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GA-8409

"TRIGA OCEANOGRAPHIC POWER SUPPLY FOR A MANNED UNDERWATER STATION,

NCEL
CR-
68.011
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v.1.

~~Volume I~~

REFERENCE DESIGN,

by

J. B. Dee, ~~J. C. Bass, R. C.~~ Campana,
and G. B. West

Power TRIGA Reactor Group
TRIGA Reactor Program

for

U.S. Naval Civil Engineering Laboratory
Port Hueneme, California 93041

CR 68.011

Contract N62399-67-C-0046

May 31, 1968"

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P.O. Box 608, San Diego, California 92112

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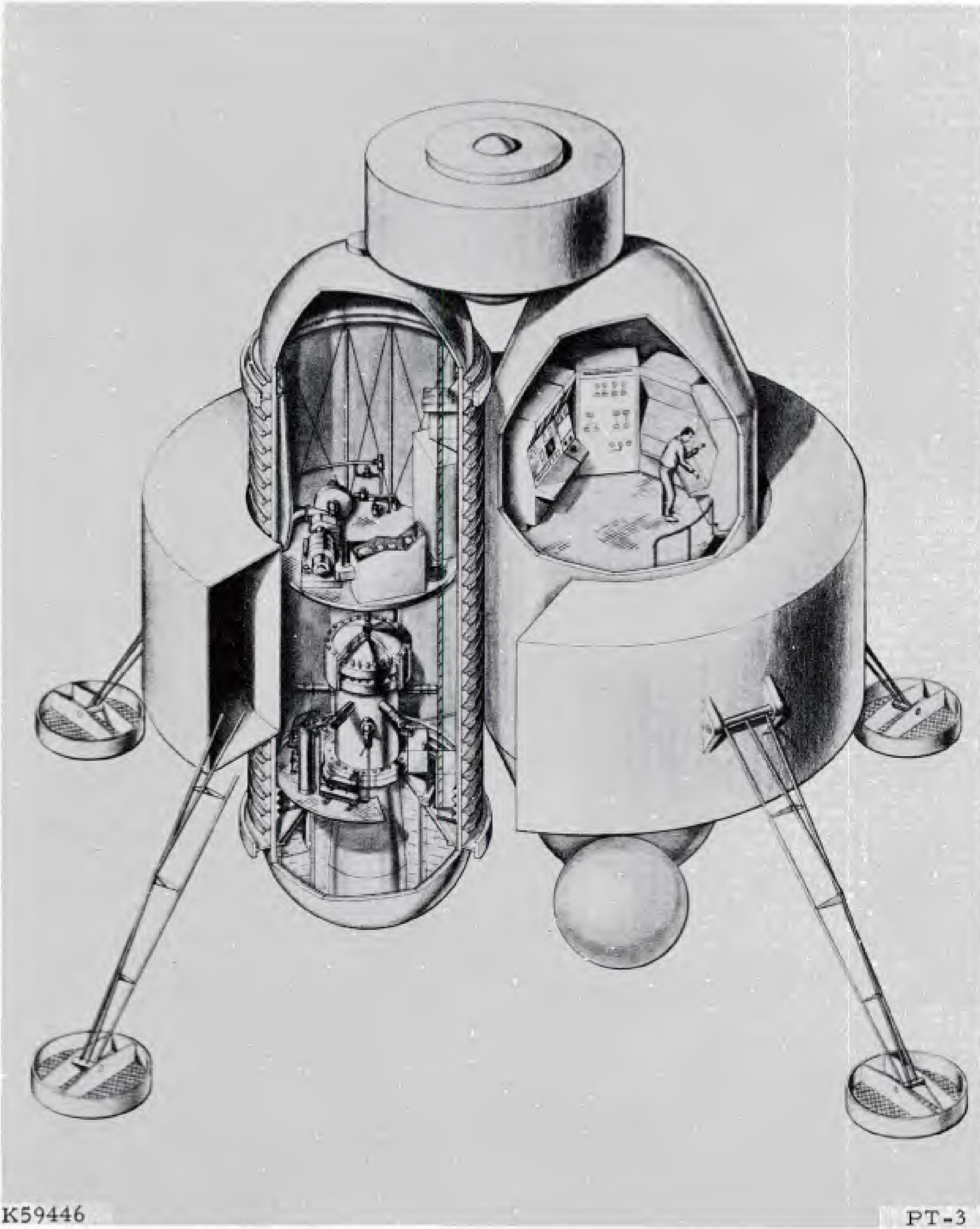

TOPS/MUS Nuclear Electric Power Plant

FOREWORD

This TOPS/MUS Reference Design Final Report covers work done under Contract NCEL N62399-67-R-0046, for a nuclear electric power plant system adapted from the TRIGA reactor which has been developed by Gulf General Atomic, Incorporated, under privately funded programs. Acknowledgment is made of the assistance of the many people at Gulf General Atomic who contributed to this report, in particular the staff of the Nuclear Analysis and Reactor Physics Department, the Electronics Division, and the Metallurgy Department.

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1. INTRODUCTION

This design study report describes a TRIGA Oceanographic Power Supply for a Manned Underwater Station (TOPS/MUS) under study by the United States Naval Civil Engineering Laboratory.

The MUS is characterized by operating conditions that have a strong bearing on the design of the electric power plant. These operating conditions include a remote and relatively inaccessible location (6,000-ft depth in ocean), moderate power requirements [100 kw(e)], a long mission duration (30 days or longer), and a small total crew (5 men). For these conditions of operation, a small nuclear reactor power plant is best suited for satisfying the power requirements of the MUS, provided that the reactor plant is safe, reliable, and sufficiently simple in operation to enable the crew to concentrate its attention on its primary function, that of conducting its naval mission.

A basic ground rule for this study is the utilization of current technology in both the reactor and energy conversion unit. The TOPS/MUS combines the extensive background of successful water reactor technology with the reliability and safety of the U-ZrH TRIGA fuel elements. The TOPS/MUS plant can be available to the Navy without need for component development and offers reliability and early availability at reasonable cost.

More than a decade of continuing TRIGA reactor development programs and experience in the manufacture and installation of 34 TRIGA reactors, combined with the use of commercially available power conversion equipment provide a basis for confidence that the TOPS plant is in the category of currently available equipment. Key features of the design include the following:

- State-of-the-art technology and commercially available components;
- Simplicity--no moving parts in the primary system;
- Inherent safety--U-ZrH/H_2O reactor is failsafe without reliance on operator;

- Continuous operator attendance not needed;
- Refueling based on manual research reactor practice.

The TOPS/MUS design philosophy has been aimed at satisfying the MUS requirements at reasonable cost through the adaptation of the highly-successful TRIGA series of research reactors to power conversion systems using components which are commercially available now. Static systems or devices are employed in the design wherever possible in preference to dynamic systems. In order to provide the least plant complexity subsystems of doubtful necessity are omitted with the understanding that they could be added later, if proven necessary, rather than adopting the alternate approach of including such subsystems now to be "conservative" and hoping to remove them later. The power and energy-producing capability of the reactor core and primary loop is sized to meet increased power requirements up to 500 kw(e) although the TOPS/MUS power conversion system has been based on an electrical power output of 100 kw(e). In order to meet Navy requirements for early availability an essential design precept was that existing machinery be used. Higher plant efficiencies could be attained if items requiring development were utilized in the design. Control simplicity and inherent nuclear safety are approached through the exploitation of the two large negative temperature coefficients of reactivity unique to H_2O-cooled, U-ZrH-fueled reactor cores. The approach to reliability is to minimize the number of devices that must be driven or controlled once the plant is in normal operation, and to provide automatic standby redundancy for pumps and blowers.

2. PLANT DESCRIPTION AND DESIGN SUMMARY

The TOPS/MUS nuclear electric power plant provides 100 kw(e) net power to a manned habitat as illustrated in the frontispiece. The power plant is generally divided into four groupings of equipment:

1. The nuclear steam supply module (primary system),
2. The power conversion equipment module (secondary system),
3. The power plant hull, and
4. The power plant instrumentation and control console in the habitat.

The general arrangement of the reactor plant is shown in Fig. 2-1.

The nuclear steam supply module includes the reactor vessel which is hermetically sealed and contains the entire primary coolant circuit, the inner shield tank containing both lead gamma-ray shielding and water neutron shielding, and a lead gamma-ray shield over the reactor vessel. The reactor core is located near the lower end of the reactor vessel, and a once-through steam generator is located near the top of the vessel. Primary water coolant flows by natural circulation (thermal siphon) up from the core to the vessel top and down through the annular steam generator to return to the bottom of the core. The core support shroud also acts to separate the two flow paths to maintain a full driving force. The primary loop is self-pressurized at essentially saturation pressure for the central riser water temperature. The six control drives are direct rack and pinion drives, each motivated by a hermetically-sealed stepping motor and are located near the vessel head. The primary system is sized for the capability to power a 500 kw(e) power conversion system so that at the 100 kw(e) rating for the TOPS/MUS the design conditions are even more conservative. The reactor vessel, fuel element cladding, end fittings, and the steam generator are made of Incoloy-800, for low corrosion rates and for freedom from stress-corrosion and radiation embrittlement. The principal remaining internal structures for which the stress levels are much lower, are made of stainless steels such as 304 L. The primary loop is cleaned and the water pretreated before filling and sealing. Further purity checks and water treatment are limited to periods of scheduled plant shutdown and maintenance.

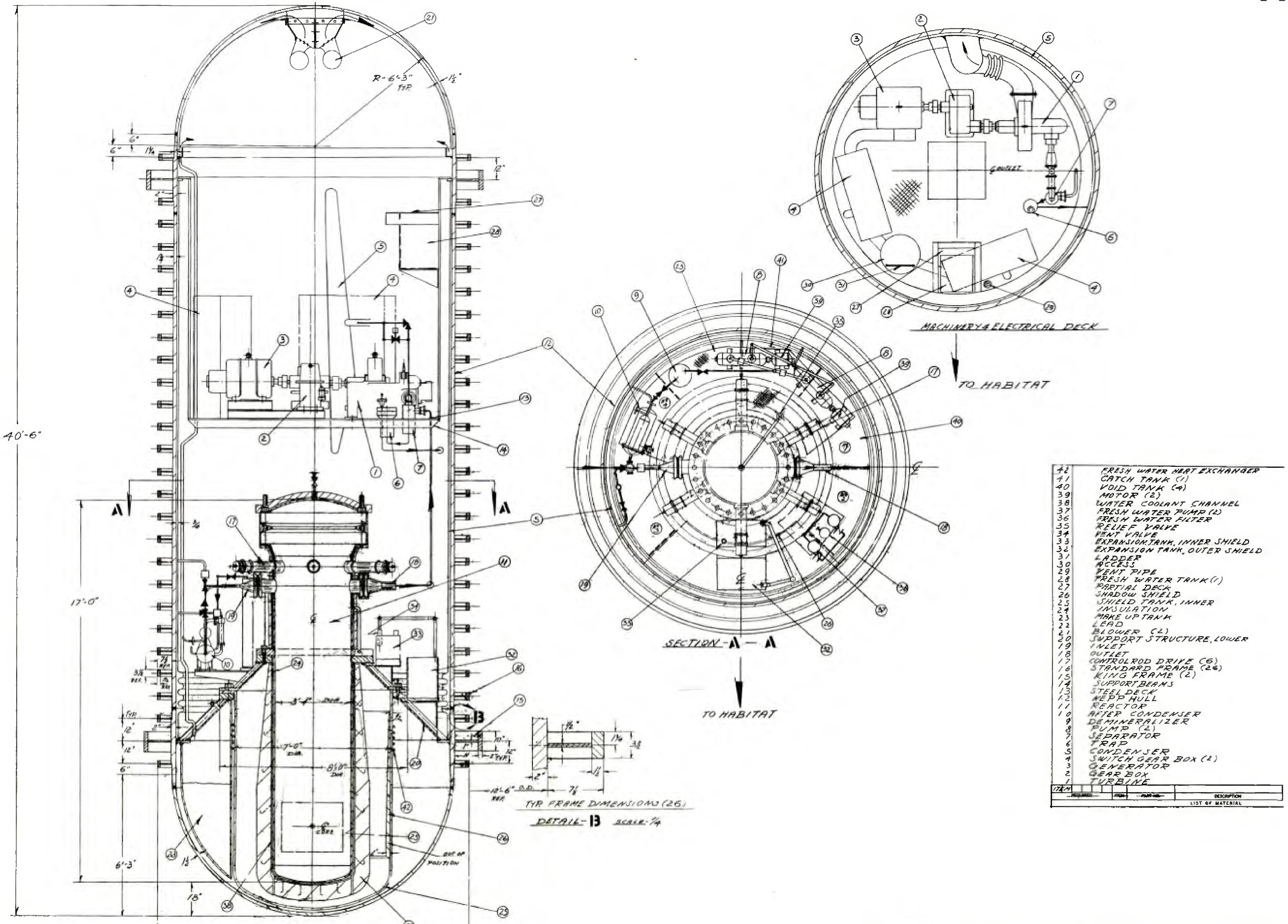

Fig. 2.1--TOPS/MUS nuclear electric power plant insullation arrangement

The power conversion system is based on the simple cycle represented by the heat balance diagram of Fig. 2-2. Feedwater entering the once-through steam generator leaves as superheated steam. Condensate that may form during startup or transient flow conditions is removed by a steam separator. Steam normally enters the turbine at 262 psi and 457°F at the design point and exhausts at 2 psia to the condenser and subsequently enters the feedwater pump. The power train includes a single-pressure-stage, two-velocity-stage Curtiss-type turbine wheel rotating at a constant speed of 6300 rpm and driving through a gear box at 1800 rpm, brushless 4-pole alternator generating 110 kw(e) gross electrical output at 60 Hz and 125/216 volts. The turbine speed is controlled by a constant-speed governor. The boiler feedpump is a multistage centrifugal pump. The power plant hull acts as the condensing surface and an insulated inner liner is used to separate condensing steam from the air space. Auxiliary systems include a full-flow demineralizer and filter; a water-cooling system for the generator, lubricating oil systems, and control drive motors; and an air blower for circulation across the top head of the hull for air temperature and humidity control.

The power conversion equipment module includes the turbine-gear box generator set with its foundations, steam separator, air ejector, and after-condenser, electrical system switchgear, and circulating pumps for the fresh-water cooling system. The power plant hull assembly includes the hull with its hatches and electrical penetrations, and internal support structures at two levels. From the upper ring the inner shroud of the condenser and the power conversion module are supported. At the lower level a support cone is attached for receiving the nuclear steam generator module. The support cone also holds the feedwater pumps at a level sufficiently below the hot well. The demineralizer is also located there to avoid unnecessary piping.

Power plant operations including startup are conducted remotely from the control console in the habitat. Startup will normally be performed at the surface prior to descent. However, provision is made for start or restart *in situ*.

A summary of some of the principal plant characteristics is given in Table 2-1, and a summary of plant auxiliary power requirements is given in Table 2-2.

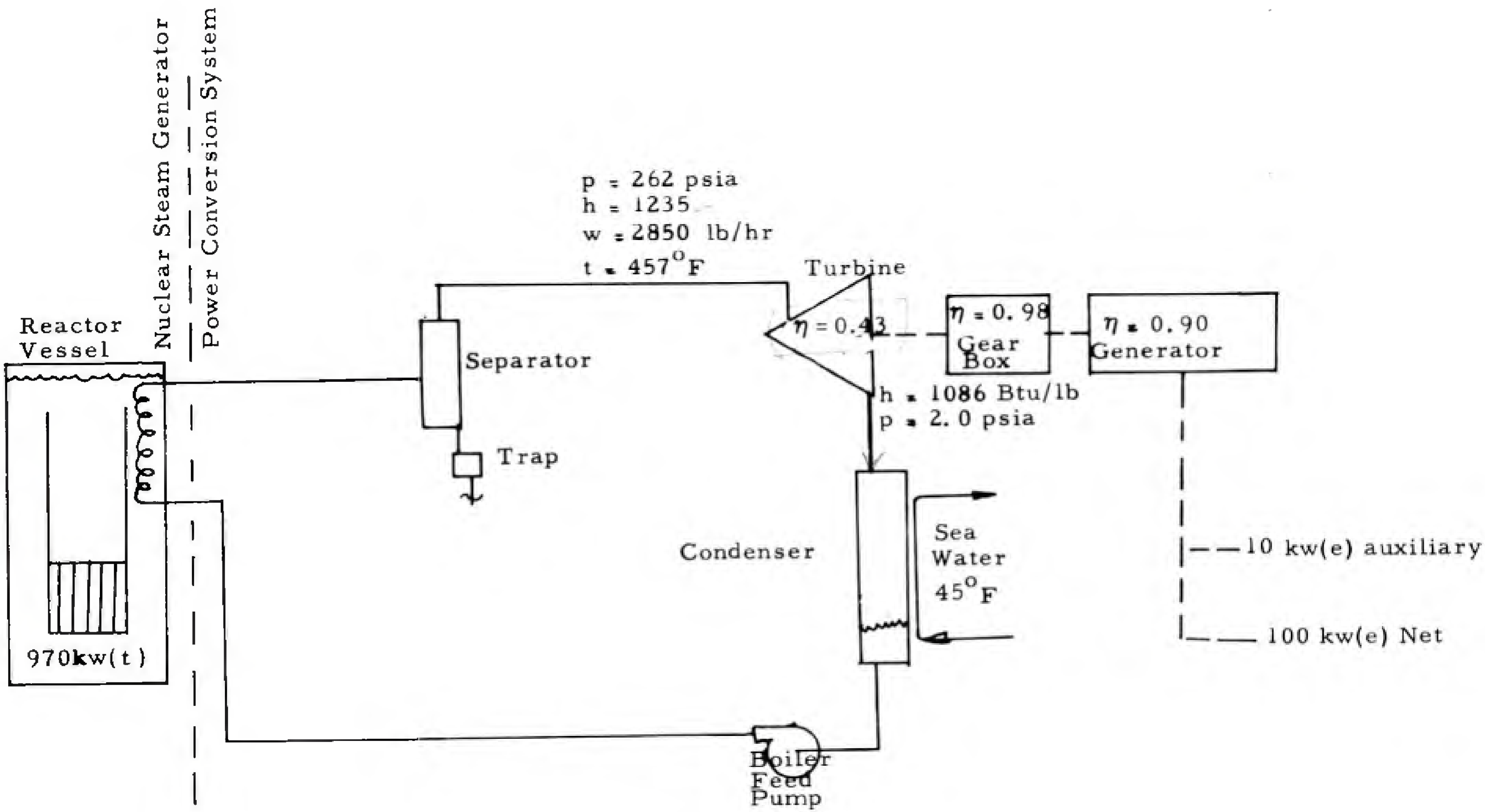

Fig. 2.2--Simplified heat balance diagram of 100 kw(e) TOPS/MUS power conversion system

Table 2-1

SUMMARY OF TOPS/MUS POWER PLANT PERFORMANCE

Electric Output	
Net power	100 kw(e)
Gross power	110 kw(e)
Voltage	125/216
Phases	3
Frequency	60 Hz
Reactor	
Thermal power	970 kw(t)
Core burnup life, mininum	3.1 Mw-years
Fuel material	$U\text{-}ZrH_{1.65}$
Uranium concentration in U-ZrH	8 wt-% (23% enriched)
U^{235} inventory	8.4 kg
Steam Power System (rated conditions)	
Throttle pressure	262 psia
Turbine steam flow	2850 lb/hr
Sea-water temperature (at 6,000 ft)	45°F
Condenser pressure	1.8 psia
Turbine exhaust moisture	3%
Efficiency of single-stage turbine	43%

Table 2-2

AUXILIARY ELECTRICAL POWER REQUIREMENTS

Item	Motor size (hp)	Power at operating point (kw)
Boiler feed pump	10	6.82
Fresh-water pump	1/2	0.25
Air blower	1/4	0.24
Total in Power Hull		7.31
Power conversion console power		0.40
Power distribution panel power		0.53
Reactor and primary console power		1.71
Total in Habitat		2.64
Total Auxiliary		9.95

The self-contained TOPS nuclear steam supply module can also be adapted to other applications by replacement of either the condenser, the power conversion module, or both. For shore-based applications or at shallow depths, a compact standard type of condenser can be used to replace the special-purpose hull-condenser employed here which simultaneously combines the functions of a leak-proof pressure hull and an adequate heat rejection surface, while providing neutral buoyancy.

For high power levels up to 300 kw(e), a similar power conversion system can be used employing the same turbine-gearbox, but with the generator, feedpump, and related components selected for higher ratings. Power levels in the range from 300 to 500 kw(e) can be obtained by increased power conversion system complexity through the use of a multiple-pressure-staged turbine with interstage condensate extraction combined with feedwater heating.

In those cases for which a more compact reactor vessel would be advantageous, the reactor pressure vessel can be reduced in height. In such a case, final selection of the vessel height would result from a trade-off of the benefits of reduced weights and dimensions compared with the increase in the equipment and maintenance required for accommodating fuel handling, top shielding, and secondary coolant activation and radiolysis for close compaction.

3. NUCLEAR STEAM SUPPLY

The components and systems associated with the nuclear steam supply module are described in this section. The nuclear steam supply includes the primary coolant loop of a two-loop indirect cycle system. The primary loop is completely closed, hermetically sealed, and there is no intermingling at any point with water in the secondary circuit.

Except for the control rod drives there are no controlled or moving parts in the primary circuit. The reactor vessel is sized to provide an adequate head for circulation at power levels well above the 100 kw(e) MUS power requirement. The system is self-pressurized so that electric heaters, quench circuits, and their controls are not required. With hermetic sealing, contaminants cannot be brought into the system with make-up water, and equilibrium water chemistry conditions can be established. The reactor vessel insulation provides for a small but sufficient heat loss by conduction and natural circulation through the shield tank to the sea for decay heat removal. In this manner, static systems are employed to satisfy necessary functions with reliability and safety.

3.1. GENERAL

The nuclear and thermal design characteristics which constitute the basis for the component designs described in the subsequent subsections are described herein.

3.1.1. Nuclear Design

The overall nuclear design objective has been to use the TRIGA fuel elements in a core configuration which has

1. Large prompt negative fuel temperature coefficient of reactivity to provide overall inherent safety and control stability with a load-following response to power demand changes,
2. A water temperature coefficient of reactivity that changes over the operating range of temperatures in such a manner as to achieve low excess reactivity requirements without compromising safety,
3. An adequate fuel burnup life, and
4. A moderate fuel inventory.

These objectives are achieved by the core described here.

The nuclear analysis has been based on reactor physics methods whose validity has been demonstrated by comparison of calculated experimental data. These comparisons have been made covering an extensive history of TRIGA reactor design modifications, critical experiments, and irradiation test configurations, and on the performance and analysis of experiments on fundamental neutronic properties of ZrH.

The fuel element dimensions are the same as for current TRIGA fuel. The number of fuel elements (193) was established primarily by thermal considerations, as outlined in the following section. The coolant water fraction and fissile loading have been adjusted to satisfy the nuclear design objectives. The principal nuclear design data are summarized in Table 3-1 and are discussed briefly below. Details of the nuclear analyses are given in Appendix A.

Power Distribution. The radial and axial distributions of power generation in the reactor core are shown in Figs. 3.1 and 3.2. The radial distribution of power generation in a fuel element at the design point is shown in Fig. 3.3.

The U^{235} loading of 8.4 kg provides for a reactivity lifetime of 3.1 Mw(t)-year for which the U^{235} consumption is 1.4 kg. To minimize the reactivity change associated with this fuel burnup, a burnable poison is incorporated in the fuel element. Erbium is used both for this purpose and to enhance the prompt negative fuel temperature coefficient of reactivity. Erbium is metallurgically similar to zirconium and has been demonstrated to alloy well in the required concentrations without interfering with the hydriding process. The nuclear analyses for this study were based on erbium, homogeneously mixed with the fuel material. Both homogeneous and lumped poisons were considered. Selection of this homogeneous method is based on the results of a current Gulf General Atomic in-house programs for which this type of TRIGA research reactor fuel is currently being manufactured.

The microscopic cross-section of Er^{167}, the absorbing isotope, at a neutron energy of 0.1 electron volts, for example, is 457 barns, which makes it of interest as a burnable poison for compensating for U^{235} burnup during core life. The enhancement of the prompt fuel temperature coefficient is produced by the shift in the thermal neutron spectrum with increasing temperature into the low-lying neutron absorption resonance, as illustrated in Fig. 3.4.

The fuel temperature coefficient of reactivity at the design point is large,

$$-1.4 \times 10^{-4} \ \delta k/k\text{-}^{\circ}C \quad ,$$

Table 3-1

REACTOR NUCLEAR DESIGN DATA SUMMARY

100 kw(e)

Active Core Configuration	
Outside diameter (mean)	29.5 in.
Height	15.0 in.
Volume	5.93 ft^3 (168 liters)
Number of fuel elements	193
Number of control rods	6
Active Core Cell Configuration	
Fuel element diameter	1.475
Fuel element pitch	1.980 in.
Volume-fraction U-$ZrH_{1.65}$	0.47
Volume-fraction cladding	0.03
Volume-fraction coolant	0.50
Active Core Composition	
Weight-fraction Uranium in U-$ZrH_{1.65}$	8%
Uranium inventory (23% enriched)	36.4 kg
U^{235} inventory	8.4 kg
Zirconium inventory	418 kg
Er^{167} concentration (core average)	6×10^{18} atoms/cc
Power Distribution	Rated Power
Cell power density peak local/average	1.15
Core radial power density peak local/average	1.78
Core axial power density peak local/average	1.25
Reactivity	
Cold, clean excess reactivity ($)	5.48
Control worth (6 rods) ($)	6.75
Burnup	
Energy removed	3.1 Mw(t)-year
U^{235} consumed	1.4 kg
Peak fraction metal atoms fissioned	<1%
Reactivity Coefficients About Rated Power	
Fuel prompt coefficient	-1.40×10^{-4} $\delta k/k$-°C
Water temperature coefficient	-0.5×10^{-4} $\delta k/k$-°C
Core water density coefficient	+0.076 $\delta k/\delta\rho$ cm^3/gm
Core Nuclear Parameters	
Neutron lifetime	45 μsec
β_{eff}	0.007
Neutron Flux	
Thermal neutron flux, core average, ($E < 1.125$ eV)	5.0×10^{12} n/cm^2-sec
Fast neutron flux, core peak ($E > 0.6$ MeV)	7.3×10^{12} n/cm^2-sec
Fast neutron current into reactor vessel	8×10^{10} n/cm^2-sec

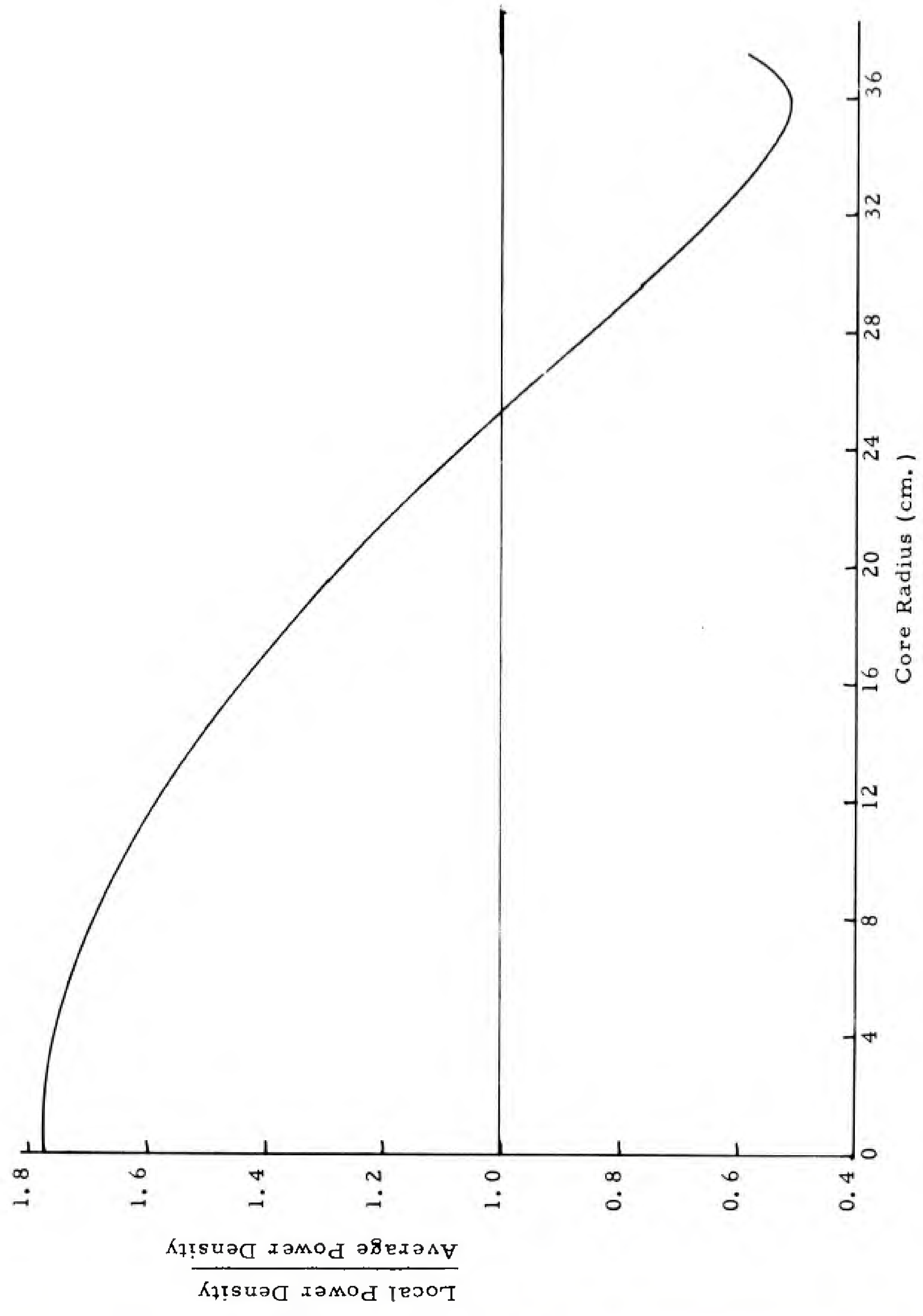

Fig. 3.1--Radial power distribution in core

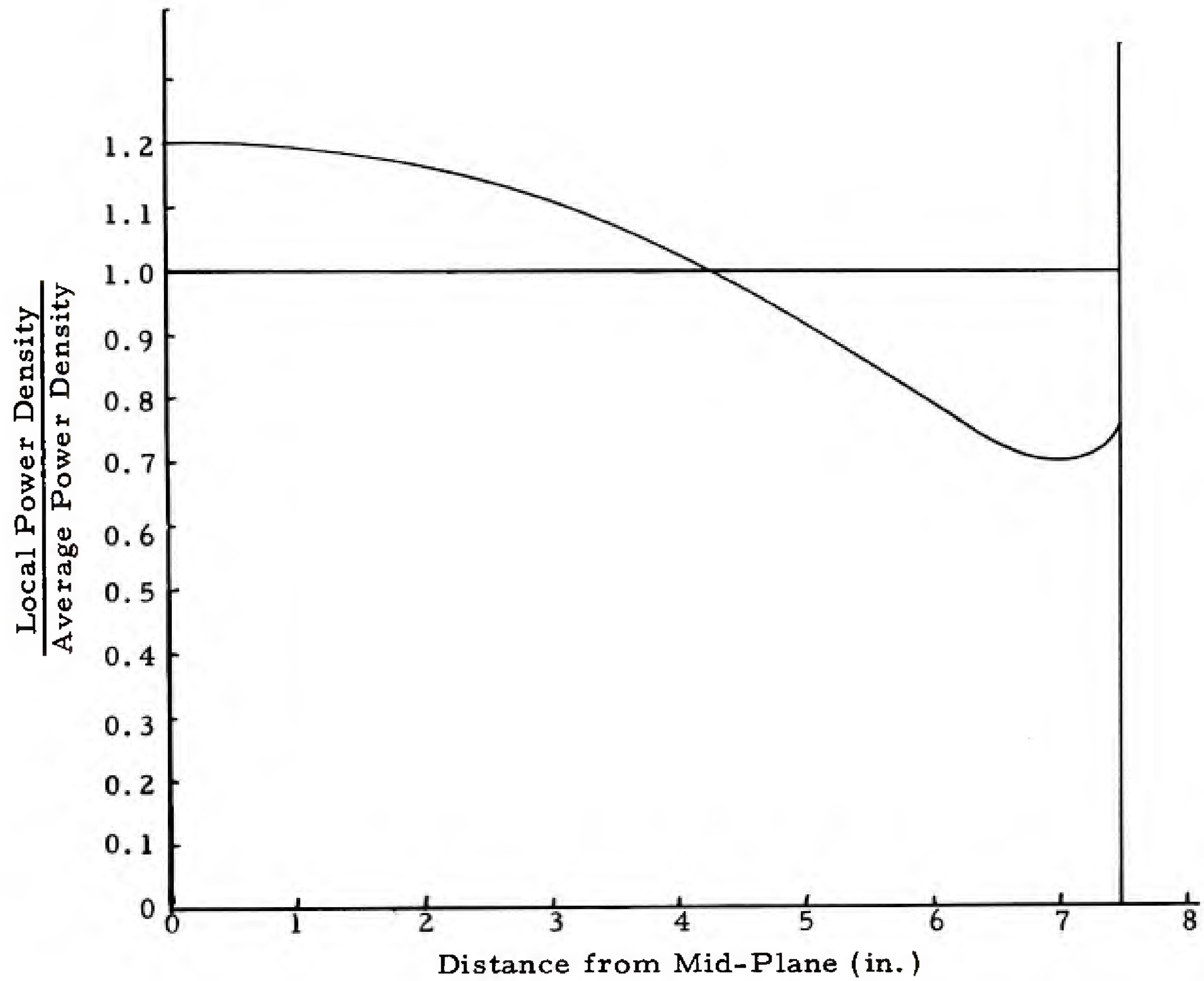

Fig. 3.2--Axial power distribution in core

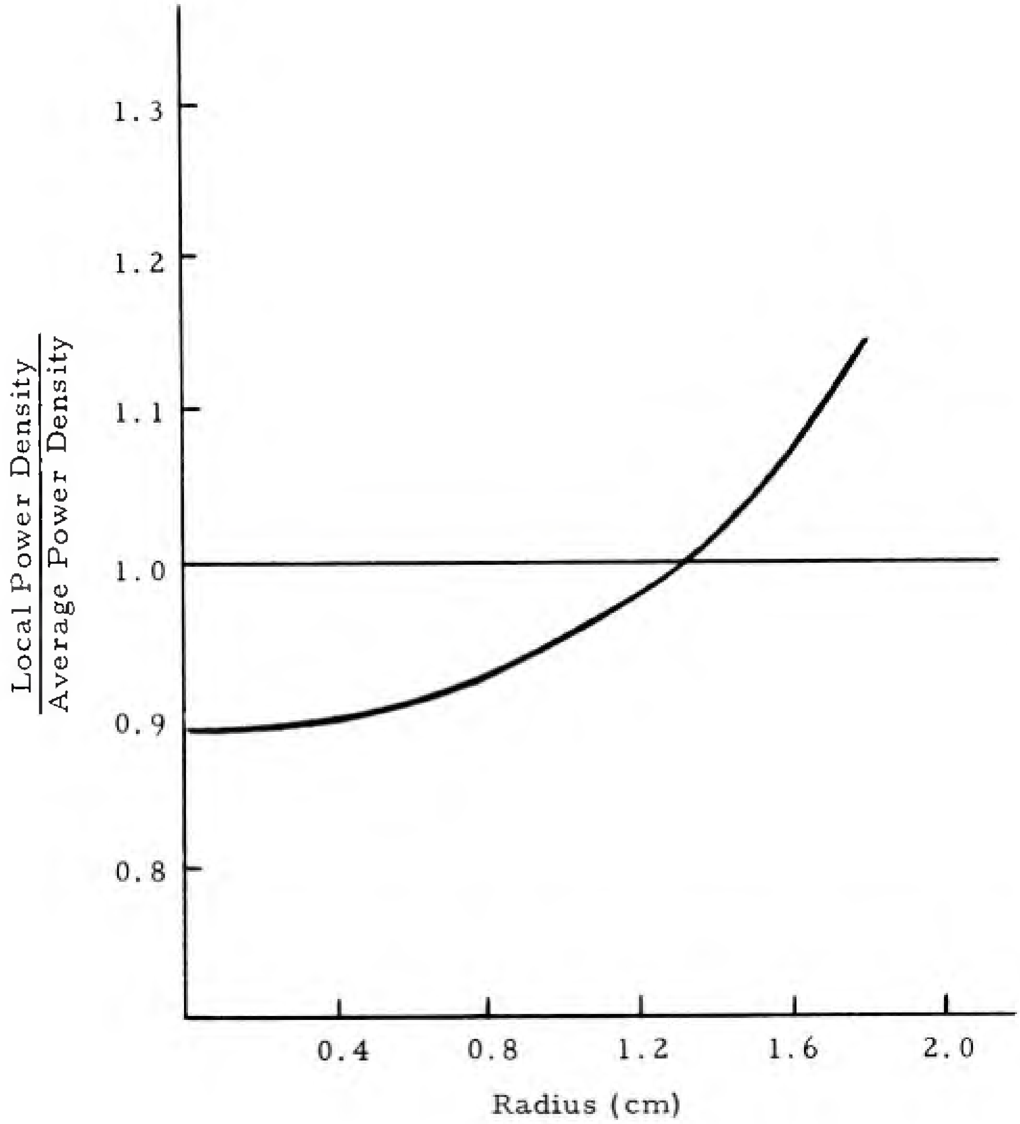

Fig. 3.3--Radial power distribution in fuel element

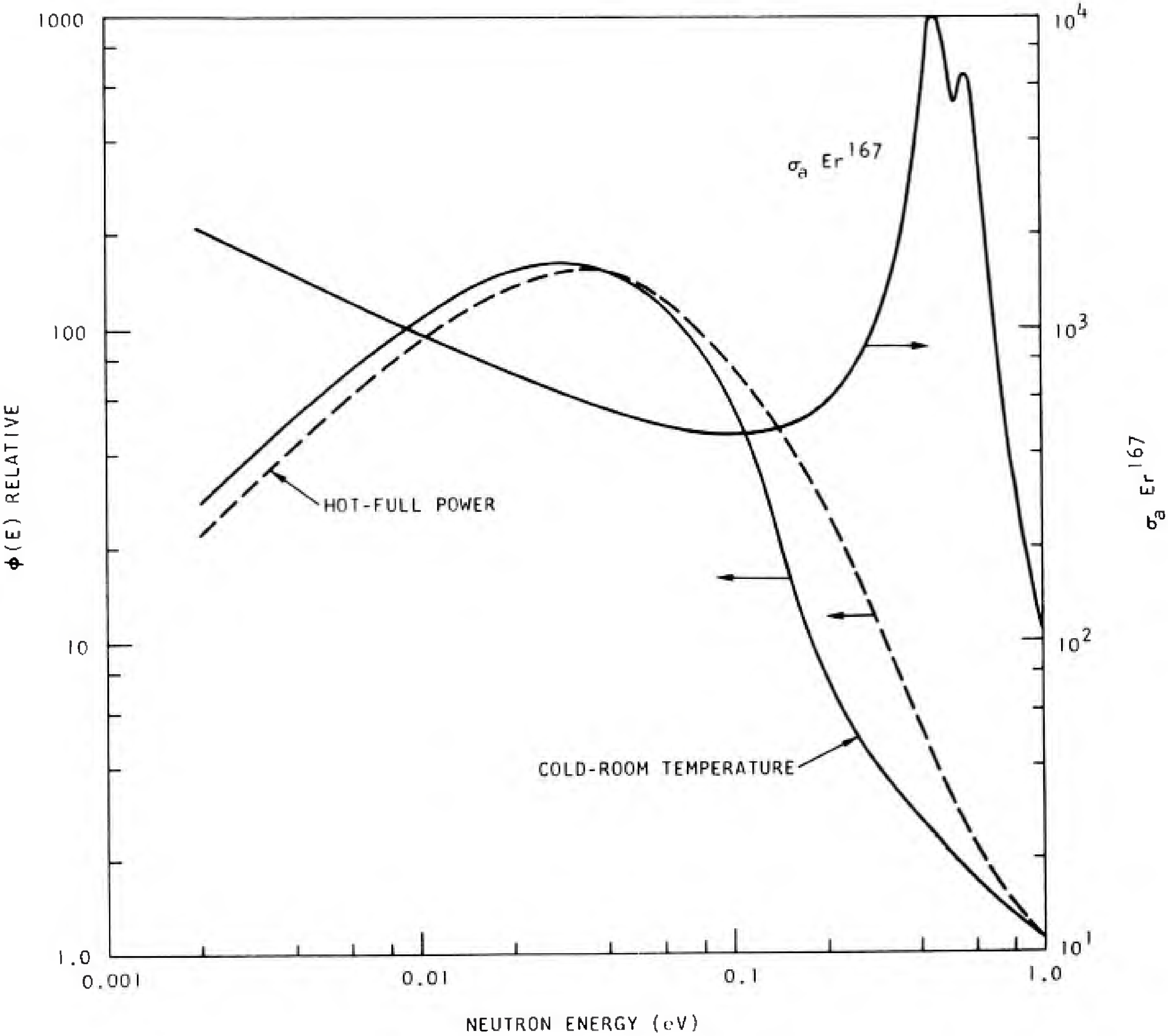

Fig. 3.4--Thermal neutron spectrum in core

and is prompt in that there is no dependence on time-dependant heat transfer from fuel to water. The core water temperature coefficient of reactivity is -0.5×10^4 $\delta k/k$-^{o}C at the same point. However, reactivity changes slowly with power changes because the core water temperature changes slowly as a consequence of the large heat capacity of the primary water inventory. Both of these coefficients are temperature-dependent, as shown in Fig. 3.5, and become larger with increasing temperature. These coefficients are determined by first generating complete neutron cross-section sets for a combination of fuel and water temperature conditions, then computing the reactivity of the system for each case, and subsequently taking differences between cases. The core water density coefficient was similarly obtained at the design point to be $+0.076$ $\delta k/\delta \rho$.

The time-dependent, equilibrium, and xenon override reactivity worths are shown in Fig. 3.6. At this power level the thermal neutron flux is sufficiently low that override after shutdown is negligible.

Reactivity requirements for the initial core loading are listed in Table 3-2. The total requirement is $5.48. Nuclear excursion analyses were performed, which show that $7.00 of reactivity can be safely inserted into the cold reactor core in a single step. Consequently it is seen that the TOPS/MUS core is completely safe against a nuclear excursion. This aspect is discussed more completely in Section 8, and in Appendix A.

Control requirement analysis indicates that 5 boron carbide control rods of 1.300-in. outside diameter, located as shown in Fig. 3.7, will provide sufficient control to shut down the reactor in case the sixth rod could not be inserted. Considerably larger control rods could be accommodated by the available space in the core positions for the control rods so that there is no question as to the capability for satisfactory final adjustment of control rod worth.

3.1.2. Thermal Design

Heat transfer in the reactor vessel takes place:

1. In the core through conduction in the fuel elements and local boiling, then
2. By transport of the heated core water by natural circulation of the primary, and subsequently
3. By heat transfer through the steam generator.

A summary of thermal conditions at the TOPS/MUS design point is given in Table 3-3.

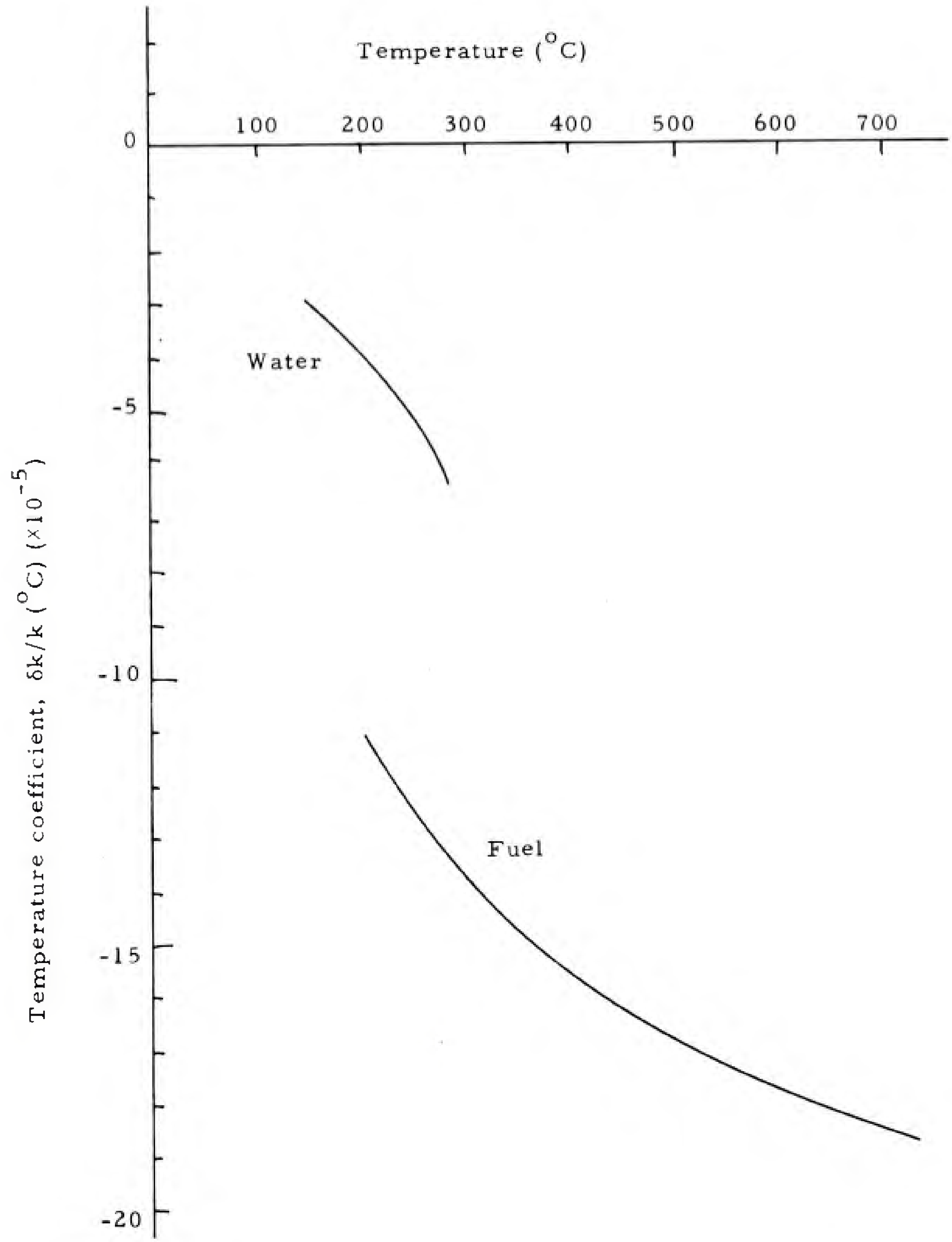

Fig. 3.5--Temperature coefficients of reactivity

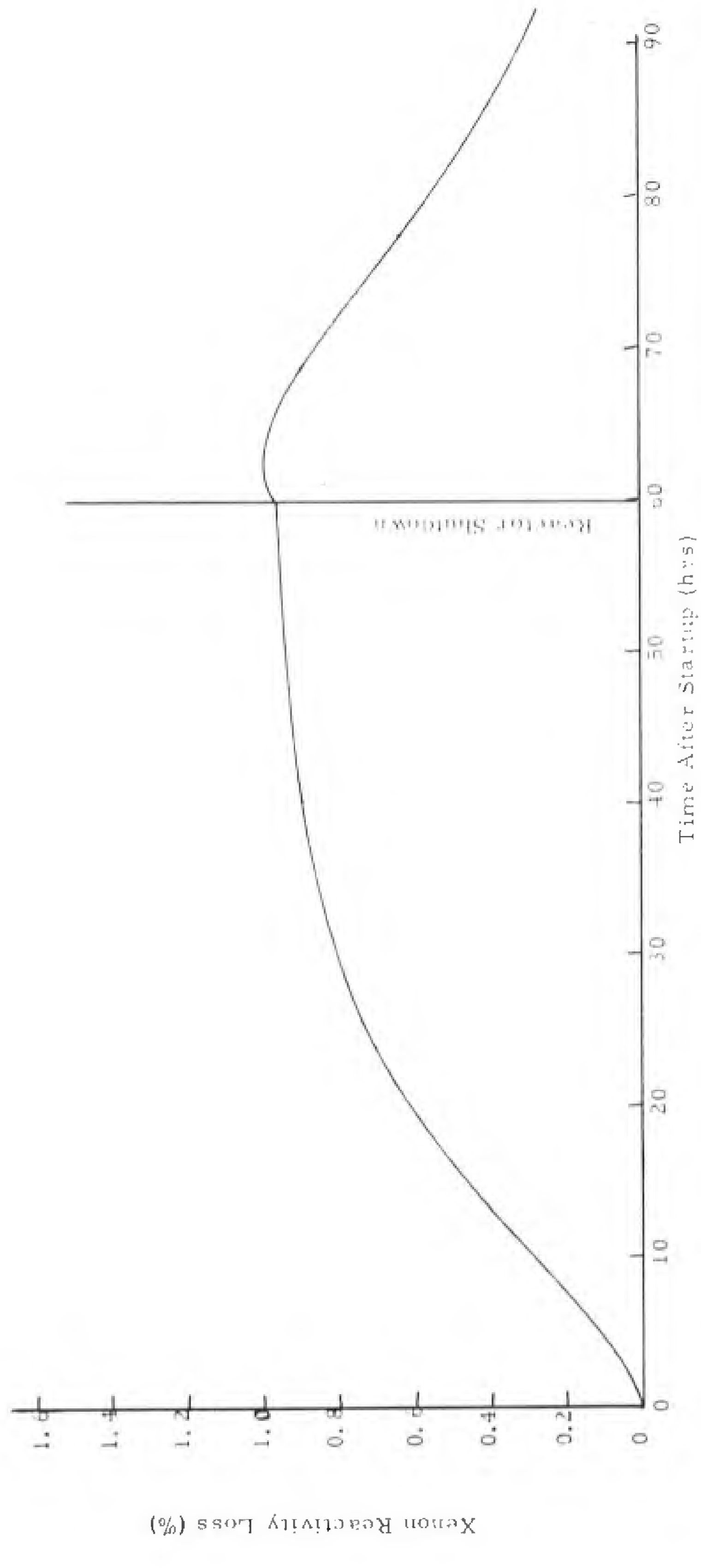

Fig. 3.6--Xenon reactivity loss

Table 3-2

REACTIVITY REQUIREMENTS

100 kw(e)

Type	δk (%)	δk ($)
Cold-to-hot (100% power)[a]	-1.87	-2.67
Equilibrium xenon	-0.97	-1.39
Contingencies		
Uncompensated burnable poison	-0.50	-0.71
Control margin	-0.50	-0.71
TOTAL	-3.84	-5.48

[a]Based on operating conditions of 310°C average fuel temperature and 250°C average water temperature.

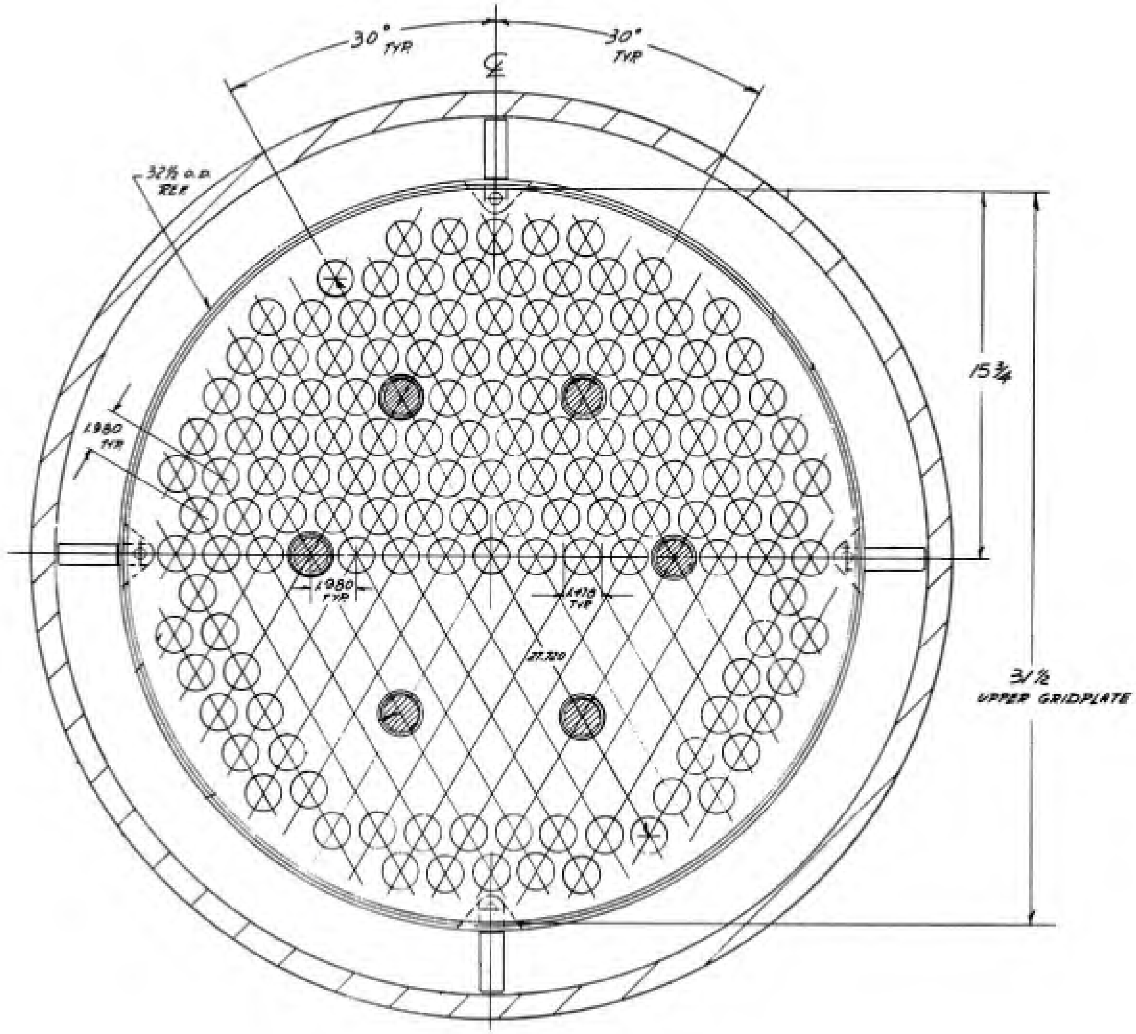

Fig. 3.7--Core layout

Table 3-3

SUMMARY THERMAL DESIGN POINT CONDITIONS

Reactor thermal power, 970 kw(t) (Btu/hr)	3.3×10^6
Coolant flow rate (lb/hr)	226,000
Coolant pressure (psia)	515
Core inlet temperature (°F)	450
Core outlet temperature (°F)	470
Channel hydraulic diameter (in.)	1.09
Heat transfer area (ft^2)	93
Heat flux	
Average (Btu/hr-ft^2)	35,600
Maximum (Btu/hr-ft^2)	76,000
Maximum during startup, 3100 kw(t) (Btu/hr-ft^2)	238,000
DNB (burnout) ratio	
Mimimum (3100 kw(t) during startup, 212°F water)	1.6
At design point	15
U-ZrH fuel temperatures	
Volumetric average	578°F (303°C)
Maximum	794°F (423°C)

Core Heat Transfer.

The fuel temperatures at the design point are mainly determined by the thermal power produced in the core and by the saturation temperature of the core water. The basic individual fuel element dimensions are unchanged from current standard practice. The spacing between fuel elements is selected on the basis of neutronic analysis to provide desirable reactivity coefficient characteristics. The resulting fuel element spacing produces a core that is open to a free flow of coolant (hydraulic diameter in excess of one inch) unobstructed by the in-core spacers or internal structural members. The number of fuel elements in the core is established by the heat transfer requirements at the higher power level of 3.1 Mw(t) (10.5×10^6 Btu/hr). For 193 fuel elements the maximum heat flux in the core at this higher power level would be 238,000 Btu/hr-ft^2 which is typical of the peak heat flux experienced in a small TRIGA research reactor similarly cooled by natural circulation of pool water. At room temperature the heat flux for departure from nucleate boiling conditions (DNB) is 380,000 Btu/hr-ft^2 so that the DNB ratio is 1.6. It is this condition that establishes the core size. At higher pressure the critical heat flux for DNB is greater and is approximately 1,200,000 Btu/hr-ft^2 for pool boiling providing a DNB ratio of 3.2 at the high design point which is considered to be within the normal design range for safety, particularly in view of the openness of the core to free coolant flow. For the TOPS/MUS design point corresponding with 100 kw(e) the DNB ratio at the peak power location in the core is 15, which is clearly quite conservative.

The heat transfer mechanism from fuel to water in the core is by subcooled and saturated local nucleate boiling so that the temperature difference between the clad temperature and the water saturation temperature is always small; e.g., $T_{wall} - T_{sat} = 12^{\circ}F$ peak for TOPS/MUS design point. At this same location, the remaining temperature differences are 21°F for the clad, 63°F for the clad-fuel interface, and 179°F in the fuel body so that the maximum fuel temperature is 737°F (392°C). For the higher power level the maximum corresponding fuel temperature is 1195°F (645°C).

Except for minor heat losses, steam cannot be removed or condensed in the expansion space (approximately 15 vol-% of the coolant volume at the design point). Consequently, at that location the steam and coolant are in thermal equilibrium at the saturation pressure (515 psia) corresponding with the coolant temperature (470°F). Coolant subsequently descending into the steam generator is subcooled 14°F and then returns to the core, where it is heated to saturation conditions during steady-state operation. The mean height between the heat source (reactor core) and the heat sink (steam generator) is 9.4 ft, resulting in a net driving force of 0.062 psia at a power level of 970 kw(t). Most of the compensating pressure drop in the loop occurs in the crossflow heat exchanger (0.057 psia) which results in part from the large flow cross-section in the core (compatible with the velocity independent nucleate boiling mode of heat transfer) and from the higher velocity of flow and larger heat transfer surface in the steam generator (required by the non-boiling water film heat transfer mode). Further discussion of heat transfer in the primary loop appears in Appendix B.

Steam Generator

Boiler feedwater entering the steam generator passes through three contiguous regions; the economizer section in which the temperature of the sub-cooled feedwater is raised to saturation, the evaporator section in which saturated nucleate boiling takes place, and the superheat region in which the steam vapor temperature is increased to approach that of the primary coolant. The heat transfer in the steam generator is reviewed in Section 3.6, and in Appendix B. Generally, the steam generator has been sized for the 3.1 Mw(t) case with the aid of a computer program used in commercial power reactor programs and by independent analyses using accepted heat transfer relations. For the TOPS/MUS application the heat transfer area is conservative by approximately a factor of four. A transient analysis was made which confirmed that the response of the reactor and primary loop system is stable to changes in thermal load, and is discussed in Appendix C.

3.2. REACTOR

3.2.1. Reactor Core

The reactor core forms a cylinder 28.5 in. in diameter with an active core height of 15 in. The active core is side-reflected by approximately 4 in. of water. End reflection is accomplished by water and by 3.4-in.-long graphite cylinders in the ends of the fuel elements. The core consists of a regular equilateral triangular array of 193 fuel elements and 6 control rods spaced on triangular centers 1.980-in. apart. The cylindrical fuel elements are 1.475 inches in diameter and the volume-fraction of the core lattice occupied by fuel elements is 0.50. The control rods are also cylindrical and are 1.30-in.-OD. The general arrangement of the reactor core within the pressure vessel is shown in Fig. 3.8.

Upper and lower grid plates are used to locate the individual fuel elements in the core. Holes in the upper grid plate are each slightly larger in diameter than a fuel element to permit insertion of an element into the core. The top end fitting has an integral tri-flute section that serves to center the top end of the fuel element in the hole in the upper grid plate while still providing a large area for flow of the primary coolant. Holes in the lower grid plate are slightly smaller in diameter than a fuel element to provide a seating surface for the tapered tri-flute on the bottom end fixture of the fuel element. The bottom tri-flute is tapered so as to be tangent to the cone having a 30 degree half-angle to ensure positive and accurate seating. Above the upper grid plate is a similarly shaped holddown plate which can be shifted 0.25-in. laterally as a safety device to prevent fuel from becoming displaced in case the entire system were to be accidentally inverted. The length of the bottom tri-flute is sufficient to ensure that the bottom end of the element cannot be removed from its grid plate hole in the event of a marine accident of this nature.

The flow pattern of cooling water in the reactor vessel is divided into two legs by a flow divider which also acts as the structural support for the reactor grid plates, as shown in Fig. 3.8. Water heated by the core rises through the central riser to the upper part of the reactor vessel. Subsequently, the hot water is cooled as it descends through the steam generator tube bank and continues down through the core reflector region to the bottom of the reactor vessel, where it is returned to the core. In order to channel full flow of water through the core, any unused grid plate holes will be plugged with dummy fuel elements and flow leakage paths between the grid plate and flow divider will be minimized.

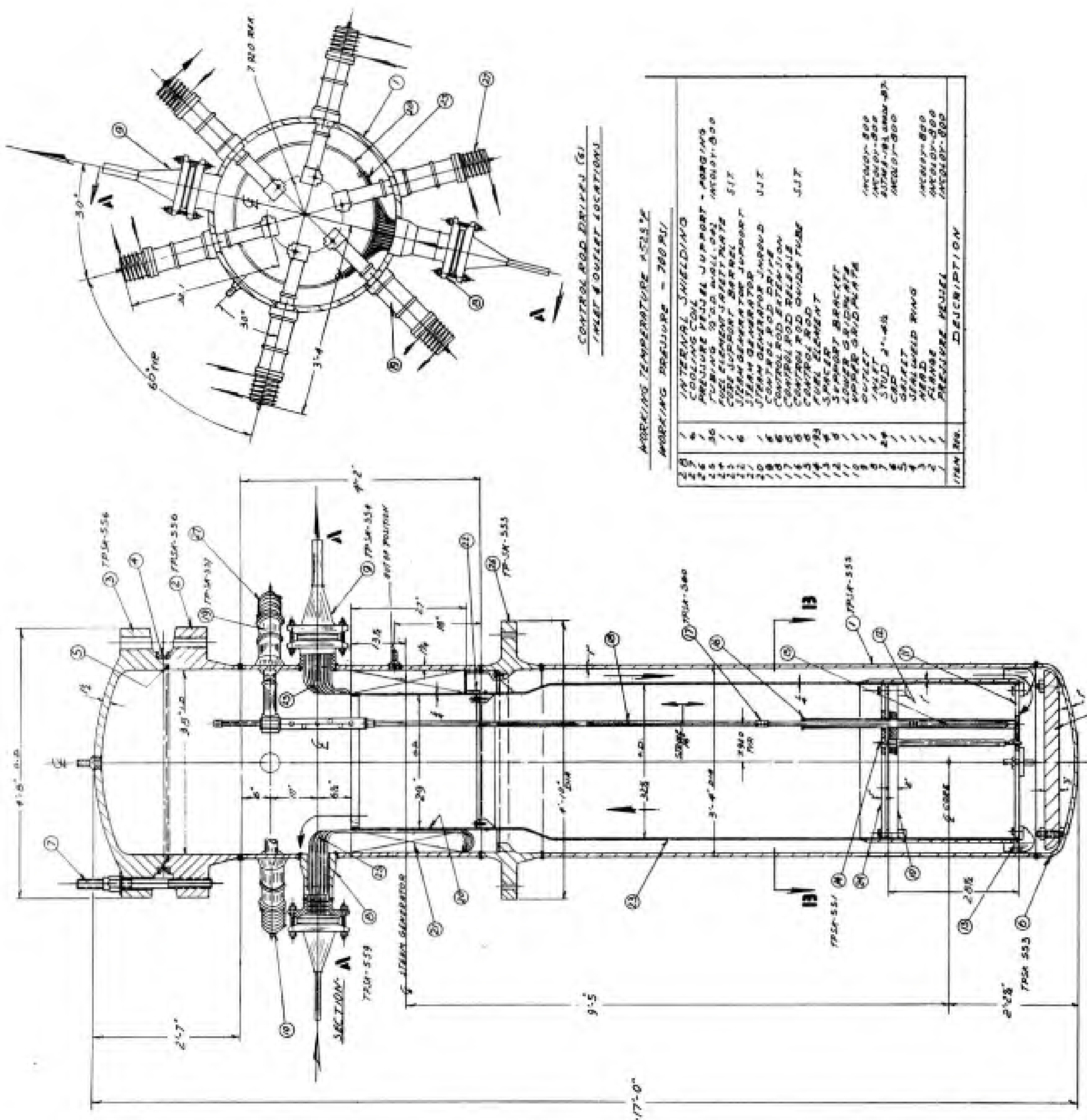

Fig. 3.8--Nuclear steam supply (vertical section, top view)

The control rods are individually operated by separate drives, which are described in detail in Section 3.4. The drive mechanism contains a rack and pinion to provide vertical movement of the control rods in the core. The control rods are guided through the core through fixed stainless-steel perforated tubes.

Type 304 or 304L stainless steel is used as the basic material for the reactor internal structures. The principal exceptions are the fuel element cladding, the steam generator, and the reactor vessel, which are more highly stressed components and are made of Incoloy-800.

3.2.2 Fuel Elements

The fuel elements are TRIGA research reactor fuel elements which act as fuel and as a neutron moderator simultaneously. The active section of the element is a cylindrical fuel body 15 in. in length and 1.435 in. in diameter and consists of a metallic alloy of uranium, zirconium, erbium, and hydrogen. The uranium content is 8 wt-% of total uranium enriched to about 23% U^{235}. The hydrogen-to-zirconium atom ratio is 1.65 to 1. The erbium content is approximately 0.015 g/cc of elemental erbium which is relatively small from a metallurgical standpoint. As shown in Fig. 3.9, graphite cylinders 3.4 in. in length and 1.4 in. in diameter act as top and bottom reflectors.

The fuel and the top and bottom reflectors are contained in cladding which consists of a 0.020-in.-thick Incoloy-800 seamless tube. The cladding is welded to the top and bottom end fittings which are Incoloy-800 castings. The top end fitting is shaped to fit and lock into a ball detent type of long-handled fuel handling tool. The regions of both end fittings that contact the grid plates have a tri-flute cross-sectional shape to provide an adequate coolant flow passage. The lower tri-flute is tapered 30° to seat in a chamfered hole in the lower grid plate, which supports the weight of the fuel element. The approximate overall weight of the fuel is 7.5 lbs and the U^{235} content is 43 grams.

Data on the physical properties of U-ZrH and the status of fuel metallurgy and fabrication technology are well established and more than 30 reactors using TRIGA fuel elements are currently in operation. The uranium loading provides for a reactivity lifetime of 3.1 Mw(t)-years. For this degree of fuel burnup, the maximum fraction of the metal atoms fissioned, including U and Zr, is about 0.004. The maximum fuel temperature is less than $800^{\circ}F$, and as a consequence, the irradiation burnup rating is extremely conservative.

The hydrogen content and distribution under these conditions is highly stable and no significant metallurgical effects can be expected in terms of hydrogen loss, migration, swelling, or phase changes.

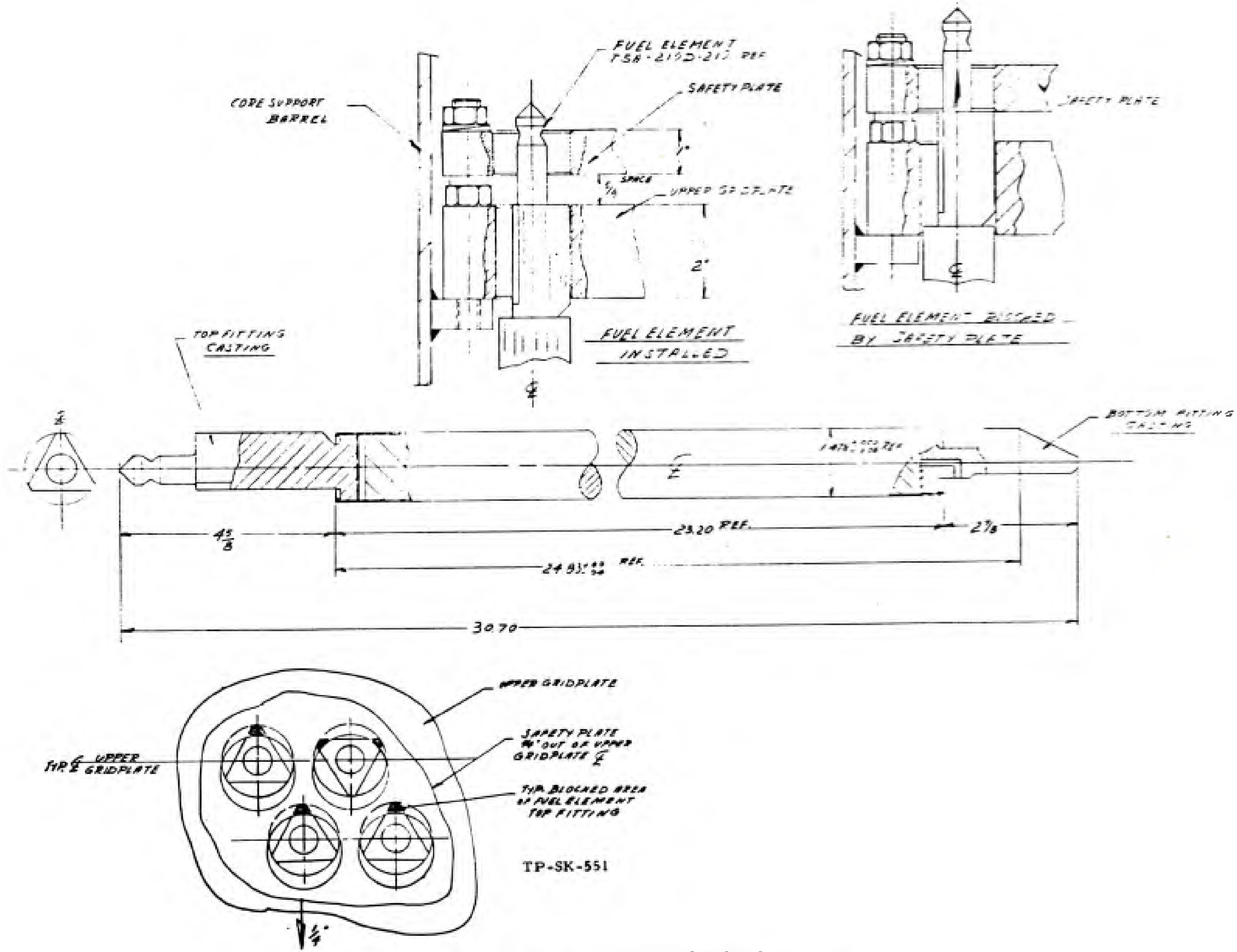

Fig. 3.9--TOPS fuel element

3.3. REACTOR VESSEL AND INTERNAL COMPONENTS

3.3.1. Reactor Vessel

The TOPS/MUS reactor pressure vessel is shown in Fig. 3.8. The general configuration of the vessel is that of a right circular cylinder nominally 40 in. in diameter by 204 in. high. The lower head is of standard 2:1 semi-elliptical design and is shown in Fig. 3.10. The upper head shown in Fig. 3.11 is also semi-elliptical, and is integral with the vessel flange forging. The reactor shell structure is shown in Fig. 3.12.

Detail design of the vessel will be in accordance with the A.S.M.E. Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Pressure Vessels. The entire vessel will be constructed of Incoloy-800. The reasons for the selection of this material over such candidate materials as 304L and 316L are discussed in detail in Appendix E. The pressure vessel will be designed to operate at a maximum pressure of 1100 psia and 560°F.

The total dry weight of the pressure vessel shell and insulation will be approximately 13,000 lbs. The entire weight of the reactor and its internals is supported by a truncated conical skirt which is part of the upper portion of the reactor vessel shell. The support skirt is located as near as possible to the center of mass of the reactor system. This was done to prevent the transmission of any extraneous moments into the reactor structure due to unexpected lateral accelerations. This will be of particular importance while the MUS system is being transported to the implantation site. For towing in high-sea states it may be advisable to take further precautions such as providing stabilizing devices which would be removed prior to the implantation of the MUS.

3.3.2. Core Support and Internal Components

Support for the reactor core and the integral once-through steam generator originates on a machined surface internal to the vessel on the same plane as the main vessel support. The core support barrel is connected by a number of vertical plates to a ring which rests on the machined surface of the support forging. Particular attention is given here to the hydrodynamic design of both the vessel and its internals in order to minimize restrictions to coolant flow.

