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Selection of a Toroidal Fusion Reactor Concept for a Magnetic Fusion Production Reactor[1]

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The basic fusion driver requirements of a toroidal materials production reactor are considered. The tokamak, stellarator, bumpy torus, and reversed-field pinch are compared with regard to their demonstrated performance, probable near-term development, and potential advantages and disadvantages if used as reactors for materials production. Of the candidate fusion drivers, the tokamak is determined to be the most viable for a near-term production reactor. Four tokamak reactor concepts (TORFA/FED-R, AFTR/ZEPHYR, Riggatron, and Superconducting Coil) of approximately 500-MW fusion power are compared with regard to their demands on plasma performance, required fusion technology development, and blanket configuration characteristics. Because of its relatively moderate requirements on fusion plasma physics and technology development, as well as its superior configuration of production blankets, the TORFA/FED-R type of reactor operating with a fusion power gain of about 3 is found to be the most suitable tokamak candidate for implementation as a near-term production reactor.

KEY WORDS: Magnetic fusion production reactor; tritium production; fusion breeder; toroidal fusion reactor.

1. STUDY OBJECTIVES

In this study we have identified the most viable toroidal fusion driver that can meet the needs of a materials production facility to be operational in the mid-to-late 1990s. The work summarized herein provides justification for the preferred concept and for the rejection of other candidate toroidal reactor concepts.

Section 2 of this paper establishes the basic requirements that the fusion neutron source must satisfy. In Section 3, we compare various types of toroidal fusion concepts for which there has been at least some significant development work. Section 4 covers our examination of certain tokamak reactor concepts and their potential application in the near term as fusion drivers for a materials production reactor.

The selected fusion driver is described in considerable detail in Refs. 1 and 2. Reference 1 discusses the integration of the breeding blankets into the fusion driver in a manner that maximizes the blanket coverage factor while retaining access to the materials production regions. In Ref. 2 we address the outstanding uncertainties in the physics and technology, as well as the development programs that must

[1] This paper represents work carried out from 1980 to 1982 and was in draft form in 1982. It was received for publication with only minor editing from its 1982 version (except for Tables II and III and Fig. 1), explaining the fact that some of the material is dated.

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be implemented to make this fusion driver operational by the mid-to-late 1990s.

2. REQUIREMENTS OF THE FUSION NEUTRON SOURCE

2.1. Fusion Power Requirement

Reference 3 establishes that, for the desired materials production rates with the types of breeding blankets envisaged, a suitable fusion power level is of the order of 500 MW, assuming a 70–80% annual capacity factor. While this power level could be met by using two or more reactors, the cost per excess neutron from a single reactor is likely to be decreasing significantly with increasing power in the range around 500 MW. Hence, all the fusion driver concepts considered herein are assumed to have a size P_{fus} of ~ 500 MW.

For a toroidal reactor intended for neutron breeding applications, the important parameters relating to cost effectiveness are

1. *Neutron wall loading*, $\phi_w = 0.8$ X (fusion power/first-wall area). This parameter is a measure of the fusion neutron production rate per unit capital cost of the reactor facility.
2. *Electrical utilization efficiency*, Q_e = (fusion power/plant electrical power input), which is a measure of the grams of neutron production per unit of operating cost. Closely related to Q_e is the fusion power amplification, Q_p = fusion power/injected heating power.

The parameters ϕ_w (neutron wall loading) and Q_p are discussed in the following sections.

2.2. Neutron Wall Loading

The most compact facility for a given fusion power will generally be the least expensive. This statement must be tempered, however, by the increasing difficulty of maintenance as reactor size is reduced and by any increase in power requirements that might result from extreme compactness (such as for high-current-density resistive magnets). Obviously ϕ_w increases with increasing degree of compactness.

Two considerations limit ϕ_w: (1) thermal hydraulic and thermomechanical problems of the first-wall and blanket system under conditions of high power loading and, in some fusion concepts, severe thermal cycling; and (2) radiation damage, which can result in more frequent downtime for maintenance or replacement of damaged components.

In regard to the above, it is worth noting that the first fission production reactors were massive installations of graphite and natural uranium slugs with relatively low power densities, compared with modern fission reactors. These early production reactors, however, were relatively easy to service and gave long, reliable operation.

We can obtain an upper limit to ϕ_w by noting that essentially all toroidal fusion concepts require a distance of at least 2 m from the central axis of the torus to the inboard edge of the plasma vessel. This 2 m includes a minimal-size inboard blanket. Knowledge of plasma confinement properties dictates a minimum radius of approximately 0.5 m for the plasma vessel. Assuming a circular vessel and P_{fus} = 500 MW gives $\phi_w = 8$ MW/m^2.

A driven reactor with a Q_p of ~ 5 will have at least 200 MW of thermal power to be removed from the plasma. Neutronic analyses[4] have shown that, for the first wall and its coolant system to be acceptably transparent to fast neutrons, the thermal wall loading ϕ_t should be no larger than about ~ 50 W/cm^2. If no magnetic divertor is provided, the minimum first-wall area must be 400 m^2, which would result in a ϕ_w of only 1.0 MW/m^2. If a divertor is actually implemented and removes 75% of the nonneutron power flow (probably an upper limit),[5] the minimum first-wall area can be as small as 100 m^2, which would give a ϕ_w of 4 MW/m^2. This value is taken as the *largest* acceptable fusion neutron wall loading. The considerations that limit ϕ_w can probably be overcome if there is only modest blanket energy multiplication (2 or less) when ϕ_w = 4 MW/m^2. Otherwise, the maximum permissible value might have to be lowered further.

For cost effectiveness, it would be undesirable to have ϕ_w much less than 4 MW/m^2. Hence, we somewhat arbitrarily establish the minimum neutron wall loading as $\phi_w = 2.0$ MW/m^2. It is recognized that, in a given fusion device, ϕ_w may actually vary considerably as a function of position on the first wall, especially for toruses of low aspect ratio.[6]

2.3. Minimum Fusion Power Amplification

For illustration, we take the cost of electricity as 30 mils/kWh, or as \$260/kWh per year at a 100% capacity factor. This relatively low cost can pertain to certain government-sponsored reactor sites with access to a high-capacity grid. Then if y is the number of excess neutrons (i.e., available for breeding) per fusion neutron, the cost per gram of excess neutrons due to electricity consumption is $\Delta C_{el} = \$19 \times P_{el}/\text{year}$, where P_{el} is the power consumption of the toroidal production reactor (TPR) in megawatts.

If $P_{el} = 500$ MW and $y = 0.5$, for example, then $\Delta C_{el} = \$19{,}000/\text{g}$. If each neutron can breed one atom of special material, just the electrical component of the production cost will be \$6300/g of tritium, or \$80/g of ^{239}Pu. These values could be significant compared with the objectives for total cost per gram of product.

The electrical power required for plasma heating is $500/\eta_h Q_p$ megawatts for 500-MW fusion power, where η_h is the efficiency of the heating system. The maximum practical value of η_h is 0.60, so that $P_{heat} \geqslant 833/Q_p$ MW.

In practice, there will always be other reactor components that consume substantial electrical power, such as resistive magnets and vessel coolant systems. If P_{el} is to be set arbitrarily at a maximum value of 500 MW, for example, then Q_p must be at least 1.7. If the plasma heating systems consume approximately half of the plant's entire electrical demand, then Q_p must be at least 3.3 to limit P_{el} to 500 MW.

Several other factors tend to weigh in favor of the highest possible Q_p:

1. The capital cost of the plasma heating equipment for a given fusion power is inversely proportional to Q_p.
2. For a given fusion power, the first-wall area that must be appropriated for injection of the plasma heating power is inversely proportional to Q_p. These penetrations of the first wall may significantly reduce the fraction of the total neutron population that can be productively absorbed.
3. The thermal wall loading is $\phi_t = 1.25\ \phi_w \times (0.2 + 1/Q_p) \times f_w$, where f_w is the fraction of the nonneutron power flow from the plasma that is not removed by a magnetic divertor. Analyses[4] have shown that, for the first wall and its coolant system to be acceptably "transparent" to fast neutrons, ϕ_t should be no larger than about 50 W/cm^2. Then, if $\phi_w = 2.0$ MW/m^2 and $f_w = 0.25$, Q_p must be at least 1.65.

On the other hand, there are several reasons why attempting to operate at very high $Q_p > 10$ is undesirable:

1. Achieving high Q_p requires a large "lawson parameter" $n_e\tau_E$. Experiments in toroidal devices indicate that this parameter increases with the density and size of the plasma. At the size and density needed to achieve high Q_p, the fusion power output may considerably exceed the 500-MW range of interest.
2. Steady-state operation of certain types of toroidal devices, including tokamaks, appears to require the injection of substantial beam or RF energy to drive the current. The high plasma density required to reach the large $n_e\tau_E$ needed for high Q_p reduces the current-drive efficiency of the beams or RF energy. Hence, steady-state current drive appears to be especially compatible with plasma operation at relatively low $Q_p \leqslant 3$.
3. Operation at lower Q_p allows the plasma to be fueled entirely by D^0 and T^0 neutral beam injection.
4. In lower Q_p operation the injected power can be tailored continually to ensure stable operation of the fusion plasma, obviating the need to develop a special control mechanism that would be required in the case of high-Q_p or ignited plasmas, where the injected power plays a minor or negligible role in controlling the plasma profiles and peak temperatures.

