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

Getting serious about Fusion[1]

Leonid E. Zakharov

Princeton Plasma Physics Laboratory, MS-27 P.O. Box 451, Princeton NJ 08543-0451

Fusion Theory Colloquium

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Abstract

This provocatively titled talk presents an unconventional view on the basic issues of magnetic fusion (excluding its nuclear issues), such as: core fueling, confinement, stability, power and He extraction from the plasma. A super-critical regime is suggested when alpha heating is not essential for sustained fusion power production.

An unusual similarity between Spherical Tokamaks and stellarators is also mentioned.

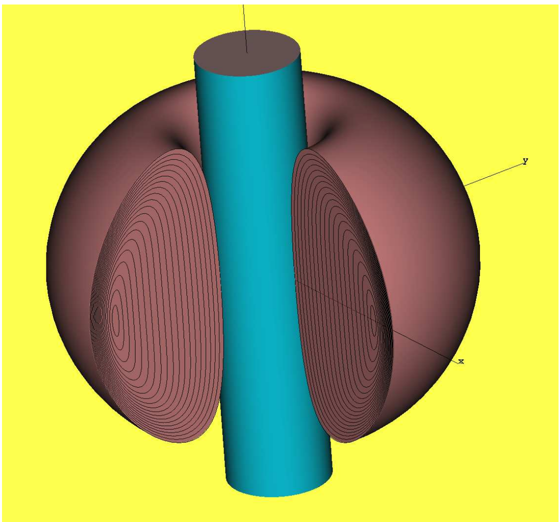
A separate national program ($\simeq$\$2-2.5 B for $\simeq$ 15 years) can realistically develop an Ignited Spherical Tokamak (IST) as a fusion neutron source for reactor R&D in 3 steps (2$\times$DD, 1$\times$DT):

1. ***A spherical tokamak, targeting achievement of absorbing wall regime with neo-classical confinement in a DD plasma and $Q_{DT-equiv} = 1$,***
2. ***Full scale DD-prototype of IST for demonstration of all aspects of stationary super-critical regime with $Q_{DT-equiv} \simeq 50$.***
3. ***IST itself with a DT plasma for reactor R&D and α-particle power extraction studies.***

Contents

1 The number $1\ kg/m^2$ of T in fusion strategy

The simple number 1 kg/m^2 of T $\equiv$ 15 MW·year/m^2 of neutron fluence uniquely specifies the fusion strategy for reactor R&D



Ignited Spherical Tokamaks (IST) are the only candidate:

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

ITER-like device ($\simeq$ 700 m^2 FW surface) would have to process 700 kg of T.

The possibility to have a unshielded copper central pole is a decisive factor in favor of IST as the reactor R&D tool

By definition, the “LiWall” regime is

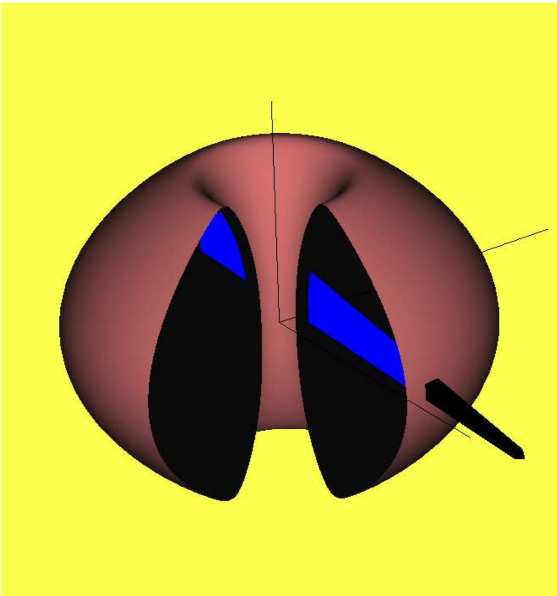
1. **Plasma fueling through the core, and**
2. **Absorbing walls $\equiv$ pumping boundary conditions for both ions and electrons**

$$\Gamma^{micro,ions}_{edge\rightarrow wall} \simeq \Gamma^{ions}_{convective}$$

$$\Gamma^{micro,electrons}_{edge\rightarrow wall} \simeq \Gamma^{electrons}_{convective}$$

Lithium plasma facing components provide, at least, the first condition of low recycling

Perfectly absorbing walls (no cold particles) would lead to a “flat” temperature, relevant to fusion



E.g, the atomic beam of 45 keV NBI will be converted into a plasma

$$E_b = \frac{3}{2}(T_i + T_e), T_i = T_e = 15\ keV$$

with collision frequencies

$$\nu_i = 68\frac{n_{20}}{T^{3/2}_{i,10}}, \quad \nu_e = 5800\frac{n_{20}}{\sqrt{T^3_{e,10}}},$$
$$\rho_i = 1.44\frac{\sqrt{T_{i,10}}}{B}\ [cm],$$
$$D_{ban} \simeq 0.016\frac{n_{20}}{B^2_p\sqrt{T_{i,10}}}\sqrt{\frac{a}{R}}\left[\frac{m^2}{sec}\right]. \tag{2.1}$$

