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A SMALL-ASPECT-RATIO TORUS FOR DEMONSTRATING THERMONUCLEAR IGNITION

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Abstract

A tokamak with 2.6-m major radius and aspect ratio of 2.0 is proposed for demonstrating thermonuclear ignition in deuterium-tritium. The 6-MA plasma current is established in part by co-injection only of 40 MW of 80-keV neutral beams (inducing ~ 2 MA at low density) and in part by the flux swing of the equilibrium-field system (inducing ~ 4 MA as the plasma pressure is increased) — there is no central current transformer and no poloidal-field coils inboard of the plasma. The core of the device consists simply of a 1.9-m-diameter steel-reinforced conducting trunk formed by coalescence of the inner legs of the toroidal-field coils. With a tensile stress at the copper trunk of 1000 kg/cm^2, corresponding to 11 T, the plasma density is sufficiently large to provide a comfortable safety margin for achieving ignition conditions.

1. INTRODUCTION

Consider two tokamak devices that have exactly the same physical characteristics, except for the major radii, R_0. Each normal-conducting toroidal-field coil has the same bore, radial build, and maximum field B_m at the coil windings. The plasma in each device has radius a_p, and is to be heated to a given temperature, T_i. Then it is expected from the commonly used "empirical energy scaling" that $n\tau_E = n^2 a_p^2\ G(T_i)$, or $n\tau_E \propto p^2 a_p^2$ for a given T_i, where p is the plasma pressure. Also, fusion power density

$P_f \propto n^2 \overline{\sigma v}$, which is nearly proportional to p^2 for the temperature range around 10 keV. Thus if $p \approx$ constant as the major radius is varied, the reactor-plasma parameters $n\tau_E$ and P_f are approximately constant, so that the performance of the device is nearly independent of R_o.

Now with increasing R_o, the facility cost tends to increase: (i) The Joule heating in the TF-coil set, (ii) the stored magnetic energy, (iii) the neutral-beam power required to reach T_i, and (iv) the tritium throughput per given pulse length are all proportional to R_o. (v) The flux swing required to establish a given plasma current increases faster than R_o. (vi) The building required to house the machine increases somewhat with R_o (although the size may still be determined principally by the length of neutral-beam lines). Thus the cost of a tokamak device of given a_p and B_m, of its auxiliary equipment, and of the associated power supplies are together roughly proportional to R_o. Hence it is desirable to operate at as small an R_o as possible, consistent with adequate access for beam injection and disassembly.

Now consider the assumption that the plasma pressure p is approximately constant with varying R_o. We define $\bar{\beta} = 8\pi\bar{p}/B_t^2$, where B_t is the magnetic field at R_o. For a given $\bar{\beta}$, $\bar{p}$ decreases as R_o decreases — i.e., as the aspect ratio $A = R_o/a_p$ decreases — because B_t/B_m decreases. However, it is known that for a given plasma shape and safety factor q, the maximum $\bar{\beta} \propto a_p/R_o$; this result follows either from the most rudimentary considerations of plasma equilibrium [1], or from the most recent results of sophisticated MHD stability codes [2] that determine the thresholds of balloning modes. Hence

$$\max \bar{p} \;\propto\; \frac{a_p}{R_o} B_m^2 \left(\frac{R_o - a_c}{R_o}\right)^2 \qquad (1)$$

where $a_c \simeq a_p + \Delta$ is the coil minor radius, and is constant. For copper TF coils, $\Delta \approx 0.1$ to 0.4 m, depending on the need for shielding to protect the coil insulation. From Eq. (1), it is found that $\bar{p}$ changes by relatively little in the range R_0/a_p = 2 to 6, for $R_0 \gtrsim 3$ m.

Thus the increase in maximum allowed $\bar{\beta}$ with increasing inverse aspect ratio permits one to profit from the advantages of small major radius, for a given minor radius — provided that one can establish the required plasma in a small-major-radius device. A principal difficulty in minimizing R_0 is the need for central current transformer, including space for the primary windings. There is also a requirement for access to the complicated core of the device, a problem that becomes especially acute in DT-burning reactors.

