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HIGHER FUSION POWER GAIN WITH PROFILE CONTROL IN DIII-D TOKAMAK PLASMAS

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ABSTRACT. Strong shaping, favourable for stability and improved energy confinement, together with a significant expansion of the central region of improved confinement in negative central magnetic shear target plasmas, increased the maximum fusion power produced in DIII-D by a factor of 3. Using deuterium plasmas, the highest fusion power gain, the ratio of fusion power to input power, Q, was 0.0015, corresponding to an equivalent Q of 0.32 in a deuterium–tritium plasma, which is similar to values achieved in tokamaks of larger size and magnetic field. A simple transformation relating Q to the stability parameters is presented.

Recent experiments investigating the confinement properties of tokamak plasmas with negative central magnetic shear (NCS) [1–3] have demonstrated greatly improved core confinement, which reaches levels comparable to neoclassical predictions for the ion heat conduction. Building on earlier studies, which showed the important role played by strong shaping [4] in improving plasma stability, together with more recent calculations [5–7], which predict that a central region with negative shear in the safety factor will enhance plasma stability, experiments were carried out on DIII-D that significantly expand the fractional volume of plasma with improved core confinement to produce much higher fusion power gain, up to $Q_{DD} = 0.0015$, in deuterium plasmas. This value of Q_{DD} corresponds to an equivalent Q in a deuterium–tritium plasma, $Q_{DT} = 0.32$.

Normalized to the square of the toroidal field times the major radius, $B_t^2 R^2$, the fusion gain results reported here are between 2 and 9 times larger than those achieved in other tokamaks. The product $B_t R$, proportional to the current through the centre post, is an important quantity in the tokamak as the dominant limitation of device construction is the stress limitation on this current. We will later show that $B_t R$ readily relates Q to the confinement and MHD stability properties of the tokamak. These results offer the prospect of reduction in the size and field required for achieving higher gain, approaching fusion ignition conditions in a plasma, and support the viability of the concept [7] of a smaller, economically attractive tokamak reactor [8] through tailoring the equilibrium profiles.

It is well known that both fusion reactivity and plasma stability are sensitive to the form of the

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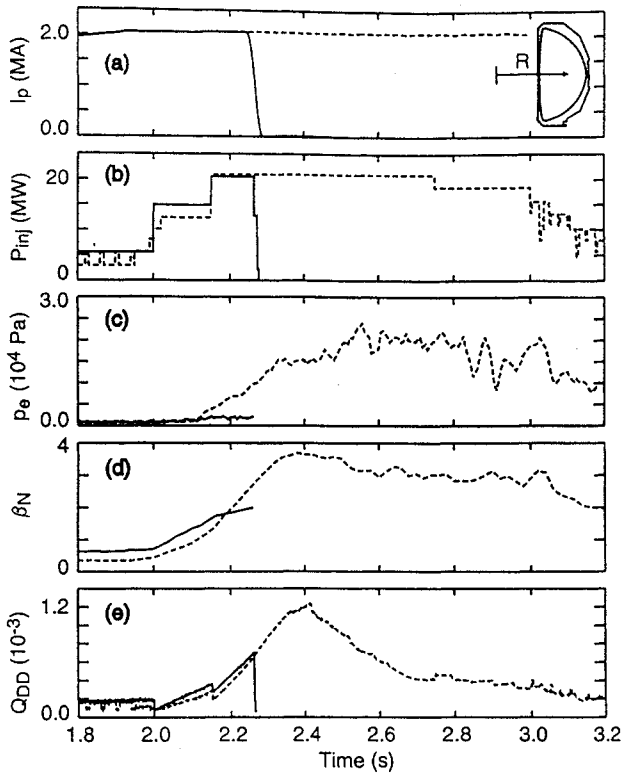[i] INRS — Énergie et Matériaux, Varennes, Quebec, Canada.

FIG. 1. Time evolution of two similar discharges, 87 887, which remains in L mode (solid line) and disrupts, and 87 937, which makes a transition to H mode (broken line): (a) plasma current, (b) injected neutral beam power, (c) edge electron pressure, (d) β_N $(= \beta a B_t / I_p)$ in units $\% \cdot m \cdot T/MA$, (e) $Q_{DD} = P_{fusion}/P_{NB}$.

