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End Product Economics and Fusion Research Program Priorities

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It is shown that deuterium based fusion fuels and reactors based on them face severe technological disadvantages in comparison with fission based systems as power sources for central station electric power plants. The author postulates the most plausible deuterium based fusion reactor consistent with the physics of the fusion reaction itself and compares this reactor (called "OMR-DT") with existing fission reactors. Since neutrons are the main problem in fusion, the author suggests that a great deal more effort should be given to the study of non-Maxwellian plasmas with the emphasis on neutron-free fuel cycles. The author also suggests that the deuterium based fusion driver may play its best role as a fissile fuel producer.

KEY WORDS: Fusion reactors.

1. INTRODUCTION

Nuclear fusion has from its very inception been viewed primarily as a source of central station power comparable to nuclear fission. The great interest in the admittedly difficult task of developing fusion was sparked by fusion's obvious advantages over fission systems with respect to fuel availability, inherent safety, and waste disposal. Although there are many possible fusion fuel cycles, the effort has concentrated almost entirely upon deuterium-based cycles and more particularly on the D-T cycle. It was realized from the beginning that the D-T cycle has substantial disadvantages (thermal cycle energy recovery, neutron damage and activation, volatile radioactive components, the need for tritium regeneration, etc.), but the advantages of the large fusion cross-section and the low temperature reactivity of the D-T cycle were apparently overwhelming. In many ways, the similarity of the D-T cycle with its neutron-based economy to the fission cycle made it easier to accept the acknowledged disadvantages of deuterium based cycles. The early emphasis on D-based cycles had fundamental implications not just for reactor designs and associated experimental programs, but also for the choice of the plasma physics regimes studied. In particular, it is easy to show that the D-based cycles operating under conditions of maximum Q (energy multiplication ratio) or maximum power density must be nearly Maxwellian. As a result, we have concentrated on the physics of such nearly Maxwellian plasmas and on theoretical and experimental studies of confinement schemes appropriate to such plasmas. Although the existence of many different reactor *concepts* is superficially comparable to the early stages of the fission power development, it is important to realize that the differences between the physics and technologies appropriate to different fusion fuel *cycles* are far larger than those between different fission fuel cycles. The fusion program's apparent diversity exists within a limited subset of the field.

The early decision to concentrate effort on the D-T fuel cycle—a calculated risk but quite the correct decision at the time—has set the scaling parame-

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ters of possible fusion utilization. D-cycle based reactors will necessarily be large in comparison with fission power systems, and the neutron fluxes in these large reactors will almost certainly be higher than in fission systems with equivalent power densities. The thermal efficiency of such central station fusion reactors will not be substantially higher than that of existing fission power plants and might in fact be somewhat lower. The mass and energy cost of the plant core will be substantially larger (possibly 20–30 times larger). These conclusions follow straigthforwardly, and inexorably, from the decision to utilize the D-T cycle, and are almost entirely independent of the particular plasma confinement scheme in use. Although the basic design features of the literally dozens of detailed fusion rector design studies span a wide range of technologies, the net size and cost of the plausible designs are surprisingly similar.

Other possible end uses of fusion are fissile fuel generation, synfuel production, fission product waste destruction, process heat sources, etc. These are even more strongly influenced by the choice of fuel cycle than is the central station power plant. Except for fissile fuel generation, there have been few detailed proposals in these areas; we will limit discussion of other end uses primarily to fissile fuel generation. Unfortunately, the particular plasma confinement schemes developed for deuterium cycle central station power plants are not well suited for use in fusion-fission devices or the other commonly mentioned fusion end uses. Although other confinement schemes, not suitable for central station use, might be able to take advantage of D-based cycles for selected end uses, there is no significant research support for such schemes.

I believe that deuterium cycle based central station fusion power plants, no matter how cleverly designed, will necessarily be technologically inferior to their competition. The "inexhaustible" nature of fusion fuel supplies will not overbalance the technological disadvantage because existing uranium reserves will almost certainly be adequate for 30–50 years, because the cost of electricity is only weakly dependent on uranium price, and because both breeder reactors and the enormous ocean water uranium reserve renders fission essentially inexhaustible also.

If deuterium based cycles do indeed lead to a technological dead end, then it is imperative that a significant portion of our resources be put to the study of other fusion fuel cycles and appropriate schemes to exploit them. Admittedly, the proper path is not obvious and there is no guarantee of success. However, it seems clear that advanced fuel cycles are highly unlikely to be of any interest whatever in those confinement systems developed for deuterium based fuels. Therefore, power producing advanced fuel systems will almost certainly have to be non-Maxwellian. We know little about such systems; we are not even sure of the best alternative fuel cycle. Although there has been a continual low level interest in "advanced" fuels, there has been little attention given to the potential of alternate fuels in systems optimized for their unique properties. The optimum combinations of confinement schemes and fuel cycles diverge at basic levels, and it is not possible to pass smoothly from one combination to another. We have been following one path; although we have made enormous progress, the path ends in a thicket. Success *may* come faster on another road.

A research and development program is aimed, as the name itself implies, at developing a product. The desirability of that product often cannot be determined within the program itself because acceptance depends on a congeries of external political and economic factors. The correct assessment of these factors nonetheless remains an essential part of a successful R&D program. In the case of a very long term development program, one must estimate both the time of completion and the state of the exogenous factors at that time. Consideration of such matters can sometimes be a difficult task, one that is viewed as far removed from the usual responsibilities of program management. Nonetheless, in long term programs, such assessments are crucial.

There is ample evidence (some of it quite recent) that misjudgment of competing technologies or of time scales can lead to development fiascos even if the internal goal is met. The Anglo-French SST is an obvious example of a case where misreading the market for a technically successful device resulted in embarrassing program failure. The SST met all of its design goals. The proponents have argued, correctly I think, that it was the best possible SST that could have been designed at that time under the existing constraints. However, the external world cares little for a program's internal constraints, and the SST failed completely to meet market requirements. As a result, only 22 planes were built (100 was to have been the breakeven number) and each one flies at a substantial loss. Possibly more serious than the monetary loss is the fact that it will be a long time before another attempt is made to design such a plane even though the technology has advanced and the ground rules have changed.

The history of the nuclear powered airplane is perhaps even more to the point. Less than two years after that project was launched in 1945, with high hopes and large budgets, several scientific review committees pointed out that the final design would be dictated by considerations of fission reactor shielding and availability of high temperature reactor structural materials. Simple scaling exercises were able to predict the final size and performance of the craft on the basis of available shielding materials and experience with high temperature heat transport limits. The last item was particularly critical because unless high temperature materials could be developed, the reactor would have to be unacceptably large and the shield, which scaled as the square of reactor dimensions would be, unacceptably heavy. In 15 years, after one billion dollars was spent on design studies, cost benefit analyses, and "proof of principal" experiments, the project was cancelled. The fundamental questions of shielding and heat transfer had not been solved and no amount of technical brilliance could overcome the scaling laws.

It is obvious that underestimating the time required to develop a particular technology can be costly. In some situations, it can also happen that such underestimation can have other important *programmatic* effects. This can occur if promising avenues of research are precluded because faster or safer paths are chosen over those with possibly greater benefits but with higher risks or longer development times. Our current dependence on the LWR-LMFBR fission reactor cycle is a particularly appropriate example. It is now suspected that other fuel cycles or other fuel/coolant pairs might well be better suited to near and intermediate term conditions than are the existing uranium based systems. The uranium LWR system was chosen in large measure because it was supposed that rapid solutions would be preferable to optimized solutions, especially in light of the anticipated rate of electrical power growth. The effect of these errors has now become obvious, but it is proving almost impossible to shift from existing systems to those that would be preferable. Because the differences between the physics and technology appropriate to individual *fusion* fuel cycles are far larger than those differences between *fission* cycles, the possibility of graceful recovery from a wrong early decision in the fusion program will be consequently even smaller.

2. METHODOLOGY AND ECONOMIC ASSUMPTIONS

The discussion below depends on predictions covering 30 or more years. It has long been apparent that fusion development must be viewed on such a multidecade basis. It is acknowledged that the present path will take at least 20 years to reach the "demo phase" and might well take substantially longer. On the other hand, both financial and intellectual discount rates make it highly unsatisfactory to expend effort on developments that might come to fruition no earlier than 40 years hence. These bracketing values make it necessary, therefore, to guess what the world will be like approximately 30 years hence. Such predictions are fraught with uncertainty, but several factors make our conclusion relatively insensitive to error. One such mitigating factor is the inertia of the very large and expensive systems under consideration. A more important factor is that it will suffice for our purpose to compare fusion with its direct competitors. The competition will take place in whatever social and economic environment exists at that time; therefore, all we need do is make relative rather than absolute comparisons.

Fusion generated energy is substitutionally competitive with fission generated energy. Both of these nuclear sources are directly competitive with the various fossil fuels. The competition between nuclear power and coal (the probable long range source of fossil energy) will be determined by a complex of factors including the cost of tapping uranium reserves, climatic effect of fossil fuel burning, public acceptance of nuclear power, rate of power demand growth, etc. Because of the substitutive nature of the fission versus fusion choice, and because deuterium based fusion is qualitatively very similar to fission from the users' standpoint, we can use "poor man's game theory" to simplify the problem. We will derive the properties of "optimum" fusion schemes. If we can then show that the optimum fusion schemes are inferior to fission based schemes *that satisfy the same market*, then the path that leads to that fusion scheme need no longer be followed no matter what the

Table I. Contributions to Electricity Cost of Nuclear Fuel Cycle Operations[a, b]

Capital charge	Fuel cycle costs, without uranium purchase				Uranium purchase cost, with UO_2 at:			Total fuel cycle cost with UO_2 at:		
rate	Enrichment	Fabrication	Store	Total	\$50/kg	\$100/kg	\$200/kg	\$50/kg	\$100/kg	\$200/kg
6	1.61	0.49	0.43	2.53	1.54	3.07	6.14	4.07	5.60	8.67
10	1.84	0.54	0.38	2.76	1.78	3.56	7.11	4.57	6.32	9.87
15	2.14	0.62	0.33	3.09	2.12	4.25	8.50	5.21[c]	7.34[c]	11.59
20	2.48	0.72	0.29	3.49	2.52	5.04	10.17	6.01[c]	8.53[c]	13.56

[a] In mills/kwh. Data from ref. 2, p. 269.
[b] Assumptions: (1) Once-through (throwaway or stowaway) cycle. (2) Enrichment to 3% at 0.25% ^{235}U tails assay. (3) Burnup at 3 MWd (heat) per kilogram of UO_2. (4) Conversion efficiency at 31%, heat to electricity. (5) Unit costs and payment schedules. Uranium: variable cost, mean payment 4 years before mean receipt of revenue; includes costs of mining, milling, exploration. Enrichment: \$100 per separative work unit (uranium), paid $3\frac{1}{2}$ years before receipt of revenue; includes chemical conversion costs. Fabrication: \$100/kg of UO_2, paid 3 years before mean receipt of revenue; includes chemical conversion costs. Storage fee: \$125/kg of UO_2, paid 3 years after mean receipt of revenue; includes transportation costs.
[c] These values give the approximate range of present market conditions.

outcome of the fossil versus nuclear decision. This process is relatively easy to carry out for central station power plants and for fusion-fission hybrids. Matters are trickier for synfuel producing systems, because deuterium based fusion might, in certain circumstances, have significant advantages with respect to fission in this case. However, a direct comparison with fossil fuel based synfuel systems would then be appropriate because substantially the same considerations of CO_2 effects would apply to both. We will not consider synfuels in this paper.

