Observation of an Improved Energy-Confinement Regime in Neutral-Beam–Heated Divertor Discharges in the DIII-D Tokamak

K. H. Burrell, S. Ejima, D. P. Schissel, N. H. Brooks, R. W. Callis, T. N. Carlstrom, A. P. Colleraine, J. C. DeBoo, H. Fukumoto,[a] R. J. Groebner, D. N. Hill,[b] R.-M. Hong, N. Hosogane,[c] G. L. Jackson, G. L. Jahns, G. Janeschitz,[d] A. G. Kellman, J. Kim, L. L. Lao, P. Lee, J. M. Lohr, J. L. Luxon, M. A. Mahdavi, C. P. Moeller, N. Ohyabu, T. H. Osborne, D. Overskei, P. I. Petersen, T. W. Petrie, J. C. Phillips, R. Prater, J. T. Scoville, R. P. Seraydarian, M. Shimada,[c] B. W. Sleaford, R. T. Snider, R. D. Stambaugh, R. D. Stav, H. E. St. John, R. Stockdale, E. J. Strait, T. Taylor, J. F. Tooker, and S. Yamaguchi[e]

GA Technologies Inc., San Diego, California 92138

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Tokamak discharges using the expanded boundary divertor in the DIII-D device exhibit *H*-mode confinement. With neutral-beam power up to 6 MW, energy confinement remains comparable to the Ohmic value at a plasma current of 1 MA. Confinement is also independent of plasma density and toroidal field. Confinement increases with plasma current, but the exact functional dependence is, as yet, uncertain. These results show that the *H* mode can be achieved in a reactor-compatible open divertor configuration.

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One goal of controlled thermonuclear fusion research is to create an ignited plasma in which thermal losses are balanced by fusion power. Although several large tokamaks have been built in recent years whose goal is to approach this ignition condition,[1-3] the energy-confinement results from the auxiliary heating experiments in these[4-10] and smaller devices[11,12] have been somewhat disappointing, because the application of either neutral-beam or radio-frequency heating has caused a degradation in the energy confinement time τ_E from the values achieved with Ohmic heating alone. This pattern of decreasing τ_E with increasing auxiliary power has come to be known as low-mode (*L* mode) confinement.

A series of experiments in the ASDEX device[13-19] has shown that a plasma condition of high confinement (*H* mode) can be achieved in which there is little, if any, degradation of confinement with auxiliary heating.[13,15-18] This result was obtained in poloidal-divertor plasmas that were formed by use of a triplet divertor coil set located inside the main vacuum chamber in the ASDEX device.[20] Although the ASDEX results have been of great significance for fusion research, the ASDEX device is too small to reach ignition and the divertor configuration is rather complicated for use in an ignition device or a reactor. Consequently, it is important to demonstrate that good, *H*-mode confinement can be obtained in a larger device with a more reactor-compatible divertor geometry.

The DIII-D tokamak[21-23] is equipped with a simplified poloidal divertor, the expanded boundary divertor,[24-27] whose design is much more compatible with the engineering requirements for an ignition device or a reactor.[28-30] Recent neutral-beam–heating experiments in DIII-D have shown the existence of the *H* mode. Up to the maximum neutral-beam power studied to date (6 MW), the energy-confinement time during the *H* mode is at or above the Ohmic level. Accordingly, our results demonstrate that good, *H*-mode confinement can be achieved in a device significantly larger than ASDEX with a reactor-compatible divertor configuration.

The DIII-D tokamak (major radius $R=1.67$ m, minor radius $a=0.67$ m, toroidal field $B_T \leq 2.2$ T, vessel elongation $=2.0$) was created by upgrading of the Doublet III device.[21,22] This paper discusses results from the single-null divertor configuration with the null at the bottom of the plasma. Deuterium plasmas with a vertical elongation of 1.8 were used for these experiments. The plasma was heated by two neutral beam lines, which injected hydrogen beams in the direction of the plasma current at an average angle of 55° with the magnetic axis. Maximum neutral-beam power was 6 MW, beam energy was 75 keV, and maximum pulse length for each of the four neutral beams was 400 ms for this work.

