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Ignited Spherical Tokamaks for development of power reactor[1]

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Abstract

On the way to power reactors, there are three, mutually linked objectives, specific to magnetic fusion, i.e., (a) development of the high fusion power density operational power reactor regime (OPRR), (b) design and development of low activation first wall (i.e., the first 15 cms of the material faced by 14 MeV neutrons) together with power extraction and helium ash exhaust, and (c) development of the tritium cycle.

This triple-objective cannot be meet based on present reactor concept (essentially non-existing). Because of lack of tritium only compact devices are suitable for reactor development, and the only candidates are spherical tokamaks (ST).

For the purposes of the first wall R&D and accumulating the necessary 15 MW·year/m^2 fluence of 14 MeV neutrons even ST require a special plasma regime, which would provide a self-generating plasma current, ignition and a self-sufficient tritium operation.

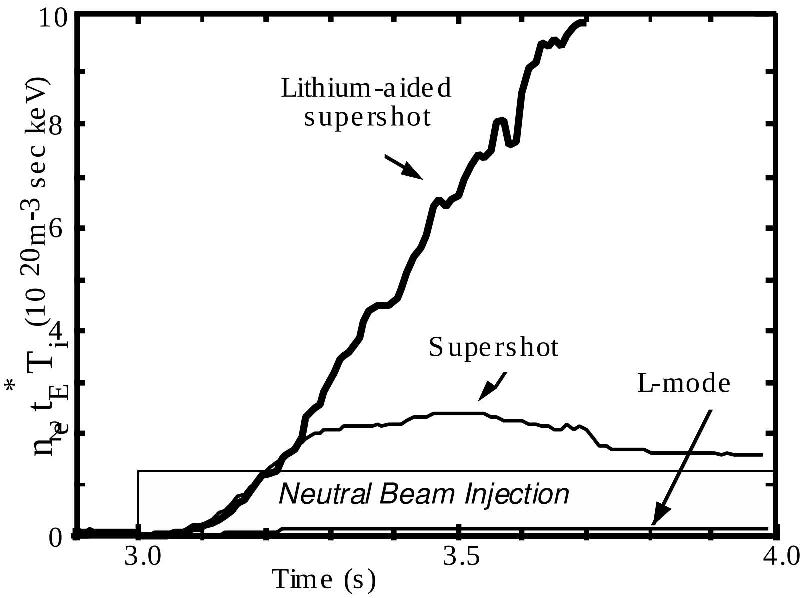
The talk compares two approaches for magnetic fusion: (a) the conventional one, based on the high recycling plasma, and (b) the LiWall approach, which utilizes the unique lithium capacity of pumping hydrogen isotopes.

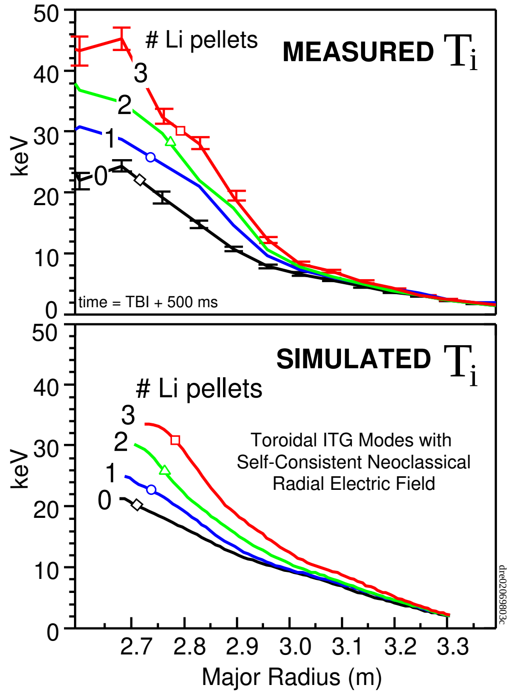
Despite its dominance during the last 35 years, the conventional approach did not resolve several basic problems of magnetic fusion even at the plasma physics level. It never approached the real, nuclear issues of the fusion power reactor. In contrast, the 6.5 years old LiWall concept (1999) has opened a way for achieving the triple objectives of magnetic fusion in a form of Ignited Spherical Tokamaks (0.5 GW of fusion power in 30 m^3).

Contents

TFTR discovered the effect of Lithium conditioning

It was the most important factor in TFTR performance





(TFTR # 83546 D.Mansfield, C.Skinner)

The increase in performance with increase in amount of lithium at the plasma edge

has never been saturated

The importance of Li for tokamak plasma was recognized in 1998

Two distinct approaches to tokamak fusion exist now:

1. Conventional (dominant) approach:

(a) *After 35 years did not resolve the issues of confinement, stability, power extraction.*

(b) *Postponing all unresolved problems to the future reactor stage.*

(c) *Short of developing a consistent fusion power reactor concept.*

2. "LiWall" approach based on unique Li ability to pump H,D,T.

(a) *Opened a way for resolving confinement, stability, power extraction issues at the pre-reactor stage.*

(b) *Based on present understanding of tokamak plasma physics, has shown the possibility of tokamak regimes with reactor relevant performance.*

(c) *Resulted in a specific Ignited Spherical Tokamak reactor development concept, consistent with basic fusion reactor physics and technology.*

Two approaches diverge at very fundamental level of fusion

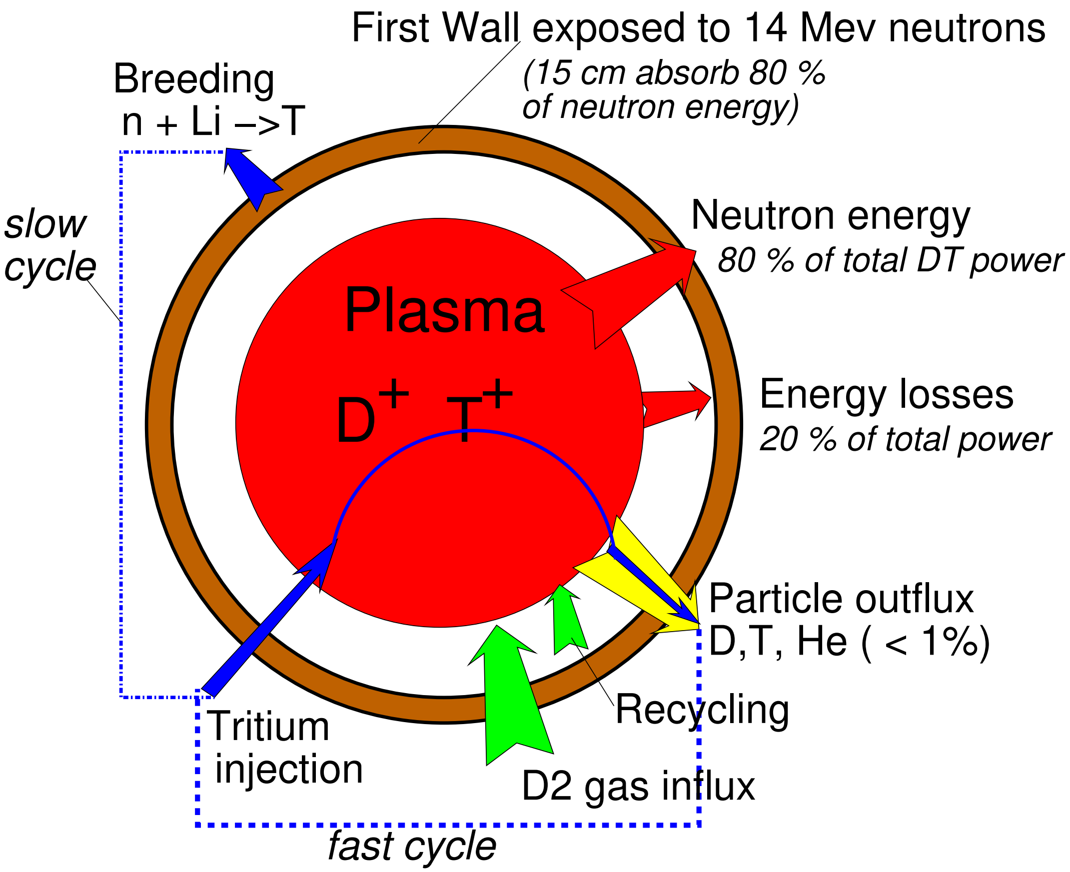
Only $D + T \rightarrow He + n$ reaction is economically suitable for the power production. All others have too small reaction cross-sections.

$$\begin{array}{ccccccc} D^{+} & + & T^{+} & = & He^{++} & + & n \\ \text{deuterium} & & \text{tritium} & & \alpha-\text{particle} & & \text{neutron} \\ 15-25 \text{ keV} & & 15-25 \text{ keV} & & 3.5 \text{ MeV} & & 14.1 \text{ MeV} \\ & & & & 20\ \%\ \text{of energy} & & 80\ \% \\ & & & & \text{left inside} & & \text{go outside} \\ & & & & \text{the plasma} & & \text{the plasma} \end{array} \tag{2.1}$$

Each neutron should be converted into tritium outside the plasma

$$n + \{Li,\ Be,\ Pb,\ etc\} \rightarrow T + \ldots \tag{2.2}$$

The problems of First Wall and Tritium Cycle are specific for DT fusion



Two loops of tritium cycle are present. First wall is being damaged by 14 Mev neutrons

Neutron fluence $\simeq$15 MW$\cdot$year/m^2 in necessary for destruction as well as for designing the First Wall of the reactor

15 MW$\cdot$year/m^2 correspond to consumption of 1 kg/m^2 of tritium.

Frequently referred as an "inexhaustible" energy source, in fact,

Fusion HAS NO tritium fuel even for designing the reactor

(*E.g., with ITER wall surface $\simeq$ 650 m^2 650 kg of T would be consumed for designing the First Wall*)

Basic reactor physics predefines “triple” unseparable objectives of magnetic fusion and its strategy

1. **Achieving the high power density Operational Power Reactor Regime (OPRR) for neutron production.**
2. **Designing the First Wall (FW) the most complicated element of the power reactor.**
3. **Tritium Cycle (TC).**

Existing concepts of "burning plasma", VNS, CTF, etc, which try to separate these objectives (in order to "simplify" the job), have little sense for reactor development.

Also targeting the sole ignition, which is a short, low power density phase, preceding OPRR, would not contribute much to the reactor R&D.

These is no step by step approach in OPRR-FW-TC development

A fusion reactor should work near "ignition" criterion

At optimal plasma temperature for fusion power the criterion is very simple

$$\boxed{\langle p\rangle\,\bar\tau_E^* \simeq 1 \qquad [\mathrm{MPa}\cdot\mathrm{sec}], \qquad \bar\tau_E^* \equiv f_\alpha f_{pk}\bar\tau_E}$$

Here,

$\langle p\rangle$ - *volume averaged plasma pressure, accounting for pressure p_e of electrons and all kind of ions, i.e., deuterium p_D, tritium p_T, helium p_{He}, impurities p_Z:*

$$p = p_D + p_T + p_e + p_{He} + p_Z$$

$\bar\tau_E$ - *overall energy confinement (accounting for radiation).*

$f_\alpha \le 1$ - *fraction α-particle energy used for plasma heating,*

$f_{pk} \simeq 1$ - *"peaking" factor*

$$f_{pk} \equiv \frac{\langle 4p_D p_T\rangle}{\langle p\rangle^2} \tag{3.1}$$

Equivalent forms of ignition criterion

$$\bar\tau_E^*\,\langle p\rangle = 1, \qquad \langle n_e T\rangle\,\bar\tau_E^* = 31\cdot 10^{20}, \qquad \beta B^2\bar\tau_E^* = 2.5, \qquad \beta \equiv \frac{2\mu_0\,\langle p\rangle}{B^2}. \tag{3.2}$$

Operational regimes and ignition phase have different contributions to $\langle p\rangle\, \bar{\tau}_E^* \simeq 1$

External power P_{ext} necessary to make ignition

$$P_{ext} > \frac{1}{4}\int_{V_0} P_\alpha dV = \frac{1}{20} P_{DT} \quad - \quad \textit{total fusion power} \tag{3.3}$$

In order to minimize the external power P_{ext}

I gnition requires: ***large*** τ_E^*, ***reduced*** $P_{DT}, \langle p\rangle, \beta$

