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POTENTIAL FOR D-³He OPERATION in ITER

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ABSTRACT

The potential for D-³He operation in ITER is examined. Without modifications to the device, except for the change of fuel, it is found that the energy multiplication, Q, of the order of 0.3 to 0.5 can be obtained using beta limits and confinement scaling expressions adopted for ITER. Breakeven with D-³He 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. This option allows one to study fusion product heating, plasma transport, and confinement issues in a burning plasma at conditions appropriate for D-³He operation.

1. INTRODUCTION

It is well known that D-³He fuel offers significant advantages for reducing 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-³He fuel can 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-³He fusion. In addition, the environmental advantages of a D-³He 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-³He operation in ITER in order to determine if the present understanding of tokamak physics will extrapolate to D-³He reactor conditions. This requires experimental testing at D-³He plasma conditions in order to study plasma transport and power balance, fueling mechanisms, impurity control, and other plasma considerations. The use of D-³He fuel in ITER 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-³He have been carried out in JET where a minority concentration of ³He in a predominantly deuterium plasma has been heated using ICRF. The enhanced tail of the ³He energy distribution driven by ICRF produced as much as 100 kW of D-³He 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 chapter, however, we wish to consider a different regime of D-³He operation, namely one where the D-³He fuel mixture is closer to 1:1 and the plasma is at temperatures appropriate for D-³He fusion. We will assume both the deuterium and the ³He are at the same temperature and have Maxwellian distribution functions. The question to be investigated is whether approximate breakeven conditions (D-³He 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. If breakeven-like conditions can be obtained, then one can study physics questions associated with a burning D-³He plasma before contaminating the device with tritium. Of equal importance is that the knowledge gained about D-³He fusion will determine if it is feasible to consider a D-³He ignition machine as the next step in the road to fusion.

2. CALCULATIONAL MODEL

The assessment of D-³He operation in ITER 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-³He fuel. In particular, the higher operating temperature of D-³He 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, ³He, protons, ⁴He, 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 ${}^3\text{He}(d,p){}^4\text{He}$, $\text{D}(d,n){}^3\text{He}$, $\text{D}(d,p)\text{T}$, and $\text{T}(d,n){}^4\text{He}$. 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 [2]. The operating density is determined by density limits or beta limits, whichever is applicable, and the beta limit is determined by the Troyon formula. 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 [3]; 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 power law empirical scaling expression for energy confinement), 2) synchrotron radiation (the Trubnikov expression as modified by Tamor for high temperature is applied locally in the plasma and volume-averaged [4,5]), 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 [3]. 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.

The figure of merit used to evaluate the performance of ITER with $\text{D}-{}^3\text{He}$ 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 $\text{D}-{}^3\text{He}$

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-³He reaction, whereas it is only 20% for the D-T reaction.

3. PURE D-³He OPERATION

First we consider pure D-³He operation of ITER. Later we consider operation in a mixed mode where a small amount of tritium is added to a D-³He plasma. Shown in Fig. 1 is the energy multiplication, Q , versus density weighted volume averaged ion temperature (all temperatures quoted in this chapter are density weighted and volume averaged). 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 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