As a result of the above considerations, we selected a minimum value of $Q_p \simeq 3$, assuming that the heating power can be injected with an efficiency of the order of 0.5 or more. Unless high Q_p can be obtained in a small machine, and steady-state current drive is feasible with relatively small injected power, it appears that the maximum Q_p should be limited to about 5, again assuming that $\eta_h \geqslant 0.5$.

In summary, the recommended basic design parameters for a toroidal fusion driver are:

1. Fusion power ≃ 500 MW.
2. Fusion neutron wall loading, $\phi_w = 2$ to 4 MW/m^2.
3. Fusion power amplification, $Q_p = 3$ to 5, assuming that plasma heating efficiency $\eta_h \geqslant 0.5$.
4. Steady-state operation, or at least long pulses with a high duty factor.

3. COMPARISON OF TOROIDAL FUSION DEVICES

3.1. Potential Advantages and Disadvantages

3.1.1. Toroidal Concepts

Of the various toroidal fusion concepts proposed and pursued over the last 30 years, the most developed are the tokamak,[7] the stellarator,[8] the Elmo bumpy torus (EBT),[9] and the reversed-field

Table I. Alternative Toroidal Fusion Devices

	Elmo bumpy torus	Stellarator	Reversed-field pinch[a]
Potential advantages vis-a-vis pulsed tokamaks	Steady-state operation allows higher duty factor and reduces mechanical and thermal fatigue Large aspect ratio allows easier access to all blanket regions	Steady-state operation No current disruptions	Ohmic heating to ignition eliminates neutral beams or RF Substantially higher β and wall loading Reduced capital cost
Disadvantages vis-a-vis tokamaks	Physically huge (major radius 20 m or more), results in larger capital cost Large circulating power in millimeter waves[b] Attainable β of bulk plasma is lower than in tokamak	Magnet fabrication is especially difficult Modularity of coils may be impractical, thus greatly complicating maintenance Ripple-induced losses of particles and energy may prevent high Q_p	Pulses are relatively short, with a low duty factor Copper coils around plasma chamber degrade neutron economy
Principal feasibility issues	Plasma energy confinement Development of efficient millimeter wave gyrotrons Minimum physical size (reactor)	Attainable β Losses by magnetic ripple Maintainability (reactor) Access to blankets	Energy confinement Attaining ignition by ohmic heating alone Achievable pulse length (reactor) Development of first-wall materials to sustain ~10 MW/m^2 for lengthy period

[a] Includes OHTE.
[b] More power than required to drive a steady-state tokamak plasma.

pinch (RFP), which includes the ZT-40 device[10] at Los Alamos National Laboratory and the OHTE device[11] at General Atomic. The tokamak has proven to be the most effective in approaching reactor-like plasma conditions. Nevertheless, proponents of the alternative (i.e., nontokamak) concepts insist that the potential advantages of their concepts, when compared with pulsed tokamaks, are so great that they should continue to be vigorously pursued both experimentally and theoretically. These potential advantages are listed in Table I.

A reactor based on any of these alternative concepts would also have serious disadvantages when compared with a tokamak reactor, as indicated in Table I. The principal feasibility issues at each concept's present stage of development, as well as in extrapolation to reactor plasmas, are also listed in Table I.

If a steady-state tokamak using noninductive current drive and operating at $Q_p > 3$ proves feasible, the potential advantages of the alternate concepts will be reduced in scope and may be eliminated. Many experiments in several tokamaks have demonstrated that plasma current can be sustained solely by the injection of raido frequency power at the so-called lower hybrid frequency.[12] To date, sustained RF-driven current has been limited to plasmas with $n_e < 3 \times 10^{13}$ cm^{-3}, or a factor of 3–5 smaller than reactor densities. However, the inherently steady-state EBT devices have operated at n_e less than 10^{13} cm^{-3} (Ref. 9).

3.1.2. *Access to Blankets*

Various schemes have been devised to permit ready access to the breeding blankets in many tokamak concepts. Access is especially feasible when there are relatively few oversized TF coils, or if the TF coils are demountable. In the case of the EBT, the large aspect ratio and simple coil system ensure good access to the blankets. If stellarator/torsatron-type reactors cannot be modularized, however, their convoluted magnetic coil configuration would make access to the blankets extremely problematical.[13]

3.2. Demonstrated Performance

Table II is a comparison of the best values of key plasma parameters achieved to date in tokamaks, stellarators, EBTs, and RFPs (including OHTE). Note that the performance parameters achieved by the tokamak some 15 to 20 year ago are comparable with the best achieved by 1986 in each of the alternative toroidal concepts.

Table II. Comparison of Key Plasma Parameters[a]

Parameters	Tokamaks	Stellarators	EBTs	RFPs (and OHTE)	Required for Tokamak MFPR
Max T_e (keV)	6.5	1.1	<1.0	0.5	15
Max T_i (keV)	12.0	1.0	0.1	0.5	30
Max $\bar{n}_e \tau_E$ (cm^{-3} s)	7×10^{13} (5×10^{12} at above temperature)	2×10^{12}	$< 2 \times 10^{10}$	4×10^{10}	$\geq 3 \times 10^{13}$
Max $\langle \beta \rangle$, spatially averaged	0.05 (0.01 at above temperature)	0.02	< 0.01	0.2	≥ 0.05
Max pulse length (s)	20 (0.4 at above temperature)	0.5	Steady	0.02	$\gg 100$
Year by which tokamaks had achieved this performance (except β)		1972	1965	1964	—

[a] Best parameters achieved as of June 1986.

The CLEO experimental facility at Culham Laboratory in the United Kingdom has been able to test four configurations in the same device by using various portions of an elaborate magnetic coil system.[14] The four configurations were the tokamak, the stellarator, the RFP, and the OHTE; the same magnetic field was used in all cases. Little difference in performance was observed between the RFP and the OHTE. These latter configurations can produce the highest β, but have poor energy confinement time τ_E. The stellarator was found to have the highest τ_E but gave the lowest β. The tokamak had the highest product of τ_E and β. For the basic feasibility of a fusion concept, the more important parameter is τ_E, but a significant β is required for reactor competitiveness.

3.2.1. τ_E in Tokamaks

The τ_E of tokamak plasmas with intense neutral beam or RF heating (PLT, PDX, DIII, ASDEX) has failed to increase with plasma size and density as markedly as it does in most ohmic-heated plasmas. However, this setback is at least partially compensated for by the strongly favorable dependence of τ_E on plasma current and on vertical elongation of the plasma.[15] While the highest values of β to date have been achieved only with very low τ_E, there is no evidence of a limit to β in vertically elongated discharges in the DIII experiments, where spatially averaged β values as large as 4% have been obtained.[15]

The best values of $\bar{n}_e\tau_E$ achieved to date at very high plasma temperatures are one order of magnitude smaller than those needed in a TPR, although the achieved $\bar{n}_e\tau_E$ at relatively low plasma temperatures are comparable with those needed in a TPR. There is every indication that TPR-level $n\tau_E$ will be reached at high plasma temperatures in the larger tokamaks that will operate in the 1980s (TFTR, JET, DIII-Upgrade).

The projections of achievable plasma parameters for each alternative to the tokamak are quite optimistic, as they have been initially for each fusion concept proposed during the last 30 years. History shows that, as devices embodying a particular concept have become larger, the projections have usually failed. It is especially difficult to understand the current enthusiasm for RFP-type devices in view of their abysmal performance despite a development history as lengthy as that of the tokamak.

3.2.2. Neutron Production

In Table III the optimal performances of 12 types of fusion devices are compared with regard to neutron production rate, neutrons per pulse, and fusion energy gain Q_p (converted to the equivalent

Table III. Record Levels of Fusion-Neutron Production in Experimental Devices[a]

Type of device	Name of device	Date of record yield	D–D neutrons per sec[b]	D–D neutrons per pulse[c]	Q for D–D[d]	Equivalent Q for D–T
Beam-injected tokamak	TFTR	1986	8×10^{15}		7×10^{-4}	0.25
Ohmic-heated tokamak	JET	1985	2×10^{14}		6×10^{-5}	0.02
Beam/gas target	U. Wisc	1976	2×10^{12}		(DT)	0.007
RF-heated tokamak	JET	1986	1×10^{14}		1.5×10^{-5}	0.005
Beam/solid target	RTNS-II	1979	4×10^{13}		(DT)	0.002
Dense plasma focus	DPF-6-1/2	1973		2.0×10^{12}	7×10^{-6}	0.002
Laser/pellet ($\lambda = 0.35\ \mu$m)	NOVA	1986		1.0×10^{13}	(DT)	0.0016
REB/exploding wire	GAMBLE II	1973		1.0×10^{11}	2×10^{-6}	6×10^{-4}
Laser/pellet ($\lambda = 0.53\ \mu$m)	GEKKO XII	1985		1.2×10^{12}	(DT)	4×10^{-4}
REB/foil	REIDEN II	1978		1.0×10^{9}	4×10^{-7}	1×10^{-4}
Tandem mirror	TMX	1980	3×10^{11}		1×10^{-7}	4×10^{-5}
Standard mirror	2XIIB	1977	4×10^{11}		9×10^{-8}	3×10^{-5}
Linear theta-pinch	SCYLLAC	1972		7.0×10^{9}	3×10^{-8}	1×10^{-5}

[a] Devices are listed in order of decreasing Q.
[b] Given only for quasi-steady devices.
[c] Given only for short-pulse devices.
[d] Devices for which D–T neutron yields are given are denoted (DT).

