EVALUATION OF A GAS-COOLED FAST BREEDER REACTOR FOR SHIP PROPULSION

by

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ABSTRACT

A gas-cooled fast breeder reactor in a closed-cycle gas turbine power plant is evaluated for merchant ship propulsion. Trends in ship design and propulsion plant requirements, and the status of gas-cooled reactor designs are reviewed to establish the motivation for investigating the proposed system.

A preliminary design of a 200,000 shp unit, including the gas turbine plant, reactor core and safety assessment, was carried out. It is concluded that use of a compact system contained within a prestressed concrete pressure vessel is an attractive approach. The composite type ship consisting of separable pusher and cargo sections was also found to have interesting safety features which commend it to nuclear ship applications.

The major technical problem encountered involved the characteristically short refueling interval of fast breeder reactors. This was solved by development of an internal blanket concept for the core, which was shown to permit a batch core life of approximately 3 years using state-of-the-art burnup calculations.

It is concluded that the proposed system has sufficient merit to warrant further evaluation: detailed economic evaluation in particular, since the present study uncovered no technically unsolvable problems.

Thesis Advisor: M. J. Driscoll

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CHAPTER I

INTRODUCTION AND BACKGROUND

1.1 FOREWORD

Preliminary design and evaluation of a gas-cooled fast breeder reactor were performed in order to determine technical problems and economic incentives in using a direct cycle system for ship propulsion. Trends leading to the development of nuclear powered shipping and the incentives for investigation of gas-cooled reactors for this purpose are discussed.

1.2 BACKGROUND

At the end of World War II, the U.S. Merchant fleet was the largest in the world and carried 57% of U.S. foreign commerce. Today this same merchant fleet is fifth in the world and only carries 5% of the U.S. foreign trade (Ul). Over three-fourths of these remaining ships are over 20 years old, which is the usual ship design-lifetime. The fleet as a whole is therefore approaching block obsolescence, which will necessitate a new shipbuilding program if further deterioration is to be averted. The last major shipbuilding program in this country occurred during World War II. Since that time there has been very little growth in the industry.

Adding to the decline has been the increased labor expenses associated with unionization since World War II. The American shipowner now pays the highest wages to the largest crews. In 1957, wages accounted for 15% of the total daily fixed costs of all ships except for passenger ships and cargo-passenger ships where the percentage was almost double (C1). Since that time crew wages have advanced far more rapidly than other costs and are a much larger percentage of the annual cost, as shown in Table 1.1. Very few American ships are automated and most, because of age, are not as well equipped as their foreign counterparts. The merchant marine does not attract the experienced personnel needed to operate highly sophisticated systems. Therefore, ship designers must keep propulsion units simple.

Realizing that if the continuing decline is not reversed the U.S. fleet will carry less than 3% of the U.S. trade tonnage in 1980, Congress and the executive branch enacted the Merchant Marine Act of 1970 (U1). This law was designed to revitalize a sagging merchant marine and achieve goals defined in accordance with the 1936 Merchant Marine Act (U1). This Act set these requirements for the merchant marine: 1) to carry domestic trade; 2) to carry a substantial portion of U.S. and foreign water borne commerce on all essential routes; 3) to act as a naval and military auxiliary in time of war;

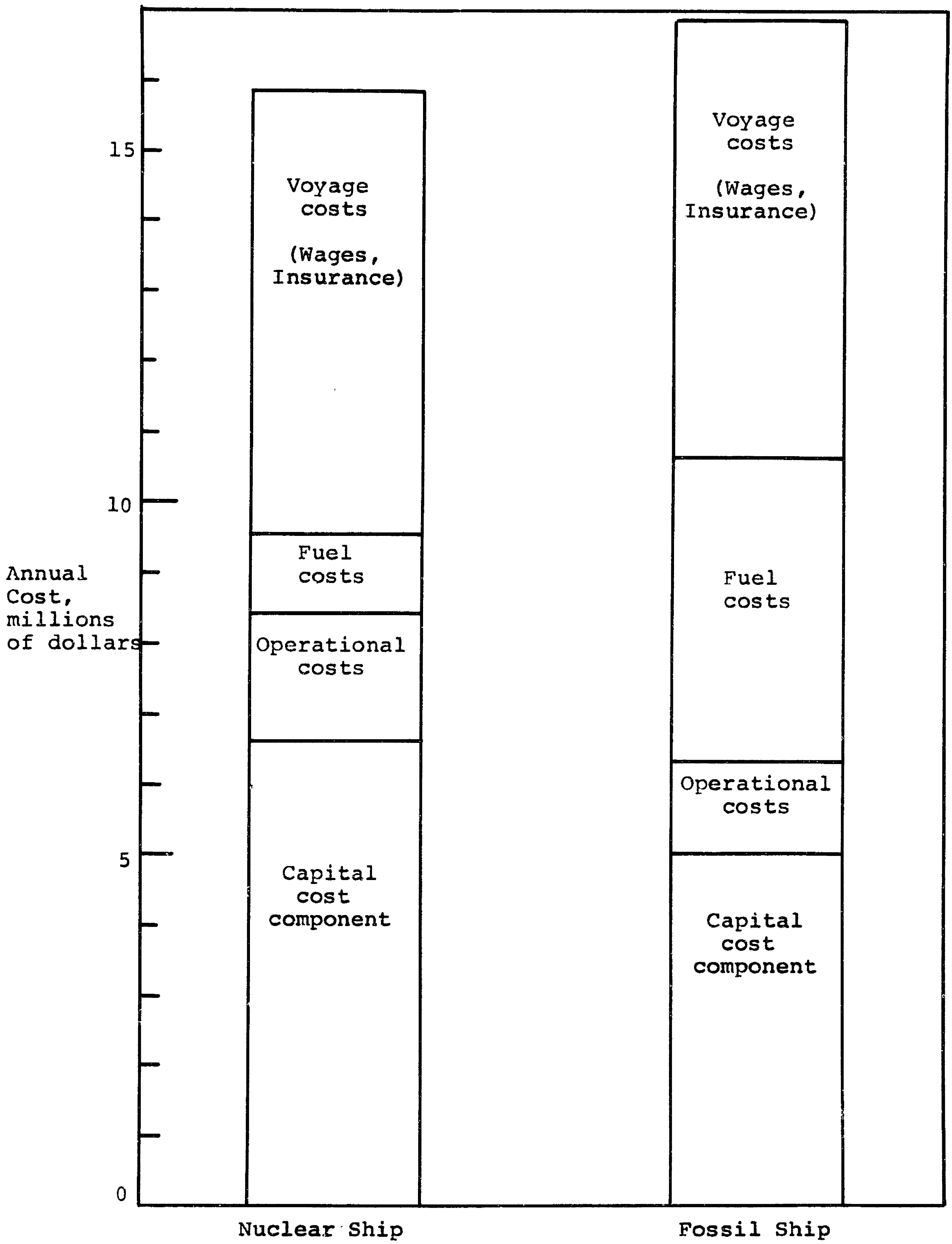


Table 1.1 AVERAGE ANNUAL COST - NUCLEAR VS. FOSSIL SHIP.

Ref. E2

and furthermore 4) all ships are to be owned and operated under the U.S. flag by American citizens; 5) the fleet is to be composed of the best, safest and most suitable vessels. The 1970 Merchant Marine Act will help the U.S. fleet achieve these requirements by providing for the construction of 300 new ships over the next several years.

Design studies for ships that would comprise a new American fleet demonstrate a desire for larger, faster ships with shorter loading and unloading times (K1, K2, P1, R1). A desire to automate engine rooms can be inferred from the greater number of new ships with bridge controls and the movement toward increasingly automatic equipment. The larger ships would gain by economies of scale in building, and also in operation, since the size of the ship's crew remains practically the same even for substantially larger ships (R1). Faster ships will be able to compete for high value, quick transport cargo. Faster turnaround times mean greater ship utilization and a higher rate of return. In brief, advanced technology must be used to the fullest if the goals of the 1970 Act are to be realized.

1.3 CURRENT DEVELOPMENTS

1.3.1 World Trade and Ship Size

As world trade rapidly increases as a result of continued population growth and industrialization of under-developed countries, larger maritime fleets will be necessary. As shown in Figure 1.1, worldwide dry cargo trade will be 2,750 million tons per year in the year 2,000, corresponding to a rate of increase of approximately 6.85% a year (P1). To meet this increase in world trade, merchant ships have increased in size, speed and number in recent years. Massive tankers and bulk carriers have been built in increasing numbers since the Suez Canal closing in 1967, and cargo ship speeds have increased by as much as 50% in the last 5 years (P1).

There is a speed-duration limit for oil-powered ships brought about by the "cubed law" phenomenon of marine propulsion plants. In other words, to double the speed we have to increase power by approximately a factor of eight. One effect is to increase the weight and volume of fuel oil that must be carried. The empirical equation for estimating power is:

$$\text{Shaft horsepower} = \frac{D^{2/3}V^3}{C}$$

where

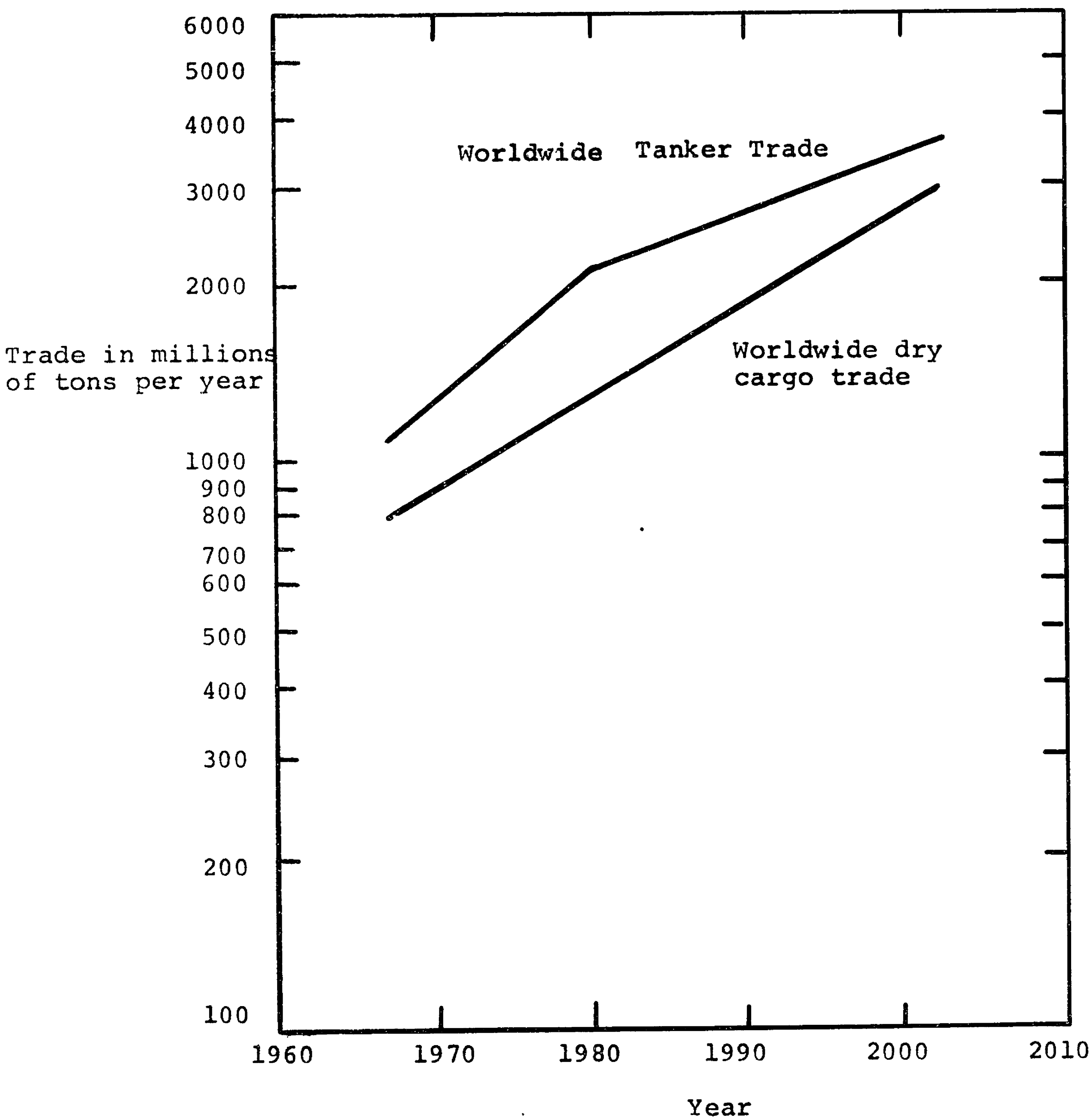
D = displacement in tons

V = speed in knots

C - "Admiralty Coefficient" a non-linear constant which gets smaller with increasing speed.

Ref. C1

Fig. 1.1 WORLDWIDE TRADE BY SEA



Ref. Pl

An example of this effect can be simply demonstrated. The "America" requires 5,000-shp to travel at a speed of 12.5 knots. But to travel at a speed of 25 knots she would require more than 50,000-shp. See Fig. 1.2.

Increases in speed and power plant rating have been generally paralleled by corresponding increases in ship deadweight. See Fig. 3 for increases in bulk tonnage. Figure 1.4 shows the trend in containership power plant rating. Not shown on Fig. 4 are the recently ordered 120,000-shp containerships for the Sea-Land Corporation. (N1) As power increases, fuel cycle costs become an increasingly larger part of the ship life-cycle costs. Fuel oil also becomes an increasingly larger percentage of the total ship deadweight and volume. Figure 1.5 shows the effects of speed and distance on cargo carrying capacity of an oil-fired ship. As distance increases, more fuel must be carried, decreasing cargo space. The higher speed end of the curve dramatically shows the adverse effect on annual cargo capacity of the additional fuel needed. Since speeds of greater than 30 knots are proposed for container ships, a reassessment of ship propulsion systems is clearly in order (N1).

An additional development motivating shipowners toward nuclear fuel is the rising cost of fuel oil used in conventionally fueled marine propulsion plants. Nuclear Industry reported that bunker C fuel oil for

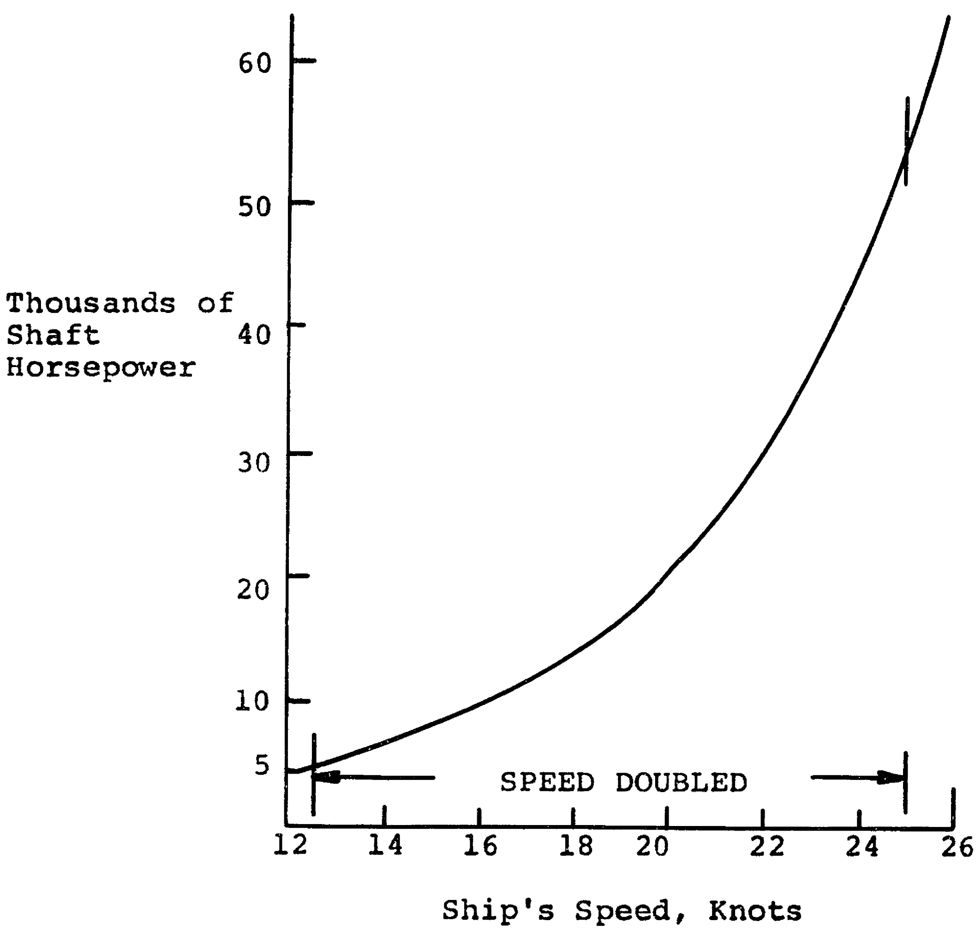
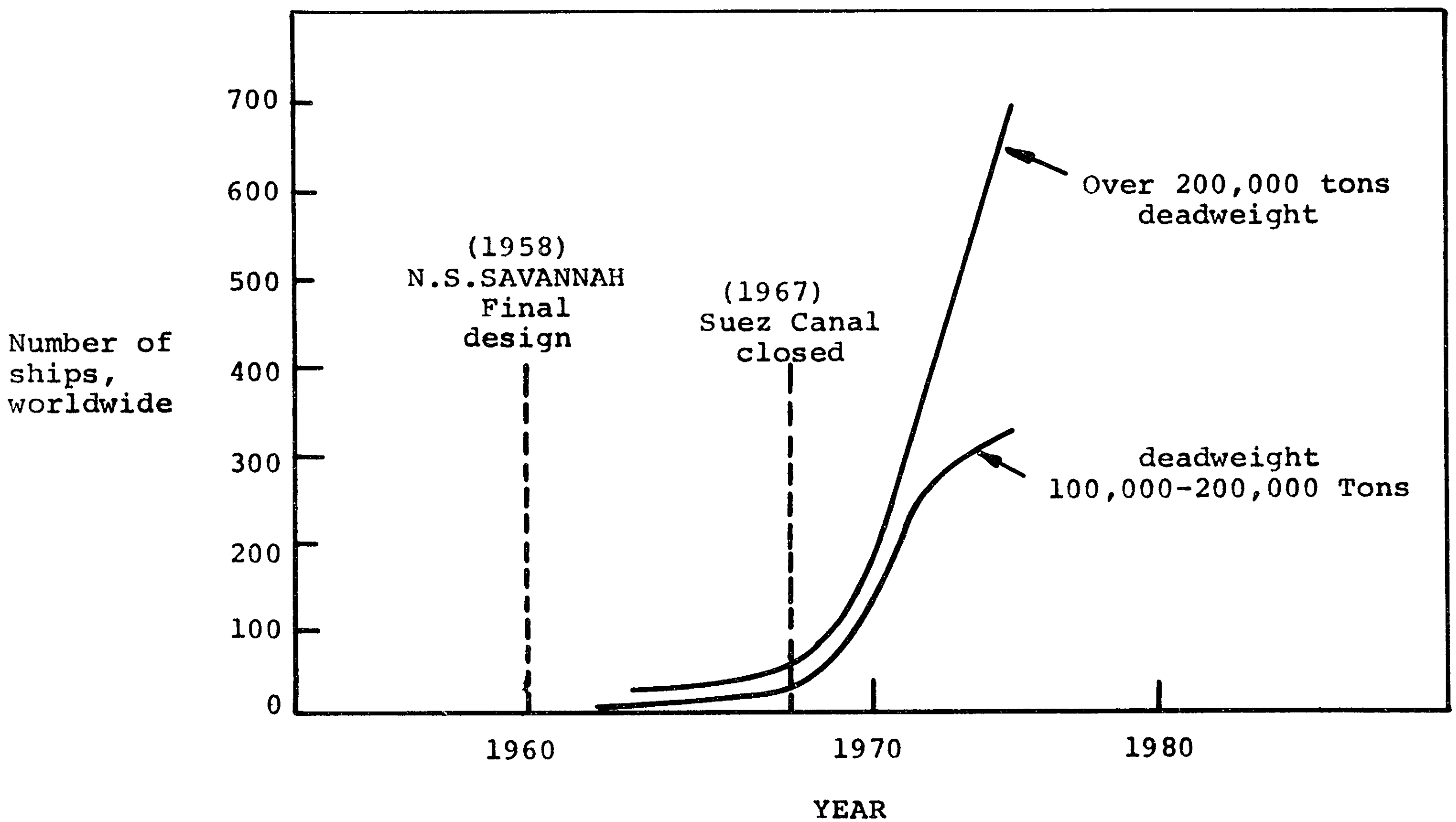


Fig. 1.2 SPEED-POWER RELATIONSHIP

TYPICAL PASSENGER LINER

Ref. Cl

Fig. 1.3 RECENT TREND IN BULK TONNAGE



Ref. P1

Fig. 1.4 TRENDS IN CONTAINERSHIP SIZE

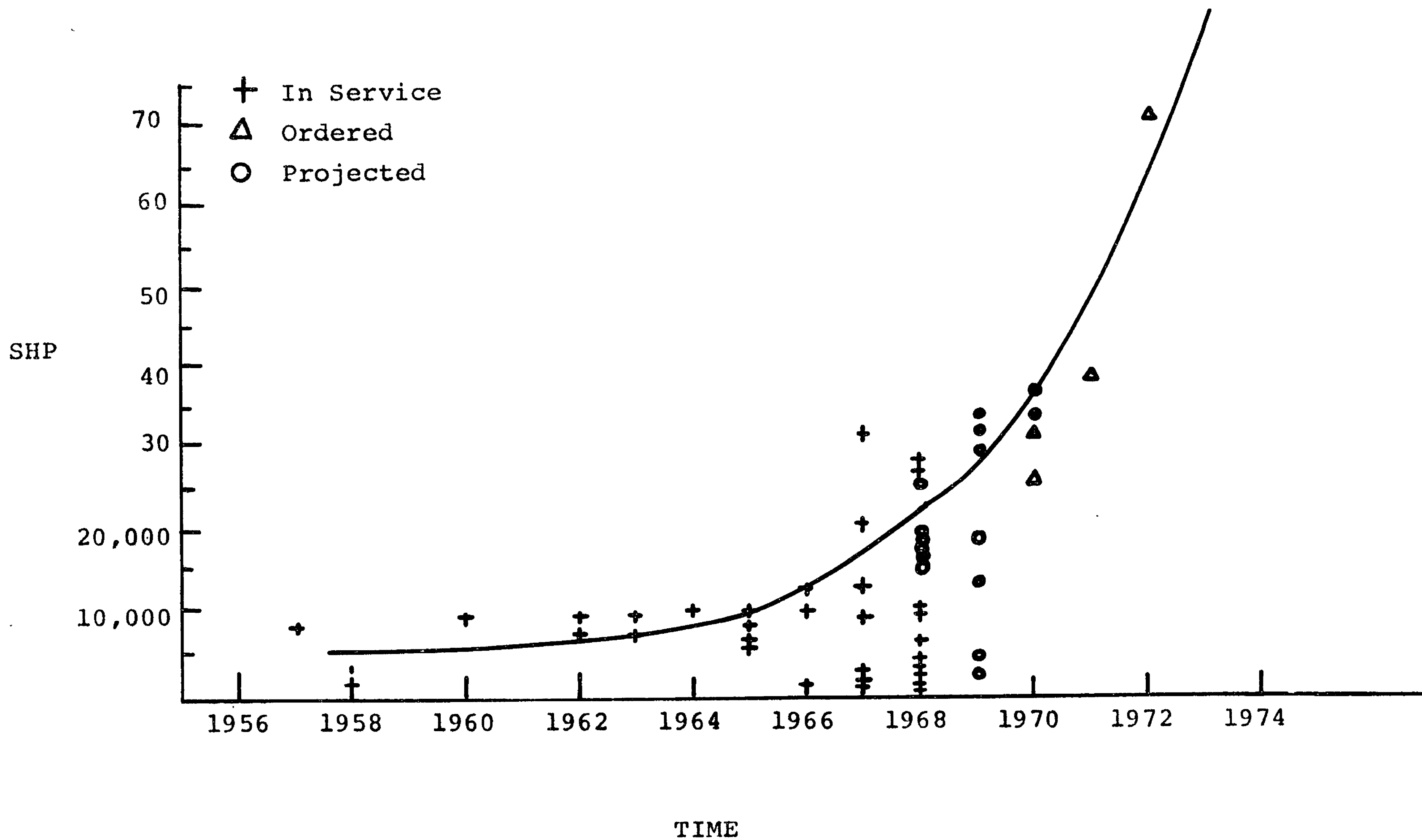


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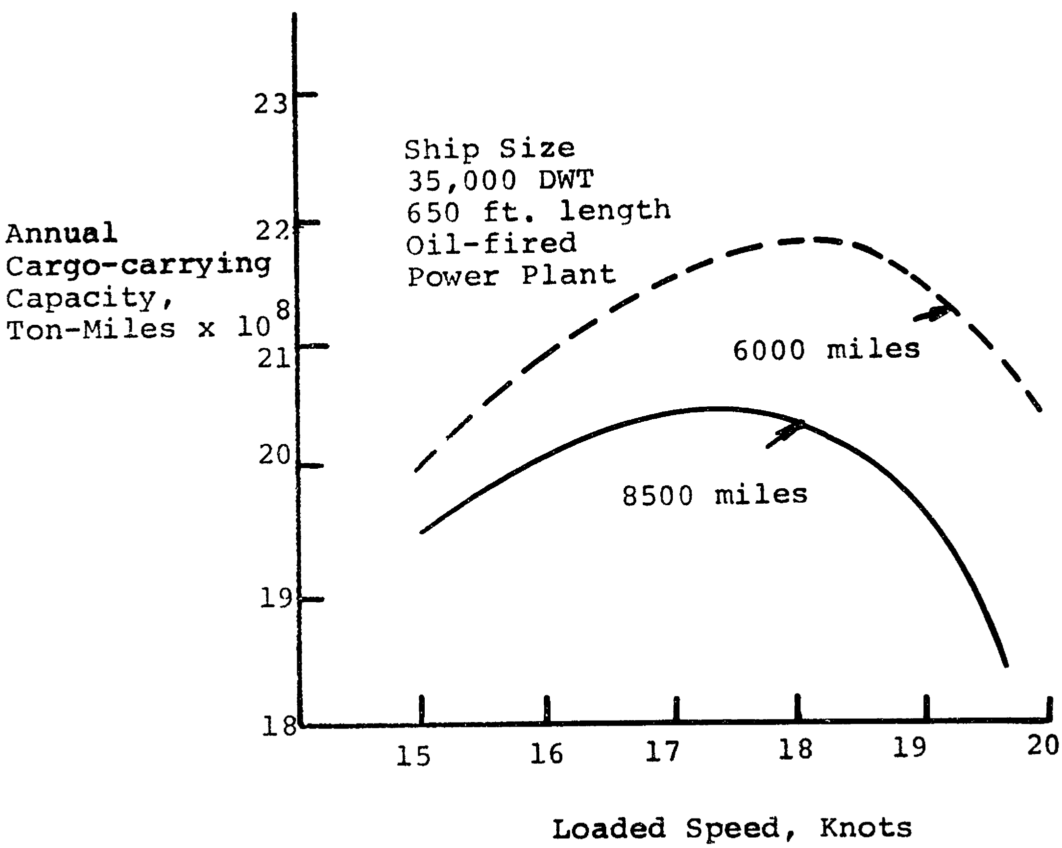
Fig. 1.5 SPEED-DISTANCE INFLUENCE ON BULK CARGO CARRYING CAPACITY



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ships has doubled in price, up to \$3.50 a barrel, during the past year and one-half, and Maritime Administrator Andrew Gibson expects the price to rise to \$4.50 a barrel in the next few years (N1). It is quite pertinent to note that Babcock and Wilcox Co. claim at \$4.00 per barrel for fuel oil, nuclear power would be competitive in the 100,000-shp range (E2).

A worldwide fleet of 2,500 newly constructed merchant ships is predicted by the Maritime Administration in the next two decades (P3). If we assume 10% of that market (250 ships) are American it can be estimated that 5 billion dollars worth of propulsion machinery and 1/2 billion dollars of fuel per year will have to be purchased. If American ships aren't built and used to carry this part of the world trade, the U.S. balance of payments will be unfavorably affected, both in terms of shipyard revenue and freight charges.

1.3.2 Developments in Ship Design

Recent developments in the transportation of goods have led to some radically different types of ships, both presently operating and proposed. Some of these concepts are particularly well suited to a nuclear powered propulsion system.

To meet the demands for a large new shipbuilding program the Maritime Administration in 1969 asked for proposals on new ships to be built for the 1970's. These designs, called the CMX designs, were solicited from two vendor teams, one based around Newport News Shipbuilding and Dry Dock Company, and the other about Bath Iron Works Corporation. The ship types, size and horsepower are listed in Table 1.2 along with the proposed Vicker's containership and the proposed General Dynamics Containership. For comparison the same parameters on the N.S. Savannah are also listed.

