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OVERVIEW OF GAS-COOLED REACTOR SYSTEMS: THEIR IMPORTANCE AND THEIR INTERACTIONS*†

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Abstract

Gas-cooled reactors considered to have a significant impact on the application of fission energy are the steam-cycle High-Temperature Gas-Cooled Reactor (HTGR), the Gas-Cooled Fast Breeder Reactor (GCFR), the Gas-Turbine HTGR (HTGR-GT), and the Very High Temperature Process Heat Reactor (VHTR). The importance of developing the above systems is discussed relative to alternative fission power systems involving Light Water Reactors (LWRs), Heavy Water Reactors (HWRs), and Liquid Metal Cooled Fast Breeder Reactors (LMFBRs). Further, the economic interactions between fueling, separative work, and capital requirements are illustrated, along with the implications such interactions have on gas-cooled reactor use. The associated interactions within gas-cooled reactor systems are also indicated. The influence of finite

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low-cost uranium resources and of extensive LWR application within the next two decades on reactor use is also discussed.

Technological developments required for the practical application of HTGRs, GCFRs, HTGR-GT and VHTRs are presented, along with the importance and environmental effects features of these applications. The technical advantages and disadvantages associated with use of the uranium and the thorium fuel cycles in HTGRs are given, including the implications a given fuel cycle has on fuel recycle and mined-fuel requirements. The influence of core design on HTGR fuel and coolant temperatures and on associated performance features are illustrated by considering prismatic and pebble-bed type cores. Finally, several scenarios relative to the development of the HTGR, GCFR, HTGR-GT and VHTR are presented. It is concluded that the long-term importance of the various GCRs is as follows: HTGR - providing a technology for economic GCFR and VHTR; GCFR - providing relatively low cost fissile fuel and reducing overall separative work needs; VHTR - providing a high-temperature heat source for hydrogen production processes; and HTGR-GT (in combination with a bottoming cycle) - providing a very high thermal efficiency system.

Evaluation of gas-cooled nuclear reactors needs to be done in the context of competing energy systems, considering the overall use of nuclear energy. Important factors associated with fission energy use include uranium ore requirements, separative work requirements, and capital investment needs. Since the above factors, as well as others, vary as a function of reactor type, not all reactors are necessarily economic at a given time. Further, the application of nuclear power will be dependent upon how nuclear plants compete economically with alternative energy generating plants, such as fossil-fuel-fired power plants. Also, the amount of energy which is generated depends upon the cost of generating that energy. Thus, there are many interrelated factors that have to be considered in projecting reactor use and importance. However, useful guidance can be obtained based on some general observations and results of previous studies. An important parameter is fission power growth, which is discussed below.

1. Fission Power Growth and Its Implications

First of all, electric energy use will be considered, since that is the primary application of fission reactors at this time and in the near future. In the United States, as well as in other countries, the near-term primary energy sources which appear practical for large-scale use are fossil fuels and fission fuels. Based on systems analysis studies using USA economic conditions and overall electric power growth estimates[1-5], the relative application of fission-reactor and fossil-fueled power plants has been calculated, considering power costs over a 50-year period as the objective function to be minimized. The results of these studies indicate that fission-power plants will be utilized extensively when competing against fossil-fueled plants. Although capital costs for nuclear plants have increased significantly since the time of the above studies, the costs of fossil fuels have risen to such an extent that fission-power plants still appear to be economically preferred.[6] Thus, it is expected that nuclear power growth will be substantial over the next several decades.

Estimates for nuclear power growth in the USA have varied but they all lead to high future installed capacity of fission power plants. The power growth curve is very significant to the application of various reactor types, primarily because of estimated differences in mined uranium and separative work requirements associated with use of different reactor types. Based on a reactor economy which builds 110,000 MW(e)/yr of fission-power plants past the year 2000, and which consists of Light Water Reactors, HTGRs (with HTGR capacity limited to 25% of converter reactor capacity), and LMFBRs, Figures 1 and 2 give the calculated U_3O_8 and the separative work demand as a function of time, with fast breeder reactor introduction date as a parameter.[4] The corresponding cumulative U_3O_8 requirements by the year 2020 for various reactor mixes and two power growth conditions are given in Table 1.[5] These results have certain implications relative to gas-cooled reactors. First of all, the LMFBR results apply equally to the GCFR if the latter reactor were introduced on the same schedule (but in place of the LMFBR), and if the GCFR and LMFBR had essentially the same fuel utilization characteristics.

Table 1. U_3O_8 Requirements for Different Reactor Mixes[5]

Case		Cumulative U_3O_8 Consumption to Year 2020 (Thousands of Tons)
1	No breeder, HTGR constrained to no more than 25% of total nuclear capacity	6313
2	No breeder, HTGR unconstrained	5248
3	Delayed LMFBR introduction (1991)	3408
4	LMFBR constrained to 200 GWe in year 2000, introduced 1988	3173
5	LMFBR constrained to 400 GWe in year 2000, introduced 1987 (base case)	2571
6	No constraints on LMFBR or HTGR, LMFBR introduced 1987	2494
7	Total energy demand reduced by 50% by year 2020; LMFBR introduced 1987	2039

- Probable Energy Demand
- Probable Uranium Reserves

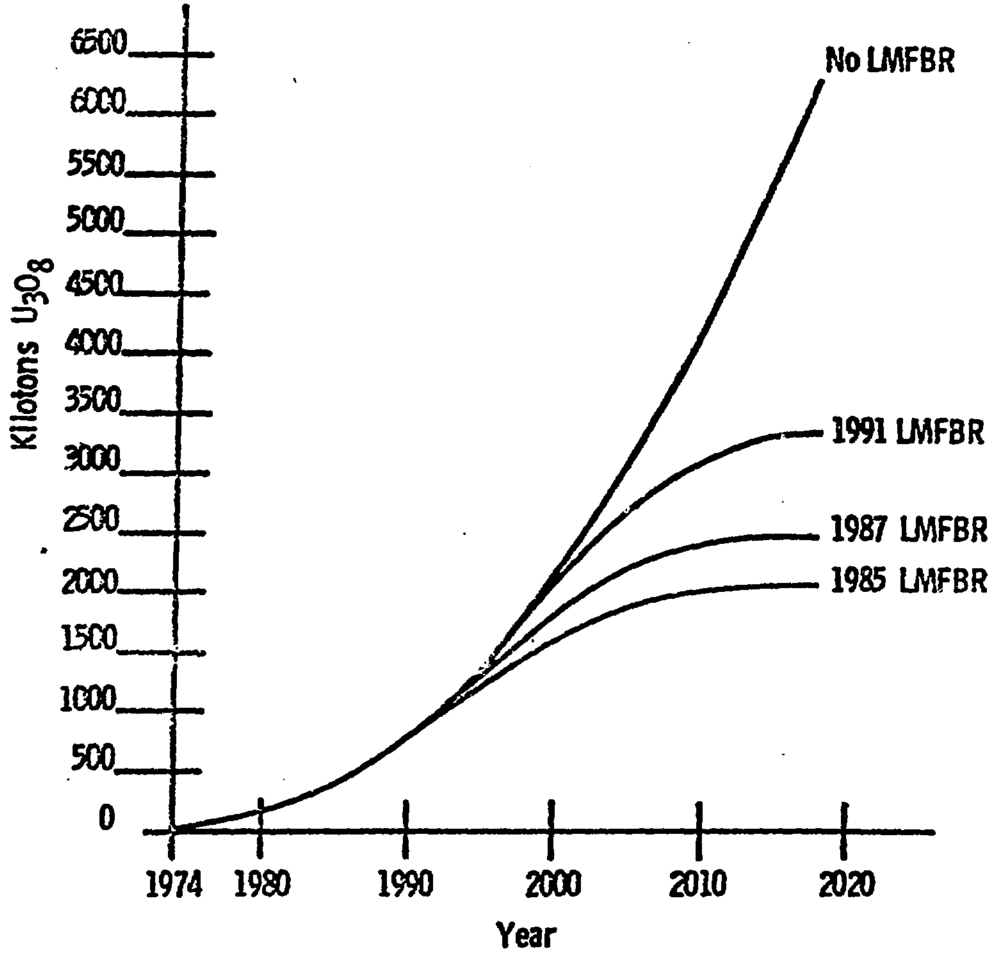

Fig. 1. Cumulative U_3O_8 Usage.[4]