FIG. 2. Pressure profiles for discharges 87 887 (L mode) and 87 937 (H mode) at 2.245 s, showing the broadening of the H mode versus ρ, the square root of toroidal normalized flux. The typical statistical error is 10%.

pressure profile. Since an H mode plasma in DIII-D is characterized by a transport barrier at the plasma edge leading to a broader pressure profile, the presence of an edge barrier (H mode) or its absence (L mode) provides a degree of pressure profile control. Earlier experiments have shown improved core confinement with NCS plasmas in both L mode [9] and H mode [2] DIII-D plasmas. However, L mode NCS plasmas with strongly peaked pressure profiles were found to disrupt at β_N values about a factor of 2 less than the values achieved in H mode [2, 10]. This lower beta limit in L mode is consistent with ideal MHD stability limits, and broadening the pressure profile is predicted to enhance stability and result in a large increase in plasma reactivity for strongly shaped plasma cross-sections [11]. Experimental confirmation of these results, by demonstrating this increase in reactivity, was a strong motivating force for these experiments, where the L–H transition timing is used strategically to moderate the peaking of the pressure profile. This controlled
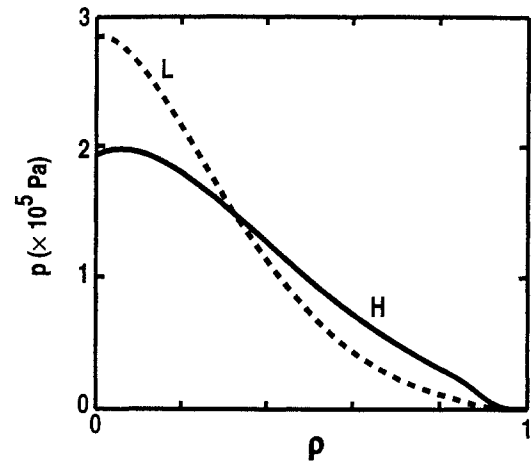
transition has led to record reactivity for DIII-D plasmas, with Q_{DD} reaching values comparable to those in the larger, higher magnetic field tokamaks, JET [12], JT-60U [13] and TFTR [14].

The increase in achievable β and Q through a controlled L–H transition is shown in Fig. 1 where the evolutions of an L mode and of an H mode plasma are compared. Low power neutral beam injection (NBI) beginning at 0.3 s produces the NCS target [15]. Small, controlled changes in plasma shape induce an H mode transition in one case at 2.1 s, indicated by the edge pressure rise (Fig. 1(c)). The L mode case disrupts at about 2.25 s (Fig. 1(a)). The H mode plasma continues to increase its stored energy and fusion reaction rate until a stability limit is reached at $\beta_N = 3.7$. For this particular case, Q_{DD} reached 0.0012. The high performance phase is terminated by a global, β-limiting instability associated with the buildup of bootstrap-driven current density near the plasma edge [16], whereupon the plasma reverts to an ELMing H mode. The broadening of the pressure profile after the L–H transition is shown in Fig. 2, where profiles are shown just prior to the disruption of the L mode plasma and 0.125 s after the L–H transition for the H mode case.

The highest Q_{DD} discharge (87 977, see Table I) was used as the basis for projecting the reactivity of a deuterium–tritium plasma under these conditions. The evolution of discharge 87 977 is similar to that of discharge 87 937, shown in Fig. 1. DT simulations based on discharge 87 977, using the TRANSP [17] analysis code, predict $Q_{DT} = 0.32$, estimated as

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Table I. Parameters of Discharge 87 977 at the Time of the Peak Neutron Rate

$I_p = 2.25$ MA	$S_n = 2.2 \times 10^{16}$ neutrons/s
$B_t = 2.15$ T	$W = 4.2$ MJ, $\tau_E = 0.4$ s
$R = 1.67$ m	$P_{NBI} = 17.75$, $dW/dt = 7.4$ MW
$a = 0.61$ m	$T_i(0), T_e(0) = 18.1, 7.5$ keV
$\kappa = 2.15$ (elongation)	$n_e(0), n_D(0) = 10.0, 8.5 \times 10^{19}$ m^{-3}
$V_p = 22$ m^3	$\tau_E/\tau^{ITER\text{-}89P} = 4.5$
$\beta = 6.7\%$, $\beta_N = 4.0$	$q_{95} = 4.2$
$p(0) = 0.33$ MPa	Thermonuclear neutron fraction: 0.6