The large containership has been studied (G1, M2) on several occasions, as its characteristics are most suited to the use of nuclear power. In addition, the lighter aboard ship, LASH, concept because of its large size and high speed may be a viable contender for a nuclear engine. One of the most interesting designs for a ship is the "Composite Ship" proposed for a nuclear power plant by J. A. Teasdale. (T1) This design proposes a separable ship with a propulsion unit which is completely independent but connectable to a cargo carrying section. This concept will be discussed in conjunction with the nuclear propulsion system later proposed.

1.3.3 Summary

Increasing world trade and population growth have created a demand for the rapid transport of large volumes of goods. To meet this demand, ship size and

TABLE 1.2 PRESENT SHIP PARTICULARS

COMPANY	CLASS	TYPE	POWER (shp.)	LENGTH (ft.)	BREADTH (ft.)	DRAFT (ft.)	DISPLACEMENT (tons)	SPEED (knots)
Bath Iron Works (KI)	Penobscot	Container	40,000	764	102	31.5	36,860	23.4
	SACO	Ore/Bulk/ Oil (OBO)	24,000	821	105.5	42.66	88,400	17.0
	Kennebec	Multi-purpose	15,000	570	75	32.33	29,300	16.0
	Machias	Tanker	24,000	821	105.5	47	91,200	16.4
	Merrimad Allagash	Container/ Breakbulk	24,000	592	86	31.5	26,400	21.2
Newport News (KI)	Crescent	OBO	24,000	871	106	42.5	90,600	17.1
	Vanguard	Container	35,000	725	103	29.5	32,200	23.3
	Voyager	Tanker	24,000	950	147.5	48.5	147,400	15.6
	Nomad	Multi	12,500	584	87	33.5	35,600	16.6
Marad Bench-mark Designs (KI)	PD 158	OBO	18,000	787	105	40.66	83,500	16.0
	PD 159	Gen'l	12,000	504	74	28.25	21,050	18.9
	PD 160	Container	40,000	758	101	28.5	32,500	24.1
	PD 161	Container	80,000 (2 screws)	941	105	32	44,650	26.7
	PD 162	Barge (LASH)	40,000	872	105	28	37,100	23.5
General Dynamics (K2)	Non Nuclear	Container	116,000	871	104	34.9	46,962	30.3
Vickers (G1)	Nuclear	Container	40,000					
General Dynamics (M2)	Nuclear	Container	100,000	871	104	30.7	39,355	30.5
NS Savannah	Nuclear	Pass/Cargo	20,000	595	78	29.6	21,840	21

References: (K1), (M2), (G1).

speed have increased and cargo handling time has decreased.

These changes have been reflected in the use of newly designed ships where greater emphasis has been placed on the propulsion unit. Greater ship utilization has increased fuel and operating costs, directing attention to a reassessment of ship energy sources. Concurrent with the increase in fuel consumption is a rise in fuel oil price, increasing the proportion of total cost attributed to fuel. A highly utilized nuclear power plant displays low fuel cycle costs. These factors indicate a need to reevaluate nuclear power for shipboard use.

1.4 POTENTIAL ADVANTAGES OF NUCLEAR POWER

The potential advantages of nuclear power for merchant ship propulsion were recognized in the early 1950's and reported by J. J. McMullen Co. for the President's Panel on the Peaceful Uses of Atomic Energy. (P3) Since then the only commercial nuclear ship built in the U.S. has been the N.S. Savannah. Since the Savannah made her maiden voyage in 1962, there have been many nuclear ship designs proposed but no additional American nuclear ships have been built. The reason for this delay is not technical, but economic.

The economics of nuclear power are very sensitive to a large number of cost uncertainties and assumptions, particularly in the case of a new system, involving both capital and operating costs. The present revived interest in nuclear power plants for ships is a result of changes in both ship design and economic climate.

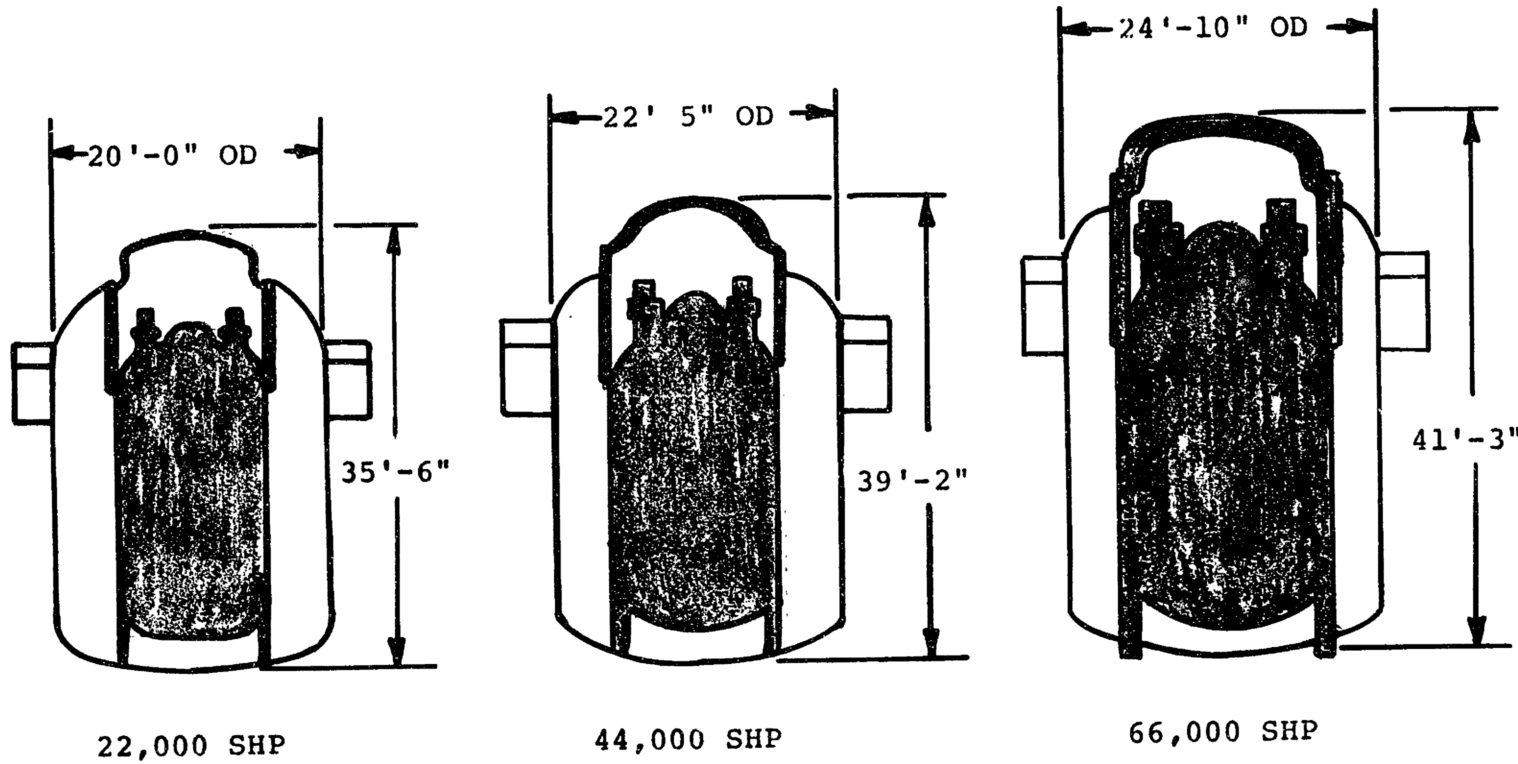
Changes in ship design in favor of nuclear fuel are: 1) high speed with corresponding high power; 2) larger size with larger power; 3) fast turn around leading to increased utilization; 4) economy of scale in nuclear plant capital cost.

Changes in the economic climate which are incentives for the use of nuclear power are: 1) rising fuel costs for oil; 2) greater certainty of nuclear costs; 3) increase in world trade and demand for ships.

As speed and power increase, the amount of fuel oil consumed also increases. Modern marine steam plants consume between 0.48 and 0.52 pounds of fuel oil per shaft horsepower hour. To illustrate the effect: a 10,000-shp ship will burn 26 tons of fuel oil per hour, and a 100,000-shp ship will consume 260 tons of fuel oil per hour. The space and weight occupied by the fuel oil displaces valuable cargo space. Nuclear power plants, on the other hand, do not change in size considerably with increasing shaft horsepower. Fig. 1.6 is an example of the increase in size of a nuclear power plant with increasing power.

Larger size ships with longer voyages mean greater ship utilization and larger power plants, with the already-discussed fuel problem. The larger ship is more adaptable

Fig. 1.6 CNSG III COMPARATIVE SIZES



Ref. S2

to the concentrated weight of a nuclear propulsion plant.

The new transportation concepts with fast turn-around were devised to get maximum utilization of the ship. The rapid loading and unloading increases the total amount of time at full power and correspondingly increases the power plant load factor. Since nuclear power plants have high capital costs and low fuel cycle costs, systems that increase plant usage are favorable to nuclear power.

Nuclear capital costs do not increase linearly with size: for example, there is a scale factor of roughly 0.7 in shore-side plants. Dollars per kilowatt are proportional to the total kilowatts of the plant raised to the 0.7 power. Thus, the larger the power plant system, the lower the plant cost per megawatt. Fossil fuel costs, which will tend to be the dominant cost item in fast ships on long trade routes, do not show this economy of scale.

The energy-resource economic climate is also rapidly changing due to environmental and conservation-related constraints. Increased awareness by oil producing countries, and increased demand by consuming countries has forced the price of oil up. This increased price of fuel oil, and the potential or actual demand for low sulfur content, is causing higher ship operating costs; and uncertainties in fuel oil supply and price are causing shipowners to reassess ship propulsion systems.

As oil fuel prices become less certain, nuclear fuel prices are becoming more certain. A large shore-based industry is providing sound data on the economics of nuclear power. Fuel cycle costs can be fairly accurately determined for marine plants, though capital cost will be more uncertain than those for the larger land-based stations until some nuclear ship construction experience is forthcoming.

An increase in world trade and commerce is conducive to new shipbuilding programs. Advantage can be taken of the advanced nuclear power technology to help increase the American portion of this trade.

Although the U.S. has not built another nuclear ship since the Savannah, two new foreign nuclear ships were built and numerous designs have been proposed.

1.5 PRESENT DEVELOPMENTS IN NUCLEAR PROPULSION

The N.S. "Lenin" was the first non-naval nuclear powered ship. The Lenin was specifically designed and built as an icebreaker to assist in keeping Russia's northern ports open during the winter. The endurance provided by her nuclear power plants (three reactors) permits the Lenin to operate for extended periods of time away from port. The icebreaker application is unique in many respects and does not really come under the merchant ship heading.

The first nuclear merchantman was the N.S. "Savannah", a combination passenger-cargo ship of 21,800 tons displacement. The Savannah was small, and therefore uneconomical from a purely commercial viewpoint. She accomplished her design purposes however, by opening ports for nuclear ships, and as a training and experimental facility for nuclear propulsion. In essence, the Savannah proved the technical feasibility of nuclear power for maritime propulsion. The particulars of the N.S. Savannah are presented in Table 1.3. One of the important factors to note is the containment weight (including the reactor). This amounts to approximately 260 lbs./shp. New reactor designs are claimed to weigh as little as 36.6 lb./shp. without the added scantlings required for a nuclear ship. (P2)

The only two merchant ships that have been built with nuclear propulsion since the Savannah are the N.S. Otto Hahn and the N.S. Mutsu. The N.S. Otto Hahn is a West German ore carrier with an integral PWR. The Otto Hahn was launched in 1964, and is used primarily as a research ship. There are extensive laboratories and experimental facilities which are not found on conventional ore carriers (P3). The reactor is very similar to the Consolidated Nuclear Steam Generator (CNSG) proposed by Babcock and Wilcox. (S2) The ship and reactor

TABLE 1.3

N.S. SAVANNAH PARTICULARS

Length Overall	595 Ft.
Breadth	78 Ft.
Draft	29 Ft. 6 Inches
Displacement	21,840 Tons
Deadweight	9,300
Cargo Holds	7
Watertight Compartments	11
Machinery Output	20,000 shp
Auxiliary Power	750 hp
Speed	21 Knots
Electric Power	1,500 kwh
Officers and Crew	109
Research Personnel and Passengers	60

Reactor

Max. Thermal Output	74 MW

Core:

Diameter	62 Inches
Height	66 Inches
No. of Fuel Elements	32
No. of Control Rods	21

TABLE 1.3 (continued)

Fuel enriched UO_2 - 4.4 wgt. % U-235

Primary Coolant:	
Core Inlet Temperature	493.6°F.
Core Outlet Temperature	519.7°F.
Primary Coolant Flow	8.6×10^6 lb./hr.
Primary Coolant Pressure	1,750 psi
Steam Generators:	
Feedwater Rate	Max 2.6585×10^5 lb./hr.
Feedwater Temperature	345°F.
Operating Pressure	455 psig@ max pwr. to 715 psig @ 0 power
Superheater Outlet Temperature	508°F
Pressure Vessel:	
Diameter (I.D.)	98 Inches
Height	27 Feet
Walls	6.5 Inches
Containment:	
Length	50.5 Feet
Diameter	35 Feet
Weight including Reactor, Machinery	3,615 Tons

specifications are given in Table 1.4. As the Otto Hahn is classed as a research ship she will provide little definitive information on the true economics of nuclear power.

The N.S. Mutsu being constructed and operated by the Japanese is also classed as an experimental ship. The Mutsu will have a Mitsubishi "dispersed" or loop type PWR where the pumps and steam generators are outside the reactor vessel. The particulars of the Mutsu are listed in Table 1.5, together with the specifications on the Mitsubishi dispersed reactor.

1.6 REACTOR TYPES

A number of different reactor types have been proposed for ship propulsion. In this abbreviated discussion the Pressurized Water Reactor will be discussed as a prelude to discussion of the breeder reactor concepts.

1.6.1 Light Water Reactors

Over the last two decades an extremely large amount of research has been done in light water reactor technology. Central station power plant core design has progressed rapidly. Linear power densities in the range of 18 kw/ft have been attained. This may be contrasted to the Savannah, which had a linear heat rating of approximately 2.42 kw/ft. The smaller cores

TABLE 1.4

"OTTO HAHN" PARTICULARS

Length Overall	564 ft.
Breadth	76.7 ft.
Draft	30.2 ft.
Displacement	26,200 tons
Dead Weight	15,000 tons
Cargo Holds	6
Watertight Compartments	13
Machinery Output	10,000 shp
Auxiliary Power	2,000 shp
Speed	15.75 knots
Electric Power	1,590 kw
Officers & Crew	66
Research Personnel & Passengers	47

Reactor Specifications

Max. Thermal Output	38 MW
Core:	
Diameter (Effective)	3.78 ft.
Height (Effective)	3.67 ft.
No. of Fuel Elements	16.
No. of Control Rods	12.

TABLE 1.4 (continued)

Fuel UO_2 - Avg. Enrichment	3.6 per cent 0-235
Primary Coolant:	
Core Inlet Temperature	511° F.
Core Outlet Temperature	532° F.
Primary Coolant Flow	5.2×10^8 lb./hr.
Primary Coolant Pressure	925 lb./in.^{2}
Steam Generators:	
Feed Water Rate	141,000 lb./hr.
Feed Water Temperature	365° F.
Operating Pressure	455 lb./in.^{2}
Superheater Outlet Temperature	523° F.
Pressure Vessel:	
Diameter	7.75 ft.
Height	28.20 ft.
Walls	2.17 in. thick
Containment:	
Height	43 ft.
Diameter	31 ft.
Weight Including Reactor	930 tons

TABLE 1.5

N.S. MUTSU PARTICULARS

Length Overall	426 ft.
Breadth	62.2 ft.
Draft	43.3 ft.
Displacement	8,350 tons
Deadweight	2,400 tons
Machinery Output	10,000 shp
Speed	16.5 knots
Electric Power	2,000 kw
Officers and Crew	59
Research Personnel	20

Mitsubishi PWR

Max. Thermal Output	36 MW
Core Life	9,000 hours @ 100% power

Core:

Diameter	3.75 feet
Height	3.35 feet
No. of Fuel Elements	32.
No. of Control Rods	12.
Fuel: Enriched UO_2	

TABLE 1.5 (continued)

Primary Coolant:	
Core Inlet Temperature	520° F.
Core Outlet Temperature	545° F.
Primary Coolant Flow	3.9×10^8 lb./hr.
Primary Coolant Pressure	1,600 lb./in.^{2}
Steam Generators:	
Feed Water Temperature	484° F.
Steam Temperature	532° F.
Steam Pressure	312 lb./in.^{2}
Steam Generation Rate	135,000 lb./hr.

are achieved by going to higher power density and permit smaller reactor vessel and shield sizes, and less power plant volume and weight.

The leading reactor type is obviously the Pressurized Water Reactor (PWR) due to its long-established usage for submarines and other naval craft. The PWR was the choice for the first three merchant ships and the icebreaker Lenin. The major advantage is the well developed technology as a result of naval research and land based applications.

Because of the economics of scale in fuel fabrication and reprocessing, all commercial maritime nuclear power use is in a very real sense symbiotically dependent on the land based reactor industry. A rough estimate indicates the equivalent of 40 x 10^3 Mwe total propulsion plant power for the shipping industry in the entire world in 1980 while land based central station power generation industry is estimated to involve some 2,000 x 10^3 Mwe in 1980. (E1) Thus, assuming equal penetration by nuclear in the land and maritime markets, the maritime nuclear system will only be 1/50th of the size of the land based system. For the foreseeable future, even this is optimistic since nuclear systems have already gained wide acceptance ashore, and are projected to constitute roughly half of the installed capacity by the end of the century.

This example serves to demonstrate the dependence of marine power plants on shore power plants not only for design developments but also for operating costs. Lower fuel cycle costs are possible with increasing fuel throughput in a reprocessing plant. Therefore deviation from land based fuel element design could be very costly. It is not surprising, therefore, that presently proposed shipboard reactor designs take advantage of the PWR land based technology, and specifically the gains made in fuel cycle economics. Because of this close link between sea and land nuclear systems one must consider future trends projected for the central station nuclear economy if one is to correctly assess the factors which would be brought to bear on shipboard nuclear prospects.

1.6.2 Fast Breeder Reactors

Economic indications show the fast breeder reactor supplanting the light water reactor due to its better fuel cycle costs. (E1) The fuel cycle cost quoted by B & W for its Consolidated Nuclear Steam Generator (CNSG IV) is 1.5 mills/shp-hr. (E2) The projected fast breeder fuel cycle cost is about half of this, or 0.8 mills/shp-hr. These numbers are, as yet, estimates, but if they are correct within a sufficiently small error range, then low fuel cycle costs are a major reason to go to the fast breeder reactor. Other long term economic incentives for selecting a fast breeder over the light water reactor are the improved resource economy and protection against rising Uranium costs. (E1)

The three major design concepts which have been proposed for the fast breeder reactor, differ in the means of cooling the fuel elements. The Liquid Metal Fast Breeder Reactor (LMFBR) is the concept most favored by the U.S. Atomic Energy Commission and all foreign countries active in FBR research and development. This concept uses liquid sodium at low pressure to cool the fuel. The second concept is the gas-cooled fast breeder reactor proposed, in the U.S., by Gulf General Atomic Corporation. Helium is circulated at high pressure to cool essentially the same general type of fuel element as is being designed for the LMFBR. The third concept is the Steam Cooled breeder reactor which employs steam as the primary coolant. This is the least promising concept as the neutronics and economics are unfavorable, and it is no longer being actively pursued.

The LMFBR is apparently the only FBR that has been previously evaluated for ship use. (M1) It is appropriate therefore to begin a more specific discussion by considering its key attributes.

Liquid metals have very good heat transfer properties, which lead to small compact cores. This also permits good emergency and post-accident cooling. In particular, natural convection shutdown cooling can easily be designed into a sodium-cooled system. There are major advantages for a ship propulsion unit. The other major advantage of the liquid metal cooled breeder reactor is the technology, which is the most advanced relative to the other FBR concepts. With ship nuclear units highly dependent on shore power plants for their economic base, this is an important consideration. There are problems, however, associated with the use of liquid metals for merchant ship reactor coolants. Naval experience with the sodium-cooled reactor on the Sea Wolf was apparently not entirely satisfactory, since the power plant was replaced by a PWR. (P2) One of the major projected disadvantages is the high weight per shaft horsepower associated with the LMFBR. (M1) A weight of 129 lbs./shp is quoted in Ref. (M1) for a container ship power plant, and even this could be optimistic. Table 1.6 lists some representative propulsion plant weight estimates. In addition to the high weight per shp the liquid metal cooled plant faces the inherent problems associated with liquid metals. Sodium is chemically reactive with both air and water. The heat transfer system design must be such, that it prevents radioactive sodium from contacting water in the event of a leak. This necessitates an intermediate sodium loop with its added weight, cost and some degradation in maximum attainable cycle

TABLE 1.6

REPRESENTATIVE PROPULSION PLANT WEIGHTS

Plant and Type	Specific Weight (lb/shp)
Fossil plant without Fueloil (100,000 shp)	55
Fossil plant with Fueloil 10 day voyage	165
N.S. Savannah (PWR)	259.5
UNIMOD Loop type PWR	43.0
CNSG 20,000 shp Integral type	67.2
630A Mark IV gas-cooled	36.32
Otto Hahn integral PWR	186

Ref. A2

temperature with an attendent loss in efficiency. Sodium reacts violently with water and air, and safety precautions must be taken to prevent contact of these ubiquitous materials with sodium: a virtual impossibility to guarantee in a severe collision.

Only a modest amount of work, all now terminated, has been done in developing the steam-cooled breeder reactor (SCBR). (W2, M4, S3) There was a considerable amount of research and development done in Germany to define systems and problems in the SCBR. Many of the SCBR problems would be accentuated if placed in a ship or mobile environment. The direct cycle system is very attractive from an economic standpoint but more difficult to accomodate from a safety and radionuclide release viewpoint. There are difficult seal points at the turbine shaft, for example. Steam, as with a BWR, would be radioactive and may require a large amount of turbine and condenser shielding, with the added weight penalty. Other problems of note are emergency core cooling and water flooding. Some of these problems are shared in common with the Gas Cooled Fast Breeder Reactor (GCFBR) so that some of the results of the SCBR development program are pertinent to the GCFBR.

The steam environment of the SCBR is poor from the reactor physics standpoint. Steam degrades the neutron energy spectrum so that the breeding ratio declines to the marginal value of approximately 1.15 (W1) The low breeding ratio leads to a long doubling time, on the order of 37 years. This is not an economically attractive feature. The high pressure of between 1250 and 3700-psia required for good heat transfer demands a heavy vessel, another detrimental feature. Finally, the high pressure and corrosive environment make the already difficult FBR fuel design problem almost insurmountable.

Gas coolants have been used in thermal reactors and have been proposed for fast reactors. There are a large number of gases that remain inert both chemically and neutronically in a fast reactor environment. Gases are visually transparent, which will assist in both maintenance and refueling. From the viewpoint of neutron economy, gas is very favorable; even the complete loss of gas will add only a small amount of reactivity. Proposed gas-cooled reactors show breeding ratios in the 1000 Mwe design studies, on the order of 1.48. There are many different thermodynamic cycles that can be used with a gas coolant as the working fluid. Some of these cycles would assist in lowering the capital cost of a gas-cooled plant.

Table 1.7 shows a comparison of some of the advantages of the alternative FBR coolants and Table 1.8 the disadvantages.

TABLE NO. 1.7

SOME ADVANTAGES OF ALTERNATIVE COOLANTS FOR FAST BREEDERS

Sodium-Cooled	Gas-Cooled	Steam-Cooled
1.) Good heat transfer characteristics.	1.) Chemically and neutronically inert coolant.	1.) Direct cycle.
2.) Low Pressure.	2.) No coolant phase-change void effect.	2.) Large steam technology base.
3.) Extensive and growing fast reactor fuel and component experience.	3.) High Internal conversion ratio and longer refueling interval.	3.) Transparent coolant facilitates refueling and maintenance.
	4.) Potential high breeding ratio and short doubling time.	
	5.) Direct cycle possible.	
	6.) Transparent coolant facilitates refueling and maintenance.	

TABLE NO. 1.8

SOME DISADVANTAGES OF ALTERNATIVE COOLANTS FOR FAST BREEDERS

Sodium-Cooled	Gas-Cooled	Steam-Cooled
1) Opaque coolant	1) No fast-reactor fuel or component experience.	1) No fast reactor fuel or component experience.
2) Chemically reactive and radioactive coolant.	2) Emergency core cooling problem.	2) Emergency cooling problem.
3) Phase change void effect.	3) Heat transfer limited.	3) Heat transfer limited.
4) Secondary heat transport system required.	4) Stringent leak requirements.	4) Flooding and coolant density coefficient uncertainty.
5) Component developments required (esp. steam generators.	5) Component development required.	5) Fission product carryover to turbine.
6) Lack of electric utility experience.	6) High coolant velocity.	6) High coolant velocity.
	7) High pressure.	7) High pressure.
	8) Flooding.	8) Corrosion.

At this point, we wish only to suggest the tentative conclusion that a gas cooled system should be evaluated for shipboard FBR applications. This task will be the central theme of the present thesis.

1.7 SYNOPSIS OF CONCLUSIONS

Nuclear propulsion for ship use is becoming increasingly competitive. In this chapter the reasons for this improving environment have been outlined using information developed by other sources. The train of logic was then further extended to suggest that maritime nuclear systems which are closely similar to land based central station systems have a significant cost advantage by being able to capitalize on some of the latter's advantages of scale: hence the present dominance of the PWR concept for ships. Since the fast breeder reactor is projected to displace the LWR on land, it then becomes of interest to assess the future of FBR's for shipboard propulsion. Finally, while sodium cooled systems are the primary choice for central station FBR's, their reactive coolant and high system weight may tip the scales in favor of the gas-cooled FBR for shipboard use. The remainder of this thesis is therefore devoted to assessment of the shipboard GCFBR concept.

Chapter 2 discusses gas-cooled reactor history and present developments. Chapter 3 presents an engineering and physics preliminary design of a gas-cooled system. Chapter 4 assesses the economic prospects and presents final conclusions.

CHAPTER II

GAS-COOLED REACTORS

2.1 INTRODUCTION

In Chapter I the factors motivating development of marine nuclear propulsion systems in general, and fast gas-cooled reactors in particular, were presented. In this chapter an abbreviated review of the technology which has been used as the basis for the proposed design will be presented.

First, selected aspects of gas-cooled reactor history are presented, with important advances highlighted. Proposed maritime designs and features applicable to a direct cycle gas turbine are then discussed.

Other proposed designs, such as the Sodium-CO_2 Breeder; the Feher cycle supercritical system, the Russian disassociating gas cycle, the direct Brayton cycle, and the indirect GCFBR are briefly discussed in order to show some comparative contemporary thoughts and to highlight some of the rationale for designing a new system.

Major developments that make the gas-cooled reactor more attractive, such as the prestressed concrete reactor vessel and the closed cycle gas turbine are discussed in more detail.

The last section of Chapter II discusses the selection of parameters for the system to be analyzed in this thesis.

2.2 GAS-COOLED REACTOR HISTORY

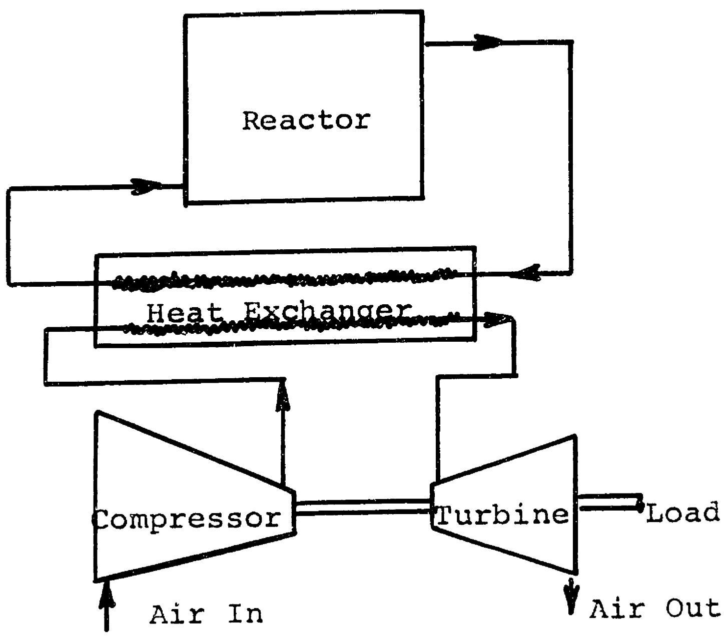
2.2.1 Possible Gas-Cooled Cycle Designs

With a gas-cooled reactor four major thermodynamic cycles are possible. There is the open direct cycle, the closed direct cycle, the open indirect cycle and the closed indirect cycle. Examples of the indirect cycles are shown in Fig. 2.1, and the direct cycles are shown in Fig. 2.2.

