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LITHIUM AS PLASMA FACING COMPONENT FOR MAGNETIC FUSION RESEARCH

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ABSTRACT

The use of lithium in magnetic fusion confinement experiments started in the 1990's in order to improve tokamak plasma performance as a low-recycling plasma-facing component (PFC). Lithium is the lightest alkali metal and it is highly chemically reactive with relevant ion species in fusion plasmas including hydrogen, deuterium, tritium, carbon, and oxygen. Because of the reactive properties, lithium can provide strong pumping for those ions. It was indeed a spectacular success in TFTR where a very small amount (~ 0.02 gram) of lithium coating of the PFCs resulted in the fusion power output to improve by nearly a factor of two. The plasma confinement also improved by a factor of two. This success was attributed to the reduced recycling of cold gas surrounding the fusion plasma due to highly reactive lithium on the wall. The plasma confinement and performance improvements have since been confirmed in a large number of fusion devices with various magnetic configurations including CDX-U/LTX (US), CPD (Japan), HT-7 (China), EAST (China), FTU (Italy), NSTX (US), T-10, T-11M (Russia), TJ-II (Spain), and RFX (Italy). Additionally, lithium was shown to broaden the plasma pressure profile in NSTX, which is advantageous in achieving high performance H-mode operation for tokamak reactors. It is also noted that even with significant applications (up to 1,000 grams in NSTX) of lithium on PFCs, very little contamination (< 0.1%) of lithium fraction in main fusion plasma core was observed even during high confinement modes. The lithium therefore appears to be a highly desirable material to be used as a plasma PFC material from the magnetic fusion plasma performance and operational point of view. An exciting development in recent years is the growing realization of lithium as a potential solution to solve the exceptionally challenging need to handle the fusion reactor divertor heat flux, which could reach 60 MW/m^2. By placing the liquid lithium (LL) surface in the path of the main divertor heat flux (divertor strike point), the lithium is evaporated from the surface. The evaporated lithium is quickly ionized by the plasma and the ionized lithium ions can provide a strongly radiative layer of plasma ("radiative mantle"), thus could significantly reduce the heat flux to the divertor strike point surfaces, thus protecting the divertor surface. The protective effects of LL have been observed in many experiments and test stands. As a possible reactor divertor candidate, a closed LL divertor system is described. Finally, it is noted that the lithium applications as a PFC can be quite flexible and broad. The lithium application should be quite compatible with various divertor configurations, and it can be also applied to protecting the presently envisioned tungsten based solid PFC surfaces such as the ones for ITER. Lithium based PFCs therefore have the exciting prospect of providing a cost effective

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flexible means to improve the fusion reactor performance, while providing a practical solution to the highly challenging divertor heat handling issue confronting the steady-state magnetic fusion reactors.

Keywords: Magnetic fusion, lithium, tokamaks and spherical tokamaks, plasma-wall interactions

I. INTRODUCTION

Lithium is the lightest metal (mass number of ~ 7), the least dense solid element (~0.53 g / cm^3), and like all alkali metals, lithium is highly chemically reactive and flammable. Lithium has a relatively low melting temperature of 180.54 °C. Interestingly, this readily melting light alkali metal may provide an attractive solution for the highly challenging plasma facing component (PFC) problem for the magnetic fusion energy development. The on-going magnetic fusion research conducted worldwide aims to create a "sun" on the Earth for a clean and safe long-term energy solution for the mankind [1]. In order to generate fusion power, as occurring in the hot interior of the sun, the plasma (highly ionized form of high temperature gas) consists of deuterium and tritium (D-T) ions must be heated to the fusion reaction temperature typically exceeding 100 million °C. In order to contain such high temperature plasmas, a number of magnetic bottles have been conceived and tested. The present day main line magnetic bottle configurations are tokamaks and stellarators. A tokamak fusion reactor prototype facility ITER is being built in France with participations from seven Members of international ITER project [2]. The PFCs are the material surfaces facing the plasma, which are subjected to the heat, particles, and neutron bombardment from the burning plasmas. Since the fundamental nature and performance parameters of the D-T burning fusion plasmas are essentially the same regardless of the magnetic bottle types, one could conclude that the PFCs for magnetic D-T fusion reactors of all types will be subjected to similar harsh conditions of intense heat, particles, and neutron bombardments. Particularly severe PFC conditions occur at the so-called divertor strike points as depicted in Figure 1 for a tokamak configuration. The colored region depicts the plasma core or the main region of magnetic confinement. The region outside is the open magnetic field line region, where the plasma heat and particles escaping from the core flow along the magnetic field lines (as depicted by the arrows) strike the material wall in the divertor region. Since the heat flows quite rapidly along the magnetic field lines, a significant fraction of the escaping heat could end up on the surfaces near the divertor strike points. The divertor strike point is typically defined by the last closed flux surface intersecting the material wall. For a tokamak, it is a ring shaped region with a rather narrow radial width of typically 1 - 10 mm, depending on the divertor configuration and device parameters. In this article, we will focus the divertor discussion to the tokamak configuration due to its geometric simplicity (i.e. axi-symmetry) and maturity of understanding. For ITER [2], a tokamak reactor with 0.5 GW of fusion power, the expected maximum divertor heat load is 10 MW / m^2. However, for a typical 1 GW electric fusion power plant producing ~ 3 GW of fusion power (or 6 x ITER), since the divertor heat load tends to go up with the fusion power for a give device size (and a typical fusion reactor size is generally expected to be similar in size to ITER), the expected heat load could be as high as 6 x that of ITER or 60 MW/m^2 for a similar divertor geometry. The tungsten-based divertor for

ITER is designed to take the projected 5 - 10 MW/m^2 of divertor heat load, but this type of solid based design is not likely to handle the much higher reactor heat load expected in the steady-state fusion reactors [3]. While tungsten has been identified as the most attractive solid divertor material, even for the ITER level heat load, many challenges including surface cracking and other deleterious modifications of the surfaces by the plasma must be overcome to develop robust plasma facing components (PFCs) [4].

In addition, for tokamaks, the edge localized modes (ELMs) (periodically bursting modes expelling typically a few % of plasma stored energy) could deposit significantly higher transient heat loads onto the divertor surfaces, further exacerbate the divertor surface deterioration. It is therefore prudent to develop an alternative divertor solution for future reactors. As an alternative divertor PFC material, lithium has received a special attention in the recent years due to a number of attractive properties of lithium-based PFCs:

1 Improved plasma performance - Lithium application onto the plasma facing components (PFCs) has been observed to improve magnetic fusion plasma performance in a number of fusion experiments which has been generally attributed to reduced "recycling". Recycling is the churning action of neutral particles between the hotter fusion plasma edge and the colder plasma facing first wall surfaces resulting in cooling of the plasma edge. The escaping plasma ions become cold neutralized particles when they reach the material wall, which are then released back into the plasma. There is mounting evidence that reducing recycling could lead to a significant improvement in fusion plasma performance. Since lithium is quite chemically reactive, it can readily combine with hydrogen (H), deuterium (D), tritium (T) and other impurities (oxygen, carbon etc.) and form more stable lithium compounds. By coating the PFCs with lithium, the lithium chemical reaction captures those neutral particles, which increases the pumping action and thereby reducing the plasma edge recycling. With reduced recycling, the edge plasma density decreases and the edge electron temperature increases, leading to a significantly lower edge plasma collisionality. The reduced recycling and collisionality results in improved fusion plasma confinement and plasma performance, as observed in many fusion experiments world-wide. This is described in Sec. II. and Sec. III. The improved plasma confinement and performance could reduce the size of a fusion reactor and its cost.

2 No obvious adverse effects – According to world-wide lithium experiments, introduction of lithium generally has not caused adverse effects on the plasma performance, including excessive plasma dilution by lithium. Due largely to its low Z (the ion electrical charge number) alkali metal property, lithium is readily ionized with an exceptionally low first ionization energy of ~ 6 eV. Its full ionization energy is ~ 120 eV with a relatively low Z = 3, so it does not radiate very much in the hot plasma core as shown in Figure 2. Interestingly, the measured core lithium concentration (dilution) tends to be very small, often well below 1% even when a large amount of lithium coating was applied to the surrounding PFCs. This lack of adverse effects on hot fusion plasmas makes the lithium applications as a PFC material in magnetic fusion plasmas much more practical, particularly for high divertor heat flux handling where a significant amount lithium is expected to be utilized.

3 High divertor heat flux handling – The lithium melting temperature is relatively low (~ 180 °C) which is likely to place lithium in a liquid state in a conceivable reactor PFC environment. As the LL temperature increases, the lithium evaporation rate also increases rapidly as shown in Figure 3 [5]. As the lithium is evaporated, lithium is quickly ionized, which could cool the divertor / edge plasma through radiative losses. The radiative edge/divertor cooling was observed in several magnetic fusion experiments as well as in test stands, as described in Sec. III. Modeling calculations indicate that the lithium radiative loss in divertor can be significantly enhanced over the coronal equilibrium value. This is because of enhanced transport (or poor confinement, low $n_e\tau$) expected in the open field line divertor region. The radiative loss can be as high as 1.2 keV per injected lithium ion, which is equivalent to over 100 MJ per mole of lithium ions [5]. Figure 4 shows the total radiation power per lithium atom/ion calculated for different electron temperature and non-stationary parameters $n_e\tau$ in suitable ranges for the divertor/edge plasmas [6]. As shown in the figure, for the expected divertor electron temperature range of 30 - 300 eV, the radiation power level of non-equilibrium (coronal) lithium can exceed the coronal limit ($n_e\tau$ = infinity) by a factor of 100 - 1000. This type of lithium behavior makes it an ideal PFC for handling very high divertor heat flux as discussed in Sec 3.3 and Sec. IV.

