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The Tokamak: Model T Fusion Reactor

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Fusion power holds promise of using cheap, abundant fuel and of being relatively low in radiological hazard potential. To date, experimental results indicate that scientific progress has been greater in the tokamak approach than in other magnetic confinement fusion concepts. For this reason, the tokamak concept enjoys generous financial support and is the main thrust of the American and world fusion programs. But the tokamak scheme for power generation is not without obstacles, which have led to concerns raised by fusion researchers (*1*) and representatives of the electric utility industry (*2*) about the commercial feasibility of tokamak power. These concerns, which were well articulated by Metz (*3, 4*), boil down to two questions.

1) Can the tokamak concept lead to an economically competitive power system?

2) Is the cost of the program required for developing the tokamak concept acceptable?

Issues raised regarding the economic potential of the tokamak concept include reactor size, power output, plant costs, and plant reliability and maintainability. Associated with these issues was the concern that a demonstration program would require a succession of increasingly large and costly devices. Metz's description of the scope of these issues was based on an examination of tokamak reactor design studies carried out between 1971 and 1976, and in particular on the tokamak reactor design study called UWMAK I evolved by the University of Wisconsin Fusion Technology Group in the early 1970's. These early studies represented initial attempts to identify problems that might be encountered in tokamak reactors. They were not intended to design economically competitive systems. Subsequent studies have benefited from the early studies and have focused more on the requirements for economic competitiveness. Moreover, they reflect scientific and technological advances which have been made over the past several years, primarily as a result of an associated, significant increase in the U.S. fusion research budget (see Fig. 1). The more recent studies indicate that many of the problems identified earlier are tractable and therefore should not be viewed as insurmountable barriers to the economic potential and development of tokamak reactors.

Summary. During the past several years there have been significant scientific and technological advances related to the tokamak magnetic confinement scheme. These are summarized in the context of a recent tokamak reactor design study which emphasizes reduced size, higher power density, and enhanced plant reliability and maintainability relative to earlier tokamak reactor design studies. The direct plant cost of the proposed reactor is estimated to be in the range \$1000 to \$1500 per electrical kilowatt. A three-phase strategy for demonstrating tokamak fusion power generation at a committed site is outlined. It is estimated that implementation of the three-phase program would require about 20 years and a total escalated expenditure of \$10 billion to \$15 billion. The tokamak power plant described here is not viewed as definitive but rather as a point of departure in the development of a plan to demonstrate tokamak power generation.

At Oak Ridge National Laboratory (ORNL), for example, we have evolved a conceptual reactor design which suggests that tokamak power systems can be economical without being monstrously large (both in electrical output and physical size), that their plant costs need not greatly exceed those of other advanced energy systems such as liquid-metal fast breeder reactors (LMFBR's) and solar electric plants, and that their reliability and maintainability should not be unacceptably low. Moreover, we have been able to identify the elements of a strategy for demonstrating significant power generation with acceptable program costs. This is not to say that our conception of the tokamak power system should be regarded as definitive. Viewed from the perspective of the ultimate tokamak fusion power system, our present concepts will undoubtedly appear as primitive as a Model T Ford compared to a modern automobile. However, just as that Model T bears a recognizable relation to its distant descendant, we believe that our present design concept of the Model T tokamak reactor can provide a realistic base for assessing the nature of tokamak fusion power systems of the future.

In this article, we will first describe the essential elements of the commercial tokamak power plant that we envision. Then we will examine the scientific and technological basis (that is, the scientific and technological progress during the past few years) for this view of a commercial tokamak power plant. Next we will consider the economic potential of tokamak power plants. Finally, we will suggest the elements of a strategy to demonstrate tokamak power generation.

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Commercial Plant

As we envision it (5), the commercial power plant would consist of multiple (for example, two to five) tokamak reactor units sharing a number of common elements, including such key elements as the pulsed power supplies for driving the plasma current and the pulsed power supplies that provide the plasma supplementary heating. Sharing these elements among multiple units would save about 15 percent in the plant costs of fusion power relative to the case where each unit has its own committed power supplies.

Each reactor unit would generate between 500 and 1000 megawatts electric (MWe) of output power. The precise value of the electrical output of each unit would be determined by both plasma physics and cost optimization considerations. Note that the large output powers characteristic of some earlier tokamak reactor studies reflected the assumption that single units of large electrical output would be desired by the utilities. Our studies show that single units of large electrical output should not be viewed as an inherent characteristic of tokamak reactors. For this discussion we will assume that an electrical utility would like to develop a particular site for a total capacity of 1500 MWe. This would be accomplished by a plant employing two tokamak reactor units, each generating 750 MWe.

Reactor unit. The major components of each tokamak reactor unit are illustrated in Fig. 2 and the essential system parameters are summarized in Table 1. The toroidal plasma core has a minor radius of about 1.5 meters and a major radius of about 6 meters. By comparison, note that the UWMAK I design had a toroidal plasma core with a minor radius of about 5 m and a major radius of about 13 m. As indicated in Fig. 2, the plasma cross section is somewhat elongated; in this case the ratio of the plasma height to width is about 1.6. The elongation is expected to result in improved plasma performance compared to the case of a circular plasma cross section. The plasma would operate with a density of $\sim 2 \times 10^{14}$ particles per cubic centimeter, a confinement time of about 1 second, and a temperature of $\sim 10^{8}$ K. It would be brought to the operating temperature in about 10 seconds, using ~ 50 to 100 MW of neutral beam injection power. Fueling would be accomplished by injecting solid fuel pellets into the plasma, and spent fuel removal would be accomplished by guiding charged particles out of the plasma chamber along diverted magnetic field lines generated by "divertor" coils.

As noted by Metz (4), the size and cost of the tokamak power system is a strong function of the fusion power density. The fusion power density in the plasma chamber of each unit is about 5.3 MW/m^3 as compared with about 0.8 MW/m^3 in the UWMAK I design. The plasma volume of each tokamak reactor unit is about 15 times smaller than that in the UWMAK I design, and the thermal power of each unit is about 45 percent of that in the UWMAK I design. We emphasize that these comparisons with the UWMAK I design are not made to criticize that design, which, as Rose and Feitag (6) said, was a problem-finder rather than a problem-solver. Instead, we wish to point to the significant reductions in projected size and power output which have occurred in the area of tokamak reactor design as a result of scientific and technological progress.