In the discussion of central station electric fusion power plants and fusion-fission systems below, we will assume that nuclear power will be politically acceptable and that the true cost of fission generated electricity will be no more than twice its present cost in suitably deflated (i.e., constant) dollars. As shown below, this assumption does not depend on the introduction of the LMFBR. From the point of view of fusion, such introduction would make the competition for central station prime mover even more severe and would reduce the value of fusion generated nuclear fuel.

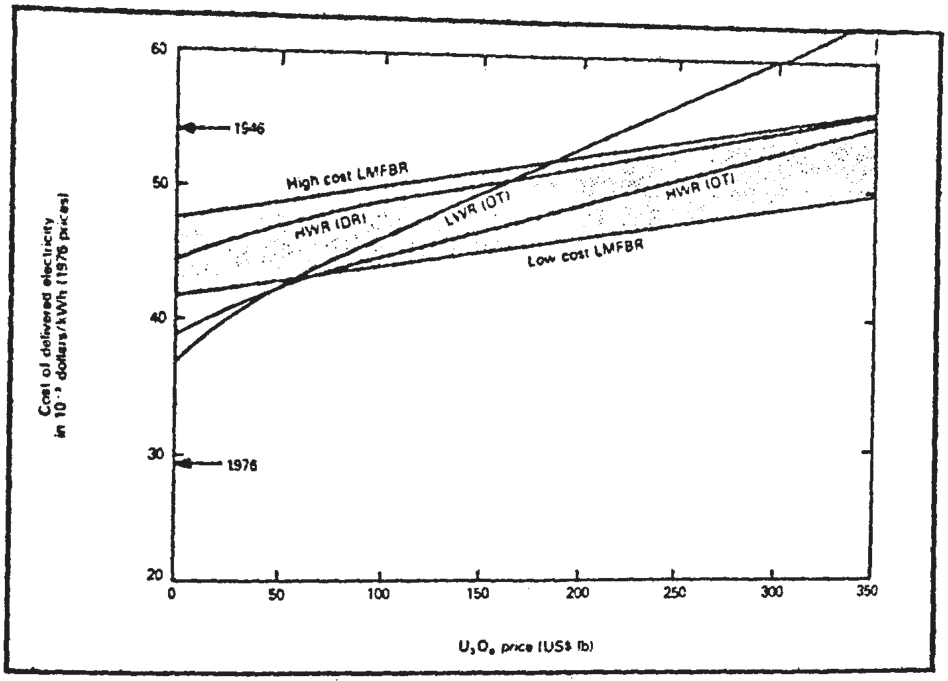
The cost of electricity (COE) is only weakly dependent on the price of U_3O_8 even for the relatively inefficient present day LWRs. A recent EPRI/Westinghouse[1] study derived an expression for COE as a function of U_3O_8 price in 1980 dollars. This calculation assumed a fixed charge on capital of 15%, an inflation rate of 6%, and a 10 year construction period. Under those circumstances, the total cost of electricity is related to the cost of U_3O_8 by

$$\mathrm{COE(mils/kWhr)} = 33 + 0.125C_u$$

where C_u is the cost of U_3O_8 in \$/lb. At current prices, the fuel purchase price represents less than 8% of the cost of electricity.

Similar conclusions were found in a recent CONAES study[2]. Table I is a detailed breakdown of fuel cycle cost based on U_3O_8 prices of \$22, \$44, and \$88 per pound. At this time (March 1982) the spot market price for U_3O_8 is 23 \$/lb; i.e., the fuel cycle contribution to the COE is almost negligible. According to the CONAES projection, the cost of electricity would double at approximately 250 \$/lb (i.e., roughly 10 times the present value). Doubling of the COE for 250 \$/pound U_3O_8 is in complete agreement with the EPRI/Westinghouse study results.

Von Hippel et al. contributed an economic analysis of alternate fission fuel cycles as part of a SIPRI symposium on nuclear energy and nuclear weapon proliferation.[3] Figure 1 shows the estimated price of delivered electricity (in 1976 dollars) levelized over the life of the power plant for alternative reactor systems, as a function of the price of U_3O_8. To offer an historical perspective, US average electricity prices (in 1976 dollars) for the years 1946–1976 are indicated by arrows on the left-hand scale. We call attention to the LWR once-through-cycle curve [LWR (OT)] which predicts an increase in COE of ~70% for U_3O_8 prices of 250 \$/lb in 1980 dollars.



Note: Here the estimated price of delivered electricity (in 1976 dollars) levelized over the life of the power-plant is shown for alternative reactor systems, as a function of the price of U_3O_8. To offer an historical perspective, US average electricity prices (in 1976 dollars) for the years 1946 and 1976 are indicated by arrows on the left-hand scale.

Fig. 1. Levelized cost of delivered electricity as a function of uranium price. (From F. von Hippel, H. A. Feiveson, and R. H. Williams, "An evolutionary strategy for nuclear power, in ref. 3.)

It is fortunate for our present purposes that the COE is so insensitive to the cost of uranium because predictions of both price and availability are complicated by political and technical factors. This is valid not just for world supplies of uranium where this situation is expected to be clouded but for the domestic U.S. reserve itself. The true magnitude of the cost-sensitive reserve and the recently revealed price elasticity of both supply and demand of electricity clearly has profound implications for nuclear program planning (LWR vs LMFBR, LWR vs HTGR, coal vs nuclear, etc.); most estimates have tended to bolster the agenda of the agency making the estimate.

Uranium resource magnitude is a strong function of allowable costs and much depends on the detailed distribution of relatively low grade ores. However, because of the extraordinary present oversupply of high grade uranium ore, there is no incentive to explore even for intermediate grade ores. There are enormous quantities of very low grade resources, but the environmental costs of mining them have not yet been accurately evaluated. It will suffice for our purposes to use the relatively conservative DOE estimates employed by CONAES (i.e., we will accept the U.S. figure of 2.4 million tons). With such limited uranium availability, we would expect prices to rise relatively rapidly and as a result, we expect nuclear growth to be relatively slow. It is possible, however, that in spite of high uranium costs, and the high cost of fission plants themselves, other factors might nonetheless mandate high nuclear installation rates. NASAP considered both high and low growth rates in conjunction with high uranium prices in arriving at its 1980 estimates of U_3O_8 price versus time. Their results are plotted in Fig. 2. Even for the projection which assumes both high growth of installed power and small uranium reserve, U_3O_8 prices remain below 250 $/lb and the cost of fission generated power is predicted barely to double in 50 years.

If U_3O_8 prices do begin to play a significant role in the total COE, then it would become reasonable to gradually switch from LWRs to more fuel-efficient reactors. With such reactors in place, NASAP predicts the true cost of electricity would not exceed the

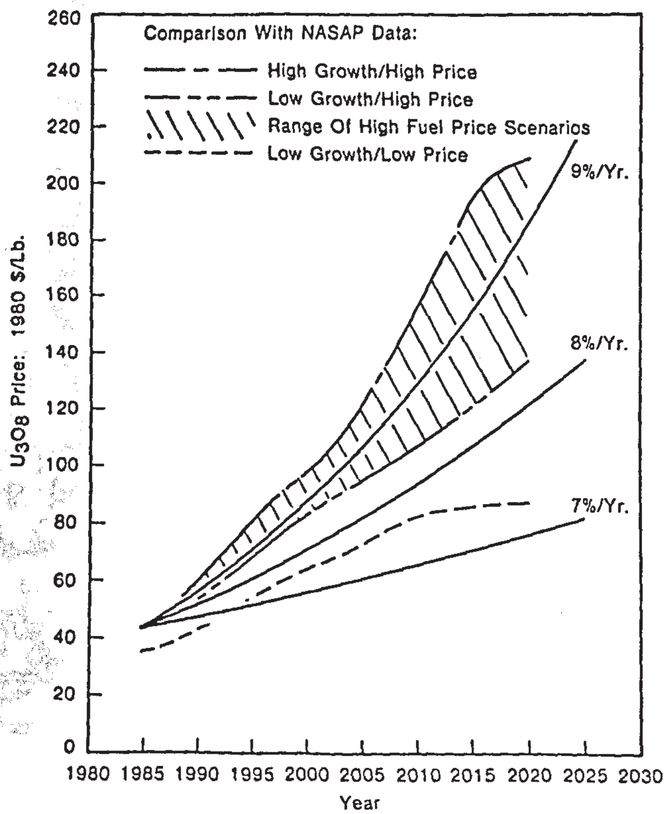


Fig. 2. Projections of U_3O_8 price vs year (rate of increase in total fuel cycle costs: 7%/yr, 8%/yr, 9%/yr). A 6%/yr inflation rate is applied to all fuel cycle costs except U_3O_8. (From ref. 1, p. 20.)

1945 value even at U_3O_8 prices of 350 \$/pound. Of course, if such fuel efficient reactors were introduced, uranium utilization would be reduced and U_3O_8 prices would tend to be depressed even in the face of such rapid electrification as might be called for by electrical replacement of fossil fuels for transportation and industrial uses. There seems to be a substantial amount of negative feedback in uranium pricing.

Finally, we note the resurgence of interest in the extraction of uranium from sea water. Although, the concentration is only 3.3 ppb, the total resource is 4×10^9 tons. Although early estimates of the cost of extracting sea water uranium were high, recent developments in plant design and adsorbent chemistry have been very encouraging. It appears that extraction at 400–500 \$/pound equivalent U_3O_8 is possible today. Several groups of workers are speculating about eventual target values in the 200 \$/pound range.(4) Although such prices hold little present interest for United States users, the idea might have economic merit even now for nations with little indigenous uranium supply. The Japanese, for example, play a major role in sea water uranium research. The seawater resources, in any event, puts a firm cap on possible uranium prices.