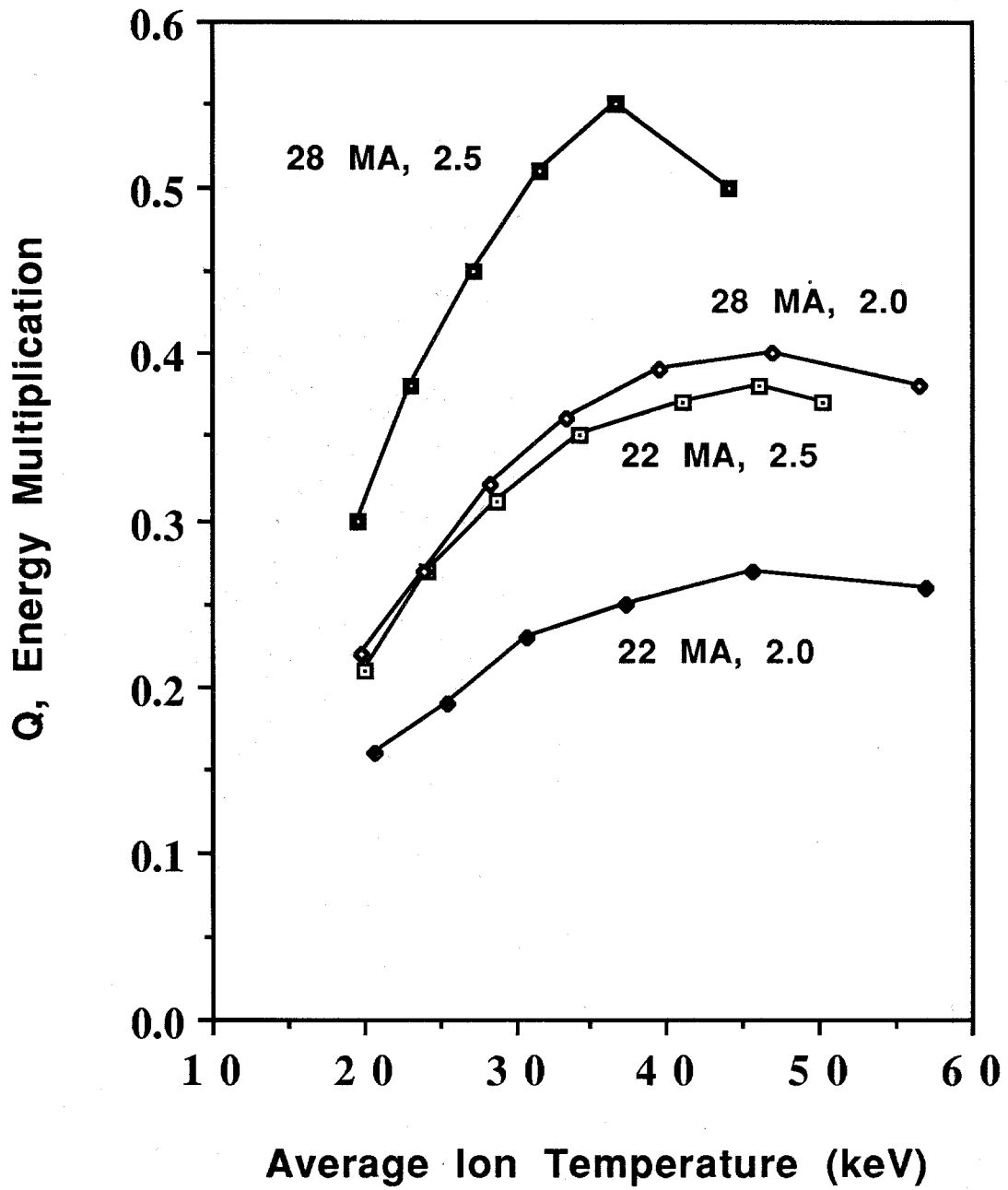


Fig. 1. Variation of Q with average ion temperature. The four cases are labeled by the plasma current and the H-mode multiplier.

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.

TABLE 1. REPRESENTATIVE PARAMETERS FOR D-³He OPERATION

Plasma current (MA)	22	22	28	28
H-mode multiplier	2.0	2.5	2.0	2.5
Q	.27	.38	.40	.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
τ_E (s)	3.4	5.4	4.3	6.7
Slowing down time (s)	5.8	6.5	5.6	6.5
Particle conf. 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

Figure 2 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-³He operation, the ohmic confinement time is relatively short.

In order to investigate the potential for D-³He 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. 3 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-³He 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

