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U.S.-DOE FUSION-BREEDER PROGRAM:
BLANKET DESIGN AND SYSTEM PERFORMANCE

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U.S.-DOE FUSION BREEDER PROGRAM--

BLANKET DESIGN AND SYSTEM PERFORMANCE

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ABSTRACT

Conceptual design studies are being used to assess the technical and economic feasibility of fusion's potential to produce fissile fuel. A reference design of a fission-suppressed blanket using conventional materials is under development. Theoretically, a fusion breeder that incorporates this fusion-suppressed blanket surrounding a 3000-MW tandem mirror fusion core produces its own tritium plus 5600 kg of ^{233}U per year. The ^{233}U could then provide fissile makeup for 21 GWe of light-water reactor (LWR) power using a denatured thorium fuel cycle with full recycle. This is 16 times the net electric power produced by the fusion breeder (1.3 GWe). The

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cost of electricity from this fusion-fission system is estimated to be only 23% higher than the cost from LWRs that have makeup from U_3O_8 at present costs (55 \$/kg). Nuclear performance, magnetohydrodynamics (MHD), radiation effects, and other issues concerning the fission-suppressed blanket are summarized, as are some of the present and future objectives of the fusion-breeder program.

Introduction

Fusion research and development is a major world-wide program funded at about \$1.5 billion per year. The objective of this research is to develop fusion science and technology to a point where we can use fusion--the process that powers the universe--directly to help solve the world's long-term energy problems. Most of the current research is focused on developing fusion-electric power plants. However, this paper discusses an alternate application for which fusion appears uniquely suited, namely, the production of fissile fuel by transmutation of the world's abundant fertile resources. For example, ^{233}U could be produced from ^{232}Th. Given an assured long-term supply of fissile material, thermal fission reactors (LWRs, CANDUs, HTGRs, etc.) with fission breeders could become a long-term energy option.

The fusion breeder program sponsored by the United States Department of Energy's Office of Fusion Energy is assessing the technical and economic feasibility of this potential use for fusion by developing conceptual designs and evaluating design's performance. This paper gives the recent results of this program plus the basic motivation behind its inception, and includes some history, present work, and plans for future work.

Motivation

The basic motive for developing the fusion breeder is to provide for an assured supply of fissile fuel so that fission can grow unemcumbered by the threat of ^{235}U resource limitations, cartels, etc. Thus, the fusion breeder program could be considered an insurance policy to protect the world's financial and technical investment in fission. Work on the fusion breeder is based on two fundamental assumptions: first, that fission will attain a significant share of the world's energy mix and second, that fusion research and development will be successful.

A uranium shortfall is a very real possibility in the future when nuclear fission is called upon to provide a large fraction of the world's energy needs. Fusion has the potential to alleviate this shortfall because fusion reactions are a prolific source of high energy neutrons [D + T → He (3.5 MeV) + n(14 MeV)]. In addition to producing the required tritium, these fast neutrons could produce fissile materials by using the excess neutrons produced by neutron-multiplying reactions such as n,2n reactions. The fusion breeder has also been called the fusion-fission hybrid and the fusion-fission fuel factory.

The ratio of fissile material to energy production in the fusion breeder can be widely varied. When optimized to emphasize fissile production by using suppressed-fission blankets, the fusion breeder would replace the uranium mining and enrichment segments of the fission infrastructure but leave the rest intact. In this scenario the nuclear utilities could continue to purchase their fuel much like they do today for their current and future generations of reactors. One fusion breeder can fuel more than 10 times the power than it produces. Thus, these breeders could be owned and/or operated by a separate entity, like the enrichment plants are today. Fuel cycle centers composed of fusion breeders and reprocessing and refabrication facilities could provide total fuel cycle services to the nuclear utilities.

Whereas the fusion breeder can help make fission a large-scale long-term energy source, it might also help in the development of fusion by providing a nearer-term application for fusion before fusion alone might be economically feasible.

Our studies indicate that the marriage of fusion and fission by the fusion breeder could result in a superior energy source, in terms of both cost and deployment, than either could on its own.

Project Goals

The ultimate goal of the Fusion Breeder Program is to develop and demonstrate the specific breeder technology needed so that the fusion process can be used to produce commercial fissile fuels at a time in the future when uranium becomes expensive and when fusion development produces a practical neutron source. The technologies that must complement fusion technology in order to produce fissile fuel involve the breeder blanket and fuel cycle. In the blanket, fusion-produced neutrons transmute fertile materials into fusile and fissile materials. Fuel cycle technologies include the separation of these fissile and fusile materials from the fertile materials, fuel fabrication, and waste management.

Our objectives at this early stage of the program are to develop and study conceptual blanket designs to improve our understanding and to expose pitfalls on paper as the designs evolve. A modest effort now on design studies and on small-scale generic experiments should save much time and money later when large-scale hardware development is begun by eliminating on paper any technological "blind alleys."

Therefore, we are currently developing and assessing fusion breeder designs based on both mainline tandem mirrors and tokamak approaches to magnetic fusion.

Project Organization

The fusion breeder project team is multidisciplinary. The participating organizations and their principal roles are listed below.

Organization	Principal Roles
Lawrence Livermore National Laboratory	Program manager, tandem mirror physics and technology, nuclear design
TRW, Inc.	Technical integration, tandem mirror plasma engineering, reactor systems modeling, design support
GA Technologies, Inc.	Fluid mechanics and heat transfer, fuel management systems, reactor safety systems, fuel reprocessing
Westinghouse Electric Company	Mechanical design, operation and maintenance, reactor system layout
Oak Ridge National Laboratory	Chemical engineering and materials
Princeton Plasma Physics Laboratory	Tokamak plasma engineering and technology
Idaho National Engineering Laboratory	Radiation damage and nuclear testing

In addition, investigators from the University of California at Los Angeles (radiation damage), and the Energy Technology

Engineering Center (liquid metals and materials) are participating in the study.

History

The fusion-fission concept has been slowly evolving for about 30 years. Progress in the controlled nuclear fusion experimental program and recent innovations in blanket designs have renewed interest in the attractive potential for this concept.