We shall now cover the earlier lithium PFC experimental results in Sec. II, and then describe the on-going world-wide lithium PFC research in magnetic fusion facilities in Sec. III. In Sec. IV, various possible solutions for high power handling for future devices, including fusion reactors shall be discussed. In Sec. V, we conclude with a discussion and conclusions.

II. Earlier Lithium Research for Fusion First-Wall Applications (1990-2000)

Lithium utilization as a wall coating material for magnetic fusion experiments started over two decades ago on TFTR. Due to the exciting results on that tokamak, lithium experiments were performed in other devices in the subsequent years. An overview of the lithium experimental results in fusion devices in this period can be obtained in a summary report of a workshop on lithium wall conditioning effects held in 1996 [7]. The workshop was attended by researchers from the major tokamaks operating at that time (TFTR, DIII-D, TdeV, Alcator C-Mod, Textor, and Tore Supra). By far the most exciting results were produced on TFTR, where deposition of a few milligrams of lithium on the bumper limiter led to a near doubling of the fusion power output.

2.1. TFTR Lithium Experiments

TFTR conducted a series of lithium PFC experiments starting with lithium pellet injection, which produced very encouraging earlier results and concluded with the

"DOLLOP" experiment, which is described below [8]. An experimental set up of the DOLLOP (deposition of lithium by laser outside the plasma) is shown in Figure 5. The system uses a laser to introduce a directed lithium aerosol into the plasma discharge scrape-off layer. Because it is light (0.52 g/cm^3 at 200°C) and has a high surface tension (397 ergs/cm^2 at 200°C), LL readily forms an upward directed aerosol when its surface is abruptly perturbed by a focused laser beam. In this work, LL was contained in a small (17.5 cm^3) boron nitride cauldron positioned near the plasma edge on a movable probe. The cauldron was heated ohmically using tantalum wire and was monitored with a thermocouple. Approximately 4 W was required to reach the lithium melting temperature. During plasma experiments, a 250°C operating temperature was maintained with a heating power of 7 W, and a directed aerosol was then created quasi-continuously by the action of a pulsed YAG laser beam directed onto LL. The laser operated at 1064 nm and delivered 1.6 J/pulse in 8 ns wide *Q* switched pulses at a repetition rate of 30 Hz. The beam was focused onto the LL surface using a lens doublet with variable spacing located just above a quartz input window. During the TFTR DOLLOP experiments, it was determined that under conditions of optimal focusing (beam diameter 1- 2 mm), an average of approximately 20 mg/s ($2 \, x \, 10^{21}$ atoms/s) of lithium was injected into the plasma edge while the laser was operating. The lithium introduced in this fashion ablated and migrated preferentially to the limiter contact points. This allowed the plasma-wall interaction to be influenced in situ and in real time by external means. As shown in Figure 6, a significant improvement in 'supershot' performance in TFTR can be seen that has been achieved through the use of lithium. The plasma confinement time and D-D fusion neutron production increased by up to 100% for lithium conditioned discharges, compared with discharges without lithium conditioning. The plasma z-effective (a measure of plasma purity) came down a nearly a factor of two. A reduction in edge density (and in carbon emission) was observed which improved the neutral beam penetration to the plasma core, which also helped improve the plasma performance. A significant reduction of ion thermal conductivity was accordingly inferred. Lithium conditioned discharges were found to produce additional beneficial effects, including limiting sawtooth behavior, producing the enhanced reversed shear regime, and enhancing the performance of high internal inductance configurations. The improved supershot performance on TFTR with lithium application can be attributable to the reduced recycling by the lithium wall coating. In an earlier experiment on TFTR, the supershot performance was observed to improve with the reduced wall recycling [9] as shown in Figure 7, where the improving plasma confinement time is shown as the edge recycling is reduced, indicated by the decreasing deuterium neutral emission line (H_α) signal. The lithium concentrations in the plasma were only a few tenths of one percent to as much as two percent for the most aggressive lithium "painting" sequence. The lithium effects were transient, lasting for only a few plasma discharges. It should be noted that the lithium technique was successful in TFTR only after extensive pre-conditioning with glow and disruptive discharge cleaning and boronization, which substantially reduced the flux of oxygen, deuterium, and hydrogen into the plasma.

2.2. Lithium Experiments in Other Magnetic Fusion Devices

While the TFTR lithium experiments have produced spectacular results, much more modest effects (or often negligible effects) were observed in other devices. On DIII-D and C-Mod, the lithium pellet injection technique was employed. While there were some incremental beneficial effects such as oxygen reduction, no significant plasma performance improvement was observed. In the TdeV tokamak in Canada, the lithium was introduced with a crucible (which is similar to the lithium evaporator discussed in Sec. 3. 1). Again, the effects of lithium were modest on plasma performance enhancement. Because of these modest effects, the lithium experiments were not continued on C-Mod and DIII-D. There was a question that the observed differences were related to TFTR being a limiter tokamak, while DIII-D, C-Mod, and TdeV were diverted tokamaks where lithium PFC conditioning may not be as effective. This question was answered nearly a decade later, where the experiments on NSTX and EAST have demonstrated strong plasma performance improvements utilizing lithium on these diverted tokamaks as described in Sec. III. After the shutdown of TFTR, the lithium experiments on fusion devices have largely stopped for several years until a new generation of lithium experiments were initiated as described in Sec. III.

2.3. Liquid Lithium APEX Study

A study called APEX (Advanced Power Extraction) was initiated in early 1998 to encourage innovation and scientific understanding for fusion plasma facing components [3]. The primary objective of APEX was to identify and explore novel, possibly revolutionary, concepts for the chamber technology that can substantially improve the attractiveness of fusion energy systems. The APEX participants conducted a study to eliminate the solid ‘‘bare’’ first wall by flowing liquids facing the plasma. An example of an APEX concept is to surround the plasma completely with a thick LL wall. The thick LL would function as PFCs, neutron shields, and s tritium breeding blanket all in one. Another APEX concept called CLiFF (the convective Liquid Flow First-Wall concept) utilizes a fast moving (convective), thin liquid metal layer flowing on the fast wall surface as shown in Figure 8. These liquid wall concepts have some common features, but also have widely different issues and merits. Some of the attractive features of liquid walls include the potential for: (1) high fusion power density handling capability; (2) higher plasma β and stable physics regimes if liquid metals are used; (3) increased disruption survivability; (4) reduced volume of radioactive waste; (5) reduced radiation damage in structural materials; and (6) higher availability. Analyses showed that not all of these potential advantages may be realized simultaneously in a single concept. However, the realization of only a subset of these advantages will result in remarkable progress toward attractive fusion energy systems. Of the many scientific and engineering issues for liquid walls, the most important are: (1) plasma–liquid interactions including both plasma–liquid surface and liquid wall–bulk plasma interactions; (2) hydrodynamic flow configuration control in complex geometries including penetrations; and (3) heat transfer at the free surface and temperature control. This APEX study has stimulated subsequent LL experimental studies on magnetic fusion experimental facilities as described in Sec. III.

III. Lithium Experiments in Present Day Fusion Devices (2000 – 2012)

Lithium research enjoyed a renewed resurgence nearly a decade following the initial TFTR lithium experiments, stimulated in part by the APEX study. Unlike earlier lithium experiments, the benefits of lithium have been observed in nearly all the magnetic confinement fusion devices which employed lithium PFC coatings including CDX-U/LTX (US), CPD (Japan), HT-7 (China), EAST (China), FTU (Italy), NSTX (US), T-10 and T-11M (Russia), TJ-II (Spain), and RFX (Italy). The devices CDX-U[10], LTX [11], CPD [12], HT-7 [13], EAST [14], FTU [15], NSTX [16], T-10 / T-11M [17] are of the tokamak type, where the plasma current is required for the plasma confinement. TJ-II [18] is a so-called heliac stellarator type, where the external windings provide the plasma confining magnetic fields. The stellarator has the advantage of disruption-free operation. RFX [19] is a reversed field pinch (RFP) configuration which operates at a much higher plasma current for a given toroidal magnetic field value than tokamaks. Of the tokamak devices, CDX-U/LTX, CPD, and NSTX are of the low-aspect-ratio ($R_0/a \leq 2$) tokamak configuration, which is also called a "spherical tokamak" (ST) because of its spherical plasma shape. HT-7 and EAST are the superconducting coil tokamaks. NSTX and EAST are tokamaks with divertor configurations, which are suited for studying lithium effects on H-mode plasmas. FTU is a high field / high density tokamak. The results from these devices generally affirmed that the lithium PFC coating enhanced the plasma performance through improving plasma confinement, and modifying plasma boundary via reduced recycling. These results have therefore confirmed the effectiveness of lithium PFC coatings in multiple devices and magnetic confinement configurations. Summary articles of the more recent lithium experiments and related research can be found in the conference reports for the 1st and 2nd workshops on lithium applications in fusion plasmas [20, 21]. In addition to the plasma performance improvement, which was the main motivation for the earlier experiments, the recent lithium research has been also trying to address the reactor divertor PFC issues as noted by the APEX study. In Sec. 3.3, the potential viability of lithium as a high heat flux divertor PFC material shall be discussed.

