CONCEPTUAL DESIGN OF AN HTGR SYSTEM FOR A TOTAL ENERGY APPLICATION

by

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ABSTRACT

A conceptual design for a small HTGR in the 100 MWe size range is described. The reactor drives indirect closed-cycle gas turbine power conversion units using helium as the working fluid and provides both electricity and thermal energy (via a 380°F hot-water utility system) to serve the projected needs of large U.S. Army installations and industrial facilities in the continental U.S. in the post 1985 time frame.

The overall system design combines many of the proven features of the Peachbottom I reactor, the Fort St. Vrain HTGR core, and Oberhausen II turbomachinery. The major unique feature is the use of an indirect power cycle, with helium-to-helium intermediate heat exchangers.

Cost estimates are summarized which indicate that the ability of the gas turbine cycle to discharge waste heat at a useful temperature gives the HTGR/GT system a significant advantage over nuclear and fossil-fired Rankine systems even though it is inferior to LWR systems on an electric-only basis. The fossil-fired gas-turbine total-energy concept is identified as its major competitor for the present application.

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Chapter 1

INTRODUCTION

1.1 Foreword

The objective of the work summarized in this report has been to develop and evaluate a conceptual design for a nuclear total utility system (NTUS) for provision of both electricity and thermal energy to large DOD installations in the continental United States in the post-1985 time frame. It has been carried out as part of a program sponsored by the U.S. Army Facilities Engineering Support Agency to determine a strategy for provision of future DOD energy needs in light of recent large increases in fossil-derived energy costs.

The concept developed in this study couples a high-temperature gas-cooled reactor (HTGR) to a closed-cycle gas turbine power conversion system using helium as the working fluid. In many respects the final design may be thought of as an updated and scaled-up version of the Peachbottom-I HTGR combined with Oberhausen-II turbomachinery. It will be shown that a NTUS of this type should be able to deliver energy cheaper than most fossil-fired units or other nuclear options.

In this initial chapter the assumed and derived groundrules governing the scope of the study will be discussed, and an outline of the body of the report presented.

1.2 Background and Groundrules

Previous studies (N1,S4) have shown that there are a dozen or more military installations in the continental United States having substantial ($\gtrsim$25 MWe) peak electrical demand, which is projected to grow into the 50-100 MWe range in the post-1985 time frame. Thermal energy demand at the same installations averages several times the electrical demand. Two important observations follow:

(a) At and above about 100 MWe small nuclear plants are potentially competitive with fossil-fired stations (at 1974 fuel costs) even when operated to produce electric or thermal power only. (I1,T2)

(b) The installations are well suited to exploitation of the total energy concept, and it is readily demonstrated (M1) that a system producing both thermal and electrical energy can deliver the combined products cheaper than a single-product plant.

These circumstances motivated the present investigation.

An important part of both the original research proposal and of subsequent work has been the development of an appropriate set of groundrules governing the study. Table 1.1 summarizes the final set of conditions, together with commentary on the significance of each item. The system described in this report conforms to the above criteria.

Table 1.1

Groundrules

ITEM	COMMENT
(1) The plant should be licensable under current AEC civilian reactor criteria; and no special provisions will be made for the military nature of the application	(a) Risk to the general public and environmental impact shall be no greater than a comparable civilian application, and conform to all applicable governmental regulations.
	(b) No site hardening required other than normal seismic resistance; no security beyond civilian norms.
	(c) The plant should be capable of being deployed at sites having the general characteristics of the larger DOD/CONUS bases, but no specific bases will be considered at the present time.
(2) The plant should be capable of assuming entire base electrical load at system mid-life and as much of thermal load as shown to be economically practicable.	(a) Normal capability of HTGR/GT reactor system is approximatel 2 MWth/MWe
	(b) The electrical load is well defined and already tied in to a centralized distribution system.
	(c) Use of heat pumps and absorptive air conditioning to optimize utility load balance is to be evaluated.
	(d) Annual total energy growth rate of 3% is assumed; with electrical tending to grow faster than thermal.
(3) A specific time frame is designated	(a) 1985 reactor startup is envisioned; costs will be quoted for 1974 and also escalated to 1985
	(b) A 30 yr. lifetime is assumed.
	(c) Reliance on state-of-the-art technology is implied; and R&D necessary should be well defined and of limited scope in terms of both financial and time requirements.

Table 1.1 (continued)

(4) DOD economics will be employed	(a) 10% annual rate used as effective cost of capital. (b) Nuclear fuel carrying charges are waived; no credit for bred fissile material. (c) Fabrication, burnup, shipping and reprocessing charges are assessed as in the civilian economy. (d) No local, state or federal taxes assessed.
(5) Reliance on the local electrical utility grid is restricted.	(a) Access to grid power provided as required for reactor safety assurance. (b) It will not be assumed that excess electric power can be sold or given to the grid, but no design features prohibiting this will be adopted. (c) Access to grid power during scheduled downtime for maintenance or refueling will be assumed, but not during unanticipated outages.
(6) The stand-alone capabilities of the system are restricted	(a) On-site cooling tower capability provided for 100% of thermal load (b) Essential or uninterruptable electrical or thermal loads (e.g., hospitals) will provide their own emergency power supplies. (c) No storage provisions will be made for electric power in the reference design, but feasibil of add-on systems will be assessed (e.g., H_2 production, flywheel storage) (d) Sufficient thermal (hot water) storage will be provided to smooth the daily load.

In addition, as a result of system reliability analyses it was decided to provide additional assurance against partial system incapacity by providing an on-site fossil-fired total-energy gas turbine unit capable of providing one reactor-loop's worth of energy (50 MWe, 100 MWth). This will insure the following capabilities:

(1) Provision of 100% of rated system power during the estimated 15% of the time when one reactor loop/ turbine plant is out of service.

(2) Provision of 150% of rated system power to meet peak load near the end of plant life -- otherwise it will be necessary to grossly oversize the nuclear unit.

(3) Provision of all or nearly all power during minimum load conditions: Spring and Fall and weekends, when scheduled inspections, maintenance and refueling operations can be conducted.

(4) Provision of essential services during unscheduled outages.

(5) Provision of the energy reconversion unit for an energy storage system using hydrogen, if such is deemed practicable.

(6) Provision of an added level of emergency power for reactor safety assurance.

The many advantages of this back-up system were considered to justify its modest cost: approximately 150 $/KWe of gas turbine installed or 75 $/KWe of total reactor power.

Finally, the results of this study were to consist of a conceptual design in sufficient detail that it could be used as the basis for discussion of specifics with a reactor vendor and architect-engineer firms, and an economic evaluation of system costs and the unit costs of power therefrom were to be made and presented in a manner which facilitated comparison with alternatives. In this latter regard, while the focus of the technical effort was to be on the HTGR/GT system, it was recognized that sufficient economic evaluation of fossil-fired and other nuclear alternatives would have to be carried out to confirm that these systems were less suited to the present application than the HTGR/GT. The magnitude of this phase of the work led to augmentation of the original contract proposal and resulted in a subtask whose efforts culminated in the report: L.J. Metcalfe and M.J. Driscoll, "Economic Assessment of Nuclear and Fossil-Fired Energy Systems for DOD Installations," Feb. 15, 1975. In the present report we will rely heavily upon this document, summarizing rather than repeating its major results and conclusions in the areas noted.

1.3 Outline of Report

The body of this report begins with a discussion of the options involved, and the considerations which support the selection of the HTGR/GT system (Chapter 2). This is followed by a detailed description of the power conversion system (Chapter 3) and nuclear-related systems (Chapter 4), core-related aspects (Chapter 5), reactor safety (Chapter 6), and economics (Chapter 7). The report concludes with a discussion and summary (Chapter 8), and appendices containing various details in support of the main text.

Chapter 2

DISCUSSION OF OPTIONS

2.1 Introduction

In response to the needs and within the constraints described in Chapter 1, a NTUS concept has been developed around a high temperature gas cooled reactor as the energy source, coupled to a closed cycle gas turbine system as the power conversion unit. Specific design details of the system will be described in Chapter 3. In the present chapter the considerations which led to selection of this particular combination of characteristics will be documented. Questions such as why not PWR?, and why a Gas Turbine cycle? will be addressed.

2.2 General Considerations

Figure 2.1 is a binary event tree which summarizes the several decisions which had to be made to narrow the choice of system characteristics prior to development of a detailed conceptual design. It is not unique since several different diagrams could have been constructed in which the same (or other) trade-offs were considered in different order. However it does provide a convenient and self-consistent outline around which the key points at issue can be discussed.

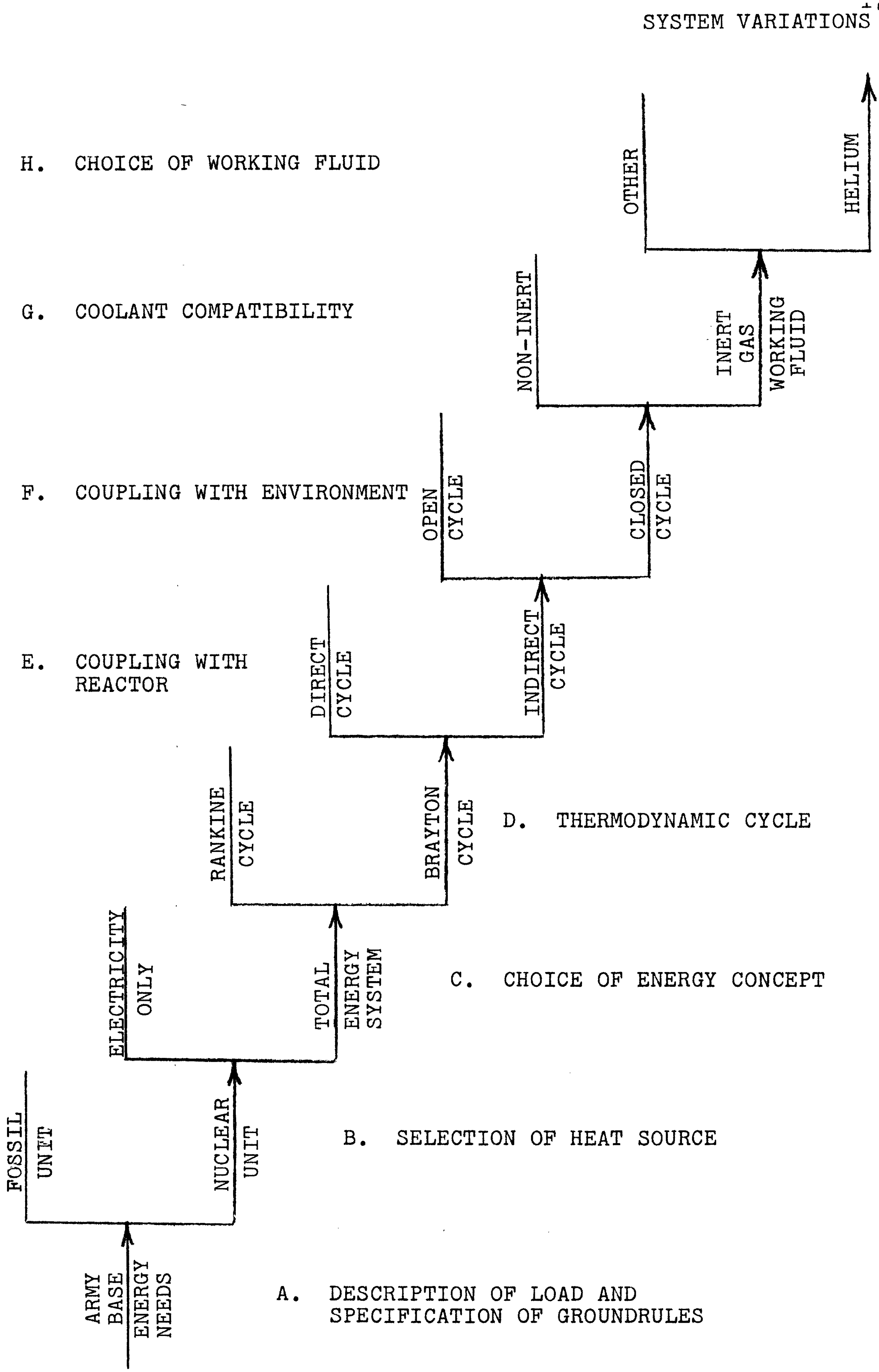

Fig. 2.1 DECISION TREE FOR SPECIFICATION OF SYSTEM CHARACTERISTICS

The initial several issues to be decided: selection of heat source, energy concept and power cycle, can best be dealt with as a package by consideration of the overall interrelation of technical characteristics and economic prospects.

Given the needs defined by the groundrules set forth in Chapter 1, one is led to prefer a high-temperature gas-cooled reactor driving Brayton cycle turbomachinery (the HTGR/GT) based upon the economic advantages stemming from two key technical considerations:

(a) The Brayton cycle can deliver "waste" heat at a temperature useful for hot-water type thermal utility systems, while with Rankine-cycle-based systems one must divert prime steam for utility use, thereby requiring an inherently larger system thermal power rating. (see Appendix B)

(b) Since Brayton cycle efficiency can match or surpass that of the Rankine cycle only above about 1400°F turbine inlet temperature, only the HTGR of currently proven reactor types can exploit this advantage.

Table 2.1 compares reactor ratings determined using the power conversion system models of Ref. (M2), which consider use of a HP turbine to extract power to the extent practicable before diverting steam to the utility system heat exchanger. We see that other reactor system ratings must be larger than that of the HTGR/GT. Since both plant capital cost and fuel

Table 2.1

Reactor Ratings Required for Equivalent

NTU Service (100 MWe, ∿ 200 MWth*) (M2)

System Type	Thermal Rating (MW)
HTGR/GT	284
Fossil-Fired Gas Turbine (FFGT)	313
HTGR/Rankine	358
Fossil-Fired Rankine	383
PWR	550

* peak electrical and thermal loads are not coincident

Table 2.2

Predicted Cost of Total Energy Alternatives in 1974 Dollars (M1)

	Electric Energy (mills/KWhr)	Thermal Energy ($/MBTU)
HTGR/GT	14.0	1.43
FFGT	14.2	1.48
PWR/Rankine	16.9	1.63
Coal/Rankine	24.6	2.77
Oil/Rankine	25.2	2.84
Outside Purchase Option (Industrial-user basis)	28.0	--

Basis: 11 $/bbl oil, 30 $/ton coal, 80% plant capacity factor, 10% cost of money, 30 year plant life consumption are roughly proportional to thermal rating, the HTGR/GT has an inherent and overriding economic advantage. Note, however, that this preference follows from the total energy nature of the application -- for an all-electric or all-thermal load the PWR/Rankine and HTGR/GT systems would have comparable thermal ratings, and the PWR would be preferred in the size range pertinent to military applications.

Table 2.2 illustrates the preceding conclusions in a more specific fashion, in terms of the results of the economic intercomparisons developed and reported in Ref. (M1). As can be seen, the fossil-fired gas-turbine (FFGT) is the most serious challenger to the HTGR/GT as the preferred system. The various pro's and con's of this alternative and appropriate recommendations for future investigations of this concept are spelled out in Ref. (M1). In the present instance we will focus on the HTGR/GT system exclusively, the preferable nuclear alternative.

A final point to be noted is that the PWR, while less attractive than the HTGR/GT in the present application, is not disqualified by a prohibitive margin. However this system has been more thoroughly analyzed in recent studies by ORNL (K1) NUS (N3) and the IAEA (I1), which again motivates our exclusive preoccupation with the HTGR/GT in the present evaluation.

2.3 Direct vs. Indirect Cycle

Considerable attention was devoted to determining whether a direct or indirect cycle should be employed. Tables 2.3 and 2.4 summarize the spectrum of considerations which led to the final decision to adopt the indirect cycle. The overriding factor proved to be the greater assurance of maintainability/reliability for the indirect cycle -- particularly important where a single reactor system is responsible for the bulk of both the electric and thermal requirements for a given installation.

Calculations of fission product release and plateout indicated that contact doses of on the order of 20 Rem/hr could be experienced on direct cycle turbomachinery unless ultra-high core integrity were achieved. Since neither the capability for decontamination under in-service conditions nor its lack of detrimental metallurgical effects over plant life has yet been demonstrated and since it was considered important to employ state-of-the-art design in an area as vital (and as expensive to research) as core and fuel element design, the use of the indirect cycle was considered to be sufficiently attractive for the present application. Some thought was given to full-flow filtration, and while found to be marginally feasible from a technical standpoint, it represents only a partial solution to the problem because radioactive noble gases can traverse the filter and contaminate the primary loop with their daughter products.

Table 2.3

Advantages of Indirect Cycle

1. Ease of maintenance; no need for on-site decontamination facility.	(a) Except perhaps for tritium the turbine plant will be uncontaminated and also outside the containment.
	(b) Machinery accessible with reactor at power; less compact arrangement is practicable
2. Better protection of reactor core during normal and accident conditions.	(a) Core is not pressure-cycled during use of helium inventory control.
	(b) Turbine plant accidents (e.g., blade ejection) do not rupture and depressurize primary system.
	(c) Precooler leaks can not introduce water into primary system
	(d) Turbine bearing lubrication and shaft-seal problems do not affect primary coolant.
3. Prototype experience available for all constituent components and systems.	(a) Oberhausen II, a 50 MWe fossil-fired helium-Brayton system, will be on line in 1975.
	(b) Location of turbine plant outside containment favors use of familiar horizontal turbomachinery.
	(c) Primary circuit similar to Peachbottom, Dragon, other gas-cooled reactors.
4. Smaller containment is practicable.	(a) Power conversion equipment located outside containment.
	(b) Higher post-accident ambient pressure improves heat removal
	(c) Can help hold down capital cost.

Table 2.3 (continued)

5.	Provides a more flexible overall system design	(a)	Steam generator can be substituted for intermediate HX unit where large steam demand occurs (e.g., industrial facility).
		(b)	It may eventually be possible to substitute chemical reactio[illegible] units for HX units to produce synthetic fuel.
		(c)	Primary coolant and turbine plant coolant chemistry can be individually optimized.
		(d)	Easier to substitute more advanced power plant cycles later: add intercooling or bottoming cycle.
6.	Potentially simpler reactor system control.	(a)	Turbine plant transients isolated from core, or their effect dampened.
		(b)	Isolates other loops from transients in one loop.
7.	Less expensive turbomachinery and associated plant.	(a)	Entire system need not be built to nuclear standards.
		(b)	Eliminates need for intermediate loop between precooler and utility system.
8.	Switch to direct cycle in second generation plants is probably feasible.	(a)	For example: adopt Shippingport-type multiple-vessel containment.
9.	Permits use of hydrogen-cooled generators	(a)	No need to put hydrogen system inside containment.
10.	Compatible with supplementary or alternative fossil-firing: the <u>same</u> power conversion system design can be used in both nuclear and fossil installations; it may be possible to install fossil-fired HXer in parallel with the reactor IHX (see Ref. U1)	(a)	One can use either nuclear or fossil heat, or both: Oberhausen II is a fossil-fired TE system.
		(b)	Some European closed GT systems burn pulverized coal-a fossil alternative of interest to DOD; but essentially any fuel can be accommodated.
		(c)	May offer an economic approach to providing backup power during nuclear unit outages, or extra power during peak demand

Table 2.4

Disadvantages of Indirect Cycle

ITEM	COMMENT
1. Slightly lower cycle efficiency. (About 3% less than GA or European large direct cycle HTGR/GT designs)	(a) Turbine inlet temperature about 75°F lower than direct cycle; temperature loss across the IHX is slightly greater than that lost in the added intermediate loop in the direct cycle, but for every 20°F loss at the turbine inlet the cycle loses 0.5% efficiency whereas per 20°F increase at the compressor inlet the cycle loses 1.5%.
	(b) Primary circulators consume about 7 MWe of electric output (equivalent in effect to another 100°F core outlet ΔT).
	(c) Forces consideration of increased core outlet temperature, greater reliance on absorptive air conditioning.
2. Requires expensive intermediate heat exchangers.	(a) Design studies indicate that high performance design is possible at reasonable cost.
	(b) Development program is straightforward and success assured if derating of operating temperature is a permissible fall-back position.
3. Departs from mainstream of currently active reactor-oriented GT system development	(a) Most of this development, particularly that by General Atomic is for much larger systems: 250 MWe/loop.
	(b) Reactor core and most auxiliary system designs are not strongly affected.

Table 2.4 (continued)

4. Rules out use of PCRV	(a) Places strong emphasis on protection against loss of coolant accident.
5. Constrains system design pressures: to insure against radioactive contamination of turbine plant its pressure should exceed that of reactor system.	(a) Peachbottom primary pressure (305 psig) is less than Oberhausen II turbine plant pressure (410 psig); but later HTGR's are higher (710 psig).
	(b) If inventory control is used turbine plant pressure will be decreased during part-load operation.
6. Complicates helium handling and purification system design.	(a) Must isolate (hence duplicate) turbine plant helium systems to prevent cross-contamination.

2.4 Choice of Working Fluids

The final two branches in Fig. 2.1 relate to the choice of working fluid in the primary and secondary systems. Here there is little room for debate as helium is the coolant of choice in all recent industrial applications of high temperature reactor technology. Extensive British experience with CO_2 appears to put the upper limit on useful operating temperatures with that fluid at about 1200°F -- too low to permit its use in gas turbine power cycles. Other non-inert gases react even more strongly with graphite or structural metals at the high temperatures of interest. Appendix A outlines a theoretical treatment which shows in a concise manner the superiority of helium to other inert (and non-inert) gases with respect to heat transfer and transport capabilities and as the working fluid in thermodynamic cycles.

If helium is used in both the reactor and turbine plants, then quite naturally both cycles are closed. A system in which a helium primary circuit was used to drive open cycle (air) Brayton turbomachinery has been investigated and found to offer no worthwhile advantages (U1). Thus we were led to select a helium/helium arrangement. This had the added advantage that the secondary system pressure could be kept higher than the primary system pressure, assuring in- rather than out-leakage and thereby providing greater assurance against release of radioactivity.

2.5 Applicable Prior Experience

Given the decision to employ an indirect cycle, several other system characteristics follow immediately, leading to specification of a plant having a strong resemblance in its various parts to three well-known prototypes: the Peachbottom I Reactor, fueled by a Fort St. Vrain type core, driving Oberhausen II turbomachinery. All of these systems in turn have been evolved from many years of related experience which we will not attempt to review here.

Use of the indirect cycle all but precludes economic application of the integral primary system concept because of the large physical size of gas-to-gas heat exchangers. (Designers of large direct-cycle HTGR/GT plants have also found it difficult to cram the entire primary system into a non-oversized reactor vessel). Therefore the principal advantage of using a prestressed concrete reactor vessel (PCRV), as in the large central station HTGR designs, is removed. Moreover, the plant size under discussion is still small enough to permit use of a shop-fabricated steel pressure vessel which can be transported to the site. Pursuit of this line of reasoning to its logical conclusion leads to a general reactor system design resembling that of the Peachbottom I Reactor, whose major characteristics are summarized in Table 2.5. In the seven years of operation prior to its recent decommissioning this

Table 2.5

Characteristics of Peachbottom I HTGR*

Rating:	40 MWe; 115.5 MWth
Net Efficiency:	34.6% (Rankine cycle)
Coolant:	Helium, outlet temperature = 1342°F; 350 psia, Flow rate = 446,900 lb/hr
Moderator:	Graphite
Fuel:	U(93% enriched) and Th carbide dispersed in graphite
Refueling schedule:	batch, 3 year cycle, off-load refueling
Primary circuit:	2 loops each containing one blower and one steam generator
Reactor Pressure Vessel:	14 ft I.D. x 35.5 ft overall height ASTM A212 grade B carbon steel.
Containment:	Cylindrical steel shell, 100 ft dia., 162 ft overall height
Owner/Operator:	Philadelphia Electric Co.
Designer:	General Atomic
Start of construction:	February 1962
Reactor Critical:	March 1966
Full Power Operation:	May 1967
Final Shutdown:	November 1974
Lifetime Forced Outage Rate:	= 5.4% (from Ref. N2)
Capital Investment:	$\$28.1 \times 10^6$ (= $703/kw) 17.0×10^6 R&D $\$45.1 \times 10^6$ Total

*Except where otherwise indicated, data from Ref. (D2) reactor performed in an exceptionally reliable manner, testifying to the basic soundness of its various design features.

