First Results from the Lithium Tokamak eXperiment (LTX)

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Abstract. The Lithium Tokamak eXperiment (LTX) is a newly commissioned, modest-scale spherical tokamak with R=0.4 m, a=0.26 m, and κ=1.5. Design targets are $B_{toroidal}$ = 3.2 kG, I_P < 400 kA, and $\tau_{flattop}$ ~ 100 msec. LTX is the first tokamak designed to investigate modifications to equilibrium and transport when global recycling is reduced to 10 – 20%. To reduce recycling, LTX is fitted with a 1 cm thick heated (300 °C) copper shell, conformal to the last closed flux surface, over 85% of the plasma surface area. The plasma-facing surface of the shell is formed of 1.5 mm thick, explosively bonded, 304 stainless steel, and will be evaporatively coated with a thin (< 100 micron) layer of molten lithium, retained by surface tension. The shell is replaceable, and a second version has been constructed, for which the inner stainless steel surface was plasma-sprayed with 100 – 200 microns of molybdenum, to form a high-Z substrate for subsequent coating with lithium. After the installation of the second shell (in 2012), a high temperature (500 - 600 °C) operating phase for LTX is planned. LTX is the first tokamak designed to operate with a full hot high-Z wall, near the projected operating temperature for reactor PFCs. The engineering design and construction of the hot high-Z shell, as well as the vessel and diagnostics to tolerate both lithium and 500 °C internal components, will be discussed. LTX will employ short-pulse fueling with a new H_2 molecular cluster injector, to transiently eliminate edge gas (between puffs). This fueling system will be briefly discussed. Diagnostics include single-pulse multipoint Thomson scattering, Lyman alpha arrays, microwave interferometers, spectrometers, and an edge Langmuir probe. LTX is now progressing through the shakedown phase, and first operation with a liquid lithium film wall is scheduled for 2010. Late in 2010, a new diagnostic (Digital Holography) for core density variations will be tested on LTX. In 2011 a 5 A, 20 kV, 1 second hydrogen neutral beam, which will provide beam-based diagnostics, and significant core ion heating in LTX, will be installed.

1. Introduction

The Lithium Tokamak eXperiment (LTX) is a low aspect ratio (A=1.6) tokamak ,with major radius R=0.4 m, minor radius a=0.26 m, elongation κ=1.5, and modest triangularity δ < 0.2. Discharges are limited on a conformal wall; there is no provision for diverted operation, and indeed, part of the aim of the lithium-walled tokamak approach is to eliminate the need for a divertor. Design targets for the device are $B_{toroidal}$ = 3.4 kG, I_P < 400 kA, and $\tau_{flattop}$ ~ 100 msec, although during initial operation the device will be limited to $B_{toroidal}$ < 2.1 kG, I_P < 150 kA, and $\tau_{flattop}$ ~ 25 msec. LTX is the first tokamak designed to investigate the modifications to tokamak equilibrium and transport when global recycling is reduced to very low values[1], near the lower limit imposed by direct reflection of deuterons from plasma-facing components (PFCs) composed of lithium. For projected LTX parameters this should impose a lower limit for the recycling coefficient of order 10%.

2. Lithium-coated heated shell

In order to reduce recycling on lithium to this low level, LTX is fitted with a conformal 1 cm thick heated copper shell, suspended within the vacuum vessel on a system of legs or risers to permit thermal expansion relative to the vacuum vessel, and provide electrical isolation from the vessel. The plasma-facing surface of the shell is clad with an explosively bonded 1.5 mm thick layer of 304 stainless steel. The shell is conformal to the last closed flux surface, and covers 85% of the plasma surface area. The shell temperature is controlled with a system of electrical cable heaters, with a total power rating of 37 kW, and is designed for a maximum operating temperature in excess of 500 °C, although routine operation will be in the 300 - 400 °C range, above the melting point of lithium (180.5 °C).

