STATUS REPORT ON FAST BREEDER REACTORS (Why Breeders?)

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World reserves of low cost uranium and growth of nuclear power require commercial fast breeder reactors some time between 1980 and 1990. High gain breeders consume small amounts of uranium and therefore will extend existing resources of economic fission fuels by orders of magnitude. While development work on fast breeders has been going on for approximately two decades, large electric power producing breeders are still some years in the future. Technical feasibility of fast breeders has been proven, but competitiveness of large breeders with more conventional fossil fired or thermal reactor plants is still to be demonstrated. Advanced breeders, such as gas-cooled fast reactors with carbide fuels and direct cycle gas turbines, promise to have power costs comparable to the most optimistic estimates made for fusion reactors by the end of this century.

INTRODUCTION

It has been recognized by early nuclear power pioneers such as E. Fermi and W. Zinn that fast neutron fission reactors could breed more fissile material than they consume, and that such breeding would eventually become necessary to extend the known resources of uranium for economic electric power. As a matter of fact, the first fast reactor, Clementine, was operated in 1946 at Los Alamos with mercury cooling. It is interesting to remember that the first electric power from fission came from the first fast breeder reactor: EBR I produced 100 Kw of electrical power in 1951 at Arco, Idaho.

Since that time, a large amount of research and development has been carried on around the world in the fast breeder field, mostly in USA, USSR and Western Europe. While a number of experimental facilities have been operated, the first large electric power producing demonstration plants are only now being built. A significant amount of development is still necessary to transform those fast breeders into commercial power plants. The timetable is still somewhat uncertain and varies from one country to another, but it is generally agreed that by the mid-1980s, large fast breeder plants should be economically competitive.

In our report, we are going first to look at the requirements for electric power and more specifically for nuclear power and consider the fuel requirements and the resources available. We will then briefly mention the principles of fast and thermal breeders, and compare the two main types of fast breeders (LMFBR and GCFR). We will try to explain the main reasons for worldwide interest in FBR and state the main remaining development problems. We will then recall the historical evolution of FBR and describe the existing programs. Finally we shall make an attempt at cost comparison between fission and fusion reactors and suggest a timetable for those various concepts.

ELECTRIC POWER REQUIREMENTS

The increasing needs for electric power around the world lead to doubling times for power generation between 7 and 10 years. The fraction of total power requirements corresponding to electric power is also rapidly increasing. In the next thirty years (1970-2000), the US electric industry will build nearly ten times as much generating capacity as exists today. The present thermal generating capacity in the US is expected to increase from 270 000 MWe in 1970, to over 1 100 000 MWe by 1990 and 2 300 000 MWe by 2000 [Ref. 1]. The electrical energy generated in thermal plants which was 600×10^9 Kwh in 1960, and is 1300×10^9 Kwh in 1970, is expected to increase to over 5500×10^9 Kwh by 1990. Therefore the level of fuel consumption in 1990 will be over four times the level in 1970. The annual fuel needs in millions of tons of coal (or coal equivalent) will increase from about 535 in 1970 to over 2200 by 1990.

The present fuels for thermal plants producing electric power are now gas, oil, coal and uranium. In the US in 1970, most of the electrical energy comes from coal (about 300×10^6 ton/year), oil (about 270×10^6 bbl/year), and natural gas (3×10^{12} ft³/year). Present forecasts show a decrease in coal uses by 1990 ($\sim 226 \times 10^6$ ton/year) and an increase in oil needs (by a factor of about 2.4) and gas needs (about doubled). This forecast assumes that by 1990 approximately half of the electrical generation capacity of thermal plants in the US will come from nuclear energy (500 000 to 600 000 MWe nuclear) and that about 70% of the total electric energy generated will come from nuclear plants. The nuclear fuel requirements in net tons of U_3O_8 , increase from approximately 8000 in 1970, to 52 000 in 1980 and 90 000 in 1990. By the end of this century, it is expected that about half of the total US energy requirements will be provided by electricity, i.e., more than one-third of the total energy produced will come from nuclear power.

While the previous data correspond to US needs, similar requirements exist in the Western world. For instance, the annual needs in EEC countries will be about 30 000 metric tons of natural uranium in 1975; by 1980, they should be between 56 000 and 80 000 tons [Ref. 2]. For 310 000 to 410 000 MWe of installed nuclear power in the year 2000 in EEC countries, a total

of 325 000 to 525 000 metric tons of natural uranium will be required between 1970 and 2000.

Let us recall that the forecasts for nuclear power growth in the US show about 150 000 MWe installed by 1980, 500 000 MWe by 1990 and 10^6 MWe by the year 2000 [Ref. 3]. Figure 1 from the EEI Fast Breeder Reactor Report for 1968 shows that the US resources at less than \$10/pound of U_3O_8 are about 0.66×10^6 short tons. Those low cost resources should be exhausted by about 1985, even with plutonium recycle in light water reactors. The total estimated resources below \$15/pound of U_3O_8 of about 10^6 short tons would be reached some time around 1990. While those 1968 AEC estimates are subject to increases as new reserves are discovered, large investments in exploration, mining, milling and diffusion plants would be needed to meet the increasing ore requirements of LWR; without breeders, a total of about 3×10^6 short tons of U_3O_8 would be needed by the year 2000. With the introduction of advanced converters such as HTGR in the 1970s, the need for uranium ores will already be reduced significantly since new fissile material (U-233) is bred in another material (thorium) and the total uranium requirements of HTGR are about half those of LWR (see Table I from Ref. 4). Furthermore, HTGR fuel cycle costs are not as sensitive to increase in ore costs as are LWR fuel cycle costs and more expensive ores could therefore be used economically in HTGRs. But, even with a mixture of HTGR, LWR and fossil-fueled plants, it is estimated that about 2.5×10^6 tons of U_3O_8 would be used in the USA in the 50-year period between 1970 and 2020. Introduction of fast breeder reactors in 1984 is estimated to reduce those requirements to 1.1×10^6 tons of U_3O_8 , which is equivalent to a saving of 4×10^{11} tons of coal [Ref. 5]. The annual ore requirements would peak below 10^5 tons of U_3O_8 by 1990 and would decrease to about half this number by the year 2020.

The previous data correspond to US needs; we have already seen that similar requirements exist for Western Europe. The situation is worse there since the uranium resources are rather meager (only about 50 000 tons of natural uranium reserves in France), while the requirements are much larger. For instance, the annual needs of EEC countries by 1980 will already be

between 56 000 and 80 000 tons [Ref. 2]. It is also estimated that nuclear energy in Japan will provide 30% of the total electric energy production by 1978 [Ref. 6]. Furthermore, those countries have no industrial enrichment facilities and now have to import enriched uranium from the US, unless they build natural uranium reactors (Magnox or HWR) which do not appear to be competitive in those countries at the present time, or build their own commercial enrichment plants. Therefore breeding appears to be even more urgently needed in Western Europe and in Japan than in the USA.

