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***Abstract* — The use of lithium to coat plasma-facing components (PFCs) or serve directly as liquid PFCs has resulted in energy confinement improvement in several devices. Coupled with the demonstrated ability of liquid Li to exhaust high power fluxes, the use of Li could resolve the two leading problems of high-Z PFCs. The effect of Li in NSTX, EAST, and DIII-D is compared, and common features are identified. A compact spherical tokamak-based Fusion Nuclear Science Facility design could deliver an attractive neutral wall loading for materials testing, with Li PFCs enabling access to the required high confinement scenarios.**

Keywords — plasma facing components; impurities; lithium; energy confinement; power exhaust

I. Introduction

Solid plasma-facing components (PFCs) are the leading candidates for future devices, and largely serve as the PFCs for present devices. ITER is relying on metallic PFCs, namely tungsten (W) in the divertor and Be on the first wall. While ITER's scenarios have been designed to work with these PFC materials, there is little safety margin on heat flux removal capability[1]. The power exhaust challenge for reactors the size of ITER is substantially harder, requiring substantially higher amounts of core and divertor radiation[2]. Moreover, studies performed over the last 5 years since the MFE ReNeW strategic planning study[3] have shown that both the steady heat exhaust and transient exhaust, during e.g. edge-localized modes (ELMs)[4, 5], is more challenging, owing in part to the inverse dependence of the scrape-off layer power flux footprint with increasing midplane poloidal magnetic field[6, 7].

Liquid metal (LM) flowing PFCs have some attractive features that could remove some of the restrictions of solid PFCs. The typical erosion and PFC performance degradation of solid PFCs can be obviated with self-healing liquid surfaces; the challenge shifts to controlling core impurity content. Similarly LM PFCs are also tolerant to neutron damage. Under the right conditions LM PFCs can exhaust very high steady and transient heat flux.

The leading LM candidate is liquid lithium (Li), which can provide access to low recycling, high confinement regimes[8, 9], e.g. at ≥ 2 times H-mode scaling laws, around which attractive core and pedestal plasma scenarios can be based. The knowledge gaps for LM PFCs include keeping the surfaces clean for reliable flow, counteracting MHD mass ejection forces, determining operating temperature windows, and demonstrating He ash exhaust. These gaps can be addressed by R&D in dedicated test stands, as well as fundamental surface science studies.

In the remainder of this paper, we will review and compare evidence on enhancement of confinement with Li conditioning on several devices, and identify an attractive scenario for a Fusion Nuclear Science Facility that makes use of the high confinement for access to a high neutron fluence scenario for component testing. We assert that, with the rest of the world fusion community focusing on evaluating and trying to extend the capabilities of solid PFCs, the development of flowing LM PFC is a transformative area in which the US can lead the world toward fusion power realization.

II. Tungsten as the Leading PFC Candidate Material

Due to its many special properties, W is the leading candidate for solid PFCs for future devices. The accepted heat flux limit for W is 5-15 MW/m^2, with the precise value depending on allowed transients. The divertor in ITER is designed with W monoblock tiles, along with Be on the first wall; the designed divertor steady heat flux limit is 10 MW/m^2. Looking ahead to future devices, W thermal and structural properties degrade with appreciable neutron fluence, such that 5 MW/m^2 is the projected acceptable upper bound for steady heat flux removal[10]. W has several additional challenges[11]: the ductile-to-brittle transition temperature is undesirably high, and increases with neutron fluence. Thus, it is a concern that W will be brittle in some regions of the wall in a fusion reactor. Also, W develops nano-structures, i.e. "fuzz", bubbles, or dust, particularly under He bombardment at elevated temperatures. These structures contribute to erosion, reduction in PFC integrity and performance, and possible enhancement of tritium retention. W also suffers from high activation.

Because it is the leading solid PFC candidate material, much of the world's PSI program is focusing on W. To help prepare for ITER R&D, ASDEX-Upgrade replaced their carbon PFCs to W-coated graphite in a stepwise fashion 1996-2007, and now use solid W PFCs[12]. Operational scenarios with W typically yield reduced overall energy confinement and pedestal temperature as compared with carbon walls, but this can be compensated with N_2 seeding in ASDEX-Upgrade.[13] In addition, gas puffing is often required to keep the W edge source down, and central ECH is often required to reduce the core impurity confinement time. Similar results have been obtained in JET[14, 15], with the details strongly sensitive to the plasma boundary shape, e.g. Figure 1. Indeed operation at lower ν*, lower density plasmas seems to be inaccessible with the ITER-like wall in JET. An extra complication for JET is that N_2 seeding cannot be used in the upcoming D-T experiments, because of incompatibility with the tritium plant, and Ne seeding does not yet give comparable performance recovery.

