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11/13/80 T.S.

11. 1981

SEPTEMBER 1980

PPPL-1700
UC-20d**MASTER**

TORFA

TOROIDAL REACTOR FOR
FUSION APPLICATIONS

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PPPL-1700

September 1980

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ABSTRACT

The near-term goal of the U.S. controlled fusion program should be the development, for practical applications, of an intense, quasi-steady, reliable 14-MeV neutron source with an electrical utilization efficiency at least 10 times larger than the value characterizing beam/solid-target neutron generators. This report outlines a method for implementing that goal, based on tokamak fusion reactors featuring resistive toroidal-field coils designed for ease of demountability.

TORFA

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1. INTRODUCTION, OBJECTIVES, AND DESIGN APPROACH

1.1 THE NEAR-TERM ROLE OF FUSION ENERGY

This section considers briefly the potential practical applications of controlled fusion energy in the *near term*, defined as the next several decades. Potential long-term applications of controlled fusion are not addressed. The statements made herein are essentially the opinions of the author.

Electricity Production

During the next 50 years the growth of electrical power capacity in the world's industrialized nations is likely to be much smaller than that envisaged in the recent past. In view of the many ways by which electricity can be produced, and because many of these electricity-producing technologies are well established, controlled fusion power plants will have to be demonstrably cheaper (e.g., in capital cost per installed kilowatt) or markedly more environmentally acceptable than competitive technologies serving the electricity market. While such advantages may accrue to fusion power in the distant future, they seem unlikely to be demonstrated in the near term.

Thus in order to help meet an inevitable demand on the controlled fusion enterprise for "repayment" of development costs, which even now have reached an accumulative value of several billion dollars, applications of fusion energy other than electricity from "pure fusion" must be pursued seriously. If alternative applications which are viable, and which are also difficult to achieve by other technologies, are indeed found, then the fate of fusion power development will be only marginally dependent on the growth rate of electrical power usage.

Fusion-Neutron Applications

1.1.6 Proposed applications of fusion energy tend to exploit the special capabilities of the 14-MeV neutrons that carry away 80% of the energy produced in deuterium-tritium fusion generators. Potential near-term applications include the following processes:

- (1) Net production of tritium.
- (2) Net production of electricity with blanket modules characterized by high energy multiplication of incident fusion neutrons.
- (3) Large-scale production of valuable isotopes that are more costly to manufacture with low-energy neutrons or by charged-particle bombardment.
- (4) Sustaining large regions of very high temperature ($\geq 1200^{\circ}\text{C}$), difficult to attain in any other way, for the generation of chemical fuels by high-temperature electrolysis, thermal decomposition, or other methods.
- (5) Generation of very high-temperature heat ($\geq 1200^{\circ}\text{C}$), for use in industrial processes external to the fusion reactor.
- (6) Programs for radiation hardening of electronic instrumentation and other neutron-sensitive components.
- (7) Production of high densities of energetic ions by energy transfer from neutrons, for use in biological, electronic, and radiolytic processes, especially those requiring large irradiation volumes.

1.1.7 Most of these potential applications have been discussed elsewhere in considerable detail. Furthermore, items (6) and (7) have for some years reached a practical stage, using 14-MeV neutrons generated by the bombardment of tritiated solid targets by high-energy deuteron beams (200 to 400 keV). Fusion machines such as the one depicted in Fig. 1-1 are presently used for neutron cancer therapy and neutron activation analysis, while larger machines such as the one shown in Fig. 1-2 are used for materials damage studies and radiation hardening programs. Of course the market for such applications is rather small.

PRACTICAL FUSION REACTOR

$$Q_p = 0.002$$

$$\text{Cost} = 10 \text{ M \$ / kW}$$

Figure 1-1. Beam/solid-target fusion-neutron source used for cancer therapy. [Modified from figure in K.A. Schmidt and H. Dohrmann, *Atomkernenergie* 27 (1976) 159.]

Figure 1-2. Beam/solid-target fusion-neutron generator at the Lawrence Livermore Laboratory. [Taken from LLL report UCRL-52000-78-3.]

On the Practicality of Driven Reactors

1.1.8 The beam-target fusion devices illustrated in Figs. 1-1 and 1-2 are characterized by fusion power amplification $Q_p \sim 10^{-3}$, where

$$Q_p = \frac{\text{Fusion Power}}{\text{Injected Heating Power}} \quad (1.1)$$

The capital cost of these fusion machines is of the order of \$10,000 per watt of fusion power ($K \sim \$10$ million per kW). These values of Q_p and K , which apply to reliable and useful fusion machines, should be kept in mind by those who insist that for controlled fusion to be of any practical use, the fusion plasma must be ignited (i.e., $Q_p = \infty$) and K can be no higher than \$1000 per kW — even if the cost of developing such a fusion machine is tens of billions of dollars. *The presentday applications cited are sufficiently unique so that a small unit cost, and reliable high-duty-factor operation, are the most important attributes of the fusion-neutron source.*

1.1.9 If practical fusion machines of increasing Q_p can be developed, then one would expect to uncover more and more opportunities for "economically viable" applications. While the attainment of even $Q_p = \infty$ (i.e., ignition) is by no means a sufficient condition for the commercially attractive production of electricity or of process heat, $Q_p = \infty$ would certainly improve the economics of those relatively unique applications of fusion that had proved potentially useful at lower values of Q_p ; the electricity consumed per fusion neutron would be that much reduced. Similarly, the achievement of D-D fusion reactors with wall power loadings comparable with those offered by D-T reactors would be expected to further enhance economic competitiveness, or lead to still more applications, because of elimination of the need to utilize a portion of the fusion neutrons to breed tritium.

1.1.10 At any rate, the examples cited above of successful fusion machines operating at $Q_p \sim 10^{-3}$ suggest that additional useful applications might be available at the $Q_p \sim 1$ level, for example, where the fusion plasma is still externally driven.

1.2 NEXT STEPS IN CONTROLLED FUSION DEVELOPMENT

Demonstration of Useful Applications

1.2.1 In view of the discussion in Sec. 1.1, the principal purpose of near-term fusion energy development should not be the single-minded one of electrical power production with "pure fusion" reactors, much less the scientific objective of studying the plasma physics of a fusion generator. The principal purpose of the immediate controlled fusion program should be *the development of an intense, quasi-steady, reliable 14-MeV neutron source with a fusion power multiplication Q_p many times larger than the value characterizing beam/solid-target systems ($Q_p \sim 10^{-3}$), and with the capability of convincing technical demonstrations of useful fusion-neutron applications.*

1.2.2 The materials radiation-damage test facilities, based on magnetic confinement fusion systems, that have been proposed from time to time are not examples of a "useful application" in the near term. These facilities have been represented as part of a development program for a fantasized sequence of fusion reactors. That is, the implementation of a materials test reactor alone is predicated on the subsequent implementation of a series of demonstration power or fuel-producing reactors. However, operation of a materials test reactor cannot provide justification for pursuing these conjectural later facilities.

1.2.3 Up to now magnetic confinement fusion research has been analogous to the development of rockets without a payload or experimental package (even the V-2 had one!), or to the continued development of particle accelerators without ever making practical use of the high-energy beam. It is time now to develop a magnetically confined fusion machine for the principal purpose of *applying* fusion energy, rather than exclusively for the further development of the fusion source. The latter development will benefit in any event, just as the building of rockets and accelerators for practical applications contributed to the technical development of rockets and accelerators themselves. If one accepts that still more potential applications will become industrially or commercially attractive with the development of reliable quasi-steady fusion sources of ever-increasing Q_p , then it becomes clear that at each stage one should attempt to exploit present fusion technology to realize practical fusion generation facilities.

Exploitation of the Tokamak Concept

1.2.4 In the development of any technology, alternate approaches often appear more attractive conceptually than the one that has so far achieved the greatest success. But the mainline approach has usually achieved its preeminence in competition with a host of other approaches that failed, and the newer alternate approaches appear more attractive on the basis of relatively little information. A working model of any kind in a given technological field can be realized only by the well-based selection of the most promising approach that lends itself to fruitful development in the near term, and then by suspending development work on alternate theoretical concepts.

1.2.5 If it is accepted that the next step in controlled fusion research is to perfect a high-duty-factor fusion device that offers a Q_p many times higher than characterizes the beam/solid-target system, then essentially all the resources available to the fusion program should be concentrated on tokamak development until success is achieved. The perceived superiority of the tokamak to meet the required fusion development goals is based on demonstrated performance to date, the high probability of vastly improved performance to be demonstrated in advanced tokamaks recently commissioned or under construction, and the much poorer (or in most cases, hopelessly poorer) performance of all other magnetic confinement and inertial confinement fusion schemes. In the very long run, one might well find fusion generators superior to the tokamak. Nevertheless, only when the above development goal is achieved can one afford to contemplate seriously other options.

1.2.6 As a corollary of the above considerations, development funds for fusion technology should be concentrated on the specific R&D needs of the 14-MeV neutron facility advocated in Sec. 1.2.1, rather than on development tasks purported to be "generic" to fusion reactors.

1.3 NEED FOR TRITIUM SELF-SUFFICIENCY

Unsuitability of Fission-Reactor Tritium Production

1.3.1 As long as tritium supplies for fusion experiments and reactors must be purchased from outside the fusion program, then one of the two basic fusion fuels will be in effect uranium-235! The presentday source of most tritium is breeding in lithium-alloy assemblies inserted in heavy-water fission reactors, and this source will be adequate for the supply of D-T fusion machines for the next 20 years or so. Nevertheless, the time-averaged fusion power production in the U.S., for whatever purpose, cannot be expected to exceed about 200 MW unless the fusion program becomes self-sufficient in tritium. The 12 kg of tritium which would be burned annually to sustain 200 MW of fusion power is presently valued at \$90 million, and its production would require the burning of at least 1200 kg of U-235 in fission reactors with a total thermal power of at least 2400 MW.

1.3.2 In heavy-water reactors, neutron capture in deuterium also produces tritium, but the energy released by nuclear burning of the tritium produced in that way would be only 1/300 of the thermal energy generated in the fission reactors producing the tritium.

