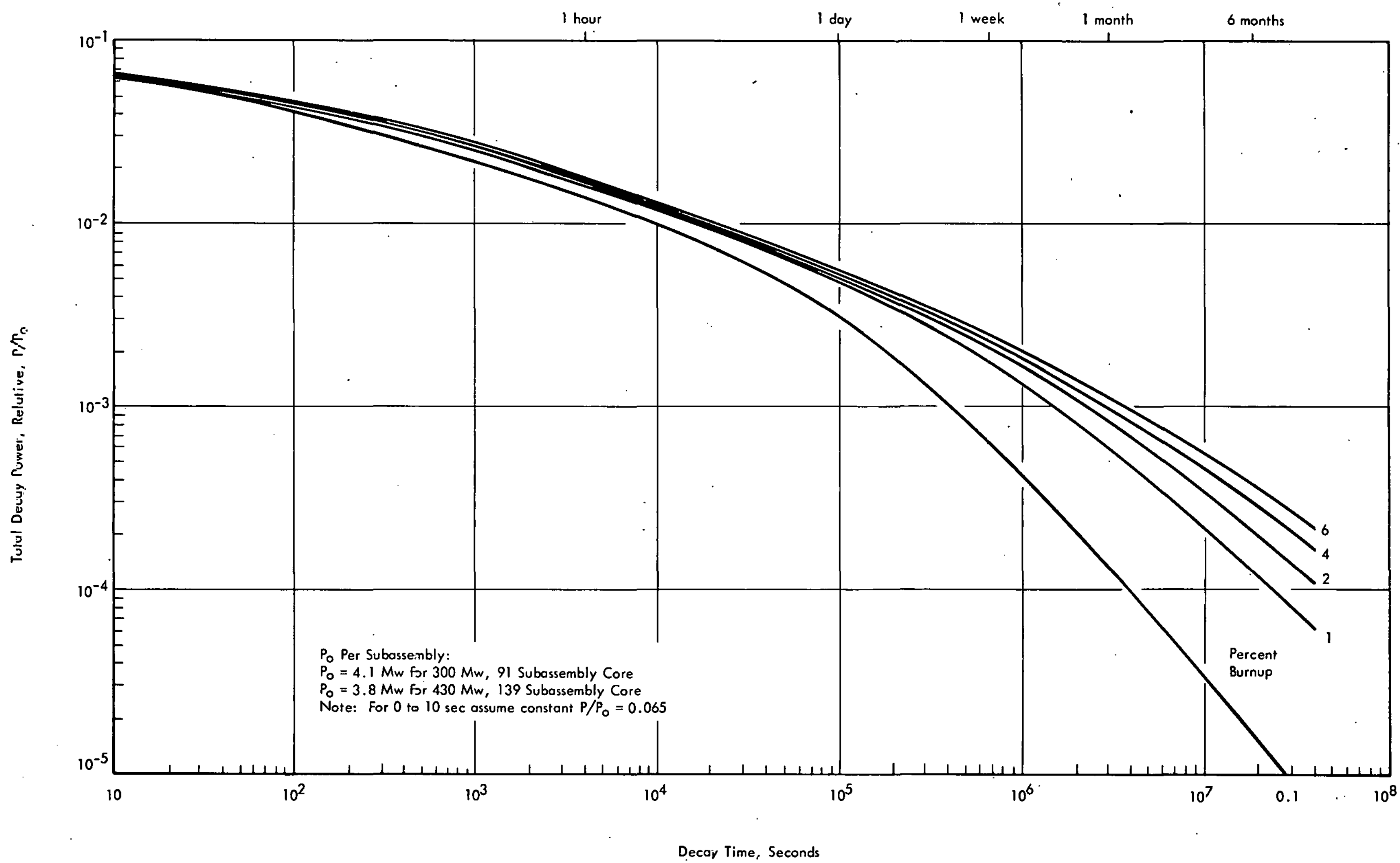


FIG. A.1 DESIGN VALUES FOR THE DECAY HEATING FOR CENTRAL CORE SUBASSEMBLIES

150 days to 300 days	Dillon and Burris, ANL-5742, with an appropriate factor to take into account the fact that the empirical curves fit the calculated data to ± 50%.

With the revised core design for operation at 200 Mwt, the decay heat values were revised in 1961 and are shown in Figure A.2.

In 1964, generalized decay heat curves were prepared, Figure A.3, that could be used to obtain the decay heat of both core and blanket subassemblies for any burnup, reactor power level, and decay time.

6. EMERGENCY COOLING

Early considerations included plans for a separate emergency cooling loop in the primary system. This loop included a heat exchanger capable of removing the decay heat of the reactor when primary pump power was lost. The loop had no valves and the secondary coolant (NaK) was directed outside the building to an air cooler. One of the disadvantages of this loop was that it could not be isolated from the system during normal operation and represented a significant heat loss from the system. It was felt that the same function could be achieved by an arrangement of the primary loops to achieve natural circulation.

The center of the core, the center of the IHX bundle, and the center of the steam generator were cascaded in elevation to achieve the natural circulation flow required. To obtain the low pressure drop in the primary loops during natural circulation, the decision was made to adopt a 30-inch pipe size instead of 24 inch, a hanging disc-type check valve, and a low pressure drop in the IHX.