The power conversion unit must inevitably have many features in common with the Oberhausen II plant in Germany -- a 50 MWe fossil-fired turbine plant. This system has been designed to simulate a 300 MWe unit for subsequent service with large HTGR's. Thus in the present application some refinement in design to reduce size is possible; this will also permit packaging the unit as a single transportable module. Table 2.6 summarizes the characteristics of Oberhausen II, which is currently undergoing commissioning exercises prior to operation as part of a total energy system. **Fig. 2.2 shows an isometric view of the unit.**

Another point to note is that even the most unique feature of the present concept -- the intermediate heat exchangers, in which we propose to use Incoloy tubing -- has precedent applications already in being:

(a) The fossil-fired Oberhausen gas heater uses Incoloy tubing at a temperature and differential pressure close to those of present interest, and in a more corrosive atmosphere on the fire-side.

(b) Incoloy is currently specified for the superheater region of steam generators in large HTGR's at a pressure much higher than the present application,

Table 2.6

Summary Description of Oberhausen II

1. System Characteristics.

Plant capacity:	50 MWe (net) plus 53.5 MW heat*
Type of cycle:	Indirect, regenerative, one stage of intercooling.
Heat source:	Fossil fired heater, coke oven gas.
Working fluid:	Helium
Helium mass flow rate:	187 lb/sec.
Pressure losses:	10.4%
Control methods:	He inventory, compressor bypass
Thermal efficiency:	32.6% gross, 31.3%** net

2. System Components.

Turbomachinery	Two shaft/reduction gear coupled, oil lubricated-Labyrinth shaft seals.
HP compressor;	5500 rpm, 15 stages, blade length-2.83 inches (inlet) 2.1 inches (outlet)
LP compressor;	5500 rpm, 10 stages, blade length-4.06 inches (inlet) 3.35 inches (outlet)
HP turbine:	5500 rpm, 7 stages, blade length-5.9 inches (inlet) 7.87 inches (outlet)
LP turbine;	3000 rpm, 11 stages, blade length-7.87 inches (inlet) 9.84 inches (outlet)
Recuperator:	Tube bundle cross counter flow, 17500 tubes, 0.47 inches O.D., 0.04 inches thickness, 73.8 ft total length, 14.8 ft shell dia., 1x10^5 sq. ft surface area, 87% effectiveness, 130 MWth duty.
Precoolers (2):	Externally finned tube, 11800 ft^2 total surface area per unit
Intercoolers (2):	Externally finned tube, 42000 ft^2 total surface area per unit.
Helium heater:	Sulzer hot gas generator plus Ljungström air heater, Incoloy 807 tubing, 92.2% efficiency, 57.4 ft length, 41 ft ft height max. tube wall temp. 1472°F.

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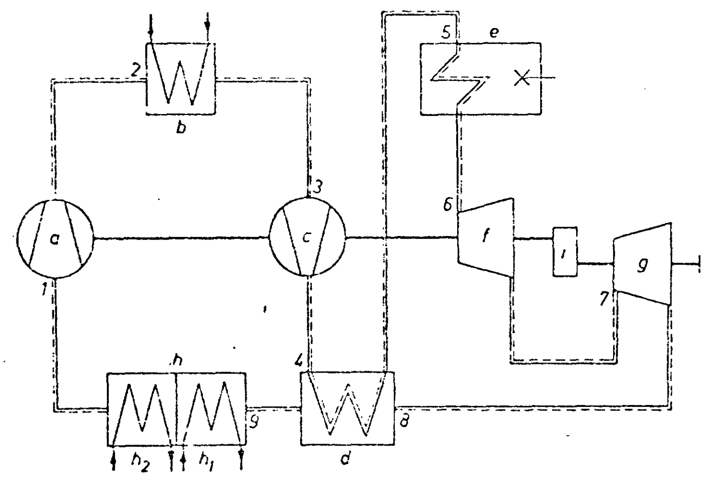
*utility delivery temp. 230°F, utility return temp. 104°F

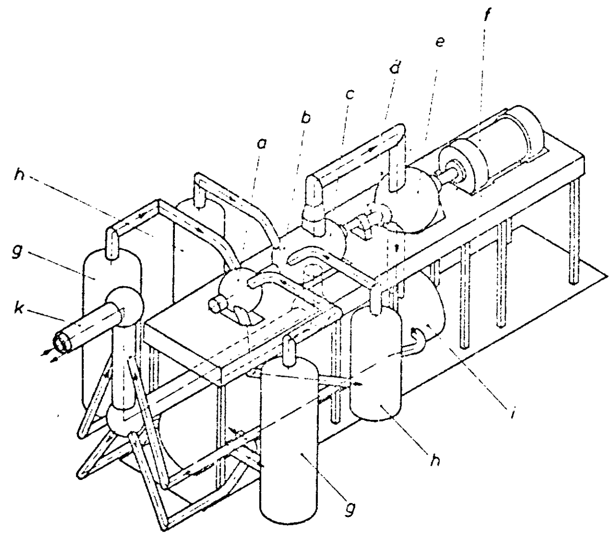
**based on total fossil energy supplied.
33.94% efficiency based on heat transferred.

Data from Refs. (B2,B3,B4,)

Table 2.7 Thermodynamic Cycle for Oberhausen II



	Inlet Temp.(°F)	Inlet Press.(psia)
a. LP Compressor	77	152
b. Intercooler	181	224
c. HP Compressor	77	223
d. Recuperator (cold)	257	416
(hot)	860	156
e. Fossil-Fired Heater	783	408
f. HP Turbine	1382	391
g. LP Turbine	1076	239
h1. Precooler(heating part)	336	154
h2. Precooler(cooling part)	113	153
i. Gear		



a. LP Compressor

b. HP Compressor

c. HP Turbine

d. Gear

e. LP Turbine

f. Generator

g. Precooler (2)

h. Intercooler (2)

i. Recuperator

k. Concentric Double Duct

Fig. 2.2 Isometric View of Oberhausen II at a slightly lower temperature and in a more corrosive environment on the steam side.

(c) Some of the newer PWR designs use Incoloy steam generators, at much higher pressure but considerably lower temperature than required by the HTGR/GT.

In view of these considerations and of the fact that even higher temperature heat exchangers are currently being developed for chemical reaction units to be used with HTGR's, the success of the present design appears assured even without an extensive R&D program.

Finally, a considerable amount of prototype experience exists in the area of core design, for which we rely heavily upon the Fort St. Vrain Reactor, currently in its start-up test program. Further discussion of this area will be deferred until Chapter 5.

2.6 Synopsis and Outline

In this second chapter we have discussed the considerations which led to selection of a reference design concept involving a total energy system centered around a HTGR reactor driving (indirectly, via intermediate heat exchangers) closed cycle gas turbines, using helium as the primary and secondary working fluids. In Fig. 2.1 we illustrated a binary decision tree for the major points at issue. The primary factors in support of the path shown in the figure can be summarized as follows:

1. Army base energy needs are projected to reach 100 MWe at a number of installations in the post 1985 time frame -

   (a) a size where nuclear reactors become competitive with fossil fired systems at current fuel prices, particularly if

   (b) total energy systems are used, to take maximum advantage of the high-capital-cost, low-fuel-cost nuclear system by maintaining a high overall system load factor.

2. The Brayton cycle has an inherent advantage over the Rankine cycle in total energy systems because it can maintain high efficiency while discharging high temperature "waste" heat.

3. An indirect cycle is selected primarily on the basis of assured maintainability,

4. The turbine plant employs a closed cycle because of the improved overall efficiency and economy possible with such systems.

5. An inert gas working fluid is desirable to permit high operating temperatures without excessive corrosion, and finally

6. helium is preferred because of its overall superior combination of physical properties.

In the next chapter a detailed conceptual design of the HTGR/GT system will be developed.

Chapter 3

POWER CONVERSION SYSTEM

3.1 Introduction

In the previous chapter the concept of a high temperature gas cooled reactor indirectly coupled to a closed cycle gas turbine power conversion unit was identified as a preferred candidate for total energy applications.

In the present chapter the characteristics of the conversion system will be developed in some detail, including establishment of cycle state points, mass flow rates, pressure losses and the cycle energy balance. Several system variations will be considered, including the extent of regeneration and compression intercooling, and a number of key issues related to system performance and economic viability addressed, such as design of the intermediate heat exchanger, and the use of absorptive air conditioning and/or heat pumps for load tailoring.

3.2 Specification of Design Constraints

There are a number of considerations based upon experience and precedent which establish an envelope within which system optimization can be carried out. The groundrules spelled out in Chapter 1 constitute a further set of constraints. The items enumerated below translate these considerations into specific requirements imposed upon the present design:

(a) A mixed mean core outlet temperature of 1500°F is established based upon primary system materials performance limitations. The ultimate determinant here appears to be fission product release from the fuel: an increase in gas exit temperature of 50°F corresponds roughly to a factor of three increase in anticipated primary circuit activity (G2). The value selected is slightly higher than the Fort St. Vrain design value of 1445°F, but less than values already achieved in practice and proposed by others for future HTGR designs. Furthermore, since an indirect design has been chosen, the incentive to achieve ultra-low activity levels to facilitate turbomachinery circuit maintenance has been reduced.

(b) Heat sink reliability requirements lead us to specify a full-capacity on-site mechanical-forced-draft wet cooling tower. A partial-capacity auxiliary heat sink would be required in any event to permit delivery of rated electric power during periods of low thermal energy demand or thermal utility system outage. The type of cooling tower selected is based upon the lower cost of this type of unit. Because the chemical treatment used in the cooling tower water, and the aeration of the water, are not compatible with optimum water treatment of the utility system water, the cooling tower is coupled to the system by a heat exchanger.

Allowing for typical temperature differences in the heat exchangers involved this leads to a practical value of 130°F for the compressor helium inlet temperature in the power cycle.

(c) Because a hot-water-type utility system exhibits optimal performance for a supply temperature in the vicinity of 380°F we were similarly led to specify a precooler helium inlet temperature of 480°F.

(d) Considerations of reliability -- from both a power provision and a reactor protection point-of-view -- led to specification of two primary loops and turbo-generator plants. In this we arrive at a design quite similar to Peachbottom I. It is also convenient in that each loop then matches the Oberhausen II rating of 50 MWe. There is considerable incentive not to employ more than the minimum number of main loops because of the dominant role of plant capital cost in determining the economic viability of nuclear units in the small size of present interest.

Given this very general set of constraints it was then possible to proceed directly to consideration of more specific options available to the power conversion system designer.

3.3 Cycle Variations

There are many versions of closed cycle gas turbine systems which could be considered for the present application. Indeed, because we are employing an indirect, non-integral design there is considerably more leeway for variation than in the integral designs proposed for large HTGR/GT systems. The major variations to be discussed here are whether or not to use regeneration and intercooling. In addition the use of combined gas/steam turbine cycles is discussed.

3.3.1 Regeneration

The use of regenerative heat exchangers between the turbine exhaust and compressor discharge streams is a fundamental design choice. This unit is large and expensive -- about the same size as the IHX between the primary and turbine plant systems. Omitting the regenerator permits higher utility water temperatures, and makes it easier to drive the steam generator of a combined cycle. However without it the system efficiency is only about 25% and the optimum compression ratio is higher than in a regenerative cycle -- leading to higher design pressures or larger equipment sizes in the design tradeoff process. All things considered the use of a regenerator is desirable, as the penalties associated with its omission were prohibitive. Oberhausen II, for example, also employs regeneration, as do most large HTGR/GT design studies.

3.3.2 Intercooling

The use of intercooling between compressor stages is a well-known approach for achieving high thermodynamic efficiency in the Brayton cycle. In the present application, however, compression intercooling has a number of disadvantages:

(a) In an optimized cycle it results in a higher compression ratio and lower utility water temperature: if de-tuned to mitigate these characteristics one loses the efficiency advantage for which it was installed in the first place.

(b) For optimum performance the coolant water discharge temperature from the intercoolers is sufficiently low that the thermal energy it contains is truly waste heat. Thus we enhance electrical utilization at the expense of thermal utilization.

(c) The intercoolers are moderately expensive and increase system size, complexity and vulnerability to malfunction.

In view of the above it was decided to omit compression intercooling. Note that this departs from Oberhausen II practice.

3.3.3 Combined Cycles

Considerable attention has been given of late to combined gas turbine/steam turbine cycles. Open cycle fossil-fired gas turbines coupled to a waste heat boiler driving a Rankine bottoming cycle are presently being marketed for utility service. Arrangements suitable for use with closed cycle systems have been published (B6,M4). General Atomic has studied an isobutane bottoming cycle for use with HTGR/GT systems (S1). We have rejected these alternatives in the present instance for several reasons:

(a) After a point, high efficiency _per se_ is no longer attractive. In particular, in a total energy system with a high thermal demand the steam turbine would only be operated during infrequent peak electrical load situations.

(b) In the present application system reliability considerations have led us to require a 50% capacity fossil-fired gas turbine backup unit -- which can also provide peaking service.

(c) Furthermore, within the design envelope established for the present applications the net improvement in system efficiency is quite small: only 38% (steam bottoming cycle) vs. our reference value of $\sim$ 33%; if isobutane cycles are developed, however, it may be possible to achieve efficiencies as high as 48%.

(d) Again, the system is complicated, and vulnerable to malfunction and misoperation.

In making the decision not to utilize a combined cycle we also considered, but rejected as unneeded, the potential capability of the system to store energy (as hot water or steam) for later conversion to electric power. Load-tailoring, as discussed in Appendix B proved to be more attractive.

Having established both by reference to previous design studies and scoping calculations of the various cycles discussed above that the simple regenerative cycle was preferable we are now prepared to consider specific items related to cycle optimization.

3.4 Cycle Optimization

The major task of the work described in this chapter involved determination of the specific state points of the turbine plant thermodynamic cycle. This work was carried out primarily through use of the in-house computer program CYCAL II, an updated and improved version of the program described in Ref. (H2).

Although pressure has a quite small effect on Brayton cycle performance when a nearly ideal gas such as helium is used, a useful first step was to select system operating pressure. Various factors were taken into account:

(a) Total capital costs are a trade-off between turbine plant cost which decreases as pressure increases (due to reduced size), and reactor vessel cost which increases with pressure. Simple economic models applied to direct cycle plants typically yield optimum pressures on the order of 800 psi, with a rather broad minimum.

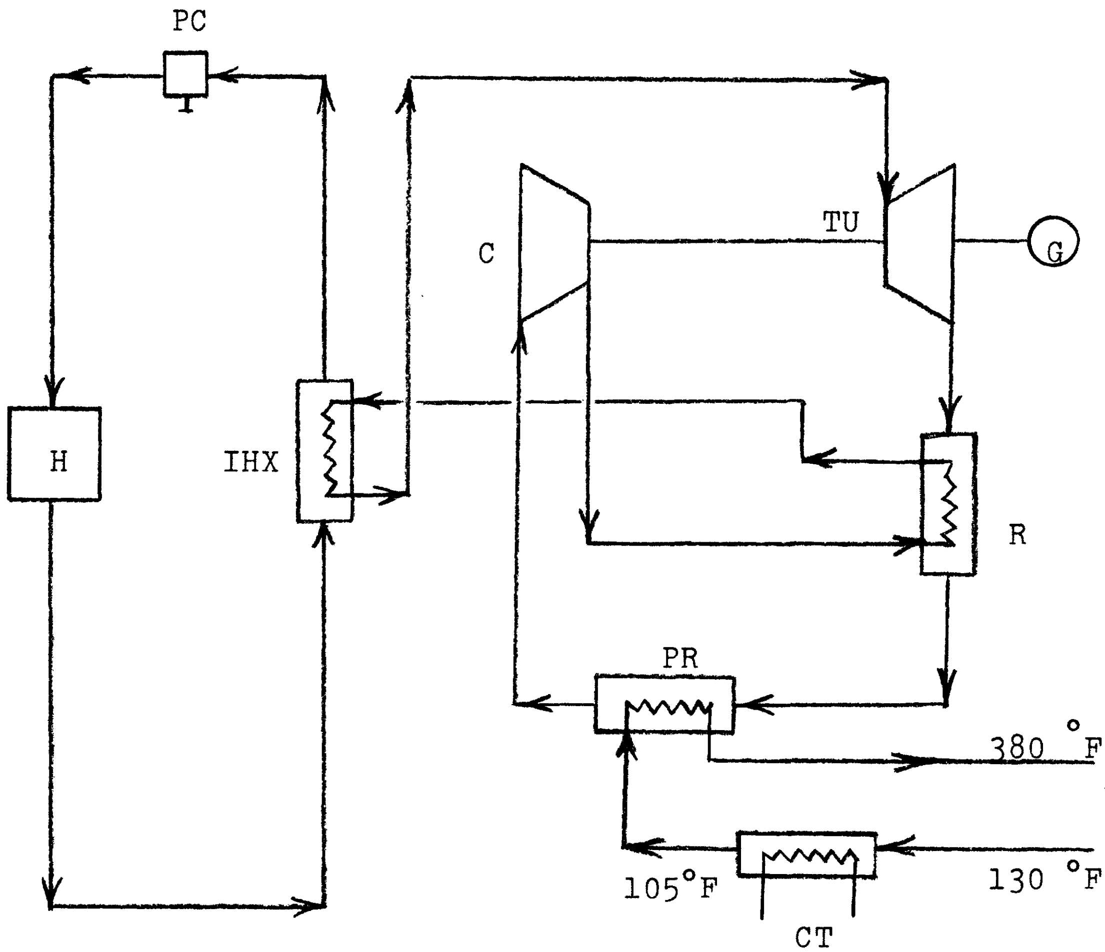
(b) It was considered highly desirable to have the turbine plant pressure exceed that of the primary circuit to provide positive assurance against leakage of radioactivity into the turbine plant. Since inventory control was also considered desirable, it was then necessary to make the pressure differential fairly substantial. This involved no economic penalty because one can exploit the advantage of decreasing turbomachinery cost at high pressure and decreasing reactor vessel cost at low pressure, as noted in (a). The limit on the differential pressure was felt to be high temperature creep of the IHX tube material, Incoloy 800.

(c) It was considered desirable not to move too far from proven state-of-the-art technology: Peachbottom I at 305 psig, Fort St. Vrain at 688 psig, Oberhausen at 408 psia.

(d) Moderately low primary system pressures also provide additional assurances against loss of coolant accidents and decreased likelihood of self-inflicted damage should the event occur.

In view of the above considerations a primary system pressure of 400 psia and a turbine plant pressure of 900 psia were chosen for the reference design.

With the foregoing preliminary points in mind we can proceed to a more detailed evaluation. Figure 3.1 shows the system layout considered and Table 3.1 lists the detailed input required for cycle performance computations: the entries all represent attainable state-of-the-art characterisitcs, some of which



C: Compressor

G: Electrical Generator

IHX: Intermediate Heat Exchanger

PC: Primary Circulator

U: Utility System

CT: Cooling Tower

H: Reactor

PR: Precooler

R: Recuperator or Regenerator

TU: Turbine

Fig. 3.1 Schematic Diagram of Reference Plant

Table 3.1
Input for Cycle Calculation

Desired Electrical Output at Bus:	100 MWe
Electric House Load:	3 MWe
Pressure Loss Coefficients	
Reactor to 1HX:	2.04%
1HX to Circulator:	0.50
Circulator to Reactor:	1.20
1HX to Turbine:	0.026
Turbine to Regenerator:	0.131
Regenerator to Precooler:	0.345
Precooler to Compressor:	0.70
Compressor to Regenerator:	0.30
Min. Precooler Inlet Temp.:	480°F
Min. Compressor Inlet Temp.:	130°F
Max. Regenerator Effectiveness:	0.87
Isentropic Turbine Efficiency:	0.90
Isentropic Compressor Efficiency:	0.88
Electrical Generator Efficiency:	0.88
Circulator Mechanical Efficiency:	0.99
Turbine Mechanical Efficiency:	0.985
Compressor Mechanical Efficiency:	0.985

represent a preview of design evaluations discussed later -- such as for the IHX.

Table 3.2 lists the constrained optimum conditions generated by the computer program. As can be seen a quite respectable efficiency of approximately 33% is developed at a rather modest compression ratio of 2.4. Note that use of an indirect cycle has cost us only 75°F in turbine inlet temperature. This loss is compensated for to some degree by a lower compressor inlet temperature (130°F) -- in a direct cycle an intermediate coolant loop would have had to be interposed between the utility water system and the turbine plant to prevent contamination of the utility water, which would have cost about 40°F added ΔT.

3.5 Heat Exchanger Design

The major heat exchangers required in the power conversion system (IHX, Regenerator, Precooler) are large and costly components which contribute in an important way to overall system cost (~110 $/Kw all-together). Their sheer bulk greatly affects plant layout and the complexity of the ductwork, most of which is concentric. Thus an important subtask of the present work involved optimization of their design using available cost function(M3) (P2) and thermal performance data on heat exchangers (F2).

Table 3.2
Cycle State Points

	Inlet Temperature (°F)	Inlet Pressure (psi)	Outlet Pressure (psi)
Reactor	953.31	403.33	400.00
1HX (Primary)	1500.00	399.63	396.82
Circulator	942.79	396.76	403.49
1HX (Secondary)	864.66	902.95	900.00
Turbine	1425.00	900.00	383.12
Regenerator (hot side)	934.87	383.09	380.23
Precooler	480.00	380.18	378.83
Compressor	130.00	378.79	906.30
Regenerator (cold side)	409.89	906.27	903.00
Thermal Efficiency (%)	33.30		
Compression Ratio	2.40		
Turbine Pressure Ratio	2.35		
Gas Flow Rate (lb sec)	444 (Primary) - Total (sum of 2 loops)		
	441 (secondary) - Total (sum of 2 loops)		
Utility Water Temperature (°F)	380		
Heat-to-Electric Power Ratio	1.8347		
Net Fuel Utilization	0.9247		

The heat exchangers in question are all of the axial counterflow, shell-and-tube type. The high pressure helium gas was put inside the tubes in the 1HX and recuperator to provide greater protection against tube failure and to reduce the shell cost. In addition, in the 1HX this meant that plugging of leaking tubes could be done from the uncontaminated tube side.

Table 3.3 presents the detailed results of the extensive, iterative design study carried out on the heat exchangers in question. It should be noted that while the analytical optimum pitch-to-diameter ratio was determined to be 1.2, the value of this parameter was set at 1.35 to facilitate the difficult process of welding tubes into such a close-packed matrix.

Of particular interest in Table 3.3 is the cost of the 1HX units (1.7 million each), which is felt to be a rather modest price to pay for the advantages they offer. In addition, while one would have to carry out a parallel detailed direct cycle design to be certain, it is felt that this cost and more is recovered elsewhere in the plant because of the lower primary circuit pressure, elimination of intermediate loops on the utility water side, smaller primary containment, etc.. It is also evident that the 1HX is nothing more than a special type of regenerator, distinguished only by its higher service temperature. In view of this latter observation it was not considered necessary to carry the design process further: a number of

Table 3.3
Heat Exchanger Characteristics

General:

Type:	Shell and Tube
Flow:	Axial Counterflow
Geometry:	Square Tube Bundles

Specific:

	1HX	Regenerator	Precooler
Heat Transfer Area, sq. ft.	59201.0	52001.2	71751.9
Number of Tubes	14699	23542	29164
Length, ft.	35.78	27.66	30.81
Diameter, ft.	6.6	5.9	6.6
Heat Transfer Coefficient (BTU/hr-ft^2-°F)	124	156	148
Log mean Temp. Diff. (°F)	75	67	40
Tube Inside Dia. (in)	0.43	0.305	0.305
Tube Outside Dia. (in)	1/2	3/8	3/8
Pitch to Diameter Ratio	1.35	1.35	1.35
Tube Material	Incoloy 800		
Shell Material	Incoloy 800		
Thermal Duty (MWth)	185	150	118
Estimated Cost ($M)*	1.7	1.7	2.0

*In 1985 dollars

excellent modular design concepts have been published for regenerators -- Swiss and GA designs are shown in Figs. 3.2 and 3.3, and a GA precooler design is shown in Fig. 3.4. We have already called attention to the use of Incoloy tubing under service conditions which bracket those of the present application.

3.6 Turbomachinery

Although we rely heavily upon Oberhausen II turbomachinery design, and their operating experience will also be of exceptional interest, there are several factors which rule out use of a "carbon-copy" unit in the present application:

(a) Oberhausen, while rated at 50 MWe was designed to simulate a 300 MWe unit and is therefore larger and more expensive than need be for a system optimized around 50 MWe service only.

(b) We have selected a design operating pressure of 900 psia, about a factor of two higher than Oberhausen.

(c) We have omitted the compressor intercoolers and operate our system at a higher overall heat-to-power ratio.