2. MODIFIED TOKAMAK CONFIGURATION

This paper proposes several modifications in the usual configuration and operation of a tokamak that should permit the attainment of thermonuclear ignition conditions in a copper-coil device with major radius $R_0 \sim 2.5$ m, using moderate magnetic field strengths. The essential features are illustrated in Fig. 1 and summarized as follows:

(1) As proposed recently for a superconducting-coil tokamak [3], there is no centrally located current transformer. A new feature is that the plasma current is established by a neutral-beam-induced current (~ 2 MA) together with the flux swing set up by the equilibrium-field coils ($\gtrsim$ 4 MA). After the final plasma pressure is attained, the current decays with a time constant greatly exceeding 10 s.

(2) The inner legs of the toroidal-field coils coalesce to form a solid 1.8 m-thick steel-supported water-cooled copper trunk. Compressive forces are thus taken up by reaction of the legs against each other, as well as against the steel center post. The small plasma aspect ratio ($R_0/a \gtrsim 2$, $\bar{\beta} \sim 0.1$) permits a large plasma pressure when the maximum magnetic field at the coils (at R = 0.9 m) is in the range 10 to 11 T.

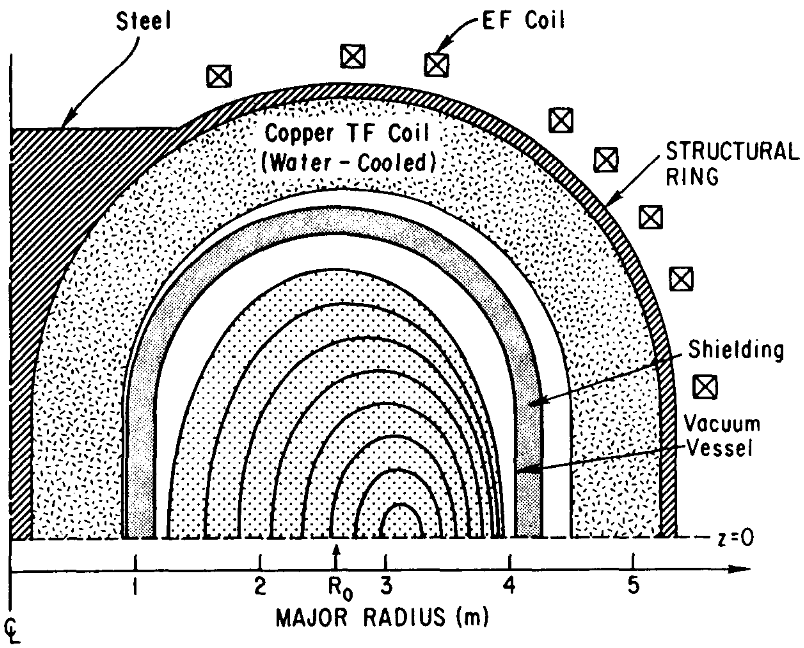

Figure 1. Elevation view of illustrative design for SMARTOR (Small-Aspect-Ratio Torus). EF coil positions are schematic only. B_m = 11 T at R = 0.9 m.

(3) The region inboard of the plasma contains none of the usual paraphernalia for a current transformer. The core is so simple that ease of access to this region for remote handling is not a limiting factor in determining the device size. (In contrast, tokamak designs with complicated core construction favor the use of large aspect ratios to facilitate access to the central region, and thereby are limited to relatively small plasma beta-values.)

It should be emphasized that the beam-induced current [4] is utilized <u>only during startup</u>. That is, there is no steady-state beam injection which would jeopardize the attainment of high fusion power multiplication, and in any event is problematical from the point of view of penetration of the final high-density plasma.

Section 3 describes the determination of parameters for the present configuration, called SMARTOR (Small-MAjor-Radius-TORus, or SMall-Aspect-Ratio-TORus). Section 4 outlines how the plasma current can be induced by moderate-energy neutral beams together with the equilibrium-field system. Section 5 summarizes the advantages and disadvantages of the SMARTOR configuration.

3. REFERENCE DEVICE PARAMETERS

The geometry of Fig. 1 has been derived from an optimization procedure [5], using the following specifications: (i) $\bar{\beta} = 0.20\ a_p/R_o$, where $\bar{\beta}$ is the spatially averaged plasma pressure divided by the magnetic field pressure at the magnetic axis. This expression for $\bar{\beta}$ is consistent with the limiting $\bar{\beta}$ found from MHD stability codes [2] for a D-shaped plasma with vertical elongation b/a ~ 1.6. (ii) Maximum tensile stress = 1000 kg/cm^2 at the coil windings for a pure-tension shape. (iii) Current density = 2.0 kA/cm^2 in the TF-coil trunk. (iv) The plasma $n\tau_E$ is found from the "empirical scaling law" for energy confinement [6], and is required to be M times the $\bar{n}\tau_E$ value for ignition with a particle-averaged temperature $\bar{T}_i$ = 8 keV, when the plasma pressure at r = 0 is $\sqrt{2}$ times the average plasma pressure.