the product of the ratio $Q_{DT}/Q_{DD} = 222$ from the simulations and $Q_{DD} = 0.0015$ from the measured neutron emissivity. The measured profiles of discharge 87 977 are maintained in the simulation and the deuterium is replaced with a nominal 50:50 mixture of deuterium and tritium. Predicted in this way, the Q_{DT} value is found to be insensitive to uncertainties in the plasma equilibrium. The dominant uncertainty in the Q_{DD} value remains the 15% statistical uncertainty in the measurement of the neutron rate, S_n. The value $S_n = 2.2 \times 10^{16}$ s^{-1} in Table I was measured with a calibrated scintillation counter. Values of $S_n = 2.3 \times 10^{16}$ s^{-1} measured with a fission product counter and $S_n = 2.5 \times 10^{16}$ s^{-1} from a TRANSP calculation using the measured temperature, density and rotation profiles provide confidence in these results.

The scaling of Q with global parameters comes about in the following way. For ion temperatures, T_i, in the range of interest, the fusion reaction rate scales approximately as T_i^2; hence, the fusion power scales as the square of the plasma pressure leading to the relation $Q_{DD} \equiv (P_{fus}/P_{NBI}) \propto f_p \beta \tau_E B_t^2$, where f_p is a profile factor $(\langle p^2 \rangle/\langle p \rangle^2)$, which increases with stronger peaking of the pressure profile. In Fig. 3, we show good correlation of $\beta \tau_E$ with Q_{DD} in this experiment. Most of the scatter in Fig. 3 arises from the definitions of Q_{DD} with a denominator of input power (P_{NBI}) and of τ_E with a denominator of loss power $(P_{NBI} - \dot{W})$, which differ under transient conditions.

We wish to express Q_{DD} in a way that incorporates the fundamental stability constraints of the tokamak, axisymmetric and kink stability. The strong shaping of DIII-D plasmas is a crucial factor in producing Q_{DD} values comparable to those of larger, higher magnetic field tokamaks. To illustrate this, we relate Q_{DD} to plasma geometry using an effective
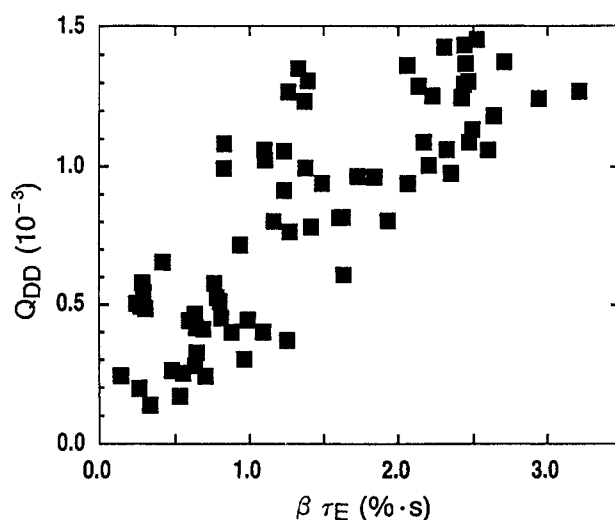

FIG. 3. Q_{DD} versus $\beta \tau_E$ for all the plasmas in this experiment. The absolute error in the neutron rate is 15%. The relative error among the data is only a few per cent. The $\beta \tau_E > 2$ values are the highest ever achieved in DIII-D. $B_t = 2.15$ T, $2.0 \leq I_p \leq 2.4$ MA, $\beta \leq 7.0\%$.

inverse aspect ratio for the torus as a shape descriptor, $E \equiv q(\mu_0 I_p/2\pi a B_t)$. (A more physical discussion of E and its relationship to axisymmetric stability is contained in Ref. [18].) For diverted plasmas q_{95} is substituted for q. A simple scaling for global confinement time in ELM-free H mode plasmas [19], DIII-D/JET scaling can be approximated as

$$\tau_E \equiv W/(P_{NBI} - dW/dt)$$
$$= 0.11 F \left(I_p R^{3/2}/\sqrt{P_{NBI} - \dot{W}} \right) \text{ (MA, m, MW, s)}$$

where F is an enhancement factor over the original relation. Combining the above relations we find $Q_{DD} \propto f_p[B_t R(E/\sqrt{\kappa})(F/q)]^2$. The fusion power gain increases with the pressure profile form factor