1.3.3 Thus the growth of fusion generation above the 200 MW level would seem to demand something approaching tritium self-sufficiency. Hence an important objective of the first practical U.S. tokamak fusion machine is that a large fraction of its tritium requirement be bred in situ, or at the very least, that a convincing demonstration be made in this facility that tritium self-sufficiency is obtainable when desired.

1.4 OBJECTIVES OF THE TORFA CONCEPT

The fusion facility that is proposed herein to meet the requirements outlined in the preceding discussion is called TORFA, for "Toroidal Reactor for Fusion Applications."

Principal Objectives

- 1.4.1 (I) *Demonstrate the engineering feasibility of large-scale fusion-neutron production of tritium, chemical fuels, and nuclear isotopes, in an integrated blanket/tokamak system that is prototypical of a production plant.*

Inherent in this objective are the achievements of (1) reliable high-duty-factor D-T burning of a large tokamak plasma, and (2) practical fusion reactor blanket capabilities. Wherever possible, TORFA will make use of tokamak technologies already proven. Thus the main technological objective will be the design, development, and utilization of large blanket modules and sectors integrated into a practical tokamak fusion-neutron generator.

- 1.4.2 (II) *Demonstrate the feasibility of a self-sufficient fusion fuel cycle by generating at least 50% of the tritium consumed in D-T reactions, using no more than 50% of the total fusion-neutron production.*

Achieving this objective using significantly less than 50% of the fusion neutron production would also help demonstrate that a fusion reactor could be an important net supplier of tritium.

- 1.4.3 (III) *Serve as an irradiation facility with large test volumes for radiation hardening programs and materials damage studies at relatively low fluence.*

- 1.4.4 (IV) *Demonstrate the production of significant electrical power (≥ 50 MW) on a high-duty-factor basis with a blanket sector characterized by large energy multiplication of incident fusion neutrons.*

Secondary Objectives

The secondary objectives of TORFA are essentially "spin-offs" that will accrue from successful realization of the primary objectives.

1.4.5 To serve as an *operational testbed* for fusion reactor subsystems:

- Quasi-steady heat and particle removal from the torus.
- Steady-state plasma heating and fueling systems.
- Heat removal from blanket modules, and effects of thermal cycling.
- Remote operations for tokamak reactor components and systems.

1.4.6 To serve as a testbed for evaluating the practical merits of various *vacuum vessel concepts*. Achieving this objective requires the capability for rapid change-out of vacuum vessel sectors.

1.4.7 To provide data on *radiation damage rates* for reactor components and materials that are integral parts of TORFA subsystems.

1.4.8 To force the development of manufacturing methods, quality assurance practices, and operational techniques that will ensure *sustained reliable operation* of all tokamak reactor systems.

1.4.9 To serve as a *prototype facility for training* engineers and technicians to operate and maintain commercial-sized tokamak production facilities.

Performance Figures of Merit

1.4.10 For a magnetic confinement fusion device intended for neutron processing applications, the important parameters relating to cost-effectiveness are:

- (I) *Neutron wall loading*, $\phi_w = (0.8 \times \text{fusion power} / \text{first-wall area})$.
This parameter is a measure of the fusion neutron production rate per unit capital cost of the reactor facility.
- (II) *Electrical utilization efficiency*, $Q_e = (\text{fusion power} / \text{plant electrical power input})$, which is a measure of the fusion neutron production per unit of operating cost. A parameter closely related to Q_e is the fusion power amplification, Q_p , defined in Sec. 1.1.8.

1.4.11 Figure 1-3 shows the ranges of neutron flux and fusion power amplification needed for potential fusion applications. Quasi-steady neutron production is required for nearly all of these applications.

TORFA

Figure 1-3. Requirements on neutron flux and fusion power amplification for fusion-neutron applications.

Objectives for Neutron Wall Loading

1.4.12 Relatively few potential applications require a large ϕ_w for technical (as distinct from economic) success. Notable exceptions are facilities for materials bulk radiation damage studies or for actinide waste burning, but neither of these areas is conceived as a vital one for TORFA.

1.4.13 The minimum value of ϕ_w required in TORFA is in fact determined by its principal objective of prototyping a large-scale fusion-neutron production facility. A critical feature of the latter will be the thermal hydraulics of the first wall and blanket systems under conditions of high power loading and severe thermal cycling. For the production of synfuels and process heat, another critical feature is the attainment of extraordinarily high temperature. Thus TORFA must be characterized by a value of ϕ_w at which the *thermal challenges* that must be resolved for successful production machines with wall loading $[\phi_w]_p$ will be encountered. Arbitrarily, we specify that ϕ_w in TORFA shall be at least 1/3 of $[\phi_w]_p$. Conceptual design and cost-effectiveness studies for many applications have indicated that $[\phi_w]_p \gtrsim 0.5 \text{ MW/m}^2$, so that we specify $\phi_w \geq 0.20 \text{ MW/m}^2$.

1.4.14 A higher minimum value of ϕ_w may be set eventually by a programmatic need to produce a minimum amount of nuclear isotopes or chemical fuel per year in TORFA, or a minimum electrical power level in a given blanket sector (Objective IV).

1.4.15 It is inappropriate to attempt to enhance the neutron wall loading in fusion reactors for the principal purpose of accelerated lifetime radiation testing. Radiation damage studies will be carried out as a matter of course, at whatever wall loading, TORFA and subsequent reactors all built for directly useful applications fusion neutrons.

Objectives for Q_p

1.4.15 For a given ϕ_w , a minimum value of Q_p can be determined by the maximum permitted thermal wall loading in the fusion-neutron generator. This criterion is selected, because thermal loading may well be the most serious engineering problem associated with sustained operation of a toroidal plasma (as it is with beam/solid-target systems!).

1.4.16 Figure 1-4 shows the relation of P_{torus}/P_f to Q_p , where P_{torus} is the thermal power that must be removed from the torus, and P_f is the fusion power production. In order not to introduce uniquely extreme demands on TORFA, we stipulate that the thermal wall loading, $\tau_t = 1.25 \tau_w \times [0.2 + Q_p^{-1}]$, shall be no larger than its value in an ignited tokamak reactor (i.e., $Q_p = \infty$) at the maximum practical neutron wall loading in such a reactor, denoted $[\tau_w]_I$. Then

$$Q_p \leq 5 \left(\frac{[\phi_w]_I}{\phi_w} - 1 \right)^{-1} \quad (1.2)$$

Taking $[\tau_w]_I = 3.0 \text{ MW/m}^2$ and $\tau_w = 0.2 \text{ MW/m}^2$ (see Sec. 1.4.13) gives $Q_p \leq 0.35$.

1.4.17 Another factor which tends to raise the minimum required value of Q_p is the maximum practical value of the plasma heating power, P_h . The three quantities (1) capital cost of the plasma heating equipment, (2) electrical power consumption of the plasma heating equipment, and (3) the wall area required for injection of the plasma heating power, are all proportional to P_h . If A_w is the area of the "first wall" (just in front of the blanket modules), then

$$\min. [Q_p] = 1.25 \cdot \tau_w \cdot A_w / \max. [P_h] \quad (1.3)$$

The maximum practical P_h for the TFTR (Tokamak Fusion Test Reactor), which has major radius $R_p = 2.5 \text{ m}$ and plasma vertical elongation $b/a = 1.0$, has been determined to be about 50 MW. Based on these numbers, we set $\max. [P_h] = 50 \text{ MW} \times (R_p/2.5) \times (b/a)$ for a general plasma. Equation (1.3) then results in a minimum value of Q_p which is usually larger than given by Eq. (1.2).

Objectives for Q_e

1.4.18 Having determined the required Q_p , one can find Q_e from the relation

Figure 1-4. For a given fusion power production, the thermal (non-neutron) power to be removed from the torus decreases rapidly with increasing fusion power amplification, Q_p .

$$Q_e = \frac{Q_p}{\frac{1}{\eta_h} + \frac{P_{coil} + P_{bl}}{P_h}} \quad (1.4)$$

where η_h is the ratio of the plasma heating power, P_h , to the electrical power consumed by the plasma heating equipment, and P_{coil} and P_{bl} are the powers consumed by the magnetic coil and blanket coolant systems, respectively. This relation assumes that no power is recovered from the torus, the coils, or the blankets in order to generate electricity.

1.4.19 In order that TORFA be overwhelmingly superior in electrical utilization efficiency to beam/solid-target systems, which are characterized by $Q_{eo} < 0.002$, we specify that $Q_e \geq 10 Q_{eo} \sim 0.02$. If $Q_p \geq 0.35$ and $\eta_h = 0.50$, this condition on Q_e is satisfied provided that $(P_{coil} + P_{bl}) \leq 16 P_h$.

1.4.20 The minimum Q_e that is commercially practical depends on the economic value of the neutron application, and may well have to be much larger than 0.02. Furthermore, a practical lower limit on Q_e even for a pilot plant may be set by the maximum electrical power that can be made available to that plant. *But for non-energy-producing applications, there is no more reason for a fusion neutron facility to have $Q_e \geq 1$ than there is for any industrial processing plant requiring a large electrical power input.*

1.4.21 A tokamak fusion machine of very modest performance will have a neutron production rate per unit capital cost of at least 10^{10} n/s/\$. For example, 10^{19} n/s or 28 MW of fusion power can easily be generated in a tokamak reactor facility of \$1 billion capital cost. This figure of merit is at least three orders of magnitude larger than that of a beam/solid-target system — typically \$1 million for 10^{12} to 10^{13} n/s. Of course, this comparison is meaningful only if both devices have comparable annual capacity factors.

Summary of Objectives for ϕ_w and Q_p

1.4.22 The design parameters of TORFA (see Sec. 1.6) are chosen so that on the basis of presently understood tokamak scaling laws, Q_p as large as 2.0 and ϕ_w as large as 0.4 MW/m^2 will be sustained, provided that impurity influx, plasma spatial profiles, and the deuterium-tritium plasma composition can be controlled. Nevertheless, TORFA will be successful if it achieves reliable high-duty-factor operation with $\phi_w \geq 0.2 \text{ MW/m}^2$ and $Q_p \geq 0.35$ (or the value given by Eq. (1.3), if higher). These latter values are adopted as the *minimum objectives*, and they must be achieved in sustained operation.