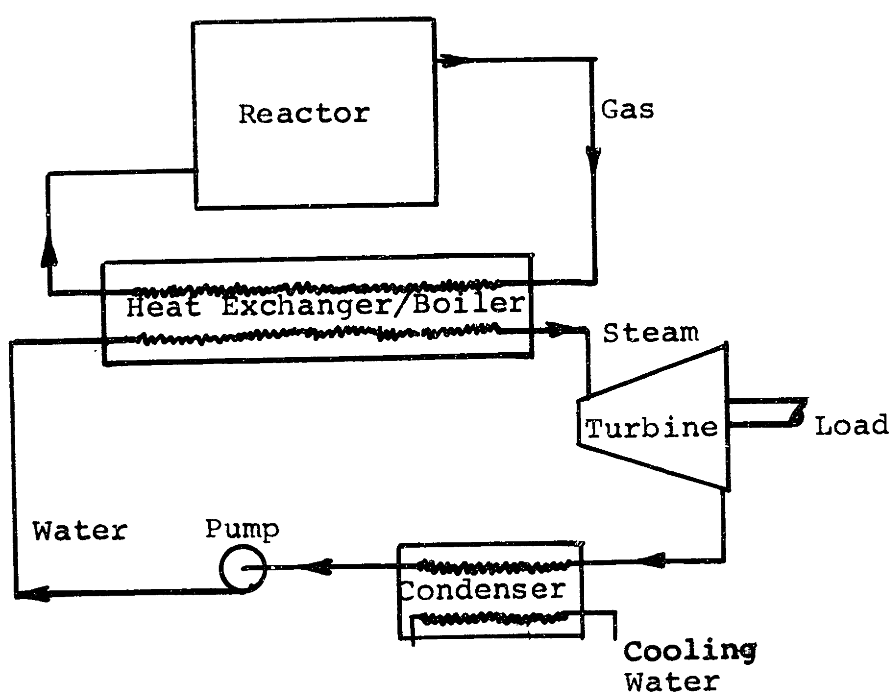
The open direct cycle involves circulating a gas, usually air, through the reactor, a turbine. amd then exhausting the gas to the atmosphere. This cycle is not currently acceptable from an environmental standpoint, as fission products from defective fuel elements and air activation products (e.g. Argon-41) would be released to the environment.

The direct closed cycle does not suffer from the problem of releasing radionuclides to the atmosphere. In this system the gas is heated in the reactor, expanded in a turbine to produce work; heat is then transferred in a recuperater and rejected in a precooler. The gas is then compressed and returned to the reactor. The

Fig. 2.1 INDIRECT GAS CYCLES



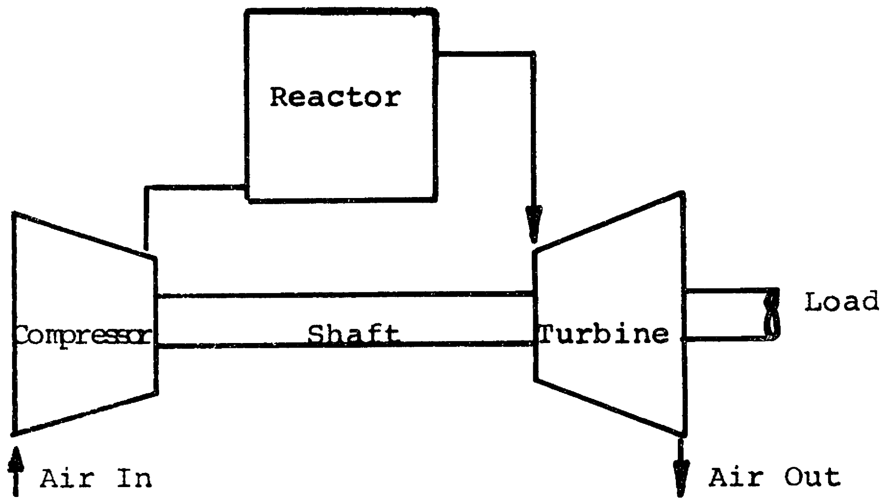
INDIRECT OPEN CYCLE



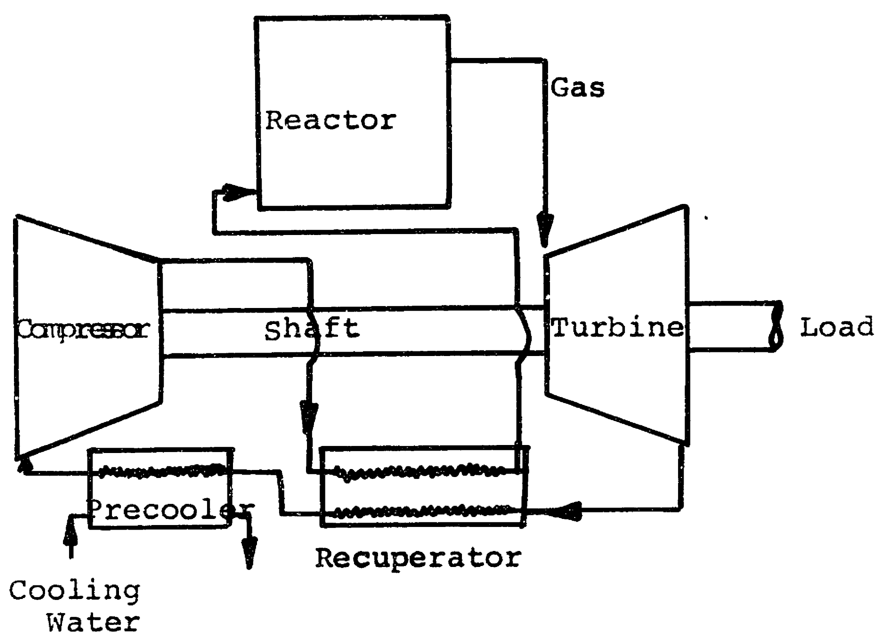
INDIRECT CLOSED CYCLE

Ref. E3

Fig. 2.2 DIRECT GAS CYCLES



DIRECT OPEN CYCLE



DIRECT CLOSED CYCLE

Ref. E3

cycle can use practically any gas, and is simple and efficient at high reactor outlet temperatures. There are a number of different variations of this basic cycle.

The open indirect cycle is used with a closed gas loop acting as the heat source for the open cycle described previously.

The closed indirect cycle is the one currently in use for gas-cooled reactors. The gas is circulated by blowers through the reactor core, steam generators and back to the reactor. The secondary working fluid can be gas or water/steam; in either case driving a turbine.

The indirect cycle gives added protection against fission product release and allows use of the more widely applied steam turbine. The direct cycle, on the other hand, has several distinct advantages for use on a ship. The most important is that the direct cycle design eliminates a large, heavy and expensive heat exchanger and all the extra associated components and piping. This simplification leads to a more compact power plant arrangement, usually of lesser weight and cost then for indirect cycle plants. Thus, there are potential capital cost savings in the direct cycle. This is a particularly pertinent consideration when considering fast reactor designs which currently are burdened by very high capital costs.

2.2.2 Early Gas-Cooled Reactors

The major initial development of gas-cooled reactors for central station power generation, as opposed to plutonium production, took place in Great Britain and, later, France.

This effort consisted of the construction and operation of natural uranium-fueled, graphite moderated and CO_2 cooled reactors starting with the Calder Hall in 1956. (E3) Since then a number of other countries have started gas-cooled reactor programs, and plants have evolved considerably from the "Calder Hall" Type. The British progressed from the Calder Hall type through the Magnox class of reactors to the present Advanced Gas-Cooled Reactor (AGR) and Dragon Type. Each was an improvement over previous designs: CO_2 is being supplanted by Helium; steel pressure vessels gave way to prestressed concrete, natural uranium was enriched, temperatures and efficiencies improved and reliability increased. Table 2.1 lists Gas-cooled reactors and their major characteristics.

The British, while experimenting with the Helium-cooled Dragon Reactor, are still marketing the AGR. The AGR is CO_2 cooled and uses slightly enriched Uranium-Plutonium fuel. The efficiency is above 40% with an indirect steam cycle. (S1) The AGR is a massive reactor due to the large amount of graphite used in the core.

TABLE 2.1 GAS-COOLED NUCLEAR REACTORS: COMMERCIAL POWER STATIONS AND POWER PROTOTYPES

Name	Location	Power MW(E)	Power date	Moderator	Coolant	P.V. Material	Fuel	Remarks
Calder Hall	U.K.	4 × 45	1956	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Chapelcross	U.K.	4 × 45	1959	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Berkeley	U.K.	2 × 138	1962	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Bradwell	U.K.	2 × 150	1962	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Hunterston	U.K.	2 × 180	1964	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Hinkley Point	U.K.	2 × 250	1965	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Trawsfynydd	U.K.	2 × 250	1965	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Dungeness 'A'	U.K.	2 × 275	1965	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Sizewell	U.K.	2 × 290	1966	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Oldbury	U.K.	2 × 300	1966	Graphite	Carbon dioxide	Concrete	Nat. U, magnox	
Wylfa	U.K.	2 × 590	1968	Graphite	Carbon dioxide	Concrete	Nat. U, magnox	
Latina	Italy	1 × 210	1963	Graphite	Carbon dioxide	Steel	Nat. U, magnox	
Tokai-Mura	Japan	1 × 154	1965	Graphite	Carbon dioxide	Steel	Nat. U, magnox	**Hollow rod F.E.'s**
G2, G3 Marcoule	France	2 × 40	1958	Graphite	Carbon dioxide	Core	Nat. U, Mg alloy	
E.D.F.1, Chinon	France	1 × 62	1963	Graphite	Carbon dioxide	Steel	Nat. U/Mo, Mg alloy	
E.D.F.2, Chinon	France	1 × 213	1965	Graphite	Carbon dioxide	Steel	Nat. U/Mo, Mg alloy	
E.D.F.3, Chinon	France	1 × 476	1966	Graphite	Carbon dioxide	Concrete	Nat. U/Mo, Mg alloy	
St Laurent I	France	1 × 487	1968	Graphite	Carbon dioxide	Concrete	Nat. U/Mo, Mg alloy	
St Laurent II	France	1 × 516	1970	Graphite	Carbon dioxide	Concrete	Nat. U/Mo, Mg alloy	
Bugey I	France	1 × 488	1971	Graphite	Carbon dioxide	Concrete	Nat. U/Mo, Mg alloy	
Windscale AGR	U.K.	1 × 28	1963	Graphite	Carbon dioxide	Steel	UO_2, S.S.	Prototype AGR
Dungeness 'B'	U.K.	2 × 610	1970	Graphite	Carbon dioxide	Concrete	UO_2, S.S.	First commercial AGR of 8000 MW(E) programme
Hinkley Point 'B'	U.K.	2 × 625	1971	Graphite	Carbon dioxide	Concrete	UO_2, S.S.	
AVR Pebble Bed	Germany	1 × 15	1966	Graphite	Helium	Steel	UC/ThC spheres	HTGR pebblebed prototype
THTR	Germany	Probably 1 × 300	1972	Graphite	Helium	Concrete	UC/ThC spheres	Pebblebed power reactor
Dragon	U.K.	1 × 20(th)	1964	Graphite	Helium	Steel	U235, Th	HTGR international experiment
Peach Bottom	U.S.A.	1 × 40	1966	Graphite	Helium	Steel	UC/ThC (prismatic)	HTGR prototype
Fort St Vrain	U.S.A.	1 × 330	1971	Graphite	Helium	Concrete	UC/ThC (prismatic)	HTGR based on Peach Bottom
Bohurice	Czechoslovakia	1 × 110	1968	Heavy water	Carbon dioxide	Steel	Nat. U	GCHWR
Brennilis EL4	France	1 × 73	1966	Heavy water	Carbon dioxide	S.S.	UO_2	GCHWR
Lucens	Switzerland	1 × 7	1966	Heavy water	Carbon dioxide	Zr Press Tubes	U, Mg alloy	GCHWR prototype
Niederaibach	Germany	1 × 100	1968	Heavy water				GCHWR

Reference: T1

This makes the Prestressed Concrete Reactor Vessel (PCRV) enormous, measuring approximately 98-feet high and 89-feet in diameter. This alone would preclude use of the AGR in a ship application. The British Dragon reactor is smaller, but is still large by ship standards.

The Dragon Reactor Experiment was carried on in England to demonstrate the major design principles of high temperature gas-cooled reactors. Helium was used to cool, and graphite to moderate, a fully enriched (93%) Uranium and Thorium fuel. The Dragon reactor produced 20 Mwth and has assisted in developing operating knowledge of helium cooled systems. (D2)

The French have built a number of gas-cooled reactors for power production commencing with EDF-1. Since that time all French reactors, with the exception of the Fast Reactor Phenix, have been graphite moderated, CO_2 cooled natural Uranium systems. Little new development work on GCR's has been occurring in France.

The German philosophy has differed from that of the English in the design and construction of gas-cooled power plants. The German industry went first to the helium cooled pebble bed 'AVR' reactor. The 'AVR' reactor delivered its design power of 15 Mw(e) to the grid in 1967. The core contains 100,000 spherical fuel elements which produces a power of 46 Mw thermal. This plant has demonstrated the safety and reliability of a helium-cooled high temperature reactor but further work is necessary to improve its economics. The plant is still in operation and is providing information on helium component design and fuel-coolant interactions.

Presently under construction by a German group is the 25 Mw(e) Geesthacht KSH Nuclear Power plant comprising a high temperature helium-cooled reactor with a direct cycle turbine. (B1) This plant, scheduled for operation in 1973, will combine for the first time a high temperature reactor and a helium turbine. This prototype plant will advance the understanding of many of the problems which will be faced by a ship reactor. Questions which should be answered by the "Geesthacht" plant are: the effects of fission-product deposition in the turbine plant; the dynamic and load-following behavior of the system; the shutdown and emergency cooling of the reactor; and the operating behavior of the helium turbine with its shaft seal system. Table 2.2 presents the major reactor and system parameters. The "Geesthacht" plant is probably the most important forthcoming source for direct cycle reactor-turbine information. (B1)

Features new to this plant which will be studied carefully for their applicability to future plants are: Helium turbine shaft seal system, helium mass flow bypass control system and the emergency core cooling and shutdown system. (B2)

TABLE NO. 2.2

CHARACTERISTICS OF THE GEESTHACHT KSH REACTOR

Thermal Power, Mw(th)	65
Electrical Power net, Mw(e)	24
Efficiency %	37
Type and Moderator	Thermal graphite
Power Density, MW/M^3	6.4
Coolant Pressure, atm (psia)	25 (367.5)
Reactor inlet temperature, ^oC	425 (797) ^oF
Reactor outlet temperature, ^oC	735 (1355) ^oF
Helium mass flow lb/sec	89.3
Core lifetime, days	900
Turbine inlet temperature, ^oC	730 (1346) ^oF
Compressor inlet temperature, ^oC	15 (59) ^oF
Heat Exchanger temperature difference, ^oC	30 (86) ^oF
Cooling water temperature, ^oC	10 (50) ^oF
Pressure ratio across turbine	2.55

Fuel

UO_2 90% enriched U-235

Thorium - Fertile

Ref. B1

In the U.S. gas-cooled reactor development for central station applications achieved practical implementation with the Peach Bottom 40 Mwe High Temperature Gas-Cooled prototype reactor plant. (G2) Peach Bottom since its inception in 1957 and commercial operation in 1967 has served as the design and operating prototype for the 330 Mw(e) Ft. St. Vrain plant, currently nearing completion, and the 1100 Mw(e) High Temperature Gas-Cooled Reactor (HTGR) plants now on order. The HTGR plants are Helium-cooled and graphite moderated using a Uranium/ Thorium fuel cycle. The present component research and development for the HTGR will help considerably in Gas-Cooled Fast Reactor (GCFR) design. Important HTGR parameters are listed in Table 2.3. (G3) The parameters are for the 330 Mw(e) Fort St. Vrain HTGR. The component development for the indirect cycle HTGR which will aid in the development of a direct cycle GCFR design will be primarily in the areas of Prestressed Concrete Reactor Vessel (PCRV) design and construction, and helium purification and handling systems.

Recent HTGR development work has had as one objective coupling the HTGR to a helium turbine. (F5) Any developmental or design work done in this area would later assist with the GCFR design.

Although it represents a now outmoded concept the ML-1 power plant, which was a military mobile gas-

TABLE 2.3

CHARACTERISTICS OF THE FT. ST. VRAIN HIGH TEMPERATURE GAS-COOLED REACTOR.

POWER OUTPUT	
Thermal rating, kW(t)	841,000
Gross Electrical, kW(e)	342,000
Net Electrical, kW(e)	330,000
Net Plant Efficiency, %	39.23
HELIUM CIRCUIT	
Temperature	
Core inlet, F	760
Core outlet, F	1,430
Pressure	
Circulator inlet, psia	686
Circulator outlet, psia	700
Flow Rate, #/hr.	3.4×10^6
CORE DESIGN	
Effective diameter, ft.	19.6
Height, ft.	15.6
Initial U^{235} loading, Kg	1,020
Initial Th^{232} loading, Kg	19,200
Power density, kW(t)/liter	6.3
Specific power Avg. (equilibrium, kW(t)/Kg)	1,100

TABLE 2.3 (continued)

Fuel life (full power, yr)	5.1
Average burnup (equilibrium), MWd/Ton (U&Th)	100,000
Conversion ratio (equiliabium)	0.62
PCRV	
Overall Height, ft.	106
Overall diameter, ft.	61
Internal height, ft.	75
Internal diameter, ft.	31
Reference pressure, psig	845
Normal working pressure,psia	700
STEAM GENERATORS	
Type	Helical coil, once-through
Number	2
Feedwater temperature, F	403
Feedwater pressure, psia	3,100
Feedwater flow rate (total), lb/hr.	2.31×10^8
Outlet steam pressure, psia	2,512
Outlet steam temperature, F	1,005

cooled direct cycle system, deserves notice here because it was a nuclear powered direct cycle system and did achieve operational status. This power plant had a nitrogen cooled core, with water moderation and 93% enriched U-235 fuel. The heated gas was used to do work in a Brayton cycle with regeneration. (A1) This plant operated from 1961 to 1965, when it was secured by the Army. Table 2.4 shows some major system parameters.

2.3 MARITIME GAS-COOLED DESIGNS

The advantages of a gas coolant in a mobile environment were recognized early as incentives favoring development of a Maritime gas-cooled reactor. These advantages are in the transparency of the coolant, its noncombustibility and low induced-radioactivity potential. In April, 1956, Nucleonics Magazine (N2) reported on offers by nine companies "to build or conduct research on, a more advanced power plant, probably a closed-cycle nuclear gas turbine, using helium as a system fluid for a second atomic tanker to go in service in 1961." (The first was to be a PWR system.) The ship was to be approximately 38,000 tons displacement, 22,000 shp and 20-21 knots. Of the nine designs proposed, two have survived in modified form past the initial stages: the Maritime Gas-Cooled Reactor (MGCR) proposed by the General Atomics Division of General Dynamics, and the 630 A Nuclear Steam Generator proposed by the General Electric Corporation.

TABLE 2.4 ML-1 DESIGN CHARACTERISTICS

GENERAL	
Gross electrical output	400 kw
Net electrical output	330 kw
Reactor thermal power (total)	3.4 Mw
Cycle efficiency	13.3 %
COOLANT	
Coolant flow	95,000 lb/hr.
Coolant	Nitrogen
Power density	700 kw/ft^3
Max. surface temp. fuel	1,420 °F
CORE LIFE, FULL POWER	10,000 hr. design
CORE SIZE	
Diameter	22 in. equivalent
Height	22 in.
Enrichment 93% V-L35	
POWER CYCLE	
Cycle characteristics	
Net power, Kw	330
Reactor inlet, °F	791
Reactor outlet, °F	1,200

TABLE 2.4 (continued)

Compressor inlet °F	132
Compressor inlet psia	117
Compressor outlet, psia	320
Reactor inlet, psia	313

2.3.1 Maritime Gas-Cooled Reactor

The proposed MGCR utilizes a helium-cooled, high temperature beryllium-oxide moderated thermal reactor directly coupled to two closed-cycle gas turbines of special design. (M1) The MGCR heats Helium to 1300°F where it is then expanded through two turbines. The first turbine drives the compressors and the second supplies shaft power. The Helium is then exhausted to a regenerator where it exchanges heat with the compressed helium stream, and then passes through the precooler prior to being recompressed. The major reactor and plant parameters are given in Table 2.5. The important feature of the propulsion plant is that all four axial turbomachines, compressors and turbines are contained in a single housing. The plant efficiency is approximately the same as a PWR. Work on the MGCR was not carried far enough to generate reliable economic analysis. The plant is compact, however, and the location of the components could be used as a guide for future designs.

2.3.2 The 630A Nuclear Steam Generator

The 630A is a nuclear fueled steam generator-superheater to be used for nuclear merchant ship propulsion. (D1) The design has evolved from land based and nuclear aircraft technologies. The Mark IV design is

TABLE 2.5 MARITIME GAS-COOLED REACTOR (PROTOTYPE)

Total Reactor Power, Mwt	74
Shaft horsepower, shp	32,000
Efficiency, %	32.1
Pressure, Reactor outlet, psia	1,080
Pressure, system maximum psia	1,120
Temperature reactor outlet °F (°C)	1,300 (704)
Fuel type	UO^2
Enrichment, % U-235	8.85
Fuel lifetime (yrs.)	1.5 - 2.7
Type of clad (not decided)	SS 316L INCONEL Hastelloy X
Type of Moderator	Be O
Containment	
Diameter, ft.	30
Length, ft.	60
Reactor Core	
Diameter, ft.	6.37
Height, ft.	6.37
No. of fuel assemblies	308

(Ref. M1)

the result of technological evolution from 1961 to date, leading to an economical, small and light reactor for ship use. The initial 630A used air as coolant but the Mark IV and subsequent designs will be helium-cooled. The 630A Mark IV system will develop 27,300 shp at 61.34 Mw(t) reactor power. The Heat Transfer Reactor Experiment (HTRE) served to develop information on fuel, coolant and shielding for the 630A system. (D1) Propulsion plant specifications are given in Table 2.6. This system was designed as an indirect cycle with the closed loop gas supplying heat to the steam generator.

2.4 OTHER PROPOSED REACTOR AND CYCLE DESIGNS

Over the past several years a number of new reactor designs and thermodynamic cycles have been proposed for use in power plant applications. A few of the concepts having at least minimal pertinence to the present application were the supercritical CO_2 or Feher Cycle, (F1, F2, F3); the Sodium-CO_2 Fast Breeder Reactor Concept, (V1); the Marine Nuclear Power Plant Design utilizing the direct Brayton cycle, (G1); the Russian Dissociating-gas cycle (K3) and the direct and indirect gas-cooled fast breeder designs. (F1, G5, G6).

2.4.1 Feher Cycle

The supercritical carbon dioxide cycle was proposed

TABLE 2.6 630A PROPULSION PLANT SPECIFICATIONS

Reactor power	61.34 Mw
Gas Flow	65.0 lb./sec.
Reactor Discharge temperature	1,200 °F
Circulator discharge pressure	830 psia
Primary loop gas pressure drop.	7.2 psi
Overall thermal efficiency	33.2 %
Rating, shp - normal	27,300
Secondary System	
Throttle temperature	1,000 °F
Throttle pressure	1,500 psig
Flow rate	$1,726 \times 10^5$ lb/hr.
Condenser pressure	1.5 inches hg. abs.
Auxiliary Power	1,446 kw
Curculator Power	482 kw
Boiler Feed Pump Power	522 kw

Ref. D1

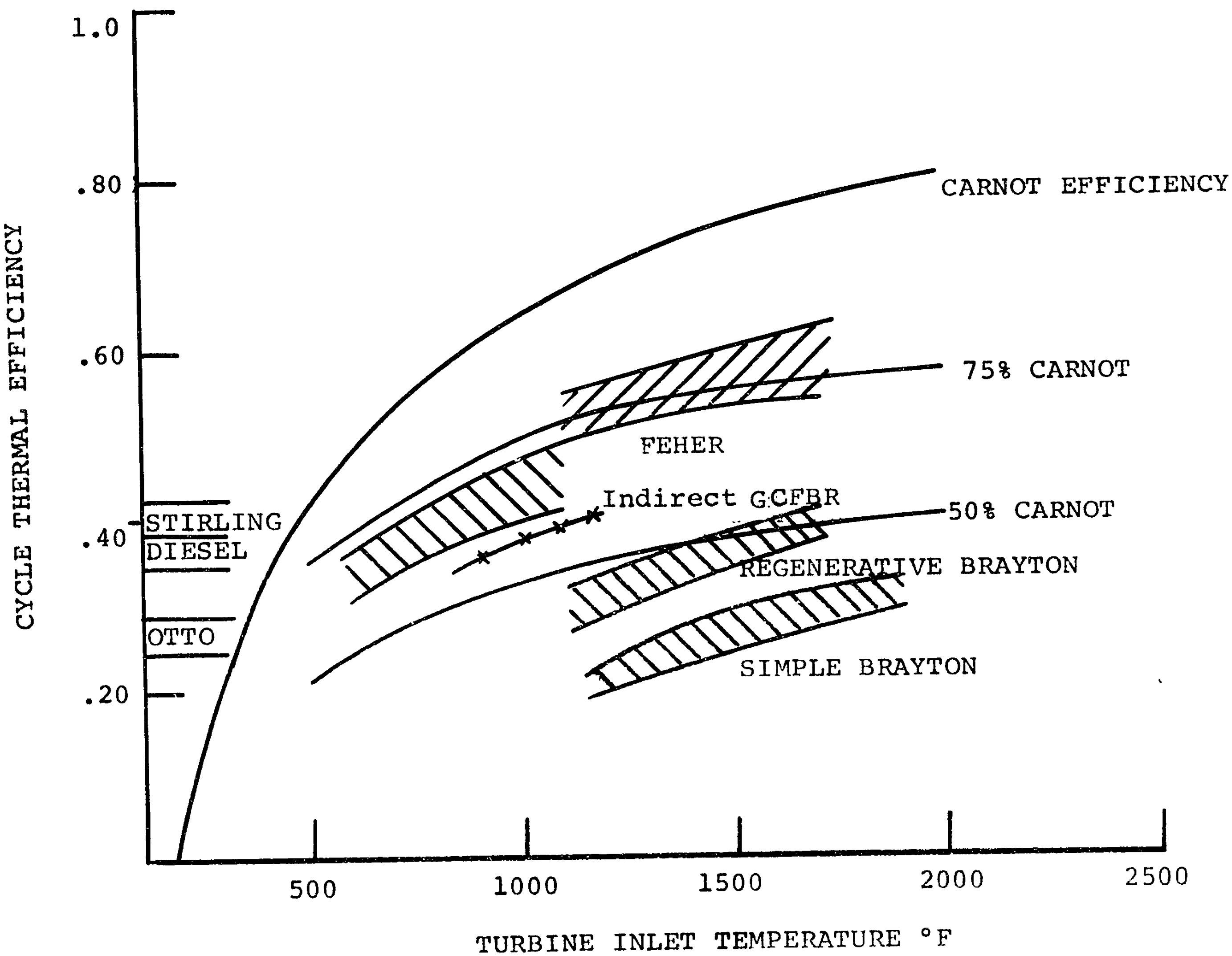
in 1962 by E. G. Feher, and since that time has been worked into a space power design (F4). No commercial system has yet been built applying the Feher cycle. The cycle has the advantages of a high overall efficiency, single phase fluid, and potential low system weight. These items are all important advantages for a mobile propulsion plant. A comparison of cycle efficiencies is given in Fig. 2.3. As can be seen from the figure, most cycles have low efficiencies in the range of feasible maximum reactor outlet temperatures. The Feher cycle has a much higher efficiency than the others shown.

The major components of the Feher cycle are a heat source, combination turbine and pump unit with an alternator, a recuperator and a precooler before the pump. The prime reasons for not applying the Feher cycle engine to ship propulsion at present are lack of component development, and high cycle pressure (2610 psia).

2.4.2 SODIUM-CO_2 FAST BREEDER REACTOR CONCEPT

The present Liquid Metal Fast Breeder design calls for a three step heat removal system: radioactive sodium to non-radioactive sodium to the water-steam prime mover circuit. This is considered essential in order to protect the primary circuit from the effects of the sodium-water interaction in the event of a leak. The sodium-CO_2 system could replace the secondary

Fig. 2.3 TYPICAL CYCLE EFFICIENCIES



REF. (F4)

sodium loop and the steam-water loop with a single CO_2 loop. Carbon dioxide and sodium react chemically, but sodium carbonate and free carbon do not appear to have major detrimental effects on the system. Carburization of the cladding may occur, but more research is necessary to determine the full effects of CO_2 leakage into the sodium. The cost of the Na-CO_2 heat exchanger is also an important item requiring further assessment.