3.1. Lithium Delivery Systems

How to deliver lithium to plasma facing components (PFCs) of fusion devices is an important component of the lithium research. There are a number of different lithium delivery systems introduced in the fusion experiments. In Figures 9 and 10, four representative types of lithium delivery systems for PFCs are shown. These systems represent important advances in lithium delivery for fusion PFC research.

The LL tray installed in CDX-U is shown in Figure 9(a) [10]. A shallow, heated, stainless steel tray was installed at the bottom of the CDX-U vacuum vessel. The tray has an inner radius of 24 cm, is 10 cm wide and 0.5 cm deep, and exposes 2000 cm^2 of LL in a pool to the plasma. It is constructed in two halves, with a single electrical break to prevent induction of large currents in the tray due to the ohmic transformer. The temperature controlled stainless steel tray is filled with LL, using a delivery system consisting of two heated tubes that were fed from lithium reservoirs outside of the vacuum vessel. After filling, the tray can be heated up to 500 °C but the usual operating temperature was ~ 300 °C well above the lithium melting temperature of ~ 180°C, which then allows an LL surface for the tokamak plasma.

The NSTX lithium evaporator (LITER) is shown in Figure 9 (b) [22]. The LITER system is essentially a temperature controlled stainless steel container filled with LL, with a nozzle to direct the lithium vapor for coating PFCs at desired locations. The nozzle is typically aimed toward the middle of the inner divertor to maximize the lithium deposition on the divertor plates. Two LITERS units were used for better toroidal PFC coverage of lithium on NSTX. The units each have a 90 g lithium capacity. The LITER consists of a main reservoir oven and an output duct to allow insertion in a PFC gap in the upper divertor region. Two heaters were used on each LITER, one heater on the output duct and one heater on the main reservoir. The heater on the main reservoir was typically operated to maintain the LL temperatures of 600–650 ℃ which enables an adequate lithium evaporation rate, as this rate increases rapidly with temperature (see Figure 3). The heater on the output duct was operated about 50–100 ℃ hotter than the heater on the main reservoir to reduce lithium condensation on the output duct aperture. Typical evaporation rates have been in the range of 1 to 40 mg/min. The lithium evaporation typically takes place between plasma discharges to obtain the desired level of lithium coating on the PFCs, which could be in the range of 30 – 500 nm thick. Lithium evaporator systems of various types have been used in CDX-U, NSTX, HT-7, and EAST devices. In NSTX, nearly 1,000 g of lithium was delivered onto the PFCs during an experimental campaign in 2010. In the CPD compact spherical tokamak, lithium evaporators were used to coat the rotating tungsten coated spherical drum limiter surface as shown in Figure 9(c) [12].

In Figure 9(d), an NSTX lithium dropper is shown [23]. The dropper drops spherical lithium granules as a powder in a controllable manner using a vibrating piezoelectric disk (PZD) with a central aperture. Lithium (Li) injection rates as low as ~1 mg/s (4.3×10^4 spheres/s) and as high as~120 mg/s (5.0×10^6 spheres/s) are attained reproducibly using this device. The reservoir capacity is ~150cm^3, corresponding to 50 g of Li powder. While the size of the lithium granule powder can be chosen according to the desired objectives, an experiment on NSTX typically utilized lithium spherical particles with a 44 μm average diameter and a thin surface coating to prevent uncontrolled reaction with air and to facilitate smooth flow by minimizing sticking. This technique can inject lithium powder at a rate which can be ~ 100 times the lithium evaporator so this dropper can be used during the plasma discharge to affect its performance. This technique also has the advantage of being able to inject lithium for long pulse (steady-state) discharges. The NSTX lithium dropper has also been used successfully on the long-pulse superconducting EAST tokamak [14]. The application of lithium to PFC surfaces by the dropper technique is similar in principle to the lithium pellet injection technique used in the earlier TFTR and NSTX experiments, and more recently, on FRX [19] and the DOLLOP system in TFTR as described in Sec. 2.1. They all use the plasma to ionize and transport the injected lithium ions to desired PFC locations surrounding the plasma.

In Figure 10, the Active Capillary-Porous Systems (CPS) on various fusion devices are shown. The CPS was developed in Russia in 1990's [24, 25]. A CPS system utilizes a capillary-porous material to wick the LL from a reservoir to the CPS plasma contacting surfaces. The CPS takes advantage of the rapid LL mobility (combination of high surface tension and low viscosity) to continuously replenish the LL saturated plasma facing porous material surfaces from the reservoir through capillary action as the high heat load evaporates lithium from the porous material. The way the lithium evaporation protects the CPS is analogous to what happens in an oil lamp. The oil is replenished through a wick from an oil reservoir, and the burning/evaporating oil actually protects the oil saturated wick material from the heat of the lamp flame. The CPS system has been tested on T10 and T-11 [17], FTU [15], TJ-II [18], and HT-7 [13]. The CPS system principle is also being applied to a new type of high heat flux CPS divertor module being developed, as discussed at the recent lithium workshop [21]. An example for the CPS divertor module planned for the KTM tokamak [24] is shown in Figure 10 (d).

3.2. Improved Plasma Confinement and Performance with Lithium

As observed in TFTR [8], lithium applications have improved plasma confinement in many devices. Improving plasma confinement is a particularly high priority for magnetic fusion research, since it can greatly influence fusion reactor performance. The thermonuclear fusion reactivity increases very rapidly with the plasma confinement enhancement factor for a given heating power. With improved plasma energy confinement, the fusion reactor size can be made more compact, resulting in the fusion reactor capital cost reduction.

In the CDX-U spherical tokamak, extensive lithium wall coatings and LL plasma-limiting surfaces on Ohmic limiter plasma discharges significantly reduced edge neutral particle recycling, which then dramatically improved the global energy confinement times by up to 6 times [26]. If this confinement improvement can be extended to larger experiments with H-mode plasmas and eventually to fusion reactors, that would certainly be quite revolutionary. The CDX-U is upgraded to the LTX [11] to test effects of very aggressive lithium pumping to achieve very low recycling regime by essentially surrounding the entire plasma surface with lithium coated PFCs. This was motivated by the theoretical prediction of a very high confinement regime by eliminating the edge recycling [27].

In the NSTX spherical tokamak, lithium evaporation has produced many intriguing and potentially important results [28]. An overview of the NSTX lithium results and their implications can be also seen in recent papers [29, 30]. In Figure 11, a schematic diagram of the poloidal cross-section of NSTX and the lithium evaporator (LITER) arrangement for NSTX is shown. The locations of two LITERs are at toroidal angles 165° and 315°, and the LITER central axes are aimed at the lower divertor. The shaded regions indicate the measured half-angle of the roughly Gaussian angular distribution at the 1/e intensity. A unique feature of the NSTX lithium research is the ability to investigate the effects of lithium in H-mode divertor plasmas. This addresses the long standing question of the effectiveness of lithium in a diverted H-mode plasma compared, for example, to the improvement observed in the limited plasma TFTR supershots. The application of LITER on NSTX has yielded a significant improvement in the electron confinement with lithium coating of carbon tiles in H-mode plasmas. Importantly, the lithium evaporation resulted in a broadening of H-mode electron temperature profiles compared to plasmas without lithium applied, as shown in Figure 12 [22]. The broadening of the electron temperature broadens the pressure profile, which helps to improve the plasma MHD stability at high beta as needed for advanced plasma operations. Analysis with the TRANSP code indicates that the electron thermal diffusivity in outer region is progressively reduced with increasing lithium evaporation, as shown in Figure

13 [31]. The improving electron energy confinement with lithium is consistent with the trend of improved confinement with reduced collisionality generally observed in NSTX. Thus far, the electron energy confinement continues to improve with the amount of lithium evaporated without reaching an apparent saturation which suggests that further improvements maybe possible. Burning fusion reactor plasmas, including ITER, rely on heating by the 3.5 MeV fusion alpha (helium) particles. These alpha particles predominantly heat electrons because of their high energy, so the understanding and eventual control of electron energy transport is of critical importance for magnetic fusion reactor optimization. The ability of lithium application to affect electron transport offers the possibility of testing competing theoretical models which may lead to a resolution of the long standing question about the mechanism of electron energy transport in fusion reactors.

On the high field FTU tokamak, introduction of lithium with a CPS limiter system has improved discharge performance, reproducibility, and recovery from disruptions [15]. In particular, lithium has reduced the radiated power in the core, raised the density limit to as high as 1.3 times the Greenwald limit (an experimentally observed density limit in tokamak plasmas), and peaked the density profile to $n_e(0)/\langle n_e \rangle \approx 2.5$ by lowering the edge density. The energy confinement has increased by up to 40%, and transport analysis shows a reduction in the electron thermal diffusivity by a factor 2.

The heliac stellarator TJ-II has studied the effects of lithium evaporated onto its vacuum chamber walls for three years and over 10000 discharges utilizing the CPS systems [18]. Boronization is also applied. The lithium / boron coating produces a remarkable reduction in the recycling of both hydrogen, to about 10%, and helium, to about 80%, as inferred from the pump-out of the density. The use of lithium has allowed routine operation with the two neutral beam injectors. Clear transitions to the enhanced confinement mode (H-Mode) and strongly peaked plasma profiles were observed under the lithiumized wall conditions [32].