It appears then that the *true* cost of nuclear fission generated electricity will perhaps double over the next 40–50 years and, in so doing, return to the true cost prevalent 35 years ago. In light of the fact that the fuel cycle costs need not, for the United States, represent a balance of trade problem, such a gradual return to historical pricing levels will be nondisruptive, especially since it can reasonably be assumed that the cost of fossil power systems will at least double over the same period of time. Thus, it seems clear that fission reactors will not be driven from the scene by questions of fuel availability. The fusion reactor of the future must compete with fission sources producing electricity at approximately 1980 costs in constant dollars, utilizing fissile feed at a price corresponding to perhaps 200 \$/pound U_3O_8 but possibly substantially less.

If for some sociopolitical reason, the fission reactor is *not* an acceptable solution to the energy supply problem, it would be highly unlikely (although not impossible) that DT based fusion would be acceptable. This conclusion follows from two qualitative aspects of reactor safety analysis. First, although the fusion reactors will have substantially lower radioactive burden, the public is not likely to make highly refined judgments as to the relative degree of radioactivity if it has turned against fission for that reason. Second, as TMI has shown, the *financial* hazard to an operator is very severe even in an accident without external energy release once the threshold for remote maintenance has been surpassed. This would be a more serious consideration for complex fusion devices with many energy sources (e.g., magnetic fields, ion beams, etc.) than for fission reactors. The occupational exposure problem is also likely to be more difficult in tritium containing systems.

3. FUSION FUEL CYCLES

The nuclear binding energy versus mass relationship shows that literally hundreds of fusion reactions are possible in principle (i.e., from simple considerations of nuclear mass differences) of yielding net power. Many of these reactions survive tighter selection criteria; the margin by which they exceed these criteria enable us to rank order various possible fusion fuel cycles. This process is discussed in detail in

Table II. Rank Order of Fuel Cycles

Reaction	$\overline{\sigma v} / \overline{\sigma v}^{P}$
D-T	1040
D-He^3	91
p-B^{11}	7–13
He^3-He^3	9
He^3-Li^6	6
p-Li^6	4

Appendix A. For ion energies less than about 5 MeV, the limiting constraint turns out to be "the ignition margin," the ratio by which total power produced exceeds the minimum radiated power of the system. In terms of this parameter, several of the more interesting fuel cycles can be rank ordered as shown in Table II.

As shown in Appendix A, the selection process used in deriving Table II takes into account such factors as maximum reasonable magnetic field strength, limiting beta in magnetic devices, self-consistent electron temperatures, etc. Nonetheless, this ordering may be considered to be based primarily on the physics considerations of power generation because the physics constraints turn out to be more stringent than the technological ones. From the point of view of the *engineering* design of fusion reactors, an important figure of merit is the constant-beta reactivity, $\langle \sigma v / T^2 \rangle$. This parameter, which is closely related to the average power density in a beta-limited magnetic confinement device, is plotted in Fig. 3. (In certain mirror systems, where velocity space scattering is the dominant loss mechanism, $\langle \sigma v / T^{1/2} \rangle$ is the appropriate figure of merit. This yields approximately the same ranking.)

In general, fusion cycles may be considered to be either deuterium-based or proton-based. The deuterium-based cycles, DT, DD, DH_e^3, etc. typically operate at much lower temperatures, have substantial energy release neutron kinetic energy (40–80%), and tend to have somewhat higher energy release per reaction. The proton based cycles p-B^{11}, p-Li^6, etc.) require a higher operating temperature for energy breakeven, have a smaller neutron energy fraction (typically 1% or less, generally in side reactions), and tend to have a somewhat smaller energy release per reaction. The D-He^3 reaction is something of an anomaly. It has the large low temperature cross-section and high energy release characteristic of the

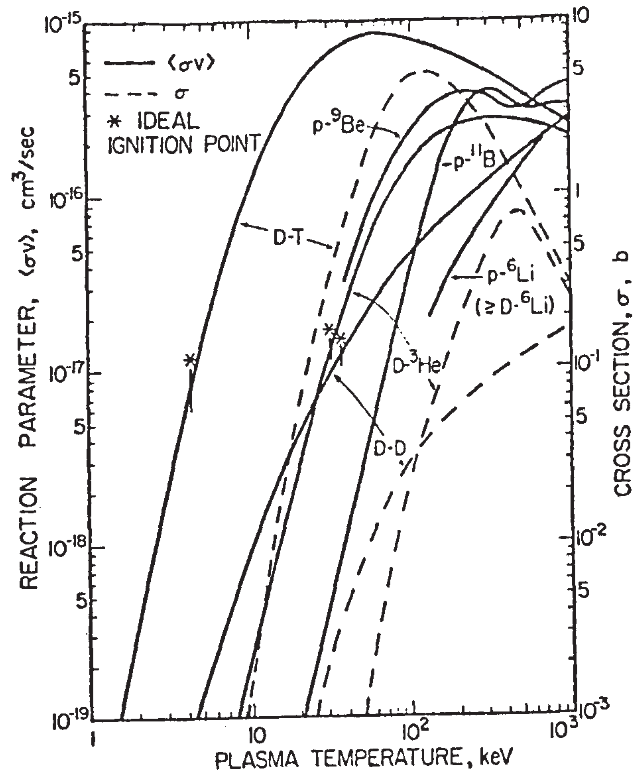


Fig. 3. Reaction parameters and cross-sections for various fusion reactions. The reaction parameter is averaged over a Maxwellian ion distribution. The curves shown for p-^{6}Li, p-^{9}Be, and p-^{11}B contain large uncertainties. Five strong D-^{6}Li reactions occur with different $\langle \sigma v \rangle$, but all lie near or below p-^{6}Li. (From The impact of alternate fusion fuels on fusion reactor technology, ANL/FPP/TM-128, November 1979.)

deuterium-based reactions, but under proper circumstances has less than 1% energy release in neutrons. This would be a nearly ideal fusion fuel cycle if only a source of He^3 existed. To date, the only source seriously considered for this material is by-product He^3 from D-D or D-T reactors. This begs the question.

For reasons made obvious by Fig. 3 and Table II, the controlled fusion program has concentrated almost entirely on the deuterium-based reactions. The result of this concentration has been a concomitant concentration of the plasma physics effort (theoretical, applied, and experimental) on the understanding and control of near-Maxwellian Coulomb collision dominated plasmas. Such concentration is clearly appropriate for investigation of the "low temperature" deuterium-based cycles because at their operating temperatures, the Coulomb scattering time is substantially smaller than the energy confinement time, the lifetime against fusion, and the time required for

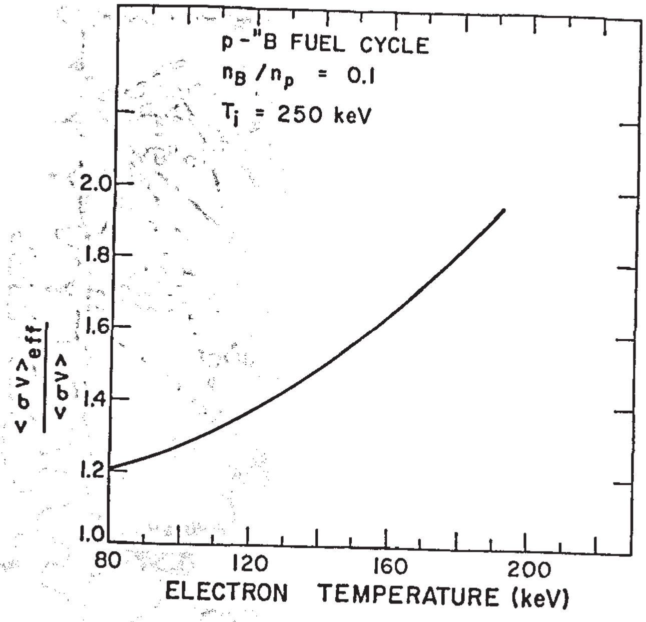


Fig. 4. Relative reactivity increase due to propagating and large energy transfer effects for Maxwellian p-^{11}B. (From Geoffrey W. Shuy, Advanced fusion fuel cycles and fusion reaction kinetics, UWFDM-335, December 1979.)

the fuel ions to transit the confinement device. Under these conditions, the plasma must be very close to equilibrium. The small variations from Maxwellian distribution caused by density and temperature gradients are of fundamental importance in determining transport coefficients and stability, but from the viewpoint of gross parameters such as pressure, reactivity, energy density, etc., the systems are basically Maxwellian.

For very high temperature operation of even such basically Maxwellian systems, some corrections are in order to reflect the behavior of the very high energy reaction products which lose energy slowly in high electron temperature, low density systems. The most important of these corrections involve fast fusion (in which one of the reaction products undergoes additional fusion reactions while slowing down), nuclear elastic scattering (which tends to impart energy directly from the products to reactive ions), and propagating reactions (in which nuclear elastic events promote individual ions into the reactive tail of the distribution). The importance of these effects is shown in Fig. 4 for the p-B^{11} fuel cycle. These effects are quantitatively significant, but, in view of the large differences in ordering parameters, are not qualitatively significant. In no case does the peak value of $\sigma v/T^2$ for any other fuel cycle operating in the near-Maxwellian mode even approach that of the D-T cycle. This simple fact, obvious from the very beginning of the program, has motivated the program's very tight focus on the physics and technology of the deuterium-based systems *in spite of the well-known technological drawbacks of such cycles.*

Table III. Acceptability Criteria

A.	Steady State Operation
B.	Diverted
C.	Ignited
D.	"Toroidal"
E.	Moderate Power Density
F.	Disruption-Free
G.	Modular
H.	Moderately Stressed
I.	"Safe"
J.	Moderate Size

4. CENTRAL STATION POWER: OMR-DT

Central station power plants based on the magnetic confinement of nearly Maxwellian plasmas will be very large compared to existing fossil and fission plants. The size is determined by a number of such disparate factors as plasma transport coefficients, neutron shielding lengths, minimal acceptable power density, maximum acceptable recirculating power fraction, etc. Somewhat surprisingly, these factors, especially such fundamental and relatively immutable ones as shielding length, heat transfer coefficients, and limiting stresses can be used to estimate the size of a power plant within remarkably close limits, almost entirely independent of the details of the plasma physics. Appendix B presents such an estimate (prepared for an earlier NSF study and subsequently subjected to intensive review). It is shown that a D-T based toroidal device (tokamak, torsatron, etc.) that satisfies fundamental physics and technological constraints will almost certain produce at least several thousand MW$_{th}$. This seemingly strong conclusion is vindicated by a number of highly detailed design studies. The minimum size of central station power plants predicted by these studies is of order 1000–1500 MW$_{th}$. The more reliable designs tend to be somewhat larger, 3000–5000 MW$_{th}$. Such reactors are comparable to or larger than the largest existing fossil and nuclear fission plants. This circumstance makes it both necessary and possible to derive a set of technology-based design criteria appropriate for such systems. These criteria are listed below (Table III) and discussed at length in Appendix B.

