Enhanced Energy Confinement and Performance in a Low-Recycling Tokamak

R. Majeski,[1] R. Doerner,[2] T. Gray,[1] R. Kaita,[1] R. Maingi,[3] D. Mansfield,[1] J. Spaleta,[1] V. Soukhanovskii,[4] J. Timberlake,[1] and L. Zakharov[1]

[1]*Princeton Plasma Physics Laboratory, Princeton, New Jersey 08543, USA*
[2]*University of California at San Diego, La Jolla, California 92093, USA*
[3]*Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831, USA*
[4]*Lawrence Livermore National Laboratory, Livermore, California 94550, USA*

(Received 5 May 2006; published 17 August 2006)

Extensive lithium wall coatings and liquid lithium plasma-limiting surfaces reduce recycling, with dramatic improvements in Ohmic plasma discharges in the Current Drive Experiment-Upgrade. Global energy confinement times increase by up to 6 times. These results exceed confinement scalings such as ITER98P(y, 1) by 2–3 times, and represent the largest increase in energy confinement ever observed for an Ohmic tokamak plasma. Measurements of D_α emission indicate that global recycling coefficients decrease to approximately 0.3, the lowest documented for a magnetically confined hydrogen plasma.

DOI: 10.1103/PhysRevLett.97.075002 PACS numbers: 52.55.Fa, 52.25.Fi, 52.40.Hf

Many experiments have indicated that plasma performance is enhanced as the global wall recycling coefficient R is reduced [1–3]. Techniques such as divertor pumping and lithium wall coatings have reduced R by 5%–15%. The resultant changes in plasma performance were dramatic. In the Tokamak Fusion Test Reactor [4], a factor of 2 increase in the confinement time was produced with extensive lithium wall coatings of the carbon wall and a 15% reduction in recycling [5]. In the DIII-D device, divertor cryopumping resulted in a strong increase in the edge pedestal temperature with only a modest decrease in global recycling [6]. However, the very low global recycling regime ($\sim$50% or less) has not been explored. Significant changes in tokamak discharge characteristics are predicted in this regime [7,8]. A candidate wall material for the minimization of recycling is metallic lithium, especially liquid lithium.

Experiments on the T11-M device in Russia have previously utilized a compact liquid lithium rail limiter, in combination with partial solid lithium coatings on the stainless steel vessel wall [9]. Initial experiments in the Current Drive Experiment-Upgrade (CDX-U) also utilized a liquid lithium rail limiter similar to the T11-M system [10]. Here we report on the results of experiments in the CDX-U with large-area (600–2000 cm^2) liquid lithium limiters, in combination with full-wall lithium coatings up to 1000 Å thick, which are applied between discharges. The use of liquid lithium as a plasma-limiting surface ensures that deuterium does not ''load'' the surface of the lithium during initial stages of plasma operation, since the high diffusivity of deuterium in liquid lithium ($>10^4$ cm^2/s at the 300–400 °C operating temperature) ensures that the surface region of the lithium is not saturated. Similarly, background gases present in the vacuum vessel (e.g., water) are dissolved into the liquid lithium rather than forming a surface layer. Loading of the solid lithium coatings cannot be avoided, but contamination of the surface is reduced by depositing the lithium coating rapidly ($<$3 min) and by minimizing the time between deposition and the plasma discharge ($<$1 min).

Tokamak discharges in CDX-U with lithium wall and limiter coatings, and the liquid lithium limiter, exhibit significantly enhanced performance compared to discharges with uncoated plasma facing surfaces. The plasma facing surfaces in CDX-U are titanium carbide, boron carbide, and stainless steel. Transport in CDX-U prior to lithium operations has been previously characterized [11] with global confinement times of approximately 1 ms, from kinetic measurements. In contrast, lithium operation has produced global confinement times of 6 ms. Confinement is strongly correlated with the level of recycling in these discharges. For the confinement data presented here, R was in the 50%–60% range. However, global recycling was reduced to approximately 30% for discharges that employed the largest available area of liquid lithium (2000 cm^2).

