LiWF & Spherical Tokamaks (part 2)

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Contents

1 The edge plasma temperature

The edge temperature pedestal and H-mode were discovered on Asdex in the early 80s

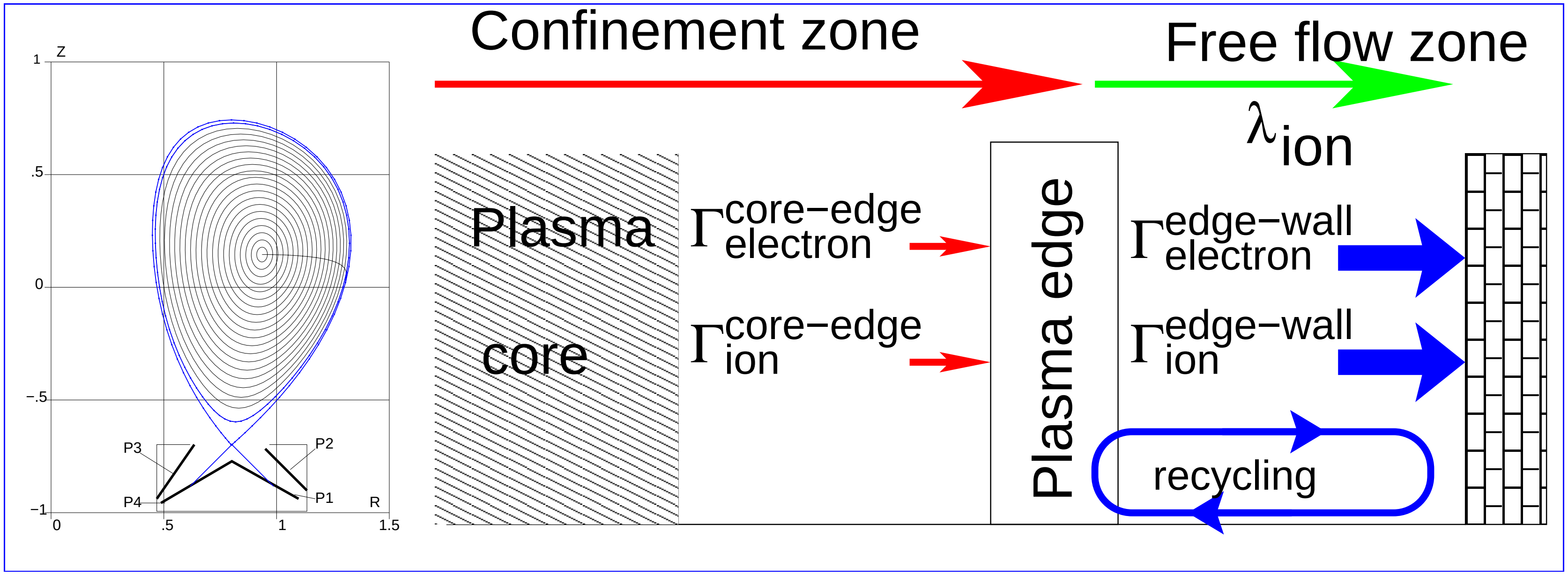
Also, the "edge transport barrier", which provides a steep temperature gradient in front of the last closed magnetic surface, was introduced.

In the LiWF regime the temperature pedestal is equal to core temperature. At the same time, the understanding of the LiWF regime gives a new view on the temperature pedestal in conventional plasma.

Apparently obvious, the concept of the "edge transport barrier" contains many hidden inconsistencies

Where is the plasma edge ?

Is it not just the separatrix by definition ?



The plasma edge, understood as a transition zone from diffusive transport to a convective one, is located approximately at one mean free path

$$\lambda_{\|,D,m} = 121\frac{T_{keV}^2}{n_{20}} \tag{1.1}$$

from the plasma facing surface. For $T_{edge} > 1$ keV the mean free path $\lambda_{\|,D,m}$ can be as large as $\simeq 1$ km or more.

T_{edge} is a boundary condition

Edge plasma temperature is determined self-consistently by the particle fluxes (Krasheninnikov)

Across the last mean free path, λ_D, in front of PFC surface the energy is carried out by the moving particles

$$\frac{5}{2}\Gamma_{e,i}^{edge-wall}T_{e,i}^{edge} = \int_V P_{e,i}dV, \qquad \Gamma_{e,i}^{edge-wall} = \frac{\Gamma^{core-edge}}{1-R_{e,i}} \tag{1.2}$$

T_{edge} serves as a boundary condition for the confinement zone

$$T_e^{edge} = \frac{2}{5}\frac{1-R_e}{\Gamma^{core-edge}}\int_V P_e dV, \qquad T_i^{edge} = \frac{2}{5}\frac{1-R_i}{\Gamma^{core-edge}}\int_V P_i dV \tag{1.3}$$

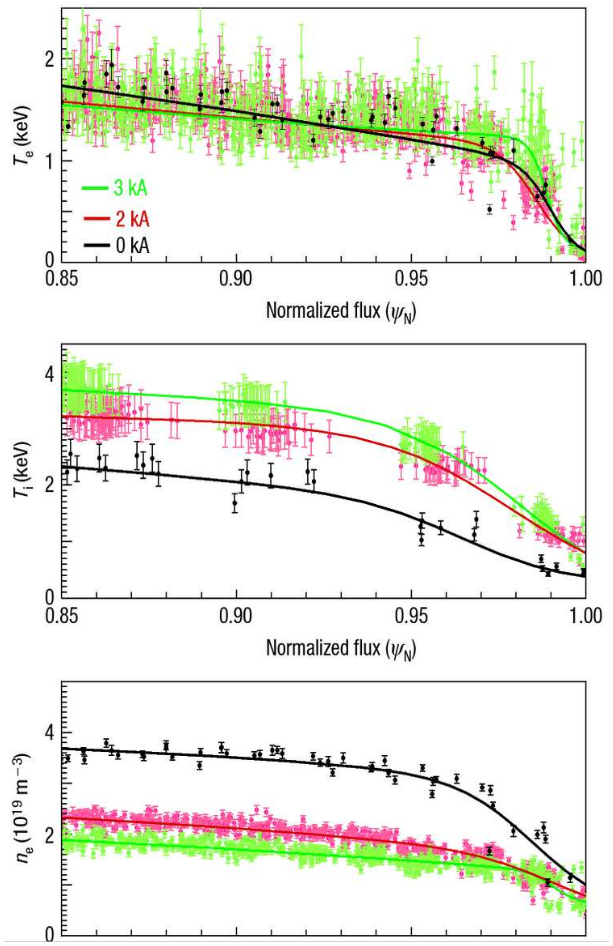
In the Lithium Wall Fusion (LiWF)

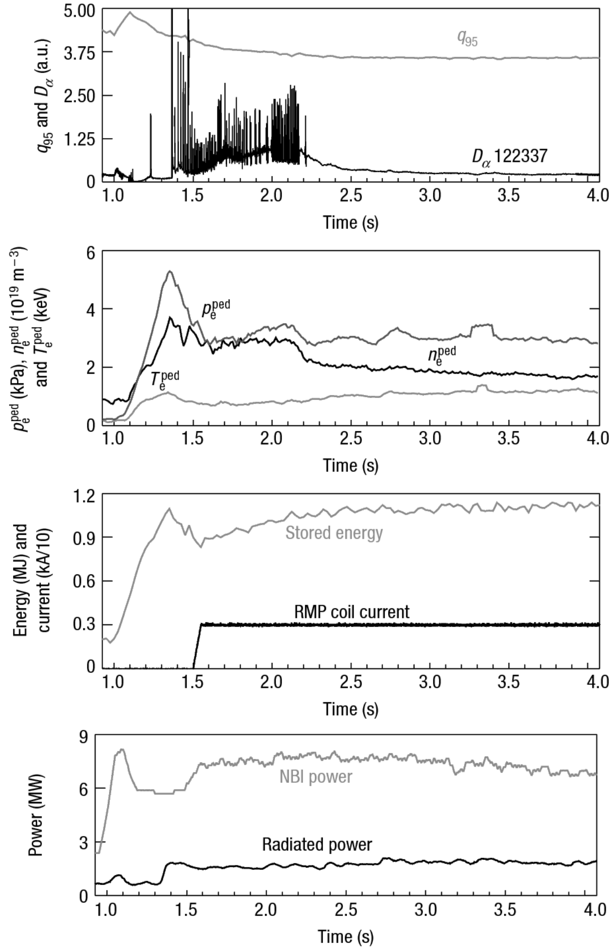
$$\Gamma_{electron,ion}^{edge-wall} \simeq \Gamma^{core-edge}, \quad \to T_{edge} \simeq T_{core}$$

The transport plasma properties near the edge do not affect T_{edge}

DIII-D made crucial input to LiWF

Resonance Magnetic Perturbation experiments have confirmed our, LiWF, views. The pedestal T_{edge} in not affected by RMP.





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

There is no confinement in the "edge transport barrier" zone

RMP interpretation

The toroidal plasma has 3 different plasma edges: two for electron and ion temperatures, and a separate for the plasma density

The edge for the electron temperature is situated at the tip of the temperature pedestal.

For the ion temperature and the plasma density, the edge seems to be at the separatrix.