The suppressed-fission blanket has emerged as a possibly superior path towards achieving a breeder that maximizes the number of client fission reactors (e.g., LWRs) that can be supported. The fission-suppressed blanket option is a less challenging technological goal than fast-fission blankets because it has superior reactor safety and institutional characteristics. Reactor safety is improved because it has a low fission rate (< 0.05 per fusion); institutional advantages result because a high support ratio fusion breeder would provide makeup fuel to more than 10 1-GWe-client LWRs, while producing only about 1 GWe locally.

More detailed discussions of the motives and history are given in Refs. 1-11.

FY82 Reference Design

Our major effort in 1982 was the conceptual design of a reference fusion breeder.[12] The fusion driver for the FY82 reference fusion breeder is a tandem mirror similar to that being developed for a reference fusion electric reactor in the ongoing Mirror Advanced Reactor Study (MARS).[13] For the breeder application, the level of fusion technology is somewhat, but not significantly, lower than that called for by MARS. The plasma energy gain Q is 19 compared to 35 for MARS.

The FY82 reference fusion breeder uses a fission-suppressed blanket in which non-fission nuclear reactions [Be(n,2n) and ^{7}Li(n,n'T)] are used to generate the excess neutrons needed for net breeding. We are concentrating on this class of blankets because its afterheat is much lower and its specific fuel production (kg-fissile per MWy-nuclear) is much higher than the fast-fission class of blankets that use ^{238}U or ^{232}Th fission induced by 14-MeV neutrons to generate the excess neutrons. The lower afterheat should lead to simpler, less risky designs, whereas the higher specific fuel production should result in more attractive deployment scenarios because fuel is the principal product, not energy.

The fission-suppressed fusion breeder can fuel (support) 12 to 15 LWRs of equal thermal power, where a uranium fast-fission fusion breeder supports only about five LWRs. Fission-suppressed fusion breeders require fusion drivers with powers approaching those for fusion electric. There is an unresolved debate about which class of blanket is best: fission-suppressed or fissioning. Comparison studies are needed so a more knowledgeable distinction can be made.

The reference fission-suppressed blanket design is based as much as possible on conventional or near-term materials and process technologies, namely, LMFBR liquid metals technologies and thorex fuel processing. We are also pursuing a simple form of pyrochemical processing that should cut the reprocessing cost by an order of magnitude. In terms of performance, technological development requirements, and risk, this design could be classified as "moderate technology." For comparison, a "low technology" blanket could be a low temperature ($\sim 100^{o}C$) water-cooled design using low-stressed well-understood materials that produce fuel but no power, while a "high technology" case might be based on molten salt breeder reactor technology in which a fertile molten salt is used for on-line fueling and processing.

The reference blanket design is based on the use of a liquid lithium coolant flowing radially through a two-zone packed bed of composite beryllium/thorium pebbles. The design (shown in Fig. 1) consists of 50 4-meter-long modules. They have a ferritic steel structure (i.e., HT-9 or similar) and operate in the 350 to 500°C temperature range. The coolant flows to the first wall plenum through a thin coolant annulus and is distributed to the packed bed through perforations in a corrugated intermediate wall which, in combination with a corrugated first wall and radial stiffeners (tied to the back of the blanket), provides structural support.

The coolant flows radially outward through two fuel zones (separated by another perforated wall), exits the bed through a third perforated wall outside of the second fuel zone into a 30-cm-thick lithium plenum, and then exits the blanket through 20 large outlet pipes. The composite fuel pebbles (3-cm-diameter beryllium pebbles with thorium snap-rings) are loaded into the top of the blanket and discharged at the bottom in a frequent batch process (i.e., fuel residence time about 3 to 6 months).

The reference blanket concept offers several potentially attractive design and performance features:

- High breeding performance per unit of thermal power production.
- Refueling without disassembly.
- Low decay afterheat and excellent provision for cooling in the event of a loss of coolant or coolant flow accident.