7. NATURAL CIRCULATION STUDIES

The first calculations were conducted when the primary loops were of horizontal configuration. A reference flow of 1.5% of the maximum operating flow was used for emergency cooling conditions. The pressure drops under these conditions were calculated as follows:

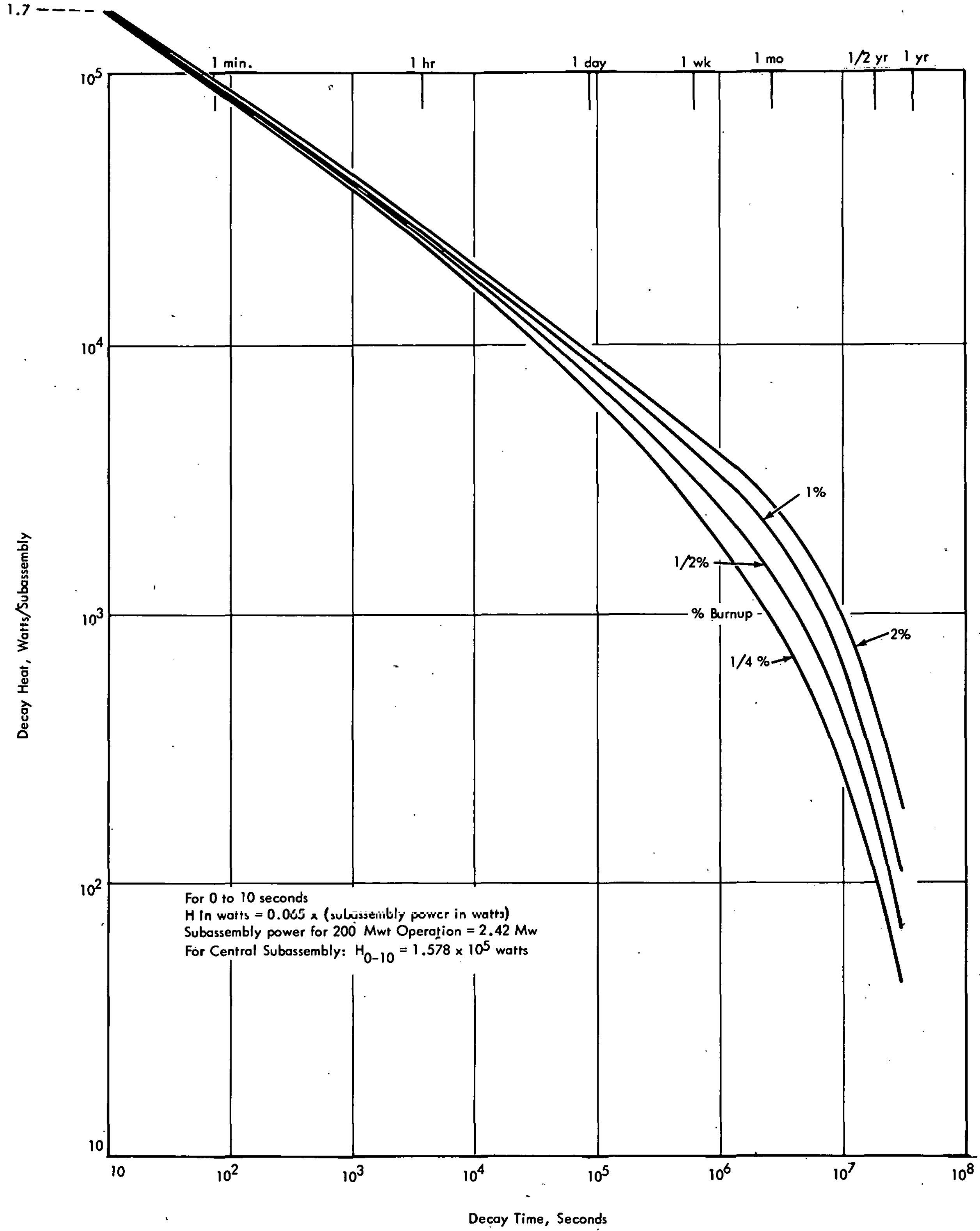


FIG. A.2 RECOMMENDED DESIGN VALUES FOR DECAY HEATING FOR CENTRAL CORE SUBASSEMBLY IN CORE A

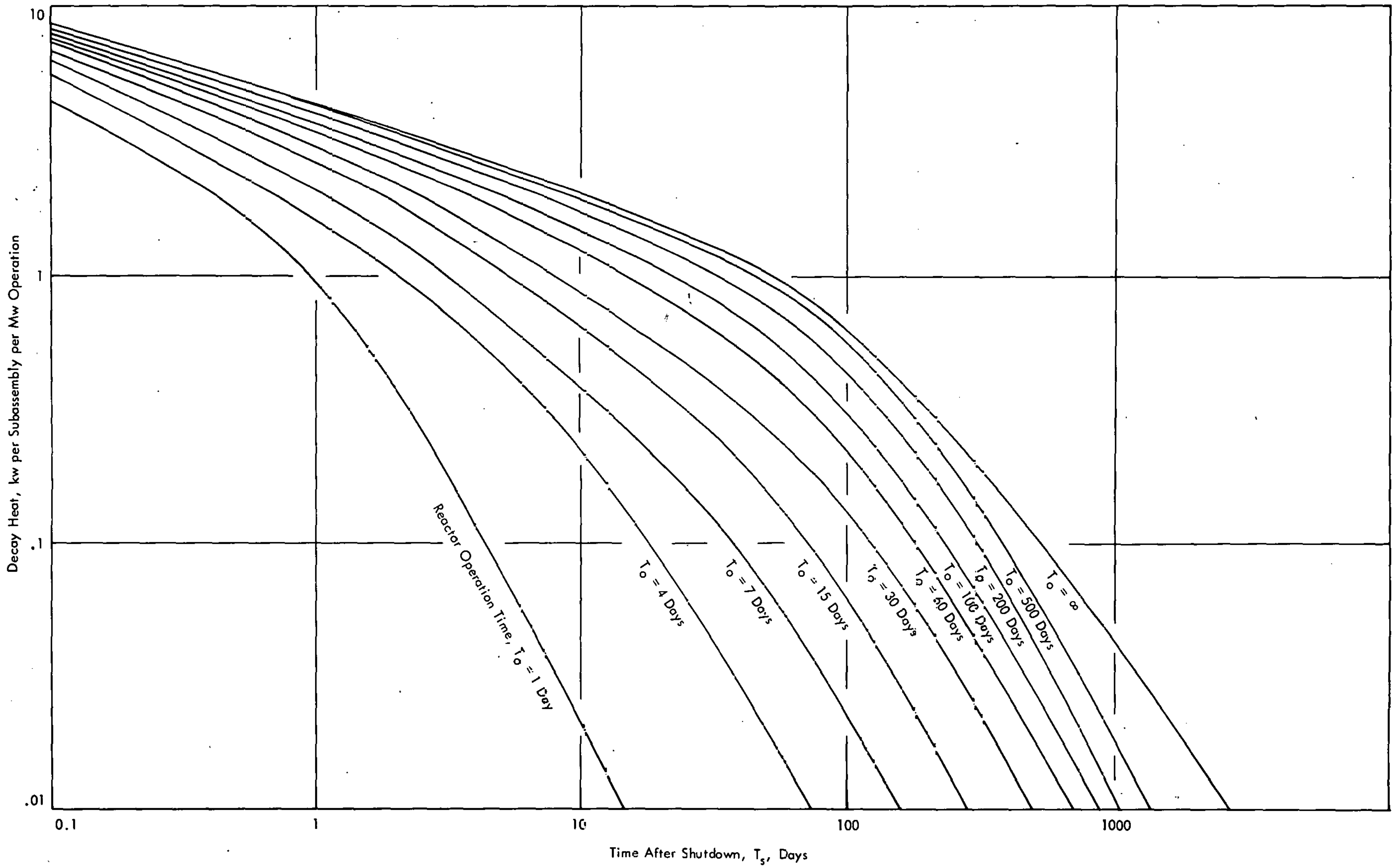


FIG. A.3 DECAY HEAT PER SUBASSEMBLY PER MW OPERATION

TABLE A.1 - NATURAL CIRCULATION PRESSURE DROPS

	Pressure Drop, Feet	
Reactor to IHX	0.000211	
IHX to Pump	0.000513	
Through Stalled Pump	0.039000	
Through Check Valve	0.004900	
Pump to Blanket Tee	0.001076	
Tee to Reactor	0.003740	
Total External		0.04944
Total Reactor Drop		0.0800
Total System		0.12944

These calculated values were used in the check valve specification where an allowable pressure drop of 0.0025 psi was specified for a flow of 72,700 lb/hr.

When more information became available on pressure drops for the reactor vessel, holddown mechanism, handling lugs, and the core subassemblies based on hydraulic tests conducted at the University of Michigan, the calculations were repeated. In this study, 2% was used instead of 1.5% for the reference flow.

A more intensive study was conducted in 1957, with the main interest directed toward the reactor outlet temperature after complete electrical failure of the power lines feeding all pumps followed by a reactor scram. This information was of significance because the primary system design temperature was established at 1000 F. This study revealed that during the first 10 seconds the temperature decreases to about 660 F and then starts to increase; at about 45 seconds, a maximum is reached that is no higher than 1000 F if the pumps do not stall before slowing down to 50 rpm. The natural circulation computations included establishing equations for the natural circulation in the primary and secondary loops as a function of time. The results showed that the influence of natural circulation became appreciable (10%) after 20 seconds.

