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Abstract
Use of a large-area liquid lithium limiter in the CDX-U tokamak produced the largest relative increase (an enhancement factor of 5–10) in Ohmic tokamak confinement ever observed. The confinement results from CDX-U do not agree with existing scaling laws, and cannot easily be projected to the new lithium tokamak experiment (LTX). Numerical simulations of CDX-U low recycling discharges have now been performed with the ASTRA-ESC code with a special reference transport model suitable for a diffusion-based confinement regime, incorporating boundary conditions for nonrecycling walls, with fuelling via edge gas puffing. This model has been successful at reproducing the experimental values of the energy confinement (4–6 ms), loop voltage (<0.5 V), and density for a typical CDX-U lithium discharge. The same transport model has also been used to project the performance of the LTX, in Ohmic operation, or with modest neutral beam injection (NBI). NBI in LTX, with a low recycling wall of liquid lithium, is predicted to result in core electron and ion temperatures of 1–2 keV, and energy confinement times in excess of 50 ms. Finally, the unique design features of LTX are summarized.

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1. Introduction

All the major tokamaks, whether limited or diverted machines, have achieved their highest performance in low recycling regimes. The aim of the lithium tokamak experiment—LTX—is to produce tokamak discharges with near-zero recycling, and determine the consequences for transport and stability of operating in this extreme limit. A fully nonrecycling first wall has been theoretically predicted to fundamentally alter the nature of plasmas in tokamaks, including ITER [1]. Similar profound changes may be expected for any magnetically confined plasma configuration [2]. The CDX-U experiments [3], which employed a liquid lithium belt or tray limiter, represented an intermediate step towards a full low recycling wall in LTX. A transport model (designated the reference transport model, or RTM) has been developed for pumping boundary conditions and employed in the ASTRA-ESC [4] code system which models all energy losses as carried by particles, at the ion neoclassical rate. This model has been successful at reproducing key features of the CDX-U discharges, including the energy confinement time increase and the observed strong decrease in loop voltage. Because of this success in modelling CDX-U low recycling discharges, the model has been used to project the performance for LTX.

LTX is the first tokamak completely designed around the use of liquid lithium as a PFC. The first tokamak discharge has now been produced in LTX.

2. Experimental results from CDX-U lithium operation

Liquid lithium limiter experiments in CDX-U demonstrated a significant, more than five-fold enhancement of confinement, when the lithium surface was in contact with the plasma edge [1, 5, 6]. The evolution of a typical low recycling discharge has previously been described [3, 7]. An equilibrium reconstruction of a typical low recycling discharge in CDX-U is shown in figure 1. The plasma stored energy and

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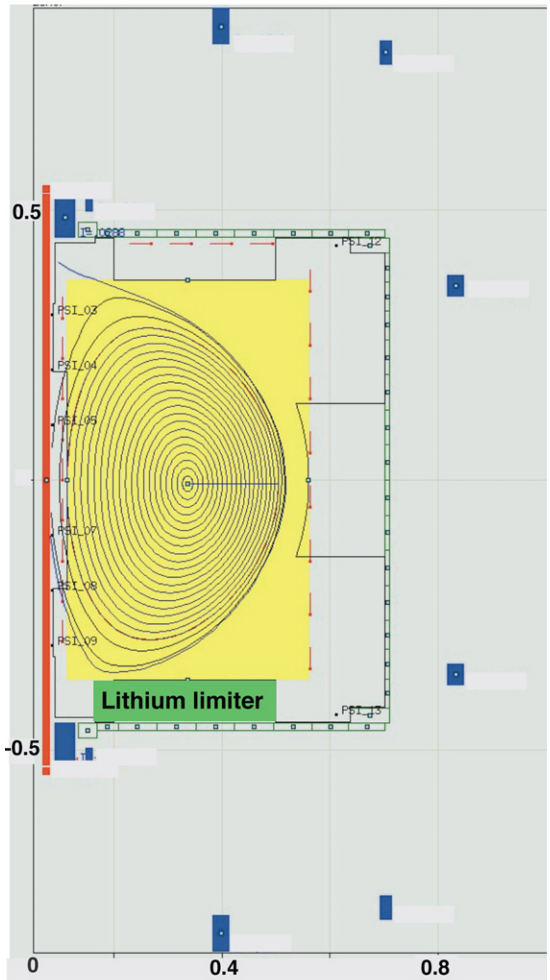