The EAST superconducting tokamak attained its first 1 MA discharges, as well as its first 100-second discharge (100kA), using lithium. The EAST tokamak also recently obtained its first H-mode plasma after wall conditioning by lithium (Li) evaporation before plasma breakdown and the real-time injection of fine Li powder into the plasma edge [14]. Similar beneficial effects from lithium have been observed in other tokamaks including CPD [20], HT-7 [13], T-10 and T-11M [17], and the reversed field pinch device RFX [19]. The benefit of lithium applications on the plasma confinement and performance has been therefore demonstrated on a number of plasma devices and with various plasma configurations.

3.3. Protective Effects of Lithium As PFCs

The divertor PFCs are expected to encounter particularly severe operating conditions, including extremely high heat flux at the divertor strike point as discussed in Sec. I. By covering the material surface with LL could provide an important protective function as demonstrated in laboratory testing and fusion experiments as described below. The effectiveness of radiative edge/divertor cooling by lithium was observed and demonstrated in several test stands and also in several magnetic fusion experiments.

The CPS has the promise of handling high heat flux as demonstrated in test stands and fusion experiments. As described in Sec. 3.1, the capillary action and high surface tension of LL keeps the CPS porous metal surface replenished, through the capillary action, from the reservoir while LL evaporates under intense heat flux from plasma. In a test stand with an electron beam, the CPS surface was observed to survive a very high heat flux of 50 MW/m^2 for a several seconds [25]. The CPS surface essentially survived the high heat flux as long as the capillary-porous material was kept saturated with LL. With a fully functioning CPS, the incoming main divertor power is radiated in the vicinity of the divertor plates (CPS) by the lithium neutrals/ions, which screen the target, substantially reducing the effective power flow density onto the CPS target [25]. The calculations show that the thermal loading onto the divertor CPS target is essentially reduced without addition of any heavy impurities, at an electron density at the separatrix of 6 x 10^{13} cm^{-4}. The shielding of the target by lithium evaporation and ionization is therefore realized. The CPS lithium limiters have been successfully tested and utilized in the experiments on various fusion facilities including T10, T11-M, FTU, and TJ-II. The protective nature of lithium is evident in the FTU experiments, where the plasma impurity content has been strongly modified by using a CPS-based liquid lithium limiter (LLL) [15]. The LLL is placed in the shadow of the toroidal TZM (titanium-zirconium-molybdenum alloy) limiter. With the lithium on the LLL heated, the only impurity present in the discharges was lithium, essentially eliminating the molybdenum (Mo) impurity, which was present without the lithium on the heated LLL. This benefit of the LLL appears in all of the operational space parameters of FTU, independent of the density and current values as well as in presence of additional heating power up to 1.6MW. The Z_{eff} ranges were generally low, between 1.5 at low density and ~ 1.0 at higher density. In Figure 14, FTU plasma discharges are shown with an extremely high amount of Li injection, using a poloidal LLL acting as the principal Li source in the shadow of the main TZM toroidal limiter. The formation of a 'virtual' toroidal lithium limiter can be seen, which redistributes the heat flux to the tokamak vessel first wall by radiative processes and therefore decreases the heat load onto the Mo limiter [33].

In CDX-U, the e-beam targeting LL in the LL tray [see Figure 9 (a)] demonstrated the effectiveness of LL in dissipating extremely high heat loads [34]. The set of vertical field coils nearest to the CDX-U mid-plane, along with the toroidal field coils, were used to guide the e-beam to the lithium in the limiter tray. The e-beam was operated at with about 1.5 kW and had an effective beam diameter of ~ 3 mm, or 60 MW/m2, on the LL surface in the limiter tray. Only a modest temperature peak at the beam spot occurs that is about 50 ℃ above the temperature of the bulk of LL. In a similar experiment at the University of Illinois (UI), thermoelectric MHD was found to cause the LL flow [35]. In both the CDX-U and UI experiments, LL was found to allow convective flows that would mitigate the effects of a localized very high heat flux.

Finally on NSTX, a liquid lithium divertor (LLD) was tested in 2010 in a high performance H-mode configuration with high divertor heat flux [36]. A picture of the LLD plates in NSTX is shown in Figure 15. The LLD consists of four plates, 22 cm wide and each spanning 80◦ toroidally. The plates are electrically isolated toroidally to prevent induction of large toroidal currents in LLD by the ohmic transformer. The quadrants were separated toroidally by graphite tiles containing diagnostics and electrodes for edge plasma biasing. The plasma-facing surface of the LLD has a 0.17 mm layer of Mo, plasma sprayed with 45% porosity onto a protective barrier of 0.25 mm stainless steel that is bonded to a copper substrate 2.2 cm thick. The Mo porosity is intended to facilitate wetting and subsequent spreading of LL over the LLD, and to make the lithium surface tension forces large relative to electromagnetic forces in the liquid layer. In the NSTX experiment, sufficient lithium was applied by the LITER system (as shown in Figure 11) to the LLD surface to saturate the sprayed Mo layer. Even though the LL layer is very thin, there was no observation of Mo line radiation from the plasma (i.e., no indication of excessive Mo material erosion), and inspection of the LLD surface after the campaign yielded no visual evidence of power or cyclic thermal stress damage to the plasma sprayed porous Mo LLD surface. In NSTX, the in-situ measurement of the divertor heat flux with a “two color” fast infrared camera of lithium coated LLD plate was significantly less (~ x 2) than those surfaces with reduced lithium coating, again indicating the benefit of a lithium coating. The reduced head load was accompanied by the increased divertor bolometric radiation as expected [37].

3.4. Benign Properties of Lithium Injection for Fusion Plasma Operations

Based on lithium experiments worldwide, introduction of significant amounts of lithium on to the PFCs generally causes no adverse effects on the fusion plasma performance . This includes the observation of very little plasma dilution by lithium. Due largely to the low Z alkali metal nature of lithium, the lithium atom is readily ionized with the low first ionization energy of only 5.9 eV, and full ionization energy of ~ 120 eV. Lithium ions therefore radiate only in the plasma edge and divertor region, and do not cause adverse effects (such as radiative cooling) in the plasma core compared to higher Z material such as fluorine as shown in Figure 2. In Figure 16, a fast camera view of the lithium line in NSTX during the lithium dropper injection in NSTX is shown where one can see the strong lithium (green light) emission along the field line of lithium injection. The green light is the dominant singly ionized (LiII) excited state transition emission at the wavelength of 548.5 nm, which is green. In NSTX, even with a rather high lithium injection rate, the plasma is relatively unperturbed by it and the lithium concentration remains very small in the plasma core. In the NSTX H-mode, the core lithium concentration (dilution) was measured to be remarkably small, typically below 0.1% even when a large amount of lithium coating (e.g., ~ 1, 000 gram of lithium evaporated onto PFC in one campaign) was applied to the PFCs [38]. This is at least partially understood by the lithium’s low charge state (Z=3 when fully ionized), which makes the neoclassical pinch forces small so no significant inward lithium transport is expected. Generally, the experimental observations have been that the more lithium that is injected, the better the plasma performance results. This lack of adverse effects on the high temperature fusion plasma core makes the lithium application as divertor PFC material much more practical, particularly for high heat flux handling where a significant amount lithium is expected to be employed.

3.5. Lithium Influence on H-Mode

The high confinement mode (H-mode) is considered crucial for high performance reactors due to its high confinement and good MHD stability property at high beta. For ITER, predictive modeling shows that the fusion Q (or the ratio of generated fusion power over input heating power) scales roughly with the square of the edge plasma or “pedestal” pressure. To fulfill its mission of high fusion gain (Q = 10), it is therefore essential for ITER to achieve an H-mode with sufficiently high pedestal pressure. On NSTX, the application of lithium helped routine H-mode operation, and the L-H power threshold (P_{LH}) is reduced by up to 40% [39]. The edge electron confinement improvement with reduced density and edge recycling is likely to be aiding the H-mode transition by increasing the electron temperature near the edge. Interesting, with aggressive lithium application, a very high confinement regime called the enhanced pedestal H-mode, with confinement enhanced relative to the predictions of the ITER H-mode scaling (up to $H_{H98y,2} \sim 1.7$ compared to around unity for a typical H-mode) has been observed following lithium application [40]. This level of high confinement is sufficient for future compact ST reactors such as a Fusion Nuclear Science Facility (FNSF) [41] or an ST Pilot Plant [42].

As mentioned earlier, the EAST tokamak achieved its first H-mode plasmas with the assistance of both evaporated lithium coatings and the real-time injection of lithium powder into the plasma scrape-off layer with an NSTX lithium dropper apparatus [14]. An H-mode lasting 6.4 seconds and limited only by the available OH flux consumption was subsequently attained. In TJ-II, application of lithium together with boronization was used to attain clear transitions to the H-mode, which essentially doubled the plasma energy confinement [32].

Another important discovery of lithium effects on the H-mode was the stabilization of edge localized modes (ELMs) in NSTX [28, 43]. The ELMs are periodic macroscopic instabilities in the H-mode barrier region. They can regulate the H-mode barrier pressure and help facilitate steady-state H-mode operation by removing impurities which otherwise could accumulate within the H-mode barrier. However, an ELM event can generate a high transient heat flux due to a loss of as much as a few precent of the total plasma stored energy when the ELM occurs. This could damage the divertor tungsten plasma-facing components of future reactors including ITER. The concern for the ELM heat damage makes ELM mitigation research a high priority for ITER, either to completely eliminate the ELMs or keep the ELM heat loss per event to a very small value well below 1%. In NSTX, the application of lithium led to a complete suppression of ELM in H-mode discharges, as can be seen in Figure 17. Interesting, as shown in the figure, the ELM stabilization is a gradual process with increasing levels of lithium application similar to the confinement improvement [43]. Detailed profile measurements, coupled with analysis of the edge MHD stability utilizing a "peeling-ballooning" model, show that the ELMs become stabilized by an effective inward shift of the H-mode pedestal pressure gradient region toward lower magnetic shear region by the lithium induced edge density reduction.