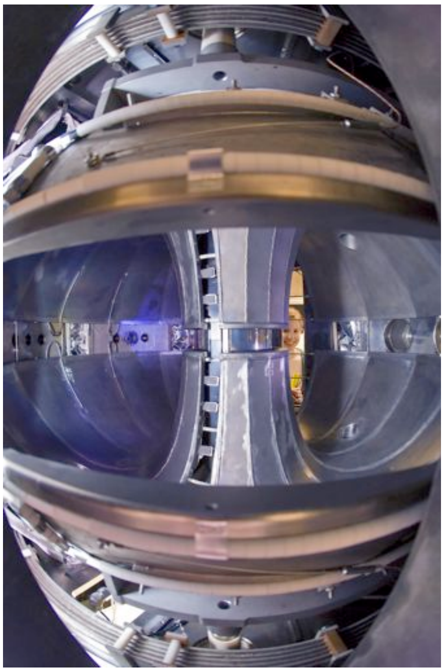

Figure 1. Fish-eye view of the LTX heated shell though a vacuum vessel port, during a vent. The interior of the shell is conformal to the last closed flux surface (LCFS); the plasma control system is designed to maintain a gap of less than 1 cm between the LCFS and the shell surface. The shell is segmented toroidally and poloidally into 4 sections, all of which are electrically isolated from each other and the vacuum vessel. The upper shell segments are separated from the lower by a gap of 5 cm at the inboard midplane, and 12.5 cm at the outboard midplane. Toroidal segmentation is provided by two gaps, 180° apart, each subtending 22.5° in toroidal angle. Also visible in the photograph as white rings encircling the shells are the flux loops, which are clamped to the outside of the shell and encased in ceramic tubing. Finally, the uppermost and lowermost structures in the photograph are the fast, gas cooled, uncased, 5-turn internal poloidal field coils.

The interior, plasma-facing surface, of the shell is primarily designed to be evaporatively coated with a thin (less than 100 micron thick) layer of molten lithium, which will be retained by surface tension and capillary forces. The surface tension of lithium in the 300 °C temperature range is approximately 400 dynes/cm, or 5 ½ times the surface tension of water at room temperature, while the density of liquid lithium is only half that of liquid water. In addition to thin, evaporated coatings of lithium over all of the upper and most of the lower shells, the lower shells were constructed with 1 cm tall welded lips which allow a pool of liquid lithium to be retained in the lower shells, similar to the pool of liquid lithium employed in the CDX-U lithium tray experiments.[2] The lower shells are also fitted with 1.1 cm tall molybdenum limiters, which are designed to wick liquid lithium from the pool to the top surface of the molybdenum limiter assembly. The shell pools will be filled using the two liquid lithium fill systems designed, and constructed, by UCSD for earlier use in CDX-U. The relative effectiveness of pools and thin films of liquid lithium at reducing recycling can then be compared in the same experiment. Experiments with a ~0.5 cm deep liquid lithium pool in the lower shell segments are planned for later in 2011.

The outer (copper) surface of the shell has been electroplated with a 200 micron layer of hard nickel. The nickel plating provides a partial barrier against chemical attack of the copper by hot lithium, and also provides uniformly low, reproducible thermal emissivity, to control radiative power losses at high temperatures. The shell is replaceable, and a second version has been constructed, which was plasma-sprayed with 100 – 200 microns of molybdenum, on the stainless-steel inner surface, to form a high-Z substrate for subsequent coating with lithium. After the installation of the second shell, a hot (500 - 600 °C) operating phase for LTX is planned, in order to investigate plasma-material interactions with a hot high-Z wall. LTX is the first tokamak in the world designed to operate with a full high-Z wall in this temperature range, near the projected operating temperature for reactor PFCs. The shell has now been tested to 300 °C, which is sufficient for intial lithium operation. During 300 °C operation, the thermal shielding installed in LTX limited the temperature rise of the centerstack to 5 °C. The maximum shell temperature at full heater power, based on this testing, projects to 560 °C, in good agreement with the results of ANSYS thermal modeling. A photograph of the shell taken through a port during a recent vent is shown in Figure 1. The results of ANSYS modeling of shell operation at 500 °C are shown in Figure 2.