Surrounding the plasma would be a blanket whose main purpose is to recover the fusion energy as heat and to breed new tritium for fueling the reactor. The blanket first wall and structure would be made of an austenitic stainless steel. The amount of steel in the blanket is about an order of magnitude less than that in the blanket of the UWMAK I design. The optimum structural alloy for the fusion reactor environment is yet to be identified, and a materials program to develop such an alloy is now under way. Nevertheless, preliminary results on the performance of austenitic stainless steels under simulated fusion neutron irradiations give us considerable optimism about the prospects for such alloys. These results will be considered further in our discussion of economic potential.

At present, there are several promising coolants for the blanket. One is a salt consisting of sodium and potassium nitrates and nitrites. These salts have been used extensively as heat transfer media in the petroleum and chemical processing industries for more than 30 years. They have relatively low melting points (~ 150°C) and are relatively inexpensive ($\sim$\$1 per kilogram), and their thermal stability and compatibility with iron-base alloys seem acceptable to temperatures around 500°C. A development program will be required to determine the ultimate acceptability of such salts in a fusion reactor environment. Other coolants under consideration are helium gas and liquid lithium. The breeding medium must consist of lithium in some form since only lithium offers any promise for

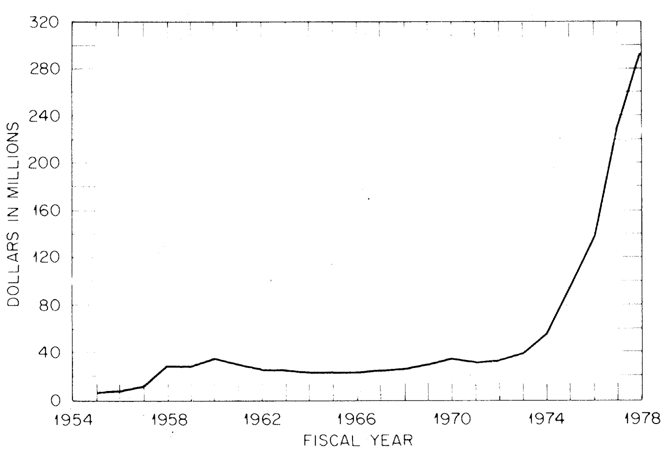

Fig. 1. The U.S. magnetic fusion energy research budget during the past 24 years in absolute dollars. If the effects of escalation were included, the drop in the budget during the period ~ 1960 to 1970 would be more dramatic. The significant increase in the budget which began in the middle 1970's is yielding substantial scientific and technological progress (see text).

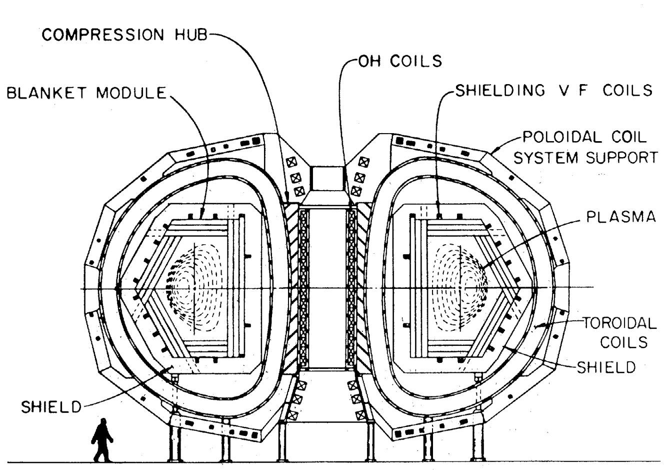

Fig. 2. Cross section through a tokamak reactor as envisioned in the ORNL study (5). The toroidal plasma core has a minor radius of ~ 1.5 m and a major radius of ~ 6 m. The superconducting toroidal field coils have inside dimensions of ~ 7 by 10 m. Such a reactor would produce ~ 750 MWe. Abbreviations: *OH*, ohmic heating; *VF*, vertical field.

tritium regeneration through neutron interactions. At present, liquid lithium seems to offer the greatest potential as the breeding material.

The blanket region is ~0.8 m thick and is surrounded by a shielding region ~0.7 m thick whose main purpose is to protect the superconducting toroidal magnet coils. A number of materials could be used for the shielding region, including iron, boron carbide, and lead. The toroidal field coils employ niobium titanium as the superconducting material. The coils are somewhat D-shaped with inside dimensions of ~7 by 10 m, approximately half the inner dimensions of the coils for the UWMAK I design (~15 by 21 m).

Balance of the plant. Each tokamak reactor unit would be housed in a circular containment building ~50 m tall and ~60 m in diameter. The volume of this containment building would be almost one order of magnitude less than the volume of the containment building suggested for the UWMAK I reactor and about the same as that of the containment building of a fission reactor with comparable output power.

Each reactor unit would have its own power conversion system. This would consist of a primary coolant loop and an intermediate salt loop coupled to a conventional steam system and turbine. The main purpose of the intermediate heat transport loop is to prevent pressurization of the blanket by high-pressure steam should there be a tube leak in the steam generator. Assuming a primary loop exit temperature of around 450°C, a steam-cycle thermodynamic efficiency of ~35 percent would be achieved with this power conversion system. Each reactor unit would operate with a burn time of ~20 minutes and a down time of ~1 minute, thus achieving an overall duty factor of ~95 percent. Salt storage tanks coupled to the intermediate heat transport loop would provide the necessary thermal energy storage to ensure continuous power to the turbine plant during the entire operating cycle.

The commonly shared power supplies for initiating the plasma current and providing the plasma heating would operate in a phased fashion for the two tokamak units. Note that these power supplies operate in a pulsed mode and need to be operating for only 10 seconds at the start of the initial plasma breakdown and heating phase. For the assumed conditions, two tokamak units and a 21-minute cycle time, these pulsed power supplies would have to be pulsed once approximately every 10 minutes. It appears that the primary energy storage requirements for these power supplies can be satisfied with motor-generator flywheel sets that are similar in rating and capacity to those being proposed for the Tokamak Fusion Test Reactor (TFTR) to be built at the Princeton Plasma Physics Laboratory.

