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A NON-INDUCTIVELY DRIVEN TOKAMAK REACTOR
BASED ON ITER

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ABSTRACT

The physics and engineering guidelines for the ITER device are shown to lead to viable physics operating points for a steady state tokamak power reactor. Non-inductive current drive is provided in steady state by high energy neutral beam injection in the plasma core, lower hybrid slow waves in the outer regions of the plasma and bootstrap current. Plasma gain Q (=fusion power/input power) in excess of 20 and average neutron wall loading, $\langle I \rangle \approx 2.0 \text{ MW/m}^2$ are predicted in a device with major radius, $R_0 = 7.5 \text{ m}$ and minor radius, $a = 2.8 \text{ m}$.

INTRODUCTION

The terms of reference [1] for the International Thermonuclear Experimental Reactor (ITER) state that ITER should demonstrate "technologies essential to a reactor in an integrated system." In addition, ITER "will provide the data base in physics and technology necessary for the design and construction of a demonstration fusion power plant." Part of the process of connecting ITER with a demonstration power plant is to perform parametric variations of steady state reactor configurations using the same current drive scenarios envisaged for ITER. At present, high energy negative-ion neutral beam injection (NBI) for current drive in the plasma core supplemented with lower hybrid (LH) slow wave current drive on the outer flux surfaces and bootstrap current, is the favored current drive scenario for the ITER design in steady state. The goal of this paper is to show that this current drive concept leads to plasma physics parameters which approach those required for a viable power producing reactor.

The neutral beams must drive current in the plasma core both to furnish the seed current

necessary to produce bootstrap current and because the LH waves can not penetrate to the axis in a reactor grade plasma. To assure penetration, the neutral beam energy, E_{nb} , is chosen to yield about a five e-fold reduction in beam current along the full beam path through the plasma. Injection tangential to the magnetic axis is used to maximize the current drive efficiency while maintaining sufficient current drive near the axis. Lower hybrid waves propagate only part of the way into the plasma from the outer edge because of a combination of an accessibility condition (primarily a function of plasma density) and a condition for avoiding strong quasi-linear electron Landau damping of the wave (a function of the plasma temperature). For reactor grade plasmas with peak electron density, $n_e \approx 10^{20} \text{ m}^{-3}$, and temperature, $T_e \approx 25 \text{ keV}$, the LH waves penetrate only to a normalized minor radius $r/a \approx 0.7$.

The remainder of this paper is organized as follows. In Section II we present the physics and engineering relations which limit the available parameter space for acceptable reactor operating points. We describe a convenient method for examining possible operating space subject to these constraints in Section III. Results are presented in Section IV including trade-offs in device performance for various operating points and the effect of various assumptions on the available operating space. Section V presents our conclusions.

CONSTRAINING RELATIONS

A fundamental limitation on plasma parameters is the critical toroidal beta beyond which the plasma is expected to become unstable to ideal ballooning or kink modes. This limit is conveniently expressed in terms of the Troyon formula [2]

$$\beta \leq \beta_T = C_T I_p / a B_0 \quad (1)$$

where I_p is the plasma current, a is the minor radius, B_0 is the toroidal field at the geometric major radius R_0 , and C_r is a coefficient, typically 3, which sets the upper limit on β . MKS units are used throughout this paper except for current in MA and temperature in keV. In addition, theoretical studies of ideal kink stability show that the plasma is likely to be unstable to $n=1$ kinks if the edge safety factor q_a drops below about 3.0. It should be noted that some experiments have been able to achieve stable operation at q_a approaching 2.5. The safety factor is given by

$$q_a = B_0 R_0 \oint (ds / R^2 B_p) \quad (2)$$

with B_p the local poloidal field, and R the major radial coordinate. The limit on q_a is typically applied where the poloidal flux is 95% of the total flux enclosed by the separatrix. The limit becomes $q_{95} > 3.0$. Since it is necessary to compute the 2-D MHD equilibrium to find the value of q_{95} , it is desirable to have a simple formula for safety factor. We use the form

$$q_a = \frac{5 a^2 B_0}{R_0 I_p} \frac{1 + \kappa_{95}^2 (1 + 2\delta_{95}^2 - 1.2\delta_{95}^3)}{2} \quad (3)$$

where κ_{95} is the elongation and δ_{95} is the triangularity of the 95% flux surface. Since q_a is typically less than q_{95} , we should take our limit lower than 3.0, but we do not yet have sufficient information to choose a precise number. There are additional conditions on κ_{95} and δ_{95} , typically $\kappa_{95} < 2.5$ and larger δ_{95} required at larger κ_{95} ($\delta_{95} = 0.5$ at $\kappa_{95} = 2.5$).