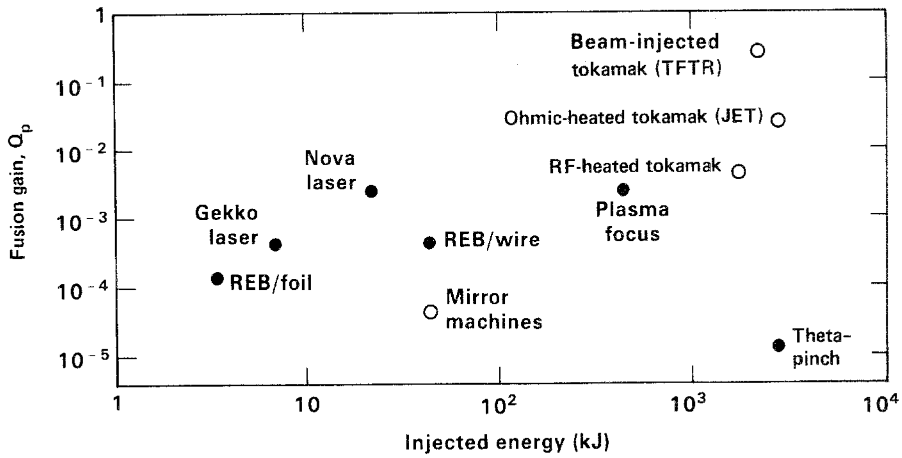

Fig. 1. Record values of fusion gain vs. energy injected into the plasma or delivered to the pellet, foil, wire, or electrodes. The equivalent Q_p in D–T is given for systems that have used only deuterium.

value for D–T operation). The record values in all categories are held by the beam-injected tokamak plasma, followed by other beam-target and tokamak systems.[16]

3.2.3. Fusion Energy Gain

Figure 1 shows the measured Q_p vs energy injected in the plasma (or pellet) for the fusion systems of Table III. The data in Fig. 1 suggest that, in almost any fusion system, Q_p can be increased by delivering more energy to the target plasma or pellet. However, this energy must be delivered in one or two energy confinement times (or in one disassembly time) so that the power requirements for systems with poor τ_E become prohibitively large. The demonstrated performances recorded in Table III and Fig. 1 show that the beam-driven tokamak system is the best near-term candidate for achieving $Q_p \sim 1$ on the basis of plasma physics effectiveness.

3.2.4. Summary

Despite the potential reactor advantages of the alternative toroidal fusion concepts, the tokamak has been selected as the fusion driver for a materials production reactor. The choice of the tokamak was based on its perceived superiority to meet the requirements of a TPR fusion driver, as determined by its demonstrated performance to date (Tables II and III), the high probability of vastly improved performance in the largest tokamaks now operating (TFTR, JET, DIII-Upgrade), and the markedly poorer performance of other magnetic confinement fusion schemes.

4. ASSESSMENT OF TOKAMAK CONCEPTS

4.1. Candidate Tokamak Reactor Concepts

Four basic tokamak reactor concepts have been examined: (1) TORFA/FED-R, (2) ZEPHYR/AFTR, (3) Riggatron, and (4) superconducting coil.

Table IV compares the principal parameters of these reactor types when designed for use as production reactors with fusion power in the range of 500 MW. Also shown are the parameters of TFTR, the largest U.S. tokamak,[17] which began operation in 1983 and which is expected to reach $Q_p \sim 1$ in the late-1980s, using 1-s pulses at very low duty factors (0.003 or less). Figure 2 shows simplified diagrams of these four reactor types.

4.1.1. TORFA/FED-R Reactors

The TORFA/FED-R reactors[18,19] were conceived specifically for blanket module testing and materials production with minimal advances required beyond expected TFTR performance for the technology of the tokamak fusion driver. The TF (toroidal field) coils are made of water-cooled copper plates and designed for rapid demountability to provide ready access to all the production regions, as well as

Table IV. Comparison of Principal Parameters of Candidate Tokamak Fusion Drivers

Parameters	TORFA/ FED-R	ZEPHYR/ AFTR	Riggatron	Superconducting Intor	Superconducting FED	TFTR (for comparison)
Major radius (m)	3.9	3.7	≤ 1	5.2	5.0	2.5
Minor radius (m)	0.95[a]	0.95[a]	0.3	1.2[a]	1.3[a]	0.85
Maximum B at coil (T)	10.0	11.0	25.0[e]	11.0	10.0	9.2
Field at plasma (T)	5.0	5.5	16.0[e]	5.5	4.6	5.0
Plasma current (MA)	5.0	5.5	6.0	6.4	6.5	3.0
Beam energy (keV)	250	175[b]	NA	175[b]	NA	120
Beam/RF power (MW)	150	60[b]	?	75[b]	50	32
$\bar{n}_e \tau_E$ (cm^{-3}s)	3×10^{13}	2×10^{14}	2×10^{14}	2×10^{14}	2×10^{14}	1×10^{13}
Req'd $\langle\beta\rangle$	0.06[c]	0.05	0.05–0.1	0.056	0.06	0.03[c]
Pulse length (s)	Steady state[d]	~ 200	~ 30	200	50 to steady state[d]	~ 1, mid-80s ~ 5, late-80s
Duty factor (%)	90–100	75	≤ 50	80	10–100	$\leqslant 0.003$
Fusion gain, Q_p	3.0	Ignited	Ignited	Ignited	Ignited	~ 1
Fusion power (MW)	500	500	> 200[e]	620	450	$\geqslant 20$
Neutron wall loading (MW/m^2)	1.7	1.8	$\gtrsim 15$[e]	1.3	1.0	$\geqslant 0.15$
Average electrical power consumption (MW)	550	400	> 500	240	185 or 300[f]	NA
Particle and heat removal from plasma	Poloidal divertor	Pumped limiter	?	Poloidal divertor	Pumped limiter	In-torus gettering

[a] Plasma vertical elongation = 1.5 to 1.6.
[b] For startup only.
[c] Approximately two-thirds in bulk plasma and one-third superthermal ions.
[d] If steady-state, noninductive current drive is feasible.
[e] Hybrid reactor mode only.
[f] With long-pulse or steady-state noninductive current drive.

for the maintenance and replacement of the internal tokamak components. The TF coils are specified to be massive enough (3000 or more tons) so that volumetric power dissipation is low and the coils can be operated in the steady state with acceptable power loss.

TORFA-type reactors are designed for driven plasma operation, preferably using neutral beam injection, but possible radio frequency waves. We assume that the plasma current can be driven in the steady state by the same injected beams or RF used for plasma heating. Because of the power needed to drive the current and for reasons discussed above, moderate values of Q_p are assumed. If noninductive current drive becomes impractical, the alternative of pulsed operation (e.g., a cycle of 500 s ON and 50 s OFF) will lead to 10% lower annual neutron production, a significant increase in reactor cost, and reduced thermal component lifetimes as a result of fatigue caused by thermal and mechanical cycling.

The ability of TORFA-type fusion drivers to operate steady state or with very long pulses is enhanced by the inclusion of a poloidal magnetic divertor for heat removal and particle control.

4.1.2. ZEPHYR/AFTR Reactors

The ZEPHYR/AFTR reactors[20] are larger versions of the high-field ignition test reactor designed by MIT and IPP-Garching in 1978–1980. The ZEPHYR/AFTR-type reactor is designed to reach

