Potential for D-^{3}He Experiments in Next Generation Tokamaks

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Abstract

The potential for D-^{3}He experiments in the proposed CIT and ITER tokamak test devices is examined. In CIT, an energy multiplication, Q, of about 0.3 can be obtained with an injection power of about 100 MW. Without modifications to ITER, except for the change of fuel, it is found that Q of the order of 0.3 to 0.5 can be obtained. Breakeven with D-^{3}He requires modification to the device to increase the elongation to 2.4, reduce the major radius to 5.6 m, and increase the magnetic field at the plasma from 4.9 T to 5.6 T. Operation with a small amount of tritium seeding can reduce the auxiliary power required to achieve breakeven and leads to Q=2 in an unmodified device.

1. Introduction

It is well known that D-^{3}He fuel offers significant advantages for reducing some of the difficult technology requirements for achieving controlled fusion (factor of 10 to 50 reduction in neutron production and associated radiation damage and activation of the structure, a similar reduction in the radioactive waste disposal problem, elimination of the need for tritium breeding, elimination of the need for periodic replacement of the first wall, possibility of direct conversion for higher overall system efficiency). The disadvantages are also well known (about a factor of 4 increase in $n\tau$ for ignition and about a factor of 4 increase in the plasma temperature). Nevertheless, it appears that a commercial tokamak reactor using D-^{3}He fuel may be economically competitive with one using D-T fuel[1]. The reduction of the engineering problems compensates for the more difficult physics problems associated with D-^{3}He fusion. In addition, the environmental advantages of a D-^{3}He fusion reactor, compared with D-T, are considerable and are more likely to lead to public acceptance of fusion power.

Given these considerations, it is prudent to investigate D-^{3}He operation in next generation tokamak facilities in order to determine if the present understanding of tokamak physics will extrapolate to D-^{3}He reactor conditions. This requires experimental testing at D-^{3}He plasma conditions in order to study plasma transport and power balance, fueling mechanisms, impurity control, and other plasma considerations. The use of D-^{3}He fuel in D-T facilities can also provide valuable information about burning plasmas before the facility is contaminated with tritium and activated by intense radiation. In this regard, it can benefit the D-T research program and provide some insurance against contaminating the facility with tritium prematurely.

Experiments with D-^{3}He have been carried out in JET where a minority concentration of ^{3}He in a predominantly deuterium plasma has been heated using ICRF. The enhanced tail of the ^{3}He energy distribution driven by ICRF produced as much as 140 kW of D-^{3}He fusion power. Experiments of this type can provide copious amounts of fast ions (14.7 MeV protons and 3.7 MeV alpha particles) which can then be used to study fast ion loss and thermalization processes.

In this paper we discuss options for D-^{3}He experiments in the CIT[2] and ITER[3] proposed tokamak test devices. CIT (Compact Ignition Torus), now known as BPX (Burning Plasma Experiment), is a small, high magnetic field tokamak with a major radius of 2.6 m (at the time of this work), an on-axis magnetic field of 9 T, and a plasma current of 11.8 MA. ITER (International Thermonuclear Experimental Reactor) is a larger tokamak; the major radius is 6 m. The on-axis magnetic field is 4.9 T, and the plasma current is 22 MA; there is an option for increasing the current to 28 MA. A previous study[4] considered D-^{3}He operation in NET; this study is similar but uses updated physics assumptions.

2. Calculational Model

The assessment of D-^{3}He operation has been done using a power balance code (DHE3TOK) similar to the physics portion of the ITER systems code, but specialized for the peculiarities of D-^{3}He fuel. In particular, the higher operating temperature of D-^{3}He requires a more careful consideration of synchrotron radiation and relativistic effects as well as different species in the plasma. The DHE3TOK code considers 6 species (deuterium, ^{3}He, protons, ^{4}He, one impurity species, and electrons) and calculates the self-consistent ion and electron temperature in the presence of external heating and internal heating due to fusion reactions. The nuclear reactions considered are ^{3}He(d,p)^{4}He, D(d,n)^{3}He, D(d,p)T, and T(d,n)^{4}He. Maxwellian averaged cross-sections are used for each of these reactions.

All ionic species are assumed to have the same temperature, but the electron temperature can be different. The density and temperature profiles are assumed to be of the form of a parabola raised to a power. The operating density is determined by density limits or beta limits, whichever is applicable, and the beta limit is determined by the Troyon formula. For the operating points of interest for D-^{3}He in CIT and ITER, the beta limit is more restrictive because of the higher plasma temperature. Included in the calculation of the plasma pressure is the thermal pressure of all ion species and the pressure of all fusion produced fast ions[5]; this is calculated using a slowing down approximation based on the Fokker-Planck equation.

The power loss mechanisms considered are 1) energy transport across the magnetic field (we use the ITER-89P[6] power law empirical scaling expression for energy confinement for ITER calculations and Goldston[7] scaling for CIT), 2) synchrotron radiation (the Trubnikov expression as modified by Tamor for high temperature is applied locally in the plasma and volume-averaged[8,9]), 3) bremsstrahlung, and 4) electron-ion energy exchange. Relativistic corrections to both bremsstrahlung and energy exchange are included. The power gains are fusion product heating and external heating. The fraction of the fusion power transferred to the ions is obtained from the slowing down approximation[4]. External heating is treated in a generic sense. The code calculates the external power necessary to sustain the plasma at a given density and temperature. A specified fraction of the heating power is assumed to go to the ions and the rest to the electrons. A 1% oxygen impurity level is assumed.

The figure of merit used to evaluate the performance with D-^{3}He fuel is the energy multiplication, Q, which is the ratio of the fusion power produced in the plasma to the injected power required to sustain the plasma. Note that Q=1 in D-^{3}He is equivalent to Q=5 in D-T from the point of view of the impact of fusion reactions on the plasma power balance. This is because essentially 100% of the fusion energy is in the form of charged particles for the D-^{3}He reaction, whereas it is only 20% for the D-T reaction. For the calculations presented in this paper, the fuel mixture is taken to be 65% D and 35% ^{3}He; this fuel mixture maximizes Q. Deuterium-rich operation reduces the electron density and thereby reduces the power losses.