1.5 DESIGN PHILOSOPHY FOR TORFA

Maximum Use of Proven Technologies

1.5.1 A practical and reliable tokamak-based fusion energy source can be developed in the near term only by exploiting the advances that have *already* been made in resistive-coil, beam-driven fusion systems — and not by introducing for the first time, in a multi-billion-dollar installation, a host of grandiose and untried technologies each one of which is to operate at its theoretical limit. It is *not* the purpose of the TORFA design to represent a tokamak configuration that is prototypical of what may be the preferred concept for fusion reactor implementation 50 years hence. Instead, the purpose of the present approach is to implement a tokamak reactor facility that is analogous in ambition to the plutonium-producing fission "piles" built at Hanford in the period 1944 to 1950. Those piles were Plain Jane reactors that had a *uniquely useful and necessary* job to do in and by themselves, without any pretense of being a step along the way toward more glorious and sophisticated facilities. TORFA is to be equivalent in simplicity and reliability to those graphite-moderated, water-cooled reactors built at Hanford 3 decades ago.

1.5.2 In fusion reactor design circles there has lately been excessive pre-occupation with compactness of design, which places extraordinary demands on many fusion technologies, and tends to eliminate engineering margins. But for many applications, as in the early Hanford reactors, there is no decisive merit in compactness; the benefits provided by compactness are often outweighed by the greater difficulties of operation, access, and maintenance. *More important objectives of practical fusion-neutron sources are the effective utilization of all the fusion neutrons, ease of accessibility to the test and production chambers, and in general maximum reactor availability.*

1.5.3 Hence the TORFA design is relatively conservative, emphasizing tokamak machine and plasma technologies that will have been demonstrated by the early 1980's, and favoring easy accessibility over compactness and complexity. The principal design features are summarized in Sec. 1.6.

The Role of TORFA

1.5.4 TORFA is not a plasma physics experiment — there will be no substantial risk with regard to TORFA achieving the minimum plasma physics performance needed to meet the objectives set forth in Sec. 1.4.

1.5.5 TORFA is not a testbed for advanced or speculative magnetics systems — only magnetic systems with proven effectiveness on tokamaks will be utilized.

1.5.6 TORFA is a testbed for the development of fusion energy applications, but not merely for blanket and other technological systems of "generic use" to all conceivable types of magnetic confinement reactors. Instead the TORFA fusion machine, blanket systems, and operational procedures will be *prototypical* of those commercial-sized *tokamak* facilities intended for the fusion applications proved out in TORFA. These prototype features, plus the development in TORFA of sustained reliable D-T operation, will eliminate any technological questions concerning the extrapolation of TORFA blanket performance to larger tokamak-based facilities.

Rationale for Neutral-Beam Heating

1.5.7 The parameters of TORFA are chosen so that Q_p up to 2.0 can be attained under favorable conditions, although a significantly smaller Q_p is adequate for meeting TORFA's objectives (see Sec. 1.4). For operation at $Q_p \gtrsim 2$, the conditions on plasma energy confinement and degree of purity are remarkably easier to satisfy when the plasma is heated by reacting ion beams (formed by neutral-beam injection), compared with heating by any other method.

1.5.8 In a D-T plasma heated by a method other than reacting ion beams, the required $n\tau_E$ for $Q_p = 1$ is 1/6 of that required for ignition at the same temperature. When reacting ion beams are used for heating, the required $n\tau_E$ is only 1/30 of that required for ignition.

1.5.9 For a given total plasma pressure, the fusion power density in neutral-beam-driven operation at $Q_p \sim 1$ is several times the power density obtained when any other heating method is used. In fact, this ratio is essentially the ratio of the $n\tau_E$'s required in the two cases.

1.5.10 If a long confinement time against transport losses is feasible, then a much larger impurity-ion concentration can be sustained in the reacting-beam case, for a given Q_p .

1.5.11 For $Q_p \gtrsim 2$, the same neutral beams used for plasma heating, fueling, and production of fusion power can in principle drive essentially all the plasma current, thus permitting steady-state operation. (The beam current drive is still speculative, however, and is not relied upon in the TORFA design.)

TORFA

Figure 1-5. Elevation view of TORFA.

1.6 PRINCIPAL FEATURES OF TORFA

The elevation view in Fig. 1-5 depicts some of the principal features of the TORFA reactor.

1.6.1 Tokamak Machine

- (1) Steady-state, water-cooled resistive TF (toroidal-field) coils constructed of plates with demountable joints allow ready access to the vacuum vessel, internal PF (poloidal-field) coils, and the blanket modules.
- (2) 12 TF coils produce a maximum field at the windings of 7.5 T with 230 MW power dissipation.
- (3) Internal water-cooled OH (ohmic-heating) coils are readily removable via the demountable TF coils. The OH coils produce a flux swing of 50 Webers, with an OH field swing of +4.0 T to -0.5 T.
- (4) A single-null poloidal magnetic divertor removes heat and particles from the torus. All divertor components are readily accessible after removing one or more upper segments of the TF coils.
- (5) The vacuum vessel consists of 12 U-shaped sectors with panels bolted on top. The sector interfaces use mechanical seals and are insulated by ceramic washers. Each sector and panel has independent water cooling.

1.6.2 Plasma Characteristics

- (1) The D-shaped plasma has major radius = 5.05 m, half-width = 1.35 m, and half-height = 2.70 m.
- (2) Maximum plasma current is 5.9 MA with a 500-s flat-top when $\langle T_e \rangle \geq 3$ keV.
- (3) Up to 150 MW of steady-state neutral beams (120 keV D^0 and 160 keV T^0) are injected for heating and fueling the plasma. (Tritium pellet injection may possibly be substituted for tritium beam injection.)
- (4) Spatially averaged $\beta = 0.065$, with $T_i \sim 2T_e$, and total ion energy $\sim 5 T_e$.
- (5) Duty factor is at least 0.85.

1.6.3 Fusion Energy Characteristics

- (1) $Q_p = 0.35$ to 2.0.
- (2) Fusion power = 50 to 300 MW.
- (3) Neutron wall loading = 0.1 to 0.5 MW/m².
- (4) Annual tritium consumption = 1 to 5 kg.

1.6.4 Blanket Test Facilities

- (1) Approximately 250 blanket chambers 0.95 to 1.0-m deep cover at least 80% of the vacuum vessel.
- (2) Each blanket module is independent of the other modules and of all reactor components, and can be readily removed by radial or vertical translation.
- (3) At least 120 blanket modules can be devoted to breeding tritium to replenish at least 50% of the reactor tritium consumption.

1.7 IMPLEMENTATION OF TORFA

Principal R&D Requirements for TORFA1.7.1 Tokamak Machine

- (1) Remotely demountable joints for resistive TF coils constructed of plate sections.
- (2) Remotely demountable high-current joints for resistive PF coils.
- (3) Methods for steady-state particle and energy removal in the burial chamber of a poloidal magnetic divertor.
- (4) Techniques for rapid remote welding on the vacuum vessel.
- (5) Ceramic dielectric breaks for vacuum vessel sector interfaces.

1.7.2 Plasma Systems

- (1) Demonstration of $Q_p \geq 0.5$ in a beam-driven tokamak plasma.
- (2) Demonstration of $\langle \beta \rangle \geq 0.05$ in long-pulse operation.
- (3) Steady-state neutral-beam injectors at 120 keV (D^0) and 160 keV (T^0), and with overall efficiency ≥ 0.4 .
- (4) Protection of first wall from major plasma disruptions.

1.7.3 Blanket Module Capability

- (1) Methods for remote installation and removal of modules inboard of the plasma.
 - (2) Structures and coolant systems compatible with pulsed magnetic fields and cyclic thermal loading.
 - (3) Ex-reactor tests of thermal hydraulics, tritium extraction, and materials compatibility.
- (Note that the development of practical blanket modules is a major task of the TORFA program itself. Only in an actual tokamak subjecting its blanket region simultaneously to realistic profiles of neutron power deposition, cyclic thermal loading, and pulsed magnetic fields can one make an adequate assessment of the dynamic performances of blanket coolant systems, structure, and fuel elements.)

Time Scale for Implementation

1.7.4 Figure 1-6 shows a possible time scale for the implementation of TORFA. The principal R&D needs for the plasma systems will be met by experimental programs on ISX-B, PDX, and TFTR, including upgrades of these facilities. If Preliminary Design of TORFA begins in mid-1981, then construction of the machine could be completed by mid-1987. The first year of operation would be devoted to systems check-out using hydrogen plasmas. Table 1-1 summarizes activities for the first several years of operation.

1.7.5 Certain blanket systems might require an extensive licensing procedure before installation. The construction of TORFA itself would not be delayed by any such licensing process, as each blanket sector would be installed as an add-on at some date after reactor start-up. Regulatory questions involving TORFA itself would involve the usage of tritium, and the disposal of activated reactor components.

1.7.6 Major Objectives II and IV will be realized simultaneously in the 4th year of operation (Phase IV). Tritium breeding will occupy 5 or 6 of the 12 major blanket sectors, while electrical power production (≥ 50 MW(e)) will utilize several sectors (see Sec. 2.6).

1.7.7 In Phase V, radiation damage studies requiring relatively low neutron fluences ($\lesssim 1$ MW-yr/m²) will be performed using 1/2 of a major sector (i.e., 1/24 of the available blanket module positions). Of course, the reactor components themselves will serve as valuable objects for materials damage studies.

DEVELOPMENT OF TORFA

Figure 1-6. Time scale for implementation of TORFA and subsequent reactors.