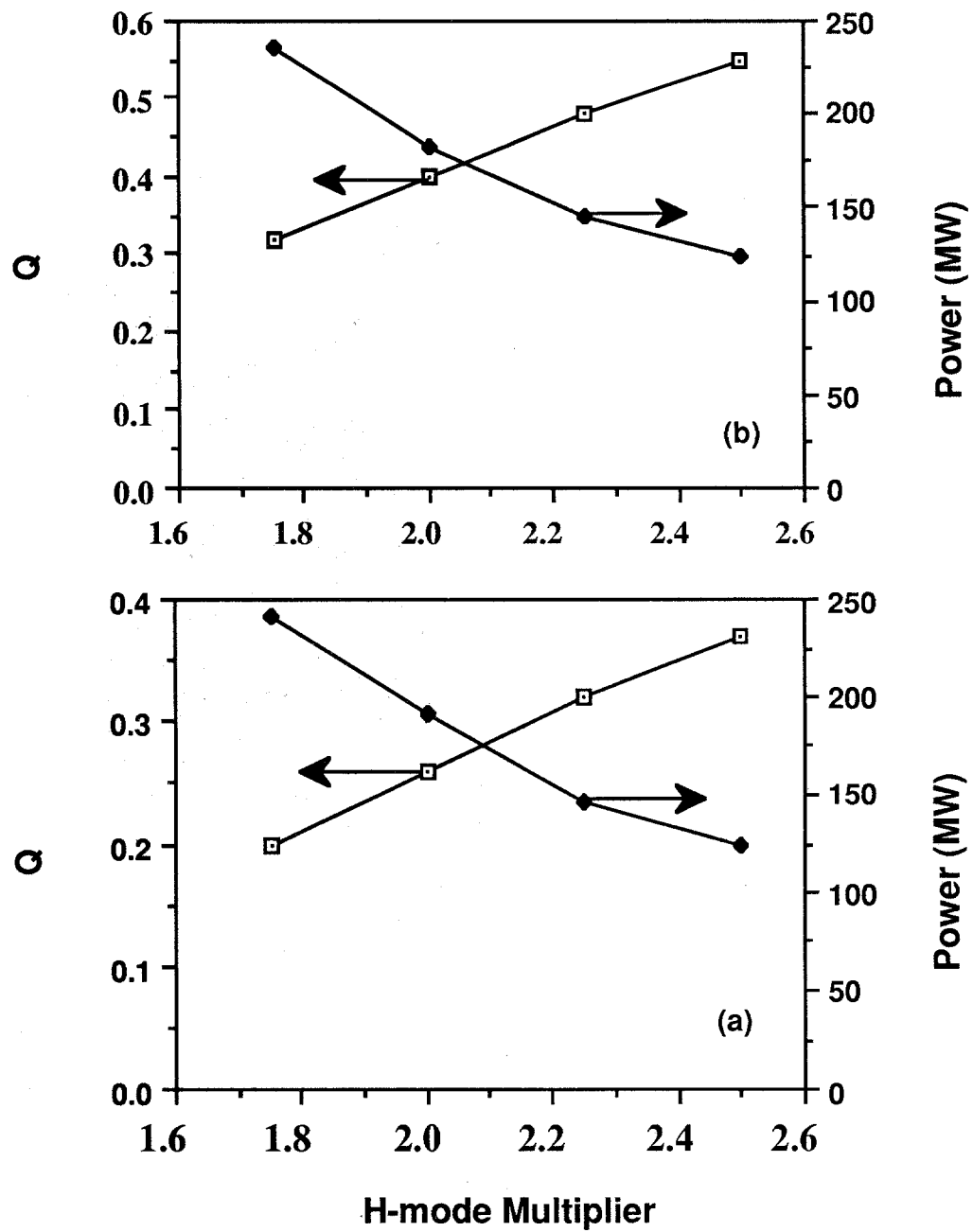


Fig. 2. Effect of the H-mode multiplier on energy multiplication and auxiliary power for 22 MA (a) and 28 MA (b) plasma current.