Table 1. Essential system parameters of the ORNL reactor design and the UWMAK I design.

Parameter	ORNL	UWMAK I
Plasma radius (m)	1.5	5
Major radius of torus (m)	6	13
Plasma shape	Elongated 1.6/1	Circular
Neutral beam injection energy (keV)	150–300	500
Plasma beta (percent)	8	3
Fusion power density (MW/m^3)	5.3	0.8
Neutron wall loading (MW/m^2)	3.0	1.25
Cycle time (seconds)	1260	5800
Duty factor (percent)	95	93
Average thermal power (MW)	2150	5000
Average electrical output (MW)	750	1500

Scientific and Technical Basis

The tokamak reactor described above is attractive on the basis of electrical output and physical size; the question of economic potential will be considered later. At this point we will address the fundamental question of whether the reactor operating characteristics are consistent with the basic physics of tokamak plasmas. A secondary question, assuming a positive answer to the first, is whether these operating characteristics can be achieved using practical technologies. At this stage of the fusion research and development program, it is not possible to give a definitive answer to either of these questions. However, it can be said that operation of a tokamak reactor in the proposed regime is consistent with our *present* theoretical understanding of tokamak plasma behavior, and that current tokamak experiments are producing results pointing toward the proposed regime of operation. Furthermore, the key technologies used to advance the plasma performance in these experiments seem to be capable of extrapolation to the requirements of reactors.

Progress toward the reactor regime. A tokamak reactor plasma will operate in the "ignition" regime. In this regime the fusion energy deposited within the plasma exceeds the energy lost from the plasma by heat conduction and radiation; thus, the fusion plasma can be self-sustaining. To operate in the reactor (that is, ignition) regime with a deuterium-tritium-burning plasma, the plasma ion temperature must exceed about 40×10^6 K and the plasma energy confinement, measured by the product of plasma density and energy confinement time, must exceed about 6×10^{13} cm^{-3} sec. Note that the plasma in the proposed reactor operates with a temperature of about 100×10^6 K and with an energy confinement of about 2×10^{14} cm^{-3} sec.

Figure 3 illustrates progress toward the reactor regime within the U.S. toka-

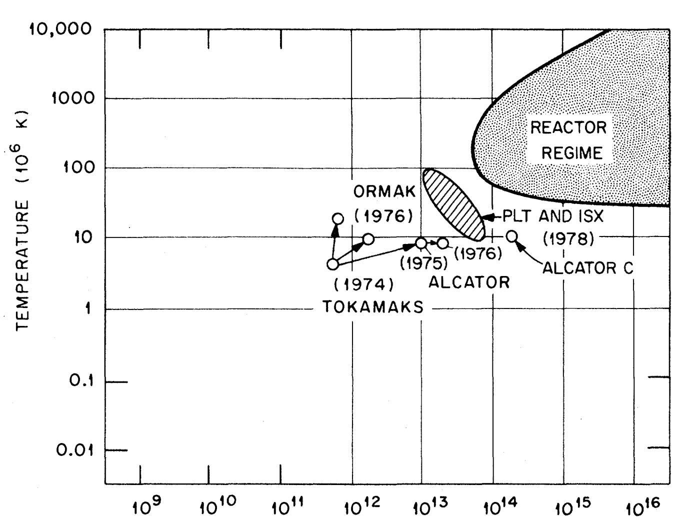

Fig. 3. Recent and projected tokamak experimental performance in terms of plasma ion temperature and plasma energy confinement; the latter is measured by the product of plasma density and energy confinement time. To enter the reactor regime, this product must exceed about 6×10^{13} cm^{-3} sec and the ion temperature must exceed about 40×10^6 K.

mak experimental program. By 1974 tokamaks heated only by the toroidal current flowing in the plasma (that is, ohmic heating) had advanced closer to the reactor regime than any other fusion confinement scheme (*7*). By 1975 the application of extremely high magnetic fields in the ALCATOR experiment at Massachusetts Institute of Technology (MIT) produced a factor of 10 increase in the energy confinement (*8*). By 1976 supplementary heating experiments at Oak Ridge (*9*) on ORMAK with newly developed neutral beam injectors had demonstrated that the plasma ions could be heated to 20×10^6 K. At the same time, experimental data from the heating experiments at Oak Ridge indicated that the improvement in energy confinement seen at MIT could also be produced by the neutral heating technique (*9*). This finding is shown by the second ORMAK point in Fig. 3, which represents operation designed to optimize energy confinement rather than heating.

The performance projected for the next generation of experiments is also shown in Fig. 3. The hatched area indicates the operating regime expected in the neutral beam heating experiments on the PLT (Princeton Large Torus) and ISX (Impurity Studies Experiment, at Oak Ridge). The ALCATOR C point indicates the maximum achievement in energy confinement anticipated in the new high-field experiment at MIT. These experiments collectively should exceed the minimum requirements for temperature and energy confinement in a tokamak reactor. Following closely on these experiments are the DOUBLET III device (at General Atomic) and the TFTR device (at Princeton). The DOUBLET III device has the goal of achieving simultaneously minimum reactor level temperatures and energy confinement (that is, the boundary of the "reactor regime" in Fig. 3) in hydrogen. The TFTR has the goal of achieving simultaneously minimum reactor level temperatures and energy confinement first in hydrogen and then in deuterium-tritium plasmas. It is emphasized that while the TFTR and DOUBLET III will approach the boundary of the reactor regime they will not enter it. It is also noted that similar tokamak experimental programs are being pursued in Europe, Japan, and the Soviet Union.

Although optimism within the program is high, there are still a number of uncertainties to be confronted by these experiments. Although ALCATOR has demonstrated extremely clean plasma operation (*8*), the mechanism for this impurity-free behavior is not well understood. All other devices contain contaminants of either light elements, such as carbon and oxygen, or heavy metals, such as iron and tungsten. These impurities radiate large amounts of energy and, without techniques for their control, could prove to be a serious impediment to reaching the reactor regime. In recent experiments on the DITE tokamak (Divertor and Injection Tokamak Experiment, at Culham, England) impurities have been actively controlled by diverting surface plasma into a burial chamber, thus shielding the inner plasma from wall-originated impurity atoms (*10*). The impurity studies on PDX (at Princeton) and ISX will further examine the evolution, transport, and control of impurities (with and without divertors) in tokamaks.

