Physics of Plasmas

Lithium, a path to make fusion energy affordable

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AFFILIATIONS

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ABSTRACT

In this tutorial article, we review the technological, physics, and economic basis for a magnetic fusion device utilizing a flowing liquid lithium divertor (molten metal velocity in the range of cm/s) and operating in a low-recycling plasma regime. When extrapolated to magnetic fusion reactor scale, the observed effects of a liquid lithium boundary on recycling reduction, confinement increase, and anomalous heat transport mitigation may offer a fundamentally distinct and promising alternative route to fusion energy production. In addition, this lithium-driven low recycling regime could accelerate fusion's commercial viability since such a device would be smaller, dramatically decreasing plant and electricity costs if all technological complexities are solved. First, the theoretical basis of the energy confinement and fusion performance as well as the related possibilities of low recycling regimes driven by flowing lithium plasma-facing components are reviewed. Then the paper emphasizes the technological obstacles that need to be overcome for developing the necessary systems for such a flowing liquid lithium solution at reactor scale and details how many of these have been overcome at laboratory and/or proof-of-concept scale. Finally, the current and planned scientific and engineering endeavors being performed at the University of Illinois at Urbana-Champaign regarding this alternative reactor option are discussed.

I. INTRODUCTION

If humankind wishes to reduce its dependence on fossil fuels in order to lessen the effects on global warming, climate change, and biosphere degradation, the only alternative capable of high energy density, 24/7 availability is nuclear power. For fusion energy to be part of the nuclear solution, it would need to follow the same trajectory as fission. This means that to merely provide 1% of the world energy demand, fusion would need to follow an exponential growth phase in which 10 fusion power plants should be built by 2060 and a hundred by 2070.[1] For this to happen, fusion reactors need to be cheaper and smaller than the ITER-like devices which are envisioned for DEMOnstration (DEMO) reactors. There must be a fusion energy device that is less complex, easier, and faster to construct.[2] Such an approach would lower the investment risk and construction period of the fusion reactor. Furthermore, it would also increase the innovation cycle as the technological and scientific knowledge, derived from the experimentation/operation of such prototypes, would be acquired faster.

Today, the conventional pathway to magnetic fusion derives from the 1990s state of the art and technology limitations. Excepting the replacement of carbon plasma-facing components (PFCs) (divertor plates with tungsten and first wall with beryllium), from the engineering and operational point of view the machine being built is very close to its first conceptual design.[3,4] Consequently, the current ITER-like DEMO power plant scenarios consider a minimum size with major radius (R) in the range of 6–9 m.[5] This solution supposes building larger and more complex reactors than ITER. Considering the enormous cost ($\sim 20 \times 10^9$ dollars) and the prolonged construction time required for the device being constructed at Cadarache (France), a larger future prototype may face even more stringent economic and practical impediments.

Electric power plants have a key metric—the cost of electricity (COE). For nuclear systems, a huge fraction of the COE is the capital cost (investment) for construction. Generally that capital cost is approximately proportional to machine volume ($\sim R^3$).[6] Therefore, the path to reduce the COE must be to reduce size while maintaining the same electrical output. Smaller size, high-fusion-power-density reactors will need better plasma performance and energy confinement.[7] In this article, we consider the effects of a possible low-recycling regime driven by a flowing lithium divertor configuration which might solve these issues and provide a path to lower COE for fusion. This would

produce burning plasma conditions at high gain factor (Q) in smaller size reactors and also utilize a plasma-facing component (PFC) solution with less replacement and maintenance requirements which are essential for a higher power plant availability. While the theoretical basis of energy confinement and fusion performance of a low-recycling regime driven by flowing lithium PFCs is reviewed, the bulk of the paper emphasizes the technological challenges that have been or need to be overcome to develop the necessary systems for such a flowing liquid lithium (FLiLi) PFC configuration.

The article is structured as follows. Section II reviews the problem of energy confinement in magnetic fusion devices that determines the minimum size required for a reactor and how lithium-driven low-recycling regime may move the paradigm to a lower size reactor. Section III explains the power exhaust problem and the positive effects of a lithium PFC solution that may relax the heat handling requirements of the reactor. Section IV analyzes the role of the proposed reactor configuration in the reduction of eventual electricity costs. Section V explores the required technologies to accomplish a low recycling scenario driven by a flowing liquid lithium PFC configuration, showing the works performed and planned at UIUC to solve the associated issues and provide an affordable reactor solution. Finally, Sec. VI outlines the main conclusions related to this innovative pathway to commercial fusion and makes a case for utilizing JET in an attempt to surpass the breakeven conditions and if the low recycling theory astounding predictions are correct perhaps achieve Q factor performance similar to that planned for ITER, in the next years.

II. PHYSICS CONSIDERATIONS OF THE FUSION PROBLEM

A. A question of energy confinement and transport

In the 1950s obtaining energy from nuclear fusion reactions on Earth in a controlled way was viewed as an extremely hard challenge that needed to overcome many formidable obstacles both from the physical basis and the required technology.[8,9] The conditions for practical energy generation from thermonuclear reactions[10] are frequently summarized in the fusion triple product that for the case of D-T plasmas is expressed in terms of temperature (T), density (n), and energy confinement time (τ_E) of the D-T ionic species,

$$n \cdot T \cdot \tau_E > 2 \times 10^{21} (\mathrm{keV\,s\,m^{-3}}). \tag{1}$$

For magnetic fusion, this implies values in the range of $n \sim 10^{20}\ \mathrm{m^{-3}}$, $T \sim 10$–$20\,\mathrm{keV}$, and τ_E around few seconds. The energy confinement time is the key parameter that represents a direct measure of the effectiveness of the plasma to be heated and isolated from energy dissipation, being defined as[11]

$$\tau_E = \frac{W}{P_h - \frac{dW}{dt}}, \tag{2}$$

where W is the kinetic energy stored in the plasma and P_h is the auxiliary heating power injected into the plasma. Its direct extrapolation from present devices to future projected reactors is based on empirical scaling laws. The parameter sensitively depends on the nature of the energy transport processes in the plasma that are intercoupled and related to intrinsic instabilities that underlay in the plasma properties, its geometrical characteristics as well as the interaction with the magnetic and electric fields. Additionally, such plasma stability and the related energetic fluxes are associated with the nonperfect confinement of the plasma and the unavoidable interaction between it and the containing device.[12]

In plasmas with noncircular cross section, the Lawson triple product (L_{TP}) is found to scale as[5,13]

$$L_{TP} = nT\tau_E \propto \frac{\varepsilon^3 H^2 \cdot \kappa^{7/2} \cdot B^3 \cdot R^2}{q^3}, \tag{3}$$

where B is the magnetic field, R the major radius, q the safety factor, ε the inverse aspect ratio (r/R, defined as the quotient between the minor and major radius of the toroidal device), and κ the elongation of the plasma loop. The H parameter is the so-called confinement factor that measures the equivalence with the projected ITER based energy confinement empirical scaling (τ_E^{ELMy}).[14] It is important to note that Eq. (3) shows an equivalent scaling of L_{TP} with H and size (R). Conditions with $H > 1$ would represent improved energy confinement with respect to the empirical as $\tau_E = H \cdot \tau_E^{ELMy}$. Therefore, one could try to achieve similar triple product in smaller devices if the H factor may be increased in the same proportion in which the reactor size was reduced. Increase plasma confinement has been pursued since the beginning of research in nuclear fusion. However, the achievement of confinement values well beyond the ITER scaling ($H > 1.5$) is not an easy and/or straightforward question. As we will see later, the use of a flowing lithium divertor scenario might enable the opportunity to achieve this high confinement route to magnetic fusion.

For practical purposes, a Q_{fus} gain factor ($Q_{fus} = P_{fus}/P_h$), which needs to be higher than unity ($Q = 1 \equiv$ breakeven) to extract net power, is also defined, easily indicating the thermonuclear efficiency of the reactor. In steady state, considering that in the D-T reaction approximately 20% of the energy is taken by the alpha particles, there is a direct (and mathematically consistent and increasing) relation between Qf_{us} and L_{TP}, B, T, and τ_E,[5,13]

$$Q_{fus} = \frac{5A \cdot L_{TP}}{5 - A \cdot L_{TP}}, \tag{4}$$

where $A > 0$ is a constant.

In a magnetic fusion reactor, the plasma needs to be heated and kept hot, isolating it from the reactor walls by using strong toroidal and poloidal magnetic fields, trying to maximize the triple product, and improving the energy confinement time and the plasma temperature (both parameters being globally coupled) until surpassing the breakeven conditions. Accomplishing such a milestone is an exceptionally difficult task as was early inferred by the first experimental approaches that demonstrated an energy transport level greatly exceeding expectations.[15–17]

First considerations assumed that the transport coefficients such as thermal diffusivity (χ) in a plasma could be directly derived from the results obtained by Chapman and Enskog for a diluted gas.[18,19] Therefore, for plasma particles under a magnetic field B, with a density n and temperature T, presenting a collision frequency ν which scales as $\nu \sim n/(T^{3/2} \cdot m^{1/2})$[20] and Larmor gyroradius $\rho \sim T^{1/2} \cdot m^{1/2}/(e \cdot B)$, the rate of such classical heat diffusion is given by[21]

$$\chi_c \propto \nu \cdot \rho^2 \sim \frac{n \cdot m^{1/2}}{T^{1/2} \cdot e^2 \cdot B^2}. \tag{5}$$

Usually, an associated inversely proportional energy confinement time (τ_E) to the heat transport coefficient is considered; thus in the classical

transport approximation, the energy confinement time is found to scale as

$$\tau_E \sim \frac{B^2 \cdot T^{1/2} \cdot e^2}{n \cdot m^{1/2}}. \tag{6}$$

Unfortunately, the real experimental values of confinement time were remarkably lower compared to those predicted by this classical model. Earlier the enhanced transport in a plasma had been observed by Bohm[22] when studying magnetic arcs for isotope separation. To explain the anomalous results, he proposed a less favorable scaling for the heat diffusion coefficient (χ_B) in the plasma where particle gyrofrequency $\Omega = e{\cdot}B/(m{\cdot}c)$ substituted collision frequency in Eq. (5), resulting in an energy confinement scaling as

$$\tau_E \propto \chi_B^{-1} \propto \left(\Omega \cdot \rho^2\right)^{-1} \sim \frac{B}{T}. \tag{7}$$

Such scaling is considered as a lower limit case for the confinement time values and the associated transport was defined by Taylor as "the maximum value which the transverse diffusion can ever attain."[23] Bohm diffusion would imply the necessity of considerably more intense magnetic fields and much larger devices to achieve fusion conditions.

The results obtained in the 1950s were more in agreement with the Bohm conjecture, but in the 1960s, experiments in the tokamak T3 in the USSR showed an increase in plasma confinement, resulting in the achievement of high plasma performance and keV range temperatures never registered before.[24] Fortunately, in the last decades, tokamak plasmas have been found to also follow generally lower diffusion rates that have originated the gyro-Bohm scaling[25,26] for thermal diffusivity (χ_{GB}) and derived energy confinement time. It is the considered for ITER and future reactor performance extrapolations being based on the ratio (ρ/r),

$$\chi_{GB} = \frac{\rho}{r} \cdot \chi_B, \tag{8}$$

$$\tau_E \sim \frac{1}{\chi_{GB}} \sim \frac{r}{\rho} \cdot \frac{B}{T}, \tag{9}$$

where ρ is the ion gyroradius and r is the minor radius of the device. Such prediction gives increased values for the energy confinement time in a factor ($r/\rho \gg 1$) compared to Bohm scaling. Consequently, larger reactors would be associated with higher confinement times. Although not as harmful as Bohm-like transport, this scaling prediction is still anomalously larger when compared to classical expectations. In toroidal devices, these observed transport levels are assumed to be the consequence of considerably more complex processes than simple, original phenomenological explanations of Bohm diffusion.

First, toroidal geometry and curvature aspects entail a magnetic field gradient and effects in the particle motion producing gyrocenter shift (neoclassical effects[26–28]) as well as trapping and bouncing of particles in banana orbit drifts[27] during their movement along field lines. Such effects provide additional contributions (enhancement) to transport coefficients. In the same way that classical transport supposes an ideal, minimum rate of heat loss in a confined plasma, neoclassical theory specifies a lower limit for the transport rate in a device where magnetic confinement is approached by means of toroidal geometry. Neoclassical transport is divided into different regimes (Pfirsch–Schluter, plateau, and banana) depending on collisionality [ratio between collision frequency (ν) and the frequency of the particles transiting around the torus, ω_T].

More importantly, the formation of coherent, macroscopic turbulent structures within the confined plasma is linked to large-scale flows that increase transport and energy losses (by a factor even beyond an order of magnitude when compared to neoclassical predictions). The creation of the turbulent structures is strongly linked to the transport processes driven by gradients.[29] They are considered as the free energy sources that trigger the nonlinear growth in the amplitude of the microinstabilities. Among the many instabilities linked to different gradients, the ion-temperature gradient (ITG) mode is considered as the main candidate for explaining the resultant, global turbulent transport, and therefore the limiting factor to the energy confinement time and the temperature profiles in the most advanced tokamaks and stellarators.[30,31] ITG mode is also expected to dominate the confinement time in ITER.[32] Eventually, turbulent transport affects the distribution and gradients within the plasma, thus creating a feedback scheme that couples the turbulence origin term and the subsequent effects[33] as shown in Fig. 1.

Consequently, the correct modification (inhibition) of gradients may be seen as a basic action to mitigate/inhibit turbulence. Reducing turbulence in fusion plasmas is a must and has been a major area of investigation within controlled fusion research. As a result, zonal flows (ZF),[34] E × B shear flow and/or plasma rotation[35,36] have been proposed and used to reduce/ameliorate turbulence and increase plasma stabilization. However, none of these actions rely on the inhibition/reduction of the turbulence source, i.e., the temperature gradients responsible for this turbulent transport in the first place. In Secs. II B–II E, we will detail the effects of a lithium boundary in hydrogen recycling and concomitant edge temperature gradient reduction.

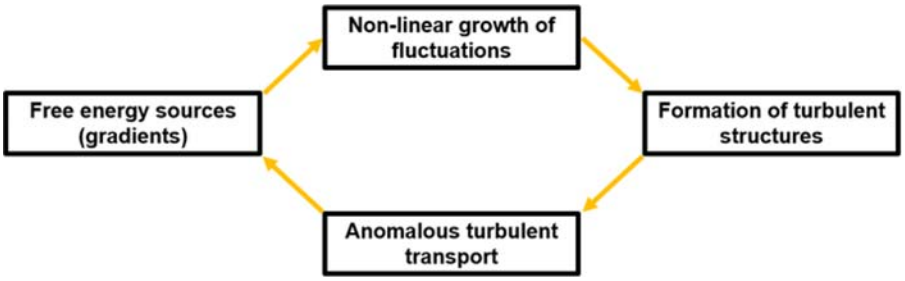

FIG. 1. Feedback scheme for the turbulent transport processes driven by gradients in magnetically confined toroidal plasmas.

B. Influence of plasma material interactions on energy transport

The physics of a confined plasma, its transport processes and ultimately the energy confinement are strongly influenced by the plasma-material boundary. In the larger machines that are the basis for ITER and DEMO, the temperature gradients are especially important in the plasma-edge region. In such devices, the more promising plasma scenario for a reactor is based on the so-called H mode confinement regime.[37] It is characterized by the creation of a narrow pedestal in the radial direction with high temperature that abruptly decreases radially toward the plasma boundary, thus presenting a very pronounced temperature gradient.[38] At the same time, the density gradient is damped in such region, normally presenting a much weaker density gradient or even a flattened profile.

Both edge temperature and density gradients are affected by the plasma material interactions with the surrounding materials. First due to the "recycling" influx of cold hydrogen atoms (thermalized to wall temperature) returning to the plasma. This source element immediately cools the plasma edge down and at the same time increases its density. The recycling coefficient is defined as the ratio between the returning hydrogen particle flux coming from the plasma-facing surface over the total hydrogenic flux from the plasma that impacts such surface,

$$R = \frac{\textit{Hydrogenic influx back to plasma from surface}}{\textit{Hydrogenic flux to surface from plasma}}. \tag{10}$$

Such a parameter depends on the nature and chemistry of the chosen plasma-facing material. In ITER-like scenarios based on a high Z refractory metal divertor (tungsten), this coefficient is generally quite high (0.95 or even close to unity when saturation of tungsten with hydrogenic atoms takes place in the range of few nm which is the penetration depth of corresponding ions). However, lithium behavior is totally opposite as it is capable of retaining large hydrogenic content that escapes from the confined plasma, lowering the recycling.

