LiWall Fusion (LiWF) and its Three step R&D Program

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1 Introduction *(to ToC)*

The Li Wall Fusion concept originated on Dec.21, 1998, when Sergei Krasheninnikov instantaneously replied "The plasma temperature will be flat" to my question "What would happen if lithium layer on the walls will absorb all the hydrogen". At that time T11-M experiments had shown the ability of a Li surface to completely deplete the discharge. The compatibility of Li with a high temperature plasma was also understood from D.Ruzic's group's measurements of sputtering. Now the concept, named a "LiWalls" or "LiWF" is complete, self-consistent and has a science-based strategy.

Since 1999, when it was presented for the first time, LiWF has had important partial validations, direct or indirect, of its theoretical basis by experimental results from DIII-D (discovery of QHM, RMP experiments), T11-M, CDX-U, FTU and, recently NSTX as well as in technology experiments at the University of Illinois Champaign, UCSD, SNL.

The LiWF concept is ready to go with a strategy toward a $P_{DT} = 0.2 - 0.5$ GW Reactor Development Facility (RDF), on a time scale competing with ITER. (1)

The concept is presented in details at `http://w3.pppl.gov/~zakharov/` in several presentations on April 11, 2007 (PPPL Colloquium), November 2, 2006 (ASP Meeting), etc.

Completion of the LiWF concept revealed how non-scientific and irrational is the presently dominated concept of magnetic fusion. It is referred here as the "BBBL-70s" (standing for "The Bib$_{\rm b}$le of the 70s").

Formulated in the 70s, it was eventually canonized by the DoE OFES program as a sort of religion with a lot of unjustified dogmas.

2 The plasma physics basis of LiWF *(to ToC)*

The essence of the LiWall plasma regime is very simple: (a) pumping plasma facing components (PFC), and (b) core fueling by Neutral Beam Injection (NBI).

The combination of the Lithium Pumping Divertor (LPD), which absorbs hydrogen (H,D,T) particles from the plasma, and core fueling by NBI can create a unique situation *with no cold particles*, which would destroy plasma thermal equilibrium. As a result, the plasma goes into the most relaxed state with a flattened temperature determined simply by

$$E_{NBI} = \left(\frac{3}{2} + 1\right)(T_i + T_e), \quad T_e < T_i, \quad \text{for} \quad T_e \simeq T_i = 16, \quad E_{NBI} = 80 \text{ keV}, \tag{2.1}$$

where E_{NBI} is the beam energy, T_i, T_e are the ion and electron temperatures. The factor 3/2 comes from the definition of the temperatures. The only effect of plasma physics here is in the addend 1, the exact value of which depends on the distribution function.

In the LiWall regime the temperature across the plasma $(T_e + T_e) \simeq 2/5E_{NBI}$ is determined by E_{NBI} independent of plasma physics details. Plasma is always in the hot-ion mode. (2)

2.1 Confinement regime. Reference Transport Model (RTM) *(to ToC)*

The mean free path λ_{D^+} of the D-ions

$$\lambda_{D^+,m} = 121\frac{T_{keV}^2}{n_{20}} \tag{2.2}$$

indicates the free streaming of plasma particles to the pumping surface, and transition from diffusive to convective transport right at the last closed surface to the pumping surface. The temperature T_{edge} at this transition is determined by

$$\frac{5}{2}\Gamma T_i \simeq P_{total,i}, \quad \frac{5}{2}\Gamma T_e \simeq P_{total,e}, \tag{2.3}$$

where Γ is the particle flux and $P_{total,i/e}$ are the volume integrated heating powers for the plasma species. This equation serves as a boundary conditions for transport in the core as soon as the mean-free path exceeds the connection length in SOL.

The temperature pedestal T_{edge} is determined by the particle flux and the total heating power, rather than by the local transport properties at the edge of the thermo-conduction zone. (3)

Recently, this understanding, which was one of contributors in developing the LiWF concept in 1998, has been unambiguously confirmed by experiments on DIII-D with Resonant Magnetic Perturbations of the magnetic configuration near the separatrix. The thermo-conduction coefficients were significantly enhanced, but even so, the electron temperature pedestal was totally untouched. This result has confirmed the scientific basis of the LiWF confinement. On the other hand, it is in sharp conflict with the notion of the mysterious "edge transport barrier" of BBBL-70s since the discovery of the H-mode in the early 80s.