Table 1-1

PROGRAM PLAN FOR TORFA OPERATIONS

<u>PHASE</u>	<u>YEAR INTO PROGRAM</u>	<u>PRINCIPAL OPERATIONS</u>
I	0 to 1.0	<u>Hydrogen Plasma Check-Out</u> - Magnetic, mechanical, and electrical systems - Plasma heating - Divertor operation - High-beta operation
II	1.0 to 2.0	<u>Deuterium Plasma Check-Out</u> - Long-pulse (500 s) plasma systems - injectors, divertor, first wall coolant - Operation at 75% duty factor - Practice in blanket tritium production and extraction techniques - Practice in remote disassembly and reassembly operations
III	2.0 to 3.0	<u>Deuterium-Tritium Plasma Check-Out</u> [The basic purpose of Phase III is to prepare for the achievement of Objectives II and IV in Phase IV.] - Long-pulse operation with D and T (500 s) - Operation at 75% duty factor - Tritium production and extraction (5 to 10 g) - Blanket heating experiments - Extensive practice in remote disassembly and reassembly - Initiate radiation hardening test program
IV	3.0 to 4.0	<u>Feasibility Demonstrations for Tritium and Electricity Production</u> - Demonstrate Primary Objective II (tritium) - Demonstrate Primary Objective IV (electricity) - Continue radiation hardening test program - Initiate testing of miscellaneous applications modules
V	4.0 to 10	<u>Production Runs</u> - Blanket module production testing (Objective I) - Radiation hardening program (Objective III) - Low-fluence materials testing ($\lesssim 1 \text{ MW-y/m}^2$)

Staged Operation

1.7.8 Because ignition of the D-T plasma is not a requirement for TORFA, a *staged operation* designed to minimize the required initial capital investment is a feasible option. This staging is to be distinguished from the phasing of H, D, and D-T plasmas discussed in the preceding. The first plasma operations would be carried out at relatively low values of magnetic field, plasma current, and injected beam power, and therefore at relatively low plasma density and fusion power output, with consequently smaller demands on tritium throughput, and on first-wall and blanket coolant systems. Table I-2 gives illustrative parameters for a 3-stage build-up in TORFA capability.

1.7.9 Initial investments in power supplies, energy storage systems, neutral beam injectors, and coolant systems, including heat exchangers and cooling towers, would be correspondingly less. Only a small number of blanket modules would be installed initially, the vast majority of blanket locations being occupied by simple inexpensive shielding assemblies.

1.7.10 The penalty to be paid for this operations scenario is reduced neutron wall loading in the early stages. Even at reduced power levels, however, it will be possible to satisfy some of the Principal Objectives set forth in Sec. 1.4, namely Objective I and, to some extent, Objective III. With the installation of a large number of blanket modules, but with reactor operation still at reduced ϕ_w , Objectives II and IV can also be satisfied.

Table 1-2

ILLUSTRATIVE STAGED OPERATION OF TORFA

(D-T Plasmas)

	<u>Stage I</u>	<u>Stage II</u>	<u>Stage III</u>
Plasma Size	Upper half of vacuum vessel only	Full vacuum vessel	Full vacuum vessel
$B_{\max}$ (T) at TF coil	5.3	5.3	7.5
P_{res} (MW)	115	115	230
Injected beam power (MW)	50	100	150
Plasma current (MA)	1.5	4.0	5.9
Pulse length (s)	250	500	500
$\langle \beta \rangle$	0.032	0.064	0.064
Max. Q_p	0.6	1.3	2.0
Max. fusion power (MW)	30	130	300
Max. outboard neutron wall loading (MW/m ²)	0.10	0.23	0.53
Annual tritium consumption	≤ 1 kg	≤ 3 kg	≤ 5 kg
Electrical power requirement during the flat-top, all systems (MW _e)	235	360	600
Percentage of blanket modules installed	25	50	100

Location

1.7.10 TORFA would be located on a large laboratory site managed under contract to the U.S. government. An existing site with a large core of engineers, physicists, and chemists, and licensed for all nuclear-relevant activities, would be preferred. This site must have at least the following characteristics:

- (1) An exclusion area for the TORFA facilities of at least 1 km^2 , located at least 10 km from population centers.
- (2) Access to a power grid for up to 600 MW(e). A smaller power capability would be adequate for the first few years of staged operation (see Sec. 1.7.8).
- (3) Cooling water supply of 50,000 gal/min.
- (4) Adequate access by highway and rail.
- (5) Capability for major remote handling operations.
- (6) Waste management facilities for storage of activated reactor components.
- (7) Low probability of major seismic activity.

Relation to Other Proposed Tokamak Fusion Facilities

1.7.11 The relation of TORFA to the proposed ETF/INTOR class of fusion reactors may be discussed elsewhere, but a few words are appropriate here. Because ETF/INTOR requires enormous advances in all phases of fusion machine technology and plasma performance, the first such machine will be concerned principally with (1) investigation of new regimes of plasma operation, especially the achievement and sustaining of ignited plasmas, and (2) the debugging in the tokamak reactor environment of advanced superconducting magnet systems and other sophisticated technologies untried under "field conditions".

1.7.12 TORFA, on the other hand, will exploit plasma and magnet technologies that have been demonstrated previously in tokamaks, in order to facilitate the achievement of sustained reliable operation, so that TORFA itself can be devoted to the advancement of fusion-neutron applications via exhaustive blanket module development. This difference in objectives is most apparent from the fact that only a small fraction of the ETF first wall (≤ 0.15) would be set aside for blanket module coverage, while TORFA will have a blanket coverage factor of at least 0.80.

1.8 SUBSEQUENT FUSION ENERGY FACILITIES

Performance Upgrading

1.8.1 A magnetic confinement fusion device based on the D-T reaction has the unique characteristic among manmade energy sources that the region of application of nearly all the energy produced is *decoupled* from the region of energy production. Hence TORFA can be a pilot plant for several fusion applications simultaneously, with no effect on the fusion neutron source. This potential is recognized in the formulation of the major objectives (see Sec. 1.4). Operational experience with TORFA will allow the first realistic estimates to be made of the economics of full-scale industrial use of certain fusion applications. Those processes which are estimated to be commercially practical can be pursued in later demonstration plants, which would evolve from TORFA in the following ways:

- (1) Somewhat larger size or magnetic field, and possibly a larger Q_p , thereby giving a higher neutron wall loading.
- (2) Capacity factor exceeding 80%.
- (3) Dedication to 1 or 2 applications.

1.8.2 On the matter of magnetic field, one notes that a factor of 1.5 increase in B_t (at the plasma) requires a factor of 2.25 increase in the TF coil power dissipation and stress, but can result in an increase in ϕ_w of up to a factor of 5, for the same reactor geometry. In fact, the TF coil dimensions could be increased so that the power dissipation would still be 200 to 250 MW for a large increase in ϕ_w , but with the penalty of higher capital cost.

1.8.3 Once a commercially viable application is identified, a demonstration or prototype reactor would be implemented, followed by a production plant. The schedules shown in Fig. 1-6 assume that at least one viable application is determined by 1986. While design of the demonstration reactor would be initiated before operation of TORFA, its construction would be dependent on favorable operational experience and blanket performance in TORFA. The design of the production reactors would not be radically different from the TORFA design, but in fact would evolve from operational experience with TORFA.

TORFA

2. DESIGN SUMMARY

2.1 RESISTIVE TOROIDAL-FIELD COILS

Disadvantages of Superconducting TF Coils

2.1.1 A superconducting coil set with highly stabilized windings (copper/superconducting volumes $\sim 10:1$) has been built for a small tokamak, the T-7 at Kurchatov I.A.E. in the U.S.S.R., and this tokamak has recently come into operation. However, no tokamak has yet been constructed with superconducting coils having a practical copper/superconductor volume ratio, and certainly the array of problems that will be encountered in a tokamak reactor environment have not yet begun to be addressed in the laboratory. Many bubble-chamber and other superconducting solenoids have been built and operated successfully, but these solenoids do not face the formidable conditions that will confront the TF coils in tokamak reactors: strangely asymmetric forces, overturning moments, rapidly pulsed magnetic fields, neutron and gamma radiation, and difficulty of access for repair of the coils.

2.1.2 As discussed in Sec. 1.5, a fundamental reason for not using superconducting coils in TORFA is that the technology of such magnets for tokamak reactor application will take decades to develop, while TORFA will use only technologies that have been proved "in the field" by the mid-1980s at the latest. Aside from the fact that there is zero practical experience with superconducting TF coils in a tokamak environment, there are many important reasons why it would be ludicrous to specify such coils for the very first engineering test reactors.

2.1.3 Nondemountability. Access to internal components such as poloidal field coils, divertor hardware, and the inboard blanket modules will be extraordinarily difficult (or impossible) unless the TF coils are readily demountable. Repair of a nondemountable coil would require the removal of an entire reactor segment. It is probably impractical to implement demountable superconducting coils.

2.1.4 Pulsed Fields. Superconducting coils with a practical copper/superconductor volume ratio may not be able to withstand (without going normal) the large rate of change of vertical fields commonly found in tokamak operation during start-up and shut-down and during plasma disruptions. In

fact, the maximum loop voltage that could be applied to start up the plasma current will be limited by the maximum dB/dt tolerated by superconducting TF coils. No such limitation is imposed by resistive TF coils.

2.1.5 Neutrons. Neutron streaming through injection and pumping ducts introduces special shielding problems for superconducting TF coils.

2.1.6 Construction. Superconducting coils of the size needed for tokamak test reactors must be built on site. They are far too bulky and fragile to be moved from an off-site fabrication plant, where quality assurance can be developed to a more advanced state than possible at a make-shift on-site fabrication facility.

2.1.7 Refrigeration. Refrigeration requirements may exceed 50 kW at 4⁰K, corresponding to 20 MW at room temperature, so that the refrigerator size must be several times the state-of-the-art. (Special concerns are eddy currents induced by pulsed fields, heat leaks to the support structures needed to counteract overturning moments and centering forces, and nuclear radiation.)

2.1.8 Maintenance Time. Maintenance of the reactor components is impossible unless the TF coils are shut off, which for superconducting coils implies a total down-time of at least 2 days even for the most trivial maintenance operation.

2.1.9 Coil Replacement. Insufficiently rapid dumping of magnetic energy following quenching is one of the paths to destruction of a superconducting magnet. The near-impossibility of replacing a damaged superconducting TF coil in any reasonable time period is acknowledged in present conceptual design studies for the ETF and similar ilk, where the designers declare that the TF coils can be made so reliable that they will never be damaged and need never be replaced! Even if one were speaking of a technology that had experienced considerable practical development and marked success in actual operations (which is decidedly not the case), the above claim must be called preposterous.

Aspects of Resistive TF Coils

2.1.1C Water-cooled resistive TF coils fabricated of copper or aluminum (1) can be made readily demountable (e.g., in the ISX device), (2) are unaffected by pulsed fields, (3) require less shielding against neutron irradiation than do superconducting coils, (4) can be factory-built and transported (disassembled) to the reactor site, (5) have simple refrigeration systems, and (6) can be activated or deactivated in less than one minute.