The CDX-U is a small low aspect ratio tokamak, with major radius $R_0 = 34$ cm, minor radius $a = 22$ cm, plasma elongation $\kappa = 1.6$, and a toroidal magnetic field of 2 kG. The device typically operates in deuterium, with plasma current $<$80 kA and pulse length $<$25 ms. For the experiments described here, the line-averaged density measured by a 2 mm microwave interferometer was $0.5–1 \times 10^{13}$ cm^{-3}. The lower plasma limiter in CDX-U is a shallow, electrically heated, circular stainless steel tray with a radius of 34 cm, a width of 10 cm, and a depth of 0.6 cm. The lithium tray limiter [12] and the system used for filling the tray with a total of 300 g (0.6 l) of lithium [13] have been extensively described elsewhere. The lithium in the tray was either heated indirectly from below by means of resistive disk heaters attached to the bottom of the tray, or directly heated by a 1.6 kW magnetically guided electron beam incident on the surface of the lithium. Lithium coatings were also produced by a resistively heated oven evaporator, which was mounted on the vac-

uum vessel wall at the outer midplane. The lithium systems installed on CDX-U are indicated in Fig. 1.

The lowest global recycling rates were inferred with the tray limiter completely filled with liquid lithium (approximately 0.6 l, or 0.3 cm deep, with a surface area of 2000 cm^2, or ~50% of the plasma-contacting area), and heated to 350 °C. The melting point of lithium is 180.5 °C, and at 350 °C the evaporation rate is sufficient to coat other plasma-limiting surfaces and the vacuum vessel inner wall with lithium at a rate of approximately 3–10 Å/ min, with higher deposition rates occurring on the center stack (inboard) limiting surfaces. Deposition rates were monitored with a commercial quartz crystal microbalance (Inficon Model XTM/2) at a distance of 1 m from the tray surface. CDX-U discharges are primarily limited on the center stack, and secondarily on the lithium tray limiter. Recycling was measured by monitoring the relative level of D_α radiation emitted at the center stack midplane, which is the primary plasma contact point. A comparison of D_α emission levels for plasma discharges with no lithium in the tray and for discharges with the tray filled with liquid lithium at 350 °C is shown in Fig. 2. The edge plasma density and electron temperature were measured with a triple Langmuir probe located at the last closed flux surface (i.e., at the outer limiter radius, located at $a = 22$ cm or $R = 56$ cm), which maps to the inner contact point at the center stack. The edge plasma density for both sets of discharges was approximately the same, 1×10^{12} cm^{-3}, although the edge electron temperature was 30 eV in the case with lithium, compared to 20 eV without lithium. In the case of a bare tray, the discharge was initiated with a vessel prefill, and no other gas fueling was required during the discharge. The bare tray discharges are therefore assumed to have a recycling coefficient R of approximately 1. For discharges using the liquid lithium-filled tray, strong gas fueling was necessary during the discharge in order to maintain the plasma density between 0.5 and 1×10^{13} cm^{-3}. The recycling coefficient for plasmas limited by liquid lithium is estimated by comparing the D_α radiation emitted in that case with the D_α radiation emitted during high recycling operation, and correcting for the difference in electron temperature [14–16]. This yields a recycling coefficient of 0.3 for full-tray liquid lithium operation (2000 cm^2), at the center stack. A comparison of the temporal evolution of the plasma current, loop voltage, density, and fueling for discharges with a bare tray and with a liquid lithium-filled tray has been previously made [17]. A diagnostic of Z_{eff} is not available on CDX-U, but modeling of the discharges indicates that $Z_{\text{eff}} \sim 2.4$ for the bare tray discharges, and <1.2 for the liquid lithium-filled tray [17].