For the density limit we use the Greenwald limit [3], given by

$$\bar{n}_e \leq \bar{n}_{Gr} = f_{Gr} \kappa_{95} I_p / A_p \quad (4)$$

where $\bar{n}_e$ is the line-averaged density, A_p is the plasma cross sectional area and the coefficient f_{Gr} is taken as 1.0. We note that the density limit is taken to be a "soft" limit in the ITER design process. Recent experiments and analysis have shown that operation near this density does not, by itself, imply that the frequency of disruptions increases. In practice, this limit should seldom be approached in steady state operation.

A variety of scaling laws are possible for energy confinement time. We use one of the models currently used for the ITER studies, namely the relation proposed by Goldston [4],

$$\tau_E = 0.037 I_p R_0^{1.75} \kappa_{95}^{0.5} p_h^{0.5} a^{-0.37} (A_1/1.5)^{0.5} \quad (5)$$

where A_1 is the average plasma mass, and the total heating power P_h is the sum of the current-drive power and the alpha power less the radiated power.

It is widely believed that the energy-confinement time cannot exceed the Ohmic, or neo-Alcator time given by [5]

$$\tau_{NA} = 0.0625 \bar{n}_e a R_0^2 q_a \quad (6)$$

In practice, τ_{NA} is always greater than τ_E under predicted steady state conditions for reactor sized devices. To assure that the confinement time used in our study does not exceed τ_{NA} we

use the formula, $\tau_E = [\tau_{NA}^{-10} + (H\tau_0)^{-10}]^{-0.1}$ where H is described below.

The Goldston scaling was developed for L-mode discharges in limiter plasmas. It was recently noticed that L-mode discharges in divertor plasmas in JET [6] displayed somewhat better confinement time, perhaps as much as 50% higher than the value predicted by τ_E . For these studies we therefore compute τ_E using the prediction from Eq. (5) multiplied by a constant $H = 1.5$.

At present, the likely candidates for non-inductive current drive are lower hybrid (LH) slow waves in the outer region of the plasma, high-energy neutral beams (NB) in the core, and neoclassical (bootstrap) current. The LH waves are limited to the outer region because the choice of $N_{||}$ dictated by the accessibility condition means that the waves are strongly Landau damped on electrons in the outer plasma region where the temperature reaches 10-15 keV [7]. The accessibility condition is given by

$$N_{||} \geq N_{||ac} = \frac{\omega_{pj}^2}{\omega_{ce}^2} + (1 + \frac{\omega_{pj}^2}{\omega_{ce}^2} - \frac{\omega_{pi}^2}{\omega_o^2})^4 \quad (8)$$

where ω_{pj} and ω_{ce} are the local plasma and cyclotron frequencies of species j and ω_o is the frequency of the launched waves. The condition for strong electron Landau damping can be written approximately as [7]

$$N_{||} \leq N_{||}^{ELD} = 7 / T_e^4(r) \quad (9)$$

In our calculations of the power required for current drive, we choose $N_{||}$ at the radius of maximum penetration into the plasma, r_{max} , to satisfy both of these conditions. The required power is reported in the form of a current-drive figure of merit

$$\gamma_{lh} = n_{e20}(r_{max}) I_{lh} R_0 / P_{lh} \quad (10)$$

where $n_{e20}(r_{max})$ is the electron density at r_{max} in units of 10^{20} m^{-3} , and I_{th} and P_{th} are the lower hybrid driven current and absorbed power, respectively. To compute γ_{th} we use the expression due to Fisch [8], multiplied by an adjustment factor to bring it into agreement with detailed calculations of ray tracing and power deposition [9]. This correction factor is typically 0.6.

For the NB figure of merit we employ an expression which can be obtained from an analytic expression for the fast ion distribution function [10]. The expression reads

$$\gamma_{nb} = \frac{\langle n_{e20} \rangle I_{th} R_0}{P_{nb}} = \frac{C_{nb} T_e J(\langle x \rangle, \langle y \rangle) \langle F_0 \rangle}{\ln A_e} \quad (11)$$

where T_e is the density-weighted volume averaged electron temperature, $\langle \rangle$ denotes a volume average and $J(x, y)$ is a function which takes into account the appropriate moment of the fast ion distribution function. It is approximated by [11]