Figure 1. Equilibrium reconstruction of a low recycling discharge in CDX-U. The positions of the poloidal field coils are indicated by blue rectangles. The lithium tray position is indicated in green. The outboard and upper limiter positions are also indicated; both were coated with lithium. Radial and vertical position is indicated in metres. For this discharge, a total coating of 13 000 Å (4 g total) of lithium had been applied before the discharge. τ_E for this discharge was 6 ms. The measured surface voltage was less than 0.5 V.

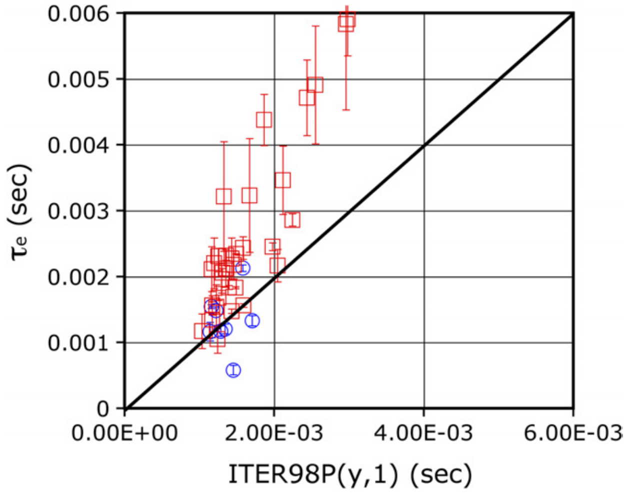

Figure 2. Measured energy confinement time versus ITER98P(y, 1) confinement scaling. Discharges with passivated lithium walls are denoted by blue circles. Discharges with active lithium evaporation are denoted by red squares.

confinement assessment were obtained from diamagnetic measurements and equilibrium reconstructions, based on magnetic measurements, and a comparison with ITER98P(y,1) has been published [3]. These results are also shown here in figure 2. The data include discharges (blue circles) which benefit from lithium gettering, but are operated against a solid, passivated lithium surface which has been exposed to base vacuum pressure for several weeks without heating or fresh evaporated coatings. These discharges are characterized by moderately high recycling, but are low in impurity content (especially oxygen), and exhibit an average energy confinement time of 1.3 ms. Discharge behaviour under these conditions was similar to earlier operation of CDX-U with extensive titanium gettering. Kinetic measurements of the energy confinement time, during operation with titanium gettering, under similar conditions of toroidal field, plasma current, shape and plasma density yielded results in the 0.7–1.1 ms range [8]. This range of confinement time is in reasonably good agreement with the more recent equilibrium reconstruction results obtained during operation with solid lithium limiting surfaces (blue circles in figure 2). In contrast, operation with a liquefied lithium limiting surface 600 cm^2 in area, in combination with continuous evaporative coating of ∼80% of the limiting surfaces and interior of the vacuum vessel, produced discharges with confinement times of up to 6 ms, from magnetic analysis (Thomson scattering was inoperable during this phase of CDX-U operations). These data are plotted in figure 2 as red squares, and represent the largest relative increase in confinement time seen in an Ohmic tokamak to date.