Thus, although W has challenges, its overall properties make it the leading substrate candidate for LM PFCs. In other words, W would be more attractive if covered with a LM.

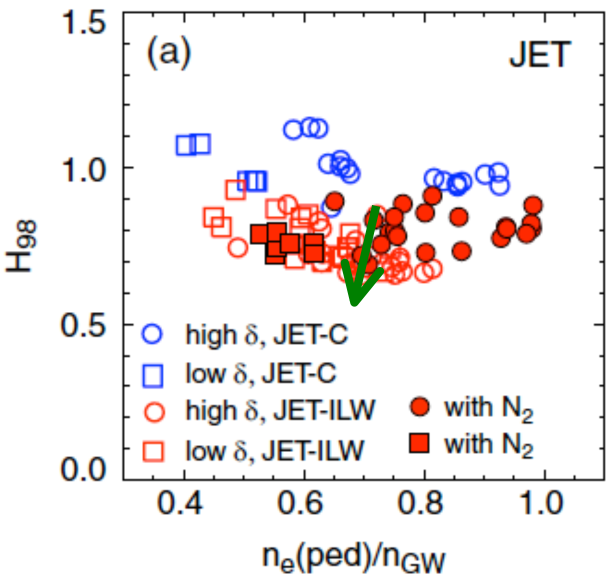

Figure 1: Change in edge operating conditions on JET with the ITER-like wall (red) from the carbon wall (blue). The solid red data correspond to N2 seeding, while the circles and squares denote low and high triangularity (Beurskens, PPCF 2013).

III. RECENT POWER EXHAUST PROJECTIONS

Heat flux exhaust for future devices is now projected to be even more challenging for future devices than was understood just 5 years ago. This applies to both steady and transient heat loads. Previous ITPA-sponsored studies of the heat flux scrape-off layer (SOL) width, λ_q, showed a dependence on major radius, i.e. that the heat flux footprint would broaden with machine size[16]. However more recent dedicated studies in low recycling, attached divertors have shown no such dependence, with footprints widths comparable between the smallest device in the study, Alcator C-Mod, and largest, JET[7], and depending only the midplane poloidal field, e.g. Figure 2. These multi-machine database results agree quantitatively with a neoclassical scaling of the heat flux width[6]. It is thought that these scalings correspond to the inherent upstream SOL transport physics, and that dissipative processes and divertor flux expansion would broaden the footprint near and below the X-point. These studies project to a heat flux width of ~ 1 mm for ITER, which is about a factor of 4-5 below previous design assumptions of the SOL width in ITER. ITER has examined the impact of narrow SOL widths on the power deposition profile[17]; sufficient divertor heat flux dissipation could be achieved with higher divertor neutral pressure, but the H-mode operating window would shrink to nearly a single point. Looking ahead to devices with higher power density than ITER, dissipation of the heat flux with solid PFCs appears feasible only with substantial core radiation[2], which would exacerbate the problem of sufficient power flow through the separatrix to remain in H-mode.

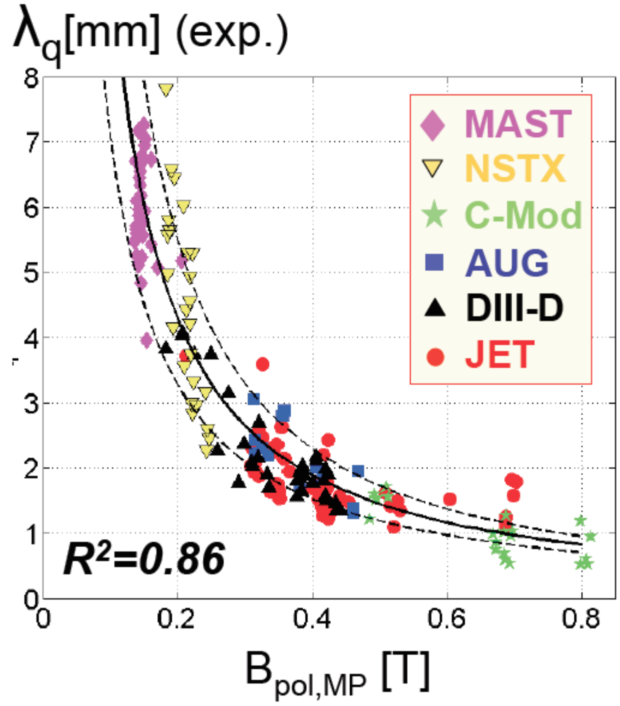

Figure 2:Dependence of scrape-off layer divertor heat flux profile width, mapped to the outer midplane via magnetic flux expansion, on midplane poloidal field (Eich, NF 2013).

