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LiWall Fusion and its Three Step R & D Program

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Abstract

The presently adopted plasma physics concept of magnetic fusion has been originated from the idea of providing low plasma edge temperature as a condition for plasma-material interaction. During 30-years of its existence this concept has shown to be not only incapable of addressing practical reactor development needs, but also to be in conflict with fundamental aspects of stationary and stable plasma.

Meanwhile, a demonstration of exceptional pumping capabilities of lithium surfaces on T-11M (1998), discovery of the quiescent H-mode regime on DIII-D (2000), and a 4 fold enhancement of the energy confinement time in CDX-U tokamak with lithium (2005), contributed to a new vision of fusion relying on high edge plasma temperature. The new concept, called LiWalls, provides a scientific basis for developing magnetic fusion.

The talk outlines 3 basic steps toward the Reactor Development Facility (RDF) with DT fusion power of 0.3-0.5 GW and a plasma volume $\simeq 30$ m^3. Such an RDF can accomplish three reactor objectives of magnetic fusion, i.e.,

1. *high power density $\simeq 10$ MW/m^3 plasma regime,*
2. *self-sufficient tritium cycle,*
3. *neutron fluence $\simeq 10 - 15$ MW·year/m^2,*

all necessary for development of the DT power reactor. Within the same mission a better assessment of DD fuel for fusion reactors will also be possible.

The suggested program includes 3 spherical tokamaks. Two of them, ST1, ST2, are DD-machines, while the third one, ST3, represents the RDF itself with a DT plasma and neutron production.

All three devices rely on a NBI maintained plasma regime with absorbing wall boundary conditions provided by the Li based plasma facing components. The goal is to utilize the possibility of high edge temperature plasma with the super-critical ignition (SGI) regime, when the energy confinement significantly exceeds the level necessary for ignition by α-particles. In this regard all three represent Ignited Spherical Tokamaks, suggested in 2002.

Abstract

Specifically, the mission of ST1, with a size slightly larger than NSTX in PPPL but with a four times larger toroidal field, is to achieve the absorbing wall regime with confinement close to neo-classical. In particular, the milestone is $Q_{DT-equiv} \simeq 5$ corresponding to the conventional ignition criterion.

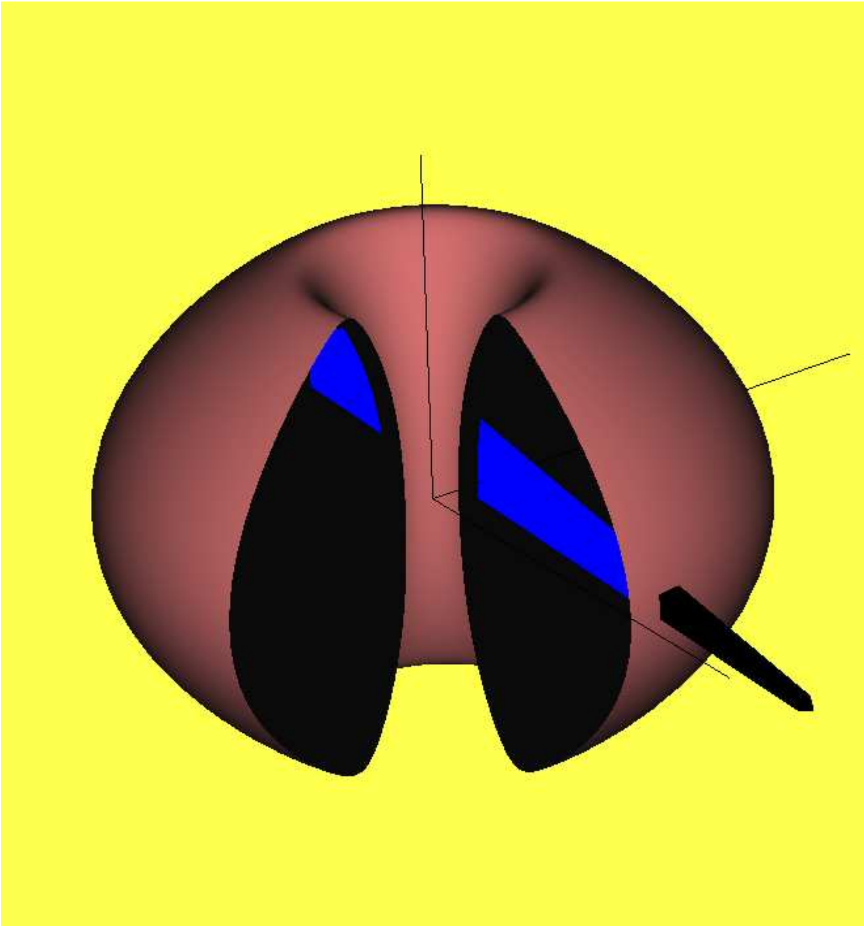
The mission of ST2, which is a full scale DD-prototype of the RDF, is the development of a stationary super-critical regime with $Q_{DT-equiv} \simeq 40-50$.

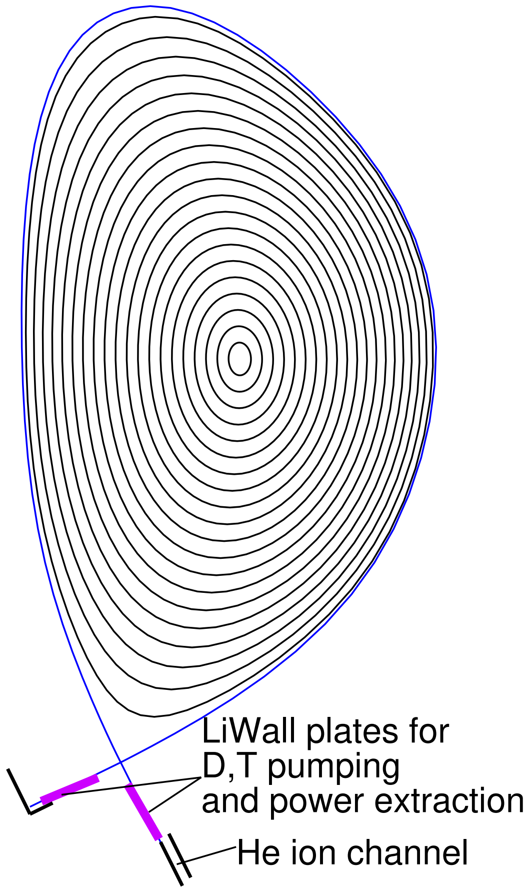
ST3 is a DT device with $Q_{DT} \simeq 40-50$ with sufficient neutron production to design the nuclear components of a power reactor. Still the mission of ST3 contains a significant plasma physics component of developing α-particle power and He ash extraction.

As a motivational step (ST0), the suggested program, assumes a conversion of the existing NSTX device into a spherical tokamak with lithium plasma facing components. The demonstration of complete depletion of the plasma discharge by lithium surface pumping, first shown on T-11M, is considered as a well-defined milestone for readiness of the machine for the new plasma regime. The final mission of ST0 would be doubling or tripling the energy confinement time with respect to the current NSTX.

What will happen if: (a) Neutral Beam Injection (NBI) supplies particles into the plasma core, while (b) a layer of Lithium on the Plasma Facing Surface (PFC) absorbs all particles coming from the plasma ?

(Assume that maxwellization is much faster than the particle diffusion.)





The answer is very simple: because there is no cold particles in the system (other than Maxwellian)

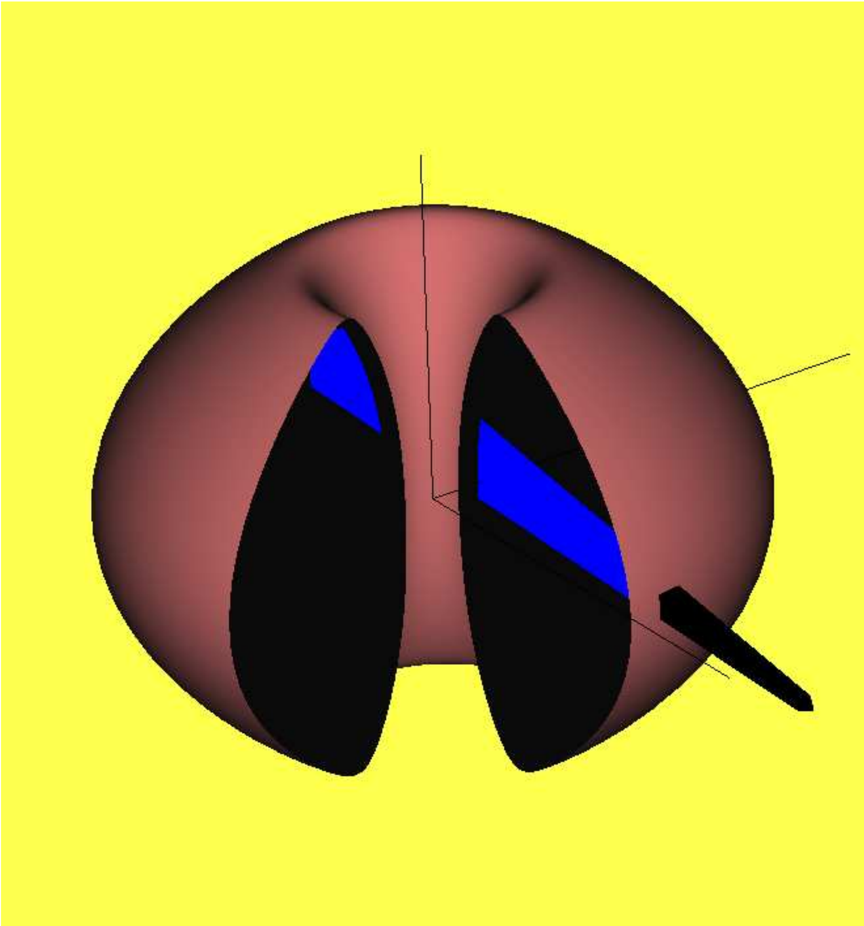
The plasma temperature will be uniform over entire cross-section.

Plasma physics is not involved into this answer.

Ion/electron temperature gradient instabilities (ITG,ETG), which are the major cause of energy losses, would be eliminated automatically.

In fact, any thermo-conduction would be eliminated. Energy from the plasma will be lost only due to particle diffusion

NBI is a ready-to-go fueling method for LiWF



The energy should be consistent with the plasma temperature

$$E_{NBI} = \left(\frac{3}{2} + 1\right)(T_i + T_e),$$

e.g., for

$$T_e \simeq T_i \simeq 16\ keV$$

$$E_{NBI} = 80\ keV$$

In absence of cold particles from the walls, after collisional relaxation

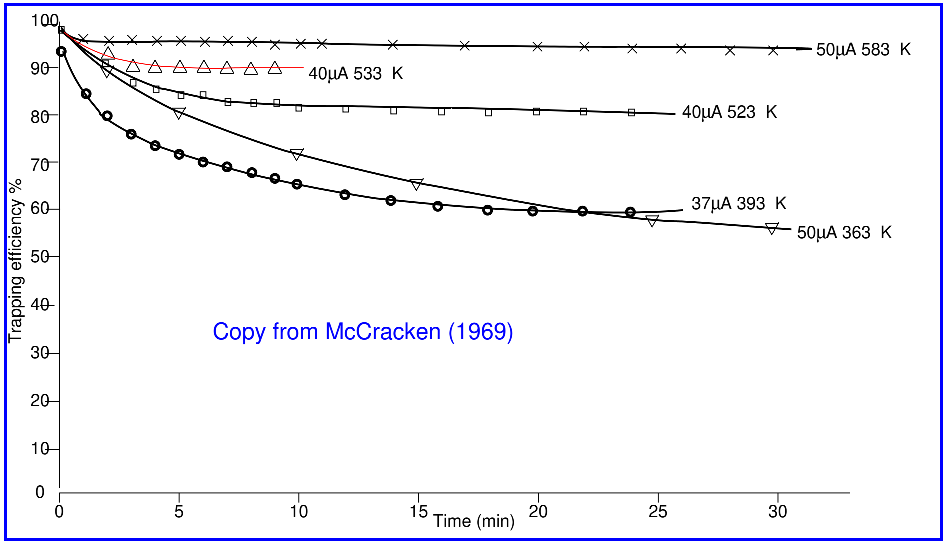
$$\nu_i = 68\frac{n_{20}}{T_{i,10}^{3/2}}, \quad \nu_e = 5800\frac{n_{20}}{T_{e,10}^{3/2}}$$

the temperature profile becomes flat automatically

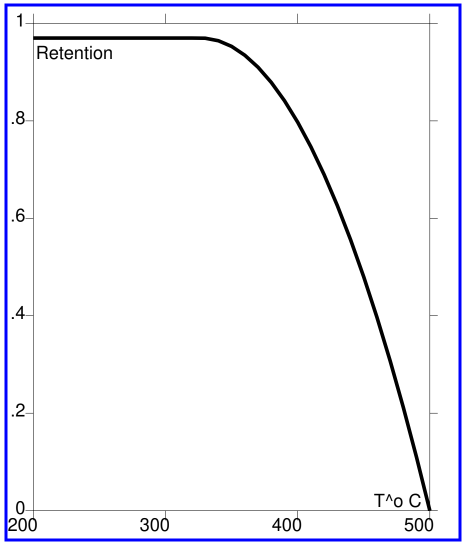
$$T_i = const, \quad T_e = const, \quad T_e < T_i$$

The plasma is always in the “hot-ion” regime

Lithium can retains $\simeq 10\%$ of H,D,T atoms per Li atoms



McCracken retention curves



Short term retention curve used in calculations

Probably the short lasting retention allows temperatures above 350^{o}C (R.Majeski)

Because of evaporation, surface temperature of Li should be limited (by $\simeq 400^o$ C)

LiWF relies on two things: (1) core fueling (by NBI), and (2) edge pumping (by Li surface)

The presently adopted concept, "The Bib$_b$le of the 70s", referred as (BBBL70), has elevated the fusion research almost to the level of ignition, BUT

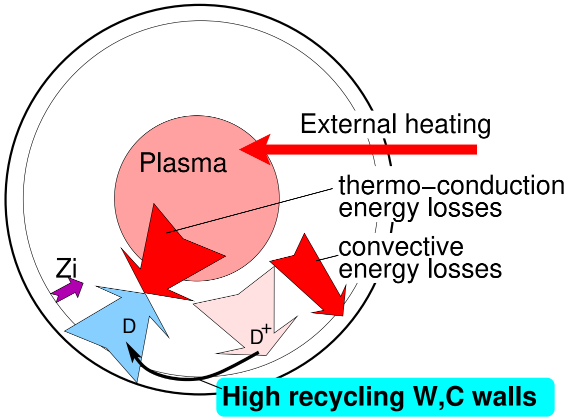
BBBL70 is incapable to deliver a meaningful power reactor

LiWF is a new concept, rather than an "improvement" of BBBL70

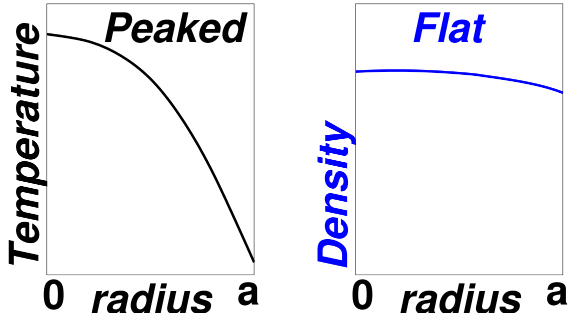
It affects the fundamental aspects of magnetic fusion, e.g.,

- A super-critical ignition regime (SCI), with $\tau_E \gg \tau^*_{E,ignition}$ is expected.
- No needs for α-particle heating. They can be lost at first orbits.
- LiWF makes the "hot-ion" mode perfect for fusion.

The right plasma-wall contact is the key to magnetic fusion

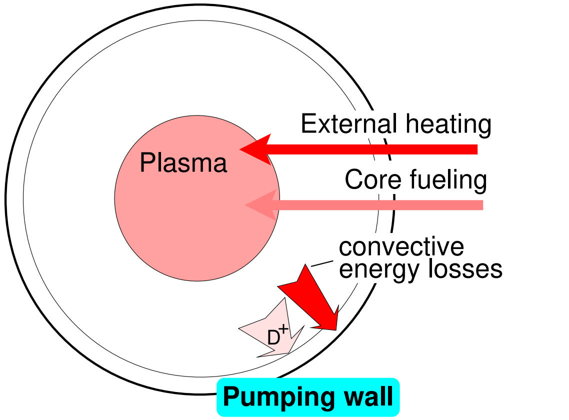


BBBL70 requires a low temperature plasma edge

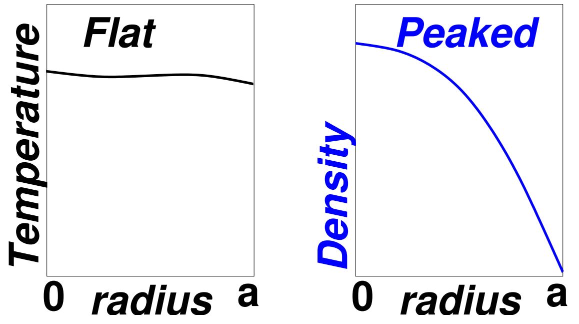


As a "gift" from plasma physics BBBL70 gets ITG/ETG turbulent transport.