FAST BREEDER REACTOR PRINCIPLES

Natural uranium contains only 0.7% of the fissionable material U-235, the remainder being U-238. In such reactors as LWR or HTGR, the uranium fuel is enriched through the gaseous diffusion process, to about 4% of U-235 in LWR for instance. While U-235 fission occurs in the reactor, a small fraction of the fertile material is transformed into new fissile material which itself is partly consumed in the reactor: U-238 yields Pu-239 in LWR and Th-232 yields U-233 in HTGR. But, in a light water reactor, less than 5% of the total weight of fissile and fertile material in the core is fissioned before the core is removed for reprocessing; furthermore, a large amount of natural uranium depleted in U-235 has been used in the diffusion plant to produce the 4% enriched uranium. Fuel utilization is better in an advanced converter such as HTGR, approximately by a factor of 2 (see Table I), but the fission fuel is still not used as efficiently as it would be in a breeder. The breeding principle consists of transforming the fertile material (U-238, Th-232) into new fissile material (Pu-239, U-233). A breeder produces more new fissionable material than it consumes, thus leading to enormous extension of the reserves of uranium and thorium: from a few decades with converter reactors, to hundreds or perhaps even thousands of years with fast breeders [Ref. 5].

Before we look in more detail at fast breeders, let us mention that breeding is thought to be feasible in thermal reactors operating with the Th-232/U-233 cycle. The ratio of neutrons produced to neutrons absorbed

in the artificial fissile material U-233 is about 2.2 in a thermal spectrum, which gives a maximum theoretical conversion ratio of 1.2 (ratio of new fissile material to fissile material consumed). Some of the excess neutrons leak out of the reactor or are absorbed in structural materials, and thus breeding will be at best marginal in thermal reactors. There are several types of such near breeders or low gain thermal breeders. The most advanced is the High Temperature Gas Cooled Reactor moderated by graphite and cooled by helium, which now operates on the U-235/Th-232/U-233 cycle: a 40 MWe prototype has been in operation at Peach Bottom since 1967, a 330 MWe plant will be operating in Colorado in 1972 and large HTGRs (600 to 1100 MWe) are now being marketed in Europe and in the US. The existing HTGRs are non-breeders, but if the ore costs justify it, their fuel cycle could be adapted to higher conversion ratios, perhaps by addition of some BeO to the graphite moderator. Two thermal breeder concepts are under development in the US, the Light Water Breeder Reactor (at Bettis Laboratory) and the Molten Salt Reactor (at Oak Ridge National Laboratory). While their breeding gain will always be very low, economic thermal breeders could significantly extend the ore resources by utilizing the important thorium reserves (comparable to the uranium reserves). The thorium cycle has also been considered with heavy water moderation in Canada. Those thermal reactors will probably only develop as high efficiency advanced converters since, at least for the time being, their potential marginal breeding is not justified by current economics.

What are now the main characteristic features of fast breeder reactors? As indicated by their name, most fissions occur at high energies (fast spectrum), and they produce more fissile material than they consume (breeding). Compared to existing thermal nuclear reactors where most fissions occur at thermal energy, the main differences with fast breeders are

- No moderator: in order for most fissions to occur at energies from, say 0.1 Mev to a few Mev, only fuel (fertile and fissile), structural materials (fuel clad, boxes, control rods) and coolant are found in the core.

- A large number of neutrons are produced per neutron absorbed in fissile material: approximately 2.3 neutrons are produced for each neutron absorbed in Pu-239, or about 2.65 neutrons if fast fission in U-238 is taken into account.
- Some of the neutrons produced in the core are absorbed in U-238 to produce Pu-239. Others are absorbed in natural uranium or depleted uranium surrounding the core (blanket) to produce more Pu-239.
- Most nonfuel materials have low neutron absorption cross section in a fast flux as compared to fuel materials (fissile and fertile). Thus, in a large reactor surrounded by blankets, losses by leakage or parasitic absorption may be kept very low.
- The breeding ratio could thus be high, 1.3 to 1.5 with oxide fuels, 1.5 to 1.6 with carbide fuels. Hence rather low doubling times could be obtained (the doubling time is the time required by the reactor to produce enough fissionable material to refuel itself and another reactor of the same power). The doubling time of large fast breeders could be made comparable to the doubling time of electric power generation needs, i.e., 7 to 10 years.
- Since fast reactors are not moderated, their fuel inventory is high (compared to thermal reactors) and their power density is of necessity very high (an order of magnitude higher than thermal reactors).
- Control of fast reactors must also take into account the fact that the characteristic times (neutronic and thermal) are smaller than for thermal reactors.
- Fast breeders are also more sensitive to small dimensional changes in the core, as they affect both reactivity and cooling.

As shown in Table II, a number of fast breeder concepts could be considered with various types of fuel and fertile materials, coolants, reactor

arrangement and power cycles. For various technical and historical reasons, the main types of fast breeder presently under consideration for power reactors are (see Fig. 2)

LMFBR: Liquid metal cooled (Na) with an intermediate sodium circuit (nonactive) and a steam cycle; $(\text{UO}_2\text{-PuO}_2)$ fuel in the core and depleted UO_2 in the blankets.

GCFR: Gas cooled (helium) with a steam cycle and the same type of fuel as LMFBR

Table III derived from Ref. 3 gives a comparison of advantages and disadvantages of the two main coolant approaches, namely sodium cooling and gas cooling. Table IV shows a simplified comparison of reactor coolants [Ref. 4]: water and helium for thermal reactors, helium and sodium for fast breeders. For typical design conditions of large LMFBRs and GCFRs, the heat transport per unit core frontal area with helium is about 80% of its value with sodium, while the heat transfer per unit fuel surface area is about 60%. Thus, while from a heat transfer viewpoint, pressurized helium is not quite as good as sodium, the gas still can effectively and economically (i.e., with reasonably low pumping power) remove the nuclear heat from the core. A comparison of typical performance of those two types of fast breeder is shown on Table V both for demonstration plants (~ 300 MWe) and large reactors (~ 1000 MWe). Comparable doubling times and cycle efficiencies are obtained with the two systems in both cases (demonstration plant and commercial reactor) and with two types of fuel (oxide and carbide). The specific inventory is lower for LMFBR (because of good sodium cooling properties), but the breeding ratio is also lower (because sodium acts as moderator); thus the doubling time is about the same for GCFR and for LMFBR.

INCENTIVES FOR FAST BREEDER REACTOR DEVELOPMENT

Let us restate the main reasons for the world-wide interest in the development of FBRs.

- The reserves of low cost uranium ores in the USA will probably be exhausted some time between 1985 and 1990 (see Fig. 1).

While existing light water reactor costs are very sensitive to increase in ore costs, the fuel cycle costs of advanced converters, and *a fortiori* breeders, are rather insensitive to such increase. Furthermore, the uranium requirements of fast breeders are very small, since, in principle, 100% of the uranium could be burnt, rather than only a fraction of a percent in LWRs.