Furthermore, although the energy confinement of the ions in current experiments is very close to that expected on the basis of classical kinetic theory, the electron energy confinement is one to two orders of magnitude less than classical (*7*). The precise mechanism for this anomaly is not established although several theoretical explanations are available. Without a firm theoretical understanding, we cannot predict with confidence the behavior of the electron energy confinement as our experiments approach the reactor regime. Fortunately, the data from all existing experiments operating over a wide range of parameters indicate that it will improve at least with the square of the plasma radius (*11*). This gives us confidence that the ignition regime can be entered in devices larger than TFTR and DOUBLET III. However, the question remains, how much larger? The answer requires some additional discussion of the scaling of energy confinement.

Scaling to the reactor regime. We can measure energy confinement in present experiments as a function of plasma parameters such as size, temperature, and density. These measurements can then be used to derive a scaling model to predict the tokamak plasma size required to achieve the reactor levels of temperature and energy confinement. This "empirical scaling model" (*9*) predicts ignition conditions in plasmas roughly 1 m in radius if the product of plasma pressure and the magnetic field strength can be made sufficiently large. However, the current empirical scaling model is based predominantly on experiments without supplementary heating. In these experiments there is an implicit dependence of the plasma temperature on the magnetic field, which disguises the actual temperature and magnetic field dependence of energy confinement. Within a few years there will be sufficient data from tokamak experiments with supplementary heating to test this scaling model at reactor temperatures.

An alternative approach is to attempt to predict the energy confinement in the reactor regime from basic principles of plasma physics. The resulting scaling based on classical collision processes indicates that ignition can be attained in very small devices (*12*). Although the ions in our experiments seem to behave in accord with this theory (*13*), the electrons do not. Their energy confinement appears to be dominated by plasma turbulence caused by a variety of fine-grained plasma instabilities. The worst of these instabilities predicted for the reactor regime are known as trapped particle modes (*14*). They have not been clearly identified in experiments to date, and as a result their effect on a reactor tokamak plasma can only be estimated from theoretical calculations. Because of the complexity of the calculations, there is a great deal of uncertainty in the result. However, the scaling model resulting from this theory is remarkably similar to the empirical scaling model in its dependence on the product of magnetic field and plasma pressure. This model also predicts ignition conditions with roughly a 1-m radius if the plasma pressure and magnetic field are sufficiently large.

The net result is that current scaling models based on either empirical or theoretical grounds predict that the reactor regime could be entered with devices only slightly larger than DOUBLET III and TFTR. (Note, for example, that the plasma radius in the TFTR is about 85 cm.) The uncertainties in these scaling laws will be reduced by the current and planned tokamak experimental programs, and within several years the exact size of a reactor plasma should be predictable with a high level of confidence. However, even at present it appears that the reactor plasma radius can be expected to be in the range 1 to 2 m and that the plasma radius of the proposed reactor ($\sim$1.5 m) is a reasonable choice for engineering design studies. This statement is based on the strong dependence of energy confinement on plasma pressure and magnetic field as evidenced in the current scaling models. This dependence should suffice to accommodate errors in our present estimates of the coefficients in the scaling models.

Limits on plasma pressure. Plasma pressure is normally measured in a tokamak in units of the toroidal magnetic field energy density. The ratio of plasma pressure to toroidal magnetic field energy density, known as beta, is an important parameter describing the performance of tokamak fusion reactors. Beta is a measure of the efficiency with which the system utilizes the magnetic field provided by the expensive toroidal field coils. Both the empirical and trapped particle mode scaling models referred to above indicate that the plasma radius of a reactor is reduced as beta (*15*) and the applied toroidal field (*16*) are increased. To produce the most economical tokamak fusion reactor, we would like to work at the lowest possible value of magnetic field and consequently the highest practical value of the plasma beta.

Until recently, it was assumed that a simple tokamak plasma could not be maintained at an equilibrium position within the surrounding reactor structure for betas greater than a few percent. At such low values of plasma beta, one would require either a very large plasma or very high magnetic fields to enter the reactor regime according to either of the scaling models given above. The early fusion reactor studies were performed in accordance with this assumed restriction on beta (see Table 1), and therefore it is not surprising that they resulted in large, expensive reactor systems.

Recent work analyzing the behavior of tokamak plasmas undergoing the intense heating required to raise them to reactor temperatures (*17*) has revealed that this equilibrium limit on plasma beta is nonexistent and that a tokamak plasma equilibrium can be maintained for beta approaching 100 percent. Experimental results from fast pinch experiments (*18*) show the existence of these equilibria at betas of ~70 percent for short periods of time.

The beta of the proposed reactor (~8 percent) lies in the range 5 to 10 percent. Although plasma equilibria exist for this range of betas, there has been some concern about the stability of these equilibria (*19*). It is known theoretically that as the plasma pressure is raised in a tokamak, a point is reached at which an aneurism called a ballooning instability develops in the magnetic field confining the plasma, and the equilibrium is destroyed. Calculations obtained from recently developed computer codes indicate that the beta limit set by this instability is in the range 5 to 10 percent for elongated plasmas (*19*). The ORMAK experiment at Oak Ridge has produced an average beta of just over 1 percent (*20*), a factor of about 3 short of the minimum beta required for ignition in typical reactor configurations and a factor of about 5 short of the minimum beta required for economic interest. Plasma shapes predicted to be compatible with the beta range of reactor interest will be explored during 1978 and 1979 in the ISX, PDX, and DOUBLET III experiments.

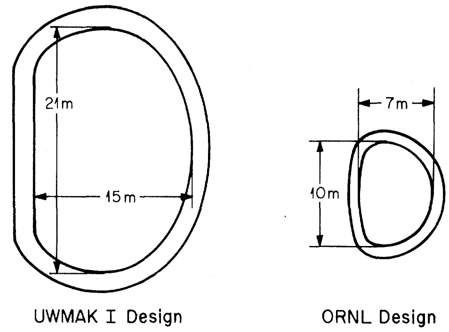

Fig. 4. If the expected value of beta is attained, it is possible to envision a compact tokamak reactor whose toroidal coil size is only a fraction of that assumed in earlier designs. This diagram shows the difference in toroidal coil size when the coil in the UWMAK 1 reactor design is compared with that in the ORNL reactor design.