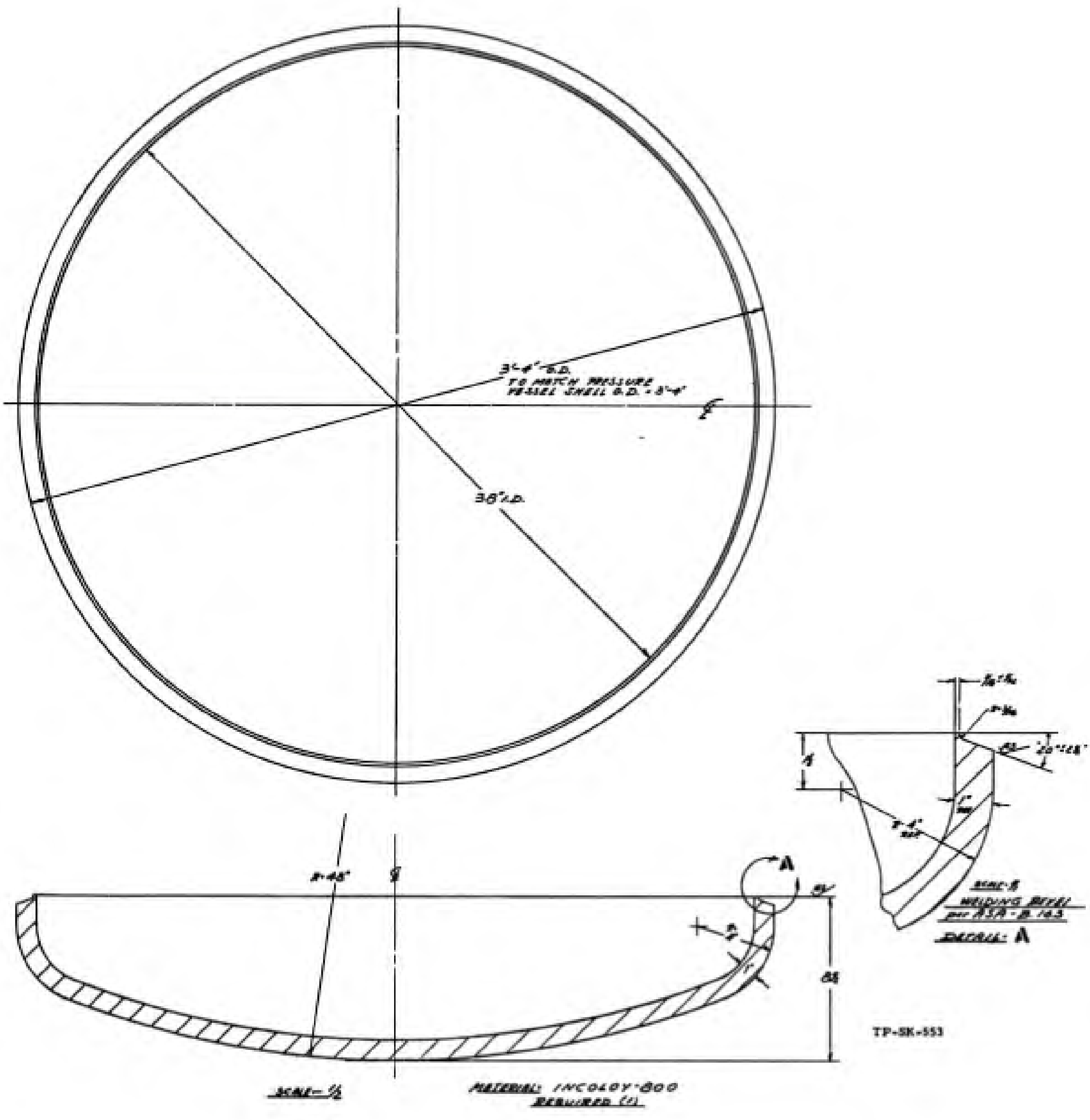

Fig. 3.10--Pressure vessel, lower head

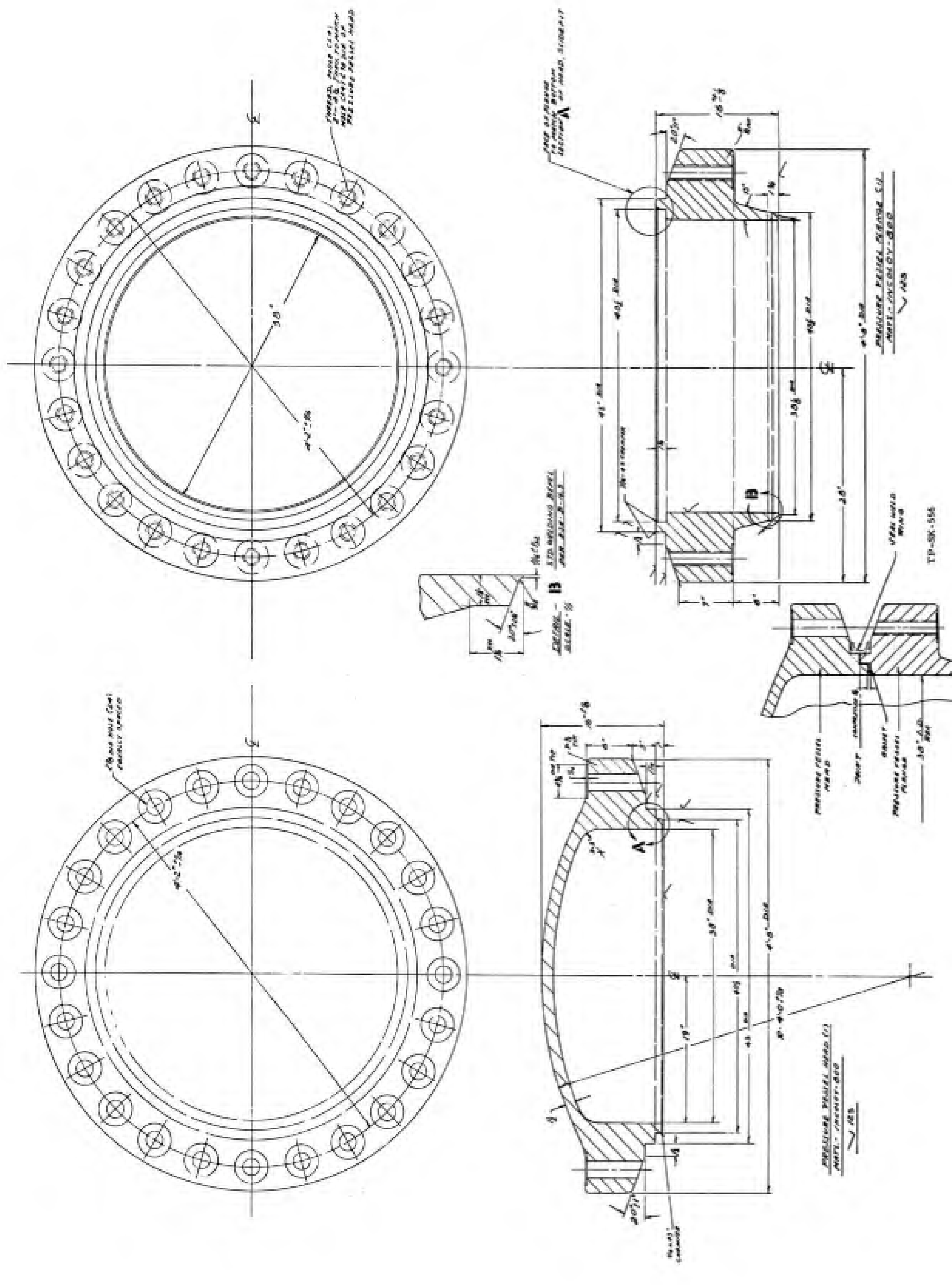

Fig. 3.11--Pressure vessel, head and flange

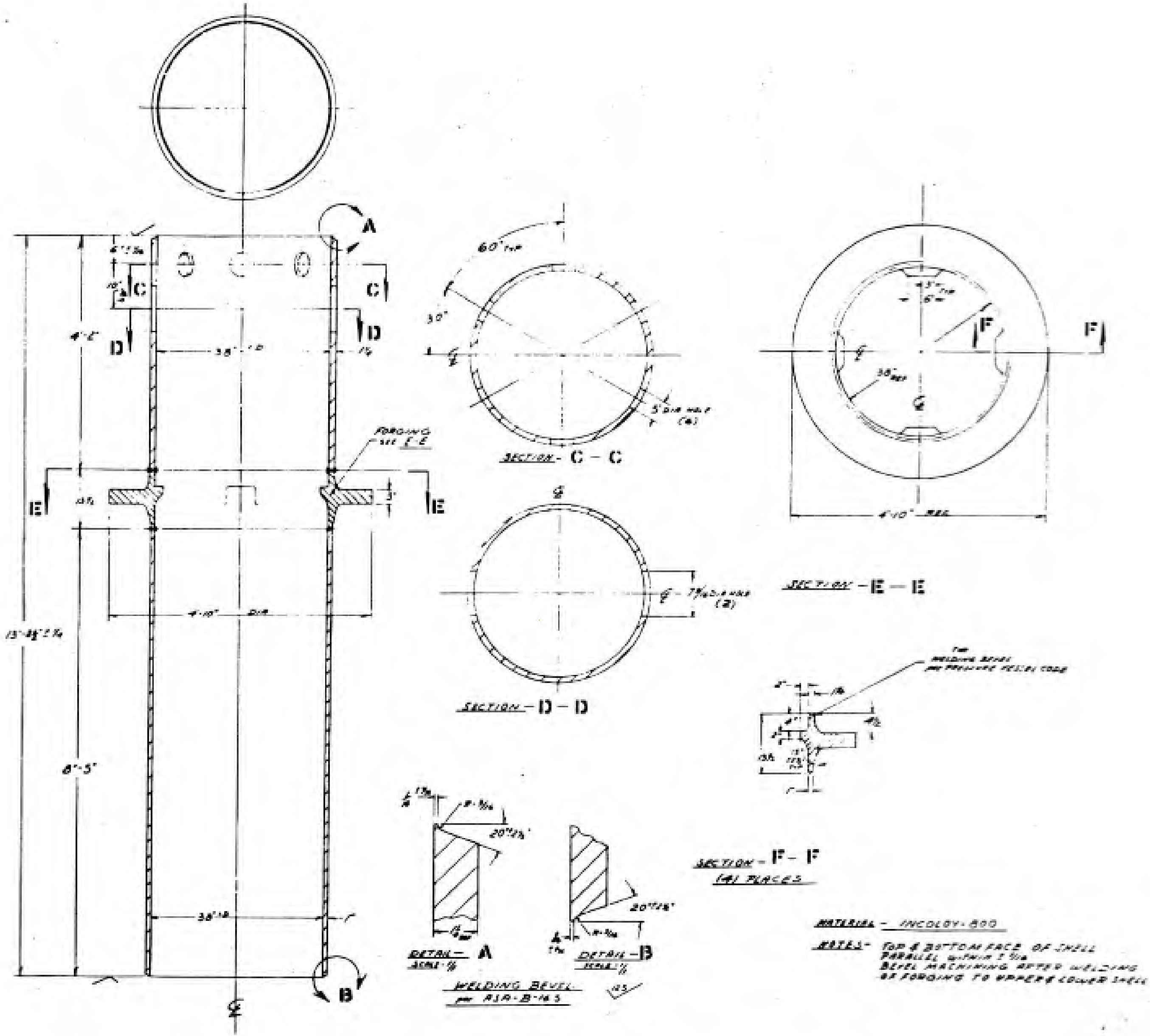

Fig. 3.12--Pressure vessel, shell

The lower core grid plate is attached to four pads at the lower end of the core barrel. The pads, which are welded to the core barrel, extend outside the core barrel at the lower end and act as spacers to minimize lateral movement of the core barrel at the lower end while allowing unrestrained vertical relative motion of the components. In order to allow assembly of the core barrel, four matching vertical grooves are machined inside the main vessel support forging.

The grid plate is 1-in.-thick. It contains 193 1-1/4-in.-diameter holes to locate the lower end of the fuel elements and allow flow of coolant through the core. In addition, there are 6 threaded control rod thimble holes. All 199 holes are located on 1.980 theoretical centers on a triangular pitch.

The upper grid and safety hold-down plates are attached to 6 support brackets located approximately 26 in. above the lower spacer pads. These brackets are also welded to the core barrel.

The upper grid plate is 2-in.-thick and acts as a guide for the upper end of the fuel elements and the control rod guide thimble, and provides some core shielding.

A safety hold-down plate is located 5/8 in. above the upper grid plate. The safety plate is 1-in.-thick and is designed to prevent fuel elements from coming out of the core in the event that the reactor becomes inverted for any reason. During fueling operations, the passages in the upper and lower grid plates and the safety plate are aligned. After fueling is completed, the safety plate will be shifted 1/4 in. out of alignment. This of course, does not interfere with control rod connecting rod motions. In the event that a fuel element becomes displaced, the misalignment of the safety plate will cause an interference to occur between the hole in the safety plate and the tri-flutes on the upper end of the fuel element. This interference will prevent removal of the fuel elements from the core.

The tubes for the integral once-through steam generator are secured to the steam generator shroud as described in Section 3.8. The construction described therein is used to suppress any induced vibrations which might occur in the generator tubes and also to allow the generator to be handled as a single unit.

The generator shroud is attached by a bolted flange to the upper end of the core barrel. The flanges will mate with sufficient precision to eliminate the need for a gasket between the surfaces. The steam generator tubes are welded and back-brazed into the appropriate pressure vessel nozzles. Further detailed discussion of the steam generator design can be found in Section 3.8.

The internal components are assembled into the reactor vessel in the following order:

1. The grid and safety plates and control rod guide thimbles are assembled into the core barrel.
2. The core barrel is lowered into the reactor vessel and bolted in place on the machined inside surface of the vessel support forging.
3. The preassembled once-through steam generator and shroud are placed on the upper flange of the core support barrel and the flanges are bolted in place.
4. The steam generator tubes are welded and back-brazed into the headers.
5. The control rods are placed in the core structure.
6. The control rod drives are assembled and the external connections seal-welded.
7. The reactor vessel and contents are cleaned.
8. The reactor vessel is filled with water.
9. The fuel elements are loaded into the core structure as a "zero power" critical experiment and criticality shimmed as needed.
10. The safety plate is shifted into the lock position.
11. Reactor water is removed.
12. The reactor head is assembled and seal-welded.
13. The necessary connections to the vent and relief system are completed.
14. Demineralized and degassed water is metered into the reactor vessel and an overpressure of hydrogen is provided.
15. Vent connections are closed and seal-welded.

Disassembly of the internals would follow in nearly the reverse order. To move the fuel elements one does not have to disturb the steam generator and shroud or the control rods and drives. The control rod drive extensions will have to be removed only if a fuel element transfer cask is used. This latter situation is discussed in Section 3.6.

3.3.3. Penetrations and Closures

An over-all view of the penetrations and closures for the reactor pressure vessel is shown in Fig. 3.8, the reactor layout. There are a total of 11 penetrations or closures.

The main closure is formed by the reactor head and pressure vessel flanges. The hydrostatic force which results from internal reactor pressure is resisted by 24 two-inch diameter studs. These studs will be fabricated from ASTMA-193 grade B7,which is a high-strength carbon steel material produced specifically for this purpose. In order to eliminate the torsional stresses in the studs, hydraulic stud tensioners will be used to pre-tension the studs during head closure operation.

Two types of seals are provided for in the reactor closure design. Under normal operating conditions, the seal will be made by seal-weld joints between the two flanges. Provision is also made for a Flexitalic type of gasket near the inside of the shell. The Flexitalic gasket will be used only for the initial fuel loading so that the system can be brought up to temperature to check out the nuclear characteristics of the core. During this preliminary check-out period, it may be necessary to open the head closure and move fuel to obtain the desired operating characteristics.

After the nuclear characteristics have been verified, the reactor head can be seal-welded in place. The seal-weld will be checked thoroughly with a helium leak detector to ensure the soundness of the seal prior to putting the reactor into normal service. In normal service, the reactor head should have to be removed only at refueling time. The actual length of time will depend on the utilization of the reactor system and should be set by the fuel element burnup criteria and would not be less than three years.

Six additional vessel penetrations are equally spaced around the reactor shell immediately below the main closure flange. These penetrations are for the control rod drives. The control rod drives are hermetically sealed with a final seal-weld made approximately 8 in. outside the reactor shell. The details of this seal-weld joint can best be seen in Fig. 3.13, the preliminary layout for the control rod drive system. The only time the control rod seal will be breached after installation will be to correct a possible mechanical malfunction of the drive system. Since the type of drive selected has already been tested in excess of the expected life of the system, and under more severe conditions, it is felt that malfunctioning will be highly unlikely.

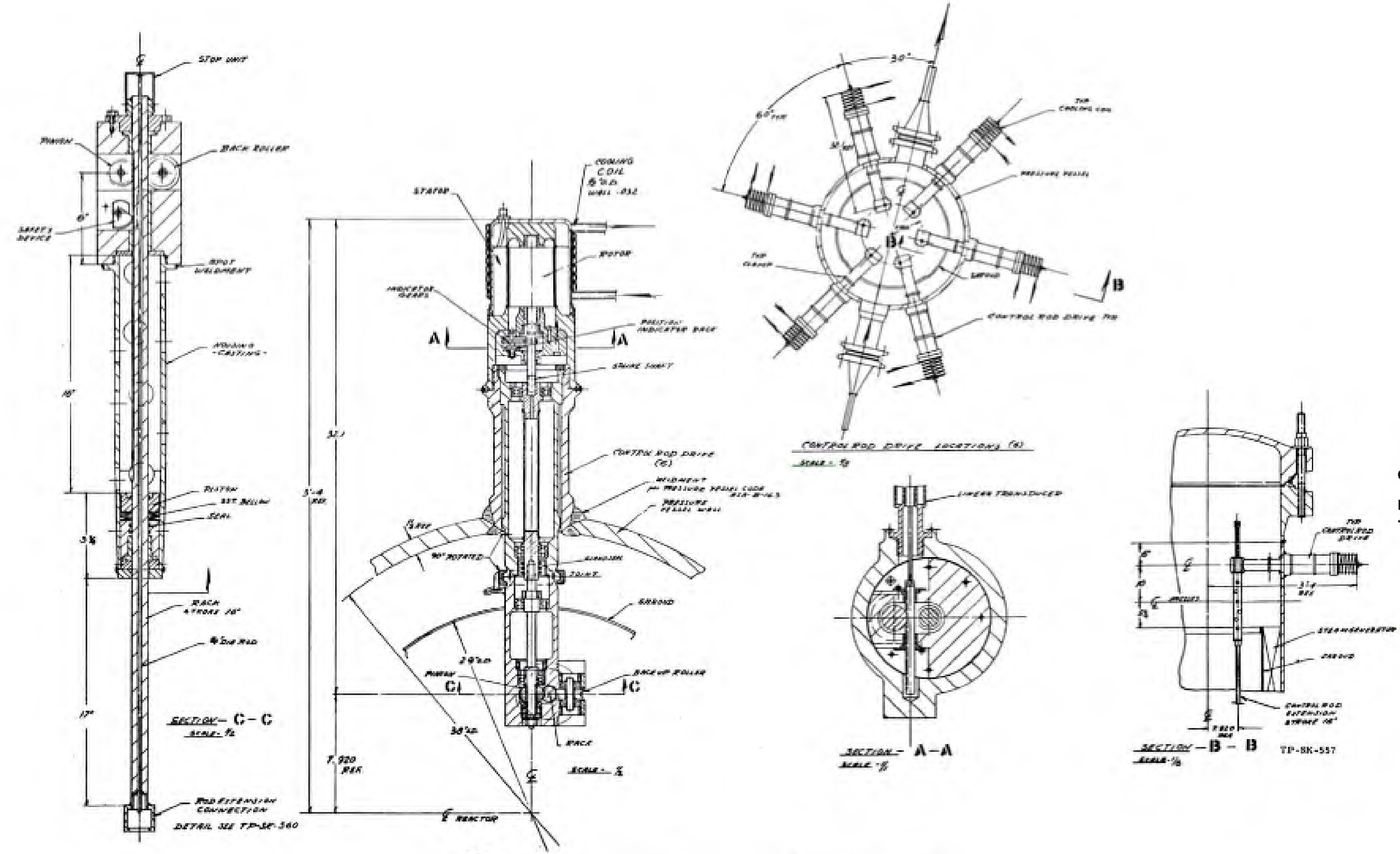

Fig. 3.13--Control rod drive layout

Two more vessel penetrations are located below the control rod drives. These are for the integral once-through steam generator inlet and outlet. All connections in these penetrations are either welded or welded and back-brazed.

The flange shown at the extremity of this penetration seals against leakage from the secondary system and is not in contact with the primary fluid. Seal-weld inserts are used here to allow for differential thermal expansions between the flanges.

The two final penetrations are for the fill and vent systems. These are standard welded nozzle penetrations. The fill nozzle is a 1/2 in. pipe connection located in the main pressure vessel shell below the plane of the steam generator nozzles. The vent nozzle is sized for 1 in. pipe and is located in the center of the reactor head. The pipes connected to these nozzles will be butt-welded and subjected to normal weld inspection. The function of these nozzles is discussed in Section 3.7.2.

3.4. CONTROL RODS AND DRIVES

3.4.1. Control Rods

The control rods are similar in construction to the fuel elements. Their cylindrical shape provides them with a high degree of structural stability and freedom from the warping and bowing problems to which plate or cruciform type controls are often subject. The nuclear poison material is boron carbide sintered into cylinders 1.25 in. in diameter and 5 in. in length. Three such cylinders comprising a total length of about 15 in. are contained in an Incoloy-800 seamless tube 1.300 in. in outside diameter with a wall thickness of 0.040 in. A diametral gap of 0.010 in. provides clearance for assembly.

The end fittings are also of Incoloy-800 and are welded to the clad material in the same manner as for the fuel elements. The tube sealed in this manner has a yield strength (0.2% offset) sufficient to contain an internal pressure 2,000 psi above the ambient pressure of the primary coolant. The control rod contains a 15-in.-long void section which acts as a reservoir for containing the helium produced by the boron n,α reaction. The wall thickness of the clad is adequate to provide for stability against buckling for an external pressure difference up to approximately 1,600 psi. The top end fitting is shaped to engage a ball-detent type of remote attachment on the control drive extension to permit remote disengagement of the rod from the drive.

3.4.2. Control Rod Drives

The TOPS/MUS reactor requires six control rod drives. The preferred location for these drives is shown in Fig. 3.8. The control rod drives penetrate the reactor pressure shell immediately below the main reactor closure flange.

The motive power for the control rod drives is derived from a canned-rotor stepping motor located at the extreme end of the drive extension. The details of the preliminary design are shown in Figs. 3.13 and 3.14.

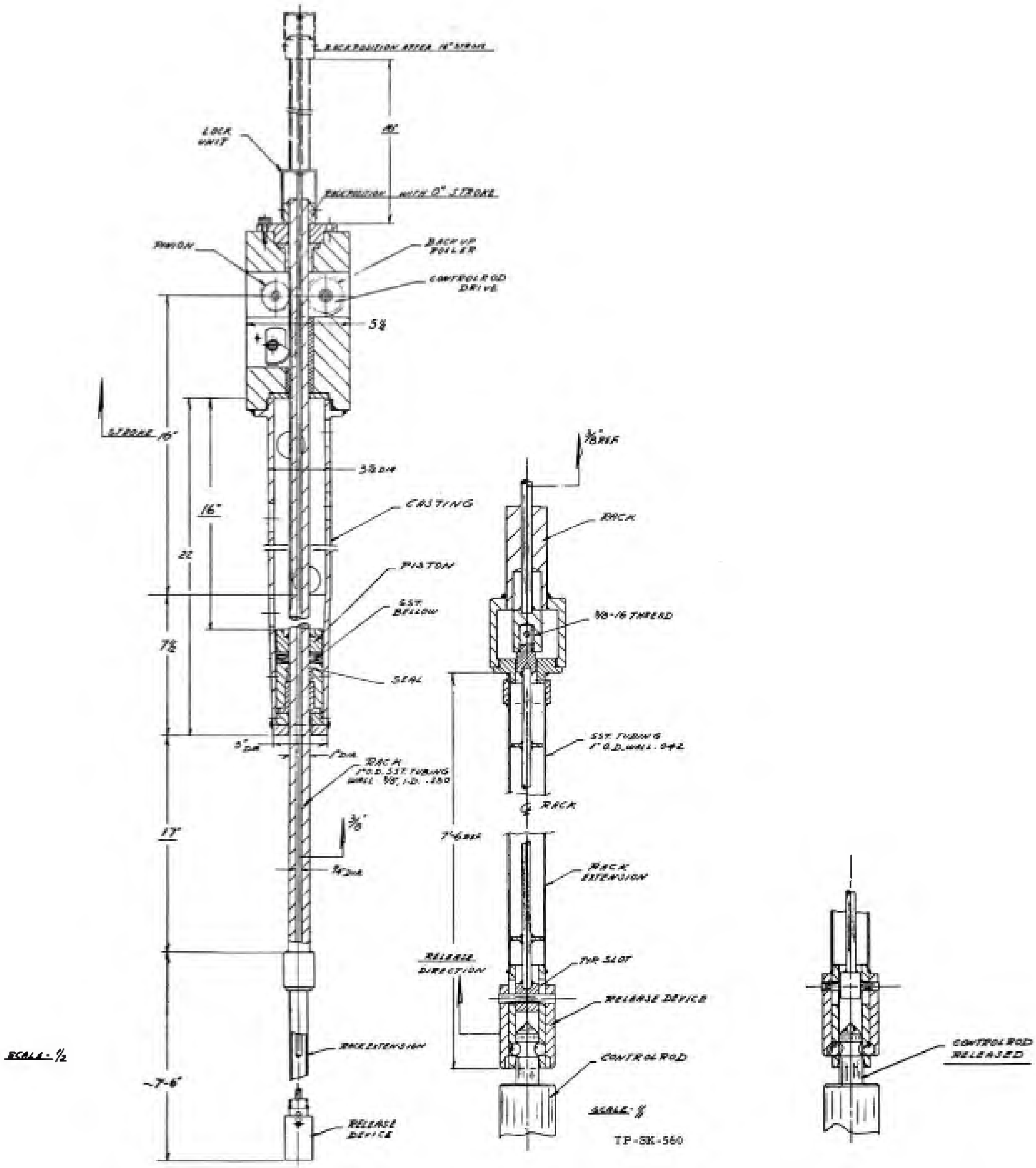

Fig. 3.14--Control rod release device

Torque from the stepping motor is transmitted through a torque tube to a pinion gear located inside the reactor vessel. A rack and pinion gear set translate the rotary motion into the linear motion required for positioning the control rod.

Position indication is derived from a linear variable differential transformer (LVDT) which is actuated by an auxiliary set of gears located between the drive motor and the torque tube. The LVDT for the reference design is manufactured by Kavlico Electronics as a custom unit. Gulf General Atomic has had excellent experience with similar units operating submerged in NASA's Plumbrook Reactor.

The stepping motor and rack and pinion drive are very similar to a unit already fabricated and tested by the Marvel Schebler Division of Borg-Warner for over 100,000 ft of rod travel under more severe conditions of load, pressure, and temperature than would be encountered in the TOPS/MUS reactor (see Fig. 3.15). The basic differences between the unit tested and the system which would be installed in the TOPS reactor have to do with configuration changes made to comply with the system's physical constraints and the inclusion of the previously-mentioned LVDT. These changes are considered to be engineering changes only and are non-developmental in nature.

An energy absorption system is included in the control rod design. It takes the form of a piston which is attached to the end of the rack gear that moves inside of a tapered cylinder. Little resistance to rack movement is allowed until the last 3-in. of the control rod stroke. At this point the exhaust area for water trapped under the piston becomes progressively restricted, thus retarding the motion of a rapidly falling rod.

In addition to the hydraulic energy absorber (dash-pot) just described, there is also a backup system. Located at the bottom of the retarder is a nest of Bellville spring washers; these spring washers are capable of absorbing all of the energy in the system over a short distance, should the primary system fail to operate properly. Such a condition could conceivably occur if control rod drive power is interrupted during the initial startup while there is little or no water in the hydraulic energy absorber system.

Under normal conditions the control rods are withdrawn one step at a time in sequence. Insertion, on the other hand, is done stepwise concurrently. This gives the effect of inserting the rods six times faster than they can be removed from the core. The details of the electrical system are discussed in Section 3.10.

The presence of an electrical signal will hold the rod in a fixed position, move the rod out, or move the rod in. Loss of an electrical signal will

Fig. 3.15--Phantom view of Marvel-Schebler control rod drive

release the drive magnetically and allow the rod to move into the core under the force of its own weight.

Several methods have been studied for preventing the control rod from falling out of the core in case of an accidental inversion of the reactor. The one shown in the preliminary sketch involves the use of a weighted pinion segment which prevents outward movement of the control should the reactor deviate from its normally vertical position by over 15°. There are several other approaches which have been used for the same purpose including the addition of a spring to overcome the weight of the control rod during an inversion, or the incorporation of a spring-loaded pawl mechanism within the stepping motor which would be held in the disengaged position by a separate motor coil. When power to the motor is interrupted, the pawl would engage the motor shaft in a way so that inversion would not allow the control rod to be withdrawn.

With either the gravity latch or the spring-loaded pawl, electric power would be supplied to the control rod drives to insert all rods simultaneously. This would be effective since the battery system should be able to function under all conditions even in an inverted position. A system similar to this is currently employed on the Marvel Schebler-built control rod drives for the N. S. SAVANNAH.

A decision as to which method is finally selected will be made prior to the submission of the "Safety Analysis Report."

Installation of the control rod drive system is accomplished in three steps. First the torque tube section of the drive is inserted in the control rod penetration and secured in place by a holddown ring which is screwed into the drive extension.

The rack and pinion portion of the drive is put into position from the inside of the reactor vessel. A female spline on the torque tube is mated with a male spline on the extension of the pinion gear. A pair of captive couplings fixed to the rack and pinion housing are moved into place and bolted to form a rigid connection between the rack and torque tube housing. Torsional moments are resisted by locating pins in the joint face. The connection between the control rod to the control rod extension is made by means of a ball latch arrangement which is shown in Fig. 3.14.

The rod, which is used to activate the ball latch, extends up through the gear rack so that the connection can be made locally near the top of the reactor. After the connection is made, a stop nut is placed over the latch extension and locked in place.

Assembly of the control rod drive is completed by positioning the drive unit at the extreme end of the control rod drive penetration. The drive unit, which also includes the LVDT for position indication, is assembled by mating the male spline at the drive unit with the female spline at the end of the torque tube. The final connection is accomplished by means of a "breach-lock" type of connection between the control rod drive penetration and the drive unit housing. After assembly of the drive unit, a final seal weld is made at the joint between the two units.

During refueling operations it may become necessary to use the large refueling cask. In order to accommodate the cask inside of the reactor, it is necessary to remove that portion of the control rod drive that carries the rack and pinion. This can be accomplished by first disconnecting the control rod from the extension at the ball latch and then removing the rack and pinion carrier by disconnecting the clamp which connects it to the torque tube extension. Since the clamp and its fasteners are all captive, a minimum number of pieces will have to be handled during the operation.

There will be two types of bearings used in the control rod drive system. The first is the ball-bearings which are used in areas where high-load carrying capacities are required. Stellite will be used exclusively for these bearings. In areas where the loads are low, graphitar will be used as a bearing material. For the most part, this will include those areas where a bearing surface is required only for a guide.

The lowest-stressed parts of the internal structure of the drive system will be constructed of 300-series stainless-steel. The higher-stressed members will use Incoloy-800. Where magnetic materials are required, such as the drive motor rotor, 400-series stainless steel will be employed.

The gear teeth within the unit, and possibly some of the splines, will use 17-4 pH-conditioned 1100. The heat treatment is specified here because for other heat treatments (condition 900), the material becomes susceptible to chloride stress corrosion cracking. The conditioned 1100 material does not have this problem.

These materials are the same, with the exception of Incoloy-800, as those successfully used in the control rod drives which were built and tested by Marvel-Schebler for identical functions.

External cooling coils are provided for the control rod drive stator. This system is supplied with water from the fresh-water-cooling system. The stator is kept cool in order to extend the useful life of the unit, since the life expectancy of the insulating material of the stator windings is a function of time and temperature. The insulation used for the stator will be of the type covered in NEMA Standard MW-16, and is rated at 220°C, but can be operated for short times at higher temperatures. These insulations include PYRE ML, a polyamide, and AI 220, a polamide-imide.

The electrical connections to the drive unit will be made with standard multipin hermetic connectors. One such connector is located on the drive motor and the other is located at the end of the LVDT.

3.5 SHIELDING

The reactor shield provides radiation protection to the crew of the MUS habitat such that federal standards cited in 10 CFR 20 will not be exceeded during normal planned operations. Access to the NEPP hull for power plant maintenance will be restricted to periods during which the reactor is shut down. Preferentially oriented shielding (shadow shielding) will be employed to provide adequate shielding between the NEPP and the MUS habitat. Radiation levels in other directions around and inside the NEPP hull will be higher since such locations are unoccupied during plant operation.

3.5.1 Reactor Shield

The reactor shield includes an inner shield tank assembly which is built as a component of the reactor module, a shield assembly in the space between the reactor module and the NEPP hull, and additional shielding over the top of the reactor vessel. These features were illustrated in Fig. 2.2. Seawater and hull structures interposed between the habitat space and the reactor module provide additional shielding, and a steel mesh to prevent close approach is employed to protect divers during shallow-water operations.

The maximum design dose rate in the hull will be 0.4 mrem per hour at full power so that a crew member stationed continuously for 90 days at that point would receive no more than half of the radiation tolerance dose permitted by 10 CFR 20 for a "restricted area" as defined in Par. 2.0.3 (14). Consequently, maintenance operations can be conducted for limited periods of time in the NEPP hull after reactor shutdown without exceeding permissible radiation exposure limits of 1.25 rem per calendar quarter.

The inner surface of the inner shield tank is exposed to a high level of fast neutron flux; and to assure freedom from radiation-induced nil ductility temperature transition problems, stainless steel is used for its fabrication. A wall thickness of 0.5 in. provides strength for its function of supporting the gamma-ray shielding. The outer wall and top of the tank utilize carbon steel for which the secondary gamma-ray production is appreciably less than for alloy steels. Water in the inner shield tank is borated.

and an expansion tank above the shield tank provides for thermal expansion. To prevent loss of water from the shield tank in case of accidental inversion, a float trap is located in the vent line.

The minimum water level over the reactor core under cold temperature conditions is 7 ft. This provides sufficient shielding so that the reactor pressure vessel head could be removed safely 24 hours after shutdowns with a maximum dose rate of 1 mrem/hr from fission products in the core before lifting the vessel head and 80 mrem/hr after removing the head but before adding additional water for shielding. The water shield tank has a conical structural top surface which is attached to the reactor vessel support ring at the approximate height of the minimum core water level.

Lead gamma-ray shielding is supported from the inner tank wall and reinforced internally with studs welded at regularly spaced intervals to the tank wall. Cooling water channels located around the lead-steel interface provide cooling for the heat conducted from the reactor vessel insulation, as well as for the heat deposited by radiation energy absorbed near the interface. The thickness of the lead is 8 in. adjacent to the core mid-plane, 7 in. below the core and tapers to a thickness of 2 in. at the tank top.

Some gamma-ray shielding is placed inside the reactor vessel in order to reduce the total shield weight. This internal shielding includes: a 1-in.-thick cylindrical shield of 304 stainless-steel surrounding the reactor core; a 3-in.-thick plate of stainless steel below the core; the 2-in.-thick top grid plate; and the 1-in.-thick holddown plate.

The space between the inner shield tank, the NEPP hull, and the reactor module support cone contains water to which potassium dichromate has been added as a corrosion inhibitor. To lighten the water shield weight, three quadrants not facing the habitat contain large void tanks. These features are illustrated in Figs. 3.16 and 3.17. The tanks are sized and located so as to permit a free flow of water for transport of heat by natural convection from the inner shield tank surface to the NEPP hull surface. This arrangement provides for neutron shadow shielding in the direction of the habitat in addition to the gamma shielding provided by the hull. Gamma shadow shielding of lead is also provided in that direction as is also shown in the figures.

Shielding of gamma-rays from 9 curies of N-16 in the upper part of the reactor vessel is provided by the reactor vessel, the two hull walls and frames, and by 4 in. of lead mounted on steel plates on the top and sides of the reactor vessel outside the thermal insulation.

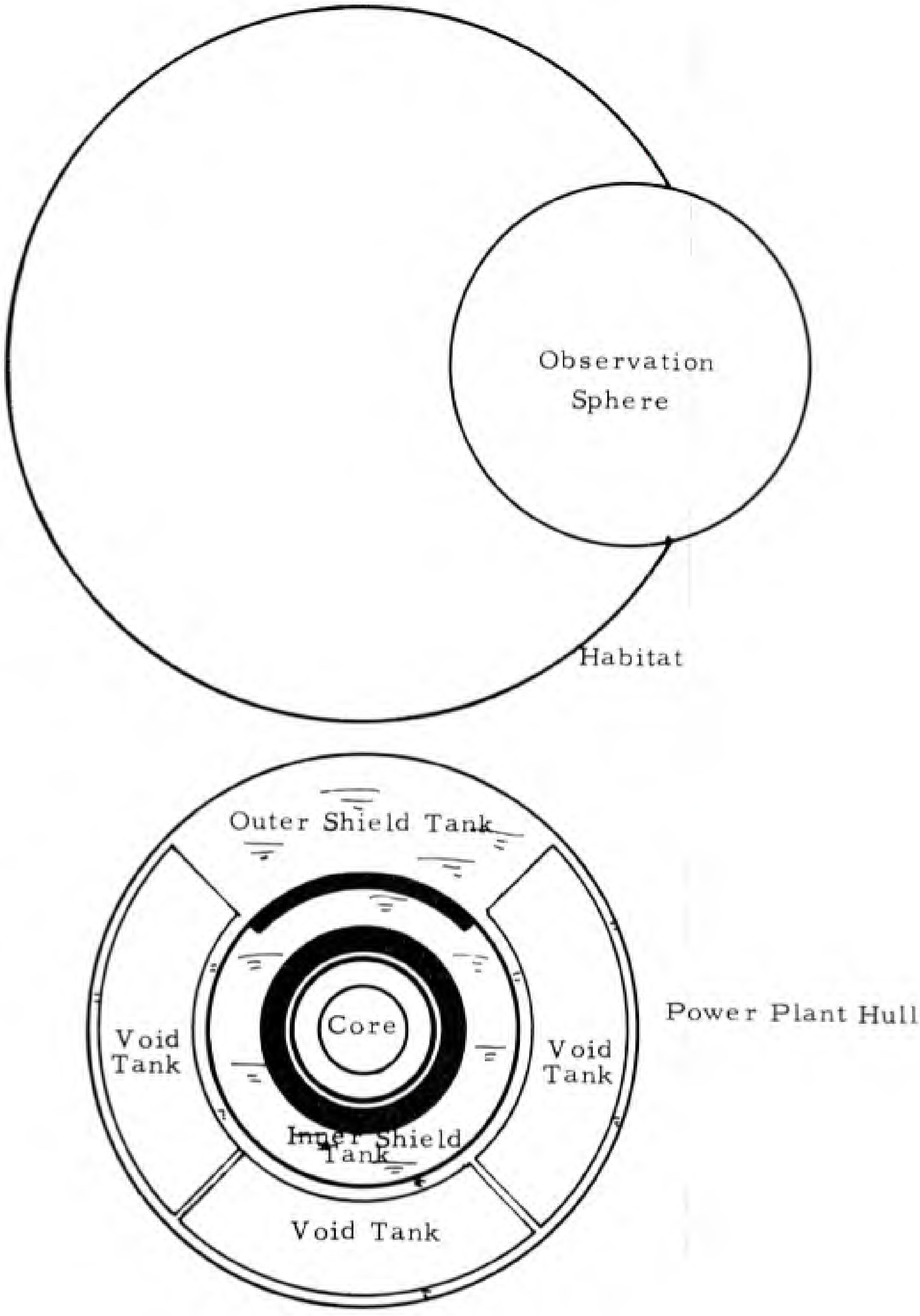

Fig. 3.16--Shielding arrangement at reactor mid-plane

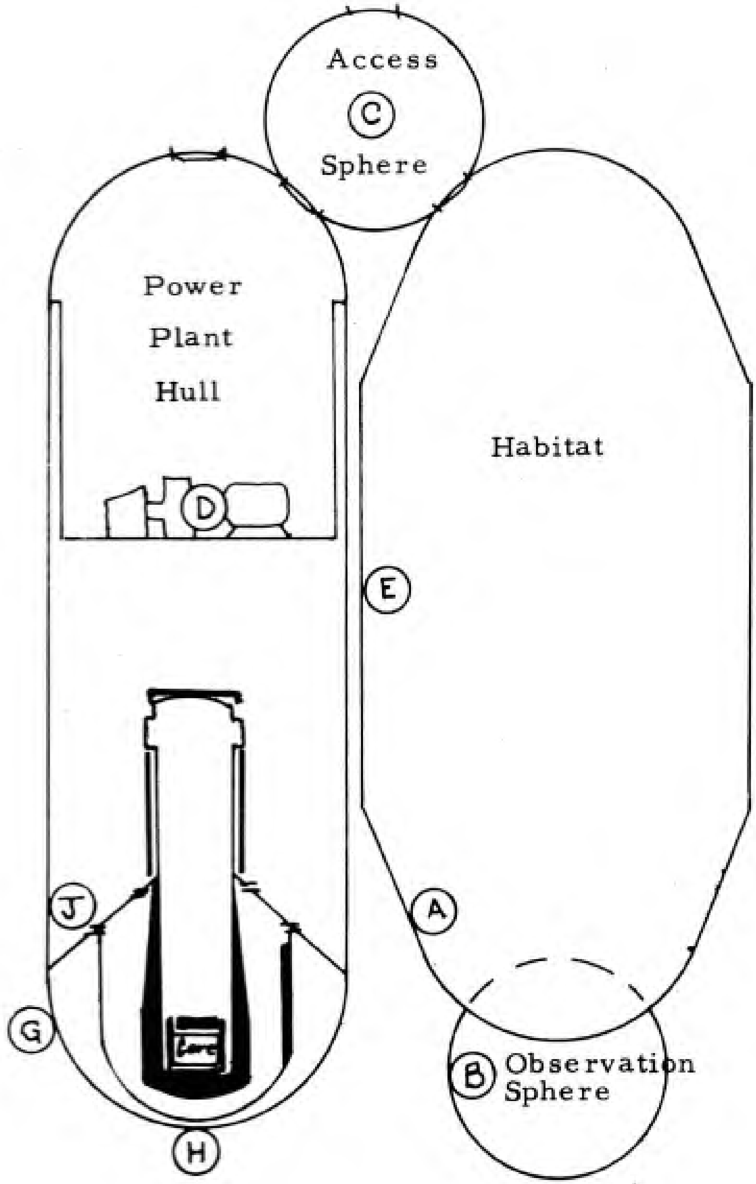

Fig. 3.17--Shielding arrangement, vertical section

Shown in Fig. 3.17 are locations in the MUS at which radiation dose rates are of interest. Radiation levels at these locations are given in Table 3-4. The dose rate is the highest at location 1, the habitat location closest to the reactor. The shielding materials and dimensions in that minimum radiation path are listed in Table 3-5 as an example case of interest.

3.5.2. Maintenance Dose Rates

After reactor shutdown the principal dose rates will result from the core fission products, shield material activation, and corrosion products transported from the reactor core to the boiler surfaces. Calculations show that the dose rate at one hour after shutdown at a point just above the nuclear module due to fission products will be 5 mrem/hr due to core contributions. Dose rates in other areas will be somewhat less.

During refueling operations, water can be added from the normal cold water level to a total height of 12 ft over the core so that fission product decay contributes less than 1 mrem/hr after a cooling-off period of 30 days.

3.5.3. Shield Cooling

The lower hemihead of the NEPP hull provides a large cooling surface (250 square ft) exposed to the cool ocean environment, and is utilized for disposal of several heat loads including thermal heat losses from the reactor vessel to the shield tank, radiation heating of the shield, and heat removed from the various power plant components by the fresh-water-cooling circuit. Except for the pump-operated fresh-water-cooling circuit, the heat transfer through the inner tank, across the outer tank, and from the hull into the sea is achieved by natural circulation processes so that heat transfer does not depend on motor-operated devices. These heat loads are shown in Table 3-6 to total 96,600 Btu/hr at the design point so that the heat flux into the sea is only ~400 Btu/hr square ft. Consequently, the temperature of the outer shield tank will be less than 85°F for a sea temperature of 45°F, typical at the design depth of 6,000 ft.

Table 3-4

RADIATION LEVELS AT SELECTED LOCATIONS

Location No.	Location	Dose rate at rated power
A	Habitat—closest to reactor	0.4 mrem/hr
B	Observation sphere	0.4 mrem/hr
D	Power conversion module	2 rem/hr
E	Habitat—midpoint of hull	0.4 mrem/hr
G	Power plant hull—reactor level	10 rem/hr
H	Power plant hull—bottom	10 rem/hr
J	Power plant hull—feed pumps	20 rem/hr

Table 3-5

TYPICAL SHIELD CONFIGURATION

(At reactor mid-plane towards habitat)

Region	Material	Thickness (in.)	Outside radius (in.)
Core	U-ZrH-water	----	14.25
Core water	Water	1.5	15.75
Flow-divider	Stainless steel	0.25	16
Thermal shield	Stainless steel	1	17
Reflector	Water	2	19
Reactor vessel	Incoloy-800	1	20
Insulation	Magnesia (85%)	1	21
Inner tank wall No. 1	Stainless steel	0.5	21.5
Gamma shield	Lead	8	29.5
Neutron shield	Water	10	39.5
Gamma shadow shield	Lead	2	41.5
Inner tank wall No. 2	Steel	0.5	42
Neutron shield (at cylinder)	Water	31.22	73.22
NEPP hull (at cylinder)	Steel	1.75	75

Table 3-6

SHIELD TANK COOLING SYSTEM

Reactor thermal energy	33,100 Btu/hr
Reactor radiation energy	15,000 Btu/hr
Fresh-water-cooling	48,500 Btu/hr
Total cooling required	96,600 Btu/hr
Cooling available at 45°F sw for 85°F outer tank	250,000 Btu/hr

3.6. FUEL OPERATIONS

All fuel operations, except in emergencies, will be performed under dry dock or dockside conditions. No fueling operations will be conducted at sea. The approach used herein is to deviate as little as possible in procedures, equipment, and facilities from normal TRIGA experience.

The general aspects of fuel operations and handling may be described as follows. Fuel elements will be fabricated in existing Gulf General Atomic facilities for the TOPS application. The fuel elements will be shipped from Gulf General Atomic in shipping packages of the type and design already under license by the Department of Transportation (DOT) and which conform to DOT and Atomic Energy Commission (AEC) regulations. Upon arrival of the fuel elements at the TOPS-MUS loading site, the fuel elements will be stored in their shipping containers until loaded into the reactor. The fuel may then be loaded into the reactor using general procedures already available and equipment of designs currently in use, and the fuel will be mechanically secured with the holddown plate. The fuel elements will remain in the core during the water pretreatment and hermetic sealing of the reactor pressure vessel, and during subsequent operation of the reactor at the station location. At the end of the TOPS core life, fuel will be removed in shielded containers which serve the purpose of handling, storage, and shipping. These will be similar in their general aspects to the TRIGA shielded shipping casks already licensed by the Department of Transportation for interstate shipment excluding air transport. Because of the substantial weight of the shipping cask (5 tons) adequate crane facilities will be required when irradiated fuel is being transferred. Shipment of the fuel elements in their shielded containers from the unloading port to a designated reprocessing plant will complete the fuel handling operations. Return inventory credit will be given for recovered uranium.

3.6.1. Handling of Unirradiated Fuel

In the initial loading of the TOPS core, the hold-down safety plate will be shifted laterally 1/4-in. to fully open holes for insertion of fuel elements. Control rods will be inserted into their positions. Control rod extension arms and drives will be attached. The reactor vessel will then be filled with pre-treated, high-purity water. This will be followed by loading of the fuel elements in accordance with normal TRIGA practice. No special precautions with respect to radiation and contamination will be required since the elements will be clean and non-radioactive. A sectionalized TRIGA tool will be used to lower individual fuel elements into their positions in the grid plate. This operation is performed from the top of the reactor pressure vessel, with the head removed, through approximately 10 ft. of water. After the core is fully loaded, the safety plate will be displaced so as to release the control rods and latch the fuel elements. Safety plate bolts will then be tightened in place to preclude inadvertent translation. Another sectionalized tubular rod with mating tool heads will be employed for this operation. This will complete the initial fueling operation. Other preparatory procedures will then follow, such as replacing the reactor-pressure vessel head, and seal-welding the head and fill connector closures.

It is possible to handle the irradiated fuel in units smaller than 19 elements at a time and thereby reduce cooling time. However, a correspondingly greater number of shipping and storage containers would be required to accommodate each core. Conversely, it would be possible by waiting for longer periods of time to handle more elements per shipping container. However, an upper limit is placed on the number of elements per container by criticality considerations. On this basis, the maximum number of elements per cask must be 30 or less. It should be noted that longer cooling times would be required since the rate of decrease in the heat generation and radiation source levels is the result of long-lived activity which is slowly decaying. It is also possible to maintain the same number of fuel elements per cask and use a cask designed with less shielding (smaller weight and size) by lengthening the cooling period to four months or more. There is obviously an economic trade-off between the design and cost of shipping containers with cooling times of the reactors during which the TOPS/MUS system is unavailable for other service. Since new shipping and handling casks must be qualified for license, the economic trade-off should be part of a study undertaken at the time of designing the new casks.

3.6.2. Unloading Fuel and Refueling Operations

Refueling the TOPS/MUS core and refueling operations will be accomplished after returning to port. A post-operation cooling period of approximately 1 month will be required to permit high-level, short-lived radioactivity to decay. Dry-dock or dockside facility utilization is assumed. Since refueling will be done only once or twice over a plant life (i.e., approximately 10 to 15 years) economy and simplicity of operation rather than minimum time or ease of performance should be the criterion of evaluation of the operational procedures used. The operations for handling of spent reactor fuel elements are presented below. The shipping cask is the key piece of equipment for these operations. Its design and principal features are discussed for TOPS service.

Sequence of Operations for Refueling TOPS/MUS

1. Wait about one month after reactor shutdown to limit the radioactivity of the fuel to levels for which the shielded containers have been designed and licensed.

2. Clear and open the 30-in.-diameter hatch in the top of the TOPS hull and attach the bumper to the rim of the port to protect against mechanical damage, particularly during passage of the shielded containers.

3. Clear the 30-in.- diameter access channel through the electrical and machinery decks of TOPS to the reactor pressure vessel.

4. Attach the water-treatment plant contained in the dock-side facilities to the reactor pressure vessel at the block valves (See Fig. 3.18). This will require that the hermetic seal welds at the valve ports be ground away.

5. Vent and dilute with inert gas to avoid explosive concentrations of the radiolytically-generated gases collected in the top of the vessel. These gases are vented through the shield tank water to remove entrained contamination.

6. Circulate the primary system water through the external water-treatment plant to remove crud and dissolved radioactivity in the filters and demineralizers and thereby reduce residual activity.

7. Pump water into the vessel and control its level to just below the top head flanges mating surfaces. Initially the pressure vessel will be only 75% filled in the cold condition corresponding to 85% full at the operating conditions. This will raise the height of water above the core from 7.5 to 12 ft, and will provide maximum shielding for subsequent operations.

8. Disconnect the vent lines from the reactor vessel to the shield tank and water treatment plant and remove the pressure vessel head.

9. Remove securing bolts and rotate the safety plate in the reactor core to latch the control rods and unlatch the fuel elements.

10. Detach and remove the control rod extension rods and drive assemblies.

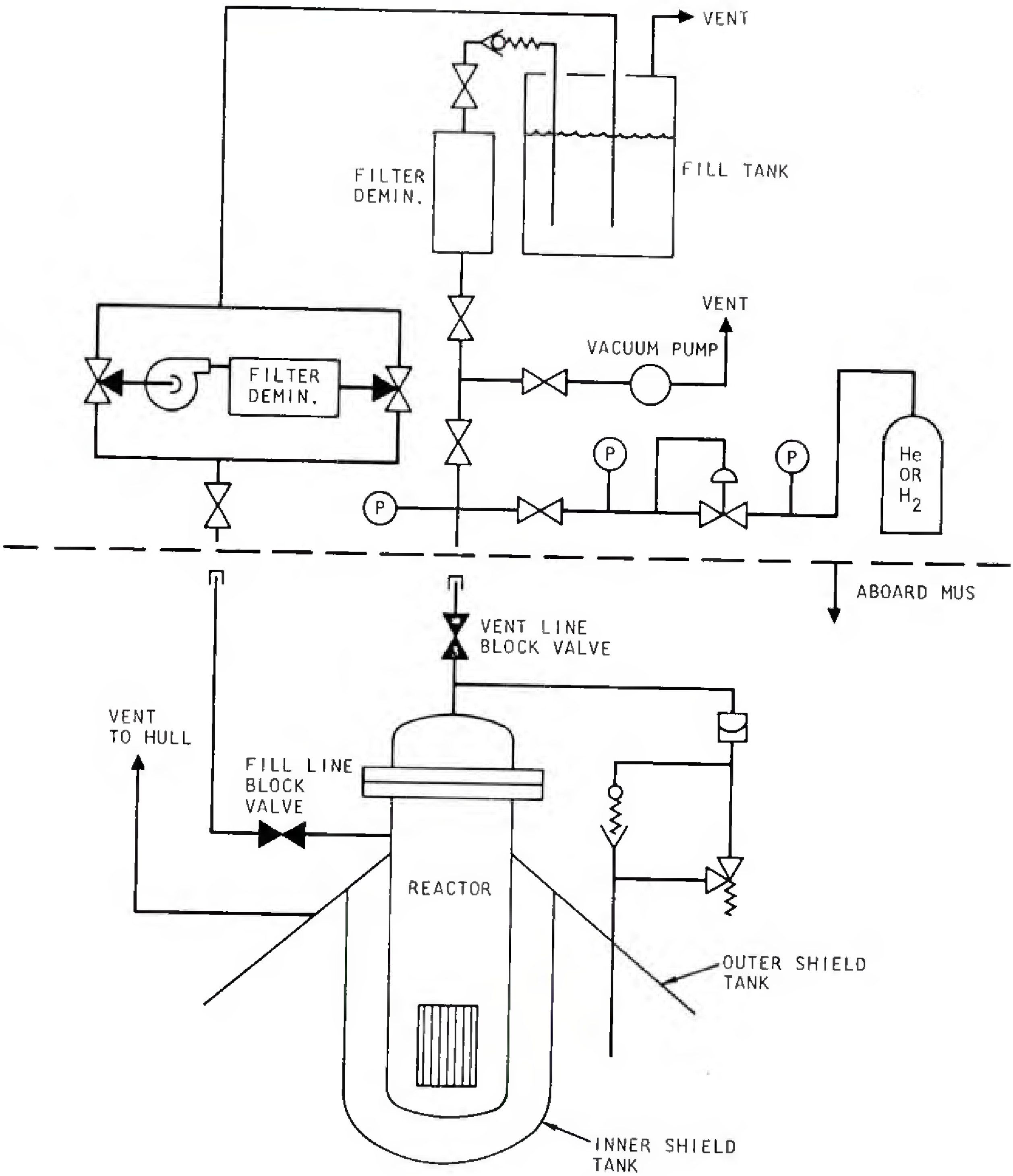

Fig. 3.18--Primary coolant fill and relief system

11. Move fuel elements, one at a time, from the core to side racks on the pressure vessel wall just above the core. This is an essential step to obviate the potential accident of a shielded container falling on a loaded core and compacting it.

12. Insert a submersible sump-type pump into the pressure vessel and connect it to the water treatment plant at the point where the line from the pressure vessel head was previously attached (see Fig. 3.18). This pump will remove water displaced by immersion of the shielded container at a rate controlled by the displacement rate during immersion so as to maintain the water at a constant level. The shielded container will displace approximately 110 gallons which must be pumped over a 20 ft maximum head. Thus a fractional horsepower pump will be adequate.

13. Load the shielded container with fresh fuel elements to be transferred to the core. While shielding is not required since the elements are not radioactive, it is convenient to perform these tasks together rather than separately.

14. Lower the shielded container into the reactor vessel. Because of limited access, visibility, etc., local control of the container by personnel at the reactor vessel is required. This will be provided by holding the container by a hoist which is held in turn by a crane hook. The crane will deliver the hoist and container to the proximity of the reactor vessel where further manipulation of the container will be done with the locally-controlled hoist. Weight of the container will be approximately 5 tons. The immersion rate of the container must be limited to prevent splashing or spilling of the water from the vessel. The rate of immersion must not exceed the capability of the pump discharging the displaced water. The plug in the bottom of the container will be opened prior to immersion to allow the container to fill during immersion. The container will be lowered to a point just above the reactor core so as to provide a maximum depth of water between the fuel and the operators who will be stationed at the top of the open vessel.

15. Fresh fuel elements will be removed from the container, one at a time, using the same tools as described in the unirradiated fuel handling operations above, and placed on the rack on the side of the vessel wall and will be replaced by spent fuel taken from the rack until the replacement is completed. The lid is placed on the top of the container using a small separate hoist.

16. The shielded container will be removed from the TOPS, allowing the water to drain from within and controlling the water level with the pumps for maximum water shielding. The lid to the container will be securely fastened. The container will be decontaminated in a wash-down tank.

17. Steps 13 through 16 will be repeated using additional containers until all spent fuel elements are removed and replaced.

18. Fresh fuel elements are moved, one at a time, using the same tools as described in the unirradiated fuel handling operations above, from the rack on the vessel wall into the core grid plates. Follow the procedures given above for unirradiated fuel handling until the refueling operations are completed.

As indicated above, the key piece of equipment in the safe and economical handling of the spent reactor fuel and its storage and shipment is the shielded container. Four shielded containers for shipment of spent TRIGA fuel were designed and built in 1958 and have been used with complete success since then. However, they were qualified under regulations which are now obsolete and their continued use is permitted under the "grandfather" clause of the newer regulations. Currently these containers are operating under DOT Special Permit 5266 and AEC license SNM-69 and are limited to "760 grams of U-235 and 1200 curies of fission and irradiated products in the form of 19 TRIGA fuel elements." Indications are that the current regulations will be revised in the near future.[(1)*] The license limits are arbitrary and unrelated to the capability of the containers. Thus it will be necessary to design and qualify shielded containers for the TOPS system.

There are three principal constraints which determine the design of the cask: (1) the neutronic constraint of criticality; (2) adequate and inherent heat removal capability, and (3) acceptable external levels of the radiation field produced. The existing TRIGA shielded container has been considered with respect to these three constraints to determine the general features of a container suitable for the TOPS system. While detailed design will be required to satisfy the regulations and qualifications for DOT licensing, the general features of a container suitable for TOPS have been obtained.

*References are listed at the end of this section.

The TRIGA shielded shipping cask or shielded container is briefly described in Reference 2 as follows:

A container has been designed to carry nineteen irradiated TRIGA fuel elements.

Each cask is ~22-1/2-in. in diameter by 4 ft high and weighs about 7600 lbs. The wall thickness is 7-1/4 in., composed of 6-3/4-in. of lead encased by 1/4-in.-thick steel shells. The interior cavity of the cask is lined with 1/8-in.-thick aluminum and contains two aluminum grid plates to accommodate the 19 fuel elements.

The entire cask is regarded as the containment vessel.

No materials are specifically used as nonfissile neutron absorbers or moderators.

Only the two 19-hole aluminum grid plates might be regarded as "protecting" the interior cavity. Primarily, however, they are present simply to hold the fuel elements in position relative to one another, rather than to protect or support the receptacle.

A drain in the bottom of the cask permits draining the water which remains inside after the cask is loaded while submerged for shielding. The drain is present as a precaution, however, and is not expected to be a routine feature of the procedures since the casks are always shipped dry.

A box of 3/4 in. thick steel covers the cask lid assembly to protect against possible accidental damage to the locking assembly for the cask lid during shipment. The screws which secure this steel box to the cask are capable of resisting a shearing force equal to 30 times the weight of the cask. It is believed that this feature effectively eliminates the possibility of inadvertent opening of the cask lid during shipment.

Two metal eyes are positioned on the opposite sides of the cask and are used to hold the cask to the bed of the carrier. One-half-inch steel chains are fastened to each eye and then are securely fastened to the bed of the carrier. A total of four such chains are used (two to each metal eye), so that the cask is secured in all four directions.

There are no special structural or mechanical means for transfer and dissipation of heat, because fuel is not placed in the cask until the fission product energy has decayed to a total of 330 watts for the 19 elements.

Maximum weight of contents is 200 lbs.

Consideration of the three major design requirements with respect to modification of the container described above to render it suitable for the TOPS-MUS application are described in the following paragraphs.

Accidental Criticality

Criticality measurements of stainless steel clad TRIGA fuel elements bearing the same U^{235} content as expected to be used in the TOPS fuel element indicated 40 elements as the minimum critical number when water-moderated and -reflected. Thus a fully-loaded shipping cask contains less than half of a critical mass. The possibility of interaction effects between the fuel elements located within this shipping cask and fissionable material contained on the outside is effectively negated by the thickness of the lead walls of the shipping cask. Therefore, there are no credible conditions of accidental criticality which could arise during the shipment of TRIGA fuel elements in these four shipping casks. The effect of Incoloy-800 replacement of the stainless steel is negligible.

Heat Removal

Heat generated after shutdown is shown in Fig. 3.19 as a function of time. After a one-month cooling period, the decay heat generated in 19 fuel elements previously operated at an average core power (193-element TOPS core) of 1 Mw(t) will be 197 watts. If all elements were as hot as the central element operating at 1.8 times the average, the 19-element heat power would be 355 w(t). Steady-state heat transfer assuming only conduction within the container (i.e., no air convection or radiation) indicates the hottest (central) element temperature will rise 650°F above ambient (i.e., 750°F). Addition of 1-in.-thickness of lead shielding to the wall adds less than 5°F since most of the resistance is in the gas layers. No credit for conduction from the fuel elements through the aluminum grid plates to the container walls was taken. Since 750°F is a very modest temperature for Incoloy-800 clad U-ZrH_x fuel elements, there is no problem with heat removal from the container.

Radiation Field

The radiation field in the vicinity of the container must be limited to permit operating personnel to work safely for reasonable periods and to protect the general public during shipment of spent fuel. AEC standards for laboratory workers permit 1-1/4 r in any calendar quarter or a dose rate of approximately 10 mr/hr for 2 hours each working day. Also

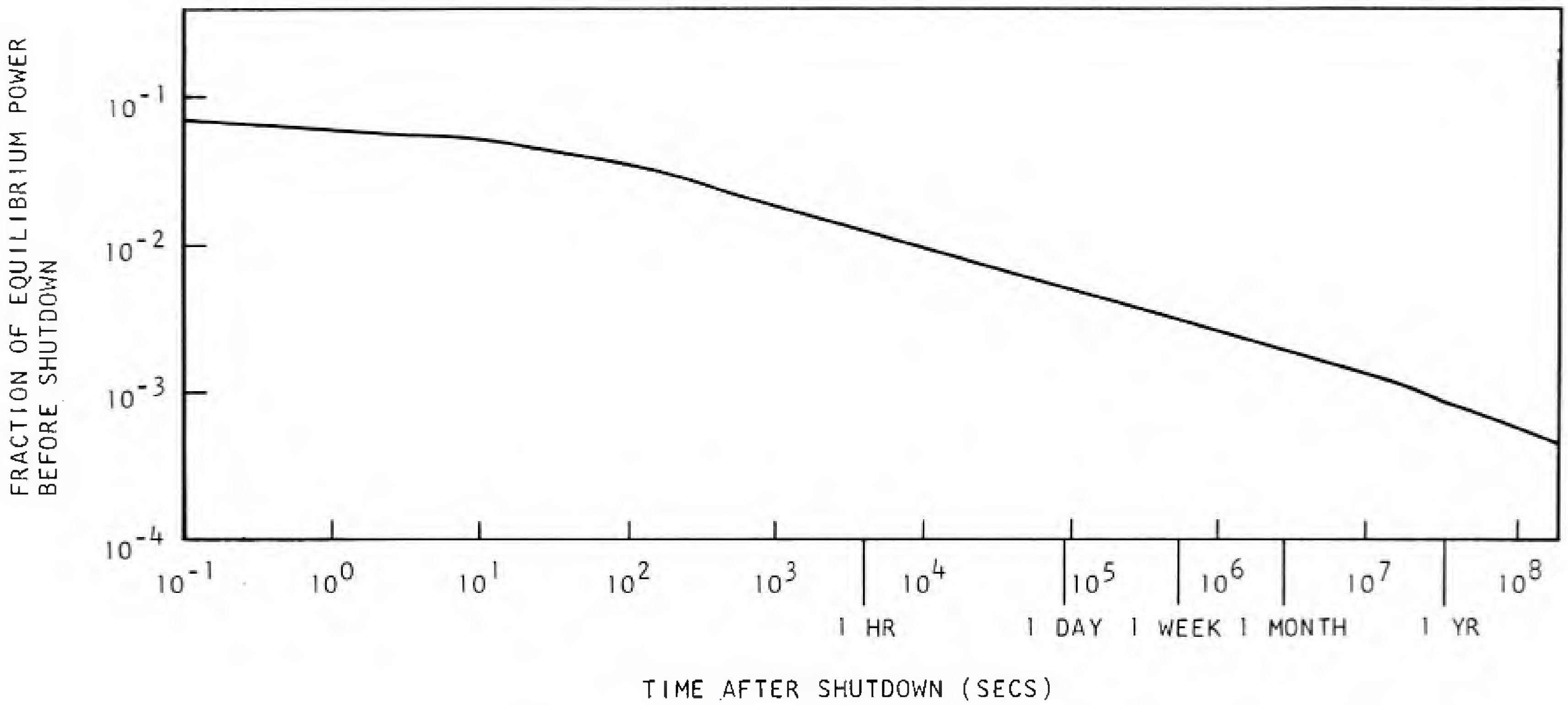

Fig. 3.19--Fission product power after shutdown

ICC regulations[3] for shipment of radioactive materials indicate a dose rate not to exceed 10 mr/hr at a point 6 ft from the surface of the container. This was adopted as the criterion for the container evaluation. The level of radiation depends on the duty cycle and power level of operation. It was assumed that the TOPS reactor was operating at 1 Mw(t) continuously for a very long time. The source strength of 19 fuel elements from the 193-element TOPS core was determined and shield attenuation and dose rate were computed as a function of cooling time to yield the resulting curve shown in Fig. 3.20. Similarly, the activity (in curies) as a function of time was computed using a computer program which keeps account of the buildup and decay of each individual nuclide during the cooling period. Results are shown by the curve of Fig. 3.21.

Times less than one hour after shutdown of the TOPS system prior to refueling are certainly not of practical interest. Considering the time required to surface the TOPS-MUS, tow it to port, prepare and arrange dry-dock facilities, and the performance of other tasks that must precede actual fuel handling, times after shutdown of less than a week are not considered to be practical and, in fact, one to three months appear to be reasonable.

Thus by the criterion indicated above (i.e., 10 mr/hr at 6 ft from the surface of the container) a shielded container of basic design similar to the existing TRIGA shipping casks will satisfy the three fundamental design requirements.

3.6.3 Radioactivity Contamination Control

Experience has indicated that during the operation of a nuclear power plant, there will be radioactive contamination arising from various sources. In the case of the TOPS/MUS system, these will occur from abnormal operation of the hermetically-sealed primary system or from unloading and refueling operations. The TOPS primary system is self-pressurized. Since pressure is a very strong function of temperature at the design operating condition, a pressure relief and safety system has been incorporated into the design. Abnormally high pressure in the primary system will cause a discharge of cover gas and steam, which may contain entrained radioactive materials, into the shield tank. Also, radioactivity from a steam generator leak into the condenser of the secondary system could be released through the air ejector into the TOPS hull. However, since two separate phase change steps are required (i.e., vaporization of primary water in the steam generator followed by condensation and re-

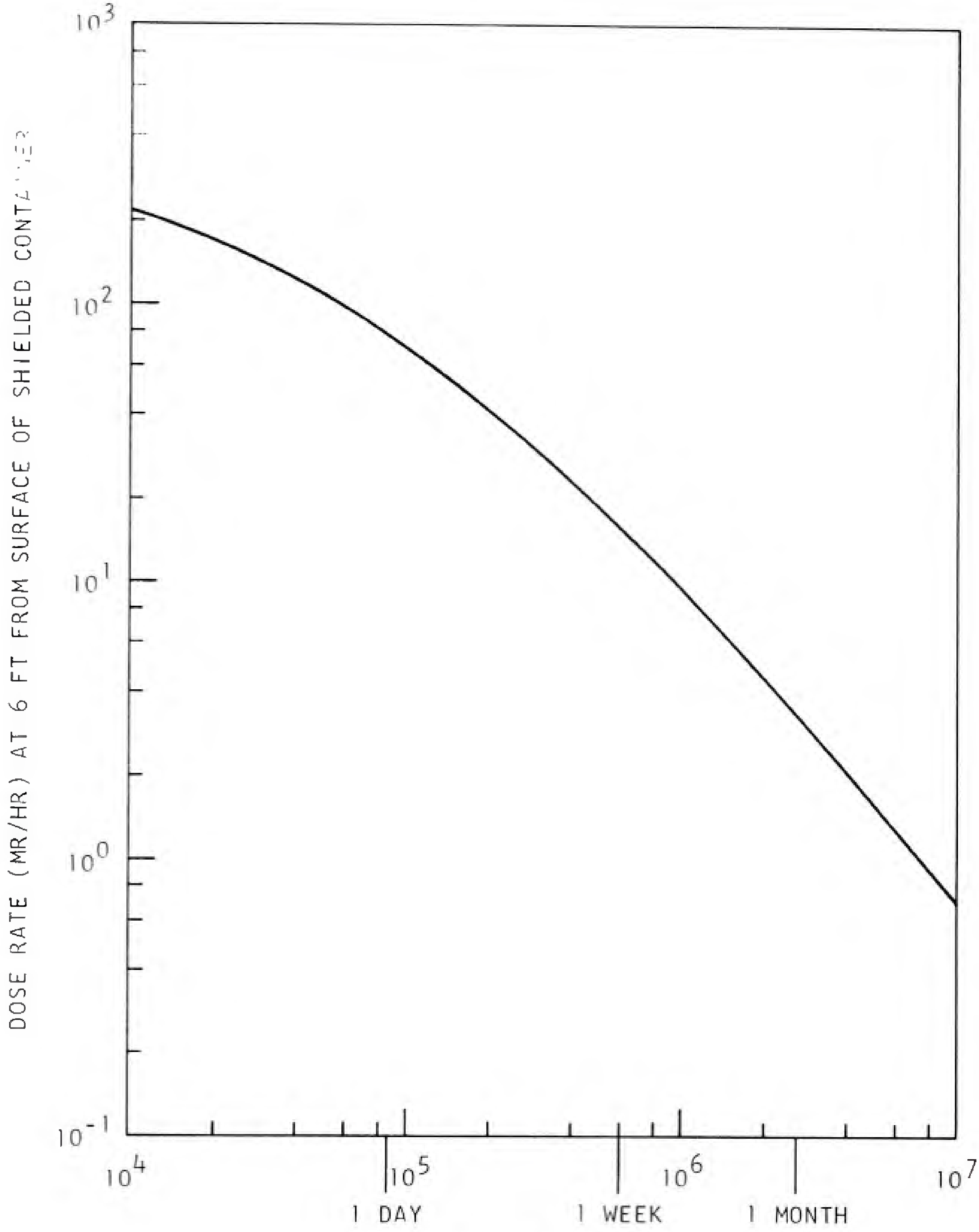

Fig. 3.20--Dose rate outside shielded container from irradiation fuel

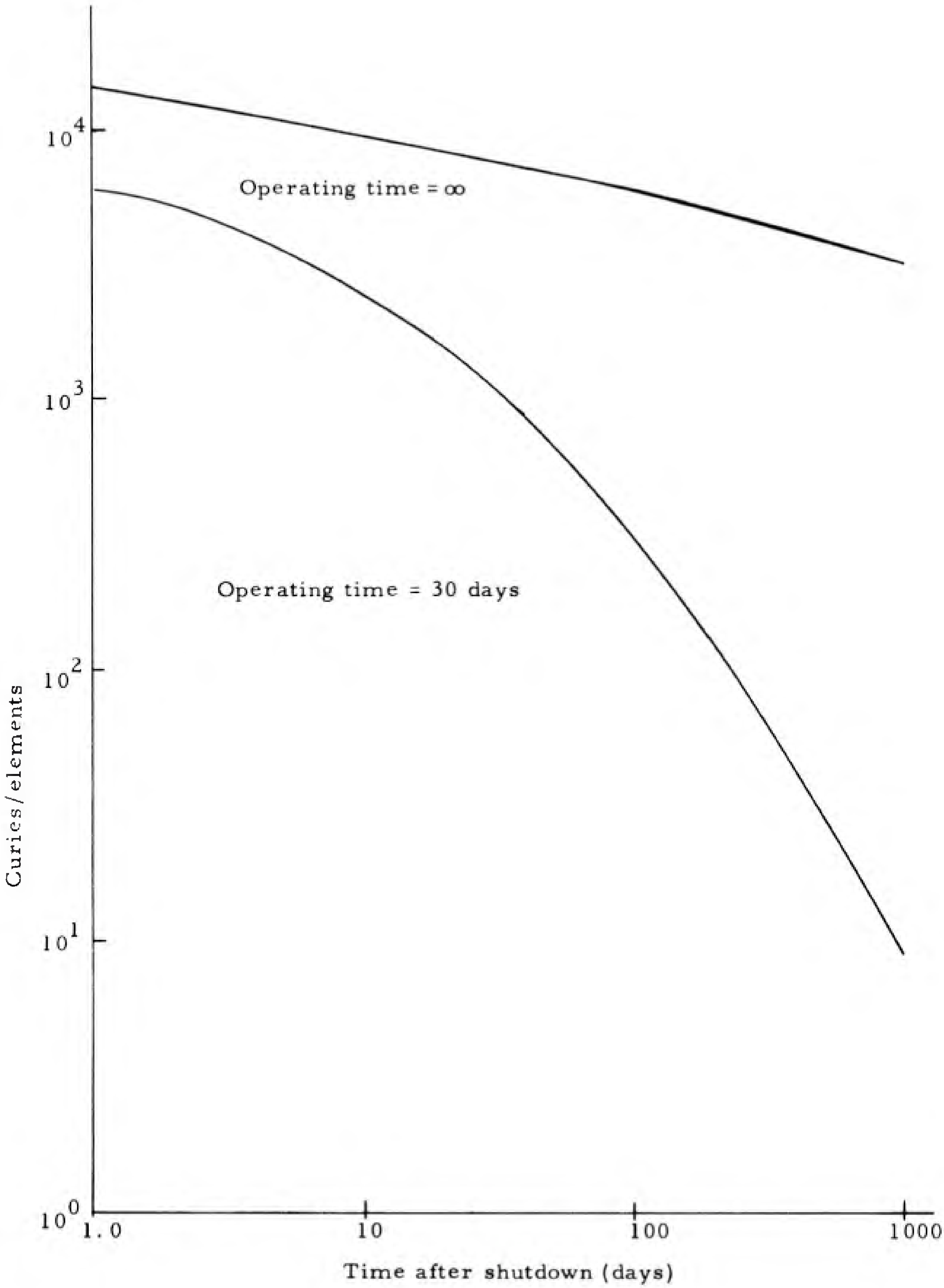

Fig. 3.21--Fission product inventory in one TRIGA fuel element after shutdown

evaporation before ejection) the level of contamination into the hull should be very low and may be handled as indicated above. It is expected that under normal conditions the TOPS power supply could be operated for extended periods with steam generator leaks before contamination would be significant. Evidence of steam generator leaks carrying radioactive contaminants will be immediately apparent by instrumentation provided in the secondary loop. Standard decontamination procedures for the secondary loop may be carried out in dry-dock or dock-side facilities. No such operations are expected to be performed in a submerged environment or on the ocean surface, when coupled with the tender.

Control of decontamination during refueling operations will be accomplished by careful planning of each individual operation and by the use of disposable plastic, drip-catch-sheets and wraps. No special or unique contamination control equipment, facilities, personnel, or situations are envisioned.

Gaseous radioactive contaminants are expected to be held to acceptably low levels. Argon release will be minimized by confinement of the air in the reactor vessel insulation space located near the core so that the confined air cannot readily mix with that in the space above. Fission product tritium release observed in pressurized water reactors that employ SS-UO_2 cermet fuels has not been observed in TRIGA fuels nor would it be expected since the U-ZrH retains hydrogen isotopes whereas stainless steel has a low hydrogen solubility and tritium fission fragments tend to diffuse out of the metal. Tritium productions in the shield water will be avoided by the use of lithium-free corrosion control additives.

3.7. PRIMARY COOLANT SYSTEMS

3.7.1. Water Treatment

Continuous primary water treatment is avoided in the TOPS system through the use of corrosion-resistant structural materials, water and system pretreatment, and by the use of hermetic sealing throughout.

Treatment of the primary system water starts with the selection of materials for the system, which has been discussed in the preceding sections, and will not be repeated here except to recall that Incoloy-800 and the stainless-steels are among the most corrosive-resistant alloys to high-purity water. Surfaces of the primary system will be cleaned and pretreated prior to hermetically sealing. The water in the system will also be pretreated and later treated *in situ* to de-aerate, deoxygenate, demineralize, and departiculate. The vessel will be purged of air and nitrogen; then the reactor pressure vessel will be prepressurized with hydrogen gas to the equivalent partial pressure of 2.5 atmospheres in equilibrium with the dissolved hydrogen at operating temperature and volume. Finally, the fill vents will be hermetically sealed by welding.

No further treatment of the primary system will be performed during operation of the reactor over at least one MUS operating cycle (~30 days) and probably much longer. Since the system is hermetically sealed, there will be no feedwater or makeup water impurities introduced, and no radiolytic gas products can escape. Under these conditions equilibrium gas pressures are low; no catalytic recombiner is required or provided. No provision is made for chemical additives to be injected, and/or sampling of the water condition during operation, so that pH is not externally controlled. A demineralizer is therefore not needed. The factors which permit such a simplified water chemistry system are discussed in detail in Appendix D.

During operation the system is expected to perform as follows. Residual dissolved oxygen in the water will initially oxidize surfaces in the primary system, forming a protective barrier over the metal, thereby inhibiting further corrosion. Corrosion rates comparable to those of stainless steel in pH-controlled water are anticipated. Removal of oxygen in the formation of the oxide barrier will produce an excess of hydrogen in the system and cover gas which will shift the equilibrium concentrations of the reactions and thereby enhance recombination rates of the radiolytic products subsequently generated. The low flow velocities associated with thermal convection should limit the rate at which crud leaves the surfaces and enters the water, and to be conducive to the setting of crud on the bottom of the reactor vessel where it would be innocuous. The pH of the initially neutral water is expected to vary between 7.0 and 9.6 (solubility limit of $Fe(OH)_2$) with a more-or-less constant value nearer 7.0. Fouling of heat transfer surfaces is not expected to be significant even at full design heat fluxes, which are ~4 times those needed in the MUS application.

At the end of an operating interval, to be determined during initial operation, the primary system will be opened and the water sampled. If the water analysis shows treatment is needed, the water will be circulated through an external treatment plant to restore it to its initial condition as nearly as possible. Chemical and mechanical flushing decontamination and de-scaling may be employed, if necessary.

3.7.2. Auxiliary Systems

A schematic of the primary coolant fill and relief system is shown in Fig. 3.18. Demineralized and de-aerated water will be used to fill the primary system.

As indicated in Fig. 3.18, the major portion of the fill system will be located either on a tender, or at the loading dock. This equipment consists of the fill tank, demineralizer, water pump, vacuum pump, and inert gas purge system. Connection to the primary system will be made through the fill and vent lines on the TOPS. These lines will be closed by block valves and capped prior to commencing normal startup procedure.

After the fill and vent lines are connected to the service facilities, the vacuum pump can be started. The vent line valves are opened between the reactor and the vacuum pump. The reactor vessel can be pumped down to a low pressure, thus removing most of the oxygen in the system.

The vacuum pump line valve can then be closed and the reactor vessel filled with purge gas at low pressure.

The fill block valve can now be opened and water pumped into the reactor vessel. The valves connecting the vent line with the fill tank can now be opened. The check valve in the vent line will maintain a slight positive purge gas pressure within the reactor vessel.

The fill pump is kept running until water is forced out of the vent line into the fill tank. At this point the vent line valve above the vacuum line connection is closed and the valves in the water pump system switched, so that water is being pumped out of the reactor vessel. As the water recedes down the vent line, purge gas is allowed to fill the void volume above the water surface.

When the reactor water level reaches the fill line inlet the water pump will lose suction and will be turned off.

The vacuum, purge gas, water fill, water removal, and purge gas cycle will be repeated several times to ensure that as much oxygen as possible is removed from the system. The system will be left with water level at the entrance of the fill line and a cover of purge gas at an over gas pressure of 2.5 atmospheres. The block valves will then be closed, the fill and vent connections removed, and the lines capped.

During operations where the reactor head is to be removed, air can be bled into the system in place of the normal purge gas. In addition, the reactor coolant will be circulated through the demineralizer system until the level activity in the water has been reduced to an acceptable limit. When this has occurred, the water level will be lowered to its normal operating level and a sufficient amount of water will be added to the system to ensure that there is at least 10 ft of water over the core, as required for shielding, before the head is removed.