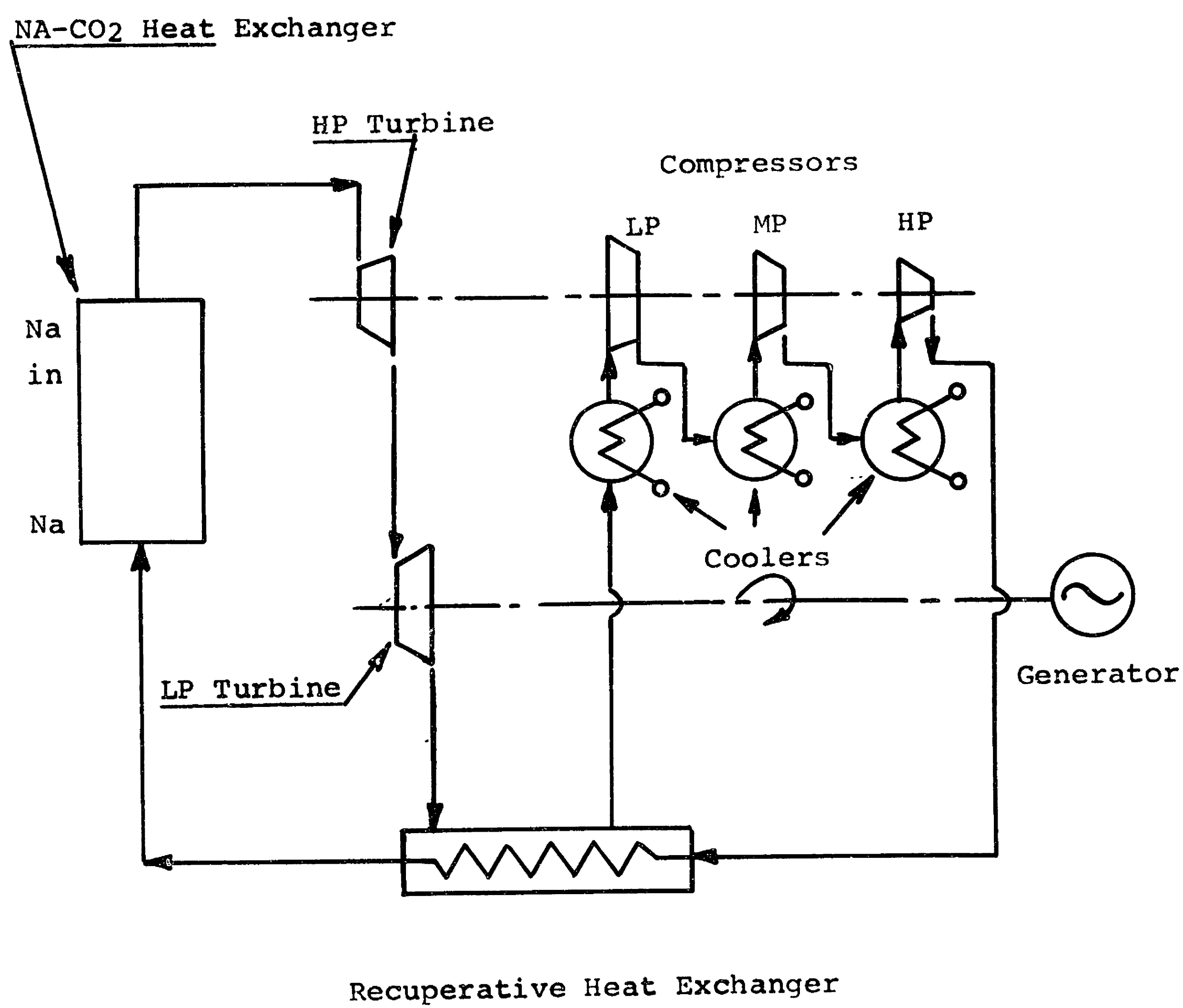
There are three basic CO_2 cycles which could be used in this concept. These are a) simple cycle with two or three compressors at a high pressure of 100-150 atm; b) a reheat cycle with three compressors, at a high pressure range of 180 to 250 atm; and c) high pressure cycle (250-300 atm) with flow derivation and work in the subcritical region of the T-S diagram for CO_2.

The circuit diagram of the simple cycle is shown in Fig. 2.4. The T-S diagrams can be found in Reference V-1.

The major advantages of this concept, compact system, simplicity, reliability and lower capital cost than the three loop sodium system may not outweigh the disadvantages.

The disadvantages of the sodium-cooled reactor still remain (i.e., chemical reactivity) and the disadvantages of gas circuits are now introduced into the

Fig. 2.4 TYPICAL NA-CO_2 CYCLE EQUIPMENT



Ref. V1

system. The problem of devising a compact and economical heat exchanger to tie these systems into an integral thermodynamic cycle will be particularly difficult.

2.4.3 Marine Nuclear Power Plant Design Utilizing the Direct Brayton Cycle.

The system, proposed for a passenger ship in Reference Gl, consists of the 630A reactor supplying heat to a direct cycle helium turbine. The propulsion system was designed to meet requirements of a specific passenger ship and the machinery sizes are for a 40,000 shp system. The Brayton cycle efficiency of 37.5% was better than a steam plant, and the designers claim a much tighter and smaller volume plant. No shielding is used around the helium turbines, compressors and heat exchangers. The cycle is designed to use existing turbo machinery and a well-studied marine thermal reactor, the 630A. The cycle state points and plant layout are shown in Figs. 2.5 and 2.6.

As with any other direct cycle plant, shielding of the turbines and other helium circulating equipment may be necessary if practical fuel defect frequencies are to be allowed, and this will increase system weight and cost.

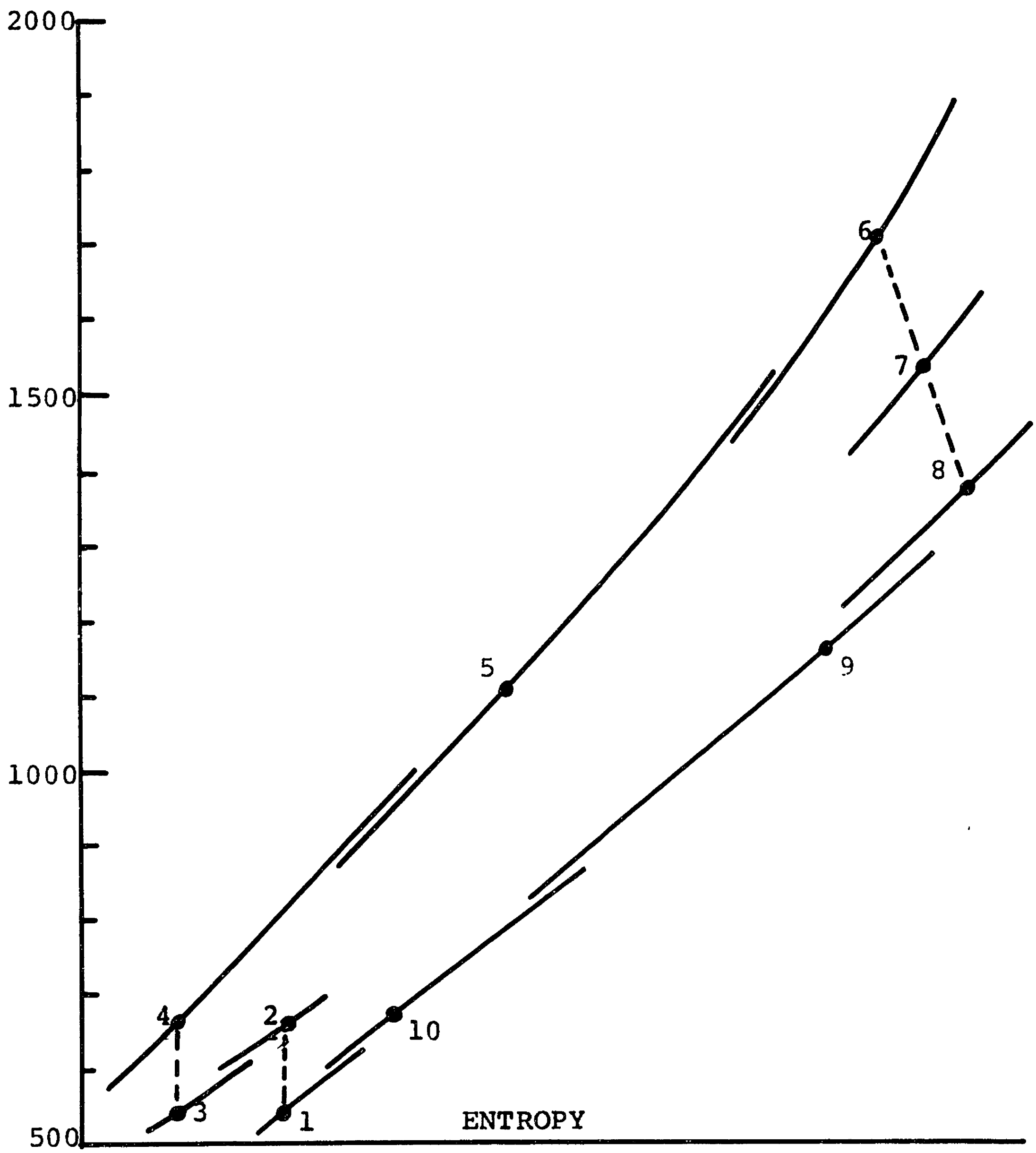
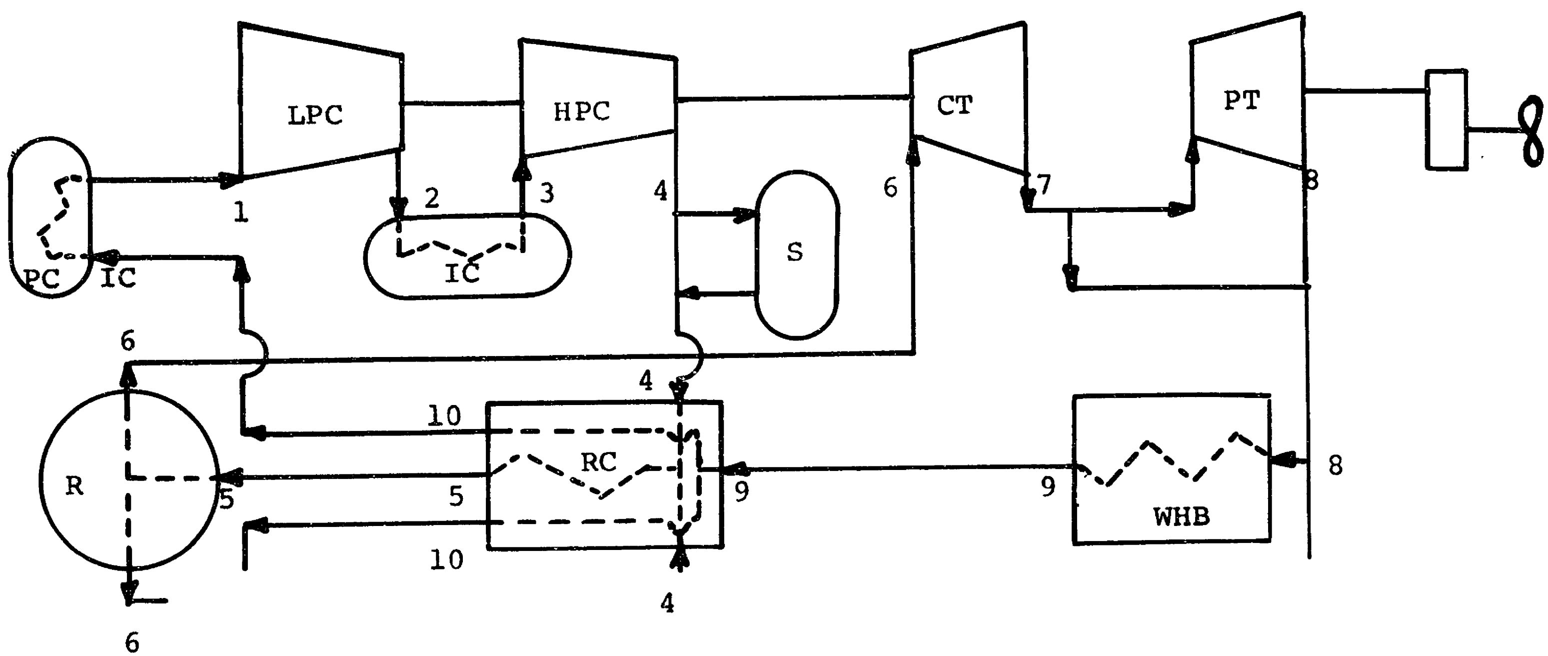


Fig. 2.5 DIRECT BRAYTON CYCLE WITH THE 630A MARINE THERMAL REACTOR

State Pt. No. 1 Low press. compressor in
2. Low press. compressor out
3. intercooler outlet
4. High press. compressor outlet
5. recuperator high temperature outlet and reactor inlet
6. reactor outlet
7. Compressor Turbine exhaust
8. Power turbine exhaust
9. Waste-heat boiler outlet
10. recuperator low temperature outlet and precooler inlet

Ref. (G3)

Fig. 2.6 Helium Flow Schematic for Direct Brayton Cycle with 630A
State points are the same as in Fig. 2.5



Legend for Figure 2

LPC	low pressure compressor
HPC	high pressure compressor
CT	compressor turbine
PT	power turbine
IC	intercooler
PC	precooler
S	helium storage
R	reactor
RC	recuperator
WHB	waste-heat boiler

Ref. G3

2.4.4 Dissociating Gas as a Nuclear Reactor Coolant

It has been proposed by a number of Russian scientists to use a dissociating gas as a coolant for a nuclear reactor. It is claimed, and calculations show, that thermal efficiencies of 50% or better are possible with dissociating gas cycles. Proposed for these cycles are N_2O_4, $A2_2CL_6$, and AL_2Br_6 gases. These gases all undergo reversible chemical reactions with an increase of mole number in disassociation.

$$\text{i.e. } N_2O_4 \leftrightarrow 2NO_2 - 13.7 \frac{KCAL}{MOLE} \leftrightarrow 2NO + O_2 - 27 \frac{KCAL}{MOLE}$$

Temp. range 300 - 450°C

The advantage is gained by an improved compression effectiveness, exothermic chemical recombination reactions during cooling and endothermic chemical reactions during heating. (K4) From the thermodynamic cycle viewpoint this system looks very good. The disadvantages lie in that too little information exists on all the chemical and thermodynamic properties of the dissociating gases, on their corrosive effect upon materials in general, and in a radiation environment in particular.

2.4.5 The Indirect Cycle Gas-Cooled Fast Breeder Reactor

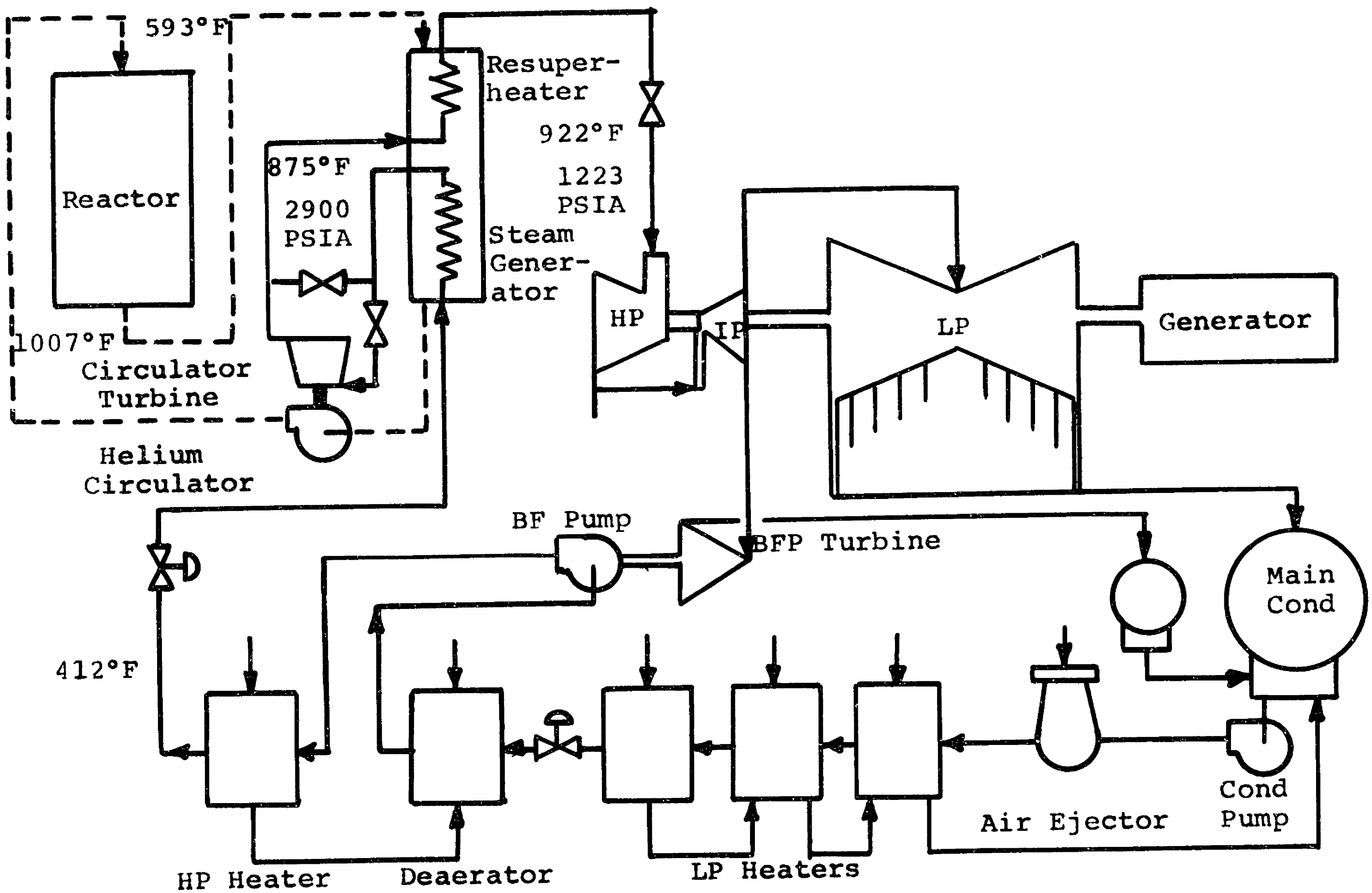
Considerable literature has been published on the Gas-Cooled Fast Breeder Reactor (GCFBR) in the past ten years (F1, P1, W1, G5).

The two countries where the most research in GCFBR technology is being carried out are the U.S., at Gulf General Atomic, and Germany, at the Karlsruhe research center.

The U.S. effort has centered on utilizing: 1) HTGR component development, and 2) Liquid Metal Fast Breeder Reactor fuel element technology, in developing an indirect cycle GCFBR. Presently effort is being expended to develop a design for a 300 Mw(e) Demonstration plant. (F1) This plant would be extremely conservative in design and would have reactor fuel and clad temperatures akin to those established in the LMFBR development program. Figure 2.7 shows the present indirect cycle design and Table 2.7 shows the corresponding parameters.

The Germans, like the Americans, are concentrating most of their GCFBR effort on the indirect cycle using FBR fuel technology. In addition, they have looked at and are doing, some basic research in, two alternate GCFBR concepts. The first of these is the Gas-turbine-connected GCFBR with oxide fuel and Vanadium-clad fuel pins. The second is the indirect cycle with

Fig. 2.7 GCFR Steam Cycle



Ref. F1

TABLE 2.7 INDIRECT GCFBR CYCLE PARAMETERS

Core Geometry	
Volume, liters	3184
Length-to-diameter ratio	0.5
Volume fractions	
Fuel, including center hole	0.300
Cladding	0.097
Gaps and structure	0.160
Coolant	0.443
Fuel rod diameter, cm	0.723
Number of fuel rods	30,700
Number of fuel elements	87
Number of control elements	31
Number of blanket elements	93
Core Physics	
Fuel average enrichment, % fissile	18.5
Radial power maximum to average	1.248
Conversion ratio	1.33
Reactivity drop per reload, $	9.00
Performance	
Reactor inlet temperature, °C	312
Reactor outlet temperature, °C	542
MWt/kg fissile	0.605

TABLE 2.7 (continued)

Thermal efficiency, %	37.6
Maximum rod rating, kW/ft	12.6
Power density, kW/liter	238
Thermal output, MW	824
Electric output, MW	310

oxide fuel in coated particle form. The two alternate concepts are heavily dependent on present thermal reactor research for both gas turbine and coated particle development.

Most of the present German work is centered on fuel element development. A prototype of the vented fuel rod pin is presently undergoing irradiation testing and additional research is being done on Vanadium cladding. Strong safety programs for all the concepts are being developed. Research is continuing on PCRV's for both thermal and fast reactors.

2.5 GAS-COOLED REACTOR DEVELOPMENTS

2.5.1 Prestressed Concrete Reactor Vessel (PCRV) Technology

The first gas-cooled reactors had steel pressure vessels, as shown on Table 2.8. These reactors were fairly small, with the largest enclosed volume at 153,800-ft.^{3} and a pressure of approximately 140 psia. Today reactors having enclosed volumes of greater than 450,000-ft.^{3} are in existence and are operating at pressures greater than 400 psia. (T2). These later reactors have PCRV's surrounding the core and, in some cases, the entire primary system. The first PCRV's were built for the Marcoule reactors in France in 1958. See Fig. 2.8.

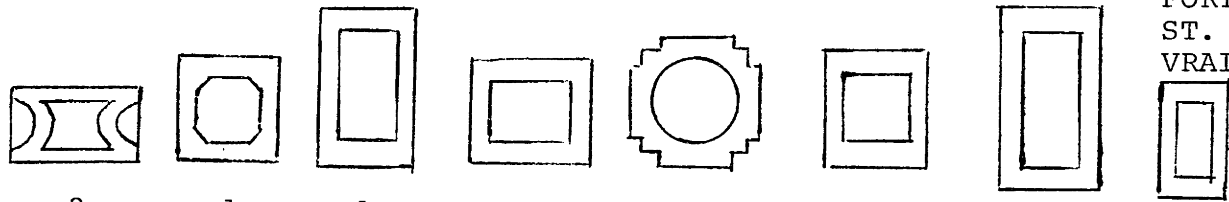
TABLE 2.8

STEEL PRESSURE VESSELS FOR GAS-COOLED POWER REACTORS

	Calder Hall	Berkeley	EDF1	EDF2	Sizewell	WAGR	EGCR	Peach Bottom
Shape	Cyl	Cyl	Cyl	Sph	Sph	Cyl	Cyl	Cyl
Diameter (m)	11.3	15.2	10	18.5	19.5	6.4	6.1	4.3
Height (m)	20.6	24.4	23	----	----	16.3	14	10.8
Thickness (cm)	5	7.5	10.6	10	10.5	4.5-11	7	6.4
Coolant	CO_2	CO_2	CO_2	CO_2	CO_2	CO_2	He	He
Pressure (atm)	6.8	9.5	28	26	19	19.5	22	24
Inlet temp (°C)	145	160	140	200	214	330	265	345
Outlet temp (°C)	345	345	355	380	410	575	565	750
Reactor Power								
Mw(t)	230	565	300	850	950	100	84	115
Mw(e)	54	138	70	200	290	27	22	40

Reference L5

FIG. 2.8--COMPARISON OF SEVERAL PRESTRESSED CONCRETE REACTOR VESSELS

PLANT	MARCOULE	EDF3	EDF4	OLDBURY	WYLFA	DUNGENESS	BUGEY	HTGR FORT ST. VRAIN
SHAPE TO SCALE								
NUMBER OF VESSELS	2	1	1	2	2	2	1	1
POWER, MW(o)/REACTOR	30	480	480	280	590	600	560	330
PRESSURE,PSI	230	440	440	385	385	485	620	700
INSULATION (HOT FACE),°F	120	770	440	475	480	1250	440	750
INTERNAL DIAM,FT	46	62	62	77	96	65.5	56	31
INTERNAL LENGTH,FT	51	66	119	60	--	58	130	75
CONCRETE THICKNESS,FT	10	13-18	15-21	15-22	11(MIN)	12.5-21	18-24.5	9-15
ARRANGEMENT	CORE INSIDE VESSEL ←		PRIMARY SYSTEM INSIDE VESSEL →					
BOILER LOCATION	OUTSIDE CORE →		BELOW CORE	AROUND CORE →			BELOW CORE →	
OPERATION	1959-60	1966	1967	1967	1968	1970	1971	1972

Ref. L5

They were small and highly overdesigned for safety. Reasons for employing concrete reactor vessels rather than steel are listed in Table 2.9.

Almost all the HTGR's being built today are using PCRV's to contain not only the core but the entire primary system as shown in Fig. 2.9.

The PCRV has been proposed for use with the gas-cooled fast reactor. The features of the PCRV which are of particular value for the fast reactor application are good resistance to radiation damage and non-explosive slow failure leading to long depressurization times.

The PCRV could also have shipboard advantages in that no extra biological shielding is necessary, and the system package can be made fairly compact even for high pressures, although not as small as steel vessels plus optimally designed shielding.

2.5.2 Gas Turbine Development

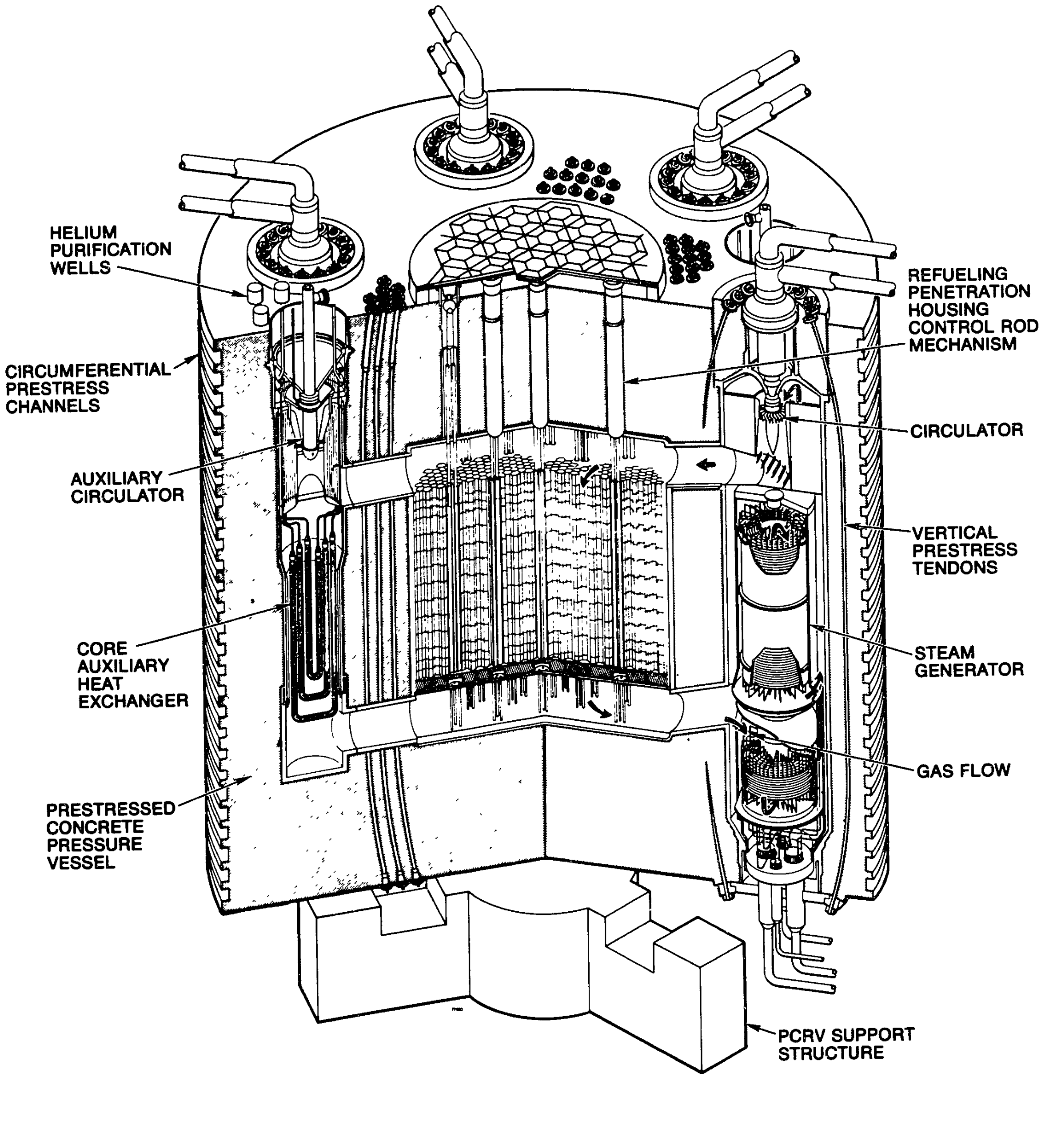
Although there is a vast amount of fossil-fired open cycle gas turbine experience, primarily in aircraft and, more recently, in central station peak power units, the less extensive closed cycle applications are most pertinent to our present interests. Closed cycle gas turbine plants have been used since 1956 in Germany. Table 2.10 lists the closed cycle gas turbine

TABLE 2.9 NUCLEAR PRESSURE VESSELS: STEEL VS. CONCRETE

	PROPERTIES REQUIRED	WELDED STEEL	PRESTRESSED CONCRETE	RELATIVE IMPORTANCE (10=VITAL, 0=USELESS)
1	Structural resistance to pressure loading	Good. (Thin welded membrane)	Good. (Thick cast-in-situ structure)	10
2	Impermeability--ability to contain gas	Good. Contains gas without difficulty	Probably fair to poor. May become impervious in time. Steel liner employed	4
3	Safe mode of failure	Potentially poor. A major failure would probably be explosive	Good. Cracking leads to nonexplosive failure	9
4	Resistance to temperature effects	Temperature must not fall below crack-arrest value	Temperature must not exceed about 70°C with normal cements. Insulation and cooling required	3
5	Small heat losses	Poor. Insulation required	Very poor, as a result of (4) above	3
6	Resistance to radiation damage	Fair	Good	1
7	Predictability of stresses	Good	Fairly good	6
8	Predictability of deformations	Good	Poor; may improve with experience	6
9	Shielding value	Poor. Additional biological shield required	Good. No additional shield required	3
10	Simplicity of construction	Fairly poor	Fairly poor	2
11	Overall cost	About equal		7

Ref. F2

Fig. 2.9 GENERAL ARRANGEMENT LARGE HTGR



Ref. (G8)

Table 2.10 CLOSED CYCLE GAS TURBINE PLANTS IN COMBINED POWER AND HEAT GENERATING STATIONS

Plant	Ravensburg Germany	Coburg Germany	Nowokashirsk Russia	Oberhausen Germany	Haus Aden Germany	Gelsenkirchen Germany	Geesthact Germany	Spittelau 1 +2 Austria	
Working medium	air	air	air	air	air	air	helium	air	
Fuel	coal	coal	lignite	coal	mine gas + coal	blast fur. gas/oil	nuclear	oil/natural gas	
Operating data:									
Compressoe inlet temp.	20	20	20	30	20	20	20	20/60	^oC
Compressor inlet pressure	7.2	7.3	7	8	9.3	10.2	9.5	7.1/8.4	ata
Turbine inlet temperature	660	680	680	710	680	711	750	720	^oC
Turbine inlet pressure	27	27 1/2	29	32	31	38 1/2	25	44	ata
Speed turbo set	12750	8220	6640	6640	8220	6650	8000	6500	r/p/m/
Speed alternator	3000	3000	3000	3000	1500	3000	3000	3000	r/p/m/
Continuous output	2.3	6.6	12	13.75	6.4	17.25	25	30/22	MW
Efficiency at terminals	25	28	28	29.5	29.5	30	39	31/24	%
Heat supply	2.1-3.5	7-14	8-10	16-24	6.7	17-25	4	25/50	$10.^3$Mcal/ hour
Design characteristics:									
Compressor type	radial	axial/ radial	axial	axial	axial	axial	axial	axial	
Compressor stages	3	6+7+1	9+10	9+10	9+6	7+8	9+9+9	8+4+9	
Turbine type	axial	axial	axial	axial	axial	axial	axial	axial	
Turbine stages	5	5	6	6	5	6	12	5+3	
Commissioning date	1956	1961	1962	1960	1963	1967	1972	1971/72	
Running time to end 1969	77000	58000	40000	60000	51000	11000			hours

power plants in operation or proposed. There is currently much interest in gas turbines for shore-side stations because their high temperature waste heat removal is more suitable for use with dry cooling towers. (F6) This interest is being manifested in increased research and development on gas turbines and associated heat exchange equipment. The special properties of gas turbines pertinent to their use in closed-cycle systems are listed in Table 2.11. These properties help make this system attractive for both marine use and for use with nuclear power. Table 2.12 lists the advantages of coupling a nuclear reactor with a gas turbine.