$$J(x, y) \approx x^2 / [4 + 3y + x^2(x + 1.39 + 0.61y^{0.7})] \quad (12)$$

$x^2 = E_b/E_c$, $y = (\sum_i \ln A_i n_i Z_i^2 / A_i) / (3 \sum_i \ln A_i n_i Z_i^2 / A_i)$, A_b is the beam mass in amu, E_b is the NB energy, $\ln A_e$ and $\ln A_i$ are the Coulomb logarithms for electrons and ions respectively, and C_{nb} is a correction factor (typically 3.4) to fit the detailed calculations of beam deposition and fast ion current performed with the ACCOME code [12]. $\langle F_0 \rangle$ is the ratio of the net current to the total fast ion current, which accounts for the partial cancellation of the fast-ion current by the electrons. To calculate $\langle F_0 \rangle$ we use a scaling formula [11],

$$\langle F_0 \rangle = 1 - (Z_b/Z_{eff}) (1 - G(\langle \epsilon \rangle, Z_{eff})) \quad (13)$$

where

$$G(\langle \epsilon \rangle, Z_{eff}) = (1.55 + 0.85/Z_{eff}) \langle \epsilon \rangle^4 \cdot (0.2 + 1.55/Z_{eff}) \langle \epsilon \rangle \quad (14)$$

and $\langle \epsilon \rangle$ is an effective inverse aspect ratio. Detailed calculations of NB current drive [12] indicate that Eq. (13) should be evaluated with $\langle \epsilon \rangle = 0.46 a/R_0$.

The neutral beam energy is set by the requirement that the beams penetrate sufficiently close to the plasma axis to produce an MHD stable current profile there. From calculations of beam attenuation, including

excitation-enhanced stopping cross sections [13], the following scaling formula has been developed for the beam energy required,

$$E_{nb}(\text{MeV/amu}) = (1.37 \times 10^{-23} (Z_{eff} + 0.2))^{0.63}$$

$$[3 \bar{n}_b / (0.2 \bar{n}_e + 1.0)]^{0.118} \bar{n}_b L / r_b \quad (15)$$

where $\xi = (0.7 + 0.022 Z_{eff})^{-1}$, L is the beam path length from the plasma edge to the tangency point, and r_b is the number of e-fold reductions in beam current over the path L . For this study, we have used $r_b = 2.4$ in this formula based on our initial calculations of NB current drive.

The geometry of the tokamak reactor considered in this study is similar to the ITER design. The inner radial build includes a central OH coil stack, TF coil inner leg, and first wall + blanket + shield. The outer bore of the OH coil stack was constrained to be $R_{OH} \leq 2.0 \text{ m}$. The elongation and triangularity on the 95% flux surface were set to $\kappa_{95} = 2.0$ and $\delta_{95} = 0.4$ respectively. A central toroidal field, $B_0 = 6.0 \text{ T}$, was chosen based on a target constraint of about 14 T maximum field on the TF inner leg, although maximum field by itself is not regarded as a fundamental constraint. Fixed parameters for our reference device are given in Table 1.

Table 1. Reference case parameters.

$R_0 = 7.5 \text{ m}$	$B_0 = 6.0 \text{ T}$
$a = 2.8 \text{ m}$	$\alpha_n = 1.0$
$\kappa_{95} (\kappa_f) = 2.0 (2.2)$	$\alpha_T = 0.5$
$\delta_{95} (\delta_f) = 0.4 (0.6)$	Deuterium beams

Another part of the radial build is the inboard first wall/blanket/shield and their associated manifolding and gaps. The thickness of these components is a function of the blanket type used. Based on work summarized in [14], this thickness varies between 1.07 m and 1.28 m over a range of blanket types. The thinnest case is for a water-cooled LiAlO_2/Be blanket while the thickest case is for a Li self-cooled blanket. For this work we will employ a He cooled Li_2O blanket with a first wall/blanket/shield total inboard radial build thickness of 1.16 m.

The TF and PF magnet systems have been designed on the basis of several engineering constraints that ensure their reliability, protection, and structural integrity. These include, sufficient current and temperature margins of the conductor in the high field regions of the coils for good stability, an

upper limit on the hot-spot temperature in a coil during quench and dump of 150 K (with 20 kV terminal voltage during dump), and maximum combined stresses at any point in the case or windings which are within allowables for the materials involved. Table 2. summarizes features of the TF system for the present study. It is a direct, non-optimized extrapolation of present ITER design guidelines. The maximum von Mises stress estimated in the nose region of the wedged TF cases is 670 MPa.