The pressure relief system, also shown in Fig. 3.18, consists of a burst disc and relief valve combination. A flow check valve is provided around the relief valve to vent any minor amounts of leakage which might occur around the burst disc, thereby maintaining atmospheric pressure in the line between the burst disc and the relief valve. In the event of a rupture of the burst disc, the flow check valve would be closed by the resulting increased pressure drop. The relief valve would then open, preventing an excessive system pressure increase.

On return to normal pressure, the relief valve would reseat. Since the pressure drop across the flow check would still be high it would remain closed.

This pressure relief system was chosen since it truly conforms to the requirements of a hermetically-sealed primary. The burst disc selected as reference is fabricated by Calmec Manufacturing Corporation, and is an adaptation of their model 712-S-1, shown in Fig. 3.22.

The Calmec burst disc has several unique features which make it very desirable for the TOPS/MUS system. The adaptation which will be used in the TOPS will incorporate a burst disc which is seal-welded in place to give a maximum helium leak rate of less than 10^{-8} scc helium. Since the burst pressure setting of the Calmec burst disc does not depend on the physical characteristics of the diaphragm, but rather on the snap-over force

INDEX NO.	PART NO.	NO. REQ.	NAME / STOCK SIZE	DESCRIPTION
14	834-3	1	FLANGE-OUTLET	18-8/316 CRES
13	834-1	1	BODY	18-8/316 CRES
12	834-5	1	STRAINER	18-8/303 CRES
11	28775-434	1	O RING	TEFLON
10	[illegible]	A/R	LOCKWIRE	CRES
9	834-7	1	RETAINER	18-8/303 CRES
8	834-6	1	NUT	18-8/303 CRES
7	834-4	1	RETAINER	18-8/303 CRES
6	834-8	1	PUNCH	17-4 PH CRES
5	834-10	5	BELLEVILLE	17-7 PH CRES
4	834-9	1	SLEEVE	18-8/303 CRES
3	834-11	1	DISC	18-8/316 CRES
2	834-2	1	FLANGE-INLET	18-8/316 CRES
1	[illegible]	8	SOCKET HEAD CAP SCREW	CRES

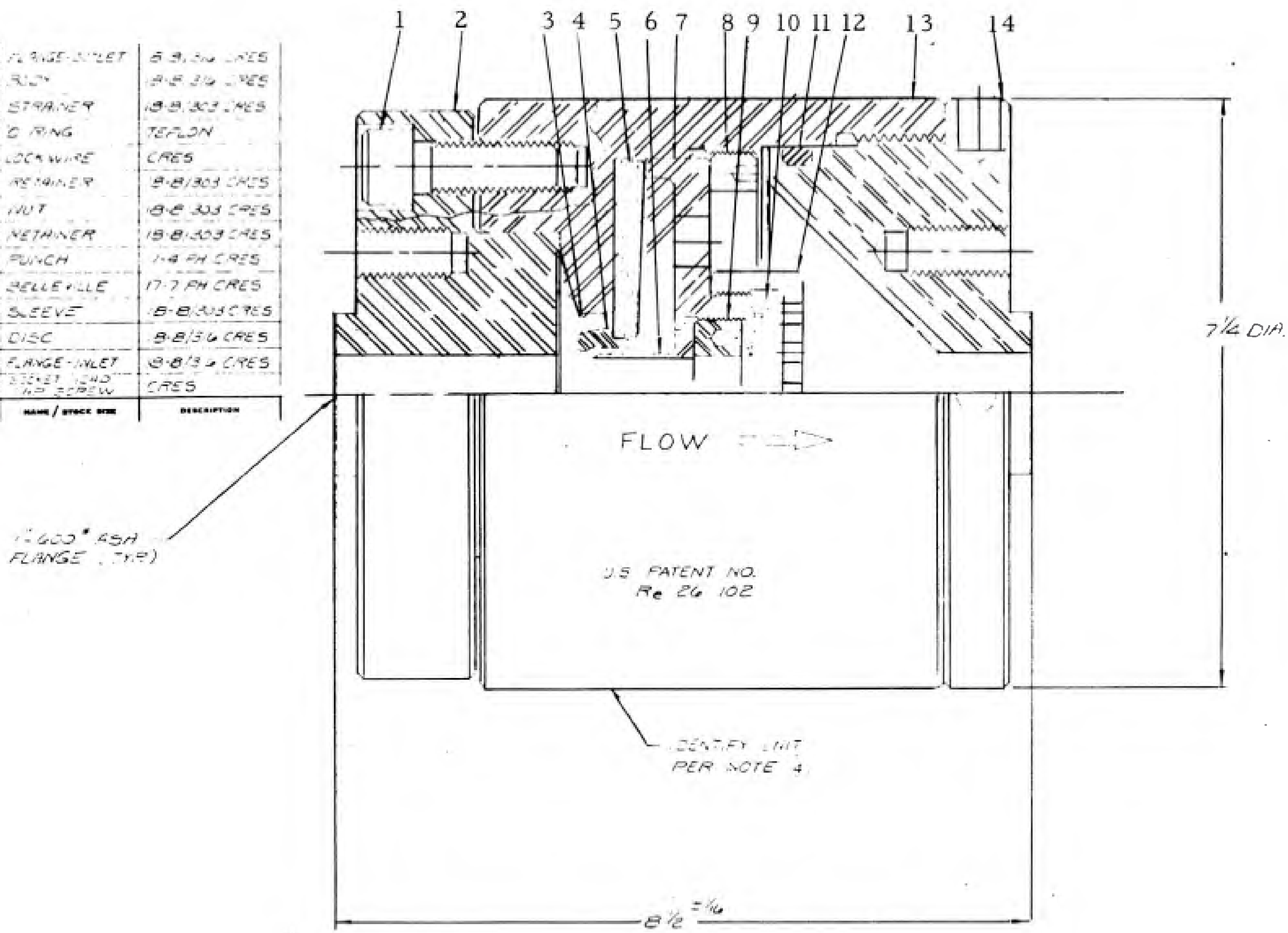

Fig. 3.22--Calmech burst disc

of a bellville spring, the unit can be reliably checked and the burst point adjusted to any desired level within a range.

Calmec has built entire combination pressure relief systems that consist of sealed burst disc, flow check, and spring-loaded relief valves, for use in space fuel tankage. Since these systems are used in space applications they must conform to more stringent leak requirements than a standard pressure relief system.

The use of such a burst disc allows the use of a more conventional spring-loaded relief valve downstream. The valve chosen for the reference design is a 3/4 × 1 JMB-A 316. This valve will have a stellite trim, a 316 stainless-steel spring, and a carbon-steel body.

A pressure switch is provided between the burst diaphragm and the relief valve to indicate the occurrence of a system overpress. This pressure switch will be set at 10 psig.

The pressure relief system will be vented into the center shield tank. It will tend to quench any steam exhausted from the system and minimize the pressure rise which would be associated with a loss of primary fluid through the pressure relief system and in addition will provide a decontamination action for the reactor steam before it can enter the hull atmosphere.

3.8. STEAM GENERATOR

The specifications require that all primary system components have heat ratings commensurate with a net plant output of 500 kw(e). As a part of the primary system, the steam generator is designed to meet this requirement. The 500 kw(e) design conditions were taken to be as follows:

Total heat transferred	2930 Btu/sec
Working fluid (steam side) flow rate	2.45 lb/sec
Working fluid inlet temperature	$100^{\circ}F$
Working fluid outlet temperature	$455^{\circ}F$
Working fluid outlet pressure	262 psia

The rates and conditions correspond to a net output of 500 kw(e) with an assumed overall net power plant efficiency of 0.16.

The steam generator bundle, shown in Fig. 3.8, consists of a rectangular array of tubes. The array is 6 tubes wide (in the plane perpendicular to the direction of primary coolant flow) with a 2D pitch. The array is coiled around the steam generator shroud in a helix.

Tubes are of Incoloy-800, 0.5-in.-OD with 20 BWG wall thickness. Since the tubes will normally operate with higher pressure on the external side, their resistance to buckling is important. The critical buckling pressure, using Incoloy properties at $700^{\circ}F$, is 26,900 psi. This should provide an adequate margin of buckling stability over the maximum allowable vessel pressure of 1250 psi. Tubes are welded to the tube sheets and back-brazed to eliminate the re-entrant discontinuity caused by rolled and welded construction. The average tube length is 50 ft. The resulting heat transfer area is approximately 230 ft^2, which gives a 20% overdesign margin. Supports and spacers will be provided to hold the tubes in their correct positions. The inlet header is preceded by an orifice plate with guide tubes to uniformly distribute the feedwater flow into the boiler tubes. This arrangement is also designed to eliminate thermal shock to the tube sheet, particularly during startup and restart.

Alternatively, instead of using a single tube bundle, the steam generator bundle could be divided into two separate arrays of 9 tubes each with separate inlet and outlet headers in order to provide redundancy. In this case, 6 additional valves would be required (4 motor-driven stop valves and 2 pressure relief valves) to take advantage of the redundancy. The added complexity and attendant maintenance and reliability problems were considered to more than offset the potential advantages in this application.

A comparison was made of two types of steam generator bundles:

1. A fully countercurrent flow unit with primary coolant circulating inside the tubes and boiling taking place in a jacket surrounding the bundle. This configuration is analogous to a "fire-tube" boiler;
2. A cross-countercurrent flow type with few, but long, tubes coiled in an annular helical bundle. Primary coolant flows across the outside of the tubes parallel to the axis of the helix. Working fluid is boiled inside the tubes.

Analysis of both of these designs is given in Appendix B. These show that for the 500 kw(e) case, the fire tube design requires 400 ft^2 heat transfer area compared with 220 ft^2 for the helical design. For this reason, the helical tube configuration was chosen. It should be noted that 40% of the heat transfer area in the fire tube design was required by the economizer.

Subsequently a detailed computer analysis of the steam generator was made for both the 500 kw(e) and 100 kw(e) cases using a proprietary computer code developed by Gulf General Atomic specifically for once-through steam generators. The final design is based on the results obtained.

A sketch of some of the details of the steam generator construction are shown in Fig. 3.23. Because the steam generator is designed for use in a 500 kw(e) steam plant with a 20% design margin, it is conservative by about a factor of 5 when used with the 100 kw(e) system. The result is that the temperature of the steam leaving the steam generator will approach the temperature of the primary fluid to within about 5°F and an adequate allowance is available to account for possible heat transfer degradations by boiling. When producing sufficient thermal energy to support a 500 kw(e) secondary system, the approach temperature will be about 15°F.

As shown in Fig. 3.23, the 6 × 6 array of tubes is bent around the upper shroud. The generator tubes are held in place by stay rods welded at intervals around the upper shroud. These stay rods not only support the weight of the steam generator but are also provided to minimize any tube oscillations which might occur because of either internal or external excitation.

Feedwater enters one of the steam generator headers where it is diverted into the 36 tubes which are welded and brazed to the tube sheet. In order to prevent excessive thermal stress which might occur as a result of

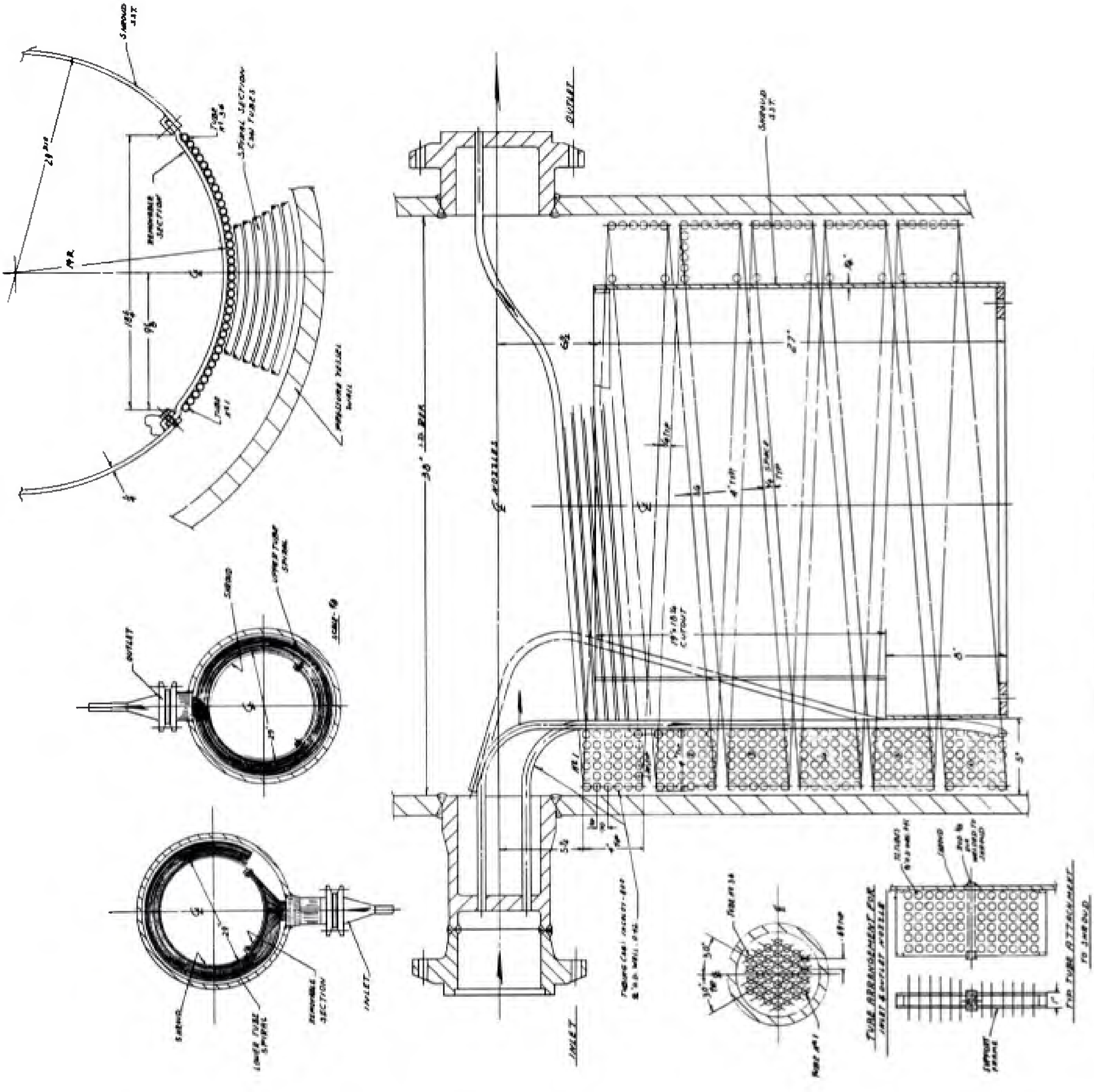

Fig. 3.23--Steam generator installation

the relatively cold feedwater impinging on the hot tube sheet, a series of baffle plates is provided to prevent impingement of the feedwater on the secondary side of the tube sheet. This essentially allows the semi-stagnant water adjacent to the secondary side of the tube sheet to act as insulation against high rates of heat transfer. This detail can be seen in Fig. 3.24.

The insulating baffles are supported on the tubes, which, as previously discussed, act as orifices for the inlet of the steam generator.

During installation or removal of the steam generator, both the inlet and outlet ends of the tubes must be deflected toward the center of the generator so that the ends of the tubes can clear the inside of the reactor vessel. A special removable partition is provided on the inlet side of the upper shroud in order to allow room for this movement. The outlet tubes which terminate at the top of the steam generator have sufficient room to allow the required deflection without special design considerations.

After the steam generator is installed in the reactor vessel, the tube ends will be fitted into their respective places in the tube sheet and TIG seal-welded into position. Following welding and a helium leak and dye penetrant test, the tubes will be back-brazed using local RF heaters and preforms of nickel, silicon, and boron.[4]

The design of the steam generator headers is such that tube plugging can be accomplished with relative ease. With the bolted inlet and outlet flanges removed, there is access directly to the tube sheets for insertion of tube plugs, should that become necessary.

The steam generator outlet nozzle is shown in Fig. 3.25 and is similar in design to that of the inlet nozzle. The tube sheet is small and compact as shown in Fig. 3.26, and is only 5.5 in. in diameter across the face.

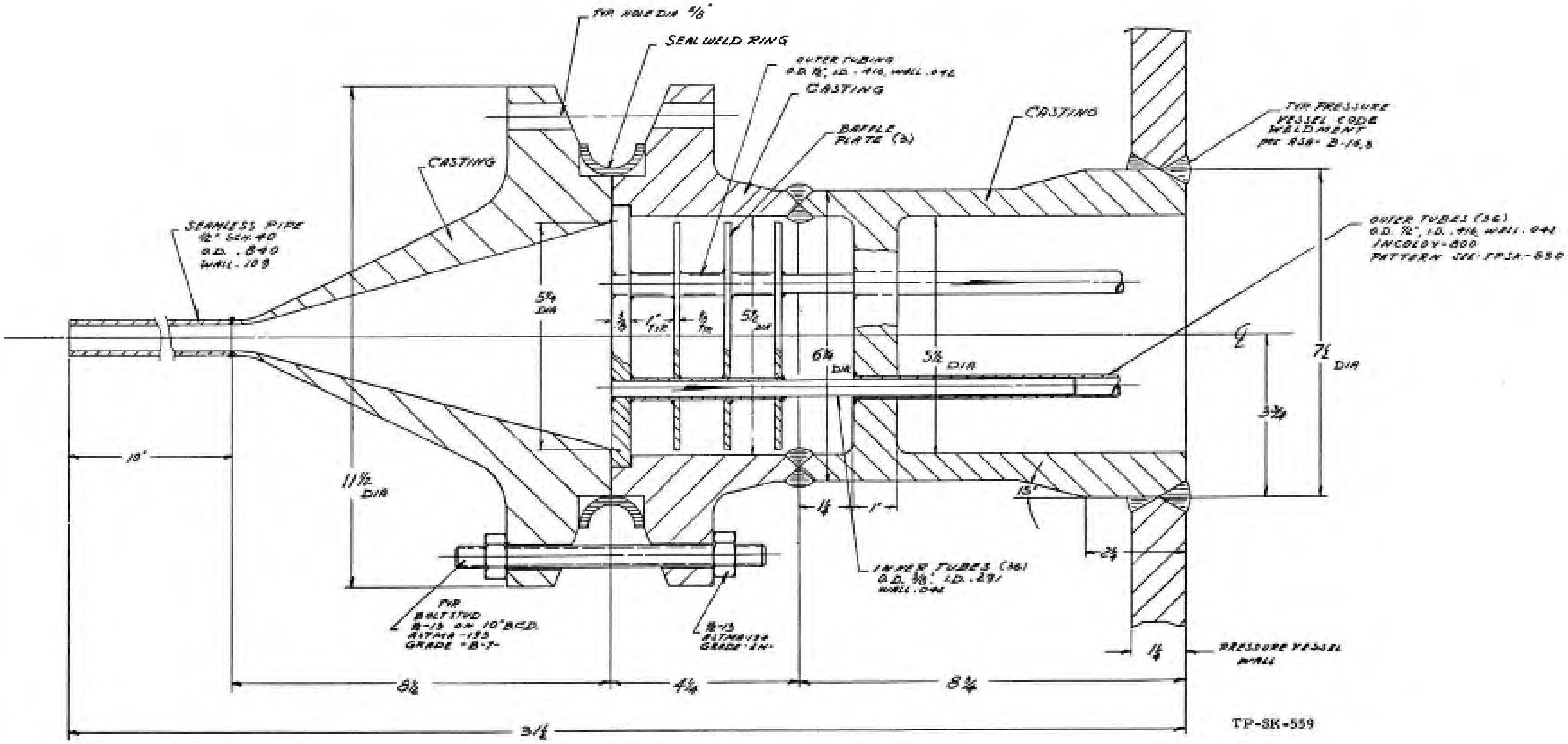

Fig. 3.24--Steam generator inlet nozzle

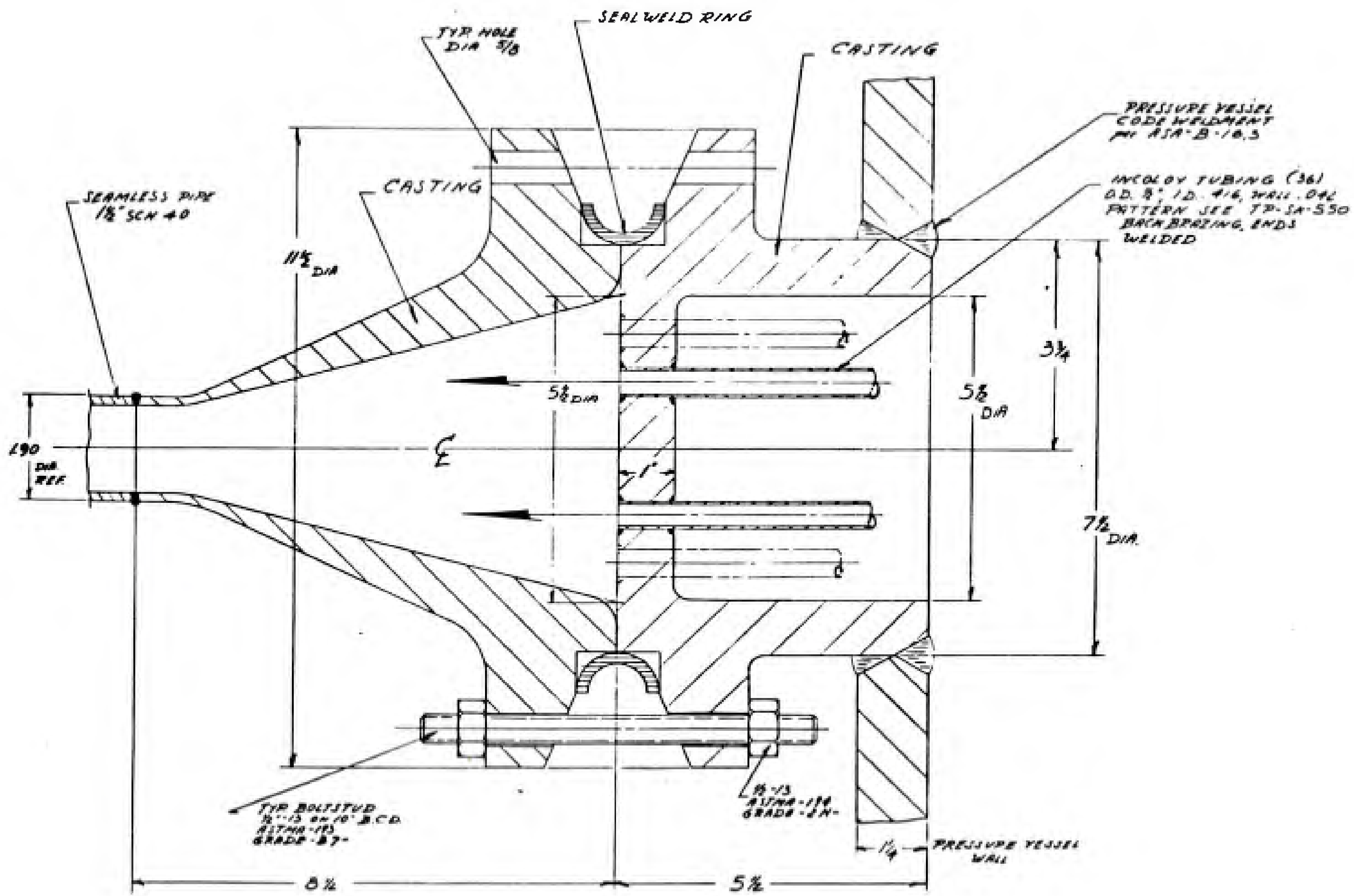

Fig. 3.25--Steam generator outlet nozzle

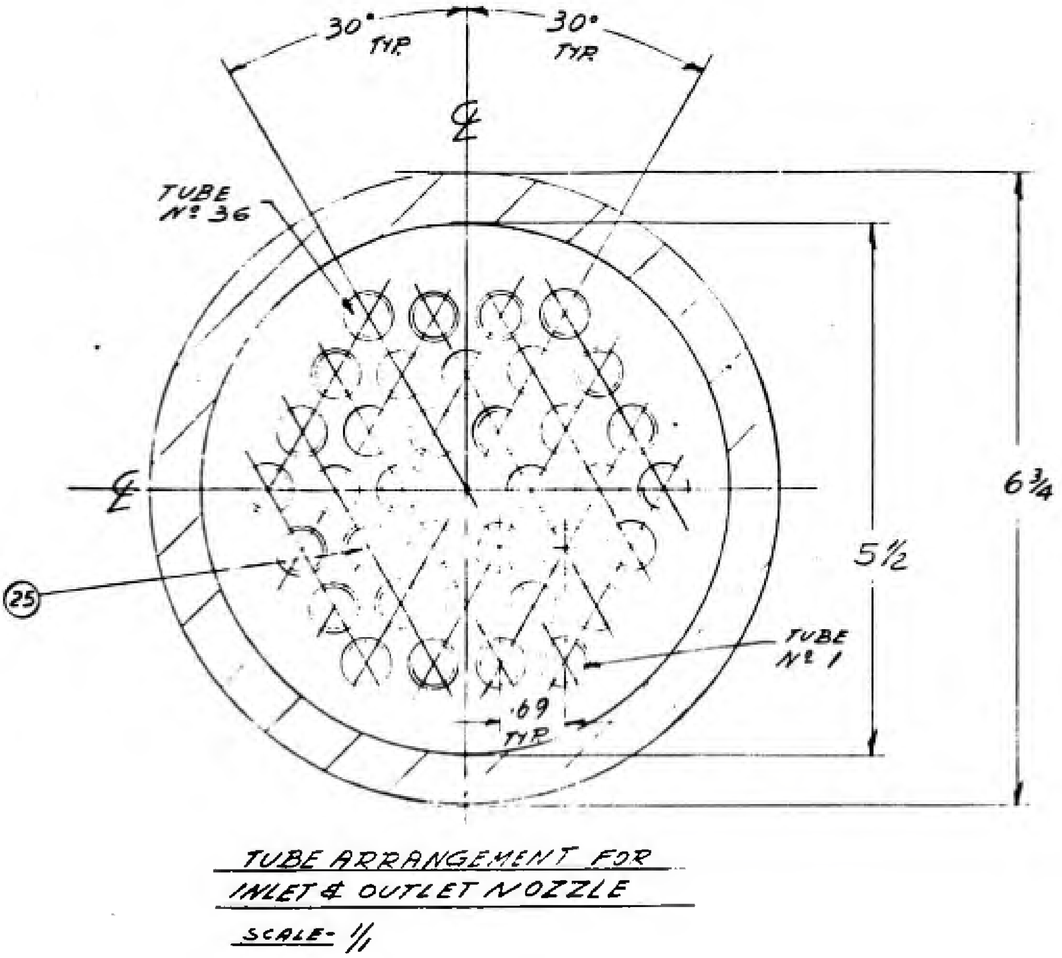

Fig. 3.26--Tube arrangement for inlet and outlet nozzle

3.9. DECAY HEAT REMOVAL

The rate of thermal production used to investigate the reaction of the reactor system to decay energy was based on the data of Shure [5]. The curve of percent of equilibrium power produced before shutdown, as a function of time after shutdown was given in Fig. 3.19.

In keeping with the philosophy of maximum system simplicity, a separate and unique decay heat removal system is not provided in the power plant. This is made possible by the reasonably large inventory of water in the primary system which provides a large amount of thermal inertia.

Figure 3.27 is a graph of reactor primary pressure as a function of time after reactor scram, assuming the reactor is operating at a power level equivalent to 100 kw(e) net output.

Five cases are represented in these curves. The first case shows what might happen to the reactor pressure if there were no external cooling and, in addition, the primary system were perfectly insulated. This actually represents an impossible situation since perfectly insulating a source of thermal energy is not possible and therefore is used as an extreme upper bound on the reactor pressure time curve. The case is not trivial, however, since it shows that even under the worst possible conditions it would take approximately 24 hours before a hazardous condition would be encountered. After this time, the reactor pressure would rise to a point where the relief valve would open and start blowing off primary coolant. This, of course, would tend to cool off the reactor and lower its pressure.

The next curve represents the pressure time history for a system insulated to such an extent that it allows only 1/2% of the thermal power produced within the core to leak out when the system is at operating temperature. In order to achieve this result, one would have to insulate the entire reactor with an equivalent of 2 in. of 85% magnesia insulation.

The maximum pressure reached by the system in the 1/2% heat leak case was not calculated, but is estimated to reach the order of 950 psia after several days, before starting to decay. This pressure represents a safe condition for the reactor pressure vessel.

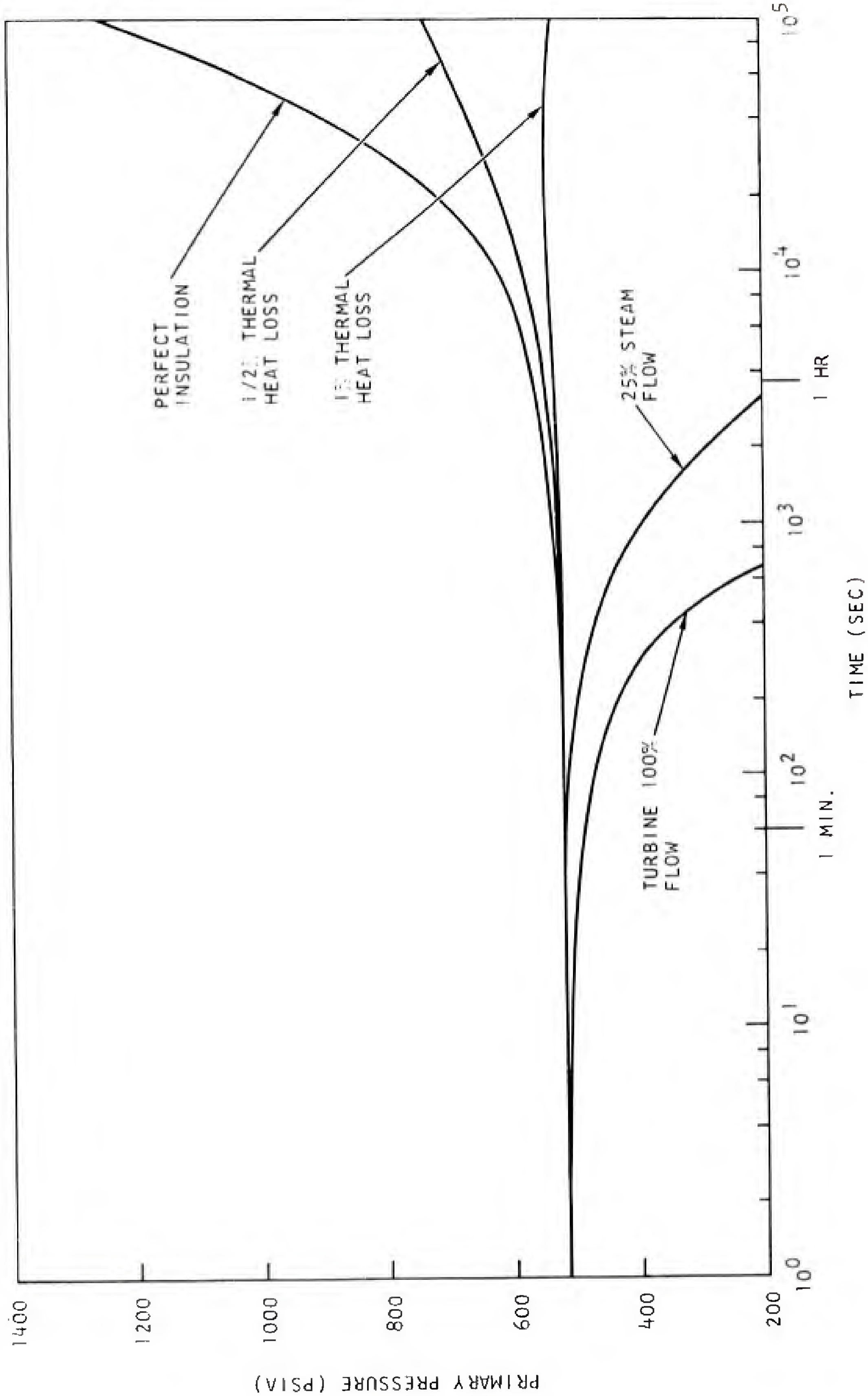

Fig. 3.27--Reactor pressure after shutdown

In actual practice, the reactor will be insulated with the equivalent of 1 in. of 85% magnesia. This will allow approximately 1% of the full thermal power produced to leak from the system through the insulation when the system is at operating temperature. This case is represented by a third curve and shows that the system pressure would rise from its initial 514 psia to a maximum of only 540 psia at approximately 4 hours after reactor scram. After the maximum is reached, the primary pressure will begin to decrease at a rather slow rate.

The above cases represent abnormal situations, that is, they represent a condition wherein the reactor has scrammed from full power and there is no power available to aid in decreasing the primary temperature at a reasonable rate. Under normal shutdown conditions, decay heat would be removed not only by the loss of the nominal 1% thermal energy through the insulation, but also by continuing to pump water through the steam generator tubes.

A combination of mechanisms is available for cooling the primary under "normal" conditions. The first would occur if steam flow to the turbine were interrupted. In this case, the steam bypass valve (see Fig. 7.1) would be opened initially allowing about 25% of the normal full flow of steam to pass directly into the condenser. This condition is represented by the curve in Fig. 3.27 labeled 25% steam flow.

As the primary pressure approaches the secondary pressure, saturated water will start to be produced. The steam separator will then allow the saturated water to flow through the steam trap and also into the condenser where its thermal energy will also be transferred to the ocean. When the primary pressure falls below the secondary pressure, no net steam will be produced in the secondary. The analyses of the detailed dynamics of the system have not been fully investigated but it would follow a pattern such as this:

1. The reactor scrams and turbine steam flow is cut off.
2. The pressure in the secondary would start to increase toward the cutoff head of the pump.
3. The steam bypass valve would be opened and the pressure would equilibriate at a point somewhat higher than the full flow pressure. Steam would flow through the steam bypass and remove heat at a rate approximately equal to 25% of the full power heat rate.

4. As the primary pressure approaches the secondary pressure, the steam leaving the steam generator will degrade from superheated to saturated and then some net water will pass through the steam trap into the condenser.

5. As the primary pressure decreases further, less net steam will be produced and the pressure in the secondary will reach a quasi-equilibrium at the normal operating pressure of 262 psia. As the condition of zero net steam is approached, a mixture of water and steam will begin to flow in the steam bypass line. This will cause a further decrease in secondary pressure determined by the head flow characteristic of the feed pump.

6. The pressure in both the primary and secondary will fall together until a minimum secondary pressure of approximately 230 psia is reached, after which point only saturated water will be flowing in the secondary. The minimum pressure will be determined by the resistance to water flow through the two parallel lines (steam bypass and steam trap) and the pump characteristic.

7. As the reactor cools off, the water level inside will begin to recede. At some point the mode of heat transfer on the primary side will change from natural convection of water to reflux condensing steam. This latter mode will, if anything, aid the heat transfer. The secondary side heat transfer mechanism will, at some point, change from boiling to forced convection which will probably offset the primary change.

It should be pointed out that there is a relief valve in the secondary which will not allow the pressure in the secondary to reach the cutoff head of the pump, so that even if the steam bypass valve cannot be opened, the reactor will be cooled off, but at a rate somewhat less than normal. No net primary pressure increase will occur in this case, and the only effect would be a longer than normal cool-down time.

If for some reason it is desired to cool the reactor down at a very rapid rate, one could operate the turbine for at least the first several minutes following reactor scram. This would result in a pressure time history which would correspond to the turbine 100% flow line shown in Fig. 3.27. The turbine should not be left on the line longer than about ten minutes at full load without further remedial action, since there is a chance that saturated water may start to flow in the steam line where damage to the turbine blading could occur.

If the steam block valve or trip valve cannot be closed, it would be necessary to do one of two things:

1. Reduce the generator output to a minimum, or
2. Open the circulation pump bypass valve to decrease the steam generator pressure and reestablish net steam production at a lower pressure. This, of course, would decrease the turbine speed and minimize the possibility of damage to the blading.

A recap of the decay heat removal system is as follows:

1. Under normal conditions the main circulation pump will be kept running. The load will be removed from the turbine and after a time, the turbine will be shut down by closing the steam block valve. The steam bypass line will be used to bring the reactor pressure down.
2. Failing operation of all systems, there will be enough thermal capacity built into the primary and heat loss from the reactor system through the insulation to the shield tank water, and ultimately through the pressure hull and into the seawater, to limit the primary system pressure rise in the extreme emergency case to the order of 25 psi. This does not constitute a hazardous condition.

3.10. REACTOR CONTROLS AND INSTRUMENTATION

Reactor controls and instrumentation include those systems which are required to start, operate, and shut down the reactor plant and to ensure its safety through normal operating phases and any credible emergency conditions.

3.10.1. Reactivity Control Requirements

Reactor control is simplified by the inherent self-controlling features of the TRIGA core. For normal operation the primary system can be brought to criticality at a temperature defined as T_0 with the system adiabatic and zero power being produced. Since the primary system is self-pressurized, with steam and water in equilibrium, the system pressure will be the vapor pressure of water at T_0. As heat is demanded from the reactor the fuel and primary coolant temperatures will swing up and down, respectively, under the influence of the fuel and water temperature coefficients. At the same time, heat dissipation through the steam generator causes a drop in primary coolant temperature at that point, balancing the temperature rise in the core, and causing circulation by thermal convection. The temperature at T_0 is set by the required fuel and primary coolant temperature at rated power and the fuel and water temperature coefficients. Figure 3.28 shows the steady-state operating points between zero and full power for a nominal 100 kw(e) net plant output. This figure shows that on a quasi-steady-state basis the primary system will be a heat source that responds to any demand between zero and full power without control action. A transient analysis has shown that the transition of plant operation from one operating point to another is accomplished within one cycle. Therefore, control action is required only to bring the reactor from a cold, subcritical condition to the T_0 condition, shim to compensate for fission product poisons or burnup, or to return the reactor to the cold, subcritical state. It is not necessary for any of these control actions to be automatic.

Startup from a cold, subcritical condition is achieved by a constant speed rod withdrawal. The rods are withdrawn in sequence at individual rates of 2 in. per minute. At this rate, it will require approximately 17 min to go to initial criticality from a cold, clean, rods-in condition. Going from initial criticality to criticality at T_0 will require an additional 20 minutes. This rate of rod withdrawal corresponds to a rate of reactivity addition that can be overridden by the negative reactivity input from the temperature coefficients and the rate of temperature rise corresponding to 3,000 Btu/sec

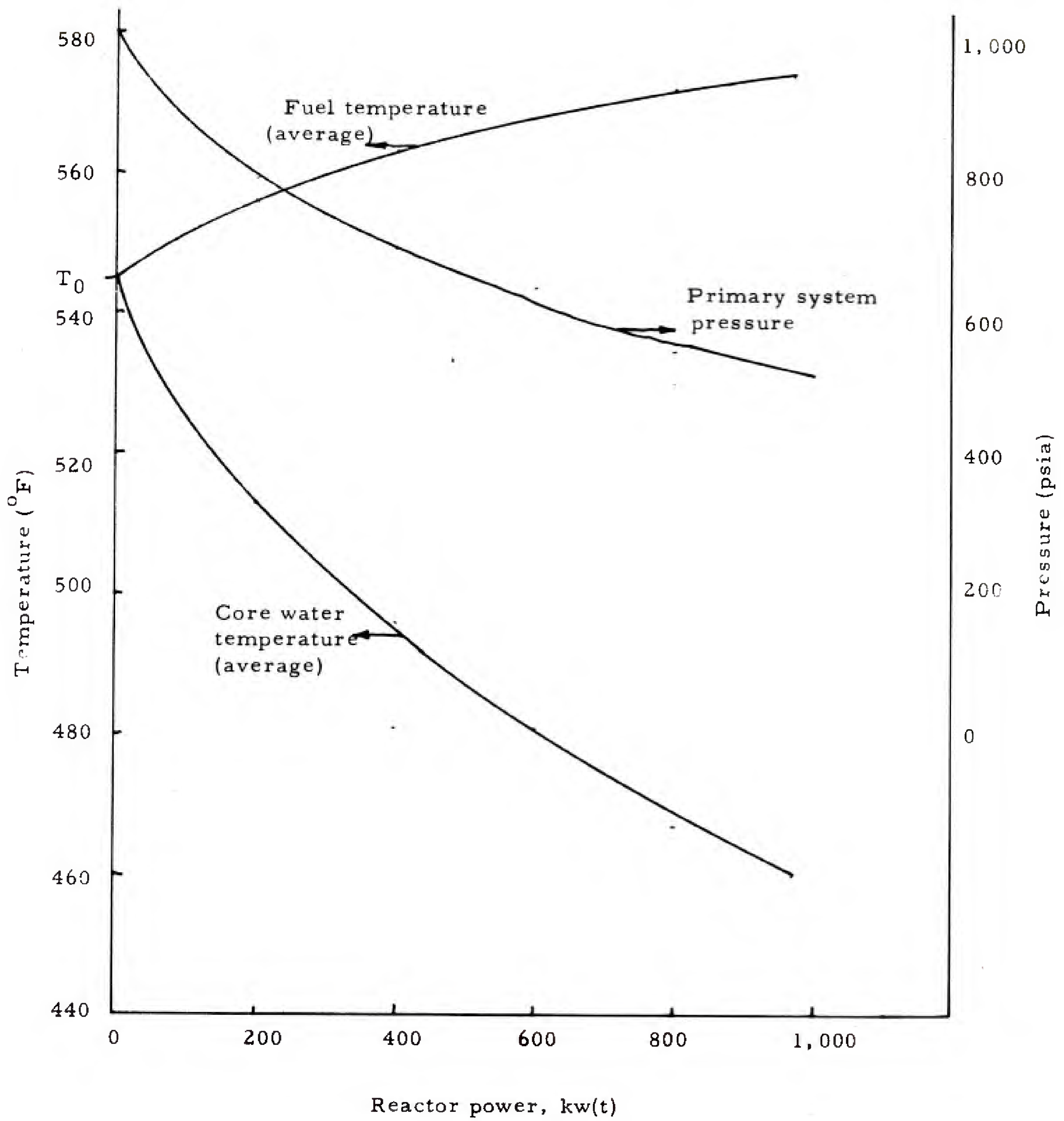

Fig. 3.28--Fuel and water temperature variations in heat load

heat output from the core. Since the ramp rate of reactivity addition is compensated by the inherent properties of the core, there is no need for servo-control of rod withdrawal.

Since the core is rated for a 500 kw(e) net plant output it can safely generate and dissipate 3,000 Btu/sec to the primary coolant, although for the initial 100 kw(e) nominal net output core it would be required to produce a maximum of 920 Btu/sec at the design point. For normal operation, the control rods will be left in a fixed position.

During early operation the buildup of fission products to their equilibrium concentration will cause T_0 to drift downward, compensating the negative reactivity of FP poisons with the positive effect of reactivity coefficients on a decreasing temperature ramp. Later in the core lifetime, a similar effect will take place owing to the fuel burn-up. In both cases, the operator will manually shim the control rods to regain the design T_0. Operating instructions will contain a chart showing T_0 as a function of reactor power and primary system pressure to guide the operator in making his adjustment. Characteristics of the TRIGA-type core minimize the amount and frequency of shimming adjustments.

Normal shutdown is accomplished by manually selecting "Rods In" on the control rod selector switch. There is no need for automatic control or limits on this operation.

In principle, the primary system can be protected against every credible incident by a logic system input from a single thermodynamic parameter. This parameter can be, optionally, primary system pressure or temperature. The former is preferred because pressure is the ultimate quantity against which the primary system requires protection, and pressure changes are greater in terms of precentage at operating conditions which give more sensitive measurements. As a backup, high fission power is also used as a safety parameter. Two automatic safety actions are initiated by primary system pressure,

1. The logic system will not permit rod withdrawal if the primary system pressure exceeds 850 psig,
2. The logic system will initiate a "Rods Drop" if the primary system pressure exceeds 1040 psig.

In addition, high fission power initiates "Rods Drop" at a power level in excess of 3.4 Mw(t).

Fission power will also be monitored and displayed as instantaneous kw(t) and integrated Mw-days. The latter will be displayed on a digital counter. Since the relation between control rod withdrawal rates and the reactivity coefficients of the TRIGA core always keep reactor power within safe operating limits, it is not necessary for any automatic control action to be initiated by the fission power parameter. Loss of primary system pressure causes the core to equilibrate at lower-than-rated power. This has the advantage of permitting approximately 1-hr of operation before control action is necessary. Also, it prevents fission power from being an effective control parameter for the loss-of-coolant shutdown. Since the plant is a demand heat source, power at any time during normal operation is fixed by the output power. Fission power is, therefore, an unnecessary parameter for automatic control.

Primary system temperature will be detected by temperature sensors installed in the steam generator tube sheets. Primary system temperature is a useful quantity during startup since the percentage change in temperature is greater than the percentage change in pressure when the system is at low temperature

3.10.2. Primary System Instrumentation

The instrumentation and safety system have been designed to conform with the objective that the "instrumentation shall be as simple as possible consistent with the control inherent in the reactor through internal reactivity coefficients." Figures 3.29 and 3.30 show the instrumentation and safety systems. Two wide-range log power channels are used to indicate power level as the reactor is brought from source level to full-power. Three linear power channels and three pressure measurement channels are used in the safety system to limit excursions. Six control rods are withdrawn sequentially at a safe rate of reactivity change and are inserted as a group.

3.10.2.1. Reactor Power Instrumentation

Wide Range Channels. Two wide-range log power channels are included to provide neutron flux information during startup from the source range to full power and above. These channels are capable of flux measurements through ten decades, using the ac signal component from a fission chamber. Each channel includes a fission chamber, preamplifier, connecting cable, and log power drawer, containing high voltage power supply, and test and calibration circuits. The output signal from either of these channels is displayed on the reactor console. Operation of either of these is sufficient for startup and power operation.

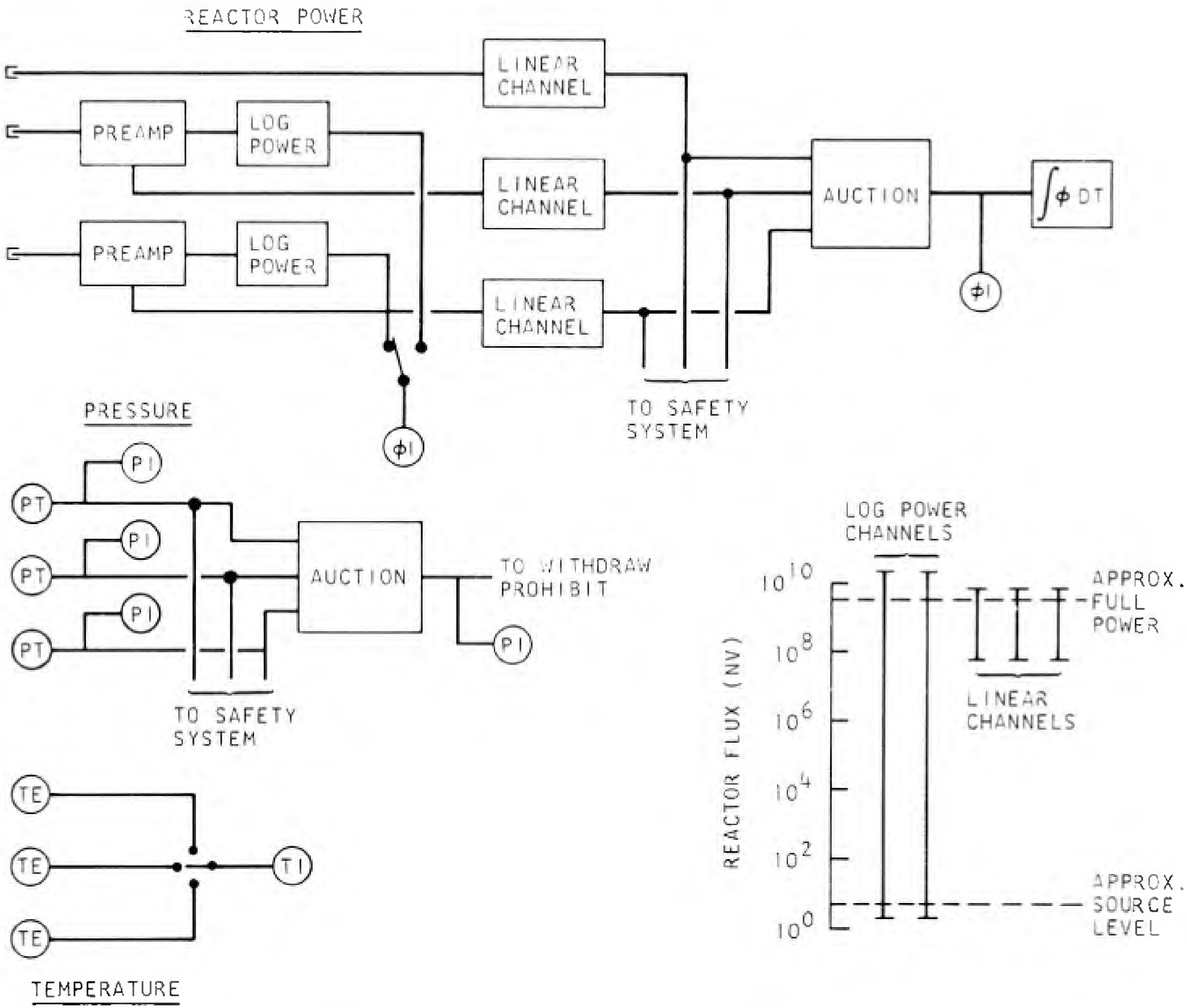

Fig. 3.29--Reactor control instrumentation

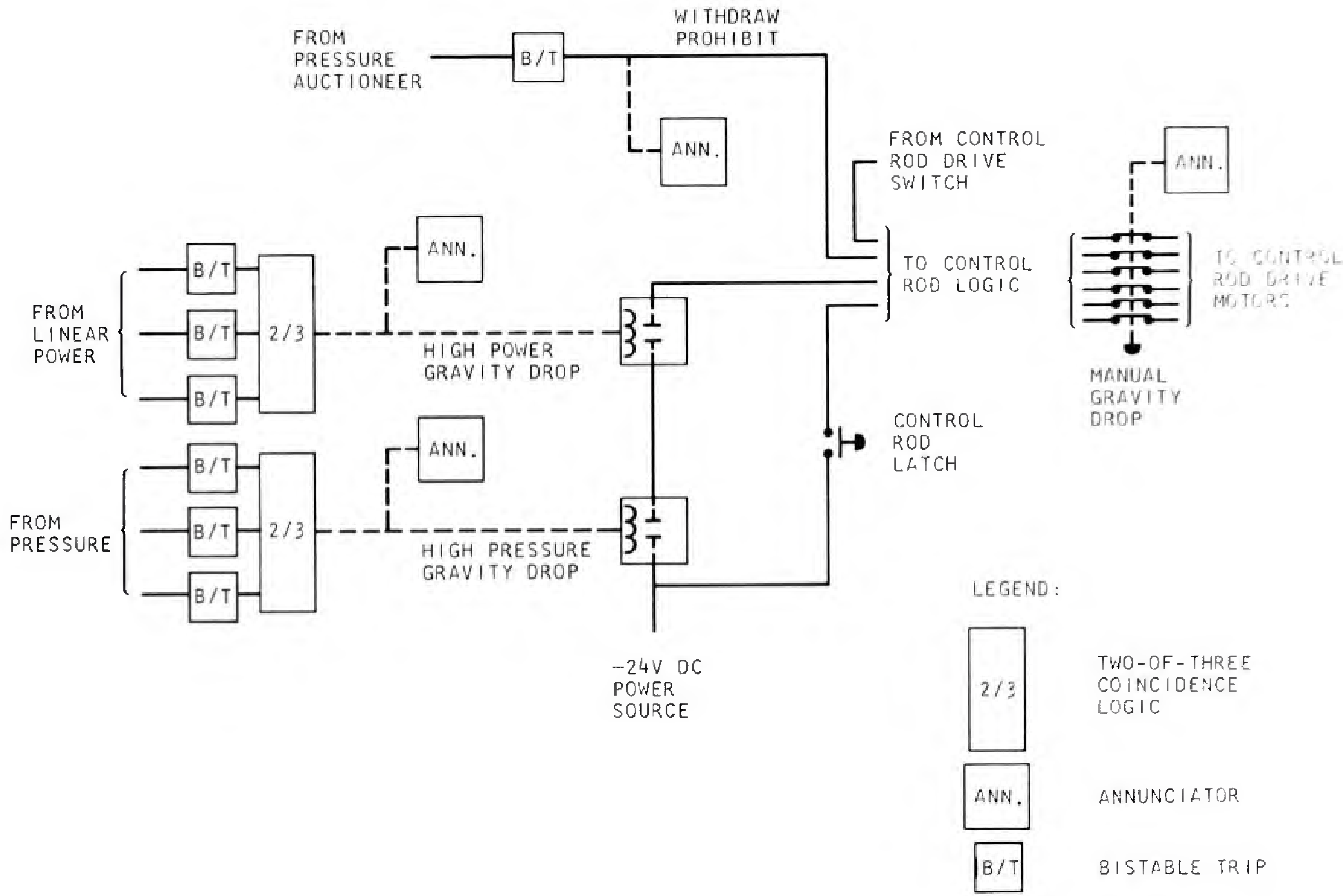

Fig. 3.30--Reactor safety system

Linear Power Channels. Three linear power channels are provided to allow for more accurate flux level measurements in the useful power range necessitated by integrated power (total energy) measurement and safety system requirements. These channels have an effective linear range of over two decades with 100% power adjustable (by internal adjustment) between 0.1 and 4 Mw. Two of these channels use dc current signals from the same fission chambers that are used for the wide-range channels described above. The third uses a similar chamber. Also included are connecting cable and test and calibration circuits. Level information is also transmitted to the safety system described later. For maximum reliability of operation, the preamplifier in the line between the detector and linear channel, as shown in Fig. 3.29, is designed so that an electronic failure in the preamplifier or in the log power channel will not cause failure of the associated linear channel. An auctioneer circuit is used to provide the highest measured power level to an indicator and to the flux integrator. Buffering is used on the inputs to the auctioneer circuit so that a failure of the auctioneer will not interfere with the safety action of the individual channels. The buffering also provides isolation between channels.

3.10.2.2. Non-Nuclear Instrumentation

Pressure Indication. Three channels of pressure measurement are provided. The transducers, using a twisted Bourdon tube driven potentiometer, are attached to control rod drive penetrations. Here the method requires no breach of the pressurized system, since the potentiometers are outside the pressure vessel wall. Pressure information is also transmitted to the safety system described later. As with the power channels, buffering provides protection here between channels and at an auctioneer circuit which provides a safety action signal and indicator signal from the highest measured pressure.

Temperature Indication. Cursory temperature information is provided using chromel-constantan thermocouples, mounted in the steam generator tube sheets, which directly drive cold junction compensated millivoltmeter indicators.

3.10.2.3. Safety System

General Philosophy. The only necessary reactor safety requirements are concerned with control-rod motion. There are two levels of safety control. At the lower level there is a control rod withdrawal prohibit; and at the higher level, a loss of drive power causing a gravity insertion.

Implementation. The power level channels are connected in a 2/3 coicidence logic system, as are the pressure transducers. Either of these systems cause the gravity insertion described above. The pressure signals are also auctioneered through a buffered auctioneer circuit and the highest is used to actuate the lower safety level of control rod withdrawal prohibit. As a result of the design, pressure follows power closely and this action is sufficient to limit pressure during normal operation. In addition, a manual switch is available to cause a gravity insertion.

Annunciation is provided to indicate the occurrence of each safety action and rupture of the burst disc.

3.10.2.4. Control

Control Rod Drives. The six control rods are each driven by a stepping motor with a maximum continuous speed equivalent to 2-in. min. They are moved out of the core sequentially with automatic transfer to the next rod after each step of a drive motor. This has the effect of keeping all control rods at the same level in the core. During reinsertion, all rods move as a group. These operations are accomplished using a three-position manual switch station, spring-loaded to the center position for no rod movement. With the control switch in the center position, holding current is provided to each rod drive to ensure no control rod movement; breaking this current or a power failure causes a gravity insertion. The drive system is also controlled by signals from the safety system described above.

Control Rod Position. Control rod position is provided using a linear differential transformer in conjunction with each control rod drive. In the type used, the windings are located on the outside of the pressure vessel wall at the control rod drive. This method precludes the necessity of breaching the integrity of the pressure vessel. The ac signal is converted to a dc level and indicated on an individual position indication meter for each control rod.

3.10.2.5. Radiation Monitoring

Two channels of area monitoring with up to three detectors per channel are provided, one channel for the TOPS hull and one for the MUS area. The TOPS module also has an air monitor to measure airborne matter activity, operation of which is remotely controlled from the MUS. All controls, readouts, and alarms for the radiation monitors are located on the Primary System Instrumentation Console.

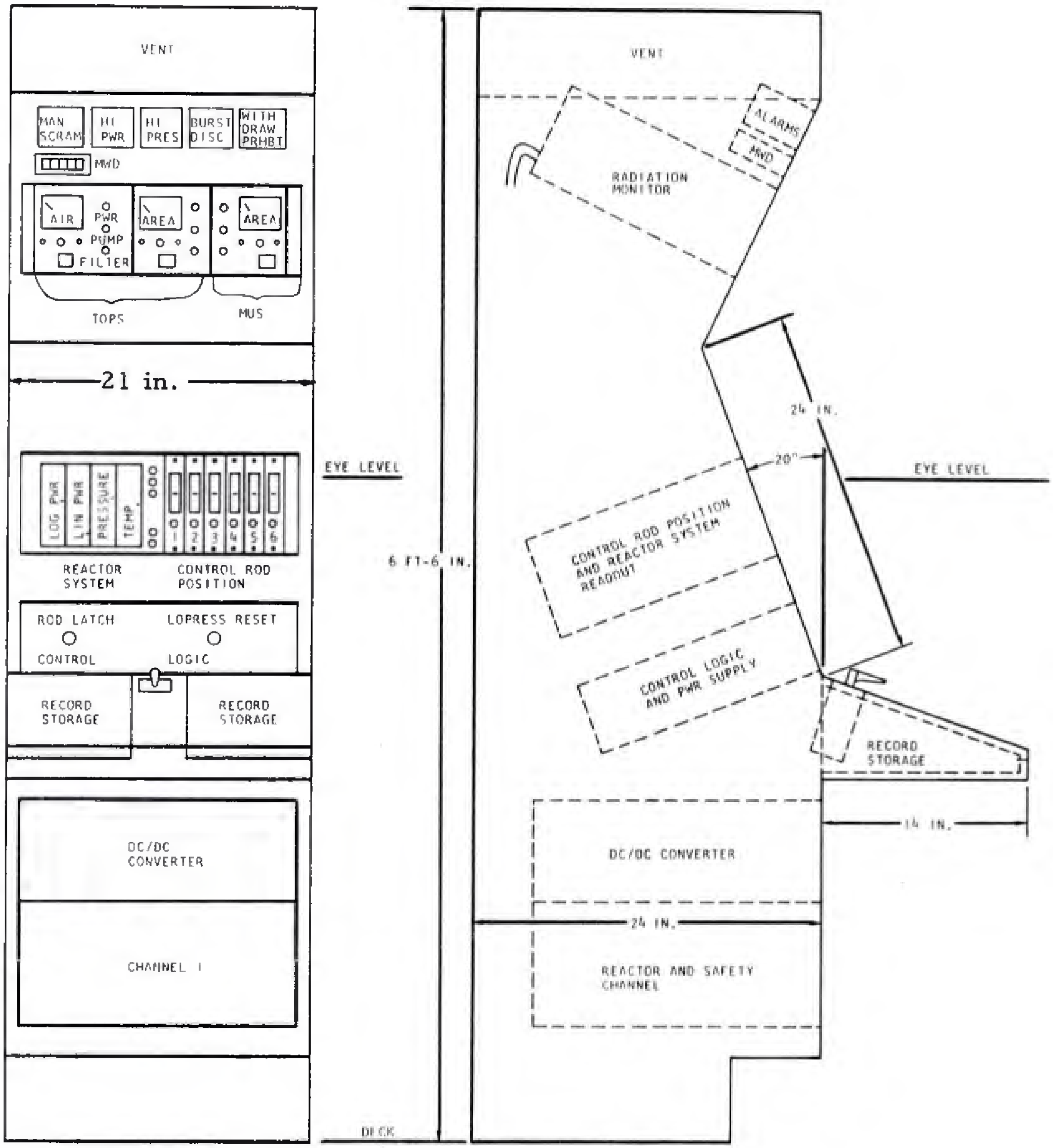

Fig. 3.31--Reactor and primary system control console

3.10.2.6. Display and Packaging

The control instrumentation package is compact, without compromising the integrity of the safety system performance. Most of the instrumentation described above is located in two standard width and depth electronics cabinets. All display and control functions are available, including safety system annunciation, in the reactor primary system control console, shown in Fig. 3.31. Mechanical shields separate the channels of the safety system, two of which are located in available space in the power conversion loop control console located adjacent to the reactor console as described in Section 6.1.

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4. TURBINE PLANT

The following section gives a description of the major components which will be used in the secondary or power conversion system of the TOPS/MUS power plant. A general arrangement drawing of the entire TOPS/MUS system was given in Fig. 2.1 and should be referred to in reference to equipment placement.

The TOPS/MUS power plant is designed in a modular form. The primary module consists of the TOPS reactor, the inner shield tank, and supported on the conical skirt which forms both the reactor support and the upper head for the shield tank is some of the secondary equipment. These pieces of equipment include the feed pumps, demineralizer, and air ejector system. The primary module is attached to the NEPP hull at a flange which forms the upper portion of the lower support cone.

The secondary module contains such equipment as the turbine-generator set, electrical switchgear, and fresh-water cooling pump. This system is supported in a basket-type structure from a location opposite the upper king frame (i.e., support ring).

Some of the auxiliary equipment is located outside the two main modules. These include the air cooling system which is located in the upper hemihead, and some of the tankage which is located in the void space of the outer shield tank.

Referring to Fig. 2.2, steam leaves the TOPS reactor through the nozzle on the right side of the reactor. It then flows up to the steam separator and trap located on the deck at the secondary module and then into the steam turbine inlet. The turbine exhausts to the in-hull condenser through a large distribution plenum.

Condensate is drawn from a point near the lower extremity of the condenser and into the inlet of one of the two boiler feed pumps. From the boiler feed pump the water flows through the demineralizer and air ejector after-condenser and then up into the steam generator inlet.

There are several reasons for locating the feed pump and some of the other secondary equipment on the primary module. This location allows a maximum positive suction head to be available to the feed pump. In addition, it minimizes the length of inter-connecting pipe and aids in lowering the center of gravity of the NEPP hull system.

A large amount of effort was expended in obtaining information on the major pieces of equipment for the secondary system. Specific manufacturers and model numbers are referred to in the text as the equipment is described and is considered "reference" equipment. It should be pointed out that this "reference" equipment may change in the final design phase as new information and equipment become available.

In some areas, such as valves and instrumentation, the selection of specific pieces of equipment was not reported. Since there are a number of vendors available and the availability of this equipment varies with time, this detail was not appropriate in the present phase. All equipment selected for the "reference" design is off-the-shelf. For most of the equipment mentioned here, there are one or more alternates available.

4.1. TURBINE GENERATOR SYSTEM

4.1.1. Turbine Generator and Control

A number of alternate turbine generator systems were considered for the TOPS/MUS application. These systems ranged from low-speed, direct-drive units producing 60 Hz power to ultra-high speed units. Production of both 60 and 400 Hz power were considered in the study. However, 60 Hz power was ultimately selected for the reference system because it would provide a minimum amount of difficulty in the selection of equipment, particularly for the experimenters who will use the MUS habitat.

The 400-Hz system, however, does have some definite advantages. These will be discussed in detail in Section 11 of this report.

Table 4-1 gives a list of turbine manufacturing companies that were contacted and were able to respond in all,or in part,to the requirements for the TOPS/MUS power plant system. Considering the information received from each of these turbine suppliers, and in the context that the power plant should use entirely off-the-shelf equipment, the turbine generator system provided by Elliott was selected for the reference design. Special consideration is also given to the turbine generator system which would be supplied by Sundstrand Aviation because of its high efficiency, compact configuration, and light weight; this particular turbine generator system will be discussed separately in Section 11, as it could provide the basis for designing an extremely compact version of the TOPS system in the power range up to 125 kw(e).

The difficulty in obtaining turbine equipment for the power range of interest (125 kw(e)) is not one of the availability of equipment but rather the efficiency of the equipment available. Most of the turbines in this range are either designed to drive auxiliary mechanical equipment such as pumps or blowers in large power or process plants, or they are designed to put out much higher powers and must be derated by decreasing the arc of steam admission in order to perform adequately at the lower powers.

In the first case, efficiency is of little basic importance in the design of the equipment since the energy is derived from essentially waste steam, and for the most part exhaust is meant to be atmospheric pressure. In the

Table 4-1

LIST OF RESPONDING TURBINE
MANUFACTURING COMPANIES

Coffin Pump Company
Coppus
Elliott Company
General Electric Corporation
MTI
Pacific Pump Company
Stal Laval Turbine A. B.
(Sweden)
Sundstrand Aviation
Terry Turbine
Westinghouse (Canada)
Worthington Corporation

second case, much of the basic efficiency is lost because of steam leakage and windage resulting from a small arc of admission. Another point to consider is that most turbines built for powers less than 1 Mw(e) are single-stage devices, which means that in order to approach the theoretically possible efficiency, one must use a large turbine wheel, go to very high wheel speeds, or both.

Large diameter turbine wheels lose in efficiency because of steam leakage and windage. In addition, they lead to large and heavy pieces of equipment. Alternatively, high-speed equipment usually runs into stress problems.

The only multistage turbine available in the power range of interest is a five-stage unit currently being built by Worthington Corporation for the U. S. Navy. This unit, however, appears to have a lower turbine efficiency at our operating conditions than does the reference unit.

The Sundstrand Aviation turbine-gearbox set is an example of a high-speed, small-diameter wheel. This unit utilizes a 6-in.-diameter turbine wheel operating at 63, 000 rpm. The estimated efficiency of the turbine is over 60%. This unit would be adapted from another usage and is discussed in Section 11. 1.

The reference design is based on the use of an Elliott-type CYR turbine. The unit which is shown in Fig. 4. 1 has a single-stage, 22-in. diameter, two-row Curtiss wheel.

The turbine will be operated at 6300 rpm with speed control provided by a type PG-PH Woodward oil relay governor. The unit, shown in Fig. 4. 2, will maintain a shaft speed consistent with NEMA Class D requirements.

Other features of the turbine include a speed changer on the governor for generator synchronization, a mechanical overspeed trip, and a high back pressure trip. The high back pressure trip, (shown in Fig. 4. 3) is used to protect the condenser from becoming overpressurized. It acts as a secondary safety backup to a standard spring-actuated relief valve located in the exhaust line between the turbine and the condenser. The entire turbine will be lagged with insulation to prevent excessive heat loss to the surrounding atmosphere. The basic weight of the type CYR turbine is 1750 lbs.

The turbine will drive an Elliott-type C1 gear unit. This unit is typical of a number of units used with steam turbines. Both the pinion and gear wheel use double helical gear teeth to eliminate end thrust. The

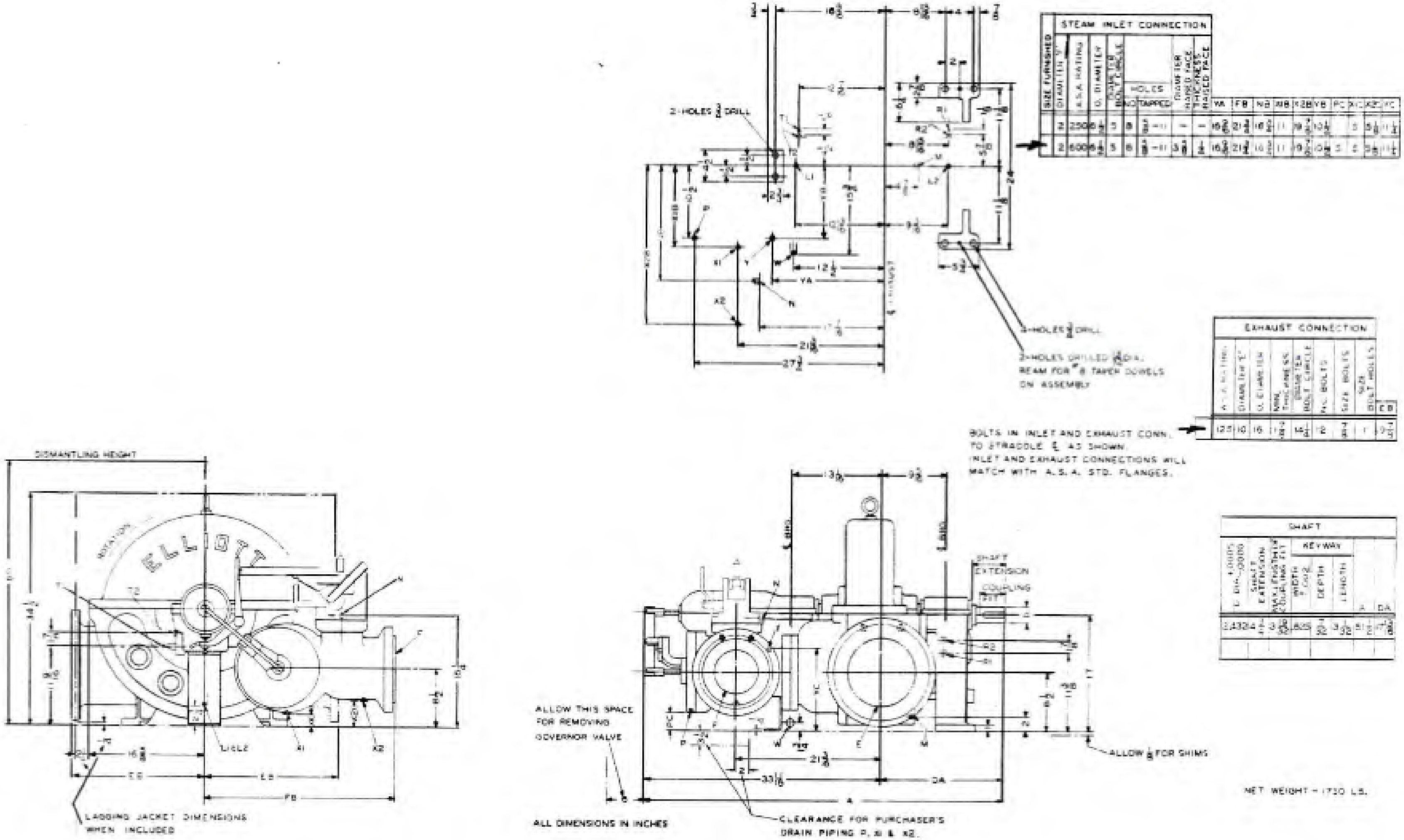


Fig. 4.1--Elliott CYR turbine

PARTS LIST

567	Valve Seat
572	Cover Assembly (LP)
573	Packing
574	Follower
585	Steam Chest Body
586	Bushing
587	Block
588	Bushing
651	Valve
652	Stem
653	Pin
654	Connection
705	Woodruff Key
706	Woodruff Key
707	Coupling Assembly
935	Trip Lever
946	Lock Nut
1001	Housing
1003	Trip Body
1004	Worm
1007	Worm Wheel
1010	Key
1011	Oil Feeder
1012	Gasket
1014	Cotter Pin
1016	Lock Washer
1017	Lock Nut
1018	Machine Bolt
1019	Shim—.005" Thick
1020	Shim—.010" Thick
1021	Shim—.020" Thick
1022	Shim—.030" Thick
1023	Machine Bolt
1024	Cap Screw
1025	Set Screw
1026	Taper Dowel
1036	Washer
1054	Pin
1055	Lever
1056	Connection
1057	Bracket
1058	Bearing
1059	Pin
1060	Set Screw
1061	Spring
1062	Elastic Stop Nut
1063	Taper Dowel
1064	Cap Screw
1065	Lock Washer
1066	Pin
1067	Shim—.031" Thick
1068	Shim—.010" Thick
1069	Shim—.005" Thick
1073	Bracket
1155	Plunger
1156	Gasket
1157	Plug
1492	Oil Pump (With Coupling and Key)

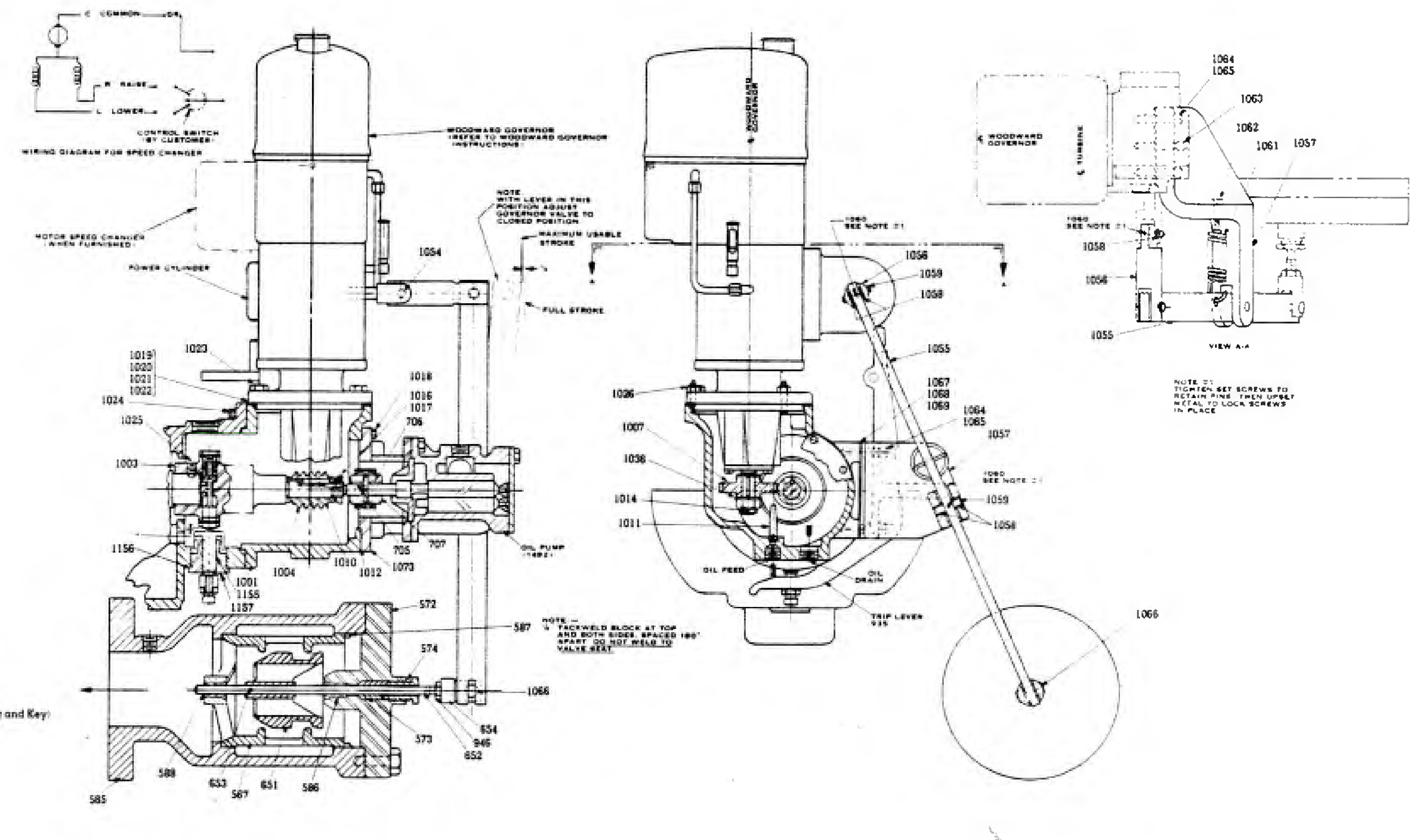
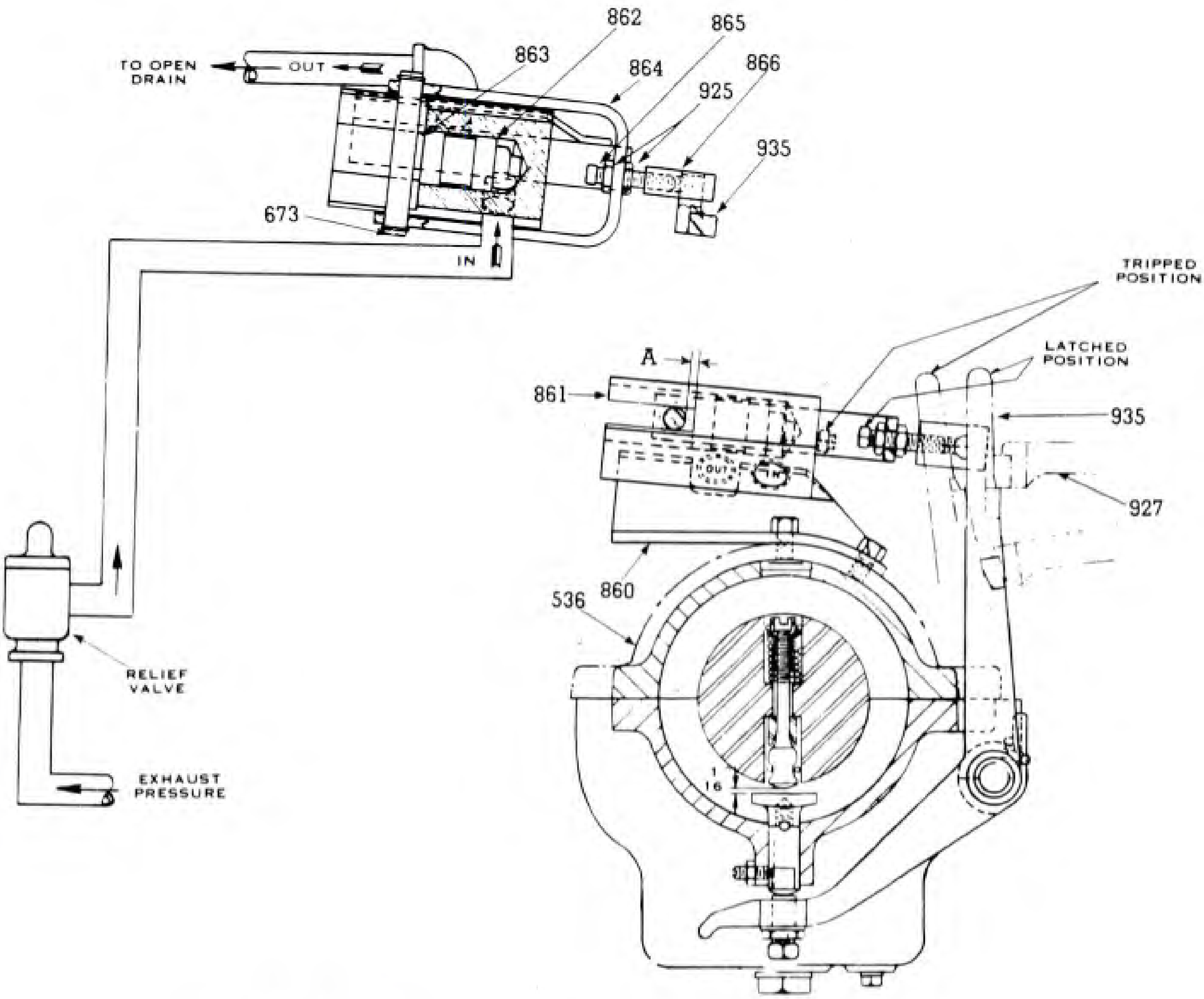


Fig. 4.2--Woodward oil relay governor



(Top view, showing piping detail)

(Front view)

Y-625879
X-616096

PARTS LIST

536	Bearing Case and Cap	864	Yoke
673	Retaining Ring	865	Set Screw
860	Bracket Assembly	866	Link
861	Cylinder	925	Jam Nut
862	Piston	927	Lever (Resetting)
863	Pin	935	Latch

Fig. 4.3--Elliott high back-pressure turbine trip

unit provides 3.5:1 speed reduction for an output speed of 1800 rpm. The weight of the type C-1 gear unit (shown in Fig. 4.4) is 1700 lb.