Second, the plasma material interaction, originated by electron, ion, impurities, and neutron fluxes impinging the plasma facing components (PFCs) produces ejection of atoms, ions, and compounds from the wall interfaces by different physical and chemical mechanisms (sputtering, evaporation, sublimation, codeposition, etc.). Such impurities reach the plasma edge and may penetrate in the confined plasma, thus contaminating it. Obviously, they also contribute to the cooling of the plasma as they will extract energy during their successive ionizations and may be transported within the confined volume. The maximum fraction of impurities that plasma may contain without collapsing strongly depends on the atomic number of the impurity (Z), being orders of magnitude larger for low Z elements (lithium, beryllium, carbon, etc.) compared to high Z elements such as tungsten.[39,40]

C. Effects of neutral recycling reduction by lithium in plasma confinement and performance

Lithium absorbs incident hydrogen at a very high rate. Therefore, a fresh lithium surface greatly reduces recycling. The impressive effects of lithium in the plasma boundary of a fusion device were first observed in the TFTR tokamak, operating in limiter mode without divertor (circular plasma cross section). It was found that pulses preceded by lithium pellet injections showed a notable increase in the energy confinement time of the discharge[41] in a keV temperature plasma edge.[42] Although several strategies were implemented in the machine trying to improve wall conditioning and thus lower recycling and increase confinement,[43] the improvements associated with lithium injection were the most successful. The resultant plasma confinement produced a combination of higher central densities and lower Hα signals in the plasma edge.

TFTR D-T operation also demonstrated that D-T fusion reaction yield could be aided by reducing the hydrogen recycling flux from surrounding walls and/or by improving energy confinement with the discovered lithium wall conditioning.[44] As demonstrated later, both actions were, indeed, intimately related.[45] Furthermore, the highest values of confinement time ever registered in the machine (factor of two larger than normal discharges with deuterium) were observed when lithium was injected in the plasma edge in the liquid state, also multiplying the triple product by a factor of 4[46] in discharges with peak density and plasma current in the core as well as higher and wider temperature profiles over the plasma cross section.

Confinement improvements have been also seen in different relevant machines when the global content of neutrals in the plasma edge was reduced and/or controlled regardless of the method utilized to do it, for example, using efficient divertor pumping.[47] In JET tokamak operating with carbon walls, the higher[48] uptake of hydrogen by the graphite surfaces was associated with lower recycling and better plasma performance when compared to the ITER-like wall (ILW) scenario[49] where the presence of tungsten negatively affected confinement.[50] Nitrogen seeding was introduced to induce detachment and limit the tungsten influx acting as a palliative for the confinement problems. Nevertheless, restoration of confinement ($H \approx 1$) was also possible in JET-ILW using other strategies directly related to hydrogen recycling reduction by means of more efficient pumping. For example, when divertor strike point was moved to a region where neutral gas pumping was favored.[51] Additionally, improved H factor (both in core and pedestal) was observed at lower averaged collisionality and peaked core density in conditions where the role of neutral content in the scrape-off layer (SOL) was pointed out.[52] In LHD stellarator, helium wall conditioning[53] and divertor cryopumping[54] reduced recycling and helped to control the edge density, increasing plasma temperature, and global confinement. Higher recycling conditions were correlated with poorer confinement in JT60 tokamak[55] and in ASDEX Upgrade, the presence of a higher density region in the plasma boundary was linked to a contribution from the neutral influx and shown to degrade the pedestal structure, pressure, and confinement.[56]

Hence, there is wide and clear scientific evidence from the most relevant worldwide machines that decreased hydrogen recycling and neutral content in the edge/SOL improves energy confinement and plasma performance. In the case of lithium, the triggering of these effects appeared to be caused by its chemical affinity by hydrogen isotopes. This fact leads to the massive and efficient trapping of hydrogenic ions escaping from confined plasma and the suppression of the subsequent returning influx of cold neutrals to the plasma edge. Consequently, plasma edge cooling is directly avoided and an automatic increase in the edge temperature may be logically expected with outstanding implications in the physics of the confined plasma and its

energy and particle transport as it is directly linked to a decrease in the temperature gradients.

D. Lithium-based reactors as possible low recycling, high-confinement route to more compact fusion

The effect of hydrogen recycling reduction driven by a lithium absorptive boundary in reactor performance was first analyzed theoretically by Krasheninnikov *et al.*[57] They considered the absorption of the plasma flux in the edge by lithium walls in a zero-recycling regime. Then, they analyzed the subsequent global heat transport from the plasma core to the edge. The work showed that in this ideal approximation, the main physics of the plasma transport processes is fundamentally distinct. As recycling is reduced, the particle influx in the edge decreases such that there are no cold fuel particles that create the temperature gradient. The subsequent low recycling plasma regime was characterized by striking predictions. These assertions included the presence of a hot, low collisionality plasma edge with strong suppression of the temperature gradient from the core and the notable expansion of the high-temperature plasma volume able to produce fusion power. At the same time, the inhibition of the strong gradients in electron and ion temperatures produce a much more stable plasma where turbulence (crucial ITG instability) would be strongly suppressed. Finally, the presence of a hot and homogeneous plasma edge would be also linked to the presence of a more even, finite current density profile. This translates into a more stable plasma with a larger beta, a parameter directly proportional to the triple product (and thus to be maximized in any fusion reactor). Beta (β) is defined as the ratio between plasma pressure and magnetic pressure, scaling as

$$\beta \sim \frac{\mathrm{n} \cdot \mathrm{T}}{\mathrm{B}^2}. \tag{11}$$

Consequently, in such an operational regime, the fusion performance in ITER-like scenarios would be considerably enhanced.

All these outstanding benefits may be possible due to the unique atomic and chemical characteristics of lithium. It is the lowest atomic number ($Z = 3$) element capable of being used as a divertor plasma-facing material, thereby minimizing the effective charge in the plasma (Z_{eff}). Its strong affinity for hydrogen isotopes, where lithium acts as electron donor, leads to the formation of Li-H chemical hydride ionic bonds. Additionally, it possesses a low melting point (180 °C), excellent heat handling capabilities, and a very low first ionization energy (5.39 eV). The total trapping of hydrogen isotopes in liquid lithium has been demonstrated in ion beam and reactor relevant flux linear plasma device experiments up to a temperature of 400 °C in the liquid lithium that was progressively converted into hydride.[58,59] Concerning the nature of this hydrogenic retention on liquid lithium, it is interesting to note that in lithium conditioned carbon walls covered by thin films, the presence of oxygen, associated with the lithium atoms, plays a major role within the retention process.[60] However, the scenario proposed here (a flowing lithium divertor with liquid thickness in the range of mm supported in a pure metallic substrate and with no presence of graphite) is quite different when compared to the research with mixed carbon-lithium walls. In any case, the unavoidable presence of impurities on the liquid lithium flowing within the divertor may actually play a role in the hydrogenic uptake, but if the lithium is renovated (thus reducing the presence of impurities and passivation layers) and carbon is not present at all, it seems logical to think that the dominant mechanism may be more in agreement with the hydride formation results found by Baldwin and Doerner.[59]

On the other hand, the diffusion of hydrogen isotopes in liquid lithium[61] and the renovation of the liquid surface induced in a flowing liquid PFC make the low recycling regime compatible with a continuous operation scenario. The 450 °C upper-temperature limit appears mandatory to reduce the lithium evaporative flux that may contribute to the accumulation of lithium in the core and concomitant dilution in the confined plasma.[62] It should be noted that if such dilution in the core exceeds a limit, the fusion power will strongly decrease, thus precluding any possible benefit of the lithium usage in terms of fusion performance. In this respect, any possible measurement of the lithium concentration in the core would be very useful to continuously monitor such key parameter, although it is necessary to keep in mind that this real-time diagnosis will not be trivial. Consequently, the most conservative and direct action to control the lithium influx will be to keep its temperature below 450 °C. Additionally, if such temperature limits are surpassed, the massive lithium vaporization (exponential with increasing temperature) will cool the plasma edge down by means of vapor shielding and/or radiation mechanisms. This will create a temperature gradient in that region. In this strongly evaporative scenario, the lithium boundary would act in the opposite manner when compared to the goal of reducing recycling and edge temperature gradients. Therefore, for a low recycling operation driven by hydrogenic absorption on the liquid PFC, the temperature of the lithium boundary must be maintained below the indicated massive evaporation threshold.

With respect to this expected flux of lithium going into the plasma by means of evaporation and sputtering, the low ionization potential of lithium is responsible for its high sputtered ion fraction (60%) experimentally determined under particle bombardment (Fig. 2[63]). In the divertor boundary, these secondary sputtered ions will interact with the plasma edge structure and then may be screened and accelerated back to the PFC surface due to the sheath potential, thus being promptly redeposited and consequently not contributing to the net erosion rate. Furthermore, even the fraction that is evaporated/sputtered in the form of neutral lithium atoms may be re-deposited in a hot plasma edge due to the very low 1st ionization energy of lithium and the high-temperature plasma structure that will interact with such neutral impurities. The usual criterion for prompt redeposition, based on general impurity transport considerations, establishes that the phenomenon will take place when the ionization length in the plasma boundary of the neutral species is smaller than its Larmor radius ($\lambda_{Li}^{+} < \rho_{Li}^{+}$). It may be expressed as

$$\frac{v_{Li}}{\langle \sigma v \rangle \cdot n_e} < \frac{102\sqrt{\mu \cdot T_i}}{Z \cdot B}, \tag{12}$$

where v_{Li} is the velocity of the lithium atoms depleted from the surface (it may be just thermal if the atom is evaporated or on the order of the binding energy if it is sputtered), $\langle \sigma v \rangle$ is the average rate coefficient for the 1st ionization of lithium by electron impact (a parameter that strongly increases with temperature), μ is the reduced mass of the lithium atom respect to the proton, Z the atomic number, and B the magnetic field (in Gauss units) on the plasma boundary. Hence, the values for both parameters basically depend on the characteristics of the plasma edge (n_e, T_i). In the practice, in a low temperature, detached

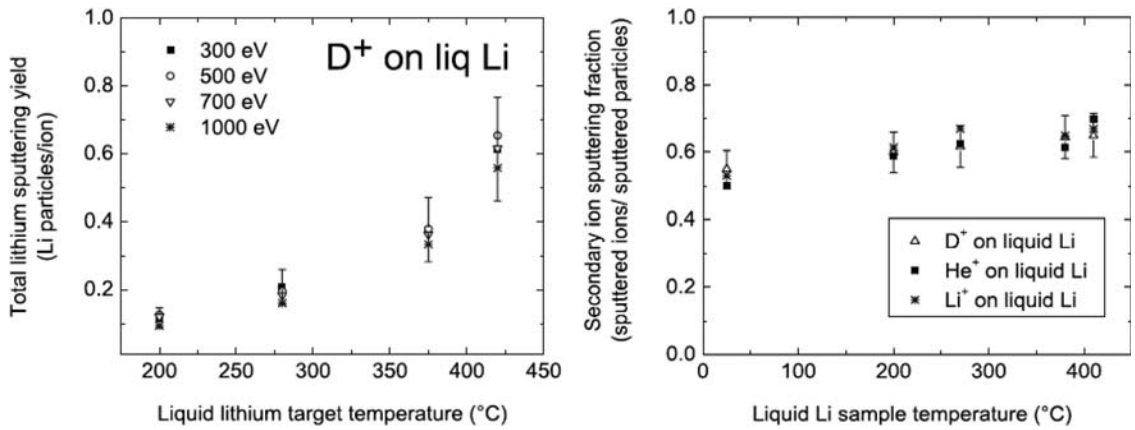

FIG. 2. Results of liquid lithium sputtering obtained by Allain and Ruzic. [Graphs adapted with permission from J. P. Allain and D. N. Ruzic Phys. Rev. B **76**, 205434 (2007). Copyright 2007 American Physical Society.] On left, it is shown the clear enhancement in sputtering with temperature for the case of D^+ bombardment, regardless the energy of the projectiles. However, the total yields tend to clearly saturate beyond 500 eV incident energy with yields well below 1. On the right, the secondary ion sputtered fraction is represented, showing that around a 60% of the sputtering is in the form of Li^+ ions in the range of 200–400 °C for D, He and Li projectiles. As explained in the main text, this particularity in conjunction with the low first ionization energy of lithium has direct consequences in the penetration of such eroded impurities within the confined plasma. At divertor relevant scale, the scenario will be dominated by a very high prompt redeposition fraction for the evaporated/sputtered lithium species, the fact that will be essential to minimize the detrimental effects in the fusion performance potentially caused by its accumulation in the core and the related D-T dilution.

plasma edge that characterizes high Z metal divertors, the criterion for prompt redeposition may be frequently not accomplished and impurities can penetrate into the plasma, affect the core as well as being transported within the plasma and/or migrate to another in-vessel surface. However, in configurations with a hotter edge that would result from the low recycling operation with lithium, a much higher temperature clearly favors the prompt redeposition criterion previously exposed.

At reactor-relevant divertor scale, a very high prompt redeposition fraction ($\geq$95%) of the sputtered lithium has been predicted by the modeling works carried out by Brooks *et al.*[64] Therefore, the problems related to net lithium erosion and concomitant core dilution would be strongly reduced. Globally, this high fraction of lithium prompt redeposition is considered to explain the extremely low contamination in the plasma core by lithium observed in every relevant fusion experiment conducted so far. At this respect, it must be remarked that even in the case of the previously mentioned high-performance discharges in TFTR with very hot (keV) plasma edge, the contribution of lithium impurities to the (eventually very low) Z_{eff} value was found to be considerably small when compared to other impurities as carbon or oxygen that penetrated further in the plasma.

Reducing recycling to zero means that all the effects of edge neutrals in temperature gradients vanish. In an ideal approach, if such gradients are totally suppressed, the plasma would behave under isothermal conditions.[65] The conditions for this theoretical case are much closer to equilibrium when compared to conventional tokamak operation since it would have fewer sources of free energy (gradients). The plasma temperature would be higher in the edge, showing a flattened profile and thus diminishing the effects of collisions in transport. Other interesting characteristics are the presence of an exponentially decaying radial density profile (decreasing with poloidal flux), peaking in the center of the plasma but with a very low density close to the separatrix. Such plasma edge structure is an absolutely opposite case compared to the conventional, high Z divertor ITER/DEMO approach and its envisioned high recycling regime where there is a strong temperature gradient close to the separatrix (beyond the pedestal) and the edge density is high due to the effect of the high recycling as an edge particle source.

In fact, in high recycling machines operating in H-mode, the presence of the plasma pedestal has been claimed to be driven by the spontaneous generation of an edge transport barrier (ETB)[66] that considerably suppresses turbulence in the edge causing higher temperature and energy confinement time of the plasma. Furthermore, different relevant machines have also shown the creation of an internal transport barrier,[67] placed closer to the plasma core in locations where the q factor had an integer value, resulting in a plasma with strong poloidal rotation[68] and regions with reduced temperature gradients[69] that significantly improved plasma performance. In analogy, low recycling regimes would be essentially an extreme case of these H-mode conditions, where suppression of external plasma cooling in the edge would allow the extension of the energy content of the "pedestal" wider with higher temperature across a larger plasma volume.

