Realizing Steady State Tokamak Operation for Fusion Energy

Presented by
T.C. Luce

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Acknowledgments

- **Contributors:**
  - US: Garofalo, Heidbrink, Humphreys, Hyatt, Kessel, Leonard, Manickam, Menard, Najmabadi, Navratil, Okabayashi, Petty, Politzer, Prater, Strait, Taylor, Turnbull, Wade, Zarnstorff
  - EU: Challis, Coda, Joffrin, Pinches, Sauter, Sips, Ward, Zohm
  - Japan: Ide, Kamada, Matsunaga, Sakamoto, Suzuki, Takase

- **ITPA IOS/SSO TG**
  - C. Gormezano (2002-2004), G. Sips (2005-2008), S. Ide (2008-)

The Role of Fusion

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  Will fusion work?

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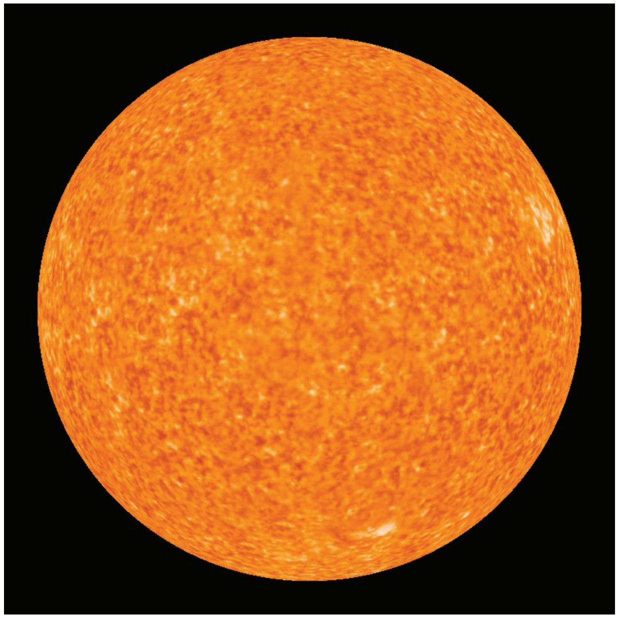

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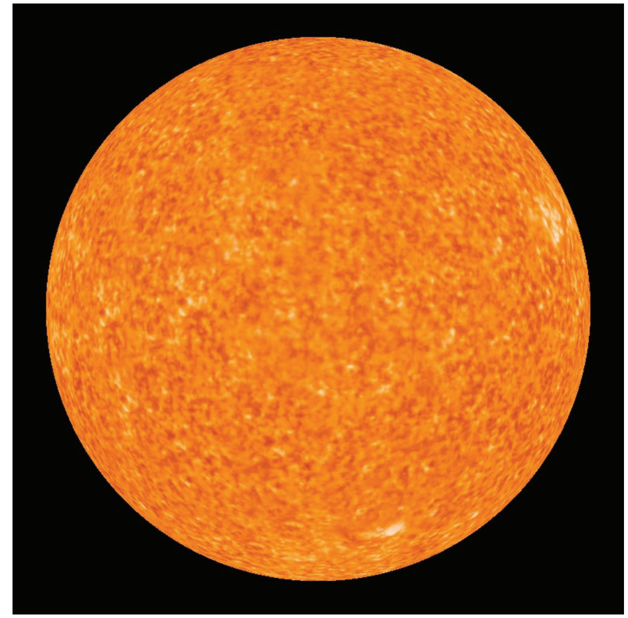
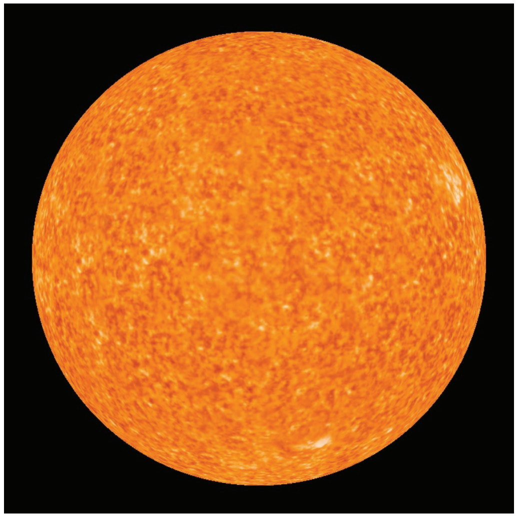


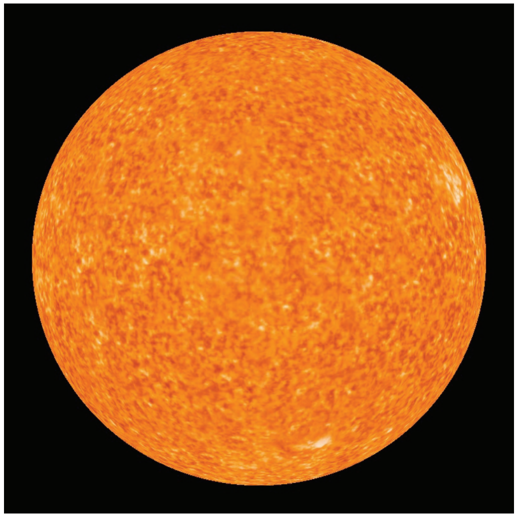
- **What is really being asked is whether we can deliver on the high expectations raised by popular culture**

The Role of Fusion



- **Between the obvious reality and the high expectations, we may lose sight of the real role of fusion--to supply the world's energy needs in an environmentally sustainable way using resources distributed globally**

The Role of Fusion



- **Between the obvious reality and the high expectations, we may lose sight of the real role of fusion--to supply the world's energy needs in an environmentally sustainable way using resources distributed globally**

- **Progress in steady-state tokamak research makes it a strong candidate as the core of a fusion power plant**

The Physics Challenges of Steady-State Tokamak Research Require a Change of Paradigm

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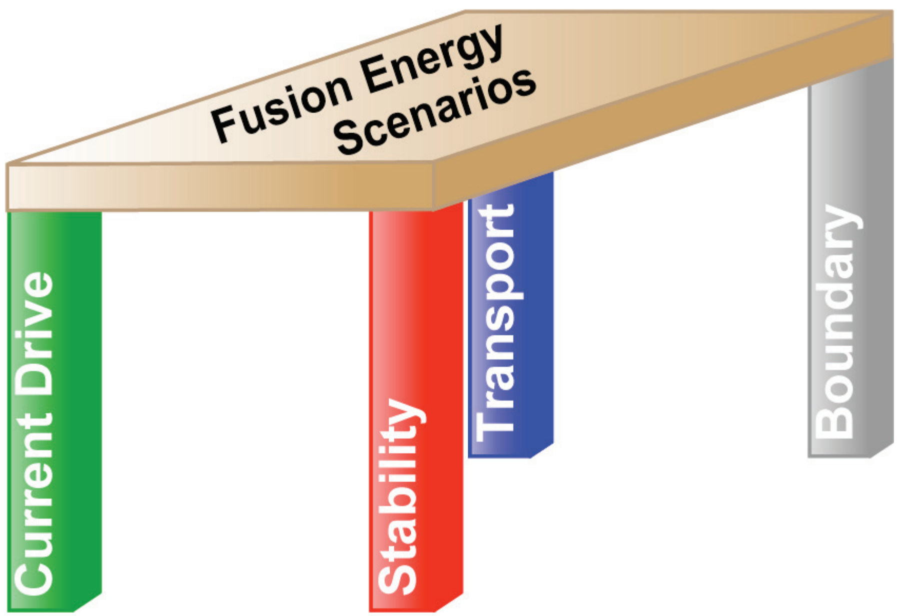

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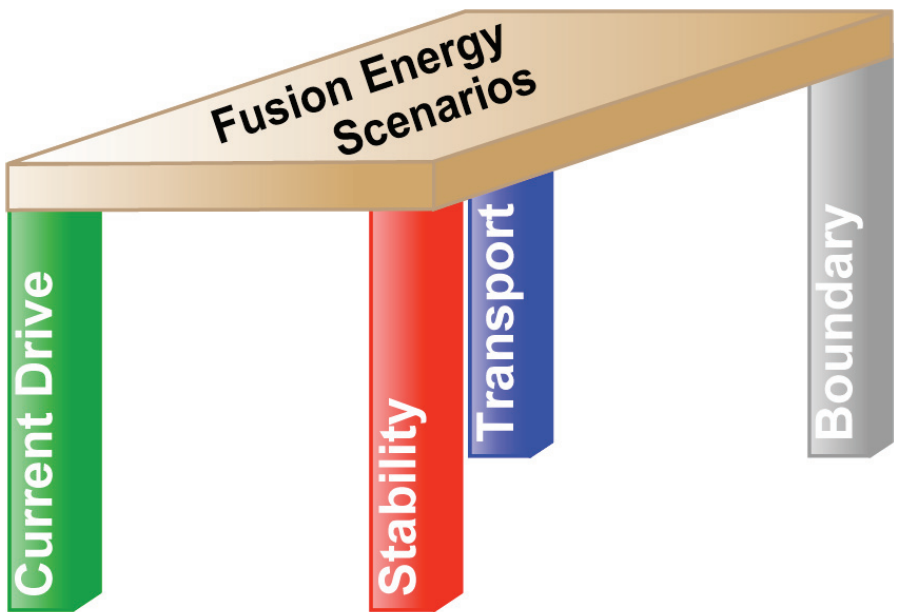
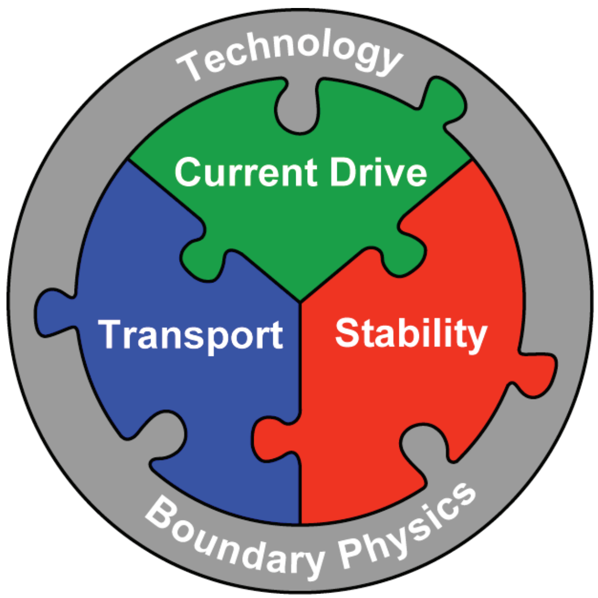
- **Steady-state tokamak research is exciting and challenging physics, but how should these challenges be met?**

- **Traditionally, fusion research is divided into areas of specialization:**

- **Steady-state tokamaks must be self-heated and self-sustained ➔ Interaction of the elements must be treated explicitly:**





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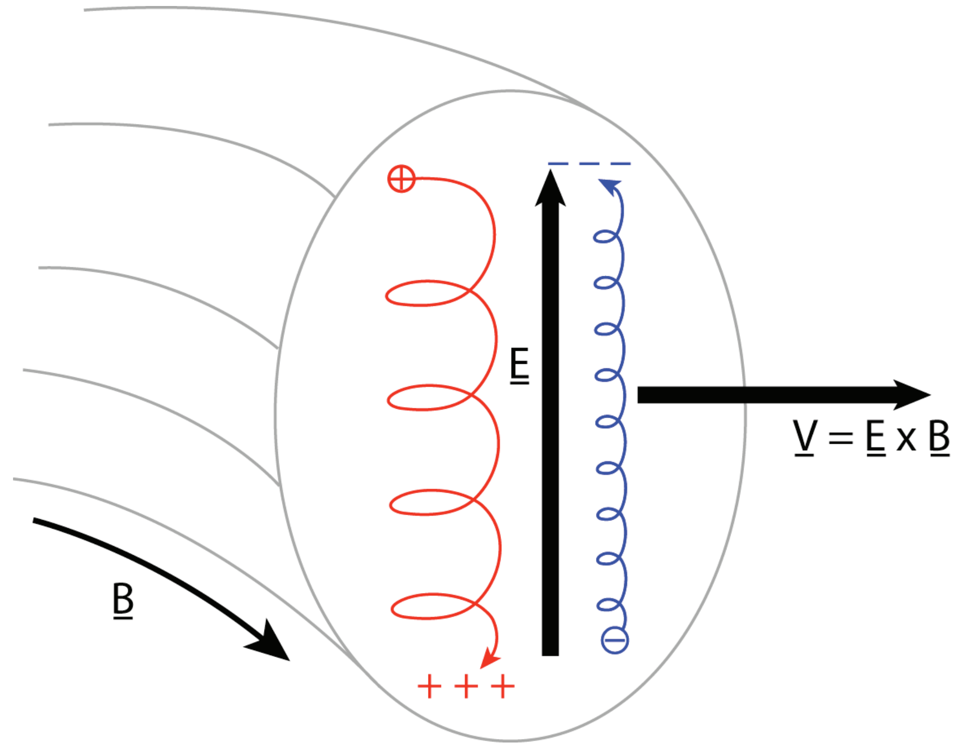
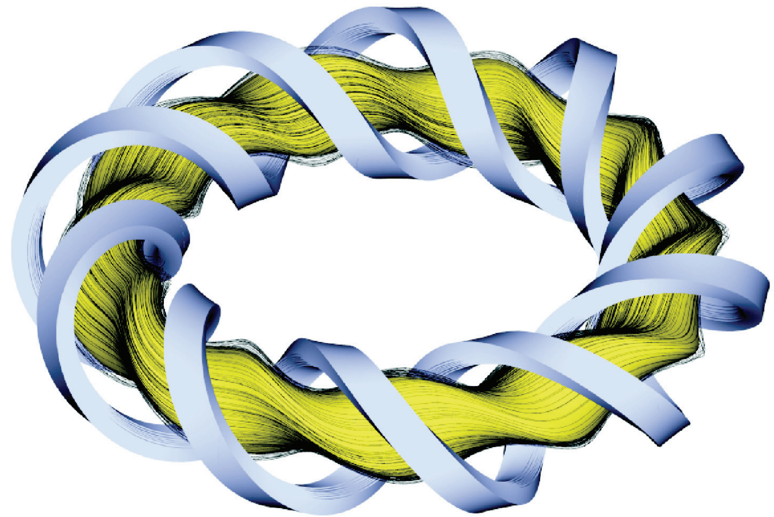
The Fundamental Challenge of the Steady-State Tokamak Starts from the Magnetic Configuration

- **Toroidal magnetic field eliminates losses along the field**
- **Single-particle motion in a purely toroidal field leads to charge separation**
- **The resulting electric field gives outward ExB drift**
  ➔ no confinement
- **Need to short out the electric field**
  ➔ use a helical magnetic field



Two Complementary Approaches to Toroidal Magnetic Equilibrium Are Being Pursued

Both schemes apply a strong external toroidal field for good confinement:



- **Stellarator achieves equilibrium with 3-D magnets**
  - Magnetic configuration is inherently steady-state
  - Confinement reduced due to loss of axial symmetry
  - Stellarator research is studying “quasi-symmetry” to improve confinement while maintaining steady state capability

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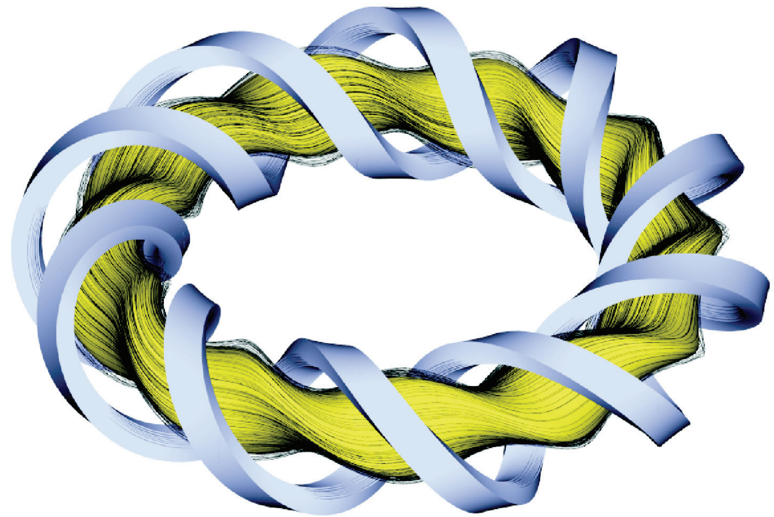
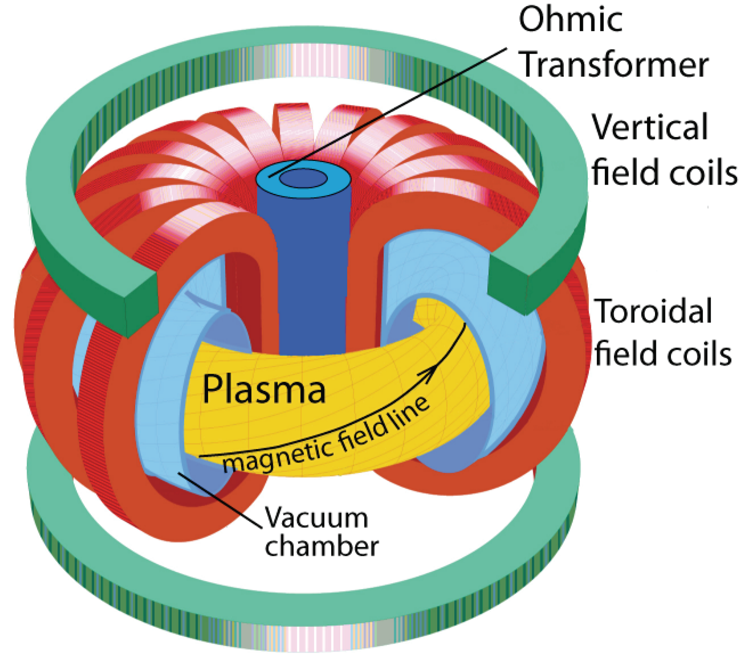
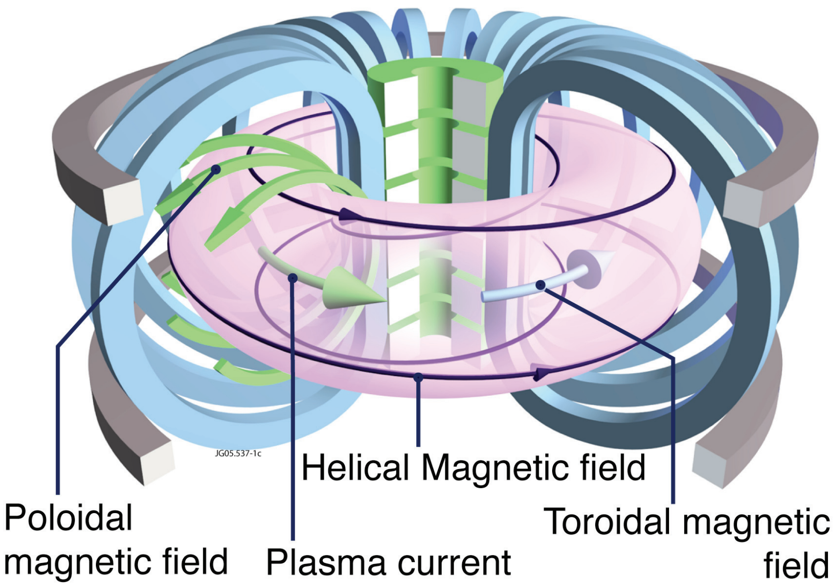
- **Tokamak uses plasma current to achieve equilibrium with axial symmetry**
  - Good confinement, but inductive current drive precludes steady state
  - Non-inductive current generation is essential for steady-state operation





Definition of Tokamak Equilibrium and Scenario Terms



Definitions:

- $\beta \equiv 2\mu_0\langle p\rangle/\langle B^2\rangle$ -- ratio of thermal to magnetic pressure
- $q \equiv$ # of toroidal turns/poloidal turn
- Stationary ≡ all plasma quantities are constant
- Steady-state ≡ all plasma and plant quantities are constant