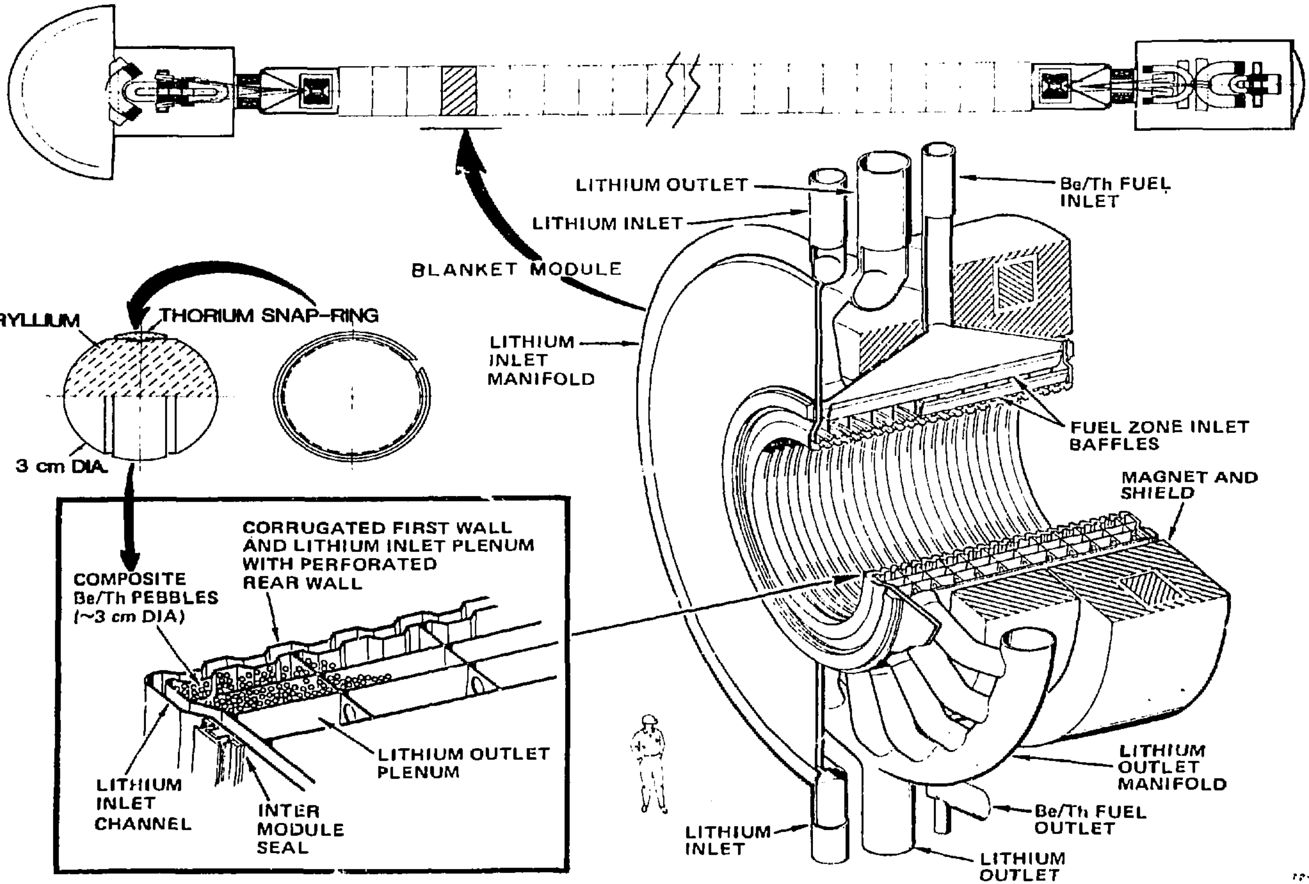

Fig. 1 Reference fusion breeder

- A beryllium multiplier form that can be easily fabricated and readily recycled.
- The extensive use of conventional materials and coolant technologies.

Nuclear performance is good for two reasons. First, the design features a high volume fraction of high efficiency neutron multipliers. The bed consists of 55% beryllium, 40% lithium, and 3% thorium. Second, the design effectively suppresses the fissioning in the blanket (< 0.04 fission per fusion neutron at 0.5% ^{233}U concentration in thorium). Fast fissions are suppressed as a result of neutron moderation in the beryllium and low thorium volume fraction. Thermal and epithermal fissions in the bred ^{233}U are suppressed as a result of fuel discharge at low concentration (< 1%) in the small volume of thorium as well as thermal neutron suppression by the ^{6}Li in the liquid lithium coolant.

As a result of the low fission rate, the fission product inventories and decay afterheat levels in the fuel are low. In fact, the fission product decay afterheat is a relatively minor contribution to the total afterheat. The afterheat associated with actinide decay through the chain

$$n + {}^{232}Th \rightarrow {}^{233}Th \xrightarrow[23\ m]{\beta^-} {}^{233}Pa \xrightarrow[27\ d]{\beta^-} {}^{233}U$$

dominates the overall afterheat level, as shown in Fig. 2. Total afterheat at shutdown is ∿ 5% of operating power. Typical fission product levels in the discharge fuel are only about 1000 ppm in thorium, or roughly 1/60 that of LWR discharge fuel. These advantages are uniquely associated with fission-suppressed blankets because fast fission blankets, with blanket energy multiplications of 6 to 10, increase the fission rate by factors of 10 to 20.

Additional reactor safety benefits for the reference design result from the use of a mobile fuel form (i.e., the composite beryllium/thorium pebbles) with provision to discharge the fuel to an independently cooled dump tank should the need arise. In addition to the primary coolant loop and the dump tank loop, the fuel handling system piping and valving provides enough coolant flow to remove the decay afterheat. Therefore, double redundancy of the cooling systems is provided in the event of a loss of coolant or loss of coolant flow accident.

The composite beryllium/thorium pebble fuel form used in the reference design provides several advantages. First, it provides a relatively simple method for achieving uniform mixing of the beryllium and thorium throughout the blanket—an advantage with respect to the thermal and nuclear breeding

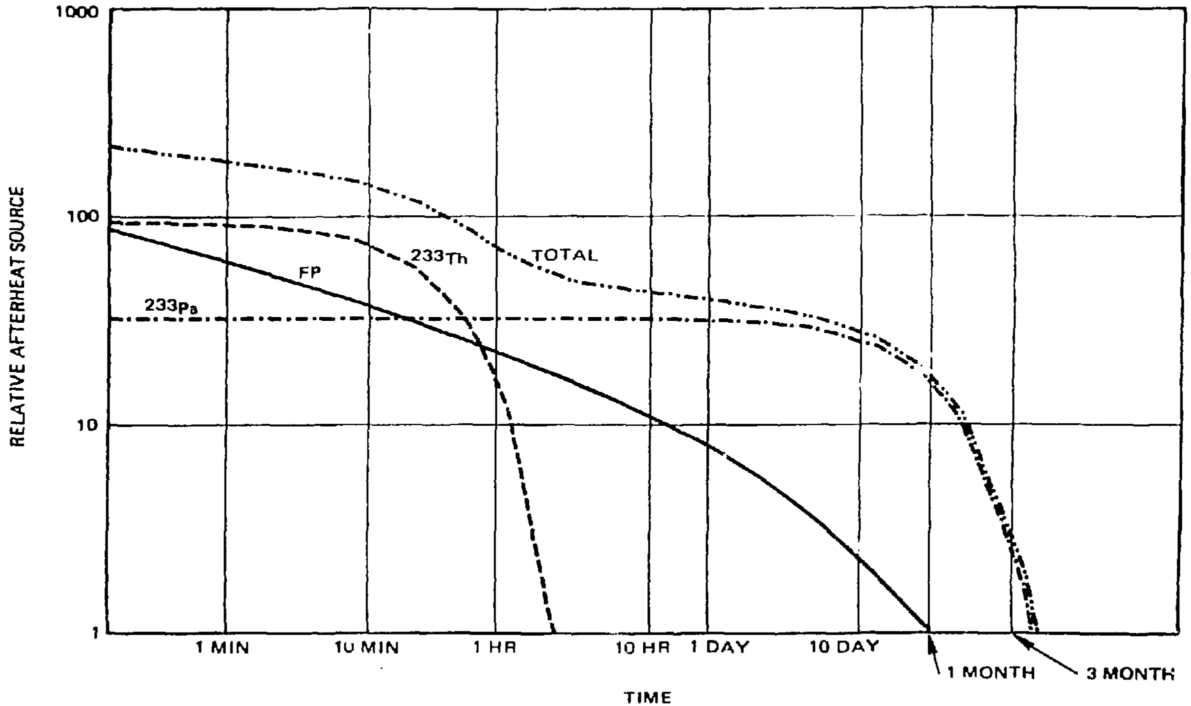

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Fig. 2 Relative afterheat sources for the reference blanket performance. Second, the design is relatively insensitive to the high rate of volumetric swelling in beryllium because the beryllium is discharged frequently and the packing density of the bed, although high, is low enough to accommodate some growth (typically 0.2% linear growth occurs over a 90-day irradiation). Finally, the small size of the pebbles (1.5-cm radius) limits the thermal- and differential-swelling-induced stress levels in the beryllium—key lifetime determinants. Our results indicate that an average beryllium in-core lifetime in excess of two years should be easily achievable, but more materials data and more accurate models are required before a more definitive lifetime estimate will be possible. The reference blanket provides a flexible design that can accommodate a wide variation in the irradiated properties of beryllium without imposing a substantial penalty on the overall level of performance.