Additionally transient heat flux exhaust appears to be more challenging than previously forecast. While magnetic perturbations[18-20] are the leading candidates for ELM suppressed regimes in present tokamaks and also for ITER, pellet ELM pace-making to increase ELM frequency, and hence reduce ELM size and peak divertor heat flux, is the main transient control strategy in the ELMy regime[21]. Peak heat flux reduction with increasing pellet induced ELM frequency, up to a factor of 12x over the natural frequency, was demonstrated in DIII-D[22]. However the peak heat flux was not reduced in JET with the ITER-like wall[23], despite a reduction in the ELM size and a 4-5x increase in the frequency. This occurred because of a narrowing of the heat flux footprint and differing ELM triggering dynamics in metal wall machines[24]. Previous estimates suggested that a 20x reduction in size and increase in frequency were needed for tolerable ELMs in ITER; more recent projections have increased the multiplier to about 1/45x at full plasma current[4]. Moreover these recent projections were made assuming a 4-5 mm upstream SOL heat flux width; if the heat flux width were ~ 1 mm as recently projected[7], with broadening in the divertor to 2-3 mm equivalent, then the ELM size would need to decrease by ~ 100x.

IV. Access to High Confinement, Toward an Attractive FNSF

As shown above, adequate power exhaust with solid PFCs for reactors will be very difficult or even impossible, given the recent projections showing an extremely narrow SOL power flow channel. Liquid metals can be used to exhaust much higher heat fluxes, both steady and transient. Experiments in which an electron beam at 50 MW/m^2 was used to heat up a lithium pool in the presence of a magnetic field showed convection forming in the pool, which distributed the heat nearly uniformly through the pool and limited the temperature rise to < 50 ^{o}C[25]. Furthermore a plasma gun device showed similar steady heat exhaust, as well as substantial transient heat exhaust, as would be experienced under ELM loads.

In addition to demonstrated high power exhaust capability, the use of Li as a plasma facing material has resulted in substantial confinement increases, both as 'coatings' on solid PFCs and also as liquid Li on top of PFCs[8, 9, 26-28]. Figure 3 shows that the confinement normalized to the ITER98y2 H-mode scaling law reached values up to 3-4 in LTX[29].

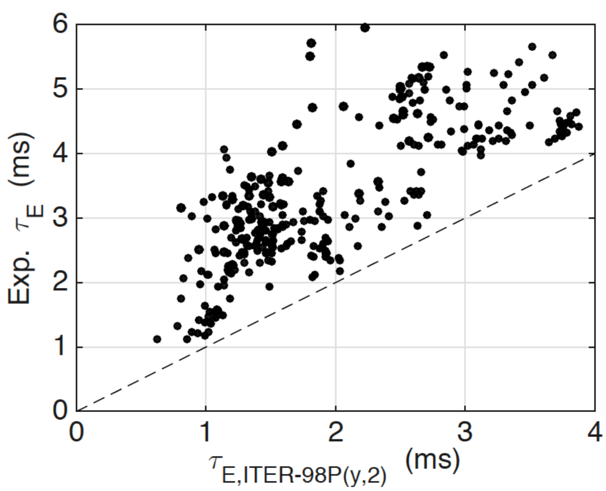

Figure 3: Comparison of τ_E with ITER98$_{y2}$ scaling, from LTX with solid and liquid lithium on the wall (Schmitt, PoP 2015).

Li was typically evaporated onto graphite PFCs between discharges on NSTX, although there were also dedicated experiments with the Li dropper with similar overall results. In NSTX a 50-100% increase in the H-factor, was observed, increasing with the amount of Li evaporation[27]. In addition ELMs were eliminated with sufficient Li dose[30], owing to the changes in the density and pressure profiles[31]. The density profile change was tied to the reduction of recycling near the separatrix at constant assumed cross-field transport[32]. The electron temperature gradient near the separatrix was unchanged, due to stronger drive for electron-temperature gradient modes, while the electron temperature a few cm inward increased substantially, due to reduced drive for micro-tearing modes[33]. The latter was conjectured to be responsible for the pressure pedestal increase, which has historically led to improved energy confinement times[34].

Li was introduced very recently into the DIII-D device via a simple gravitationally driven apparatus[35]. In discharges with a pre-existing edge instability, the injection of microscopic Li spheres resulted in long ELM-free phases with improved pedestal height and width (increased by 100%), and H-mode confinement (increased by 50-60%)[36]. Analysis showed a relaxation of the pressure gradients near the separatrix, leading to improved edge stability. It is noteworthy that no change in recycling was observed, i.e. it was not required for the observed performance improvements.