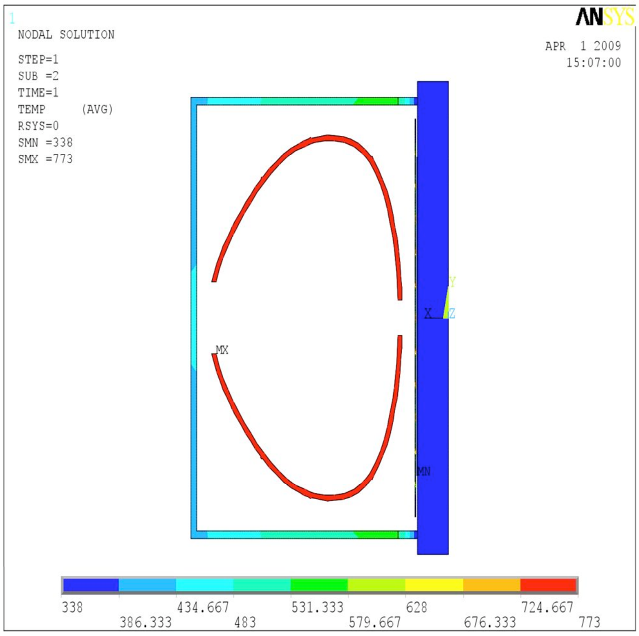

Figure 2. ANSYS modeling for 500 °C operationof the LTX shell. 29 kW of electrical heater power is required. The centerstack and vessel wall are fitted with external water cooling tubes. The centerstack is fitted with a passive heat shield of polished stainless steel over multilayer silicon-bonded mica. Future installation of a helium-cooled shield is planned. The shell now baked routinely to 300 °C; at this temperature the vacuum vessel did not require water cooling, since the vessel temperature rise could be limited to 20 °C with external air circulation. The centerstack ΔT was < 4 °C. The projected continuous wall temperature limit is 560 °C.

Although the shell has two poloidal and two toroidal gaps, the high electrical conductivity of the 1 cm thick copper shell, combined with the low aspect ratio (A=1.6) geometry introduce significant eddy current shielding of poloidal fields during startup. We have developed approaches to modeling nonaxisymmetric eddy currents flowing in the thick copper shell, and the effects of these eddy currents on null formation and poloidal field evolution. The primary tool used to evaluate the effect of the eddy currents in the copper shells on null formation is a 2D axisymmetric code, LRDFIT, which represents the shells by axisymmetric coupled current elements with a finite (polygonal) cross section. Each element is assigned an electrical conductivity; the conductivity of the elements is varied individually to produce the best fit to the observed magnetic signals in LTX, due to the nonaxisymmetric shell currents. The total axis-encircling current is constrained to be zero, since the shell is toroidally segmented. The calculated shell currents and the fit of the modeled fields to the magnetic data is shown in Figure 3.[3] A fully nonaxisymmetric model of the conducting shell (CBSHL) has been developed by Zakharov. A CAD (ProEngineer) model of the shells, vacuum vessel, poloidal field coils, and diagnostics has now been imported into CBSHL, and preliminary results have been obtained with the new code. Future development of CBSHL will include a predictive mode of operation, where the code can be used to design the discharge including the effects of all nonaxisymmetric eddy currents.[3] Analysis of shell current data taken during disruptions will also be performed, and compared to current models for tokamak disruptions.