In a lithium low recycling regime,[70,71] the plasma energy losses would be dominated by particle diffusion rather than conductive thermal diffusivity. Combined with full neutral beam injection (NBI) core fueling (no external gas puffing), the plasma would have only a source of hot particles in the core and one efficient particle sink in the edge, i.e., the absorbing flowing lithium elements. As formulated in Ref. 72, considering that particle flux involved from the core to the edge would depend on the NBI input,

$$\Gamma^{\text{core-edge}} = \Gamma^{\text{NBI}} = P^{\text{NBI}}/\left(e \cdot E^{\text{NBI}}\right), \tag{13}$$

being P^{NBI} and E^{NBI} the power and energy of the NBI fueling system, e the elemental electron charge, and a corresponding flux for from the edge to the wall ($\Gamma^{\text{edge-wall}}$) affected by recycling (R),

$$\Gamma^{\text{edge-wall}} = \Gamma^{\text{core-edge}}/(1 - R), \tag{14}$$

the power evacuated by plasma particles, considering a plasma with averaged temperature (T_i, T_e) would be

$$5/2 \cdot (T^e + T^i) \cdot \Gamma^{edge-wall} = P^{NBI} + P^{\alpha} - P^{rad}, \quad (15)$$

with P^{rad} and P^{α} the power terms given by radiation losses and alpha particle generation. Then, we can obtain the basic low recycling regime relation between plasma temperature, NBI heating, and recycling proposed by Zhakarov,[71]

$$1/2 \cdot (T^e + T^i) = 1/5 \cdot E^{NBI} \cdot (1 - R) \cdot \left(1 + (P^{\alpha} - P^{rad})\right)/P^{NBI}. \quad (16)$$

Equation (16) shows that if the particle source in the edge is minimized, then the average temperature value of ions and electrons, extended widely along the edge due to the reduced gradient, increases to hot values in the range of keV with lower recycling. Intuitively, this expression may be also seen as an immediate consequence of the high diminution of the cooling term in the edge caused by suppression of the cold recycled neutrals, thus creating a plasma that is effectively isolated from the wall boundary. The influx of neutrals may be also considered as a source of the so-called plasma-beam instability.[73] Again, it is important to remember that the necessity of isolating the plasma for its efficient confinement and heating was a major requirement since the very early magnetic fusion ideas,[8,9,15] being a clear objective of the plasma surface interaction (PSI) discipline. Because of the isolation from recycled neutrals, the edge temperature would be decoupled from the heat transport characteristics of the core, and the heat flux from the edge to the wall would be dominated by particle diffusion of ions rather than thermal conductivity. This is a substantial difference when compared to high recycling regimes, where heat conduction of ions is frequently affected by ITG modes and concomitant turbulence, showing associated heat losses that are greatly enhanced. Such heat losses originated by the nature of the plasma edge have motivated the use of more intense heating power trying to increase the temperature of the plasma and improve the performance of the device. However, this "brute strength" strategy alone does not solve the original question as further heating of the plasma core accompanied by strong recycling in the edge may even worsen the temperature gradients and hence the turbulent transport and heat dissipation problem regardless the level of external power injected into the plasma.

Moreover, considering a hot, low recycling plasma with low collisionality (banana regime), in a tokamak geometry, and approximating the perpendicular velocity of the ions as the ratio between Larmor radius and the ion–ion collision time, Zhakarov gives an expression[74] for the heat flux attributed to ion particle diffusion by integrating over the toroidal volume, yielding a result that is independent of the high edge temperature. This heat flux would be directly proportional to the square of the edge density, whose profile would decrease exponentially with poloidal flux.[65] Following this formulation, the heat flux to the walls would depend on the edge density but not on the (high) edge temperature resulting from the low recycling regime. It means that a high edge temperature does not proportionally increase the power density to the divertor. Therefore, if the edge density can be externally adjusted (for example, by means of resonance magnetic perturbations as suggested in Ref. 74) up to a sufficiently low and controlled value, the heat flux reaching the divertor PFCs may be made compatible with the power density limits of the lithium boundary and consequently consistent with the previously comments related to the possibility of an excessive temperature rise and the related problems of massive evaporation, edge radiative cooling, and/or core dilution.

The postulated flattened temperature profiles along the plasma edge as a consequence of the changes in energy transport caused by low recycling have been recently observed in lithium Tokamak experiment (LTX) tokamak during the termination phase of Ohmic-heated discharges where the gas fueling was ceased in a machine operating with plasma boundaries massively covered by lithium.[75] The energy confinement time (measured for electrons) was enhanced up to a factor of 3 (200% enhancement) during the discharge when compared to Ohmic heating scaling law with global recycling coefficients decreasing up to 60%. Moreover, during the termination of the shot (where fueling was suppressed and the temperature profile became flattened), the energy confinement time did not decrease with the global density decay. These observations were the first proof of principle confirmation of the low recycling regime's basic predicted feature. However, the extrapolation of such results to reactor scale cannot be directly done without further research at more reactor-relevant scales (in terms of size, geometry, plasma stored energy as well as during longer timescales). In this respect, LTX has been upgraded to allow NBI heating, thus planning to go deeper in the understanding of the low recycling regime and trying to demonstrate that the achieved novel regime may be supported by total core (NBI) fueling.[76]

More recent works have simulated the hypothetical performance of a JET size tokamak operating with D-T fueling and liquid lithium walls. In this remarkable and striking research, it is theorized that a decrease in the global recycling up to $R = 0.5$ would be translated into a spectacularly increased thermonuclear performance, producing fusion power of 23–26 MW and a fusion gain factor up to 5–7[74] close to the maximum Q objectives of ITER. This would occur despite ITER being double JET in size (R) and magnetic field. The continuous (not limited in time due to passivation/saturation on static lithium surfaces) recycling reduction in JET by using flowing lithium on the divertor plate would increase fusion output by one order of magnitude compared to the previous D-T JET results ($Q \approx 0.7$)[77] obtained with a high recycling plasma. Although such astonishing results obviously need a sound experimental validation, the possibility of attempting a more modest goal (i.e., only overpass breakeven conditions increasing the Q factor in a 50%, beyond $Q = 1$) by using a flowing lithium divertor to pursue a pronounced confinement gain within the low recycling conditions seems noteworthy. With the proper development of the incipient technology that will be widely exposed in Sec. V, these experiments could be attempted in a device that is already constructed and operable.

On the other hand, examples of the exceptional plasma performance theorized by Zhakarov are exactly what has been seen in LTX[75] (and previously in its predecessor CDX-U, as we will comment later, where the energy confinement was enhanced in factor 3), the only machines where sufficiently low recycling for the achievement of the claimed regime has been achieved, but also in the primal experience gained in D-T TFTR operation where L_{TP} was increased by a factor up to 64 with lithium wall conditioning.[45] Concerning this question, it should be recalled that the mentioned machines (TFTR, CDX-U, and LTX) were limiter devices and the direct extrapolations to a diverted device like JET may not be straightforward. Further experimentation in diverted devices with very low recycling conditions driven by flowing lithium components at a relevant scale appears essential to demonstrate technology integration as well as confirm the proof of principle to extend the low recycling framework that LTX provided in nondiverted plasmas to divertor-relevant devices.

Any device operating with a high temperature, low recycling regime would also have a higher β operation and a natural and strong enhancement of fusion power as the burning zone of the confined plasma would be increased to a much larger plasma volume, thus opening the path to more compact, smaller, easier and faster to construct, and less expensive fusion reactors, where learning, innovation, and feedback time lapses may be strongly reduced as well. The increase in the edge temperature and beta would also entail a higher conductivity of the plasma[78] in the boundary that would have beneficial effects related to stability, as already theorized in 1963 for high temperature, collisionless plasmas.[79] Furthermore, tearing modes[80] are mainly caused by high resistivity values in the plasma edge. The eventual effects of such instabilities are linked to plasma disruptions that need to be virtually suppressed in a reactor for economically feasible operation. Disruptions cause dramatic damage of the inner walls of the device. Associated repair and replacement of the PFCs will negatively affect the operational cost and the availability of the reactor. Stabilization of the MHD activity in the edge has been already observed in low recycling experiments with lithium.[81] Additionally, ELMs suppression has been achieved with lithium as well because of decreased recycling and the pedestal profile stabilization.[82,83]

Going back into the original theoretical studies and challenges of the physics of confined plasmas, it is obvious that the first key question in the fusion problem was to minimize the energy dissipation to benefit triple product and fusion performance. Low recycling particularly achieved by fuel-absorbing lithium walls may achieve much better fusion conditions. If the input of cold neutrals to the edge is avoided, the free energy source associated with the created temperature gradient is damped. As postulated in the 1960s, in a plasma scenario where the temperature gradient induced transport is vanished, the diminution of energy losses and heat flux in the divertor may be automatically expected.[84] Additionally, if temperature gradients are significantly reduced and, at the same time, they are accompanied by exponentially decaying density gradients, the conditions would be the opposite compared to those that drive high thermal diffusion and intrinsic unstable modes (relative ion T gradients exceeding corresponding density gradients).

Moving deeper into the theory, Galeev and Sagdeev[85] found a universal, intrinsic instability that is present in any plasma with a finite value for the thermal diffusivity. However, they also showed that if thermal conductivity is removed from the analysis (i.e., temperature gradient vanishes and thermal diffusion does not play any role in the energy flux), then this instability is damped, opening the door to a more stable high-temperature plasma scenario. This is exactly the basic premise of the isothermal tokamak theory and the recent results obtained by Zhakarov where the dependence of fusion gain obtained in a lithium walled JET tokamak would be small with respect to the values of the heat transport coefficients, even in cases where such values are increased by two order of magnitude with respect to the neoclassical ones. A low recycling regime in the ideal case of an isothermal structure is also characterized by an automatic strong poloidal rotation[65] that may be identified as a sign of natural stability, having the same effect compared to $E \times B$ shear flows present in the plasma edge for particular conditions on electron density profiles where turbulence and transport are reduced.[86] The natural, strong peak in density profile at the center will be a characteristic as well in the exposed lithium-driven regimes with core fueling. This density peaking is a factor that has been recently claimed as possible cause of the mitigation of ion heat transport and the concomitant observed improved confinement in W7-X stellarator with pellet fueling.[87]

Certainly, a low recycling device may be expected to present other microinstabilities, mainly driven by density gradients [although other ones such as trapped electron modes (TEM) may be expected as well]. Another universal instability is also associated with high temperature, collisionless plasmas where a density gradient is present.[79] As the low recycling theory framework infers, the radial density profile will exhibit an exponential decay and then these density gradient instabilities will be automatically present. However, the stability conditions to control it (potentially shear flows and/or induced rotation) appear more benevolent when compared to lower temperature collisional regimes so any necessary plasma stabilization seems easier to be driven in such lithium driven collisionless regime[88] than in present main-line fusion devices with high recycling.

E. Confinement improvements worldwide driven by lithium plasma boundary

The use of lithium in fusion devices has been considerably extended worldwide as a powerful wall conditioning technique and/or conforming PFCs.[89] Utilizing lithium in both ways has basically shown to increase energy confinement in all cases. Russian teams started to work in concepts of liquid lithium limiters based on capillary porous systems (CPS) where liquid lithium was supported in a porous mesh[90] for liquid surface stabilization via capillary forces. Soon thereafter experiments were conducted in the Russian T-11 tokamak using a rail CPS limiter and showing the reduction in hydrogen recycling and the good capabilities and robustness of lithium CPS when handling nominal and transient heat loads.[91,92] Among many other tests, experiments with liquid lithium have included its utilization in conventional and spherical tokamaks as well as stellarators by means of coatings or Li injection,[81,82,93–95] additional CPS limiters[96,97] (in TJ-II stellarator and FTU tokamak) and flowing ones,[98,99] showing different grades of benefit and improvement in the plasma performance mainly depending on plasma-surface area covered by lithium and/or its amount present in the plasma boundary, factors directly linked to the recycling reduction. Notable results in relevant machines include the achievement of quiescent, novel edge localized mode (ELM) free mode operation in long duration discharges in Experimental Advanced Superconducting Tokamak (EAST)[100,101] and the suppression and pacing of ELMs by lithium coatings and pellet injection in the NSTX-U plasma edge.[81,82]

Regarding flowing liquid metal solutions, slow-medium flow with both flat surface (FLiLi, Flowing Liquid Lithium Limiter)[98] and trenched [Liquid Metal Infused Trenches (LiMIT) concept that will be more widely introduced in Sec. V] plates have been tested in the EAST tokamak. Experiments with three different generation FLiLi plates have been conducted. The concept is based on the flow of a thin layer ($\sim$1 mm thickness) of lithium over a rectangular plate that is inserted in the midplane of EAST for plasma exposure. The flow of lithium was provided by external electromagnetic (EM) pumping (velocities in the range of cm/s) and is aided by gravity.[98,99] The testing has shown beneficial effects in terms of confinement and reduced recycling, handling heat fluxes up to 3.5 MW/m^2.[102,103] For example, the L-H transition threshold was found to progressively decrease during the operation with the flowing limiter, enabling longer H modes and improving the energy content of the plasma. At the same time, hydrogen recycling and impurity plasma content were reduced and notably,

ELM size and frequency were also progressively diminished. LiMIT concept incorporated a characteristic trenched surface that improved the wetting and global coverage of the plate substrate also offering [thermoelectric magnetohydrodynamic (TEMHD)] capabilities for self-flow of the liquid lithium along the PFC surface. LiMIT also increased the heat handling capabilities clearly beyond 4.5 MW/m^2 (Ref. 104) appearing as a robust limiter that did not present any noticeable damage after the testing. Complete details about the LiMIT concept, its testing as PFC and the technological evolution of the prototypes are addressed throughout Sec. V.

On the other hand, the strong effects of reduced recycling in energy confinement time were observed in the CDX-U tokamak. In experiments combining full solid coatings on the walls and liquid lithium limiters, they found that decreasing recycling (with R_H coefficients around 50%–60%)[105] increased the energy confinement time up to values that exceed the ITER scaling by a factor of 3. These high confinement conditions were driven by the suppression of the plasma edge cooling by thermalized neutral fuel atoms that characterize high recycling regimes, where the formation of strong temperature gradients and concomitant turbulent structures that enhance energy transport and losses take place. These CDX-U results showed that conditions for the low recycling plasma regime may be created by using lithium surfaces that actively absorb the particle fluxes in the edge, thus eliminating the cold particle source responsible for the thermal conduction driven by gradients and the associated poor confinement.[81] Those results were later amplified by measuring a flat temperature profile in its successor LTX as commented in Subsection II D.[75]

III. POWER EXHAUST IN FUSION DEVICES AND RELATED SCENARIO WHEN USING A FLOWING LIQUID LITHIUM BOUNDARY

A. General considerations

In fusion devices, the nominal power loading from plasma will be mostly concentrated on the divertor PFCs. They will need to survive enormous heat and particle loads, not only in steady state, but also during transients [vertical displacement events (VDEs), disruptions, runaways, and ELMs[106]] During operation, the unavoidable erosion of exposed materials must be limited to ensure a sufficient lifetime when using solid PFCs. For ITER, its tungsten divertor establishes a maximum nominal power load around 10 MW m^{-2}.[107] For DEMO-like devices the longer pulsed operation and the associated higher neutron fluence will constrain the limits to a value of 5 MW m^{-2}.[108] Given a maximum allowed erosion for the tungsten divertor elements of 5 mm thickness in two years' lifetime,[109] slow transients surpassing the limit will be only allowed for short times (10 s) for a limited number of cycles. Type I ELMs must be suppressed, and disruptions and/or vertical displacement events totally avoided.[110] High Z tungsten elements are not compatible with a hot plasma edge, so to accomplish this operational scenario, the temperature in front of the divertor plates needs to be sufficiently low (5 eV) to stay below the sputtering threshold.

For a tentative, high Z, high recycling DEMO reactor operating with $Q_{fus} = 10$, where heat losses to the divertor will be dominated by strong gradients beyond the pedestal, this power to be exhausted will be in the range of 500 MW.[109] To limit the heat flux to the divertor, a high fraction of the power must be radiated. Radiated power exhaust fraction (f_{rad}) needs to be around 90%–95% both in the divertor (detachment[111,112]) and core (higher Z seeding).[113] Obviously, this will degrade the confinement in the core and needs to be carefully controlled so as not to surpass the L-H threshold and/or induce radiative collapse [such as Multifaceted Asymmetric Radiation From the Edge (MARFE) events[114]] and at the same time be sufficient to mitigate the heat flux to divertor elements until the 5 MW/m^2 established limit. This high radiation scenario and its tight safety margins are yet to be demonstrated and will need an extremely efficient control as any failure will be notoriously harmful for the reactor operation and the integrity of the inner walls.

