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FINAL REPORT

300 KWe CAPSULE NUCLEAR POWER PLANT STUDY

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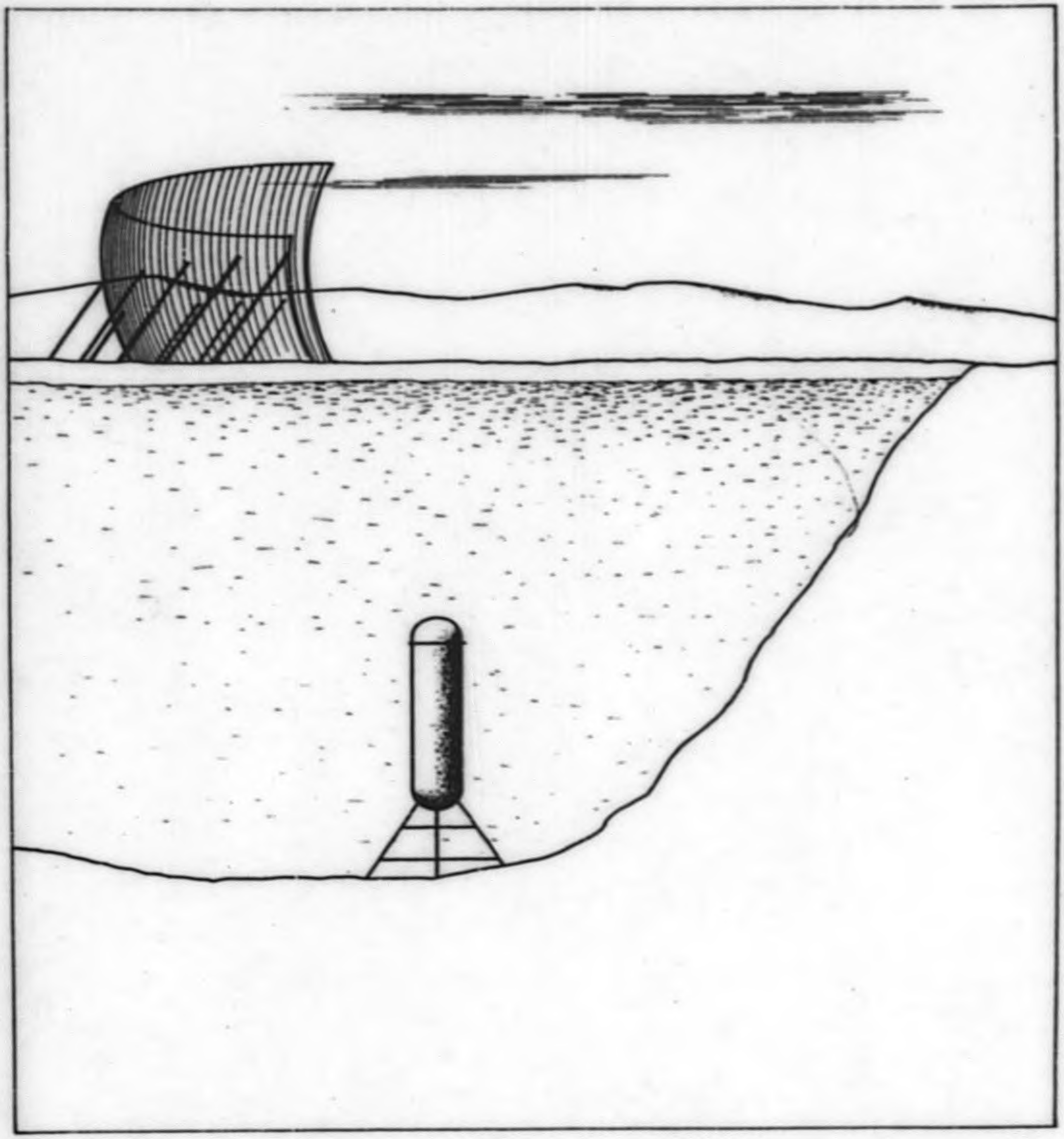


ACKNOWLEDGEMENT

The data presented herein are the result of the combined efforts of many persons in the General Electric Company. The Office of Army Reactors and the San Francisco Operations Office of the U.S. Atomic Energy Commission contributed significantly to the successful completion of this study by their capable guidance and program direction.

TABLE OF CONTENTS

TABLE OF CONTENTS
(Cont'd)

SECTION 1.0

INTRODUCTION

This document presents the results of investigations concerned with the conceptual design of a 300 KWe "Capsule" nuclear power plant.

"Capsule" is an unattended, self-supporting power plant described as an arrangement whereby the nuclear reactor, turbine, generator, feed pump, and condenser are all contained in one vessel, which is still small enough to be handled as a single package. The reactor is a natural circulation, light water reactor operating on its over-all temperature coefficient without the impediment of control rods. Operation is planned to be continuous on an unattended basis at rated power throughout life of the plant.

The scope of this study included the following:

A. Study various plant arrangements which retain the Capsule concept of the unattended sealed nuclear power plant, under the following general guide lines:

   1. 300 KWe is required.

   2. Men are available for periodic plant adjustment.

   3. The core can be removed and replaced.

   4. The plant may be maintained by replacement of major components.

   5. Analyze plant arrangements utilizing one or more capsules although a plant arranged in one capsule is deemed desirable.

   6. Direct and indirect cycle systems are to be considered.

   7. Advanced control concepts may be used.

B. Recommend a plant arrangement from which a conceptual design is to be developed.

C. Determine major physical dimensions and performance capabilities of the power conversion equipment. Investigate the possibility of having all rotating equipment on one shaft.

D. Prepare cost estimates and time schedules as follows:

   1. Cost of engineering development program necessary to achieve an operating prototype.

   2. Cost of fabrication and installation of prototype.

   3. A schedule of the time required for conducting the development program, and the fabrication and installation of an operating prototype.

4. An economic study of equipment, operating, and maintenance costs of a nuclear plant based on the conceptual design. This study will include an economic comparison of the capsule power plant, using cost data prepared under this project agreement, with the cost data made available by the Commission of other small size nuclear power plants and a diesel electric plant.

The General Engineering Laboratory of the General Electric Company collaborated with the Atomic Power Equipment Department in this project by investigating the design of power conversion equipment and plant control systems. A summary of this work is provided herein, and a separate report, 60 GL 144, "Unattended Power Plant" (Reference 1), covers in greater detail the work accomplished on the design of power conversion equipment and plant control systems.

It is important to recognize that the evolution of a new concept requires a much greater breadth of investigation than is involved in producing a preliminary design of a reactor where the type has been fairly well ascertained, and one is concerned with varying the rating, size, or number of loops. While the intent is to use in Capsule only components where engineering feasibility has been proven, the new ground rule of unattended operation makes mandatory the re-investigation of parameters normally considered as fairly well defined. To accomplish the above scope within the time allotted, it was sometimes necessary to restrict the depth of detail in evaluating alternates or to close out a promising alternate in order to pursue the main objective. (An example of this is the two-capsule arrangement which offers advantages and some disadvantages. This can be further investigated if or when the single capsule becomes too big or too heavy.)

This work was performed for the Office of Army Reactors under Contract AT(04-3)-189, Project Agreement No. 19, with the San Francisco Office of the United States Atomic Energy Commission.

SECTION 2.0

BACKGROUND AND DESIGN PHILOSOPHY

2.1 Background

The concept of an unattended nuclear power supply first came into being when man found it necessary to provide power in environments where he could not exist or, at best, could exist for only limited periods of time.

The continuing conquest of space has shown that man can provide power in a new environment on an unattended basis. These systems, however, must still be improved to provide more power for longer periods.

The need for electrical power in the ocean depths is another application to a difficult environment receiving increasing attention. In fact, it was this application that first evolved the "Capsule" nuclear power plant upon which the present study is based (Ref. 2,3).

The Army Reactors Office, with a broad charter to provide for base electrical power needs for all services, recognizes the ultimate need for completely unattended nuclear power supplies in certain areas. While initiation of programs for such a purpose must await the establishment of a firm requirement by a potential user, the Army Reactors Office has recognized that the availability of such a plant would have significant importance in the reduction of the size of the operating force at remote bases where, although men and machines can exist, the cost of existence is high and conditions at best are difficult. At most of these bases, personnel are stationed to perform other functions, and at first glance, the need for a truly unattended power plant does not appear to hold much edge over a partially attended or reduced attendance type of power plant. While such potential gains are of sufficient significance to justify exploratory development, the real gains will come when the military planners can start with an unattended, long life power plant and use it where needed, to perform new functions, to establish unattended bases, to survive attack damage, or to provide a power base for emergency rebuilding of disaster areas.

Since this is not merely a new design but also a new concept of operation, it is imperative that the reviewer understand the design philosophy and not judge by existing reactor standards. For example, Capsule is not presented as an "improvement" over SL-1 or PL-2, but as a complementing concept for those applications where PL-2 requires too much site preparation, transportation, or operating support. The PL-2 type of mission (manned, maintained, logistic supportable) will continue where personnel can be supported. As operating conditions and environments become more difficult for man, the true purpose of the capsule concept comes into greater perspective and usefulness.

2.2 Design Philosophy

Using a plant concept evolved for deep submergence in the ocean, General Electric has studied the feasibility of adapting this design to an application requiring an unattended power generation plant in a remote area where there is a source of water for use as a heat sink. Such a water source could be either a natural lake, the ocean, or an artificial lake created by ice melted in the course of startup and operation of the plant. The only requirement for such a site would be that the terrain would be such as to prevent the runoff of the water, so that the plant would be continuously submerged in the heat sink.

The proposed conceptual design of the unattended power plant does not require research-type investigations to prove feasibility of the over-all design or any reactor or power plant component thereof.

The elimination of the usual maintenance and accessibility requirements will allow a great simplification to be made in the design of the plant, and will also increase the reliability of the over-all concept and the compactness of the plant arrangement. It is within the framework of this new philosophy of operation, maintenance, and application that this study has resulted in a specific design preference and recommendations for proceeding with the development of this concept. The primary design benefits of this concept are:

A. A single type of fluid can be used for cooling the reactor power generation equipment, for bearing lubrication and cooling, and for the transfer of excess heat to the heat sink. Water has been selected as such a coolant. All of the plant components and the coolant would be sealed into the package with no requirement for makeup. As a result, a number of auxiliaries, such as air and steam separators and ejectors, gas pressurizing systems, separate lubrication systems, and the like, would not be required with this plant concept.

B. The entire plant package would be immersed in a large water sink, thereby eliminating the need for a separate emergency cooling system. The condenser can be cooled by natural circulation, which with the aid of a natural circulation primary reactor cooling system, eliminates the need for the usual electrical auxiliaries required to maintain proper reactor temperatures at all times, as well as for special components for emergency circulation.

C. A sufficiently large temperature coefficient of reactivity can be designed into the reactor so that the reactor power may be regulated throughout the life of the plant without the need for moving control rods and mechanical drives.

D. The reactor can be operated at essentially a constant power output for its entire life, and since control rods and the provision for scramming the reactor have been eliminated, the reactor control and safety circuit equipment could be eliminated entirely, adding greatly to the simplicity and reliability of the plant. The history of operating performance of nuclear reactors indicates that the reactor control and safety circuits are those which provide the most shutdowns and spurious scrams.

E. The number of items of rotating equipment would be reduced to a maximum of two - the turbine-alternator and the circulating pump. The radial and thrust bearings for the pump and turbo-alternator would be lubricated with high-pressure water from the feed pump. Since the plant would be required to undergo only one startup operation in its lifetime, the wear on water-cooled bearings that normally occurs during startup would be reduced to the minimum associated with the single initial startup operation. Such wear is sufficiently small that it is not considered necessary to supply an initial source of high-pressure water for the pump bearings at this one startup condition.

F. Since there is sufficient nuclear reactivity in the reactor to provide several years of operation at full power, it is assumed that the alternator would operate under full load at all times in order to minimize the complex equipment normally required to regulate the reactor and power plant output. External resistance loads would be used to supplement the regular electrical power load, and to maintain the load on the turbo-alternator at a constant level.

G. No shielding would be built into the plant other than that provided by the coolant itself. The plant would not be operated prior to shipment to the site, and once the plant was placed in its operational location, its water environment would provide the shielding for any facilities at the site that would use the electrical output from the plant.

H. With the anticipated minimization of auxiliary systems, with the compactness of the arrangement of all equipment, with the elimination of reactor shielding during the shipment phase of installation of the reactor, and with the use of a single type of coolant which is completely sealed and therefore requires no makeup, it is possible to put a plant of sizeable output in a single container. For a power output up to a few hundred electrical kilowatts, the total power package is of a size and weight within the capability of transport by large military air transport planes. The plant would be delivered to a remote site, and put in place so that it could be immersed and cooled by a water pool. The plant would be capable of full-power operation for several years, and provision is made for refueling and replacement of major components.

The elimination of movable control rods, drives, and electronic control and safety systems would contribute substantially to the reliability of the over-all plant system. It is believed possible to design into such a reactor a sufficient temperature coefficient of reactivity to compensate for the change in reactivity over the full operating life of the plant.

The fuel for the reactor could be a dispersion of UO_2 in stainless steel, clad with stainless steel. This fuel, as well as other fuel types, has been experimentally shown to be capable of the required burnup and corrosion resistance for the type of plant and plant conditions contemplated for this application. The temperatures and pressures in the primary coolant loop are modest enough so that conventional construction materials can be used for the reactor and pressure vessel. By proper "over-design" of the turbine-alternator and condensate pump, it should be possible to provide high reliability over the operating life of this equipment. Turbine-alternator packages have been operated for many years, even under changing loads, without maintenance or shutdown. Water-lubricated bearings have been developed, and are in use for large rotating equipment. It is felt that this knowledge can be used to provide reliable bearings for the intended application. The windings of the generator can be maintained at a fairly low temperature, providing a wide latitude in the selection of insulating materials. The generator and the reactor are separated by the natural circulation water system, which contributes to minimizing radiation damage to the insulating materials on the generator.

The heat exchanger, condenser, and other plant equipment would be of a conventional design so as to permit high reliability of operation of this equipment.

It is appreciated that, although the technology associated with the individual pieces of this equipment is fairly well established, the total reliability cannot be demonstrated until a complete plant is put together and tested.

SECTION 3.0

SUMMARY OF RESULTS

A. General

It is concluded from the work accomplished during the study period covered by this report that the unattended nuclear power plant concept is feasible to design and build, and that it offers economic advantage over conventional manned nuclear plants in some remote areas, in the 300 KW rating studied.

B. Technical

1. The direct and indirect boiling, natural circulation concepts were investigated, and both are judged to be feasible to design and build.

2. The reference design is a direct cycle, natural circulation boiling water reactor without control rods, having the turbine-generator and feed pump located on a single shaft, all contained in a single capsule.

3. Several control concepts are described. In the interests of simplicity, low cost, and increased reliability, the reference system requires periodic attendance; the advanced concept is a more complete control system that requires no attendance. Both concepts use a positive (as opposed to inherent) method of controlling reactor pressure.

4. A core life of 3 years can be realized in a field-operated unit. Provision is made in the design to refuel the reactor and to replace major components.

5. The plant as described can be designed using concepts of reactor fuel, power conversion equipment, and plant control that do not require research-type programs to confirm the feasibility of operation of these components.

6. A program to design, fabricate, install, and test a prototype within a time cycle of 30 months is recommended to demonstrate the feasibility of all components to work together in a completely unattended power plant system.

7. The scope of work did not include analysis and evaluation of safeguard problems. However, safeguard considerations were recognized during the course of the work, and areas of potential interest are pointed out. It is believed that the unattended plant design can be developed to the point where destructive nuclear excursions and gross reactor vessel failures are not credible for the field units.

C. Economics

1. Estimate of the cost of design, development, fabrication, and startup of the prototype reactor is $4,118,000. A capital facility to house the prototype with full containment is estimated at $470,000 additional.

Not included in the above is the cost of the six-month power generation test program.

2. Production field units could be built, following prototype operation, with a capital equipment selling price of $600,000 each (based upon an initial order of 10 plants). Fuel could be purchased for $135,000.

3. Annual charges vary from $145,000 to $200,000, depending upon installation and site.

4. On the basis of equipment prices quoted above, the power cost is 56 mills/kwh without considering installation cost. For this plant, installation costs are not a major factor as can be seen from the analysis in Section 8.3, where installation effects are considered. Specific installation costs will of course vary considerably with the relative availability of a water pool.

5. Areas for further reduction of capital and operating costs are still available from design refinements to come and from investigations of specific applications. For example, the cost of demineralizers for removal of soluble boron and the weight of the package suggest more effort be spent on startup and shutdown mechanisms.

SECTION 4.0

RECOMMENDATION

The engineering work performed to date on the concept of a reliable unattended nuclear power plant has indicated that a small power plant of the required rating, 300 KWe, can be designed using components of a type that have already been demonstrated to perform satisfactorily under similar, but not identical, circumstances; has a power generation cycle containing a minimum of moving parts (turbine, generator, and feed pump on one shaft), and can be completely assembled at the factory. All of these factors are in the direction of increasing the reliability of the plant and eliminating site construction and operational costs.

The extent of the feasibility investigation has been confined to engineering design studies, and discussions with vendors having experience in the design of components similar to those proposed for use in the unattended plant. The complete feasibility of such a plant can best be established by building and operating a prototype plant so that safe operation and long, reliable life can be demonstrated. This document describes a logical program of design and fabrication, leading to a demonstration of reliable operation, that should be the next step toward the objective of producing a new line of simple, portable power units.

The reference concept of the unattended power plant is given in detail in Section 6.0 of this document. The program leading to the successful operation of an unmanned power station of the type described above is given in considerable detail in Section 7.0. A summary of the program follows.

- Phase I - Conceptual design, analysis, initial confirmatory tests.
- Phase II - Detailed design, additional confirmatory tests, component fabrication and assembly of the prototype plant and non-nuclear tests.
- Phase III - Cold and hot critical tests of the reactor.
- Phase IV - Full power tests of the entire plant.

It is believed that the plant described in Section 6.0 of this document can be designed using concepts of reactor fuel, power conversion equipment, and plant control that do not require research-type programs to confirm the feasibility of operation of those components. Rather, the program envisions confirmatory-type tests that will provide design data, life expectancy, and reliability information.

The over-all prototype program described in Section 7.0 appears reasonable to complete in 30 months, including several months of full power operation. Life and reliability testing of the complete plant could extend beyond that point as required to completely demonstrate success in those areas.

The General Electric Company believes that the ultimate in freedom from logistical problems has not yet been achieved by nuclear power plants now under development by the Armed Forces, and that the logical next generation of logistic freedom will come by the use of greater design simplicity and unattended long-lived operation. A program of the type described in this document should be pursued immediately to achieve the objectives listed above.

SECTION 5.0

SELECTION OF THE MOST PROMISING CONCEPT

5.1 Summary

A. Basic Feasibility and Cycle Selection

Various thermal cycles were considered for application to the capsule reactor concept. Two cycles, namely the indirect and the direct boiling cycles, were investigated extensively. It was determined from these investigations that both cycles are feasible. However, it is recommended that the direct cycle be pursued for the reasons of lower weight, potentially lower cost, capability to grow to other ratings, the use of developed reactor technology, and ease of shipment and installation. The indirect cycle offers a slight advantage in requiring less non-standard hardware to build the prototype. A summary of the factors resulting in these conclusions is given below, and further detail is given in succeeding portions of this section of the report.

B. Plant Size and Weight

The direct cycle plant is approximately 20 feet long and 7 feet in diameter, and weighs about 30,000 pounds. The indirect cycle plant has the same dimensions, but weights in excess of 40,000 pounds. The weight difference is due to the shorter chimney and lower operating pressure of the direct cycle, which results in a shorter, thinner wall reactor vessel, and in the elimination of the primary heat exchanger required for the indirect cycle. The lower weight is an advantage permitting air transportation to remote sites.

C. Plant Control

Both cycles require a positive means for controlling pressure of the primary system; i.e., a pressure regulator. (The indirect cycle might use a compact pressurizer for this regulation.) Both cycles also require some means of controlling water level and electrical load variations. In addition, the direct cycle requires a throttle valve or orifice to supply constant pressure to the turbine. The hardware required for these control functions is more standard for the indirect cycle, and for this reason it would probably be easier and less expensive to build the control system for the first indirect cycle plant than for the first direct cycle plant.

D. Capability for Increasing Rating

The direct cycle promises to provide the capability for increased power rating because of the design margins and flexibility in the use of both nuclear and hydraulic parameters. The initial core for a boiling water reactor of a specific design is very conservative in order to provide assurance of satisfactory operation until operating experience is gained. This design approach provides room for growth within the design margins. The direct cycle BWR for Capsule offers the additional variable of the reactivity in voids. The indirect cycle is inhibited in its growth potential because of the flow restrictions imposed by the heat exchanger in the downcomer and the comparatively low thermal driving head available to move water through the core by natural convection.

E. Power Conversion Equipment

The turbine and feedwater pump are considered to be more difficult to design for the direct cycle, but within the state of current technology. The single-stage turbine requires short buckets, and exhausts steam with a high moisture content. The problems concerned with designing a feedwater pump having low flow and back pressure and high head are well known. On the other hand, the design of these components for the indirect cycle is fairly conventional. Work accomplished thus far has shown that the turbine, alternator, and pumps can be arranged on a single shaft for either cycle.

F. Core Design

The physics and thermal hydraulics of the core are similar for both cycles and, in all cases, fall within conventional design margins. Emphasis will be placed on the distribution of burnable poison which will most successfully flatten the core, and on verification of the prediction of the variation of reactivity with moderator temperature.

G. One Versus Two Capsules

A study was made of arrangement of the plant into two capsules. It was determined that arrangement of the plant into either one or two capsules, using either the direct or indirect cycle designs, was feasible. Arrangement of the reactor into one capsule, and the power conversion equipment and external heat exchanger into a second capsule, offered the advantages of elimination of the spray condenser and circulating water pump. The spray condenser would be replaced by a falling film heat exchanger.

The disadvantages of the two-capsule arrangement are that access for servicing is more difficult, the piping and header connections are complex, and the feedwater pump must either be separately motor-driven or be driven through a long shaft extension from the turbine-generator alternator shaft. In addition, the problems of transportability and ease of installation favor the single capsule.

H. Partial Attendance

Consideration has been given to the possibility of obtaining advantage from the use of a man in partial attendance for periodic plant adjustment for the purposes of reducing cost, extending plant life, or increasing reliability.

At this stage of design, the emphasis has been placed upon inherent control where outside influences, either manual or automatic, are at a minimum. Where outside influence has been required, a reliable and simple device can be conceived. It is recognized that in bringing concepts to fruition, details may arise where the reliability of equipment will be questioned. The possibility of periodic manual adjustment will be kept continually in mind during the plant detail design stage.

There are two major areas of the design where plant performance and reliability might be improved by partial attendance. These areas are plant control and reactor water cleanup.

It is considered feasible to design plant control systems of varied complexity, ranging from simple manual valves operated by reach rods to completely automated systems that will control and trim the plant for maximum safety and efficiency. Since cost and the number of moving parts increase with increased complexity, and reliability decreases with an increased number of moving parts, it is simply a question of which is more important for the application; complete unattendance or higher reliability and lower cost. The path chosen here is toward simplicity, low cost, and high reliability by reducing the number of parts insofar as is possible. In other parts of this report and in Ref. 1, various systems ranging in complexity from simple manual control by periodic adjustment to completely unattended are described. Considerable thought has been given to cleanup of the primary water system. It is concluded that there is a possibility that by initial cleanup during startup operation, and for a short period thereafter, the reactor may run unattended for its core life without harmful effects from accumulation of corrosion products. Other means of controlling corrosion product buildup are recommended for evaluation during the prototype testing phase. The final course of action will be established by the results of prototype testing.

5.2 Summary of Preliminary Plant Performance Data and Design Criteria

A. General Plant Data	DIRECT		INDIRECT	
	Start of Life	End of Life	Start of Life	End of Life
Reactor Type	Boiling Water		Boiling Water	
Cycle	Direct, Natural Circulation		Indirect, Natural Circulation	
Design Life	3 Years		3 Years	
Gross Output, KWe	310	310	310	310
Thermal Power Output, KW*	1740	1740	2400	2400
Steam Flow Rate, lb/hr* (No Bypass)	5680	5520	7050	7350
Reactor Pressure, psia	1200	600	2000	930
Reactor Outlet Temperature, °F	567	486	636	536

* The thermal power output and steam flow rates are lower here than reported later due to the use of higher condenser pressure required to satisfy pump conditions when the pumps are mounted on the same shaft with the turbine-generator, and also because early predictions of steam requirements were optimistic.

B. Fuel and Core Assembly	DIRECT Start of Life	DIRECT End of Life	INDIRECT Start of Life	INDIRECT End of Life
Fuel Material	Fully enriched UO_2 dispersed in stainless steel		Fully enriched UO_2 dispersed in stainless steel	
Fuel Density	$10_g\ UO_2/cm^3$		$10^g\ UO_2/cm^3$	
Total Weight of UO_2 in Reactor, kg	16	13.1	16	12
Metal/Fuel Ratio	12:1		12:1	
Clad Material	Type 304 Stainless Steel		Type 304 Stainless Steel	
Clad Thickness, in.	.010		.010	
Fuel Plate Thickness, in.	.045		.045	
Active Length of Each Plate, in.	23.0		.23.0	
Active Width of Each Fuel Plate, in.	2.5		2.5	
Side Plate Material	Stainless Steel		Stainless Steel	
Side Plate Thickness, in.	.025		.025	
Number of Plates per Ass'y	10		10	
Cross Sectional Size of Ass'y	3.0 in. x 3.0 in.		3.0 in. x 3.0 in.	
Distance Between Plate Centers, in.	.30		.30	
Number of Assemblies	37		37	
Total Number of Plates	370		370	
Fuel Ass'y Weight, lbs. (approx.)	10		10	
Over-all Length of Fuel Ass'y, in.	26		26	
UO_2 in Meat of Element, Weight %	21		21	
Stainless Steel in Meat of Element, Weight %	79		79	

		DIRECT		INDIRECT	
		Start of Life	End of Life	Start of Life	End of Life
C.	Nuclear Characteristics				
	Core Configuration	Right Cylinder		Right Cylinder	
	Equivalent Core Diameter, in.	20		20	
	Equivalent Reflector Thickness, in.	4		4	
	Enrichment, %	93		93	
	Fuel Burnup, kg		4		4
	Reactivity Balance, K				
	Moderator Temp. Coefficient	.04 to .05		.06 to .07	
	Void Fraction	.02 to .03		.01 to .02	
	Burnable Poisons	.02 to .03		.02 to .03	
	Total		.08 to .10		
	UO_2 in Element, Volume %	1.1		1.1	
	Stainless Steel in Element, Volume %	13.0		13.0	
	Water in Element, Volume %	85.9		85.9	
D.	Heat Transfer				
	Total Heat Transfer Area, ft^2	333		333	
	Heat Flux, Average, $Btu/hr/ft^2$	18,000		25,000	
	Minimum Burnout Factor	22		--	
	Maximum Fuel Temperature, °F	645		--	
	Average Fuel Surface Temp., °F	575	495	648	544
E.	Hydraulics				
	Coolant Flow Rate, lb/hr x 10^{-5} (Total Core Flow)	9.25	11.1	1.53	1.84
	Core Inlet Velocity, ft/sec.	2.3	2.5	.12	.14
	Feedwater Return Enthalpy, Btu/lb.	134	134		

	Start of Life	End of Life	Start of Life	End of Life
Steam Volume Fraction				
Average	0.045	0.055	.021	.026
Exit	0.094	0.105	.042	.052
Fuel Assembly Flow Area, in.^{2}	8.54		8.54	
Hydraulic Radius of Fuel Ass'y, in.	1.262		1.262	
Fuel Ass'y Flow Friction Length, in.	24		24	
Minimum Effective Chimney Height, ft.	4		8	
Average Downcomer Liquid Velocity (adjacent to chimney), ft/sec.	2.1	2.3	.65	.78

F. Component Design Data

Pressure Vessel		
Inside Diameter, in. (approx.)	30	30
Length, ft. (approx.)	11	15
Design Pressure, psia	1320	2200
Working Pressure, psia	1200	2000
Head Opening, in.	30	30
Weight, lbs.*	7000	13,200
Primary Heat Exchanger		
Tube I.D., in.	--	.50
Tube, O.D., in.	--	.75
Tube Length, ft.	--	40
Number of Tubes	--	21
Material	--	Stainless Steel

* The vessel weight was previously reported as 5800 pounds based on preliminary calculations.

	DIRECT		INDIRECT	
	Start of Life	End of Life	Start of Life	End of Life
Alternator (Direct and Indirect Cycle)				
Type	Homopolar Inductor Alternator			
Speed, rpm	12,000			
Poles	8			
Freq., cps	800			
Weight, lbs. External SCR inverter required to convert 800 cps to 400 cps	1700			
Turbine				
Speed, rpm	12,000		12,000	
Turbine Shaft Power, KW	356		356	
Inlet Pressure, psia	600		150	
Inlet Temperature, ^{o}F	486		450	
Exhaust Pressure, psia*	8		8	
Pitch Diameter	17.4		21	
Quality, Exhaust, %	85		95	
Turbine Eff.	64		70	

* Separately driven pumps.

5.3 Plant Description

A variety of arrangements, combinations of equipment, and control schemes are possible for the capsule reactor based upon study of the concept to this time. These include a choice of thermal cycles, arrangement in one or more capsules, separate pumps or pumps integral with the turbine-generator shaft, and various combinations of steam relief, pressure control and steam flow divider controls. These conditions have been investigated to determine which combinations provide the most advantages from the standpoint of simplicity, cost savings, weight, compactness and accessibility, consistent with the unattended concept.

Typical arrangements that point out the various characteristics and advantages and disadvantages of the plant designs investigated are described below.

A. Plant Arrangement

Indirect Cycle, Single Capsule (Fig. 1)

The indirect cycle plant arranged in a single capsule is the original concept as described in Reference 3. Dimensions are approximately 7 feet outside diameter by 20 feet long. Total dry weight is approximately 40,000 pounds. More recent weight calculations indicate that the plant may weigh several thousand pounds more than this figure. The primary system consists of the reactor core and chimney, the primary heat exchanger or once-through boiler, and the primary water, all enclosed and sealed in the reactor pressure vessel. A sheet metal shroud surrounds the pressure vessel, providing an air space for insulation and forming a portion of the condensate sump. The turbine and alternator are integral, and mounted to the top of the reactor vessel. The feed pump and circulating pump, driven by a single electric motor, are located in the condensate sump.

All of the equipment described is enclosed by a metal capsule which seals it from the water of the heat sink in which the capsule is submerged. The lower portion of the capsule also forms the outer wall of the condensate sump, and the top of the capsule becomes the condenser for turbine steam. A heat exchanger surrounds the outside of the capsule, and provides for cooling the condensate water. A sheet metal shroud surrounds and protects the heat exchanger tubes, and aids in the flow of water past the tubes.

Access to the turbine-generator is by unbolting and removal of the capsule flanged head. Access to the core is gained by removal of the turbine-generator and by removal of the bolted head from the reactor pressure vessel.

When the plant is in operation, the circulating pump takes water from the condensate sump, and supplies the larger portion of it to the external heat exchanger. The water (which condenses the turbine steam) is cooled and enters the condenser in the dome of the capsule through spray nozzles. The spray water and condensed steam flows downward to turn to the condensate sump. The smaller portion of water from the circulating pump supplies back pressure to the feed pump. The feed pump supplies water to the primary heat exchanger or once-through boiler through the boiler inlet header. As the water flows through the flash boiler, steam is formed and enters the turbine through pipes penetrating the top of the reactor vessel. The steam leaving the turbine is condensed by the spray nozzles, as previously described, and the cycle is repeated.

Control of the reactor and electrical load is described in Section 5.3C, and descriptions of the turbine-alternator and pump design are given in Section 5.3D.

Direct Cycle, Single Capsule (Fig. 2)

Figure 2 is an over-all assembly drawing of the plant. Dimensions are approximately 7 feet outside diameter by 16 feet long. Total dry weight is approximately 30,000 pounds. The difference in weight and length between the indirect and direct cycle plant is due to the 4 foot shorter chimney length of the direct cycle reactor, which is reflected in a shorter reactor pressure vessel. Also, the upper operating pressure of the direct cycle results in a lighter pressure vessel. The direct cycle does not have a primary heat exchanger.

The arrangement of reactor pressure vessel, core and chimney, turbine-alternator, pumps, and spray condenser is the same as for the indirect cycle. Freer access to the core is provided through the larger flanged pressure vessel head because of elimination of the primary heat exchanger.

The operation of the spray condenser and pump is similar to the indirect cycle except that high-pressure steam flows directly to the turbine and the feedwater pump must return the water to the reactor at high pressure. This results in more difficult feed pump design.

Direct Cycle, Single Capsule, Pumps Integral with Turbine-Alternator

Figure 3 is an over-all assembly drawing of the direct cycle plant in a single capsule, with the pumps on the same shaft as the turbine and alternator. This has the advantage of eliminating the separate motor drive for the pumps. The plant weight would be about the same as for the direct cycle with separately driven pumps, but the over-all capsule length would increase by about 2 feet to a total of 18 feet. The increase in length is due to adding the pump height to the turbine-alternator.

The turbine is located above the alternator, which permits the alternator to operate out of the direct path of steam exhausting from the turbine. The sump water level is brought up high to provide at least 3 feet of head for the first stage of the pump. A required increase in condenser pressure increases steam flow rate and decreases plant efficiency.

Arrangement of the Plant into Two Capsules, Indirect Cycle (Fig. 4)

Figure 4 is an arrangement of the direct cycle plant into two capsules. One capsule contains the steam-generating equipment, and the other contains the power conversion equipment and has the cooling water heat exchanger mounted on the outside.

The drawing shows the indirect cycle, but the direct cycle would be subject to approximately the same considerations. The main advantage is the possible use of a falling film heat exchanger which eliminates the circulating water pump stage. The falling film heat exchanger requires that the feedwater pump be located low in the capsule to obtain pump back pressure. Thus, the falling film heat exchanger would require that the pump be located below the reactor core in the use of the single-capsule arrangement. For this reason, the falling film heat exchanger is not considered practical for the single-capsule arrangement.

In the two-capsule arrangement, the pump may be directly connected to the turbine-alternator shaft by a shaft extension, or be separately driven by an electric motor.

Access to the equipment in the capsules is by removal of the capsule heads.

In the case of the reactor capsule, the piping connection to the headers is broken before the vessel head is unbolted to permit access to the core. The primary heat exchanger is removed with the head. In the case of the capsule containing the rotating equipment, the header connections to the turbine are unbolted before the turbine-generator can be unbolted and removed. The feed pump is then accessible. In comparing ease of accessibility of the two-capsule arrangement with the single capsule, it is judged that access to the two-capsule system is more difficult from the standpoint of removing two capsule heads and all of the header connections before getting to the major components.

Even though each of the two capsules is lighter in weight than the single capsule, the total weight of the two-capsule arrangement will be greater. The shipping advantage of having two smaller capsules instead of one large one may be overcome by designing the indirect cycle so that the reactor pressure vessel with the turbine-alternator mounted to it could be shipped separated and mounted in the field. It appears that the direct cycle, single-capsule plant arrangement would be within maximum size and weight considerations for air transportation without disassembly into two parts.

The advantage offered by the two-capsule arrangement are elimination of the spray condenser and circulating pump stage. This advantage is balanced by loss of simplicity and compactness, increased total weight, complexity of piping and header connections for the falling film heat exchanger, and more difficult access for servicing. Removal of electrical components from the radiation zone is not considered important since the radiation level at the top of the single-capsule arrangement is within tolerable design limits. In consideration of these facts, the selection of a one or two capsule arrangement is dependent upon how the advantage of increased reliability by reduction of parts and of difficulty of access for servicing is weighed.

To preserve the simplicity of the concept, the single-capsule arrangement is recommended.

B. Preliminary Plant Heat Balance

Indirect Cycle (Fig. 5)

Figure 5 is a preliminary heat balance for a 150 psia indirect cycle plant. This is inconsistent with the selected turbine inlet pressure of 120 psia developed in Section 5.4, but is typical of the system and will be refined when the design is better established. The drawing shows that the turbine is supplied with 7350 lbs/hr of steam which produces 310 KWe gross alternator output. The steam is condensed to 8 psia, and flows to the condensate sump where the circulating pump raises the pressure to 50 psia. The outlet from the circulating pump divides two ways. The major portion flows through the cooling water heat exchanger, where the temperature is reduced to 90° F, and supplies the turbine-generator bearings and the condenser spray nozzles. The smaller portion of the circulating pump outlet supplies back pressure to the feed pump which raises the pressure to 180 psia. The feedwater flows to the flash boiler, where it becomes superheated steam, and enters the turbine at 150 psia. The turbine exhausts to 8 psia in the condenser, where the spray nozzles provide cold water for condensing. The condensed steam and spray water flow to the condensate sump and the cycle is repeated.

The condition shown is for end of life. The turbine steam temperature drops off as reactor fuel burns up. Steam mass flow is increased by control of feed pump speed to compensate for loss of energy in the steam. Control schemes are discussed in detail in Section 5.3C.

Direct Cycle (Fig. 6)

Figure 6 is a preliminary heat balance for 600 psia conditions (end of life) for the direct cycle with separately driven pumps. The steam flow rate and thermal power rating of the reactor are based upon preliminary estimates of steam flow requirements, and do not include bypass steam required for control.

The turbine is supplied with 5520 lbs/hr of steam directly from the reactor. The steam is condensed to 8 psia, and flows to the condensate sump, where the circulating pump raises the pressure to 50 psia. The outlet from the circulating pump divides two ways, as for the indirect cycle. The major portion flows through the cooling water heat exchanger, where the temperature is reduced to 90° F, and supplies the turbine-generator bearings and the condenser spray nozzles. The smaller portion of the circulating pump outlet supplies back pressure to the feed pump which raises pressure to 700 psia for delivery to the reactor vessel. The reactor generates steam which flows directly to the turbine, and the cycle is repeated.

C. Control and Transient Performance

The direct cycle and the indirect cycle boiling water reactors have been proposed for this application. In this section of the report, control systems for both cycles are described; this includes both existing systems on operating plants and proposed new concepts. A reference control concept is described for each cycle. The concepts for both cycles are feasible. Briefly, both methods use a positive (as opposed to inherent) method of controlling reactor pressure and power.

Control Concepts

Previous BWR experience and studies have yielded the following control concepts:

1. Dual Cycle. Subcooling and voids are controlled by withdrawing heat from steam generators in the recirculation flow loops. Reactor pressure is controlled by a pressure-regulated turbine control valve that controls steam flow from the reactor. The control range is limited.

2. Feedwater Temperature Control. Subcooling and voids are controlled by varying the feedwater temperature. Reactor pressure is controlled by a pressure-regulated turbine control valve. The control range is limited.

3. Recirculation Flow Control. Subcooling and voids are controlled by varying recirculation flow. Reactor pressure is controlled with the recirculation flow control system or by a pressure-regulated turbine. The control range is limited.

4. Control Rods - the most versatile control device.

5. Burnable poisons in the fuel and/or the water. This is a limited-range, long-term, or steady-state feature. Burnable poisons in the fuel are used in the initial core design to the extent of approximately 6% $\Delta k/k$.

6. Moderator Temperature Control. A method of control inherent in the core design; it offers the possibility of inherent pressure control and load response. In a BWR, the moderator temperature is determined by the operating pressure. In the initial core design, moderator temperature is used to control reactivity over the 3-year life by permitting pressure to decrease.

Direct Cycle Control System

The main control requirements are:

Reactor power
Reactor pressure and turbine pressure
Turbine speed or generator frequency
Reactor water level

Figure 7A is a simplified flow diagram with a representative concept for the direct cycle. At the start of operation, the reactor will be at 1200 psia and about 110/105% power. The excess power 10/5% will be bypassed to the condenser through a pressure-regulated bypass valve; thus, reactor pressure is controlled by a positive method. The turbine will be designed for a constant pressure of 600 psia. Therefore, a pressure-reducing valve will be required to maintain the turbine inlet pressure at 600 psia over a range of 1200 to 600 psia reactor pressure. Water level will be maintained in the reactor and sump by controlling a valve on the discharge of the feed pump or by a simple standpipe arrangement in the condenser. The turbine speed will be controlled by modulating a dummy electrical load to absorb small disturbances in the actual load or the turbine steam flow.

With operating life, reactor power will decrease because of fuel burn-up. The loss of core fuel burnup reactivity will be compensated to a limited degree by the use of burnable poisons; however, this effect will not be adequate for the 3-year life. The additional required positive reactivity will be provided by the moderator temperature coefficient. Reactor pressure will be lowered to the value necessary to raise power back to the 110/105% level. This step will be accomplished by reducing the set point on the bypass pressure regulator. Preliminary calculations show that by reducing pressure from 1200 psia (567° F) to 600 psia (486°F), rated reactor power can be maintained.

Indirect Cycle

The control requirements are the same as for the direct cycle. Figure 7B is a flow diagram of the indirect cycle with a representative control concept. Primary pressure is controlled by controlling the power absorbed by the steam generator. In turn, the power from the steam generator is controlled by controlling the feedwater flow supplied to it from the feedwater pump. The water level in the steam generator and the sump should be maintained inherently. Turbine speed will be controlled in the same way as in the direct cycle.

Reactor power will be maintained, as in the direct cycle, by permitting reactor pressure to vary from 2000 psia (636° F) to 930 psia (536° F). Changes in pressure will be made by changing the set point on the control system between primary and secondary feedwater flow.

D. Power Conversion Equipment

Generator Design Requirements

Subsequent to the broad analysis of the turbine made and reported on later in this report, a much more specific analysis is made to pinpoint the actual design specifications.

One point, not originally required, is the apparent need for 400-cycle output from the power plant. Our original specifications of the generator had been as follows:

Homopolar inductor alternator

6 pole - 600 cycles
12000 rpm
400 to 600 volts
3 phase - power factor 1.0

We discussed this new need at length with electrical machinery specialists, and submit a resume of the deliberations.

Homopolar Inductor Alternator

Advantages:

1. Simple and reliable
2. No rotor windings
3. Uses stationary rectifiers
4. Operated under variable power factor load
5. Turbine/generator unit construction

Disadvantages:

1. Must have symmetrical inertia about any cross-section axes; i.e., no less than 6 poles.
2. Magnetic field initially exerted by surge of current at startup.
3. External rectifiers for exciting d-c field windings
4. High specific weight

Induction Alternator

Advantages:

1. Freedom to construct any even number of poles
2. Low specific weight

Disadvantages:

1. Bar windings in rotor
2. Capacitors across windings
3. May require additional capacitors in series with output leads to correct for load power factor.
4. Requires rotating rectifiers
5. Turbine rotor may be separable from generator shaft.
6. Lower reliability due to rotor windings and need to cool capacitors below 75° C

Both machines may be potted with Urathane compound. This material has been found compatible for duty such as we envision in steam and radiation environments. It is necessary, however, to maintain the temperature of the Urathane below 200° F. This may be done by supplying water coolant if needed.

It is strongly recommended that the Homopolar Inductor Alternator be used. A tabulation of weights and sizes of the two types of machines is shown below:

SPEED	CYCLES	POLES	ROTOR DIAM.	STACK LENGTH	COIL LENGTH	STATOR DIAM.	WT.
Homopolar Inductor Alternator - 320 KW - 3Ph. - 400/600V - PF = 1.0							
12000	800	8	15-3/4	10-1/2	19-1/2	27	1700
24000	1600	8	11	7-3/4	14-1/2	19	700
Induction Generator - same specs.							
12000	400	4	8-1/2	16	26	14-1/2	800
	Plus rotating capacitors (3) - 150# wt. - 1.38 ft3 vol.						

A 6-pole homopolar generator to produce 400 cycles must run at 8000 rpm and, from the above figures for an 8-pole unit, is quite large and heavy. As will be obvious later, the turbine will also be very large. In fact, it would be quite impractical to design such a turbine with any reasonable performance. Its pitch diameter would be 31 inches, bucket heights would approximate .25 inches, and each stage of a two-stage machine would weigh about 450 pounds.

For these reasons, it was felt that the homopolar generator would have to be run at a higher speed, thus producing more than 400 cycles. This implies the use of inversion equipment to convert the higher output frequency to the required 400.

An investigation was made concerning the feasibility of reliable static inverters. It has been determined that circuitry has been developed, at lower levels, that could handle as much as 1200 cycles inverted to 400 cycles with a reliability equivalent to that of the generator for a 3 to 5 year period. Silicone Controlled Rectifiers would be used. For 320 KW load, a volume of 1 to 2 cubic feet would be required. This equipment may be installed inside the capsule, with cooling if needed outside the capsule or at the terminal ends at the load connection. We therefore recommend:

1. Selection of the homopolar induction alternator having the following characteristics:

   320 KW output
   3 phase, 400/600 volts
   Power factor - 1.0
   12000 rpm
   8 poles
   800 cycles

2. Use of SCR circuitry for inverting alternator output to 400 cycles.

3. Select a 12000 rpm, 400 cycle induction alternator only if 800 cycles or use of SCR inverter found unacceptable by the customer.

Water-Lubricated Bearings

Preliminary discussions were held with specialists of the General Engineering Laboratory Bearing Center concerning design of the bearings for the turbine-generator unit. After considering the range of component weights and speeds, and the state of bearing art, it became apparent that to operate above 12000 rpm would require an undue development effort. Althrough at this writing no specific bearing design work has begun for this application, we conclude that 12000 rpm, from the standpoint of the bearing, is a reasonable speed selection.

Turbine-Generator Design Concepts for Indirect Cycle

Specific analyses were made of the turbine for direct coupling to the generator. Power level was determined on the basis of:

Net Electric Output	300 KW
Pump Power	20 KW
Generator Efficiency	90%
Turbine Shaft Output	356 KW
Turbine Inlet Pressure	150 psia
Turbine Inlet Temperature (end-of-life conditions)	450° F
Turbine Exhaust Pressure	8 psia
Turbine Flow Rate	8910 lbs/hr

Approximate rotor size and other characteristics for a 2-stage turbine were estimated as:

Rotor speed (rpm)	8000	12000	16000	24000
Pitch diameter (in.)	31.5	21.0	15.75	10.50
Bucket height on 2nd stage (in.) (full admission)	0.273	0.410	0.546	0.818
Rotor weight per stage (lbs)	497	115	49	18.2
Bucket root stress (psi)	2360	5300	9420	21200

A cross plot was made at the 12000 rpm speed to show the effect of varying turbine inlet pressure. This was done for the case of exhaust pressure - 8 psia.

Inlet pressure (psia)	100	75	50
Flow (lbs/hr)	9800	10630	12300
Pitch diameter (in.)	21.0	21.0	21.0
Bucket root stress (psi)	6120	7670	10760

It is apparent that improvements can be made in the aerodynamics of the flow path by increasing bucket heights, and by decreasing inlet pressure. It is a conclusion that there would be advantages to selecting an inlet pressure of 50 to 75 psia.

No specific analysis was made to couple the turbine exhaust and spray condenser flow requirements to the feedwater pump.

Turbine-Generator Design Concept for the Direct Cycle

The conclusions reached concerning the selection of speed and generator type apply also to the direct cycle.

However, as the feedwater pump for the direct cycle is very critically tied into the selected operating conditions of the turbine, an extensive analysis of this unit has begun.

The feed and circulating pump will consist of a number of stages, the first stage of which will handle both the feedwater and spray condenser flow requirements. Subsequent stages will handle the feedwater flow component.

Two conditions must be met:

1. Suction conditions at the inlet of the first stage require essentially low speed, low flow, and high suction head to minimize inlet cavitation.

2. Design practice of the pump impeller implies reasonable speeds, flow, and moderate heads.

It turns out, in this application where all stages of the pump may be on one shaft, that the conditions tend to be incompatible. Several possibilities present themselves:

1. All pumps on same shaft driven by their own motor located at extreme bottom of capsule. This permits running at slow speed (6000 rpm), favoring the suction conditions and consequently permitting operation of turbine exhausts as low as 8 psia. High-pressure stages, however, are severely handicapped and in order to supply feedwater up to 1200 psia reactor pressure would require numerous stages. The design would be an open impeller type and require intensive development.

2. All pumps on same shaft driven by a high-speed (24000 rpm) motor. This concept favors the high-pressure stages to the extent that they are considerably easier to design. The first-stage suction conditions are such that this speed may only be considered for the case of using a falling film type heat exchanger rather than a spray condenser; or by increasing the turbine exhaust pressure and hence the suction head on the pump.

3. Possibility of mounting all the pumps onto the 12000 rpm turbine-generator shaft. Pump stage conditions then fall somewhere between the first two mentioned.

   To explore this further, studies were begun of the pump mounted to the main unit shaft. In addition, a minimum static water head was considered so the turbine-generator package could still be located at the top of the capsule. At this location, the spray condenser has to be used. A minimum of 3 high-pressure stages of an open impeller design and a first-stage centrifugal are tentatively selected to supply the 1350 psi head rise in the direct cycle system.

These pumps would be mounted at the lower end of the main unit, with a minimum water height above the pump inlet of 3 feet. This implies that at the range of conditions imposed, a turbine exhaust pressure of 20 psia or more must be considered.

