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CIMPA/ICTP Geometric Structures and Theory of Control

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Progress and Outstanding Challenges in Tokamak Research

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Topics and Preamble

- Tokamak fundamentals
- Tokamak stability
- Confinement and transport
- DT experiments in TFTR and JET
- The leap to ITER

Disclaimer:

- A single lecture cannot encompass all areas of tokamak physics
  - Tokamaks have been intensely studied for almost 50 years
- Even within the subset of topics, I have had to be very selective

Acknowledgements:

R. Fonck, T. Luce, G. Matthews, J. Menard, H. Qin, J. VanDam

Essential Features of the Tokamak

- Toroidal configuration *symmetric about its major axis* formed by a strong applied *toroidal field* plus a *poloidal field* generated by both toroidal plasma current and external coil currents
  - The poloidal field is necessary for compensating particle drifts
  - The toroidal field is necessary for plasma stability
- The configuration need not be symmetric poloidally
  - External poloidal field coils can modify the shape of the minor cross-section
- A tokamak plasma can be described by many different coordinate systems
  - relative to the *major axis* (R, Z, ϕ)
  - relative to the *minor axis* (r, θ, ϕ)
  - various magnetic coordinates which can simplify calculations

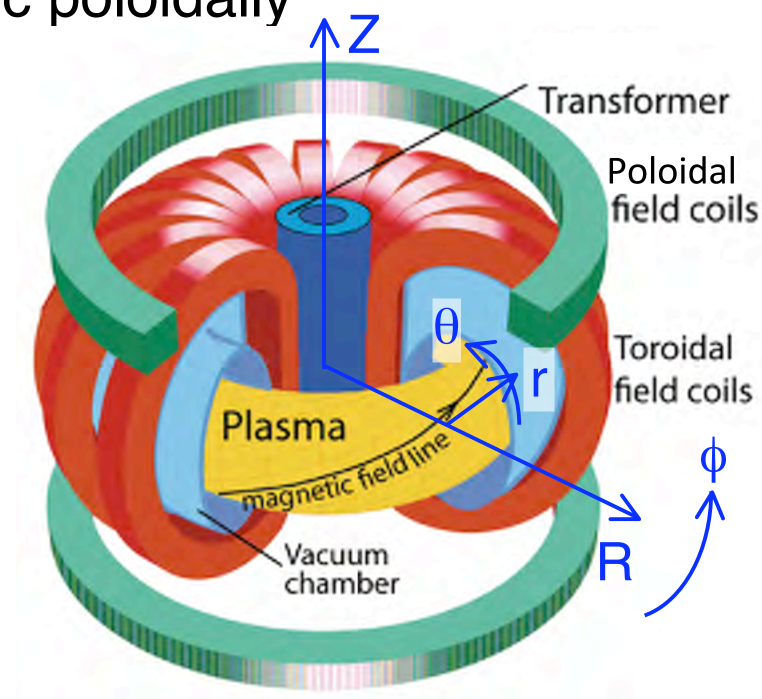


Figure courtesy T. Luce

Tokamak MHD safety factor *q*

- ***q*** = *number of toroidal transits of a field line around the major axis to complete one poloidal transit around the minor axis*
- In a stable tokamak plasma, magnetic field lines trace out nested *flux surfaces* each characterized by a **poloidal flux** ψ ($\propto I_p$), an enclosed **toroidal flux** χ ($\propto B_T$) and value of *q*

$$q = \frac{d\chi}{d\psi} \qquad q_{edge} \propto \frac{RB_T}{\mu_0 I_p} f(a/R, \text{boundary shape, profiles})$$

- For magnetohydrodynamic (MHD) stability *q* must be > 1 everywhere
- This places an upper bound on the plasma current for a given toroidal field, plasma size and cross-section shape
- A tokamak plasma has a *"last closed flux surface"* beyond which the field lines intersect some material surface
- In practice, *q* must be >~2 near the last closed flux surface

Tokamak MHD Equilibrium

- On timescales > Alfvén timescale a/v_A, $v_A = B/(\mu_0\rho)^{1/2}$, equilibrium is determined by static pressure balance

$$\mathbf{J} \times \mathbf{B} = \nabla p$$

- In a tokamak, axisymmetry reduces this to the 2D "**Grad-Shafranov**" eqn.

$$\Delta^*\Psi = \mu_0 R J_\phi = -[\mu_0 R^2 dp/d\Psi + F dF/d\Psi]$$

where $\Delta^*\Psi = R^2\, \nabla\bullet\, (\nabla\Psi / R^2)$

Ψ is the poloidal flux $= RA_\phi$, (A the magnetic vector potential)

$p(\Psi)$ is the plasma pressure

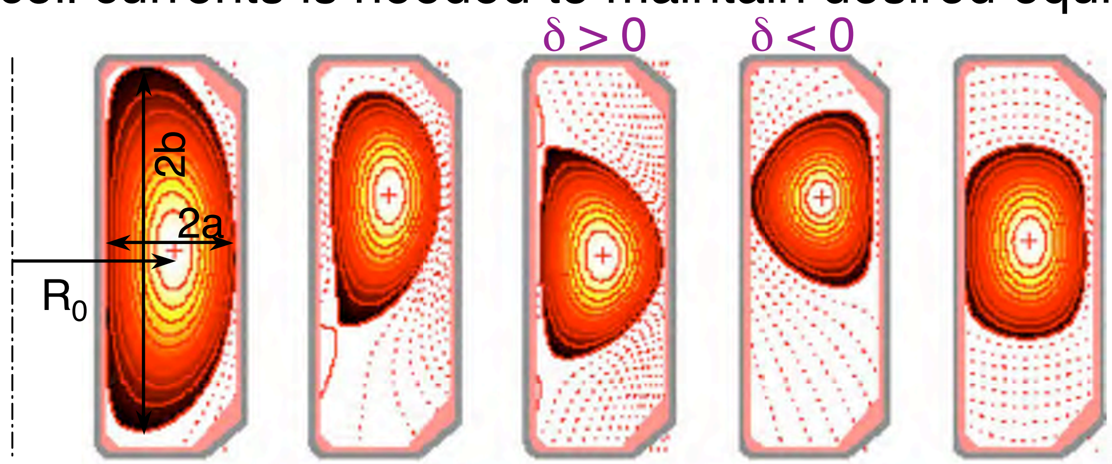
$F(\Psi)$ the poloidal current $= RB_\phi$, (B_ϕ the toroidal magnetic field)

- In principle, there is an infinite number of solutions
- In practice, the solutions are constrained by experimental data
  - Total plasma current and coil currents
  - External magnetic measurements (fluxes, field components)
  - Internal measurements of plasma pressure and magnetic field
  - Geometry of surrounding structures
- Important MHD parameter: β = plasma pressure/magnetic pressure ($B^2/2\mu_0$)

Controlling and Shaping the Plasma Cross-section in a Tokamak

- A tokamak requires a major axial (usually vertical) magnetic field to resist major radial expansion forces on the plasma
  - Electromagnetic: a current loop tries to maximize its area
  - Hydrodynamic: plasma pressure tries to expand the torus
- A uniform axial field produces a nearly circular cross-section
- In modern tokamaks, the equilibrium field is generated by many nearby coils to push and pull on the plasma and shape its cross-section
  - **Aspect ratio**: R_0/a (R_0: major radius of toroidal axis, a: minor radius on R)
  - **Elongation** κ: axial height / width = b/a
  - **Triangularity** δ: (inward) displacement of top, bottom points from axis
- Feedback control of coil currents is needed to maintain desired equilibrium

Equilibrium control in the TCV tokamak (EPFL, Lausanne)



Creating a Magnetic Separatrix to Produce a Divertor in a Tokamak

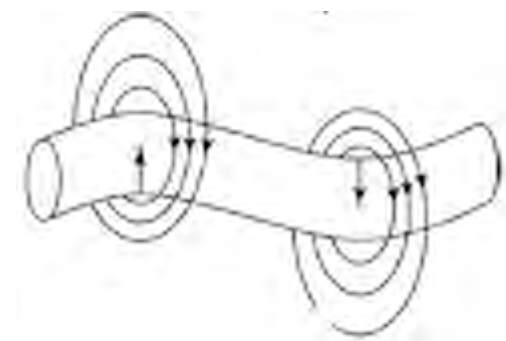
- Between two parallel conductors carrying current in the same direction there is a magnetic null point: $B_\perp = 0$
- Through the null there is a surface (**separatrix**) which separates flux surfaces which encircle only one conductor from those that encircle both
  - the null point forms an **X-point** in a cross-section of the separatrix
- In a tokamak **divertor**, a separatrix is formed between the plasma, carrying toroidal current, and a poloidal field coil with current in the *same* direction
  - on the separatrix $q \to \infty$ because $B_{pol} \to 0$
- Particles diffusing from the plasma across this separatrix are then tied to field lines which are diverted away from the main plasma
  - these field lines are made to intersect some more distant material surface
- Divertors were originally incorporated in tokamaks to reduce the influx of impurities ejected by plasma impinging on surrounding material surfaces
  - the divertor plate can also be angled to spread the heat over a wider area
- Divertors are now used primarily because they allow easier access to the **H-mode of confinement**

Topics

- Tokamak fundamentals
- **Tokamak stability**
- Confinement and transport
- DT experiments in TFTR and JET
- The leap to ITER

Tokamak Equilibria Can Be Unstable to Many Modes

- An (axially) elongated tokamak plasma is unstable to axisymmetric major-axial displacement
  - Divertor coils strongly attract the plasma
  - Stability requires fast feedback on radial field
- A current-carrying plasma may be subject to a kink instability
  - Higher poloidal field on inside of bend reinforces initial displacement
  - In a tokamak, the strong toroidal field helps to stabilize the kink
  - A surrounding perfectly conducting wall can also stabilize the kink because poloidal field is compressed on the outside of the bend
  - A wall with finite conductivity of the wall slows growth of the instability unless the plasma is moving relative to the wall



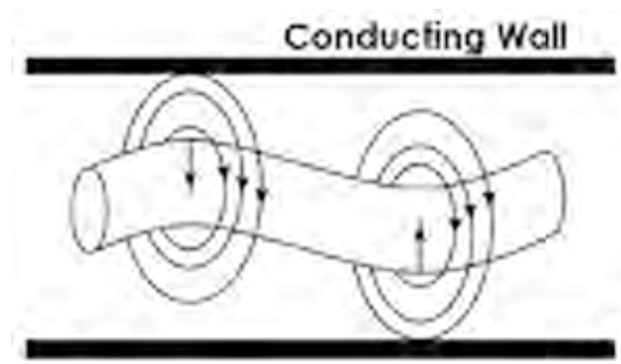


Figure courtesy T. Luce

Finite Plasma Pressure and Non-Ideal Plasma Behavior Introduce Other Instability Modes

- Can assess MHD stability by perturbing equilibrium fluid elements searching for displacement vectors ξ (perpendicular to the magnetic surfaces) which reduce the potential energy of the system

$$\xi = \xi_0 exp[i(n\phi + m\theta)]$$

where n, m are toroidal and poloidal mode numbers

  - For kink-like modes ($n < \sim 10$) need full 3D displacement
  - For high toroidal mode number/short radial wavelength, calculation reduces to ODE $\Rightarrow$ **“ballooning” modes** on low field side
  - Flux surfaces where $q = m/n$ are susceptible to instability

- In “ideal” (infinite conductivity) plasma, flux surface topology is preserved
- Finite plasma conductivity allows reconnection of field inside plasma to form **magnetic islands**
  - Radial excursion of field lines in magnetic islands “short circuits” the isolation of perfect surfaces
  - Causes radial transport and flattens profiles