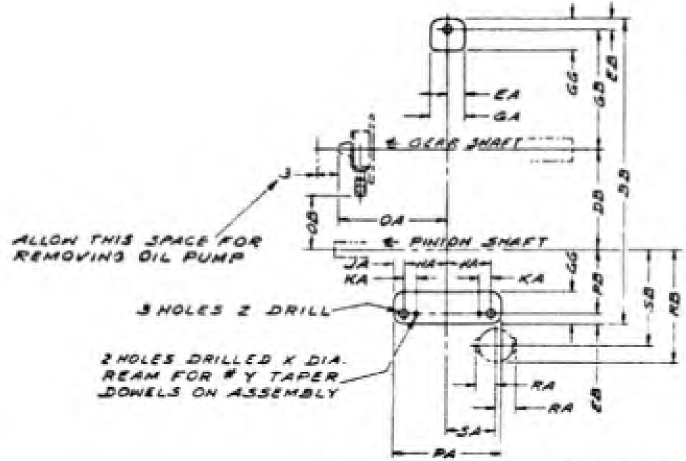
The generator used to complete the turbine generator (TG) set is supplied by the General Electric Company. The unit shown in Fig. 4.5 is a brushless, synchronous, self-excited, two-bearing generator. The selected unit is frame number 445Z which is rated at 125 kw at 1800 rpm. A more complete description of its electrical characteristics is given in Section 4.7.

Since the generator is air-cooled, it will be necessary to duct the exhaust so that the thermal energy picked up in the air as it passes through the generator can be removed by passing the exiting air over finned coils cooled by the fresh-water cooling system. This is done to minimize the temperature rise within the machinery space both for the protection of the equipment and to allow easy access by operating personnel when the plant is not operating on nuclear-generated steam. The General Electric unit weighs 1540 lb.

Figure 2.1 showed the NEPP hull of the MUS with the above-mentioned turbine-generator set installed. It can be seen that the unit fits within the confines of the NEPP hull and allows space for a 30-in. refueling passage to the reactor. There are several other alternates possible using existing equipment available from various manufacturers. Some of these are discussed in Section 11.

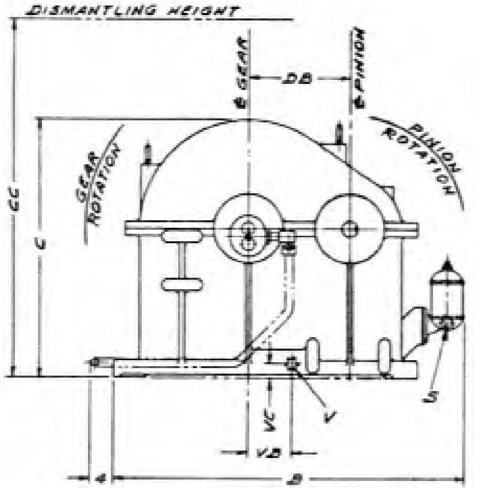
The turbine-generator set will be installed on a common bedplate to maintain proper alignment between units. Installation of the unit will be made prior to the attachment of the upper hemi-head to the NEPP hull. Removal or replacement of the unit once the hemi-head is welded in place will have to be accomplished through the 30-in. access hatch in the top of the hemi-head. Since neither the turbine nor the gear reducer will fit through the hatch, it will be necessary to disassemble each major item in situ, and pass the pieces through the hatch. Removal or replacement of some of the large outer casings for the turbine and gear reducer may require that they be cut and re-welded if replacement becomes necessary. During the normal course of events, the replacement of the casings should not be required over the lifetime of the system.

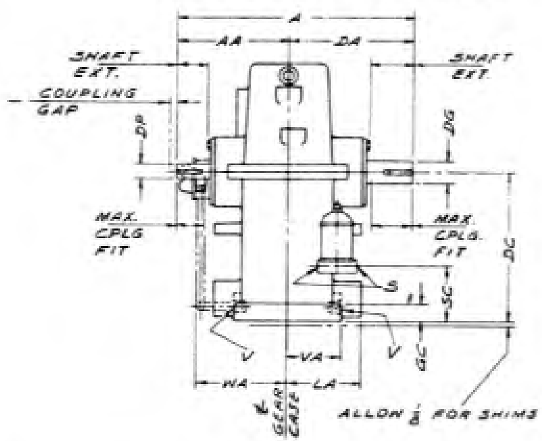
The lubrication system for the turbine and gear reducer is self-contained. A typical arrangement drawing for this system is shown in Fig. 4.6 with a diagrammatic arrangement of the lubrication system (shown in Fig. 4.7).



PLAN VIEW OF CONTACT SURFACES

GEAR		PINION SHAFT					
		DP DIA +.0005 -.0000	SHAFT EXTENSION	MAX LENGTH OF COUPLING FIT	KEYWAY WIDTH ±.002	KEYWAY DEPTH	KEYWAY LENGTH
	C1	1.9320	$3\frac{15}{16}$	$3\frac{1}{4}$	.500	$\frac{3}{16}$	$2\frac{7}{16}$



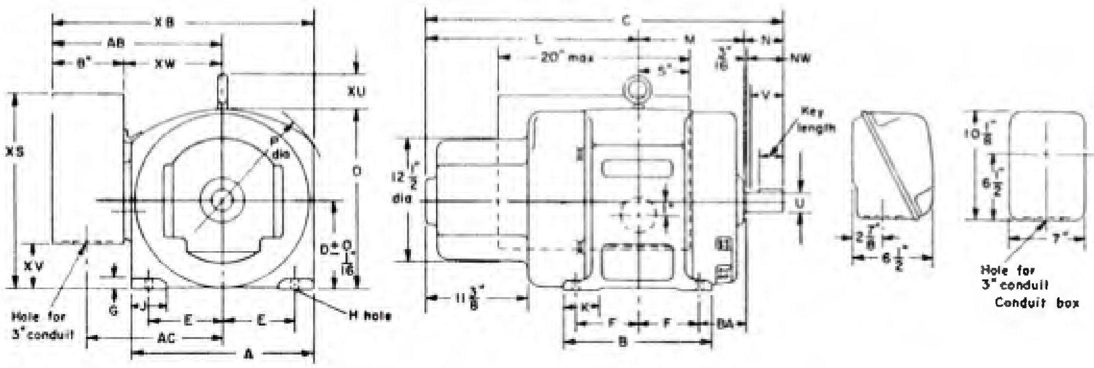


GEAR		GEAR SHAFT					
		DG DIA.	SHAFT EXTENSION	MAX LENGTH OF COUPLING FIT	KEYWAY WIDTH ±.002	KEYWAY DEPTH	KEYWAY LENGTH
	C1	1.9320 +.0006 -.0000	$4\frac{9}{16}$	$3\frac{7}{8}$	.500	$\frac{3}{16}$	$2\frac{7}{16}$

ALL DIMENSIONS IN INCHES.
SHAFT BEARINGS : PRESSURE LUBRICATED

GEAR		A	AA	DA	EA		GA	HA	JA	KA	LA	OA	PA	RA	SA	VA	WA	B	BB	DB	EB	GB	OB	PB	RB	SB	VB		C	CC	DC	GC	SC	VC		GG	S	V	X	Y	Z	NET WT.
	C1	$25\frac{1}{2}$	$12\frac{1}{2}$	13	$2\frac{1}{4}$		$4\frac{1}{2}$	$4\frac{3}{8}$	$1\frac{3}{8}$	$1\frac{5}{8}$	$7\frac{5}{8}$	$12\frac{5}{8}$	$11\frac{1}{2}$	$2\frac{9}{16}$	5	$5\frac{3}{4}$	12	$34\frac{1}{8}$	26	8	$1\frac{3}{4}$	$10\frac{7}{8}$	$4\frac{3}{16}$	$6\frac{3}{8}$	14	$11\frac{3}{4}$	0"		$24\frac{1}{2}$	34	15	$2\frac{1}{4}$	$6\frac{1}{4}$	2		$4\frac{1}{2}$	1	$1\frac{1}{2}$	$\frac{1}{2}$	#9	1	1700#

Fig. 4.4--Elliott C-1 gear unit



Frame	Approximate Net Wt in Lb of Generator and Regulator	Dimensions in Inches																
		KEYWAY		Key Length	MOUNTING					SHAFT				A	B	C	D *	J
		Width	Depth		E	F	G	H	BA	N	N-W	U □	V †					
364Z	850	1/2	1/4	2 3/4	7	5 5/8	1	21/32	5 7/8	4 7/16	4 1/4	2 1/8	4	17 7/8	13 3/4	36 1/8	9	3 1/4
364U	850	1/2	1/4	5	7	5 5/8	1	21/32	5 7/8	6 9/16	6 3/8	2 1/8	6 3/8	17 7/8	13 3/4	38 1/4	9	3 1/4
365Z	900	1/2	1/4	2 3/4	7	6 1/8	1	21/32	5 7/8	4 7/16	4 1/4	2 1/8	4	17 7/8	14 3/4	37 1/8	9	3 1/4
365U	900	1/2	1/4	5	7	6 1/8	1	21/32	5 7/8	6 9/16	6 3/8	2 1/8	6 3/8	17 7/8	14 3/4	39 1/4	9	3 1/4
404Z	1050	5/8	5/16	3 1/4	8	6 1/8	1 1/4	13/16	6 5/8	4 15/16	4 3/4	2 3/8	4 1/2	19 7/8	15 1/4	38 3/8	10	3 1/2
404U	1050	5/8	5/16	5 1/2	8	6 1/8	1 1/4	13/16	6 5/8	7 5/16	7 1/8	2 3/8	6 7/8	19 7/8	15 1/4	41 1/4	10	3 1/2
405Z	1140	5/8	5/16	3 1/4	8	6 7/8	1 1/4	13/16	6 5/8	4 15/16	4 3/4	2 3/8	4 1/2	19 7/8	16 3/4	40 3/8	10	3 1/2
405U	1140	5/8	5/16	5 1/2	8	6 7/8	1 1/4	13/16	6 5/8	7 5/16	7 1/8	2 3/8	6 7/8	19 7/8	16 3/4	42 3/4	10	3 1/2
444Z	1340	3/4	3/8	4 1/8	9	7 1/4	1 1/2	13/16	7 1/2	5 11/16	5 3/4	2 7/8	5 1/2	21 7/8	17 3/4	43 3/16	11	3 7/8
444U	1340	3/4	3/8	7	9	7 1/4	1 1/2	13/16	7 1/2	8 13/16	8 3/8	2 7/8	8 3/8	21 7/8	17 3/4	46 1/16	11	3 7/8
445Z	1540	3/4	3/8	4 1/8	9	8 1/4	1 1/2	13/16	7 1/2	5 11/16	5 3/4	2 7/8	5 1/2	21 7/8	19 3/4	45 3/16	11	3 7/8
445U	1540	3/4	3/8	7	9	8 1/4	1 1/2	13/16	7 1/2	8 13/16	8 3/8	2 7/8	8 3/8	21 7/8	19 3/4	48 1/16	11	3 7/8
447Z	1750	3/4	3/8	4 1/8	9	10	1 1/2	13/16	7 1/2	5 11/16	5 3/4	2 7/8	5 1/2	21 7/8	23 1/4	48 11/16	11	3 7/8
447U	1750	3/4	3/8	7	9	10	1 1/2	13/16	7 1/2	8 13/16	8 3/8	2 7/8	8 3/8	21 7/8	23 1/4	51 9/16	11	3 7/8

Frame	Dimensions in Inches											
	K	L	M	O	P	AB	AC	XB	XS	XU	XV	XW
364Z	3 3/8	20 3/8	11 5/16	18 3/16	20	17 5/8	13 5/8	26 5/8	20	2 15/16	5	9 5/8
364U	3 3/8	20 3/8	11 5/16	18 3/16	20	17 5/8	13 5/8	26 5/8	20	2 15/16	5	9 5/8
365Z	3 3/8	20 7/8	11 13/16	18 3/16	20	17 5/8	13 5/8	26 5/8	20	2 15/16	5	9 5/8
365U	3 3/8	20 7/8	11 13/16	18 3/16	20	17 5/8	13 5/8	26 5/8	20	2 15/16	5	9 5/8
404Z	3 3/4	21 3/8	12 9/16	20 3/16	22 1/4	18 3/4	14 3/4	28 3/4	21	3 7/16	6	10 3/4
404U	3 3/4	21 3/8	12 9/16	20 3/16	22 1/4	18 3/4	14 3/4	28 3/4	21	3 7/16	6	10 3/4
405Z	3 3/4	22 1/8	13 5/16	20 3/16	22 1/4	18 3/4	14 3/4	28 3/4	21	3 7/16	6	10 3/4
405U	3 3/4	22 1/8	13 5/16	20 3/16	22 1/4	18 3/4	14 3/4	28 3/4	21	3 7/16	6	10 3/4
444Z	4 1/4	22 11/16	14 9/16	22 1/4	24 1/2	19 7/8	15 7/8	30 7/8	22	3 11/16	7	11 7/8
444U	4 1/4	22 11/16	14 9/16	22 1/4	24 1/2	19 7/8	15 7/8	30 7/8	22	3 11/16	7	11 7/8
445Z	4 1/4	23 11/16	15 9/16	22 1/4	24 1/2	19 7/8	15 7/8	30 7/8	22	3 11/16	7	11 7/8
445U	4 1/4	23 11/16	15 9/16	22 1/4	24 1/2	19 7/8	15 7/8	30 7/8	22	3 11/16	7	11 7/8
447Z	4 1/4	25 7/16	17 5/16	22 1/4	24 1/2	19 7/8	15 7/8	30 7/8	22	3 11/16	7	11 7/8
447U	4 1/4	25 7/16	17 5/16	22 1/4	24 1/2	19 7/8	15 7/8	30 7/8	22	3 11/16	7	11 7/8

* "D" will not be exceeded. When exact dimension is required, liners up to $\frac{1}{16}$ of an inch may be necessary.

⊕Standard conduit box Fig. 2 should be used when generator is furnished with panel-mount regulator.

† "V" represents length of straight part of shaft.

□Shaft diameters $1\frac{5}{8}$ inches and larger +0.000 inch, −0.001 inch.

Fig. 4.5--General Electric brushless alternator

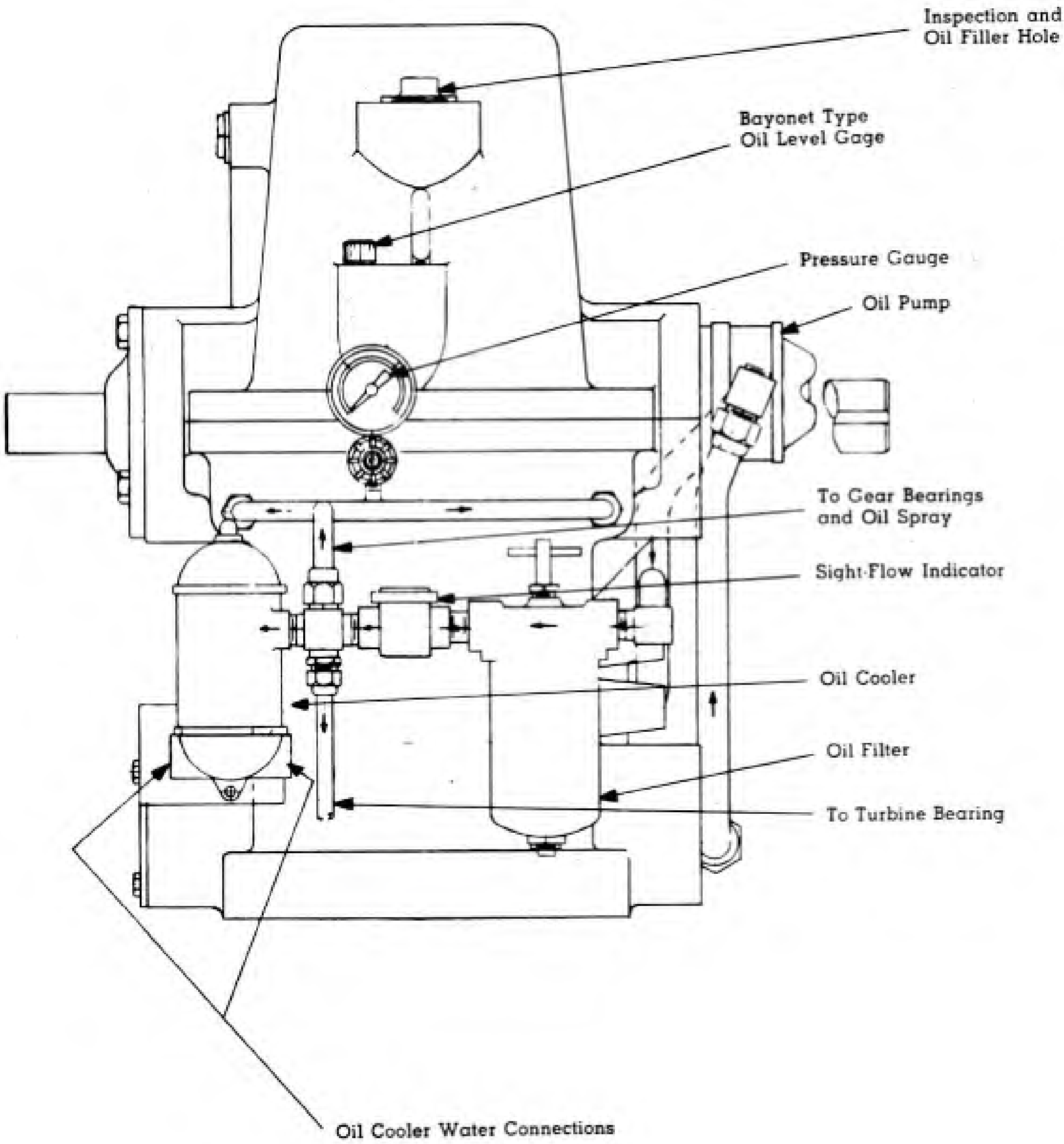


Fig. 4.6(A)--Elliott lubrication arrangement drawing

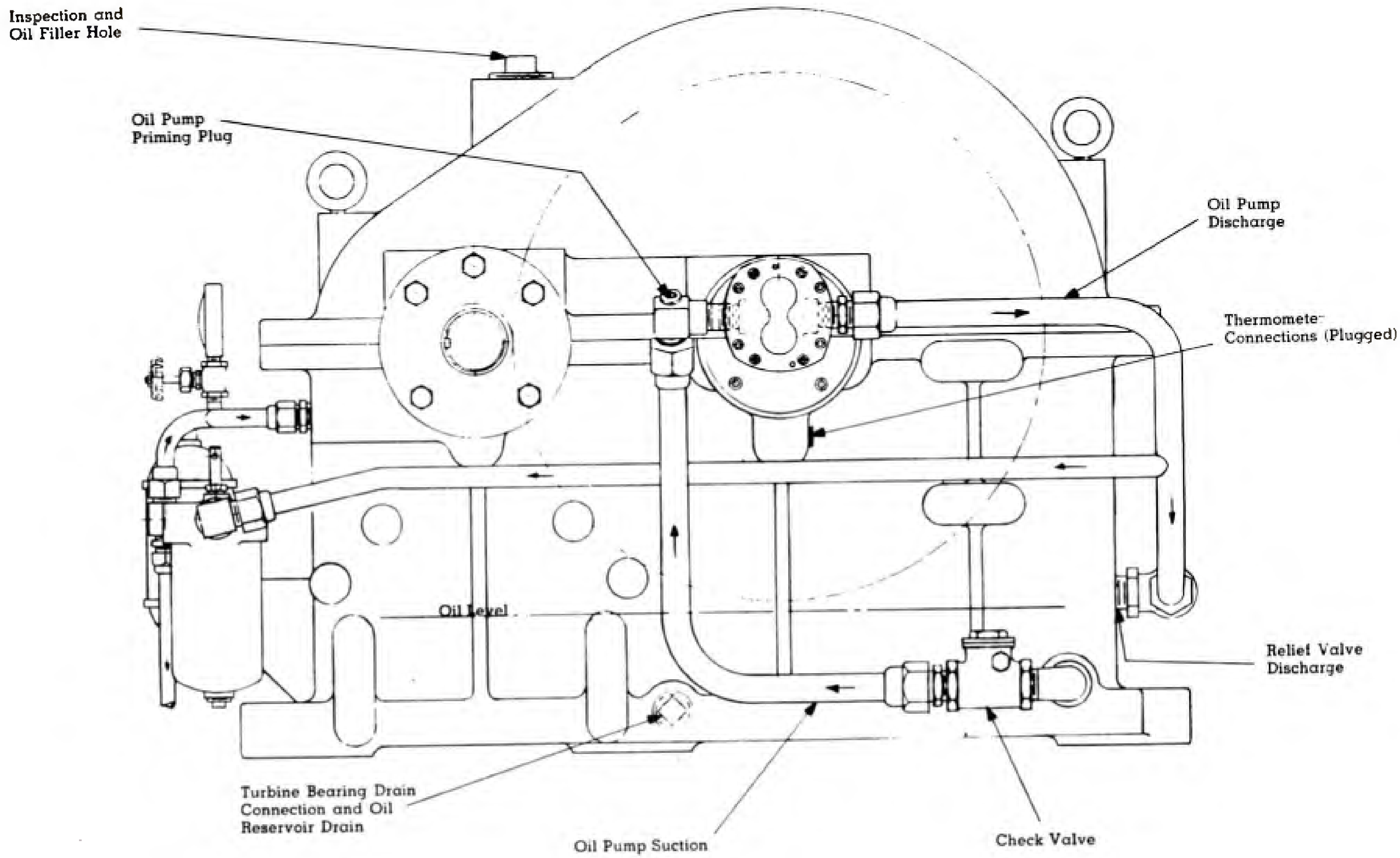


Fig. 4.6(B)--Elliott lubrication arrangement drawing

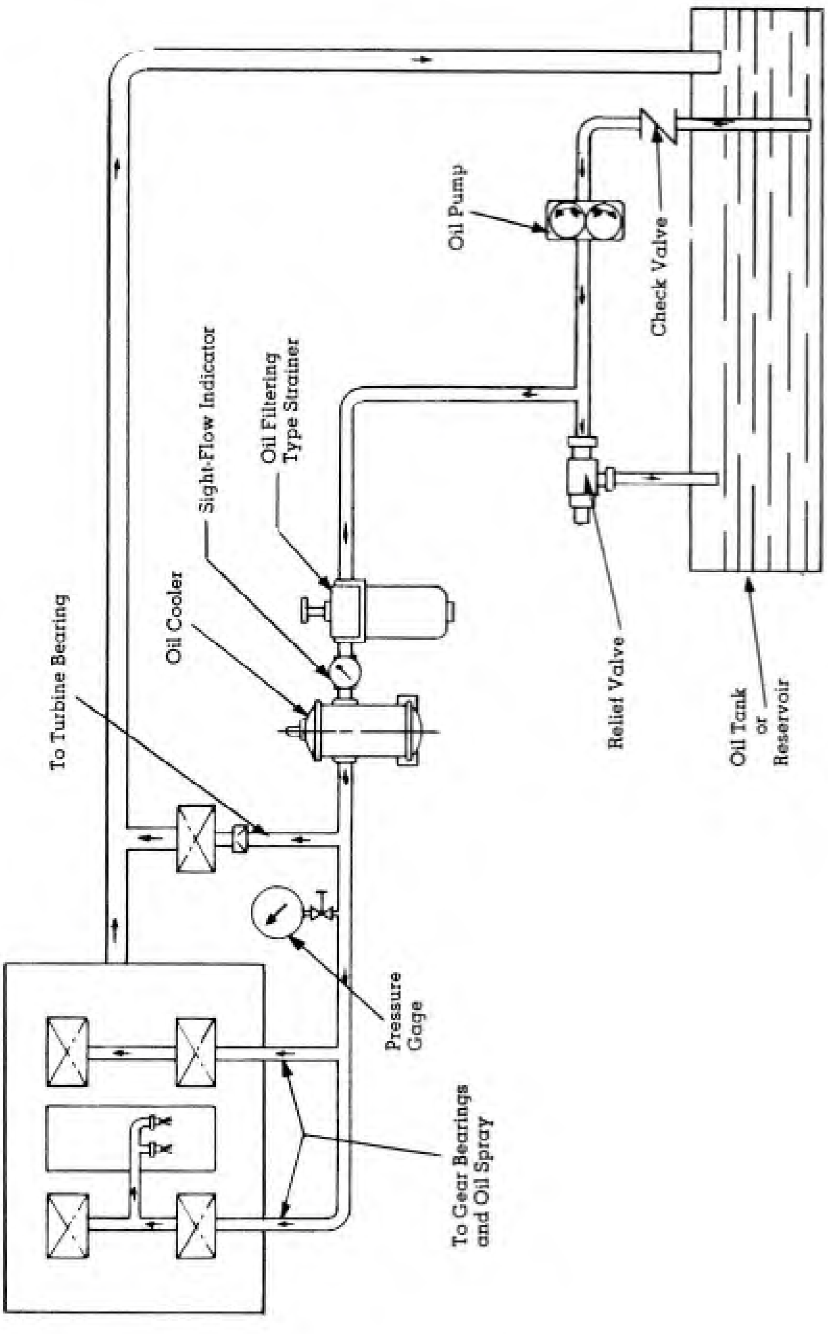


Fig. 4.7--Elliott lubrication system diagram

The oil reservoir for the system will be the gear-reducer case. Sufficient volume is provided in the case for 4.8 gallons of high-grade turbine gear oil.

Both the turbine and gear-reducer bearings are pressure lubricated during normal operation. In addition, these bearings also have a ring-lubrication system which will supply lubricant to the bearings during startup.

The gears in the gear reducer and the governor drive are supplied with spray nozzles which eject lubricant directly at the gear teeth. The lube pump which is driven directly from the low-speed shaft of the gear reducer also supplies the oil for operation of the Woodward governor.

The lubrication system includes both an oil filter and an oil cooler. The oil cooler will be water-cooled with fluid supplied from the fresh-water coolant system (see Section 5.3).

4.2 CONDENSERS

The plant employs a main condenser, which is actually a major portion of the power plant hull surface, and a small after-condenser which is used for condensing the effluent from the main condenser air ejector. These condensers are described in the following subsections.

4.2.1 Main Condenser

A portion of the cylindrical surface of the power plant hull is used as the condensing surface of the main condenser. The condenser was designed to conform to the configuration of the hull as established by GD/EB. The hull is made of HY-130 carbon steel 1-3/4-in.-thick. Stiffening ribs are provided on the outside surface only, and are spaced at regular intervals (as shown in Fig. 2.1). The interior surface of the hull is completely unencumbered; there are no structures attached to the internal condensing surface between the upper and lower support rings. The external surfaces are exposed to the sea and are painted for corrosion protection. The stiffening rings act partially as extended surfaces for added heat transfer which is assumed to compensate for the added flow resistance of the ribs to full convective flow. However, recent progress in optimization of the hull design for maximum structural efficiency has led to a closer spacing of stiffening frames and this tends to limit the heat rejection by natural convection flow of external sea water. This problem is amenable to solution either by modifying the hull framing structure while considering the closely coupled effects upon heat transfer or, alternatively, by the addition of a nonstructural lightweight flow control baffle as shown in Figure 4.8. Analysis shows that adequate horizontal flow would be induced by the "chimney" as shown. It is unlikely that any further external cooling measures would be required but if such were the case the arrangement would be conducive to the addition of a small motor-propeller circulator mounted in the chimney as a potential backup approach.

Turbine exhaust steam and condensate are confined to an annular space 3/4-in.-thick bounded on the outside by the carbon steel hull surface and in the inside by a concentric stainless steel shell 3/32-in.-thick over a height of 25 ft between the upper and lower support rings to which they are welded. The lower end is corrugated to act as a bellows to accommodate differential axial expansion. The lower two feet of the enclosed annular space acts as an integral hot well to provide a gravity head to satisfy the net positive suction head (NPSH) minimum requirements for the feedwater pumps. The hot well also acts to provide subcooling of the condensate which acts to significantly increase the NPSH margin. The effective condenser surface area is 903 square ft. The shell is insulated on the inside surface to minimize heat transfer into the air space. A pressure relief vent, which opens when the condenser pressure exceeds ambient air pressure, protects the condenser shell from collapse due to excessive condensing pressure.

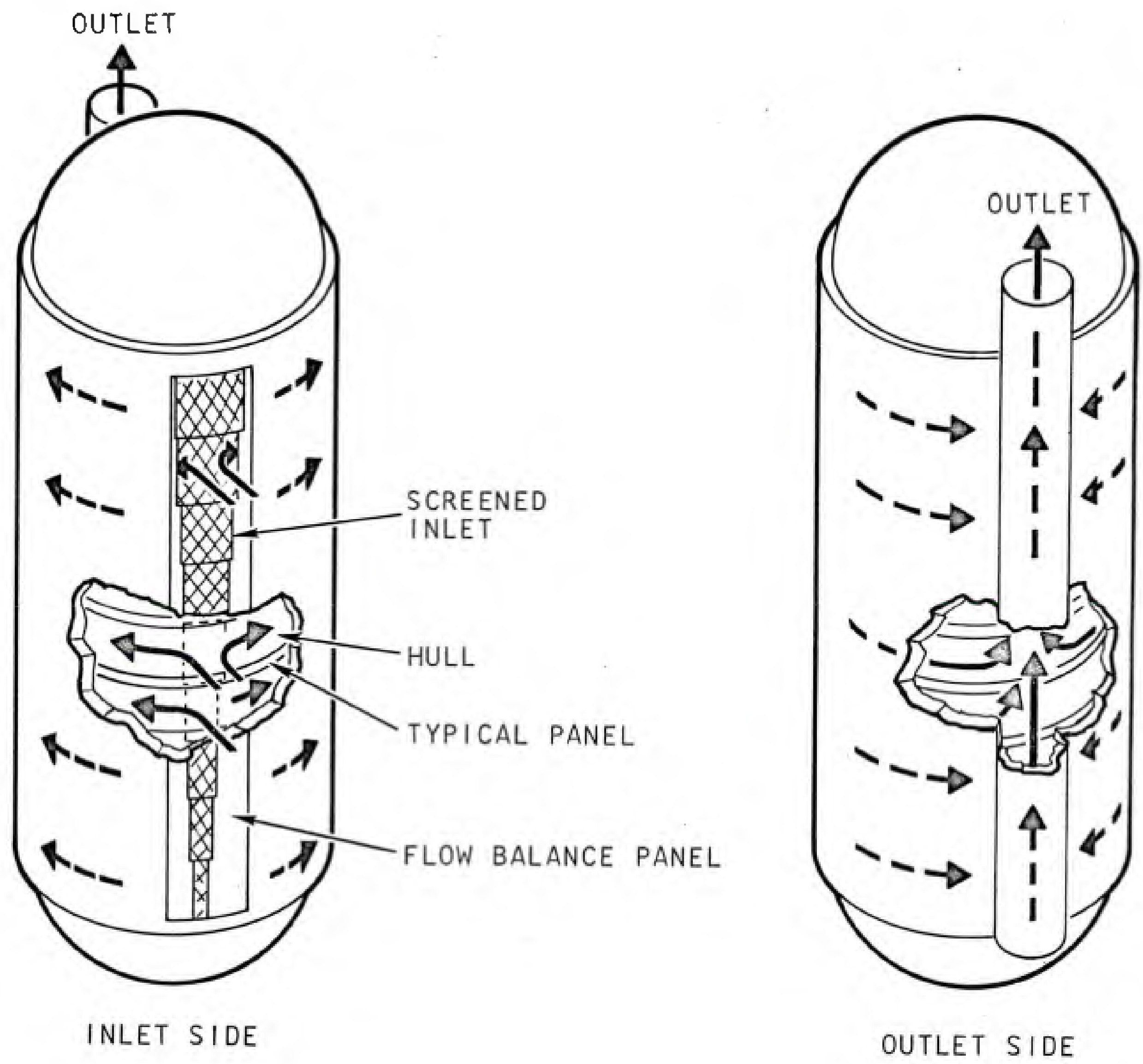


Fig. 4.8--Schematic arrangement for a seawater flow augmenting baffle

The design ambient sea-water temperature at a depth of 6000 ft is taken to be 45°F and the condenser temperature under these conditions is approximately 120°F at a condensing pressure of 1.8 psia, corresponding with a condensing heat flux of 3100 Btu/ft^2-hr at full load.

The heat transfer coefficient from the hull to the sea water was taken to be $0.17\,k\,(g\beta\,\rho\,r\,\delta\,T/V^2)^{1/3}$ as given in NCEL report AD-626185.(1)* Other heat transfer coefficients are as follows:

$h_{condensing\ steam}$	1000 Btu/ft^2-hr-°F
h_{hull} (k - Btu/hr ft-°F)	125 Btu/ft^2-hr-°F
h_{paint}, outside	500 Btu/ft^2-hr-°F
$h_{fouling}$, outside	500 Btu/ft^2-hr-°F
$h_{fouling}$, inside	1000 Btu/ft^2-hr-°F
Combined U (excluding sea film)	72 Btu/ft^2-hr-°F

The effects of these heat transfer coefficients on condensing temperature are given in Fig. 4.8(A). For operation at the surface where sea water temperatures are higher and where part of the hull may be exposed to air, the maximum net power that can be provided by the system may be obtained from Fig. 4.8(B).

4.2.2 Air Ejector and After-Condenser

In the evaluation of the condenser air ejector system, both mechanically-driven and steam-jet air ejectors were considered. The two mechanical systems considered used electric motor-driven pumps. One was a system using a turbine vacuum pump manufactured by Nash Engineering Co., and the other a turbine pump manufactured by SIHI Pump Co. Both of these pumps are of the water-seal variety.

The steam-jet ejectors investigated were all either one- or two-stage devices. There are a number of these which are essentially functionally identical. The only problem encountered was in selecting a unit which the vendor would fabricate from materials which are compatible with the system and whose performance most closely matched the system requirements.

The steam-jet ejector system was selected for the reference design for several reasons. The steam system is lower weight, lower cost, more reliable, uses less complicated controls, and improves the overall system efficiency over a comparable mechanical system. In

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*References are listed at end of this section.

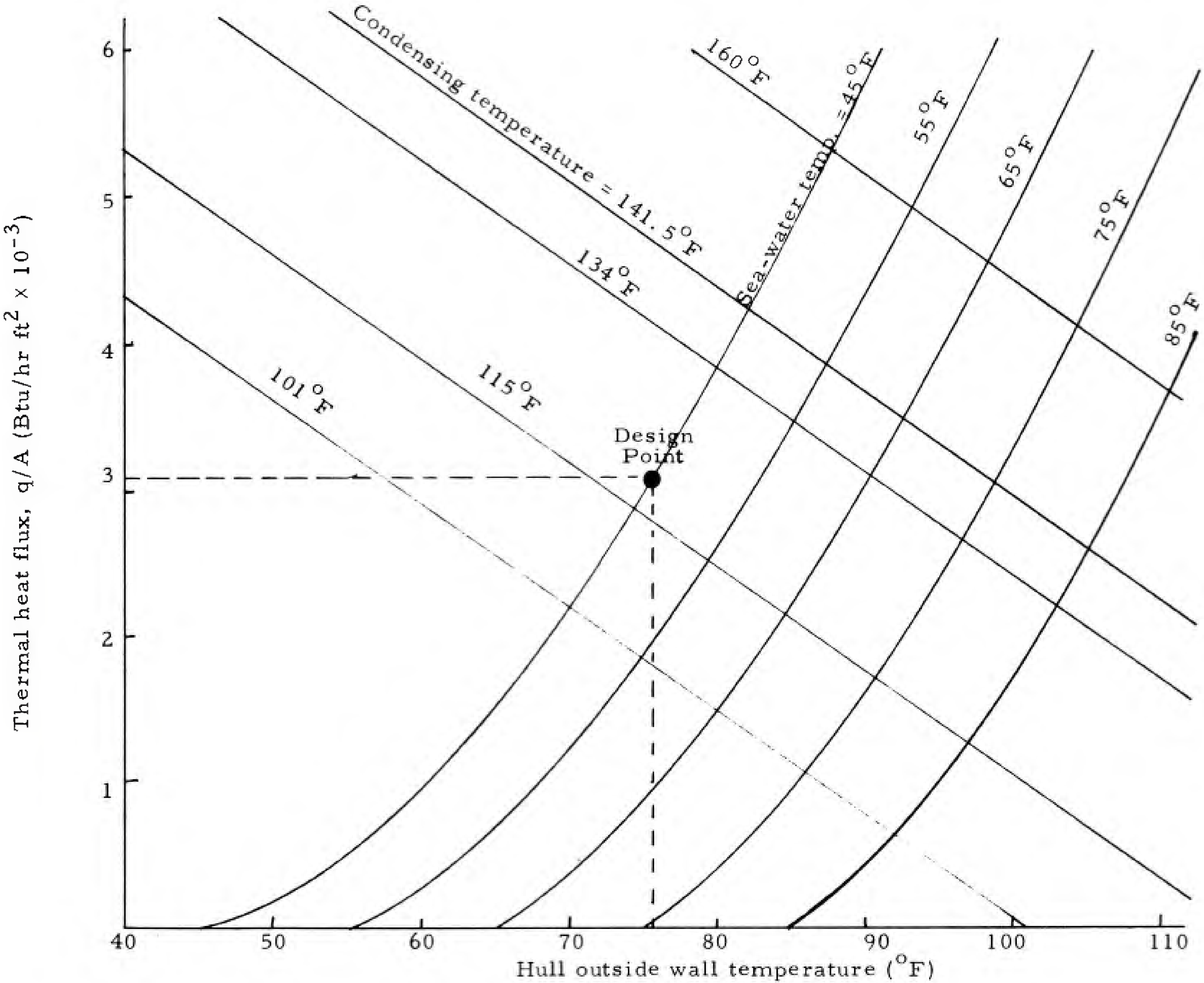


Fig. 4.8(A)--Hull condenser heat transfer

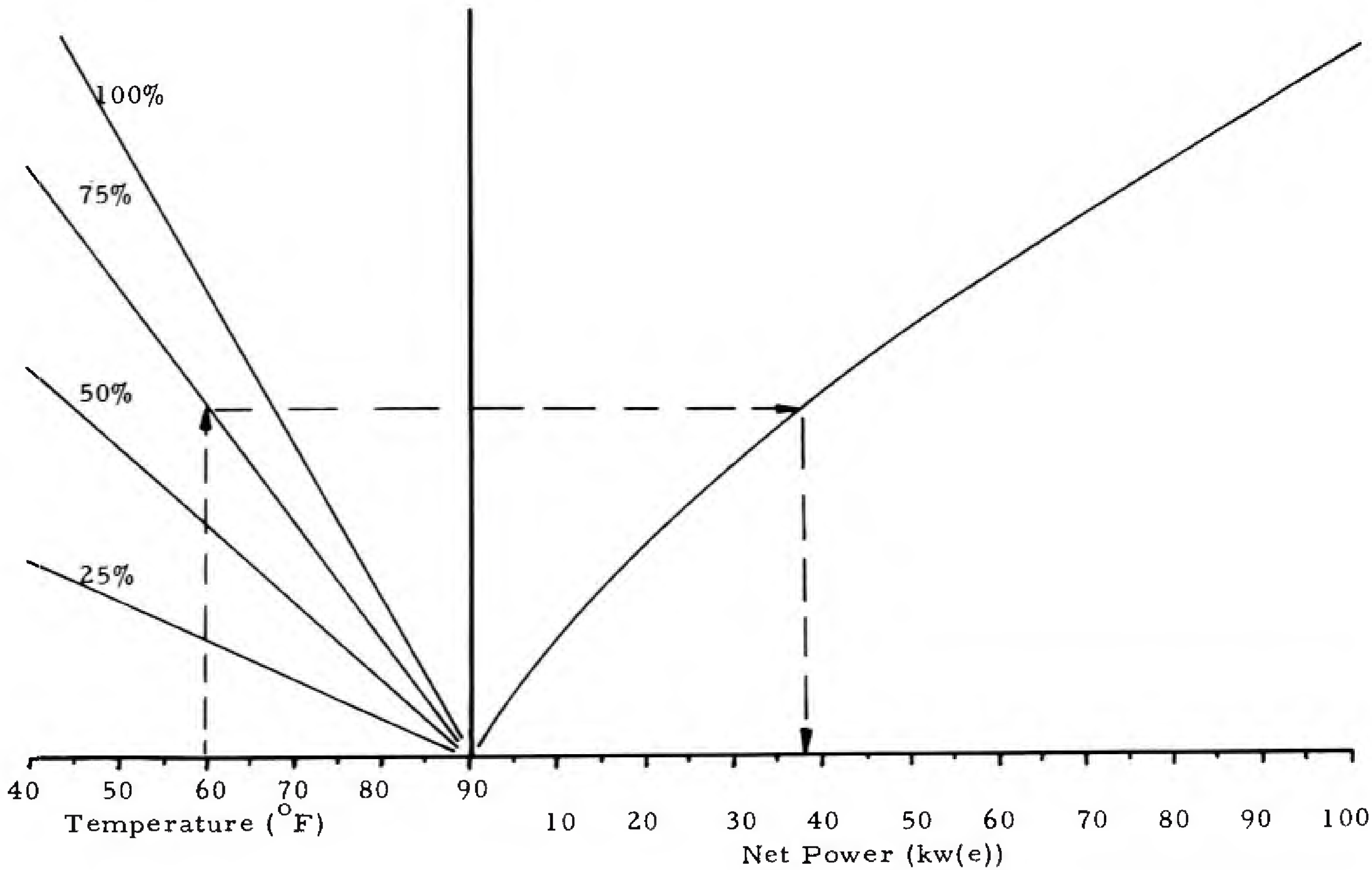


Fig. 4.8(B)--Net power out as a function of seawater temperature and % condenser submergence

addition, it is very easy to completely separate the boiler feedwater from the fresh water cooling system with the steam-jet ejector.

A schematic of the condenser air ejector subsystem is given in Fig. 4.9. The energy is derived from a constant flow of steam bled from the main turbine steam line. The flow of steam will be controlled by a fixed orifice which will be sized to deliver approximately 33 lb/hr of steam to the air ejector nozzle.

The air-steam mixture which is withdrawn from the condenser through a precooler is passed over a set of heat exchanger coils cooled by the fresh water cooling system. This condenses a large portion of the steam entrained in the air-vapor mixture in order to increase the pumping efficiency of the steam-jet ejector.

The reference steam-jet ejector selected is supplied by Schutte Koerting (model S-1) as are both the precooler and after-condenser. Since the unit has no moving parts and therefore should be highly reliable, only a single unit is installed. The only part in the unit which may need replacing at infrequent intervals would be the high-pressure nozzle. With a 2-in. exhaust line this unit should be able to pull up to 40 lb/hr of air while maintaining condenser pressure of 2 psia.

The after-condenser is a tube-in-shell heat exchanger. Primary coolant from the boiler feed pump is supplied to condense the steam in the mixture leaving the steam-jet ejector exhaust. The available energy in the mixture is thus transferred to the boiler feedwater. As a result there is a net gain in over-all system performance compared to a mechanical system which cannot use cooling water with temperature as high as that of the feedwater.

All non-condensible vapors evolved in the process are vented to the atmosphere of the NEPP hull. This vent is connected to the air cooling system which is discussed in Section 5.3.2.

The entire air-ejector system will be fabricated from 300-series stainless steel with the exception of the high pressure nozzle in the steam-jet ejector. This will be stellite in order to minimize erosion. The weight of the entire unit will be 400 lbs. An outline drawing of the unit is shown in Fig. 4.10.

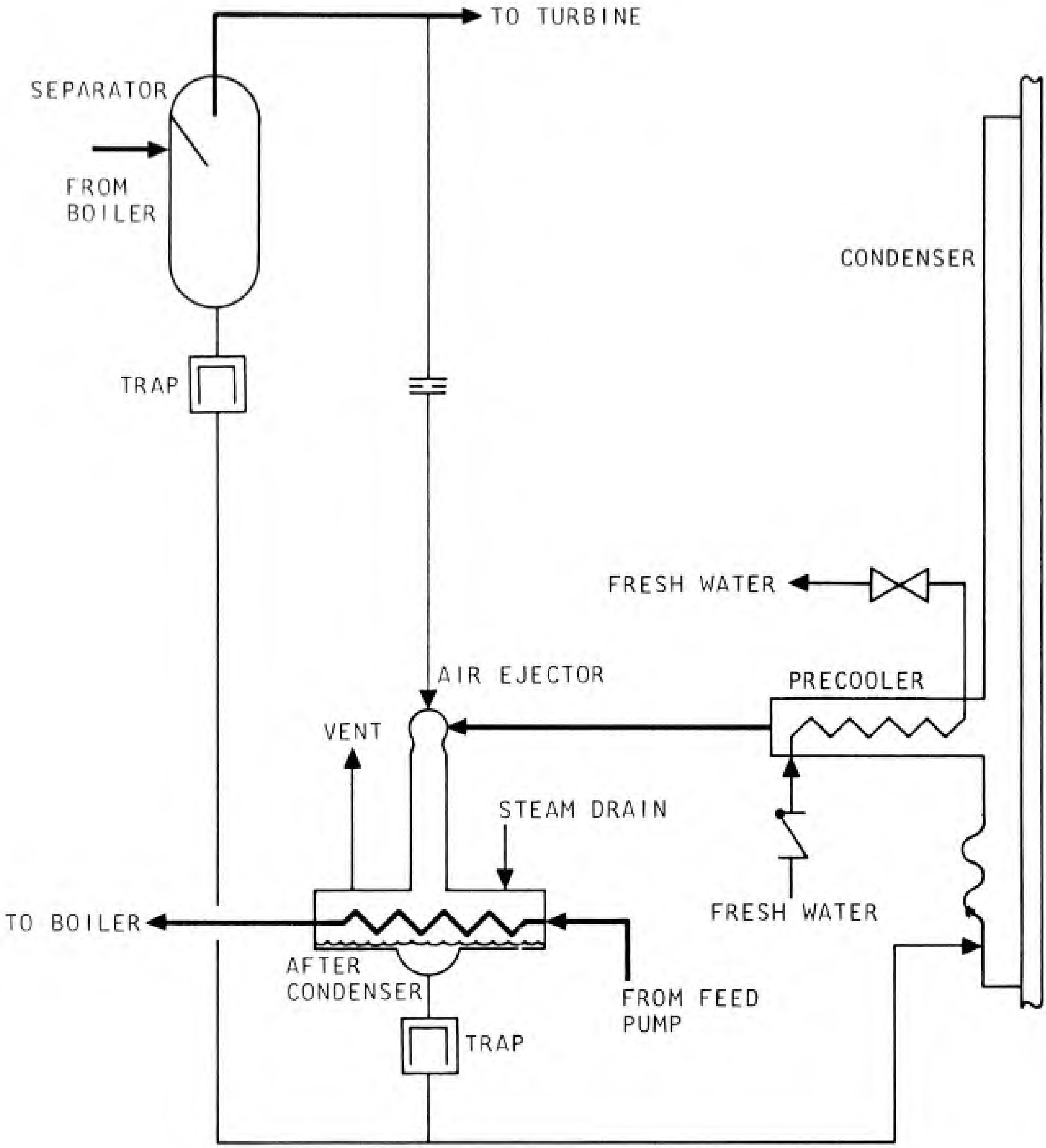


Fig. 4.9--Air ejector system schematic

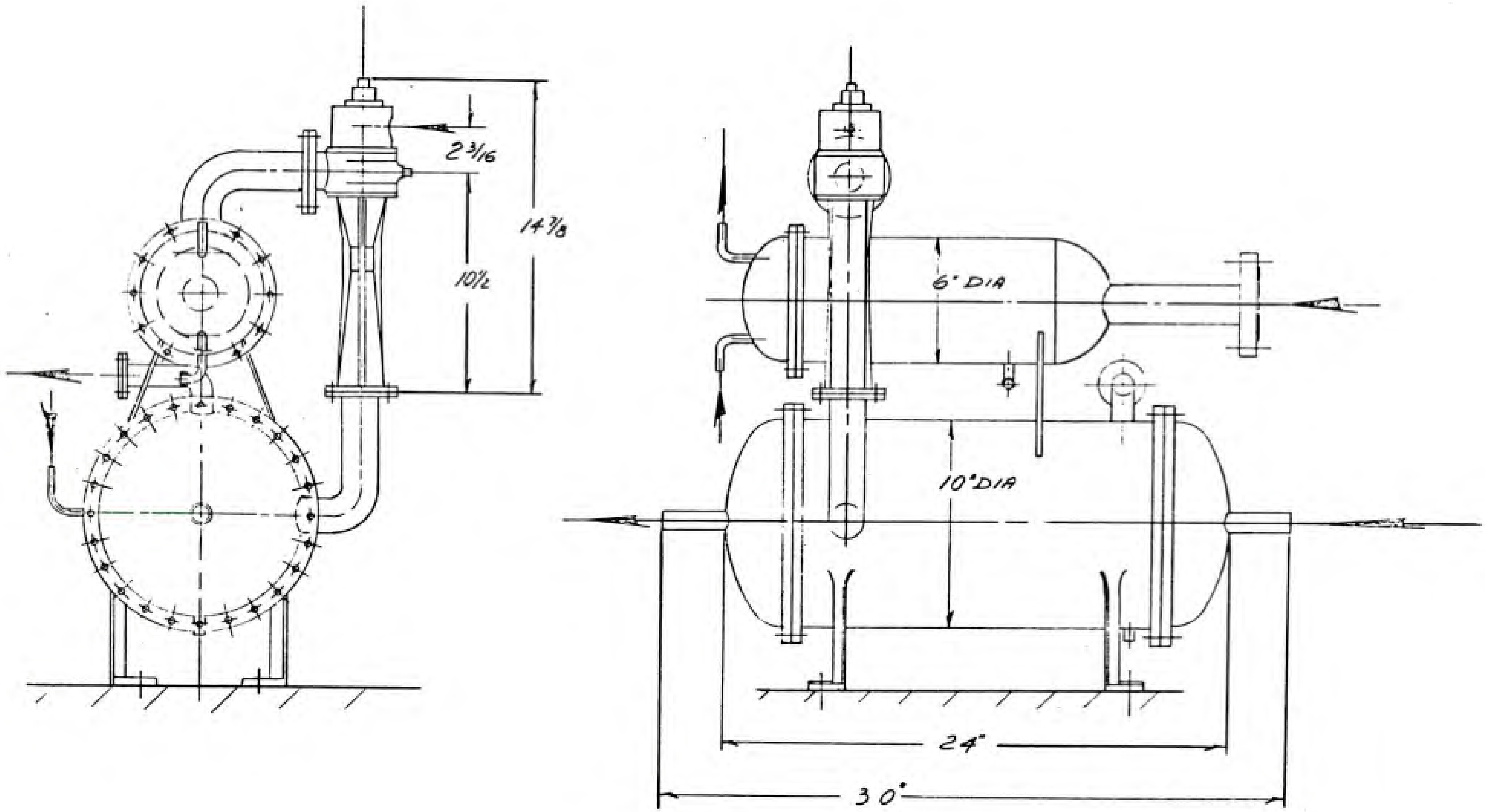


Fig. 4.10--Schutte-Koerting air ejector and after-condenser

4.3 BOILER FEED PUMPS

The boiler feed pump selected for the reference design is an SIHI Model 3105. The pump will be of all stainless steel construction in order to maintain the water purity requirements of the secondary system, which are discussed in Section 4.4.1. The motor is driven at 1750 rpm by a 10 hp, 240 V, 60 Hz 3Φ motor supplied by General Electric Corporation.

Several alternate pumps are also available which have the desired performance characteristics.

Pump characteristic curves for the SIHI 3105 are shown in Fig. 4.11. It will be noted that this particular pump has an almost linear relationship between available head and flow and power and flow. This type of pump curve is characteristic of regenerative turbine pumps. These characteristics lead to very stable system performance.

One minor drawback of a regenerative turbine pump is that it is relatively inefficient at low flows and high heads. This means that at cut-off flow, the pump consumes maximum power. This disadvantage can be partially overcome, however, by proper system design and operation.

Figure 4.12 shows the portion of the flow diagram that relates most directly to the boiler feed pump subsystem. In the TOPS/MUS power plant, two boiler feed pumps will be provided in a parallel arrangement, one for normal operation and one for standby. The electrical system has been set up so that only one pump operates at a time. In the event of a pump failure, the loss of pressure will be sensed and the standby pump will come on the line automatically.

Under normal operating conditions the diverter valve, V4, will be set to provide flow directly to the steam generator. When operating at full load, the flow to the steam generator will be about 7.5 gpm while the head required at the pump will be about 700 ft of fluid. In order to match the pump characteristics to the system and to reduce the auxiliary power requirements at full load, a fixed area orifice sized to bypass approximately 3 gpm at 700 ft differential is provided. This will allow the pump motor power to be reduced from approximately 9.1 hp to 8.3 hp at the 7.5 gpm flow. Since the bypass is through a fixed area orifice, the bypass flow at lower pump flow and higher differential heads will be somewhat higher.

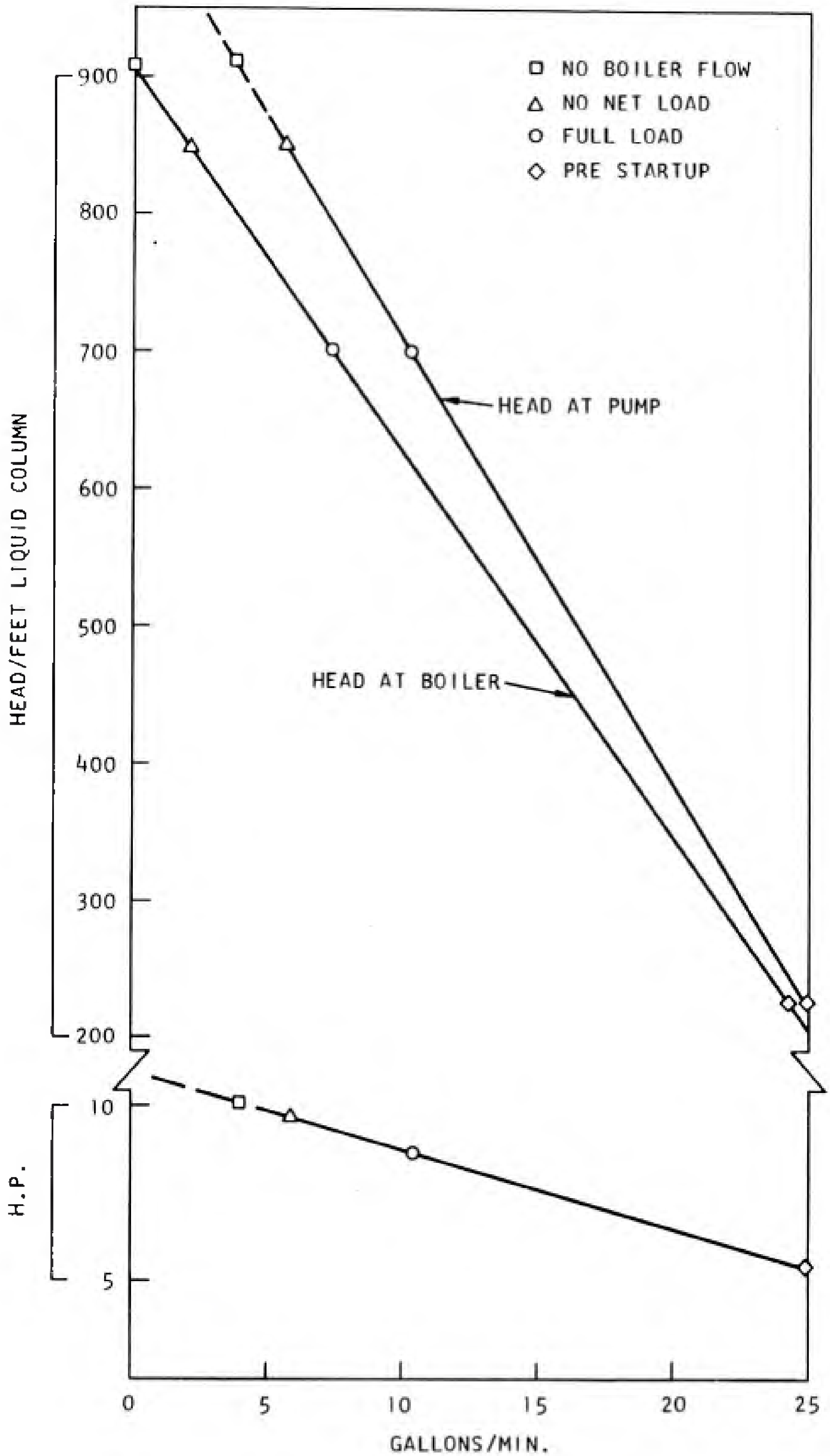


Fig. 4.11--SIHI pump characteristics

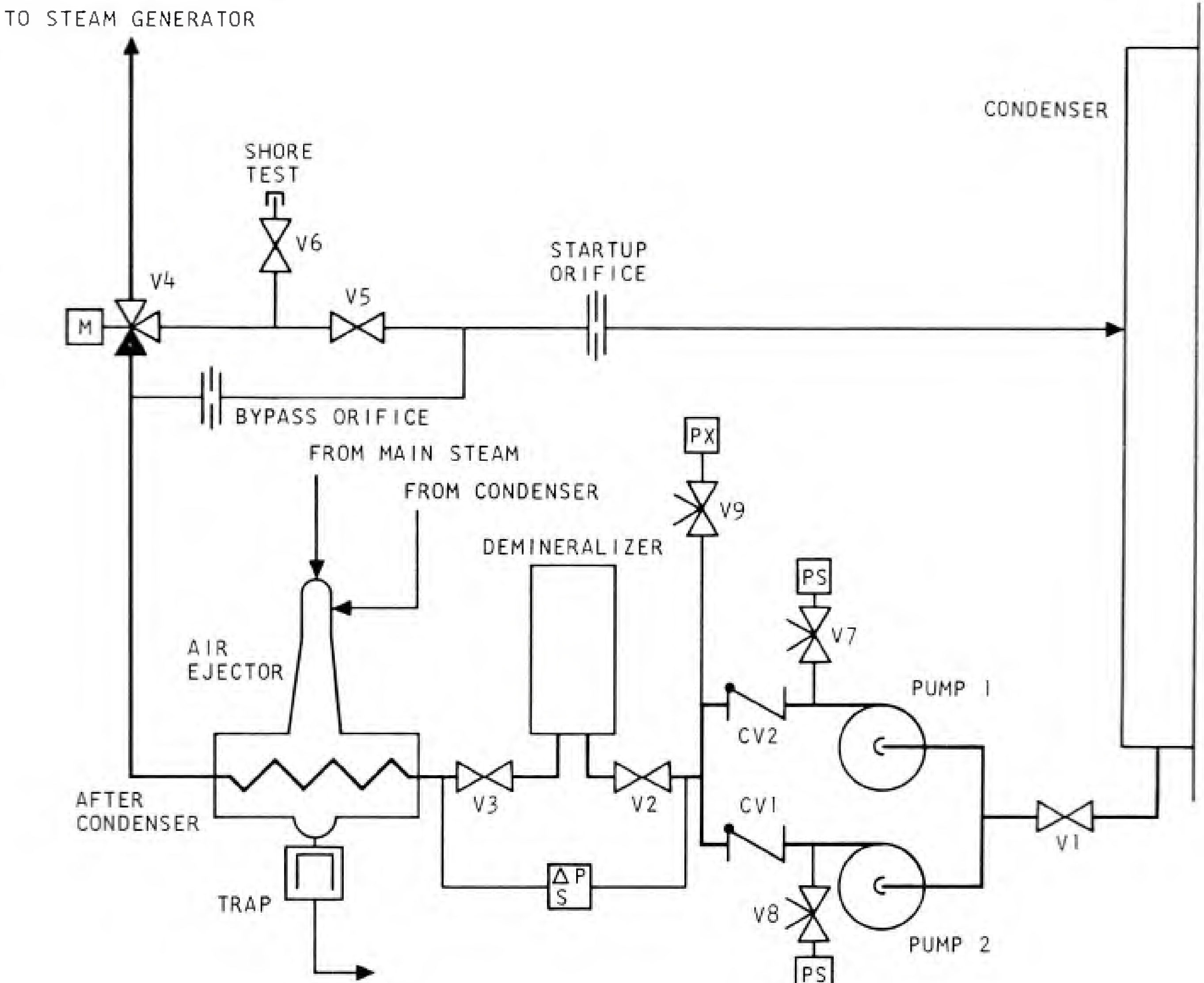


Fig. 4.12--Boiler feedpump subsystem diagram

At cutoff, for instance, the bypass flow will be up to about 3.9 gpm and the pump motor output will be up to about 10.1 hp instead of the 11.2 hp which would normally be required at cutoff without bypass flow.

An added advantage of this fixed area bypass is that it will preclude the possibility of pump damage due to overheating. With the bypass there will always be enough flow to keep the pump operating in a safe condition.

During normal system operation where the turbine generator system will be operating between 0 and 110 kw(e), the pump pressure will vary from approximately 850 ft of head at no load to 700 ft at full load. The input power will likewise vary from about 9.7 hp at no load to 8.3 hp at full load.

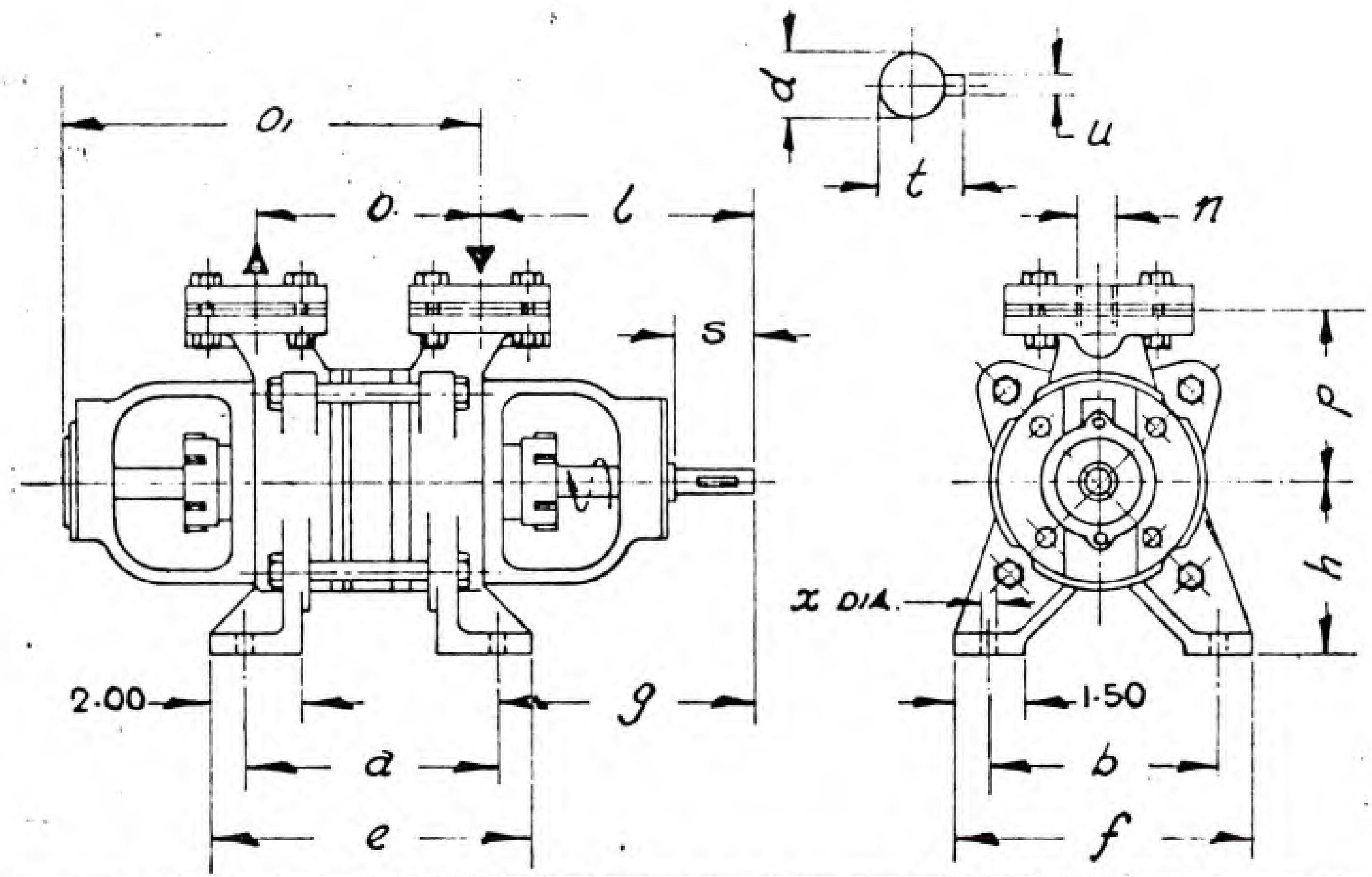
Valve V4 is provided primarily for startup operation. Prior to startup, this valve will be in position to divert the pump flow back to the condenser. The flow rate under these conditions will be about 25 gpm, as set by both the normal system flow resistance and a fixed orifice downstream of V4.

By increasing the pump flow to 25 gpm, which corresponds to a decrease of pump head to 224 ft, the pump power requirement decreases to 4.1 hp. This will decrease power drain and extend the capability of the system to restart on batteries, by about a factor of 2 when compared with the use of a normal shutoff valve.

The value of 25 gpm was selected for the bypass flow to prevent cavitation at the pump inlet. This is necessary because the suction specific speed, and therefore the NPSH requirements of the pump, increase as the flow rate increases. There is not much chance of pump damage even when operating under reasonably high NPSH requirements. However, in order to increase over-all system reliability, the condition is avoided.

The startup and bypass orifices are arranged (as shown in Fig. 4.12) so that there will always be a positive backpressure on the startup valve stem to prevent air leakage into the system at this point.

The general physical arrangement of the SIHI Model 3105 pump is shown in Fig. 4.13.



PUMP	n	a	e	o	o1	l	s	g	b	f	h	p	u	t	d
3105	1-1/4	11.4	13.0	11.2	18.4	8.7	1.5	8.6	5.2	6.7	4.5	5.2	0.25	0.87	0.75

Fig. 4.13--SIHI boiler feedpump

4.4. SECONDARY SYSTEM WATER TREATMENT

4.4.1. Power Conversion Loop Water Treatment

Treatment of the water in the power conversion loop depends on the purity of the water. Protection of the once-through steam generator from deposition of corrosion products is the chief concern in treatment of the TOPS power conversion loop water. All major loop components are composed of corrosion-resistant alloys— Incoloy-800 steam generator, HY-130 condenser, stainless steels— all other components.

It appears that boiler designers make allowances for scale and scale build-up in boiler design on an empirical basis where manufacturers have accumulated proprietary sets of data. (Manufacturers seldom indicate the design basis but only the maximum concentrations of impurities permitted in the water.) Thus previous experience with the water treatment of other systems must be relied upon here.

The low pressure, low temperature once-through boiler design to be employed in TOPS is characteristic of power plant practice about the turn of the century. However, at that time water purity control and specifications were essentially unknown. Thus, there is no direct correlation between the water purity to be maintained in the TOPS system with any existing data. In addition, subcritical once-through boiler data is sparse or nonexistent. Lacking data for subcritical boiler systems, supercritical once-through boiler technology was resorted to as a point of departure in the establishment of the TOPS secondary loop specifications of water purity. Recommended specifications for once-through boilers based upon supercritical operation over a period of more than five years with a large number of sizable power plants is presented in Table 4-2, as taken from Ref. 2. There is little doubt that if this specification were adopted, problems of scale build-up in the boiler, turbine blade deposition, and condenser corrosion would be eliminated for all practical purposes. However, the question is: are these specifications too stringent and do they require unwarranted expenditures and equipment installation for the TOPS system? Before that question could be answered, further consideration on the differences between subcritical and supercritical pressure operation needed to be considered.

The criterion for establishment of water purity for once-through, super-critical boilers is essentially that of elimination of turbine deposition which results in loss of turbine efficiency. In fact, it has been repeatedly

Table 4-2

RECOMMENDED WATER QUALITY LIMITS FOR ONCE-THROUGH BOILERS[a]

Impurity	Maximum concentration (ppb)
Total dissolved solids	50
Silica	20
Iron	10
Copper	2
Nickel	1
Dissolved oxygen	7
Carbon dioxide	0
Organic	0
Hardness	0
pH	8.5 to 9.2

[a] From Reference 2, based on super-critical pressure once-through boiler experience.

stated in the literature that the turbine, then the superheater, then the boiler, in that order of importance, establish the water purity levels required in supercritical boiler operation. The reason for this is that the solubility of important impurities is markedly different at supercritical pressures from that at subcritical pressures.[3] There is a discontinuity in solubility at the boundary of the two regions. Thus, in supercritical boiler operation, these impurities are carried by the steam through the boiler and superheater and are deposited in the turbine as the temperature and pressure in the system fall. While the general notion of differences in solubility between the supercritical and subcritical regions is well established, the amount and precision of the data available are not sufficient to permit any realistic extrapolation of the once-through boiler data to subcritical once-through boiler operation. However, in the following paragraphs an attempt has been made based on the best available information to determine the extent of relaxation in the supercritical pressure system water specifications that can be made for the TOPS application.

Table 4-3 is a preliminary recommendation for the water purity in the secondary loop of the TOPS system. There are two basic reasons for the tentative qualification attached to the specification: (1) the components and materials of construction of the components which will be in contact with the secondary-loop heat transfer fluid have not been sufficiently established to determine the impurities to be contended with and the corrosion rates in the various regions of the loop, and (2) further effort will be required to extrapolate from supercritical once-through boiler application to subcritical boiler application and from subcritical boilers with high recirculation rates to the subcritical once-through boiler operation. That is, in drum-type boilers, there is a recirculation rate of the water in the drum which is many times the steaming rate, while in the once-through boiler, the steaming rate and recirculation rates are identical. Table 4-4 shows typical water characteristics recommended for drum-type boilers at TOPS/MUS conditions.[4]

In the early evolution of supercritical once-through boilers, severe degradation of turbine performance resulted from copper deposition. In these systems, condenser materials were characteristically copper, brass, and monel metal. Good control of dissolved oxygen resulted in markedly decreased copper deposition in turbines of these systems. In later systems, using once-through supercritical boilers, copper-bearing materials were eliminated from the condensers. At lower temperatures and subcritical pressures, copper carry-over is not likely to be a problem for the turbine but is potentially a deposition problem in the boiler, and therefore it is recommended that copper-bearing materials be excluded from the loops of the TOPS system.

Table 4-3

PRELIMINARY TOPS WATER PURITY SPECIFICATION FOR SECONDARY LOOP

Item	Maximum concentration (ppb)
Total dissolved solids	1,000
Chlorides	
(boiler)	500
(steam)	15
NaOH	No excess
Oxygen (feedwater before hydrogen addition)	15
Hydrazine hydrate	3,000
Iron	20
Copper	0
Oxygen (condensate)	40
Hardness (as $CaCO_3$)	20
Morpholine (condensate)	3,000
pH (boiler)	9 to 10
pH (condenser)	9.5 to 10.5

Table 4-4

CHARACTERISTICS OF FILTERED WATER IN BOILER OPERATING AT 500 PSIG[a]

Hardness	$\leq$2 ppm (as $CaCO_3$)
Excess NaOH	$\geq$150 ppm (as $CaCO_3$)
Na (PO_4)	$\geq$50 ppm (as PO_4)
Na_2SO_3)	50 ppm (as Na_2SO_3)
Dissolved solids	$\leq$3,000 ppm
Iron	100 ppb
Oxygen	30 ppb
Silica	1,000 ppb

[a]Taken from Reference 4, at TOPS conditions.

In Reference 5, it is stated that sodium concentrations in steam should be maintained at less than 50 ppb to avoid turbine degradation. Similarly, silica in the range of 5 to 20 ppb concentration in the steam will avoid turbine deposition and degradation. The carry-over of these constituents is a function of pressure. Curves of References 6 through 8 indicate that at the TOPS condition, carryover will be negligible and that the criteria for sodium and silica maximum concentrations in the feed-water should not be based on turbine deposition, but on boiler deposition. Thus, the maximum concentration levels may be increased to a new maximum value, but less than those acceptable in drum-type boilers.

Corrosion of the stainless steel and low-carbon-steel components of the loop can be arrested by proper maintenance of pH. Values of pH of 10 to 11 are desirable for this purpose. In loops containing copper, a lower pH is desirable to prevent corrosion. However, since copper is to be excluded from the TOPS secondary loop, pH values are recommended from 9.5 to 10.5.

Ammonia, morpholine, or cyclohexylamine are used to control the pH of the condensate. Based on PM-3A experience, morpholine is preferred. However, these substances must be volatile to pass through the vapor phase regions of the loop. Their boiling points are reported(4) to be 264° and 273°F, for morpholine and cyclohexylamine, respectively. However, cyclohexylamine is considered more volatile than morpholine since it forms an azeotrope with water that boils at 207°F while morpholine does not. Thus, depending on the temperature in the loop, cyclohexylamine or ammonia might ultimately replace morpholine.

With respect to chloride concentration the boiler composed of Incoloy-800 is immune to chloride(9) and intergranular stress corrosion; however, much of the remainder of the loop is to be composed of stainless steel; thus, the chloride concentration must be controlled. Reference 10 indicates that 500 ppb are acceptable in austentic stainless steel boilers. In the secondary loop of TOPS downstream of the Incoloy boiler, concentrations of 15 ppb should be maintained according to information presented in Reference 10.

Cooling water (sea water) leakage into the condenser, one of the usual criteria in power plant operation for establishing the purity and chemical treatment of water, will not be a consideration in the TOPS system since the pressure hull must resist sea water pressure at depths to 6,000 ft and any degree of condenser leakage would be regarded as a catastrophic failure to the total system.