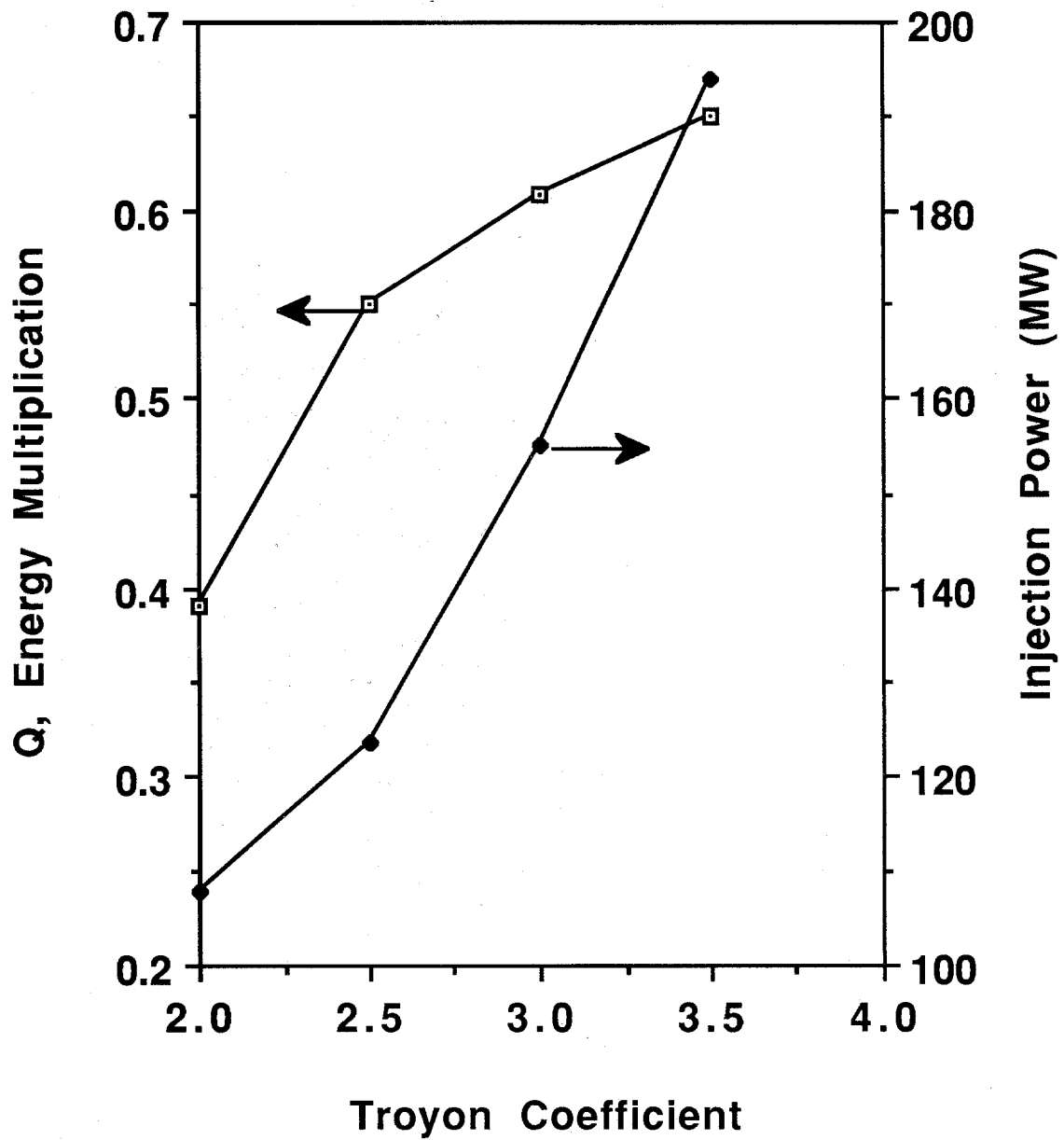


Fig. 3. Effect of beta on D-³He energy multiplication for a plasma current of 28 MA. The H-mode factor is 2.5.

of this variation can be determined. Shown in Fig. 4 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 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 $D-^3He$ in ITER, further modification is required. One possibility is to increase the toroidal magnetic field. Figure 5 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. 4. 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.

4. $D-^3He$ OPERATION WITH TRITIUM SEEDING

An important limitation to achieving $D-^3He$ 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 $D-^3He$ 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 6 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 $D-^3He$ 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.