Graphic courtesy S. Pinches, CCFE

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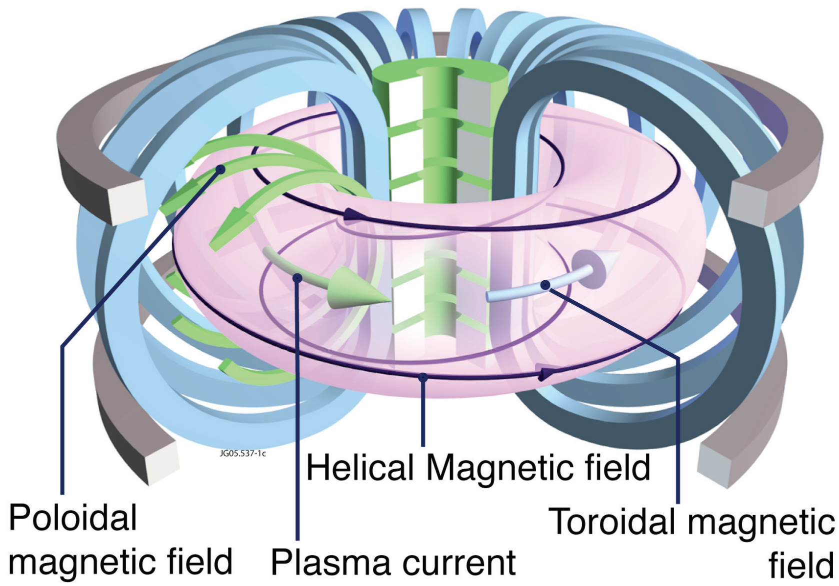

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Plasma physics timescales in a power plant:
Ideal MHD (< 1 ms), Resistive MHD (~ 10 ms),
Energy confinement (~ 1 s), Current relaxation (≥ 100 s)

Graphic courtesy S. Pinches, CCFE

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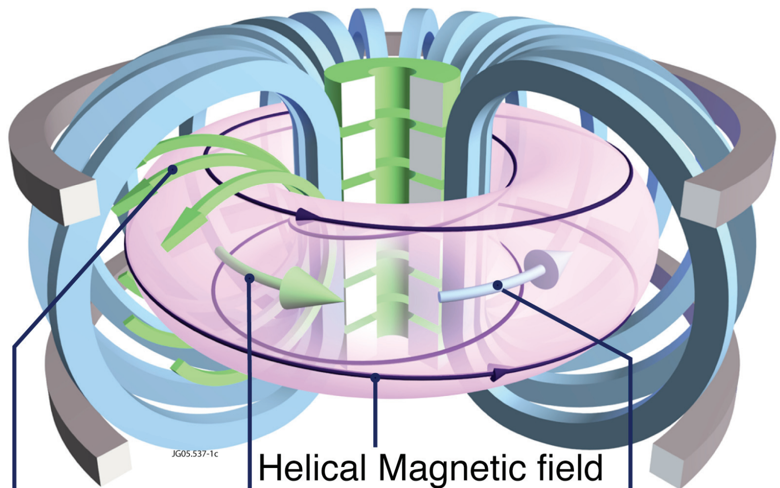
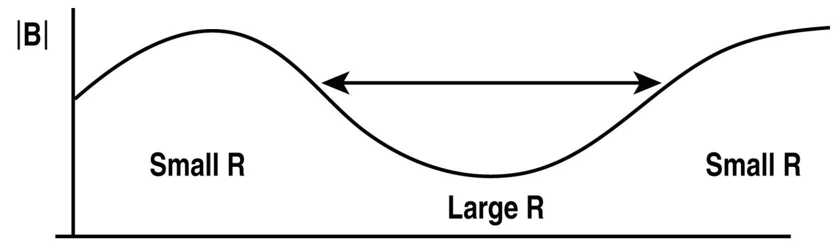


Particles can be trapped near B_{min} due to conservation of energy and magnetic moment:

- $\mu = mv_{\perp}^2 / 2B$
- $E = m(v_{\parallel}^2 + v_{\perp}^2) / 2$



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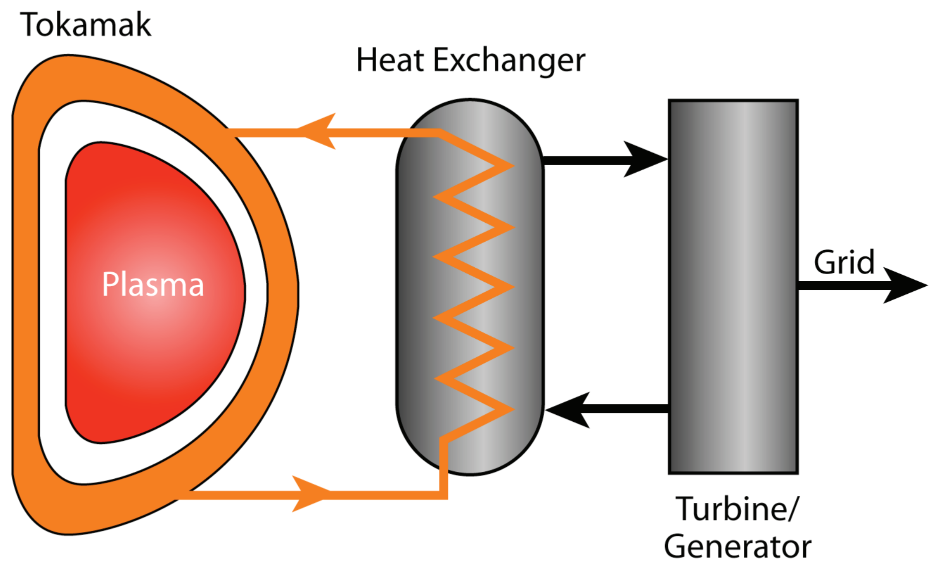
Power Plant Optimization Motivates Steady-State Tokamak Operation

- **Efficient conversion of heat to electricity requires a constant heat source**
  - Lifetime of heat exchangers limited if temperature deviates by more than a few degrees
  - Time without fusion power between tokamak pulses is too long (>100 s) for a thermal reservoir to be practical
  - Probability of unscheduled outage must be very small
    ➔ high reliability needed



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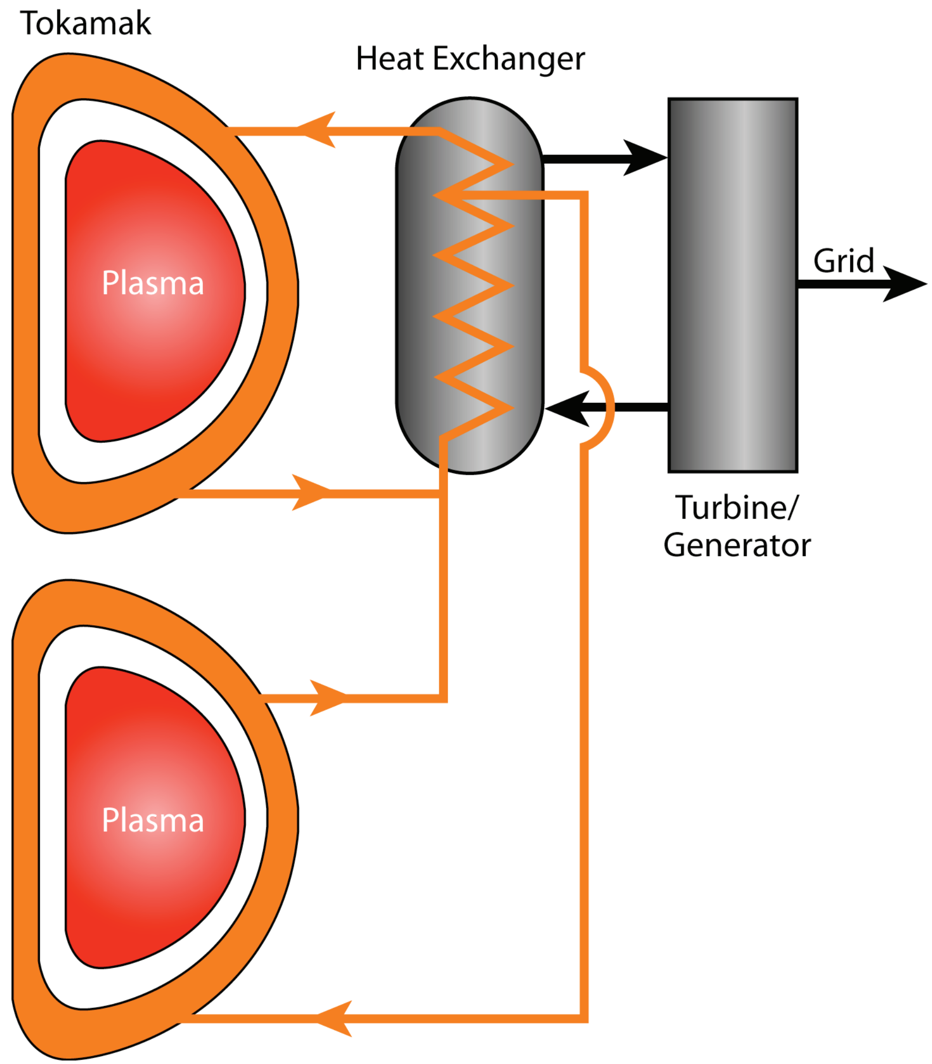
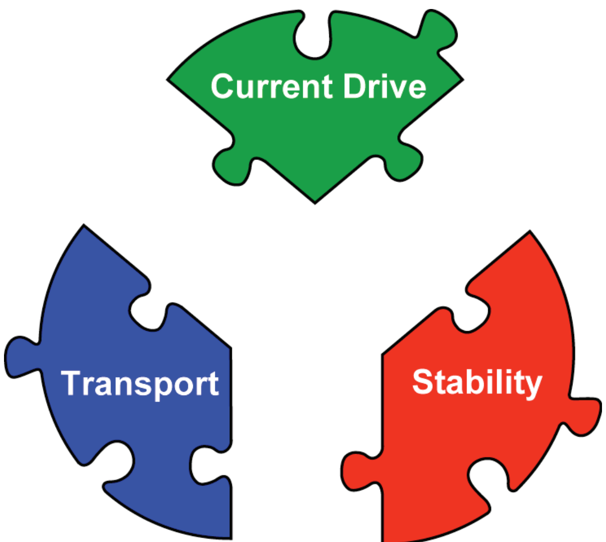
- **Pulsed tokamaks can be used in pairs to yield constant heat source ➔ capital cost increases for fixed power**



Outline

- Introduction (complete!)

- How is current generated in steady-state?



- Are transport and current profile related?

- How does stability depend on transport and current profile?

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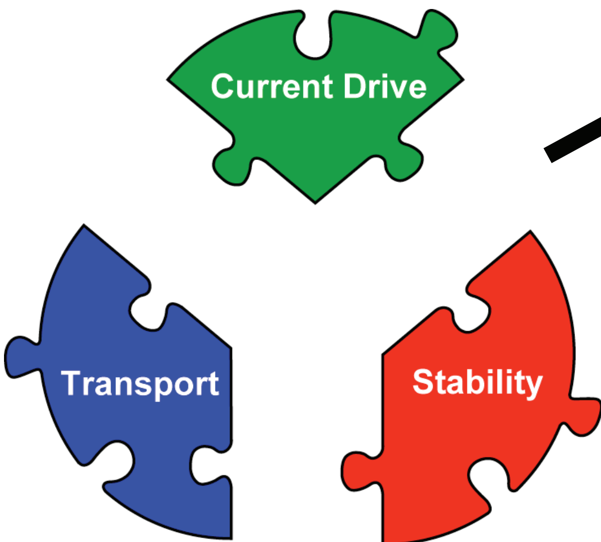

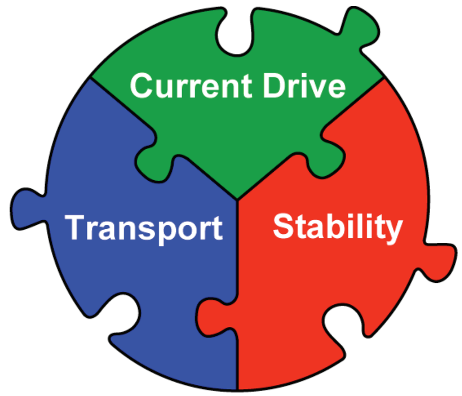

- Integration of the elements into a stationary core scenario
- How does stability depend on transport and current profile?
- Are transport and current profile related?

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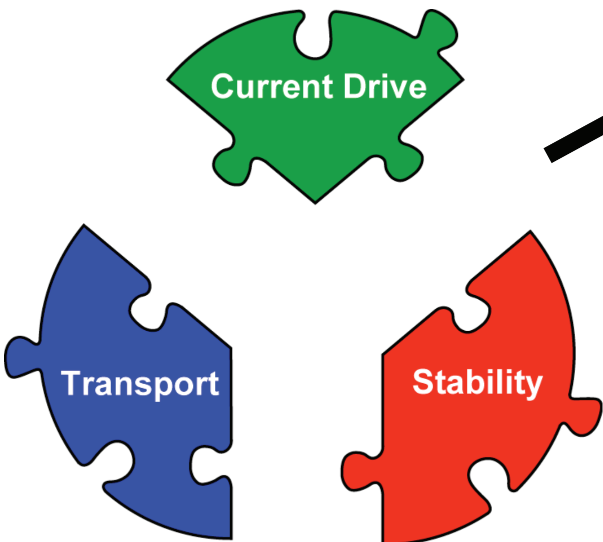

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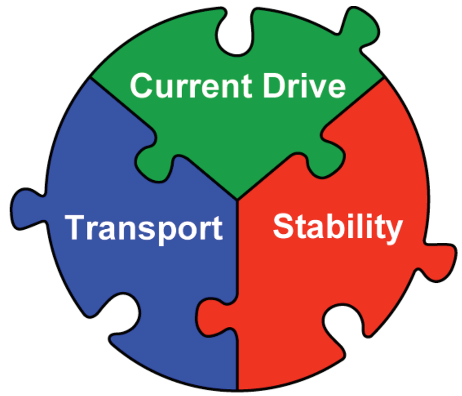
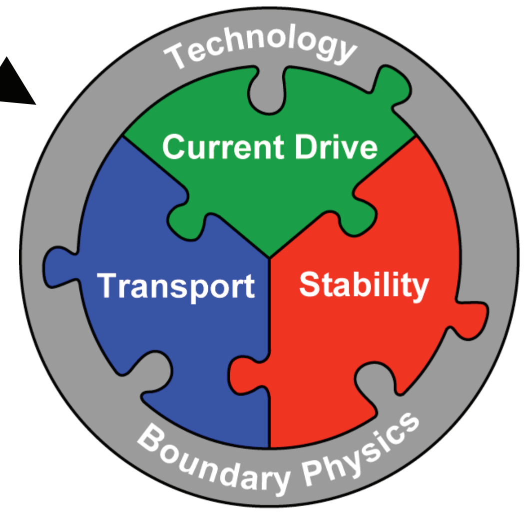


- Integration of the elements into a stationary core scenario
- Integration of the core scenario into an attractive energy system



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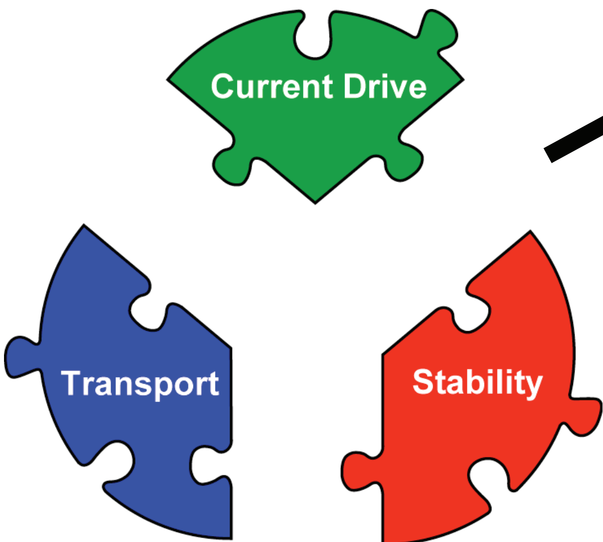

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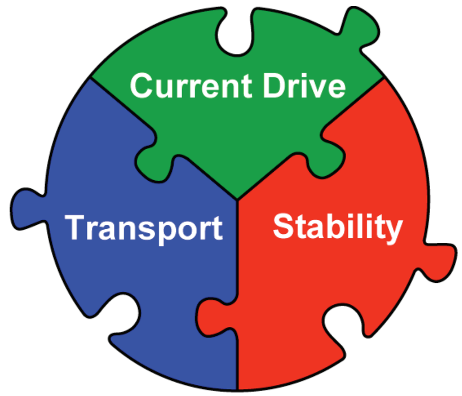

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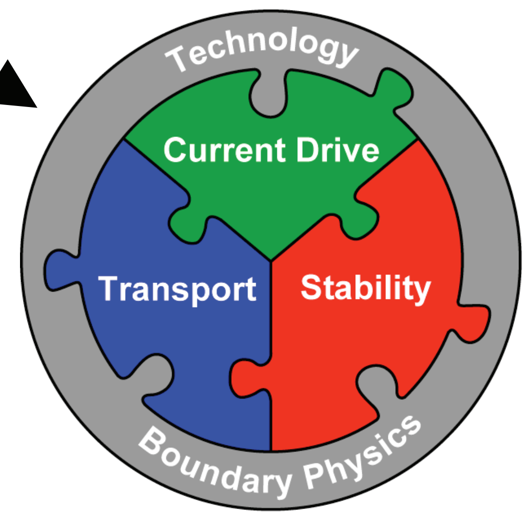
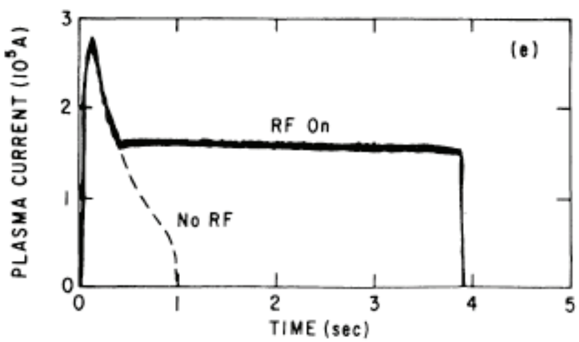


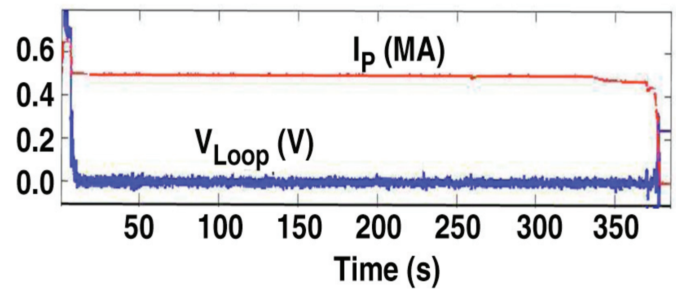
- Near-term challenges
- Conclusions

Steady-State Tokamak Operation by Non-Inductive Current Drive Has Been Clearly Demonstrated

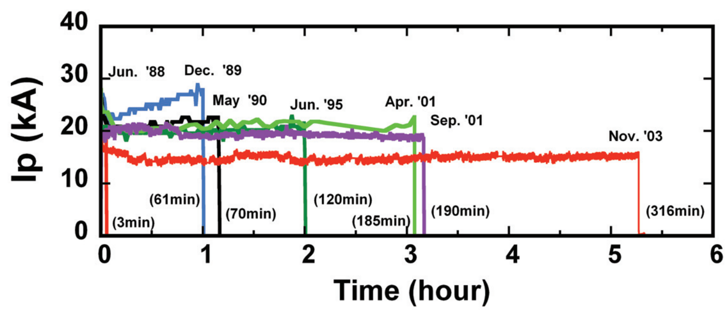
Sustaining the plasma current by non-inductive means is essential to steady-state tokamak operation



PLT (1982)
>3 sec



Tore Supra (2004)
>300 sec



TRIAM-1M (2003)
>18000 sec

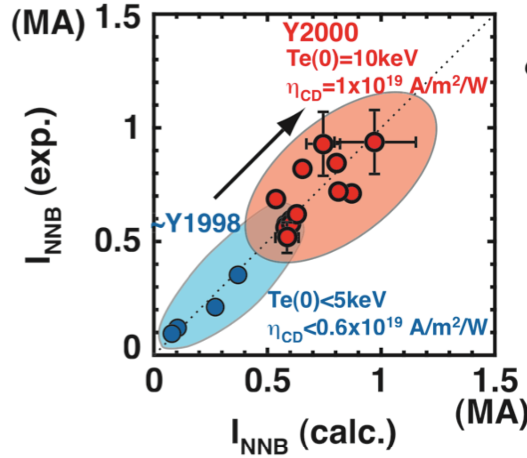
All of these results were achieved with lower hybrid waves, but at conditions far from those required in fusion power plants.

