Ignition in Near Term D-^{3}He Tokamak Reactors

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IGNITION IN NEAR TERM D-^{3}He TOKAMAK REACTORS

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Abstract

The prospects for achieving breakeven and ignition in near term ETR type tokamaks under D/He-3 relevant conditions are considered. Using present scaling laws for beta in the first stability regime, it is found that CIT may be close to breakeven with the presently planned toroidal magnetic field system, if the ASDEX H-mode scaling law is used. With Kaye-Goldston scaling, Q = .22 can be attained, but this requires an excessive amount of RF heating power. Larger devices, such as NET/INTOR, can ignite with ASDEX H-mode scaling with an increase of the toroidal field by 20% and removal of the blanket and reduction of the inboard shield to that required for D/He-3.

Introduction

The recent realization [1] that the moon contains an enormous amount of He-3 has rekindled interest in fusion reactors based on the D/He-3 cycle. The higher operating temperature and reduced reactivity of D/He-3 relative to D/T requires better energy confinement and higher beta operation to be economical. In order to establish the physics data base necessary for D/He-3 reactors, it is desirable to consider whether significant plasma performance with D/He-3 can be obtained in near term devices planned for D-T operation. Since tokamaks represent the presently leading concept for magnetic confinement, we consider the possibility of breakeven and ignition in proposed ETR devices, such as CIT, TIBER, FER, and NET/INTOR using D/He-3 fuel. These devices span the range from small major radius (1.2 m) and high field (10 T) to large major radius (5.2 m) and moderate magnetic field (5 T). The physics assumptions in this analysis are the same as those used for predicting their performance with D/T fuel. The question of whether an economical reactor can be made with these assumptions is left for others to consider (the possibility of a D/He-3 tokamak reactor using the second stability regime is considered in a companion paper [2]).

Plasma Model

The performance of a tokamak reactor operating with D/He-3 fuel can be estimated using the DHE3TOK computer code, which is essentially a global power balance code. This code calculates the fusion power produced for given plasma density and temperature profiles. The plasma beta, which is determined by the plasma density and temperature and the pressure of the fast fusion produced ions, is constrained to satisfy MHD equilibrium and stability considerations. The loss mechanisms included are bremsstrahlung and synchrotron radiation, with relativistic corrections, and transport across the magnetic field. The electron and ion temperatures are allowed to separate; rethermalization, with relativistic corrections, is included.

In this analysis we consider operation only in the first MHD stability regime. The volume-averaged beta is determined by the Troyon [3] formula,

$$\beta = \frac{C\ I\ (MA)}{a\ (m)\ B_T(T)}$$

where the coefficient C is normally in the range 3 to 4. We use the same value as that used in determining the D-T performance of the device.

The biggest uncertainty in our analysis is the energy confinement time for transport across the magnetic field. We use empirical scaling laws to estimate this loss. These are based on present experiments, which operate at a similar density but an order of magnitude less temperature than that required for D/He-3 fuel. In order to see the sensitivity of the results to a change in the energy confinement scaling law we consider two different scaling laws: Kaye-Goldston scaling [4] with an H-mode factor of 2, and ASDEX H-mode scaling [5]. The loss is assumed to be in the electron channel. The loss in the ion channel is assumed to be neoclassical (or a multiple of it), which is a negligible loss at D/He-3 conditions.

The DHE3TOK code calculates the ignition margin, which is defined as the fusion power produced divided by the total power loss, and the energy multiplication, Q, which is defined as the fusion power divided by the externally injected power. If the plasma is ignited, then the ignition margin is above unity and one has a safety factor against increased losses. If the plasma is not ignited, then Q determines the power that must be injected into the plasma (presumably by an RF heating system) to maintain the thermal balance of the plasma. The Q-value computed assumes the ions have a Maxwellian distribution function. Any enhancement of Q due to the RF system driving "tails" in the ion distribution is not included.