Nevertheless with Oberhausen experience in hand it should be a fairly straight forward task for turbomachinery manufacturers to design a unit for the present application. Several have

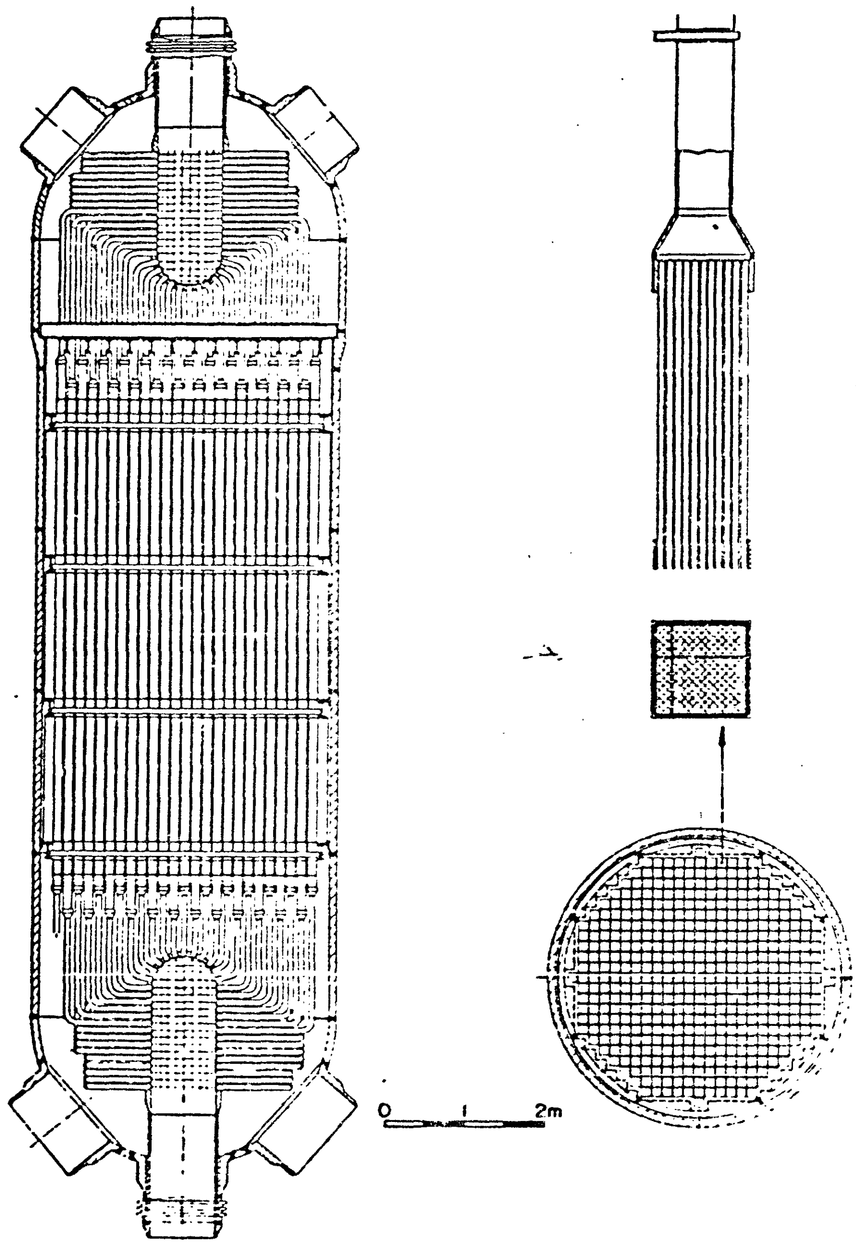


Fig. 3.2 Swiss Federal Institute's Proposed Nuclear Gas Turbine Regenerator (L1)

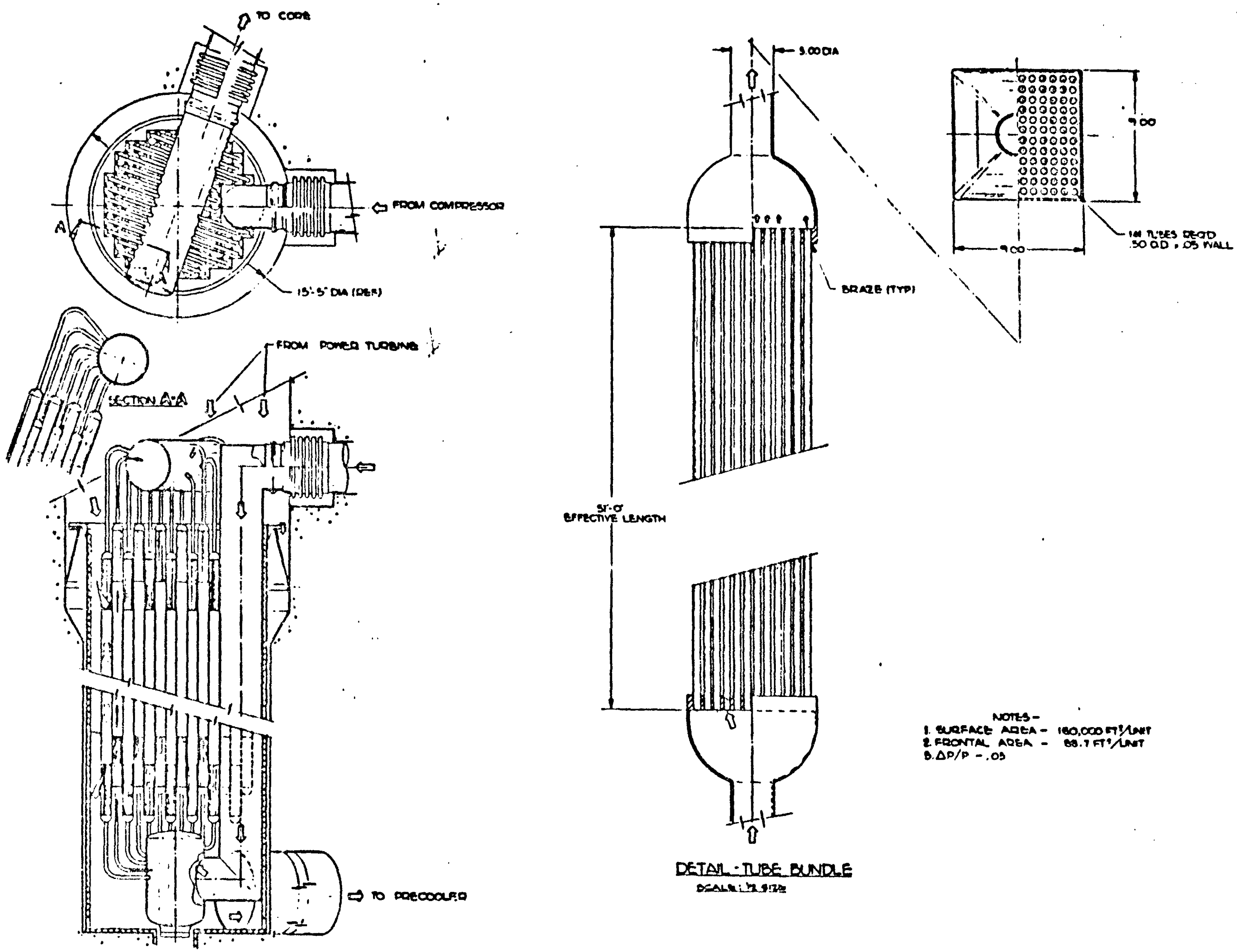


Fig. 3.3 General Atomic HTGR/GT Regenerator (G3)

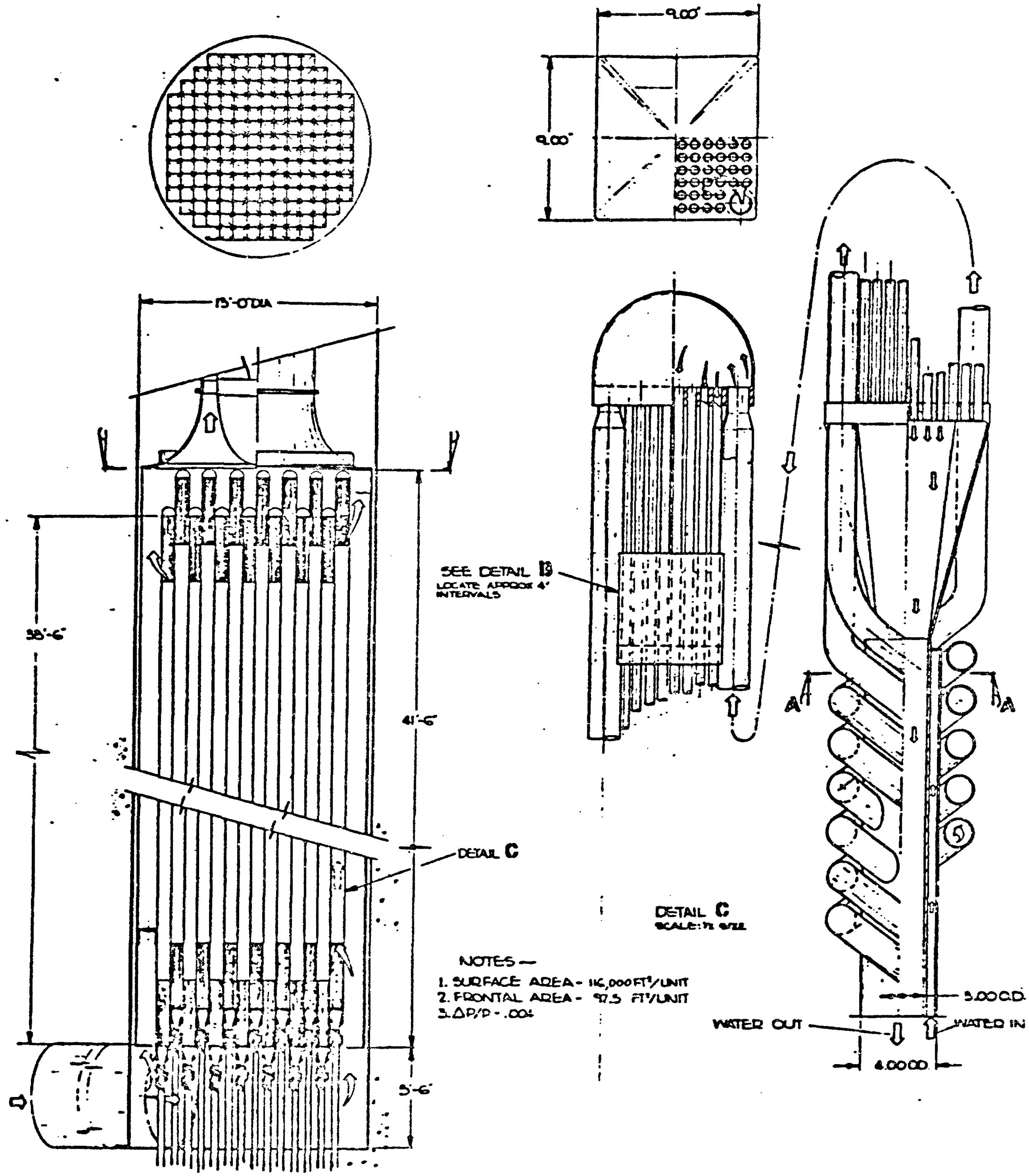


Fig. 3.4 General Atomic HTGR/GT Precooler (G3)

already participated in design studies with GA on the larger units required for central station electric utility service. Following the practice adopted for these larger designs we will also employ a single shaft arrangement, but in the present case we prefer a geared-shaft arrangement following Oberhausen (and conventional gas turbine practice) as it allows a better match between requirements and machine characteristics because the compressor and high pressure turbine are not restricted to the same speed of rotation. Single shaft machines, moreover, have a better inherent protection against overspeed during loss-of-load transients than split shaft machines due to the natural damping effect of the compressor and the greater system inertia. A single shaft design also has fewer components - hence lower capital costs, and a lower starting motor power suffices.

On the other hand, the single-shaft arrangment is somewhat less flexible from a control standpoint, and requires higher bypass flow rates than the split-shaft design.

Axial flow machines are specified because of their high efficiency, design flexibility and ease of manufacture.

One additional design objective is worthy of note. Considering the component sizes, weights and the compact layout possible, an attempt should be made to shop-fabricate the turbine plant loops as modules which can be transported to the site as a unit.

3.7 System Control and Load Tailoring

Control of the present HTGR/GT unit differs in two important respects from that of the larger units being designed for central station electric utility system service:

(a) the subject unit is load following, not base-loaded, since we have assumed stand-alone service and (at least for the reference design) no storage provisions for electric or thermal power.

(b) the subject unit is a total energy plant, hence must simultaneously satisfy both an electric and a thermal demand.

In spite of these differences, proven techniques are available for closed-cycle turbine plant control which appear adequate for the proposed application, namely:

(1) Part-load following:

rapid response -- compressor bypass control

slow response -- helium inventory (pressure level) control

(2) Loss-of-load transient:

compressor bypass control to limit turbine overspeed to $\leq$ 120%

(3) Emergency shutdown:

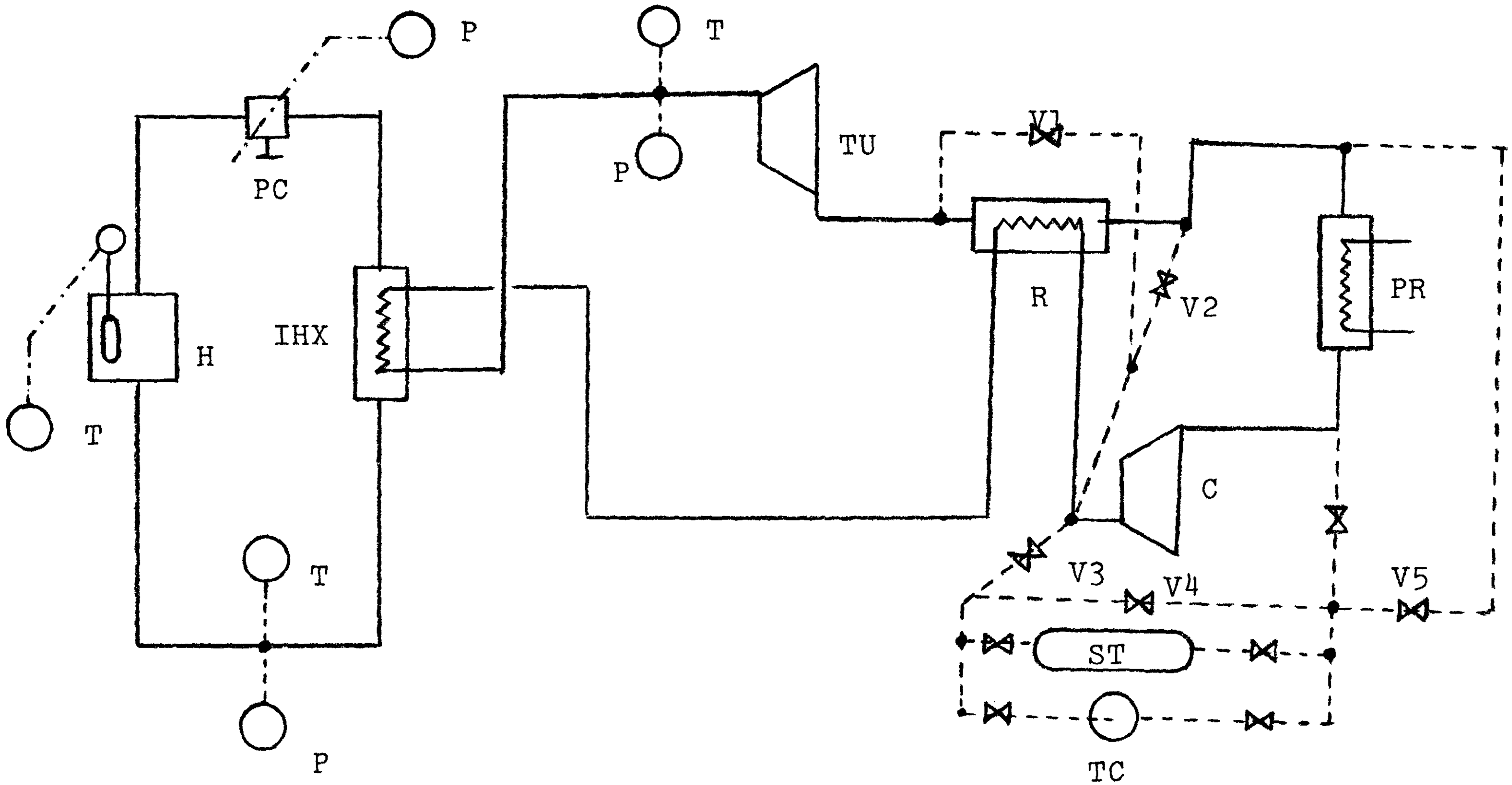
shutdown bypass valve

(4) Normal startup and shutdown:

starting motor or alternator with starting system electric power provided from grid, standby FFGT, or emergency diesels.

Additional flexibility can be achieved by the ability to vary reactor outlet (hence turbine inlet) temperature and the precooler water (hence compressor inlet) temperature. However the full range of control required can be obtained by the combined use of inventory and compressor bypass: the former permits nearly constant plant efficiency over a power range spanning more than 50% of design output and the latter permits a 40% increase in thermal energy available to the utility system at 20% bypass flow and essentially constant reactor power. Figure 3.5 shows a schematic of the power conversion system control features -- the resemblance to Oberhausen II is evident.

Although inventory control is employed, it should be noted that this option will only be used for slow variations in turbine plant operating conditions - over a minimum interval of half-a-day or so: for example, in preparation for weekend load reductions, or on a much longer time scale for adjustment to match seasonal load patterns or long term changes in the demand-growth spectrum. Two reasons motivate this restriction: the desire to limit pressure cycling of the 1HX units, and to use only the normal helium makeup and storage system - thereby avoiding the expense and complexity of installing a rapid gas transfer system and additional storage tanks.



C: Compressor

H: Reactor

IHX: Intermediate Heat Exchanger

PR: Precooler

PC: Primary Circulator

T: Temperature Measurement

P: Pressure Measurement

R: Regenerator

V1: Compressor Bypass/Power Shutdown Valve

V2: Compressor Bypass/Temperature Control Valve

V3: Helium Inventory Valve

V5: Recirculating Valve

ST: Helium Storage Tank]- Part of Normal

TC: Transfer Compressor]- Storage/Makeup System

Fig. 3.5 Circuit and Control Schematic of Power Conversion Unit

We will discuss the related subject of reactor system control in a later chapter, but call attention to the fact that the large heat capacity of the graphite in the reactor core make the constant-temperature control mode a natural choice for implementation.

Finally, the desirability of using absorptive air conditioning and/or heat pumps to tailor the load to match the normal output of the power plant, and in particular to achieve a better seasonal balance, is noted. A more detailed discussion on this aspect is presented in Appendix B.

3.8 Summary

Table 3.4 presents a condensed summary of the major features of the system developed up to this point. Perhaps the most interesting observation is that the use of an indirect cycle has involved a lower price than might at first be imagined: something on the order of four million dollars in direct component costs, but only 75°F in turbine inlet temperature. The efficiency is comparable to that of a CNSG-type PWR, but efficiency per se does not tell the whole story since the HTGR waste heat is directly useful, and since the ratio of thermal to-electrical demand will usually exceed 2:1 in most prospective applications -- making higher efficiency unnecessary.

Table 3.4
Summary of Design Features

Power conversion:	Indirect Brayton Cycle
Layout:	Non-integral, two loops
Capacity:	100 MWe, $\leq$ 200 MWth
Efficiency:	33%
Pressure Vessel:	Steel Pressure Vessel
Turbomachinery:	Single (geared) Shaft 50 MW Turboset per loop
Regeneration:	87%
Compressor Intercooling:	None
Max. System Temp.	1500°F (Primary) 1425°F (Secondary)
Max. System Pressure:	900 psi (Secondary) 400 psi (Primary)
Core Thermal Rating:	278 MWth

Chapter 4

PRIMARY AND AUXILIARY SYSTEMS

4.1 Introduction

In this chapter we will consider key aspects of the design of the various reactor systems. The general approach has been to rely heavily upon the concepts proven in practice in the operational prototype HTGR plants: Peachbottom I (recently decomissioned) Dragon and AVR. In addition the system designs of Geesthacht II, a HTGR/GT unit carried through to a point just short of construction, and the JAERI HTGR, presently on the drawing boards, proved to be useful. Various characteristics of interest from these units are summarized in Table 4.1.

In addition to the obvious savings achieved by not having to totally re-engineer the plant design this philosophy is particularly attractive because of the high reliability achieved in these prior designs. For example, Peachbottom's lifetime forced outage rate was less than 5%, and the AVR load factor was an outstanding 88% during 1973.

A final point to note is that even though the present design, unlike Peachbottom, AVR or Dragon, is mated to a gas turbine plant, the HTGR is not particularly sensitive to power cycle design. Both GA and European designers note in reference to their <u>direct</u> cycle designs that the balance of the plant other than the main coolant loops is largely unaffected. In the present instance the use of an <u>indirect</u> cycle gives even greater assurance of similitude.

Table 4.1 Summary Description of Small HTGR Reactors

	Peach Bottom I	Fort St. Vrain	Dragon	AVR	Geesthacht II	JAERI--VHTR	THTR
1. Power Conversion	Indirect Rankine	Indirect Rankine	Indirect	Indirect Rankine	Direct Brayton	Rankine/ Brayton	Indirect Rankine
2. Rated Output (MWt/MWe)	115.5/40	842/330	20/-	49/13.2	65/24	50/4.5 (Indirect) /7.0 (Direct)	750/300.
3. Net Eff. (%)	34.6	39.2	NA	27	37	21.36(Direc.)	40
4. Core Dimensions (ft.), Dia/Ht	9.16/7.5	19.5/15.6	3.5/8.3	9.8/9.8	8.1/6.9	15.3/8.2	18.3/ 19.7
Power Density (KW/liter)	8.3	6.3	14	2.3	6.4	4.7	6.0
Outlet Temp. (°F)	1380	1445	1382	1562	1355	1382	1382
Coolant Flow	Up	Down	Up	Up	Up	Down	Down
5. Reactor Vessel	Steel	PVRV	Steel	Steel	Steel	Steel	PCPV
Dimensions (ft), Dia./Ht	14/35	49/106 (Inside)	11.5/58	19/82.4	13/32.5	16.8/36.1	52/50 (Inside)
Operating Pressure (psig)	335	688	294	131	368	569	588
6. Containment	Vertical Cylindrical Steel Shell	NA	Steel Shell (Inner) Concrete (Outer)	Steel Cylinder	Concrete Building	Cylindrical Concrete Structure	NA
Dimensions (ft), Dia./Ht	100/162	76x120/161 (Outside)	66/106 (Inner)	52.5/134.5	100x100/ 140	114.8/213	81/83 (Outside)
Operating Pressure (psig)	-.2~8	-.2	10(Inner) .5(Outer)	.3	0	0	NA
7. Primary System Control Rod Drives	Hydraulic	Electrical Cable and Drum Type	Electrical Gear Box Wire, Drum	Pneumatic Slip on Guide Rails	Hydraulic	Electrical Magnetic Clutch and Drum	Pneumatic

Table 4.1 Continued

	Peach Bottom I	Fort St. Vrain	Dragon	AVR	Geesthacht II	JAERI--VHTR	THTR
Circulators	Centrifugal Blowers driven by 1.402 MW/ 3600 RPM Motors	Axial Flow Circulators driven by Exhaust Steam from HP Turbine	Six Blowers; Variable-frequency Squirrel--cage Motors rated 100 h.p. at 1200 r.p.m.	Two Blowers driven by Electrical Motor	NA	Centrifugal Horizontal Circulators driven by Induction Motor (Indirect)	Electric Motor driven Blower
Ducting	Concentric	NA	Concentric	NA	Concentric	Concentric	NA
8. Emergency Shutdown	Thermally Released Absorbers	Releasing Boron Carbide Spheres by Gravity Fall	None	Charging Graphite Spheres Containing Boron into Core	Reactor Scram and Shut-down of the Turbine	Boron Balls	Long-Stroke Pneumatic Absorber Rod Drive System
9. Emergency Shutdown Cooling	Natural Convection Cooling	Provision for Flooding Evaporator and Super-heater of Steam Generator	Guaranted Electrical Supplies, Natural Circulation in Cooling Circuit	Heat Loss through Vessel Walls	He and N_2 Emergency Cooling Systems (2)	Emergency Core Cooling System (5% of Full Flow)	Rupture-Safe PCPV
10. References	P1, D2	P4,D4,P5	D5, R1	D6	B1	J1,I2,S6	M10

4.2 Primary Coolant System

The primary coolant system flow diagram is shown in Fig. 4.1. Important features are:

(a) A downflow arrangment is used in the core to permit installation of seismic hold-down structure in the colder inlet plenum; this then leads to installation of control rod drives at the top of the core. In this regard we are following Fort St. Vrain practice rather than that of Peachbottom.