However, comparisons of the CDX-U confinement results with scaling laws show little promise for predicting confinement in LTX. As can be seen in figure 2, the energy confinement during CDX-U lithium operation exceeds ITER98P(y,1) by 2–3×. The energy confinement times observed in low recycling discharges are up to 25× the values derived from neo-Alcator scaling. We note that results from a similar sized diverted spherical tokamak which employed high recycling carbon divertor targets, START, also typically exceeded neo-Alcator scaling, but by a factor not exceeding 2–4 [9]. Comparisons of CDX-U with simulation codes such as TSC [10], which was extensively used to model CDX-U as part of a benchmarking effort [11] for M3D [12] and NIMROD [13], have been successful at reproducing the sub-millisecond energy confinement times typically seen in high recycling discharges [14], but underpredict low recycling confinement by an order of magnitude or more. The failure of existing scaling laws and numerical codes to predict the confinement time in very low recycling discharges implies that existing approaches are inadequate to predict the performance for LTX.

In CDX-U, the primary variable determining confinement time was recycling, which is estimated to be of order 50–60% for CDX-U lithium discharges [3]. Recycling is not a parameter entering into any of the existing confinement scaling laws. Indeed, for the discharges analysed for figure 2, the toroidal field was fixed at 2.1 kG, the plasma current varied only from 61 to 78 kA, the line-averaged density varied only from 0.4 to 0.7 × 10^{19} m^{-3}, and the plasma minor radius and shape varied by only ~10–20%. Most of the parameters which are important inputs to confinement scaling laws were therefore approximately constant across all the discharges.

The primary factor determining the variation in ITER98P(y,1) energy scaling evident in figure 2 is the power input, which in an Ohmic tokamak such as CDX-U, under conditions of roughly constant plasma current, is determined by the surface loop voltage. The surface loop voltage at the current flattop in CDX-U was reduced from 2 to 3 V, for high recycling discharges, to less than 0.5 V for low recycling discharges. The resulting factor of 6 variation in Ohmic power P input enters ITER98P(y, 1) as $P^{-0.69}$. If the loop voltage had not varied for the data shown in figure 2, but remained fixed at the value for high recycling discharges (2–3 V), the ITER98p(y, 1) scaling would yield ~1 ms confinement time for all the discharges shown. The confinement scaling shown in figure 2 therefore has no predictive capability, since for discharges at similar plasma current the surface loop voltage is not an externally adjustable factor, but determined by the plasma temperature and $Z_{\mathrm{effective}}$. Thus lowered loop voltage is a consequence of increased confinement, not an actuator.

Although Thomson scattering was not available for low recycling discharges on CDX-U, the carbon IV impurity ion temperature was measured to be 70–80 eV, from Doppler broadening of the 466 nm emission line (compared with 20–30 eV for high recycling discharges). The peak electron temperature can be estimated to be ~300 eV from the Spitzer resistivity, for moderately broadened electron temperature profiles, with a $Z_{\mathrm{effective}}$ of 1. This compares with a measured value of 100–150 eV, from Thomson scattering, on earlier high recycling discharges [14]. Gas puffing was always terminated several milliseconds before the energy confinement time was analysed, so that the edge neutral density is dominated by recycling (still estimated at 50% in these discharges). CDX-U operated in a low collisionality regime, with $\upsilon^*_{\mathrm{i,e}} < 0.1$. The ion–electron equilibration time was always much longer than the energy confinement time, with values ranging from 5 to 8 ms for high recycling discharges, up to 15–20 ms for low recycling discharges. Thus the electrons and ions were poorly coupled and charge exchange losses are not a dominant factor in the power balance for electrons. Radiated power was significantly less than the Ohmic input power, and we note that radiated power as a fraction of Ohmic input power was actually higher in the low recycling discharges, due to the greatly reduced loop voltage during lithium operation. Thus, for Ohmically heated discharges, energy confinement in CDX-U was determined by electron thermal transport and particle diffusion.