Current Drive

Physics of Current Drive Methods Used Routinely In Scenario Development Has Been Validated

- **Injection of neutral beams ($E_{beam} >> T_i$)   (ex. JT-60U)**
  - Current deposition likely fixed in power plant environment

0-D validation



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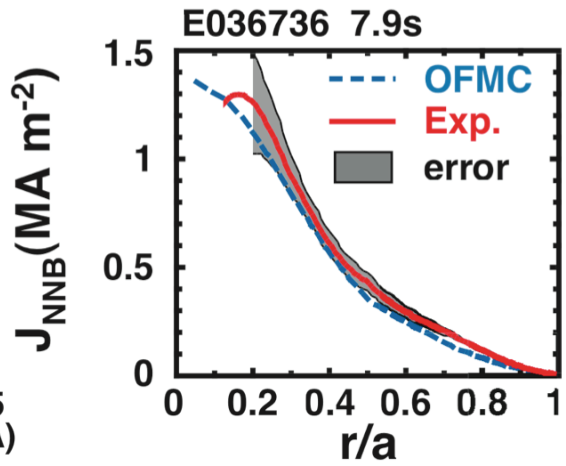

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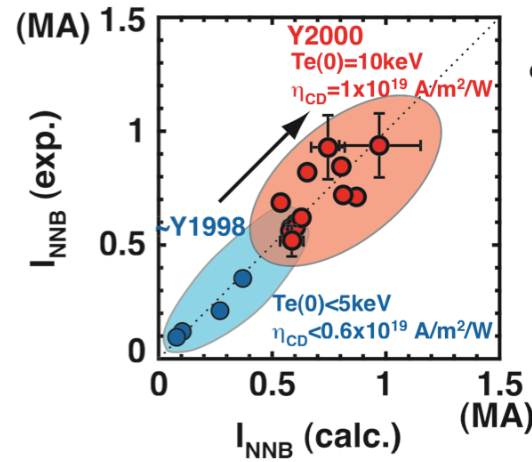

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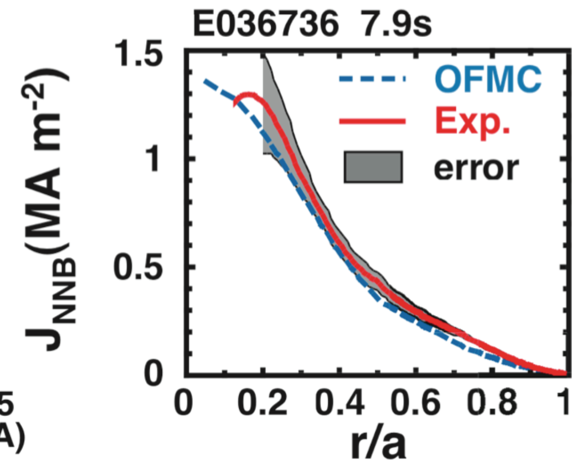

- **Injection of waves to the electron cyclotron resonance (ex. DIII-D)**
  - Current deposition could be varied in power plant environment

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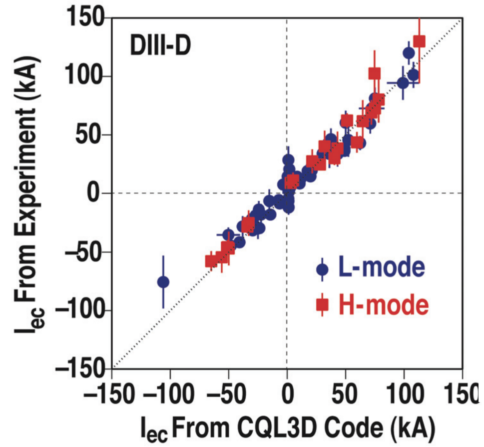

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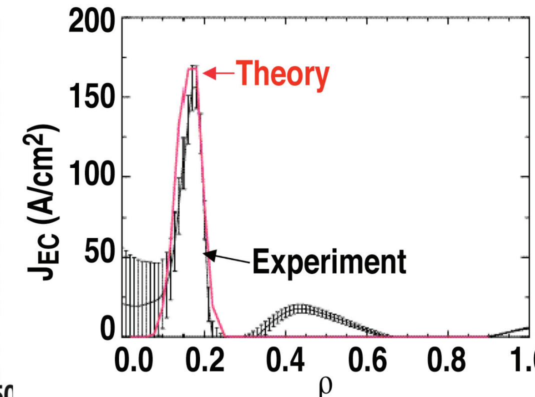

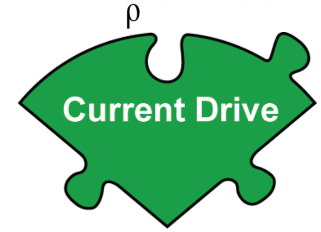

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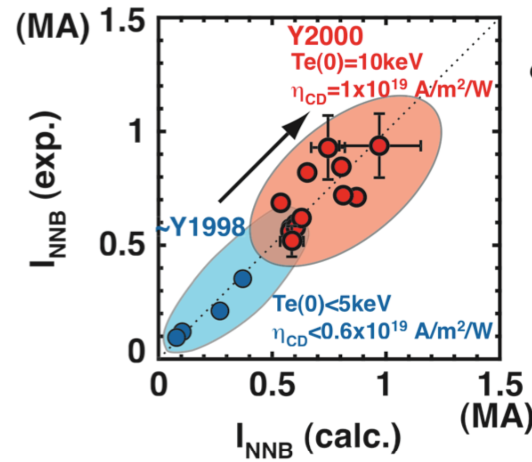

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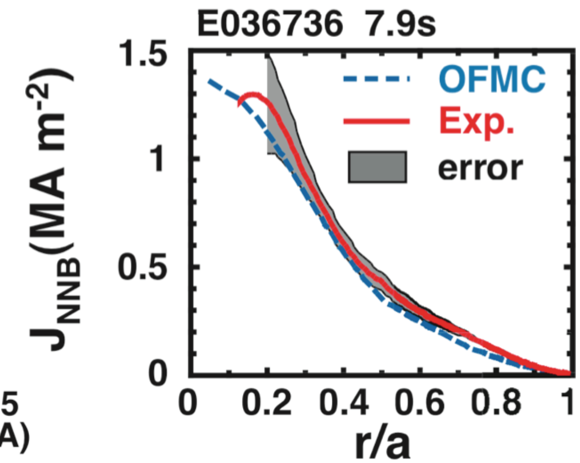

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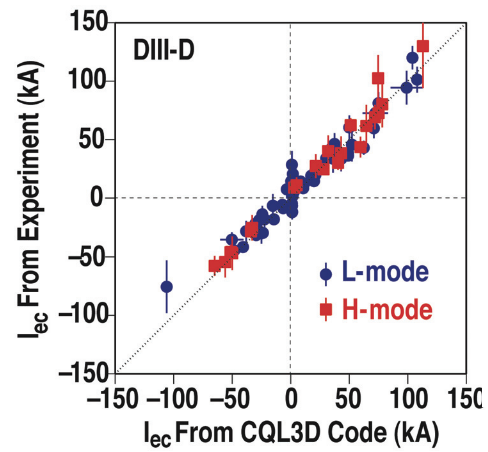

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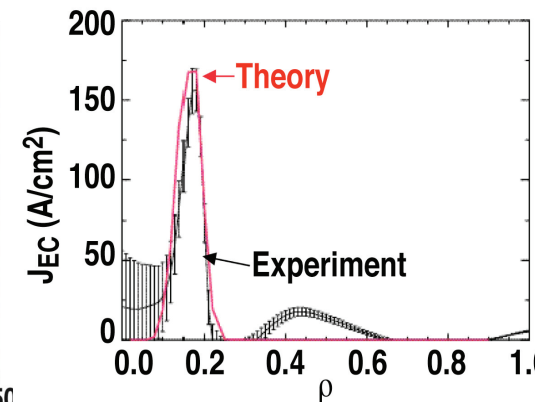
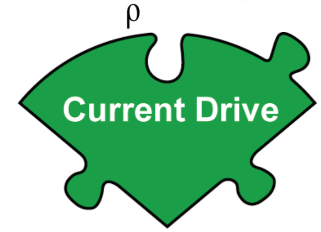


- **Both methods require only line of sight and no structures near the plasma**



External Current Drive Schemes Lead To High Recirculating Power Fraction

- **Both neutral beam and electron cyclotron current drive efficiencies under fusion conditions yield ~ 0.05 A/W**
- **Recirculating power fraction is given by:**
  - $f_{recirc} = P_{CDe} / P_{plant,}$ where $P_{CDe} = (I_{CD} / \gamma_{CD}) / \eta_e$
  - Take I_{CD} = 12.5 MA and $\eta_e = 0.5$
  - For P_{plant} = 1000 MWe, the recirculating power fraction f_{recirc} = 50% ➔ much too high for a practical power plant
- **To make the target of f_{recirc} = < 20%, it is necessary to find a factor of 2.5:**
  - Increase in I/P, or
  - Reduction in the demand for external current drive

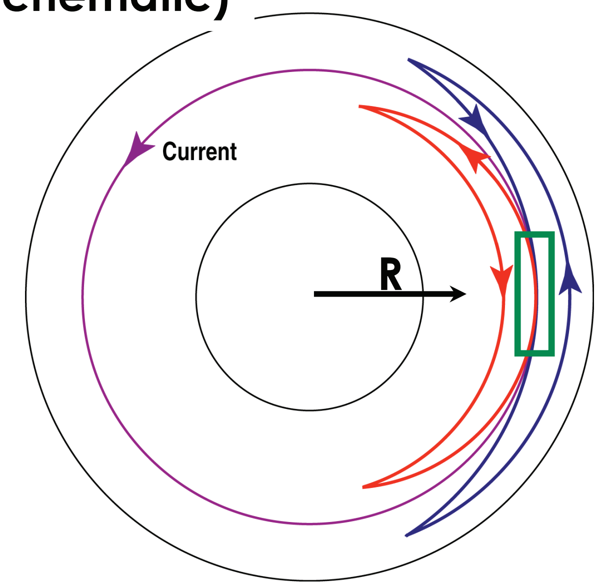
Tokamaks Have the Amazing Capacity to Generate Their Own Toroidal Current

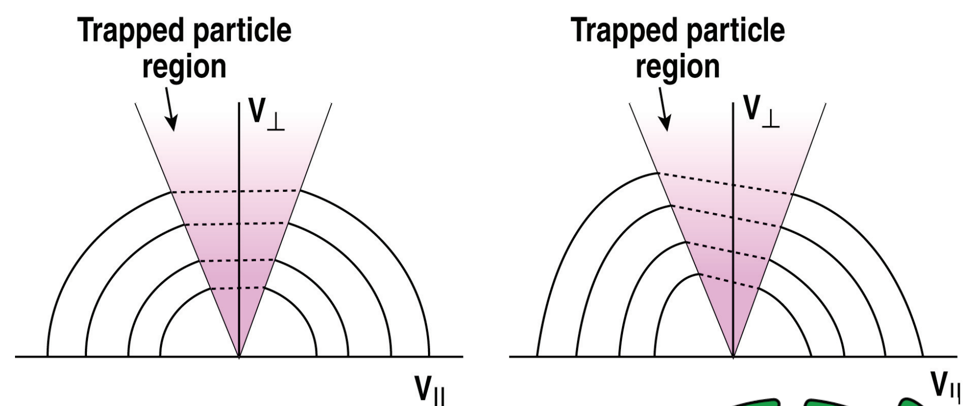
- **Due to gradients in density and temperature, more trapped particles are moving in the direction to drive current at any location**

- **The asymmetry in trapped particles (which can't carry current) is transferred to the passing particles by collisions**

- **"Bootstrap current" theory gives:**
  $j_{BS} \propto p_e \, q \;\; f(\nabla n/n, \nabla T/T)$

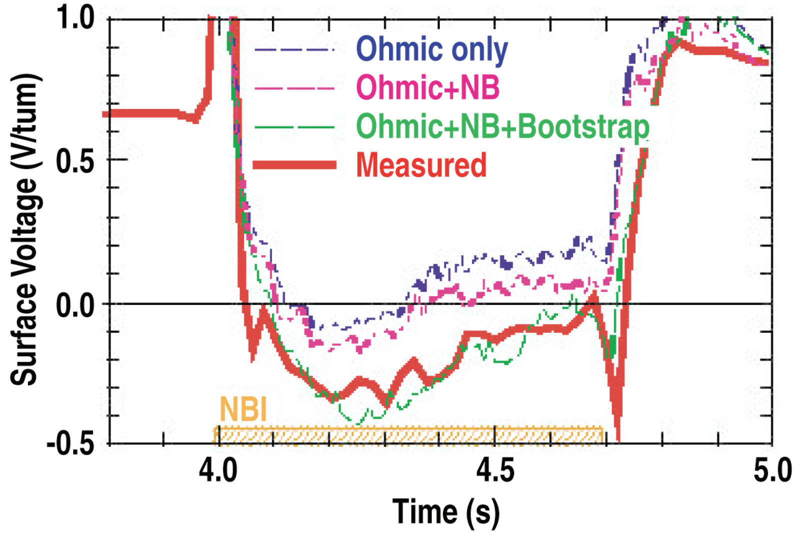
Tokamak Top View with Ion Orbits (Schematic)





Bootstrap Current Theory Has Been Validated on Several Toroidal Devices

- **Clear experimental verification of bootstrap current in tokamaks**
  - Verification also on toroidal multipoles and stellarators



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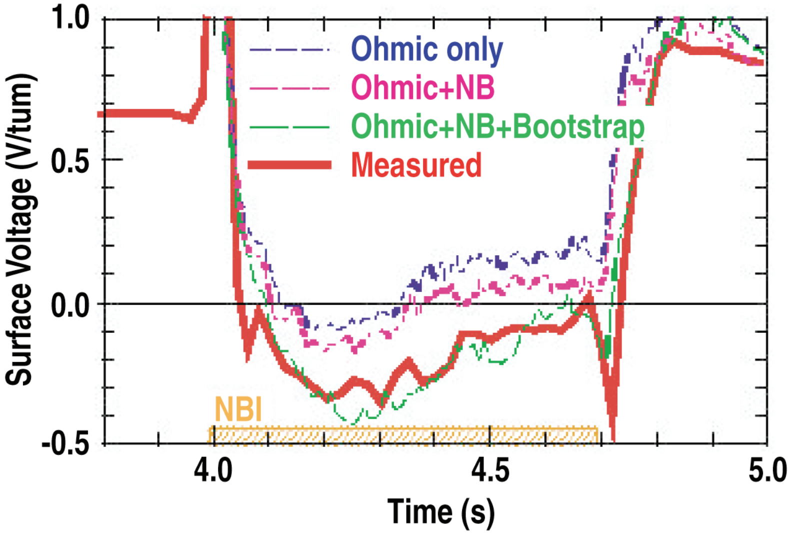
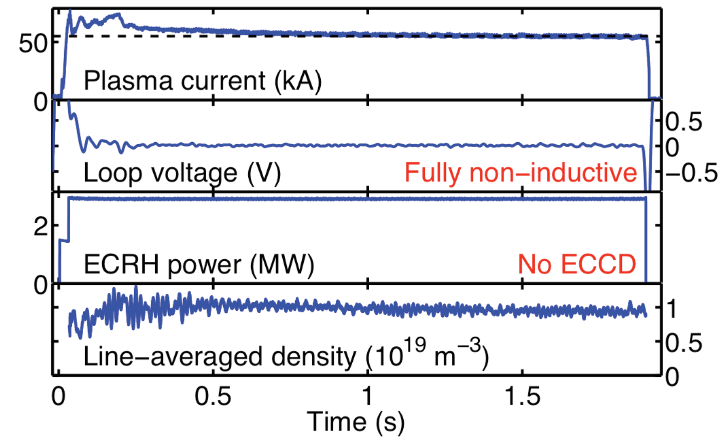


TFTR

0-D validation TCV



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  - Balanced EC wave injection

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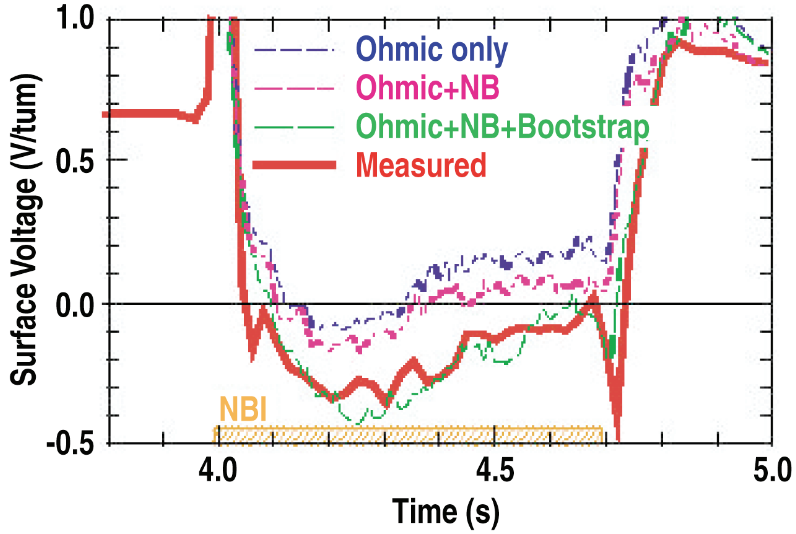
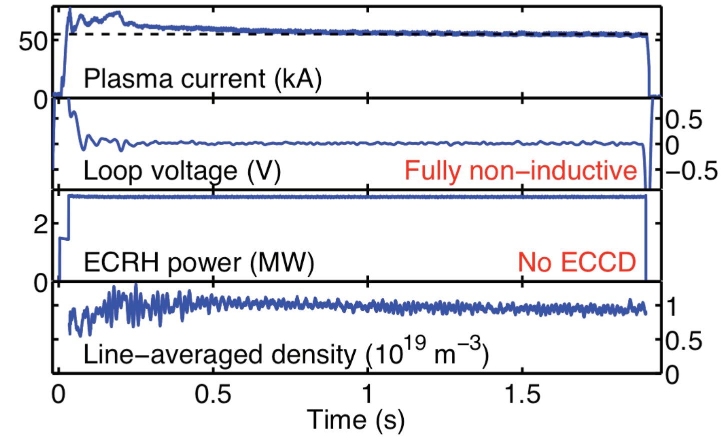


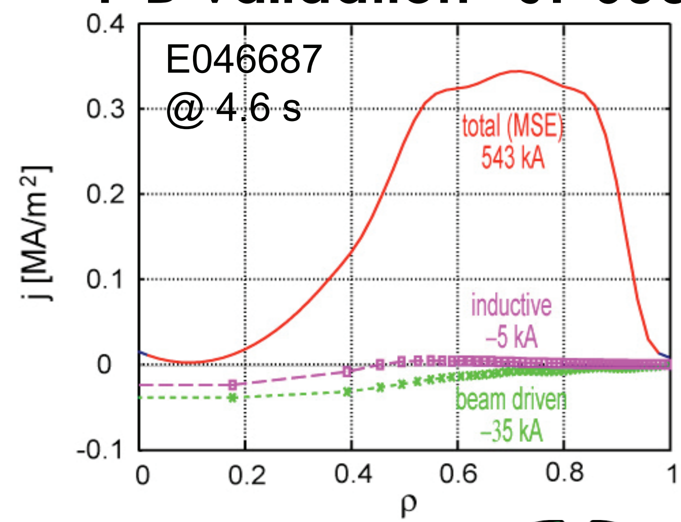
TFTR

- **100% bootstrap current demonstrated under two very different conditions**
  - Balanced EC wave injection
  - Balanced neutral beam injection

0-D validation TCV



1-D validation JT-60U



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High Bootstrap Current Requirement For Steady State Links Strongly The Current Profile and Transport

• **Physics of external current drive is well understood, but:**

• **External current drive has low efficiency**
➔ f_{BS} > 0.6 required

High Bootstrap Current Requirement For Steady State Links Strongly The Current Profile and Transport