In the zone of the electron temperature pedestal the confinement is essentially absent. Instead of mysterious "transport barrier" properties, the $T_e(x)$ profile is determined by recycling (Simple Recycling Model)

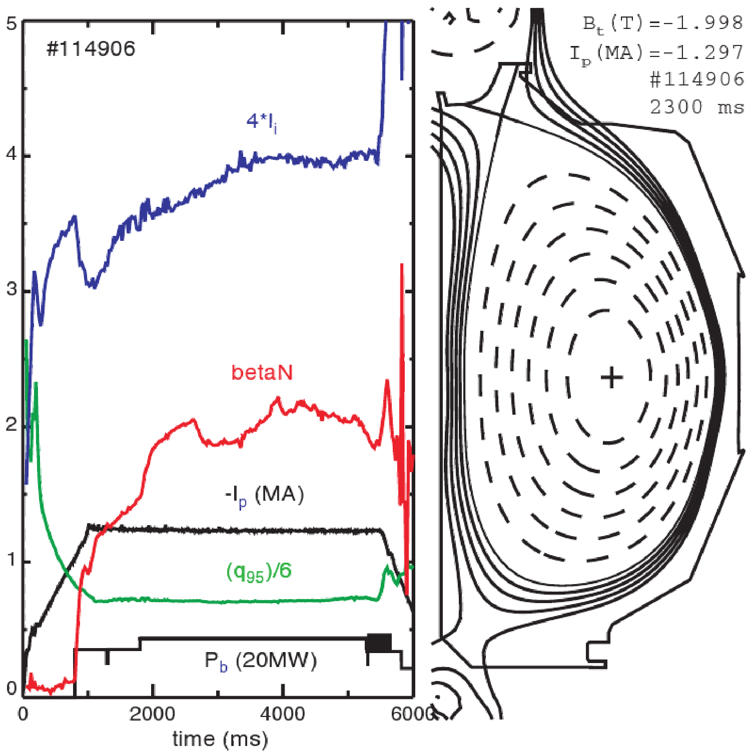
$$T_e^{edge}(x) = \frac{2}{5}\frac{1 - R_e(x)}{\Gamma^{core-edge}} \int_V P_e dV, \qquad R_e(x) = 1 - R_{edge}\frac{x}{x_{edge}}, \tag{1.4}$$

where $x = 0$ is at the separatrix.

Scrape Off Layer Currents

SOLCs are present even in the most quiet plasma

SOL current and MHD activity in DIII-D



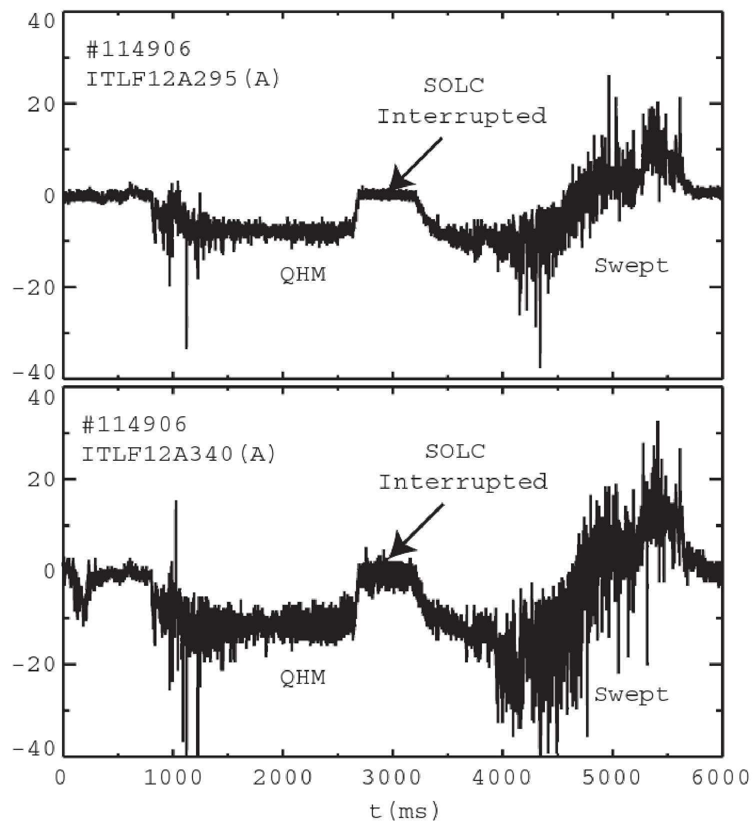


Figure 3. Pictorial discharge summary; the left-hand panel shows I_p in units of megaamperes, P_b in units of 20 MW, q_{95} divided by 6, β_N, and the nominal no-wall limit (here, 4 li). The right-hand panel sho
SOL
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c

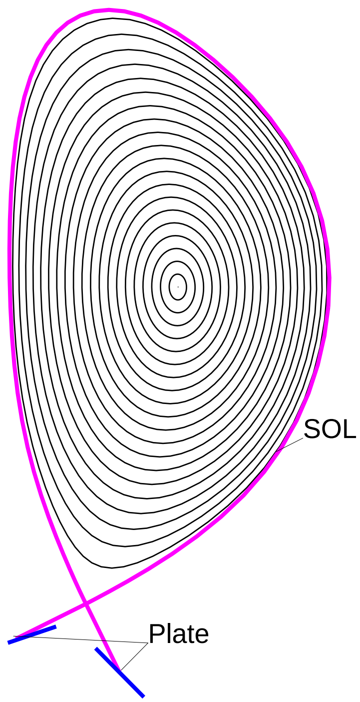
Figure 4. Signals from tile current sensors in tile ring #12 A in the

Todd Evans, Hiro Takahashi and Eric Fredrickson (NF,2004) have found a link between SOLC and MHD activity on DIII-D

Probably SOLC determine the width of the temperature pedestal

2 Sheath potential

Collisionless Scrape Off Layer introduces new physics



Conventional estimate of sheath potential

$$\varphi_E \simeq 3T_e \qquad (2.1)$$

is not applicable. The mirror ratio along field lines in the SOL and confinement of trapped particles in SOL determine the sheath potential

$$\varphi_E \simeq T_e. \qquad (2.2)$$

A blanket of trapped particles is expected between the SOL and wall

Lithium PFC satisfies, at the very least, the condition of low recycling, $R_i \ll 1$

The importance of the secondary electron emission is not yet known

The scales

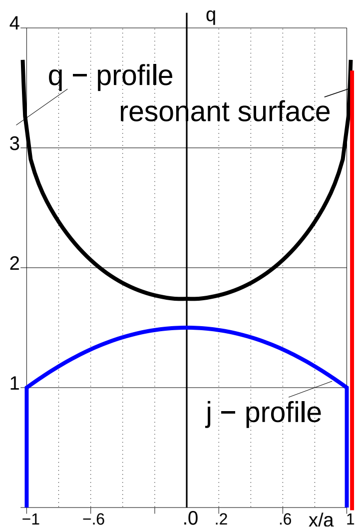
$$\rho_e^{se} = \frac{4.76}{B_T} \ll \rho_e^{SOL} = 238\frac{\sqrt{T_{e,10keV}}}{B_T} \ll \rho_D = 14100\frac{\sqrt{T_{i,10keV}}}{B_T} \ [\mu\mathrm{m}] \qquad (2.3)$$

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

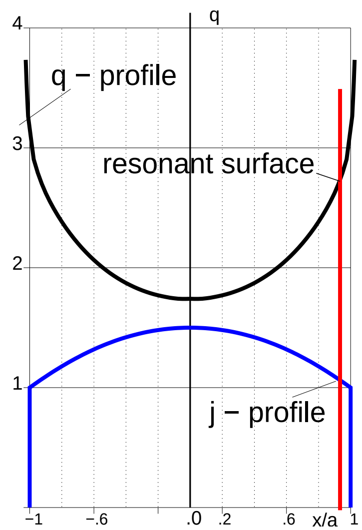
3 No ELMs, blobs in LiWF regime

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

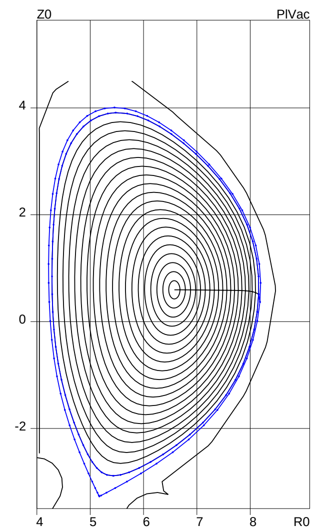
$$W \propto \int \frac{j' R \psi^2 d\rho}{B_{tor}\left(\frac{1}{q}-\frac{n}{m}\right)} \simeq \frac{j_{edge}}{B_{tor}\left(\frac{1}{q_{edge}}-\frac{n}{m}\right)}\psi^2$$