The primary factor to evaluate the power finally reaching the surface of the divertor is the parallel heat flux across the separatrix (q_{sep}) that follows this expression:

$$\mathbf{q_{sep}} \sim \mathbf{P_{sep}}/\left(\mathbf{R} \cdot \boldsymbol{\lambda}_\mathbf{q} \cdot \mathbf{B_p}/\mathbf{B_t}\right), \tag{17}$$

where P_{sep} is the total power across the separatrix, R is the major radius of the machine, B_p and B_t are the poloidal and toroidal magnetic fields, and λ_q is the exponential power decay length in the SOL. This parameter is crucial for the estimation of the peak heat load deposited on the divertor target. Smaller values of λ_q will cause higher heat fluxes and more challenging power exhaust handling in the divertor.

Predictions or estimations of this parameter are essential to evaluate the design of future fusion devices (i.e., ITER and DEMO). Multimachine investigations (Eich scaling) have demonstrated that λ_q scales inversely with poloidal magnetic field[115] and does not scale favorably with machine size (R).

Since power at the separatrix scales faster than ~R, and the power decay length does not follow the P_{sep}/R scaling,[108] these results have a significant impact in future machines as P_{sep} will be significantly increased, so to achieve a constant heat flux, an increase in λ_q would be needed. ITER experts group assessed values of $\lambda_q = 5$ mm to be achieved in partial detachment conditions for an acceptable plasma performance with tolerable material damage and suitable power dissipation.[107] Unfortunately, that value is a factor 5 larger than direct extrapolations of the Eich scaling and consequently determining a highly radiative operation scenario as previously commented.

B. Liquid lithium and high temperature SOL scenario

Theoretical work on the scaling of the divertor power decay width has been carried out by Goldston considering a heuristic-drift model[116] in low puffing H-mode plasmas in which the divertor would significantly act as a particle sink. The model estimates the resultant SOL λ_q width as the product of the residence time of the ions in the SOL and their perpendicular averaged drift velocity, resulting in a scaling $\lambda_q \sim \rho_p \cdot \varepsilon$, where ρ_p is the ion poloidal gyroradius ($\sim T_i^{1/2}/B_\theta$) and ε the inverse aspect ratio, then the scaling may be assessed as

$$\boldsymbol{\lambda}_\mathbf{q} \sim \boldsymbol{\varepsilon} \cdot \mathbf{T_i}^{1/2}/\mathbf{B}_\theta, \tag{18}$$

where B_θ is the poloidal magnetic field. This expression poses an explicit relationship that analytically agrees with the Eich scaling[115] with respect to the magnetic field dependence and it also favors configurations where ion temperature is enhanced.

In the lithium low recycling proposed configuration, the liquid surfaces would act as an outstanding particle sink resulting in an edge temperature much higher compared to high recycling ITER-like devices. The ITER-like SOL will be highly collisional with maximum temperatures on the order of 100 eV beyond the pedestal, which needs to

diminish to <5 eV in front of the divertor targets. In a lithium regime, the SOL conditions would be almost collisionless with a high fraction of trapped particles.[117] Consequently, the particles reaching the divertor plates by pitch angle scattering, in reactors with a very hot (∼10 keV) temperature and conversely tiny density in the SOL, would have λ_q that would also scale with Larmor poloidal ion gyroradius.[118] Such differences increase the power width by a factor of $\sqrt{\frac{T_{hot\,SOL}}{T_{cold\,SOL}}}$. That value may be as high as an order of magnitude $\left(\sqrt{\frac{\sim \mathrm{keV}}{5-10\,\mathrm{eV}}}\right)$. The resultant hot edge-SOL structure, where ion temperature gradients and associated turbulent transport would be strongly reduced, would have had one order of magnitude lower divertor heat loads, relaxing the power exhaust handling requirements of the divertor solution. Therefore, a low-recycling lithium-divertor machine could be smaller with a higher power density and have no worse of a power load than that imagined for ITER and DEMO. It appears plausible to optimize such benefits in order to compensate the power exhaust increase derived from the smaller inner area of an envisioned more compact reactor or caused by potential higher magnetic field.[119–121] Likewise, it is also worth mentioning that the liquid lithium solution might also be combined with advanced divertor magnetic configurations that offer the possibility of spreading the heat fluxes over larger divertor areas as double null, X, Super-X, Snowflake, or long-legged ideas[122–125] as well as with, otherwise, potentially required seeding scenarios.[62]

These implications on future fusion reactors and associated divertor designs should be of prime importance since the heat exhaust problem in the divertor continues to be one of the most important challenges. Rather than attempting to solve the problem by cooling down the plasma that surrounds the divertor elements to induce detachment and assure the survival of the PFCs, a lithium walled reactor would suppose a completely different approach that would attack the problem from the root. In a completely different physics scenario, the heat flux from the core is reduced as turbulence decreases and confinement of energy improves. Therefore, the heat loss rate, driven by particle flux, entering the SOL is damped as a result of the combination of trapping of particles in the region, the very low density values, the reduced temperature gradient, and an exhaust mechanism of pitch angle scattering for the majority of particles.

C. Additional power handling benefits

The utilization of flowing liquid lithium surfaces as PFCs has additional advantages since the heat load extraction will not be limited to conduction as in the case of conventional solid PFCs. On the contrary, liquid lithium streams can dissipate power by means of convection, evaporation and that lithium vapor can shield the divertor through radiation.

Enhancement of the heat handling capabilities by convection has been observed in flowing liquid lithium experiments when exposed to reactor-relevant plasmas.[99] Massive evaporation at sufficiently high temperatures may produce the creation of a vapor cloud that may be screened in the SOL due to the prompt redeposition of lithium in the plasma boundary.[64] Cyclic vapor shielding in lithium PFCs has been observed in Magnum PSI in cold divertor-like plasmas when the temperature of the lithium surface increased up to 700 °C.[126] If this vapor cloud is sufficiently thick, the heat dissipation may be important especially when combined with lithium radiation produced by the ionization/excitation of these vaporized atoms. Figure 3 shows such lithium radiation in the plasma boundary (color red for Li I transition and green for Li II) of EAST tokamak during the previously mentioned experiments with LiMIT plate.

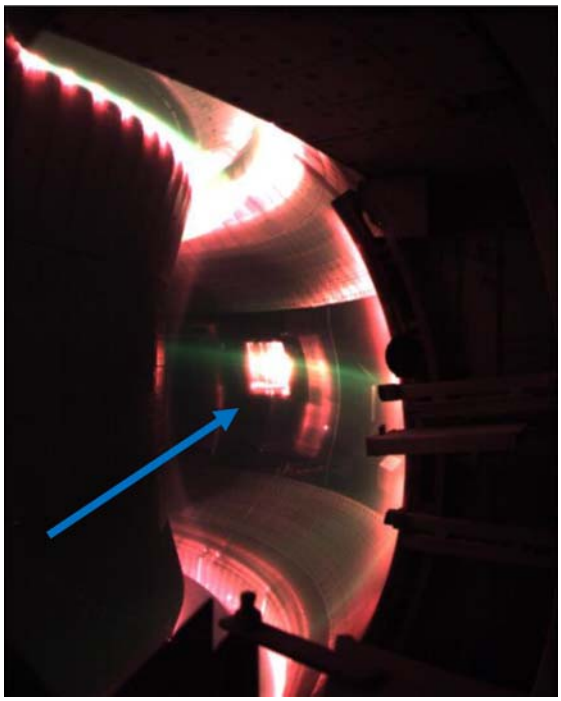

FIG. 3. Lithium radiation (showed in color) within the plasma boundary observed during the LiMIT operation in EAST tokamak. The LiMIT limiter plate is the rectangle at the end of the arrow.

The theoretical considerations[127–130] have shown that a significant power dissipation is possible by means of noncoronal radiation of lithium in plasmas where high evaporative fluxes are present, requiring temperatures clearly beyond 600 °C on the liquid metal surface. At such temperature, however, no net retention of hydrogen particles may be expected on the flowing lithium divertor, thus totally precluding the achievement of lower recycling, but also the lithium contamination and accumulation in plasma and/or solid first wall may compromise the reactor operation due to the commented core dilution but also due to fuel codeposition problems on the first wall that, additionally, might threaten the tritium inventory limits inside the reactor (see more details about this specific drawback issue in Subsection V E). For such reasons, this vapor shielding scenario will not be compatible with a low recycling divertor configuration able to continuously flow and absorb the hydrogenic plasma exhaust flux. Nonetheless, these phenomena may be extremely helpful in a reactor, when the lithium elements face a possible off-normal transient event. Immediate, strong evaporation that will follow the intense localized heating of the lithium surface will be a natural defense mechanism that will protect the in-vessel elements. The volumetric dissipation of the transient heat will evolve as a radiative collapse rather than as a strong disruptive episode concentrated on the solid wall that will produce irreversible damage on solid PFCs.[131] In this sense, the exceptional characteristics of lithium to protect substrates have been observed in experiments exposing lithium PFCs to extremely high, reactor-relevant transient

heat loads at the level of what may be expected in the more catastrophic transient events.[132,133]

Finally, it is interesting to mention the lithium vapor box concept proposed by Goldston *et al.*[134] as perhaps an extreme case of the lithium radiative divertor scenario. In this concept, a liquid lithium pool would be statically confined in a differentially pumped divertor box presenting different sections operating at distinct temperatures. Operating at the proper temperatures it would prevent from excessive lithium vaporization to the plasma by means of lithium condensation on cold (∼300–400 °C) baffles and, at the same time, would be able to detach the plasma from the box strike points operating at hotter temperature (∼600 °C). Recently, the authors have determined that a lithium evaporative temperature of 580 °C would be sufficient to volumetrically dissipate the divertor power exhaust reaching the SOL (∼200 MW is estimated in this study as conservative value) of a relevant fusion power plant operating with this detached divertor configuration driven by lithium vaporization and radiation. This conceptual study was carried out developing a power balance and a detachment model in the lithium vapor box.[135] The proposed technology is now under development in order to be tested at laboratory and linear plasma device scale. However, the reader should note that the idea is conceptually incompatible with the flowing lithium divertor scenario, as the liquid lithium would be intrinsically static and operating clearly beyond the temperature limits for proper hydrogenic absorption by lithium and associated conversion to hydride addressed in Ref. 59. Consequently, in this concept, the claimed (and outstanding) benefits in recycling and concomitant edge cooling suppression, energy confinement, and plasma performance enhancement of a low recycling boundary driven by a flowing divertor configuration may not be certainly expected as the vapor box concept aversely aims to induce the cooling (detachment) of the divertor plasma edge.

IV. ECONOMIC CONSIDERATIONS OF A FLOWING LIQUID LITHIUM DIVERTOR SOLUTION

Several studies have approached the economics of fusion energy, focusing on the dependence of the electricity cost (COE) with physical and technological aspects.[136,137] Such estimations can be averaged and evaluated in terms of COE percentage: direct cost (investment) for the construction of the reactor and associated installations (∼60%), cost for the replacement of divertor and blanket elements (20%–30%) and sum of fuel, operation, maintenance, and decommissioning costs (10%–20%). The direct investment cost is clearly increased in larger size machines, being considered directly proportional to machine volume ($\sim R^3$).[10] At this respect, it is worth mentioning that the projected size for ITER and DEMO is considerably larger than the actual size of fission reactors. A light-water-reactor (LWR) core can actually fit inside the ITER center stack. LWRs are a mature technology where a $5B capital cost can make a 1000 GWe power plant in 5 years which will last for 40 years of near continuous operation. If the world turns to a nuclear option to replace fossil fuels, fusion must be on the same order of costs, although it is also necessary to keep in mind that the much higher external cost related to the nuclear fission wastes will also affect to the comparative economic feasibility of both technologies.

In a lithium-based low recycling regime, burning conditions and high fusion Q gain could be achieved in smaller reactors compared to the case of high recycling DEMO scenarios. Just considering the predictions[74] for JET tokamak operation where it might produce a similar fusion performance compared to the expected in ITER-like Q factor could be obtained in a machine with half the size and approximately half the magnetic field value, resulting in a construction cost that would be only an eighth when compared to a doubled size machine. As the capital cost of reactor construction and its depreciation is around 60% of COE[137,138] the associated reduction in a factor 7/8 might imply an outstanding decrease in COE around 50%. This assertion will be valid if we consider fusion devices with the same magnetic field whose construction cost would approximately scale with R^3 as inferred in Ref. 6. For devices with increasing magnetic field, however, the effect of such parameters should be taken into account as the higher field will increase the associated cost of the magnets and other related subsystems. In this respect, perhaps a direct cost scaling with $R^3 \cdot B^2$ (expression proportional to the stored magnetic energy) may be more realistic for the case of higher magnetic field reactors.

It is also important to note that the reactor size and concomitant plasma volume will also affect the fusion power that may be extracted from the reactor. The dependence of the fusion power with engineering (external values of toroidal field, major radius, and inverse aspect ratio) and internal performance parameters such as safety factor q and normalized beta may be written as[121]

$$P_{\text{fus}} \sim \frac{\beta_N^2 \cdot B^4 \cdot R^3 \cdot \epsilon^4}{q^2}. \tag{19}$$

On the other hand, as inferred in Ref. 139, the maximum reactor performance given a minimum device size will be reached by operating at highest possible values of Greenwald density fraction, normalized beta, and also H factor. Both beta and H are expected to be clearly increased in a lithium low recycling regime,[42,57,71] hence the influence of the lower size in the fusion power might be compensated through such effects. Additionally, Costley *et al.*[139] scanned the fusion performance parameters (triple product and Q_{fus}) depending on size for fixed values of $P_{\text{fus}} = 200$ MW, $Q \sim 5$ and usual Greenwald density, aspect ratio, and beta values (being all these values ITER range). They showed that operating at high confinement ($H = 1.5$), ITER range Q factors might be achieved in smaller devices around $R = 3$ m (JET size) with a higher magnetic field in the range of 8 T, being 5.3 T the toroidal on axis magnetic field of ITER. This magnetic field value is smaller compared to other scenarios where engineering and stability advanced, high field tokamaks are considered.[121]

More interestingly, if confinement might be increased up to H values of 1.9 being accompanied by high elongation spherical shape, the magnetic field requirements would be clearly relaxed until very plausible levels comparable to JET metrics. Such confinement increase is clearly within the enhancement seen in machines operating with walls massively coated by lithium[75,105] than even increased the H factor threefold. In this way, the doors would be open to improve generated power (compensating in this way the effect of a lower machine size) by moderately increasing the magnetic field [$P_{\text{fus}} \sim B^4$ as Eq. (16) shows] up to ITER values that would not compromise the main actual concerns about plasma stability and would not worsen the power exhaust scenario where λ_q is expected to scale inversely with poloidal B. In this aspect, a liquid lithium solution would ameliorate such λ_q constraints giving an opportunity to increase the magnetic field without compromising the power exhaust requirements.

Another important economic/technological challenge for fusion reactors is related to the extreme heat/particle exhaust conditions and

the very energetic generated neutron fluence and their effects in the PFCs and structural materials. Radiation damage on breeders and progressive deterioration/destruction of solid PFCs will determine the global machine duty cycle and their possibilities to be economically competitive. Consequently, the use of advanced materials with longer lifetime will be clearly necessary to reduce the cost of fusion electricity up to affordable levels.[140] The approach based on a flowing liquid lithium divertor can increase the lifetime of such components, as the liquid surface exposed to plasma will be immune to permanent deterioration and destruction. If liquid loop technologies[141] for injection, pumping, tritium recovery as well as cleaning and recirculation of the liquid PFC are developed, the liquid lithium component can be continuously replaced, purified, and recycled back inside the reactor in real-time. This advanced configuration would be especially important in the case of the divertor material, whose replacement frequency is established at two years even considering the most optimistic scenarios.[112,113] However, it is important to bear in mind that the technological readiness level (TRL) of the mentioned systems currently corresponds to prove of concept. Future research will need to raise the technologies until higher TRLs consistent with more realistic reactor scenarios when compared to potential power plant applications. Additionally, the effects of these incipient technologies on the operating costs may be expected, although the arguments presented in this chapter suggest that the benefits in reactor size and direct construction cost would be more important.

Conceptual studies have suggested a minimum availability of 75% for a fusion power plant to be competitive.[142] Furthermore, engineering analysis envisions as baseline scenario a two-year lifetime for divertor and four-year replacement of the breeder blanket elements, being this replacement coincident with the second one of the divertor.[143] The remote handling replacement strategy entails six months period to replace just the blankets being equal compared to the time to replace the blankets and divertor because the divertor elements must be removed first in any case. It also considers a shutdown time for replacing only the divertor of four months, also adding a one-month cooling period before each replacement and one month of conditioning and pumping before reactor restart. The resultant approach will consist of a total, periodic cycle of 48 operational months in a total period of 62 months that would fulfill the availability demands with a global value of 77.8%.