(b) Single isolation valves are installed in the core inlet/outlet ducting to permit isolation of (but not maintenance on) an inoperable loop at system design pressure. As in Peachbottom,concentric ducting is employed--but in our case bottom rather than top entry is called for.

(c) Two independent oversized purification systems are employed: they also provide the important function of auxiliary shutdown cooling. (Normal shutdown cooling uses the IHX; ultimate shutdown cooling relies upon heat loss through the pressure vessel. These aspects are discussed further in Section 4.4.)

Large diameter piping is required if coolant velocities are to be kept to an acceptable level (160 fps, as in Peachbottom). To accommodate a total core flow of 1.60 x 10^6 lb/hr in two main coolant loops the outlet pipe diameter must be 4.83 ft, which leads to an annular duct having an OD = 6.33 ft. Since Peachbottom has a larger temperature rise across the core (730°F vs. 540°F) flow accommodation is somewhat more

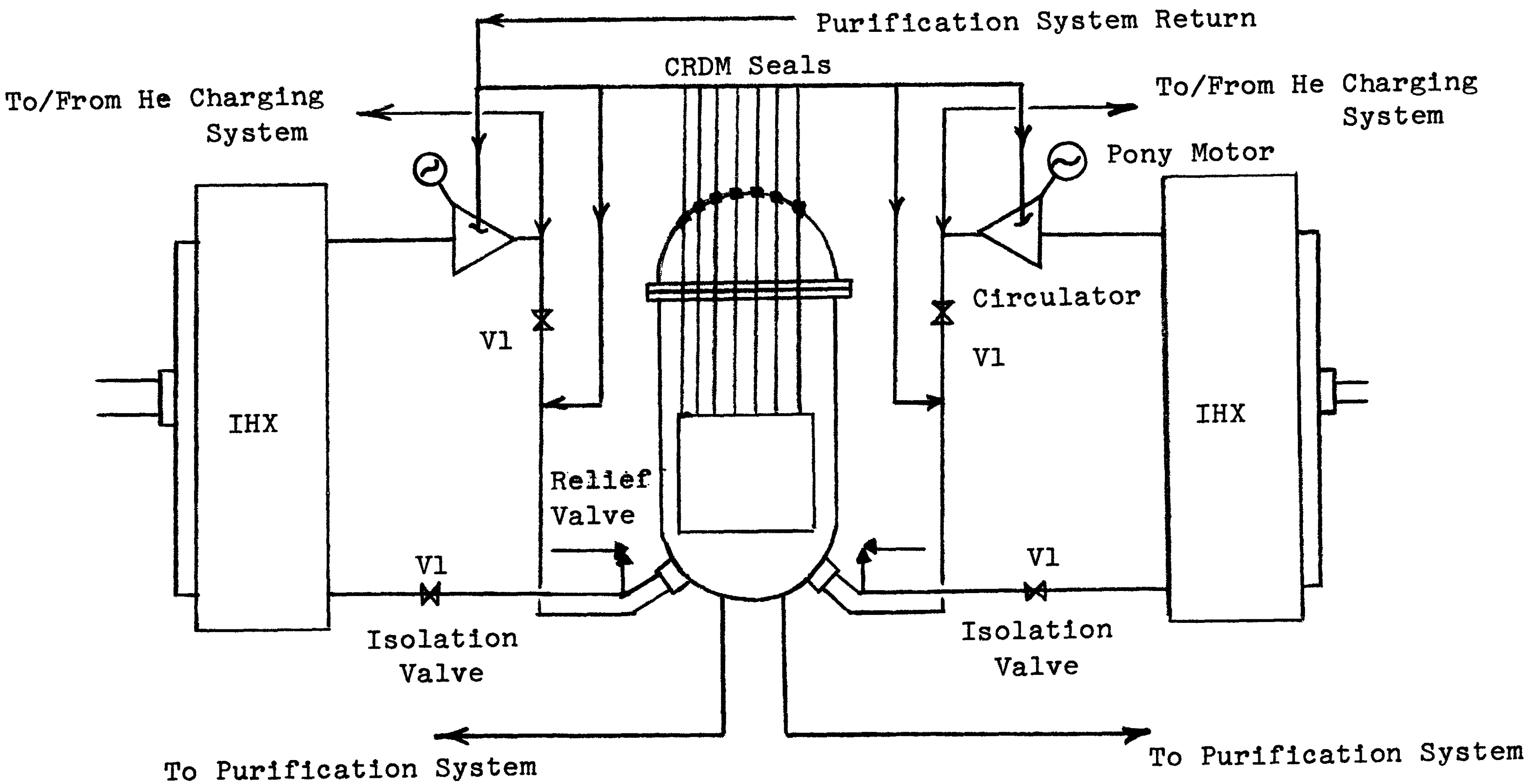


Fig. 4.1 Primary Coolant System Schematic

difficult despite our higher primary pressure (400 vs. 320 psi).

The primary coolant is pumped around the main loops by electric-driven single stage axial flow circulators--one in each loop. Each circulator drive is rated at 4.4 MWe (2.36x that of Peachbottom). Pony motors are used on each shaft to provide circulation during shutdown conditions.

As we have previously noted, a steel pressure vessel is employed since the size is within shop fabrication and transport-to-site capabilities. The size is roughly comparable to the vessels used for 1000 MWe PWR reactors, but the design pressure is lower by a factor of approximately five.

Table 4.2 summarizes the primary component characteristics.

4.3 Major Auxiliary Systems

The auxiliary systems to be discussed here include those supporting systems required to insure the safe and reliable operation of the overall reactor plant. They include:

(a) The helium purification system

(b) The shutdown and emergency cooling systems

(c) The refueling systems

Each of these key systems are described further in the subsections which follow:

4.3.1 Purification System

The purification system in the present design is unique only in that it is also used to provide auxiliary shutdown cooling; otherwise it differs in no essential way from the proven designs used elsewhere in both HTGR and LWR practice. Fig. 4.2 is a schematic diagram of the purification system flow path. Components on the right-hand side of the flowsheet employ the usual

Table 4.2

Main Component Characteristics of

Primary Coolant System

Circulators	
Type:	Electrical Motor Drive
Total He Flow Rate, lb/sec:	222 x 2
Number of Circulators:	2
He Inlet Temp., °F:	943
He Temperature Rise, °F:	10
He Outlet Pressure, psia:	404
He Pressure Rise, psia:	7
Circulator Power, MWe	4.4 x 2
Pressure Vessel	
Type:	Steel Vessel
Inside Diameter, ft	20
Height, ft	36
Thickness, in	6-7
Estimated Cost, $M	9.7
Ducts	
Diameter, ft	
Reactor to 1HX:	4.83
1HX to Circulator:	4.1
Circulator to Reactor:	4.08 / 6.33 (Annular)
Length, ft (Approx.)	
Reactor to 1HX:	15
1HX to Circulator:	15
Circulator to Reactor:	50

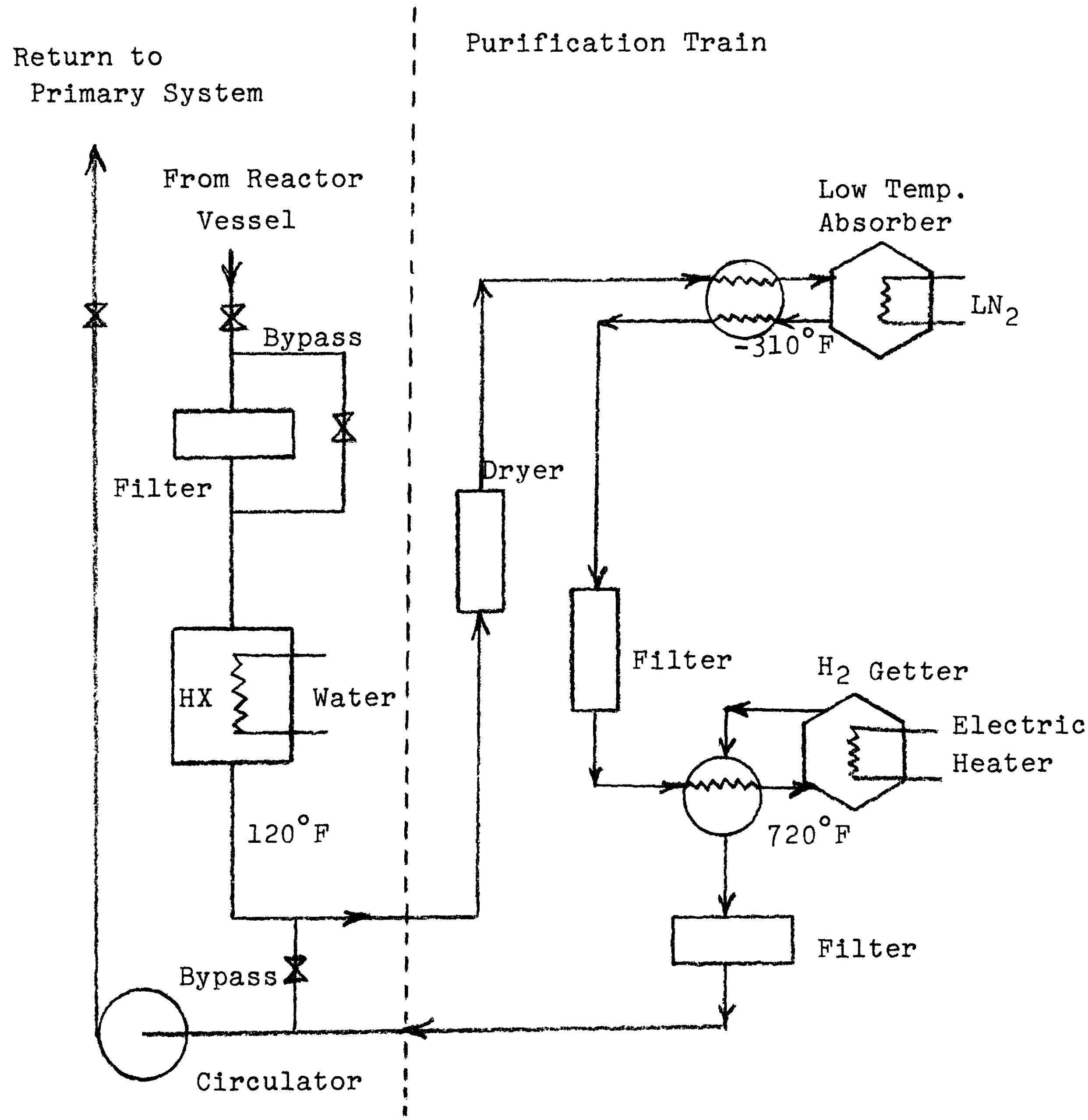


Fig. 4.2 Schematic of Purification System

cryogenic and gettering absorption techniques to purify the helium. The heat exchanger and circulator on the left-hand side of the flowsheet, however, serve a dual function--to provide flow to the purification train during normal operation; and to remove energy from the primary coolant should both main loops be rendered incapable of doing so. In view of the latter function, duplicate independent purification systems are provided to assure reliability.

We will not dwell further on the normal design features or functions of the purification system except to note that packaged units are now being constructed in modular form for LWR offgas service. These units are shop fabricated and transported as a package to the site; a similar design approach is recommended in the present instance. One other point worth nothing is that hydrogen gettering may not prove to be a severe problem in the HTGR/GT from a chemical standpoint since water leakage into the primary current is highly unlikely: we do retain this capability, however, because of the desire to remove tritium.

Auxiliary shutdown cooling is accomplished by bypassing the inlet filter (to decrease pressure drop and preclude loss of function due to plugging) and the purification train (again to decrease pressure drop and permit increased flow). In the shutdown cooling mode each of the two independent purification systems is capable of removing 2% of rated core thermal output. Although the decay power immediately upon shutdown can be as high as 5% of rated output, it will fall to less than 2% in one hour, and the large heat capacity of the reactor graphite will readily store the excess energy over the initial period without overheating the fuel inventory (see additional discussion in Chapter 6, Safety and Reliability Analysis). One purification loop at 50% capacity can hold the core mean temperature rise to an insignificant 150^{o}F.

The purification system return flow is used to cool the control rod drive mechanism and provide gas to the circulator seals.

4.3.2 Shutdown and Emergency Cooling Systems

The following modes of energy removal from the primary coolant are provided:

Operational:	Normal --	use of gas turbine loops via 1HX units
Shutdown:	Normal --	use of GT loops in shutdown mode use of shutdown cooler loop on secondary side of 1HX
	Auxiliary --	use of purification system
	Ultimate --	use of heat leakage through reactor vessel, removed by reactor cavity and/or containment air cooling systems

Post-Accident (Primary System Blowdown)

Same modes as during shutdown.

The post-accident aspects of heat removal will be discussed in Chapter 6. Here we will confine our remarks to the normal and ultimate modes during operation and shutdown.

The gas turbine loops can be operated to remove energy in a self-sustaining mode down to about 10% of rated core thermal power. At this point the fuel temperature will have decreased from its normal operating level by about 500°F, providing a substantial cushion against overheating in the transition to the shutdown cooling mode. Once the reactor reaches 10% power a rapid shutdown is effected by control rod drivedown or scram. If all shutdown cooling modes are operational the maximum decay heat rate can be accommodated without an increase in fuel temperature: if only <u>one</u> (and the least effective) of the several redundant paths is operational the fuel temperature increase will be less than 300°F before the heatup transient is turned around -- hence the fuel will not exceed its normal full power operating temperature.

At lower power the turbomachinery must be spun using a starting motor (or the alternator if equipped to do so): in this role it can still be used to remove on the order of 1% of rated thermal power per loop. To permit total shutdown and isolation of the turbine plant a bypass loop complete with circulator and gas-to-water heat exchanger has been provided across the inlet/outlet of the secondary side of each IHX. This can also remove ~1% rated power per loop when the primary circulators are driven by their pony motors. Should the pony motors become inoperable there is still the possibility of heat removal (following flow reversal in the core) by natural convection. This is not relied upon however; instead the purification system is employed as previously described.

An "ultimate" mode of heat removal is also provided. By properly designing the reactor vessel interior insulation to permit a non-negligible heat loss (but sufficient to avoid overheating or excessive thermal stress during normal operation) it is possible to remove long-term decay heat by conduction through the vessel walls. This mode of cooling has been successfully demonstrated on the AVR reactor during a simulated total loss of circulator power without scram (G8).

4.3.3 The Refueling Systems

Although it is intended that refueling systems similar to those used on Fort St. Vrain and Peachbottom I will be adopted, this area is too important to totally escape mention, at least of the following aspects (see also Table 4.3):

(a) Because of the waiver of fuel carrying charges and disallowance of credit for bred material, there is less incentive for rapid reprocessing of spent fuel than for commercial units.

(b) Because of the unit's stand-alone status and batch core loading, rapid refueling is even more attractive than on the larger civilian HTGR's.

(c) Based on Peachbottom experience and the need for high availability, it is recommended that tools be available for removal of broken fuel blocks.

(d) Given on the order of nine units and a 3-year core lifetime it would probably be possible to employ a full-time itinerant refueling crew to refuel the various base reactors in turn at the rate of 3/year.

Table 4.3

Refueling Systems

1. Refueling Concept		
Refueling philosophy	:	Off-load, Batch refueling
Refueling Cycle	:	3 years
Operational mode	:	Each machine involved performs only one function.
2. Refueling Machine		
Machine Location	:	Above the Pressure Vessel
Machine Control	:	Remote/Manual
Machine	:	Identical to that used for Fort St. Vrain
3. Storage of Fuel		
New Fuel Storage	:	Dry and He Atmosphere
Spent Fuel Storage	:	Dry and He Atmosphere, Inside Primary Containment.
4. Environment of Reactor during Refueling		
Temperature	:	Below 1000°F to prevent graphite oxidation should air leak in
Pressure	:	He gas, slightly below atmospheric pressure

4.3.4 Other Systems

Table 4.4 is a brief synopsis of the various systems required to supplement the major reactor systems: the most important are discussed elsewhere in this report; in most other cases standard HTGR design practice is implied. We call attention to the fact, however, that an attempt has been made to simplify and combine systems and overall plant design: for example, the use of all electrical controls in place of a combination of electrical-hydraulic-pneumatic. While this has many obvious advantages, particular attention will have to be paid to implementation of compensatory design strategies which guard against common mode failure.

4.4 Plant Layout

While much of this aspect of system design lies within the province of the architect-engineer it is important that it be addressed here. As noted in the section on containment design (see Chapter 6), the HTGR is inherently large--which can lead to an expensive balance-of-plant if a compact arrangement is not devised. It is also desirable, and particularly so in the present instance, to facilitate rapid maintenance--which implies good accessibility. Finally, because a non-integral design is used, particular care must be taken to protect the primary circuit ducting against excessive thermal and seismic stresses.

Figure 4.3 illustrates the general features recommended for the system layout. Points which deserve mention include:

(a) off-center reactor location coupled with vertical

Table 4.4

Auxiliary Systems Checklist
(Nuclear Island Only)

1. Cooling	
(a) shutdown cooling	see text
(b) pressure vessel cooling	passive: internal insulation
(c) emergency core cooling	see text
(d) containment cooling	no special system: uses air recirc. system
(e) component cooling	uses part of purification system flow
2. Gas Handling Systems	
(a) helium charge/discharge	apply conventional HTGR practice
(b) helium purification	apply conventional HTGR practice
(c) helium storage	apply conventional HTGR practice
(d) buffer and shaft seal	apply conventional HTGR practice
(e) gaseous radwaste	apply conventional HTGR practice
(f) leak detection	two methods: He leak detector; airborne radioactivity
3. Power Supply Systems	
(a) Instrument and control air	not used for Category I instruments
(b) Emergency component power	diesel engines
(c) Emergency instrument power	batteries
(d) Hydraulic systems	not used for Category I instruments
4. Component Handling	
(a) Refueling machine	Fort St. Vrain type
(b) Spent fuel storage	intra-containment pit
(c) New fuel storage	ex-containment vault
5. Plant Service	
(a) bearing lube	
(b) containment spray	not used
(c) decontamination system	no water used inside containment; CO_2 and powder-type extinguishers used
(d) fire protection (other than core graphite)	no water used inside containment; CO_2 and powder-type extinguishers used
(e) service water	no water used inside containment; CO_2 and powder-type extinguishers used
(f) drain and vent	no water used inside containment; CO_2 and powder-type extinguishers used
6. Emergency Reactor Shutdown	
(a) boron injection	boron steel shot: gravity-drop into in-core channels

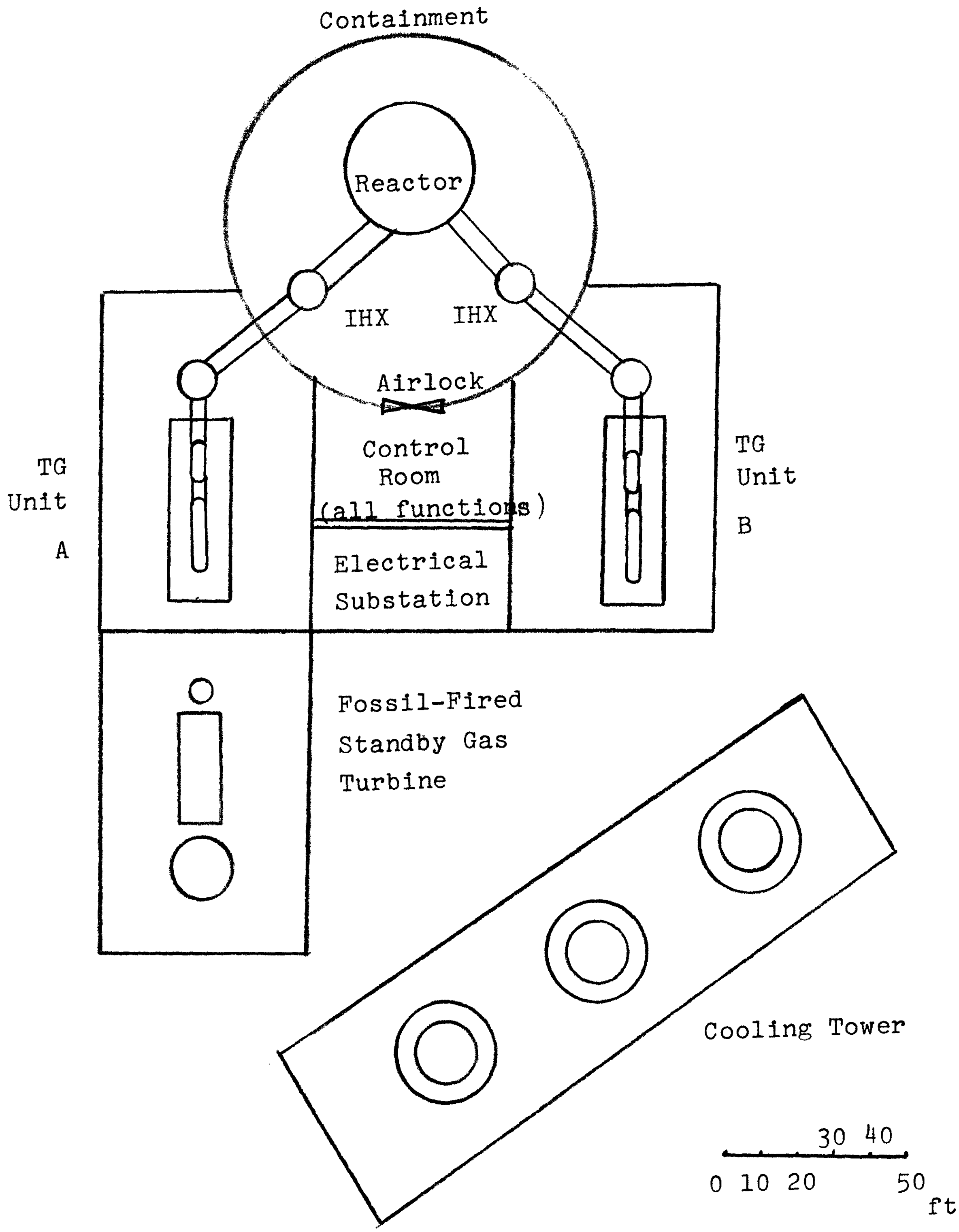


Fig. 4.3 Schematic of Suggested Plant Layout

IHX arrangement-centerlines on a triangular grid--to minimize containment volume.

(b) turbomachinery shafts perpendicular (approximately) to containment to avoid direct-line missile trajectories: but note that control room requires a shadow-shield.

(c) all subsystems accessible to and visible from the control room insofar as practicable.

(d) ventilation stack (not shown) downwind from, and one stack-length from, all subsystems; likewise oil tank farm for FFGT unit (not shown) safe distance from plant, constructed with fire-control moat.

Figure 4.4A and 4.4B show a vertical section through the primary containment vessel--the similarity of the present layout to the Geesthacht and JAERI designs described in Ref. (F1) is evident.

4.5 Conclusion

While development of a detailed design is more within the province of the reactor vendor and architect-engineer, in this chapter we have sketched a preliminary design in order to provide some assurance of feasibility and compatibility with past practice and to call attention to some of the features which our review suggests should be incorporated in a final design. Many are not essential to the concept and a final design may well adopt approaches proven on some of the other units shown in Table 4.1.