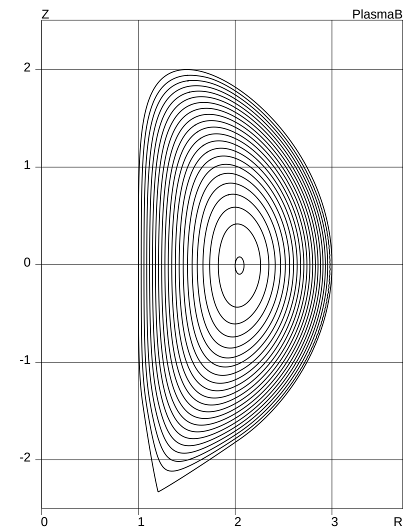
case 1: $mq_a < n$
Ideally unstable



case 2: $mq_a > n$
Tearing stable



Ideally & tearing stable

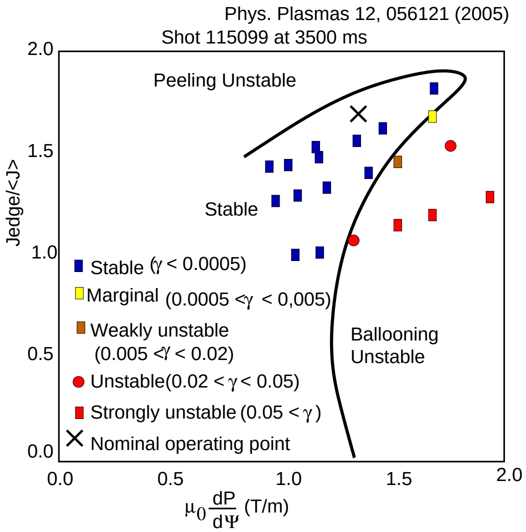


$\mathrm{j/B}$ =const equilibrium, $j_{edge} \neq 0$

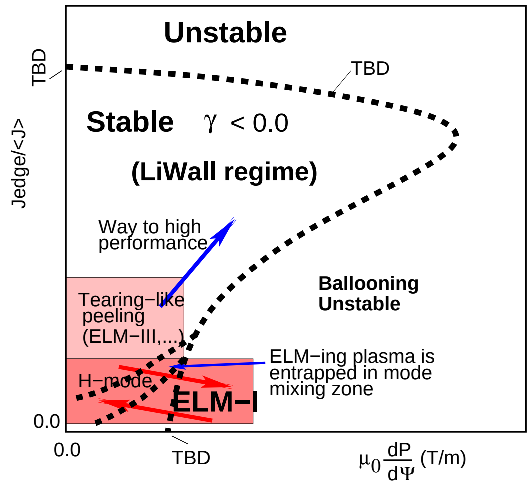
In presence of a separatrix, the finite edge current density is stabilizing as well as the low edge density. No ELMs, blobs.

KINX code Stability Diagram

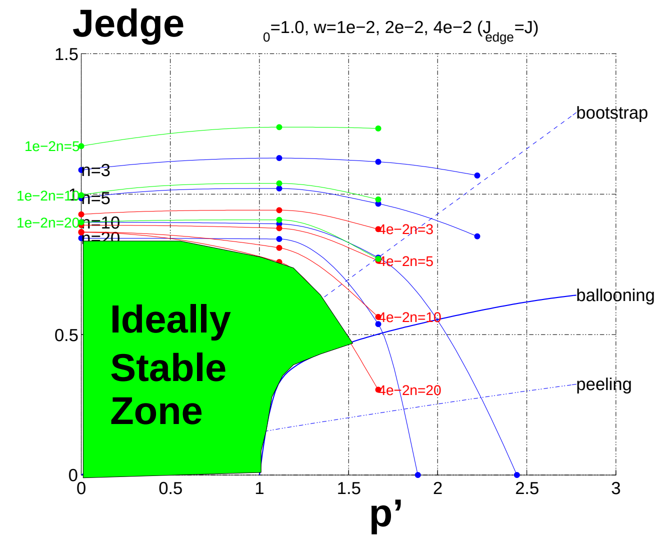
Peeling-ballooning diagram of Phyl Snyder initiated theory of ELMs



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 $p = 0, j_{egde} \neq 0$

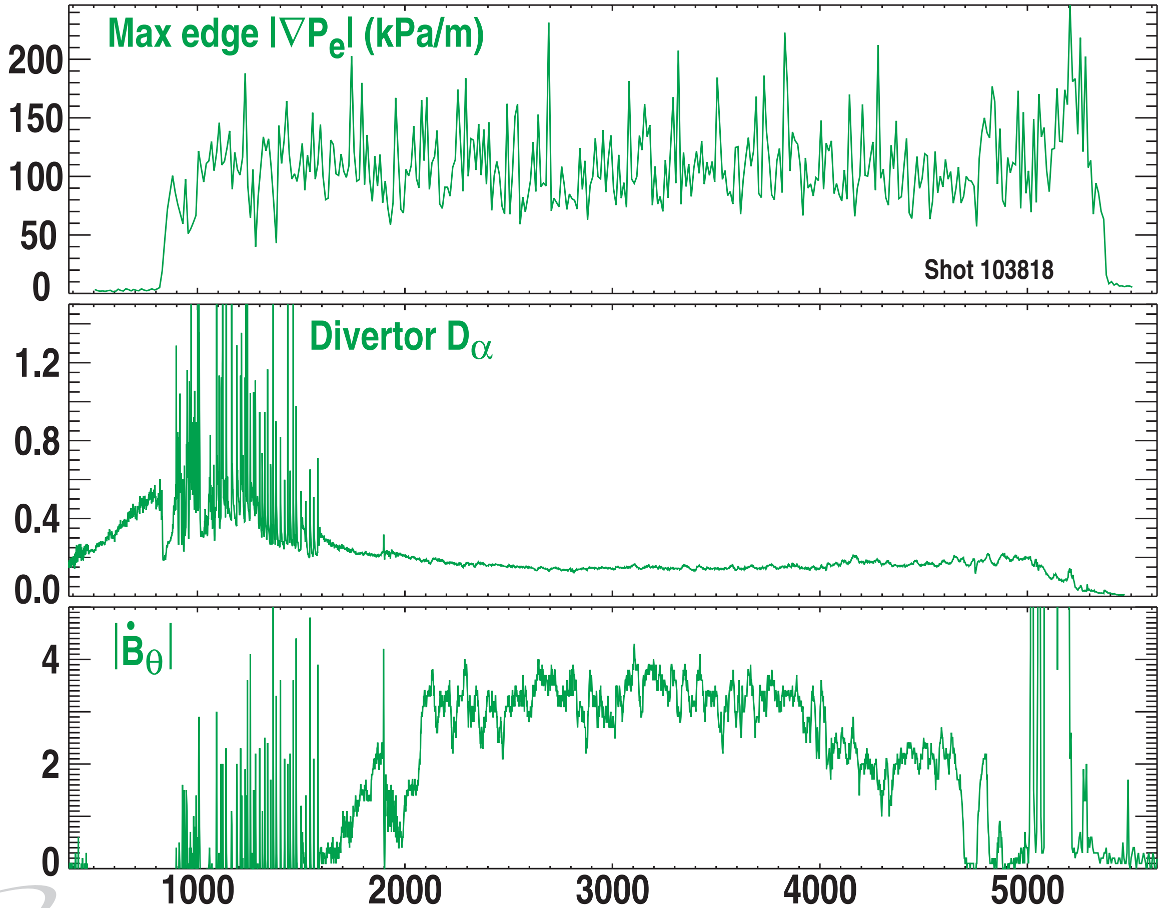
$$W = \oint \oint \psi(l) i_{ll'} \psi^*(l') dl dl' - \frac{\bar{\jmath}_\varphi}{B_\varphi} \oint \frac{\psi^* u' + \psi u'^*}{2} dl, \quad \psi \equiv -\frac{B_p r}{B_\varphi} u' -$$

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

DIII-D reported the QHM regime in 2000

Taken from "Quiescent Double Barrier H-mode Plasmas in the DIII-D Tokamak" by K.H.Burrell, APS-2000, Quebec City, Canada