Most of the plasma volume does not produce fusion



Molten Li pumps the plasma out. High edge T is OK



No "gifts" from plasma physics (ITG/ETG, sawteeth, ELMs) are expected or accepted.

LiWF relies only on external control.

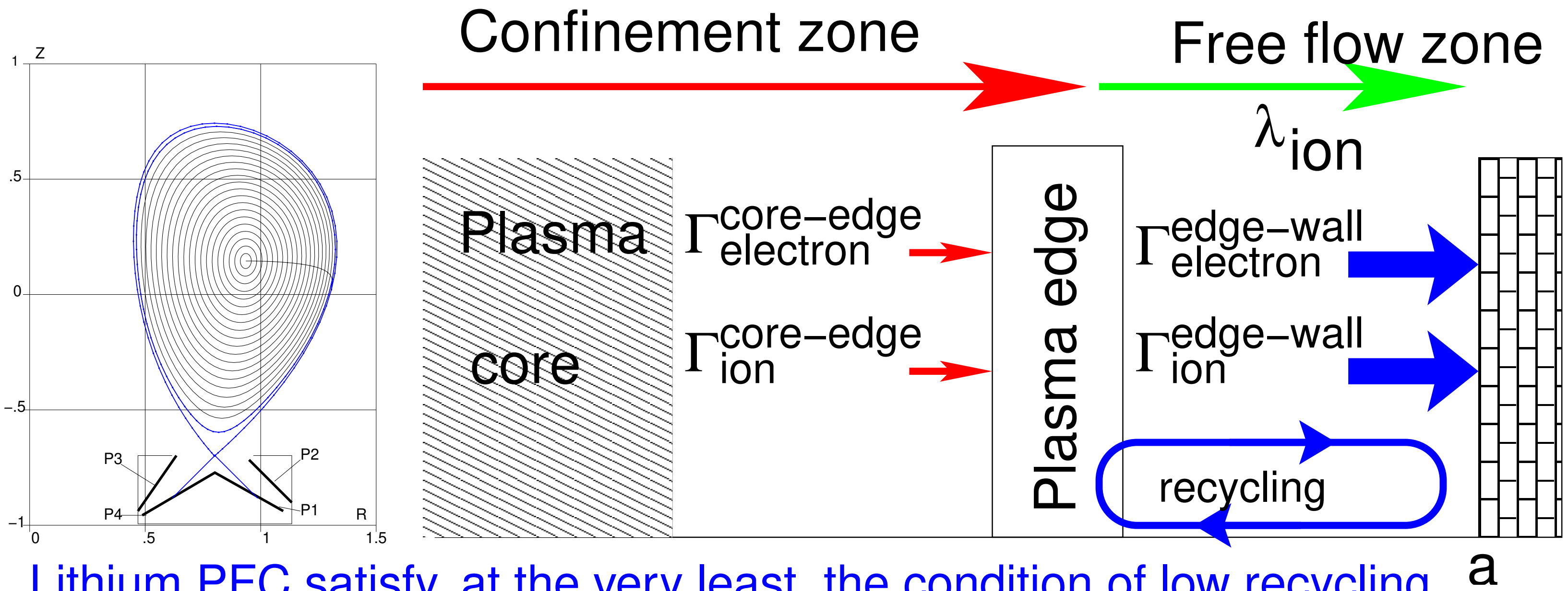
The entire plasma volume produces fusion

Lithium based PFC are incompaible with BBBL70. On the other hand

Li is unique for pumping wall idea (the LiWF)

LiWF requires recycling coefficient $R \ll 1$, i.e.

$$\Gamma_{ion}^{edge-wall} \simeq \Gamma_{ion}^{core-edge}, \quad \Gamma_{electron}^{edge-wall} \simeq \Gamma_{electron}^{core-edge}$$



Lithium PFC satisfy, at the very least, the condition of low recycling.

The importance of the second condition is not yet known. The scales

$$\rho_e^{se} = \frac{4.76}{B_T} \ll \rho_e^{SOL} = 238\frac{\sqrt{T_{e,10keV}}}{B_T} \ll \rho_D = 14100\frac{\sqrt{T_{i,10keV}}}{B_T} \ [\mu m]$$

give a chance to magnetic insulation (upon its necessity).

The edge plasma temperature is determined by the particle fluxes rather than by near edge transport properties

Across the last mean free path, λ_D, in front of PFC surface

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

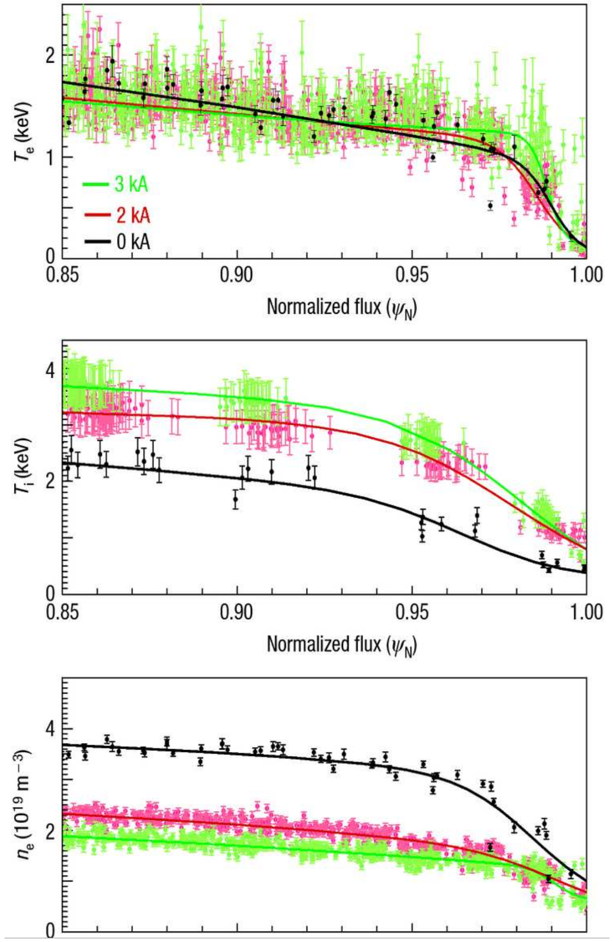
the energy is carried by the moving particles

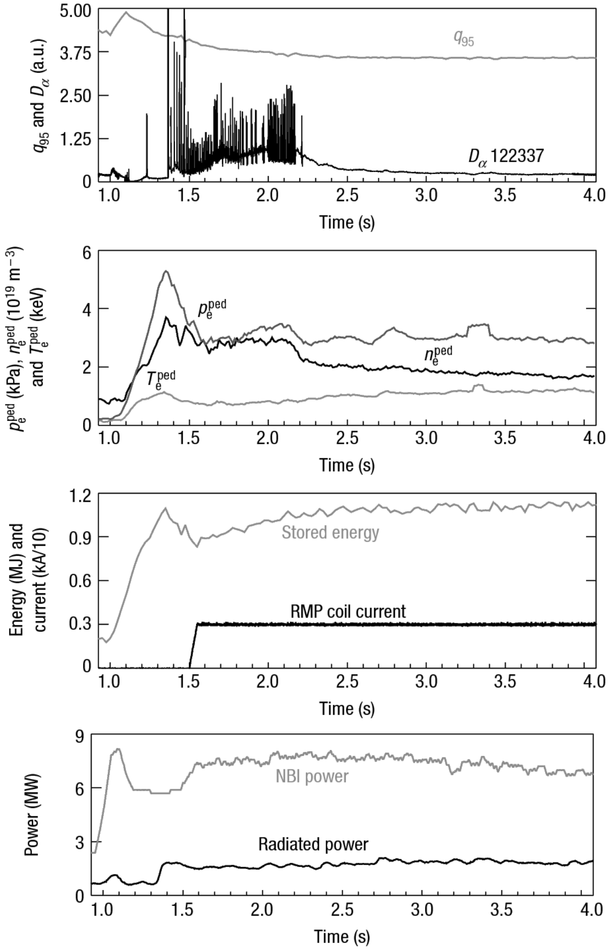
$$\frac{5}{2}\Gamma_{electron}^{edge-wall}T_e^{edge} = \int_V P_e dV, \qquad \frac{5}{2}\Gamma_{ion}^{edge-wall}T_i^{edge} = \int_V P_i dV$$

For edge temperature $T_{e,i} \simeq 1$ keV (low collisionality H-mode) the mean free path λ_D is very long $\simeq$ km's

This Krasheninnikov's boundary condition determines the edge temperature pedestal

RMP experiments on DIII-D have confirmed our, LiWF, views





0 kA, 2 kA, 3 kA $I_{RMP-coil}$ T.Evans at al., Nature physics 2, p.419, (2006)

These observations are in conflict with one of misconceptions of about the "edge transport barrier"

The pumping PFCs deeply affect the core confinement

The core particle flux $\Gamma_{e,i}^{core-edge}$ *and the flux to the wall* $\Gamma_{e,i}^{edge-wall}$ *are related through the recycling coefficient* R

$$\Gamma_{e,i}^{edge-wall} = \frac{\Gamma^{core-edge}}{1-R_{e,i}}, \qquad \frac{5}{2}\Gamma^{core-edge}T_{e,i}^{edge} = (1-R_{e,i})\int_V P_{e,i}dV$$

In the case of pumping PFC

$$R_{e,i} \ll 1, \qquad \Gamma_{e,i}^{edge-wall} \simeq \Gamma^{core-edge},$$

the Krasheninnikov boundary conditions lead to the temperature profile $T_{i,e}(a)$*, which eliminates the* thermo-conductive *energy losses*

$$\underbrace{\oint q_{i,e}dS}_{thermo-conduction} \simeq 0, \qquad T_{i,e}^{edge} \simeq T_{i,e}(0)$$

in the transport equations inside the core

$$\underbrace{\frac{5}{2}\oint \Gamma^{core}T^{i,e}dS}_{convection} + \underbrace{\oint q_{i,e}dS}_{thermo-conduction} = \underbrace{\int_0^V P_{i,e}(V)dV}_{Power\ source}, \qquad \underbrace{\oint \Gamma^{core}dS}_{convection} = \underbrace{\int_v SdV}_{particle\ source}$$

The energy losses are caused exclusively by particle losses

The confinement can be predicted in a straighforward way

The thermo-conductive losses of the conventional plasma have no limits because of turbulence.

Plasma diffusion is limited by the best confined component, i.e. ions.

The LiWF is the only concept, which does not depend on anomalous behavior of electrons and associated mysteries

A simple Reference Transport Model (RTM) is relevant for projections of LiWall regime

$$\Gamma^{core} = \chi_i^{neo-classics} \nabla n$$

$$q_i = \chi_i^{neo-classics} \nabla T_i, \qquad \text{not important}$$

$$q_e = \chi_i^{neo-classics} \nabla T_e, \qquad \text{not important},$$

RTM predicts the feasibility of the siper-critical ignition (SCI) regime with $\tau_E \gg \tau_E^*$

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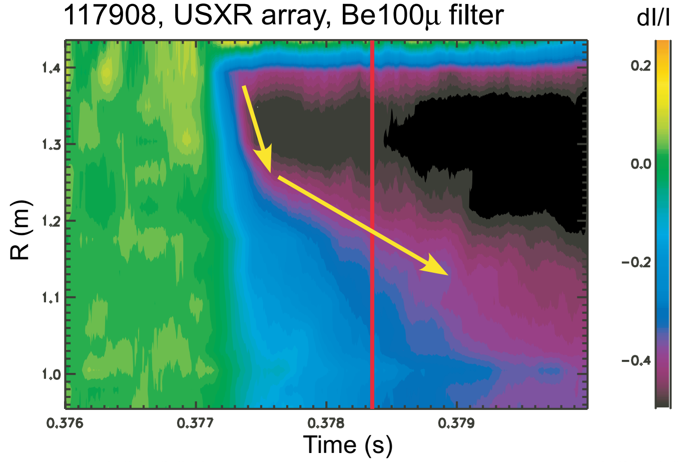
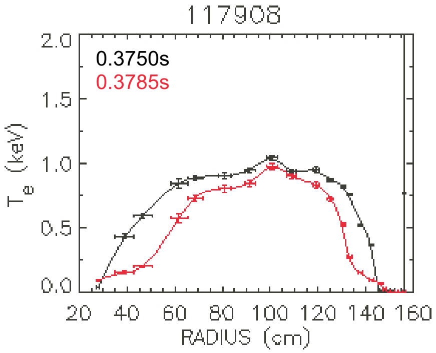
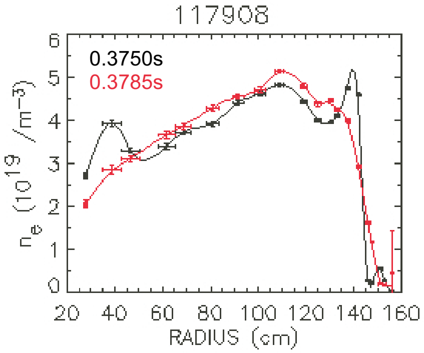
Perturbation Analysis Indicates Two Regions of $\chi_{e,pert}$

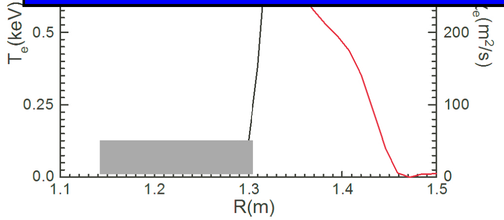


- T_e crash propagates from edge to core, n_e globally unperturbed
- Difference in propagation speed corresponds to differences in pertur

NSTX experiments:
Ions are neo-classical,
Electron are anomalous,
Density profile is not "stiff"
(K.Tritz, APS-06)



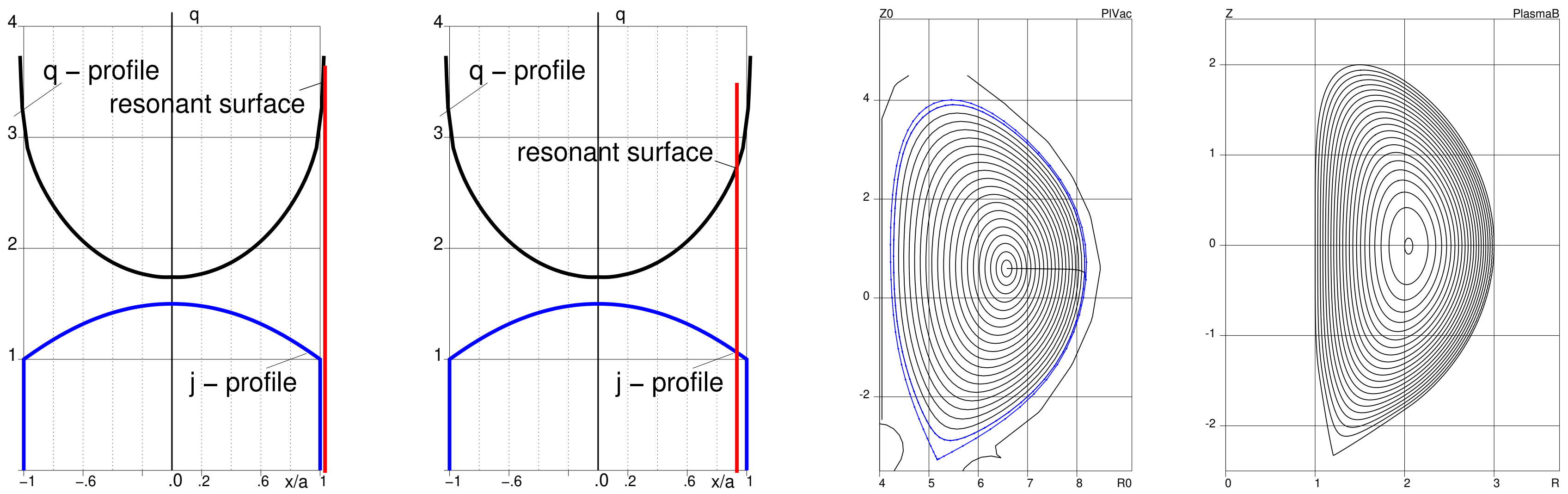




- Dependence of $\chi_{e,pert}$ on T_e gradient suggests critical gradient threshold

A widespread belief in MHD theory is that the high edge current density is destabilizing (“peeling modes”)

$$W \propto \int \frac{j' R \psi^2 d\rho}{B_{tor}\left(\frac{1}{q} - \frac{n}{m}\right)} \simeq \frac{j_{edge}}{B_{tor}\left(\frac{1}{q_{edge}} - \frac{n}{m}\right)} \psi^2$$



case 1: $mq_a < n$ *Ideally unstable*

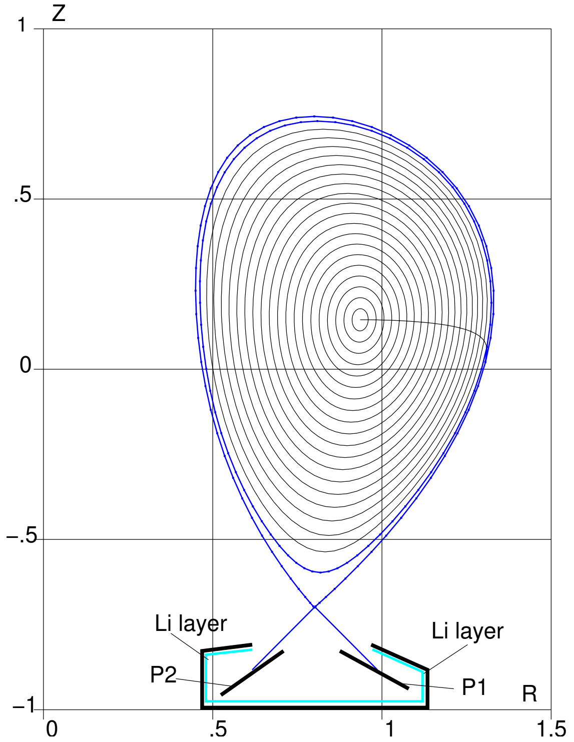
case 2: $mq_a > n$ *Tearing stable*

Ideally & tearing stable

$\mathbf{j/B}$ *=const equilibrium,* $j_{edge} \neq 0$

In presence of a separatrix, the finite edge current density is stabilizing. No ELMs.