- FBRs, as well as HTGRs, will have a much higher thermal efficiency than LWRs: 39 to 42% versus 32%, which reduces the thermal effects of the nuclear power plants. The steam conditions will be comparable to the most modern fossil-fueled plants.
- FBRs will offer a premium market for all the plutonium produced in LWRs. As shown on Fig. 3 (from Ref. 3), about 250 metric tons of plutonium will come out of US light water reactors by 1984. At a price of \$8/gram, it represents two billion dollars worth of plutonium.
- The large stockpiles of depleted uranium, estimated to be approximately 400 000 tons of U-238 by 1980, could be used as fertile material in FBRs: in a 30-year period, about 50% of the U-238 in a FBR will be converted into fissile plutonium. Thus the otherwise wasted depleted uranium could greatly increase the world energy resources.
- The projected power costs of FBRs will be lower than those of other reactors (which are already competitive with fossil fuel plants). Table VI adapted from Ref. 7 shows the range of projected costs of 1000 MWe plants after appropriate R&D programs. LMFBRs should give cheaper power than LWRs some time between 1984 and the year 2000 according to the various estimates. Similarly, LMFBRs could become competitive with HTGRs as early as 1986 by assuming high costs for HTGR and low costs for LMFBRs. By taking median values, the data from Table VI show that LMFBRs

should become competitive with LWRs by about 1989 and with HTGRs by about 1995. Similar results obtain with the Gas Cooled Fast Reactor which should be the cheapest system by the end of this century [Ref. 7] since the running costs of GCFR and LMFBR are the same and the plant costs of GCFR should be lower.

A Cost-Benefit Analysis of the US Breeder Program has been performed by the US AEC [Ref. 8]. This study compares the anticipated R&D costs of the breeder program to the expected savings in energy costs. This evaluation mainly considers the Liquid Metal Fast Breeder Reactor but it also mentions the effect of a parallel breeder program. With LMFBRs introduced in 1984, rising uranium costs, constant fossil fuel costs and a 6.3% per year growth rate of electrical energy demand (11.4 years doubling time), the benefit-to-cost ratio is 3.64 (for a discount rate of 7%). This is the ratio of anticipated savings to development costs, discounted to 1970. If GCFRs are introduced at the same time as LMFBR, the benefit-to-cost ratio may be shown to increase to 4.47, assuming a LMFBR installed capacity double that of the GCFR. Those results correspond to gross discounted savings compared to a non-breeder economy from 9 to 12 billion dollars, or more than 200 billion (1970) dollars benefits from 1984 to 2020 [Ref. 5]. In this analysis, the electrical energy demand increases from 2000×10^9 Kwhr per year in 1980 to $18,500 \times 10^9$ Kwhr in 2020. These studies assume a mixture of LWRs, HTGRs and FBRs introduced at different rates and dates.

DEVELOPMENT PROBLEMS OF FAST BREEDER SYSTEMS

The development problems of the two FBR systems are somewhat different, except for fuel development since both LMFBR and GCFR use the same type of UO_2 - PuO_2 stainless steel clad fuel pins. Tables VII and VIII show the status of FBR technologies [Ref. 9] and the major development areas for LMFBR and for GCFR [Ref. 3]. No basic feasibility problem is involved, and those developments are only required to obtain economic power from fast breeder power plants. FBR developments, system and component design, fuel

considerations, and core heat removal were discussed at the 1968 American Nuclear Society Meeting in Washington and are reported in Refs. 10 and 11. Before we emphasize the most important remaining problems, let us briefly describe these two FBR systems.

Figure 2 shows very schematically the sodium and the gas fast systems. In the LMFBR, the reactor core is cooled by a flow of primary sodium which exchanges its heat in a primary heat exchanger with a secondary sodium coolant. This nonradioactive secondary sodium raises steam in another heat exchanger and the steam expands in a steam turbine driving a generator. The primary sodium loop which has to be heavily shielded is sometimes contained in a large pool of sodium where one finds the reactor core, blankets and shields, the primary pumps, and the primary heat exchangers. In a GCFR, the primary loop comprising the reactor core and internals, the helium circulators and the steam generators, is contained within a Prestressed Concrete Reactor Vessel which holds the gas pressure (70 to 120 atm). The steam raised in the heat exchanger then expands in a steam turbine. In some advanced concepts, the hot gas out of the reactor expands directly into gas turbines which drive both helium compressors and generators [Ref. 12]. As previously mentioned, the cores of the LMFBR and GCFR are very similar, consisting of assemblies of vertical fuel elements, each containing a number (200 to 300) of fuel pins cooled by axial flow of sodium or helium respectively.

The advantages claimed by the proponents of these two systems, which both have high thermal efficiency and potentially low doubling times, are mainly

- LMFBR: a) Good heat transfer characteristics with low coolant pressure
- b) Potential for low specific fissile inventory
- c) Good emergency cooling possibilities, especially if the core can always be kept under sodium.

- GCFR: a) Inert, and transparent coolant which does not interfere with core neutronics
- b) Potential for high breeding gain

- c) Simplified operation and maintenance, and no need for intermediate coolant loop.

The GCFR design contemplates commonality of fuel with LMFBR, obtained by equalizing the pressure on both sides of the cladding [Ref. 13]. GCFR fuel will use artificial surface roughening to improve the surface heat transfer coefficient [Ref. 14]. Furthermore, as indicated in Table VII, a great deal of the HTGR technology is applicable to GCFR.

Besides the fuel problems which are common to both systems and will be discussed later on, the special development problems seem to be in the area of steam generators for sodium cooling and helium circulators for gas cooling. In view of the chemical reactions between sodium on one hand, water, steam and air on the other hand, the design of reliable, safe and economic sodium-cooled steam generators is a difficult problem which is now under development in a number of countries, such as France, Germany, Japan, UK, US and USSR [Refs. 10 and 11]. Core heat removal does not appear to be a problem in helium-cooled fast breeders under normal operation, but cooling must be insured under all circumstances, including the removal of afterheat once the reactor is shut down. Therefore reliable and redundant helium circulators (and drives) must be provided. A large amount of circulator design and testing is now taking place for thermal helium cooled reactors (HTGR) which should also be applicable to fast systems [Ref. 15]. Extension of the PCRV experience is also required since helium pressure in a GCFR is about double that found in thermal systems [Ref. 12]. Both LMFBR and GCFR require fast and reliable in-core instrumentation, although of a different nature: thermocouples are needed in both systems for measurement of coolant temperature at the exit of the fuel elements. But sodium might boil and be ejected out of the core in case of supersaturation, and acoustic monitors may be needed for early detection of boiling in order to avoid failure propagation and the effects of a positive void coefficient of reactivity. Detailed experimental verifications of core thermal and mechanical behavior are required for both FBR systems.

The maximum fuel burnup required for large economic fast breeders varies from 7-8% to about 10%, according to European or US estimates [Ref. 7]. Satisfactory experience has been obtained in fast test facilities in the US (EBR II) and in Europe (Rapsodie in France and Dounreay in UK), but only up to 5 to 8% with mixed oxide fuels. While the desired burnup has nearly been reached, the fast fluence (which indicates the amount of radiation damage) was usually a factor 3 to 4 lower than the fluence desired for large breeders ($2-4 \times 10^{23} \text{ n/cm}^2 > 0.1 \text{ Mev}$). Even with demonstration plants of approximately 300 MWe, the fast flux will be only about 1/3 and the fast fluence/burnup ratio only about 1/2 of the values for 1000 MWe reactors. The Fast Flux Test Facility (FFTF) will have a fast flux comparable to that of large FBRs ($\sim 7 \times 10^{15} \text{ n/cm}^2\text{-sec}$ above 0.1 Mev), but this large test reactor (400 Mwt) will only start operation in the US in 1974-75.