2.6 SELECTION OF POWER PLANT FOR FURTHER EVALUATION

The direct cycle gas-cooled fast breeder reactor system has been proposed for central station power plants. (D3, G6) Many of the advantages that can accrue for land based plants can also benefit mobile power plants, and potentially to an even greater degree. There are at least three important criteria involved in selecting a particular type of power plant for a ship: economics, safety and weight and size.

Economics analyses of nuclear plant performance are usually broken down into three areas: capital cost, operating cost and fuel cycle cost. The operating cost of a GCFR can be considered as being approximately the

TABLE 2.11 SPECIAL PROPERTIES OF CLOSED CYCLE PLANTS

1. Load-regulation by pressure level variation.
2. High and constant efficiency over wide power-range with simple cycles.
3. Constant temperatures at all loads (low thermal stresses).
4. Turbo-machinery blading working at constant design point and elevated Reynolds numbers.
5. Low pressures combined with high temperature service.
6. No regulating mechanism (valves) at elevated temperatures.
7. Small dimensions of rotating machinery and heat exchangers.
8. Low or no cooling water requirement.
9. Free choice of working mediums, e.g. air, helium, CO_2.
10. Possibility of using new materials for components (No oxidation in inert gases.)
11. Clean cycle, no contamination or fouling.
12. Can use all fuels: gas, oil, nuclear, solid.
13. High unit outputs possible.
14. Direct waste heat utilization at elevated temperatures for heating purposes without affecting power cycle efficiency.

Ref. K4

TABLE 2.12 INCENTIVES AND PROPERTIES OF HIGH TEMPERATURE REACTORS COMBINED WITH CLOSED CYCLE GAS TURBINES.

1. Simple one-loop cycle with corresponding simple plant layout.
2. High temperature operation with high efficiency.
3. Constant efficiency over a wide power range.
4. Simple load variation by pressure level change in the common reactor cooling and working cycle.
5. Inherent safety of system. No danger of explosion.
6. Choice of appropriate inert gas.
7. Absence of oxygen allows use of new very high temperature reactor materials with high stress properties at elevated temperatures.
8. No corrosion or fouling problems.
9. Small dimensions of components due to elevated pressure and improved heat transfer.
10. Small number of auxiliaries.
11. No additional circulators for reactor.
12. Small space requirement.
13. Small, medium and large unit output possible.

Ref. K5

same (or slightly less, due to its amenability to automation) as any other reactor system and only slightly higher than a fossil-fueled propulsion system (due, for example, to added insurance costs and a higher level of operator training). Therefore the prime advantages or disadvantages are in capital and fuel cycle costs.

Capital costs are higher for nuclear systems than fossil systems and breeders are more costly than light water reactors using the same cycle. (E1) The capital cost component of the cost of power generated is very large with a nuclear power plant. Three factors reduce the capital cost component of a nuclear propulsion unit. First, the basic system design itself must be chosen to reduce capital cost. This motivates selection of a simple direct cycle, as there are no intermediate coolant loops. Secondly, power plant utilization must be high. This is now possible due to the development of ships having faster loading and unloading capabilities. Thirdly, higher ship shaft-horsepower has enabled economy of scale to become a more important factor in propulsion plant cost. The need for higher shaft-horsepower has reduced the dollar per shaft-horsepower cost for nuclear. Of the nuclear systems proposed for ship propulsion, the LWR's have lower capital costs than the FBR's. Of the FBR's it appears that the direct cycle GCFR has the lowest capital cost.

Capital cost is only one factor in the total cost assessed against a propulsion plant; the other major factor is fuel cost. As discussed previously nuclear fuel costs are lower than fossil fuel costs. Current predictions, however, show the LWR being supplanted by the FBR on the basis of the even lower fuel cost for the FBR. Due to its high breeding ratio (1.48) as a result of a hard spectrum, the GCFBR appears to have better fuel cycle costs than the LMFBR, which has a lower breeding ratio of about 1.3. These low fuel cycle costs combined with the lower capital cost possible with the direct cycle make the GCFBR direct cycle particularly attractive for assessment.

There are no weight estimates made for the GCFBR direct cycle. But the improving PCRV technology and the simple cycle indicate a reasonable if not light weight for this system relative to other propulsive units. The smaller size means less volume is occupied by the propulsion system leaving greater volume for cargo.

The major safety analysis aspects of the GCFBR will be discussed in the final sections of Chapter 3. There are some difficult engineering problems to be solved for the depressurization or loss of coolant accident and water flooding of the core, but these seem to be solvable problems.

The LMFBR has a different safety problem, and one that practically precludes its use at sea. Liquid metals are chemically reactive with air and water. There is no practical solution to this problem. LWR's also have safety problems, but none appear to be as severe. This is in part due to the large design and development effort that has gone into shoreside LWR's. With a similar land based design and development effort, FBR safety problems may also be relegated to be a lesser consideration in the long run.

The prime reason for designing a helium-cooled direct cycle system lies in its economic potential. It appears that, if a system of this type can be made technically sound, it can be economically viable. A proposed closed-cycle system design with these objectives in mind is described and analyzed in Chapter Three. Some of the obstacles to achievement of a technically sound design are also discussed, and the proposed design incorporates some suggested solutions to the engineering questions posed.

CHAPTER III

PROPULSION PLANT DESIGN

3.1 INTRODUCTION

As discussed in Chapter 2, the direct gas-cooled power cycle was selected as the basis for the propulsion plant design. Although a detailed system optimization would be necessary to arrive at the lowest power generating cost, the plant components were independently optimized by Starkus in Reference S4, to achieve a low capital cost system, albeit one with a low cycle efficiency, as discussed later in this chapter. Likewise, the reactor thermal hydraulics have not been completely optimized, but GGA studies (F1, D6, M7, W1) were used to keep the design within a near-optimum envelope. The primary effort in the present thesis was expended in achieving a long reactor core lifetime with reasonable control requirements: a task identified as one affecting the basic feasibility of the concept, and hence having higher priority than optimization.

The thermodynamic cycle and the design constraints used to determine the state points are discussed in section 3.2

The reactor design and the analytical methods used to determine the core parameters are presented and discussed in section 3.3

Safety problems, as a special design constraint, are discussed in section 3.4

3.1.1 BACKGROUND

Before proceeding directly to detailed design considerations, there are several "customer related" specifications which must be established for the proposed design. These relate to plant rating requirements and compatibility with ship design.

In Chapter 2 it was shown that power plant ratings were rapidly increasing; one hundred and twenty thousand shaft horsepower ships are now on order. In the early 1980's, when a shipboard GCFBR could become operable, 200,000 shp ships may be required. It was therefore decided to base the present design on this rather large unit size. It is interesting to note that the thermal rating of this size plant (560 Mw(t)) is roughly comparable in size to the GCFBR "demonstration" plant proposed for central station application by GGA (824 Mw(t)). Thus it may be possible to have the shipboard plant serve both functions and thereby obtain A.E.C., Utility and Maritime Administration support for this new concept.

In choosing the potential application, it was apparent that the pragmatic approach of insuring a broad spectrum of compatible ship types was in order. Thus, satisfying the most restrictive application was considered as a goal. New ship types have been proposed for future use employing separate cargo and propulsion sections (T3). This composite type ship places severe weight and location constraints on the power plant. This type ship would appear to be the most difficult to outfit with a nuclear power plant from the standpoint of weight, volume, and location. (On the other hand, nuclear propulsion could also be ideal for this type ship, due in part to its almost constant weight over voyage length). Provision of maximum flexibility for total power system arrangement also led to selection at the outset of electric drive.

Within this broad envelope, a considerable number of other design decisions were made, as will be discussed in subsequent sections of this chapter.

3.2 CYCLE DESIGN PHILOSOPHY

3.2.1 Selection of Working Fluid

As discussed in Chapter 2, many different fluids have been proposed for use in gas-cooled reactor designs. Helium has been chosen for this design. Nuclear, thermal, mechanical, and metallurgical design of the reactor core are greatly influenced by the type of coolant gas used. The choice of a primary coolant working fluid affects not only the cycle efficiency and cost, but also the design of all major components.

For a direct cycle application certain gases can be immediately eliminated from consideration as they do not meet the requirements necessary for good nuclear or thermal design; such as sufficient chemical and radiolytic stability under irradiation. In particular, the gas must not be corrosive to the fuel cladding, the hottest exposed surface, and one which must withstand high burnup. If clad integrity is maintained then little radioactive contamination of the turbomachinery should occur. This is important in order to permit maintenance and inspection of turbomachinery. Air as a working fluid can be eliminated due to Argon-41 activation (1.83 hr half-life) and to oxidation problems. Of the three other gases which have been used in reactors: nitrogen, helium, and CO_2, nitrogen can also be eliminated: although it was used in ML-1, it has demonstrably poorer properties than CO_2 or helium and can lead to deleterious nitriding.

The choice between CO_2 and helium is a difficult one to make. The major criteria are chemical stability at high temperatures and heat transfer effectiveness and pumping power.

Table 3.1 shows the properties of Helium and CO_2 near the required operating conditions. The prime reasons for not using CO_2 are its reaction with stainless steel at high temperatures, and induced coolant radioactivity (O^{16}-N^{16} half-life 7.4 sec). A comparison of the heat

TABLE 3.1

PROPERTIES OF CARBON DIOXIDE AND HELIUM AT 500p.s.i. AND 1000°F

Gas	Molecular Weight (M)	Specific Heat (Cp) joule/g$^{\circ}$C	Prandtl No. (Np)	Thermal Conductivity (K) 10^{-3} W/CM$^{\circ}$C	Viscosity M(10^3) g/sec cm	Molar Heat Capacity (MCp) BTU/$^{\circ}$F
CO_2	44.01	1.19	0.65	0.64	0.356	52.5
He	4.0	5.2	0.675	2.94	0.382	20.8

Ref. M5

transfer effectiveness and pumping power requirements can be found in References E3, M4, D5, and L4. Helium has the advantage of a greater heat transfer effectiveness leading to smaller heat exchangers than CO_2. However, CO_2 requires less pumping power and usually smaller turbo-machinery than Helium. These factors directly affect the design of the system from a size standpoint. With Helium, smaller heat exchangers are possible, assisting in the integrated design, while the increased length of the turbo-machinery makes an integrated design more difficult. Table 3.2 shows a comparison of CO_2 and Helium at a pressure of 588 psi and a gas temperature of 600°C at the channel hot point.

CO_2 can deliver approximately 20 percent more power at a fixed flow area than Helium, but for a fixed heat transfer area about 10 percent less. The prime ther-modynamic factors in favor of Helium are: CO_2 requires two to three times the pressure drop to do the same job; second, CO_2 requires a greater heat transfer surface area for a given flow area. The above reasons, coupled with its growing use in present technology make helium appear to be the best overall choice.

3.2.2 Brayton Cycle vs. Steam Cycle

The Brayton or direct cycle gas-cooled power cycle is much simpler in design than the indirect steam cycle

TABLE 3.2 COMPARISON OF CO_2 AND HELIUM GAS COOLANTS

	Ratio of CO_2 Value to He Value	
	Smooth Surfaces	Rough Surfaces
Core pumping power	1.49	1.58
Core flow area	0.69	0.75
Core gas force	2.6	3.0

Ref. M5

proposed for gas-cooled fast reactors. This simplicity should lead to a lower weight, as the steam generators and their attendant circuit are eliminated; one should also note that cost is roughly proportional to weight. It has been estimated for a 100,000 shp plant the helium turbomachinery would weigh 33 percent of the steam turbine. (S4)

The Gulf General Atomic (GGA) demonstration fast breeder plant has a reactor outlet temperature of 1010°F (543°C). (F1) With the use of Vanadium clad, reactor outlet temperatures could be raised to 1328°F (720°C). With ceramic coated particles it is believed reactor outlet temperatures of 1706°F (930°C) would ultimately be possible. These temperature increases would be highly favorable to a helium Brayton cycle, since the efficiency of the Brayton cycle increases with reactor outlet temperature faster than does the Rankine cycle. Higher pressures and temperatures in a steam system put much greater demands on the materials, and it appears that approximately 1000°F steam is about the optimum for present and near-term steam plant materials. Higher temperatures require more expensive turbine and superheater alloys, raising the cost significantly. Steam at low quality, as found in the last turbine stages, is also highly corrosive and erosive, and it appears steam turbomachinery will require more maintenance than the closed cycle helium machinery.

The final advantage of the helium cycle over the steam cycle is that it rejects heat at a much higher temperature.

A smaller waste heat rejection exchanger (i.e., precooler) can be used with helium. This high temperature rejected heat can be used to provide steam for the ship's auxilliary services.

3.2.3 Cycle and Pressure Ratio Selection

The basic direct cycle is shown in Figure 2.3. There are many variations on this cycle which may be proposed to achieve a design optimum for a particular application. The design optimum for ship propulsion would be that overall package which yielded the lowest cost for freight transport. This requires consideration of the combination of the capital, operating and fuel costs for the propulsion plant. There are two design extremes deserving of analysis in a Brayton cycle plant.

The cycle can be designed for very high thermodynamic efficiency or the designer can aim for a compact, simple plant with a moderate efficiency. As GCFBR's are predicted to have fairly insensitive fuel costs as a function of efficiency (L4), it pays in the present application to opt for a simple plant with moderate efficiency.

The reason for the weak dependence of fuel cost on thermodynamic efficiency is attributed by the authors of Ref. 24 to the fact that fuel burnup is balanced by fuel production (breeding) and the major costs incurred are carrying charges on the vessel inventory. See Figure 3.1. Since a major stumbling block to the use of GCFBR's and LMFBR's is their high capital cost, simplifying the system and reducing the capital cost is an obvious economic strategy. To reemphasize a previously discussed fact, a lighter

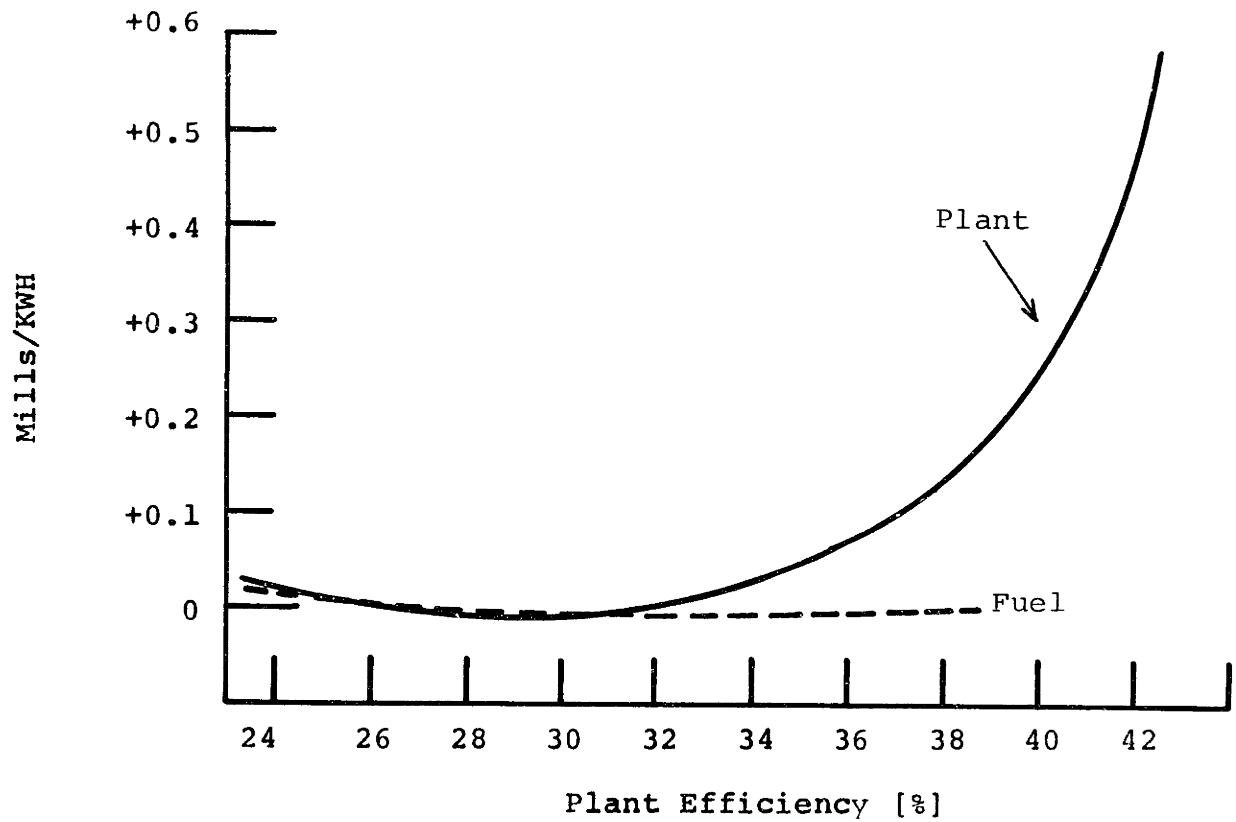


Fig. 3.1 Typical Differential Power Generation Cost for GCFBR

Ref. 12

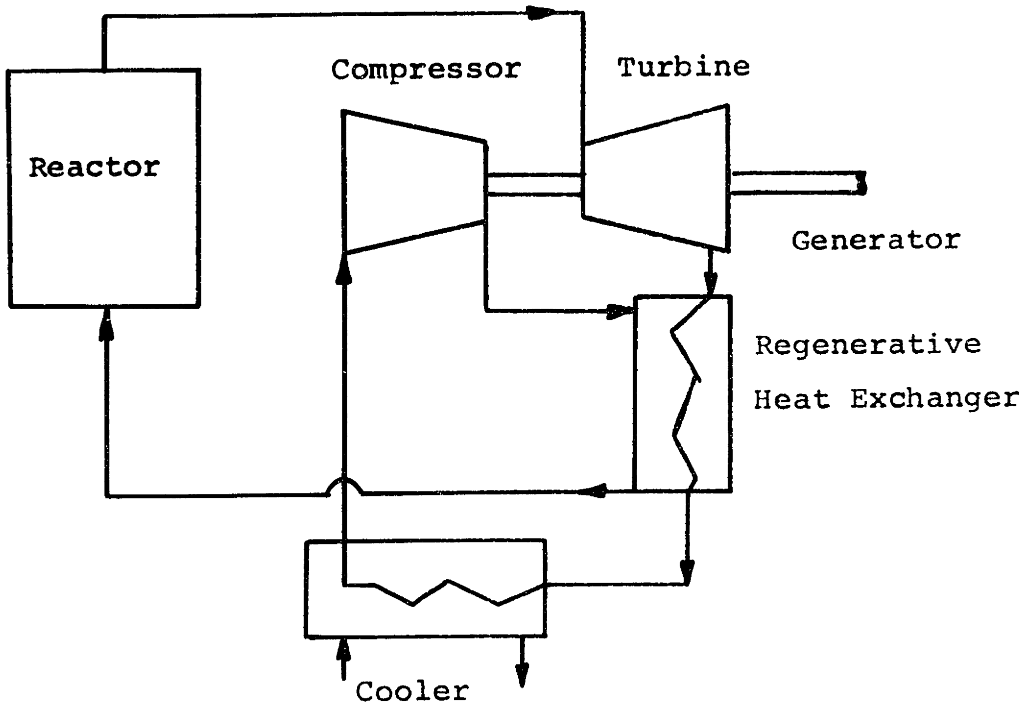
more compact plant is an additional advantage for a shipboard plant, which will also ultimately be reflected in economic advantages in terms of increased cargo revenue.

In keeping with the philosophy of minimized cost and simplicity, a single-shaft turbomachinery system was chosen over a double shaft. See Figure 3.2 for a comparison of system layout. The two-shaft arrangement eases startup, but for the present application start-up frequency would be low.

The maximum cycle temperature is determined by materials limitations of the reactor fuel cladding, and the maximum pressure is set by reactor pressure vessel technology. The maximum fuel clad temperature limitation sets a maximum reactor outlet temperature of about 1200^{0}F (649^{o}C). Pressure level must be selected according to the maximum pressure deemed safe in current pressure vessels. PCRV pressure vessles have been operated in nuclear service at pressures of 610psig. Swedish investigators are designing PCRV vessels in BWR's operating at a pressure of 1250 psia (M8). Calculations done by Starkus (S4) show the thermodynamic optimum as being too high (i.e., 2,500psia) for current PCRV pressure vessel technology, although steel vessels for current PWRs have design pressures of 2,000 psia.

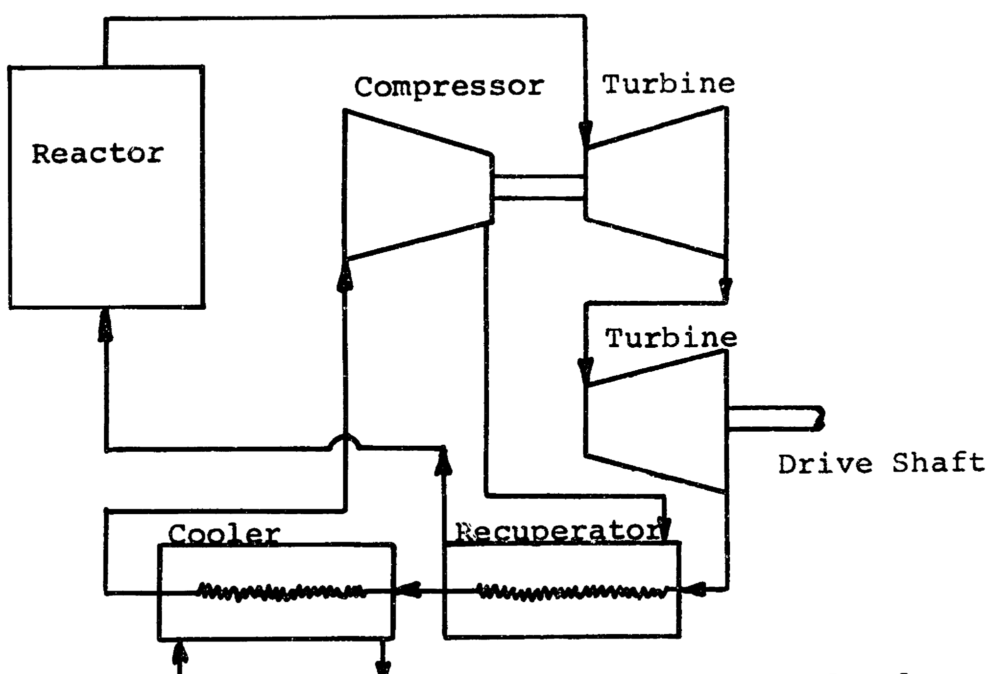
Heat exchanger size can be varied in order to obtain optimum thermal efficiency. It is questionable, however, that the higher thermal efficiency leads to the lowest generating cost. The optimization of the regenerative heat exchanger

Fig. 3.2 COMPARISON OF SHAFT SYSTEMS



SINGLE SHAFT

DOUBLE SHAFT



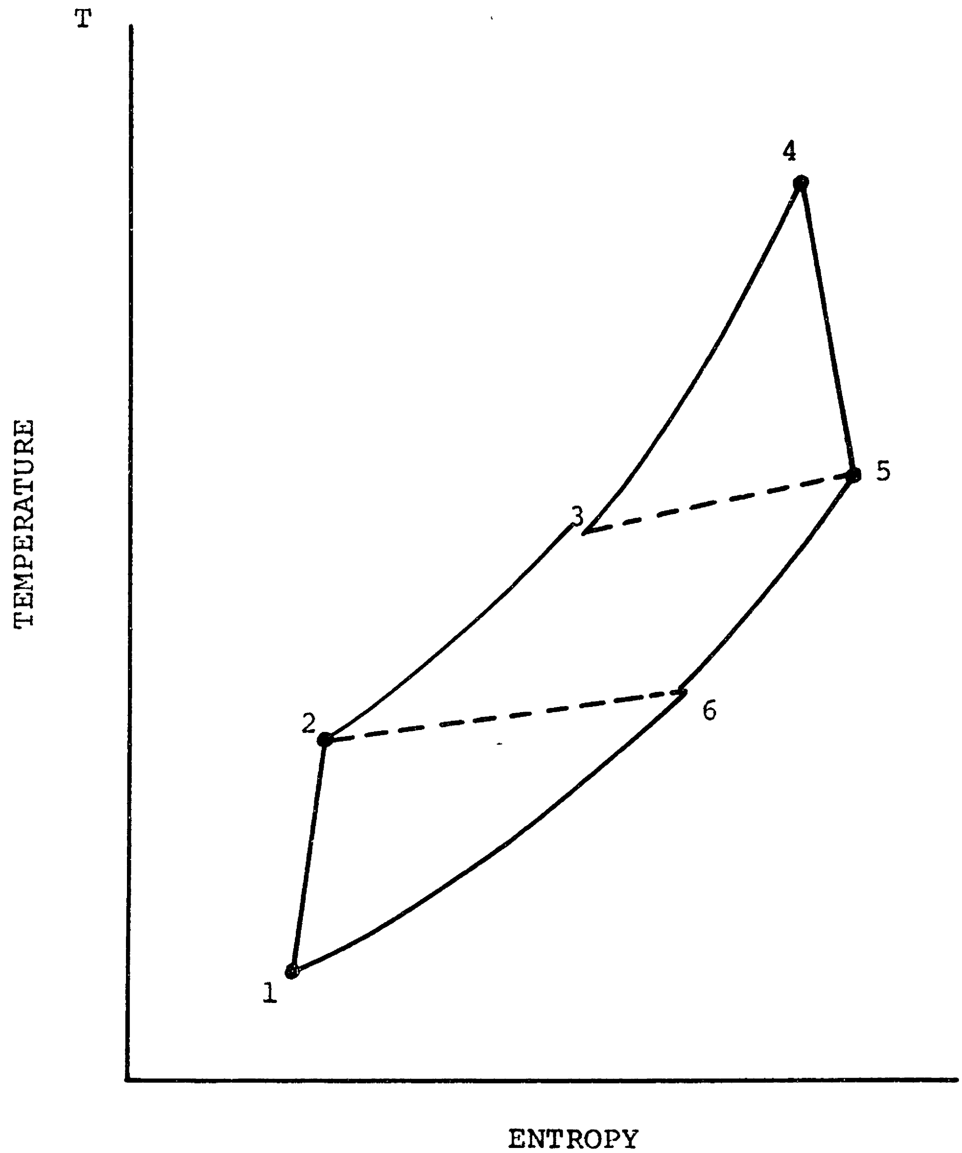
Ref. K1

effectiveness was accomplished in reference S4 and was found to be 83 percent. No intercooling is to be used. The regenerator effectiveness was set to insure a reactor inlet temperature maximum of 750°F. The pressure ratio was set at 2.32, leading to an overall plant efficiency of 26.5 percent. This, according to Reference S4, gives reasonable turbomachinery blade sizes and stresses. The cycle temperature-entropy diagram is shown in Figure 3.3. The plant layout is presented in Figure 3.4

Other design constraints accounted for in the system are integrated vs. non-integrated design and control requirements.