To permit operating the first-stage pump at reasonable suction inlet heads under these conditions requires a relatively high turbine exhaust and pump pressure with adequate subcooling.

Analysis was made of a multi-stage turbine for the direct cycle application for the end of the cycle operation; i.e., 600 psia turbine inlet pressure at saturated temperature conditions over a range of back pressures to determine turbine characteristics.

Conditions assumed:

Turbine shaft power - 356 KW

Turbine inlet pressure - 600 psia

Inlet temperature - 486° F

Rotor speed - 12000 rpm

3-stage turbine, equal pressure ratios

Specific speed ratio - 0.45

Water extraction at each stage

P_{exh} (psia)	8	20	30
Stage pressure ratio	4.22	3.105	2.72
Rotor Pd (in.)	20.6	17.4	16.3
Minimum quality (%), 1st stage	92.1	94.0	94.5
Flow rate (lbs/hr)	5970	7400	8300
Turbine efficiency (%)	63.7	64.4	64.4

A preliminary concept of the direct cycle turbine-generator - pump assembly is shown on Figure 3.

5.4 Reactor Design

A. Thermal Hydraulics, Primary System

The thermal hydraulic characteristics of the primary systems of the indirect and direct cycles have been investigated. In reviewing the indirect cycle core design, particular emphasis has been placed on steam void accumulation; also considered were design of the primary heat exchanger, or once-through boiler, and determination of allowable steam turbine inlet pressure. For the direct cycle, emphasis has been placed on estimating the steam void volume fraction likely to exist in the core over the range of expected operating conditions.

Indirect Cycle Core Design

Results

1. The reactor appears to have adequate heat transfer surface area in both the core and the downcomer heat exchanger for the thermal duty required.

2. The concept is based on saturated conditions at the core outlet and 36° F sub-cooling at the core inlet (Ref. 3). The corresponding flow rate at rated conditions is to be provided by natural convection.

3. The over-all average steam volume fraction is approximately 0.021 for rated conditions at the beginning of life and approximately 0.026 for rated conditions at the end of life.

4. It was determined that for adequate heat transfer area, 40 feet long, 3/4" x 1/2" boiler tubes would be required. This length would allow approximately 35% extra area as a factor of safety for fouling, error in heat transfer coefficients, etc.

5. The minimum turbine inlet pressure was found to be 120 psia.

Direct Cycle Core Design

Results

1. Preliminary sizing of the core results in the average core heat flux being of the order of magnitude of 15,000 to 25,000 Btu/hr-ft^2. With this low heat flux, and with the flat plate fuel design, neither fuel temperature nor boiling burnout appear to be limiting factors. Consequently, effort at this time was restricted to estimating the chimney height required for internal natural circulation in accordance with a preliminary limit on over-all core average steam volume of about 5%.*

* Note: This was later relaxed.

2. Two core sizes have been considered: the original reference design using 25 elements 2 feet long, and a proposed design using 37 elements 2 feet long.

3. It has been decided to use 600 psia as the lowest reactor operating pressure (at the end of the fuel cycle).

Calculated and Recommended Design Chimney Height

	Calculated Height	Recommended Design Height[(a)]
25 - Assembly Core	5.3 ft.	8 ft.[(b)]
37 - Assembly Core	2.6 ft.	4 ft.

(a) The recommended design height allows approximately 50% margin for present uncertainties of actual steam volume fractions and pressure drops at the very low steam qualities required and for possible steam entrainment in the downcomer.

(b) The present reactor vessel will not permit a chimney of appreciably greater height than 8 feet.

B. Physics Analysis

The fuel element under investigation is a flat plate, highly enriched UO_2 stainless steel cermet fuel, stainless steel clad. There are ten plates in each 3" x 3" assembly and 37 assemblies in the core.

Two significant parameters have been investigated in the preliminary scoping. These are:

1. Moderator temperature coefficient

2. Core size

By changing moderator temperature through changes in system pressure, compensation may be made for the effect of burnup. This compensation results from the increase in K_{eff} which accompanies an increase in moderator density when temperature is decreased. The effect of temperature is much more pronounced in the high temperature region (600° F) than in lower regions.

Core size has very significant effects on K_{eff} in this small size range. This results from the fact that a slightly smaller core has significantly more neutron leakage. The optimum effective core diameter has been found to be in the range of 20 to 23 inches.

SECTION 6.0

DESCRIPTION OF THE MOST PROMISING CONCEPT

6.1 General Description (Figures 8 and 9)

The information given in this section of the report deals with the field production model of the capsule reactor as the reference design. Performance is based upon anticipated successful operation of a full-scale prototype.

The plant is rated at 300 KW net electrical output. It consists of a natural circulation, direct cycle boiling water reactor supplying steam directly to a single-stage turbine. The alternator produces 3-phase power at 800 cycles per second, 400/600 volts, and unity power factor. Silicone-controlled rectifiers invert tne alternator output to 400 cycles per second.

The turbine, alternator, circulating pump, and feed pump are mounted on a single shaft which rotates at 12,000 rpm.

The reactor primary vessel and turbine-alternator-pump unit are enclosed in a cylindrical capsule. The top of the capsule forms the steam condenser. Cooling water is circulated from the condensate sump through a heat exchanger which surrounds the capsule and is sprayed into the top of the capsule to condense the turbine exhaust steam. The spray water and condensed steam are collected in the condensate sump. The feed pump also takes water from the condensate sump, and pumps it at high pressure into the primary vessel to maintain reactor water level.

The entire plant is submerged in a lake or other suitable heat sink which provides the means for dissipating waste heat. The heat exchanger is cooled by natural convection.

The reactor is self-controlled by its negative temperature coefficient, and no control rods are provided in the reactor structure. The reactor will initially be brought to criticality by the removal of soluble poison from the primary coolant by circulation of the coolant through an ion exchange system. The ion exchanger, its circulating pump, and the equipment provided to control the turbine-alternator are intended to be assembled in a second package that may be located on land adjacent to the power unit.

The plant is arranged to provide for refueling and maintenance of major components. The heat sink for the plant also provides the shielding for these operations and during startup when personnel will be in attendance.

A total weight of about 30,000 pounds and a size of 7 feet diameter and 20 feet length permit the power plant to be shipped by presently available cargo aircraft.

A detailed description of the features of the design and method of operation are given in the remainder of this section.

6.2 Plant Performance and Design Criteria

A. General Plant Data

	Start of Life	End of Life
Reactor Type	Boiling Water	
Cycle	Direct, Natural Circulation	
Design Life	3 Years	
Gross Output, KWe	300	300
Thermal Power Output, KW	3340	3340
Steam Flow Rate, lbs/hr (5% bypass)	11,655	11,340
Reactor Pressure, psia	1200	600
Reactor Outlet Temperature, ° F	567	486

B. Fuel and Core Assembly

Fuel Material	Fully enriched UO_2 dispersed in stainless steel	
Fuel Density, 10g UO_2/cm^3		
Total Weight of UO_2 in Reactor, Kg		
w/o Boron	13.8	12.0
w Boron	16	11.8
Metal/Fuel Ratio	14:1	
Clad Material	Type 304 stainless steel	
Clad Thickness, in.	.010	
Fuel Plate Thickness, in.	.045	
Active Length of Each Fuel Plate, in.	23.0	
Active Width of Each Fuel Plate, in.	2.5	
Side Plate Material	Stainless steel	
Side Plate Thickness, in.	.025	
Number of Plates per Assembly	10	
Cross Sectional Size of Assemblies	3.0 in. x 3.0 in.	

	Start of Life	End of Life
Distance Between Plate Centers, in.	.30	
Number of Assemblies	37	
Total Number of Plates	370	
Fuel Assembly Weight, lbs. (approx.)	10	
Over-all Length of Fuel Assembly, in.	26	
UO_2 in Meat of Element, Weight %		
w/o Boron	18	
w Boron	20.5	
Stainless Steel in Meat of Element, Weight %		
w/o Boron	82	
w Boron	79.5	
Assembly Drawing	Figure 12	

C. Nuclear Characteristics

	Start of Life	End of Life
Core Configuration	Right Cylinder	
Equivalent Core Diameter, in.	20.6	
Equivalent Reflector Thickness, in.	4	
Enrichment, %	93	
Fuel Burnup, U_{235}, Kg, w/o Boron		1.58
w Boron		3.70
Reactivity Balance		
K, hot, operating, clean, no voids, no boron	1.125	1.129
k, boron	-.063	-.003
k, fission product	-.03	-.03
k, void	-.032	-.036
k, burnup, 3 years		-.100
k, temperature change		+.040

	Start of Life		End of Life
UO_2 in Element, Volume %	1.05		.83
Stainless Steel in Element, Volume %	15.5		
Water in Element, Volume %	83.4		
Control Strength Required		30%	
Control Strength Available with Boron Distributed in Water		30%	
D. Hydraulics			
Coolant Flow Rate, lb/hr x 10^{-5} (Total Core Flow)	7.95		8.76
Core Inlet Velocity, ft/sec	1.1		1.3
Feedwater Return Enthalpy, Btu/lb.	205		205
Steam Volume Fraction			
Average	0.068		0.081
Exit	0.161		0.195
Fuel Assembly Flow Area, in.^{2}		7.53	
Hydraulic Radius of Fuel Assembly, in.		1.175	
Fuel Assembly Flow Friction Length, in.		26	
Minimum Effective Chimney Height, ft.		4	
Average Downcomer Liquid Velocity (adjacent to chimney), ft/sec.	.8		.9
Average Steam Surface Separation Velocity, Ft/sec.	0.24		0.50
E. Heat Transfer			
Total Heat Transfer Area, ft.^{2}		364	
Heat Flux, Average Btu/hr/ft^2		31,300	
Minimum Burnout Factor		22	
Maximum Fuel Temperature, $^{\circ}F$		645	
Average Clad Surface Temperature, $^{\circ}F$	590		506

	Start of Life	End of Life
F. Hot Spot Factors		
Gross Flux Peaking		
Radial	1.39	
Axial	1.24	
Local Flux Peaking		
Variation in Enrichment and Fuel Element Dimension	1.10	
Variation in Distribution of Burnable Poison (for flattening)	1.10	
Allowance for Overpower and E uipment Malfunction or Deterioration	1.15	
TOTAL	2.39	
G. Component Design Data		
Pressure Vessel		
Inside Diameter, in.	30	
Length, in.	125	
Design Pressure,psia	1320	
Working Pressure, psia	1200	
Head Opening, in.	30	
Weight, lbs.	7000	
Alternator		
Type	Homopolar Inductor Alternator	
Speed, rpm	12,000	
Poles	8	
Freq., cps	800	
Weight, lbs. External SCR inverter required to convert 800 cps to 400 cps	1700	

	Start of Life	End of Life
Turbine		
Speed, rpm		12,000
Turbine Shaft Power, KW		380

6.3 Reactor and Internal Mechanical Design (Figure 8)

A. Reactor Pressure Vessel (Figure 10)

Vessel Size

Since the key requisites in the 300 KW plant are portability and compactness, the pressure vessel and associated components were designed with this in mind. The vessel diameter is determined by the core diameter and the width of the downcomer annulus surrounding it. The core circumscribed diameter is taken as 23 inches with a 3-1/2 inch downcomer annulus resulting in a 30 inch I.D. for the vessel. Since the chimney located above the core is only 20 inches in diameter, there is ample space about the chimney for downcomer flow. The chimney height has been set by analysis at four feet. Two feet are allowed above the chimney to the normal water level and 2-1/2 additional feet from the water level to the top of the vessel. The core length is 26 inches. Allowing circulating space below the core, the over-all inside length of the pressure vessel is 11 feet 4 inches.

Material Selection

The material selected for the pressure vessel is 304 stainless steel. An insulating vessel surrounds the reactor pressure vessel such that during reactor operation a layer of superheated steam acts as the insulator. Stainless steel, rather than carbon steel clad with stainless steel, was selected since there will be times when there will be moisture on the outer surface of the vessel as well as on the inside. A solid stainless steel vessel of this size, then, would be more economical than a carbon steel vessel with double cladding. The 304 stainless steel of the vessel matches the thermal coefficient of expansion for the core and reactor internals which are also 304 stainless steel.

Design Basis

The design code followed is the ASME Unfired Pressure Vessel Code - Section VIII, 1959 edition, with latest addenda and code case rulings which are currently in force. In order to keep the pressure vessel weight at a minimum, the higher design stress for 304 stainless steel was used, which allows a slightly increased deformation. Using the lower design stress would increase the wall thickness by 1/2 inch, resulting in a direct weight increase of over 2300 pounds for the reactor vessel alone. Increasing the pressure vessel thickness would also increase the diameter of the insulation, the containment vessel, and the outer shroud.

Reactor Vessel Data

The reactor vessel data has been compiled and tabulated under Component Design Data, Section 6.2G.

B. Insulation

In the present plant arrangement, the reactor is located in a heat sink pool. In order to minimize the heat loss to the pool, a containment vessel surrounds the reactor pressure vessel, houses the turbine-alternator, and also provides the condenser dome and condensate sump. Additional insulation which isolates the reactor pressure vessel from the condensate sump is in the form of a metal can immediately located about the pressure vessel. This metal can is attached directly to the reactor pressure vessel and vented to the condenser dome. Venting the can to the condenser provides a layer of superheated steam during reactor operation.

C. Containment Vessel (Figure 11)

The containment vessel, as mentioned earlier, surrounds the reactor pressure vessel and wholly contains the power-generating portion of the nuclear steam supply system. The condensate sump fills the lower part of the containment vessel surrounding the reactor pressure vessel, and reduces the heat flux transmitted from the reactor through the reactor vessel.

The turbine back pressure in the condenser dome is 25 psia. The containment vessel has been designed to withstand 140 psia to accommodate any accidental pressure excursions. No safeguards analysis, however, has been made to establish the criteria for pressure containment. The containment vessel thickness was made on the basis of minimizing plant weight and still maintaining a significant safety margin. The containment vessel thickness, then, is subject to change after accidental pressure buildup has been evaluated and reviewed by safeguards.

D. Core Internals

Fuel (Figures 8 and 12)

The fuel design was based primarily on the physics and core design information tabulated in Section 6.2B. The fuel material is fully enriched UO_2 dispersed in stainless steel, with stainless steel cladding. The stainless steel used is type 304. The over-all dimensions of each fuel assembly are 3 in. x 3 in. x 26 in. There are 37 fuel assemblies arranged in a 5 x 5 array, with 3 located on each side.

Chimney

The chimney height was set at four feet based on an analysis given in Section 6.4B. The inside diameter is 20 inches, and the wall thickness is 1/8 inch. The material used is type 304 stainless steel.

Core Support Arrangement

The core support is integral with the reactor pressure vessel, and supports a load of less than 500 pounds. This is the combined weight of the fuel and chimney. The core support frame, surrounding the core, rests on the vessel core support. Mounting pins are attached to the core support frame which guide the chimney into place onto the core support and maintain its position.

E. Reject Heat Disposal System

The reject-heat disposal system condenses the turbine exhaust steam, and subcools the water to provide sufficient inlet suction head for the circulating pump. The major components of this system are the spray condenser and the external heat exchanger. A schematic flow diagram is given on Figure 13. Steam from the turbine is condensed in a dome by a spray of cold water. This water flow is that required to just condense the steam at the desired turbine exhaust pressure. The water is then subcooled by mixing with additional cold water. The total water flow then passes through the circulating pump. From the circulating pump discharge, the flow divides, with the major portion passing through the external heat exchanger and the remainder to the feed pump suction. The external heat exchanger transfers heat by free convection to the sea or pond in which the unit operates, which may be either fresh or saline water. Upon leaving the heat exchanger, the water is split into two flows. One goes to the spray condenser, and the second is that used for subcooling.

This section of the report discusses the physical requirements of the spray condenser and external heat exchanger. Several schemes for providing a heat sink for the field plants are described in Section 6.10, Heat Sink Requirements.

Spray Condenser

The spray condenser is essentially a dome into which the turbine exhaust steam and a spray of cooled water are mixed so that the steam is entirely condensed. The characteristics which must be provided for in the design are mainly the total volume of the dome, the water droplet size, and steam volume flow.

Cooling water flow rate has no significant bearing on dome size, since the volume of water is small compared with that of the steam at condensing conditions. It does affect the detailed spray nozzle design, but the effects will not appreciably affect the cost of the nozzles. The prime factor required of the cooling water is that it be delivered at sufficiently high pressure to produce an acceptable droplet size. The 50 psi pressure head, available from the circulating pump, should be able to produce droplet diameters under 500 microns.

Figure 14 shows theoretical dome size requirement versus steam volume flow and water droplet diameter. For a steam mass flow rate of 12,000 lbs/hr, the volume flow would be 46 cfs at a pressure of 30 psi to 157 cfs at 8 psi. Applying a factor of two in droplet diameter for pessimism, the figure may be entered at 1000 micron droplet size, and these volume flows to obtain dome volumes of 18 cubic feet for 30 psi and 67 cubic feet at 8 psi. Thus, theory says that steam pressure is very important in sizing the dome, that any future changes to an exhaust pressure less than 30 psi would require a significant increase in dome size.

An alternate approach to sizing would be to use a lump value heat transfer rate evolved from a range of experiments. A value of 600,000 Btu/hr-ft^3 is quoted in the literature, so this might be reduced by a factor of two to provide design margin. For a total heat rate of 12 million Btu/hr, this approach gives a required dome size of 40 cubic feet, which is in the same general range as the theory.

It is recommended that a design volume of 40 cubic feet be used so long as exhaust pressure stays to 25 psi or more. Such a volume should provide adequate space based both on theoretical and experimental data. A circulating pump discharge pressure use of 50 psi should be adequate to obtain useable droplet diameters.

<u>Sea Water Heat Exchanger</u>

Heat transfer area and total volume are the major design factors determining cost and layout of the heat exchanger. The number of factors having potential technical effect is much greater than for the spray condenser. The salient factors may be grouped for discussion as follows:

1. <u>Factors set by reactor and turbine</u>

    a. Heat rate
    b. Condensing temperature

2. <u>Factors set by circulating pump</u>

    a. Total circulating flow rate
    b. Sump temperature

3. <u>Factors set by heat sink</u>

    a. Temperature
    b. Scaling and corrosion

Currently, the nominal heat rate is around 12 million Btu/hr. While future iterations may alter this value, the probable variation is small with respect to the possible effects of other items.

Condensing temperature currently is at 250° F, corresponding to a pressure of 30 psia. It has been as low as 183° F, corresponding to a pressure of 8 psia. This temperature represents the maximum hot-side temperature available to the heat exchanger if no subcooling is required. However, some subcooling will be required to provide a reasonable inlet suction head for the pump. Low sump temperature is obtained, however, only at the expense of low heat exchanger discharge temperature or large water flows. Figure 15 shows the relationship between the flows and the temperatures. Condensing, subcooling, and circulating flows are plotted versus sump temperature for lines of constant heat exchanger discharge temperature. The external heat transfer coefficient is small compared to the interior, so a nominal heat flux can be calculated by assuming 3 feet per second for interior flow and free convection externally. Figure 16 shows this heat flux plotted versus over-all temperature differential. Using a log mean temperature difference with a mean water temperature, heat transfer area can be computed.

Since low heat exchanger discharge temperature means large surface area and large water flow means large inlet suction head, a minimum heat transfer area occurs for a maximum inlet suction head. Figure 17 shows heat transfer area plotted versus sump temperature for lines of constant heat exchanger discharge temperature and circulating water flow. Included also is a limit line for pump inlet suction head of 12,000 for a shaft speed of 12,000 and a static head of 3 feet. This line reaches a minimum at the following conditions:

Heat transfer area	1,300 ft^2
Circulating flow	190,000 lbs/hr
Heat exchanger discharge temperature	128° F
Sump temperature	193° F

From Figure 17:

Spray flow	85,000 lbs/hr
Subcooling flow	95,000 lbs/hr

This analysis has been performed for specific values of pump speed and inlet suction head and water temperature. However, it can be stated that any forseeable changes would not alter the required heat transfer area beyond the range of 1000 ft^2 as a minimum or above 2000 ft^2 as a maximum.

For a specific heat transfer area and pipe size, the physical volume of the heat exchanger depends upon the allowable water temperature use on the sea water side. A tube outside diameter of 1-3/8 inches appears satisfactory. Figure 18 shows required tube spacing versus sea water temperature rise for heat transfer areas of 1000 and 2000 ft^2, and for tube bundle heights of 10 and 20 feet. Also shown are the outside diameters of the tube bundle for a nominal inner diameter of 4 feet. A temperature rise of 10° F could be tolerated, but the small effect on size would indicate 5° F to be a good design value.

The orientation of the tubes with respect to vertical is open to question. The external flow will be in the turbulent range, indicating some gain with horizontal or inclined tubes. However, the eddies so resulting may aggravate chemical and biological fouling. Thus, it is advised that inclination of the tubes from vertical only be used where fouling problems will not be aggravated.

The heat transfer rates and free convection mass flow rates are calculated for clean tubes. Since the free convection heat transfer coefficient is already quite low, application of nominal fouling factors does not seriously alter the required heat transfer areas. However, exposure to untreated water can cause serious accumulations in specific locations, which would have to be met in evaluation of any actual potential site. A possibility is to completely treat small ponds and, for large seas, to build a separating dike to allow treatment of a finite volume (see Section 6.10).

6.4 Reactor Thermal and Hydraulic Analysis

A. Introduction

The reference design is for a natural circulation reactor. As such, it depends on the balance between the available density head gain and the pressure losses in the system. Therefore, the most important thermal-hydraulic factors to be determined in this reactor are the flow rates, void fractions, and exit steam qualities in conjunction with core size and chimney height. It is also necessary to determine the maximum fuel temperature and minimum boiling burnout margin in conjunction with an estimate of hot spot factors to establish that none of the design limits have been exceeded. By maintaining the largest reasonable margin between design values and design limits, it should be possible to maintain the highest reliability of the plant.

B. Core Pressure Drop and Chimney Height

The objective of this investigation was to determine the reactor flow rate, void fraction, and exit steam quality for the chosen system at a thermal rating compatible with the power conversion equipment. Preliminary scoping (Reference 5) provided a reactor size and chimney height. The analyses shown below describe the method used in the sizing and in the final determination of the fluid flow characteristics of the core.

The hydraulic components of this reactor are the core, the chimney, and the downcomer. The core was made up of 37 elements with 10 plates per element, each having a uniform flow area.

The chimney might be characterized as a stove pipe, or cylinder, 4 feet in length and approximately 20 inches in diameter. The downcomer is an annulus with an inside diameter of approximately 20 inches and an outside diameter of 30 inches. Since the downcomer surrounds both the core and the chimney, it will be assumed to be 6 feet long.

The water enters the core slightly subcooled, passes through the channels between fuel elements, and exits as a two-phase steam-water mixture. In going through the core, it will be subjected to an entrance loss, an exit loss, a friction loss, and an acceleration or momentum change. The Martinelli-Nelson two-phase pressure loss multipliers were utilized to determine the friction losses for the two-phase mixture. The pressure losses in the chimney and downcomer were similar to the above, and were handled in the same manner.

To determine the average and exit void fractions in the core, the data of Dingee, Egan & Chastain (Reference 6) and of W.H. Cook (Reference 7) has been investigated. By interpolating between the interpretation of the results of these two investigations, it was possible to obtain a preliminary estimate of the void fraction-quality-heat transfer relationship at low quality and over a range of pressures. Spot checks of the results indicated good agreement between the void fraction-quality relationship used and the relationship proposed by W.W. Goodwin in NCR Thermal Design Memorandum #3.

The local quality-void fraction relationships which were used are shown in Figures 19, 20, and 21. The over-all average core steam volume fraction was estimated with the above data, using the approximate relation:

$$\bar{\alpha}_o = 1/2 \left[\frac{L_{NB}}{L} \alpha_s + \frac{L_B}{L} \left(\alpha_s + \alpha_e \right) \right]$$

where:

$\bar{\alpha}_o$	=	over-all average core steam volume fraction
α_s	=	exit steam volume fraction at exit steam quality, Xe (Figure 19)
L	=	core active zone length
L_{NB}	=	average length of core where subcooled nucleate boiling occurs
L_B	=	average length of core where bulk boiling exists

Preliminary calculations indicate that in all cases, the fuel surface temperature with no nucleate boiling would be in excess of the maximum liquid superheat temperature. Therefore, it would seem likely that subcooled nucleate boiling would commence immediately after the core inlet. It was assumed that

$$L_{NB} = L - L_B$$

The calculations were made for an excess steam flow of 5% for control, and an allowance of 50% was made in the available driving head provided by the chimney. That is, flow rates, void fractions, and steam qualities have been determined for a chimney height of 2.67 feet with a recommended height of 4 feet to allow some margin for inaccuracies in the predicted void fractions and pressure losses.

A summary of the basic thermal-hydraulic results is given in the following table.

TABLE V
Summary of Thermal-Hydraulic Data

	Start of Life	End of Life
Pressure	1200	600
Reactor Power, KW	3340	3340
Feedwater Return Enthalpy, Btu/lb.	205	205
Total Core Flow, lb/sec.	221	243
Bulk Exit Steam Quality	.0137	.0131
Steam Volume Fraction		
Core Exit	.161	.195
Over-all Average	.068	.081
Chimney Height, ft.	4	4

C. Hot Spot Factor

The foregoing thermal design was based on an assumption of uniform heat generation and of uniform flow in the reactor core. In reality, variations in the power generation, both axially and radially, and variations in fabrication, plus uncertainties in analysis, would not allow completely uniform cores. To be sure that the highest temperature location in the hottest channel did not exceed design limits, a series of hot spot factors were determined which expressed numerically the effect of the above variations.

The hot spot factors in a boiling water reactor are usually divided into three categories. These are:

1. Gross flux peaking
2. Local flux peaking
3. Overpower allowance

The gross peaking hot spot factor is the product of the peak-to-average flux distributions in the radial and axial directions. Low values (see Table VI) are obtained in this reactor because of the greater flattening effect of the reflector on such a small core. As a supplement to the inherently low peak-to-average ratio, it has been proposed that burnable poisons be distributed to flatten the core even more. This brings up one of the considerations in local flux peaking. Since burnable poison will be utilized to flatten the core, the absence of burnable poison on a fuel element could provide a spike or peak in flux. Therefore, a local flux peaking factor of 1.10 is assigned to the variation in distribution of burnable poisons for flattening.

Similarly, allowance was made for the possibility of variation of enrichment and of fuel element dimensions. A concentration of fuel in a given location could have the same effect as the omission of burnable poison; i.e., it could provide a spike in the thermal flux.

The two factors considered above comprise the local flux peaking factor for this design.

The last hot spot factor provides an allowance for overpower operation. In this reactor, such operation might result from malfunction or deterioration of the plant equipment.

TABLE VI
Hot Spot Factors

Gross Flux Peaking	
Radial	1.39
Axial	1.24
Local Flux Peaking	
Variation in Enrichment and Fuel Element Dimensions	1.10
Variation in Distribution of Burnable Poison (for flattening)	1.10
Allowance for Overpower and Equipment Malfunction or Deterioration	1.15
TOTAL	2.39

Transient Behavior

This system is designed to operate at constant load. External means are used to maintain a constant electrical load on the turbine-generator regardless of the demands of the active load on the system. Because of this, the reactor should operate always at near full load. The regulation of this position will be such that the 5% bypass flow will provide sufficient extra power for variations resulting from changes in the actual load-dummy load split and for variations resulting from reset of the pressure regulator, etc.

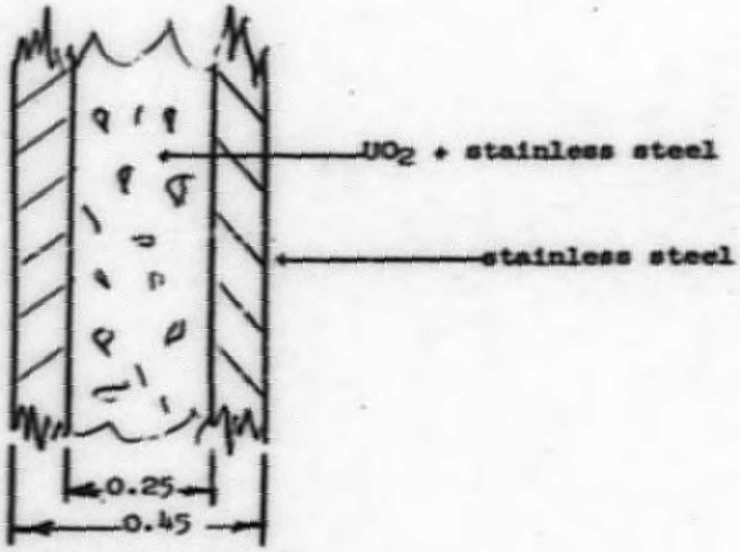
Realizing the above, it becomes evident that any significant transient behavior can result only from an accident, and should be considered in such context.

D. Fuel Temperature

Because of the low heat transfer rate, q/A, employed in this reactor configuration, it would not be expected that the fuel temperature would limit the reactor. Nevertheless, it is desirable to make a preliminary evaluation of the temperatures to verify this belief.

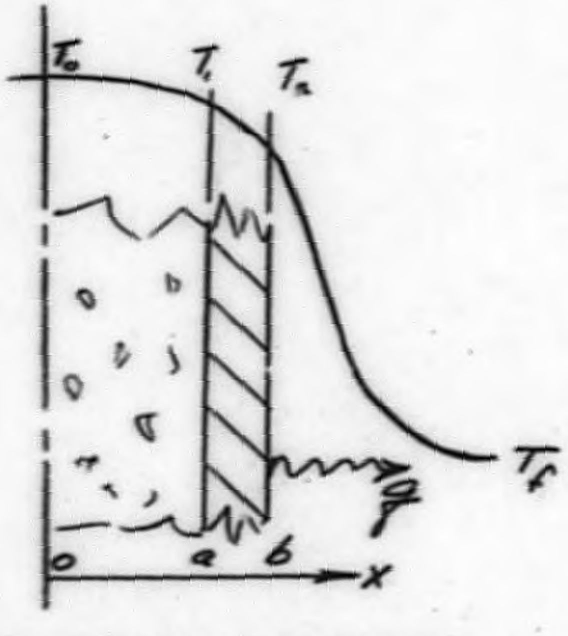
The fuel element in the subject reactor consists of a slab of stainless steel impregnated with UO_2 (uranium dioxide) and clad on each side with a thin sheet of stainless steel.

A sketch of the element is shown on the following page (see also Figure 12).



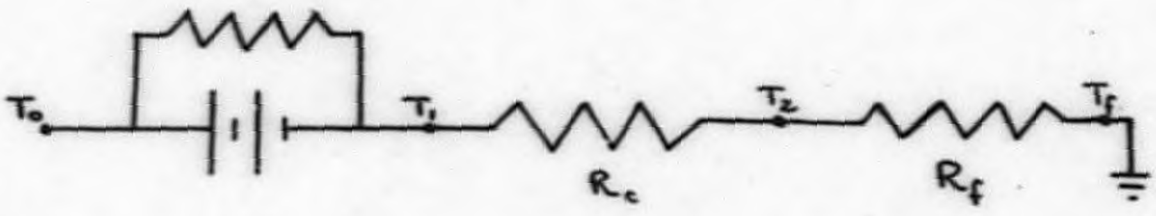
PROPOSED FUEL ELEMENT

Neglecting any axial conduction and all end effects, it is noted that the element is symmetrical around its centerline and may be represented as shown below.



HALF SLAB REPRESENTATION OF FUEL ELEMENT

The foregoing may be represented as a one-dimensional thermal circuit, made up of a source (the fuel), a thermal resistance from the cladding, and a thermal resistance resulting from the film.



THERMAL CIRCUIT REPRESENTATION

The above assumes that none of the resistance results from the interface between the fuel and the clad.* If desired, this resistance could be added between the fuel and clad, and would merely be a linear addition. In the fuel region, the general equation for steady state in a system becomes

$$-k \nabla^2 t = Q (\bar{r}) \tag{1}$$

and for a one-dimensional system, such as is given here, this becomes

$$-k \frac{d t}{dx^2} = Q (x), \; 0 \leq x \leq a \tag{2}$$

Assuming that the internal heat source is uniform, the general solution becomes

$$t = -\frac{Qx^2}{2k} + C_1 x + C_2 \tag{3}$$

* This is consistent with the current practice.

Applying the boundary conditions

$$\frac{dt}{dx} = 0 \text{ @ } x = 0;\ t = t, \text{ @ } x = a$$

which yields the solution

$$t - t_1 = \frac{Q}{2k} (a^2 - x^2) \tag{4}$$

Solving for $t_{max.}$, let $x = 0$ and

$$(t_{max.} - t_1) = \frac{Q}{2k} a^2 \tag{5}$$

The thermal conductivity of the UO_2-stainless steel cermet may be approximated by considering the volumetric contribution of UO_2 in the stainless steel. The volumetric ratio of stainless steel to UO_2 is approximately 14:1; therefore

$$k_{fuel} = 9.5 \text{ Btu/hr-ft}^2 \text{ °F/ft.}$$

The source term, Q, may be approximated by assuming uniform heat generation, and thereby using the relationship

$$Q(\bar{r}) = (q/A) \left(\frac{Area}{Vol}\right)$$

$$= (31,300) \frac{(3.0)\ (2)}{2.5(.025)} (12)$$

$$= 3.6 \times 10^7 \text{ Btu/hr-ft}^3$$

and

$$a = \frac{.045}{2(12)} = 1.875 \times 10^{-3} \text{ ft.}$$

The temperature drop across the fuel might therefore be found to be

$$(t_{max.} - t_1) = \frac{Q}{2k} a^2$$

$$= 6.7 \text{ °F}$$

Cladding

The thermal resistance in the cladding is given by

$$R_c = \frac{b-a}{k_c}$$

$$= 8.25 \times 10^{-5} \text{ hr-ft}^2 \ {}^\circ\text{F/Btu}$$

Film

The resistance in the film is given by

$$R_c = 1/h$$

where the conductance, h, may be taken to a conservative estimate as the conductance for turbulent flow with saturated water; i.e.,

$$h = 0.023 \, R_e^{\ 0.8} \, Pr^{\ 1/3} \, \frac{k}{D_e} \qquad (6)$$

where as previously determined

Re* = 72,100

Pr* = .96

k* = .314

De = .0392 ft.

Substituting into (6) above yields

$$h = 1350 \text{ Btu/hr-ft}^2 \ {}^\circ\text{F}$$

and

$$1/h = 7.4 \times 10^{-4} \ {}^\circ\text{F hr-ft}^2/\text{Btu}$$

Assuming that the maximum heat flux is 2.39 times the average heat flux

$$(q/A)_{max.} = (\text{hot spot factor}) \ (q/A)$$

$$= 74{,}800 \text{ Btu/hr-ft}^2 \ {}^\circ\text{F}$$

* (Reference 8)

Then the temperature drop across the fluid film and the cladding would become

$$t_{max.} = (q/A)_{max.} \, R_{total}$$

and $\quad t_{clad} = 6.2\ ^{o}F$

with $\quad t_{film} = 55.4\ ^{o}F$

The maximum temperature drop in the fuel would become

$$(2.39)\ (6.7) = 16.0^{o}F$$

The temperature difference would therefore be

$$t_{max.} = 55.4 + 6.2 + 16.0 = 77.6\ ^{o}F$$

$$t_{fuel,\ max.} = 567 + 77.6 = 645\ ^{o}F$$

It may, therefore, be seen that the maximum fuel temperature is quite low and well below all design limits.

Boiling Burnout

Boiling burnout is characterized by the rather abrupt formation of a thin insulating blanket of steam over the heat transfer surface which causes a large increase in thermal resistance. This phenomena causes excessive overheating to the point that the clad of the fuel element is melted.

Unlike most boiling water reactors, this reactor should not be limited by the phenomenon of boiling burnout. An extremely low average heat flux, plus a very low exit quality, cause this reactor to have heat transfer characteristics more like those of a single-phase convective system than like a boiling or two-phase system. The expected exit quality of this system will be of the order of 1%, and the average heat flux approximately 30,000 Btu/hr-ft^2. Extrapolating the reference burnout design line (Reference 9) to a quality of 1%, it is seen that the burnout heat flux would be about 3.1×10^6 Btu/hr-ft^2. With uniform heat flux and no hot spots, the average burnout margin would be about 100. Considering the highly improbable event that all of the malfunctions described in the hot spot factor would occur simultaneously, the maximum heat flux might be 74,800 Btu/hr-ft^2, and the exit quality would become 0.034. Again using the design line referred to above, it is seen that the burnout margin would be 18.

F. Pressure Level, Steam Quality, and Inlet Subcooling

One of the unique aspects of this reactor is that it operates over a range of pressure. Identification of the pressure range was the result of an optimization of several factors. These are presented below.

1. As the pressure, and therefore the temperature, increase, the slope of k_{eff} versus temperature becomes steeper. This indicates that a given range of pressure would provide much more Δ k if the range were at high pressure than if the range were at low pressure.

2. As pressure is raised, the thickness of the vessel wall is increased, and therefore reactor vessel weight is increased. This provides a restrictive limitation on maximum pressure. To indicate the order of magnitude involved, an 80 psi increase in pressure to 1280 psia increases the weight 600 lbs.

3. Minimum pressure has been primarily a judgment point. A conservative choice of 600 psi has been made for the lower limit of reactor pressure.

4. The turbine for this plant is designed to operate at a constant inlet pressure, and to have a constant back pressure in the condenser. To obtain constant turbine inlet pressure, a throttle valve must be placed upstream of the turbine. Wet steam, therefore, will be introduced to the turbine for pressure greater than 450 psi. As the pressure is increased, the inlet moisture will increase and therefore the exit moisture will increase.

5. As pressure in the reactor system is increased, the complexity of the feedwater pump increases. The design of the feedwater pump is limited to about 1320 psi at rated flow conditions.

From consideration of the above advantages and disadvantages, a pressure range of 1200 to 600 psi was selected.

Steam quality has been determined by core size, inlet subcooling, thermal rating, and steam flow requirements. The design exit steam quality is considerably lower than any currently designed boiling water reactor. The primary reason for this is to obtain the maximum change in k_{eff} from a change in moderator temperature.

The design variation in average steam exit quality with life is only 0.06 of 1%. However, an allowance for variation in flow, as provided by the longer than required chimney, might allow for a 0.41% variation in average exit quality. It was shown in Section C that maximum steam quality might be as high as 3.4%.

Inlet subcooling is primarily determined by the condenser pressure. Preliminary heat balance results show that heat losses in the condenser and gains in the pump are approximately equal; therefore, as condenser pressure is decreased, inlet subcooling is increased and average void fraction goes down. The design condenser pressure was fixed by the required circulating pump back pressure. A back pressure of 25 psia was required at a pump shaft speed of 12,000 rpm. Condenser pressure might be decreased if a separate circulating pump were utilized. Though such a maneuver would increase system efficiency, and therefore life, it would add another moving component to the system.

6.5 Physics Analysis

A. Introduction

Physics analysis of this reactor core has emphasized the determination of the proper core geometry and design to provide 3300 KW of thermal energy for the longest possible core life. With burnable poisons incorporated in the core operating at 3.3 MW, the calculated lifetime is 2.3 years, whereas the non-borated core is limited to one year.* The present core of 37 stainless steel clad, plate-type elements containing a mixture of fully enriched UO_2 powder and stainless powder "sandwiched" between stainless plates represents the final choice.

Each element measures 3" x 3" x 24", and is composed of 10 plates, each .045" in thickness. Each plate contains an active fuel region .025" in thickness by 2.5" wide by 23" long.** The fuel elements were chosen because of the high exposures experienced with prior elements of this type (similar elements have operated in VBWR with burnups in excess of 30% U^{235}) and the high resistance of stainless steel to corrosion. Once the number of elements had been determined that permitted symmetric core geometry, then reactivity change with change of moderator density dictated the maximum core size permitted (consistent with heat transfer requirements). It was recognized that core size was very important in obtaining the desired change in reactivity, and that a small core was more satisfactory.

Two control aspects have been investigated. These are startup and prototype scram. For startup, a soluble reactor poison in the water of the primary system will hold the core subcritical. The poison in solution would be removed through use of system deionizers.

To scram the prototype, the nine center elements are arranged so that they may be lowered from or raised into the core. Investigations show the reactor to be just subcritical (K = 0.97) when cold and clean with the center nine elements removed. More shutdown margin is provided in the hot operating condition with these same elements removed.

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* Further design optimization should yield a three-year life. See Section 6.12 for discussion of potential for increasing plant life.

** Two loadings were investigated: 13.75 kg fully enriched UO_2 (12.1 kg U^{235}) for a core without boron, and 16 kg fully enriched UO_2 (14.1 kg U^{235}) for the core with burnable poison.

The nuclear characteristics of the proposed 300 KWe core have been determined for the expected operating conditions of temperature, pressure, and void content. The reactor has been designed to operate for the length of its life without attention and outside control at a constant power rating of 300 KWe. There were a number of design methods by which this concept could be realized. In a small core, reactivity is quite sensitive to moderator temperature changes because of high neutron leakage. These changes are more important at higher temperatures. This effect, coupled with burnable poisons in the fuel, permits the use of no control rod and the attainment of reasonably long life.

The Δ k gain due to moderator temperature difference during core lifetime is .04. Burnable poisons are used to incorporate 6.3% in k_{eff} at beginning of life to realize the two to three year lifetime.

Characteristics of the core were then obtained. It is important to the safety and long life of the unattended reactor that temperature and void excursions tend to be self-limiting. Temperature and void coefficients were found to be strongly negative in all cases from cold to hot operating conditions.

B. Calculational Techniques

Some of the core reactivities are calculated using a one-dimensional diffusion theory code (WANDA for the IBM-704), using three neutron energy group cross sections consisting of two epithermal group cross sections (condensed from 54 epithermal group cross section files by the MUFT-4 code), and thermal group cross sections obtained by means of the Wilkins formula for thermal spectrum averaging. However, most of the core reactivities were obtained by application of a neutron leakage term to k of the system. K_{∞} was tabulated from three group cross sections obtained in the manner just mentioned as follows:

$$k_{\infty} = \frac{\nu_1 \Sigma_{f_1}}{\Sigma_{SD_1} + \Sigma_{a_1}} + P_1 \frac{\nu_2 \Sigma_{f_2}}{\Sigma_{SD_2} + \Sigma_{a_2}} + P_1 P_2 \frac{\nu_3 \Sigma_{f_3}}{\Sigma_{a_3}},$$

where

ν_i = neutrons liberated/fission from group i into group 1

Σ_{f_i} = macroscopic fission cross section in group i

Σ_{SD_i} = macroscopic slowing down cross section from group i into group i + 1

Σ_{a_i} = macroscopic absorption cross section in group i

P_i = resonance escape probability from group i

Then, using these cross sections in a diffusion theory case, buckling, B^2, for the reactor was found for different reactor temperature conditions by backing it out of diffusion theory calculations. K_{eff} was found by means of the following method:

$$k_{eff} = \frac{k_{\infty}}{(1 + B^2 \tau)\ (1 + B^2 L^2)}$$

τ = fermi age

L^2 = thermal migration area

Core reactivities found by this approximation were normalized to a diffusion theory case for the identical cross sections, and this normalization was applied to all similar cases.

C. Burnup

Analysis of the one region core with its uniform boron and fuel distribution lent itself ideally to an application of perturbation theory.

Since the reactor has as its variable with time the density of the system water moderator, the density change with time was the dependent variable. If the parameters of the system - cross sections and diffusion constants - were made functions of water density, then the perturbation integral equation could be made entirely an equation in water density change. Each nuclear parameter was represented as $\Sigma = A + B\rho$, where ρ = water density. The constants were chosen such that the equation represented the value of Σ at the beginning and end points of life. Hence, a change in Σ with water density was represented

$$\frac{\Delta\Sigma}{\Delta T} = B \frac{\Delta\rho}{\Delta T}, \text{ where } \Delta T = \text{temperature interval}$$

Two neutron energy group perturbation analysis then involved obtaining actual neutron fluxes in two groups and their adjoint fluxes. The diffusion equation for two energy groups may be written in the matrix form:

$$\nu_c J \Phi = K \Phi$$

where $\phi \equiv \begin{pmatrix} \Phi_1 \\ \Phi_2 \end{pmatrix}$, $K = -\begin{pmatrix} \nabla \cdot D_1 \nabla - \Sigma_r & 0 \\ 0 & \nabla \cdot D_2 \nabla - \Sigma_{r_2} \end{pmatrix}$, $J = \begin{pmatrix} 0 & \Sigma_F \\ 0 & 0 \end{pmatrix}$

The eigenvalue ν_c represents the number of neutrons per fission required to maintain criticality. Since the actual number of neutrons per fission of U^{235} is found experimentally to be 2.46, a value of ν_c less than this represents a supercritical condition.

If the core parameters are changed, the new critical equation becomes:

$$\mathcal{V}_c' \quad J' \quad \phi' \quad = K' \quad \phi'$$

The change in the eigenvalue can be expressed in the following form:

$$\Delta\mathcal{V} = \frac{\int \Phi^+ \Delta K \Phi \, dv - \mathcal{V}' \int \Phi^+ \Delta J \Phi' dv}{\int \Phi^+ J \Phi' dv}$$

where

$$\Delta\mathcal{V} = \mathcal{V}_c' - \mathcal{V}, \quad \Delta K = K' - K, \quad \Delta J = J' - J$$

and ϕ^+ is the adjoint flux in the unperturbed reactor. The value of ϕ and ϕ^+ are respectively the solutions of the two-group diffusion equation and its adjoint.

Then, in our case, $\Delta\mathcal{V}$ was set equal to 0, and ϕ' (flux in the perturbed system) assumed to be equal to ϕ.

This results in the equation involving flux-flux adjoint integrals:

$$\Delta\mathcal{V} = 0 = \int (\delta\Sigma_a) \Phi_2^+ \Phi_2 \, dv - \mathcal{V}' \int \Phi_1^+ \Phi_2 (\delta\Sigma_f) \, dv +$$

$$\int (\delta\Sigma_{sD}) \left[\Phi_1^+ \Phi_1 - \Phi_2^+ \Phi_1 \right] dv + \int \frac{\delta D_2}{D_2} \left[\Phi_2^+ \Phi_1 \Sigma_{sD} - \right.$$

$$\left. \Phi_2^+ \Phi_2 \Sigma_a \right] dv + \int \left(\frac{\delta D_1}{D_1} \right) \left[\Phi_1^+ \Phi_2 \, \nu \Sigma_{f_2} - \Phi_1^+ \Phi_1 \Sigma_{sD} \right] dv$$

Flux normalization was achieved by assuming 1.25 grams U^{235} burned per megawatt-day at a power of 3.3 megawatts.

Burnup estimates then showed a lifetime of 12.0 months operating at 3.3 MW with no boron in the core, and 27.6 months for the core incorporating 6.3% in k in boron at the beginning of life and also operating at 3.3 MW.

D. Temperature and Void Coefficients

The reactor shows throughout its lifetime negative temperature and void coefficients. In the operating range of 565° F to 485° F, the void and temperature coefficients are quite substantially negative, and will tend to be a limit to any temperature and power excursions the reactor might encounter.

Temperature coefficients (k/k/°F)for various reactor conditions are as follows:

Temperature and Void Fraction:	k/k/°F
100° F, 0%	-1.70×10^{-5}
300° F, 0%	-1.47×10^{-4}
565° F, 7.3% operating	-1.22×10^{-3}

Void coefficients are more constant, and are as follows:

Temperature:	$\Delta k/k/\%$ voids
65° F	-2.25×10^{-3}
300° F	-2.74×10^{-3}
565° F	-4.32×10^{-3}

E. Description of Accompanying Graphs

Figure No. 22 shows k_{eff} versus temperature for a clean 13.75 kg fully enriched UO_2 (11.2 kg U^{235}) loading core, incorporating no burnable poison. Leakage core reactivities are plotted, and are normalized to reactivities from diffusion theory. The graph shows the strong dependence of k_{eff} on temperature (water density at that saturated temperature). The temperature coefficient for various conditions is given also.

Figure No. 23 is a cross plot of the values in Figure No. 22 showing k_{eff} versus moderator density, where the density of the moderator is that of water at saturated pressure without voids.

Figure No. 24 shows k_{eff} versus percent voids in the moderator at three different temperatures. A table of the void coefficient ($\Delta k/k/\%$ void) is also given.

Figure No. 25 shows the effect of fully enriched UO_2 content on k_{eff} for a hot (565° F), clean core. Diffusion theory core reactivities are graphed, and none include boron.

Figures 26, 27, 28, 29, and 30 show three group fluxes as a function of distance from the center of the core. Figures 26, 28, and 29 are plotted on a semi-logarithmic scale, and 27 and 30 show fast and thermal fluxes on rectangular coordinate paper. It should be mentioned that for points more than a few inches away from the edge of the core, the fluxes become approximate, as the assumption has been made that the reactor may be represented by an infinite cylinder with proper axial buckling yielding the correct spectrum. This is not exactly true, but the flux values given are certainly correct to within an order of magnitude. The flux values are given for a thermal power rating of 3.3 MW. The radial peak-to-average (thermal flux) equals 1.39.