3. D-^{3}He Experiments in CIT

An illustrative example of the performance of CIT with D-^{3}He fuel is given in Figures 1-3; these use Goldston scaling for energy confinement with an enhancement factor of 3 for H-mode operation. This corresponds to an enhancement factor of about 2.5 for ITER-89P scaling using CIT parameters. Figure 1 shows contours of constant injection power in a space

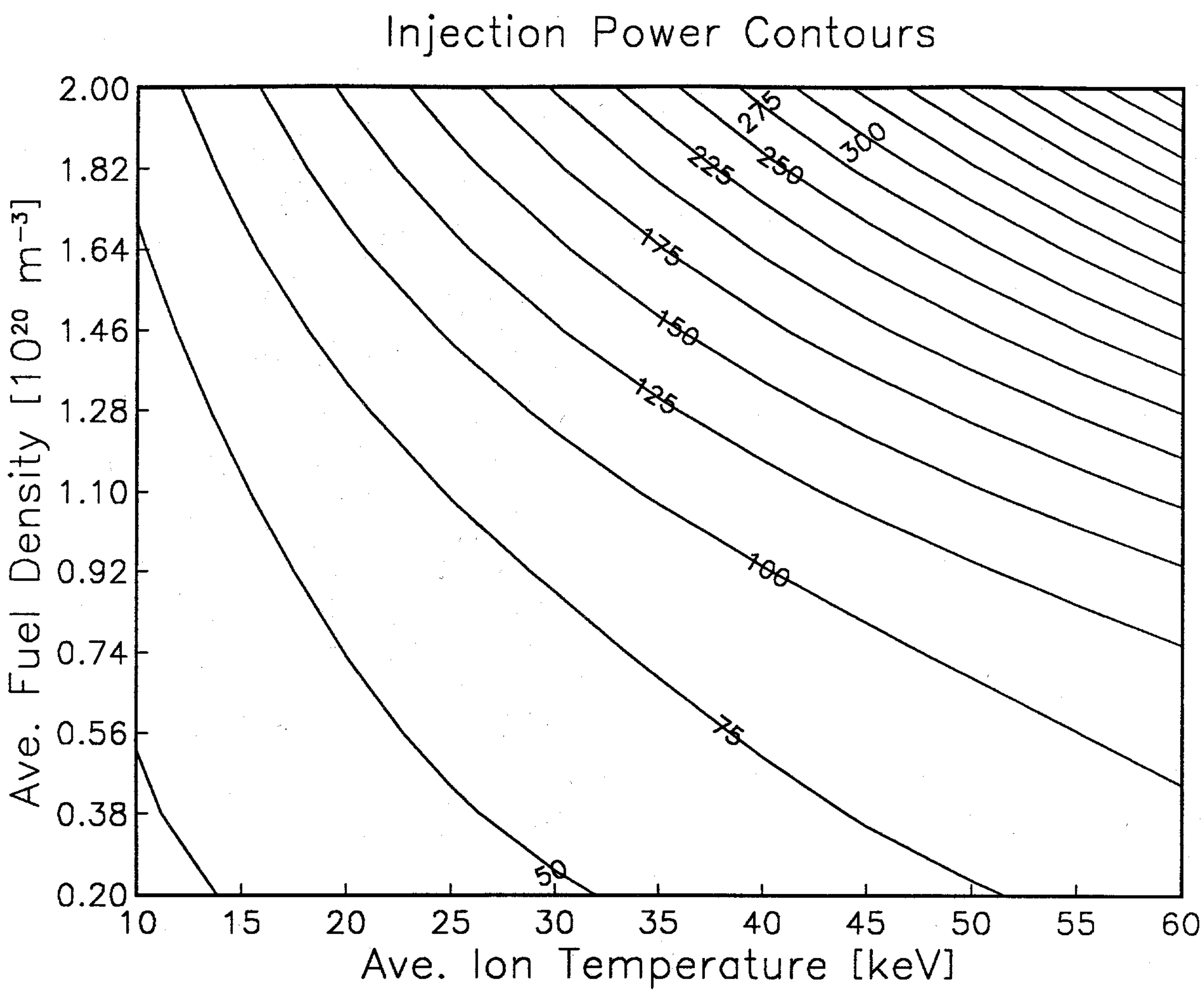

Fig. 1. Auxiliary heating power contours (in units of MW) for D-^{3}He in CIT.

of average fuel density versus the average ion temperature (all temperatures quoted in this paper are density weighted and volume-averaged and all densities are volume-averaged). Since, for pulse lengths of a few seconds, the CIT device can handle a total power of about 100 MW, the region above P_{aux} = 100 MW should be regarded as inaccessible for practical purposes. This power limit nearly coincides with the conventional Troyon β-limit using a coefficient of 3.0. Contours of constant β are shown in Fig. 2. Figure 3 shows contours of constant energy multiplication, Q. The maximum Q value that can be obtained for 100 MW of injection power is about 0.3 and the corresponding fusion power is about 30 MW. The maximum Q occurs for an average ion temperature of about 35 keV and a density of about 1×10^{20} m^{-3}.

The performance of CIT is clearly critically dependent on the enhancement factor h, just as it is in the case of D-T operation. Since the maximum Q ~ h^2, achieving Q = 1 would require h ~ 5, far outside of the present realm of experience.

3. D-^{3}He Experiments in ITER

For ITER we consider both pure D-^{3}He operation and operation with a small mixture of tritium in addition to D and ^{3}He. The question to be investigated is whether approximate breakeven conditions (D-^{3}He fusion power about equal to the injection power sustaining the plasma) can be achieved in ITER using the same physics scaling laws used in determining the performance of ITER with D-T fuel. Pure D-^{3}He operation is discussed in Section 3.A and mixed mode operation with a small amount of tritium in a D-^{3}He plasma is discussed in Section 3.B.

3.A Pure D-^{3}He Operation

Shown in Fig. 4 is the energy multiplication, Q, versus density weighted, volume averaged ion temperature for pure D-^{3}He operation. We see that Q improves with ion temperature and reaches a maximum at about 35 to 50 keV depending on the confinement scaling expression and the beta limit. The Q-values for the 28 MA case are considerably better