Studies in 1980s Produced a Simple Criterion for Stability to Pressure-Driven Instabilities

- Across a range of tokamak shapes, theory showed

$$\langle\beta\rangle_{max} = C \cdot I_p / aB_T$$

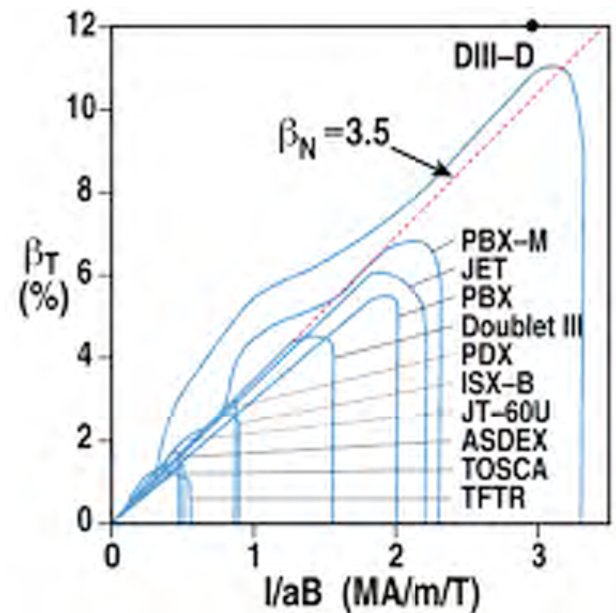
where $\langle\beta\rangle = 2\mu_0\langle p\rangle/\langle B^2\rangle$ [$\langle\rangle$ indicates volume average] and C is a constant: $C \approx 3.5$ mT/MA

- This expression was usually approximated by experimentalists as

$$\beta_{T,max} \;(= 2\mu_0\langle p\rangle/B_{T0}^2) = C \cdot I_p / aB_{T0}$$

where B_{T0} is the applied toroidal field at the minor axis

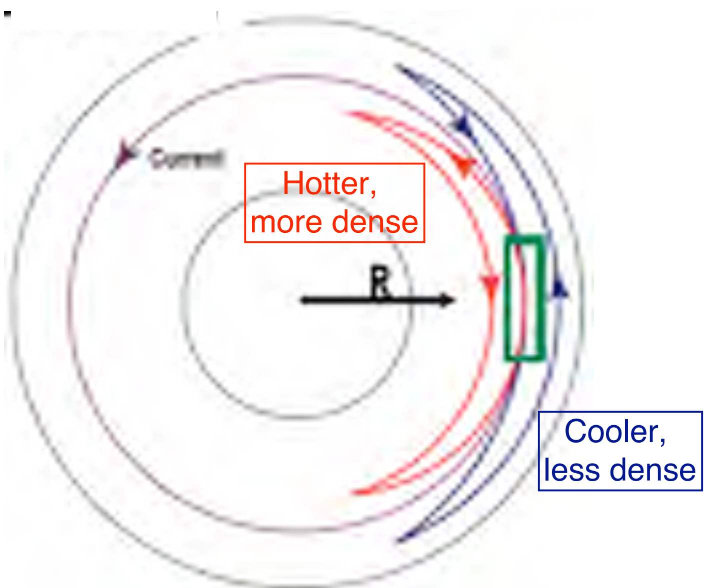
- The **normalized beta** $\beta_N = \beta_T / (I_p/aB_{T0})$ could then be compared to the constant C
- Scaling was confirmed across many tokamaks with auxiliary heating
- To maximize $\beta_T \Rightarrow$ operate at lowest q stable to current-driven kink
- Pushed tokamak design to achieve **high elongation and triangularity**



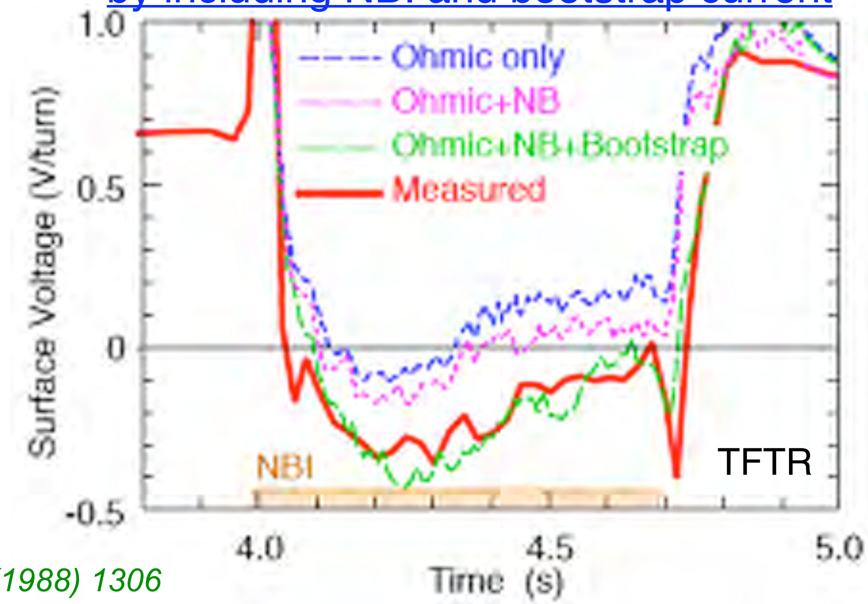
 F. Troyon, et al., *PPCF* ***26****, 209 (1984); figure T. Luce, APS-DPP 2009*

A Consequence of Toroidicity with Important Practical Applications is the "Bootstrap" Current

- In a tokamak, only untrapped (passing orbit) electrons carry toroidal current
- **Bootstrap current** arises from differential friction between untrapped electrons and trapped particles on co-parallel (larger r) and counter-parallel (smaller r) legs of their orbits in presence of a radial pressure gradient
- $I_B/I_{tot} \approx \varepsilon^{1/2}\beta_P$; $\varepsilon = a/R_0$ - inverse aspect ratio, $\beta_P = 2\mu_0\langle p\rangle/B_P^2(a)$ - poloidal-β
- "Supershots" in TFTR achieved sufficiently high β_P to confirm the effect
- Important for possibility of a steady-state tokamak reactor



Can only match measured loop voltage by including NBI and bootstrap current

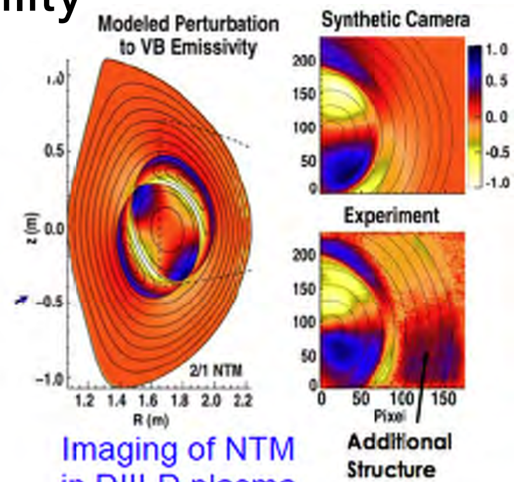


M. Zarnstorff et al., *PRL* **60** (1988) 1306

While Potentially Beneficial, Bootstrap Current Can Destabilize High-Pressure Plasmas

- Local perturbations to the bootstrap current cause growth of the **Neoclassical Tearing Mode** (NTM) instability

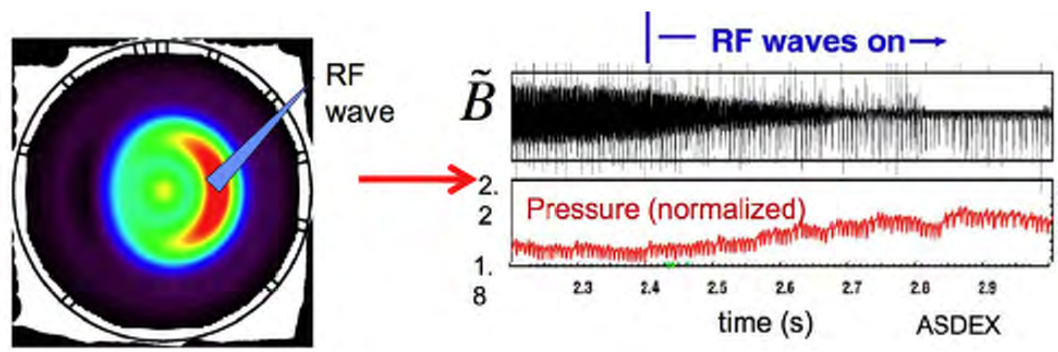
Current perturbation forms an island
↓
island alters pressure
↓
alters bootstrap current



Imaging of NTM in DIII-D plasma

- NTMs of concern to ITER because there is evidence that their threshold for instability decreases with tokamak size

- NTMs can be controlled by feedback stabilization using local heating in the island to counteract the perturbation to bootstrap current



Courtesy R. Fonck

Although We Have Learned to Avoid Many MHD Modes, Two Important Instabilities Persist

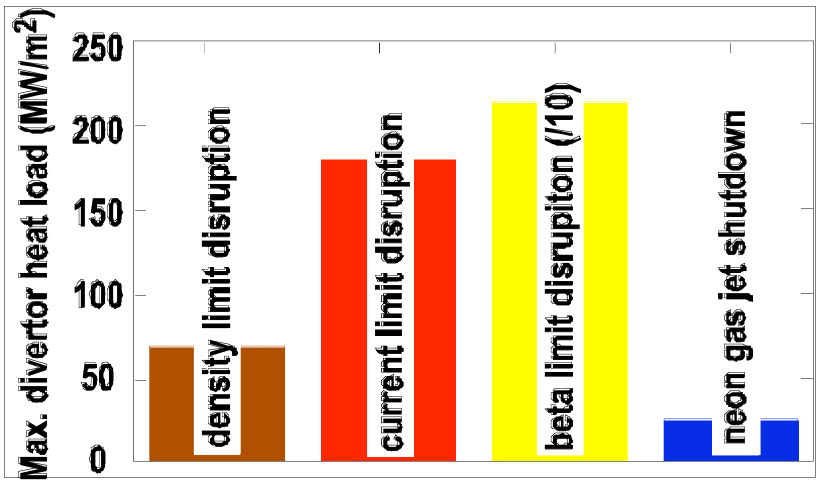
- **Disruption**: a significant rapid (~ms) loss of plasma confinement followed by termination of the plasma current (0.01 – 0.1s)
  - Ubiquitous feature of tokamak operation
    - First described over 40 years ago
  - May be triggered by many different conditions
    - low q_{edge}
    - too high or too low density
    - high β
    - impurity influx
    - unfavorable pressure or current profiles
- **Edge-Localized Mode (ELM):** periodic, rapid losses of energy from the edge of plasmas in the “high confinement” mode of operation (**H-mode**)

Disruptions are Particularly Dangerous for Burning Plasmas: Must be Minimized and Mitigated