A self-replenishing liquid lithium PFC reactor solution may entail a considerable reduction in the cost and shutdown time associated with divertor replacement, also increasing the duty cycle and the availability of the power plant. Considering as conservative assumption that the lifetime of the divertor may be increased up to the level of the blanket one (thus passing from two years to four years), the maintenance cycle would contain only one reactor shutdown of six months after 4 years operation. In Fig. 4, a radial chart comparing both operational/maintenance scenarios for conventional DEMO and flowing liquid lithium option is presented. It shows that availability of the plant would be increased up to 85.7% for an equivalent operational time (48 months) that would be completed in a total period of only 56 months, result that implies an 8% equivalent reduction in COE. The dwell time (pumping, conditioning, etc.) is reduced a 50% (from 4 months to 2 months), the shutdown time for replacements is reduced a 40% (from 10 months to 6 months), and the number of divertor replacements would be a half, so the associated cost would be reduced

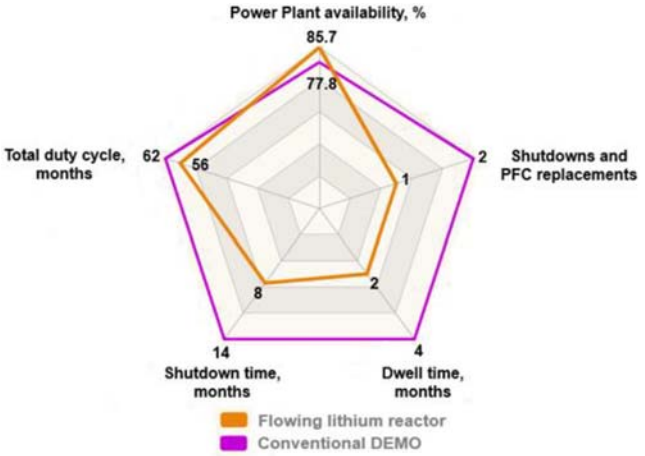

FIG. 4. Radial chart presenting the tentative two different operational and maintenance scenarios in a 4-year operation cycle (48 months). With a self-healing, self-replenishing liquid lithium divertor, contingency aspects (dwell and shutdown time) could be considerably reduced (factors 50% and 40%), dividing by two the PFC replacement (and related costs), finally increasing plant availability from 77.8% to 85.7% and giving an extra security margin for the minimum fulfillment of 75% that is considered to make fusion energy minimally competitive.

in the same factor. The cost of divertor and blanket replacement is approximately assessed around 23%[137] and 30%[138] of the COE, considering as conservative estimation that the specific contribution of divertor replacement may be a half of the blanket one, the 50% reduction in divertor replacement would be directly translated in an additional 3.8%–5% COE reduction.

Globally, a high confinement smaller reactor solution with a liquid lithium divertor, in which power generation might be similar compared to ITER size machines by means of beta and H improvement as well as a possibly conservatively larger magnetic field, would be associated with reductions in COE up to 50% for the smaller size and lower construction cost, 5% for the lower divertor replacement requirements, and 8% for the related higher availability of the plant resulting from a less time-consuming replacement and maintenance. In this respect, it must be specified that the considerations made herein are not based on existing experimental data (as flowing lithium divertors have not operated and flowing lithium limiters have only been used a few times). On the contrary, they are based on conservative estimations taking into account the characteristic self-healing nature of the liquid PFCs that theoretically opens the possibility of enlarging the PFC lifetime. For such flowing liquid metal (LM) solutions, the incipient technologies (widely presented in Sec. V) need to be demonstrated at proper TRL in order to show that they are capable of providing a larger operational time and also to prove that they do not introduce additional problems that might limit the reactor duty cycle.

Another possible technical benefit of this lithium configuration might be related to the extraordinary impurity gettering and pumping capabilities of lithium. Pumping of impurities (mainly residual vacuum and/or seeded species) is carried out in the subdivertor region of tokamaks by means of cryogenic units that saturate quickly and whose regeneration may compromise the reactor duty cycle. The technical needs of cryopump regeneration (possibly affected by ammonia

generation), where the necessary massive outgassing would obligate to stop the plant operation, is already threatening the much shorter duty cycle of ITER if nitrogen is used as divertor radiator[144] and similar constraints might be expected in future reactor prototypes operating in complex seeding and/or strong cryopumping requirements. Any reduction in the cryopumping needs would be translated in relaxation in the regeneration needs and/or in the number of necessary units with immediate benefits in the reactor availability and also makes the pumping system simpler with cheaper and less time-consuming maintenance. Regarding this question, the direct comparison and extrapolation of the Li divertor impurity pumping capabilities respect to the usual cryopumping systems in a reactor are intrinsically difficult, being a question that seems to not be addressed yet in the specialized literature. In principle, one could guess that lithium gettering will help in the approached rector scenario but it is also true that the presence of the liquid lithium on the divertor surfaces may be problematic for the cryopump placement/installation and therefore might reduce the efficiency of the cryopumping system (loss of vacuum conductance, direct obstruction of the pumping channels by lithium…). Therefore, in order to assess extra cryopumping needs due to the lithium presence, but also to guarantee that the flowing lithium configuration is consistent with the cryogenic system, it is important to note that engineering/design efforts toward an optimization/integration of both divertor subsystems will be necessary to eventually develop a proper impurity pumping scenario compatible with a flowing lithium divertor.

Finally, when fully operative, an important requisite for high availability of fusion power plants will be an almost total absence of disruptions.[110] In this respect, lithium wall configurations may significantly lower the risk and probability of disruptions. The enhanced flattened temperature profiles over the plasma volume would entail a decrease in the turbulent transport and associated plasma instabilities such as resistive tearing modes that drive transient collapse. Even if a disruption happens the liquid material will protect the structural components of the reactor, thus preventing destructive damages that, otherwise would need to be replaced, thus dramatically effecting reactor availability and the COE.

V. TECHNOLOGICAL CHALLENGES OF FLOWING LIQUID LITHIUM PFCs

While this paper has made a case for lithium PFCs, it has not yet examined their shortcomings or how to actually make it all work. This section will examine this subject, highlighting the work at Illinois.

A. Self-flowing liquid metal PFC concepts and power handling

At UIUC, flowing liquid metal PFCs have been developed by employing thermoelectric magnetohydrodynamic (TEMHD) effects to drive self-generated movement on liquid lithium infused in small trenches where a stabilized flow can be used to mitigate high heat fluxes and provide a clean lithium surface to the plasma. This phenomenon was first observed by Jaworski and co-workers.[145] Because of such effects, the thermoelectric current created between two dissimilar metals when subjected to a thermal gradient (being provided by the plasma heat flux and the internal cooling of the element) can produce a driving Lorentz force when held in a transverse magnetic field.[146] Following this approach, in 2011 UIUC researchers designed and constructed the first LiMIT (liquid-metal infused trenches) PFC. The complete theoretical analysis of the TEMHD self-flow concept is detailed in Ref. 147.

The first LiMIT design was based on narrow trenches (width ~mm) infused in a liquid lithium bath (depth ~ 5–10 mm). This configuration was tested at laboratory scale in MW/m^2 range electron beam exposure experiments, demonstrating stable flow under magnetic fields in different (vertical, horizontal, and obliquely oriented) configurations,[148,149] with flow velocities in the range of 5–15 cm/s regardless of the geometrical orientation of the plate. Figure 5 shows both 3D and cross-sectional schematics of this first LiMIT design as well as results from calculations of the thermoelectric currents and the specific force induced in the liquid lithium bulk. Successful testing of the LiMIT system has been carried out in the HT-7 tokamak,[150] the Magnum PSI linear plasma device[151] and very recently in the EAST superconducting tokamak[104,152] where a LiMIT plate was coupled to the preexistent full liquid lithium loop of FLiLi antecessors.

In Magnum PSI, the tests showed that the circulating liquid lithium was capable of safely handling heat loads of 3 MWm^{-2} during timescales of 5 s[151] with velocity measurements (up to 70 cm/s) that were consistent with the predicted TEMHD based models. Testing in HT-7 (2012) was the first full-scale test of the system at high toroidal magnetic fields (1.6 T). The observed flow velocity ~4 cm/s matched the theoretical prediction within the error bars.[153] Higher fields slowed the flow velocity, but sufficient speed remained to ensure a clean absorptive surface. Even with only a partial fill on the lithium plate (that anyway covered a very small area of the in-vessel area), the global plasma confinement properties were shown to improve with a 10% increase in confinement time.[150]

The experiment in EAST aimed to investigate the lithium effect on the plasma and to explore if the TEMHD concept and the trenched surface coupled to a loop may help in an actual fusion device regarding key issues as heat handling and wettability. The LiMIT plate (Fig. 6) was manufactured from molybdenum [titanium-zirconium-molybdenum alloy (TZM)] with dimensions of 320 mm × 300 mm × 20 mm, with 0.5 mm depth trenches that were orientated vertically and perpendicular to the toroidal magnetic field. The plate was coupled to a full lithium loop system including an EM pump, a distributor placed on the top of the plate, a lithium collector on the base and an external reservoir with an injection system for the lithium.

Forty shots were done to test the LiMIT plate against different plasma heating power, distance from the separatrix, cooling pressures in the plate for TEMHD, and operating temperature. Figure 3 shows the plate inserted on the midplane of EAST during plasma operation. Analysis of the data is still ongoing but has demonstrated heat handling of 4.5 MW/m^2 (being the analysis of high power NBI shots, with expected much larger handled heat flux, still pending), improved plasma performance, and was not damaged by the plasma.[104,152] This LiMIT plate has many similarities to the flowing liquid lithium (FLiLi) system which uses gravity to produce a slow thin flow.[98] Improvements to accomplish a power handling even higher are presented in Subsection V C.

B. Liquid lithium surface stability and dryout

The first fundamental problem derived from using a flowing liquid lithium PFC is the stability of its surface under intense heat loads, magnetic fields, MHD activity, and induced $j \times B$ Lorentz forces. In such a scenario, Rayleigh–Taylor instability (RTI) and

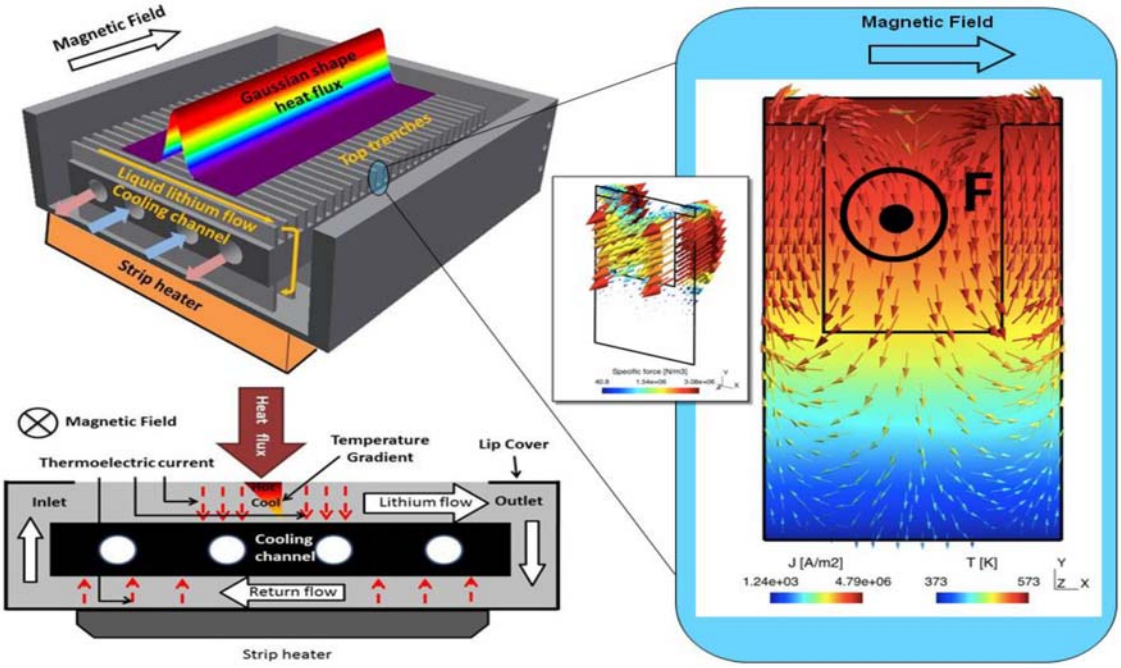

FIG. 5. 3D and sectional schematics of liquid-metal infused trenches (LiMIT) first concept explaining how the temperature gradient produces a thermocurrent which in-turn creates flow due to a $j \times B$ force. Upper right shows current density and specific force calculations for typical operating conditions.

Kelvin–Helmholtz instability (KHI) on the liquid PFC can be triggered. These instabilities can lead to ejection of liquid metal droplets from the melt unless stabilized by surface tension.[154,155] Generally, RTI is gravity driven events produced when the interface between a lighter and a heavy fluid is perturbed, thus producing bubbling on the low-density fluid and spikes on the high-density one as a consequence of the conversion of potential energy into kinetic energy.[156] On the other hand, KHI instabilities occur when two inviscid fluids are in relative and irrotational motion characterized by a discontinuity on the density and tangential velocity profile at the interface. Such discontinuity generates a shear flow and induces vorticity on the boundary layer that becomes unstable after the instability grows creating vortex/spiral structures that eventually eject. The process entails a conversion of kinetic energy (taken from the mean flow) into potential energy that is translated in a relative movement (up and down) of the heavy and light fluids, respectively.[157] To combat those phenomena, capillary porous systems, being explored mostly in Russia and Europe, utilize the strong surface tension effects of the porous substrate-liquid metal interface to avoid such instabilities. However, to operate continuously in a low recycling regime a macroscopically flowing system to provide a fresh, clean absorptive boundary will be needed. By varying trench dimension and exposing a LiMIT system to pulsed plasmas a stability criterion was developed[155] which explained previous results in DIII-D and NSTX.[154,158] It also showed that thin enough trenches ($\sim$1 mm in width) will not eject droplets.

Depending on the average velocity of the liquid metal stream, flowing schemes can be classified in slow-medium flow (the only flowing PFC technology tested to date, with velocity in the range of few to tens of cm/s) and fast flow solutions being proposed as future configurations (velocity in the range of m/s with much thicker, $\sim$ few cm,

FIG. 6. The LiMIT plate used in the EAST winter campaign of 2019/2020. The front face trenches (left) are vertical and perpendicular to the magnetic field while the Langmuir probes are circled in red. Right: back side of the plate with the cooling lines, heater elements, and thermocouples.

liquid metal layers[159]). Higher velocity flows have a much higher heat handling capability as the fast speed would also increase the contribution of convection in the dissipation, with the liquid lithium stream acting as a coolant itself. However, such velocity and the nature of the related flow also interplay on the surface stability, so ejection concerns are much more important as speed increases. Consequently, for future divertor concepts considering fast, annular flow of liquid lithium, along the divertor, much more advanced external methods/configurations would need to be developed to assure a stable fast-flowing solution in a high magnetic field divertor environment.[160–162] In this respect, a recently published paper has addressed a new interesting concept called "divertorlet"[163] that pretends to reduce the velocity of the liquid metal at the free surface by means of multichannel design that will minimize this flow length. Besides, the fast flow will be maintained in the vertical direction as opposed to the one that faces the plasma, thus trying to combine the high heat flux capabilities of fast flow schemes with a more stable and conservative liquid metal surface facing the plasma.