The capability of natural circulation to remove decay heat was rechecked in a 1959 calculation. It was concluded that boiling would occur in the hottest channel and that there was a probability that fuel damage would occur. The decision was made to provide pony motors on two of the primary and secondary loops and a later decision resulted in provision of pony motors on all three loops.

8. CAPABILITY OF PONY MOTOR DESIGN

The pony motors were designed to operate at 70 rpm on both the primary and secondary pumps; the following factors were considered in the design:

- Decay Power - The decay of reactor power due both to fission products decay and neutron decay were included.
- Decay Flow - This information was taken from pump coastdown information obtained during Reactor Components Test Facility operation.
- Characteristics of Pump Operation With Pony Motor - This information was calculated using data from tests performed at 46.6, 70, and 90 rpm on pump 3. These data showed good agreement with the curve calculated by application of the affinity laws.
- Natural Circulation of Primary System - It was assumed that differential circulation was occurring in the core area as well as the overall loop.
- Check Valve Closing Backflow - The closing backflow requirement of the check valve was taken into account.
- Maximum Temperatures - The power distribution and hot channel factors were taken into account for both 200-Mwt and 430-Mwt conditions. It was determined that a temperature of 1100 F would be reached 26 seconds after a 200 Mwt scram if only one of the three pumps were operating.