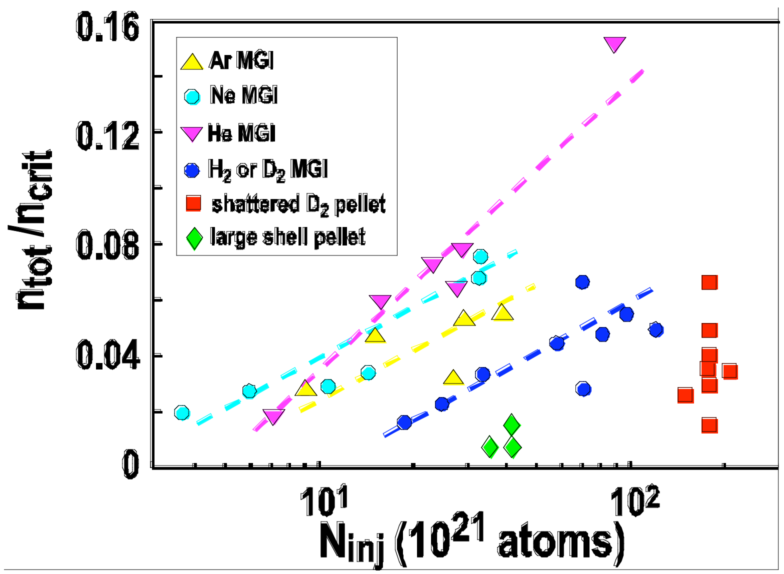
- In ITER, thermal energy in plasma and poloidal field energy ~ 1GJ
  - In current tokamaks ~10MJ
  - "Thermal quench" can damage **plasma-facing components** (PFCs)
    - Difficult to make PFCs handle both steady-state and transient heat loads
  - "Current quench" can produce damaging electromagnetic forces
    - Currents can be induced in conducting elements surrounding plasma
    - These currents may be non-axisymmetric: $\mathbf{J} \times \mathbf{B} \neq 0$
  - Can create large population of energetic (>10MeV) "**runaway electrons**"
- ITER will need to achieve disruption frequency ~1% of discharges
  - Identify disruption precursors in real time and take avoidance actions
    - e.g. reduce β (heating power) or density (fueling), apply MHD mode control
  - Once a disruption starts, use measures to mitigate harmful effects
    - Dissipate plasma energy through radiation over entire first wall
    - Increase density with massive gas injection, liquid jet or pellet injection

Several Tokamaks Have Demonstrated Mitigation of Disruption Heat Loads, Vessel Currents and Forces

- MGI with argon provoked disruptions in Alcator C-Mod, *but*
- Resulting divertor heat loads were significantly reduced *[Whyte, APS 09]*



- Large density increases with Massive Gas Injection (MGI), shattered pellets and shell pellets in DIII-D, *but*
- Critical density for runaway electron suppression not yet reached *[Hollman, APS 09]*

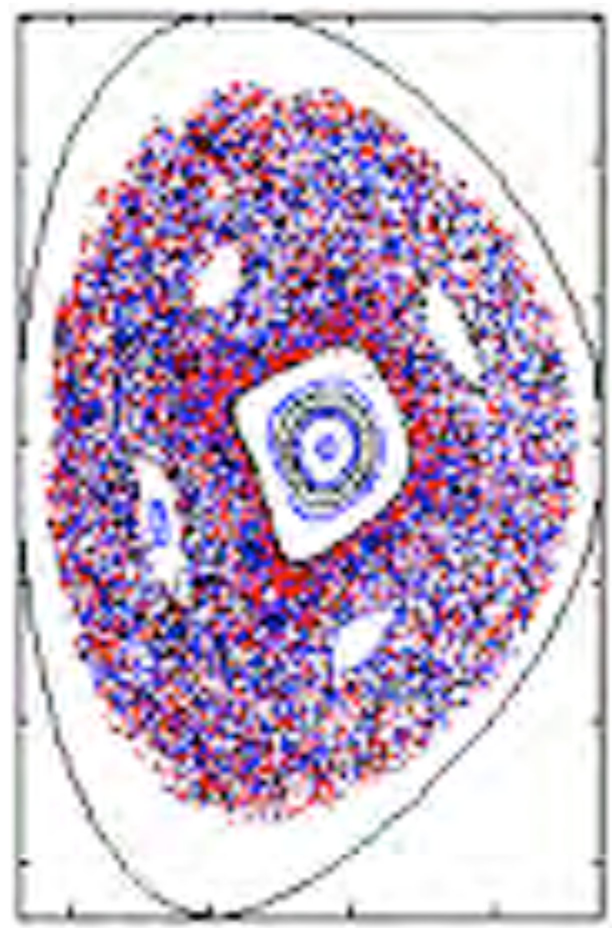


- Method adopted for ITER will need achieve minimal number of false negatives (→ damage) and positives (→ wasted shots)

Formation of Stochastic Field Structure Following MGI May Inhibit Runaway Electron Avalanche

- Runaway electrons are generated initially by the Dreicer mechanism
  - In presence of sufficient electric field, some thermal electrons can be accelerated faster than they lose energy by collisions ($\propto v^{-3}$)
- Runaways can multiply by direct "knock-on" collisions $\Rightarrow$ **runaway avalanche**
- Suppressing runaway avalanche by collisions alone would require a critical (Connor-Hastie-Rosenbluth) density equivalent to several hundred grams of gas in ITER
- 3D resistive MHD modeling shows that formation of **stochastic fields** triggered by MGI can cause rapid loss of runaways
- May not be necessary to attain CHR density limit to avoid runaway damage in ITER

Simulation with NIMROD code of Alcator C-Mod following MGI

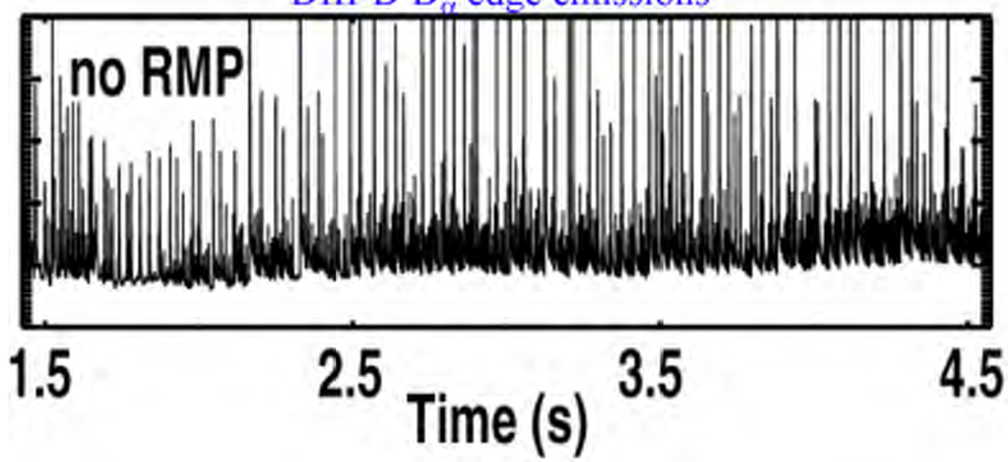


V. Izzo, APS-DPP meeting (2009)

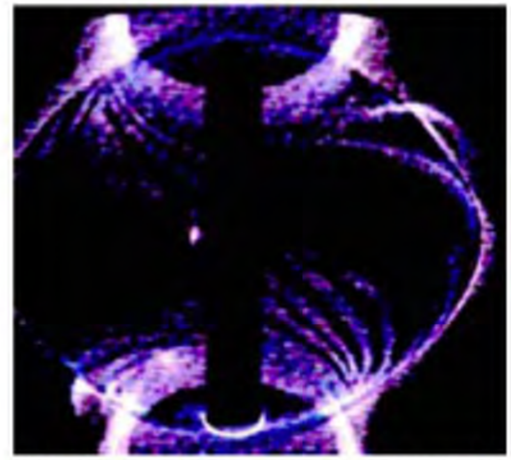
Steep Pressure and Density Gradients in H-mode Plasmas Destabilize Edge Localized Modes (ELMs)

- ELMs readily observed as "spikes" in D_α line emission from plasma edge

DIII-D D_α edge emissions

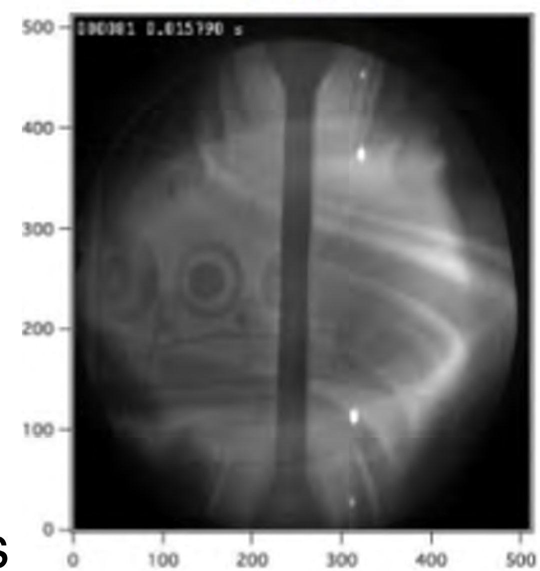


- Each spike is correlated with large, coherent filamentary instability at edge
  - Periodic ELMs represent a relaxation instability
- Many different types of ELM have been found
- Can reduce impulsive load by operating in regimes with (or triggering) more frequent ELMs



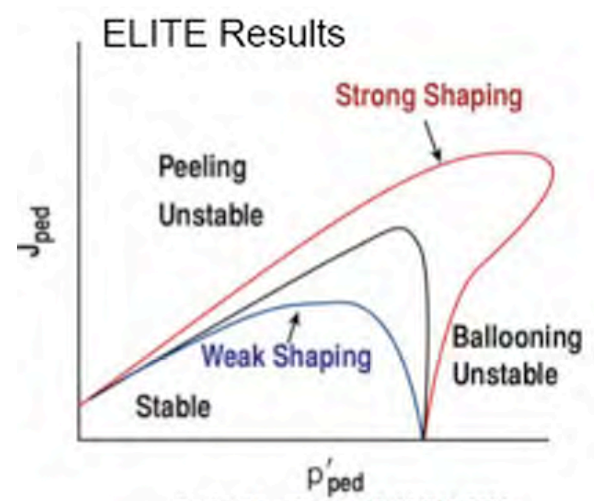
Kirk, 2004 Proc. 20th Int. Conf. on Fusion Energy 2004

MAST

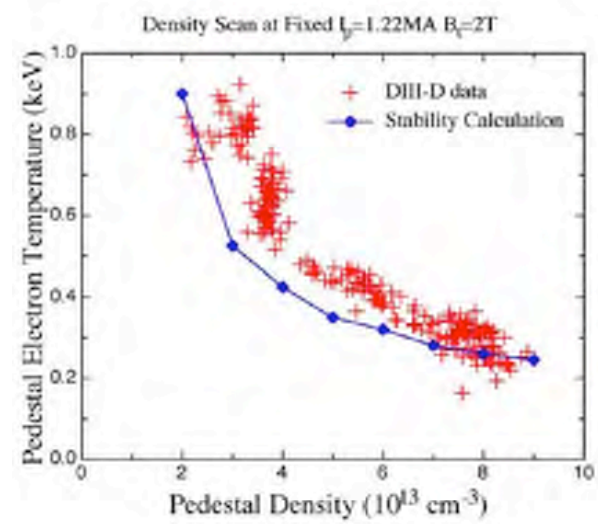


Edge Localized Modes are Well Described by Theory of "Peeling-ballooning" Modes

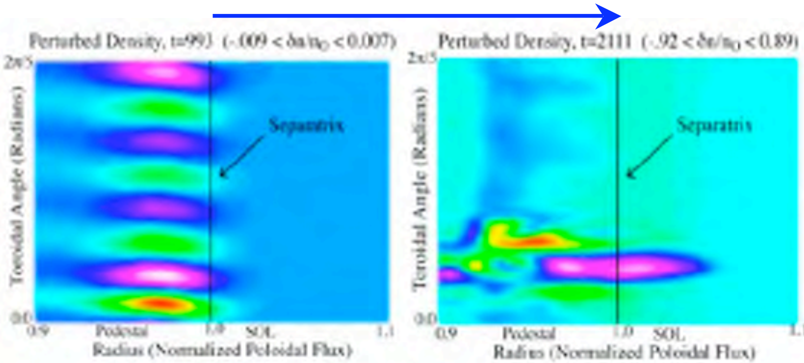
- High edge current density drives "peeling"
- High edge pressure drives "ballooning"
- Bootstrap current plays crucial role linking pressure and current
- ELM then relaxes unstable gradients
- Theory describing peeling-ballooning modes reproduces ELM threshold and observed mode structure