No fusion reactor concept under active study has shown itself capable of meeting all these criteria, but we can extrapolate from current studies well enough to imagine a device that might. The immediate goal is to imagine the "best possible" deuterium based central station fusion reactor. On the basis of the selection criteria of Table III such a reactor could be a moderate-aspect-ratio, steady-state, toroidal device operating in the ignited mode. Because of its reliance on near optimum temperature deuterium based plasmas, the plasma will be Maxwellian. Therefore, relying on a somewhat optimistic extrapolation of plasma physics and on substantial development of existing technology, we will postulate the existence of an Optimum Maxwellian Reactor operating on the D-T cycle: OMR-DT. In the interest of having a concrete exemplar, we will suggest a model for "OMT-DT"; however, the argument is not sensitive to the exact nature of the device so long as it operates on a deuterium based cycle, meets the selection criteria of Table III, and exhibits the fundamental properties of near Maxwellian plasmas.

OMR-DT is a small aspect ratio, steady-state, ignited torsatron with discrete divertors. For present purposes, it could just as well be a steady state current-drive tokamak. OMR-DT has aspect ratio of 6 and an average β of 12%. We assume that the plasma does not disrupt and that the ions obey neoclassical transport theory. The superconducting magnetic field coils are so arranged that the internal elements of the reactor (first wall, blanket, shield, etc.) can be removed without disturbing the magnetic field coils. Of course, experienced fusion researchers will recognize that the plasma physics assumptions are very optimistic and that the physical configuration has yet to be demonstrated.

How does this "best" plausible D-T fusion reactor compete with an LWR? Because we have assumed that there will be no shortage of uranium and that fission will be "acceptable," OMR-DT will have no market unless it has substantial advantages with respect to the LWR. (In view of the time scale envisaged, comparison of OMR-DT with advanced fission reactors such as HWRs or HTGRs might be more to the point, but the simpler test will be sufficient.) Although it is difficult to make a quantitative comparison between a reactor which does not yet exist and another which is still evolving, nonetheless, such comparisons can be made because certain generic features of D-T fusion are fundamental and because it is reasonable to assume that future fission reactors will be at least as good as existing ones.

The microscopic differences between D-T fusion and heavy element (e.g., uranium) fission are manifested as fundamental macroscopic differences in the reactors that exploit these fuels. The D-T reaction has a single path

$$D + T \rightarrow \mathrm{He}^4(3.5\ \mathrm{MeV}) + n(14.1\ \mathrm{MeV})$$

whereas the fission reaction must be represented statistically. The most probable fission reaction is

$$\mathrm{U}^{235} + n \rightarrow \mathrm{FP}_1(100\ \mathrm{MeV}) + \mathrm{FP}_2(65\ \mathrm{MeV}) + 2.5n(5\ \mathrm{MeV}) + \mathrm{rad.}(20\ \mathrm{MeV})$$

where FP_1 and FP_2 are fission products and the last term primarily represents prompt γ reactions.

In contrast to the 80% of the energy released carried by the 14.1 MeV neutron in the fusion reaction, the lower energy neutrons of the fission reactor carry less than 3% of the available fission energy. The fission products (FP_1 and FP_2) that do carry the bulk of the energy are usually massive and heavily charged. The range of these products in typical dense fuel material is only 5 μm (5×10^{-4} cm). The "range" of a fusion neutron is hard to define, but it is clearly measured in tens of centimeters: 50 cm blankets are required for complete energy attenuation, even thicker ones for radiation shielding. Fundamentally then: (1) energy release is local in fission, distributed in fusion; (2) the energy weighted neutron flux is substantially higher in fusion (approximately 14 MeV per 20 MeV total release) than it is in fission (approximately one 2 MeV neutron per 80 MeV energy release).

A particularly important consequence of these considerations is that the coolant can, and indeed must, be in close proximity to the reaction zone in fission reactors. This makes possible the rod-bundle and plate-bundle fuel element configurations common to almost all fission reactors. Such configurations allow an almost arbitrarily large ratio of heat transfer surface to reactor volume.

For obvious reasons the coolant in a fusion reactor must be external to the reaction zone. The total heat removal from a fusion device for given average heat flux is proportional to the surface to volume ratio for the entire plasma chamber. This is much less than the surface to volume ratio of multiple internal element systems. Thus, if the limiting heat fluxes are equal, the fission reactor can have substantially higher volumetric energy density. Note that this conclusion does not even take into account the large and complex external blanket required in a D-T fusion reactor. The external volume of the fusion blanket tends to have very low power density.

Although neutron economy is important to both OMR-DT and the LWR, the number of possible coolants is more restricted in OMR-DT than in the fission reactor. This surprising result comes about because it is essential to exploit high energy neutron multiplying reactions if tritium breeding requirements are to be met, and this limits both the thickness of the first wall and the allowable coolant pressure. Furthermore, because of the difficulty in separating tritium from large quantities of water, it is far more difficult to utilize water cooling anywhere in D-T systems than it is in fission systems. Although the neutron economy is important in fission reactors, the use of slightly enriched uranium allows the use of water as coolant.

Although it may be possible to mitigate heat transfer problems at the fusion reactor first wall by the development of high efficiency divertors, the first wall will always operate at heat fluxes at least equal to that of fission reactors and in a flux of neutrons whose number and energy *both* far exceed that of equivalent fission reactors. This follows from the higher ratio of neutrons to energy per fusion reaction, the simple scaling of heat transfer area with plasma radius, and from the necessity to drive the 1 m thick blanket to technologically interesting power densities. Neutron energy fluxes of 10 MW/m^2 would be desirable from an economic standpoint, but almost all recent designs have assumed the energy throughout to be in the range of 1.5–3 MW/m^2. Even these lower values correspond to 14 MeV neutron fluxes of $\approx 10^{14}$ n/cm^2 sec.

The first wall of a fusion reactor is the functional equivalent of the fuel cladding in a fission reactor. Both of these interfaces tend to be the design limiting features of their respective reactors. The fusion reactor first wall is necessarily subject to much higher neutron flux and, in the absence of divertors, much higher heat flux than fission reactor fuel cladding. Simultaneously, the first wall is subject to more restrictive physical demands; vacuum integrity, high stress, etc. Fission reactors have proven capable of

Table IV. Comparison of LWR With OMR-DT

Parameter	LWR	OMR-DT
Neutron flux at interface (n/cm^{2}sec)	10^{13}	10^{14}
Mean energy at interface (MeV)	0.1	14.1
Average heat flux at interface (W/cm^2)	100	100
Power density (W/cm^3)	100	10
Power density (MW$_{th}$/kton)	5	.2

continued operation with limited failures of the cladding/coolant interface, but such operation will be much more difficult in a fusion reactor. The implications for system reliability are obvious, especially in view of the fact that it also appears it will be more difficult to replace individual portions of a fusion reactor first wall than it is to replace fission reactor fuel elements.

If the heat and neutron fluxes at the first wall of a D-T fusion reactor are limited to values typical of those in fission reactors, then it must follow, because of surface/volume considerations, that the overall energy density of the reactor will be substantially lower. This will tend to make the fusion reactor more expensive; in the absence of a detailed design, it is difficult to be precise as to how much more expensive. However, it does seem clear that the structure of a fusion reactor will be at least as complex as that of a fission reactor, and the construction materials will be more energy intensive. Use of a simple energy density scaling factor is, therefore, somewhat weighted in favor of the fusion system; nonetheless, as can be seen in Table IV, which sums up the several criteria discussed above, the fusion reactor is still at a factor of 25 disadvantage.

Table IV demonstrates that, even though the fusion reactor has a substantially lower average energy density, its critical components will operate at substantially higher neutron flux. In view of the particularly critical nature of the first-wall/plasma interface, this is a very important consideration. Table IV was constructed assuming that the average heat flux at the interface is the dominant design criterion. This is not the only possibility and, therefore, it is important not to use this particular case to justify a simplistic rebuttle. For example, it is possible to imagine fusion reactors with substantially higher power densities (e.g., some versions of the reverse field pinch or very high field tokamaks.) However, if the postulated power density in the fusion device is set equal to that of the LWR, then the heat flux at the interface of the fusion device exceeds that of the LWR by a factor of at least 10, and the fusion neutron flux is higher than that of the LWR by a factor of at least 100. The reaction kinetics of D-T imposes a 2–3 order of magnitude handicap on the fusion reactor. *This handicap can be redistributed but it cannot be made to disappear*. In comparison with the LWR, OMR-DT is larger, more highly stressed, technologically more complicated, and has substantially lower power density. It seems hard to identify a customer.

5. FUSION-FISSION

The idea of using fusion generated neutrons to produce fissile fuels is not a new one. The idea was apparently first suggested in 1956 in classified studies and was proposed for the first time in the open literature in 1969. Although the first published work described what has since come to be known as a "stand-alone fuel factory," the bulk of the early work concentrated on the use of fast fission of U^{238} in subcritical blankets to raise the total energy production of hybrid fusion-fission machines. Some attention was given also to the use of fusion generated neutrons for treatment of fission product wastes. Waste transmutation and high gain fast fission systems for significant power multiplication were found to have significant technological disadvantages (very high neutron fluxes and power densities). Thus attention now is concentrated on fissile fuel production with electricity as a byproduct in some cases.

Fusion-fission studies have been supported for the last decade at a level of perhaps one to two million dollars per year by DOE and EPRI. The Soviet Union maintains an aggressive effort in this area (the hybrid was viewed for a time in the Soviet Union as a possible competitor to the LMFBR), and there have been several joint US/USSR workshops on the subject. The current status of fusion-fission research is summarized in Report RP-1463, "Preliminary Feasibility Assessment of Fusion-Fission Hybrids", prepared for EPRI by Westinghouse Electric Corporation.(1) Other useful information is contained in the 1978 and 1981 versions of the "Source Book of Fusion-Fission Hybrid and Related Technologies,"(6) as well as the proceedings of the EPRI/DOE fusion-fission hybrid review meeting held on June 4, 1981, in Washington, D.C.(7)

The energetic neutrons born in the D-T reaction are capable of exciting a number of (n,2n), (n,3n), and (n, Tn') reactions. Therefore, it is possible, even in blankets designed with engineering realities in mind, to have a neutron excess of 0.4–0.6 neutrons per fusion reaction, i.e., 0.4–0.6 neutrons in addition to those required to regenerate the tritium fuel. These excess neutrons are available in the energy range of 1–2 MeV. In other words, there is effectively one "MeV range" neutron available for each 40 or so MeV of fusion energy released. The neutron excess in the D-D reaction is remarkably similar. There are two neutrons available in the D-D reaction for each 68 MeV of fusion energy released, *if* all reaction products are recycled and burned. This corresponds to one "MeV range" neutron per 34 MeV of fusion energy released. In light of the reactivity difference between the D-D and D-T cycles and of the similarity of the fissile breeding blanket required for either of these systems, it is not surprising that fusion-fission hybrid studies to date have concentrated almost exclusively on the D-T cycle. The concentration on D-T is strengthened by the conviction that fusion-fission hybrids are a near term option, a stepping stone on the way to a "pure" fusion power economy. It makes sense for this agenda to rely on relatively near term plasma physics capabilities.

