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DESIGN CONSTRAINTS FOR RF-DRIVEN STEADY-STATE TOKAMAK REACTORS

by

David A. Ehst

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DESIGN CONSTRAINTS FOR RF-DRIVEN STEADY-STATE TOKAMAK REACTORS

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ABSTRACT

Plasma current density profiles are computed due to electron Landau damping of lower hybrid waves launched into model tokamak density and temperature profiles. The total current and current profile shape are chosen consistent with magnetohydrodynamic equilibrium for a variety of temperature and density distributions and plasma beta values. Surface current equilibria appear attractive and are accessible to waves with n_2 as low as 1.2. By suitably choosing the spectrum location and width it is possible to drive the 9.8 MA current of a 7.0-m reactor with as little as 2.8% of the fusion power recirculated as rf input from the waveguides.

1. INTRODUCTION

Steady-state fusion reactors are a desirable goal in view of the large demands on materials and components in a pulsed reactor and in view of the significant amounts of energy (~ 10 GJ) that must be stored and transferred in pulsed devices. This report is a study of steady-state tokamak reactors in which radio-frequency (rf) wave sources essentially replace the conventional ohmic-heating transformer windings. The rf waves thus serve two primary functions: the plasma is heated to ignition, and the plasma is contained by the steady-state toroidal current generated by the rf waves. Radio-frequency-driven tokamaks may offer distinct economic advantages over alternatives such as neutral beam-driven reactors since rf sources operate at high efficiencies and may be remotely located relative to the reactor containment structure. The emphasis of this study is to outline the constraints and benefits associated with rf-driven reactors. While the result is by no means an optimized design for such a reactor, the study does suggest such reactors can be competitive with more conventional tokamak reactors.

2. THEORY OF LOWER HYBRID WAVES IN A TOKAMAK PLASMA

2.1 The Lallia-Brambilla grill

An array of waveguides has been shown to be an effective structure for radiating lower hybrid waves into tokamaks [1-3]. The guides are oriented with the electric field vector parallel to the toroidal magnetic field, and, if each guide is phased $\pi/2$ relative to its neighbor, a travelling wave is launched which carries momentum in the toroidal direction. The waveguide dimensions determine the parallel wavelength while the number of guides in the array influences the spectral width.

Figure 1 displays a typical spectrum in the near-vacuum region at the waveguide opening; the spectral intensity is given as a function of the parallel index of refraction, $n_z = ck_z/\omega$, where c is the speed of light, $2\pi/k_z$ is the wavelength in the toroidal direction, and $\omega/2\pi$ is the wave frequency. (We assume the tokamak's poloidal magnetic field, B_p , is small compared to the toroidal component, B_t , such that the electron guiding centers essentially travel in the toroidal direction.) Also shown in the figure is a rectangular approximation to the actual spectrum which indicates a many-wave spectrum with parallel phase velocities in the range

$$c/n_{z1} < \omega/k_z < c/n_{z2}.$$

2.2 Quasi-linear theory of current generation

A narrow rectangular spectrum of waves has a parallel electric field E_z , where

$$E_z^2 \approx \int_{-\infty}^{\infty} dk_z k_z^2 |\phi_{kz}|^2$$

$$\approx \Delta k_z k_{z1}^2 |\phi_{kz}|^2.$$

Here we set $\Delta k_z = k_{z1} - k_{z2}$, $k_{z1,2} = \omega n_{z1,2}/c$, and assumed $\Delta k_z \ll k_{z1}$. The spectral intensity is thus

$$|\phi_{kz}|^2 = \begin{cases} E_z^2 / (k_{z1}^2 \Delta k_z), & k_{z2} < k_z < k_{z1} \\ 0, & \text{elsewhere.} \end{cases}$$

According to quasi-linear theory the resonant electrons (those with parallel velocities in the range of parallel wave-phase velocities) will be randomly

accelerated and decelerated by the E_z field, and the distribution of resonant electrons will diffuse in parallel velocity, v . The diffusion coefficient is [4]

$$D^{QL} = \frac{e^2}{m_e^2} \int_{-\infty}^{\infty} d\left(\frac{k_z v}{\omega}\right) \frac{k_z^2}{v} |\phi_{kz}|^2 \pi \delta\left(\frac{k_z v}{\omega} - 1\right),$$

where e and m_e are the electron charge and mass, respectively. Substituting the expression for the spectral intensity,

$$D^{QL} = \begin{cases} \frac{e^2}{m^2} \frac{E_z^2}{\Delta k_z} \frac{\pi}{\omega/k_z}, & \omega/k_{z1} < v < \omega/k_{z2} \\ 0, & \text{elsewhere,} \end{cases}$$

where k_z is averaged over the (narrow) spectrum.

Although the wave electric field tends to diffuse the resonant electrons in v , collisions try to impose a Maxwell-Boltzmann distribution upon the electrons. This competition is approximately described [5,6] by the following kinetic equation for the electron distribution function $f(v)$:

$$\frac{\partial f}{\partial \tau} = \frac{\partial}{\partial w} D \frac{\partial f}{\partial w} + \frac{\partial}{\partial w} \left(\frac{1}{w^3} \frac{\partial f}{\partial w} + \frac{f}{w^2} \right),$$

where

$$D = D^{QL}/v_T^2 v_0,$$

$$v_T = \sqrt{T_e/m_e},$$

$$w = v/v_T,$$

$$\tau = v_0 t, \text{ and}$$

$$v_0 = (Z_i + 2)n_e e^4 \ln \Lambda / 4\pi\epsilon_0^2 m_e^2 v_T^3 .$$

Here T_e , m_e , and n_e are the electron temperature, mass, and density, $\ln \Lambda$ is the Coulomb logarithm, ϵ_0 is the permittivity of free space, and Z_i is the charge of the ion species in the plasma. This differential equation has been averaged over velocity perpendicular to the magnetic field and is zero dimensional in real space. In the steady state its solution is

$$f = N \left(\exp - \int \frac{w dw}{1 + w^3 D} \right) . \quad (1)$$

The coefficient N must vary with w to keep f continuous at the boundaries of the resonant region (where D is discontinuous).

