What Fusion Energy Science (FES) do we have 15 years after TFTR*

Leonid E. Zakharov

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

PPPL Experimental Seminar

December 18, 2012, PPPL, Princeton NJ

*This work is supported by US DoE contract No. DE-AC02-09-CH11466.

Abstract

In defiance to many "fantastically incorrect statements"[1] of opponents fusion propaganda (which is intended to provide energy from "seawater"[2] while being "unaware of any major project failure in magnetic fusion research"[3]) often uses a trick of making people feel fool in front of "computer simulations of plasma turbulence which helps scientists predict plasma behavior"[3].

In fact, these simulations and the three decade long obsession of FES with the core transport, were critical in termination progress in fusion. During the last 15 years the fusion program followed exactly the path understood and predicted by "The theory of the failure of magnetic fusion"[4] (LZ, 2004), i.e., from progress to stagnation, and then to degradation, when science no longer plays a role.

At this point the result is devastating. After 3-4 decades of development:

(a) confinement theory with its 3-5-D numerical codes has no idea where the confinement zone is in tokamaks,

(b) the macroscopic stability codes simulate the free boundary plasma as "salt" water, mixed with halo-currents,

(c) there is not even a basic understanding of the plasma edge and pedestal region,

(d) the "miraculous" edge transport barrier has created an entire industry of cooking shear flow stabilizations, pedestal bootstrap currents, peeling-ballooning edge stability, screening of RMP, etc.

The energy "vision" of FES (except its energy from "seawater") is simply ridiculous. After 15 years of existence, FES failed not only in the energy aspects, but even in of science. The situation with FES can only get worse.

In contrast, the basic level of science of magnetic fusion has been created in a separate, essentially underground effort. It provided a much deeper understanding of the tokamak plasma and now raises the necessity of a separate program which would aim toward a P_{DT}*=100-200 MW DEMO device with the electric Q factor exceeding unity.*

[1] Stewart Prager, Richard Hazeltine, "Rohrabacher's Comments on Fusion Research Are "Misinformed"", APS News, August/September 1995 (Volume 4, Number 8) `http://www.aps.org/publications/apsnews/199508/letters.cfm`

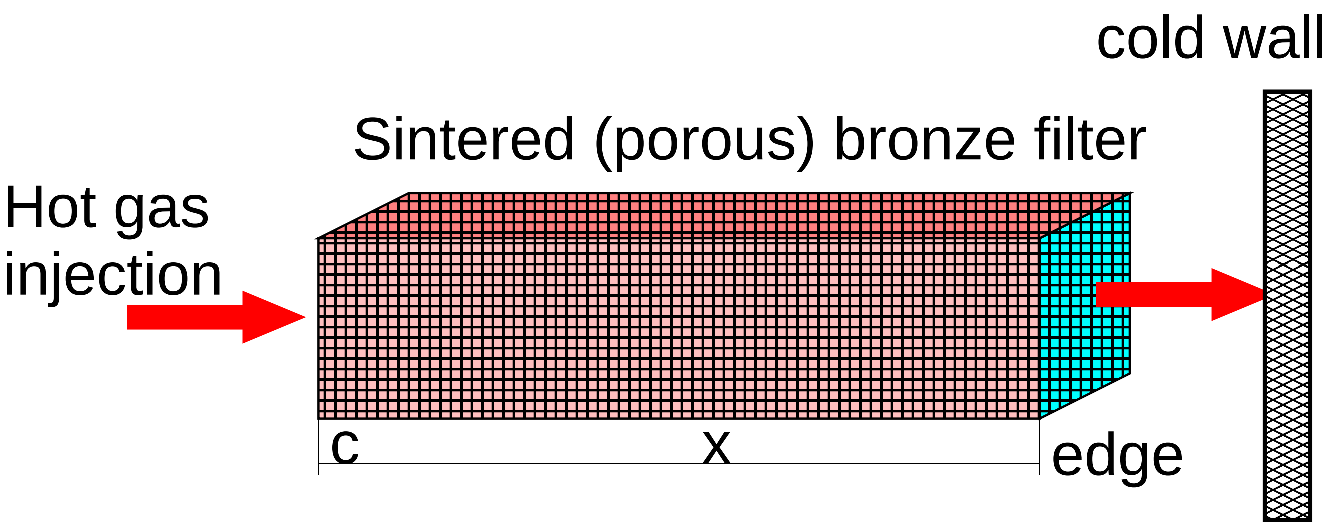
[2] Stewart Prager, "How Seawater Can Power the World", NYTimes, The Opinion Pages, 07.11.2011 `http://www.nytimes.com/2011/07/11/opinion/11Prager.html?_r=0`

[3] Stewart Prager, "The Way Forward with Magnetic Fusion Energy", NYTimes, The Opinion Pages, 11.19.2012 `http://dotearth.blogs.nytimes.com/2012/11/19/in-defense-of-sustained-research-on-fusion/`

[4] Leonid E. Zakharov, "The theory of the failure of magnetic fusion", APS DPP-2007 `http://http://w3.pppl.gov/~zakharov/APS-07F.pdf`

Contents

1 Energy confinement in simple terms



1. ***Hot gas is injected into the porous metal filter from left;***
2. ***Heat is transferred to the right by thermal conduction and with gas diffusion;***
3. ***Side surfaces are assumed to be thermally insulated.***

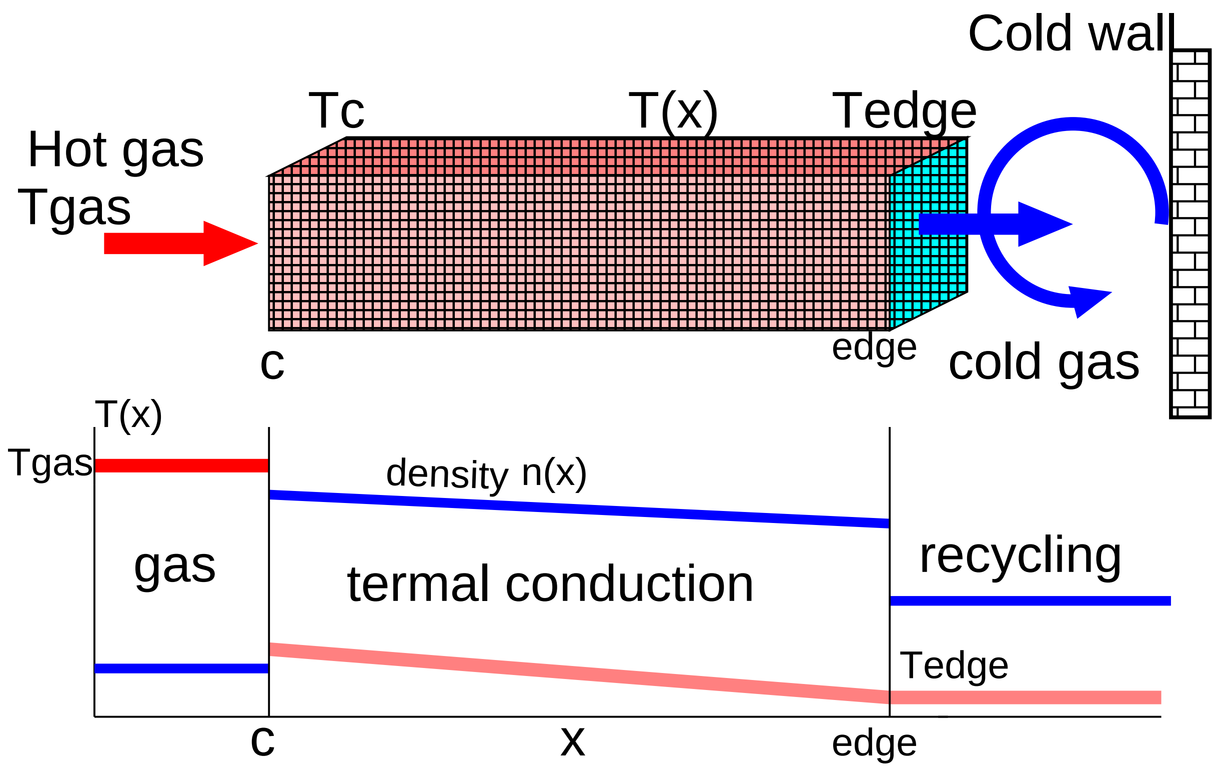
$$q^{heat} = \frac{5}{2} T^{gas} \Gamma^{hot\ gas} = -\kappa \frac{dT(x)}{dx} - \frac{5}{2} T(x) D \frac{dn(x)}{dx} = \frac{5}{2} T^{edge} \Gamma^{edge \to wall}, \qquad (1.1)$$

where $\Gamma^{hot\ gas}, -D\frac{dn(x)}{dx}, \Gamma^{edge\to wall}$ ***are the particle fluxes.***

The temperature profile depends on boundary conditions on the right surface.

1.1 Regime controlled by thermal conduction

High recycling leads to low edge temperature



$$T^{edge} = \frac{2}{5} \cdot \frac{q^{heat}}{\Gamma^{edge \to wall}}, \qquad \nabla T(x) = \frac{q^{heat}}{\kappa}, \qquad T(x) \ll T_{gas}. \qquad (1.2)$$

The core temperature T_c is determined by the heat flux and thermal conduction, rather than by the hot gas temperature

$$q^{heat} = n\chi\frac{T_c}{L}, \qquad \kappa \equiv n\chi, \qquad W = \frac{3}{2}\frac{nT_cL}{2}. \tag{1.3}$$

The energy confinement time is determined by size L of the system the temperature conduction coefficient χ

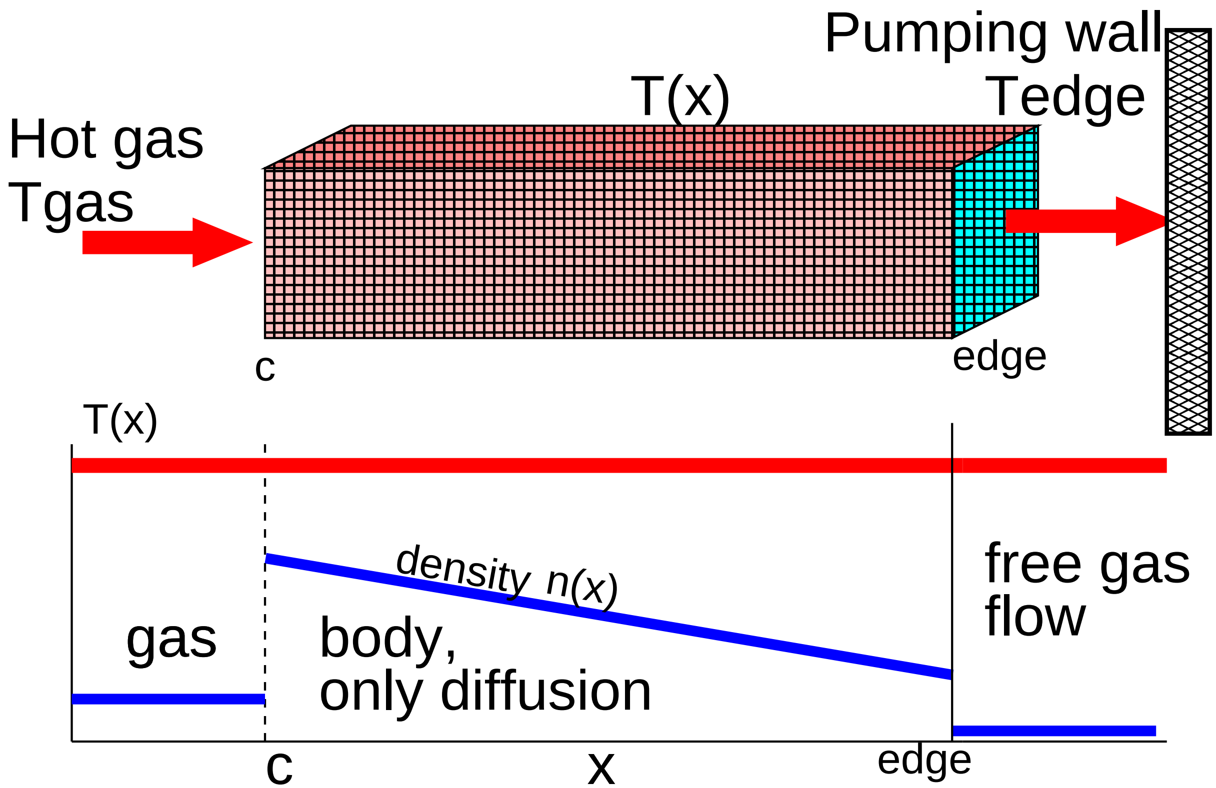
$$\boxed{\tau_E \equiv \frac{W}{q^{heat}} = \frac{3}{4}\frac{L^2}{\chi}, \qquad (1.4)}$$

where χ is determined by the physical properties of the core.

For anomalously large χ the only way to have a good τ_E is to increase the size of the system L

1.2 Pumping walls. Diffusion based confinement regime

Pumping walls prevent edge cooling



Everything is very simple: $$T(x) = T_{gas} \qquad (1.5)$$

No dependence on thermal conduction χ. Wall is invisible.

