

Performance Projections for the Lithium Tokamak eXperiment (LTX)

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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. Numerical simulations of CDX-U low recycling discharges have now been performed with the ASTRA code, utilizing a model with neoclassical ion transport and boundary conditions suitable to a nonrecycling wall, with fueling via edge gas puffing. This transport model has successfully reproduced the experimental values of the energy confinement (5 – 6 msec), loop voltage (<0.5V), and density for a typical CDX-U lithium discharge. In addition, DEGAS 2 modeling has now been performed for low recycling CDX-U discharges. The transport model benchmarked with CDX-U data has also been used to project the performance of the new Lithium Tokamak eXperiment (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 msec.

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 toward a very low recycling wall. A transport model (designated the Reference Transport Model, or RTM) has been developed and employed in the ASTRA-ESC [4] code 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 modeling CDX-U low recycling discharges, the model has been used to project performance for LTX.

LTX is also nearing first plasma, scheduled for the end of September as of this writing. LTX is the first tokamak designed around the use of liquid lithium as a PFC.

2. Modeling of CDX-U with ASTRA

Liquid lithium limiter experiments in CDX-U demonstrated a significant, 5-10 fold enhancement of confinement, when the lithium surface was in contact with the plasma edge.[1,5] These experiments, although not equipped with detailed profile diagnostics, nonetheless offer the possibility of testing different transport models against CDX-U data.

Figure 1 shows the simulation results for CDX-U lithium discharges, using the RTM in the ASTRA-ESC code system. 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. Zero recycling boundary conditions were used in the simulation, which yields a high edge electron temperature.