3.6. Improving Plasma-Wave-Based Heating and Current Drive Performance

Magnetic fusion reactors require a tool to heat the plasmas to fusion reacting temperatures (over 100 million °C). For tokamak type reactors, in addition to heating, it is also essential to drive some fraction of the plasma current by an external means (i.e., current drive). While a neutral beam injection (NBI) based heating and current drive system has been the mainline tool in fusion experiments, plasma-wave-based (PWB) heating and current drive concepts have been pursued actively as alternatives. A typical PWB system consists of a radiofrequency ("rf") power source (typically ~ 100 MHz - 200 GHz range depending on the plasma wave types), a power transmission line (waveguide or coaxial line), and a launcher (an antenna or a waveguide) to couple the rf power to the plasma waves. For fusion reactor applications, since the PWB source can be located well away from reacting plasmas and the transmission line can be made to minimize the penetration profile across the fusion blanket and the neutron shielding, PWB systems have certain technical advantages over the NBI based systems. The application of lithium to PFCs to control the plasma edge turned out to be generally useful for PWB research. Some examples from NSTX, FTU, and EAST are given in this section.

In NSTX, the ability of lithium to reduce the edge scrape-off-layer density turned out to be highly beneficial for High-Harmonic Fast Wave (HHFW) heating and current drive [44]. The HHFW is a magneto-sonic wave operating at multiples of the ion cyclotron frequency, and is designed primarily to heat electrons and drive plasma current. The plasma waves launched by an antenna must reach the main plasma region (to the interior of the last closed flax surface) with sufficient efficiencies to be a viable tool for reactor application. The plasma "scrape-off" region, a lower temperature and lower density region between the rf antenna and the main plasma, can often present various challenges for plasma wave propagation. It was discovered that the edge density (in the vicinity of the antenna) must be kept below the cut-off density to insure that the wave propagation starts well away from the antenna to avoid edge parasitic effects. Another form of plasma wave coupling investigated on NSTX involved the electron Bernstein wave (EBW) [45]. The EBW heating and current drive approach is a promising tool to drive tokamak plasma currents in a desired location to improve advanced tokamak reactor performance. The EBW is a hot plasma wave, which requires a mode-conversion of the launched electromagnetic electron cyclotron wave to EBW at the so-called mode-conversion layer located in the plasma edge region. It was found that in NSTX H-mode plasmas, collisional absorption near the mode-conversion layer significantly reduced the EBW coupling efficiency. With application of lithium, and the resulting reduced density and increased electron temperature, the edge collisional absorption in the mode-conversion region was indeed reduced sufficiently to improve the EBW coupling efficiency significantly from ~10% to ~ 60%.

In the EAST ICRF experiment (where ICRF is a class of magneto-sonic plasma waves in the ion cyclotron frequency regime), the use of lithium has also resulted in a dramatic reduction in the H/(H+D) ratio to as low as 7%, and has thus allowed significantly improved ICRF H minority heating efficiency [14]. The ICRF H-minority heating efficiency goes down significantly as the H-minority fraction (in predominantly deuterium plasmas) increases, so it is important to keep the hydrogen (H) level low. The hydrogen mainly comes from the residual water retained in the vacuum vessel walls after venting to atmosphere and its level is a rough indicator of the cleanness of the plasma. The application of lithium has been observed to be effective in reducing the hydrogen level in the plasmas in all fusion experiments mentioned in this article, since lithium reacts strongly with atomic hydrogen.

In FTU, with Lower Hybrid (LH) and Electron Cyclotron Resonance (ECR) heating applied at total power levels up to 1.6 MW, preliminary results indicate that a strong internal transport barrier (ITB) was created with lower additional power than in pure metallic or boronized wall machine conditions [15]. The reduced recycling as well as the low Z_{eff} values, along with increasing the LH current drive efficiency, could help the formation of an optimal current radial current profile. Recently, quite flat electron temperature profiles up to 4 keV have been obtained with 800kW of ECRH, which is half the power normally needed to reach a comparable temperature without lithium.

IV. Viability of Lithium PFCs for Magnetic Fusion Reactors

At the second Lithium Symposium [21], a panel discussion was held addressing the previously identified question, "Is a lithium PFC viable in magnetic fusion reactors such as ITER?" The following specific technical issues for lithium reactor applications were identified in the first Lithium Symposium [20]: 1. Handling high divertor heat flux, 2. Removal of deuterium, tritium, and impurities from LL, 3. Removal of high steady-state heat flux from divertor, 4. Flowing of LL in magnetic fields, 5. Longer term corrosion of internal components by LL, 6. Safety of flowing LL, and 7. Compatibility with LL with a hot reactor first wall. It was noted that LL has tremendous potential as a PFC material in the highly challenging fusion reactor environment. It can handle an extremely high heat flux due to its exceptionally high rate of heat dissipation through radiative cooling (Sec 3.3). The advantage of LL is its resilience from any mechanical damage, which is problematic for solid PFCs. It can melt, vaporize, and ionize yet it can be collected and re-purified for renewed PFC application. Lithium is also compatible with a high neutron environment. The isotope ^{6}Li (constituting 7.5 % of natural lithium), for example, produces tritium from the n + ^{6}Li interaction but the resulting tritium can be then removed along with other non-lithium ions as described in Sec. 4.2. It is also possible to utilize a pure form of ^{6}Li for an LLD if that proves to be advantageous from the tritium production point of view, as a closed divertor chamber with a significant amount of LL may have a rich spectrum of slowed down neutrons suitable for the tritium production. As noted in ref. 3, there are a number of LL based PFC concepts for reactors. An intriguing idea is to use the energy of vaporization in a closed divertor chamber [46]. While the heat of vaporization for LL is quite substantial (~ 11 MJ / liter), the required amount of LL would also be quite substantial (~ tens of liters per second) to support ~ few hundred MW of steady state power influx by just evaporation alone. The radiative cooling idea as discussed in Sec. 3.3 would make this concept much more practical, since the radiative cooling can be up to 1,000 times more energy efficient than evaporative cooling. In that case, the amount of lithium evaporation/ioninzation required inside the closed divertor would be relatively modest (~ tens of cc per second), so this approach is much more practical. Therefore in this section, as an example, we shall consider a closed LL divertor (LLD) concept as shown in Figures. 18 - 19 [47, 48]. It should be noted that since the solid material based design for the "first wall" (the terminology for PFCs that are not in the divertor) maybe acceptable due to the relatively modest anticipated heat load, we shall focus here on the utilization of LL for the highly challenging divertor PFC issue. The closed LLD has several potentially attractive features:

1 Having been placed within the closed divertor chamber, the LL within the LLD can avoid very fast disruptive electro-magnetic forces as the closed chamber could provide some degree of electro-magnetic shielding from fast disruptive events. Even if the LL is "splashed", it can be largely contained within the chamber..

2 The closed divertor chamber could provide some degree of particle partition from the main plasma chamber due to the strong divertor lithium retention. The in-flowing plasma from the main chamber into the divertor chamber also tends to push back the

lithium outflow from the divertor chamber via the force of friction. While the main plasma is likely to tolerate a considerable amount of lithium, it would be advantageous to be able to control and limit the lithium transport out of the divertor chamber from the particle inventory point of view particularly for tritium.

3 The closed divertor chamber with a high population of lithium ions within the chamber could be used as a radiative chamber to reduce the peak divertor heat load and spread the heat load throughout the divertor chamber, where the heat can be removed more readily by much larger volume/surface area heat exchanger structures (see Figure 19).

4 The vaporized and ionized lithium can be collected within the divertor chamber, and the entire divertor chamber wall coated with LL can function as a strongly pumping chamber consistent with the attainment of the ultra-low recycling regime for advanced high performance plasma operations (see Figure 19).

We shall now go over each technical question for the closed LL divertor system for reactors in some detail in the following sections.

4.1. Handling High Divertor Heat Flux

As mentioned in pervious sections (see Sec. I and Sec. 3.3), lithium has a promise for handling high localized heat flux through evaporation and ionization processes. The LL can also provide a "sacrificial" surface to protect the substrate material from high transient heat loads such as the ones caused by ELMs (Sec. 3.5). If the heat flux is high, that much more lithium would be evaporated and ionized, which would then increase the radiative cooling until an equilibrium condition is reached (see Figure 19). Since the projected lithium radiation is very high (i.e., ~100 MJ / mole of lithium ions), it should be readily feasible to provide an adequate amount of LL in the high heat flux divertor region. If the LL surface layer could reduce the direct divertor heat flux down to the vicinity of 5 MW /m^2, then the solid substrate material (perhaps made out of tungsten) could become viable as long as it is coated with lithium. As for the actual substrate, the requirement is to ensure an adequate amount of LL available at the location of intense heat flux (i.e., near the divertor strike point). The Active Capillary-Porous System (CPS) might be a good divertor surface, since the surface can be kept saturated with LL through rapid capillary action. There is a question of how to keep the CPS surface free of lithium compounds, which would not flow as well as pure LL. A CPS concept based on a large surface area divertor is now under development to be used in the KTM tokamak [24]. If the CPS surface function is just to provide LL available for evaporation/ionization into the divertor plasma, the CPS surface maybe kept clean by the plasma "scrubbing" action of the plasma. Perhaps, in terms of the LL circulation point of view, a flowing thin LL layer on smoother substrate surfaces may be more practical, even though there is an increased chance of LL splashing during disruptive events. However, as noted earlier, the closed divertor system should minimize the consequences of any LL splashing. The thin LL layer tends to adhere to the divertor surface because of strong surface tension. This has been previously studied, taking into account electromagnetic forces [3]. One can also consider the thin plasma-sprayed Mo coating to stablize the thin LL layer as used in NSTX LLD [36]. Finally, it would be of interest to utilize the thermoelectric and the

Marangoni effects to keep the LL flowing, as demonstrated in CDX-U [34] and the UI lithium test stand [35].