Energy confinement time for diffusion based regime

The core temperature T is determined by hot-gas temperature

The core temperature is well determined $T_{core} \simeq T^{hot-gas}$

Thermal energy and losses have amazingly simple expressions, containing only global parameters,

$$q^{heat} = \frac{5}{2}\Gamma^{hot-gas}T^{hot-gas}, \qquad W = \frac{3}{2}NT_c. \tag{1.6}$$

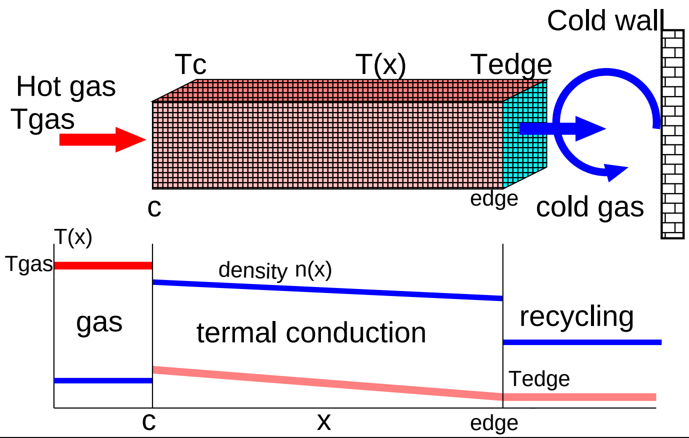
The energy confinement time is determined the diffusion coefficient D, rather than by χ

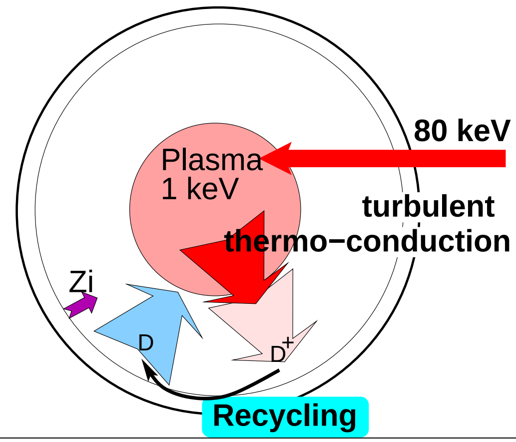
$$\tau_E = \frac{W}{q^{heat}} = \frac{3}{5}\frac{N}{\Gamma^{hot-gas}} = \frac{3}{5}\tau_D, \tag{1.7}$$

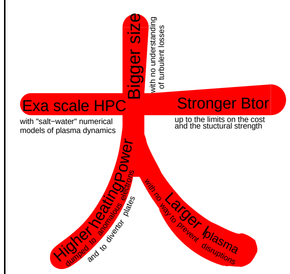
Diffusion is determined by a different physics than χ and can be orders of magnitude smaller than thermal conduction

$$D \ll \chi. \tag{1.8}$$

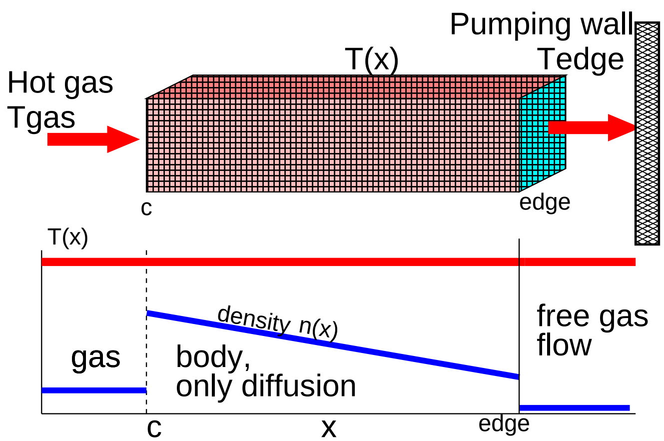
Analogy with the plasma

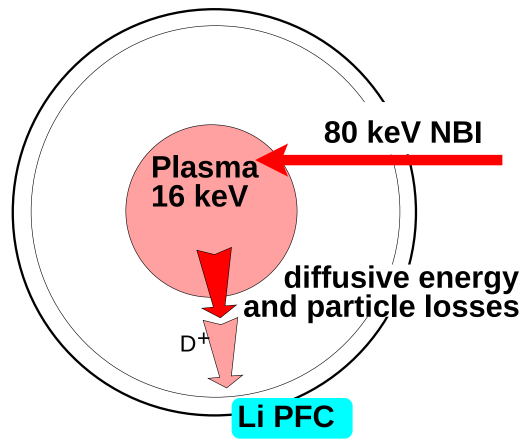






Fusion of 5 "Bigs" (FES)





Diffusion based confinement

$$\tau_E \simeq \frac{3}{5}\tau_D(1-R^{ecycle})$$

pumping divertor, LiWF (Lithium Wall Fusion)

For magnetically confined plasma, it is much more efficient to prevent plasma cooling by pumping out neutrals recycled from the walls, rather than to rely on extensive heating power in order to compensate the essentially unlimited energy losses from the turbulent plasma core.

Diffusion based confinement regime is not affected by anomalous electrons

Fusion needs good confinement

Misrepresentation of the confinement problem as a "core transport" problem is the biggest, root-level mistake of the current program.

It costed tens of Bs $ and decades of wasted research time.

Yes, it would be remarkably good if the turbulent thermal conduction coefficients were orders of magnitude smaller. No such luck, TFTR and JET proved this.

In contrast, the confinement with pumping walls is determined by the BEST confined component (e.g., by ions even in the case of anomalous electrons).

Confinement is NOT the same as the core transport.

Confinement is much more sensitive to the plasma edge conditions (which, in fact, control the core transport)

The "understanding" of plasma turbulence for fusion purposes and relevant control of the core transport is a fantasy.

In contrast, arranging NBI and plasma-wall interaction is a practical task for technology, engineering, design and experiments.

Initially poor TFTR confinement was rescued by carbon PFC conditioning and discovery of supershots (Jim Strachan).

Then, Lithium assisted supershots (Jim Strachan, Dennis Mansfield, others) elevated fusion tripple product in the core $\boxed{n(0)T(0)\tau_E}$ ***by a factor of*** **56 !!!**

These brightest experimental facts have been ignored by theory and management. The program attention was on turbulent transport ("Bohm", "gyro-Bohm", PPPL-IFS model).

The TFTR program was lost at $Q_{DT} = 0.25 < 1$.

Soon, in Dec. 1998 (14 years ago) the potential effect of lithium was understood (S.Krasheninnikov, LZ)

For TFTR, it would be sufficient to double τ_E ***in order to get the targeted*** $Q_{DT} = 1$***. (Later on, with liquid lithium (LiLi) CDX-U easily quadrupled*** τ_E***.)***

But the fusion establishment dreamers of energy from seawater and newly PPPL enthusiasts of 3D core confinement by a single magnetic surface (NCSX) has destroyed the capable device.

The key to good confinement

The message to Chinese:

The key to the fusion relevant confinement is in development of the diffusion based confinement regime, which is insensitive to always anomalous electrons

The misinterpretation of the confinement problems is a multi-billion dollar mistake of conventional fusion. Do not repeat it.

What was achieved after TFTR in the environment of obsession the core transport (confirmed last week by the OFES Assistant Director E.S.) ?

2 Plasma edge and the confinement zone

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

Energy fluxes $Q_{i,e}$ are transported to the wall by the particle flux:

$$
\begin{aligned}
\frac{5}{2}\Gamma_e^{edge-wall}T_e^{edge} &= Q_e^{core-edge} = \underbrace{\int_V P_e dV}_{\substack{heat\ source\\ for\ electrons}} - \frac{\partial}{\partial t}\int_V \frac{3}{2} n T_e dV, \\
\frac{5}{2}\Gamma_i^{edge-wall}T_i^{edge} &= Q_i^{core-edge} = \underbrace{\int_V P_i dV}_{\substack{heat\ source\\ for\ ions}} - \frac{\partial}{\partial t}\int_V \frac{3}{2} n T_i dV.
\end{aligned}
\tag{2.1}
$$

Edge temperature does not depend on transport coefficients near the edge. Potential ∇n-driven turbulence (e.g., TEM) also would have no effect on T^{edge}.

This property of T^{edge} allows to determine the real position of the plasma edge and the size of the energy confinement zone

The confinement zone is not what TTF experts are thinking and what the "first principles" codes are simulating

2.1 Edge “transport barrier”: a fantasy or a reality

In figure, the normal person see the same sudden drop of ion and electron temperature at the plasma edge. The certified experts of TTF see a remarkable “transport” barrier

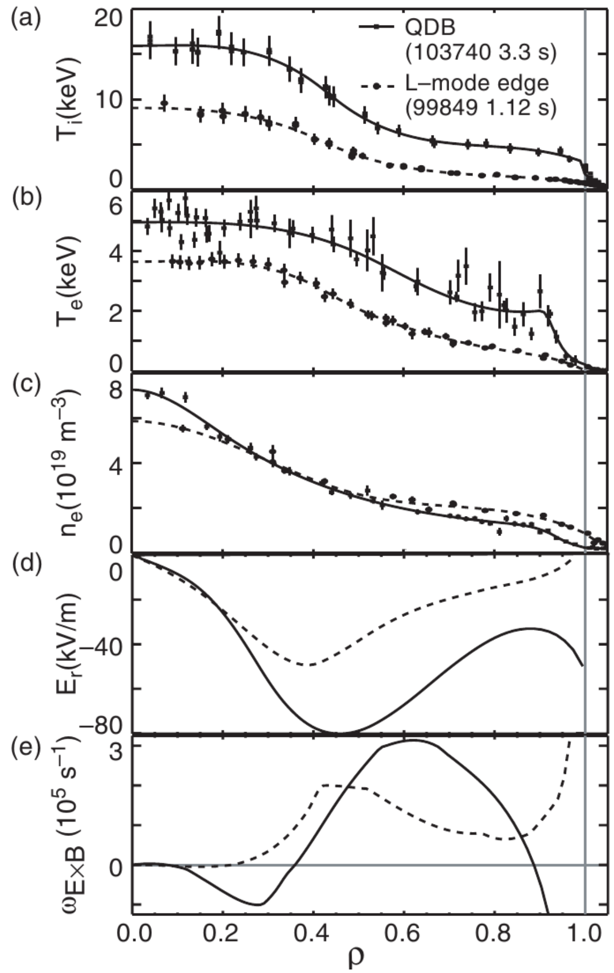


Figure 4 Kinetic profiles from a QDB (103740) and ITB with an L-mode edge (99849). (a) Ion and (b) electron temperatures, (c) electron density, (d) radial electric field, and (e) E x B shearing rate. *The picture of an H-mode below was taken arbitrarily from paper “The quiescent double barrier regime in DIII-D” by C. M. Greenfield, K. H. Burrell, E. J. Doyle et al. Plasma Phys. Control. Fusion 44 (2002) A123-A135. There are many similar pictures from different regimes on DIII-D and from other machines.*

What GK theory sees on these plots is a sharp gradient of electron temperature in the pedestal region, which is located inside the separatrix ($\rho = 1$). For GK this automatically means the presence of two zones of confinement: a core and the “edge transport barrier” (ETB) with suppressed radial transport.

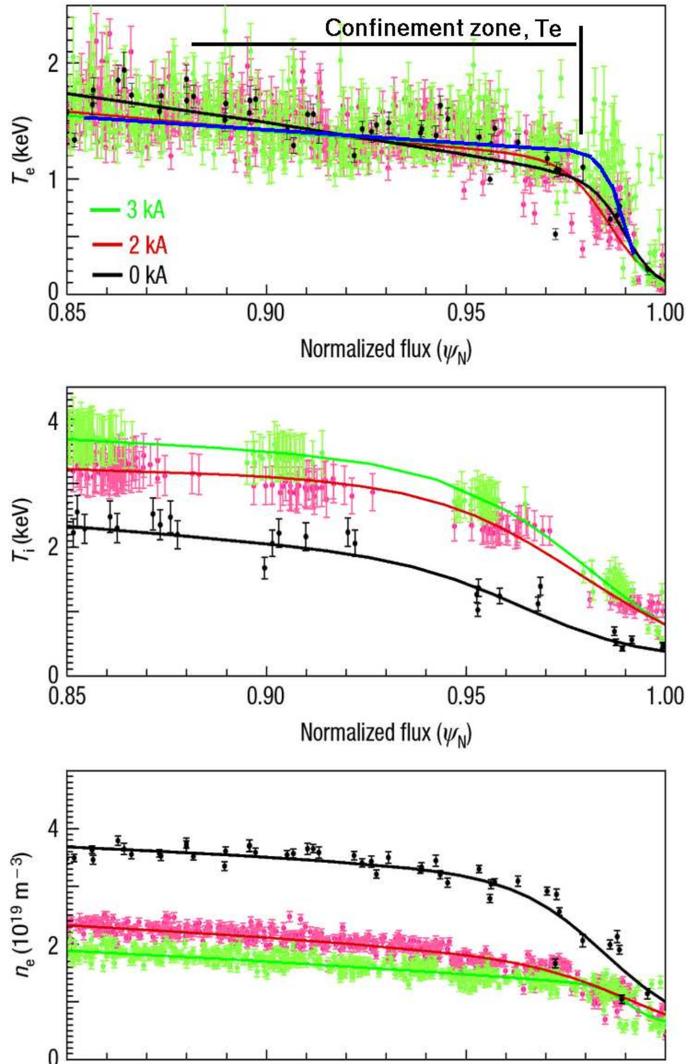
At the same time, a normal physicist would notice a similar sharp gradient on the ion temperature. In this example it is clearly located outside the separatrix. Nobody would suggest a transport barrier in the open field line region where there is no confinement. The normal physicist would reasonably suggest that the sharp electron temperature gradient has the same reason - open field lines, rather than mythical “edge transport barrier”. Accordingly the pedestal region has no electron confinement.