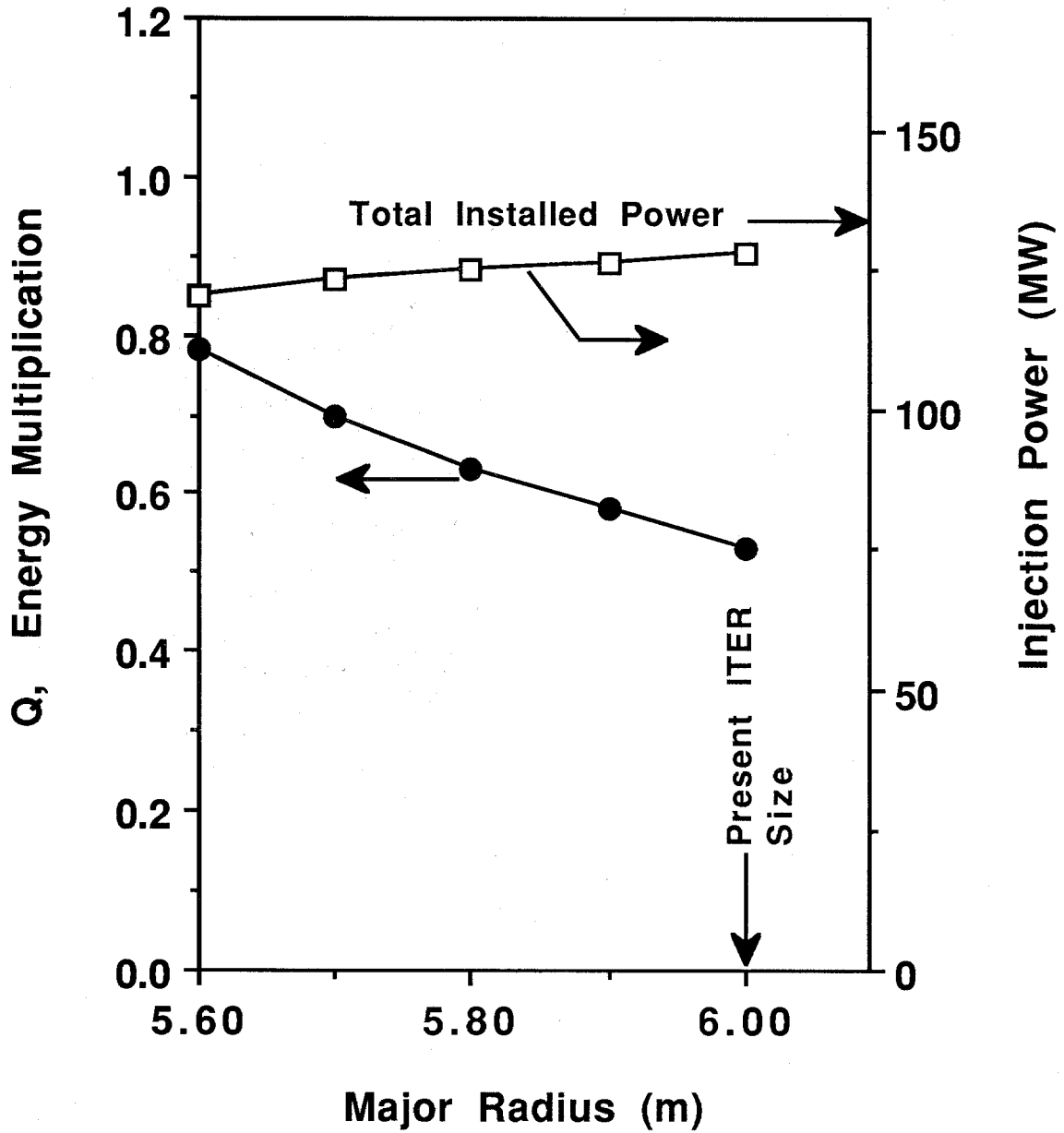


Fig. 4. Effect of decreasing the major radius on Q and the injection power. The H-mode factor and Troyon coefficient are both 2.5. The plasma current is 28 MA.