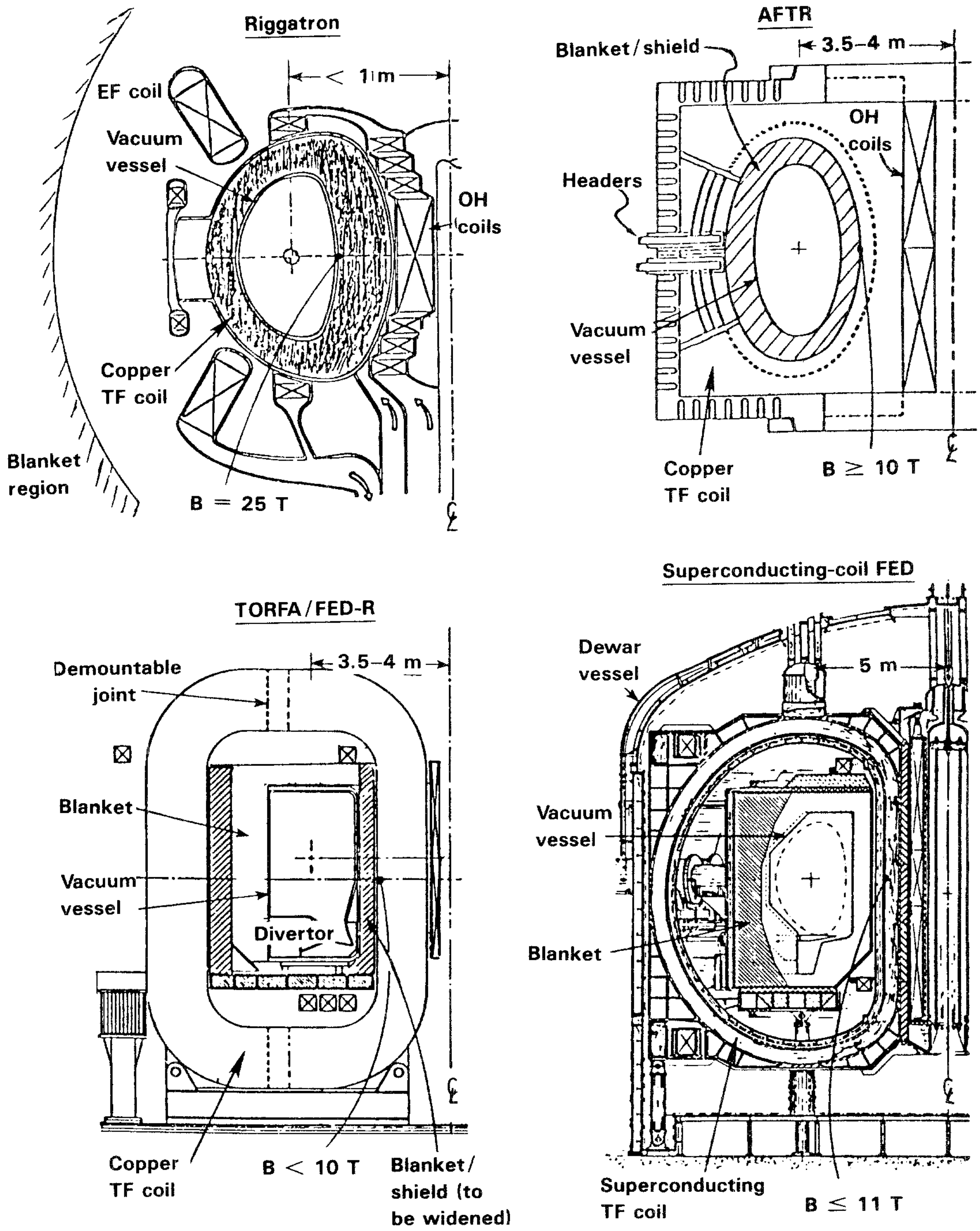

Fig. 2. Candidate tokamak reactor concepts.

ignition ($Q_p = \infty$) using neutral beam or RF heating, with achievement of the required $n\tau_E$ assisted by the use of very high magnetic fields (up to 15 T at coil windings). The magnetic field is significantly lower (see Table IV) for application to a production reactor with nearly full blanket coverage and for relatively moderate power levels. The TF coils are of Bitter-plate-type construction. Operation is pulsed, with steady-state current drive rendered difficult by access constraints and, in any event, incompatible with ignited operation.

The plasma parameters for an AFTR device of the required fusion size are similar to those of TORFA/FED-R. The principal distinction is that proponents of the AFTR concept assume ignited operation on the basis of the so-called Alcator scaling law; however, this law is known to overestimate confinement times in nonohmic-heated plasmas, and it is unlikely that the relatively low density of the AFTR plasma would permit ignition. The most marked differences between the TORFA and AFTR concepts derive from the TF coil designs. The Bitter-plate coil concept used in AFTR allows minimal access to the bore of the TF coils; maintenance of the machine components and removal of production blankets are achieved only by retracting an entire sector of the tokamak. The consequence could be prolonged downtime for maintenance or replacement of blankets.

The ZEPHYR/AFTR design concept is also not suitable for the inclusion of a poloidal magnetic divertor because of the constraints of the Bitter-plate coils. Inherently pulsed operation and uncertain impurity control probably limit the duty factor to about 75%.

4.1.3. Riggatron Reactors

The Riggatron copper-coil devices[21] are, according to their proponents, capable of reaching ignition by ohmic heating alone. Riggatron R&D was pursued at INESCO, a private company, but the concept derives from the Alcator tokamaks developed at MIT. Ohmic heating would be especially strong because of the unusually large plasma current densities made possible by extraordinarily high magnetic fields (25–30 T at the TF coils) and small major radius. Nevertheless, final INESCO plans apparently called for auxiliary heating by ion cyclotron waves.

The success of the Riggatron is predicted on the validity of the Alcator scaling law, which states that τ_E is proportional to plasma density and to the square of the minor radius. However, this law has apparently broken down in experiments on Alcator C, the closest existing experimental device to a Riggatron, as well as in experiments in other large tokamaks.[22] More careful analysis of older tokamak τ_E data, together with the new data, strongly suggests that the dependence on minor radius is weaker than a_p^2 and that there is a substantial dependence of τ_E on major radius R_p. In "standard" Riggatron designs R_p is only slightly larger than in Alcator C, while a_p is significantly larger. Given the newly favored scaling with R_p, it appears that the Riggatron will not achieve the $n\tau_E$ required for ignition.

If cyclotron heating were applied, the consequent smaller reliance on ohmic heating would permit an increase in R_p as required to obtain the τ_E needed for ignition. In fact, a larger device would probably be required to accommodate any supplementary heating apparatus. This step would bring the Riggatron in the direction of the ZEPHYR/AFTR class of reactor concepts.

All of the Riggatron magnet systems are pulsed. Impurity control methods have been not been identified, but impurity control is an especially critical issue in view of the high thermal wall loading ($\gtrsim 300$ W/cm^2). Pulse lengths will not exceed a few tens of seconds, and the duty factor will be 50% or less.

4.1.4. Superconducting-Coil Tokamaks

These tokamaks form the basis of most conceptual design studies of tokamak reactors such as INTOR[23] and the FED baseline.[24] The attraction of superconducting coils is the reduced power requirement when compared with resistive coils (200–300 MW would be saved in a production reactor), but the capital cost of a superconducting-coil production reactor will be much larger than that of a resistive-coil reactor. (On the other hand, superconducting TF coils are probably essential for "pure fusion" electrical power reactors with no product but electricity.)

No superconducting coil for a tokamak has yet been tested in the United States, Europe, or Japan, although the Large-Coil Test Facility[25] scheduled to begin operation before the end of 1983 will have at least three coils of 2.5-m×3.5-m bore size. A small superconducting-coil tokamak called T-7 ($R_p = 1.2$ m, $a_p = 0.25$ m, $T_t = 2.2$ T) was put into operation in the Soviet Union in 1978. Although the coils appear to operate satisfactorily, they contain a ratio of copper to superconductor (for cryostabilization) that is several times larger than would be practical in full-sized reactor coils. Because the geometry of the TF coils severely limits access to the vacuum vessel, making it difficult to rectify persistent vacuum problems, few plasma physics results are available from T-7 even after four years of operation. Larger superconducting-coil tokamaks are presently under development in France and the Soviet Union, but operation is not expected before 1988.

The disadvantages of using superconducting coils for application to tokamak production reactors include higher cost, nondemountability, the need for perfect radiation shielding, susceptibility to pulsed magnetic fields, complicated refrigeration requirements, and the difficulty of coil maintenance or replacement. It may be possible to eliminate these disadvantages for large fusion power reactors because the necessarily large physical size of such reactors permits considerable space for radiation and electromagnetic shielding and for access without undue cost penalty. For a power reactor application, the dissipative loss in resistive coils may well be unacceptable.

The INTOR test reactor[23] has been designed for repetitive pulsed operation. Versions of the FED concept specify pulse lengths that, depending on the effectiveness of neutral beam or RF current drive, range from 50 s to steady state. Although the FED plasma is supposed to be ignited, or at least to operate at high Q_p, the application of noninductive current drive may limit Q_p to the range 5–8, which is acceptable for a production reactor. Because the superconducting-coil FED designs specify a pump limiter rather than a magnetic divertor, there is some uncertainty about the ability to control erosion and impurity buildup to the extent necessary to achieve quasi-steady operation. If noninductive current drive is used, total electrical power consumption will increase to at least 300 MW.

4.2. Deployment and Maintenance of Blanket Assemblies

Materials production will take place in blanket regions surrounding the plasma chamber. In any tokamak reactor concept, there are certain common concerns related to blanket performance:

1. Minimizing the effective thickness of the first wall, including such in-vessel components as protective plating and limiters, to maximize fusion-neutron transmission into the blankets, as well as the hardness of the transmitted neutron spectrum
2. Maximizing isolation of the blanket assemblies from the pulsed tokamak fields by eliminating large current paths and by ensuring that blanket components will not have to sustain arcing in the event of plasma disruption
3. Minimizing the consequence of exposure of sensitive blanket materials to water or to oxygen by careful material selection and leakage precautions

Table V summarizes anticipated difficulties and special advantages in deploying and maintaining

Table V. Anticipated Difficulties in Configuration and Maintenance of Blanket Assemblies

Tokamak reactor type	Anticipated difficulties	Special advantages
TORFA/FED-R	Loss of effective blanket coverage by inclusion of magnetic divertor	Demountability of TF coils allows ready access to all production blankets
ZEPHYR/AFTR	Region inboard of plasma has inadequate space for blankets Access to outboard blankets is difficult Access to inboard blankets requires severing reactor and retracting entire reactor module	
Riggatron	Neutron economy is poor because most neutrons must penetrate copper TF coils to reach blankets Tritium self-sufficiency is in doubt	Blanket regions are external to the tokamak
Superconducting coil	Access to inboard blankets is difficult and may require retraction of entire reactor module	

materials production blanket assemblies for each of the four tokamak reactor concepts considered herein. Special attention is given to the issue of adequate access to the blankets for maintenance or replacement, either to remove the product or because of failure due to radiation damage, thermal fatigue, excessive coolant leakage, or accidents.

Inclusion of a magnetic divertor may well be essential for quasi-steady-state operation. However, it can result in a significant reduction in atoms bred per fusion neutron because of attenuation and moderation in the divertor hardware.

For the AFTR reactor, the inboard blanket/shield thickness for the dimensions shown in Table IV is only about 50 cm, which may be too small for a good production blanket. In any event, the reactor will have to be severed to gain access to this region.

The Riggatron advantage of external blanket regions may be more than offset by the requirement that most source neutrons penetrate the TF and PF coils.

