1371

SEPTEMBER 1977 PPPL-1371

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This work was supported by U. S. Energy Research and Development Administration Contract EY-76-C-02-3073.

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SMARTOR — 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 1.9 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.

Alternate designs are presented, each with an aspect ratio of 1.9, with R_o = 2.6 m and a plasma density sufficiently large to provide a comfortable safety margin for achieving ignition conditions. The first design features higher beta ($\bar{\beta}$ = 0.10, b/a ~ 1.6) with low tensile stress at the copper trunk (1000 kg/cm^2), while the second features lower beta ($\bar{\beta}$ = 0.06, b/a ~ 1.2) with high tensile stress (1800 kg/cm^2). Extension of this small-major-radius, small-aspect-ratio configuration to an economically practical fusion reactor is also examined.

1. INTRODUCTION

This paper describes 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 $R_o \sim 2.5$ m, using moderate magnetic field strengths. The essential features are illustrated in Figures 1 and 2 and summarized as follows:

(1) As proposed previously for a superconducting-coil tokamak,(1) 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 (~ 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 1.9 m-thick steel-supported water-cooled copper trunk. Compressive forces are thus taken up by diametric reaction of the legs against each other. The small plasma aspect ratio ($R_o/a \lesssim 2$, $\bar{\beta} \sim 0.1$) permits a large plasma pressure when the magnetic field at the plasma center is in the range 3.5 to 5T.

(3) The region inboard of the plasma contains no poloidal-field coils, or special structural supports. The core is so simple that ease of access to this region for remote handling is not a limiting factor in 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.)

(4) The small major radius results in energy and power requirements for the toroidal field coils that are greatly reduced compared with the corresponding requirements for a device of more conventional design with comparable plasma parameters.

Section 2 describes the determination of parameters for this configuration, called SMARTOR (Small-MAjor-Radius TORus, or SMall-Aspect-Ratio TORus).

Section 3 outlines how the plasma current can be induced by moderate-energy neutral beams together with the equilibrium-field system. Section 4 examines the applicability of the SMARTOR configuration to an economically practical tokamak fusion reactor.

2. DETERMINATION OF DEVICE PARAMETERS

The purpose of the SMARTOR design is to minimize the major radius R_o of the tokamak, while eliminating complicated construction on the inboard side of the plasma. The present approach features (i) elimination of all current-transformer paraphernalia inboard of the plasma (i.e., no coils or iron core); (ii) coalescence of the inner legs of the toroidal field coils into a solid trunk, thus providing their own support structure in this region; (iii) limiting the tensile stress in the trunk to the 15,000 psi range (for the higher-beta option); (iv) the smallest possible plasma aspect ratio for achieving the required plasma confinement.

The principal parameters used in this analysis are the following (with units):

a_c = radius of the conducting portion of the trunk (m)

a_p = plasma half-width (m)

B_t = toroidal magnetic field at $R = R_o$ (T)

B_m = toroidal magnetic field at $R = a_c$ (T)

I_c = total trunk current (A)

J_c = current density in the trunk (A/m^2)

n = plasma density (cm^{-3})

τ_E = energy confinement time (s)

$\bar{\beta}$ = (average plasma pressure)/(magnetic pressure at $R = R_o$)

σ = tensile stress (kg/cm^2)

2.1 PARAMETRIC VARIATION

The process of plasma start-up by beam-induced current and equilibrium-field programming is discussed in Sect. 3. For the present we assume that the conventional centrally located current transformer can be eliminated, and investigate the optimal device parameters. These parameters are determined according to the following constraints.

Power Dissipation. J_c = constant

$$\text{or} \quad I_c = \pi J_c a_c^{\;2} \tag{1}$$

Tensile Stress. Assuming a pure-tension TF-coil configuration,[3]

$$J_c B_m a_c \approx 2\times10^5\sigma \quad \text{or} \quad I_c B_m = 2\pi\times10^5\,\sigma a_c \tag{2}$$

Ampere's Law.

$$a_c B_m = \mu I_c \tag{3}$$

$$R_o B_t = \mu I_c \tag{4}$$

where $\mu = 2\times10^{-7}$.

Plasma Confinement. The "empirical scaling law" for confinement of the total plasma energy is[2]

$$\bar{n}\tau_E = 3\times10^{-15}\, q_a^{1/2}\bar{n}^2\langle a_p^{\;2}\rangle \tag{5}$$

Also $2\,\overline{nT_i} = 2.5\times10^{15}\,B_t^2\bar{\beta}$, with B in Teslas, T_i in keV, and n in cm^{-3}. It is known that the attainable "beta" is $\bar{\beta} = C_{beta}(a_p/R_o)$,[4] where C_{beta} is maximum at an optimal q_a (the limiter safety factor) ~ 3 and b/a (plasma vertical elongation) ~ 1.5 to 2. At the present time it is not known whether Eq. (5) will hold in tokamak plasmas where ion heat conduction is the dominant energy loss mechanism. Present experiments indicate that τ_{Ei} is nearly independent of density and temperature, so that $\bar{n}\tau_{Ei} \propto \bar{n}$.

Spatial peaking of $n(r)$ and $T_i(r)$ results in an increase in average fusion power density by a factor R_f, so that the required $\bar{n}\tau_E$ for ignition is reduced by a factor R_f.[5] To take into account the possibility of enhanced diffusion at high plasma temperature, or significant radiation loss, the required $\bar{n}\tau_E$ is multiplied by a "safety margin" $M > 1$. Taking $b/a = 1.6$, Eq. (5) becomes

$$\frac{(a_p B_t)^2}{R_o} = \frac{\bar{\bar{T}}_i}{C_{beta}} \; \frac{(M\bar{n}\tau_E)^{1/2}}{8.8\times10^7 \; q_a^{\;1/4}} \tag{6}$$

where $\bar{n}\tau_E$ is the value required for a particular application and includes the factor R_f. Note that $\bar{n}\tau_E \propto (a_p B_t)^4/R_o^{\;2}$, so that increasing the plasma radius is as effective as raising the magnetic field — provided that R_o can be kept small.

<u>Geometry</u>

$$R_o = a_c + a_p + \Delta_t \tag{7}$$

where Δ_t is the total distance between the surface of the conducting portion of the trunk and the inner edge of the plasma "scrape-off" layer ($\Delta_t = R_{vl} - R_{cl} + \Delta_1$ in Fig. 1).

In Eqs. (1) to (7), J_c, σ, C_{beta}, M and Δ_t are constants. There are 6 variables: a_c, a_p, B_m, B_t, I_c, R_o. Equations (1) to (3) give a_c, I_c, and B_m:

$$a_c = 5.6\times10^5 \; \frac{\sigma^{1/2}}{J_c} \tag{8}$$

$$I_c = 10^{12} \, \frac{\sigma}{J_c} \tag{9}$$

$$B_m = 0.35 \; \sigma^{1/2} \tag{10}$$

(Note that B_m corresponds to a magnetic pressure p_m, where B_m(Tesla) = 0.50 $p_m^{1/2}$(kg/cm^2). That is, σ = 2.0 p_m.) Equations (4), (6) and (7) give a_p, B_t, and R_o. To find a convenient solution, we assume that the plasma aspect ratio, A, is $R_o/a_p = 2 + \delta$, where $\delta << 2$. It is found later that for reasonable values of J_c and σ, R_o/a_p is indeed near 2. Then

$$a_p = \frac{a_c+\Delta_t}{1+\delta} \tag{11}$$

where

$$\delta \approx \frac{k-1}{1-\frac{3}{2}k} \text{ if } \delta << 2. \tag{12a}$$

with

$$k = \frac{\bar{T}_i \ (M\bar{n}\tau_E)^{1/2}}{1.1\times10^7 \ q_a^{1/4} \ C_{beta}} \ \frac{(a_c+\Delta_t)}{(\mu I_c)^2} \tag{12b}$$

Then R_o is found from Eq. (7), and B_t from Eq. (4). To display a range of possible solutions, with R_o/a_p differing significantly from 2, it is necessary to retain terms up to δ^2. Then two plasma sizes are consistent with each set of constants.