Tritium production, energy recovery, and fissile fuel breeding in fusion reactors all take place in the blankets surrounding the plasma region. At the existing level of sophistication, there is little attention given to optimizing the fusion driver/breeding blanket combination; blankets and drivers are often considered in arbitrary combinations. The blanket concepts are conveniently divided into three categories: pure fusion electrical devices, hybrids, and symbiotic (fission-suppressed) systems.

Pure fusion devices have blankets which capture the neutron energy and breed makeup tritium. The product of these reactors is electric power and possibly tritium destined for additional fusion reactors.

Hybrids are essentially source driven, subcritical fission reactors. Their blankets are loaded with fertile and in some cases fissile material. The fission reactions induced by the 14.1 MeV neutrons and subsequent lower energy fissile captures in bred material lead to energy multiplication factors from 5 to as much as 50. The main product of the hybrid was initially considered to be electricity with fissile fuel as a more or less valuable byproduct. More recent design studies have emphasized fuel production over power multiplication. Such designs almost always emphasize the U-Pu fuel cycle because of the high fast fission cross-section of both U^{238} and Pu.

Symbiotic systems are intended to be exclusively fissile isotope producers although some versions produce sufficient electrical power for internal use and external sale. The design goal of these devices is either to eliminate fission reactions in the blanket, ("fission suppressed systems") or, at least, to separate a fission plate neutron multiplier from the region in which the fissile fuel is bred. In either case, the fissile fuel breeding region must be rapidly cycled to prevent *in situ* burning of the newly bred fuel.

Both hybrid and symbiotic systems can employ separate neutron multiplying and fertile zones; these functions are more easily separated in fusion reactor blankets than they are in fission reactor cores because of the source/blanket geometry and because of the properties of the 14.1 MeV source neutrons. These high energy neutrons are more valuable, in both the economic and technical sense, than the lower energy fission spectrum neutrons because they exceed the threshold of several important neutron multiplying reactions. The 14.1 MeV fission of U^{238} yields 4.5 neutrons (4.0 neutrons in the case of Th^{232}). In contrast, Pu^{239} or U^{235} yield slightly less than 2.5 neutrons per fission for thermal energy fission. Furthermore, the 14.1 MeV neutrons lie above the (n, 2n) and (n, 3n) reaction thresholds for U^{238} and Th^{232} and that of most structural materials. This facilitates additional neutron multiplication. Beryllium can be used as multiplier in both systems.

In fission plate systems, U^{238} with its large fast fission and (n, 2n) cross-sections is a clear choice for the plate itself. This choice takes advantage of the availability of the very large stocks of high purity U^{238} in diffusion plant tailings. The choice of the product fissile isotope depends on the fission reactor economy presumed to be in existence when the plant becomes operational. All three fissile isotopes, U^{233}, U^{235}, and Pu^{239}, have been used as fission reactor fuels. Of these isotopes, U^{233} has the largest η, defined as (neutrons produced)/(neutrons absorbed) in thermal or epithermal neutron spectra. For this reason it would be the fuel of choice for breeding or "near-breeding" thermal reactors if suitable recycling facilities were available.

In the event that fast breeders do not become either a necessary or acceptable solution to the uranium supply question, thermal breeders or high conversion ratio reactors offer an attractive way to extend fuel resources significantly. The actual number of fissile atoms which can be utilized per fissile atom introduced into the fuel cycle is proportional to $1/(1-C)$, where C is the internal conversion ratio. This assumes, of course, that an efficient reprocessing technology is available. Thus, for example, an HTGR with $C \simeq 0.95$ would yield 20 fission events per fissile atom introduced into the system (either mined or produced in a hybrid device), as compared to the 2.5 fission events per fissile atom introduced possible with current LWRs operating at $C \simeq 0.6$. If suitable reprocessing facilities are available, then U^{233} is probably the most desirable isotope to breed because it is a premium fuel in light water reactors, a superior performer in high conversion efficiency HWR or graphite moderated systems, and because there are other sources of Pu^{239}. If suitable reprocessing facilities for U^{233} are not available, then it may be more economical to breed U^{235} or even to breed reactivity into already fabricated fuel elements (the "value breeder" concept first advocated by W. B. Lewis in conjunction with linear accelerator bred neutrons).

Table V. Classes of Hybrids and Typical Performance Parameters

Parameter	Fast-fission U-Pu cycle	Fast-fission Th-U cycle	Fission-suppressed Th-U cycle
Energy multiplication, M	11.0	5.0	1.5
Fissile Breeding, F	1.5	0.8	0.7
F/M	0.14	0.16	0.47
Breeding rate:			
kg/MW_{fusion} year	6.6	3.5	3.1
$kg/MW_{blanket}$ year	0.77	0.88	2.57
$kg/MW_{nuclear}$ year	0.73	0.83	2.2

6. THE OPTIMUM FUEL FACTORY: OFF-DT

There have been many hybrid and symbiotic blanket designs. The goals, engineering assumptions, and level of design sophistication vary widely. Table V shows the general features of typical designs operating on different fuel cycles. The beryllium-multiplier, high F/m systems of Moir and Maniscalco have not been considered in detail here but are included in the Workshop proceedings.[5] Table VI is a more detailed summary of the reference designs used in recent EPRI/Westinghouse study.[1] Table VII compares various critical design parameters of

Table VI. Summary of Reference Hybrid Blanket Characteristics[a]

Blanket characteristic	Uranium fast fission	Thorium fast fission LM cooled[3]	He Cooled	Thorium fission suppressed
Fertile fuel zone				
Thickness, cm	16	25	16	80
Fertile fuel	UC^2	Th	Th	MS
Clad & Strt.	SS	HT9	SS	C
Moderator		Na		Be
Coolant	He	Li	He	Molten salt (MS)
Reflector zone				
Thickness, cm	10	72		
Reflector	Pb	C		
Structure	SS	HT9		
Coolant	He	Li		
Lithium fuel zone				
Thickness, cm	45	70	45	
Lithium fuel	Li_2O	Li	Li_2O	
Structure	SS	HT9	SS	
Coolant	He	Li	He	
f, $\frac{\text{net fissile atoms}}{\text{fusion neutron}}$	0.97	0.97	0.88	0.82
T, tritium breeding ratio	1.20	1.05	1.13	1.05
T & F	2.17	2.02	2.01	1.87
M, blanket energy multiplication	7.92	4.28	5.2	1.62
Exposure, MW-yrs/m^2	4.8	5.0	9.6	0.1
Fertile fuel density, g/cm^3	11.9	11.3	11.3	2.0
Coolant Δt, °C	250	150	250	50
Coolant exit temperature, °C	550	470	550	650
Coolant inlet pressure, MPa	6.50	0.10	6.50	0.69
Coolant pressure, drop., MPa	1.0	0.003	1.0	0.10

[a]From ref. 6.

hybrid systems with both fission and fusion electric devices.

Some of the design features are highly critical. Blanket power density, which varies linearly with the first wall neutron flux in any given system, is of paramount importance. Light water reactors typically operate at ~ 100 W/cm^3. Hybrid blankets with power densities which exceed this range might be expected to be difficult to cool. Fission plate systems at this power density would be particularly vulnerable to loss of coolant accidents. On the other hand, blankets with power densities significantly below the LWR range could be expected to be so large, and hence costly, that economical production of either power or fuel would be unlikely.

M is defined as the energy deposited in the blanket per fusion event divided by the fusion reaction energy. M is thus a measure of the energy amplification capabilities of the blanket. Designs with very high M values (i.e., ~ 50) must be examined thoroughly for the possibility of the blanket going critical either from isotopic buildup or unexpected geometrical deformations. F is the number of fissile atoms produced per fusion reaction. If cooling either during normal operation or during accident conditions is a dominant design constraint, then F/M is an important parameter. In general, a high value of F/M is desirable.

The choice of an "optimum" fusion-fission system depends on the weight given to these technical design considerations and to the institutional/environmental circumstances predicted for advent of these systems. In light of this complexity, the various panels and review groups that have been called together within the last few years to assess the status of fusion-fission have shown remarkable consistency. There is strong consensus that the "best" fusion-fission device is a stand-alone, remotely sited, fuel

Table VII. Nuclear Technology Technical Comparison[a]

	LWR	LMFBR	Hybrids			Fusion-elec.
Parameter			UFF	TFF	TFS	
Peak neutron flux (neutrons/cm^2/s)	10^{12}	7×10^{14}	2×10^{15}	10^{15}	9×10^{14}	9×10^{14}
Average neutron flux (neutrons/cm^2/s)	10^{11}	2×10^{14}	2×10^{14}	10^{13}	9×10^{13}	9×10^{13}
Neutron spectrum (meV)	0.1	1	14	14	14	14
Blanket/core energy density (w/cm^3)	98[(a)]	350[(a)]	350	100	10	7.5
Blanket/heavy metal inventory (kG)	28×10^3	23×10^3	8.7×10^2	6.7×10^2	3×10^3	0
Tritium inventory (kG)	0.00337		0.56	1.166	3.78	4.527
Blanket/core temp. (°C)	327	537	500[(g)]	500[(g)]	500[(g)]	500[(g)]
External driver power input (MW)	0	0	198[(e)]	254[(e)]	361[(e)]	380 (tokamak) 625 (Y-Y)
Blanket/core thickness (m)	3.4	3	0.76	0.575	0.63	0.4

[a]From ref. 6.

factory producing U^{233}. This fuel factory may, and probably should produce sufficient electrical power for its own needs but it should not be a supplier to the electrical grid. Systems that do supply the grid will benefit from the sale but questions of political and financial infrastructure will add enormous complications.

Therefore, to evaluate the potential of fissile fuel breeding systems, we consider a fusion reactor blanket, which, through use of U^{238} fission plates, is neutronically optimized for breeding relatively fission-product-free U^{233} and which has sufficient energy gain that stand-alone operation is possible with low Q fusion drivers. The technical and economic parameters of such a device are summarized in Appendix C. The design described in this Appendix uses a uranium fission plate neutron multiplier followed by a graphite reflected molten salt breeding region. The use of liquid breeder avoids the fabrication and reprocessing costs associated with solid fuel elements and facilitates rapid cycling of the breeding material to optimize isotopic impurity and minimize fission product contamination of the breeding zone. In the optimized design, the net breeding rate was 3.33 and the energy multiplication factor was 8.3. The design was shown to be operable under reasonable engineering constraints and has wide design margin (a LOCA analysis was performed) for the design value of 1 MW/m^2 incident on the first wall. The blanket design was performed independently of fusion driver choice (although a linear system is preferable) and has sufficient energy gain that stand-alone operation would be possible even with relatively low Q fusion drivers. A fusion driver with $Q = 0.27$ would result, for the efficiencies chosen, in stand-alone operation. This configuration with a $Q = 0.3$ linear driver will be designated OFF-DT.