Figure 2 illustrates the electron distribution function when $w^3 D \gg 1$, for which case the resonant region, $w_1 \equiv \omega / (k_{z1} v_T) < w < w_2 \equiv \omega / (k_{z2} v_T)$, is a nearly flat plateau. As the wave electric field is reduced the value of D becomes small, and collisions effectively erode the plateau in phase space. In the limit $w^3 D \ll 1$, Eq. (1) becomes the Maxwell-Boltzmann distribution. The momentum carried by the electrons is obtained by taking the first moment of f , and the plasma's parallel current density is

$$j = en_e \int_{-\infty}^{\infty} dv v f(v) . \quad (2)$$

Neglecting the small contribution to Eq. (2) from the regions $w < -w_1$ and $w_2 < w$ and assuming the wave electric fields are so large that the plateau in f has zero slope, the current density is approximately

$$\begin{aligned}
j &\approx 2.1 \times 10^{-12} n_e \sqrt{T_e} f(w_1) \left(\frac{w_2^2 - w_1^2}{2} \right) \\
&\approx 2.1 \times 10^{-10} \frac{n_e}{\sqrt{T_e}} \exp\left(-\frac{255.9}{n_{z1}^2 T_e}\right) \left(\frac{1}{n_{z2}^2} - \frac{1}{n_{z1}^2} \right). \quad (3)
\end{aligned}$$

In these formulas j is in A/m² when n_e is in m⁻³ and T_e is in keV;
 $f(w_1) \approx \exp(-w_1^2/2)/\sqrt{2\pi}$.

In the steady state the power transferred by the wave to the resonant electrons is passed on to other particles via collisions. Hence, denoting the collision operator on f by $C(f)$, energy density is dissipated at a rate

$$\begin{aligned}
P_{rf} &= - \int_{-\infty}^{\infty} dv K \frac{n_e m_e v^2}{2} C(f) \\
&= \int_{-\infty}^{\infty} dw K \frac{n_e m_e v_0^2}{2} w^2 \frac{d}{dw} D \frac{df}{dw} \\
&= 1.7 \times 10^{-31} K \frac{n_e^2}{\sqrt{T_e}} \ln \Lambda f(w_1) \ln \left(\frac{1 + w_2^3 D}{1 + w_1^3 D} \right)^{1/3}. \quad (4)
\end{aligned}$$

When $w_1^3 D \gg 1$ (large electric fields) and for $\ln \Lambda \approx 18$, this expression is approximately

$$P_{rf} \approx 1.2 \times 10^{-30} K \frac{n_e}{\sqrt{T_e}} \exp\left(-\frac{255.9}{n_{z1}^2 T_e}\right) \ln \left(\frac{n_{z1}}{n_{z2}} \right) \quad (5)$$

All units are MKS and keV. K is a correction factor which is obtained from a more realistic two-velocity dimension analysis [7]. In Ref. 7 the authors numerically solved the kinetic equation with quasilinear and Fokker-Planck

operators and found $K = 0.392$. This value will be used in the remainder of our work.

In designing tokamak reactors the local current density must be consistent with MHD equilibrium requirements, and the goal of achieving economical wave-driven currents is essentially the minimization of the power dissipated in doing this. Thus, a useful parameter for study is the local ratio p_{rf}/j , which is approximately given by Eqs. (3) and (5):

$$\frac{p_{rf}}{j} = 2.2 \times 10^{-21} n_e \frac{\ln(n_{z1}/n_{z2})}{(n_{z2}^{-2} - n_{z1}^{-2})} . \quad (6)$$

This formula points to the important principle that wave-driven reactors be designed to have the highest current density in regions of low-particle density; the ratio p_{rf}/j is independent of temperature. In reactor analysis three strategies will generally be considered in order to drive currents in low-density plasma regions. First, the average density may be lowered while increasing the average temperature, keeping the product $\overline{n_e T_e}$ constant (the bar denotes a spatial average). Since the plasma beta is proportional to $\overline{n_e T_e}$ this strategy will reduce the rf power while keeping the fusion power density relatively constant. Second, the density profile might be varied. As an example, for a given average density a broad, flat profile tends to reduce the central density, which is advantageous if the equilibrium requires a centrally peaked current density. Third, equilibria can be studied which require currents in low-density regions, e.g., surface currents. Finally, Eq. (6) shows p_{rf}/j is influenced by the spectrum location and width. To proceed it becomes necessary to explicitly treat plasma profile effects on the total rf power and current.

2.3 Spatial gradients

We use a cylindrical model of tokamak density and temperature profiles given by

$$n_e(r) = n_{e0} [1 - r^2/a^2]^{\alpha_n} \quad (7a)$$

and

$$T_e(r) = T_{e0} [1 - r^2/a^2]^{\alpha_T}, \quad (7b)$$

with r being the distance from the minor axis and a the minor radius. The total current and power dissipated in the torus are

$$I = \int_{r_c}^a j(r) 2\pi r dr \quad (8)$$

and

$$P_{rf} = \int_{r_c}^a p_{rf}(r) 4\pi^2 R r dr, \quad (9)$$

where r_c is the radius of furthest wave penetration and R is the major radius.