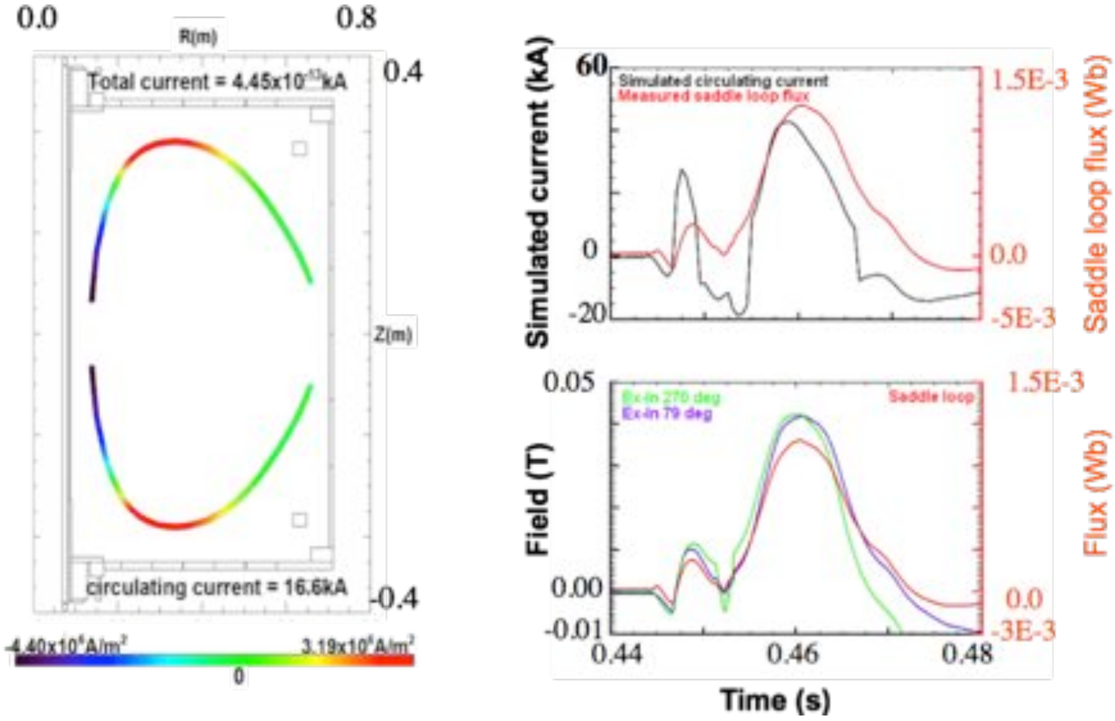

Figure 3. (a) Distribution of shell currents calculated with LRDFIT. (b) Comparison of the simulated current waveform and the signal from a set of saddle coils located in one of the shell toroidal gaps, and (c) a similar comparison with the Mirnov coils located in the shell gap.

LRDFIT has been employed for plasma reconstructions. A time sequence of reconstructions of an early, low current discharge in LTX using LRDFIT is shown in Figure 4.

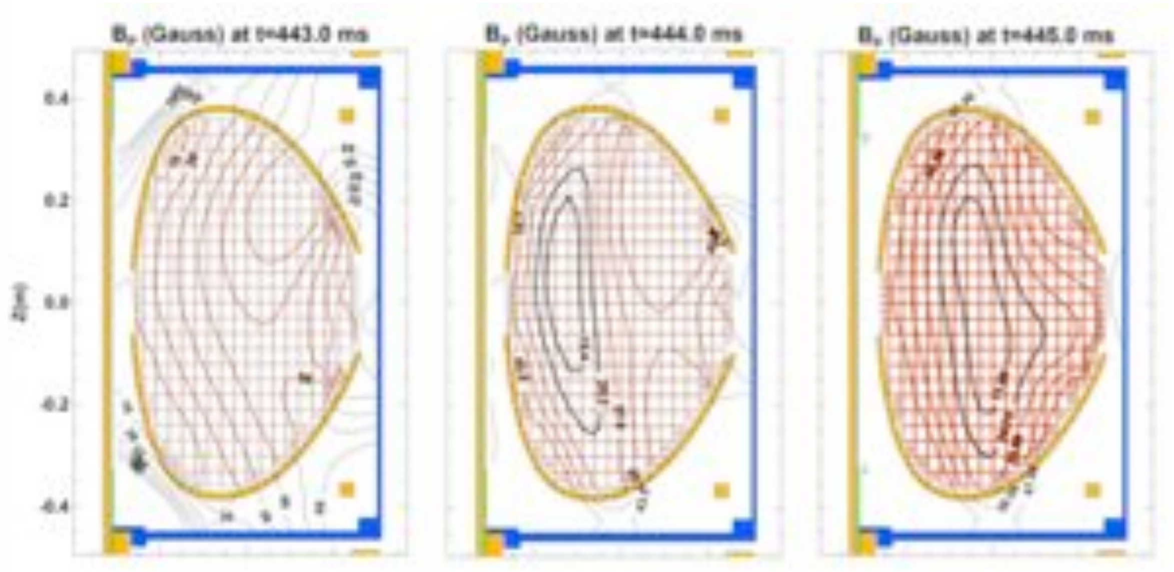

Figure 4. Sequence of LRDFIT reconstructions for an early, low current LTX discharge. Peak current was ~10 kA.