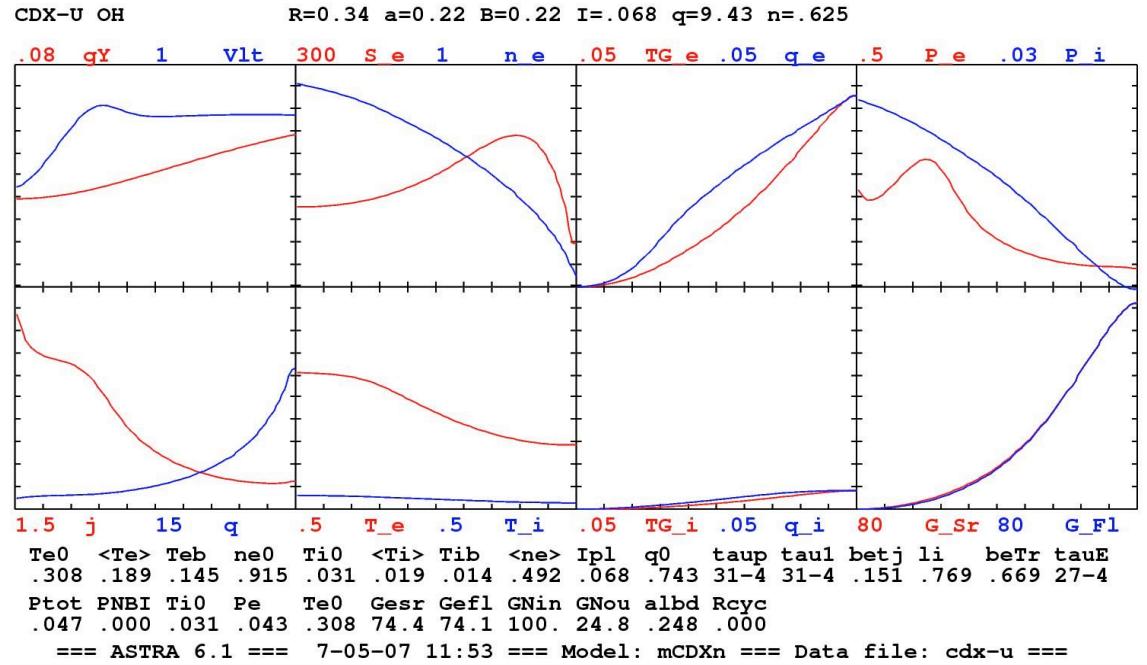


FIGURE 1. STATIONARY PLASMA PROFILES AS FUNCTIONS OF THE MINOR RADIUS, PREDICTED USING RTM FOR CDX-U PLASMA. BEGINNING WITH THE UPPER LEFT FRAME, GY IS THE FLUX CONSUMPTION, VLT IS THE LOOP VOLTAGE, S_E IS THE PARTICLE SOURCE DUE TO EDGE FUELING, N_E IS THE ELECTRON DENSITY, TG_E, Q_E, ARE THE CONVECTIVE AND TOTAL HEAT FLUX IN THE ELECTRON CHANNEL, P_E, P_I ARE THE POWER DEPOSITION PROFILES INTO BOTH SPECIES, J, Q ARE THE CURRENT DENSITY AND Q-PROFILE, T_E, T_I ARE THE TEMPERATURE PROFILES, TG_I, Q_I, ARE THE CONVECTIVE AND TOTAL HEAT FLUX IN THE ION CHANNEL AND G_FL, G_SR ARE THE PARTICLE FLUX AND INTEGRATED PARTICLE SOURCE, WHICH COINCIDE IN THE STATIONARY PHASE. THE VARIABLE RANGE IS INDICATED ABOVE THE UPPER ROWS OF PLOTS, AND BELOW THE LOWER ROW.

A comparison of ASTRA-ESC simulations using the RTM to CDX-U parameters is shown in 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 long energy confinement time observed in the CDX-U lithium experiments. The model is much more sensitive to the particle diffusion model than to thermo-conduction. Thus, with a small reduction in diffusivity (to $0.8\chi_i^{\text{neoclassical}}$) it can reproduce all the CDX-U reference parameters listed in Table 1. For a comparison, the GLF23 transport model has been included in the ASTRA-ESC modeling, as an additive transport term to the RTM. GLF23 was developed to model anomalous transport regimes in conventional tokamaks. Since conduction losses modeled by GLF23 are virtually “turned off” by the lack of a significant electron temperature gradient, 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. We therefore use the RTM to project performance for LTX.

Parameter	CDX-U	RTM	RTM-0.8	RTM-0.65	GLF23+ RTM	Comment
Fueling, 10^{21} /sec β_i	1-2 0.16	0.98 0.15	0.5 0.15	0.3 0.151	3 0.145	Gas puffing rate adjusted to match measured β_i
ℓ_i	0.66	0.77	0.702	0.671	0.877	Internal inductance
V, Volts	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	~1	0.9	0.7	0.596	0.9	Central density
$T_e(0)$, keV	Not measured	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. COMPARISON OF CDX-U LOW RECYCLING DISCHARGE WITH THE ASTRA REFERENCE TRANSPORT MODEL (RTM), RTM WITH REDUCED NEOCLASSICAL TRANSPORT (SCALED BY 0.8 OR 0.65), AND RTM COMBINED WITH THE GLF-23 TRANSPORT MODEL. THE LATTER COMPARISON INDICATES THAT ANOMALOUS TRANSPORT IS NOT A SIGNIFICANT CONTRIBUTOR.

The time history of the CDX-U discharges was not reproduced. There is no reliable equilibrium reconstruction data for the ramp-up phase of the discharge. We note also that transport simulations of a plasma discharge with a flattop duration comparable to the energy confinement time are difficult. The time dependence of the gas fueling, 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 $v_{i,e}^*$ < 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 fueled lithium discharges. It is possible that the somewhat reduced confinement time, relative to the experimental results, seen in modeling 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. Finally, we note that numerous Tokamak Simulation Code (TSC) simulations of CDX-U under high recycling conditions have been performed, primarily as input to the M3D code in order to develop numerical models for sawtoothing and other MHD activity.[6] TSC modeling indicated that energy confinement times with a high recycling edge should range from 0.15 ms [6] to 0.4 ms, [7] 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 LTX. Note that simulations of the LTX plasma, which will have a controlled current flattop, will be more reliable.