4.2. Removal of Deuterium, Tritium, and Impurities from Liquid Lithium

For LL to be viable for steady-state reactor operation, it is essential to continually purify the LL by removing D, T, and various impurities including the lithium compounds (see Figure 20). The IFMIF group [49] has a promising concept for a purification loop. The circulation requirements for impurity removal from LL is relatively modest, and likely to be only about 1 liter/sec. The IFMIF technology being developed should be quite relevant for the LL purification process in addition to the on-going LL test stand research and development (RandD) activities.

4.3. Removal of High Steady-State Heat Flux from Divertor

Divertor heat removal is a difficult challenge, since the steady-state heat removal requirement is very high (e.g., ~ a few hundred MW for a 1 GW electric power plant as discussed in Sec. I). If we were to allow some lithium vaporization and ionization in the closed lithium divertor chamber, then the divetor radiation would spread the heat more uniformly onto the surrounding divertor wall surfaces and greatly facilitate the heat removal (see Figure 20). For more near-term fusion facilities such as FNSF [41], the heat removal requirement is an order of magnitude lower (since the fusion power output is only about 1/10 of a 1 GW electric fusion power plant), so it would serve as a cost effective demonstration of a LL divertor system for future fusion power reactors.

4.4. Flowing of Liquid Lithium in Magnetic Fields

Since LL is a liquid metal, it encounters electromagnetic forces if it tries to flow across a magnetic field. A simple estimate indicates that it would likely take prohibitively large amount of power to move a sufficient amount of LL to carry out the divertor heat load to an outside heat exchanger as in the ALPS study [3]. As noted above, since it is possible to spread the divertor heat load through radiation, the heat can be then removed through secondary divertor structures. Such structures can be cooled, for example, by circulating high-pressure helium gas system which is considered a relatively safe approach in many reactor studies. There is, however, a need to circulate some amount of LL to keep the LL sufficiently pure by removing lithium compounds formed due to interaction with the plasma and residual gases in the reactor chamber. The circulating LL required for impurity removal is likely to be relatively modest, since the amount of lithium compounds and impurities formed within the reacting plasma chamber is expected to be low (estimated to be only a few grams or a mole per second). One could therefore envision a modest flowing LL loop (perhaps ~ a liter per second) for purification, which would then only require a negligible amount of power for circulation.

4.5. Longer Term Corrosion of Internal Components by Lithium

The longer term lithium corrosion issue is something we need to investigate in the future. This area has also been studied by IFMIF [49] and in fusion blanket RandD activities where a large amount of LL or its compounds are expected to be used. The LITER stainless steel containers operated for multiple years at high temperature ~ 600 °C without any corrosion or structural issues. The CPS systems also have operated over similar time scales without operational issues. Lithium is also known to be compatible with refractory metals such as molybdenum and tungsten. Lithium could chemically react with materials such as copper, aluminum, and ceramics. It is therefore important to design a system to segregate LL from materials that could be corroded by LL.

4.6. Safety of Flowing Liquid Lithium

The safety issue of a large quantity of flowing LL is being address by the IFMIF group, since IFMIF handles ~ 1 ton of flowing LL [49]. An effective safety barrier using inert gas such as argon is being designed and qualified at IFMIF. It would be logical to take advantage of the RandD performed by the IFMIF group on this topic. If an inert cooling gas such as pressurized helium gas is used for the divertor heat exchanger, the safety of the lithium system is also enhanced.

4.7. Compatibility with Lithium with Hot First Wall

It is generally envisioned in magnetic fusion reactor studies that the reactor first wall temperature will be relatively high (~ 500 - 700 °C) to keep the first wall surface relatively clean, particularly of tritium to keep the tritium inventory reasonably low. On the other hand, the divertor LL temperature is likely to be lower (~ 200 – 450 °C) to avoid excessive lithium evaporation (see Figure 3). Because of this temperature discrepancy, LL is often thought to be not compatible with the reactor environment. However, the lower operating temperature of LL may in fact make it suitable for reactor divertor operation. With the low melting temperature of ~ 180°C, it is very practical to keep lithium in a liquid state in a reactor environment, provided that the LL can be kept relatively clean as discussed in Sec. 4.2. If the LLD operates below the first wall temperature as noted above, the lithium and associated impurities should migrate toward the lower temperature LLD chamber, and keep the higher temperature first wall relatively clean. There was an interesting experiment in T11-M where a lower operating temperature CPS was able to actually collect lithium from the higher temperature CPS via plasma interactions [17]. The experimental set up of the T11-M lithium system is shown in Figure 21. The purpose was to demonstrate how the particles and gas could migrate toward the lower temperature region in the vacuum-plasma system within a fusion reactor. The approach would be similar to a dehumidifier (which has a cold collecting surface) collecting the water vapor within a room. In addition, the particles tends to end up in the divertor chamber because of the net particle flow from the main chamber to the closed divertor chamber with frictional forces in the direction of the closed divertor chamber.

Therefore, an LLD operating below the first wall temperature, together with a purifying system, could serve as the gas pumping, tritium recovery, and impurity control system for the entire reactor plasma - vacuum system.

V. Discussions and Conclusions

In magnetic fusion research, the use of lithium in magnetic confinement experiments started in 1990's in order to improve tokamak plasma performance with a higher pumping plasma-facing component (PFC) [7]. It was indeed a spectacular success for TFTR, where a very small amount of lithium coating (~ 0.02 gram per second) on the PFCs resulted in the fusion power output and the plasma confinement improving by nearly a factor of two [8]. This success was attributed to the reduced recycling of cold gas surrounding the fusion plasma by the highly reactive lithium covering the wall. The plasma confinement and performance improvements have since been confirmed in a large number of fusion devices with various magnetic configurations [20, 21]. Additionally, lithium was shown to broaden the plasma pressure profile, which is advantageous for achieving high performance H-mode operation for tokamak and ST-type reactors [22]. It is also noted that even with significant application of lithium on PFCs (e.g., up to 1000 grams of lithium in NSTX), no adverse effects on plasma operations were evident. Indeed, very little contamination (< 0.1%) of lithium in the main fusion plasma core was observed even in the H-mode. Lithium therefore appears to be a highly desirable PFC material for improving magnetic fusion plasma performance and operations. An exciting development in recent years is the realization of lithium as a potential solution for handling intense steady-state divertor heat loads, which could be as high as 60 MW/m^2 at the divertor strike points. Because of the low melting temperature of ~ 180°C, lithium can be readily kept in a liquid state in a fusion reactor environment. By placing LL surface in the path of main divertor heat flux (i.e., the divertor strike point), the LL will be evaporated from the surface. The evaporated lithium is quickly ionized by the plasma and the ionized lithium ions can radiate strongly, reducing the heat flux to the divertor strike point surfaces and protecting the substrate material. The protective effects of LL have been observed in many experiments and test stands as described in Sec. 3.3. For a closed radiative LL divertor system, as described in Sec. IV as an example of a possible divertor system for a tokamak reactor, the ionized lithium could reduce the divertor heat flux through radiation and simultaneously provide a strong pumping action by coating the entire divertor chamber wall surfaces. The lithium condensed on the divertor chamber in liquid form is eventually collected and purified, and is then fed back into the divertor strike point region for divertor heat flux handling. In Figure 22, a conceptual radiative LLD prototype that could be tested in NSTX-U is shown [47]. NSTX-U can provide very high divertor heat fluxs (~ 40 MW/m^2), comparable to that is expected in future tokamak reactors. The KTM lithium divertor module as shown in Figure 10(c) would be another exciting tool in this area of research [24]. Finally, it should be emphasized that lithium PFC applications are quite flexible and diverse. There are other divertor configurations that could greatly reduce the heat flux at the divertor strike point through expanding the divertor flux lines [50,51]. Lithium application should be quite compatible with various divertor geometry and magnetic confinement configurations providing the same benefits of lithium as noted in Sec. IV.

Application of lithium may also be considered for protecting the tungsten based solid PFC surfaces such as the ones for ITER, as long as a means to purity/refresh LL can be provided. In summary, lithium based PFCs have the exciting prospect of providing a cost effect flexible means to improve the fusion reactor performance, while providing a practical solution to the highly challenging divertor heat handling issue confronting steady-state magnetic fusion reactors.