Technological implications. If betas in the range 5 to 10 percent can be achieved, relatively compact tokamak reactors can be constructed with maximum field requirements (that is, the field at the coil) in the range of 8 to 12 tesla (*5*). Figure 4 illustrates the reduction in toroidal coil size that has resulted from the improved prospects with regard to achievable plasma beta. In Fig. 4 the toroidal coil in the UWMAK I reactor design is compared with the coil in the ORNL reactor design. Both coils are assumed to operate with a maximum field of about 8 T at the coil winding; however, a plasma beta of ~8 percent is assumed in the ORNL design while a beta of ~3 percent is assumed in the UWMAK I design.

As the size of the superconducting coils required for the reactor is reduced, the technological extrapolation required to produce these coils becomes much more reasonable and likely to be achieved in the near term (*21*). The Large Coil Program at Oak Ridge has just let contracts for the design and construction of superconducting coils about half the size required for our compact reactor. Coils made of niobium titanium (with the potential of operating in the range 8 to 10 T) and coils made of niobium tin (potential operating range, 10 to 12 T) will be constructed. These coils will be assembled and tested in a toroidal geometry in the early 1980's. The goal of the Large Coil Program is to provide the technological base for constructing coils of the size required in reactors.

The development of tokamak reactors will also require a technique for heating the plasma to reactor temperatures. Injection of powerful beams of neutral hydrogen atoms into a confined tokamak plasma provides such a technique. The proposed reactor would require 50 to 100 MW of neutral beam power at an energy of roughly 150 to 300 kiloelectron volts to achieve sufficient heating (*22*). Because of the reduced plasma radius, the beam energy required to penetrate to the center of the plasma is a factor of 2 to 3 lower in this reactor design than in the earlier reactor studies (see Table 1). To date, low-power 120-keV beams have been operated in initial experiments with the TFTR beam test facilities at Oak Ridge and Lawrence Berkeley laboratories. In addition, neutral beam injectors at 40 keV with 700 kW of power and capable of almost continuous operation have been produced at Oak Ridge (*23*). Data from these experiments lead us to conclude that we will be able to supply the 20 MW of 120-keV neutral beams required for the TFTR experiment and that the higher-energy beams required for the power reactor can be developed from this technology base.

In addition to the injection of energetic hydrogen particles to heat the plasma, one must also deal with the problem of injecting cold hydrogen to refuel the plasma during the extended burn. The most promising technique proposed for fueling is the injection of pellets of frozen hydrogen at high velocity, greater than 10^3 m/sec (*24*). Recent experiments in which hydrogen-ice pellets were injected into ORMAK demonstrated the basic feasibility of this technique (*25*). In these experiments 30-μm pellets were accelerated in a hypersonic gas jet to roughly 10^2 m/sec. One potential technique for producing the higher velocities and larger pellets needed for reactor-size plasmas is mechanical acceleration in an advanced centrifuge. The first test of this advanced fueling technique is scheduled for 1978 (*26*).

Superconducting magnets, neutral beam heaters, and fueling mechanisms have been singled out for discussion because they represent the most novel technologies that will be required for reactor operation. In addition to these technological developments, there are a host of engineering problems which must be solved for successful operation of the reactor. Over the past few years, studies of tokamak devices following the TFTR have been undertaken in order to uncover these engineering problems (*27, 28*). Major components of a device which would demonstrate all of the characteristics of the fusion core of a power reactor were evaluated with regard to engineering feasibility and cost, and comprehensive systems models were developed so that changes in the operating characteristics of the device could be evaluated in terms of their effect on individual components. These systems models provide us for the first time with tools for evaluating the sensitivity of an overall system design to the precise operating details of the tokamak plasma.

As part of these studies, the development problems which will have to be addressed in order to provide the required device components were evaluated in the context of the range of possible plasma operating parameters. A careful study of the technological requirements to build this fusion core demonstration device in the light of these plasma uncertainties was completed (*29*). The general conclusion of this analysis is that, with a few exceptions, the major technological issues are the subject of active development programs which are scheduled for completion in the early 1980's. The exceptions identified in the analysis are such that they could also be addressed in this period if the relevant programs are initiated soon. Consequently, it appears that with adequate support we will be able, by the early 1980's, to develop the technologies necessary to demonstrate the characteristics of the fusion core of a tokamak power reactor.

Economic Potential

Power generation cost is generally divided into three components: capital cost, fuel cost, and operation and maintenance cost. For example, the light-water fission reactor power generation cost is presently distributed with about 70 percent capital cost, 20 percent fuel cost, and 10 percent operation and maintenance cost. Fission power generation is said to be highly "capital-cost intensive"; that is, the power cost is to a large extent determined by capital-cost investment and to a lesser extent by fuel cost. Fusion reactors will be even more capital-cost intensive: about 90 percent of the power generation cost for fusion reactors will be capital expenditures and about 10 percent will be operation and maintenance; the fuel cost of fusion power should be less than 1 percent of the total. The capital cost (in mills per kilowatt-hour) is derived from the installed plant cost (dollars per kilowatt electric) and the plant capacity factor (the ratio of the average generated load for a period to the total rated capacity of the plant). The capacity factor is a measure of the plant performance which reflects system reliability and maintainability. In this section we will consider the plant costs and the reliability and maintainability of tokamak power systems.

Table 2. Representative plant cost estimates for a tokamak and an LMFBR. The estimates are in 1976 dollars and do not include interest during construction or escalation.

	Cost ($/kWe)	
	Tokamak	LMFBR
Reactor systems	450	100
Balance of plant	800	700
Total	1250	800

Plant costs. At the outset it is emphasized that estimates of plant costs for power systems based on advanced energy sources must be regarded as approximate and somewhat speculative at this time. Nevertheless, such estimates are instructive in comparative assessments of alternate energy sources.