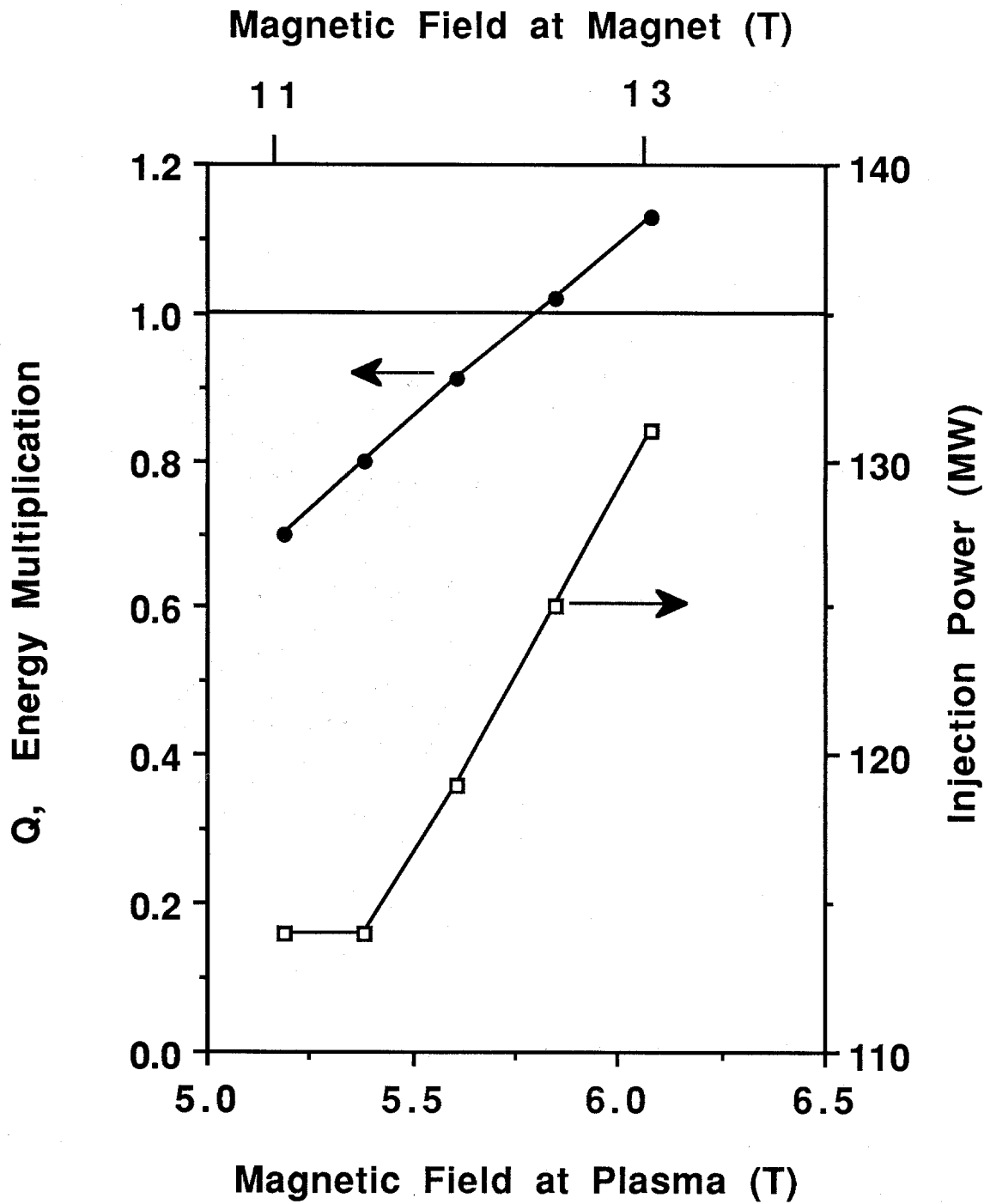


Fig. 5. Effect of increasing the magnetic field on the D-³He Q and injection power. The Troyon coefficient is 2.3 and the H-mode factor is 2.5. Breakeven is achieved at a magnetic field of 12.5 T at the magnet.