There has been some question with respect to analysis of the impurity concentrations in the ppb range. Determination in water of chloride nitrite, ferrous, ferric, and ammonium ions, is possible in the range of a few ppb, with a sensitivity of a few ppb. At a concentration of 10 ppb, precision is two to three percent.[11] Further, it has been indicated[12] that silica may be measured to ±5 ppb. The chloride ion is somewhat of a problem with the lowest detectable level of measurement by standard spectrochemical techniques being 20 ppb. Some analysts claim that 10 ppb of chloride is determinable. Precision of chloride ion analysis is estimated to be ±50% near the low end of the measurement range.

Power conversion loop water treatment will start with pretreatment to clean the system and fill it with high-purity water. Continuous treatment will include a full-flow demineralizer to deionize and mechanical departiculate the water; hydrazine hydrate additions to getter O_2; and morpholine additions to neutralize carbonic acid in the condensate and control the pH. All makeup water will be deionized by a second demineralizer in the makeup water line prior to being added to the loop.

The loop demineralizer is sized to accommodate the full feedline flow (approximately 7-1/2 gpm) through a mixed cation-anion resin bed of the disposable cartridge type. Three cubic ft of bed volume, 3 ft deep in the flow direction, should provide greater than 90 days of useful life based on life limited by increasing differential pressure to flow across the demineralizer to 25 psi at the end of life. The demineralizer is designed to operate at full feedline pressure (320 psi maximum) and temperature (120°F), thus obviating temperature and pressure reducing and raising before and after passing through the demineralizer, respectively.

Addition of water treatment chemicals to the system will be made to the condensate. Quantities to be added will be determined during initial plant checkout by evaluating water samples, subsequent to additions, to reduce oxygen concentration to zero and retain 20 to 30 ppb of hydrazine at the steam generator inlet.

4.4.2. Effluent Control

Unlike the primary system of the NEPP, the power conversion loop is not hermetically sealed, and water and steam escaping from the loop must be collected and returned to the system. The smaller the leakage rate and the more efficient the collection system, the smaller the volume and weight of makeup water tankage that is required. As shown in Fig. 7.1, there are several water and steam escape paths from the main loop and recovery paths terminating in either the condenser or the makeup tank.

During startup, particularly, water and steam will be separated in the steam separator. Water separated from the steam is released through a trap (preventing steam flow) connected by a return line to the condenser.

Leakage of water and steam through the turbine packing gland is minimized (or reversed) by maintaining a small differential pressure across the packing. This is accomplished by a pressure-regulated steam supply to an exterior gland housing. Effluent steam from the gland housing then passes to the NEPP compartment's ambient atmosphere via the steam ejector, where some additional water is condensed and heat is returned to the feedwater.

The air-ejector is a steam-driven aspirator sucking air and some steam from the condenser. Steam loss from the condenser is reduced by a freshwater precooled section. Condensate in the precooler returns to the condenser to which it is attached by gravity flow. Downstream of the aspirator, the steam enters an after-condenser where much of its heat content is recovered by heating the feedwater, and its water is condensed. Nevertheless, some vapor is vented to the NEPP compartment's ambient atmosphere. The condensate is released through a trap blocking steam flow to the main condenser.

Steam vented to the NEPP compartment ambient atmosphere would raise the relative humidity until random condensation and uncontrolled dripping of water would result if nothing were done to prevent it. This is avoided by the provision of a dehumidifier, using the upper NEPP hull hemispherical head as a condenser cooled from the outside ambient sea water. Compartment air is blown over the surface of the head through an insulated shroud by one of two identical blowers. One blower is used in normal operation for which it has been adequately sized, while the other is a redundant standby unit which automatically operates on failure of the first. Condensed water vapor is accumulated via gravity flow into drip troughs around the base of the hemispherical head and shroud from which it is piped to the makeup tank. Makeup tank water is demineralized when it is added to the condenser so that any contamination introduced through the dehumidifier will be removed without entering the power conversion loop.

There will undoubtedly be other small leaks which are not readily identifiable in advance of an existing plant. Also, some components will change leakage rates. Catch tanks of small size will be used locally to recover highly contaminated (oily) drippings. Water vapor will be recovered by the dehumidifier as described above. While water recovered in the catch tanks can be returned to the system through the makeup tank, it can only be done during NEPP shutdown. Usually this will occur when an external source of high-purity water will be available instead.

4.4.3 Chemical Cleaning

Water treatment prior to and during normal operation is reviewed in the preceding sections. After a long period of operation there may be some plating of solids on the steam generator tube walls. If plating on tubes takes place to the extent that the reserve heat transfer surface is no longer adequate for full performance, chemical cleaning will restore the boiler to a satisfactory condition. The recommended approach is reviewed in the following discussion.

Water treatment for a once-through steam generator system differs from that which is customary for recirculating boiler systems. By way of comparison, consider first a recirculating boiler. In a recirculating boiler, a steam separator (steam drum) is employed which separates the steam for the turbine from the water which recirculates back to the boiler heat transfer surface. To a first approximation, any solids which are dissolved or entrained in the feedwater do not carry over with the steam but remain behind in the recirculating water and tend to become more concentrated. Normal power plant practice for such systems is to use additives which will keep the solids in suspension as a sludge instead of plating out (scaling) on the boiler heat transfer surfaces. To maintain the content of solids within an acceptable range of concentrations, water is removed from the recirculating loop, either periodically or continuously, in a process called blowdown.

In a once-through steam generator system such as that described here, there is no recirculation of boiler water as liquid. All of the liquid entering is evaporated and leaves as dry steam. Excessive buildup of solids in the boiler is prevented by maintaining the concentration of dissolved solids at a level which is several orders of magnitude less than is required for recirculating boilers. At the same time, only non-solid additives are employed for corrosion control. Typically, gaseous ammonia is used for pH control and hydrazine or other volatile liquids for oxygen scavenging. Loop materials with very low corrosion rates are selected so as to minimize the rate of addition of solids by corrosion into the boiler feedwater. Consequently, in an ideal once-through steam generator system there would be no dissolved solids, and no blowdown would be necessary for the prevention of excessive sludge buildup. Because practical systems do pick up small but finite amounts of solids, some minimal plating does occur. When warranted, once-through boilers are restored to original condition by periodic cleaning by circulation of a dilute acid (e.g., 3% citric acid and formic acid at 150 to 250°F) until the concentration of dissolved solids, principally iron, ceases to increase. The time required depends on the specific situation but is of the order of an hour. Subsequent flushing with a basic solution followed by flushing with pure condensate restores the plant to operating condition. Thus when needed, this chemical cleaning process for the once-through system replaces the blowdown used with recirculating boilers.

4.5. STEAM SEPARATOR

Of the several types of steam separators considered, the one selected as best fulfilling the requirements of the TOPS/MUS system utilizes the centrifugal principle. Several companies fabricate such devices and their operation is very similar; varying, for the most part, only in the details of construction.

The separator selected as the reference unit is a Schutte and Koerting tangential steam separator, size 2-1/2. This unit contains no moving parts and consists basically of an 8-5/8-in.-diameter cylinder approximately 25-in. -high.

Fluid from the once-through steam generator enters the lower portion of the cylinder perpendicular to its centerline and tangential to the cylinder wall. This configuration causes a vortex to form inside of the main cylinder. In normal operation liquid is drawn from the lower periphery while dry steam is withdrawn from the upper central portion of the cylinder at the eye of the vortex.

The water withdrawn from the separator flows through an inverted bucket steam trap (Armstrong Model 5155), back to the main condenser. During startup all of the fluid entering the steam separator will leave through the steam trap system.

The dry steam which remains after the entrained moisture has been removed flows through the main steam line to supply the turbine.

In actual operation the separator should operate dry, since the steam leaving the once-through steam generator should be superheated by 60°F. The separator system is included to prevent any "slugs" of moisture, which might remain in the fluid stream as a result of power transient, from entering the turbine. Such an occurrence could result in damage to the turbine blades. As previously mentioned, the separator also aids in the startup sequence.

4.6 TURBINE PLANT PIPING AND VALVES

The pipe sizes selected for the NEPP were based on the water rates which correspond with plant operation at 100 kw(e) net. The velocities are well within the criteria suggested by Crane(13) for typical steam power plant service.

Referring to Fig. 7.1, the major system nominal line pipe sizes are as follows:

1. From the condenser to the boiler feeder pump inlet, 1 in.;
2. From the pump outlet to the boiler inlet, 1/2 in.;
3. From the boiler outlet to the steam stop valve, 1-1/2 in.;
4. From the steam stop valve to the turbine inlet, 2 in.; and
5. From the turbine outlet to the condenser, 10 in.

All piping in the system will be 300-series stainless steel in keeping with the water purity requirements. All piping connections 1/2-in. or larger will use either welded or flanged connections, while some of the smaller lines will be either screwed or welded.

Particular attention will be given to pipe stresses during the final design stage. This will be of greatest importance in the design of the piping runs between the reactor and the steam turbine since the greatest degree of motion will occur in this section. The placement of the equipment, as shown in the NEPP installation drawing, allows piping flexibility in every plane, as defined in NEMA SM 20-1958.*

All valves chosen for the system are standard, commercially available items. A list of these valves and their service is given in Table 4-5, along with the reference supplies where specific selections have been made.

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*NEMA SM 20-1958, "Standard Publication for Mechanical Drive Steam Turbine".

Table 4-5

LIST OF SECONDARY SYSTEM VALVES, REGULATORS AND CHECKS

Valve No.	Maximum temperature (°F)	Maximum pressure (psia)	Minimum pressure (psia)	Line size (pipe)	Type service	Operation	Remarks	Manufacturer	Model No.
1	220	14.7	1	3/4	Water shut off	Manual	Suction valve minimum pressure drop	Jamesbury	1-D15OF 36M
2	220	350	14.7	1/2	Water shut off	Manual		Jamesbury	1/2 DZF 30S 36M
3	220	350	14.7	1/2	Water shut off	Manual		Jamesbury	1/2 DZF 30S36M
4	250	350	14.7	1/2	Water diverter	Motor	2-minute close limit switches 3-way	Honeywell	1617
5	250	350	1	1/2	Water shut off	Manual		Jamesbury	1/2 DZF 30S36M
6	250	350	1	1/2	Water shut off	Manual		Jamesbury	[illegible] DZF 30S36M
7	220	350	14.7	1/4	Water	Manual	Instrument calibration	Hoke	7165F45
8	220	350	14.7	1/4	Water	Manual	Instrument calibration	Hoke	7165F45
9	220	350	14.7	1/4	Water	Manual	Instrument calibration	Hoke	7165F45
10	550	350	14.7	1-1/2	Steam shut off	Manual		Honeywell	9108
11	550	350	14.7	1/4	Steam	Manual	Instrument calibration	Dragon	303-3
12	550	350	14.7	2	Steam shut off	Motor	2-minute close limit switches	Honeywell	1107
13	550	350	1	1/2	Steam shut off	Motor	30-sec close limit switches	Honeywell	1407
14	550	100	14.7	3/8	Steam relief	None	20 psig set	Sage	220-6B-8MC
15	550	100	14.7	1/4	Steam	Manual	Instrument calibration	Dragon	303-3
16	550	350	1	1-1/2	Steam relief	None	350 psia set	Powell	2333 1/2 × 3
17	220	40	1	1/2	Water	Manual shut off		Jamesbury	1/2 D15OF 36M
18	220	14.7	1	1/4	Vapor	Manual	Instrument calibration	Hoke	7165F45
19	220	350	14.7	3/8	Water shut off	Solenoid	N. C.	Skinner	X52 HDB-20502
20	160	35	1	1/2	Water shut off	Solenoid	N. C.	Skinner	X52 HDB-20502
21	160	35	1	1/2	Water shut off	Manual		Jamesbury	1/2 DZF 30[illegible] 36M
22	160	35	1	1/2	Water shut off	Manual		Jamesbury	1/2 DZF [illegible] 36M
R1	550	350	14.7	3/8	Steam regulator		8-10 psig	Fisher	57-[illegible]
CV1	220	350	14.7	1/2	Water			Sage	[illegible]
CV2	220	350	14.7	1/2	Water			Sage	93[illegible]
[illegible]	160	35	1	3/8	Water		1 psid crack	Nupro	8C4

4.7 ELECTRICAL SYSTEM

The electrical system provides a capability of 100 kw of three-phase, 60 Hz power at 125/216 Y volts at the generator to assure 120/208 volts at the MUS outlets. The information presented below applies primarily to the nuclear electric power plant but, because of the integrated nature of the power plant and habitat, some aspects of the habitat electrical system are included here.

The TOPS electrical system is based on components and equipment designed to meet commercial marine service specifications (e.g., IEEE-45). This is believed to be necessary since the TOPS will be towed on the ocean surface to the station site where, in the process of preparation for submergence, the hull may be open to sea atmosphere. These components and equipment will also satisfy operational requirements in 100% relative humidity.

The TOPS electrical system is shown in its relationship to the MUS electrical system in the single-line diagram (Fig. 4.14). The electrical output from the TOPS main bus connects to the MUS bus by way of four submarine cables, each rated at 30-kw power capacity, and each comprised of a 3-phase, 4-wire circuit. Circuit breakers are shown at either end of the cables to isolate cable faults. Since the TOPS hull will not be occupied during normal operations in the submerged mode of operation, all instrument output and controls will be available in the MUS habitat. Power circuits necessary for TOPS plant operation are restricted to the TOPS hull and related control and instruments leads only are routed into the MUS hull. An external connection is provided from the TOPS main bus to an external source in a tender, shore-based or buoy-based plant. This connection might have been more directly made to the MUS bus, but it was desirable to locate the switchgear associated with it in the TOPS hull, thereby retaining more space in the habitat. Also, by this arrangement, one less hull penetration is required in the larger of the two hulls. During normal startup operation of the TOPS plant, an external source of power will be used for TOPS plant loads. However, under submerged conditions or otherwise, it will be desirable to start or restart the TOPS plant using the auxiliary battery provided in the MUS hull. This may be accomplished by reversing the normal direction of power flow, so

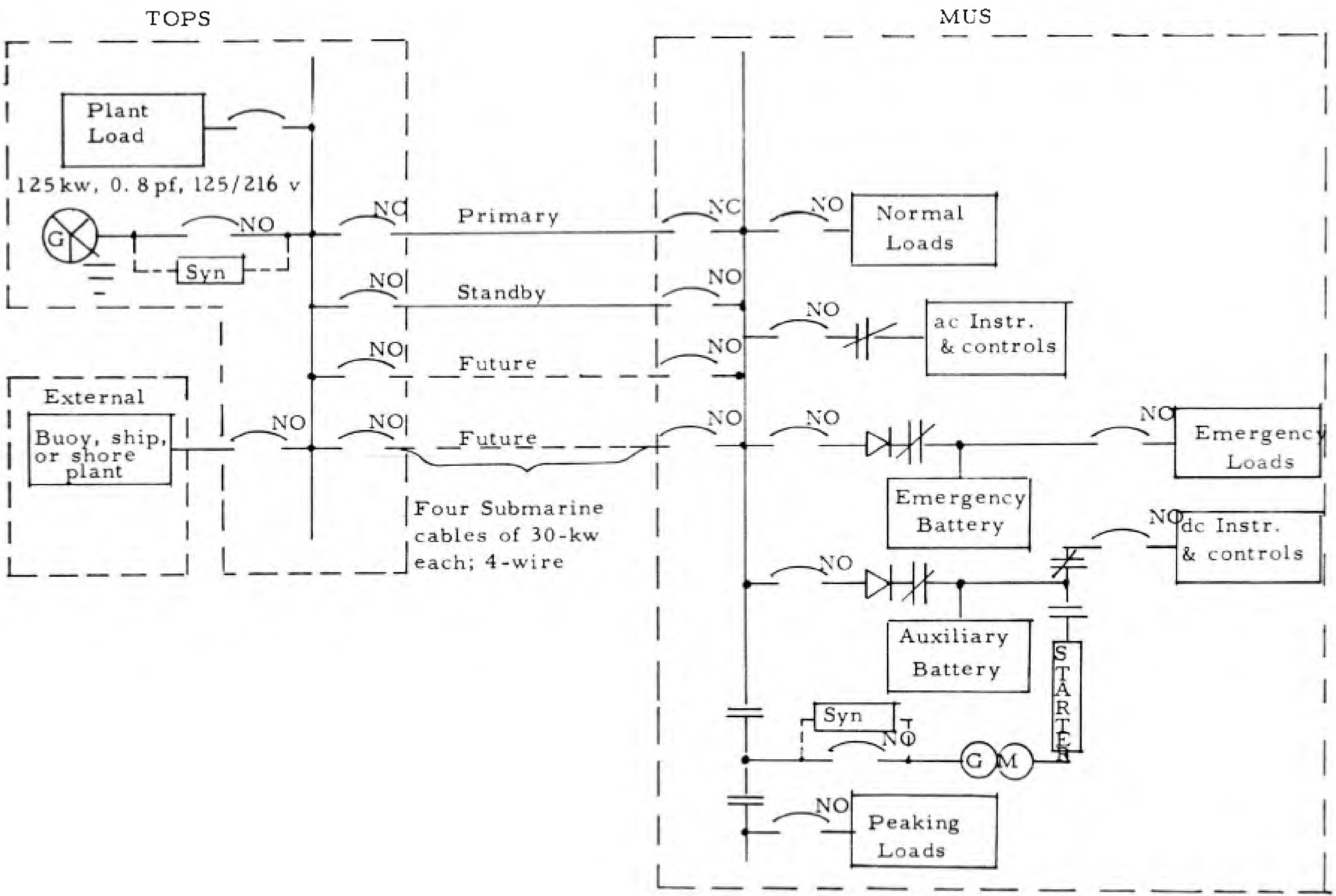

Fig. 4.14--Single-line electrical distribution system diagram

that the MUS bus, energized by the motor-generator set driven from the auxiliary battery, will supply power to the TOPS bus.

During system startup, approximately 10-kw of 3-phase, 60-cycle, ac power will be required from the auxiliary source located in the MUS module for a period of one hour. The auxiliary battery has sufficient capacity when fully charged for several hours of supply to the plant loads. As the TOPS turbogenerator is synchronized and the generator circuit breaker is closed, the motor-generated set driven from the auxiliary battery and the turbogenerator will share the plant load momentarily, after which time the motor-generator set can be shut off and the auxiliary battery recharged from the TOPS plant. Since the switchgear in the TOPS plant is to be controlled from the MUS module, remote electrical operation of all circuit breakers is provided.

Certain critical instrumentation and controls in the TOPS plant will be needed to maintain the plant in the standby condition during "quiet operation." "Quiet operation" is the operational mode where all rotating machinery or other noise and vibration sources are shut down so as to provide a minimum noise level for experimentation where external, natural, and/or man-made noise detections are important. In particular, the control rod drives of the nuclear reactor and the primary system pressure indication will be required. Source power for these systems is dc and is provided from the dc bus in the MUS module.

The turbogenerator set of the power conversion loop is the primary source of power for the TOPS electrical system. The generator is a 125-kw, Wye-connected, 3-phase, 125/216Y volt generator. Other characteristics of this generator are given in Table 4-6. The generator as supplied is provided with a brushless excitation system and silicon-controlled rectifier (SCR) voltage regulation by varying the field current to the generator's exciter winding.

To bring the turbogenerator onto the line, generator frequency and voltage will be brought to the operating point according to the setting of the throttle valve on the turbine. Synchronization of the generator will be achieved by signal and feedback between the synchroscope and electrically-adjusted Woodward governor on the turbine. The automatic synchronizing device will close the generator circuit breaker when synchronism is achieved.

The Wye-connected generator provides both 125 volts phase to ground, single phase, and 216 volts, three-phase, power at standard utilization voltages so that no stepping or isolation transformers are required.

Table 4-6

GENERATOR DATA

Type	Brushless synchronous
Rated output (kw)	125
Power factor	0.8
Speed (rpm)	1800
Voltage (volts)	125/216 Y
Phases	3
Connection	Wye
Frequency (cycles)	60
Temperature rise ($^{\circ}$C)	60 (in 50°C ambient for marine applications)
Efficiency	90
Weight (lb-mass)	1445
Coolant	Air
Excitation	Direct-connected, self-excited
Construction	Drip-proof

Generator Voltage Characteristics

Regulation	
Steady-state (no load to full load)	±2%
Transient (apply or remove full load at pf = 1.0)	10%
Unbalance	
Balanced load	1%
25% unbalanced load	7.5%
Waveform	
Rms total harmonics, (line-line)	5%
Rms total harmonics, (line-neutral)	7%
Maximum single harmonics, (line-line)	3%

TOPS plant loads are shown in Fig. 4.15, where it can be seen that the major load is one of the feedwater pumps, while the other is on standby. Since the pump is driven by a 3-phase motor, balanced loading is inherent. Similarly, the remainder of the plant load consists of fractional horsepower, three-phase motor-driven pump and blower and their standbys. Lighting and receptacle equipment normally in use only when the NEPP is shut down are distributed from a separate panel and may be provided either as single or three-phase (3 Φ is shown in Fig. 4.15.) It is assumed that the loads in the MUS module have been engineered so as to achieve acceptable load balance under all predictable operating conditions.

The TOPS electrical system will utilize air circuit breakers. The main TOPS circuit breaker is a draw-out, air-type of 25,000 amp interrupting capacity. Draw-out construction provides maximum safety to personnel. All cubicles are deadfront design. No contact with an energized breaker is possible, and removal of the breaker for service disconnects the circuit completely so a separate disconnect switch is not required. In addition to remote control operation, manual operation of the breakers is provided. The use of air breakers eliminates the need for storage and conditioning of electrically insulating oil.

For protection of the power plant from faults in the distribution system, a protective relaying scheme is employed which provides overcurrent, overvoltage, and undervoltage protection. The overcurrent relays are set to trip the generator breaker instantaneously for large currents and provide inverse time delay for fault currents below the instantaneous values. Maximum utilization of the thermal capacity of the generator is thereby achieved for temporary overloads. Differential relays trip the breaker and de-energize the field supply, in case of generator internal faults.

Switchgear (shown in Fig. 4.14) will be located on the machinery deck in a metal enclosure 36-in.-wide (2-unit) by 48-in.-deep, by 6 ft 6-in.-high (78 ft^3). Weight of the switchgear and enclosures is given in Table 4-7, along with other estimated weights of the NEPP electrical system. Power cabling weight is based on four 30-kw, 3Φ, 4-wire cables. Standard cables are used within the TOPS and MUS hulls, but submarine cable of recommended construction[14] is used between hulls. An allowance of 20 ft/cable and 4 ft/cable was assumed in the TOPS and habitat hulls, respectively. All wiring was assumed to be metal-conduit enclosed.

The electrical system instrumentation and controls of the NEPP are located in the power distribution panel located on the control deck in the habitat immediately to the right of the power conversion console. Figure 4.16 shows the physical arrangement of the panel. Instruments include a voltmeter with phase switch, a frequency meter, a synchroscope indicator, a

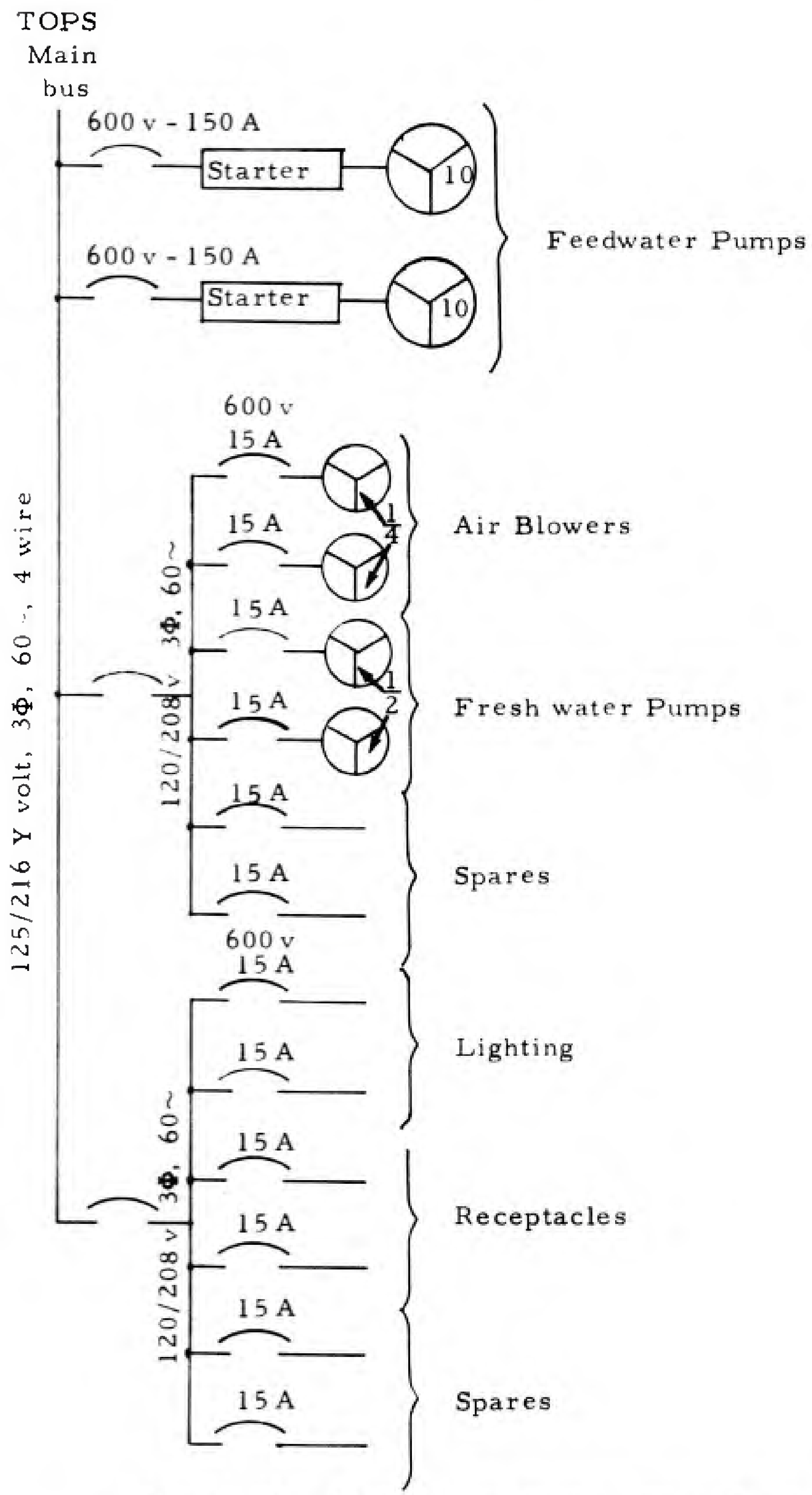

Fig. 4.15--TOPS power plant auxiliary loads

Table 4-7

ELECTRICAL SWITCHGEAR AND WIRING WEIGHT ESTIMATES

Item	Weight (lb-mass)
Switchgear	670
Metal enclosure (marine specification)	1730
Power cables	200
Lighting and receptacle fixtures and wiring	400
Miscellaneous (20% contingency)	600
Total	3600

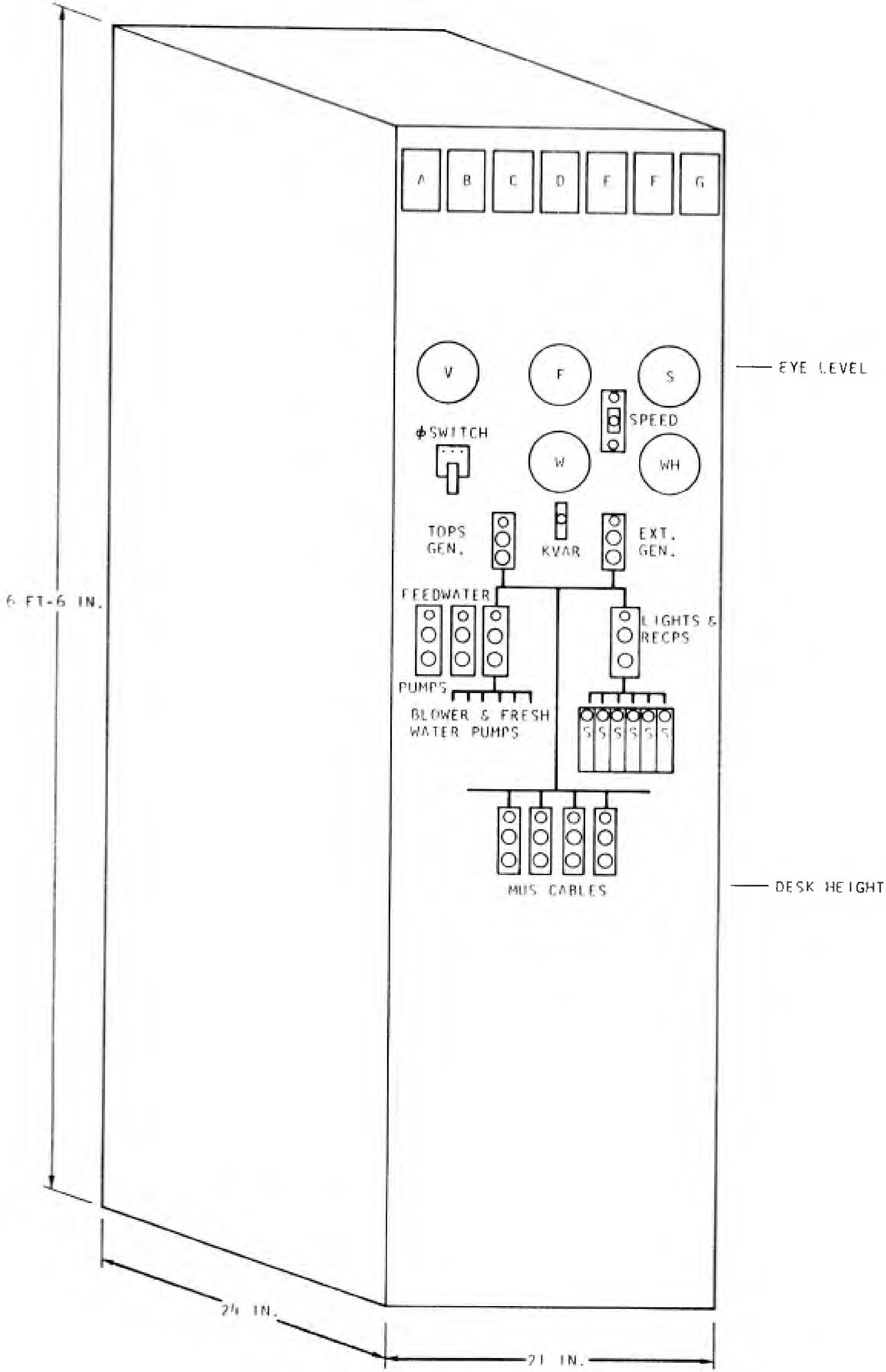

Fig. 4.16--TOPS power distribution panel

wattmeter with autotransformer and switch to permit KVAR to be read, and a watt-hour meter. Necessary current and voltage transformers are provided.

Also located on the panel are close-open contacts for remote control of the circuit breakers located in the NEPP compartment, as indicated above. Closed breakers are indicated by panel lights. Contacts for operation of the circuit breakers to the air blowers, freshwater pumps, and spare circuits are located on the power conversion loop console, rather than the power distribution panel, since these units are started across-the-line. Also provided on the power distribution panel is a contact for remote operation of the electrical adjustment of turboalternator speed via the Woodward governor on the steam turbine. This will permit the alternator to be brought on the line by manual synchronization, if desired, even though automatic synchronism is regarded as the normal procedure.

TOPS power profiles for various operating modes are shown in Table 4-8. Information contained in the table is further explained by the following paragraphs.

TOPS Power Profile Notes

1. TOPS plant auxiliary loads are 3-phase, Y-connected motors,

   1 - 10 hp feedwater pump,
   1-1/2 hp fresh-water pump,
   1-1/4 hp air blower.

2. Standby operation is defined as the existence of the following conditions for "quiet operation"

   a. Reduced primary coolant temperature, 250°F;
   b. All rotating machinery off;
   c. Reactor at operating temperature and pressure condition, but zero power extraction (except for inherent compensation of heat losses);
   d. Reactor and primary system instrument power is required for readout of existing system conditions;
   e. Power conversion loop requires only pressure, temperature, liquid level, and annunciator power;
   f. TOPS power distribution panel requires only power for annunciator lights.

3. Emergency operation is defined to mean that TOPS is inoperable and only life-support and recovery system power will be permitted. Thus, power to monitor the MUS habitat for nuclear radiation is the only demand made by the TOPS system during emergency operation.

Table 4-8

TOPS POWER PROFILE

	V (volt)	Φ/Hz	Startup Load (kw)	Startup Duration (hr)	Normal Operation Load (kw)	Normal Operation Duration (hr)	Shutdown Load (kw)	Shutdown Duration (hr)	Standby Load (kw)	Standby Duration (hr)	Emergency Load (kw)	Emergency Duration (hr)
TOPS plant Auxiliary loads (motors)	125/ 216 Y	3/60	6.75 3.55	0.67 0.33	6.75	2200	6.75	2.5	0	<24	0	—
Reactor and primary system instruments and controls (control rod drives and instruments)	125	1/dc	1.71	1	1.71	2200	1.71 0.285	0.2 2.3	1.71	<24	0.02	24
Power conversion loop instruments and controls	115	1/60	0.40	1	0.40	2200	0.40	2.5	0.17	<24	0	—
Power distribution switchgear (solenoids, annunciators, and indicator lights)	115	1/60	0.53	1	0.53	2200	0.2	2.5	0.2	<24	0	—

4. During the first 10 minutes of shutdown procedure, the control rod drives will operate to drive all rods to full "in" positions. During the remainder of the 2.5-hr period only MUS monitoring instrumentation will require power in the reactor and primary system. Although the TOPS plant will cool off safely without any auxiliary power supplied, circulation in the power conversion loop will shorten the time, and is preferred.

5. Solenoid holding coils in circuit breakers and control elements have a power factor of 0.5.

6. Power requirements for instrumentation and control for reactor and primary system and for power conversion loop are shown below. The total connected loads are also indicated.

Reactor and Primary System Instruments and Controls

	Power (watts)
Annunciators	40
Air radioactivity monitors	100
Area radiation monitors (2)	50
Log power channels (2)	80
Linear power channels (3)	90
Integrator	10
Control logic	10
Dc/dc converter	42
Pressure transmitter channels (3)	3
Total	425

Power Conversion Loop Instruments and Controls

	Power (watts)
Motor valves (3)	70
Solenoid valves (2)	50
Annunciators (10)	30
Relay coils (18)	18
Instruments and indicator lights	20
Control indicator lights	15
Power supplies, pH conductivity, humidity, pressure, and temperature	130
Total	333

Power Distribution Panel

	Power (watts)
Synchroscope, wattmeter, watt-hour meter	120
Annunciators (7)	45
Control indicator lights (15)	15
Circuit breaker solenoid coils (20) . . .	260
Total	440

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5. PRESSURE HULL AND FACILITIES

The pressure hull containing the TOPS/MUS nuclear power plant is connected to the habitat pressure hull by an access sphere. A hatch is located at each entrance. During normal operation the hatches are closed and the atmosphere in the two hulls is separated so that separate atmosphere control systems are required. The design of the hulls and air control systems is provided by GD/EB. However, a brief description is given here for completeness.

5.1. PRESSURE HULL

The power plant pressure hull is a vertical cylinder with hemispherical heads as shown in Fig. 5.1. The hull is fabricated from HY-130 and all penetrations are welded. Normal access to the power plant hull is through a standard size hatch, 22-in.-diameter opening, in the upper head between the hull and the access sphere. A second hatch of larger diameter, 30 in., is located at the top of the upper head for access during refueling operations. The cylindrical portion of the hull is stiffened with external ribs to provide rigidity to resist buckling by external sea pressure. The hemiheads are unribbed. Internal support of the reactor power plant components is made available at two locations, one at an internal support ring located near the connection between the cylindrical section and the upper head, and at a similar support ring at the lower end.

Electrical penetrations for cables to the habitat hull are to be provided by the hull contractor.

The power plant also acts as a containment vessel for the nuclear power plant in the event that there is any release of radioactivity from the primary system. Consequently, the plant can be operated or tested at the surface or at a shore location without need for additional containment. As discussed in Section 8, the maximum pressure that would result from a complete failure of the primary water loop would be about 125 psia. The power plant hull has adequate strength for containing this pressure with a very considerable safety margin with the exception of the access hatches which are designed only to withstand an external pressure. Consequently, double hatches will be used, with the inner hatches designed to seal against an internal pressure of 300 psia.

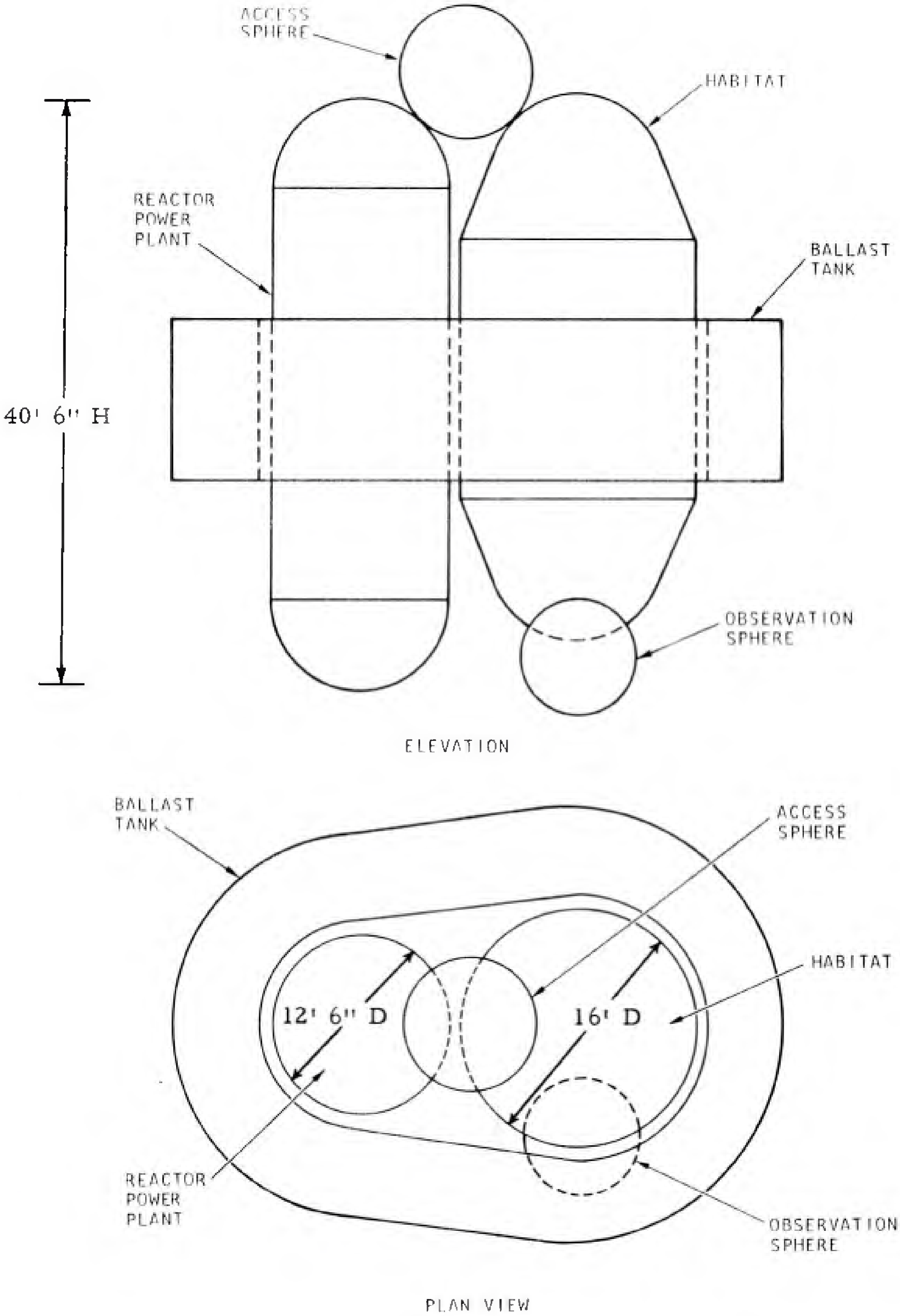

Fig. 5.1--MUS layout showing location of reactor power plant

5.2. ENVIRONMENTAL CONTROL

To permit access for maintenance on station with the convenience of permitting intermingling of air through the open hatches, the air temperature, pressure, composition, and trace concentrations of foreign materials, must be controlled within acceptable limits to be established by others based on operational requirements. These systems were not investigated in detail during the study; however, preliminary considerations indicate the following information:

Normal air is strongly recommended for the atmosphere in the power plant hull in preference to an inert gas such as nitrogen or helium because of the additional fill systems and maintenance access problems associated with a non-air atmosphere. The chief disadvantage of air is its contribution to oxidation of surfaces, particularly iron, in the secondary loop. This disadvantage can be adequately handled by continuous water treatment, which is reviewed in Section 4.4.1, and by selection of corrosion-resistant materials in the higher temperature parts of the secondary system.

Temperature and humidity are maintained within acceptable limits by the cooling system described in Section 5.3. The oxygen content is not expected to decrease appreciably due to iron oxidation during the minimal 30-day mission period, and no provision is included for make-up. Before entering the power plant hull the pressure hatch must be opened by a crew member; this then reveals the inner reactor containment hatch in which a valved air sample connector can be located to permit withdrawal of an air sample to be analyzed with habitat equipment.

5.3. MISCELLANEOUS SYSTEMS

5.3.1. Fresh Water Cooling System

A fresh water cooling system is provided for several of the major pieces of equipment. A schematic of this system is given in Fig. 5.2 and a compilation of the estimated heat loads removed from the various pieces of equipment is shown in Table 5-1, below.

Table 5-1

HEAT LOADS FOR THE FRESH WATER COOLING SYSTEM

Item	Load (Btu/hr)
Control rod drives	3,070
Turbine bearings	4,090
Gear reducer	8,860
Generator	27,950
Air ejector precooler	4,500
Total	48,470

Water circulation for the system is provided by two Worthington 1 DN 2 monobloc centrifugal pumps with 3-in. impellers. These pumps (Fig. 5.3) will be driven by 1/2-hp, 3500 rpm, 60-Hz motors. The two pumps, each capable of pumping 28 gpm at 30 ft of head, are connected in a hydraulic parallel. Under normal conditions only one pump is in operation with the other on standby. Switching from one pump to the other can be done either automatically or manually, on failure of the operating pump.

The fresh water cooling system is essentially a closed loop. A small reservoir and expansion tank is provided to supply the system. Heat absorbed in the fresh water cooling system is transferred to the outer shield water tank by means of a natural convection cooling coil located within the tank. There will normally be no communication between the fresh water cooling system and the outer shield tank water. The heat rejected to the outer shield tank water will ultimately be transferred through the lower hemihead and into the ocean.

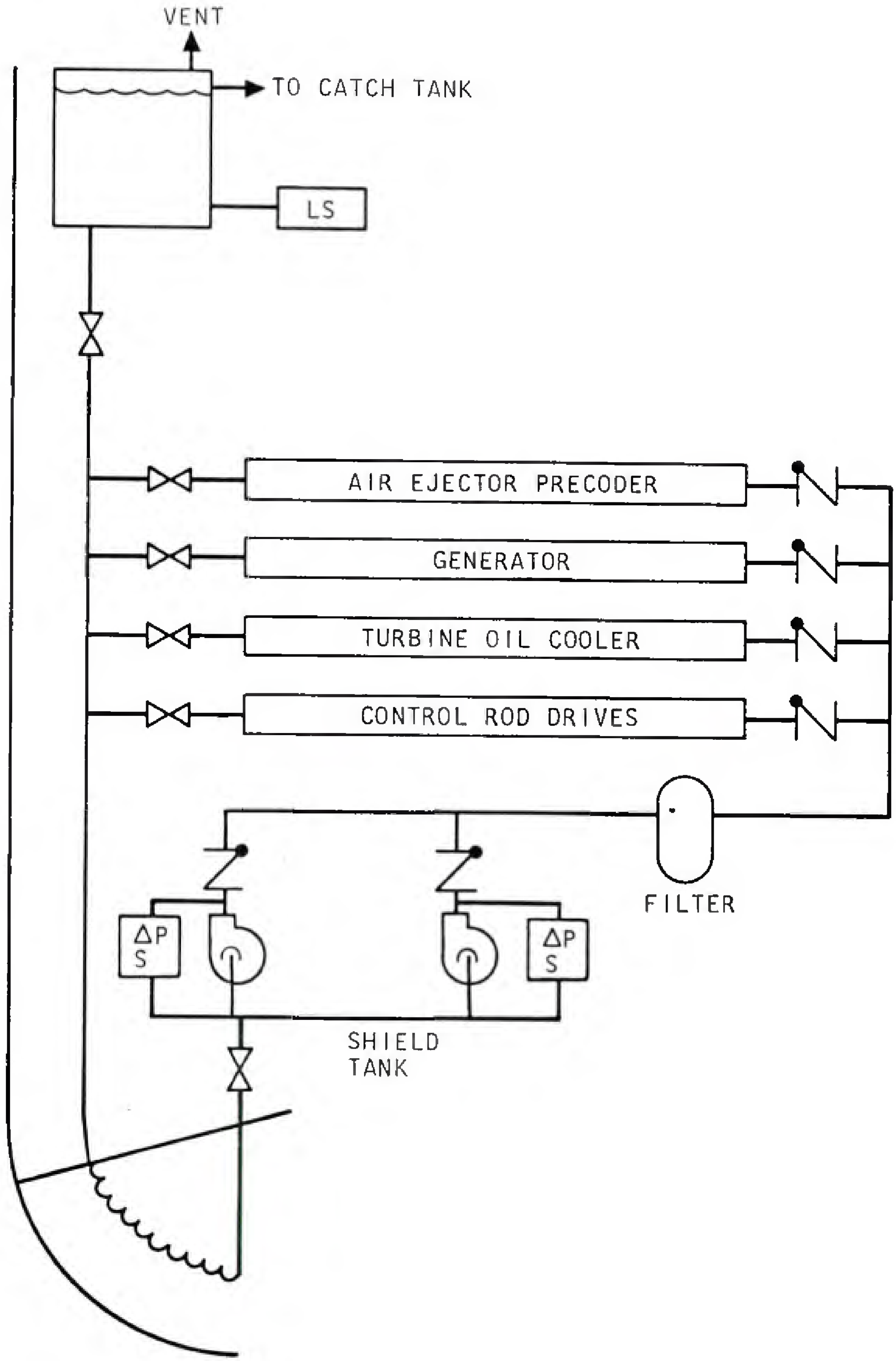

Fig. 5.2--Fresh-water cooling system schematic

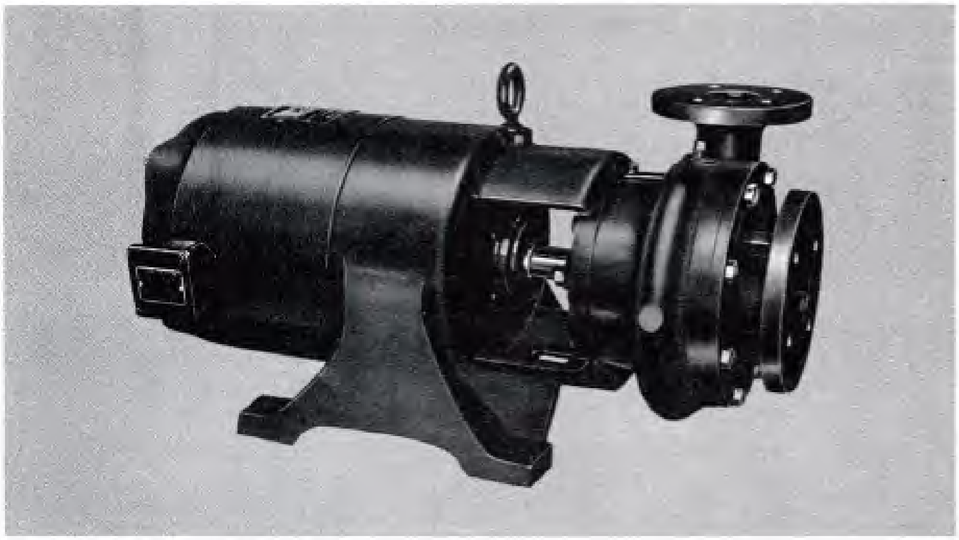

Fig. 5.3--Worthington Monobloc pump

5.3.2. Air-Cooling System

A system is provided in the NEPP hull to cool the air within the space. There are several sources of energy, which are not accounted for by the other cooling systems, which tend to increase the temperature within the hull.

The largest source of heat into the air space will be the main condenser. Since one wall of the condenser is contiguous with the interior of the hull and is insulated, it will still transfer approximately one percent of its energy to the air space. If the energy is not removed, the air space temperature would ultimately equilibrate with the condenser.

Other sources of heat are the turbine, the feedwater pump motors, high-temperature piping, upper portion of the reactor, fresh water pump motors, gear reducer case, generator case, air ejector vent, and fan motors. A compilation of these estimated heat loads is given in Table 5-2.

Table 5-2

ESTIMATE OF AIR COOLING SYSTEM HEAT LOADS

Item	Load (Btu/hr)
Condenser	30,000
Valves and piping	18,000
Turbine generator set	4,100
Feed pump	2,100
Fresh water pump	200
Air ejector vent	100
Fan	100
Reactor	7,000
Miscellaneous electrical	300
Total	59,900

The air-cooling system consists of a pair of one-quarter horsepower motor-driven squirrel cage blowers, mounted in the space below the 30-in. access hatch. They are connected through barometric louvers to a plenum and ducting system which directs the flow of air over the interior of the upper hemihead.

The blowers, louvers, and inlet plenum are designed to be removed through the 30-in. -hatch to allow access into the machinery space below. An illustration of this system can be seen in Fig. 2.1.

Each of the 1/4-hp blowers installed should be capable of delivering approximately 200 cfm of air to the plenum. This will be sufficient to remove not only the heat deposited in the air to maintain the air space temperature below 120°F, but will also remove about 1/2-lb of water per minute from the air space under saturated conditions. The water removed from the air space will condense on the inside of the upper hemihead, drain down to the gutters at the edge, and then into the system make-up tank. This will provide a supply of fresh water to make up for any leakage in the system.

5.3.3. Make-up and Catch Tankage

Both the make-up and catch tanks are located in space allotted to the outer shield void tank system. The locations are shown in Fig. 2.1.

Each of these tanks has a capacity of about 21 ft^3. The catch tank will be initially empty, and during the normal course of events, should never become full.

The make-up tank will be partially-filled initially with about 10 ft^3 of demineralized water. This will occur after the proper amount of water has been introduced into the main condenser. The make-up tank will have a level transmitter associated with it, so that the operator can see the level in the tank at any time (see Fig. 7.1).

Water will be pumped into and drawn out of the make-up tank from the secondary system, depending on the condenser hot-well level. A mixed bed demineralizer will be provided in the system between the make-up tank and the condenser hot well to ensure that only high-quality water is drawn into the secondary system. After entering the secondary system the water subsequently passes through the secondary system full-flow demineralizer before entering the steam generator so that, in effect, make-up water is demineralized twice.

An electrical safety circuit is included in the make-up tank valve system to prevent drawing water out of the system, should it become empty, or adding water to the system if the tank is full.

The make-up tank will receive water from the condensate formed in the air-cooling system, since this will be a source of relatively pure water. In addition, it will be connected to selected drains such as the boiler feed-pump packing gland, which will also be sources of relatively pure water.

All leakage sources of water that could be expected to contain contaminates, particularly oil, would be drained to the catch tank. A system that could evaporate the catch tank water was taken into consideration; however, it was rejected as it was an unnecessary system; the vapor formed by this evaporator would have been condensed by the fresh water cooling system and returned to the make-up tank to supply an additional source of make-up water.

Water collected in the catch tank or any of the void tanks would have to be removed at the normal maintenance periods when the MUS is at the surface. No provision is made to pump these tanks while the system is under water.

6. CONTROLS AND INSTRUMENTATION

Power conversion plant and auxiliary power supply controls and instrumentation information is provided below. Controls and instrumentation for the reactor and primary system of the NEPP, and dose rate and area radiation monitoring within the TOPS and MUS hulls were considered in Section 3.10. Electrical system instrumentation and controls information is presented in Section 4.7.

6.1. POWER CONVERSION PLANT INSTRUMENTATION & CONTROLS

Plant instrumentation and controls include those systems which are required to start, operate, and shut down, the power conversion plant and to ensure its safety through normal operation and all credible emergency conditions. The general requirements of the instrumentation and controls of the power conversion plant are:

1. To provide the console attendant with a rapid, complete, concise, and accurate picture of conditions of the power conversion system at all times, and

2. To operate reliably.

More specific requirements are given in the following paragraphs.

6.1.1. Instrumentation and Control Requirements

The instrument and controls required for the power conversion plant are shown schematically in their relationship to major plant components in Fig. 7.1. The functional requirements of the instrumentation and controls are reviewed here in sequence, starting with the steam generator and proceeding around the main power conversion loop.

Steam Generator and Steam Separator

The temperature and pressure of the turbine steam are monitored at the effluent of the steam separator, and provide the operator with information defining the thermodynamic state of the steam delivered to the turbine.

Turboalternator

Built into the steam turbine is a Woodward governor for turbine speed regulation and an overspeed trip mechanism designed to protect against excessive speed which might damage the turbine or alternator. In addition, the governor speed is electrically adjustable. Speed adjustment for alternator synchronization is made automatically from the synchroscope or manually by the power distribution panel attendant. In addition, a motor-driven stop valve is provided in the main steam line. This valve will be operated for 2 to 5 minutes between fully open and fully closed positions during startup and shutdown sequences so as to protect the turbine from wet steam. "Open" and "closed" limit indication is provided at the console. The stop valve will be closed during startup until the temperature and quality of the steam have been raised, and the other components of the system are operating properly; it will also be closed during the shutdown sequence as the quality of the steam decays. When the stop valve is closed, the steam bypass valve will be motor-driven to the open position, permitting passage of steam around the turbine to the condenser. A fixed orifice in the bypass steam line regulates the rate at which bypass steam can flow. In addition, the pressure at the steam turbine gland is regulated by a pressure regulator, pressure switch, and spring-actuated relief valve, which control the pressure and leakage through the gland.

Main Condenser

The temperature and pressure in the condenser are measured to provide the operator with condenser performance. The liquid level in the condenser is measured at two diametrically opposite points (at feedwater pump suction pipes). Because of the large diameter of the annular condenser, erroneous indication of the condensate level would tend to be indicated whenever the axis of the plant was off from its vertical position. As a solution to the problem, the two sensors are used and their level signals averaged. This is accomplished by taking the output signal from the LVDT transmitters and electronically rectifying, summing and averaging them. The averaged level is displayed for the console attendant. Automatic dumping or filling of condenser water to or from the make-up water tank to regulate the level in the condenser is controlled from the averaged level values. Console-controlled fill and drain operation are also provided. The purity and condition of the water in the condenser are measured and displayed on the console continuously with pH and conductivity meters.

Make-up Water Tank

The water level in the make-up tank is monitored to provide the operator with knowledge of the amount of make-up water available. Fill and drain operations of the make-up tank to the condenser are normally automatic, utilizing solenoid actuated valves and make-up tank level switches in series with set-point switches on the average condensate level indicator. However, these operations may also be performed manually from the control console. Low-level limit is annunciated by audible and visible indicators.

Feedwater Pumps

The pressure at the discharge of the feedwater pumps is measured, electrically transmitted, and displayed for the console attendant. In addition, pressure switches are provided so that either the failure or degraded output of one pump will automatically start and substitute the operation of the standby pump. Since the pumps are interchangeable, a choice may be made as to which pump will operate and which will stand by. Typical control circuitry for selection of the pump for primary operation and automatic switching to the standby pump with loss of output from the primary pump is shown in Fig. 6.1. Orifices are provided in the bypass line from the feedwater line to the condenser, in order to tailor the characteristic pressure-flow curve of water delivered to the steam generator, to satisfy turbine requirements and to protect the pumps from damage.

Demineralizer Pressure Drop

The demineralizer in the feedwater line is to function both as a mechanical filter of loop crud and as a mixed ion-exchange bed. The life of the demineralizer is limited by the pressure differential developed across it as crud fills the pores. The pressure difference will be measured and a high-limit alarm provided to announce the end of the demineralizer life. Approach to the limiting ΔP will be visually displayed by a meter reading in comparison with a set point.

Feedwater Startup Valve

A motor-driven diverter valve is provided in the feedwater line for operation during startup to provide control of the feed rate to the steam generator and to permit pump start in a low-load condition. Console control and "open" and "closed" limit lights are provided.

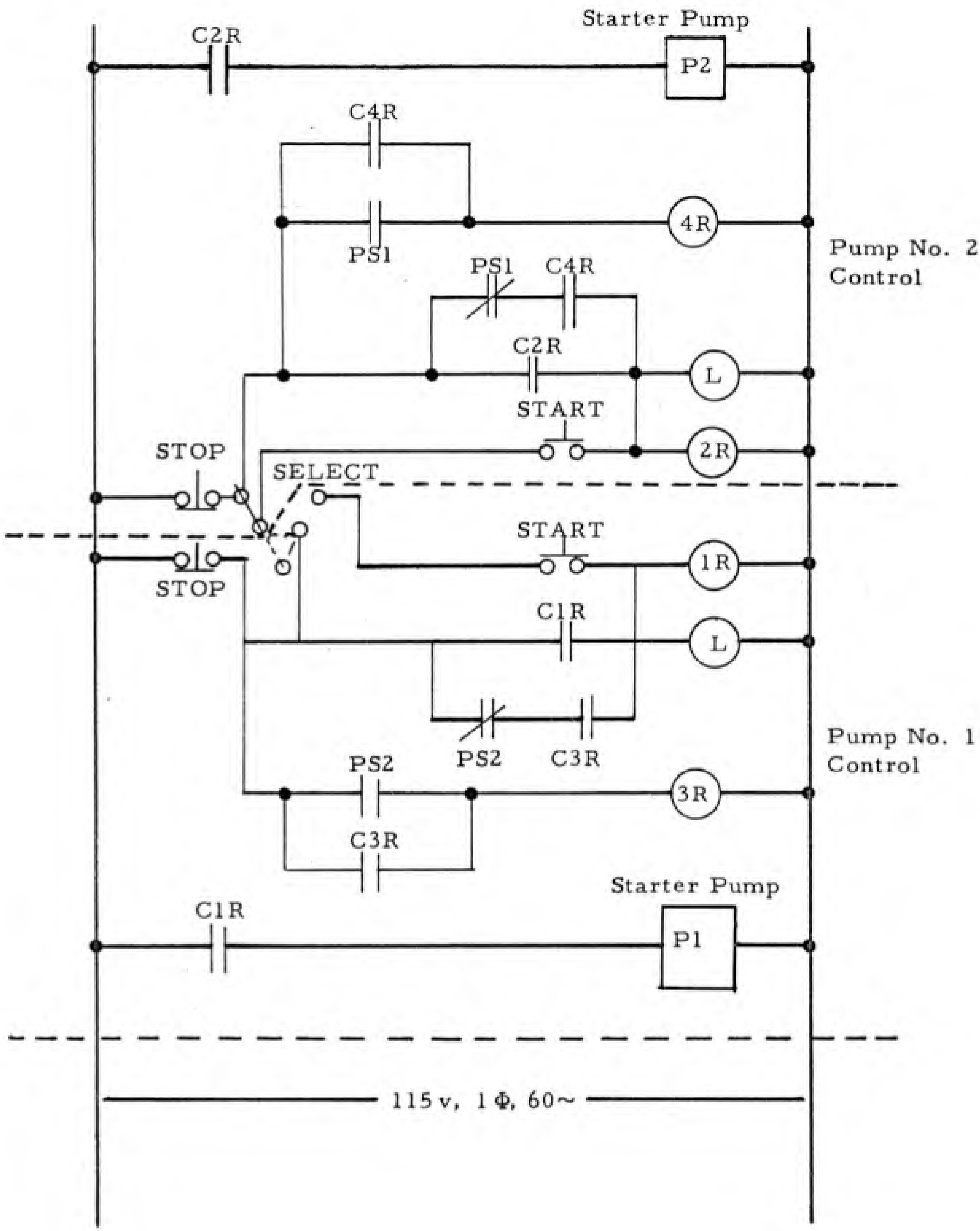

Fig. 6.1--Typical control circuit for automatic switching of standby units

Air Blowers

Air blowers will be controlled from the console by start and stop contacts since across-the-line starting is employed. Control circuitry, similar to that shown in Fig. 6.1, for the feedwater pumps will be employed to permit selection of primary and standby functions and automatic switching from primary to standby in cases of primary blower malfunctions. (In particular, the circuit diagram need only be changed by substituting the blower circuit breakers for the feedwater pump motor starters.)

Ambient Atmosphere

The ambient atmospheric conditions, temperature, pressure, and relative humidity within the NEPP hull are measured and electrically transmitted to the control console for display.

Fresh Water Pumps

A fresh water system for cooling control rods in the primary loop, after-condenser, air-ejector precooler, turbine bearings, gear box, and alternator coolant air was shown in Fig. 5.3. A one-half horsepower motor-driven pump provides circulation with a matching pump provided for standby. Selection of either pump for operation or standby service may be made from the console. Automatic switching from operating to standby pump operation is provided for cases of malfunction of the pump initially selected for operation. Fresh water pumps are controlled from the console by start and stop contacts since across-the-line starting is employed. Control circuitry is the same as that shown for the feedwater pumps except that the motor starters are replaced by the fresh water pump circuit breakers. A low water level pressure switch to actuate an annunciator is also provided.

6.1.2 Instrumentation and Controls

The instruments and associated controls are commercially available, pre-engineered, and selected for marine service. A list is provided in Table 6-1. Specific instruments will be indicated in the detailed design phase.

Thermocouples

Thermocouples will be chromel-constantan type directly connected to compensated junction meters.

Table 6-1

INSTRUMENT AND CONTROLS LIST

Identification	Function	Element	Range	Accuracy (±%)
TX 1	Steam supply temperature	Cr-C thermocouples	0 to 800°F	1
TX 2, TX 3, and TX 4	Pressure vessel wall temperature	Cr-C thermocouples	0 to 800°F	1
TX 5	Condensate temperature	Cr-C thermocouples	0 to 200°F	2
TX 6	Ambient air temperature	Cr-C thermocouple	0 to 200°F	2
PX 1	Steam supply pressure	Bourden LVDT	0 to 1,000 psi	1
PX 2	Condenser pressure	Bourden LVDT	30-in. vac, -30 psi	1
PX 3	Feedwater pump discharge	Bourden LVDT	0 to 1,000 psi	1
PX 4	Demineralizer pressure differential	Bourden LVDT	0 to 60 psi	1
PS 1 (V15)	Turbine gland pressure	Bourden microswitch		1
PS 2	Feedwater pump No. 1, discharge pressure	Bourden microswitch	0 to 1,000 psi	1
PS 3	Feedwater pump No. 2, discharge pressure	Bourden microswitch	0 to 1,000 psi	1
PS 4	Freshwater pump No. 1, pressure differential	Bourden microswitch	0 to 30 psi	1
PS 5	Freshwater pump No. 2, pressure differential	Bourden microswitch	0 to 30 psi	1

Table 6-1 (Continued)

Identification	Function	Element	Range	Accuracy (±%)
LX 1	Condensate level No. 1	Bellows LVDT	0 to 50-in. H_2O	1
LX 2	Condensate level No. 2	Bellows LVDT	0 to 50-in. H_2O	1
PX 5	Ambient atmosphere pressure in NEPP hull	Bourden LVDT	0 to 60-in. Hg	1
HX 1	Ambient air relative humidity		0 to 100%	1
LX 3	Makeup tank level	Bellows LVDT	0 to 50-in.	1
PHX 1	Condensate pH	pH-cell	6 to 11	0.2
CX 1	Condensate conductivity	Electrodes bridge	10^5 to 10^7 mho	1
RS-13, RS-14, and RS-15	Air blower No. 1 control and automatic standby switching	Solenoids		
RS-16, RS-17, and RS-18	Air blower No. 2 control and automatic standby switching	Solenoids		
S-1	Selector switch for feedwater pumps	Contactor		
S-2	Selector switch for freshwater pumps	Contactor		
S-3	Selector switch for air blowers	Contactor		
A-A to A-J	Annunciator	Lights and buzzer		
S-4 to S-9	Start-stop for pumps and blowers	Contactors and lights		

Table 6-1 (Continued)

Identification	Function	Element	Range	Accuracy (±%)
S-10	Steam stop valve switch	Contactor and limit lights		
S-11	Startup valve switch			
S-12	Turbine by-pass valve switch			
E-1	Control console envelope			
SV-1	Makeup tank fill valve	Coil	On-off	
SV-2	Makeup tank drain valve	Coil	On-off	
MV-1	Steam stop valve	Motor	Open-close	
MV-2	Turbine bypass valve	Motor	Open-close	
MV-3	Startup valve	Motor	Open close	
RS-1, RS-2, and RS-3	Feedwater pump No. 1 control and automatic switching	Solenoids	On-off	
RS-4, RS-5, and RS-6	Feedwater pump No. 2 control and automatic standby switching	Solenoids	On-off	
RS-7, RS-8, and RS-9	Freshwater pump No. 1 control and automatic standby switching	Solenoids	On-off	
RS-10, RS-11, and RS-12	Freshwater pump No. 2 control and automatic standby switching	Solenoids	On-off	
PWRS-1	Power supplies			
SYN-1	Synchronizer	Motor		

Table 6-1 (Continued)

Identification	Function	Element	Range	Accuracy (±%)
V-1	Generator output voltage	Meter	0 to 500 V	0.5
F-1	Frequency meter	Meter	0 to 3,600 rpm	1
Sync	Synchroscope	Meter	0 to 90°	0.5
W	Wattmeter	Meter	0 to 300 kw	1
WH	Watthour meter	Meter	0 to 50 Mwh	1
S	Voltmeter switch	3 pt contact	---	
A	Annunciator (7 positions)			
S	Speed control switch	Lights and contactor		
S	KVAR switch	Contactor		
S	Start-stop contactor for CB control (14 items)	Contactors and lights		
SCB	Circuit breaker solenoids holding coils (20)	Coils		

Pressure Transmitters

Pressure will be measured and transmitted using Bourdon tube-driven linear variable differential transformer (LVDT) devices. Pressure will be indicated by calibrated meters with high and low limit set point switching and relay control, where applicable.

Pressure Switching

Pressure switches will be composed of Bourdon tube-actuated microswitches with relay control of operating elements.

Liquid Level

Liquid level and pressure differentials will be measured using d/p cells with linear variable differential transformer transmission to calibrate meters with limit and alarm switches for relay operation where necessary.

Console

Preliminary layout of the control console for the power conversion system is shown in Fig. 6.2. This console will be located in the habitat on the control center deck between the console (on the left) containing the nuclear reactor and primary system and radiation monitor instruments and controls, and the NEPP power distribution panel (on the right). Descriptions and figures relating to these other two instrument and control centers may be found in Sections 3.10 and 4.7. Table 6-2 is a list identifying the console panel instruments and controls, and Table 6-3 shows the annunciator functional assignments.

Power for the power conversion plant instrumentation and controls is supplied from the MUS ac bus. Approximately 400 watts of power at 115v, 60 Hz, 1Φ, is required. While some of the ac power may be converted to dc within the instrument and control power packs contained in the console, no external source of dc power is required for the power conversion plant instrumentation and controls.

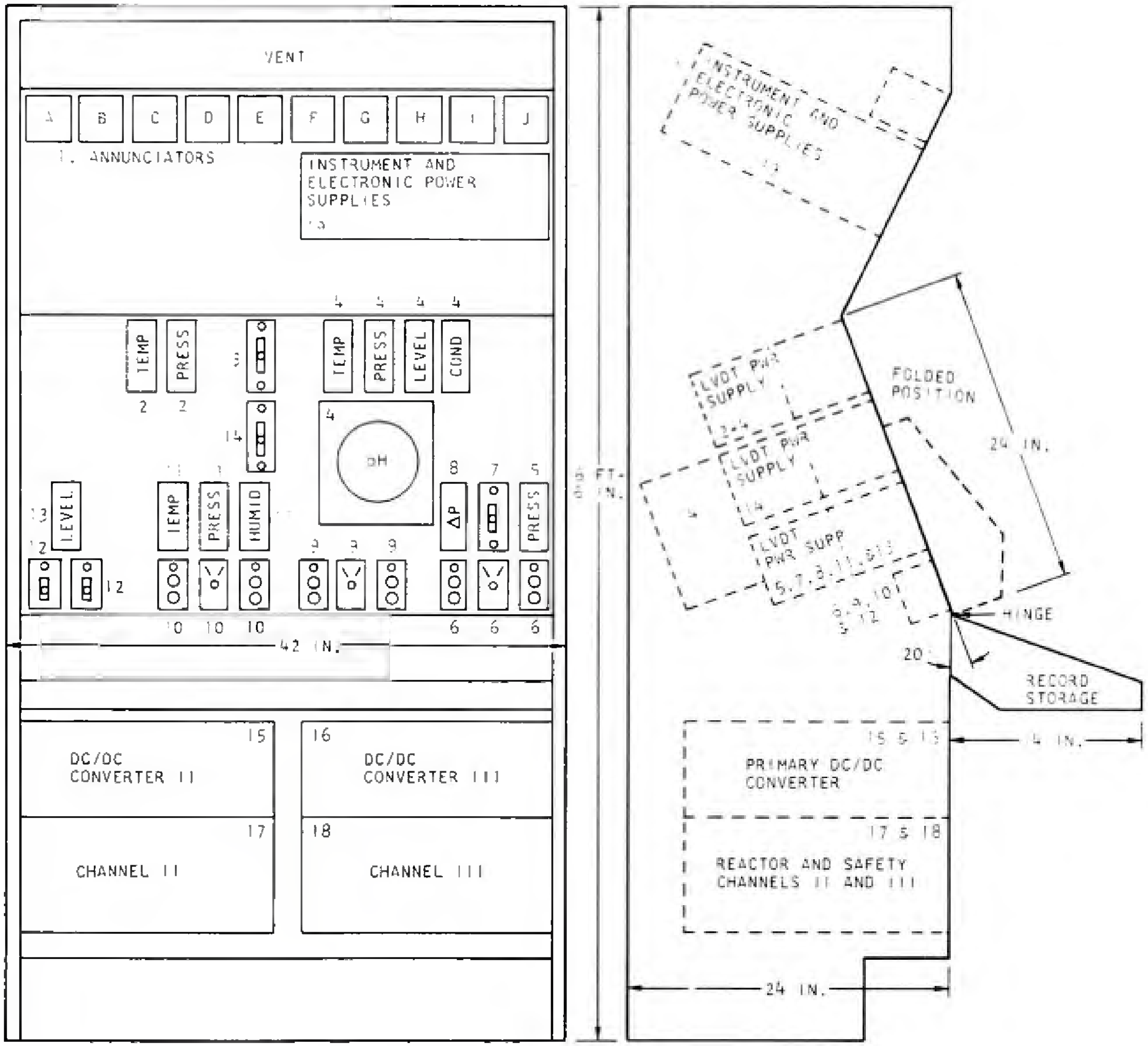

Fig. 6.2--Power conversion system control console

Table 6-2

POWER CONVERSION PLANT CONSOLE
IDENTIFICATION LIST

1. Annunciators
2. Turbine steam supply parameters
3. Steam stop valve two-directional switch and open-closed limit lights
4. Condenser parameters
5. Feedwater pumps discharge pressure
6. Feedwater pump selector switch and start, stop, and power on lights, for operating and standby pumps
7. Start valve two-directional switch and open-closed limit lights
8. Demineralizer ΔP indicator and set point contactor
9. Fresh water pump selector switch and start, stop, and power on lights, for operating and standby pumps
10. Air blower selector switch and start, stop, and power on lights, for operating and standby blowers
11. Ambient air condition indicators for NEPP hull
12. Makeup tank fill and drain switches to solenoid valves with power on lights
13. Makeup tank liquid level indicator
14. Bypass valve two-directional switch and open-closed limit lights
15. Primary system and reactor dc/dc converter power supply for safety channel II
16. Primary system and reactor dc/dc converter power supply for safety channel III
17. Reactor safety channel II electronics
18. Reactor safety channel III electronics
19. Instrument and electronic power supplies for conductivity meter, relative humidity, etc.

Table 6-3

POWER CONVERSION PLANT CONSOLE ANNUNCIATOR LOCATION AND FUNCTIONS CHART

A. Low-liquid level in condenser

B. High ΔP across demineralizer (end-of-life)

C. Low, or high, make-up tank water level

D. Feedwater pump low discharge pressure - standby pump operation required

E. Air blower low output - standby blower operation required

F. Fresh water pump loss of discharge pressure - standby pump operation required

G. pH outside of set range

H. Low fresh water reservoir level

I. (spare location)

J. (spare location)

6.2. AUXILIARY POWER SUPPLY

The auxiliary power supply is a battery of lead-acid cells with a capacity of 1500 amp-hr or, at 125 vdc, a capacity of 187.5 kw(hr). This battery occupies 93 cubic ft of the habitat hull volume and weighs ~ 14,700 lb-mass. The battery powers a motor-generator set of 50 kw, 0.8 power factor capacity which can supply ac power for NEPP startup and other auxiliary loads such as winch (30 hp) operation during descent to the ocean floor. The auxiliary battery power supply is located in the MUS and is the responsibility of the MUS designer (General Dynamics/Electric Boat Division) and thus it is not considered further here. For additional information on the NEPP and its relationship to the auxiliary battery, see Section 4.7, Electrical System.

7. OPERATION AND PERSONNEL REQUIREMENTS

The TOPS has been designed for simple and reliable operation with minimum attendance. Operational control is located at a single console stationed in the MUS module. It is recognized that TOPS is only one of several systems which requires attention during the pre-submergence countdown sequence. It is possible for TOPS to be operated during startup, normal operation, and shutdown, by a single operator. Maintenance to be performed between MUS cycles will, however, require additional personnel, as discussed below.

In the following subsections, consideration is given to the various TOPS operations and personnel requirements. This is followed by consideration of the maintenance tasks to be performed insofar as they are definable at this time. Finally, the support facilities required for TOPS operation and maintenance will be identified and defined.

7.1. STARTUP

Two basic startup modes must be considered in this section. The first will be referred to as "normal" startup and the second will be referred to as "restart."

The difference between these two startup modes is that during normal startup it is assumed that MUS is at the surface and is attached to an external source of auxiliary power provided from a tender. The restart situation would occur when the MUS was submerged and could result from a planned shutdown for quiet operation. In this case, the only power available would come from the auxiliary power supply system of the MUS.

A secondary system piping and instrumentation diagram, to which reference should be made as the startup procedure is described, is shown in Fig. 7.1.

In the following description, it is assumed that the system has been thoroughly checked-out, all water levels are normal, and all valves are in their prestartup positions. The nuclear instrumentation and radiation monitor equipment has been turned on and checked out. The various steam lines should be drained and certain drain lines subsequently closed.

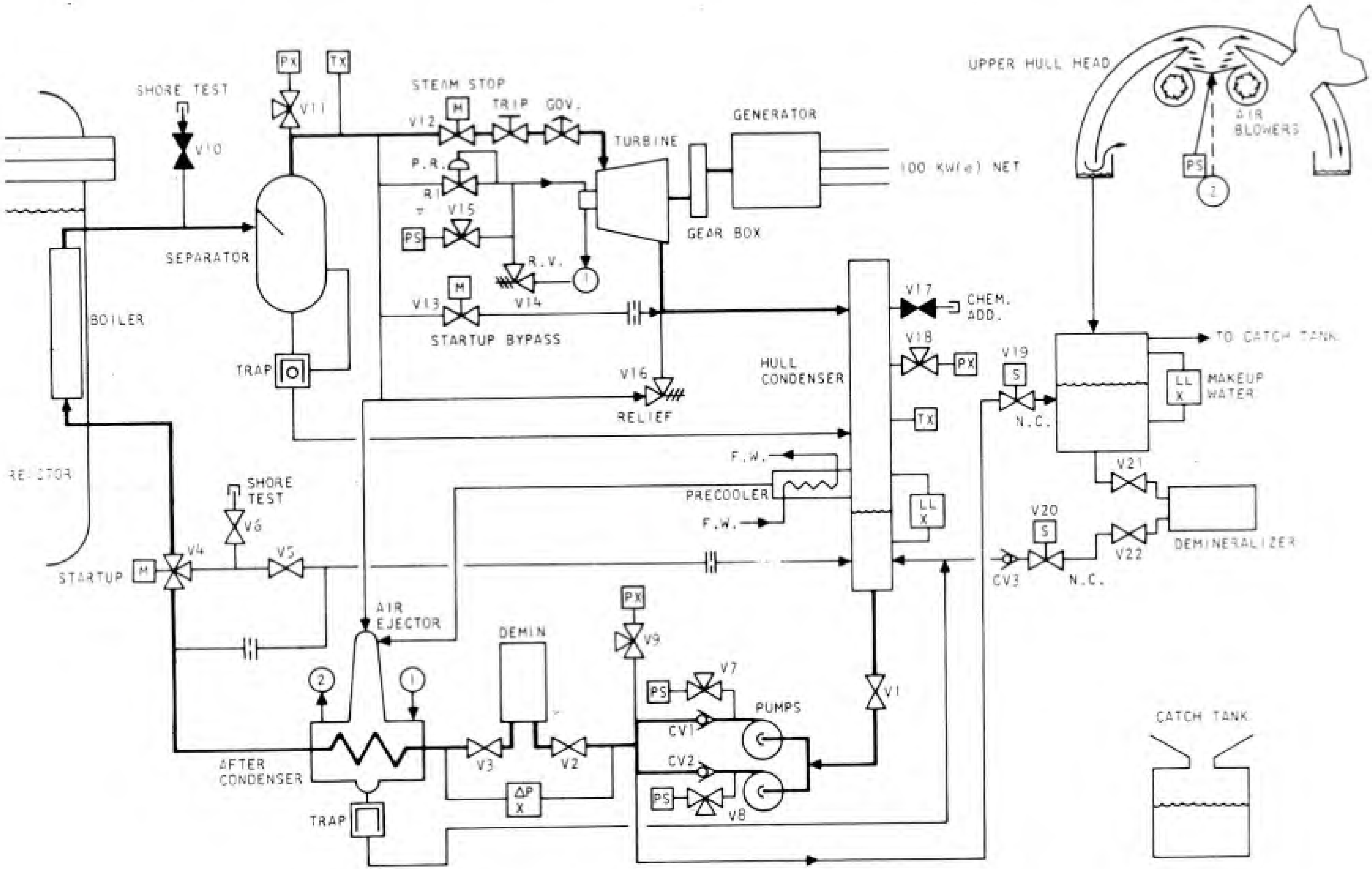

Fig. 7.1--Secondary system piping and instrumentation diagram

At time t-2 minutes, the fresh water cooling system pump will be turned on. This system is not shown in total in Fig. 7.1, but it is used to cool the control rod drives, lube oil system, generator air, and air-ejector precooler. For details of this system see Section 5.3.