However, resistive coils do have two notable disadvantages:

2.1.11 Power Dissipation. The power consumption of resistive coils will always be considerably greater than that of superconducting coils (about 20 MW for the cryogenic refrigerator). However, if sufficient copper is used in the resistive coils so that they can operate steady state, then the power dissipation can be reduced to the range 150 to 300 MW, for full-scale D-T reactors. This power level will be acceptable for applications other than "pure fusion" electrical power production.

2.1.12 Mass of Material. Several thousand tons of copper will be required for a resistive TF coil set that can operate steady state. This mass represents about 1 day's U.S. production of copper. On the other hand, cryostable superconducting coils require a mass of copper or aluminum of at least several hundred tons, in addition to large masses of niobium and titanium or tin. The prices of the latter metals have of late reached extraordinarily high levels.

2.1.13 A disadvantage of presentday resistive TF-coil sets is that they are pulsed at the same frequency as the pulsing of the plasma discharges, so that the coil lifetime is limited by fatigue. *This fatigue problem will be eliminated in the steady-state TF coil set of a tokamak reactor.*

2.1.14 While a resistive magnet may be destroyed in the event of loss of water coolant, the replacement cost of a TORFA-type resistive magnet (see below) will be many times smaller than that of a superconducting TF coil. Furthermore, the demountable design of the TORFA coils allows for their rapid removal and replacement.

2.1.15 It is clear from the preceding discussion that water-cooled resistive TF coils can be the only reasonable choice for TORFA and indeed for the following facilities mentioned in Sec. 1.8.

Long-Term Prospects for Resistive Coils

2.1.16 It will be argued that in the long term, all toroidal reactors must have superconducting coils. But even if reliable superconducting coils of acceptable cost, and permitting easy reactor maintenance, are developed in 50 years' time, certain other factors will continue to weigh in favor of retaining the simpler resistive TF-coil sets, at least for D-T reactors:

- (1) It seems probable that one will know how to generate toroidal plasmas of higher beta, so that the magnetic field required for a given reactor application will be reduced.

- (2) It is not clear that economically recoverable helium will be available in 50 years. Even if it is, there are many other applications requiring helium with which magnetic fusion reactors will have to compete, such as dirigibles, superconducting transmission lines, levitated trains and as a general-purpose coolant. What helium is available for fusion reactors should be reserved for cooling blankets and for cryopumping.
- (3) The price of niobium, like that of most relatively rare metals, could become exorbitant in 50 years.

2.1.17 The discovery of a practical superconductor that could operate at high current density at temperatures above that of liquid hydrogen might change the outlook mentioned in Sec. 2.1.16, No. (2). But one can equally hope for the development of a resistive alloy that will have twice the conductivity of copper!

2.1.18 It seems likely that superconducting coils will be relegated to use with fusion reactors using "advanced fuels" such as D-D, where the power densities are so low that the coil power dissipation must be minimized. But resistive TF coils could be used even for advanced fuel reactors of exceedingly large size and operating at very high beta.

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Design Considerations for TORFA TF Coils

2.1.19 Figures 2-1 and 2-2 depict the TF coils of TORFA, and the principal parameters are summarized in Table 2-1. The TF coil set comprises 12 coils of rectangular shape. (It is possible that a D-shape would be chosen ultimately, however.) Each coil consists of 20 copper plates with milled grooves for the water coolant. The plates are insulated by G-10 or ceramic sheets and the entire assembly bolted together. A very small number of plates is desirable, in order to minimize potential problems with the joints, but the need to limit the current carried by the external leads dictates a minimum of 20 plates per coil. All plates and coils are series connected.

2.1.20 The inboard coil legs are nested up to $R = 1.8$ m. The upper and lower segments are nested up to 2.4 m (coil width = 1.40 m). The bursting forces at the inboard corners are counteracted by hydraulically-set clamps.

2.1.21 The width of the plasma chamber is 3.2 m, or one-half the horizontal bore of the TF coils. This geometry represents an effective division of the magnetic real estate between the fusion-neutron generator, and the region of fusion-neutron applications.

2.1.22 The TF coil power dissipation is 230 MW at maximum field. Another 5 MW is consumed by the water coolant system. The heated water (at about 60°C) can be used to provide site heating needs.

2.1.23 The joints in the TF coil legs allow disassembly for maintenance and replacement of interior components (see Sec. 2.2, 2.3, and 2.6). These joints also enable easy replacement of a coil segment in case of damage, without having to disassemble any of the PF coils, the vacuum vessel, or any other internal component. The upper segments are lifted straight up and held by machinery integrated into the roof of the Reactor Cell. The outboard segments are moved radially outward on rails.

2.1.24 The most probable source of a fault in a TF coil, requiring its repair, will be failure of a joint, perhaps caused by bolt fatigue, coolant leakage, or a large increase in contact resistance. Hence the greatest attention must be given to the concept and design of reliable joints that will still permit easy disassembly. This goal is most difficult for the inboard segments which nest together. The bolted joints depicted in Fig. 2-1 are merely illustrative, and other concepts, such as hydraulically clamped joints, are being developed.

TORFA

Figure 2-1. Elevation view of TORFA.

TORFA

Figure 2-2. Plan view of TORFA.

Table 2-1

TORFA TOROIDAL-FIELD COILSA. Geometry

No. of Coils	12
Horizontal bore	6.4 m
Vertical bore	9.7 m
Major radius of coil	5.0 m
Radius at maximum field	1.80 m
Radial thickness, outboard leg	1.40 m
Azimuthal thickness, outboard leg	1.25 m
Overall diameter	19.2 m
Overall height	12.5 m
Minimum access gap between coils	3.4 m
Weight per coil	325 met. tons
Ripple at $R = R_p$	0.002 p-t-p
Ripple at outboard plasma edge	0.036 p-t-p

B. Electrical

Field at coil, B_m	7.5 T
Field at plasma, B_t	2.67 T
Ampere-turns per coil	5.62×10^6
Turns per coil (no. of plates)	20
Coil current	281 kA
Stored energy of coil set	9800 MJ
Copper conductor	CDA 102 Alloy
Conductivity	101% IACS
Resistivity at 50°C	$1.9 \times 10^{-8} \Omega m$
Max. current density (inboard leg)	0.99 kA/cm^2
Packing factor - outboard leg	0.86
- inboard leg	0.73
Resistive power dissipation	230 MW

Table 2-1 (continued)

C. Coolant System

Coolant	Pressurized water
Plates per coil	20
Cooling channels per plate	10
Turns per channel	1
Coolant velocity	2.0 m/s
Channel pressure loss	20 psi
Total volume flow (12 coils)	25,000 gal/min.
Coolant inlet temp.	20°C
Coolant temp. rise	33°C
Max. ΔT , conductor to coolant	8°C
Max. ΔT , to local hot spots	18°C

D. Magnetic Stresses

Maximum bending stress	16,000 psi
Tensile stress, inboard leg	8,300 psi
Tensile stress, horizontal leg	3,600 psi
Bearing stress at inboard region	3,500 psi

2.1.25 Of the 3 experimental tokamaks with jointed TF coils presently in operation (ISX, Doublet III, and PDX), only ISX has rectangular coils with joints that permit rapid disassembly and reassembly. However, there is no nesting of the inboard legs in ISX, so that the design of the inboard joints is facilitated. There has been no failure of the coil joints in any of these machines.

2.1.26 The FM-1 "spherator" at PPPL was equipped with rectangular TF coils featuring readily demountable joints. Disassembly and reassembly of the entire TF coil set could be completed in several days, with hands-on operation.

Toroidal-Field Ripple

2.1.27 Figure 2-3 shows the TF ripple on the midplane of TORFA as a function of major radius. The peak-to-peak ripple is 3.6% at the plasma edge and 0.2% at the magnetic axis ($\beta = 0$ position). (Because of the rectangular shape of the TORFA TF coils, the ripple at the plasma is much smaller than for circular and even for D-shaped coils.) While this ripple is somewhat larger than would be acceptable for obtaining an ignited plasma, its magnitude is certainly satisfactory for TORFA whose objective is $Q_p \lesssim 2$, or $n\tau_{Ei} \lesssim 3 \times 10^{13} \text{ cm}^{-3}\text{s}$. The neutral-beam injectors must be oriented for injection at an angle of at least 12° from the normal to the plasma surface.

2.1.28 The small amount of ripple at the TORFA plasma is actually beneficial for expelling fusion alpha particles as they slow down below 500 keV and pitch-angle scatter. The ripple will also enhance the hypothetical beam-driven plasma current by increasing the fraction of electron trapping.

Basic Parameters of TORFA

2.1.29 Table 2-2 gives the important plasma and reactor parameters of TORFA (other than the TF-coil parameters). Assuming a beam injector efficiency of 0.50, the electrical power requirements of the injectors will exceed that of the TF coils. The installed beam power is unlikely to be increased much beyond the 50 MW postulated in Stage I (see Table 1-2) until an effective development program raises the overall injector efficiency considerably above its presentday value of about 0.25 for 120-keV beams.

Figure 2-3. Toroidal-field ripple in the midplane of TORFA.

A. Geometry

Major radius, R_p	5.05 m
Plasma half-width, a_p	1.35 m
Plasma half-height, b_p	2.70 m
Aspect ratio, R_p/a_p	3.74
Vacuum vessel horiz. bore	3.20 m
Vacuum vessel vert. bore	7.40 m
Height of divertor chamber	1.50 m
Width of scrape-off layer	0.25 m
Blanket thickness, inboard and top	0.95 m
bottom	1.00 m
outboard	1.00 m
Outboard shield thickness	0.40 m

q at plasma edge	3.0
Plasma current	5.9 MA
Plasma inductance	6.3 μ H
Inductive flux for start-up	40 Webers
Radius of OH throat	2.1 m
Field swing in core for start-up	3.0 T
Loop voltage at $\langle T_e \rangle = 8$ keV	0.040 V
Pulse length	500 s
Field swing in core to drive flat top	1.5 T
Range of OH field	+4.0 T to -0.5 T

Injected neutral-beam power	$\lesssim 150$ MW
Spatially averaged β	0.065
β_p	2.2
Effective area of first wall	575 m^2
Q_p	$\lesssim 2.0$
Fusion power at $Q_p = 2.0$	300 MW
Neutron wall loading at $Q_p = 2.0$ (spatially averaged)	0.42 MW/m^2
Maximum outboard neutron wall loading	0.53 MW/m^2
Duty factor	0.85

2.2 POLOIDAL-FIELD (PF) COILS

Resistive PF Coils

2.2.1 As in the case of superconducting TF coils, the viability of a superconducting OH (ohmic-heating) solenoid or of superconducting EF (equilibrium-field) coils has yet to be demonstrated on a tokamak device, and there is no prospect for such tests for at least a decade. Hence the OH solenoid, the EF coils and the divertor coils of TORFA will all use copper conductors with internal water cooling passages.