When the density level becomes stationary, $T_i = T_e = const$

In “flat” temperature there is no mystery nor plasma physics

Absorbing walls lead to the best possible confinement situation

1. **No reasons for ITG or other turbulence**
2. **Thermo-conduction losses are essentially eliminated**
3. **τ_E is the same as particle confinement time, which is always determined by the best confined component.**

The neo-classical diffusion coefficient

$$D_{ban} \simeq 0.016 \frac{n_{20}}{B_p^2\sqrt{T_{i,10}}}\sqrt{\frac{a}{R}}\left[\frac{m^2}{sec}\right], \quad \tau_p \simeq \frac{a^2}{D_{ban}} \qquad (2.2)$$

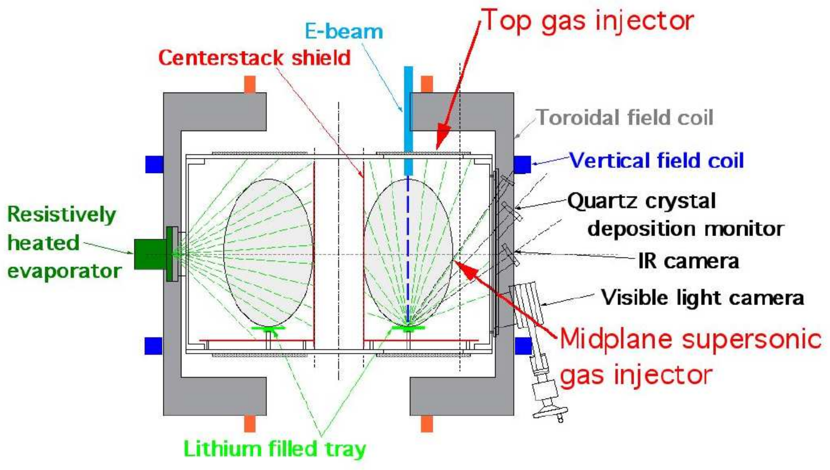
suggests the energy confinement time $\tau_E \simeq \tau_p \gg 10$ sec for $a \simeq 0.4,\ B > 2$ T.

To my knowledge, there is no indication of "profile-stiffness" for the density profile

CDX-U spherical tokamak research focuses on investigating lithium as plasma facing component

CDX-U

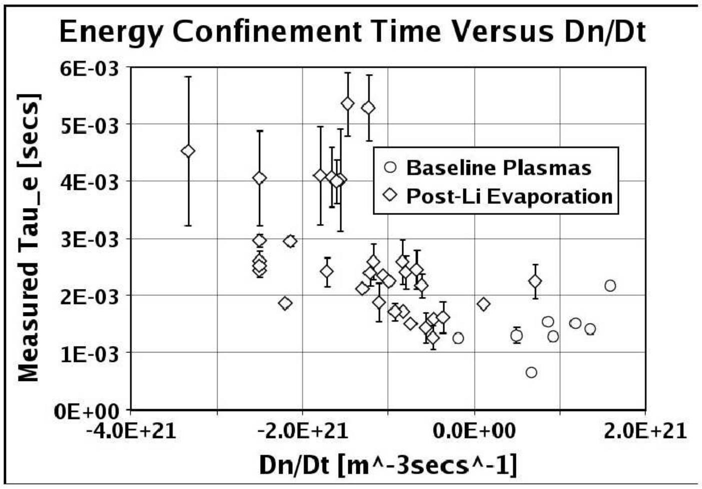
CDX-U: R_0=34 cm, a = 22 cm, κ≤1.6, $B_T(0)$~2.2 kG, $I_P \le 80$ kA, t_{disch}<25 msec, $T_e(0)$~100 eV, $n_e(0)<6\times10^{19}$ m^{-3}



R_0= 34 cm, width = 10 cm
6 mm deep

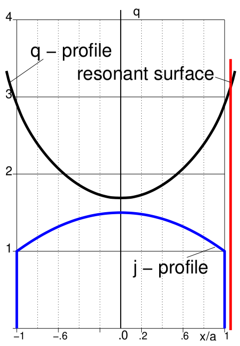
Longest τ_E correlated with lithium evaporated on plasma facing components

CDX-U LTX

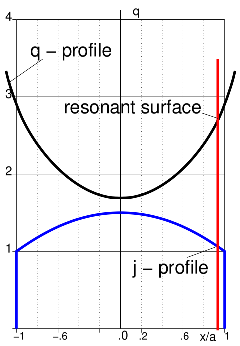


- Plasmas after lithium evaporated onto PFC's have highest "pumpout rate" (or fastest decrease in plasma density) after SGI
- Lithium wall related to energy confinement improvement of nearly 6x over values for plasmas without active lithium PFC's
  - Largest increase in Ohmic tokamak confinement ever observed