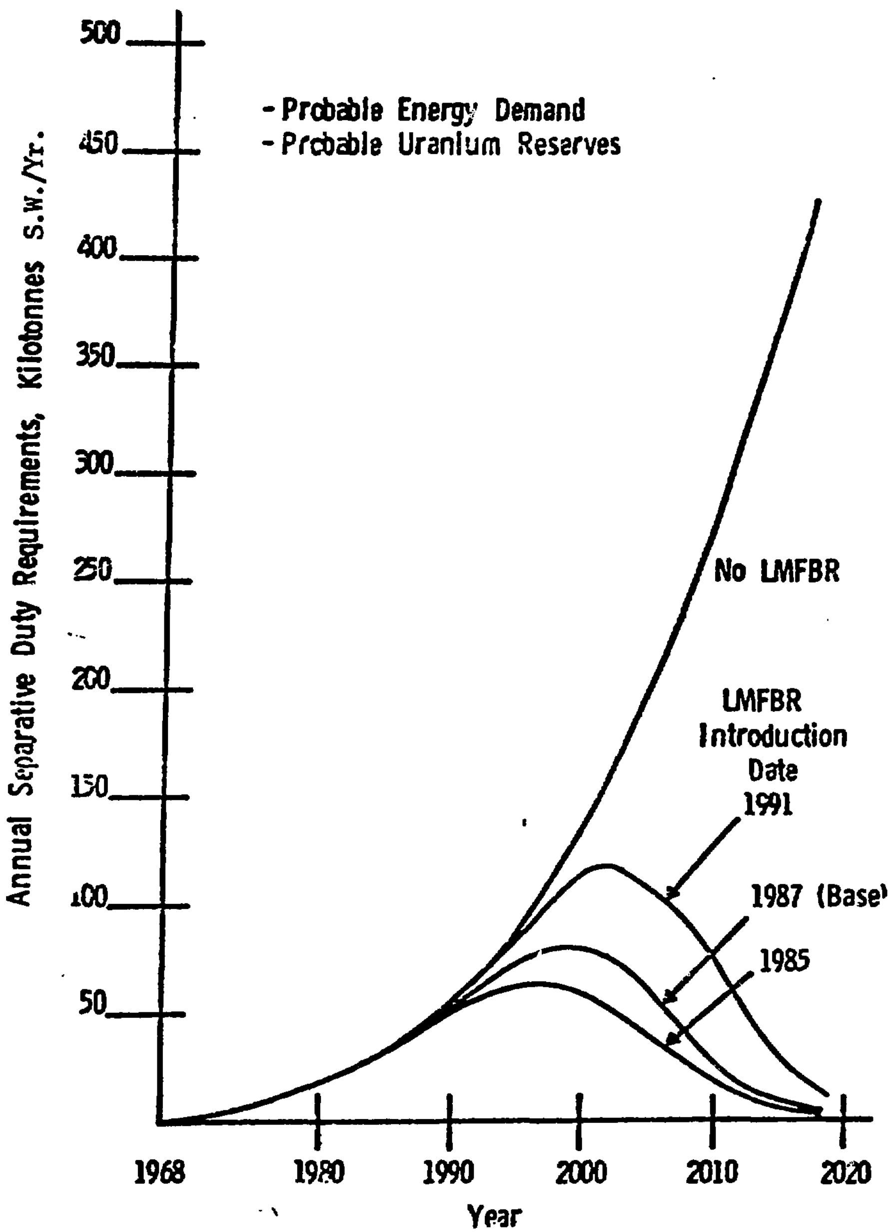

Fig. 2. Separative Work Demand Versus LMFBR Introduction Date.[4]

Since the latter condition appears to be the case, the above results can pertain to the influence of either the LMFBR or GCFR on mined U_3O_8 and separative work requirements; thus, the term Fast Breeder Reactor (FBR) rather than LMFBR appears appropriate when discussing the above results. Second, the results from Table 1 indicate that for the case where LWRs and HTGRs are utilized, and where HTGR use is unconstrained, mined U_3O_8 requirements are reduced by over a million tons over the time period to the year 2020, relative to constrained use of HTGRs. Table 1 also indicates the strong influence that delay of FBR use has on mined fuel requirements for the conditions employed. Further, decreasing the total energy demand by 50% did not have a proportionate decrease on mined fuel requirements when the reference breeder was introduced. This results because most of the fissile fuel requirements occur in the later years when the FBR dominated the nuclear economy, and because the mined fuel requirements of the breeder itself are not large.[7] However, a 50% decrease in energy usage has a proportionate effect on mined fuel requirements when no breeder is introduced.

The importance of the above is related to the amount of uranium ore available at reasonable prices. Recent estimates[8] of USA U_3O_8 resources give 700,000 tons of U_3O_8 as the reasonably assured resources at reasonable costs, and corresponding potential resources of 3.4 million tons. Based on these estimates, and the mined ore requirements given in Fig. 1 and Table 1, there is an urgent need for early introduction of FBRs to assure the long-term practicality of fission power plants having low fueling costs. Further, although HTGRs required less U_3O_8 than did LWRs (c.f. Cases 1 and 2, Table 1), the amount required was still high and more than twice as much as for the reference case with the FBR introduced in 1987.

More recent projections of U. S. nuclear power growth[9] indicate somewhat lower estimates for growth than previously. Again, considering that a linear power growth with time occurs beyond the year 2000, and that power growth beyond 2000 is most significant to application of various reactor types,[8] the fission power growth estimate corresponds to the equation

$$P = 100{,}000t \qquad (1)$$

where P = nuclear power capacity, MW(e)
t = time in years measured from 1990.

Since most of the fuel requirements in the above expanding economy are associated with converter reactor use and not with reference breeder reactor use, the fueling requirements of a nuclear power economy made up of a given reactor type will be considered. The power capacity is that given by Eq. (1) and the load factor is taken as 65%. The nominal reactor characteristics considered for specific reactor types are given in Table 2. Thus, the approximate characteristics of a

Table 2. Reactor Characteristics Considered in Evaluating Converter Reactor Fueling Requirements

Approximate Type	Specific Inventory [kg fissile/MW(e)]	Conversion Ratio	Thermal Efficiency
"LWR"	3	0.6	0.33
"HTGR-1"	2	0.65	0.39
"HTGR-2"	3.5	0.8	0.39
"HWR"	1.7	0.8	0.30
"LWBR"	6	1	0.33

Light Water Reactor are designated "LWR". "HTGR-1" refers to approximately reference HTGR parameters; "HTGR-2" refers to HTGR parameters which give a higher nuclear performance. "HWR" refers to approximately reference natural-uranium heavy-water reactor parameters. "LWBR" refers to a LWR with a conversion ratio of unity, and the approximate equivalent inventory of fissile material needed. The fueling requirements for the above reactors under the given conditions are summarized in Table 3. As shown in the table, of the reactors considered, the mined fuel requirements up to the year 2020 are lowest when HWRs are utilized; however, the results assume recycle of bred fissile material, and if that were not done, mined fuel requirements would be about 10 million tons U_3O_8 by the year 2020. Even with recycle, mined fuel requirements would be such as to lead to mining of low-grade ores. Thus, the above results also concur with the previous conclusion, i.e., FBRs are needed to maintain low fueling costs in the future.

Table 3. Approximate Fueling Requirements of Converter Reactors as a Function of Time, Based on Linear Power Growth from 1990 to 2020, with All Reactors Having 30-Yr Life, and Assuming Recycle of Bred Fissile Material

	Approximate Mined Fuel Requirements at Given Time (10^6 tons U_3O_8)					
	Year			Next 30 Years		
Reactor Type	2000	2010	2020	Burnup	Recovered Inventory	Net Use
"LWR"	1.3	3.3	6.4	4.1	2.4	8.2
"HTGR-1"	0.8	2.4	4.6	3.1	1.6	6.2
"HTGR-2"	1.1	2.6	4.5	1.7	2.8	3.4
"HWR"	0.7	1.9	3.6	2.3	1.4	4.5
"LWBR"	1.6	3.2	4.7	0	4.7	0

The results in Table 3 illustrate that so long as nuclear capacity is increasing significantly with time, inventory requirements will markedly influence mined fuel requirements. For example, improving the conversion ratio of the HTGR at the expense of specific inventory *did* not change mined fuel requirements through the year 2020, even though overall, considering reactor operation for 30 years with recovery of inventory, "HTGR-2" was significantly better than "HTGR-1." Also, the high specific inventory associated with the "LWBR" led to about the same mined fuel requirements as the "HTGR" in 2020, even though over the entire lifetime the net use would be zero for the "LWBR." It should also be noted that improved fuel utilization does not necessarily lead to improved economic performance; this is due to the timing of fueling needs, with inventory charges occurring independent of reactor use, and burnup charges occurring only with reactor use.

2. Cost Considerations

As indicated above, primary advantages for breeder development are associated with limiting the mined-fuel and separative-work requirements. Such benefits occur primarily in the future, and their large magnitudes are due to the large projected amount of energy use rather than to a large change in unit power cost. For example, considering linear power growth for 30 years in accord with Eq. (1), a 65% load factor, and 30-year utilization of all reactors which are built, results in nuclear plants generating about 255 x 10^{12} kWhr(e) by 2020; the associated subsequent energy use (until 2050) is also 255 x 10^{12} kWhr(e). A 1-mill/kWhr(e) cost savings gives undiscounted savings of about $510 billion; the discounted savings are given below.