During operation of the electron beam heating system in combination with the resistive oven evaporator, the net recycling coefficient on the center stack plasma-limiting surfaces was generally higher. In this case only half the lithium-filled tray was liquefied and heated to 400–450 °C by the electron beam, which provided thick (up to 1000 Å) lithium coatings, covering 180° of the center stack plasma-contacting surfaces, in the 5–7 min interval between plasma discharges. Lithium coatings were produced on the remaining 180° of the center stack surface by the resistive oven evaporator, but the small (<10 g) lithium inventory in the oven limited the deposition rate and duration. D_α emission levels from center stack limiting surfaces coated by the resistive oven evaporator are also indicated in Fig. 2. The average D_α emission level indicates an approximate 20% reduction in recycling when corrected for electron temperature differences for the half of the center stack coated by the resistive oven evaporator. Thus global recycling under these conditions is estimated at 0.5–0.6. Data on energy confinement were only available under these conditions.

The effective particle confinement time $\tau_p^* \equiv \tau_p/(1 - R)$, where τ_p is the particle confinement time, is estimated by measuring the density decay rate when gas fueling is

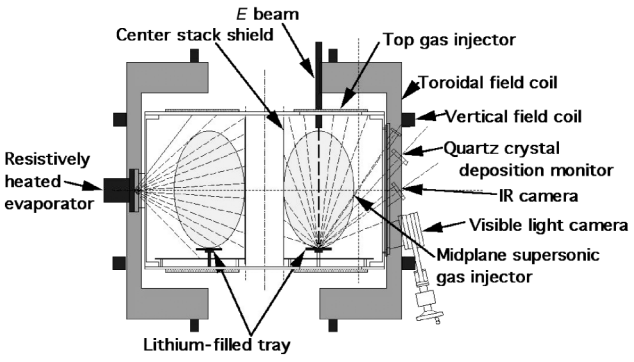

FIG. 1. Schematic of CDX-U, with the lithium and fueling systems, and the electron beam injector.

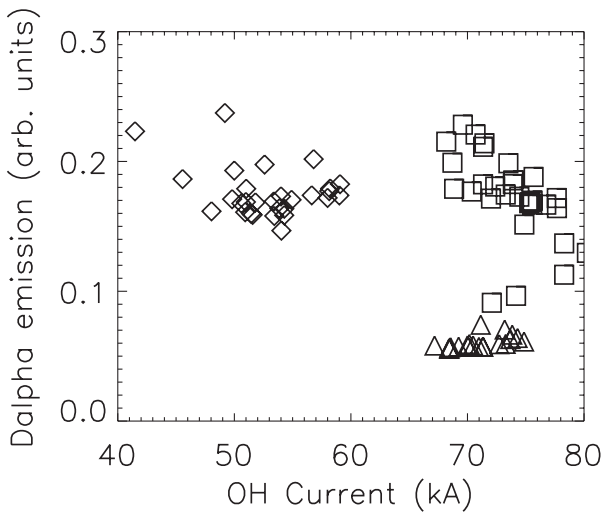

FIG. 2. D_α emission at peak plasma current for discharges without lithium wall coatings (diamonds), with modest levels of lithium wall deposition (6–10 Å/ min, squares) and with extensive lithium wall coatings produced by evaporation from the lithium tray limiter (triangles).

terminated. In CDX-U, when lithium limiters and wall coatings are not employed, the density decay rate is too long to allow an estimate of $\tau_p{}^*$. Discharges which were initiated within a few minutes of a lithium coating cycle exhibited very rapid density pump-out after cessation of gas fueling, with $\tau_p{}^*$ as low as 2–3 ms. Operation with a full lithium tray limiter at a temperature above 300 °C produced similar values of $\tau_p{}^*$. Discharges operated with older, chemically inactive lithium wall coatings exhibited intermediate values of $\tau_p{}^*$. Values of $\tau_p{}^*$ obtained under various operating conditions are indicated in Fig. 3. The deuterium pumping rate represented by the lowest values of $\tau_p{}^*$ shown in Fig. 3 is in the range of $2–3 \times 10^{21}$ particles/s.