Using the approximations (3) and (5) and the profiles given by (7a) and (7b)

these integrals may be evaluated analytically:

$$I = 6.8 \times 10^{-10} a^2 n_{e0} T_{e0}^{-1/2} (n_{z2}^{-2} - n_{z1}^{-2}) \alpha_T^{-1} b^{-k} \Gamma(k, d)$$

and

$$P_{rf} = 9.2 \times 10^{-30} R a^2 n_{e0} T_{e0}^{-1/2} \ln(n_{z1}/n_{z2}) \alpha_T^{-1} b^{-h} \Gamma(h, d),$$

where

$$b = 255.9 n_{z1}^{-2} T_{e0} , \quad d = b \left(1 - r_c^2/a^2\right)^{-\alpha_T} ,$$

$$k = 1/2 - \alpha_T^{-1} - \alpha_n/\alpha_T , \quad h = k - \alpha_n/\alpha_T .$$

The ratio of total rf power to current is thus

$$\frac{P_{rf}}{I} = 1.3 \times 10^{-20} R n_{e0} \left(1 - r_c^2/a^2\right)^{\alpha_n/\alpha_T} \frac{\ln(n_{z1}/n_{z2})}{(n_{z2}^{-2} - n_{z1}^{-2})} G(k, h, d) , \quad (10)$$

where G is approximated by a ratio of continued fractions,

$$G = \frac{\left(\frac{1}{d+} \frac{1-h}{1+} \frac{1}{d+} \frac{2-h}{1+} \frac{2}{d+} \dots \right)}{\left(\frac{1}{d+} \frac{1-k}{1+} \frac{1}{d+} \frac{2-k}{1+} \frac{2}{d+} \dots \right)} .$$

As an illustration, consider a wave which penetrates half the distance to the center of a nearly flat ($\alpha_n = 0.3$) density profile with $n_{z2} = 1.45$, $n_{z1} = 1.55$, $n_{e0} = 9.15 \times 10^{20} \text{ m}^{-3}$, $R = 9.11 \text{ m}$, and $I = 12.2 \text{ MA}$. This formula predicts $P_{rf} \approx 147 \text{ MW}$. Compared to a fusion power output of 2500 MW this represents about 6% of the output which must be recycled as rf input power to maintain a steady state. The form of Eq. (10) emphasizes the importance of having low values of n_{e0} . Assuming β_t is fixed this corresponds to high temperature operation. We define $\beta_t = \sum_j 2\mu_0 n_j T_j / B_0^2$, where B_0 is the toroidal field at the major radius and where the bar denotes a spatial average. In the approximate geometry of concentric circular flux surfaces with equal ion and electron densities and temperature our pressure parameterization yields

$$B_0^2 \beta_t / 2\mu_0 = 2n_{e0} T_{e0} (1 + \alpha)^{-1} ,$$

and

$$\frac{P_{rf}}{I} = 17R B_0^2 \beta_t \frac{(1 + \alpha)}{T_{e0}} \left(1 - r_c^2/a^2\right)^{\alpha n/\alpha_T} \frac{\ln(n_{z1}/n_{z2})}{(n_{z2}^{-2} - n_{z1}^{-2})} G. \quad (11)$$

To improve the accuracy of our results the electric field amplitude of lower hybrid waves was followed numerically in tokamak-like plasmas. In cylindrical coordinates we assume a uniform wave intensity at the surface and solve

$$p_{rf} = -\frac{1}{r} \frac{d}{dr} r \left(v_{gr} \omega \frac{\partial \epsilon_r}{\partial \omega} \frac{E^2}{2} \right),$$

where v_{gr} is the radial group velocity and the remaining factor in parentheses is the wave energy density. For lower hybrid waves [8,9] the real part of the dielectric function is

$$\epsilon_r = \epsilon_0 \left[\frac{k_z^2}{k^2} \left(1 - \frac{\omega_{pe}^2}{\omega^2} \right) + \frac{k_r^2}{k^2} \left(1 - \frac{\omega_{pe}^2}{\omega^2 - \Omega_e^2} - \frac{\omega_{pi}^2}{\omega^2 - \Omega_i^2} \right) \right],$$

The dispersion relation ($\epsilon_r = 0$) is

$$\omega = \omega_{LH} \left(1 + \frac{m_i}{m_e} \frac{k_z^2}{k^2} \right)^{1/2},$$

and

$$v_{gr} = \frac{\partial \omega}{\partial k_r} = \frac{(\omega_{LH}^2 - \omega^2) k_r}{\omega k^2}.$$

This is a nonlinear differential equation since p_{rf} , given by (4), is a function of E . In solving this the cyclotron frequencies, Ω_e and Ω_i , are assumed constant, the plasma frequencies, ω_{pe} and ω_{pi} , and lower hybrid frequency,

$\omega_{\text{LH}} = (\omega_{\text{pi}}^{-2} + \Omega_e^{-1} \Omega_i^{-1})^{-1/2}$, vary with r according to (7), and the radial wave number, k_r , is an additional function of r , given by the dispersion relation with $k^2 = k_r^2 + k_z^2$. Both ω and k_z are constants; the ion mass, m_i , is set at 2.5 times the proton mass to simulate a DT plasma. From the solution to this equation, $E(r)$, the diffusion coefficient, $D(r)$, and distribution function, (1), are known as functions of radius. Consequently the current and power density profiles are found from (2) and (4), and the total current and dissipated power are known from (8) and (9).

3. MHD EQUILIBRIUM

A constraint on reactor design is that the current density and pressure profiles be consistent with MHD equilibrium. A number of equilibria have been computed by solving the Grad-Shafranov equation for the poloidal flux function, $\psi(\rho, h)$ [10,11]:

$$\frac{\partial^2 \psi}{\partial \rho^2} - \frac{1}{\rho} \frac{\partial \psi}{\partial \rho} + \frac{\partial^2 \psi}{\partial h^2} = -4\pi^2 \left(\mu_0 \rho^2 \frac{dp}{d\psi} + \frac{1}{2} \frac{dF^2}{d\psi} \right). \quad (12)$$

We have defined ρ as the distance from the symmetry axis and h as the distance from the equatorial plane; a scalar pressure, $p(\psi) = \sum_j n_j(\psi) T_j(\psi)$, is assumed with the density and temperature being constant on flux surfaces, $\psi(\rho, h) = \text{constant}$. We require $\psi = \text{constant}$ on the boundary

$$\begin{cases} h = \kappa a \sin t \\ \rho = R + a \cos(t + d \sin t) \end{cases},$$

where $0 \leq t \leq 2\pi$. The current and pressure profiles, plasma beta, and total current in equilibrium depend on the pressure and diamagnetic function, $F^2(\psi)$, specified in (12). We set $p(\psi) = p_0 \hat{\psi}^\alpha = \sum_j n_{j0} \hat{\psi}^\alpha n_{j0} \hat{\psi}^\alpha T_j$, where

$\hat{\psi} = (\psi_{\text{limiter}} - \psi) / (\psi_{\text{limiter}} - \psi_{\text{magnetic axis}})$; near the magnetic axis these density and temperature profiles have the approximate form given by (7a) and (7b) if $\kappa = 1$ and $R/a \gg 1$. Note $\alpha = \alpha_n + \alpha_T$.