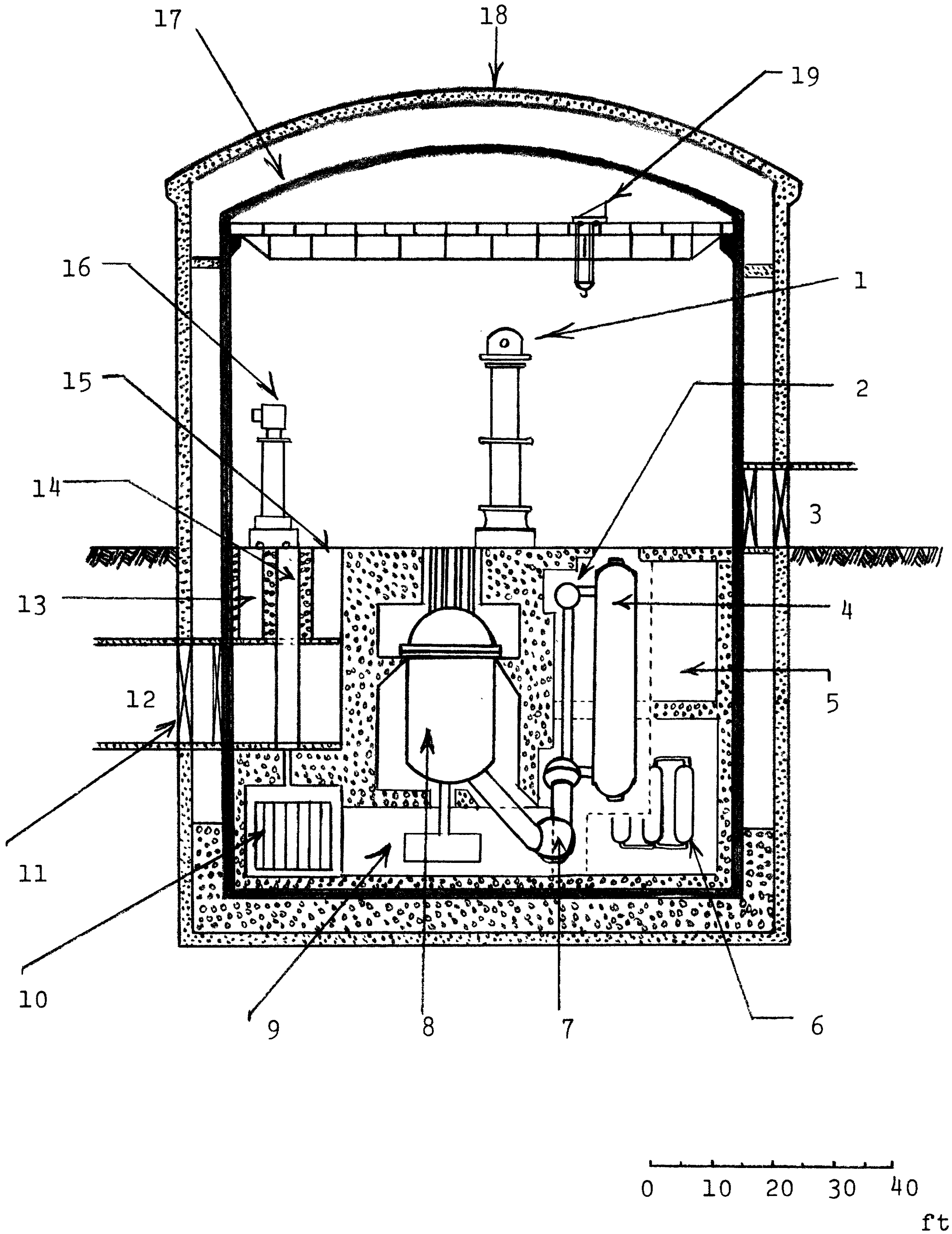


Fig. 4.4A Vertical Section through the Primary Containment Vessel

1. Fuel Handling Machine
2. Circulator
3. Airlock Entrance
4. IHX
5. Containment Ventilation/Electric Supply System
6. Helium Storage Tanks
7. Isolation Valve
8. Reactor
9. Helium Purification System
10. Spent Fuel Storage Room
11. Airlock Entrance
12. Fuel Element Transfer Facility
13. Spent Fuel Canning Room
14. Spent Fuel Disposal Chute
15. New Fuel Preparation Room
16. Fuel Transfer Machine
17. Inner Steel Vessel
18. Outer Concrete Wall
19. Crane

Fig. 4.4A (Continued)

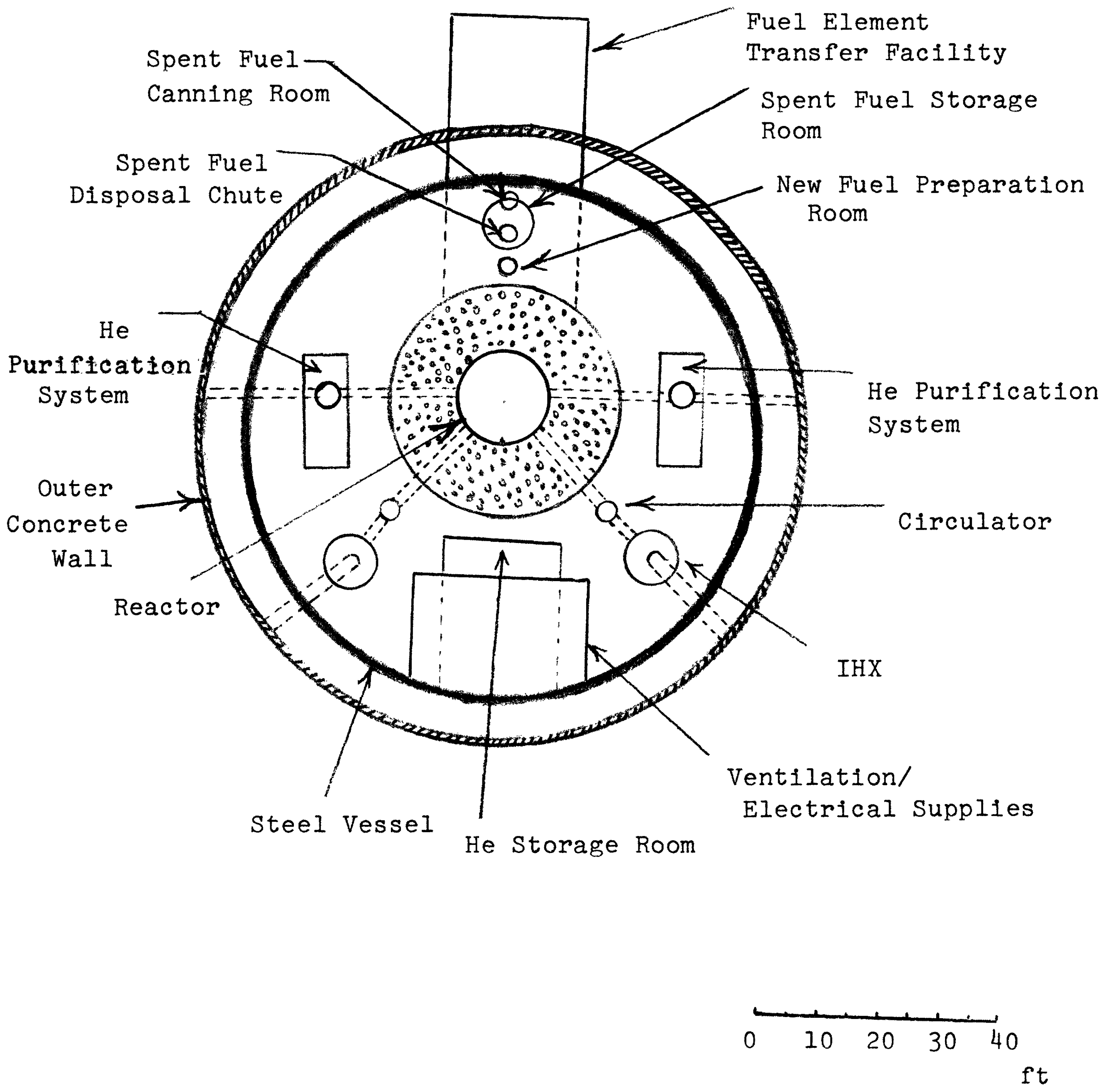


Fig. 4.4B Horizontal Section through the Primary Containment Vessel

It is essential that cost reduction be addressed in any future design work because of the heavy capital cost burden assessed against small nuclear units. Particular emphasis should be placed on reduction in containment volume, on modular construction of subsystems at the factory and on borrowing intact of component and subsystem designs from other reactors. Further efforts should be made to simplify the design and combine similar functions where this can be done without compromising safety-related redundancy and diversity.

Chapter 5
CORE DESIGN

5.1 Introduction

Considerable effort has gone into all aspects of HTGR fuel design, both in the U.S. and abroad. As a result, many variants of the basic concept, which involves coated-particle-type fuel in a graphite matrix, have been examined. However, only the GA-type hexagonal block fuel and the AVR-type pebble bed fuel will have been subjected to extensive proof testing through 1985; and there is a growing consensus among the major developers of the HTGR (or HTR as it is designated in Europe) that block-type fuel should be adopted as the standard fuel type for all but very high temperature service. Because this type of fuel appeared to be eminently suitable for the present application and because development of new fuel concepts is prohibitively expensive in terms of both financial and temporal requirements, it was decided at the outset to specify Fort St. Vrain type fuel for the HTGR/GT unit. Figure 5.1 illustrates the main generic features of such fuel.

In this chapter we will examine the many subsidiary considerations required to specify a core design and determine its various neutronic characteristics. It should be noted that more detailed examinations of both the reactor physics

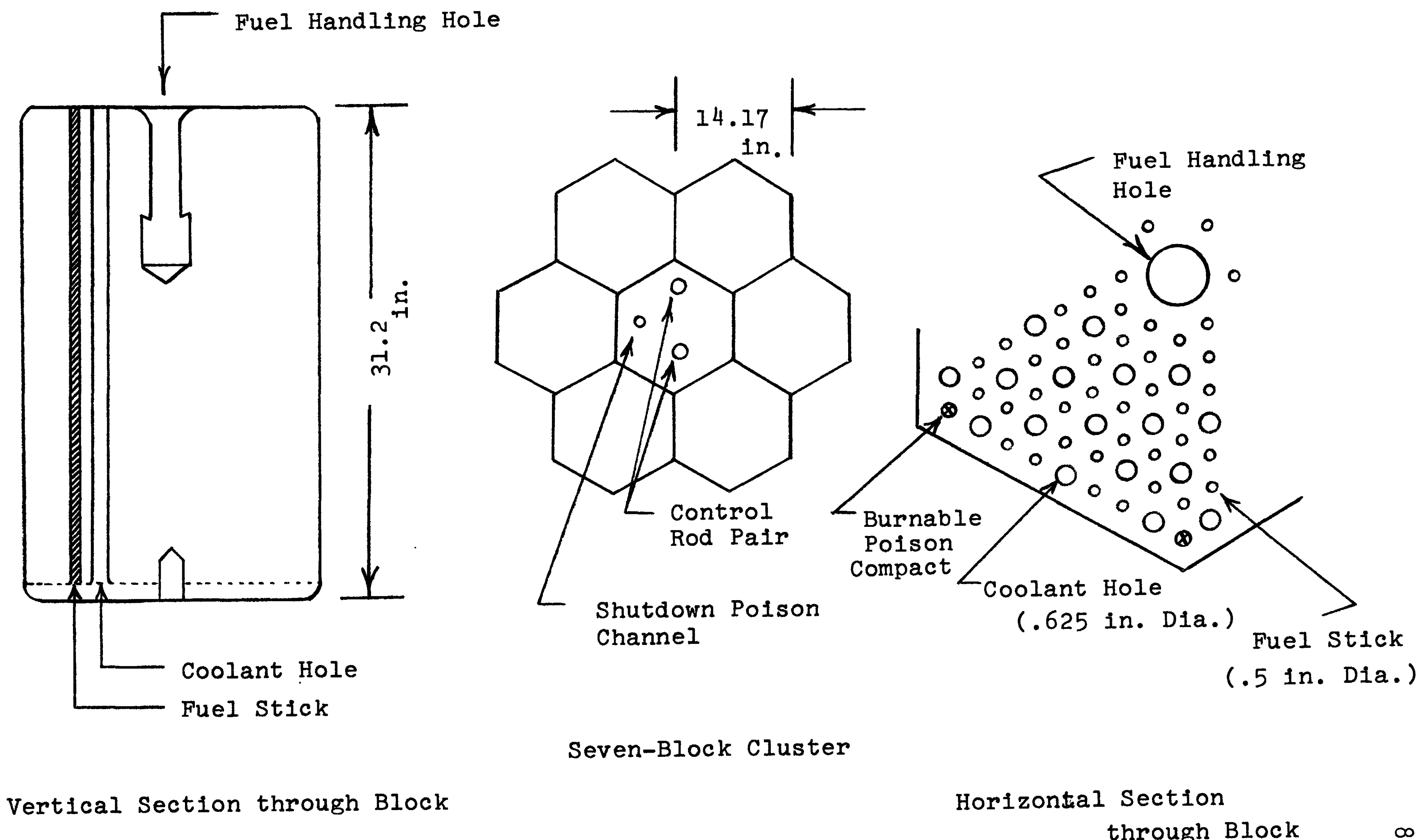


Fig. 5.1 Fuel Element Design

and thermal-hydraulic aspects of core design are in progress, and will be reported separately as Sc.D. and S.M. theses by Ribeiro and Stengle, respectively.

5.2 Design Constraints

The HTGR/GT is subject to two requirements which distinguish it from the large HTGR/Rankine units being designed for electric utility service: the core outlet temperature must be higher to facilitate operation of the gas turbine cycle; and a batch-loaded long-life core is preferable to reduce refueling outages and on-site fuel handling. These requirements are by no means unique: HTGR's for process heat applications require even higher temperatures, and the Peachbottom I HTGR was designed to have a 2.2 full-power-year batch core life.

In view of the fact that Fort St. Vrain is designed to have a coolant outlet temperature of 1445°F, it was not considered that the value of 1500°F selected for the present application represented a significant change over state-of-the-art capability. Various strategies are available to the designer to achieve even higher temperatures as summarized in Table 5.1. As can be seen there is considerable room for improved performance should the need arise. Temperature limits are more likely to be set by duct and vessel insulation, IHX tube and turbine blade materials limits, rather than by the core. Most of these areas are presently being worked on

Table 5.1

Methods for Increasing HTGR Core Outlet Temperature (Q1)

	Design Change	Gas Temperature, °F
(1)	Optimize core loading and orificing	1500
(2)	Use Fort St. Vrain fuel blocks (210 fuel holes instead of 132 as used in later designs)	1680
(3)	As in (2) plus use of TRISO coating on all fuel particles	1730
(4)	As in (3) plus use of cluster control rods	1740
(5)	As in (4) plus 3 instead of 4 year fuel life	1770
(6)	As in (5) plus reduced power density: from 8.4 to 7.3 w/cc	1800
(7)	As in (5) plus axial-pushthrough fuel management	2070
(8)	In addition one can usually gain 100-300°F by switching to pebble-bed type fuel.	

as part of the effort to develop HTGR's for process heat applications. In addition, fossil-fired gas turbines of advanced design are being developed for high temperature service: use of ceramic blades may even permit operation up to 2500°F. While this ultra-high temperature capability is of considerable technological interest, it is probably significant for the present application only in that it confirms the modest and readily achievable goals of our design. For a total energy application having a appreciable thermal energy demand, high thermodynamic efficiency -- which is the primary benefit of higher operating temperatures -- is not necessarily advantageous.

In view of the above factors, and the equally important question of future commercial availability, it was concluded that the Fort St. Vrain fuel design should be adopted for the Army HTGR/GT system.

5.3 Selection of Fuel Cycle

While the HTGR is a particularly flexible concept from the viewpoint of its adaptability for consumption of various combinations of fissile and fertile fuels, from a practical standpoint only the high enrichment U^{235}/Th^{232} cycle and the low enrichment U^{235}/U^{238} cycle deserve consideration here. Options such as the use of plutonium, which has the desirable feature of providing long batch lifetime (G4), are insufficiently developed to permit their use by the Army in the absence of commercial precedent and an industrial base.

In regard to the comparison of high and low enrichment fuel cycles, recent analyses of low enrichment fuel cycles for the large commercial HTGR's appear to be converging to the consensus that this cycle is less economic than the high enrichment cycle (B9) (G5). Waiver of carrying charges on U^{235} should enhance the economic advantage of the high-enrichment fuel cycle for DOD service. In any event, if the high enrichment cycle gains commercial favor, the resulting economies of the associated large-scale fabrication facilities will give this cycle a large economic advantage for the Army application as well.

In spite of the many obvious advantages of arbitrarily requiring use of the GA fuel cycle, the low enrichment cycle was carried a substantial way through the design process. One motivation for continued interest was the possibility of using a single fuel particle, thereby providing added diluent to help retain fission products. The results of in-depth evaluation, however, provided even more conclusive evidence of the superiority of the U^{235}/Th^{232} cycle: in particular, the low enrichment cycle was found to have on the order of a 30% shorter batch-loaded reactivity lifetime. Even if this short-coming could be overcome, work carried out in Europe on the HTR low enrichment cycle has shown that optimal fuel loadings are such that fuel burnup lifetimes are about 25% shorter than for the high enrichment cycle (G4). Finally once power cycle design had proceeded to the point where an indirect cycle was specified and IHX design was optimized to achieve a low ΔT between the primary and secondary circuits, it no longer became necessary to pursue ultra-high fuel integrity. Hence the high enrichment U^{235}/Th^{232} GA fuel cycle was adopted.

In addition, while a reference fuel cycle has been selected, it is important to note that in general it is possible to switch from one fuel cycle to another over the life of an as-built plant in HTGR-type reactors without system redesign (T1)(G4).

5.4 Selection of Fuel Particle Type

There is considerable latitude in the choice of a specific particle design for use in HTGR-type fuel. The current reference design fuel for the commercial HTGR's involves the use of two particles: a smaller "feed" particle of the TRISO type (i.e., employs a SiC barrier layer) which contains the fully enriched U^{235} carbide, and a larger "breed" particle (pyrocarbon coating only) containing thorium oxide. While this fuel appears suitable for use in the present application, a number of improvements could be achieved using technology already in the test phase in the U.S., Europe and Japan (B8,P3,G6)

(a) use of a single intermediate-sized particle kernel of mixed UO_2/ThO_2 to provide a greater volume of matrix for the dilution of fission products, and enhanced stability against kernel migration (amoeba effect) from ThO_2. The fima (fissions per initial metal atom) is reduced by a factor of about 6 from that of the GA feed particle.

(b) use of TRISO coating on all particles to provide improved fission product retention: coating candidates are SiC and ZrC (a new coating under study because of its potential for improved high temperature performance).

(c) addition of oxygen and fission product getters to the kernel matrix to reduce gas-induced stress on the particle coating.

Improvements are also under active investigation on improved methods for compaction of the particles into a fuel stick (rod) and sealing the rods into the graphite block. In both areas the objective is to increase the effective thermal conductivity and thereby allow the fuel to operate at the lowest possible fuel temperature in the lowest possible temperature gradients. Again, while adoption of improvements of this nature is not essential, it would be desirable to do so if they prove to be effective and become commercially available.

The proposed changes in particle design also have the potential for reducing fuel fabrication charges, since the small TRISO particles in GA-type fuel are more difficult to fabricate to acceptable quality control standards. On the other hand, the single particle design will result in a larger amount of U-233 mixed in with the U-235 in the spent fuel. It is not clear how much (if any) of a penalty this would prove in the present application where the conversion ratio is quite low and where the use to which the spent fuel will be put is unresolved.

It is important to note that the details of particle design are largely decoupled from the neutronic design of the core. Hence one can readily change among and interchange fuel employing the various particle designs under current investigation.

5.5 Reference Core Design

Using nuclear methods (M6) and cross sections developed and validated by GA for the Fort St. Vrain core (modified to account for our smaller carbon-to-uranium ratio) a reference design was developed for a 300 MWTh HTGR core. A detailed description of the methodology and various intermediate results will be documented in the Sc D thesis by Ribeiro: a summary description is presented in Tables 5.2 and 5.3; Figs. 5.2 and 5.3 illustrate the general configuration constructed from the sub-units shown in Fig. 5.1. Table 5.2 also shows Fort St. Vrain data for comparison.

As shown in the table, a core lifetime of 4.8 effective full power years - approximately 6 calendar years - should be within reach based upon fast fluence and time-at-temperature exposure. However a reference, assured value of three calendar years (2.4 efpy) has been assumed in this report because of the difficulty involved in achieving sufficient reactivity lifetime within practicable control limits. It is anticipated that future work will confirm the capability of the core described here to sustain a six year batch reactivity lifetime, in which case the economic prospects of the HTGR/GT will be improved.

Table 5.2

Comparison of FSV and HTGR/GT Core Design Parameters

	FSV	HTGR/GT
Reactor core output, MW(Th)	851	300
Core dimensions, dia/ht, ft	19.5/15.6	14.3/10.4
Number of fuel elements/columns	1482/247	532/133
Primary coolant flow, 10^6 lbs/hrs	3.39	0.8 x 2 = 1.6
Primary coolant inlet pressure, psig	688	394
Ave. coolant temp., reactor inlet, °F	762	953
Ave. coolant temp., reactor outlet, °F	1445	1500
Core orifices	37 variable	19 fixed
Maximum fast fluence (E>0.18 Mev) 10^{21} nvt	8	6.68*
Ave. power density, W/cc	6.3	6.34
Fuel life, full power years	4.8	{2.4 assured 4.8 anticipated
Number of refueling regions	37	Batch refueled
Element (hexagonal prism): across flats/length, in	← 14.17/31.2 →	
Fuel holes per element, std/control	← 210/100 →	
Fuel hole diam., in	← 0.5 →	
Coolant channels per element, std/control	← 108/57 →	
Coolant channel dia., in	← 0.625 →	
Reflector thickness, cm, top/bottom/side	118.9/118.9/135.9	118.9/118.9/100
Max. fuel burnup, MWD/T	100,000	98,000*
Max. fuel centerline temperature, °F	2300	2100

* at 4.8 efpy

Table 5.3

Summary of HTGR/GT Referenced Design Characteristics

1. Compositions at BOL - refer to Fig. 5.2

	Constituent	Concentration (10^{-5} nuclei/barn cm)	mass, kg
Core	carbon	6190.	$58.42.10^3$
	U^{235}	4.200	775.7
	U^{238}	0.316	59.1
	Th^{232}	34.68	6.323×10^3
	Si	73.7	815.8
	B^{10}	0.1218	0.958
Reflector	carbon	8876.	-
	B^{10} (bottom)	0.2436	0.219

2. Power density and related parameters

Maximum power density (at core center)	(W/CC)
at BOL	10.44
at 850 days	8.60
at 1690 days	7.55

Radial peaking factor (BOL) = 1.21

Axial peaking factor (BOL) = 1.28

Avg. burnup in 1752 days (4.8 years) = 63,000 MWD/T

Max. burnup in 1752 days (4.8 years) = 98,000 MWD/T

3. Channel heat generation rate, $MWth/cm^2$

	BOL	1690d	Ratio BOL/1690d
Central Channel	2.597×10^3	2.244×10^3	1.16
Peripheral Channel	1.637×10^3	1.800×10^3	0.91
Ratio Central/ Peripheral	1.59	1.25	1.27