The deposition of fusion-neutron energy in the structure beyond the vacuum vessel wall places significant demands on the tokamak cooling system. It is preferable to enlarge the radius a_c of the copper trunk so that it can remove <u>all</u> the nuclear heat, as well as the Ohmic power dissipation, rather than to employ a smaller trunk surrounded by water-cooled shielding. The reasons are the following: (1) For the same plasma size and B_t, I_c is constant, so that the stress on the coils $\propto a_c^{-2}$. (2) For the same I_c the Ohmic power dissipation in the trunk $\propto a_c^{-2}$. (3) The core construction is simplified by the elimination of an additional cooling system.

(For the outer legs of the TF coils, where access is much easier, sufficient space can be specified between the vacuum vessel and the TF coils for water-cooled shielding to absorb most of the neutron energy. In the trunk region, about 15 cm of uncooled shielding may be added just beyond the vacuum vessel to provide some protection for the insulators in the trunk, as indicated in Figs. 1 and 2. The shielding might consist of graphite, which would reach a temperature of 1500°C or more during a power pulse, but which would radiate its heat to the water-cooled copper TF coils between pulses. At any rate, the duty factor of an ignition test reactor is expected to be sufficiently small so that serious radiation damage to the insulator will not occur.)

Figure 3 shows R_o, a_p, and B_t as a function of tensile stress for pure-tension coils, for $\Delta_t = 0.35$ m, $\bar{T}_e = \bar{T}_i = 8.0$ keV, and $R_f = 2.0$. The smallest of the two solutions for R_o is given in this figure. (Note that $\bar{T}_i$ is the particle-averaged temperature. In a plasma with realistic profiles, $T_i(0) \sim 12$ to 15 keV. For $T_i \sim 10$ keV, $R_f = 2.0$ is equivalent to $\beta^* \approx 1.4\ \bar{\beta}$, where β^* is the root-mean-square plasma beta.) The required $\bar{n}\tau_E$ is $M \times 2.5\times10^{14}\ cm^{-3}s$. It is found that solutions do not exist for very small σ and small beta.

For $q_a \approx 3$ and $b/a \approx 1.6$, which are perhaps the optimal values for maximizing beta, the plasma current I_p must be in the range 7 to 9 MA, an excessive current for an ignition test reactor. Hence we propose to operate at $q_a \lesssim 4$ with $I_p \sim 6$ MA. Preliminary results from MHD stability analyses indicate that at $q_a \lessapprox 4$, $C_{beta} = 0.20$ is a reasonable expectation when b/a is optimized.[4] (Note that for plasmas with aspect ratio ~ 2, a current of at least 4 MA is required to confine most fusion alphas in the hot central region.)

Figure 4 shows how the plasma parameters vary with the safety margin, M, for a tensile stress of 1000 kg/cm^2 (14,200 psi), a rather comfortable value. Solutions do not exist for very large M-values, when beta is small. Evidently, if the presently observed empirical scaling law continues to hold at high $T_i \sim T_e$, and if radiation losses are small (i.e., M ~ 1), then ignited tokamak plasmas with $R_o \lesssim 2$ m are possible with moderate magnetic fields.

Table 1 shows two sets of reference design parameters, with the same plasma aspect ratio, major radius, and pressure, and both calculated for M = 3.5. The higher-beta option features considerable vertical elongation, but the tensile stress and current density in the trunk are moderate. The lower-beta option places relatively little demand on plasma shaping and MHD stability, but at the expense of large tensile stress and current density in the trunk in order to achieve the same plasma pressure.

2.2 TOROIDAL-FIELD COILS

In order to accommodate the plasma, the radius R_1 at which the vertical bore of the TF coils is largest should be near R_o. If R_1 = 2.6 m for the examples of Table 1, then the horizontal bore of a pure-tension D-shaped coil[3] is $R_1^2/a_c - a_c = 6.6$ m. For a TF coil set of that size, the magnetic energy requirement would be uncomfortably large. Consequently, it would seem more appropriate, at least for the higher-beta lower-field option, to use coils of racetrack shape, similar to those of PDX,[6] and with a horizontal bore of only 3.6 m. Then the stress in the curved sectors of the coil is likely to be in the range of 20 to 24 kpsi, but still tolerable. For the lower beta option, on the other hand, the stress at the coil is 80% greater, so that a coil shape approaching the pure-tension variety may be essential. In that case, the stored magnetic energy could be three times that of the higher-beta racetrack design.

Figure 5 depicts the racetrack TF coil for the higher-beta option of SMARTOR. Table 2 compares the coil parameters with those of other large copper-coil devices: PDX,[6] Doublet III,[7] and TFTR.[8] While the maximum field at the coil windings is considerably larger for SMARTOR, the tensile stress is no greater than in the other devices, and as pointed out above, the solid trunk construction of SMARTOR enables centering forces to be counteracted much more easily.

The SMARTOR coil set has 70% more stored energy than the TFTR coil set. Beginning from a vertical distance of 1.5 m from the midplane, the axial and radial builds of the TF coils can be made to increase in order to reduce power dissipation as well as stress in these regions. Then the resistive power loss of the SMARTOR coils is significantly less than that of the TFTR coils. Because of the large conductor volume, it is anticipated that flat-top times exceeding 10 s are realizable.

The 34-cm-diameter steel center post takes up 4% of the trunk area, so that the average current density of 2000 A/cm^2 in the trunk is achieved with J_c = 2100 A/cm^2 in the water-cooled copper region. In the absence of shielding, the fusion-neutron power deposited in the 2.4-m tall portion of uniform cross section is about 30 MW, which is comparable with the Joule heating in this region (~ 45 MW).

2.3 NEUTRAL-BEAM INJECTION

The ripple at the outside edge of the plasma (R = 3.95 m) is about 2.5% when the number of TF coils is 16 and the axial build of the copper is 0.65 m in each outer leg.* The axial distance between the coils, including structural support, is 0.72 m for $4.4 \lesssim R \lesssim 5.4$ m. This space is adequate for tangential injection of 100 A-equiv. of the 80-keV beams used for current induction (see Sect. 3.2). Five of these beam-lines are required.

The ripple-trapping injection method (using near-vertical beam injection) is especially suited for final heating of the SMARTOR plasma, in that the required injection energy is proportional to $(R_oB_t)^{2/5}$, for a given plasma opacity and applied ripple strength. For the relatively small value of R_oB_t in SMARTOR, a deuterium beam energy of 80 keV is suitable. Hence the same power supplies used to operate the current-inducing beams can later be switched to the heating beams. However, in the present analysis it is assumed that the heating beams are 200-keV D° injected perpendicularly in the conventional manner.

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* To reduce the ripple at (R = 3.95 m, z =0) to less than 1.0%, the outer legs of the TF coils in Fig. 5 must be displaced outward by 30 cm. The ripple decreases very rapidly at smaller R.

3. START-UP

3.1 GENERAL APPROACH

In order to dispense with the centrally located current transformer, the plasma current is established by a neutral-beam-induced current acting in concert with the change in flux-swing $\Delta\phi_V$ set up by the vertical magnetic field, B_V. In most large tokamak devices, $\Delta\phi_V$ is used to augment the flux-swing $\Delta\phi_H$ established by the conventional plasma current transformer (i.e., via the "OH coils", or iron-core transformer): typically, $\Delta\phi_V \sim \frac{1}{3}$ to $\frac{1}{2}$ $\Delta\phi_H$. In the preliminary design of a 7.7-MA, R_O = 6 m tokamak reactor, it was proposed[1] that $\Delta\phi_H$ could be replaced by the flux change resulting from reducing the current in "nulling coils" located on the outboard side of the plasma, but inside the TF coils; just before discharge initiation, the nulling coils cancel out the B_V set up by steady-state, superconducting EF (equilibrium-field) coils. However, this technique turned out to be grossly inadequate because of insufficient area between the inner edge of the plasma and the outer edge of the superconducting TF coils, from which rapid flux change had to be excluded by special shielding coils.