4.3. Most Suitable Near-Term Tokamak Reactor Concept

4.3.1. Copper-Coil Concepts

Table VI is a comparison of the important relevant characteristics of the TORFA/FED-R, ZEPHYR/AFTR, and Riggatron reactor concepts. A tokamak TPR can be competitive with alternative sources of special nuclear materials only if careful attention is given to neutron economy so that as close to 100% as possible of the fusion neutrons are beneficially absorbed. It appears that the Riggatron reactor, even if feasible from the standpoint of plasma physics and technology, is ruled out as a serious TPR by reason of its disastrous neutron attenuation and spectral softening in the TF coils. While uranium could conceivably be added to various regions to multiply the neutron population, it does not seem possible, for example, that the "standard" Riggatron configuration could be a net breeder of tritium.

Table VI. Relevant Characteristics of Copper-Coil Tokamak Candidates

Characteristics	TORFA/FED-R	ZEPHYR/AFTR	Riggatron
Neutron economy	Excellent	Fair to good	Poor (neutrons lost in coils)
Magnetic field requirements	Moderate	Moderate if $n\tau_E$ scaling is favorable	Severe
Plasma heating	Straightforward	Feasible	—[a]
Duty factor	Near 100% if noninductive current drive is feasible	≈ 75%	50% or less
Radiation damage to reactor components	First wall only	First wall and magnets	Severe damage to all components
Access to blankets	Very good	Poor	Excellent
Maintenance and availability	Good access to all in-bore components will allow fast turn-around	Poor access to in-bore components will lengthen downtimes	Damage due to radiation or severe cyclic stress necessitates frequent reactor replacement (weeks to months). Availability depends on time to replace. Blanket regions easily serviced.

[a]Appears impossible to attain reactor temperatures with ohmic heating alone. Not clear that auxiliary heating can be applied to Riggatron configuration.

Table VII. Ranking[a] of Candidate Tokamak Concepts for 1990s Deployment

Concept	Reasons for ranking
TORFA/FED-R with $Q_p \sim 3$	● Plasma performance closest to that already proven in tokamak devices ● Many plasma and machine parameters close to those anticipated for the TFTR and DIII-Upgrade in the mid-1980s ● TF coils use proven technology ● Excellent access to materials production regions ● Duty factor can be 100% if noninductive current drive is feasible; required injected power is compatible with $Q_p \sim 3$
ZEPHYR/AFTR ignited	● TF coils are more advanced than for TORFA, but are proven in principle ● Proposed ignition operation is desirable, but no near-term ignition test is in the offing ● Difficult access to blanket regions ● Achievable duty factor is limited by pulsed ohmic current drive and uncertain impurity control
Superconducting coils (INTOR/FED) ignited	● Superconducting TF coils pose high risks ● No near-term ignition test is in the offing ● Capital cost may be much larger than resistive coil options ● Achievable duty factor is uncertain unless noninductive current drive is used; plasma will then have $Q_p = 5$ to 8
Riggatron ("standard" version) ignited	● Achievement of ignition is highly uncertain ● Loss of neutrons in TF coils makes even tritium self-sufficiency appear problematical ● Achievable duty factor is highly uncertain ● Severe cyclic stresses limit lifetime

[a] In order of suitability.

4.3.2. Relative Ranking

Table VII ranks the four tokamak reactor concepts in order of their feasibility and desirability for implementation as a TPR in the 1990s. The main considerations are:

1. Required plasma performance comparable with that expected to be demonstrated by the mid-1980s in TFTR, DIII-Upgrade, and other large experimental tokamaks
2. Advances beyond the state of the art required for the implementation of the TF coils
3. Achievable duty factor, taking into account pulsing of the TF or current drive systems and means of particle and impurity control
4. Neutron economy and degree of access to materials production regions

In view of the comparisons shown in Tables V through VII, the most suitable near-term reactor concept is TORFA/FED-R. The plasma specifications for this fusion source are closest to those anticipated for TFTR (see Table IV), although a TPR must have a pulse length and duty factor that are orders of magnitude greater. While an ignited reactor such as ZEPHYR/AFTR, with a smaller investment in plasma heating equipment and smaller electrical power consumption, would probably result in a more cost-competitive TPR than one with $Q_p \sim 3$, the choice of an ignited reactor at this time would carry considerable risk because of its significantly greater requirements on plasma confinement and on the β of the *bulk* plasma. In TORFA/FED-R, approximately one-third of the plasma pressure is due to superthermal deuterons and tritons.

Recent experimental results[5] and extensive analyses for test reactors[23] indicate that a magnetic divertor may be essential to reduce surface erosion in the plasma chamber and consequent impurity buildup in tokamaks with high thermal wall loading. The AFTR and Riggatron concepts are not amenable to inclusion of a magnetic divertor, so their achievable pulse length and duty factor are very uncertain. A poloidal divertor is inherent to the TORFA/FED-R concept, although its inclusion entails some loss in the effective blanket coverage factor.

The ignited superconducting-coil tokamaks are advantageous with respect to electrical power consumption. However, in addition to the uncertainties of realizing ignited plasma operation, considerable risk can be incurred by the paucity of operational experience with superconducting coils on tokamak devices even by the late 1980s.

4.3.3. *Neutron Wall Loading*

None of the three highest ranked concepts has $\phi_w = 2\ \mathrm{MW/m^2}$ or more, which was suggested earlier (in Sec. 2.2) as being the minimum desirable for economic competitiveness. For a given fusion power, the superconducting-coil options have significantly smaller ϕ_w than do the TORFA or AFTR options, suggesting an inability of superconducting-coil tokamaks to be cost-competitive for the size range appropriate to a production reactor (~ 500 MW). Future design work should investigate how ϕ_w can be increased by about one-third for the same fusion power level while retaining good access to the production blankets.

4.3.4. *Relative Capital Costs*

The FEDC (Fusion Engineering Design Center) in Oak Ridge has recently designed and costed a version of TORFA, called FED-R, which is intended to serve as a near-term fusion test reactor.[4] The FEDC has also designed and costed superconducting-coil test reactors of comparable fusion power. Their estimated total capital costs for the various reactor types, excluding blankets, have turned out to be rather similar. The reason is that approximately half of the direct cost is accounted for by facilities and equipment that are required for *any* tokamak reactor concept, or indeed for almost any magnetic confinement fusion concept. These common facilities and equipment include (1) buildings; (2) heat-exchangers and cooling towers for the first wall, divertor, and shield/blanket; (3) tritium- and fuel-handling systems; (4) plasma heating systems; (5) vacuum pumping; (6) instrumentation and control; and (7) remote-maintenance equipment. The total direct cost of these items is typically \$400 to \$500 million (1982 dollars).

The direct cost of components peculiar to the type of tokamak under consideration, such as the magnet systems and plasma chamber, ranges between \$300 and \$700 million (1982 dollars) depending on the concept type and fusion power level in the range 150 to 450 MW. Although the AFTR devices have not been costed by the FEDC, there is no reason to expect any deviation from the above results.

4.3.5. *Blanket Systems*

These systems were not costed by the FEDC for their test reactor designs since all of these devices can have only partial coverage of the plasma chamber wall (10 to 30%). For larger devices with complete blanket coverage, such as those listed in Table IV, concepts with smaller wall loadings will require correspondingly larger wall areas for the same fusion power and will, therefore, need more massive blankets of higher total cost. Whether this consideration has significant impact on the cost trends discussed above depends on the relative cost of the production blankets to the total cost of the reactor. Superconducting-coil tokamaks of production reactor size will tend to be penalized on this account, although this drawback may be compensated for by their lower operating cost for electricity consumption.

Since *machine-dependent* capital costs are not an overwhelming factor in determining overall production costs, which reflect both total capital cost and operating cost, cost comparisons can play only a secondary role in the selection of the preferred tokamak concept. The primary bases for selection, which have been discussed in the preceding sections, include realistic prospects for successful plasma operation and blanket performance, as well as ease of access to the production blankets.

4.4. Impact of Power-Producing Blankets

4.4.1. *Motivation*

The electrical power consumed by the copper-coil candidate fusion drivers is of the order of 500 MW or higher. The net power consumption can be reduced by converting the nuclear heat deposited in the blankets to electricity. Efficient conversion requires that the blankets be operated at a temperature of at least 250°C.

If the spatially averaged blanket power multiplication M is assumed to be 1.5 and the thermal-to-electrical conversion efficiency is assumed to be 0.33, the electrical power generated by the production reactor blankets is 0.4 times the fusion power. Subtracting this value from the average electrical power consumption listed in Table IV gives a net power consumption of 350 MW for the TORFA/FED-R case and 200 MW for the AFTR case.

The net power consumption of TORFA/FED-R can be reduced to about 160 MW if M can be increased to 2.5, corresponding to 35 MeV per fusion neutron. Attaining this magnitude of energy gain necessitates recourse to a fissionable multiplier. For example, a one-sixth sector of the blanket operating at $M=10$ with all other sectors operating at $M=1.5$ would provide 390 MW of electrical power.

Whatever the acceptable level of power consumption, should Q_p turn out to be less than the design goal so that additional neutral beam or RF power is required, a certain number of blanket assemblies containing depleted uranium (that would also produce net tritium) could be retrofitted to reduce the net power drain as required.

4.4.2. *Costs*

If hot blankets are used, the capital cost of the plant will increase because of more expensive energy conversion systems, requirements for pressure tubes or vessels in the blanket assemblies and enhanced safety equipment. Safety systems would have to be further upgraded should a fissionable blanket section be installed. However, detailed analysis may show that these capital costs are more than offset by significantly reduced operating costs.