EDGE ∇P_e DOES NOT CHANGE WHEN ELMS DISAPPEAR

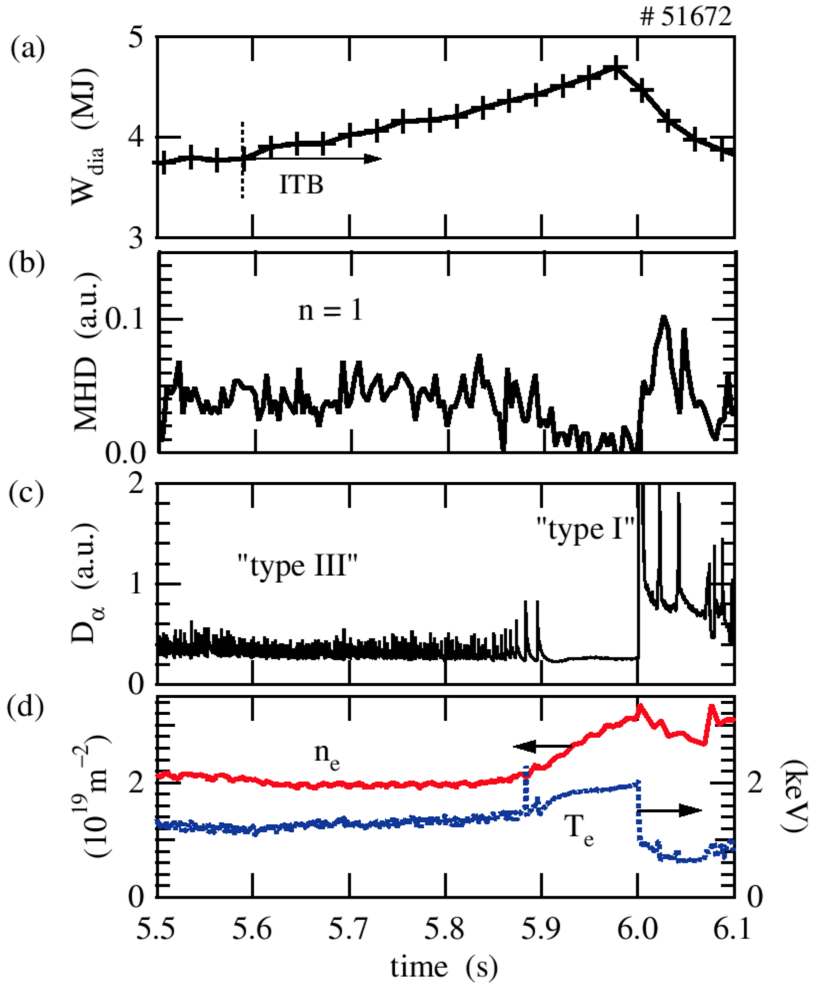


255-00/rs

DIII-D NATIONAL FUSION FACILITY SAN DIEGO

JET exhibited ELM free periods

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

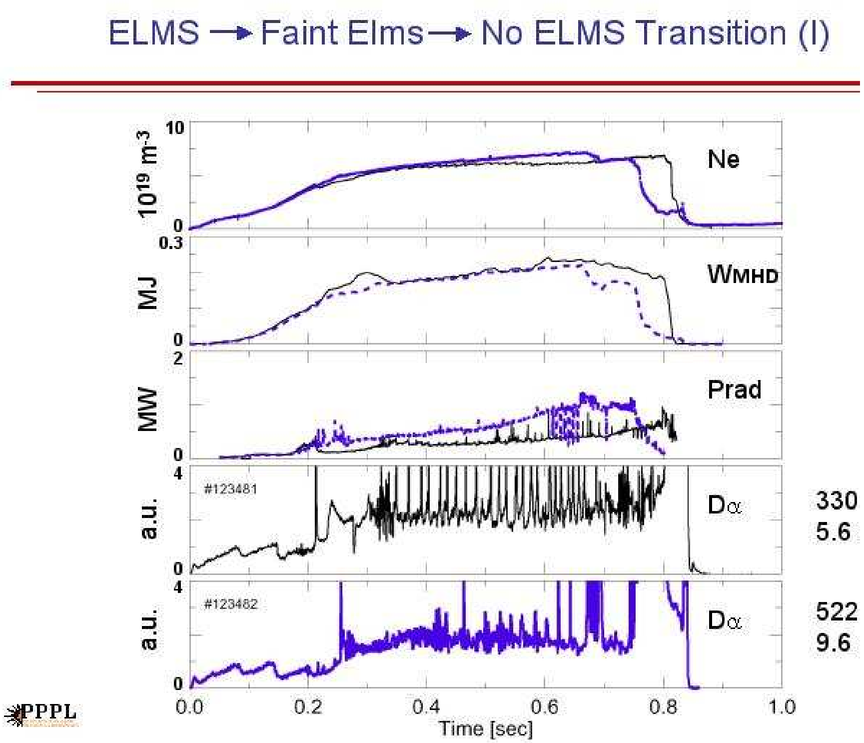
"Edge issues in ITB plasmas in JET"

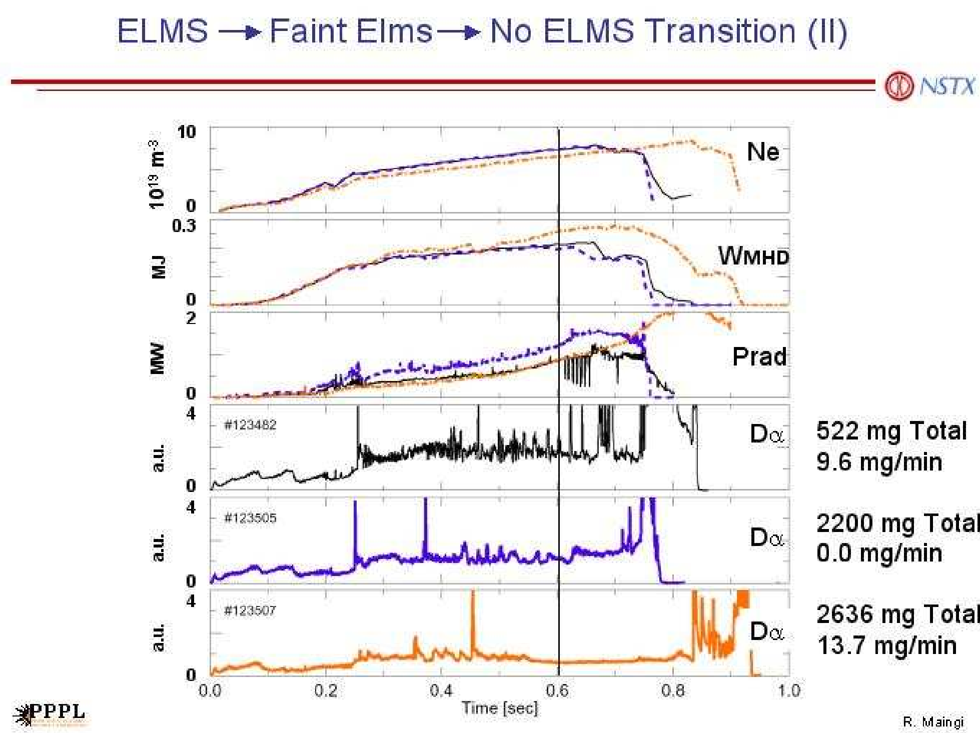
Plasma Phys. Control. Fusion 44 (2002) 2445-2469 Y. Sarazin, M. Becoulet, P. Beyer, X. Garbet, Ph. Ghendrih, T. C. Hender, E. Joffrin, X. Litaudon, P. J. Lomas, G. F. Matthews, V. Parail, G. Saibene and R. Sartori.

The authors emphasized the crucial role of the edge current density

Li on NSTX eliminated ELMs

ELMs were suppressed after Li conditioning on NSTX



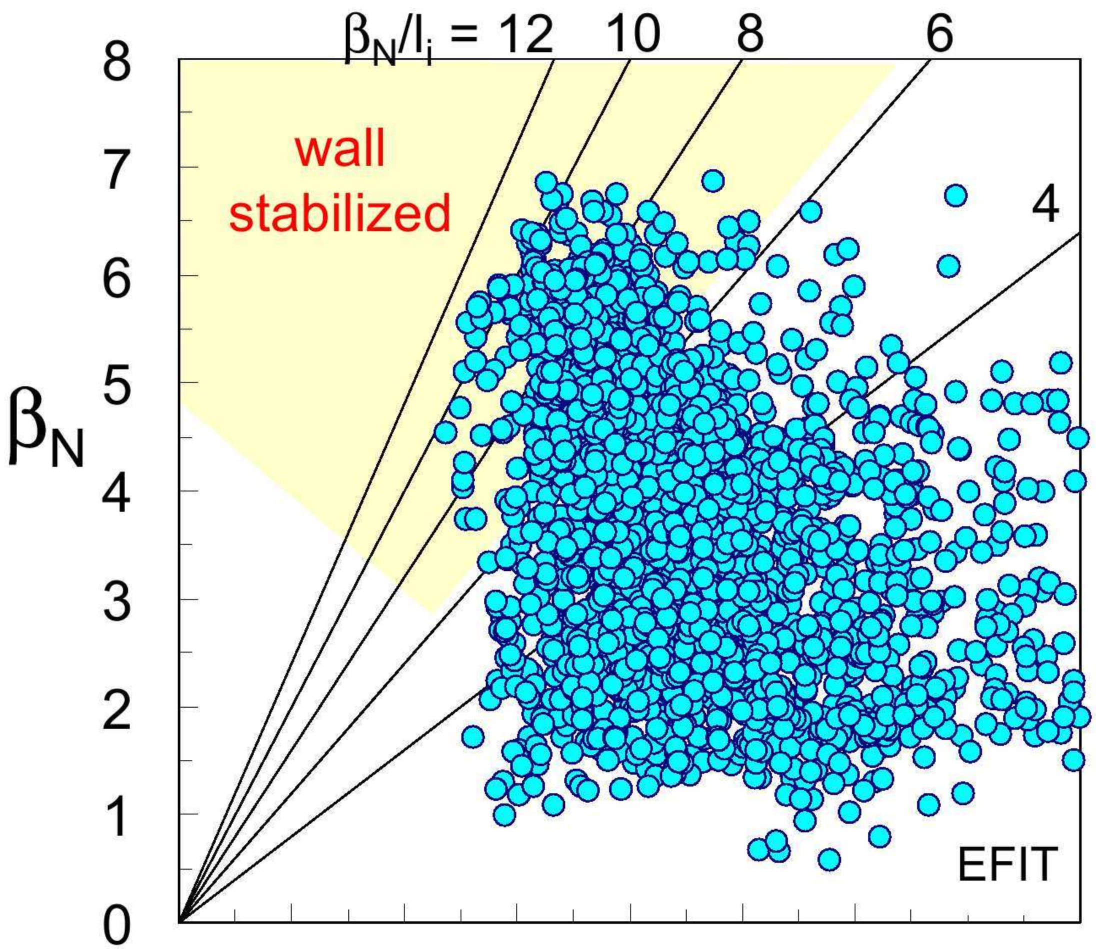


Four shots are shown (D.Mansfield): before Li evaporation, after depositing $\simeq$200 mg, then +1700 mg, and +400 mg.