LiWF regime eliminates the effects driving impurities to the plasma core. Introduces the mirror-machine physics into SOL



Three forces are acting on impurities on the way from PFC to the plasma

1. *A small electro-static force ZeE_{SOL}, directed back to the plate.*
2. *Friction $R_V \propto Z^2$ with the ion flow, also directed back to the plate.*
3. *Thermo-force $R_T \propto Z^2$, driving impurities into the plasma.*

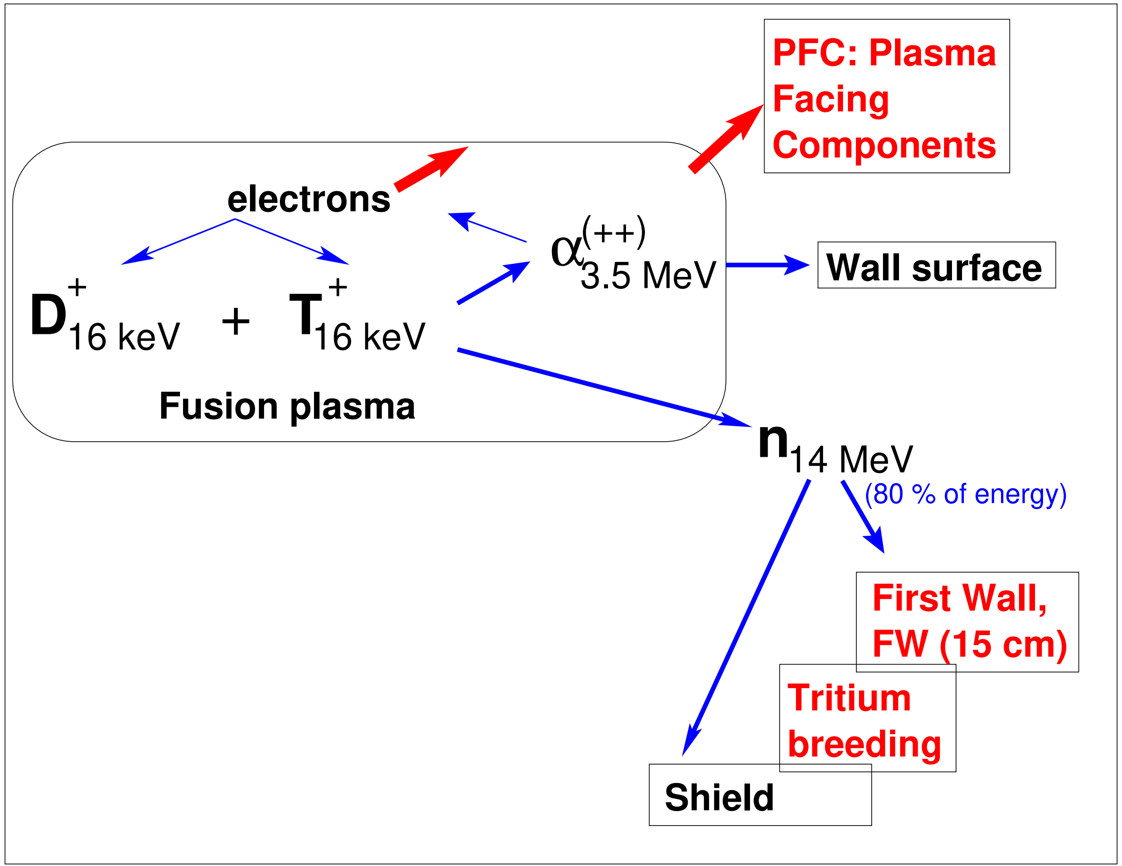
In addition, there is a direct plasma-wall interaction through the radial bursts of blobs

In collisionless SOL the thermo-force is absent, leading to $Z_{eff} \simeq 1$

Blobs are not expected (as in QHM regime on DIII-D)

Magnetic fusion based on LiWF concept is capable of approaching reactor development

The BBBL70 relies on plasma heating by α-particles



Flow pattern of fusion energy (since the 50s)

Ignition criterion:

$$f_{pk} \cdot \langle p \rangle \cdot \tau_E^* = 1$$
[MPa · sec]

Peaking factor f_{pk}:

$$f_{pk} \equiv \frac{\langle 16 p_D p_T \rangle}{\langle p \rangle^2}$$

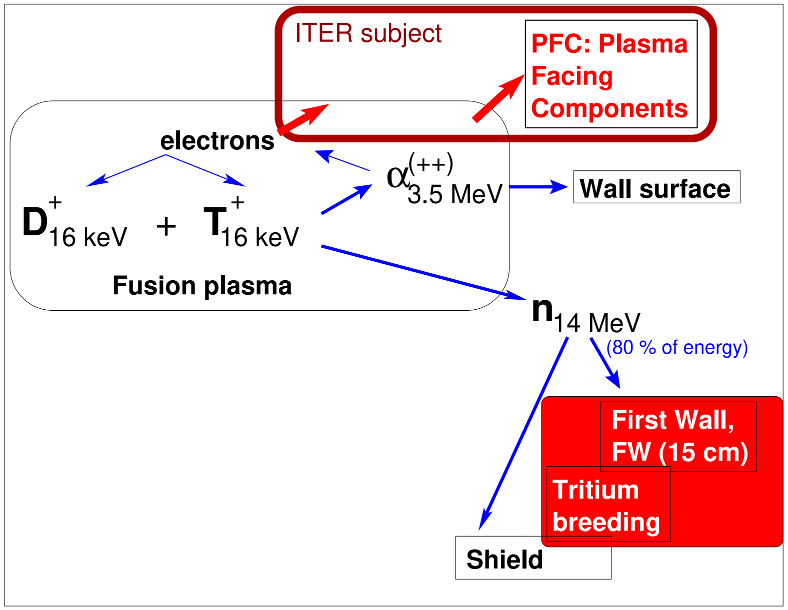
Plasma pressure p:

$$p = p_e + p_D + p_T + p_\alpha + p_I$$

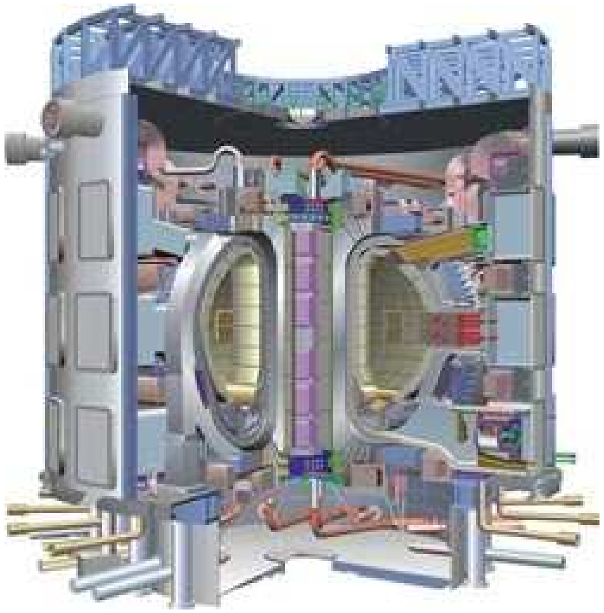
The plasma is in the “hot-electron” regime, the worst one.

All present day machines work in the “hot-ion” mode

ITER is the first machine targeting the α-heating regime



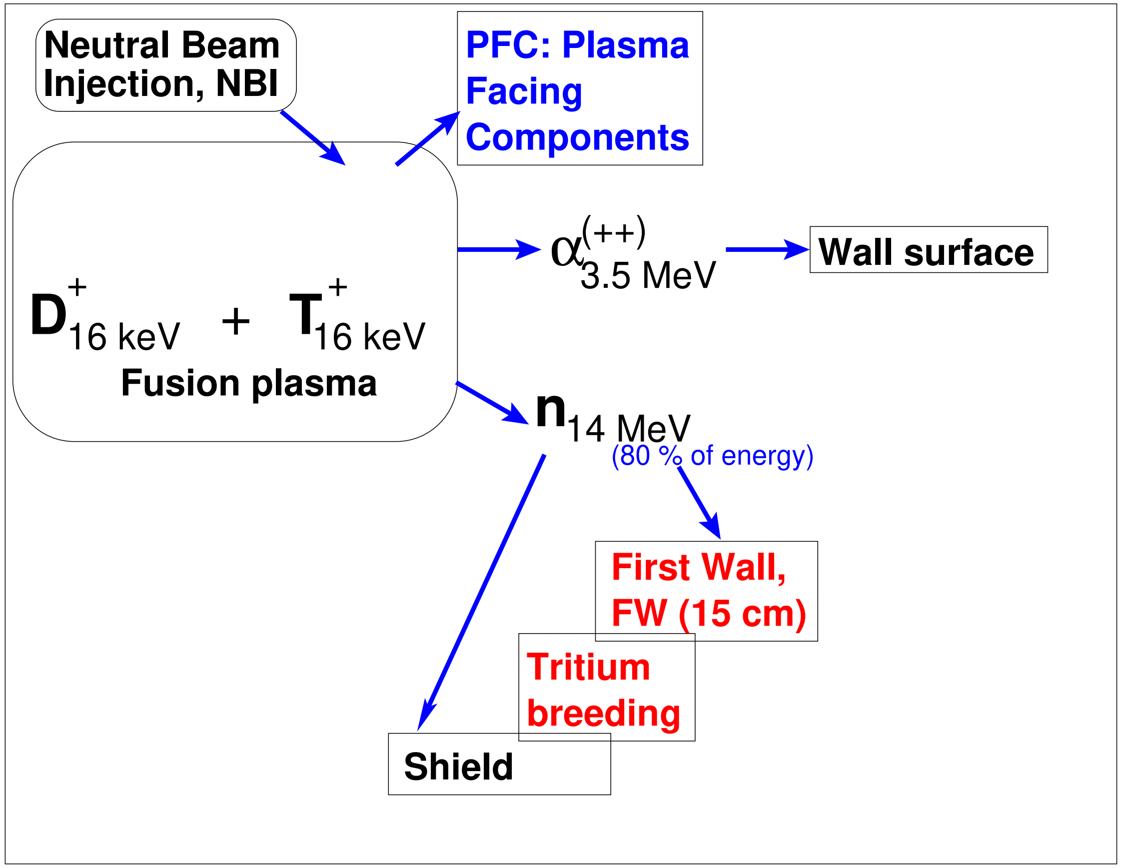
$\Longrightarrow$



Even in expected “burning plasma” regime ITER is still dealing mostly with plasma physics issues.

Being an implementation of the old concept, ITER only barely touches the reactor aspects of fusion

The LiWF is insensitive to major unknowns in plasma physics



LiWall concept has a clean pattern of flow of fusion energy

α-particles are free to go out of plasma

NBI controls both the temperature and the density

$$P_{NBI} = \frac{3}{2}\frac{\langle p\rangle V_{pl}}{\tau_E},$$

$$\frac{dN_{NBI}}{dt} = \Gamma^{ions}_{core\rightarrow\ edge}$$

Super-Critical Ignition (SCI) confinement is necessary to make NBI work this way

$$\tau_E >> \tau_E^*$$

LiWF conceptually resolves fundamental issues, intractable for BBBL70

The criterion of conceptual relevance to reactor R&D is very simple: ability of delivering

15 MWa/m^2

of neutron fluence,

or burn-up of

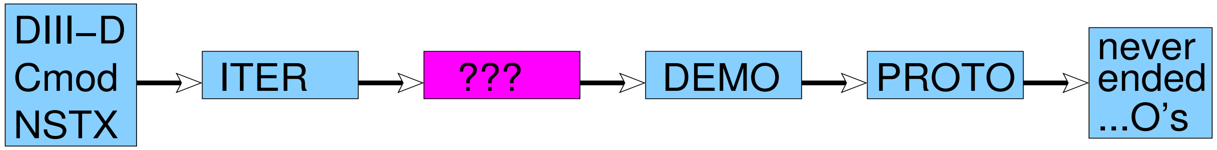
1 kg(T)/m^2(FW)

A compact Reactor Development Facility (RDF) with new plasma regimes is absolutely necessary

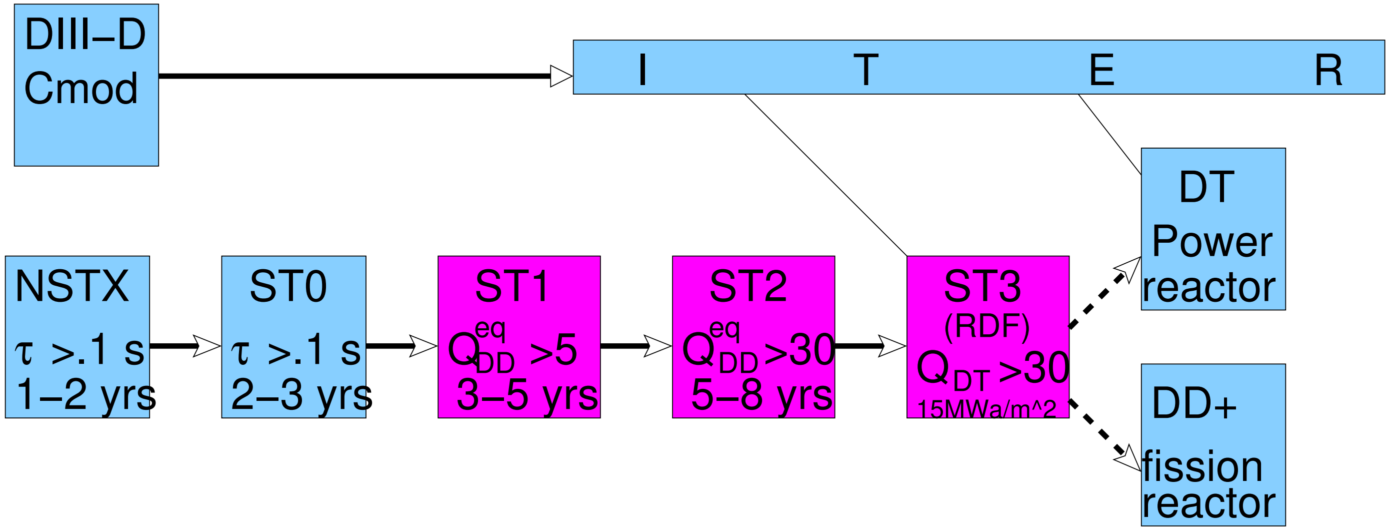
(ITER is capable of only 0.3-0.4 MWa/m^2 (burn-up of 10-15 kg of T, instead of 650 kg)