Plasma equilibrium reconstructions were performed with the Equilibrium and Stability Code (ESC) [18], which has been modified to include the effects of vessel eddy currents on the magnetic signals. The energy confinement time τ_E is given by [19]

$$\tau_E = \frac{W_{\text{kinetic}}}{(-I_p \frac{d\Psi_{\text{edge}}}{dt} - \frac{dW_{\text{mag}}}{dt} - \frac{dW_{\text{kinetic}}}{dt})}, \quad (1)$$

where W_{kinetic} is the stored plasma kinetic energy, and ψ_{edge} is the edge poloidal flux, which yields the surface voltage $V_{\text{edge}} = (d\psi_{\text{edge}}/dt)$. Since the CDX-U Ohmic transformer is driven by capacitor banks, the discharge is not stationary, and so the time derivative of the stored magnetic energy W_{mag}, and the time derivative of the stored kinetic energy W_{kinetic} must be included in Eq. (1). We evaluate τ_E when $(dW_{\text{mag}}/dt) = 0$ from the ESC reconstructions, which corresponds closely to the peak in the plasma current. A compensated [20] diamagnetic loop is used in combination with magnetic reconstruction of the plasma boundary to measure the stored plasma kinetic energy. ESC also is used to determine the poloidal flux and hence the surface voltage near the time of peak plasma current. The plasma current is measured with a Rogowski coil internal to the vacuum vessel. A plot of the measured values of confinement time versus ITER98P(y, 1) [21] is shown in Fig. 4. This scaling was the first to incorporate data from the START low aspect ratio tokamak [22], which was similar in size to CDX-U. Prior to the introduction of lithium plasma-facing components to CDX-U, the measured confinement time fell in the range of 0.7–1.1 ms [10]. Although this estimate was derived from measurements of the electron temperature and density rather than from magnetic reconstructions, it is in agreement with the passivated lithium results shown in Fig. 4. However, the confinement time during active lithium operation exceeds previous results by up to a factor of 6 or more, and ITER98P(y, 1) ELMy H-mode scaling by a factor of 2–3. Note that other than the incorporation of the lithium tray limiter and lithium wall coatings, and additional fueling capability (a second gas puffing system), no other changes in the CDX-U configuration were made in order to obtain this improvement in confinement. The plasma current and toroidal magnetic field, as well as the size of the plasma (determined by the limiter positions) were identical for the pre- and postlithium discharges. The discharge electron density was similar, although in many cases the lithium discharges ran at somewhat lower density. The operating gas (deuterium) was the same in both cases. A transition back to lowered confinement could be reliably produced by allowing the lithium surfaces to passivate (collect background gases and acquire a high recycling coating) over days or weeks.

The error estimate is provided by a second calculation of τ_E when $(dW_{\text{kinetic}}/dt) = 0$, which occurs earlier in the discharge than the peak in magnetic stored energy. The time interval between the peak in kinetic stored energy and the peak in magnetic stored energy generally increases as the confinement time increases, which may contribute to the increase in the error estimate for discharges with active lithium evaporation and long confinement times. The larg-

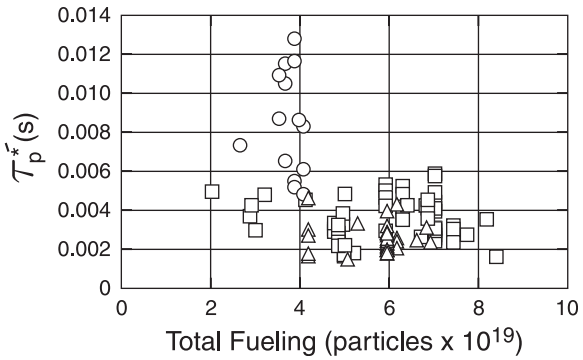

FIG. 3. Effective particle confinement time $\tau_p{}^*$ versus total number of deuterons injected for discharges with varying levels of lithium wall conditioning. The triangles denote results with the full lithium tray liquefied (2000 cm^2), squares denote electron beam evaporation and a 600 cm^2 liquid lithium area, and circles denote operation with passivated lithium (no active evaporation or heating).