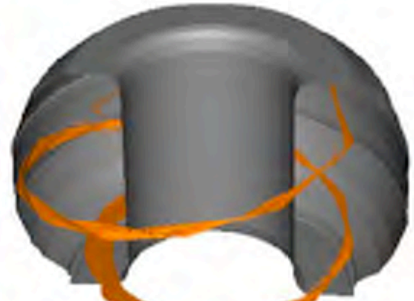


Snyder, Phys. Plasmas **12**, 056115, 2005



Simulation of time evolution

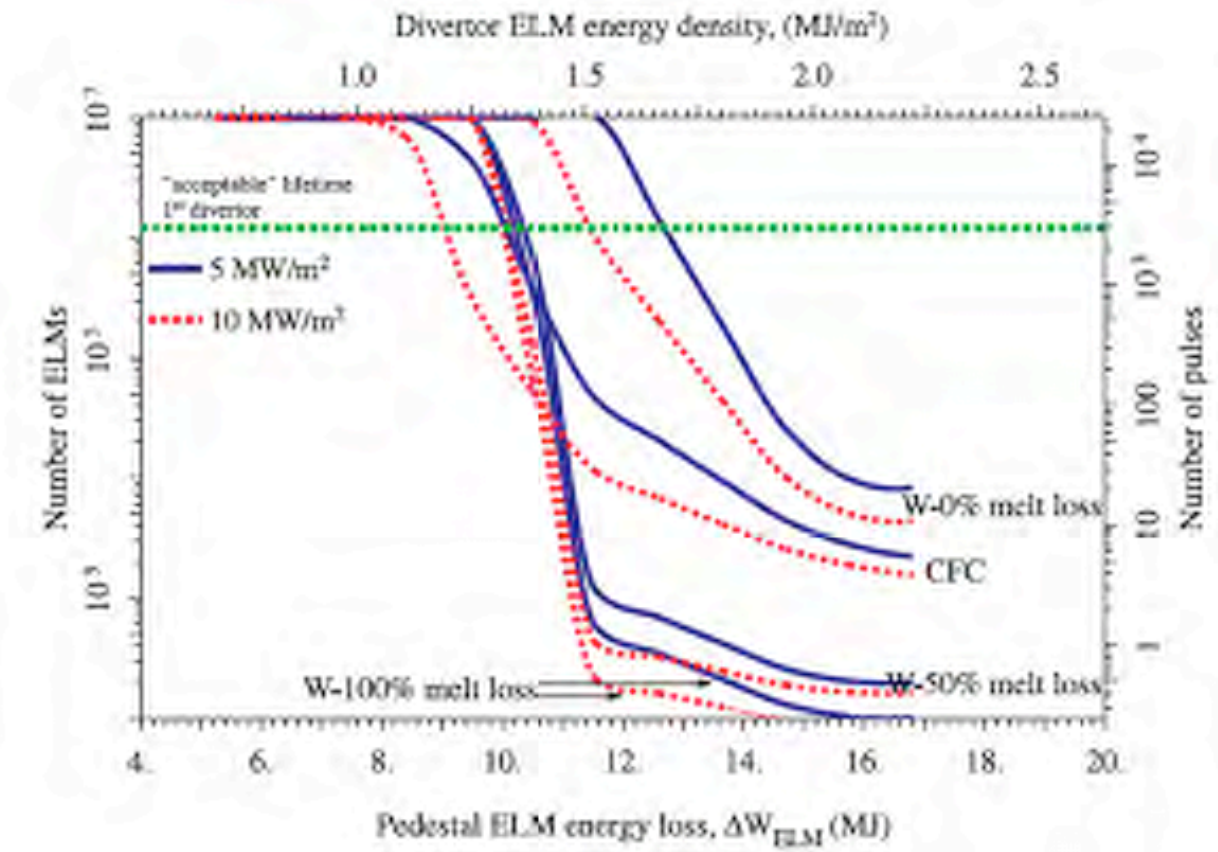




P.B. Snyder et al, Phys. Plasmas 12 056115 (2005).

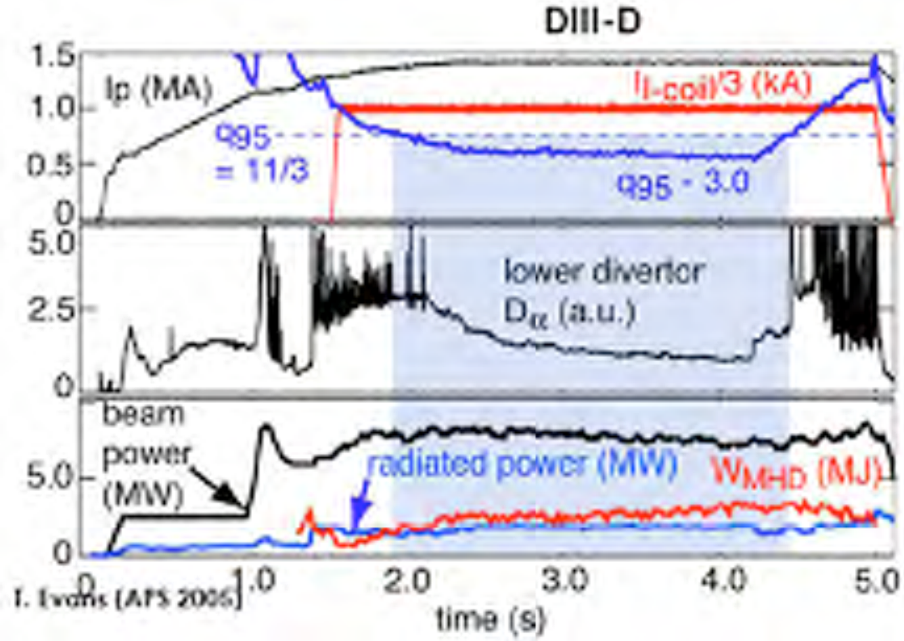
Repeated Large ELMs Will Damage the Divertor Target in ITER and Limit Its Lifetime

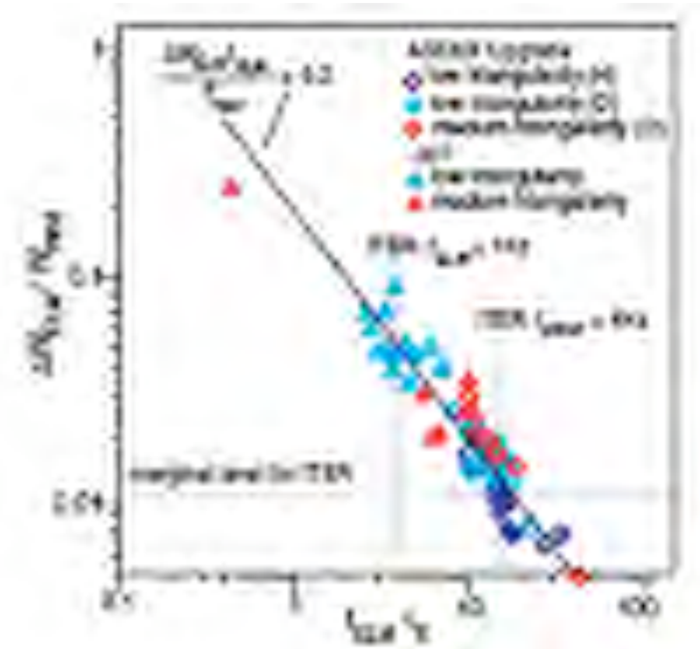
- Calculated erosion lifetime of a tungsten target (10mm thick) or CFC target (20mm) as a function of ELM energy loss from the pedestal
- Heat loads between ELMs are 5 MW/m^2 (—) and 10 MW/m^2 (…)
- Curves are shown for different fractions of tungsten lost by melting



G. Federici, PPCF (2003)

Several Tokamaks are Investigating ELM Control Methods for ITER





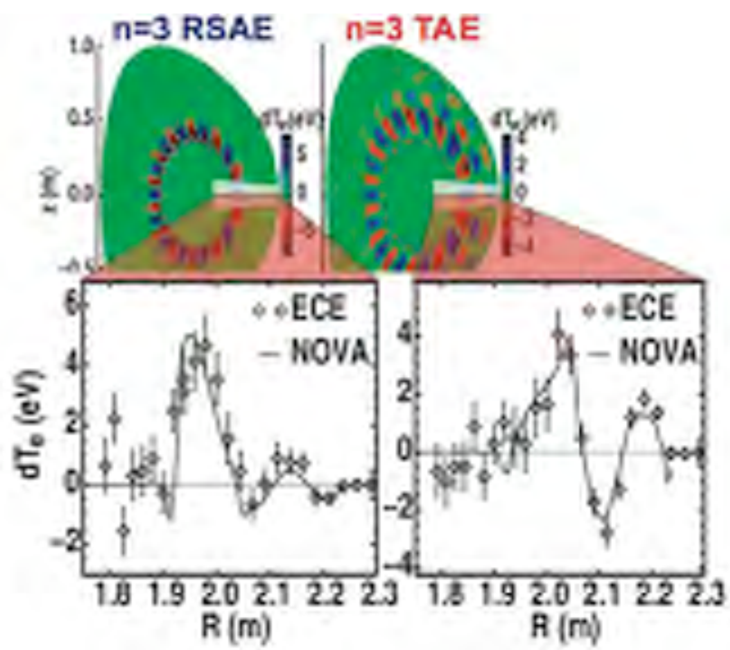
- Applying Resonant Magnetic Perturbation (RMP) with non-axisymmetric external coils can suppress ELMs in certain conditions
  - RMP creates region with stochastic field lines (overlapping islands) at edge
  - Additional transport relaxes edge pressure gradient
- Repetitively injecting small solid H, D pellets can trigger ELMs
  - ELM size reduces with frequency
  - **Issues**: minimum pellet size and penetration; compatibility with fueling requirements

- ITER will be equipped with non-axisymmetric coils to control ELMS

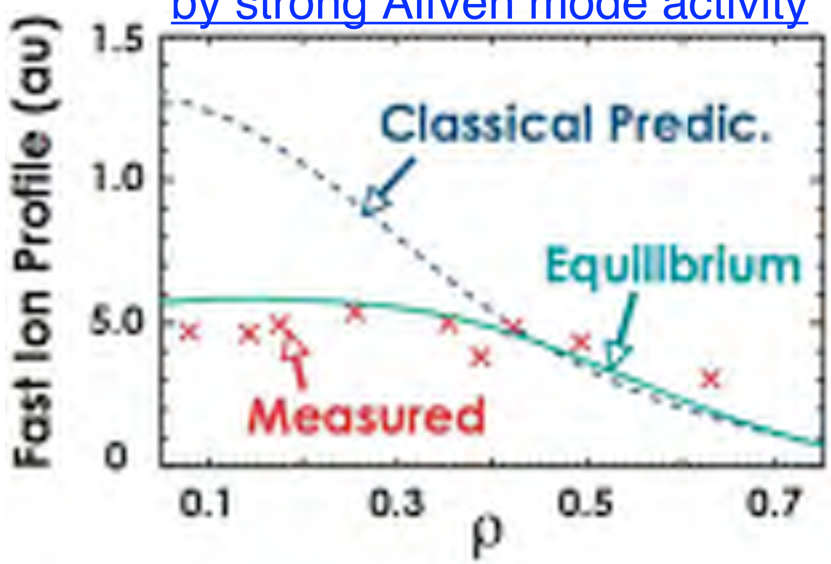
 T. Evans, APS-DPP meeting (2005) V. Mukhovatov, PPCF (2003)