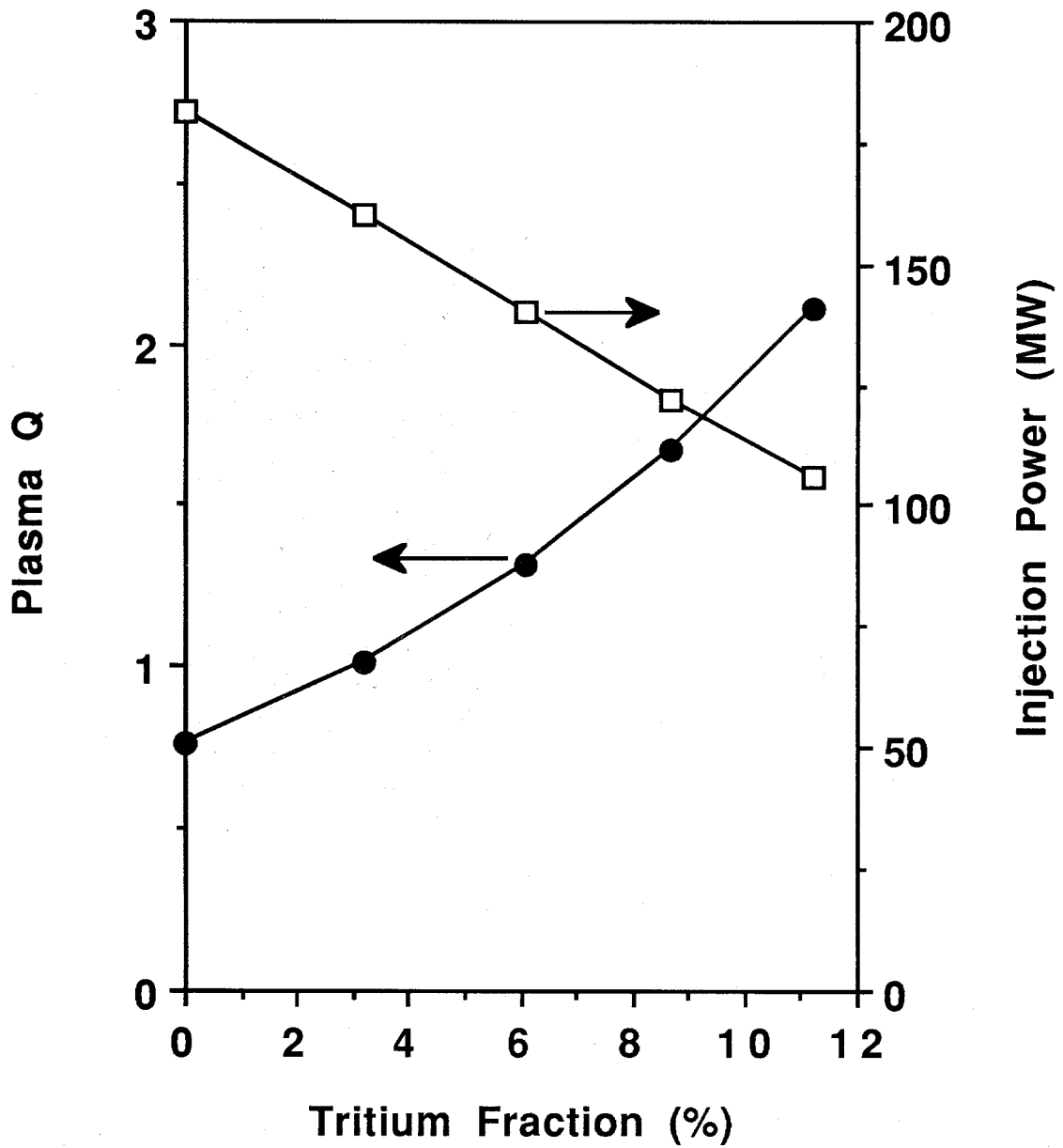


Fig. 6. Plasma Q (ratio of fusion product heating to injected power) with tritium seeding. The plasma current is 28 MA, the Troyon coefficient is 3.5, and the H-mode factor is 2.5.

The resulting fast ion fraction is more complicated than for either pure D-³He 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-³He 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 transport, and confinement issues in a burning plasma at conditions appropriate for D-³He 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-³He and pure D-T operation.

5. CONCLUSIONS

We have shown that conditions for near breakeven D-³He operation in ITER may be possible and that, with some modification to the machine, actual breakeven may be attained. In addition, seeding a D-³He 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-³He can provide valuable data on confinement and transport of burning plasmas at D-³He relevant conditions and determine whether a next step machine with the objective of D-³He ignition is feasible.

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