As shown in Table IX the design characteristics of fuel elements for fast breeder reactor prototypes are very similar, even with gas cooling. The fuel pin is small (0.5-0.6 cm pellet diameter), long ($\sim 100 \text{ cm}$) with a thin stainless steel clad (OD/ID: 1.12-1.16). The maximum linear rating is 13-18 Kw/ft (425-585 w/cm), the design hot spot clad temperature is 665-730C and the maximum burnup is 50-100 Mwd/Kg (fuel). A core plan for a 300 MWe GCFR demonstration plant is shown on Fig. 4 which is very similar to a LMFBR core.

Although fuel swells at high burnups, swelling of the stainless steel clad under high fast neutron fluences may even be more limiting. This radiation swelling of the steel depends both on fluence and temperature and may therefore cause serious distortion and bowing of the fuel elements. Close to 10% volume expansion of 20% cold-worked 316 stainless steel could be expected at fluences of about $10^{23} \text{ n/cm}^2 (>0.1 \text{ Mev})$ at 500C. Figures 5 and 6 show the strong dependence of radiation swelling upon fluence (significant only above $\sim 0.5 \times 10^{23} \text{ n/cm}^2$) and upon temperature (maximum for stainless steels around 450C). Some metallurgical improvements could be expected, but the nonuniform swelling of the fuel elements may have to be accommodated by leaving a clearance between elements (less detrimental to

the neutron economy in GCFR than LMFBR) and by periodic rotation of the fuel elements [Ref. 16].

HISTORICAL FBR DEVELOPMENTS AND WORLD PROGRAMS

As previously mentioned, the first fast reactor (Clementine) was operated in the US in 1946. Since then a number of fast reactors have been operated or are being built in the US and abroad as seen on Table X (from Ref. 8). The first fast breeder, EBR-I was started in the US in 1951 while the first sodium cooled breeders in the USSR and in the UK were started in 1959 (BR 5 and Dounreay respectively). The first fast reactor in France, Rapsodie, went critical in 1967 and has recently been uprated from 20 to 40 Mwt. Some characteristics of those first generation fast breeder reactors are shown in Table XI (from Refs. 10 or 11). Most of these first reactors use metallic fuel, except for Rapsodie which uses $\text{PuO}_2\text{-UO}_2$ fuel, and BR 5 which had a PuO_2 core and also a UC core; those fast reactors are cooled by liquid metal, Hg, NaK or Na.

Table XII updated from Ref. 3 shows the milestones of the LMFBR programs around the world. Germany, together with Belgium and Holland, Italy and Japan have started ambitious FBR programs which follow by a few years the programs already started in France, UK, USA and USSR. It may be noted that, although the first demonstration plant went critical in 1963 (Enrico Fermi in the US), the next demonstration plant is not scheduled to start until 1970 (BN 350 in the USSR). Figure 7 shows the Fermi reactor which has not yet operated at its design power (200 Mwt) since some of the fuel elements experienced melting in 1966. Table XIII shows the second generation of LMFBRs, which are demonstration plants with electric power production in the 250 to 600 MWe range, and not commercial reactors. All of them will have stainless clad $\text{UO}_2\text{-PuO}_2$ fuel cooled by sodium. As shown in this table, the net cycle efficiency is high (35 to 42%), and the power density, specific power and linear rating are very similar for all plants. In most cases, the breeding ratio is rather low and therefore the doubling time is high. The assumed maximum burnup varies from 50 to 100 MWd/Kg. Both pool and loop designs are utilized. Figure 8 shows an example of the pool concept

for the PFR under construction in the UK. Figure 9 from Ref. 17 shows a concept of a 1000 MWe sodium-cooled fast breeder with the same pool design, which should have a net efficiency of 41%, a specific power of 1.3 MWt/Kg(fissile) and an average power density of ~ 0.6 MWt/liter (of core). With a breeding ratio of 1.4 with $\text{PuO}_2\text{-UO}_2$ fuel, the doubling time should be approximately 8 years.

Rather large sums of money are being spent now on LMFBR programs: close to \$125 millions per year in the US and about the same amount abroad, mainly in France, Germany, Italy, Japan and UK. The spending level is also high in the USSR where two large reactors are presently under construction, BN 350 and BN 600. It is felt that total expenditures of 2 to 3 billion dollars would have to be spent in the US before the advent of true commercial breeders some time around 1990. As could be seen in Table XIII, the development philosophy is quite different in the US and abroad. A conservative approach is taken by the US AEC [Ref. 18] whereby large component development and fuel testing programs (including full-size loops in a Fast Flux Test Facility) precede the construction of one or more demonstration plants. In other countries, such as France, Germany, UK and USSR, construction of demonstration plants follows closely the operation of first generation reactors (Rapsodie, DFR); a great deal of the fuel testing will be performed in the demonstration plants which therefore are expected to go up to full power rather slowly. This latest approach should lead to earlier commercial development of fast breeders as shown on Table XIV established from recent estimates [Refs. 7, 11, 18, 19, 20]. The first large FBR should operate in USSR before the first US demonstration plant. There now appears to be a lag of several years between the US and the European LMFBR programs.

A review of the US fast breeder reactor program was recently published in *Energie Nucleaire* [Ref. 21]. The US FBR program was discussed by M. Shaw and J. Yevick of the US AEC and by H. Fenech of the University of California at Santa Barbara. Four US manufacturers described their LMFBR programs, and gas cooled fast reactor developments were also presented (see also Ref. 4) Reports on the US and European GCFR work were recently given at Oak Ridge National Laboratory [Ref. 22] and are discussed by C. Rennie [Ref. 23].

The potential of the GCFR system has previously been discussed in two US AEC reports on alternate breeders [Refs. 24 and 25]. The utilization of a closed cycle helium turbine coupled with a GCFR is presented in Refs. 12 and 26. Figure 10 shows a conceptual design of a GCFR 300 MWe demonstration plant [Ref. 16] whose characteristics are given in Table XV, together with typical design parameters for a 1000 MWe plant. These GCFR designs are based on LMFBR fuel technology and, as shown on Table IX, the fuel element parameters are very similar. The Prestressed Concrete Reactor pressure Vessel containing the core, blankets and shields in a central cavity and the steam generators and helium circulators in satellite cavities is similar to designs for other thermal gas-cooled reactors such as AGR or HTGR. Design and safety studies for a GCFR demonstration plant are being pursued by Gulf General Atomic Incorporated together with 41 US and one foreign utility company. Some fuel irradiation is being performed under US AEC sponsorship and work in heat transfer and fluid flow is under way in a joint program with the Swiss Federal Institute (EIR). Besides government studies in Europe, a Gas-cooled Breeder Reactor Association has been set up in Brussels in 1969 by 11 industrial groups, utilities, and government organizations with members from seven European countries. It appears that the development of an alternate breeder system could proceed with a timetable not very different from that of the LMFBR, i.e., a large commercial GCFR with steam cycle could be in operation some time between 1985 and 1990.

CONCLUSIONS

We have tried to show the status of fast breeder reactors which have now been under development for about 20 years in this country and will probably be built in large numbers by utilities some time in the coming twenty years. The timetable for introduction of FBRs is different in Europe and in the US because of different needs. Uranium and thorium reserves are much larger in the US where enrichment facilities are also readily available and therefore it does not appear that there is an urgent need for breeders in this country before about 1990, while they may be needed in Europe a few years before (1980-1985). Most of the development work up to now has been performed for Liquid Metal Fast Breeder Reactors, but Gas-Cooled Fast breeder Reactors based on similar fuel technology will benefit from the advent of High Temperature Gas-cooled Reactors and should be competitive at about the same time. Fast breeders will considerably extend the use of fission fuels (uranium and thorium) since they make a very efficient use of fissile material and breed new fissile material from fertile material. The plutonium produced by existing Light Water Reactors will be used in FBRs and thermal and fast reactors will probably coexist for a number of years, together with fossil fuel plants (especially for small plant sizes).

