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CHARACTERISTICS OF A PRE DEMONSTRATION FUSION DEVICE*

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ABSTRACT

Low beta reactor system studies have shown that criteria necessitated by the physics lead to difficult engineering problems. The low power density produced in these systems also results in large reactors and makes the question of economic viability a serious one. It is clear that if higher pressure plasmas can be confined and operated, many of these problems could be alleviated. The incentives for high plasma density as a means of achieving higher pressures are discussed. The possible methods of achieving high density are outlined with particular attention to the high beta approaches. The requirements which are imposed on the reactor by using the flux-conserving tokamak concept are described.

The characteristics of a pre-demonstration fusion device which can test the necessary aspects of such a system are developed. The plasma size required to attain ignition is found to be relatively small provided other criteria can be satisfied. These criteria are described and the technology developments and operating procedures required by them are given. The dynamic behavior and parameters of the reference system during the operating cycle are also outlined.

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INTRODUCTION

The power density produced in a reactor plasma depends on the density and temperature of the fuel ions burned. In the case of a deuterium-tritium fueled system, this quantity is given by $N_D N_T \langle\sigma v\rangle Q_F$. Tokamak reactors will have to operate under ignition conditions to attain economic viability. This requirement imposes parametric constraints on the containment system since ignition depends on density, temperature, and energy confinement time. The energy confinement time in turn depends on other variables, but at present this dependence is not known. There are indications, however, that confinement time will scale favorably with an increase in density. This result is suggested both by experiments and by theoretical analyses. Thus from several points of view there are incentives to operate the reactor plasma at high density.

Several approaches for achieving high density are envisioned. A method which requires a moderate magnetic field (or high beta) is desirable for economic reasons. This may be possible using the concept of a flux-conserving tokamak.(1) Noncircular plasmas may offer a means to high density.(2) Finally, high magnetic fields (or low beta) may be necessary.(3) In this paper a reference system which includes a moderate magnetic field, a moderately elongated plasma, and high beta will be developed. Since the operating temperature of the plasma will be in the range of reactor interest, i.e., $\sim 10 - 15$ keV, high beta implies high density and vice versa in this discussion.

High beta (density) imposes constraints on the design and operation of the system. For example, stable-high beta plasma equilibria must be established and maintained. A compatible poloidal field system must be developed. High power density means that the first wall will be subjected to high neutron fluxes (> 1 MW/m^2) as well as a high thermal flux. The breakdown and current rise phases of start-up may require high voltages when increased filling pressures are used. Supplemental heating using neutral beams may be difficult due to the beam energies required to achieve adequate beam particle penetration. Finally, impurity and burn product effects in a dense plasma must be examined. In some cases technological developments may be needed to meet design criteria while in other cases it may be possible to develop operating poocedures which provide the desired solutions. Specific features of this type are discussed below.

II. SYSTEM MODELING

There are many ways to simulate the detailed energy balance processes in a tokamak. In the present work the same model used previously in the ORNL Experimentl Power Reactor Design Study[4] has been adopted, and the impact of high density has been assessed. The time-dependent model includes ~~deuterium, tritium, alpha, electron, and impurity particle and energy balance equations. It also includes beam-plasma fusion and heating, cold neutrals to simulate both recycling and fueling, and the effects of impurities on radiation and transport.~~ Pseudoclassical, neoclassical, and dissipative trapped particle mode scaling laws as well as radiation effects and cold fueling are included. Conduction and convection times are functions of plasma collisionality.

Impurities affect plasma energetics through the creation of a fuel ion defect and by increasing Z_{EFF} and enhancing radiation losses. The sources of high-Z (iron) and low-Z (carbon) impurities are due to sputtering of the wall by deuterons, tritons, alphas, and impurities as well as by charge-exchange neutrals and neutrons produced in the plasma.