If one or more sectors contains depleted uranium, the production of additional fissile material (e.g., an additional 0.5 ^{239}Pu atom/fusion neutron entering this sector) will result in greater plant revenue. A larger blanket neutron multiplication also allows a reduction in the desired value of ϕ_w below the 2 MW/m^2 recommended in Section 2.

5. SUMMARY

A toroidal materials production reactor (TPR) should have a fusion neutron wall loading of 2–4 MW/m^2 and a fusion energy gain Q_p of at least 3, preferably with steady-state operation. Ease of access to the production blankets is an important requirement. From the combined considerations of state-of-the-art performance, present development programs, and projected reactor characteristics, the tokamak is by far the most viable toroidal reactor candidate for meeting the requirements of a TPR that could be implemented in the 1990s.

Of the various tokamak reactor concepts, the TORFA/FED-R type of reactor with $Q_p \sim 3$ is the most suitable candidate for a TPR from the point of view of minimal extrapolation of plasma parameters and fusion technology beyond the TFTR level, as well as ease of access to the production blankets. This concept was recommended and adopted as the reference toroidal reactor in companion papers in this issue.

"Fusion Technology for a Magnetic Fusion Production Reactor"[2] makes specific recommendations for modification of or additions to the DOE magnetic fusion energy program to expedite the capability of implementing a competitive tokamak materials production reactor in the 1990s. Resistive-coil tokamaks appear to offer much greater flexibility than do superconducting-coil tokamaks in configurational changes that might result in reduced cost. Hence, future work should analyze suggested approaches for reducing the production cost per gram of fusion neutrons in modified versions of the TORFA/FED-R concept.

APPENDIX A: UPDATE FOR TOROIDAL SELECTION, 1983

Introduction

This appendix updates (through December 1983) the evaluation of toroidal fusion reactors used for the production reactor mission reported here and in a companion paper.[(2)] Reference 2 discusses the fusion technology for the preferred concept and some familiarity with the concepts therein is presupposed. Here more recent developments are discussed with regard to their possible impact on the capability, costs, time scale for implementation, and technological risks of a toroidal fusion reactor designed for the materials production mission.

Overview of 1983 Events

The year 1983 saw an extension of the dominance of the tokamak concept in toroidal magnetic confinement fusion research, and indeed probably in all of fusion research. The principal experimental event was the start-up of the TFTR (Tokamak Fusion Test Reactor) at Princeton Plasma Physics Laboratory (PPPL), and the determination that its energy confinement scales according to the most favorable of the various scaling laws that had been derived from results on smaller tokamaks. (Only ohmic-heated plasmas were operated in 1983, however.)

There was relatively modest progress in the development of fusion reactor technologies. On the theoretical front, the most important new development was the demonstration with numerical plasma codes that tokamak plasmas with appropriate shaping and elaborate poloidal-field systems can achieve β of several tens of percent. (β is defined as the ratio of plasma pressure to confining magnetic field pressure.)

Here we discuss the implications of all these results for the toroidal Magnetic Fusion Production Reactor (MFPR) concepts.

Impact of More Recent Developments in Experimental Plasma Physics

The most important experimental development in 1983 was the bringing of TFTR on line, and the determination that its confinement properties scale according to the most favorable of the various scaling laws that had been derived from results on smaller tokamaks.[26] However, only ohmic-heated plasmas have been operated in TFTR through 1983. Experiments in smaller tokamaks with intense beam or RF heating have generally revealed a degradation in energy confinement time τ_E with increasing heating power density. This degradation has been shown to be avoidable in tokamaks with a poloidal magnetic divertor, which in fact was included in TORFA-D2, the preferred tokamak MFPR concept. All these results taken together provide support for the design specifications given in the main body of this work.

Another important development in 1983 was the demonstration in the Princeton Large Torus device at PPPL that the plasma current can be started up at lower density entirely with RF power in the lower hybrid frequency range. This demonstration provides credence to the backup operational mode for the MFPR plasma, which would be adopted in the event that the neutral-beam injector development required for the baseline operational mode is not realized. (Experimental demonstration of sustained current drive by injected neutral beams was realized on TFTR in 1985 and 1986.) In the backup operational mode, RF energy is injected to start up the plasma current, which is then maintained by the central transformer.

Impact of More Recent Developments in Theoretical Plasma Physics

The most important theoretical development in 1983 was the demonstration with plasma equilibrium and stability codes that very high β operation (tens of percent) is feasible in principle in tokamaks with appropriate plasma shaping and poloidal field (PF) coil systems.[27,28] Operation at higher plasma β means that the toroidal magnetic field will be reduced for the same fusion power production, leading to a reduction both in cost and in construction difficulties of the TF (toroidal-field) magnets. However, this advantage is partly offset by the more elaborate PF coil systems that are required to realize very high-β plasmas. In addition to much higher current requirements for the regular PF coils, a set of high-current pusher coils must be located at the midplane as close as possible to the inboard edge of the plasma in order to help generate the required plasma "bean shape." The aspect ratio R/a (major radius divided by minor radius) of TORFA-D2, the reference tokamak MFPR concept, is near the minimum value of 3.5–4 required to allow entry into the stable very high β regime (the so-called "second region of stability").

If tokamaks of very high β can be realized, there would be little impact on the viability of any superconducting-coil MFPR option. While the cost of the tokamak device itself might be reduced somewhat, the cost of auxiliary components such as power supplies and remote handling equipment, of shielded buildings, of heat conversion systems, of tritium processing systems, etc., are (in total) much greater than the cost of the fusion device and would not be reduced.

On the other hand, the various copper-coil options may become more viable with plasmas of higher β, because the reduction in magnetic field accompanying the increase in plasma β can result in a significant reduction in the circulating electrical power to operate the magnets. Taking into account the increased power needed to operate the PF coil system, the savings in circulating power could be 100–150 MW. Reduction in the maximum stresses experienced by the TF coils also allows greater freedom in the design and location of the demountable joints, and therefore can reduce downtime for replacement of in-bore components. However, some design concepts of high-β tokamak reactors have an increased number of in-bore components that might eventually need maintenance.

More Recent Developments in Alternative Toroidal Concepts

The year 1983 was unfavorable for nontokamak toroidal concepts. The only changes that need be made in Table II to account for developments in 1983 are reductions in the temperature and $n\tau_E$ values for the EBT entry as a result of more accurate plasma diagnostics. The planned next step device in the EBT program, called EBT-P, has been cancelled by the Department of Energy/Office of Fusion Energy. Small EBT programs are continuing at Oak Ridge National Laboratory (ORNL) and elsewhere.

Stellarators

No results showing improved parameters have been reported from 1983 stellarator experiments. A new large stellarator, called the ATF, is currently under construction at ORNL. In the late 1980s it should provide critical data concerning the viability of the stellarator approach.

Reversed-Field Pinches

Reversed-field pinches (RFPs) continue to demonstrate $n\tau_E$ values that are two to three orders of magnitude smaller than have been achieved to date in tokamaks. There is still no experimental justification for considering RFPs as potential near-term fusion neutron sources. The largest RFP experiment to date, called RFX, will be built in Italy by Euratom with completion at some indefinite date. Another large RFP-type experiment may be built at Los Alamos National Laboratory (LANL) in the late 1980s. Following operation of one or both of these devices, a reevaluation of the prospects for the RFP may be in order.

Compact Tori

Initial work has been reported at LANL and PPPL on compact toruses, such as the spheromak,[29] which were not discussed in the main body of this paper. However, these devices are still at an extremely early stage of development (e.g., $T_e < 100$ eV and pulse length < 1 ms), and it will be many years before their prospects for use as the basis of an MFPR can be examined seriously.

Impact of Recent Developments in Fusion Technology

In 1983 there were no dramatic new developments on the technology front relevant to toroidal fusion devices. Progress continued in supplying the superconducting TF coils for the ORNL Large Coil Project, but no tests were initiated in 1983. This project is of only tangential relevance to the preferred MFPR concept, which is based on copper TF coils.

The very high-field tokamak option requires high-stress, high-conductance copper alloys for fabrication of the magnets (see the discussion in Appendix B). The strongest such alloys contain Cu, Be, and Ni, but are commercially available only in very thin sheets. In the last year, INESCO and Brush-Wellman have developed a high-purity CuNiBe alloy, which has been made in plates up to 1 m^2 in size. The greater the copper content, the higher the alloy's conductivity but the lower its strength. For example, the alloy has a conductivity of 55% of that of OFHC copper at a stress of 150 ksi and 74% at 97 ksi.

Tables III and IV of Ref. 2 summarize the fusion technology development requirements for TORFA-D2. Inorganic insulation is needed for the TF coils in order to minimize the amount of shielding required to protect the coils and therefore the reactor size and cost. The reference MFPR design specifies the magnesium aluminate inorganic called SPINEL. In 1983 INESCO demonstrated that a suitable inorganic 250-μm ceramic oxide coating will remain attached to CuNiBe under conditions of high temperature and voltage drop. While the development of the CuNiBe alloy is not relevant to the baseline MFPR, the results on inorganic insulators are an important feasibility demonstration for any copper coil MFPR.

Turning to the other areas listed in Tables III and IV of Ref. 2, there has been steady progress made in the development of high-frequency gyrotrons (to be used for plasma initiation) and in practical magnetic divertor operation (for disposal of plasma thermal flux and impurity ions). Work continued in developing long-pulse positive-ion-based neutral beams that are to be used in the largest existing fusion devices. The development of negative-ion-based neutral beams proceeded at a low funding level in the fusion program, but is expected to receive increasing support in the space-based defense program. Important work on high-current negative-ion beams is proceeding in Japan. There has been relatively little support for the development of remote maintenance systems specifically for fusion devices.