It is interesting to speculate, now that we have attempted to answer the question *why breeders?* how those advanced fission reactors will fit in the timetable for fusion reactors. In an article published in the June 1970 issue of Fortune [Ref. 27], entitled "The Hot New Promise of Nuclear Power," Mr. T. Alexander writes "The closest to hand is the fast breeder fission reactor, the technical and economic feasibility of which seems assured. The other, and potentially far better, is controlled thermonuclear fusion. In some ways complementary to the breeder, in some ways competitive, the fusion principle is inherently safer and cleaner. And if direct conversion of fusion energy to electricity could be achieved, as many now believe, civilization might obtain its energy at a price far lower than ever imagined." He continues by stating that a Controlled Thermonuclear Reactor has not yet reached the stage of the first fission reactor in 1942.

The timetable for fusion reactors varies according to the degree of optimism, but it was pretty much agreed at the Culnam meeting [Ref. 28] that a true controlled nuclear reaction (producing more power than is provided to it) could be expected to occur during this decade. In view of the difficult problems of economic fuel injection and material behavior under high neutron bombardment [Ref. 29], especially for the most promising (D-T) reaction, it is questionable that an acceptable commercial fusion reactor could be built before the end of this century. As we have seen, there is no immediate need for fusion reactors from the viewpoint of fuel resources. It is true that the inventory of radioactive materials will be smaller in a fusion reactor than in a fast breeder of the same power, and also that a fusion reactor does not contain several critical masses. But a large amount of radioactive tritium circulates in the various loops of a fusion reactor and the materials in the blankets (niobium or molybdenum) become quite active. Therefore fusion reactors will also most probably require a secondary containment like fast breeders. The much lower thermal pollution ($\sim 90\%$ overall efficiency) claimed for fusion reactors will only be obtained in the next century with direct conversion of thermonuclear power to electricity from such reactions as (D-He³) which require a much higher ignition temperature than (D-T) reactions. In the present systems with intermediate fluids such as lithium [Refs. 30, 31] or helium [Ref. 32], and a steam turbine or gas turbine cycle, the net efficiency of fusion systems is comparable to that of advanced fission reactors ($\sim 40\%$) when the energy required for coolant pumping and fuel injection is taken into account.

At this point, it may be interesting to look briefly at the world reserves of fission and fusion fuel and also at a brief comparison of extrapolated power costs. We first remark that approximately the same amount of energy is obtainable per unit mass of fission fuel (U or Th) or fusion fuel (lithium needed to breed tritium): about 1 MW-day/gram. Table XVI derived from Ref. 33, shows that the energy content of (U + Th) and of lithium are about the same; even if more lithium were discovered, the content of natural lithium in the earth's crust (65 ppm) is only four times that of U + Th (16 ppm). Therefore the amount of energy obtainable from

the (D-T) fusion reaction is of the same order of magnitude as the energy obtainable from fission reactions, i.e., sufficient for a large number of centuries. Deuterium out of the oceans will then carry on the burden for millenia. The last remark concerns power costs; which, it is recognized, are much harder to ascertain for an unknown technology such as fusion reactors than for an extrapolation of known technology, namely fast breeders. Using numbers presented by Werner et al. in Ref. 31, the capital cost of a 4200 MWe fusion reactor is about \$235/Kwe, while the running costs, including fuel, are about 1.75 mill/Kwh: therefore the net power costs are approximately 6.45 mill/Kwh. The data given by the latest EEI report [Ref. 7] for 1000 MWe GCFR plants in the year 2000 are \$150-180/Kwe and 0.7-0.9 mill/Kwh running costs, leading to total power costs of only 3.7 to 4.5 mill/Kwh. Even using the more optimistic data from Carruthers et al. [Ref. 33], the capital costs for a 2100 MWe fusion reactor is 175-188 \$/Kwe and the running costs are about 0.21 mill/Kwh, leading to power costs of 3.71-3.97 mill/Kwh, compared to about 3.73 mill/Kwh for a fast breeder of the same electrical output. With the previous data it appears that the first large (D-T) fusion reactors could only be marginally competitive with fast breeder reactors by the year 2000. There most probably will be room for advanced thermal fission reactors, fast breeders and fusion reactors for many years, with possibly even symbiosis between the various types of reactors, such as breeding of thermal fission fuel (U-233) in fast breeders, or breeding of fast breeder fuel (Pu-239) in fusion reactors [Ref. 34].

Let us conclude by quoting from Chairman Seaborg's speech of November 26, 1969 [Ref. 5]. "If the energy potential of breeders is so good, why bother with trying to achieve controlled thermonuclear fusion?...I mentioned earlier that fission through breeders would supply us with energy for centuries or perhaps thousands of years. With the successful utilization of controlled fusion, man will have virtually an unlimited energy resource at hand. Even at a power consumption rate many times that of today's, he will have an energy reserve that will last for millions of years."

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TABLE I

URANIUM UTILIZATION FOR 1000 MWe PLANTS*
AT 80% CAPACITY FACTOR

	LWR	HWR†	Gas Cooled Reactors		
			Magnox†	HTGR	GCFR**
Natural U required for first core, tonnes U_3O_8	580	300	1130	450	700
Natural U required for yearly feed, tonnes U_3O_8	170	150	330	55	—
Natural U equivalent of yearly U discharge, tonnes U_3O_8	40	—	—	5	—
Natural U equivalent of yearly Pu discharge, tonnes U_3O_8	20	—	—	—	+75
Net yearly requirements tonnes U_3O_8	110	150	330	50	-75
Total 30-yr commitment tonnes U_3O_8	3770	4650	10,700	1900	-1500

Similar results obtain for LMFBRs.

- * LWR = light water cooled and moderated reactor
 HWR = heavy water cooled and moderated reactor
 Magnox = CO₂ cooled, graphite moderated, natural uranium reactor
 HTGR = high temperature gas cooled reactor (thorium cycle)
 GCFR = gas cooled fast breeder reactor (PuO₂-UO₂ fuel)

†Throw-away cycle

- ** Assuming fissile plutonium equal to 100% enriched U-235.
 (Fertile material: depleted uranium ~1 tonne U_3O_8 /yr)

TABLE II
DESIGN VARIABLES FOR FAST BREEDERS

Fissile material:	U-235, Pu mixtures
Fertile material:	U-238, depleted U, Th-232
Type of fuel:	metal, oxide, carbide
Reactor coolant:	liquid metal (Na, NaK), gas (He, CO ₂) steam
Reactor arrangement:	integrated concept (pool), loop design
Power cycle:	direct (gas turbine), indirect (gas/steam) intermediate loop (Na/Na/steam)

TABLE III

Comparison of FBR Coolant Approaches
(from EEI, April 1968)