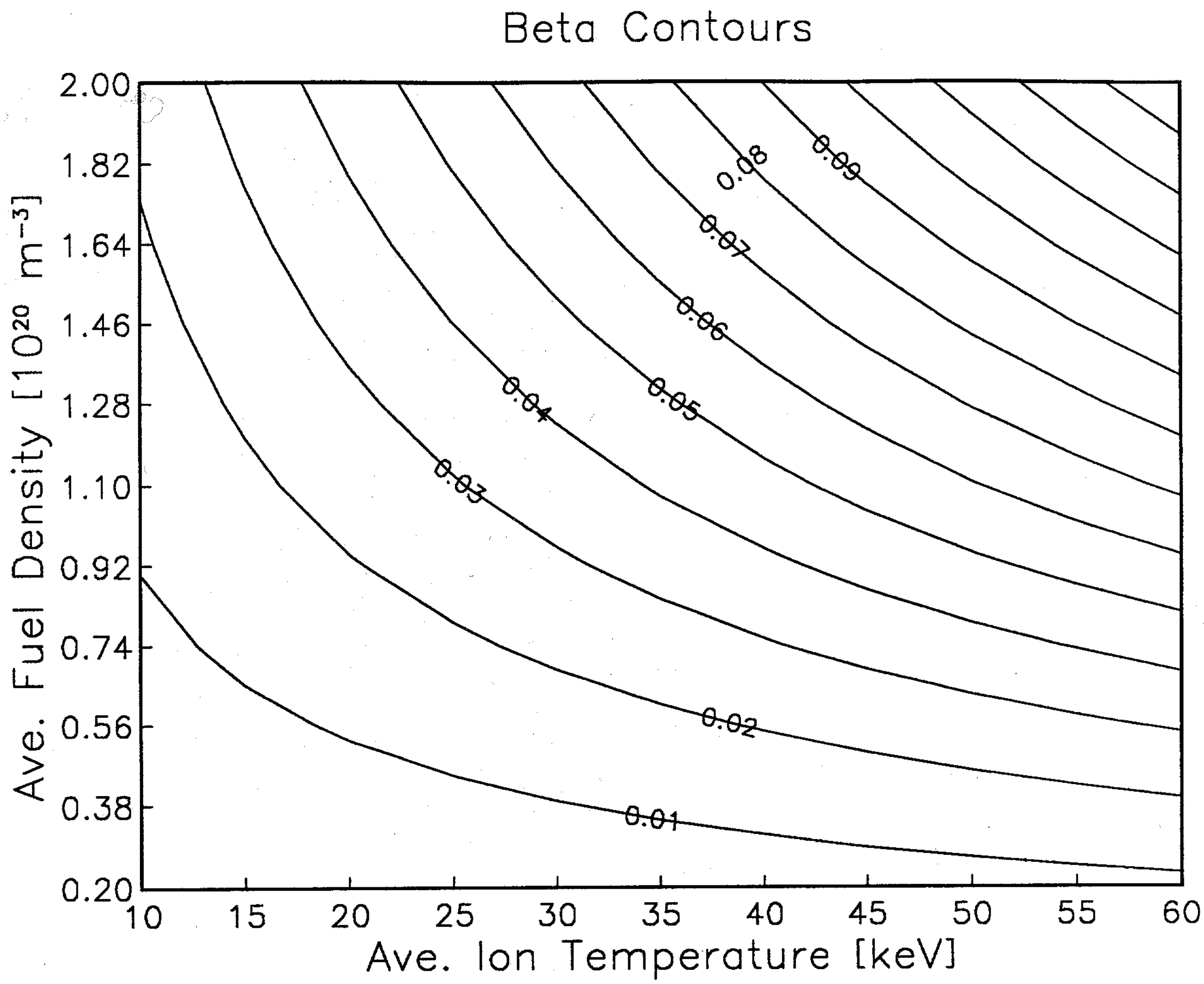

Fig. 2. Constant beta contours for D-^{3}He in CIT.

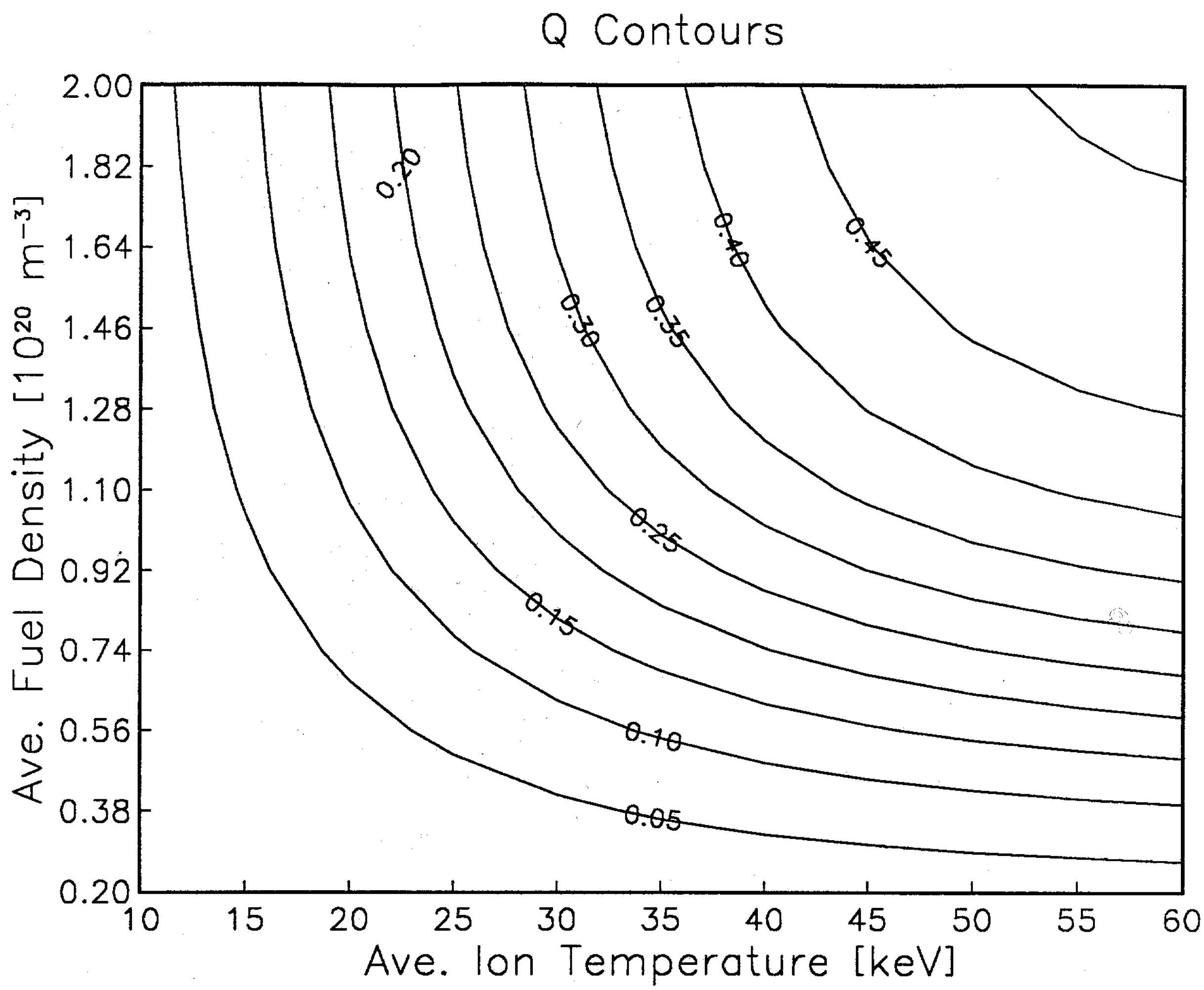

Fig. 3. Constant energy multiplication, Q, contours for D-^{3}He in CIT.