Figure No. 31 shows the axial flux distribution for the same core as above, and is subject to the same limitation of actual size versus mathematical representation. Axial thermal flux peak-to-average is 1.24. A term $\frac{(\int \Phi^3)}{(\int \Phi^2)}$ reflecting non-uniform burnup in the axial direction which is important in perturbation theory burnup of the core is equal to 1.066.

Figures No. 32 and 33 show adjoint fluxes for two groups in the core and reflector. Adjoint fluxes were used in the perturbation theory burnup prediction for the lifetime of the core, and tend to be an indication of the importance of the corresponding real neutron to the neutron economy or reactivity of the core.

Figure No. 34 provides a burnup prediction for the lifetime of the non-borated 13.75 kg UO_2 core with plot of reactor system pressure versus core lifetime. Perturbation of the core shows that the reactor will fall in pressure below the design minimum in 12 months of 3.3 MW thermal power output operation. The sharp decrease in pressure in the first few weeks is due to samarium poison buildup. This might be compensated by maintaining some boron in solution the first few weeks.

Figure No. 35 is a prediction of the lifetime of the core, and shows reactor system pressure versus core lifetime for the reactor core with burnable poison. 6.3% in $\Delta k/k$ is incorporated in boron at the beginning of life and 4.0% $\Delta k/k$ accrues from temperature change during core lifetime. At 3.3 MW, 27.3 months core lifetime is predicted.

6.6 Control and Transient Performance

The direct cycle boiling water reactor is proposed for this application. In this section of the report, control systems for this cycle are described - this includes both existing systems on operating plants and proposed new concepts. The reference control concept described is a minimum control system that requires periodic attendance; the advanced control concept described is a more complete control system that requires no attendance. Both concepts are feasible. Briefly, both concepts use a positive (as opposed to inherent) method of controlling reactor pressure. Reactor power is maintained by dropping reactor pressure on a definite schedule over the three-year life of the core; by dropping pressure, the moderator temperature is decreased, and positive reactivity is added through the negative temperature coefficient to offset the loss of reactivity caused by fuel burnup. In the minimum control system, pressure is dropped periodically by an operator. In the unattended control system, pressure is dropped automatically.

A. BWR Operation and Control

Nine boiling water reactors (BORAX I, II, III, IV, SPERT I, EBWR, VBWR, SL-1, Dresden) have been designed, tested, and operated successfully. With the exception of Dresden, each of these reactors employs the direct cycle; i.e., steam flow from the reactor flows directly into the turbine-condenser. In Dresden, steam flows directly from the reactor, and indirectly from steam generators in the recirculation flow downcomers in a unique dual cycle system.

Control of these BWR plants is in most ways comparable to the control of conventional power plants. Positive methods are used to control the power generated, system pressure, and the turbine load. The major difference is in the core; wherein fuel flow controls power in a conventional plant, reactivity controls power in a nuclear plant. In either plant, pressure must be controlled by matching the power generated to the load absorbed.

In a BWR, reactivity is a function of the control rods, enrichment, void fraction, xenon-samarium poisoning, a burnable poison, moderator temperature coefficient, and Doppler coefficient. In this unattended, long life capsule reactor, the core size, fuel burnup, and the core lifetime dictate the use of burnable poisons and full enrichment. Full enrichment practically rules out the Doppler effect. Burnable poison is a slow, long term effect. No control rods are used because of their complexity and their incompatibility with the ultimate goal of a simple, unattended, inherently controlled reactor design. Xenon-samarium poisoning is inherent in the fuel; it is not an independent reactivity function available for control.

Thus, the designer is left with the void and moderator temperature coefficients to control reactivity and thereby reactor power. By design (for safety and control), these coefficients are negative.

B. Void Reactivity

First, let us consider the void coefficient. At a constant pressure, the variation between reactivity in voids and power is definite; in the power range, a specific increase in core reactivity will result in a specific change in void reactivity and power level. During steady-state operation, the core reactivity is balanced by the reactivity in voids. To increase power, core reactivity is increased and then power and steam voids increase until the negative void reactivity cancels the positive core reactivity.

This fraction of steam voids in the core is a major parameter affecting control of reactor power, and an understanding of its effect on reactor performance is essential. From a given steady-state power level with the reactor exactly critical, a decrease in voids will increase reactivity and power. This power increase will continue until the net core reactivity is returned to zero by an increase in voids or the insertion of some other form of negative reactivity. Conversely, an increase in voids will decrease reactivity and power. This power decrease also will continue until the net core reactivity is returned to zero.

The void fraction can be changed by reactor power, pressure, subcooling, core flow rate, and a number of secondary quantities which must first affect these primary quantities. All of the quantities are coupled closely in the reactor and its thermal system, but for ease of explanation, they will be considered separately.

An increase in reactor power increases voids, and so causes a decrease in reactivity and power. This decrease is the reason for the term "negative coefficient of reactivity."

A pressure increase decreases voids, and causes power to increase. Thus, pressure has a positive effect on reactor power.

Subcooling, the difference between saturation enthalpy and the enthalpy of the core inlet flow, affects voids by changing the boiling boundary. An increase in subcooling raises the boundary, decreases voids, and increases power.

The core flow rate also affects the boiling boundary. Increasing flow raises the boundary, decreases voids, and increases power.

In all cases, a reversal in the power change can be explained by an opposite set of events. An important characteristic to grasp from this discussion is that the BWR will not necessarily have an inherent load following ability because the positive effect of pressure generally overrides any lesser negative effects. However, when pressure regulation is provided as described in the following control concepts, this characteristic does not limit BWR performance.

C. Moderator Temperature Coefficient

Second, let us consider the moderator temperature coefficient. This coefficient is the primary source of inherent load response in a pressurized water reactor, PWR. Our reasoning here may follow the reasoning used to explain the effect of this coefficient in a PWR. Changes in the average moderator temperature cause proportional, negative changes in reactivity. On a load demand, the moderator temperature drops, density increases, reactivity increases, and power increases to meet the demand. On a load rejection, the moderator temperature rises, density decreases, reactivity decreases, and power decreases to meet the rejection. Thus, the moderator temperature coefficient is a function of moderator density, and it provides the negative feedback necessary to make the reactor respond to load changes.

In a BWR, the reactivity associated with the moderator also may be assumed a function of moderator density. Such an assumption neglects the effects of temperature on the moderator cross section and the distribution of the density caused by steam voids. The temperature effects are second order, and are neglected justifiably. The void distribution is more significant, but its neglect does not overly detract from the performance of the analytical model in studying trends rather than true magnitudes.

With these assumptions, the average moderator density is:

$$\rho_m = \bar{R}_g \rho_g + (1 - \bar{R}_g) \rho_f$$

ρ_g = density of saturated steam

ρ_f = density of saturated water

$\bar{R}_g$ = void volume fraction

If the moderator has no voids, $\bar{R}_g = 0$ and $\rho_m = \rho_f$.

If the moderator is all voids, $\bar{R}_g = 1.0$ and $\rho_m = \rho_g$.

In this core design, $\bar{R}_g$ is small, so that $\rho_g \bar{R}_g \simeq 0$.

Therefore: $\rho_m \simeq (1 - \bar{R}_g)\rho_f$

$$\delta k \simeq f\left[(1-\bar{R}_g)\rho_f\right] = f\left[(1-\bar{R}_g)\rho_f(T_m)\right]$$

The density of saturated water ρ_f , is a function of temperature or pressure; it represents the moderator temperature coefficient. The void fraction is a complex function of power, pressure, rate of change of pressure, and subcooling; it represents the void coefficient.

Assuming small oscillations:

$$\delta k = \left(\frac{\partial f}{\partial T_m}\right)\Delta T_m + \left(\frac{\partial f}{\partial R_g}\right)\Delta \bar{R}_g$$

$\left(\frac{\partial f}{\partial T_m}\right)$ = moderator temperature coefficient

T_m = average moderator temperature

$\left(\frac{\partial f}{\partial R_g}\right)$ = void coefficient

These coefficients may be calculated from the slopes of the curves of k_{eff} in Physics Section 6.5

In a BWR, let us assume the same method to determine the average moderator temperature. However, the core outlet flow is a saturated, boiling water mixture, and so the outlet temperature is determined by pressure. The core inlet temperature change is a function of feedwater flow and temperature; as in a PWR, inlet temperature will decrease with increasing load.

Therefore, the negative feedback necessary for inherent load response may be available also in a BWR. To be effective though, this negative feedback from pressure must be large enough to override the positive feedback of pressure on voids.

In existing BWR designs, this moderator temperature coefficient is generally assumed negligible by comparison with the void coefficient. This assumption is valid in view of experiments performed on VBWR to determine whether or not a direct cycle BWR will inherently maintain pressure and respond to a load or control rod disturbance. To explain: With VBWR at steady state, the load was increased slightly by opening the turbine control valve a fixed amount. Pressure decreaced at a steady rate, and the control rods were not withdrawn. Flux decreased with pressure, and over a range of 100 psi, the reactor showed no tendency to recover and stabilize. This result was true for load rejections, control rod insertions, and control rod withdrawals.

Conversely, a similar experiment on EBWR exhibited a limited self-controlling effect. The reactor was stabilized manually with the turbine load; over a period of an hour, with no load or rod disturbances, the reactor maintained pressure within a band of 50 psi. Starting at an equilibrium of 574 psi, pressure drifted slowly up to 584 psi within the first half hour, and then decreased to 540 psi during the second half hour.

Obviously, there is a promise of a BWR design with inherent pressure regulation and load response, but as yet it is only a promise. The ability to design a core with temperature and void coefficients that would guarantee a long term, self-controlling plant is beyond the present level of BWR technical knowledge. Therefore, this capsule reactor design must include a positive method of controlling pressure, not an unusual requirement in view of the fact that all PWR designs include a pressurizer, and all BWR designs include an automatic pressure control or continuous manual rod control.

D. Control Concepts

Previous BWR experience and studies have yielded the following control concepts:

1. Dual Cycle. Subcooling and voids are controlled by withdrawing heat from steam generators in the recirculation flow loops. Reactor pressure is controlled by a pressure-regulated turbine control valve that controls team flow from the reactor. The control range is limited.

2. Feedwater Temperature Control. Subcooling and voids are controlled by varying the feedwater temperature. Reactor pressure is controlled by a pressure-regulated turbine control valve. The control range is limited.

3. Recirculation Flow Control. Subcooling and voids are controlled by varying recirculation flow. Reactor pressure is controlled with the recirculation flow control system or by a pressure-regulated turbine. The control range is limited.

4. Control Rods. The most versatile control device.

5. Burnable poisons in the fuel and/or the water. This is a limited range, long term, or steady-state feature. Burnable poisons in the fuel are used in the initial core design to the extent of approximately 6% $\delta k/k$.

6. Moderator Temperature Control. A method of control inherent in the core design; it offers the possibility of inherent pressure control and load response. In a BWR, the moderator temperature is determined by the operating pressure. In the initial reactor design, moderator temperature is used to control reactivity over the core life by permitting pressure to decrease.

E. <u>Minimum Control System</u>

The minimum control system was designed to be as simple a control system as practical. Thus, narrow restrictions are required in permissible load variations. Periodic attendance to maintain reactor power is also required. The preliminary specifications noted below define the performance of this minimum control system.

1. Specifications:

Generator	Load: The electrical load shall be 300 KW. Permissible variations shall not exceed +0%-10% at a rate less than 3 KW/minute. Frequency: The frequency shall be 800 cps ± 10%. Voltage: Voltage variation with life shall not exceed 10%.
Turbine	Inlet Pressure: Turbine inlet pressure shall be 550 psia. Permissible variations shall not exceed ± 15 psi. Speed: The turbine speed shall be 12,000 rpm ± 10%.
Reactor	Pressure: The reactor pressure will be variable between 1200 psia and 600 psia. Regulation: The pressure regulation due to the short term variations described in these specifications shall not exceed 25 psi.
Bypass	Flow: To insure adequate control of pressure and compensate for changes in flow required during life, a bypass flow of 5% to 10% of rated flow shall be maintained. Valve Capacity: The bypass valve shall be capable of handling 50% of rated flow at 1200 psia.
Water Level	Variation: The height of water in the spray condenser shall not vary by more than 1 foot.
Duty Cycle	Continuous: Once startup has occurred, the power plant shall deliver rated power continuously until the end of the core life is reached. Shutdown: Should a malfunction occur, the power plant will be designed to enter an emergency shutdown phase automatically. If the fault has been cleared, and no internal damage has resulted, startup will again be possible.

Attendance	No continuous attendance shall be required. Periodic attendance for the purpose of maintai-ing rated reactor power shall be req red.

2. System Description

Figure 36 is a schematic diagram showing one of the simplest configurations which may be used to adequately control the power plant. The manner in which the control system functions will become apparent by considering the contributions of each of the various control elements individually.

Turbine Inlet Throttling. As mentioned previously, reactor pressure will be varied from 1200 psia at the start of life down to 600 psia at the end of life in order to maintain reactor power. Since turbine loading will remain practically constant throughout life, the flow of steam required must remain essentially constant. Thus, the steam must be throttled as pressure changes. The amount of throttling required will vary as a function of life. Throttle valve V-1 shown in Figure 36 is the means by which this throttling is achieved. Since no access is available to elements within the capsule vessel, provisions must be provided for repositioning the valve remotely.

Reactor Pressure Controller. Excess steam is generated in the reactor and shunted to the spray condenser through a bypass line. The pressure control is used to modulate a bypass valve which controls this bypass steam and thereby controls reactor pressure. The pressure controller is of the simple, self-actuated, proportional type. Provisions will be made for resetting the pressure reference about which pressure is being controlled by a remote manual means; this is indicated in Figure 36. The pressure controller will make a continuous comparison between the actual pressure existing within the reactor vessel and the desired value as determined from the set point. Any variation between the actual pressure existing and that desired is converted into a power signal (derived from reactor pressure) that modifies the steam flow in a direction to reduce the pressure difference. Since this is a proportional type control, some error is always required to reposition the valve. The amount of error required is defined by the gain which is: 25 psi/25% steam flow change at 600 psia.

Feedwater Flow and Level Control. The feedwater flow   pressurized by a pump located on the end of the turbine shaft.   valve V-3 located in the line connecting the output of the pump to the reactor vessel is the method by which feedwater flow is varied. In principle, a nearly constant flow of feedwater is required since a nearly constant flow of steam is desired from the reactor vessel. Because of small disturbances which may occur and because of the change in core characteristics with life, this sensitive flow balance cannot be insured without a positive means of control. The system has a constant amount of water enclosed within the capsule. Any unbalance between the feedwater inlet and the steam flow output will result in a change in reactor water level and condensate sump water level. A drop in water level in the reactor could upset the natural circulation and, hence, the reactivity balance within the reactor. For this reason, it is necessary that the water level be controlled to within acceptable limits. The lower control illustrated in Figure 36 is the means by which this is accomplished. Since the water in the reactor is in a turbulent state due to boiling, a more accurate indication of water distribution is obtained by sensing the water level in the spray condenser. The water level sensor provides a signal to the level controller which is proportional to the change in water level. This signal is compared to a desired reference setting in the level controller, and an output is provided which is used to modulate the area of valve V-3. By varying the flow restriction presented to the feedwater pump with this valve, it is possible to control feedwater flow.

Speed Control. As long as the inlet pressure to the turbine remains constant and a constant electrical load is applied to the generator output terminals, there will be no change in turbine speed. Because of slight disturbances which can occur in a reactor power system, as well as variations in electrical output loading, some type of speed control must be provided. The speed control for this system consists of a means of automatically controlling a dummy generator load to hold speed constant. This type of speed control can be achieved external to the capsule. As indicated in Figure 36, a speed reference signal is provided to the turbine speed control. A comparison is made between this reference signal and the frequency of the voltage generated. A difference between these two signals is amplified and used to control the amount of dummy load presented to the generator. This approach presents the generator with a constant electrical load.

The size of this external speed control equipment varies directly with the amount of dummy load that must be dissipated. The specifications have been written so as to limit this dummy load to approximately 15% of rated power. This means that the dummy load must be capable of dissipating approximately 45 KW.

3. Startup

The power plant will be supplied with the amount of water which it will use throughout the remainder of its life. The necessary poison will be added to the water to prevent the reactor from going critical initially. Before poison removal begins, it is necessary that the steam throttle valve V-1 be fully closed. This must be done so that the initial power developed by the reactor may be used to heat up the system and increase pressure. The pressure reference to the pressure controller must be set at 1200 psia to be sure that the valve V-2 remains fully closed. The level reference to the level controller must also be set to insure that the flow valve V-3 is fully closed. Once these valves are closed, the power buildup can be begun by the process of removal of the soluble poison from the reactor water.

After the reactor has been raised to 1200 psia, steam flow will be passed through the bypass valve. At this time, an external source of electric power will be connected to the generator in order to drive it as a motor, thereby turning the turbine, the feedwater pump, and the circulating pump. Once the pumps have been started, it is permissible to start opening the main steam valve V-1. The steam entering the turbine will assist the generator in supplying the power required to overcome frictional energy and the pumping power. When the reactor power level is sufficient to supply the power requirements, the external electric power source will be disconnected from the generator, and the turbine will supply the circulation and feedwater pump power. By additional removal of the reactor poison, opening the main steam throttling valve, and loading the generator, the reactor will be brought up to full power. Close attendance will be maintained until Xe-Sm equilibrium is reached.

4. Normal Operation

It is assumed that the plant has been brought to steady-state rated power. All manual adjustments have been made for 1200 psia operation, 105% reactor power, 100% turbine-generator power, and 5% bypass flow. Under these conditions, no immediate attention need be given the plant. It is capable of providing continuous power for long periods of time.

As time progresses, the negative reactivity created by fuel burnup will decrease reactor power. This decrease will result in less bypass flow until the bypass valve is closed. With this valve closed, the pressure controller cannot function. Any further reduction will cause pressure to drop, and thereby reduce the power to the turbine-generator. Therefore, prior to a complete closure of the bypass valve, the reactor operator must lower the bypass pressure controller set point to:

Decrease reactor pressure
Decrease moderator temperature
Increase reactivity
Increase reactor power, and
Increase bypass flow back to .5%.

Simultaneously, the reactor operator must manually adjust the throttle valve to maintain a turbine inlet pressure of 550 psia.

To make these adjustments, the reactor operator will require the following indications:

Reactor pressure
Turbine inlet pressure
Total steam flow, and
Dummy load power

This method just represents one operating procedure; alternate methods of operation are possible.

5. Evaluation

The control system just described has certain characteristics which should be considered. To facilitate comparison, these characteristics will be listed as advantages and disadvantages:

Advantages:

1. The minimum control scheme is not complex. This should aid materially in the design of a reliable system.

2. The number of control parameters used are a minimum.

3. The pressure-controlled bypass valve is a proven concept (EBWR).

4. Only small disturbances are corrected automatically. Long term changes are corrected manually.

5. The control devices will be self-actuated, or powered by reactor pressure or the feedwater pump pressure.

Disadvantages:

1. Partial attendance will be required.

2. The generator must always be operated at 90%-100% load.

3. Manual adjustments must be made periodically on two or three of the control devices.

4. Loss of electrical load will shut the plant down.

F. Unattended Control System

As used here, the definition of an unattended control system is one in which no periodic corrections are required. Figure 37 is a schematic diagram of such a system. As will be seen in the description which follows, the scheme contains some features beyond automatic reset. The advantages gained by the inclusion of these additional features are reflected in the specifications.

1. Specifications:

The specifications are the same as for the Minimum Control System except for:

Generator	Load: The rated electrical load shall be 300 KW. Permissible variations shall not exceed 0%-100% at a rate less than 3 KW/minute. A full load rejection will be absorbed automatically.

Bypass	Valve capacity: The bypass will be capable of handling rated steam flow.

2. System Description

A comparison between Figures 36 and 37 will indicate the major difference between the two control methods. The additional items added internal to the capsule are:

Turbine speed control
Main steam flow divider
Automatic turbine inlet pressure control

Turbine Speed Control. The output of the speed control error sensor is a mechanical motion which is used to reposition a steam flow divider. All rejected steam is bypassed to the spray condenser. The flow divider is designed to present a constant flow restriction. Therefore, diversion of flow has no effect on the upstream pressure. The amount of flow rejection is determined by the amount of electrical load being supplied. Because of the constant flow restriction characteristic of the flow divider, a constant thermal load is presented to the reactor regardless of electrical loading. This device will be designed to permit the plant to absorb a full load generator tripoff.

Pressure Controller. The reactor vessel pressure control is essentially the same as that described for the Minimum Control System. In this case, the basic difference is that the trimming of the pressure reference signal is accomplished automatically. The intelligence for this automatic trim is derived from the spray condenser water level sensor. The signal is in the form of proportional mechanical displacement which is used to reposition the pressure reference. Any decrease in reactor power will show up as a decrease in total steam flow to the condenser and a decrease in condensate water level.

3. Evaluation

Advantages:

1. No attendance is required after the plant is placed in operation.

2. Complete loss of electrical load can be sustained without an emergency shutdown occurring.

3. No dummy load is required. The plant is more efficient.

4. The human error element is eliminated.

Disadvantages:

1. More equipment must be placed in the capsule.

2. The control system is complex. The automatic trimming of the pressure controller reference will require considerable study.

G. Control Elements

1. Feedwater Flow Regulator

Figure 38 is a schematic of a constant flow regulator. Flow from the feedwater pump is forced to flow through a fixed area orifice, the bottom of which is connected to a weight by means of a spring. Equilibrium is reached when the pressure drop across the orifice is sufficient to balance the weight. The purpose of the spring is to improve the dynamic response of the valve. If vernier control overflow is required, a needle may be inserted from the top to change the fixed orifice equivalent area.

2. Steam Control Valve

A single valve which incorporates the features of pressure regulation, throttling, and flow division is shown in Figure 39. The steam supply entrance is shown in the upper right-hand corner. Movement of the pressure regulator spool varies the flow area directly connecting the high pressure line with the condenser. In this manner, bypass flow may be regulated.

The purpose of the center spool is to provide throttling to insure a turbine inlet pressure of 550 psia. The positioning of this spool may be done either manually or automatically. The net outward force on the spool is proportional to turbine inlet pressure.

The remaining spool provides flow division. The bands of the spool have been machined so that a constant flow of steam flows from the valve outlets regardless of spool location.

3. Inlet Pressure Regulator

Figure 40 is a schematic of the Inlet Pressure Regulator that controls the center spool on Figure 39 and maintains 550 psia turbine pressure. The reference used is a weight which is connected to a mechanical torque summer by means of a spring. The force input to the summer is obtained from the unbalanced force existing across the inlet pressure regulator valve spool. An increase in pressure will result in an increase in the unbalanced force on the spool. Equilibrium is reached when the force developed produces a torque on the summer which just balances the torque produced by the reference weight.

4. Reactor Vessel Pressure Regulator

A schematic diagram of the Reactor Vessel Pressure Regulator is shown in Figure 41. As can be seen, this device is a mechanical torque summer with a variable ratio input. Rotation of the float input varies the fulcrum, which modifies the torque developed by the supply pressure force. If the variable fulcrum point is considered fixed, then the operation of the device is identical to the Inlet Pressure Regulator previously described. Movement of the fulcrum serves to readjust the pressure reference.

5. Turbine Speed Regulator

Speed control is achieved through the combined action of the Flow Divider, Turbine Speed Actuator, and the Turbine Speed Sensor. The Speed Actuator and Sensor are shown schematically in Figure 42. The actuator is a double-ended diaphragm actuator whose gain is set by the taper of the plugs at each end. The Speed Sensor is a centrifugal pump about 4 inches in diameter that operates at a small percentage of its maximum flow. This is the region of the ΔP versus flow curve, where a small change in pressure causes a large change in flow. This causes the pressure rise across the pump to be primarily a function of the square of speed.

The actuator is connected to measure the pressure rise developed across the pump. This causes the actuator to produce a force which is proportional to the pressure rise. This force is used to move the main steam valve Flow Divider spool. The position of force balance determines the amount of steam which is permitted to flow through the turbine.

6.7 Power Conversion Equipment

Turbine-Generator Assembly

As a result of a comprehensive analysis of the thermodynamic cycle, a selection has been made of a turbine-generator configuration that best meets the requirements of the over-all system.

Consideration was given to use of single staged, velocity compounded and multi-staged turbines in which speed, inlet, and outlet conditions were investigated to appraise the turbine type and potential performance and the effects of such variations on other system components. The resulting turbine flow and operating conditions were inter-related with the spray condenser, heat exchanger, circulating and feedwater pumps, as well as with the reactor requirements.

Each of the components, though not in themselves appearing as an optimum device, when integrated into this particular application contribute to a system representing an optimum concept.

Figure 43 is a sketch of the turbine-generator assembly. As shown, the turbine-generator circulating and feedwater pumps are mounted on a common 12,000 rpm shaft mounted in two radial bearings. The equipment is vertically mounted, with the turbine at the top, to permit exhausting directly into the spray condenser.

This arrangement also favors a maximum water sump head over the circulating pump inlets. The thrust and one of the radial bearings is located between the turbine and alternator.

At the lower end is mounted the circulating pump overhung from the end bearing. The feedwater pump is located above the turbine. This location resulted in a simpler assembly than if the pump were attached at the bottom. At the top location, it also turns in a steam vapor environment with minimum friction losses.

The general specifications for the turbine-generator package had been tentatively established as follows:

Single shaft machine, including all rotating components
Simple and reliable
Net 300 KW electric power output continuous operation
Voltage and frequency - at vendor's discretion

The resulting design is felt to meet the intended need, and represents the type of machine that can be fabricated and demonstrated within the allowable time schedule of the succeeding phase.

Specifically, there will be several developmental problems:

Feedwater Pump - technically feasible - past experience at lower heads per stage. Major difficulty in physical design of rotor due to high stress conditions and in appreciating a reasonable efficiency.

Circulating Pump - feasible - operating conditions close to prior experience. Major difficulty in securing optimum inducer design and material selection for impeller to withstand near cavitating conditions.

Radial and Thurst Bearings - development required in view of high bearing surface speeds and resulting operation in turbulent region.

Homopolar Alternator - will require some development, particularly in selection and application of the potting or insulating material to withstand slight radiation damage and wet steam environment.

None of these problems are of the category that raise questions as to inability to be overcome. Rather, they are technically feasible, and their solutions depend upon slight extensions of the state of the art.

The specifications for each of the rotating components are as follows:

A. Electric Generator

Homopolar Inductor Alternator
12,000 rpm
300 KW, 3 phase, 400/600 V, P.F. 1.0, efficiency = 92%
8 pole, 800 cps
Rotor diameter = 15-3/4"
Stack length = 10-1/2"
Coil length = 19-1/2"
Stack diameter = 27"
Weight = 1700 lbs.

Sizes, weight, and other data are approximate. Alternator selection and prior alternator experience largely determine speed selection.

B. Turbine

Single stage, single row turbine
Speed = 12,000 rpm
Pitch diameter = 25"
Pitch line velocity = 1310 ft/sec.
Bucket height = 0.570"

Turbine Stage Power Requirements

Alternator output = 300 KW
Alternator efficiency = 92%
Alternator input = 326 KW
Pump power requirement = 67 KW
Bearing losses = 24 KW

Total Turbine Stage Power = 417 KW

Reactor pressure (psia)	600	800	1200
Nozzle inlet pressure (psia)	550	550	550
Exhaust pressure (psia)	25	25	25
W/Vo	.338	.391	.395
Flow rate (lbs/hr)*	11100	11430	12350
Turbine efficiency (%)	53.4	52.5	49.5

C. Bearings

Radial bearings

Diameter = 4.5"
Clearance modulus = 0.0015 in./in.
L/D = 1.25
Full circular design with 3 helical grooves

Thrust bearing

Total thrust about 1600 lbs.
7 tilting pads
O.D. = 9.0"
I.D. = 4.5"

* Allowing condenser pressure to rise from 8 psi to 25 psi to satisfy pump inlet conditions accounts for these higher steam flow rates as compared to those reported earlier (Reference 5).

Materials

Bearings and shoes - graphitar
Journals - 17-4 Nitrided

Water flow

36 lbs/min/journal
206 lbs/min/thrust
266 lbs/min. total

Water pressure - 10 to 20 psi above ambient.

Power losses

7 HP/journal
18 HP/thrust
32 HP total = 24 KW

Critical speed approximately 14000 rpm.

D. Circulating Pump

Flow requirement

Turbine flow = 24.75
Nominal bypass flow = 1.0
Cooling water flow = 250
Bearing flow = 26

Total flow = 302 gpm

Sump water temperature = 180° F
Spray condenser pressure = 25 psia
Water head above inlet = 3 ft.
Net suction head = 43.4 ft.
Speed = 12,000 rpm
Suction specific speed = 12,400 rpm

Specific speed = 4200 rpm

Double inlet, mixed flow impeller

Pump pressure rise = 50 psi
Pump efficiency = 70%
Pump power = 11.5 HP
= 8.6 KW

E. <u>Feedwater Pump</u>

Speed = 12,000 rpm
Single stage hollow rotor pump
Flow rate = 25.7 gpm
Discharge pressure = 1300 psia
Scoop tube radius = 5.25"
Pump efficiency = 25% (estimated)
Pump power = 77.8 HP, 58.2 KW

F. <u>Turbine-Generator Assembly</u>

Total length = 44.5"
Diameter = 29"
Approximate weight = 2500 lbs.

6.8 Plant Instrumentation

Minimum instrumentation is provided for normal plant operation. Critical temperatures and pressures, and electrical output characteristics, are indicated and recorded for the purpose of diagnosing trouble in the event of plant malfunction.

Portable nuclear instrumentation and more precise plant process instrumentation is provided for plant startup. It is assumed that a trained and equipped crew will be available to start up and refuel installed production models. Such a crew would bring equipment to a new site to start up a plant, and remain long enough to insure that it is functioning properly and will continue to function on an unattended basis for its anticipated core life. The crew would then remove the nuclear and detailed plant process instrumentation, and move on to another new site where the same instrumentation would be used again.

6.9 Refueling and Servicing Procedure (Figures 8 and 44)

The plant may be refueled at the end of core life, and is designed and arranged so that major components may be removed for repair or replacement.

Reference to the direct cycle plant arrangement (Figure 8) shows that the top of the capsule is flanged for removal and access to the turbine-alternator-pump unit. This unit is removable, permitting access to the primary vessel for refueling. Figure 44 is a refueling flow sheet providing details of this procedure.

The entire spent core is removed as a unit to minimize shutdown time. This is accomplished by lowering a transfer cask over the open top of the primary vessel. Poison segments having been previously inserted in the core to insure subcriticality, the core is hoisted into the transfer cask. The transfer cask weighs approximately 15 tons. The reverse procedure is used to place a new core into the reactor, except the new core is not highly radioactive, making the cask unnecessary for shielding.

The startup crew is equipped with all necessary tools for refueling and maintenance, which are available for reuse at suceeding sites.

6.10 Heat Sink Requirements

The capsule reactor dissipates waste heat through an external heat exchanger to the body of water in which it is submerged. This may be a lake, pond, harbor, or the ocean, or any body of fresh or salt water large enough to accept the heat and dissipate it to the atmosphere. The limitations specified for dissipation of heat were that ambient air conditions would range from 80° F to -20° F. This temperature range might be experienced in the temperate zones, and result in a heat sink water temperature of 90° F to 32° F. Based upon the preliminary design work reported in Section 6.3E, it is estimated that sufficient heat transfer area can be provided in the sea water heat exchanger to permit adequate cooling, since there is a large temperature difference between the condenser sump (25 psi, 238° F) and the heat sink.

A number of possibilities exist for heat sinks to dissipate the reactor waste heat.

A. Pond or Sea (Figure 45)

Heat transferred to the local heat sink water must eventually pass on to more remote sinks or thermal pollution will raise the temperature above acceptable limits. The earth is a very poor conductor, so the ultimate sink must be considered to be the sky and air. For temperate climates, an annual evaporation rate of 70 inches of water may be assumed. Such an evaporation rate means an average pond surface area of 7 acres to dissipate 1.2×10^7 Btu/hr. Of course, in humid climates, there would be some periods on which atmospheric conditions would essentially prevent any heat removal from the pond. However, for an area of 7 acres and a depth of 50 feet, 1.2×10^7 Btu/hr would only heat the water 2.1° F per week without heat removal. Thus, short term hot spells with no heat removal, of a month in duration, would not present a serious situation.

B. Icecap (Figure 46)

Discussion with the Corps of Engineers regarding Camp Century experience with obtaining water from the icecap has suggested the possibility of dissipating reactor waste heat to the ice.

At Camp Century in Greeland, potable water is obtained by boring a hole approximately 150 feet deep in the ice and melting the ice at the bottom of the hole by means of a steam jet. The water thus obtained is contained in the resulting cavity, and may be pumped to the surface for use as required. It is considered feasible to obtain 3000 gallons of water per day in this manner.

Some analytical work was accomplished to determine the feasibility of dissipating the heat to the ice from an ALPR type plant. The heat load was 2.7×10^7 Btu/hr, and heat was rejected at 130° F. The analysis showed that steady-state conditions were obtained after several years with the water temperature reaching approximately 34° F, with a cavity diameter of approximately 750 feet. Using a similar approach, it was estimated that the heat of 1.2×10^7 Btu/hr from the unattended reactor would result in equilibrium conditions with a cavity diameter of approximately 438 feet.

Experimental data has shown that the ice varies in density from .3 to .7 from the surface to a depth of 200 feet. It is therefore expected that the pool of water would be retained and not flow away into the ice as it is formed. It is also known that the ice tends to creep and fill in cavities. This is not expected to be a problem since heat from the surface of the water would tend to melt the ice in the top of the cavity until some equilibrium water level condition was reached.

It is appreciated that problems of lowering the plant to the proper depth, conducting power away from the plant, and refueling would exist. These problems have not been studied, and the scheme is only offered as a suggested means of establishing a suitable heat sink.

6.11 Safeguard Considerations

The scope of work for this study did not permit evaluation of anticipated safeguard questions. However, safeguard problems which can reasonably be expected to arise in the design, construction, and operation of a sealed, unattended plant of the type described herein can be outlined.

Three general problem areas are of potential interest:

1. Safety of capsule during storage or shipment to destination. Storage and shipment would probably be evaluated in a manner analogous to storage and shipment of fuel in a conventional container.

2. Safety of access to plant during startup and following a period of operation becomes amenable to safeguard evaluation when the design has progressed far enough to show the instrumentation and protective devices to be used during plant startup at the site.

3. Plant safety during normal operation will require detailed evaluation of the design, especially the means by which the reactivity and pressure would be controlled.

Specific accident conditions that are anticipated to require more detailed study include the following:

1. Failure of ion exchange system.
2. Leakage from primary to secondary system.
3. Leakage from secondary system to outside.
4. Failure of circulating or feed pumps.
5. Stoppage of turbine-alternator.
6. Loss of electrical load.
7. Error in checking reactivity worth of core during production.
8. Failure of instrumentation used to start production plant in field.

The proposed remote locations of the field units will reduce containment requirements. Three lines of containment will still be present, however - the high integrity fuel elements, the reactor vessel, and the capsule vessel. The fuel elements are stainless steel clad, stainless steel-uranium dioxide cermet core, plate-type elements which have low corrosion rates, a high melting point, and high strength at operating temperatures. The reactor and capsule vessels will be designed with appropriate corrosion allowance and safety factors to make failure unlikely.

The need for a scram device for the field plants will be evaluated on the basis of prototype tests and performance, and detailed accident analyses, that will be required before final design of the field units is completed. It is believed that the reactor system can be developed to the point where destructive nuclear excursions and gross reactor vessel failures are not credible for the field units.

6.12 Potential for Increasing Plant Life

After determining the other parameters of the plant design selected as the reference, the fuel life was re-checked and found to be 2.3 years instead of the initially assumed 3 years. This difference results from the trade-offs made in firming up the other plant design parameters. Working from better knowledge of these other parameters, the three year life can be re-established and possibly exceeded. The methods which can be used to do this are: increased thermal efficiency, refinement of end of life reactor pressure, use of non-uniform distribution of fuel and/or burnable poisons in the core, and optimization of beginning of life pressure. These areas are described below:

1. Separate Motor-Driven Circulating Pump

   The reference design has the turbine, generator, feed pump, and circulating pump on a common shaft. This was done to improve plant reliability by having only one rotating part. However, it was achieved at the expense of increasing condenser back pressure to 25 psia to satisfy inlet conditions for the circulating pump. The higher condenser back pressure brings with it the advantage of permitting higher heat sink temperatures. The reason for this is that the corresponding condensate sump temperature may be higher, thus resulting in a higher temperature difference across the external heat exchanger. Removing the circulating pump from the turbine shaft and driving it separately with an electric motor at 6000 rpm would permit condenser back pressure to be lowered to 8 psia with a resulting increase in plant efficiency. The reference design thermal rating of the core is 3300 KW. Separately driving the circulating pump would permit thermal rating of approximately 2600 KW. The lower thermal rating results in a direct increase in core life amounting to approximately 6 months. The final selection of a single shaft power conversion unit versus a separately motor-driven circulating pump would be made based on the results of detail design and trade-offs among parameters of reliability, core life, and heat sink temperatures.

2. Investigation of the Effects of Dropping End of Life Reactor Pressure Below the Selected Value of 600 psi

A gain in reactivity of 1-1/2% due to temperature change would result by lowering end of life pressure to 400 psi. Some of this gain would be lost by an increase in void fraction at the lower pressure. The reference design is conservative when compared to BORAX performance which operated with 3% reactivity in voids at 300 psi. However, the resulting void fraction would require investigation to determine its effects on control and its approach to the design limit of exit steam volume fraction in the hot channel. In addition, tradeoffs with turbine design parameters would have to be made to determine optimum steam inlet conditions. A gain of 1% reactivity would result in a 3-month increase in core life.

3. Non-Uniform Distribution of Fuel and Burnable Poisons in the Fuel

A number of refinements could be made with non-uniform distribution of fuel and burnable poisons in the core. Proper distribution of boron in the high flux areas and higher concentrations of U_{235} in the periphery of the core would tend to yield a flatter flux distribution, more uniform burnup, and thus extend core life. It is probable that other burnable poisons with smaller effective cross sections or a lumped poison such as boron could more nearly match the reactivity loss due to burnup. It is estimated that improved design optimization in this respect could extend the core life by as much as six months.

4. Optimization of Beginning of Life Pressure and Reactivity Control

It will be noted from the physics analysis that pressure takes a sharp drop from 1200 psi to 1100 psi in life in approximately the first 2 weeks. This is the result of the buildup of fission product poisons (principally samarium) in the fuel. A way to correct this condition would be to increase reactivity initially to overcome the fission product buildup effect. The design pressure at beginning of life could be retained by bypassing the excess steam during the startup operation or until the control requirements of 10/5% bypass were met and the plant was ready to go on its unattended life cycle. Another promising method would be to fabricate samarium into the fuel element initially, and compensate for the initial high cross section fission product buildup.

6.13 Plant Weight

The estimated weight of the capsule is less than 30,000 pounds. The total weight of the plant, including the capsule and supporting equipment, is less than 60,000 pounds. These figures are believed to be conservative based upon the conceptual design work accomplished up to this time. It is recognized that supporting equipment will be required for starting the plant and for shutting it down. The approach taken in the design as described throughout this report is that soluble boron will be used in the primary water to shut the plant down, and that a demineralizer system will be used to remove the boron for startup purposes. These systems are therefore required as supporting equipment, and are included in the over-all weight of the plant. In addition, a base to support the reactor in an upright position in the heat sink is required, as well as basic instrumentation and electrical load control and disconnecting equipment.

No attempt was made to optimize the weights or arrangements of supporting equipment. The weights given are therefore very rough estimates.

Without having studied arrangements of equipment or the problems involved with mounting equipment on skids, it is expected that the total plant could be arranged on a maximum of three skids. A general grouping of the total plant equipment follows.

		Pounds	
1.	Capsule Weight		
	Primary vessel	7000	
	Reactor vessel insulation	700	
	Reactor internals, chimney, side mounts, and braces	100	
	Reactor core	400	
	Containment vessel	6700	
	Sea Water heat exchanger	5500	
	Shroud	6500	
	Power conversion unit	2500	
	CAPSULE TOTAL WEIGHT		29,400
2.	Supporting Equipment		
	(a) Reactor Demineralizer System		
	Demineralizer	9000	
	Resin add tank	2000	
	Demineralizer cooler	3000	
	Demineralizer pump and motor	800	
	REACTOR DEMINERALIZER TOTAL WEIGHT		14,800
	(b) Poison System		
	Tank	3500	
	Pump and motor	1000	
	Piping and valves	500	
	POISON SYSTEMS TOTAL WEIGHT		5,000
	(c) Support Base		
	Frame	3000	
			3,000
	(d) Electrical Equipment	2000	
			2,000
	OVER-ALL PLANT WEIGHT CAPSULE AND SUPPORTING EQUIPMENT		54,200

It will be noted that the weight of the capsule is 29,400 pounds exclusive of the skid or shipping crate. It is estimated that the capsule weight can be reduced by as much as 10% by design optimization in the following areas:

(a) Shortening of reactor vessel length by reducing chimney height. This will result in an over-all reduction in plant length, with a subsequent reduction in weight.

(b) Reduction in thickness of the outer shroud, or complete elimination of same.

The problems of optimizing plant weight and arrangement of total plant equipment for shipment to a selected site by air transport or other means would be the subject of study during the prototype phase of the program.

SECTION 7.0

DESCRIPTION OF PROTOTYPE PLANT AND DEVELOPMENT PROGRAM

7.1 Description of Prototype Plant

A. General

The prototype plant design is based on the most promising concept as described in Section 6.0. The selected design is a direct cycle boiling water reactor contained in a single capsule having the turbine-generator and pumps on a single shaft.

It is assumed that the plant as described can be installed and tested at Vallecitos Atomic Laboratory in Pleasanton, California. Selection of this site is contingent upon approval of the project by the General Electric Company and Federal authorities. In the event of difficulty in obtaining such approvals, it is presumed that these plans could be adopted to carry out the program at Idaho Falls. In order to meet safeguard requirements, the prototype is equipped with a scram device in addition to the liquid poison injection system. A complete nuclear and process instrumentation system is provided for plant safety, and to supply information to verify design analysis and criteria.

The plant is contained in a pool of water which acts as a heat sink and shielding for servicing operations. Heat is removed from the heat sink by means of circulating water through a heat exchanger to a cooling tower.

A pool makeup system and demineralizer are supplied for pool water clean-up purposes. A reactor demineralizer is provided to remove boron from the reactor water and to clean up the reactor water system.

B. Prototype Scram Device (Figure 47)

The prototype scram device is shown in the reference drawing given above. The purpose of the scram device is to provide an additional emergency shutdown scheme other than liquid poison injection. It is provided to insure a fail-safe system while the reference design is being confirmed.

For startup of the prototype plant, liquid poison withdrawal will be used, and the scram device will be employed only for shutdown. Sub-criticality is established by the sudden withdrawal of the center nine fuel assemblies. A rack and pinion are used to raise and lower the center fuel assemblies, and a spring is used to supplement the force of gravity upon scramming. A drive shaft extends through the reactor pressure vessel, insulation, and containment vessel walls utilizing rotary seals. The shaft is attached to a drive sprocket, which in turn is connected to the drive by means of a drive chain. The scram device is actuated by de-energizing the magnetic clutch to the drive motor. The system is fail-safe, and scrams in the event of the drive chain breaking or an electrical failure.

C. Auxiliary Systems (Figure 48)

Heat Sink Cooling System

The cooling system for the reactor pool consists of a closed primary loop that circulates demineralized water from the reactor pool through a holdup tank and external heat exchanger. Heat absorbed by the primary loop is transferred to a secondary system in a shell and tube heat exchanger. The secondary water system uses normal service water circulated through a cooling tower to dissipate the system heat.

This cooling system is the same thermal rating and approximate duty as the conventional systems employed in General Electric pool-type reactors.

D. Makeup and Pool Cleanup Demineralizer System (Figure 49)

The initial charge of water for the pool and for the reactor must be demineralized. Pool water must also be continually demineralized to minimize activated corrosion product buildup in the pool.

To provide the initial charge of demineralized water, a 25 gpm water softener and mixed-bed demineralizer are adequate. The mixed-bed demineralizer is a package unit complete with auxiliaries for manual regeneration of the resins. The softener is a package complete with brine tank and controls for manual regeneration. This system will provide initial demineralized water requirements, as well as continuing makeup requirements.

If the prototype is located elsewhere than at VAL, the suitability of the above arrangement must be investigated with respect to the quality of the water supply. Some different system may be required which, for example, could include the addition of a filter.

The mixed-bed demineralizer is also connected so it can be used to maintain pool water purity. Following makeup of the initial charge of water for the system, the resins are regenerated and the demineralizer is ready for cleanup use. After the reactor is in use, the pool water may contain sufficient activity to cause the exhausted resins in the demineralizer to be discarded and replaced. However, regeneration facilities are also available for use during initial testing.

E. Reactor Demineralizer (Figure 49)

The reactor demineralizer serves several purposes in the system:

1. To remove the boron poison control so the reactor will start up.

2. To operate during the initial period of operation to remove the corrosion products produced during the reactor startup so the system will start off clean. This period is expected to be 300-400 hours after temperature is achieved.

3. In the prototype, the demineralizer can also be used repeatedly if desired for successive startups.

The reactor demineralizer system is a high pressure system which includes the following:

1. A recirculation pump
2. A heat exchanger
3. Two demineralizers in parallel
4. A fresh resin addition tank
5. A spent resin storage tank
6. Flowmeter
7. Control valves
8. Shutoff valves and piping
9. Anion and cation resins

To start the reactor, cold water from the reactor is pumped through the heat exchanger, through the control valve and flowmeter to one demineralizer and back to the reactor. The reactor demineralizer is initially filled only with anion resin, which removes the boron poison from the reactor water. The water can be pumped at a constant rate, in which case boron is removed from reactor water at an exponential rate.

A specific case is shown in Fugure 50, where initial poison concentration is 75 pounds ammonium pentaborate in 10,000 pounds water (2025 ppm boron). For a final boron concentration of 10 ppm at the end of 30 hours, the flow rate is 1773 pounds reactor water per hour. A very long time would be required to remove the last traces of boron.

If it is desired to remove boron at a constant rate, the demineralizer flow rate can be varied from 1/2 to 5 gpm as shown on Figure 51. Any other boron removal rate can be achieved as desired and within flow capacity of the equipment.

As boron is removed, reactor temperature and pressure is increased; then heat is removed by the heat exchanger to produce an outlet temperature of 120° F (140° F maximum) since the demineralizer resins are sensitive to temperature. Cooling water flow can be varied manually to maintain the desired temperature.

To remove 75 pounds of ammonium pentaborate will require about 130 cubic feet of anion resin and about 10 cubic feet of cation resin. The demineralizer tank is sized to contain about 13 cubic feet of resin to make the size small for a distant location, to minimize the capital cost, and to achieve a reasonable unit flow rate through the demineralizer. Exhausted resins will have to be removed and replaced with fresh resins about ten times during the boron removal process. If boron removal is carried out at an exponential rate, per Figure 50, resin beds would initially have to be replaced at about half-hour intervals while the final bed would last indefinitely. Alternately, if the boron removal rate were kept constant, per Figure 51, then resin bed changes would occur at 3-hour intervals. Only one demineralizer is needed if the boron removal process is stopped each time a resin bed is exhausted. However, two demineralizers are provided so that resins can be replaced in one bed while the other is in use, since the exhaustion cycle of one bed can be quite short.

The last two resin charges (Nos. 10 and 11) will be mixed cation and anion resin to remove cations as well as anionic boron. Also, the mixed bed will remove corrosion products formed during the startup cycle, as well as adjust the reactor water pH to a neutral value. The final bed will be able to operate for a long time after the reactor is operating to remove final traces of boron as well as corrosion products to produce a clean system. Exhausted resins from the prototype unit are discarded to a storage tank for final disposition. Resins from the initial reactor startup will contain little radioactivity, and could be regenerated for re-use. However, the economics and waste disposal aspects at the prototype site would first have to be studied.

F. Liquid Poison Injection System

The liquid poison injection system supplies soluble boron to hold down the cold-clean core and to shut down the reactor during normal operation.

The system consists of the following components:

1. High pressure positive displacement pump
2. Open tank with cover, 250 gallon capacity, with mixer
3. Control valves and piping
4. Tank level indicators
5. Ammonium pentaborate

A batch of ammonium pentaborate is mixed with water in the tank. For initial operations of the reactor, the batch mixture is pumped into the primary vessel. Another batch is prepared for standby. The reactor is started as previously described, by removal of the boron with the reactor demineralizer system.

When it is desired to shut down the reactor, the valving of the system is arranged to permit the standby batch mixture to be pumped into the primary system. The pump is rated to pump against reactor rated pressure. Sufficient shutdown control is provided by a solution of 2025 ppm of boron.

CONTROL AND INSTRUMENTATION

General Features

The control and instrumentation system includes:

1. Nuclear instrumentation, to measure neutron flux at the core and to supply signals required for startup, for operation at power, and for alarm and scram circuits.