Sodium-Cooled	Gas-Cooled
ADVANTAGES	
1. Good emergency and post-accident cooling	1. Chemically and neutronically inert
2. Low pressure	2. No coolant phase-change void effect
3. Extensive fast reactor fuel and component experience	3. High internal conversion ratio and long refueling intervals
4. Good heat transfer characteristics	4. Potential high breeding ratio and short doubling time
5. Vented fuel possible	5. Vented fuel possible
6. Low specific inventory potential	6. Small positive void coefficient
DISADVANTAGES	
1. Opaque fluid	1. No fast reactor fuel and component experience
2. Chemically and neutronically active	2. Emergency cooling problem
3. Phase change-void effect	3. Heat transfer limited
4. Secondary heat transport system required.	4. Stringent leak requirements
5. Component development required	5. Component development required
6. Lack of electric utility experience with sodium	6. High coolant velocity
	7. High pressure

TABLE IV
SIMPLIFIED COMPARISON OF REACTOR COOLANTS

COOLANT	WATER	HELIUM		SODIUM
Reactor Type	(PWR)	(HTGR	— GCFR)	(LMFBR)
Average temperature (°C)	300	600	450	450
Average pressure (atm)	150	50	100	10
ΔT (coolant rise) °C	25	400	300	165
Density ρ (g/cm ³)	0.70	0.003	0.0062	0.80
Specific heat C_p (wsec/g-°C)	5.2	5.2		1.25
Flow velocity V (cm/sec)	400	5000	8000	600
Heat transport per unit frontal area $\rho V C_p \Delta T$ (w/cm ²)	36,000	31,000	78,000	100,000
Heat transfer coefficient h (w/cm ² -°C)	3	0.3	1.5*	10
Average film drop Δt (°C)	25	200	100	25
Heat transfer per unit surface area $h \Delta t$ (w/cm ²)	75	60	150	250

*With artificial surface roughening doubling the heat transfer coefficient.

TABLE V
TYPICAL PERFORMANCE OF FAST BREEDER REACTORS

	DEMONSTRATION PLANT*		COMMERCIAL REACTOR**			
	Na	He	Na	He	Na	He
Coolant						
Fuel	UO_2-PuO_2		UO_2-PuO_2		UC-PuC	
Power (MWe)	250-500	310	1000	1000	1000	1000
Inventory $\frac{\text{Kg-fiss(core)}}{\text{MWe}}$	2.5-3.5	4.4	1.55	2.6	1.45	2.0
Breeding ratio	1.1-1.3	1.33	1.31	1.51	1.50	1.60
Doubling time (year)	15-20	~20	8	8	4.5	4.5
Coolant exit temp. (C)	540-590	540	650	635	585	590
Cycle efficiency	0.39-0.41	0.38	0.43	0.40	0.39	0.38
Capital costs (\$/Kwe)			150	130	150	130
Fuel costs (mill/Kwh)			0.8	0.8	0.4	0.4
Power costs (mills/Kwh)			4.2	3.8	3.8	3.4

*LMFBR demonstration plants in France, Germany, UK and USA. GCFR designs from GGA.

**The costs are taken from US AEC studies (1969)

TABLE VI
PROJECTED NUCLEAR POWER COSTS (1970 dollars)*

Year of Startup	LWR	HWR	HTGR	LMFBR	GCFR	MSBR
<u>1975</u>						
Plant (\$/Kwe)	200-240	280-340	230-270			
Energy (mill/Kwhr)	6-7	6.8-8.2	6.1-7.1			
<u>1980</u>						
Plant (\$/Kwe)	170-210	270-330	180-220	Demon- stration plant		
Energy (mill/Kwhr)	5.2-6.2	6.6-8.0	5.1-6.1			
<u>1985</u>						
Plant (\$/Kwe)	160-190	220-260	160-190	220-260	Demon- stration plant	
Energy (mill/Kwhr)	4.9-5.7	5.5-6.5	4.6-5.4	5.5-6.5		
<u>1990</u>						
Plant (\$/Kwe)	155-185	200-240	150-180	180-220	225-275	Demon- stration plant
Energy (mill/Kwhr)	4.8-5.6	4.8-5.8	4.3-5.1	4.5-5.5	5.6-6.8	
<u>2000</u>						
Plant (\$/Kwe)	150-180	180-200	135-165	160-190	150-180	150-180**
Energy (mill/Kwhr)	4.7-5.5	4.2-5.2	4.0-4.8	3.9-4.7	3.7-4.5	3.7-4.5

* from Ref. 7, Edison Electric Institute 1970 report

** beyond 2000

TABLE VII
Status of FBR Technologies

Technological area under study or development	GCFR	LMFBR
Coolant	Considerable operating experience with British reactors (CO ₂) and US and European HTGR (helium)	Operating experience with EBR-II, SEFOR and Fermi in the US, DFR in England and Rapsodie in France
Fuel	GCFR fuel design will be based primarily on LMFBR experimental data. Heat transfer results with roughened surfaces available mostly from European GCRs.	Experimentation is under way. High burnup on fuels will be tested upon completion of the Fast Flux Test Facility and the demonstration plant.
Pressure vessels	Considerable PCRV experience in England and Europe. Fort St Vrain should provide any added US experience required.	High pressure primary containment not required.
Pumps and circulators	HTGR experience will be directly applicable to GCFR. Fort St Vrain circulator development program has been completed. Also, UK and European gas-cooled reactors from which to draw experience.	Both mechanical and electromagnetic pumps are under study. Mechanical pumps seem to be favored. Pumps to about 20,000 gpm will be needed for demonstration plants. Larger sizes will be needed for 1000 MWe plants. At present, 12,000 gpm is largest size operated.
Steam generators	Steam generators for Peach Bottom HTGR are operating satisfactorily at design output. Steam generators for use at Fort St Vrain are of a different design. Final proof testing of the design will occur when Fort St Vrain starts up in 1971. Since GCFR steam generators will be the same design principles as the Fort St Vrain generators, Fort St Vrain will provide most of the necessary steam generator experience.	Present steam generators use bi-metallic tubes to mitigate corrosion problems. Only the Fermi reactor has a single tube steam generator and it has a history of leakage. However, successful single tube steam generators will probably be required for economical operation. Considerable amount of sodium steam generator testing now under way.

TABLE VIII
MAJOR DEVELOPMENT AREAS FOR LMFBR AND GCFBR*
(from Ref. 3)

Item	Na	Gas		Na	Gas
<u>Physics</u>			<u>Fuel and Materials</u>		
Large Pu criticals	X	X	Fuel irradiation	X	X
Doppler confirmation	X	X	Clad, structure swelling	X	X
Data methods	X		Fuel/cladding interaction	X	X
Coolant reactivity			Cladding support/cycling		X
Void coefficients	X				
<u>Safety</u>			<u>Instrumentation/Control</u>		
Realistic DBA**	X	X	Fuel failure monitor	X	X
Loss of coolant		X	Impurity monitoring	X	
Emergency decay cooling		X	Boiling detection	X	
Coolant voiding effects	X				
Fuel failure propagation	X	X	<u>Coolant Technology</u>		
<u>Core Design</u>			Chemistry and Purification	X	
Fuel element structure		X	Na-H ₂ O reaction	X	
S/A irradiation testing	X	X	<u>Fuel Recycle</u>		
Coolant boiling effects	X		Fuel fabrication	X	X
Heat transfer/hydraulics		X			
Core clamping and hold-down	X	X			
<u>Plant Design</u>					
Emergency decay coolant		X			
Temperature optimization	X	X			
Fuel transfer system	X	X			
System transients	X	X			
<u>Components</u>					
Steam generator	X	X			
PCR/V		X			
Pumps/blowers	X	X			
Fuel handling	X	X			

* Areas where effort is required are indicated by an X.