Discount factor (from 1975):	7.5%	10%
Approx. discounted savings to year 2020 ($'s billions)	20	9.6
Approx. discounted savings to year 2050 ($'s billions)	25	11

Increasing the cost savings from 1 to 2 mills/kWhr(e) from 2010 to 2020 would add about $6.6 billion to savings when discounted at 7.5%, and about $2.6 billion when discounted at 10%. (To place such cost differentials in perspective, it is useful to note that increasing the price of U_3O_8 to $100/lb from $8/lb increases the fuel cycle cost of an LWR by about 5 mills/kWhr(e) and of an HTGR by about 3 mills/kWhr(e), based on reference designs.) If such savings can be effected by early development of a breeder, then clearly it would be advantageous economically. Also, there are related factors which are important, one being the ability to decrease the amount of low-grade ores which need to be mined. The cost of recovery of U_3O_8 from low-grade ores is highly uncertain, as is the requirement for acceptance from an environmental-effects viewpoint of such recovery. However, if very large quantities of U_3O_8 could be obtained at approximately $100/lb, it is very likely that converter reactors would still compete favorably with future fossil-fueled power plants; at the same time, there is large uncertainty in the quantity of U_3O_8 which can be obtained at $100/lb or less. Without high performance breeder reactor use, converter and break-even breeder reactors have substantial U_3O_8 requirements under the reference nuclear growth rate. Only if the rate of power growth is substantially less than postulated, and this might indeed be the case, will the need date for economic breeders be delayed.

Taking a world-wide view, the conclusions reached above would still be valid. This is so because the planned fission power growth on a world basis relative to world U_3O_8 resources is such that the ratio of need to resources is at least as great as obtained for the USA analysis. Taking into consideration fission energy requirements for process heat needs in addition to those for electric energy production would increase the need for improved fuel utilization.

The most important component of power cost is that associated with capital costs. Present estimates[6,10] of capital costs for large LWR plants range from approximately $500–600/kW(e) for December 1981 startup, and approximately $650–800/kW(e) for December 1985 startup. Because of their large contribution to power costs, it is important to emphasize development of systems having low capital costs. The savings associated with a decrease in capital costs of $100/kW(e) for a nuclear capacity of 3×10^6 MW(e) in 2020 amounts to $300 billion (undiscounted). At 14%/yr capital charge rate, the above corresponds to a change in power costs of about 2 mills/kWhr(e). Based on a linear increase in power capacity in accordance with Eq. (1), the discounted savings are approximately $37 billion when discounted at 7.5%/yr from 1975. Reactor systems with significantly lower capital costs have a marked advantage for utility acceptance, since capital costs are the major component of power cost, and since raising capital funding is a difficult task. Yet, the relative capital costs of the various reactor systems are not known very well. Estimates of HTGR capital costs[11] indicate that total costs of large plants are essentially the same as those for LWR plants, although the distribution of those costs is different. However, experience to date has indicated that HTGR capital costs are higher than those for LWRs. How much of that difference is associated with developing a new reactor system is difficult to evaluate. At the same time, if a legal limit exists for the thermal capacity of a given reactor as is presently the case in the USA, HTGRs having higher electrical power capacity than those of LWRs can be built because of their higher thermal efficiency, and it is likely that under such circumstances HTGRs will have unit capital costs competitive with those of LWRs.

With regard to HWRs, the key factor is the capital cost. Estimates of HWR capital costs are generally higher than those of LWRs, and the economic acceptance of such systems in the USA depends upon a significant reduction in capital costs;[12] the estimates made at that time (1970) indicated that HWR reactors had capital costs about $100/kW(e) greater than did LWRs. Escalations since that time have doubled LWR capital costs, and if similar escalations apply to HWRs, the cost differential would be $200/kW(e) (including D_2O requirements). More recent estimates of HWR capital costs indicate a $180/kW(e) capital cost penalty relative to LWR costs;[13] at the same time, W. B. Lewis[14] projects HWR capital costs to be at least as low as LWR capital costs. Use of natural-uranium HWRs would, of course, not require separation of uranium isotopes, but would require production of D_2O. The capital requirements for D_2O separation plants are considerable, and although limited in a linear-growth economy, the discounted capital cost needs would be comparable with those for uranium enrichment plants. In summary, it appears that HWRs are the best converter reactors from the viewpoint of fuel utilization and ease of required separation processes; how they will contribute to fission power use will be determined by their capital costs (including the cost of D_2O).

Relative to separative work requirements, Fig. 2 gives values for a specific HTGR/LWR/LMFBR reactor industry, with introduction of a breeder reducing separative work (S.W.) needs significantly. Changing the specific mix of HTGRs and LWRs will change the S.W. requirements, but not to a significant extent. This is indicated in Table 4, which gives the S.W. requirements of HTGRs and LWRs based on estimated initial and annual makeup fuel requirements,[15] considering a power growth for each reactor as given by Eq. (1) and a 65% load factor.

Table 4. Estimated Separative Work Requirements for LWRs and HTGRs

		Annual S.W. Required,* 10^3 M.T. of S.W.		
Reactor Type	Year:	2000	2010	2020
HTGR		(110)	(184)	(258)
PWR		123(103)	224(183)	325(264)
BWR		115(95)	206(166)	297(237)

*Values in parenthesis are with recycle of bred fuel, while those not in parenthesis refer to no fuel recycle.

The results given in Table 4 indicate that while the overall S.W. needs of HTGRs are less than for LWRs when fuel is not recycled in LWRs, the magnitude of the requirement is still very large, and that HTGR use will not significantly reduce the need for breeders from the viewpoint of reducing S.W. requirements. When bred fuel is recycled in LWRs, the S.W. requirements of LWRs and HTGRs are essentially the same. The primary advantages of lower S.W. needs are the economic benefits including lower capital investment requirements, and the reduction in electric power use associated with decreasing S.W. needs. (There is some feedback between separative work needs and mined fuel needs, inasmuch as gaseous diffusion plants require significant amounts of electricity; however, that feedback is relatively small (∿ 5%). Further, if centrifuge separation becomes economic, that feedback would be reduced.) The cost benefits associated with reduced separative duty costs because of breeder introduction are significant, particularly on the basis of S.W. costs of $75/kg, which represents S.W. costs estimated to exist within the next decade.[6,16] However, there does not appear to be an uncertainty with regard to the technical ability of providing S.W. units if economically needed, in the same way there is uncertainty in the U_3O_8 resources available at different recovery costs.

3. Influence of Cost Considerations, Mined-Fuel Needs, and Separative Work Requirements on Gas-Cooled Reactor Importance

As indicated above, mined-fuel requirements become very high when extensive fission power is utilized and there is not widespread use of FBRs; under such circumstances and limited U_3O_8 resources at reasonable costs, fission power may not be competitive with alternate energy sources. Although HTGRs have better fuel utilization characteristics than LWRs, they are not sufficiently better to change the above conclusion, even if HTGRs were the only converter reactor utilized. Moreover, greater numbers of LWRs than HTGRs will be operating in the next two decades, and so the impact of HTGRs on overall fuel utilization cannot be great. In addition, the economic factors are such that if converter reactors are operated with higher conversion ratios than "reference" values, the overall fuel cycle cost will generally be adversely affected. As a result, while the better fuel utilization of HTGRs relative to LWRs is an advantage, its impact is primarily an economic one to be considered by utilities when choosing between LWRs and HTGRs, and not one which justifies HTGR development in itself. Further supporting that view are the S.W. requirements of HTGRs, which are not significantly different than those of LWRs.

An important factor influencing gas-cooled reactor use is their capital costs. As indicated above, it appears likely that HTGRs will have unit capital costs competitive with LWRs. Under such conditions, there is reasonable assurance that GCFRs also can have acceptable capital costs because of the similarity of major components utilized in the two reactor types. This is a very important point, since differences in capital costs between FBRs and LWRs can significantly delay the date when FBRs compete economically with LWRs. Thus, development of economic HTGRs gives assurance that GCFRs will be economic breeders, and in that way furthers breeder development while obtaining near-term benefits.

Introduction of the HTGR-GT and/or VHTR does not change the above to any great degree, because the fuel utilization performance of these reactors is basically no better than that of the HTGR. Further, their development schedule does not permit them to be built immediately. The primary influence on fuel utilization would be through development of reactor systems having higher thermal efficiency, which requires essentially the use of combined cycles (e.g., HTGR-GT with a bottoming cycle). While the potential importance of the HTGR-GT and VHTR appear to be very great, particularly the application of the VHTR to hydrogen production processes, widespread future use of these systems will be strongly dependent upon the success of breeder reactor development. More discussion concerning these systems is given below in Sections 4.3 and 4.4.

4. Gas-Cooled Reactor Systems and Their Significance

A discussion is given below of HTGRs, GCFRs, HTGR-GT, and VHTRs, considering application, status, economic features, and overall significance and evaluation (including factors discussed in previous sections). Certain parameters, however, are discussed primarily under one reactor even though they can be applied to others; thus, comparisons of thorium and uranium fuel cycles and of prismatic and pebble-bed fuel designs are done mainly in the section on HTGRs.

4.1 HTGR

The term HTGR applies to various fueled-graphite reactor designs, including the various prismatic and pebble-bed types.

4.1.1 Application. The HTGR makes use of the steam cycle and its primary application is the generation of electric power. Because of its relatively good fuel conservation characteristics in comparison with LWRs and its high thermal efficiency, the HTGR shows promise of having widespread use in base-loaded power plants.