Based on a model developed at Oak Ridge (*5*), the direct plant cost of the conceptual tokamak power plant described earlier has been estimated to be in the range $1000 to $1500 per kilowatt electric, in 1976 dollars. On the same basis (1976 dollars) the direct plant cost of the LMFBR is estimated to be in the range $600 to $1000 per kilowatt electric (*30*) and that of solar electric plants is estimated to be in the range $1500 to $3000 per kilowatt electric (*31*). It is noted that the *direct* plant cost does not include interest during construction or the effects of escalation. Such factors are usually expressed as percentage increments relative to the direct plant cost and should be independent of the energy source. Thus, current estimates suggest that tokamak plant costs would be somewhat higher than LMFBR plant costs and somewhat lower than solar electric plant costs.

Since a tokamak reactor represents a more extensive set of advanced technologies than does the LMFBR, it might at first appear that tokamak plant costs should be much greater than LMFBR plant costs. In this context it is most illuminating to examine the figures in Table 2, which show representative, estimated direct plant costs for a tokamak and an LMFBR broken down into reactor systems costs and balance of plant costs. The following points are emphasized.

1) The reactor systems cost of the tokamak is greater than that of the LMFBR by about a factor of 4 to 5, reflecting differences in the respective technologies.

2) The balance of plant costs is comparable for both systems.

3) The balance of plant costs is the dominant cost of both systems.

Thus, although the reactor systems cost of a tokamak may indeed be much greater than that of the LMFBR, the total plant costs of the two systems might differ by only ~50 percent.

System reliability and maintainability. Although it is too early in the development of fusion power to project a meaningful value for the plant capacity factor, it is very important to consider the areas of system reliability and maintainability in conceptual design studies. In the initial fusion reactor design studies, reliability and maintainability were examined by extrapolating experience gained from the operation of fission reactors and small-scale plasma confinement experiments. Although such an approach was useful in identifying potential problems, it did not allow for creative approaches to system reliability and maintainability based on the unique features of large fusion devices. As discussed below, recent experimental results in the area of materials performance and current design studies that take advantage of the unique features of fusion systems indicate that the prospects for reliability and maintainability of fusion devices are encouraging.

A major issue associated with the question of fusion power plant reliability is the limitation on the service life of the blanket structure arising from neutron irradiation effects. It is well known from fission reactor experience that deleterious changes in physical and mechanical properties occur in structural materials as a result of neutron-induced atomic displacements and transmutation gas products—particularly helium, which has a very low solubility in the metal lattice. Calculations indicate that, in general, atomic displacement rates will be comparable in the first-wall region of a fusion reactor blanket and in the high-flux region of advanced fission reactors; however, the helium production rates will be much higher in the fusion reactor environment because of the difference in neutron energy spectrum between fusion and fission reactors.

Early projections (*32*) of the performance of stainless steel when used as the structural material of a fusion reactor blanket were based on extrapolations of low-fluence fast fission reactor data. Because of the nature of the neutron energy spectrum of a fast fission reactor, the helium generation rates associated with these data were orders of magnitude too low for simulating fusion reactor conditions. These extrapolations suggested that even without a high helium content, loss of ductility might limit the service life of a stainless steel first wall to about 2 years when it is operated at a temperature of ~500°C. Thus, the early projections of blanket structure performance had ominous implications for fusion power reliability.

In 1975 it was noted (*33*) that mixed-spectrum fission reactors (reactors in which both the fast flux and the thermal flux are high) could simulate the ratio of the fusion displacement rate to the helium production rate very closely in alloys such as stainless steel which contain significant amounts of nickel. Helium is produced by the reaction sequence with thermal neutrons

$$^{58}\mathrm{Ni} + n_{\mathrm{th}} \rightarrow {}^{59}\mathrm{Ni} + \gamma$$

$$^{59}\mathrm{Ni} + n_{\mathrm{th}} \rightarrow {}^{56}\mathrm{Fe} + {}^{4}\mathrm{He}$$

Simultaneously, displacement damage is produced by the fast neutrons. Extrapolations of recent data obtained in a mixed-spectrum fission reactor, at high fluence and with high helium production, suggest that the service life of a stainless steel first wall might approach 10 years if it is operated at ~400°C (*34*). The reduction in first-wall temperature does not seriously affect the plant thermal efficiency, and an efficiency of ~35 percent should be possible. Thus, the prospects for extended wall life appear encouraging at this point, and although a major alloy development program will be required to identify and qualify the most suitable blanket structural material, structure life should not adversely affect fusion power reliability.

In previous tokamak reactor design studies, it was generally assumed that the first wall of the blanket would also serve as the major vacuum boundary between the plasma and atmospheric pressure. This usually required that the first wall contain hundreds to thousands of lineal meters of welds. Should an operating failure such as a pinhole leak develop in a radioactive first wall, it is doubtful that it could be repaired without unreasonable difficulty. Therefore, in the reference power plant it has been proposed that the tokamak reactor system be enclosed in a vacuum building. This completely changes the character of the first-wall surface from a welded structure with absolute vacuum integrity to one that can have mechanical joints and only requires high pumping impedance (it can be slightly leaky because the pressures on both sides are about the same). It is believed that such an approach has significant assembly, disassembly, and repair advantages over the vacuum first-wall approach.

Vacuum enclosures of the physical size needed to house a fusion reactor are structurally reasonable, as demonstrated by the Plum Brook facility of the National Aeronautics and Space Administration (NASA). A cross section of that facility is reproduced in Fig. 5, which also shows the approximate dimensions of the reference tokamak reactor. The facility volume and the radiation shielding provided by the concrete structure appear to be consistent with the reactor requirements. It is also noted that such a building is comparable in size to the containment building of fission reactors.

To further minimize downtime and facilitate maintenance, a modular approach has been taken for the blanket design, which eases the problems of remote maintenance. Small, easily replaced blanket modules have been emphasized (see Fig. 6) to eliminate the need for remote maneuvering and welding of massive components in tight quarters.