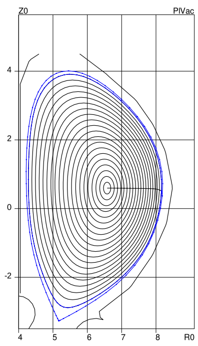
A widespread belief in MHD theory is that the high edge current density is destabilizing



case 1: $mq_a < n$
Ideally unstable



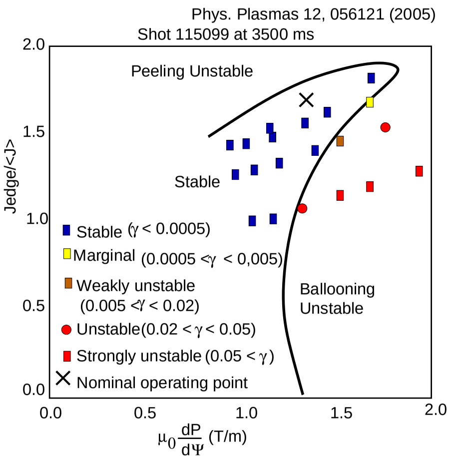
case 2: $mq_a > n$
Tearing stable



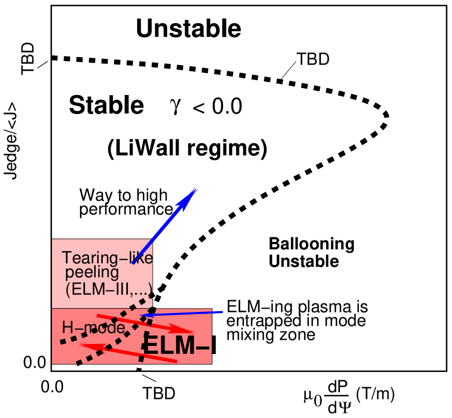
LiWall + Separatrix: $q_a = \infty$
Ideally & tearing stable

In presence of separatrix, the high edge current density is stabilizing

High edge temperature is stabilizing for ELMs.



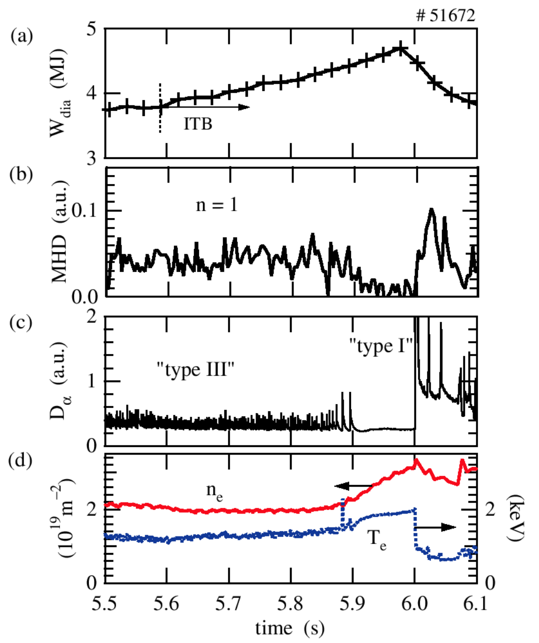
DIII-D interpretation



"Zakharov's" diagram (TBD)

The "peeling-ballooning" diagram is misleading

There is no "peeling" modes for the separatrix limited plasma if $j_{edge} \neq 0$



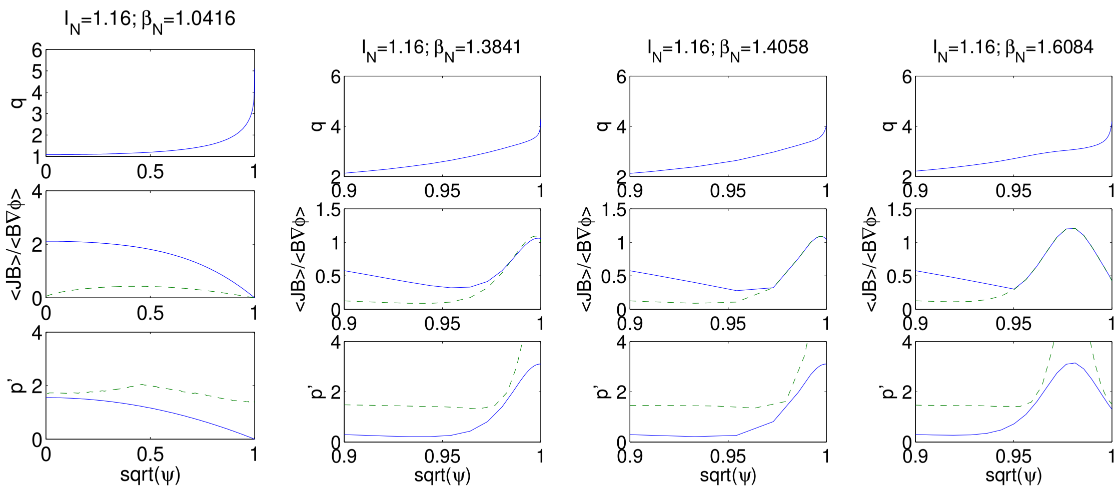
JET has a quiescent regime as a transient phase from ELM-III to ELM-I