Location of PF Coils

2.2.2 Figure 2-1 shows the locations of the PF coils. The OH flux must be well-isolated from the plasma, because of the appreciable change in this flux during the very long flat-top (~ 500 s). The OH solenoid and all the EF and divertor coils are located just outside the blanket modules. This location allows adequate shielding for the coil insulation, and at the same time the coils are sufficiently close to the plasma so that effective plasma control is maintained with minimum PF-coil current, and with minimum overturning moments on the TF coils. The internal location of the PF coils is practical only because of the readily demountable TF coils. For example, once the upper legs of the TF coils have been removed, the OH solenoid can be withdrawn and replaced by straight-up lifting, without any disassembly of the solenoid, and without prior removal of the inboard blanket modules. The divertor coils can also be removed by straight-up lifting, without any disassembly.

2.2.3 The throat of the OH solenoid has sufficient area so that a flat-top of 500 s is easily attained at moderate OH fields, even if the loop voltage is significantly higher than the 0.04 V calculated at $\langle T_e \rangle = 8$ keV.

2.2.4 The outboard EF coils are constructed in two 180-degree segments, which are united by means of "plug-in" joints. The two segments are removed by pulling them radially outward, after removal of the appropriate shield blocks, and the inverse procedure is used for installation. To permit removal of an outboard blanket module, one or more of the outboard EF coils is translated vertically without disassembly.

2.2.5 The lower EF coils (below the vacuum vessel) are also constructed in 180-degree segments. Their removal requires the prior removal of the lower TF coils legs. To reduce the frequency of this task, redundant EF coils are specified for this region.

TORFA

2.3 CHOICE OF MAGNETIC DIVERTOR

Poloidal Versus Bundle Divertors

2.3.1 The inherent advantages of a poloidal divertor, compared with a bundle divertor, are that it preserves the axisymmetric magnetic configuration of the tokamak, mechanical stresses on the divertor coils are relatively low (if the coils are located internally), and a large plasma throughput is easily accommodated. At high plasma beta, a poloidal-field separatrix tends to form naturally, thus alleviating performance demands on the divertor coils. The principal disadvantage of a poloidal divertor in a tokamak with nondemountable TF coils is the difficulty of access to the divertor coils and energy collectors for replacement or repair. In TORFA, this disadvantage is eliminated.

2.3.2 The inherent advantages of a bundle divertor, which is situated outboard of the vacuum vessel between two TF coils, are its ease of incorporation into the tokamak configuration, and the ease of access to the divertor coils and energy collectors. The principal disadvantages are the serious degradation of the tokamak magnetic topology (which is known to cause deterioration in plasma particle and energy confinement), severe mechanical stresses, and high power dissipation in the case of resistive divertor coils. The latter two problems are aggravated by the need to provide space for adequate shielding of the divertor coils. The advantage of easy access, compared with the poloidal divertor case, is less pronounced in those designs where the outboard TF coils legs are positioned several meters from the vacuum vessel.

2.3.3 Favorable performance of the poloidal divertor, for particle and energy exhaust as well as for impurity control, has been demonstrated in two small tokamaks, DIVA and T-12, and most impressively in one large tokamak, PDX.

A bundle divertor has been implemented to date only on one small tokamak, DITE, and with unsatisfactory results. By 1985, poloidal divertors will have been tested and optimized in 3 large tokamaks with high-temperature plasmas — PDX, ASDEX, and JT-60. Relatively minor testing will have been carried out with bundle divertors by 1985. These considerations alone are sufficient to rule out the bundle divertor for TORFA, which will incorporate only proven technologies in accordance with the discussion in Sec. 1.5.

Impracticality of External Divertor Coils

2.3.4 There have appeared proposals to place poloidal divertor coils external to the TF coils, for application specifically to tokamaks with nondemountable TF coils. Such proposals seem impractical for a number of reasons: (1) Inadequate control of the positions of the separatrix and diverted field lines, and even of the plasma vertical position; (2) enormous inductive power requirements, because the stored poloidal-field energy would be comparable with that of the toroidal field; (3) much larger overturning moments on the TF coils, necessitating greater structural support and increasing cyclic stresses; (4) poor time response.

2.3.5 [In the case of superconducting TF coils, still more decisive disadvantages of external divertor coils are the large pulsed fields from which the TF coils cannot be shielded, and the increased thermal leaks caused by the massive structural support needed to counteract the greater overturning moments.]

2.3.6 The readily demountable TF coils of TORFA eliminate the usual disadvantages of interlocking divertor and TF coils and of difficult access to the hardware in the divertor chamber. External divertor (or EF) coils need not be considered.

Particle and Energy Removal

2.3.7 The most attractive scheme for disposing of the energy of the diverted plasma in TORFA is to maintain a relatively high pressure in the divertor chamber, so that much of the inflowing energy is removed by charge exchange and radiation, and the remaining energy is distributed over a large-area target. Water-cooled collectors, such as the swirl tubes used in neutral-beam line calorimeters, can serve as neutralizing plates. Particles would not be collected in the divertor chamber, but would be pumped away by high-capacity vacuum pumps located between all the TF coils. If the plasma is fueled entirely by neutral beams, the particle throughput is $1 \times 10^{22} \text{ s}^{-1}$ or less.

2.4 PULSE LENGTH ASPECTS

2.4.1 The principal motivation for increasing the pulse length of the burn in TORFA and follow-on reactors is to minimize the thermal cycling of the first wall and blanket modules as well as the cyclic stresses on many reactor components, and thus minimize the likelihood of thermal or mechanical fatigue.

Limitations to Pulse Length

2.4.2 The factors that potentially may limit the length of the burn in each cycle are (1) difficulties in heat removal from the torus, (2) build-up of impurities, (3) build-up of fusion ash, (4) inadequate fueling, (5) loss of thermal stability, and (6) exhaustion of the OH current drive.

2.4.3 Limitations (1), (2) and (3) can in principle be eliminated by the poloidal divertor. Satisfactory fueling can be realized by using both T^0 and D^0 beam injection, if pellet or gas injection is inadequate. Maintenance of thermal stability is not a problem in a beam-driven plasma with $Q_p \leq 2$. Hence, there is considerable incentive to remove the limitation imposed by the OH current drive capacity. The reference TORFA design has an OH system which, together with the EF system, establishes a plasma current up to 5.9 MA and maintains it for a flat-top of 500 s.

Neutral-Beam-Driven Current

2.4.4 If the results of tokamak experiments indicate that the TORFA neutral beams, together with the EF system, will be able to establish the prescribed plasma current, I_p , and that the beam-driven current itself could be as large as I_p , then the OH solenoid in TORFA would be eliminated, R_p would be reduced to 4.55 m, and B_m would be reduced to 7.0 T. The pulse length could be made arbitrarily long. However, it seems unlikely that such experiments will be done in time to impact the Final Design of TORFA, if the design proceeds according to the schedule of Fig. 1-6.

2.4.5 If during the course of TORFA operation, it is demonstrated that the injected beams can in fact drive a significant fraction of I_p , then the pulse length of the burn can be made significantly longer than 500 s. This possibility is taken into account in the prescribed orientation of the TORFA neutral-beam injectors.

2.4.6 The D^0 beams are injected quasi-tangentially (35 to 40° from the normal to the plasma surface), to facilitate establishment of a beam-driven current if such a current is indeed feasible. The D^0 beams will also be used for plasma heating and fueling, as well as for fusion production by beam-target reactions.

2.4.7 If T^0 beams are required for fueling, they will be injected about 15° off-normal, also in the co-direction. The T^0 beams will contribute insignificantly to plasma current drive (or to plasma rotation), but will enable fusion production by energetic-ion ("beam-beam") reactions with the fast deuterons. Near-perpendicular injection will allow the T^0 beams to have the same penetration effectiveness as the D^0 beams, when $E_b \approx 120$ keV for D^0 and 160 keV for T^0 .

2.4.8 Each neutral-beam injector will deliver up to 20 MW, and be capable of steady-state operation. In Stage I (see Table 1-2), the beam injectors will be located above the horizontal midplane. In later stages, injectors will be positioned symmetrically above and below the midplane.

2.4.9 Stress limitations may allow TF-coil operation at a B_m somewhat higher than the nominal maximum value of 7.5 T, thereby allowing a higher plasma pressure at a correspondingly higher I_p . For the same beam power, but using a larger plasma density, Q_p and ϕ_w are likely to increase, but the pulse length of the plasma current will be reduced.

2.5 VACUUM VESSEL

Design Principles

2.5.1 The design of the TORFA vacuum vessel, like those of the TF and PF coils, is predicated on the assumption that each component may require maintenance or removal at some time or another. (This assumption is to be contrasted with that used in certain other tokamak reactor designs, namely that a particular component or subsystem can be made so reliable that nothing will ever go wrong.)

2.5.2 The following guidelines are applied to the choice of vacuum vessel concept:

- (1) A leak in the primary vacuum vessel wall must not necessitate reactor shutdown.
- (2) A coolant leak in a blanket sector must not necessitate reactor shutdown.
- (3) The vacuum vessel must not have a markedly unproven construction, even if a conventional type of construction results in greater neutron energy degradation.
- (4) Rapid remote maintenance or removal of divertor hardware must be feasible.
- (5) Repairs to the vacuum vessel must be feasible to execute, by remote means, in an acceptable length of time.

Location

2.5.3 The main vacuum vessel is placed inside the blanket sectors, in order to avoid the disastrous consequences for reactor availability of a coolant leak in a blanket sector. Blanket modules can also be replaced without breaking the main vacuum, an essential requirement for a test reactor as well as for a production reactor requiring batch processing of its modules.