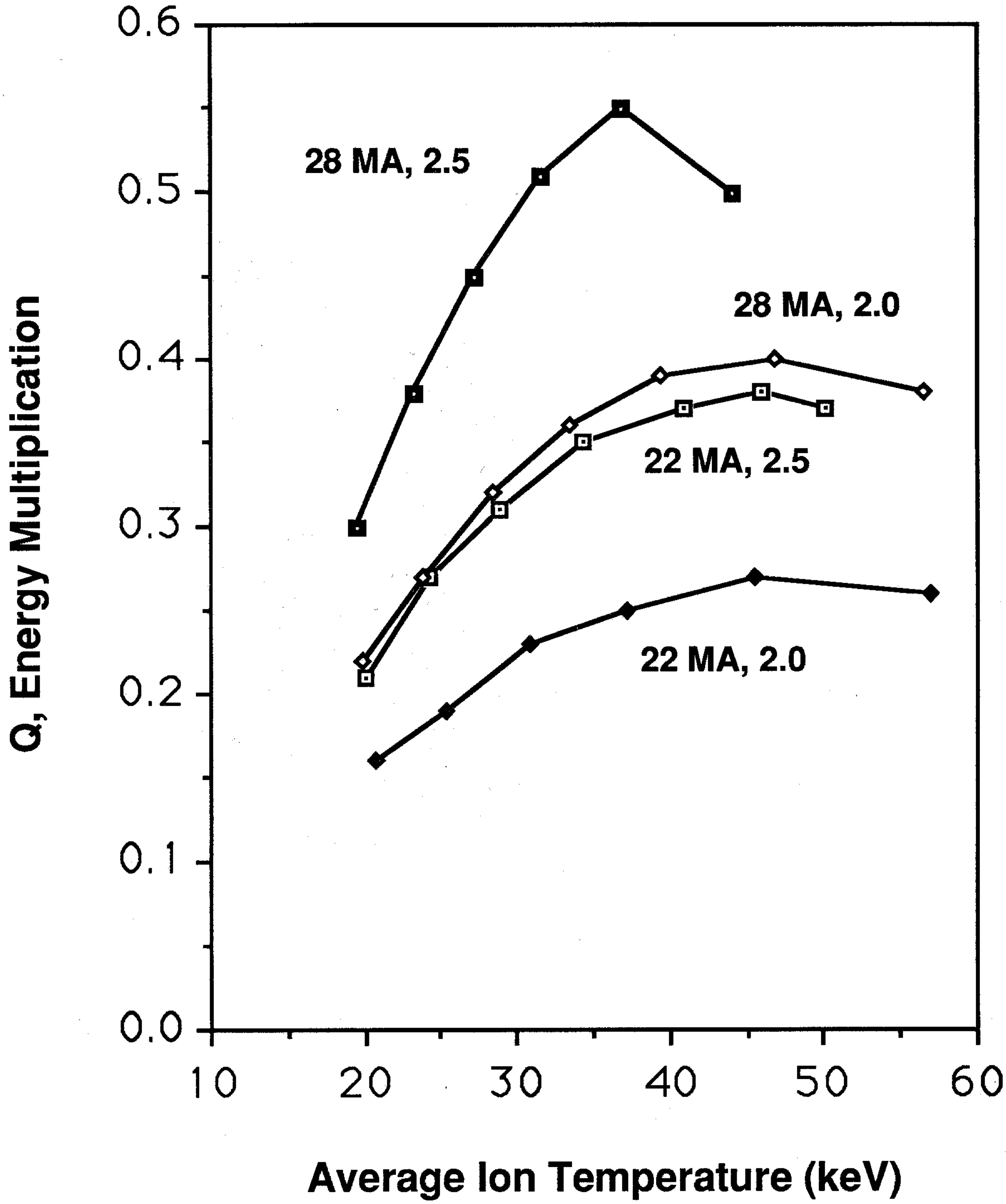

Fig. 4. Variation of Q with average ion temperature in ITER. The four cases are labeled by the plasma current and the H-mode multiplier.

Table 1. Representative Parameters for D-^{3}He in ITER

Plasma current (MA)	22	22	28	28
H-mode multiplier	2.0	2.5	2.0	2.5
Q	0.27	0.38	0.40	0.55
Injection power (MW)	186	122	183	124
Fusion power (MW)	49	46	72	68
Ion density (10^{13}/cm^3)	3.1	3.0	3.7	4.5
Ion temperature (keV)	46	46	47	37
Electron temperature (keV)	30	32	33	30
Energy confinement time (s)	3.4	5.4	4.3	6.7
Slowing down time (s)	5.8	6.5	5.6	6.5
Particle confinement time (s)	10	16	13	20
Troyon coefficient	2.5	2.5	2.5	2.5
Transport power (MW)	195	124	198	131
Synchrotron power (MW)	28	32	39	33
Bremsstrahlung (MW)	13	12	19	27

than those for the 22 MA reference case; this is because confinement improves with plasma current. The dominant energy loss mechanism for these plasmas is transport across the magnetic field. Consequently, it is essential to optimize the system to minimize plasma transport.

Representative parameters are shown for the optimum ion temperature for these 4 cases in Table 1. The transport losses dominate the total energy loss, with both synchrotron and bremsstrahlung considerably smaller. As is to be expected, these calculations are sensitive to uncertainties in confinement scaling, which essentially determine the "error bars" in these results. The total external heating power currently planned for ITER is 135 MW, which is sufficient for the injected power required. The ion densities are in the low 10^{13} cm^{-3} range. Consequently the slowing down time of the fast ions is comparable to the energy confinement time, but still less than the particle confinement time, which is assumed to be three times the energy confinement time. A potential problem with operation at the ion temperature for optimum Q is transport of the fast ions before they have given up their energy to the plasma by

Coulomb collisions. If this is a problem, then operation at a lower ion temperature (and therefore higher ion density) will reduce the slowing down time without serious reduction of Q.

Figure 5 shows the effect of variations in the H-mode multiplier used in calculating energy confinement. As expected, Q improves with the H-mode multiplier. Above an H-mode factor of about 2.5, however, no improvement in Q is seen. This is because the confinement time is limited by the ohmic confinement scaling expression. At the elevated temperatures and low density for D-^{3}He operation, the ohmic confinement time is relatively short.