2. Area radiation monitors outside the pool.

3. Pool cooling system instrumentation.

4. Reactor process instrumentation.

5. Reactor auxiliary systems process instrumentation.

Nuclear Instrumentation (Figure 52)

Startup Channels

The startup channels each use a proportional counter. The counter has an effective range from a low of 0.12 nv to a high of 4.2×10^4 nv (0.12 nv corresponds to about 4 microwatts in the core, 4.2×10^4 nv to 1.3 watts).

The pulses from the proportional counter are amplified and then counted in the log count rate meter and recorder. The meter indicates log count rate.

The counter pulses go to a scaler which can be set either to count the number of neutron pulses occurring during a preset time interval or to register the time required for a preset number of pulses. Pulse signals from the scaler operate a loudspeaker to give an audible indication of flux level. It is used primarily to check very low count rates during startup.

Log-N Period Channels

These channels monitor power level of the reactor over the range from about 0.1 watt to 10^6 watts (flux level from 4.5×10^3 nv to 4.5×10^9 nv at the chamber location). The system integrates the current from a gamma compensated ionization chamber, amplifies this signal by means of an amplifier having a logarithmic characteristic covering a 7-decade range, and differentiates the log-N signal to give the reactor period. Two recorders permanently record log-N and reactor period. The period signal is fed to the trip amplifier for possible scram.

Safety Channels

Three safety channels monitor power level over the entire flux range of 4.5×10^1 nv to 4.5×10^{10} nv. They therefore overlap the startup channel range, and cover the log-N period channel range. Each safety channel contains a gamma compensated ionization chamber located in the pool, a micromicroammeter, and a trip actuator.

The safety channel signals are combined with signals from the log-N and period amplifier in a trip amplifier. The trip amplifier scrams the reactor on high flux or fast period.

Area Radiation Monitors

Twelve radiation monitors are located in the reactor installation to give audible and visual indication when radiation levels exceed those specified in the range of 0.1 to 100 mr/hr.

Pool Cooling System Instrumentation

The cooling system instrumentation is used for high-power operation when forced circulation is required for reactor cooling (above 100 KW). Primary and secondary coolant water flow, primary heat exchanger inlet and outlet temperatures, and cooling water conductivity are measured and recorded. Separate strip chart recorders for recording these measurements are located on a separate control panel in the reactor control center.

Reactor Process Instrumentation

Pressure, steam and water flow, and temperature measurements within the reactor vessel and its primary cooling system are indicated in the control room for the reactor operator. Recorders on the process instrument panel provide permanent records of these parameters for observation of current trends and for future study. In addition, reactor water level is recorded and controlled, and reactor containment vessel low water level is alarmed.

Reactor Auxiliary Systems Process Instrumentation

The liquid poison system is instrumented for level measurement, level alarm, and pressure and conductivity measurements. Readout instruments for these parameters are mounted on the process instrument panel.

The reactor and pool demineralizer systems are instrumented for flow, temperature, conductivity and differential pressure. Indicating, recording, and integrating instruments as required are also mounted on the process instrument panel.

D. Prototype Building and Structural Arrangement (Figure 53)

The prototype reactor power plant will be installed in a pressure-tight containment vessel below ground. The facility is enclosed in a service building as shown on the reference drawing.

The prototype test building is divided into three functional areas: the heat sink, the service area, and the control area.

The heat sink consists of a pool of water contained in a pressure-tight vessel below ground. The vessel is fabricated of carbon steel, and is surrounded by reinforced concrete. During reactor operation, the pool will be covered and sealed by the vessel head. Energy-absorbing material is located under the head to be capable of absorbing missile energy in the unlikely event of a catastrophic accident to the power plant.

The service area portion of the building is divided into two levels, one at approximately the top level of the pool, and the other at the next level down.

Refueling and maintenance operations will be carried out at the top level. The structure here is a steel-framed building with steel panel wall construction. Lapped seams will be used in the wall construction to provide better than average leakage resistance.

At the next level down are located the heat exchangers, pumps, demineralizers, and service equipment for the pool and the reactor power plant. The equipment will be arranged and shielded so that maintenance access can be had to this level during plant operation.

The control room is located at the refueling service elevation in an adjoining structure.

Containment is achieved in this plant by a form of direct pressure suppression with the reactor located in a water pool inside a pressure-tight containment. Vessel failure will release the vapors directly to the pool, where they will be suppressed. The design pressure of the containment was established on the basis of taking no credit for suppression. This approach provides a large margin in the design since it is probably more realistic to assume that suppression will be complete.

Radioactive gas leakage, either at an incident or during refueling from a ruptured element, will be handled by the special exhaust duct at the top of the pool. Isolation valves are provided in the exhaust duct that close on signal in the event of an incident during operation, thus isolating the pool and insuring condensation of steam energy release.

The building above the pool will be of leak-resistant construction, and the ventilation will be such that air leakage will normally be into the building.

7.2 Description of Development Program

A. Introduction

The objective of the program to be described is the successful development of a 300 kilowatt power plant for unattended operation over several years of operating life. The concept of unattended operation requires that the design be made as simple as possible in the interest of obtaining maximum reliability of operation of the few components of the plant. In addition, known and proved technology must be used throughout the design of the reactor and power system in order to establish the feasibility of the design in advance of expensive developmental testing and model fabrication.

Reactor and heat transfer equipment design and testing would be done by the Atomic Power Equipment Department of the General Electric Company. Reliable heat exchanger manufacturers would be called on to assist in the design and manufacture of the special heat exchanger equipment used in the fluid systems. The design and development of the turbine-alternator equipment and circulating pumps and motor would be placed with the Mechanical Engineering Laboratory Section of the General Engineering Laboratory of the General Electric Company for detailed design, analog studies, analysis of the transient performance of the power plant, and manufacture of the initial control equipment required for the power generation equipment. Over-all program responsibility and coordination would reside with the Atomic Power Equipment Department.

Because of the relatively small size of the reactor and power plant involved in this development, it is considered practical to perform the nuclear experiment and prototype operation at the Vallecitos Atomic Laboratory of General Electric Company at Pleasanton, California. The advantages of performing the experimental work on this reactor power plant at the California site are obvious because of the nearness of the laboratory to the design and manufacturing center in San Jose. In the event that approval to perform the experiment at Vallecitos is not obtained from the necessary governmental bodies, it would be proposed to perform this experiment at the National Reactor Testing Station in Idaho on a site provided by the Atomic Energy Commission.

B. Development Program Schedule (Figure 54, see next page)

The schedule is divided into two major sections - the over-all program and the design areas requiring confirmatory testing. The design areas are again sub-divided into the areas associated with the reactor and primary system, the turbine-alternator and secondary system, and the power plant in general. The time cycle of the program is divided into four phases: The first phase is conceptual design, analysis, and initial confirmatory engineering tests; the second phase is detailed design, additional confirmatory tests, and component fabrication; the third phase is critical reactor tests; and the fourth phase is full power reactor tests. The entire time cycle shown on this schedule covers a period of 2-1/2 years, and is divided into months.

The design area portion of the schedule shows the major tasks that will be undertaken in the course of preliminary and detailed design. In many cases, the items are shown in two portions of the schedule. Where tasks should be started in Phase I, this is shown; at a later date, in Phase II, the task is taken up at the appropriate time for more detailed investigation.

Following is a task breakdown according to the development program schedule. This breakdown may be more easily followed by referring to the schedule as the various tasks are studied by the reader. The schedule time scale is shown in consecutive months so that the first month may be any point in time. Critical schedule dates may then be referenced to the beginning of the time cycle, and calendar or fiscal years may be superimposed upon the time cycle.

C. Task Breakdown

The program covers design and development work aimed at the design, fabrication, installation, and test of a capsule nuclear power plant capable of unattended operation for a period of several years. The objectives of the proposed 2-1/2 year program are:

1. To complete conceptual design efforts currently under way, and obtain theoretical data in support of basic feasibility.

2. To develop detailed designs for the reactor and power plant components.

3. To establish, by component testing, data pertaining to plant reliability and life.

4. To obtain experimental proof of reactor physics calculations through critical experiment testing.

5. To construct and operate a full-scale prototype in order to fully evaluate the capabilities of the unattended reactor and its associated power plant equipment.

This program is divided into three major tasks to enable precise definition and control of the development effort. The tasks are:

A. Reactor and Primary System Development
B. Turbine-Generator and Secondary System Development
C. Power Plant General Development

The Reactor and Primary System Development task encompasses reactor physics calculations to establish core configurations, method of control, reactivity calculations, and critical experiment testing. It also included design of the reactor mechanical design and experimental verification of heat transfer coefficients, hydraulic parameters, etc.

The Turbine-Generator and Secondary System task encompasses mechanical design and proof-testing of high-speed rotating machinery in a sealed container. Necessary investigations include materials selections, bearing designs, critical speed relations, etc. The secondary system also includes mechanical development work on hot water circulating pumps with very low NPSH, experimental verifications of hydraulic ~~pump~~ performance, and establishment of heat transfer coefficients.

The Power Plant General Task entails the establishment of the plant control system, development of assembly and initial startup procedures, determination of total plant performance data by prototype operation. Further detail for each of these tasks is presented separately.

Expected Results - First Year

(a) Complete conceptual designs.
(b) Complete major portion of detailed designs and component testing.
(c) Procure materials and begin fabrication of prototype plant.

Problems Anticipated During the Second Year

Detailed designs and component testing will be completed. The prototype plant will be constructed and critical experiments completed. A prototype facility will br erected, critical experiment tests will be completed, and reactor tests initiated.

Task A - Reactor and Primary System Development

Under Task A, the work will include, but is not limited to, the following:

1. Reactor Physics

   a. Core Calculations. Provide design analysis and computations to determine optimum core geometry, core loadings, core and fuel enrichments, poison solution requirements for adequate control and nuclear performance data over plant life.

   b. Critical Experiment Testing. Physics parameters will be verified by critical experiment tests of the prototype core to verify such things as critical mass, temperature coefficient, void coefficient. Tests will be conducted on both a cold-clean core and a hot critical core in the prototype installation.

2. Fuel Fabrication Development

   Efforts will be directed toward the successful application of ARB developed fuel technology to the 300 KW capsule plant. Investigations are expected to lead to the establishment of process standards (i.e., fabrication specifications) governing the manufacture of the fuel-bearing plates, cladding specifications and method of application, and techniques for assembly into suitable fuel elements. Prototype testing of sample fuel elements in an operating reactor will verify the capability of the fuel to perform under design temperature, heat flux, and power levels. These tests will determine effects of burnup and temperature on the proposed fuel.

3. Soluble Poison Development

   The present conceptual design proposes use of soluble poison as a means of preventing criticality prior to startup. This poison would be boric acid or similar boron compound in water solution. The investigations that will be conducted under this sub-task will determine methods for filling the dry reactor with the solution, means of removing the poison from the water (including percentages and rate of removal that could be accomplished). See Section 7.1E for more details.

4. Water Treatment - Water Decomposition

   The problems of corrosion product buildup, water purity, and water decomposition will be the subject of an extensive program to be conducted by out-of-pile testing and finally by operation of the prototype.

5. Reactor Design and Fabrication

   Development efforts required during design and fabrication of a prototype reactor will include significant investigations of reactor and primary system hydraulics, as well as special emphasis on mechanical factors contributing to plant reliability over the projected life. The key parameters would appear to be circulation flow rates and distribution, and their effects on reactor output and heat transfer. Final verification of the resultant parameters will be obtained during prototype tests.

Expected Results - First Year

(a) Conceptual designs will be completed to establish basic feasibility.

(b) Detailed designs will be initiated and critical components tested to establish performance data and reliability.

(c) Prototype reactor materials will be ordered and fabrication of reactor and primary system components initiated.

Problems Anticipated During the Second Year

The program will continue through the second year. It is anticipated that the prototype reactor will be completed and installed by the 20th month. Critical experiment testing is scheduled to be complete and full scale reactor testing under way by the 24th month. The completion of the reactor testing and full plant testing are projected for the 24th to 30th months.

Task B - Turbine-Generator and Secondary System Development

Task B encompasses the design and initial testing of the power conversion system, including the turbine-generator and secondary system. This effort is sub-divided into the following programs:

1. Detailed Design

   Designs of the turbine-generator, condenser, circulating pumps, and associated secondary systems will be completed. Basic performance criteria will be established and fabrication specifications prepared.

2. Water-Lubricated Bearing Development

   The application of water-lubricated bearings to the design of the turbine-generator and the circulating water pumps requires experimental evaluation of various designs to determine optimum configurations and materials reliability.

3. Generator Development

   Design of the generator entails an evaluation of permanent magnet versus inductor alternator types in general, and in particular the ability of electrical insulations and magnet materials to withstand radiation effects over the life of the plant. Investigations will be conducted in these areas, and, based on experimental results, an optimized design selected. Final design of the generator will incorporate results obtained in Sub-Task 2 above as well.

4. Turbine Design and Construction

The turbine is expected to be of generally conventional design. It is anticipated that the turbine and generator will be assembled on a single shaft and tested as a unit. Necessary testing will determine that performance criteria have been met.

5. Pumping Element Development

The capsule plant envisions use of both a secondary circulating pump and a primary feed pump. Investigations will be made to determine the capability of each pump to function under conditions of relatively high temperature and low NPSH and the feasibility of the single-shaft concept (turbine-generator and pumps on one shaft).

6. Secondary System Hydraulics

Detailed analysis of condenser design and predictions of flow and heat transfer parameters will be verified through mockup testing.

7. Condenser Scaling

Materials evaluations will be made to determine corrosion and scaling characteristics of various condenser tubes operating in fresh water. The information obtained will lead to selection of optimum materials, and will suggest design configurations for minimum effect of scale formations. Consideration will be given to corrosion and scaling problems in salt water if it is determined that these conditions prevail at the site selected for eventual application.

Expected Results - First Year

(a) Detailed design of turbine-generator set, pumps, and secondary system will be completed.

(b) Water-lubricated bearing tests will be conducted and optimum design established.

(c) Component tests on the turbine and generator and fabrication of prototype unit initiated.

(d) Pumping element analyses will be completed, prototype units constructed, and confirmatory testing initiated.

Problems Anticipated During the Second Year

(a) Testing of pumping elements will continue in the 12th to 16th month.

(b) Tests of assembled turbine-generator unit will be conducted, with completion scheduled by the 18th month.

(c) Hydraulic analysis of secondary system and resolution of heat exchanger scaling considerations will be completed.

Task C - Power Plant General Development

Task C is proposed to provide integration of reactor and primary system with the power conversion equipment and secondary system. The programs proposed herein include:

1. Plant Control System Design and Analyses

Detailed analyses and component designs will be evolved to provide safe and reliable operation. Problems to be considered include startup correlations of reactor and turbine-generator systems, normal operation control of turbine-generator frequency and feed pump speeds, system response to transient load changes, ability to safely accommodate load rejections and/or loss of auxiliary power, and ability to compensate for long-term drift in reactor output. The analytical results will be verified by testing of the prototype equipment.

2. Application to a Specific Site

The complete program includes construction and operation of a full-scale prototype. Presumably, this prototype would be erected at General Electric's Vallecitos Atomic Laboratory. In order to complete this phase of the program, it will be necessary to design a heat sink, analyze the site and the plant safeguards requirements associated therewith, and obtain AEC approval for construction and operation.

3. Plant Assembly and Shipping Procedures

The proposed plant is expected to be an extremely compact unit which may be fully assembled at the factory and shipped as a package to the installation site. The requirements for compactness and shipment as an assembly introduce several questions which must be satisfactorily answered. The unit must be designed to fit together easily and accurately. It must also be designed to maintain proper alignments and tolerances during shipment, at which time it would be turned from its normal vertical position to a horizontal position. One of the problems which must be carefully analyzed is the problem of bearing damage from shipment of assembled units. Adequate provision must be made to prevent brinelling, fretting, corrosion, etc., of pump and turbine-generator bearings. Three dimensional studies will be made, probably including full scale wooden mockup assemblies in order to verify assembly and alignment procedures.

4. Reactor Startup Procedures

Reactor startup investigations will involve a study of both the methods of starting a production-type reactor and the startup procedure for the prototype. In addition, the complete subject of reactor operation will receive a searching investigation to be sure that all areas of safety have been adequately covered before the nuclear testing program is initiated.

The reactor startup procedure is closely allied with over-all plant control, and will be carefully coordinated with the work in Sub-Task 1 above to insure safe operation at all times.

5. Reactor Testing

Construction and operation of a full scale prototype plant is proposed to experimentally verify the performance and capabilities of the 300 KW unattended plant. The first phase of this testing will consist of reactor tests conducted without the turbine-generator. During this phase, the reactor will be operated at varying conditions and over a wide range of power output. Measurement of various nuclear parameters will serve to finally verify the reactor design.

6. Plant Testing

After verification of reactor performance under Sub-Task 5, the turbine-generator system will be added and the complete prototype plant tested. Operating experience under accelerated life tests will serve to determine such basic parameters as ability to maintain load, response to load changes, transient effects and component reliability.

Expected Results - First Year

(a) Plant control system analyses will be carried out.
(b) Preliminary site analysis will be completed.
(c) Facility designs will be initiated.
(d) Reactor assembly procedure development will be under way.

Problems Anticipated During the Second Year

Major tasks delineated above will continue in the second year. Control system analysis should be complete by mid-year. Plant assembly and startup procedures will be completed. Reactor tests will commence at the completion of the critical experiment program described under Task A. It is anticipated that reactor test will begin about the 24th month. The reactor testing and subsequent total plant testing will extend into the third year.

D. <u>Development Program Costs</u>

These costs are estimated on a CPFF basis, and include all applicable adders and fee. This schedule was arbitrarily split into fiscal years, and had assumed a start date of August 1, 1960. While slippage in start date will shift the split by year, the total estimated costs cited below apply to the point where the prototype has been built, checked for hot and cold criticality, and is ready for power test operations.

Design and Development	FY '61	FY '62
a) Reactor and Facilities Design	$ 415,000	$ 318,000
b) Fuel Design	19,300	32,700
c) Water Treatment	65,000	111,000
d) Power Conversion Unit	766,000	511,000
	$1,265,300	$ 972,700
Prototype		
Materials and Equipment		$1,000,000
Installation		410,000
		$1,410,000
Operation		
Startup and Criticality Tests		470,000
GRAND TOTAL	$1,265,300	$2,852,700
Capital Facility		
Reactor Pool and Building	$165,000	$305,000

SECTION 8.0

FIELD PLANT COSTS AND SCHEDULES

8.1 Field Plant Costs

The realistic basis for evaluating the advantages to be gained by pursuing the development of any new system will inevitably focus on the question of foreseeable cost savings. Obviously, there will be features of the Capsule which cannot be measured in comparable dollars; e.g., if an unattended plant is truly necessary in an isolated location, the question of cost becomes secondary to the question of whether there is a power supply available that can do the job. Thus, not all, by any means, of the savings achievable in an unattended plant can be measured on a direct equipment cost basis, but this certainly serves as a starting point.

To provide a basis for true equipment costs of a small plant of this nature, it is necessary to consider beyond the first-of-its-kind plant in order to avoid excessive charges for special tooling and engineering, and to take advantage of a basic manufacturing "learning curve" that occurs on any new design. For this reason, the equipment costs cited are actually unit prices for an initial order of ten plants. Pricing is done on the basis of standard industrial practice, and the quoted numbers include all adders and profit as normal for a fixed-price bid.

The price of the plant on the above basis is $600,000. The price of the initial fuel load is $135,000.

An estimate of power costs in mills/kwh is provided as follows:

	$	Est. Life or Frequency Years	Annual Cost	Mills/kwh
Capital Cost				
Reactor Equipment	360,000	20	18,000	6.9
Turbine-Generator Feed Pump	240,000	9	26,700	10.1
	$600,000		$44,700	17.0
Fuel Cost				
Fabrication	135,000	3	45,000	17.1
Burnup	63,300	3	21,100	8.0
Reprocessing	96,500	3	32,200	12.2
	$294,800		$98,300	37.3
Operating and Maintenance				
Personnel (startup and refueling)	2,100	3	700	
Materials	9,000	3	3,000	
			$3,700	1.4

TOTAL - 55.7 mills/kwh

8.2 Field Plant Schedule (Figure 55)

The over-all schedule time cycle is 46 months. The total time is divided into two major sections: the development program, and the time required for production of ten field units. The development program portion has been previously described in Section 7.0.

The schedule for production of field units assumes that the ten units are built consecutively, an order being placed for all ten units at one time; and delivery is f.o.b., on skids, San Jose, California.

Two months are allowed after the completion of the development program to incorporate design changes in the field unit. Drawings are released to the shop at the 32nd month. The first production field unit is completed at the 40th month, based on completion of the first two turbine-generator feed pump units at the 38th month. The remaining nine plants are completed at the 46th month, based on completion of one turbine-generator feed pump unit per month after completion of the initial two units.

8.3 Comparative Cost Analysis

The best means of evaluating comparative costs of different systems is to apply them all to a specific site and measure total costs and annual costs necessary to meet a given power need. In this manner, each system bears the weight of its own auxiliaries. However, the application of Capsule to a specific site is outside the scope of this study, as well as any application of other nuclear plants to such a site. At the direction of the AEC, a meeting was arranged with Kaiser Engineers to discuss the applicability of their recently completed "Study of Remote Military Power Applications" (Reference 10) in providing comparative nuclear data. It was pointed out at this meeting that the South Pole Station was the application having the closest comparable power rating (500 KWe). Since this is a manned base where extreme reliability is required for life support, a high degree of backup is specified. For comparative purposes here, the data is used to show total plant comparisons as divorced as possible from the fact of a pole site and neglects backup power. The comparison of a 300 KWe Capsule to other 500 KWe plants may appear unfair. However, one could assume that the improved heat sink conditions at the Pole would improve Capsule output while the reduced operating crew would reduce the electrical heating load required to be supplied by the reactor. This comparison serves at best to give only an order of magnitude estimate, and the disparity in ratings is within proper range for this purpose. Tables 3 and 4 of the Kaiser Engineers report on the Pole Station, Antarctica, are reproduced in total with a column added to show Capsule.

TABLE 3
SUMMARY OF CHARACTERISTICS
PORTABLE AND MOBILE NUCLEAR POWER PLANTS

CONCEPT	ML-1	PL-2	PM-1	PM-2A	Capsule
Type of reactor	Gas-cooled	Boiling Water	Press. Water	Press. Water	Boiling Water
Design contractor	Aerojet-General Nucleonics	Combustion Eng.	The Martin Co.	Alco Products	General Electric
Nominal capacity	330 KW (net)	1,000 KW (net) + 0.4 MWT steam	1,000 KW (net) + 2.0 MWt	1,560 KW (net) + 0.3 MWt steam	300 KW (net)
Auxiliary power, KWe	70	237	250	315	0
Reactor thermal rating, MWt	3.4	8.8	9.37	10	3.3
Core lifetime, MWt-years	3.9	30.0	18.7	8.0	9.9
Heat dissipation	Gas-to-air	Steam-to-air	Steam-to-air	Steam-to-glycol	Steam-to-water
Design ambient air temp., °F	100	60	70	Arctic conditions	80
Design altitude, ft. above sea level	0	6,000	6,500	Arctic conditions	
Plant lifetime, yrs.	6	20	20	20	20 (9) turbine
No. of operating personnel	6	7	8	8	< 1
No. of shipping packages (excluding foundations and turbine building)	4	17	18	26	2
Shipping weight, lbs. (excluding foundations and turbine building)	120,000	485,000	508,000	711,000	60,000
Delivery time (f.o.b. plant)	15 mos.	15-18 mos.	14 mos.	15-18 mos.	10 mos.
Status of project	Prototype, operation scheduled in 1961. Component tests, GCRE and GTTF are now operating.	Ref. design of PL-2 complete. Stationary, 300 KWe prototype (SL-1) now operating. PL-2 type core to be tested at 8-10 MWt starting July 1961.	Design complete. Operation scheduled for summer 1961 at Sundance, Wyo.	To be shipped May 1 for installation and operation at Camp Century, Greenland, by September 1, 1960	Prototype can be operating July, 1962. Field unit deliverable February 1964.

TABLE 4

COMPARISON OF COSTS

NUCLEAR POWER PLANT COSTS MODIFIED FOR POLE STATION

	ML-1	PL-2	PM-1	PM-2A	Capsule
Plant Net Capacity, kwe	500	500	500	500	300
Reactor Thermal Power, mwt	4.2	3.6	3.5	3.5	3.3
Electric Energy per Core, kwhrs	4.4×10^6	33.9×10^6	28×10^6	11.3×10^6	7.9×10^6
Electric Energy per Year, kwhrs (0.9 plant operating factor)	3.93×10^6	3.93×10^6	3.93×10^6	3.93×10^6	2.63×10^6
Core Lifetime, yrs.	1.1	7.9	6.0	2.9	3.0
Core Fabrication Cost	$ 240,000	$ 330,000	$ 250,000	$ 240,000	$ 135,000

ESTIMATE OF PROJECT COST

	ML-1	PL-2	PM-1	PM-2A	Capsule
Structures and Improvements	$ 160,000	$ 315,000	$ 286,000	$ 350,000	$ 10,000
Nuclear Power Plant Equipment	2,950,000	1,788,000	1,952,000	2,600,000	547,000
Total Direct Construction Cost	$3,110,000	$2,103,000	$2,238,000	$2,950,000	$ 557,000
Indirect Cost	1,280,000	1,560,000	1,660,000	2,170,000	15,000
Total Direct and Indirect Construction Cost	$4,390,000	$3,663,000	$3,898,000	$4,920,000	572,000
Escalation	260,000	220,000	234,000	300,000	58,000
Total Including Escalation	$4,650,000	$3,883,000	$4,132,000	$5,220,000	$ 630,000
Design Engineering	200,000	400,000	400,000	400,000	53,000
Total Excluding Contingency	$4,850,000	$4,283,000	$4,532,000	$5,620,000	$ 683,000
Contingency	970,000	857,000	908,000	1,120,000	137,000
TOTAL PROJECT COST	$5,820,000	$5,140,000	$5,440,000	$6,740,000	$ 820,000

TABLE 4 (Cont'd)

	ML-1	PL-2	PM-1	PM-2A	Capsule
ANNUAL COSTS					
Nuclear Fuel					
Fabrication	$ 218,000	$ 42,000	$ 41,000	$ 83,000	$ 45,000
Burnup	32,000	18,000	27,000	28,000	21,000
Reprocessing	418,000	11,000	32,000	51,000	32,200
Transportation	10,000	7,000	5,000	9,000	5,000
Subtotal - Nuclear Fuel	$ 678,000	$ 78,000	$105,000	$ 171,000	$103,300
Operating and Maintenance					
Labor w/escalation	$ 142,000	$ 184,000	$210,000	$ 210,000	$ 700
Supplies	46,000	30,000	32,000	42,000	3,000
Subtotal - Operating and Maintenance	$ 188,000	$ 214,000	$242,000	$ 252,000	$ 3,700
Annual Depreciation	875,000*	257,000	272,000	337,000	56,000**
TOTAL ANNUAL COST	$1,741,000*	$ 549,000	$619,000	$ 760,000	$163,000

* Based on a plant life of 6 full power years.

** Based on replacement of rotating equipment every 9 years.

SECTION 9.0

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SECTION 10.0

DRAWINGS AND FIGURES

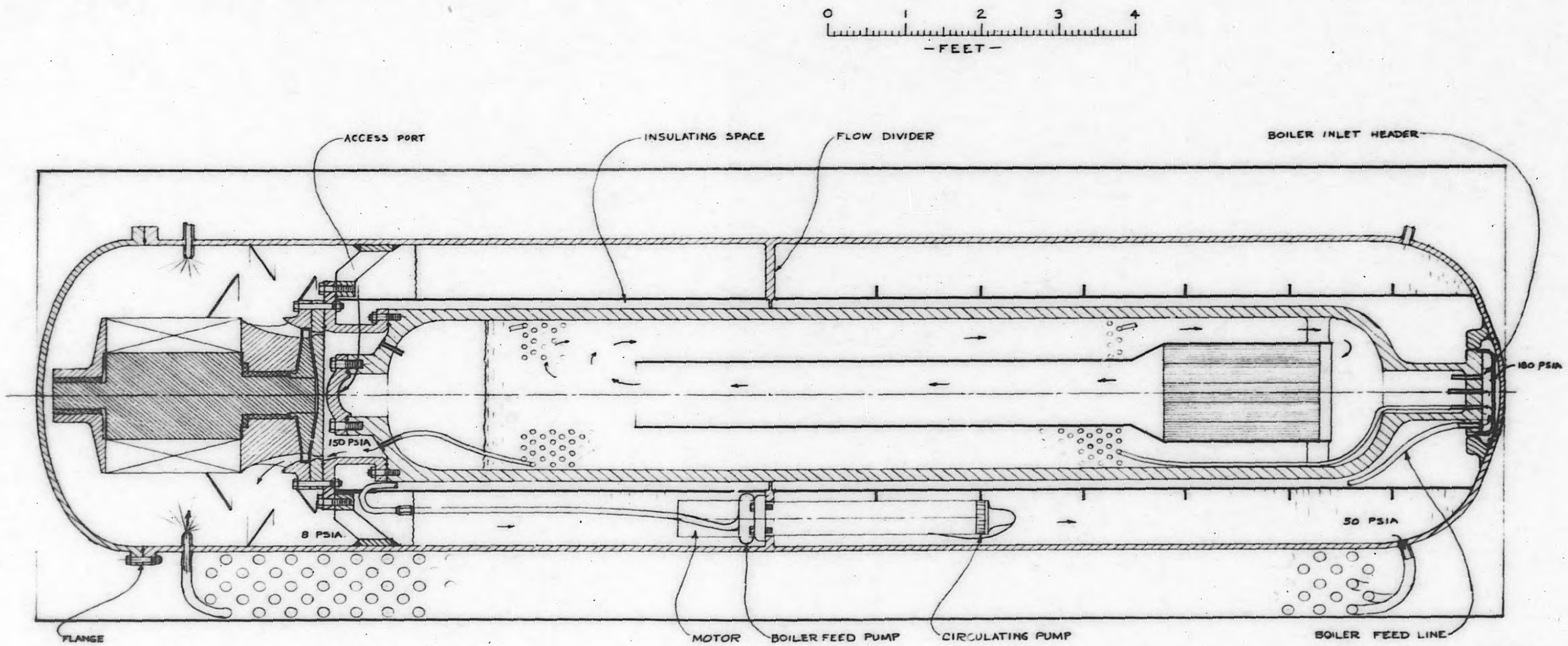


FIGURE 1

J.H. GERMER JUNE 30, 1960	APPROVED			GENERAL ELECTRIC ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
	BY	FOR	DATE	TITLE UNATTENDED CAPSULE REACTOR INDIRECT CYCLE ARRANGEMENT
				MADE FOR 300 KW STUDY
				SCALE	612 D 839

Issued Jul. 19 '60

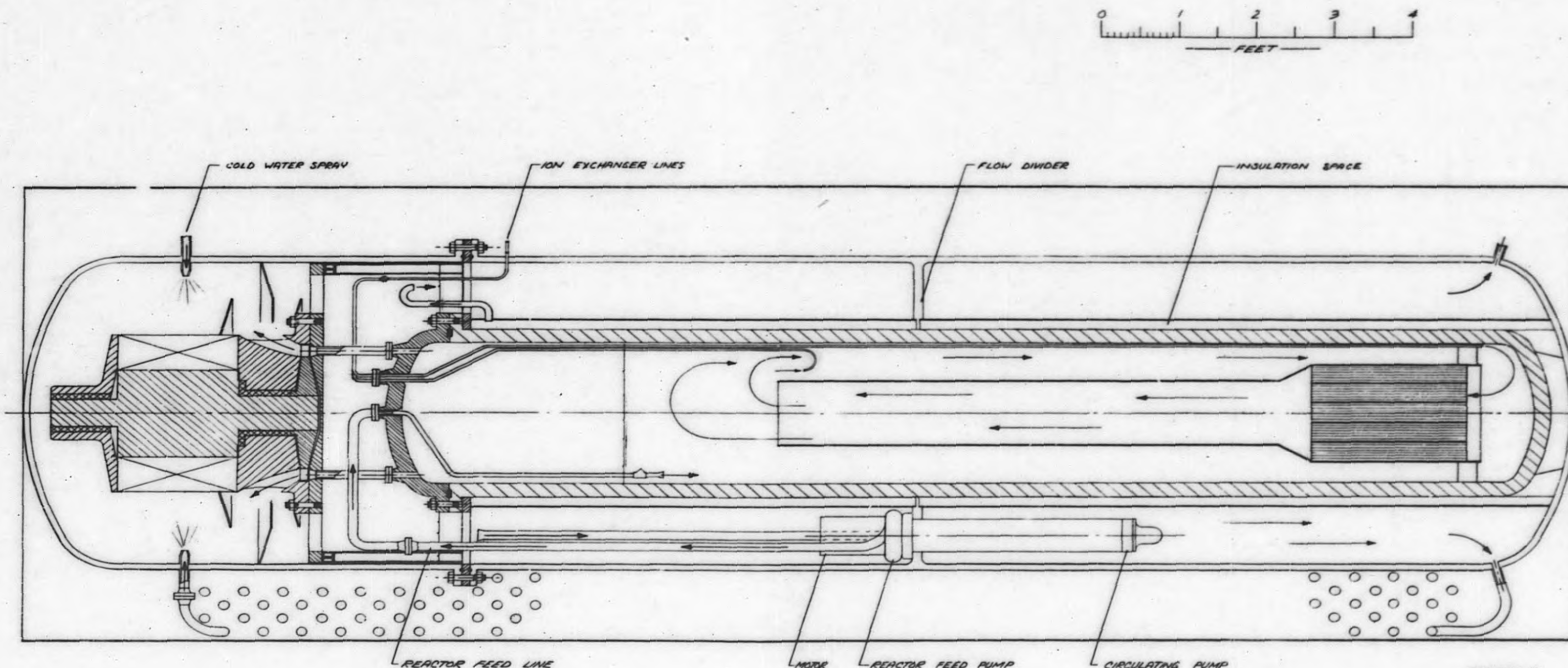
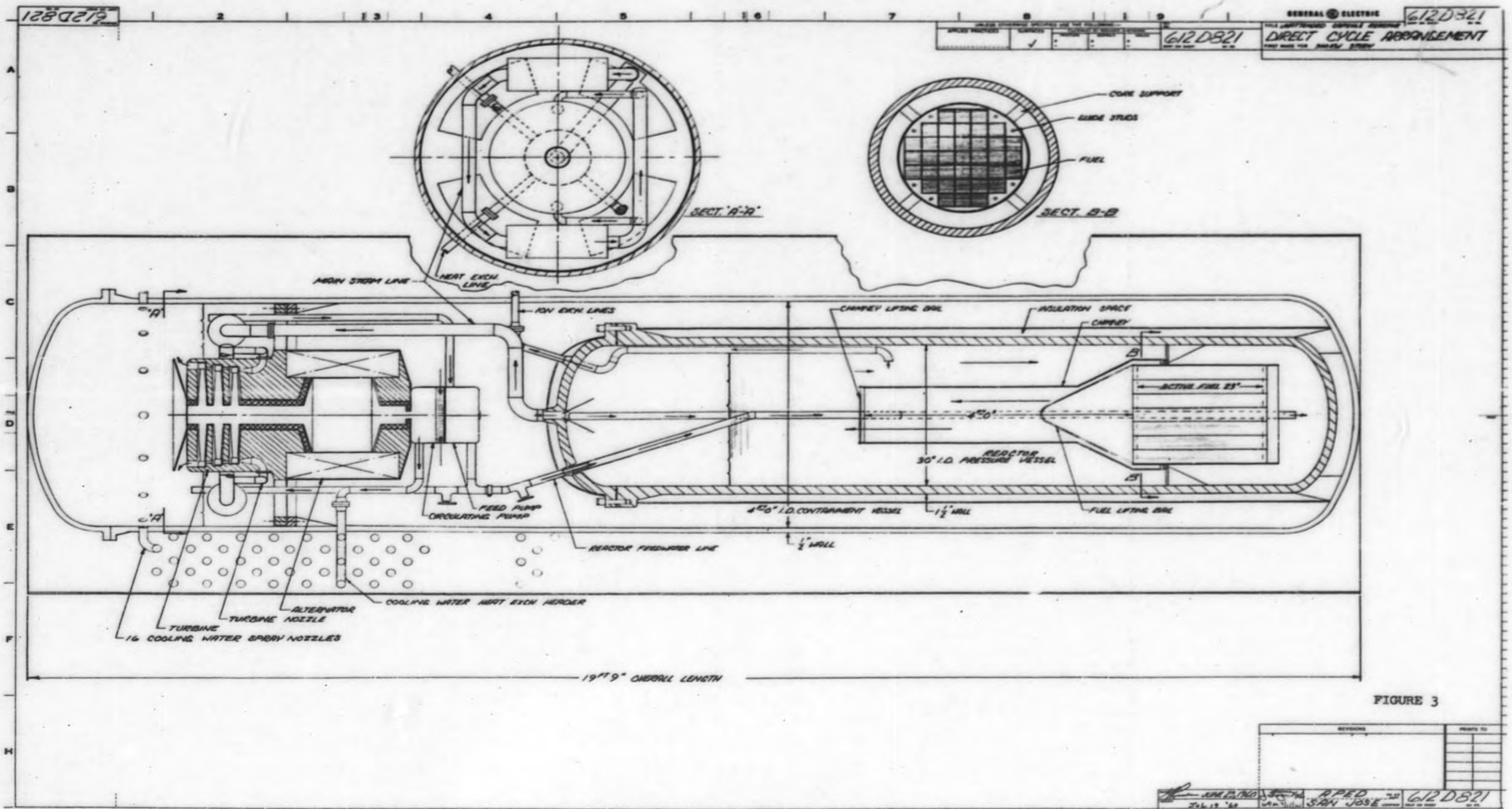


FIGURE 2

DRAWN BY	APPROVED			GENERAL ELECTRIC
	BY	FOR	DATE	ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
M.K. [illegible] MAY 26, 1960				TITLE UNATTENDED CAPSULE REACTOR DIRECT CYCLE ARRANGEMENT
CHECKED BY				MADE FOR 300 KW STUDY
ENGINEER J. [illegible]				SCALE / DWG. NO. 612 D 838 / REV.

revised July 19 '60



612D821
GENERAL ELECTRIC
DIRECT CYCLE ARRANGEMENT
612D821
CORE SUPPORT
GUIDE STUDS
FUEL
SECT. 'A-A'
SECT. B-B
MAIN STEAM LINE
HEAT EXCH LINE
ION EXCH LINES
CHIMNEY LIFTING BAIL
INSULATION SPACE
CHIMNEY
ACTIVE FUEL 23"
4'-0"
REACTOR
30" I.D. PRESSURE VESSEL
4'-0" I.D. CONTAINMENT VESSEL
1 1/2" WALL
FUEL LIFTING BAIL
FEED PUMP
CIRCULATING PUMP
REACTOR FEEDWATER LINE
1/2" WALL
COOLING WATER HEAT EXCH HEADER
ALTERNATOR
TURBINE NOZZLE
TURBINE
16 COOLING WATER SPRAY NOZZLES
19'-9" OVERALL LENGTH
APED
SAN JOSE
612D821

FIGURE 3

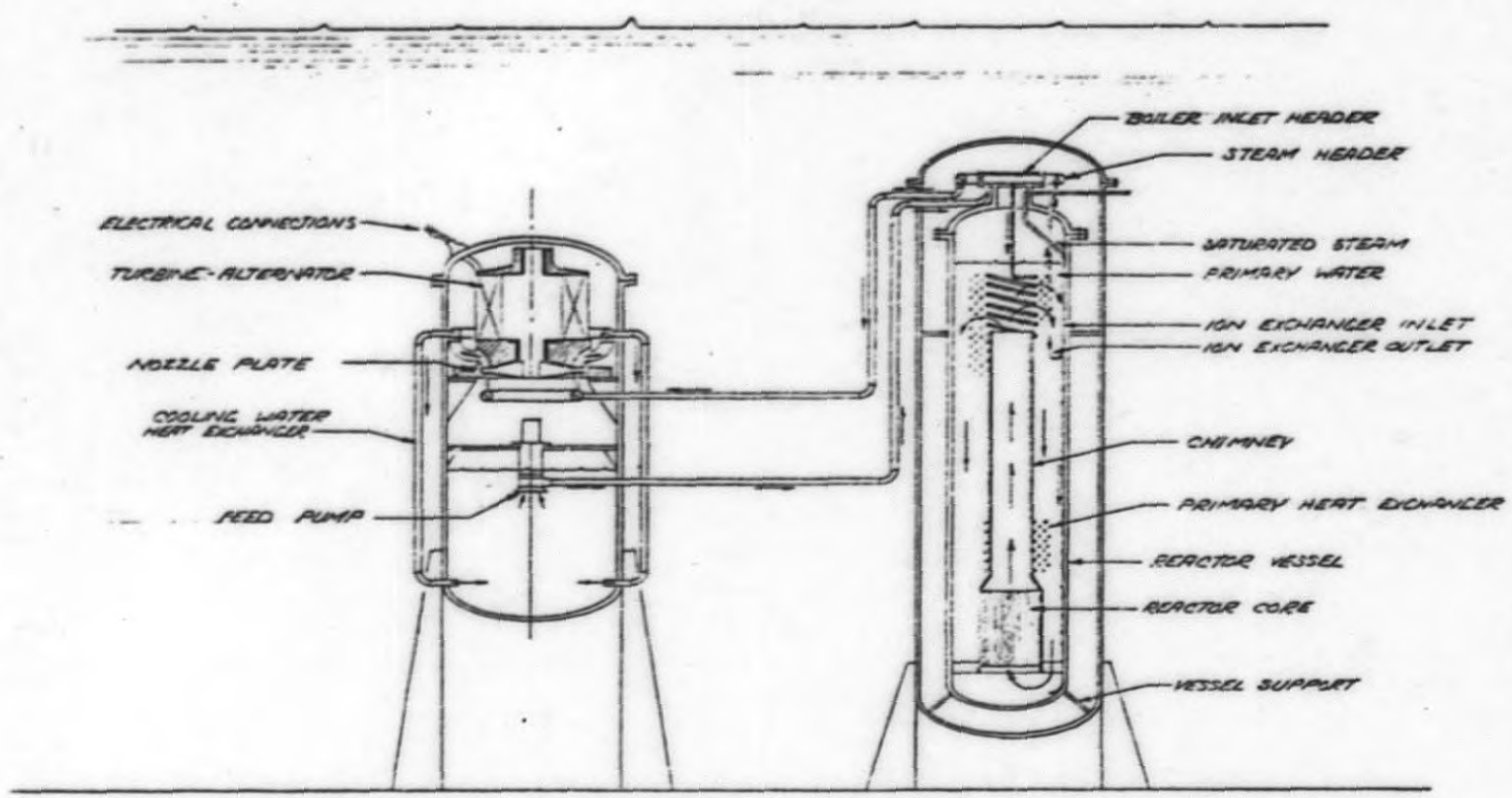


FIGURE 4

DRAWN BY	APPROVED BY	FOR	DATE
CHECKED BY			
ENGINEER			

GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE UNATTENDED CAPSULE REACTOR
TWO CAPSULE ARRANGEMENT
MADE FOR 300 KW STUDY
SCALE
DWG. NO. 932C531
REV.

REVISIONS

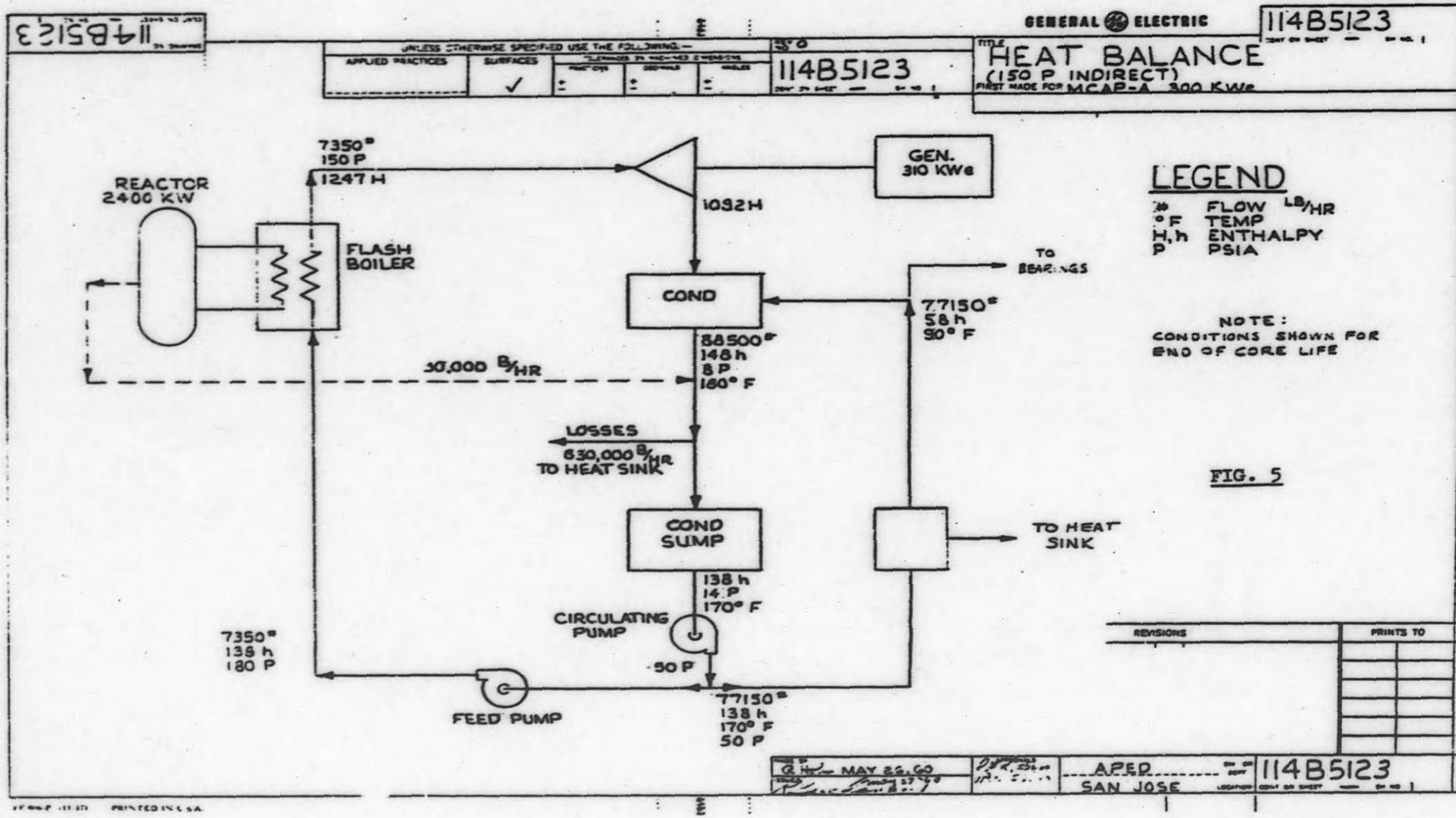

114B5123
GENERAL ELECTRIC
114B5123
HEAT BALANCE
(150 P INDIRECT)
FIRST MADE FOR MCAP-A 300 KWe
UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING—
APPLIED PRACTICES
SURFACES
114B5123
7350#
150 P
1247 H
GEN.
310 KWe
REACTOR
2400 KW
FLASH
BOILER
1032H
LEGEND
FLOW LB/HR
°F TEMP
H,h ENTHALPY
P PSIA
TO
BEARINGS
COND
77150#
58 h
90° F
NOTE:
CONDITIONS SHOWN FOR
END OF CORE LIFE
88500#
148 h
8 P
180° F
30,000 B/HR
LOSSES
630,000 B/HR
TO HEAT SINK
COND
SUMP
TO HEAT
SINK
138 h
14 P
170° F
CIRCULATING
PUMP
7350#
138 h
180 P
50 P
FEED PUMP
77150#
138 h
170° F
50 P
REVISIONS
PRINTS TO
MAY 26, 60
APED
SAN JOSE
114B5123
PRINTED IN U.S.A.