The Orbach charge (Feb. 2007) can be interpreted

as another chance to ignore the basics of strategy and follow the old teaching

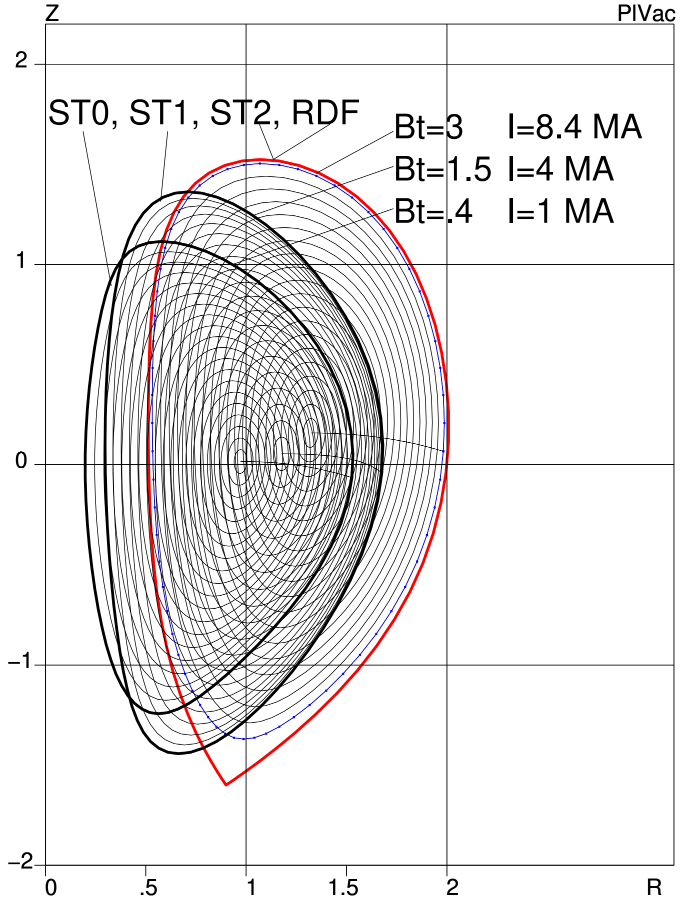


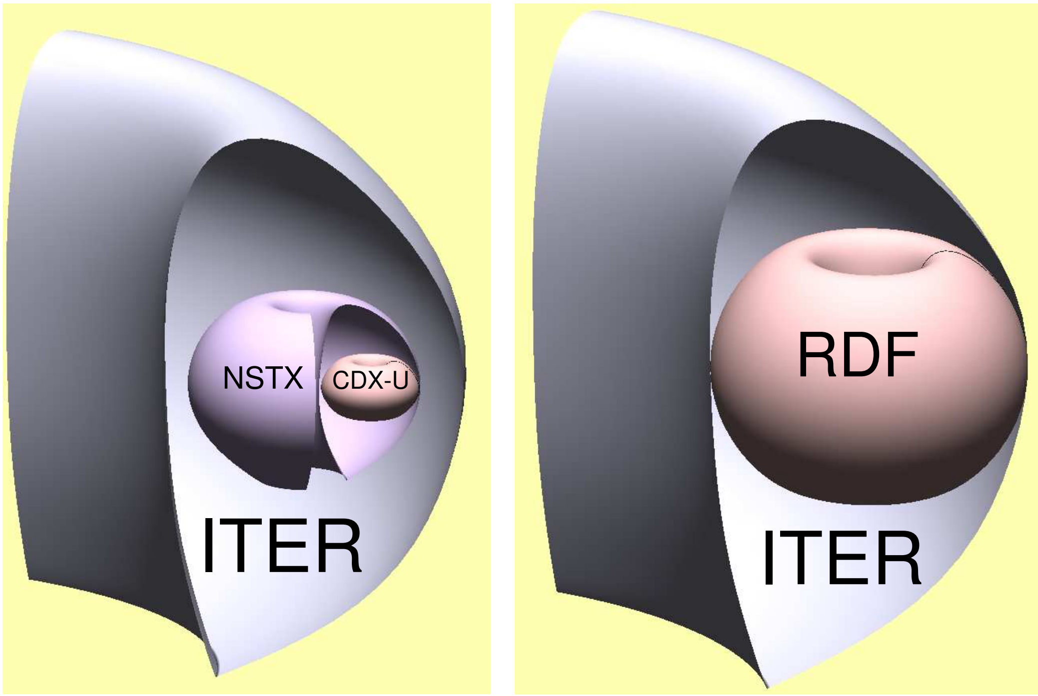
or as an opportunity to develop the LiWall plasma regimes for RDF on the time scale competitive with ITER



LiWF strategy does not need fusion power ("burning plasma") until step 3

Increase in performance of STs is provided by the increase in magnetic field and I_{pl}





RDF with $P_{DT} = 0.2 - 0.5$ GW is 27 times smaller than ITER

3 steps rely exclusively on the "present understanding of fusion" and existing technology. No big leaps.

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, $R_{outer} \simeq 1.65$ m.	Achieve RTM-like confinement: $\tau_E \rightarrow 2-3 \times \tau_{E,NSTX}$.	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.

The success of ST0 in the RDF program would bootstrap the necessary funding of fusion

LiWF is consistent with common sense in all reactor issues

Issue	LiWF	BBBL70 concept of “fusion”
The target	RDF as a useful tool	Political “burning” plasma
Operational point:	$P_{NBI} = E/\tau_E$	ignition criterion $f_{pk} p \tau_E = 1$
Hot-α, 3.5 MeV	”let them go as they want”	“confine them”
Cold He ash	residual, flashed out by core fueling	“politely expect it to disappear”
$P_\alpha = 1/5 P_{DT}$	goes to walls, Li jets	dumped to SOL
Power extraction from SOL	conventional technology for $\frac{\tau_E^*}{\tau_E} P_\alpha$	no idea except to radiate 90 % of P_α by impurities
Plasma heating	“hot-ion” mode: NBI $\rightarrow i \rightarrow e$	first heat useless electrons: $\alpha \rightarrow e \rightarrow i$
Use of plasma volume	100 %	25-30 %
Tritium control	pumping by Li	tritium in all channels and in dust
Tritium burn-up	10%	fundamentally limited to 2-3 %
Plasma contamination	kill the Z^2 thermo-force, clean plasma by core fueling	invites all “junk” from the walls to the plasma core
He pumping	Li jets, as ionized gas, $p_{in} < p_{out}$	gas dynamic, $p_{in} > p_{out}$
Fusion producing β_{DT}	$\beta_{DT} > 0.5\beta$	diluted: $\beta_{DT} < 0.5\beta$

As a reactor concept, the BBBL70 is not consistent with common sense

LiWF has a robust plasma physics and technology basis. It contributes to present understanding of fusion in unique way

Issue	LiWF	BBBL70 concept of "fusion"
Physics:		
Confinement	diffusive, RTM≡ $\chi_=\chi_e = D = \chi_i^{neo}$	turbulent thermo-conduction
Anomalous electrons	play no role	is in unbreakable 40 years old marriage with anomalies
Transport database	scalable by RTM (Reference Transp. Model)	religious beliefs on applicability of scalings to "hot e"-mode
Sawteeth, IREs	absent	unpredictable and inavoidable
ELMs, $n_{Greenwald}$-limit	absent	intrinsic for low T_{edge}
p'_{edge} control	by RMP through n_{edge}	through T_{edge} and reduced performance
Fueling	existing NBI technology	no clean idea yet
Fusion power control	existing NBI technology	no clean idea yet
Operational DT regime	identical to DD	needs fusion DT power for its development
Time scale for RDF:	$\Delta t \simeq 15$ years	$\Delta t \simeq \infty$
Cost:	$\simeq$ $2-2.5 B for RDF program	$\simeq$ $20 B with no RDF strategy

3 step RDF program of LiWF suggests a way for bootstraping its funding

With no tangible returns the BBBL70 is irrational and compromizes credibility of fusion

In the late 80s, when ITER project failed with mission of R&D for nuclear components of a reactor

magnetic fusion made a transition from a phase of "difficult" problem to the phase of "complicated" problem

In science, the phase of "difficult" problem is always linked to the progress.

The phase of "complicated" problem means stragnation and fragmentation. With time, the situation in this phase becomes only worse. There is no "self-organized"-like backward phase transition.

LiWF concept is able to return magnetic fusion back to its "difficult" phase, where unresolved problems are well-specified and most of them will be resolved step by step

After 40 years since acceptance of tokamaks as a mainstream approach for magnetic fusion it is the time to switch into a reasonable reactor concept

Ray Orbach and Sam Bodman give us a unique chance to do this in time

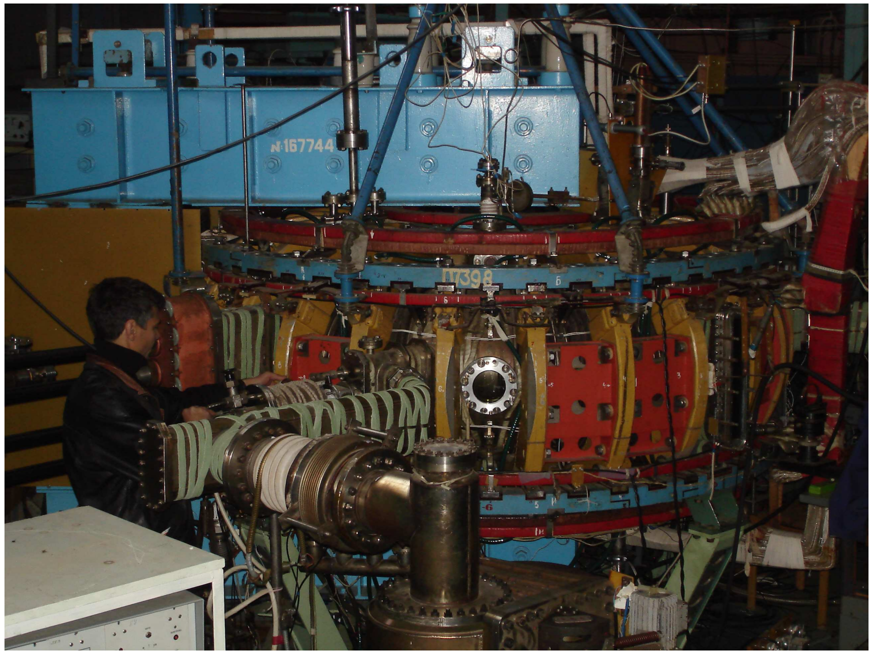
So far, there is no implementation of the pumping PFC surface together with core fusion.

At the same time with only one exception of ill-fated Dimes experiment on DIII-D, the effects of lithium conditioning on confinement, stability, radiation, Greenwald limit were exclusively positive.

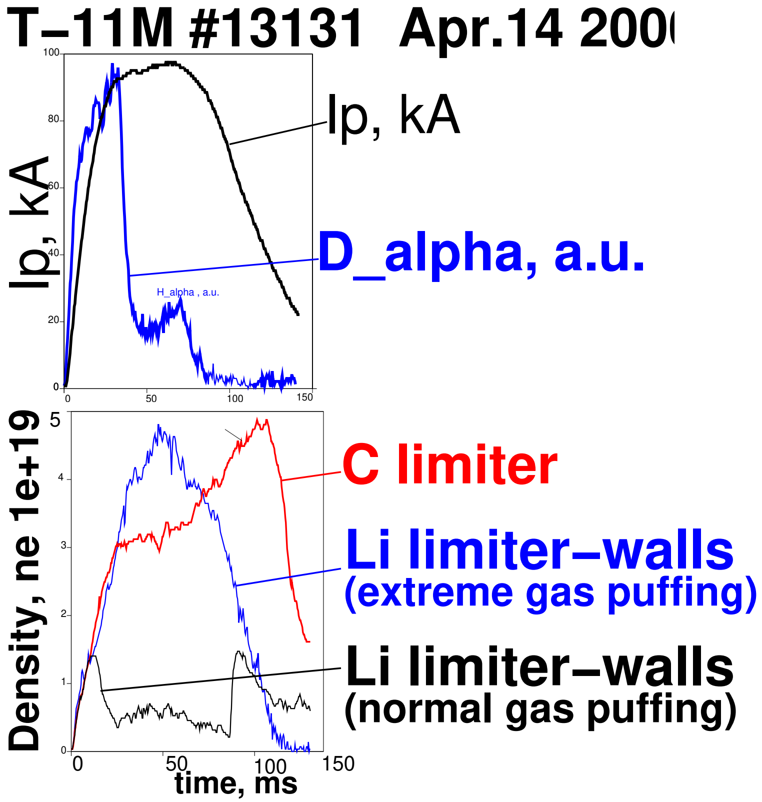
NSTX is the most ready device for making a conclusive experiment

In 1998 T-11M tokamak (TRINITI, Troitsk, RF) demonstrated outstanding plasma pumping by Li coated walls

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



T11M and DoE's APEX/ALPS technology programs triggered the idea of LiWalls



Lithium completely depleted the discharge in T-11M

In PPPL, CDX-U demonstrated similar pumping capabilities

Reference Transport Model (RTM) is natural for LiWall regime

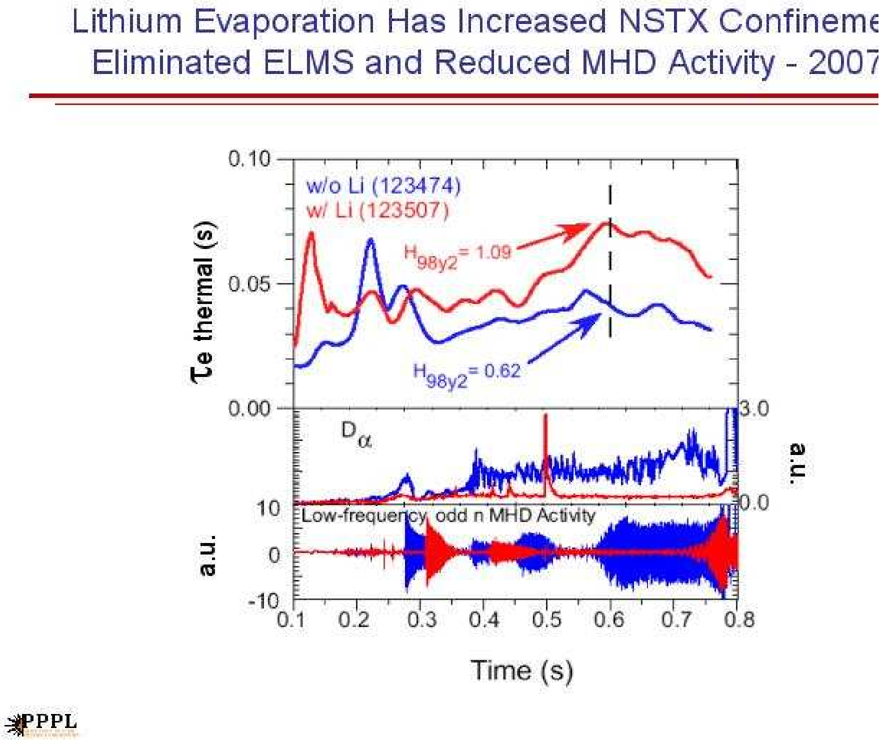
$$
\begin{aligned}
q_i &= \chi_i^{neo-classics} \nabla T_i, && \textit{not important}, \\
q_e &= \chi_i^{neo-classics} \nabla T_e, && \textit{not important}, \\
\Gamma_{i,e} &= \chi_i^{neo-classics} \nabla n &&
\end{aligned}
\qquad (6.1)
$$

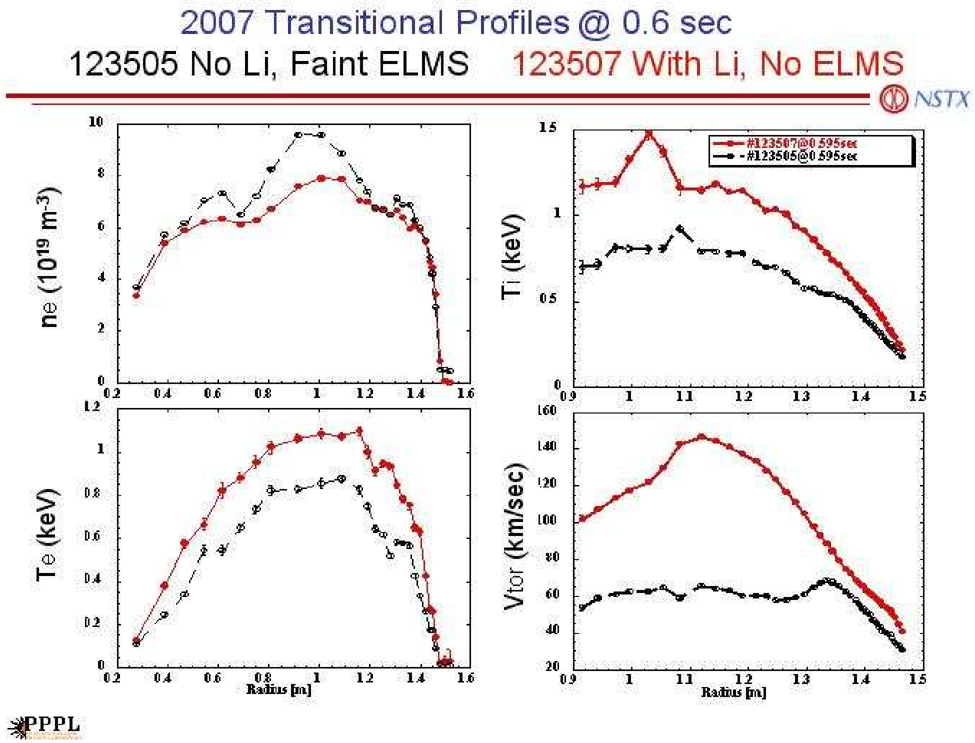
Table 1

Parameter	CDX-U	RTM	RTM-0.8	glf23	Comment
$\dot{N}$, 10^{21}part/sec	1-2	.98	0.5	0.8-3	Gas puffing rate adjusted to match
β_j	0.160	0.151	0.150	0.145	measured β_j
l_i	0.66	0.769	0.702	0.877	inernal inductance
V, Volt	0.5-0.6	0.77	0.53	0.85	Loop Voltage
τ_E, msec	3.5-4.5	2.7	3.8	2.3	
$n_e(0)$, $10^{19} part/m^3$		0.9	0.7	0.9	
$T_e(0)$, keV		0.308	0.366	0.329	
$T_i(0)$, keV		0.031	0.029	0.028	

RTM does not contradict CDX-U measurements and equilibrium reconstruction

NSTX had 2 campaigns with Li conditioning by evaporation



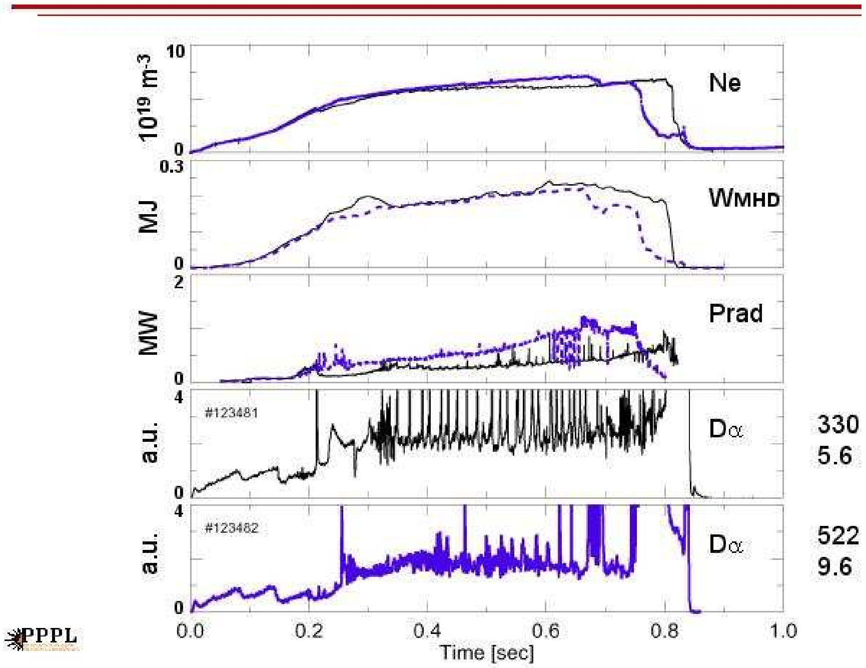


There are indications of improved confinement with Li conditioning on NSTX after evaporation.