"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 crucial role of the edge current density was emphasized in the paper

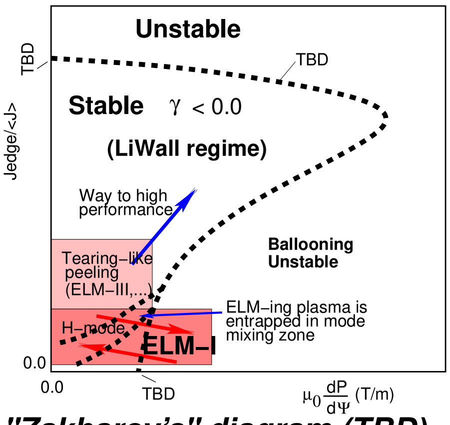
S.Medvedev’s group shown the absence of peeling modes with KINX code in 2003



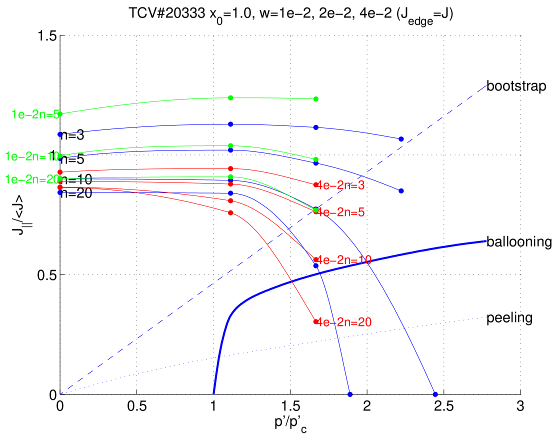
TCV-like profiles were used as a reference for KINX.

“Heuristic” and numerically calculated diagrams are identical

S.Yu. Medvedev, A.A.Martynov, et al. Plasma Phys.Control Fusion 48 927-938 (2006)



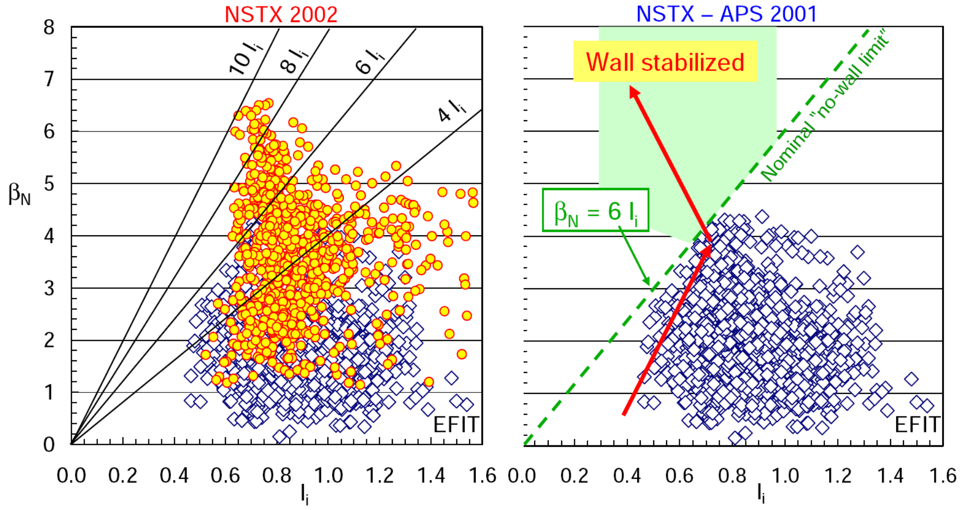
"Zakharov’s" diagram (TBD)



Keldysh Institute calculations

“Flat” temperature makes plasma stability robust and independent from the core physics

