FUSION BREEDER REACTOR DESIGN STUDIES*

R. W. MOIR, J. D. LEE, M. S. COOPS, F. J. FULTON, W. S. NEEF, JR.
Lawrence Livermore National Laboratory, P.O. Box 5511, L-644
Livermore, CA 94550
(415) 422-9808
D. H. BERWALD, R. B. CAMPBELL, B. FLANDERS, J. K. GARNER, N. GHONIEM (Consultant, UCLA)
J. OGREN, Y. SAITO, A. SLOMOVIK, R. H. WHITLEY, TRW, Inc.
One Space Park
Redondo Beach, CA 90278
K. R. SCHULTZ, G. E. BENEDICT, E. T. CHENG, R. L. CREEDON
I. MAYA, V. H. PIERCE, J. B. STRAND, C. P. C. WONG, GA Technologies, Inc.
P.O. Box 81608
San Diego, CA 92138
J. S. KARBOWSKI, R. P. ROSE, Westinghouse Electric Corp.
Advanced Energy Systems Division, Fusion Engineering Dept., P.O. Box 158
Madison, PA 15663
J. H. DEVAN, P. TORTORELLI, Oak Ridge National Lab
P.O. Box X, Mail Stop 4500S
Oak Ridge, TN 37830
L. G. MILLER, P. Y. S. HSU, J. M. BEESTON, Idaho National Engineering Lab (INEL-EG&G)
P.O. Box 1625
Idaho Falls, ID 83415
N. J. HOFFMAN, Energy Technology Engineering Center
P.O. Box 1449, 8900 De Sota Ave.
Canoga Park, CA 91304
D. L. JASSBY, Princeton Plasma Physics Laboratory
P.O. Box 451
Princeton, NJ 08540

ABSTRACT

Studies of the technical and economic feasibility of producing fissile fuel in tandem mirrors and in tokamaks for use in fission reactors are presented. Fission-suppressed fusion breeders promise unusually good safety features and can provide make-up fuel for 11 to 18 LWRs of equal nuclear power depending on the fuel cycle. The increased revenues from sales of both electricity and fissile material might allow the commercial application of fusion technology significantly earlier than would be possible with electricity production from fusion alone. Fast-fission designs might allow a fusion reactor with a smaller fusion power and a lower Q value to be economical and thus make this application of fusion even earlier. A demonstration reactor with a fusion power of 400 MW could produce 600 kg of fissile material per year at a capacity factor of 50%. The critical issues, for which small scale experiments are either being carried out or planned, are: 1) material compatibility, 2) beryllium feasibility, 3) MHD effects, and 4) pyrochemical reprocessing.

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INTRODUCTION

This paper will describe a reference fission suppressed blanket design,[1] and review recent fusion breeder studies. Goals and design choices will be discussed as we go along. Prior to 1979 we studied fast-fission, Pu-producing hybrids.[2] High energy multiplication allowed economical operation at lower Q values[a] of 2 to 4 depending on cost of equipment handling recirculating power. The local blanket energy multiplication doubled during the typical 4 year irradiation time (7 MW·y/m^2 integrated first wall loading) producing 2% Pu/^{238}U at discharge. The designs resulted in fuel forms, power densities, radioactive inventories, and afterheat cooling requirements much like fission reactors. Fissioning hybrids could fuel or support about 5 LWR's of equal thermal power.

In 1978, we started looking into fission-suppressed ^{233}U breeders.[3] We prefer ^{233}U over Pu because conventional fission reactors (LWR, HWR and HTGR's) utilize the former much more efficiently, and secondly, ^{233}U can be substituted for ^{235}U with little change in the present day fuel cycle. The fission-suppressed designs resulted in power densities, radioactive inventories and afterheat cooling requirements much like pure fusion reactors. Fission-suppressed hybrids could support about 15 LWR's of the same nuclear power.

BREEDING BLANKET DESIGN

We have made a number of studies of the tandem mirror as a hybrid (interchangeably called fusion breeder). In 1982, we began applying the fission-suppressed blankets to the tokamak.[4] Any blanket which works on a tokamak will probably work on a tandem mirror, but the reverse may not be true. The tandem mirror has relatively simple geometry and has a uniform, steady-state, low magnetic field (4-5 Tesla), whereas the tokamak has a complex geometry and a nonuniform field with a high value on the inside of the torus (∿10T). Fig. 1 illustrates the tandem mirror blanket geometry, and Fig. 2, the tokamak.