With pumping walls the temperature profile in the core adjusts itself to eliminate thermo-conduction energy losses (as well as temperature-gradient driven turbulence).

The LiWF relies on the best possible confinement regime, when energy losses are determined by the particle diffusion, which is limited by the best confined component. (4)

In contrast, the BBBL-70s concept imposes a low edge plasma temperature, thus automatically leading to turbulence driven energy losses. The anomalously high thermo-conduction of electrons has remained intractable in the BBBL-70s fusion for 40 years already.

In LiWF, assuming electrons not confined at all, the diffusion is determined by ions, which in the absence of turbulence are neoclassical. In NSTX the ions are neoclassical even in the presence of turbulence.

The predictive Reference Transport Model (RTM) of LiWF is simply $\chi_e = \chi_i = D = \chi_i^{neoclassical}$. RTM reflects the fact that LiWF is the only magnetic fusion concept, which is insensitive to the anomalous behavior of electrons. (5)

RTM model is quantitatively consistent with data from the CDX-U (PPPL) experiments, where confinement has been *enhanced by a factor of 4-6 with a liquid lithium surface.*

2.2 The super-critical confinement regime. *(to ToC)*

The elimination of the anomalous electron transport impliess such good confinement that LiWF will not need plasma heating by fusion α-particles. A super-critical regime with plasma parameters exceeding the ignition criterion:

$$f_{pk} \langle p \rangle \tau_E > 1, \quad Q_{DT} \equiv \frac{P_{DT}}{P_{NBI}} = \frac{5P_\alpha}{P_{NBI}} > 5 \tag{2.4}$$

becomes possible for LiWF. Here, f_{pk} is the peaking factor converting the averaged plasma pressure $\langle p \rangle$ into the fusion relevant pressure $p_D p_T$, $P_{DT}, P_\alpha, P_{NBI}$ are the NBI, fusion powers and its α-particle fraction.

LiWF makes the hot-ion regime, the best in present machines, perfectly relevant to the power reactor and its R&D. (6)

In contrast, even for ITER, the BBBL-70s does not have the database for "hot-electron" regime in its reactor or "burning plasma" projections.

The RTM model predicts $Q_{DT} \simeq 40 - 50$ in a Spherical Tokamak (ST) with $P_{DT} = 0.2 - 0.5$ GW with plasma volume 27 times smaller than that of ITER. The α-particles can be thrown away from the plasma on their first orbits.

By this elimination of α-particles from the plasma the LiWF regime eliminates numerous problems associated with them. For the first time, this also suggests a real solution to the otherwise intractable power extraction problem. Plasma temperature, density and fusion power are controlled exclusively by NBI. (7)

2.3 Pumping and power extraction. *(to ToC)*

Technically the pumping surface will be a thin ($\simeq 0.1$ mm) layer of liquid Li slowly ($V_{Li} \simeq 1$ cm/s) moving due to gravity, $\mathbf{j}\times\mathbf{B}$ force, or Marangoni effect, and replenished by an external supply of fresh Li. The flow pattern is controlled by wicking and capillary forces. The actively cooled guide plates of such a Lithium Pumping Divertor (LPD) keep the surface temperature of the Li surface below evaporation limit $T_{Li} < 400 - 450^oC$.

The rate of replenishment is not a problem. E.g., for pumping the ITER discharge it would be only 10L/hour, with a total inventory of Li inside the chamber of 3-4 L.

The limitation on the Li surface temperature represents only a design issue. The power extraction capabilities are determined by the coolant side of the PFC, rather than by the plasma facing surface temperature.

The LPD is compatible with both pumping and power extraction requirements of the present machines and the reactor. Only $P_{NBI} \ll P_\alpha$ should be extracted by the PFC.	(8)

2.4 Stability. *(to ToC)*

A flattened temperature profile as well as a high fraction of (or complete) non-inductive current drive, which is simplified for the LiWall regime, eliminates the $q = 1$ surface and the possibility for sawteeth or internal reconnection events (which remain unpredictable for BBBL-70s since 1971).