Theoretical analyses [1, 2] of low recycling tokamaks have indicated that as recycling drops below 50% the edge electron temperature will rise and the core electron temperature gradient, and consequently the impact of thermo-conduction, will be greatly reduced. In such a diffusion dominated confinement regime, energy transport will be due to particle losses and is not sensitive to thermo-conduction. We assume that electrons are poorly confined, and further assume that the coefficient of particle diffusion is equal to the ion thermo-conduction coefficient, which is close to the neoclassical value in spherical tokamak experiments. The same level of electron thermo-conductivity was also assumed. We refer to this model as the RTM. It provides a valuable point of comparison to the CDX-U results. The RTM yields values for the energy confinement time which are somewhat shorter than the experimentally determined values. The experimental values of the confinement time approach the duration of the current flattop in CDX-U. If electrons were also confined neoclassically, the energy losses in the regime would be reduced by approximately the square root of the ion to electron mass ratio, as indicated by an analysis of neoclassical transport with no significant temperature gradients [15]. The experiment was of course unable to address confinement times which significantly exceed the discharge flattop. Therefore, the RTM, with low recycling boundary conditions, and a Fokker–Planck model for the gas puff source of neutral atoms, has been implemented in the ASTRA-ESC code system for simulation of CDX-U and LTX discharges.

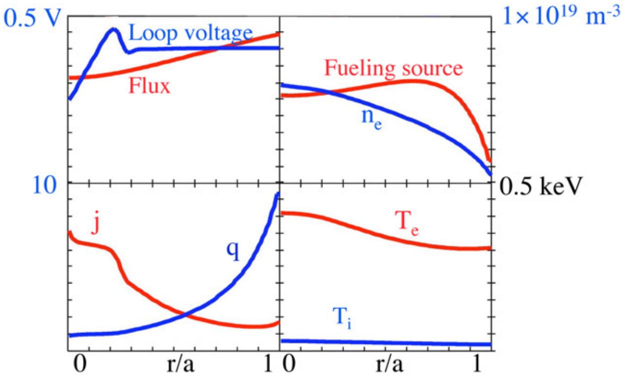

Figure 3. Stationary plasma profiles as functions of the normalized minor radius, modelled with the RTM in ASTRA-ESC for CDX-U lithium-walled discharges. Profiles are for the results summarized in table 1 under the 'RTM' entry.

3. Modelling of CDX-U with ASTRA-ESC and the RTM

Figure 3 shows an example of the modelled radial profiles for a CDX-U lithium discharge, assuming perfectly pumping boundary conditions, using the RTM. An Ohmic plasma discharge (CDX-U shot number 0818051533), near the flattop of the plasma current, has been simulated. The only fitting parameter in the RTM was the intensity of the gas puffing, which was adjusted to fit the value of plasma β_{j} (in Shafranov's definition) obtained from the CDX-U equilibrium reconstruction. The particle source due to gas puff was simulated using the Fokker–Planck equation. This is most appropriate for a small, low density device like CDX-U, since neutral shielding is poor and neutrals are not limited to the edge region. This model for the particle source was used in combination with the zero recycling boundary condition for the transport equations, when the energy through the last closed surface is transported with the particle flux. Transport studies with intermediate levels of recycling, appropriate to larger devices with good edge shielding of neutrals, have also been performed recently, and indicate that transport is not substantially affected by finite recycling, as long as the recycling coefficient remains below 50%.

A comparison of ASTRA-ESC simulations, using the RTM, with the available CDX-U parameters is shown in

Table 1. Comparison of CDX-U low recycling discharge with the ASTRA RTM, RTM with reduced neoclassical transport (scaled by 0.8 or 0.65) and RTM combined with the GLF23 transport model. The latter comparison indicates that anomalous transport is not a significant contributor.

Parameter	CDX-U	RTM	RTM-0.8	RTM-0.65	GLF23 + RTM	Comment
Fueling, 10^{21} (s)	1–2	0.98	0.5	0.3	3	Gas puffing rate adjusted
β_j	0.16	0.15	0.15	0.151	0.145	to match measured β_j
ℓ_i	0.66	0.77	0.702	0.671	0.877	Internal inductance
V (V)	0.45	0.77	0.53	0.4	0.85	Loop voltage
τ_E (ms)	3.2	2.7	3.8	5.3	2.3	Confinement time
$n_e(0)$ (10^{19} part m^{-3})	$\sim$1	0.9	0.7	0.596	0.9	Central density
$T_e(0)$ (keV)	0.3 (Spitzer, est.)	0.30	0.366	0.413	0.33	Central electron temperature
$T_i(0)$ (keV)	>0.06–0.07	0.03	0.029	0.030	0.028	Central ion temperature