START, NSTX achieved the reactor R&D levels of beta

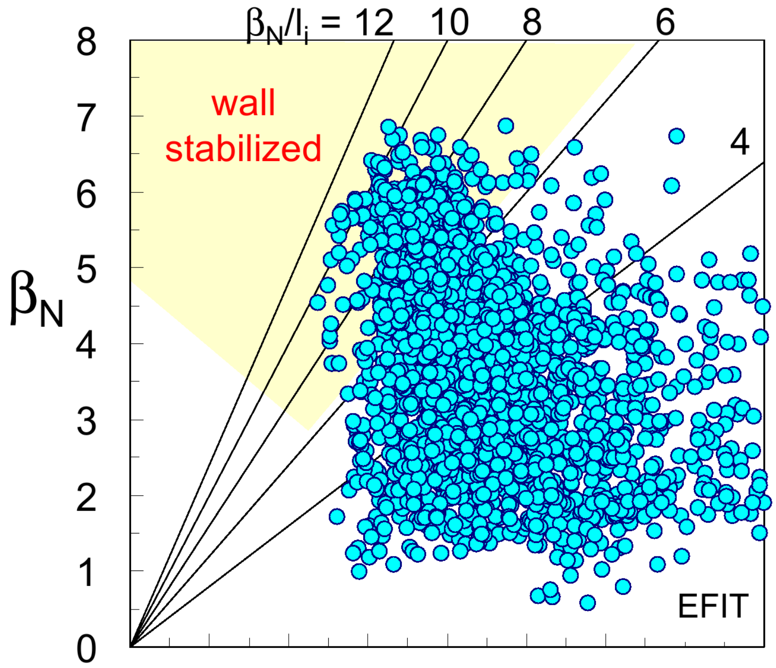


- Normalized beta, $\beta_N = 6.5$, with $\beta_N/l_i > 9.5$; $\beta_N > 30\%$ over $\beta_{N\ \text{no-wall}}$
- Toroidal beta has reached 34%

NSTX

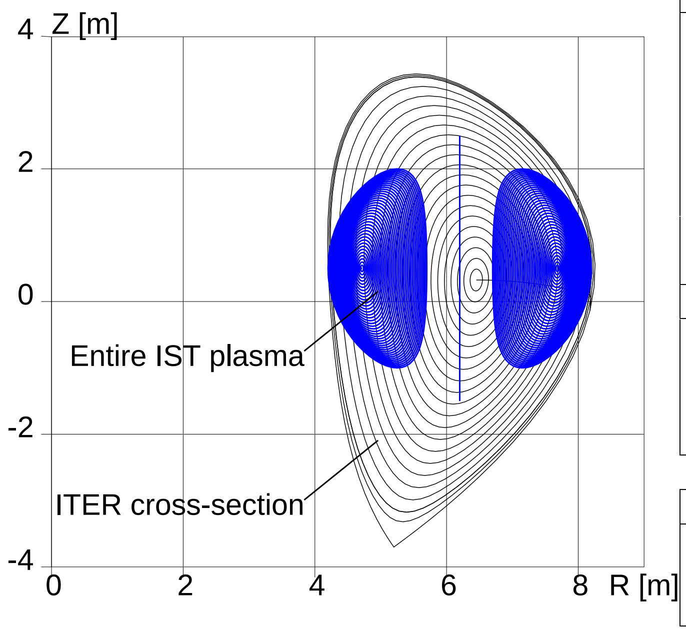
Tendencies in stability in NSTX are consistent with the LiWall concept

STs already have a relevant stability data base



In 2004 beta in NSTX approached the necessary 40 % ($\beta = 39\%$)

Only compact devices are suitable for reactor R&D

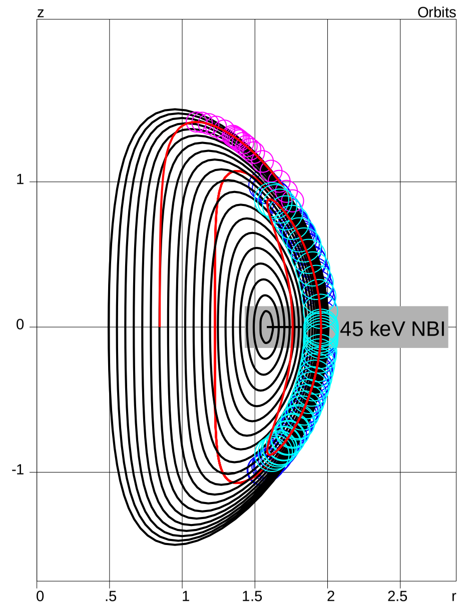


IST Parameters				
CenterPole R	m	0.5	0.5	0.5
CenterPole B	T	7.5	7.5	7.5
Plasma R_1	m	0.5	0.5	0.5
Plasma R_2	m	2.0	2.0	2.0
Height	m	3.0	3.2	3.4
Volume	m^3	26.1	27.8	29.6
Surface	m^2	53.4	55.9	58.5
I plasma	MA	11.1	11.9	12.7
IST Plasma performance				
P_{DT}	MW	388	490	606
τ_E	sec	0.75	0.69	0.64
$F_{neutron}$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

IST rely on β=0.4 and in "flat" $T_{i,e} \simeq$15 keV

Large Shafranov shift in ST makes core fueling possible



α-particles orbits in 8.4 MA IST

"Core" fueling is crucial for the density profile control.

The charge-exchange penetration length

$$\lambda_{cx} \simeq \frac{0.3}{n_{e,20}} \frac{V_b}{V_{b,40\ keV}} [m] \qquad (3.1)$$

The distance between magnetic axis and plasma the surface in IST

$$R_e - R_0 = 0.3 - 0.5\ [m] \qquad (3.2)$$

45 keV NBI can provide core fueling

Toroidal geometry of magnetic surfaces is favorable for core fueling

The NBI beam is quasi-1D while the volume $V(a)$ of magnetic surfaces is $\propto a^2$, where a is the minor radius of magnetic surface.