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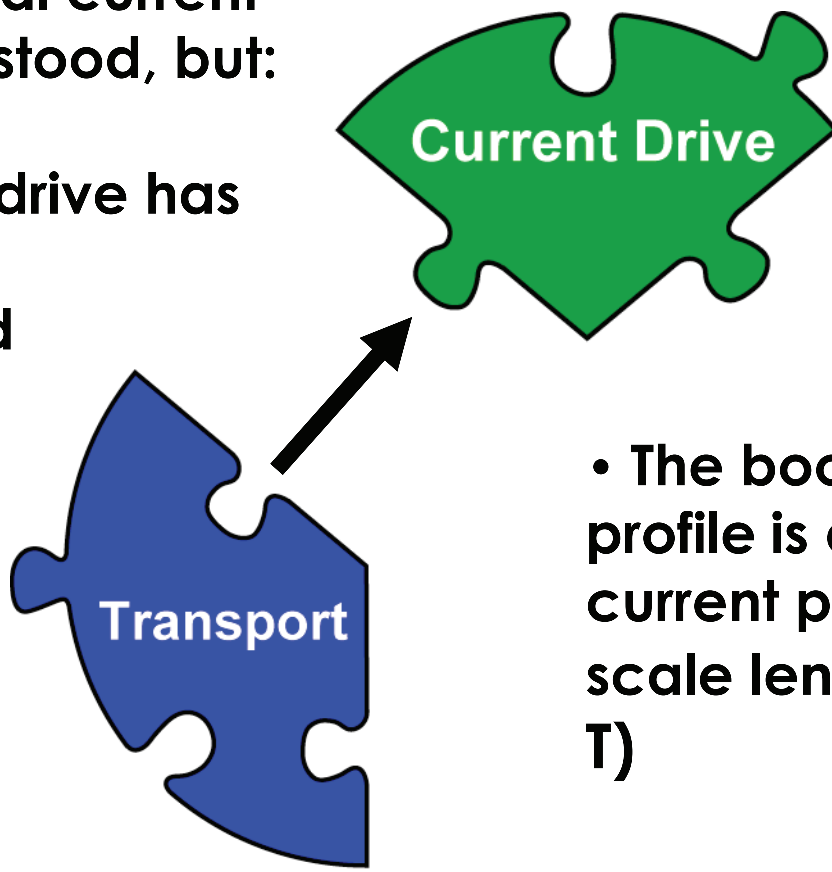

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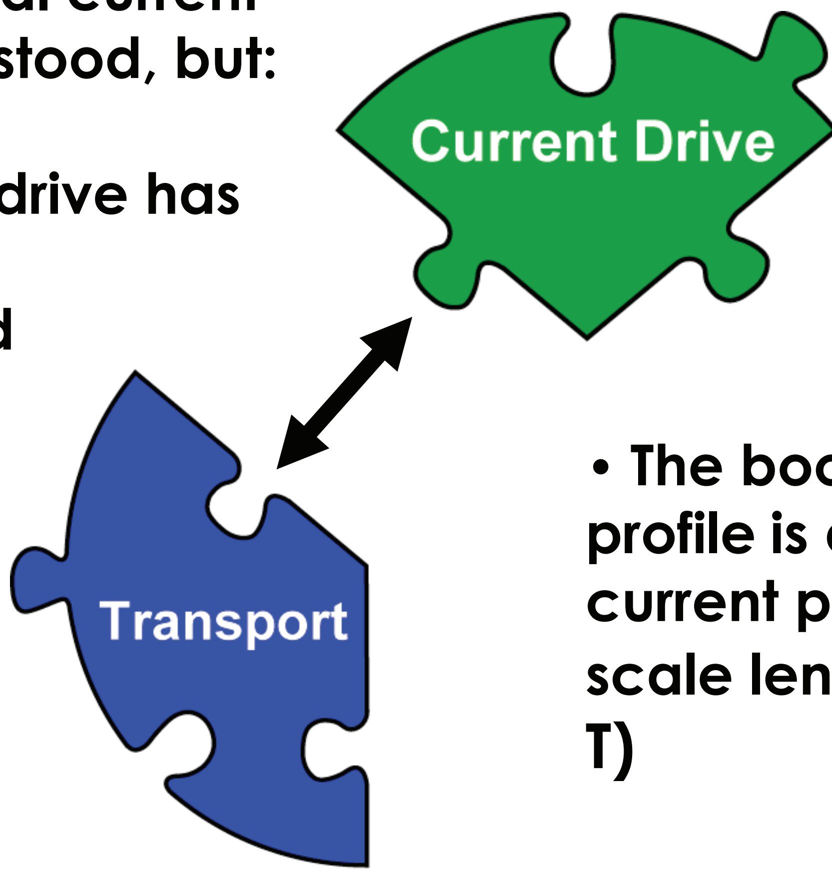


- **The bootstrap current profile is determined by the current profile (∝ *q*) and the scale lengths (∝ $\nabla n/n$, $\nabla T/T$)**

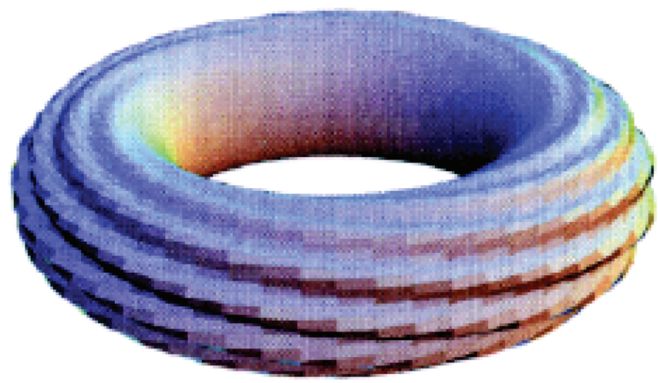
- **At high f_{BS}, the current profile depends strongly on transport ➔ motivates study of the dependence of transport on the current profile**

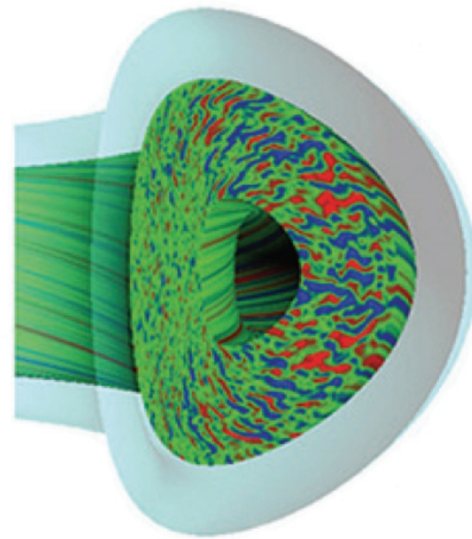
Basic Picture of Transport in Tokamaks

- **Cross-field transport in tokamaks is larger than predicted by collisional theory**
  ➜ Transport is due to plasma turbulence

- **Fluctuations likely responsible for transport are strongly aligned with the magnetic field and have a "ballooning" character**
  ➜ Transport may be sensitive to current profile

- **Transport of density, momentum, and temperature are seen to depend on each other--further challenge for optimization, but not discussed here**

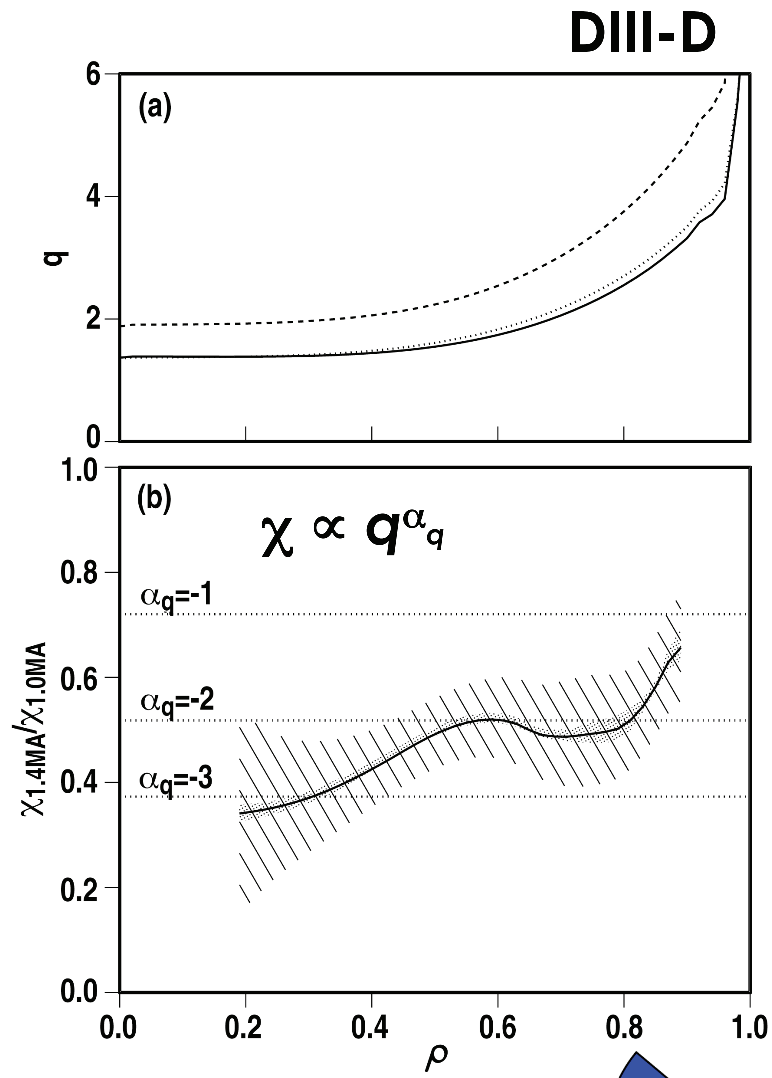




Transport

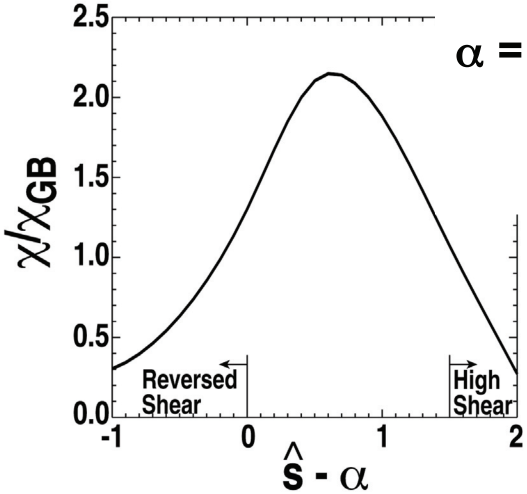
Dependence of Transport on q Is Well Documented

- **Linear scaling of global confinement with plasma current has been known for decades ($\tau_E \propto I$)**
- **Dimensionless parameter scaling experiments show a strong q dependence ($\chi \propto q^2$)**
- **Theoretical explanation of this dependence is challenging since the basic equations depend on |B|**
  - Leading explanation is that the linear damping terms have a $1/q$ dependence



Transport Depends Strongly on Magnetic Shear

- **Radial perturbations are strongly inhibited by magnetic shear $\hat{s}$ (variation of *q* with radius)**



α = normalized pressure gradient

Transport Depends Strongly on Magnetic Shear

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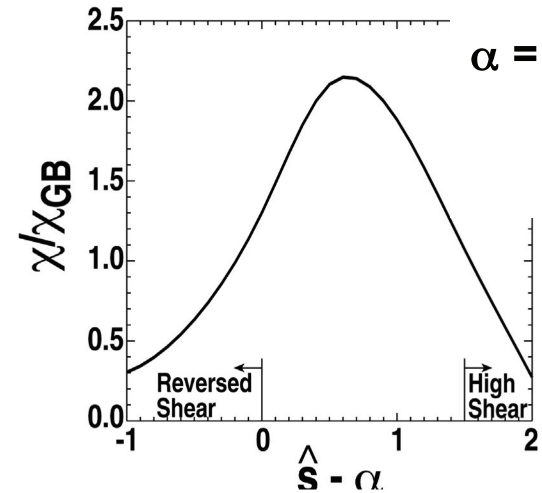
- **Reduced transport at high and low magnetic shear have been routinely demonstrated experimentally**



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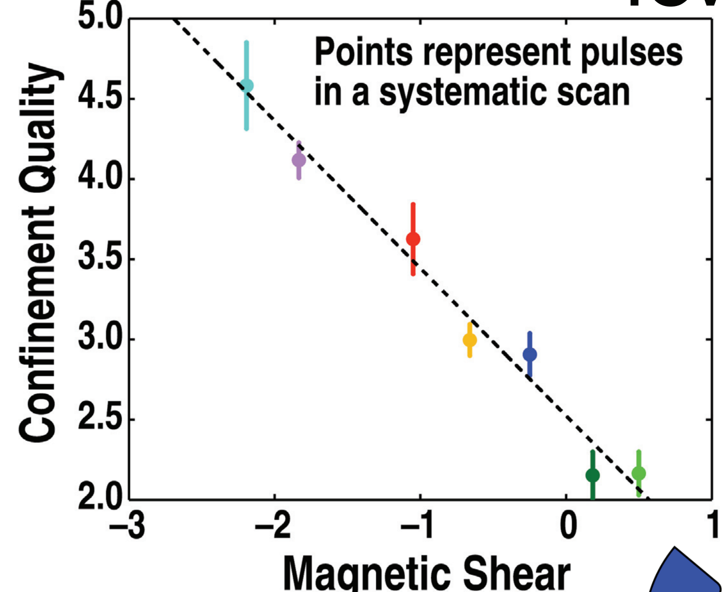
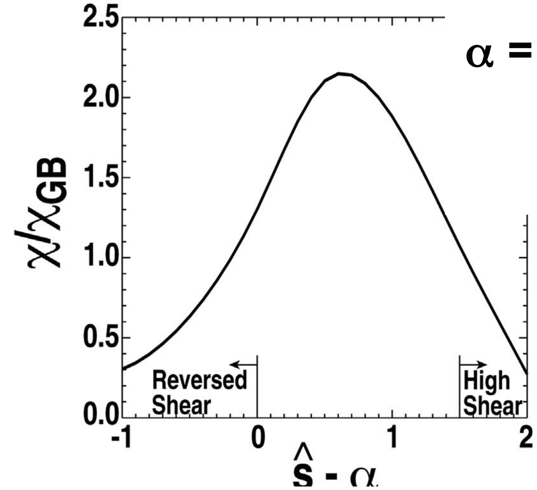
TCV



Transport

Transport Depends Strongly on Magnetic Shear

- Radial perturbations are strongly inhibited by magnetic shear $\hat{s}$ (variation of *q* with radius)
- Reduced transport at high and low magnetic shear have been routinely demonstrated experimentally
- At low shear, the steepening profiles (larger α) can lead to positive feedback, called an Internal Transport Barrier (ITB)
- Velocity shear can play the same role, but a means for control in a power plant is unknown
  ➔ opportunity for breakthrough



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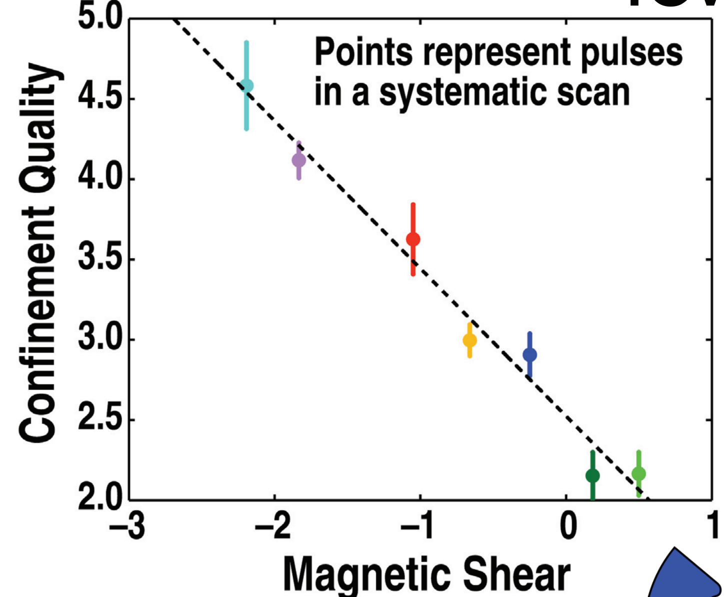
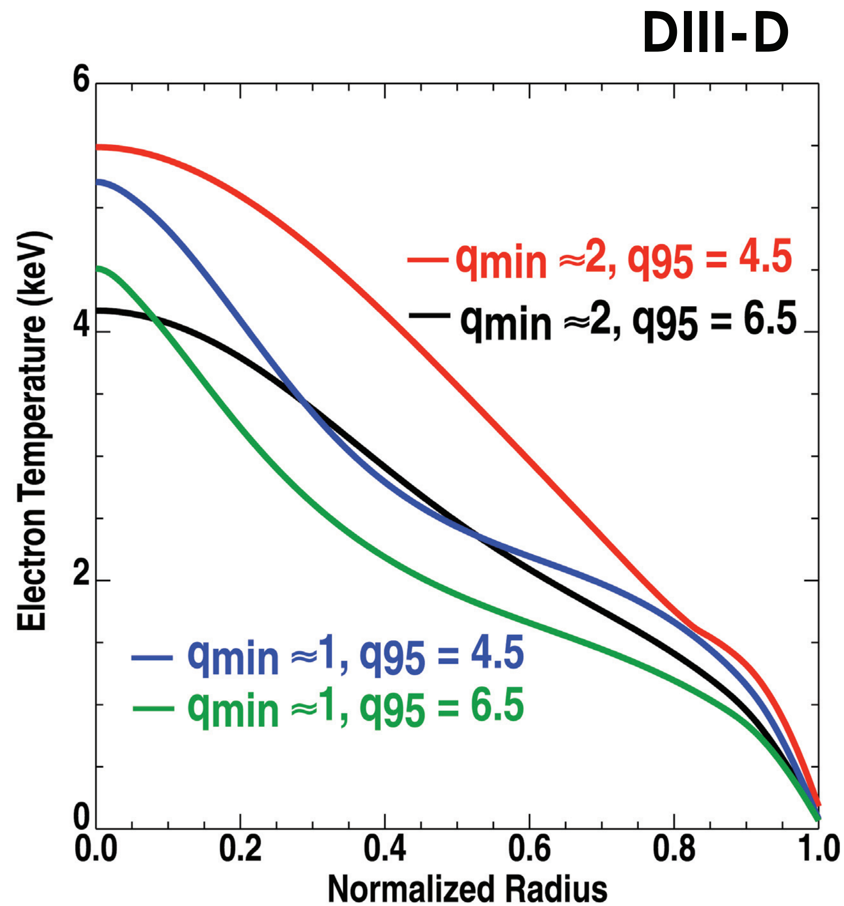


Transport

Connection Between Transport and Current Profile Is Key to Optimization of the Bootstrap Current

- **Experiment clearly shows response of temperature profiles to changes in the current profile**
  - Profile shape responds to *q* and shear in the core
  - Profile height responds to total current through the pedestal height
- **At high f_{BS}, there will be a strong interaction of the transport and the resulting equilibrium current profile**



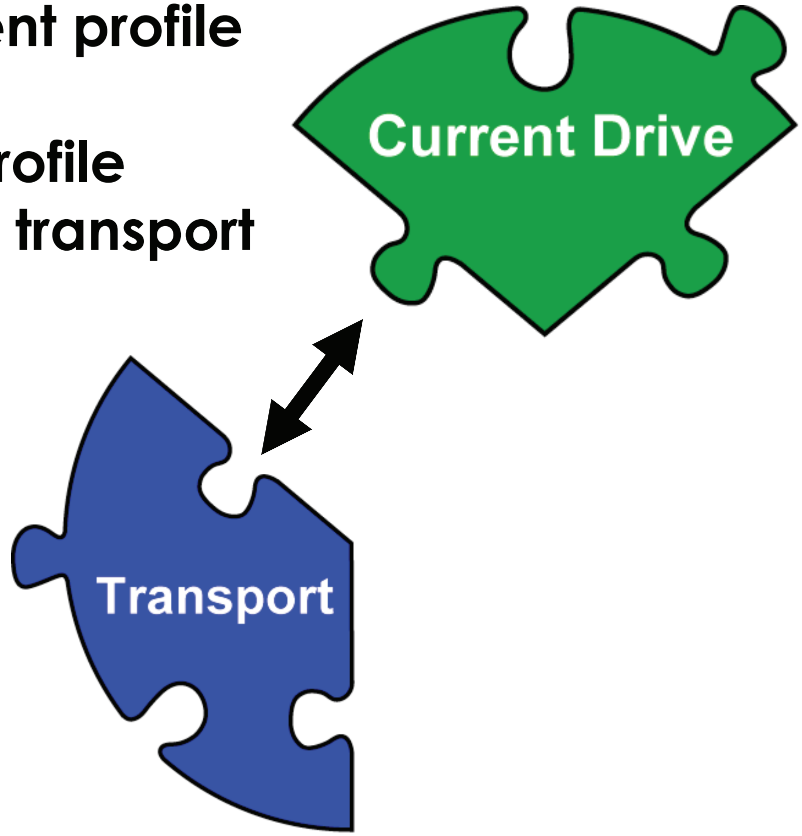
➔ **Optimization of the bootstrap current requires an accurate model of the transport dependence on *q*, $\hat{s}$**