Figure 1 shows the plasma radius required for ignition as a function of temperature. The density is 2×10^{20} m^{-3}, the plasma elongation (height to width ratio) is 1.6, and the axis magnetic field for three cases is shown. The minimum size for ignition occurs at about T = 13 keV. Below this temperature, the $N\tau$ required for ignition increases sharply and an increase in size is necessary to attain it. Above 13 keV, trapped ion mode transport ($N\tau \propto \beta^2 T^{-11/2}$) dominates the scaling and an increase in plasma size again is required. The neutron wall loadings associated with various operating temperatures are also given. Note that at $N = 2 \times 10^{20}$ m^{-3}, T = 13 keV, and B_T = 4.5 T, the wall loading exceeds 3 MW/m^2, and the plasma radius necessary for ignition is 1.25 m. With a field strength of 6.5 T, the size required is reduced to ~ 1 m and the wall loading is ~ 2.5 MW/m^2.

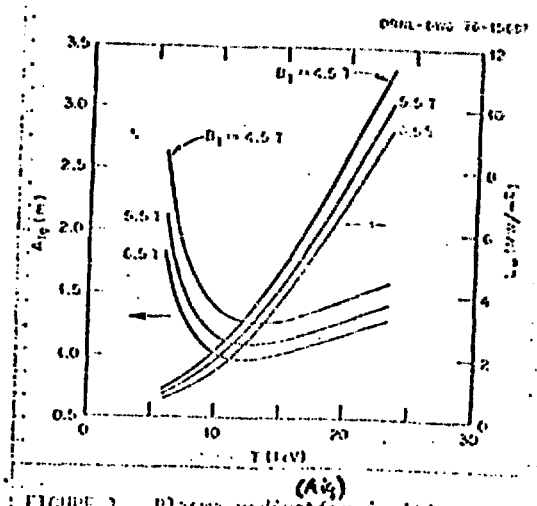

FIGURE 1. Plasma radius for ignition and corresponding neutron wall loading as a function of temperature.

function of density. In these cases T = 13 keV. At lower densities the collisionality of the plasma is such that transport is dominated by the trapped ion mode which requires an increased plasma size. As density is increased the size required for ignition decreases, since the impact of the trapped ion mode becomes less pronounced.

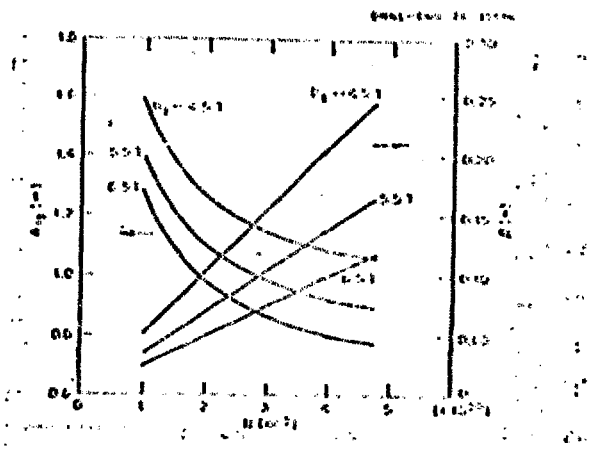

FIGURE 2. Plasma radius for ignition and β as a function of density.

There is no quantitative answer to the question of what maximum density is acceptable. Also, theoretical work[5] suggests that β ~ 20% or higher may be attainable. For this analysis an operating density of about 2 x 10^{20} m^{-3} is assumed. This results in a β ~ 14% at 13 keV, a wall loading less than 4 MW/m^2, and an ignition size of 1.25 m. Table 1 summarizes the reference system parameters computed from a time-dependent, self-consistent simulation model.