Energetic Ions Including α-Particles Can Destabilize Alfvén Wave Eigenmodes in Toroidal Plasmas

- In a torus, the shear Alfvén wave ($\omega = kv_A$, $v_A = B/(\mu_0\rho)^{1/2}$) develops an eigenmode structure as a result of toroidal and poloidal periodicity
- Fusion α-particles with $v_\alpha > v_A$ can excite Toroidal Alfvén Eigenmodes (TAEs) which then affect the α-particle orbits and cause losses
- Theory of Alfvénic modes is now highly developed and successful
  - Many modes beyond basic TAEs have been found in shaped, high-β plasmas
- Existing tokamaks can use NBI ions to excite modes at low magnetic field



Fast-ion profile becomes flattened by strong Alfvén mode activity



M. Van Zeeland, PRL (2006) 135001

Topics

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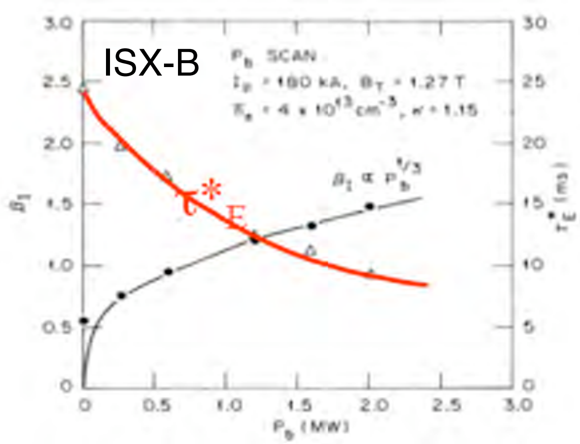
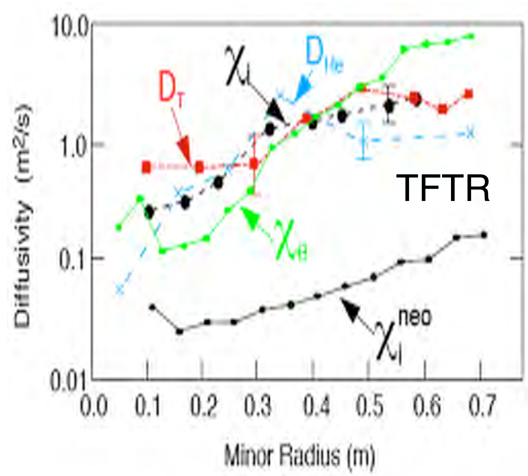
For Plasmas Stable to Large-Scale MHD Modes, Transport From Micro-Turbulence is Dominant

- Until 1990s, transport understanding was largely empirical
- Despite better confinement in tokamaks, transport was anomalous
  - Diffusion exceeded predictions of "**neoclassical**" (toroidal) theory
- Turbulence was blamed but theoretical and simulation tools were not yet sufficiently developed to tackle the problem quantitatively
- Measured fluctuations were reduced when plasma underwent transitions from low (L-mode) to high (H-mode) confinement

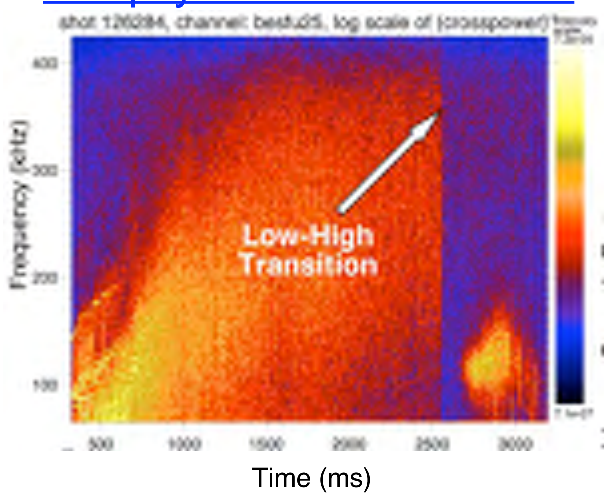
Confinement degraded as NBI power increased



Transport coefficients exceeded neoclassical

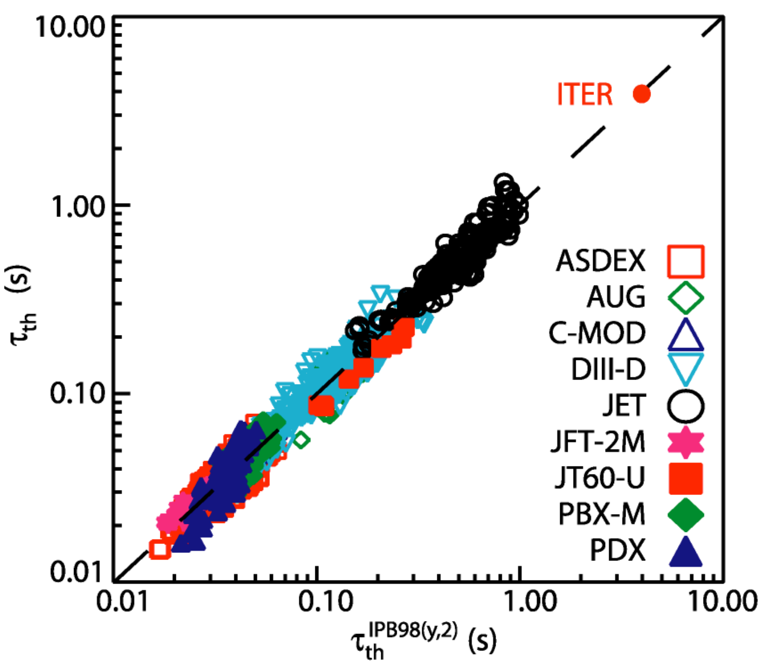


Broadband turbulent fluctuations abruptly fall at L-H transition



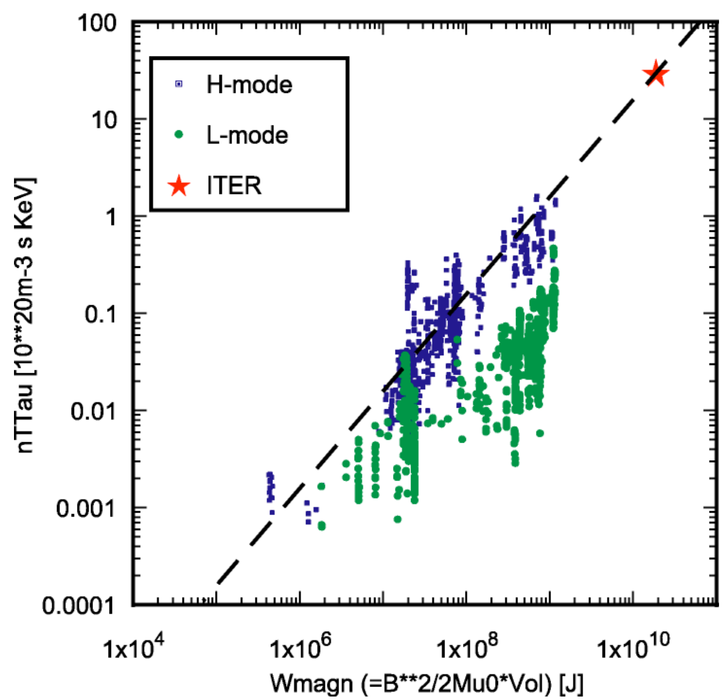
Data from Many Experiments Combined to Produce an Empirical Scaling for Design of ITER

- ITER needs a confinement time of ~4s to achieve Q ≈ 10
- 1998 data from H-mode divertor plasmas with "Type I" ELMs



$$\tau_{E,98} \propto I_P^{0.93} B_T^{0.15} \overline{n}_e^{0.41} P_{Loss}^{-0.69} R_0^{1.97} \varepsilon^{0.58} \kappa^{0.78}$$

- Can also examine scaling of "fusion triple product" nTτ with tokamak size and magnetic field

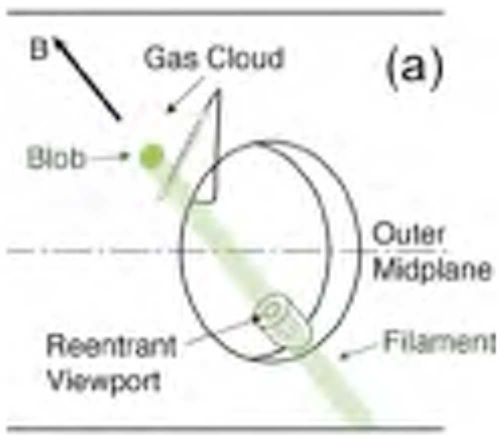


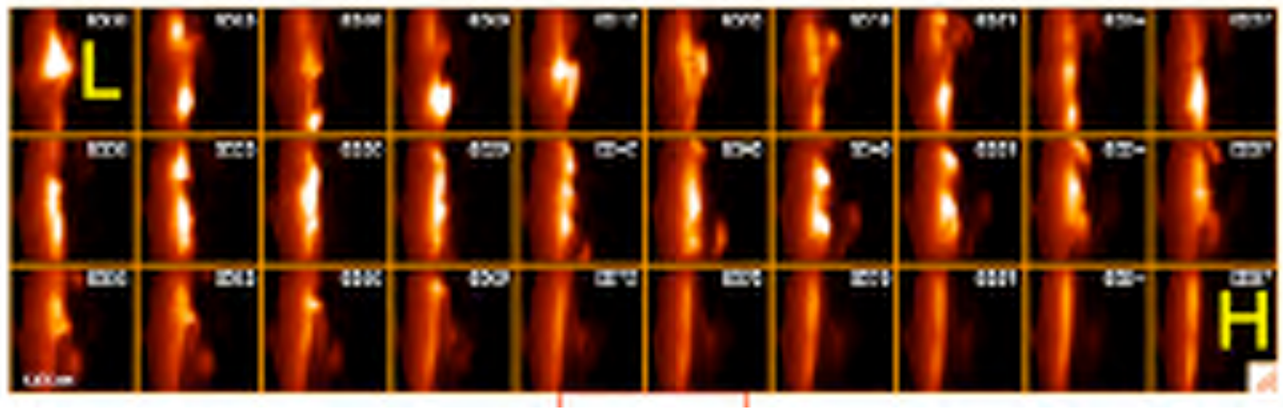
Can we put confinement on a firmer footing than a purely empirical scaling?