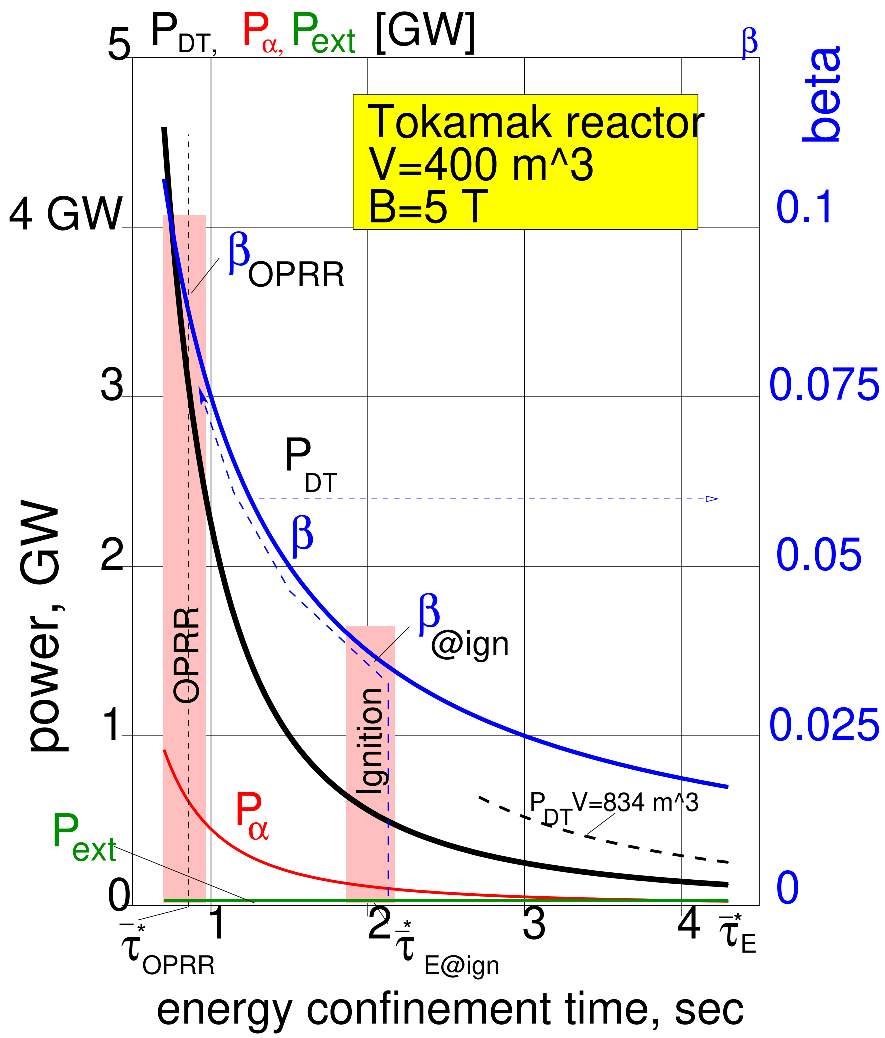
In contrast, for high power production P_{DT}

Operational regime requires: ***high*** $\langle p\rangle$, β ***reduced*** τ_E^*

Total fusion power

$$P_{DT} = 5\int P_\alpha dV = 7.5 f_{pk} \langle p\rangle^2 V_0 = 1.2 f_{pk} (\beta B^2)^2 V_0 = 7.5 f_{pk} \frac{V_0}{\bar{\tau}_E^{*2}}. \tag{3.4}$$

Ignition and OPRR are well separated $\bar{\tau}_E^*$'s in the power reactor



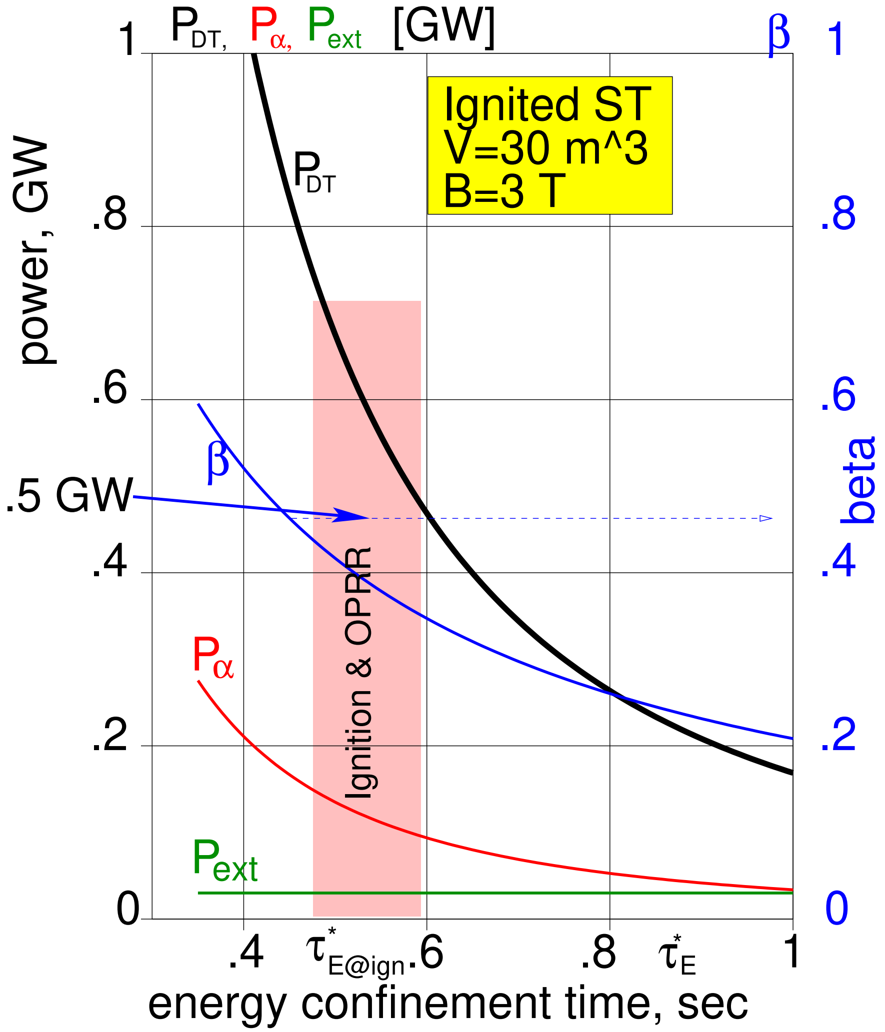
It is not possible to ignite the power reactor at its operation parameters

Between ignition and operation there is a transition phase

At transition phase confinement degradation is limited by

$$\bar{\tau}_E^* \propto \frac{1}{\sqrt{P_{DT}}}$$

Ignited Spherical Tokamaks (IST) serve for reactor development

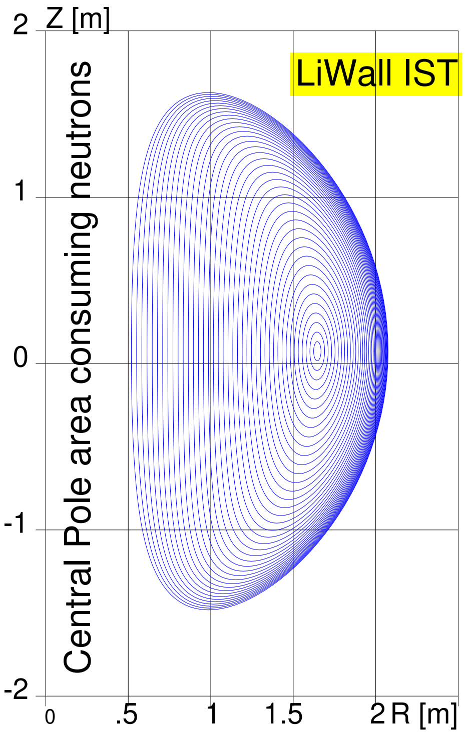


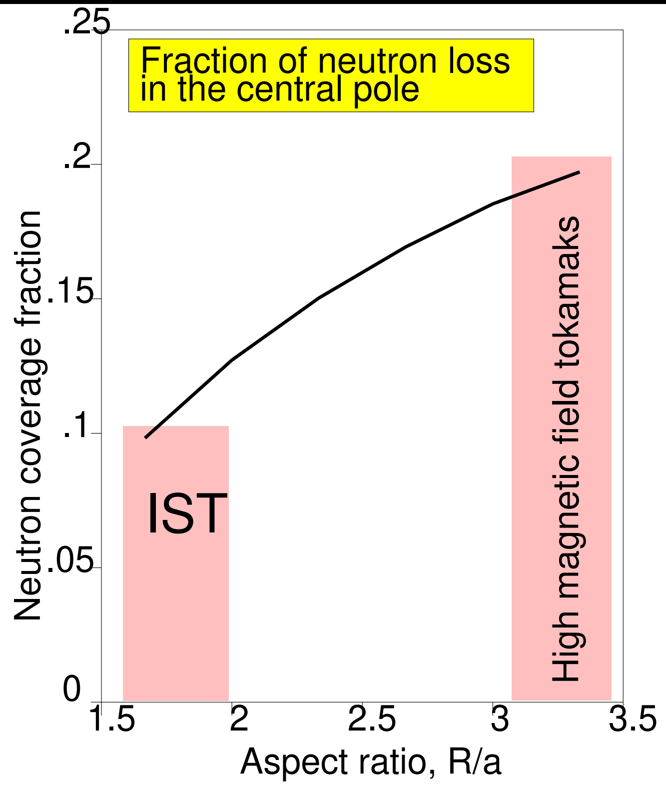
Ignition in IST can be made at operation parameters with a moderate external power

$$P_{ext} \simeq 25-35 \; [MW]$$

Then, all NBI ports can be filled with tritium breeding material

Ignited Spherical Tokamaks are the candidate for reactor R&D





1. *High magnetic fields are not the option for reactor development (unfavorable geometry for neutrons, no data on stability limits, etc.)*
2. *Philosophy of externally driven “Component Test Facility” does not work.*
3. *There is no plasma physics reasons NOT TO ignite the high-beta device.*

Ignited ST suggests use of full FW area for tritium breading

New regimes are required for Ignited Spherical Tokamaks

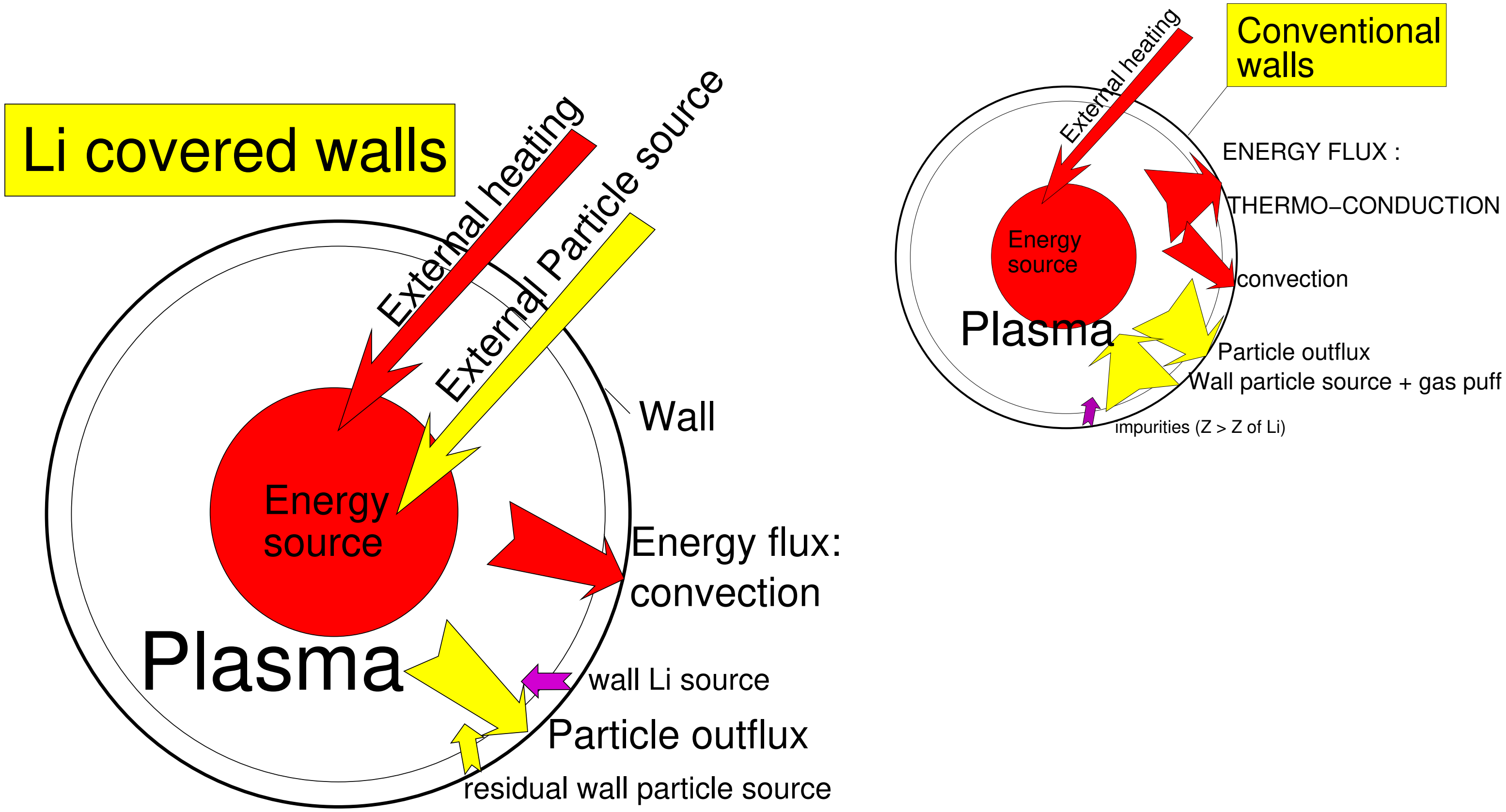
“Conventional”:

- High recycling plasma edge
- Low edge plasma temperature - insufficient plasma performance
- Divertor based concept for particle control
- Localized power extraction from the plasma - not scalable to reactor
- Gas-dynamic concept of He ash pumping

“LiWalls”: utilizes unique properties of Li of pumping H,D,T isotopes

- Low recycling plasma edge
- High edge plasma temperature - new plasma regimes
- High performance
- Consistency with reactor development requirements

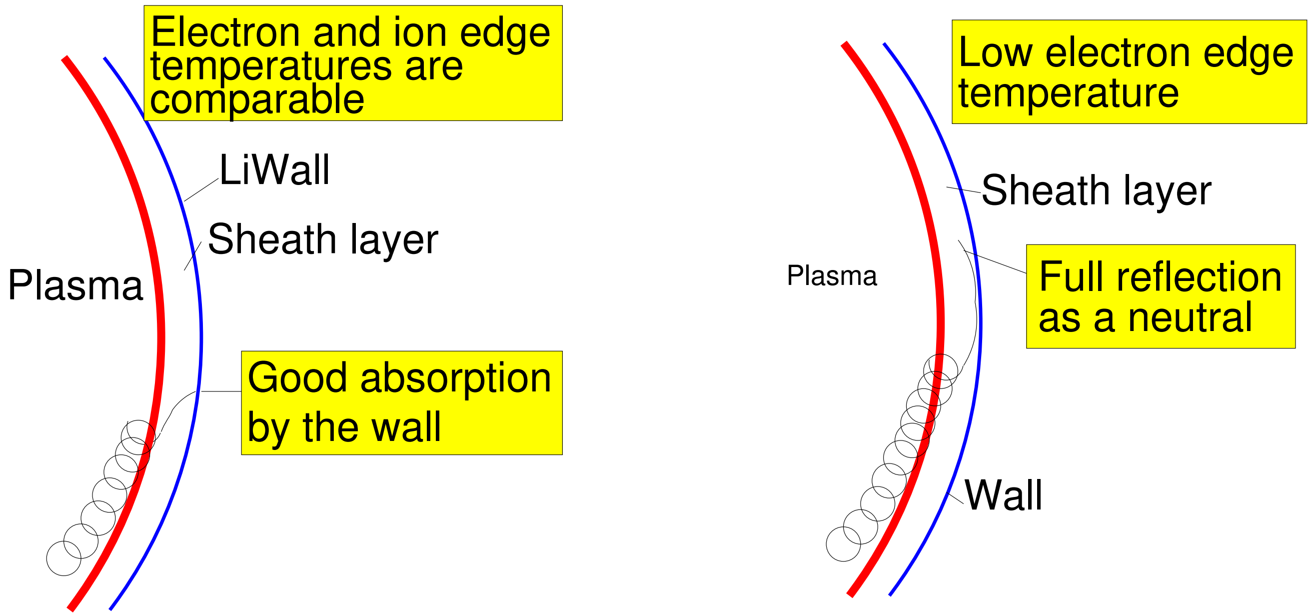
Li is an outstanding getter for hydrogen plasma particles.



Li wall serves as a "trash can" for plasma particles and impurities

He is the only exception

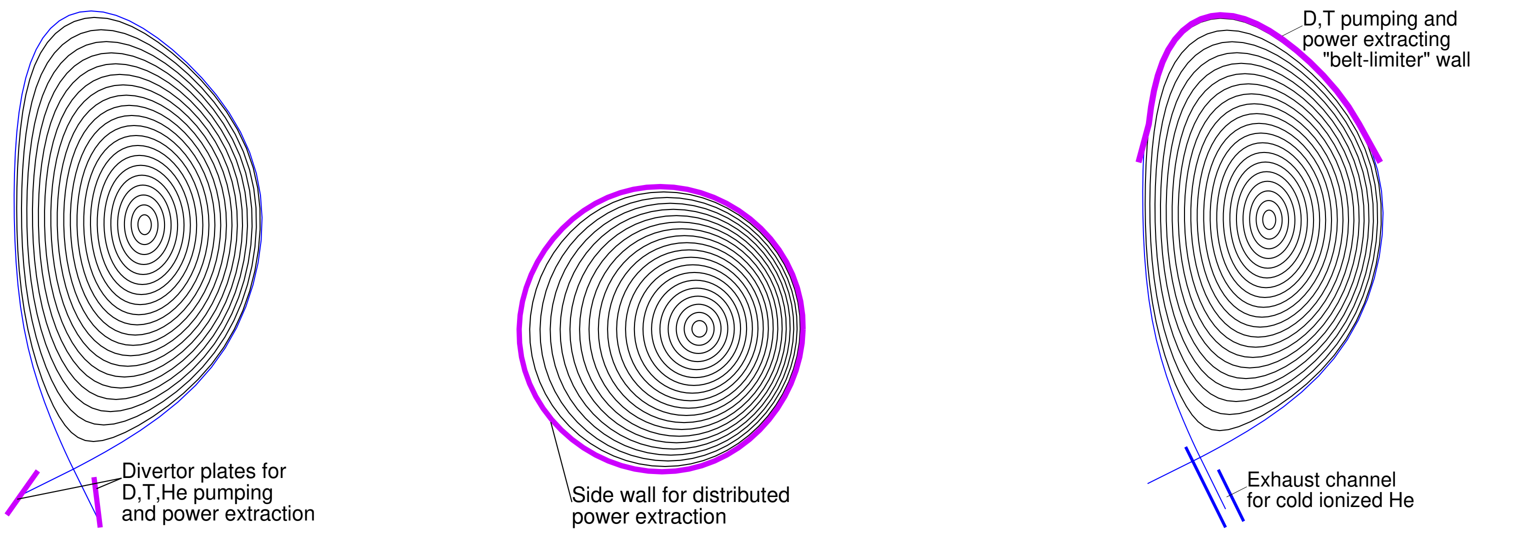
Plasma has to be aligned with the Li wall surface (like belt limiter)



Sheet potential near the walls determines the physics

$$\phi_E \simeq 3T_e/\rho_i \tag{4.1}$$

The entire tokamak program is built around the single idea of a divertor



(a) **(b)** **(c) LiWalls case**

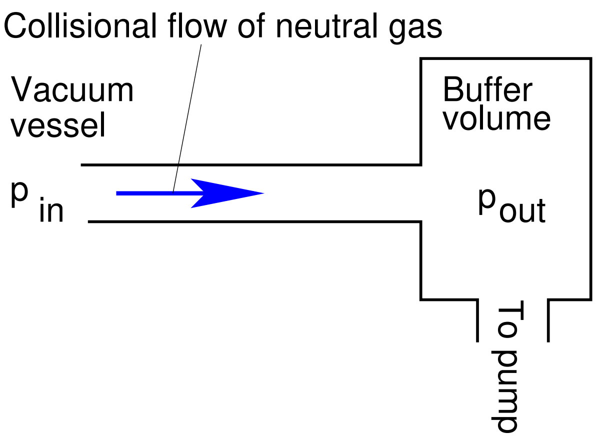
(a) conventional divertor: *all problems are well known; Not scalable to reactor*

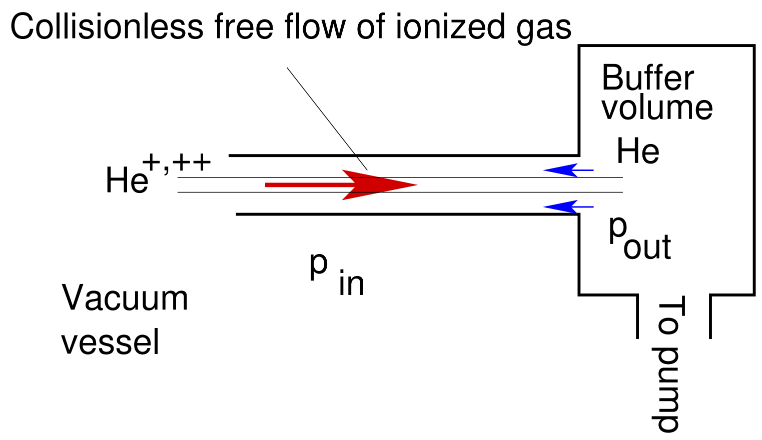
(b) the side walls: *inconsistent with particle, impurities and helium pumping:*

both requiring low edge plasma temperature (turbulence, ELMs, disruptions, etc).

LiWalls absorb the power and D,T from the plasma and automatically distill the Helium ash

"So-far-unbeatable" objection against LiWalls was its "illusory" inconsistency with He pumping





Dominant gas-dynamic scheme:

a) high pressure in the divertor

$$p_{in} > p_{out}$$

b) D,T,He are pumped out together

LiWall scheme:

a) Free stream of He$^{+,++}$ along $\mathbf{B}$,

b) Back flow is limited by

$$\Gamma_{He} = Dn'_x, \quad D = hV_{termal}$$

c) Helium density in the vessel plays no role, while D is in the hands of engineers.

LiWall concept is perfectly consistent with pumping He

Low plasma edge temperature is the major reason of fusion failure as the energy program

At high power, the temperature gradient excites the Ion Temperature Gradient (ITG) turbulence resulting in so-called "profile consistency".

The temperature profile becomes very sensitive to the edge temperature, which has to be kept low (divertor requirements).

Conventional divertor and poor performance of tokamaks are linked to each other

As a result

"Profile consistency" leads to large size tokamak devices

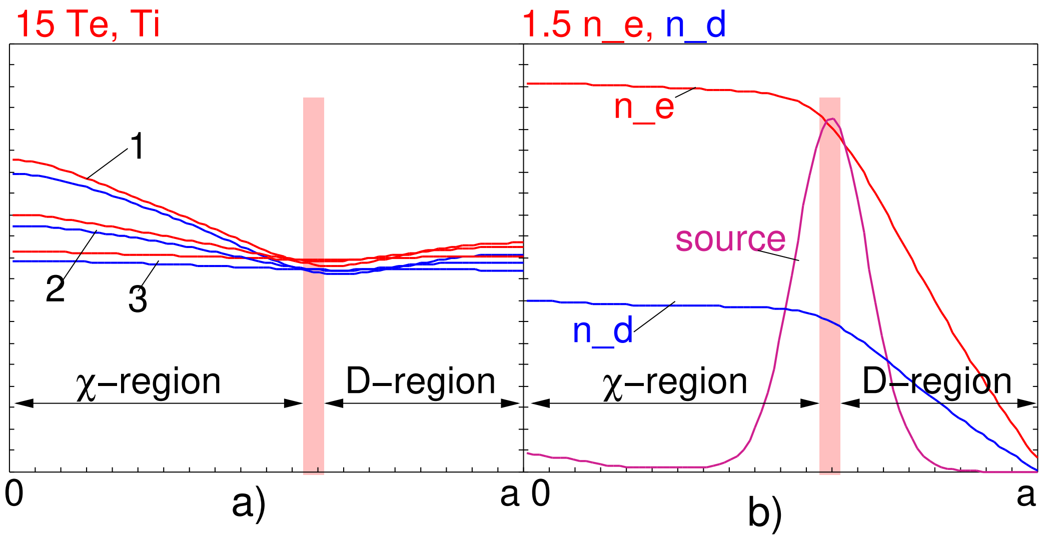
irrelevant to the reactor R&D as well as to the power reactor itself.

ITER is a reference example: low power density, low neutron flux.

Absorbing LiWalls automatically lead to the high edge temperature

The edge temperature is determined by the microscopic particle flux $\Gamma^{micro}_{edge\to wall}$ to the wall. ***In conventional (high recycling) plasma***

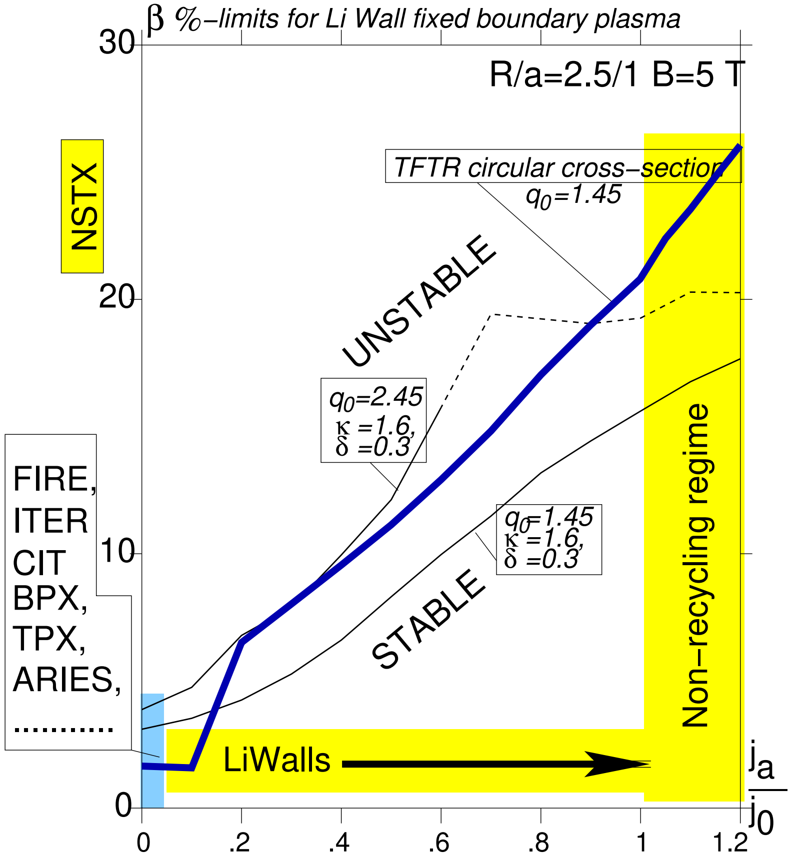
$$T_{edge} \simeq \frac{1}{\Gamma^{micro}_{edge\to wall}} \int P_{heat} dV, \quad \Gamma^{micro}_{edge\to wall} \gg \Gamma_{convective}, \quad T_{edge} \ll T_{core}. \qquad (4.2)$$



χ- and D- confinement regions in the low recycling regime. (a) Electron and ion temperatures for three values of thermo-conduction coefficients. (b) Electron, ion density and the particle source.