Transport

Optimizing the Bootstrap Current For Steady State Must Include Stability of Current and Pressure Profiles

- Transport sets the bootstrap current profile

- The current profile strongly affects transport



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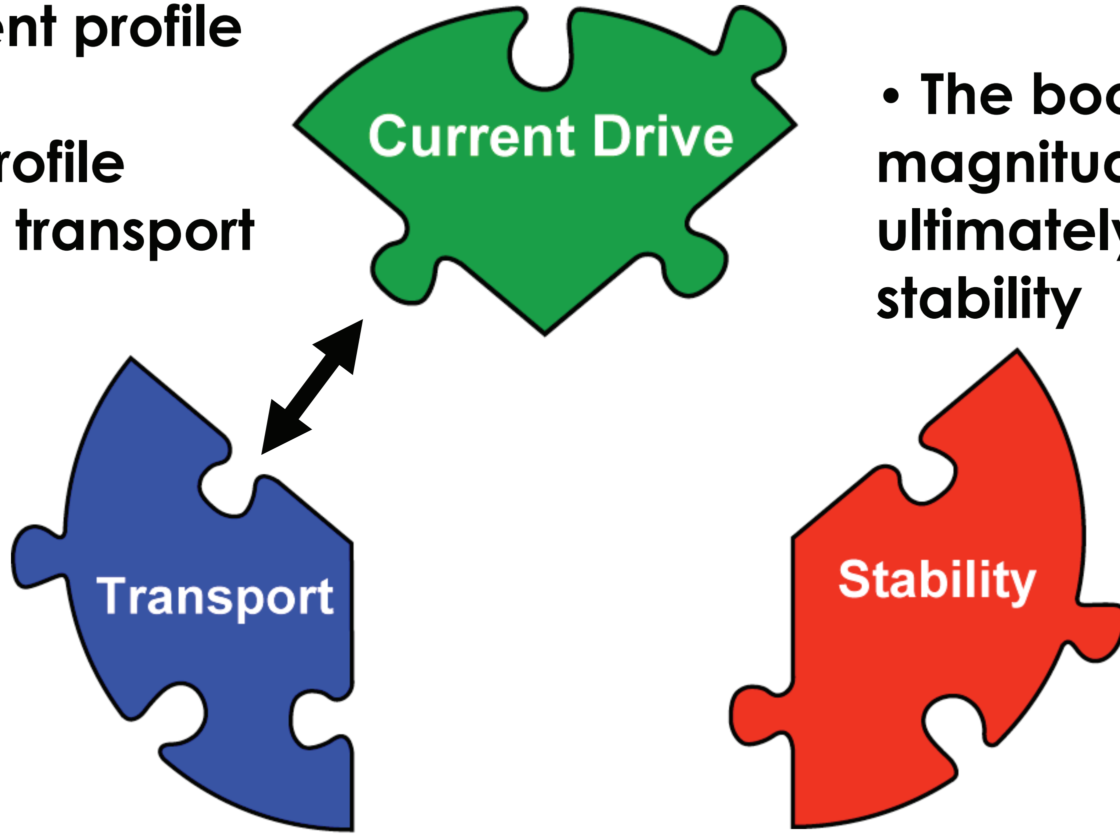

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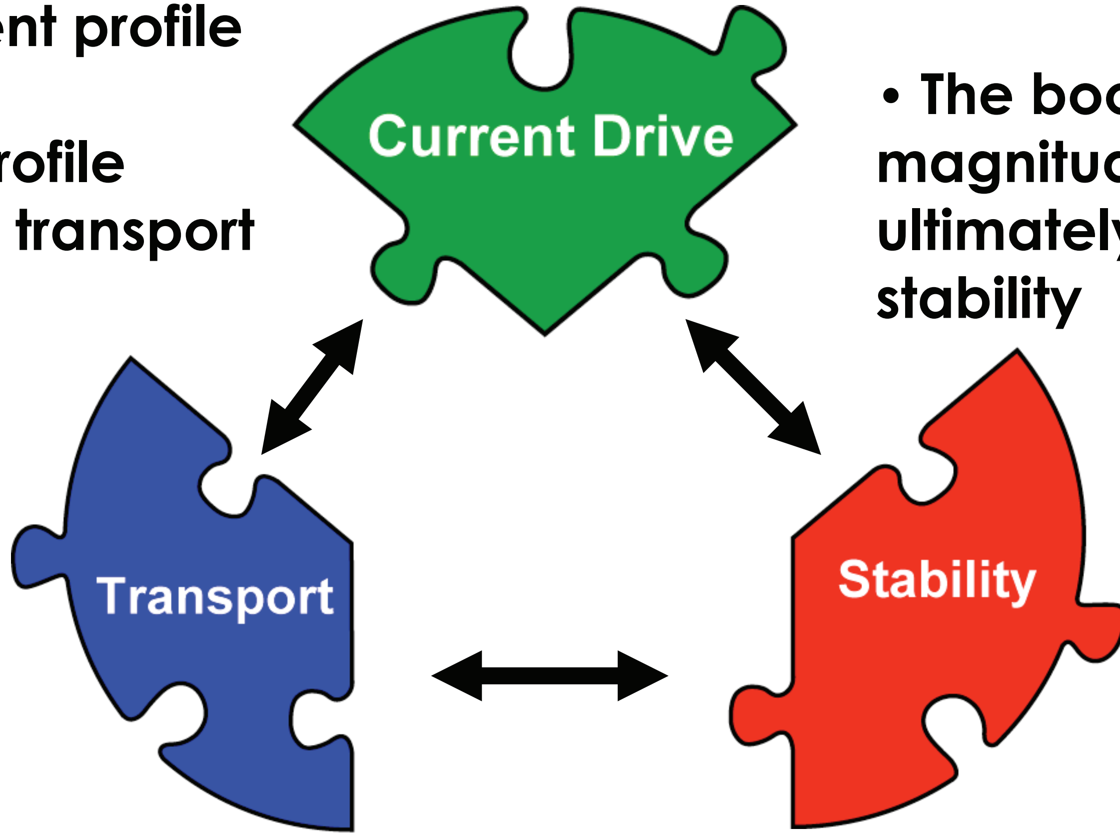


- The bootstrap current magnitude ($\propto$ p) is ultimately limited by stability

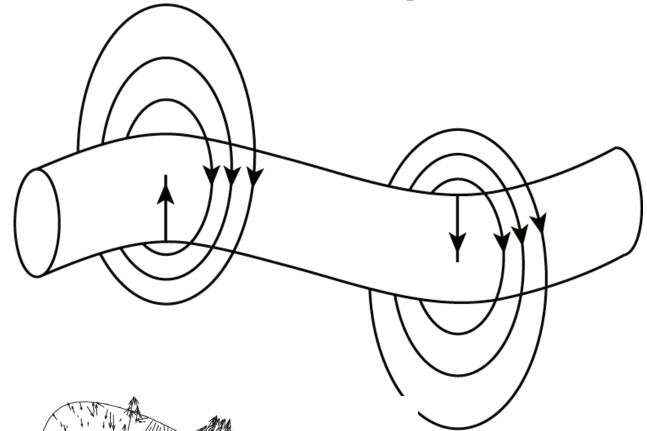
- To obtain high f_{BS}, the dependence of stability on the current and pressure profiles must be considered

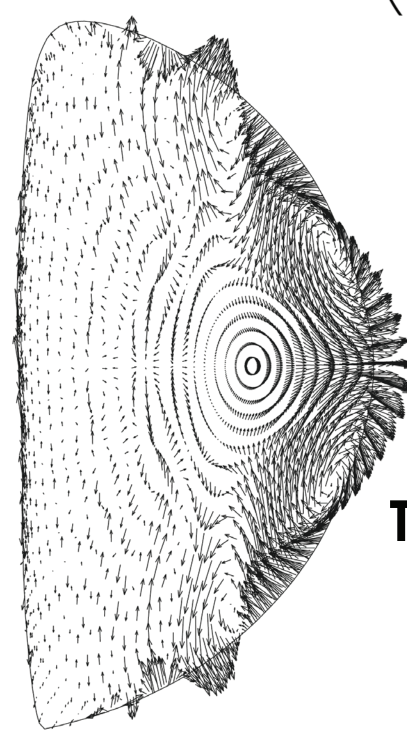
Tokamak Pressure Is Usually Limited by Long-Wavelength Kink Modes

- **Basic kink in a cylinder is unstable because the higher magnetic energy on the inner bend reinforces the initial displacement**

- **In a torus with finite pressure, the mode structure is more complex**
  - Mode structure is strongly concentrated to the low-field side
  - Mode is not radially localized

Cylinder ($\beta = 0$)



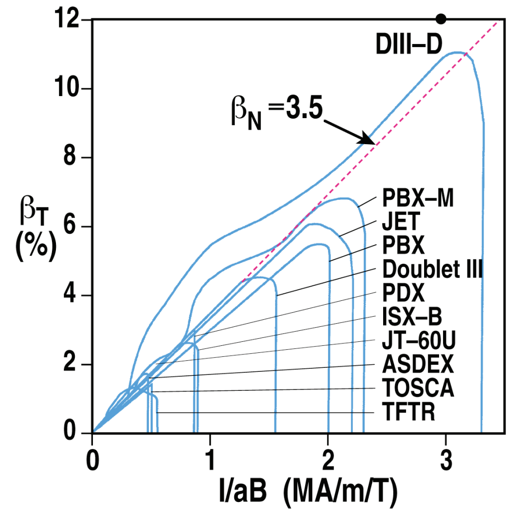


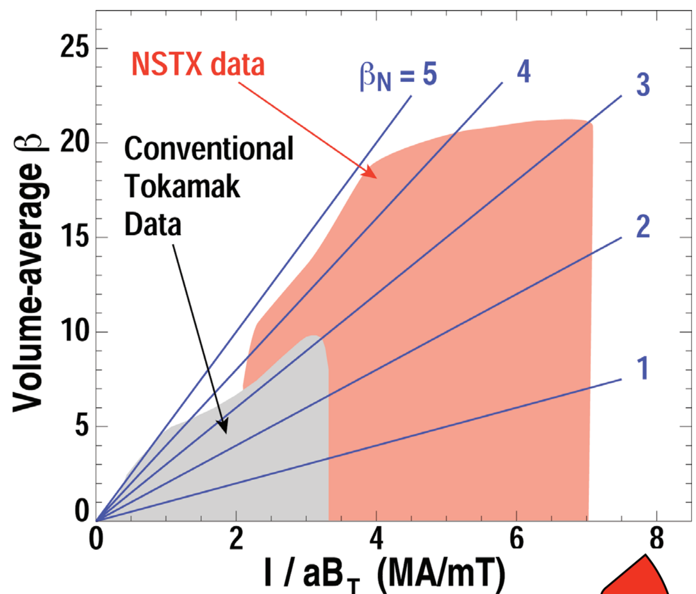
Torus ($\beta > 0$)

Stability

General Trends in Pressure Limits Are Characterized Well by Ideal MHD

- **Theoretical studies in the early 1980's showed that global stability limit on pressure was characterized by a single parameter $\beta_N \equiv \beta_T$ / (I/aB) = 3.5**
  - Maximum I/aB in experiment is limited by the kink limit at low safety factor *q*
- **Improvement beyond β_N = 3.5 can be attributed to several effects:**
  - Optimized cross-section shape
  - Optimized current profile
  - Optimized pressure profile
  - Wall stabilization

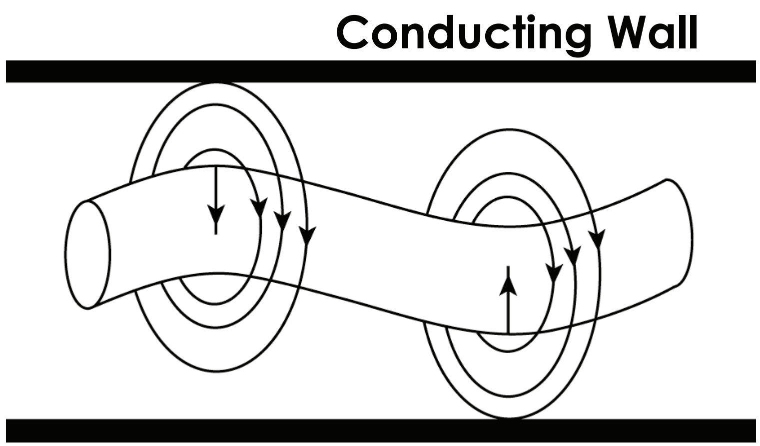




Stability

Conducting Wall Near the Plasma Allows Access to Higher Pressures

- **A perfectly conducting wall stabilizes the kink due to higher magnetic energy between the wall and the mode**



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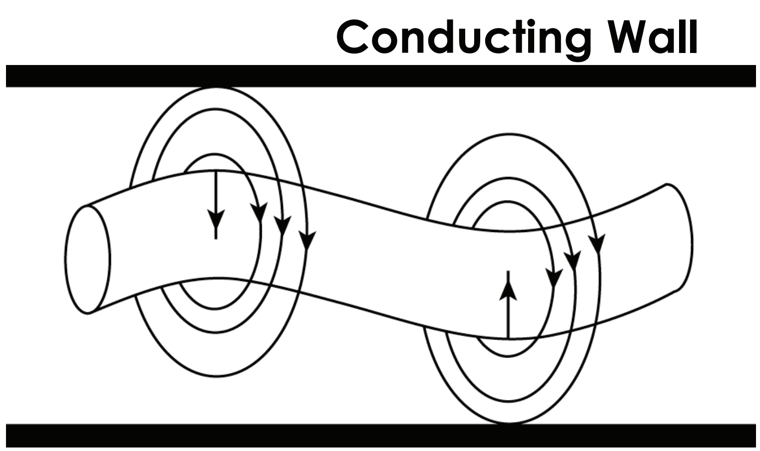
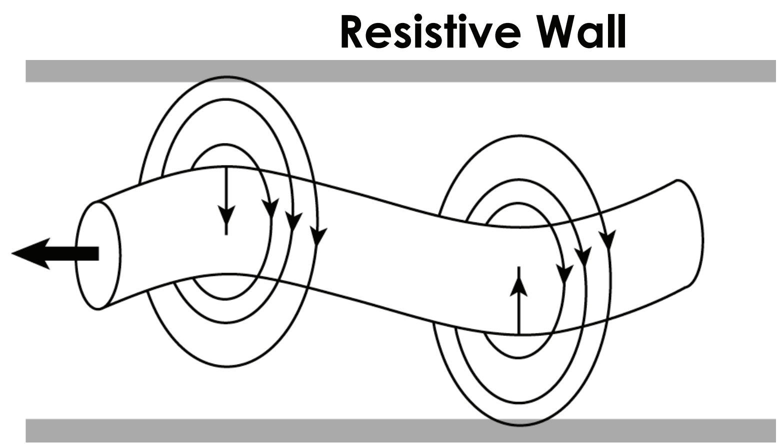
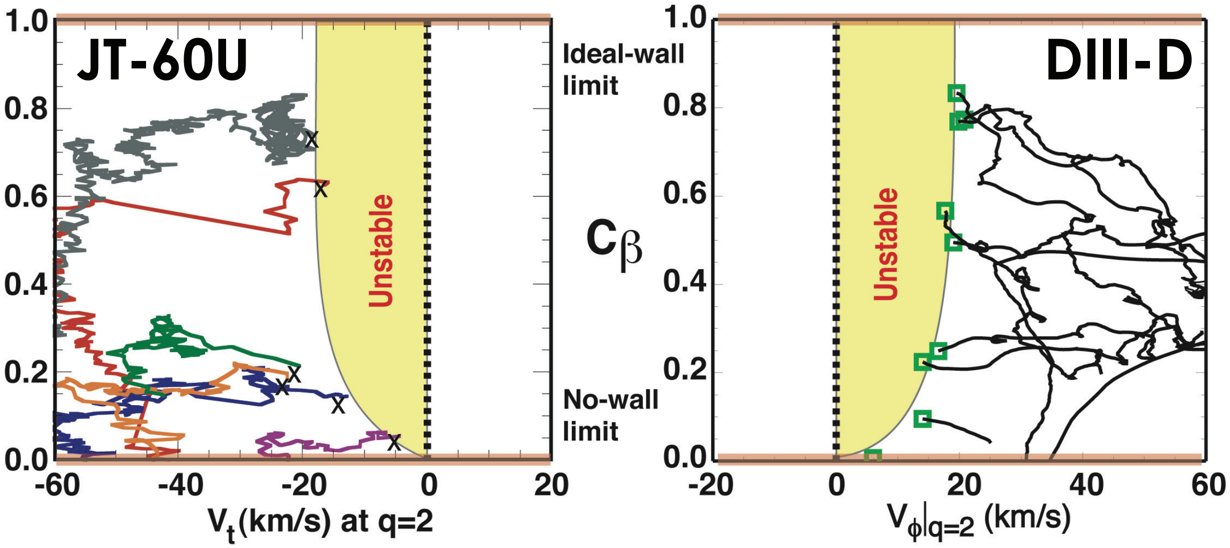
- **Resistive wall allows magnetic field to "leak" through**
  ➔ the instability is slowed, but not stabilized
  - Resistive wall can be made "perfect" by moving the plasma relative to the wall
  - Slower growth due to resistive wall opens possibility of feedback control





Operation with Pressure Above the No-Wall Limit Has Been Confirmed in Several Tokamaks

- **Wall stabilization works in rotating plasmas**



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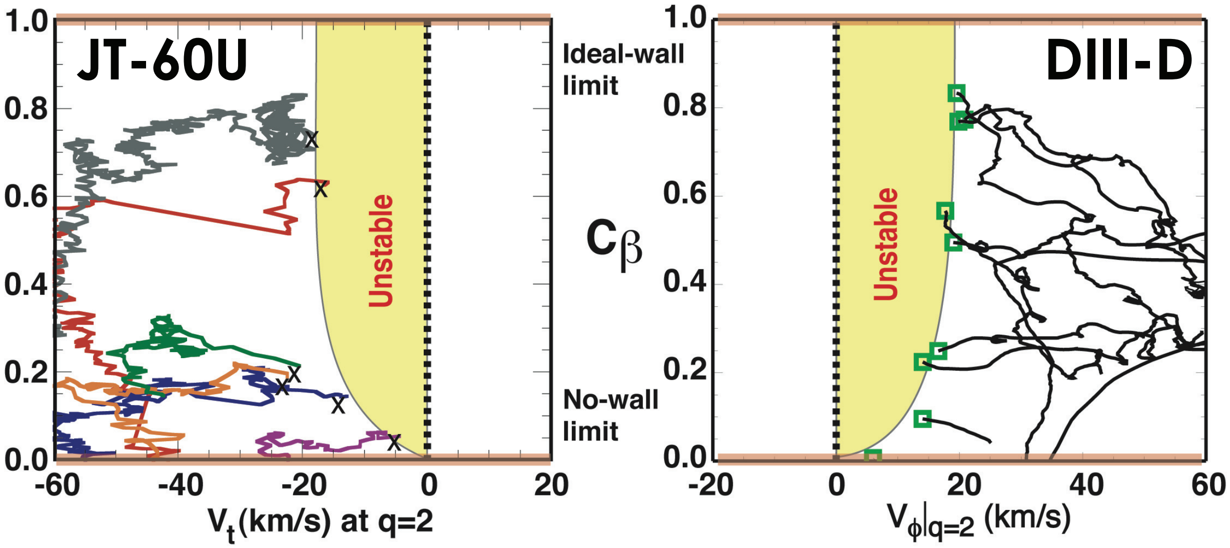