At time t-1 minute, the air-cooling system is activated. This blows air over the inside of the upper hemihead (see Section 5.3). The boiler feed pump is also turned on at this time. The startup valve (V4) should be in the position which diverts all of the boiler feedwater to the condenser.

At time t=0, the reactor control rod system is latched,and rod withdrawal begun. The circuitry for the control rod drives is designed so that the rods move one step at a time in a sequential fashion. The reactor core should become critical (K=1) at approximately t+17 minutes. At this point, both the primary temperature and primary pressure will begin to increase. The primary temperature will indicate the greatest initial increase.

The primary temperature should reach 220^{o}F at about t+22 minutes. At this time, the startup valve (V4) motor can be energized to change the flow of boiler feedwater from the condenser to the steam generator. Water will flow through the once-through steam generator and into the steam separator; from the steam separator, the water will flow through a steam trap back to the condenser.

The secondary system pressure should stabilize at about t+24 minutes. At this time, the turbine bypass valve (V13) is opened. This will initially allow both water and steam flow through the steam bypass line, but as the quality of the fluid leaving the steam generator increases, the flow in the bypass line will become entirely steam. Steam will also flow into the steam-jet air ejector, which will result in a decrease in condenser pressure.

At about t+37 minutes, the reactor power and temperature should stabilize with the core power at about 250 kw(t).

The system should remain in this state of equilibrium until the condenser pressure has been reduced to at least 10 psia. This should occur by t+40 minutes. At this time the drive motor for the steam stop valve (V12) will be energized and dry steam will start to flow into the turbine.

At t+42 minutes, the stop valve (V12) will be fully opened and the turbine will be turning at its rated speed. The turbine bypass valve (V13) can now be closed. By t+43 the system should be ready to accept a load. The generator can then be loaded as described in Section 4.7. The system should now be ready to function in a normal manner.

A restart would most likely begin with the reactor hot and the control rods into a position of K=1. This position will be determined by both the nuclear considerations (burnup) and the manner in which the system was shut down. The preferred method of shutting down, as in the case of quiet operation, would be to insert the control rods about 4-in., and cool the reactor down to about 250°F by combining operation of the turbine and opening the turbine bypass valve (V13) with the boiler feed pump operating (see Section 7.3). The desired conditions should be reached in less than 1 hr.

Prior to restart, the startup valve (V4) would be moved to divert the full pump flow to the condenser. The steam stop valve (V12) and startup bypass valve (V13) would both be closed.

At t-2 minutes, the fresh water cooling system and air-cooling system would be started. At t-1 minute, the boiler feed pump would be turned on.

At t=0, the control rods would be withdrawn sequentially. At about t+1 minute, the startup valve (V4) will energize and start the boiler feed flow to the steam generator. The flow and pressure in the secondary system should stabilize by about t+4 minutes.

At t+8 minutes, the turbine bypass valve (V13) can be opened. As soon as the condenser reaches 10 psia and superheated steam is flowing in the startup bypass line, the steam stop valve (V12) can be opened. The turbine should be up to speed and ready to accept the load by about t+15 minutes.

7.2. NORMAL OPERATION

TOPS, as designed, is load-following and has inherent nuclear reactivity stability and therefore requires a minimum of operational monitoring and control once the plant has been started and has achieved steady operation. Monitoring to ensure continued normal operation will be required at intervals of once every four hours or less. Depending upon the level of power operation, occasional adjustments of the control rod positions will be required to compensate for burnup of nuclear fuel.

Similarly, the conversion plant must be monitored and adjusted but the interval of nonattention between checks must be significantly shorter. Intervals of two hours are believed to be reasonable since the power conversion system is made up of rotating machinery components in contrast to static reliability of the primary loop. However, since the primary and reactor instruments and controls are mounted on the same console as those of the power conversion system, they would be monitored simultaneously. Critical TOPS operational parameters trip annunciator circuits at limiting conditions. During the period where the console is unattended, remote visible and audible annunciation may be desired in the location of the MUS crew activity.

Attendance at the TOPS console will be required for about 5 minutes out of every 2 hrs. The attendant's job is to check the status of the TOPS plant. In particular, he would check and adjust,or correct, as necessary a few important items affecting the efficient and reliable operation of the plant. Examples are such items as follows:

1. Compare primary loop pressure and temperature to setback and scram levels. The pressure, along with temperature, indicates the thermodynamic state of the loop and thus the efficiency potential of the plant. Pressure compared with vapor pressure is also an indicator of radiolytic gas content of the loop. Unusual pressure attributable to these gases would indicate accelerated corrosion rates.

2. Electrical Load and Generator Frequency. Turbine speed is directly proportional to the frequency. When compared with the load, the performance of the turbogenerator is indicated.

3. Identification of the Operating Feedwater Pump. Standby parallel feedwater pumps are provided. Failure or malfunction of one pump would automatically start the second pump, and this action would be visible on the control panel. Such an event might require other actions for the continued well-being of the crew or continuation of the mission.

7.3 SHUTDOWN

7.3.1 Scheduled Shutdown

The sequence of operations to effect shutdown is:

1. Drop the main load. Plant and vital nonplant loads may continue during initial cooling of the plant.

2. Make the reactor subcritical by driving the control rods into the core to overcompensate for positive reactivity of the temperature drop.

3. The turbine-generator can be operated for approximately 10 minutes for the purpose of lowering reactor system temperatures. This also conserves battery energy and reduces thermal shock to the turbine during shutdown.

4. Drop the remainder of the load and switch to auxiliary power source for the plant load during cooling.

5. Bypass steam around the turbine to the condenser.

6. Stop steam supply to the turbine.

7. Cool off the plant through the condenser and pressure vessel insulation.

The above steps describe a normal shutdown operation; in addition, there are scrams and setback conditions which are considered elsewhere. A more detailed description of the shutdown operation and sequence is contained in Section 3.8, "Decay Heat Removal." Since the plant is designed for 1% thermal leakage through the insulation on the pressure vessel, the plant can be cooled without power and unattended when the decay heat reaches this level. The time at which the 1% level will be reached following a very long operating time at full power has been determined from Fig. 3.19 to be approximately 2-1/2 hr. Thus, the maximum shutdown time is less than 2-1/2 hr when defined in this way. Simultaneous with the 2-1/2 hr cooling-off period, the other steps of the shutdown sequence may be performed. Continuous console attendance will be required for a period estimated to be approximately 30 min. The remaining 2 hr. of cooling where pumping is required need only occasional checks of the system to assure normal operation. As indicated in Section 3.9, no hazardous condition will result even with the complete loss of auxiliary cooling.

7.3.2. Unscheduled Shutdown

Considerable effort in failure mode analysis which is beyond the present scope of work, would be required to determine the procedures to be followed in an unscheduled shutdown of the NEPP. However, a minimum time estimate can be made based on a reactor scram in which the control rods are dropped, since a minimum of 45 minutes is required to withdraw them in the restart procedure. The fault initiating the scram will be identified on the control console by annunciator action. If total loads are reduced to approximately 10 kw, power may be supplied for approximately 1 hr from stored heat in the primary system, otherwise only approximately 10 minutes is available. After these times the M/G set energized by the auxiliary battery would supply power for diagnoses, correction, restart, shutdown, etc., depending on the action required. A time estimate for various actions is shown below.

		Time Required (Min)
1.	Determine that reactor scram has occurred and verify that it was not a spurious action. Observe that rods have been dropped by position indicators.	5
2.	Reduce loads and start M/G set and connect onto line.	10
3.	Assimilate the meaning of the situation and determine alternative actions and priorities that will contribute to the ultimate decision.	15
4.	Investigate (enter TOPS compartment if necessary), correct fault or repair, and replace faulty equipment if possible.	240
5.	Restart the NEPP by withdrawing control rods to the hot critical position.	45
	Total	315 min (or ~5 hr)

Item 4 in the table is obviously the largest and most nebulous, but after 4 hrs in which to recover, it would seem that it would be clear that repair or correction would be successful in the next few hours, or that aborting the mission was required. On this basis, an 8-hours time limit is suggested as a reasonable basis for design.

7.4. MAINTENANCE

The maintenance philosophy established for the TOPS/MUS system will ultimately be reflected in the system reliability or failure rate. In order to determine a detailed maintenance policy for the system, one must know the mode of failure for each active element in the system, its functional relationship to the system, and the allocated reliability for the particular element. Much of this information will be obtained during the detailed design phase of the program.

A preliminary maintenance plan has been prepared and will be used as a point of departure for the detailed maintenance plan which will be generated during the detailed design phase. This preliminary plan is shown in Table 7-1.

Development of a sound maintenance policy is of particular importance to the TOPS/MUS system since the maintainability of the system, once it is implanted, is restricted not only in material and available personnel, but, more importantly, in time available for active repair. The majority of the maintenance actions on the TOPS system will be performed in the period between operating cycles of the TOPS/MUS. The full extent of these maintenance actions will be determined by the maintenance plan to be generated in the design phase.

Maintenance of the primary system water conditions must be considered. The primary system will be hermetically sealed and will be operated for 30 days or more without chemical additives or pH control. If it is decided that water samples are required after each thirty-day run, then the procedures indicated under fuel operations (Section 3.6) for opening the loop will be followed. Samples of water and crud will be withdrawn for analysis and equal volumes of water replaced. The pressure vessel cover gas must also be repressurized with hydrogen. If more extensive maintenance than taking water samples is determined to be required, primary loop water will be circulated through an external treatment plant which must be aboard the tender. (Without this requirement, the water treatment plant is restricted to dock or shore-based facilities.) Decay heat and radiation fields may also limit active maintenance time for significant periods for some equipment. On a quarterly basis the primary circuit will be opened and the water reprocessed. Other maintenance actions will include calibration checks on the primary system instrumentation and pressure-relief devices. Some of these will

Table 7-1

PRELIMINARY MAINTENANCE PLAN

System	Each Dive	30 days	Quarterly	Annually
Primary System				
Core				Fuel element change after 3.1 Mw-yr
Control drives		Visual check	Operational check	Remove, disassemble, and inspect at refueling
Water		Check water level and purity	Process core water	
Pressure relief system		Visual inspection	Check pressure relief system for operation	
Pressure vessel	Visual leak check		He leak check all seals	Decontaminate vessel at fuel change
Miscellaneous valves	Visual leak check	He leak check fill and vent system seals		
Instrumentation			Calibrate and replace, as required	
Secondary System				
Steam separator		Visual check for leak		Clean out collected solids
Trap (steam)		Check for leak		Disassemble and clean

Table 7-1 (Continued)

System	Each Dive	30 Days	Quarterly	Annually
Steam stop valve	Operational check	Check packing		Remove and inspect motor
Turbine and gearbox	Inspect visually; check oil; drain water	Change oil; check linkage	Check operation on external steam supply	Clean governor; replace after 3 yrs; disassemble, check clearances, and reassemble
Generator	Visual check	Clean exterior	Lubricate bearings; electrical check	Disassemble and inspect
Gland seal system	Visual check		Check gland pressure setting	Clean and inspect regulator
Air ejector		Visual check	Recheck steam flow	Check orifice and ejector for erosion
Condenser	Visual check		Calibrate instruments and check for leaks	
Feed pumps	Visual check	Adjust gland lube bearings	Motor electrical check	Disassemble and inspect
Demineralizers	Visual check	Replace cartridge		
Bypass valve	Operational check	Check packing		Remove and inspect
Orifices		Visual check for leaks	Calibrate flow	Remove and inspect
Miscellaneous valves and piping	Visual check			
Water		Sample for pH and solids; adjust additives	Reprocess or replace	

Table 7-1 (Continued)

System	Each Dive	30 Days	Quarterly	Annually
Auxiliary Systems				
Air-cooling system		Visual check	Electrical check; check drains; check louvers	Remove, clean, and replace
Makeup water system	Check level	Sample water	Check operation of valves	Clean tank
Catch tank	Remove any water collected			Clean and inspect
Void tank	Remove any water collected			Clean and inspect
Inner shield tank	Check level		Sample water	Replace water and purge system
Outer shield tank	Check level		Sample water	Replace water and purge system
Hatches				
22-in.	Visual check			Leak check
30-in.	Visual check			Leak check
Electrical				
Switchgear		Visual check	Operational check	Clean and inspect
Electrical connections		Visual check		Clean and inspect
Miscellaneous instrumentation			Calibrate and replace, as required	

be required by the manufacturers and some by the results of the safety analysis. It is expected that detailed control rod drive inspection will be conducted at very infrequent intervals because of their low demonstrated failure rate.

In the secondary,or power conversion loop, water samples will be taken at shutdown and again prior to the next startup. Chemical additives will be injected at each of these times to control water conditions and purity, and feed equipment for chemical additives will be recharged. The full-flow demineralizer will require periodic replacement. Replacement once every thirty-day cycle is considered to be the most frequent rate anticipated; replacement once every ninety days is the minimum design objective. Liquid-level checks and additions will be made to the make-up and storage tanks as required. Pumping of the bilge water overboard may also be necessary, depending on the rates of water escape from the TOPS system. Other secondary loop components will require maintenance as specified by their manufacturers. Such items might include inspection and replacement of bearings, packings, lubricants, electrical contacts, vacuum tubes, belt drives, etc.

Other equipment in the TOPS hull may require only the maintenance indicated by their manufacturers. However, the expected high relative humidity in the hull will require the use of components that have been designed for use in such an environment. Some maintenance may need to be performed to maintain the full level of environmental protection of components, such as replacement of gaskets or repainting surfaces.

Since the facilities and equipment, either shore-based or on tenders, are used primarily between MUS cycles, the maintenance on such equipment will be performed during MUS dives rather than between them. In addition to the normal maintenance requirement of manufactured equipment, disposition of contaminated and radioactive materials,and decontamination of equipment will be one of the principal tasks of the support facility.

7.5. SUPPORT FACILITIES

Support facilities and equipment have been identified as follows:

A. Commercial Facilities

1. Fuel fabrication,
2. Fuel reprocessing,
3. Radioactive Waste Disposal Service.

B. Navy Supplied, Joint TOPS-MUS Facilities

1. Docks,
2. Tender ships,
3. Instrument calibration & repair shop,
4. Repair machine shop,
5. Auxiliary power source (30 kw, 125/216 Y-volt, 3Φ, 60 Hz),
6. Towing vessel.

C. TOPS Support Facilities

1. Water treatment plant on-dock, buoy, or tender,
2. Radiochemical and water laboratory for water and contamination analysis,
3. Decontamination equipment and waste storage,
4. Five-ton crane for handling shielded containers during loading and unloading of TOPS reactor cores,
5. Shipping, storage, and handling shielded containers.

Personnel to man the support facilities are listed below. Facilities and equipment items under A will be performed by separate and independent commercial vendors and personnel for these items have been omitted from the tabulation given. Personnel required to operate the support facilities are:

1. Towing and tender ship crews;
2. Machinists (3);
3. Electronics and instrument technicians (2);
4. Crane operator (1);
5. Radiochemist (1);
6. Chemical plant technicians (6);
7. Stevedores (5);
8. Health Physicist.

8. NUCLEAR SAFETY

Inherent in the TOPS/MUS design is nuclear safety through inherent characteristics of the system in preference to the use of engineered safety systems employing supplemental systems. Thus the TOPS/MUS reactor is a logical utilization of the well-known TRIGA reactor technology. There are currently 34 TRIGA reactors in operation with an outstanding record of safety attesting to the effectiveness of the inherent safety features of the U-ZrH-H_2O, natural convection cooled TRIGA core,and to the conservative design and good quality control in manufacture.

8.1. SAFETY FEATURES AND CHARACTERISTICS

Major features of the TOPS/MUS system which contribute to the safety of the plant arise from the physical properties of the U-ZrH fuel material, the nuclear properties of the U-ZrH-H_2O core, from the thermal characteristics of the primary loop, and from the elimination of the necessity for pumps either for primary loop circulation or for afterheat removal.

During in-pile transient fuel tests at Gulf General Atomic, U-ZrH fuel material has been repetitively pulsed to peak adiabatic temperatures in the range from 1100° to 1200°C without creating an unsafe condition. The U-ZrH fuel material has a high heat volumetric capacity (5400 joules/cc from 23° to 1200°C) and occupies a relatively large volume (79 liters of U-ZrH in the TOPS/MUS), so that it is capable of absorbing a large energy release (1.1×10^{8} joules) without exceeding a peak adiabatic temperature of 1200°C, a temperature which tests have demonstrated is not an unsafe condition. It should also be noted that this temperature is well below the melting point of the fuel or clad materials. The maximum fuel temperature during steady-state operation at the design point is 423°C. Consequently, there is a large factor of safety at the design point.

The U-ZrH-H_2O core lattice provides: (1) The large prompt negative fuel temperature coefficient of reactivity, which depends on the interplay of moderating characteristics of the U-ZrH fuel bodies and that of the interstitial water, and (2) The separation of the smaller water temperature coefficient from the fuel temperature coefficient, which dichotomy provides a means for reducing the excess reactivity allowance needed in the cold, clean conditions to compensate for the reactivity lost in raising system temperatures to the operating point. It should also be noted that U-ZrH fuel temperatures are relatively low when used in a water-cooled system, and low U-ZrH fuel temperatures are more compatible with long core life than are high fuel temperatures associated with other types of reactor coolants.

The heat transfer from the reactor core to the steam generator is by natural circulation of water and no primary coolant pump is required. Even if partial loss of coolant occurred to the extent that the steam generator became uncovered by coolant, efficient heat transfer would take place by condensing primary coolant steam on the steam generator tubes. Nevertheless, if during normal TOPS/MUS conditions the heat transfer to the steam generator were to cease entirely and no change in control rod position were made, reactivity coefficients would compensate for the loss of heat load at a primary coolant temperature of approximately 286°C at zero net power corresponding with a primary coolant pressure of 1020 psi.

Similarly, the removal of afterheat requires no active mechanism but relies on thermal conduction through the reactor vessel and thermal insulation, and subsequently by natural convection of shield tank water, as discussed in Section 3.10.

8.2. POTENTIAL ACCIDENTS

A set of potential accident situations for the TOPS/MUS is reviewed in this section. No nuclear transient accident was conceived that could represent a safety problem either to a surrounding populace or to the operating crew.

8.2.1. Maximum Reactivity Accident

The maximum amount of excess reactivity available at any time during the life of the reactor core is calculated to be $5.48. The initial core loading will be shimmed to provide the design reactivity requirement. Nuclear analysis of the reactor, described in Appendix A, shows that the cold core can tolerate a maximum excess reactivity of $7.00 in a single fast step, without exceeding a maximum transient adiabatic temperature of 1200°C, a temperature that U-ZrH fuel material has been tested to and demonstrated to be acceptable. The power and temperature change with time for this case is shown in Fig. A-3 in Appendix A. Consequently, the reference design has a margin of $1.52, which is adequate to account for reasonable design refinements during the detailed design phase without incurring a change in the safety analysis for maximum reactivity insertion.

No mechanism has been identified capable of inserting the maximum reactivity in a single step. The control rod withdrawal rates are limited by control system design to full insertion of $5.48 in 40 minutes, in connection with startup procedure simplification. The control system will stop withdrawal of control rods when the primary system reaches a pressure of 865 psia and a scram will occur if the pressure reaches 1055 psia. If both of these safety actions fail to occur and if, in addition, the reactor operator also has failed to terminate rod withdrawal, then the maximum effect would be as follows. (Since three failures are required, this accident is not considered to be credible.) Nuclear analysis, as discussed in Appendix A, shows the maximum power level for an unterminated control rod withdrawal to be 2.2 Mw(t), which is less than the 3.1 Mw(t) basic thermal design capability. The increase in reactor water temperature is monotonic, as shown in Fig. A.4, and approaches a temperature of 360°C. The pressure relief system would intervene, however, at a pressure of 1100 psia (290°C), and the reactor would continue to generate power at a low level in equilibrium with the thermal power carried out through the steam relief, so that a significant amount of response time would be available in which to accomplish a reactor shutdown.

The maximum step increase in reactivity conceived could occur during refueling operations if a control rod were to be inadvertently pulled from the core while removing a control rod extension rod subassembly. In this case, the energy release would be about 5 Mw-sec for a control rod worth of $1.10. The minimum cold water depth over the core is 7 ft. If the operations were conducted for the minimum depth of water, then the total integrated radiation dose to the crew engaged in the operation would be less than 2 rads, so that accidental removal of a control rod during refueling does not constitute a major hazard.

8.2.2. Loss-of-Coolant Accident

In principle, a major loss of reactor coolant could occur either at a low point in the reactor vessel or at a high point. There are no penetrations or connections to the vessel below the support skirt, and a break below that point is considered to be exceedingly improbable. However, if it did occur at the design point temperature, all of the reactor coolant could be forced from the reactor vessel at a rate dependant on the size of the opening. Primary coolant water leaving the vessel would partially flash into steam and finally come to an equilibrium temperature and pressure. The maximum pressure in the power plant hull in this case would be about 125 psia, corresponding with a steam temperature of about 171 $^{\circ}$C. The core would be uncovered and become sub-critical. The strength of the pressure hull as a containment vessel is fully adequate for containing this pressure.

The more plausible type of failure, which is unlikely, however, is a failure at the top of the vessel where the penetrations and attachments are made. In case of complete release of pressure, primary coolant would flash into steam as before, and could produce a maximum pressure up to 125 psia in the hull, depending on the rate of release compared with the cooling mechanisms in the hull. Subsequently, the reactor would continue to boil the remaining water in the core until finally the water level is reduced to the level of the top of the core. If shutdown were not effected during the interim period, the loss of reactivity due to loss of water over the top of the core would shut down the reactor, and afterheat would be removed first by the water and subsequently by free convection directly to the ambient air or steam. It is believed that a detailed analysis will show that in this case, maximum fuel temperatures would remain sufficiently low so that fission products would not be released.

8.2.3. Primary Coolant Leakage

If a leak were to occur in the primary system, coolant would be lost either through the pressure relief into the outer shield tank, into the secondary coolant system through the steam generator, or into the air space. Operation of the pressure relief is annunciated in the habitat. Otherwise, all primary coolant system leaks must be detected by one of several indirect means including the radiation level measured over the reactor vessel, the airborne activity measurement, unusual changes in the temperature and pressure of the condenser, the liquid level of the condenser hot-well, or in the liquid level of the make-up tank. If the leak rate were sufficiently large, unusual changes in the temperature and pressure of the air space would be measured.

Except for a major break in the reactor vessel, for which the implications are discussed in the preceding sections, the type of single-failure providing the maximum leak rate would be complete severance of a steam generator tube. The maximum flow through the two opened ends of the 1/2-in. steam generator tube would be 3300 pph, assuming critical flow. If permitted to continue unchecked, 1.45 hr would be required before the primary coolant liquid level would lower to the top of the core. In such a case, the reactor operator would first be alerted by annunciation of an increase in radiation level in the power plant hull over the reactor corresponding with a reduction in water level of one ft. If the flow rate were the maximum, the additional heat loss into the condenser would be 1.2×10^6 Btu/hr, or about 43% of normal full load, so that if it occured while the plant was operating at full load, this would provide additional overpressure and overtemperature indication. The operator's procedure would be to shut down the reactor, but continue operation of the feedwater flow to cool the primary system and to return water into the reactor vessel through the tube failure when the primary system pressure has been reduced below that supplied by the feedwater pump.

8.2.4. Station Upset

If the MUS were to be completely inverted accidentally while operating at full power for a long period of time, the reactor core would be uncovered, become subcritical, and the fuel temperature could rise due to heating by fission product decay energy. If removal of afterheat by natural convection of the steam were not sufficient to prevent fuel clad failure due to excessive internal hydrogen gas pressure, then fission products could be released into the reactor vessel but not into the hull. In this case the reactor core would be damaged, but no radioactive release from the reactor vessel would occur so that including the containment provided by the hull, any possible release would be doubly contained.

8.2.5. Equipment Malfunctions

The primary and afterheat removal systems have no moving parts except for the control rod drives. The control rods are sized so that if a rod cannot be inserted, the remaining five can shut down the reactor.

Failure of the complete secondary system or any component, cannot compromise the safety of the nuclear system, since, with complete loss of load, the reactor coolant pressure will slowly increase to a safe maximum equilibrium pressure of about 1,000 psia. Further protection is provided by annunciation to the operator of high primary coolant pressure and additional protection is provided by the subsequent automatic high pressure shutdown if action were indicated and not taken.

9. EQUIPMENT LIST AND WEIGHTS

An equipment list for the TOPS/MUS system is given in Table 9-1. This list includes the major pieces of plant equipment and is divided into 6 major categories of equipment.

All of the equipment shown will not be in the TOPS/MUS system at the time of a dive, but some will be required to service the system between dives.

Table 9-2 gives a compilation of the weights of various pieces of equipment which will be in the TOPS/MUS hulls during a dive, and the calculated center-of-gravity of each, with respect to the theoretical center of radius of the lower hemihead. This information will be used by the hull designer to estimate the center of gravity of the entire MUS system.

Table 9-1

EQUIPMENT LIST

	Number required
Primary Equipment	
Reactor tool (set)	1
Reactor vessel	1
Steam generator assembly	1
Core support	1
Grid plate, lower	1
Grid plate, upper	1
Safety plate	1
Internal shield	1
Fuel elements	193
Control rods	6
Control rod drive assembly	6
Pressure burst disc assembly	1
Bleed valve	1
Relief valve, primary	1
Fill valve	1
Vent valve	1

Table 9-1 (Continued)

	Number required
Stud tensioner	1
Inner shield tank	1
Refueling Equipment	
Fuel cask (19 elements)	1
Chain hoist (7-1/2-ton)	1
Fuel handling tool (set)	1
Submersible pump system (portable)	1
Fuel cask cover	1
Radiation monitor (portable)	2
Breathing apparatus (portable)	3
Water treatment facility	1
Secondary Equipment	
Mechanics tool set	1
Auxiliary steam supply and test equipment	1
Shore test valve, inlet	1
Shore test valve, outlet	1
Steam separator	1
Steam stop valve	1
Steam gland pressure regulator	1
Steam gland relief valve	1
Steam turbine	1
Gear reducer and lube system	1
Generator	1
Turbine bed plate	1
Relief valve, secondary	1
Air ejector system	1
Steam trap	1
Water trap	1
Demineralizer, secondary	1
Feed pump	2
Startup valve	1
Electrical Equipment	
Switchgear A	1
Switchgear B	1
Submarine cable	4
Nuclear system console	1
Power distribution console	1
Electrical test equipment set (portable)	1

Table 9-1 (Continued)

	Number Required
Auxiliary Systems	
Demineralizer, makeup	1
Fan	2
Barometric louver	2
Fresh-water pump	2
Fresh-water heat exchanger	1
Generator air cooler	1
Fresh-water filter	1
Air monitor, portable	1
Miscellaneous Tankage	
Outer shield tank	1
Void tank	4
Catch tank	1
Makeup tank	1
Fresh water tank	1

Table 9-2

NEPP EQUIPMENT WEIGHT ESTIMATE AND CENTER OF GRAVITY LOCATION

		Weight (lb)	$\overline{X}$[a]
Primary System			
Reactor vessel and insulation		13,000	7-ft 9-in.
Internals		5,400	1-ft 11-in.
Water		5,610	2-ft 4-in.
Fill valve		8	5-ft 5-in.
Fill line		5	5-ft 8-in.
Vent valve		12	13-ft 1-in.
Relief valve system		120	6-ft 1-in.
Vent and relief line		15	6-ft 0
Total		24,170	
Shielding			
Inner shield tank		6,360	1-ft 8-in.
Lead		53,580	-1-ft 0
Water		14,100	-1-ft-3-in.
Void tanks		1,520	0 0
Water		7,300	0 0
Upper reactor shielding		9,580	9-ft 5-in.
Total		92,440	
Secondary System			
Instrumentation		112	6-ft 0
PS (3)	5		
PX (3)	40		
ΔPX	40		
LLX	25		
TX (2)	2		
Total	112		
Pipes and Flanges		180	13-ft 7-in.
Pipe	150		
Orifice flange (3)	30		
Total	180		

[a] $\overline{X}$ measured in the positive vertical direction from the center of radius of the lower hull hemihead.

Table 9-2 (Continued)

		Weight (lb)	$\overline{X}$[a]
Secondary System (Continued)			
Steam separator		132	15-ft 9-in.
Separator trap		210	15-ft 11-in.
Turbine		1,750	16-ft 11-in.
Gear reducer		1,730	17-ft 4-in.
Generator		1,540	17-ft 5-in.
Bed plate		560	16-ft 0
Air ejector (after condenser)		400	5-ft 0
After condenser trap		45	4-ft 3-in.
Demineralizer		340	5-ft 6-in.
Boiler feed pump (2)		700	4-ft 10-in.
Valves and regulator		135	
V-1	6		3-ft 0
V-2	1		5-ft 10-in.
V-3	1		5-ft 10-in.
V-4	10		6-ft 0
V-5	3		6-ft 3-in.
V-6	3		6-ft 6-in.
V-7	1		4-ft 8-in.
V-8	1		4-ft 8-in.
V-9	1		4-ft 10-in.
V-10	5		16-ft 7-in.
V-11	1		17-ft 8-in.
V-12	40		16-ft 10-in.
V-13	15		19-ft 6-in.
V-14	12		17-ft 3-in.
V-15	1		16-ft 8-in.
V-16	20		20-ft 2-in.
V-17	3		20-ft 6-in.
V-18	1		19-ft 0
V-19	1		4-ft 6-in.
Pressure regulator	5		17-ft 0
Check valves (2)	4		4-ft 8-in.
Total	135	7,834	

[a]$\overline{X}$ measured in the positive vertical direction from the center of radius of the lower hull hemihead.

Table 9-2 (Continued)

		Weight (lb)	$\overline{X}$[a]
Electrical System			
Switchgear		670	19-ft 6-in.
Enclosures		1,730	19-ft 0
Cable		200	16-ft 3-in.
Lighting and receptacle		400	14-ft 0
Miscellaneous		600	19-ft 0
Reactor control console		750	[b]
Power plant control console		350	[b]
Total		4,700	
Auxiliary Systems			
Air cooling system air blower (2)		30	31-ft 9-in.
PS instrumentation		1	32-ft 0
Vent pipe		100	13-ft 3-in.
Plenum		720	29-ft 6 in.
Drain line		50	14-ft 5-in.
Makeup tank			
Tank		610	0 0
LLX Instrumentation		10	-2-ft 0
Valves		5	5-ft 0
V19	1		
V20	1		
V21	1		
V22	1		
CV3	1		
Total	5		
Demineralizer		150	5-ft 6-in.
Piping		25	3-ft 9-in.
Water		624	-1-ft 0
Catch tank		610	0 0
Fresh water system			
Tank		320	23-ft 7-in.
Pump (2)		200	16-ft 8-in.
Valves		10	16-ft 0
Pipe		30	14-ft 0
Tubing		25	1-ft 4-in.
ΔPS (2)		40	17-ft 0

[a] $\overline{X}$ measured in the positive vertical direction from the center of radius of the lower hull hemihead.

[b] Located in habitat.

Table 9-2 (Continued)

	Weight (lb)	$\overline{X}$[a]
Auxiliary Systems (Continued)		
Fresh water system (Continued)		
LS	10	27-ft 0
Filter	15	17-ft 3-in.
Water	350	23-ft 4-in.
Total	3,935	
Miscellaneous Structure		
Machinery deck and structure	2,200	21-ft 11-in.
Lower support structure	2,650	2-ft 8-in.
Condenser insulation	750	14-ft 4-in.
Total	5,600	
Total (Overall weight)	138,679	

[a] $\overline{X}$ measured in the positive vertical direction from the center of radius of the lower hull hemihead.

10. RELIABILITY

The reliability of the NEPP is expected to be high for the TOPS/MUS nuclear electric power plant. This expectation is based on the use of commercially available components and existing industrial technology. Therefore, no development programs are necessary. However, the quantitative reliability of the system is dependent not only on the reliability of the components and subassemblies included in it, but also the detailed engineering design and integration of components into the system. Thus, quantitative reliability estimates must be worked into the detailed design phase. Nevertheless, considerable attention was given in this study to the reliability and simplicity-to-achieve-reliability aspects of the design and arrangements. A conservative approach has been employed with most critical components selected to operate at derated conditions,or provisions have been made for standby redundancy.

While specific quantitative reliability information is not available, some general reliability information may be obtained by considering generic failure rates and life expectancy of some of the major components of the NEPP in the expected TOPS/MUS environment. The information generated may be used to focus attention on the critical components with respect to obtaining specific reliability data for subsequent design, use of alternate components and/or system arrangements, use of redundant components and provision of stringent manufacturing and quality assurance specifications for component suppliers.

For a preliminary reliability analysis, generic failure rates and life expectancy were obtained from Reference 1*, and supplemented by data from References 2 through 4. Data are presented for laboratory environment and had to be modified by a failure rate multiplier for alternate environments. Mean data for shipboard and ground installation environment are shown in Table 10-1. Comparison of the two environments indicates the principal difference to lie in the mechanical shock to which shipboard equipment is subjected. This results in a failure rate multiplier to the laboratory environment failure rates of 15,versus 8 for ground installations. If it is assumed that most of the mechanical shock intensity occurs as a result of surface wind and wave action (rather than dock impact or weapon blasts), then the TOPS/MUS environment while it is stationary on the ocean floor should be more like that of a ground installation since most of its operational life will occur there. During surface towing from port to

*References are listed at end of this section.

Table 10-1

ENVIRONMENTAL PARAMETERS FOR RELIABILITY DATA

	Environment	
	Shipboard	Ground
Temperature ($^{\circ}F$)	40	40
Vibration (g, rms)	3	2
Shock (g, rms)	28	1
Humidity (%)	83	70

implantation sites equipment may be secured to prevent damage. A laboratory multiplier of 11.0 was estimated to be a conservative value to apply to the laboratory environment failure rates under these circumstances.

The relationship among the failure rates of generic components used in the TOPS system is shown in Fig. 10.1. Mean-time-between-failures (MTBF) for the 9 most unreliable major components is given in Table 10-2.

One can easily see that the generator, which is used functionally in a nonredundant manner, will be one of the major controlling factors in system reliability. For this reason the generator selected for the system will be obtained from a quality manufacturer of known reputation in the field, and the particular piece of equipment selected will be the most reliable unit available.

Another important item in the system is the motor-driven feed-water pump. Here the method of standby redundancy has been employed in order to increase the resultant over-all system reliability. The MTBF is approximately doubled by the single standby unit of similar design and reliability. Information in Fig. 10.1 suggests that solenoid valves might be replaced by motor valves since the latter's failure rate is $\sim 1/3$ that of the solenoid valves.

Since the reliability of any system depends both on the complexity and functional relationship of equipment, the design was conceived to eliminate all systems or subsystems possible in order to enhance the system reliability. This has been done to the extent that, where necessary, system efficiency and weight has been sacrificed for system reliability.

The reliability of the primary system should be extremely high since almost all dynamic systems have been eliminated. The control rod drives selected have already been tested without failure for lifetimes in excess of that expected for the system. In addition, the system can function with one inoperative control rod out of six. From the use of the binomial theorem[5] for the reliability of operating parallel system, the reliability of the control rod drive subsystem can be expressed as

$$R_s = 6R^5 - tR^6 \quad ,$$

where R is the reliability of the individual control rod drive. The non-failure test time of 10^5 ft of travel will give an equivalent control rod drive reliability for a 30-day mission of 0.9959 with 90% confidence assuming a random failure mode.

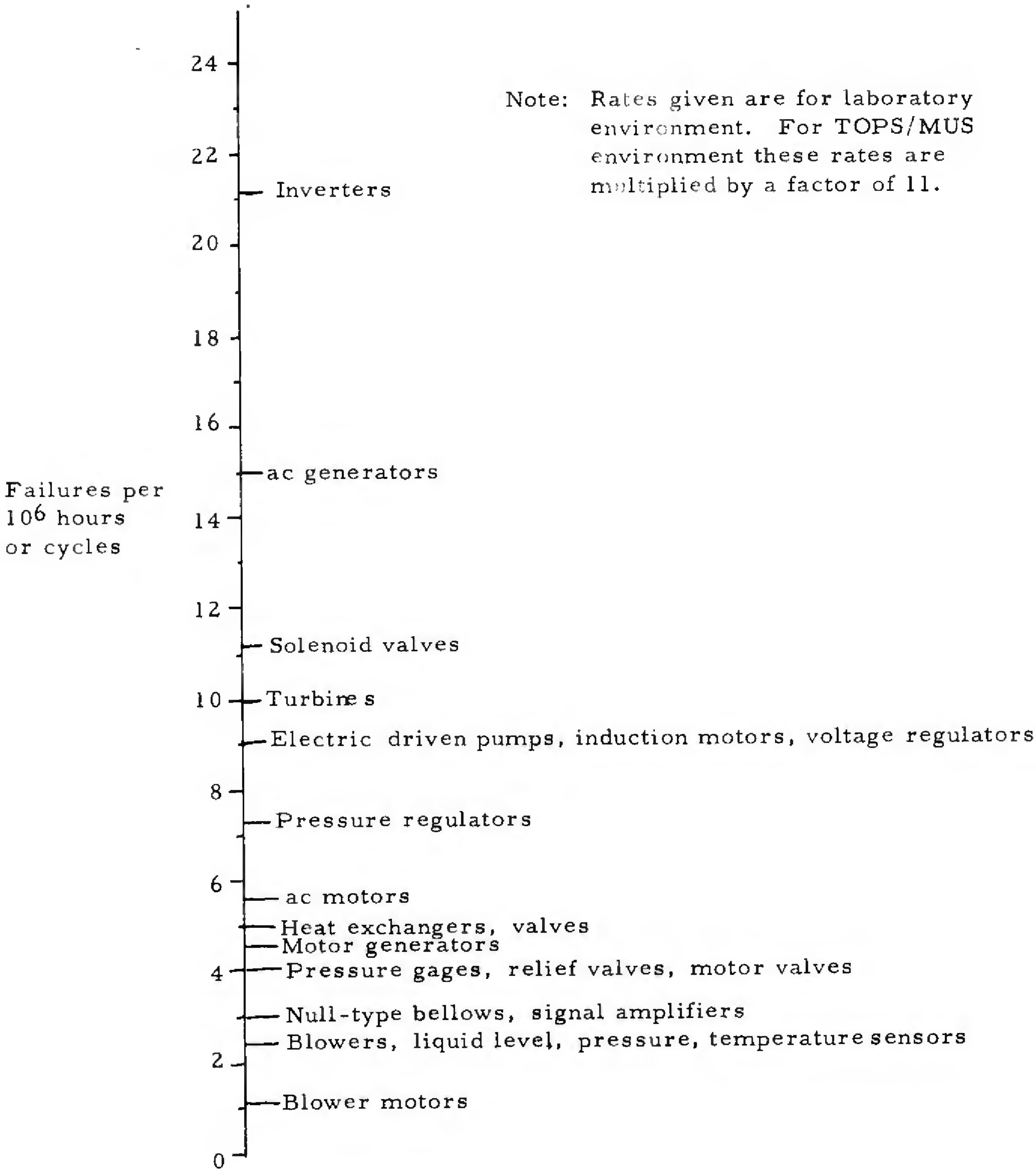

Fig. 10.1--Generic failure rates for typical components

Table 10-2

MEAN-TIME-BETWEEN-FAILURES OF GENERIC TOPS COMPONENTS

(TOPS/MUS Environment)

		MTBF (hr)
1.	ac generators	6,060
2.	Solenoid valves	8,250
3.	Turbines	9,100
4.	Voltage regulators.	10,200
5.	Electric-driven pumps . . .	10,340
6.	Pressure regulators	12,100
7.	ac motors	17,100
8.	Heat exchangers	18,200
9.	Valves.	18,200

The reliability of TRIGA fuel elements is very high. The failure mode to be expected in the TOPS/MUS application is leaking of the cladding which is not catastrophic, nonpropagating, and not even serious with respect to corrosion or fission gas release. Since such failures do not result in loss of powerplant performance nor do they cause the mission to be aborted, the reliability of the fuel element is considered to be essentially 1.0.

The reactor pressure vessel will be designed, manufactured, inspected and maintained in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels. Since the code by necessity is conservative and incorporates substantial factors of safety, and stringent manufacturing and inspection specifications, the pressure vessel is expected to have a high reliability for both mission and plant lifetimes. Although there has been repair and maintenance required of vessels built under this code, there are no known failures which would result in an aborted mission.

The steam generator and pressure relief burst disk are regarded as the primary system components that essentially establish the overall reliability of the primary system. Assuming the burst disk failure rate to be $\leq$ that of typical bellows because of general fundamental similarities, its rate is $\sim 1/2$ that of the steam generator (heat exchanger). For a mission time of 1 month, the reliability of the steam generator with 90% confidence is 0.955 or better. If further increases in reliability are required, the steam generator design can be modified in such a way that it is divided into two independent tube banks with separate tube sheets and pressure vessel nozzles. In this circumstance isolation stop valves would be required at the inlet and outlet of each tube bank. In addition, each bank would require a pressure relief valve to accommodate expansion of water if any became trapped between closed valves. Thus, seven additional motor-operated valves would be required.

A better reliability estimate of the TOPS/MUS system will be conducted as part of the detailed design phase and should include failure mode analyses; component reliability allotments to achieve system goals and definition and implementation of a quality assurance program.

REFERENCES

1. Earles, D. R., and M. F. Eddins, "Reliability Engineering Data Series: Failure Rates," AVCO Corporation, Wilmington, Massachusetts, April 1962.

2. RADS Reliability Notebook, PB-161894, U. S. Department of Commerce, (RADS-TR-58-11), Revised, January 1963.

3. Reliability Stress and Failure Rate Data for Electronic Equipment, Militiary Standardization Handbook, MIL-HDBK-217, August 1962.

4. Failure Rate Data Handbook, (SD 63-470), Bureau of Naval Weapons, June 1962.

5. Bazovsky, Igor, Reliability Theory and Practice, (Prentice-Hall Space Technology Series, 3rd Ed.,) March 1963.

11. ADAPTABILITY

The TOPS/MUS power plant is adaptable to other power levels and other applications. The reference design described in this report is divided into four parts which include the TOPS nuclear steam generator module, the power conversion equipment module, the hull-condenser and feedwater systems, and the control consoles. The hull-condenser is specific to the MUS application so that a substitute condenser would be required for another application without significantly altering the remaining units. Clearly, rearrangement of the respective modules may be made in another application provided; that accounting is made for such things as the secondary steam and condensate flow requirements and for the NPSH requirements of the hot-well effluent into the feedwater pump.

For applications in which the reactor vessel dimensions should be reduced, the use of forced circulations in the primary loop would permit reduction from 17 ft to 8 ft, while still retaining the capability for natural circulation of the primary coolant at a sufficient level to remove after-heat from the core. Refueling would be more difficult and would also make seal-welding of the vessel head a potential problem. Similarly, more efficient compact power conversion equipment can be provided for applications in which the increase in cost is justified.

For application to higher power levels, the primary system is designed to be adaptable with no major changes, except for increased attention to after-heat removal. The secondary system, however, will require a higher capacity condenser and a power conversion system modified, or newly designed, for the higher power level. The instrumentation and control system changes would be expected to be minimal.

In applications where continuity of operation is a primary requirement the steam generator can be made of two parallel arrays of tubes coiled in a double-helix in the manner of a double thread. The principle of redundancy can be extended by having two separate turbine-generator sets connected to the two steam generator arrays. It is also possible to provide for personnel access to the secondary system where maintenance personnel and equipment are available to make use of such access.

The adaptability of the TRIGA power reactor preliminary design presented in this report can be summarized by saying that the primary system components can supply steam having availability equal to or higher than steam at 262 psia/455°F, at rates up to 2930 Btu/sec. These components can be arranged for forced circulation or natural circulation, single or multiple steam generator bundles, and compacted or extended arrangements. The secondary system can use single or duplicated power conversion equipment, regenerative feedwater heating can be used, and different condenser pressures can be chosen to suit the heat sink available for a projected application. Turbine speed can also be chosen to match requirements of an application. It should be noted that thermodynamic cycle efficiency would be improved and the size of rotating machinery would be reduced if the ac frequency were increased above 60-Hz and the turbine speed increased accordingly; at 400-Hz the gearbox could be eliminated. The possible combinations of the variables listed above give a range of applications whose principal limit is the maximum power output.

11.1 COMPACT CONFIGURATION

In those cases for which a more compact reactor vessel would be advantageous, the reactor pressure vessel can be reduced in height, by a factor of two (see Figure 11.1). In such a case, final selection of the vessel height would result from a tradeoff of the benefits of reduced weights and dimensions compared with the increase in the equipment and maintenance required for accommodating fuel handling, top shielding, and secondary coolant activation and radiolysis for close compaction. For operation at power levels exceeding 100 kw(e), natural circulation flow would need to be boosted by the use of a jet pump subsystem activated by a canned rotor pump. Outline drawings are shown in Figure 11.1 of the TOPS nuclear steam supply module for both the natural circulation primary system and for the boosted flow primary system. The taller vessel was chosen as the basis for this design study in order to provide maximum simplicity of reactor operation and maintenance consonant with the special requirements of the MUS application. The reactor configuration is designed with sufficient height to allow natural circulation of the primary coolant for power levels up to 100 kw(e) as well as for after shutdown cooling for all power levels up to 500 kw(e).

The control rod drives would be as described in the text, with some minor changes in configuration, to allow their use in this system. The primary circulation pumps would be canned rotor centrifugal pumps which would operate in parallel at all times. The pumps would supply high-pressure water to multiple water eductors located slightly above the level of the core mid-plane. The area provided for the eductor diffusers will provide sufficient flow space to allow natural convection with the pumps shut down.

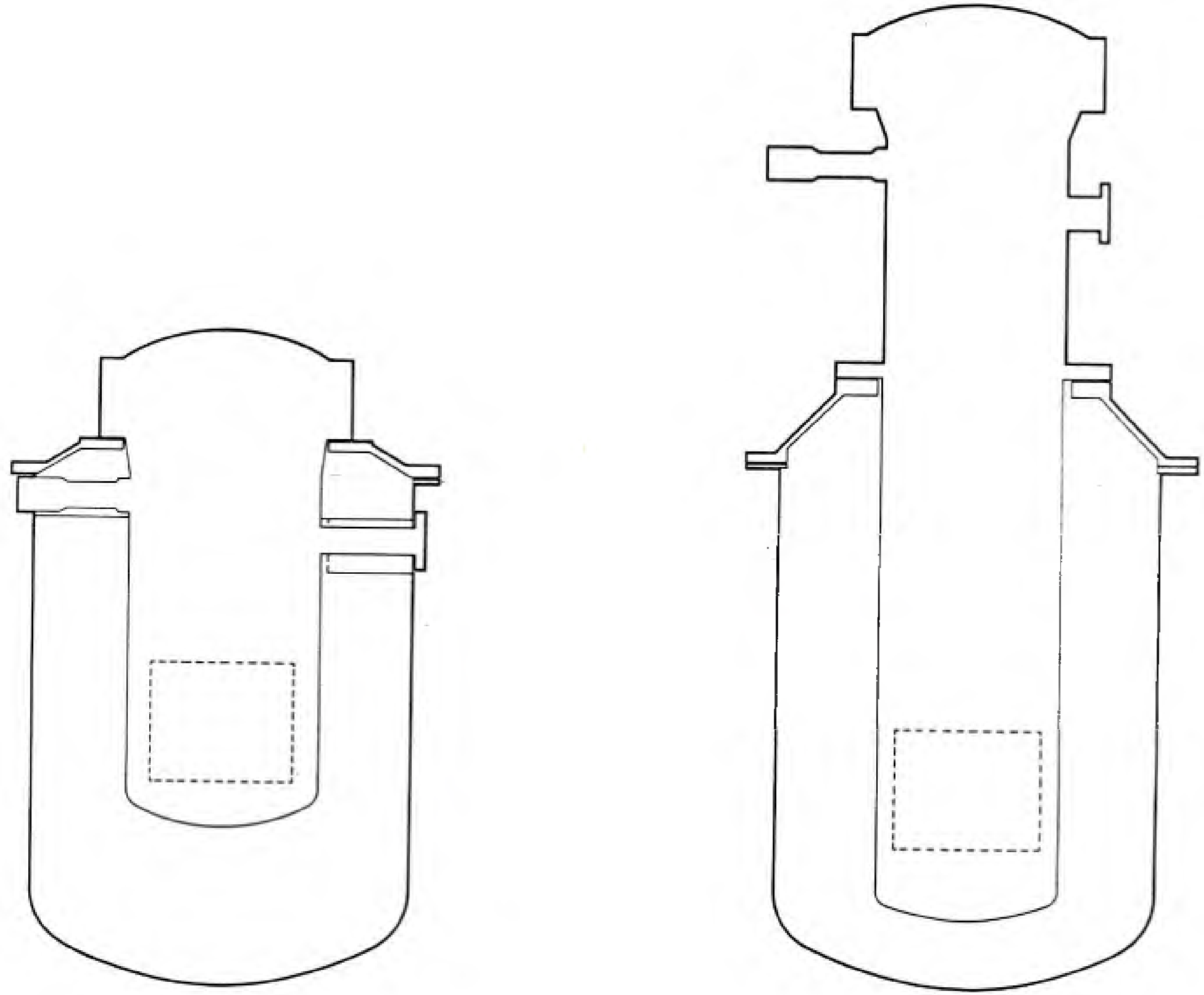

Fig. 11.1--TOPS nuclear steam supply module showing comparative size of basic arrangement and compact arrangement

The secondary, or power conversion system, selected for use with the compact TOPS reactor would be supplied by Sundstrand Aviation. It is of particular interest since it is based on actual existing hardware now in production. The details of the system, as proposed to Gulf General Atomic[(1)*] will not be included here, but a brief system description and performance summary will be given.

The heart of the Sundstrand system is a turbine and gearbox, developed by Sundstrand under contract to Westinghouse Electric Corporation, Undersea Division. This contract is to provide a power plant and tankage for the Mark 48 torpedo, an advanced ASW weapon. The development and prototype phase of this contract has been completed and the production phase is now underway. A photo of this piece of equipment is shown in Fig. 11.2. This photo shows the unit in its torpedo configuration and it therefore contains some equipment which would not be used in the system described here.

The following is a brief description of the major components of the secondary system.

Turbine. The turbine, which rotates at 63,000 rpm, is a single-pass impulse wheel presently in production for the Mark 48 torpedo engine. It is a high-strength forged wheel, capable of running at higher temperatures and with higher pressures than the TOPS application requires. The turbine is integral with its supporting shaft and transmits its power through a small spur gear which is actually the first component of the speed-reduction gearbox. The shaft is supported on oil film journal bearings which provide long life and quiet running.

Gearbox. The proposed gearbox is also a Mark 48 torpedo engine component. Modifications for this job are small; no changes in gear meshes or bearings are necessary, and the housing will also be used in its present form. The gearbox is a multiple-reduction unit, affecting the overall gear ratio through three internal speed reductions. Power takeoff pads are available at the intermediate speeds. The reactor boiler feed pump will be mounted on one of these. Input power from the turbine gear is taken through three spur gears spaced around it. These gears are close-coupled to three helical cut gears, which, in turn, mesh with three companion gears; this provides the second reduction. Splined to these gears are long lay shafts which are designed to "wind up" to a sufficient degree and thereby create equal load division among the gears. The other ends are splined to the output pinions which mesh with the single output gear. The alternator is directly coupled to this gear by means of a suitable coupling. All gears, with the exception of the first mesh, are helical cut with integral thrust runners

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*References are listed at the end of this section.

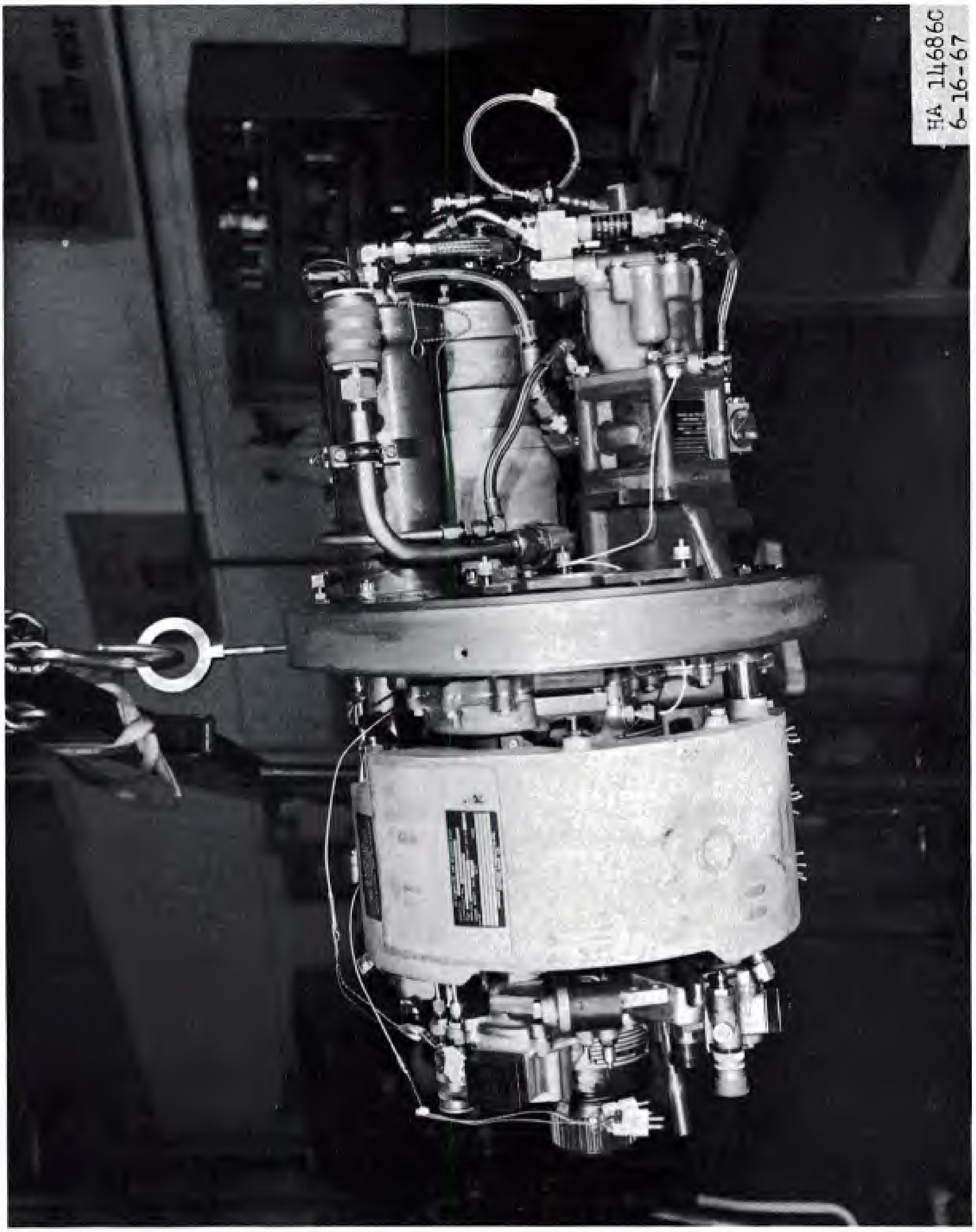


Fig. 11.2--Sundstrand turbine gearbox (Mark 48 torpedo)

to take up the axial loads created by the gears. The gear train is low in both noise and vibration. Rotating members are supported on fluid film journal bearings throughout. Lubrication is by force-feed from a small oil pump. A water-oil heat exchanger provides for oil cooling. The fresh-water-cooling system would be available for the heat exchanger.

Boiler Feed Pump. The TOPS boiler will be supplied with feedwater pumped by a single-stage Sundyne centrifugal pump. This is essentially a Sundstrand shelf item and is found in numerous applications of various jobs. It is simple in design and construction,and has proved reliable over long durations. Lubrication of the bearings will be from the gearbox lube.

Alternator. The alternator selected for this system is the 1800-rpm General Electric unit described in Section 4.1.

A weight estimate for the combined rotating unit is given in Table 11-1. This can be compared with similar pieces of equipment for the reference system.

As shown in Table 11-2, the system performance with the Sundstrand turbine will be considerably higher than with the reference Elliott turbine. The reason for the increase in efficiency is that the 63,000 rpm turbine wheel can operate near peak theoretical efficiency. Additional efficiency is gained over the Elliott because the Sundstrand turbine will be operated as a nearly full admission unit, thus eliminating a large amount of side leakage and windage.

The system component establishing MTBF is the gearbox. The unit proposed uses a production-type gearbox which has not been optimized for the particular speeds and load involved in a MUS-type application so that lifetime is reduced at the 100 kw(e) power level. If this unit is pursued, an evaluation of a gearbox modified design for continuous 100 kw(e) duty might be desirable.

Figure 11.3 shows the expected MTBF for the current Sundstrand systems without any design changes. Since operation of the system will not necessarily be at a mean load of 100 kw(e) (approximately 176 SHP equivalent in turbine power) but at a much lower mean load, the actual MTBF of the system should be of the order of several thousands of hours. Because the exact operating power profile of the MUS is not know, an exact prediction cannot be made.

Although the Sundstrand turbine system was not selected as the reference design, it is felt that the system has considerable merit, and further study of this unit appears warranted. Figure 11.4 shows the overall arrangement of compact high-speed Sundstrand turbine system coupled to a 1800 rpm generator.

Table 11-1

WEIGHT ESTIMATES

		(lbs)
1.	Gearbox and turbine assembly	250
2.	Sundyne pump	40
3.	Generator and regulator	1,540
4.	Controls	10
5.	ac submersible motor	40
6.	Start pump	30
7.	Governor and steam valve	160
8.	Miscellaneous plumbing	200
9.	Base mounting plate	300
	Total	2,570

Table 11-2

EFFICIENCY PREDICTIONS

Type	(%)
Cycle	14.5
Turbine	60
Gearbox	91
Alternator	92
CRU	49

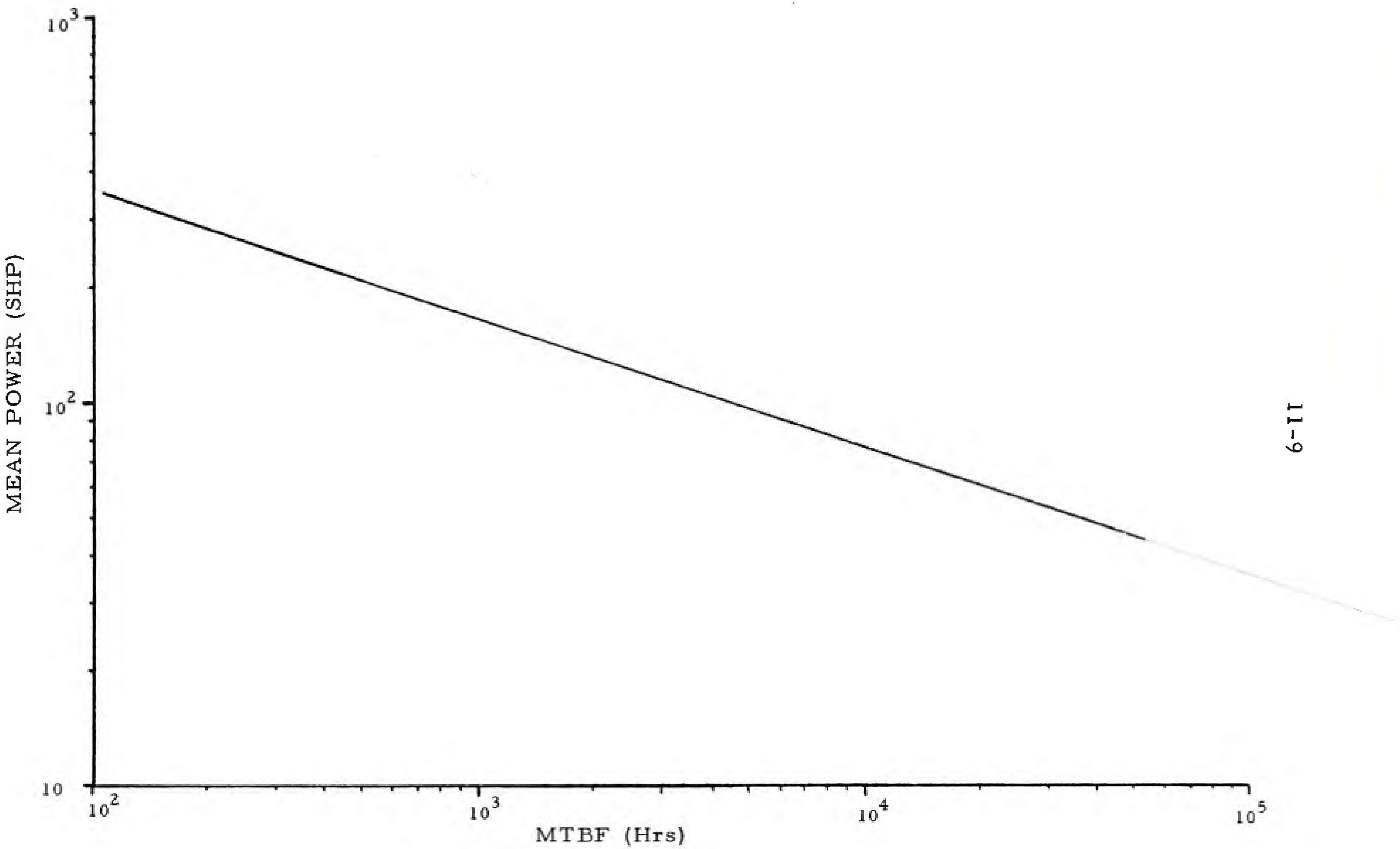


Fig. 11.3--Expected MTBF for high-performance Sundstrand system

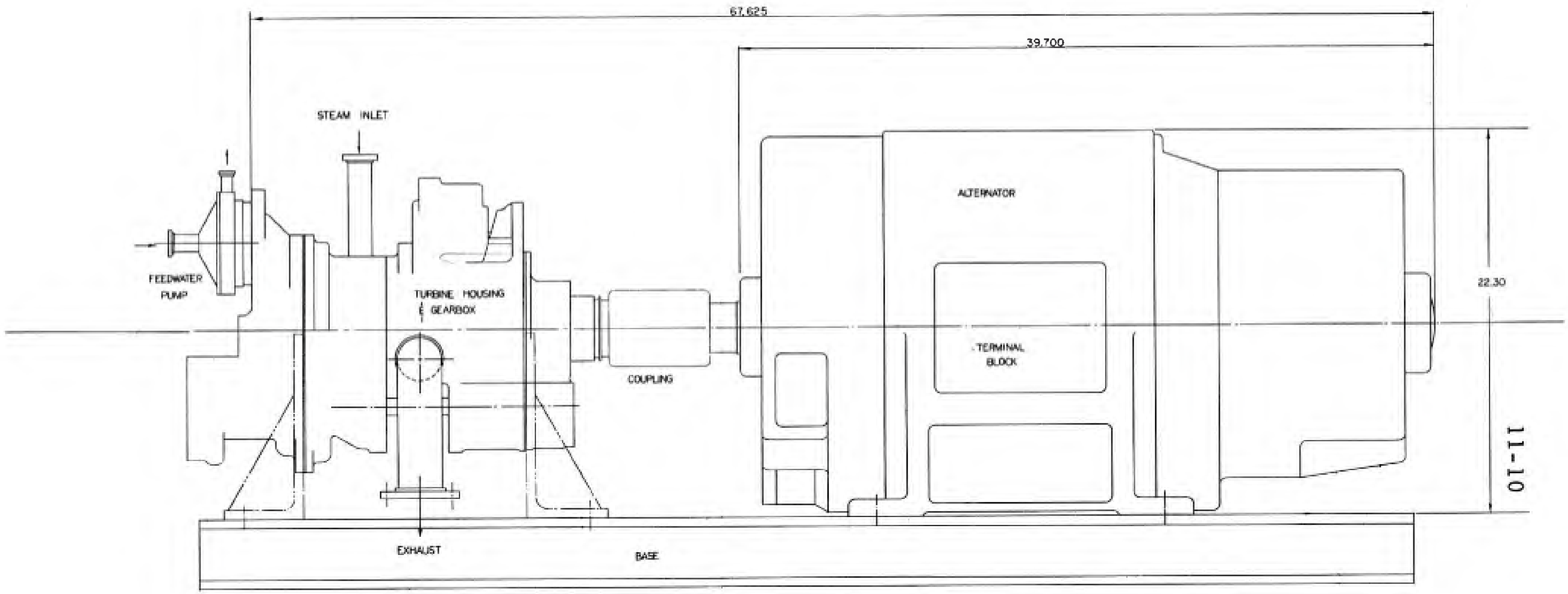


Fig. 11-4--Overall arrangement of compact high-speed Sundstrand turbine system coupled to a 1800-rpm generator

11.2 500 kw(e) OPERATION

The primary loop for the TOPS/MUS has been designed for the capability of generating 3.1 Mw(t) to provide thermal power for a power conversion system delivering 500 kw(e) at an assumed net efficiency of 16.1%.

The secondary loop has not been designed for this power level and in the MUS application the limited condenser surface area would preclude the possibility of 500 kw(e) operation.

For 500 kw(e), a multistage reaction turbine would be needed with interstage water extraction used for feedwater heating. This type of turbine has been used to provide net plant efficiencies in excess of 18% in existing nuclear power plants (e.g., SM-1 and SM-1a) at net power levels just under 2,000 kw(e).

Thermal conditions in the primary loop at 3.1 Mw(t) would provide a core outlet temperature of 498°F, for which the saturation pressure is 670 psia, and the core inlet temperature would be 462°F corresponding with a primary coolant flowrate of 232,000 lb per hour. The maximum fuel element temperature is 660°C, which is compatible with the long-term irradiation properties of the fuel. The secondary steam conditions are 262 psia at 458°F at a flowrate of 8820 lb of steam per hour.

The after-heat removal system (e.g., thermal insulation from reactor vessel to shield tank) would require modification to increase the heat removal to account for the increase in thermal power.

The reactivity requirements for this power level are given in Table 11-1 and total to a maximum value of $9.92. This excess reactivity requirement is the same as that approved by the USAEC for an existing TRIGA open-pool reactor of approximately the same core size (TRIGA/ACPR). As in the TOPS/MUS application, no method has been conceived by which a step change in reactivity can be made greater than for a single control rod which would have a worth of $2.00 in compliance with the "stuck rod" criterion and could not cause any core damage. The control rod diameter in this case would be 2.00 inches. For a rod withdrawal rate selected to provide the same rate of reactivity insertion as in the TOPS/MUS application, the power and temperatures would also change in the same manner (see Fig. A-4 in Appendix A) until if unchecked by either the operator, or by either of the two safety actions controlled by primary coolant pressure level, the pressure would reach a level at which the pressure relief system would function and coolant would begin to boil off from the primary system until the level is reduced to where the core becomes subcritical. Subsequently, fission product decay heating would be removed by heat transfer through the reactor vessel, insulation, and the shield tank. It appears reasonable that a detailed analysis of this accident would show that fuel temperatures would not be sufficient to cause a break of fuel cladding so that fission products would not be released. Even if this were not so, the loss of criticality and the attendant power reduction would cause a primary system pressure reduction to permit closing the pressure relief valve so that a barrier would be restored to the flow of gases from the reactor vessel.

11.3. 400 CYCLE POWER

There are several very definite advantages to the selection of 400 Hz power for the TOPS/MUS system. The main advantage is that use of 400 Hz equipment allows the flexibility of using high-speed rotating equipment with its inherently smaller volumes and lighter weights. Another advantage would be a somewhat greater overall system efficiency compared with that enjoyed by the 60 Hz design which comes from two sources:

1. Elimination of the turbine gearbox speed reducer,

2. Use of high efficiency auxiliary equipment.

The 400 Hz system which would be proposed for use in the TOPS/MUS system could utilize the Elliott AYR turbine. This unit is similar to the CYR described in Section 4. In the 400 Hz system, the turbine would operate at 8000 rpm and would be directly coupled to a Westinghouse 6QM312B (420-Hz at 8400-rpm) air-cooled, brushless alternator, a drawing of which is shown in Fig. 11. 5.

The Westinghouse alternator is a 312KVA unit which was originally designed to U. S. Navy specifications for shipboard use. The unit is designed to have a service life in excess of 20, 000 hours. This means that the combination of Elliott turbine and Westinghouse generator could be used for powers up to 250 kw(e) with high reliability.

The combination of this turbine and generator would weigh in at about 1225 lb compared to 5020 lb for the equivalent 60 Hz equipment selected. In addition, the package would occupy a cube of space about 80 in. long by 40 in. wide by 34-1/2 in. high, compared to 123 in. long by 40 in. wide by 34-1/2 in. high. A comparison of the two systems is shown in Fig. 11.6.

A gain in efficiency can be realized by the elimination of the gearbox which is 98% efficient, and also by the reduction in power required by the boiler feedpump.. This can also be reduced by using a J.C. Carter single-stage, high-speed pump P/N 6353-4, shown in Fig. 11.7, which is driven by a close-coupled 12, 000 rpm, 4-pole, 400-Hz motor. The pump head/flow characteristics are shown in Fig. 11.8. This unit requires about 5. 2 SHP at the design point compared to 8.3 SHP for the SIHI.