table 1. The model reproduces well the measured or reconstructed values of the central density, the internal inductance, and especially the low loop voltage and enhanced energy confinement time observed in the CDX-U lithium experiments. Thus, with a small reduction in particle diffusivity (to $0.8\chi_i^{\text{neoclassical}}$) it can reproduce substantially all the CDX-U reference parameters listed in table 1. For a comparison, the GLF23 transport model has been included in the ASTRA-ESC modelling, as an additive transport term to the RTM. GLF23 was developed to model anomalous transport regimes in conventional tokamaks [16]. Since conduction losses modelled by GLF23 are virtually 'turned off' by the lack of a significant electron temperature gradient with nonrecycling walls, the addition of the GLF23 model predicts no significant change, as can be seen from a comparison of the confinement times for GLF23 + RTM, and the RTM alone.

One parameter which is not well reproduced by the model is the ion temperature. The measured impurity ion temperature (70–80 eV) is well above the model's prediction of 30 eV. Also, the impurity ion temperature derived from C IV line widths is unlikely to represent the peak deuterium ion temperature in the discharge, which is expected to be higher. Ionization states higher than C IV could not be analysed for Doppler line width with the available instrumentation. Thus the discrepancy between the modelled and the measured ion temperature is likely to be even larger than shown in table 1.

The time history of the CDX-U discharges was not reproduced in these simulations. There is no reliable equilibrium reconstruction data for the ramp-up phase of the discharge. We also note that transport simulations of a plasma discharge with a flattop duration comparable to the energy confinement time are very challenging even for well diagnosed plasmas. The time dependence of the gas fuelling, which in CDX-U was terminated 1–2 ms prior to the flattop in plasma current (when τ_E is measured), was not simulated. CDX-U discharges during the low recycling experiments were operated at modest density, hence the normalized collisionality was $\nu^*_{i,e} < 0.1$, for the measured impurity ion temperature of 70–80 eV, and assuming $T_e > T_i$. Although the plasma density was low, we stress that these were not slideaway discharges. The production of fast electron populations is clearly indicated in CDX-U by a marked increase in x-ray emissions, which was never present in the heavily fuelled lithium discharges. It is possible that the somewhat reduced confinement time given by ASTRA-ESC, relative to the experimental results, seen in modelling CDX-U with the RTM is due to either the reduction in edge neutral pressure, as a result of the cessation of gas puffing during the measurement of τ_E, or the higher ion temperature observed in the experiment. However, no MHD activity (which could heat the ions) in the form of either internal reconnection events or, interestingly, sawteeth, were observed in the CDX-U lithium discharges [5]. A possible explanation for the observation of higher ion temperature, which cannot be verified experimentally, would be provided by a small region of enhanced confinement in the plasma core. The global estimate of confinement time from magnetics is volume-weighted, and so a high confinement core region cannot be excluded. TSC modelling indicated that energy confinement times with a high recycling edge should range from 0.15 to 0.4 ms [11], which is at least an order of magnitude less than the experimentally measured confinement time during low recycling lithium operations. These studies further justify the adoption of the RTM to model lithium wall regimes and LTX. Note that simulations of the LTX plasma, which will have a controlled current flattop, will be more reliable. LTX will also have multipoint Thomson scattering available at an early stage.