Fuel Form - Mobile versus Fixed

In all cases we have mobile fuel (pebbles in these two examples, but molten salt in other examples). The reason for the mobile fuel is so we can remove fissile material without blanket removal before it builds up enough to fission and cause the blanket power to increase significantly. The blankets shown in the figures have the common feature of using pebbles for the following reasons: 1 - dump fuel for safety; 2 - radiation damage consequences for small pebbles are less troublesome; 3 - fuel can be removed at low burnup to reduce power swing (in prior fast-fission designs,[2] the blanket power density doubles); and 4 - the blanket can last much longer than the fuel lifetime.

REFERENCE BLANKET - LIQUID LITHIUM COOLED PEBBLE BED

After an extensive scoping phase, a reference blanket concept based upon the use of a liquid lithium coolant flowing radially through a two zone packed bed of composite beryllium/thorium pebbles was selected. This design is described in a detailed report.[1] The design shown in Fig. 1 uses a ferritic steel (i.e., HT-9 or similar) structure and operates in the 350-450°C temperature range. In this concept, the coolant flow resembles that of a conventional oil filter. Specifically, 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, and exits the blanket through 20 large outlet pipes. The composite fuel pebbles (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 ∿ 3-6 months).

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

- high breeding performance per unit of thermal power production;
- low decay afterheat and excellent provision for cooling in the event of a loss of coolant or coolant flow accident;
- a beryllium multiplier form which can be easily fabricated and readily recycled;
- the extensive use of conventional materials and coolant technologies with a low wall loading.

[a] ($Q = P_{fusion}/P_{injected}$)