In the present scheme, $\Delta\phi_V$ is in fact adequate for producing the desired final current because of 3 important features:

(1) Soon after discharge initiation, a plasma current of at least 2 MA is built up directly by tangential injection of moderate-energy neutral beams.[9] The use of injected beams to drive toroidal currents in the steady state has been discussed previously.[10,11] The new feature here is the application of the

beam-induced current to relieve the inductive volt-seconds demand on the poloidal-field coil system during start-up. There is no beam-induced current in the steady state. (Intense beam injection near the beginning of start-up also reduces the resistive volt-seconds that must be supplied by external coils.)

(2) The ratio of the flux-swing $\Delta\phi_V$ to the inductive flux-swing $\Delta\phi_I$ required to establish a given current I_p is

$$\frac{\Delta\phi_V}{\Delta\phi_I} \sim \frac{B_V \; 2\pi R_o(a_p+\Delta_V)}{L_p I_p} \sim \frac{a_p+\Lambda_V}{R_o} \; \frac{2+\beta_p}{2.5} \tag{13}$$

where $a_p + \Delta_v$ is the radial extent of the region containing significant vertical flux, and is approximately constant for most high-density ignition devices. This ratio approaches unity for SMARTOR, but is considerably less for a "conventional" superconducting-coil device with the same plasma current. That is, in small-aspect-ratio devices, the EF system is much more effective in inducing the required plasma current.

(3) Because the TF coils are normal conductors, the rapidly rising vertical field can be permitted to penetrate the trunk, which at any rate is located at a small major radius and includes relatively little area.

Finally, we recall a previous suggestion for obtaining a small-aspect-ratio tokamak without a current transformer — the so-called "spherator-astron", in which both the plasma current and the toroidal-field current are to be driven by means of relativistic electron beams.[12]

3.2 TIME SEQUENCE OF START-UP EVENTS

This section presents an outline of the proposed plasma start-up procedure; the details will be found in Ref. (9). Figure 6 and Table 3 give an example of the results of Ref. (9) for a plasma with uniform density, temperature, and current density, and with a geometry approximately that of Fig. 5. The following is a description of the start-up events corresponding to Fig. 6 and Table 3.

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

(ii) <u>0.01 to 0.05 s</u> A current of 150 kA is established in a 40-cm radius plasma centered at R_o = 3.3 m, by an extremely rapid increase of current in several "initiation" EF coils. At t = 0.05 s, one has $T_e \sim 100$ eV, $n_e = 2.2\times10^{13}$ cm^{-3}, oxygen content = 3%, neutral density = 10^8 cm^{-3}.

(iii) <u>0.05 to 2.0 s</u> During this period, 500 A-equiv. of 80-keV D^o beams (40 MW) are injected parallel to I_p. The fast deuterons give up their energy to background ions and to electrons, and augment the plasma density. The fast deuterons also pitch-angle scatter while alowing down. The plasma current induced at time t by N_h beam ions injected at $t = t_o$ is

$$\Delta I_p(t_o,t) = eN_h \langle v_{||} \rangle \left[1 - \frac{1}{Z_{eff}} + R_{tr} \right] \times \left[1 - e^{-(t-t_o)/\tau_d}\right] \qquad (14)$$

where R_{tr} is a term that accounts for banana-trapped electrons.[9,11] The exponential factor is due to the electron return current, which decays with a time constant determined by the neoclassical skin resistivity, enhanced arbitrarily by a factor of 2. The neutral density, which causes charge-exchange loss of fast ions but also indirectly suppresses T_e, is maintained at 10^8 cm^{-3}. (Energy loss of bulk ions and electrons is by transport according to Eq. (5), charge exchange, and oxygen radiation.)

As I_p increases, the plasma radius is increased to maintain q_a = constant, with R_o 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 m, $n_e = 4.2\times10^{13}$ 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 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 flux at $R < R_o$ results in an additional ΔI_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. 6), and results in significant acceleration of the fast ions.(13) This "energy clamping" effect increases the beam-induced current, and thus can be used to minimize the injected beam power. However, it 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 down, 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. During the period of heating by the 200-keV beams, deuterium and tritium gases are puffed into the torus to increase the plasma density. 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. (In practice, R_o would be somewhat larger, with the vertical elongation of the plasma gradually increasing, as in Fig. 5. Beam energies of at least 300 keV may be required at the higher densities.) At $t = 4.0$ s, one has $T_i \approx T_e = 12.7$ keV, with $n_e = 1.0\times10^{14}$ 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 down. For $t > 4$ s, T_e is clamped at 12.7 keV. At this temperature, only 20% of the fusion-alpha energy is given up to the plasma ions, so that T_i plunges to 10.5 keV at $t = 5$ s (see Fig. 6). The plasma current drops slightly because of a small decrease in plasma pressure and then decays with a time constant $\tau_I \sim L_p/R_p$, where L_p is the plasma inductance. In fact, τ_I 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 volt-seconds to sustain the final current in an ignition test reactor; the pulse length will be determined by the rate of temperature rise of the TF coils.

Thus the present scheme appears to give the required build-up of plasma current without a centrally located current transformer. As evident from Fig. 6, the beam-induced current, $I_b - I_{ret}$, is limited by the finite decay time τ_d of the electron return current, I_{ret}. This problem is analogous to the "skin effect" in the ordinary current start-up procedure. It appears that

$I_b - I_{ret}$ could be raised from 2 MA to 3 MA if the plasma column were built up in a manner analogous to the "moving limiter" technique, so that τ_d always remains small.

3.3 SAVINGS IN VOLT-SECONDS

3.3.1 Resistive Volt-Seconds. As illustrated in Fig. 6, injection of 40 MW of 80-keV beams, beginning when I_p = 150 kA and $T_e \sim 100$ eV, raises T_e to about 2 keV in about 200 ms. If T_e were to be raised to a similar temperature by Ohmic heating alone, a current of several megamperes would be required, and the "resistive flux-swing" would be at least 3 volt-seconds, even when $Z_{eff} = 1$. If R_o greatly exceeds 2.5 m, as may well be the case for an ignition-sized reactor with a centrally located transformer, or if $Z_{eff} \sim 3$ during start-up, then a resistive flux-swing of at least 10 volt-seconds may be encountered during start-up. Thus initiating beam heating as soon as practical — when $I_p \sim 150$ kA — can result in a considerable savings in volt-seconds, even without the beam-induced current.

3.3.2 Inductive Volt-Seconds. A zero-order calculation of $L_p I_p$ indicates that the flux-swing required to generate 6.0 MA in the plasma shown in Fig. 5 is 19.5 volt-seconds. A very crude estimate of that portion of the flux-swing produced by the EF coils which is available to induce a plasma current is found by multiplying the final vertical field by the area between 0.9 m and 2.6 m; the result is 21 V-seconds. Because considerable resistive volt-seconds are normally required during start-up, it appears that the EF system by itself is not adequate. Hence the important role of the beam-induced current in providing about 30% of the final plasma current, with the accompanying savings in both inductive and resistive volt-seconds.

(For a moderately elongated plasma operated at $\bar{\beta} \sim 6$ to 10%, the EF system must be considerably strengthened, so that there may result in additional volt-seconds not taken into account here.)

It is relevant to note that neutral-beam currents of 300 A-equiv. at 20 keV have already been injected into the 2XIIB device at the Lawrence Livermore Laboratory,[14] with beam currents up to 500 A at higher voltage being planned for operation within 2 years. Another possibility for setting up the beam-induced current in SMARTOR is to make use of the IPINS beam sources,[15] also under development at Livermore. Each IPINS source is to supply a 1- μs pulse of 200-kA beam at 100 keV. This pulse length is about one-seventh of the transit time around a torus with $R_o \sim 3$ m, and thus represents a toroidal current of about 30 kA (after the decay of the electron return current). About 100 IPINS sources operating sequentially would be required for SMARTOR. Furthermore, the use of a large number of sequentially fired short-pulse beams would allow the beam focus to follow the plasma major radius as it decreases from 3.3 to 2.7 m.