Figure 2 displays the current density j and pressure p in the equatorial plane of a circular plasma ($\kappa = 1$, $d = 0$) with an aspect ratio $A = R/a = 3$, $\alpha = 1.4$, $\beta_t = 0.06$, and safety factor, $q(\psi)$, varying between unity at the magnetic axis and 3.0 at the boundary. This particular equilibrium results from choosing

$$F^2(\psi) = F_0^2 (1 - 0.097 \hat{\psi}^{1.7}),$$

where F_0 is the product of ρ and the toroidal vacuum magnetic field. This equilibrium is typical of those with what we shall call centrally peaked densities; although $j(\rho)$ is asymmetric due to toroidal effects, the current peaks at only one interior location.

By setting $F^2 = F_0^2 (1 - 0.299 \hat{\psi}^{2.9})$ and $\alpha = 2.0$ the equilibrium in Fig. 3 results. This particular case is for $A = 3.0$, $\kappa = 1.3$, and $d = 0.25$. The non-circular plasma shape may be conducive to higher beta stability [12-16]; if, for example, $q(\psi)$ varies between 1.3 and 3.9 then $\beta_t = 0.07$ while if q varies between 1.0 and 3.0 then $\beta_t = 0.12$. This current profile is hollow in the center and is typical of what may be termed a surface current equilibrium.

Even higher beta equilibria exist if "force-free" paramagnetic currents flow on the plasma surface [17,18]. Figure 4 is an example, taken from Ref. 17, with $A = 2.5$, $\kappa = 1.0$, $d = 0.0$, $\alpha = 2.0$, and

$F^2 = F_0^2 [1 - C_1 \hat{\psi}^{2.0} + C_2 (\hat{\psi}^{1.9} - \hat{\psi}^{2.0})]$, where $C_2 \approx 20$. This example has $\beta_t = 0.16$ when $q(\psi) \approx q(0) = 1.0$. We note that peak betas of twenty per cent have been experimentally demonstrated in SPICA [19], a torus with $A = 3.0$, $\kappa = 1.0$, $d = 0.0$, and $q(\psi) \approx q(a) = 1.7$. While the force-free surface currents

in the experiment are obtained by implosive heating of the plasma, the intriguing possibility exists that these currents could be driven in a steady-state by rf wave damping.

In our examination of rf-driven tokamaks both centrally peaked and surface currents will be considered, and in each case the correct total current, I , for equilibrium will be computed from Eq. (12). Likewise, the appropriate flux surface geometry, $\psi(\rho, h)$, will be used to create two-dimensional density and temperature profiles from which the fusion power production will be found:

$$P_f = \int_{\psi_{\text{magnetic axis}}}^{\psi_{\text{limiter}}} n_{DT}(\psi)^2 \frac{\langle \sigma v [T_{DT}(\psi)] \rangle}{4} U \frac{dV}{d\psi} d\psi, \quad (13)$$

where dV is the differential toroidal volume, $dV/d\psi$ is the Jacobian between ρ , h , and ψ coordinates, and $U = 17.58$ MeV. The DT density and temperature, n_{DT} and T_{DT} , are computed from the zero-dimensional steady-state particle and power balance equations [11], for which the average electron temperature, impurity content, and profile parameters, α_n and α_T have been specified. The alpha particle density is determined from the fusion rate, and β_t is consistent with the MHD equilibrium. Our subsequent analysis of the required rf power for a given reactor configuration will be approximate to the extent that the profiles given by (7a) and (7b) are in disagreement with the two-dimensional profiles used in (13). However, in the large aspect ratio limit, near the magnetic axis, $\psi \propto r^2$ and the profiles are nearly the same. We stress that the equilibrium I and $\psi(\rho, h)$ are functions of the major radius R and vacuum toroidal magnetic field; we assume a toroidal field strength of 9.0 T at the inner magnet surface and a blanket/shield/vacuum gap spacing of 1.2 m on the inside of the torus.

4. REACTOR DESIGN OPTIONS

4.1 Geometric trends

From Ref. 11 we may approximate

$$P_f \approx C R^3 \kappa A^{-2} \beta_t^2 B_0^4 \left[(1 + \alpha)^2 \alpha^{-3/4} \langle \sigma_v(T_{10}) \rangle T_{10}^{-2} \right] \quad (14)$$

and

$$I \approx C \kappa A^{-2} R B_0 q^{-1}, \quad (15)$$

so

$$\frac{P_f}{I} \approx C R^2 \beta_t^2 B_0^3 q \left[(1 + \alpha)^2 \alpha^{-3/4} \langle \sigma_v(T_{10}) \rangle T_{10}^{-2} \right], \quad (16)$$

with $C \approx 6 \times 10^{23}$.

Consider first the influence of aspect ratio on reactor design, recalling that an rf-driven reactor may require no central solenoid with primary Ohmic heating (OH) coils and thus could be designed with smaller A than conventional tokamaks. Since the maximum stable β_t is thought to scale inversely with some power of A [16] we see from (14) that reactor size, R , is reduced for a fixed P_f and B_0 by going to small A . However, for a fixed maximum toroidal (TF) field at the magnet the value of B_0^4 decreases rapidly upon going to small A : $B_0 = 9.0 \text{ T}(1 - A^{-1} - 1.2 R^{-1})$. Minimum machine sizes occur for $A \approx 2.5$ for typical β_t scalings [11]. In addition, (16) indicates that I tends to large values for fixed P_f as A becomes small. These trends have been numerically confirmed [11] and are illustrated in Fig. 5; based on the figure, $A = 3$ is chosen for the remainder of our investigation, this representing a compromise between reasonable major radius and plasma current.