3. The Lithium Tokamak eXperiment (LTX).

The ASTRA model has also been used to project the performance of 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 400kA, with a 50 msec flattop. The toroidal field will also be increased to 3.4 kG. A comparison of CDX-U and LTX is shown in Table 2. Initially LTX will be fueled via edge gas puffing, using either conventional wall mounted

Parameter	CDX-U	LTX
Major radius	0.34 m	0.4 m
Minor radius	0.22 m	0.26 m
Toroidal field	0.21 T	0.34 T
Plasma current	100 kA	400 kA
Current flattop	5 ms	>100 ms
Ohmic flux	30 mV-s	160 mV-s (centerstack maximum: 225 mV-s)

TABLE 2. COMPARISON OF CDX-U AND LTX PARAMETERS.

puffers or supersonic gas injectors (SGIs) close-coupled to the plasma for improved fueling efficiency. Use of the SGI resulted in a factor of three improvement in fueling efficiency, compared to conventional wall mounted gas puffers, in CDX-U. ASTRA modeling for this initial phase of operation has been performed, and indicates that confinement times of ~ 25 msec and core electron temperatures of ~ 1.5 keV will be produced. Pulsing the gas sources off to transiently remove the neutral gas load from the edge is predicted to result in an electron temperature profile with $Te(a) > Te(0)$. This technique may be repeated many times during a discharge, to produce an effect similar to multiple pellet fueling. Neutral beam injection is planned for a later phase, in order to provide core fueling of hot ions. ASTRA modeling of LTX with NBI has also been performed. Neutral beam heating of the ions is expected to 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.

3.1. Modeling of the Ohmic regime for LTX with 0.3 MA plasma current.

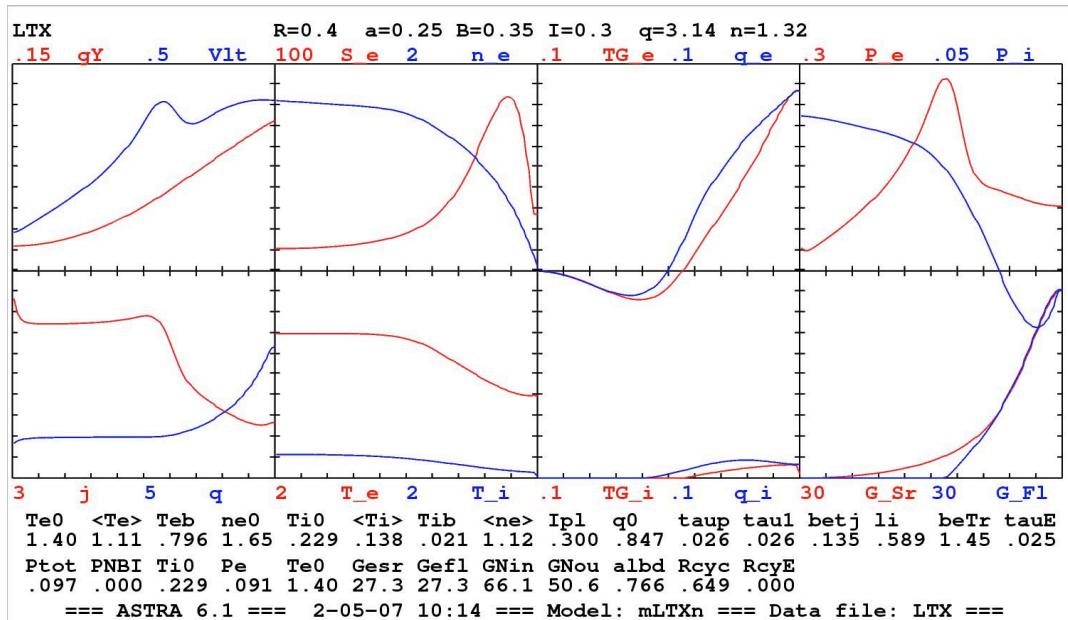


FIGURE 2. STATIONARY PLASMA PROFILES AS FUNCTIONS OF THE MINOR RADIUS FOR OHMIC HEATED LTX PLASMA (SEE FIG. 1 FOR THE FRAME NOTATION).

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 fueling, even the Ohmic heating regime is expected to offer significant performance for a small tokamak. Figure 2 shows the ASTRA-ESC evaluation of one of the possible (low beta) regimes, with electron temperature $T_e(0)=1.4$ keV, density $n_e(0) = 1.65 \times 10^{19}$ 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 weak coupling of the ions to the electrons. The code shows also 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 fueling 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.

3.2 Initial neutral beam injection (NBI) heated regime in LTX

LTX is expected to employ a neutral beam originally intended as a diagnostic beam for the now-cancelled NCSX project. Therefore, ASTRA with the RTM has been used to model modest beam heating in LTX, up to the ~100 kW level.