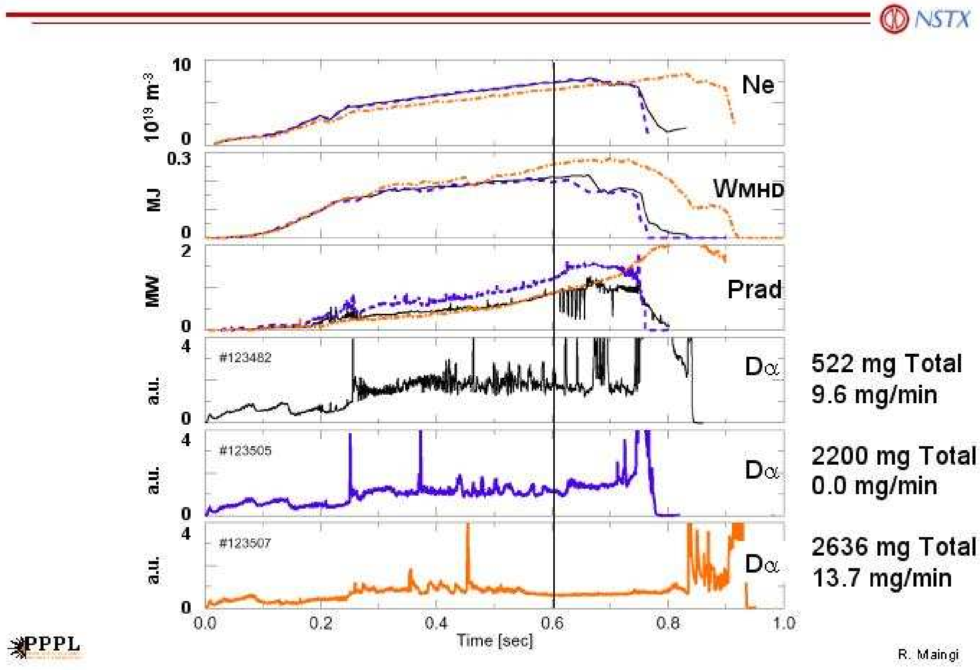
NSTX is not yet in the LiWall regime. There is no effect on the density rise

ELMs were suppressed after Li conditioning on NSTX

ELMS → Faint Elms → No ELMS Transition (I)



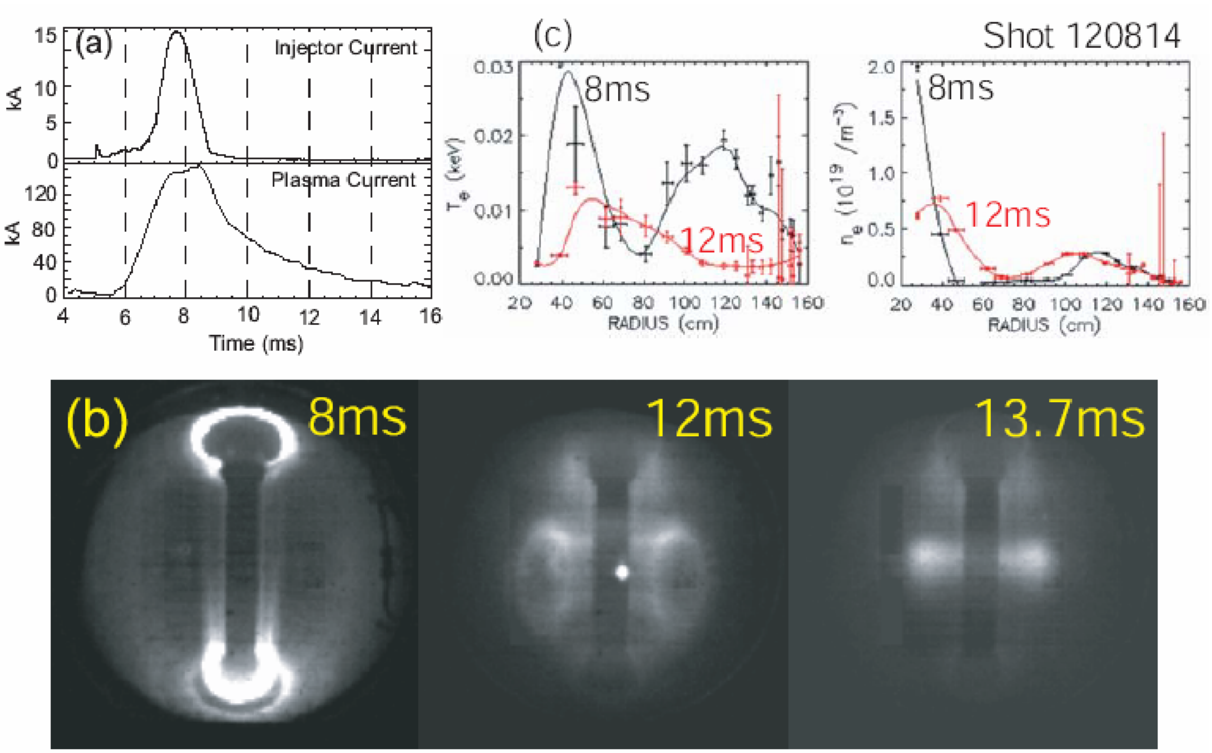
ELMS → Faint Elms → No ELMS Transition (II)

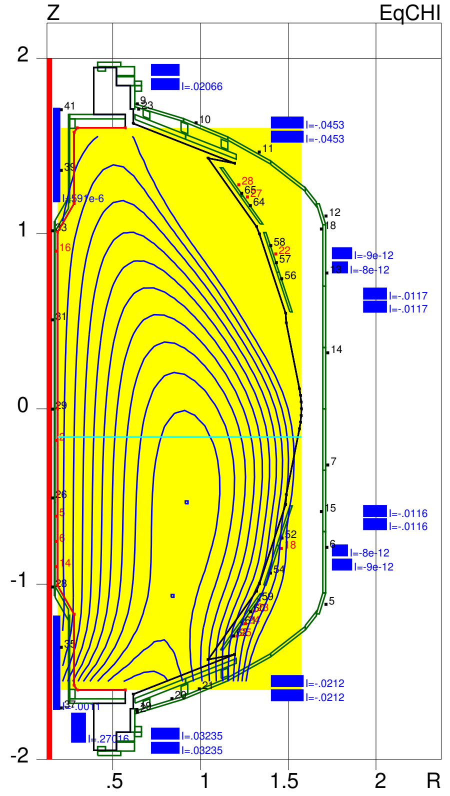


Four shots are shown (D.Mansfield): before Li evaporation, after depositing ≃200 mg, then +1700 mg, and +400 mg.

It was a surprise, although consistent with tendencies, how easy ELMSs were suppressed

LiWF is compatible with both inductive and CHI start-up



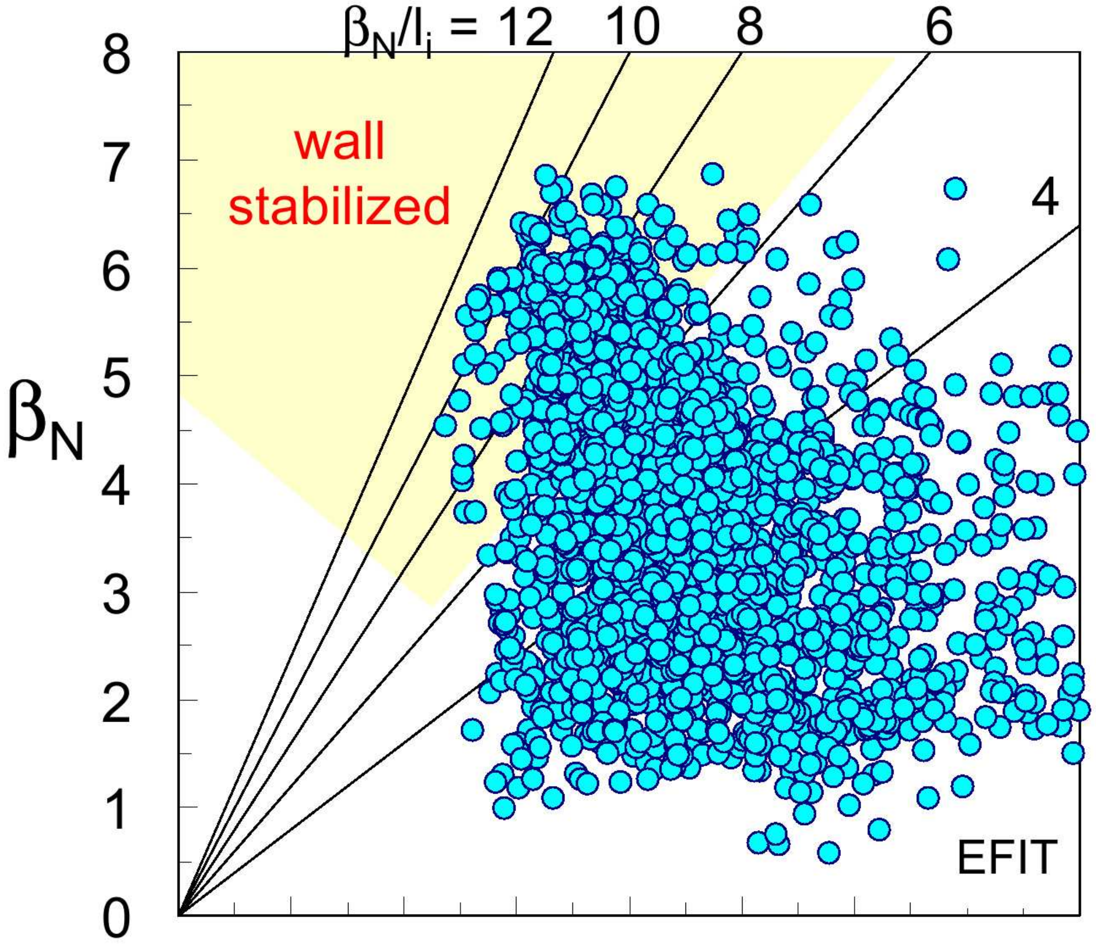


In 2006 CHI startup generated 160 kA current in NSTX

From R.Raman at al., PPPL-4207 (2007)

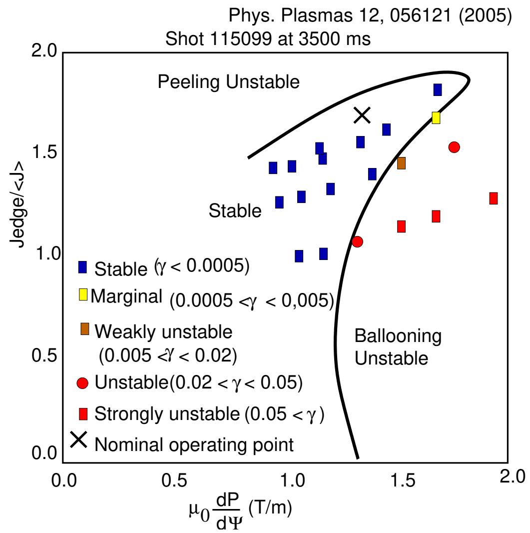
With Li electrodes, even in the worst case scenario, CHI will create a perfect, transient Li plasma with Z_{eff}=3 (typical for C-wall machines)

The stability data base for RDF is already in a good shape

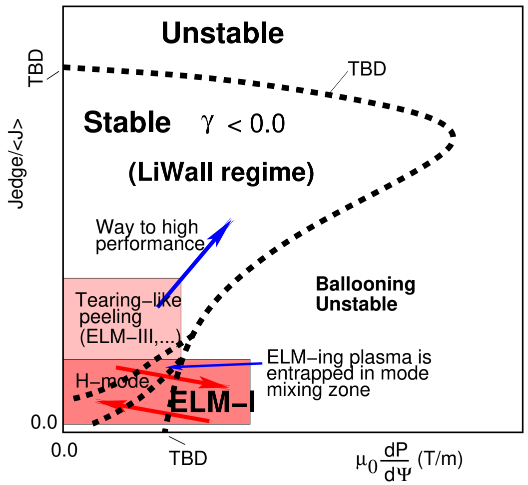


In 2004, beta in NSTX has approached the record level of 40 %

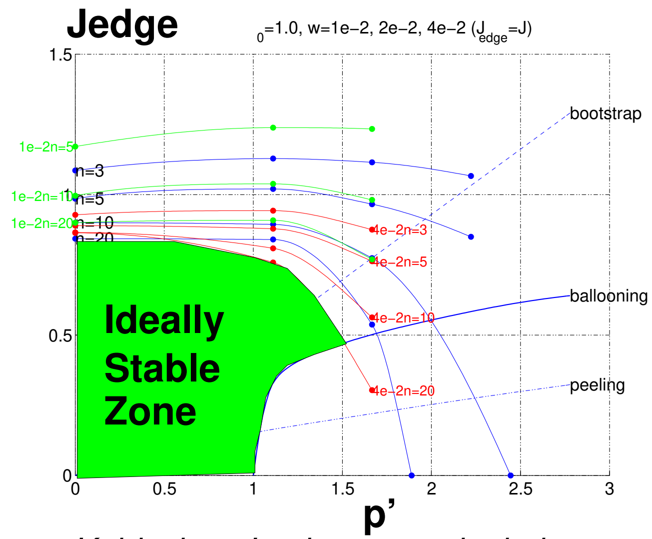
Peeling-ballooning diagram of Phyl Snyder initiated theory of ELMs



Peeling-ballooning diagram (P.Snyder)



"Heuristic diagram" (Zakharov, 2005)



Keldysh Institute calculation, (Medvedev, 2003)

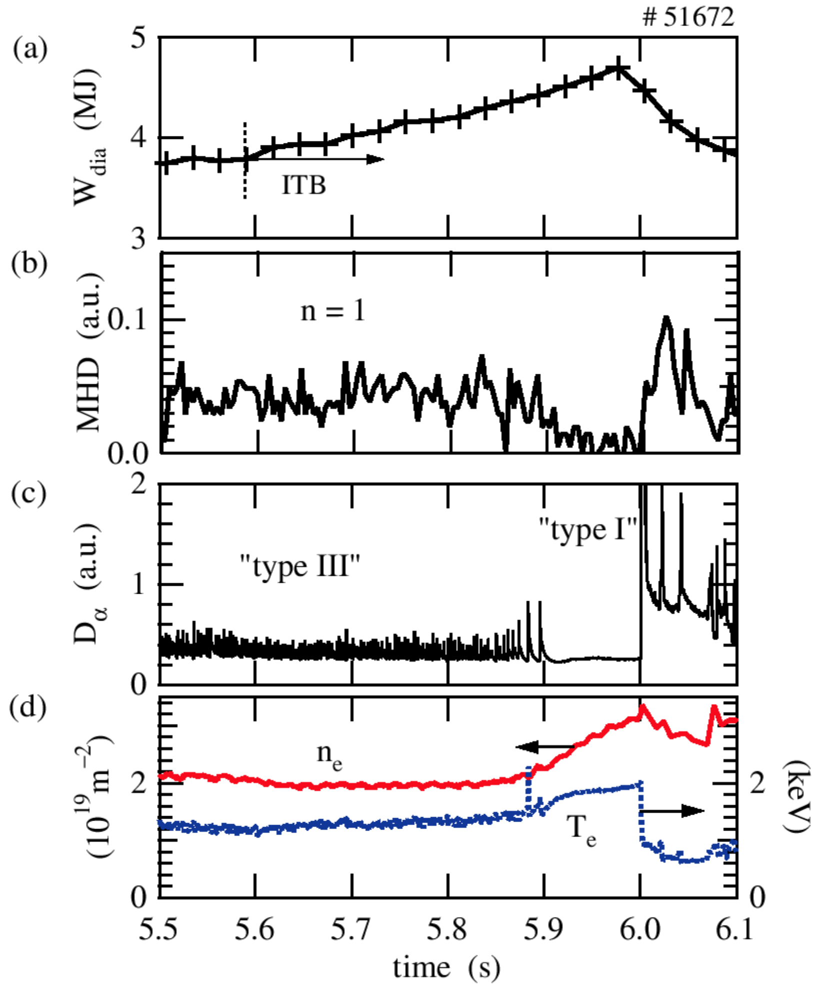
New understanding is that the finite current density at separatrix is stabilizing for ELMs, while pressure remains destabilizing.