** Design Basis Accident

TABLE IX

CHARACTERISTICS OF STAINLESS STEEL CLAD, MIXED OXIDE FUEL ELEMENTS
FOR FAST BREEDER REACTOR PROTOTYPES

	PFR UK	Phenix France	SNR Germany	BN-350 USSR	BN-600 USSR	GE USA	AI USA	Westinghouse USA	GGA USA
Coolant	Sodium	Sodium	Sodium	Sodium	Sodium	Sodium	Sodium	Sodium	Helium
Power, MWe	250 (275)*	233	295	350	600	310	500	300	~310
Pellet diameter, cm	0.50	0.55	0.52	~0.52	0.60	0.51	0.54	0.56	0.62
Cladding OD/ID	1.15	1.16	1.15	1.13	1.135	1.16	1.135	1.125	1.15
237 Maximum linear rating, Kw/ft	13.5	~14.5	~17.5	14	13 (av)	15	15	13.5	13.8
Hot spot cladding temperature at mid-radius, C	650 (700)*	700 (730)*	700	710	690	~700	~700	665	700
Maximum burnup, 10 ³ MWd/tonne	70	≥50	85	≥50	100	100	75 (av)	100	50-100

* For later cores.

TABLE X
LIQUID METAL COOLED FAST REACTOR PROJECTS (from Ref. 8)

<u>Name</u>	<u>Country</u>	<u>Power (MWt)</u>	<u>Coolant</u>	<u>Initial Operation</u>
Clementine	United States	.025	Hg	1946
EBR I	United States	1.0	NaK	1951
BR 2	USSR	0.1	Hg	1956
BR 5	USSR	5.0	Na	1959
Dounreay	Great Britain	60.0	NaK	1959
LAMPRE	United States	1.0	Na	1961
EBR II	United States	60.0	Na	1963
Fermi	United States	200.0	Na	1963
Rapsodie	France	20.0 ¹	Na	1967
SEFOR	United States ²	20.0	Na	1969
BR 60 (BOR)	USSR	60.0	Na	1970
BN 350	USSR	1000.0	Na	1970
PFR	Great Britain	600.0	Na	1972
JEFR	Japan	100.0	Na	1973
Phenix	France	600.0	Na	1973
FFTF	United States	400.0	Na	1974
PEC	Italy	140.0	Na	1975
NA 2	W. Germany ³	750.0	Na	1975
BN 600	USSR	1470.0	Na	1976

¹Being increased to over 40 MWt in 1970.

²With Germany and Euratom.

³With Belgium and Netherlands.

TABLE XI
First Generation Fast Breeder Reactors

	USA				USSR			UK	France
	CLEMEN-TINE	EBR-I	EBR-II	EFFBR	BR-1	BR-2	BR-5	DFR	RAPSODIE
<i>Reactor Power</i>									
Thermal, MW(th)	0.025	1.2	62.5	200	0	0.1	5	72	20
Electrical, MW(e)	0	0.2	20	66	0	0	0	15	0
<i>Core</i>									
Fuel	Pu-Metal	U-Metal	U-Metal	U-Metal	Pu-Metal	Pu-Metal	PuO ₂	U-Metal	PuO ₂ /UO ₂
Core volume, liters	2.5	6	65	420	1.7	1.7	17	120	54
Fuel rating av, MW(th)/kg fissile	0.0016	0.02	0.3	0.37	0	0.008	0.1	0.24	0.14
Power density av, MW(th)/liter	0.01	0.17	0.8	0.45	0	0.06	0.3	0.5	0.32
Linear rod power max, W/cm	(av 50)	300	450	250	0	150	200	(av 320)	(av 210)
Neutron flux max, n/(cm ² sec)	(av 5 × 10 ¹²)	1.1 × 10 ¹⁴	3.7 × 10 ¹⁵	4.7 × 10 ¹⁵	5 × 10 ¹⁰	1 × 10 ¹⁴	1 × 10 ¹⁵	2.5 × 10 ¹⁵	1.8 × 10 ¹⁵
<i>Primary Heat Transfer System</i>									
Coolant	Hg	NaK	Na	Na	-	Hg	Na	NaK	Na
Coolant temperature									
Core inlet, °C	40	230	370	290	-	30	375	200	410(450)
Core outlet, °C	120	320	470	430	-	60	450(500)	350	500(540)
Coolant mass flow, m ³ /h	0.6	80	2200	5500	-	6	240	1800	800
Number of coolant loops	1	1	2	3	-	1	2	24	2
<i>Time Schedule</i>									
Design	1945	1945					1956		1958
Construction	9/1946	1949	1957	8/1956			1957	3/1955	1962
First criticality	11/1946	8/1951	10/1961	8/1963			6/1958	11/1959	1/1967
Full operation	3/1949	12/1951	4/1965	8/1966	1955	1956	7/1959	7/1963	3/1967
Shutdown	6/1953	1963	-	-	1956	1957	-	-	-
<i>Remarks</i>	1. fast reactor, 1. Pu-fueled reactor	1. nuclear electricity generation Pu-Core since 1962	Reactor Plant with integral fuel processing facility	Since 10/1966 out of operation			UC-Core since 1965		(RAPSODIE is not really a reactor of the first generation, it belongs, to a large extent, to the second generation)

TABLE XII
FBR PROGRAMS — Milestones (from Ref. 3)

	USA	USSR	UK	France
Start initial studies	1945	1949-1950	1951	1956
First critical :small	1955	1955	1954	--
facility operation :large	1968	1961	1962	1966
Experimental reactor critical	1946 (Clementine) 1951 (EBR-I) 1961 (LAMPRE) 1963 (EBR-II) 1963 (EVESR)	1956 (BR-2) 1958 (BR-5)	1959 (DFR)	1967 (Rapsodie)
5% burnup in :thermal flux	1960			
FBR pins :fast flux	1967	1961	1964	1968
Test reactor critical	1969 (SEFOR) 1974 (FFTF)	1969 (BOR) (60 MWt)	--	--
Large test :(>5 MWt or	1966 (SCTI)		1963 (7000 gpm pump)	1962 (10 MWt, IHX, pump)
loops operation : >5000 gpm)	1970 (SPTF)	1966	1967 (large Na-H O)	1964 (5 MWt SG) 1969 (20 MWt SG)
Demonstration reactor critical	1963 (Fermi) ~1978 (Na FBR)	1970 (BN 350)	1972 (PFR)	1973 (Phenix)
Large reactor critical (estimate)	~1984	1975 (600 MWe)	1979	1980
Professional Staff 1966-1967	1000	?	~650	~700
R&D funds :through 1967	400		?	?
millions \$:1967 annual	~70	?	~35	~30
:70 - 75 annual	big increase?		increase	~30
Reactor construction millions \$:already built	EBR I-II LAMPRE 90 Fermi	?	24 DFR	45 Rapsodie
:being built or discussed	330 SEFOR FFTF 3 Dems.?		70 PFR	100 Phenix

Table XII contd.