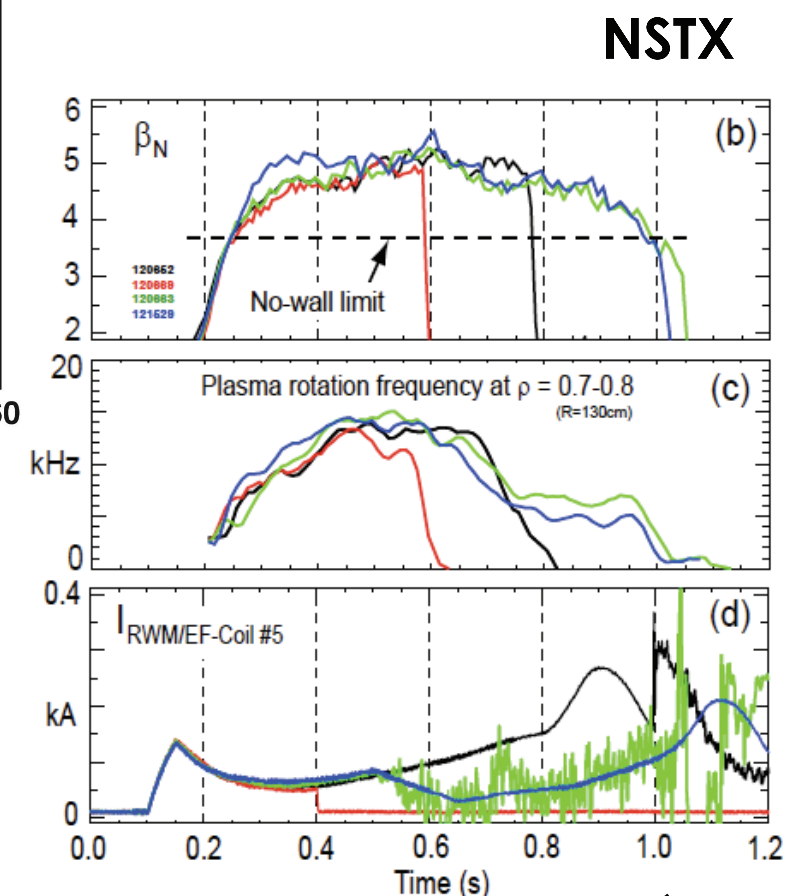

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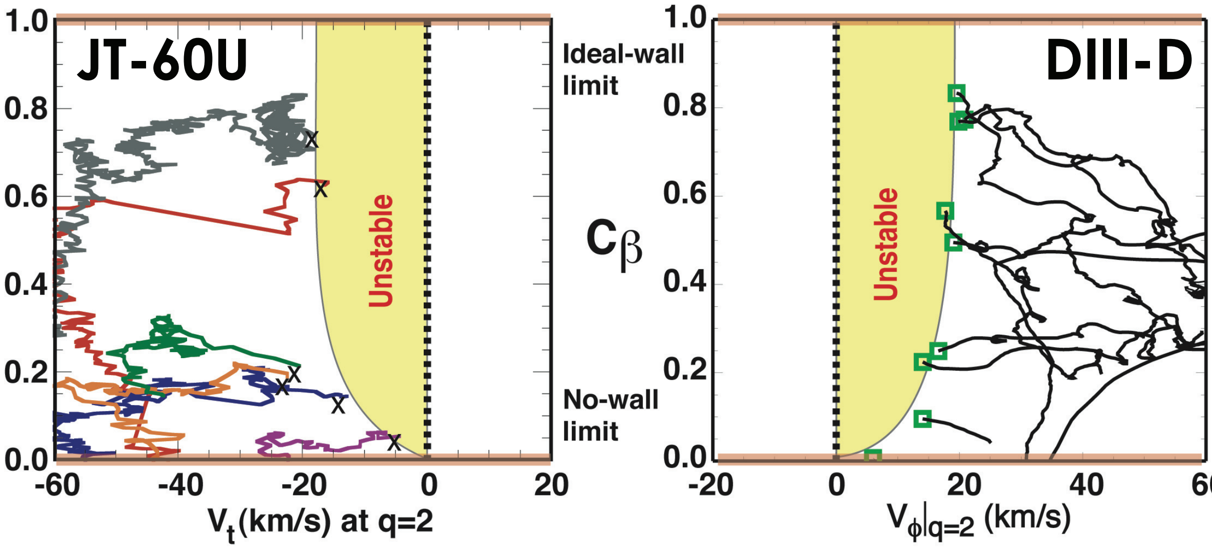

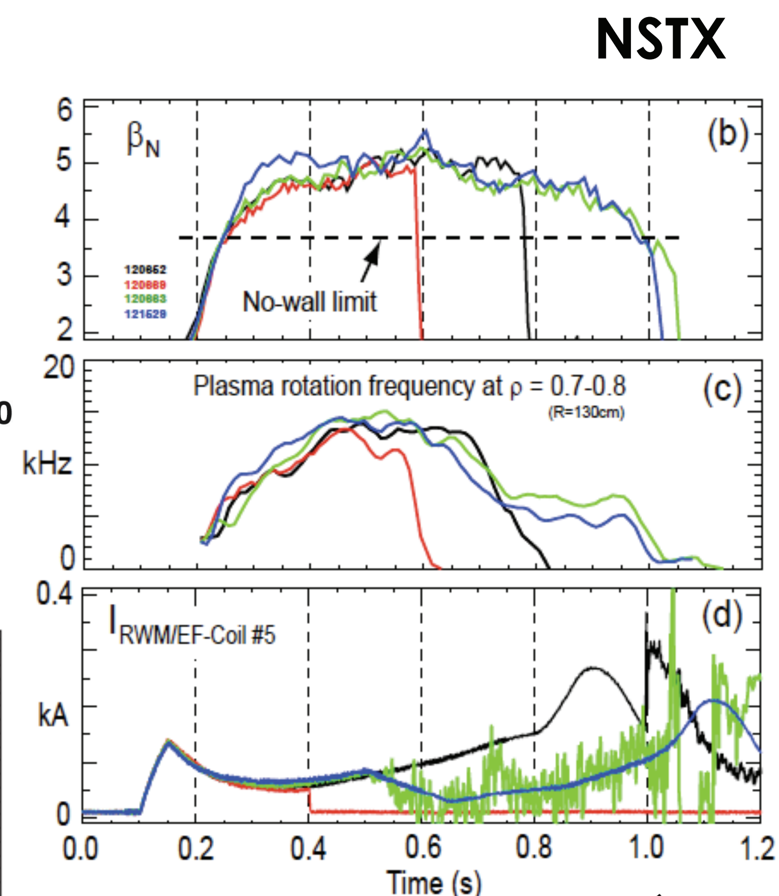

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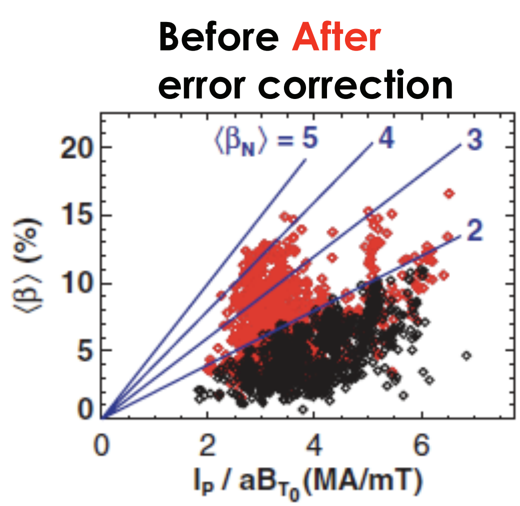

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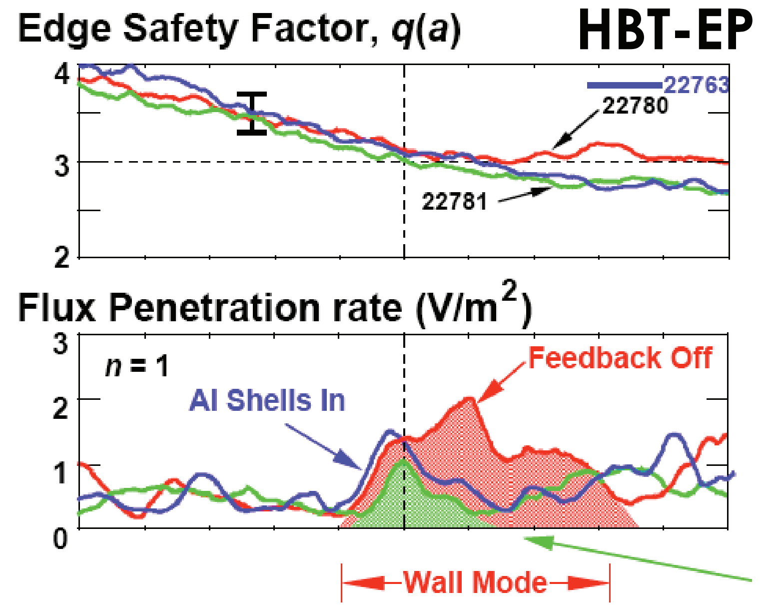
Stability

Active Feedback Control May Be Required To Operate Above The No-Wall Limit Without Rotation

- **Active feedback control allows operation at the ideal-wall limit without rotation**
  - Application of active feedback has the same stabilizing effect as a highly conducting wall
  - Note that active control of the n=0 mode is routine



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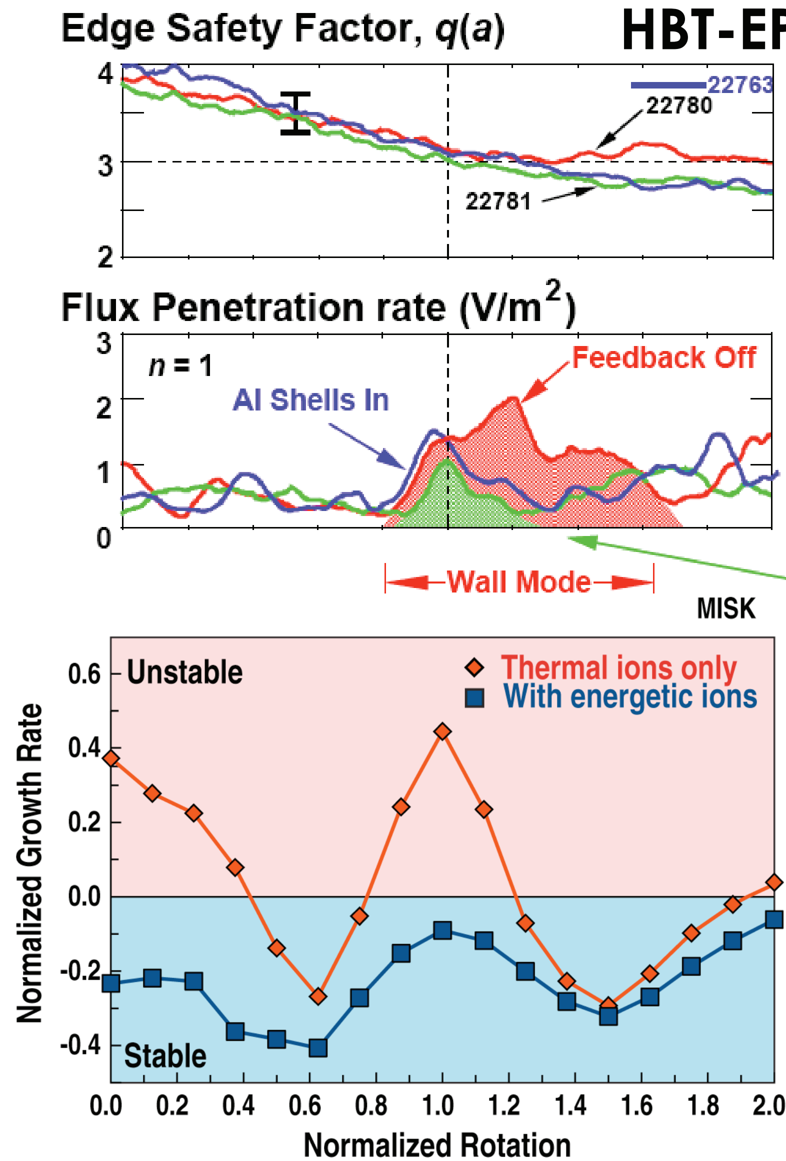
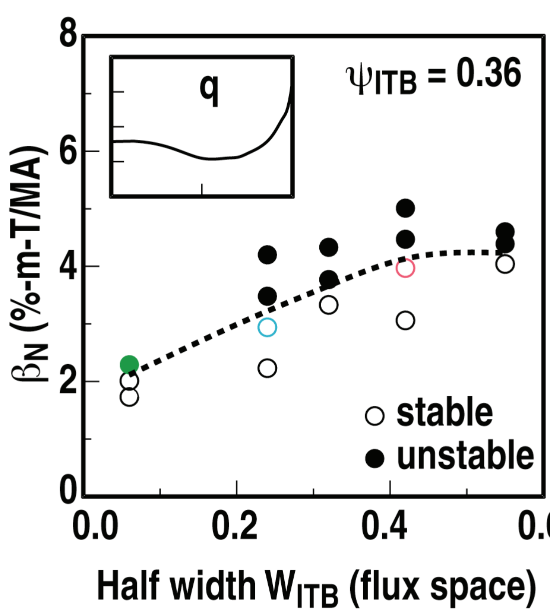
- **Emerging theory indicates that kinetic stabilization may allow kink stability without rotation**
  - Critical to validate this theory, including scaling to smaller normalized gyroradius

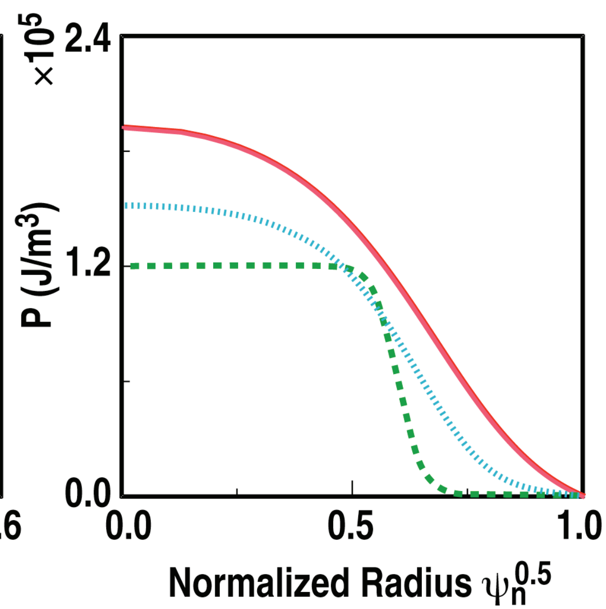


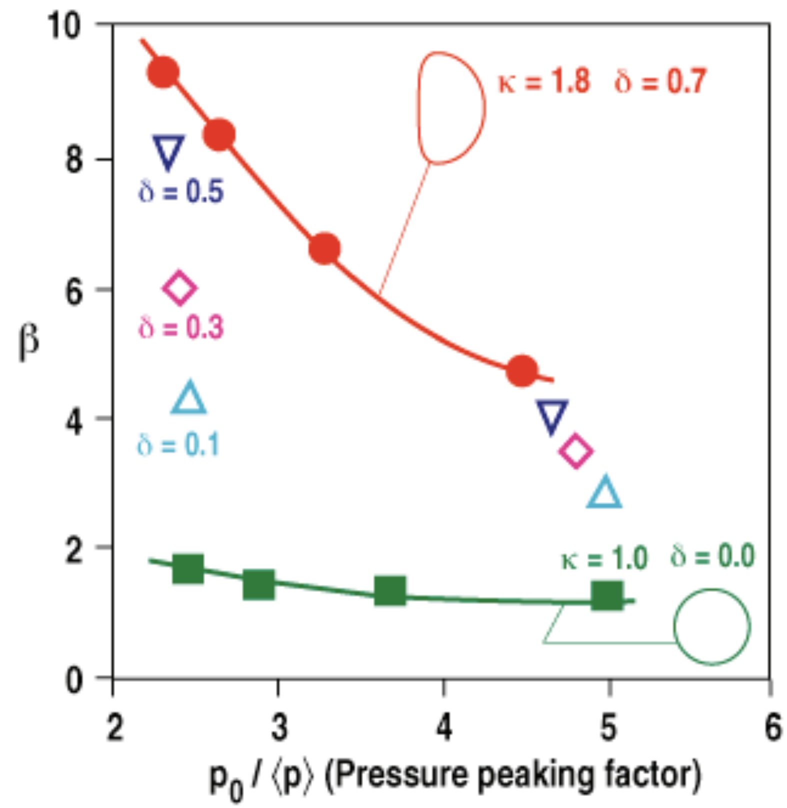
Stability

Theoretical Pressure Limits Are Sensitive to Cross-Section and Pressure Profile Shapes

- **Optimized cross-section shape allows much higher β with similar pressure profiles**
  - Due to higher I/aB and higher β_N
- **Broader pressure profiles allow higher pressure at all radii**







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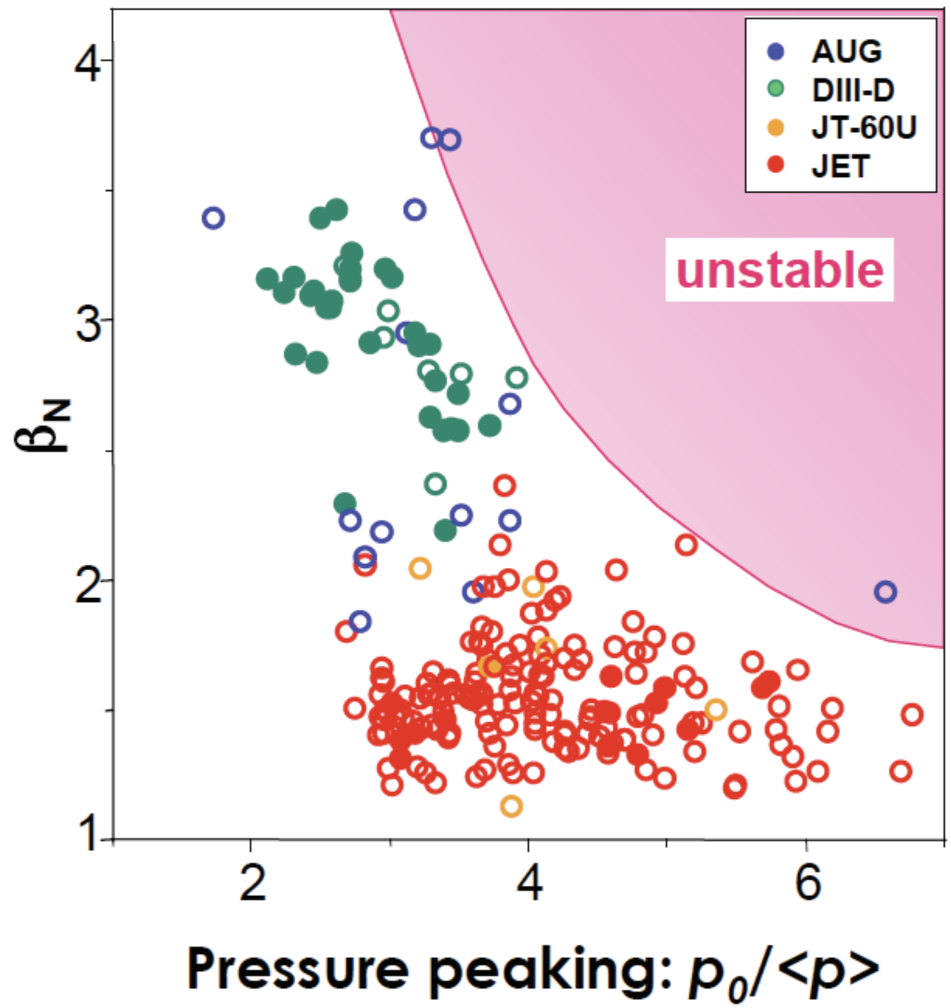
Stability

Experiments Validate Ideal MHD Prediction of Variation of the Pressure Limit with Pressure Peaking