Table 1. Reference Parameters

Major radius, R_0(m)	5
Minor radius, a(m)	1.25
Elongation, b/a (-)	1.6
Toroidal field at winding, B_{max}(T)	8
Toroidal field on axis, B_T(T)	4.32
Plasma current, I(MA)	4
Electron density, n_e(m^{-3})	2.2 x 10^{20}
Ion density, n_i(m^{-3})	2.0 x 10^{20}
Ion temperature, T_i(keV)	13.5
Energy confinement time, τ_E(s)	1.2
Toroidal beta, β (-)	0.14
Average neutron wall loading, L_w(MW/m^2)	3.7
Thermal fusion power, P_{th}(MW)	1510

Parameter Searching and System Dynamics

III. OPERATING CONDITIONS AND DYNAMICS

The operating parameters shown in Table 1 represent steady state conditins. Since the model used includes the trapped ion mode where $n\tau_E \propto B^2 T^{-11/2}$, the temperature increases to the point where the energy loss rate equals the heating rate due to thermonuclear alpha particles and a steady state is achieved. Figure 3 shows $n\tau_E$ vs T_e for three different values of magnetic field. The system size is that given in the table and Z_{EFF} = 1 is assumed. As the temperature incrdases, $n\tau_E$ is dominated first by pseudoclassical, then by trapped electron mode, and finally by trapped ion mode diffusion losses.[6] $n\tau_E$ required for ignition is also shown.

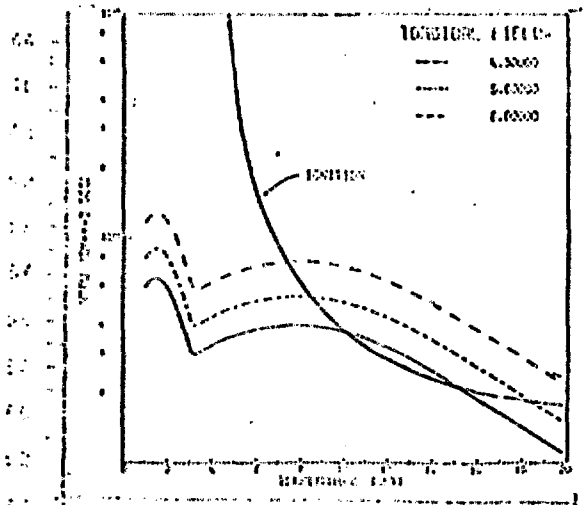

FIGURE 3. $n\tau$ required for ignition and system $n\tau$ operating characteristics as a function of temperature and various field strengths.

For example, when B_T = 4.3 T ignition is attained at 10 keV, a thermal excursion occurs after ignition since P(alpha) exceeds P(loss) and steady state conditions are reached at ~ 15 keV, i.e., $n\tau\{\text{operation}\} = n\tau\{\text{ignition}\}$. There is a minimum field for this size device which can provide ignition. It is about 4.0 T and results in an $n\tau$ (operation)

EXXXE

curve which is tangent to the $N\tau$ (ignition) curve. Note that this would occur at about 13 keV and 2×10^{20} m^{-3} which in turn implies a fusion power output of 1200 MW. This is an important point since it means that (assuming this scaling model) the minimum power output associated with an ignited device of this size is greater than 1000 MW. A heat removal capability compatible with this power level would have to be installed in such a device.

The system could be operated at a higher field. In the 6.0 T case, ignition occurs at 7 keV, but the thermal excursion persists to a temperature greater than 20 keV. This is the result of improved confinement at the higher field. Under these conditions the system could achieve ignition at a density less than 2×10^{20} m^{-3}. The power produced is proportional to N^2 so that ignition conditions at the lower density would be compatible with a power output less than 500 MW. This, of course, represents a low beta system and is less attractive economically. Alternatively, a smaller size system could be used to achieve ignition under high field conditions. Figure 1 showed that a 1 m plasma radius may be sufficient. In this case the power output would be 630 MW when $N = 2 \times 10^{20}$ m^{-3} and $T_i = 12$ keV.

The ignition calculations above assumed a pure ($Z_{EFF} = 1$) plasma. When impurities are included, both the ignition requirement and the system operating characteristics change. Figure 4 shows this for the reference (Table 1) system. Iron is assumed to be the impurity. Due to enhanced radiation losses, ignition cannot be achieved above $Z_{EFF} \sim 1.5$. This establishes the impurity control criteria which will have to be met during the burn cycle of the device. Note also (see Fig. 4) that in the second trapped electron mode $N\tau \propto 1/Z_{EFF}$ and in the trapped ion mode $N\tau \propto Z_{EFF}$ so that the confinement characteristics change in Figure 4

cross over after the transition into the trapped ion mode.