4.1.2 Status of Development. The HTGR concept has had three highly successful experimental reactor demonstrations:[17] the Dragon Reactor, the AVR Reactor, and the Peach Bottom HTGR. Further, power operation of the Fort St. Vrain HTGR is planned the latter part of this year, and the THTR is scheduled for power operation in 1977. A strong research and development program is being carried out in HTGR technology including areas such as fuels and materials, fuel recycle, chemistry and fission product behavior, graphite, PCRVs, reactor surveillance, and components testing.[17] Primary USA parties engaged in this activity are General Atomic Company, Oak Ridge National Laboratory, and Idaho National Engineering Laboratory. The United Kingdom has also been carrying out a strong program on HTGR technology and design development, and Germany has a strong program, although development emphasis has been on the HTGR-GT. France is also becoming significantly involved in

HTGR technology development; further, a number of nations participate through the OECD Dragon Project. Development has proceeded to the point where HTGRs are being offered commercially to utilities as power producers, with research and development continuing in various areas. The HTGR fuel recycle area is one of concern, inasmuch as the technology has not yet been completely developed; the largest effort is being carried out in the USA where plans are being made for demonstrating fuel recycle technology in a demonstration plant. Germany is also carrying out significant work in fuel recycle technology, and plans a more extensive program in the future.

4.1.3 Economic Features. As discussed above in Section 2, the unit-capital costs of HTGRs appear to have the potential of being essentially the same as those for LWRs. Further, HTGR fuel cycle costs are potentially lower than those for LWRs. However, HTGR fuel cycle costs using the reference thorium cycle are very dependent upon the cost of recycling fuel and these costs are not known with confidence. At the same time, the costs of natural uranium ore and of separative work have increased substantially in the past few years, and estimated costs in the 1980's are approximately \$30/lb for U_3O_8 and \$75/kg S.W., which tend to improve the position of HTGRs relative to LWRs. Taking into consideration all these factors, it appears that the cost of power from an HTGR should be at least 0.5 mill/kWhr(e) lower than that from LWRs; however, the uncertainty associated with the above is estimated to be 1 mill/kWhr(e), since potential benefits do not always materialize. The above situation undoubtedly contributes to the present reluctance to "go HTGR," although other factors such as the present high cost of money, the high capital costs for all nuclear power plants, and the cost of "breaking into" a more established industry also have a strong influence. Overall, the long-term economic features of HTGRs look good, but the near-term economics are not as favorable.

4.1.3.1 Relative performance of thorium and uranium fuel cycles. Work in the USA has been devoted toward development of the thorium fuel cycle in HTGRs, and most of the emphasis in other countries is also on that cycle at this time. However, especially in Europe, there has been and still is a continuing interest in the uranium cycle.

Interest in the thorium fuel cycle stems primarily because that cycle provides improved fuel utilization over the uranium cycle; at the same time, use of the thorium cycle places importance on developing an economic fuel recycle technology. On the other hand, interest in the uranium cycle stems from the ability of utilizing low-enriched uranium as the initial fuel (rather than the highly-enriched uranium required in the thorium cycle), and to the reduced need for recovering fissile-fuel from the irradiated elements. If irradiated fuel elements from HTGRs could indeed be disposed of inexpensively in the form they leave the reactor (throw-away cycle), use of the uranium cycle in HTGRs would appear economically attractive for some time. However, a socially-acceptable, long-term disposal method has not been established to date. If separation of plutonium from the fission products is necessary, many of the same problems associated with fuel recycle would have to be faced. For either cycle, fuel recycle would improve the fuel utilization of HTGRs, with a more beneficial effect upon the thorium cycle.

A large number of studies have been conducted comparing (or relating to) the uranium and thorium fuel cycles in HTGRs.[18-31] Many of these studies compare the two cycles in the same reactor design, or only consider a given cycle, while others are based on different designs having differing degrees of optimization of the two cycles; thus, it is difficult to compare results directly. Further, comparison of results is difficult because different economic bases were often used which influenced design values and performance. Nonetheless, these studies generally show that under the design and economic conditions employed, use of the thorium fuel cycle gave fuel cycle costs which were 0.1–0.3 mill/kWhr lower than those for the uranium cycle, on the basis that economic fuel recycle plants were available. These studies considered natural-uranium ore costs and unit separative work costs which are lower than those expected in the next decade; using projected economic conditions would generally aid the thorium cycle relative to the uranium cycle.

In the uranium cycle, it is advantageous to "lump" the fuel so as to decrease the needed uranium enrichment and thus lower fuel inventory costs. In the thorium cycle this is not needed since the fissile particle is always highly-enriched uranium. As a result, a core design well suited for the thorium cycle may not necessarily be well suited to the uranium cycle. Nonetheless, changeover from one cycle to the other in a given reactor may be desirable at some time; in a study of this, Gutmann et al[29] compared the two cycles in a Fort St. Vrain Reactor design, utilizing a large fuel kernel in the uranium cycle to provide some "lumping." The results indicated that for the economic conditions employed and assuming an economic fuel recycle technology, the thorium cycle had fuel cycle costs about 0.2 mill/kWhr less than the uranium cycle. This study also indicated that a changeover from the uranium to the thorium cycle (or vice versa) could be readily accomplished if desired.

More recently, Teuchert et al[31] have made an interesting study of the uranium and thorium fuel cycles in pebble-bed reactors. In these studies, the thorium-based fuel was essentially dispersed throughout the fuel sphere; the uranium-based fuel, however, was arranged in a "shell" within the sphere and utilized relatively large fuel kernels, thus achieving significant fuel "lumping." The results indicated that the thorium fuel cycle cost was about 0.2–0.3 mill/kWhr(e) less than for the uranium cycle. At the same time, no significant penalty was associated with fuel refabrication costs relative to fresh fuel fabrication costs; imposing a reasonable penalty would lead to about the same fuel cycle costs for the two cycles. However, increasing the natural-uranium ore costs and S.W. costs to values projected in the future would favor use of the thorium cycle. With regard to changing from one fuel cycle to another in this reactor design, Teuchert et al also found such a changeover could be readily accomplished.

Overall, it appears that the thorium fuel cycle would be economically preferred if a fuel recycle technology is available; if direct disposal of irradiated fuel is acceptable, the uranium cycle would be preferred.

Another factor to be considered in the choice of fuel cycle is the irradiation performance of fuel. Thoria is less subject than urania to fuel migration under high-temperature and high-temperature-gradient conditions. Further, for pure fissile particles, the uranium dicarbide or mixed dicarbide/dioxide has better irradiation stability than urania. As a result, use of "feed-breed" type fuel in the thorium cycle appears to provide higher temperature and temperature-gradient capabilities than use of urania in the uranium cycle.

The difference in economic performance between the thorium and uranium fuel cycles is primarily due to the different fuel utilization characteristics of the two cycles. The thorium cycle generally has a fuel conversion ratio about 0.05 to 0.08* higher than the uranium cycle (with recycle in both cases), leading to net fuel feed requirements about 10–20% less for the thorium cycle. The fissile inventory needs of the two cycles vary significantly with the fuel exposure requirements (which is influenced markedly by fuel fabrication and recycle costs); it appears that practical exposure requirements lead to fissile inventory needs for the uranium cycle which are about 80–85%* those for the thorium cycle. Thus, fuel utilization differences between the two cycles favor the thorium cycle since the fuel-makeup needs dominate the 30-year fueling requirements of a given reactor. However, in an expanding economy, the inventory needs have much more of an impact; if the conditions implicit in Table 3 are considered, use of the uranium rather than the thorium cycle in "HTGR-1" would increase the total mined U_3O_8 requirements by about 5% through the year 2020.

4.1.3.2 Comparison of prismatic and pebble-bed fuel designs. The primary "prismatic-type" fuel design is that developed by General Atomic Company, while the primary "pebble-bed" fuel design is that developed by Germany, and these are the designs considered here; both involve fueled-graphite systems. The prismatic design consists of a graphite block containing coolant holes and holes filled with fuel rods, with fabrication of the fuel rods separate from that of the moderator block. The pebble-bed design consists of mixed

*Specific values can lie outside this band depending on the economic conditions and cost parameters employed.

fuel and moderator material, with an outer layer of fuel-free graphite. The prismatic design provides more positive control of coolant flow in the various regions of the reactor under design conditions; the pebble-bed design maintains reasonable coolant flow distribution even with severe graphite shrinkage or expansion, but tends to have a higher core pressure drop. The prismatic design maintains positive control over fuel location, while the pebble-bed design provides ease of fuel movement and on-line refueling. The prismatic design provides convenient and effective space for control rods, whereas the pebble-bed design has more difficulty in providing rapid control rod movement involving large shutdown margins.

Overall, it appears that the primary advantages of the prismatic design are associated with its fabrication of fuel rods independent of the graphite moderator fabrication, along with its provisions for reactivity control. The former circumstances tend to give relatively low fuel fabrication and refabrication costs. The primary advantages of the pebble-bed design are its on-line fueling capability (which leads to reduced neutron losses, uniform fuel burnup, relatively low fuel temperatures, and less power peaking due to control rod movement), and its use of dispersed fuel in the graphite moderator such that temperature gradients in the fuel sphere are relatively small (an advantage since fuel failures are adversely influenced by high temperaturees and high temperature gradients). The above indicates that it is more important to use the thorium fuel cycle in the GAC design than in the pebble-bed design (see Section 4.1.3.1). (However, it should be noted that the dispersed fuel concept is not limited to pebble-bed designs but could also be utilized in prismatic designs.) The use of pebbles would also be advantageous with regard to fuel head-end reprocessing, since a large fuel block crusher would not be needed with pebble-bed fuel; further, segregation of fuels in different spheres is possible.