Finally, the reference design uses conventional and well-known materials and coolant technologies. Our selection of ferritic steels was based on their irradiated and unirradiated properties (e.g., high strength, high thermal conductivity, low neutron swelling, excellent liquid metal compatibility), as well as the extensive industrial experience in the fabrication of components from ferritics (principally 2-1/4 Cr - 1 Mo) and the current interest of the nuclear materials community in these alloys.

Our choice of liquid lithium as the blanket coolant was primarily derived from nuclear, heat transfer, and tritium extraction advantages. We also considered the operational and safety implications of liquid lithium vs the obvious alternative, $Li_{17}Pb_{83}$. It is our considered opinion that liquid lithium systems can be designed to operate more economically and more reliably than lead-lithium systems, and they will have the advantage of lower normal tritium releases. An acceptable level of lithium safety appears to be achievable based on the development of liquid sodium coolant safety systems in the LMFBR program. The recognition that fusion breeder reactors would not, most likely, be sited near population centers but rather in remote, safeguarded, fuel cycle centers provides additional confidence in the choice of a liquid lithium coolant.

Our choice of thorium metal as a fertile fuel form rather than thorium dioxide (thoria) or another thorium form is primarily based on fuel cycle considerations. Although thorium oxide would provide fewer compatibility concerns, thorium metal is less expensive to reprocess (either aqueous or pyrochemical) and is more amenable to the selected fuel form. Key design and performance parameters for the reference fusion breeder are listed in Table 1.

Table 1 Key design and performance parameters for the reference fusion breeder.

Output (net)	
^{233}U @ 70% capacity factor (kg/yr)	5600
Electric power, net	
minimum (MW)	970
average (MW)	1320
maximum (MW)	1660
Thermal efficiency, average (%)	29
Central cell parameters	
Central cell fusion power (MW)	3000
Central cell fusion neutron power (MW)	2400
Maximum blanket thermal power (MW)	4728
Average blanket thermal power (MW)	3864
Central cell length (m)	200
Number of blanket modules	50
Number of central cell coils	50
Central cell coil B field strength on axis(T)	4.2
Blanket first wall radius (m)	1.5
Wall loading, P_n/first wall area (MW/m^2)	1.3
Plasma radius (m)	0.58
Blanket module mechanical design	
Structural material	HT-9 ferritic steel
Module length (m)	4
Inter-module vacuum seal arrangement	metal omega seal
Number of pebble bed zones	2
Pebble bed volume fractions	
Beryllium (%)	55
Lithium, 0.2 a/o vol/(%) ^{6}Li (%)	40
Thorium, incl. bred fissile (%)	3
Ferritic steel (%)	2
Thickness of each pebble bed (cm)	20
Lithium reflector thickness (cm)	30
Blanket outer radius (m)	2.34
Shield thickness (cm)	75
Magnet inner bore (m)	6.7
Magnet pitch (m)	4

Table 1 (Continued.)

Nuclear design parameters

Net fissile breeding ratio	0.62
Net tritium breeding ratio	1.06
Minimum blanket energy multiplication	1.25
Maximum blanket energy multiplication	1.97
Maximum thorium power density (W/cm^3)	182
Maximum beryllium power density (W/cm^3)	5.4
Maximum lithium power density (W/cm^3)	3.3
Maximum average power density (W/cm^3)	7.7
Zone 1 fuel residence time (days)	78
Zone 1 uranium discharge concentration (%)	0.86
Zone 1 protactinium discharge concentration (%)	0.53
Zone 2 fuel residence time (days)	156
Zone 2 uranium discharge concentration (%)	0.74
Zone 2 protactinium discharge concentration (%)	0.20
Average fission rate per fusion	∿0.04
Average burnup at fuel discharge (MWD/MT)	∿500

Blanket module heat transfer and thermal design parameters

Coolant inlet temperature (°C)	340
Coolant outlet temperature (°C)	490
Lithium flow rate (m^3/s)	0.31
Lithium pressure drop[a] (MPa)	∿2 (300 psi)
Lithium pump power all modules[a] (MW)	∿35
First wall pressure[a] (MPa)	∿1.7 (250 psi)
Minimum first wall temperature(°C)	361
Maximum first wall temperature(°C)	409
Maximum structure temperature(°C)	490
Maximum beryllium surface temperature(°C)	475
Maximum beryllium internal temperature(°C)	<483
Maximum beryllium ΔT(°C)	< 38
Thermal conversion efficiency, net (%)	37

[a]Actual values are likely to be a factor of 2 lower.

Technical Issues

Although the blanket design appears feasible, the following performance issues must be resolved by further study and experimentation:

- Nuclear performance--Uncertainty in beryllium neutron multiplication must be reduced and better methods must be used to account for the blanket's heterogeneity. Analysis is underway and experiments are being planned to reduce these uncertainties.
- Material compatibility--The initial results from static capsules containing beryllium, lithium, and steel support our choice of materials, but more static and dynamic tests are required to improve confidence.
- Radiation damage--Swelling and loss of ductility will limit the lifetime of the beryllium pebbles and the blanket structure. Tests with beryllium samples irradiated in EBRII at $425^{\circ}C$ to 10^{22} n/cm^2 ($E > 1$ MeV) are showing good ductility, namely, 15% at $450^{\circ}C$ increasing to 30% after annealing. This should lead to longer beryllium pebble lifetime than we have assumed. Radiation effects on structures is a major issue for all fusion electric blankets, but is less of a concern for the

fusion breeder with only a 1.3-MW/m^3 wall load. We are developing designs that are especially tolerant of swelling.