4. The LTX

The LTX will be somewhat larger than CDX-U ($R_0 = 0.4$ m, $a = 0.26$ m, $\kappa = 1.6$), and will also operate with a limited, rather than a diverted, discharge. However, LTX is designed to employ a thin-film liquid lithium wall covering 90% of the plasma facing area (5 m^2). The maximum plasma current will be increased to 400 kA, with a 50 ms flattop. The toroidal field will also be increased to 3.4 kG. Initially LTX will be fuelled via edge gas puffing, using a combination of conventional wall mounted puffers, supersonic gas injectors (SGIs), or molecular cluster injection [17]. Use of the SGI resulted in a factor of three improvement in fuelling efficiency, compared with conventional wall mounted gas puffers, in CDX-U. Except for the prefill by wall-mounted puffers, the gas fuelling systems for LTX are all designed to produce sub-millisecond initiation and termination of gas flow. Fueling will be pulsed, in a fashion similar to pellet fuelling, so that in the interval between gas pulses the neutral gas density in the plasma edge will be reduced to low levels. The ASTRA-ESC code has been used to project the performance of LTX. For the initial phase of LTX operation, the simulations indicate that confinement times of $\sim$25 ms, and core electron temperatures of $\sim$1.5 keV, can be expected. Pulsing the gas sources off to transiently remove

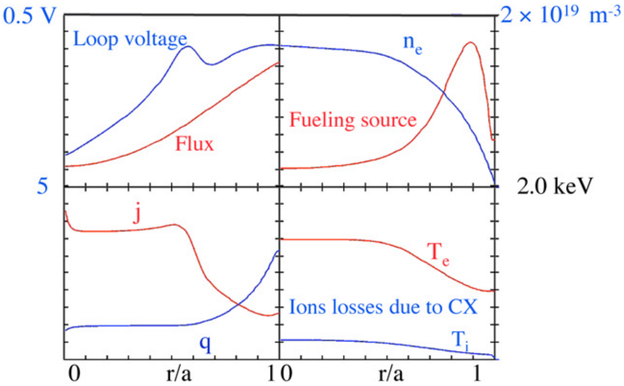

Figure 4. Stationary plasma profiles, as functions of the normalized minor radius, predicted for LTX with the RTM in ASTRA-ESC for Ohmic discharges.

the neutral gas load from the edge is predicted to result in an electron temperature profile with $T_e(a) > T_e(0)$, and an increase in confinement to >30 ms.

Neutral beam injection (NBI) is planned for a later phase, in order to provide core fuelling of hot ions. This would constitute the possibility of achieving the innovative, Li wall fusion regime, which combines core fuelling with pumping wall conditions. ASTRA modelling of the LiWF regime in LTX with NBI indicates that the ion heating should be very effective in LTX, since the total Ohmic input power is small (due to the very low loop voltage), and the ions are not strongly coupled to the electrons in this modest density regime.

4.1. Modelling of the Ohmic regime for LTX with 0.3 MA plasma current

We first consider LTX with Ohmic heating alone, which is relevant to first lithium operation.

For a toroidal field of 0.35 kG and plasma current $I_{pl} = 0.3$ MA, with gas fuelling, even the Ohmic heating regime is expected to offer significant performance for a small tokamak. Figure 4 shows the ASTRA-ESC evaluation of one of the possible (low beta) regimes, with electron temperature $T_e(0) = 1.4$ keV, density $n_e(0) = 1.65 \times 10^{19}\,\mathrm{m}^{-3}$, and energy confinement time $\tau_E = 25$ ms. The ion temperature, $T_i(0) = 0.22$ keV, remains relatively low (although much higher than in CDX-U) because of the weak coupling of the ions to the electrons. The code also shows that the volt-second requirements for this regime are well within the capacity of the central solenoid of LTX. A full ASTRA-ESC survey of the available equilibria, with variations in the gas fuelling rate to explore the available density range, plasma current, etc has not yet been performed. In fact, a wide range of regimes with higher beta can be obtained in LTX even with Ohmically heated plasmas.