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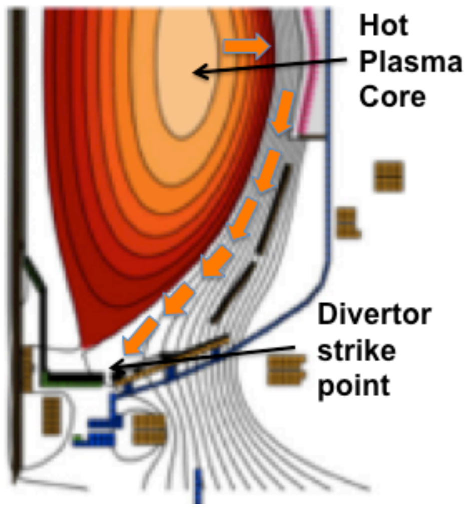

Fig.1. Tokamak divertor configuration cross section view. The main escaping plasma heat flux path from the plasma core to divertor surface is depicted by arrows for NSTX spherical tokamak.

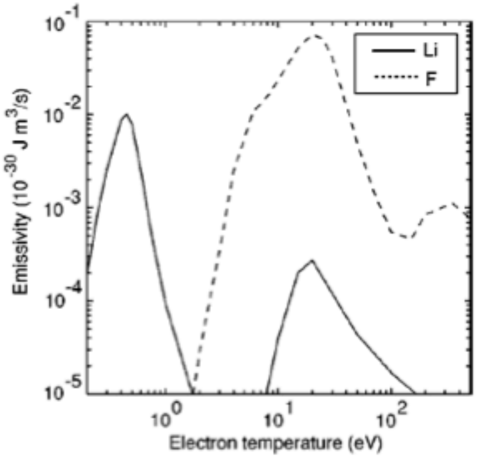

Fig. 2. Emissivity for lithium and fluorine in coronal equilibrium, i.e., no transport or charge-exchange recombination.

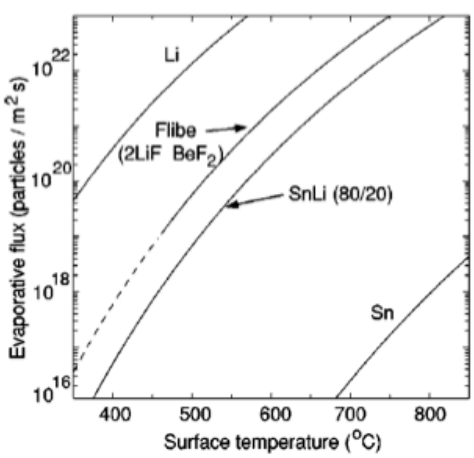

Fig. 3. Evaporation rates of four candidate liquid-wall materials. The dashed line indicates solid flibe; other salt mixtures may be liquid in this region.

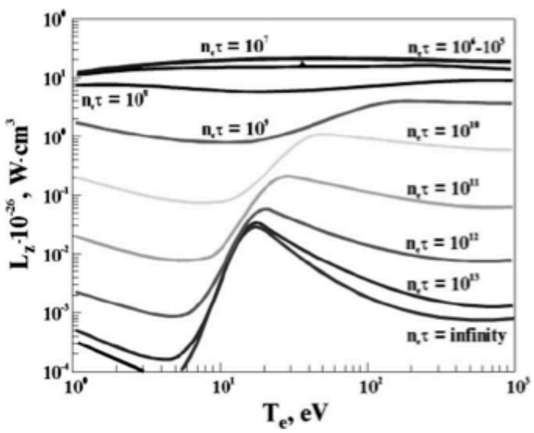

Fig. 4. The Li radiation power per one atom and one electron in coronal equilibrium ($n_e\tau$ = infinity) and non-equilibrium regimes.

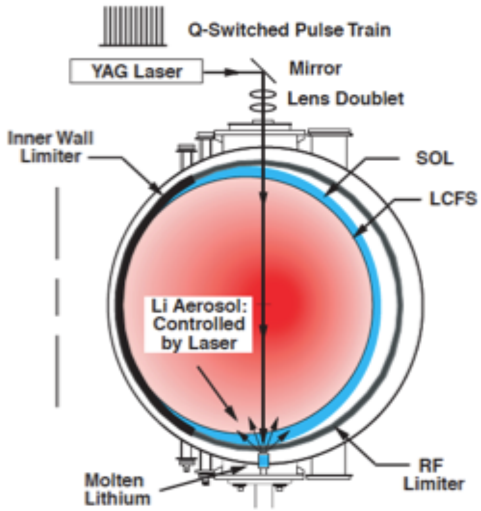

Fig. 5. DOLLOP, shown schematically, delivered a directed Li aerosol into the plasma SOL. The computer controlled YAG laser was located about 50 m from the TFTR vessel. The distance from the cauldron to the centre of the vacuum vessel cross-section was 1.1 m.

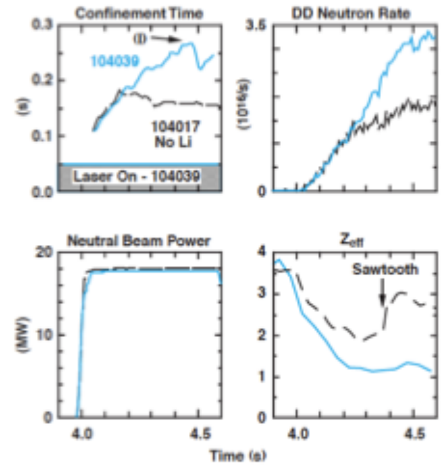

Fig. 6. Improvement in plasma performance brought about by Li conditioning. This particular discharge (104039) represents the frstt attempt to improve a high power discharge using DOLLOP to infuence the plasma-wall

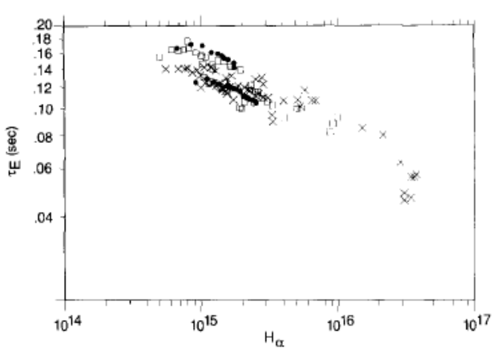

Fig. 7. The energy confinement time of supershot plasmas during the beam heating plotted against the hydrogen influx, Hα.

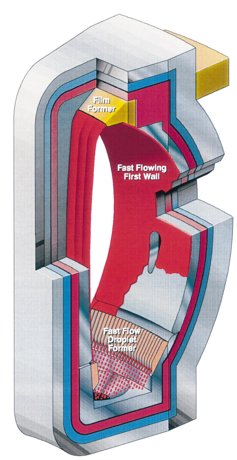

Fig. 8. Conceptual sector schematic of CLiFF implementation in ARIES-RS reactor.

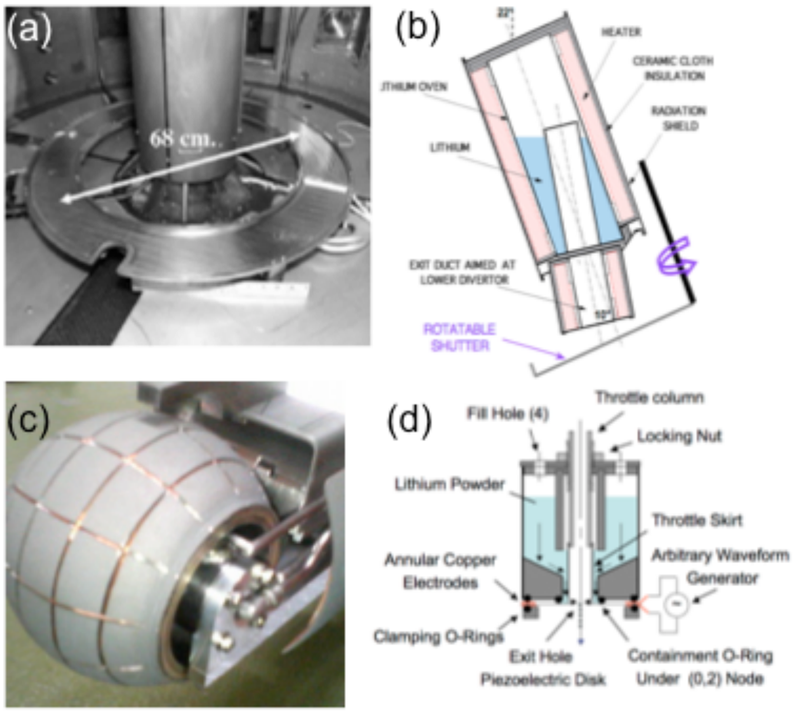

Fig 9. (a) Liquid lithium tray in CDX-U. (b) Lithium evaporator (LITER) for NSTX. (c) Rotating in-situ lithium coated drum limiter in CPD. (d) Lithium dropper for NSTX and EAST.