- Magnetohydodynamics—The pressure required to flow lithium through the blanket, especially the pebble bed in a 4-T magnet field, is uncertain; thus, the structural requirement of the blanket and its effect on breeding are uncertain. Analysis is underway and experiments are being planned to reduce this uncertainty.
- Beryllium/thorium fuel fabrication—The beryllium industry will have to be expanded significantly to produce the beryllium metal for one reference blanket (about 900 tonnes) per year. Also, a low-cost remote method must be developed to fabricate and refabricate the beryllium/thorium fuel pebbles. Neither of these appear to pose major problems.
- Beryllium resource—The 900 tonnes of beryllium per blanket are about 1.5% of the known U.S. deposits (60,000 tonnes), about 0.36% of undiscovered U.S. deposits (250,000 tonnes), and about 0.12% of the known plus undiscovered world deposits (740,000 tonnes).[6] Beryllium "burnup" is very small, only about 0.02% per year. Beryllium loss during refabrication will be the principal beryllium loss mechanism. For the reference

case, a pessimistic estimate based on only a two-year life and current fabrication practices (7% loss) would give a yearly loss of 3.5%. Reducing this loss rate at least an order of magnitude by using better fabrication practices and longer service life is considered straightforward. At a loss rate of 0.35% per year, the world's estimated beryllium resource could support 240,000 fusion breeder-years of operation, which in turn could support on the order of 3,000,000 GWe-years of fission reactor energy. The beryllium resource appears more than sufficient to support a large-scale long term fusion breeder deployment.

Economic Analysis

The economics of the fusion breeder are evaluated by examining a symbiotic fusion-breeder/fission-burner electricity generation system. In this concept, the fusion breeder is typically incorporated into a remotely sited and safeguarded fuel cycle complex along with fuel reprocessing plants, fuel fabrication facilities, and possibly a waste disposal facility.

The following quantities are used for both the fusion breeder and the client fission burners to develop a consistent estimate of the symbiotic cost of electricity:

- Fixed capital costs.

- Variable operating costs.
- Fissile fuel production and consumption.
- Fissile fuel inventories.
- Net thermal-to-electric conversion efficiencies.

Summaries of these data are given in Tables 2 through 6. We use this information to estimate the year-by-year costs of electricity as well as a transfer price for the bred fissile fuel of the fusion breeder and its fission reactor clients. These year-by-year costs are also combined to provide average present value costs. The breeder's net specific fissile production rate of 1.81 kg/MW_n-yr (Table 2) is 14.4 times the net specific fissile consumption rate of its client LWRs on the denatured thorium fuel cycle. Thus, 21 1000-MWe LWRs can be supported by one reference fusion breeder.

Table 3 gives the total extimated capital cost for the fusion breeder and its reprocessing (thorex) and refabrication (thorium and beryllium) facilities, including indirect and time-related costs based on LWR construction experience. An LWR, when costed consistently, has a specific capital cost of 540 \$/kW_n (Table 5), so the fusion breeder cost of 1826 \$/kW_n is 3.4 times the LWR cost on this basis.

This fusion breeder vs LWR comparison can be lumped into a simple and convenient figure of merit for the symbiotic system.

Table 2 Reference fusion breeder performance parameters.

Fusion power (MW_f)	3000
Average blanket energy multiplication (M)[a]	1.61
Average total nuclear power (MW_n)[a]	4464
Average net electrical power (MWe)[a]	1317
Average net nuclear-to-electrical efficiency[a]	0.295
Net fissile fuel production rate (kg/yr)[b]	5646
Net specific fissile fuel production rate (kg/MW_n-yr)[c]	1.81
In-core fissile inventory (kg)	1032
Ex-core fissile inventory (kg)[b]	2874
Total fissile inventory (kg)[b]	3816
Specific fissile inventory (kg/MW_n)	0.854
Average fissile discharge enrichment (%)[a]	1.24
Heavy metal throughput, MT/yr thorium[b]	604
Average plant capacity factor (%)	70

[a]Average over blanket operation.

[b]At 70% capacity factor.

[c]Value for full power operation.

Table 3 Summary of fusion breeder fixed charges.

Direct cost ($M)	3744
Indirect cost ($M)	3179
Time-related cost ($M)	1232
Total capital cost 1982 $ ($M)	8155
Specific cost ($/$kW_n$)	1826
Total annual charge at 18.04%/yr ($/$kW_n$-yr)	329

Table 4 Summary of fusion breeder variable charges in year zero

Fuel cycle operating cost ($M/yr)	89
Blanket structure replacement cost ($M/yr)	2
Miscellaneous operation and maintenance cost ($M/yr)	132
Total variable (direct) charge in year zero ($M/yr)	223
Specific variable (direct) charge in year zero ($/kW$_n$-yr)	50

Table 5 Summary of LWR fixed charges

Direct cost ($M)	789
Indirect cost ($M)	585
Time-related cost ($M)	245
Total capital cost 1982 dollars ($M)[a]	1620
Specific cost ($/kW$_t$)	540
Total annual charge at 18.04%/yr ($/kW$_t$-yr)	97

[a]Basis: informal Ebasco cost estimate for 1000 MWe (3000 MW$_t$) LWR; 1982 $.

Table 6 Summary of LWR variable charges in year zero

	Fuel cycle	
	Denatured thorium	Denatured uranium
Annual fuel processing charge ($\$/kW_t$-yr)	12.36	12.02
Annual fuel reprocessing cost ($/kghM)	600	558
Annual fuel fabrication cost ($/kghM)	865	865
Annual fuel transportation cost ($/kghM)	22	22
Annual back end fuel cycle cost ($/kghM)	75	75
Annual total unit processing cost ($/kghM)	1562	1520
Annual fuel throughput (kgHM/kW_t-yr)	0.0077	0.0079
Operation and maintenance charge ($\$/kW_t$-yr)	9.16	9.16
Total year zero variable charge ($\$/kW_t$-yr[a])	21.5	21.2

[a]Includes 70% capacity factor.