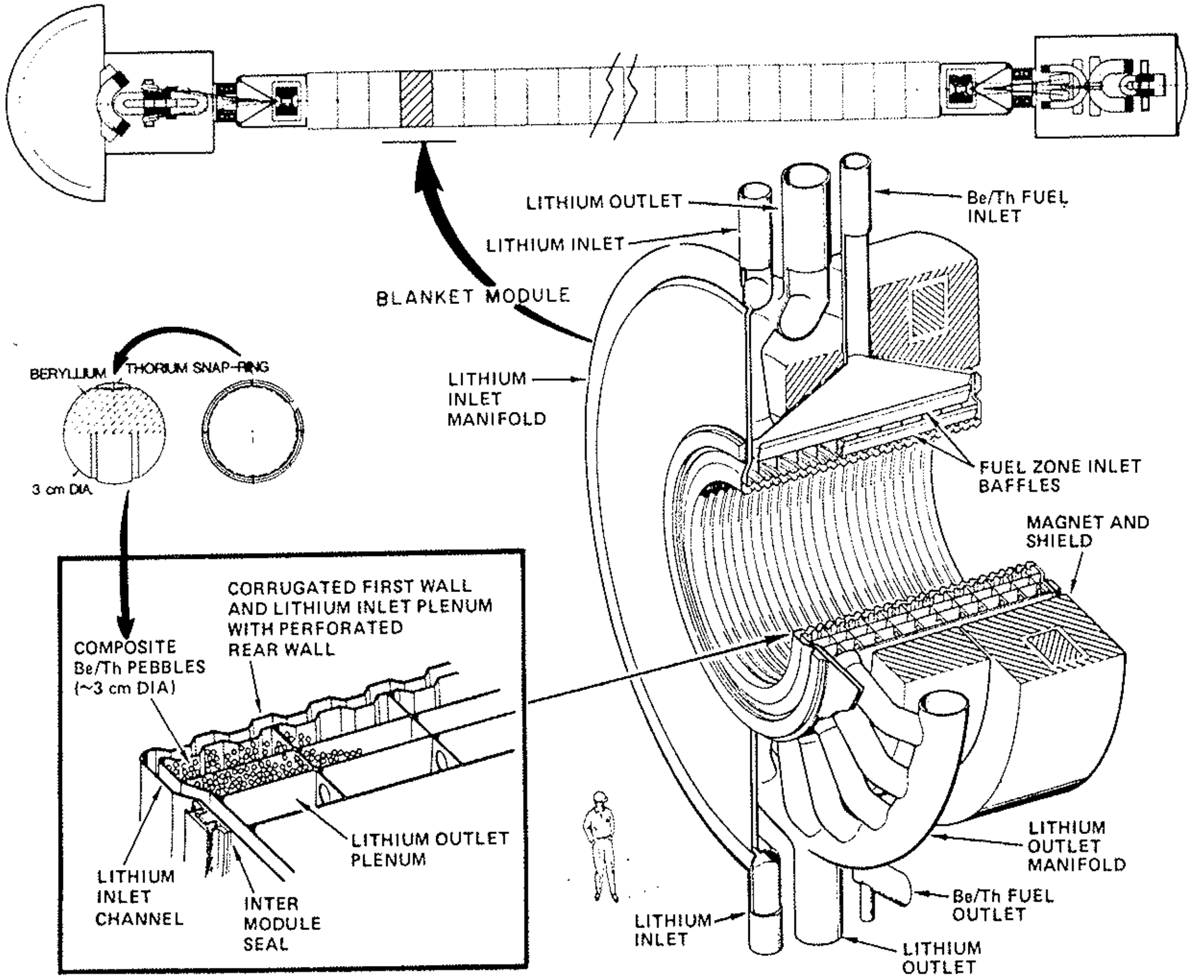

Fig. 1. Tandem Mirror Fusion Breeder Blanket

The breeding performance is good for two reasons. First, the design features a high volume fraction of high efficiency neutron multipliers. The bed volume fractions in Fig. 1 include about 55% beryllium, 40% lithium, and 3% thorium -- all excellent neutron multipliers. The remainder of the fuel region following the two corrugated walls is less than 2% steel excluding the zone separator. 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 due to neutron moderation in the beryllium and low thorium volume fraction. Thermal and epithermal fissions in the bred ^{233}U are suppressed due to fuel discharge at low concentration (< 1%) in the small volume of thorium. Also, thermal neutron depletion, due to the large 1/V neutron absorption cross section of the ^{6}Li in the liquid lithium coolant, inhibits fissioning of the bred fissile fuel.

As a result of the low fission rate, the fission product inventories and decay afterheat levels in the fuel are very low. In fact, the fission product decay afterheat is a relatively minor contribution to the total afterheat, and

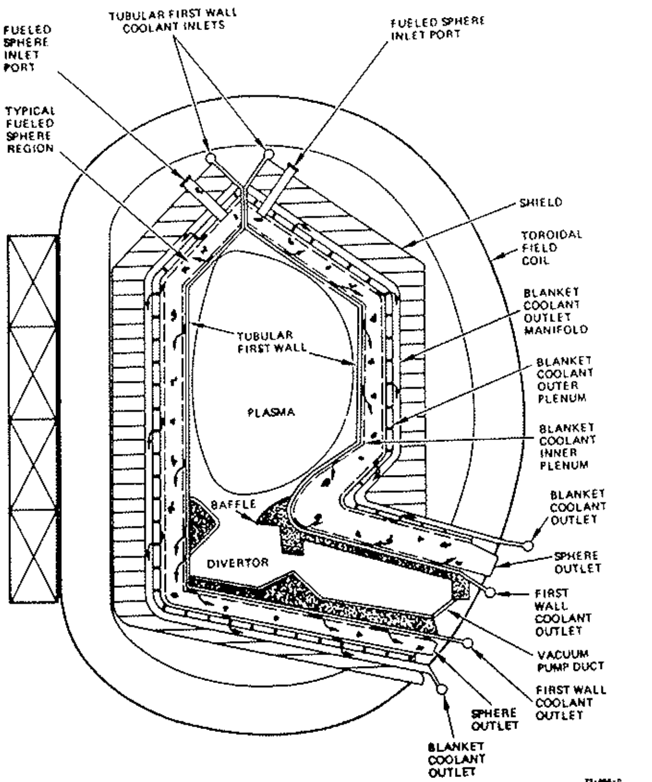

Fig. 2. Cross-section elevation view of a suppressed fission blanket concept for a tokamak reactor (not to scale).

the afterheat associated with the actinide decay chain dominates the overall afterheat level. 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 since fast-fission blankets, with blanket energy multiplications of 6-10, increase the fission rate by factors of 10-20.

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 a coolant flow sufficient 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 employed in the reference design provides several advantages in comparison with previous designs. First, this form provides a relatively simple method to achieve uniform mixing of the beryllium and thorium throughout the blanket - an advantage with respect to the thermal and nuclear breeding performance.