Integrated design places all the helium containing components inside the prestressed concrete reactor vessel (PCRV) with only a rotating shaft carrying the power out. Maintaining all the major helium components in the PCRV lessens considerably the possibility of a loss of coolant accident. A major design decision has to be made in whether to use a multicavity PCRV in an integrated design or a PCRV solely for the reactor. PCRV's can be made large enough to hold the entire Helium-containing system. This option provides more, hence heavier, shielding and structure than appears necessary. Since it is the more conservative approach, however, it may be the more acceptable alternative for initial designs. In the long run, it would appear that a PCRV vessel solely for the reactor is all that is necessary, together

Fig. 3.3 Temperature-Entropy Diagram for Shipboard Direct Cycle Nuclear Power Plant

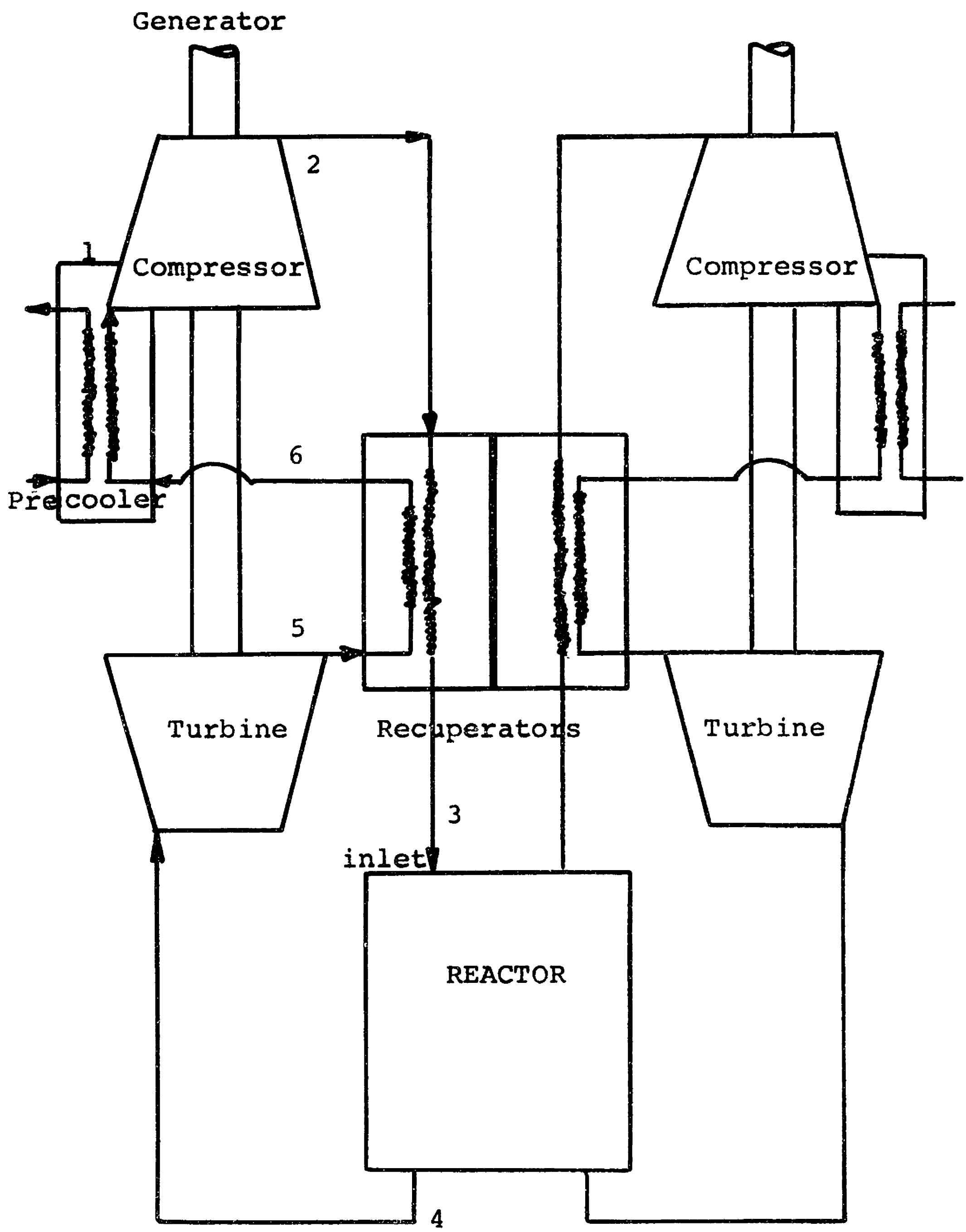


1. (113°F, 561 psi - 1.58 BTU/lbm °R)
2. (379°F, 1300 psi - 1.52 BTU/lbm °R)
3. (753°F, 1284 psi - 1.07 BTU/lbm °R)
4. (1220°F, 1223 psi -.653 BTU/lbm °R)
5. (830°F, 575 psi - .593 BTU/lbm °R)
6. (457°F, 568 psi - .964 BTU/lbm °R)

ENTROPY based upon 0°F and 14.5 psi

Ref. S4

Fig. 3.4 PLANT COMPONENT LAYOUT



with a lower design-pressure secondary containment to provide back pressure and contain a reactor loss of coolant accident. This will be discussed in Section 3.4.

With the non-integrated direct nuclear Brayton cycle, there are problems with developing quick-acting valves to isolate the reactor in the event of a pipe break. Gasparovic, in Reference G7, has cited the nearly impossible speeds with which isolation valves must respond in order to be effective.

To meet ship demands for minimal supervision, load following and simplicity, the control system must be taken into account at the early stages of the design. Of the possible control schemes proposed for the direct cycle system, the helium mass flow bypass system (B2) for controlling power is simple and maintains thermal efficiency closely constant throughout the power range. This system will be further discussed in the section under safety.

3.3 REACTOR CORE DESIGN

3.3.1 Introduction

The core design for the shipboard GCFBR is based in part on proposed GGA designs for a 300 Mw(e) demonstration plant and a 1000Mw(e) commercial plant (F1,W1). The following section describes the important characteristics of the GGA designs.

Section 3.3.3 then discusses the design method used to develop the present core configuration, and the major design assumptions. Section 3.3.4 is a summary of the design features determined by the analytical methods, satisfying constraints placed by thermal-hydraulics, materials, and safety considerations.

3.3.2 Background on GGA - Proposed GCFBR's (F1)

The Gulf General Atomic Company is developing an indirect cycle gas-cooled fast reactor as an alternative to the LMFBR. In an effort to gain maximum advantage from the LMFBR development program, GGA is initially hoping to apply LMFBR fuel and clad developments while independently developing other components necessary for gas-cooled reactors. Table 3.3 presents the basic design data for the 300 Mwe Demo plant recently proposed. The fuel clad for this design are conservatively rated from the standpoint of European designs. The Europeans are investigating Vanadium-clad fuels, cermets and ceramic coated particles which are capable of considerably higher temperatures than the GGA-proposed 1000°F.

The current GGA designs are for a indirect cycle, and therefore the component arrangement is not laid out appropriately for a direct cycle system. The GGA design for the 300 Mw(e) system is approximately 50 percent larger in terms of equivalent thermal rating than the proposed shipboard

Table 3.3

300 MW(e) GCFR DEMONSTRATION PLANT DATA SUMMARY

GENERAL	
Average breeding rate	1.33
Maximum fuel burnup, MWd/Te heavy metal	100,000
Net electrical power, MW(e)	311
Plant efficiency, %	37.6
Steam conditions at main turbine	
Throttle pressure, psia	1223
Throttle temperature, °F	922
Condenser pressure, in Hg absolute	1.75
Reactor coolant	Helium
Reactor coolant pressure, psia	1250
Reactor vessel and primary containment	PCRV
PCRV dimensions, ft.	84 diam by 71 high
REACTOR	
Reactor geometry	
Core height, in	39.2
Core length-to-diameter ratio	0.5
Axial blanket length, each end, in	17.7
Core volume fractions, %	
Fuel	30.1
Helium coolant	44.6
Cladding	10.0
Structure	6.0
Caps (box interspace, control rod channel)	9.3
Reactor heat transfer	
Helium temperatures	
Reactor inlet, °F (°C)	593 (312)
Mixed mean outlet, °F (°C)	1007(541)
Average Power density, kWt/liter of core	238
Maximum linear rating (10% overpower), kW/ft	13.8
Hot-spot cladding temperature, °F (°C)	1290 (700)
Radial maximum-to-average	1.30
Axial maximum-to-average power ratio	1.20
Rod surface roughening	
Fraction of average core length roughened,%	75
Roughening heat-transfer multiplier	2
Roughening friction-factor multiplier	3
Maximum heat flow, Btu/(hr) (ft^2)	520,000
Core and axial blanket power fraction,%	95.55
Radial blanket power fraction, %	4.45

Table 3.3, continued

Nuclear characteristics, (midcycle)	
Fissile core loading (Pu),kg	1320
Average fast neutron flux (E> 0.1 MeV), n/cm^2 - sec	2.2 x 1
Reactor rating, Mw(t)/kg fissile	0.605
Doppler constant, TdK/dT (T in °K)	1
Fuel lifetime, full power days	750
Partial refueling cycle, yr	1
Fuel element	
Distance across hex flats, external in	6.642
Element overall length,in	118.25
Number of rods, standard element	271
Rod outside diameter, in.	0.282
Rod pitch triangular lattice, in.	0.386
Rod cladding material	316 SS
Cladding OD/ID	1.15
Fuel Material	PuO_2 - UO_2
Blanket element	
Number of rods	127
Rod outside diameter, in.	0.464
Blanket material	Depleted UO_2
PRIMARY COOLANT SYSTEM	
Number of loops	3 main, 3 auxilliary
Main helium turbocirculator (each of 3)	
Type	Single-stage axial
Drive	Steam turbine
Pressure rise, psi	60
Brake horsepower (per circulator)	22,300
Steam generators (each of 3)	
Type	Helical once-through
Heavy duty, Btu/hr	8.45 x 10^8
Surface area, ft^2	33,400
Feedwater temperature, °F	412
Steam outlet temperature, °F	875
Steam pressure, psi	2900

plant design.

Many aspects of reactor physics and core thermal-hydraulic design can be borrowed from the 300 Mw(e) design. The major carry-over is in the area of nuclear cross-sections, effective fuel-rod roughening coefficients and performance characteristics of the pressure-equalized fuel rods.

The design data of most use from the GGA studies were core material concentrations, fuel rod and subassembly design and control requirements. The reactivity control requirements for the 300 Mw(e) reactor core are listed in Table 3.4; they will be referred to shortly for comparison to the proposed shipboard plant.

3.3.3 Core Design

3.3.3.1 Physics Design

The proposed 560 Mw(t) gas-cooled fast breeder reactor has approximately the same core volume as the higher-rated GGA demonstration plant because of its more conservative thermal design (no clad roughening). The average fissile plutonium enrichment of approximately 22 percent is about the same.

It is important to note that the physics design is in many ways more difficult for small LMFBR and GCFBR cores. Small cores usually have high neutron leakage, low internal breeding ratios, a large reactivity swing due to burnup,

TABLE 3.4

300 Mwe Demonstration

Plant Control Requirements

$1 = 0.00351 K

		Dollars
Cold to Hot Operations		
Doppler		$1.30
Grid plate expansion and distortion		1.05
Fuel-length expansion		.60
Radial distortion (thermal bowing)		.05
	SUB TOTAL	$3.00
Reactivity Losses During the Core Life cycle		
Burnup		$8.20
Axial Swelling		.25
Radial Swelling		.58
	SUB TOTAL	$9.00
Other Allowances		
Compensation for removal of Helium		$ .40
Compensation for Np^{239} decay		.60
	SUB TOTAL	$1.00

Minimum shutdown:

One stuck rod and standard 0.01 K margin

TOTAL CONTROL REQUIREMENT	$16.85
TOTAL CONTROL CAPABILITY PROVIDED	$17.85
EXCESS CAPABILITY	$ 1.00

and hence a short refueling interval.

A survey of some of the very small fast reactors currently in operation can demonstrate the problem of obtaining long refueling intervals. The Enrico Fermi Power Plant has control sufficient for only 15.4 cents worth of burnup, necessitating an almost weekly partial refueling of the core (Y1). The Fermi Reactor is designed for 65.9 Mw(e) and 200 Mw(t) at 30.5 percent efficiency. The British Dounreay reactor at 60 Mw(t) and a load factor of 70 percent must be refueled every 20 days.

The Southwest Experimental Fast Oxide Reactor (SEFOR) is a small 20Mw(t) fast reactor built to obtain nuclear safety information on oxide core design. The refueling interval for this reactor is quoted at one and one half years, but at the extremely low load factor of 7 percent: at 70 percent it would be only two months.

It is quite clear from these figures why fast reactors have not been serious contenders for shipboard propulsion systems in the past, when 50,000 ship ratings were under consideration. No ship could bear the economic burden of such frequent refueling, either in port or through provision of an on-board capability.

Fortunately, larger fast reactors will improve considerably on the one month refueling intervals for the very small FBR's. The 300--1000 Mw(e) designs can be expected to have six months to one year (partial-core) refueling intervals at 0.8 load factors. Fuel lifetime will be 100,000 megawatt days per tonne (about 3X that for LWR's) and is set by the ability of the fuel to withstand the deleterious effects of high burnup and irradiation. Reactivity control will also be an important factor. In particular, if the safety-related constraint is imposed that no regulating control rod is to be worth more than one dollar, then one may consider the design as being reactivity limited. The one dollar per rod restriction is to prevent prompt critically from occurring in the event of a rod ejection accident; it also eases the problem of insuring shutdown with one rod stuck out. The difficulty of this problem is compounded by the fact that there are no effective soluble poisons or burnable poisons for FBR's, as there are in LWR's, to ease the reactivity control problem. In order to insure technical feasibility of the subject GCFBR reactivity control system, the objective was established that a batch burnup core must be devised which had a burnup reactivity swing less than that provided in the GGA demonstration plant.

It is essential to obtain a long batch core burnup on the order of 1000 days in order to become competitive with other fuels used for propulsion. For example, the N.S. Savannah sailed for three years on a single core loading. Because of the importance of this objective, and the demonstrable weakness of FBR's in this area, a major effort has been put into designing a long life batch-fueled core, as will be discussed in the following paragraphs.

Detailed core nuclear design procedures used state-of-the-art approaches, as follows. The ANISN multigroup code was used in the S8 option to solve an approximate one-dimensional problem using a 26 group cross section set in the ABBN format.For cross-section information see Refs. B3 & B4. The twenty-six group cross-section set was then reduced to four groups by collapsing over the calculated core neutron spectrum. This four group set was then used in the two dimensional multigroup diffusion burnup code 2DB (L1). The Energy structure of the 4 group set is shown in Table 3.5.

The 2DB Code was used to determine K effective, flux distributions, power densities, and material inventories over a 1000 day burnup period. A number of cases were run, varying material concentrations (i.e., enrichment), core zone configurations and physical size to determine the base case or "regular" core. The "regular" core was a uniform enrichment core whose material enrichment and physical configuration appeared reasonable from the physics and engineering viewpoints. Table 3.6 shows the major parameters of the regular core design and Figure 3.5 shows the core configuration. Note that the core volume and fissile inventory (3080 l and 1388 kg) are quite close to the GGA demonstration plant values in Table 3.4 (3460 l and 1320kg). A major core length-to-diameter ratio of 1.0 in the present

TABLE 3.5

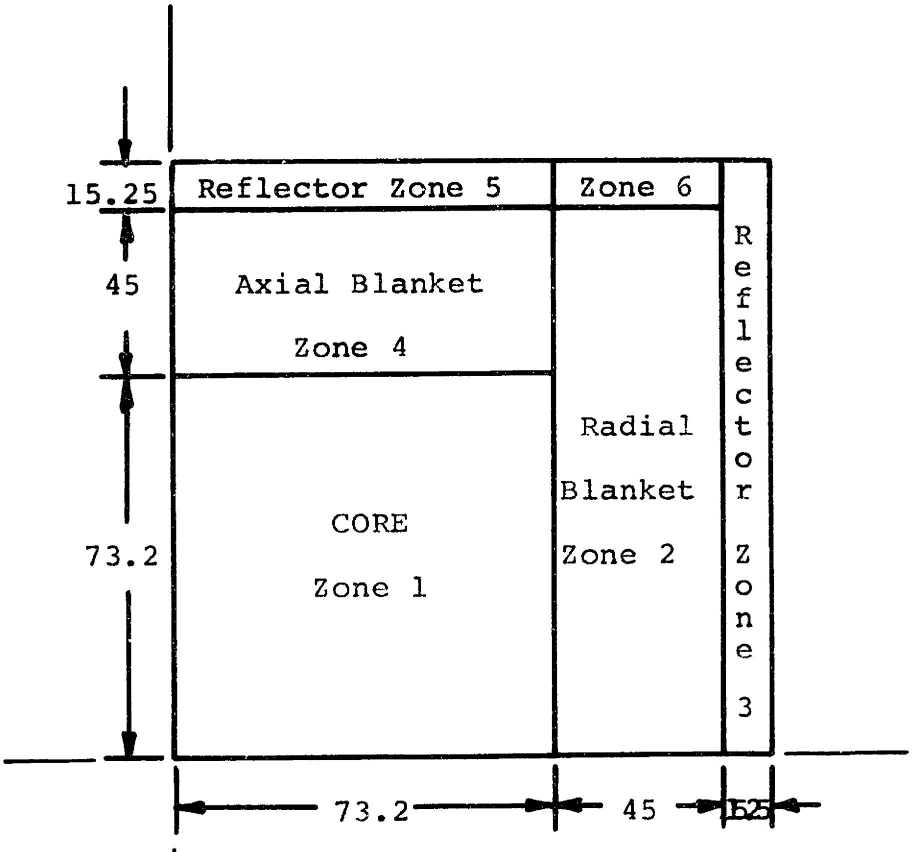
Collapsed Group's Energy Structure

Energy Range (MeV)	New 4 Group Set Group No.	26 Group Set Group No.
0.8-10.5	1	1-5
0.4-0.8	2	6
0.1-0.4	3	7.8
0.025-100(Kev)	4	9-26

TABLE 3.6

Proposed "Regular" Core Characteristics

General	
Reactor Power, Mw(t)	563
Shaft horsepower, (shp)	200,000
plant efficiency, (%)	26.5
Reactor Geometry	
Core Height, cm (in)	146.4 (57.7)
Core length-to-Diameter Ratio	1.0
Axial Blanket length, each end, cm (in)	45 (17.72)
Core Volume, l	3,080
Core Composition, Volume %	
Fuel	30
Helium Coolant	60
Cladding and Structure	10
Axial Blanket Composition, Volume %	
Fuel	30
Helium Coolant	60
Cladding and Structure	10
Radial Blanket Composition, Volume %	
Fuel	50
Helium	36
Cladding and Structure	14
Fissile enrichment, %Pu-239+241	22
Beginning-of-Life Core	
K effective	1.182
Fissile Pie Loading, Kg.	1388.4
End-of-Life Core (1000) days	
K effective	1.135
Fissile Pie Loading	1298.4
K over life	0.047
reactivity in dollars	13.20



NOTE: All Dimensions in CM.

Fig. 3.5 QUARTER CORE DIMENSIONS OF REGULAR CORE design, as is opposed to the flattened-core value of 0.5 in the GGA design.

A rather simple relation (D8) can be used to estimate the refueling interval for a burnup of 100,000 magawatt days per tonne:

$$d = \frac{B}{LPEN} \qquad (3.1)$$

where B = burnup, Megawatt days per Metric tonne (100,000)

L = System load factor (0.8)

P = specific power rating, KW/Kg fissile (400)

E = fractional enrichment (0.22) and

n = fuel reshuffles per cycle (1.0 for a batch core)

Values for the proposed regular core (shown in parentheses above) yield a refueling interval of approximately 1400 days. Thus, considering only materials limits the core can meet our design requirements. However, this core exceeds the known-to-be achievable reactivity control value based on GGA's design. For a calculated burnup of approximately 1000 days a change in k of 0.047 occurred, corresponding to a 13.20 dollar change in reactivity. This can be compared to the $8.20 control for this purpose provided in the GCFBR demonstration plant, which, it should be noted, reduces the reactivity swing by employing partial refueling at one year intervals, a solution not open to us. In addition to the high reactivity swing over life, the core power peaking factor (maximum-to-average power density ratio) is rather high: 1.79 at the beginning of life (and essentially the same at 900 days). Beginning and end of life core radial and axial flux distributions are shown on Figures 3.6 and 3.7

These two inherent defects strongly indicated that some other type of core arrangement was necessary. The concept adopted to avoid these problems involved use of a central blanket region in the core as shown in Figure 3.8. The core with the central axial blanket has been named the "Parfait" core. The effect of this central blanket is two-fold. Firstly, it decreases the core reactivity swing over life by allowing Plutonium concentration to increase substantially with time in the high worth central region. Secondly, power peaking factors are reduced both axially and radially.

A second set of iterations was performed to evaluate the effect of creating a central axial blanket region in the regular core. The effect achieved was to decrease the control requirement to K=0.027, or approximately 7.72 dollars of reactivity. This is a decrease of $5.48 over the single zone core and is 50 cents less than the GGA multiple-zone core. The refueling interval according to Eq. 3.1 remains at around 1400 days.

The power peaking factor for the regular core of 1.79 has been reduced to 1.51. This is slightly less than the GGA demonstration plant, which has a peaking factor of

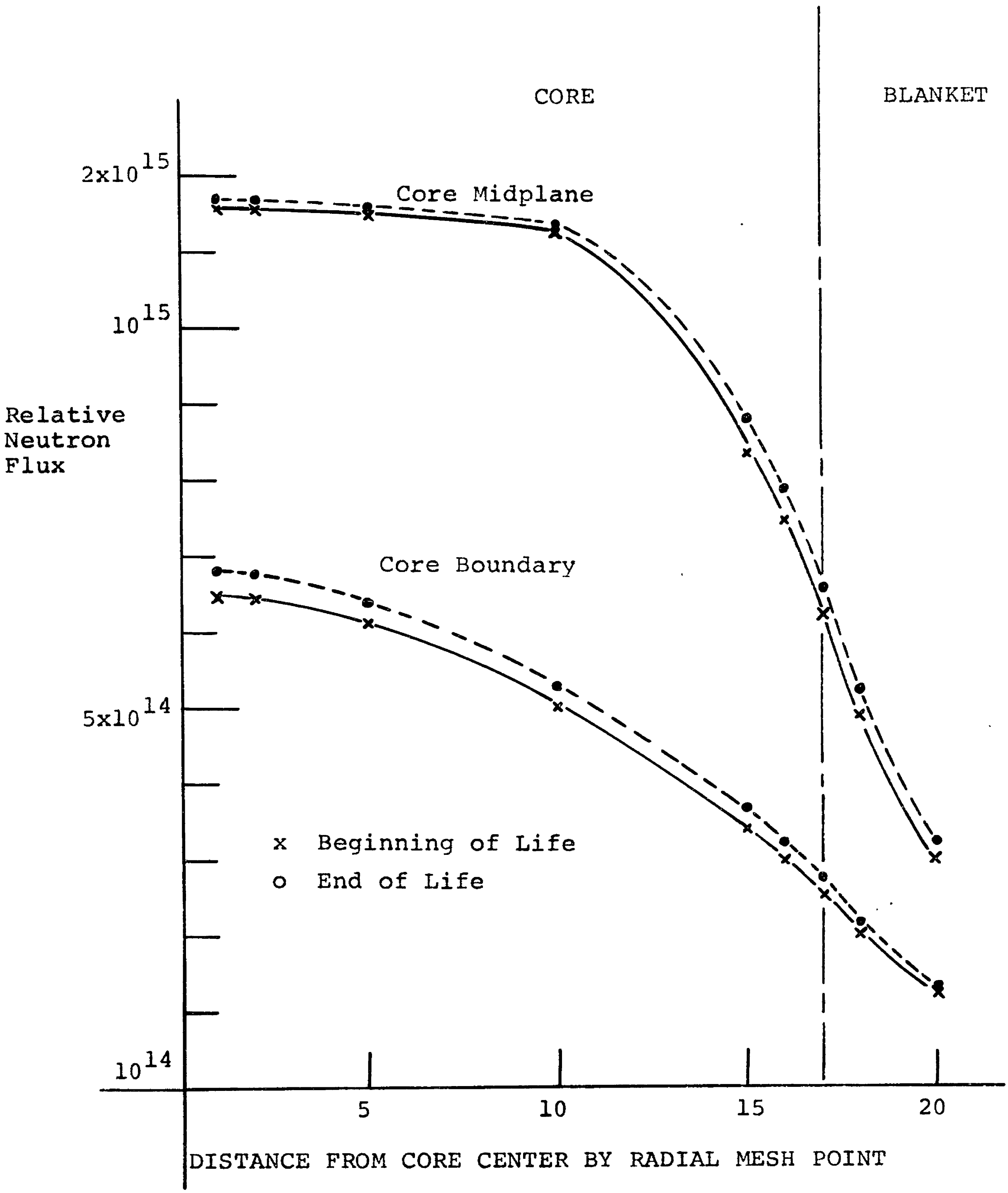


Fig. 3.6 Regular Core, Radial Flux Distribution

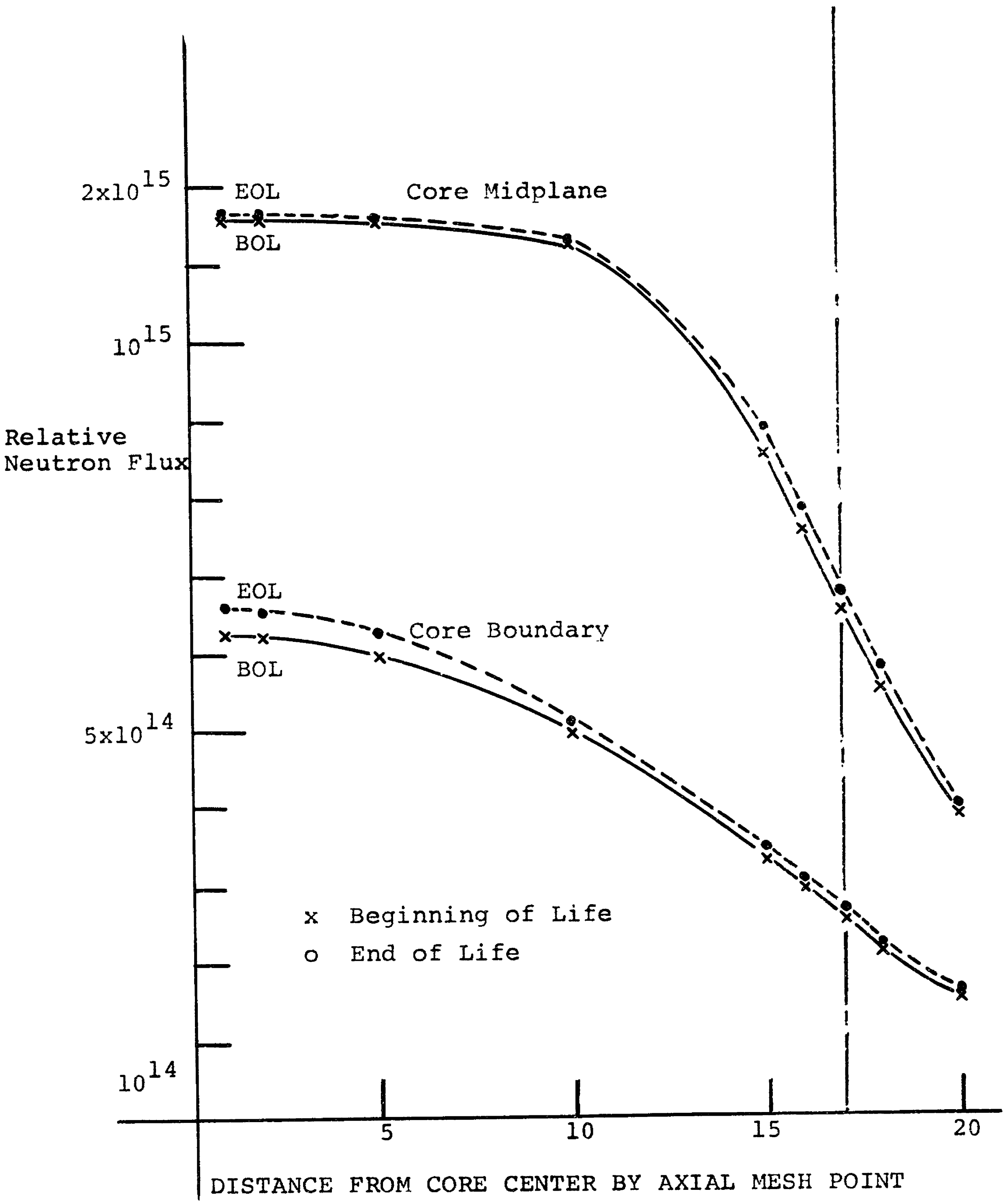
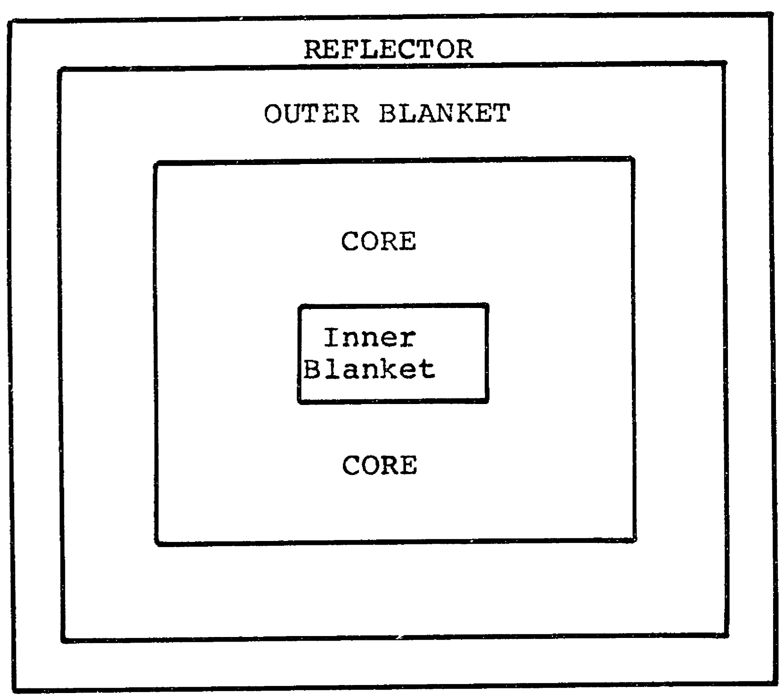


Fig. 3.7 Regular Core, Axial Flux Distribution

Schematic Of

Fig. 3.8 CORE WITH CENTRAL BLANKET



1.56. This is at the beginning of life which is the worst case. Figure 3.9 and 3.10 show the radial and axial flux for both the beginning and end of life Parfait Core.