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

4 Global stability

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



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

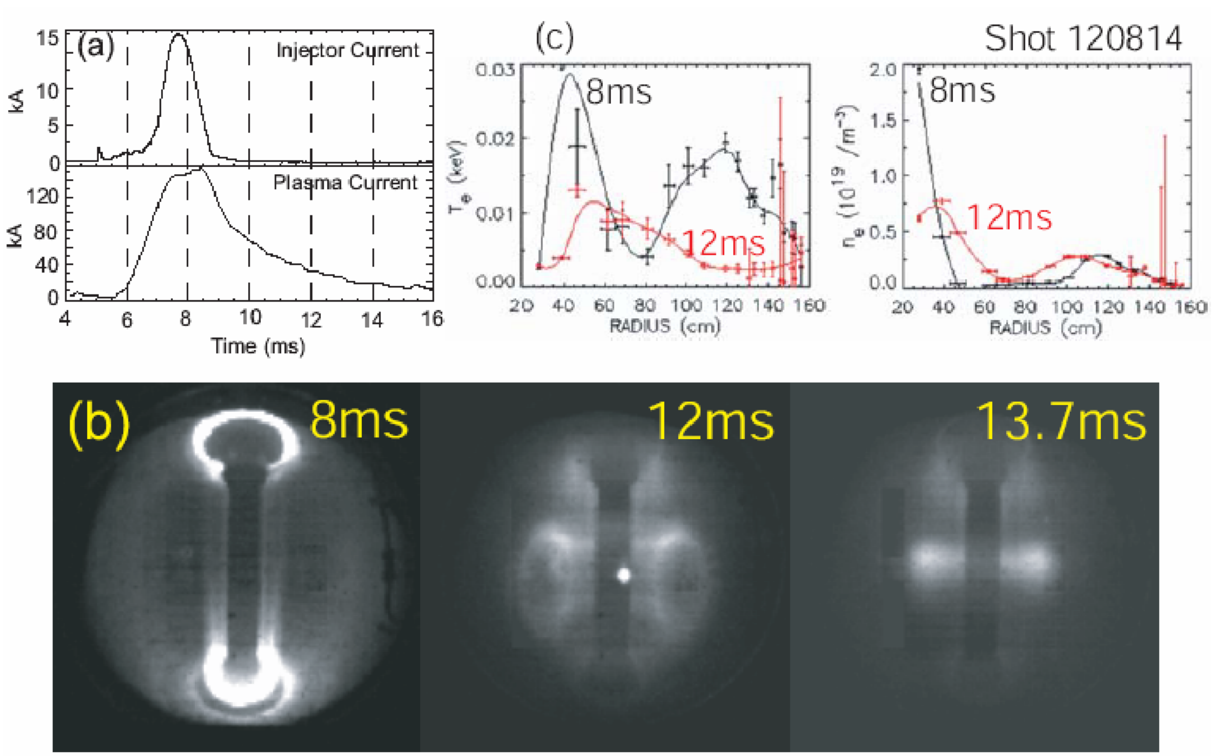
Stability with respect to global ideal kink modes of LiWF plasma is not different from the conventional plasma.

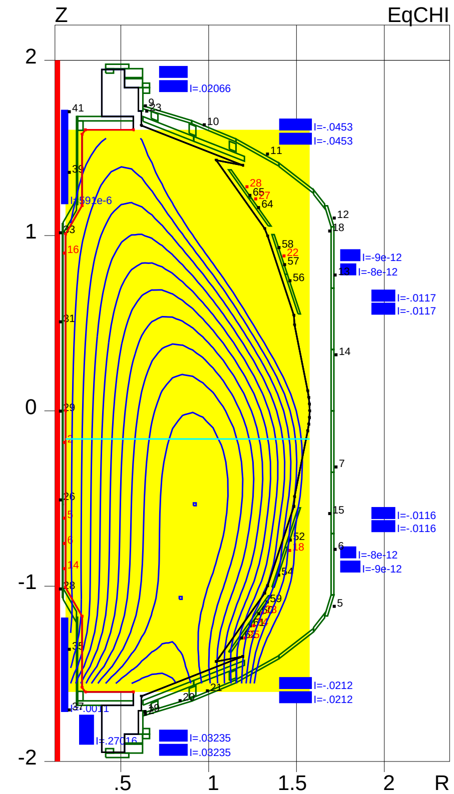
No Greenwald limit in LiWF

LiWF regime eliminates q=1. No sawteeth, no internal reconnection events. In all aspects stability is better (or the same).

5 Non-inductive startup. Li & CHI

LiWF is compatible with both inductive and CHI start-up



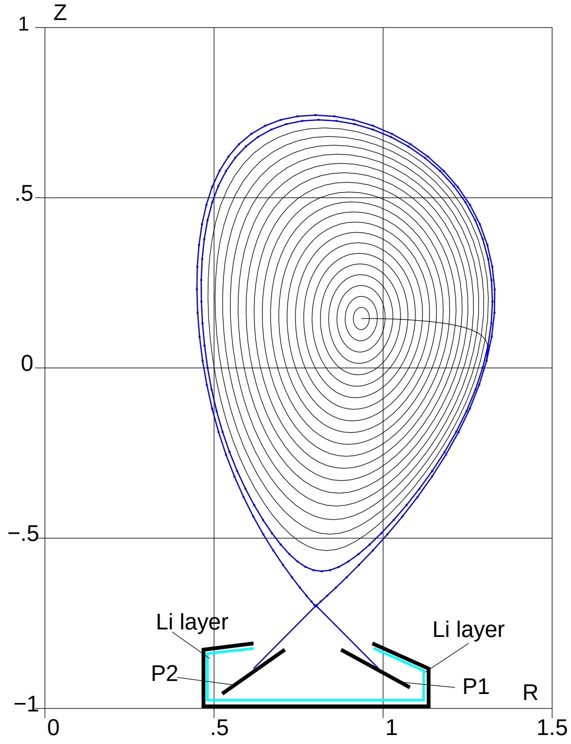


In 2006 CHI startup generated 160 kA current in NSTX *From R.Raman at al., PPPL-4207 (2007)*

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)

6 LiWF and stationary plasma

LiWF suggests the self-consistent approach to the stationary plasma



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

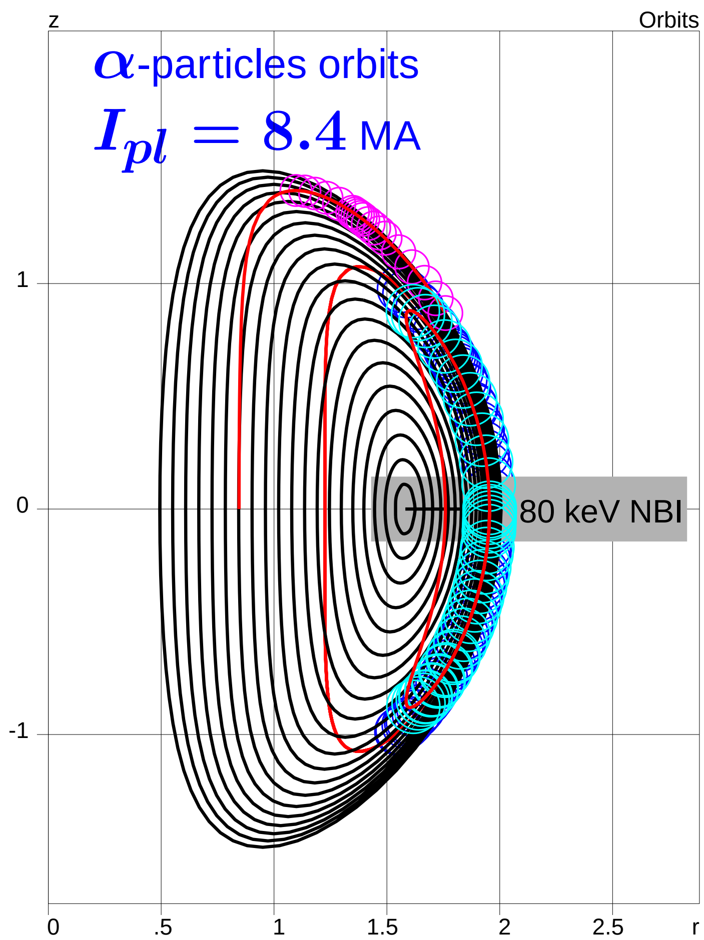
1. **A small electro-static force ZeE_{SOL}, directed back to the plate.**
2. **Friction $R_V \propto Z^2$ with the ion flow, also directed back to the plate.**
3. **Thermo-force $R_T \propto Z^2$, driving impurities into the plasma.**

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

At high T_{edge} the thermo-force is absent in the SOL, leading to $Z_{eff} \simeq 1$

Interaction with side walls is not expected (blobs are absent)

7 Alphas are not confined in ST



Large Shafranov shift in STs makes core fueling possible

The charge-exchange penetration length at $E = 80$ *keV*

$$\lambda_{cx} \simeq \frac{0.6}{n_{e,20}} \, [m]$$

The distance between magnetic axis and the plasma surface in projected RDF

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

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

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

8 Burn-up of tritium

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

$$f_{TB} = n \langle \sigma v \rangle_{DT,16keV} \bar{\tau}_E = 0.03 n_{20} \bar{\tau}_E \quad (8.1)$$