DIII-D experiments with QHM and especially with RMP has confirmed the common sense and the interpretation of the normal physicist: for electrons the confinement zone extends from the magnetic axis to the top of the temperature pedestal. In the pedestal region not only there is no any transport “barrier”, there is no confinement at all.

Fusion establishment advertised shear flow stabilization of the plasma edge as a great achievement of gyro-kinetic theory. In contrast, this is an outstanding example in a series of failures of heavily funded worthless theory.

2.2 RMP on DIII-D have determined the edge position

RMP experiments on DIII-D have determined the size of the confinement zone



0 kA, 2 kA, 3 kA $I_{RMP-coil}$

T.Evans at al., Nature physics 2, p.419, (2006)

1. ***The pedestal*** $T_e^{pedestal}$ ***is found insensitive to RMP*** $\rightarrow T_e^{pedestal}$ ***is the*** T_e^{edge} $\rightarrow$ ***The tip of the*** T_e ***pedestal is the boundary of the confinement zone for electrons.***
2. ***RMP do penetrate into the confinement zone: The gradients***
$$n'(x),\ T_e'(x)$$
in the core are reduced by RMP - indication of "screening".
3. ***Different positions of the "edge" for*** T_e, T_i, n_e ***are possible***

Claims about flow shear "stabilization" of turbulence and suppressed transport in the pedestal are baseless.

It is just opposite: there is no electron confinement in the pedestal region.

The pedestal is situated outside the confinement zone

SOLCs exist even in the most quiet plasma. They are the key to the understanding of the plasma edge.

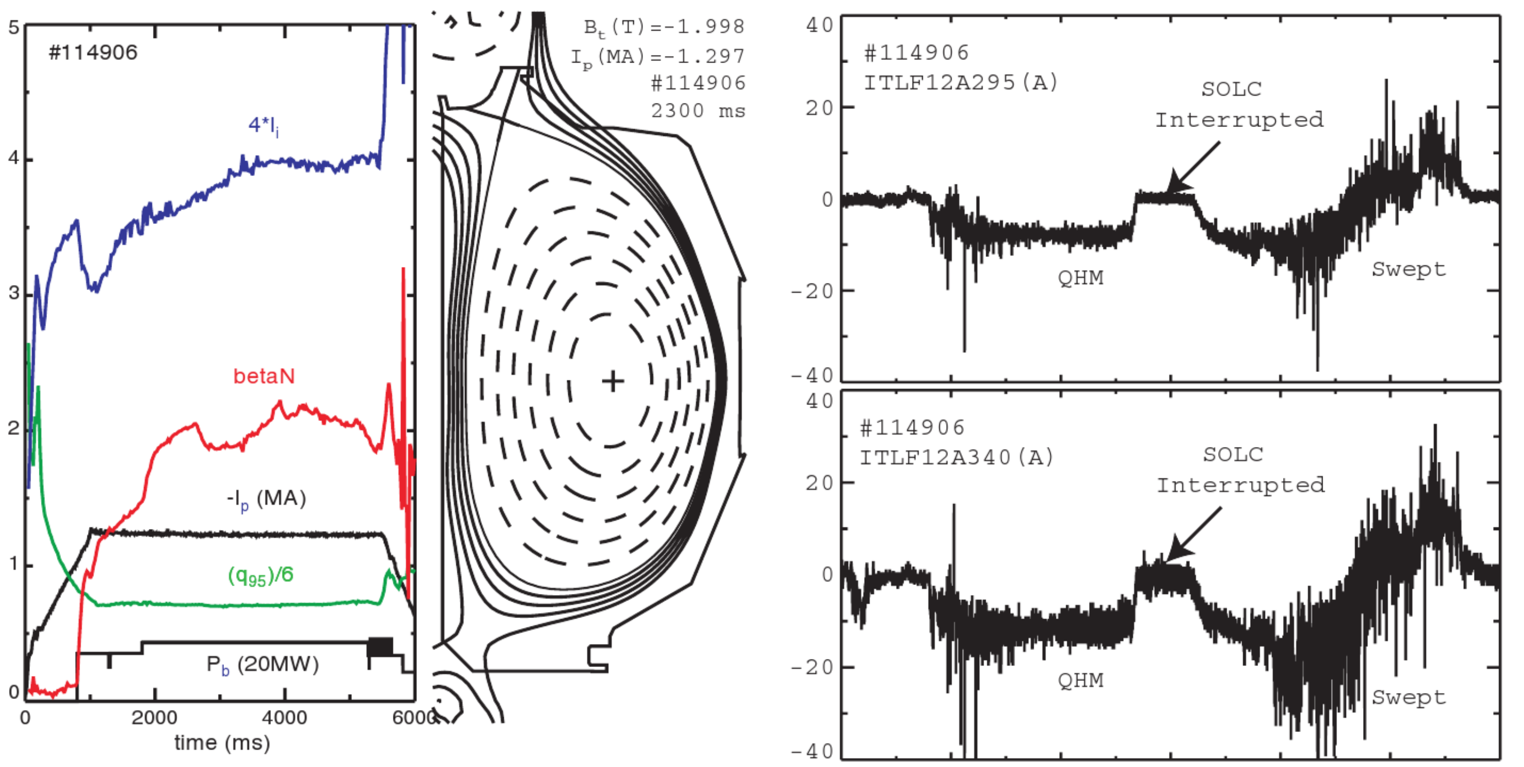


Figure 3. Pictorial discharge summary; the left-hand panel shows I_p in units of megaamperes, P_b in units of 20 MW, q_{95} divided by 6, β_N, and the nominal no-wall limit (here, 4 li). The right-hand panel shows the plasma boundary and four exterior flux surfaces in the

Figure 4. Signals from tile current sensors in tile ring #12 A in the discharge shown in the previous figure. It has a period of QHM over

Todd Evans, Hiro Takahashi and Eric Fredrickson (NF,2004) have found a link between SOLCs and MHD activity on DIII-D. SOLCs are the first candidate for intrinsic perturbations, which determine the width of the temperature pedestal.

3 ETB based industry of cooking FE“Science”

The most prominent examples:

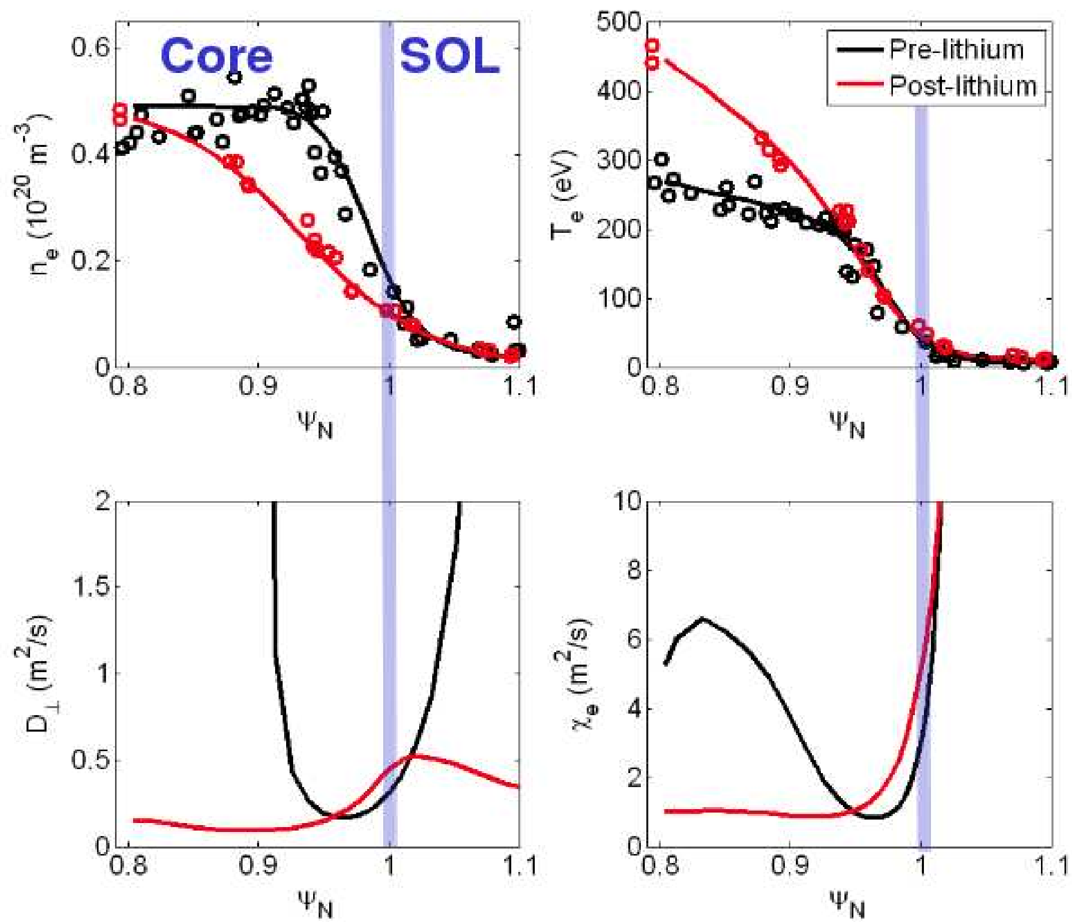
1. ***Shear flow stabilization** of turbulence in the pedestal region (to the level below neo-classical transport);*
2. ***Screening** the external magnetic field perturbations (RMP) by plasma sheared flow;*
3. ***Huge edge localized bootstrap current**, “confirmed” by GIGO 5-D kinetic simulations;*
4. ***“Peeling-ballooning”** model of ELM stability;*
5. ***EPED model of the width/height** of the pedestal.*

In addition

1. ***H-, I-modes** as an example of a remarkable cooperation of the plasma with gyro-kinetic physicists (who have no idea where is the confinement zone in tokamaks.)*
2. ***Confinement scalings** with 1 % precision in the exponents of core parameters, all missing the key physics of the plasma edge and instead introducing a free H-factor in order to hide the failure of theory.*

3.1 “Suppressed” transport at the edge

Profiles in discharges with lithium coatings only reproduced with recycling *and* transport reduction



R_p=0.98,
P_{loss}=3.7 MW

R_p=0.92,
P_{loss}=1.9 MW

(2D Plasma Neutral Transport Modeling With SOLPS)

R_P=divertor recycling coefficient

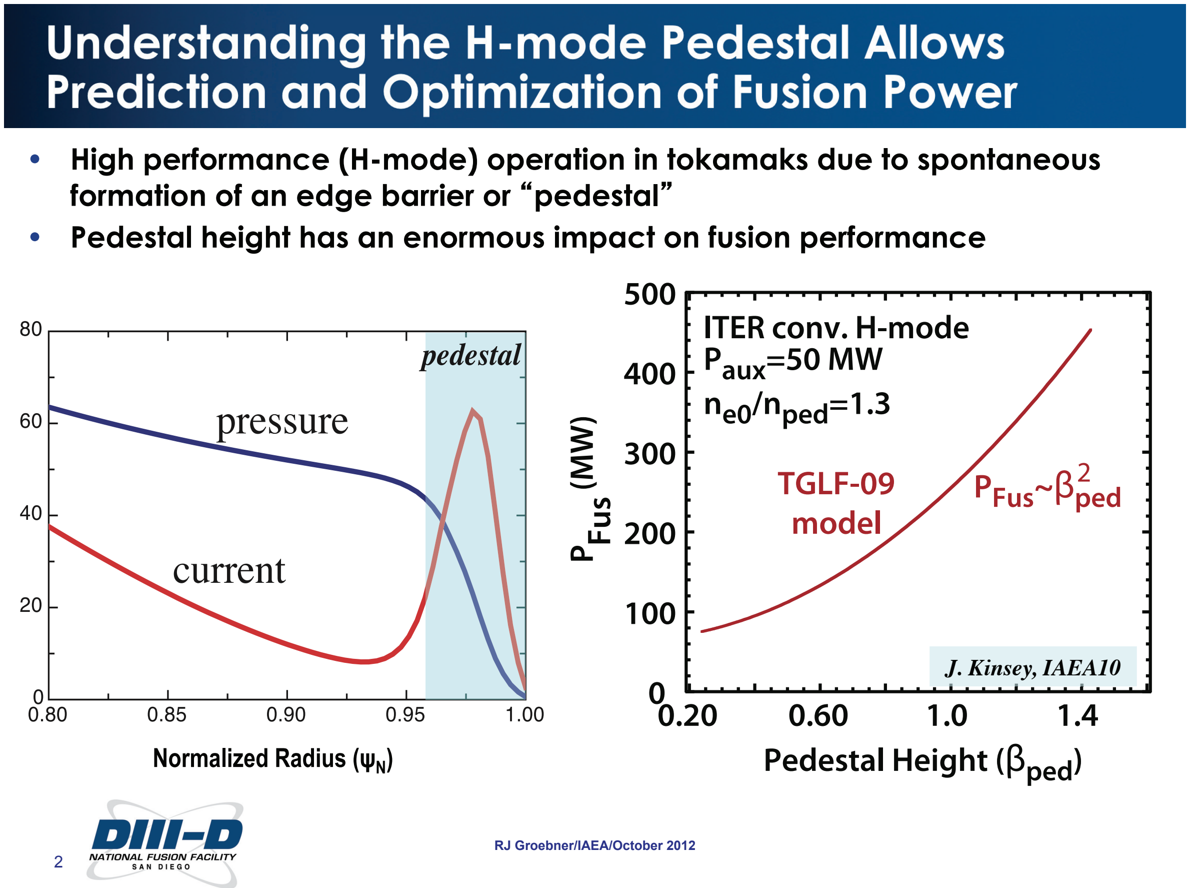
J. Canik, PSI 2010

NSTX OAK RIDGE IAEA FEC 2010 meeting: ELM suppression with lithium R. Maingi EXC/2-2 11-16 Oct. 2010 9

Plasma is cooperating with core transport physicists by exceptional thermal insulation

3.2 "Outstanding" edge localized bootstrap current

"Improved Understanding of Physics Processes in Pedestal Structure, Leading to Improved Predictive Capability for ITER" (author+61 contributors) -IAEA-24 FEC



In magnetic fusion a beautiful huge "bootstrap current" in the pedestal was cooked with a theoretical formula for perfect magnetic surfaces, not applicable for the plasma edge.