Figure 2 shows the variation of plasma size with M, calculated using the above specifications. The principal geometric and plasma parameters for M = 3.5 are given in Table 1, which corresponds to the layout shown in Fig. 1. For a circular plasma of the same parameters of Table 1 except for the elongation factor, the $\bar{n}\tau_E$ calculated from the "empirical" relation is approximately equal to that required for ignition.

4. PLASMA START-UP

The plasma current to be established in the reference SMARTOR design is 6 MA. The EF (equilibrium-field) coils must provide the flux swing to induce the major portion of this current at the same time that they provide the required vertical field. Provided that the plasma current profile is not

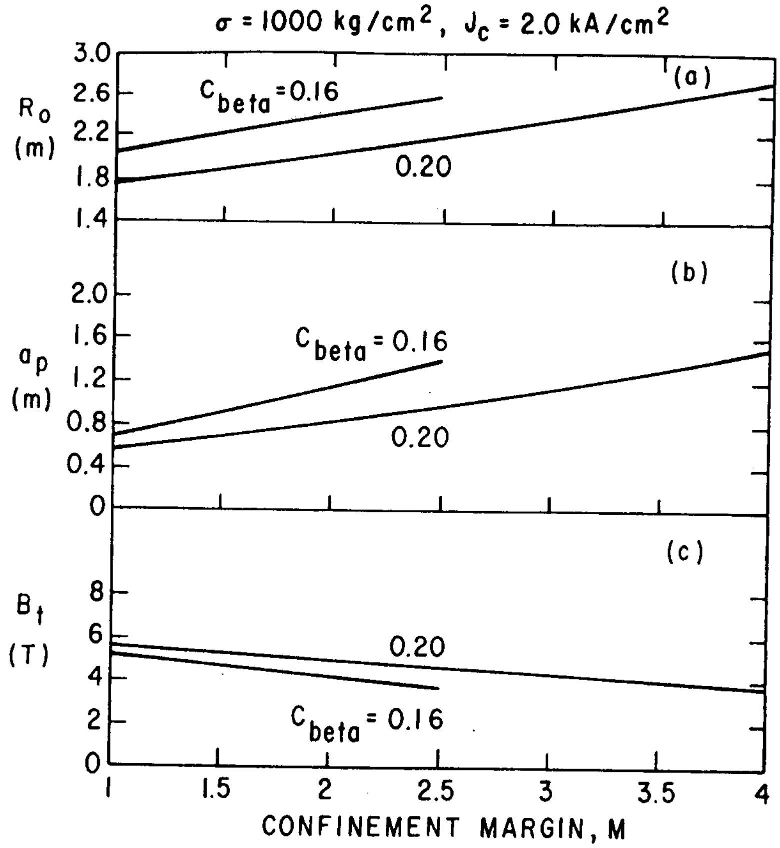

Figure 2. Required (a) major radius, (b) plasma radius, and (c) magnetic field at the plasma for thermonuclear ignition, when the tensile stress in the coil is 1000 kg/cm^2 and B_m = 11 T. M is the degradation in $n\tau_E$ from the value given by "empirical scaling". Beta = $C_{beta}\, a_p/R_o$.

too peaked, it appears possible for the EF coils to provide the entire flux swing required for current start-up. In any case, the plasma pressure must rise in a fairly well-prescribed fashion, with neutral-beam heating initiated at the early stages of current start-up (e.g., I_p ~ 200 kA). The vertical field is increased to accommodate the increasing plasma pressure, and the incremental flux swing further increases the plasma current. Finally, the EF coils must also provide the proper shaping field for the final D-shaped plasma. (There is also the option of using iron or magnetic steel for the