As a result, the particle source $S_{part}(a)$ distribution

$$\frac{d\dot{N}_b}{dx} = -\lambda_{cx} N_b, \quad S_{part}(a) \equiv \frac{d\dot{N}_b}{dV} = \frac{d\dot{N}_b}{dx}\frac{dx}{dV} \propto \frac{1}{a}, \tag{3.3}$$

despite the attenuation of the beam.

Without relying on other fueling ideas (e.g., HFS pellet injection)

ISTs allow a variety of NBI combinations for flexible fueling

A new, super-critical ignition (SCI) regime is possible in ST

The power balance in the plasma is given by

$$f_\alpha \int P_\alpha dV + P_b = \frac{E_{pl}}{\bar{\tau}_E}, \tag{3.4}$$

where $f_\alpha \le 1$ is a fraction of used α-particles.

For ignition at $f_\alpha = 1$

$$f_{pk} \langle p_{pl} \rangle \bar{\tau}_0 = 1, \quad f_{pk} \equiv \frac{\langle 4 p_D p_T \rangle}{\langle p \rangle} \simeq 1 \tag{3.5}$$

IST would need

$$\bar{\tau}_0 \simeq 0.7\ sec. \tag{3.6}$$

With a "flat" temperature and "excessive" τ_E

$$\bar{\tau}_E \gg \bar{\tau}_0 \tag{3.7}$$

IST can be in a "super-critical" ignition regime with $f_\alpha \ll 1$

Super-critical regime would change the philosophy of ignition

1. **No confinement of α-particles is necessary. They can be expelled to the wall at full energy.**

2. **Burn-up of tritium is enhanced**

$$n\left\langle\sigma v\right\rangle_{DT,16keV}\bar{\tau}_E = 0.03 n_{20}\bar{\tau}_E \rightarrow 1. \qquad (3.8)$$

3. **No issues with Helium dilution of the DT plasma.**

4. **Density profile and bootstrap current are controlled by NBI.**

5. **Power regime and fueling are externally controlled**

6. **A reasonable NBI can provide a high Q $\simeq$ 50 factor, e.g,**

$$P_b = \frac{E_{pl}}{\bar{\tau}_E}, \quad \text{e.g.,} \quad P_{DT} = 0.5\ GW \quad \int P_\alpha dV = 100\ MW$$
$$\bar{\tau}_E \simeq 10\tau_0, \quad P_b = 10\ MW, \quad Q \equiv \frac{P_{DT}}{P_b} = \frac{5\tau_E}{\tau_0} = 50. \qquad (3.9)$$

Only NBI power goes to the SoL in the SCI regime

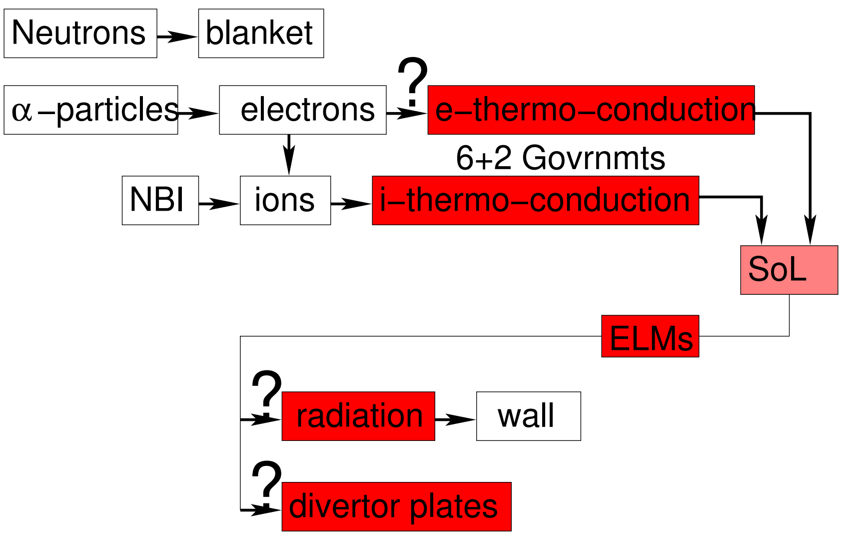
7. **$\simeq$ 60 % of α-particles can be intercepted by the wall in IST at first orbits.**

8. **Although non-uniform, their power is distributed over the wall surface, rather than being localized.**

9. **The α-particle expulsion reminds the ion losses during counter injection in DIII-D QHM.**

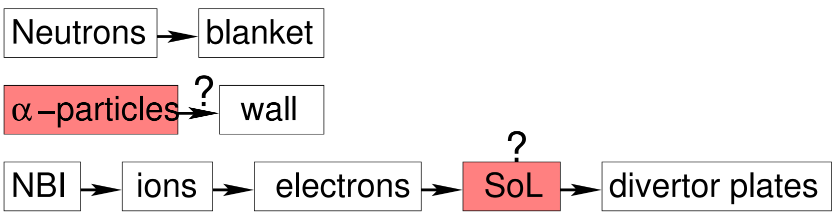
10. **A natural "Hot ion mode", $T_i > T_e$, is provided in SCI with NO high-tech involved.**