For a fixed A and P_f and for β_t increasing with κ [16], (14) suggests that R decreases by going to more elongated plasmas. However, the required

plasma current increases rapidly for a fixed P_f once κ exceeds about two [11], as shown in Fig. 6. The optimum κ depends on detailed cost breakdowns for a power plant; our studies focus on values in the range $1.0 \leq \kappa \leq 1.65$.

A critical economic parameter of the rf-driven tokamak reactor is the ratio of rf power to fusion power, which is roughly given by the ratio of (11) to (16):

$$\epsilon \equiv \frac{P_{rf}}{P_f} \approx \frac{28T_{i0}\alpha^{3/4}(1 - r_c^2/a^2)^{\alpha n/\alpha_T} \ln(n_{z1}/n_{z2})G}{RB_0\beta_t q(1 + \alpha)10^{24}\langle\sigma v(T_{i0})\rangle(n_{z2}^{-2} - n_{z1}^{-2})}, \quad (17)$$

where we assumed $T_{e0} = T_{i0}$. In the range of interest to fusion power $\langle\sigma v(T_{i0})\rangle \sim T_{i0}^2$, and we can illustrate strong influences on reactor design by rewriting (17) as

$$\epsilon \propto (RB_0\beta_t T_{i0})^{-1}. \quad (18)$$

This last expression confirms the claim in Sec. 2.2 that higher plasma temperatures are beneficial to reactor economics for fixed β_t . In addition, large R , B_0 , and β_t are desirable, which suggests large, high power reactors are more favorable. Formula (18) is a generalization of the ϵ expression found in Fisch's work [5,6]. Fisch did not obtain the complete temperature dependence since he ignored the temperature variation of $\langle\sigma v\rangle$ in his approximate formula. In addition Fisch adopted a particular scaling law for β_t : $\beta_t \propto q^{-2}k^{-1}$.

4.2 Centrally peaked current profiles

An extensive survey was made of reactors with equilibria typified by Fig. 2 with $A = 3.0$, $\kappa = 1.0$, $d = 0.0$, $\alpha = 1.4$, $q(a) = 3.0$, and $\beta_t = 0.06$. The reactor major radius and plasma current were first computed for a fusion power output $P_f = 5000$ MW and for a variety of average electron temperatures

and density profiles: $\bar{T}_e = 10, 18, 26, 34$ keV and $\alpha_n = 0.3, 0.7, 1.1$. The second step in the survey process was the computation of the current profiles and rf power dissipation in these candidate reactors; by varying the incident wave intensity, spectrum center, and spectrum width it was possible to match the required total plasma current with centrally peaked current densities in each reactor and to minimize the required rf power.

The survey results may be briefly summarized. Increasing $\bar{T}_e$ above 8 keV is beneficial at first because the lower density plasma dissipates less power, as suggested by (18). However, above 20-40 keV the fusion power density starts to decrease so rapidly that the reactor volume must increase, and the larger major radius requires higher total plasma currents. Numerically it is found that ϵ is a minimum for $\bar{T}_e \approx 18$ keV. For all peaked currents with $P_f = 5000$ MW the best value found for ϵ was about 3.8%. The smallest such reactor (with $\alpha_n = 1.1$, $\bar{T}_e = 18$ keV, $B_{\max} = 9$ T) had $R = 9.8$ m. In view of the large major radii required for 5000 MW reactors, the remainder of our investigation concentrated on smaller reactors.

The parameter survey was repeated for the same centrally peaked current equilibria in 2500-MW reactors. $\bar{T}_e \approx 18$ keV was again found optimum, but in this case the best value of ϵ was about 4.2%. This best case (with $\alpha_n = 1.1$, $B_{\max} = 9$ T) requires $R = 8.2$ m, which confirms the trend to larger ϵ at smaller R , suggested by (18). The increase in circulating power, represented by ϵ , may well be offset by the decrease in reactor capital cost at the smaller sizes.

The chief disadvantage of all centrally peaked current reactors is the low index of refraction required for the rf to penetrate to the hot central plasma regions. In order to achieve the minimum values of ϵ , it is necessary to choose the n_z spectrum below about 1.3. In fact the central plasma will be inaccessible to lower hybrid waves with such high parallel phase velocities [5,8]; such

waves penetrate a limited distance and are reflected. This consideration quite naturally leads to the examination of reactor plasmas with surface currents, described in the next section.

4.3 Surface current profiles — reference reactor

A survey of reactors with surface currents (Fig. 3 with $A = 3.0$, $\kappa = 1.3$, $d = 0.25$, $\alpha = 2.0$, $q = 1.3-3.9$, and $\beta_t = 0.07$) was done for a moderate-sized radius, $R = 7.0$ m, and for a maximum toroidal field typical of NbTi superconductor, $B_{\max} = 9.0$ T. The inner blanket/shield thickness was set at 1.0 m [20], and the vacuum scrape-off region was assumed to be 0.2 m, which yields a field on axis of 4.46 T. For the calculations, an impurity content of 0.5% oxygen and 5.0% beryllium is assumed which results in $Z_{\text{eff}} \approx 1.7$.

The survey results for various $\bar{T}_e$ and α_n values are given in Table I. Fusion power density is a maximum for average ion temperatures approximately equal to $13 \text{ keV}/(1 + \alpha_T)$ [11]; for $\alpha_T = 1.1$, 0.7, and 0.3 this corresponds to $\bar{T}_i \approx \bar{T}_e = 6.2 \text{ keV}$, 7.6 keV, and 10.0 keV, respectively. For the higher temperatures in the table the power density and total fusion power accordingly decrease with $\bar{T}_e$. Note likewise that for a fixed $\bar{T}_e$ the low values of α_T bring the power output closest to the maximum. On the premise that fusion outputs less than about 2000 MW are impractical for reactors with $R = 7.0$ m and based on our previous results for centrally peaked currents we will focus on the subset of reactor candidates with $\bar{T}_e = 18 \text{ keV}$.