At this point in the commissioning of LTX, a variety of poloidal field coil programming sequences have been developed which allow for reliable null formation and development of a tokamak discharge.

3. Wall Conditioning

Most of the techniques commonly employed for wall conditioning in tokamaks cannot be used in LTX. The use of carbon, boron carbide, titanium carbide and similar low-Z materials as limiters is not tolerated, since these materials are reactive with liquid lithium at elevated temperatures. Coatings composed of these materials would be expected to form over clean liquid lithium, and could provide recycling sites, and increase the global recycling coefficient. Techniques such as boronization, titanium gettering, and other wall conditioning techniques will not be employed in LTX. This strongly limits the range of available techniques for wall conditioning, and reduction of impurity influx.

As a result, it is not expected that high performance discharges will be obtained in LTX until lithium is introduced. In order to provide reference increased recycling discharges for comparison to discharges against liquid lithium walls, we will employ lithium evaporation into a helium glow internal to the shell structure, with the shell at room temperature. Solid lithium will provide a low-Z wall in LTX, but was shown in CDX-U to only modestly reduce recycling, if at all.[2] Two glow electrodes are now installed in LTX, and will be biased at 480 V.A.C. in 10 – 30 mTorr of helium to produce the discharge (A.C. glow was used to successfully condition the stainless steel lithium-filled tray in CDX-U). One of the two sets of crucible heaters and Y_2O_3 crucibles for lithium evaporation is shown in Figure 5. Initial operation will involve the evaporation of ~ 5 g of lithium from each crucible, although the crucible capacity is 30 g. The crucible and heater system has been tested for resistance to liquid lithium at a temperature of approximately 700 °C, for rapid evaoration of the lithium charge. As of this writing, final preparations for lithium wall conditioning with evaporation into a helium glow have been completed, and preliminary results are expected to be available at the Fusion Energy Conference 2010.

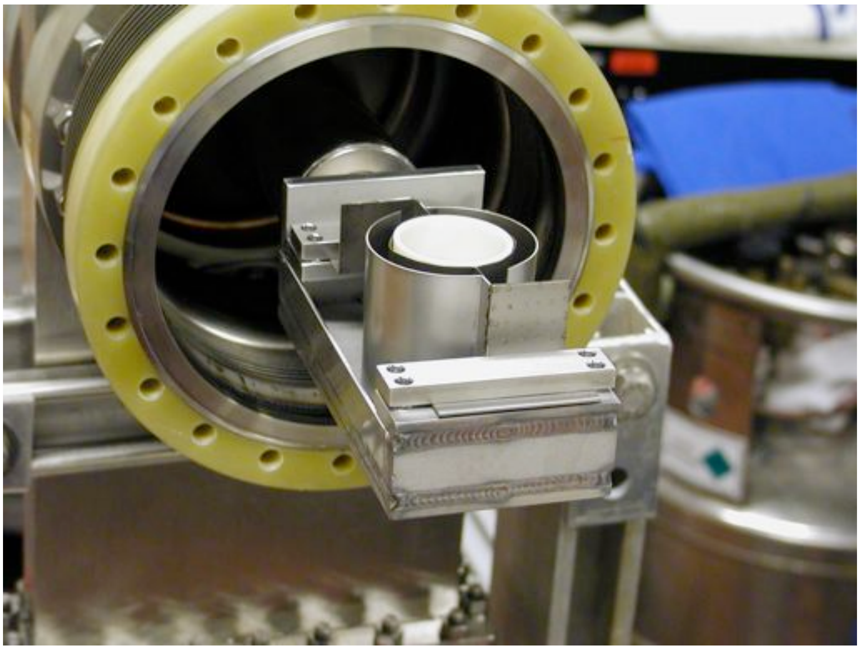

Figure 5. Photograph of a yttria crucible in a tantalum crucible heater. Two such lithium evaporation systems are installed on LTX.