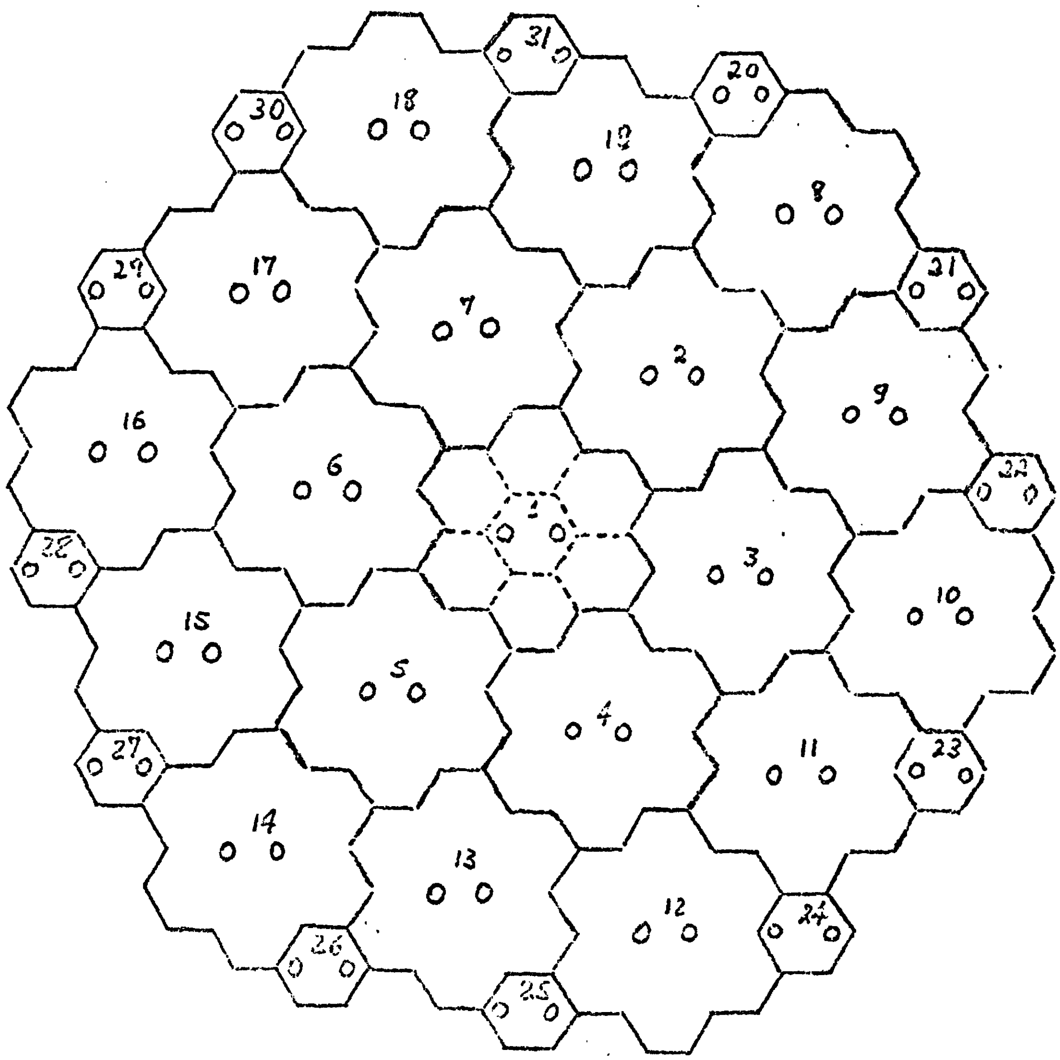
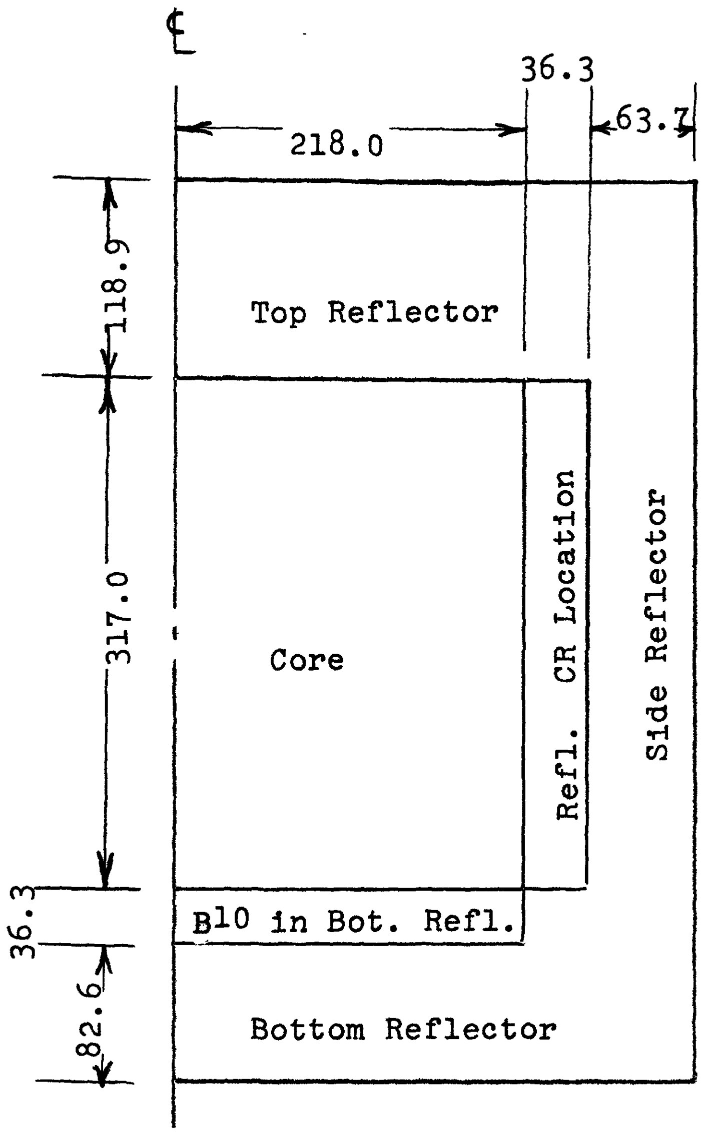


Fig. 5.2 Horizontal Section through Reference Core Design



(all dimension in cm)

Fig. 5.3 Vertical Section through Reference Core Design

The distinctive features of the subject design are as follows:

(a) Control rods are employed in the reflector as well as in the core. This approach is a useful one because of the relatively small size of the present core - in the even smaller Dragon reactor only reflector control is used. Table 5.4 illustrates the value of these rods in enhancing the reactivity swing which can be compensated for by movable poison. These rods can be used to compensate for the reactivity change between cold shutdown (including margin) and hot full power (with equilibrum Xenon), hence they will not increase the radial peaking factor.

(b) Reactivity lifetime considerations have also led to use of higher fissile and burnable poison loadings than Fort St. Vrain (factors of approx. 2.5 and 3 respectively). In addition, a poison-loaded zone is included in the lower reflector to depress the power in the hot fuel near the core exit and to provide still another increment of reactivity control. Calculations show that this stratagem reduces the hot spot fuel temperature by about 150°F.

Table 5.4

Reactivity Control Requirements
and Methods for the HTGR/GT Core

Requirements (Δk/k)		Control Poison (Δk/k)	
(1) cold clean to hot	0.037	(1) 0.14	19 core rods
(2) hot to hot-full-power	0.021	(2) 0.08	12 reflector rods
(3) case 2 plus equilibrum xenon	0.112	(3) 0.17	burnable poison
(4) case (3) plus 3 years batch burnup	0.170		
(5) stuck rod allowance	0.045		
(6) 10% margin	0.005		
TOTALS	0.39	0.39	

(c) A single fuel loading is used throughout the core. This is possible because of the relatively small size of the core, which permits a relatively good peak-to-average power ratio without resort to zoning the fuel loading. This will also be favorable from an economic standpoint - larger batches of identical fuel should cut costs. (In addition one can contemplate ordering for several reactors at once.) Should future performance demands be increased, zoning could be reconsidered.

(d) A single set of fixed orifices is used throughout core life. This simplification is made practicable by the use of a batch, uniformly-loaded core and, again, the small size of the core. Stengle will report a detailed thermal/hydraulic analysis for the core over its design lifetime in his SM thesis. In addition, we have not taken credit for control rod programming, which could improve the temperature profile control over lifetime.

It is important to note that the reference core design combines in a compatible manner features whose design and operation have already been proven by experience. Hence the design is quite orthodox in terms of both the configuration employed and the requirements demanded of the constituent materials. Furthermore some degree of fall-back margin is built in: in the extreme one could settle for the assured lifetime of 3 years instead of the target span of 6 years, in which case a near-duplicate of (a scaled-down) Fort. St. Vrain core could be adopted.

5.6 Discussion and Conclusions

The reference core design described in this chapter is characterized by its inherent simplicity. A minimum batch lifetime of 3 calendar years is assured based on reactivity control limitations - if these can be overcome the fluence and burnup - limited lifetime of around 6 years can be achieved as a refueling interval. Complete adoption of Fort St. Vrain technology for fuel and control rod design has been shown feasible; hence by 1985 nearly ten years of in-service experience should be available. There is, moreover, already sufficient irradiation experience on similar fuel in the Peachbottom I and Dragon reactors to assure basic feasibility.

In brief, the Army HTGR/GT core can fit right into the commercial fuel cycle of the larger HTGR's being designed for central station utility service and take advantage of their large economy of scale for both fabrication and fuel processing.

CHAPTER 6

CONSIDERATIONS RELATED TO SAFETY AND RELIABILITY

6.1 Introduction

Because of the close relation of the present reactor design to that of Peachbottom I there are many aspects of performance analysis under normal operation, anticipated transients and accident conditions which are similar if not identical. Hence Peachbottom I precedent is an important reference point on which one can base the discussion of the present design. In the discussion which follows we will reference rather than repeat most of this background material and focus our discussion on the differences in design which must be reflected in the safety- and reliability-related performance analyses. Table 6.1 summarizes the key differences of present interest, about which more will be said later.

We also call attention to the topical report (the final version of which will also be published as a SM thesis): M. R. Doyle, "Comparison of Environmental Impacts of Coal and Nuclear Systems for Military Utility Applications; and Consequences of Reactor Accidents". This report discusses the public consequence aspects of the highly improbable accidents in which there is significant fission product release external to the reactor containment. In the present report, therefore we will confine our remarks to the in-plant and design aspects of safety.

Table 6.1

Safety-Related Differences: HTGR/GT vs Peachbottom I

ITEM: HTGR/GT uses	COMMENT
(1) IHX in place of Steam Generator (SG)	(a) Eliminates accidental H_2O ingress into primary system.
	(b) Reduces ΔP across IHX/SG tubing, less corrosive secondary side environment: potential for improved integrity.
(2) Downflow core; concentric ducting enters bottom of reactor vessel.	(a) Flow reversal required before natural circulation becomes effective--this may be hard to establish because of configuration or may in fact work against forced circulation under low flow conditions.
	(b) Gravity assists control rod scram.
	(c) Favors more rugged seismic design.
(3) Ft. St. Vrain type fuel	(a) Potentially better fission product retention.
	(b) Elements less subject to breakage and problems resulting therefrom.
(4) Higher operating pressure: 400 vs. 320 psia; larger component sizes.	(a) Potentially higher blowdown rates during accidents.
(5) Different approach to shutdown and emergency cooling.	(a) Relative performance differences not entirely clear.
	(b) We do provide ≥ 4 separate, on-line (hence continuously tested) cooldown circuits, any one of which suffices.
(6) No containment inerting.	(a) Hydrogen explosion not a concern due to lack of water, other means available to preclude graphite fires; large amount of helium available, including turbine plant inventory.
	(b) If required, inerting capability could be backfit without undue bother.

6.2 Containment and Associated Systems

The containment building for the HTGR/GT reactor is comparable in size to that for a 1000 MWe PWR, i.e. approximately 1.2×10^6 ft^3 free volume. This follows from the fact that the reactor vessel is the same diameter as a PWR pressure vessel, and the IHX units are equivalent in size to steam generators for the more highly rated PWR. The 40 MWe Peachbottom I Reactor, for example, has a 1.14×10^6 ft^3 total contained volume (7.2×10^5 ft^3 free volume) using a vessel of 100 ft dia, 162 ft overall height. Proportionally, then, the containment is a more important component cost-wise on the HTGR than on the PWR. Since each percent of added containment volume represents an expense of an order of $130,000 it is particularly important to devise a compact layout (compatible with design and maintenance requirements). There are also a number of safety-related trade-offs to consider. Large contained volume leads to low internal pressure following a primary coolant blowdown--which helps reduce the problem of post accident confinement; on the other hand a higher post-blowdown pressure makes core cooling easier because of the increased gas density. The design pressure selected in the present instance is 10 psig--lower than comparable PWR's at ≈40 psig--but about the same as Peachbottom at 8 psig. While primary system blowdown alone would not require a 10 psig allowance, sufficient margin has been incorporated to permit simultaneous rupture of tubes in both lHX units and complete discharge of both turbine unit helium inventories into the containment, plus discharge of all on-line primary and secondary reserve helium (one system volume each) to allow for post-blowdown helium injection as part of air-exclusion procedures.

Figure 6.1 presents a schematic of the containment and associated systems. They represent an amalgam of current PWR and HTGR practice. The containment consists of an interior free-standing steel vessel (cylindrical shell with formed top and bottom heads) surrounded by a concrete shield vessel, with an annular air-space in between. Since the annular space is processed through the off-gas treatment facility, the design provides, in effect, double containment. This advantage is achieved at very modest cost: both concrete and steel are required in any event--the former to provide post-accident shielding and external missile protection and the latter to provide leak-tightness. It should be noted that this design should substantially reduce the largest normal release category--primary system leakage--attributed to an earlier version of our system by Doyle in Ref. (D1). During both normal and accident conditions a slight negative pressure is drawn in the annulus air space to insure in--rather than out--leakage. The contained volume is normally at a slightly lower pressure; however if a blowdown accident occurs, primary pressure remains high until plant cooldown is effected, following which the containment is depressurized via the offgas system.

Also shown in Fig. 6.1 is the containment ventilation system, which is a simplified version of that used in the latest HTGR designs (K3) and provides the following services:

(1) Containment purge and recirculation--to provide temperature-controlled and filtered air to (and distribute it within) the containment, and to exhaust containment air.

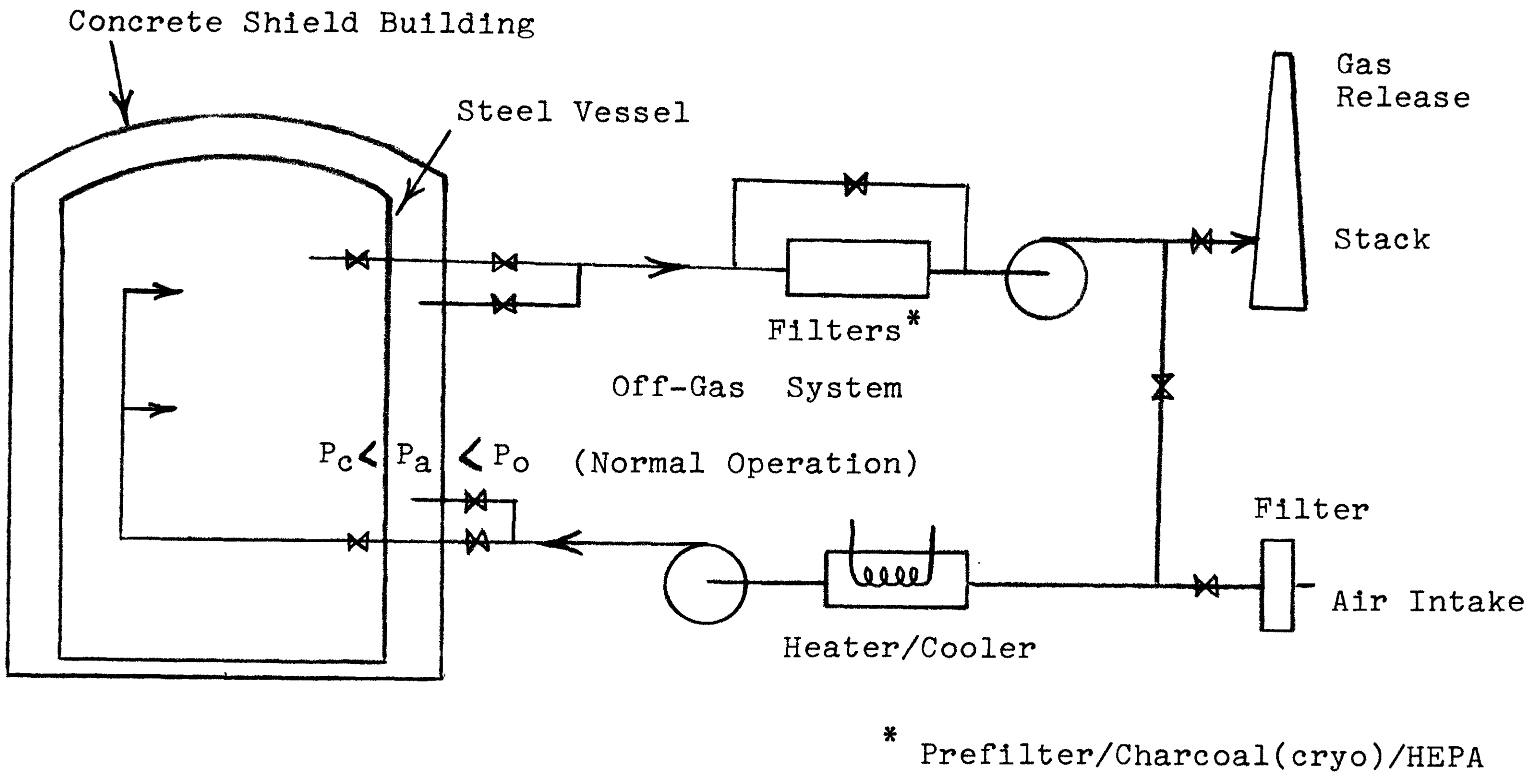


Fig. 6.1 HTGR/GT Containment and Associated Systems

(2) remove radioiodine and radioactive particulates from the containment atmosphere: redundant, independent subsystems are provided to handle the maximum hypothetical accident.

All ventilation system hardware is standard equipment of proven design; seismic hardening is required only in the subsystem involved in post-accident cleanup.

Note that unlike the PWR, post-accident containment cooling is not required and no containment spray is needed because no steam is released during blowdown. The complete absence of water (i.e., no steam generators either) means that no hydrogen will be produced by metal-water reactions or radiolytic decomposition. Hence recombiners and containment inerting systems are not required on this account.

While Peachbottom employed a containment inerting system, motivated primarily by the concern over post-blowdown graphite fires, it was considered desirable not to employ this approach in the present design. Without inerting, maintenance activities on the reactor systems can be carried out more rapidly -- and without the danger of accidental asphyxiation to personnel. We have less need for inerting because of the absence of hydrogen, as noted above, and because of the large helium inventory available to keep air out of the core following a system rupture: if needed, the helium from the turbine plants can also be employed. More attention will have to be paid to control of argon activity induced in containment air than in an inerted design, but this is considered feasible since there is no essential difference in this respect from the situation with PWR reactors.

6.3 Accident Categorization

The major accidents of concern in the present design are the following:

(1) rupture of primary system pressure boundary;

(2) core graphite oxidation;

(3) loss of forced circulation through the primary system;

(4) severe reactivity excursion;

(5) core channel blockage.

This breakdown is somewhat arbitrary since, for example, (2) is only of concern as a consequence of certain categories under item (1); and (3), (4), and (5) are of concern only in that they can lead to fuel overheating. As long as one can keep the fuel cool in a non-oxidizing atmosphere, fission product retention is assured. Other connections exist: the extremely high flow rates encountered during the depressurization following a large break might create debris (e.g., loose insulation) which could lead to core channel blockage or control rod malfunction. Because primary system blowdown plays such a key role in HTGR accidents we will discuss it separately (see Section 6.4). Here we will briefly discuss the other entries, with emphasis on aspects where the present design differs from its predecessors.

Severe reactivity excursions are not a particularly important concern in the present design. The core has a negative temperature coefficient and both the coated fuel particles and moderator graphite can withstand sizable temperature excursions: graphite has high heat capacity, reaches maximum strength near 4500^{o}F and provides structural strength at even higher temperatures (W1). The in-core sections of the movable and burnable poison elements are the same as proven for Fort St. Vrain and other HTGR service. In addition the use of top-entry control rods provides a conceptual advantage over Peachbottom, since both gravity and flow assist scram. We also propose to employ cannisters of poison-loaded shot which empty into special core channels as a last ditch shutdown system. The major difference lies in the control rod drive mechanisms. We favor the use of PWR-type electro-mechanical units in place of the Peachbottom hydraulic-type or Fort St. Vrain cable-and-drum drives. This is feasible in the present case because of the reasonable stroke length, about 12 ft. At the present stage of design it is left open as to whether the reflector region control elements should employ a different drive design to provide additional assurance of control diversity. The experimentally demonstrated inherent ability of AVR to handle an anticipated transient without scram (via its negative temperature coefficient and high heat capacity) argues against the need for going to this level of complexity. Finally, while not yet assured, it is a design objective to have each rod worth less than one dollar to assure against prompt criticality should excessively fast removal occur (for which, however, no mechanistic sequence has been identified).

Loss of forced circulation through the primary system has already been discussed in Section 4.2 of Chapter 4, where it was shown that the large heat capacity of the HTGR core makes this a much more benign occurrence than in other reactors. Small HTGR's are even more favored because of their larger relative inventory of reflector graphite.

Core channel blockage is of concern in any reactor. In the present instance the downflow core design requires particular attention to this possibility. Since the core inlet must be provided with an upper grid structure to provide seismic hold-down, to align control rods and support the channel orificing for each block, we will also build in anti-blockage inlets (e.g., standpipes with side ports). The same structure can support a coarse-mesh filter on the incoming coolant, which flows <u>up</u> an annulus between the reflector and the vessel (which also insures fall-back of large debris). Thus, only structure in the upper plenum (or the core itself) can contribute to blockage. The chief concern is typically with the insulation or insulation support sheeting and whether it can be damaged by pressure or temperature cycling or flow-induced forces--particularly during blowdown. In the present instance the use of an indirect cycle at least protects the core against pressure cycles caused by the turbomachinery; only the blowdown accident appears to be of concern as potentially more severe than in large HTGR's.

6.4 Primary System Blowdown

Rupture of the primary system pressure boundary is by far the most serious accident of concern in the present design. There are various factors which both mitigate and aggravate this accident with respect to the present design, as noted in Table 6.2. The most serious version of this accident would involve massive rupture of the main coolant ducting between the pressure vessel and isolation valves. This would lead to a very rapid depressurization of the primary system--complete blowdown within several seconds. Indeed it is unlikely that isolation valve response would be rapid enough to prevent blowdown should rupture occur anywhere in the primary circuit. During this initial period the main concern is that the high helium flow rate within the primary system not damage key components such as the control rods, vessel insulation, or helium circulators.

Upon receipt of a low primary system pressure signal (or equivalent alternative high primary flow or high containment pressure/temperature/activity signals) an automatic scram would occur: if not, then the operator could initiate emergency scram and allow the boron shot to enter the core. Once subcritical several hours exist before core thermal margins would be exceeded. Thus there is ample time to bring into action the various modes of shutdown heat removal. The more immediate problem is to avoid ingress of air into the hot core because of the danger of graphite oxidation--i.e., a "graphite fire".

Table 6.2

Factors Affecting the HTGR/GT Blowdown Accident

Mitigating

1. Indirect Cycle removes turbine/compressor-generated missiles as accident initiators.

2. Pressure is lower than PCRV-Type HTGR's (400 vs ~ 700 psia), hence lower flow for a given break size: mass flow through break is proportional to product of primary pressure and break area.

3. Satisfactory primary system blower performance after depressurization is more readily assured than that of direct cycle Turbomachinery with respect to providing adequate flow through core.

4. Service pressure of steel vessel and piping (400 psia) is much lower than for PWR's (~2000 psia).

5. Unlike Fort St. Vrain, the present reactor is contained: unlike Geesthacht, two main coolant loops are provided.

Aggravating

1. Large vessel and duct dimensions compared to Peachbottom, Dragon, AVR (but not other gas-cooled reactors).

2. No easy way to limit rupture flow areas to ~100 sq. in. as in integral/PCRV Type HTGR's. Hence more rapid blowdown is possible.

3. Containment inerting not used (in contrast to Peachbottom)

Two approaches are possible to avoid graphite fires: keep the containment atmosphere inerted at all times during power operation, or inject sufficient makeup helium to keep primary system pressure slightly above that of the containment atmosphere. The latter approach is selected here. In conjunction with this procedure it will be necessary to monitor the core inlet oxygen content. Should primary circulator operation (in the intact loop) draw containment air into the system this mode of cooling should be stopped in favor of the use of the purification system mode--which adds and removes helium directly from the reactor vessel. Note that the ruptured loop will have both isolation valves shut and its circulator tripped to reduce the possibility of either drawing or forcing air into the core.

Ample inert gas exists for injection into the primary system to maintain a differential pressure: the entire primary helium inventory and the entire turbine plant helium inventory (both including reserve storage); in addition the nitrogen from the liquid nitrogen system used for the cryogenic absorbers is available at the discretion of the station staff.

The prohibition against air in the core is not absolute: a small amount (<5 vol. %) can be tolerated at high temperatures (P1), and below about 900°F the graphite is cold enough to reduce the reaction rate to a tolerable level (Y1). Thus it is only during the interim period between blowdown and cooldown that caution is necessary.

Assuming that overheating and oxidation are avoided, blowdown will release on the order of several hundred curies of activity into the containment, consisting of primary circuit circulating activity and a small percentage of volatiles deabsorbed from the primary circuit surfaces. This can easily be handled by the off-gas treatment system. Note that even if the noble gas component of this inventory were entirely released to the environment it would be considerably less than releases during <u>normal</u> operations from BWR plants in the years prior to adoption of high efficiency off-gas systems (D1).

6.5 Maintainability and Reliability

Because the concepts of maintainability and reliability played such an important role in many of the design decisions discussed in this report, a brief summary of some of the pertinent considerations in this regard is called for.