For such comparative, lower complexity reasons, the slow-medium flow has been the more developed to date. Two different varieties of plates (flat plate FLiLi and trenched LiMIT) have been tested as flowing liquid lithium limiters in EAST tokamaks in experiments where the PFCs were integrated in a EM pumping scheme. For both plate designs, the system was able to pump lithium and continuously refill the liquid surface facing the plasma within the tokamak at the plasma edge. The slow-medium flow solutions provided a stable flowing surface able to operate in relevant tokamak environments, although a number of technological constraints (related to wetting and material compatibility, topics approached in Subsections V D and V G) were found and need to be improved and solved for a longer duty cycle, real reactor solution.[103,152]

A second concern related to surface stability is the perturbation of the thickness of the lithium layer, leading to the underlying plate or structures being exposed to the plasma when bombarded by a nonhomogeneous, highly localized heat flux. This phenomenon called dryout[164] and produces a depression of the flowing liquid lithium surface where the heat load is concentrated. The resultant thermal gradient on the lithium surface interplays with magnetic field generating local acceleration that depress the liquid interface, exposing the solid surface beneath and producing a local increase in the level of the liquid in the downstream lithium region. Figure 7 shows a sketch of the surface dryout phenomenon.

C. New LiMIT geometries that ameliorate dryout, improve surface stability, and power handling

The lessons learned from the results of LiMIT testing in the variety of devices exposed in previous subchapters revealed some technological concerns about its feasibility. Main difficulties were related to dryout and nonfull coverage with lithium of the PFC, aspects both technologically key. In order to ameliorate these issues, the dryout of liquid lithium layers on the components must be mitigated for improving the stability of the liquid metal surface, also assuring a homogeneous coverage of the substrate. These enhancements have been shown essential in order to advance configurations able to be tested at longer timescales and in more relevant devices. Consequently, during the last few years, Center for Plasma Material Interactions (CPMI) at Illinois has developed different experimental and modeling works as well as manufacturing techniques in order to explore potential novel geometrical designs and engineering for the LiMIT elements that can minimize these problems and potentially handle more relevant reactor power loadings ($\geq 10\,\mathrm{MW/m^2}$) without deterioration of the elements. Computational modeling is being implemented for analyzing the TEMHD flow in the proposed elements by using COMSOL Multiphysics in order to predict experimental behavior and select the most advantageous plate configurations for further experimentation and designing. The experiments carried out were centered in the exposure of different designs of LiMIT plates to $\mathrm{MW/m^2}$ range heat fluxes provided by an electron beam in the SLIDE (solid–liquid divertor experiment) facility widely described elsewhere.[165]

The first prototype consisted of a two-dimensional (2D) surface with different rectangular posts and spaces placed along the plate (see Fig. 8). Three different size configurations were tested within this geometry: $1 \times 1\,\mathrm{mm^2}$ posts with 2 mm separation, $2 \times 2\,\mathrm{mm^2}$ posts with 2 mm separation, and $2 \times 2\,\mathrm{mm^2}$ posts with 4 mm separation. The other solution proposes the utilization of a modular 3D hybrid CPS-LiMIT system with partial use of the capillary effect to decrease dryout and flow depression and thus improve the liquid lithium surface stability. The advanced 3D porous "ordered foam" configurations were prepared by using laser methods to 3D print stainless steel structures (Fig. 8). The experimentation was carried out with three different geometrical sizing configurations (see figure caption for details).

Regarding the 2D postgeometry, the measured velocities were in the range of 2–8 cm/s in good agreement with the modeling works. Crosstalk flow and swirling along perpendicular direction to the main flow was predicted by simulations and observed in the test (Fig. 9).

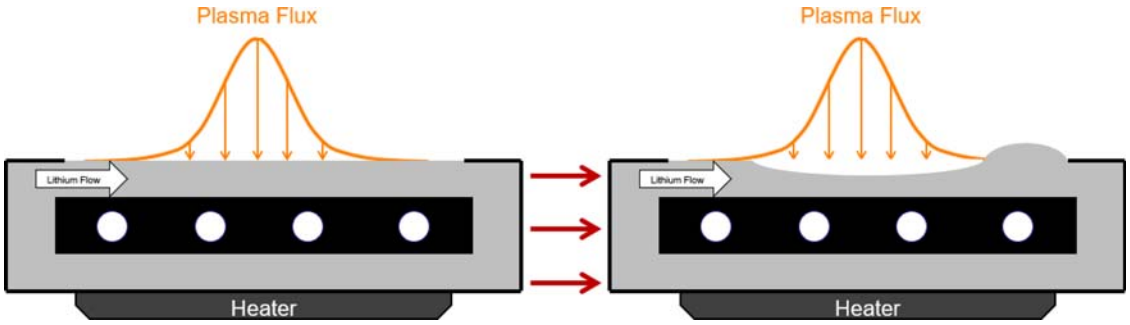

FIG. 7. Schematic of dryout caused by localized plasma flux in a plate with a flowing liquid lithium layer. [Adapted and reproduced with permission from M. Szott and D. N. Ruzic, Fusion Eng. Des. **154**, 111152 (2020). Copyright 2020 Elsevier B.V.[164]]

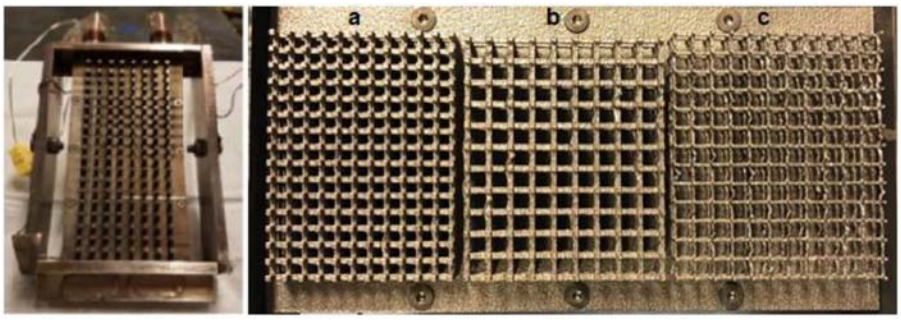

FIG. 8. Left: Example of 2D post geometry consisting in $2 \times 2\,mm^2$ rectangular posts with $4\,mm^2$ separation used in the experiments and modeling works. Right: The three different 3D porous LiMIT geometries tested in SLiDE fabricated by 3D printing technology: (a) design with $1 \times 1\,mm^2$ structure with 2 mm separation, (b) the $1 \times 1\,mm^2$ structure with 3 mm separation, and (c) the $0.5 \times 0.5\,mm^2$ structure with 3 mm separation.

It resulted in an amelioration of the dryout problem that, however, was not totally suppressed.[165,166]

The overall performance of the 3D ordered foam LiMIT plates was excellent in terms of improved wetting and coverage and also regarding surface stability and dryout mitigation. Additionally, the heat handling capabilities of the solution were improved respect to previously tested geometries.

Figure 10 shows a COMSOL simulation of the experiment showing the velocity profile of the lithium flow along the LiMIT module consisting of a $1 \times 1\,mm^2$ 3D modular structure with separation of 3 mm gaps.

Full coating with effective wetting was achieved on the texturized surface. Rapid swirling of the liquid was monitored showing bulk flow throughout the structure. Enhanced surface stability beyond the standard LiMIT design was provided by excellent capillary action that, otherwise, was compatible with TEMHD drive. Power exhaust handling showed an increase in the known LiMIT-style PFC operating window by 127%, being enhanced from 3 to 6.8 MW/m^2 of impinging heat flux. At the same time, the robustness of the element against damage or liquid ejection/dryout was excellent. No dryout or ejection was observed.[165]

Future testing of this PFC concept will occur as a flowing liquid lithium divertor plate for the ST40 tokamak,[167] in a project with Tokamak Energy Ltd. that will pair the lithium low recycling benefits with the high field, spherical torus approach. The LiMIT element will be integrated into a loop containing a complete set of accessory elements that are being designed as well: a load lock section for lithium, EM pumps, a lithium reservoir, pumping lines, and distribution and collection units placed within the plate, as well as sensors, flow meters, and feedback systems for a safe, continuous operation. Figures 11 and 12 show sketches of the different designed elements.[168]

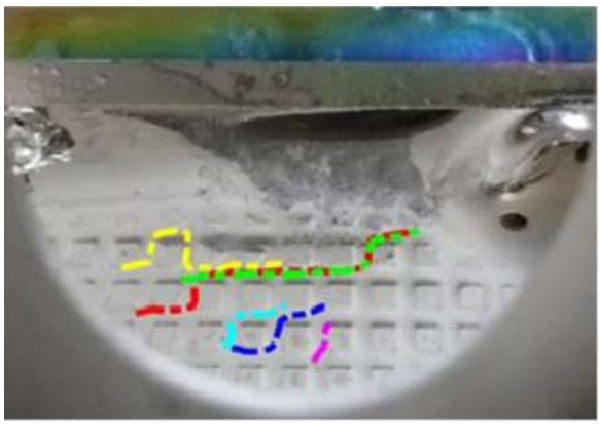

FIG. 9. Crosstalk flow (shown in color dotted lines) in the 2D LiMIT postgeometry.

D. Wetting, spreading and distribution of liquid lithium

In flowing liquid metal PFCs, proper wetting on the underlying solid support structure and homogeneous distribution for flow needs to be achieved at temperatures compatible with the material vapor pressure and potential substrate corrosion limitations. As liquid metals pose high values of surface tension this adequate wetting behavior is not trivial. For lithium low recycling, the upper operational temperature limit is set at 450 °C approximately. Additionally, the wetting control will be selective depending on the surface location. Regions of plasma exposure must always be covered by liquid metal; however, other surfaces should not wet to prevent wicking of the liquid metal away from the desired zones. As seen in experiments performed on HT-7 and EAST tokamaks with both flowing liquid lithium limiters (FLiLi and LiMIT)[102–104,150,152] absence of total lithium coverage on the substrate was reported and resulted in non-optimal operation of the PFC.

During the last few years, CPMI at Illinois has studied the static wetting properties of liquid lithium on fusion-relevant substrates and lithium compounds.[169,170] Minimum temperatures about 300–350 °C were required to wet with contact angle below critical value (defined as 90° between liquid droplet and substrate) stainless steel, molybdenum, tungsten, tantalum, and TZM and lithium was found to wet its oxide, nitride, and carbonate compounds at lower temperatures, a result that implies that the unavoidable passivation of the lithium layers will not worsen the wetting. During flowing operation, however, spreading and homogeneity in the lithium stream may demand a smaller contact angle of the liquid possibly driven by higher temperatures, although experimentation also demonstrated that flowing liquid lithium temperature may be reduced once the metal wets. Additionally, mirror finishing (by means of fine grade, surface polishing) of surfaces has shown that lithium may wet a surface as soon as it melts. Other successful methods to improve wetting are glow discharge conditioning of the substrates and the evaporation on these substrates of a thin lithium film.[169] Local and efficient heating of the liquid lithium surface in contact with the filled plate by an electron gun seems also interesting to improve the wetting as has been already showed in tin filled CPS.[171] Preventing lithium flow to undesired locations in the machine can be

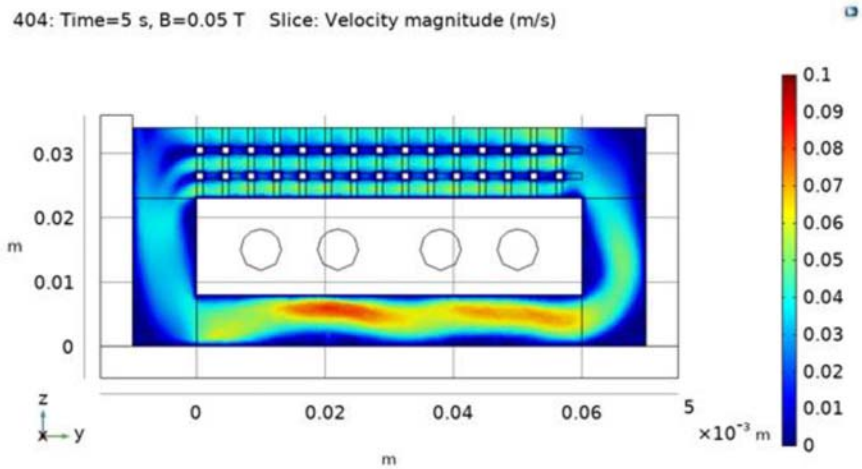

FIG. 10. Distribution of lithium velocities on the 3D modular structure obtained with COMSOL. Values were in the range of 2–5 cm/s with extensive cross flow and distribution in both perpendicular directions of the surface.

achieved through local surface modification. By controlling the roughness of substrate materials, the temperature at which liquids will wet those materials can be adjusted. At Illinois, a laser structuring process was developed that induces microstructure and nanostructure formation on the surfaces of stainless steel and molybdenum. On both substrates, structuring of the surface was observed to produce a hydrophobic effect that increased the wetting temperature by 80° C.[172]

Following these laboratory scale lessons, LiMIT trenched geometry operation in EAST showed good wetting behavior that also assured surface stability, having a global wetting pattern favored by the texturized surfaced when compared to FLiLi. Nevertheless, the total, homogeneous distribution of the lithium layer along the plate surface remained challenging. About 87% of the plate[104,152] seemed to have had lithium entering and flowing along the trenches, improving the global wettability respect to the flat FLiLi plate (around 70%). Some areas remained absent of lithium coverage, probably due to deficient lithium spreading from the distributor. This finding seemed to be originated by the clogging of a part of the distribution holes due to the possible formation of solid lithium impurities due to passivation. Consequently, to improve the lithium spreading and thus enabling the real full coverage of the PFC with lithium, the design of the distributor will need to be improved in order to avoid such problems. Such activities are being developed at CPMI to design, fabricate and test different distributor designs containing texturized surfaces aimed to enhance the spreading of lithium over all the plate channels by using different

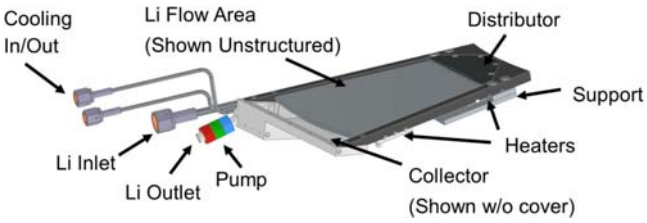

FIG. 11. Preliminary designs of a LiMIT plate (with an approximated plasma exposed area of 500 cm^2) and accessory elements for testing in ST 40 tokamak. The installation of the PFC solution, which will cover approximately 1/16 of the lower divertor outer tiles area, is programmed to start in late 2021.

two-dimensional geometries as rectangular, rhomboidal, or cylindrical posts.

E. Pumping of liquid lithium/lithium hydride (Li-LiH) and hydrogenic extraction

As the liquid metal is a conductive fluid, the j × B Lorentz force can be used to induce the movement by providing a current in the liquid bulk that would also interact with the magnetic field of the fusion device and/or other possible external fields. Through this electromagnetic (EM) pumping scheme, the magnetic drag of the flow on the liquid lithium divertor elements is found to scale as

$$\Delta p_{MHD} \sim \sigma \cdot v \cdot L \cdot B^2, \tag{20}$$

where σ is the conductivity of the liquid metal, v its average velocity on the element, L is its equivalent (in a hydraulic sense) length, and B the toroidal magnetic field. The total power to produce this flow will be strongly reduced in slow-medium flow PFC solutions when compared to fast flow ones as the velocity is reduced by two orders of magnitude (∼cm/s vs ∼m/s average velocity ranges). As the formulation of the general MHD pumping scheme shows, the absolutely dominant component of the total MHD drag will be the contribution of the piping system rather than the plates/distributor or reservoir as the involved velocity will be clearly larger.[98] Simple extrapolations of this formulation considering a tentative DEMO-like prototype with major radius of 5 m, minor radius of 3 m, total equivalent length of the piping system (that dominates the total MHD drag) of 200 m (pretty large and conservative value) with a total lithium mass flow of 0.5 kg/s (value that surpasses the necessary rate expressed later in this chapter for continuous hydrogenic extraction in the power plant prototype) flowing through pipes with 5 mm radius and 1 mm wall thickness gives a power requirement (usually calculated multiplying the MHD drag by the total lithium volumetric flow) in the range of 150–200 kW, which is clearly an insignificant fraction of the envisioned GW range power output.[110,141] Additionally, lithium-driven low-recycling regimes and its confinement enhancement will minimize the requirements in magnetic field of the fusion device for a given performance. Hence, use of liquid lithium will pose an advantage in terms of MHD

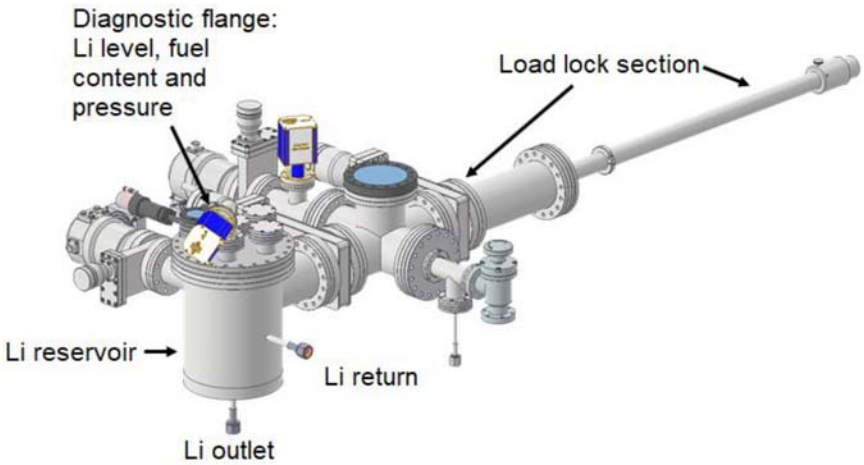

FIG. 12. External elements∗ of the loop being designed for the testing of the LiMIT divertor module in ST40 tokamak. *EM pump, flow meters, valves, and lithium pumping lines to the machine are not shown for simplicity.

drag and magnetic pumping requirements respect to any other liquid metal option such as tin, tin-lithium, or galinstan alloys where low recycling would not be feasible.