Following the cold-wall phase, LTX will be operated with hot walls and thicker lithium films, produced by evaporation of up to 25 g of lithium from each of the two crucible systems. Following the campaign to investigate evaporated lithium film walls, a pool of liquid lithium, of up to 150 g in each of the two lower shell segments, will be introduced into the lower shell segments.

4. Fueling and diagnostics

Other novel LTX systems include short-pulse fueling with supersonic gas jets, and a new H_2 molecular cluster injector. These short-pulse, closely coupled fueling systems allow for the transient elimination of edge gas (between puffs). This is only the second test of a molecular cluster injector on a tokamak [4], and the first in an ST. The cluster injector has been operated up to a fueling rate of 700 Torr-liters/sec. Measurements of the gas plume from the cluster injector with a low energy electron beam show clear evidence of condensation when the gas reservoir is cooled to approximately 80 °K. The expansion of the gas plume seen at higher operating temperatures is absent. The cluster injector valve and nozzle, along with electron beam excitation data, is shown in Figure 6.

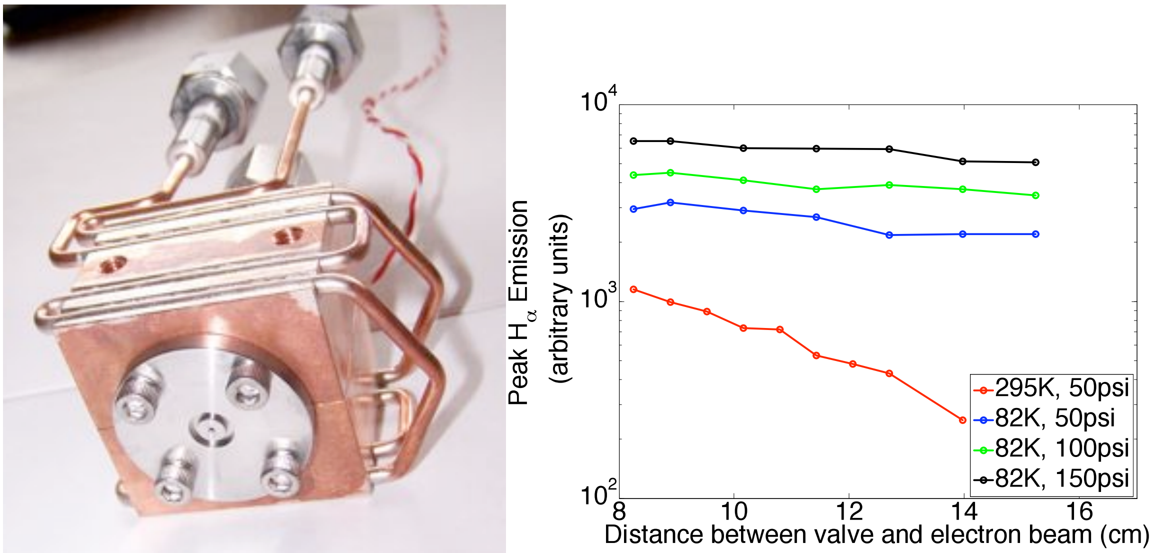

Figure 6. Photograph of the fast valve, with cooling jacket, employed in the hydrogen cluster injector, and H_α *data obtained with low energy electron beam excitation, showing a strong decrease in dispersion of the gas plume as the gas reservoir is cooled.*

Diagnostics installed on LTX include single-pulse multipoint Thomson scattering at up to 16 radial locations, multiple Lyman alpha arrays for recycling determinations, a fixed 1mm and a movable 2 mm microwave interferometer, a VUV survey spectrometer, filterscopes, visible spectrometers, and an edge Langmuir probe. A new diagnostic (Digital Holography) for core density variations will be tested on LTX. Improvements to LTX to be made in the 2011 – 2012 time frame include 1) An upgrade of the Ohmic power supply to enable full current (400 kA) operation, 2) The installation of 5 high resolution (1 – 2 mm) Thomson scattering channels viewing the edge plasma, 3) An upgrade of the toroidal field coil cooling to allow 3.2 kG operation, and 4) The installation of a 5 A, 20 kV, 1 second hydrogen neutral beam, which will provide beam-based diagnostics, and significant core ion heating in LTX [1].

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