New Instruments and Computational Tools Are Revolutionizing the Study of Turbulence

- Fast cameras (up to 10^6 fps) can visualize turbulent structures

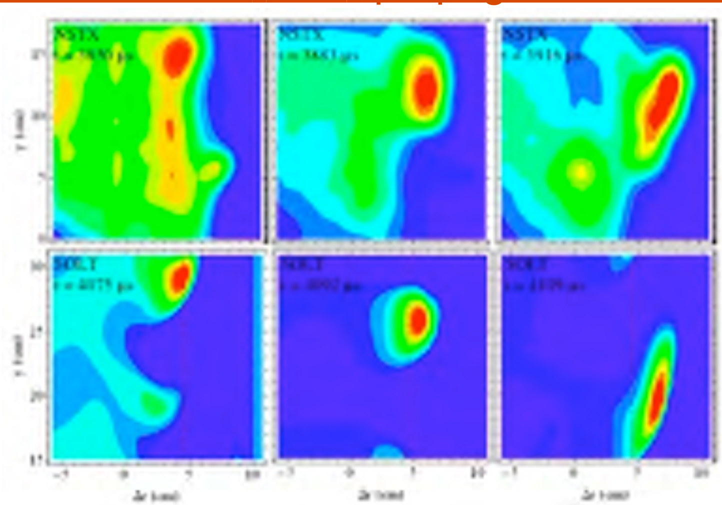
Over 100μs, turbulent edge becomes quiescent at L-H transition





- Developments in theory have improved computation schemes
- Massively parallel computers allow realistic simulations of turbulence from first-principles
- Codes incorporate “synthetic diagnostics” to compare simulations with measurements

Simulations of “blob” propagation in NSTX



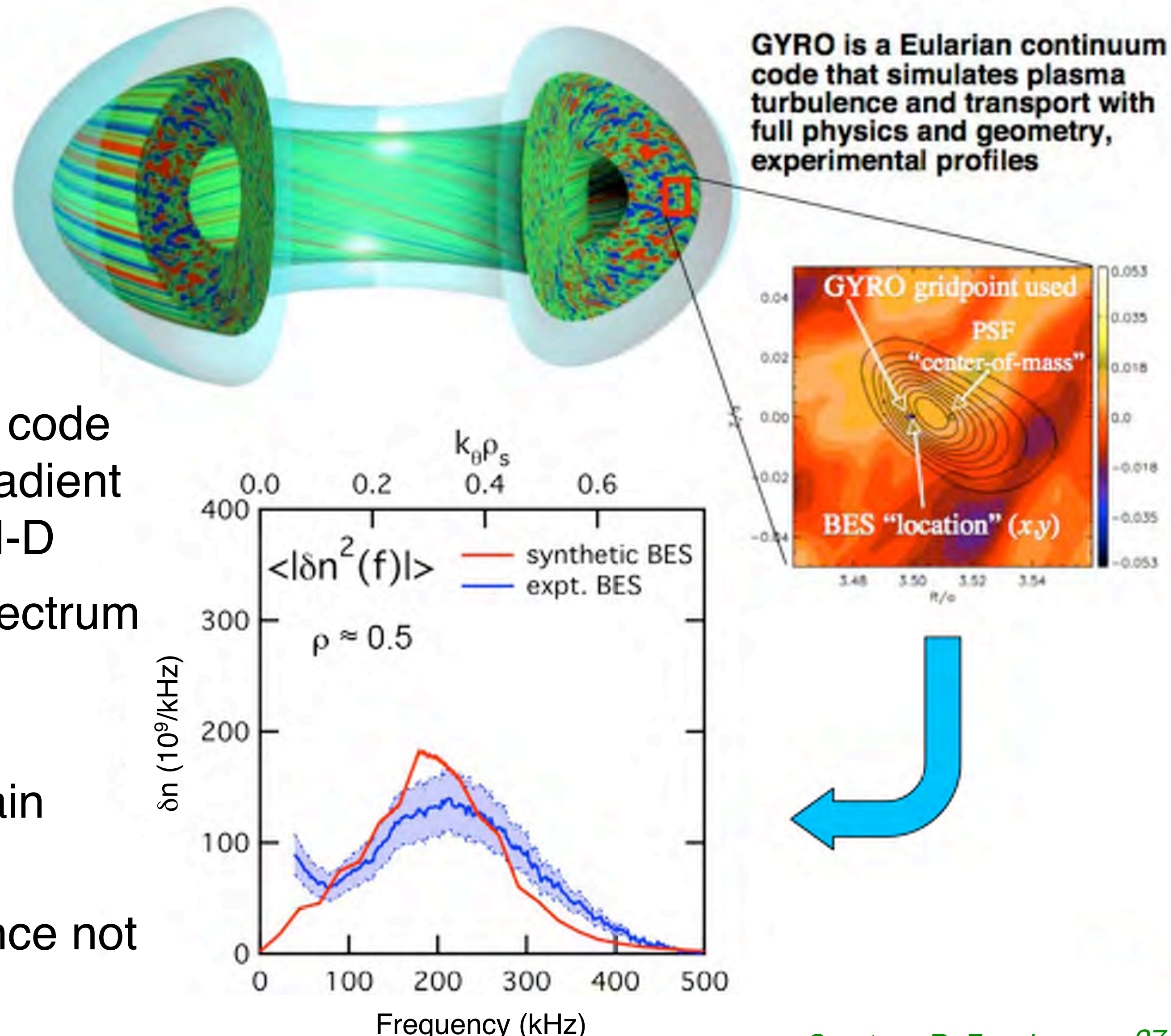
Courtesy S. Zweben, R. Maqueda, D. D’Ippolito

Simulations and Measurements of Ion-Scale Turbulence Have Attained Excellent Agreement

- In last 15 years, a "standard model" of ion turbulence and transport has emerged

  Example

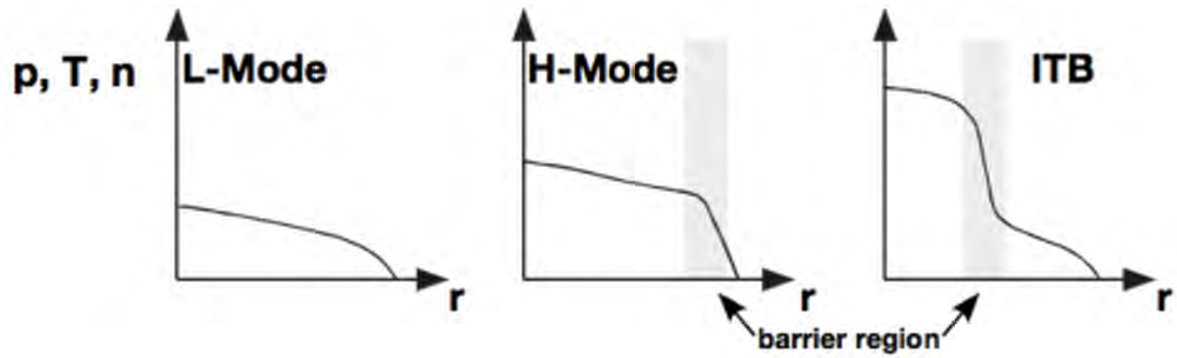
- Simulation with GYRO code of Ion Temperature Gradient (ITG) turbulence in DIII-D
- Matches fluctuation spectrum from Beam Emission Spectroscopy (BES)
- But, some details remain unresolved *and*
- Electron-scale turbulence not yet accessible



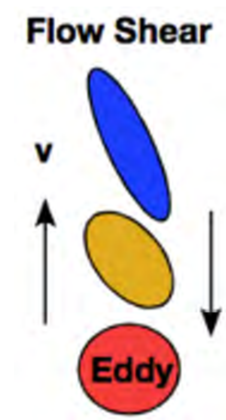
Courtesy R. Fonck

Understanding and Controlling Transport Have Led to Improved Confinement

- **Transport barrier**: region of locally reduced transport in radial profile
  - Edge transport barrier → "H mode" (high confinement)
  - Internal transport barrier (ITB) in core of plasma



- Transport barriers form with suppression of turbulence by
  - **Flow shear** ($\partial v/\partial r$): driven by plasma gradients and external momentum sources
  - **Negative magnetic shear** ($\partial q/\partial r<0$) : created by current drive including bootstrap current
  - **Zonal flows**: flows created by fluctuations themselves



Dependence of Tokamak Confinement on Plasma-Wall Interactions is Not Well Understood

- Many techniques have been applied in tokamaks to modify the interactions between a plasma and its surroundings
  - Limiters (object defining the last closed flux surface) *vs* divertors
  - Refractory metallic surfaces (high-Z) *vs* carbon (graphite, low-Z)
  - Baking PFCs and the vacuum chamber (reduces adsorbed H_2O)
  - Discharge cleaning (pulsed or glow discharge) by noble gases
  - Surface coatings: titanium (gettering), boron, silicon, lithium
- All have been claimed to produce benefits!
  - Reduced impurities in the plasma – *fairly obvious connection*
  - Better confinement – *how?*

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"Conditioning"

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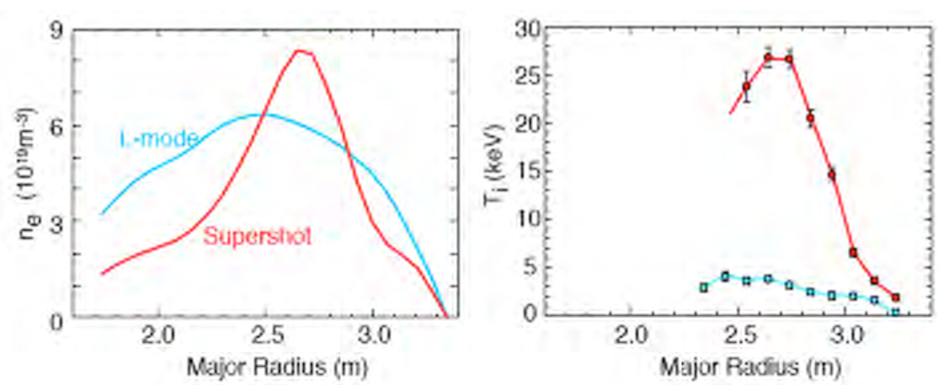
  - Better confinement – *how?*
- A common thread in the claims related to "conditioning" is that confinement improves with reduced "**recycling**" from walls
- **Recycling** describes ions which diffuse from the plasma, impinge on the PFCs, become neutralized and return to the plasma edge

Effects of Wall Conditioning Were Dramatic in TFTR

- Originally used repeated tokamak discharges in helium to deplete the graphite limiter surface of adsorbed hydrogen isotopes ⇒ *lower recycling*
  - With centrally deposited NBI, density profile became peaked
  - Ion temperature increased by factor >5 and became very peaked
- Injecting lithium into the plasma edge further improved confinement
  - Benefits of lithium have since reproduced in tokamaks T-11, NSTX (divertor), EAST and stellarator TJ-II

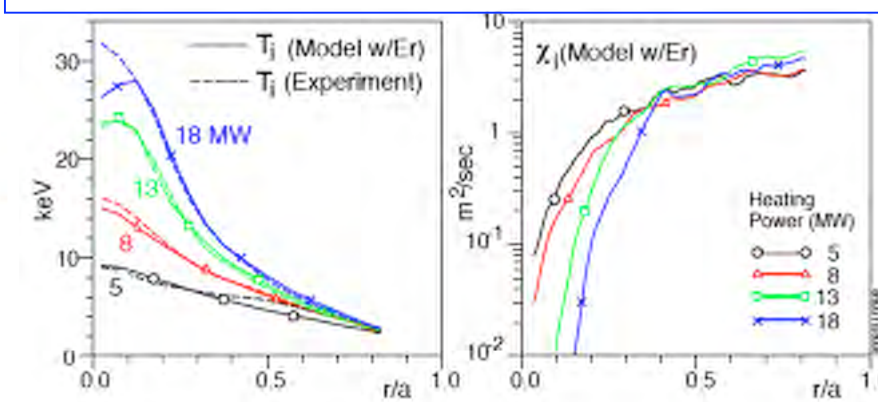
Theory-based model with ITG turbulence suppressed by self-consistent flow shear matches data in supershot power scan



τ_E(s): 0.06 → 0.18

$n_e T_i \tau_E$: 0.15 → 4.3

($10^{20}m^{-3}$keVs)