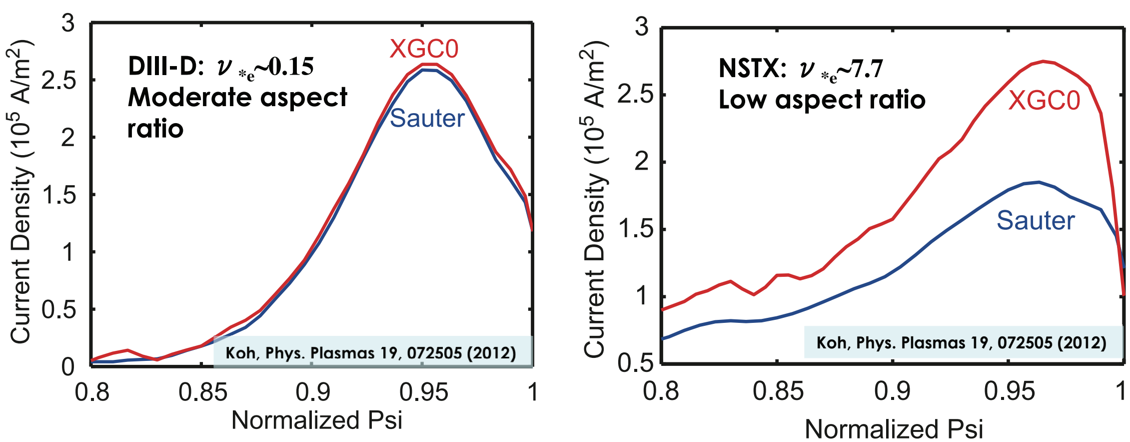
In fact, there is no mechanism for a significant bootstrap current in the pedestal zone

First Principle GIGO 5-D kinetic simulations

"Improved Understanding of Physics Processes in Pedestal Structure, Leading to Improved Predictive Capability for ITER" (author+61 contributors) -IAEA-24 FEC

Kinetic Codes for Neoclassical Bootstrap Current Have Been Used to Benchmark Simpler Models

- **NEO: ~10%–20% differences in the bootstrap current from simplified models**
- **XGC0: Agreement with Sauter model in banana-plateau regime**
  - Some differences in collisional regime
- **MIT Global Pedestal DK Code: Agreement with Sauter in banana**
  - Some disagreement in plateau [Landreman & Ernst, PPCF 2012]

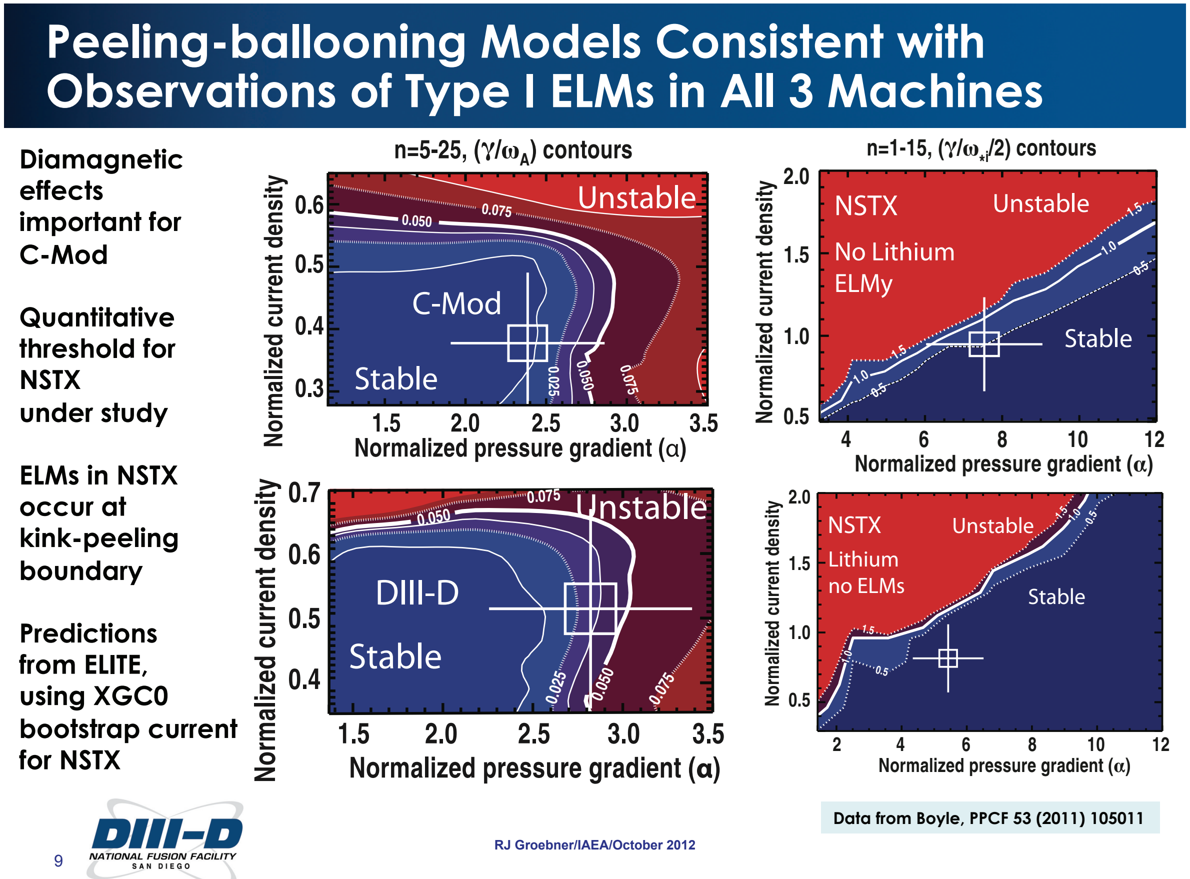


8 DIII-D NATIONAL FUSION FACILITY SAN DIEGO RJ Groebner/IAEA/October 2012

Misunderstanding of the basic physics of the plasma edge has been translated into GIGO (garbage-in-garbage-out) kinetic simulations generating the false bootstrap current.

3.3 "Peeling-ballooning" models of ELMs

"Improved Understanding of Physics Processes in Pedestal Structure, Leading to Improved Predictive Capability for ITER" (author+61 contributors) -IAEA-24 FEC

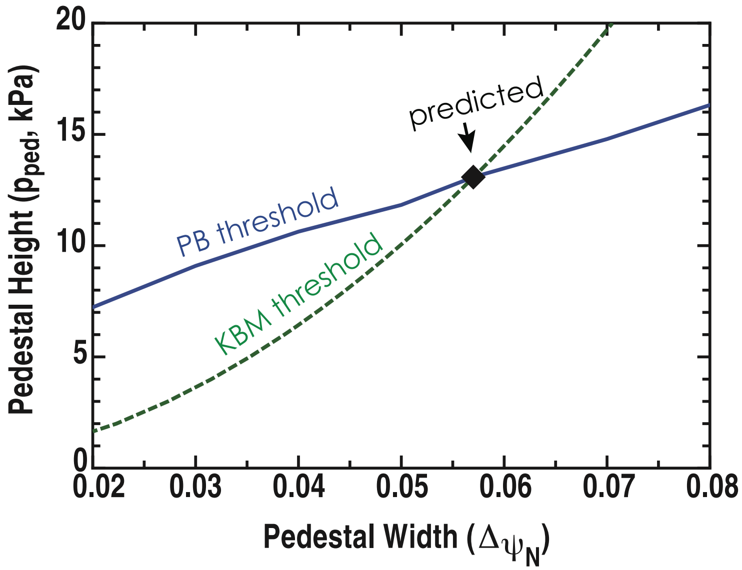


Based on ideal MHD, inapplicable for the plasma edge, full of massaging of plasma profiles, the "peeling-ballooning" model heavily relies on huge value of bootstrap current, which is solely a product of fantasy of gyro-kinetics and its GIGO codes.

3.4 “Explanation” of Height/Width of the pedestal

“Improved Understanding of Physics Processes in Pedestal Structure, Leading to Improved Predictive Capability for ITER” (author+61 contributors) -IAEA-24 FEC

EPED Model Combines PB and KBM Constraints to Predict Maximum Achievable Height and Width



Combines models for bootstrap current, PB stability, KBM stability

Inputs: B_T, I_p, R, a, κ, δ, m_i, n_{ped}, β_{global}

Outputs: Pedestal height and width
(no free or fit parameters)

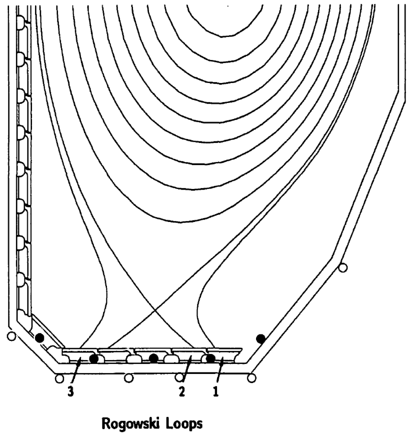
6 DIII-D NATIONAL FUSION FACILITY SAN DIEGO

RJ Groebner/IAEA/October 2012

Exceptionally beautiful fantasy of gyro-kinetics and core transport about plasma cooperation:
shear stabilization of electro-static turbulence provides thermal insulation, while invisible electro-magnetic ballooning modes gently adjust the shape of the pedestal.

4 “Halo” current based interpretation of VDE

E.J. STRAIT, L.L. LAO, J.L. LUXON, E.E. REIS. “Observation of poloidal current flow to the vacuum vessel wall during vertical instabilities in the DIII-D tokamak”, Nucl. Fusion v. 31 p. 527 (1991)



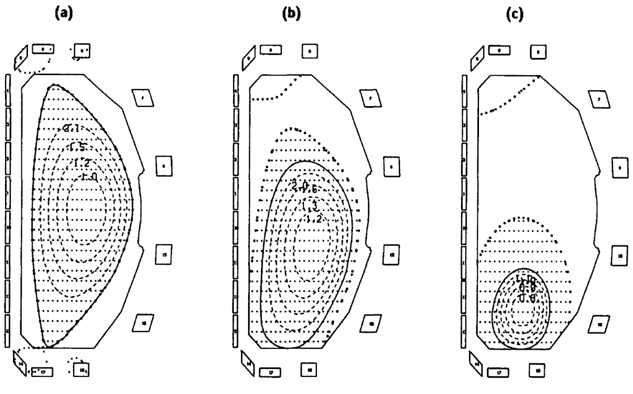


FIG. 3. Equilibrium flux plots from EFIT at three times during the vertical instability: (a) 2660 ms, (b) 2675 ms and (c) 2684 ms. Plasma current was allowed in the hatched region, including part of the SOL.

R.S. GRANETZ, I.H. HUTCHINSON, J. SORCI, J.H. IRBY, B. LaBOMBARD, D. GWINN “DISRUPTIONS AND HALO CURRENTS IN ALCATOR C-MOD NUCLEAR FUSION, Vol. 36, No. 5 (1996)

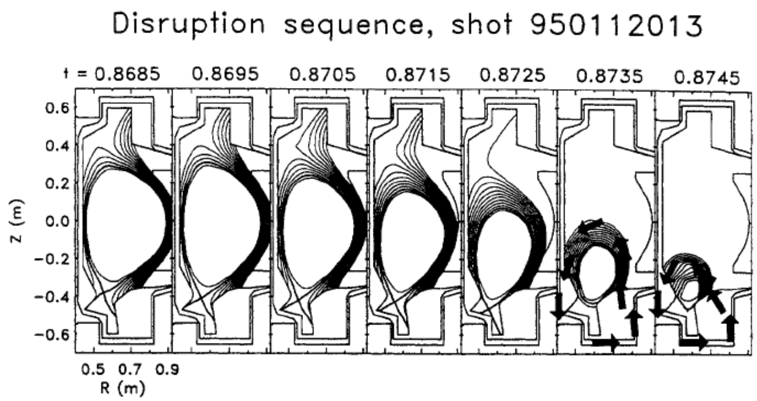


FIG. 3. Magnetic flux reconstructions at 1 ms intervals for the disruption shown in Fig. 1(a). The arrows show the poloidal projection of halo current flow, which exists primarily during the last two frames. In order to be force-free, the portion of the halo circuit in the plasma scrape-off must actually follow a helical path.