Several cases varying the size of the internal blanket were investigated. That discussed herein was the best achieved in the present work. Further optimization may result in further small reductions in both the peaking factors and the control requirements. This can be done by optimizing the axial thickness of the blanket, its radial diameter and perhaps by varying initial inner blanket enrichment. One should also note that the above power peaking factors are for the clean core condition (no control poison) and can therefore be improved by control rod programming.

The proposed Parfait core configuration is shown in Figure 3.11 and Table 3.7 presents the Parfait core parameters proposed as the basis for the shipboard GCFBR design. The core region is surrounded by a radial and axial blanket of depleted Uranium. These blankets are 45 cm. in thickness and are themselves surrounded by a Beryllium Oxide reflector 15.25 cm. thick. The Beryllium Oxide reflector is used to improve breeding in the blanket and to reduce the neutron flux incident on the PCRV liner.

A comparison of the regular core and the last two iterations on the Parfait type core is shown in Table 3.8. The Parfait core No. 9 showed the best Plutonium production and the lowest fissile inventory. It also has the lowest reactivity swing over life and is presented as the proposed design.

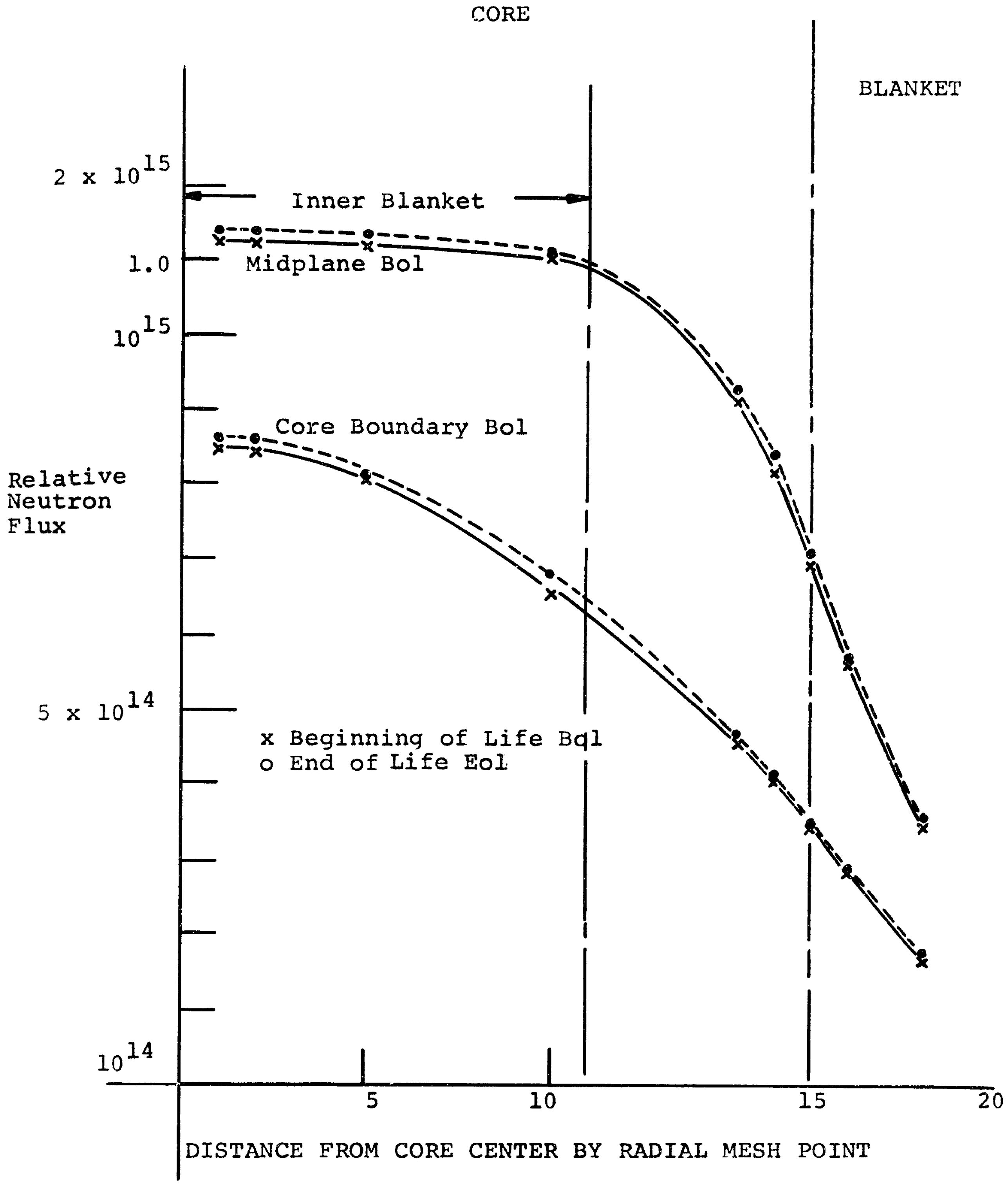


Fig.3.9 PARFAIT CORE, RADIAL FLUX DISTRIBUTION

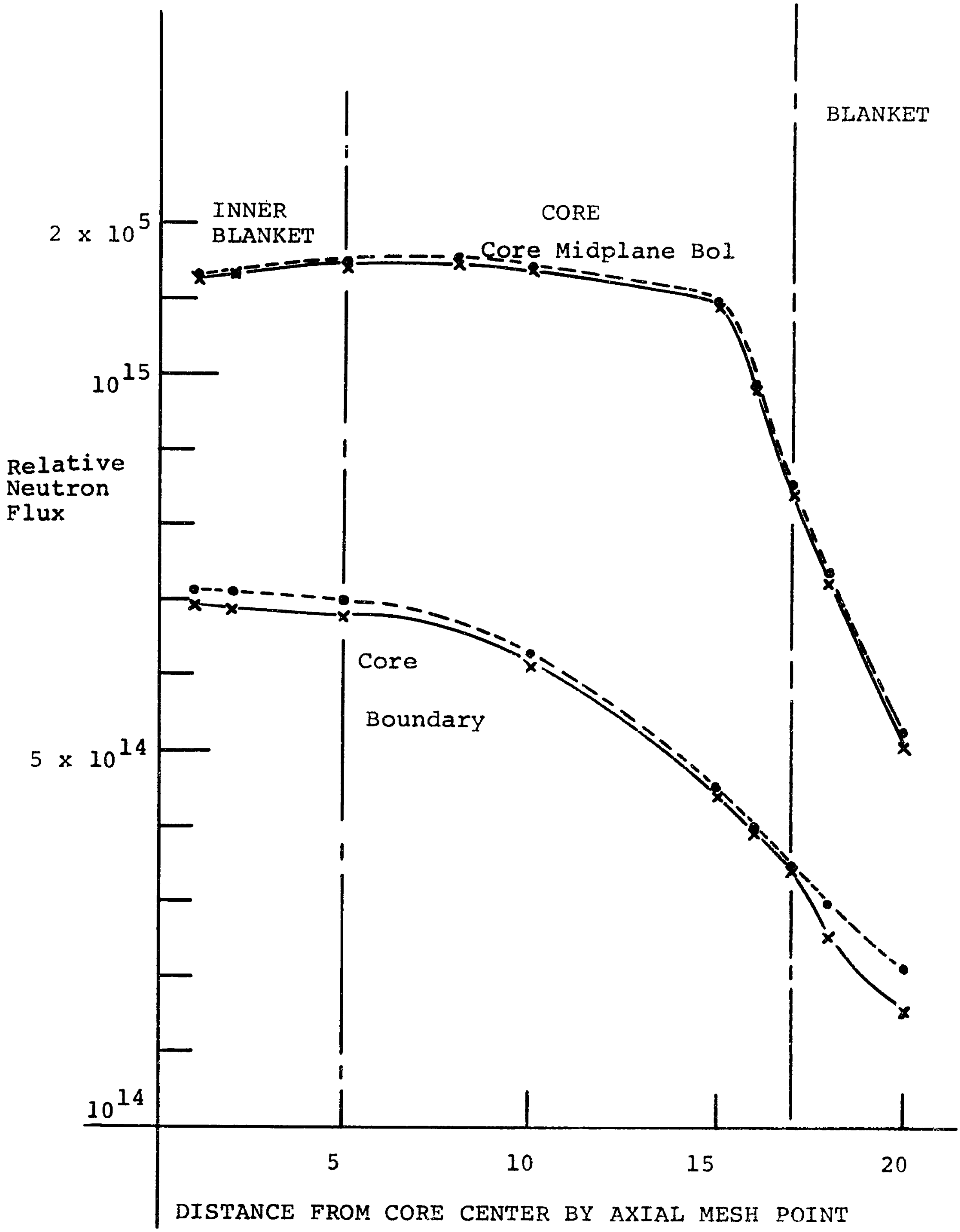


Fig. 3.10 PARFAIT CORE, AXIAL FLUX DISTRIBUTION

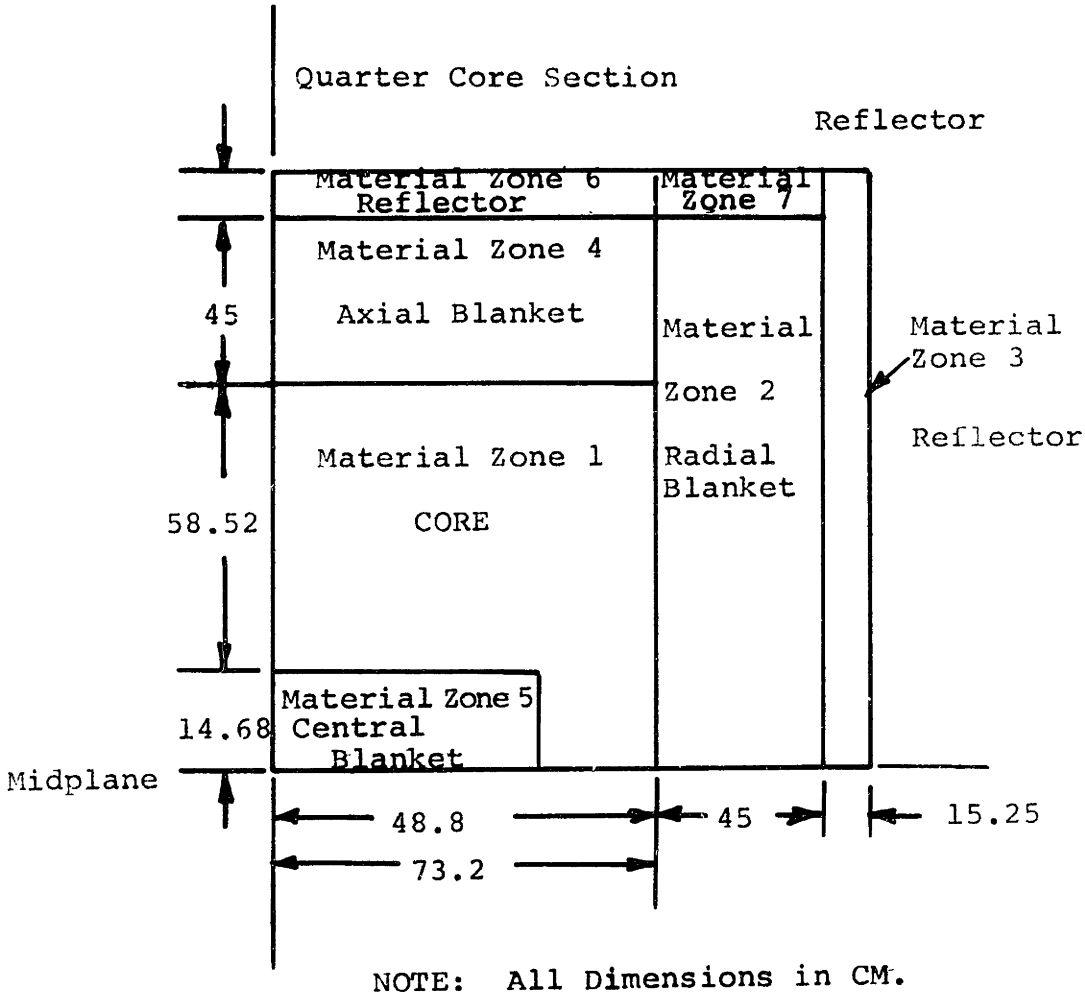


Fig. 3-11 PARFAIT CORE CONFIGURATION

TABLE 3.7

Proposed "Parfait Core" Characteristics

General	
Reactor Power, Mw(t)	563
Shaft horsepower, shp	200,000
Plant efficiency, %	26.5
Reactor Geometry	
Core Height	146.4 (57.7)
Core Length to Diameter Ratio	1.0
Axial Blanket length, each end, cm (in)	45 (17.72)
Core Volume	2720
Core Composition, Volume, %	
Fuel	30
Helium	60
Cladding and Structure	10
Axial Blanket Composition, %	
Fuel	30
Helium	60
Cladding and Structure	10
Radial Blanket Composition, %	
Fuel	50
Helium	36
Cladding and Structure	14
Axial Central Blanket	
Volume, l	176.7
Height cm.	39.04
diameter, cm	107.36
Composition, depleted U-238 Oxide stainless steel clad	
Beginning of Life	
Fissile enrichment, Pu-239 & Pu-241	22
Fissile Loading Core, Kg.	1246.56
K effective	1.045

TABLE 3.7 CONTINUED

End of Life	
Fissile Loading core, Kg	1148.62
K effective	1.018
K overlife	0.027
reactivity in dollars	7.72

TABLE NO. 3.8

SHIPBOARD REACTOR CORE COMPARISONS

Parameter	Regular Core	Parfait #8	Parfait #9
Length to Diameter Ratio	L/D = 1.0	L/D = 1.0	L/D = 1.0
Enrichment , %	22	22	22
Radius, cm.	73.2	73.2	73.2
Blanket Size			
Central Internal	None	r=39.04 z=19.52	r=48.8 z=14.64
BOL Keff.	1.181	1.058	1.045
EOL Keff. 1000 days	1.135	1.028	1.018
Core			
Pu. TOT. BOL,Kg	1641.2	1483.6	1452.8
Pu. TOT. EOL,Kg	988.7	1305.0	1305.9 K_e=1.00
TOTAL			
Pu-239, BKT. PRODUCED	140.44	187.9	192.92
Pu-239, CORE BURNED,(Kg)	333.6	143.5	108.43 K_e=1.00
ΔK	0.047	0.030	0.027
ΔK dollars	13.40	8.55	7.70
Peaking factors Radial x Axial	1.79		1.51

Recommendations for further development of the physics design of the reactor will be discussed in Chapter 4. The next section will discuss the engineering of the reactor core.

3.3.3.2 Engineering Design

The core engineering can be considered a direct carryover of the GGA design work for the demonstration and commercial GCFBR's in many respects. In essence the core is derated from the 300 Mwe GGA demonstration plant design by not using roughened fuel elements and by using sealed fuel rods rather than pressure-equalized fuel rods.

The pressure-equalized fuel rods are GGA's prime or preferred fuel rod design. At present a fuel rod development program is underway to test both pressure-equalized rods and the back up sealed rod design. The sealed rod is very thick and is neutronically less desireable than the pressure-equalized rod. The reactor outlet temperature required is 1220°F (660°C) which seems to be at the upper limit of present fuel technology.

A core outlet temperature of 1220°F usually corresponds to a cladding hot spot temperature of around 1400°F (760°C) if one includes approximate values for typical hot channel factors. Hot channel factors for a GCFBR are discussed in Ref. W1. This would be at the design limit for strainless steel clad. The few sealed rods that have been tested with SS clad have performed well at temperatures to 1292°F (700°C) (W3) in thermal fluxes to 60,000 MWD/T. At present rods are being irradiated under fast flux in EBR II. The fuel element for GGA's proposed 1000 Mw(e) plant has a total length of 289.5 cm. The proposed shipboard design has an element length including the blanket of 236.4 cm. An additional 90 cm. gas plenum must be added to accomodate fission product gases, to give a total length of approximately 326 cm. this adds 3.5 p.s.i. pressure drop to the GGA fuel elements (smooth surfaces) 38.05 p.s.i.

The gas turbine system was designed to provide a much larger core pressure drop of 60 p.s.i. Therefore there is a margin for increasing by surface roughening the heat transfer effectiveness of the fuel rods. It may be possible to shorten the rod length by pressure-equalization of the fuel rods. This would remove the fission products and reduce the length of the fission gas plenum shortening the rod length. Decreasing rod length and core pressure drop would increase plant efficiency. Use of the pressure-equalized rod would also reduce cladding thickness, thereby improving neutron economy in the core.

Due to the large coolant void spaces, element swelling will be less of a design problem than in the LMFBR. Again the present design uses GGA developed approaches and analysis to insure compatibility in this area.

One of the major thermal design problems forseen is the difficulty in removing heat from the blanket region, as shown in Figure 3.12. The relative channel powers remain fairly constant over core life, and it will be possible to use fixed orificing to maintain a constant T across most of the core. This is not the case with the radial blanket, and research and development are necessary to devise a means of extracting blanket heat without mixing excessive cold gas from the blanket into the main gas stream, thereby lowering the reactor outlet temperature. This problem is more severe in a shipboard design because orifice adjustment during refueling shutdowns is less practicable. The blanket is usually under a separate fuel management scheme than the core and usually has a lifetime greater than the core by a factor of two. Additional study is necessary to optimize radial blanket design and management.

Since many of the engineering design decisions are safety-oriented, further dicussion of a number of features of the proposed design is postponed until the following section, which discusses this important topic.

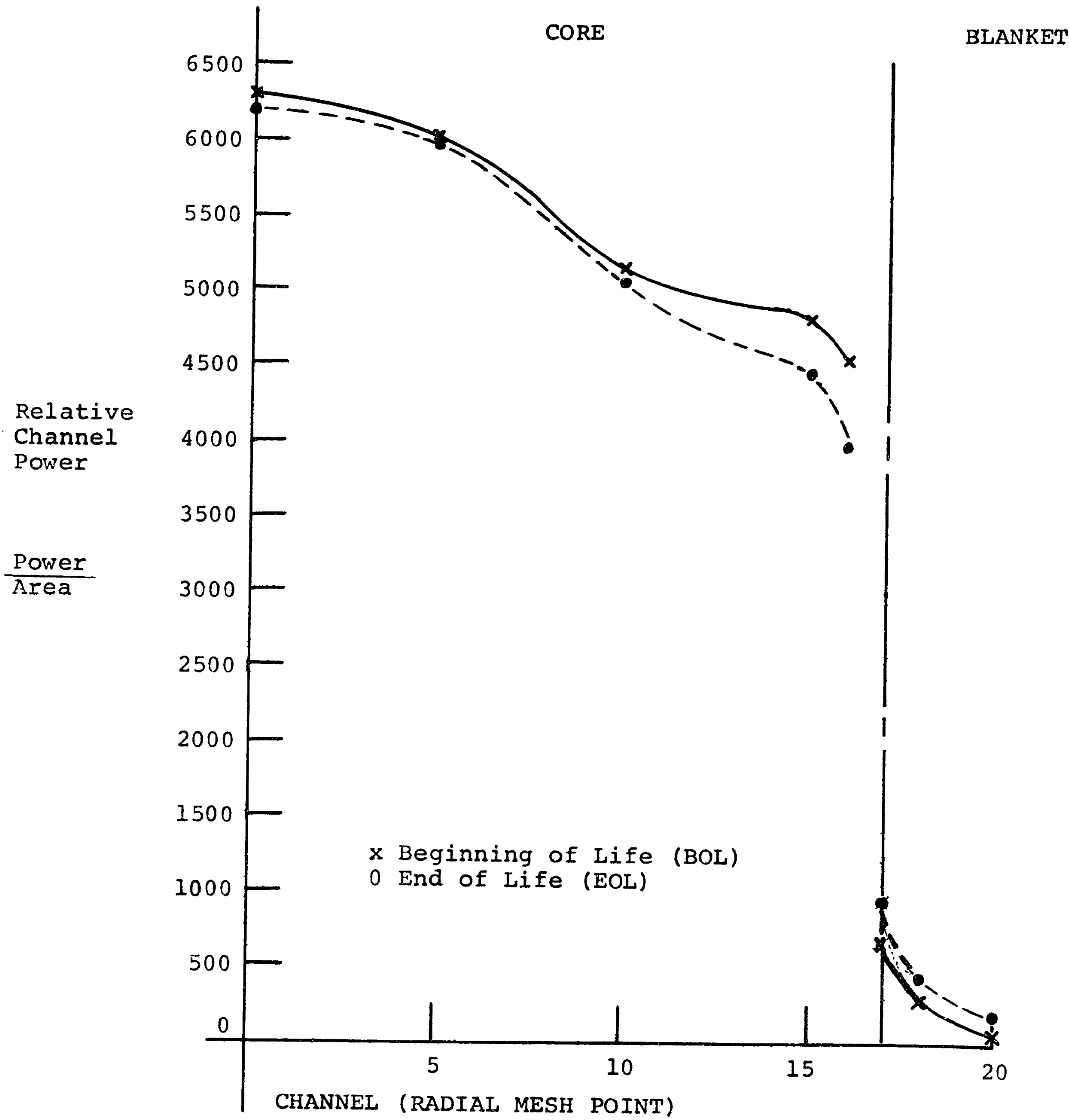


Fig. 3.12 RADIAL VARIATION OF RELATIVE CHANNEL POWER

3.3.4 SAFETY CONSIDERATIONS

3.3.4.1 Introduction

Nuclear reactors in mobile applications face a different set of safety constraints than for central station power plants. A major advantage of a mobile reactor is that it may be moved away from large population centers if the need arises. On the other hand, many problems arise as a result of the mobility. This section outlines some of the problems, and the solutions devised to safeguard the public and crew for the case of a shipboard nuclear propulsion system of the GCFBR type.

Section 3.3.4.2 discusses navigational and movement-related safety problems and features. Section 3.3.4.2 evaluates four of the major direct cycle design safety constraints: Emergency core cooling, plant control, coolant and component activation and helium leakage. Section 3.3.4.4 deals directly with specific gas-cooled reactor problems including water flooding of the core.

The concluding section summarizes the safety features inherent in the design.

3.4.2 Ship-Related Safety Requirements

3.4.2.1 Movement-related Problems

The N. S. Savannah served to define in part the criteria a nuclear ship must meet in order to gain entrance into harbors, an important task since there are many attendant dangers which land based plants do not have to contend with.

Navigational problems, such as grounding or collision, can result in containment damage, loss of ship control, sinking, and loss of core cooling. Other internal effects which are important design considerations are fire, explosion, missile generation, and ship motion in waves.

The consequences, according to Reade in Ref. R2, can be divided into two categories; 1) structural damage, and 2) an adverse change in normal operating conditions. The first of these includes penetration of the containment from collision or grounding, damage to equipment (e.g., pipes, valves, electrical distribution, pumps, control systems, etc.) due to fire or missiles, bulkhead failure following a cargo explosion, damage due to ship motion, and containment collapse from external pressure following sinking. The category of "adverse change in normal operating conditions" includes the loss of sea water for emergency core and containment cooling following a grounding that leaves the ship high-and-dry or clogs the sea water intakes, and the loss of power caused by fire, flooding, or structural damage. Key ship design criteria are set by the regulatory agencies such as the American Bureau of Shipping (ABS) and the U.S.

Coast Guard, or other agents such as Lloyds of London. (A2)

Most of the movement-related accidents are considered to have the highest probability in the port area or the heavily congested approaches. In these areas lie the greatest probability of collision or grounding. Also, the greatest population density encountered by ships is found in these areas. Since ports put the severest restrictions on nuclear powered ship operations, means of eliminating or reducing port traverse would lessen both the risk of ship accident and the consequences thereof. At present the principal means of dealing with movement-related accidents is two-fold. The first means lies in prevention of accidents by provision of highly trained crews, sophisticated navigation equipment and stringent administrative controls. The second lies in construction of the ship to higher-than-average ship standards, in provision of collision-absorbing barriers, and in containment of the reactor and primary systems.

As previously noted, a composite ship design was selected for the present evaluation based in part on its likelihood of having the most stringent design constraints. A second reason for the choice, enhanced safety potential, will be developed here.

The composite type ship would rarely enter inner port facilities, and would not have to make river passages to major ports. Loaded outgoing cargo sections could be brought by tug to outer roadsteads or anchorages and then the pusher ship (propulsion unit) could disengage from the incoming loaded cargo unit, and engage the outgoing loaded cargo unit. This could significantly reduce the hazard to the public of a nuclear plant being in a crowded harbor in a large city. This could be a good reason for 1) lessening collision barrier requirements, as the probability of an ex-port accident is much lower. Secondly, with a reliable, multiloop nuclear power plant, auxilliary propulsion units could be eliminated, as one reason given for their incorporation in a design is to move a ship from a harbor in the event of an accident. Additional safety advantages of the composite ship are: 1) in the event of a grounding it may be possible to decouple and float the nuclear pusher ship away; 2) there is a buffer space provided by the coupling mechanism between cargo and pusher ship in the event of fire, missiles, etc., and again it may be possible to decouple the nuclear pusher ship from a burning cargo unit; and 3) if the cargo unit is struck it may be possible to decouple and prevent the pusher ship from sinking with the cargo unit.

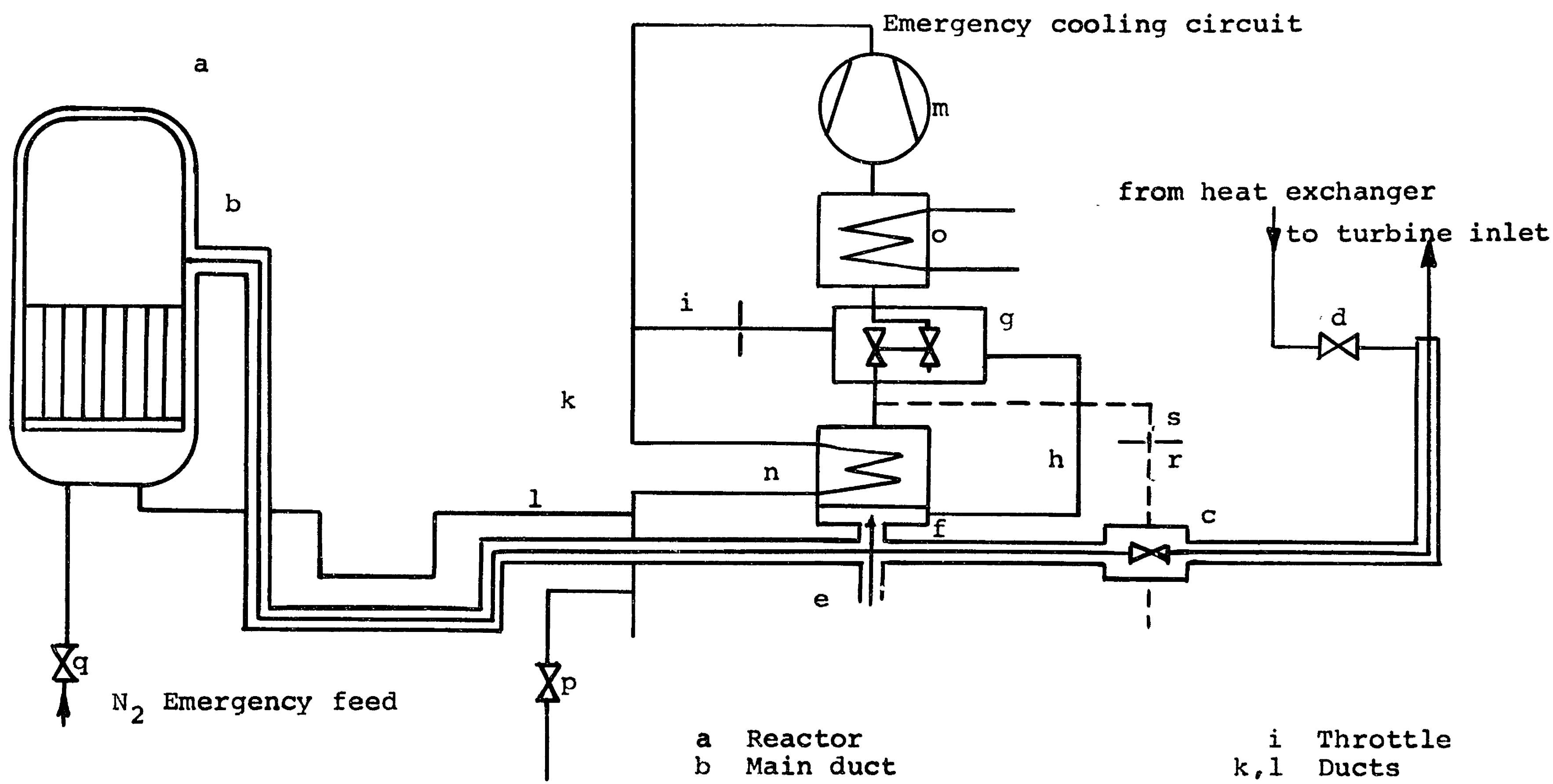
The above advantages do not lessen the necessity for keeping the pusher vessel of the highest quality. They may, however, relax or lessen the standards for the cargo end, a distinct economic advantage. The need for protection against navigational hazards must be maintained. Good ship construction with the design goal of maintaining the containment surrounding the reactor intact is a major safety consideration regardless of the type of reactor system employed. The composite ship may permit reduction in design standards at a later date, but for the present, the major effect of adoption of this design would be to lower the probability of a nuclear ship accident affecting the public.