Finally, we note that the beam power required for heating to ignition is proportional to R_o, while the injected power required to induce a given current is also proportional to R_o (i.e., induced current $\propto$ (transit time)$^{-1} \propto 1/R_o$). Recalling also the flux-swing argument of Sect. 3.1-(2), it is evident that the small-R_o SMARTOR configuration is "self-consistent", in that the required neutral-beam and EF functions are performed more readily than in tokamak configurations of larger major radius and aspect ratio.

4. COMMERCIALLY PRACTICAL SMARTOR REACTORS

4.1 SUPERCONDUCTING COILS

In order that the overall plant efficiency (= net electrical energy/thermal energy production) of a tokamak power reactor be at least 30%, superconducting toroidal-field coils must be employed. There are several reasons why a superconducting trunk is unattractive for the SMARTOR concept. (1) The shielding requirement (about 0.9 m vs 0.15 m in Fig. 5) results in a large increase in major radius, and a substantial drop in plasma aspect ratio and beta. (2) Increased structural requirement for a superconducting trunk results in a further increase in major radius. (3) It might well be necessary to shield the trunk from the rapidly changing vertical field.

A small major radius is the principal key to reduction in the unit cost and unit power output of a tokamak reactor. In order to retain this great advantage of the SMARTOR approach ($R_o \lesssim 3$ m), it seems worthwhile to consider using a copper or aluminum trunk attached to superconducting outer legs. With some additional shielding, the dimensions inboard of the plasma would be modestly larger than those of Fig. 5. The power dissipation of this "hybrid" coil would be approximately one-third of a copper-coil device of similar size, with 1 m of space on the top, bottom, and outboard sides for a tritium breeding blanket. The principal difficulty with this scheme arises from the heat leaks at the joints between the superconductors at 5 to 10° K and the normal conductors at room temperature. Each conductor carries a current of the order of 100 kA.

Reference (16) indicates that the liquid-helium refrigeration power required to compensate the heat transport across the normal conductor-superconductor junctions can in principal be made as small as 0.4 W/A, or 20 MW for a 50-MA coil set. Taking a less ideal value of 50 MW, the total power consumption of the TF coils would be at least 100 MW. The total fusion power production for a device with $R_o \sim 3$ m is in the range 500 to 750 MW, so that the coil power dissipation is probably prohibitively large.

Regardless of the fate of this type of "hybrid" coil design, the SMARTOR configuration with a completely superconducting TF coil set might well be attractive for non-tritium-burning, long-pulse experimental tokamaks, where no neutron shielding is required.

4.2 ULTRA-SMALL-ASPECT-RATIO SMARTOR (ULSAR)

The analysis of Sect. 2.1 indicates the possibility of an ignited copper-coil reactor with aspect ratio as small as 1.5, and corresponding $\bar{\beta} \sim 0.15$. However, the major radius of such a device would exceed 4 m, and the size of the plasma would greatly exceed the minimum requirement for ignition. Such devices can be found by extrapolation of Fig. 4 to very large M, or by adopting the solution for δ that gives the larger R_o (see Sect. 2.1).

About 40 cm of shielding would be required to retard radiation damage to the insulators in the trunk, although this problem is not critical considering the relatively small voltages that must be withstood. The fusion-neutron flux in this region is also expected to be substantially smaller than the average value around the torus, mainly because of the outward displacement and compression of the plasma pressure surfaces.(17) Furthermore, the entire trunk could be replaced once every several years. Because of the extremely small aspect ratio, as well as the outward plasma displacement, less than 15% of the total fusion-neutron production impinges on the wall enclosing the copper trunk, so that tritium breeding in this region can be dispensed with.

Unlike the small reference design of Sect. 2.2, the horizontal bore of the ULSAR coils would necessarily be sufficiently large so that a pure-tension design is most suitable. For $|z| > 2$ m, the radial and axial builds of the copper can be made very large, in order to minimize power dissipation. The large size of ULSAR results in a fusion power output that, after thermal conversion, exceeds the power dissipation in the copper coils by a large factor; the coils can also be designed for steady-state operation in this instance. Whether this power ratio is sufficiently favorable for a power reactor has not yet been determined.

4.3 PARTIALLY CRYOGENIC TF COILS

For the ULSAR design of Fig. 7, $J_c = 1.75$ kA/cm^2, $B_m = 11$ T, aspect ratio = 1.68, $\bar{\beta} = 0.12$, and M = 5. The TF coils are sufficiently oversized so that the EF coils can be located inside the TF coils, but outside the 0.75 to 1-m tritium-breeding and heat-conversion blanket. This placement reduces the equilibrium-field power, while minimizing activation and maintenance problems. There are still no poloidal-field coils inboard of the plasma.

The ripple at the outside edge of the plasma, R = 5.95 m, is at most 1% using just 12 TF coils with an axial build of 2 m at R = 10 m. The toroidal field at the outside edge is only 1.85 T, so that it is conceivable to install a bundle magnetic divertor(18) for impurity control and pumping during the burn. The decay time of the plasma current is expected to be several hundred seconds.

The top, bottom, and outboard legs of the TF coils can be operated steady-state at liquid-nitrogen temperature, as indicated in Fig. 7, resulting in a reduction in Joule heating of a factor of 7 in these legs. Taking into account the refrigeration power, the total power consumption for these sections of the TF coils can be reduced by a factor of 3 from the room-temperature case. The power dissipation in the entire TF coil assembly is estimated to be approximately 140 MW, assuming that complete shielding of the liquid-nitrogen-cooled region from fast neutrons is provided by the blanket. For the design shown in Fig. 7, the fusion power production is approximately 1050 MW. After thermal energy conversion and subtraction of various power drains, the net electrical power is approximately 250 MWe, with a net plant efficiency of 24%.

The insulating layers in the trunk can be eliminated, if all TF coils are operated in parallel with just one turn per coil. For the same plasma parameters, the thickness of the copper trunk can then be increased by several tens of cm (into the "shielding" region), with consequent reductions in Joule heating and stress. However, the coil current would then be of the order of 4 MA, compared with 100 kA for the design of Table 2.

5. CONCLUSIONS

The tokamak design proposed herein makes use of an initial beam-induced current to assist the equilibrium-field system in establishing the large plasma current required for a comfortably ignited plasma. With elimination of the centrally located current transformer, it is possible to design a copper-coil ignition test reactor with small major radius (~ 2.5 m) and very small aspect ratio ($\lesssim 2$), thus allowing a plasma equilibrium with $\bar{\beta} \sim 0.1$. The SMARTOR design, however, is not readily applicable to an economically practical tokamak reactor, particularly one with superconducting TF coils, although some concepts in this direction may be worthy of evaluation.

The advantages of the SMARTOR ignition test reactor 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 very small aspect ratio permits a large plasma beta, and consequently a considerable reduction in the magnetic field.

(4) The small major radius results in significant cost savings.

(5) 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 if the higher beta option ($\bar{\beta} = 0.10$, $B_t = 3.8T$) proves feasible.

(6) The toroidal electric field is large only during production of the initial 150-kA current, so that it is more difficult for a large population of runaway electrons to be formed.

The principal disadvantages of the SMARTOR design are the following:

(1) A second set of neutral-beam injectors is required for establishing the beam-induced current (and minimizing the resistive volt-seconds requirement).

(2) The injection direction of the current-inducing beams may have to accommodate the changing position of the plasma major radius.

(3) For a given magnetic field at the trunk, the field at the plasma is lower than in conventional tokamak designs.