Figure 1 illustrates the characteristic features that are associated with the *H* mode in DIII-D. When the neutral beams start, the H_α/D_α radiation from the plasma edge and the divertor region first increases and then decreases abruptly. At this drop, the plasma line-averaged density begins to increase rapidly. The drop in H_α/D_α and the rise in density indicate a sudden, significant improvement in particle confinement time. In addition, the energy confinement time drops when the neutral beams first come on (*L* mode), then rises again after the H_α/D_α drop to a level near the Ohmic value (*H* mode). The duration of the *L* phase decreases with increasing neutral-beam power. Later in the *H*-mode phase of the discharge, short, sharp bursts of H_α/D_α radiation occur. These transient events have been named the edge-

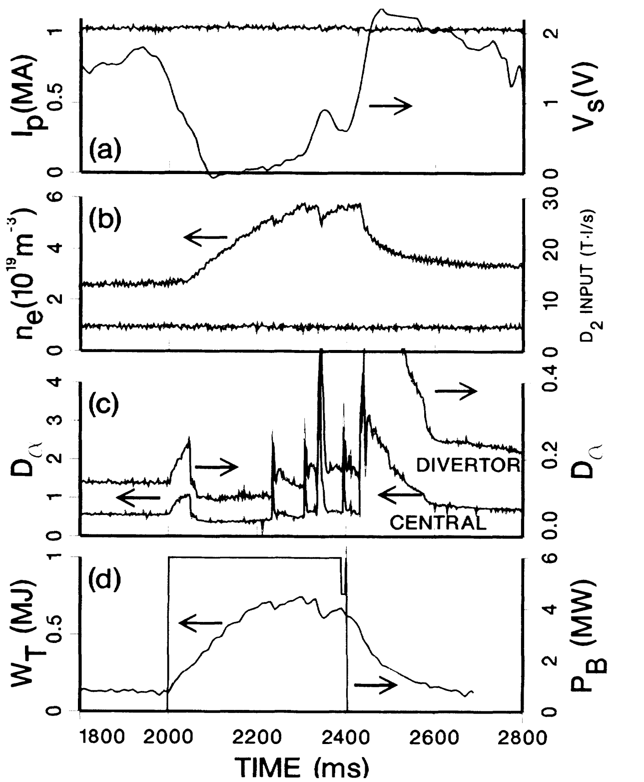

FIG. 1. Various diagnostic signals as functions of time in the discharge: (a) Plasma current and voltage at the plasma surface, (b) line-averaged density and deuterium gas injection rate, (c) H_α/D_α radiation along a horizontal chord through the plasma midplane and along a chord viewing the divertor region, and (d) neutron-beam power and plasma stored energy.

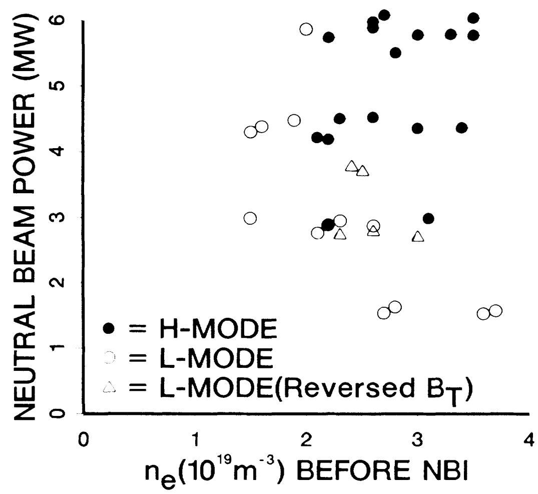

FIG. 2. Region of parameter space where the H mode exists. The density plotted is the plasma line-averaged density just before the neutral beams are turned on. Most of the cases are for the ion ∇B drift towards the divertor X point. Those marked reversed B_T have the ion ∇B drift away from X point. Plasma current is 1 MA; the magnitude of the toroidal field is 2.1 T.

localized mode (ELM) by the ASDEX group.[15,16] This whole group of features is characteristic of the H mode in DIII-D and appears to be very similar to the H-mode behavior in ASDEX.[13,15] In spite of the difference in divertor geometry and machine size, the H mode in DIII-D has all the qualitative features of the ASDEX H mode.

In order to obtain the H mode in the expanded boundary, the neutral-beam–heating power and the line-averaged density must both exceed certain thresholds. The threshold levels are illustrated in Fig. 2. At plasma current $I_P=1$ MA and toroidal field $B_T=2.1$ T, the density threshold is about 2×10^{19} m^{-3} and the power threshold lies near 2.8 MW. Exactly how these thresholds depend on plasma current and toroidal field is not yet known; however, with use of beam power $P_B=2.8$ MW, the H mode has been achieved over the range $0.7\ \mathrm{MA}\le I_P\le 1.6$ MA at $B_T=2.1$ T and over the range $0.9\ \mathrm{T}\le B_T\le 2.1$ T at $I_P=1$ MA. Accordingly, no H-mode threshold in current or toroidal field has been seen to date. In addition to the power and density threshold, the presence of sawtooth oscillations in the soft x-ray flux during the Ohmic phase correlates well with the appearance of the H mode. To date, no H-mode behavior has been seen in discharges that had no sawteeth before the neutral beams started even though the density and power thresholds were exceeded.