The phenomenal discovery of the Quiescent H-Mode (QHM) on DIII-D in 2000, which in many aspects is a prototype of the LiWall regime, demonstrated elimination of ELMs in the situation with the high temperature pedestal ($T_{edge,e} > 1,\ T_{edge,i} > 4$ keV).

This observation, crucial for LiWF, has been understood only recently (2005), when it was noticed theoretically that a plasma limited by the separatrix will be stable to ELMs, if the current density at the edge is finite.

This conclusion destroyed a long standing misconception of BBBL-70s about the peeling mode instability.

The theory revealed a large operational space for LiWF with no ELMs, sawteeth. Greenwald density limit does not exist for LiWF.	(9)

Experimentally, the CDX-U plasma became quiescent as soon as lithium tray was introduced into the chamber. ELMs disappeared on NSTX after lithium conditioning by evaporation. DIII-D and JET data on ELM-free regimes are consistent with the LiWF stability concept. FTU exceeded the Greenwald density limit by a factor 1.4 in the first experiments with Li. Not this factor is approaching 2.

2.5 Stationary plasma. *(to ToC)*

According to theoretical simulations, high beta spherical tokamaks in the LiWall regime can have excessive bootstrap current (whose profile can be controlled by NBI). The NBI itself can generate the current in the plasma center where the bootstrap current is deficient.

The thermo-force ($\propto Z^2$), driving ionized impurities from PFC to the plasma core, is intrinsically absent in the collisionless scrape off layer (SOL). The QHM regime in DIII-D showed no indication of blob formation and interaction of the high-edge temperature plasma with the side walls.

Core fueling creates the situation where particles are going from the plasma center to the edge, rather than vice versa.

LiWF is unique in having a science based concept of the stationary plasma.	(10)

In this regard, in BBBL-70s with its edge fueling, blobs and instabilities, everything is upside-down. Whatever is generated between plasma and walls sooner or later will be in the core.

3 Reactor R&D aspects *(to ToC)*

The reactor R&D objective of magnetic fusion is to develop (a) the high fusion power density ($P_{DT} \simeq$ 10MW/m^3) regime, (b) the long lasting First Wall (FW), which is the first 15 cm of material faced by 14

MeV neutrons, and (c) tritium cycle.

The reactor strategy at the present point of fusion development is determined by a simple number

$$15\ \mathrm{MW} \cdot \mathrm{year/m}^2 \equiv 1\ \mathrm{kg}_{tritium}/\mathrm{m}^2, \tag{3.1}$$

which translates the reference neutron fluence necessary for damaging the first wall samples into consumption of tritium. Large machines (wall surface $S_{ITER} \simeq 650\ \mathrm{m}^2$) are obviously irrelevant for reactor R&D.

Only Spherical Tokamaks in the LiWall regime are suitable for the mission of a Reactor Development Facility (RDF). Existing technology is sufficient for RDF based on the LiWF concept. (11)

Incapable of meeting the reactor R&D requirement, and with no real reactor concept, the BBBL-70s substitutes for fusion objectives by the politically motivated "road map" through the "burning plasma", light weighted for the practical fusion development.

4 Three step program toward the Reactor Development Facility. ***(to ToC)***

The program includes a motivational step (ST0), based on the existing NSTX (modified for Lithium Pumping Divertor) and LTX in PPPL, and three new STs: two of them (ST1,ST2) with DD plasma and one (ST3) with DT plasma, which will provide neutron flux and fluence for developing the FW and tritium cycle.

The LiWF is the only concept which does NOT need the DT plasma for development of the plasma regime for RDF. PPPL has everything necessary for the motivational and ST1,ST2 steps. (12)

4.1 Transition to molten lithium in NSTX ***(to ToC)***

NSTX device as well as extensive lithium experience in PPPL are well suited for the motivational step. In 2006, 2007, NSTX conducted Li conditioning experiments using intensive evaporation of lithium. Being successful in enhancing the energy confinement and elimination of ELMs, these experiments failed in pumping the plasma to the extend necessary for LiWall regime. The lithium pumping divertor with molten lithium is desperately needed for NSTX to be relevant to this mission. In this regard, the Lithium Loaded Target Plate (LLTP) was proposed in 2006 and comprehensively analyzed.