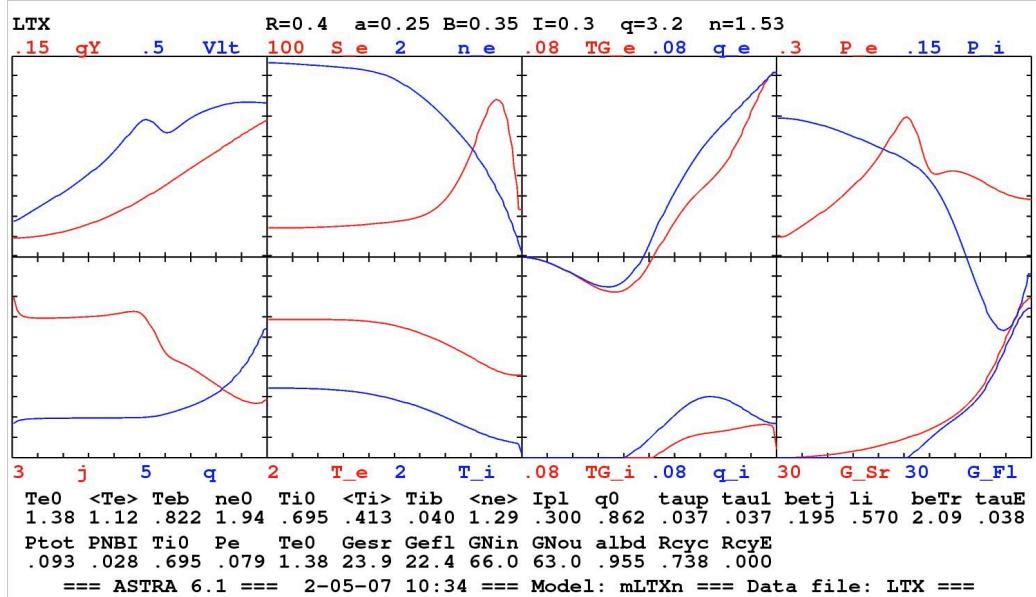


FIGURE 3. STATIONARY PLASMA PROFILES AS FUNCTIONS OF THE MINOR RADIUS FOR AN OHMIC AND NBI HEATED LTX PLASMA, WITH $P_{NBI} = 0.028$ MW DEPOSITED IN THE PLASMA (SEE FIG. 1 FOR THE FRAME NOTATION).

Figure 3 shows the prediction by ASTRA-ESC for LTX with the same gas fueling as the Ohmic regime (Fig.3.17), but with a small injected NB power (only 30 kW). Even such a small neutral beam power represents a significant increase over the power coupled from the electrons (note P_e , P_i in Fig. 1), so a substantial increase in the ion temperature, to $T_i(0) = 0.7$ keV, and the energy confinement time, to $\tau_E = 38$ ms, is expected.

3.3 Hot-ion regime with Ohmic + NBI heating in LTX

If we consider the injection of 90 kW of NBI heating power, then LTX will access the hot-ion regime, with $T_i > T_e$. Figure 4 shows the predicted regime, which requires partial gas