Second, the design is relatively insensitive to the high rate of volumetric swelling in beryllium since 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 determinates. Our results indicate that an average beryllium in-core lifetime in excess of two years should be easily achievable, but that 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 which 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 utilizes conventional and well known materials and coolant technologies. Our selection of ferritic steels was based upon their irradiated and unirradiated materials 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. Since the time integrated fluence for a 6 year blanket life is only 6 x 1.3 x 0.7 = 5.5 $MW \cdot y/m^2$, the design is conservative with respect to radiation damage.

Our choice of liquid lithium as the blanket coolant primarily derived from nuclear, heat transfer, and tritium extraction advantages, but also considered the operational and safety implications of liquid lithium versus 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 will have the advantage of lower normal tritium releases. An acceptable level of lithium safety appears to be achievable based upon 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 upon 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. There is considerable experience in the use of thorium metal in boiling water fission reactors (e.g., Indian Point BWR).

The direct cost of the plant[1] was estimated at $3744M (1982 Dollars) including $372M for beryllium and thorium fabrication and reprocessing facilities and had a peak thermal power of 5340 MW (3000 MW_{fusion}). The direct cost of an LWR was estimated[1] at $788M for 3000 MW_{th}. Thus, the plant cost 2.4 times an LWR. Typical parameters are given in Table 1.

Table 1. Reference Blanket System Parameters

P_{fusion}	3000 MW (15 MW/m)
Blanket energy multiplication	1.6 ave. (1.97 peak)
^{233}U production	5600 kg/yr (@ 70% C.F.)
Center cell length	200 m
Center cell mag. field	4.2 T
First wall radius	1.5 m (1.3 MW/m^2)
Li inlet temp.	340°C
outlet	420°C
Pressure on first wall	1.0 MPa or 150 Psi
Structural material	HT-9
Blanket thickness	0.85 m
Shield thickness	0.75 m
Net fissile breeding ratio	0.62
Peak power density	182 W/cm^3 Thorium
	5.4 W/cm^3 Be
	3.3 W/cm^3 Li
Average fission rate per fusion	0.04
Average fission burnup at fuel discharge	500 MWD/MT
Average net power	1300 MW_e
	(1660 MW_e peak)
Recirculating power	720 MW_e
Fusion power gain ($\eta_{trap}Q$)	14.6

Beryllium Pebbles

The reference design calls for beryllium and thorium at a 20 to 1 volume ratio. First we considered a mixture of balls, but experiments carried out by W. S. Neef (see p. 3-35 of ref. 1) with two masses of balls showed a severe segregation problem. To maintain uniformity, we propose using composite balls (See Fig. 1).

The disadvantages of these composite balls are the necessity of remotely removing and replacing the snap-ring, and the groove around the beryllium ball could leave it more susceptible to radiation damage.

Beryllium Lifetime

The expected lifetime of the Be pebbles, discussed more fully in ref. 5, is predicted to be greater than 2 $MW \cdot y/m^2$.

Beryllium Radiation Damage Experiments

Beryllium samples irradiated in EBR-II for about 5 years to a fluence of 1×10^{22} n/cm^2 (E > 1 MeV) at a temperature of 450 °C have been analyzed.[6] The samples showed a swelling and decrease in ductility, but at an equivalent of about 2 $MW \cdot y/m^2$ they maintained their integrity, leading us to believe that we can get enough lifetime before remanufacturing is necessary to be economically viable. Prototypical irradiations of beryllium should be carried out at high temperatures with fission neutrons and high energy neutrons for example from FMIT (Fusion Materials Irradiation Test Facility) or a fusion engineering test facility such as the proposed TDF.[7]

Beryllium Material Compatibility

Beryllium will attack many other metals by interdiffusion and forming compounds. The rate is highly temperature dependent. Experiments carried out at ORNL have shown that beryllium interacts with elemental nickel in 316 SS at temperatures as low as 450°C in static lithium. Non-nickel-bearing ferritic steels are expected to afford acceptable compatibility with beryllium at temperatures below 500°C and tests are being conducted to measure the extent of interaction of beryllium with Cr-Mo steels in both lithium and sodium. Thorium will also be included in these ongoing compatibility experiments. This work also discusses the use of coatings for inhibiting mass transport and weight loss. This subject is discussed in refs. 1 and 8.

ALTERNATIVE BLANKET COOLANTS

Helium Cooling. Ability to keep radioactive contaminants at a low level is the big virtue of helium coolant. Corrosion is only by impurities and neutronics are unaffected by the low density of helium. The large film temperature drop and low heat capacity are disadvantages. The high pressure, about 50 atmospheres, is a disadvantage in that it leads to increased structural material which hurts the neutron economony. The blankets shown in Figs. 1 and 2 can be modified to be helium cooled using the naturally tensioned skin containment structure shown in Fig. 3, which was adapted from ref. 9.