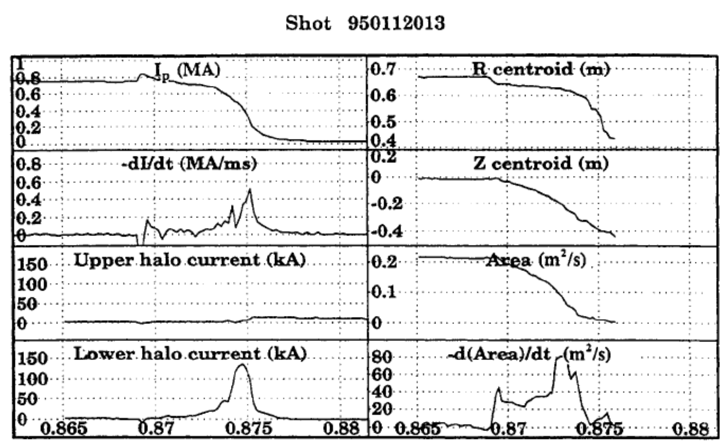


FIG. 2. Evolution of the plasma current, halo currents, position and cross-sectional area during the thermal quench disruption shown in Fig. 1(a).

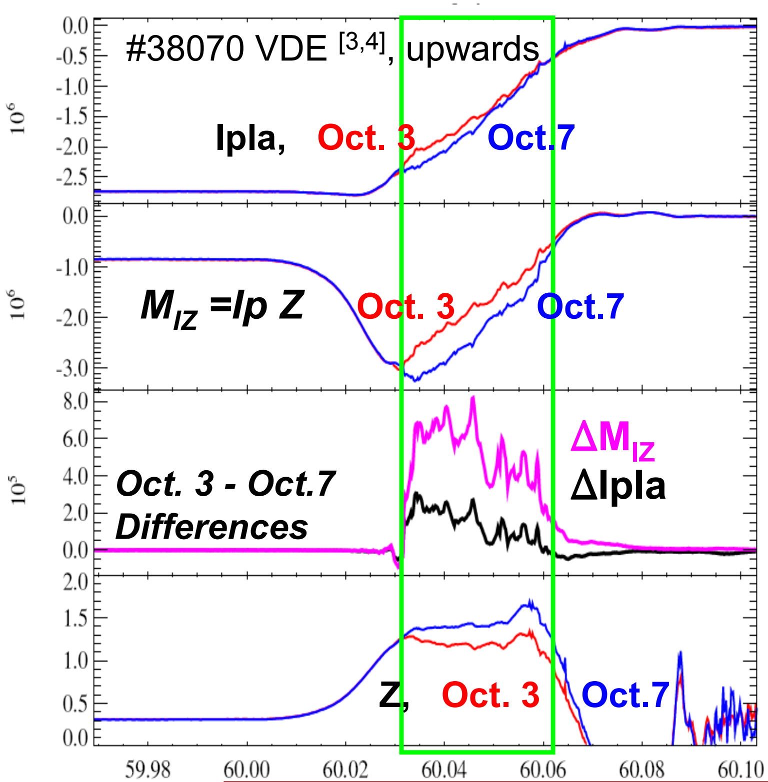
Relying on magnetic reconstruction, otherwise baseless interpretation seems to be un-ambiguous and is widely accepted. It required only time to be challenged.

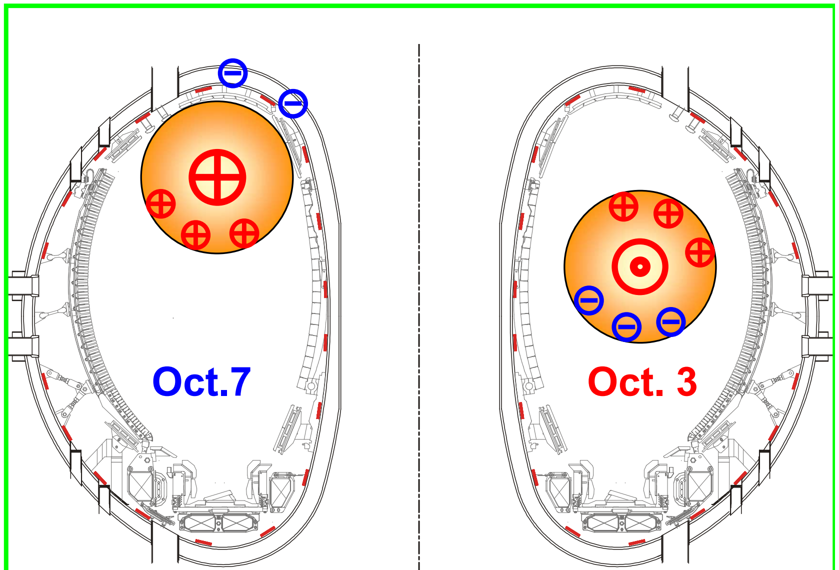
4.1 VDEs on JET. Breakthrough in understanding

Community adopted halo current explanation have been ruled out unambiguously

EFDA JET

Vessel current during VDE, #38070





- In octant 7 the plasma is closer to top of the vessel than in octant 3.
- **The current from plasma flows on vessel in octant 7.**

The measured Ipla in octant 7 is higher then in octant 3 → the missing vessel current in octant 7 is OPPOSITE to Ipla!

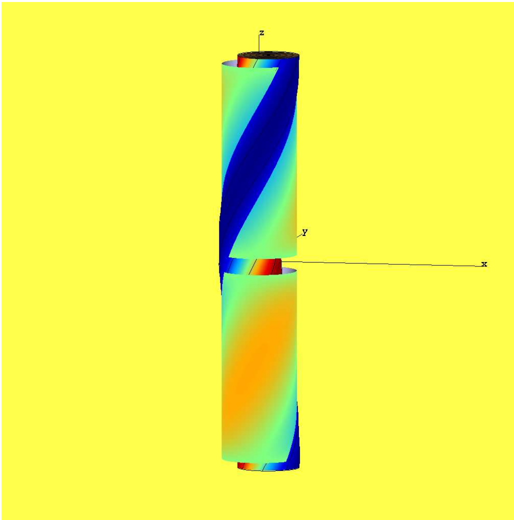
The "halo" current based interpretation predicts the opposite sign of asymmetry in the current measurement and contradicts JET Ipla's.

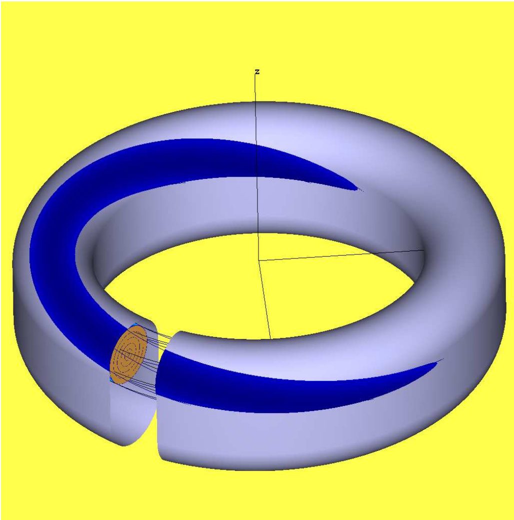
UKAEA

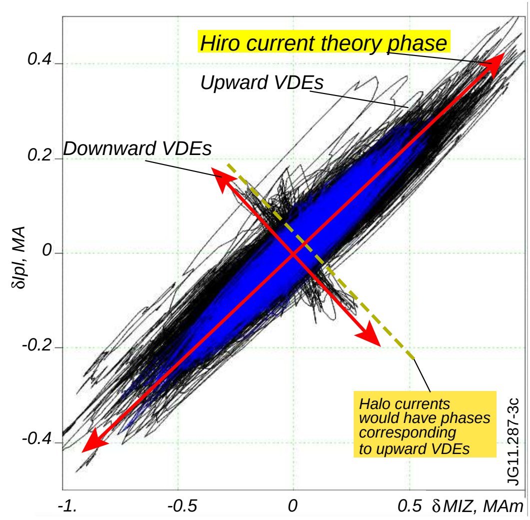
S N Gerasimov et al, Scaling JET Disruption Data to ITER. W70 7/10/09 8/24

Hiro currents explain toroidal asymmetry in Ipl

Toroidal asymmetry in the plasma current measurements during VDE on JET was explained in 2007 by the theory of the Wall Touching Kink Mode. Its Hiro currents are responsible for asymmetry. The halo currents would lead to the opposit sign of the effect.







As a side result, the "salt-water" boundary condition $V_{normal} = 0$, irrelevant to the tokamak plasma, was revealed in all 3-D MHD codes.

The theory of WTKM multiplies by zero the applicability of the 3-D MHD codes (M3D, NIMROD) for disruption simulations.

Since 2007 the boundary condition remains uncorrected. The inability to correct it does not stop very capable cooks to fool the IO with disruption simulations based on the "salt-water" MHD.

Two PPPL Theory Dept. reports (2011-2012, 11 authors) have been fabricated in order to hide the failure on multi-M$ M3D, NIMROD, TSC.

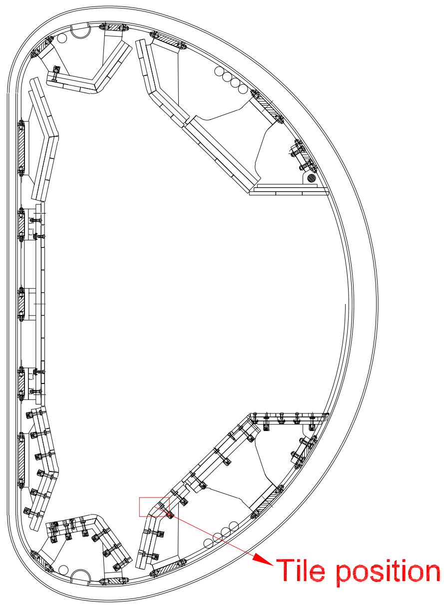
1. ***"The disruption simulations that were considered in the answers to the two questions are the TSC[1] [2], DINA [3], and M3D [4] [5] simulations."***
2. ***"The boundary conditions used in the TSC, DINA, and M3D simulations are appropriate for obtaining an estimate of the maximum of the total force exerted on the wall by the halo current under certain approximations, such as axisymmetry in the TSC and DINA codes, by varying assumed values for the resistance and width of the halo."***
3. ***"The assumption in existing simulations that the plasma cannot flow into the wall, $v_n = 0$, is un-physical. . . . the impact of this boundary condition on current simulations is limited to essentially the inertia of the halo plasma, which is NEGLIGIBLE in the overall simulation."***

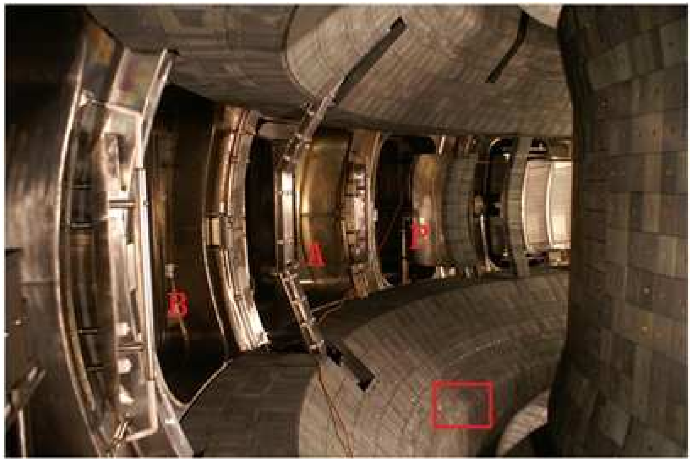
Every New Jersey plumber knows a big difference between water flow in a good and raptured pipes. They have much better knowledge of the value of boundary conditions than the crowd of PPPL experts in MHD and disruptions.

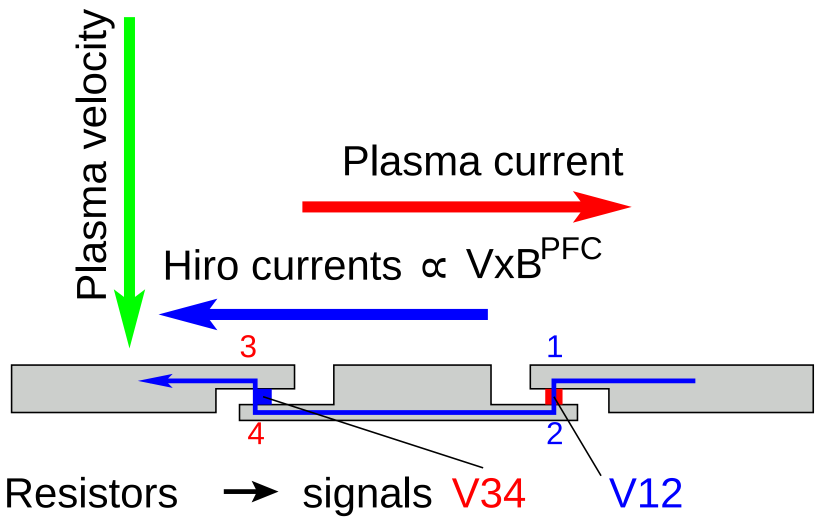
For 11 certified experts in MHD from PPPL and IO (led by A.Boozer and M.Bell) this difference is beyond the understanding. The political objectives of this reports, coming from the Director Office of PPPL, have eliminated the sense of scientific ethics.