In order to investigate the potential for D-^{3}He operation in ITER, we will focus on the 28 MA plasma current case, which gives considerably better performance than the 22 MA case. All of the above calculations are for a Troyon coefficient of 2.5. Because of the high temperatures the resulting density is less than that given by the density limits. Shown in Fig. 6 is the effect of varying the Troyon coefficient. Higher Q can be obtained by raising beta, but the required injection power also increases. This is because Q is small and the plasma is far from ignition.

In the above, we have not considered any modifications to ITER in order to improve its performance with D-^{3}He fuel. Reduction in the plasma volume can reduce the required injection power. Reduction in the major radius will also increase the magnetic field at the plasma and may improve the performance. One way of incorporating both of these effects without affecting the vacuum chamber is to reduce the major radius but keep the inside edge of the plasma at the same radius. Consequently, one reduces the horizontal half-width of the plasma at the same time. The vertical height can be held constant if one permits the elongation to increase. At constant q the plasma current increases slightly in this variation. Whether the poloidal field magnet system will permit this variation needs to be studied before the practicality of this variation can be determined. Shown in Fig. 7 is the effect of reducing the major radius by up to 40 cm in this manner. Q improves from 0.55 to about 0.8 and the required injection power is reduced because of the smaller plasma volume. The elongation varies from the

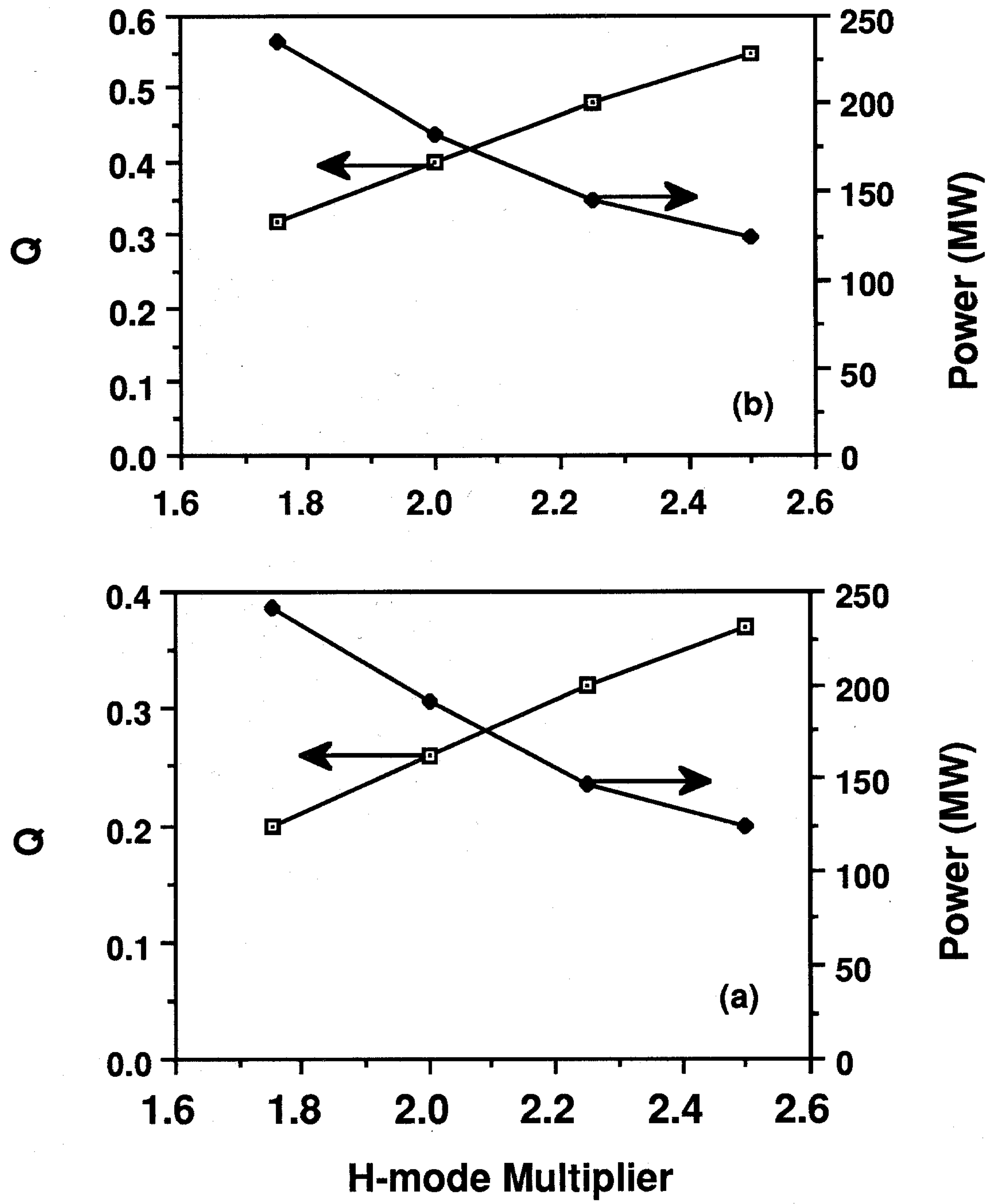

Fig. 5. Effect of the H-mode multiplier on energy multiplication and auxiliary power for 22 MA (a) and 28 MA (b) plasma current.