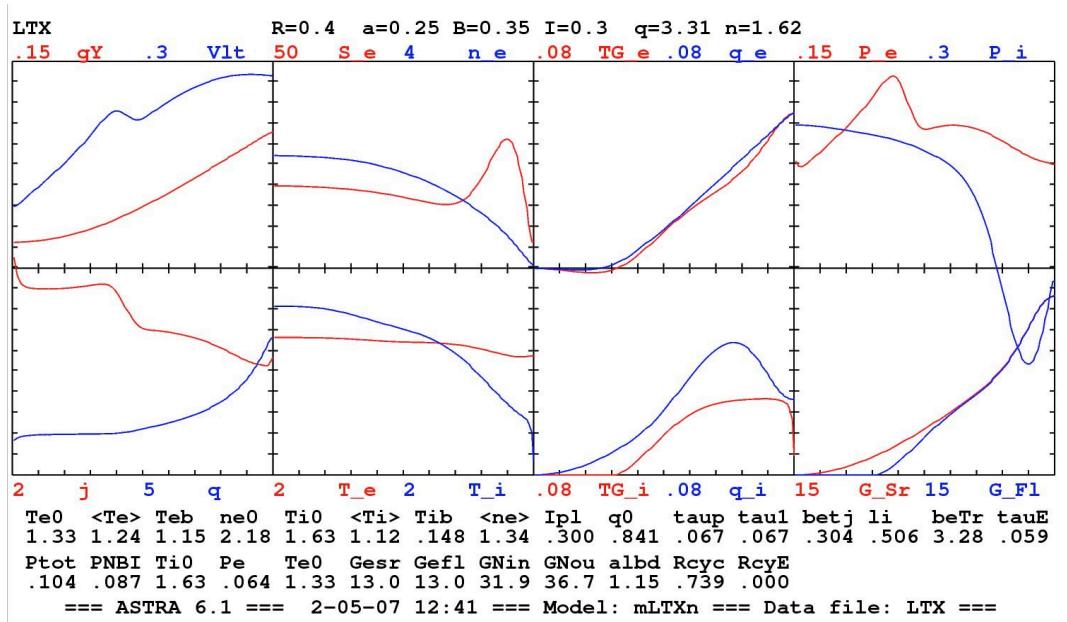


FIGURE 4. HOT-ION REGIME IN LTX FOR OHMIC AND NBI HEATED LTX PLASMA WITH $P_{NBI} = 0.09$ MW DEPOSITED IN THE PLASMA. SEE FIG. 3 FOR THE FRAME NOTATION.

fueling to maintain the density. 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 very high energy confinement time, $\tau_E = 59$ ms. The flattop loop voltage drops to < 0.3 V. Note that in the simulation the plasma is not fully sustained by NBI, although less gas fueling is required to maintain the density (in part because of the increase in confinement time). 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. However, 20 – 30 A of NBI, at 10 – 20 keV, is required to fully fuel the discharge. We are exploring options for full beam fueling of LTX.

4. Design and status of LTX

LTX, at this writing, is expected to achieve first plasma at the end of September. Initial operation will utilize a CDX-U capacitor bank for an Ohmic power supply, until the new IGBT controlled system is completed. Lithium operation is expected in mid 2009. Elevation and cutaway views of the machine are shown in Figure 5. Two views of LTX are shown in Figure 5, which illustrate the essential features of the device. The poloidal field coil set is visible in the external view of the device (Fig. 5A). The blue, red, yellow, and a new uncased internal coil comprise the PF set; all but the orange coils (which control vertical position and κ) are new for LTX. Equilibrium calculations indicate that the new PF coil set will support discharges with plasma currents > 400 kA, with a wide range of current profiles.

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 6); the outer equatorial plane break also provides toroidally continuous diagnostic access. The shell is seen mounted in the vessel in Figure 5B. 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

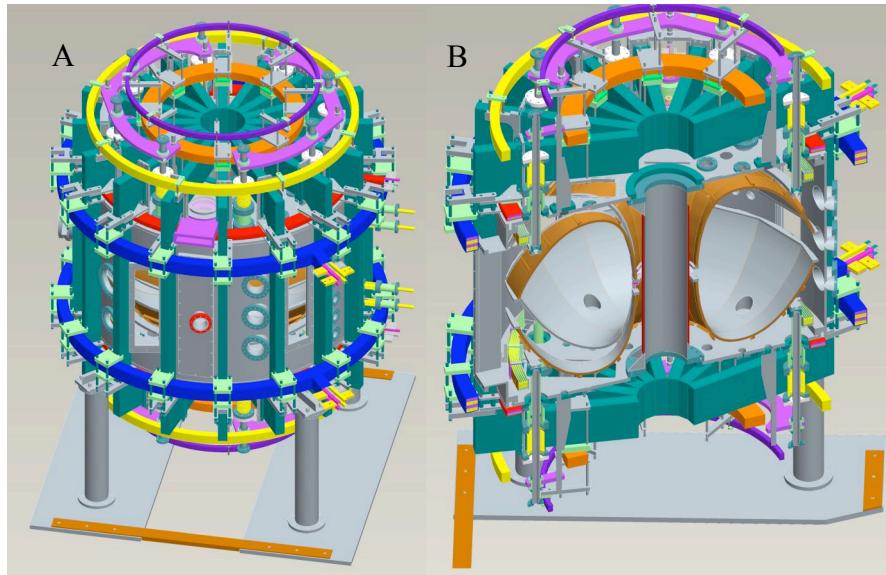


FIGURE 5.
ELEVATION (A)
AND SECTION (B)
OF LTX, SHOWING
THE INTERNAL
HEATED SHELL

formed bellows, and is supported externally off the vacuum vessel. This approach 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 shell itself, with support legs, is shown in Figure 6A, along with the calculated distribution of forces during a disruption, in Figure 6B.