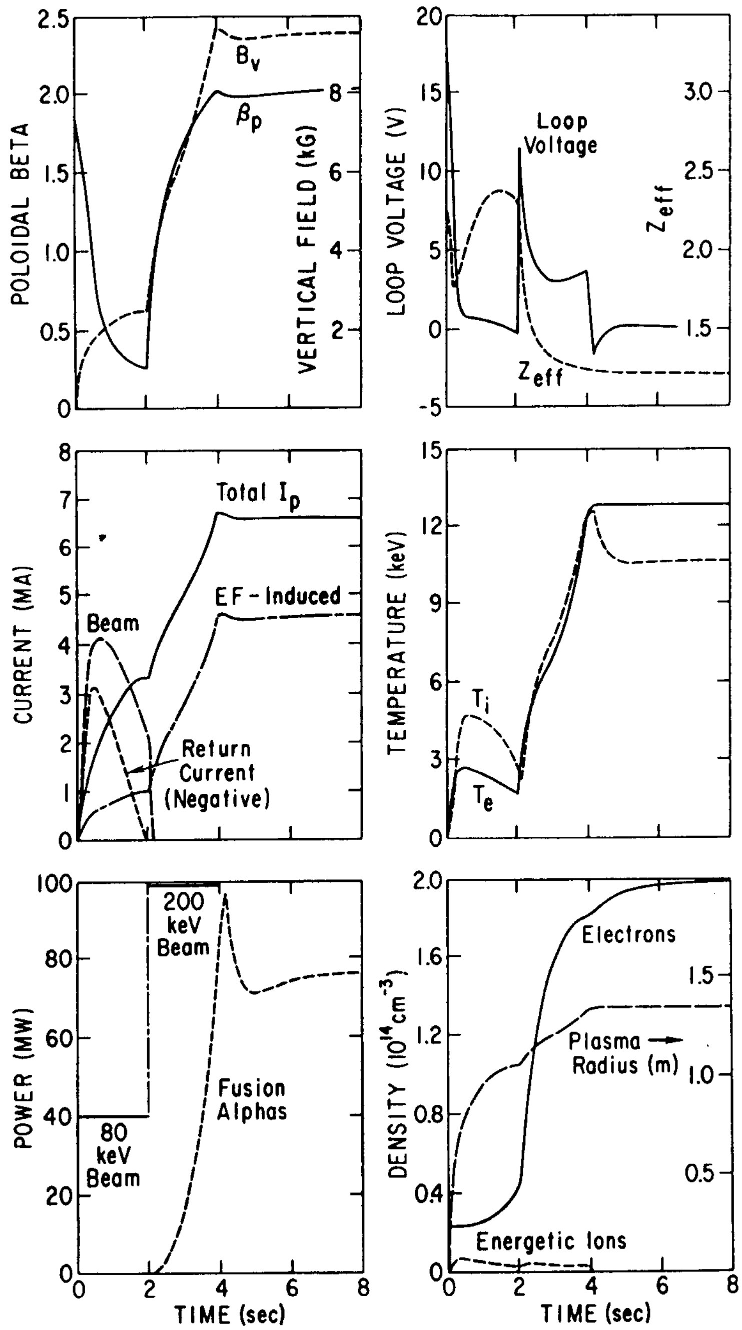

Figure 3. Illustrative time evolution of plasma parameters during start-up. Neoclassical skin resistance is enhanced by a factor of 2.

central structure, to form part of a small transformer providing the initial flux swing.)

This section summarizes the results of a simple zero-dimensional numerical treatment of start-up [7], which indicates that the EF system can supply at least 4 MA. The time variation of important plasma parameters is shown in Fig. 3, for a geometry slightly different than that of Table 1.

(i) 0 to 0.01 s The filling gas is broken down at R = 3.3 m by a pulse of radiation at the electron cyclotron frequency.

(ii) 0.01 to 0.05 s A current of 150 kA is established in a 40-cm radius plasma centered at R_0 = 3.3 m, by an extremely rapid increase of current in several EF coils. A hexapole null expels the vertical field from this region at low current. This start-up procedure is similar to that used in the ATC device, which also had no poloidal-field coils inboard of the plasma [8].