Status of the Next-Step Tokamak and and Relation to the MFPR

The Department of Energy/Office of Fusion Energy is currently considering designs for a tokamak ignition test device that, if approved, would come on line in the early 1990s. One concept for this device, called TFCX,[30] features copper TF and superconducting PF magnet systems that are essentially the same as for the baseline MFPR concept (TORFA-D2). Another approach under consideration favors more compactness and makes use solely of copper coils.[31] The major parameters of the preferred TFCX, as of October 1983, are given in Table AI. This concept features a "D"-shaped plasma of moderate β similar to that in the baseline MFPR. However, there is an important difference in the operational procedures for driving the plasma current I_p. In the TFCX, I_p is to be initiated by RF power and sustained by the central solenoid for a 300-s pulse. In TORFA-D2, I_p is to be initiated by the ohmic-heating solenoid and sustained in the steady state by injected neutral beams. The backup

Table AI. Comparison of Reference Tokamak MFPR and TFCX Concept

Parameter	MFPR (1982 design)	TFCX (1983 concept)	JET (operating)
Geometry			
Major radius (m)	3.9	3.00	2.95
Minor radius (m)	0.95	1.20	1.25
Aspect ratio	4.1	2.5	2.35
Elongation	1.5	1.6	1.6
Inboard blanket/shield (m)	0.8	0.2	0.1
Plasma			
B at plasma axis (T)	5.0	3.8	3.4
$\langle\beta\rangle$	0.06	0.083	0.05
⟨Temperature⟩ (keV)	20	12	7.0
⟨Density⟩ (10^{13} cm^3)	7.5	8.0	7.0
Plasma current (MA)	5.5	11.0	6.0
Solenoid flux (Wb)	13.0	12.0	25.0
Auxiliary heating method	Beams	RF	Beams & RF
Heating power (MW)	150	60	25
Plasma heat removal	Poloidal divertor	Pumped limiter	Limiter
Magnets			
TF horizontal bore (m)	4.5	3.9	3.1
TF vertical bore (m)	6.75	5.2	4.9
TF coil material	Cu plates	Cu plates	Cu, wound
Maximum B at coils (T)	9.8	7.8	7.0
PF coil material	NbTi & Cu	NbTi	Cu
Power flow			
Fusion power (MW)	450	230	> 25
⟨Neutron wall load⟩ (MW/m^2)	1.4	1.0	0.2
Duty factor	> 0.95	0.1	0.01
TF coil loss (MW)	220	350	280
PF coil loss (MW)	50	10	—
Circulating power (MW)	575	425	650

mode for TORFA-D2, discussed in Ref. 2, is similar to the proposed TFCX operational mode.

Table AI also gives the parameters of the Joint European Torus (JET), which very recently came into operation. This tokamak is somewhat larger than the TFTR and is expected to demonstrate still more advanced plasma physics and fusion neutron performance by the late 1980s.

In the Soviet Union leaders of the fusion program are proposing to construct a tokamak test reactor that would actually produce about 150 kg of Pu per year as well as 300 MW of electricity. It is not clear whether it would be a net consumer or producer of tritium. This tokamak would be very large (having a 5.5-m major radius), use superconducting TF coils, have a fusion power of approximately 500 MW, and a duty factor exceeding 80%. Evaluations in the United States of similar-sized fusion demonstration plants indicate that the proposed Soviet facility would cost at least several billion dollars. While extensive design studies will no doubt continue, there is no indication that the Soviet government would approve the construction of such an ambitious project.

Recommendations for New Design Variants

Operation at Higher β

The impact on the MFPR design of operating at much higher β should be examined. This option would facilitate the technology aspects associated with the TF magnets and reduce electrical power consumption, but a much more elaborate PF coil system would be required to implement "bean shaping" of the plasma. All the components inboard of the plasma would have to be resized, the effective blanket coverage might be reduced slightly, and reactor maintenance might be complicated. Table AII compares the most important plasma parameters of an illustrative bean-shaped tokamak configuration with those for the reference TORFA-D2 tokamak.

Table AII. Illustrative Parameters of a High-β Tokamak MFPR

Parameter	High β	MFPR (1982)
Geometry		
Major radius (m)	3.4	3.9
Minor radius (m)	0.90	0.95
Aspect ratio	3.8	4.1
Elongation	1.4	1.5
Inboard blanket/shield (m)	0.8	0.8
Plasma		
B at plasma axis (T)	3.4	5.0
$\langle\beta\rangle$	0.20	0.06
⟨Temperature⟩ (keV)	20	20
⟨Density⟩ ($10^{13}/cm^3$)	12	7.5
Plasma current (MA)	6.4	5.5
Solenoid flux (Wb)	24	13
Heating power (MW)	150	60
Magnets		
TF horizontal bore (m)	4.5	4.5
TF vertical bore (m)	6.7	6.75
TF coil material	Cu plates	Cu plates
Maximum B at coils (T)	6.75	9.8
PF coil material	NbTi & Cu	NbTi & Cu
Power flow		
Fusion power (MW)	450	450
⟨Neutron wall load⟩ (MW/m^2)	1.8	1.4
Duty factor	> 0.95	> 0.95
TF coil loss (MW)	130	220
PF coil loss (MW)	60	40
Circulating power (MW_e)	520	575

The conceptual design work on various high-β copper-coil tokamak test reactors will have direct bearing on an alternative design of the MFPR tokamak driver for higher-β operation.

Experimental investigations of moderate-β bean-shaped plasmas began in 1984 on the PBX device at PPPL.[27] Information on the feasibility of $\langle\beta\rangle \gtrsim 10\%$ plasmas will be available by the late 1980s.

Plasma Current Drive

Present Department of Energy/Office of Fusion Energy plans call for rather slow development of high-current, high-energy, negative-ion-based neutral beam injectors. (But development of the ion sources and neutralizers is garnering increased support in the strategic defense program, and a strong negative-ion beam program is underway in Japan.) To hedge against the possibility of these injectors not being available in the 1990s, a design modification of TORFA-D2 should be worked out to accommodate the proposed TFCX operational scenario. As discussed in Ref. 2, that scenario would result in a duty factor of the order of 0.9 and an increase in the production cost per fusion neutron. However, the MFPR fusion technology would be simplified by elimination of steady-state operation of the complicated and radiation-vulnerable neutral beam injectors.

New Approach to Fabrication

In the course of the FY-82 study, a new way to construct a tokamak MFPR was conceived. While this scheme cannot be discussed herein, it can be stated that the magnet coil and blanket are integrated in a system that is directly exposed to the fusion neutron source. The entire assembly may be processed chemically to recover the special material. The construction approach offers a potential means of reducing the capital and operating costs of the MFPR, and perhaps simplifying material recovery as well.

APPENDIX B: UPDATE ON PROSPECTS FOR A VERY HIGH-FIELD TOKAMAK MFPR

Introduction

Here we define a "high-field tokamak" as one having a field at the plasma center of 10 T or more. The very high-field tokamak option (such as that embodied in the proposed Riggatron device) was examined in the main body of this paper with regard to its potential use as the neutron source of a magnetic fusion production reactor (MFPR). The high-field concept was found to be inferior in performance and prospects to tokamak drivers with copper coils operating at lower fields. However, because there appears to be increasing interest in high-field tokamaks of extreme compactness (major radius R less than 1.5 m), we have examined new information on this topic to determine whether the high-field approach has become more attractive for the MFPR mission. The conclusion of this reexamination is that this option remains unattractive for fundamental reasons in each of the three critical areas of plasma physics, plasma engineering, and nuclear engineering:

1. New data on energy confinement scaling continues to raise serious doubt that the conditions for ignition can be achieved in very small devices.
2. There is inadequate space in very compact tokamaks for the complex poloidal-field coil systems that are needed for operating high β plasmas; that option would alleviate the overwhelming engineering challenges of the toroidal field magnets.
3. Even with a working device, the excessive loss of fusion neutrons in the magnets raises doubt about the feasibility of generating substantial net tritium or fissile material.

Progress in Compact Tokamak Development Programs

We are aware of three programs underway to implement very high-field, extremely compact, tokamak test reactors. All of these programs have existed for several years, and none received increased funding in 1983. However, there appears to be wider interest in their prospects, especially because of the increasingly poor prospect of funding being made available to construct a large superconducting-coil tokamak test reactor. There is presently renewed interest in both compact and moderate-sized copper-coil tokamak test reactors. The following discussion pertains to very compact devices ($R < 1.5$ m).

T-14

The T-14 device in the Soviet Union is designed only to reach fusion energy breakeven in a short pulse utilizing plasma compression.[32,33] All fusion neutrons will be absorbed in the massive coil system immediately surrounding the plasma. A small model of T-14 has successfully undergone magnet performance testing. As of mid-1983, the actual construction of T-14 had not been approved by the Soviet authorities.

Ignitor

The Ignitor device, which is under construction by Euratom, is similar in concept to T-14 but is somewhat larger, having as its objective the demonstration of ignition in a short pulse. Like the T-14, all fusion neutrons would be absorbed in the massive close-fitting magnet structure. The prospects for funding of Ignitor beyond the conceptual design stage are still dim. The proponents of T-14 and Ignitor have made no proposals for a follow-on device that could generate fusion neutrons for useful application. Neither device lends itself to practical neutron utilization because of the lack of space for a blanket, the enormous power drain of the magnets, and the inherently short duty factor that results from inertial cooling of the magnets.