4.2. Initial NBI heated regime in LTX

Starting in 2010, LTX will employ a neutral beam (5 A, 15–20 keV, in H) originally intended as a diagnostic beam for the now-cancelled NCSX project. ASTRA with the RTM has been used to model modest beam heating in LTX, up to the ∼100 kW level. Under these conditions, LTX is predicted to access a hot-ion regime, with $T_i > T_e$. Figure 5 shows the simulation results. Significant supplemental gas fuelling is required to maintain the plasma density, since at the projected confinement time the total required fuelling current is 20–30 A, well in excess of the 5A supplied by the neutral beam. The simulation shows the possibility of achieving a central ion temperature $T_i(0) = 1.63$ keV, exceeding the electron temperature $T_e(0) = 1.33$ keV and a high energy confinement time, $\tau_E = 59$ ms. The flattop loop voltage drops to <0.3 V. The reduction in edge gas results in a nearly flat predicted electron temperature profile, and a broad ion temperature profile with a pronounced edge pedestal.

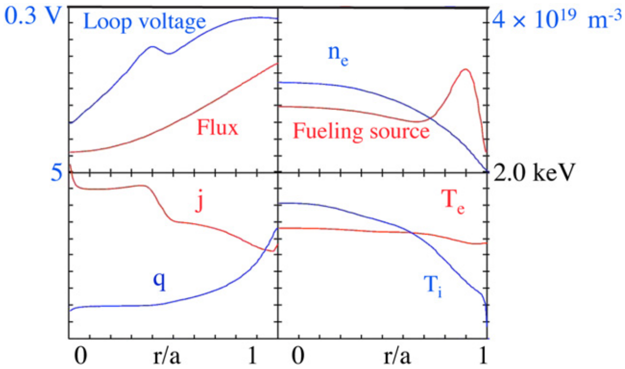

Figure 5. Stationary plasma profiles, for the hot-ion regime in LTX. A neutral beam heated LTX discharge is modelled by the RTM in ASTRA-ESC, with $P_{nbi} = 0.09$ MW deposited in the plasma.

In the future, full core fuelling of LTX would be possible with the addition of two 15 A, 8–12 keV neutral beam ion sources. These beam systems have been developed and are available [18].

5. Design and status of LTX

The first construction phase of LTX has now been completed, and the machine achieved first plasma on 3 October, 2008. Discharge development is underway, and installation of the remaining initial diagnostic set, as well as implementation of a new IGBT-based Ohmic power supply, is expected in late spring, 2009. Lithium operation is expected to follow, in mid-2009. Elevation and cutaway views of the machine are shown in figure 6. Two views of LTX are shown in figure 6, which illustrate the essential features of the device.

Central to the LTX concept is a heated, conformal shell, coated with molten lithium. The shell is formed of 1/16″ 304 stainless steel explosively bonded to 3/8″ OFHC copper, and is heated with commercial resistive cable heaters. The shell has two toroidal breaks and two poloidal breaks (best seen in figure 7); the outer equatorial plane break also provides toroidally continuous diagnostic access. The shell is seen mounted in the vessel in figure 6(*b*). Both views in the figure also show the shell support structure, which is designed for both mechanical and 1 kV electrical isolation of each of the four shell segments from the vacuum vessel. Mechanical support for the shell segments is provided by four legs per segment. Each leg extends through the upper and lower vessel flanges via a vacuum electrical break and a formed bellows, and is supported externally off the vacuum vessel. This approach

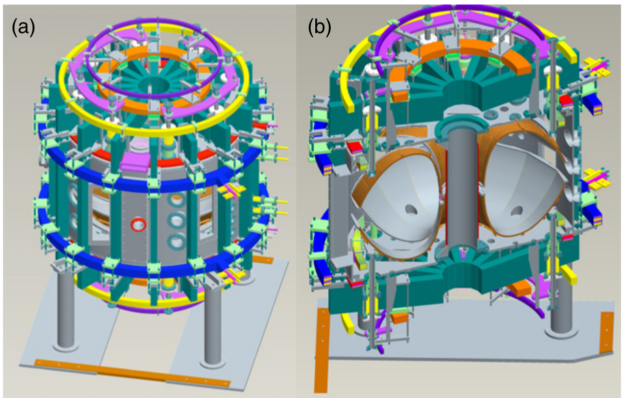

Figure 6. Elevation (*a*) and section (*b*) of LTX, showing the internal heated shell.