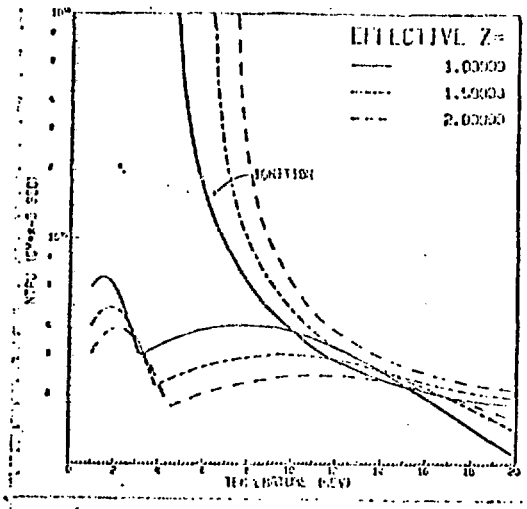

FIGURE 4. $N\tau$ required for ignition and system $N\tau$ operating characteristics as a function of temperature and various impurity (iron) levels.

The desired reactor plasma performance can be attained at B_T = 4.3 T, $N = 2 \times 10^{20}\ m^{-3}$, and T_i = 13 keV prpvided that a plasma with a beta of about 0.15 can be stably confined. The equilibria of such plasmas are discussed in the next section.

IV. MHD EQUILIBRIUM AND STABILITY

The value of beta in a stable toroidal plasma is influenced by the plasma cross sectional shape,[7,8] safety factor profile and toroidal current profile. The conventional approach to and results of MHD equilibria are discussed first and compared to those of the high beta flux-conserving tokamak (FCT) equilibria.[1,5]

Under the assumptions of large aspect ratio and nearly circular concentric flux surfaces, it has been shown that β_p = A is an upper limit in an upper limit in an axisymmetric equilibria.[9,10,11] If β_p exceeds

poloidal field separatrix would intersect the plasma boundary. For nearly concentric flux surfaces, the local values of poloidal field strength can be used to relate q, A, β_p and β as follows,

$$q = \frac{B_T}{AB_p} \quad , \quad \beta = \frac{\beta_p}{q^2 A^2}$$

Thus then implies that beta is limited to $1/q^2A$, which using typical values for q and A yields $\beta \sim 1 - 5\%$.

It has been shown numerically that when these assumptions are not made, $\beta_p \sim A^2$ equilibria can exist in an ideal, circular shell.(12) These equilibria were computed in the usual way, i.e., the pressure function, $p(\psi)$, and the toroidal diamagnetism function, $F(\psi)$, were chosen arbitrarily so that reasonable, bell-shaped toroidal current distributions resulted. The solutions are consistent with those observed in present experiments.

The high beta profiles obtained in this way typically include reversed currents at small values of R, that is on the inside of the plasma cross section. Also, to prevent q < 1 at the plasma center so that satisfy local (Mercier[13]) stability criteria, as β_p is increased, it is necessary to increase q at the plasma boundary. This is accomplished by increasing B_T or decreasing the total plasma current. Because of this, as β_p is increased, β first increases but ultimately decreases. Figure 5 shows this behavior for optimized equilibria with D-shaped cross sections. The maximum β(2.5 – 5.6%) depends on the shape of the current profile assumed but occurs at $\beta_p \sim 2.4$.

In the same beta optimization calculations(4) it was found that there is a limit to increasing beta by elongating the plasma. As the elongation is increased, successively more peaked current profiles are

required to satisfy local stability criteria. The increase in on beta obtained from the slight noncircularity is nullified as the current profile becomes more peaked. The peaked profile enhances the importance of the plasma center where the flux surfaces remain approximately circular concentric. In the highly elongated cases the peripheral regions of the plasma are found to contribute little to the average beta. A moderate elongaton of ~ 1.6 is found to provide the maximum benefit with respect to beta within the assumptions used here.