1-D energy principle is now written to check a single point $p = 0, j_{egde} \neq 0$

$$W = \oint\oint \psi(l) i_{ll'} \psi^*(l') dl dl' - \frac{\bar{J}_\varphi}{B_\varphi}\oint \frac{\psi^* u' + \psi u'^*}{2} dl, \quad \psi \equiv -\frac{B_p r}{B_\varphi} u' - inu$$

High plasma T_{edge} in LiWF is consistent with the high performance spot on stability diagram

Quiescent period in JET ITB experiments is consistent with this theory



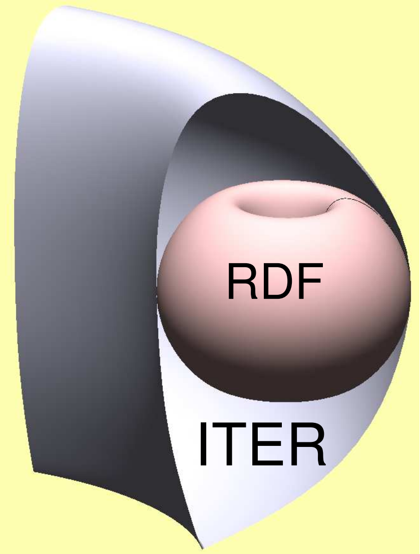
JET has a quiescent regime as transient phase from ELM-III to ELM-I

"Edge issues in ITB plasmas in JET"

Plasma Phys. Control. Fusion 44 (2002) 2445Â-2469 Y. Sarazin, M. Becoulet, P. Beyer, X. Garbet, Ph. Ghendrih, T. C. Hender, E. Joffrin, X. Litaudon, P. J. Lomas, G. F. Matthews, V. Parail, G. Saibene and R. Sartori.

The authors emphasized the crucial role of the edge current density

RDF is a powerful neutron source (0.2-0.5 GW) for reactor development

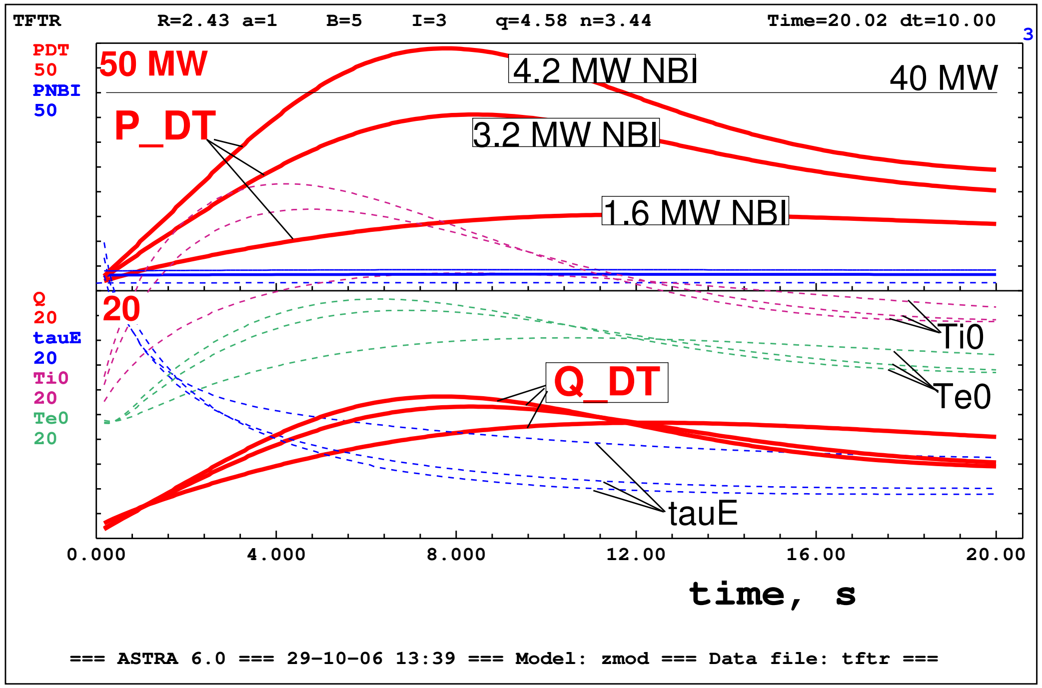


RDF should target three mutually linked objectives of magnetic fusion

1. *High power density plasma regime, $\simeq$ 10 MW/m^3*
2. *Fluence of neutrons 15 MWa/m^2 for designing the First Wall*
3. *Self-sufficient Tritium Cycle*

LiWF approach, together with existing technology, seems to be capable of accomplishing this mission

ASTRA-ESC simulations of TFTR, B=5 T, I=3 MA, 80 keV NBI



Even with no α-particle heating:

$$P_{NBI} < 5 \text{ [MW]},$$
$$\tau_E = 4.9 - 6.5 \text{ [sec]},$$
$$P_{DT} = 10 - 48 \text{ [MW]},$$
$$Q_{DT} = 9 - 12$$

within TFTR stability limits, and with small PFC load (< 5 MW)

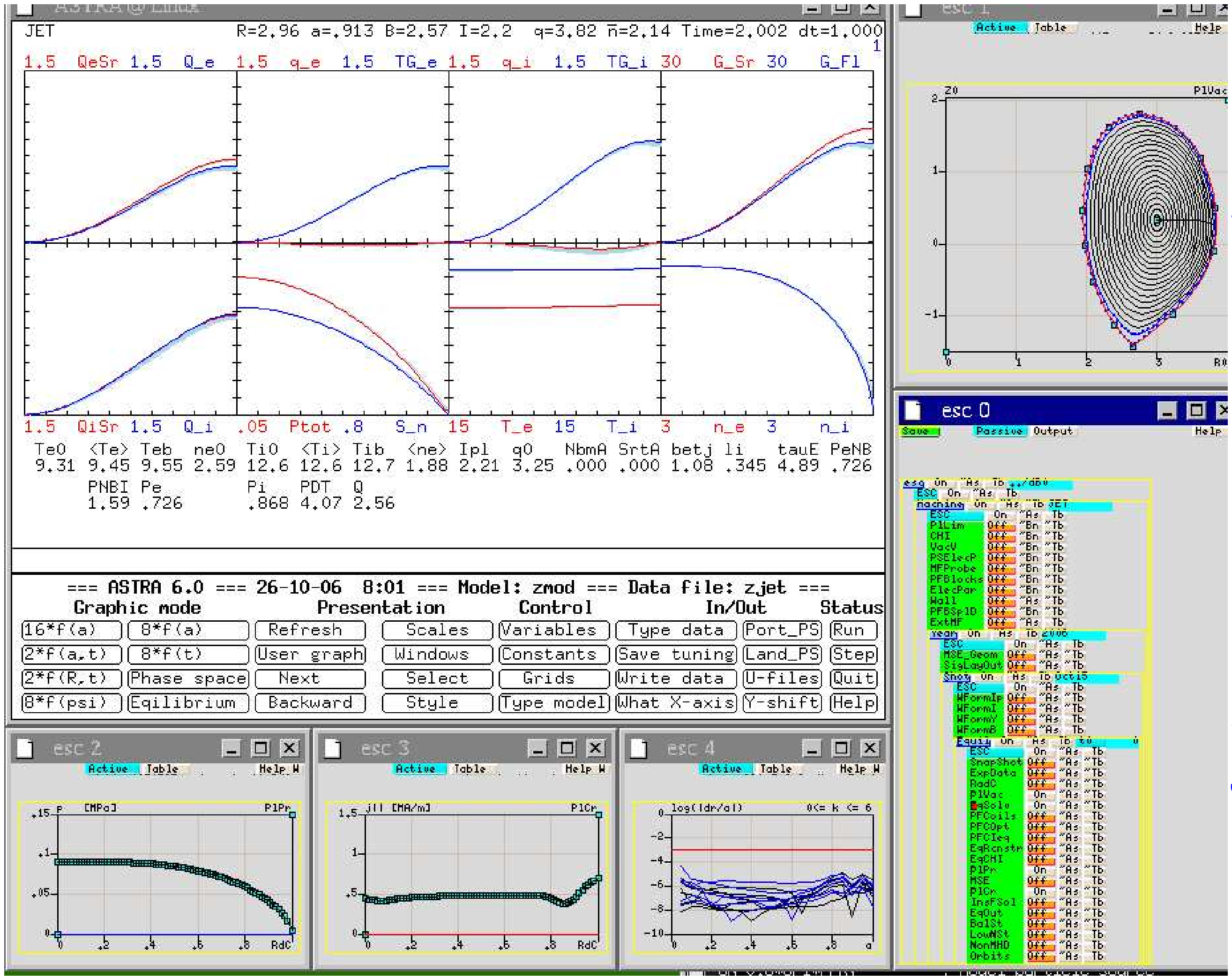
	PNBI	n	T	P DT	Q DT	tauE	nend	Ti0	Te0	gb %
(a)	1.65	0.3	10	15.4	9.34	6.54	0.42	18.7	14.8	1.64
(c)	3.30	0.3	10	35.5	10.6	4.04	0.55	17.6	13.6	1.96
(d)	4.16	0.3	10	48.9	11.6	3.58	0.59	17.5	13.4	1.96

The "brute force" approach ($P_{NBI} = 40$ MW) did not work on TFTR for getting $Q_{DT} = 1$. With $P_{DT} = 10.5$ MW only $Q_{DT} = 0.25$ was achieved.

In the LiWall regime, using less power, TFTR could challenge even the $Q = 10$ goal of ITER

(Ignition criterion corresponds to $Q = 5$)

ASTRA-ESC simulations of JET, B=2.6 T, I=2.2 MA, 50 keV NBI



Hot-ion mode:

$T_i = 12.6$ *[keV]*,

$T_e = 9.45$ *[keV]*,

$n_e(0) = 0.3 \cdot 10^{20}$,

$\tau_E = 4.9$ *[sec]*,

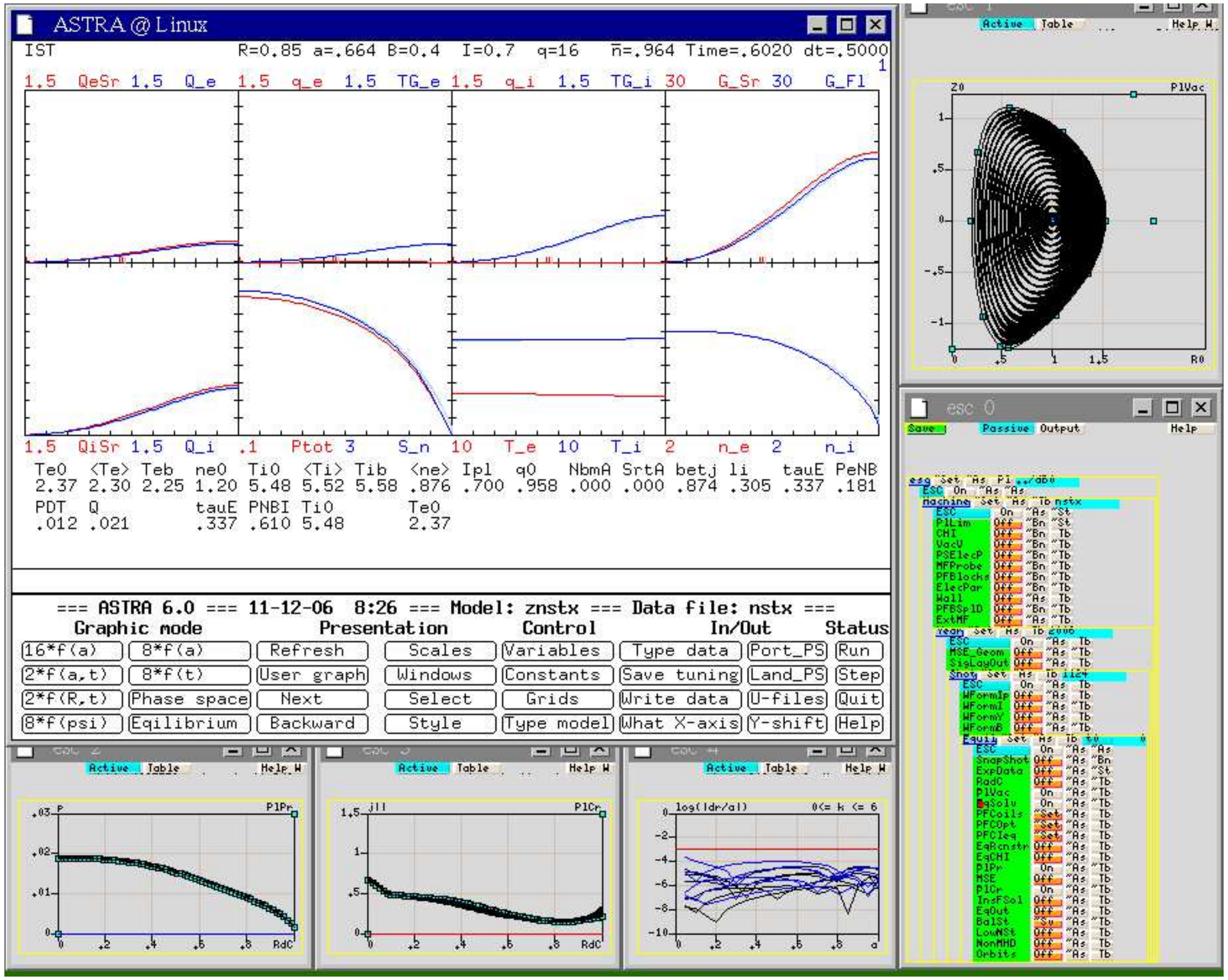
$P_{NBI} = 1.6$ *[MW]*

For 50 keV NBI,

3+2 MWs are available

Can be experimentally tested on JET with intense Be conditioning

ASTRA-ESC simulations of ST-0, B=0.4 T, I=0.7 MA, 0.6 MW, 20 keV NBI



Hot-ion mode:

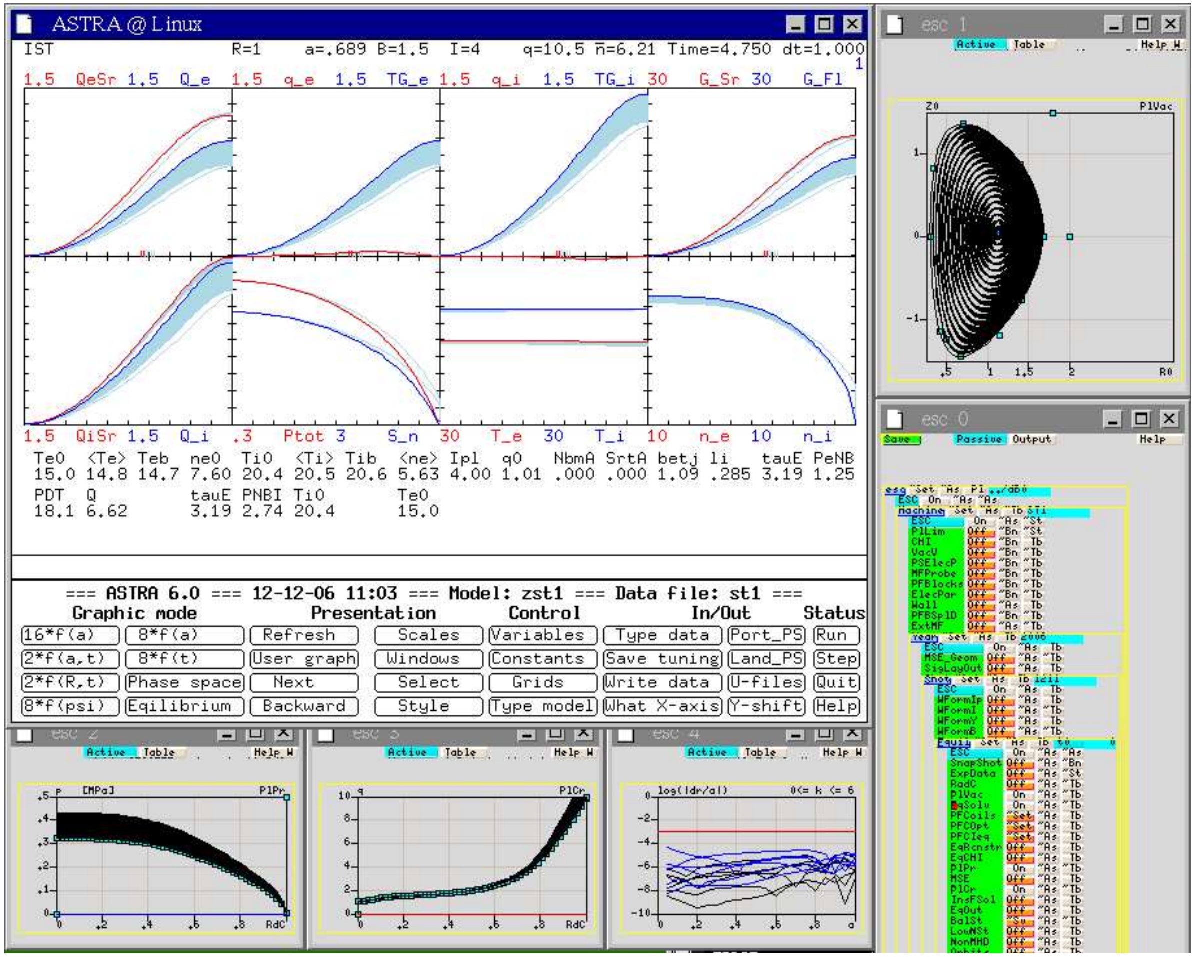
$$T_i = 5.5 \text{ [keV]},$$
$$T_e = 2.5 \text{ [keV]},$$
$$n_e(0) = 0.14 \cdot 10^{20},$$
$$\tau_E = 0.33 \text{ [sec]},$$
$$P_{NBI} = 0.61 \text{ [MW]}$$

NBI energy should be consistent with the plasma temperature:

$$E_{NBI} = 2.5(T_i + T_e)$$

ST0 should reach at least 1/3 of τ_E predicted by the Reference Model

ASTRA-ESC simulations of ST-1, B=1.5 T, I=4 MA, 2.7 MW, 80 keV NBI

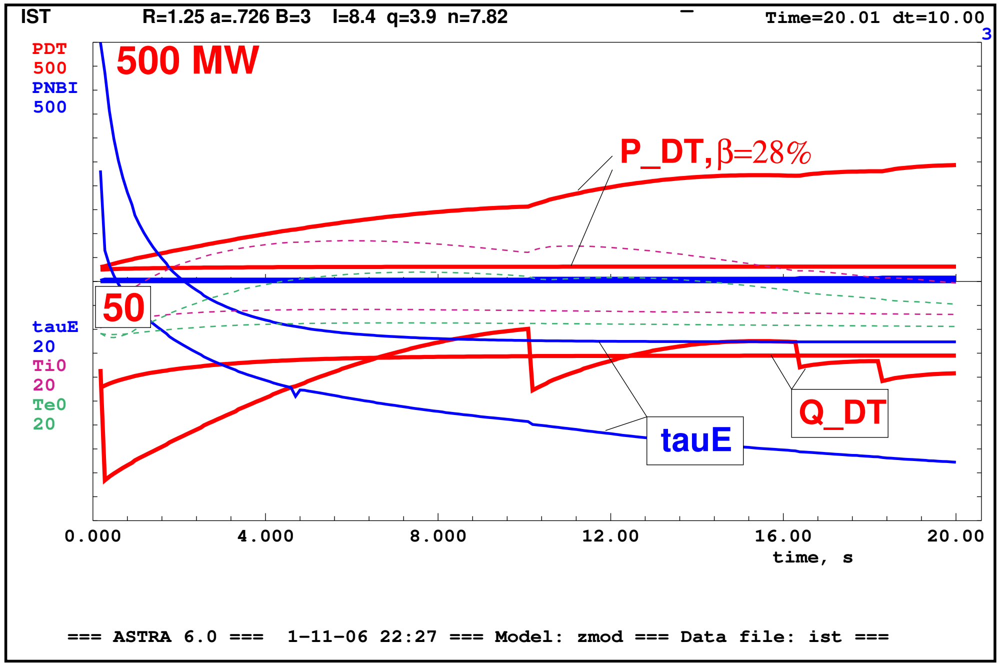


Hot-ion mode:

$$\beta = 0.35,$$
$$T_i = 20 \text{ [keV]},$$
$$T_e = 15 \text{ [keV]},$$
$$n_e(0) = 0.75 \cdot 10^{20},$$
$$\tau_E = 3.19 \text{ [sec]},$$
$$P_{NBI} = 2.7 \text{ [MW]},$$
$$P_{DT}^{equiv} = 18,$$
$$Q_{DT}^{equiv} = 6.6$$

ST-1 could be the first machine in super-critical regime, $Q_{DT}^{equiv} > 5$

ASTRA-ESC simulations of ST2, B=3 T, I=8.4 MA, 80 keV NBI



$$P_{DT}^{equivalent} \simeq 250\ MW,$$
$$\beta = 28\ \%,$$
$$Q_{DT}^{equivalent} \simeq 40,$$
$$P_{NBI} < 6\ MW,$$
$$\tau_E = 5 - 16\ sec$$

The heat load of divertor plates is small

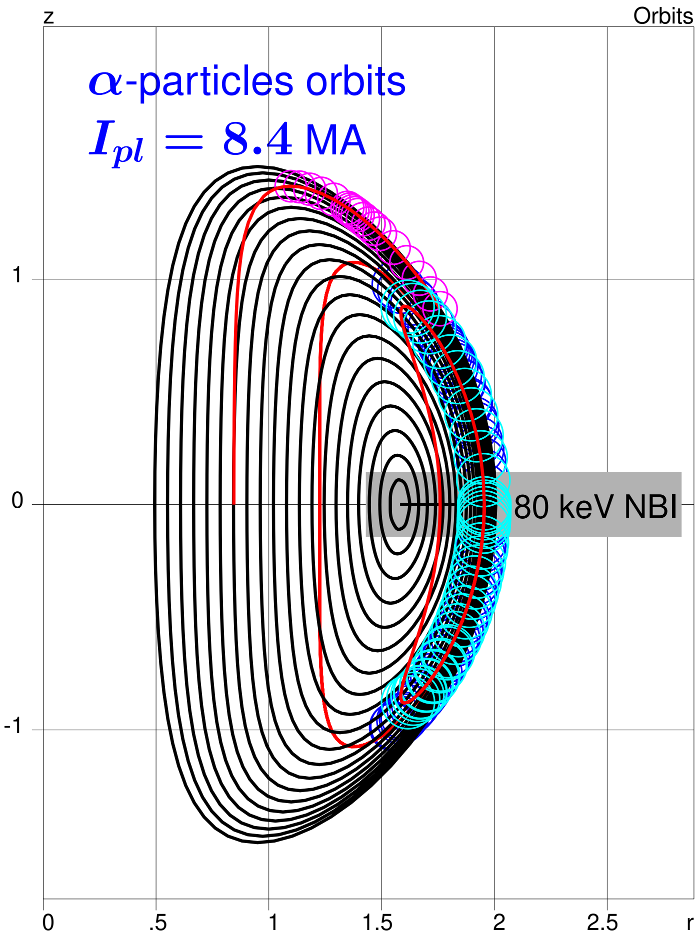
$$P_{NBI} \simeq 6\ MW$$

The regime of ST2 (with no fueling by tritium) is identical to RDF

The mission of ST2 is complete development of the stationary plasma regime for its DT-clone, RDF, (except extraction of α-particles).

Only LiWF approach allows the development of the full regime for RDF even in Princeton area

Large Shafranov shift makes core fueling possible in RDF



The charge-exchange penetration length

$$\lambda_{cx} \simeq \frac{0.6}{n_{e,20}} \frac{V_b}{V_{b,80\ keV}} \ [m]$$

The distance between magnetic axis and the plasma surface in IST

$$R_e - R_0 = 0.3 - 0.5 \ [m]$$

80 keV NBI can provide core fueling and control of fusion power

Even at 8.4 MA 60 % of alphas intersect the plasma boundary and can be intercepted at first orbits (e.g. by Li jets)

Burn-up of tritium is proportional to the energy confinement time, and can be very efficient in LiWF

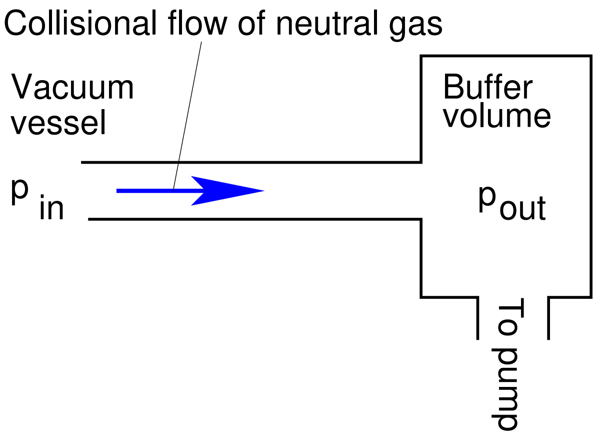
$$n \left\langle \sigma v \right\rangle_{DT,16keV} \bar{\tau}_E = 0.03 n_{20} \bar{\tau}_E$$

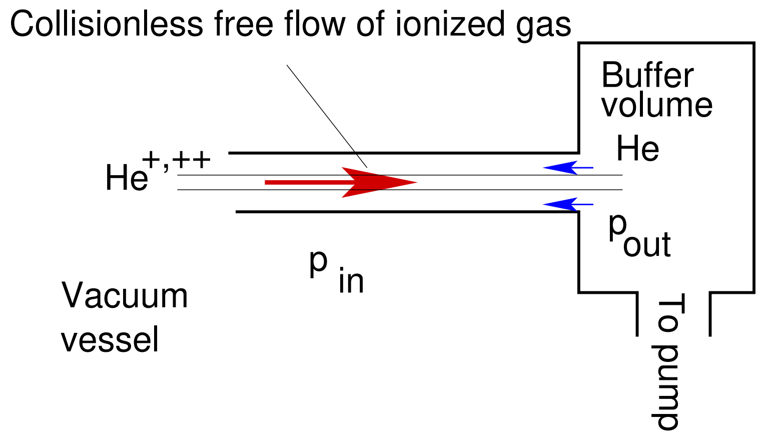
In LiWF the burn-up of tritium could be a significant fraction of unity

On the other hand, due to reliance on ignition criterion $nT\tau_E^*$

BBBL70 is locked into very low, 2-3 %, rate of tritium burn-up

Conventional approach is based on gas-dynamic method





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* $\mathbf{B}$,

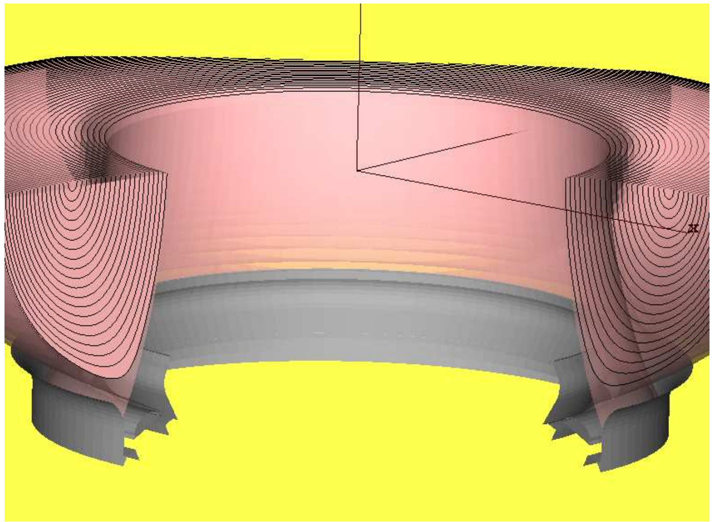
b) Back flow is limited by

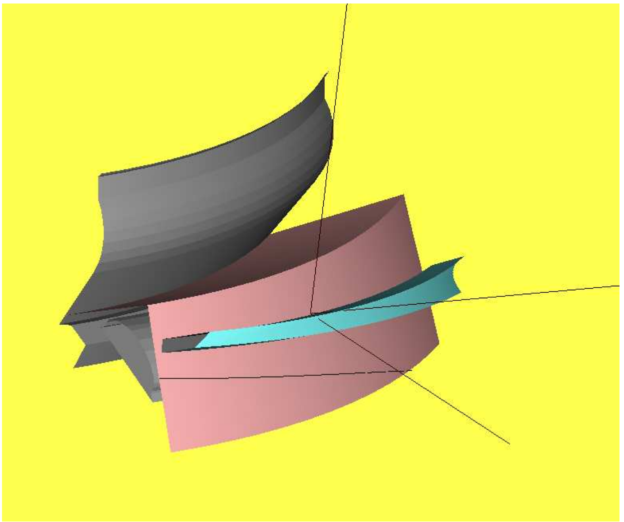
$$\Gamma_{He} = Dn'_x, \quad D = hV_{thermal}$$

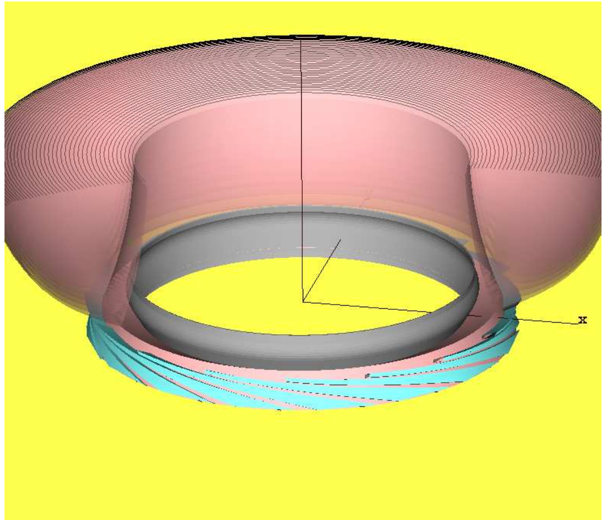
c) Helium density in the vessel plays no role, while D *is in the hands of engineers.*

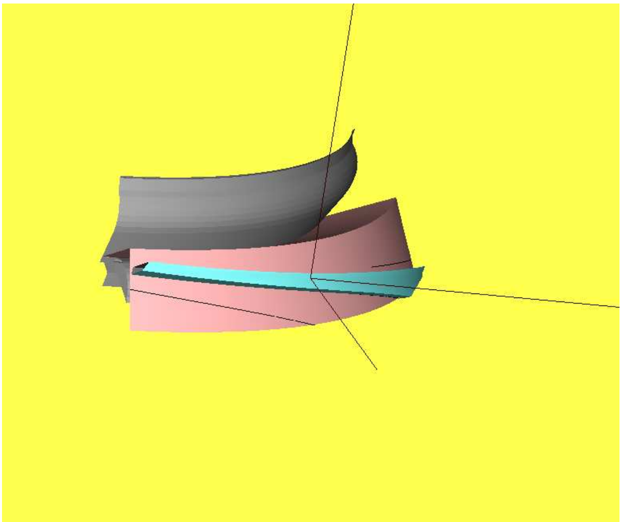
The second scheme is appropriate for the low recycling regime

Honeycomb channel duct utilizes condition $B_{pol} \ll B_{tor}$









Hot-ion regime and expulsion of the fusion products is suitable for DD fusion

Fusion reactions

$$\begin{aligned} D+D &\underbrace{\Longrightarrow}_{50/50\%} \begin{cases} T_{1.01\ MeV} & +p_{3.02\ MeV} \\ He^3_{0.82\ MeV} & +n_{2.45\ MeV} \end{cases}, \\ D+He^3 &\Longrightarrow \quad He^4_{3.6\ MeV}+p_{14.7\ MeV}, \\ D+T &\Longrightarrow \quad He^4_{3.5\ MeV}+n_{14.1\ MeV} \end{aligned} \tag{6.2}$$

Ion Larmor radii of charged products

$$\begin{aligned} &\rho_{T,cm}=\frac{10}{B_T}\sqrt{3}, \quad \rho_{p,cm}=\frac{10}{B_T}\sqrt{\{3,14.7\}}, \quad \rho_{\alpha,cm}=\frac{10}{B_T}\sqrt{3.5}, \\ &\rho_{He^3,cm}=\frac{10}{B_T}\sqrt{1.23} \qquad \text{– can be confined} \end{aligned} \tag{6.3}$$

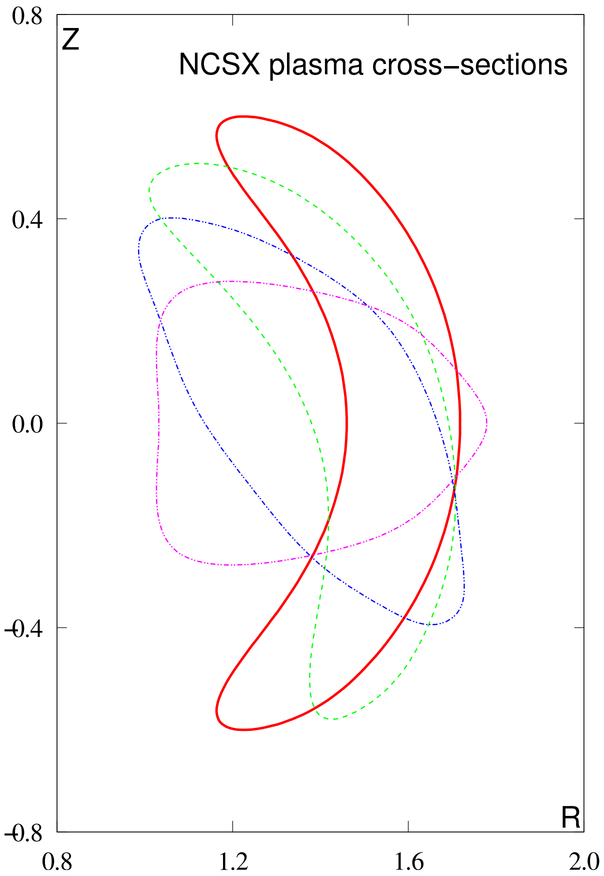
In $D+D, D+He^3$ fusion, the ash products have the same Larmor radii

$$\rho_{T,cm} \simeq \rho_{p,cm} \simeq \rho_{\alpha,cm} \tag{6.4}$$

and can be expelled on the first orbits.