4.3 Xiong tiles on EAST - New diagnostics for VDE

Hiro currents in VDE were measured on EAST in ASIPP (Hefei, China) using specially designed tiles

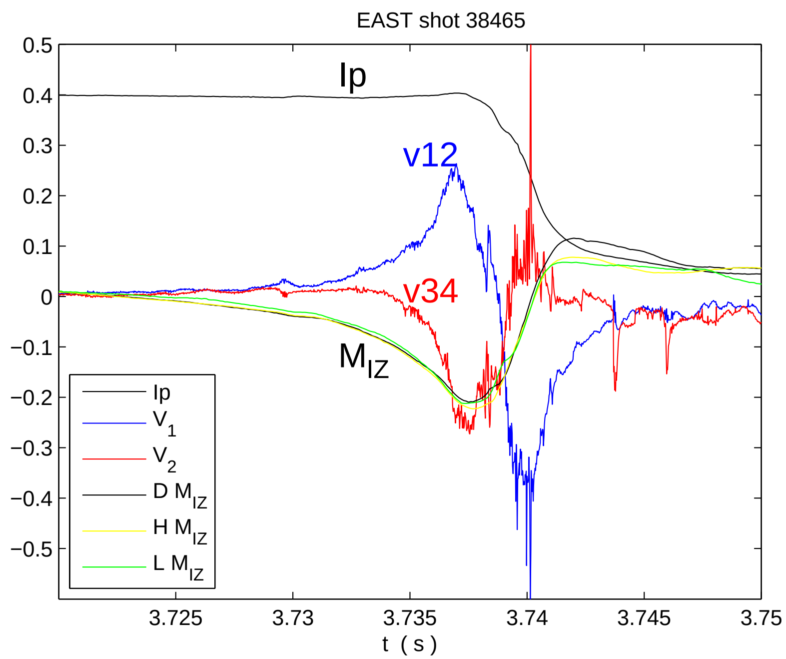




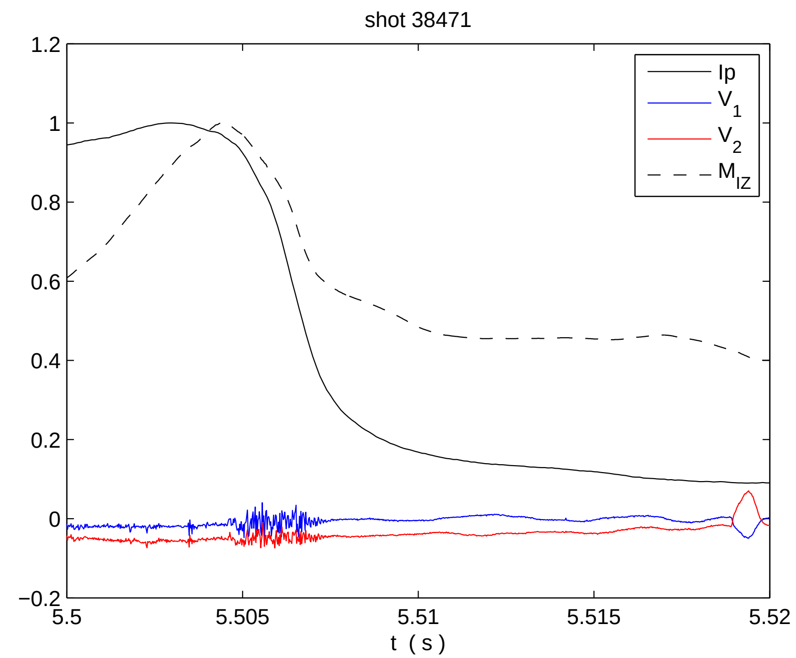


First measurements of Hiro currents in VDE

Toroidal currents, opposite to the plasma current, predicted by theory (L.Zakharov) and for 2 decades being overlooked in interpretations and simulations of Vertical Disruptions, were measured on EAST in May 2012 (H.Xiong)



Downward VDE



Upward VDE

No toroidal asymmetry.

Hiro currents in VDE are NOT SHARED between plasma and the tiles.

Despite all community attempts, Hiro currents cannot be to confused with "halo" currents.

Boozer's cooking the "halo" currents from the Hiro currents failed miserably

4.4 The attack on science blessed by PPPL Director

Page 2 from S.Jardin presentation to the ITPA-MHD Meeting (Padova October 4-7, 2011)

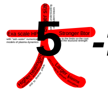
In 2010, a single scientist in the U.S. fusion community was repeatedly making the following claim (and being quite vocal about it)

"…….the present numerical codes (M3D, NIMROD) are not capable of simulating disruptions because of their "salt-water" boundary condition Vnorm = 0, irrelevant to tokamak plasma. For almost 4 years this boundary condition was not corrected. In fact, it represents a fundamental flaw of numerical scheme, making it not suitable for plasma dynamics in tokamaks."

This claim was not backed-up by any mathematical, physical, numerical, or experimental analysis, but arose primarily because the code's results did not support that scientist's theory of disruptions.

In fact, mathematics and physics (LZ 1979, 1981, 2008, 2010, 2012 papers), numerical simulations (DSC) and experimental measurements (entire JET data base, EAST) prove the GIGO nature of not only 3-D M3D but of 2-D TSC as well.

FES state of the art in macroscopic stability

***The success of theory and science based experiments on EAST revealed the full depth of the failure of 

-fusion in addressing the stability problem***

ITER at this moment has neither physics based model nor valid numerical codes even for simplest 2-D vertical instability.

3-D simulation codes, after 3.5 decades of development are worthless due to their "salt water" boundary condition $V_{normal} = 0$ for the plasma velocity to the wall, which is uncorrected since 2007.

This is a disaster in the key problem due to incompetence of magnetic fusion establishment and management, which is always ready to attack the science, while protecting and promoting the outdated activities.

Physics of Hiro, Evans and halo (if any) currents

The basic understanding of mechanisms driving disruption currents to the wall was created. Further development required.

The physics of Hiro and Evans currents is different from the "physics" of halo currents and summarized in the Table.

	Hiro currents	Evans currents:	Halo currents:
1	*Both result from magnetic flux conservation.*		*Derived from questionable use of equilibrium reconstruction. No strong reason for existence.*
2	*Driven by instability acting as current generator.*	*Driven by instability acting as voltage generator.*	*Assumed to be driven by a residual voltage outside the last closed magnetic surface.*
3	*Highly concentrated at the plasma edge.*		*Diffused in space with open field lines.*
4	*Big in amplitude, proportional to plasma deformation.*		*Limited by the ion saturation current.*
5	*Absolutely necessary to slow down the instability.*	*Force-free, little, if any, effect on stabilization.*	*Secondary, if any, effect on stabilization.*
6	*Opposite to I_{pl}.*	*Same direction as I_{pl}.*	*Same direction as I_{pl}.*
7	*Consistent with toroidal asymmetry in JET VDEs.*		*Ruled out as a reason of toroidal asymmetry.*
8	*The real plasma physics objects*		*Most probably the result of misinterpretation*
		May 2012	
9	*Consistent with EAST VDE measurements.*	*No indication of presence*	*No indication of presence*

5 Some other “achievements” of FES

After TFTR, magnetic fusion has lost its status as a potential near term energy source.

- ***Because of very a bad energy confinement time the 5 FES fusion has no solution of the power extraction problem.***
- ***Fundamentally non-stationary plasma-wall interaction are incompatible with the stationary plasma.***
- ***High edge plasma density and low temperature prevents efficient coupling with the current drive systems.***
- ***There is no basic concept of fueling the large size dense plasma.***
- ***The practical concept of the burning plasma is absent (in fact impossible)***
- ***The realistic concept of DEMO is absent (and impossible)***

The worse possible budget situation has been created:

(a) about 50 % of a huge $0.4B budget represents taxation to ITER, which negative impact on fusion is obvious but the positive value is highly questionable;
(b) the domestic fusion is fragmented and has no priorities allowing to recover from the lost credibility.

6 LiWall Fusion (LiWF) as a contrast to FES

Utilization of the pumping abilities of liquid lithium (LiLi) is the simplest implementation of the "pumping walls" and diffusion based confinement regime.

The predicted tendencies toward much better confinement have been confirmed even at the level of Li conditioning experiments:

CDX-U quadrupled τ_E (2003-2005);
NSTX enhanced τ_E by 50 %, broadened T_e (2006-2011);
EAST achieved H-mode exclusively due to Li (2010).

Earlier in the mid 1990s, Li conditioning rescued TFTR: all its supershots were obtained with Li conditioning.

The LiWall Fusion (LiWF) with use of flowing LiLi (FLiLi) goes far beyond simple conditioning.

Its diffusion based confinement regime allows for the use of NBI for plasma fueling.

The existing and growing power extraction problem from the tokamak plasma cannot be resolved by solely material developments.

The key to its solution is in enhanced confinement suggested by the LiWF.

6.1 Key aspects of LiWF

- ***The best stability: no sawteeth (LZ), no ELMs (LZ, NSTX), no Greenwald density limit (LZ, FTU);***
- ***The best plasma edge for the current drive: low edge density, high edge temperature;***
- ***The best stationary plasma-wall interactions: no thermal force in SoL (which drives impurities to the plasma), no dust, no wall/target plates erosion;***
- ***The simplest physics of the tokamak plasma: temperature is determined by the NBI energy, density is determined by NBI current and diffusion coefficients.***
- ***Predictive Reference Transport Model (RTM)***

$$\Gamma = -\chi_i^{neo} \nabla n_e, \quad q_i = -n_e \chi_i^{neo} \nabla T_i, \quad q_e = -f n_e \chi_i^{neo} \nabla T_e \tag{6.1}$$

for a LiWF regime, which is insensitive to the thermal conduction.

In addition:

- ***Use of NBI for plasma fueling by tritium;***
- ***Use of the entire plasma cross-section for fusion production;***
- ***The best possible recycling of tritium used for plasma fueling;***
- ***Self-consistent practical concept of the burning plasma;***
- ***Clean objectives and a practical concept of the fusion DEMO ($Q_{DT}^{electric} > 1$).***

6.2 FLiLi - solution of the Li technology problems

In July 2011 a practical concept of FLiLi system was invented.

There is no other system known (CPS, LIMIT, power, pellets) to satisfy the requirements.

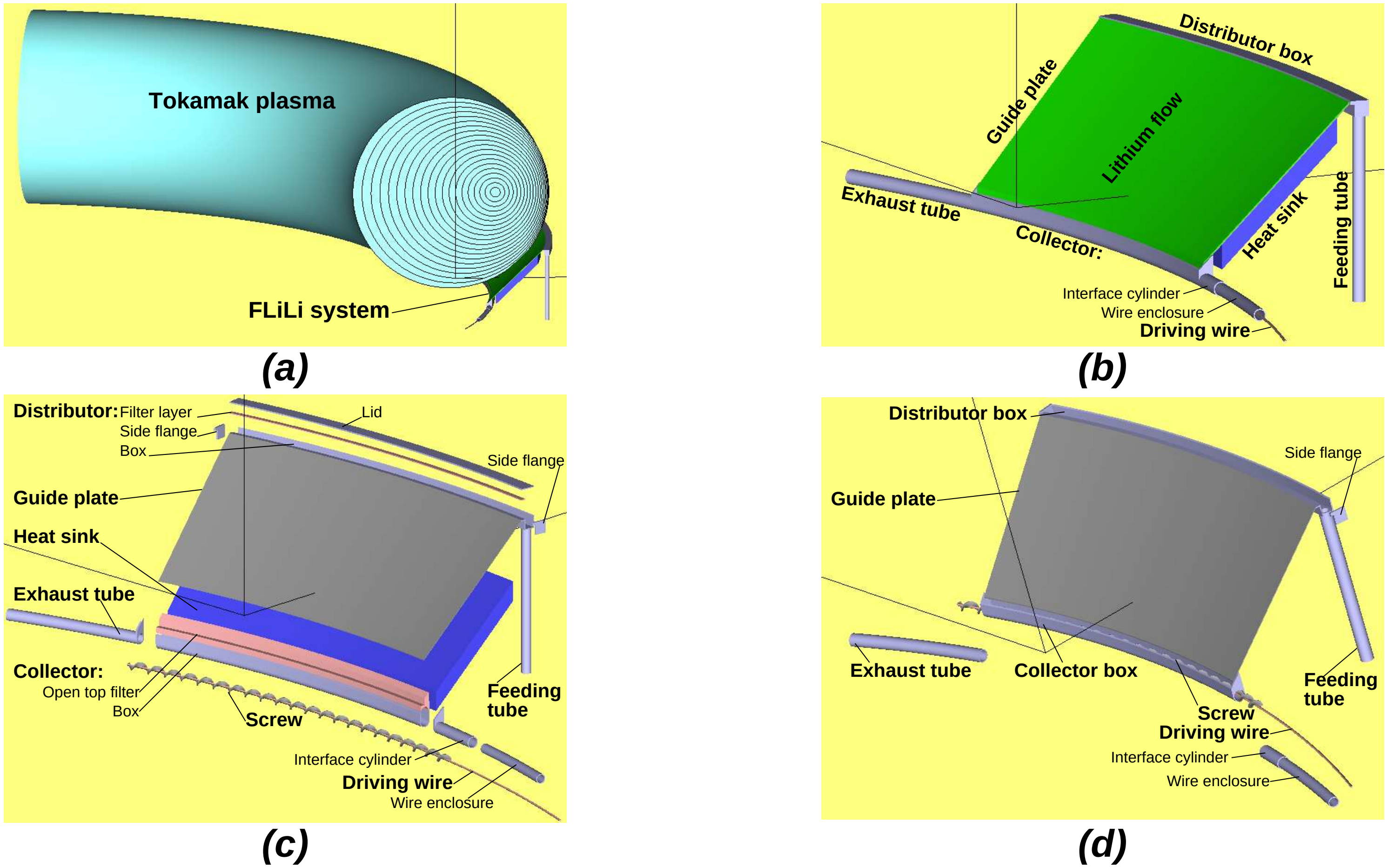
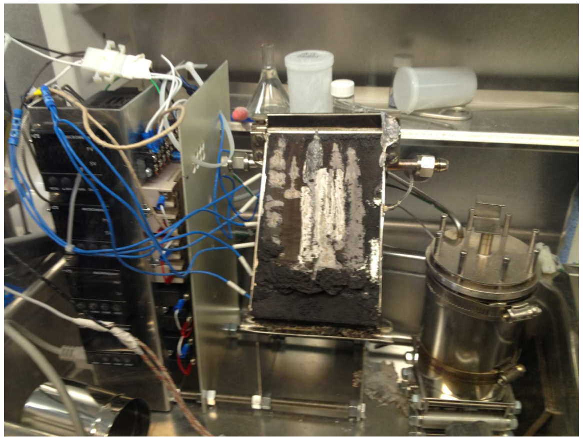


Fig.5 Conceptual design of the FLiLi system. (a) Example of FLiLi system as a limiter for a tokamak with a circular cross section (e.g., HT-7). (b) Assembly of distributor, feeding pipe, guide plate with LiLi flow (green), heat sink, collector and exhaust mechanism. (c) Separate parts of FLiLi system. (d) guide plated with open boxes of distributor and collector with a flexible screw lithium propelling mechanism.