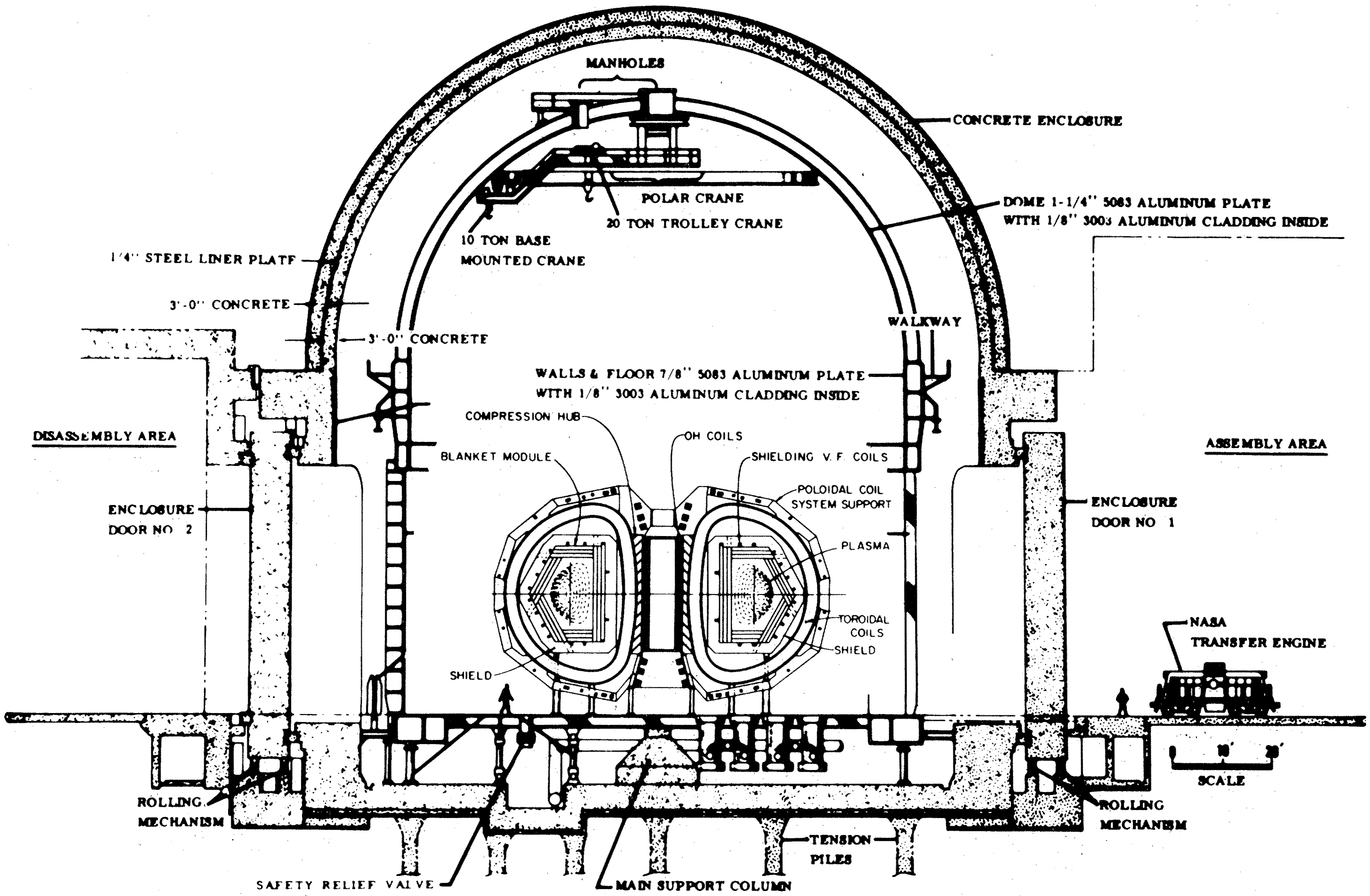

Fig. 5. The ORNL reactor design superimposed on a cross section of the NASA Plum Brook vacuum facility at Sandusky, Ohio (*36*). A vacuum building of the approximate size needed to house a fusion reactor is structurally reasonable. Such a building is comparable in size to the containment building of fission reactors. As discussed in the text, it is believed that a vacuum building approach would enhance the reliability and maintainability of fusion reactors.

Strategy to Demonstrate Power Generation

Initial estimates of the facility and cost requirements for demonstrating tokamak power generation reflected the large size of the early reactor designs and the concerns about first-wall service life. When these early designs (plasma radius, ~5 m) were compared with the TFTR (plasma radius, ~1 m), it appeared that a succession of increasingly larger and more expensive devices would be required to reach reactor-size systems. At the same time, the pessimistic projections about the performance of stainless steels as first-wall materials suggested that a very extensive and expensive materials program would be needed to identify a first-wall material. As part of this program, several expensive high-energy neutron-irradiation facilities would have to be developed and built.

As a result of the continued progress in understanding tokamak physics, the projected plasma radius of power reactors is now about 1.5 m, or within a factor of 2 of that of TFTR. Moreover, recent data on the performance of stainless steels as first-wall materials are encouraging, and it appears that much of the irradiation testing could be accomplished in existing mixed-spectrum fission reactors, although some high-energy neutron facility (*35*) will be required for correlation purposes.

In view of the current situation, we believe that the facility and cost requirements for demonstrating tokamak fusion power can be significantly reduced compared to those reflected in the initial planning exercises. In particular, we suggest that a committed-site, multiple-unit strategy be adopted to demonstrate power generation after the operation of DOUBLET III and the TFTR. This strategy would consist of three phases: (i) a fusion reactor core demonstration, (ii) a power technology demonstration, and (iii) a commercial prototype demonstration. These phases are described below.

During the reactor core demonstration phase, a central pulsed electrical plant would be built to provide pulsed power for all the units. Concurrently, a single tokamak would be built and connected to the central pulsed electrical plant. The device would consist primarily of a torus, a nonbreeding blanket, a shield, toroidal field coils, poloidal field coils, neutral beam injectors, and a divertor system. The scope of this phase is to establish a controlled fusion energy source.

During the power technology demonstration phase, lithium would be introduced into the blanket and a heat transport system, a turbine system, and a tritium recovery system would be added to the basic facility. Electrical power produced during this phase could be fed into a commercial grid. The scope of this phase is to establish technical feasibility.