2.5.4 Satisfying the guidelines of Sections 2.5.1 to 2.5.3 results in a vacuum vessel and blanket module system that has a more detrimental effect on fusion-neutron transparency than do certain other design concepts. However, a conservative vacuum vessel design is essential to insure reliability until extensive operational experience, in TORFA, lends confidence to the eventual substitution of a more daring vessel concept.

Sectoring

2.5.4 Several electrical breaks are required in the stainless steel or aluminum vacuum vessel to permit high-voltage start-up of the plasma current, and rapid penetration of applied poloidal magnetic fields. The TORFA vacuum vessel comprises 12 sectors bolted together, with the sectors insulated from each other by ceramic "picture-frame" washers, as indicated in Fig. 2-4.

2.5.5 The main part of each sector is a U-shape rolled from a single sheet, with a nearly flat plate bolted on top and sealed with metal O-rings. Each sector has no curvature in the toroidal direction; adjacent sectors meet at 150° .

2.5.6 The ceramic washer will be effective even after a year's neutron irradiation ($\sim 10^{12}$ rads), especially if it remains under compression. The washers can be replaced annually, if necessary, or whenever a sector joint is unbolted.

2.5.7 The ceramic washer is not cooled directly, so that it must be positioned relatively distant from the edge of the plasma.

2.5.8 Small leaks in the O-ring seals (Fig. 2-4) can be tolerated because there is vacuum pumping around the interface, in the space between adjacent structural compartments for the blanket modules.

2.5.9 The proposed method of sectoring lends itself to periodic substitution of vessel sectors for the purpose of evaluating different vacuum vessel concepts, such as one that may have a milder effect on the neutron energy spectrum.

Upper Plates

2.5.10 The top plates of the vessel sectors suspend the shielding for the central divertor coil. This shielding in turn suspends the divertor heat collection assemblies (see Fig. 2-1). The plates are removed by simple lift-off after unbolting, thus allowing easy access for maintenance of the divertor hardware. If necessary, the upper plates can be indented to allow the divertor coil to be positioned closer to the separatrix.

2.5.11 The upper plates provide access to the divertor chamber for the high-capacity turbopumps that exhaust the divertor particle influx.

Figure 2-4. Schematic drawing of interface between vacuum-vessel sectors of TORFA. (Not to scale.)

Vessel Coolant System

2.5.12 The thermal (non-neutron) load on the first wall will be of the order of 10 W/cm^2 . A candidate coolant system consists of water-carrying tubes brazed to the vacuum vessel sectors. A leak in these tubes will have no effect on the torus vacuum. The coolant channels will also act as stiffeners. Water is probably the most effective means of removing heat from the first wall, although a penalty is paid in neutron energy degradation in passing through the coolant channels.

2.5.13 A second approach to the coolant system, illustrated in Fig. 2-4, is to machine the first wall and coolant channels out of thick steel or aluminum plate, in a manner analogous to the fabrication of the plates of the TF coils. In this case no external stiffeners are needed. Leaks from the coolant passages are likely to affect only the secondary vacuum region.

2.5.14 During regular operation, TORFA will experience about 100 burn cycles per day, or on the order of 15,000 cycles per year. The deleterious effects of thermal cycling can be reduced in a beam-driven reactor by controlling the ramp-up and ramp-down times of the thermal wall loading to make the temperature gradients across the wall as linear as possible. This control can be achieved by a slow ramp-up and ramp-down of the injected beam power, at least 30 s for startup and 30 s for shutdown.

2.5.15 Sacrificial armor plating must be installed to protect the first wall from neutral beam impingement or from dumping of the plasma energy during a major disruption. This plating will be cooled by radiation and by thermal conduction to the main vacuum vessel by a high-conductance path, such as a copper bar with low-Z cladding.

2.6 BLANKET SECTORS

Need for Inboard Blanket Coverage

2.6.1 In order that a production reactor of any type be self-sufficient in tritium, useful blanket coverage will have to be at least 80%. Even when tritium self-sufficiency is not required, economic pressures will demand that practically every fusion neutron be captured for some useful purpose. Hence, blanket modules must be emplaced inboard of the plasma in order to utilize at least a large fraction of the neutrons impinging on this region.

2.6.2 TORFA is to prototype all the important features of commercial production reactors (see Sections 1.5.6 and 1.8). Hence it is a vital element of the TORFA design that provision be made for full-sized blanket segments inboard of the plasma. *Operations with TORFA must demonstrate the practicality of rapid access to the inboard blanket regions for removal of bred tritium or other blanket production and for removal of the blanket modules themselves.*

2.6.3 TORFA could be made slightly smaller in major dimensions by forgoing inboard blanket segments for simple shielding assemblies. Such a design alternative is out of the question, however, for the reasons discussed above. In fact the design of the TF coils, and particularly of the inboard joints, is influenced somewhat by the need for a demonstration of the rapid removal of inboard blanket modules. This same design approach will be adopted in commercial production facilities.

2.6.4 A blanket sector is defined here as the blanket region extending between the center planes of adjacent TF coils, including outboard, inboard, and bottom compartments. Each blanket sector corresponds to one vacuum vessel sector. The total surface area of a sector facing the fusion neutron flux is approximately 40 m^2 .

2.6.5 Blanket modules are also placed in compartments above the divertor chamber. These modules are irradiated by neutrons whose spectrum is degraded seriously by passage through the divertor hardware.

Rapid Replacement of Blanket Modules

2.6.6 As indicated in Figs. 2-1 and 2-2, each blanket module has dimensions of approximately 1.5 m in width, 1.0 to 2.0 m in height, and 1.0 m in depth, with a weight between 5 and 50 metric tons, depending on the composition. There are approximately 30 modules in a sector.

2.6.7 The blanket sectors are designed for rapid change-out of modules, even on the inboard side, without disturbance to the plasma vacuum system. Each of the outboard modules is removed by simple translational motion in the radial direction, without displacing the outboard legs of the TF coils. To replace an inboard blanket module, two of the upper TF coil legs must first be detached and pulled away. The inboard modules are then pulled straight out vertically by machinery integrated into the roof of the Reactor Cell.

2.6.8 Clearances are allowed for module loading, coolant headers, and thermal expansion; 15 cm is available for the inboard modules, and whatever is needed for the outboard modules is taken from the outboard shielding (40 cm thick). Contact maintenance is possible at the outside of this shielding.

Secondary Vacuum Region

2.6.9 Design philosophy is discussed in Sec. 2.5.2. Each blanket module is bolted into a separate vacuum-sealed compartment, as indicated in Fig. 2-5. A leak in the plasma vacuum vessel does not necessitate shut-down of the reactor. Furthermore, a leak in a blanket module will not lead to any dangerous escape of material into the primary vacuum region or into the Reactor Cell. By using at least one independent pumping system for each blanket sector, this scheme also aids the tracking down of a leak in any blanket module.

Small Test Volumes

2.6.10 Relatively small irradiation test volumes could be placed in the shielding region between adjacent blanket module compartments (see Fig. 2-2), but withdrawal of an outboard TF coil leg would generally be necessary to gain access to one of these test volumes.

2.6.11 To help achieve Objective II (see Sec. 1.4.2), 20-cm-thick blankets of solid lithium breeder could be placed around beam ducts and other penetrations, for a distance extending about 1 m outward from the vacuum vessel.

OUTBOARD BLANKET MODULE COMPARTMENT PLAN VIEW

Figure 2-5. Schematic view of outboard blanket module compartment in TORFA. (Not to scale.)

APPENDIX A

LIST OF SYMBOLS

a_p	Plasma minor radius
A_w	Area of the "first wall"
b/a	Vertical elongation of the plasma
B_t	Toroidal magnetic field at the plasma magnetic axis
B_{max}	Toroidal magnetic field at the TF coil windings (maximum value)
I_p	Plasma current
n	Plasma electron density
P_f	Fusion power production
P_h	Injected plasma heating power
P_{res}	Resistive power dissipation of the TF coils
P_{torus}	Thermal (non-neutron) power to be removed from the torus
Q_e	Fusion power/plant electrical power input
Q_p	Fusion power/injected plasma heating power
R	Radial distance from major axis of the reactor
R_p	Plasma major radius
T_e	Electron temperature
T_i	Ion temperature
β	Average plasma pressure/magnetic field pressure
ϕ_w	Neutron wall loading
τ_E	Energy confinement time of the plasma

APPENDIX B

ADAPTATION OF THE TORFA CONCEPT TO AN INEXPENSIVE D-T-BURNING TEST REACTOR

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ABSTRACT

Less expensive versions of the basic TORFA concept can be realized by simplifying the region between the inboard TF coil segments and the inboard edge of the plasma, with a consequent reduction in plasma major radius and in magnetic field strength at the TF coils. These versions have the same plasma performance as the reference TORFA concept and retain its primary objectives. Higher Q versions of the standard TORFA can be realized, with modest cost savings, by keeping the same magnetic field, but with simplification of the inboard region and with optional reduction in plasma major radius.

TM-341

August 1980

Presented at the ATP Workshop in San Diego, California, 18-20 August 1980.

1. INTRODUCTION

In the spring of 1980 the present author issued a report describing a concept for a D-T-burning tokamak reactor that could be built using early-1980s technologies, and, equally important, had objectives that made sense. The TORFA concept, described in Ref. 1, is the only type of D-T reactor that can be built in the foreseeable future, at any cost, and that also can be expected to operate. Aside from the likes of TFTR and JET that will experience only short occasional pulses of D-T operation, the TORFA concept is probably the only type of D-T tokamak reactor that can be built at all.

A reactor along the lines of the TORFA concept as described in Ref. 1 could possibly be implemented for less than \$1 billion (1980), provided that fabrication and operation were taken no farther than Stage I (see Table 1-2 of Ref. 1). Nevertheless, there is certainly incentive to examine less expensive versions of the basic TORFA concept, while retaining most of the objectives given in Ref. 1 for the "standard" TORFA facility. Indeed, the TORFA concept lends itself, by suitable modification of dimensions, fields, and secondary design features, to a large variety of technological purposes that require the use of tokamak fusion reactors. As the inherent flexibility of the basic TORFA concept will not occur to most readers upon their perusal of Ref. 1, various reports will be devoted to these different versions of the TORFA concept.