Next the concept of flux-conservation and its impact on attaining high beta equilibria will be outlined. As described above, high beta equilibria calculations conventionally have used relatively arbitrary profiles for $p(\psi)$ and $F(\psi)$. When heating leads to a pressure excursion on a time scale which is short compared to the magnetic diffusion time, the toroidal and poloidal fluxes are frozen into the plasma. The $p(\psi)$ and $F(\psi)$ functions which describe FCT equilibria are then not arbitrary. Rather $p(\psi)$ is determined by the heating and transport processes and $F(\psi)$ by the requirement that flux be conserved. Also, since

$$q = \delta\phi / \delta\psi ,$$

the q profile established at low beta will persist during the pressure excursion due to flux conservation. The beta limitations observed in

F 5 Fig. 5 due to the necessity to increase q(limiter) to prevent q < 1 at the plasma center should no longer occur. Numerical procedures have been developed[5] to preserve $q(\psi)$ as $p(\psi)$ is increased.

A typical FCT equilibria sequence where q(0) = 1, q(a) = 4.8,

F-6 and $\beta \leq 0.18\%$ is shown in Fig. 6. In these equilibria it is not

necessary to decrease the total plasma current or increase B_T to maintain local stability as β_p is increased. Thus, unlike the conventional high beta equilibria, the FCT approach permits β to increase monotonically with β_p.

The FCT equilibria configurations do not show reversal currents; however, the total plasma current does increase. It has been estimated analytically[14] that the total plasma current scales as $p^{1/3}$. This has been observed numerically[5] as well. The increased current in a high beta FCT equilibrium is produced by diamagnetic effects and does not have to be driven by the ohmic system primary. However, the poloidal field system must be designed specifically to accommodate the rapid changes in the plasma current distribution and magnitude. A low impedence poloidal coil configuration which can provide the needed current wave forms during the FCT pulse is required.

Only local stability criteria satisfied by $q(0) \geq 1$ have been addressed. There is experimental evidence that high beta plasmas can operate stably from the belt-pinch[15] and SPICA.[16] There is, nonetheless, considerable effort required to resolve the nonlocal and more general stability questions associated with FCT equilibria. This work is under way.

When the coils are properly located they will respond in the same way as the ideal shell.

V. POLOIDAL FIELD SYSTEM

The high beta equilibria computed in the previous section were obtained using a fixed boundary MHD code. In this model the eddy currents induced in the shell which surrounds the plasma responds to changes in the plasma motion. The eddy currents provide the proper equilibrium shaping field, cancel the poloidal field which otherwise would exist outside of the shell, and have a sum equal to the plasma current but flow in the opposite direction.[17]. The shell can be replaced by a set of discrete coils in the vicinity of the plasma which produce the ideal equilibrium field pattern. ~~The currents in the coils will be inductively driven by changes in the plasma, will produce the proper equilibrium fields, will cancel the flux (due to the plasma current) which otherwise would exist outside of the coils, and will have an amp-turn total equal and opposite to that of the plasma current.~~ The shape of the current induced in the shell on eitehr side of the "tip" of the D-shaped plasma remains unchanged for large changes in β and I_p. Thus, the coils on either side of the tip can be located[17] to carry equal currents which reduces the number of power supplies and the degree of control required. Finally, the coils must be placed a finite distance from the plasma boundary, and it is found that a divertor-like pair of coils must be located near the tips of the D to reproduce the proper field pattern.[18] Such a poloidal field system retains the features of the STATIC system[17,19] but in addition it is compatible with the equilibrium requirements of the rapid changes of the D-shaped plasma during heating.