This figure of merit, the capital cost ($/kWe) of the fusion breeder plus supported LWRs relative to the LWR, is defined as

$$C^{ave} = \frac{C^B + R}{\eta_{rel} + R} \quad ,$$

where C^B (3.4) is the breeder capital cost relative to the LWR cost per unit nuclear power, R (14.4) is the nuclear support ratio, and η_{rel} is the electrical efficiency of the breeder relative to an LWR (0.295/0.33 = 0.89). So, in the reference case we obtain an average capital cost ratio of

$$C^{ave} = \frac{3.4 + 14.4}{0.89 + 14.4} = 1.16 \quad .$$

In other words, for a 16% capital cost increase, the LWRs, would have an assured fuel supply.

In the more detailed analysis, summarized in Table 7, the average present value of the cost of electricity for the symbiotic system is 38.7 mil/kWeh (1982 dollars). The symbiotic cost of electricity is only 11% higher when we compare it to a consistent calculation for the cost of electricity from an LWR fueled with conventionally mined uranium costing 55 $/kg in the first year (with 3% real escalation above 7% inflation in each of the succeeding years). In the case of the conventional LWR, we assumed full reprocessing and fissile recycling of uranium and plutonium. In the first year of operation the electricity cost is 23% higher.

This result indicates that if fusion breeders were introduced into the nuclear economy, they could preserve the economic viability of the LWR by putting an upper limit on the cost of fissile fuel.

Sensitivity Studies and Comparison with Fusion Alone

Here we present sensitivity studies that indicate the change in the present value of the symbiotic cost of electricity as a function of the uncertainties in the cost of the fusion breeder, or the fusion power gain (Q).

Table 7 Summary of results for baseline economics analysis.

Year zero bred fissile fuel cost[a] (\$/g)	198
Ave. present value of bred fissile fuel (\$/g)	93
Year zero symbiotic electricity cost (mil/kW_eh)	76.3
Ave. present value of symbiotic electricity cost (mil/kW_eh)	38.7
Year zero conventional electricity cost (mil/kW_eh)	61.8
Ave. present value of conventional electricity cost (mil/kW_eh)	34.8
Symbiotic/conventional cost of electricity[b]	1.11

[a]All costs in 1982 \$ (year zero is first year of operation).

[b]Basis: average present value.

The sensitivity in the electricity cost to a $\pm$ 50% change in the capital cost of the breeder corresponds as a $\pm$ 13.6% change in the system cost of electricity.

For a pure fusion electric tandem mirror that is modeled consistently using the same physics, design, and economics codes, a 50% change in the capital cost gives a 49% change in the cost of electricity. Therefore, the fusion electric case is at a disadvantage with respect to both the reference cost of electricity (71% higher than the symbiotic system) and its sensitivity to cost uncertainties.

The uncertainty in fusion gain ($\eta_{trap}Q$) affects the recirculatiing power requirement and the size and cost of the plasma heating systems. Figure 3 shows the effect of fusion gain uncertainties for both the fusion breeder and fusion electric cases. As shown, the symbiotic cost of electricity is insensitive to fusion gains above about 10 and does not increase significantly until the gain falls to about five—threefold below the predicted value. Conversely, for fusion electric generation, insensitivity occurs above 30 and very substantial increases in the cost of electricity occur for gains below about 15.

The economic effect of operating on the denatured uranium fuel cycle instead of the denatured thorium fuel cycle was also considered. The latter fuel cycle provides a 21% larger number of LWR clients, but the fissile inventory cost and fuel reprocessing cost are lower for the denatured uranium fuel cycle. As a result, the calculated difference in the electricity cost (a decrease of 0.15 mil/kWeh) is insignificant, and the two fuel cycles are equally attractive from an economics perspective. A choice between these must derive from noneconomic considerations such as the larger LWR support and a smaller fraction of plutonium burners for the denatured thorium fuel cycle. The denatured uranium fuel cycle preserves PUREX reprocessing on the LWR side of the system and

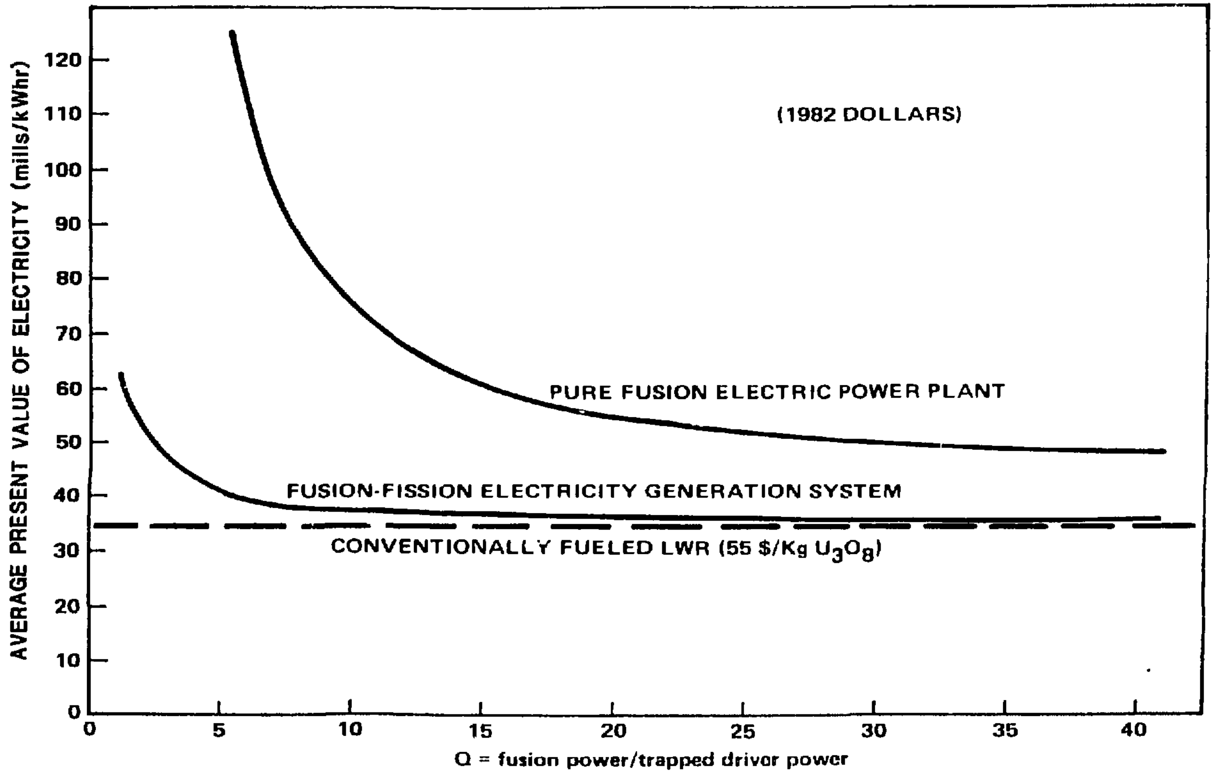

Fig. 3 The cost of electricity for fusion-fission electricity generation and fusion electric versus the fusion gain

minimizes the LWR fissile inventory per unit electrical power generation.