The duration of the H mode has so far been limited by the neutral-beam pulse length. By turning on pairs of sources sequentially, we have been able to achieve H-mode durations of 740 ms at modest (3 MW) power levels.

Theory predicts[31] and experiment confirms[32] that the power threshold of the H mode depends on the direction of the ion ∇B drift. This threshold is higher if the drift is away from the divertor X point. As is shown in Fig. 2, in DIII-D the H-mode threshold is near 2.8 MW with the ∇B drift toward the X point. In spite of the presence of sawteeth, no H mode has been found up to the maximum 3.8 MW available during the experiments with the ∇B drift away from the X point.

The character of the ELM's depends on the neutral-beam power level. At powers just above the H-mode threshold, the ELM's exhibit small H_α/D_α bursts which repeat every 10 to 30 ms. At powers above 4.0 MW, the ELM's are infrequent and the associated H_α/D_α bursts are much larger. Usually, one to three ELM's occur during a neutral-beam pulse; the inter-ELM periods are typically 100 ms. ELM-free periods of 300 ms have been seen on some shots. As shown in Fig. 1, the total plasma energy content W_T drops 10% to 20% during one of these large ELM's. Since the ELM's are so infrequent at high powers, we have the opportunity to study

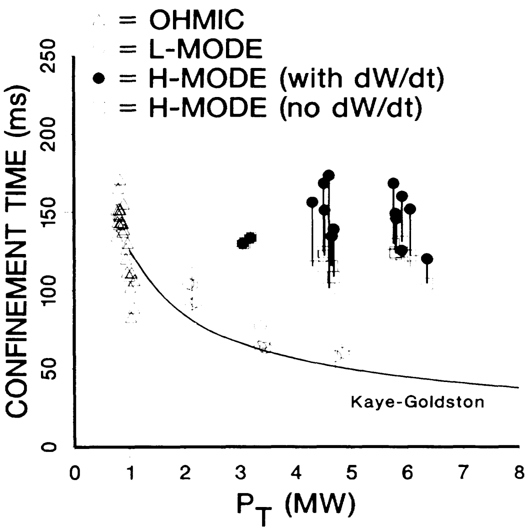

FIG. 3. Global energy confinement time vs the sum of the Ohmic and neutral-beam powers. Plasma current is 1 MA and toroidal field is 2.1 T. For the H-mode plasmas, the confinement time is given both with and without the correction for time-varying stored energy. For each H-mode shot, a heavy straight line connects the two points showing the confinement time determined by using each definition. Also shown for comparison with the L-mode data is the prediction of the Kaye-Goldston scaling law (Ref. 12).

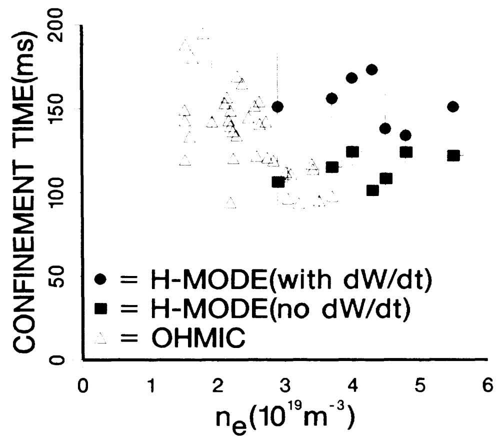

FIG. 4. H-mode and Ohmic energy confinement time vs plasma line-averaged density for a case with plasma current of 1 MA and a toroidal field of 2.1 T. For the H-mode cases, the total input power is in the range 4.5 to 5.0 MW. H-mode confinement times both with and without the correction for time-varying stored energy are given.

the intrinsic plasma confinement during the ELM-free periods. However, since these periods are usually comparable to the inferred τ_E, we must include the correction for time-varying W_T in the determination of τ_E. Accordingly, we use $\tau_E = W_T/(P_T - dW_T/dt)$ where P_T is the sum of the Ohmic and neutral-beam input powers. The dW_T/dt value is determined from magnetic data taken between ELM's. For some purposes (e.g., calculating achievable β_T), one simply needs the value of W_T obtained under given conditions. Accordingly, in our subsequent discussion, we have also quoted confinement times ignoring the dW_T/dt correction. The confinement scaling trends turn out to be the same using either measure of τ_E.