FIG. 5

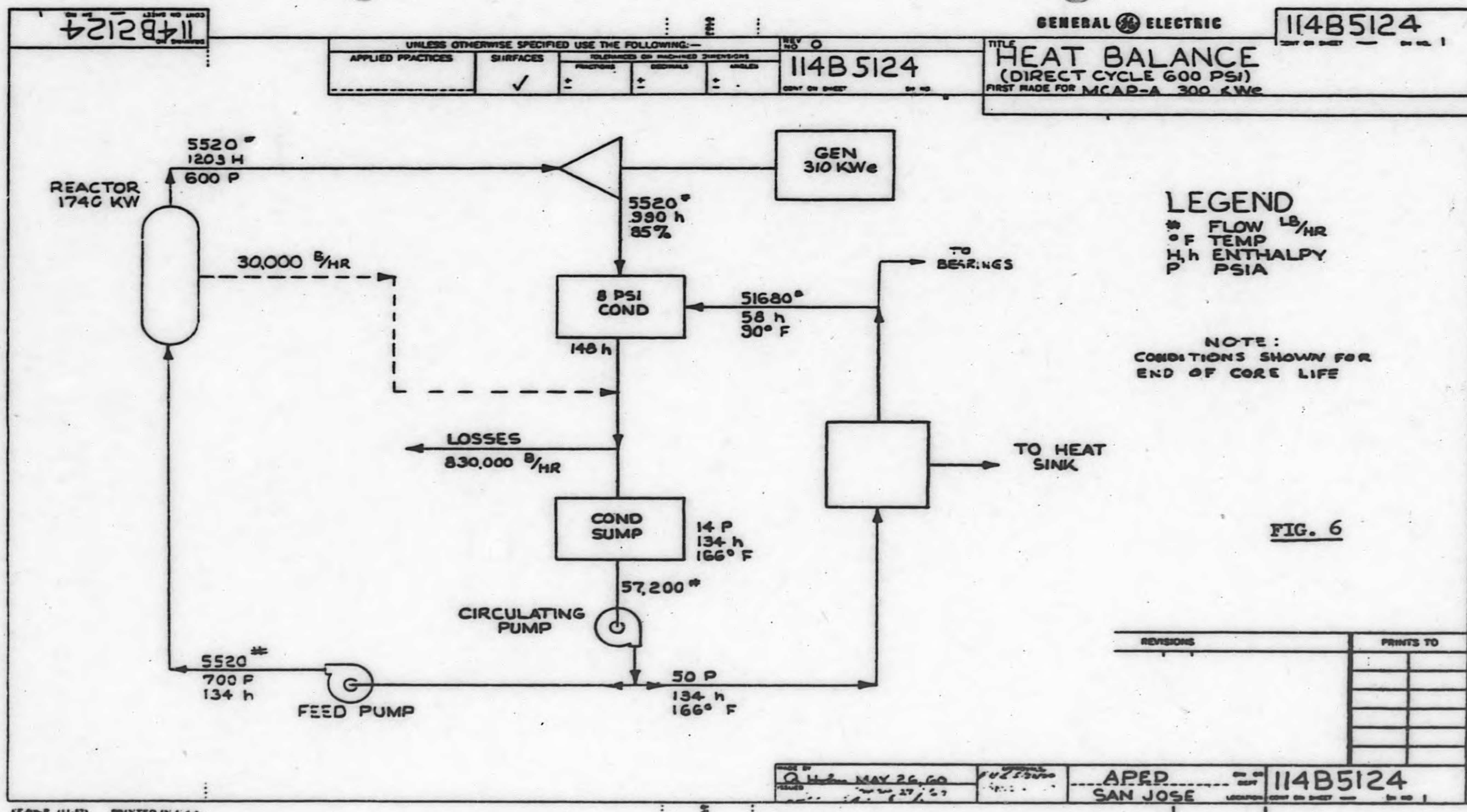
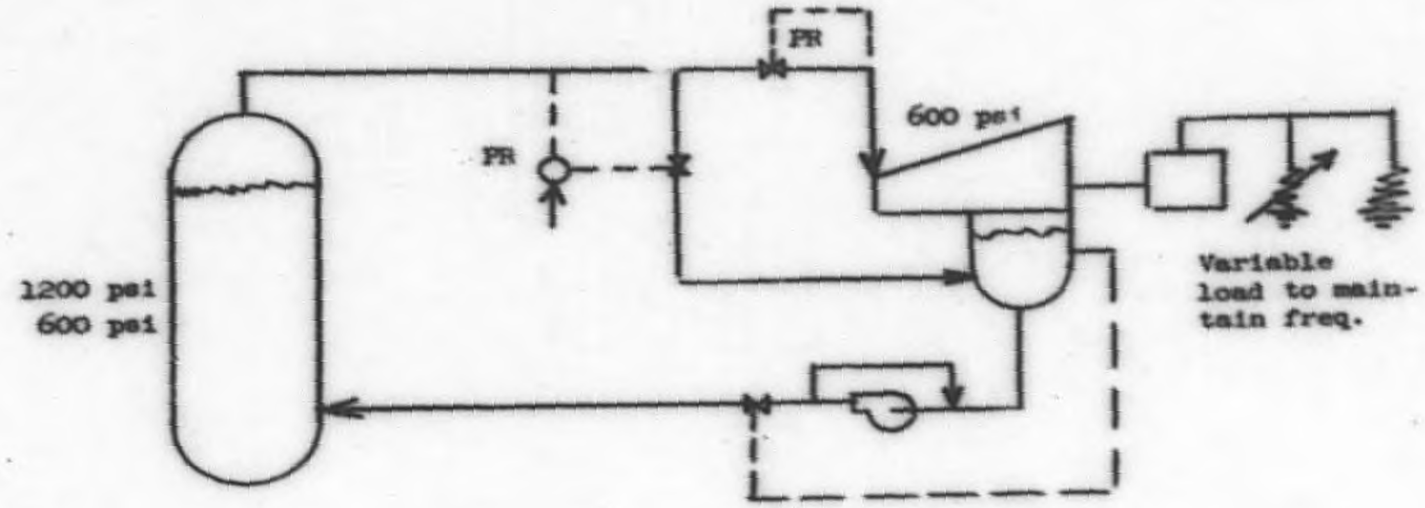
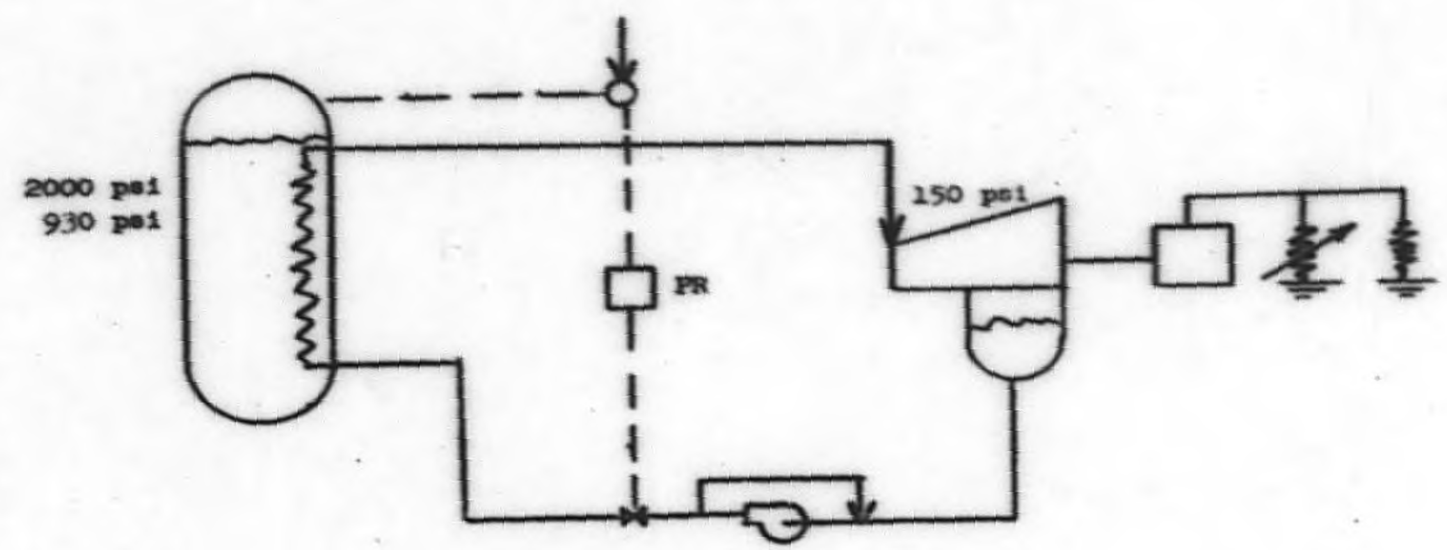

GENERAL ELECTRIC
114B5124
UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:—
APPLIED PRACTICES
SURFACES
TOLERANCES ON MACHINED DIMENSIONS
FRACTIONS
DECIMALS
ANGLES
REV NO 0
114B5124
HEAT BALANCE
(DIRECT CYCLE 600 PSI)
FIRST MADE FOR MCAP-A 300 KWe
5520 #
1203 H
600 P
REACTOR
1740 KW
GEN
310 KWe
5520 #
990 h
85%
LEGEND
FLOW LB/HR
°F TEMP
H, h ENTHALPY
P PSIA
30,000 B/HR
TO BEARINGS
8 PSI
COND
51680 #
58 h
90° F
148 h
NOTE:
CONDITIONS SHOWN FOR
END OF CORE LIFE
LOSSES
830,000 B/HR
TO HEAT
SINK
COND
SUMP
14 P
134 h
166° F
57,200 #
CIRCULATING
PUMP
REVISIONS
PRINTS TO
5520 #
700 P
134 h
FEED PUMP
50 P
134 h
166° F
MAY 26, 60
APED
SAN JOSE
114B5124
PRINTED IN U.S.A.

FIG. 6



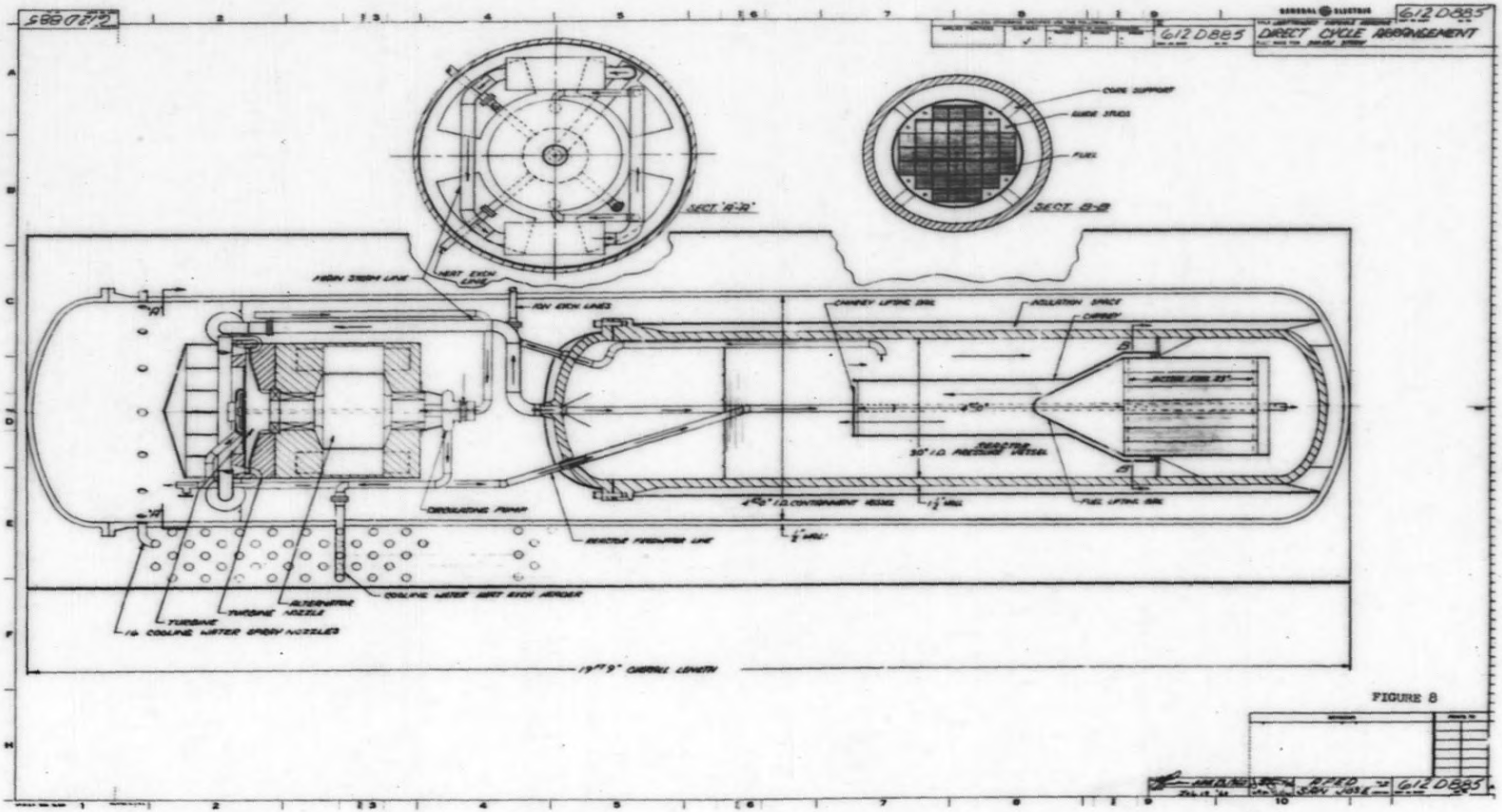
PR
PR
600 psi
1200 psi
600 psi
Variable
load to main-
tain freq.

Fig. 7A



2000 psi
930 psi
150 psi
PR

Fig. 7B



612D885
GENERAL ELECTRIC
DIRECT CYCLE ARRANGEMENT
SECT 'A-A'
SECT 'B-B'
CORE SUPPORT
GUIDE STUDS
FUEL
MAIN STEAM LINE
HEAT EXCH LINES
ION EXCH LINES
INSULATION SPACE
CHIMNEY LIFTING BAIL
CHIMNEY
CIRCULATING PUMP
REACTOR FEEDWATER LINE
FUEL LIFTING BAIL
COOLING WATER HEAT EXCH HEADER
ALTERNATOR
TURBINE NOZZLE
TURBINE
16 COOLING WATER SPRAY NOZZLES
19'-9" OVERALL LENGTH
APED
SAN JOSE

FIGURE 8

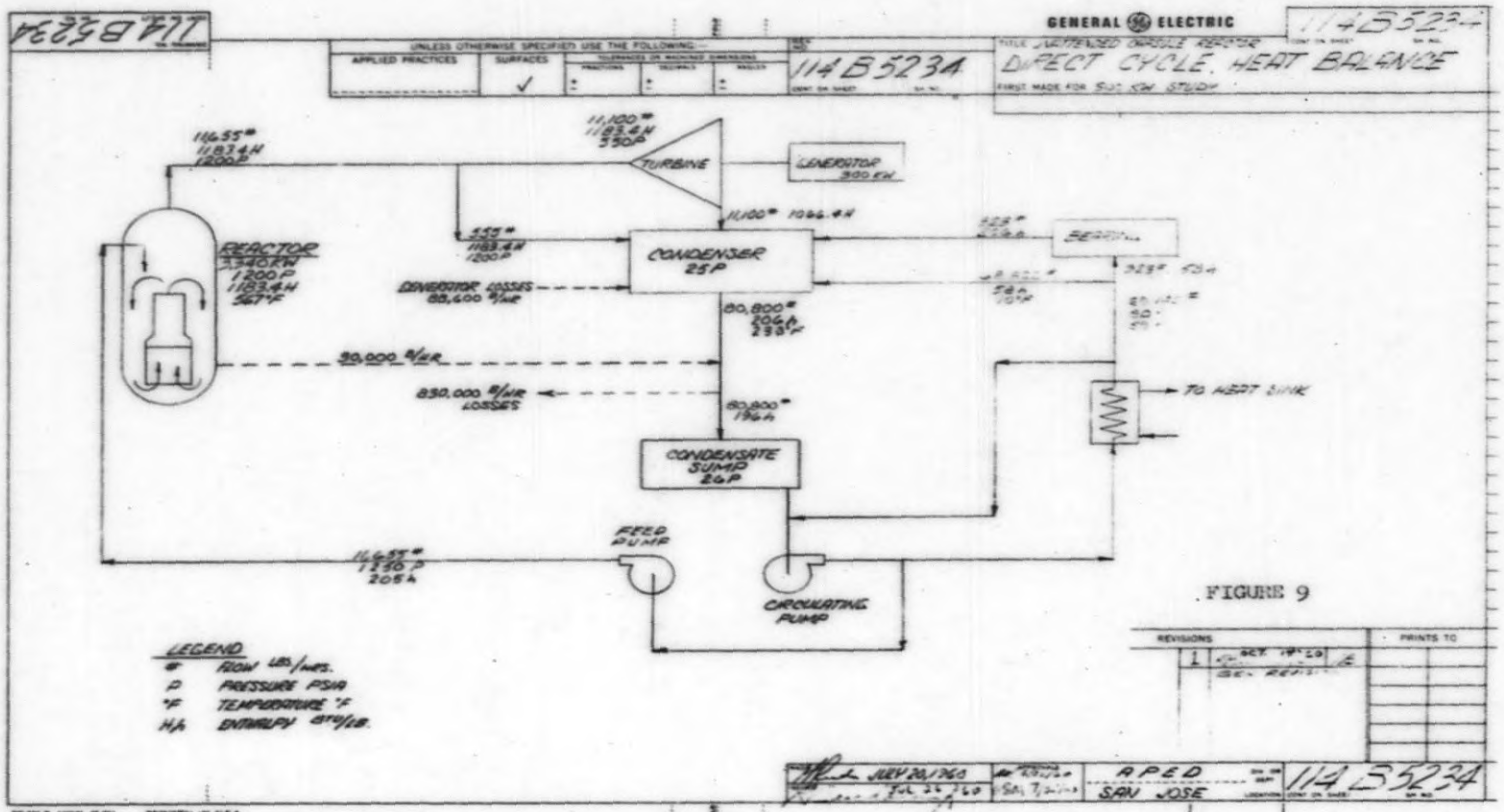

GENERAL ELECTRIC
114B5234
UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING
APPLIED PRACTICES
SURFACES
114B5234
DIRECT CYCLE HEAT BALANCE
TURBINE
GENERATOR 300 KW
REACTOR
3340 KW
1200 P
1183.4 H
567°F
11,655 #
1183.4 H
1200 P
11,100 #
1183.4 H
330 P
11,100 # 1066.4 H
555 #
1183.4 H
1200 P
CONDENSER 25 P
GENERATOR LOSSES 88,600 #/HR
80,800 #
206 h
239°F
30,000 #/HR
830,000 #/HR LOSSES
80,800 #
196 h
CONDENSATE SUMP 26 P
TO HEAT SINK
FEED PUMP
CIRCULATING PUMP
11,655 #
1250 P
205 h
LEGEND
FLOW LBS/HRS.
P PRESSURE PSIA
°F TEMPERATURE °F
H/h ENTHALPY BTU/LB.
REVISIONS
PRINTS TO
JULY 20, 1960
APED
SAN JOSE
114B5234
PRINTED IN U.S.A.

FIGURE 9

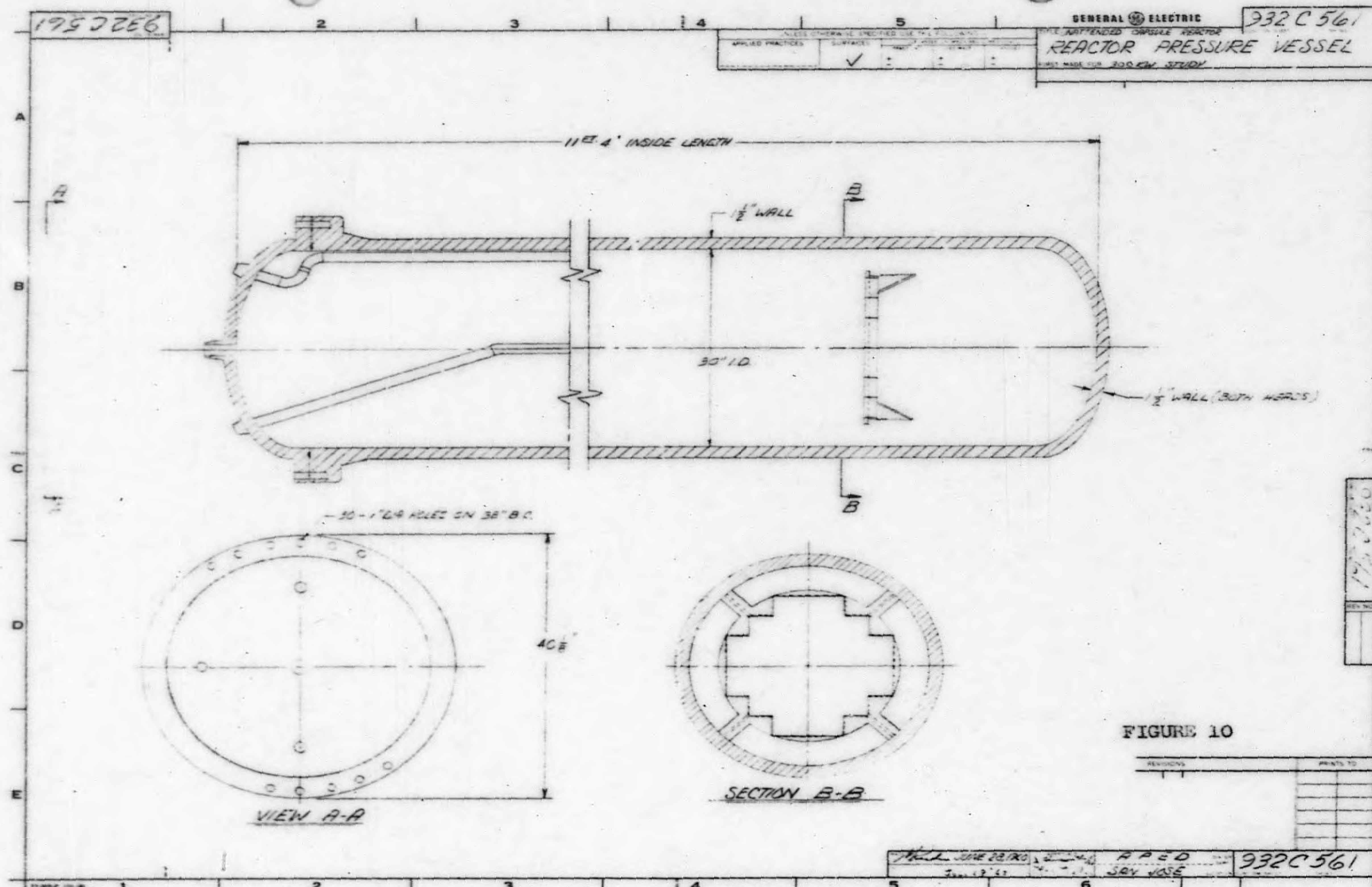

932C561
GENERAL ELECTRIC
UNATTENDED CAPSULE REACTOR
REACTOR PRESSURE VESSEL
FIRST MADE FOR 300 KW STUDY
11 FT. 4" INSIDE LENGTH
1 1/2" WALL
30" I.D.
1 1/2" WALL (BOTH HEADS)
30 - 1" DIA HOLES ON 38" B.C.
40 1/2"
VIEW A-A
SECTION B-B
A.P.E.D.
SAN JOSE
932C561

FIGURE 10

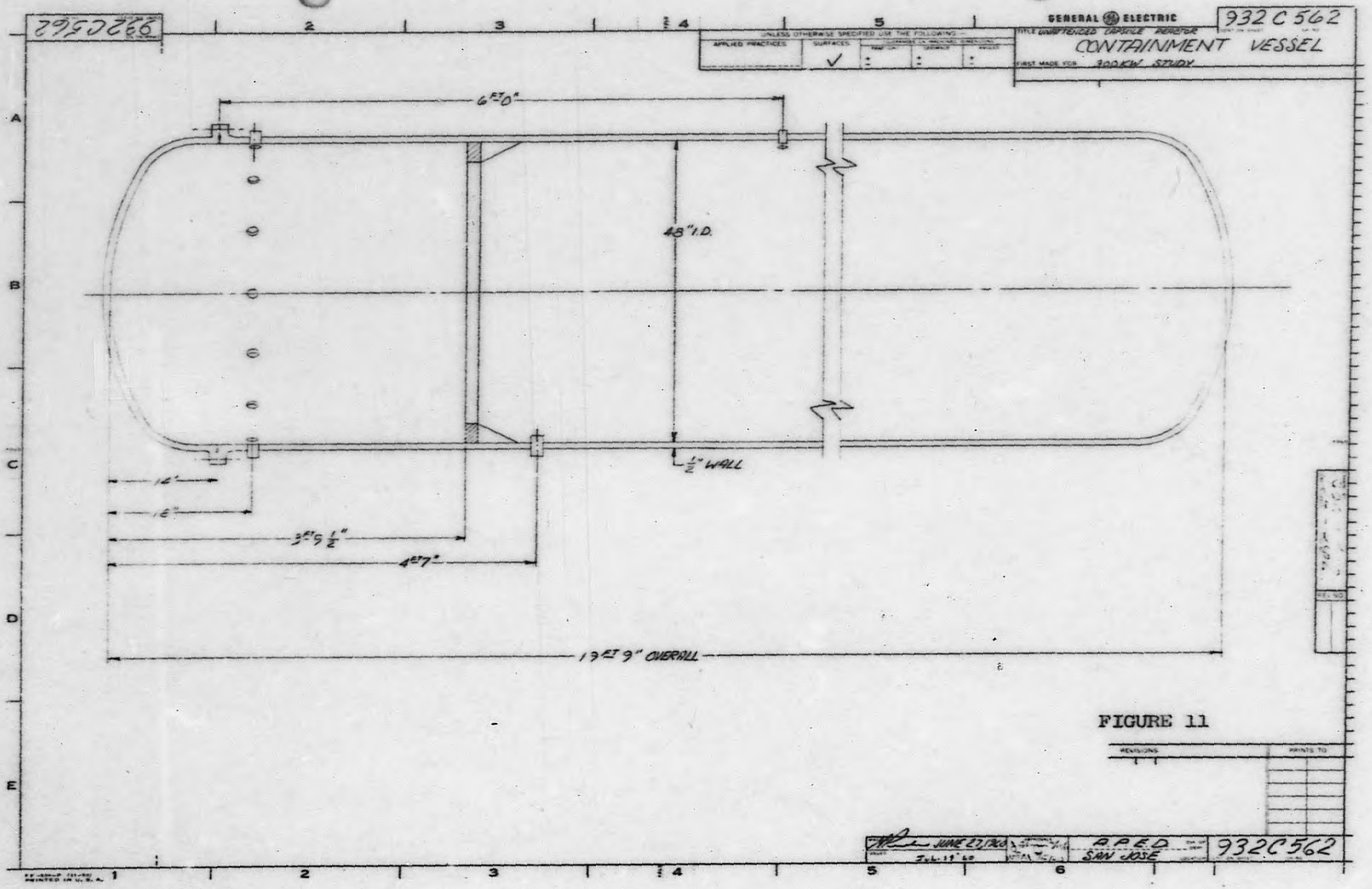
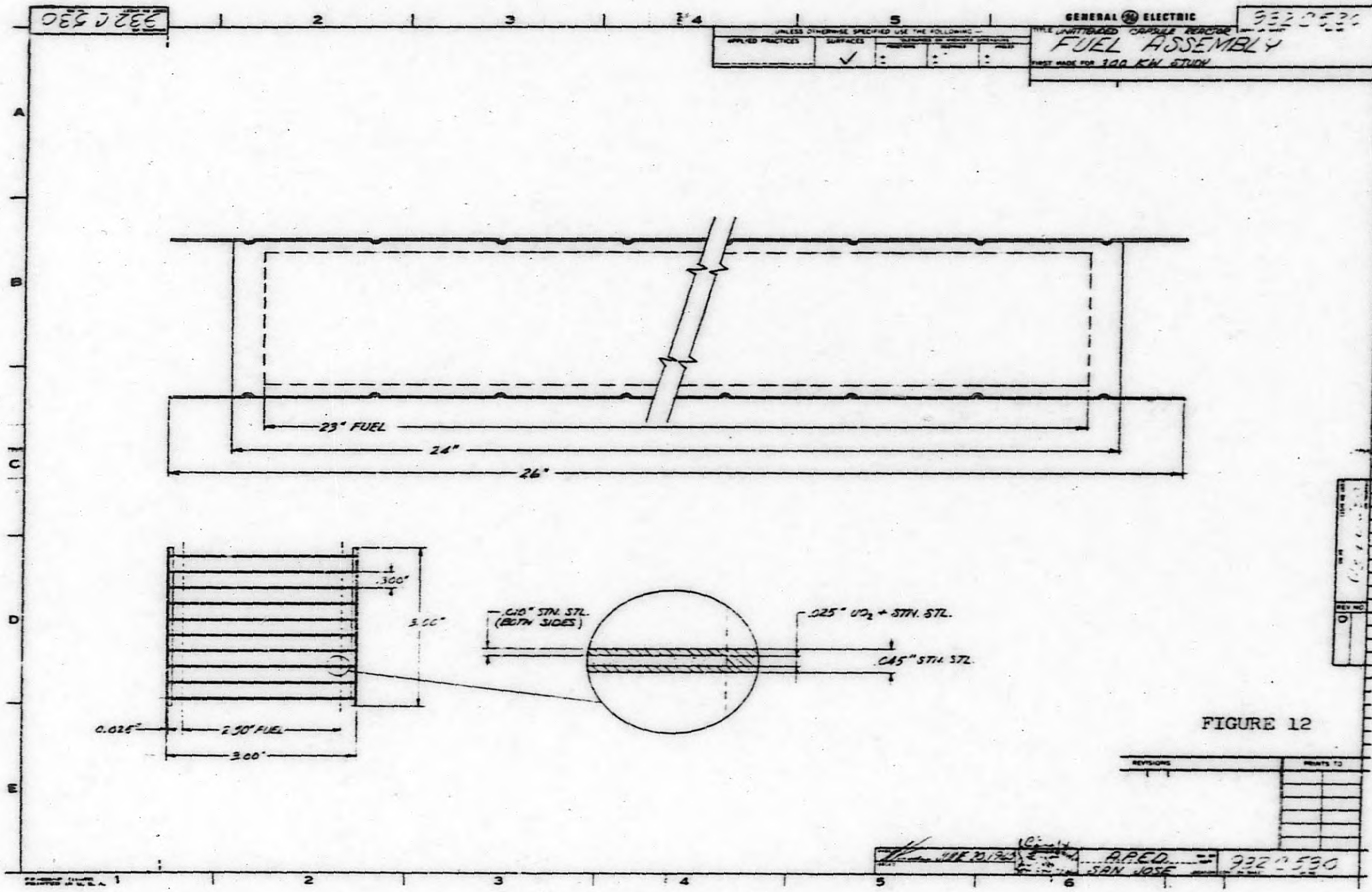

GENERAL ELECTRIC
932C562
CONTAINMENT VESSEL
300KW STUDY
6FT 0"
48" I.D.
1/2" WALL
14"
18"
3FT 9 1/2"
4FT 7"
17FT 9" OVERALL
JUNE 27 1960
A.P.E.D.
SAN JOSE
932C562

FIGURE 11



GENERAL ELECTRIC
UNATTENDED CAPSULE REACTOR
FUEL ASSEMBLY
FIRST MADE FOR 300 KW STUDY
23" FUEL
24"
26"
.300"
3.00"
.010" STN. STL. (BOTH SIDES)
.025" UO_2 + STN. STL.
.045" STN. STL.
0.025"
2.50" FUEL
3.00"
A.P.E.D.
SAN JOSE
932C530