Present and Future Work

In addition to addressing some of the technical issues already discussed, our work this year includes:

- Tokamak fusion breeder—Adapting or developing a fission-suppressed blanket for the tokamak. Its toroidal geometry (shown in Fig. 4) and its higher magnetic field (inboard) may require significant changes, such as not having the beryllium/thorium pebbles in the inboard blanket or switching to helium cooling.
- Molten salt blanket--We are continuing to look for ways to incorporate a fertile molten salt into the blanket in a technically feasible way, to take advantage of its potential for online refueling and online low-cost reprocessing. It also needs no thorium refabrication. The basic features of a design concept we are presently pursuing are shown in Fig. 5. It has a beryllium pebble bed for neutron multiplication. The bed also has tubes containing a slowly flowing fertile molten salt such as $LiF + BeF_2 + ThF_4$. The coolant is helium and the structure and tubes are steel. The tube walls are

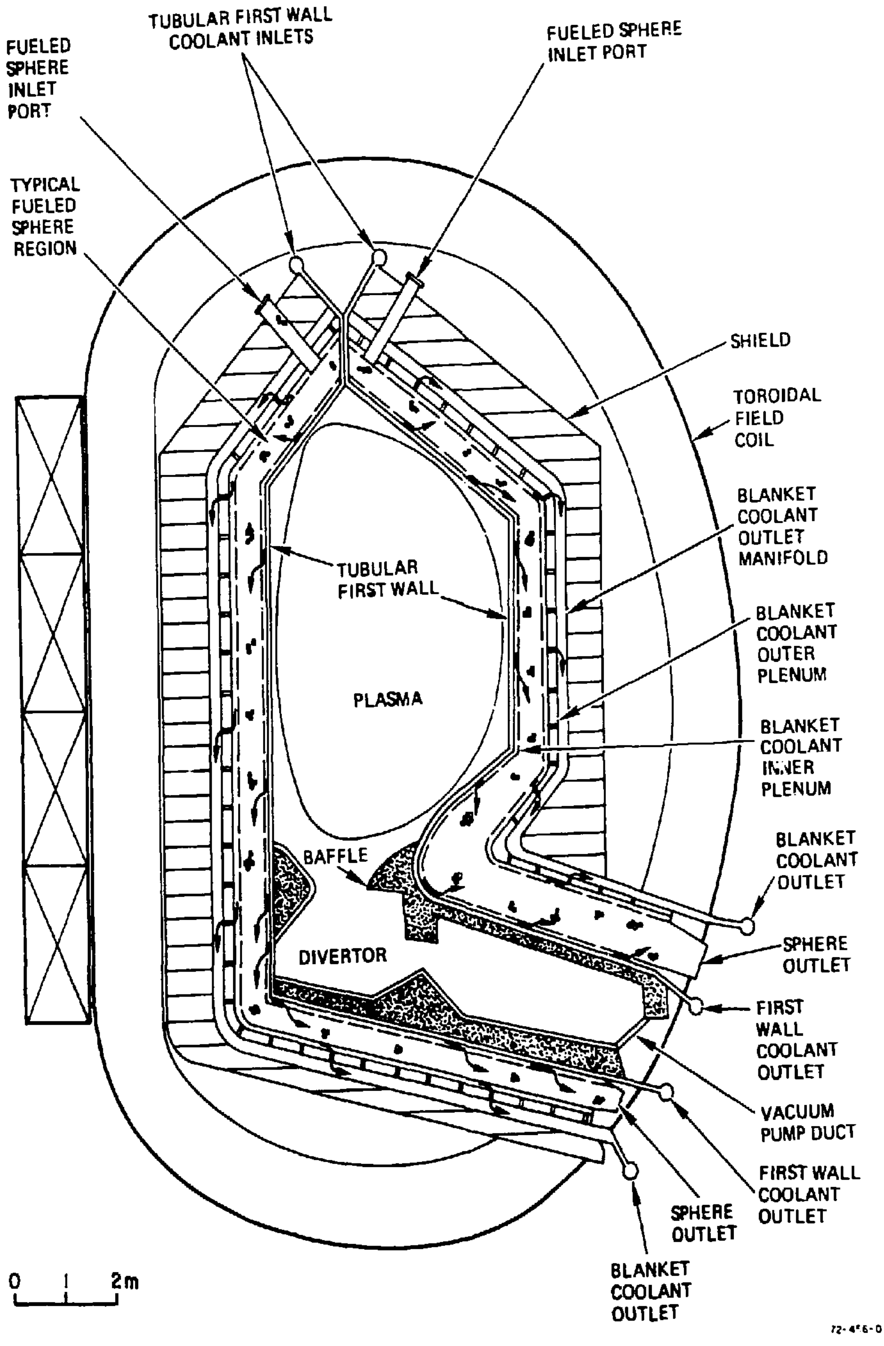

Fig. 4 Tokamak geometry

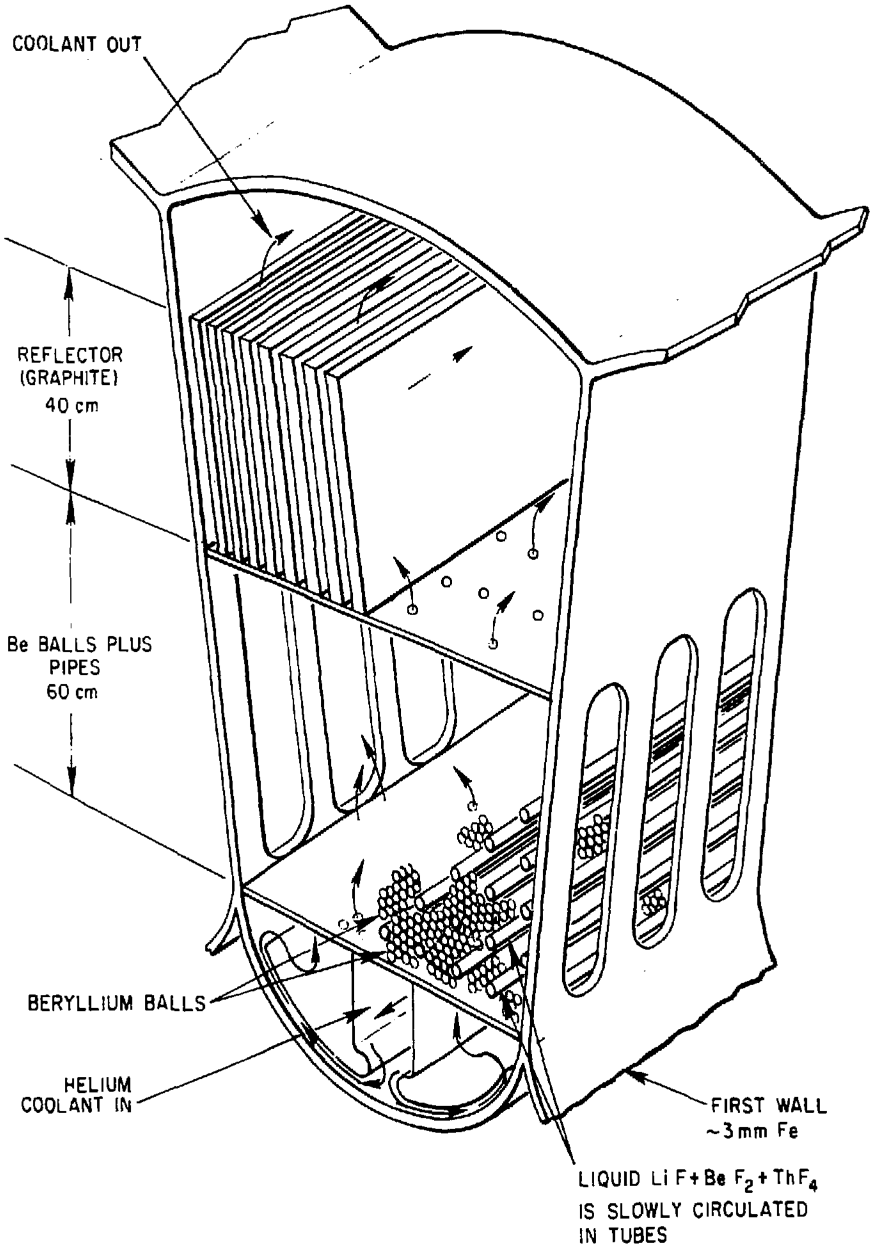

Fig. 5 Conceptual Be/molten salt blanket

maintained at about T-melt of the salt ($\sim 565^{o}C$) to minimize corrosion. We believe this design concept can overcome the materials problem encountered with our first beryllium/molten salt design (developed in 1979) and still have good performance.

- Fast fission blankets--We are applying the pebble bed concept to fast fission blankets for both the tandem mirror and tokamak fusion drivers. Our preliminary design has UC fuel and helium cooling. This is a low level effort; our objective is to develop designs with acceptable safety characteristics so we can evaluate the relative merits of fast-fission vs fission-suppressed blankets with low performance fusion drivers.
- Cold blankets--We may also investigate the tradeoffs of using low temperature blankets, probably cooled by low pressure water. The question is whether the disadvantage of not using blanket and other heat to produce electricity in offset by lower costs and higher availability. To get some idea of what might happen, let's take the reference breeder, eliminate thermal conversion, and assume this reduces capital cost by 10% and increases availability by 10%. This will also reduce the breeder's thermal efficiency to a negative 6.3% because it must now import power (280 MWe). Using the same figure of merit developed