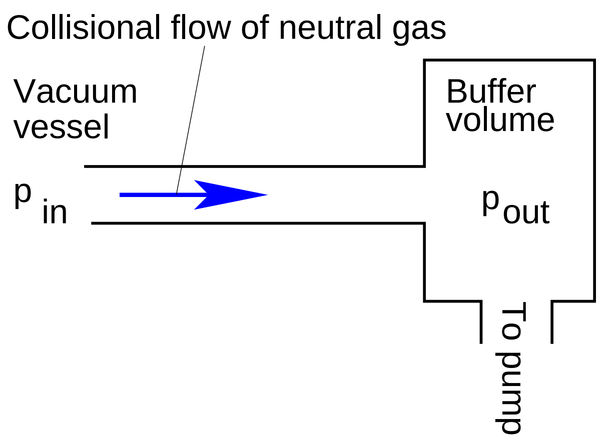
With $\tau_E \simeq$10 sec in the LiWF regime, the burn-up of tritium could be a significant fraction of unity ($f_{TB} \simeq$0.3)

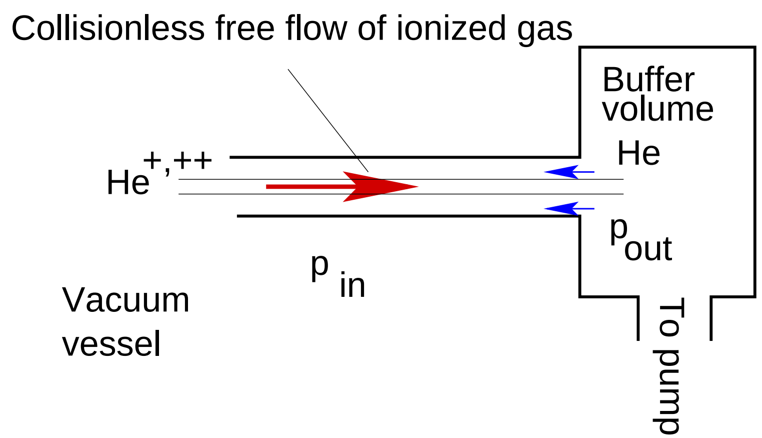
On the other hand, due to reliance on ignition criterion $nT\tau_E^* \simeq$ const,

With $\tau_E^* \simeq$1 sec, BBBL70 is locked into very low, $f_{TB} \simeq$0.02-0.03 rate of tritium burn-up

9 Helium pumping

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

LiWF scheme:

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

b) Back flow is limited by

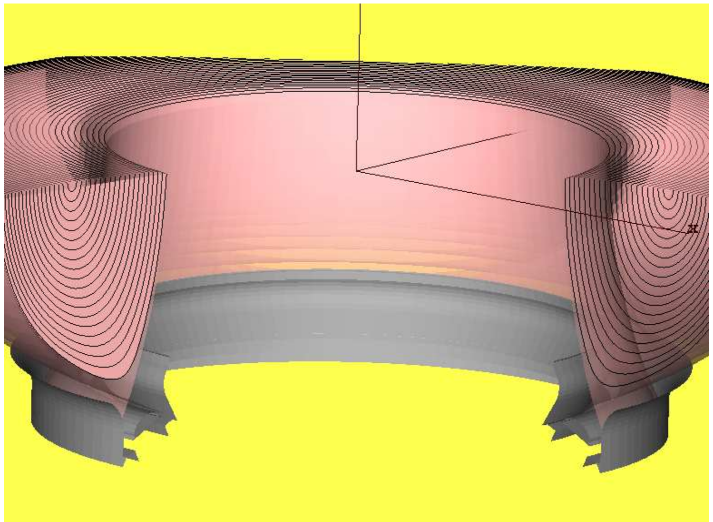
$$\Gamma_{He} = Dn'_x, \quad D = hV_{thermal}$$

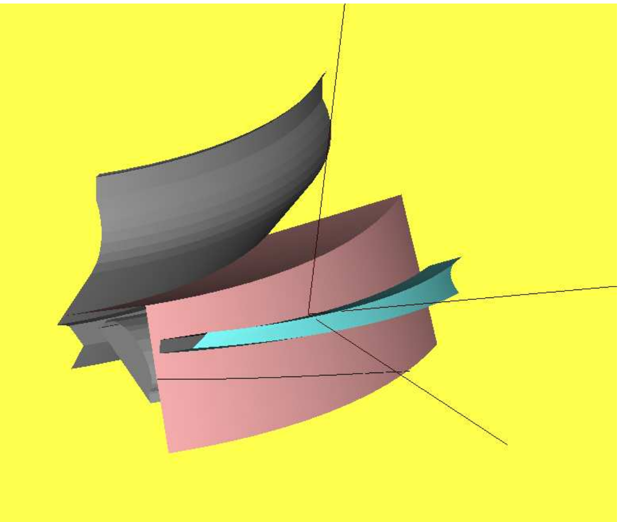
c) Helium density in the vessel plays no role, while D *is in the hands of engineers.*

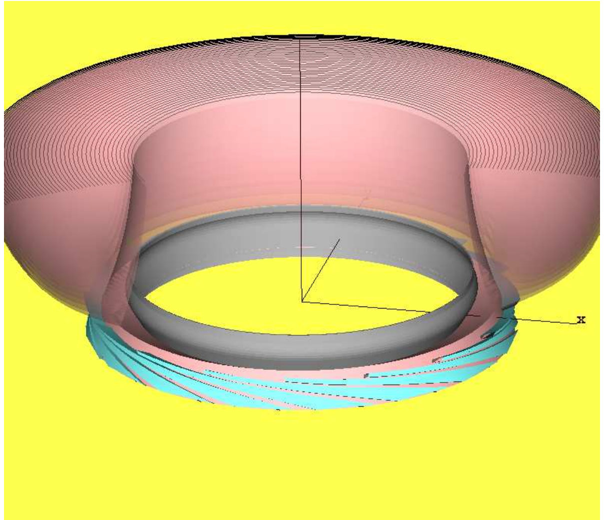
The second scheme is appropriate for the low recycling regime

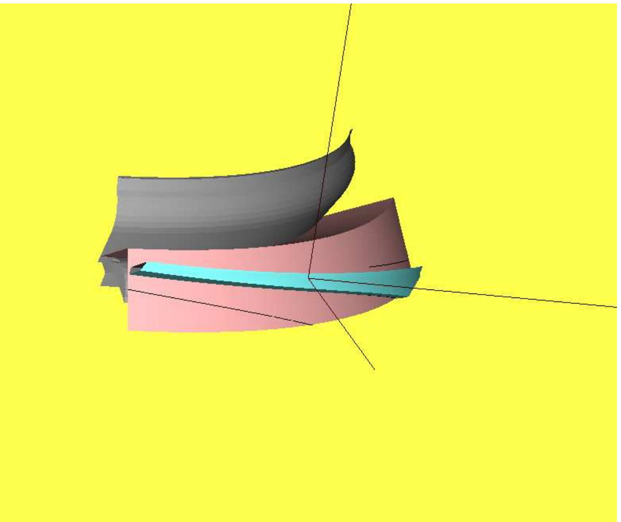
Compact “honeycomb” membrane

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





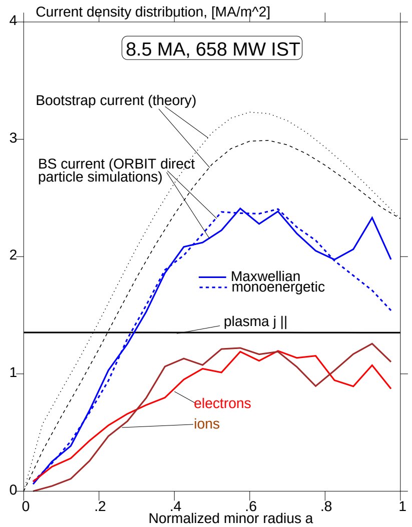


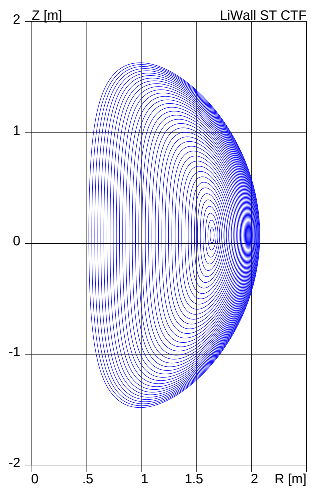


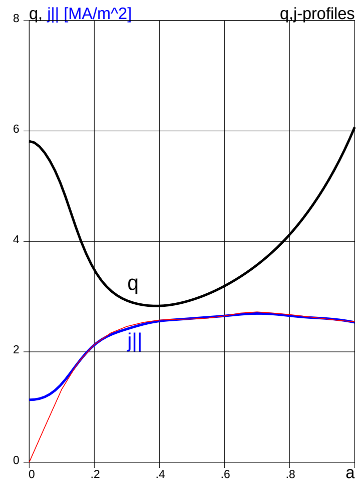
The blanket of trapped particles outside SOL helps to pump He

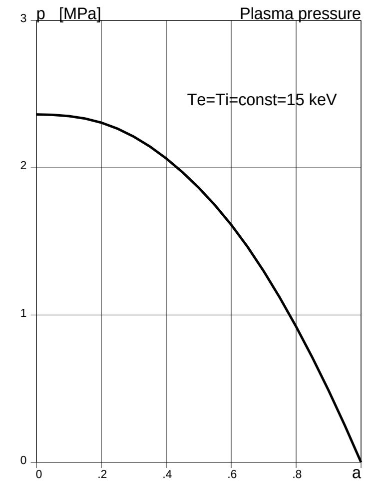
10 Bootstrap current

Bootstrap current is required for a stationary regime









j_{BS} profile cross-section q- and j-profiles pressure profile

Ballooning stable high-beta configuration with a self-consistent bootstrap current