Results

Table 1 gives the basic parameters of the devices studied in this report. Figure 1 shows the energy multiplication, Q, predicted for CIT with D/He-3 fuel versus average ion temperature for both Kaye-Goldston and ASDEX H-mode scaling. The Q value maximizes at an ion temperature of about 40 to 50 keV; the injected power required to sustain the plasma at this temperature for Kaye-Goldston scaling is 106 MW, which is beyond the power levels considered for CIT; for ASDEX H-mode scaling the required power is only 20 MW, which is in the range considered for startup. Similar calculations have been done for FER, TIBER, and NET. The peak Q-value and the corresponding required injection power are shown in Table 2. A device of the size and parameters of NET can achieve breakeven (Q = 1) with ASDEX H-mode scaling. Note that Q = 1 in D/He-3

Table 1. Basic Parameters of the Devices

Parameter	CIT	TIBER	FER	NET
Magnetic Field (T)	10.4	5.5	5.3	5.0
Plasma Current (MA)	9.0	10.	6.0	10.8
Major Radius (m)	1.22	3.0	5.2	5.2
Aspect Ratio	2.71	3.6	4.64	3.8
Elongation	1.9	2.4	1.5	2.2

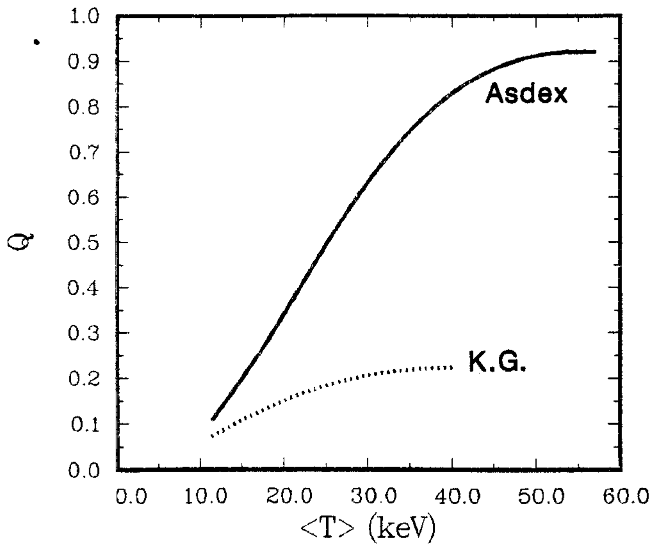

Fig. 1. Energy multiplication versus average ion temperature for CIT using both Kaye-Goldston and ASDEX H-Mode energy confinement scaling.

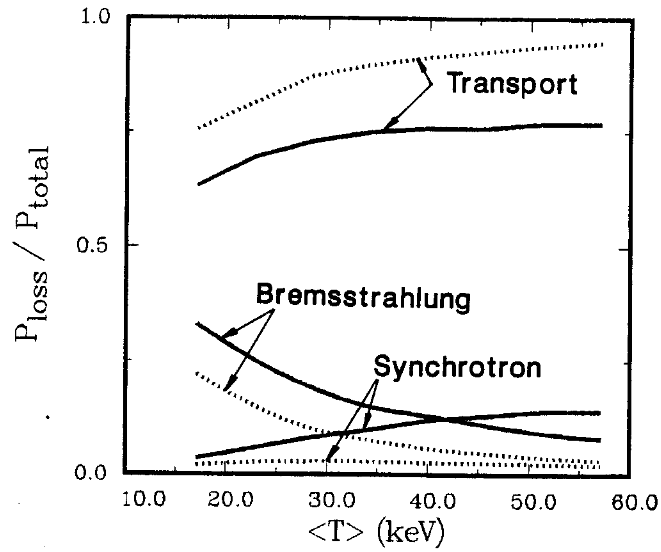

Fig. 2. Distribution of power losses versus average ion temperature for NET using both Kaye-Goldston and ASDEX H-mode scaling.

Table 2. Q-Values and Required Injection Power

Device	K.G. Q	K.G. P (MW)	ASDEX H-Mode Q	ASDEX H-Mode P (MW)
CIT	.22	106	.92	17
TIBER	.25	92	.83	21
FER	.20	88	.49	27
NET	.37	123	.99	33

is equivalent to Q = 5 in a D/T plasma from the point of view of the impact on the plasma power balance. The distribution of the various losses is shown in Fig. 2 for the NET parameters. Transport is the largest loss mechanism; synchrotron and bremsstrahlung radiation are considerably smaller.