Glove box based prototype of FLiLi (April, 2012)

FLiLi prototype in PPPL



Failure with wetting SS by LiLi

The goal is to get a contiguous LiLi flow:

$$h_{LiLi} = 0.1 \text{ mm}, \quad V_{LiLi} \simeq 1 \text{ cm/s}. \tag{6.2}$$

The flow rate, required for the existing and future devices (for plasma pumping out), is very small

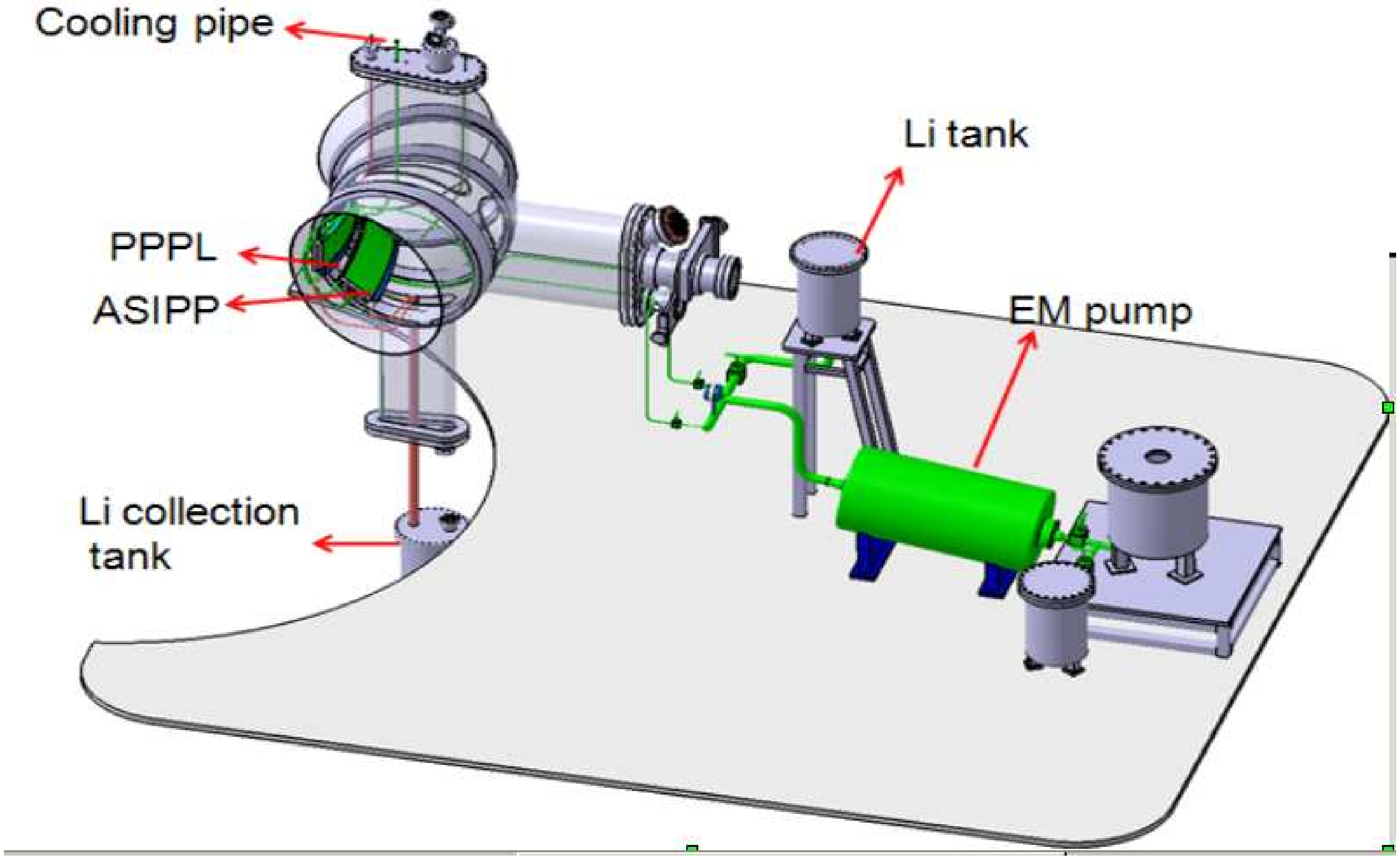
$$\Gamma = 2 \text{ cm}^3/\text{s}, \tag{6.3}$$

thus, making the stationary FLiLi system designable.

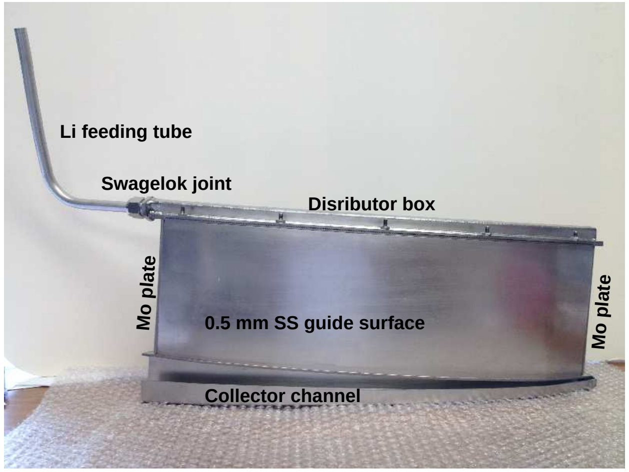
Imperfect Ar atmosphere in the glove box did not allow to wet SS guide plate and to experiment with flowing lithium.

Instead, it gave the experience sufficient to design, manufacture and a FLiLi limiter and install it in HT-7 tokamak (ASIPP, Hefei)

Two (ASIPP, PPPL) FLiLi limiters on HT-7 (August, 2012)



Operational FLiLi limiter for HT-7 (October 4, 2012)

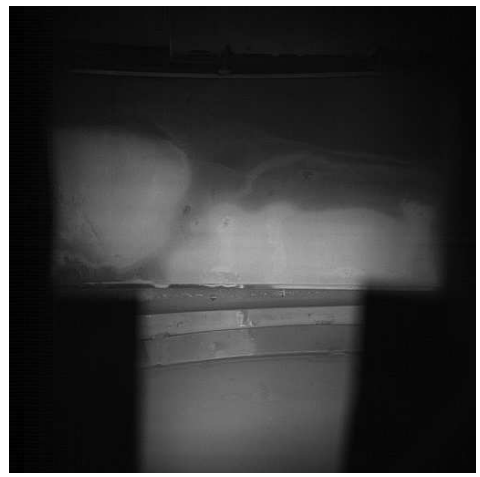


PPPL July 29, 2012

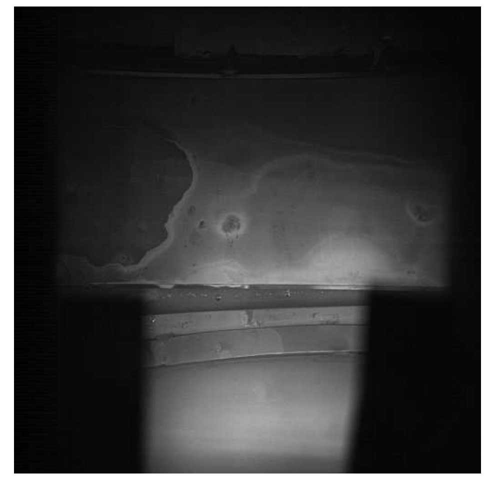
HT-7 ASIPP, August 19, 2012

$P_{Ar} = 15$ **kPa**

$P_{Ar} = 15$ **kPa**



$P_{Ar} = 25$ **kPa**

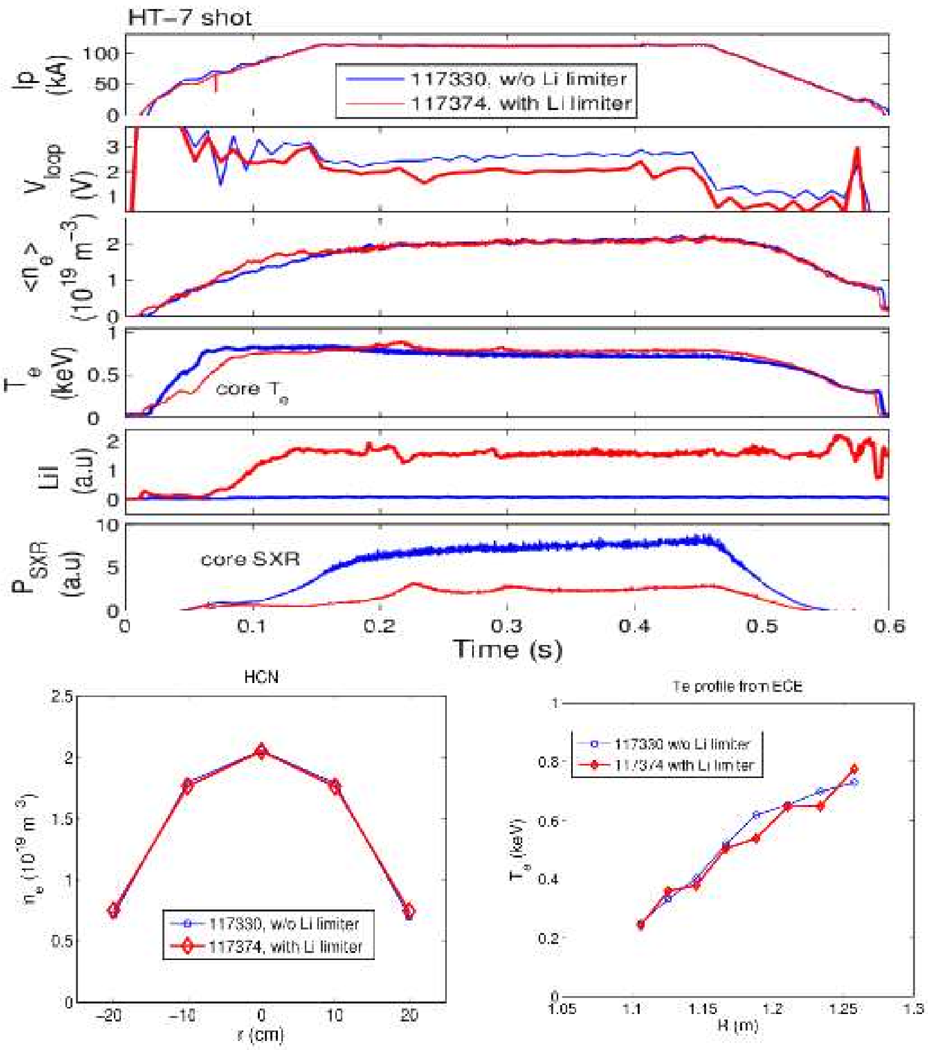


$P_{Ar} = 40$ **kPa**

Full success with two (PPPL, ASIPP) FLiLi limiters on HT-7, October 4, 2012 (55 years after Sputnik launch on Oct. 4, 1957)

FLiLi is consistent with a tokamak plasma

Comparison w/o and with ASIPP Li limiter



- Confinement improves about 25% due to the decrease of radiation.
- Similar results of PPPL flowing Li limiter experiment

Properties of FLiLi system is simply amazing

In terms of physics:

- ***Scalable in both poloidal and toroidal directions;***
- ***No flow interaction with the tokamak magnetic field (all Hartmann numbers < 1);***
- ***Scalable from a workbench to tokamaks;***
- ***Reliable control of flow rate by the external pressure at the level of fraction of atm;***
- ***Insensitive to the SoL currents at the level of the ion saturation current;***

The physics of FLiLi is clean, simple, easy to analyze and works as theory predicts.