9. DECAY HEAT TEST, 1964

The object of this test was to determine whether setting losses in the primary system were sufficient to dissipate the decay heat.

It was postulated that all three loops of the secondary cooling system were disabled and severed. It was further hypothesized that the below-floor air system was inoperative and that power for the primary pump and pony motor was not available. Thus, without circulation of sodium and air, the system temperature would rise to an equilibrium point where losses matched decay heat input.

This test was proposed as a simulation of the decay heat generation (scaled down by 0.5) with observations of the time-temperature behavior of all primary system components. Results would be scaled up for comparison. Since reactor operation was not feasible with the above constraints, decay heat was simulated by equivalent energy inputs from the primary pumps and from vessel resistance heaters and piping induction heaters. Heat was reduced in accordance with the decay heat curves.

The data obtained from this test resulted in a more accurate estimate of final conditions. The findings of this test were that the primary sodium system temperature peaked out at 550 F (from 734 F) in 28 hours. Although the below-floor ventilation system blowers had to be turned on after 18 hours because of concrete temperatures at the peak temperatures, the heat rejection was 380 kw or 1,300,000 Btu/hr.

10. SETTING LOSS TESTS, 1966

During the heat balance tests, it was calculated that all three loops of both primary and secondary systems at 530 F had a setting loss of 500 kw. The test was continued with one of the secondary loops isolated and the setting loss was 392 kw, a difference of 108 kw. It was therefore assumed that each secondary loop system has 108-kw setting loss; therefore, the remainder, or 176 kw, represents the primary system setting loss. It was estimated that the below-floor heat removal system was removing an additional 180 kw from the induction heating system, giving a cooling load of 356 kw or 1,210,000 Btu/hr.

11. STRESS ANALYSIS OF PRIMARY SYSTEM

Stress calculations were made of the various primary system arrangements to determine piping stresses and bending and torsional movements on the component nozzles. In all cases, the piping stresses ranged between 12,000 and 13,000 psi, whereas the allowable stress range was 25,500 psi. Therefore, the criterion in evaluating all piping arrangements was the moment effects rather than piping stress.

The piping arrangement with horizontal expansion loops indicated that too much space would be required in the plant layout to accomodate the lengths that were necessary. The decision was made to forego drainability and use the vertical loop arrangement. Vertical loops blended well with the primary system concept of component tanks since they would be of the same general configuration. It was also ideally suited to the penetration of the secondary shield wall where the shielding requirements precluded direct penetrations.

The following allowable combined bending moments and torsional moments for the 30-inch, 14-inch, and 6-inch connections to the reactor vessel were used as a baseline:

Connection, diameter in inches	Bending Moment, ft-lb	Torsional Moment, ft-lb
30	50,000	80,000
14	10,000	7,500
6	2,000	1,000

The vertical loop length was selected both on the basis of nozzle moments and on the floor levels in the building. Both the M. W. Kellogg Co. and Mr. S. W. Spielvogel conducted stress analyses on the final system. Although large-diameter, thin-walled piping was in common usage, the temperature of 900 F and 1000 F was an unusual service for this type of piping. The normal elbow flexibility was adjusted to be a significant contributor to system flexibility due to ovalation at the midplane. A maximum moment of 57,700 ft-lb was calculated for the cold erected condition that would reduce to 19,200 ft-lb in the hot condition with 100% cold spring. The deflections at each point were calculated, and this information was used in specifying gaps and clearances for erection of secondary containment and the shielding column.

Calculations were also made of the U-type gas equalizer lines between the pump and IHX tanks and it was determined that two additional out-of-plane bends were needed for adequate flexibility. The arrangement of the 6-inch overflow line was also revised when it was found that inadequate clearance was available in the shield wall penetration. This condition was remedied by vertically anchoring the piping inside the shield wall.

The skirts on the pump and IHX tanks were carefully insulated to attain essentially equal thermal growth under all conditions. With uneven growth, severe moments could be applied to the tank nozzles. To monitor the thermal behavior, thermocouples were attached to all skirts and differential movements were calculated under all conditions. The allowable differential has not been exceeded.

12. STUDIES OF PIPING AND CONTAINMENT SUPPORT

Combined Support Study - Before secondary containment was a part of the primary system design, the vertical piping loops were to be supported at a midway point on the risers. With secondary containment added in close proximity to the piping, the support studies were magnified to include a similar support system for secondary containment. Figure A.4 shows that the hanger load changed by a factor of three for piping hangers and containment hangers between the filled and the empty condition. Such hanger performance was not practical. Therefore, it was decided that support of both piping and containment could be achieved with a single hanger as shown in Figure A.5. By this method, the change in load between empty and filled containment was approximately 10% and the nozzle movement effect was minimized. This arrangement was selected for the final design. Automatic load-sensing