Li is used for coating the PFCs in the EAST device, both via evaporation in the morning before an experiment, and also injection with a Li dropper. The first access to H-mode in EAST was attributed in part to the Li conditioning[37]. Use of the Li dropper contributed to the record > 30 sec long pulse H-modes[38]. Recently it was reported that use of the Li dropper also eliminated ELMs in ICRF heated discharges[39]. Profile measurements needed for edge stability analysis were unavailable during these previous experiments; these should be available in upcoming experiments.

Figure 4 compares and contrasts the main results with Li conditioning between NSTX, DIII-D, and EAST in tabular form. While edge stability is improved in all devices, confinement is improved above normal H-mode scaling only in NSTX and DIII-D, while recycling is reduced only in NSTX and EAST. On the other hand, edge fluctuations increase in DIII-D and EAST, while they decrease in NSTX. Thus it is likely that both the operating scenario and the Li delivery method affect the global characteristics within particular devices.

	DIII-D	NSTX	EAST
Delivery method	Dropper	Inter-shot evaporation, (Dropper)	Dropper, (Morning evaporation)
Pedestal Width	**Increased**	**Increased**	**?**
Pedestal Height	**Increased**	**Increased**	**?**
H-factor	**Increased**	**Increased**	**Unchanged**
Edge fluctuations	**Increased**	**Decreased**	**Increased**
Radiated power	**Steady during EF**	**Ramp during EF**	**Steady during EF**
Effect on ELMs	**Delayed**	**Eliminated**	**Eliminated**
Recycling	**Unchanged**	**Reduced**	**Reduced**

Figure 4: Comparison of Li effects in DIII-D, NSTX, and EAST.

It is instructive to compare the profiles changes observed in NSTX and DIII-D during Li conditioning. Figure 5 compares the electron density, electron temperature, ion temperature, and total pressure profiles as a function of normalized poloidal flux, ψ_N, for a reference ELMy discharge without Li in black/blue, and an ELM-free discharge with Li in red[40]. These profiles changes are thought to be due to reduced fueling from reduced recycling, and changes to the edge micro-turbulence from the resulting density profile changes. Figure 6 shows a comparable set of profiles from DIII-D, with (red) and without (green) active Li injection[36]. In this case recycling and fueling do not change, but the edge profiles change in a similar way to NSTX. This opens up the possibility that there may be an additional, perhaps yet unidentified mechanism, that leads to the inward shift of the density and pressure profiles in both devices, in substantially different experimental scenarios. This speculation will have additional circumstantial evidence if the EAST profiles change in a similar way with Li injection, the data for which will be available in upcoming EAST experiments.

The enhanced confinement resulting from Li can be used as

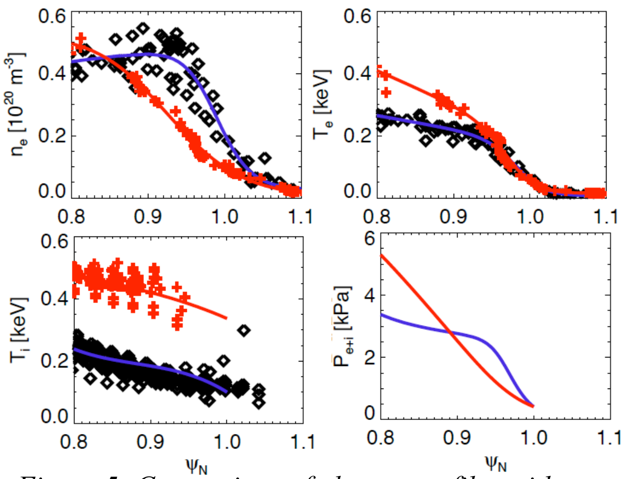

Figure 5: Comparison of plasma profiles with (red – ELM-free) and without (black - ELMy) Li conditioning in NSTX (Maingi, NF 2012).