During the commercial prototype demonstration phase, improved units (for example, ~750 MWe per unit) would be added and tied into the central pulsed electrical plant. The purpose of this phase is to demonstrate system reliability under practical utility conditions. The scope of this phase is to establish economic feasibility.

The strategy outlined above is motivated by the following considerations.

1) The plasma size requirements are essentially the same for the reactor core demonstration device and the commercial prototype. A succession of increasingly larger devices is not required to demonstrate power generation.

2) Capital equipment is conserved during all phases of the demonstration program.

3) Increasing amounts of power are supplied to the electrical grid as each additional unit is added at the site.

The facility cost for the three phases is estimated to be between \$2 billion and \$3 billion (in 1976 dollars). This does not include engineering or contingency costs, nor does it include indirect capital costs. In addition, it is noted that development costs would accompany the demonstration program. The total program cost including engineering, contingency, and development, but excluding escalation, is estimated to be ~\$5 billion to \$8 billion (in 1976 dollars), or ~\$10 billion to \$15 billion if escalation is included. The escalated figures are based on preliminary projections which suggest that prototype operation could commence approximately 20 years after a commitment had been made to the three-phase demonstration program. It appears that a \$10 billion to \$15 billion expenditure represents an acceptable cost for developing a new energy source.

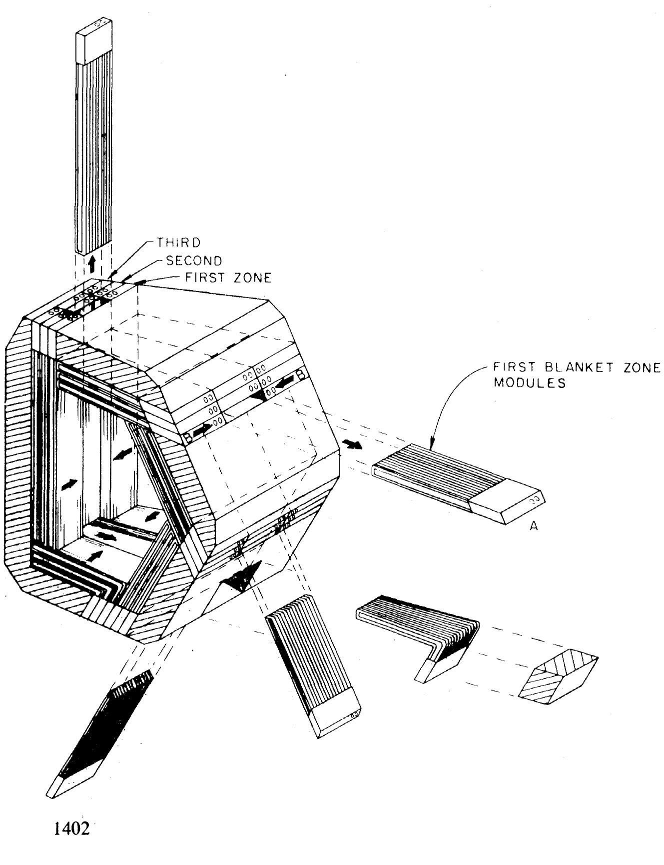

Fig. 6. Remote maintenance of the blanket is receiving major emphasis in current reactor designs. This diagram shows remote removal of individual blanket modules that have been damaged by radiation. In this arrangement there is a five-step sequence to remove the middle module (*A*). The outer modules (*B*) are then moved to the free space and the sequence is repeated. Such a procedure could be carried out without having to move the toroidal field coils.

Concluding Remarks

The tokamak power plant described in this article is based on a specific set of assumed technological directions, design approaches, and plasma characteristics. It is emphasized that there is no unique set of technologies, engineering approaches, and plasma characteristics that will lead to commercial fusion power. Several acceptable sets no doubt exist, and the set described here should be viewed in this context.

The technology base required to realize the reference plant described here is, for the most part, being addressed in ongoing programs. At this point it appears that uncertainties about tokamak fusion power are primarily related to the expected plasma physics performance and not to foreseeable limitations in the areas of technology and engineering. Within the next 3 to 5 years these uncertainties should be resolved.

A three-phase strategy for demonstrating fusion power generation at a committed site has been proposed in this article. It is emphasized that this is a strategy and not a detailed plan. Nevertheless, the strategy outlined here suggests that tokamak fusion power could be demonstrated with reasonable expenditures of money.

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Engineering Limitations of Fusion Power Plants

Problems related to radiation damage and plant costs may prevent the practical application of fusion.

W. E. Parkins

During the past few years, increasing attention has been directed to preliminary engineering studies of possible fusion reactor power plants. These studies have employed the best available theoretical extrapolations of results of confinement experiments in order to predict operational parameters for the power-producing plasma. While no net power-producing fusion plasma has yet been demonstrated, the engineering studies are important in providing guidance to the development program and in pointing out significant practical problems to be faced, once it is learned how to achieve useful thermonuclear reacting conditions. This article does not deal with the difficulty or probability of success of plasma confinement, but instead focuses on engineering aspects of proposed full-scale plants believed to be of critical importance to the future of fusion power.

One problem area that has been stressed is the difficulty of processing and containing the tritium in a fusion plant employing the deuterium-tritium (D-T) reaction. This subject was evaluated in a section of the first report of the Atomic Industrial Forum Committee on Fusion (*1*). The conclusion was that the technology is available to meet the operational requirements and that the principal concern is the impact that plant design features for tritium handling might have on total capital costs. Another problem area treated was related to the acceptability of plant operation from the environmental and safety standpoints. Again, it was concluded that fusion plants will be able to meet all environmental and safety requirements. The greatest difficulty appears to be that of adequately limiting the release of tritium during normal plant operation and as a result of postulated accidents. This then reflects back on the plant design features that will ensure adequate tritium containment, and the effect on capital costs again becomes a principal point of concern.

There are other features inherent in a fusion reactor plant that will force increases in the cost of the initial installation, and the magnitude of the total investment is recognized to be a problem of critical importance to the eventual successful application of fusion power (*1*). In this article some of the engineering factors bearing on capital costs will be evaluated. A second very serious engineering problem area discussed is that of the limited operating life of the reactor vessel, caused by the deleterious effects

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