6.5.1. Fission Products

Primary circuit fission product inventory has an important bearing on both maintainability and post-accident conditions. Detailed calculations can become quite complex, but the following simple model can provide useful, approximate results.

The steady state inventory (in curies) of a given fission product of yield y in the fuel of a reactor rated at P MWth is:

$$C_f = 0.8 \times 10^6 \; y \; P \text{ , curies} \qquad (6.1)$$

The steady state inventory of the same fission product in the coolant, assuming a fraction r is released and that the half life is much longer than the time required to escape the fuel, is (in the absence of purification or if the half life is much shorter than the mean time required to process one primary coolant volume):

$$C_c = 0.8 \times 10^6 \; r \; y \; P \quad \text{curies} \qquad (6.2)$$

Thus for Kr 85m, for which

$r = 6.5 \times 10^{-5}$ (GA's "expected" value)

$y = 0.013$ (for mass 85 chain)

$P = 300$ MWth,

equation 6.2 gives $C_c = 200$ curies.

Thus we can expect several hundred curies of noble gas nuclides in our primary circuit.

Since the plateout mean time (on the order of seconds) is much shorter than most half lives of interest, Eq. 6.2 also approximates the plateout inventory of non-noble gas constituents. Again tens to hundreds of curies may be plated out on primary coolant surfaces--a consideration which contributed to the decision to adopt an indirect cycle and minimize the amount of rotating machinery exposed to primary coolant.

6.5.2 Reliability

Again one can demonstrate in a rather simple manner the key considerations involved without undue complexity. Consider a unit having two independent Turbo generator loops, each of which have a non-failure probability (availability) p; then the following probability table can be constructed:

	State of loop	
Fraction of full power deliverable	#1	#2
100%	operational	operational
50%	operational	failed
	failed	operational
0%	failed	failed

Or in terms of Probabilities:

	#1	#2	Fraction of time in state
100%	p	p	p^2
50%	p	(1-p)	2p(1-p)
	(1-p)	p	
0%	(1-p)	(1-p)	$(1-p)^2$
			sum = 1

For example, if p = 0.9 (typical commercial gas turbine availability):

100%	available	81% of time
50%	available	18% of time
0%	available	1% of time.

Since even a 1% rate of complete forced outage would mean a total of approximately four days a year without electricity or thermal power, we opted to provide a 50% capacity fossil-fired gas turbing standby unit. The need for a backup is even more understandable when one considers that the above analysis does not take into account outages caused by the reactor plant used to drive the turbine units--whether forced outages (about 5% on Peachbottom) or, what is even more important in the present instance, refueling/maintenance downtime. One can readily envision a net effective value of p as low as 0.8, which would correspond to around two weeks of total plant outage per year unless backup fossil capacity were provided. Similar considerations induce commercial utilities to restrict individual units to on the order of ten percent of system capacity and to maintain around twenty percent reserve in order to achieve a high degree of supply reliability. In the present case we cannot hope to match commercial performance, and considerable attention will have to be paid to keeping outages to a tolerable level.

6.6 Conclusion

Use of an indirect cycle has created a safety picture having a close resemblance to that of the Peachbottom I reactor. Furthermore we have incorporated into our design features which should enhance the safety-related aspects of this type of design. Thus the major significant difference in this regard is the scaled up rating (factor of 2.5). In addition to this useful U.S. licensing precedent there is the far more extensive British and European experience with similar designs (Dragon, AVR, THTR) and a large number of graphite-moderated gas-cooled predecessors.

Thus the key question of licensability boils down to whether Peachbottom I could be re-licensed today. While we have no indication that it could not, only NRC review could establish this point in an unequivocal manner. Since no plant of this type has been licensed in the U.S. in some ten years now, the review would probably focus on whether in developing the present design, in our attempt to employ proven concepts, we have instead adopted outmoded practices. The integral-design (PCRV) units--Fort St. Vrain, Delmarva and Philadelphia Electric--have unquestionably established a new set of precedents, the most important of which from our point of view is the reduced vulnerability to primary coolant blowdown. Thus this accident must be the focus of any subsequent efforts to carry this design further down the road to deployment.

Chapter 7

ECONOMIC EVALUATION

7.1 Introduction

An extensive analysis of the economic prospects of a HTGR/GT unit for total utility service has been carried out by L. J. Metcalfe and reported in Ref. (M1); a more comprehensive report (SM thesis) is also being prepared on this subject (M5). Therefore, only a brief summary of the findings will be presented here.

7.2 Background

Table 7.1 summarizes the groundrules governing the economic environment under which the cost evaluation was carried out. Capital costs were evaluated using the AEC's (now ERDA) computer program CONCEPT III (B7). Because of the uncertainty in future fossil fuel prices, the results have been developed in terms of breakeven prices for coal and oil relative to the HTGR/GT system. Fuel cycle costs for the nuclear alternatives have been developed from available industrial data, modified to account for government ownership, and operation and maintenance costs have been estimated using published estimates for similarly rated units in civilian service (I1).

Table 7.1

Groundrules for Economic Studies

Plant Power Rating--100 Mwe, approximately 200 Mwth

Plant Types--nuclear: HTGR/Brayton, PWR/Rankine

fossil : Coal- and oil- fired Rankine
Oil-fired Gas Turbine/Total Energy Unit

Site--AEC's "Middletown", USA

Construction time--fossil-fired Rankine 5 years:

fossil-fired Gas Turbine: 3.5 years

nuclear: 7 years

Date of Operation--1985

Cooling--Mechanical Draft Cooling Towers

Environmental--near zero rad waste systems for nuclear stations; SO_x removal systems for coal stations

Work Week--40 hours, no overtime

Cost of Money--10%

Escalation--8% labor

5% materials

Single unit on site; two 50 MWE units for FFGT

80% operating capacity factor

30 year plant lifetime

NUCLEAR FUEL CYCLE:

carrying charges waived on in-core fuel

no credit for bred fissile material in spent fuel

batch-loaded core with assured lifetime of 3 calendar years

7.3 Results of Economic Evaluation

Table 7.2 summarizes the results of the economic studies reported in Ref. (M1). Several points are of interest:

(a) the PWR can deliver electric power more economically than the HTGR/GT unit in the electric-only mode of operation, but the HTGR/GT is superior for total energy applications. As shown in Appendix B, the PWR rating required for total energy service is inherently larger than that of the HTGR/GT unit because prime steam must be diverted to heat utility water in the former, whereas in the latter the energy used for this purpose is truly "waste heat" from the power conversion cycle.

(b) the fossil-fired gas turbine (FFGT) is the principal competition for the HTGR/GT for the proposed application.

(c) fossil-fired Rankine cycle units do not appear to be cost competitive since their breakeven fuel costs in 1974 dollars are below values already being experienced.

Thus the choice comes down to a selection between two quite different alternatives: the high capital cost, low fuel cost HTGR vs the low capital cost, high fuel cost FFGT. Because of the uncertainty in both the present capital and fossil fuel markets, it is difficult to project the status of the comparison over the thirty year design life of the plants (starting from 1985). What now appears as essentially a standoff could well develop into a clear choice over such a long time span.

Table 7.2

Summary of Economic Comparisons (In 1985 $)
For Units Delivering ≤ 100 MWe, ≤ 200 MWth

Plant Type	Cost of Electric Power	Cost of Thermal Energy	Fossil Fuel Breakeven Price
HTGR/GT (100 MWe equivalent rating @ 35% efficiency)			
Electric only	41.5 mills/kwhr		
Thermal only		4.28 $/MBTU	
Total Energy	25.2	2.58 $/MBTU	
PWR (173 MWe equivalent* rating @ 33% efficiency)			
Electric only	30.3 mills/kwhr		
Thermal only		2.93 $/MBTU	
Total Energy	30.3	2.93 $/MBTU	
OIL/RANKINE (147 MWe equivalent* rating @ 38.5% eff.)			8.83 $/bbl
COAL/RANKINE (147 MWe equivalent* rating @ 38.5% eff.)			15.60 $/Ton
FFGT (100 MWe equiv. rating @ 31.8% eff.)			18.13 $/bbl (∼ 10.60 $bbl in 1974

*equivalent to 100 MWe total energy system satisfying same thermal and electrical loads as the HTGR

7.4 Discussion

Under the terms of the comparison we have seen that the HTGR/GT is the superior nuclear alternative and better than all but one of the fossil alternatives, against which it is essentially a stand-off. In some respects, however, the framework has been prescribed in a manner more, rather than less, favorable to the nuclear option in general and the HTGR/GT in particular. In order to appreciate this aspect it is necessary to consider the following points:

(a) System reliability is a key factor for a stand-alone unit. It is unlikely that a single nuclear unit can be relied upon to provide more than about 90% availability, hence a 50% capacity FFGT has been specified for standby service. FFGT units, however, can be provided in modular blocs, and thereby achieve very high availability factors for the power system.

(b) The economic viability of the capital-intensive nuclear system is more sensitive to system capacity factor than the fuel-cost-intensive fossil systems. In a situation where demand is growing, one is faced with the dilemma of either oversizing the nuclear unit at beginning-of-life to meet end-of-life demand, and thereby incurring a low capacity factor over life, or sizing the nuclear unit to meet BOL demand and thereby defaulting on a large sector of the future load, which must be met by other means.

(c) The system has been designed to satisfy a thermal/electrical demand ratio of roughly 2/1. If the load is all electrical or all thermal, then the PWR is the preferable nuclear alternative. Work to establish the potential size of the thermal demand and the fraction which can be economically reached by a central utility system is being carried out in parallel with the present study, hence a definitive demand schedule has not been factored into the economic evaluation. Note, however, the analysis summarized in Appendix B which shows that use of absorptive air conditioning and heat pumps can often tailor the load to match the power conversion system's capabilities.

(d) Due notice should be taken of the large increases in and uncertainties in nuclear power plant capital costs of late. Otherwise comparable units have quadrupled in cost in less than a decade and contemporary units exhibited a factor of two range in unit costs ($/Kwe). While this uncertainty can undoubtedly be narrowed in the present case by use of a standardized design and by modular construction where practicable, the capital cost uncertainty will still be considerably greater than for the FFGT units, where turnkey purchase contracts are currently available.

On the other side of the balance, the nuclear option has been penalized somewhat by overconservatism in one area: a 3-calendar-year core lifetime has been assumed because it is certain that this level of performance could be achieved under warrantee. Work continues at MIT, however, to extend the core design life to six calendar years-an objective now considered practicable.

In conclusion, because of the dominant role of capital costs for the nuclear units (which contribute on the order of 2/3 of the total cost of power) it is a logical next step to obtain a more detailed design and cost estimate from a reactor vendor/architect engineer before making an unqualified commitment to the nuclear alternative for the present application. It would also be desirable to carry out an in-depth assessment of FFGT units consuming various fuels: domestic or imported oil, shale oil, syncrude from coal and high or low BTU gas from coal, since these options appear to offer the strongest competion to nuclear.

CHAPTER 8

SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

8.1 Introduction

The central objective of the study summarized in this report was the development of a conceptual design for nuclear power plants to provide total utility service for some of the larger DOD bases and industrial facilities in the continental U.S. in the post-1985 time frame. The present report focuses on the development of the technical description of the reactor and power conversion systems. Separate reports deal with the subjects of economic evaluation of the various alternatives (M1) (M5), environmental and public safety impact (D1) (D3) and thermal utility system optimization (S3). In the present report, however, we have summarized and/or previewed the pertinent aspects of these parallel efforts.

8.2 Summary Description

The concept developed in response to the defined needs involved a high temperature gas-cooled reactor (HTGR) indirectly coupled (via intermediate heat exchangers) to twin closed-cycle gas turbine units employing helium as a working fluid. Table 8.1 summarizes the key features of the HTGR/GT system. In many respects the plant may be considered a hybrid of the technology proven-out on Peachbottom I (reactor systems) Fort St. Vrain (core) and Oberhausen II (turbomachinery). As such, the design can be built without appreciable additional R&D effort.

Table 8.1

Summary Description: Military Base HTGR-GT

Reactor	
Nominal Rating:	100 MWe; 200 MWth
Efficiency:	33%
Core:	Hexagonal block graphite U^{235}/Th^{232} GA-type fuel cycle $\geq$ 3.0 effective full power year batch lifetime.
Outlet Conditions:	1500°F, 400 psia.
Primary System	Steel pressure vessel, non-integral design. Two main coolant loops. Indirect cycle, helium coolant. Electrically driven circulators.
Turbine Plant	Two independent turbomachinery groups. Horizontal arrangement. Regenerative, no intercooling. Delivers precooler effluent to utility system at 380°F. Turbine inlet temperature = 1425°F.
Heat Sink	Forced-draft wet cooling tower, 200 MWth (100% of plant full power waste heat rating).
Utility System	Hot water type; 380°F supply, 150°F return. Heat pumps and adsorptive air conditioning used to extend reactor capabilities.
Backup Systems	Fossil-fired GT total energy unit rated at 50 MWe, 100 MWth.

Although the plant efficiency for conversion of thermal to electrical energy is scarcely better than that of a LWR, two points should be kept in mind:

(a) the HTGR/GT unit provides "waste" heat at a directly useful temperature for the hot-water-type thermal utility system, while the PWR must divert prime steam to service this load.

(b) conversion efficiency is not the proper measure of total energy system efficacy--since the planned applications have projected thermal to electrical demand ratios generally in excess of 2:1, even lower efficiency would be adequate.

Thus it is because of the unique characteristics of the total energy application that the HTGR/GT is favored and can meet all requirements using conservative technology.

8.3 Conclusions

As a result of the work carried out to produce and evaluate a conceptual system design, the following key conclusions were developed:

(a) The HTGR/GT system is considerably superior, in an economic sense, to fossil-fired Rankine cycle systems, and significantly superior to LWR units for total energy loads which match the natural thermal/electrical ratio of the unit, $(1-\eta)/\eta$, or which can be tailored to match by use of absorptive air conditioning or heat pumps.

(b) The HTGR/GT and FFGT systems are essentially an economic stand-off, but some of the key groundrules favor the nuclear unit - in particular the assumption of an 80% capacity factor averaged over the life of the plant.

(c) Reliability considerations favor use of a 50% capacity FFGT on standby to backup the nuclear unit.

(d) While all aspects of the design are state-of-the art and no extensive R&D program is required to bring the concept to a deployable status, several key components are one-of-a-kind designs: the helium turbomachinery and 1HX units in particular. Otherwise extensive "borrowing" can be done from off-the-shelf stock for other applications.

(e) The HTGR has exceptionally good safety-related characteristics; however a concerted analytic effort will have to be devoted to the primary system blowdown accident because of the non-integral design employed. No plant of this general type has been licensed in the U.S. in over ten years. This, plus the fact that up to a dozen units would be built to the same design, suggests that intensive scrutiny in the licensing effort is also in prospect.

(f) In a similar vein, a conscious effort has been made to achieve simplicity of design and operation and economy of cost--in part by using minimum practicable redundancy (e.g., two main loops) and by calling upon systems to do double-duty (e.g., normal, shutdown, and emergency cooling; normal and emergency containment air purification). Hence reliability and susceptibility to common-mode failures must be given due attention in further design efforts.

A final observation is that the plant design is inherently flexible in several respects: alternate subsystems can be substituted for those called for in the reference design (e.g., hydraulic in lieu of electro-mechanical control rod drives); and major units in the power conversion train can be interchanged--steam generators or chemical reactors can be employed in parallel to or in place of the IHX (as in the proposed JAERI HTGR).

8.4 Recommendations

Based upon the conclusions developed as a result of the work documented in this report and the other contributions referenced herein, the following recommendations are made:

(a) The HTGR/GT system be carried through to the point of vendor/architect engineer evaluation to develop a more firm capital cost estimate, since this accounts for most of the product costs for the capital-intensive nuclear system. A prospective licensing effort and schedule should be discussed with the NRC. Finally, the advantages to the HTGR of a longer core life should be factored into the evaluation.

(b) Detailed evaluation be made of the various fossil-fired gas turbine (FFGT) total energy systems which could be used to replace or supplement the HTGR/GT unit: in particular the base-loaded-nuclear, fossil-peaking mode.

(c) Various thermal utility scenarios be examined to resolve the following issues:

(1) whether primarily electric or primarily thermal systems are likely or practicable, since the HTGR/GT plant loses its advantage over LWR's in the all-electric or all-thermal limits;

(2) whether substantial differences exist in utility systems designed to be coupled to FFGT as opposed to HTGR/GT power units -- for example: amenability to solar supplementation, use of heat pumps, adsorptive air conditioning, energy storage, supplementary trash incineration.

In brief, our results tend to support the conclusions of recent studies that small nuclear plants can now even compete with fossil-fired units for all-electric or all-thermal service (I1) (T2). With the added advantage of a total energy application, the HTGR/GT system reduces unit energy costs to about 60% of single-product costs. It is perhaps not surprising, then, that only the fossil-fired gas turbine/total energy unit appears competitive with the nuclear system. Moreover, the FFGT does not necessarily have to bear the now-heavy burden of being suited only to use of natural gas or petroleum-based fuels. Synthetic fuel from coal or shale oil now appear to be viable alternatives, and an even more suitable system in which low BTU gas is generated from coal on site is now practicable - a demonstration unit having operated since 1972 in Germany (A2).

Thus, before making a final selection of an energy system for large military installations and industrial facilities it is recommended that FESA evaluate the competition between the HTGR/GT and FFGT in greater depth.

APPENDIX A

SELECTION OF GASEOUS WORKING FLUIDS

A.1 Introduction

Although much has been written on the subject, (M7, M8), it is considered a useful exercise to display an independent evaluation of the physical bases underlying the choice of the gaseous working fluids in the primary and turbine circuits.

The following groundrules will be applied:

(a) All temperature state points will be kept the same in order to maintain comparable thermodynamic efficiencies. Thus the temperature change through all heat exchangers and across all heat transfer boundaries will be held constant.

(b) Similarly, the ratio of pumping power expended in overcoming pressure drop to energy transferred will be held constant. This latter quantity is also proportional to turbine or compressor work (i.e., energy transformed), hence other constraints are also implied.

(c) Several ideal gas approximations will be employed: constant Prandtl number (same value for all gases) and the ideal gas law.

(d) We will assume system pressure is fixed, since this is a key determinant of system cost.

(3) Two variations of heat exchanger or core design will be considered: variation in channel length

at constant diameter; and variation in diameter at constant length

A.2 Derivation

The following relations are employed -- using conventional notation throughout:

(a) heat transport per channel

$$q_t = \rho \cdot \frac{\pi D^2}{4} V\, C_p\, \Delta T \tag{A.1}$$

(b) heat transfer per channel

$$q_F = h \cdot \pi D L \cdot \Delta t \tag{A.2}$$

(c) heat transfer coefficient

$$\frac{hD}{k} = 0.023 \left(\frac{DV\rho}{\mu}\right)^{0.8} Pr^{0.4} \tag{A.3}$$

(d) Prandtl number

$$\frac{C_p \mu}{k} = 0.7 \tag{A.4}$$

(e) Ideal Gas law

$$\rho = \frac{MP}{RT} \tag{A.5}$$

(f) Pressure drop

$$\Delta P = 4f \frac{L}{D} \frac{\rho V^2}{2g_c} \tag{A.6}$$

(g) Friction factor

$$f = \left[\frac{DV\rho}{\mu}\right]^{-0.2} \tag{A.7}$$

(h) Pump work

$$W = \Delta P \cdot N \cdot \frac{\pi D^2}{\mu} V \qquad (A.8)$$

where N is the number of channels

$$N = Q/q_t \qquad (A.9)$$

and Q is the total thermal rating -- same for all cases.

The derivation may now proceed as follows, considering the case in which D is held constant and L is varied:

(1) Equate Eqns. (A.1) and (A.2); employ Eq. (A.3) for h and use Eq. (A.4) to remove Pr and to replace k by the equivalent product $C_p\mu$. Finally use Eq. (A.5) to replace $\rho$by M (recalling that P and T are held constant). Then we can solve for L in terms of the remaining variables.

$$L \sim M^{0.2}\ V^{0.2}\ \mu^{-0.2} \qquad (A.10)$$

(2) Now apply the condition (from Eqs. (A.8) and (A.9))

$$\frac{W}{Q} \sim \Delta P \cdot \frac{D^2}{q_t} \cdot V = \text{constant} \qquad (A.11)$$

Inserting Eq. (A.1) we can solve for V

$$V \sim M^{1/9} Cp^{5/9} \mu^{-1/9} L^{-5/9} \qquad (A.12)$$

Using Eq. (A.10) to eliminate L:

$$V \sim Cp^{1/2} \qquad (A.13)$$

, a remarkably simple result.

From Eq. (A.10)

$$L \sim M^{1/5} Cp^{\ 1/10}\ \mu^{-1/5} \qquad (A.14)$$

Equation (A.9) allows us to solve for the number of channels:

$$N \sim M^{-1} Cp^{-3/2} \tag{A.15}$$

Finally the heat exchanger surface area can be determined:

$$S \sim LN \sim M^{-6/5} Cp^{-7/5} \mu^{-1/5} \tag{A.16}$$

as can the frontal flow area:

$$A \sim ND^2 \sim M^{-1} Cp^{-3/2} \tag{A.17}$$

a final parameter of interest is the duct diameter. For the same duct length and pumping power expenditure, Eqns. (A.6, A.7, A.8) combine with:

$$Q = \rho d^2 V \; Cp \; \Delta T = \text{constant}$$

to give

$$d \sim M^{-5/12} \; Cp^{-7/12} \; \mu^{-1/24} \tag{A.18}$$

A similar derivation can be carried out for fixed L and variable D. The table below summarizes the results:

Parameter	Vary L, Fix D	Vary D, Fix L
Channel flow velocity, V	$Cp^{1/2}$	$Cp^{1/2}$
Number of tubes, N	$M^{-1}Cp^{-3/2}$	$M^{-2/3}Cp^{-4/3}\mu^{-1/3}$
Channel surface area, S	$M^{-4/5}Cp^{-7/5}\mu^{-1/5}$	$M^{-5/6}Cp^{-7/12}\mu^{-1/6}$
Channel length, L	$M^{-1/5}Cp^{-1/10}\mu^{-1/5}$	---
Tube diameter, D	---	$M^{-1/6}Cp^{-1/12}\mu^{1/6}$
Frontal flow area, A	$M^{-1}Cp^{-3/2}$	$M^{-1}Cp^{-3/2}$
Duct diameter, d	$M^{-5/12}Cp^{-7/12}\mu^{-1/24}$	$M^{-5/12}Cp^{-7/12}\mu^{1/24}$

Further simplifications are possible. For a gas molecule (mono-, di- or triatomic) having n atoms per molecule the heat capacity is given by

$$Cp \sim FM^{-1} \tag{A.19}$$

where $F = (1 + n + n^2)$ (A.20)

and the viscosity (very approximately) varies as

$$F^{-0.5}M^{0.07} \tag{A.21}$$

Substitution of these relations into the above table gives the simpler comparison:

Parameter	Fixed D	Fixed L
V	$M^{-0.5}F^{0.5}$	$M^{-0.5}F^{0.5}$
N	$M^{0.5}F^{-1.5}$	$M^{0.643}F^{-1.167}$
S	$M^{0.586}F^{-1.3}$	$M^{0.572}F^{-1.33}$
L	$M^{0.08}F^{0.2}$	---
D	---	$M^{-0.07}F^{-0.017}$
A	$M^{0.5}F^{-1.5}$	$M^{0.5}F^{-1.5}$
d	$M^{0.17}F^{-0.604}$	$M^{0.17}F^{-0.604}$

Note that if the comparison is restricted to the monoatomic inert gases (almost a necessity for the present high temperature application) further simplification is possible: the F dependence can be suppressed since n = 1, F = 3.

From this last table we see that tube or channel length and diameter are very weak functions of gas properties. Hence we are justified in comparing systems having the same length and diameter for the channels (tubes) in the core and other heat exchangers. Moreover N, S, A have roughly the same dependence. Thus we can finally reduce the comparison to:

$$\text{Coolant velocity} \quad \sim \left[F/M\right]^{1/2} \tag{A.22}$$

$$\left.\begin{array}{l}\text{Number of tubes}\\ \text{Heat transfer surface area}\\ \text{Frontal flow area}\end{array}\right\} \sim \left[\frac{M}{F^3}\right]^{1/2} \tag{A.23}$$

Since the overall cost and size of heat exchange equipment are closely related to the parameter of Eqn. (A.23), we can readily see the superiority of helium with its low molecular weight and monoatomic molecule. Equation (A.22) indicates that high flow velocities are to be expected: however this is largely compensated for by the high sonic velocity in helium which leads to tolerable Mach numbers.