3.3.4.3 Power Plant Safety Requirements

Direct cycle power systems have inherent safety problems which differ substantially from those in the well-studied indirect cycle plants. The following four major areas will be considered: 1) depressurization, emergency core cooling and afterheat removal; 2) Plant control for startup, shutdown and load following; 3) activation of turbomachinery components, and 4) system helium coolant leakage.

Emergency core cooling for a direct cycle system has been studied by the Germans, and solutions proposed for use with a helium turbine (B5). If no damage has occurred in the system, then it is possible to keep the system at no-load and remove the after heat slowly by the use of the main system. This can be done for a normal shutdown and to keep the plant at hot standby.

In cases where component failure has occurred some other method of core cooling is necessary. Figure 3.13 shows the circuit designed in Reference B5 for the removal of afterheat in the HTR.



a	Reactor	i	Throttle
b	Main duct	k,l	Ducts
c	Hot gas valve	m	Emergency Cooling blower
d	Cold gas valve		
e	Emergency cooling connection	n	Heat exchanger
f	Emergency cooling double duct	o	Cooler
g	Double valve	p,q	Valves
h	Duct	r	Duct
		s	Throttle

Fig. 3.13 GAS-COOLED DIRECT CYCLE
EMERGENCY COOLING

The circuit appears complicated, but this is primarily due to the use of concentric ducts, with the hot gas in the center pipe and the colder gas in the annulus between the two pipes. The other complicating feature is the need to keep the circuit hot in order not to thermally shock the reactor in an emergency. Details of the design can be found in Ref. B5.

Plant control schemes for a direct cycle helium gas turbine are in the early design stages. Designs of closed cycle air turbines provided the basis for present systems. A system has been designed for the 25 Mw(e) Geesthacht plant. (Ref. B2 & B6)

Gas turbines must be accelerated to a self-sustaining speed by some external means (i.e., electric motor). Further, they require gas at a suitable temperature to allow operation. Since a single shaft (per turbine) system has been selected for reasons previously discussed, start up will be more difficult, as will be shaft design. However, the shipboard application is such that startup and shutdown cycles will be kept to a minimum. There will be no reason to routinely shutdown while in port as steam ships do. The composite type ship again favors this concept.

Electric drive also assists in maintaining load on the system, either for supplying the auxilary power or just turning the alternator. In addition to assisting in control by simplifying startup and shutdown, reversing the propellor is made relatively easy by use of electric drive. It is also conducive to quick and easy maneuvering.

Rapid large maneuvering changes should not be necessary, but turbine control through variation of the total mass of gas in the system (i.e., pressure control) and throttle and by-pass valves, will permit a wide range of power plant maneuvering. The by-pass valve system will also provide protection against loss of load or overspeed of the turbine (H1).

At present, the combined by-pass and pressure level control system have proven satisfactory in air closed-cycle plants. By-pass control is characterized by high control speed; and pressure level control by economy with adequately high speed. The main characteristic of importance is that control information is received from the circuit and not the electric end. The reference input is turbine inlet pressure and the quantity controlled is the heated (reactor) outlet temperature. This type of control maintains the temperature changes in the reactor within an acceptable range. Excellent descriptions of the circutry and control systems can be obtained from references B2 and B6.

An additional reason control is important, is its role in protecting against fuel cladding failure. Fuel failure in a closed cycle system leads to fission product carryover into the turbomachinery and heat exchangers. This carryover, together with activation of the impurities in helium may necessitate secondary shielding and increased maintenance and inspection times. Both effects are undesireable. Neutron-induced activation in pure helium is essentially nonexistent. The problem occurs due to the small amounts of impurities present in grade A helium: 5 parts per million (ppm) of H_2, 45 ppm of H_2O and 73 ppm of air. Of these Argon-41 activation (half-life 1.83 hrs) is of most concern. Removal of argon may be possible and is under study. (V2)

Fission product release and deposition could be a major problem, but one for which there may also be numerous solutions. As stated previously, much information on this aspect will come from the 25 Mw(e) Geesthacht power plant. The pressure-equalization fuel rod proposed by GGA as their preferred fuel rod for the GCFBR would serve to eliminate most of the gaseous and volatile fission products from the rod. Then, in the event of a failure, little of the fission product inventory would be available for release into the system; in addition the driving force (differential pressure) leading to mass flow would be very small. The sealed fuel rod, on the other hand, relies on high cladding integrity to prevent failure and subsequent release. This problem is another reason for use of an integral PCRV for the initial design. If a fission product problem does exist, then shielding is already present for all components. The helium coolant system will also include a purification loop to maintain the helium in the cleanest possible condition. The problem common to many thermal spectrum GCR's of carbon being circulated by the helium and causing deleterious effects upon materials is not a problem in the fast reactor as no exposed graphite is present. Care must be taken in turbine and duct insulation, selection, design and installation not to compromise decontamination or to create an activation problem.

It appears that no difficulty should occur due to radioactive materials in the system and that in the long term turbomachinery will only need minimal shielding during operation, inspection, or maintenance.

Helium is an expensive coolant costing about five times as much as CO_2 per SCF. Therefore, it is in the designers interest, for two reasons, to maintain helium leakage at a very low value. In the first instance the helium will be carrying some activated impurities or fission products, and from the radiological safety standpoint, it would be undesireable to have these leaking out of the primary containment. The second factor is the makeup cost, not only for the replacement helium but in terms of the cost of maintaining the excess inventory on board ship.

There are many design ideas which have been proposed to decrease helium leakage from such systems. The use of concentric ducting helps to decrease the total system surface area available for leakage to the containment. Probably the most important design feature in reducing helium losses is placement of the high pressure, high temperature turbine seal on the inlet end of the turbine against, or next to, the hot side or outlet seal of the compressor. This lowers the total Δp across the seals, reducing both the leakage and the complexity of the seal design. Fairly elaborate seals and bearings have been designed, as referenced in B6. In addition to seals, the casings, ducts, valves, etc. must all be carefully manufactured and inspected thoroughly. All flanged connections must have welded lip seals. Through-bolts are provided with welded caps. Contact faces have channels for helium leakage removal.

3.3.4.4 Reactor Safety

Considering the predicted short-term viability of the gas-cooled reactor for indirect-cycle central station power plants, more research and development on safety features has been done for it compared to direct-cycle plant in general, and a shipboard application, in particular. The discussion in this section will briefly outline gas-cooled fast reactor safety problems and concentrate on those few which affect the technical feasibility of the design. Much is still unknown about the performance of gas-cooled fast reactors, since none have ever been build and operated.

As previously discussed, fuel rod development tests are continuing and much of the safety will be dependent on fuel rod design.

It would be difficult and beyond the scope of this report to make the decision which fuel rod is safer, sealed, or vented. It should suffice to say that both sealed and vented rods appear feasible, and either may be better suited for use in the shipboard GCFR.

The major items most affecting the design feasibility are water ingress into the core in the event of a sinking, PCRV technology, and depressurization and/or loss of flow.

Water flooding of the core is almost certain in the event of a sinking in deep water, since the PCRV must be relief-valved to allow flooding before its crushing strength is exceeded.

At a specific seawater depth the concrete reactor vessel will begin to fail and will either crumble or crush the reactor. It is usually considered preferable to install a set of relief valves to allow the containment to be pressure-equalized prior to the depth where collapse will occur but at a depth where little hazard will be created.

Solutions to the problem of water making the core more reactive have been proposed in regard to the Steam-Cooled Fast Breeder Reactor design (S3). It appears from the results of those studies that a poison such as Gadolinuim, or Europium or some other rare earth, which has large absorption resonances in the epithermal to thermal region could be incorporated in the fuel to poison the core. Gadolinium oxide was chosen initially by GGA (F8) when wet refueling was considered for the GCFR. A concentration ratio of 1 atom of Gadolinium to every three fissile atoms was considered sufficient to maintain the core subcritical when fully flooded <u>with all rods out</u>. The loss in conversion ratio is 0.05. Gadolinium Oxide can be placed in the fuel during fabrication.

The problem of water in the reactor core does not present a problem in terms of technical feasibility, but will have a modest effect on the economics.

PCRV technology has rapidly advanced in the past few years. Multicavity PCRV's are being tested for future use, and the Ft. St. Vrain PCRV has been certified as a containment concept <u>not</u> requiring a secondary containment building. This means the area surrounding the PCRV can be designed for controlled leakage and maintained at a pressure slightly above atmospheric to assist in emergency core cooling. The controlled leakage area could serve as an exclusion zone and would be less costly than an additional containment.

Containment of the entire helium containing system in a PCRV would be highly desireable from the safety aspect: of prime concern is the depressurization accident leading to a reduction of coolant in the primary system. With all ducts, turbomachinery and heat exchangers within the prestressed concrete reactor vessel, catastrophic failure leading to a rapid depressurization is reduced to an almost negligible probability. Double seals are required for the PCRV for the Ft. St. Vrain reactor (D 10). With double seals the probability of use of emergency core cooling systems is further reduced.

There are several major decisions which affect the feasibility and safety with respect to the PCRV. Non-integrated and integrated or semi-integrated design are all options that are possible. The integrated design would obviously be the heaviest system, but scoping calculations have shown the partial weight (PCRV only) to be approximately 127 lbs/shp. This calculation is based on determination of the approximate enclosed volume of all major components and then using the known weight of an already designed PCRV of the same enclosed volume. The Fort St. Vrain PCRV was used for the present purpose: even though it was only a single cavity vessel it has the highest working pressure (T4, A3). The gas turbomachinery had a weight which is nearly a factor of three lower than the estimated weight of the equivalent single shaft steam turbine for the indirect cycle (L3). This would partially offset the weight of the PCRV.

It appears possible that semi-integrated designs, with just the reactor inside the PCRV are possible in the future. In this case a secondary containment would be required.

Other nuclear accidents such as control rod ejection and refueling accidents are the same for the proposed design as for the GGA design. Discussion of these accidents can be found in references F8 and O1.

The shipboard GCFR does not appear to have any unsolvable technological safety problems. Many areas need additional research and development in order to insure a safe design, but most areas are shared in common with proposed land **bas**ed GCFBR design proposed for central station power generation.

CHAPTER IV

CONCLUSIONS AND RECOMMENDATIONS

4.1 Introduction

Advances in gas-cooled reactor technology, pre-stressed concrete vessel construction, and gas turbine design have combined to create a climate favorable for the appraisal of a gas-cooled fast reactor with a direct cycle helium turbine. At the same time increasing world trade and expanding markets have created a demand for new, fast, large ships. At present, new ship types, transportation systems, and propulsion units are being studied and developed for future employment. As the fast breeder reactor is predicted to gradually assume the role of the major central station electrical energy producer, an evaluation of the GCFBR direct cycle system for nuclear ship propulsion was considered to be an appropriate objective for this thesis.

4.2 Conclusions

The gas-cooled fast breeder reactor appears to be a viable power source for shipboard propulsion units. Coupled with a direct-cycle helium power conversion system it offers promise of safe, economically attractive power.

Reactivity burnup control appeared to be a major problem affecting feasibility in terms of obtaining adequate time between refuelings. Batch loading of a core having a central blanket (the Parfait core) was studied and found to be a sound design approach to achieving refueling times on the order of 1000 days. The central axial blanket also helps to decrease power peaking factors.

Engineering of all major systems appears practical and the overall system appears very safe. The PCRV will add considerable to the protection of the reactor and prevention of a catastrophic loss of coolant accident. Two parallel loops, as proposed, offer some degree of redundancy; a four-loop design could also be made compatable with the PCRV internal layout.

Water ingress to the reactor in the event of sinking seems to be an easily solvable problem. The same holds true for most ship-related problems. Selection of the composite ship and PCRV design options appeared to offer a particularly favorable combination of advantages. There are acceptable alternatives to both the composite ship and the PCRV, however,

4.3 Recommendations

Additional research and development on a number of phases of ship, power plant and reactor design are necessary before a system of this kind can be built. Further optimization and detailed design work is necessary to evaluate specific design trade-offs to determine the economically viable ship,the regulatory-acceptable ship and the ship owner-attractive ship.

There is very little firm economic information currently available to back up the proposed design. Detailed economic information must be generated for the ship, reactor and power plant. The economics must be generated within a transportation system, such as that proposed for the Composite Ship by Teasdale (T3).

Gas-cooled fast reactor design is in its early stages: no system of this kind has yet been build. The 300 Mw(e) GCFBR Demonstration design proposed by GGA could be derated and serve dual use as both the demonstration plant for larger land based central station plants, and as a prototype for additional ship reactors.

More detailed calculations mating the Parfait core gas-cooled reactor and the gas turbine system must be done.

Design optimization of the Parfait core is needed to optimize the effect of the central blanket for power flattening and reacitvity control.

A parallel effort is now underway on the use of this type core in sodium-cooled reactor (D9), which should contribute to this objective. Reference W1 discusses in detail further work which is necessary for the physics and engineering development of the gas-cooled fast breeder reactor apart from the shipboard environment.

Gas turbines for closed-cycle application are just emerging. Additional information is being developed by the 25 Mw(e) Geesthacht Power plant on the interaction of nuclear reactors and gas turbines. Increased development of closed and open cycle gas turbines will greatly assist in designing future units for use aboard ship.

The PCRV, while eliminating the need for separate primary and secondary biological shielding, is heavy when compared with recent PWR CNSG designs. Although the lighter helium turbomachinery will save weight, the PCRV is by far the largest and heaviest propulsion component. The weight of the PCRV alone is 127 lbs/shp compared to around 67 lbs/shp for a CNSG. However, the shielding and reactor weight has been optimized to the lowest practical value for the CNSG system: the Savannah's shipboard PWR's weighed in at a very heavy 360 lbs/shp. The PCRV has so far only been used in land based central station plants where weight is not an important consideration. PCRV weight and design has to be optimized for shipboard applications. This optimization along with increased data measured on operational PCRV's, which will allow reduction in design margins, can significantly reduce the PCRV weight. Unquestionably, a lower PCRV weight will be necessary to make the design more attractive. However, even the present weight is tolerable, and it will be more system economics than weight per se which is the final determinant.

The future American merchant marine fleet will need greatly increased financial and technical support in order to compete for world trade. The gas-cooled fast reactor coupled to a direct-cycle nuclear gas turbine may provide a competitive propulsion plant. The Parfait core and composite ship are two technically innovative ideas which will assist in making this an economically viable system.

REFERENCES

A.1. Aerojet - General Nucleonics, "The ML 1 Design Report," IDO - 28550, (May, 1970).

A.2. Andrea, M.I. "Economic Considerations for the Application of Nuclear Power to Merchant Ships" PB - 175510 (May, 1967).

A.3. Arthur, I.P., "Prestressed Concrete Pressure Vessels," GA - 9016 (October, 1968).

B.1. Bohm, E. et al., "The 25 Mw(e) Geesthact KSH Nuclear Power Plant," Proceedings of a Symposium on Advanced and High Temperature Gas Cooled Reactors, IAEA Julich (October, 1968).

B.2. Bammert, K. et al., "Performance of a High Temperature Reactor with Helium Turbines," Ibid.

B.3. Brewer, S.T., E.A. Mason, and M.J. Driscoll, "The Economics of a Fuel Depletion in Fast Breeder Reactor Blankets," COO-3060-4, MITNE-123,(August, 1972)

B.4. Bondarenko, I.I., editor, "Group Constants for Nuclear Reactor Calculations," Consultants Bureau, N.Y. (1964).

B.5. Bammert, K., "Emergency Cooling System for a Nuclear Power Gas Turbine," Atomkernenergie (ATKE) Bd. 18 (1972).

C.1. Conklin, D.L., "Economics of Nuclear and Conventional Merchant Ships," USAEC, (June, 1958).

C.2. Crouch, Holmes F., Nuclear Ship Propulsion, Cornell Maritime Press, Cambridge, Maryland (1960).

C.3. Cautius, W. et al., "Operational Experience with the AVR Reactor," Proceedings of a Symposium on Advanced Temperature Gas-Cooled Reactors, IAEA, Julich (October, 1968).

D.1. Delson, E.B. and E.C. Hunt, "The 630A - Mark IV, A Nuclear - Fueled Steam Generator - Superheater for Merchant Ship Propulsion," "Marine Technology," Vol. 2, No. 4, (October, 1965).

D.2. "Directory of Nuclear Reactors," IAEA, Vol. 5 (1964).

D.3. Dalle, Donne, M. and K. Wirtz, "Problems of Gas-Cooled Fast Breeder Reactors," Annual Meeting, American Nuclear Society, Boston, Massachusetts (June, 1971).

D.4. Dalle, Donne M. et al., "High Temperature Gas Cooling for Fast Breeders," Proceedings of the Symposium on Advanced and High Temperature Gas-Cooled Reactors, IAEA, Julich (1968.)

D.5. Dibelius, G., "Gas-Cooled Reactors with Gas Turbines using Helium, Neon, or Carbon Dioxide," Ibid.

D.6. Dee, J.B., P. Fortescue and J.A. Larrimore, "Gas-Cooled Fast Breeder Reactor Studies" GA-10678 Gulf General Atomic Co., California (June, 1971).

D.7. Directory of Nuclear Reactors, Volume No. 4, IAEA, Vienna (1962).

D.8. Driscoll, M.J. "Supplement to M.I.T. 22.94 Notes (June, 1969).

D.9. Ducat, G.A., "SCD Thesis Prospectus; Evaluation of the Parfait Blanket Concept for LMFBR Core Design," Department of Nuclear Engineering, M.I.T. (March, 1972).

D.10. Docket 50 - 267, Ft. St. Vrain "Final Safety Analysis Report" Amendment 14 Vols. 1 - 4.

E.1. Edison Electric Institute "Fast Breeder Reactor Report" April, 1968 EEI, N.Y.

E.2. Esleeck, S.H. "Application and Economics," Marine Nuclear Steam Propulsion Executive Briefing, Babcock and Wilcox Company.

E.3. El Wakil, M.M. Nuclear Power, McGraw HillBook Co.(1962).

F.1. Fortescue, P. and W.I. Thompson, "The GCFR Demonstration Plant Design," GA - 10036 (1970).

F.2. Feher, E.G., "The Supercritical Thermodynamic Power Cycle," Douglas Paper 4348, Douglas Aircraft Co., Missile and Space Systems Division.

F.3. Feher, E.G., "Application of the Supercritical Cycle to Electric Power Generation in Space," Douglas Paper 4932 Douglas Aircraft Co. (May, 1968).

F.4. Feher, E.G., "Feher Cycle" Astropower Laboratory, California.

F.5. Fortescue, P. "Gas Turbines and Nuclear Power", Gulf General Atomic Co. (April, 1971).

F.6. Fortescue, P., "Tomorrow's Plant; Gas Turbines, Nuclear Power, Dry Cooling," Power Engineering, (August, 1971).

F.7. Frame, A.G. et al."Design of the Prototype Fast Reactors," Paper 3-3 Fast Breeder Reactors, Proceedings, P.V. Evans, Editor, Pergammon Press, Oxford (1967).

F.8. Fortescue, P., et al., "Safety Characteristics of Large Gas-Cooled Fast Power Reactors," GA-6732 (October, 1965).

G.1. Gaunt, I.A.B. and G.R. Wilkinson, "Nuclear Propulsion for Merchant Ships," International Shipbuilding Progress, Vol. 16, No. 178.

G.2. Gulf General Atomic Corporation "Peach Bottom HTGR," Information Brochure.

G.3. Gathy,B.S. "A Marine Nuclear Power Plant Design Utilizing the Direct Brayton Cycle," U.S. Coast Guard Academy, New London, Connecticut.

G.4. Gulf General Atomic, "1100 Mw(e) HTGR Nuclear Steam System," Gulf General Atomic Co.

G.5. General Atomic Corporation, "Abstract of Design Study of 1000 Mw(e) Gas-Cooled Fast Reactor Plant," GA - 7240 (December, 1967).

G.6. Gratton, C.P. et al., "A Gas- Cooled Fast Reactor with Direct Cycle Potential," Proceedings of the Symposium Advanced and High Temperature Gas-Cooled Reactors, IAEA, Julich (1968).

G.7. Gasparovic, N., "Nuclear Helium Gas Turbines," Energy International (April, 1971).

G.8. Gulf General Atomic Corporation, "PCRV (Prestressed Concrete Reator Vessel)" Information Brochure (1970).

H.1. Hurst, J.N. "Some Considerations Affecting the Integration of Gas Turbines with Direct-Cycle Gas-Cooled Reactors," SM 111/41, Proceedings of a Symposium, Advanced and High Temperature Gas-cooled Reactors, IAEA, Julich (1968).

K.1. Kiss, R.K. and J.J. Garvey, "CMX Designs - Merchant Ships for the '70's," Marine Technology, Vol. 7 (October, 1970).

K.2. Kesterman, F. "Nuclear Containerships," Nuclear Engineering International, Vol. 14 No. 153 (February, 1969).

K.3. Krasin, A.K., et al., "Prospects of Using Chemically Reacting Systems As Coolantand Working Substance in Atomic Power Stations with Fast Nuclear Reactors," JPRS - 48330, CEMA Symposium on Atomic Power Stations with Fast Reactors (Volume 1) (July, 1969).

K.4. Keller, C., "The Origin and Development of the Closed Cycle Gas Turbine System," Escher Wyss News, Vol. 39, No.1 (1966).

K.5. Keller, C., The Use of the Closed-Cycle Gas Turbine on Atomic Power Plants, Ibid.

L.1. Little, W.W., Jr. and R.W. Hardie, "2DB User's Manual... Revision 1" BNWL - 831 Rev. 1, (August, 1969).

L.2. Lys, L.A. and J.H. Broido, "Principal Considerations in the Design of Gas-Cooled Fast Reactors with Direct Cycle Gas Turbines," Proceedings of a Symposium, Advanced and High Temperature Gas-Cooled Reators, IAEA, Julich (1968).

L.3. Lys, L.A. "Helium Cooled Reactors," Energy International, (May, 1970).

L.4. Lyall, H.G."A Comparison of Helium and CO_2 as Reactor Coolants," Journal of Nuclear Energy, Vol. 26, No. 2 (February, 1972).

L.5. Lanning, D.D. "Power Reactors" 22.32 Course Notes, M.I.T. (1972).

M.1. Moore, W.C., "Nuclear Safety Considerations Affecting the Design of the Maritime Gas-Cooled Reactor," Nuclear Ship Propulsion, IAEA Proceedings Series, (1961).

M.2. General Dynamics Co.,"Digest of the Advanced Nuclear Cargo Ship Study," MA-4192 (June,1968).

M.3. Mueller, R.A., et al., "Design and Evaluation of a Steam-Cooled Fast Breeder Reactor of 1000 Mw(e)," Fast Breeder Reactors, BNES, (May 1966).

M.4. Meleese - d'Hopital, G. and P. Fortescue "Thermodynamic Comparison of a Gas Coolant for Nuclear Reactors," High Pressure Gas as a Heat Transport Medium, The Institution of Mechanical Engineers, Proceedings 1966 -1967, Vol.181, Part 3I.

M.5. Meleese, G. "Influence of the Choice of Coolant on Reactor Thermal Performance," Nuclear Applications, Vol. 2, No. 3 (June, 1966).

M.6. M.I.T. 22.26 Nuclear Reactor Design Solution Book (Spring, 1969).

M.7. Meleese, G. and J.B. Dee, "Gas-Cooled Fast Breeder Reactor Designs, GA - 10325, (January, 1971).

M.8. Menon, S.K., "Prestressed Concrete Pressure Vessels for Water Reactors," First International Conference on Pressure Vessel Technology, Delft, (1969).

N.1. Nuclear Industry, Vol. 19, No. 4 PP. 36-37 (April, 1972).

N.2. The Nuclear Industry, 1970, United States Atomic Energy Commission

P.1. Pitkin, M., "Introduction Address," Marine Nuclear Steam Propulsion Executive Briefing, Babcock and Wilcox Co. (1971).

P.2. Pocock, R.F., Nuclear Ship Propulsion, Ian Allen LTD., Shepperton, Surrey, United Kingdom, (1970).

P.3. Pelland, B., "Physics Design of the Gas-Cooled Fast Breeder Reactor Demonstration Plant," GA - 10509 (August, 1971).

R.1. Ruhe, W.J., "Potentials of Nuclear Power at Sea," Nuclear News (August, 1969).

R.2. Reade, J.M., IV.,"Design Criteria for Nuclear Powered Merchant Ships," Naval Engineer Thesis, Department of Naval Architecture and Marine Engineering, M.I.T.(1969).

S.1. Silk, E.F. and R. Cameron, "Progress on the Engineering and Construction of the Dungeness 'B' Nuclear Power Station," Proceedings of a Symposium Advanced and High Temperature Gas-Cooled Reactors, IAEA, Julich (1968).

S.2. Sawyer, G.A., "Review of CNSG Marine Reactor Program," Proceedings of the Conference on Nuclear Marine Propulsion, Blacksburg, Virginia, USAEC CONF 640810 (August, 1964).

S.3. Sofer, G. et al. "Steam-Cooled Power Reactor Evaluation, Steam Cooled Fast Breeder Reactor," NDA - 2148 - 4, USAEC (April, 1961).

S.4. Starkus, J. "Preliminary Analysis of a Closed Brayton Cycle for Nuclear Power," M.S. Thesis Department of Mechanical Engineering, M.I.T., (1971).

T.1. Thorn, J.D., "The Use of Gas-Cooling in Nuclear Power Reactors," High Pressure Gas as a Heat Transport Medium. Proceedings of a Symposium, Institution of Mechanical Engineers, Vol. 181, Part 3I (1966- 1967).

T.2. Turner, F.H., "Prestessed Concrete Pressure Vessels For Nuclear Power," British Ropes, LTD., Doncaster, England (1966).

T.3. Teasdale, J.A., "The Modern Composite Ship, a Competitive Nuclear-Powered Merchantman,Proceedings of the Royal Institution of Naval Architects, (1967).

T.4. Tan, C.p., "Prestressed Concrete in Nuclear Pressure Vessels - A Critical Review of Current Literature," ORNL - 4227 (May, 1968).

U.1. United States Code Congressional and Administrative News, November 20, 1970, No. 12, Legislative History of PL 91 -469 Merchant Marine Act of 1970.

V.1. Van Dievoet, J.P., "The Sodium - CO_2 Fast-Breeder Reactor Concept," ANL-7520 Part 2.

V.2. Von Der Decker, C.B., "The Program of Plate Out Investigations in the Gas Turbine Project. Proceedings of the GCR Information Meetings at the Oak Ridge National Laboratory. (April, 1970).

W.1. WASH 1089, "An Evaluation of Gas-Cooled Fast Reactors," Division of Reactor Development and Technology, U.S. A.E.C. (April, 1969).

W.2. WASH 1088, "An Evaluation of Steam Cooled Fast Breeder Reactors," Division of Reactor Development and Technology USAEC. (April, 1969).

W.3. Washburn, T.N. et al., "Fuel Element Development for the Gas-Cooled Fast Breeder Reactor, Part 1, Sealed Fuel Rod Design, ORNL - TM - 3427, (June, 1971).

Y.1. Yevick, J. Fast Reactor Technology: Plant Design, M.I.T. Press (1966).

In addition to the above specific references, the following bibliography is valuable:

1) Bibliographies in Nuclear Science and Technology, Section 2.2. Nuclear Ship Propulsion, AED - C- 22 - ol, through 11.