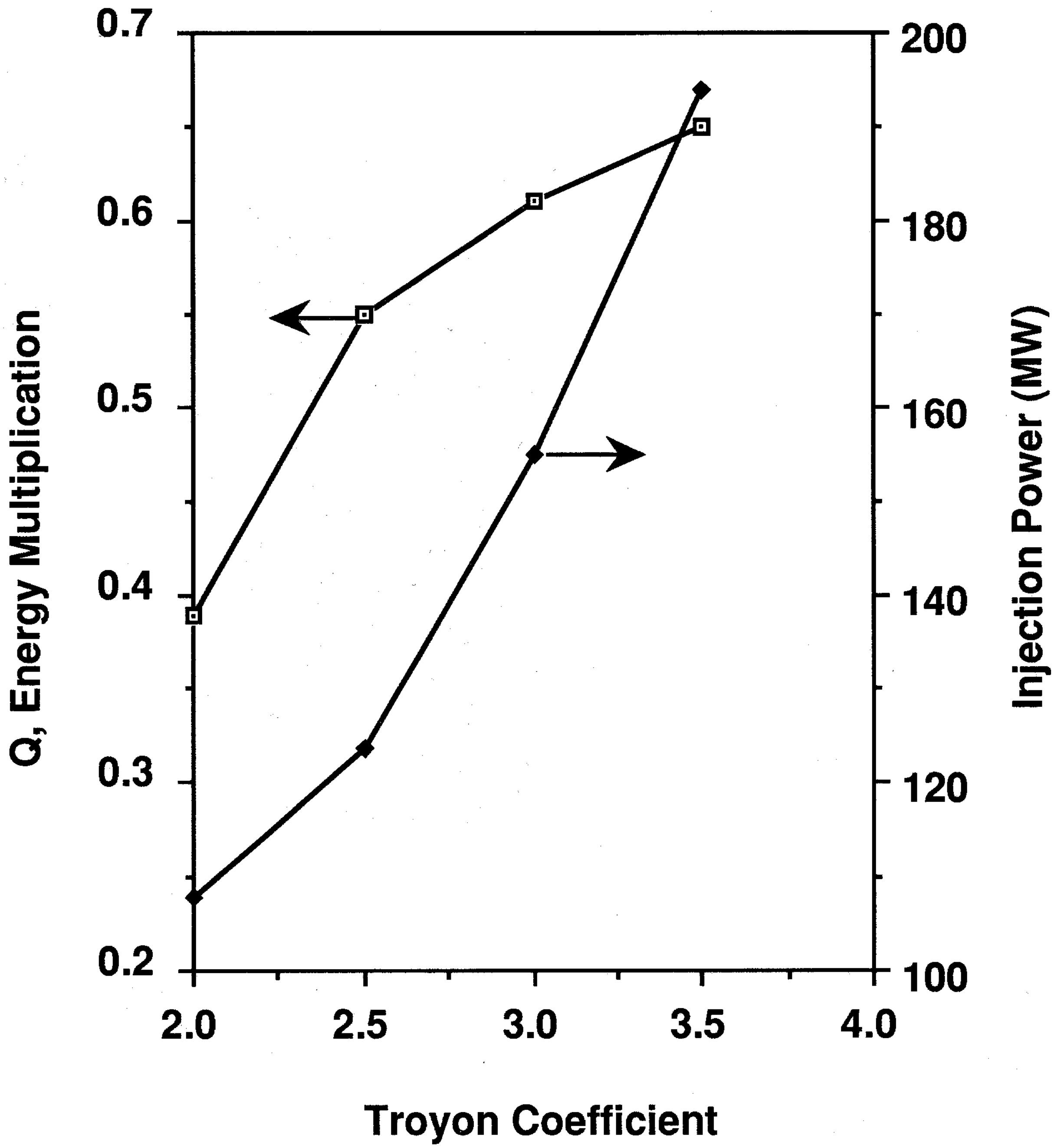

Fig. 6. Effect of beta on D-^{3}He energy multiplication for a plasma current of 28 MA. The H-mode multiplier is 2.5.

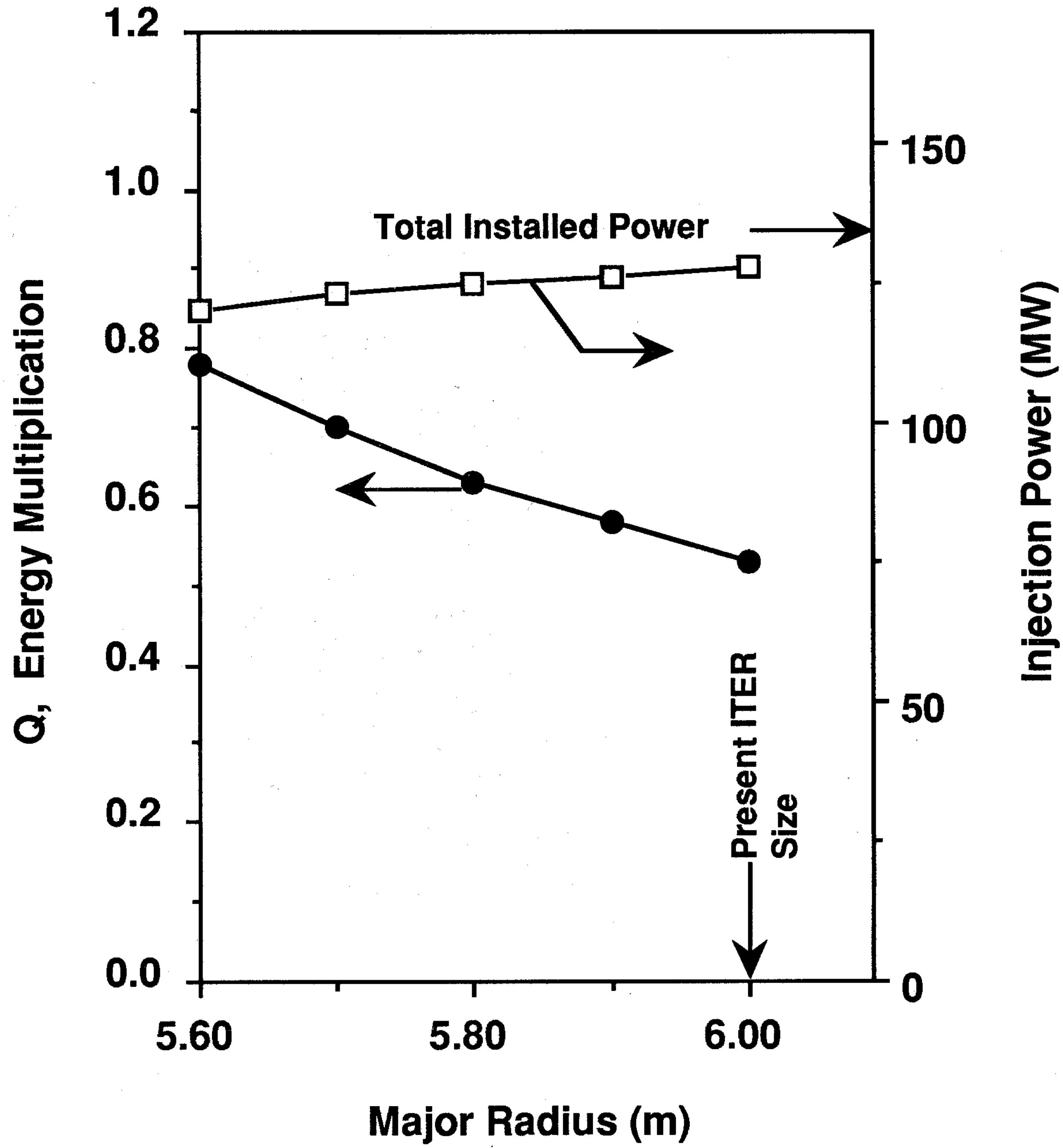

Fig. 7. Effect of decreasing the major radius on Q and the injection power. The H-mode factor and Troyon coefficient are both 2.5. The plasma current is 28 MA.

original value of 1.98 to about 2.4 and the plasma current increases from 28 MA to 29 MA as the major radius is reduced from 6.0 m to 5.6 m.