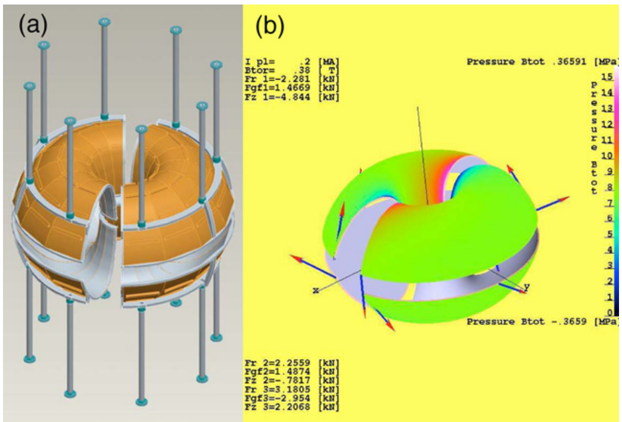

Figure 7. CAD view of the shell and support structure (*a*), and calculated forces on the shell during a disruption (*b*). The total overturning moment on the shell is approximately 5 kN.

avoids supporting the shell segments on internal high voltage ceramic breaks, which would be subject to repeated mechanical shock during disruptions, due to the overturning moment on the shell segments. The support structure also allows for external compensation for thermal expansion. The shell itself, with support legs, is shown in figure 7(*a*), along with the calculated distribution of forces during a disruption, in figure 7(*b*).

Tubular cable heating elements (not shown) are clamped onto the outer, copper surface in order to maintain a temperature of up to 400, or 500 °C for short periods. LTX is the first tokamak able to operate with a wall in this temperature range. The cable heaters are constructed with long cold sections at the terminating ends; all sections of the heater not in good thermal contact with the shell are unheated. Vacuum isolation is through Swagelok fittings so that all electrical connections for the heaters are made outside the vessel. The shell segments are individually electrically isolated through insulating supports and electrical breaks on the heater feedthroughs in order to facilitate glow discharge cleaning (GDC) of the inner shell surface. A photograph of the assembled LTX is shown in figure 8. LTX achieved first plasma on 3 October 2008.

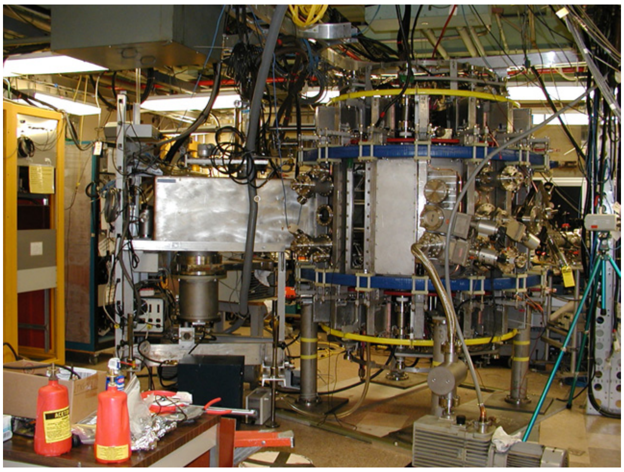

Figure 8. LTX assembled for the first plasma, on 3 October 2008.

6. Summary

Modelling of Ohmic discharges indicates a diffusion-based confinement regime resulting from the lithium low recycling plasma facing surfaces in CDX-U. The global energy confinement time is consistent with, or somewhat better than, modelling with the RTM, which assumes particle transport at the ion neoclassical rate, and implements nonrecycling boundary conditions.

Use of the same model to project the performance of LTX indicates the possibility of achieving a hot electron plasma ($T_e > 1.5$ keV, $\tau_E > 20$ ms) with Ohmic heating, as well as of attaining the Li wall fusion regime with NBI and a hot ($T_i \sim T_e \sim 1.5$ keV), well-confined ($\tau_E > 50$ ms), low collisionality ($\nu^*_{i,e} \sim 0.01$) plasma. The first tokamak discharge in LTX has now been produced.

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