Two important questions which require detailed evaluation are (1) the actual plasma current induced by the equilibrium-field coils, and (2) whether the vertical field configuration required for moderate plasma elongation can be obtained with all the poloidal-field coils located outside the TF coils. Finally, it must be verified in present beam-injection experiments that a large beam-induced plasma current with acceptable radial current-density profile can be established.

ACKNOWLEDGEMENT

The calculations of plasma current start-up were performed by L. Bromberg. This work was supported by U. S. Energy Research and Development Administration Contract EY-76-C-02-3073.

TABLE 1. REFERENCE DESIGN PARAMETERS

	Higher Beta Low Stress	Lower Beta High Stress
C_{beta}	0.20	0.12
σ (kg/cm^2)	1000	1800
R_o (m)	2.60	2.60
a_p (m)	1.35	1.35
b/a	1.6	1.2
$\bar{\beta}$	0.10	0.06
J_c (kA/cm^2)	2.0	2.6
a_c (m)	0.90	0.91
B_m (T)	11.1	14.8
B_t (T)	3.8	5.2
I_p (MA)	6	6
$\bar{T}_e = \bar{T}_i$ (keV)	8.0	8.0
n_e (cm^{-3})	2.3×10^{14}	2.6×10^{14}
$\bar{P}_f$ (MW/m^3)	4.4	5.6
Fusion power (MW)	525	525
Ave. neutron power loading (MW/m^2)	2.1	2.2

M = 3.5

TABLE 2. TOROIDAL-FIELD COIL PARAMETERS

	SMARTOR*	PDX	DOUB.III	TFTR
Internal bore (m)	3.6×5.6	1.7×3.5	2.1×4.0	2.8
Major radius of coil (m)	2.7	1.4	1.4	2.8
Ampere turns (MA)	50	18	28	64
No. of coils	16	20	24	20
Turns per coil	32	20	8	38
Coil current (kA)	98	44	195	85
Average current density (kA/cm^2)	2.1	2.8	2.4	2.6
Radial thickness of copper (m)	0.75	0.20	0.35	0.44
Field at major radius (T)	3.7	2.5	4.0	4.6
Max. field at windings (T)	11.1	6.2	9.3	9.5
Max. stress (kpsi)	14**	14	18	26
Stored energy (GJ)	2.4	0.15	0.46	1.4
Resistive power (MW)	240	71	130	360

* Beta = 0.10 option

** Tensile stress

TABLE 3. PLASMA PARAMETERS DURING A 10-s PULSE*
(see Fig. 6)

TIME (seconds)	0.01	0.05	2.0	4.0	10.0
I_p (MA)	0	0.15	3.5	6.7	6.6
R_o (m)	3.3	3.3	2.7	2.35	2.35
a_p (m)	0.2	0.40	1.05	1.35	1.35
T_e (keV)	10^{-3}	0.1	1.7	12.7	12.7
T_i (keV)	10^{-4}	0.05	2.2	12.6	10.6
n_e (10^{13} cm^{-3})	0.1	2.2	4.2	18	20
β_p	-	1.7	0.25	2.0	2.1

* Current-inducing beams initiated at t = 0.05 s.

Heating beams initiated at t = 2.0 s.

Ignition attained by t = 4.0 s.

(The parameters in this table are not the reference design values of Table 1.)

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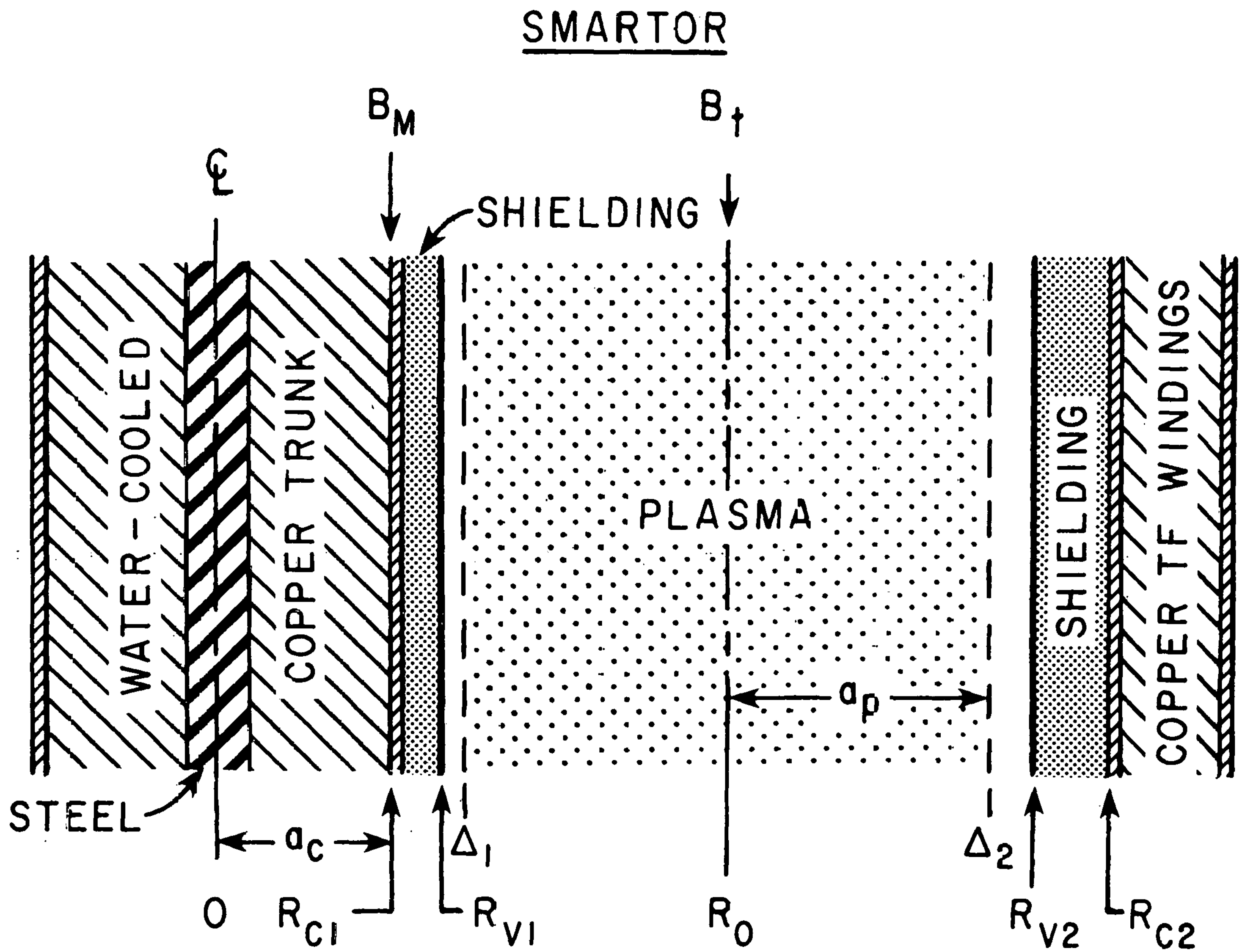

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Fig. 1. Schematic layout of SMARTOR in the horizontal midplane.