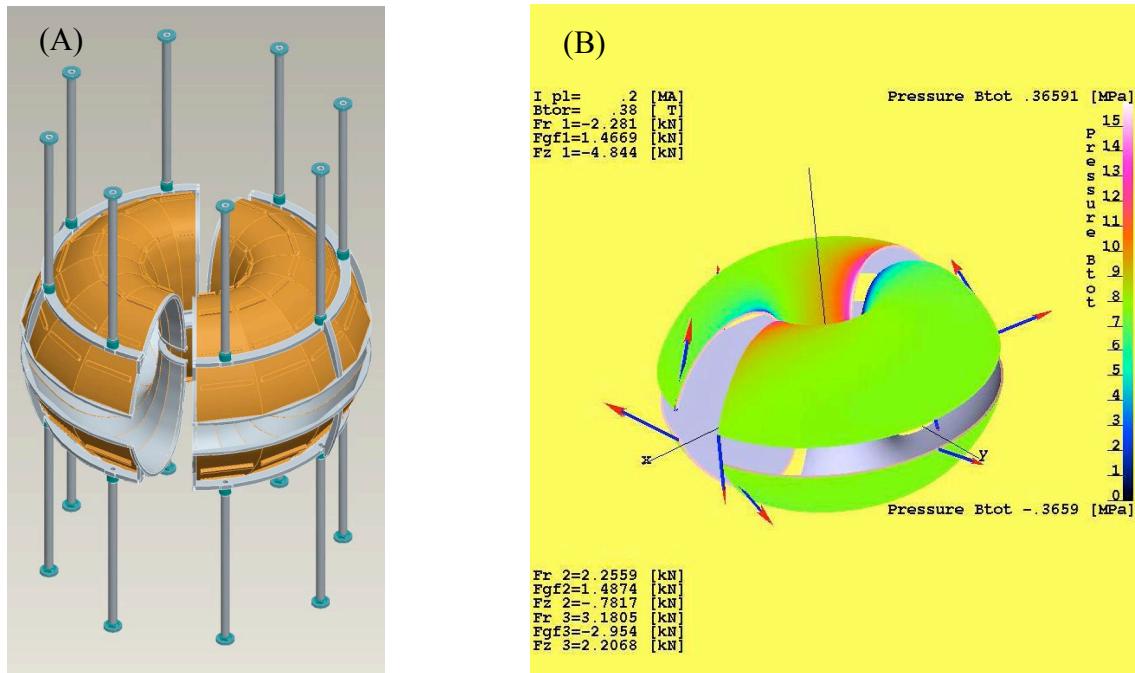


FIGURE 6. CAD VIEW OF THE SHELL AND SUPPORT STRUCTURE (A), AND CALCULATED FORCES ON THE SHELL DURING A DISRUPTION (B). THE TOTAL OVERTURNING MOMENT ON THE SHELL IS APPROXIMATELY 5 KN.

Tubular cable heating elements (not shown) are clamped onto the outer, copper surface in order to maintain a temperature of up to 400 °C, or 500 °C for short periods. These 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.

5. Summary

Modeling of low recycling Ohmic discharges in CDX-U indicates that the global energy confinement time is consistent with, or somewhat better than, a transport model which assumes that particle transport, at a rate no greater than ion neoclassical, is responsible for global energy loss. However, we note that experimental results cannot distinguish between the model employed here, and competing models for transport in low recycling tokamaks, which predict even longer confinement times (e.g. the “Isothermal Tokamak model of Catto and Hazeltine [8]). Use of the same model to project the performance of LTX indicates that hot ($T_i \sim T_e \sim 1.5$ keV), well confined ($\tau_e \sim 60$ msec) plasma regimes, at relevant low collisionality ($v_{i,e}^* \sim 0.01$), with Ohmic and very modest NBI heating.

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