The above features indicate that the pebble-bed fuel is basically better suited for very high temperature applications than is the GAC prismatic design. At the same time, the prismatic design appears adequate for HTGR applications, and it may be better suited for recycling of bred fuel if its fuel refabrication costs are relatively low.

4.1.4 HTGR Significance and Evaluation. The HTGR is the most developed of the Gas-Cooled Reactors (GCRs) considered here, and the only one of them that might be introduced commercially on a large scale within the next 15 years. It also constitutes a versatile energy source inasmuch as its component technology is closely related to the other similar, but less developed, GCR concepts having complementary characteristics. Thus, successful penetration of the

HTGR into the commercial power market in a major way catalyzes efficient, cost effective, and timely development of GCRs. The above features are believed to be the most significant ones relative to the long-term importance of HTGRs. In addition, on a shorter time scale, the HTGR is important because it provides a high thermal efficiency system, gives improved fuel utilization relative to LWR use, gives diversification to the nuclear industry, and appears to have advantageous safety characteristics. At the same time, penetration of the HTGR into the utility market will be dependent upon successful operation of the Fort St. Vrain Reactor, assurance that fuel recycle technology will be demonstrated and have acceptable costs, and upon bringing unit capital costs down to a level competitive with LWRs. There appears to be a reasonable probability that the above conditions will be met. Once established, HTGRs would provide utilities with more diversity of choice in fission-power plants and their siting; further, there would be increased assurance that GCFRs would be economic breeders.

4.2 GCFR

4.2.1 Application. The primary application of the GCFR will be as an energy source for electric power production, while providing excess fissile fuel and associated separative work capacity to the nuclear power industry.

4.2.2 Development Status. There has been a limited development and design effort on the present GCFR concept for about the past 13 years, with most work being carried out in the USA and several European countries. Nearly all of the recent work in the USA has been supported by the government, utilities, and GAC, and centered on the GAC reactor concept, with GAC as the lead contractor. Elements of the present program[32] include fuels and materials development and testing, heat transfer and fluid flow studies, plant design and associated parameter studies, core design and engineering, reactor physics and criticality studies, fission product behavior studies, shielding investigations, component development and design, PCRV closure development, plant cost estimates, and safety studies. The present work primarily involves selected technology development, and at the present financial support level will not bring GCFR technology to the level required for a demonstration plant on the time schedule envisioned by GAC (demonstration plant[33] in the mid-1980's). At this time, there is little indication that the GCFR program support will be increased to the required level in the immediate future. However, the GCFR benefits significantly from the LMFBR fuel development effort, and the present program considers oxide fuel pellets as developed and tested in the LMFBR Program to be the reference fuel. Further, the component technology development under the HTGR Program applies generally to the GCFR.

Other GCFR work is being carried out primarily by Germany[34] (which is constructing and testing an in-pile loop for irradiation testing of a vented-fuel-rod bundle in the BR-2 reactor), by Switzerland (which is carrying out a detailed heat transfer and fluid flow test program for GCFR fuel bundles), and by the Gas Breeder Reactor Associates[35] (which carries out design and safety studies on GCFR plants and components, safety studies, and limited development work).

Results of development work to date have indicated that the vented-fuel-rod concept will perform successfully, and that the reference oxide fuel will perform satisfactorily at GCFR temperatures and significant fast fluence. Core design characteristics, heat transfer and fluid flow relations, reactor physics characteristics, and fuels and materials behavior for the GCFR are such that a high performance breeder appears practical with oxide-based fuel; however, integrated experimental studies are needed to verify the above. Component development and testing is limited at this time, and much verification of component design and behavior is needed, even though it builds on HTGR experience. Present analyses of reactor behavior under loss-of-cooling conditions indicate that abnormal conditions can be controlled satisfactorily; however, extensive experimental testing is required to verify that the analyses and conditions considered are adequate. Overall, however, GCFR development has progressed significantly on a very limited budget.

4.3.3 Economic Features. As with all nuclear power plants, the economic performance of the GCFR is primarily dependent upon capital and fuel cycle costs. As given in Sections 1 and 2, introduction of an economic high-performance breeder has a major impact upon mined-fuel and separative work requirements, reducing such needs markedly. Since these results are based on carrying out fuel recycle, and since the fuel recycle needs of the GCFR and LMFBR are virtually the same, both FBRs have the potential of having low fuel cycle costs when economic fuel recycle takes place.

As illustrated in Section 2, relative capital costs of nuclear plants can have a significant impact on plant application. Since component costs are an important part of capital costs, and GCFR and HTGR components have many similarities, the GCFR capital cost should be reasonably close to that of HTGRs.[35] Under such circumstances, GCFR capital costs should be competitive if HTGR capital costs are competitive. On the above bases (including economic fuel recycle), the GCFR should compete economically with converter reactors when ore prices are still relatively low.

4.2.4 GCFR Significance and Evaluation. The GCFR has the potential of being a high performance breeder which is also an economic power producer. Because of the importance of breeder development, it might be argued that the GCFR should be developed at the expense of the HTGR. However, there are many potential difficulties with such an approach, and it is doubtful that the time schedule for GCFR development under such circumstances would be much different than with it following the HTGR. Successful development of the HTGR is not yet assured, and if the HTGR is not successful, the discussion in Section 4.2.3 implies that the GCFR may find it difficult to compete economically. (There are, of course, circumstances where the above is not valid.) Further, it appears to be an efficient use of resources for the GCFR to build on HTGR technology.

Building on HTGR component and LMFBR fuel and fuel-recycle technology as it does, the date of GCFR introduction may be soon enough to contribute significantly to fueling and separative work needs. If USA nuclear power growth follows the lower estimate of growth as given in Ref. 9 [800 GW(e) in 2000], introduction of commercial GCFRs in 2000 could still limit mined U_3O_8 requirements to estimated potential USA resources at reasonable costs (3.4×10^6 tons).

During the first five or more years after FBR introduction, fuel utilization factors require that the FBR expand as rapidly as available fuel permits; once the FBR system growth leads to producing excess fissile fuel, this excess could profitably be ^{233}U rather than plutonium by utilizing thorium fuel in the blanket, with the product ^{233}U employed in HTGRs.[36-38] This would be beneficial to both systems once the need for FBR fuel is more than satisfied; prior to that time, producing ^{233}U in the blanket of the FBR would lead to increased mined fuel requirements.

Overall, the major advantages of the GCFR are its ability to utilize HTGR component technology in general and its high breeding performance using oxide-based fuel. Because of cladding temperature limitations, the GCFR will have a lower thermal efficiency than the HTGR. Major development areas include specific large-scale component development, fuel and reactivity behavior under loss-of-cooling conditions, and development of fuel recycle technology.

4.3 HTGR-GT

4.3.1 Application. The HTGR-GT power plant combines the basic HTGR with a closed cycle helium gas turbine power conversion system; it appears well-suited for sites where cooling water is very limited and dry cooling towers are required. In addition, for sites where cooling water is available, a "bottoming cycle" can be provided so as to generate additional power from the heat rejected from the helium turbine cycle, thereby achieving very high overall cycle efficiency. In both cases, electric power production is the primary product. In special cases, the energy rejected from the helium-turbine cycle can be utilized effectively in low-temperature process-heat applications such as desalination of brines or seawater, or in space heating.

4.3.2 Development Status. The primary HTGR-GT development work is sponsored by the USA and Germany; the USA work emphasizes the HTGR-GT design as developed by GAC,[39,40] safety evaluations,[41] and involves a limited technology development program at this time.[42,43] A larger overall program is presently being carried out in Germany, involving a number of industrial and government research organizations; development areas include plant layout, component development, fission product behavior studies, basic research work, fuel development, materials development, and testing of gas turbine components in both development and pilot-plant demonstration facilities.[44] The USA program includes plant design parameter studies, analysis of plant performance and control requirements, study of plant response characteristics, evaluation of plant safety, planning for large-scale component testing facilities, review of materials and turbomachinery technology, materials testing, evaluation of fission product behavior in the reactor circuit, and specific component and plant systems design studies. Studies are also being performed relative to use of a "bottom cycle" using a compound such as isobutane as the secondary working fluid; addition of a bottom cycle could increase the overall cycle efficiency to about 48% when water cooling is available, in a cost effective manner.[45]

The HTGR-GT builds on HTGR and gas-turbine technology, and thus is in a relatively-advanced state of development. However, it is dependent upon successful economic development of the HTGR, particularly with regard to having acceptable capital costs; also, extensive testing of large-scale components is required, particularly with regard to the helium turbine, and gas circulators and compressors. Further, present emphasis is on a core outlet coolant temperature of about 815°C, which is relatively low with regard to efficient use of gas turbines. A higher temperature is more desirable, but in order to employ significantly higher values, extensive materials development work remains to be done. With regard to the use of a bottom cycle with the HTGR-GT, extensive experimental studies of the bottom cycle itself are needed, including material behavior, operating conditions, and associated component development.

4.3.3 Economic Features. The HTGR-GT is based largely on HTGR component technology, and should have similar capital costs in the applicable areas. The fuel cycle costs should be about the same as for the HTGR, except for changes in the core so as to increase outlet gas coolant temperature for a given fuel temperature (these latter changes would tend to increase HTGR-GT costs slightly but not significantly). Since unit capital costs for HTGRs today appear to be higher than those for LWRs, the HTGR-GT will find it difficult to compete until HTGR capital costs are competitive.