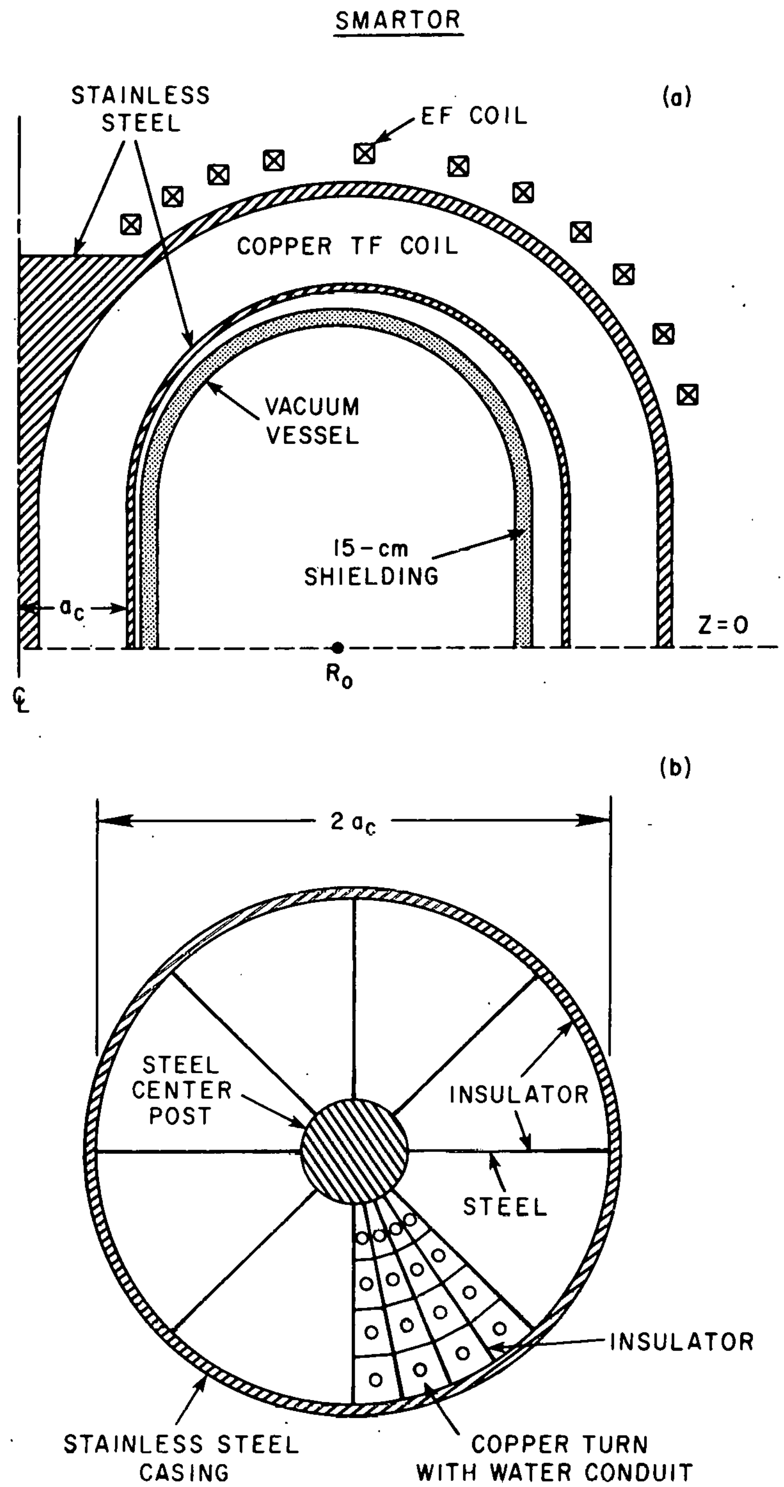

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Fig. 2. (a) Schematic layout of the SMARTOR magnetics system. (b) Plan view of the trunk of SMARTOR. (In practice there would be 16 coils, with 32 turns per coil.)

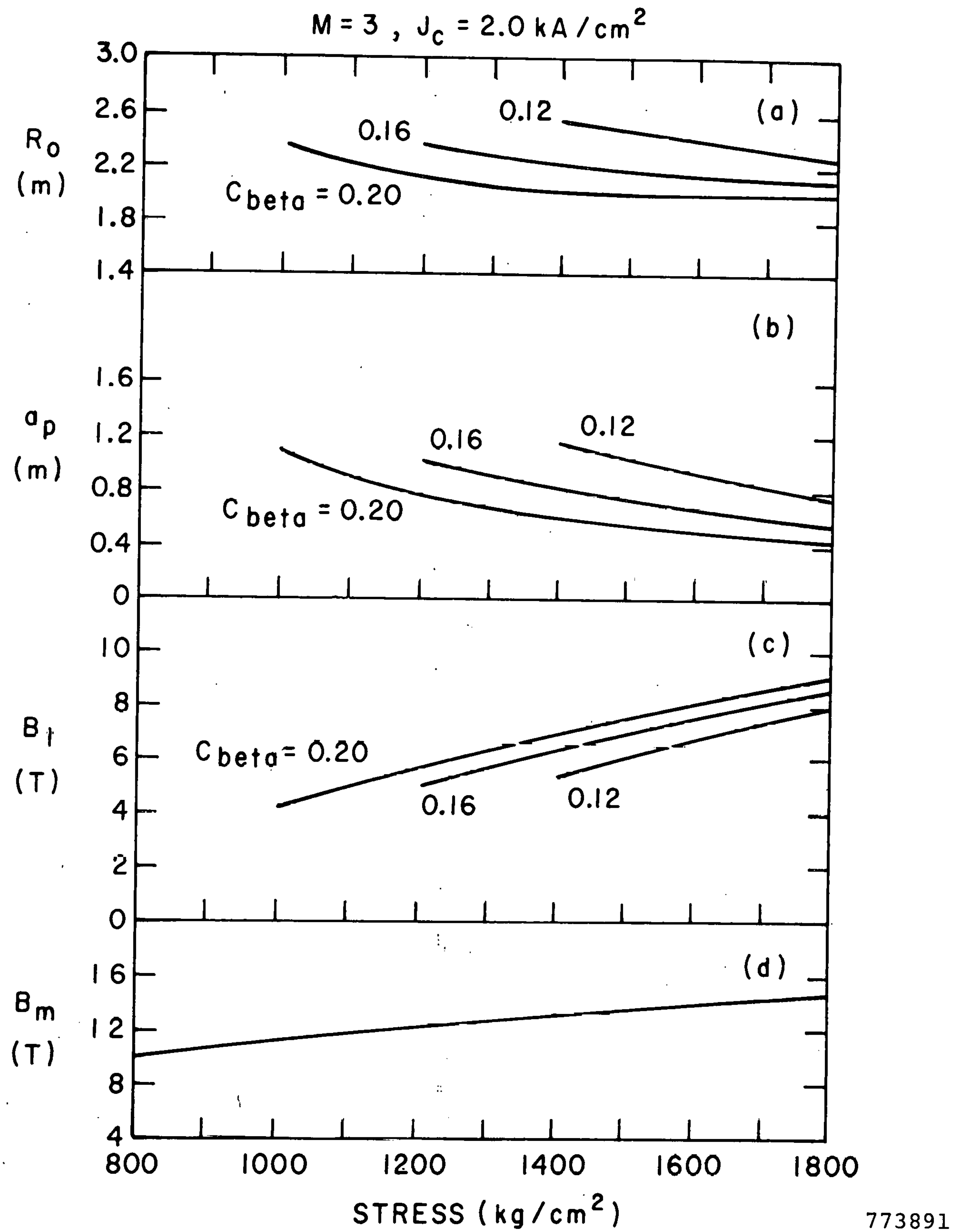

Fig. 3. Variation with tensile stress of (a) major radius, (b) plasma radius, (c) magnetic field at the plasma, and (d) maximum magnetic field at the trunk for a pure-tension coil design. The plasma parameters give thermonuclear ignition when $n\tau_E$ is degraded by a factor of 3 from "empirical scaling", for $\bar{T}_i = 8$ keV. Beta = $C_{beta}a_p/R_o$.