Table 2. TF System Parameters

Number of coils	16
Total current	225 MA
Winding pack geometry	
Cross-section (approx.)	(1.15 m x 0.61 m)
Mean radius of the inboard leg	2.9 m
Mean radius of the outboard leg	13.0 m
Mean height above the midplane	8.9 m
Total TF system stored energy	105.3 GJ
Maximum field at the windings	14.3 T
Winding pack current density	21.2 A/mm ²
Total TF system cold mass	10,400 t
Conductor descrip. (Nb ₃ Sn, cable-in-conduit)	
Operating current	32.9 kA
Critical current	54.8 kA
Current sharing temperature	6.2 K
Effective area of a turn	1552 mm ²
Area of steel in the conduit	666 mm ²
Area of insulation around conduit	154 mm ²
Area of conductor inside conduit	439 mm ²
Area of helium inside conduit	293 mm ²
Fraction of copper in conductor	0.67

The PF system is a closer extrapolation of the present ITER design since the critical central solenoid is essentially unchanged (1.4 m inner radius and 2.0 m outer radius). The present design provides a volt-sec capability similar to ITER but the total stored energy is higher, $W_{PF} = 12$ GJ. The increase results from the increased radii of the divertor and equilibrium field solenoids. Design of the PF system is considerably eased by the dramatic reduction in fatigue cycles and ac losses in this steady state device, compared with ITER which must be designed to operate in pulsed ignited mode.

Combining these components of the radial build places the radius of the first wall on the

inboard side at approximately $R_{TW} \geq 4.5$ m. With this constraint, we chose a geometric major radius, $R_0 = 7.5$ m and a plasma minor radius, $a = 2.8$ m. The inner scrapeoff layer thickness is $\Delta_1 = 0.2$ m. A typical set of equilibrium flux contours for this device operating with toroidal current, $I_p = 25$ MA, is shown in Fig. 1.

Figure 1. Equilibrium flux contours for the baseline case.

STEADY STATE OPERATING CONTOURS (SOPCON's)

We have found that plots of performance parameters (eg. Q = fusion power/current drive power, $\langle \Gamma \rangle$ = average neutron wall load, and current-drive power, $P_{cd} = P_{lh} + P_{nb}$), in the I_p - C_T plane are convenient for comparing steady state operating points. We refer to these Steady state OPERATING CONTOUR plots as SOPCONs. One such SOPCON, is shown in Fig. 2.

The power balances for electrons and ions are performed separately for each point on the SOPCON. Alpha power and current drive power are balanced against transport losses and radiation power (Bremsstrahlung and synchrotron). Transport losses are calculated using empirical energy confinement scaling laws for τ_E . These equations determine the electron and ion temperatures. For a fixed configuration, each point in I_p - C_T space corresponds to a unique

Figure 2. SOPCON for the baseline case.

value of beta through the Troyon formula. This uniquely determines the plasma density at power balance. These equations and the other physics expressions described above are incorporated into the MUMAK code [15].

Radial profiles of the density and temperature are taken to be of the form, $X(r) = X(0) [1 - (r/a)^2]^x$, where $x = n, T$. Power balance terms in MUMAK are integrated over these model profiles using a circular flux surface model to yield profile correction factors. The volume of the fully shaped plasma is used to compute the magnitude of the power balance terms.

RESULTS AND DISCUSSION

Fixed parameters for our reference device are given in Table 1. The major radius and aspect ratio are consistent with the ITER engineering design guidelines. Optimization of the device geometry for reactor performance or minimum cost has not been done. The SOPCON for this reference case is shown in Fig. 2. Goldston energy confinement scaling with an enhancement factor, $H = 1.5$ (limited by the Neo-Alcator confinement time), has been used. The shaded region shows where the density for power balance exceeds the Greenwald density limit. The density and temperature profile exponents are $\alpha_n = 1.0$ and $\alpha_t = 0.5$ respectively (the same pressure profile as in ITER), and the plasma effective charge is taken to be $Z_{eff} = 1.5$ with oxygen impurity. We have adopted the interim ITER guideline for bootstrap current fraction, $I_{bs}/I_p = 0.3$, though higher fractions may be possible in scenarios with high poloidal beta ($\beta_p \geq 1.0$).