Because heat is rejected over a wide temperature range in the Brayton cycle, the HTGR-GT is well suited for sites requiring dry cooling towers. Cost estimates have been made comparing the HTGR and HTGR-GT at such a site; both systems had about the same thermal efficiency but the capital costs of the HTGR-GT were estimated to be about 15% less than those for the HTGR.[39] At the same time, a large uncertainty is present in such cost estimates until much more large-scale component development and testing has taken place relative to helium-turbine systems as employed within HTGR-GT systems.

The HTGR-GT concept which appears to have the best chance of competing economically is the combined-cycle system utilizing a bottom (condensing) cycle along with the Brayton cycle. The addition of the bottom cycle should be very cost effective, and the resulting very high thermal efficiency system has advantageous environmental and fuel resource implications, and significantly raises the electric power output of a plant having a legal thermal limit.

4.3.4 HTGR-GT Significance and Evaluation. The HTGR-GT builds on HTGR technology and permits development of a power generation system well suited for sites requiring dry cooling towers, as well as for dual-purpose power/low-temperature process-heat applications. Its primary importance, however, is believed to be its ability to utilize combined cycles resulting in electric power plants having very high thermal efficiencies. The combination of high power output per plant and very high thermal efficiency appears to have the highest probability of providing an economic system and one with improved fuel utilization. At the same time, the HTGR-GT is largely dependent upon successful introduction of the HTGR, and even with the use of combined cycles it does not significantly change the need for developing a successful FBR. The HTGR-GT is a logical and useful extension of the HTGR, and would provide fuels and materials technology useful to the VHTR if the core-outlet coolant temperature of the HTGR-GT were increased.

4.4 VHTR

4.4.1 Application. The industrial sector is the largest industry user in the United States, accounting for about 40% of the total primary energy consumption. Natural gas and petroleum are the primary fuels currently used by industry. Of the direct process (nonelectric) uses, 51% use natural gas, 27% use oil, and 22% use coal. Due to scarcity, price increases, and long-term supply problems of natural gas and petroleum, USA industry will rely in the future more and more on the most abundant domestic fuel resources, namely, coal and nuclear. From a national energy viewpoint, the use of coal and nuclear fuel in industry would release gas and oil for other uses and would make an important step toward national self-sufficiency in energy.

The VHTR is a gas-cooled graphite-moderated reactor operating at temperature levels higher than those required for the steam-generating reactor. There are a number of large industrial process heat applications that could utilize the VHTR. These include coal conversion to synthetic gas or liquid fuels, hydrogen production by thermochemical water splitting, and hydrogen or synthesis gas (H_2 + CO) production for use in direct reduction of iron ore, refining of petroleum, and refining of liquids derived from oil shale and tar sands.

In concept, the VHTR could also produce electricity at high efficiencies by using an advanced topping cycle, or could be used as a chemical energy storage system for load following and meeting peak electrical loads. Chemical energy from a VHTR could be piped long distances to dispersed locations where the energy would be released into local power or heat producing facilities.

4.4.2 Development Status. In April 1974, the U.S. Atomic Energy Commission (now U.S. Energy Research and Development Administration [ERDA]) authorized General Atomic Company (GA), General Electric Company (GE) and Westinghouse Astronuclear Laboratory (W) to assess the available technology for producing process heat utilizing very high temperature nuclear reactors.[46-48] The GA fuel design was similar to that of the HTGR design; the GE fuel design utilized the pebble-bed concept, and was based on a joint study by KFA-Jülich and GE; the W fuel design utilized dispersed fuel in an extruded, prismatic design based on nuclear-rocket fuel technology. Table 5 provides a summary of the features of the three designs.

The concepts and technology were evaluated for producing process stream temperatures of 649°C, 760°C, 871°C, 982°C, and 1093°C. The current technology limits to the process temperature that can be obtained from VHTR systems are associated with materials and safety. The basic limitation is the temperature at which the fuel and structural materials can operate; the temperature required of the fuel and materials depends on the process temperature desired, whether an intermediate heat exchange (IHX) is required between the reactor coolant (helium) and the process heat exchanger (PHX), and to some extent the concept selected.

Three levels of technology can be identified as a function of process temperature. First, process temperatures in the range of 538 to 649°C can be achieved with current technology within perhaps 10 to 12 years. Process temperatures from 649 to 871°C represent near-term technology, i.e., commercial application, allowing 7 to 10 years for R&D, could be achieved within 15 to 18 years. Finally, process application from 871 to 1093°C appears to represent long-term technology which would require in the order of 15 to 25 years of R&D and perhaps 25 to 35 years for commercialization.

Table 5. Comparison of Concepts for 1600°F Process Temperatures

	General Atomic	General Electric	Westinghouse
Basis of very high temperature nuclear reactor design	Modification of HTGR concept	Pebble bed concept based on German technology	Prismatic fuel concept based on nuclear rocket technology
Reactor core type	Hexagonal graphite blocks, solid cylindrical fuel rods	Pebble bed core Graphite sphere fuel element	Hexagonal graphite blocks, hollow cylindrical fuel rods with central coolant channel
Intermediate heat exchanger	No	Yes	Yes
Reactor coolant	Helium	Helium	Helium
Reference thermal power	3000 MW(t)	3000 MW(t)	3000 MW(t)
Pressure vessel concept	Prestressed concrete reactor vessel	Prestressed concrete reactor vessel	Prestressed cast iron reactor vessel
Fuel composition	• Fully enriched U feed (UC_2) with thorium (ThO_2) fertile material • No recycle of ^{233}U • Triso coating for both fissile and fertile particles • Carbon to thorium ratio = 200	• Low enriched (9.01%) ^{235}U fuel • No recycle of bred plutonium • Triso coated UO_2 fuel particles • Graphite sphere fuel element • Carbon to heavy metal ratio = 350	• Fully enriched U feed (UC_2) with thorium (ThO_2) material • Recycle of ^{233}U • Triso coated fissile particle • Biso coated fertile particle • Carbon to thorium ratio = 206
Average fuel residence time	3 years	3.8 years	4 years
Power density	8.4 watts/cm^3	5 watts/cm^3	10 watts/cm^3
System pressure	49 atm	41 atm	68 atm
Core inlet temperature	500°C	250°C	430°C
Core outlet temperature	982°C	950°C	1010°C
Maximum fuel temperature	1406°C	1110°C	1181°C

Studies are currently under way by ERDA contractors, NASA and the American Iron and Steel Institute relative to various process applications and to design of the VHTR. Application to methane reforming or other hydrogen-producing processes appears to be very important. Use of the VHTR for the above rather than coal could reduce coal use by 25-30%, and have environmental and economic benefits relative to fissil fuel use. Oak Ridge National Laboratory is evaluating these various applications and an evaluation report is due in early 1976.

In addition to the limited USA VHTR effort, relatively strong programs are being carried out in Germany[44] and Japan[49] including process development, design studies, fuels and materials technology development, and safety studies. Extensive and strongly supported programs in these areas will be required before the potential of the VHTR can be realized.

4.4.3 VHTR Economic Features. The economic performance of the VHTR is based on the referenced studies as evaluated by ORNL. Figure 3 gives the range of estimated costs of nuclear process heat when supplied from a 3000 MW(t) VHTR, considering the presence or absence of an intermediate heat exchanger as a parameter. The economic ground rules used in obtaining these estimates are given in Table 6. The range of cost is comparable to the estimated cost of process heat derived from fossil fuels.

Table 6. Economic Ground Rules Employed in VHTR Evaluation

Reference Plant Size, MW(t) - 3000		
Process Heat Cost is evaluated assuming all energy from the reactor has the same value independent of the form of the energy or how it is used.		
July 1974 Dollars - No Escalation		
80% Plant Factor		
25% Fixed Charge Rate		
Capital Costs Include:	Direct Costs Indirect Costs Interest During Construction - 5%/yr	
Fuel Cycle Cost Basis:	U_3O_8, \$/lb	30
	Enrichment, \$/SWU	75
O&M Costs - 9×10^6 \$/yr.		