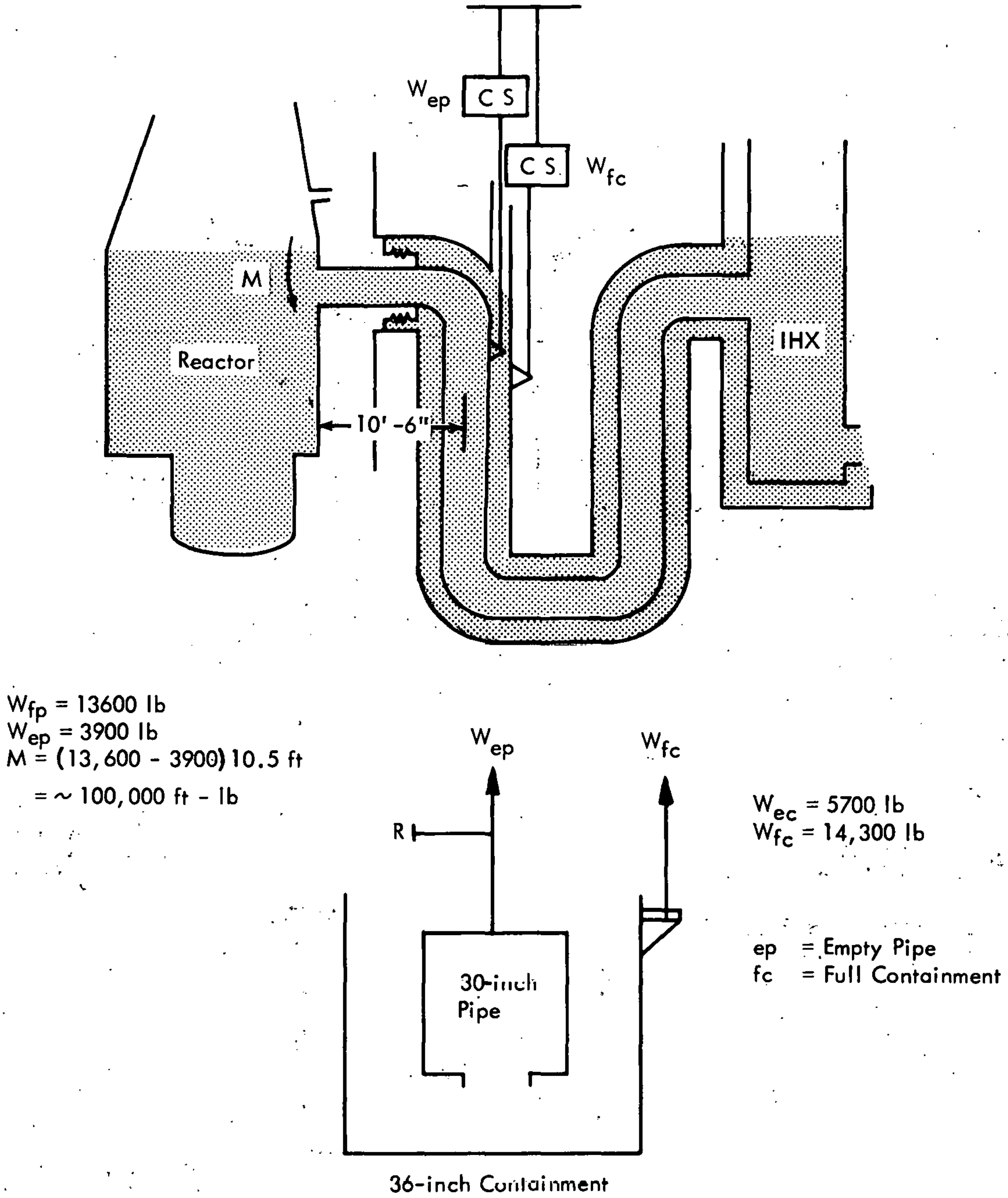


FIG. A.4 PIPING AND CONTAINMENT SUPPORT

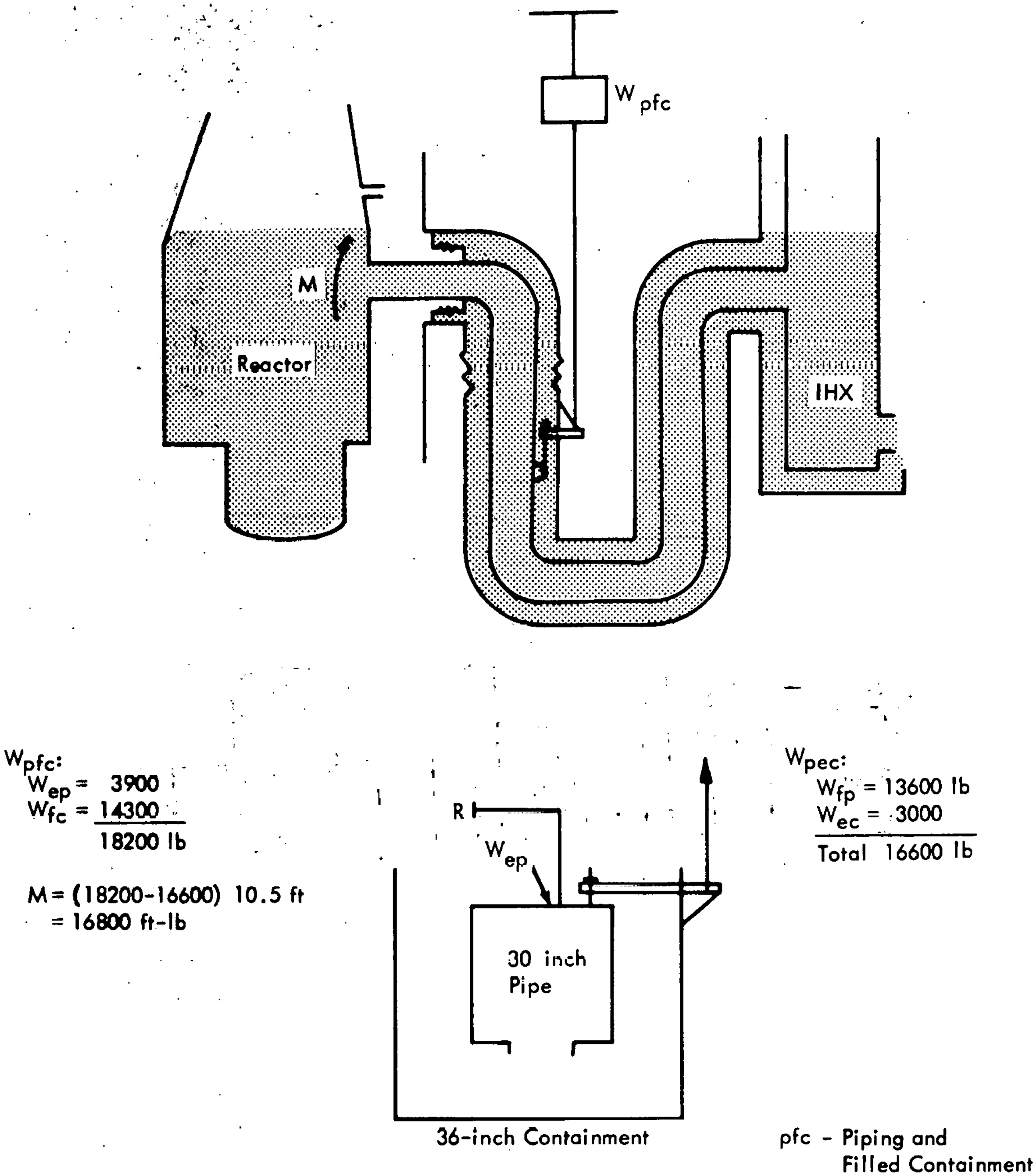


FIG. A.5 PIPING AND CONTAINMENT SUPPORT

hanger equipment was investigated as an alternate arrangement; however, it was determined that high values of moments would be imposed on equipment nozzles before the automatic load computing devices could be energized.

Hanger Setting During Sodium Fill - Studies indicated that the dry load for the 30-inch pipe hangers was 9600 lb, whereas the filled load was 17,600 lb since the hanger could not be adjusted during a drain or fill operation.

During the fill operation, there were two alternatives for hanger settings. The first alternative was to set the hanger loading at dry load conditions thus imposing at the time of fill a downward moment on the reactor nozzle of (8000 lb at 10 ft) 80,000 ft-lb. The second alternative was to set the hanger loading for the filled load, thus resulting in the same high moment being applied in an upward load direction before the fill operation. Both alternatives were discarded and a compromise plan was adopted. The hangers would be set at a load midway between the dry and the filled loads. Stress calculations were performed to determine the moments that would be imposed on equipment nozzles both at the dry cold preloaded condition and after being filled. It was found that half loading in the dry condition gave a moment of 53,000 ft-lb and the filled condition gave 25,000 ft-lb. It was therefore decided to equalize the two conditions by decreasing the initial load value so that the final moment and the initial moment were both approximately 39,000 ft-lb.