More important than the energy required to pump the lithium is the amount of deuterium/tritium (D/T) trapped in the lithium and how one can extract it. The composition and nature of the lithium stream to be pumped will be progressively and unavoidably changed. The most important compound that will be formed is lithium hydride[173] that precipitates as solid salt in the liquid bulk when solubility limit (1% molar ratio approximately at 400 °C[174]) is overpassed. Solubility of other lithium compounds as oxides, hydroxide, or nitride in liquid lithium is very low as well, then they will also contribute to the formation of a slurry stream with different physical properties that will affect to its circulation as a fluid. Those impurities pose a much lower value (orders of magnitude) of electrical conductivity,[175] a property that is essential and needs to be high to induce efficient and effective EM pumping. Second, the solid impurity particles will settle to the bottom of the PFCs/pipes/elements of the loop, thus producing possible clogging/blocking problems that will affect the overall flow capabilities. Furthermore, beyond a given threshold, the system might be incapable of pumping such solid impurities at high concentration, thus needing extra, auxiliary systems for solid filtration and separation. Therefore, experiments have been performed on Li/LiH mixtures to ensure rapid extraction while avoiding LiH buildup.

Tritium self-sufficiency is desired in fusion pilot plants due to the extremely limited external inventories of this isotope worldwide. From a commercial standpoint, this isotope is only produced in Canada Deuterium Uranium (CANDU) fission reactors at approximate rates of 130 g/year[176] (although some production exists related to defense applications), requiring a complicated separation process to extract it from heavy water. Only two facilities worldwide are operative as potential suppliers. For such reasons, tritium is also very expensive (with prices even beyond 100 000 $/gram).[177] Therefore, any real fusion reactor will need to breed its own tritium fuel by using a lithium-containing compound that will be transmuted into tritium when interacting with the fusion neutrons in the reactor breeding blanket units.

Tritium is radioactive by means of negative β emission with a lifetime of 12.32 years, so its storage in the reactor must be limited and it needs to be fully used and minimally lost in the components and auxiliary systems. The administrative limits for the in-vessel tritium accumulation at ITER are determined as 700 g of total moveable inventory.[178] The tritium breeding rate in the blanket modules is also affected by a low supply margin that limits the total tritium that can be lost from the fuel cycle to approximately 0.1% of the fueled tritium.[179] For reactors based on the traditional ITER-DEMO approach (big size, high recycling solid walls, with fusion reactions happening only in the very plasma core), the maximum burning efficiency in the reactor will be around 1%–1.5%.[180] The remaining unused tritium will need to be removed from the gas exhaust system and in-vessel components where hydrogen isotopes may be unavoidably retained by means of different mechanisms (implantation, bubble trapping, codeposition, bulk diffusion). To carry out such crucial actions, external tritium processing plants are being designed with a duty cycle that needs to be carefully synchronized with the tritium inventory requirements in PFCs and pumping system. On the other hand, the higher performance of the low-recycling regime is expected to increase the burning efficiency[57,71,74] in the reactor as temperature and triple product may be extended over much larger plasma volumes. Such more efficient burning will directly relax the recovery requirements of any tritium processing plant. However, in this sense, the most extraordinary possibility of a reactor solution based on flowing liquid lithium PFCs is that efficient trapping of hydrogen isotopes in flowing lithium components can uniquely offer a possibility for its control, mobilization, continuous recovery, and reinjection into the plasma in real time, also adding the advantage of maintaining the lithium plasma facing surface fresh and clean from impurities and thus achieving a stationary operation throughout a full liquid lithium loop as proposed by Ono *et al.*[141]

Within this scheme, the extraction of hydrogen isotopes and impurities absorbed by the floating lithium layer would be carried out continuously by means of lithium/lithium hydride distillation,[175] a technology whose development is underway at CPMI with the rest of mentioned liquid loop technologies. The concept envisions a stationary loop with mass flow of liquid lithium in the range of hundreds of g/s circulating at moderate velocities (cm/s) as capable to be sufficient to absorb the hydrogenic flux from plasma, compensate evaporation/erosion, and conform a steady state liquid lithium absorptive boundary in the divertor. To achieve such solution, development of suitable technologies at reactor relevant scale needs to be accomplished to

enable the critical demonstration of tritium real time recovery and inventory limit control in a lithium PFC configuration, showing that the massive absorption of tritium in flowing liquid lithium components is not a major drawback, and on the contrary, provides the singular possibility to be used to recuperate and refuel the tritium in real time.

It is important to remember that tritium recovery and concomitant fuel self-sufficiency are conditions that any future reactor will need to accomplish regardless of its plasma-facing material choice. Such continuous, real time recovery possibility does not exist for configurations based on solid traditional materials such as tungsten. Even considering the small short-term hydrogenic retention of tungsten, the long-term diffusion and permeation of H-isotopes will produce the much larger long-term retention (even up to 10% atomic ratio) of radioactive fuel[176] also implying a more difficult outgassing from the solid walls at much higher temperatures. Real concerns about the efficiency of such thermal recovery processes in considered solid materials have been found with investigations showing a difficult time (up to 1 month treatments) and energy-consuming procedure (with desorption needing temperature rises beyond 1000 °C for significant release) for the recovery of tritium from solid divertor PFC and codeposits.[181–183] Additionally, it will be impossible to outgas, handle and refuel this in-vessel inventory during reactor operation as the trapped gaseous fuel cannot be mobilized out of the vessel and consequently will be released directly into the confined plasma, dramatically affecting its performance.

The implementation of the auxiliary tritium recovery technologies from lithium will not require a major additional requirement as a tritium gas processing plant will be needed anyway to process the gaseous tritium removed from the pumping system and PFCs. Likewise, the temperature requirements for the total outgassing of tritium from lithium are less important (lithium hydride totally decompose below 700 °C and hydrogen release peaks at lower temperatures) when compared to the removal from solid materials, and the liquid loop may directly transform the tritium inventory into a moveable one, prone to be processed faster and more efficiently. However, to approach the proposed reactor scenario in an integrated way, it is necessary to consider that the evaporation/erosion and the coupled migration of lithium out of the flowing divertor may be a possible source of uncontrolled Li-fuel codeposition in remote hidden or plasma-shadowed metallic, solid first wall regions. Such issue might jeopardize the tritium inventory and safe operation of the reactor in the similar way compared to the current concerns with solid materials, but also aggravated by the high undesired hydrogenic retention in such regions out of the lithium loop. Concerning this issue, however, experiments in both linear plasma and gas exposure[184,185] have inferred that the operation with a hot tungsten first-wall ($T \geq 400\,°C$) would form a very thin (sub-micron size) lithium film where lithium hydride is unstable and thus the long-term fuel retention may be similar when compared to pure tungsten. Consequently, if most of the particle flux is concentrated and absorbed in the flowing liquid lithium divertor, the influence of this remote tritium codeposition would be very minimal and would not significantly aggravate the problem with respect to a pure W walled reactor.

After the exposure to the plasma, the flowing lithium stream will contain a mix of Li and LiH/LiD/LiT as well as different impurities unavoidably originated by gettering/passivation processes (Li compounds as oxides, carbonate, nitride, etc.) and/or corrosion. Beyond a solubility limit (determined by temperature[174]) of the hydride in the liquid, the formation and precipitation of solid hydride take place with the separation of two phases: the alpha (α) phase where liquid lithium is in equilibrium with a minor temperature-dependent fraction of dissolved hydrogen and the beta (β) phase where stoichiometric hydride is present. Within this system, dilution or transition of hydrogen from β to α phase, stability of hydride and hydrogenic desorption from both phases depends on thermodynamic equilibrium determined by Sievert's law,[173] whose main parameters are external hydrogen pressure and volumetric temperature, with the thermodynamic decomposition of pure hydride taking place at a temperature around 690 °C.

As commented, the fuel recuperation is envisioned to be carried out by using a thermal method for the distillation of lithium–lithium hydride mixtures.[175] First investigations regarding hydrogen outgassing from this kind of mixtures showed that the hydrogen recovery rate strongly depends on the hydrogen fraction and Li-LiH proportion of the mixtures,[186] indicating that higher LiH fractions determine a larger desorption flux of hydrogen. Consequently, the main idea is to treat LiH enriched streams (previously separated from liquid lithium by centrifugal, filters and/or cold trap technologies) that can be heated up to 700 °C in order to recuperate hydrogen at a rate that may compensate the tritium losses expected in future reactors. For such purpose, the first generation of Li/LiH distillation column was developed, assembled, and operated at UIUC.[187] The distillation apparatus basically consists of a base bucket where the Li-LiH mixture is heated by using an inductive system up to 700 °C and two modular condensation stages (at temperature of 350 and 320 °C) for pure Li recovery. The outgassed hydrogen escapes from the top of the column through a small sniffer tube that directs it to an analysis chamber. The temperature and hydrogen evolution with time is recorded with thermocouples and differentially pumped mass spectrometry (residual gas analyzer, RGA). Absolute calibration for hydrogen gas correlates the obtained RGA signals to total recovery rates of hydrogen. A cross-sectional sketch of the column and its internal elements after testing, showing almost total lithium removal in the base bucket and Li presence mostly concentrated on the first condensation stage, can be visualized in Fig. 13.

In Fig. 14, the evolution of desorbed hydrogen flux in the column depending on temperature[188] is shown where three different regions are visible (marked in roman numbers), characterized by a different hydrogen outgassing rate and associated temperature. The first one appears at lower temperature and probably corresponds to hydrogen outgassing associated with impurity depletion (mainly residual water). The second one shows a peaked hydrogen depletion rate (approximately 4.3×10^{22} H_2 molecules/m^2 s) at a temperature close to 700 °C. It is the highest desorption rate that probably combines hydrogen desorption from alpha phase as a result of $\beta \rightarrow \alpha$ transitions but also including outgassing from pure hydride regions of the mixture. The operation of the column at this point would need only a distillation area of 0.35 m^2 to compensate the fuel losses previously inferred in an ITER-like lithium walled reactor (around 3×10^{22} atoms/s [**571**]). The experimental values (size of the heated bucket section of 62 cm^2 and necessary heating power of 2640 W) allow to calculate the necessary power density that may be used to extrapolate the necessary power to heat up the previously obtained, reactor-scale distillation

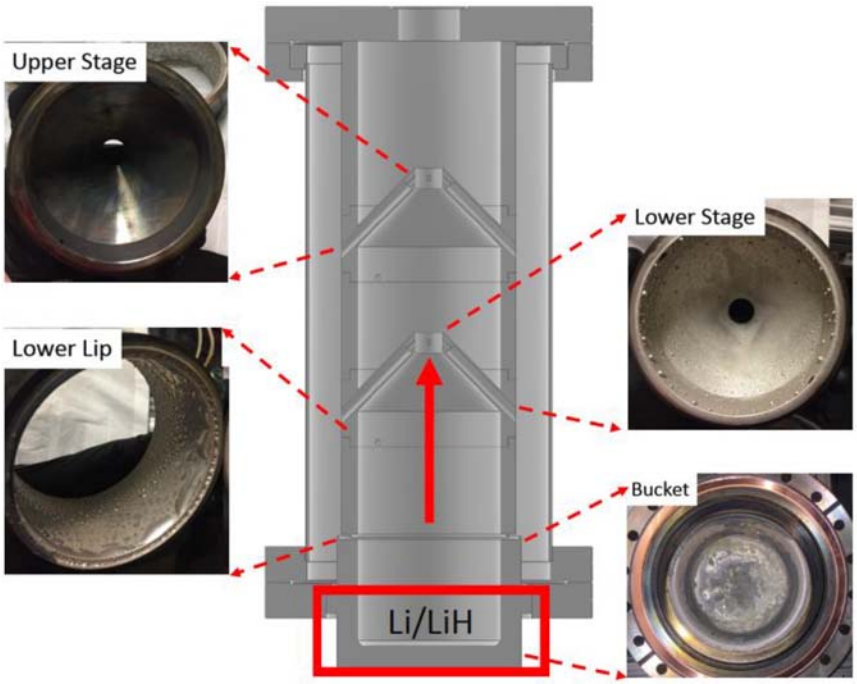

FIG. 13. Cross-sectional sketch showing the column and their modular elements and pictures showing the aspect of the internal elements after Li/LiH distillation experimentation.

area, giving a value around 150–200 kW. The third region shows a more constant outgassing rate that may be mainly produced due to decomposition of pure hydride after the total depletion of hydrogen from the alpha phase. It corresponds to a hydrogen outgassing rate of 1.72×10^{22} H_2 molecules/m^2 s. The system operating at this point would need an active area around 0.87 m^2 and power heating around 500 kW. This heating power for the Li-LiH distillation will unavoidably impact the economy of the plant with an additional cost. However, the operation scenarios here inferred would suppose (in the worst case) an affordable energetic cost that will be only a 1% fraction

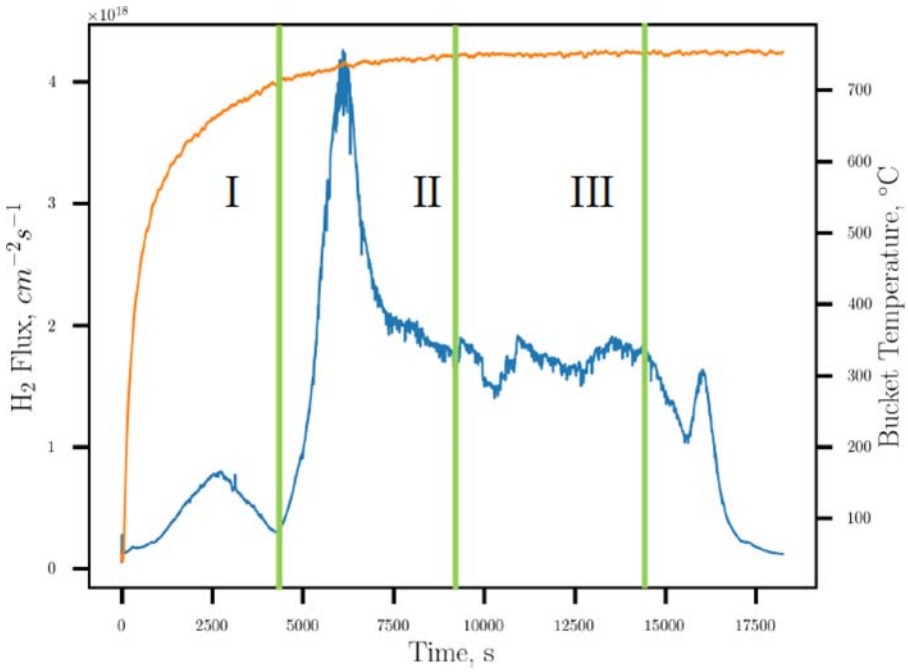

FIG. 14. Time evolution of hydrogen outgassing from Li/LiH registered during a test performed in the distillation column.

of the auxiliary heating power of the plasma (500 kW power for the Li-LiH distillation within a ITER-like reactor, a device that will use 50 MW of heating power for 500 MW of total generated fusion power).