7. OFF-DT VERSUS THE URANIUM MINE

Although it is possible to consider OFF-DT as part of a dedicated power system (i.e., by invoking such concepts as the support ratio), it is in fact nothing other than a supplier of highly enriched fissile fuel in direct competition with other sources of such fuel. Based on the postulate that there will be no shortage of fissile material in the intermediate term, if

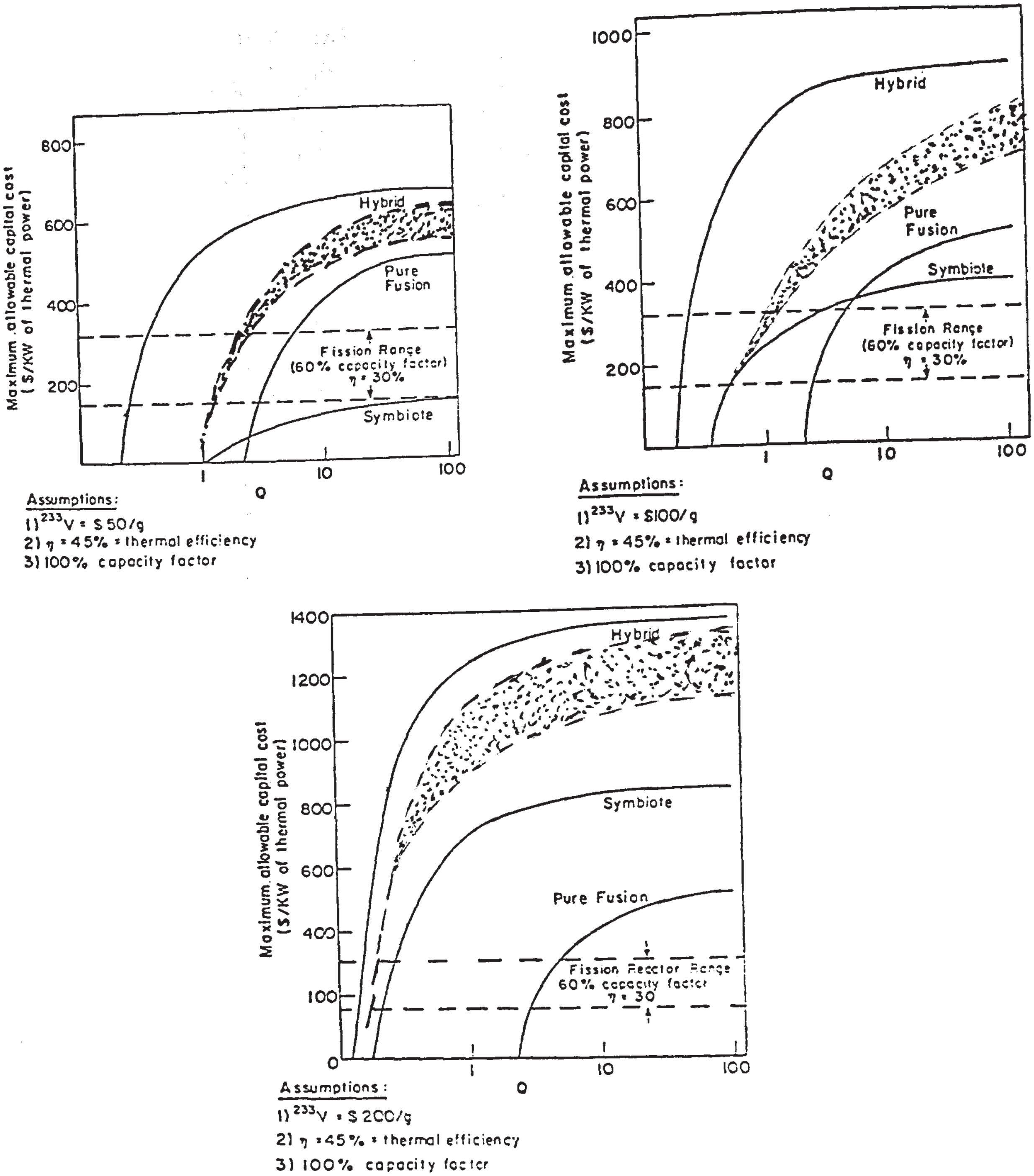


Fig. 5. Maximum allowable capital cost per kW of thermal power versus Q, the fusion reactor gain (est. 1978 fission reactor costs are also plotted). From ref. 9, Feasibility of ^{233}U Breeding in Deuterium-Tritium Fusion, Appendix E. The shaded areas are estimates of allowable cost for the Be-multiplier suppressed fission systems described by Moir and Maniscalco. These systems produce both fuel and electricity for off-site use.

at all, then the fusion-fission hybrid will succeed or fail simply on the basis of net fuel costs. The particular configuration chosen for Off-DT does produce highly enriched fuel and thus to some extent replaces an enrichment facility, but this consideration is of relatively minor import. Can OFF-DT produce fissile fuel more cheaply than a uranium mine?

The blanket/breeding zone system described in Appendix C has a net breeding ratio of 3.33 and an energy multiplication factor of 8.3. If we assume a 15% return on capital and price electricity at 20 mils/kwh, then we can derive the maximum capital cost allowable for both the blanket and fusion driver as a function of fusion driver Q and the value U^{238}.

These results are tabulated in Appendix C and plotted in Fig. 5. Figure 5 also shows the maximum allowable capital cost curves for pure fusion and symbiotic systems as well as the capital cost range for LWR fission reactors. We conclude from Fig. 5 that unless the price of uranium *substantially* exceeds the current value, the hybrid economics for systems with Q less than 3 are not appreciably superior to that of pure fusion devices and that symbiotic systems are definitely more tightly constrained. For U^{233} values in the range of \$100/g, symbiotes become comparable with pure fusion devices and the economics of hybrids appear markedly superior. Of course, this conclusion does not include the external costs of the hybrid.

Although most studies have considered the addition of fissile generated blankets to moderate to high Q fusion devices, the benefits of this combination are limited for reasonable fissile fuel prices. It is hard to image that the specific capital cost of the fusion driver/breeding blanket system could possibly compare to that of an LWR unless the fusion drivers were relatively small (approximately 25–50 MW), probably linear, steady-state, and extremely simple (both technologically and geometrically). *Presumably, these desirable features would be achieved as trade-off for low Q*. Obviously, if the driver had all these desirable characteristics and the additional potential of high Q operation, then it would be of prime interest to the "main line" fusion program. There is, of course, no such recognized device.

The low Q driver is of no use to the main line program, nor are any of the main line drivers, with the possible exception of the Tandem Mirror, reasonable candidates for fissile factory drivers. The national program has stated that development of pure fusion for central station power is its "highest goal" and has ignored, for the most part, the possibility of fissile fuel breeding. In light of the basic incompatibility of fusion drivers for central station power plants and fuel factories, this neglect is justified by the policy. However, there is at this time no federally supported research aimed at developing a suitable driver for OFF-DT.

8. SUMMARY

The reaction kinetics of deuterium-based fusion fuels imposes severe technological disadvantages on deuterium-based fusion reactors in comparison with fission-based systems as power sources for central station electric power plants. It has been assumed that these technological disadvantages would be outweighed by the potentially lower radioactive burden of fusion reactors and by the "essentially inexhaustible" nature of fusion fuels. Sufficient information is now in hand to allow this assumption to be tested. Although it is not possible to develop an absolute cost analysis of a fusion reactor for comparison with fission systems, it is possible by choosing limiting cases to determine relative rankings and scale factors. In the case of deuterium-based systems, this comparison is sufficient to suggest changes in policy.

In the following paper, we postulate the existence of the most plausible deuterium-based fusion reactor consistent with the fundamental physics of the fusion reaction itself. In other words, we assume that questions of plasma confinement and stability have all been satisfactorily answered. We then compare this best plausible reactor (called "OMR-DT") with existing fission reactors. We conclude that even for this optimistic case, OMR-DT is larger, more highly stressed, technologically more complicated and has substantially lower power density than existing fission reactors. If we also assume that fission reactors are politically acceptable, and that the true cost of U_3O_8 is not much greater than 10 times the current price, we conclude that deuterium-based fusion reactors cannot compete with fission reactors as central station power sources.

This conclusion is not totally orthogonal to current thought. The proponents of the existing fusion program claim only that a necessarily long and necessarily expensive development program will produce a fusion reactor with a *quantitative* advantage with respect to fission. We cannot argue that such an outcome is impossible; however, based on fundamental considerations, such quantitative advantage is unlikely. Nonetheless it may be decided that the possibility of such success is sufficient to justify substantial support for the fusion program. If so, this justification must be clearly projected to the public if the fusion program is to have the long term support required.

All the deuterium based fuel cycles share the same fundamental disadvantage; the large fraction of the reaction energy carried by neutrons. Neutrons are singularly intractable bits of matter; the problems of radiation damage, induced activity, shielding and reliance on thermal cycle appear unavoidable. Therefore, the only prospect of a substantial, *qualitative*, possibly near-term "breakthrough" must somehow utilize the neutron-free proton-based fuels. Fundamental considerations of relative cross-sections dictate that deuterium based systems operating in regimes compatible with the technological/economic requirements of central station power producers must be Maxwellian. Equally fundamental considerations of plasma physics and reactor technology have shown that confinement schemes developed for Maxwellian plasmas are inadequate for use with proton-based fuels. Therefore, we suggest that substantially more effort should be given to the study of the physics of highly non-Maxwellian plasmas with particular emphasis given to the neutron-free fuel cycles because only this combination offers any possibility of a qualitative, possibly near-term advance.

The "neutron-rich" nature of the deuterium based fusion cycles that places them at such disadvantage with respect to fission for central station power use can be turned into substantial advantage when the fusion reaction is viewed as a fissile fuel producer. In light of the fact that there will be no absolute shortage of uranium, fusion produced fuels will compete purely on a basis of price. The optimum fusion driver for this purpose is not likely to be the optimum driver for a central station power plant. The development of such an optimized driver should be considered as a possible short term goal for the fusion program. Support of such development is not consistent with current policy.

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APPENDIX A

The binding energy versus mass relationship shows that literally hundreds of fusion reactions are capable in principle (i.e., from simple consideration of nuclear Q values) of yielding net power. Many of these reactions survive tighter selection criteria; the margins by which they surpass these criteria enable us to rank order the various possible fuel cycles. We follow here a scheme suggested by Crocker in 1970. A potential fusion reactor fuel cycle must satisfy criteria of (1) physics—energy balance, and (2) technology—energy density, scale size.