As an example we may compare argon (M=40) and helium (M=4) heat transfer surface areas: according to Eqn. (A.23) argon will require $\sqrt{40/4}$ = 3.2 times the area (number of tubes, etc.). This result is confirmed by other studies (A1), and helps explain the preference given helium as a working fluid.

A.3 Turbomachinery Design

While the preceding considerations, relating as they do to the important variables governing heat exchanger costs, are an essential ingredient in the design trade-off process, it is also important that one consider turbomachinery design characteristics, since there are significant differences related to the properties of the working fluid which in many respects tend to act in an opposite manner compared to their effect on heat exchanger design. In addition to the information on Oberhausen referenced in the body of this report, Refs. (E1) and (B10) provide useful general background material on the subject of nuclear and closed-cycle gas turbines.

The change in enthalpy across a compressor or turbine can be written in the form:

$$n \, \phi \, \frac{u^2}{g_c} = C_p \, \Delta T \tag{A.24}$$

n = number of stages

u = average blade tip velocity

ϕ = flow coefficient ($\sim$ 0.4 for the compressor, $\sim$ 1.0 for the turbine) - an indicator of necessary blade stagger

For comparable* turbomachinery, then:

$$n \sim \frac{C_p}{u^2} \sim \frac{F}{Mu^2} \tag{A.25}$$

*i.e., fixed ratios of all velocity vectors

Precise specification of the effect of the choice of working fluid on turbomachinery characteristics is complicated by the rather broad spectrum of acceptable design trade-offs involved. Nevertheless some approximate comparisons can be made by restricting considerations to several limiting cases:

Case A - Fixed Rotational Speed

This case would arise if all machinery were mounted on a single shaft and/or constrained to match generator specifications (without gearing). Blade tip speed, hence axial flow rate is proportional to the product of radius (diameter) and speed of rotation:

$$u \sim D\omega, \text{ or } u \sim D \text{ for constant } \omega \qquad (A.26)$$

But $\overset{\circ}{m}\, Cp\, \Delta T \sim \rho D^2 u Cp \Delta T = \text{constant}^*$, and since $\rho \sim M$ and $Cp \sim M^{-1}$ (ignoring the differences among mono-di- and triatomic gases):

$$D^2 u = \text{constant} \qquad (A.27)$$

Hence from Eqs. A.26 and A.27, both u and D are constant.

From Eq. A.25 we then have:

$$n \sim Cp \sim M^{-1}\,, \qquad (A.28)$$

or the number of stages required vary inversely as the molecular weight of the gas.

If we further assume that the ratio of blade thickness

*We assume a fixed ratio of blade tip to rotor hub diameters - hence flow area is a fixed fraction of machine diameter.

to length is fixed, then the length of the machine is also proportional to L.

Thus a low mass gas such as helium will tend to require a large number of turbomachinery stages: Eq. A.28 shows that helium would require ten times as many stages as argon.

To summarize; for fixed-speed turbomachinery:

$$\left.\begin{array}{l}\text{gas velocity}\\ \text{diameter of machine}\\ \text{volumetric flow rate}\end{array}\right\} \quad \text{constant}$$

$$\left.\begin{array}{l}\text{number of stages}\\ \text{length of machine}\end{array}\right\} \quad \sim M^{-1}$$

$$\text{mass flow rate} \quad \sim M$$

Case B Stress-limited design

Changes in generator design or use of reduction gearing can permit some flexibility in the choice of rotational speed. Ultimately, however, we are limited by the centrifugal stress in the blade hub.

Assuming that the ratios of hub-to-tip diameter and blade thickness-to-length remain fixed the centrifugal force is:

$$F \sim \frac{mv^2}{r} \sim (D \cdot D^2) \frac{D^2 \omega^2}{D} \sim D^4 \omega^2$$

while the blade root area, $A \sim D \cdot D \sim D^2$

Thus the stress, $\sigma = \frac{F}{A} \sim D^2 \omega^2 = \text{constant};$

but since $u \sim v \sim D\omega$, $\sigma \sim u^2 = \text{constant}$.

If $u = \text{constant}$, since $pD^2uCp = \text{constant}$ and $pCp \sim \text{constant}$, we also have $D = \text{constant}$; thus again:

$$u, D = \text{constant}. \tag{A.29}$$

As before, therefore, the number of stages and the length of the turbomachinery is:

$$L \sim n \sim Cp \sim M^{-1} \tag{A.30}$$

Case C Gas-Velocity Limited design

We are not free, other constraints being satisfied, to set an arbitrary gas velocity, since excessive Mach numbers must not be achieved. Operation at equal Mach number requires

$$u \sim M^{-1/2} , \qquad (A.31)$$

where we have again ignored the differences among mono-, di- and triatomic gases by introducing only the mass dependency for the speed of sound.

Combining Eqs. (A.25) (A.31) and $Cp \sim M^{-1}$, we find that $n \sim$ constant, or all gases would require the same number of stages: in strong contrast to the more familiar and more often cited constant rpm results.

Furthermore, since D^2u = constant, $D \sim M^{1/4}$, and heavier gases would require larger machine diameters. Finally, mass flow rate $\sim M$; volume flow rate $\sim$ constant, as before. One can find studies which support these uncommon conclusions on turbomachinery design: Reference (H3) suggests that the number of stages of turbomachinery necessary for air and helium (indeed, any gas) will be approximately equal, but the diameter of the helium machine will be less - because the Mach Number constraint is applied.

In actual practice it is seldom that the designer opts for one of the above limiting cases, but rather seeks a compromise solution which combines several features of each case. He also has the added flexibility of varying the percentage of reaction blading in the turbine and the blade stagger in the compressor. A good example of a compromise design is contained in reference (A1), which compares units designed for the same service, individually optimized to use helium and argon, respectively; in their final design the helium turbomachinery had roughly double the number of stages and operated at twice the rpm of the argon turbomachinery. Again, while these gross generalizations should be applied with some caution, we also note that in Ref. (A1) the helium machinery is longer and the argon machinery of greater diameter, as predicted. However, in the compromise solution the differences are not all that significant.

* * *

Another consideration related to gas properties is the pressure ratio across the turbine/compressor units. The optimum cycle compression ratio changes both with the nature of the working fluid (atoms/molecule) and the type of cycle (regenerative/non-regenerative): in general monatomic gases and regenerative cycles have lower optimum compression ratios. For a simple Brayton cycle we have:

$$\ln\ r_{opt} \sim \frac{\gamma}{\gamma-1} = \frac{Cp}{R} = \frac{F}{M} \qquad \text{(A.32)}$$

Thus helium can operate at a lower pressure ratio than heavier gases: in Ref (A1) a ratio of approximately 2.9 was used for helium vs. 4.5 for argon. This advantage manifests itlsef through higher gas densities (smaller ducts) and/or lower pressures (thinner duct walls).

* * *

As a compromise between the heat-exchanger-favoring properties of low mass gases and the turbomachinery-favoring properties of heavier gases, it is sometimes proposed (B11) that mixed gases be employed for cycle working fluids (He - Ne, He - Ar, He - Xe). Helium's lower cost and freedom from induced radio-activation, however, have led us not to pursue these options.

* * *

In conclusion, while it is clear that helium turbomachinery will assume design characteristics substantially different from the familiar open-cycle gas turbines operated on combustion gas, the principles (and experience) required to design and build suitable turbomachinery at a reasonable cost is available. Reference (G7) quotes prices for the first-of-a-kind Oberhausen II turbomachines (including foundations and auxiliaries) equivalent to $\$12.43 \times 10^6$ (U.S.)[*]. While this represents 248 \$/kwe, Ref (G7) also makes the explicit point that "You should bear in mind that the dimensions of these components represent a 300 mwe turboset" - which would imply a scaled cost of around 41 \$/kwe. The same source indicated a cost for the heat exchange components (except for the fossil-fired heater) of $\$3.79 \times 10^6$ (U.S.).

[*]converted using 1 DM = \$0.4075 U.S. as of 12/5/74

APPENDIX B

CHARACTERISTICS OF NUCLEAR TOTAL ENERGY SYSTEMS

B.1 Introduction

The fact that the present application involves total energy service has a profound effect on several key design decisions, including choice of reactor type, specification of required power conversion cycle efficiency, and the use of ancillary equipment in the utility system to match the demand spectrum of the energy sink to that of the energy source. In the sections which follow a somewhat oversimplified treatment will be presented in order to clarify the various issues.

B.2 Selection of an HTGR/Brayton Unit over a LWR/Rankine Unit

The widespread use of light water reactors in both commercial and military applications raises the question as to why they would not be preferable as well in the current application. The superiority of the HTGR/Brayton unit is due to the fact that the thermal energy supplied to the utility system is truly waste heat from the power conversion system.

Let us compare the reactor rating, Q_R, required to satisfy utility hot water and electric loads. Qu and W_e respectively, for the LWR/Rankine (subscript L) and HTGR/Brayton (subscript H) systems.

$$Q_{RL} = Qu + W_e/\eta_L \tag{B.1}$$

$$Q_{RH} = W_e/\eta_H \text{ if } (\frac{Qu}{W_e}) < (\frac{1-\eta_H}{\eta_H}) \tag{B.2}$$

$$Q_{RH} = Qu + W_e \quad \text{if} \quad (\frac{Qu}{W_e}) > (\frac{1-\eta_H}{\eta_H}) \tag{B.3}$$

The ratio of LWR to HTGR reactor ratings is:

$$\frac{Q_{RL}}{Q_{RH}} = \eta_H \quad (\frac{Qu}{W_e}) \quad + \quad \frac{\eta_H}{\eta_L} \quad \text{if} \quad (\frac{Qu}{W_e}) \quad < (\frac{1-\eta_H}{\eta}) \tag{B.4}$$

$$\frac{Q_{RL}}{Q_{RH}} = \frac{(\frac{Qu}{W_e}) + \frac{1}{\eta_L}}{(\frac{Qu}{W_e}) + 1} \quad \text{if} \quad (\frac{Qu}{W_e}) > (\frac{1-\eta_H}{\eta_H}) \tag{B.5}$$

These relations, Eqs. (B.4) and (B.5), are sketched in Fig. B.1.

Figure B.1 illustrates several key points. Since η_H and η_L are both approximately 1/3 for systems of current interest, it can be seen that for either all electric (Qu/W_e=0) <u>or</u> all thermal ($Qu/W_e \rightarrow \infty$) loads the LWR/Rankine and HTGR/Brayton systems would require roughly equivalent core ratings. In such cases the LWR would be favored because of the far greater deployment and maturity of LWR technology. However for a total energy system tailored to match the natural output of the HTGR/Brayton unit ($Qu/W_e \simeq 2$), the LWR core would have a thermal rating some 1.7 times that of the HTGR according to our simple model. Since many plant costs would scale very nearly in direct

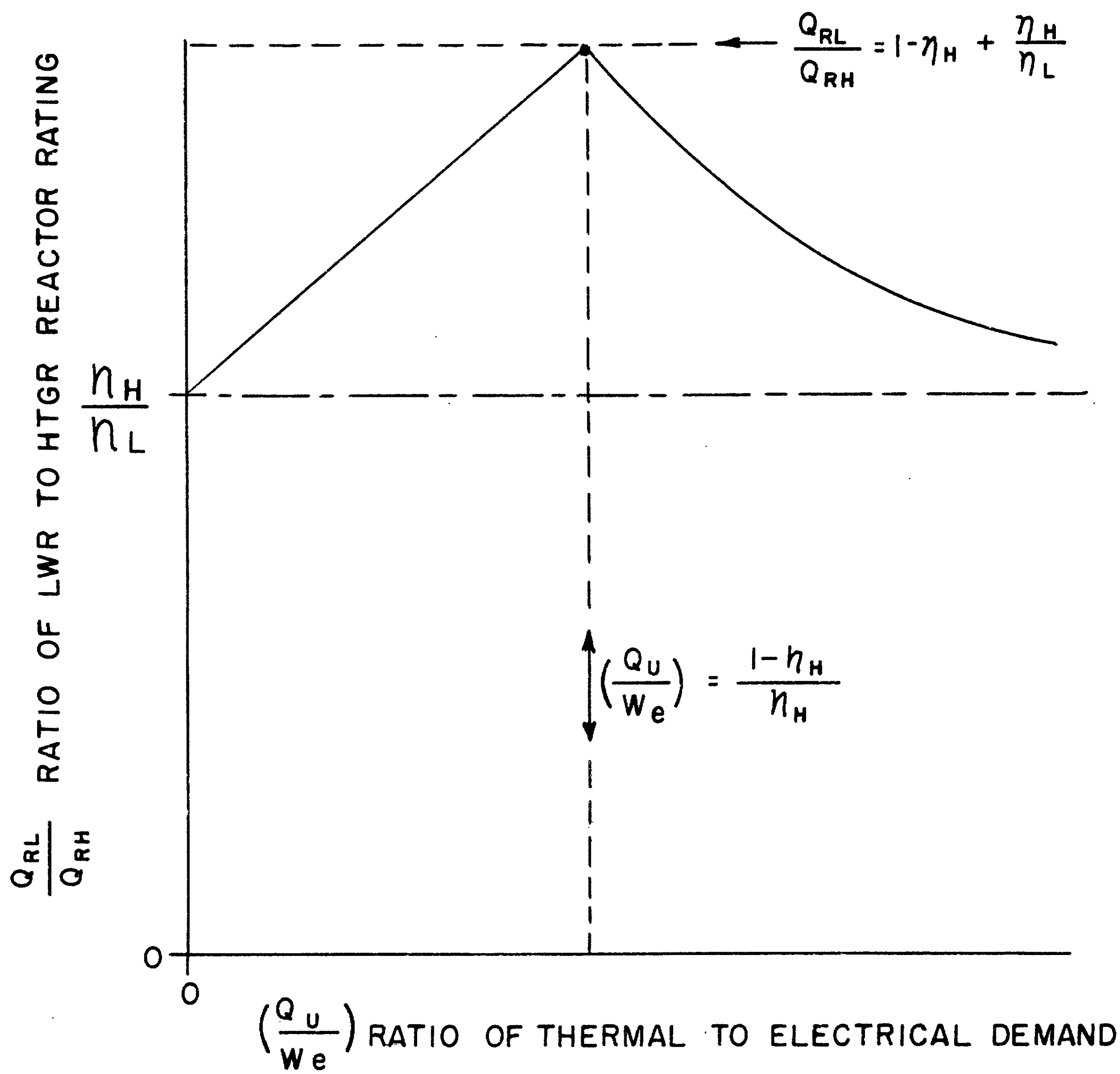


FIG. B.1 RELATIVE REACTOR RATING REQUIRED TO SERVE TOTAL ENERGY SYSTEMS proportion to the rating, this translates into a substantial cost advantage for the Brayton-type system.

The case against the PWR is not quite as overwhelming as that presented above for several reasons: some energy can be extracted from the prime steam and converted into electrical energy using a HP turbine before diverting the exhaust to heat utility water; the peak thermal and electrical loads do not occur simultaneously (on eiher a daily or a seasonal basis) as implied in Eqs. (B.1) - (B.5); and thermal energy storage can be used to smooth out the thermal demand schedule. Nevertheless, while a more sophisticated analysis would not penalize the LWR quite so severely, McRobbie has reported that even an HTGR/Rankine unit would have to be about 1.25 times as large as the HTGR/Brayton unit (M2). Differences this large give the Brayton-based system a sufficient inherent cost advantage to warrant preference over the more familiar Rankine systems for total energy applications.

B.3 Load Tailoring for HTGR/GT Systems

Given any set of simultaneous loads, We and Qu, the required reactor rating, Qe, is given by the larger of:

$$Q_R = We + Qu \qquad \text{(thermal demand dominates)} \qquad (B.6)$$

or

$$Q_R = We/\eta \qquad \text{(electric demand dominates)} \qquad (B.7)$$

We can, however, use adsorptive air conditioning or heat pumps to tailor the load.

Consider replacing 1 MWe of compressive AC delivering C_1 MWth of cooling capacity by adsorptive AC which delivers C_2 MWth cooling per MWth supplied. If δWe is the decrease in electric demand, the increase in thermal demand is $\delta Qu = -C\,\delta We$, where C is an over-all coefficient of performance, $C = C_1/C_2$ ($C_1 \approx 3.38$, $C_2 \approx 0.52$, therefore $C \approx 6.5$).

Alternatively, let us assume that the thermal demand dominates and we wish to reduce δQu by installing a heat pump such that $\delta We = -(\delta Qu/C)$, where C is again the coefficient of performance (now $C \approx 3.6$).

In both cases, then, we have

$$W'e = We + \delta We \qquad (B.8)$$

$$Q'u = Qu - C\delta We \qquad (B.9)$$

Inserting Eqs. (B.8) and (B.9) into Eqs. (B.6) and (B.7) and equating Q_R values gives:

$$\delta We = \frac{Qu - We\left(\frac{1-\eta}{\eta}\right)}{C + \left(\frac{1-\eta}{\eta}\right)} \qquad (B.10)$$

$$Q'_R = \frac{Qu + CWe}{1 + \eta(C-1)} = \frac{W'e}{\eta} \qquad (B.11)$$

Or in other words, we have transformed the load such that:

$$\frac{Q'u}{W'e} = \frac{1-\eta}{\eta} \quad \text{, the "natural" load matching condition} \qquad (B.12)$$

The value of Q'_R given by Eq. (B.11) is smaller than the Q_R given by Eqs. (B.6) or (B.7). Since heat pumps or adsorptive air conditioning are less expensive per kw (equivalent basis) than a nuclear reactor, it will always be preferable to use such devices to tailor the load to match the natural output of the heat engine.

Two examples will suffice:

A. Dominant electrical load

We = 100 MWth

Qu = 100 MWth

Q_R = 300 MWth

η = 1/3

C = 6.5 (C_1 = 3.38; C_2 = 0.52)

* * *

δWe = -11.8 MWe

Q'_R = 264.6

δQ_R = -35.4 MWth

B. Dominant thermal load

We = 100 MWth

Qu = 400 MWth

Q_R = 500 MWth

η = 1/3

C = 3.6

* * *

δWe - 35.6

Q'_R = 406.8

δQ_R = -93.2 MWth

In case A, use of 77 MWth of absorptive AC in lieu of 11.8 MWe of compressive AC (both providing 40 MWth of cooling capacity) reduces the required reactor rating by 35.4 MWth. If absorptive AC costs \$35 per KW cooling capacity and the reactor 500 \$/KWth, the net savings is:

$$35.4 \times 10^3(500) - 40 \times 10^3(35) = \$16.3 \times 10^6.$$

In case B, installation of heat pumps consuming 35.6 MWe to produce 128 MWth of heating capacity, will reduce the reactor rating by 93.2 MWth. If we assume that the heat pump costs \$35 per KW heating capacity and the reactor 500 \$/KWth, the net savings is:

$$93.2 \times 10^3(500) - 128 \times 10^3(35) = \$42.1 \times 10^6.$$

These two examples illustrate an important point, namely that it always pays to tailor the load to match the capabilities of the power conversion cycle. The importance of this conclusion is enhanced by the results of the preceding section: allowing the system to operate at its maximum η provides the greatest region of cost effectiveness, and insures that the HTGR/Brayton system is in the region where it is most superior to Rankine systems.

Thus even in the nearly all-electric or nearly all-thermal cases the HTGR should be able to preserve some degree of competitiveness with the PWR by the expedient of load tailoring. We also note that use of compression intercooling and an organic bottoming cycle could boost HTGR/GT efficiency to near 50% - which would make it very attractive for the all-electric application.

B.4 Allocation of Total Energy Costs

In the HTGR/GT system the "waste" heat from the gas turbine can be used to heat utility system water to an adequate temperature ($\sim 380^{\circ}F$), providing an essentially free source of thermal energy. By charging for the thermal energy thus produced, the cost of electricity can be partially defrayed, and the HTGR can be made more economically attractive. However, the portion of the cost of electricity which should be alloted to the thermal load is arbitrary.

An equal-fractional-savings model for the HGTR/GT can be developed by comparing the HTGR/GT operating in a Total Energy mode, to the same HTGR operated in an all-electric or all-thermal mode.

We require that:

$$\frac{e_o - e}{e_o} * \frac{d_o - d}{d_o} \quad \text{or} \quad \frac{e}{e_o} = \frac{d}{d_o} \tag{B.13}$$

where d_o = unit cost of thermal energy from a thermal-only plant, \$/MBTU (also equipped with turbines and electrical equipment to provide electric-only service, if desired)

The cost of the thermal energy, d_o, must completely assume the cost of electricity forgone:

$$d_o = \frac{e_o \eta_o}{3.413} \tag{B.14}$$

The allowable price combination is set by the requirement that one recover the same return as in the all-electric mode:

$$e_o W_o = e_o \eta_o \, Q_R = eWe + 3.413 \, Q_u \cdot d \tag{B.15}$$

where

e_o = unit cost of electricity from an electric-only system, mills/Kwhr

W_o = electric-only output, megawatts

η_o = thermodynamic efficiency

Q_R = reactor thermal rating, megawatts

e = unit cost of electricity from dual purpose plant, mills/Kwhr

d = unit cost of thermal energy from dual purpose plant, \$/MBTU

We = electric output from dual purpose plant, megawatts

Q_u = thermal energy output from dual purpose plant, megawatts

By combining Eq. (B.15) with Eqs. (B.13) and (B.14), the equal-fractional-savings model yields:

$$\frac{e}{e_o} = \frac{d}{d_o} = \frac{1}{2 - \eta_o} \tag{B.16}$$

Since $\eta_o = 1/3$, we have

$$\frac{e}{e_o} = \frac{d}{d_o} \approx \frac{3}{5} \tag{B.17}$$

Hence the product costs from a dual purpose plant are only about 60% of those from a single product plant.

It should be noted that separating the HTGR costs in this manner is only a convenience - the total dollar costs are the same regardless of how the individual product costs are allocated.

The advantages of Eq. (B.16) for dual product cost allocation are its simplicity, its lack of dependence on ill-defined alternative costs, and its reasonable pro-rata distribution of electric and thermal savings relative to other options.

Application of a similar model to a PWR, in which prime steam is diverted to heat utility water, gives $e = e_o$, $d = do$, or no savings results because of the total energy nature of the application.

B.5 Capacity Factor Over Life

The competitive cost picture presented for the HTGR/GT system in this report depends upon the ability to utilize the system at a high capacity factor over its life. This involves more than designing reliability and maintainability into the unit, as we believe we have done. An essential part of the scenario is determined by how one incorporates the nuclear unit into a given base's energy demand history, as the following simple illustration will show.

In order to approximate the lifetime average capacity factor, consider a total energy demand, D, growing continuously at a constant rate of r percent per year. (We assume both peak and average growth rates are the same).

Then the demand at time t, if the initial demand was D(0), is:

$$D(t) = D(0) \exp(rt/100) \tag{B.18}$$

and the average demand over a period T is:

$$\overline{D} = \frac{[D(T) - D(0)]}{rT/100} \tag{B.19}$$

Thus, the average system capacity factor, $\overline{C}$, over the period T for a plant sized to meet the projected demand at the end of the period, D(T), is:

$$\overline{C} = \frac{\overline{D}}{D(T)} = \frac{1 - \exp(-rT/100)}{rT/100} = \frac{1 - [\, D(0) \,/\, D(T) \,]}{\ln[D(0) \,/\, D(T)\,]} \tag{B.20}$$

For a 30 year plant lifetime (T) and a 5% energy demand growth rate (r), $\overline{C}$ is 0.52, or at best one can extract about half the rated output from the unit over its lifetime. The capacity will be further degraded by the average annual load factor attainable.

On the other hand, the unit can be sized to meet beginning-of-life peak demand. This plant can then be run at an average load factor approaching the plant availability factor, thereby achieving more economic operation. However, now the reactor cannot meet the growing demand, and alternative (e.g., fossil) sources must be periodically added to the system to satisfy the growth of demand.

In the above example, over the 30 year life span of the nuclear unit, approximately 50% of the energy would be provided by these supplementary fossil units! The first option involves paying twice as much for the nuclear energy; the second option solves only half of the fossil fuel problem.

An in-depth investigation would probably identify an optimum nuclear-fossil mix with the nuclear plant oversized at beginning-of-life but undersized at end-of-life, with a base-load-nuclear/peak-load-fossil type operation over the mid-range period. For the present report, however, all calculations were conducted with an 80% capacity factor.

APPENDIX C

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