With Li Walls $\Gamma^{micro}_{edge\to wall} \simeq \Gamma_{convective}, \quad T_{edge} = \frac{1}{\Gamma_{convective}} \int P_{heat} dV \simeq T_{core}$

With Li surface a stabilizing conducting wall can be placed at the plasma boundary



High edge plasma temperature leads to
(a) high current density at the edge, and
(b) second stability in the plasma core

β - limits for the second stability regime

- *fixed boundary plasma*
- *n=1,2,3 + ballooning modes (DCON,PEST-2,BALLON)*
- *current density with an edge pedestal*

$$j_{\|} = j_a + (j_0 - j_a)\left(1 - \frac{r^2}{a^2}\right)$$

LiWall tokamaks are consistent with high fusion power density

IST can be stable at the reactor relevant plasma pressure

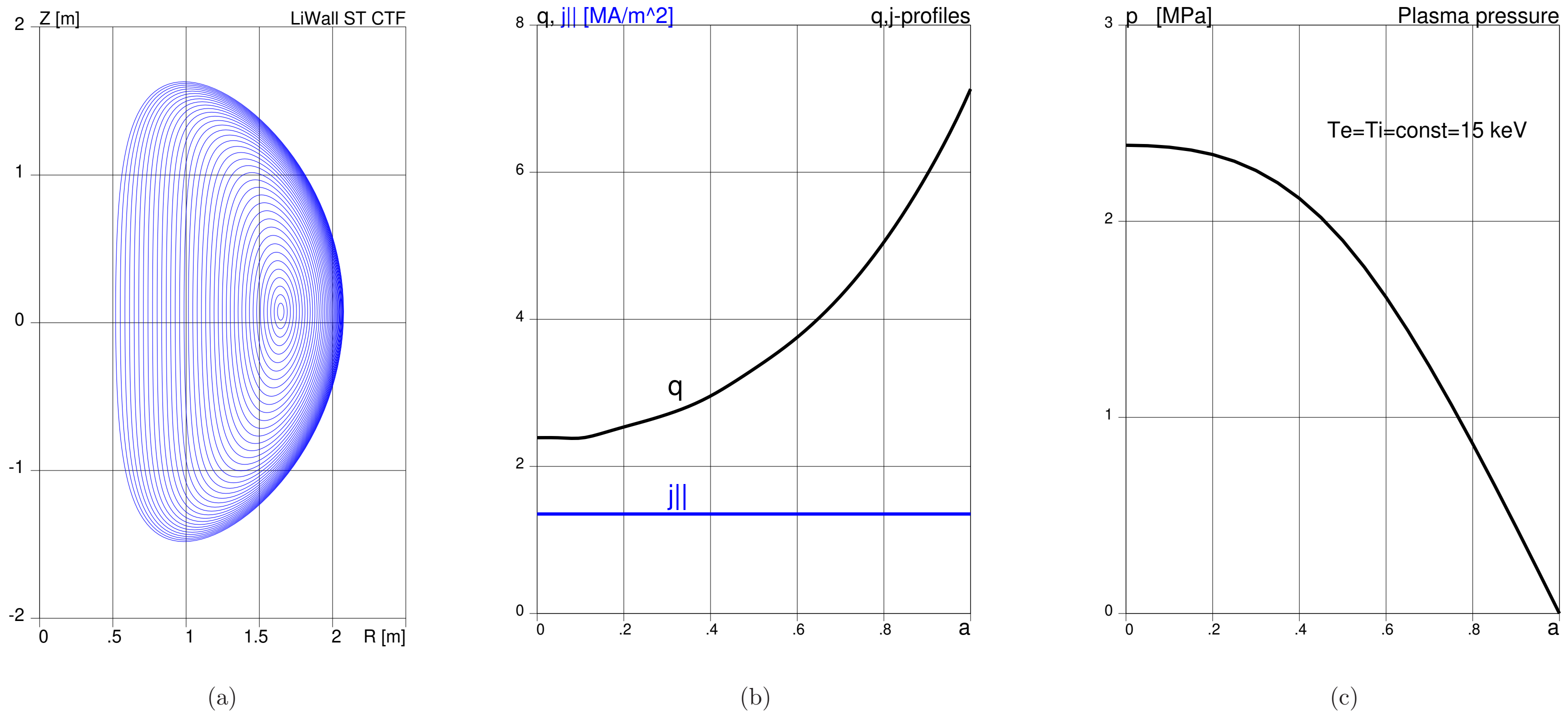
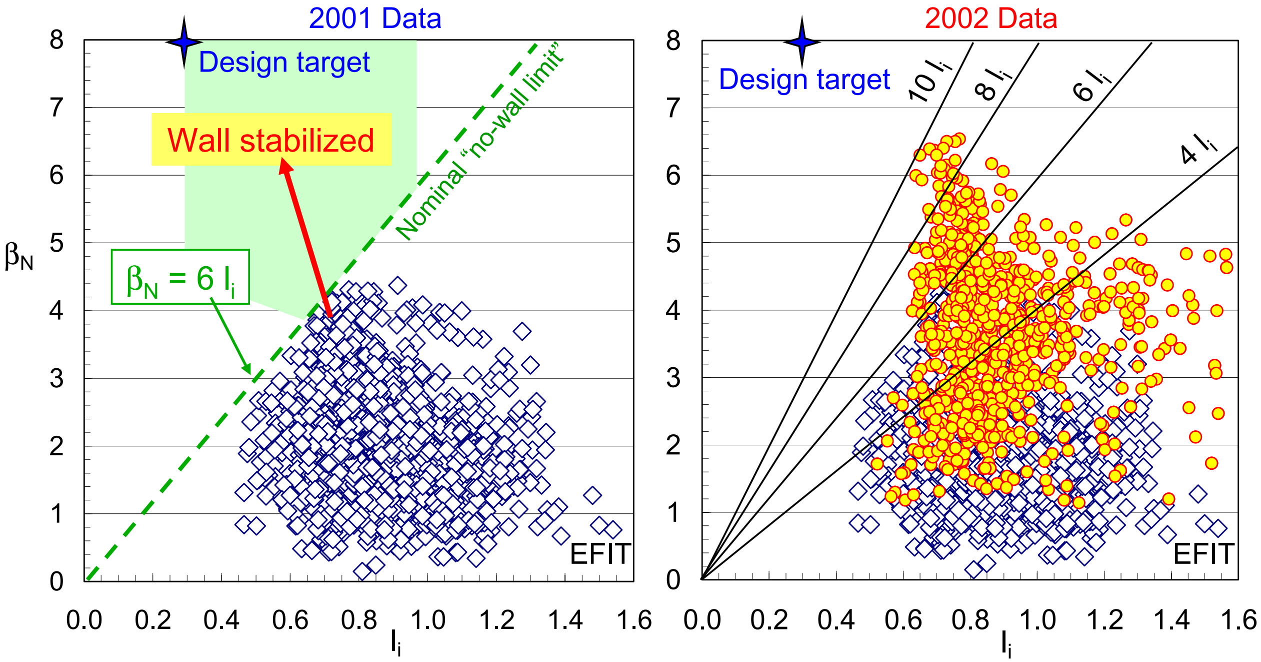


Figure 1: (a) Stable magnetic configuration of Ignited Spherical Tokamak with $I_{pl} = 8.5$ MA, $\beta = 0.46$. (b) Parallel current density and q-profile. (c) Pressure profile (exceeding OPRR level).

High-β of ST overcompensates a lack of magnetic field strength

Plasma operation in low l_i, wall-stabilized space



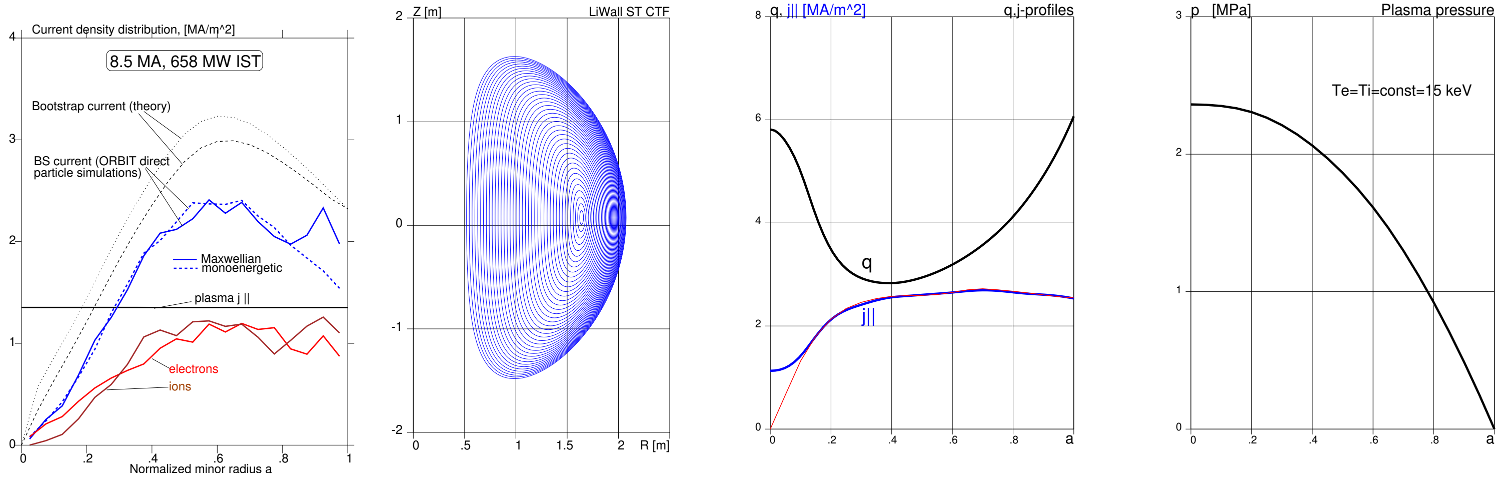
- Normalized beta, $\beta_N = 6.5$, with $\beta_N/l_i = 9.5$; β_N up to 35% over $\beta_{N\ \text{no-wall}}$
- Toroidal beta has reached 35% ($\beta_t = 2\mu_0\langle p\rangle / B_0^2$)

NSTX

ST already have a relevant stability data base with tendencies consistent with LiWall concept

No profile control is needed for a bootstrap current required for the stationary regime in IST