We have studied the dependence of energy confinement time τ_E on n_e, I_P, B_T, and P_T for H-mode discharges in DIII-D. For the data presented here, W_T is determined from a full MHD equilibrium analysis.[33] Comparison with W_T determined from diamagnetic loop and a full kinetic analysis shows agreement within the experimental error.[23] The fast-beam ion contribution to W_T is less than 10%.

The variation in τ_E with total heating power P_T is shown in Fig. 3 for $I_P = 1$ MA. As can be seen here, the H-mode τ_E is independent of heating power up to the maximum power studied. In addition, τ_E in the H mode is equal to or slightly above the Ohmic values given in the figure. Also shown in Fig. 3 are confinement points for L-mode discharges. Our L-mode results are slightly above the values calculated from the Kaye-Goldston[12] scaling law. DIII-D is the only device larger than ASDEX which has obtained τ_E independent of P_T.

The density dependence of τ_E differs in the Ohmic and H-mode plasmas. Because of the equilibrium analysis used to determine τ_E, the statistical error in τ_E is about 30% for Ohmic cases, which explains most of the scatter seen in the Ohmic points in Fig. 3. The rest of the variation is due to the dependence of Ohmic τ_E on density. As is shown in Fig. 4, with $I_P = 1$ MA, Ohmic τ_E falls slowly with increasing density over the range studied. Within the experimental error, there is no dependence of τ_E in the H mode on density. This insensitivity of τ_E to density is quite similar to our previous divertor results from Doublet III.[34]

Our toroidal-field scaling results are also similar to our Doublet III results[34]: As is shown in Fig. 5, at a plasma current of 1 MA, H-mode τ_E values are independent of toroidal field over the range 1.3 to 2.1 T. At B_T values near 1.4 T we again have a case where the H-mode τ_E values including the dW_T/dt correction are systematically above the Ohmic values, although the error bars again overlap.

In previous divertor experiments,[15,16,34-36] the key parameter controlling τ_E was plasma current: Confinement increased approximately linearly with current. In H-mode discharges in DIII-D, we have seen clear evidence of an increase in τ_E with current. We have not yet established the current scaling because the data set is small and because, in the few shots we have at currents away from 1.0 MA, the ELM amplitude and frequency appear to be correlated with the current. Further work, with careful attention to ELM characteristics, will be re-

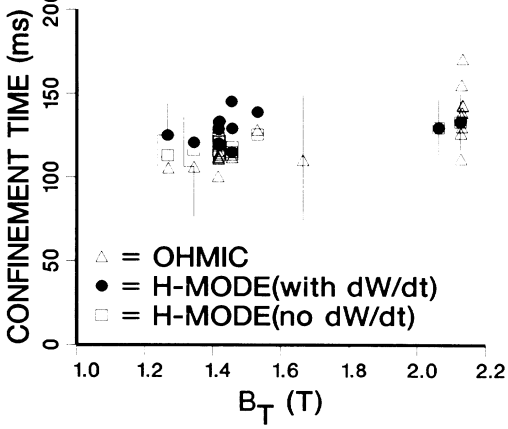

FIG. 5. H-mode and Ohmic energy confinement time vs toroidal field for a case with plasma current of 1 MA. For the H-mode data, the total input power is in the range 3.0 to 3.3 MW. The Ohmic data have line-averaged density in the range $(2.6-3.0)\times10^{19}$ m^{-3}. At this power level, the H-mode plasmas exhibit frequent, small-amplitude ELM's. Accordingly, the H-mode confinement times including the dW_T/dt correction do not show the ELM-free confinement time. The slight difference between the two H-mode confinement times at lower toroidal fields is due to a neutral-beam pulse length that was slightly too short for the plasmas to reach steady state.

quired to elucidate the current scaling.

In summary, we have produced H-mode confinement in the reactor-compatible expanded boundary divertor in DIII-D. This divertor exhibits H-mode confinement characteristics very similar to those in the ASDEX device,[13-19] including little if any degradation of confinement with auxiliary heating power. This result indicates that it should be possible to utilize the good, H-mode confinement in open divertor configurations in ignition devices and fusion reactors.

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[(a)]Permanent address: Hitachi, Ltd., Ibaraki, Japan.

[(b)]Permanent address: Lawrence Livermore National Laboratory, Livermore, CA 94550.

[(c)]Permanent address: Japan Atomic Energy Research Institute, Ibaraki, Japan.

[(d)]Permanent address: Max-Planck-Institüt für Plasmaphysik–EURATOM Association, D-8046 Garching bei München, Germany.

[(e)]Permanent address: Mitsubishi Electric Corp., Tokyo, Japan.

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