	Germany/Belgium/Holland	Italy	Japan
Start initial studies	1956 (Belgium), 1960 (Germany)	1962	1962
First critical :small	--	--	--
facility operation :large	1966	--	1967
Experimental reactor critical	1970 (KNK-Na)	--	1973 (JEER)
5% burnup in :thermal flux		1968	
FBR pins :fast flux	1968	1968	1969
Test reactor critical	--	1975 (PEC) (140 MWt)	1972 (~100 MWt)
Large test :(>5 MWt or	1965 (5 MWt SG)		
loops operation :>5000 gpm)	1969 (88000 gpm pump)	1972 (50 MWt SG)	(under consideration)
	1968 (large Na-H ₂ O)		
	1970 (50 MWt SG)		
Demonstration reactor critical	1975 (Na)	1978?	1977?
Large reactor critical (estimate)	~1980	1983?	1987
Professional staff 1966-1967	200-250 (Germany) 150-200 (Belgium + Holland)	~100	~150
R&D funds millions \$:through 1967	~60 (Germany), ~10 (Belgium & Holland)	~15	?
:1967 annual	~40 (+15 Belgium & Holland)	~10	~10
:70 - 75 annual	increase	big increase	big increase
Reactor construction millions \$:already built	0	0	0
:being built or discussed	HNK , Na Dem. (300)	PEC Na Dem. (140)	JEER Na Dem. (150)

TABLE XIII
LMFBR Demonstration Plants*

	USA			USSR		UK	France	Germany
	GE	Westing-house	AI	BN-350**	BN-600	PFR	Phenix	SNR
<u>Reactor Power</u>								
Thermal, MWt	750	770	1250	1000	1470	600 (670)	563	730
Electrical, MWe	310	300	~500	350	600	250 (275)	233	295
Net efficiency (%)	41.4	39	40	35	40.8	~41	41.5	40.4
<u>Core (reference)</u>								
Fuel: pin length (cm)	90	76	125	105	70	91.5	85	95
diameter (cm)	0.58	0.64	0.64	~0.60	0.69	0.585	0.66	0.60
Core volume (liter)	2000	~2000	3100	1900	~2300	1320	1225	1750
Fuel rating av, MWt/kg fissile	0.82	0.85	0.93	~1.28	1.22	~0.7	0.78	0.84
Power density av MWt/liter (core)	0.31	0.39	~0.37	0.5	0.55	~0.4	0.43	0.37
Linear rod power max, W/cm	500	450	490	470	~500	450	~480	580
Breeding ratio	1.2	1.22	1.3	1.5	~1.35	1.15-1.25	1.16	1.33
Burnup, MWd/Kg (max)	100	~100	75 (av)	50	100	70	>50	85
<u>Primary Heat Transfer System</u>								
Type	pool	loop	loop	loop	pool	pool	pool	loop
Number of coolant loops	3	2+1	3	5	3	3	3	3
<u>Coolant temperature</u>								
Core inlet, C	425	400	405	300	380	400 (430)	405 (420)	375
Core outlet, C	590	550	570	500	550	560 (600)	565 (590)	545
<u>Steam conditions</u>								
Temperature, C	510	480	~480	435	505	515 (565)	510 (540)	505
Pressure, atm	170	170	~170	45	140	162	163	165
Date of Operation				1970	1976	1972	1973	1975

*PuO₂-UO₂ fuel, stainless steel clad.

**UO₂ for the first load (950 Kg of U-235)

TABLE XIV

ESTIMATED LMFBR POWER PLANTS TIMETABLE

	<u>Demonstration Plant</u> (250 to 500 MWe)	<u>Large Plant</u> (600 to 1000 MWe)	<u>Commercial Plant</u> ($\geq$ 1000 MWe)
France	1973*	1980	~1984
Germany	1975*	1980	~1986
Japan	1977		1987?
UK	1972*	1979	~1983
USA	1977-78	1984	~1990
USSR	1970*	1976*	~1980

*Plants under construction or committed

TABLE XV
TYPICAL PARAMETERS FOR GAS-COOLED FAST BREEDER REACTORS*

Plant	Commercial Reactor	Demonstration Plant
Electrical power (Mw)	1000	311
Thermal power (Mw)	2530	826
Net cycle efficiency (%)	39.5	37.6
Helium pressure (atm)	85	85
Reactor inlet temperature (C)	340	312
Reactor outlet temperature (C)	635	541
Midclad hot spot temperature (C)	800	700
Maximum linear rating (w/cm)	580	450
Maximum fuel burnup (Mwd/t)	100,000	50,000-100,000
Active core volume (liters)	8600	3200
Active core length (cm)	160	100
Fuel pin outside diameter (cm)	0.8	0.715
Clad OD/ID	1.10	1.15
Fuel volume fraction	0.30	0.30
Coolant volume fraction	0.54	0.45
Reactor pressure drop (atm)	3.4	2.7
Average rating Mw/kg (fissile)	0.90	0.60
Average power density kw/l (core)	270	240
Total pumping power (Mw)	~100	~50
Average fuel enrichment (%)	18	12
Fuel life (year)	2.3	2.4
Doubling time (year)	9	21
Conversion ratio	1.5	1.33
Steam conditions:		
Temperature, C (F)	538 (1000)-538 (1000)	468 (875)-496 (925)
Pressure, atm (psi)	163 (2400)	196 (2900)

* $\text{PuO}_2\text{-UO}_2$ fuel with stainless steel clad and surface roughening.

TABLE XVI
ENERGY CONTENT OF WORLD FUEL RESERVES
(1Q = 10^{18} Btu)

	<u>High Grade Ores</u>	<u>Low Grade Ores (estimate)</u>
Uranium	76	5×10^6
Thorium	48	7×10^6
U + Th	124	<u>12×10^6</u>
Lithium	18*	5×10^6
(Li ⁶ in oceans)		6×10^6
Total Li		<u>11×10^6</u>

Assumptions: 100% utilization of U and Th by breeders;
Li⁶ content (7.4%) burnt in D-T reactors.

* very conservative figure because of low demand. (From
Ref. 33)

Fig. 1 — Cumulative U_3O_8 Requirements.

SODIUM FAST SYSTEM

GAS FAST SYSTEM

FIGURE 2. Fast breeder reactor systems

Fig. 3 — Fissile Plutonium Recovered from Light Water Reactors.

CORE PLAN

- STANDARD FUEL ELEMENTS
- CONTROL FUEL ELEMENTS
- BLANKET ELEMENTS

FIGURE 4. Plan view of 300 MWe GCFR core

FIGURE 5. Swelling of stainless steel. [Ref. 10]

DEPENDENCE OF VOLUME EXPANSION AT A CONSTANT FLUENCE ON IRRADIATION TEMPERATURE (EXPERIMENTAL DATA - AUSTENITIC STAINLESS STEELS)

FIGURE 6

FIGURE 7. The Enrico Fermi fast breeder reactor has an offset fuel handling machine permanently installed within the reactor vessel. (Courtesy "Solid Fuel Reactors," by J. R. Dietrich and W. H. Zinn, published for the USAEC by the Addison-Wesley Publishing Co., Inc., Reading, Mass., 1958.)

FIGURE 8. PFR Primary Circuit [Ref. 20]

FIGURE 9. Sodium Breeder Reactor Plant. [Ref. 17]

FIGURE 10. Nuclear steam supply of 300 MWe GCFR