Stability limit calculations with a self-consistent bootstrap current



(a) bootstrap current

(b) cross-section

(c) q- and j-profiles

(d) pressure profile

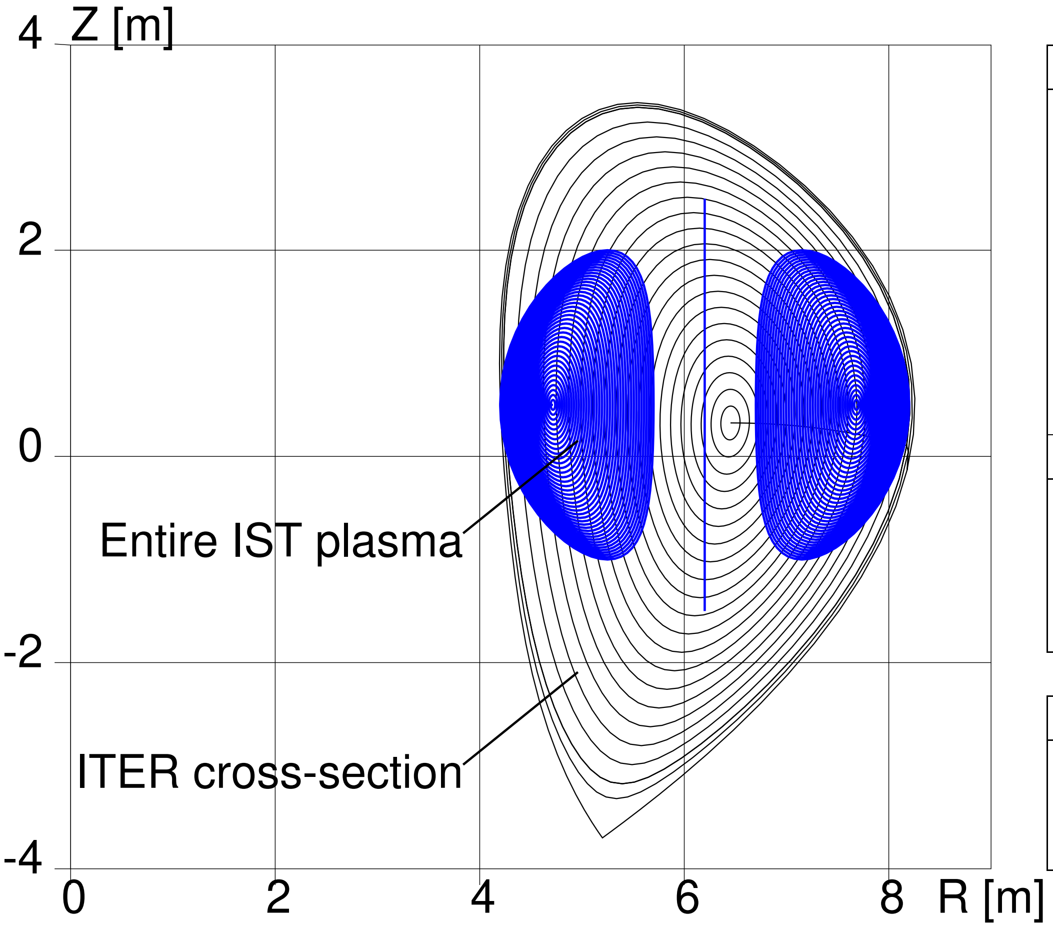
According to theory,

In the LiWall regime IST is ”over-driven” with bootstrap current.

No needs in externally driven “burning plasma” operational mode

Ignited Spherical Tokamaks can easily outperform ITER

Crucial difference is in β=0.4 (vs 0.03) and in "flat" $T_{i,e}$ =15 keV (vs peaked T)



IST Parameters				
CenterPole R	m	0.5	0.5	0.5
CenterPole B	T	7.5	7.5	7.5
Plasma R_1	m	0.5	0.5	0.5
Plasma R_2	m	2.0	2.0	2.0
Height	m	3.0	3.2	3.4
Volume	m^3	26.1	27.8	29.6
Surface	m^2	53.4	55.9	58.5
I plasma	MA	11.1	11.9	12.7
IST Plasma performance				
P_{DT}	MW	388	490	606
τ_E	sec	0.75	0.69	0.64
$F_{neutron}$	MW/m^2	5.8	7.0	8.3
$Loss_{neutron}$	%	9.4	9.6	9.8

ITER				
P_{DT}	MW	410	V	834 m^3
τ_E	sec	3.7	S	680 m^2
$F_{neutron}$	MW/m^2	0.5		

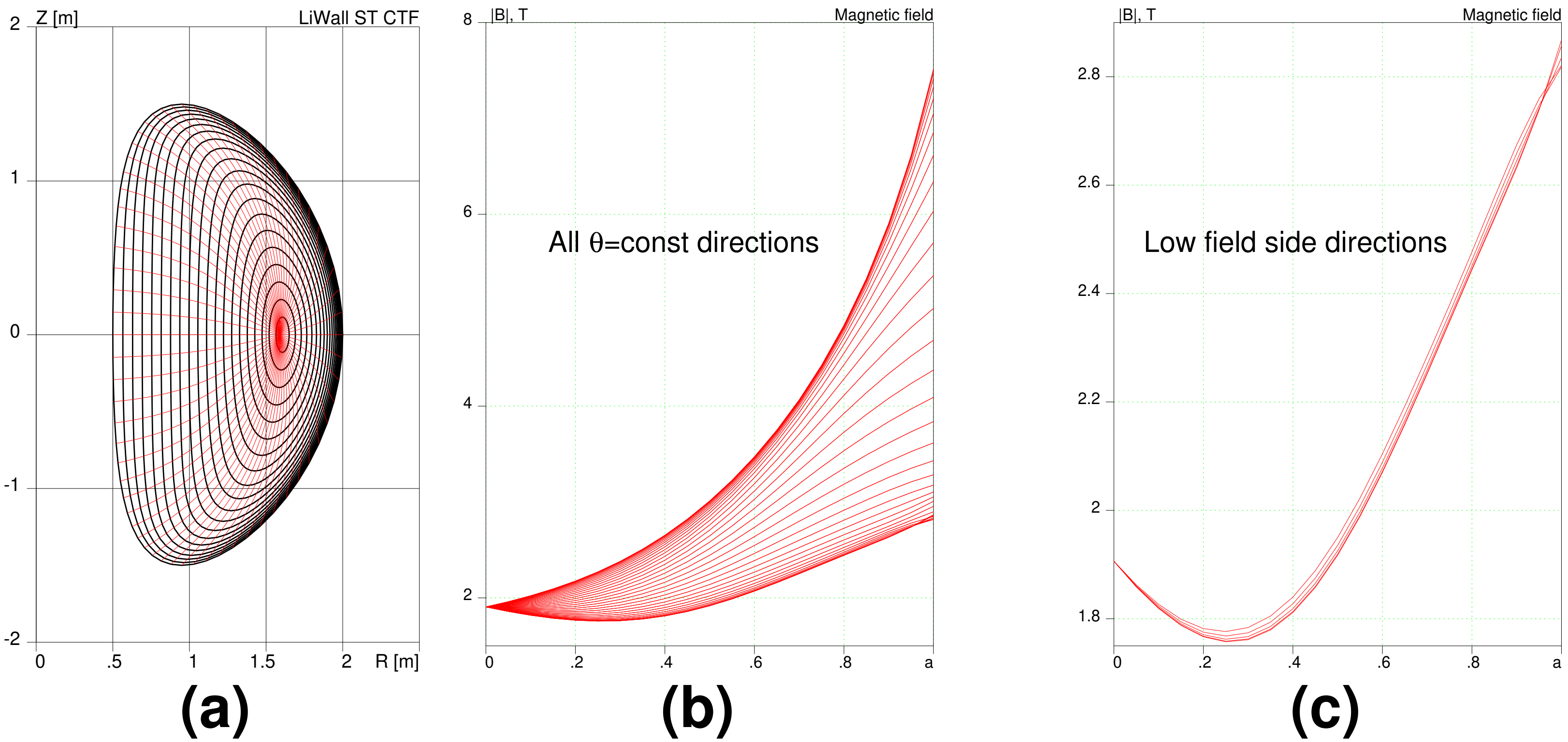
ISTs are suitable for developing Operation regime, First wall and Tritium cycle

Expected unique confinement and stability physics of IST makes reasonable even the question on feasibility of DD ignition

Based on "our best understanding of fusion"

IST can challenge confinement relevant to DD fusion

Absolute magnetic well situation can be created in IST

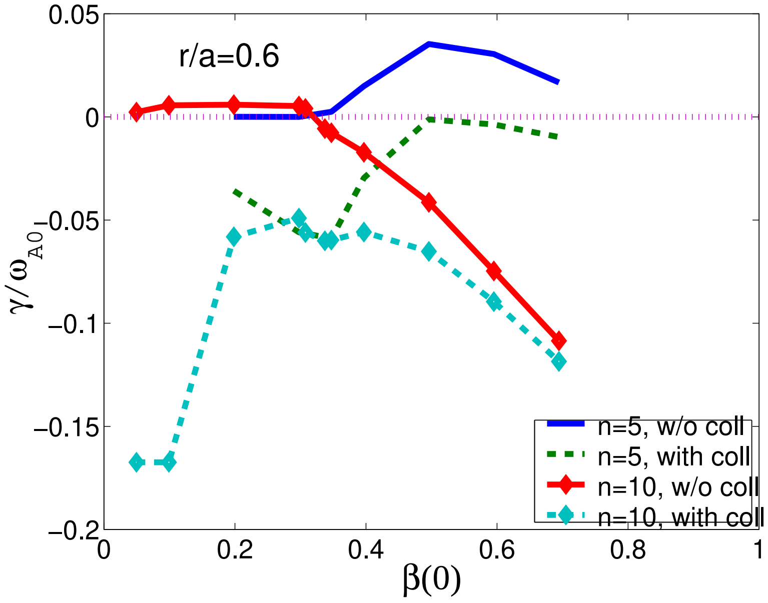


(a) Stable magnetic configuration of Fig. 1 (IST with $I_{pl} = 8.5$ MA, $\beta = 0.46$). Red lines correspond to $\theta = const.$ (b) $|B|$ as a function of a for 64 equidistant θ values. (c) $|B|$ as a function of a for 5 θ values near the outer middle plane.

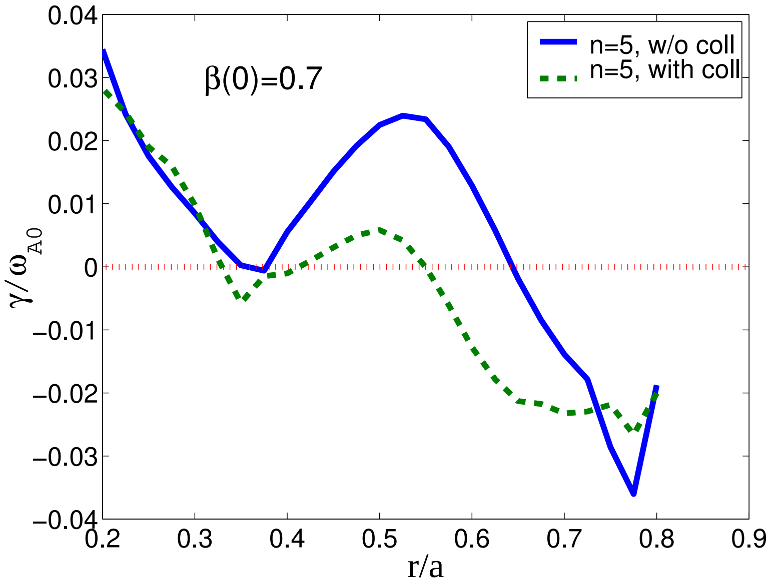
Large inverse gradient of $|B|$ reverses particle precession

Electron trapped modes are stabilized by reversed particle precession

Gorelenkov's HINST calculations of stability



Increase in β stabilizes modes



Even $n = 5$ mode is stable at $a > 0.6$

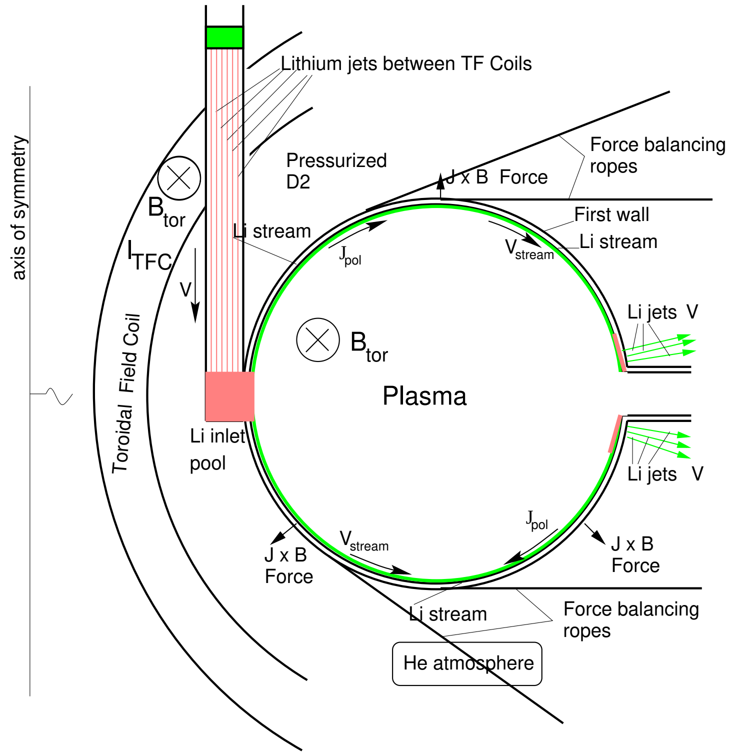
In LiWall regime particle losses are determined by the best confined component.

With no micro-turbulence DD fusion might be possible (needs $\tau_E \simeq 20$ sec)

but probably unpractical because of low power density.