In these calculations, there is no change in the plasma dimensions or magnetic field. Only the fuel and the operating temperature is changed. With D/He-3 fuel, however, the amount of neutron shielding required is much less since the neutron production is much reduced from the D/T value. In addition, there is no need for a tritium breeding blanket. Consequently, improved performance can be obtained by removing the inboard breeding blanket and reducing the shielding to that required for D/He-3. This allows the major radius of the plasma to be reduced and/or the minor radius to be increased. Figure 3 shows the Q-value for a plasma with a minor radius of 1.7 m for various major radii. The MHD safety factor, q, is held constant as the major radius is reduced. The reduction of major radius has three benefits: increased magnetic field at the plasma, increased plasma current and therefore energy confinement, and increased beta.

Finally, we consider the possibility of achieving ignition in D/He-3. With Kaye-Goldston scaling, ignition appears to be out of reach for the parameters of these devices. With ASDEX H-mode scaling it may be achieved by increasing the magnetic

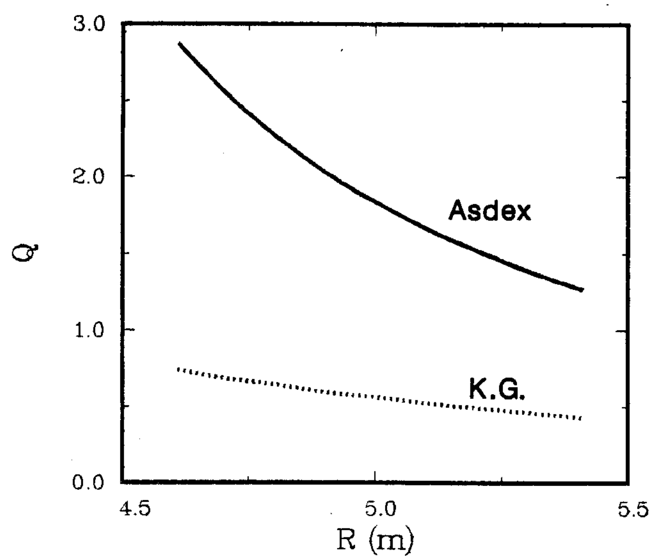

Fig. 3. Energy multiplication versus major radius; the plasma half-width is 1.7 m, and the magnetic field at the TF magnets is 10.4 T.

field and the elongation. Shown in Fig. 4 is the ignition margin, Q/(1+Q), versus magnetic field at the plasma using an elongation of 2.4, which is the TIBER value; the major radius is held at 4.7 m and the plasma aspect ratio is 2.7. Ignition is achieved at a magnetic field of about 7 T; with the space required for neutron shielding, this corresponds to 13 T at the magnet.

Acknowledgement

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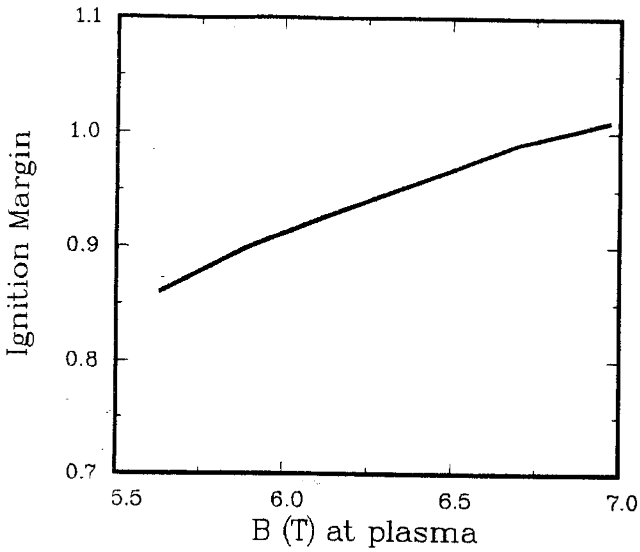

Fig. 4. Ignition margin versus magnetic field at the plasma for a major radius of 4.6 m, plasma aspect ratio equal to 2.7, and elongation of 2.4. Ignition is achieved at a magnetic field of 7 T.

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