In order to achieve breakeven (Q=1) with D-^{3}He in ITER, further modification is required. One possibility is to increase the toroidal magnetic field. Figure 8 shows the effect of increasing the toroidal field at the magnet from 11 T to 13 T. The major radius for this calculation is 5.6 m, corresponding to the left boundary in Fig. 8. An increase of the toroidal field at the magnet from 11 to about 12.5 T is required to achieve breakeven conditions with an injection power less than the total external heating power available on ITER.

3.B D-^{3}He Operation with Tritium Seeding

An important limitation to achieving D-^{3}He breakeven in ITER is the power required to sustain the plasma at the necessary density and temperature. Since ITER is designed for D-T operation, seeding a D-^{3}He plasma with a small amount of tritium can be done and will increase the plasma reactivity substantially and reduce the amount of external power required. Figure 9 shows the effect of increasing the tritium fraction from zero to about 11%, at which point the alpha heating from D-T reactions equals the charged particle heating from the D-^{3}He reaction. The machine parameters for this calculation are those of an unmodified ITER with plasma current of 28 MA. We see that a plasma Q (ratio of charged particle heating in the plasma to injected power) of about 2 can be attained in this manner.

The resulting fast ion fraction is more complicated than for either pure D-^{3}He or pure D-T operation. At a tritium fraction of 11% the fusion products are 88 MW of 14.7 MeV protons and 22 MW of 3.67 MeV alpha particles from D-^{3}He reactions, and 110 MW of 3.5 MeV alpha particles from D-T reactions. The different energy of the two alpha groups is not significant since the alpha particles have about a 1 MeV energy spread and thus the two groups overlap to a large extent.

Since the plasma Q is about 2, fusion product heating is the largest power input in the power balance of the plasma. Consequently one can study fusion product heating, plasma

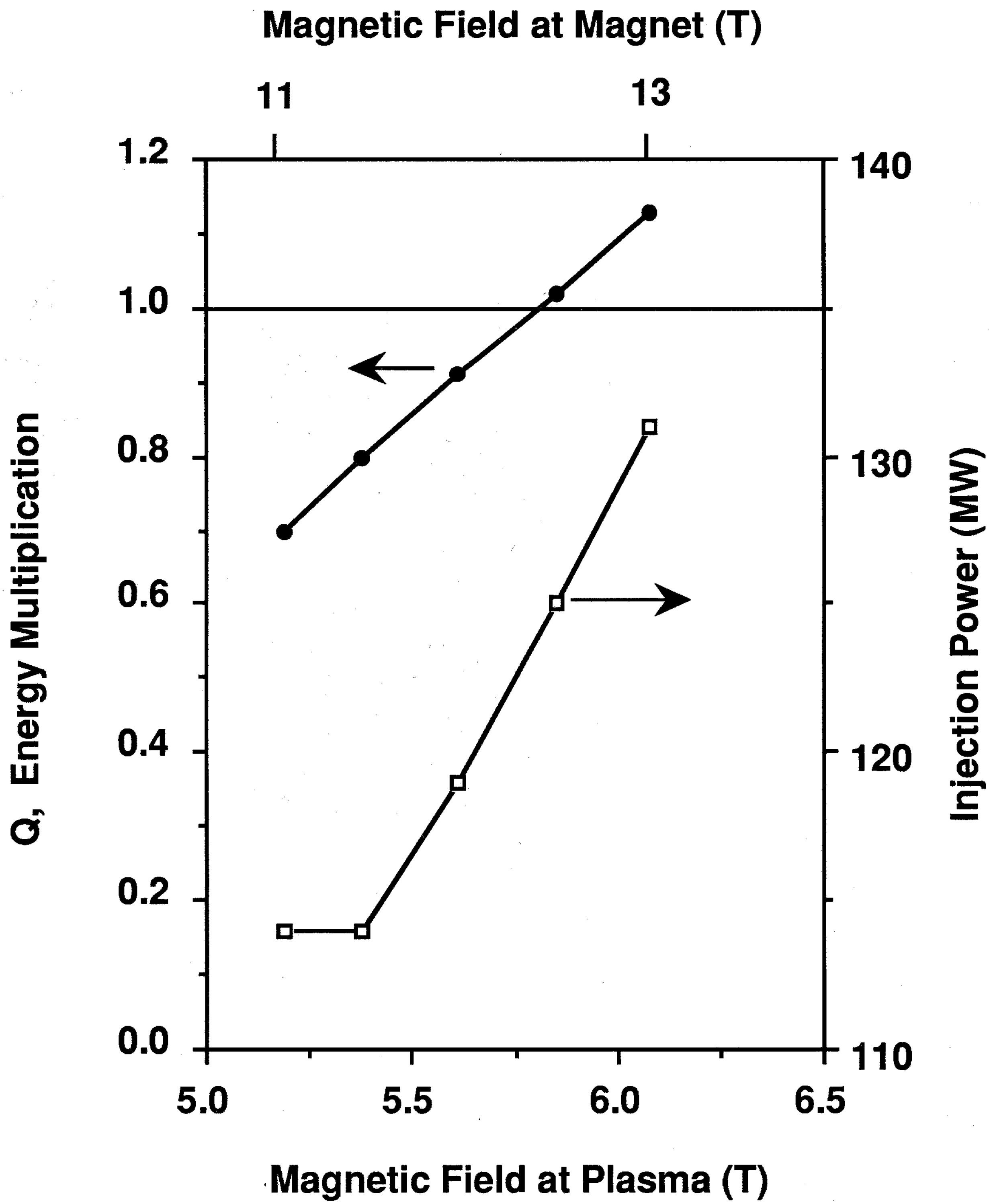

Fig. 8. Effect of increasing the magnetic field on the D-^{3}He Q and auxiliary heating power. The Troyon coefficient is 2.3 and the H-mode factor is 2.5. Breakeven is achieved at a magnetic field of 12.5 T at the magnet.