The present report is concerned with only one direction of modification — toward inexpensive D-T reactors that easily satisfy the criteria of the recently issued ERAB report² for the FED (Fusion Energy Device). The author emphasizes, nevertheless, that he has no use for the stated objectives of the FED as a definition or justification of that facility, because those objectives can be met by any D-T reactor capable of achieving a fusion energy multiplication, Q_p , exceeding unity in short pulses. (The ERAB panel discourages not only high reactor availability, a realistic gesture, but also high duty factor even when the reactor is available for operation!) The prescribed FED goals are (1) to provide a burning plasma, defined as one with $Q_p > 1$; (2) to provide a focus for developing reactor technologies; (3) to explore and delineate aspects of reactor safety (a concession to the post-TMI world).

Even the TFTR can marginally satisfy the objectives for the FED! Such bland and essentially meaningless goals for the next major step beyond TFTR will fail to unlock the billions of dollars needed for a stepped-up fusion development program. The reader is referred to Ref. 1 for a meaningful set of objectives for the next D-T reactor.

2. APPROACH TO COST REDUCTION

In the reference TORFA concept shown in Fig. 1, full-scale inboard blanket modules are included in order to satisfy a basic TORFA objective of prototyping a commercial production facility. A conventional current-driving solenoid is also included in the inboard region, because of the present uncertainty concerning the effectiveness of the beam-driven plasma current. (In TORFA, fast ions formed from neutral beam injection provide most of the plasma heating and fueling, and also are directly responsible for a significant fraction — sometimes the majority — of the fusion power production.)

The two obvious methods for reducing reactor cost involve a reduction in the thickness Δ_1 between the inboard TF coil segments and the inboard edge of the plasma:

- (1) Replacement of the inboard blanket modules by water-cooled shielding of minimal thickness, which is determined by the need to provide sufficient attenuation of neutron flux to protect the TF coil insulation, and also to remove the neutron energy deposition. A total thickness of 45 cm is adequate, resulting in a reduction in Δ_1 of 50 cm. [The resistive TF coil construction prescribed for TORFA lends itself to the use of ceramic plate insulation, (Ref. 4) whose tolerance to neutron irradiation is uncommonly high.]
- (2) If the plasma current can in fact be driven entirely by injected neutral beams, then the current-driving solenoid can be eliminated, leading to a further reduction in Δ_1 of 45 cm.

Adoption of either or both of the above steps results in a significant reduction in R_p , the plasma major radius. In addition to this reduction, to retain the same plasma performance as in the reference TORFA,¹ B_m can be reduced from its standard value of 7.5 T, or R_p can be reduced still further

while keeping $B_m = 7.5$ T. Table 1 shows machine parameters for a number of illustrative examples. Obviously, other cases with intermediate values of B_m , R_p and a_p are possible.

3. ILLUSTRATIVE MODIFICATIONS OF THE REFERENCE TORFA CONCEPT

Models B through G in Table 1 are less expensive versions of the reference TORFA concept (model A) that provide at least the same plasma performance — n_{TE} , $Q_p = 0.5 - 2$, and neutron wall loading $= 0.1 - 0.5$ MW/m². Elevation views of these models are compared in Fig. 2. Because the TF coils are identical for all the models (except G) in Table 1, all these configurations are accessible by changes in the vacuum vessel dimensions and by the inclusion or removal of the current-driving solenoid. In model G, which has the smallest plasma major radius, the TF coils are also reduced significantly in size, in order to suggest the maximum cost savings achievable; the penalty is loss of flexibility for upgrading performance.

The TF coil power losses and the bending stresses in the coils are relatively small in B and exceptionally small in E. These two models are capable of higher performance (i.e., models C and F, respectively), if the TF coils are at some later stage brought up to the same operating conditions as in model A ($B_m = 7.5$ T). This performance will demand an upgrading of the TF and EF power conversion systems (the coils are powered directly off the grid), all the water coolant systems, and the neutral-beam injector power.

The total cost of tokamak reactor facilities of the same B_m and comparable a_p , including TF coils, power supplies, beam injectors, blanket modules, and tritium facilities, varies between R_p and R_p^2 . Hence model D, for example, will cost between 75 and 85% of model A, while model G will cost between 55 and 75% of model A. The cost of model B will be comparable with that of D, because the higher R_p of the former is compensated by its smaller B_m and injector capacity. Similarly, the cost of model E will be comparable with that of G, although it may be necessary to contract the outboard TF coil legs of E (as in G) to realize this cost benefit. At any rate, TORFA models can be found with capital cost between one-half and three-quarters of that of the reference model A, or well under \$1 billion (1980). All these models will have the same plasma performance.

For each of models B through F, it is evident that the outboard segments of the TF coils can be moved radially inward (as in G), which would provide some savings in capital and operating costs for the TF coils. However, if the outboard leg positions of Fig. 2 are retained, then one always has the option to implement the rather large plasma of model H which, given favorable transport scaling laws, is capable of attaining ignition by the SMARTOR technique (see Ref. 3). In determining fields and sizes for models B through G, it was assumed that $n_{TE} \propto n_a^2$. To obtain ignition in model H, it is necessary to invoke more favorable scaling, such as $n_{TE} \propto n_a^2$ or nTa_p^2 .

4. EVOLUTIONARY APPROACH TO IMPLEMENTATION

(1) At the time of initiation of engineering design of the first TORFA device, the effectiveness of the beam-driven current may well be unclear, so that a current-driving solenoid must be included. In that case the minimum cost system could be a modified model D, with the outboard legs of the TF coils moved radially inward by 0.7 to 0.9 m from the model A case.

(2) If following machine commissioning, the beam-driven current proves to be fully effective, one could remove the current-driving transformer, and increase the size of the vacuum vessel radius and the plasma minor radius by 25 cm, thus evolving to model F. The result would be an increase in plasma Q_p and some increase in neutron wall loading, and these benefits would be obtained at the cost of a new vacuum vessel. With the larger plasma, one also has the option to reduce both B_m and injected neutral-beam power in order to reduce operating costs, for the same Q_p as previously (i.e., model E).

(3) If the machine as commissioned is model D without modification, then upon removal of the current-driving solenoid one has the option to implement model H and attempt to achieve ignition at the cost of a much larger vacuum vessel.

(4) The TORFA TF coils consist of 4 segments, each consisting of 20 plates bolted or clamped together (Ref. 4), and this fabrication technique allows upgrading of individual segments whenever desired. Hence even if the upper and lower segments are initially made as short as possible in order to save both capital and operating costs, at a later time one can replace these segments with longer segments, thus enabling the implementation of a larger vacuum vessel, such as that in model H.

5. TF COIL OPERATING COSTS

The first TORFA would be constructed at a remote site with good access to a strong electric power grid (see Ref. 1), and where the cost of electricity at off-peak hours (2200 to 0700) can be expected to be as low as 25 mills/kWh. If the reactor were operated for 10 hours per day for 300 days per year at full TF coil power (i.e., capacity factor ≈ 0.34), the annual cost of electricity for the TF coils would be \$17 million for the 230 MW cases, and only \$9 million for model E. These numbers should be compared with an estimated total annual operating cost of the order of \$100 million for the entire facility. Furthermore, even 230 MW is less than the sum of the estimated electrical power drains of the remaining reactor systems (poloidal field coils, OH solenoid, beam injectors, blanket coolant systems, pumps, and HVAC).

6. CONCLUSIONS

The basic TORFA concept of Ref. 1 can be adapted to the design of inexpensive D-T test reactors that readily satisfy DOE constraints on cost, (<\$1 billion), schedule for implementation (by 1990), and goals. More importantly, these inexpensive versions of TORFA retain the primary objectives established for the TORFA facility, which are the most concrete and sensible objectives yet proposed for the near-term controlled fusion program. One secondary TORFA objective must be abandoned in these less expensive facilities, namely the prototyping of the blanket region that would be used inboard of the plasma in a commercial production reactor. By retaining the same TF coil dimensions as in the reference TORFA, the lower field versions (Models B and E) provide the option of a later increase in magnetic field leading to a higher $n\tau_E$ and Q_p than available in the reference TORFA.

ACKNOWLEDGMENT

This work was supported by the U.S. Department of Energy, under Contract No. DE-AC02-76-CHO-3073 with Princeton University.

TORFA

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Table 1
ILLUSTRATIVE MODIFICATIONS OF THE REFERENCE TORFA CONCEPT

TORFA MODEL	REFERENCE TORFA	← INBOARD BLANKET MODULES REPLACED BY MINIMAL SHIELDING →				← CURRENT-DRIVING SOLENOID ELIMINATED →		
		Higher Q_p		Higher Q_p		Higher Q_p		High Q_p or Ignition**
		A	B	C	D	E	F	G
R_p (m)	5.05	4.55	4.55	4.35	4.10	4.10	3.75	4.58
a_p (m)	1.35	1.35	1.35	1.15	1.35	1.35	1.00	1.82
B_m (T)	7.50	6.42	7.50	7.50	5.50	7.50	7.50	7.50
B_t (T)	2.67	2.54	2.97	3.10	2.41	3.29	3.60	2.94
R/a	3.75	3.37	3.37	3.78	3.05	3.05	3.75	2.52
Δ_i (m)	1.65	1.15	1.15	1.15	0.70	0.70	0.70	0.70
*Relative $n\tau_E$ $\propto \beta B_t^2 a_p$	1.00	1.0	1.4	1.0	1.0	1.9	1.0	3.3 or 6.0**
TF coil power loss (MW)	230	170	230	230	125	230	190	230
Max. bending stress (psi)	24,000	17,500	24,000	24,000	13,000	24,000	24,000	24,000
*Max. Beam power (MW)	150	135	150	150	120	150	110	150

Notes

1. TF coils have rectangular shape (see Fig. 1), and in all cases are no larger than in the standard TORFA.
- *2. To achieve highest possible β , Q_p , and neutron wall loading.
- **3. Ignition using SMARTOR concept (See Ref. 3), but requires $n\tau_E \propto n^2 a_p^2$ or $n\tau_E \propto n^2 a_p^2$.
- *4. Assumes $n\tau_E \propto n a_p$.
5. In all cases, plasma elongation = 2.0, except in model H where $b/a = 1.5$.

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August 1980

TORFA

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Figure 1. Elevation view of the reference TORFA (Model A).
[Taken from Ref. 1.]

Figure 2