The divertor-like coils described above carry currents which flow in the same direction as the plasma current. The separatrix formed as a result of this can be shifted to the edge of the plasma by making

modest changes in the poloidal field system. This divertor configuration is "natural" for D-shaped plasmas. The coil locations in this case are determined as described above except that the shell used is D-shaped and has a sharp tip similar to the geometry of a separatrix. The resulting coil system is shown in Fig. 7. Using these coil locations, typical FCT equilibrium ($\bar{\beta}$ = 0.15) which includes the natural divertor has been computed using the free boundary MHD code and also is shown in Fig. 7. The current required in a divertor coil is 1.54 I_p compared to 1.14 I_p for an equivalent equilibrium without the natural divertor feature. The current level is small compared to conventional divertor coil requirements. Since the coils in this system simultaneously maintain the equilibrium and the separatrix configuration, the location of the separatrix is fixed when the plasma is centered. The difficulty of controlling the separatrix location ~~[illegible]~~ should be minimized.

Finally, the current wave forms in the coils during the pulse are of interest. The pulse is composed of start-up, ohmic heating, FCT heating, ignition excursion, burn, and shutdown phases. Start-up ($0 < t \leq 1$ s) is defined as the time required for breakdown and for the plasma current to reach about 0.5 MA. During the ohmic heating phase ($1 \text{ s} \leq t \leq 2$ s) the plasma current raises to its final value and a low beta discharge is established. The FCT heating phase ($2 \text{ s} \leq t \leq 6$ s) is when rapid beam heating occurs and the equilibrium evolves in a way dictated by flux conservation. Ignition occurs and a thermal excursion follows until the burn phase is established ($10 \text{ s} \leq t \leq 30$ s). Finally, the shutdown phase ($30 \text{ s} \leq t \leq 40$ s) is entered and results in termination of the discharge. Figure 8 shows the

current time histories in the plasma and the equilibrium field and ohmic coil systems which are required to maintain the ~~FCT~~ plasma equilibrium during the cycle. The average beta which is also consistent with the cycle for the pre-demonstration device (cf. Table 1) is also shown as a function of time. This typical case does not include a natural divertor. Table 2 summarizes the results.

Table 2

PARAMETERS FOR PDFD REFERENCE CYCLE

$\bar{\beta}(\%)$	$\beta_{axis}(\%)$	$\bar{\beta}_p(\%)$	I_p, ~~14~~ 7 x I_{EF-D}(MA)	I_{EF-O}(MA)	I_{EF-I}(MA)
0.8	2.7	0.5	4.1	2.43	1.94
1.8	6.4	1.1	4.2	2.68	1.85
5.7	17.3	2.6	4.8	3.60	1.54
11.7	33.6	4.0	5.7	5.02	1.06
14.7	41.2	4.4	6.1	5.75	0.82

The conclusion of these analyses is that the high density desired may well be accessible at moderate magnetic field strengths (high beta) by operating in the flux-conserving mode. The poloidal field system required to permit such operation is reasonable and, in fact, may faciliate impurity control as a result of its natural divertor configuration.

VI. PLASMA START-UP

The gas breakdwon and current rise phase in a large tokamak will establish certain design criteria for the system. The parameters of special interest are voltage and current wave forms, maximum filling pressure and plasma density, volt-second consumptions, tolerable impurity level and line radiation, wall interactions, and runaway electron flux.

Innovative start-up schemes may be required for large tokamaks ($R_0 \approx$ 5 meters) because high density operation may need extremely large voltages to ionize the gas; high runaway electron fluxes may occur; and large tokamaks are very sensitive to impurities.

Previous research in this area[19-21] has been concerned with both gas discharge physics, and atomic and molecular physical processes. The model used in this analysis includes these processes as well as wall interactions, runaway electrons, and time-dependent impurities. These are incorporated into a set of coupled rate equations. The equations are used to compute the time-dependent average values of the density and temperature of electrons, H_1^+, H_2^+, H_1^0, H_2^0, the non-equilibrium ionization states of oxygen, plasma current from a circuit equation, and the runaway electron flux.