The incorporation of the distillation technology into a full liquid lithium loop at pilot plant scale will require the coupling of this technology with centrifugation and/or filtration capable to concentrate a hydride-rich slurry that will maximize the tritium recovery. In this way, this LiH enriched stream can be diverted from the loop toward the distillation column, and a fraction rich in liquid lithium may be purified in yttrium filters and/or surface cold traps before returning to the flowing liquid lithium divertor. For such purpose, the incorporation and coupling of the distillation technology with innovative ideas that may take advantage of the magnetic field of the reactor to induce centrifugal separation[189] may be contemplated. A schematic of the processes, considered for a 3 GW-thermal power plant where tritium recovery needs to balance the fueling requirement (1% burn), is shown in Fig. 15.

Based on such scheme proposed by Ono *et al.*, using the measured recovery rates and incorporating mass flow global balances, extrapolations of the total tritium inventory within the loop scheme have been performed. Complete details of such calculations will be published in a specific and separate article being prepared. Generally, the mass flow formulation considers centrifugation to preconcentrate LiT up to 33% and 0.5 g/s tritium fueling rate, resulting in 3.5 m^2 of distillation area and 12 MW heating power that would be needed for the separation process. Additionally, assuming a machine with $R = 5$, divertor area of 150 m^2, total lithium flow rate ~0.4 kg/s circulating with 2 mm thickness, and lithium average velocity of 1 cm/s (based on a 3 m inner radius geometry), the total lithium content of the loop will be around 55 L with a tritium inventory that will eventually depend on the efficiencies of the centrifugal and distillation separation processes. Scans performed varying such parameters show that it would not exceed 1.2 kg at plausible efficiency values around 60%. Such quantity is within the order of magnitude of the ITER tritium inventory and considerably small when compared to the inventories expected to be necessary for the D-T operation start-up in conventional DEMO scenarios. In them, the low tritium burnup (1%) expected within a high Z solid wall-3 GW power reactor will determine a derived scenario dominated by tritium fueling/exhaustion that would require higher tritium inventories (up to 21 kg) to assure a continuous operation as processing and reserve time in the range of 6 h are expected.[181,190] To ameliorate such constraints, research priorities in the direction of improving the burnup fraction and tritium processing technology have been claimed.[191,192] In this sense, lithium low recycling ideas envision a burning efficiency much higher, a consequence of the higher temperate and enhanced confinement over a more extended confined volume and larger beta,[43,57,70] thus more efficiently using the fuel and minimizing the related tritium requirements to start-up the reactor. Continuous recovery with distillation is a faster and more efficient option that can greatly decrease processing and reserve time as well as the derived start-up inventory. These are all arguments about how the liquid lithium configuration can also help in these nontrivial questions that are common regardless of the considered fusion reactor configuration.

F. Helium ash exhaust

Continuous and efficient exhaustion of helium (He) out of the reactor will be necessary to maintain steady state conditions in the

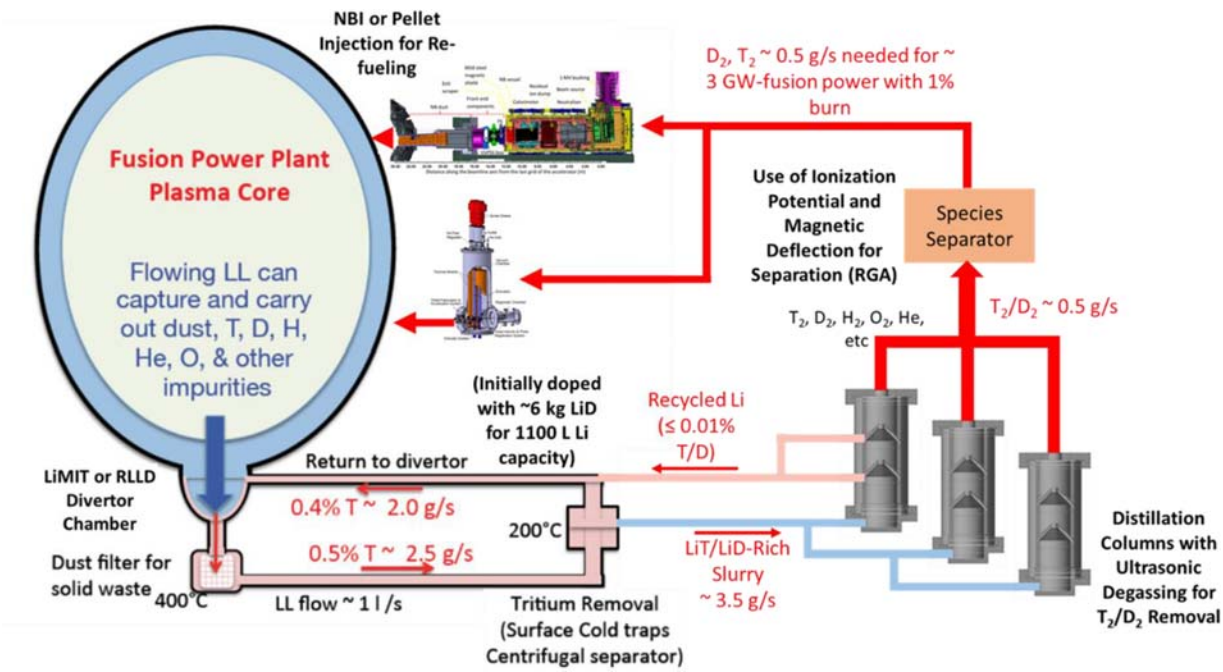

FIG. 15. Integration of the distillation column technology into a stationary loop necessary for a flowing liquid lithium configured fusion reactor envisioned to produce 3 GW-thermal power with 1% tritium burnup fraction. [Figure adapted with permission from Ono *et al.*, Nucl. Fusion **57**, 116056 (2017). Copyright 2017 International Atomic Energy Agency.[141]]

plasma core in any proposed prototype. Projected pumping requirements appear important at reactor while helium atoms are hard to pump by cryopanels due to their low evaporation point and chemical inert nature. However, the gettering of helium by the wall materials of the reactor would unavoidably help in this task, being the role of this codeposition claimed as crucial for any power plant scenario.[193] Trapping of helium in tungsten is limited in time due to material saturation at very low retention levels below 10^{20} atoms/m^2 at temperatures up to 700 °C.[194] Higher temperatures moderately increase the retention but aggravate the formation of fuzz tendrils and bubbles that would be irreversible and considerably damage the tungsten tiles. While a possible toroidal pumping duct idea[71] to externally pump the helium ashes in a flowing lithium configuration has been proposed; on the other hand, conceptual studies postulated that flowing liquid metal components may be also advantageous for inducing continuous helium trapping on lithium driven by strong liquid convection.[195] Experimentally, even in slow-flowing systems, He has been shown to be retained in liquid lithium, potentially at impurity boundaries. This reduction in the helium recycling was observed at Illinois in its flowing lithium retention experiment (FLiRE) facility with liquid lithium streams.[196–198] It has also been shown in tests performed by Hirooka *et al.*, employing moving liquid lithium coatings.[199] Although the proper scaling of such benefits at relevant tokamak scenario needs to be demonstrated, any contribution in helium pumping by the flowing liquid lithium solution will not be limited in time. When integrated and accommodated with the suitable pumping systems in an optimized divertor, the configuration might help in relaxing the external pumping requirements necessary for helium removal at steady state operation.

G. Material compatibility, temperature limitations, and safety

Liquid lithium is known to corrode many commonly used metals such as copper or aluminum and leaches chromium from steel.[200–202] This situation may be aggravated in flowing configurations due to the induced erosion/abrasion on the solid surfaces. Therefore, these corrosion considerations are important when approached the proposed reactor scenario in an integrated way. Materials such as tungsten, molybdenum, and 316 stainless steel have shown good or fair compatibility with liquid lithium at laboratory time scales, thus possibly offering plausible options as PFC substrates. Nonetheless, the utilization of such options for the structural materials might be inappropriate considering neutron activation, cost, or other engineering issues. To study the compatibility with low activation structural candidates and to confirm the good perspectives with the PFC substrates at longer, reactor-relevant time scales, specific experimental efforts to determine unknown lithium corrosion patterns in unexplored materials and alloys are necessary. In a collaboration with General Fusion company, CPMI has recently developed a novel dynamic lithium corrosion testbed where eight samples of four different materials (privately and confidentially determined by the company) are immersed and spun in liquid lithium during timescales of 100 h per run. After the testing, the mechanical properties and the chemical changes induced on the sample surfaces are studied by tensile stress testing and surface characterization techniques. First results have been satisfactory in the question of detecting changes in the material surfaces associated with lithium deposition and in their mechanical properties, showing the usefulness of the facility in discerning what material options seem incompatible with lithium and which candidates may be initially considered. However, the 100 h laboratory timescale is not representative when compared to the contact time between the molten metal and the structural materials that are envisioned for a fusion reactor. The duration of contact of reactor materials with liquid metals is clearly longer by a few orders of magnitude. Therefore, for the encouraging materials screened, much longer timescale experimental validations would be the next step within the selection of structural candidates in order to validate the potential substrates that may be compatible with lithium within the temporal window in which an eventual reactor will operate.

For a low-recycling PFC solution to work, the lithium must retain the deuterium and tritium except during off-normal events. This limits the exit temperature of the flowing lithium to the 400–450 °C range. Therefore, if the lithium is introduced at 200 °C, the average temperature of the divertor structure would be around 300 °C. Thermodynamic power extraction efficiency through a boiling water cycle (T-Cold = 100 °C) when T-Hot is only 300 °C is at most 35%, and likely lower. This does not preclude efficient electrical generation since 80% of the fusion energy is carried by neutrons, and the blanket can operate at much higher temperatures than the divertor plate system. The inside of a fusion device is a vacuum, so some degree of thermal isolation between the divertor and the blanket is feasible. After all, any fusion device would have to be superconducting, and therefore keep the magnetic field coils at cryogenic temperatures. If it is possible to maintain a temperature gradient between the blanket and the coils, it is possible to maintain a much smaller temperature gradient between the blanket and the divertor.

To prevent risks associated with water, cooling systems will probably need to be gas based. Although He cooling capabilities are normally considered below water cooling systems, advanced He cooling concepts with very high heat transfer rates[203,204] are being developed. The exploration and optimization of these innovative technologies appear paramount to open the possibility of achieving divertor relevant power exhaust capabilities in the range of the ITER divertor needs ($\geq$10 MW/m^2) using He cooling instead of more conventional liquid water systems. Of course, these technologies will need to demonstrate the projected performance in experimental tests up to the relevant technological readiness level (TRL) corresponding to potential reactor installations. In this sense, in a collaboration with Micro Cooling, Inc., advanced prototypes of such cooling systems will be tested, coupled to a LiMIT plate. The configuration basis is the use of texturized plates with trench dimensions and spacing in the submillimeter range ($\sim$100 μm) and innovative manufacturing techniques to configure an actively cooled heat sink that will maximize the contact area of the coolant (helium) flowing internally through microchannels, thus improving the extraction of the heat flux deposited on the lithium surface. Modeling and simulation works have pointed to potential heat handling capabilities at the level of 20 MW/m^2 without raising the lithium temperature beyond 400 °C. Remaining below this temperature limit on the flowing liquid lithium components is essential to achieve a low recycling regime, assuring a stationary absorptive boundary also compatible with the limited evaporation and concomitant edge cooling and core dilution. The planned experimentation at the laboratory scale is focused on determining if the proposed geometry is compatible with proper wetting and flow pattern and will experimentally determine the

real heat handling capabilities of the design, thus checking if the power exhaust performance predicted by the simulations (ITER divertor relevant) can be achieved.

In terms of safety, molten sodium has been used commercially to cool fission power plants. The key is that the molten metal is not under high pressure. Every slow to medium flow system attempting to operate in the low-recycling regime has a free surface of lithium exposed to vacuum. Pressure relief is therefore automatic and the maximum pressures in the lithium pumping systems will need to be low. Finally, related to diagnostics and other in-vessel components of the reactor, it is necessary to take in mind that even local and temporal temperature excursions on the liquid lithium surfaces might produce strong evaporation of liquid Li that might directly affect those elements. This may result especially important for the case of windows, copper gaskets, brazing joints, and so on. Clever design/engineering of their emplacement, as well as proper utilization/replacement strategies, will be essential to assure their long-term compatibility within the lithium divertor environment.

VI. CONCLUSIONS

Lithium's ability to reduce recycling, increase energy confinement, and mitigate anomalous heat transport theoretically opens a different and promising alternative route to fusion energy that might accelerate the development of commercially viable fusion power through more affordable, smaller size, yet higher-power-density devices. Using this approach, electricity costs might be greatly reduced if all the technological constraints can be resolved. The scientific arguments, observed physics, and phenomenology of the lithium low-recycling framework in this narrative offer the possibility of favoring the thermonuclear efficiency of the reactor, also helping in technological hurdles related to plasma surface interaction and combating many of the physical/technological challenges that any fusion reactor will face. Lithium as PFC has shown to dramatically improve plasma stabilization and energy confinement, ameliorating the problems of turbulence and instabilities.

Therefore, the utilization of a flowing liquid lithium divertor is a technological scenario that may open the pathway to a less expensive approach to fusion reactors. More compact, smaller reactors will need significant beta and confinement improvements greatly beyond the ITER scaling and require the finding of operable, high confinement, high-performance scenarios. In this sense, the flowing lithium divertor approach is an incipient technology that attempts to enhance the fusion performance by means of a continuous and notable reduction of neutral recycling on the divertor plasma boundary, being an advanced plasma performance scenario that needs to be corroborated at reactor-relevant scale. With this developing technology operating, the continuous low recycling operation may be envisioned and, if the theoretical regime is corroborated, the fusion reactor approach would be based on smaller machine size, increasing energy confinement time, higher edge plasma temperature, burning fraction, and volume as well as enhanced fusion power and Q gain factor. Additionally, a high-temperature edge configuration driven by the low recycling may diminish the power exhaust handling constraints and the liquid nature of the divertor elements may help in increasing the PFC lifetime and the power plant availability, utterly decreasing the PFC replacement costs. It even poses a method to recover tritium in real time and relax its inventory start-up requirements. All these technical considerations appear fundamental to finally establish the feasibility of a future fusion power plant that may be economically competitive. These benefits of flowing lithium can be combined with higher magnetic field configurations and perhaps spherical geometries (high elongation, low aspect ratio) and/or advanced divertor magnetic structures/configurations such as double null, super X, or snowflake.

To accomplish this approach, new, emergent, and developing technologies need to be demonstrated and developed to the maximum technological readiness level corresponding to a continuous reactor operation. Cutting-edge technological prototypes have been successfully tested at laboratory and/or mid-size tokamak scale showing potential and encouraging solutions to most objections claimed against lithium utilization. Work at the University of Illinois at Urbana-Champaign is pioneering and leading endeavors with national, international, and commercial partners to demonstrate and meet all the critical major engineering/technological challenges at higher TRL and move to the reactor scale. This paper has reviewed the work on many of these aspects and shows that flowing liquid-lithium technology is ready for the next step.

One possible next step could be to experimentally explore the low-recycling theory, trying to investigate the accessibility to such regimes and the enhanced performance at the largest existent relevant tokamak (JET), by means of a LiMIT or FLiLi style flowing lithium divertor PFC solution. Such kinds of divertor plates are currently being designed to be tested on the ST-40 tokamak divertor within a full lithium loop configuration. To add such a device to JET as a final lithium campaign for the machine follows the same technological and scientific roadmap of previous magnetic devices: the HT-7 and TFTR tokamaks. These final runs achieved their highest levels of performance. The scientific arguments and the accumulated research experience exposed in this review provide the necessary motivation to aggressively advance the engineering/technology necessary to enable flowing lithium-enhanced magnetic fusion devices.

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DATA AVAILABILITY

The data that support the findings of this study are available from Tokamak Energy Ltd., General Fusion, and Micro Cooling, Inc. Restrictions apply to the availability of these data, which were used under license for this study. Data are available from the authors upon reasonable request and with the permission of Tokamak Energy Ltd., General Fusion, and Micro Cooling, Inc.

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