previously, these changes would give a relative capital cost of

$$C^{ave'} = \frac{3.4(0.9) + 14.4\ (1.1)}{-0.063/0.33 + 14.4(1.1)} = 1.21\ .$$

This is only 4% higher than the reference case, which has a relative capital cost (breeder + supported LWRs relative to the LWRs alone) of 1.16. Based on this simple exercise it is possible that low technology "cold" blankets might compete with higher technology "hot" blankets. The cold blanket case may also be able to run at a higher wall loading and/or fusion power level, both of which would reduce unit costs.

Summation

Fusion has the potential to produce large quantities of fissile material, thus eliminating fission's dependence on the uranium ore and enrichment markets. Our fusion breeder program is assessing this potential.

Recent work has concentrated on developing a blanket design that uses beryllium for neutron multiplication and suppresses in situ fissions. The fission-suppressed class of blankets has attractive safety and deployment advantages. When this blanket is coupled with a 3000-MW fusion tandem mirror driver similar to the design being developed for a conceptual commercial electric plant, it produces 5600 kg per year of ^{233}U. This amount of ^{233}U can provide the fission makeup requirement (or support) for 21 GWe of LWRs on the denatured thorium ($^{233}U/^{238}U/^{232}Th$) fuel cycle with recycle. This is 16 times the breeder's average net power (1.3 GWe). The cost of power from these fusion-breeder-fueled LWRs is predicted to be 23% higher in the first year of operation than conventionally fueled LWRs with U_3O_8 at 55 $/kg. When averaged over the breeder's 30-year life, the cost increase drops to 11%. Fueling higher gain reactors such as CANDUs or HTGRs would result in support ratios of 30 to 50 and would most likely produce lower costs.

In summary, in our present work we are refining the reference design, addressing its unresolved technical issues, developing tokamak-based breeder designs, looking at the fast-fission option, and considering a new molten salt blanket design concept.

If fission is called on to meet a major fraction of the world's expanding energy demand, a uranium shortage could occur in about 20 years. The goal of the fusion breeder project is to develop the technology so fusion will be available to eliminate the shortage. It could also be an earlier, economically competetive application for fusion.

Acknowledgement

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