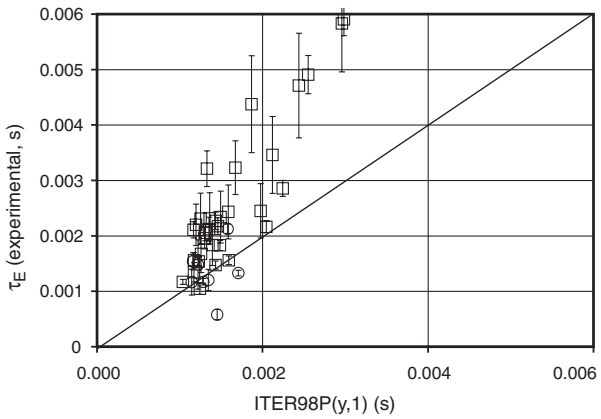

FIG. 4. Measured energy confinement time versus ITER98P(y, 1) confinement scaling. Discharges with passivated lithium walls are denoted by circles. Discharges with active lithium evaporation are denoted by squares.

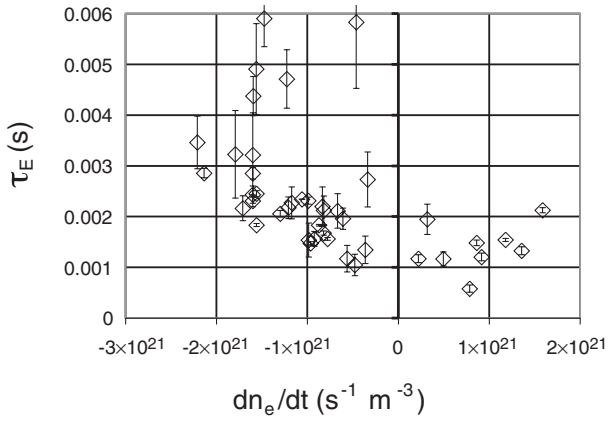

FIG. 5. Measured energy confinement time versus the density pump-out rate (dn_e/dt). The density pump-out rate is a good measure of the degree to which recycling is suppressed.

est contributor to the confinement time increase for discharges with active lithium evaporation is the reduction in loop voltage, by up to a factor of 4. Stored energy also increases in discharges with active lithium evaporation, and the stored energy peaks later in the discharge than is the case without evaporation.

We emphasize that the plasma-limiting surfaces for all discharges for which confinement time data are available were modified by lithium deposition, since the full set of magnetic diagnostics were only available after electron beam evaporation of the lithium in the limiter tray had been underway for several months. The high recycling regime could only be revisited by allowing the lithium coatings to passivate. However, even in the high recycling regime, oxygen and other impurities were greatly reduced compared to prelithium operation.

We find that confinement time is most strongly correlated with the density pump-out rate. We characterize the density pump-out rate by the time rate of change of the density dn_e/dt after fueling ceases, but before the plasma current begins to decrease and the equilibrium is modified. This characterization allows the inclusion of high recycling discharges, for which dn_e/dt is positive even after fueling is terminated, whereas $\tau_p{}^*$ is not meaningful under these circumstances. Figure 5 shows this correlation. High recycling discharges with constant or increasing density after cessation of fueling evidence modest confinement times. The production of discharges with enhanced confinement time is correlated with lowered recycling, as measured by the density pump-out rate. Note that, for all discharges plotted in Figs. 4 and 5, gas fueling was terminated 1–2 ms before the peak in plasma current, when the measurement of confinement time occurred.