The power density can be written in terms of the rate limiting reaction (reactants 1 and 2) of the cycle $[W_n = n_1 n_2 \overline{\sigma v}_{12} Q_t]$, where Q_t is the total nuclear output of the cycle. Similarly, we may write the volumetric power loss as

$$W_L = W(\text{particle}) + W(\text{thermal cond.}) + w(\text{Bremss}) + W(\text{cyclotron})$$

If the volumetric power generation is balanced against the volumetric loss, and all losses other than Bremsstrahlung are considered negligible or are recoverable at 100% efficiency, $\overline{\sigma v}$ for energy balance is

$$\overline{\sigma v}^{P} \geqslant \frac{1\times 10^{-19} Z_B^3 T_e^{1/2}}{Q_t} \ \text{m}^3/\text{s for } T_e \& Q_t \text{ in eV} \quad (1)$$

where Z_B is the effective ionic charge for electron Bremsstrahlung. Equation (1) is simply an idealized ignition condition for 100% efficient energy recovery and recycle. The neglect of cyclotron radiation deserves particular attention because this term is important for the high temperature, low density conditions which prevail for most of the so-called advanced fuel cycles.

It is difficult to derive an equivalent absolute *technological* criterion, but reasonable assumptions regarding technological limitations yield a result adequate for present discussion. If one assumes $\beta = 1$ and temperature equilibrium between the ion species, then it is easy to show that the reactivity is related to the minimal acceptable power density W_* and the maximum attainable magnetic field intensity B_* by

$$\frac{\overline{\sigma v}^{T} \geqslant 1.0\times 10^{30} W_*}{B_*^4 \dfrac{(T_i + \bar{Z} T_e)^2}{Q_t} \ \text{m}^3/\text{s for } T_e Q_t \text{in eV}} \quad (2)$$

where W_* is expressed in W/m^2 and B_* in teslas. $\bar{Z}$ is the average ionic charge. For reactions at low T_i, the density is such that $T_i \approx T_e$ but at large values of T_i ($> \approx 100$–500 keV), the electron temperature is clamped by both Bremsstrahlung and cyclotron radiation (when present). The lower limit to electron temperature is set by considerations of ion-electron thermal transfer. At values of T_e much less than 20 keV, the ion-electron thermal transport and consequent ion cooling is inconsistent with the energy balance assumptions of $\overline{\sigma v}^P$. If $W_* = 10^6$ W/m^3 and $B_* = 25$ T, then

$$\frac{\overline{\sigma v}^P}{\overline{\sigma v}^T} = 3.9\times 10^{10}\frac{Z_B^3 T_e^{1/2}}{\left(T_i + \bar{Z} T_e\right)^2} \tag{3}$$

For all but the DT cycle, it is safe to assume that T_e is clamped at 100 keV and $T_i \gg T_e$. For these assumptions

$$\frac{\overline{\sigma v}^P}{\overline{\sigma v}^T} \simeq 4\times 10^{12}\frac{Z_B^3}{T_i^2} \tag{4}$$

Since the larger of $\overline{\sigma v}^P$ and $\overline{\sigma v}^T$ is the limiting constraint, we arrive at the somewhat surprising conclusion that for T_i less than 5–10 MeV (taking Z_B into account), $\overline{\sigma v}^P$ is the limiting criterion.

It is possible now to rank order possible fuel cycles by the ratio of the left- to right-hand sides of inequality (1), i.e., by the approximate value of $\overline{\sigma v}$ at the peak of the cross-section divided by the value of $\overline{\sigma v}^P$ (see Table II). The physics advantage of the D-T reaction is obvious. There are several technological disadvantages: T breeding and concomitant lithium requirement, 14 MeV neutrons with attendant radiation damage and structural activation, forced reliance on the thermal power cycle because the bulk of the energy is carried by neutrons, the need to maintain substantial inventories of tritium. However, these well-known disadvantages are not entirely eliminated in the runner-up D-D cycle, which generates both tritium and neutrons. In fact the neutron flux in strongly confined, or tritium recycling, D-D systems is comparable in both damage potential and power weighted intensity to that in the D-T reactor. The only qualitative difference in the D-D cycle is elimination of the tritium breeding requirement. This is certainly an important consideration, but it does not outweigh the two order of magnitude difference in constant-β reactivity (i.e., in peak value of $\overline{\sigma v}/T^2$).

The other reactions listed in the Table II do have potentially important technological advantages, but the margin over the idealized physics ignition requirements is disturbingly small. In effect, the assumptions of negligible losses and high efficiency energy recovery must be realized. Values of $n\tau$ and β substantially larger (factors exceeding 10 for $n\tau$) than postulated for on D-T systems are required. Additionally, these cycles, because of their reliance upon very high particle energies and high voltages for direct recovery, call upon extremely difficult ultra high voltage (UHV) technology. The difficulties experienced in implementation of UHV power transmission for trunk power lines affords ample evidence of the difficulty in harnessing megavolt potentials in nonlaboratory uses. Reliable 5–10 MV operation in the radiation field of a fusion reactor looks to be a particularly difficult and often overlooked technological requirement.

Although it is hard to prove a technological impossibility, the simple arguments above imply a very high order of technological implausibility for the advanced fuel cycles, at least when considered in Maxwellian or near-Maxwellian plasma systems. An effort to identify non-Maxwellian, possibly "non-plasma" schemes to exploit the advanced fuel cycles is certainly justified, but until plausible schemes can be identified and analyzed, the cost of such studies should be small.

APPENDIX B: DESIGN CRITERIA

An acceptable fusion reactor must simultaneously satisfy requirements imposed by physics, technology, and economics. The physics requirements are compatible with many different conceptual confinement schemes ($n\tau$ is the dominant "physics criterion"), but in reactor embodiment, many of the confinement schemes capable of achieving satisfactory $n\tau$ values have uneconomically low power densities, are inherently extremely large, employ unrealistic technological systems, etc.

The limits of technological possibility are much better known and more tightly drawn than those of physics. Operating tensile stress radiation damage limits, maximum operating temperatures, superconductor shielding requirements, etc., are well known because they have been studied in many other contexts. Relaxation of those limits would have immediate commercial consequences and, therefore, much time and effort has gone into their study, much more than has been expended in the fusion program.

These technological limitations are extremely complex, multiply constrained, and seem unlikely to change substantially. Although economic limits have proved to be substantially more flexible than previously assumed, it is nonetheless becoming clear that an economic system apparently capable of coping with the energy costs of heavy crude and tar sand development will not require the deployment of one of the "inexhaustible" energy resources within the next 30 to 50 years. This consideration, along with the existence of a well-developed fission reactor system, limits the cost of conceptual fusion reactor designs, unless overweighing political or environmental concerns become dominant.

Every large scale fusion effort has seen the wisdom of using conceptual design studies of proposed reactor schemes to guide their research and development programs. In the U.S. such studies have ranged from the early D-Stellarator reactor of Lyman Spitzer to the detailed series of tokamak designs by R. Conn[1] and his group at the University of Wisconsin. Such studies have played a particularly prominent role in the history of the theta pinch and mirror programs. Similar studies have been carried out in other national programs, but usually in somewhat less detail; the early tokamak designs by Golovin, et al.,[2] the torsatron reactors by Iiyoshi,[3] and the reversed field pinch design by the Culham group[4] are typical. These reactor studies form a rich set of "design experiments"; it is possible to draw from the set of such studies a list of requirements that must be substantially satisfied if any given reactor concept is to be commercially acceptable. These engineering requirements are analogous to the list of physics precepts (e.g., desirability of short magnetic connection lengths, avoidance of large drift velocities, minimization of velocity space gradients, etc.) that have evolved over the last two decades of physics experiments and theoretical developments. Table III gives the engineering desiderata, which are arranged in approximate order of importance with the more important requirements listed first.

Steady-State Operation

Steady-state operation has proved to be the most important design consideration in many large scale engineering facilities (with large cyclic stress fluctuations becoming the lifetime limiting parameters). Because of the particular requirements of a fusion reactor, this constraint is much more important than it would be in an equivalent fossil fueled plant or even a fission reactor.

Effects on Materials

Because the fusion fuel is at high temperature and separated from the structural material, every fusion reactor configuration possesses a very highly stressed "first wall." Both the thermal loading and the radiation damage peaks at this front wall and drops rapidly with distance into the blanket. Every conceptual design finds economics primarily determined by the allowable first wall lifetime. This lifetime is determined in large measure by the magnitude and time history of cyclic stresses.

The lifetime limiting effects, both brittle failure and enhanced creep, are aggravated by radiation damage, which both embrittles the material and augments the creep. An upper limit for the lifetime can be estimated by neglecting the effects of radiation damage. Such an approach is exemplified in the study done by J. Meyer.[8] Meyer applies the ASME code of values for uniradiated material to a simplified reactor first wall model. His report is general scaling laws specific examples will suffice here. For example, in comparison with steady state operation at the same average power level, the lifetime of a type 316 stainless steel wall at 600°C is reduced by a factor of nearly 40 for repeating 2 h burn cycles at 5 MW/m^2 and by a factor of 200 for a repeated 10 min burn cycle at only 3 MW/m^2. The lifetime is measured in days for 10 min cycles of 5 MW/m^2. A penalty this severe in the design limiting parameter cannot be tolerated.

Effect on Cycle Efficiency

Plasma energy losses are usually highest during the startup phase (confinement is highly collisional, impurities are high), and energy injection is often quite inefficient. The energy stored in the plasma and associated magnetic fields is comparable to the gross thermal output for several to 10 s operation and is difficult to recover with high efficiency. Therefore, unless the burn phase is appreciably longer than the startup phase, the net cycle efficiency can be severely reduced.

Effect on Capital Cost

The energy injection equipment operates at high instantaneous power levels for periods of time short compared to the burn cycle, but usually long enough to require complex design. The injection equipment will, of course, require energy storage equivalent to several seconds thermal output or else provisions must be made to allow substantial direct drain from the grid. Because of the importance of thermal stresses, it is imperative that the equipment work with high reliability, especially in a rapidly cycling reactor because a missed shot will cause substantially larger thermal fluctuations in the balance of the plant.