FIGURE 12

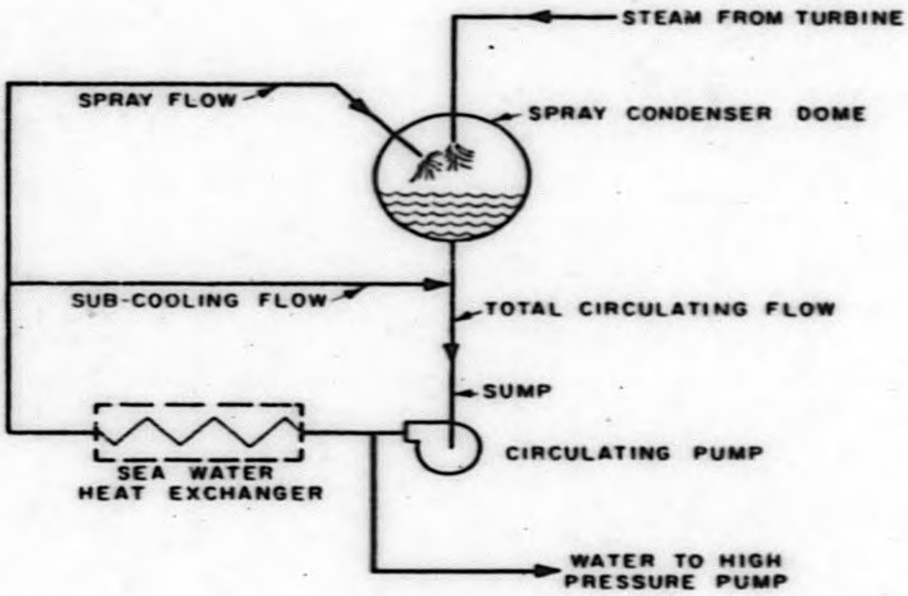


FIGURE 13

SCHEMATIC OF HEAT REJECTION SYSTEM

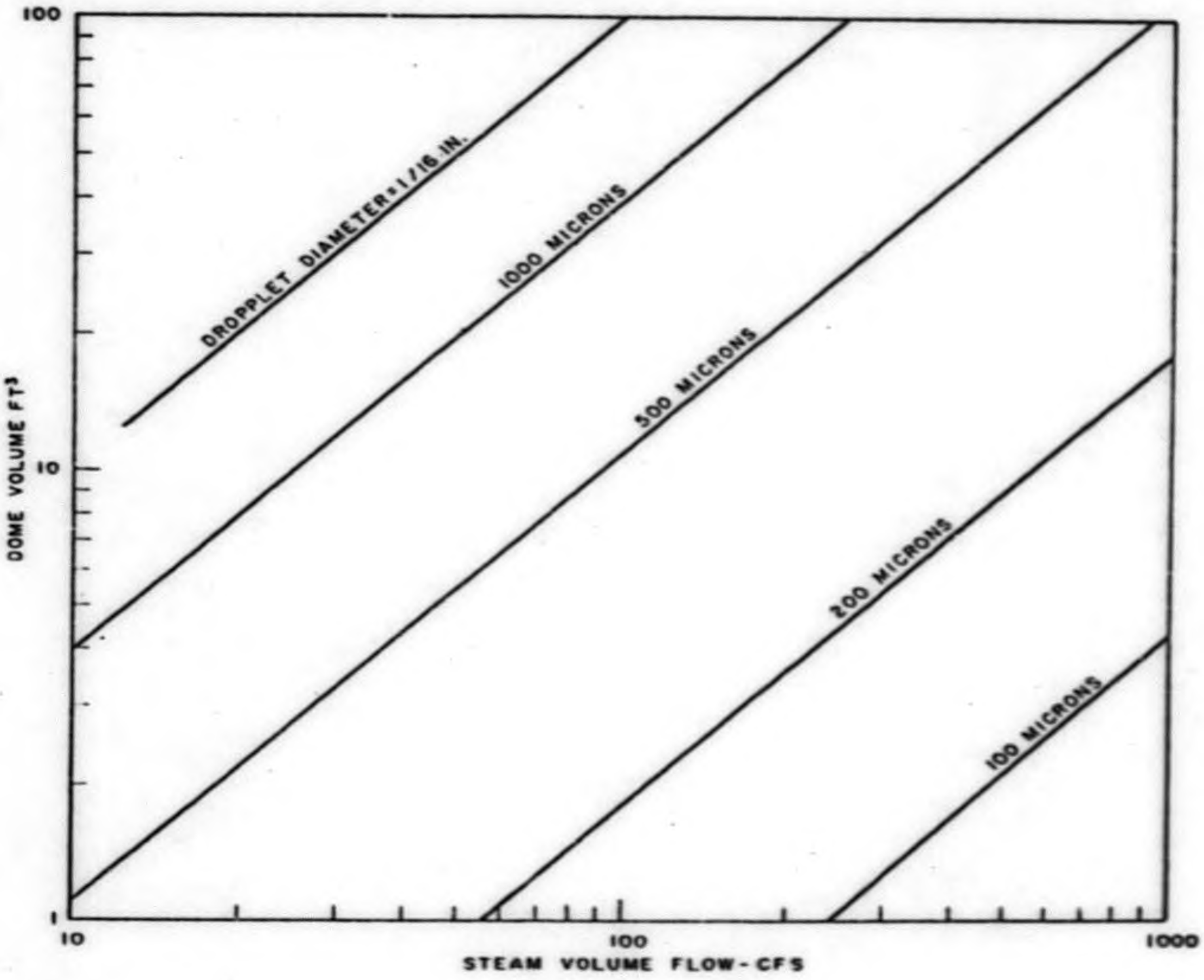


FIGURE 14

THEORETICAL DOME VOLUME FOR SPRAY CONDENSER

FOR: $\frac{\text{TEMPERATURE RISE OF WATER}}{\text{INITIAL WATER-TO-STEAM TEMPERATURE DIFFERENTIAL}} = 0.95$

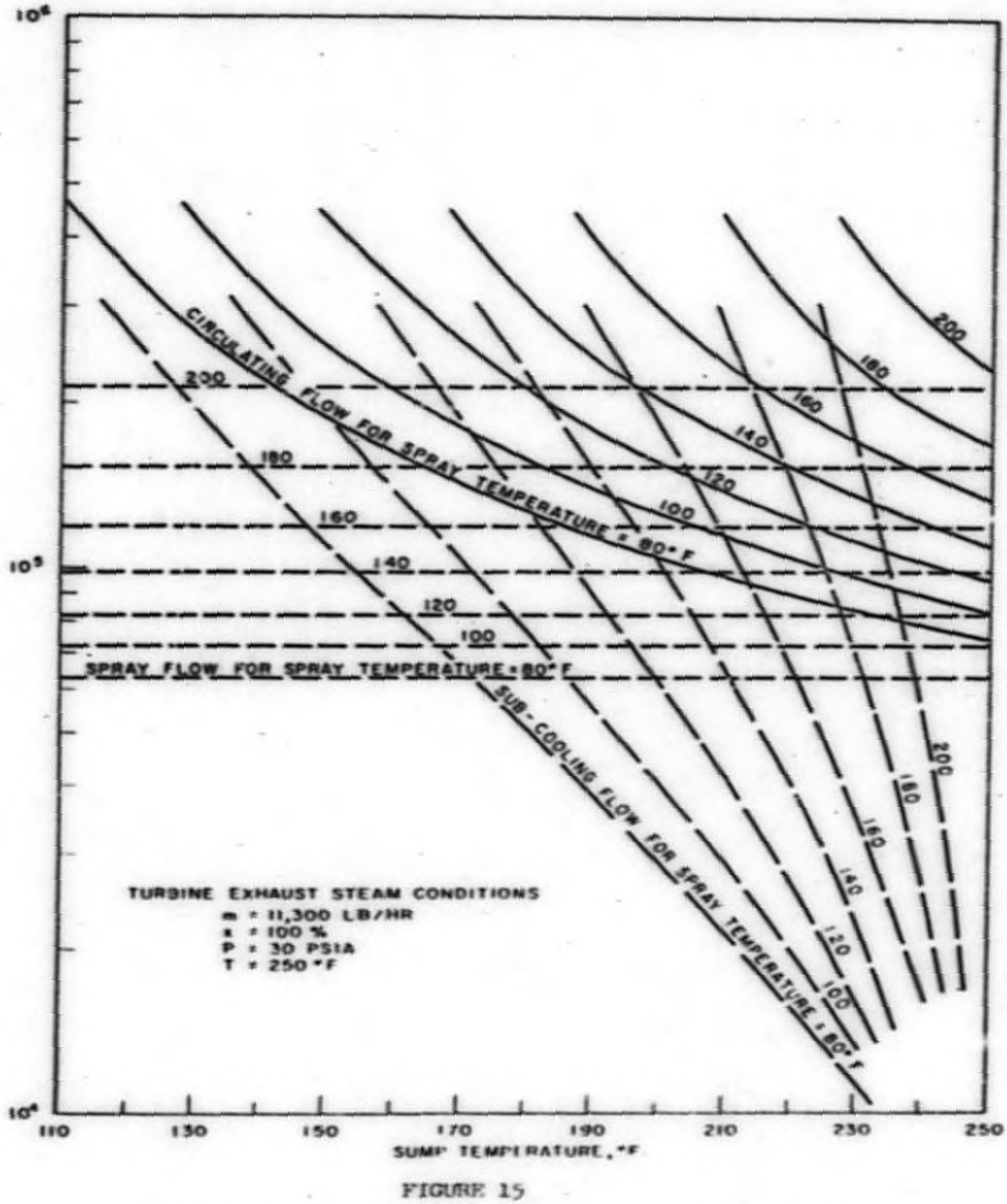


FIGURE 15

EFFECT OF SUB-COOLING ON TOTAL CIRCULATING FLOW

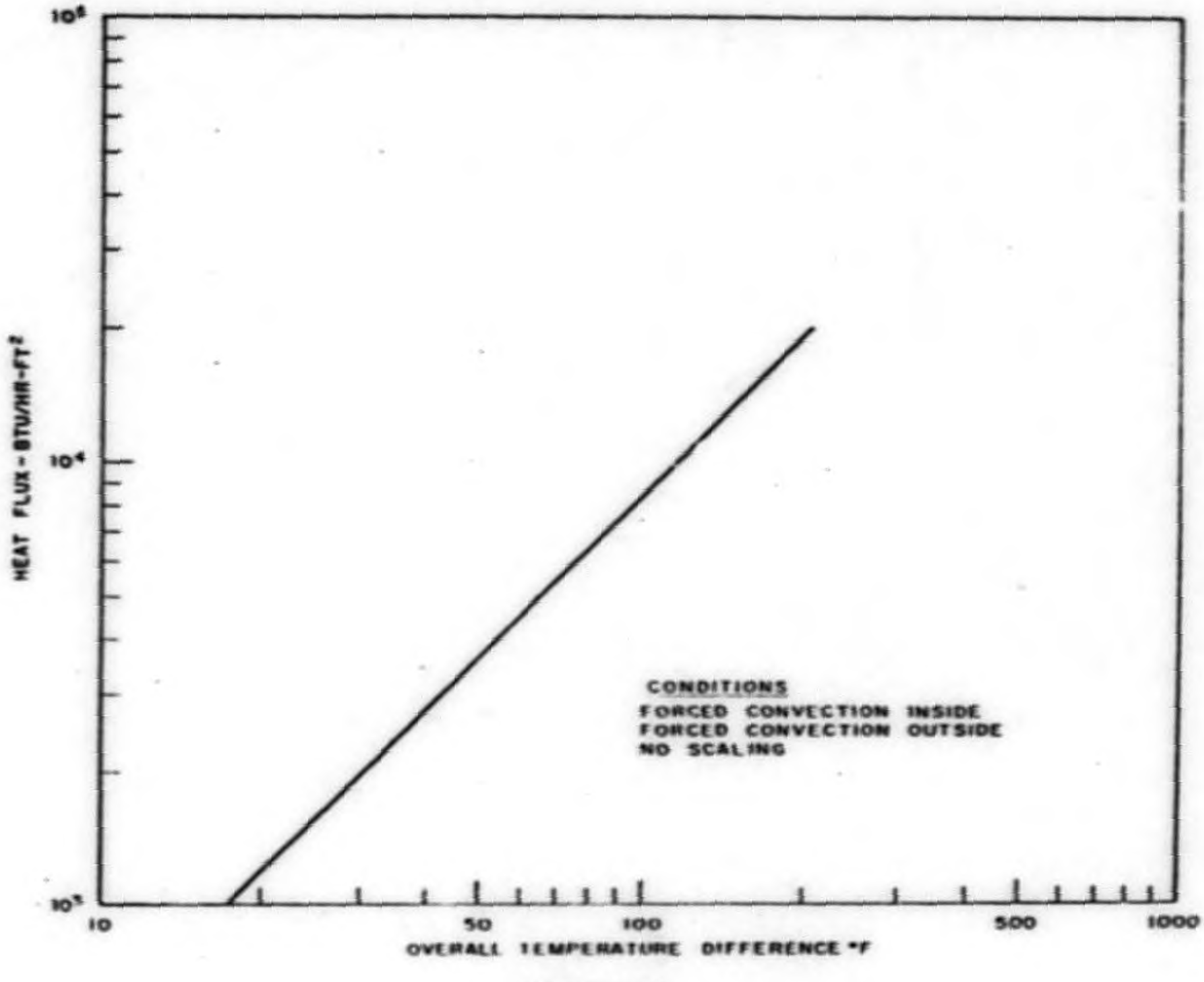


FIGURE 16

HEAT FLUX vs TEMPERATURE DIFFERENCE FOR SEA WATER HEAT EXCHANGER

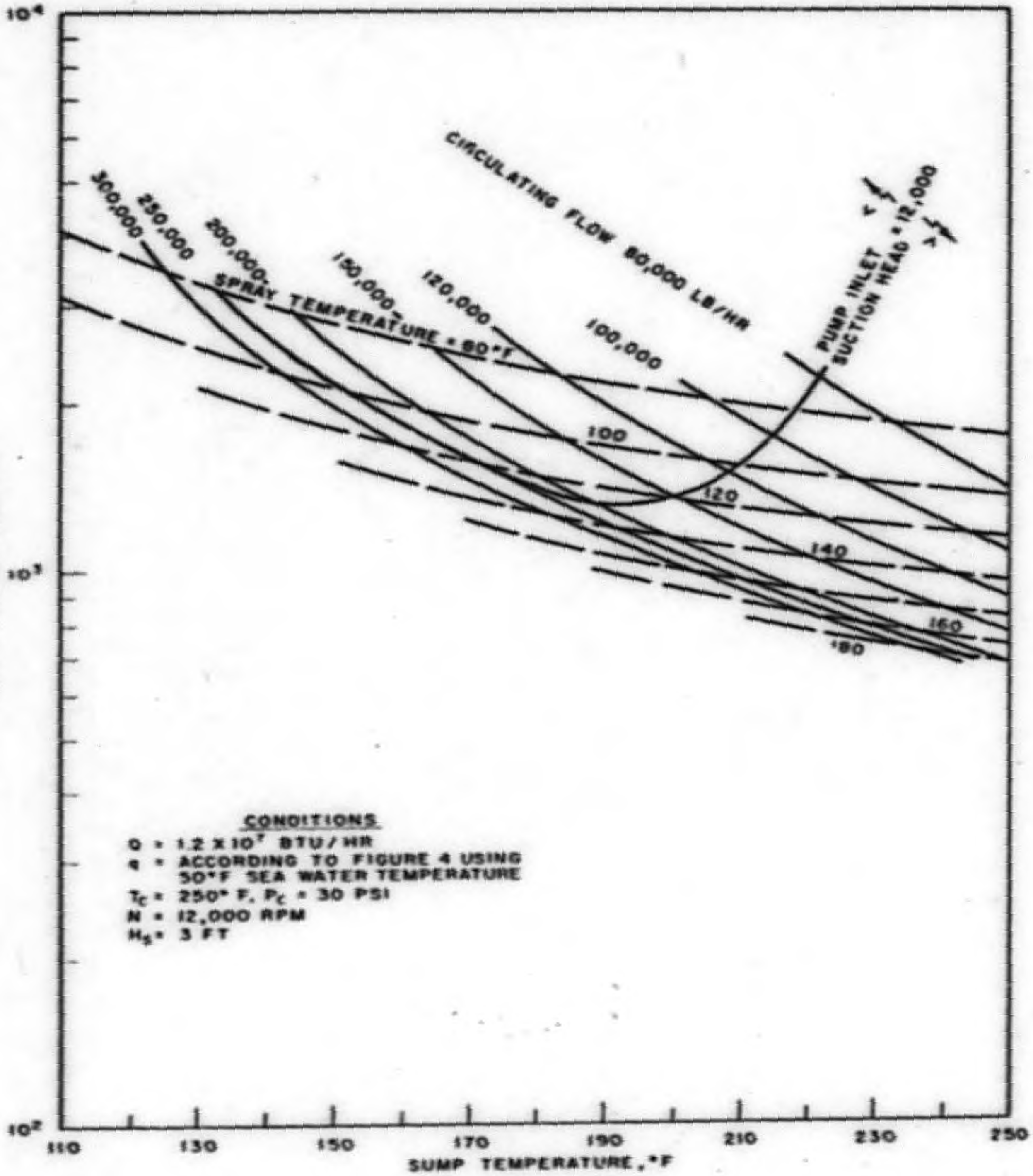


FIGURE 17

MINIMUM HEAT TRANSFER AREA FOR CIRCULATING PUMP INLET SUCTION HEAD OF 12,000

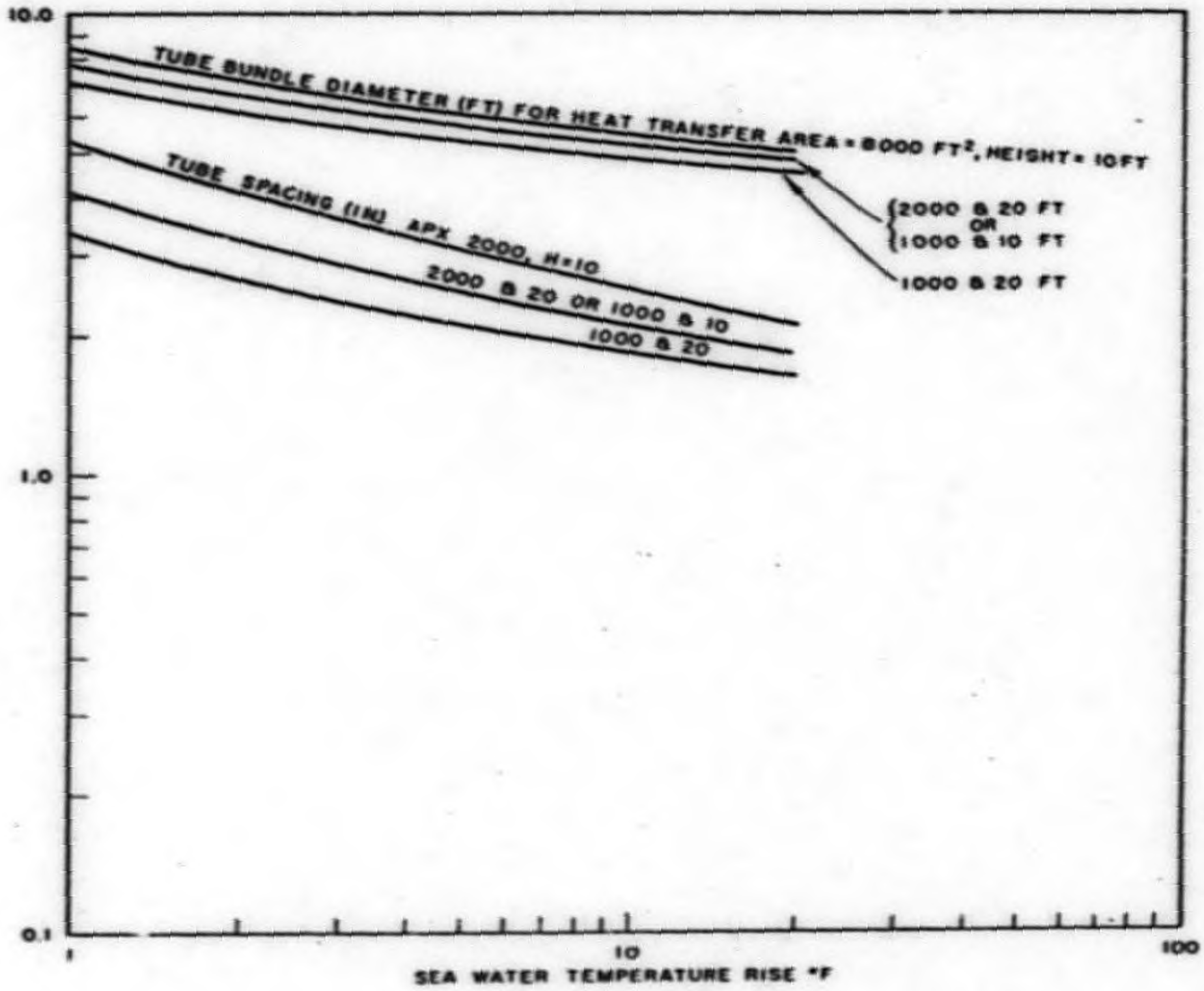


FIGURE 18

SIZE OF SEA WATER HEAT EXCHANGER

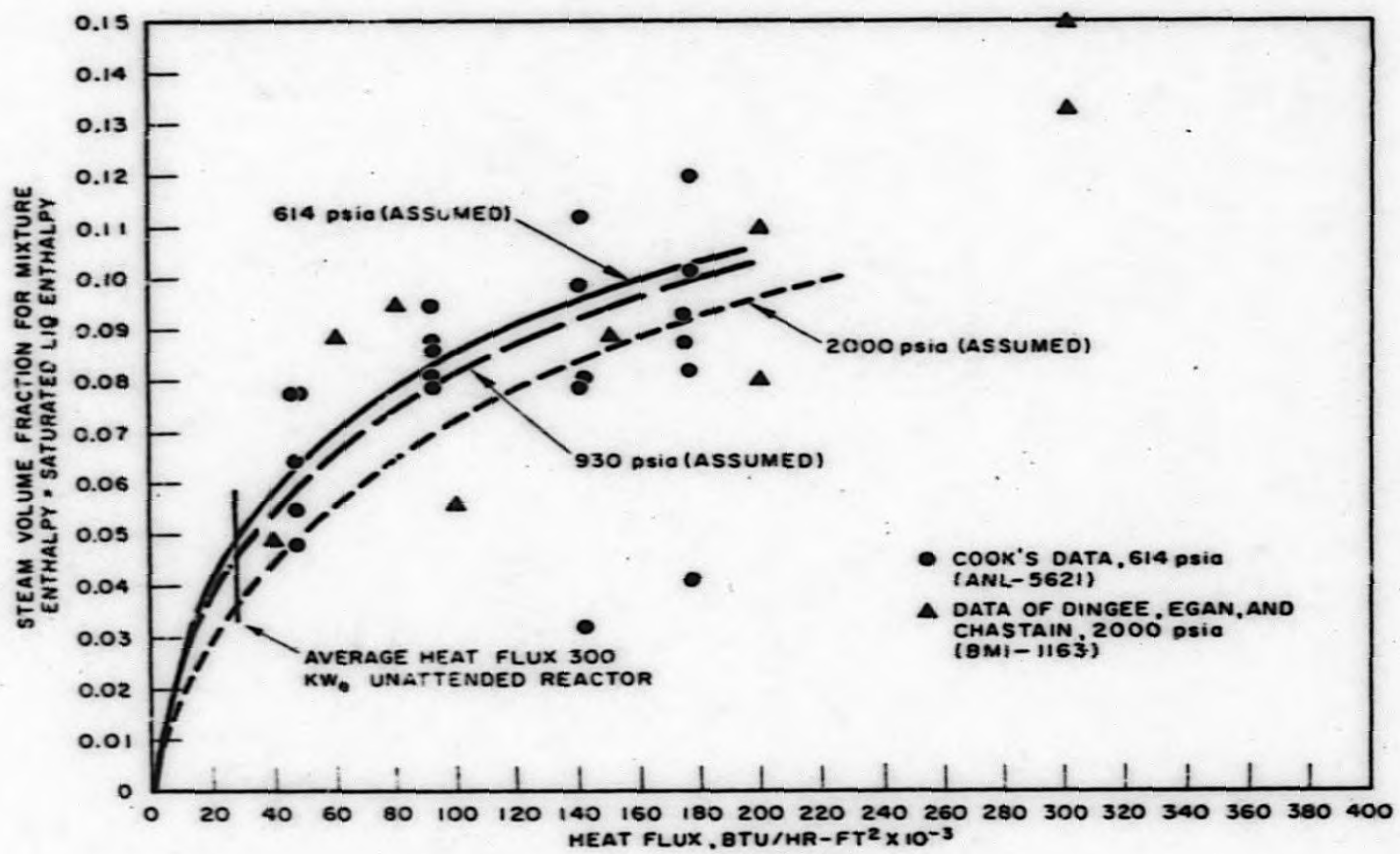


FIGURE 19

PRELIMINARY DESIGN ESTIMATE OF STEAM VOLUME FRACTION AT SATURATED BULK ENTHALPY

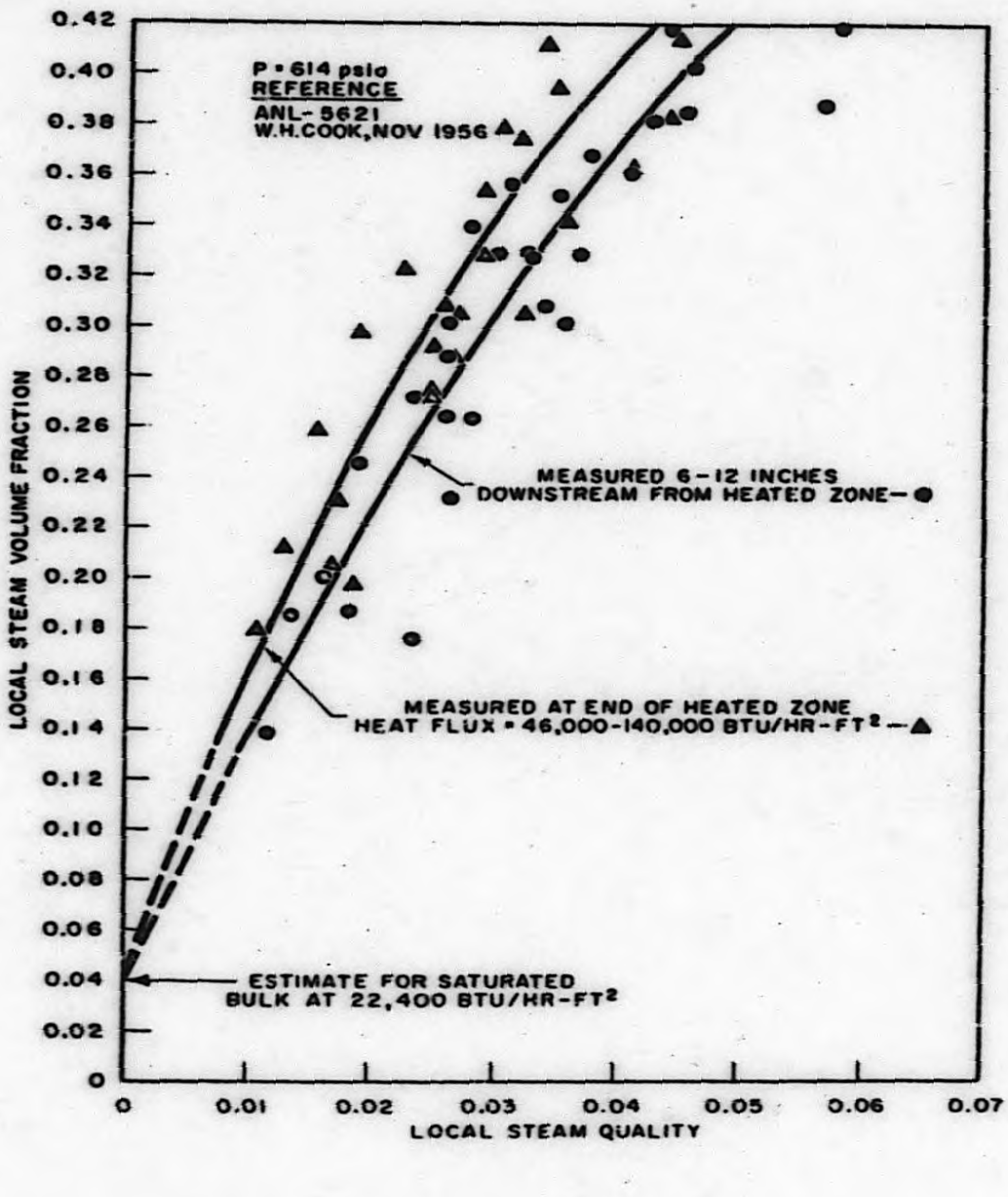


FIGURE 20 PRELIMINARY DESIGN ESTIMATE OF STEAM VOLUME FRACTION

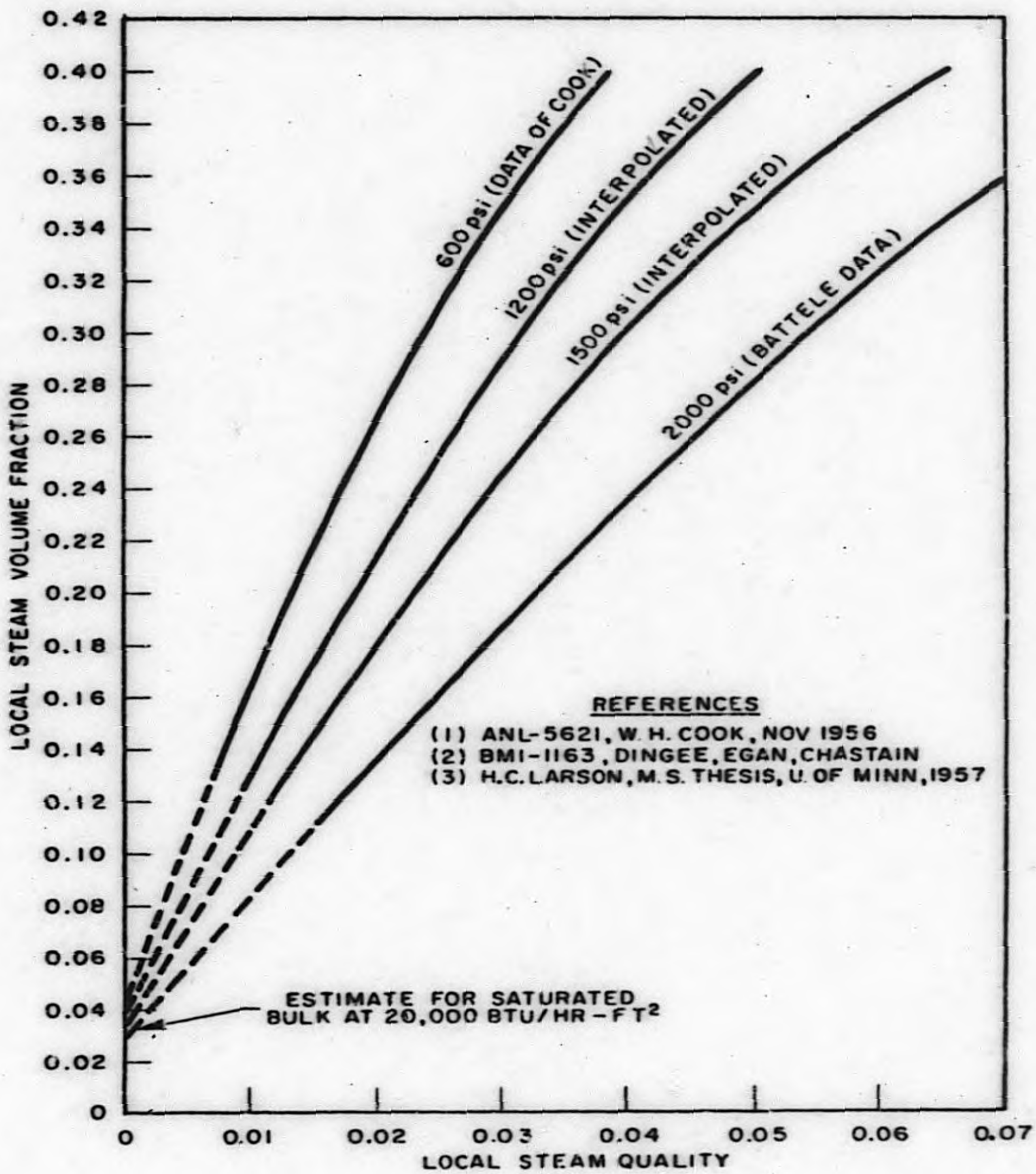


FIGURE 21

PRELIMINARY DESIGN ESTIMATE OF STEAM VOLUME FRACTION

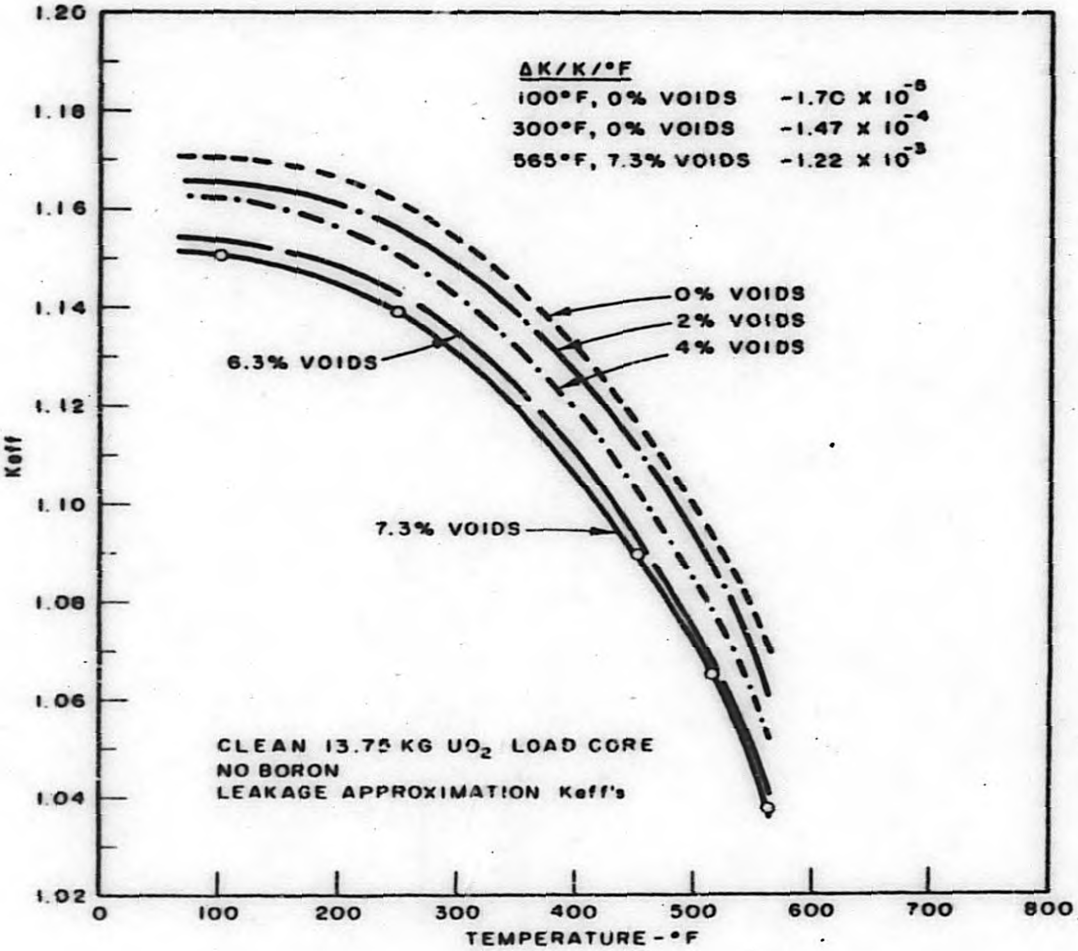


FIGURE 22

Keff vs TEMPERATURE

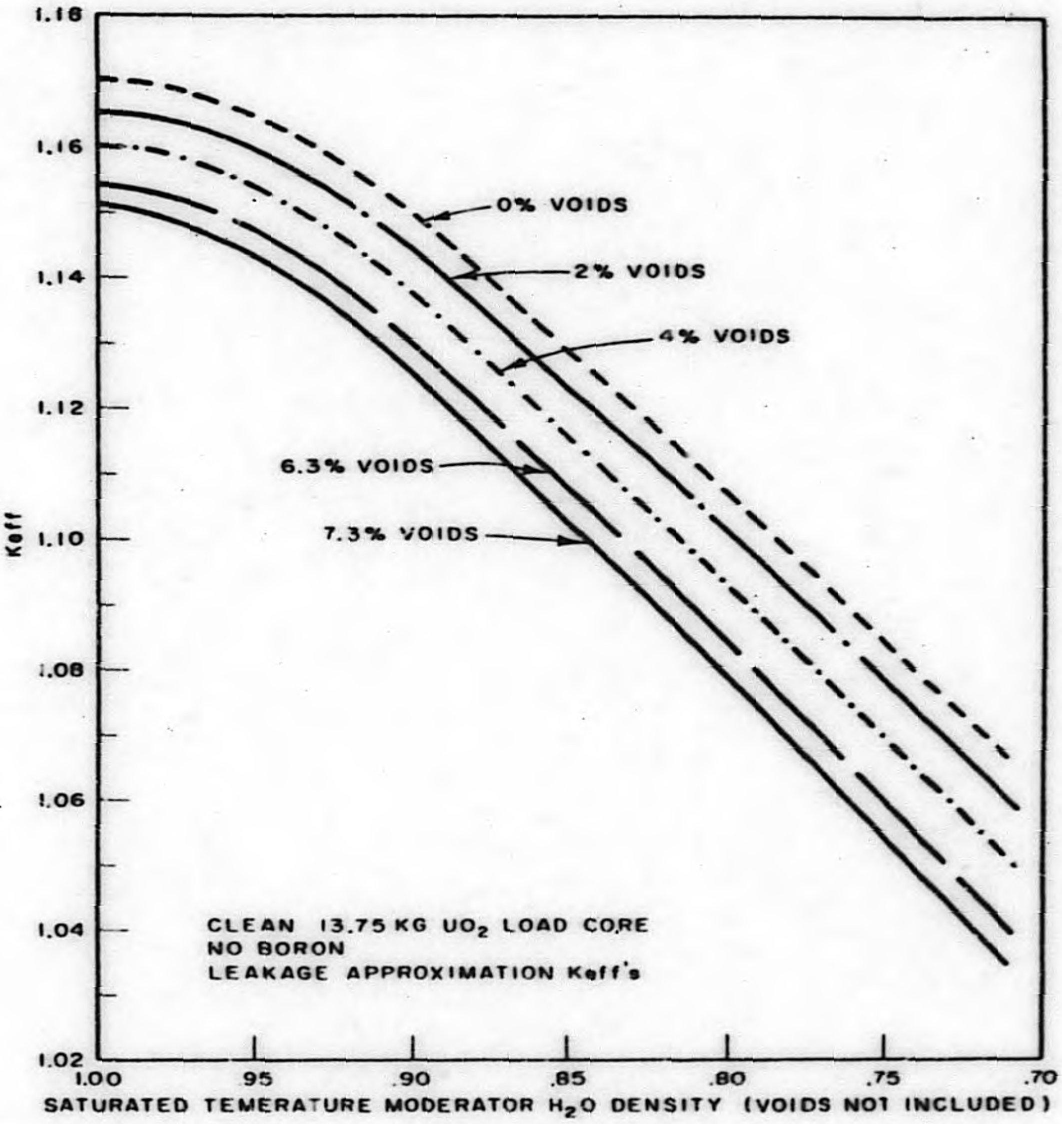


FIGURE 23

Keff VS DENSITY OF MODERATOR

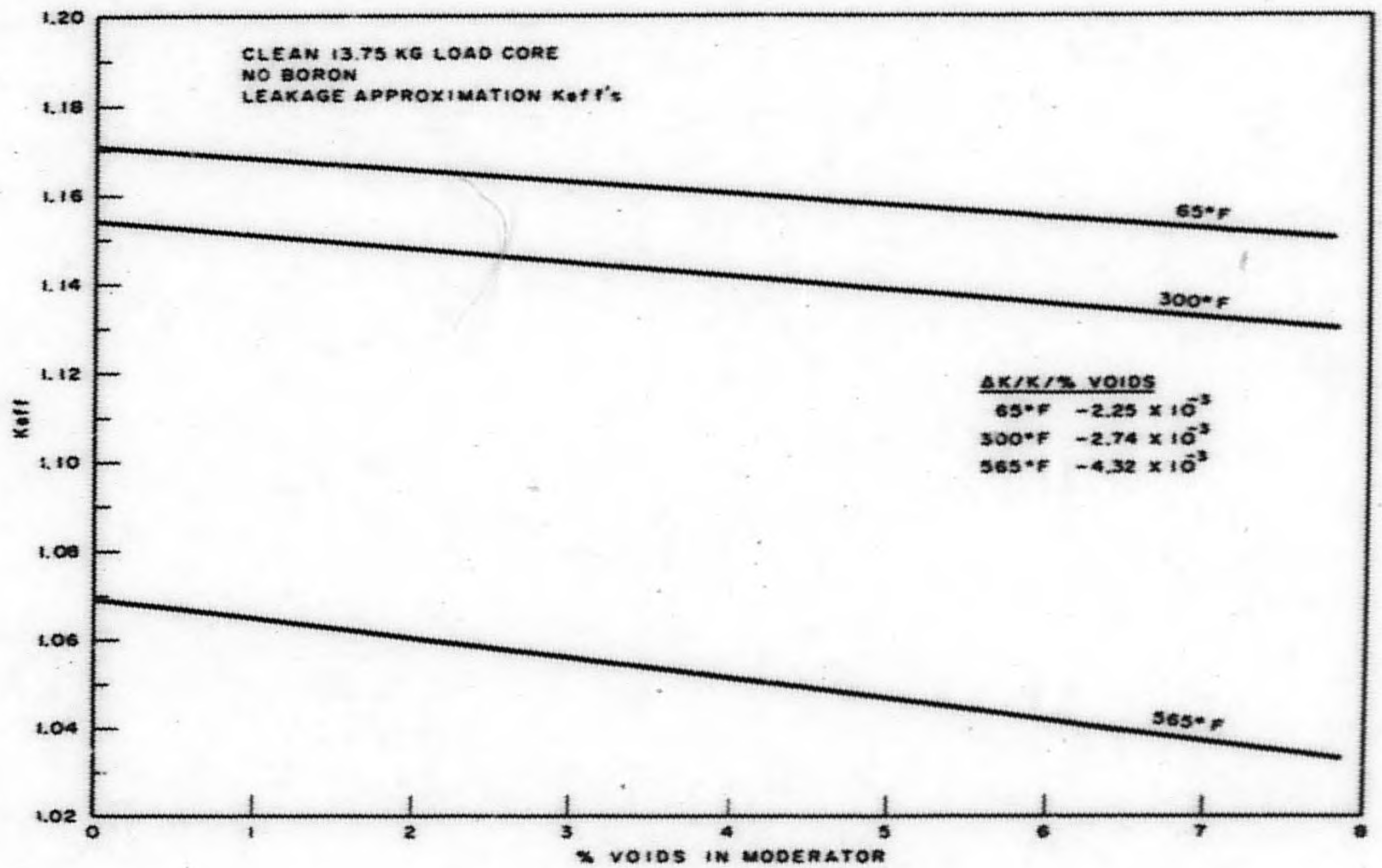


FIGURE 24

Keff vs % VOIDS IN MODERATOR

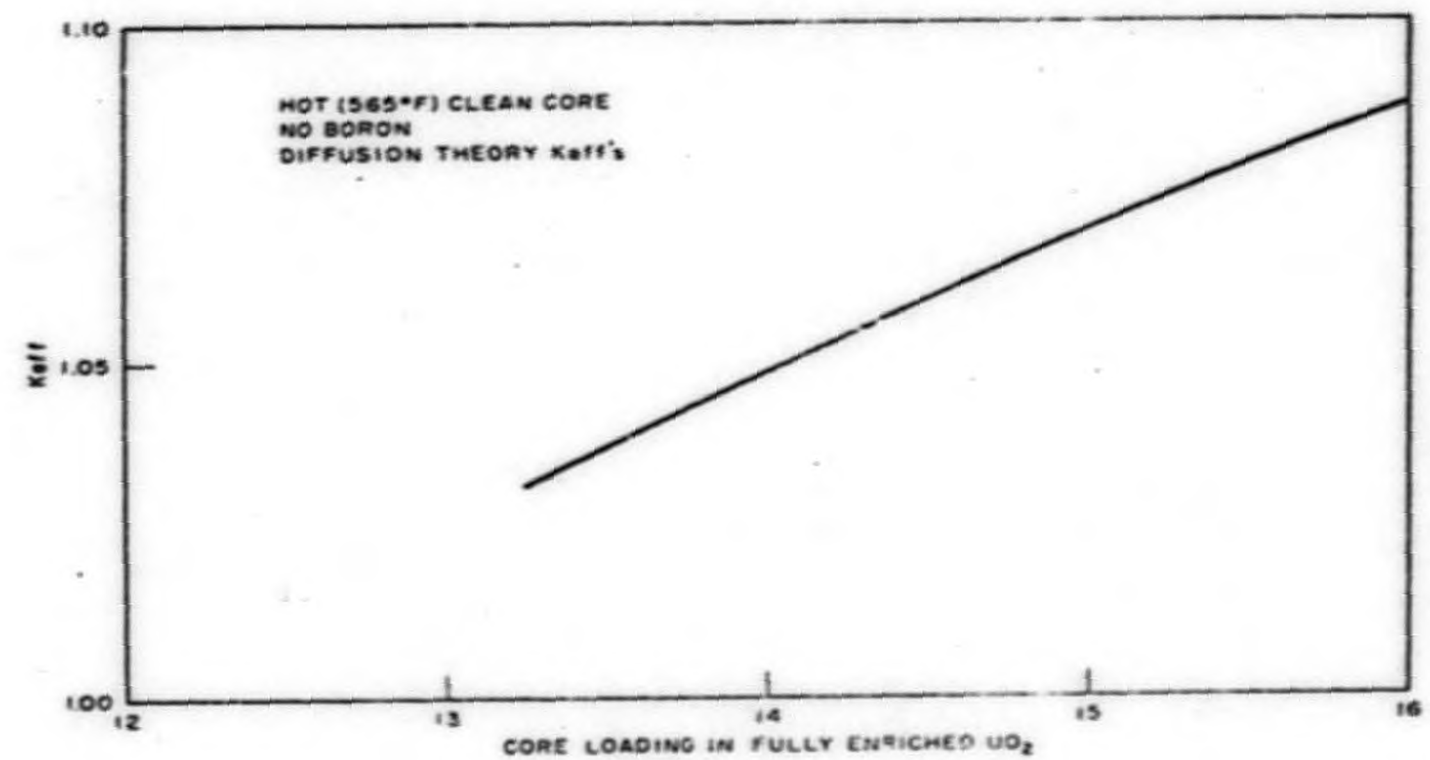


FIGURE 25

Keff VS LOADING 7.3% VOIDS

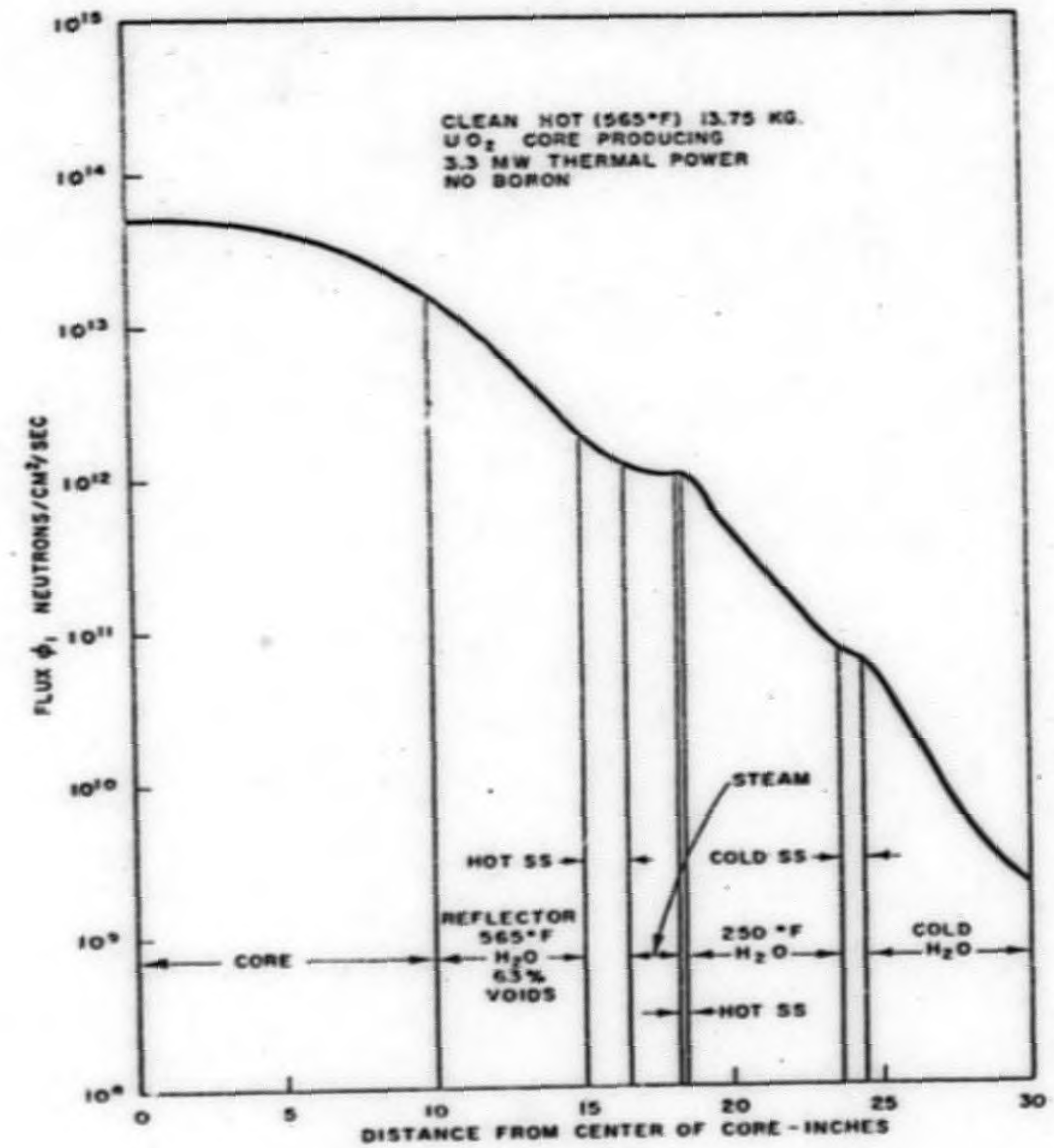


FIGURE 26

FLUX ϕ_1 VS DISTANCE FROM CENTER OF CORE

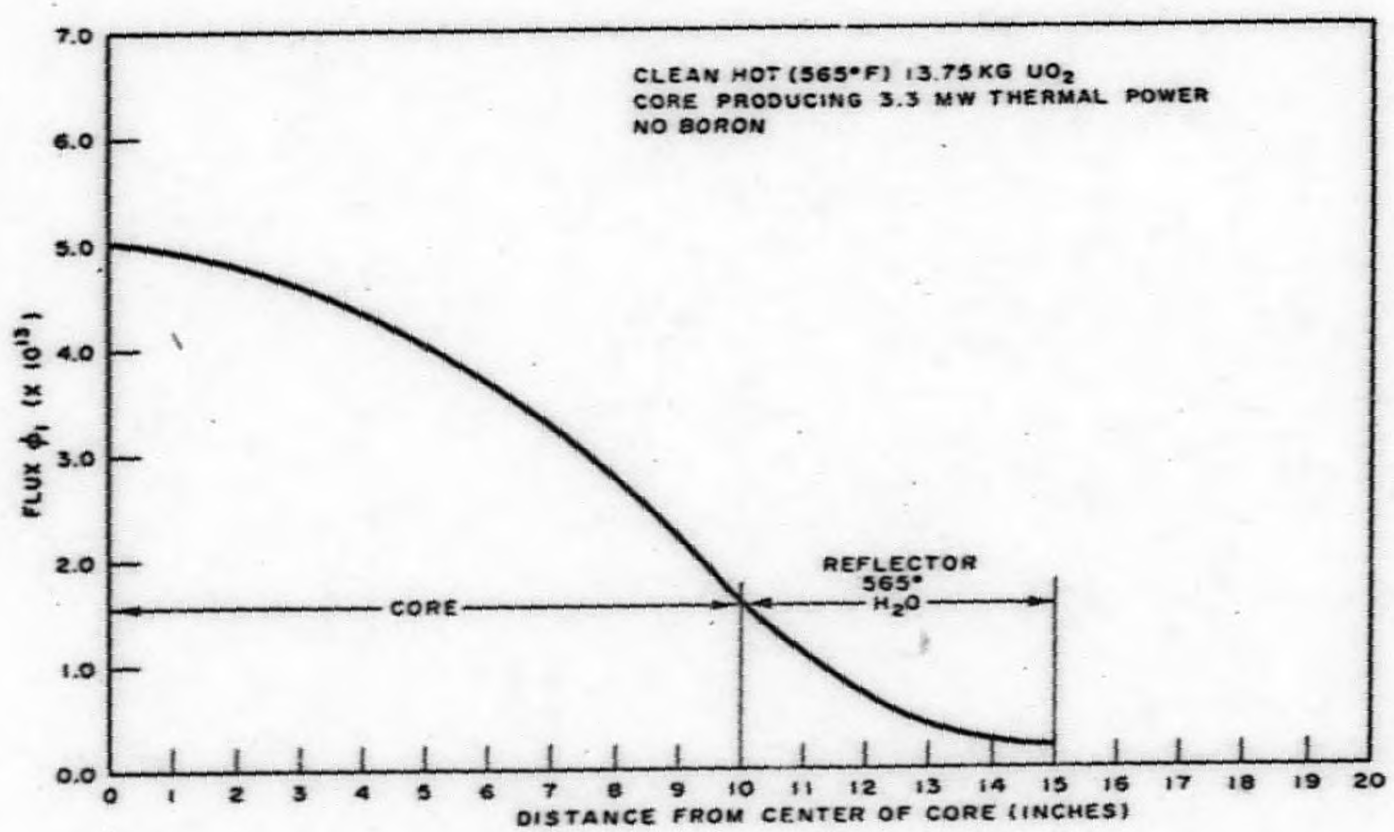


FIGURE 27

FAST FLUX ϕ_1 vs DISTANCE FROM CORE CENTER

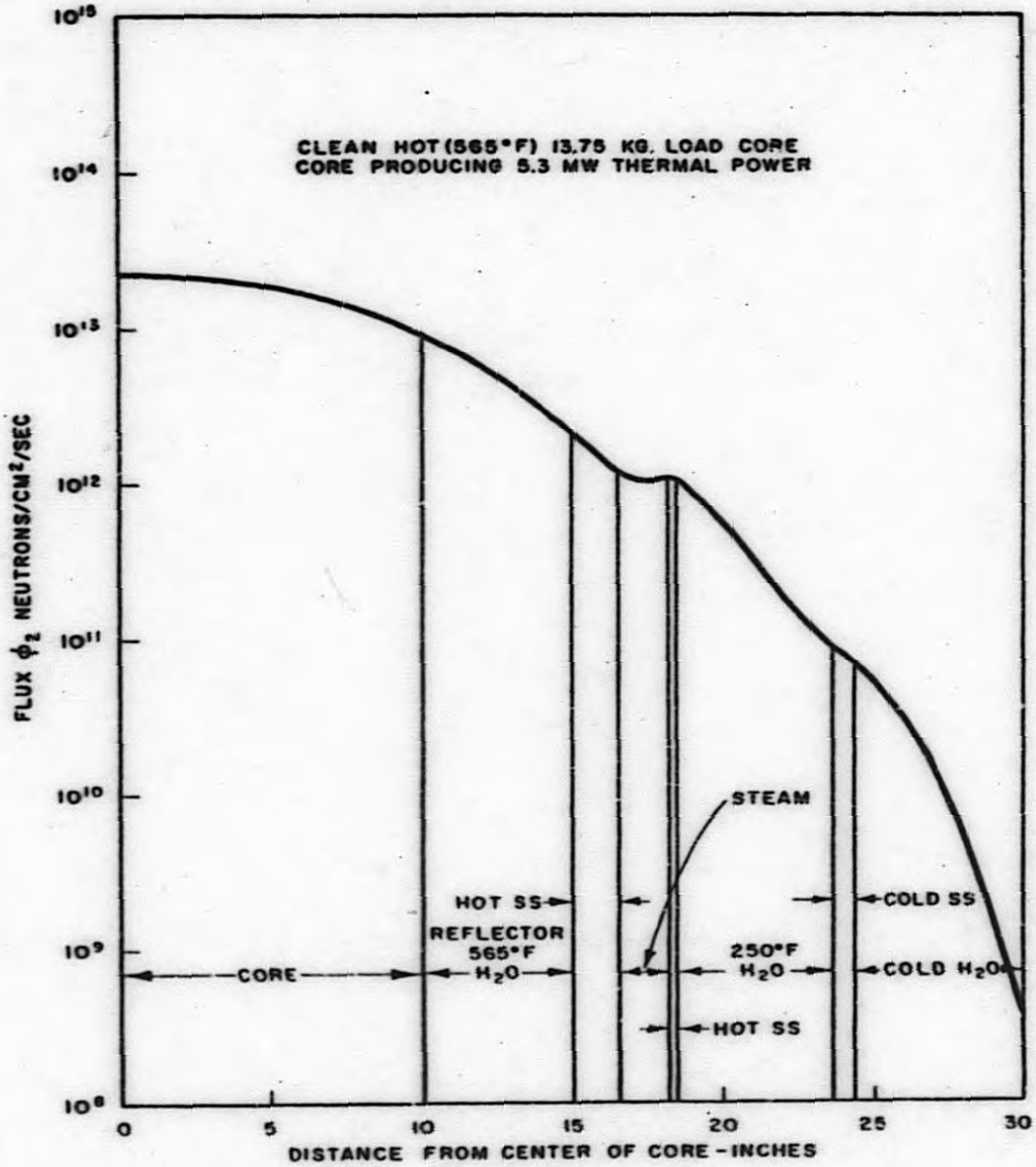


FIGURE 28

FLUX ϕ_2 VS DISTANCE FROM CORE

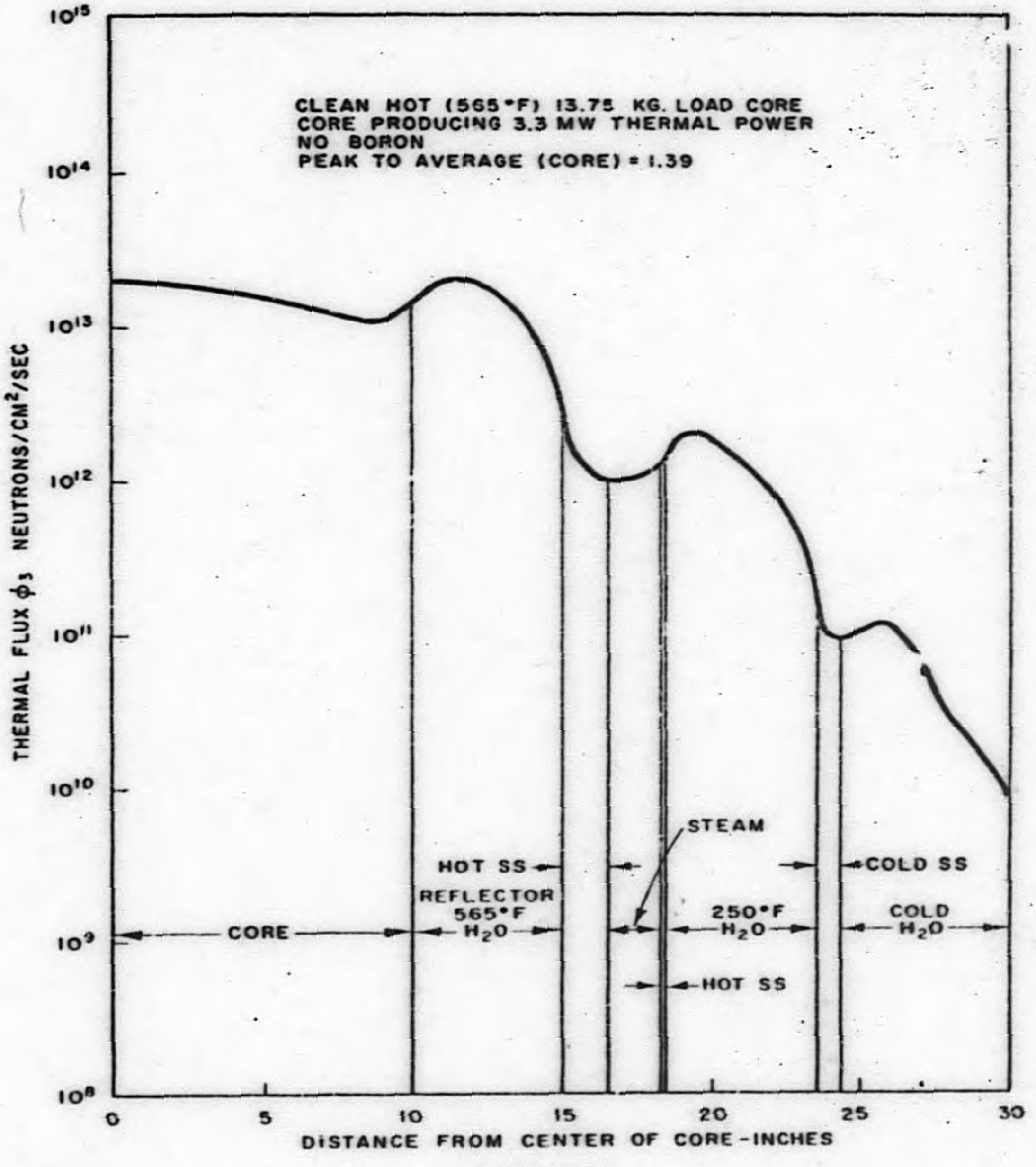


FIGURE 29

THERMAL FLUX ϕ_3 VS DISTANCE FROM CORE

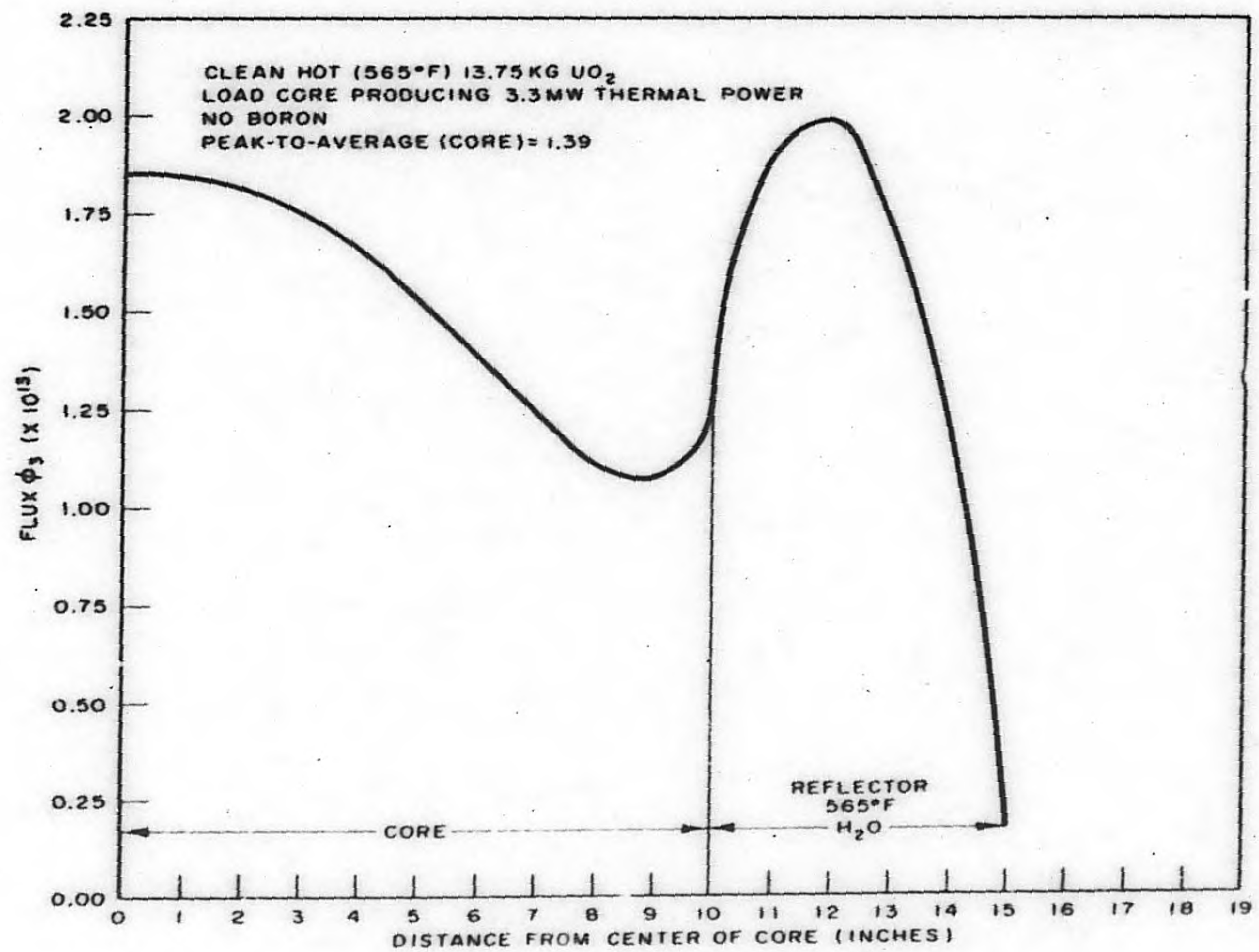


FIGURE 30

THERMAL FLUX vs DISTANCE FROM CENTER OF CORE

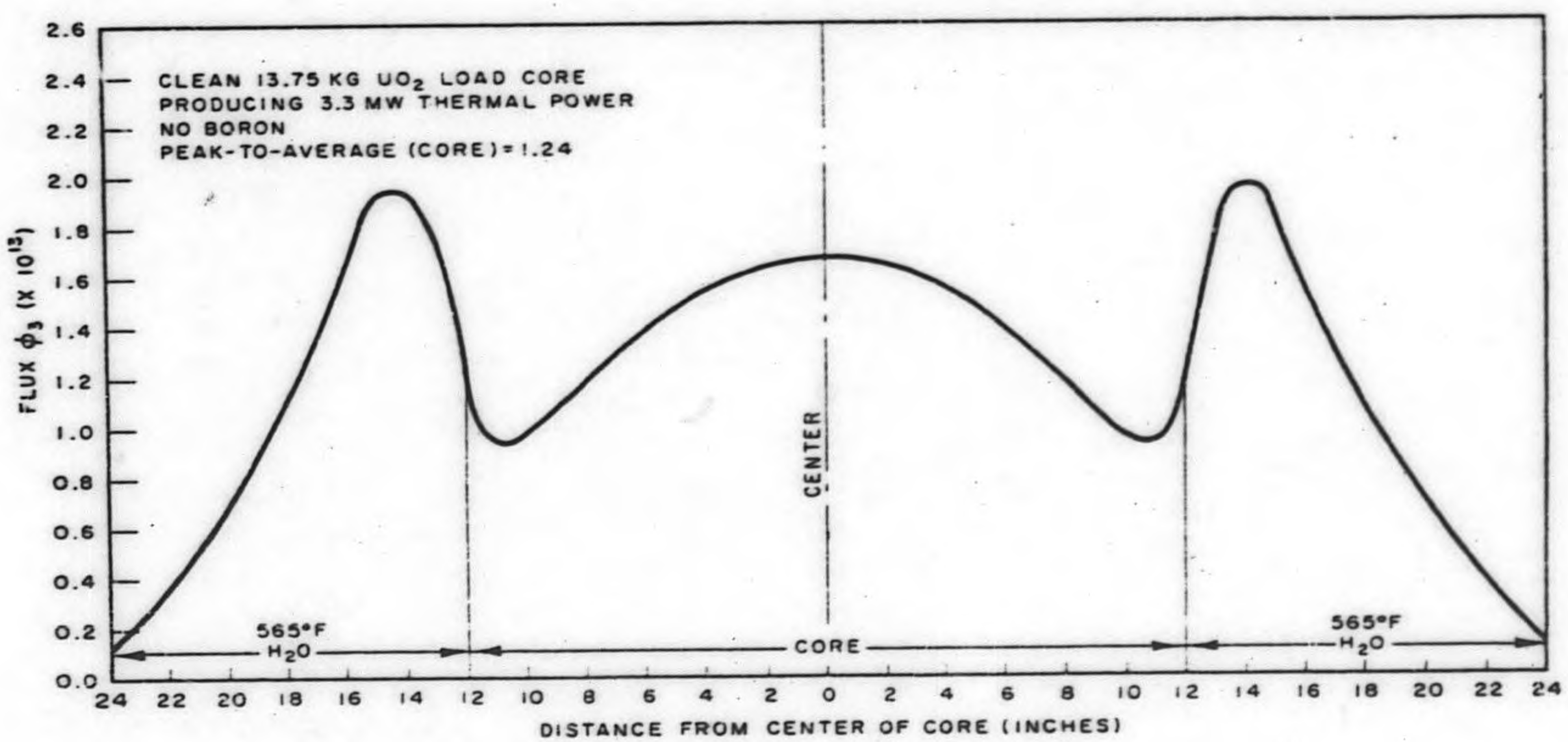


FIGURE 31

THERMAL FLUX ϕ_3 (AXIAL DISTRIBUTION)