LiWF is uniquely compatible with J.Sheffield's view on DD fusion

The 3 steps strategy has a vision beyond the RDF



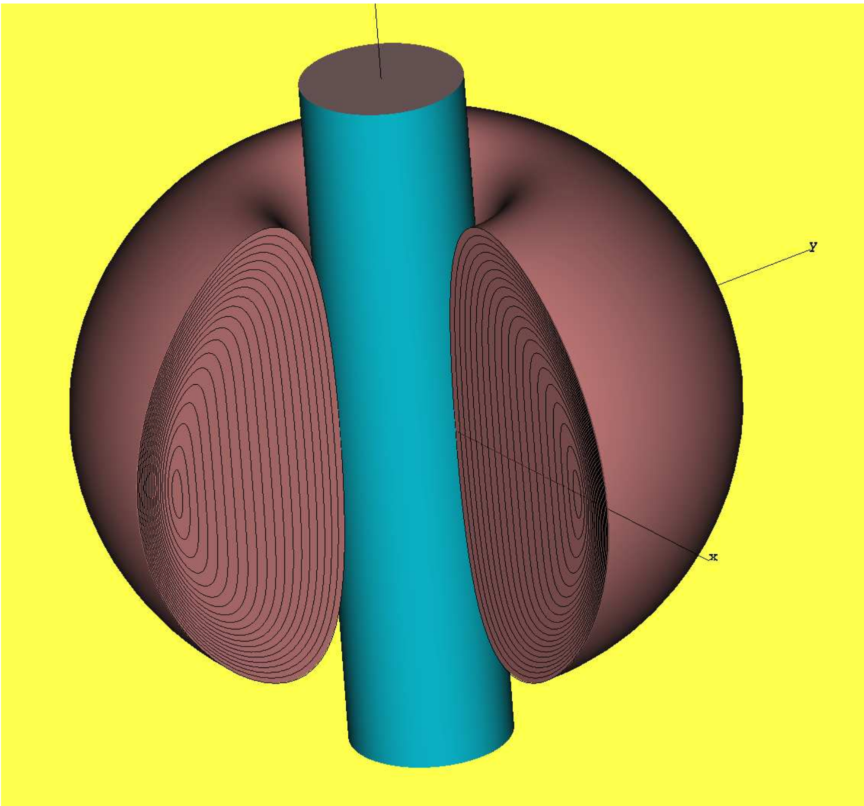
Regarding LiWall regime, Spherical Tokamaks are more similar to stellarators rather than to tokamaks:

1. *Both are suitable for low energy NBI fueling*
2. *Both are "bad" for α-particle confinement and good for SCI regime*

While STs cannot serve as a reasonable power reactor concept, the stellarators have no obvious obstacles to be a power reactor.

The LiWF strategy is consistent with both R&D and power production phases of fusion energetics

Spherical Tokamaks are the only candidate for RDF



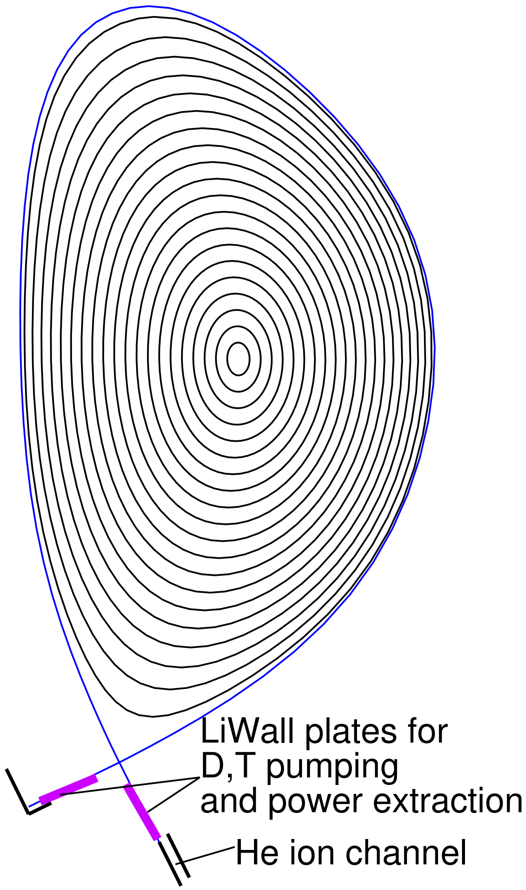
1. Volume $\simeq$30 m^3.
2. DT power $\simeq$ 0.2-0.5 GW.
3. Neutron coverage fraction of the central pole is only 10 %.
4. FW surface area 50-60 m^2

   On properties of insulation, see [1] R.H. Goulding, S.J. Zinkle, D.A. Rasmussen, and R.E. Stoller, "Transient effects of ionizing and displacive radiation on the dielectric properties of ceramics," J. Appl. Phys. 79 (6), 2920 (1996).

ITER-like device ($\simeq$ 700 m^2 surface) would have to process 700 kg of tritium for developing the First Wall.

The possibility of an unshielded copper central stack is a decisive factor in favor of IST

“Bleeding” (R. Goldston) Li target plate (belt limiter) with 0.1 mm thick Li is the concept of the pumping lithium divertor.



Replenishment of Li by gravity flow

$$\nu_{Pa\cdot sec} \simeq 5\cdot 10^{-4}$$

$$V_g = \frac{\rho g h^2}{2\nu}\sin\theta \simeq 0.05\frac{h^2}{0.01\ mm^2}\sin\theta,$$

Marangoni flow

$$\frac{d\sigma(T)}{dT} = -1.62\cdot 10^{-4}$$

$$V_M = \frac{d\sigma(T)}{dT}\frac{h\nabla T}{\nu} \simeq 8\cdot 10^{-5}\frac{h}{0.1\ mm}\nabla T$$

with Li supply controlled by capillary and wicking forces.

No rivers, water-falls of Li, evaporaton, dust, trays, or thick ($\simeq$ 1 mm) layers of Li on the target plates

Inventory of lithium for pumping purposes is not the issue

E.g., for the ITER size plasma 3-4 L of lithium (0.1 mm $\times$ 30-40 m^2) with the rate of replenishment

$$10L/hour, \quad V_{Li} < 1 \text{ [cm/sec]}$$

is sufficient.

Existing technology of capillary systems (“Red Star”, T-11M, FTU, UCSD), gravity and Marangoni effect provide a solid design basis for pumping surfaces. Everybody has his own experience with solder and copper wire.

The issue is only in the oxidation (hydrolyzation) of the Li surface during the idle period of the machine.

In LiWF molten lithium can be used to control the inventory of unburned tritium

There is very little in open literature on wetting/wicking by Li

The RDF program assumes conversion of NSTX in PPPL into ST0 with Li based PFC

- *The current NSTX program is essentially exhausted.*
- *It is focused mainly on self-improvements and is trailing the achievements of other teams, rather than advancing fusion energy.*
- *The program already has been twice explicitly warned about possible shutdown.*
- **On the other hand, the experience accumulated on NSTX, and the machine itself, are extremely valuable for developing the next steps in magnetic fusion.**

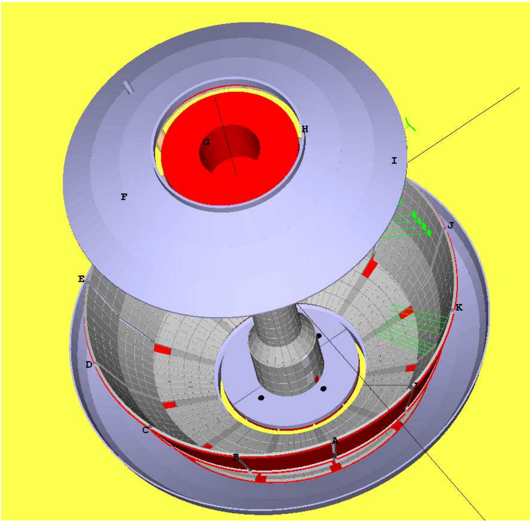
For ST0, the criterion for readiness of the machine to LiWall regime can be well-defined:

Demonstration of complete depletion of the plasma discharge by wall pumping, as on T-11M in 1998

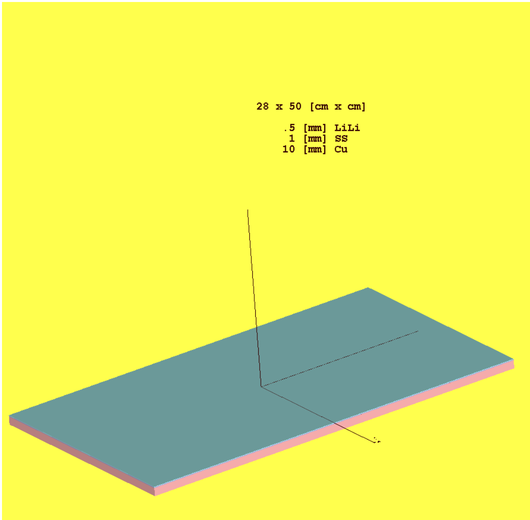
The mission of the ST0 is

To demonstrate feasibility of the LiWall regime with $\tau_E \simeq 0.1 - 0.15$ sec, ($\simeq 2 - 3\tau_{E,NSTX}$)

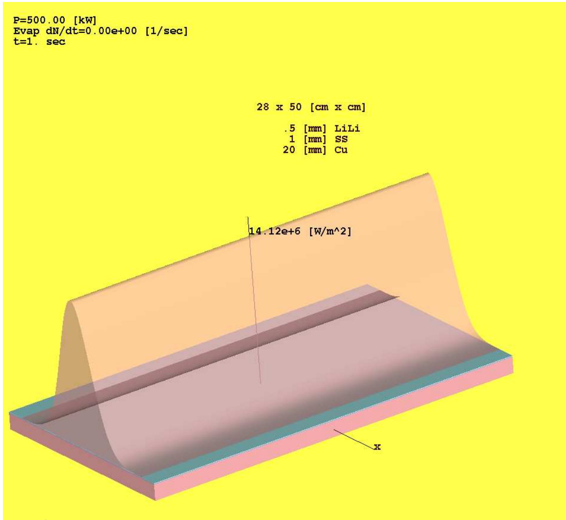
Molten Li is necessary to provide 10000 active monolayers or $\simeq 3\mu m$ of Li for pumping NSTX plasma



Li coated plate in low inner divertor



Li/SS/Cu (0.5mm/1mm/10mm) sandwich with a trenched surface

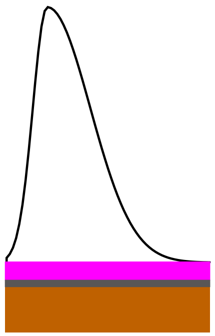


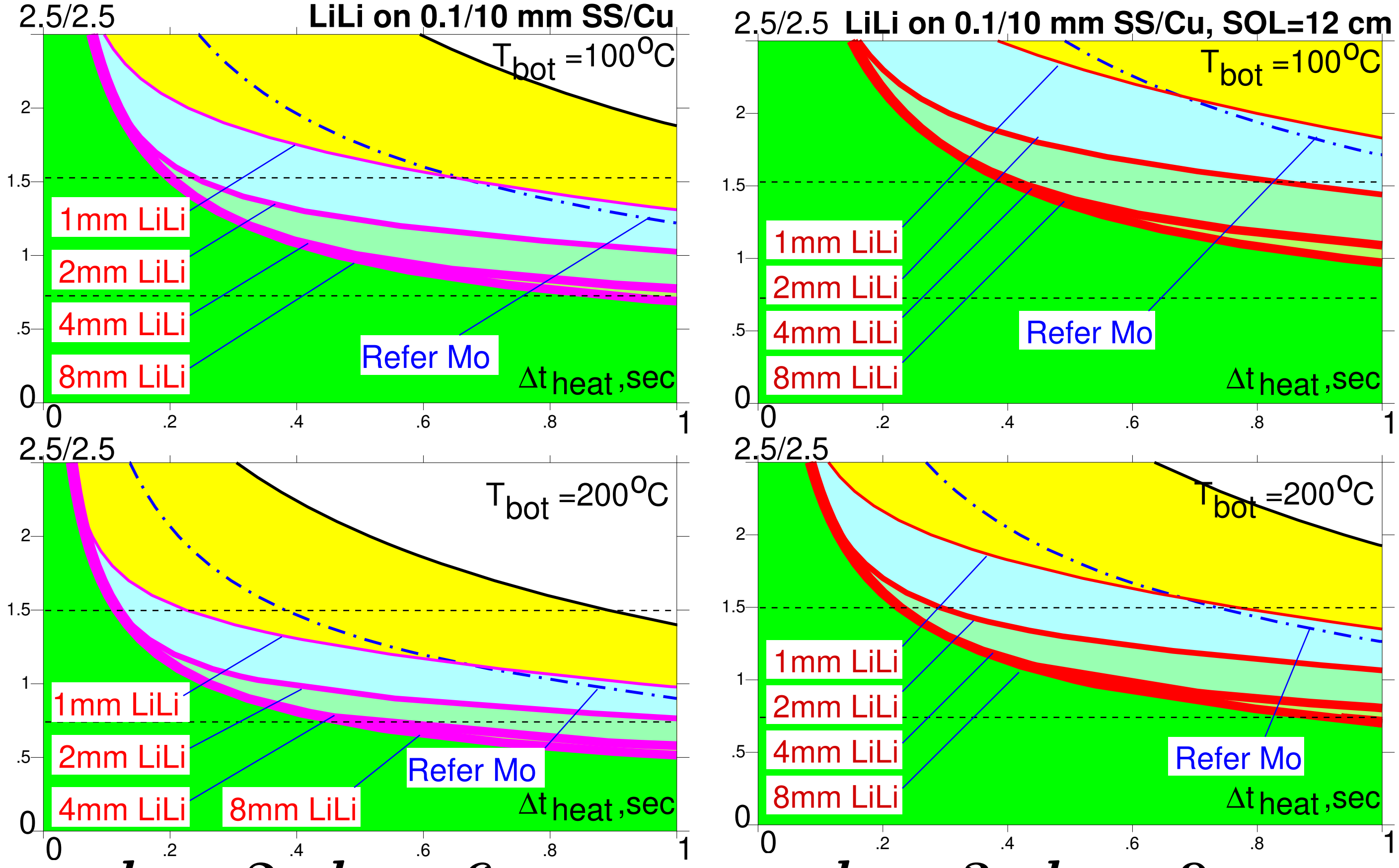
Gaussian (8 cm wide) heat deposition profile

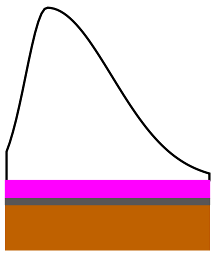
$$
\begin{aligned}
&S \simeq 0.75\ [m^2], \quad L_{SOL,m} = 2.5, \quad V_{Li} \simeq 0.35\ [L], \quad M_{Li} \simeq 175\ [g],\\
&\nu_{Pa\cdot sec} \simeq 5\cdot 10^{-4}, \quad I_{ion,MA} = \frac{(0.4-1)\cdot 10^{-3}}{1.6},\\
&V_{Li,cm/sec} = (2-5)\cdot B_{tor}\frac{h^2_{Li,mm}}{0.01}\frac{0.1}{w_{SOL}}\frac{I_{SoL,MA}}{I_{ion}}
\end{aligned}
\qquad (6.5)
$$

Li/SS/Cu plate could be the real first step toward Li PFC and LiW regime

The plate 0.1-1 mm of Li on 0.1/10 SS/Cu provides the operational space for the LiWall regime







$d_i = 2, d_e = 6$ cm $\quad$ $d_i = 3, d_e = 9$ cm

Within 1-2 campaigns, experiments with plate could provide the data for ST0