Table II. Comparison of DD Fusion Reactivity for Several Tokamaks

Tokamak	DIII-D	TFTR Ref. [14]	JT-60U Ref. [13]	JET Ref. [12]	DIII-D Ref. [20]
Discharge	87 977	68 522	17 110	26 087	78 136
B (T)	2.15	5.00	4.40	2.80	2.15
R (m)	1.67	2.50	3.05	2.95	1.68
$E/\kappa^{1/2}$	0.98	0.35	0.39	0.60	0.84
F	2.4	1.2	1.8	2.3	2.1
q	4.2	3.8	4.0	3.8	5.1
τ_E (s)	0.40	0.19	0.54	1.30	0.23
β (%)	6.7	1.0	1.5	2.2	4.8
$\langle p^2 \rangle / \langle p \rangle^2$ [a]	1.6	3.0	2.0	1.8	1.6
Q^*_{DD}	0.0020	0.0021	0.0037	0.0051	0.0006

[a] Estimated from the ratio of peak pressure to average pressure. A radial profile of the form $p(\rho) \propto (1 - \rho)^\alpha$ is assumed, yielding $\alpha = p_0/\langle p \rangle - 1$ and $f_p = \langle p^2 \rangle / \langle p \rangle^2 = (\alpha + 1)^2/(2\alpha + 1)$, where $\langle\rangle$ denotes the volume average.

f_p, and as the square of the magnetic field strength B_t, machine size R, shaping $E/\sqrt{\kappa}$, inverse safety factor q^{-1} and energy confinement enhancement F. All but F are determined by low-n stability considerations.

These parameters for determination of Q_{DD} are displayed in Table II for discharge 87 977, compared with an earlier high performance VH mode plasma [20, 21] in DIII-D as well as published data from other tokamaks. The dramatic improvements over earlier DIII-D results in fusion gain produced in these experiments derive in approximately equal measure from improved shape factor, lower q and improved confinement. For purposes of comparison with other published values we have plotted

$$Q^*_{\mathrm{DD}} \equiv P_t/(P_{\mathrm{NBI}} - \dot{W}) + (P_{\mathrm{bt}} + P_{\mathrm{bb}})/P_{\mathrm{NBI}}$$

where P_t, P_{bt} and P_{bb} are the fusion powers from thermal–thermal, beam–thermal and beam–beam reactions, respectively. Here, both the thermonuclear fusion reaction rate and the energy confinement time are referenced to the input power that would be required to sustain the plasma's thermal energy in steady state. As shown in Table II, DIII-D has smaller B_t and R than the other tokamaks listed, but this is counterbalanced by the strong shaping and associated enhanced confinement [20] that allow it to operate at higher beta with modest input power. To more clearly demonstrate how these results extrapolate to requirements for achieving higher gain approaching fusion ignition conditions in a plasma, we separate the
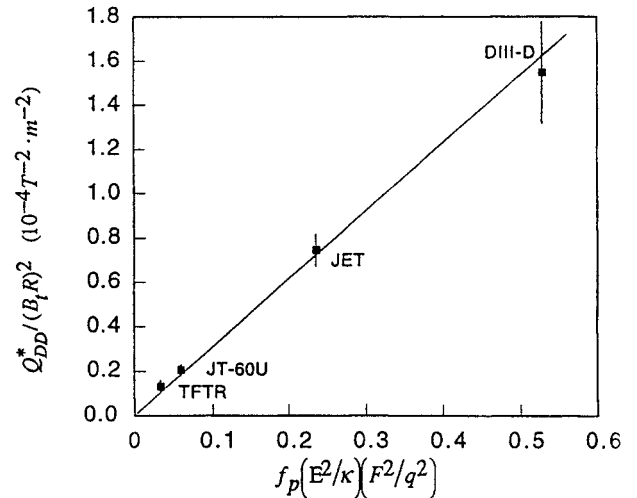

FIG. 4. $Q^*_{\mathrm{DD}}/(B_t R)^2$ versus the scaling relation discussed in the text for the highest DIII-D Q_{DD} and the highest values reported at TFTR, JET and JT-60U.