The plasma cross-sectional area for this mildly noncircular reactor with $R = 7.0$ m, $A = 3.0$, $\kappa = 1.3$, and $d = 0.25$ is $X = 21.9 \text{ m}^2$, and an effective minor radius is used in the cylindrical model for rf current generation:

$$a \equiv \sqrt{X/\pi} = 2.64 \text{ m}.$$

Figure 7 displays the total current and rf power for the reactor with $\bar{T}_e = 18 \text{ keV}$ and $\alpha_n = 1.7$ for a variety of spectra. The percentage depth of

rf penetration is indicated along the curves. Note that as the spectrum narrows deeper penetration occurs with the total I remaining constant. The penalty accompanying the deeper penetration is an increased P_{rf} . The results also indicate that $\log I$ varies linearly with $\log P_{rf}$, the slope depending on α_n . During the start-up phase of reactor operation with a fixed spectrum the total current and depth of penetration increase with P_{rf} . The intersection of the curves with the dashed line at 9.81 MA determines the required P_{rf} and depth of penetration. This information for the reactor with $\alpha_n = 1.7$ is in turn plotted in Fig. 8 for several values of n_{z1} .

The open circles in Fig. 8 correspond to $n_{z2} = 1.2$ and demarcate the approximate accessibility limit for surface penetration. Referring to Fig. 3 we observe the current peaks 34% of the way in on the outer edge of the plasma and 17% on the inner side. Taking the average value, we shall require that rf-driven currents penetrate 25% of the distance into the plasma. Figure 8 shows that the spectrum with $n_{z1} = 1.45$ requires at least 90 MW to generate currents. The spectrum with $n_z = 1.56-1.80$ achieves 25% penetration with only 65 MW. The best case, $n_z = 1.20-1.60$, achieves the required penetration with about 56 MW. From Table I we find the corresponding $\epsilon = 2.8\%$. For the reactors with $\alpha_n = 1.3$ and 0.9 the best results were $P_{rf} = 70$ MW and $P_{rf} = 85$ MW, corresponding respectively to $\epsilon = 4.1\%$ and $\epsilon = 5.9\%$. The parameters of the best reactor candidate are summarized in Table II, and the profiles are shown in Fig. 9. The broad temperature profile seems to be consistent with both interior heating from alpha particles and surface heating from the rf.

Accessibility to lower hybrid waves demands [5] $n_z \geq 1 + \omega_{pe}^2 / \Omega_e^2$, where we compute the local plasma frequency from (7a) and the cyclotron frequency from $B(\rho) = B_0 R / \rho$. This condition is plotted in Fig. 10 for the reference reactor, assuming waves are launched from the outside ($\rho > R$). The figure confirms

that spectra with $n_z \geq 1.2$ will propagate 25% of the distance into the plasma. Accessibility is somewhat easier for waves launched on the inside ($\rho < R$), which is a definite possibility for a reactor design lacking a central solenoid. A more detailed study of accessibility in DT plasmas [21] yields a less optimistic result on this point; waves with $n_z = 1.2$ would require low frequencies ($f = 0.836$ GHz, $\omega = 5.2$ GHz) to penetrate 25% of the distance from the outside. Unfortunately this frequency is below the local lower hybrid frequency ($\omega_{LH} = 8.26$ GHz at $\rho = R + 0.75 a$), a condition which may allow parametric instabilities to grow, preventing lower hybrid wave penetration [22]. To avoid parametric decay we choose to operate at $f = 1.5$ GHz with the consequence that those modes at the low n_z end of the spectrum may not fully penetrate. Referring to Fig. 8 and Ref. 21 we find the whole problem can be avoided by selecting $n_z = 1.55-1.80$, except that in this case $P_{rf} = 65$ MW and $\epsilon = 3.2\%$.

5. REACTOR ENGINEERING

The rf system parameters in Table II are based on a system [23] designed to deliver 50 MW of auxiliary heating power for tokamak startup. Although the system in Ref. 23 has different spectral characteristics ($f = 2.4$ GHz, $n_z \approx 3$), it was planned to handle the same CW power that we are considering and was designed to satisfy the neutronics, materials, and electronics requirements of a tokamak reactor. Note the modest area of waveguide grill required, $A_{WG} = 0.83$ m², compared to the first wall area, $A_{wall} = 792$ m². The rf system was found to cost 75% as much as a 50 MW neutral beam system for auxiliary heating. In addition, the reactor containment building will be considerably smaller for rf compared to neutral beams; as shown in Fig. 11 the waveguides may be easily routed to a convenient external location.

By not needing the Ohmic heating (OH) solenoid in the central region it will be possible to place equilibrium field (EF) coils along the inner leg of the toroidal field (TF) magnet. By situating EF coils in this strategic location highly shaped plasmas may be formed with relatively modest amounts of energy stored in the EF coils [24]. Elimination of the OH coils also allows the installation of more structural support for the TF coils, which opens the option of using higher toroidal magnetic fields [25].

Operation without an OH transformer poses the problem of how to initiate the discharge using the rf system. An emf develops around the torus from the changing equilibrium field during startup; let us consider a simple example in which the only external currents are those in the EF coils, denoted I_{EF} , which have a mutual coupling M to the plasma current, I . The emf is then

$$V = - \frac{d(LI)}{dt} + \frac{MdI_{EF}}{dt} ,$$

where L is the plasma self-inductance. In the presence of both an emf and the rf wave, the electron current will consist of both the conventional dc component and the plateau current carriers, so Ohm's law must be modified as

$$I = \frac{V}{R^N} + I_{RF} ,$$

where R^N is the neoclassical resistance and I_{RF} is the plateau current. These circuit equations can be solved for $I(t)$ with $I_{RF}(t)$ given as an input, subject to two constraints. First, the current in the EF coils must be consistent with that required for MHD equilibrium of the plasma current,

$$I_{EF} = I_{EF}(I, n, T_e) .$$

This is assured by solving the Grad-Shafranov equation for each time step during startup. Second, the resistance R^N depends on plasma properties such as T_e , which are computed by simultaneously solving the dynamic zero-dimensional plasma transport equations for n , T_e , and T_i .

