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LTX: Exploring the advantages of liquid lithium walls

LTX is a low aspect ratio tokamak with a heated liner or shell, which covers 80% of the plasma surface area (5 m^2). The shell can be heated to 300 - 400 C, and coated with lithium. The research goals for LTX are, first, the investigation of tokamak equilibrium and confinement with low recycling, liquid lithium walls. Second, LTX is necessarily developing the technology necessary for safe deployment of liquid lithium plasma-facing components (PFCs) in a tokamak.

We have recently developed a new approach to coating the preheated (300 C) shells with lithium, to generate large-area, liquid lithium, plasma facing surfaces. An electron beam system is used to evaporate lithium from a pool of liquid at the bottom of the lower shell. The e-beam system produces a clean, 50 – 100 nm thick coating of liquefied lithium in < 5 minutes. At present, this system can only coat approximately half the shell inner surface. Tokamak discharges are now thus limited by 2 m^2 of liquid lithium. Analysis of diamagnetic loop and other magnetic data for these discharges indicates that confinement in LTX Ohmic discharges is now improved by 10×, compared to best previous results, which approximately obeyed ITER89P L-mode scaling. Confinement time with liquid lithium PFCs now exceeds ITER98P(y,2) ELMy H-mode scaling by 4-5×. This is the first experimental evidence that high performance tokamak discharges are compatible with large-area liquid lithium walls. Note that the newest confinement results have not yet been confirmed with kinetic measurements (Thomson scattering + ion temperature estimates), although for comparison discharges, which obeyed ITER89P, there was good agreement between kinetic and magnetic estimates of confinement time. We are installing a second electron beam system to coat the remainder of the shell area, which should expand liquid lithium coverage to 4 m^2, or roughly 80% of the total plasma surface area. Confinement results with both electron beam systems and Thomson scattering are expected by mid – late summer 2014. New results should be available for the final version of this whitepaper.

These recent results from LTX confirm, and extend, earlier (2005) results on confinement with low recycling lithium walls from CDX-U.

The next step for LTX would involve the installation of a neutral beam to heat the plasma with auxiliary power levels approximately 10× the Ohmic power input, and efficient core fueling. Presently LTX uses ducted gas injectors for plasma fueling, both with and without Laval nozzles to collimate the gas jet. These approaches provide a fueling efficiency up to 35% - much higher than traditional wall-localized gas puffing or recycling, but much less than can be obtained with core neutral beam fueling. LTX operates at present with Ohmic heating only, and a test of confinement with ion heating is needed to provide confident extrapolation to larger devices or reactors. NBI will also greatly extend diagnostic and physics capabilities. A beam has been offered to the LTX group by Tri-Alpha Energy.

To summarize, here we propose a modest initiative which would allow extension of Ohmic results to a determination of the confinement increase afforded by low recycling liquid lithium walls in a tokamak with strong auxiliary heating, core fueling, and a fast ion population, as well as deployment of additional diagnostics afforded by a neutral beam. This latter includes CHERs to determine plasma rotation and core lithium concentrations at various wall temperatures (liquid lithium PFC temperatures). This program will address a number of ReNeW thrusts, among them Thrust 5 (Expanding the limits for controlling fusion plasmas), Thrusts 9 and 10 (Understanding boundary layer plasmas and plasma-surface interactions), Thrust 12 (Demonstrating an integrated solution for plasma-material interactions with high core performance), and Thrust 16 (Develop the spherical torus to advance fusion nuclear science).