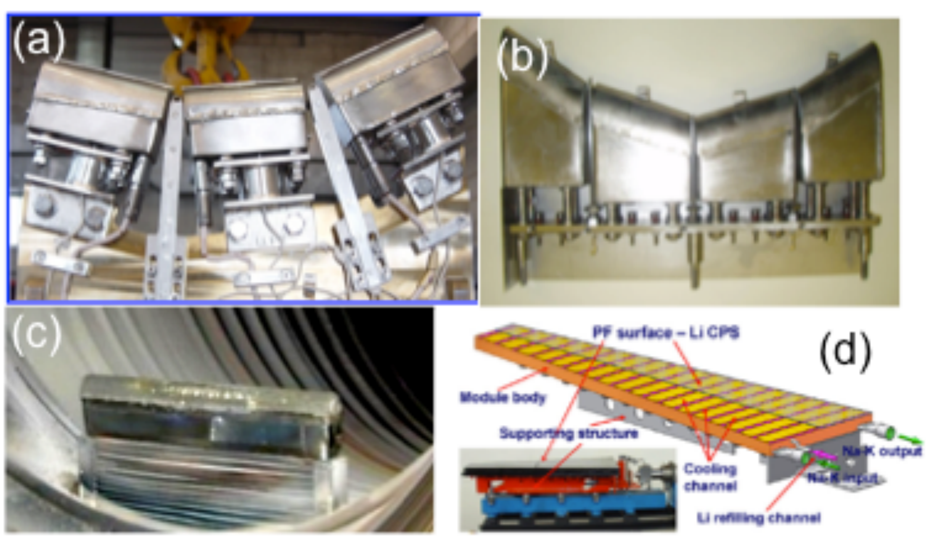

Fig 10. CPS systems. (a). Active Capillary-Porous System (CPS) installed in FT-U. (b) Liquid lithium structure in TJ-II. (c) Li Limiter in T-11M. (d) Design of the lithium in-vessel unit of KTM.

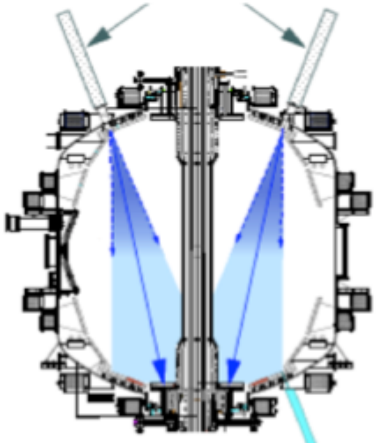

Fig. 11. A schematic of the lithium evaporators (LITERs) injecting vapor which condenses on the room-temperature plasma facing components in the lower part of the vacuum chamber, including the lower divertor plates.

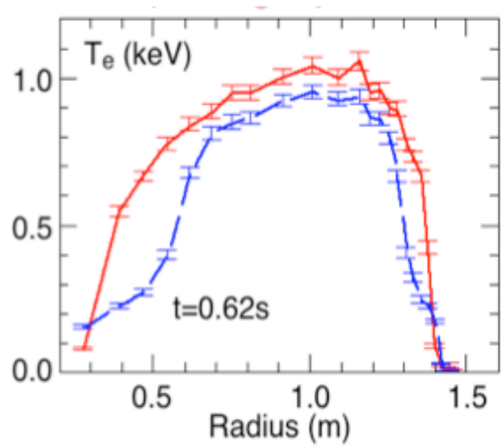

Fig. 12. Radial profiles of electron temperature before (blue) and after 260 mg lithium deposition (red) in the NSTX H-mode discharges.

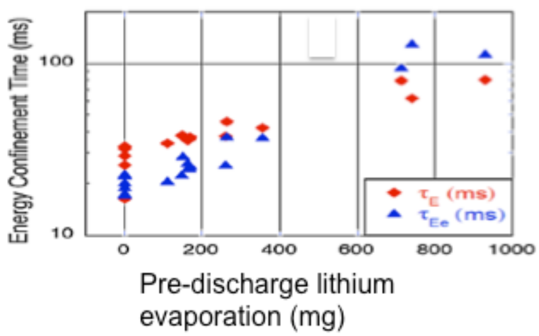

Fig. 13. Total and electron energy confinement time as a function of pre-discharge lithium evaporation.

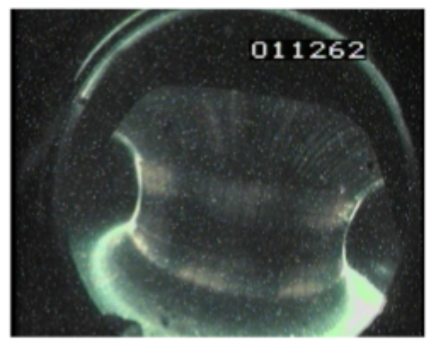

Fig. 14. Enhanced Li injection forms a ‘radiative’ toroidal Li limiter in FT-U.

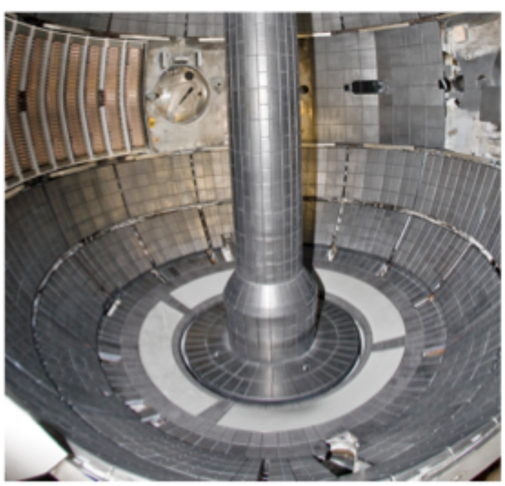

Fig. 15. Photo of the interior of NSTX before the start of the 2010 experimental campaign. Shown is the location of the Liquid Lithium Divertor (LLD).

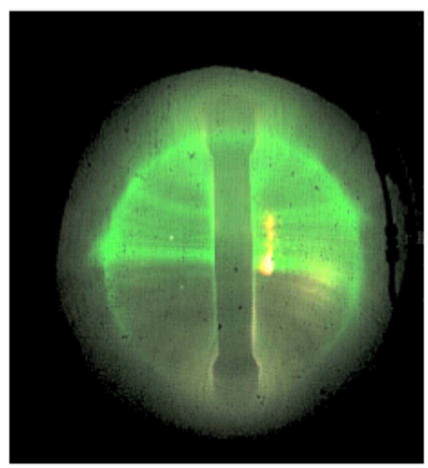

Fig. 16. Enhanced Li injection with lithium dropper forming lithium “radiative mantle” in NSTX.

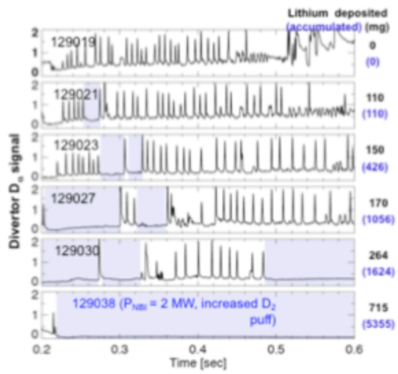

Fig.17. Temporal edge D-alpha signal for various lithium deposition rate. The regularly occurring spikes represents the Edge Localized Modes (ELMs).

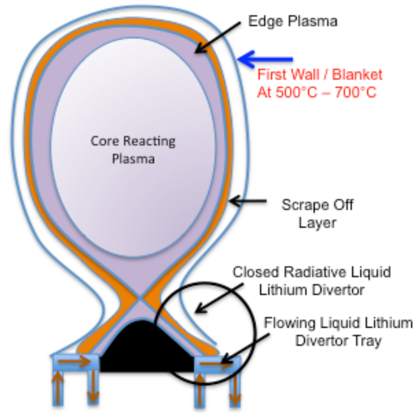

Fig.18. Schematic view of a possible liquid lithium divertor system.

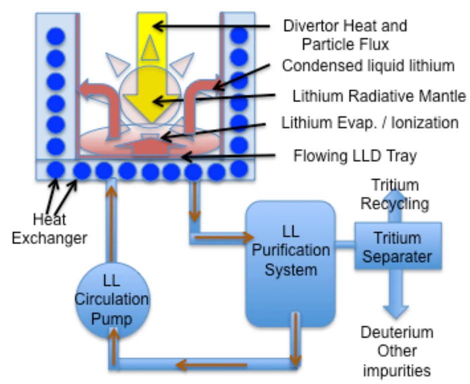

Fig. 19. Schematic view of closed radiative mantle based liquid lithium divertor with a closed loop liquid lithium purifying and tritium extraction system.

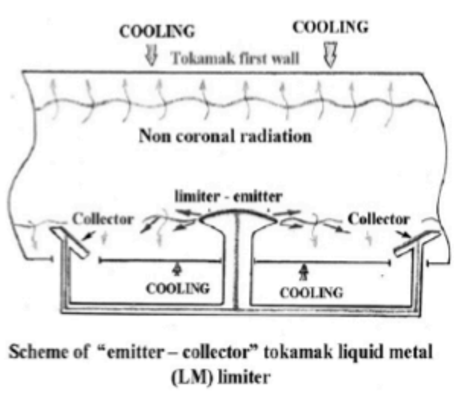

Fig. 20. Li-emitter- collector scheme.

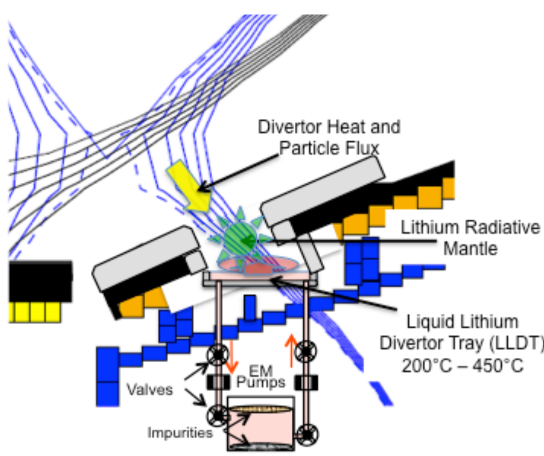

Fig. 21. Schematic view of a possible radiative mantle based closed liquid lithium divertor system for NSTX-U.

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