Due to increasing plasma current during startup, there is a reverse emf that induces a return current of bulk electrons that tend to cancel the plasma current [5,6]. As a result the plasma current requires a time L/R^N to reach its steady-state value. At $T_e = 8$ keV, $L/R^N > 10^3$ s. Startup time can be reduced by maintaining a large plasma resistance R^N to the return current of bulk carriers during startup, as will be shown in the example to follow.

This simulation was made for a reactor similar to the reference case; the parameters are not consistent with Table II, but the results are indicative for full-scale reactors. It was assumed that the rf-driven current is of the form $I_{RF} = I_0 P_{rf}/P_0$, where $I_0 = 11.6$ MA and $P_0 = 20$ MW. Figure 12 shows the results of the simulation. The period of 0-32 s is the main plasma current initiation phase, the period of about 32-41 s is the main heating phase, and from 41 s on, the plasma is in a steady-state configuration. The plasma density is composed of equal parts of deuterium and tritium and is ramped up from an average value of 0.5×10^{20} to $1.53 \times 10^{20} \text{ m}^{-3}$ in about 10 s. It was found that the key to a successful startup is to control the applied rf power to the plasma very carefully. During the period 0-32 s, P_{rf} was controlled according to the algorithm: $P_{rf} + P_\alpha - P_R - P_{TR} = 1.0$ MW, where P_α is the alpha heating power of the plasma, P_R is the radiated power, and P_{TR} is the thermal transport power out of the plasma. Physically, this formula ensures that there is a net heating power of 1 MW to the plasma, a relatively small amount. This keeps T_e low enough so that the resistivity can be kept high.

As shown in the figure, the rf heating power needs to be varied from about 5 MW to 40 MW during the 32-s plasma current induction period. $\bar{T}_e$ varies from 30 eV to about 300 eV, which is much lower than would be obtainable if the rf power would just be kept constant at 40 MW. The plasma current starts at $I = 0$ at $t = 0$ and reaches a value of 10 MA at about 32 s. During this period, the current in the EF coils has been ramped up continuously to keep the plasma in MHD equilibrium. At 32 s, P_{rf} is raised to 100 MW in order to heat the plasma to ignition. During this main heating period, the EF current is raised to its final value, and this induces a further 1.6 MA of plasma current to bring I to its equilibrium value of 11.6 MA. When ignition is reached, $\bar{T}_e = 8.5$ keV and P_{rf} is lowered to 20 MW to maintain the current in steady-state operation.

It must be emphasized that, although the current versus power algorithm used in this exercise ($I_0/P_{rf} = 11.6 \text{ MA}/20 \text{ MW}$) is unrealistic, the results qualitatively assure us that startup is achievable on times much shorter than $L/R^N \approx 10^3 \text{ s}$.

During the initiation of rf currents it will be desirable to adjust the n_z spectrum to match the rf current profile to the changing plasma temperature and density profiles. Although a one-dimensional, time-dependent calculation is beyond the scope of our present investigation, we note that the rf system in Ref. 23 has been designed with electronic phase shifters which can be used to program $n_z(t)$.

Additional engineering advantages accrue to our reactor. The power supply configuration is simplified by the removal of the OH system and by the elimination of the energy storage system required by pulsed tokamaks. It may be possible to fill the central hole with nonsaturated iron. This would tend to attract the plasma towards the center and, therefore, reduce the EF system requirements.

The long startup times possible with a steady-state reactor would mean lower power requirements on the EF power supply. Steady-state operation also reduces fatigue problems in reactor materials [26], allowing longer first-wall lifetimes while using readily available austenitic stainless steel. Steady-state operation can alleviate the serious ratcheting problem which occurs at the uranium-cladding interface of any uranium compounds used in the blanket for fissile-fuel breeding; this problem, which is due to the high thermal expansion coefficient of uranium, is one of the main reasons that fission plants are not used by utilities to follow fluctuating loads. Startup without the emf from an OH coil eliminates the need for resistive bellows or ceramic breaks in the first wall, which can improve the reactor reliability. Substitution of rf for neutral beams also eliminates the need for a beam dump inside the torus.

The thermal-to-electric conversion efficiency, $\eta_{th/ac}$, in Table II is typical of a high-temperature, lithium-cooled reactor [27].

Energy production in the blanket region from exothermic neutron reactions is assumed here to multiply the plasma's thermal power by a factor $f_B = 1.2$. The net electric output, $p_{net}^{rf} = 780$ MW, should be compared with that of a conventional tokamak reactor with, for example, a 95% duty factor [27]:

$$p_{net}^{conventional} = 2010 \text{ MW} \times 1.2 \times 0.38 \times 0.95 = 870 \text{ MW}.$$

6. CONCLUSIONS AND FUTURE WORK

Steady-state tokamak reactors driven by rf appear to offer a number of advantages over ohmically driven pulsed devices, despite the penalty paid in recirculating power. One of the most attractive possibilities is operating at very high stable β_e values with force-free surface currents, a possibility which deserves additional attention. The whole question of current density stability and transport remains unexplored. Bootstrap currents and the emf

induced by the EFC during startup will influence the current profile evolution. We note that hollow current profiles have been inferred for some present-day experiments [28]; it may be possible to simulate $j(r)$ evolution using one-dimensional transport codes with MHD-type transport processes [29].

Additional outstanding problems are associated with steady-state fueling and impurity control. Impurity buildup may, in fact, limit pulse lengths, even though transformer volt-second limitations can be removed with the use of rf-driven currents. Even in such a situation, an rf-driven reactor may prove competitive with a transformer-driven model; the engineering and economic tradeoffs between these alternatives must await further detailed analysis.