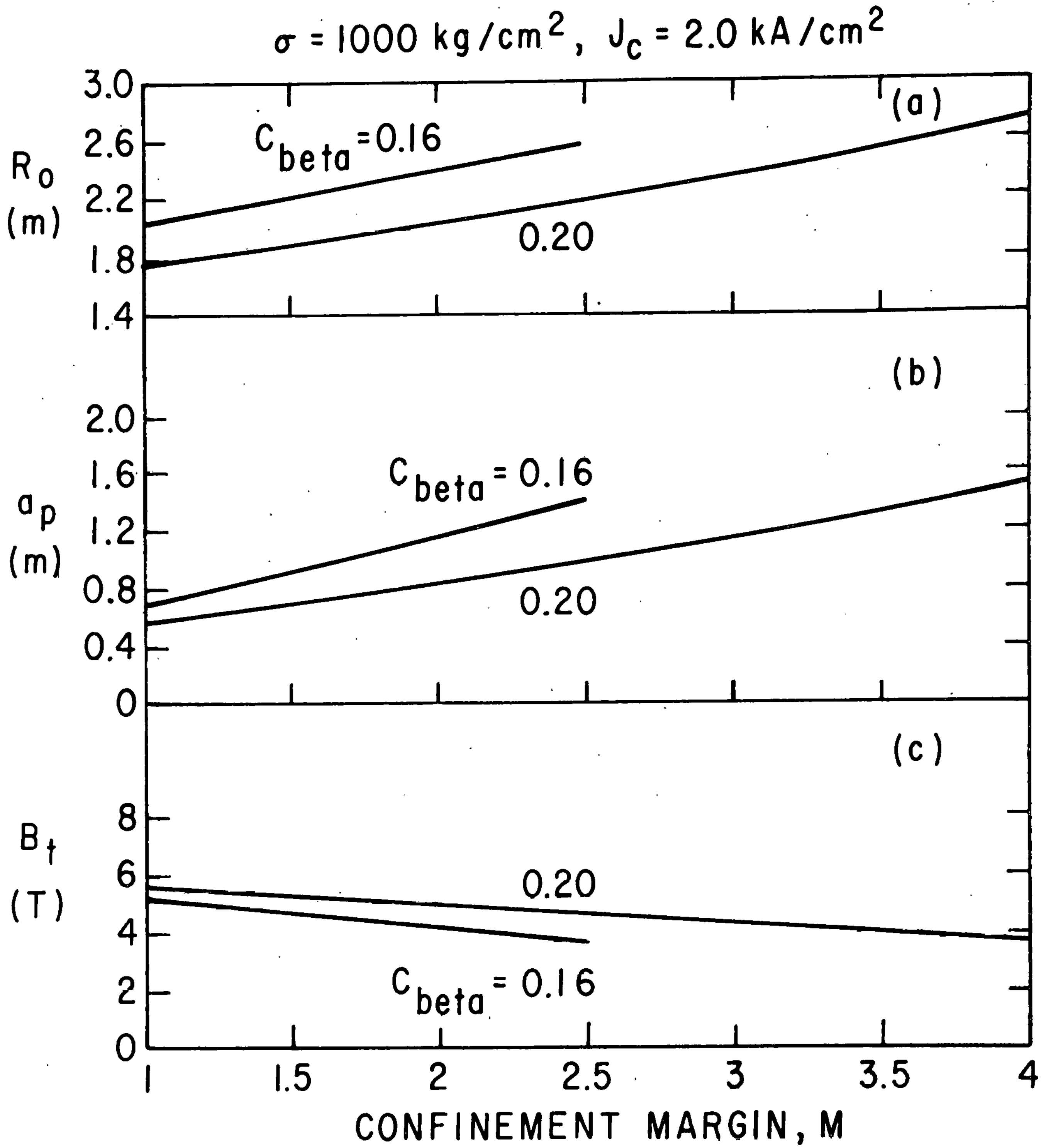

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Fig. 4. Required (a) major radius, (b) plasma radius, and (c) magnetic field at the plasma for thermonuclear ignition, when the tensile stress in the coil is 1000 kg/cm^2 and B_m = 11 T. M is the degradation in $n\tau_E$ from the value given by "empirical scaling". Beta = $C_{beta} a_p / R_o$.

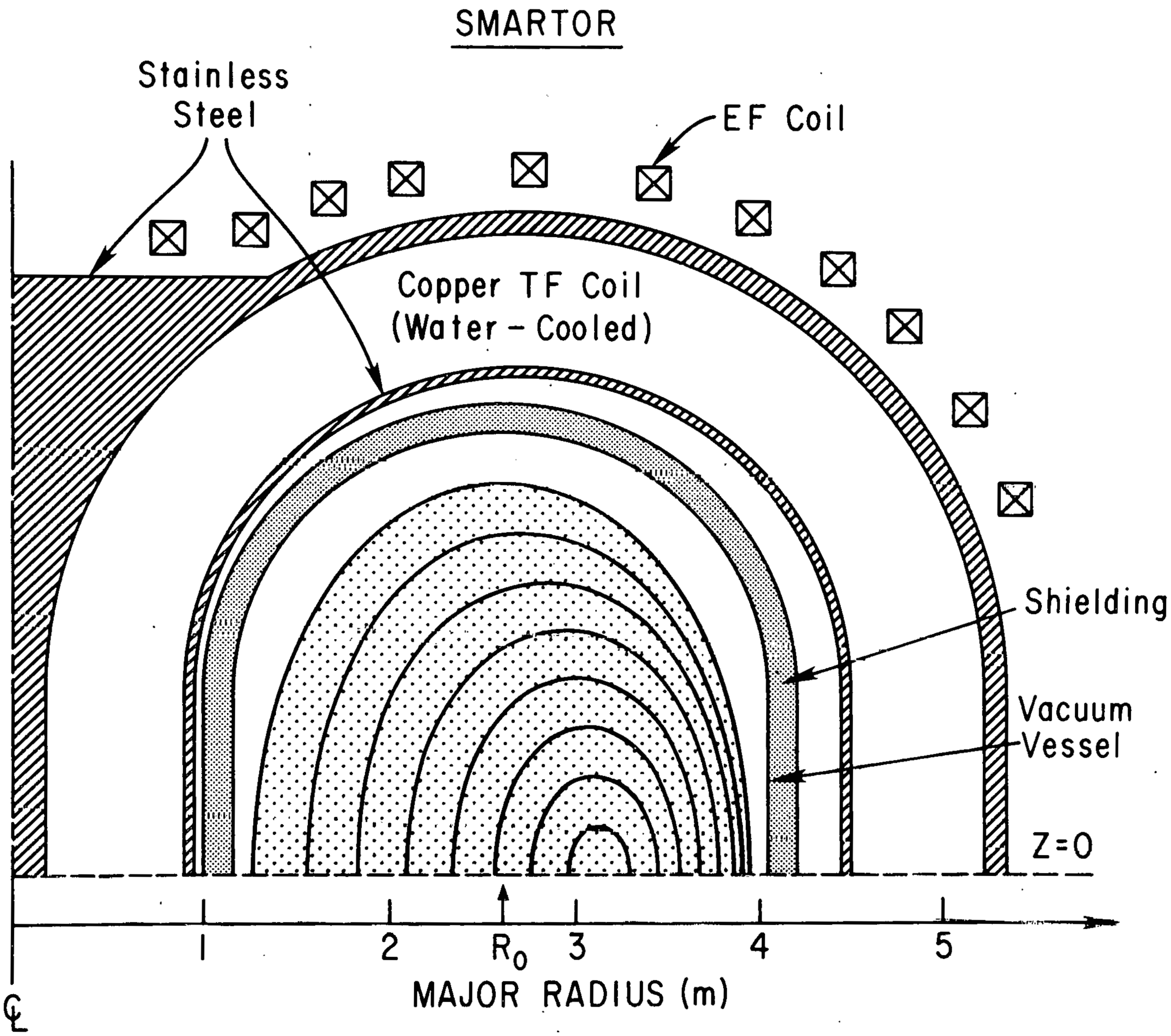

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Fig. 5. Elevation view of SMARTOR with racetrack coil for higher-beta option. EF coil positions are schematic only. B_m = 11 T at R = 0.9 m. Magnetic flux surfaces in the plasma are indicated for b/a = 1.6.