Riggatron

The Riggatron concept once pursued by INESCO Inc., is the only high-field device whose proponents claim will have serious application viz. the production of fissile material. This device was covered fully in the main body of this paper and is given the dominant consideration here.

In 1983, major changes were made in the Riggatron design.[33] These include (1) injection of up to 10 MW of RF power that was accepted by INESCO as being essential for reaching ignition temperature, and (2) increase of the reference plasma major radius from 0.7 to about 1 m, with still larger sizes under consideration.

Considerable design work was done on the proposed first test device, called FDX-1, which is intended to operate for at least 1000 cycles at full field (16 T) or 10,000 cycles at 75% of full field. Other parameters include a plasma current up to 8 MA, vertical elongation up to 1.4, and a pulse length of several seconds.

Impact of Recent Theoretical Developments

In 1983, work with plasma equilibrium stability codes demonstrated that very high β operation is feasible in principle to tokamaks with appropriate plasma shaping and PF coil systems. Here β is defined as plasma pressure divided by magnetic field pressure. The consequent reduction in toroidal magnetic field for the same fusion power production would alleviate the fabrication and operational difficulties of the magnets in high-field tokamaks, as well as significantly reduce their electrical power requirements. However, to realize high-β operation requires the use of an elaborate poloidal-field coil system inboard of the plasma center. There is apparently no physical space for such coils in the ultracompact high-field tokamaks considered here. In fact the 1-m Riggatron is unable to accomodate coils that will allow plasma elongation greater than 1.4, so that the volume-averaged β will be restricted to 4 or 5%. Because of space limitations, the current density in the ohmic-heating coils must be 30 KA/cm^2.

Impact of New Data on Confinement Scaling

Size Scaling

Recent experimental results from the TFTR device at Princeton Plasma Physics Laboratory, the largest operating tokamak in the United States, have consolidated and extended previous data for the size scaling of energy confinement time τ_E in ohmic-heated tokamaks.[26] The parameter τ_E increases approximately linearly with minor radius and quadratically as the major radius R_p, and obviously favors larger machines.

Density Scaling

Data from most tokamaks show that at low to moderate densities n, τ_E is proportional to n. However, this relation tends to break down at higher densities, and in particular has stymied very high-field experimental devices from realizing larger $n\tau_E$ values. But in late 1983, the Alcator-C device at MIT was able to extend the proportionality to higher density by fueling the center of the plasma with pellets injected at 1 km/s.[34]

Relevant Developments in Fusion Technology

The three principal materials development requirements for very high-field compact tokamaks are (1) high-strength, high-conductivity copper alloy for the TF (toroidal-field) and PF (poloidal-field) coils; (2) an inorganic insulator for these coils; and (3) first-wall protection. Whereas the Ignitor and T-14 magnets are to operate at liquid-nitrogen temperature and are inertially cooled, the Riggatron magnets are water-cooled and are intended for steady-state operation in "commercial" reactor versions, although not in the test devices.

High-strength copper–nickel–beryllium alloy is currently available commercially only in very thin sheets. INESCO and Brush-Wellman have developed proprietary high-strength, high-conductance CuNiBe alloys, which have been formed into plates up to 1 m^2 in size.[33] An alloy with a conductivity of 55% of that of OFHC copper can withstand stress of 150 ksi, whereas an alloy with 74% OFHC copper conductivity can withstand 97 ksi stress. This material could allow steady-state operation of the FDX-1 magnets at the design field of 16 T at the plasma center and almost 30 T in the central solenoid.

According to INESCO, a suitable insulator is a 250-μm thick layer of an undisclosed type of ceramic oxide. INESCO tests reportedly have shown that this insulator will remain bonded to the above alloy at 200°C and at large voltage differentials.

Table BI shows the best results from Alcator-C and compares them with what one might expect to get using the field and geometry of the proposed FDX-1. The $n\tau_E$ that is apparently achievable in ohmic-heated plasmas is sufficient for ignition given a central plasma temperature of about 12 keV. However, there remain several obstacles to attaining the required temperature. These are:

1. The required $\langle\beta\rangle$, although only moderate, may not be sustainable because of lack of space for the required PF-coil, plasma-shaping system.
2. Ohmic heating alone will be insufficient to reach the temperature at which fusion/α particle heating can become important (about 7 keV in the plasma center).
3. Because FDX-1 designers are aware of item 2, they are specifying the use of RF heating to achieve ignition temperature. But the FDX-1 will then be subject to the degradation in τ_E observed in tokamaks with intense neutral-beam or RF heating, and possible with α-particle heating as well. In these auxiliary heated regimes, τ_E is found to be essentially independent of density but increases with plasma current. Experiments in the ASDEX (at Garching), PDX (at Princeton), and DIII (at General Atomic) tokamaks have shown that this degradation can be overcome by using a poloidal magnetic divertor, but the versions of the Riggatron presently under consideration (such as the FDX-1) are much too small to accommodate a divertor. Thus the only recourse to ensure attainment of ignition would seem to be still larger R_p to take advantage of the increase of τ_E with size and current.

Table BI. Comparison of Alcator-C with FDX-I

Parameter	Alcator-C (achieved)	FDX-I (design)
Major radius	68 cm	100 cm
B at plasma	11 T	16 T
$\langle n\rangle$	$1\times10^{15}/cm^3$	$2\times10^{15}/cm^3$
$\tau_E{}^a$	50 ms	100–200 ms
$n\tau_E^a$	$5\times10^{13}/cm^3$	$2\text{–}4\times10^{14}/cm^3$
Temperature at center	1.5 keV	12 keV needed
Average temperature	0.5 keV	5 keV needed
$\langle\beta\rangle$ (%)	0.4	3.5

[a]Applies only to ohmic-heated regime.

Ignitor Concept

In the present design of the Ignitor device, the plasma in the compressed stage has the same dimensions as those of the earlier Riggatron plasma (major toroidal-plasma radius $R = 0.75$ m, minor toroidal-plasma radius $a = 0.25$ m). Although the prescribed radiofrequency preheating followed by compression is capable of bringing the plasma to ignition temperature, the probable scaling of τ_E in the auxiliary heated regime indicates that at best ignition could be obtained transiently (for perhaps one second immediately after compression), but could not be sustained. It is possible that ignition could be maintained for several seconds if a way were found to increase the plasma current while the plasma expands after compression.

Breeding Prospects of the Riggatron

The Riggatron blanket is to be located completely outside the tokamak device. In the main body of this paper, it was argued that the Riggatron would have marginal breeding performance. A paper published in 1983 by INESCO authors claimed that the TBR (tritons bred per fusion neutron) would be in the range 1.0–1.2.[35] Our analysis indicates that this result is a serious overestimate, for the following reasons:

1. The thickness of the TF coils (through which the neutrons must pass on their way to the blanket) is taken only as 7 cm in the entire outer half of the tokamak. This thickness is several times too small for a quasi-steady-state TF coil system with the enormous fields that must be generated by the Riggatron. Even if a miraculous cooling system could be engineered, the electrical power consumption in the TF coils would be of the order of 1 GW.
2. The poloidal-field coils were apparently omitted from the neutron model (see Fig. 5 of Ref. 5), although Fig. 1 of Ref. 5 shows that these coils have notable size and will absorb significant neutron flux.
3. The blanket consists entirely of lithium; all structural components were omitted in the neutronic model.

Our rudimentary analysis continues to indicate that net fissile or tritium breeding in the Riggatron is unlikely. Substantial fissile breeding would be possible *only* if an *external* source of tritium could be made available. (By comparison, the total breeding ratio in the baseline MFPR is about 1.55.)

Technical Prospects

Although the reference Riggatron is somewhat larger than it was in the late 1970s, it is still too small to achieve sustained ignition conditions, to permit installation of the PF coil system needed to achieve high β, or to permit utilization of a practically large fraction of the fusion neutron generation.

The consequence of having to operate at relatively low values of β is that these very compact tokamaks will always have electrical power requirements for the magnets exceeding what the reactors themselves could produce even with uranium blankets having significant neutron energy multiplication (the preferred Riggatron blanket concept).

As ohmic heating to ignition was eventually abandoned by INESCO, there was actually little point in retaining a tiny device size. In fact INESCO began to parameterize machines of size in the range up to $R_p = 2$ m. Similarly, the design of the Ignitor device featuring compression and short-pulse ignition will probably undergo continued evolution to larger size. Compact copper-coil tokamaks with $R_p > 2.5$ m and magnetic field $B < 8$ T have been under consideration by other groups, so that a search for the optimal size seems likely to result in a merging of the very high-field concepts with moderate-field ones. (Note that our baseline MFPR design has $R_p = 3.9$ m.)

Overall Conclusion

As of 1983, it appeared that a moderately high-field copper-coil device emerging from the convergence of design concepts discussed herein and elsewhere would prove to be the most cost-effective vehicle for demonstrating thermonuclear ignition of a magnetically confined plasma. However, there remains great uncertainty concerning the time scale for implementing an ignition demonstration device, and it is likely 10 years away. Furthermore, the tokamak fusion driver for an MFPR would have to be somewhat larger and have a much lower field than is characteristic of an ignition test device in order that the electrical power consumption of the MFPR be acceptable. The breeding blanket would have to be contained within the device itself to avoid unacceptable neutron loss. Thus in our opinion the very high-field ultracompact tokamak approach is not an option for the production reactor mission.

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