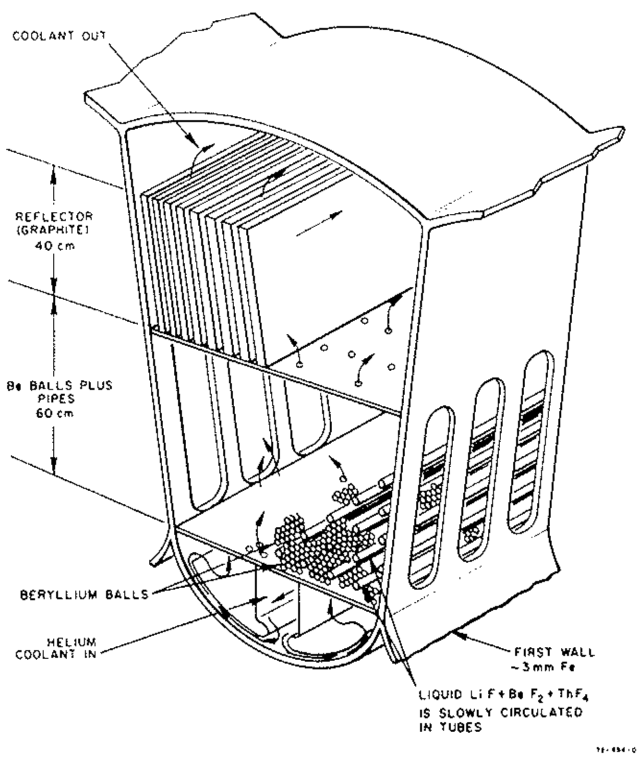

Fig. 3. Helium cooled blanket. This blanket will work in either the Tandem Mirror geometry (Fig. 1) or the Tokamak geometry (Fig. 2). This figure also shows the breeding concept using molten salt which slowly circulates in the pipes.

Water Cooling. Water, if used as a coolant, must occupy less than 10% of the volume of a blanket to minimize moderation of neutrons before neutron multiplication reactions occur. If the water temperature is kept low (< 100°C), good breeding occurs because little structural material is needed to contain the water, and tritium diffusion into the water can be kept small. Such a blanket suffers economically from lack of electricity production, but will have a lower plant cost and higher plant availability due to the simpler, low-temperature balance of plant, and will also incur low technological risk. Modest temperature water-cooled blankets deserve more attention.

Molten Salt Cooling. Use of molten salt as a coolant has not worked out well for three reasons:

1) The hot spot temperatures usually require refractory metals such as TZM.

2) The large amount of fluorine hurts neutron economy with use of pebbles.

3) The salt must not come into contact with the beryllium or else bred uranium will deposit on the beryllium balls causing hot spots. Coatings would be attacked at pin hole defects.

PIPE BLANKET CONFIGURATION

The use of internal pipes in blanket designs has the advantage of forming a barrier to radionuclides much as the clad of conventional fission reactor fuel. In the case of the blanket shown in Fig. 3, the coolant is helium and the pipes contain the fertile fuel along with actinides, tritium and fission products. The small amount of radionuclides in the helium such as trace amounts of tritium and activated corrosion products can be kept to a low level. Other pipe designs have the coolant flowing in the pipes and a static or slowly circulating liquid metal in the fuel pebble bed to conduct the heat into the pipes.[13]

HELIUM-COOLED MOLTEN SALT BLANKET OPTIONS

In 1969, Lidsky discussed a molten salt fission-suppressed breeder.[10] The liquid ^7Li as the neutron multiplier gave a local breeding ratio of 1.45 (T+F). The structural material was TZM, and the cooling was both by liquid lithium and molten salt.

In 1977, Blinkin and Novikov suggested replacing the ^7Li neutron multiplier with beryllium[11] which gave a local breeding ratio of 1.63. In 1978, Lee[12] reported on a homogeneous one-zone design in which the local breeding ratio was 2.2. A rather detailed study in 1979 uncovered problems with fabrication of TZM, radiation damage to the beryllium and to the graphite cladding of beryllium.[3] In 1981, a design with steel was used to contain the molten salt where corrosion was virtually eliminated by keeping the steel cool.[13] However, it used ^7Li instead of Be to produce excess neutrons, thus its breeding was only 1.5.