13. SYSTEM PRESSURE DROP STUDIES

With the decision to use gravity flow from the reactor to the pump tank, the pressure drop studies consisted of two separate systems: the gravity flow and the pump discharge flow. The levels in the IHX and the pump tanks were dependent upon the pressure drop between the two. These pressure drops were kept to a minimum for natural circulation purposes, minimum variation in heat transfer surface as a function of flow, and avoidance of a low level in the pump tank where gas gulping could occur. The pump discharge head had to accommodate the piping system pressure drop but more particularly the core pressure drop, which changed frequently with the development of the subassembly design.

The early gravity flow calculations predicted the following drop in levels after a start with all tanks at the reactor level datum:

	300 Mwt	430 Mwt
IHX Level Drop, ft	1.0	1.5
Pump Level Drop, ft	3.0	4.4

The pump discharge piping pressure drop calculation used to procure the primary pump was stated as follows:

	Pressure Drop, psi
Core	50
Axial Blanket	14
Vessel	10
External System	15
Static Head	3
Total Primary Loop Pressure Drop	92

To establish the specification for the primary pump, a reserve of 20% head capacity was added to the above requirement, resulting in 110 psi (310 feet at 1000 F) for the design value. The final design specification for the pump was for 110% flow at 110 psi or 4, 840, 000 lb/hr/loop.

Revisions were made in the design of the fuel subassembly in 1957, and it was calculated that the core subassembly would have a pressure drop of 91.4 psi. Based on this value the system pressure drops were calculated and are shown in Table A. 2.