A factor of 3 increase in carbon IV impurity ion temperature, from 24 to 71 eV, was also found for lithium limited discharges, as determined by Doppler broadening of the 466 nm C IV emission line. Carbon and oxygen emission were reduced by a factor of 10 in the lithium discharges. The loop voltage required to maintain a 70 kA plasma current was reduced from 2–3 V in prelithium discharges to 0.5 V or less in lithium discharges.

In conclusion, the use of successively more aggressive levels of lithium wall coatings coupled with large-area liquid lithium limiting surfaces has a profound effect on energy confinement in a tokamak. Extensive coatings applied with a minimal time delay before discharge initiation produced a factor of 6 improvement in energy confinement time, the largest increase ever obtained in an Ohmic tokamak. The observed increases in impurity ion and edge electron temperature suggest that low-recycling lithium operation modifies the temperature profile. However, core electron temperature measurements were not available in CDX-U. Core electron measurements via Thomson scattering measurements are planned for the follow-on to CDX-U, the Lithium Tokamak eXperiment, which will begin operation in early 2007.

We thank Daren Stotler for assistance in the recycling estimates. Jonathan Menard, Michael Bell, Henry Kugel, and Michael Ulrickson also contributed valuable advice. This work supported by U.S. DOE Contract No. DE-AC02-76-CH0-3073.

[1] J. D. Strachan, Nucl. Fusion **34**, 1017 (1994).
[2] J. Ongena *et al.*, Nucl. Fusion **33**, 283 (1993).
[3] S. V. Mirnov *et al.*, Fusion Eng. Des. **65**, 455 (2003).
[4] D. K. Mansfield *et al.*, Nucl. Fusion **41**, 1823 (2001).
[5] R. V. Budny *et al.*, J. Nucl. Mater. **196–198**, 462 (1992).
[6] W. P. West *et al.*, Phys. Plasmas **9**, 1970 (2002).
[7] L. E. Zakharov *et al.*, Fusion Eng. Des. **72**, 149 (2004).
[8] S. I. Krasheninnikov *et al.*, Phys. Plasmas **10**, 1670 (2003).
[9] V. Evitkin *et al.*, Fusion Eng. Des. **56–57**, 363 (2001).
[10] G. Antar *et al.*, Fusion Eng. Des. **60**, 157 (2002).
[11] T. Munsat *et al.*, Phys. Plasmas **9**, 480 (2002).
[12] R. Majeski *et al.*, J. Nucl. Mater. **313–316**, 625 (2003).
[13] R. Majeski *et al.*, Fusion Eng. Des. **72**, 121 (2004).
[14] L. C. Johnson and E. Hinnov, J. Quant. Spectrosc. Radiat. Transfer **13**, 333 (1973).
[15] R. K. Janev *et al.*, J. Nucl. Mater. **121**, 10 (1984).
[16] DEGAS2 User's Manual, http://w3.pppl.gov/degas2.
[17] R. Majeski *et al.*, Nucl. Fusion **45**, 519 (2005).
[18] L. E. Zakharov and A. Pletzer, Phys. Plasmas **6**, 4693 (1999).
[19] I. H. Hutchinson, *Principles of Plasma Diagnostics* (Cambridge University Press, Cambridge, 2002), p. 22.
[20] L. E. Zakharov and V. D. Shafranov, in *Reviews of Plasma Physics*, edited by M. A. Leontovich (Consultant Bureau, New York, 1986), Vol. 11, p. 153.
[21] M. Wakatani *et al.* (ITER Physics Expert Groups on Confinement and Transport and Confinement Modelling and Database Collaboration), Nucl. Fusion **39**, 2137 (1999).
[22] O. J. W. F. Kardaun, in *Proceedings of the Eighteenth IAEA Fusion Energy Conference, Sorrento, 2000* (IAEA, Vienna, 2001), CD-ROM file ITERP/04 (unpublished).