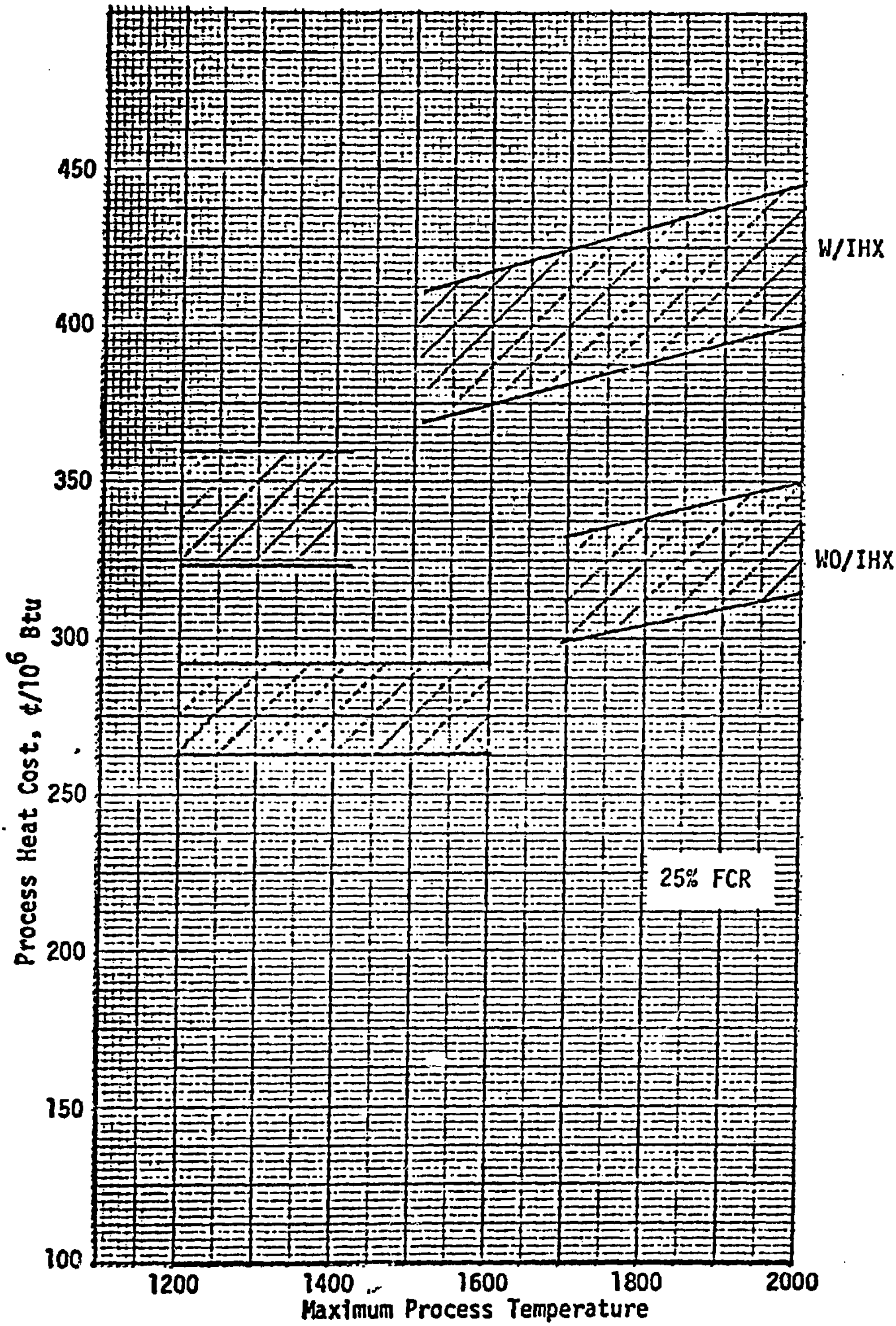

Fig. 3. Process Heat Cost - VHTR

4.4.4 VHTR Significance and Evaluation. It appears that there is a major market potential for the VHTR in both synthetic fuels and in associated electric power production. It is the only type of fission reactor which appears to have the potential for reaching the desired process temperature range of 750°C to 1100°C. Further, process temperatures in the 760 to 870°C range are achievable with near-term technology. The major development considerations are high temperature materials, the safety questions (especially regarding the need for an IHX) and the process heat exchanger. The VHTR temperature capability is expected to be particularly useful to the synthetic fuels industry.

The major advantages of the VHTR over competing fossil energy sources are conservation of fossil fuels and reduced atmospheric impacts. It is expected that at future fossil-fuel prices the VHTR will also be economic as well; however, that will be dependent on the price of fissile fuel. Without breeder reactor introduction and limited economic uranium resources as presently estimated, the VHTR will probably find it difficult to compete with fossil fuels economically. Thus, the VHTR appears dependent upon the successful development of breeder reactors, particularly since the VHTR will probably take at least two decades to resolve the difficult materials problems and to develop a commercial reactor/process system. By that time the fuel resource situation will be much better resolved. Recognizing the difficulties of high-temperature-materials technology development and the future importance of high-temperature process heat, significant efforts should be carried out now and emphasize development of materials and processes; at the same time, recognition needs to be taken that the VHTR will not conserve fuel any better than the HTGR, and that future widespread application of the VHTR will require successful breeder development, or discovery of additional U_3O_8 resources having reasonable recovery costs.

5. Summary Evaluation of the Interaction and Development of Gas-Cooled Reactor Systems

The HTGR is presently being offered commercially to utilities, and is in a much more advanced state of development than the other GCRs considered here. At the same time, a high-performance, economic breeder appears needed as soon as possible, and the GCFR is a likely candidate. Further, the HTGR-GT is a logical near-term extension of the HTGR, and when developed and combined with a bottom power cycle has higher fuel utilization characteristics than the HTGR and probably lower costs for power production. The VHTR in turn is a logical extension of the HTGR-GT, as the core outlet coolant temperature is increased to improve the thermal efficiency of the helium turbine cycle, with important potential benefits regarding long-term fossil-fuel conservation; however, its timing is such that widespread VHTR use appears dependent upon successful breeder introduction. Under the above circumstances, a number of scenarios for logical GCR development can be envisioned, dependent upon the specific assumptions made. For example, if the need for an economic FBR (as illustrated in Section 1) is filled by the LMFBR, the importance of GCFR development is greatly reduced. GCRs would then be logically developed in the order HTGR, HTGR-GT, (GCFR), VHTR. In a related scenario, it could be assumed from the start that the LMFBR will be introduced successfully and in a timely manner, such that importance is placed early on developing a high-temperature energy source for use in fossil-fuel conservation; GCR development would then logically concentrate on the VHTR. Another scenario could consider FBR introduction into the economy at a much delayed date (e.g., 2010); under such circumstances, HTGR-GT (using combined cycles) development would be emphasized because of the important beneficial impacts this concept could have on mined-fuel and separative-work requirements as well as on power costs. In still another scene, it could be assumed that LMFBR development did not lead to an economic reactor system; GCR development would then concentrate on the GCFR.

A number of other scenarios could also be envisioned, but the above sufficiently illustrate that quite different GCR development programs can logically be drawn up, dependent upon the assumptions made concerning future events and conditions. Further, it is clear that the above scenarios generally imply much simpler conditions and choices than what have to be faced in drawing up practical reactor development programs, and that a reasonable program has to be flexible enough to adapt to changing conditions with time. Thus, there is little point in trying to cover all possible scenarios. Instead, a short discussion is given below concerning what is believed to be a prudent course of action for GCR development based on present-status information and expected conditions, with allowance for contingencies (additional information pertaining to the rationale employed is given previously in Section 4). In doing this, it is recognized that much subjective judgment is associated with the rationale and the implied program emphasis. Because of this and because implicit conditions might be different, other knowledgeable persons can arrive at conclusions and schedules different than those presented.

Highest priority at this time should be given to HTGR technology development and demonstration since the HTGR is relatively far advanced and benefits can be obtained reasonably quickly; further, that course of action benefits all the GCRs. Once the HTGR is commercially accepted, first priority should be shifted to GCFR development so as to increase significantly the probability that an economic FBR will become a reality. Second priority should be given to the HTGR-GT utilizing combined cycles; this provides some assurance of improved fuel utilization and economics in case the FBR is delayed, and also contributes to VHTR technology development as efficiency of the helium turbine cycle is improved through use of higher core outlet coolant temperatures. The VHTR program should be restricted at first to a technology development program with emphasis on development of high temperature materials and processes for hydrogen production (this latter work of course is not restricted to a VHTR program). As the probability of FBR commercial success increases, along with successful development of the HTGR-GT (combined cycles), a VHTR component technology development program should be emphasized. Based on commercial acceptance of HTGRs by 1980 (as evidenced by reactor sales and commitments), GCFRs could reasonably be built commercially by about 2000, and VHTRs by about 2010. The timing of the HTGR-GT (combined cycles) is dependent upon the emphasis given that concept, which is related to funding available for all reactor development. The GCFR should have priority over the HTGR-GT (combined cycles), but the latter concept should not be neglected since very high thermal efficiency systems are needed. Because of its close relation to the HTGR, the HTGR-GT (combined cycles) might become commercial before the GCFR even on the basis of the above priorities, with 1995 a reasonable date. Development of the HTGR-GT (helium turbine cycle) concept is of lesser importance, but it occurs naturally along with that of the HTGR-GT (combined cycles); use of the former concept is dependent upon associated economics.

REFERENCES

1. "Potential Nuclear Power Growth Patterns," USAEC Rep. WASH-1098, (Dec. 1970).

2. "Cost-Benefit Analysis of the U.S. Breeder Reactor Program," USAEC Rep. WASH-1126 (Apr. 1969).

3. "Updated (1970) Cost-Benefit Analysis of the U.S. Breeder Reactor Program," USAEC Rep. WASH-1184 (Jan. 1972).

4. "Environmental Statement: Liquid Metal Fast Breeder Reactor Program. Vol. III: Implication of LMFBR Program Implementation," USAEC Rep. WASH-1535 (Mar. 1974).

5. STRAUCH, S., FLETCHER, J., and BENNETT, L., "An Examination of LMFBR Power and Resource Cost Projects," paper presented at ANS Topical Meeting on Advanced Reactors: Physics, Design and Economics, Atlanta, Georgia (Sept. 8-11, 1974).