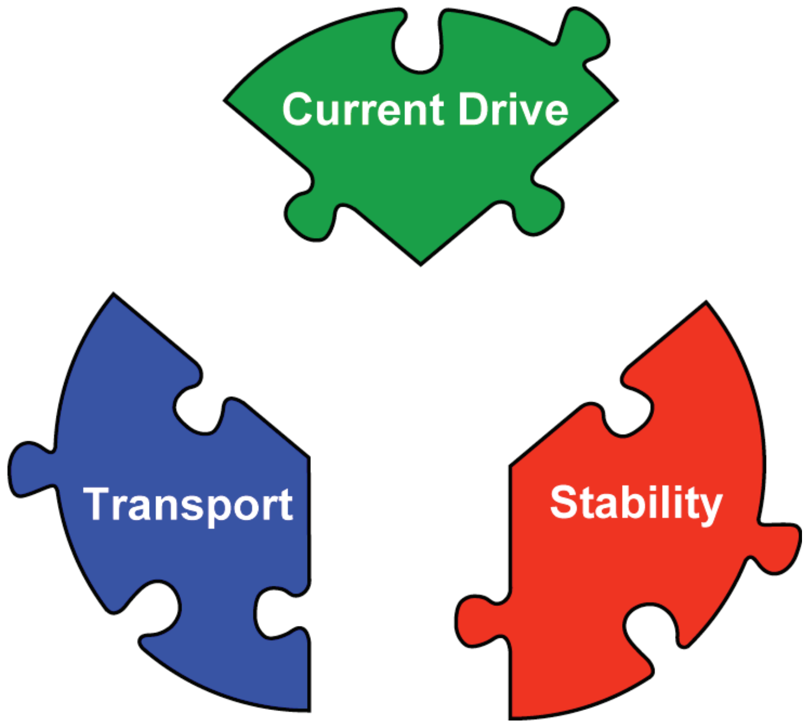
- **Lower pressure limits with peaked pressure profiles is confirmed experimentally on several tokamaks**
  - Limit is lower because the mode couples weakly to the wall



Insights from Studying the Individual Elements Need To Be Integrated Into the Steady-state Scenario

Current sustained in steady-state with minimal external sources

- Key metric: $f_{BS} > 0.6$



Good alignment of bootstrap current profile with total current profile

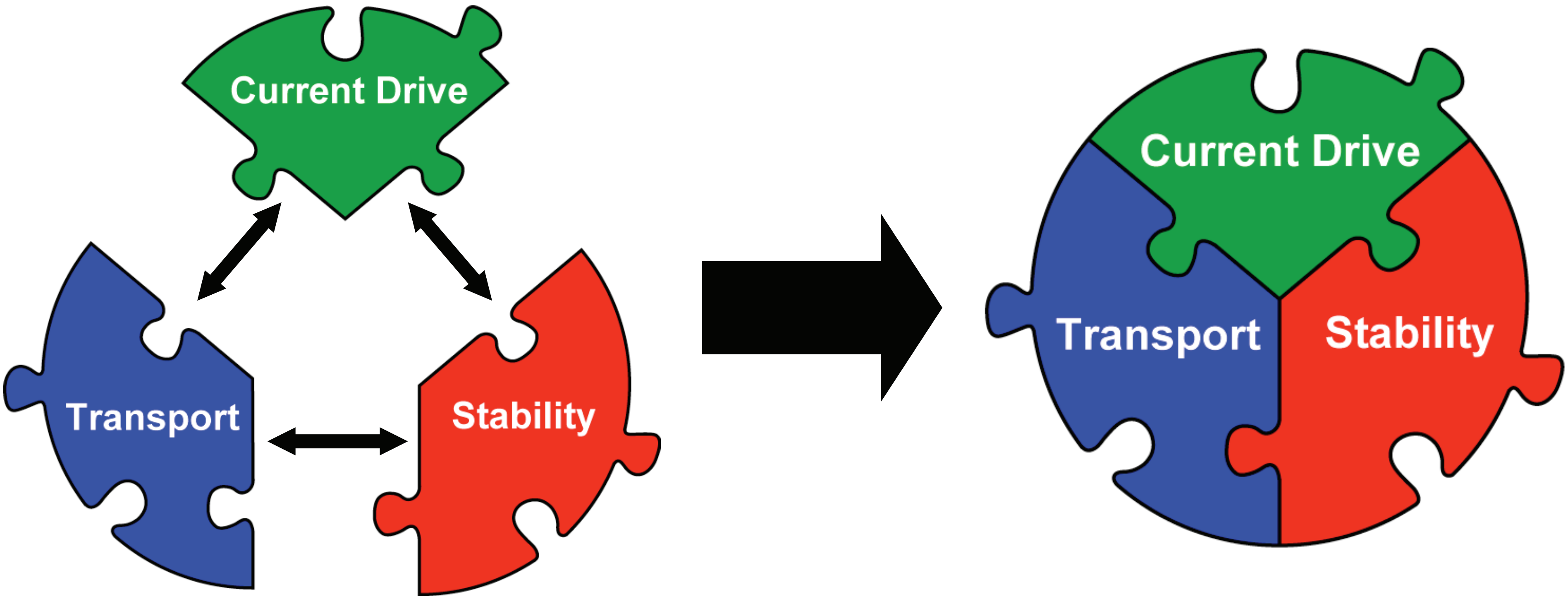
- Key metric: Stationary profiles on τ_E timescale

Broad pressure and current profiles for wall stabilization

- Key metric: $\beta_N > 3.5$

Integrating the Individual Elements Into a Solution for the Steady-state Scenario Is Essential

- **Each of the elements has a clear connection with the other two elements ➔ strong interaction implies integrated empirical optimization is needed**



- **Demonstration of integrated solutions stationary on times approaching the resistive timescale is the area of greatest progress in the last decade**

Simple Power Balance Provides Insight Into Optimization Priorities

- **0-D formula for fusion gain Q based on transport losses:**

$Q \propto \beta_N H / q_{95}^2$

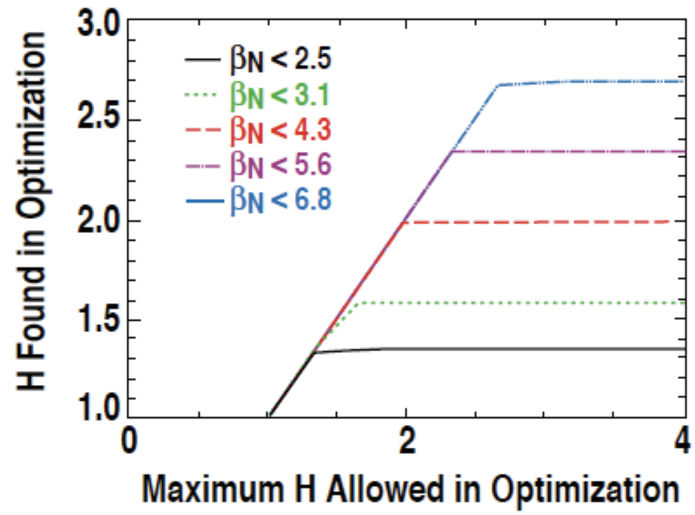
  - H is the ratio of the confinement time to a multi-tokamak database projection (IPB98y,2 scaling used in this talk)
  - Equal weight to improving pressure limits and confinement
  - Emphasis on high current

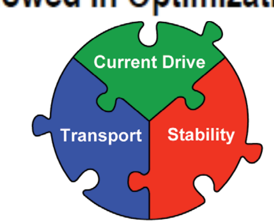
- **0-D formula for fusion gain based on self-heating and self-generated current:**

$Q \propto \beta_N^3 / [f_{DL}^2 (1 - f_{BS})]$ $\quad\quad f_{BS} \propto \beta_N q_{95}$

  - f_{DL} is the ratio of the density to the empirical density limit $I/\pi a^2$
  - Weights strongly improvement of the pressure limit
  - H absent because the formula assumes no power required beyond that for external current drive

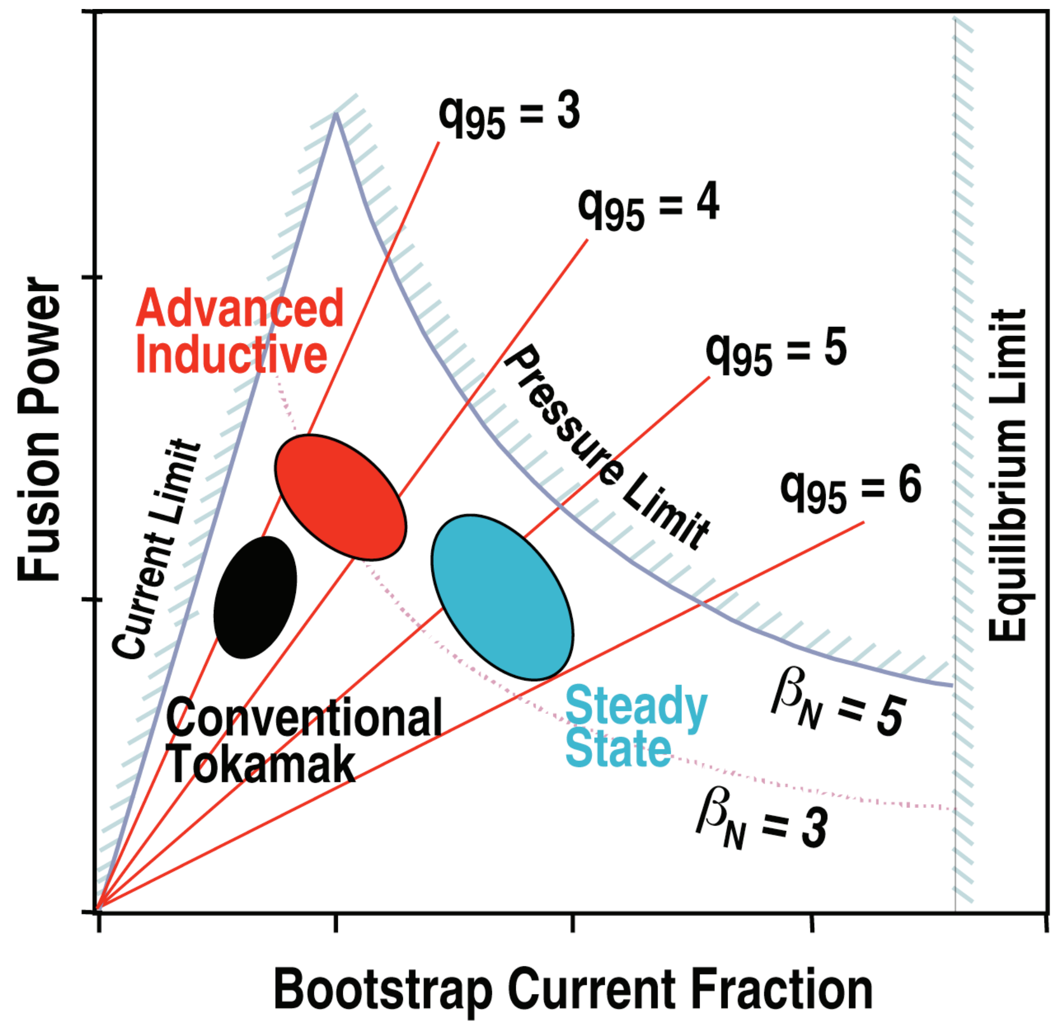
L. J. Perkins, "It's Beta Stupid!" Seminar, March, 1993





Tokamak Operational Landscape Shows Clearly the Different Optimization Strategies

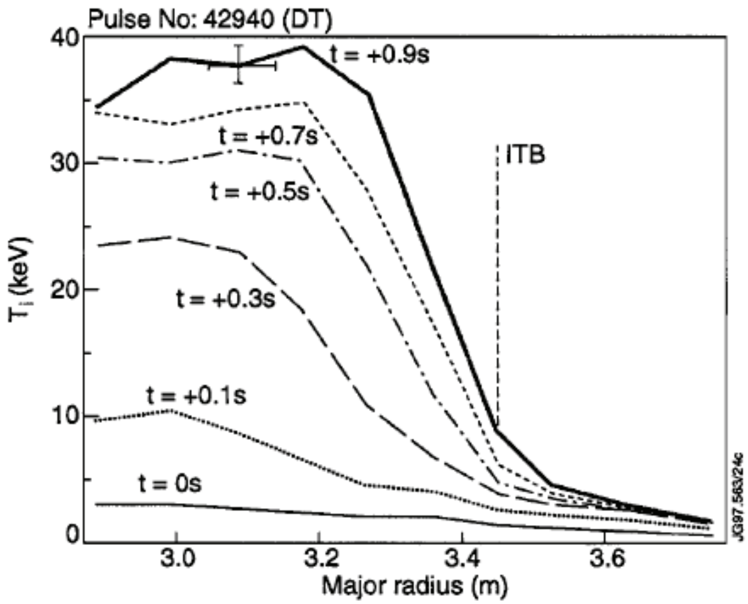
- **At low q_{95}, high fusion power is possible, but not in steady-state operation**
- **At high q_{95}, steady-state operation is easier, but at low fusion power**
- **Achieving steady-state while maintaining high fusion power implies operating near the pressure limit at modest q_{95}**
- **Figures of merit for steady-state operation:**



$\beta_N > 3.5$, $H > 1$, $f_{BS} > 0.6$, $q_{95} = 4 - 6$

Fusion Power > 8 MW Was Achieved Transiently Using An Internal Transport Barrier

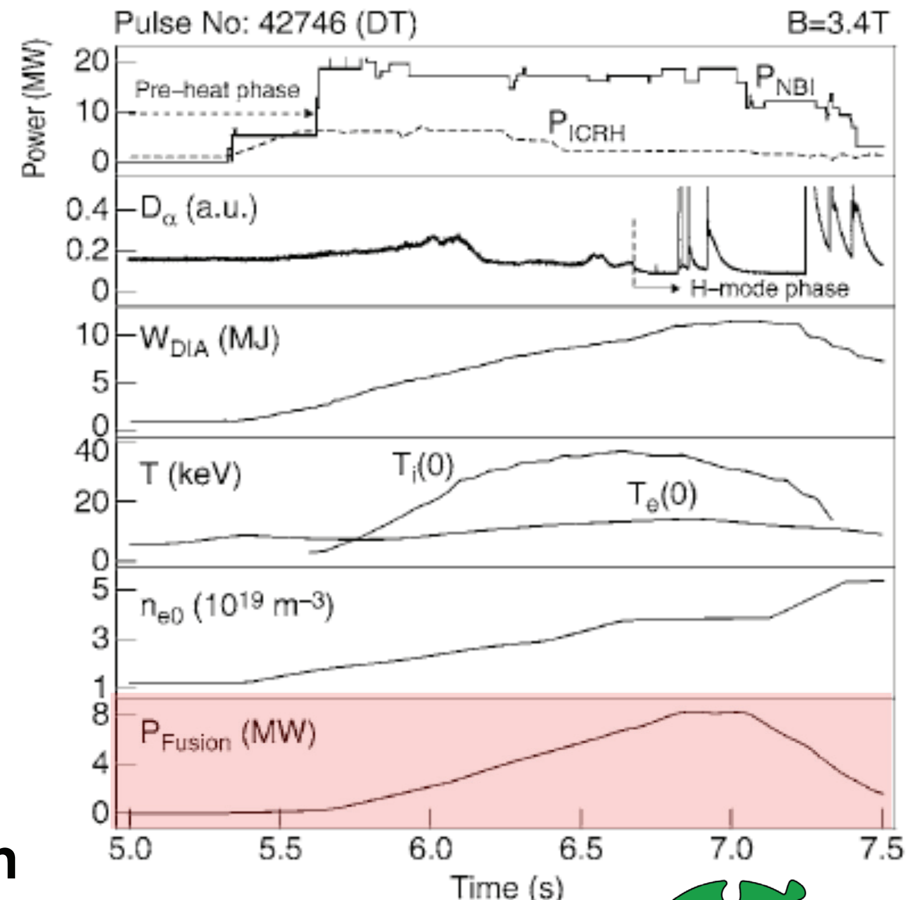
- **Strong ITB leads to high ion temperature and significant fusion power**



- **Active control of the pressure peaking on the transport timescale would be very difficult at high fusion power (∝ p^2) and gain**

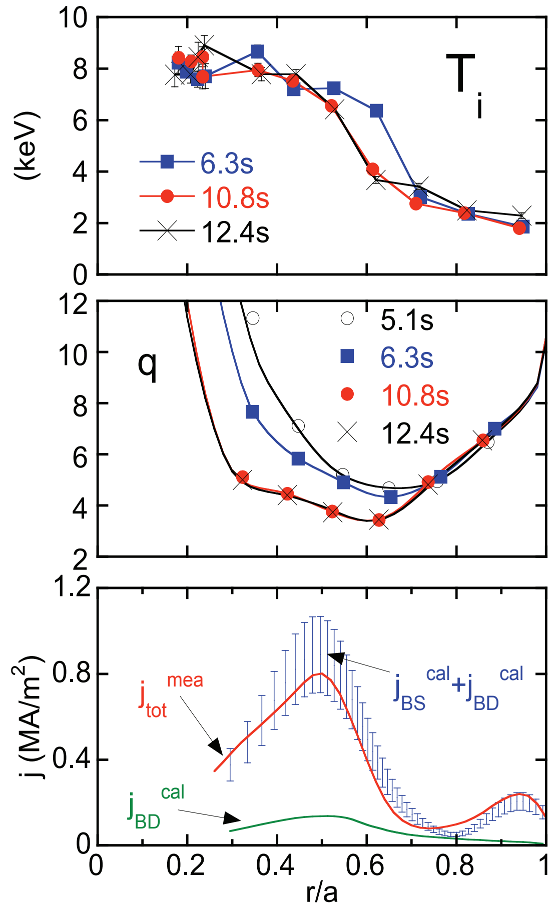
JET

β_N=1.9, H=1.5, f_{BS}=0.15, q_{95}=3.9



Gormezano, PRL 1998

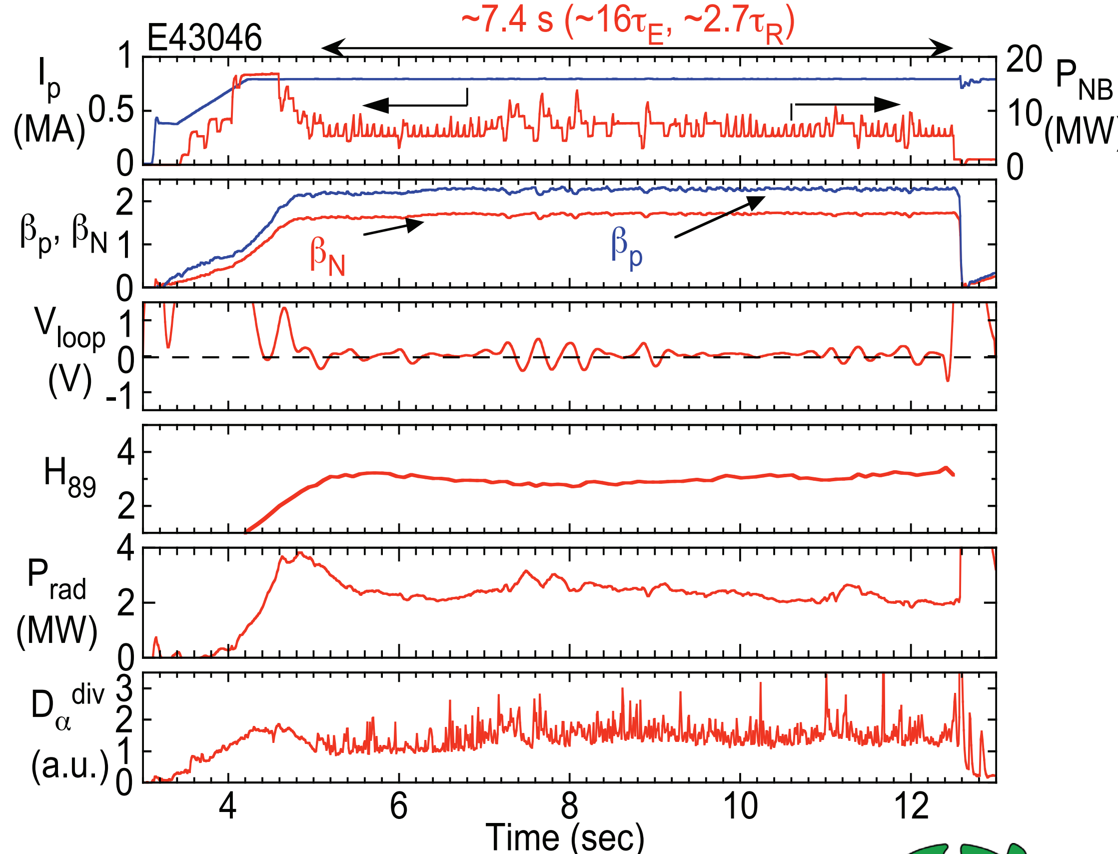
Stationary ITB Regime Has Been Achieved With Non-Inductive Current Sustainment