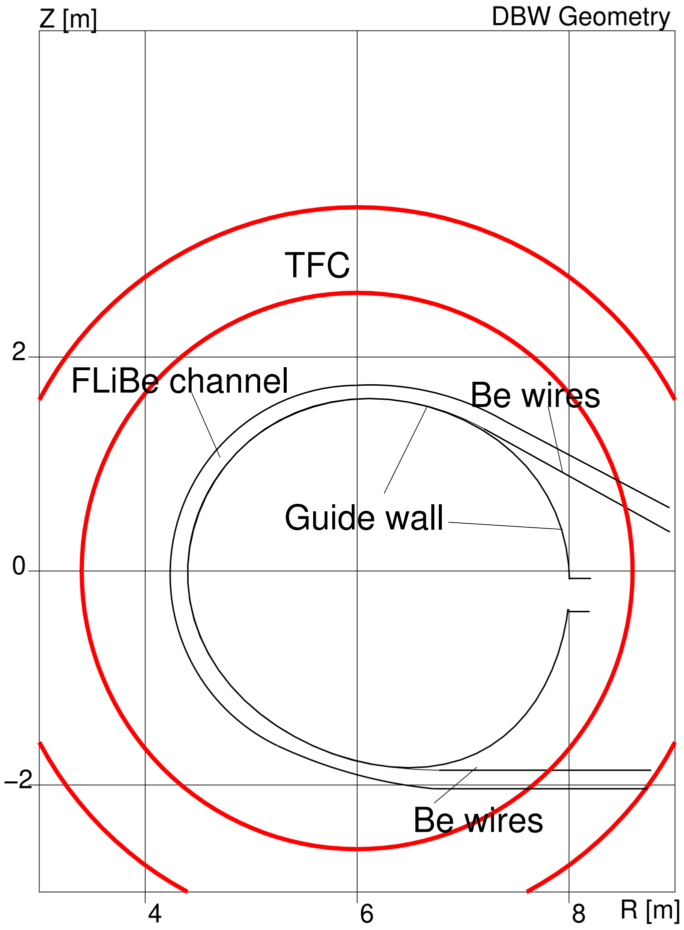
Intense Li Streams affect the very fundamentals of reactor design.

Electrodynamic pressure creates a stable situation for the first wall.



- **Guide wall works against expansion $\Longrightarrow$**
- **Guide wall can be made as a thin shell (like a car tire).**
- **Inner surface is sealed by the lithium streams (insensitive to cracks) $\Longrightarrow$**
- **Vacuum barrier can be moved to the plasma boundary (giving access to the neutron zone).**

Topology of Be wires can be made consistent with the presence of the FLiBe Blanket



Equation for poloidal curvature of the guide wall

$$d\frac{T}{\rho} = p_{JxB} - p_{ext} - g\rho_{FLiBe}(z - z_0).$$

Both radial force on both lines of wires

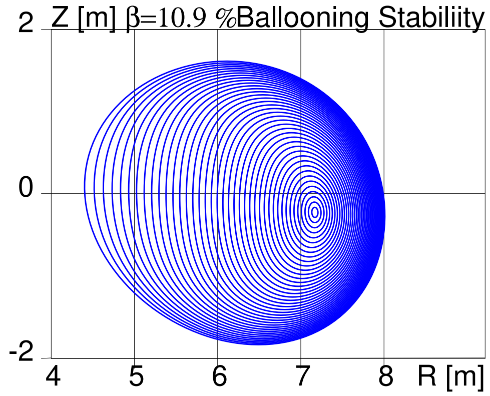
$$F = 1.5 \text{ [MN/m]}$$

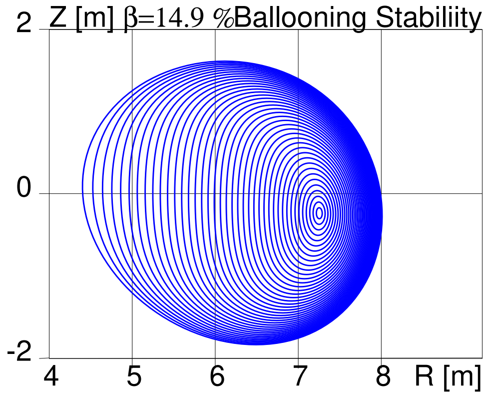
and tension in wires

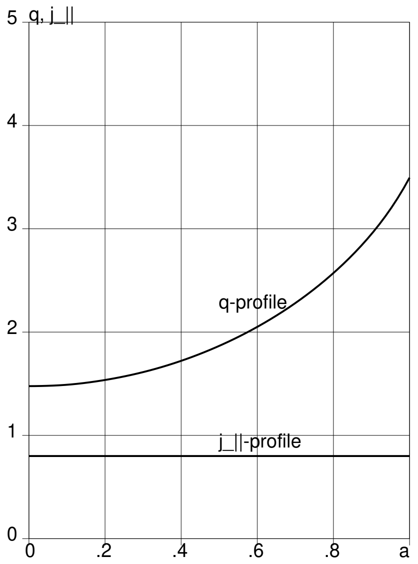
$$d \cdot T = 0.75 \text{ [MPa} \cdot \text{m]}$$

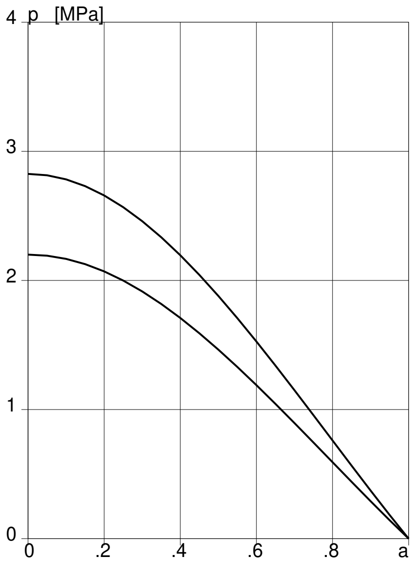
are reasonable.

Plasma shape is consistent with the wall stabilized high-β.

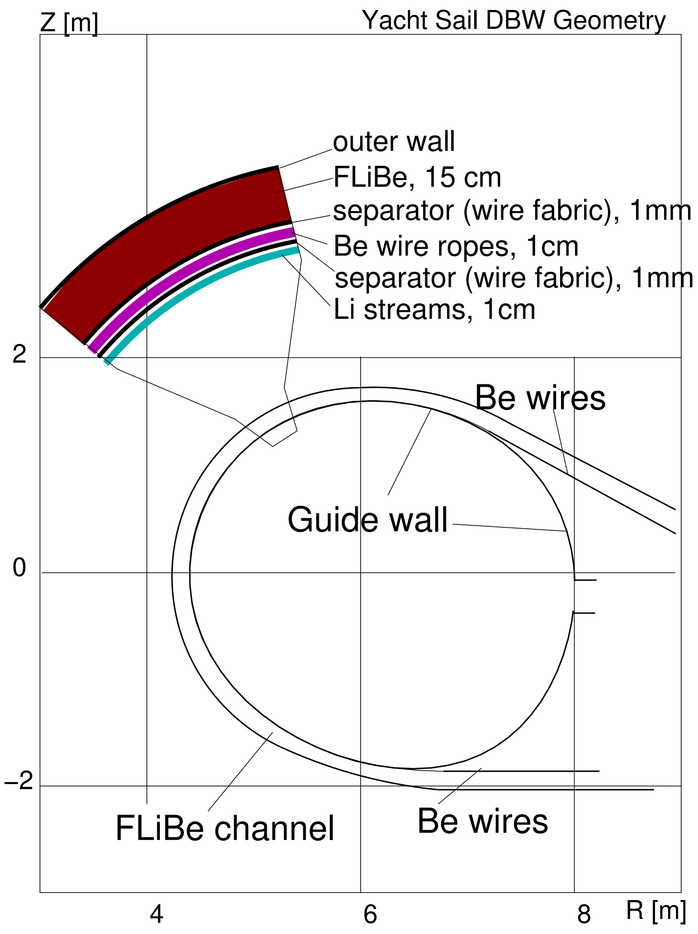








Yacht Sail design opens a way for low-activation DT fusion



- Intense Li Streams keep low temperature of the FW plasma side
- Guide (patchy wire fabric) wall serves as a separator between Li streams and wire ropes.
- Wire ropes provide the FW force balance.
- Second patchy wire fabric layer separates the wire ropes from FLiBe.
- FLiBe blanket is an element of FW.

Consistency with a power reactor is outstanding:

- Excellent energy extraction from the plasma and the blanket.
- Wires can withstand any plasma disruptions.
- Be wire ropes multiply neutrons.
- Minimal amount of high-Z materials.
- Vacuum barrier at the plasma boundary.
- Extremely high reliability, no damage, replacement on the fly.

There is no visible fundamental plasma physics or technology problems to develop the high power density regime, first wall, and tritium cycle for the fusion power reactor.
IST is a key tool.

Entrapment of the convention fusion program by a single, reactor irrelevant plasma physics concept, represents the real obstacle, which made fusion an extraordinary failure in physics of the 20 century.

Essentially in middle of 1980s, when the plasma physicists were not capable of providing the neutron fluence for the ITER project, the fusion program felt down into "activity trap".

The reactor relevant problems were put under the "rug", while the research was fragmented in uncorrelated activities covered by intense propaganda of scientific achievements with their illusory relevance to fusion energy.

6.5 years since the formulation of the LiWall concept lead to a firm indication that the current fusion program is essentially in an irreversible state of "thermal death" with scientific leadership structure vanished.

It is time for a separate program relying on new plasma regimes, new management approach, and explicitly targeting the power reactor development

New, fresh tokamak fusion programs may have a big advantage in

1. ***not only on relying on already existing knowledge and experience, but also***
2. ***on absence of not yet accumulated fragmentation and erosion of leadership which should be avoided by all means.***

With already exisitng scientific and technology background, the tokamak fusion is just one of the practical physics problems with no mystery. It can be developed as the energy source.

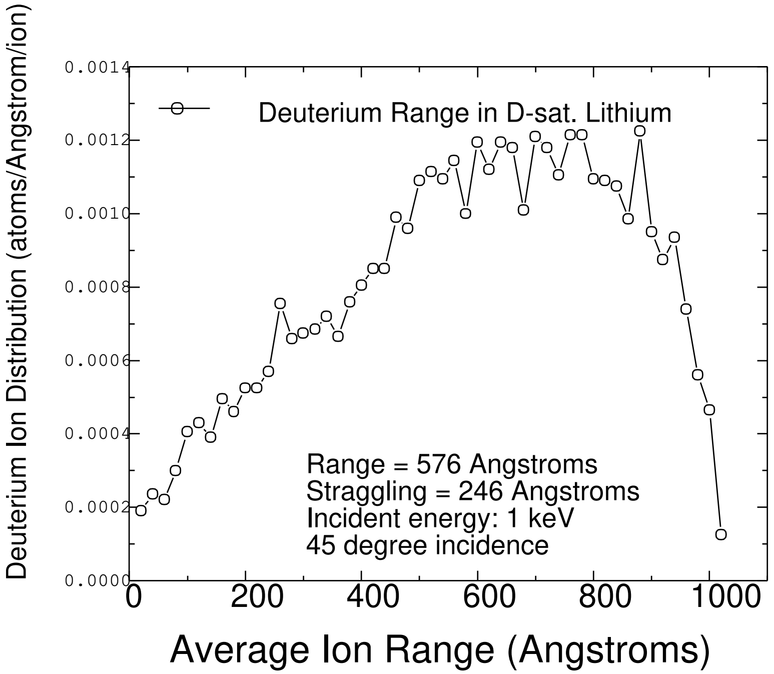
This requires the appropriate management preserving the structure of scientific and technical leadership, which would keep the program from the danger of "activity trap".