The model has been applied to the PDFD reference design (cf. Table 1). For an applied voltage of 50V/turn, which is typical of present tokamaks, and an initial H_2^0 density of 0.2×10^{20} m^{-3} which is five times less than is desirable for high density operation, the density, temperature, and plasma current stagnate at low levels. This indicates that much higher voltages are required in larger tokamaks to achieve start-up, even at relatively low filling pressures. With $N_{20} = 1.0 \times 10^{20}$ m^{-3} and 450 V/turn, the plasma is fully ionized after 14 msec. However, even a small amount of impurities would cause line radiation at these very high densities so the plasma must be kept extremely pure.

The ionization and plasma current initiation in large tokamaks requires very large applied voltages and low impurity levels. Runaway electron creation may be an important effect. Research is required on small radius and low density start-up methods.

VII. PLASMA HEATING

The fundamental aspects of neutral beam heating during the plasma start-up are discussed in this section. The supplemental heating power level necessary to achieve ignition in the reference system in about 6 s is found to be 75 MW. This calculation is based on energy balance considerations using the final density ($\sim 2 \times 10^{20}$ m^{-3}) and does not include the problems of beam penetration into a dense plasma. High density and high impurity content make penetration into a large plasma increasingly difficult. Beam focusing, perpendicular injection, and high beam energies may be necessary to overcome the penetration problem. Small radius start-up and low density start-up may provide other means to solve the problem. Here the concept of low density start-up, which has previously been suggested[22,23] is outlined.

Beam deposition can be characterized by the function $H(\bar{r})$ which was first introduced by Rome et al.[24] $H(r)$ is a geometric measure of the deposited fast ion number density profile which results from injection. It is desirable to have the beam deposition peaked in the central region of the plasma for maximum heating efficiency. In these calculations a deuteron beam, 30 cm in diameter, is assumed to be injected into a $Z_{EFF} = 1$ plasma.

Figure 9 shows the deposition profiles for several different beam energies. In these cases injection is nearly perpendicular (16^0 from perpendicular to prevent ripple trapping of the fast ions), and the density profile is assumed to be $N = 3 \times 10^{20}$ $(1 - (r/a)^2)m^{-3}$. Adequate penetration can be achieved in this way if a beam energy of $\sim$ 250 keV can be made available. Impurity effects dictate an every higher energy.

High density and temperature are required at the ignition point where the beams are no longer needed to sustain the plasma energetically.

It may be possible to establish the temperature conditions at low density, increase the fueling rate to build up the necessary density while continuing injection, and tolerate the less favorable injection profiles which will occur near the end of the heating power. Figure 10 shows H(r) for 150 keV tangential injection into average densities of 7.5, 10, and 20 x 10^{19} m^{-3}. The injection deposition profile is favorably peaked when N = 7.5 x 10^{19} cm^{-3}, is relatively flat for N = 1 x 10^{20} m^{-3} and becomes peaked at r/a = 0.75 when N = 2 x 10^{20} m^{-3}. Therefore, heating efficiency will decrease when N > 1 x 10^{20} m^{-3}, but if this part of the heating cycle occurs over a time on the order of an energy confinement time, the less efficient heating process may well be tolerable.

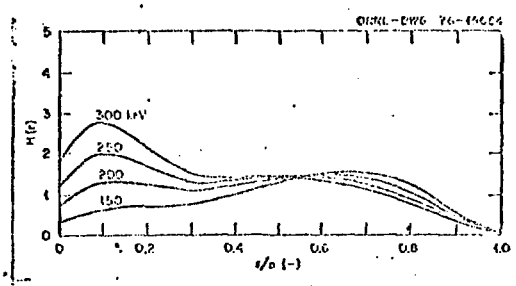

FIGURE 9. Neutral injection fast ion deposition profiles for various injection energies (perpendicular injection).

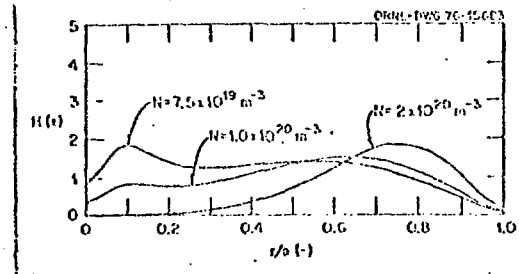

FIGURE 10. Neutral injection fast ion deposition profiles for various plasma densities (tangential injection).