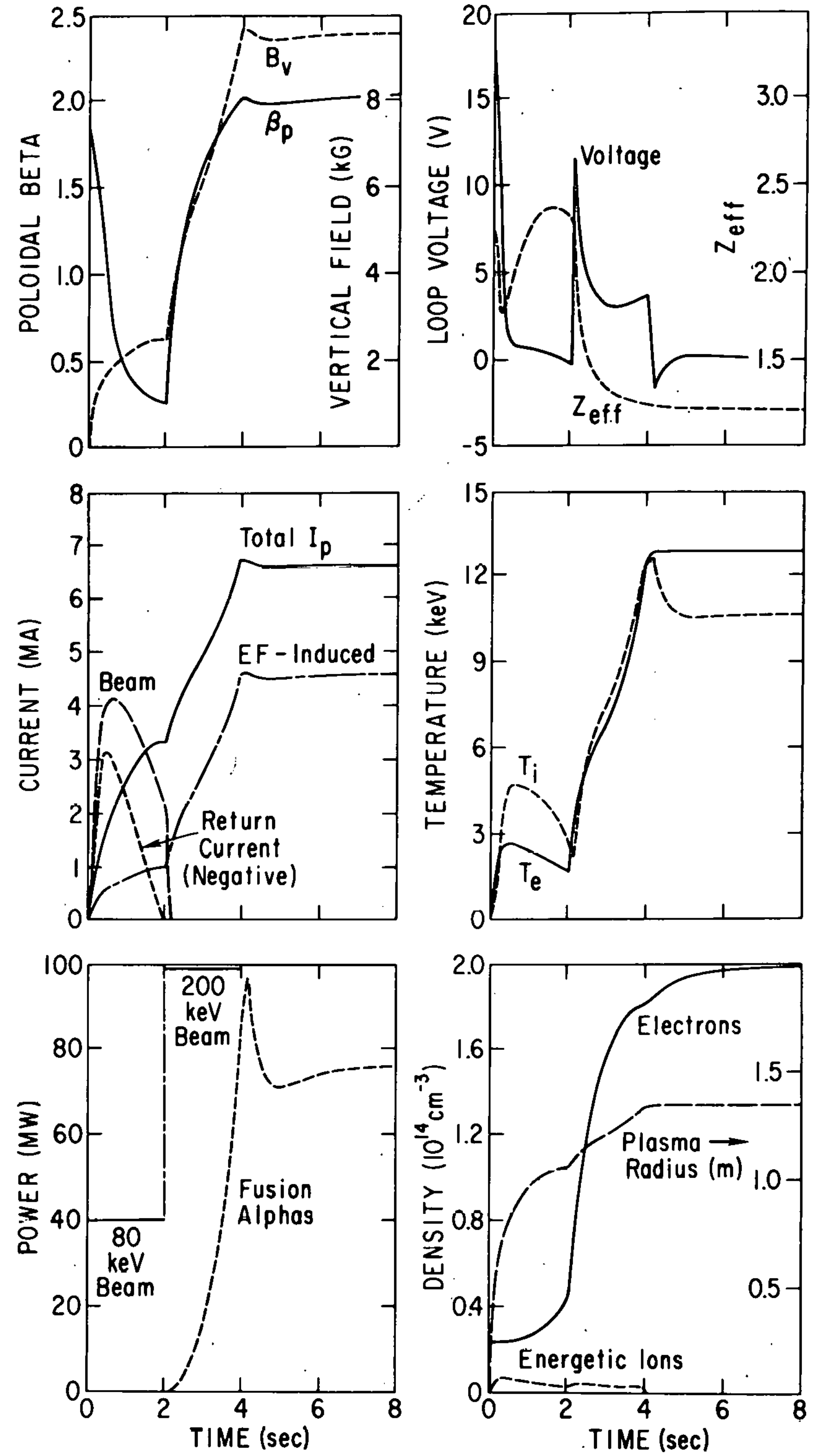

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Fig. 6. Illustrative time evolution of plasma parameters during start-up. Neoclassical skin resistance is enhanced by a factor of 2. [This calculation is due to L. Bromberg.]

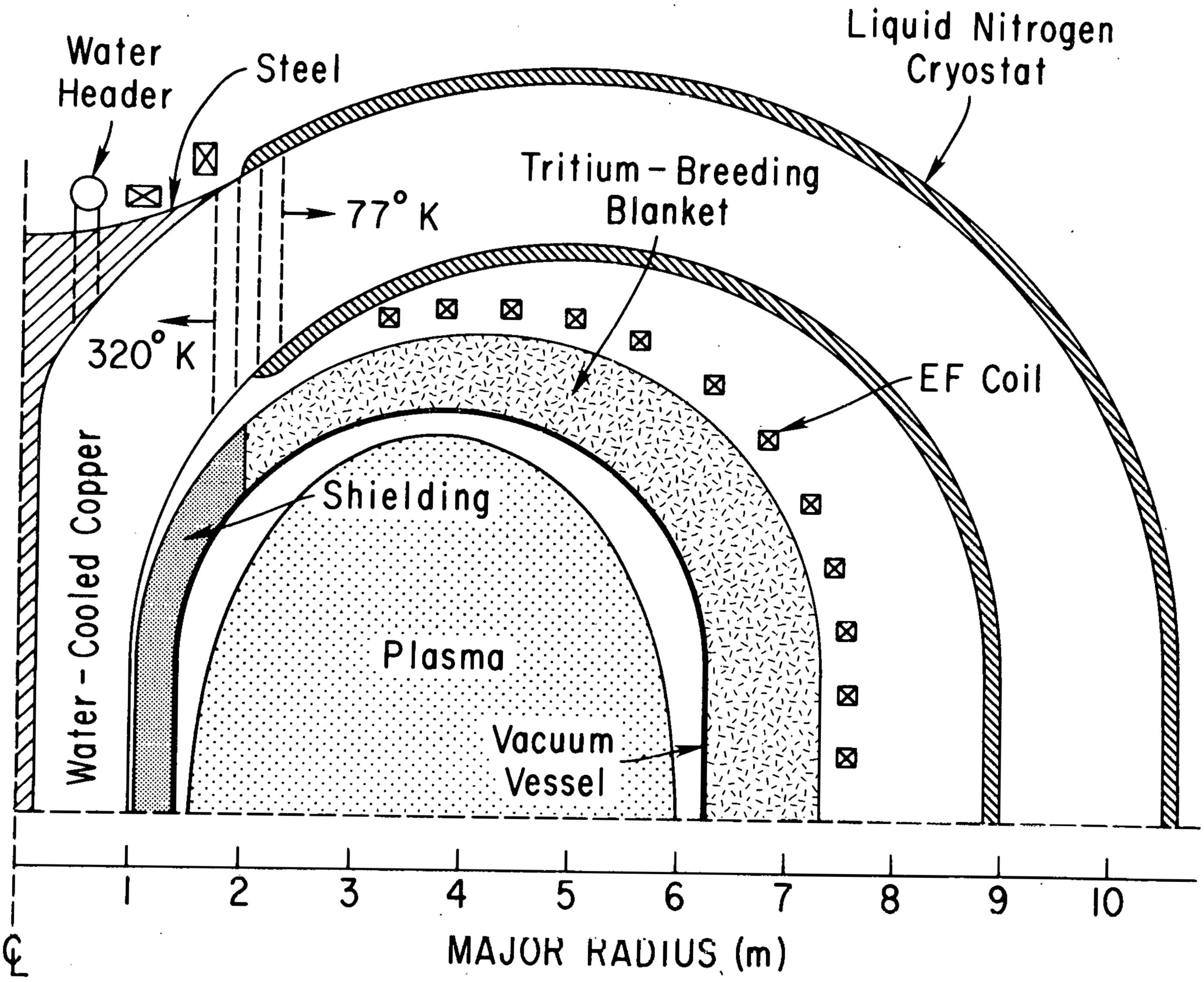

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Fig. 7. Possible geometry of commercial SMARTOR reactor with aspect ratio = 1.68 and partially cryogenic TF coils. EF coil positions are schematic only. B_m = 11 T at R = 1.0 m.