We are now considering a design using helium as the coolant with the thorium containing salt in pipes somewhat like the helium cooled pipe design of ref. 9. Corrosion is inhibited by a frozen salt layer on the inside of the pipe or at least a salt-steel interface temperature which is cool enough. This design concept (see Fig. 3) looks promising in that it could be a relatively low technology, high performing, economical fusion breeder.

BREEDING RATIO AND LWR SUPPORT RATIO

We define the support ratio as the number of fission reactors which can be fueled by one fusion breeder where each reactor has the same maximum nuclear power. We assume the ^{233}U makeup for a conventional LWR is 460, 380 or 300 kg for each full power 1 GWe year of operation on the denatured uranium cycle (^{233}U + ^{238}U with Pu recycled), the denatured thorium cycle (^{233}U + Th + ^{238}U with Pu recycled) or the thorium cycle (^{233}U + Th), respectively. This ^{233}U consumption rate at 33% thermal efficiency equates to 0.15, 0.125, and 0.10 kg $^{233}U/MW_{nuclear} \cdot y$.

In each fusion reaction, the nuclear energy release is 14.1 MeV x M + 3.52 MeV. The number of fissile atoms bred is F and of fusile (tritium) atoms is T. If T equals 1.05 and the beryllium blanket has an average energy multiplication of 1.6, then each fusion reaction results in F fissile atoms bred and 26 MeV energy release. The fissile production for ^{233}U is then:

$$\frac{F\ 233 \times 1.67 \times 10^{-27}\ \text{kg}}{26\ \text{MeV} \times 1.6 \times 10^{-19}\ \text{j/eV}} \times \frac{3600\ \text{sec}}{\text{hr}}\ \frac{24\ \text{hr}}{\text{d}}\ \frac{365\ \text{d}}{\text{y}}$$

$$= \frac{2.95\ F\ \text{kg}\ ^{233}U}{\text{MW yr}}$$

The support ratio then is 19.7 F, 23.6 F, and 29.5 F for the three different fuel cycles. For the reference design shown in Fig. 1, the breeding ratio is 0.62 (T+F = 1.68). The support ratios on the three LWR fuel cycles then are 12, 14, and 18. It may be possible with improved design to increase the breeding ratio to 0.75 (T+F = 1.8) or more. The support ratios would then become: 15, 18, and 22. On the other hand, practical engineering considerations may result in lower performance than shown above.

ECONOMICS

There are a number of ways of looking at the economics of a fusion breeder:

1 - Incremental System Capital Cost

If the fusion breeder costs 2.5 times an LWR of equal nuclear power and produces as much electricity, and supports 12 LWRs, then the combined system cost per unit power ratio is:

$$\frac{2.5 + 12}{1 + 12} = 1.115$$

The incremental cost of the electrical producing system is 11.5%. This represents the extra cost of the fusion breeder fuel source. Using a discounted cash flow method (p. 8-38, ref. 1), the incremental cost of electricity varies from 5% to 11% depending on assumptions made. Fig. 4 shows the average present value of the system electricity cost versus Q for the optimized reference design. We can see that Q should be equal to or greater than about 6 so that the recirculating power does not significantly add to the product cost.

2 - Equivalent Cost of U_3O_8

The 30-year discounted average cost of the material produced in the reference design is equivalent to \$165/kg U_3O_8 (\$75 per pound). The price of U_3O_8 reached a high of over \$40 per pound in 1978. This would be about \$60 in 1983 dollars. The price of uranium at present is depressed to under \$30 per pound; however, some studies predict the price may be well over \$100 per pound early in the next century (see Jan. 1982 Nuclear News p. 61).

3 - Cost of ^{233}U (or Pu) per gram

The present value of ^{233}U averaged over 30 years in our model (pp. 8-30, ref. 1) is \$93/g.