TABLE A. 2 - SYSTEM PRESSURE DROPS
March, 1957

	300 Mwt	430 Mwt
Fuel Subassemblies, psi	91.4	56.8
Handling Lug, psi	4	5.8
Holddown, psi	3	4.35
Upper and Lower Sodium Pool, psi	3.7	5.35
Vessel to Pump, psi	1.1	1.6
Pump to Vessel, psi	10.4	15.1
Total, psi	113.6	89
feet	296	293

This increased core pressure drop was also shown in Reference 9 which listed system pressure drops as follows:

Component Head Loss at Design Flow, Check Valve and Piping to Tee, ft	14.1
14-inch Piping, 90%, ft	9.5
Reactor Vessel Inlet Plenum to Core, ft	13.5
Core and Axial Blanket, ft	254
Holddown Mechanism, ft	5.5
30-inch Piping, ft	1.0
IHX, including nozzles, ft	1.9
Total, ft	299.5

For planning purposes, the conventional pressure drop constants were used for elbows (0.2), exits (0.5), entrances (1.0 or 1.5), and the friction factor was based on the Reynolds number. The IHX pressure drop ranged from 0.76 feet at 200-Mwt conditions to 2.48 feet at 430-Mwt conditions. In the final calculations, which considered nozzle shapes and enlargements due to shield plates, etc., the following levels were predicted.

	Sodium Flow, lb/hr x 10^6	IHX Level Drop, ft	Pump Level Drop, ft
200 Mwt	2.95	0.45	0.81
300 Mwt	4.4	0.98	1.77
430 Mwt	5.29	1.37	2.72
150% of 300-Mwt Flow	6.6	2.26	4.08
Refueling Flow	1.7	0.14	0.26

In preoperational testing and operation up to 100 Mwt, there was good general agreement between the predicted and actual pump tank level. The performance of the pump discharge piping system is also felt to be in agreement with predicted values, although it can not be checked because there is no pressure detector on the inlet plenum of the reactor

14. CAPABILITY OF HANDLING 150% OF 300-Mwt FLOW

An investigation was conducted to determine whether the components of the primary system could accommodate a flow 50% greater than the reference (300 Mwt) flow of 4.4 x 10^6 lb/hr. This investigation included pressure drop calculations and pump and IHX studies.

Using 6.6 x 10^6 lb/hr, 800 F, and 550 F for primary system temperatures, the level drop between the reactor and IHX and pump was 2 feet 9 inches, and the level drop between the IHX and pump was 2 feet 8 inches, giving a pump tank level of 5 feet 5 inches below reactor level datum. In a more refined calculation conducted in 1961, this pump tank level was calculated to be 4.08 feet. This level was well above the impeller which is located 7 feet 4-3/8 inches below reactor level datum. Pressure drop calculations on the pump discharge piping were not performed.

The pump manufacturer was informally asked for an opinion on the capability of the Fermi pump to handle 15,100 gpm with a discharge head of 226 ft at 600 F; the answer was that a larger impeller and new pump case would be required, but the same tank could be used. Also the horsepower rating of the motor would increase to 1100 hp. If two-loop operation was hypothesized, a larger pump tank would be required.

The IHX manufacturer was asked to supply information on the performance of the unit under 150% flow conditions; Table A.3 lists this information.

TABLE A.3 - IHX PERFORMANCE AT 150% FLOW

	Shell Side	Tube Side
Flow, lb/hr	6.6 x 10^6	6.6 x 10^6
Gravity, liquid, lb/cu ft	52.7	53.7
Specific Heat, Btu/lb/F	.3035	.307
Temperature In, F	1000	530
Temperature Out, F	650	880
Operating Pressure, psi	50	300
Velocity, ft/sec	4.22	5.72
Pressure Drop	4.5 ft Na	7.25 psi
Conductivity, Btu/hr - ft^2 - F	40	41.7
Heat Exchanged, Btu/hr	702,500,000	
MTD, corrected, F	120	
Transfer Rate, Service, Btu/hr	983	
Clean, Btu/hr	1225	
Surface, Effective, ft^2	5960	

It can therefore be concluded that for 150% flow the pressure drop of 4.08 feet is satisfactory, the present pump tanks could be used but the pump internals would have to be modified and a larger motor would be needed, and the IHX has the capability to operate at this level. However, based on low power level performance, this performance (at 150% flow) may not be achieved. (See Table 18, p. 140).

15. PLANT CONTROL STUDIES

A committee on control was formed to achieve coordination between groups concerned with the design of controls for the plant. The basic tenet of this committee was that the requirements and/or limitations of the reactor plant take precedence over steam system considerations. Safety was the paramount criterion.

On of the major decisions of this committee was the determination of whether the sodium flow should be a constant value or should vary with load demand. The decision was made in favor of constant sodium flow (see Section III-G).

Maximum rate of loading of the Fermi plant turbine-generator was another topic reviewed by the committee. It was decided that the control system should be designed to accommodate a 5 Mw/min electrical output step change.

The committee also investigated the reactor safety philosophy pertaining to reactor shut-down rates. There was basic agreement on a progressive shut-down sequence to minimize thermal shock to the reactor.

Another basic decision of the control committee was the allocation of operational authority over the feedwater controls between PRDC and DECo. This was resolved by assigning the major functions to PRDC in keeping with the policy that reactor safety is the ruling consideration.

16. REMOTE MAINTENANCE STUDY

The system with replaceable internals for the primary system components permits use of the building crane for handling the removable internals. It is necessary that these internals be removed by remote control to prevent radiation exposure to the operators. A remote maintenance system was developed incorporating television cameras in the building and a control console in the plant control room.

A coffin or bag-type container is placed over the component, then the component internals are drawn up into the inert gas filled container and sealed. The opening to the component is replugged. The component internals are then transferred by building crane to a decay tank. These items are then allowed to decay for a time sufficient to reduce the radiation to acceptable levels. New component internals are installed into the primary system tanks with the same equipment.