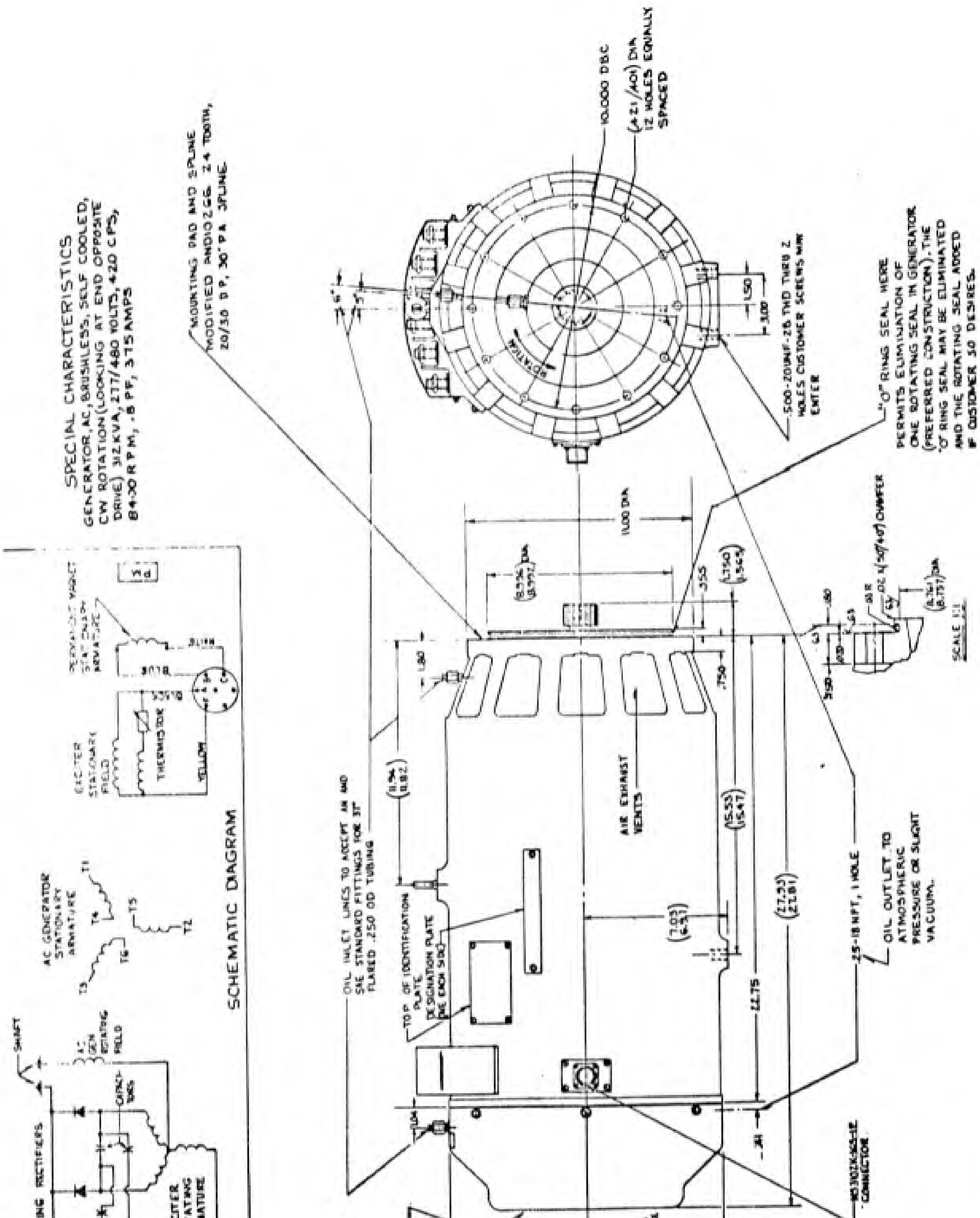
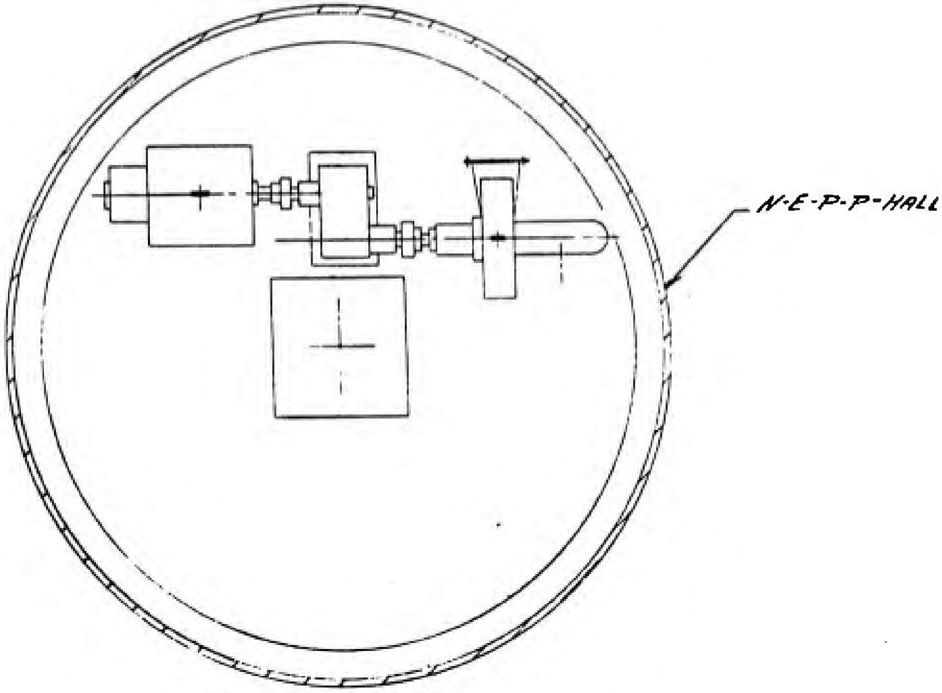
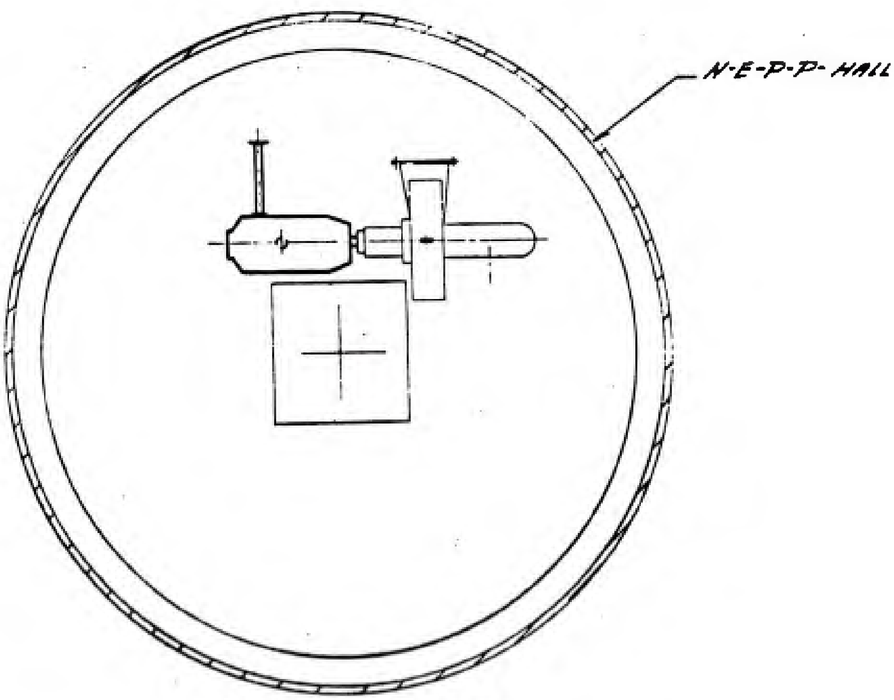


Fig. 11.5--Westinghouse 400-Hz generator



60 Hz SYSTEM



400 Hz SYSTEM

Fig. 11.6--Size comparison of 60-Hz and 400-Hz systems

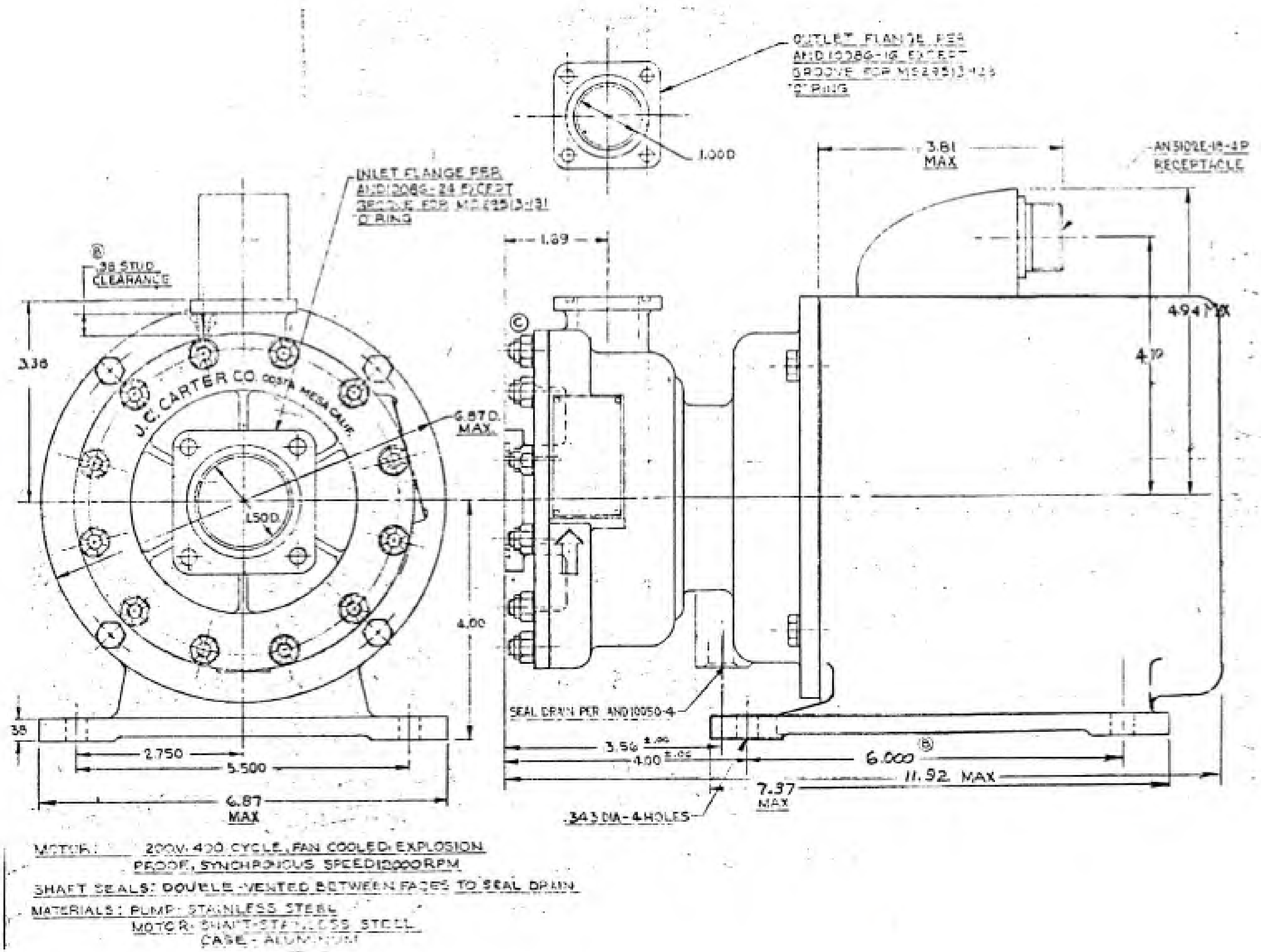


Fig. 11.7--J.C. Carter single-stage, high-speed pump
(P/N 6353-4)

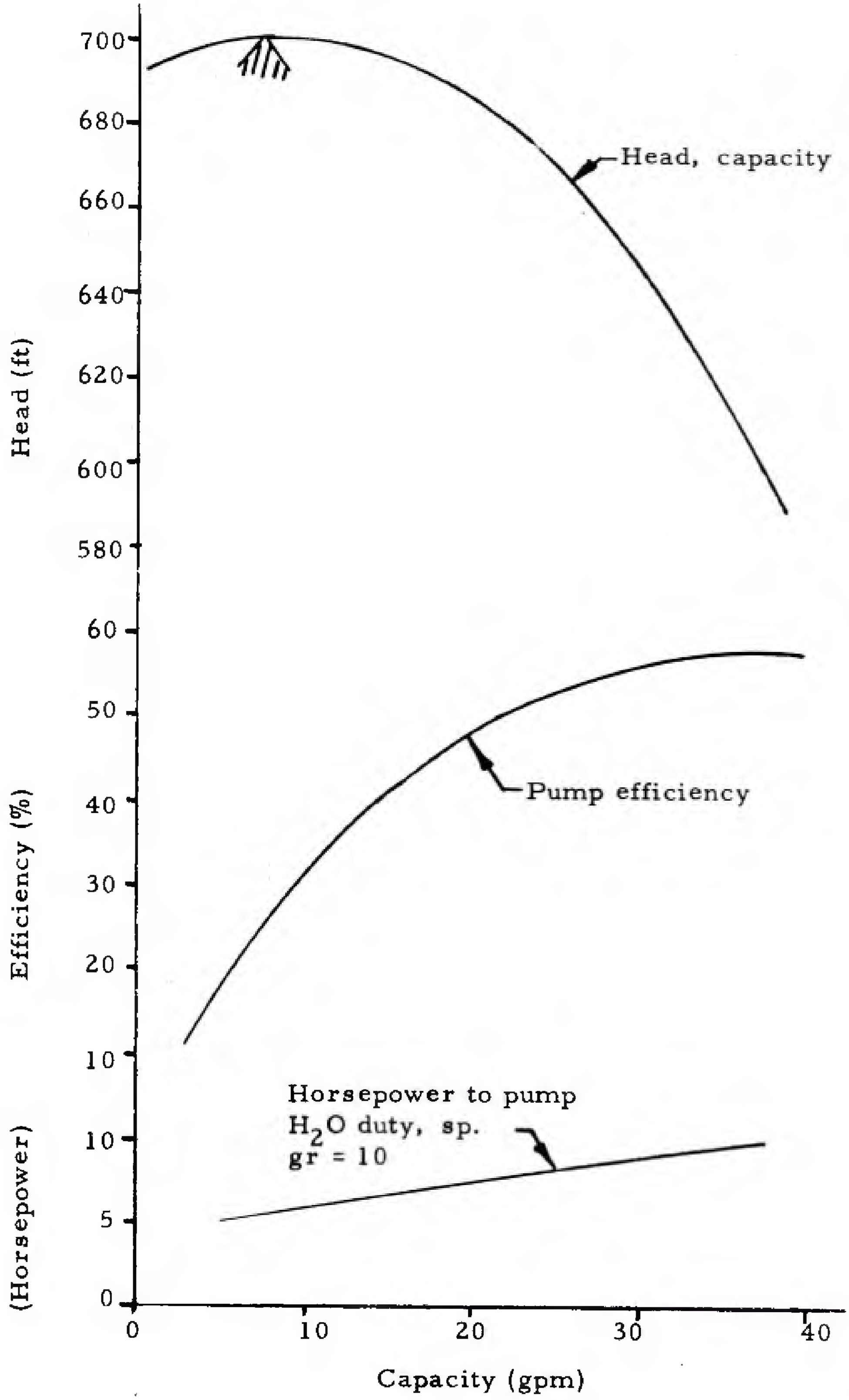


Fig. 11.8--Performance characteristics for J. C. Carter single-stage, high-speed pump

The Carter pump weighs an estimated 29 lb compared to 350 lb for a SIHI. Since a redundant system would be used in either case, the savings in weight would be 642 lb.

Comparable weight savings can be expected in the other system auxiliaries. A cursory investigation led to the conclusion that all of the power electrical equipment for a 400 Hz system could be obtained with slightly more difficulty than comparable 60 Hz equipment.

The problem of providing dc and 60 Hz power with a 400 Hz system was also investigated. It was found that Lear Seigler makes a solid-state frequency changer which is capable of converting 400 Hz power to both dc and 60 Hz power in power ranges of interest to the MUS application. Since 60 Hz was selected by the MUS program, the 400 Hz system was not investigated beyond the point of establishing its current availability and performance.

If, at a later date, 400 Hz power were thought to be desirable for a particular application in the 100 to 500 kw(e) range, a system could be put together which would be very light and compact by mating a compact TOPS reactor, as described in Section 11.1, with a 400 Hz conversion system, such as described in this section, with little or no development.

REFERENCES

1. Sundstrand Engineering Proposal No. 2301A-P6-125 kw(e) Steam Turbine System. November 30, 1967.

APPENDIX A

NUCLEAR ANALYSIS

This section describes in further detail the results of the nuclear analysis for the TOPS/MUS and presents the major computational results..

A.1. SUMMARY

Final calculations were made for a reactor containing 200 fuel elements. The unit cell was composed of 50 vol-% H_2O and a standard size (1.475-in. -OD) TRIGA fuel element, composed of 8 U(23)*-$ZrH_{1.65}$, containing homogenized Er^{167} and including a 20-mil clad of Incoloy-800. Two power levels were investigated; 3.1 Mw and 0.83 Mw. It should be noted that the lower power was raised from 0.83 to 0.970 Mw in order to meet the requirement of 110 kw(e) gross. The conclusions reached, however, on the basis of the calculations described are still valid. The core loading requirements were based on the high power for a one-year lifetime. The calculated cold clean excess reactivities needed for operation are $9.92 at 3.1 Mw and $5.48 at 0.83 Mw. Six rods of 2.0-in. maximum diameter would be required to shut down the 3.1 Mw core, allowing for one stuck rod. The calculated prompt negative temperature coefficient in the range of operating power levels is approximately 1.5×10^{-4} $\delta k/\delta T(^{\circ}C)$. A step reactivity insertion of approximately $7.00 into the cold reactor would be necessary to produce a peak fuel temperature of 1200°C. Fuel tests conducted at Gulf General Atomic in which U-ZrH fuel has been repetively pulsed to about 1200°C have demonstrated that these temperatures do not constitute a safety hazard. An estimate of the rate of control rod poison burnup for the 2-in. -OD rod in the 3.1 Mw core gives a value of approximately 10 mils of boron burned off the radius of a control rod for each 30 days of exposure to the average core thermal flux. To first approximation, the control rod worth varies directly with the radius, so the decrease in rod worth is indicated to be somewhat on the order of 12% during the 1 yr at 3.1 Mw. In actuality, only a portion of the rods would extend into the core average thermal flux, the remainder being in the lower, top reflector flux, so that the expected change would be less.

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*The notation, 8 U(23), will be used throughout this report to represent 8 wt-% Uranium 23% enriched.

A.2 ANALYSIS OF 3.1 Mw AND 0.83 Mw SYSTEMS WITH FUEL AND BURNABLE POISON LOADINGS BASED ON A ONE-YEAR LIFETIME AT 3.1 Mw

A.2.1 Cell and Reactor Description

This analysis was conducted for a reactor whose radius was 37.4 cm and contained 200 fuel elements. A water reflector 10-cm thick surrounded the core in the one-dimensional GAZE diffusion calculations. The reflector water was assumed to be at the average core water temperature and density with no voids. The group structure used for all calculations (GAZE and 1-DF cells) is given in Table A-1.

A unit cell of 50 vol-% water and 8.0 U(23)-$ZrH_{1.65}$ fuel-moderator material containing homogenized Er^{167} and clad with 0.020-in. Incoloy was chosen for this study. A more complete description of the cell is given in Table A-2 and plots of the cell power distribution are shown in Fig. A.1. The uranium loading was determined by estimating the necessary excess reactivity for the system, finding the loading to produce the excess, and then adding 1.43 kg of U^{235}--the fuel burnup to produce 3.1 Mw for 1 yr. The Er^{167} loading was that homogeneous concentration needed to offset the increased excess reactivity due to the 1.43 kg of fuel added for burnup.

A2.2 Calculational Results

Table A-3 gives a summary of the GAZE diffusion problems run and results obtained for a core using the 8.0 U(23)-$ZrH_{1.65}$ cell described in Table A-2. The radial power distribution and thermal neutron spectra have been shown in Section 3, in Figs. 3.1 and 3.4, respectively.

Complete cross-section sets were generated for each of the different fuel and water temperature conditions in the core and reflector (Table A-3, cases 1 through 5). When only the core density was changed to reflect the voids, the unvoided cross-section set was used with the water density changed to reflect the voids. Also, for the cases where fuel and erbium were removed to reflect burnup, no reaveraging of cross sections was done.

Table A-1

TOPS/MUS ENERGY GROUP STRUCTURE

Group	Energy
1	15 MeV to 0.608 MeV
2	0.608 MeV to 9.12 keV
3	9.12 keV to 1.125 eV
4	1.125 eV to 0.75 eV
5	0.75 eV to 0.57 eV
6	0.57 eV to 0.42 eV
7	0.42 eV to 0.26 eV
8	0.26 eV to 0.14 eV
9	0.14 eV to 0.06 eV
10	0.06 eV to 0.002 eV

Table A-2

TOPS/MUS BASIC CELL DATA FOR CORE

8 U(23)-$ZrH_{1.65}$ fuel element; 50 vol-% H_2O in core; homogenized (over core) $Er^{167} = 6 \times 10^{18}$ nuclei/cc; fuel height 15-in. (38.1 cm).

Cell region	Radius (in.)	Radius (cm)	Area (cm^2)	Volume (cm^3)	Volume fraction
8 U(23)-$ZrH_{1.65}$	0.7175	1.822	10.429	397.34	0.4731
Incoloy-clad	0.7375	1.873	0.592	22.56	0.0269
Water	1.042	2.648	11.021	419.90	0.5000
Total cell			22.043	839.8	1.000

Fuel and Burnable Poison Loading

U^{235} 42.2 g/element = 0.106 g/cm^3

Er^{167}[a] 1.40 g/element = 0.00351 g/cm^3

	Cell atom densities (nuclei/cc)	Homogenized atom densities (nuclei/cc)
Fuel		
H(ZrH)	0.056776×10^{24}	0.02686×10^{24}
Zr	0.03568×10^{24}	0.01688×10^{24}
U^{235}	0.000272×10^{24}	0.000129×10^{24}
U^{238}	0.000898×10^{24}	0.000425×10^{24}
Cladding		
Incoloy[b]	0.0937×10^{24}	0.00252×10^{24}
Water		
$H(H_2O)$	0.0668ρ[c] $\times 10^{24}$	$0.0334\rho \times 10^{24}$
Oxygen	$0.0334\rho \times 10^{24}$	$0.0167\rho \times 10^{24}$
Poison		
Er^{167}	12.7×10^{18}	6.0×10^{18}

[a]Erbium will be used in natural isotopes form, which is 22.9% Er^{167}.

[b]Stainless steel cross-sections used with atomic density to produce correct 2200 meter/sec absorption cross-section (0.299 cm^{-1}).

[c]ρ = water density.

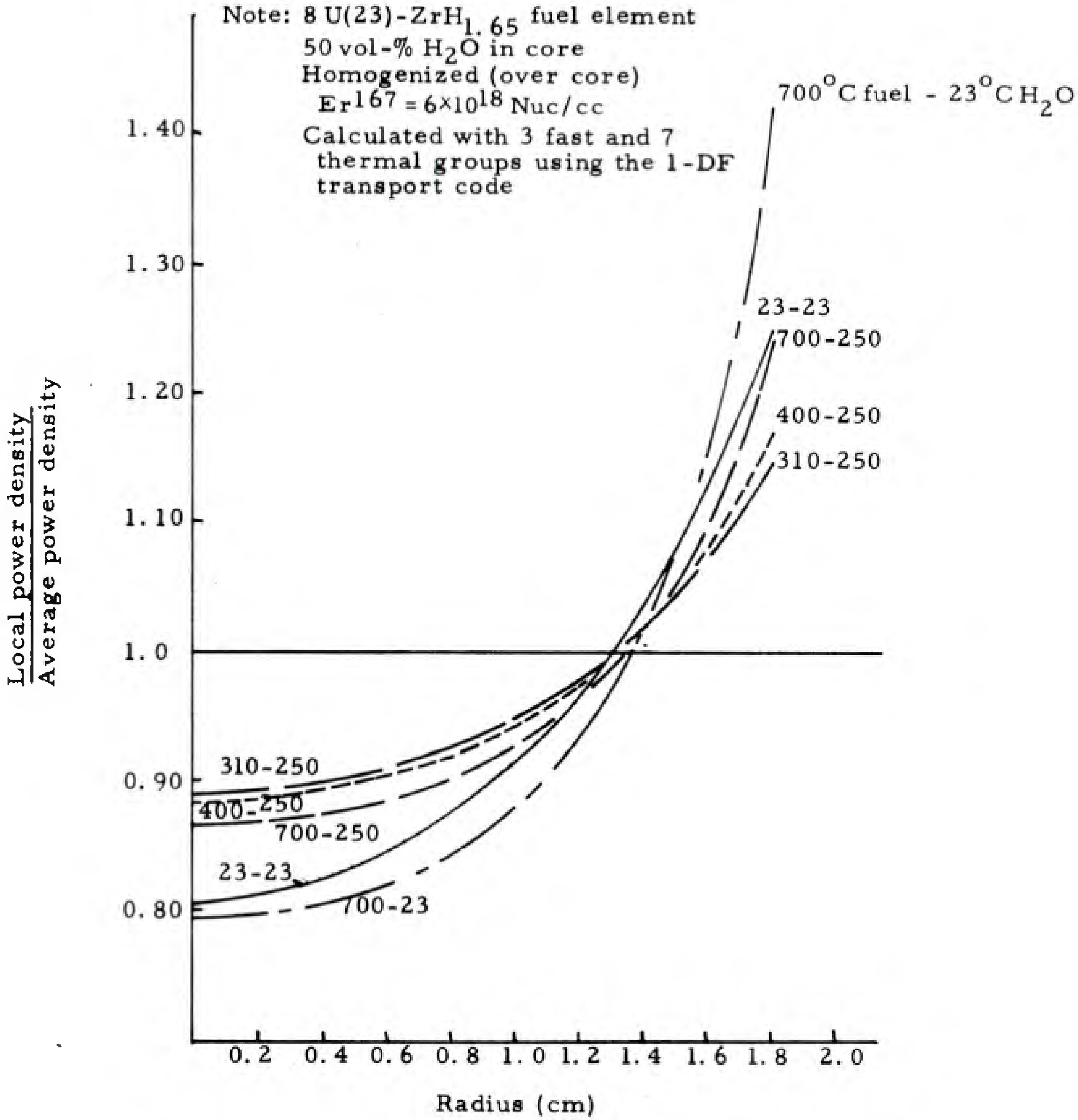


Fig. A.1--Cell power distribution versus fuel and water temperature

Table A-3

TOPS/MUS GAZE (ONE-DIMENSIONAL, DIFFUSION THEORY) RESULTS

8 U(23)-$ZrH_{1.65}$ fuel element; 50 vol-% H_2O in core; homogeneous (over core) $Er^{167} = 6 \times 10^{18}$ nuclei/cc; 200 element core.

Calculated Effective Multiplication (k_{eff})

Case No.	Core			Reflector		k_{eff}
	Fuel temperature (°C)	Water temperature (°C)	Water density	Water temperature (°C)	Water density	
1	23	23	1.00	23	1.00	1.05326
2	250	250	0.80	250	0.80	1.04265
3	310	250	0.80	250	0.80	1.03459
4	400	250	0.80	250	0.80	1.02108
5	400	280	0.75	280	0.75	1.01942
6	700	23	1.00	23	1.00	0.94738
7	700	250	0.80	250	0.80	0.96891
8[a]*	310	250	0.72	250	0.80	1.02836
9[b]*	400	250	0.72	250	0.80	1.01511
10[c]*	400	250	0.80	250	0.80	1.04490

[a] 10% density change (from Case 3) for steam voids in core.
[b] 10% density change (from Case 4) for steam voids in core.
[c] No Er^{167} and 3.1-Mw-yr burnup (1.43 kg U^{235})

* Cross-sections from Cases 3 or 4 used without reaveraging.

Table A-3 (Continued)

Reactivity Change

	Initial Condition				Final Condition					
Case No.	Description	Fuel temperature (°C)	Core H_2O temperature (°C)	Reflector H_2O temperature (°C)	Case No.	Description	Fuel temperature (°C)	Core H_2O temperature (°C)	Reflector H_2O temperature (°C)	δk (%)
1	Isothermal cold	23	23	23	2	Hot	250	250	250	-1.6
1	Cold	23	23	23	3	Hot (830 kw)	310	250	250	-1.87
1	Cold	23	23	23	4	Hot (3.1 Mw)	400	250	250	-3.22[a]
3	Hot (830 kw) ($\rho = 0.8$)	310	250	250	8	Hot (830 kw) ($\rho = 0.72$)	310	250	250	-0.62[b]
4	Hot (3.1 Mw) ($\rho = 0.8$)	400	250	250	9	Hot (3.1 Mw) ($\rho = 0.72$)	400	250	250	-0.6[b]

Reactivity Coefficient

Initial Condition			Final Condition			
Case No.	Temperature (°C)	Item	Case No.	Temperature (°C)	Item	δk/δT
2[c]	250	Fuel	3	310	Fuel	-13.4×10^{-5}
3[c]	310	Fuel	4	400	Fuel	-15.0×10^{-5}
4[c]	400	Fuel	7	700	Fuel	-17.4×10^{-5}
2[c]	250	Fuel	7	700	Fuel	-16.4×10^{-5}
1[d]	23	Fuel	6	700	Fuel	-15.6×10^{-5}
4[e]	250	Water	5	280	Water	-5.5×10^{-5}

[a]Actual 3.1-Mw temperature = ~420°C; δk to 420 = -3.22 - 20 × 15.8 × 10^{-5} = -3.54%.
[b]δk/δρ (at operating temperature of 250°C) = 0.0061/0.08 = +0.076.
[c]Initial and final water temperature is 250°C.
[d]Initial and final water temperature is 23°C.
[e]Initial and final fuel temperature is 400°C (includes change in water density).

A. 2. 3. Temperature and Void Coefficients

Figure 3. 5 showed the shape of the prompt negative fuel temperature coefficient and the core water temperature coefficient in the reactor operating range, as derived from the results shown in Table A-3. The fuel coefficient comes directly from the temperature ranges covered in the study, but the water coefficient is derived from problems 4, 5, and 9. This series of problems indicates that the water coefficient gets all of its negative component from the change in water density associated with a temperature change. The effect of the microscopic cross-section change with temperature, reflecting changes in the moderating characteristics of the water, is positive in its influence on reactivity--about 55% of the density effect in absolute magnitude. With this relationship, the core water temperature coefficient was derived by relating changes in core water temperature with its corresponding density change.

From problems 3 and 8, or 4 and 9, a core water density coefficient of the following magnitude is indicated:

$$\frac{\delta k}{\delta \rho} = +0.076 \quad .$$

A. 2. 4. Xenon Reactivity Loss

The time-dependent, equilibrium and override xenon reactivity worth values were calculated for both power levels based on the operating temperature conditions estimated at the beginning of this study. (Cross-sections come from problems 3 and 4 of Table A-3.) The values are shown in Fig. A.-2, and should not change to any major degree if recomputed to reflect any minor changes in core fuel and water temperatures.

A. 2. 5. Reactivity Requirements

On the basis of the calculations summarized in Table A-3, and the xenon results shown in Fig. A. 2, the reactivity requirements for a reactor with either a 3. 1 Mw or a 0. 83 Mw power level are given in Table A-4. The contingency values for the 3. 1 Mw system represent best estimates of reasonably attainable goals in balancing the reactivity effect of fuel and burnable poison depletion, and providing some margin for operational reactivity requirements. The lower contingency values for the 0. 83 Mw system were used to reflect the fact that balancing the reactivity changes should be less difficult with the system that has less fuel and burnable poison loss.

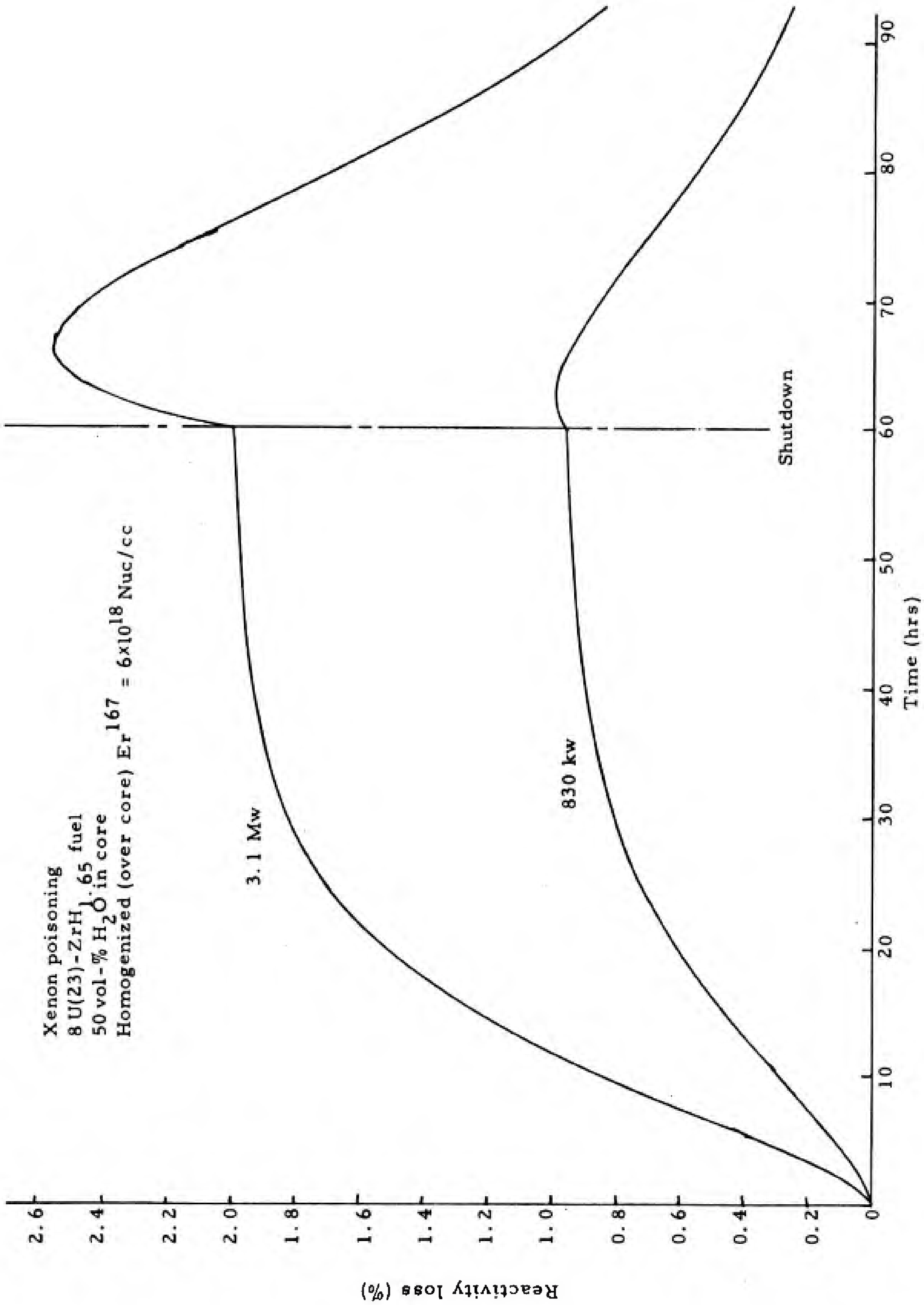


Fig. A.2--Xenon reactivity requirements

Table A-4

TOPS/MUS REACTIVITY REQUIREMENTS

8 U(23)-$ZrH_{1.65}$
50 vol-% H_2O core
Homogenized (over core) $Er^{167} = 6.0 \times 10^{18}$ Nuclei/cc

	830 kw(th) core (100 kw(e) net)		3.1 Mw(th) core (500 kw(e) net)	
	δk (%)	δk ($)	δk (%)	δk ($)
Cold-hot swing	-1.87	-2.67	-3.54	-5.06
Equilibrium xenon	-0.97	-1.39	-2.00	-2.86
Contingencies:				
Burnup--burnable poison imbalance	-0.50	-0.71	-0.70	-1.00
General	-0.50	-0.71	-0.70	-1.00
Total	-3.84	-5.48	-6.94	-9.92
Assumed Operating Conditions:				
Average fuel temperature	310°C (590°F)		420°C (788°F)	
Average core and reflector water temperature	250°C (482°F)		250°C (482°F)	

The loading in the 8 U(23)-$ZrH_{1.65}$ elements appears to be adequate to compensate for both unburned Er-167 and fission product poisoning after 1 yr at 0.83 Mw. Some small reduction may even be possible. However, more excess reactivity is needed for the 3.1 Mw power level than is provided by the loading in 200 8 U(23)-$ZrH_{1.65}$ elements. To give the 7% reactivity requirement indicated by Table A-4 and allow for fission product poisoning, a fuel element with approximately 25% enrichment--8 U(25)-$ZrH_{1.65}$ will be needed for the 3.1 Mw power level.

As mentioned earlier, the final value for the power level for the TOPS/MUS application was 0.97 Mw. Also, the average fuel temperature was revised downward slightly as a result of more detailed thermal analysis. The effect on reactivity requirements due to the decreased temperature override is of the same order of size of the increase in equilibrium xenon reactivity loss associated with the increase in neutron flux. Consequently, the error due to the assumption that these two effects compensate is believed to be sufficiently small that the results obtained for the 0.83 Mw calculations are valid for the reference design at 0.97 Mw.

A.2.6. Pulse and Ramp Reactivity Insertions

The fuel temperature coefficient shown in Fig. 3.5 was used in several time-dependent kinetics calculations which showed that it would take approximately a \$7.00 step reactivity insertion in a cold, clean reactor to produce a peak fuel temperature of 1200°C. Table A-5 gives a summary of the results of this set of kinetics calculations and the power and temperature versus time are plotted in Fig. A.3 for a \$5.48 pulse (the excess reactivity requirement for 0.83 Mw operation). The calculated results are shown in Fig. A.4 for the case of all rods being run out at the design removal rate--40 minutes for removal of all rods--in the 0.83 Mw core. Calculations, such as for the runout of rods just mentioned, where heat transfer characteristics and the details of the water temperature coefficients are important, can benefit significantly from the use of more refined heat transfer models and more information on the water temperature coefficient, both of which were considered outside the scope of this study.

A.2.7. Control Requirements

By using the reactivity requirements as estimated in Table A-4 for both power level cores, placing a tentative upper limit of 6 on the number of control rods, and demanding that the core be shut down even if one of the rods is stuck in the out position, the control requirements as shown in Table A-6 were derived.

Table A-5

TOPS/MUS SUMMARY OF TIME-DEPENDENT BLOOST KINETICS CALCULATIONS

8 U(23)-$ZrH_{1.65}$ Fuel Element
50 vol-% H_2O in core
Homogenized (over core) $Er^{167} = 6 \times 10^{18}$ nuclei/cc
200-element core

Reactivity insertion ($)[a]	Time for insertion (sec)	Minimum period (sec)	Peak power (Mw)	Energy Release		Peak Fuel Temperature	
				Prompt burst (Mw-sec)	At 2 sec (Mw-sec)	Prompt burst (°C)	At 2 sec (°C)
1.12	0.01	0.01	18.5	4.0	9.7	88	158
5.48	0.01	1.43×10^{-3}	21×10^{3}	113	123	1025	1005
7.00	0.01	1.07×10^{-3}	36×10^{3}	149	159	1223	1195
9.92	0.01	0.72×10^{-3}	78×10^{3}	220	230	1561	1519

[a] $\beta_{eff} = 0.007$ (estimated).

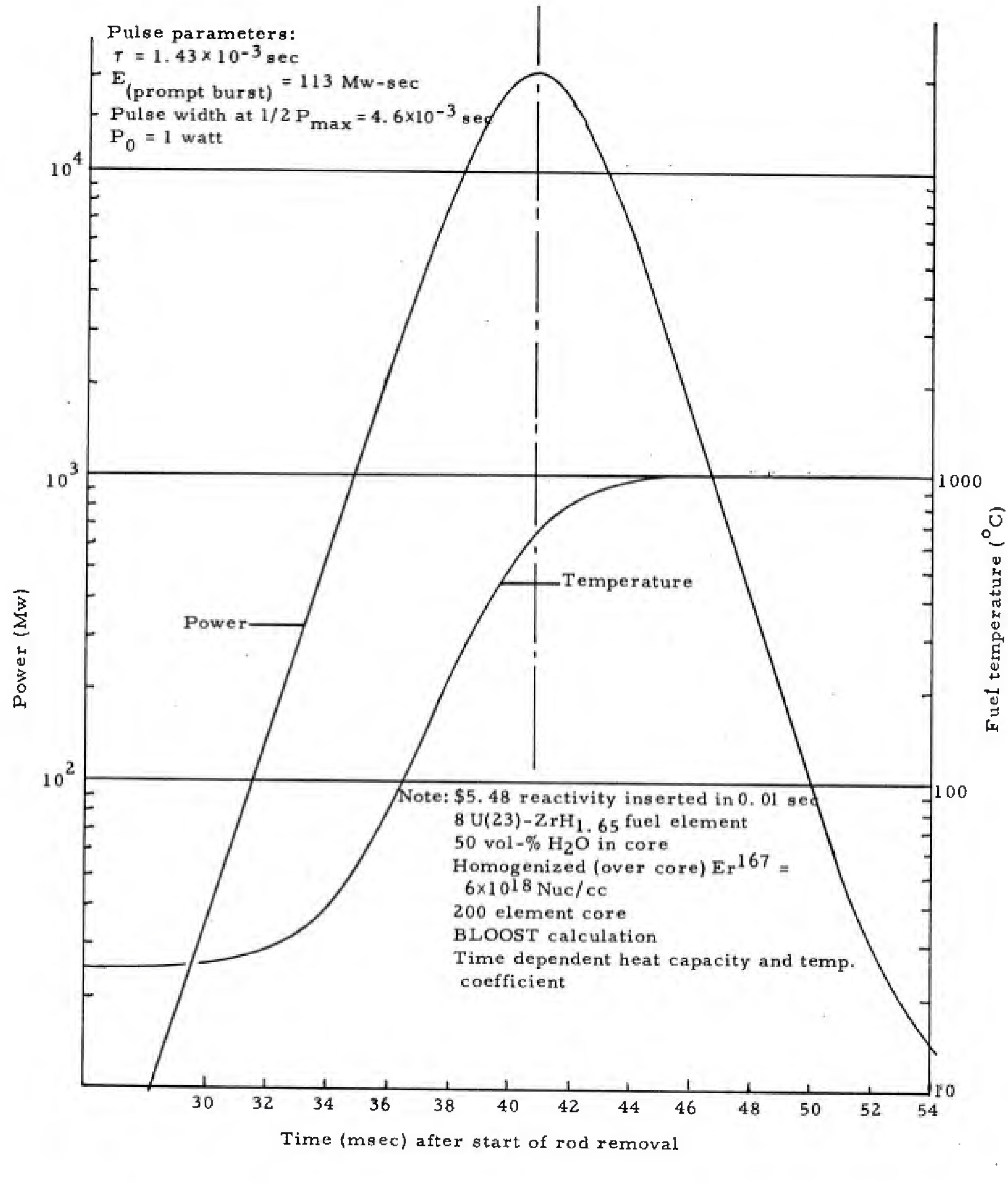


Fig. A.3--Power and temperature versus time

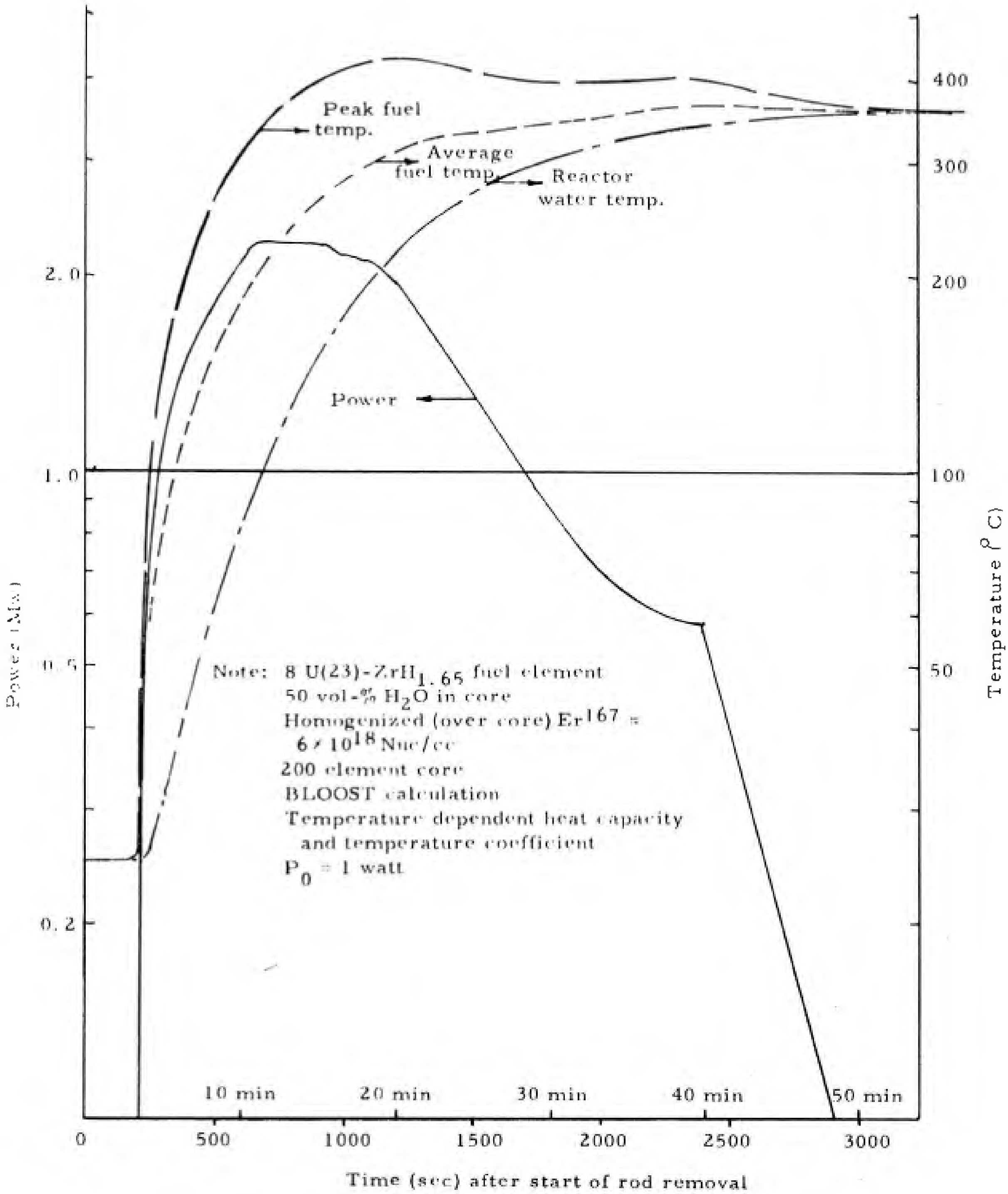


Fig. A.4--Power and temperature versus time after rods run out

Table A-6

TOPS/MUS CONTROL REQUIREMENTS

8 U(23)-$ZrH_{1.65}$ fuel element; 50 vol-% H_2O in core; homogenized (over core) $Er^{167} = 6 \times 10^{18}$ nuclei/cc

	830 kw-core ($)	3.1 Mw-core ($)
Core excess reactivity	5.48	9.92
Worth of 5 rods to shutdown[a]	5.60	10.00
Per rod	1.12	2.00

[a]For satisfaction of stuck rod requirement with a total of 6 control rods for the core.

If the diffusion length, L, of the unrodded, homogenized core is small compared with the effective control rod radius; i. e., if $r_{eff}/L >> 1$, then the cell method developed by Hurwitz and Roe with area absorption theory can be applied in calculating the approximate worth of a rod in a thermal reactor. For a cylindrical rod the area of absorption A_1 is given by

$$A_1 = 2\pi r_{eff} \lambda \frac{K_1 (r_{eff}/\lambda)}{K_0 (r_{eff}/\lambda)} ,$$

where r_{eff} represents the extrapolated radius inside the rod at which the flux vanishes, λ is the transport mean free path of the surrounding media and the Bessel functions K_1 and K_0 vanish at infinity. The reactivity worth is:

$$\delta k = A_1/A_0 ,$$

where A_0 = area of core/number of rods.

This method, which gives a rod worth at the position of average importance in the core, was used to compute the rod worth values shown in Table A-7, for an earlier study that used a higher loading of U^{235} and Erbium, 8 U(30)-$ZrH_{1.65}$, 50 vol-% H_2O core. These calculations are considered to be valid for estimating the control rod worth for the reference design core composition. Rod worth values for the 8 U(23)-$ZrH_{1.65}$, 50 vol-% H_2O core would be somewhat larger (~15%). Note that the rod worth values vary directly with rod diameter in our range of interest. As a check, this same method was used to calculate the worth of a 1-1/4-in.-diameter rod in an 80 element, 1-Mw TRIGA core. The result was a value of $1.50. This compares with an actual value of about $1.80 for a D-ring rod of the same radius containing 25 wt-% boron in graphite. The value would be ~$2.00 if it contained full-density B_4C. The D-ring in a TRIGA is close to the position of average flux in the core and thus for the rod worth estimates in this study, the "corrected worth" values, shown in Table A-7 are the Hurwitz-Roe method values multiplied by $2.00/1.50--the ratio of the actual maximum worth D-ring 1-Mw TRIGA rod to the worth calculated by the Hurwitz-Roe method. The largest rod that can be handled without sizeable difficulties in the Power TRIGA (TOPS/MUS) core is ~2-in.-diameter.

From the results shown in Table A-7, the rod worth requirements for the 0.83 Mw core with 8 U(23)-$ZrH_{1.65}$ fuel are satisfied with a rod diameter of ~1.25 in., while a 2-in.-diameter rod should be the maximum size necessary in the 3.1-Mw core under the assumptions of the study.

Table A-7

CALCULATED CONTROL ROD WORTH VERSUS DIAMETER

8 U(30)-$ZrH_{1.65}$; 50 vol-% H_2O core[a];
homogenized Er^{167} = 15.0 × 10^{-6} (×10^{24}) nuclei/cc

Rod diameter (in.)	Calculated[b] δk ($)	Corrected Worth[c] δk ($)
1.25	1.16	1.54
1.75	1.70	2.26
2.00	1.96	2.61
2.25	2.21	2.94
2.50	2.47	3.28

[a]Rod worth values for the 8 U(23)-$ZrH_{1.65}$, 50 vol-% H_2O core would be somewhat larger (~15%).

[b]Hurwitz and Roe--area absorption theory.

[c]Based on a calculation, using the same method, of the worth of a 1.25-in. rod in an 80-element TRIGA Mark F or Mark III core. Calculated value was $1.50 and experimental worth of a 25 wt-% boron-in-graphite rod is about $1.80, or $2.00 if full density (2.54 gm/cc) B_4C is used. Correction factor is 2.00/1.50 = 1.33.

A. 2. 8. Control Rod Burnup

An estimate was made of the rate of boron burnup in a control rod assuming it was exposed to the core average thermal flux in the core containing 8 U(23)-$ZrH_{1.65}$ fuel. The time for the control rod poison concentration to decrease by 1.5% was calculated using $N = N_0 e^{-\sigma_a \Phi t}$, then this was interpreted to mean the rod volume had decreased by 1.5% by a decrease in the radius. The result of this calculation was an estimate that the rod radius decreased 10 mils each 30 days. This would give a decrease in rod worth of ~12% after 1 yr in the 3.1-Mw core, assuming that rod worth is proportional to its diameter. On the basis that only some portion of the rods would extend into the core average thermal flux, the remainder being in the lower flux of the top reflector, the rod worth loss estimate should be more of an upper limit.

APPENDIX B

THERMAL-HYDRAULIC ANALYSIS STEAM GENERATOR DESIGN

When circulation of primary coolant is carried out by thermal convection, the rate of circulation is a dependent variable that is a function of primary loop geometry and the temperature "swing" in the primary coolant. For any given primary coolant temperature entering the core and secondary coolant, or working fluid, temperature there will be a family of steam generators that can effect the balance between heat generated in the core, heat transported to the steam generator, and heat delivered to the working fluid. The optimum heat exchanger will satisfy the heat balance with minimum surface area (which is assumed analogous to minimum cost) while keeping fuel temperature and primary system pressure within acceptable limits.

Steam generator tubes were arbitrarily fixed at 0.5-in.-OD with 20 BWG wall thickness on the basis that this is a standard tube size that is easy to work with. Mean steam generator bundle height above the core was taken as 10 ft, since this space was made available by the pressure vessel layout which provided sufficient water depth over the core for shielding during refueling operations. No attempt was made to optimize either tube diameter or bundle height.

The first configuration "fire tube" steam generator was optimized by assuming a total heat transfer area and varying the length and number of tubes to obtain maximum Reynolds' number inside the tubes. The use of Reynolds' number as a factor of merit for optimization is based on the fact that primary side heat transfer is by forced convection* which is Reynolds' number dependent while the secondary side heat transfer is principally by boiling and dependent on the surface-to-bulk fluid temperature difference. Therefore, that configuration giving the highest primary side

*Even though the primary coolant circulation is caused by thermal convection, the fluid flow condition in the steam generator tubes is not one of natural convection since the flow is undirectional and caused by external forces as distinguished from the local multidirectional eddies which characterize natural convection.

Reynolds' number should give close to the maximum heat flux. Determination of flow rate and Reynolds' number for each tube length is an iterative process since friction factor is Re dependent.

Having chosen a tube length, the heat fluxes from primary coolant to inner tube wall, inner wall to outer wall, and outer wall to secondary coolant, were balanced by a trial and error solution with wall temperature as the dependent variable. If the balanced heat flux multiplied by assumed area failed to give the desired heat flow rate, the area was adjusted accordingly and the process repeated until a check was obtained.

Primary side heat transfer coefficients were calculated by the Dittus-Boelter equation

$$Nu = 0.027\ Re^{0.8}\ Pr^{0.4}\ .$$

Secondary side boiling heat transfer has been calculated by Rohsenow's equation(1)*

$$\frac{c_\ell (T_w - T_{sat})}{h_{fg}} = C_{sf}\left[\frac{(q/A)_b}{\mu_\ell h_{fg}} \sqrt{\frac{g_0 \sigma}{g(\rho_\ell - \rho_v)}}\right]^{1/3} Pr_\ell^{1.7}\ ,$$

which is rearranged as follows, to solve for the heat transfer coefficient

$$h_b = \frac{\mu_\ell}{(h_{fg})^2} \sqrt{\frac{\rho_\ell - \rho_v}{\sigma}} \left[\frac{c_\ell}{Pr^{1.7}}\right]^3 (T_w - T_{sat})^2 \left(\frac{1}{C_{sf}}\right)^3\ ,$$

where

μ_ℓ = viscosity of liquid

h_b = boiling heat transfer coefficient

h_{fg} = latent heat of vaporization

ρ_ℓ = density of liquid

ρ_v = density of vapor

σ = surface tension

c_ℓ = heat capacity in liquid

T_w = boiling side wall temperature

T_{sat} = saturation temperature of liquid, and

---

*References are listed at the end of this section.

C_{sf} = dimensionless coefficient depending on surface and liquid combination (= 0.013 for water-stainless steel consistent units).

Secondary side heat transfer in the economizer region was calculated by the Martinelli-Boelter equation which superimposes forced and free convection effects[1]

$$N_u = 1.75 F_1 \left[\frac{\pi}{4} \text{Re} \, \text{Pr} \, \frac{D}{L} \pm 0.0722 \left(\frac{D}{L} \text{Gr} \, \text{Pr} \right)^{0.84} F_2 \right]^{1/3}$$

where

Re = Reynolds number

Pr = Prandtls number

Gr = Grashofs number

D = tube diameter

L = tube length, and

F_1 and F_2 = temperature dependent coefficients .

Flow in the economizer section is nominally laminar.

Flow in the secondary side superheater region is turbulent, and the steam side coefficient can be calculated by the Dittus-Boelter equation

The optimized heat exchanger has 640 tubes 6 ft long for a total heat transfer area of 420 ft^2 with 89 ft^2 in the superheater, 138 ft^2 in the boiler, and 172 ft^2 in the economizer sections. The pinch point diagram for the steam generator is shown in Fig. B-1. This diagram is applicable to both steam generators studied.

The second configuration helical bundle was not optimized, but was designed to give the same heat transfer rate as the first configuration unit with the same primary and secondary fluid temperatures and pressure drop on the primary side. The mass velocity on the secondary side (inside the tubes) was arbitrarily taken as 100 lb/ft^2-sec, which is in the design range of conventional steam-tube boilers. Rectangular tube array was chosen for ease in construction. The friction factor, h, for a rectangular array of tubes in cross flow is[2]

$$f = \left(0.044 + \frac{0.08 \, x_\ell}{(x_T - 1)^n} \right) (\text{Re})^{-0.15} ,$$

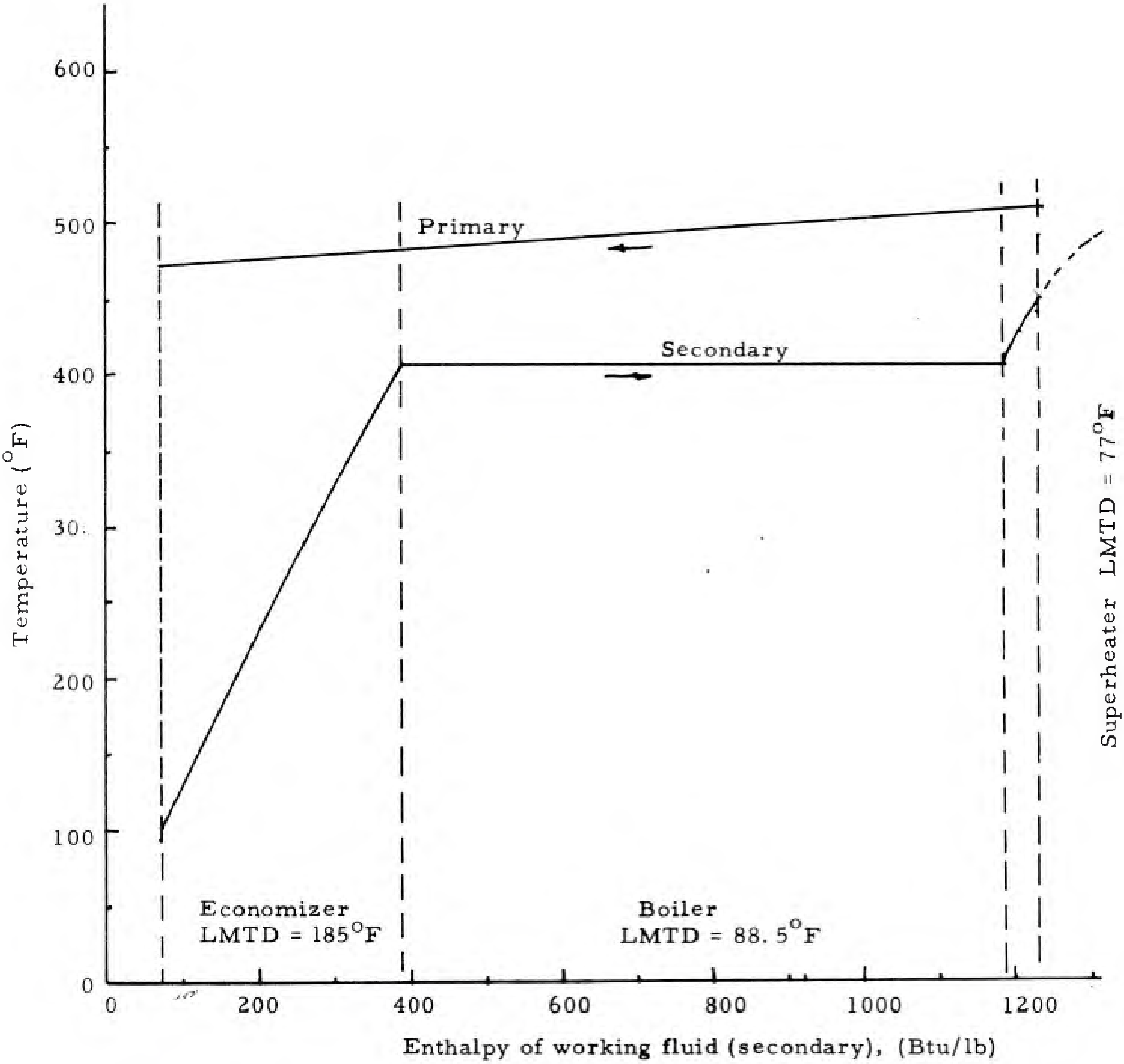


Fig. B.1--Pinch point diagram for steam generator (3.1 Mw(t)) where x_ℓ = pitch/diameter in longitudinal direction

x_T = pitch/diameter in transverse flow

$n = (0.43 = 1.13/x_\ell)$, and

Re is evaluated at the point of minimum clearance between tubes.

The optimized first configuration steam generator had a pressure drop of 17.8 lb/ft^2, therefore[2]

$$\Delta P = 17.8 \text{ lb/ft}^2 = 4 f N_B \frac{G^2_{max}}{2 g \rho_\ell} ,$$

where N_B is the number of tube banks

The total heat transfer area = $A = A_t \times \ell_t + N_t$, where A_t = area per ft of tube, ft; ℓ_t = length per tube, ft. The minimum flow area is

$$A_f = \frac{N_t}{N_b} (P - D) \ell_t \cos \theta ,$$

where θ = helix angle measured from the transverse plane.

By rearrangement,

$$\ell_t N_t = A/A_t ,$$

and

$$A_f = \frac{A (x_t - 1) D \cos \theta}{N_B A_t} .$$

If tubes are specified at 0.5-in.-OD, D = 0.0417 ft, A_t = 0.131 ft^2/ft, N_t = 25 tubes, G = 100 lb/ft^2-sec, and if the helix angle is small, so that $\cos \theta \simeq 1$, then

$$A_f = \frac{A (x_t - 1)\, 0.0417}{0.131 N_B} .$$

A series of solutions, shown in Fig. B-2, show that the flow area at which design ΔP occurs is insensitive to transverse pitch. Since the heat transfer coefficient for cross flow is

$$h = 0.26 \frac{k}{D} Re^{0.6} Pr^{0.33} ,$$

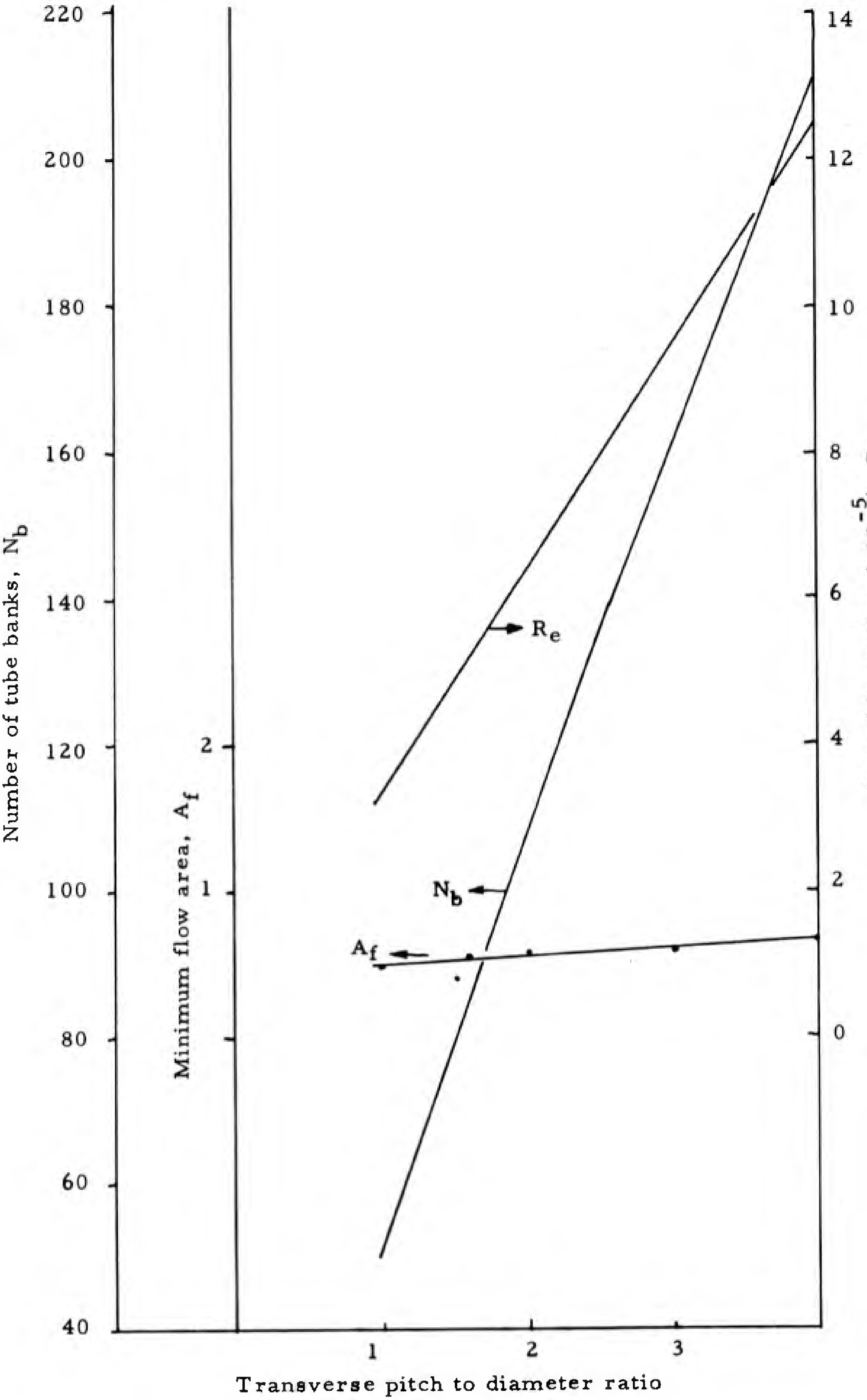


Fig. B.2--Effect of transient pitch/diameter ratio on bundle parameters where k = thermal conductivity of the liquid of the film temperature, it can be seen that h will increase as pitch decreases. Therefore, pick the smallest practical transverse pitch from a fabrication standpoint, or $x_t = 1.5$.

On the secondary side, inside the tubes, Reynolds' number is well up in the turbulent range because of the higher mass velocity in the helical bundle. Therefore, coefficients in the economizer and superheater regions can be calculated by the Dittus-Boelter equations. The boiling coefficient was calculated by Rohsenow's equation which was used in the boiling region of the "fire-tube" bundle. The resulting bundle had a total area of 223 ft^2 with 15 ft^2 in the superheater, 169 ft^2 in the boiler, and 39 ft^2 in the economizer. This hand calculation was checked later by a computer program, for steam generators which produces heat balances across incremental lengths of the bundle using local properties and number of phases. This check calculation gave 193 square ft total required bundle area. An arbitrary 20% allowance was added to this number giving an approximate effective heat transfer area of 230 ft^2.

REFERENCES

1. Rohsenow, W. M., and H. Y. Choi, Heat, Mass, and Momentum Transfer, (Prentice-Hall, Inc.) 1961.

2. McAdams, W. H., Heat Transmissions, 3rd Ed., (McGraw-Hill Book Company, Inc.), 1954.

APPENDIX C

STEADY STATE AND TRANSIENT CONTROL ANALYSES

The transient control analysis described here was conducted to evaluate the stability of the total power producing system and specifically to show the extent to which the reactor will inherently follow an electrical load change without requiring any control drive or other mechanical action. The results of this analyses are affirmative and show that the system is adequately load following for all normal variations in load. In this analysis a constant volume flow rate feedwater pump was assumed (see Fig. C-1).

Subsequent to the transient analysis, a modification was made to the feedwater system which consisted of changing the constant volume feed pump to a centrifugal pump. This of course does not affect the primary objective of the analysis which has to investigate the response of the primary loop and reactor and which "see" only changes in heat removal by the steam generator regardless of the cause. However, additional study is recommended for investigating the steam generator behavior, particularly at the maximum potential steam generation rates.

Steady-state operating points over range from zero to maximum power must be determined to find the quasi steady-state performance of the plant as a load follower and to provide input for the transient analysis which shows how the plant will respond to perturbations. For a given configuration of core, steam generator and primary system loop temperatures throughout the primary system will equilibrate at certain values depending on the temperature and rate of heat removal through the secondary system. Since the primary system is self-pressurized, the system pressure will also be a dependent variable.

Primary system temperatures are determined by simultaneous solution of heat transfer and transport equations for the successive steps in the flow of heat from its origin in the fuel elements to its delivery to the secondary coolant. A steady-state solution demands that the heat flows at each step are the same.

By definition:

Q_1 = heat flowing from interior of elements to element surface.

Q_2 = heat transferred from element surface to primary coolant,

Q_3 = heat transported by primary coolant from core to steam generator,

Q_{4s} = heat transferred from primary coolant to working fluid in superheater section.

Q_{4b} = heat transferred from primary coolant to working fluid in boiler section ,

Q_{4e} = heat transferred from primary coolant to working fluid in economizer section .

At steady-state $Q_1 = Q_2 = Q_3 = (Q_{4s} + Q_{4b} + Q_{4e})$.

For a pressurized TRIGA there will be a series of states of $\bar{T}_f$ and $\bar{T}_p$ where k = 1. Here $\bar{T}_f$ = mean fuel temperature and $\bar{T}_p$ = mean primary water temperature. Neglecting the effect of reflector water temperature, steady-state operation can be defined by

$$\dot{K}_f \partial \bar{T}_f + \dot{K}_w \partial \bar{T}_p = 0 \quad , \qquad (1)$$

where $\partial \bar{T}_f$ and $\partial \bar{T}_p$ are mean temperature deviations from a steady-state datum point (e. g., full load) and where $\dot{K}_f$ and $\dot{K}_w$ are the components of the temperature coefficient of reactivity for fuel and water temperatures.

Rewrite equation (1) as

$$\dot{K}_f (\bar{T}_f - \bar{T}_{fo}) + \dot{K}_w (\bar{T}_p - \bar{T}_{po}) = 0 \quad ,$$

in which the subscript o refers to the initial condition, and rearrange to obtain

$$\bar{T}_f = \bar{T}_{fo} = \frac{\dot{K}_w}{\dot{K}_f} (\bar{T}_p - \bar{T}_{po}) \quad . \qquad (1a)$$

For cylindrical fuel rods, neglecting the effect of a center hole

$$\bar{T}_f = \bar{T}_s + \frac{1}{8\pi} \frac{Q}{L} \frac{3600}{K} + \Delta T_{wg}$$

where

$\frac{Q}{L}$ = Btu/linear ft-sec ,

K = conductivity of fuel body, Btu-ft/ft^2 hr ^{o}F

ΔT_{wg} = ΔT across cladding and fuel-to-cladding gap ,

$\bar{T}_s$ = average fuel body surface temperature

For N fuel elements producing a total of Q_1,

$$\overline{T}_f = \overline{T}_s + \frac{1}{8\pi}\,\frac{Q_1}{NL}\,\frac{3600}{K} + \Delta T_{wg} \quad . \tag{2}$$

Combining equations (1a) and (2) gives

$$\overline{T}_s = \overline{T}_{fo} - \frac{1}{8\pi}\,\frac{Q_1}{NL}\,\frac{3600}{K} - \Delta\overline{T}_{wg} - \frac{\dot{K}_w}{K_f}\,(\overline{T}_p - \overline{T}_{po}) \tag{3}$$

Using the above reactivity coefficients and heat flow temperature conditions for a 200 element core at 3,000 Btu/sec and assuming ΔT_{wg} of 100°F at rated conditions and incorporating these in the second term, gives

$$\overline{T}_s = 788 - 0.0792\, Q_1 - \frac{\dot{K}_w}{\dot{K}_f}\,(\overline{T}_p - \overline{T}_{po}) \quad , \tag{3a}$$

or

$$Q_1 = 12.6 \left[788 - \overline{T}_s - \frac{\dot{K}_w}{\dot{K}_f}\,(\overline{T}_p - \overline{T}_{po}) \right] \quad . \tag{3b}$$

Heat transfer from fuel element surface to primary coolant given by

$$Q_2 = h_e\, A_c\, (\overline{T}_s - \overline{T}_p) \quad . \tag{4}$$

Where

h_e = heat transfer coefficient in core, Btu/ft^2 sec °F ,

A_c = heat transfer area in core, ft^2 .

Primary coolant circulation rate by thermal siphon given by

$$\dot{v} = \left[\frac{2g\,\dfrac{y\,\Delta\rho}{\overline{\rho}}}{\dfrac{1}{A_{fc}^2}\left[4f_c\left(\dfrac{L}{D}\right)_c + C_{\ell c}\right] + \dfrac{1}{A_{fe}^2}\left[4f_e\left(\dfrac{L}{D}\right)_e + C_{\ell e}\right]} \right]^{1/2} \quad . \tag{5}$$

where

$\dot{v}$ = primary coolant flow rate, ft^3/sec

y = center-to-center height of steam generator above core, ft

$\Delta\rho$ = density difference between hot and cold legs of primary loop, lb/ft^3

$\overline{\rho}$ = average primary coolant density, lb/ft^3 ,

f_c = friction coefficient in core

$(L/D)_c$ = effective length/diameter ratio in core

A_{fc} = flow cross sectional area in core, ft^2

$C_{\ell c}$ = parasitic loss coefficient in core

f_e = friction coefficient in steam generator

$(L/D)_e$ = effective length/diameter ratio in steam generator

A_{fe} = flow cross sectional area in steam generator, ft^2

$C_{\ell s}$ = parasitic loss coefficient in steam generator .

Taking the following values

$$\left.\begin{array}{l} y = 10 \text{ ft} \\ A_{fc} = 1.29 \text{ ft}^2 \\ (L/D)_c = 25 \\ C_{\ell c} = 2.0 \\ f_c = 0.006 \end{array}\right\} \begin{array}{l} \text{200 elements} \\ \text{0.8 in.-diameter} \\ \text{20 in. long} \end{array}$$

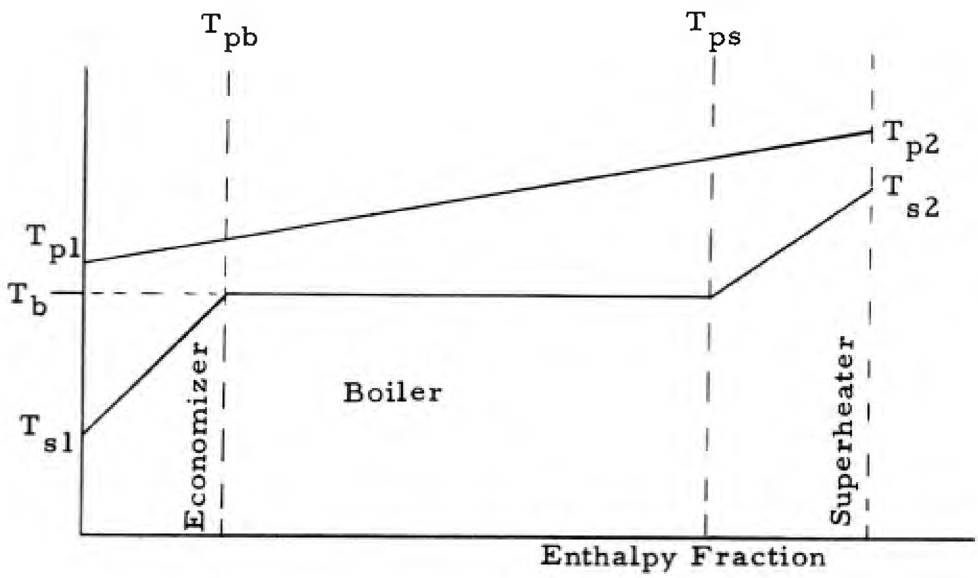
$$\left.\begin{array}{l} A_{fc} = 0.65 \text{ ft}^2 \\ (L/D)_e = 141.7 \\ C_{\ell s} = 1.5 \\ f_e = 0.0065 \end{array}\right\} \begin{array}{l} \text{400 ft}^2 \\ \text{660 tubes} \\ \text{60 in. long} \\ \text{0.424 in. ID} \end{array}$$

gives

$$\dot{v} = \left[\frac{2g\,\frac{\Delta\rho}{\overline{\rho}} \times 10}{13.82}\right]^{1/2}$$

$$\dot{v} = 6.82\left(\frac{\Delta\rho}{\overline{\rho}}\right)^{1/2} \quad . \tag{5a}$$

The sketch below shows the point of reference in the coolant circuits for temperatures in the following analysis of the coolant circulation



Pinch-point diagram showing definition of symbols

Heat transport by thermal siphon is

$$Q_3 = \bar{\rho}\,\dot{v}\,\overline{C}_p\,(T_{p2} - T_{p1}) \quad . \qquad (6)$$

Between 450° and 520°F $\rho \cong 75.9 - 0.054T$, where

$\rho = \text{lb/ft}^3$

$T = {}^{\circ}\text{F}$

then

$$\Delta\rho = 0.054\,(T_{p2} - T_{p1})$$

and

$$\bar{\rho} = \frac{\rho_1 + \rho_2}{2} = 75.9 - 0.054\,\frac{T_{p1} + T_{p2}}{2}$$

also,

$$\bar{C}_p = 1.16\ \text{Btu/lb-}^{\circ}\text{F}$$

between 450° and 510°F. Combining the above with equations (6) and (5a) gives

$$Q_3 = 1.84\,(T_{p2} - T_{p1}) \left[75.9\,(T_{p2} - T_{p1}) - 0.027\,(T_{p2}^2 - T_{p1}^2)\right]^{1/2} . \tag{6a}$$

Heat balance across the superheater, boiler, and economizer sections of the steam generator give

$$Q_4 = Q_{4s} + Q_{4b} + Q_{4e} .$$

In the superheater,

$$Q_{4s} = \dot{W}_p C_{pp} (T_{p2} - T_{ps}) \tag{7}$$

$$= \dot{W}_s C_{ps} (T_{s2} - T_{sb}) ,$$

where $\dot{W}_s$ is the secondary coolant mass flow rate and $\dot{W}_p$ is the primary coolant mass flow rate and hence,

$$T_{s2} = \frac{\dot{W}_p \bar{C}_{pp}}{\dot{W}_s \bar{C}_{ps}} (T_{p2} - T_{ps}) + T_b .$$

In the boiler,

$$Q_{4b} + \dot{W}_s L_v = \dot{W}_p \bar{C}_{pp} (T_{ps} - T_{pb}) ,$$

where L_v is the latent heat of vaporization, so that

$$T_{ps} = \frac{\dot{W}_s L_v}{\dot{W}_p \bar{C}_{pp}} + T_{pb} ;$$

and, in the economizer,

$$Q_{4e} = \dot{W}_s \bar{C}_{ps} (T_b - T_{s1}) = \dot{W} \bar{C}_{pp} (T_{ps} - T_{p1}) ,$$

where

$$T_{pb} = \frac{\dot{W}_s \, \overline{C}_{ps}}{\dot{W}_p \, \overline{C}_{pp}} (T_{sb} - T_{s1}) + T_{p1} \ .$$

Heat flux equations are the following:

In the superheater,

$$Q_{4s} = U_s A_s \Delta T_s \ ,$$

where U_s is the overall heat transfer coefficient in the superheater.

In the boiler,

$$Q_{4e} = U_b A_b \Delta T_b \ ,$$

where U_b is the overall heat transfer coefficient in the boiler.

In the economizer

$$Q_{4e} = U_e A_e \Delta T_r \ ,$$

where U_e is the overall heat transfer coefficient in the economizer, and

$$A_s + A_b + A_e = A = \text{constant} \ .$$

Finally, there results the following equations using numerical values of K_w and K_f.

$$\overline{T}_f = T_o - \frac{21.0 \times 10^{-4} \overline{T}_p + 0.162}{6.89 \times 10^{-4} \overline{T}_f + 0.462} (T_o - \overline{T}_p) \ ;$$

$$\overline{T}_s = \overline{T}_f - 0.0774 \, Q \ ;$$

$$Q_2 = h_c \times 98 \, (\overline{T}_s - \overline{T}_p) \ ;$$

$$Q_3 = 1.84 \, (T_{p2} - T_{p1}) \left[75.9 \, (T_{p2} - T_{p1}) - 0.027 \, (T_{p2}^2 - T_{p1}^2) \right]^{1/2} \ ;$$

$$Q_4 = Q_{4s} + Q_{4b} + Q_{4e} \quad ;$$

$$Q_{4s} = \dot{W}_p \bar{C}_{pp} (T_{p2} - T_{ps}) \quad ,$$

$$= \dot{W}_s \bar{C}_{ps} (T_{s2} - T_{sb}) \, ,$$

$$= U_s A_s (\Delta T_s) \quad ;$$

$$Q_{4b} = \dot{W}_p \bar{C}_{pp} (T_{ps} - T_{pb}) \quad ,$$

$$= \dot{W}_s L_v \quad ,$$

$$= U_b A_b (\Delta \bar{T}_b) \quad ;$$

$$Q_{4e} = \dot{W}_p \bar{C}_{pp} (T_{pb} - T_{pl}) \quad ,$$

$$= \dot{W}_s C_{pe} (T_{sb} - T_{sl}) \quad ,$$

$$= U_e A_e (\Delta \bar{T}_e) \quad ;$$

also,

$$A = A_s + A_b + A_e \quad ;$$

$$\dot{W}_p = 1.584 \left[75.9 \, (T_{p2} - T_{pl}) - 0.027 (T_{p2}^2 - T_{pl}^2) \right]^{1/2} \quad ;$$

and ,

$$Q_1 = Q_2 = Q_3 = Q_4 \quad .$$

Solution of these equations gives the following results:

	Nominal power output (kw(e))		Heat rate (Btu/sec)	
	120	500	820	2860
T_f ($^\circ$F)	647	856	647	856
T_{p2} ($^\circ$F)	470	498	470	498
T_{p1} ($^\circ$F)	454	462	454	462
$\dot{W}_p$ (lb/sec)	41.9	64.6	41.9	64.6

Off-design operation for the 120 kw(e) nominal plant was calculated from the following simplified equations. For an impulse turbine running at constant speed, but varying load, the pressure drop across the turbine is directly proportional to steam weight flow through the turbine. Pressure drop through the turbine throttle valve can be approximated by

$$\Delta\rho_v = \bar{K}_v \frac{A_o}{A} \frac{\dot{W}^2}{\bar{\rho}_v} ,$$

and pressure drop through the remaining circuit can be approximated by

$$\Delta\rho_c = \bar{K}_c \frac{\dot{W}^2}{\bar{\rho}_c} .$$

This gives a relation for boiler pressure

$$\rho_b = \rho_c + \bar{K}_t \dot{W}_t + \bar{K}_v \frac{A_o}{A} \frac{\dot{W}_t^2}{\bar{\rho}_v} + K_c \frac{\dot{W}^2}{\bar{\rho}_c} .$$

The constants $\bar{K}_t$, $\bar{K}_v$, and $\bar{K}_e$ are evaluated at rated conditions, and from these the boiler pressure in the steam generator at any off-design point can be calculated.