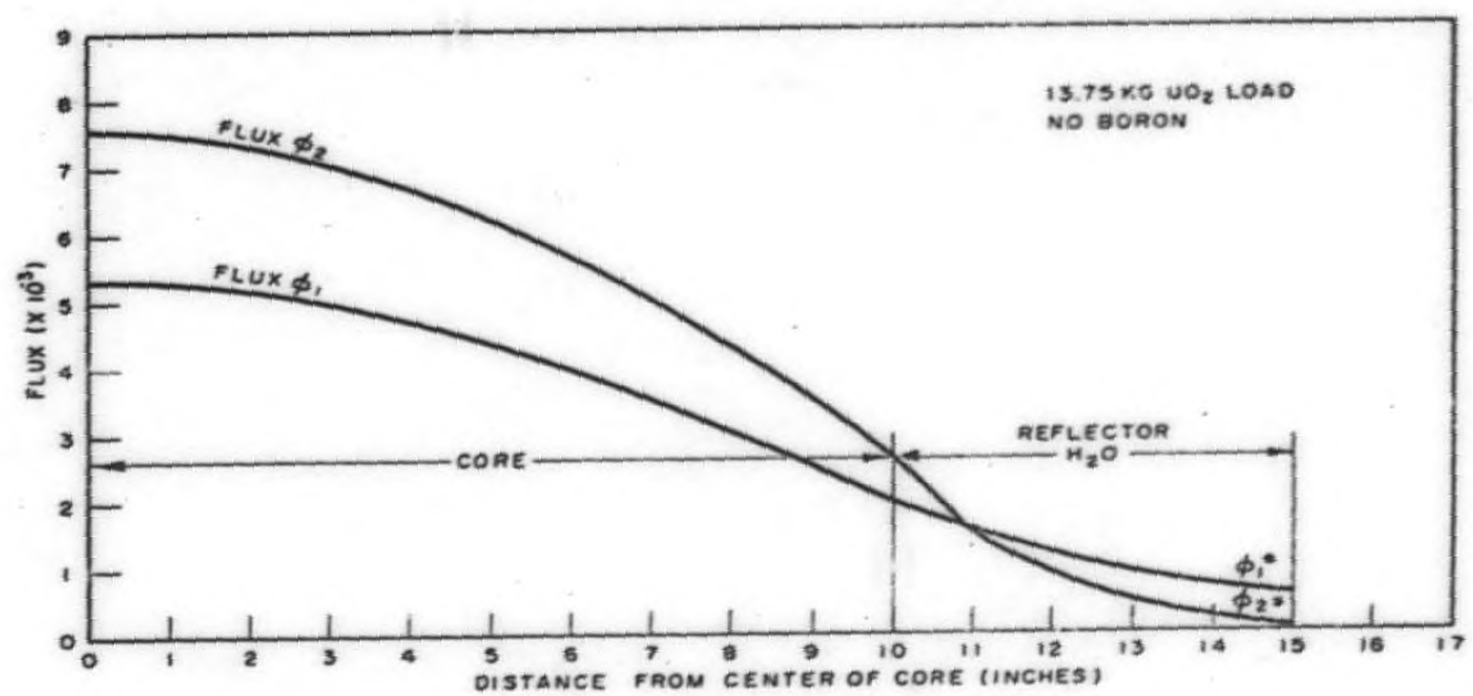


FIGURE 32

RELATIVE ADJOINT FLUX VS DISTANCE FROM CENTER OF CORE

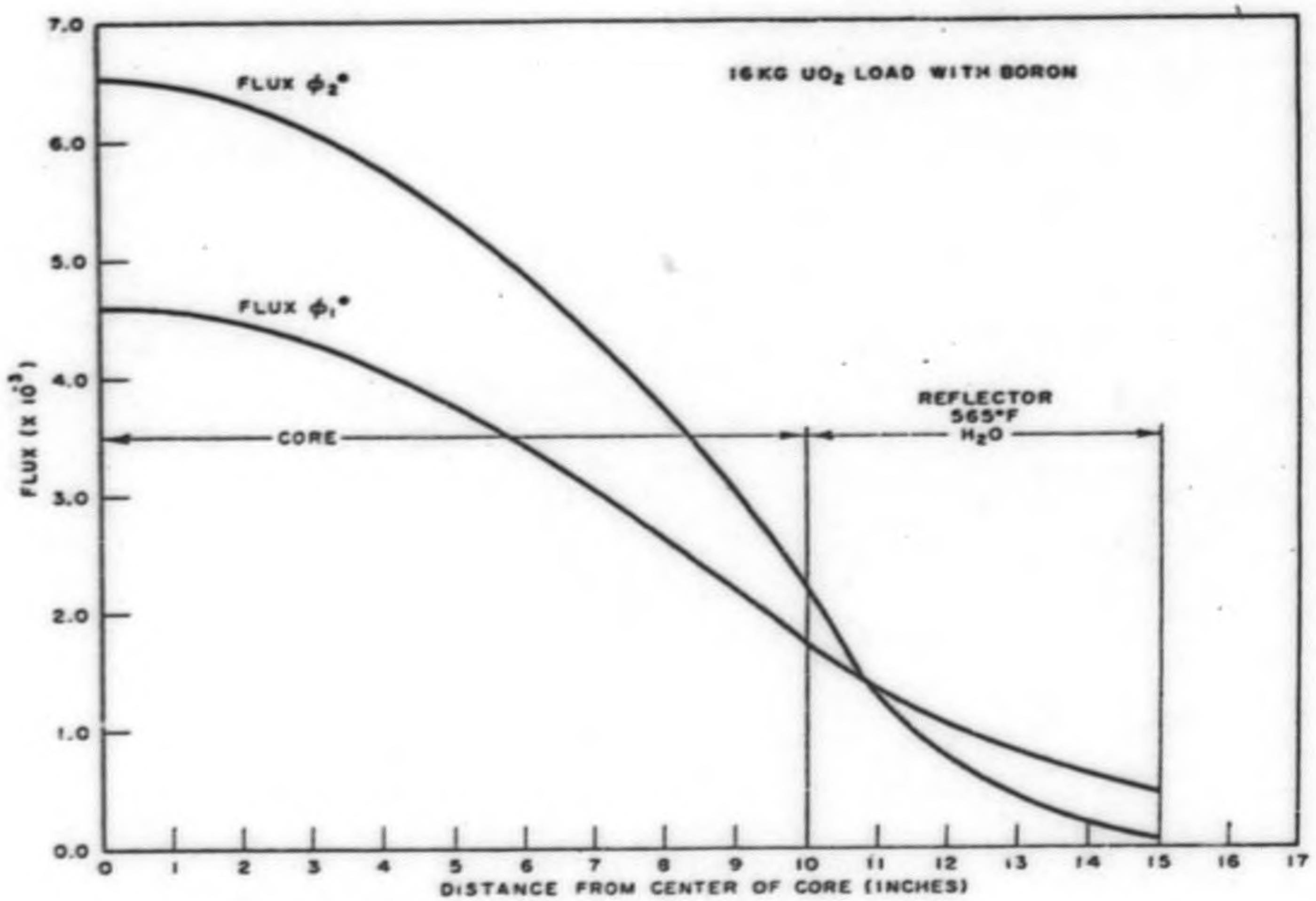


FIGURE 33

RELATIVE ADJOINT FLUX vs DISTANCE FROM CENTER OF CORE

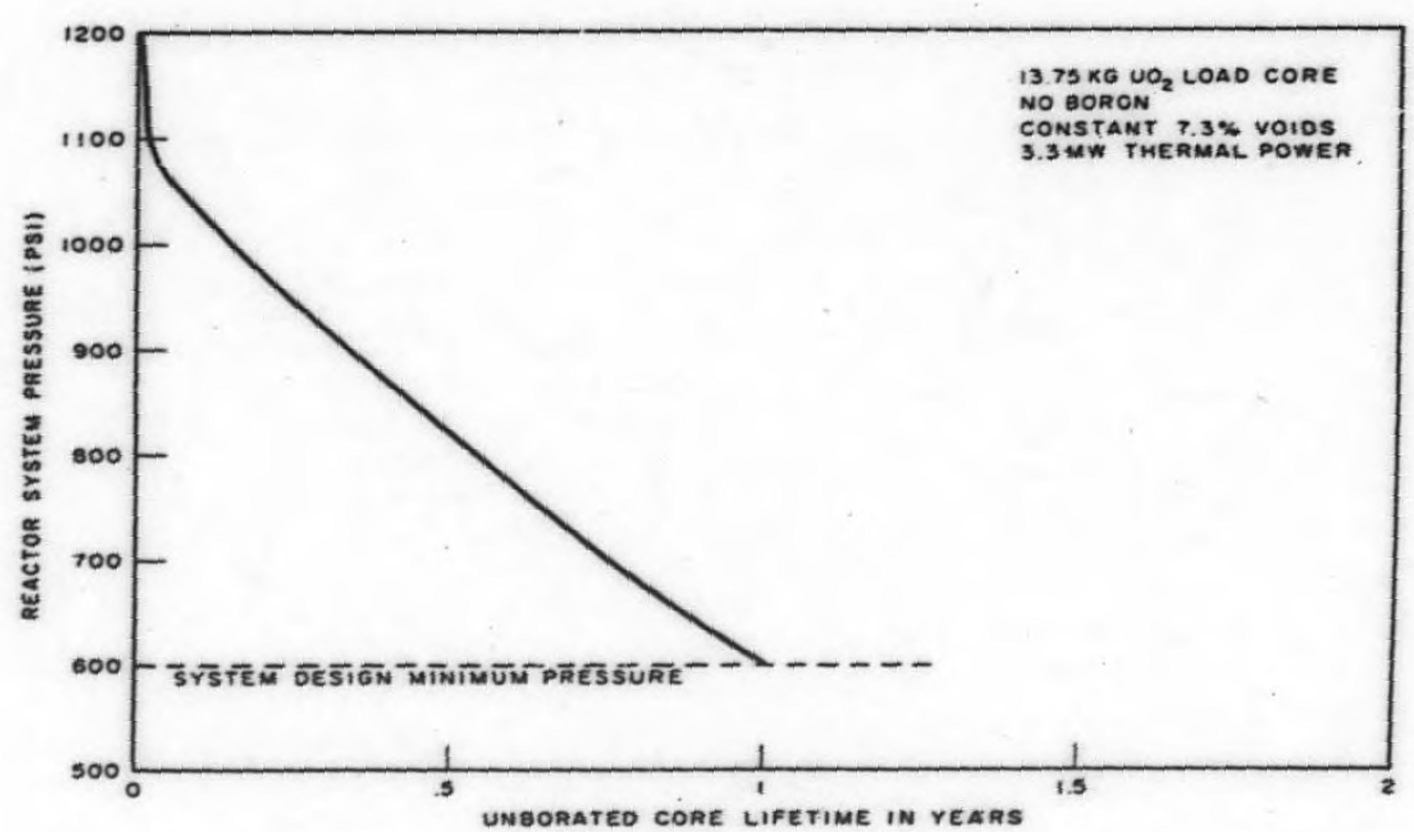


FIGURE 34

REACTOR PRESSURE VS CORE LIFETIME

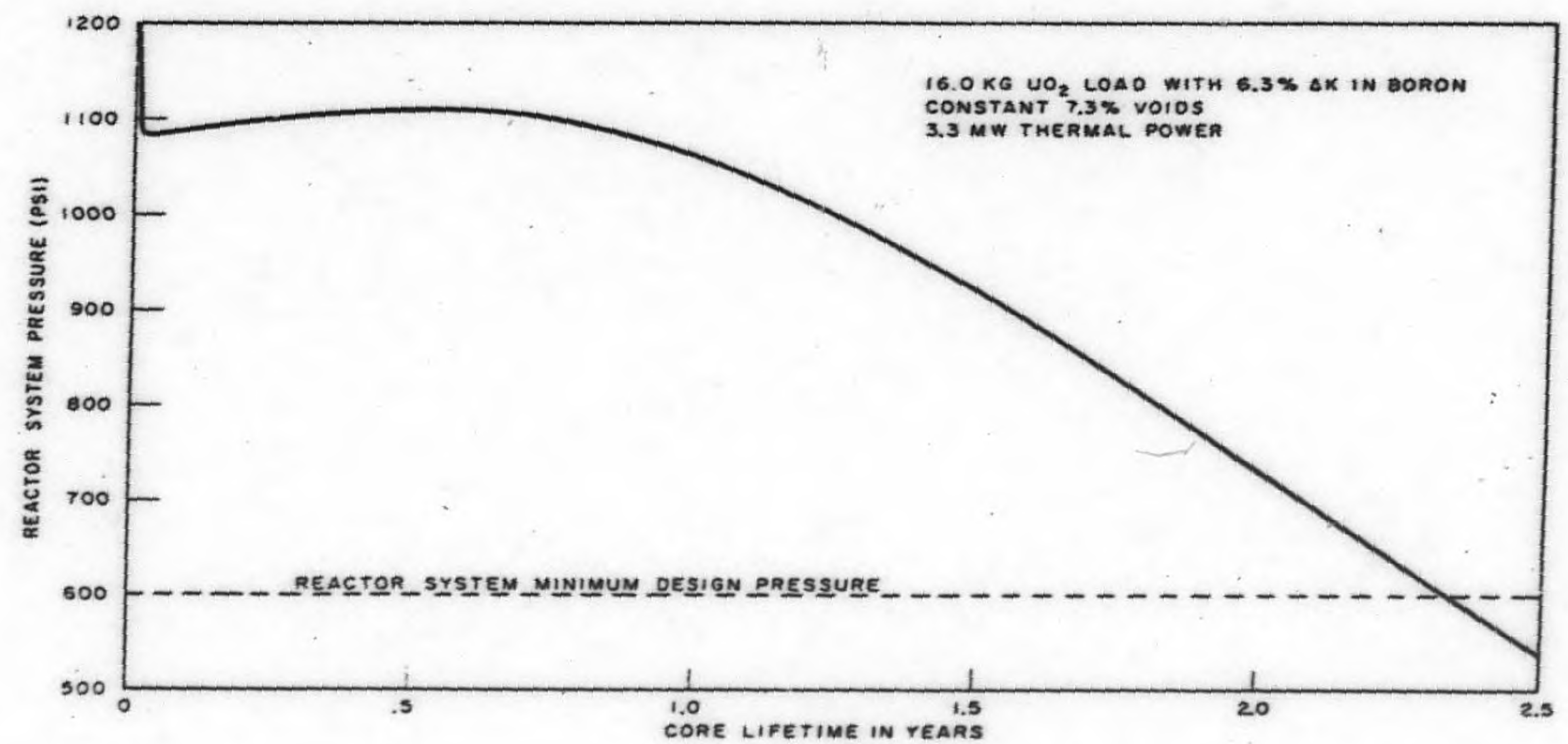


FIGURE 35

REACTOR PRESSURE VS CORE LIFETIME

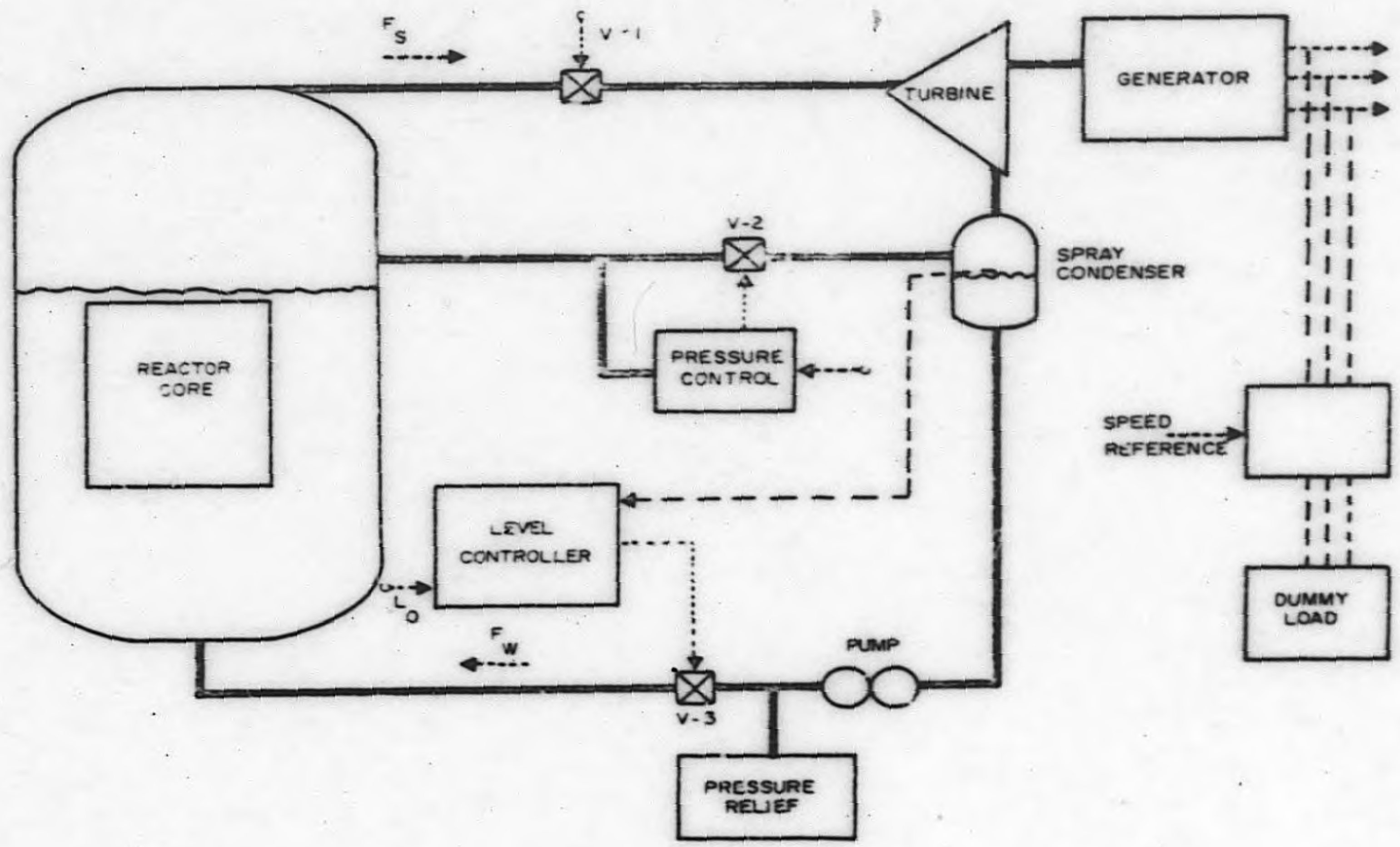


FIGURE 36
MINIMUM CONTROL SYSTEM

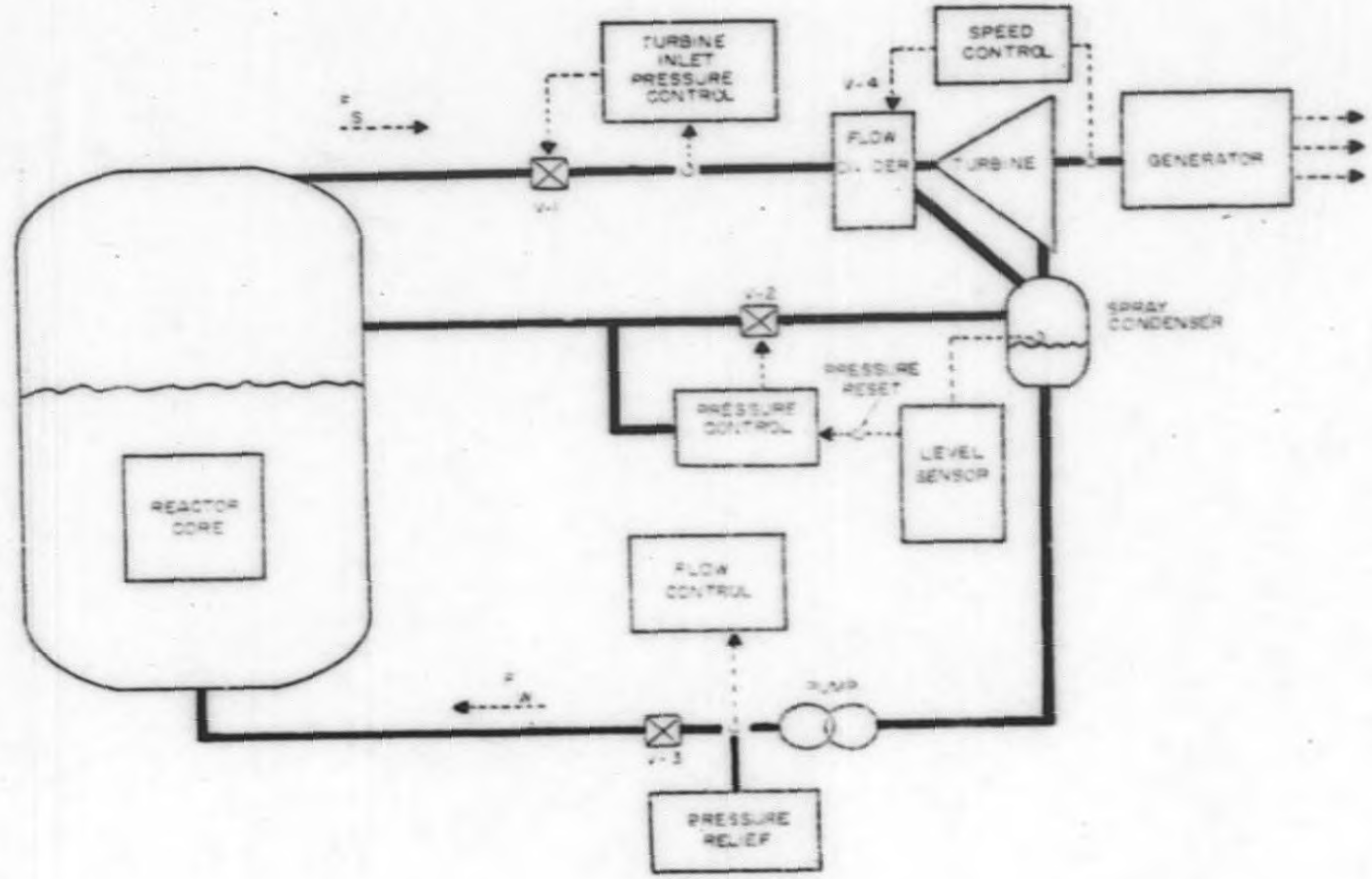
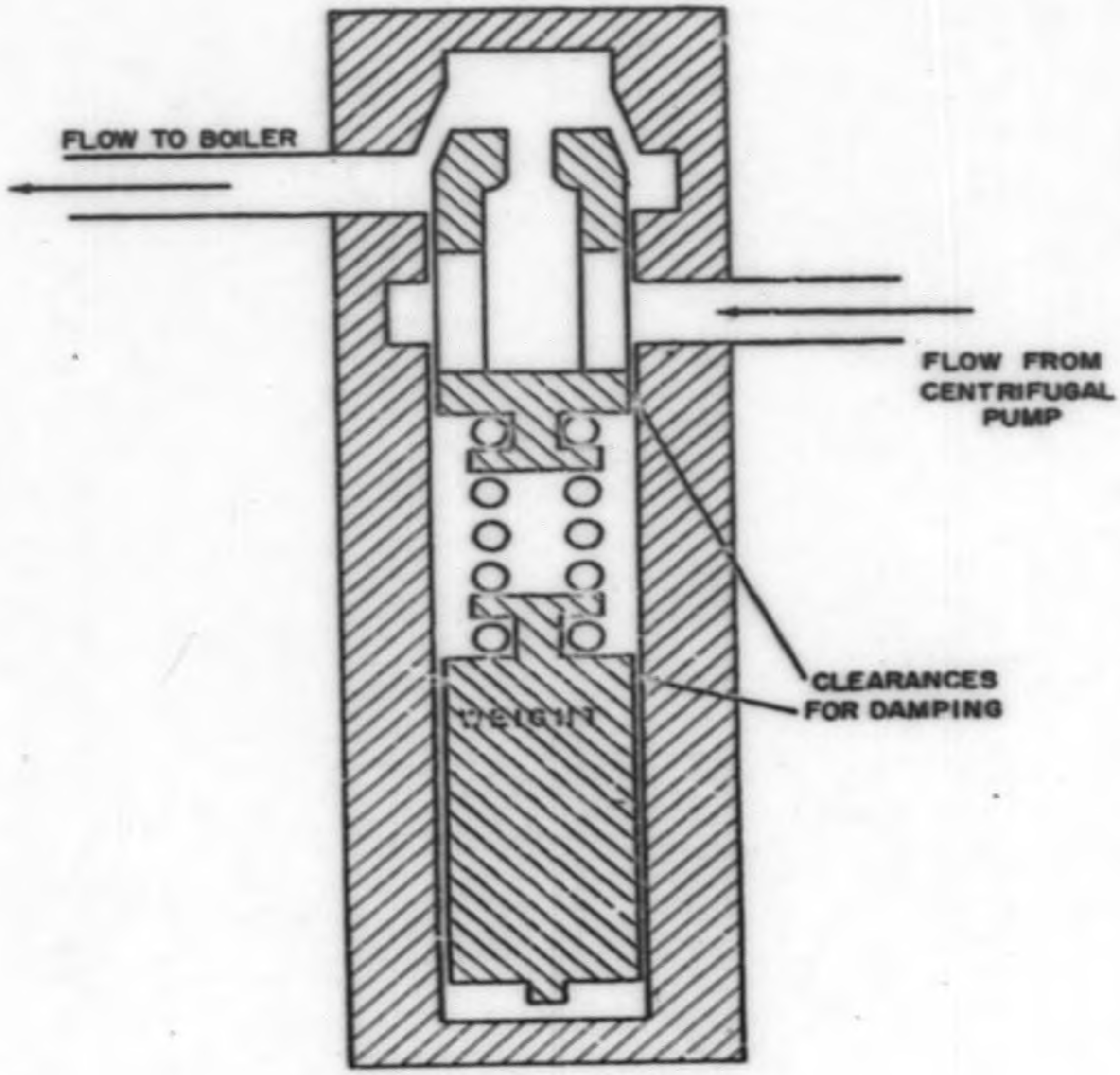
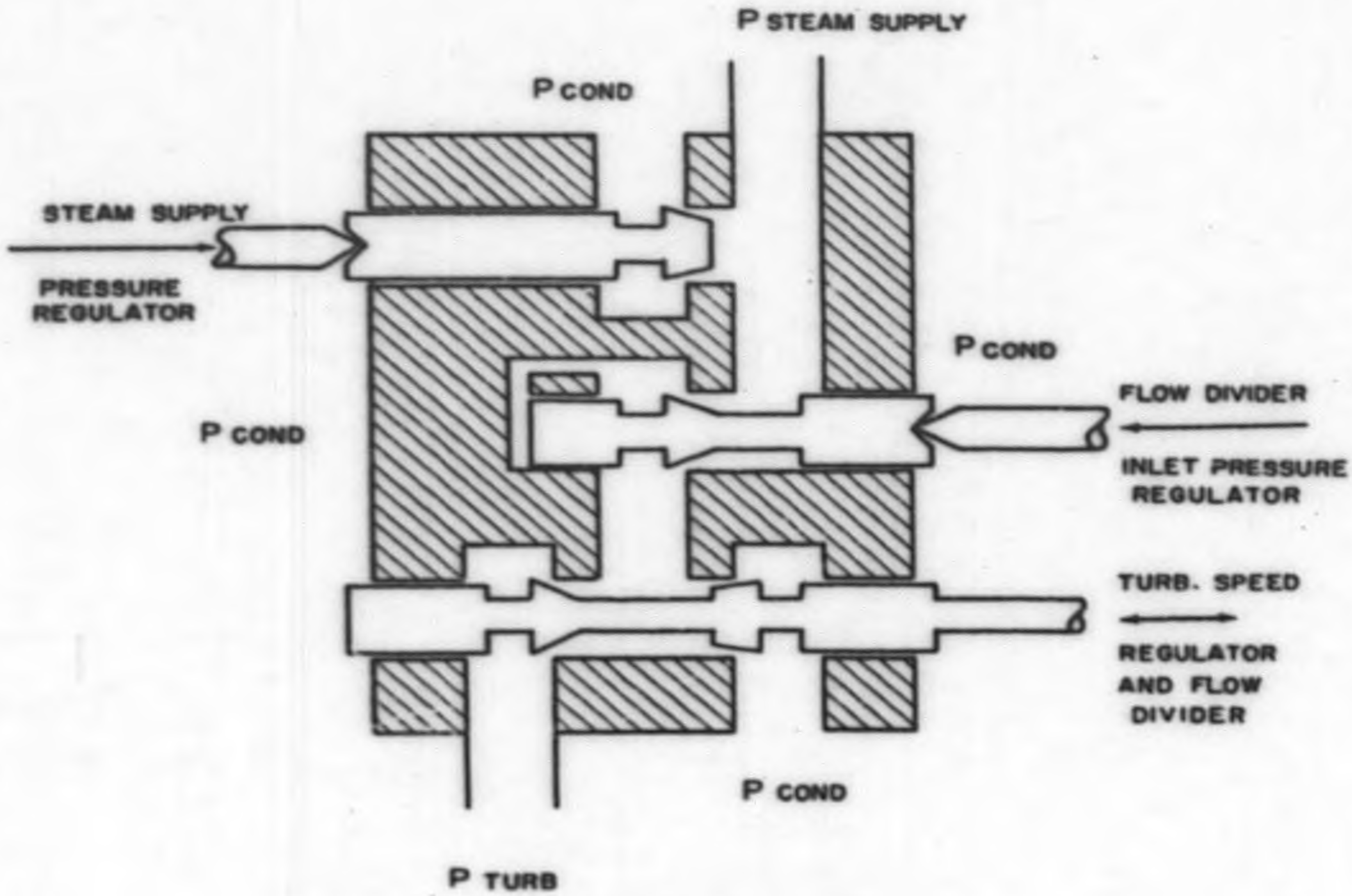


FIGURE 37
UNATTENDED CONTROL SYSTEM

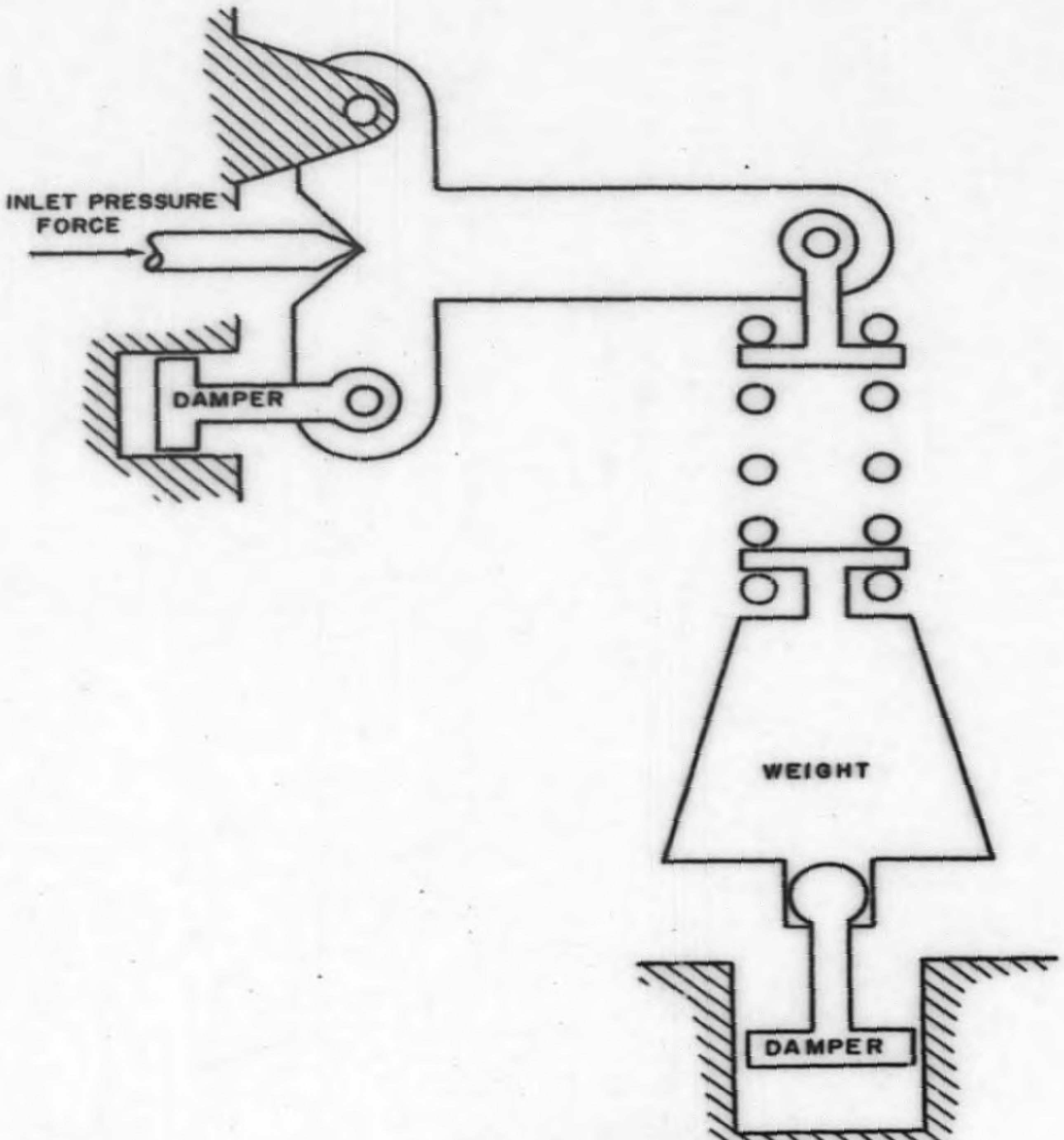


FEEDWATER FLOW REGULATOR
SCHEMATIC

FIGURE 38

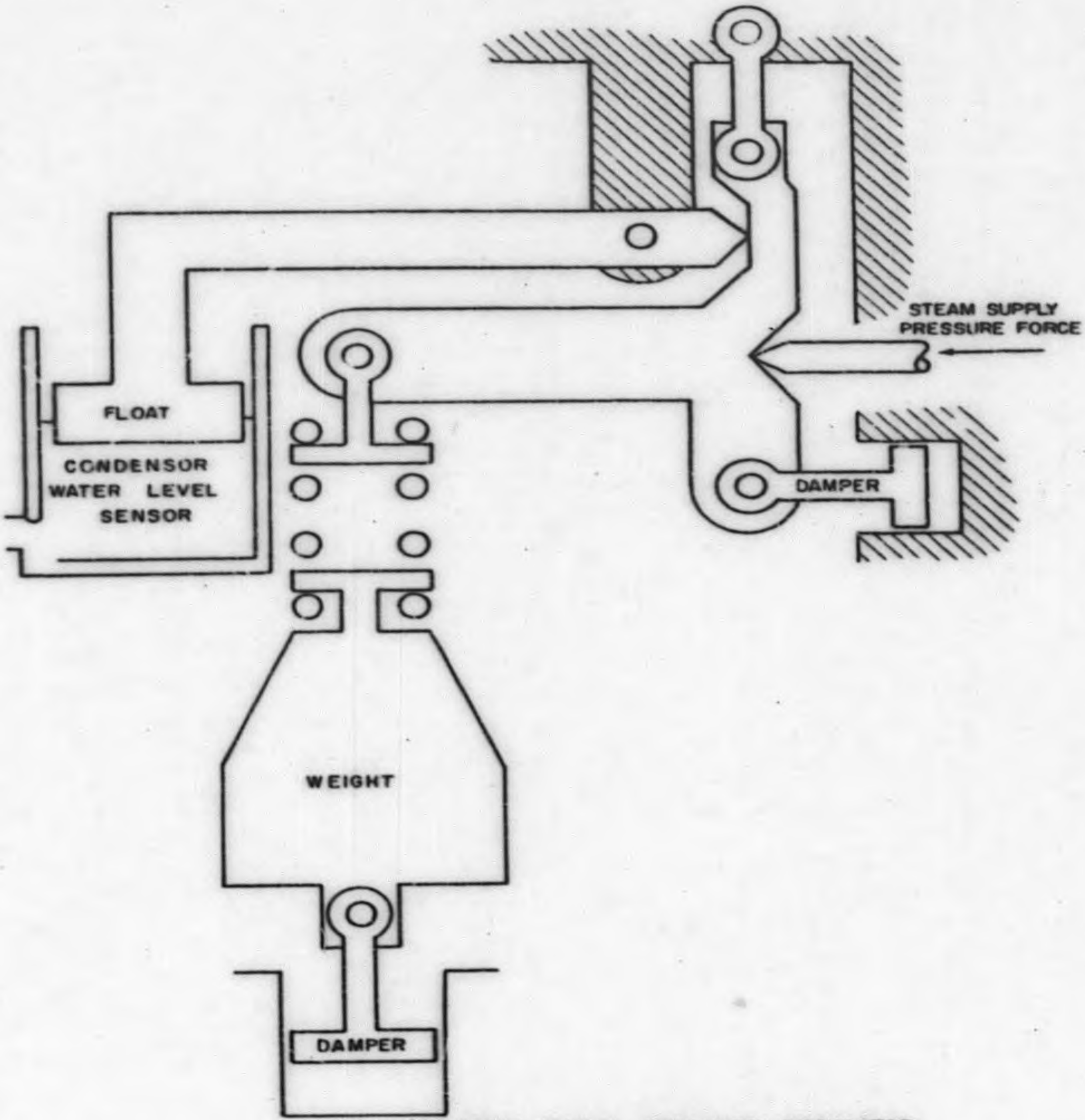


STEAM CONTROL VALVE
SCHEMATIC
FIGURE 39



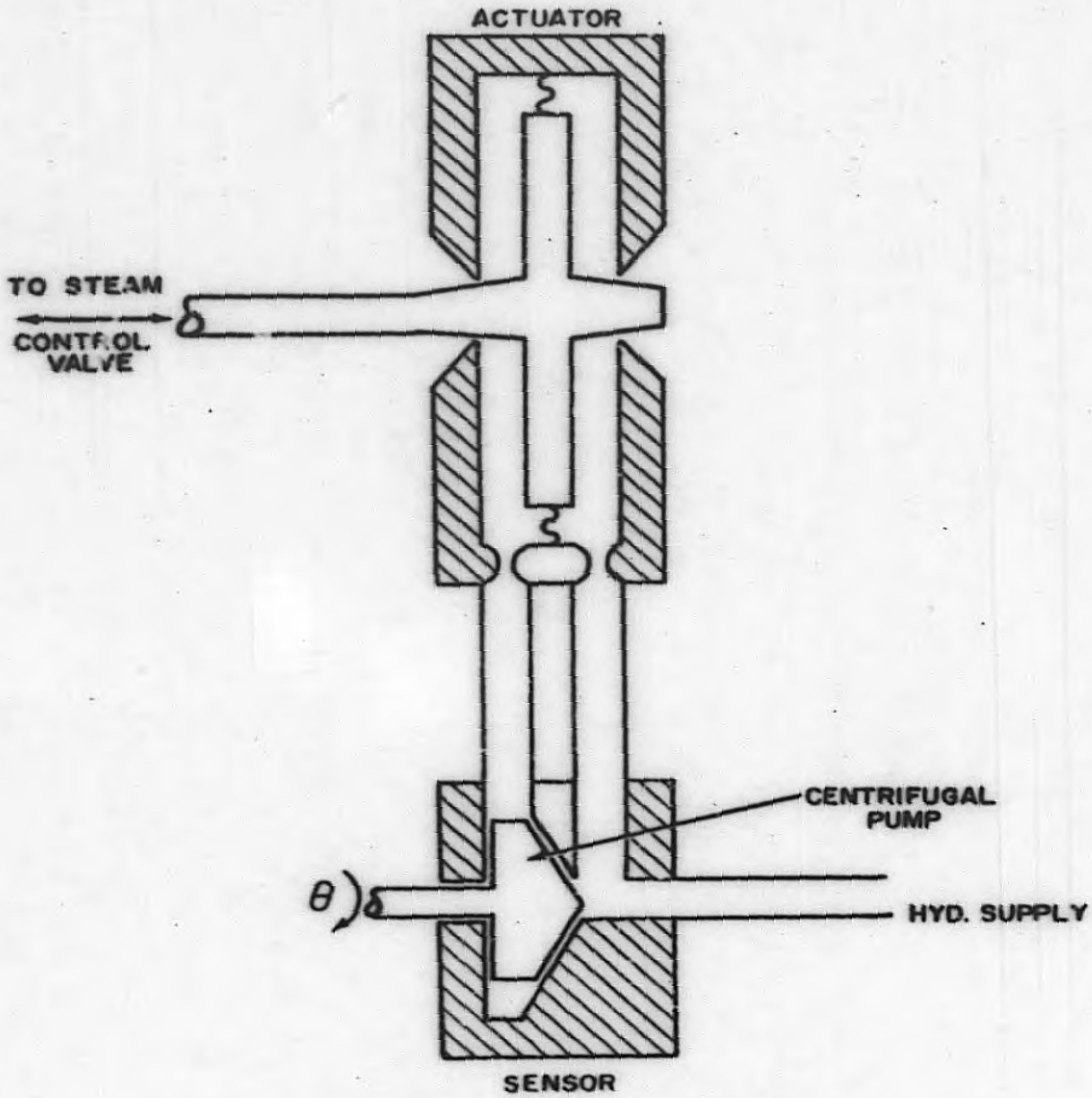
INLET PRESSURE REGULATOR
SCHEMATIC

FIGURE 40



REACTOR VESSEL PRESSURE REGULATOR
SCHEMATIC

FIGURE 41



TURBINE SPEED REGULATOR
SCHEMATIC

FIGURE 42

ROTOR SPEED: 12000 RPM
APPROX WT: 2500 #
VERTICAL MOUNT

ALL DIMENSIONS: APPROXIMATE

SCALE: 1/8" = 1"