Power extraction in α-heating- and SCI-based fusions are totally different



α-heating-based fusion has a pile of problems on a way to PFC

Power extraction in SCI-based fusion is conceptually clean



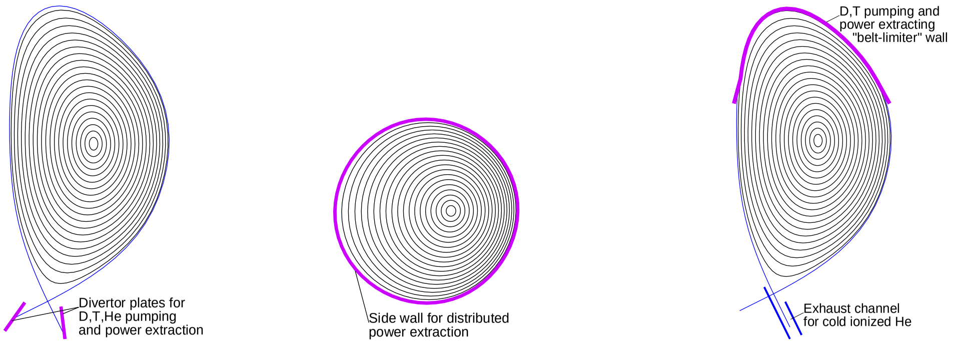
The new physics of α-particle losses (which are favorable) and of the collisionless SoL becomes essential.

The goal is to remove trapped particles from SOL. It is opposite to confinement related goal in former mirror machines.

Collisions, cone and flute instabilities work for expelling particles.

Plasma physics of IST is scalable to the power (SCI) reactor

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



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

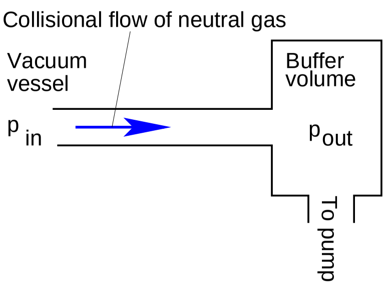
(a) conventional divertor: *all problems are well known; Not scalable to reactor*

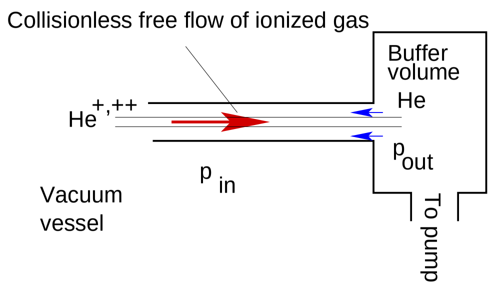
(b) the side walls: *inconsistent with particle, impurities and helium pumping:*

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 (as a cold gas) from D,T

The gas-dynamic scheme of He exhaust is currently adopted





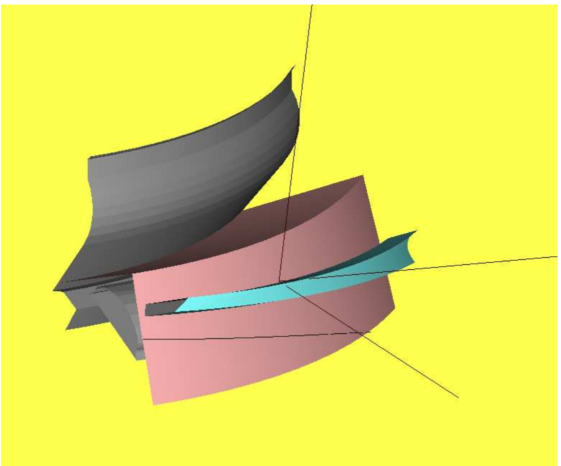
Conventional, gas-dynamic scheme:
a) collisional neutral gas in "pipe",
b) requires pressure drop

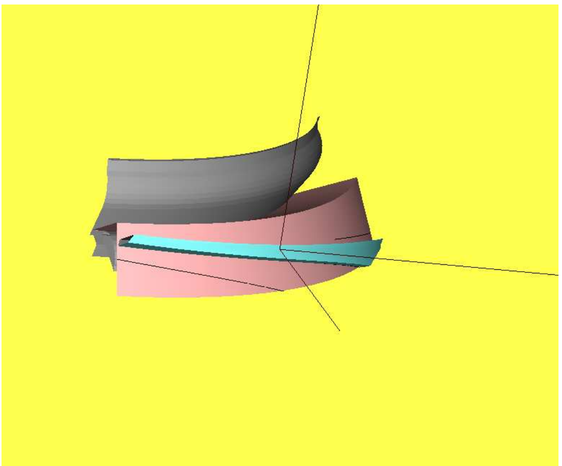
$$p_{in} > p_{out}$$

A scheme for ionized gas in tokamaks:
a) Free stream of He$^{+,++}$ along $\mathbf{B}$,
$\lambda \simeq \frac{1}{n\sigma_{cx0+}} \simeq \frac{1}{10^{12}\cdot 3\cdot 10^{-15}} \simeq 30\ [\mathrm{m}]$
b) Back flow is limited by
$\Gamma_{He} = Dn'_x, \quad D = hV_{thermal}$
c) Helium density in the chamber plays no role, while D is in the hands of engineers.