Diverted

Divertors and magnetic limiters have been incorporated in most reactor designs despite the complexity they often introduce because they serve a multiplicity of functions. Although it is possible in some cases to "design around" one or another of these functions, it does not appear possible to achieve a self-consistent design without divertors. The divertor influences:

1. *First wall heat load*. The first wall in an undiverted system absorbs approximately 10% of the direct neutron throughput (possibly 8% of the total thermal power) and all of the conduction and radiation power (greater than 20% of the total). An efficient limiter can reduce the first wall heat flux by more than a factor of two.
2. *Impurity control and sputter protection*
3. *Gas recycle control*. The particle confinement times and maximum permissible plasma edge pressure in a reactor are still undetermined. However, even for particle confinement times on the order of several energy confinement times and allowable edge pressures of order 10^{-3} central plasma density, simple scaling calculations show that neutral gas pumping ducts would require an area comparable to or greater than the total first wall area. Plasma pumping via divertors is the only possible way of removing the gas load.
4. *Magnetic limiters*. An active divertor simplifies startup requirements, reduces power demand during startup, and offers the possibility of both thermal stability control and wall protection during unplanned shutdowns.

Ignited

Reactor system studies indicate that the provision of substantial power *input* to an operating reactor reduces reliability, system efficiency, and economic desirability. Much of this adverse impact results from the multiplier effect that modest conversion efficiency has on recirculating power in even moderate Q systems ($Q \approx 6$–12). There are additional technological difficulties associated with the heating of an operating reactor. Neutral beam lines allow neutron streaming (greatly increasing shielding requirements), are very sensitive to direct neutron streaming, and are subject to power density limits because of breakdown in severe radiation environments. Low frequency rf systems requiring internal antennae are subject to breakdown as well as severe neutron heat loading.

"Toroidal"

The mirror and theta pinch programs have conclusively demonstrated that energy and particle confinement times must be substantially ($< 10\times$) greater than the classical collisional angular relaxation time. This requirement implies that "velocity space confinement" is not adequate and that the particle must be constrained to either diffuse across a real spatial barrier in toroidal devices or, equivalently, over a potential barrier as in the tandem mirror reactor.

Moderate Power Density

D-D and D-T reactors will both require blankets of at least 1 m thickness. Plasma radii will be at least 2 m and probably significantly more. Technological considerations of power peaking and other difficulties associated with high powered systems in which the radius of curvature is comparable to unit thickness all imply a total radial scale size of 3–5 m. If the average power density in a fusion reactor is to be comparable to that in other highly developed energy sources (1–3 MW/m^3), then the power density *in the plasma* must be in the range of 5–10 MW/m^3.

Disruption Free

The energy confinement time in an operating reactor will be on the order of several seconds. The kinetic energy content will thus be comparable to

25–50% of the energy produced per second. The magnetic energy content will vary from values roughly comparable to the plasma kinetic energy to approximately 5–10 times greater. No front wall design consistent with thermal gradient and system neutronics constraints is capable of surviving the deposition of this much energy on a small fraction of wall area in a short time. Major disruptions are troublesome in existing experiments, will probably be survivable in specially armored $Q \simeq 1$ experiments (TFTR), and will be intolerable in a full scale operating reactor.

Modular

A fusion reactor will become too radioactive for hands-on maintenance after only a few hours operation. After extended periods of operation, it will require extensive shielding even for remote maintenance. The necessity of providing for remote maintenance is often acknowledged, but the difficulty of actually accomplishing this is underestimated. The problems associated with refueling fission reactors (in which the task is the well-defined planned replacement of vertical fuel rods in a vertical cylindrical assembly using a device integrated into the total system design) are illustrative. Manipulations more complex than simple vertical or radial motion of moderate size components are inconsistent with claims to modularity.

Moderately Stressed

It is a truism that an operating reactor must have substantial margins of safety with respect to magnetically imposed stresses. It is more important, however, that the stresses imposed by off-normal operation (e.g., a single failed magnet) be tolerable without requiring active control. A system which depends on large counterbalancing forces is subject to catastrophic failure in the event of relatively minor component failures.

"Safe"

Because of the decoupling of the plasma and structure and because the principal radioactive burden of a fusion reactor resides in induced radioactivity of structural components rather than in unavoidable fusion products, it is possible to design fusion reactors whose total radioactive burden, measured in units of biological hazard potential, can range over 3 orders of magnitude. The worst case is comparable to an equivalent light water reactor. The long term biological hazard potential varies over an even larger range, but it is always substantially lower than equivalent fission systems. The important consideration in determining the risk is the biological hazard potential multiplied by the probability of release. An acceptable fusion reactor should not aggravate the possibility of substantial release. For example, a reactor design which requires high pressure water cooling of the first wall and a liquid lithium tritium-generating blanket would almost certainly be unacceptable.

Moderate Size

The combination of physics, wall loading, and average power constraints imposed on potential fusion reactors indicates that they will almost certainly be very large devices. Although there is no question that modest (≈ 100 MW$_e$) units would be highly desirable, the question is likely to be that of maximum acceptable size. The answer to this question will, of course, depend on the reliability of the units; highly reliable units of approximately 800–1500 MW$_e$ size would clearly be acceptable, units of three times that size would strain the distribution network expected to be in place, and units five times larger would almost certainly be unacceptable. It is quite likely that smaller units, noneconomical but subsidized as a development incentive, would be first to be deployed.

APPENDIX C

In the study by A. Cook and L. M. Lidsky[9] an optimized fusion reactor blanket consisting of a lithium cooled, U^{238}, neutron multiplying region followed by a molten salt region breeder, U^{233}, is described and its economic parameters, neutronic performance, and safety aspects examined. It is shown that a fissile isotope breeding blanket is neutronically feasible and that the particular design here is operable under realistic engineering constraints. The blanket breeds 1.06 atoms of tritium and 1.20 atoms of Pu^{239} and U^{233} per incident neutron. The total energy deposited in the blanket is 8.3 times the incident fusion neutron energy. Based on an analysis

Table C1. Design Parameters of the Blanket[a]

First wall:	
Stainless steel (modeled as 100% Fe for the neutronics)	
density = 7.92 g/cc	
thickness = 0.5 cm	
Multiplier region:	
i. Composition-volume fractions	
Uranium, 63%	
Lithium, 24%	
Clad, 13%	
(stainless steel)	
ii. Densities	
Uranium (100% ^{238}U) = 19 g/cc	
Li (70% ^{6}Li–30% ^{7}Li	
mole percent) = 0.43 g/cc	
Stainless steel = 7.94 g/cc	
iii. Dimensions	
Coolant channels, 0.53 cm (the end channels are 0.26 cm)	
Clad thickness, 0.14 cm	
U slab thickness, 1.386 cm	
Molten salt region:	
i. Bounded by two 0.5 cm thick hastelloy walls which were modeled as 100% Fe for the neutronics	
ii. Salt composition-mole %	
LiF(depleted in ^{6}Li), 71%	
ThF_4, 24%	
BeF_2, 2%	
iii. Dimension, 57 cm thick	
iv. Density, 277 lb/ft^3	
Graphite reflector:	
Density = 1.60 g/cc	
Thickness = 30 cm	

[a]From ref. 9.

of fission reactor fuel cycle economics, including mining, fabricating, and reprocessing costs and projections of fissile isotope values, it is concluded that blankets consisting of fertile isotopes in solid form are economically unattractive. The fuel cycle costs of a molten salt system are significantly lower than those of a solid fuel system, and hence the molten salt system is shown to have the potential of breeding fissile isotopes economically.

Based on these results it is concluded that a symbiotic ("fission-free" fissile breeding) fusion reactor would be economical only if its construction costs per MW thermal of plant capacity were substantially lower than those of conventional fission reactors or, possibly, if they were integrated into a system particularly engineered to utilize U^{233} in very high conversion efficiency reactors. Hybrid designs have greater potential of being economic devices especially if a large fraction of the generated heat is converted to electricity. The reliability requirements are, however, very much more stringent, and most analyses consider this a strongly negative factor. It is estimated for a U^{233} value of $50/g and electric power prices of 20mils/kwh that the maximum allowable capital cost would be $150/kW of thermal capacity for a sym-

Table C2. Economic Parameters of the Blanket[a]

Neutron fluence	1.4×10^{25} n/m^2/yr
^{233}U bred	182,146 g/yr
^{239}Pu bred	209,537 g/yr
Heat produced	$2{,}285 \times 10^6$ kwh/yr
Equivalent electric power	$1{,}030 \times 10^6$ kwh/yr
Total salt inventory	2548.6 ft^3
which costs	$\$5.382 \times 10^6$
Make up ThF_4	182 kg/yr
which costs	$3,913/yr
Salt processing costs	$\$300 \times 10^3$/yr
Total ^{238}U inventory	43×10^3/yr
Multiplier region fuel cycle cost in $/kg[b]	$(275.75 + (17.46 - 4.88v)n)$

[a]From ref. 9.
[b]Note this value is slightly different from the value computed earlier since the breeding rate is slightly different.

Table C3. Economic Assessment of Various Modes of Reactor Operation[a]

Q	Revenues/yr Fissile	Electric	Maximum allowed capital cost = c	Maximum allowed c/kW of thermal power
	Mode: symbiotic Assumption: $^{233}V = \$50/g$			
1	5.506 m	−5.506 m	0	0
10	5.506 m	−0.5506 m	33.035 m	$127
100	5.506 m	−0.05507 m	36.34 m	$139
	Mode: symbiotic Assumption: $^{233}V = \$100/g$			
0.38	14.606 m	−14.606 m	0	0
1	14.606 m	−5.506 m	60 m	$230
10	14.606 m	−0.5506 m	94 m	$360
100	14.606 m	−0.05506 m	97 m	$372
	Mode: symbiotic Assumption: $^{233}V = \$200/g$			
0.17	32.8 m	−32.8 m	0	0
1	32.8 m	−5.506 m	182 m	$698
10	32.8 m	−0.5506 m	215 m	$825
100	32.8	−0.05506 m	218 m	$836
	Mode: hybrid Assumption: $^{233}V = \$50/g$			
0.27	5.506 m	0	36.7 m	$140
1	5.506 m	15.06 m	137.1 m	$526
10	5.506 m	20.02 m	170.2 m	$653
100	5.506 m	20.52 m	173.5 m	$665
	Mode: hybrid Assumption: $^{233}V = \$100/g$			
0.27	14.606 m	0	97.4 m	$374
1	14.606 m	15.063 m	198 m	$760
10	14.606 m	20.02 m	346 m	$885
100	14.606 m	20.52 m	351 m	$898
	Mode: hybrid Assumption: $^{233}V = \$200/g$			
0.27	32.8 m	0	291 m	$839
1	32.8 m	15.063 m	319 m	$1224
10	32.8	20.02 m	352 m	$1350
100	32.8 m	20.52 m	355 m	$1363
	Mode: Pure fusion Assumption: no energy multiplication, no fissile breeding			
2.22	0	0	0	0
10	0	2.41 m	16.07 m	$409
100	0	3.03 m	20.2 m	$514

[a]From ref. 9.

biote and $650/kW for a hybrid if the breeder were to be analyzed on a stand-alone basis. The design and economic parameters for an optimized fusion reactor blanket are given in Tables C1 and C2. Table C3 gives an economic assessment of various modes of reactor operation.

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