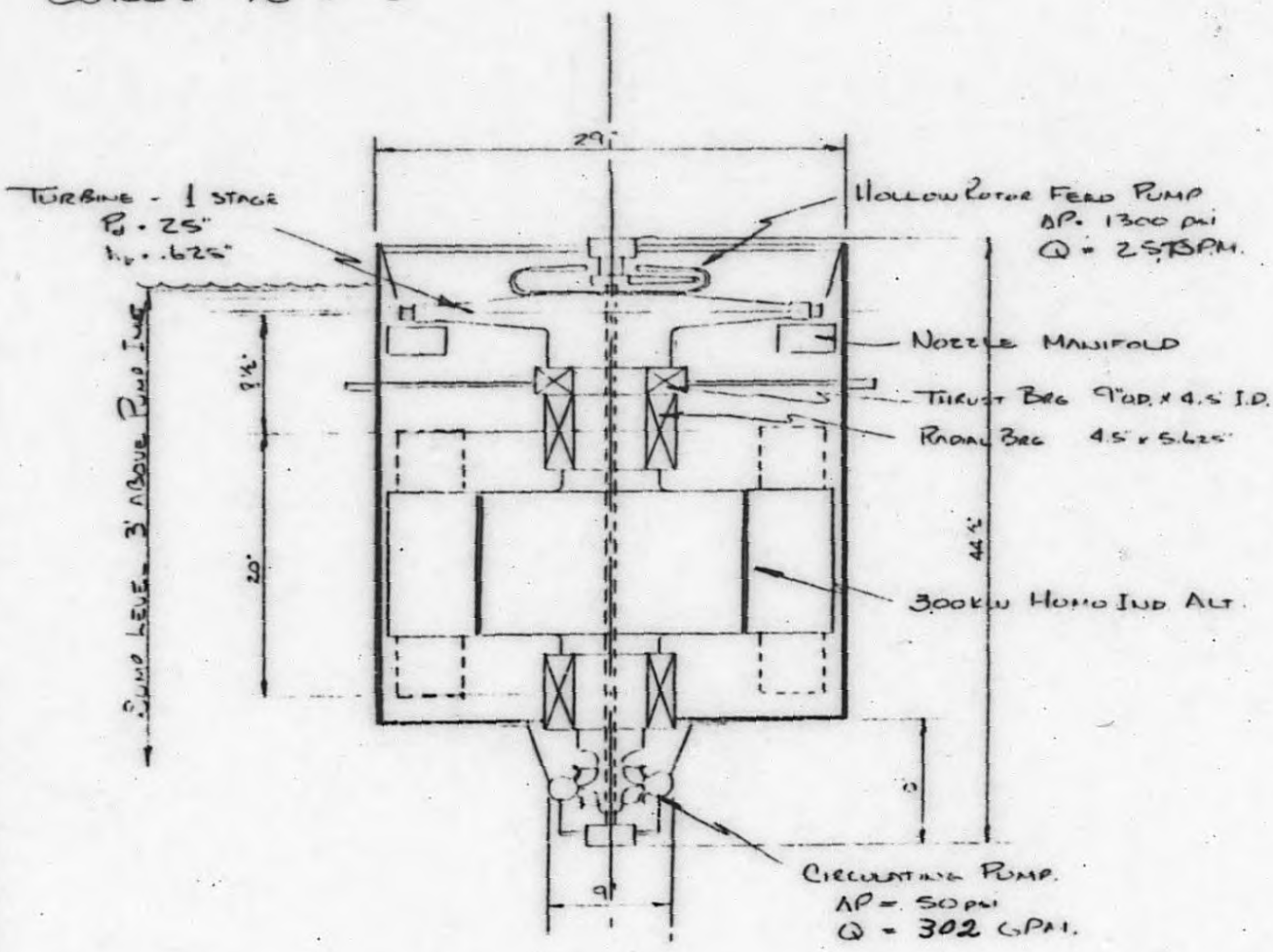


FIGURE 43

TURBINE/GENERATOR ASSEMBLY - FOR
300 KW. UNATTENDED POWER PLANT

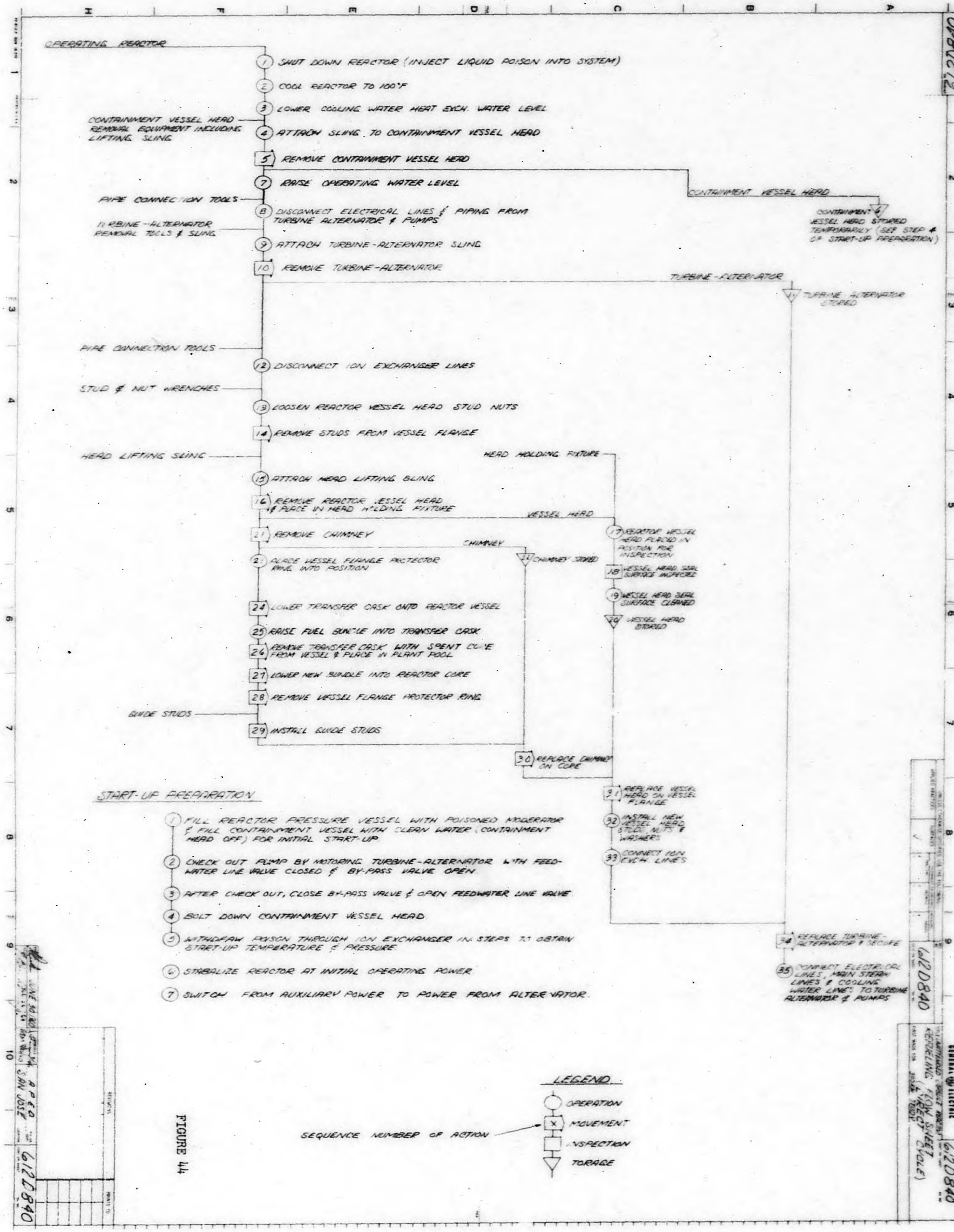

612D840
OPERATING REACTOR
1 SHUT DOWN REACTOR (INJECT LIQUID POISON INTO SYSTEM)
2 COOL REACTOR TO 100°F
3 LOWER COOLING WATER HEAT EXCH. WATER LEVEL
CONTAINMENT VESSEL HEAD REMOVAL EQUIPMENT INCLUDING LIFTING SLING
4 ATTACH SLING TO CONTAINMENT VESSEL HEAD
5 REMOVE CONTAINMENT VESSEL HEAD
7 RAISE OPERATING WATER LEVEL
CONTAINMENT VESSEL HEAD
CONTAINMENT VESSEL HEAD STORED TEMPORARILY (SEE STEP 4 OF START-UP PREPARATION)
PIPE CONNECTION TOOLS
8 DISCONNECT ELECTRICAL LINES & PIPING FROM TURBINE ALTERNATOR & PUMPS
TURBINE-ALTERNATOR REMOVAL TOOLS & SLING
9 ATTACH TURBINE-ALTERNATOR SLING
10 REMOVE TURBINE-ALTERNATOR
TURBINE-ALTERNATOR
11 TURBINE ALTERNATOR STORED
PIPE CONNECTION TOOLS
12 DISCONNECT ION EXCHANGER LINES
STUD & NUT WRENCHES
13 LOOSEN REACTOR VESSEL HEAD STUD NUTS
14 REMOVE STUDS FROM VESSEL FLANGE
HEAD LIFTING SLING
HEAD HOLDING FIXTURE
15 ATTACH HEAD LIFTING SLING
16 REMOVE REACTOR VESSEL HEAD & PLACE IN HEAD HOLDING FIXTURE
VESSEL HEAD
17 REACTOR VESSEL HEAD PLACED IN POSITION FOR INSPECTION
21 REMOVE CHIMNEY
CHIMNEY
22 PLACE VESSEL FLANGE PROTECTOR RING INTO POSITION
23 CHIMNEY STORED
18 VESSEL HEAD SEAL SURFACE INSPECTED
19 VESSEL HEAD SEAL SURFACE CLEANED
20 VESSEL HEAD STORED
24 LOWER TRANSFER CASK ONTO REACTOR VESSEL
25 RAISE FUEL BUNDLE INTO TRANSFER CASK
26 REMOVE TRANSFER CASK WITH SPENT CORE FROM VESSEL & PLACE IN PLANT POOL
27 LOWER NEW BUNDLE INTO REACTOR CORE
28 REMOVE VESSEL FLANGE PROTECTOR RING
GUIDE STUDS
29 INSTALL GUIDE STUDS
30 REPLACE CHIMNEY ON CORE
31 REPLACE VESSEL HEAD ON VESSEL FLANGE
32 INSTALL NEW VESSEL HEAD STUDS, NUTS & WASHERS
33 CONNECT ION EXCH. LINES
34 REPLACE TURBINE-ALTERNATOR & SECURE
35 CONNECT ELECTRICAL LINES, MAIN STEAM LINES & COOLING WATER LINES TO TURBINE ALTERNATOR & PUMPS
START-UP PREPARATION
1 FILL REACTOR PRESSURE VESSEL WITH POISONED MODERATOR & FILL CONTAINMENT VESSEL WITH CLEAN WATER (CONTAINMENT HEAD OFF) FOR INITIAL START-UP
2 CHECK OUT PUMP BY MOTORING TURBINE-ALTERNATOR WITH FEED-WATER LINE VALVE CLOSED & BY-PASS VALVE OPEN
3 AFTER CHECK OUT, CLOSE BY-PASS VALVE & OPEN FEEDWATER LINE VALVE
4 BOLT DOWN CONTAINMENT VESSEL HEAD
5 WITHDRAW POISON THROUGH ION EXCHANGER IN STEPS TO OBTAIN START-UP TEMPERATURE & PRESSURE
6 STABALIZE REACTOR AT INITIAL OPERATING POWER
7 SWITCH FROM AUXILIARY POWER TO POWER FROM ALTERNATOR
LEGEND
OPERATION
MOVEMENT
INSPECTION
STORAGE
SEQUENCE NUMBER OF ACTION
GENERAL ELECTRIC
REFUELING FLOW SHEET
612D840
APED SAN JOSE
612D840

FIGURE 44

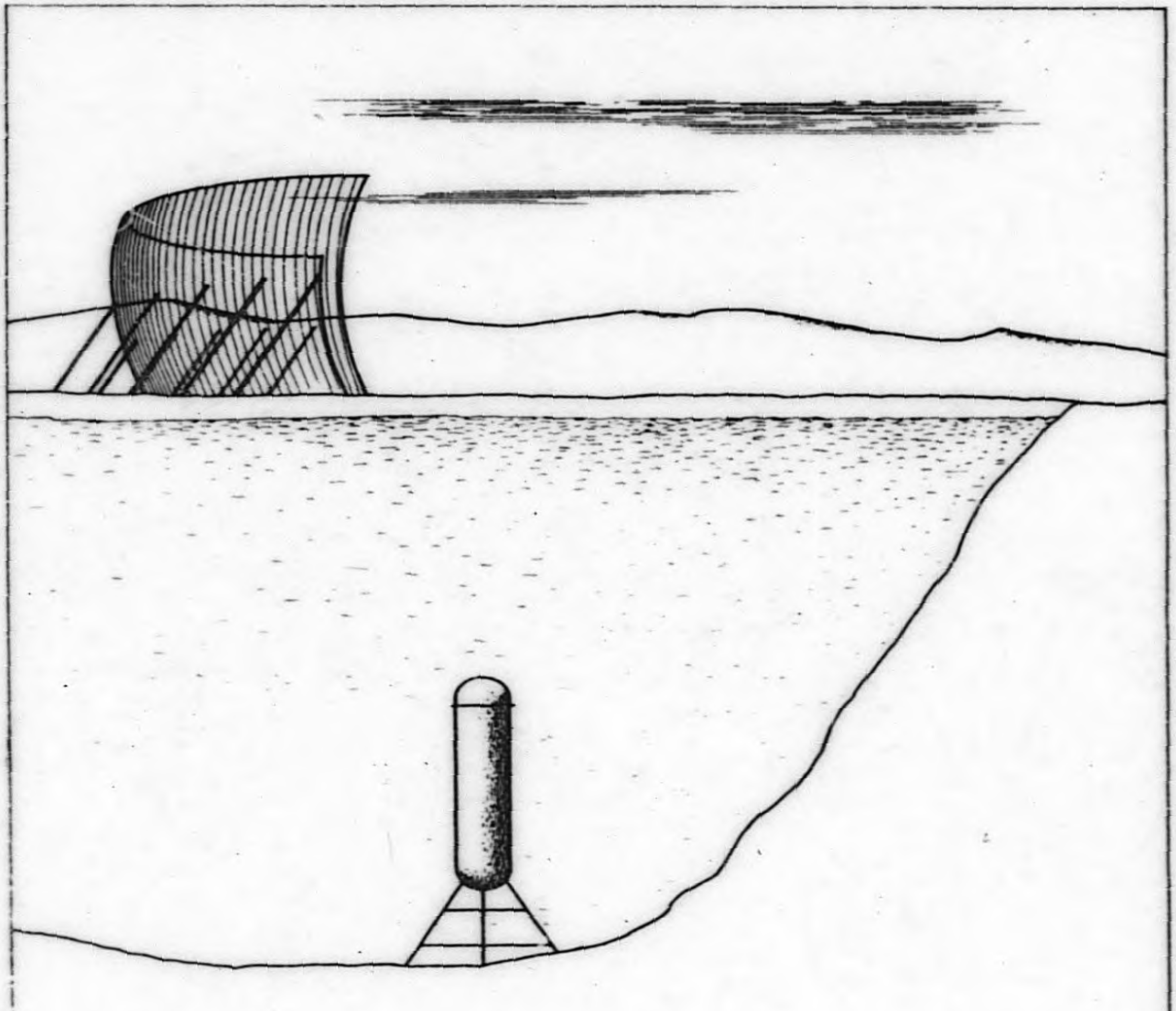


FIGURE 45

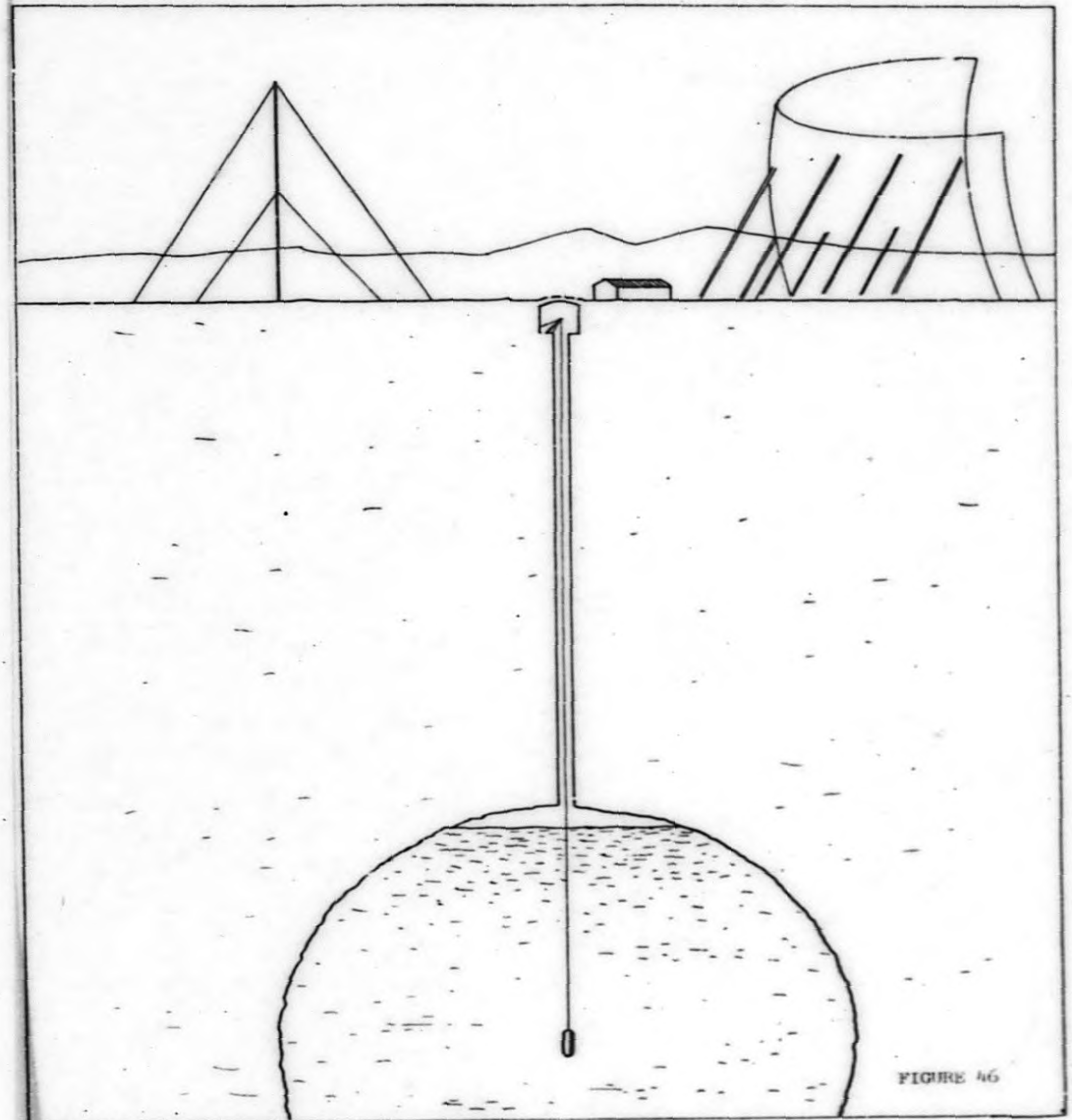


FIGURE 46

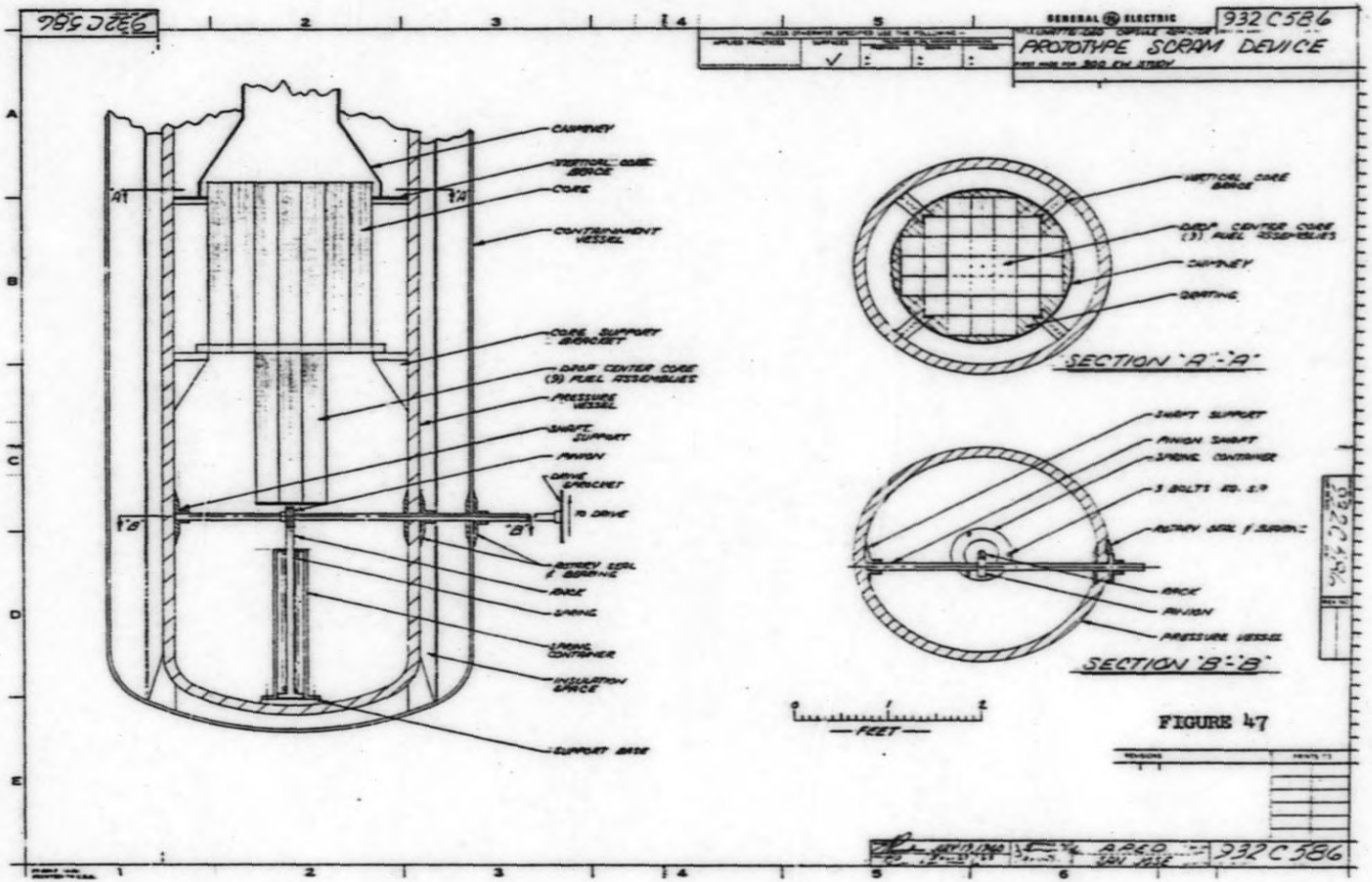
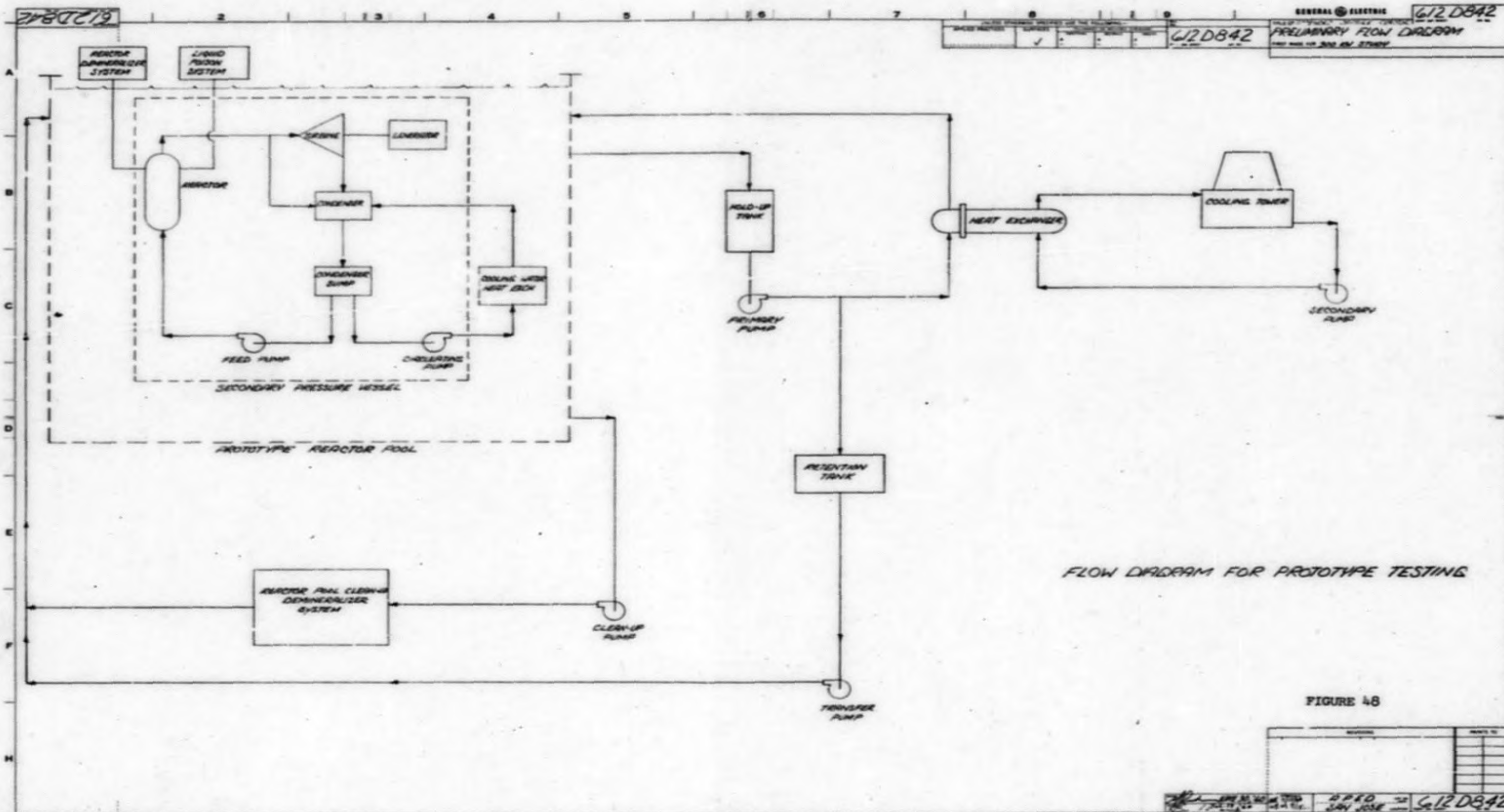

GENERAL ELECTRIC
932C586
PROTOTYPE SCRAM DEVICE
CHIMNEY
CORE
CONTAINMENT VESSEL
CORE SUPPORT BRACKET
DROP CENTER CORE (9) FUEL ASSEMBLIES
PRESSURE VESSEL
SHAFT SUPPORT
PINION
DRIVE SPROCKET
TO DRIVE
ROTARY SEAL & BEARING
RACK
SPRING
SPRING CONTAINER
INSULATION SPACE
SUPPORT BASE
"A"
"B"
VERTICAL CORE BRACE
CHIMNEY
GRATING
SECTION "A"-"A"
SHAFT SUPPORT
PINION SHAFT
SPRING CONTAINER
ROTARY SEAL & BEARING
RACK
PINION
PRESSURE VESSEL
SECTION "B"-"B"
FEET
A.P.E.D.
SAN JOSE
932C586

FIGURE 47



FLOW DIAGRAM FOR PROTOTYPE TESTING

FIGURE 48

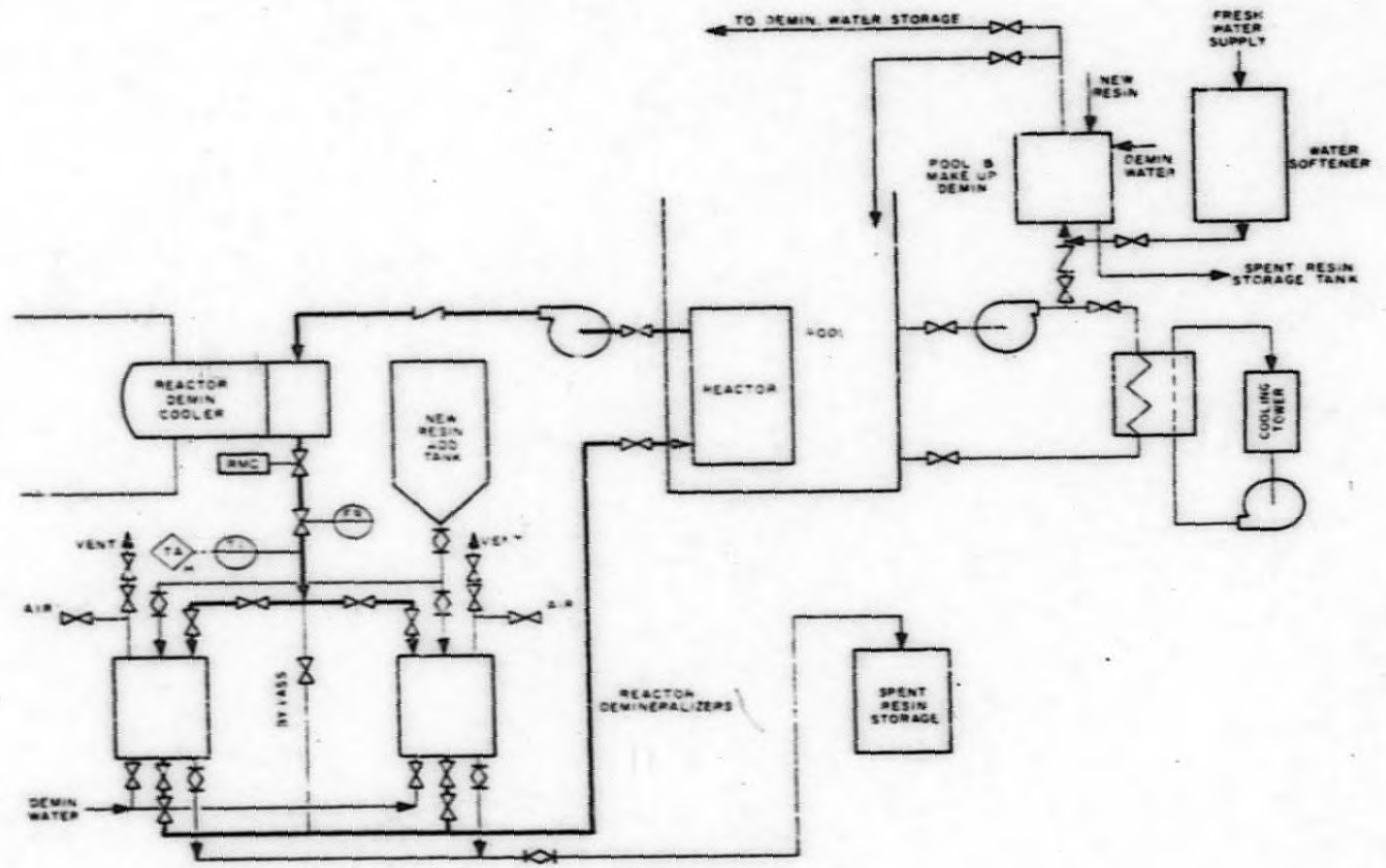


FIGURE 49

300 KW UNATTENDED REACTOR DEMINERALIZER SYSTEMS

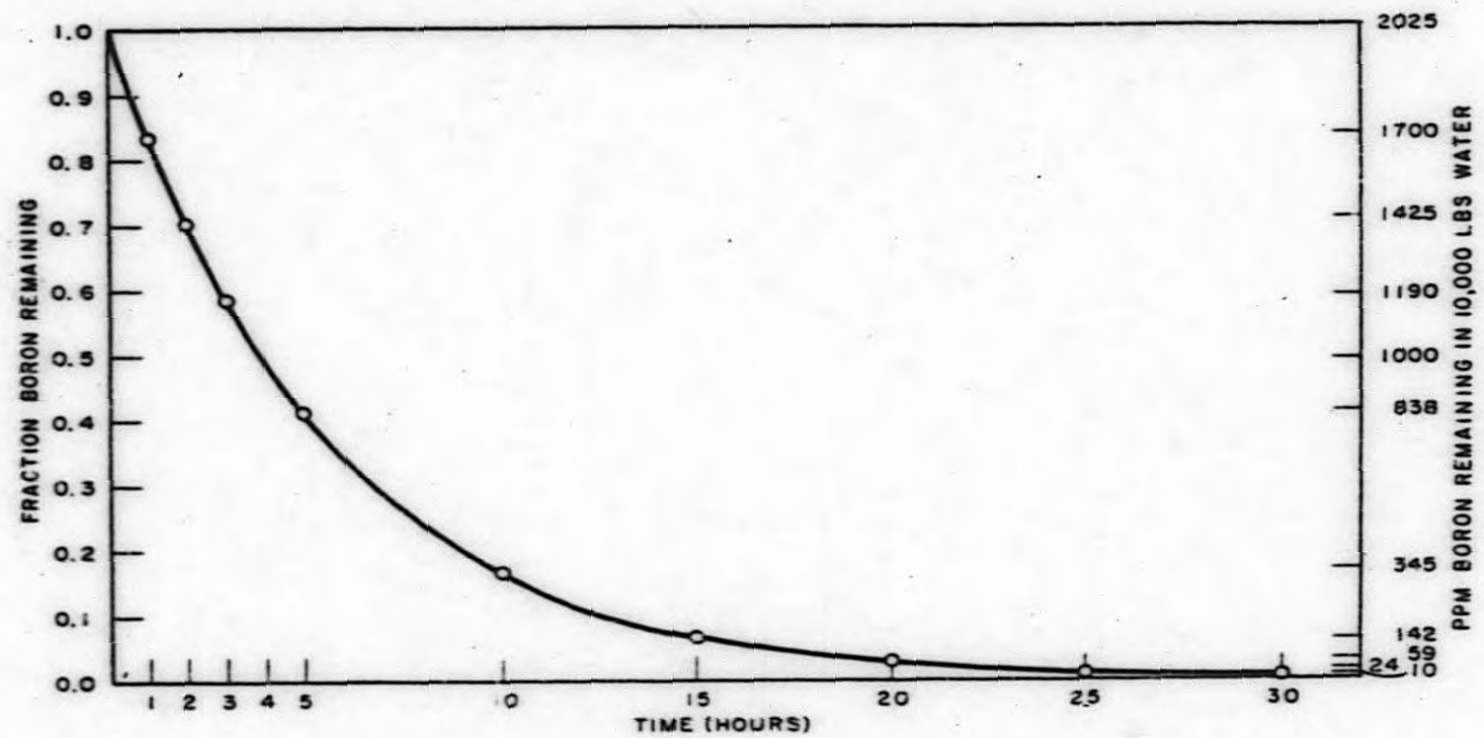


FIGURE 30

RATE OF BORON REMOVAL FROM REACTOR FOR CONSTANT DEMINERALIZER FLOW OF 1773 LBS/HR (3.51 GPM) COLD
(ASSUMED 30 HOURS TO GO FROM INITIAL BORON CONCENTRATION TO FINAL CONCENTRATION OF 10 PPM)

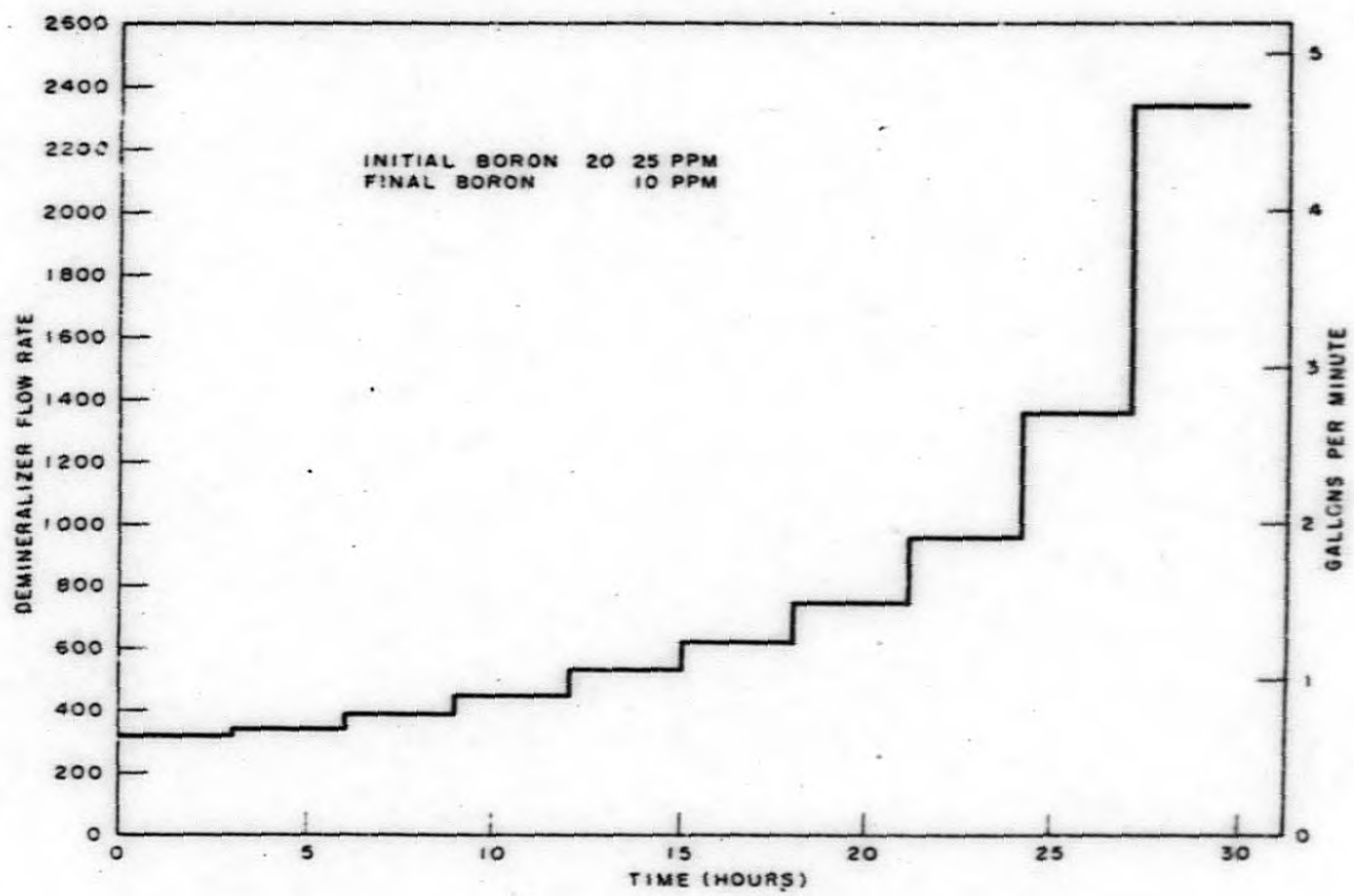


FIGURE 51

REACTOR DEMINERALIZER FLOW RATE VARIATION WITH TIME TO PRODUCE A RELATIVELY CONSTANT RATE OF BORON REMOVAL OVER A 30 HOUR PERIOD

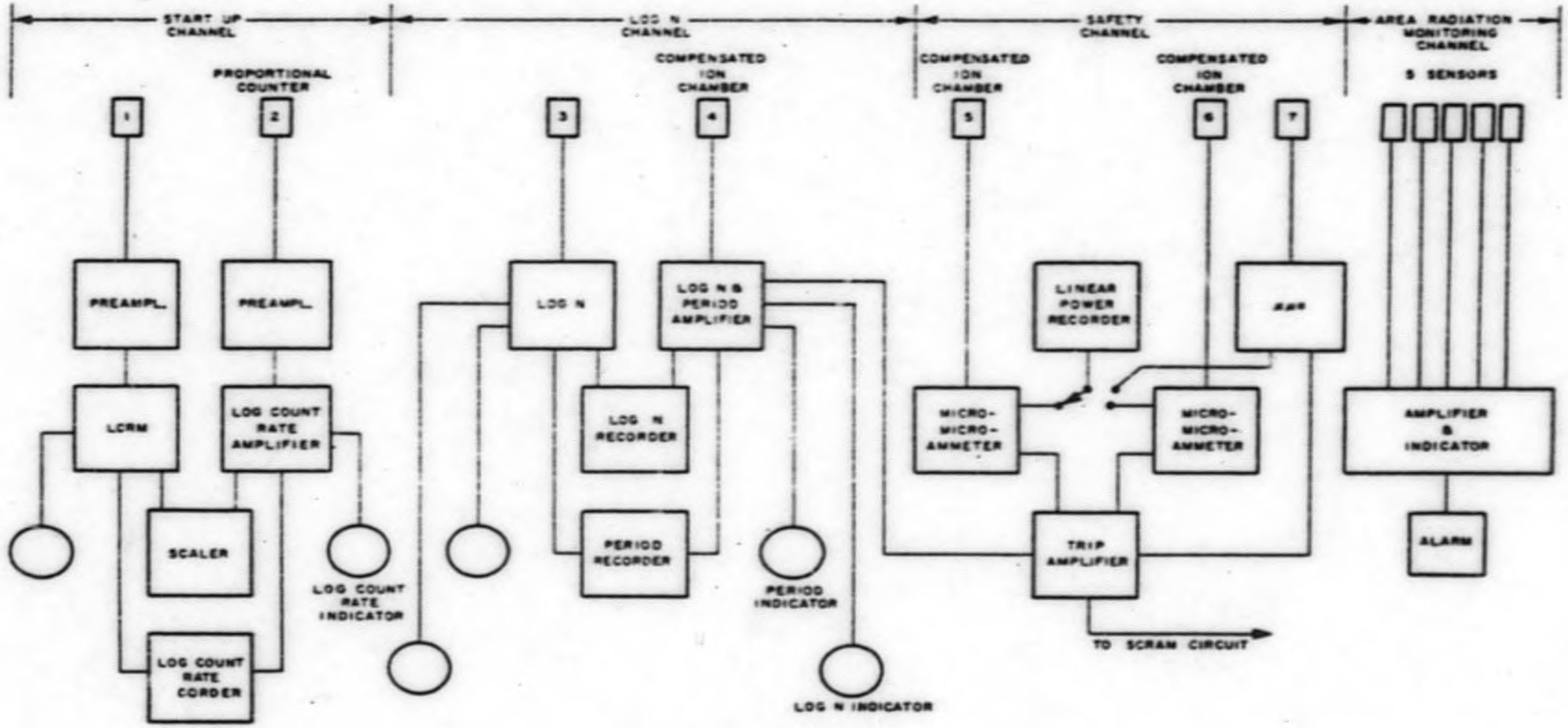
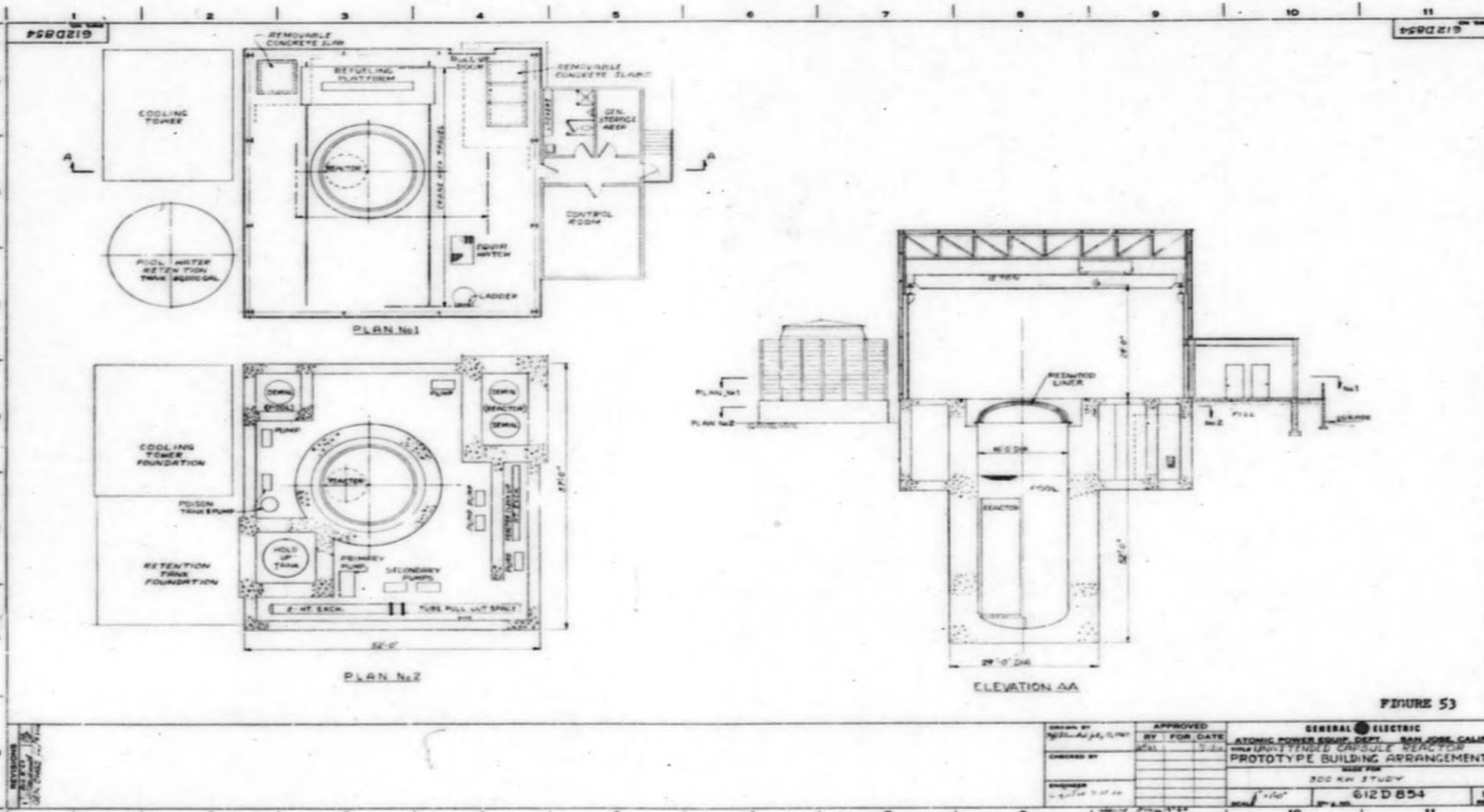


FIGURE 52

NUCLEAR INSTRUMENTATION



612D854
REMOVABLE CONCRETE SLAB
REFUELING PLATFORM
COOLING TOWER
REACTOR
CONTROL ROOM
EQUIP. HATCH
LADDER
POOL WATER RETENTION TANK
PLAN No 1
COOLING TOWER FOUNDATION
POISON TANK & PUMP
RETENTION TANK FOUNDATION
HOLD UP TANK
PUMP
PRIMARY PUMP
SECONDARY PUMPS
TUBE PULL OUT SPACE
52'-0"
PLAN No 2
PLAN No 1
PLAN No 2
REACTOR
POOL
28'-0" DIA
ELEVATION AA

FIGURE 53

DRAWN BY
APPROVED
BY
FOR
DATE
CHECKED BY
ENGINEER
GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
UNATTENDED CAPSULE REACTOR
PROTOTYPE BUILDING ARRANGEMENT
MADE FOR
300 KW STUDY
612D854
1
REVISIONS

DEVELOPMENT PROGRAM SCHEDULE

300 KW UNATTENDED NUCLEAR POWER PLANT

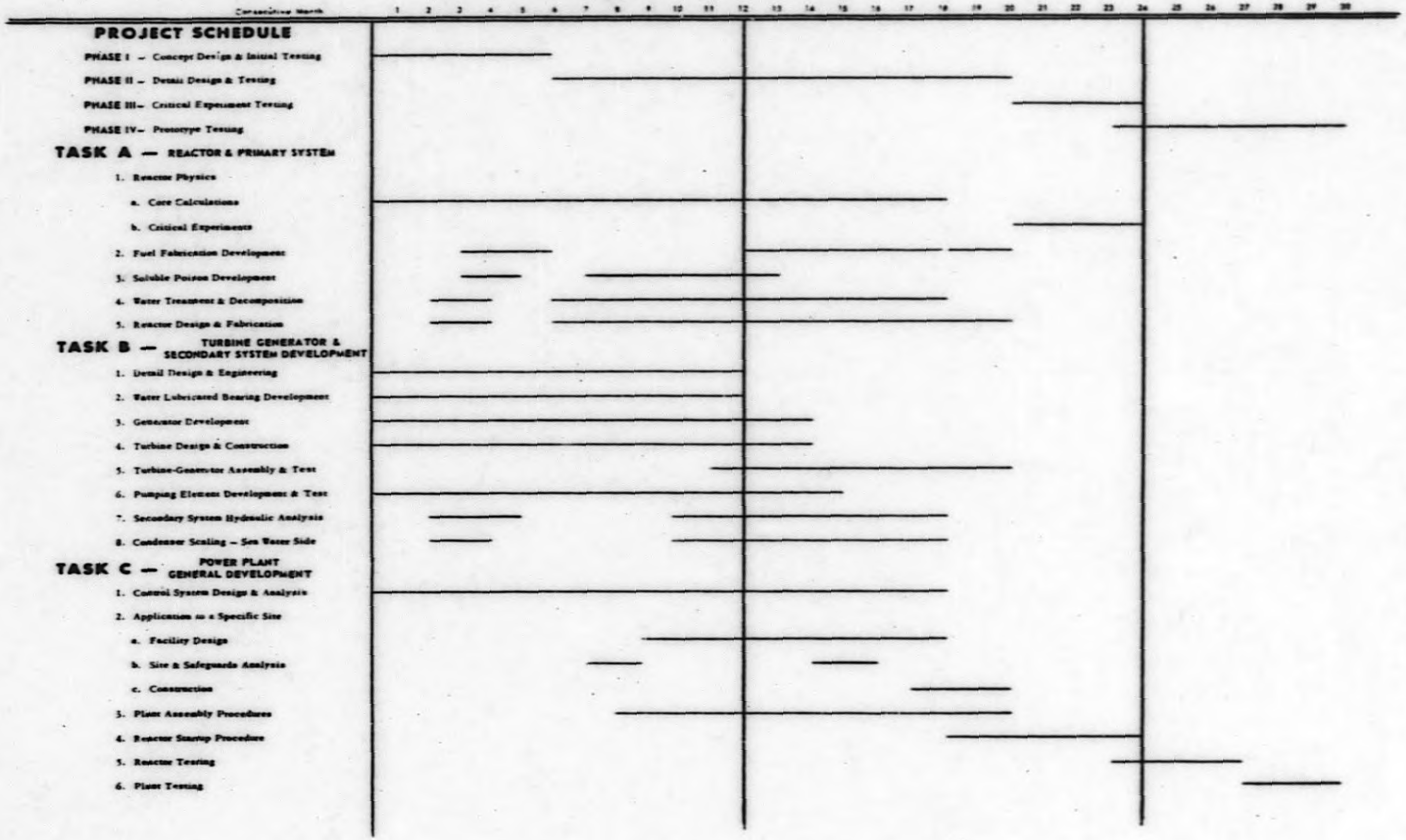


FIGURE 54

OVERALL SCHEDULE

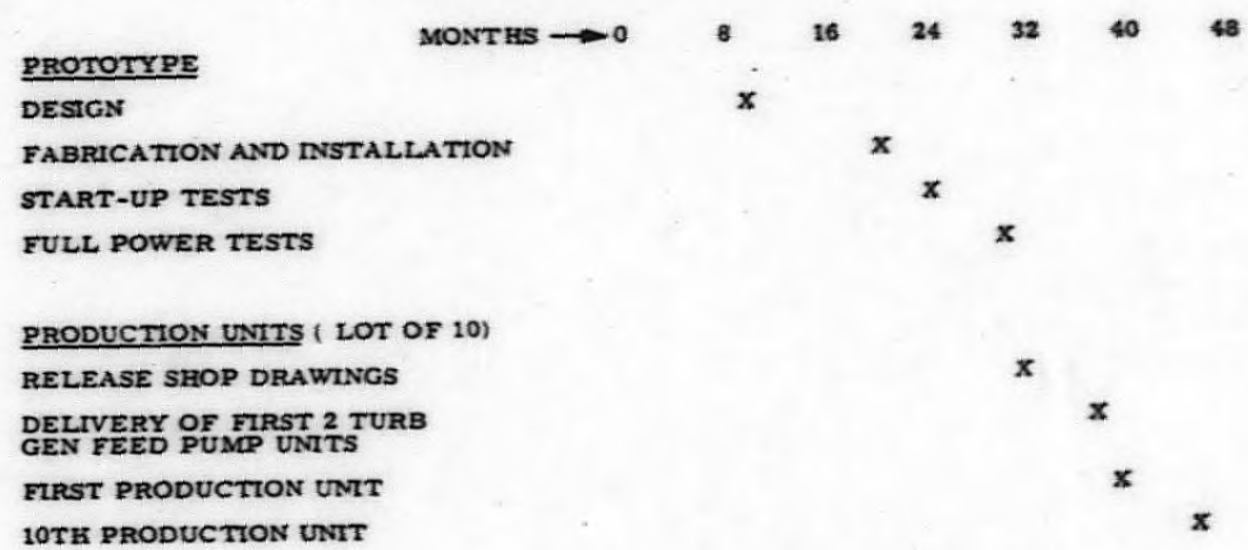


FIGURE 55