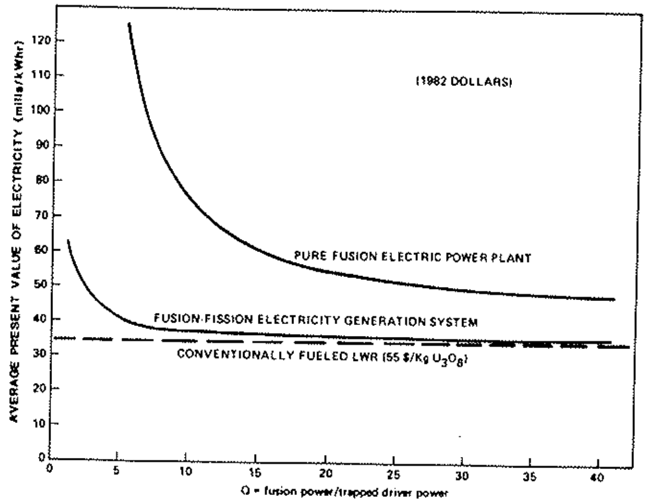

Fig. 4. The cost of electricity for fusion-fission electricity generation and fusion electric versus the fusion gain.

BURNER REACTOR FUEL CYCLE

Either ^{233}U or Pu can be produced. For the fission-suppressed blanket, we prefer ^{233}U production because of its higher utilization efficiency in thermal neutron spectrum reactors and because suppression of fissioning is easier since the fast-fission cross-section is 2 to 3 times lower for ^{232}Th than for ^{238}U. For the fast-fission blanket we prefer Pu production because both the breeding ratio and energy multiplication is higher than for ^{233}V and this results in a lower Q required for electrical break-even.

The four fuel cycles for use of ^{233}U or Pu are:

1 - ^{233}U+^{238}U:	denatured uranium fuel cycle
2 - ^{233}U+Th+^{238}U:	denatured thorium fuel cycle
3 - ^{233}U+Th:	thorium fuel cycle
4 - Pu+^{238}U:	plutonium fuel cycle

The first fuel cycle is unique in that it is almost indistinguishable from the present day fuel - ^{235}U + ^{238}U. No thorium chemistry is needed whatsoever in the burner fission reactor's fuel cycle. It's one in which no readily accessible materials are available for weapons manufacture, and therefore it would be suitable for supplying reactors in foreign countries with the spent fuel being shipped back for reprocessing. Reprocessing of spent fuel is necessary for good fuel utilization efficiency. The first two fuel cycles produce Pu. This bred Pu could then be recycled in Pu burning reactors on the fourth fuel cycle. ^{232}U is produced in small quantities along with ^{233}U. Associated with ^{232}U are gamma radiation emitters which require shielded handling - an expense which, however, makes this fuel much less suitable for weapons use than plutonium. The third fuel cycle is the most fuel efficient. These fuel cycles are discussed more fully in the fusion breeder context in Chapter 7 of ref. 1.

FUSION BREEDER FUEL CYCLE

The feed stock for the blanket of the fusion breeder will either be Th or ^{238}U. The discharge will be Th + ^{233}U + fission products or ^{238}U + Pu + fission products. As mentioned earlier, there will be small quantities of other elements, for example, ^{232}U or ^{238}Pu.

REPROCESSING OF FUSION BREEDER FUEL

In order to suppress fissioning of bred fuel the concentration must be kept less than 1% of the fertile in the fission suppressed case. Therefore, a goal is low cost reprocessing so that we can easily afford to reprocess at this low fissile concentration. Aqueous reprocessing of Pu (Purex) or Thorium (Thorex) fuels are well known and expensive processes routinely used for oxide fuel forms requiring little or no further development. Pyrochemical[19] (or pyrometallurgical) processes are well founded in laboratory scale proof tests and have the potential to save almost an order of magnitude on reprocessing costs but will require a relatively expensive development effort. The preferred solid fuel form is metallic thorium. In the case of molten salt (in which fluorination should be a low-cost process[13]), the form is ThF_4. The molten salt case should have the lowest fuel cycle costs as well as allow for on line refueling and reprocessing.

D-D CYCLE FUSION BREEDERS

For the same fusion power on the D-D cycle, fusion breeders can produce almost twice the amount of material as can D-T cycle fusion breeders. However, this gain is approximately offset by the increased cost per unit power of D-D fusion over D-T fusion based on a particular example of a breeding version of the Wildcat and Starfire Tokamak studies, according to a recent study by Greenspan and Miley.[14] Even though D-D fusion would make a better breeder, the more advanced fusion technology required might make this a somewhat later rather than an early application of fusion.