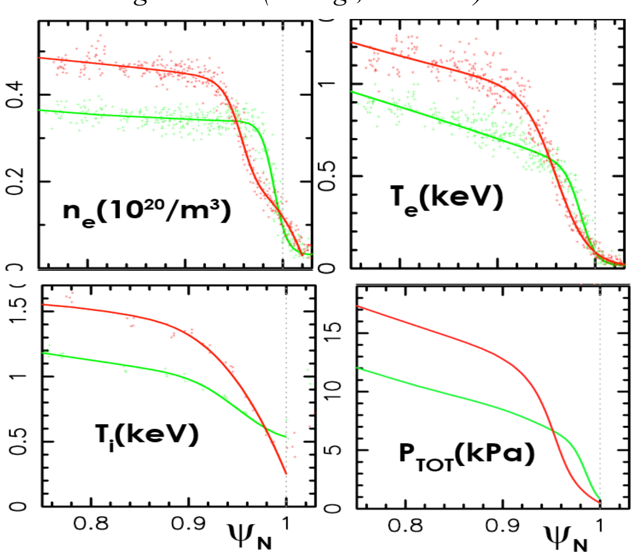

Figure 6: Comparison of plasma profiles with (red) and without (green) Li conditioning in DIII-D (Osborne, NF 2015).

the basis for a different, more compact design point for a Fusion Nuclear Science Facility. An example[41] of an ST-based FNSF design point that can make use of this enhanced confinement is shown in Figure 7. The computed bootstrap fraction is between 0.7-0.8 for H98y2 values between 1.5 and 2.0; this is sufficiently high to reduce external current drive, but with enough margin below unity bootstrap faction to allow control. The computed peak outer wall neutron flux exceeds 1.5 MW/m^2 over this same H98y2 range. Moreover, the scenario is computed to transition from transport-limited to stability limited over this range of H98y2, which enables investigation of the physics across this important transition point.

The most likely candidates for a comprehensive high

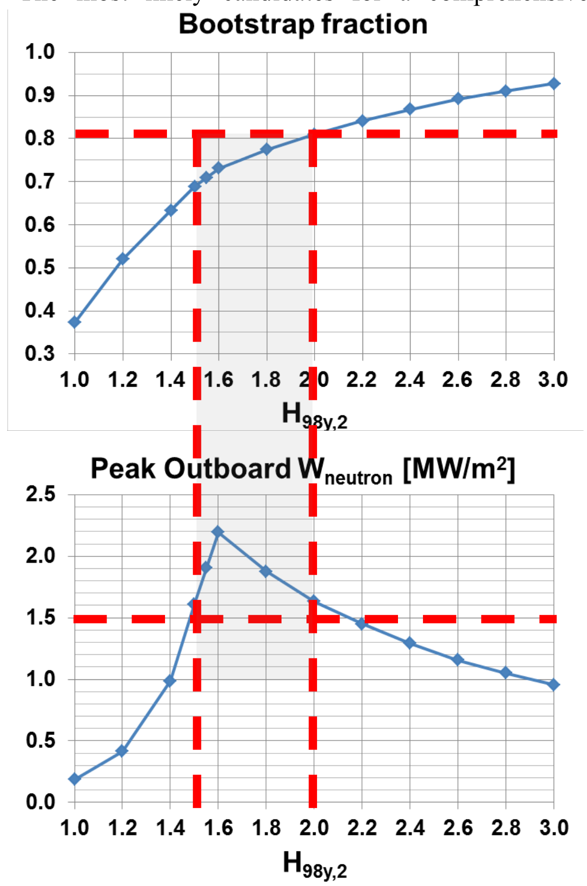

Figure 7: Operating points for an ST-based Fusion Nuclear Science Facility as a function of confinement multiplier, H98y2.

confinement scenario initiative are NSTX-U[42] and EAST[38, 43], which both deploy Li PFCs as an integral part of their scientific programs. These would complement the LTX program[44], which has a similar goal on a smaller scale device. On NSTX-U, this would be facilitated by acceleration of the baseline schedule, which is presently projected for liquid Li divertor deployment in ~ 2021. On EAST, US-based teams would contribute by providing designs for liquid Li PFCs, which would be built and deployed by EAST. The basis for such collaborative activities is already established: a flowing liquid Li system (FLiLi)[45] was found to be compatible with H-mode discharges on EAST in 2014, and a Liquid Metal Infused Trenches (LiMIT) system[46] will likely be installed in 2015, both in strong collaboration with US participants.

V. Conclusions

The use of Li to access high conifnement regimes would lead to compact next step device designs, with less auxiliary power needed to enter high confinement regimes. Such scenarios offer attractive design points for nuclear science facilities, and even reactors, as described conceptually in several papers[47, 48]. Extension of the the present set of studies to long pulse H-modes, likely in the EAST device, will give credence to the use of Li for future devices. Substantial R&D is needed in the areas of regulation of Li flow, controlling the Li chemistry and safety aspects, including tritium retention, and identifying and optimizing the acceptable operational temperature windows. A broad program to address these issues is needed for Li PFCs to be considered as viable candidates for future devices.

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