6. DAVIS, W. K., "Economics of Nuclear Power," paper presented at the International Symposium on Nuclear Power Technology and Economics, Taipei, Taiwan (Jan. 13-20, 1975).

7. KASTEN, P. R. and BENNETT, L. L., "Economic Interactions Between Fueling, Separative Work and Capital Requirements for Fission Power Reactors," paper presented at the International Symposium on Nuclear Power Technology and Economics, Taipei, Taiwan (Jan. 13-20, 1975).

8. "Report of the Liquid Metal Fast Breeder Reactor Program Review Group," USERDA Rep. ERDA-1 (Jan. 1975).

9. "Nuclear Fuel Cycle," report of the Fuel Cycle Task Force, USERDA Rep. ERDA-33 (March 1975).

10. "Power Plant Capital Costs - Current Trends and Sensitivity to Economic Parameters," USAEC Rep. WASH-1345 (Oct. 1974).

11. UNITED ENGINEERS AND CONSTRUCTORS, INC., "High Temperature Gas-Cooled Reactor Plant - 770 MW(e) Central Station Power Plants Investment Cost Study," USAEC Rep. WASH-1230 (Vol. V) (Dec. 1973).

12. "Report of the EEI Reactor Assessment Panel," Edison Electric Institute, New York N. Y. (1970).

13. WEINBERG, A. M., "The Maturity of Nuclear Energy: An Overview," paper presented at the International Symposium on Nuclear Power Technology and Economics, Taipei, Taiwan (Jan. 13-20, 1975).

14. LEWIS, W. B., "Heavy Water Power Reactors," paper presented at the International Symposium on Nuclear Power Technology and Economics, Taipei, Taiwan (Jan. 13-20, 1975).

15. "Nuclear Energy Growth 1974-2000," USAEC Rep. WASH-1139 (1974), (Feb. 1974).

16. BENEDICT, M., "Fuel Cycles for Nuclear Reactors: Uranium Enrichment and Reprocessing," paper presented at the International Symposium on Nuclear Power Technology and Economics, Taipei, Taiwan (Jan. 13-20, 1975).

17. ANS Topical Meeting on Gas-Cooled Reactors: HTGR and GCFBR, May 7-10, 1974, Gatlinburg, Tenn., CONF-740501 (May 1974).

18. KASTEN, P. R., "The Role of Thorium Power Reactor Development," Atomic Energy Review VIII (3), 473-534, IAEA, Vienna (1970).

19. LANE, R. K., GEORGE, C.H., and DAHLBERG, R. C., "Comparative Fuel Utilization in the HTGR and PWR," Gulf-GA Report A12592 (Apr. 15, 1973).

20. OEHME, H., "Comparative HTGR Designs," ANS Topical Meeting on Gas-Cooled Reactors: HTGR and GCFBR, May 7-10, 1974, Gatlinburg, Tenn., p. 72, CONF-740501 (May 1974).

21. CARLSMITH, R.S., PODEWELTZ, C.M., and THOMAS, W. E., "Fuel Cycle Cost Comparisons for High Temperature Gas-Cooled Reactor Fuels," ORNL Rep. ORNL-TM-1112 (Apr. 14, 1965).

22. THOMAS, W.E., "Blanket Study for a 1000-MW(e) HTGR," ORNL Rep. ORNL-TM-2477 (Dec. 5, 1968).

23. FISCHER, P.U., JAYE, S., and STEWART, H.B., "Fuel Cycles for the HTGR," International Survey Course of the International Atomic Energy Agency, Vienna, Austria (Sept. 1-12, 1969).

24. CARLSMITH, R.S. and THOMAS, W. E., "Fuel Cycles for High-Temperature Gas-Cooled Reactors Utilizing Plutonium," ORNL Rep. ORNL-TM-1111 (Apr. 14, 1965).

25. DAHLBERG, R.C., "Comparison of HTGR Fuel Cycles for Large Reactors," ANS Topical Meeting on Gas-Cooled Reactors: HTGR and GCFBR, May 7-10, 1974, Gatlinburg, Tenn., CONF-740501 (May 1974).

26. SOUTHWORTH, B.W., LEE, D.H., and DAHLBERG, R.C., "The Use of Low Enrichment Uranium in the HTGR," General Atomic Rep. GA-9715 (Sept. 30, 1970).

27. DAHLBERG, R.C., "Comparison of HTGR Fuel Cycles for Large Reactors," Proc. of the Gas-Cooled Reactor Information Meeting, Oak Ridge, Tennessee, April 27-30, 1970, Rep. CONF. 700401 (1970).

28. GUTMANN, H., LARSEN, H., HANSEN, U., NEEF, H. J. and BROGLI, R. H., "Alternative Fuel Cycles for HTGRs," General Atomic Report GA-A13055 (June 1974).

29. GUTMANN, H., HANSEN, U., LARSEN, H., NEEF, H. J., "The High Temperature Reactor in the Future Fuel Market," paper presented at European Nuclear Conference, Paris, France, April 21-25, 1975.

30. INSCH, G.M. and JOYCE, A.J., "The Low Enriched H.T.R.," ANS Topical Meeting on Gas-Cooled Reactors: HTGR and GCFBR, May 7-10, 1974, Gatlinburg, Tenn., CONF-740501 (May 1974).

31. TEUCHERT, E., BOHL, L., RÜTTEN, H.J., and HAAS, K.A., "The Pebble Bed High Temperature Reactor as a Source of Nuclear Process Heat: Vol. 2, Core Physics Studies," Rep. Jül-1114-RG (Oct. 1974).

32. Project Staff, "Gas-Cooled Fast Breeder Reactor Quarterly Progress Report for the Period February 1, 1975 through April 30, 1975," Rep. GA-A13458 (July 17, 1975).

33. SIMON, R. H., DEE, J. B. and MORITA, W. I., "Gas-Cooled Fast Breeder Demonstration Plant," ANS Topical Meeting on Gas-Cooled Reactors: HTGR and GCFBR, May 7-10, 1974, Gatlinburg, Tenn., CONF-740501 (May 1974).

34. DALLE DONNE, M. and GOETZMANN, C. A., "Gas-Cooled Fast Breeder Reactor Design, Development, and Safety Features," ANS Topical Meeting on Gas-Cooled Reactors: HTGR and GCFBR, May 7-10, 1974, Gatlinburg, Tenn., CONF-740501 (May 1974).

35. Gas Breeder Reactor Association, "The Gas Cooled Breeder Reactor - Economic Power with Breeding," Nucl. Engr. Intl. July 1974, 566 ff.

36. BROGLI, R. H. and SCHULTZ, K. R., "Thorium Utilization in an FBR/HTGR Power System," Proc. of the Am. Power Conf. 36, 247-258 (1974).

37. BROGLI, R. and SCHLUETER, G., "Fuel Economics of GCFR/HTGR Symbiotic systems," Rep. GA-A13110 (Aug. 30, 1974).

38. KASTEN, P. R., BENNETT, L. L., and THOMAS, W.E., "An Evaluation of Plutonium Use in HTGRs," Rep. ORNL-TM-3525 (Oct. 1971).

39. KOUTS, S. L., KRASE, G. M., MEYER, AL, "HTGR Gas-Turbine Power Plant Preliminary Design," paper contributed by the Gas-Turbine Division of ASME and presented at the Gas-Turbine Conference and Products Show, Zurich, Switzerland, March 30-April 4, 1974.

40. Project Staff, "Gas Turbine HTGR Trade-Off Studies: Turbomachinery Orientation and Rating," Rep. GA-A13097 (Aug. 1, 1974).

41. NEILL, J. M., "Gas Turbine HTGR Safety Summary Report," Rep. GA-A13326 (March 14, 1975).

42. Project Staff, "Gas-Turbine HTGR Program, Quarterly Progress Report for Period October through December 1974," Rep. GA-A13301 (Jan. 31, 1975).

43. CHMIELEWSKI, R. D. and HOOT, C. G., "Fission Product Plateout Calculations for a Gas Turbine HTGR," Rep. GA-A13213 (Nov. 1, 1974).

44. KRAEMER, H., SCHULTEN, R. et al, "HTGR for Direct Cycle and Process Heat Applications," ANS Topical Meeting on Gas-Cooled Reactors: HTGR and GCFBR, May 7-10, 1974, Gatlinburg, Tenn., CONF-740501 (May 1974).

45. FORTESCUE, P. and QUADE, R. N., "Direct Cycle and Process Heat HTGR Development," ANS Topical Meeting on Gas-Cooled Reactors: HTGR and GCFBR, May 7-10, 1974, Gatlinburg, Tenn., CONF-740501 (May 1974).

46. "High-Temperature Nuclear Heat Source Study," Rep. GA-A13158 (Sept. 27, 1974).

47. "The VHTR for Process Heat," Rep. GEAP-14018 (Sept. 1974).

48. "The Very High Temperature Reactor for Process Heat," Rep. WANL-2455-1 (Sept. 1974).

49. MURATA, H. and ISHIKAWA, H., "Design and Development of an HTGR for Multi-Purpose Utilization in Japan," paper presented at ANS National Topical Meeting on Nuclear Process Heat, Los Alamos, New Mexico, October 1-3, 1974.