According to theory,

In the LiWall regime ST can be "over-driven" with bootstrap current

11 LiWF and DD fusion

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

Fusion reactions

$$
\begin{aligned}
D+D \quad & \underset{50/50\%}{\Longrightarrow} \begin{cases} T_{1.01\ MeV} & +p_{3.02\ MeV} \\ He^3_{0.82\ MeV} & +n_{2.45\ MeV} \end{cases}, \\
D+He^3 & \Longrightarrow \quad He^4_{3.6\ MeV}+p_{14.7\ MeV}, \\
D+T \quad & \Longrightarrow \quad He^4_{3.5\ MeV}+n_{14.1\ MeV}
\end{aligned}
\tag{11.1}
$$

Ion Larmor radii of charged products

$$
\begin{aligned}
& \rho_{T,cm}=\frac{10}{B_T}\sqrt{3}, \quad \rho_{p,cm}=\frac{10}{B_T}\sqrt{\{3,14.7\}}, \quad \rho_{\alpha,cm}=\frac{10}{B_T}\sqrt{3.5}, \\
& \rho_{He^3,cm}=\frac{10}{B_T}\sqrt{1.23} \qquad \text{– can be confined}
\end{aligned}
\tag{11.2}
$$

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

$$
\rho_{T,cm} \simeq \rho_{p,cm} \simeq \rho_{\alpha,cm} \tag{11.3}
$$

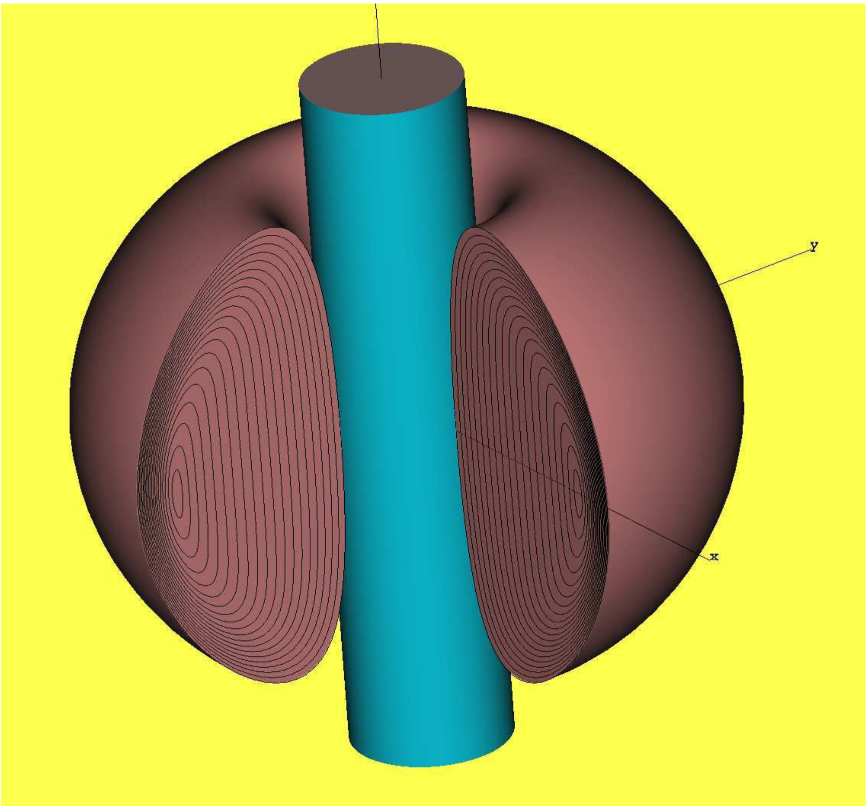
and can be expelled on the first orbits.

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

Unfortunately the cyclotron radiation makes the scheme unrealistic

12 Spherical Tokamaks and RDF

STs together with the LiWF regime 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

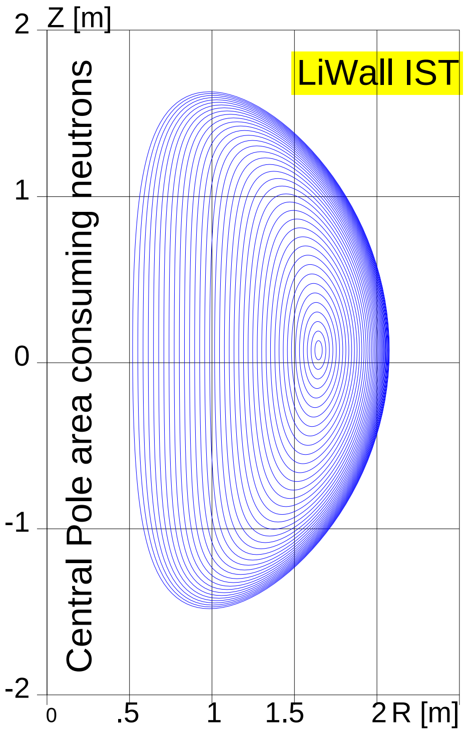
700 kg of tritium for developing

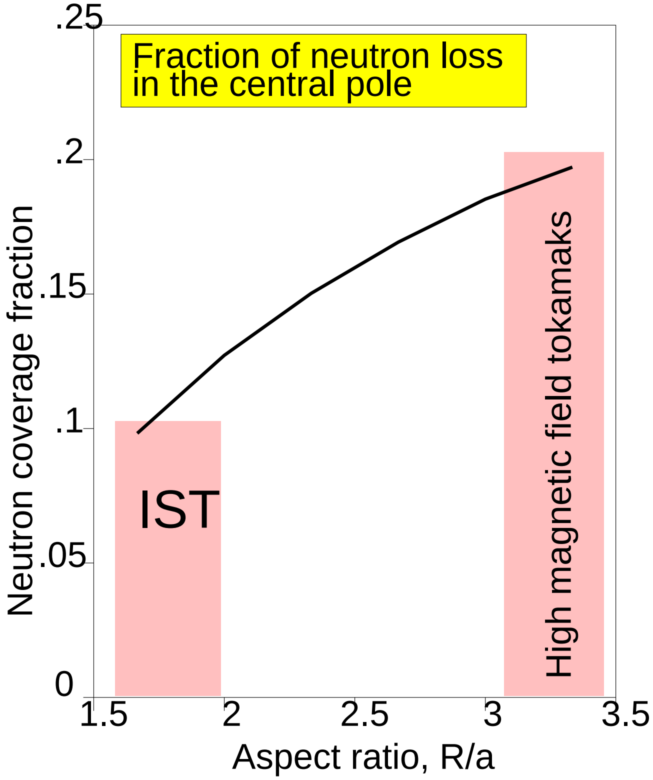
the First Wall.

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

Neutron coverage fraction

Spherical Tokamaks are suitable for the mission of RDF





1. High magnetic fields are not the option for reactor development (unfavorable geometry for neutrons, no data on stability limits, etc.)

2. Philosophy of an externally driven "Component Test Facility" based on conventional regime does not work.

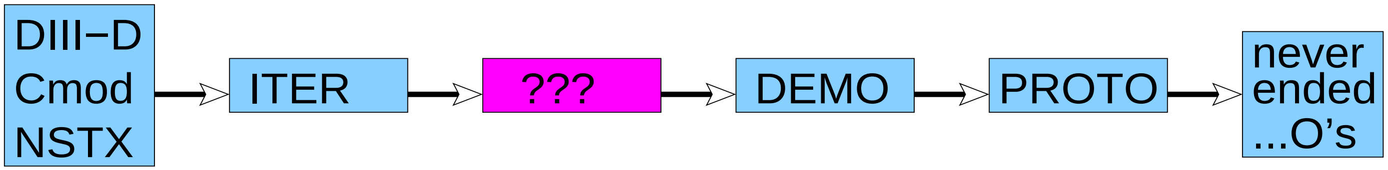
3. There is no plasma physics reasons NOT TO ignite the high-beta device. In this regard, the LiWF suggests different options.