- Model reproduces observed *inverse* dependence of ion thermal diffusivity on ion temperature

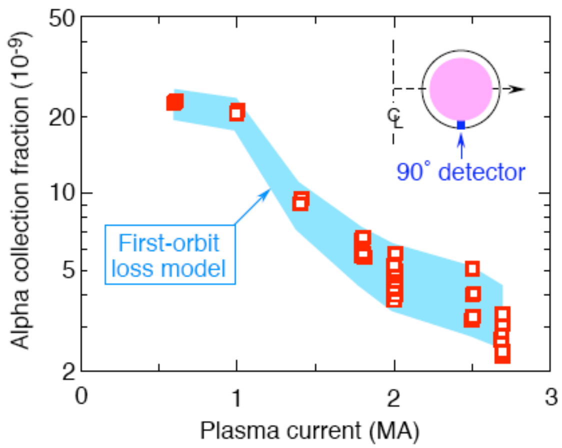
D. Ernst et al., PRL **81** (1998) 2454

Topics

- Tokamak fundamentals
- Tokamak stability
- Confinement and transport
- **DT experiments in TFTR and JET**
- The leap to ITER

TFTR Measured Confinement and Thermalization of Fusion Alphas in DT Plasmas

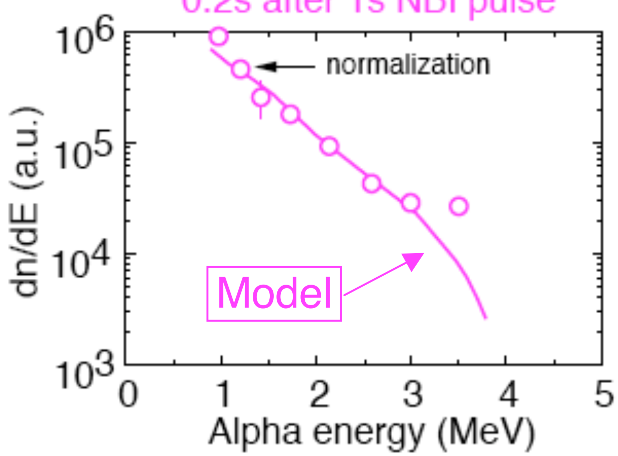
Flux of α-particles to detector agrees with calculated loss for unconfined orbits



- Shading shows result from an orbit-following code based on calculated alpha-particle birth and plasma current profiles
- At 2.5MA, ~3% of alphas lost on first orbit after birth

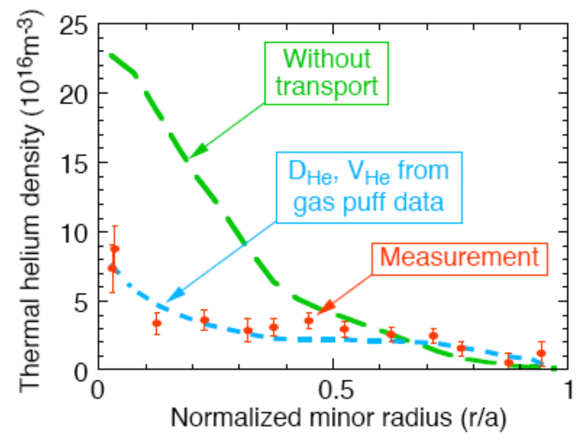
Confined α-particles show classical slowing-down energy spectrum

0.2s after 1s NBI pulse



- Calculated spectrum from Fokker-Planck calculation using measured plasma parameters

Profile of thermalized α particles matches model for helium puff



- Concern was that with central source, α's might accumulate in core and dilute fuel

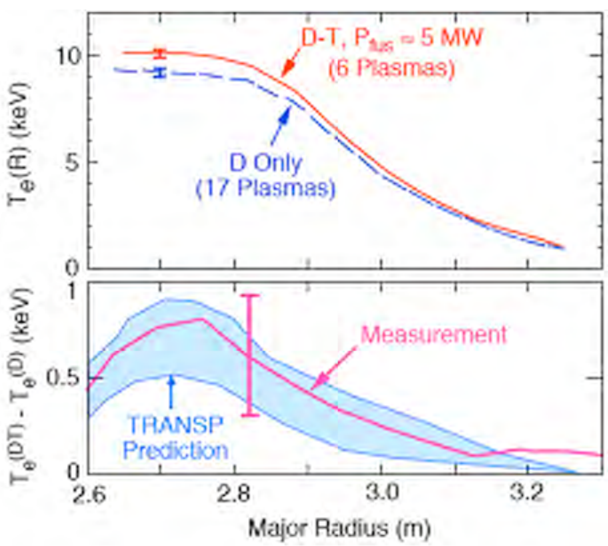
- In TFTR and JET, the achieved fusion power was modeled quite accurately based on measured plasma parameters and classical ion thermalization

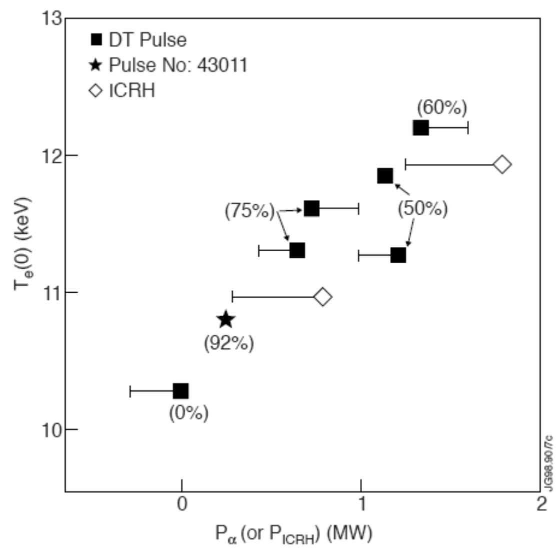
R. Hawryluk et al., *PoP* **5** (1998) 1577

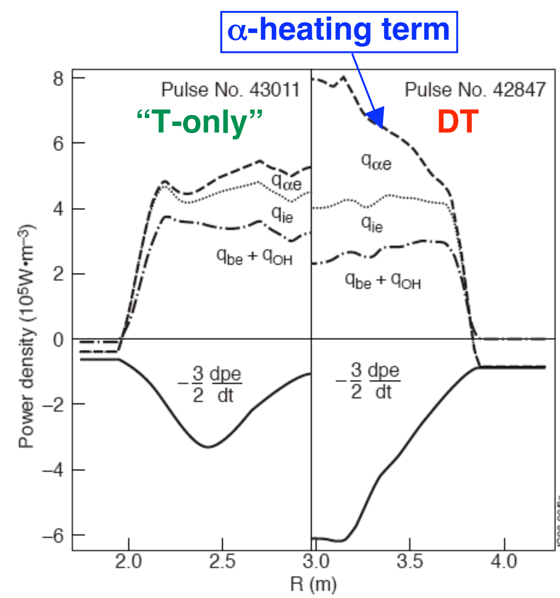
TFTR and Later JET Confirmed Electron Heating by DT Alpha Particles

DT plasmas in TFTR showed an increase in electron temperature compared to D-only plasmas

With higher Q, JET provided a more definitive demonstration of α-particle heating
- **D → DT → T variation**







- Prediction includes model for isotopic dependence of electron thermal transport

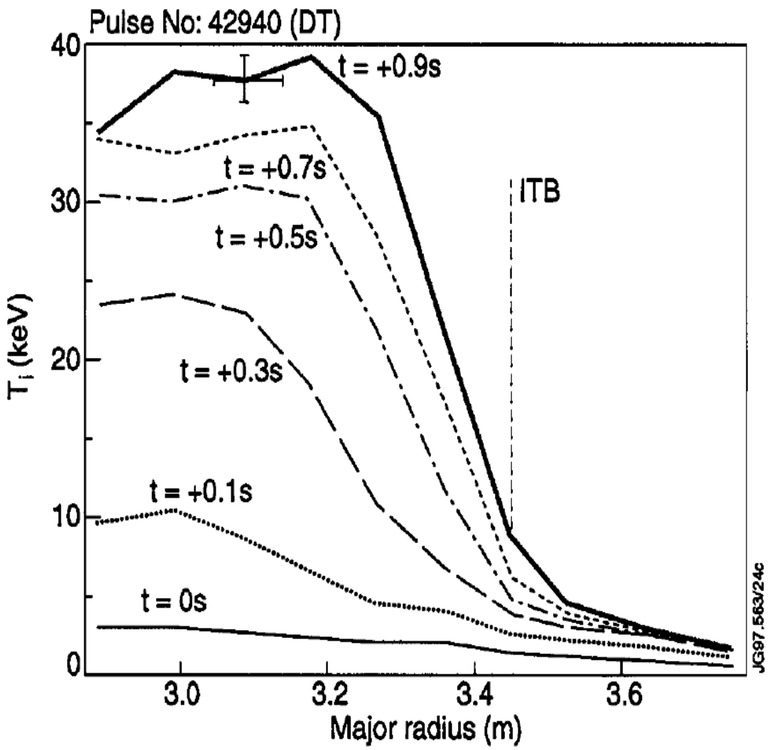
- JET experiment also included a comparison discharge in which electrons were heated by energetic ions from ICRH to mimic α-heating

G. Taylor et al., *PRL **76** (1996) 2722*   *P. Thomas* et al., *PRL **80** (1998) 5548*

"Advanced Operating Modes" Also Achieved in DT Plasmas

- Both **JET** and **TFTR** investigated DT plasmas with q-profile modified to produce q(0) > 1 and low magnetic shear

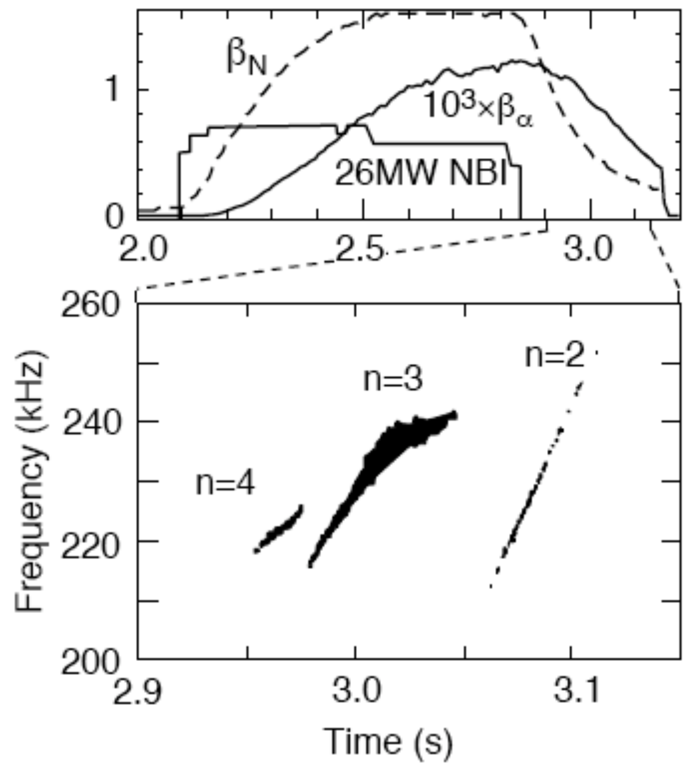
JET "Hybrid Mode" DT plasma with Internal Transport Barrier