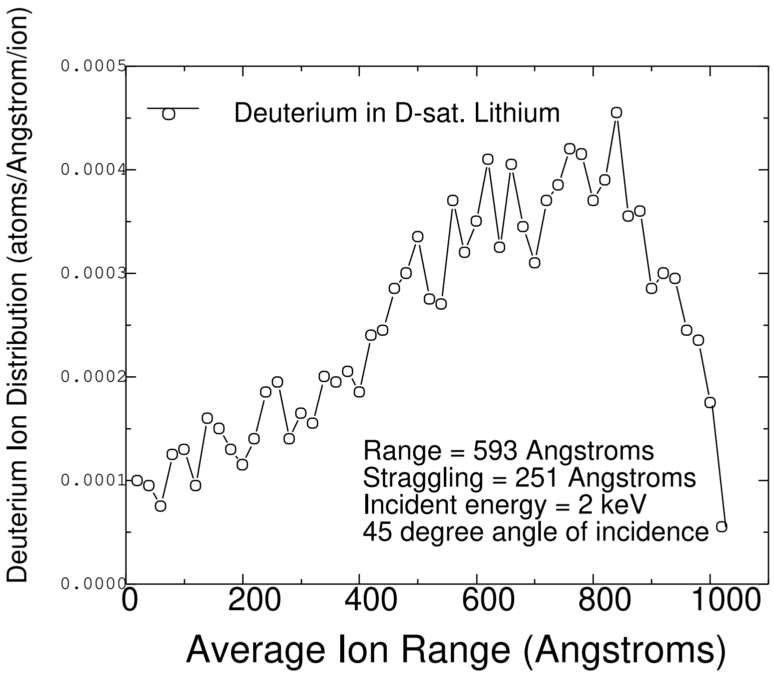
Only this condition can provide the efficient use of the government funds and uninterruptible progress.

web page: http://w3.pppl.gov/~ zakharov

Averaged Ion Range Of Deuterium incident on D-sat. Lithium

(TRIM calculations with 10000 flights by J.P.Allain, University of Illinoice, April, 2000)

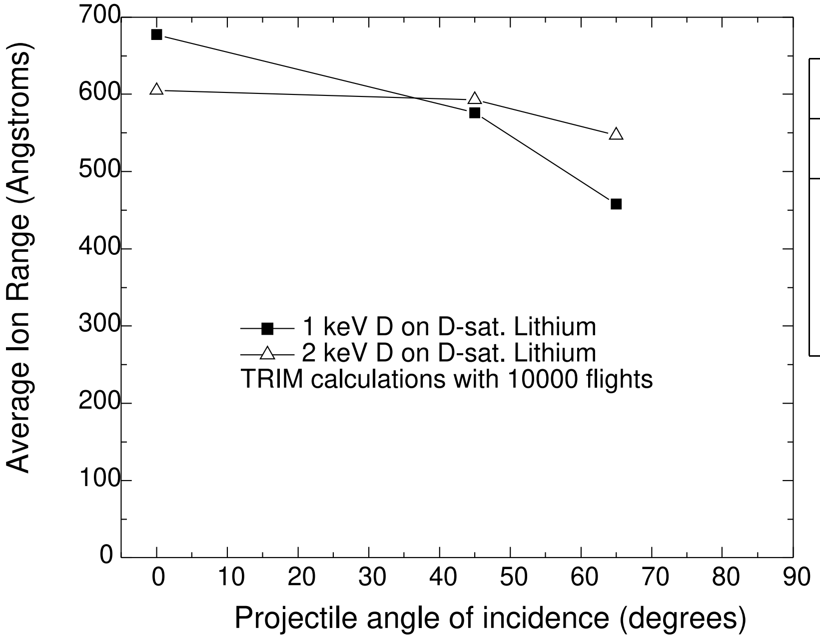




1 keV D on D-sat Li, Average Ion Range = 576 A
2 keV D on D-sat Li, Average Ion Range = 593 A

Averaged Ion Range as a function of angle of incidence

(TRIM calculations with 10000 flights by J.P.Allain, University of Illinoice, April, 2000)



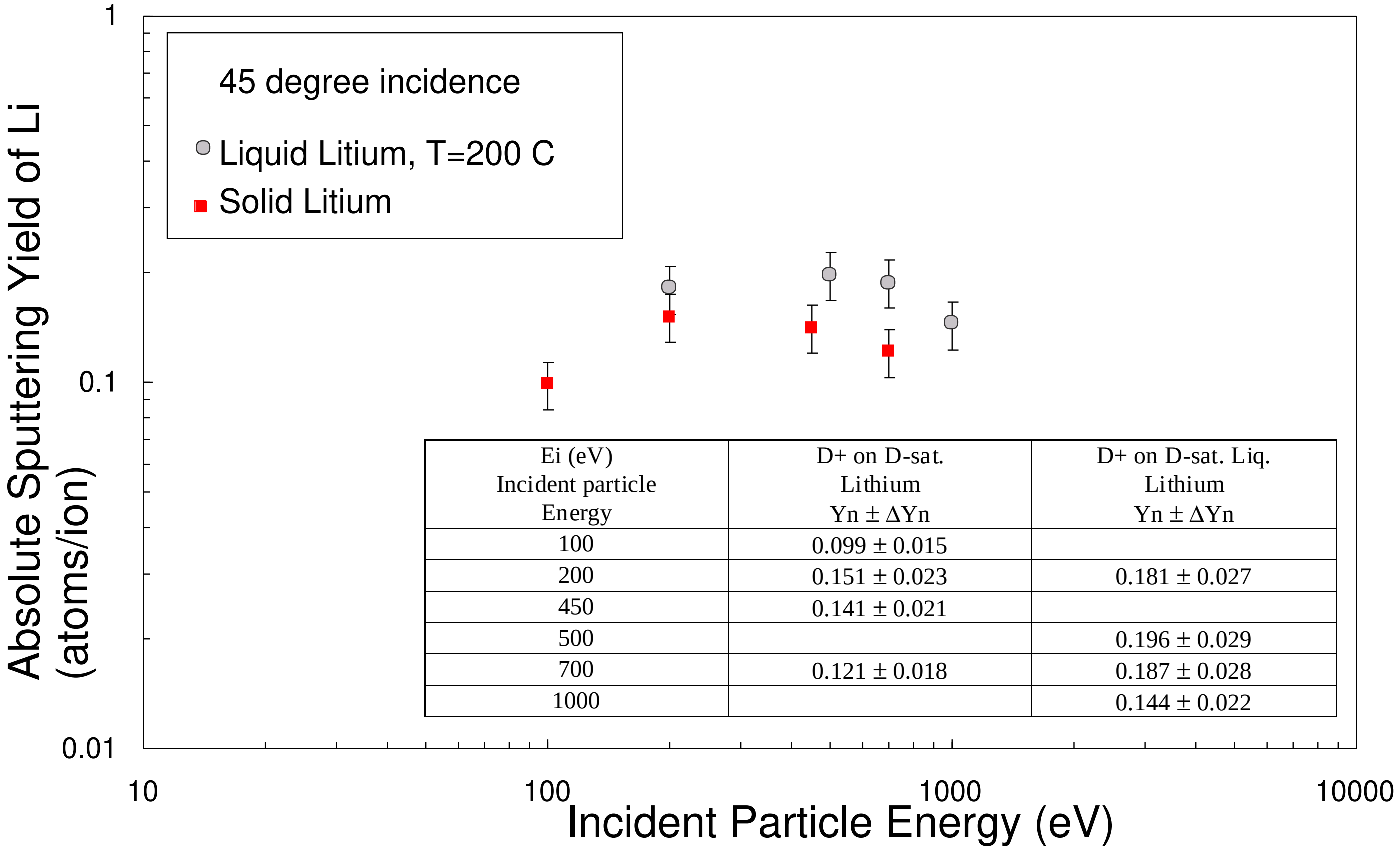
	Average Ion Range	
Angle	**at 1 keV**	**at 2 keV**
0	**677 A**	**605 A**
45	**576 A**	**593 A**
65	**458 A**	**547 A**

For 1 keV deuterium ion more than 150 Li monolayers participate in absorption.

D+ sputtering on Li

(http://starfire.ne.uiuc.edu/iiax/iiax.html, page 33)

D+ on D-saturated Solid and Liquid Lithium Measurements
(IIAX Data, J.P.Allain & D.N.Ruzic)



Ei (eV) Incident particle Energy	D+ on D-sat. Lithium Yn ± ΔYn	D+ on D-sat. Liq. Lithium Yn ± ΔYn
100	0.099 ± 0.015	
200	0.151 ± 0.023	0.181 ± 0.027
450	0.141 ± 0.021	
500		0.196 ± 0.029
700	0.121 ± 0.018	0.187 ± 0.028
1000		0.144 ± 0.022

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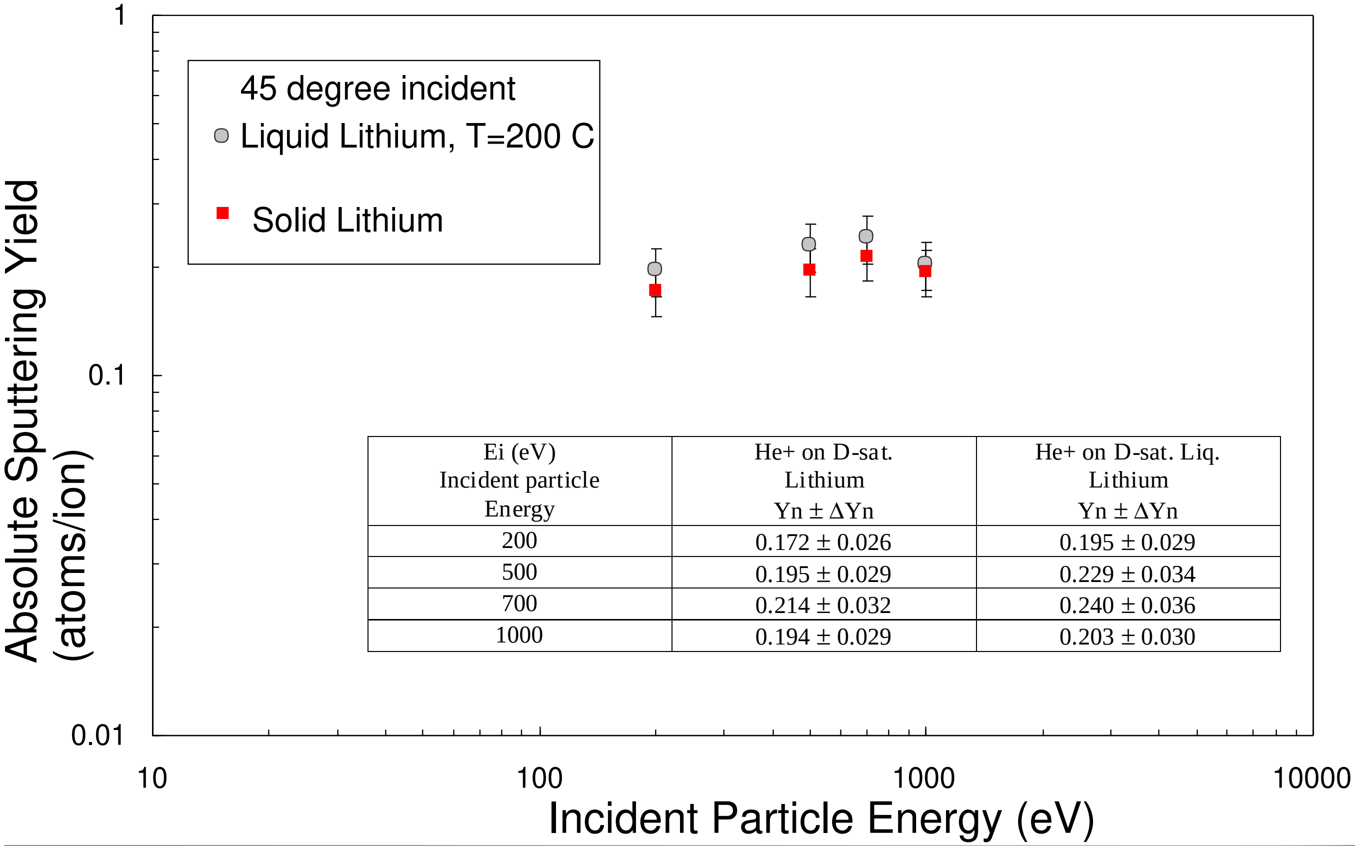
He sputtering on Li

(http://starfire.ne.uiuc.edu/iiax/iiax.html, page 34)

He+ on D-saturated Solid and Liquid Lithium Measurements
(IIAX Data, J.P.Allain & D.N.Ruzic)