In ST large area can be used for tritium breeding and designing the FW

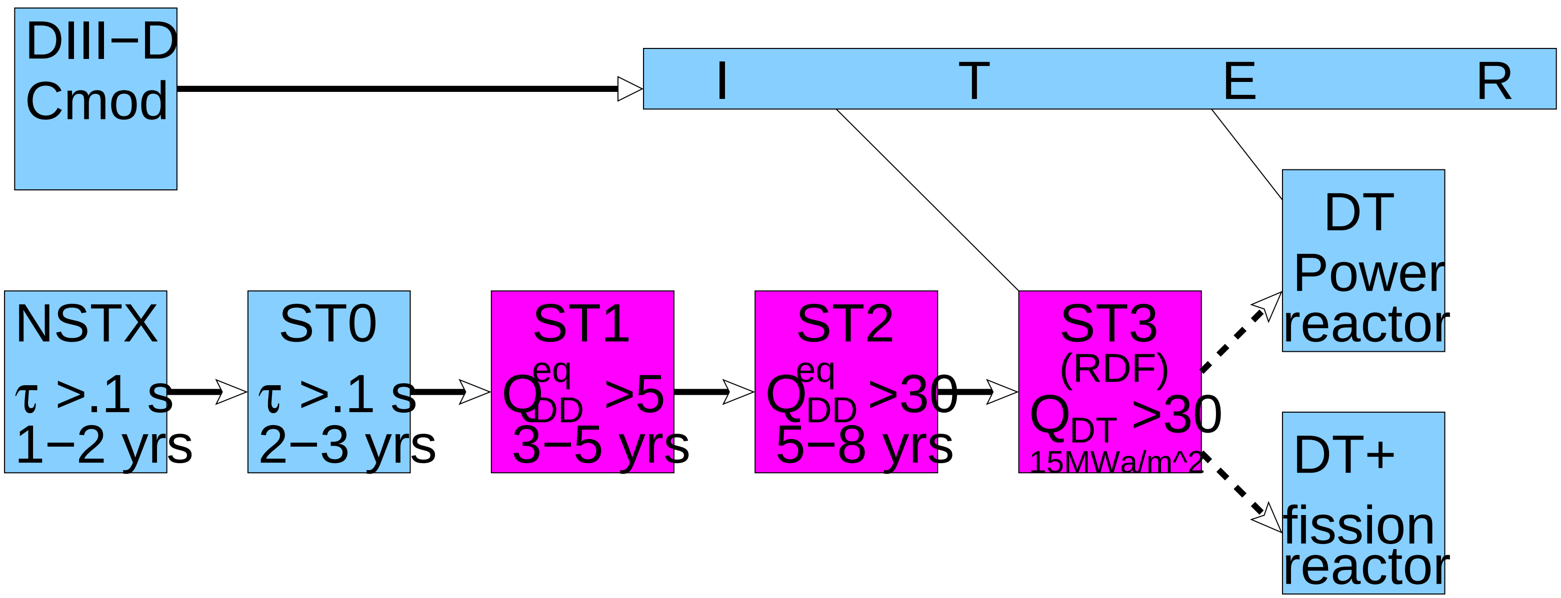
13 The LiWF path toward a reactor

The BBBL70 endless path is unacceptable for the society

According to old teaching, at least, next two generations will not see the fusion power



The LiWF concept stratifies the path to the power reactor



No “Demos”, only useful devices.

No needs in P_{DT} for R&D of LiWF

The LiWF plasma regime of either RDF or power reactor can be developed without assistance of fusion power (even in the Princeton area).

The phase of “burning plasma” (as it is introduced presently) is not necessary.

Tritium can be introduced just at the last stage of development before the real operation.

LiWF vs BBBL70

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 $\rightarrow i \rightarrow e$	to heat first useless electrons, then ions: $\alpha \rightarrow e \rightarrow i$
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	eliminates 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: $\beta_{DT} < 0.5\beta$

Currently adopted BBBL70 concept has little in common with controlled fusion and its power reactors

LiWF vs BBBL70 in plasma issues

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	plays no role	is in unbreakable 40 year old marriage with anomalous electrons
Transport database	easyly scalable by RTM (Reference Transp. Model)	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 plasma	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

14 Summary.

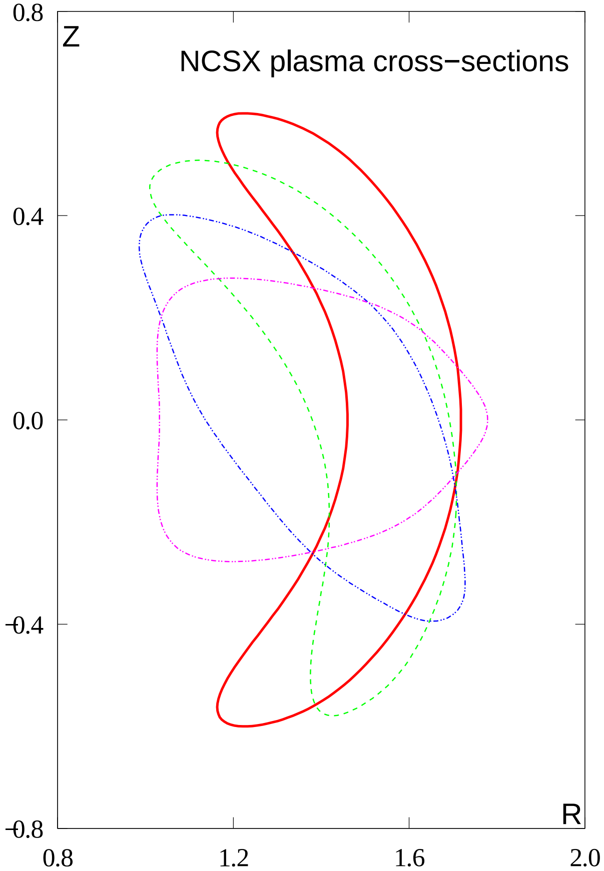
LiWF is a separate, self-consistent magnetic fusion concept, rather than an “improvement” of the old one.

The old one cannot be improved. It is not possible to make progress in magnetic fusion based on existing plasma regimes

New regimes and approaches, suggested by the LiWF concept, can put the power reactor development on a practical basis

Looking beyond RDF

The 3 steps strategy has a vision beyond the RDF



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

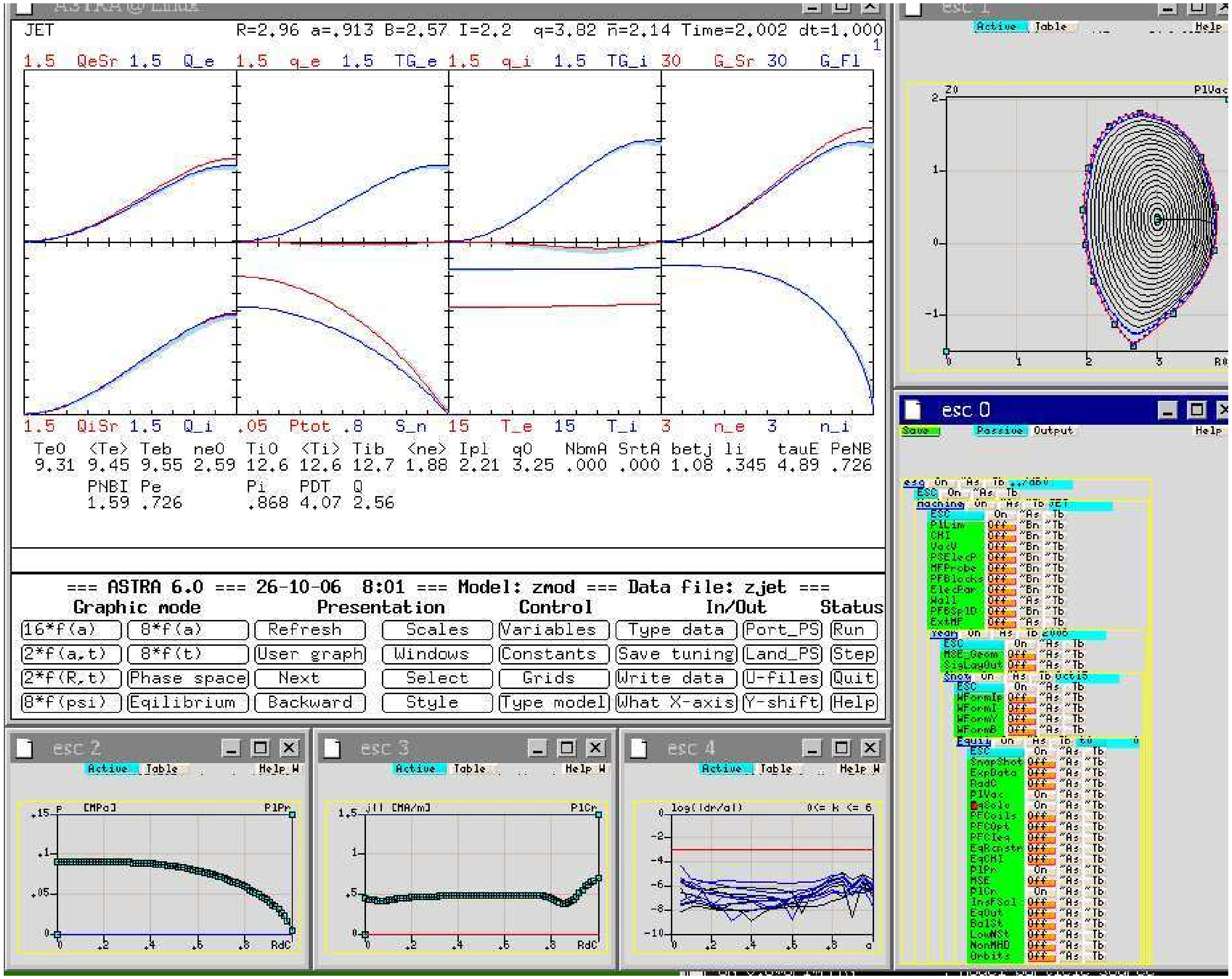
1. *Both are suitable for low energy NBI fueling*
2. *Both are "bad" for α-particle confinement and good for SCI regime*

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

Simulation of LiW regime for JET

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



Hot-ion mode:

$$T_i = 12.6 \text{ [keV]},$$
$$T_e = 9.45 \text{ [keV]},$$
$$n_e(0) = 0.3 \cdot 10^{20},$$
$$\tau_E = 4.9 \text{ [sec]},$$
$$P_{NBI} = 1.6 \text{ [MW]},$$
$$P_{DT} = 4.07 \text{ [MW]},$$
$$Q_{DT} = 2.56$$

3+2 MWs 50 keV NBI, are available

Can be experimentally tested on JET with intense Be conditioning