LiWall concept is consistent with pumping He using the second scheme

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

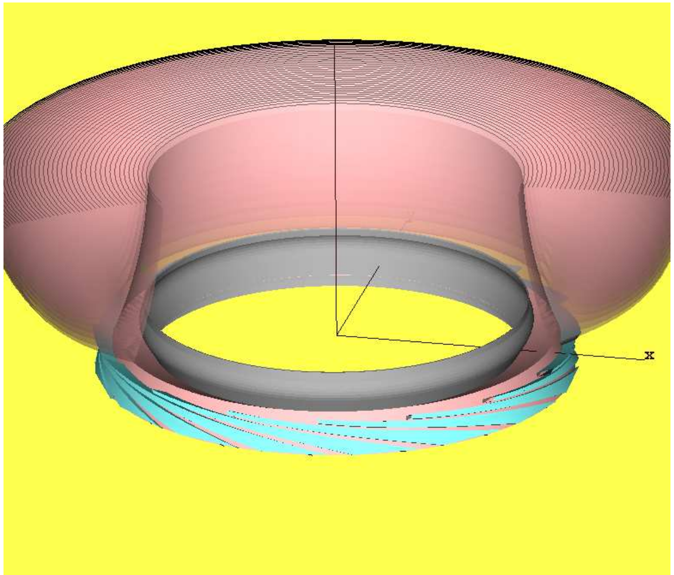




Both He$^{+,++}$ ions and incoming neutrals $n_{He} \simeq 10^{18}/m^3$ are collisionless

$$\textit{Flux}\quad \Gamma_{He^{+,++}} \ll \Gamma_{He}, \quad \textit{mean free path}\quad \lambda \simeq 10\ m \tag{3.10}$$

The poloidal extend of the duct is smaller than toroidal $L_{pol}/L_{tor} < 0.05$



The approach relies on

1. *Power absorption by divertor plates or bumper limiter*
2. *D,T pumping by lithium coating*
3. *Ionization of He, released from the Li surface, near the separatrix*

The size of honeycomb channels is exaggerated. Also, B_{tor} is reduced by a factor of two.

Honeycomb channel ducts allow to pump He$^{+,++}$ ions, while trapping neutral He backflow

4 The 3 steps strategy toward the power reactor

The IST concept opens a way toward power reactor R&D

All three mutually linked objectives of magnetic fusion, i.e.,

1. ***Development of the high power density Operational Power Reactor Regime, $\simeq$ 10 MW/m^3 (0.5 MW/m^3 in ITER, 1000 MW/m^3 in a fission sub-critical cell),***
2. ***Development of the "First Wall" (FW), i.e., first 15 cms of the structure faced by 14 MeV neutrons,***
3. ***Tritium Cycle***

can be achieved with ISTs.

The Quiescent H-Mode discovered on DIII-D gives a basis for optimism

Three steps (2$\times$DD, 1$\times$DT) are necessary to develop an IST

1. **ST, targeting achievement of absorbing, LiWall regime with neo-classical confinement in a DD plasma and**

$$Q_{DT-equiv} \simeq 1$$

2. **A full scale DD-prototype of IST for demonstration of all aspects of a stationary super-critical regime with**

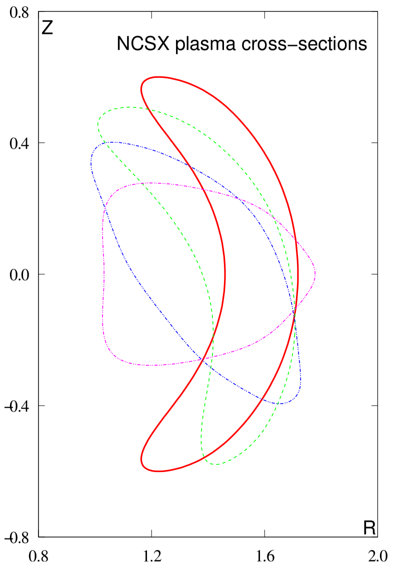
$$Q_{DT-equiv} \simeq 50$$

3. **IST itself with a DT plasma as a neutron source for reactor R&D and α-particle power extraction studies.**

$$Q_{DT} \simeq 50$$

15 years for a separate $\simeq$\$2-2.5 B program is a reasonable time interval for implementation of 3 steps

The 3 steps strategy has a vision beyond the IST based R&D



Regarding SCI 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 SCI-based strategy includes both R&D and power production phase of fusion energetics