DEVELOPMENT

During the next 10 to 15 years, the pacing or long lead time items for the fusion breeder is fusion technology itself. Blanket technologies if not addressed early in the program could become pacing items. To prove fusion breeders will be practical, an expanded studies and experimental program is needed. The ultimate proof test of breeding blanket technology will require exposure to 14 MeV neutrons up to fluences of at least 5 $MW\cdot y/m^2$ at a wall loading greater than 1 MW/m^2. This subject has been studied in an EPRI sponsored fusion breeder development study for tandem mirrors,[15] tokamaks,[16] and inertial fusion.[15]

Fusion research and development has become costly because the size of experimental facilities is large and getting larger (e.g., TFTR $\sim$ \$0.5 B, MFTF-B $\sim$ \$0.3 B). Provided we have a healthy fission industry, the fusion breeder could be an early application of fusion research and development which would help justify the large expenditures which will be necessary to construct and operate even larger facilities in the future.

DEPLOYMENT

As discussed earlier, the fusion breeder produces primarily fuel; 5600 kg/y compared, for example, to an equal nuclear power fission breeder that would produce 220 kg/y at a breeding ratio of 1.2. The question we have posed is how early an impact and how large an impact can the fusion breeder have when the long predicted uranium shortage forces breeding technology to be deployed.

We show as an illustrative example how fusion can go through an orderly set of development steps and expand, limited by traditional, new technology learning curves. Our example shows how 50% of the projected electrical demand in the year 2050 can be met by nuclear with the help of the fusion breeder. A possible set of development steps which assumes significantly increased funding from the present level are shown below:

- Integral neutronic tests to verify breeding: TFTR, mid to late 1980s
  and
- Heat removal at temperature: Tritium burning upgrade of MFTF-B, early 1990s
  and
- Blanket component and material lifetime testing,[b] 1-10 $MW\cdot y/m^2$: Engineering Test Reactor,[c] start mid to late 1990s
  and
- Prototypic blanket testing and system demonstration with breakeven or better power > 1000 kg ^{233}U/yr (production rate): Fusion Power Demonstration, Phase 1; mid-1990s Phase 2, 2000
- First commercial fusion breeder (3000 MW fusion) > 6 Tonnes ^{233}U/yr by 2015

We optimistically assume based on successful operation of the first commercial fusion breeder starting in 2015 and a clear need for fuel, that 5 more units could be ordered and put into operation by 2030 providing fuel for over 120 GW_e of LWRs. By 2040 (15% growth rate), there could be 24 fusion breeders providing fuel for 500 GW_e of LWRs. By comparison we assume the first commercial LMFBR to be operational in 2005 in the U.S.[d] By 2020 five more plants could be put in operation, accounting for 9 GW_e. At a 15% growth rate, there would be 36 GW_e of LMFBRs by 2030 and 150 GW_e in 2040. In summary, by 2030 the high LWR support ratio of the fusion breeder could provide about a 10-year lead in deployment relative to the LMFBR, and by 2050, 50% of the electrical demand could be met by fusion breeder supported LWRs. If the first commercial fusion breeder were delayed by ten years (2025), then the fusion breeders impact would be only comparable by 2030 to the LMFBR example used here.

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[b]Material testing can be carried out concurrently with the following step, or with some delay and extra risk but lower total cost, the material testing can be carried out in the prototypic blanket testing facility.

[c]This machine could be based on a tandem mirror operated at 25 MW fusion such as TDF or a tokamak operated at 250 MW of fusion power such as FED-R.

[d]We argue that the fusion breeder may be preferred over the fission breeder and therefore may displace the fission breeder at some future time. However, we must remember that fusion feasibility is not even proven, whereas the fission breeder is proven (although in a somewhat costly form) and its development should be carried up to the point where deployment is possible because of the possibility that it will both be needed and fusion will not be available when needed.

CONCLUSION

Fusion breeder studies, being carried out with increasing attention to details, are laying the foundation for an early and economic application of fusion. The concept of fission-suppression has been shown to be advantageous due to its extra safety. The fission-suppressed fusion breeder is fast to deploy due to its extra high support-ratio, and it is based on relatively modest extensions of conventional nuclear technology, which however will require considerable R and D. Experimental studies are needed to resolve key issues such as establishing material compatibility by carrying out tests on liquid metal loops including MHD effects, integral neutronics tests to verify breeding predictions, and verification of pyrochemical processing of low fissile discharge fuels. We show how fusion by breeding can allow nuclear to expand to 50% of the electrical demand by 2050, for example, if needed.

We are presently including in our work fissioning blankets whose goal is to achieve good safety and economics by the use of pebble fuel. Such designs could make a fusion breeder practical based on fusion performance, expected to be achieved within the next decade.

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