Figures 11 and 12 show the time-dependent density and power histories during heating. Initially $N = 7.5 \times 10^{19}\ m^{-3}$, $T = 100$ eV, and $P_{beam} = 75$ MW. At $t = 2$ s, $T_i = 12$ keV; and the fueling rate is increased to begin the density buildup phase. The plasma ignites and beam injection is terminated after $t = 4.4$ s. Density buildup occurs over a period of 2.4 s, and the average energy confinement time during this phase is $\tau_E \approx 1.5$ s. Energetically this scenario is feasible since 75 MW is sufficient to heat the plasma to ignition at full density.

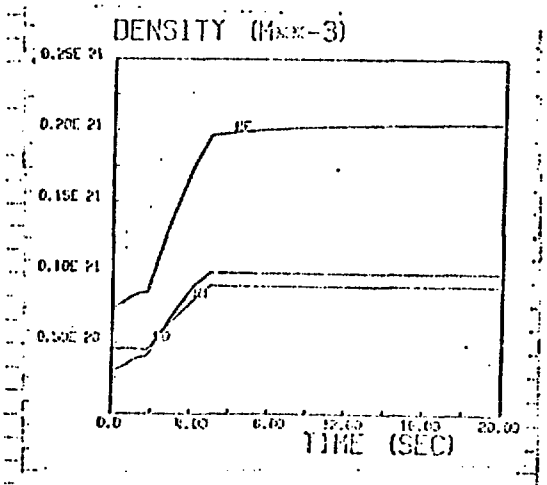

FIGURE 11. Plasma density buildup during low density start-up procedure.

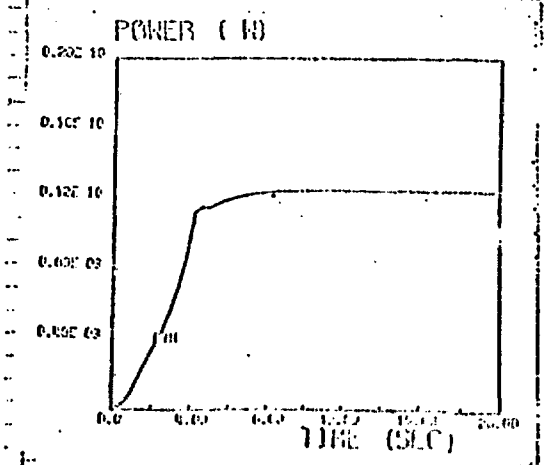

FIGURE 12. Fusion power output during low density start up.

In fact, more detailed studies of the low-density approach using a coupled beam deposition-spatially dependent transport calculation are necessary to optimize the start-up procedure. Also, as the flux surfaces shift when high beta is attained, beam penetration should improve.

VIII. CONCLUSIONS

There are incentives for high density operation in a tokamak reactor. Several features of a PDFD which will demonstrate the feasibility of attaining such operation in a flux-conserving tokamak were described. Other key aspects such as fueling, impurity control and the effects of magnetic field ripple were not included here. Suitable high beta equilibria were found and a poloidal field system design compatible with the equilibria was developed. The natural divertor configuration inherent in the system led to including a poloidal divertor in the PDFD. The start-up analyses showed that further innovations are required unless very large voltages are applied to achieve breakdown and to establish the plasma current. Neutral beam energies of $\sim$ 300 keV were found to be necessary when the plasma is heated under full bore and full density conditions. A low density start-up procedure was outlined which, if feasible, suggests that beams with energies similar to those being developed for TFTR may be adequate to achieve ignition in the PDFD. Successful operation of a PDFD of this type will lead to high power density, high beta fusion reactors of reduced physical size. These features, in turn, enhance the possibility of attaining economic viability in the tokamak.

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