For the particular case where the boiler feed pump produces constant flow at all boiler pressures, as typical of a displacement pump, and where water leaving the steam separator re-enters the boiler at no loss in enthalpy, the quality of steam leaving the steam generator is given by two heat balances

$$h_{f2} = (1 - q)\, h_{f3} + q\, h_{f1} \quad ,$$

and

$$h_{f2} + \Delta h = q \times h_{g3} + (1 - q)\, h_{f3} \quad ,$$

giving

$$q = \frac{\Delta h}{h_{g3} - h_{f1}} \quad ,$$

where

h_{f1} = water enthalpy leaving the condenser ,

h_{f2} = water enthalpy entering the steam generator ,

h_{f3} = water enthalpy leaving the steam generator,

h_{g3} = vapor enthalpy leaving the steam generator ,

Δh = total enthalpy change in steam generator,

q = quality in steam generator effluent ,

Q = core heat rate, and

$\dot{W}$ = weight flow rate of working fluid .

For any given heat rate the average fuel temperature, average fuel surface temperature, and steam generator boiling temperature are obtained by simultaneous solution of the heat flux equations for flow of heat from fuel body to surface, surface to primary coolant, and primary coolant to secondary coolant, respectively. Then

$$\Delta h = \frac{Q}{\dot{W}} \quad ,$$

and h_{g3} is fixed by the boiling temperature. These quantities, in turn, fix the effluent steam quality. Assuming the steam separator effects 100%

separation of steam and water, the steam weight flow entering the turbine throttle will be

$$\dot{W}_t = \dot{W} \times q \ .$$

Since the turbine forepressure is directly proportional to weight flow at constant speed,

$$\rho_f = \rho_{fo} \frac{\dot{W}_t}{\dot{W}} \ ,$$

where ρ_f = turbine fore pressure at any off design point,

ρ_{fo} = turbine fore pressure at rated conditions .

Turbine work can then be calculated by expanding the calculated quantity of dry-saturated steam at h_{g3} along an isenthalp to ρ_f and then expanding on a suitable turbine expansion line to the condenser pressure.

If the boiler feed pump has a characteristic in which flow decreases with back pressure, the change in flow must be incorporated in calculation of Δh. If the effluent from the steam generator is superheated it will be necessary to calculate boiling pressure from the system pressure drop equation. This pressure and the known Δh in the steam generator fix the amount of superheat in the effluent steam. The TOPS-MUS plant has so little superheat (50°F) at rated conditions that it quickly drops into the wet steam region as power is reduced.

Given the quasi-steady state operating conditions, the next step in stability and control analysis was the development of a mathematical model for a pressurized water reactor power conversion system including provisions for the simulation of a natural circulation primary loop with coolant circulation rate as a function of reactor core temperature rise, and a once-through steam generator with constant secondary flow rate. A simplified system drawing is shown in Fig. C. 1; a block diagram of the simulation program is shown in Fig. C. 2 ; and an analog computer diagram of the complete mathematical model is shown in Fig. C. 3. The more complex dynamics of the system are considered beyond the scope of the present study; however, conservative assumptions are made where simplification is used.

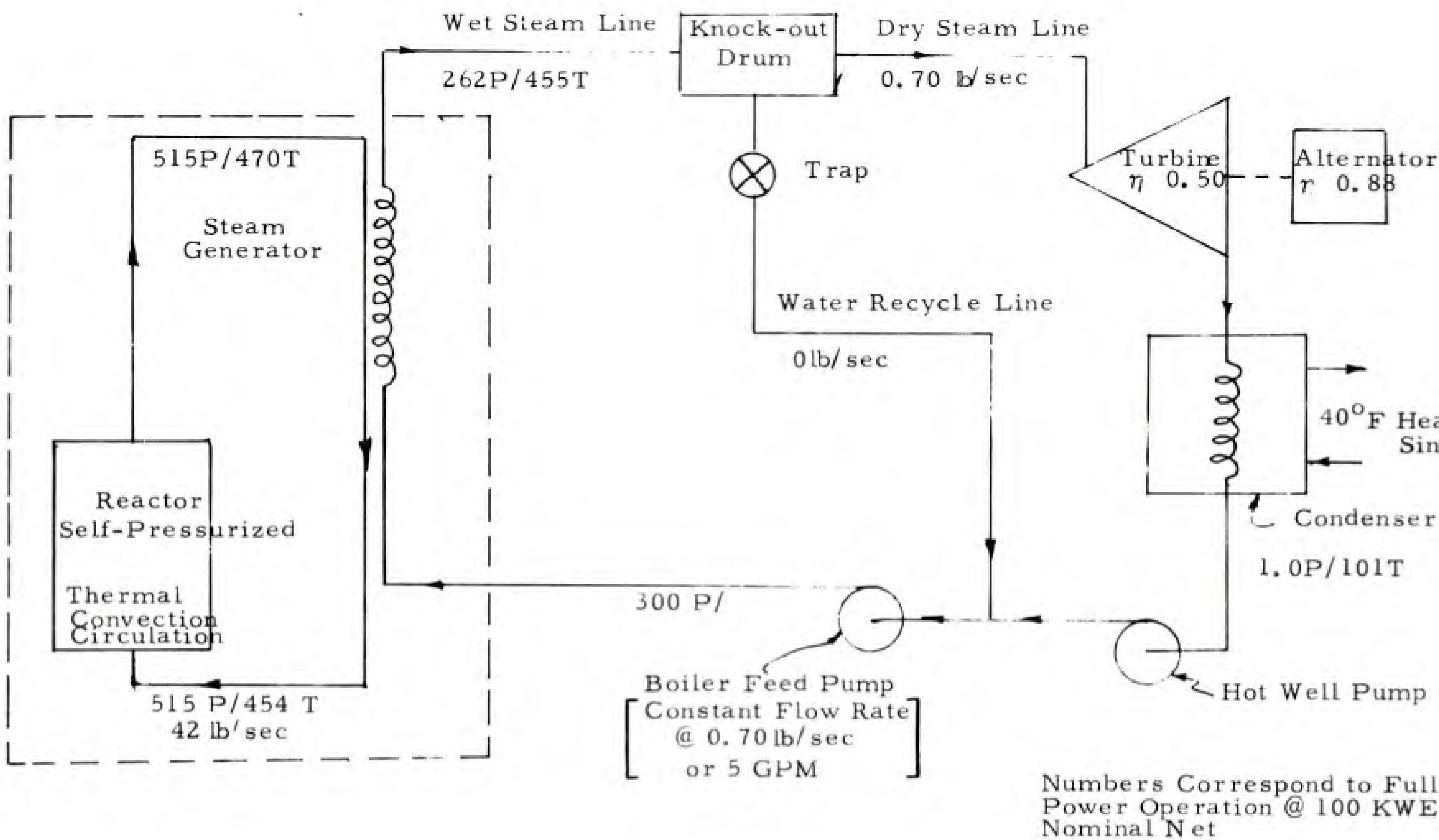


Fig. C.1--System flow diagram model for transient analysis

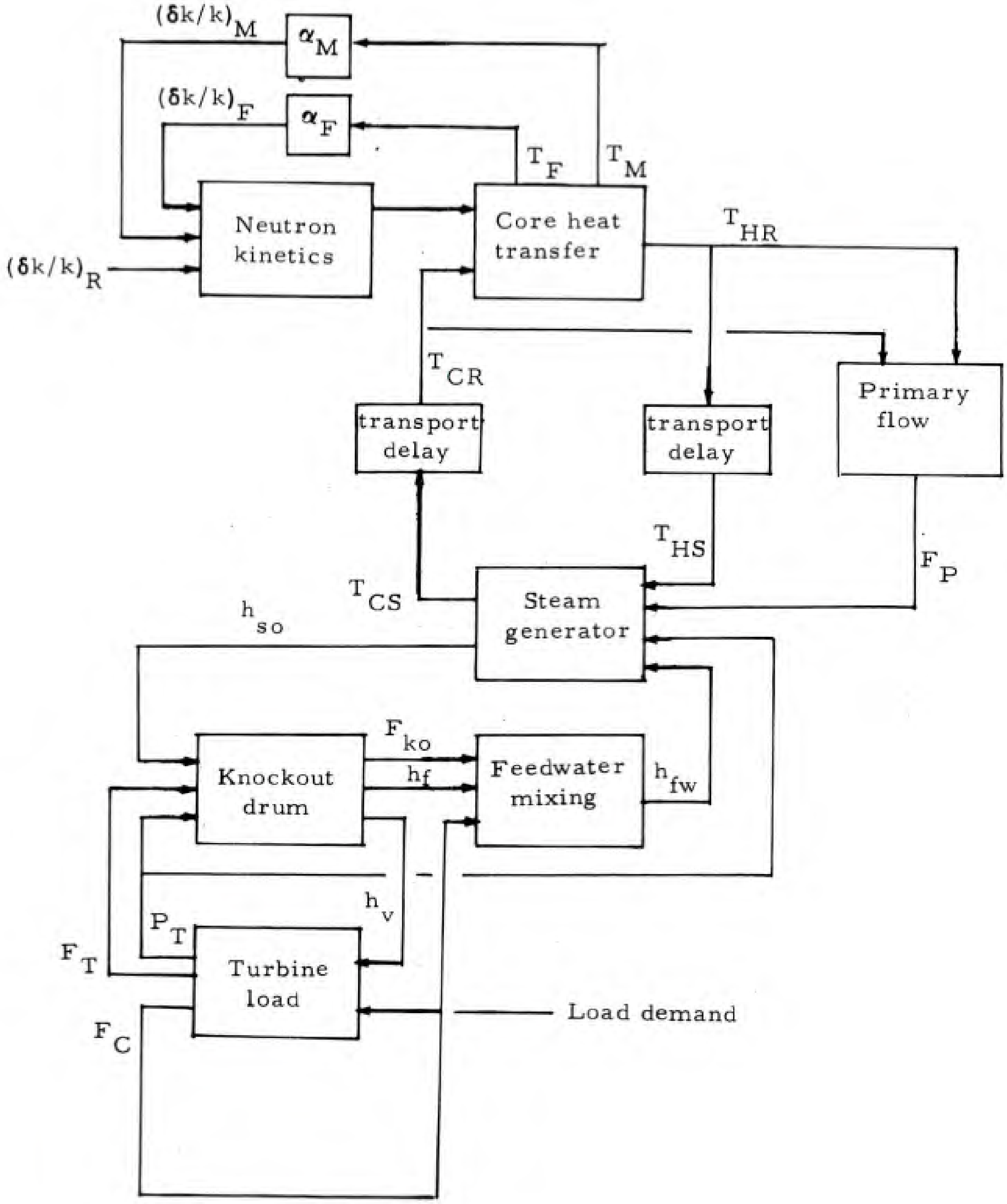


Fig. C.2--Block diagram for transient analysis

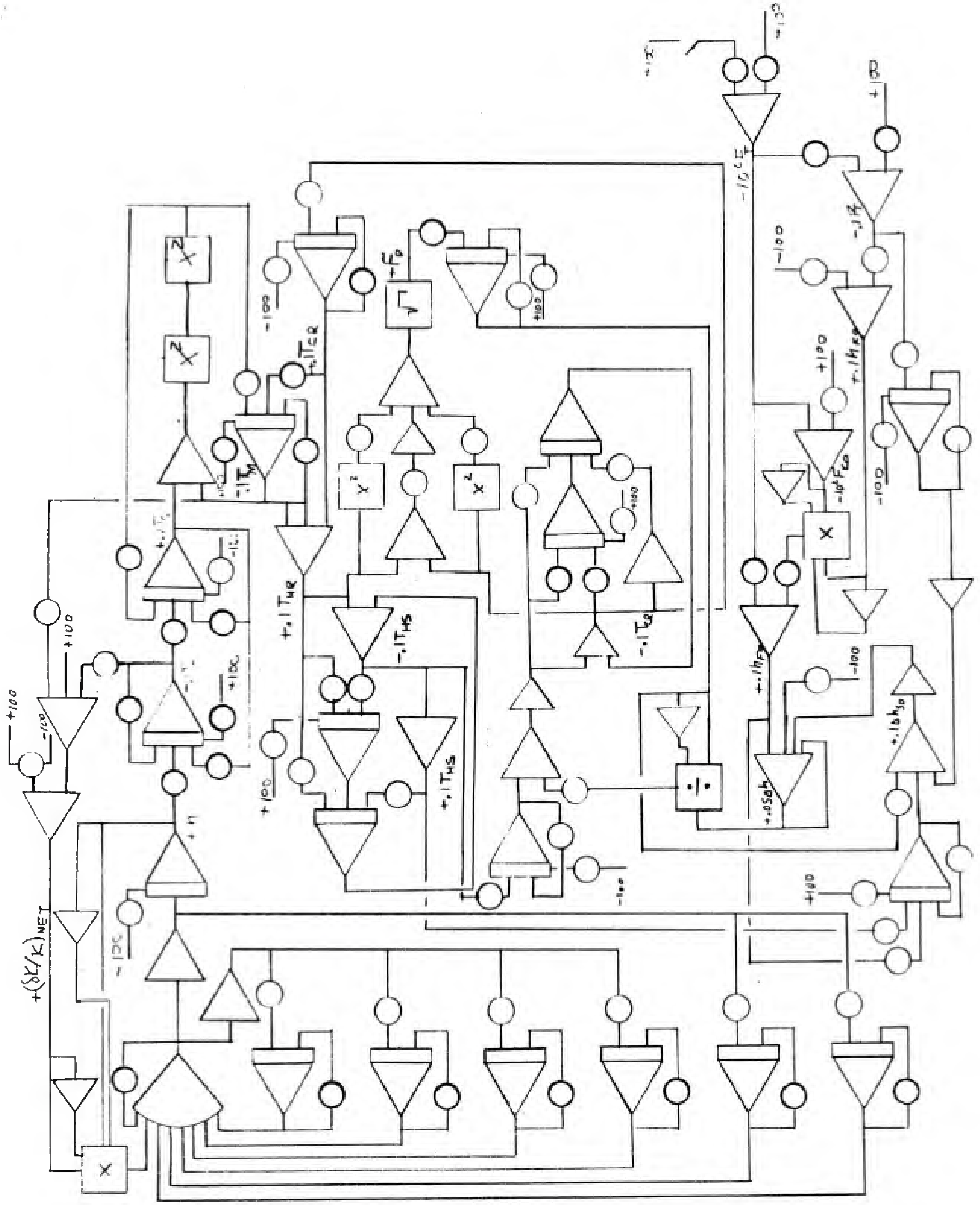


Fig. C.3--Diagram of analog computer network

C.1. PRIMARY SYSTEM

The nomenclature for the following transient analysis is defined in Section C.5.

C.1.1. Neutron Kinetics

The neutron kinetics simulation is described by

$$\frac{dn}{dt} = \frac{\delta k/k}{\ell *} \times n - \sum_{i=1}^{i=6} \frac{dc_i}{dt} ,$$

$$\frac{d^2 c_i}{dt^2} = \frac{\beta_i}{\ell *} \frac{dn}{dt} - \lambda_i \frac{dc_i}{dt} ; \qquad i = 1, 6 .$$

This is derived from the standard time variant, point source kinetics relationship.

C.1.2. Core Heat Transfer

The heat transfer equations used are shown below. The core is assumed sufficiently small and circulation sufficiently high to allow an average fuel element heat-transfer simulation.

$$C_F \frac{dT_F}{dt} = -K_{FC} (T_F - T_C) + n ,$$

$$C_C \frac{dT_C}{dt} = K_{FC} (T_F - T_C) - Q_{CW} ,$$

and

$$Q_{CW} = K_{CW} (T_C - T_M)^x ,$$

where the exponent x is calculated from steady-state temperature data. The equation

$$C_M \frac{dT_M}{dt} = \text{heat transferred from core - heat removed}$$
$$= Q_{CW} - F_P \Delta h = Q_{CW} - F_P C_P (T_{HR} - T_{CR}) .$$

Assuming an arithmetic mean temperature, T_M, for the coolant in the core,

$$T_{HR} = 2T_M - T_{CR} \quad .$$

C.1.3. Reactivity Coefficient

For this study, the reactivity is assumed to be determined by fuel temperature, T_F, and mean core water temperature, T_M.

$$(\delta k/k)_{net} = (\delta k/k)_{T_F} + (\delta k/k)_{T_M} + (\delta k/k)_{rods} = 0 \quad ,$$

where

$$(\delta k/k)_{T_F} = \alpha_F \,(T_F - T_{F_0}) \quad ,$$

$$(\delta k/k)_{T_M} = \alpha_M \,(T_M - T_{M_0}) \quad .$$

C.1.4. Flow, Transport, and Mixing Dynamics

Natural circulation flow is initiated as a result of a positive convection head generated when reactor power is increased and a temperature difference across the core is established.

$$F_P = K_1 \left[(T_{HR} - T_{CR}) - K_2 \,(T_{HR}^2 - T_{CR}^2) \right]^{1/2} \quad .$$

A finite time is required for the primary coolant to travel the distance from the reactor to the steam generator and return. This transport delay is simulated using Pade's approximation:

$$L^{-1}\left(\frac{\tau^2 S^2 - 6\,\tau S + 12}{\tau^2 S^2 + 6\,\tau S + 12} \, F(S) \right) \approx f(t - \tau) \quad ,$$

where S is the Laplace operator. Additional first order lags are used to simulate mixing in plenums and during passage through the core and steam generator volumes;

$$\frac{dT_{out}}{dt} = \frac{1}{\tau} \,(K\,T_{in} - T_{out}) \quad .$$

C. 1. 5. Primary System Pressure

The system being simulated is one employing self-pressurization; i. e., no external means of primary pressure control is used, and the steady-state value of this pressure is determined by the average temperature of the coolant leaving the reactor, assuming saturation conditions. This type of operation requires that a steam volume or steam dome be present in the primary loop. In steady-state, net boiling occurs in the coolant only in sufficient amount to maintain a constant steam content by replacing that steam which condenses due to heat leakage from the system. Thus, a steady pressure is maintained.

Under transient conditions the pressure deviates from the saturation relationship due to compression and decompression of steam in the steam dome resulting from changes in volume of primary water with temperature. The exact type of transient variation which occurs depends on the volume of the steam dome, the nature of the heat transfer occurring between steam and water, the extent of water mixing which occurs in the upper plenum, and the rate at which water flashes to steam on a decreasing pressure transient. These relationships are quite complex, and no attempt to simulate them is made in the present study because their time constants are small compared with those of other parts of the system. Instead, a continuous saturation relationship is assumed between reactor coolant outlet temperature and primary pressure.

A factor which deserves consideration, in a Phase II investigation, is the reactivity effect of the voids introduced in the reactor core by coolant boiling. Even though in steady state very little net boiling occurs, there is appreciable local boiling in the hotter channels of the core. Under steady-state conditions, the reactivity effect of this boiling is compensated by water in the cooler channels which remain subcooled, and by fuel temperature coefficient effects. Under transient conditions of decreasing pressure, however, additional void volume is produced by flashing of water into steam. The effect of this void on reactor kinetics could be significant.

C. 2. STEAM GENERATOR

C. 2. 1. Digital Computer Analysis

Previous experience at Gulf General Atomic indicates that the simulation of a once-through steam generator by the direct solution of the energy and mass conservation equations, with the appropriate variations applied to steam properties and heat transfer coefficients as functions of pressure, enthalpy, flow rate, and phase, requires computer programs that are more complex than the program described here. For this reason, and because a digital computer program for simulation of a once-through steam generator exists in a developed state, it was decided to modify the existing code,as required,to represent the design being considered here, and to determine dynamic characteristics of the steam generator using the digital code. Then a relatively simple analog computer program was developed to produce similar static and dynamic responses as the digital program, but without the same internal complexity.

The digital code has been written for nuclear heated helium on the primary side, and includes provisions for an integral reheater. This code was changed to provide for pressurized water on the primary side, and was simplified in some respects to conform to the less complex design. The program includes the classical mass and heat transfer relationships and accounts for the varying quantity of heat stored in primary and secondary water, and in the tube metal. Correlation between pressure, enthalpy, temperature, and specific volume for the water on both sides and in each phase are determined by double interpolation of steam table data. In addition, the program includes the calculation of locations at which the secondary water passes the saturated liquid and saturated vapor enthalpies, such that different heat transfer relationships may be used for each phase. These transition locations are also printed out at each time so that movement of the transition points during transients may be observed.

Steam generator design data for input to the dynamic model of the steam generator was taken from the results of a steam generator sizing digital code. The dynamic responses of the steam generator to various perturbations in feedwater enthalpy, secondary pressure, and primary inlet enthalpy and flow were computed to obtain the necessary response information to develop a simple analog computer model of the once-through steam generator for incorporation into the analog computer program of the complete system. The responses were obtained at reactor power levels of full power and again at minimum load to note the differences at the extremes of the operating range.

C. 2. 2. Analog Model

Following the determination of transient response of the once-through steam generator with the digital computer program, an analog computer model of the steam generator was devised such that this model demonstrated similar dynamic behavior to that of the digital program. This analog model produces output responses for the cold primary temperature, and hot secondary enthalpy only, and does not provide any information regarding steam generator internal temperatures or phase transition locations. In the equation below, the variables are considered to include the correct time response relationships as determined in the analysis of the digital computer results. For a heat balance across the steam generator,

$$F_P C_P (T_{HS} - T_{CS}) = F_{FW} (h_{SO} - h_{FW}) \ .$$

The proper time lags among the variables are included in the analog model.

C. 3. SECONDARY SYSTEM

C. 3. 1. Steam/Feedwater System

At reduced loads wet steam is produced by the steam generator. This passes through the knockout drum which separates dry steam sent to the turbine and saturated liquid which is mixed with condensate flow. For heat balance:

$$F_{FW} h_{SO} = F_{KO} h_f + F_T h_v \ ,$$

and

$$F_{FW} h_{FW} = F_{KO} h_f + F_C h_C \ .$$

For mass balance:

$$F_T = F_C = F_{FW} - F_{KO} \ .$$

For the system under study, feedwater flow is fixed by the constant feedwater pump speed and condensate enthalpy by the condenser size. Both are considered invariant. Saturated liquid and vapor enthalpies are calculated from secondary pressure which is a known function of load and adjusted as the turbine speed governor operates with load change.

C. 3. 2. Turbine Load System

In order to use the program for a study of turbine speed control capabilities, additional information is required, including turbine inertia, turbine load, and torque characteristics, and magnitude and rate of load changes. Again, this is beyond the preliminary nature of this study and the assumption made is that of step changes in load and pressure which is not unreasonable for a small machine with a fast-acting speed governor.

C. 4. ANALYSIS

To simplify the analysis, many of the nonlinearities of the system were made piece-wise linear in the two operating limits of interest. This resulted in two versions of the analog model with different coefficients.

Steps in load were attempted in the regions of full load and minimum load. Figures C. 4 and C. 5 show the parameters of the system transient response to approximate 10% load changes near full power and minimum load. Although a sophisticated control system would probably improve the transient response, there is no indication of instability or dangerous overshoot at either level. It is thus assumed that even without control of any kind the proposed system is load-following and stable over its full range of operation.

C. 5. DEFINITION OF TERMS

C_C = thermal capacity of cladding

C_F = thermal capacity of fuel

C_i = concentration of delayed neutrons of group i

C_M = average heat capacity of coolant water present in core

C_P = specific heat of water

F_C = condensate flow

F_{FW} = feedwater flow

F_{KO} = saturated water flow from knockout drum

F_P = primary coolant flow

F_T = turbine flow

h_C = condensate enthalpy

h_f = saturated liquid enthalpy

h_{FW} = feedwater enthalpy

h_{SO} = steam generator outlet enthalpy

h_v = saturated steam enthalpy

K_{CW} = film heat-transfer coefficient

K_{FC} = fuel-to-cladding heat-transfer coefficient

K_i = proportionality constant for primary flow calculation; $i = 1, 2$

$\delta k/k$ = reactivity

Δh = enthalpy rise of coolant in passing through steam generator

ℓ^* = mean effective neutron lifetime

n = reactor power

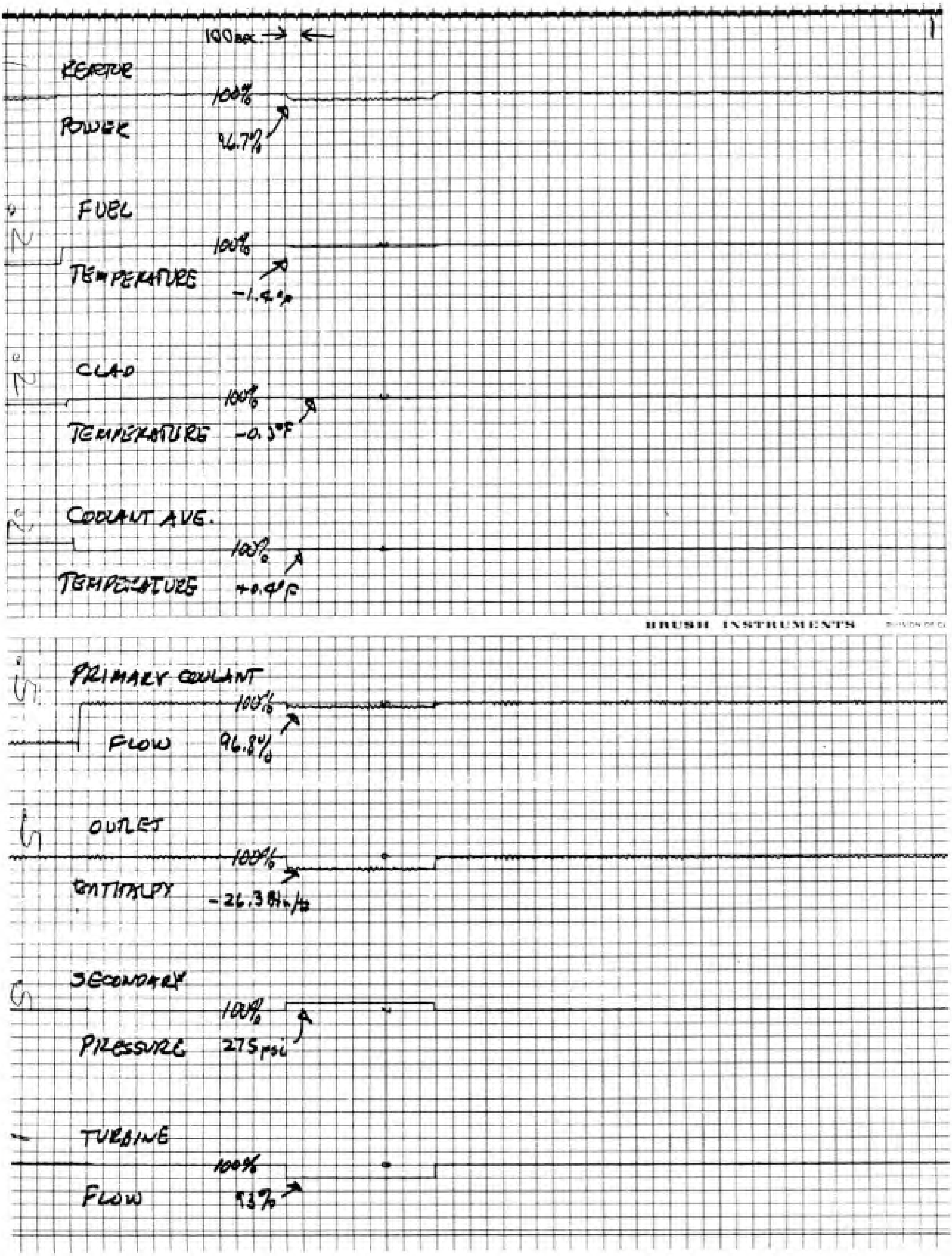


Fig. C.4--Dynamic response to a step turbine load from full load

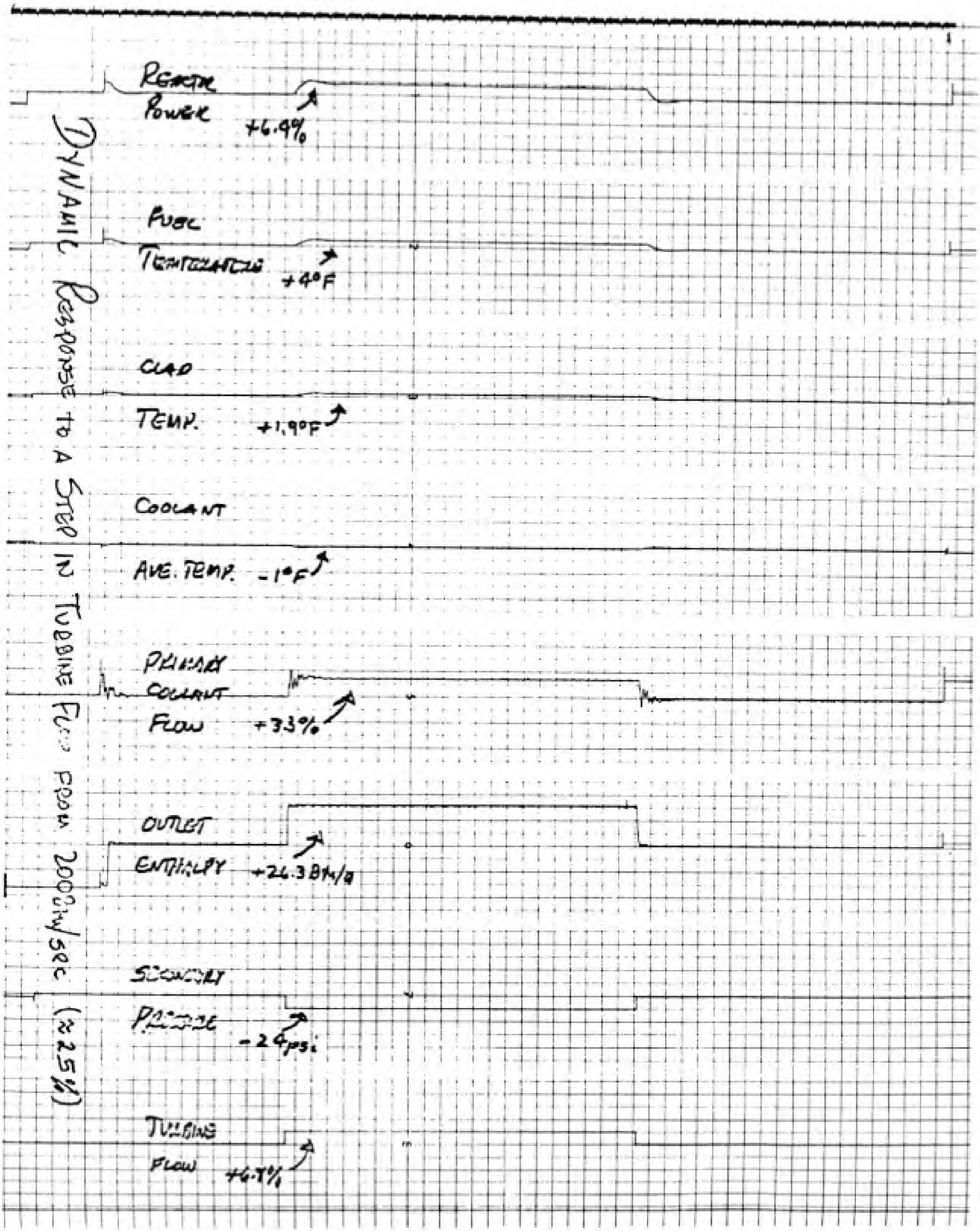


Fig. C.5--Dynamic response to a step in turbine load from minimum load (~25%)

P_T = turbine pressure

Q_{CW} = rate of heat transfer to coolant

t = time

T_C = cladding average temperature

T_{CS} = temperature, cold steam generator (primary)

T_{CR} = temperature, cold reactor

T_F = fuel average temperature

T_{F_0} = reference value of average fuel temperature

T_{HS} = temperature, hot steam generator (primary)

T_{HR} = temperature, hot reactor

T_M = average temperature of coolant in core

α_F = fuel-temperature reactivity coefficient

α_M = moderator temperature reactivity coefficient

β_i = effective neutron fraction of delay group i

λ_i = decay constant of delay group i

τ = time delay .

APPENDIX D

PRIMARY SYSTEM WATER TREATMENT

In the determination of the treatment of the primary system water, requirements of minimum operational maintenance, plant simplicity, and reliability are dominant. Thus, the approach used has been one where justification must be made for each procedure, component, or additive to be incorporated into or used on the primary loop, rather than justification of omissions of usual water-treatment measures. Nevertheless, in the following discussion, some reasons are presented which permit deviation from what is normally considered standard, utility-power plant, water-treatment practice. It should also be noted that as operational experience has accumulated in conventional and nuclear power plants, there has been significant relaxation of the initially over-stringent procedures and specifications. This trend is continuing; thus, while the course taken here in the interest of plant operational and maintenance simplicity has been one of complete justification of requirements, allowances should be included for additions to the system as demonstrated to be required during initial testing of the system.

Conventional water-treatment involves the addition of LiOH, morpholine, cyclohexylamine, or NH_4OH to control the pH, an O_2 getter such as hydrazine, filters to remove particulate and suspended solids, and a demineralizer to reduce dissolved solids. In nuclear power plants the primary circuit is also subjected to radiolytic decomposition of H_2O into H_2 and O_2. Usually a catalytic recombiner is included in the circuit in the gas phase to accelerate recombination. In addition, H_2 cover gas is often used so as to promote recombination in the water through adjustment of the equilibrium concentrations for the reactions.

The general reasoning that has been adopted and that is believed will permit elimination of primary water treatment is described as follows: The primary circuit in initially clean condition will be filled with pre-treated water and hermetically sealed. Residual dissolved oxygen in the water will initially oxidize surfaces in the primary circuit forming a protective barrier over the metal to inhibit further corrosion. Corrosion rates comparable to those of stainless steel in pH-controlled water are anticipated. Removal of O_2 in the formation of the oxide barrier will leave a surplus of H_2 in the loop water and cover gas which will shift the equilibrium concentrations of the reactions and thereby enhance recombination rates of the H_2 and O_2 subsequently generated in radiolytic decomposition.

The design of the primary loop has several general features which affect the water-treatment requirements. One of the primary features is the selection of Incoloy-800 and 304 stainless-steel as the exclusive materials of construction of the primary circuit. The very high resistance of these materials to corrosion in pressurized water is a distinct advantage. Second, hermetic sealing of the primary loop limits the oxygen in the system to that residual oxygen in the fill water plus any oxygen generated in radiolytic decomposition. The oxygen and impurities introduced in the feed-water of conventional systems and nuclear reactor power plants, the major source of impurities, have been eliminated. Next, the primary circuit is a natural convection loop. As such, the flow velocities are low and the available driving pressure head, which is dependent on the density differences in the hot and cold water, are small, thereby eliminating the possibility of using demineralizer beds or mechanical filtration. In addition, the minimum temperature of the primary circuit is approximately 455°F, a temperature exceeding the capabilities of proven demineralizer and filter beds. Finally, the low velocities of flow associated with natural convection should limit the rate at which crud leaves the surfaces and enter the water, and perhaps enhance the settling of crud on the bottom of the reactor pressure vessel, where it would be innocuous to heat transfer surface fouling and scale buildup.

While it is apparent that the primary loop water is treated to limit general corrosion, the specific objective in the water treatment of this system is to prevent fouling of the heat transfer surfaces which might result in unacceptable degradation of the plant performance or failure of the power plant. Corrosion of the primary circuit should be uniform without pits and at a rate of approximately 5 mg/dm^2-month.[(1-3)*] Design allowances for pitting, wall-thinning, and other structural deterioration resulting from corrosion are adequately included by other design requirements of the primary loop. The principal concern is that of fouling of the heat transfer surfaces and possibly the activation and distribution of radioactive crud as it would affect repair and maintenance services. Accumulated radioactive crud deposits can be removed between MUS cycles by decontamination procedures that have already been developed.

Pretreatment of Loop and Water

Treatment of the primary loop water essentially is a pretreatment to be accomplished prior to hermetically sealing the loop. No treatment will be required during operation as long as the seal is not broken for refueling, unforseen maintenance and repairs, or seal failure. Pretreatment will consist of cleaning of the loop according to standard procedures,

*References are listed at the end of this Appendix.

filling the loop with pretreated water, in situ cleaning of the water, preconditioning of the system, and pressurizing with an initial cover gas of hydrogen. The final procedure will be completion of the hermetic seal at the last opening, and its testing for tightness.

It is assumed that in the construction of the reactor, particularly the primary loop, high standards of construction cleanness characteristic of reactor assembly will be maintained. Initial pretreatment will require filling of the primary circuit with pretreated water; that is, de-oxygenated, distilled, and/or demineralized water. Water will be circulated in the primary circuit and externally through equipment attached for pretreatment. This equipment will include particulate filters, full flow demineralizer, and mechanical de-aerators. During this procedure the temperature of the water will be raised and controlled by an electric heater. A provision will be included for addition of water treatment chemicals and for taking water samples for analysis and control. Trapping of residual air in the top of the reactor vessel will be avoided and the space will be filled by an inert gas or hydrogen at a pressure exceeding ambient to ensure the exclusion of all oxygen and nitrogen. This will be accomplished as described in Section 3.7.2. A hydrogen partial pressure over water of about 2.5 atmospheres will be required. If the final closure of the hermetic seal is made in the gas space above the water, a small addition of helium to the cover gas will permit the use of the helium-mass-spectrometer leak detector to ensure hermetic tightness.

The TOPS/MUS primary water treatment can be characterized as:

1. Precleaning primary surfaces;
2. Pretreatment of fill water to:
   a. De-aerate,
   b. Deoxgenate,
   c. Demineralize,
   d. Mechanically filter,
   e. Distill, and
   f. Form inhibiting oxide coating;
3. H_2 overpressure of a few atmospheres;
4. No chemical additives to control pH or getter O_2 and CO_2;
5. No demineralizer;
6. No mechanical filter (possibility of magnetic or electromagnetic filtering);
7. No catalytic recombiner.

Discussion of Water Treatment Factors

In the treatment of the primary loop water there are many factors that need to be considered; the principal ones are:

1. Chloride stress corrosion,
2. Intergranular stress corrosion,
3. Caustic cracking,
4. Cathodic depolarization,
5. Radiolytic decomposition,
6. pH control,
7. Heat transfer surface fouling,
8. Filters and traps,
9. Fuel rod failures, and
10. Water condition monitoring,

Chloride stress corrosion is not expected to be a problem in the primary circuit. This corrosion mechanism requires three ingredients:

1. Chloride ions in concentrations of 25 ppm or greater with associated oxygen concentrations of 10 ppm or more,[(4)]
2. Stresses approaching the yield stress, and
3. Alloys sensitive to the mechanism.

Materials of construction of the primary circuit have been selected to avoid chloride contamination, and pretreatment of the water should eliminate any initial chloride ions. The major components (steam generator, fuel elements, and pressure vessel) of the primary circuit are made from Incoloy-800, an alloy practically immune to this corrosion type. Some core structural parts, such as the core barrel and grid plates, are 304-stainless steel, but these members do not carry critical loads and thus are lightly stressed. Therefore, at least two of the three requisites for chloride stress corrosion are absent from all points of the primary circuit.

Intergranular stress corrosion has also been observed[(5)] in high-nickel alloys, particularly Inconel-600, in high-purity water at elevated temperatures. Susceptibility to this stress corrosion effect increases with Ni content. It has been stated[(5)] that alloys of 35 to 45% nickel content should withstand stress corrosion cracking in chloride media and in pure water at high temperatures as well. Incoloy-800 therefore should avoid both chloride and intergranular stress corrosion.

Caustic cracking is another corrosion mechanism which is not a concern in the primary circuit. In some power plant water treatments where NaOH or LiOH is used to control alkalinity, this may become a problem if excess caustic is not eliminated, particularly from regions of high stress level and in crevices or cracks where circulation and mixing are inhibited. These chemicals will not be added to the primary water of the TOPS/MUS.

Cathodic depolarization is the usual corrosion mechamism found in power plant systems, even those with complete water treatment. Unless inhibited, it is this mechanism that permits metal to continuously enter the water as ions, that is, to corrode. Depolarization occurs according to the following reactions:

$$2H^+ + 2e \rightarrow H_2 \qquad (A)$$

$$0_2 + 2H_2O + 4e \longrightarrow 4OH^- \quad . \qquad (B)$$

Reaction (A) is typical of acid water conditions with H_2 leaving the cathode of the galvanic cell as hydrogen bubbles, as dissolved gas, or by reaction with O_2 to form water. Reaction (B) occurs in neutral or alakline water conditions and is the more usual form encountered in power plant practice. The hydroxyl ions formed typically combine with metal ions at points removed from their origin, thus avoiding anode polarization which would inhibit further corrosion. The $Fe(OH)_2$ formed in this manner is subsequently reduced to Fe_3O_4, which becomes one of the major suspended solid constituents with the potential to foul the heat transfer surfaces of the fuel element and the steam generator. The addition of hydroxyl ions to the water in reaction (B) will cause the pH to rise to values as high as 9.6, corresponding to the saturated solution of $Fe(OH)_2$. With these conditions and full oxygen consumption, corrosion by reaction (B) would essentially be terminated. If, however, additional oxygen is available, then the $Fe(OH)_2$ will be oxidized in the presence of water to $Fe(OH)_3$, which is insoluble. Precipitation of $Fe(OH)_3$ will effectively remove the hydroxyl ions and the pH will fall back toward the neutral condition. While the residual amount of oxygen in the pretreated water may be kept to very low values, radiolytic decomposition of water must be considered as another source of oxygen, which may enter into reaction (B).

Cathodic depolarization may also be accomplished by hydrogen according to reaction (A) in water with pH values of less than 7. To prevent corrosion with hydrogen depolarization, the pH value of water should not be below 7; this means that free carbonic acid must be eliminated (i. e., CO_2) in the pretreatment of the water. The use of a hydrogen cover gas above the liquid in equilibrium with hydrogen dissolved in the water will drive reaction (A) far to the left. This will prevent hydrogen depolarization corrosion and will maintain the pH of the water at a value of at least 7.

Radiolytic decomposition of water has been mentioned above as an alternate source of oxygen to that residually contained in the pretreated water by which continuous corrosion of the metal may occur. In the radiolysis of water, the initial decomposition products include free radicals, as shown in the following reaction:

$$H_2O \rightsquigarrow H,\ OH,\ H_2,\ H_2O_2,\ HO_2\ . \qquad (0)$$

It is noted that there is no molecular oxygen among these initial reaction products to further corrosion. However, along the trajectories of the ionizing particles or rays, recombination reactions occur

$$OH + H_2 \longrightarrow H^1 + H_2O\ , \qquad (1)$$

$$H + H_2O_2 \longrightarrow OH + H_2O\ , \qquad (2)$$

$$H^1 + H_2O_2 \longrightarrow OH + H_2O\ , \qquad (2^1)$$

where H^1 indicates a solvated electron (after Allen[6]). Other reactions follow which may be more spatially distributed. These reactions which inhibit the reformation of water are:

$$OH + H_2O_2 \longrightarrow H_2O + HO_2\ , \qquad (3)$$

$$H + O_2 \longrightarrow HO_2\ , \qquad (4)$$

$$H^1 + O_2 \longrightarrow HO_2\ , \qquad (5)$$

$$2HO_2 \longrightarrow H_2O_2 + O_2\ . \qquad (6)$$

In the above reactions, the unionized H and OH forms are free radicals with their characteristic short life and large reactivity. It has been found[6] that the initial decomposition product HO_2 is not characteristic of reactor (neutron and gamma) radiation, but is prevalent in alpha and fission fragment induced radiolytic decomposition.

The generation of oxygen through radiolytic decomposition can be accomplished only through the combination of Equations (3) and (6), since combinations of Equations (4) and (5) with Equation (6) result in the yield of only one mole of oxygen for every 2 moles of oxygen entering the reaction. When an excess of hydrogen is introduced into the water, then reaction (3) suffers in comparison to reaction (1) where H_2O_2 and H_2 are competing for the hydroxyl radicals, respectively, since the ratio of their reaction rate constants, $k1/k3 = 1$. Also, reaction (5) has a rate constant many times that of reactions (2), (2^1),or (4) so that consumption of oxygen and increasing concentration of H_2O_2 continue until very low oxygen concentrations are reached (10 μmoles/l, according to Reference 6). Then, at very low oxygen concentrations and low ratios of O_2 to H_2O_2 concentrations, H_2O_2 is eliminated via reactions (2^1) and (2). As a result of these reactions, it is reported[4, 6] that O_2 and H_2O_2 have been completely eliminated from irradiated water. Thus, beneficial effects of prepressurizing the TOPS primary with H_2 gas in equilibrium with H_2 in solution will result. Nevertheless, Allen[6] indicates that low steady-state concentrations should be expected since radicals diffusing out of different spurs and tracks and finding little to react with will react with each other. The corrosion rate in the TOPS primary can therefore be expected to be limited by the source of oxygen, resulting from secondary radiolytic effects since hermetic sealing eliminates feedwater as a source. Associated with the corrosion will be slow evolution of H_2 gas which may obviate initial pressurization.

Continued generation of hydrogen gas by radiolytic decomposition of water could result in increasing pressures within the pressure vessel unless an equilibrium condition were attained within the structural limits of the pressure vessel design and code. Molecular hydrogen is eliminated through reaction (1). In pure water, equilibrium concentrations of hydrogen in solution are attained at very low concentration levels (about 1 millimole per liter). However, radical scavengers which compete for the radicals in the back reactions (1) and (2), have been shown[4, 7] to have a dramatic effect on the hydrogen concentrations at which equilibrium is attained. Typical radical savenging reactions proceed as shown:

$$RS^- + OH \longrightarrow RS + OH^- \qquad (11)$$

$$RS + H \longrightarrow H + RS^- \qquad (12)$$

where RS is a generalized radical scavenger molecule. Rate constants for scaverngging reactions are unavailable, and only qualitative information is known. The presence of multivalent cations or of anions of low electron affinity tend to raise the equilibrium hydrogen concentration, while alkali hydroxyde behaves in a manner equivalent to pure water.[4] It has been stated[4] that the presence of excess hydrogen has a stabilizing effect, and that under this condition, no measurable amounts of either peroxide or oxygen were formed during the irradiation of water in hermetically-sealed capsules, and the hydrogen concentration remained constant. In addition it is reported[1] that a series of recombination reactions occur above 200°C that result in a net recombination formation of water at a rate equivalent to the production of primary free radicals of H and OH. This deduction was based on the work reported in Reference 8. It has been reported[7] that when water is contaminated with iron, manganese, and chromium, to levels up to 100 mg/liter, the saturation pressures at temperatures of 200°C (~400°F) and above, will be of the order of one atmosphere. Since such high solute concentrations will not occur in practice, saturation pressures in heterogeneous water reactors operating at water temperatures of 200°C or higher will be extremely small. It is noted that over the full range of electric power generation from zero to design capacity, primary water temperature will exceed 400°F in all regions of the flow circuit. In addition, it has been reported,(4) with respect to the Shippingport PWR, that at temperatures up to 300°C when the electrical conductivity of the water is less than 1×10^{-6}/ohm-cm, and no radiolytic decomposition of water is observed, accumulation of hydrogen occurs at a rate which corresponds to that of the corrosion of the stainless steel surfaces and oxygen concentrations are always vanishingly small. Furthermore, it was reported(9) that the Elk River reactor has been operated with its catalytic recombiner in the primary circuit being inoperative. During this operation, the primary loop cover gas pressures rose slightly to an acceptable equilibrium level.

The foregoing suggests that it should be possible to operate a hermetically-sealed primary loop of the TOPS/MUS system without a catalytic recombiner and still remain well within the pressure limitations of the reactor vessel. The equilibrium pressure level will depend primarily on the kind of radical scavengers and their concentration as a result of the corrosion processes in the vessel. With the level of purity expected to be attained in the primary system (conductivity equal to or less than 10^{-6}/ohm-cm), equilibrium partial pressures of hydrogen in a range of 2 to 5 atmospheres are expected. However, because of the inadequacy of fundamental information with respect to radical scavengers, their reaction constants and concentrations, confirmation of the equilibrium pressures developed by radiolysis of primary water will have to be obtained during the initial checkout of this reactor system. If,in the unlikely event that a catalytic recombiner should prove to be required,it will be added.

The pH level of the water in standard power plant practice appears to be the single-most effective parameter for the control of corrosion. In all-ferrous systems this is usually controlled to the range of 9.5 to 12. When copper-bearing materials are used in condensers and other components, lower pH values (7.5 to 9) are frequently found to be advantageous. It has been shown(10) that in static and very low velocity systems, corrosion rates of mild steel at a pH of 7 are lower than at high pH values. However, the velocities and Reynolds numbers in the critical regions of the TOPS primary loop exceed the conditions required for the low pH protective mechanism involved. On the other hand, large reactor power plants and in-pile loops have been operated without chemical treatment of the primary water during operation with neutral water.(7,11) Corrosion of the primary system is reported as very limited after approximately 2-1/2 years of operation. Suspended solid content was high but resulted from impurities incoming with feed water, a situation which is not applicable to a hermetically-sealed primary system. Recent experiments in boiling water and superheated steam mockups of reactor operation[12] were conducted with high-purity neutral water recirculating with 0.2 ppm oxygen and no chemical additives. Corrosion rates less than 10 mg/cm^2-month were observed in a carbon steel system over periods exceeding 10,000 hrs. In Reference 13, where the pH varied from 7.5 to 8.5, the corrosion rate was lower than 5 mg/dm^2-month. In a hermetically-sealed system where the feedwater as a source of oxygen has been eliminated, the pH of initially neutral water is expected to rise with the consumption of any residual oxygen in the corrosion process, as indicated above. With hydrogen overpressure used to suppress oxygen generated by radiolysis of water, the pH of the water is expected to vary between 7 and 9.6, with a more-or-less constant value nearer 7.

Surface fouling of heat transfer surfaces of the primary loop was indicated above to be the major feature of the general corrosion phenomena to be of concern in TOPS. In the primary loop composed of Incoloy-800 and 304 stainless-steel, corrosion will be uniform at a rate of approximately 5 mg/cm^2-month.[1-3] Thus, design allowances for pitting, wall-thinning and other structural deterioration resulting from corrosion are adequately included by other design requirements for the primary loop leaving fouling as the principal concern. Although substantial efforts have been made over the past decade or two with respect to fouling of heat transfer surfaces, it is still very difficult to assess this factor for the TOPS/MUS primary circuit. Little or no work has been done on Incoloy-800 corrosion in pressurized water, even though a significant program is current for superheated steam conditions.[14,15] Furthermore, previous work[1] has shown large differences in the fouling of zircoloy and stainless steel surfaces, which, as far as is known, have not been satisfactorily explained. Thus, extension to the conditions of present interests are not possible. However, the TOPS/MUS application requires a heat load of only 20% of the design levels of the system at full electrical output of 100 kw. If it is assumed that

1. A corrosion rate of 10 mg/dm^2-month,
2. All corrosion products are released to the water as crud,
3. All crud is deposited on
   a. Fuel elements, or
   b. The steam generator,
4. The density of the deposit is 2.6 grams/cubic centimeter,[1]
5. The thermal/conductivity of the scale is 0.39 Btu/hr-$^{\circ}$F,[1] then

the temperature rise across the scale amounts to approximately 1 $^{\circ}F$ after 1 year of full-power operation. All assumed values were chosen so as to make the calculated temperature difference approach the upper limits of expected values. Thus, in view of the results of the computation, it is difficult to postulate how surface fouling could lead to loss of performance or failure of the system by over-temperature of the heat transfer elements of the system (fuel elements and steam generator).

Filters and Traps to remove particulate and suspended matter from the naturally-convecting stream of the primary loop have not been included. Pressure losses through filter beds would result in additional and undesirable riser heights to maintain circulation at rates necessary for cooling. Furthermore, since the temperature rise across the reactor is quite small (approximately 17 $^{\circ}F$), the minimum temperature of the loop is approximately 450 $^{\circ}F$; this temperature is a deterrent to the use of many conventional types of mechanical filters.

The low velocity and small density difference between water and crud might permit settling of some crud to the bottom of the reactor tank. Trapping of this material could effectively remove it from water circulation and as potential material for deposition. Magnetic trapping using permanent or electro-magnets or a combination might be considered. A large percentage of crud even from stainless-steel loops[11, 16] is reported to be Fe_3O_4; Fe_3O_4 and gamma Fe_2O_3, two of the main corrosion products, are both ferro-magnetic materials.[17] In addition, many ferrites and spinels, including many nickel and chromate-bearing compounds, are similarly ferromagnetic.[17] Since the flow forces driving or suspending the solids are not great in the low-velocity stream (less than 0.5 fps) a significant fraction of the ferromagnetic component of the crud might be trapped and held, possibly at the bottom of the reactor vessel, to preclude their deposition on the heat exchanger surfaces. Flow resistance of a trap of this nature would be expected to be very low (i. e., since the forces on the particles are small, the converse should also be true). Water temperatures are well-below the Curie point for high permeability, permanent, and electro-magnet core materials, as well as for the suspended materials (Fe_3O_4). It would seem that by magnetic trapping and/or pumping, it should be possible to create a concentration gradient which would permit removal of the suspended ferromagnetic solids (and possibly non-ferromagnetic materials by an interference mechanism) from the water stream. While traps suitable for this service are not presently available,[18] development of such a trap should be considered for inclusion in a research and development program.

Failure of fuel element cladding so as to expose the U-ZrH$_x$ fuel-to-water corrosion is a condition which will have a finite probability of occurring even with stringent quality control procedures. Operation of TRIGA reactors over a period of more than ten years in steady and pulsing operation modes with deliberately compromised cladding sheaths and infrequent failures have provided an adequate background of information to assure continued operation of the system without danger of corrosion damage over extended periods of time.

The hydride fuel has excellent corrosion resistance in water. Bare fuel specimens have been subjected to a pressurized water environment at 570°F and 1230 psi during a 400-hr period in an autoclave.[19] The average corrosion rate was 350 mg/dm^2-month weight gain, accompanied by a conversion of the surface layer of the hydride to an adherent oxide film. The maximum extent of corrosion penetration after 400 hr was less than 2 mils.

Experiments carried out at Gulf General Atomic show that the zirconium hydride systems have a relatively low reactivity in water, steam, and in air, at temperatures up to 850°C. These tests have involved the quenching in water of U-ZrH specimens after heating to as high as 850°C.

Monitoring of primary water conditions will be difficult with a hermetically-sealed system. Continuous monitoring is discouraged or eliminated by the desire to minimize the number of penetrations and seals which must be formed. Similarly, intermittent monitoring is discouraged by the number of sampling ports required and the number of seals which must be repetitively destroyed and resealed without contamination of the system. As presently designed and planned, it will be impossible to monitor the primary water during operation of TOPS/MUS in the submerged environment. In practice, water samples could be obtained every 30 days during the surfacing of the MUS for crew changes and other logistic support requirements. However, to avoid the possibility of the closed cycle where contamination during monitoring leads to corrosion which requires more careful control and further monitoring, it is desirable to make the sampling of primary loop water as infrequent as possible. The required frequency of water condition measurements will be determined during the checkout and demonstration operation of the system. An objective of 1 to 3 years between measurements is considered to be reasonable.

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APPENDIX E

REVIEW OF INCOLOY-800

Incoloy-800 was developed by International Nickel Company about 25 years ago as an alloy with high creep strength and good resistance to oxidation and carburization at elevated temperatures. Incoloy-800 has been used extensively in the petrochemical field for reformer and cracker tubes (over ten years of continuous operation at 1500°F has been reported). It has also been used in the heat-treating field for mufflers, radiant tubes, and as a sheath material for electrical resistance heating elements. Incoloy-800 has been selected as a construction material in many chemical plants where stress corrosion has been a problem with stainless steels. Over 14 million pounds of this alloy were sold in 1964.

In recent years Incoloy-800 steam superheater tubes have been installed in a number of power plants. This trend has resulted because Incoloy-800 is free from the trouble experienced with stress corrosion of stainless steel superheater tubes. No failure of Incoloy-800 has been reported in superheated or wet steam or in high temperature water (neutral or alkali pH), even when contaminated with chloride and oxygen. The tests have included high-pressure steam temperatures up to 1500°F for periods of up to 18 months. The Incoloy-800 reheater tubes in the Oak Creek Power Plant (Wisconsin Electric Power) are reported to have been in operation for 4 yrs at a steam temperature of 1000°F, 550 to 600 psi. Incoloy-800 alloy has been selected by the General Electric Company Atomic Power Equipment Department, for nuclear superheat applications as a result of extensive stress corrosion tests in contaminated superheated steam at temperatures up to 1300°F for 10,000 hours.

In the General Electric tests, Incoloy-800 was also corrosion-tested under heat-transfer conditions at metal temperatures up to 1410°F in superheated steam facilities. The hydrogen and oxygen contents of the steam were controlled to simulate those found in boiling-water reactors. The corrosion data from the 4,000 hr heat-transfer tests indicated good corrosion resistance up to at least 1300°F.[1]

The following is excerpted from General Electric Report GEAP-4633, "Incoloy-800 for Nuclear Fuel Sheaths," compiled by C. Spalaris, dated July 1964.

"Incoloy-800* was selected in 1961 as fuel cladding material for nuclear superheat reactors. At that time, the alloy was one of a group of possible candidates. Because of its lower neutron cross section - relative to other high-nickel alloys - and because of its successful use in petrochemical and household appliance industries, Incoloy-800 was considered first in relation to the other alloys selected for study. Since 1961, fuel elements sheathed in Incoloy 800 performed satisfactorily in a superheat environment for 135 days without failure, whereas fuel elements clad in Type-304 stainless steel failed within 29 days. In other tests, Type-304 stainless steel failed after even shorter periods of time (10 to 15 days). In corrosion loops where superheated steam conditions were simulated, Incoloy-800 performed satisfactorily for 63 days without failure. Type-304 stainless steel failed within 2 to 3 days in similar experimental runs. Under stress conditions in a corrosion loop that included temperatures of 1150°F and initial chloride concentrations of 70 μgm/in.^{2} on the specimen surface, Incoloy-800 performed satisfactorily up to 2,000 hours of exposure.

Because of the encouraging results obtained to date, Incoloy-800 was chosen as the reference clad material for fuel clad applications in superheat reactor cores under study by GE-APED. It was decided to group the available data into one report, a monograph, to enable fuel technologists to locate the information needed for fuel design or fuel performance analysis. Although detailed information is included, the primary objective of this report is to summarize properties and behavior of Incoloy-800.

a. Availability

Incoloy-800 is available in all forms of wrought products such as tubing, bar stock, flat bars, sheet, pipe, and other standard products. Ingots of the material have been purchased and reduced to wrought products without difficulty. The alloy is described in ASTM Specifications B-407, B-408, and B-409, designated as "Nickel-Iron-Chromium Alloy."

b. Fabricability

Fabricability of Incoloy-800 is comparable to that of Type-304 stainless steel. Welded and drawn tubing has been readily produced in large quantities, and end plug welds for fuel elements have been made without difficulty. Impurities such as sulfur should be excluded from the material or cover atmosphere when welding.

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*ASTM designation: Nickel-Iron-Chromium Alloy. In the absence of a compact, convenient, generic designation, "Incoloy-800" is used throughout this report. "Incoloy," "Inconel," and "Ni-O-Nel" are trademarks of the International Nickel Company, Inc.

c. <u>Long-Term Stability at High Temperatures</u>

The alloy is relatively stable at temperatures of interest to superheat fuel designs. Exposures at 1050° and $1150^{\circ}F$ for periods of over 10,000 hours resulted in reduced ductility, but values lower than 20 to 25 percent elongation were not observed. The principal metallurgical changes detected by electron microscopy were carbide precipitation at grain and twin boundaries. The observed tensile property changes appeared to be related to the combined effect of a strengthened matrix and formation of a continuous intergranular carbide network.

d. <u>Effects of Neutron Flux Upon Tensile Properties</u>

Neutron exposures to 4×10^{20} nvt ($E_n \geq 1$ MeV) at 700° to $800^{\circ}F$ do not affect the room temperature tensile properties of Incoloy-800. When tested at $1100^{\circ}F$, irradiated tensile specimens showed no change in 0.2 percent offset yield strength but a drop in ultimate tensile strength and elongation. Minimum elongation values recorded were 23 percent.

e. <u>Strain Cycle Properties</u>

Fractures after strain cycling (slow frequencies, 30 min./cycle) showed plastic deformations. Irradiation decreased the number of cycles to failure at a given strain range by a factor of approximately 1/3.

f. <u>Electron and Optical Microscopy</u>

Carbide precipitates in various morphologies, which were observed after prolonged exposures at 1050° to $1150^{\circ}F$, have been identified as $Cr_{23}C_6$.

g. <u>Corrosion</u>

Cycle exposure with imposed stress in a loop operating with water and steam, in which 20 ppm oxygen, 2 to 3 ppm H_2, and 0.5 ppm chloride (iron chloride) were added to simulate reactor superheated steam conditions, showed Incoloy-800 to remain intact. Under similar conditions, Types 304, 347, 316, and 304 vacuum-melted grades of stainless steel developed localized cracking within two weeks. The tests with Incoloy-800 were carried out for 63 days without sign of cracking.

Under steady-state corrosion at $1300^{\circ}F$ in ex-reactor oxygenated steam, Incoloy-800 showed a corrosion rate which gave a predicted three-year maximum weight loss

> of 6520 mg/dm^2, or 3.3 mils metal loss (95 percent confidence limits). Under similar conditions, nonlinear extrapolation of the data showed a metal loss of 1 mil per three years.
>
> Cupons under stress and with chloride salts on their surface exposed to 1150°F in steam showed no cracking or localized corrosion attack. "

The results of irradiation tests on Incoloy-800 were reported in AEC Research and Development Report GEAP-5100-9, by D. H. Coplin, et al., (June 1966). The following section is reproduced from this report.

4.2 INCOLOY-800

The influence of neutron irradiation on Incoloy 800 properties varies distinctly with the tensile test temperature. At low test temperatures the normal neutron damage is observed that increases both the yield and ultimate strengths. The usual loss of ductility is also observed. At high test temperatures the yield strength is virtually unchanged to an exposure of 2.2×10^{21} nvt while the ultimate strength decreases with exposure. The elongation, both total and uniform, appears to be increasing slightly at the high exposures. The results are shown in Figures 4-3 and 4-4. The temperature at which the change in damage mechanism occurs is greater than 316°C (600°F), since the data at this intermediate temperature indicate the same changes in properties with irradiation as were observed in the room temperature tests.

Even though these coupons were irradiated at a relatively low temperature 180°C (356°F), the results are similar to those for higher irradiation temperature. Busboom[7] has shown that for irradiation temperatures above 400°C (750°F), the effect of neutron irradiation on the yield strength is very small over a range of tensile testing temperatures from ambient to 705°C (1300°F). For the same irradiation temperature the elongation continually decreases with exposure for the same range of test temperatures. This decrease in ductility has been associated with the change in mode of fracture from transgranular (ductile) to intergranular (brittle). The same phenomena has been reported by Spalaris[8] for Incoloy at exposures of $\sim 4 \times 10^{20}$ nvt at 700 - 800°C and by Pesel[9] for high nickel alloys after irradiation at 649°C (1200°F) to 2.5×10^{20} nvt and 1.2×10^{21} nvt. Hughes and Caley[10] made similar observations of a change in fracture characteristics for austenitic alloys (20 Cr-25 Ni-0.7 Nb and 25 Cr-20 Ni-0.3T) after irradiation at 593°C (1100°F) to 748°C (1380°F) to an exposure of 1×10^{20} nvt.

One postulated mechanism for radiation hardening in Incoloy 800 is the production of He in the lattice as a result of thermal neutron (n, α) reaction with boron. The He molecules coalesce and form relatively large bubbles which block the flow of dislocations, causing the strengthening of the material. At high temperatures, however, the mobility of the helium gas may cause bubble accumulation at the grain boundaries. The weakened grain boundaries would then be expected to fail before the matrix. The helium formation mechanism is not a completely satisfactory explanation. Busboom[7] reported experimental results from tensile coupons irradiated in the experimental Breeder Reactor II that also showed a reduction in elevated temperature ductility. Yet, these coupons were exposed to a predominately fast neutron flux and thus, the helium formation by thermal neutrons with B^{10} should have been sharply reduced. More experimental work is required to define the type of defect structure that is causing the irradiation damage.

As with Zircaloy-2, an effort was made to derive an empirical relation that could be used to predict changes in strength with neutron exposure. The changes in yield strength and in ultimate strength are plotted as a function of integrated neutron flux in Figure 4-7. The slope of the lines drawn through the data indicate the applicability of either $(\phi t)^{1/2}$ or $(\phi t)^{1/3}$ power functions. Based on the ambient temperature results, the change in yield strength can be best approximated by a $(\phi t)^{1/3}$ function. The data was insufficient to obtain a similar plot for the other temperatures. In Figure 4-8 the data for the room temperature changes in yield stress are shown to also agree reasonably well with an exponential type equation. This relation between the change in yield stress and the integrated neutron flux is however, not the same equation that was used for the Zircaloy-2 data. In this case, the exponential term is not raised to the one-half power. There is no theoretical reason for the change; only the fact that the equation seems to provide a better fit for the data.

To summarize, the results from the Incoloy-800 tensile tests are increased yield and ultimate strength with increasing exposure for tests at 316°C (600°F) or lower. At 593°C (1100°F) testing temperature the yield stress remains nearly the same as the unirradiated samples to an exposure of 2.2×10^{21} nvt. The ultimate strength of these same 593 °C (1100 °F) coupons decreased with increasing exposure. This loss of ultimate strength can be related to the change in fracture characteristics of the material. The ductility for all testing temperatures decreased with increasing exposure. The elongations measured at 593 °C (1100 °F) indicate a possible trend of increasing ductility after going through a minimum, but the increase at the highest exposure is not large enough to make a definite conclusion. The change in room temperature yield strength can be represented about equally by either a $(\phi t)^{1/3}$ or the exponential relation, but the latter is preferred because it predicts eventual damage saturation.

TABLE 4-2

INCOLOY-800 TENSILE RESULTS

Coupon Number	Exposure (E(n) > 1 MeV)	Test Temperature (°F)	0.2 Percent Offset Yield Strength (ksi)	Ultimate Strength (ksi)	Total Elongation (%)	Uniform Elongation (%)
L-A	None	75	44.80	87.21	51.31	42.81
L-20	None	75	42.40	86.71	52.44	43.15
L-21	None	75	46.80	85.94	51.82	42.24
L-15	None	75	42.72	86.61	50.61	41.90
L-16	None	75	45.26	86.51	45.92	38.46
L-17	None	1100	26.29	63.59	44.62	38.44
L-18	None	1100	26.23	61.98	40.36	36.19
L-8	0.6×10^{21}	75	86.05	104.59	34.36	25.90
L-14	1.25×10^{21}	75	94.89	109.06	33.25	25.03
L-12	2.4×10^{21}	75	113.14	120.65	27.49	20.60
L-13	2.5×10^{21}	75	114.15	121.03	27.63	21.52
L-1	0.6×10^{21}	600	58.69	84.12	31.05	25.81
L-7	1.25×10^{21}	600	72.93	90.46	28.57	23.35
L-9	1.5×10^{21}	1100	31.45	50.20	7.24	6.52
L-10	1.9×10^{21}	1100	29.74	48.42	8.34	7.84
L-11	2.2×10^{21}	1100	31.43	43.70	9.83	9.00

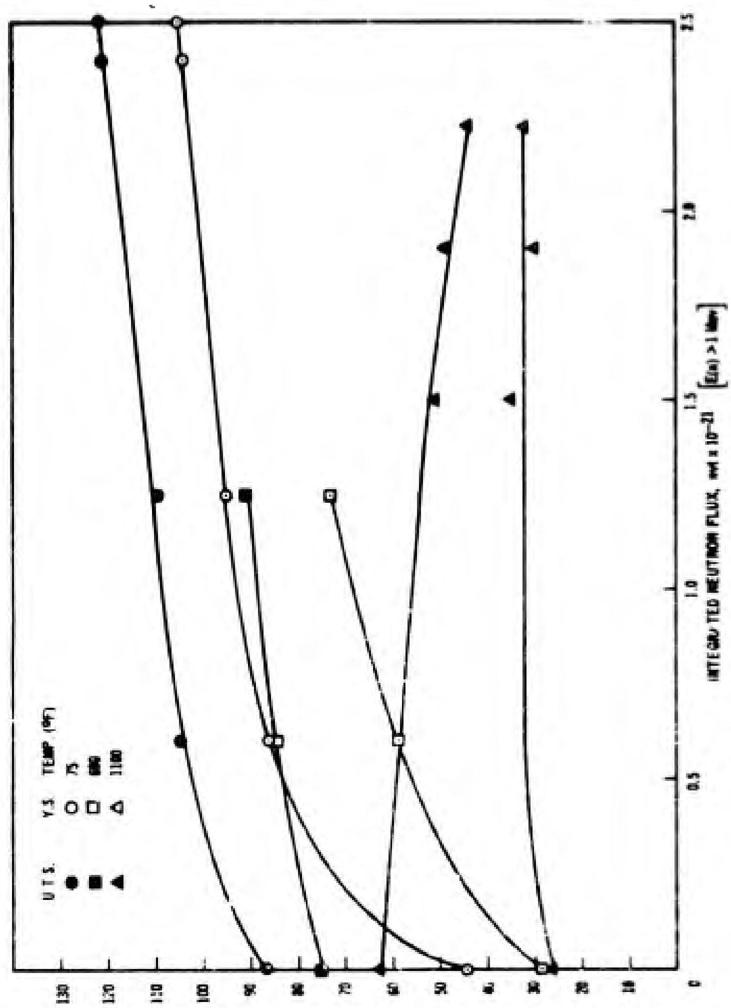

Figure 4-6. Effect of Irradiation on Yield Strength and Ultimate Tensile Strength of Incoloy-800

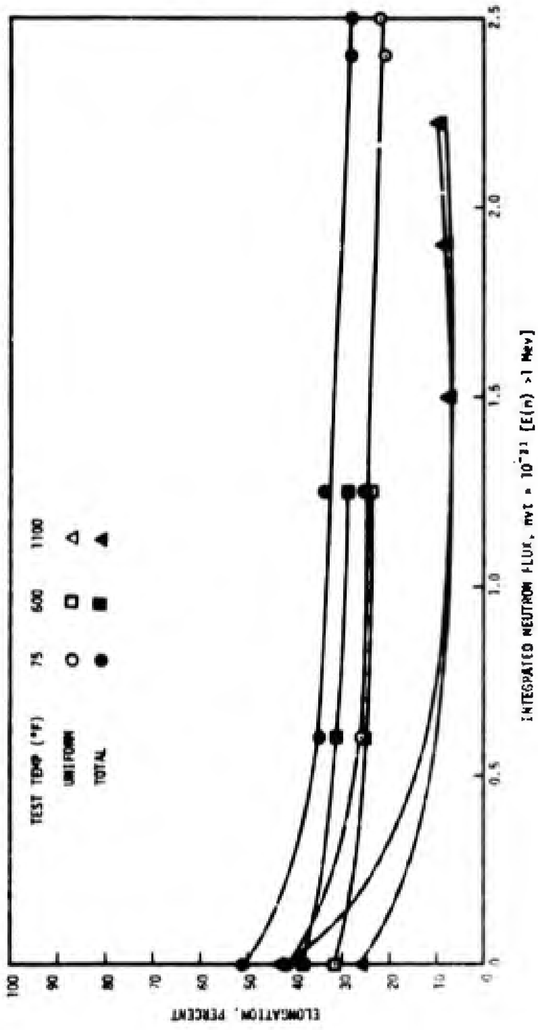

Figure 4-6. Effect of Irradiation on Uniform Elongation and Total Elongation of Incoloy-800

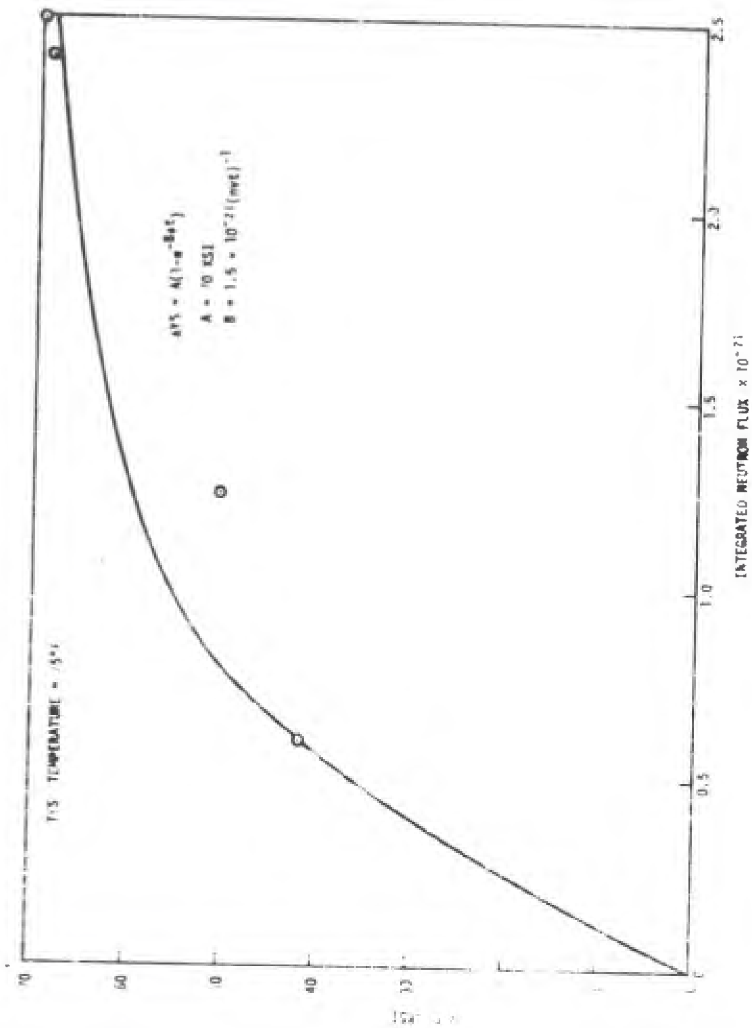

Figure 4-8. Change in Incoloy-800 Yield Strength with Irradiation

A number of utilities have installed Incoloy-800 tubes in parts of the heat exchangers in the steam generator plants. For example, in the Fisk Station of the Commonwealth Edison Company, six tubes of Incoloy-800 were installed, bent cold into "serpentine" configuration, and the tubes were not stress-relieved after bending. The first sample tube was taken out of the superheater after about 15 months' operation with a steam output temperature of 1050°F and outside metal temperature of 1200°F. This sample tube contained a weld joint. The tube was described as being in excellent condition. In the Oak Cree Power Plant of the Wisconsin Electric Power Company, heat exchanger tube sections of Incoloy-800 have been operating with steam temperature of 1000°F, metal temperature of 1300°F, for over 4 years without any sign of trouble. In the Philco Unit 6 station of American Electric Power Company, test lengths of Incoloy-800 have been in service for over two years in 1050°F steam at 4800 psi pressure, and have remained in good condition.

Recently, the ASME Research Committee on High Temperature Steam Generation has investigated the performance of established and candidate superheater tube materials, including Incoloy-800, when exposed to the action of flowing, high pressure steam preheated to 1200°F, 1350°F, and 1500°F. The work was done at the Philip Sporn Station of American Electric Power Service Corporation. The test duration was three years. The Incoloy-800 was found to be suitable for this service at 1200°F and 1300°F.[2]

During the past two years, the corrosion resistance of Incoloy-800 tubes and welded sections in high temperature steam and wet-steam has been examined in autoclave tests carried out at BNWL for Gulf General Atomic. In these tests the water was intentionally contaminated with chloride and oxygen. Under these test conditions the Incoloy-800 tubes and weld sections have shown no sign of stress-corrosion cracking after many months of exposure, whereas, specimens of the 300-series stainless steel failed by stress-corrosion cracking quite rapidly.

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1. Pearl, W. L, et al., Nucl. Appl. 1, 235-45, June 1965.

2. Cordovi, M. A., et al., and F. Eberle, et al., papers presented at the Winter Annual Meeting of the ASME, November 29, 1966.

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13. ABSTRACT

The TRIGA Oceanographic Power Supply (TOPS) is designed to provide 100 kw of electrical power for a Manned Underwater Station (MUS). The design is based on utilization of existing technology both in the reactor and energy conversion unit.

The nuclear steam supply module is based on the widely used TRIGA research reactor. It contains the entire pressurized water primary coolant circuit and the steam generator. The U-ZrH-fueled, water-cooled core is characterized by unique features of inherent safety with load-following capability. The result is a simple system requiring a minimum of attendance by an operator.

The active mechanisms of the steam-Rankine cycle power conversion system utilize only commercially available components. The entire power plant except for the control console is contained in a one-atmosphere environment at a depth of 6000 feet by a pressure hull which is adapted to function also as condenser surface for the thermodynamic cycle.

Design emphasis throughout the study has been placed on reliability, minimum risk, and utilization of off-the-shelf equipment, with plant efficiency taking a subordinate position. The basic TOPS nuclear steam supply module is designed to support alternative power conversion systems for producing up to 500 kw net electrical power.

14. KEY WORDS	LINK A		LINK B		LINK C	
	ROLE	WT	ROLE	WT	ROLE	WT
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