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TABLE I. Reactor Survey

$\kappa = 7.0$, $B = 9.0$, $A = 3.0$, $\kappa = 1.3$, $d = 0.25$, $\alpha = 2.0$, $q = 1.3-3.9$,
 $\beta_t = 0.07$, $I = 9.81$, $\beta_t = 4.46$, $\Delta_{B/S}^f + \Delta_v = 1.2$, $Z_{eff} = 1.7$

α_n	α_T	$\bar{T}_e$ (keV)	$\bar{T}_i$ (keV)	$\bar{n}_e$ (10^{19} m^{-3})	τ (s)	P_f (MW)	P_w (MW/m ²)
0.9	1.1	10	9.86	13.9	1.91	2420	2.44
0.9	1.1	18	19.1	7.03	2.90	1450	1.46
0.9	1.1	26	31.3	4.52	5.85	772	0.781
0.9	1.1	34	—	—	>10	—	—
1.3	0.7	10	9.82	14.4	1.89	2540	2.56
1.3	0.7	18	18.6	7.32	2.50	1690	1.71
1.3	0.7	26	30.4	4.67	4.70	936	0.947
1.3	0.7	34	45.0	3.42	10.8	512	0.517
1.7	0.3	10	9.84	16.0	1.99	2580	2.61
1.7	0.3	18	18.1	8.31	2.15	2010	2.03
1.7	0.3	26	28.7	5.29	3.43	1250	1.26
1.7	0.3	34	41.8	3.85	6.46	738	0.746

τ is containment time.

P_w is neutron wall loading.

TABLE II. Reactor with Surface Currents at Moderate β_t

<u>Equilibrium:</u>	$A = 3.0, \quad \kappa = 1.3, \quad d = 0.25, \quad \alpha = 2.0,$ $\beta_t = 0.07; \quad q(0) = 1.3, \quad q(a) = 3.9$
<u>Plasma:</u>	$\bar{T}_e = 18.0 \text{ keV}, \quad \bar{T}_i = 18.1 \text{ keV}$ $\bar{n}_e = 8.31 \times 10^{19} \text{ m}^{-3}, \quad \alpha_n = 1.7, \quad \alpha_T = 0.3$ $Z_{\text{eff}} = 1.7, \quad \tau = 2.15 \text{ s}$
<u>Design:</u>	$R = 7.0 \text{ m}, \quad B_{\text{max}} = 9.0 \text{ T}, \quad A_{\text{wall}} = 792 \text{ m}^2$ $\Delta_{B/S}^i + \Delta_v = 1.2 \text{ m}, \quad B_0 = 4.46 \text{ T}$
<u>RF System:</u>	$n_z = 1.2-1.6, \quad P_{\text{rf}} = 56 \text{ MW}$ $I_{\text{WG}} = 67 \text{ MW/m}^2$ $A_{\text{WG}} = 0.83 \text{ m}^2$ $f = 1.5 \text{ GHz}, \quad \eta_{\text{ac/rf}} = 0.41$
<u>Performance:</u>	$P_f = 2010, \quad I = 9.8 \text{ MA}$ $\epsilon = 2.8\%, \quad \text{Duty cycle} = 100\%$ $f_B = 1.2, \quad \eta_{\text{th/ac}} = 0.38$ $P_w = 2.03 \text{ MW/m}^2$ $P_{\text{net}}^{\text{rf}} = P_f \times f_B \times \eta_{\text{th/ac}} - P_{\text{rf}} \eta_{\text{ac/rf}}^{-1} = 780 \text{ MW}$

FIG. 1. Typical spectrum from Brambilla grill.

FIG. 2. Centrally peaked current equilibrium: $A = 3.0$, $\kappa = 1.0$, $d = 0.0$, $\alpha = 1.4$, $\beta_t = 0.06$, $q(0) = 1.0$, $q(a) = 3.0$.

Fig. 3. Surface current equilibrium: $A = 3.0$, $\kappa = 1.3$, $d = 0.25$, $\alpha = 2.0$; $\beta_t = 0.12$ when q varies between 1.0 and 3.0; $\beta_t = 0.07$ when q varies between 1.3 and 3.9.

Fig. 4. Force-free current equilibrium: $A = 2.5$, $\kappa = 1.0$, $d = 0.0$, $\alpha = 2.0$; $\beta_t = 0.16$, $q(a) \neq q(0) = 1.0$; this figure is reproduced from Ref. 17.

FIG. 5. Major radius and plasma current for various beta scalings; $P = 2500$ MW, $B = 9.0$ T, $q_a = 3.0$, $\alpha = 1.4$, $\Delta_{B/S}^1 = 1.0$ m, $\kappa = 1.65$, $d = 0.5$, centrally peaked current equilibria. This figure is reproduced from Ref. 11.

FIG. 6. Major radius and plasma current for various beta scalings; $P = 2500$ MW, $B = 9.0$ T, $q_a = 3.0$, $\alpha = 1.4$, $\Delta_{B/S}^1 = 1.0$ m, $A = 3.5$, centrally peaked current equilibria. This figure is reproduced from Ref. 11.

FIG. 7. Total current and rf power for $\bar{T}_e = 18$ keV, $\alpha_n = 1.7$, and $n_{z1} = 1.60$. Percent of wave penetration is indicated on curves; $I = 9.8$ MA is required for equilibrium.

(ANL Neg. No. 190-79-44)

FIG. 8. Required rf power for reactor with $\bar{T}_e = 18$ keV and $\alpha_n = 1.7$ for several spectra. Percent penetration is indicated along curves.

(ANL Neg. No. 190-79-42)

FIG. 9. Reference reactor profiles: $\alpha_n = 0.3$, $\alpha_I = 1.7$, $\bar{n}_e = 8.31 \times 10^{19} \text{ m}^{-3}$, $\bar{T}_e = 18.0 \text{ keV}$, $I = 9.8 \text{ MA}$, $P_{rf} = 56 \text{ MW}$.

FIG. 10. Lower hybrid wave accessibility.

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FIG. 11. RF system with waveguides routed to external location; note neutron shielding around waveguides. EF coils are shaded.

FIG. 12. Typical reactor parameters during startup.

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