parametric dependences of Q_{DD} into primarily economic and technological factors, $B_t^2 R^2$, and primarily plasma control parameters, $f_p(E^2/\kappa)(F^2/q^2)$. Calculations similar to those in Ref. [8], with the reactor design systems code (SuperCode), show that the capital cost of the tokamak reactor core increases approximately linearly with $B_t^2 R^2$. As shown in Fig. 4, the ratio of fusion gain to this cost factor, $Q^*_{\mathrm{DD}}/(B_t^2 R^2)$, for the highest performance plasma in each device is adequately described by this simple expression. Details such as impurity concentration, individual
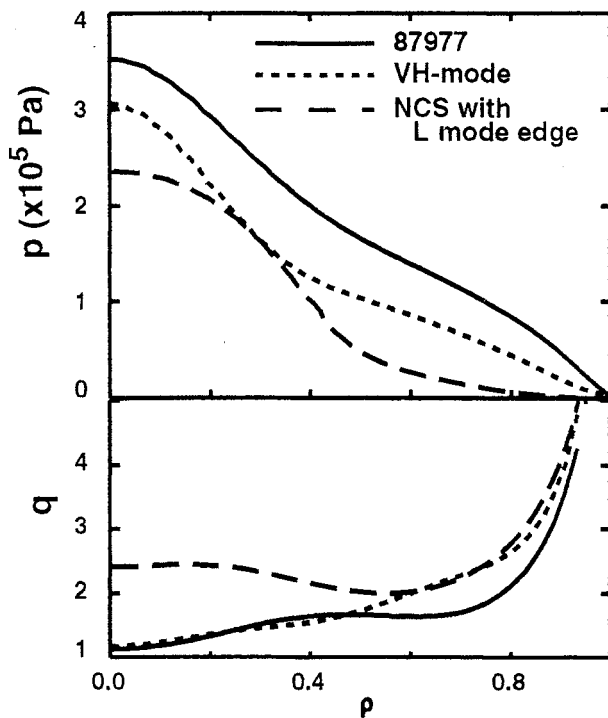
FIG. 5. Pressure and safety factor profiles at the time of peak neutron emission.

form factors for temperature and density profiles, T_i/T_e, the thermal fraction of fusion power and neutral beam deposition profile are sufficiently similar to be ignored. Additionally, all are free of sawteeth. The confinement factor, F, is determined from the experimental data and no predictive capability is implied. For plasmas discussed here, the dominant power flow is through the ion channel which has confinement comparable to the neoclassical level [22] and an F value of 2.4. Thus, one does not anticipate much further enhancement of this quantity. The range of F in modern tokamaks is from about 1 to about 2. The remaining factors are bounded by ideal MHD stability. F is known [20] to exhibit strong dependence on E, q and the neutral pressure in the vessel. The relative importance of the individual terms in the abscissa is shown in Table II.

Our present results represent a first attempt at control of the pressure profile in conjunction with current profile control. The L–H transition was used to provide stability through pressure broadening. Consistent with this, the observed limit in β_N is raised to about 4 from about 2 in the L mode. It is not known at present whether the H mode is playing a synergistic role as well in increasing the volume of plasma having

reduced core transport. In Fig. 5, we show the pressure and safety factor profiles for discharge 87 977, the highest Q_{DD} plasma. For comparison, we show the VH mode, which produced the highest neutron rate (discharge 78 136 in Table II), and one of the best L mode plasmas. In comparison with the VH mode, we see that the region of low shear is expanded. The VH mode is calculated to be second stable [23] within $\rho = 0.37$, whereas discharge 87 977 is second stable out to $\rho = 0.7$. This appears to be reflected in the pressure profile. The most dramatic difference is in the width of the ion temperature profile. The density profile is somewhat more peaked than for a VH mode, but remains very broad. While the L mode NCS plasma also has a large region with negative to neutral magnetic shear, as mentioned before, these plasmas with an L mode edge which show an internal transport barrier characteristically terminate in disruption at β_N about 2.

Future experiments will attempt to regulate the heating power to avoid beta limits while exercising improved edge gradient control to maintain an attractive pressure profile and thereby extend these results to quasi-stationary operation. In a reactor embodiment of the tokamak, similar plasmas should be sustainable in steady state with modest radiofrequency driven current requirements [2]. These results are favourable for scaling to a compact fusion reactor [24].

In summary, fusion gain can be increased in four ways: increased size and field, $B_t^2 R^2$, increased shaping, E^2/κ, improved confinement, F^2, and increased peaking of the pressure profile, f_p. By taking advantage of these improved stability properties that come from strong shaping and pressure profile control, combined with the improved confinement achieved in plasmas with negative central shear when high power neutral beam heating is applied, DIII-D has achieved high values of $Q_{DD}/(B_t^2 R^2)$.

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