17. CONCRETE POURING HAZARD STUDIES

A question was raised about the potential hazard of pouring concrete in the vicinity of tanks and piping filled with sodium. A failure in the steel forms could cause overstresses and cause a failure in the sodium-containing vessels resulting in a sodium-water reaction. Therefore, an independent check was made of the form supports for both the secondary shield wall and the 5-foot-thick operating floor for adequate strength during pouring. This investigation was completed, and it was concluded that a potential hazard did not exist. A recommendation was also made that concrete pouring be done in finite steps, rather than all in one pour. These recommendations were incorporated into the final pouring procedure, and the operation was completed without any difficulties.

DISTRIBUTION LIST

USAEC - Chicago Operations Office
- Director, Contracts Division (2)
- G. H. Lee

USAEC - Washington, RDT
- Director
- Asst. Director, Program Management
- Asst. Director, Reactor Engineering
- Asst. Director, Reactor Technology
- Asst. Director, Plant Engineering
- Asst. Director, Nuclear Safety
- Project Manager, LMEC
- Project Manager, FFTF
- Program Manager, LMFBR
- Liquid Metal Projects Branch
- Chem & Chem Separations Branch
- Reactor Physics Branch
- Fuels and Materials Branch
- Applications and Facilities Branch
- Components Branch
- Instrumentation and Control Branch
- Systems Engineering Branch
- Core Design Branch
- Fuel Handling Branch
- Special Technology Branch
- Reactor Vessels Branch

USAEC-RDT Site Representatives
- Site Representative, APDA
- Senior Site Representative, ANL
- Senior Site Representative, AI
- Acting Senior Site Representative, IdOO

USAEC-DTIE
- R. L. Shannon (3)

USAEC - New York Operations Office
- J. Dissler

USAEC - San Francisco Operations Office
- J. Holliday

Director, LMFBR Program Office, ANL
- A. Amorosi

Director, LMEC, AI
- R. W. Dickinson

Aerojet - General Corporation
- H. Derow

Argonne National Laboratory
- R. Bane
- L. W. Fromm
- S. Greenberg
- L. J. Koch
- S. Lawroski √ 5 —
- M. Novick
- F. Smith

Atomics International
- R. Balent (2)
- S. Golan

Babcock & Wilcox Company
(Box 1260, Lynchburg, Va 24505)
- M. W. Croft

Babcock & Wilcox Company
(Barberton, Ohio)
- P. B. Probert

Baldwin-Lima-Hamilton Corp.
(Industrial Equipment Div., Eddystone, Pa)
- J. G. Gaydos
- R. A. Tidball

Brookhaven National Laboratory
- O. E. Dwyer
- D. Gurinsky (2)
- K. Hoffman
- C. Klamut
- L. Newman
- A. Romano

Combustion Engineering, Inc.
(Box 500, Windsor, Conn)
- W. P. Staker
- W. H. Zinn

General Electric Company
(175 Curtner, San Jose, Calif 93125)
- K. P. Cohen (3)

General Electric Company
(310 DeGuigne, Sunnyvale, Calif 94086)
- A. Gibson

Gulf General Atomic, Div. of Gulf Oil Co.
(San Diego, California)
- P. Fortescue

M. W. Kellogg Company
(711 Third, New York, New York)
- E. W. Jesser

Lewis Flight Prop. Laboratory, NASA
(21000 Brookpark, Cleveland, Ohio)
- C. A. Barrett

Los Alamos Scientific Laboratory
- D. B. Hall (2)
- G. Waterbury
- W. R. Wykoff

MSA Research Corporation
(Callery, Pa 14024)
- C. H. Staub

Nuclear Materials & Equipment Corp.
(Apollo, Pennsylvania)
Z. M. Shapiro

Oak Ridge National Laboratory
(Box X, Oak Ridge, Tennessee)
F. L. Culler (2)
J. H. Devan
D. Gardiner
J. White

Oak Ridge National Laboratory
(Box Y, Oak Ridge, Tennessee)
R. E. MacPherson, Jr.

Pacific Northwest Laboratory, BMI
E. Astley (5)

Power Reactor Development Company
(1911 First, Detroit, Michigan 48226)
A. S. Griswold

Southwest Atomic Energy Associates
(Box 1106, Shreveport, La 71102)
J. R. Welsh

United Nuclear Corporation
(Box 1583, New Haven, Conn)
A. Strasser (2)

Westinghouse Electric Corporation
(Box 158, Madison, Pa 15663)
J. C. R. Kelly, Jr. (2)

Westinghouse Electric Corporation
(Box 158, Madison, Pa 15663)
C. A. Anderson

USAEC-UKAEA Exchange
UKAEA
Reactor Group Headquarters
Risley, Warrington, Lancashire
England
J. Stephenson (12)

USAEC-EURATOM Exchange
EURATOM
51, Rue Belliard
Brussels 4, Belgium
A. deStordeur (10)

CNEN
Via Mazzini 2
Bologna, Italy
F. Pierantoni (4)

CEN Saclay
Boite Postale 2
Gif-Sur-Yvette (Set 0) France
G. Vendryes (10)

Kernforschungszentrum Karlsruhe
7500 Karlsruhe, Germany
W. Haefele (10)