Achieved P_{DT}=8MW, β_N=1.9, H=1.5

First observation of α-driven TAE

- Mode develops in core when damping by sub-Alfvén NBI-ions decays

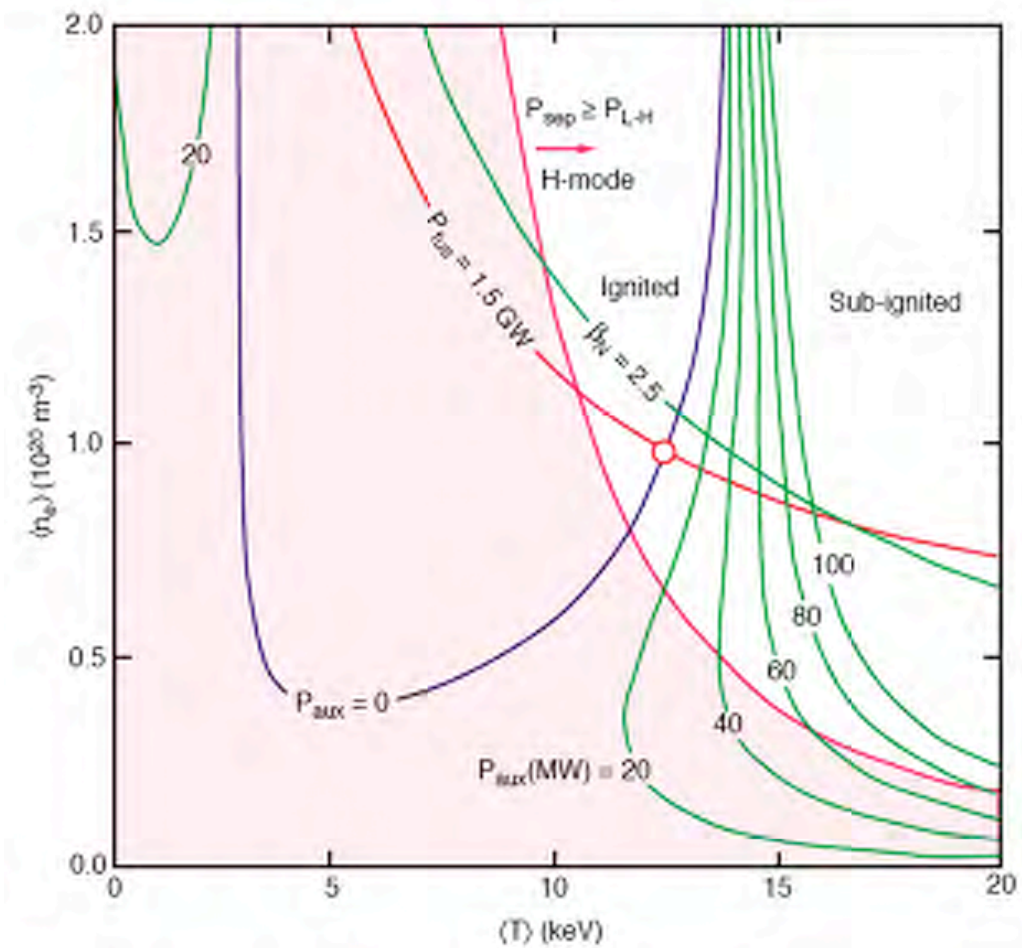


C. Gormezano et al., *PRL (1998)*
M. Bell et al., *PoP* ***4*** *(1997) 1714*

Topics

- Tokamak fundamentals
- Tokamak stability
- Confinement and transport
- DT experiments in TFTR and JET
- **The leap to ITER**

Can Use the Empirical Scaling to Assess Fusion Burn Control and Thermal Stability in ITER



Contours depend on I_p, B_T, scaling of confinement and assumed profile shapes

- Vary plasma temperature and density to generate Plasma Operation Contours (POpCon)
- Sustained fusion ignition (P_{aux}=0) and finite-Q (P_{aux} > 0) are accessible
- Need to achieve H-mode ($P_{sep} \geq P_{L\text{-}H}$) and stay below the beta limit
- Plasma burn will be stable since ITER operates near the stable (right) branch of the ignition curve
  - Power loss increases faster than fusion power as temperature rises

What New Physics Should We Anticipate in ITER?

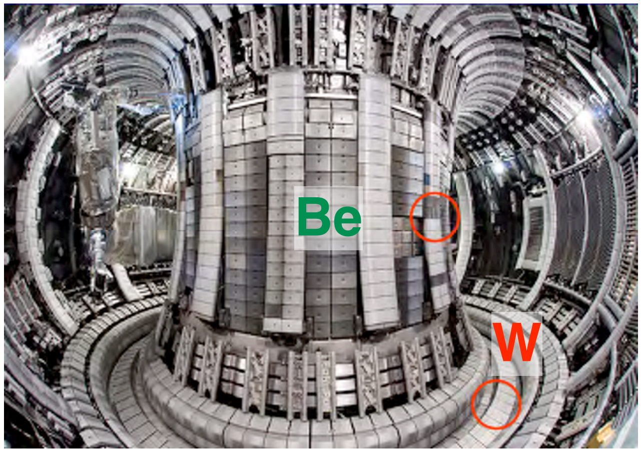
- ITER requires high energy NBI to penetrate its large plasma
  - ~1MeV NBI will dominantly heat electrons (like α-particles)
  - JT-60U has demonstrated good performance with 0.4MeV NBI
  - Will good confinement in plasmas with $T_i > T_e$ (hot-ion modes) persist?
  - Will TAE activity affect confinement of NB injected ions?
- Physics of wave heating (ICRH, ECRH, LHH) is reasonably well understood but there are practical issues
  - Coupling power to the plasma is often the limiting factor
  - Wave couplers must operate in a more hostile environment
  - Large α-particle population may affect wave absorption
- Dominant self-heating by fusion α-particles creates challenges, particularly for achieving and maintaining high-confinement modes
  - Equilibrated ion & electron temperatures
  - Low rotation (reduced momentum input)
  - Profiles (n, T, q) become self-organized

All factors involved in controlling confinement and MHD stability

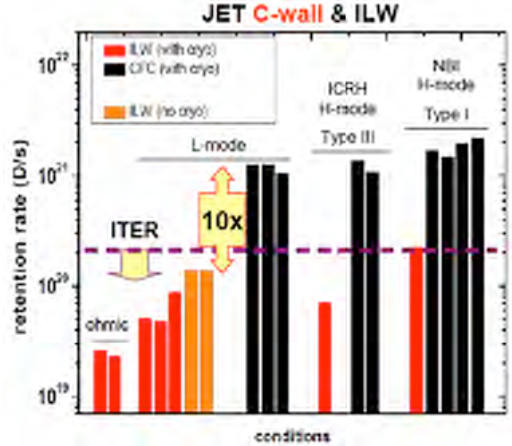
Choice of Plasma Facing Materials is Critical

- Until recently, most high-power, high-performance tokamaks operated with carbon PFCs in high-heat flux regions
  - Carbon is extremely “forgiving” of transient heat loads and low-Z
- Carbon retains too much tritium for use in ITER
  - Experience in TFTR, JET showed retention of up to 50% of T fuel
- ITER planned to use tungsten for its divertor targets during DT
  - Other areas would be covered with beryllium tiles (JET experience)
  - Concerns about damage to tungsten and tungsten impurities (high-Z)
- Several tokamaks are now investigating metal divertor PFCs
  - Alcator C-Mod has operated with Mo walls and will soon switch to W
  - ASDEX-U has applied W coating on all its graphite PFCs
  - JET is now operating with an “ITER-Like Wall”: W divertor, Be elsewhere
- Other tokamaks also investigating liquid metals, *e.g.* lithium in NSTX, for future beyond ITER

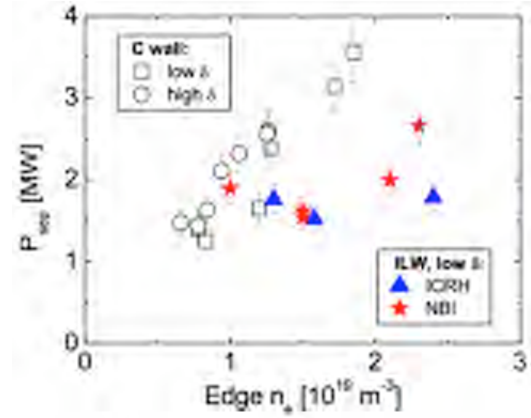
JET has Completed the First Year of Operation With Its ITER-Like Wall (ILW)

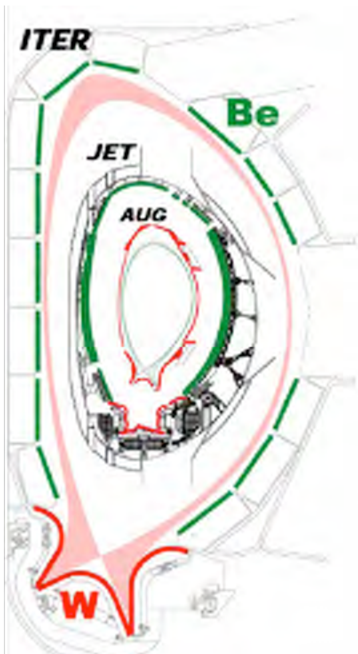


Lower D retention

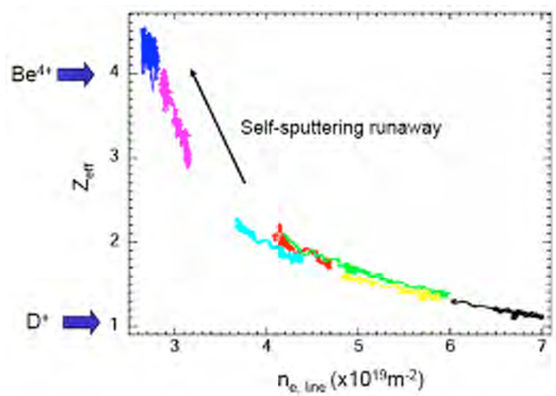


Lower H-mode threshold





Low Z_{eff}



- Similar results in ASDEX-U

Concerns

- higher disruption loads,
- narrower window for good H-mode

G. Matthews, 20 PSI Conf. (May 2012)

Tokamaks Remain the Most Successful Confinement Concept

- They emerged because they demonstrated better confinement *and*
- They were simpler than stellarators – a few, axisymmetric coils
  – Allowed larger devices with auxiliary heating and good diagnostics
- We have made great strides in understanding confinement & stability
  – Advances in diagnostic techniques allowed much of this progress
- We are developing the capability to predict tokamak plasma behavior from first principles: ***theory → computation → experimental test***
- Some of the original simplicity of tokamaks has had to be sacrificed to operate them with high power heating and near stability limits
  – Many poloidal field coils are needed for plasma shaping and divertors
  – They require advanced feedback involving magnets, heating and fueling systems and real-time measurements of many plasma parameter
  – Even axisymmetry has been modified for MHD mode (including ELM) control
- The first experiments with DT fusion fuel were a resounding success
  – We learned how to operate tokamaks in a fusion nuclear environment
  – The fusion rates were consistent with our understanding and simulations
  – The alpha particles behaved as expected and heated the plasma effectively

Tokamaks are Ready for the "Leap to ITER"

- The knowledge we have gained from several generations of tokamaks has given us confidence to proceed to ITER, *but* ...
- There remain unresolved issues in some areas
  - Validation of the choice of PFC material
    - Alcator C-Mod, ASDEX-U and JET-ILW provide grounds for optimism
  - Adequacy of the auxiliary heating systems to achieve the H-mode in order to reach plasma self-heating
  - Adequacy of the schemes proposed for ELM control
  - Ability to eliminate damaging disruptions reliably
- The ultimate success of ITER still depends on research underway now in many tokamaks
  - We cannot rely on just one experiment to answer critical questions
- Existing tokamaks also need to train the next generation of physicists and engineers who will operate ITER