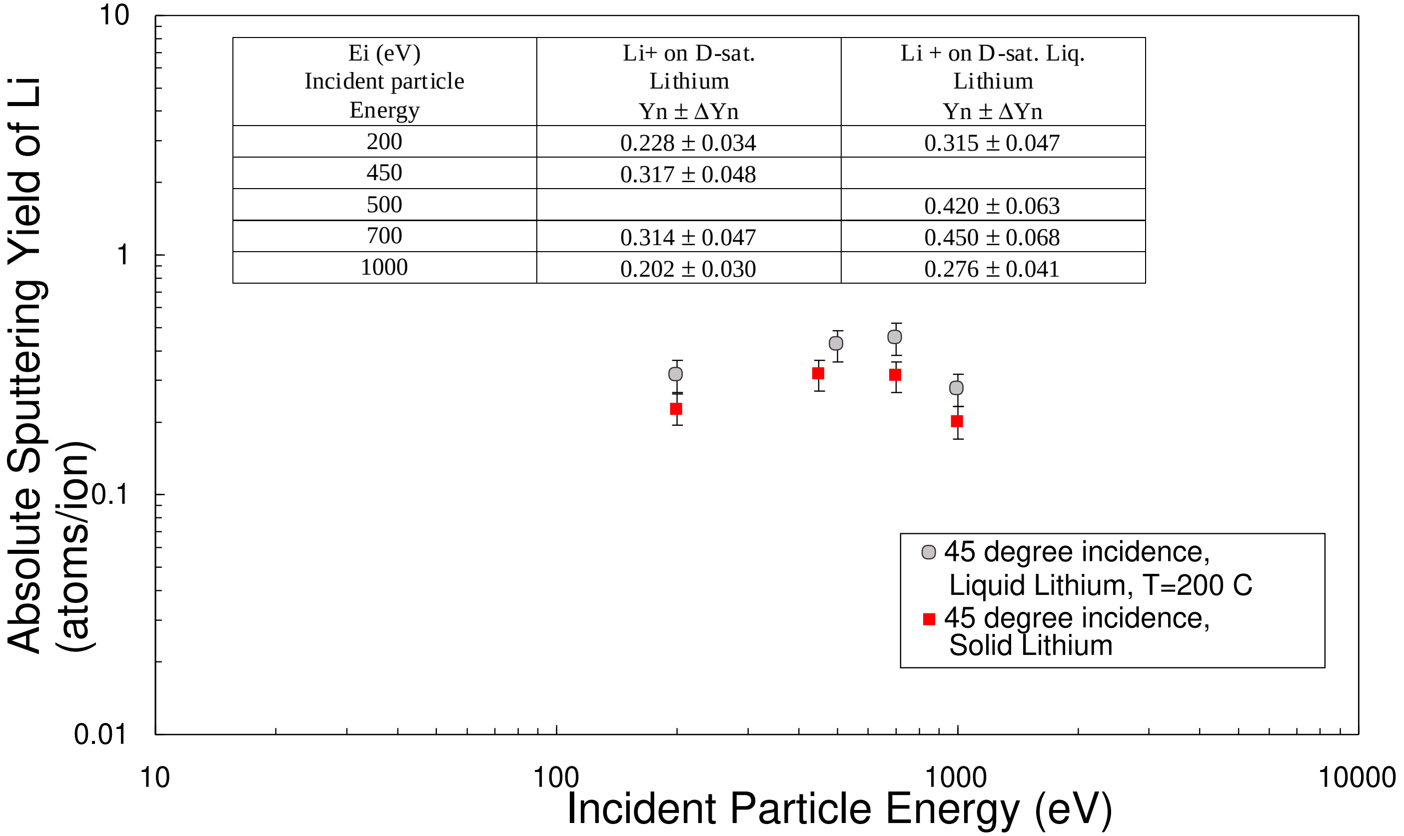


Ei (eV) Incident particle Energy	He+ on D-sat. Lithium Yn ± ΔYn	He+ on D-sat. Liq. Lithium Yn ± ΔYn
200	0.172 ± 0.026	0.195 ± 0.029
500	0.195 ± 0.029	0.229 ± 0.034
700	0.214 ± 0.032	0.240 ± 0.036
1000	0.194 ± 0.029	0.203 ± 0.030

Li sputtering on Li is less than 1 (no runaway)

(http://starfire.ne.uiuc.edu/pmi/IIAX%20Summary.pdf, page 32)

Li on D-saturated Solid and Liquid Lithium Measurements
(IIAX Data, J.P.Allain & D.N.Ruzic)



Ei (eV) Incident particle Energy	Li+ on D-sat. Lithium Yn ± ΔYn	Li + on D-sat. Liq. Lithium Yn ± ΔYn
200	0.228 ± 0.034	0.315 ± 0.047
450	0.317 ± 0.048	
500		0.420 ± 0.063
700	0.314 ± 0.047	0.450 ± 0.068
1000	0.202 ± 0.030	0.276 ± 0.041

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T.D.Rognlien, M.E.Rensink (LLNL) “Liquid-Walls Temperature Limits” (http://www.td.anl.gov/ALPS_Info_Center/alps/rogn_impur.pdf, *ALPS/APEX Meeting, Argonne Nat. Lab., May 8-12, 2000)*

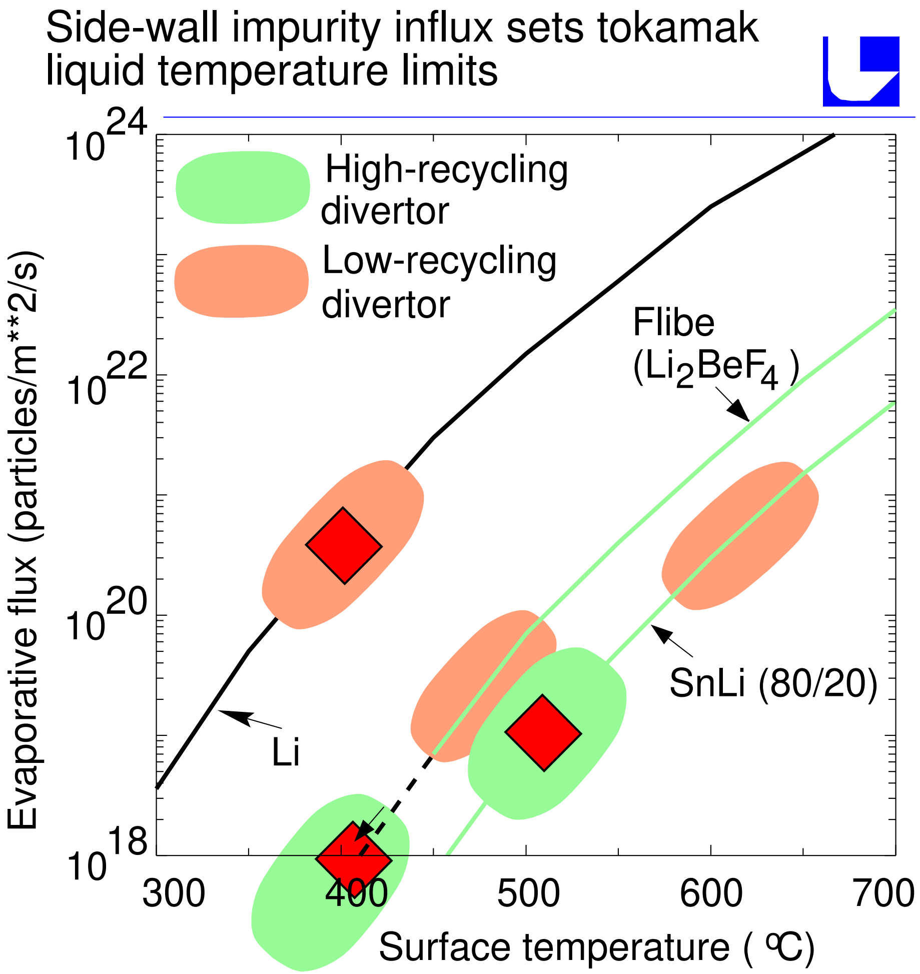
Evaporation Li flux Γ

$$\Gamma \simeq 3.5 \cdot 10^{18} \cdot 10^{\frac{T^o - 300^o\ C}{50^o}} \left[\frac{1}{m^2 \cdot sec}\right]$$

At $T = 200^o\ C$

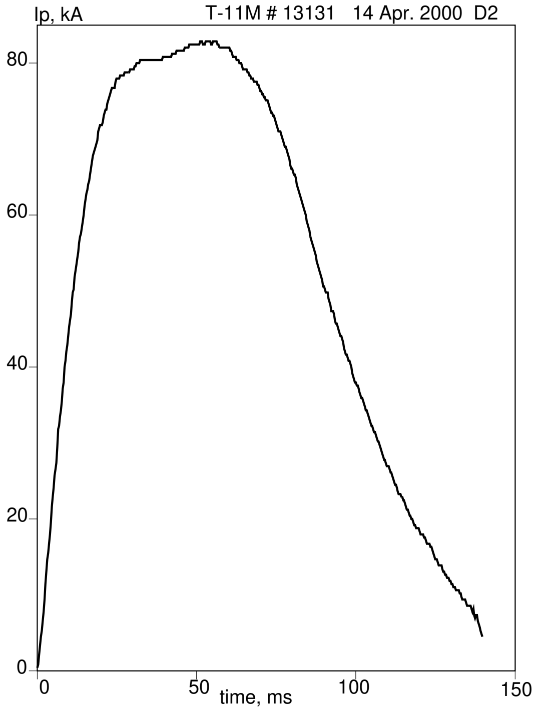
$$\Gamma \simeq 5 \cdot 10^{15} \left[\frac{1}{m^2 \cdot sec}\right]$$

T_{Li} should be less than $400^o\ C$.

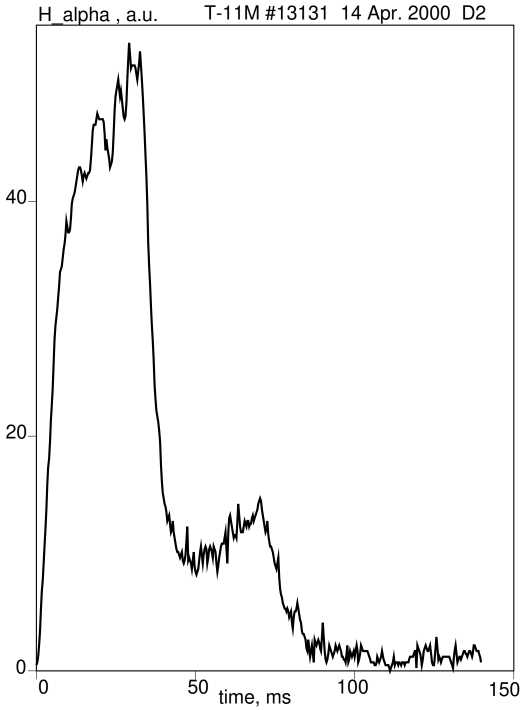


T-11M, CDX-U demonstrated outstanding pumping capabilities of Li

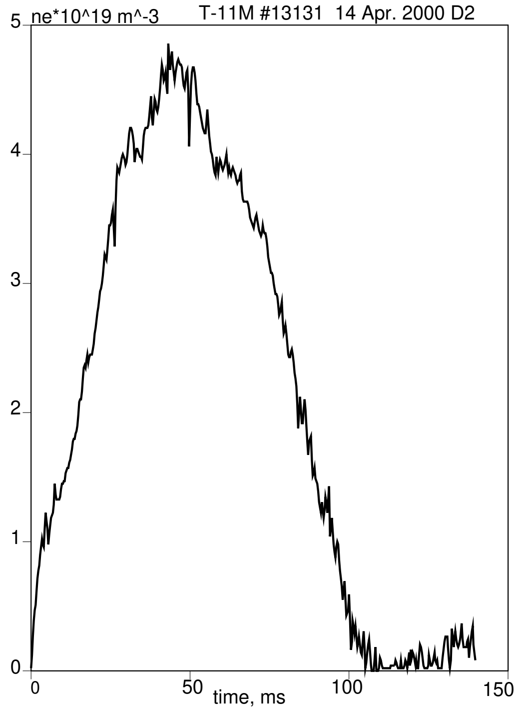
(http://w3.pppl.gov/~zakharov/Mirnov010221/Mirnov.ppt, *p.19, Exper. Seminar PPPL, Feb. 21, 2001)*



Plasma current



Gas puffing

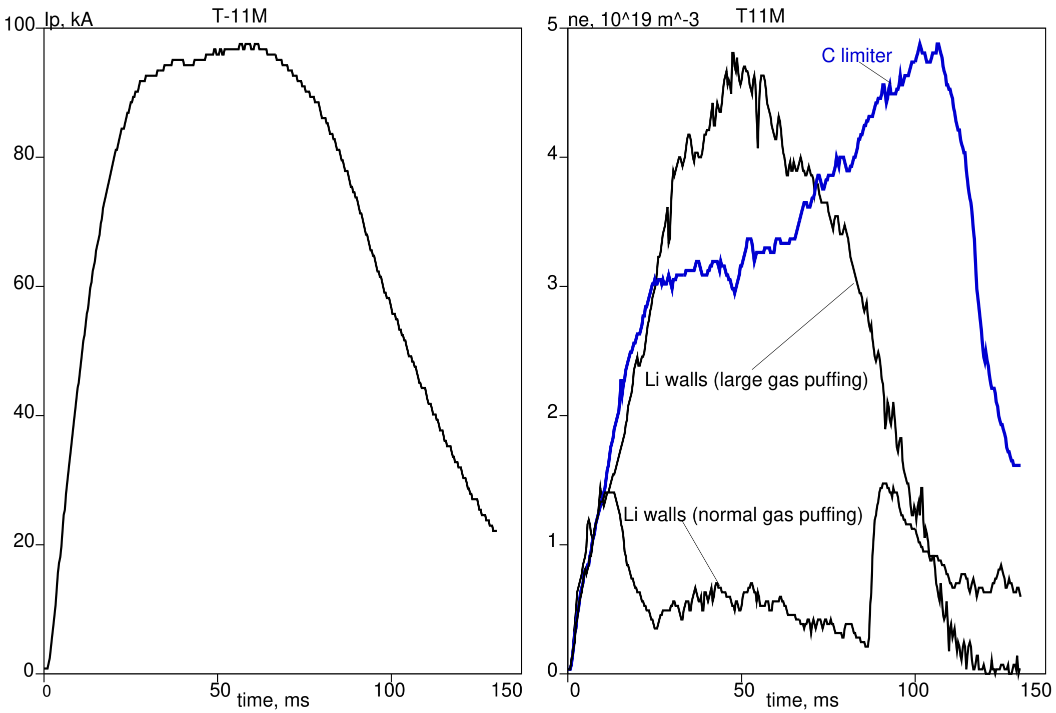


Averaged density

Density decays, presumably, with the particle confinement time.

T-11M, CDX-U demonstrated outstanding pumping capabilities of Li

(http://w3.pppl.gov/~zakharov/Mirnov010221/Mirnov.ppt, *p.18, Exper. Seminar PPPL, Feb. 21, 2001)*



Plasma current **Averaged density**

Density decays, presumably, with the particle confinement time.