Contours shown on the SOPCON include $Q = 20$ and 25, $P_{cd} = 110$ and 150 MW, $q_{95} = 4.0$ and 4.5

(scaled from the MHD calculation at $I_p = 25$ MA shown in Fig. 1), $E_0 = 1.8$ and 2.4 MeV, and $\langle \Gamma \rangle = 1.5$ and 2.0 MW/m². Several possible operating points could be selected. For fixed neutron wall loading, $\langle \Gamma \rangle = 2.0$ MW/m², the operating point with the lowest current is located at $I_p \approx 23$ MA, and $C_T \approx 2.6\%$. The lower limit on the current in this case is due only to the soft density limit. The plasma Q is 21 at this point, with a current drive power requirement, $P_{cd} = 170$ MW, and a beam energy requirement, $E_0 = 2.7$ MeV. Alternatively, a high current operating point with the same wall loading exists at $I_p \approx 36$ MA and $C_T = 3.0\%$. The plasma gain at this point is $Q = 28$ with current drive requirements, $P_{cd} = 120$ MW and $E_0 = 1.4$ MeV. In this case the upper bound on the current, for fixed wall loading, is set by the beta limit. A more detailed list of parameters for an intermediate point are given in Table 3.

We note that the safety factor, at 95% flux, for the operating point in Table 3, is quite a bit higher than in the ITER design ($q_{95} = 5.3$ compared with $q_{95} = 3.2$ in ITER). Since ripple induced losses of fast alphas scale strongly with the safety factor, the use of ferromagnetic inserts might be necessary to reduce the ripple for low current operation. Our calculations have not included fast alpha losses, under the assumption that the required low level of ripple can be incorporated in the design.

The SOPCON shown in Fig. 3 was generated with the same assumptions as for Fig. 2 except that both the density and temperature (and therefore the pressure) profiles were assumed to

Table 3. Steady State Operating Point

P_{fus} [MW]	3500
Q	26
$\langle \Gamma \rangle$ [MW/m ²]	2.0
P_{nb} (P_{th}) [MW]	110 (22)
T_e (T_i) [keV]	19 (18)
n_0 (n_{Gr}) [10^{20} m ⁻³]	0.64 (0.69)
β (C_T) [%]	4.0 (2.7)
β_p ($\beta_{p,th}$)	1.31 (0.99)
n_0 (n_{z1}) [10^{18} m ⁻³]	3.0 (0.45)
Z_{eff} (Z_1)	1.5 (8.0)
τ_E (τ_{MA}) [s]	2.2 (30)
I_p (I_{bs}) [MA]	25 (7.5)
I_{nb} (I_{th}) [MA]	14.7 (2.8)
γ_{nb} (γ_{th}) [10^{20} A/Wm ²]	0.65 (0.50)
q_{95} (q_0)	5.3 (1.05)
$N_{ ,th}$ ($\Delta N_{ }$)	1.6 (0.1)
E_{nb} [MeV]	2.1

Figure 3. SOPCON with peaked density and temperature profiles.

be more peaked, $\alpha_n = 2.0$ and $\alpha_r = 1.0$. This shows the improvement in reactor performance which could be achieved if methods for central fueling and heating were fully developed. For the same intermediate current, $I_p = 25$ MA as in Table 3., and Troyon coefficient, $C_1 = 2.8\%$ (which may be close to the limit for this pressure profile), the plasma Q value increases to $Q = 37$ with the corresponding decrease in current drive power, $P_{CD} = 90$ MW at an average wall loading, $\langle \Gamma \rangle = 2.0$ MW/m². The beam energy requirement is also somewhat lower, $E_b = 1.7$ MeV, from the scaling of Eq.(15).

CONCLUSIONS

The steady state operating scenario for ITER leads to viable plasma parameters for a tokamak power reactor. An operating point has been identified, below the beta and density limits, in a device with $R_0 = 7.5$ m and $a = 2.8$ m which produces 3500 MW of fusion power, an average neutron wall loading $\langle \Gamma \rangle = 2.0$ MW/m² with a plasma $Q = 26$. Preliminary reactor economics analysis indicates that an operating point with higher power density may need to be selected for economically attractive reactor operation. The current is provided by neutral beam injection in the plasma core, lower hybrid slow waves in the outer regions and bootstrap current. The neutral beam power required is $P_{nb} = 110$ MW at a deuterium beam energy of $E_{nb} = 2.1$ MeV. The lower hybrid power requirement is $P_{lh} = 22$ MW. The lower hybrid frequency should be high ($f_{lh} \approx 8$ GHz) to avoid both damping on alphas and parametric instabilities. A He-cooled Li₂O blanket is used with a total inboard radial thickness of 1.16 m. A PF coil set has been found which produces an equilibrium with $\kappa_{95} =$

2.0 and $\delta_{95} = 0.4$ at the 95% flux surface. The total stored energy in the PF set is approximately 12 GJ. A TF coil set has been examined using Nb₃Sn conductor, which produces a 6 T field on the geometric axis of the plasma. The maximum field on the inner leg of the TF coil is approximately 14 T.

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