I am asking FESAC panel to revitalize the 2008 NSTX program with the installation of the LLTP. There is no technical or design issues with LLTP. (13)

The milestone for this experiment is to reproduce the T11-M, CDX-U, FTU plasma pumping capabilities. The priority is to develop, for the first time, the plasma pumping by lithium surface in the SOL area and get all characteristics on electric currents, power and particle deposition. *The mission is to clarify the device operational compatibility with the molten Li.*

The successful accomplishment of this mission should be the end of the NSTX program, while the in-vessel components of the same device should be made compatible for the long lasting LPD. Accordingly the new program should be initiated with a new mission and milestones as a motivation for three step RDF program.

In parallel, LTX will perform the compemenary studies of the LiWall regime with plasma pumping by the side walls.

4.2 ST0, LTX and ST1-ST3 steps of the program *(to ToC)*

The following Table summaries the devices and the milestones, priorities and mission of their programs.

Steps toward RDF	Milestone	Priorities and Mission
NSTX with molten LLTP (Li Loaded Target Plate), B=0.4 T, $I_{pl} = 1$ MA, A=1.2, $R_{outer} = 1.5$ m	Reproduce T11-M, CDX-U, FTU plasma pumping experiments	Plasma pumping. Low energy NBI. Stability. Clarify the system compatibility with molten Li
ST0 (modified NSTX): B=0.3-0.5 T, I_{pl}=0.7-1 MA, A=1.2, $R_{outer} = 1.5$ m. LTX (modified CDX-U) B=0.3 T, I_{pl}=0.3 MA, A=1.6, $R_{outer} =$ 0.65 m.	Achieve RTM-like confinement: $\tau_E \longrightarrow 2-3 \times \tau_{E,NSTX}$.	Plasma boundary. Stability. Start-up. Core fueling by low energy NBI. Collisionless SOL/PFC interaction. Role of C-walls. Creating a design concept of LPD for ST1.
ST1: B=1.5 T, I_{pl}=2-4 MA, A $\simeq 5/3$, $\beta = 0.2-0.3$, $R_{outer} =$ 1.65 m	Achieve Super-critical regime: $Q_{DT}^{equiv} > 5, \quad f_{pk} p \tau_E > 1$	Plasma boundary. Stability. Physics and technology of LPD. Secondary electron emission. Role of TEM. Creating concept of a Startup and stationary LPD
ST2: DD-prototype of ST3, B=3 T, I_{pl}=4-8 MA, A $\simeq 5/3$, $\beta = 0.3-0.4$, $R_{outer} = 2$ m, $Vol_{plasma} \simeq 30$ m^3	Achieve RDF stationary regime: $Q_{DT}^{equiv} = 30-50$	High $\beta \simeq 30-40$ %. Noninductive current drive. Integrate the stationary plasma regime for RDF. Assess the feasibility of DD fusion.
ST3: DT neutron source. B=3 T, I_{pl}=4-8 MA, A $\simeq 5/3$, $R_{outer} = 2$ m, $Vol_{plasma} \simeq 30$ m^3	Achieve DT-stationary regime: $Q_{DT} = 30-50, \; P_{DT} = 0.2-0.5$ GW	Power extraction from α-particles, He exhaust. Integrate the stationary neutron producing regime for RDF mission.

5 Summary *(to ToC)*

LiWF targets a practical, highly needed tool for power reactor development, i.e., the 0.2-0.5 GW neutron source as a Reactor Development Facility in the form of an ST.

In the present environment of desperate energy needs, LiWF relies on bootstrapping its program and funding ($\simeq$\$2-2.5 B total) after demonstration of new confinement and stability regime at the ST0 stage. (14)

In this regard the help of FESAC to initiate the program at the NSTX level is highly appreciated.