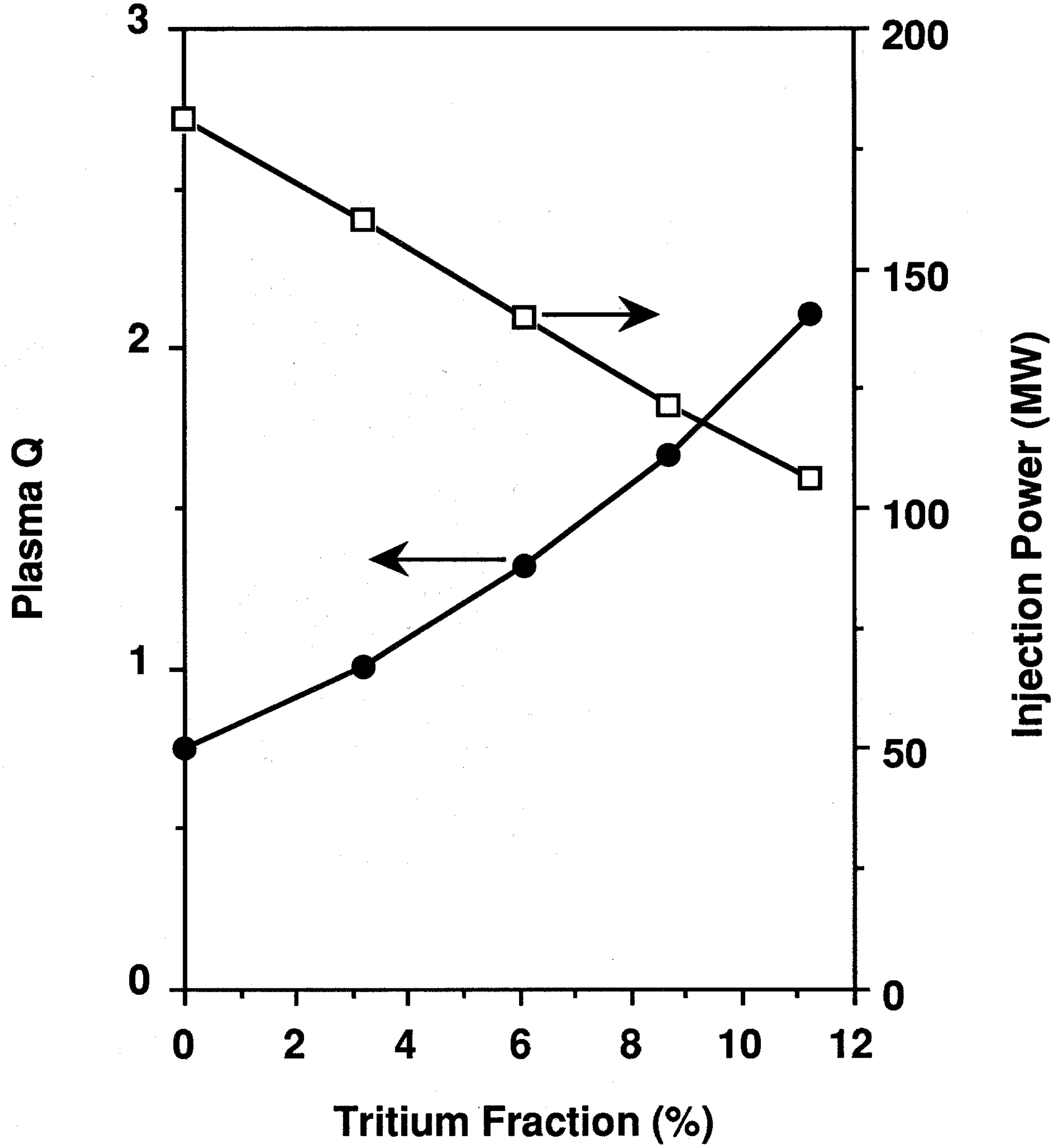

Fig. 9. Plasma Q (ratio of fusion product heating to injected power) with tritium seeding. The plasma current is 28 MA, the Troyon coefficient is 3.5, and the H-mode factor is 2.5.

transport, and confinement issues in a burning plasma at conditions appropriate for D-^{3}He operation. In this mode of operation the 14 MeV neutron power is 440 MW and the machine becomes contaminated with tritium. Consequently, this mode of operation should be considered as an intermediate step between pure D-^{3}He and pure D-T operation.

4. Conclusions

We have shown that an energy multiplication, Q, of about 0.3 can be obtained for D-^{3}He operation in CIT. The energy multiplication increases with beta; at Q = 0.3 the beta limit is reached and the injection power is 100 MW, which is the the maximum the machine is expected to be able to handle for pulse lengths of a few seconds.

We have shown that conditions for near breakeven D-^{3}He operation in ITER may be possible and that, with some modification to the machine, actual breakeven may be attained. In addition, seeding a D-^{3}He plasma with a small amount of tritium, will allow operation at a plasma Q of about 2 within the limits of the total amount of external heating power currently planned for ITER. Operating ITER with D-^{3}He can provide valuable data on confinement and transport of burning plasmas at D-^{3}He relevant conditions and determine whether a next step machine with the objective of D-^{3}He ignition is feasible.

Acknowledgement

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References

1. KULCINSKI, G.L., EMMERT, G.A., BLANCHARD, J.P., et al., "Apollo-L3, An Advanced Fuel Fusion Power Reactor Utilizing Direct and Thermal Energy Conversion," *Fusion Technology,* **19**, 791 (1991).

2. SCHMIDT, J.A., MONTGOMERY, D.B., "Status of CIT", *Fusion Technology,* **19**, 594 (1991).

3. SALPIETRO, E., "Overview of the ITER Project", *Fusion Technology*, **19**, 608 (1991).

4. EMMERT, G.A., EL-GUEBALY, L.A., KULCINSKI, G.L., et al., "Possibilities for Breakeven and Ignition of D-^{3}He Fuel in a Near Term Tokamak", *Nuclear Fusion*, **29**, 1427 (1989).

5. DENG, B.Q., EMMERT, G.A., "Fast Ion Pressure in Fusion Plasmas," UWFDM-718, Fusion Technology Institute, University of Wisconsin, Madison WI (1987).

6. UCKAN, N.A., "ITER Physics Design Guidelines," *Fusion Technology,* **19**, 1493 (1991).

7. GOLDSTON, R.J. "Energy Confinement Scaling in Tokamaks", *Plasma Physics and Controlled Fusion,* **26**, 87 (1984).

8. TRUBNIKOV, B.A., "Universal Coefficients for Synchrotron Radiation in Plasma Configurations" in *Reviews of Plasma Physics,* Vol. 7, Plenum Press, New York (1979) 345.

9. HEINDLER, M., NASSRI, A., KERNBICHLER, W., MILEY, G.H., GRATTON, F.T., "Synchrotron Radiation in Fusion Power Reactors", 1989 Annual Sherwood Theory Conference, San Antonio TX (1989), paper 2C25.