(iii) 0.05 to 2.0 s During this period, 500 A-equiv. of 80-keV D° beams (40 MW) are injected parallel to I_p. The fast deuterons augment the plasma density, and pitch-angle scatter while slowing down. The plasma current induced at time t by N_h beam ions injected at $t = t_0$ is

$$\Delta I_p(t_0,t) = eN_h\langle v_\parallel\rangle \left[1 - \frac{1}{Z_{eff}} + R_{tr}\right]\left[1 - e^{-(t-t_0)/\tau_d}\right] \tag{2}$$

where R_{tr} is a term that accounts for banana-trapped electrons. The exponential factor is due to the electron return current, which decays with a time constant τ_d determined by the neoclassical skin resistivity, enhanced arbitrarily by a factor of 2.

As I_p increases, the plasma radius is increased to maintain q_a = constant, with R_0 reduced continuously so that the outer edge of the plasma remains at R = 3.7 m. At t = 2.0 s, one has a_p = ~~0.8~~ 1.0 m, $n_e = 4.2 \times 10^{13}$ cm^{-3} (the increase is due to gas puffing plus decelerated fast ions), T_e = 1.7 keV, and T_i = 2.2 keV. The injected beams have induced ΔI_p = 2.1 MA. During

TABLE 1. ILLUSTRATIVE DEVICE PARAMETERS
(See Fig. 1)

R_o (m)	2.60
a_p (m)	1.35
b/a	1.6
$\bar{\beta}$	0.10
J_c (kA/cm^2)	2.0
B_m (T)	11.1
B_t (T)	3.8
I_p (MA)	6.0
$\bar{T}_e = \bar{T}_i$ (keV)	8.0
$\bar{n}_e$ (cm^{-3})	2.3×10^{14}
$\bar{P}_f$ (MW/m^3)	4.4
Fusion power (MW)	525
Average neutron power loading (MW/m^2)	2.1

this period, B_v has been raised to 0.25 T because of the increase in plasma current and pressure. The increase in applied vertical-field flux at $R < R_o$ results in an additional $\Delta I_p = 1.1$ MA, so that the total plasma current is 3.3 MA at t = 2.0 s.

The toroidal electric field, which is due to the change in applied vertical field, can be quite large (see Fig. 3), and results in significant acceleration of the fast ions. This "energy clamping" effect increases the beam-induced current, but is not taken into account in the present analysis.

(iv) <u>2.0 to 4.0 s</u> At t = 2.0 s, the 80-keV beams are shut off, and

500 A-equiv. of 200-keV D° beams (100 MW) are injected nearly perpendicularly to the magnetic axis. An electron "return current" now counteracts the decay of the beam-induced current. In fact, this decay must proceed very slowly, because T_e is now rapidly increasing. The fusion alpha power becomes significant at $t \approx 3$ s. The continuous increase in applied vertical field results in a plasma current of 6.7 MA at $t = 4.0$ s. To keep q_a = constant as I_p increases, a_p is increased to 1.35 m with R_o decreasing to 2.35 m, and b/a increasing to 1.6. At $t = 4.0$ s, one has $T_i \approx T_e = 12.7$ keV, with $n_e = 1.8 \times 10^{14}$ cm^{-3} and $\beta_p = 2.0 \approx R_o/a_p$.

(v) Beyond 4.0 s Thermonuclear ignition is reached at $t = 4.0$ s, so that the 200-keV heating beams can be shut off. For $t > 4$ s, T_e is clamped at 12.7 keV. The decay time of the plasma current is of the order of 100 s, so that even at $t = 15$ s, $I_p = 6.5$ MA. Hence there is no need to provide flux swing to sustain the final current in an ignition test reactor.

5. SUMMARY

The configuration and operating mode proposed herein make it possible to realize a copper-coil ignition test reactor with small major radius (~ 2.5 m) and very small aspect ratio (~ 2), thus allowing a stable plasma equilibrium with $\bar{\beta} \sim 0.1$. The principal advantages are the following:

(1) The core of the device (inboard of the plasma) contains no poloidal-field coils, is extremely simple, and should present no serious problems for remote handling.

(2) No separate ohmic-heating power supply is required.

(3) The small major radius results in significant savings in the TF-coil power requirements, neutral-beam injector costs, and tritium inventory.

(4) The limit to the pulse length is determined by the temperature rise of the water-cooled TF coils, and can be made many tens of seconds.

The principal disadvantages of the proposed design are the following:

(1) A second set of neutral-beam injectors is required for establishing the beam-induced current, in the event that the EF flux swing is insufficient for inducing the entire plasma current.

(2) The same set of poloidal-field coils perform 3 functions, so that each pair of coils requires individual programming.

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