In terms of technology:

- ***The smallest possible flow rate;***
- ***No high pressure, no mechanical or EM Li pumps;***
- ***Separation of the plasma pumping and the power extraction;***

The transition to a stationary FLiLi is straightforward. For the first time, FLiLi resolves the problem of contamination of the Li surface by outgasing from the walls.

In terms of safety:

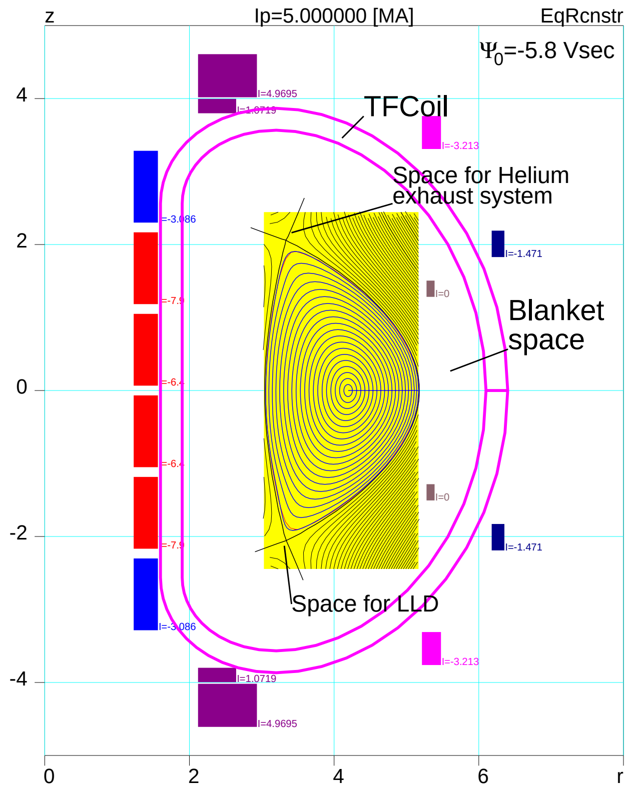
The inventory of Li with an exposed free surface is minimal (< 0.5 L). It has no ability to create a detonation or ignition level of hydrogen in the vacuum vessel.

Inconsistency of FLiLi with seawater energy of FES

FLiLi as a LDRD project survived only a half year. In Sept. 2012, the Director of PPPL terminated its extension. At the same time, a very "productive" activity (see the picture) with no idea what kind of Li is needed for tokamaks was fully funded and extended.

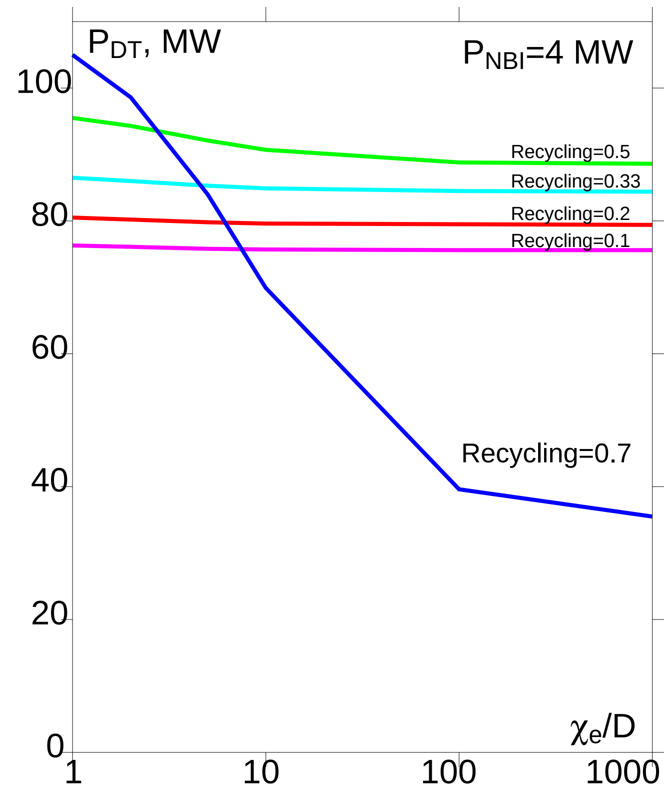
Empty J.Timberlake's room (with his stuff) full of (imperfect) vacuum. The vacuum chamber itself is absent after 1 year of heavily funded R.G. and M.J. LDRD proposal.

Fusion-Fission Research Facility (FFRF) as a potential next step device for China



Parameter	FFRF
$d_{blanket,m}$	**1**
a_m, R_m	**1.0, 4.0**
$V^{pl}_{m^3}, S^{pl}_{m^2}$	**130, 230**
n_{20}	**0.4**
E^{NBI}_{keV}	**120**
$\frac{T_i+T_e}{2}\|_{keV}$	**24-27**
$B_{t,T}$	**4-6**
$I_{pl,MA}$	**5**
$\Delta\Psi_{f-top,Vsec}$	**40**
$\Delta t^{inductive}_{f-top,s}$	$>$**4000**
$W_{th,MJ}$	**42**
$\tau^{ind}_{E,sec}$	**20-7**
P^{NBI}_{MW}	**2-5**
P^{DT}_{MW}	**50-100**

Active fission core power 80-4000 MW.



P^{DT} ***vs electron anomaly***

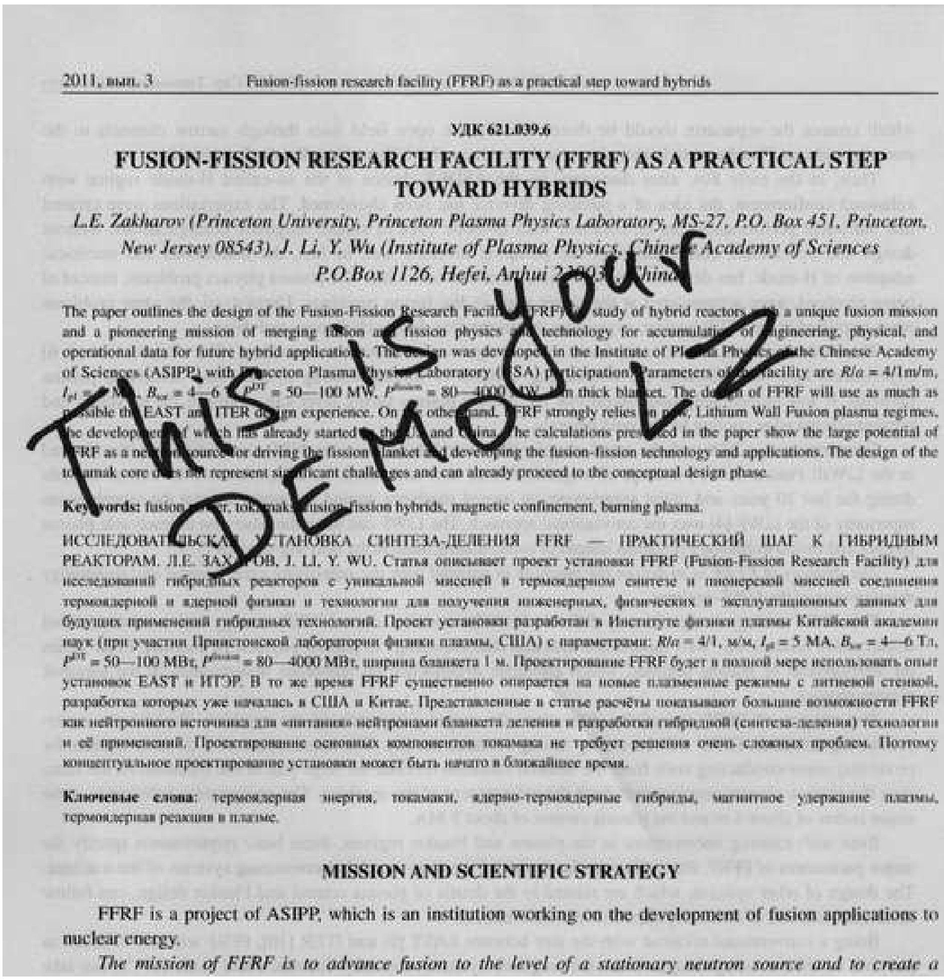
At the practical level of $R^{ecycl} < 0.5$, the burning plasma regime with $P^{DT} = 50 - 100$ MW is possible in FFRF

Remarkably, a robust "hot-ion" regime was found (thanks to G.Hammett) where the cyclotron radiation keeps $T_e < T_i$ even with the α-particle heating.

What is the fusion DEMO ?

In conventional fusion there is no a valuable DEMO concept.

The 100-200 MW FFRF of the LiWF with its innovative burning plasma regime is the first realistic model of DEMO. It has both fusion and fusion-fission missions



2011, вып. 3 Fusion-fission research facility (FFRF) as a practical step toward hybrids

УДК 621.039.6

FUSION-FISSION RESEARCH FACILITY (FFRF) AS A PRACTICAL STEP TOWARD HYBRIDS

L.E. Zakharov (Princeton University, Princeton Plasma Physics Laboratory, MS-27, P.O. Box 451, Princeton, New Jersey 08543), J. Li, Y. Wu (Institute of Plasma Physics, Chinese Academy of Sciences P.O.Box 1126, Hefei, Anhui 230031, China)

The paper outlines the design of the Fusion-Fission Research Facility (FFRF) for study of hybrid reactors with a unique fusion mission and a pioneering mission of merging fusion and fission physics and technology for accumulation of engineering, physical, and operational data for future hybrid applications. The design was developed in the Institute of Plasma Physics of the Chinese Academy of Sciences (ASIPP) with Princeton Plasma Physics Laboratory (USA) participation. Parameters of the facility are $R/a = 4/1$m/m, $I_{pl} = 5$ MA, $B_{tor} = 4$—6 T, $P^{DT} = 50$—100 MW, $P^{fission} = 80$—4000 MW, 1 m thick blanket. The design of FFRF will use as much as possible the EAST and ITER design experience. On the other hand, FFRF strongly relies on new, Lithium Wall Fusion plasma regimes, the development of which has already started in the US and China. The calculations presented in the paper show the large potential of FFRF as a neutron source for driving the fission blanket and developing the fusion-fission technology and applications. The design of the tokamak core does not represent significant challenges and can already proceed to the conceptual design phase.

Keywords: fusion power, tokamak, fusion-fission hybrids, magnetic confinement, burning plasma.

ИССЛЕДОВАТЕЛЬСКАЯ УСТАНОВКА СИНТЕЗА-ДЕЛЕНИЯ FFRF — ПРАКТИЧЕСКИЙ ШАГ К ГИБРИДНЫМ РЕАКТОРАМ. Л.Е. ЗАХАРОВ, J. LI, Y. WU. Статья описывает проект установки FFRF (Fusion-Fission Research Facility) для исследований гибридных реакторов с уникальной миссией в термоядерном синтезе и пионерской миссией соединения термоядерной и ядерной физики и технологии для получения инженерных, физических и эксплуатационных данных для будущих применений гибридных технологий. Проект установки разработан в Институте физики плазмы Китайской академии наук (при участии Принстонской лаборатории физики плазмы, США) с параметрами: $R/a = 4/1$, м/м, $I_{pl} = 5$ MA, $B_{tor} = 4$—6 Тл, $P^{DT} = 50$—100 МВт, $P^{fission} = 80$—4000 МВт, ширина бланкета 1 м. Проектирование FFRF будет в полной мере использовать опыт установок EAST и ИТЭР. В то же время FFRF существенно опирается на новые плазменные режимы с литиевой стенкой, разработка которых уже началась в США и Китае. Представленные в статье расчёты показывают большие возможности FFRF как нейтронного источника для «питания» нейтронами бланкета деления и разработки гибридной (синтеза-деления) технологии и её применений. Проектирование основных компонентов токамака не требует решения очень сложных проблем. Поэтому концептуальное проектирование установки может быть начато в ближайшее время.

Ключевые слова: термоядерная энергия, токамаки, ядерно-термоядерные гибриды, магнитное удержание плазмы, термоядерная реакция в плазме.

MISSION AND SCIENTIFIC STRATEGY

FFRF is a project of ASIPP, which is an institution working on the development of fusion applications to nuclear energy.

The mission of FFRF is to advance fusion to the level of a stationary neutron source and to create a

This is your DEMO-2

On the left is my recommendation to Jiangang Li on the concept for the next-step (two) DEMO devices in China

Two similar devices, DEMO-D (no tritium) and DEMO-T (with DT power) are necessary, in order to assure the success and resolution of potential operational problems in activated DEMO-T.

8 Summary

Except of great contribution of DIII-D, FES failed in all essential aspects of magnetic fusion science: confinement, macroscopic and edge plasma stability, plasma edge, power extraction, fueling, numerical simulations.

The science based concept of burning plasma as well as a vision for fusion power are simply absent in FES. The programs is kept hostage of incompetence of the current fusion establishment.

The scientific credibility of magnetic fusion was undermined. In fact, the physics idea of magnetic fusion itself was put at the risk of extinction. With the tungsten divertor at the H-phase, ITER is going to complete the job in this regard sooner than later.

LiWF returns magnetic fusion to its basic concept, i.e., insulation of the plasma from the wall. (FES is doing just the opposite).

The only visible chance for magnetic fusion to survive could be

(a) to develop the LiWF regime on EAST within 2-3 years

(b) to convince JET to perform DT experiments in this regime and obtain $Q_{DT} > 5$

(c) to initiate a mutual US-China 100-200 MW DEMO program aiming to the DEMO-T (including a U-blanket) in China and the complementary DEMO-D in the US.