Deeply reversed q profile may be a problem for energetic particles

β_N=1.7, H=1.8, f_{BS}=0.75, q_{95}=8.3

JT-60U



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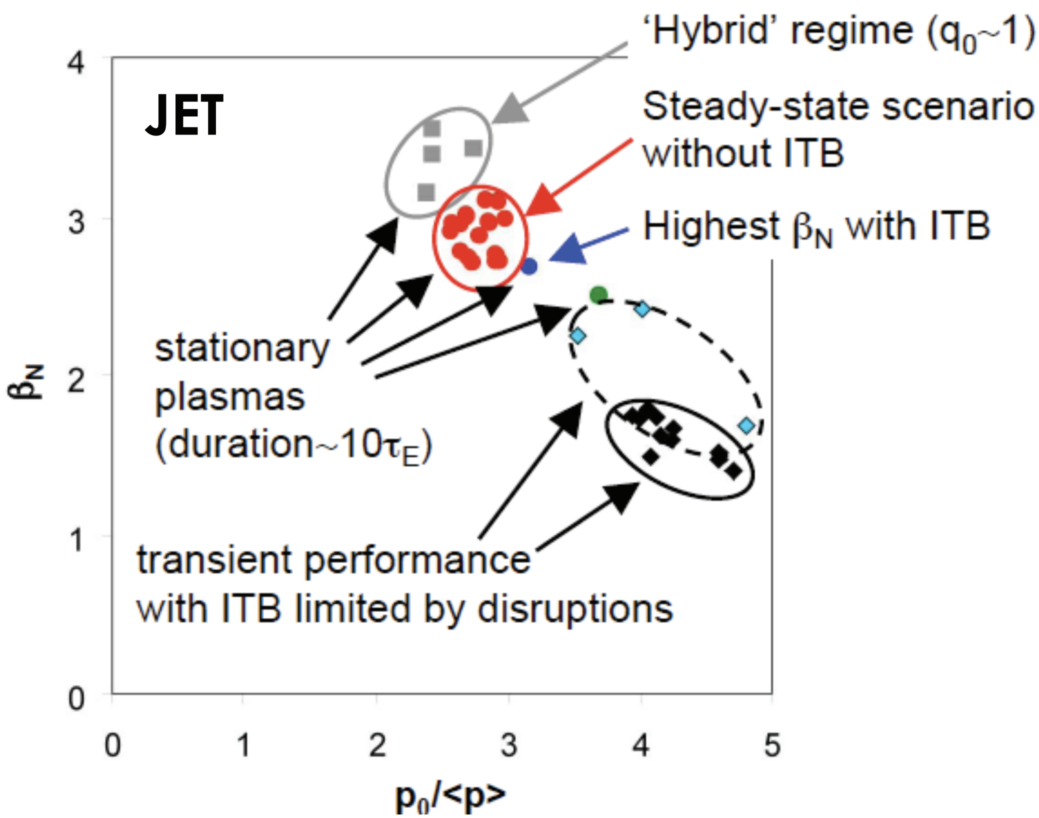
Broadening of the Pressure and Current Profile Led to Higher β_N in JET

- **Broader pressure profiles allow higher β_N in both the ITB and non-ITB regime**



Challis, APS 2007

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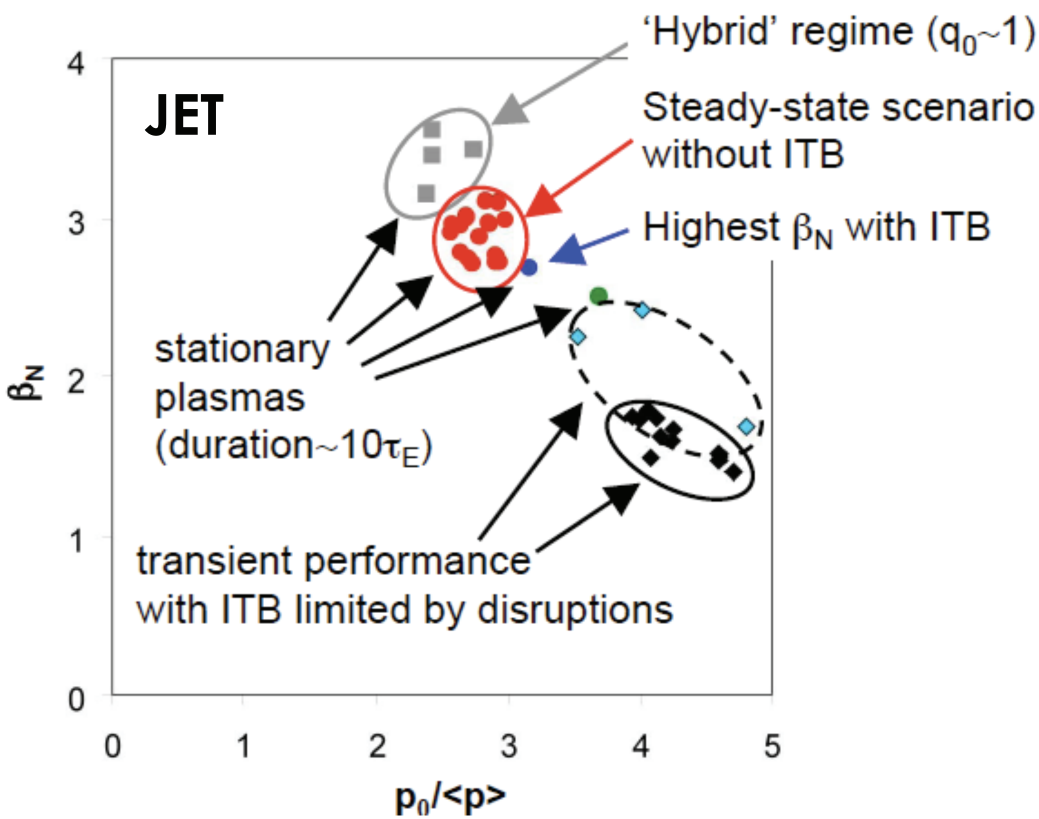
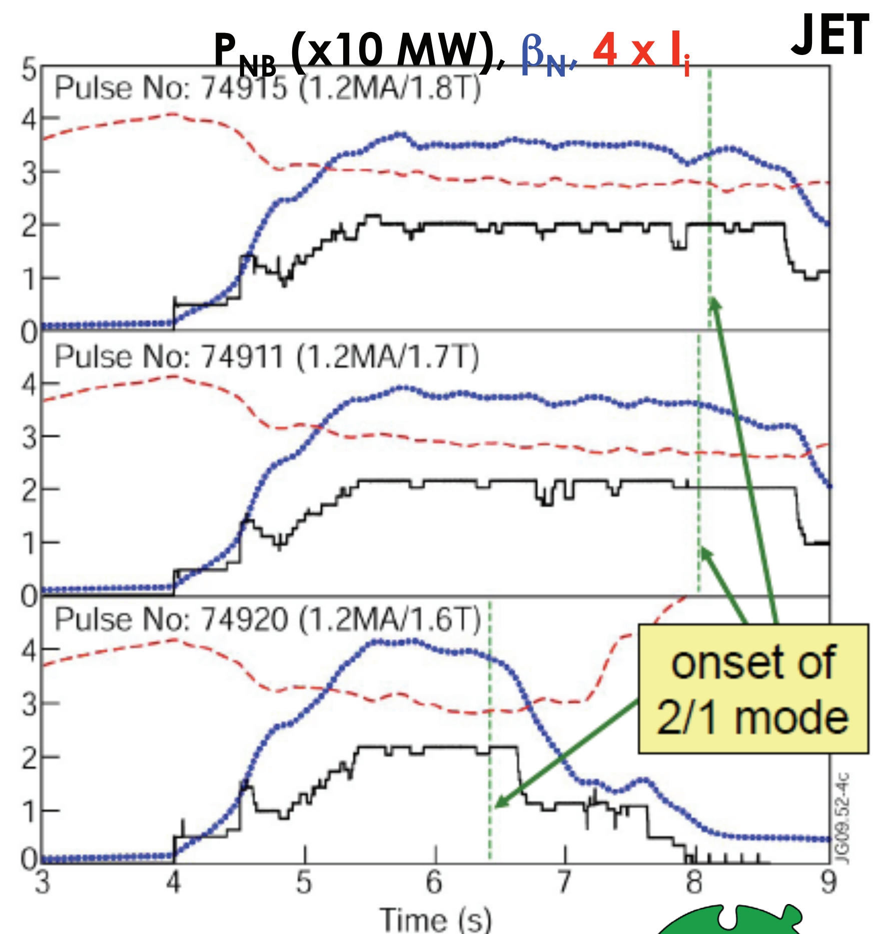


Challis, APS 2007

- **$\beta_N \geq 4$ achieved with broader pressure profiles in 2008**



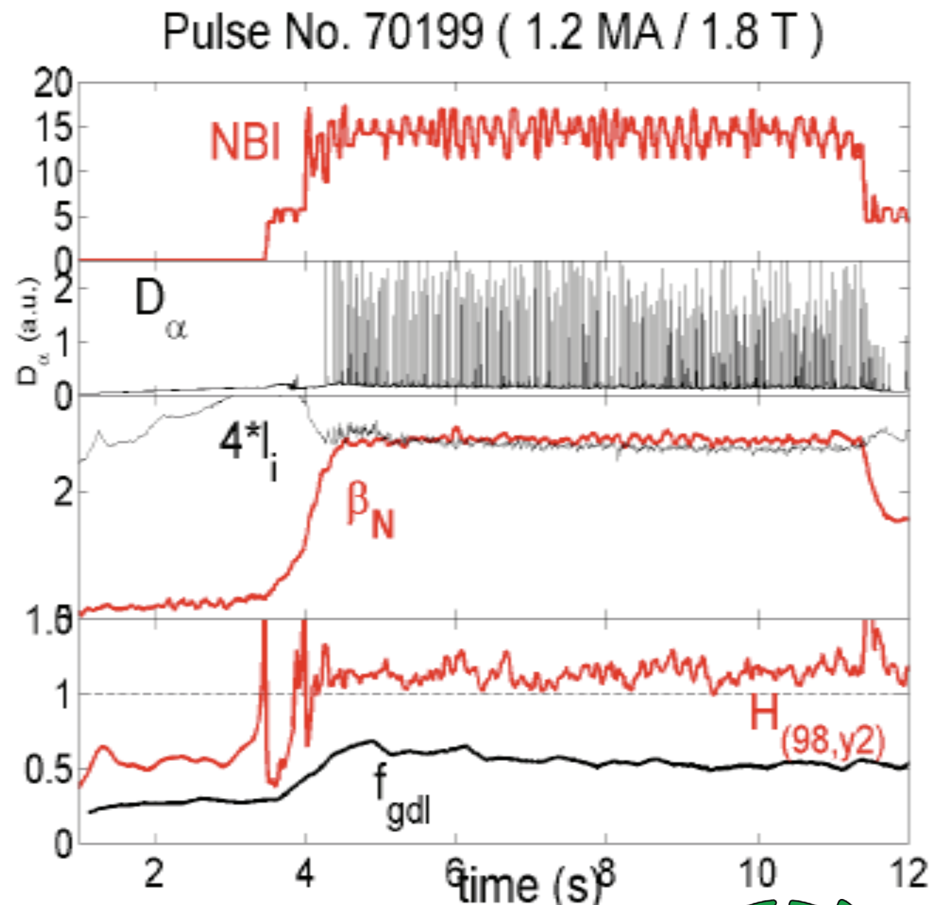
Challis, EPS 2009

Stationary Performance Above Maximum β With ITB Is Obtained With Broader Pressure and Current Profiles

- **Active control is applied only to the total stored energy, not the profile**
- **Optimization of the current profile was carried out by varying the current ramp rate and the heating power and timing**
- **Performance is stationary on the energy confinement timescale and duration approaches the resistive time scale**

JET

β_N=2.9, H=1.2, f_{BS}=0.3, q_{95}=4.8

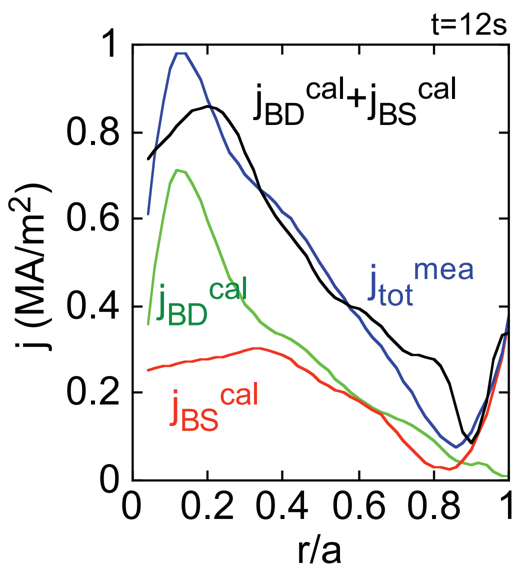


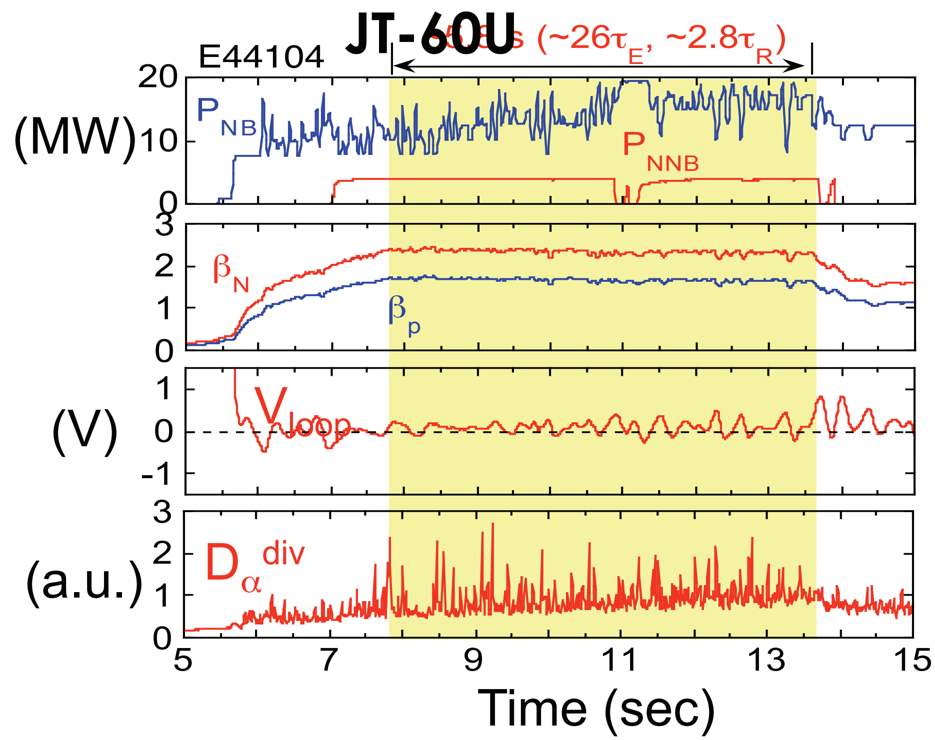
Rimini, IAEA 2008

Non-Inductive Sustainment with Good Plasma Performance Has Been Achieved in JT-60U

β_N=2.4, H=1.0, f_{BS}=0.45, q_{95}=4.5

- **Performance achieved with weak shear and no ITB**

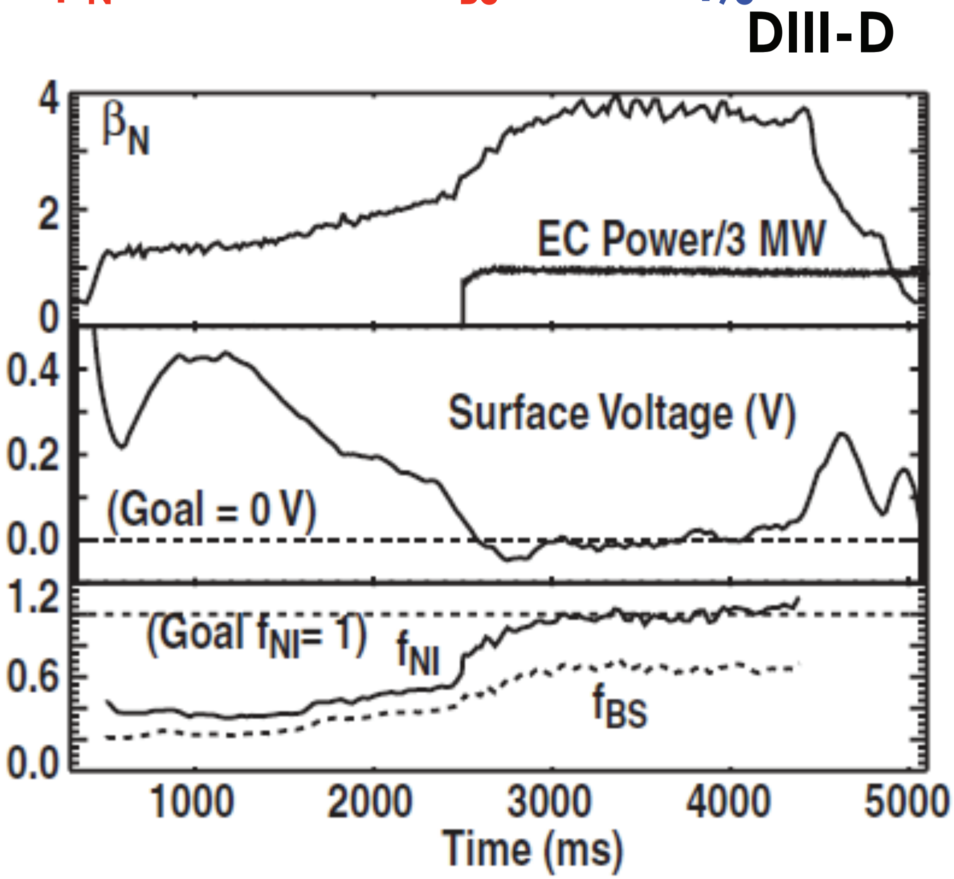


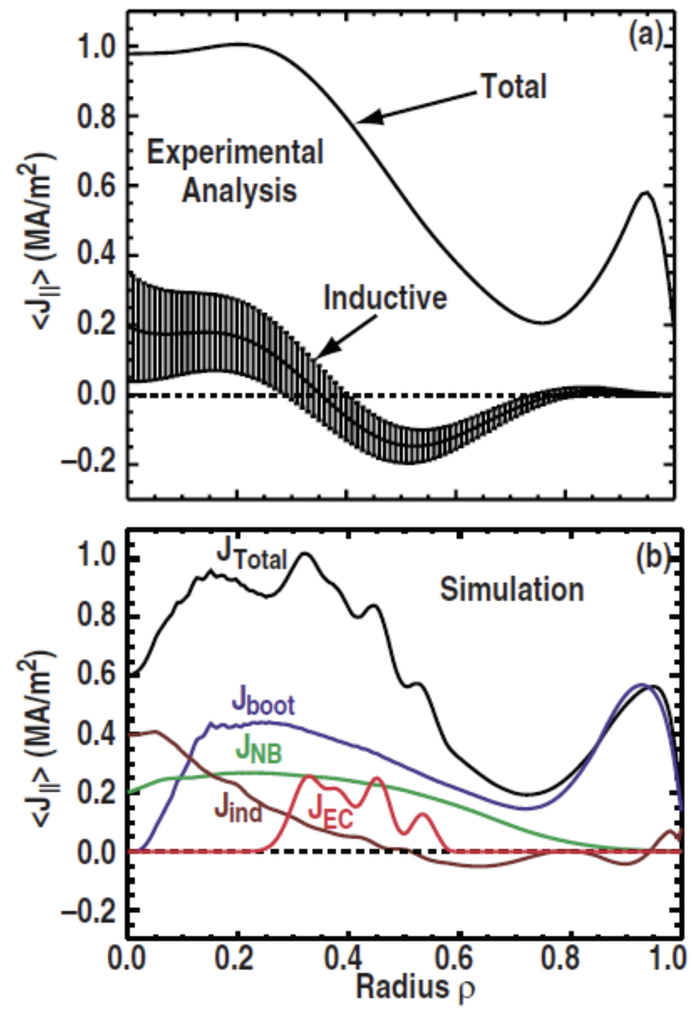


- **Note that JT-60U could not take advantage of wall stabilization for these experiments ➔ higher β_N is possible**

Wall Stabilization Allows Higher Performance Non-Inductive Discharges in DIII-D

β_N=3.7, H=1.5, f_{BS}=0.65, q_{95}=6.3





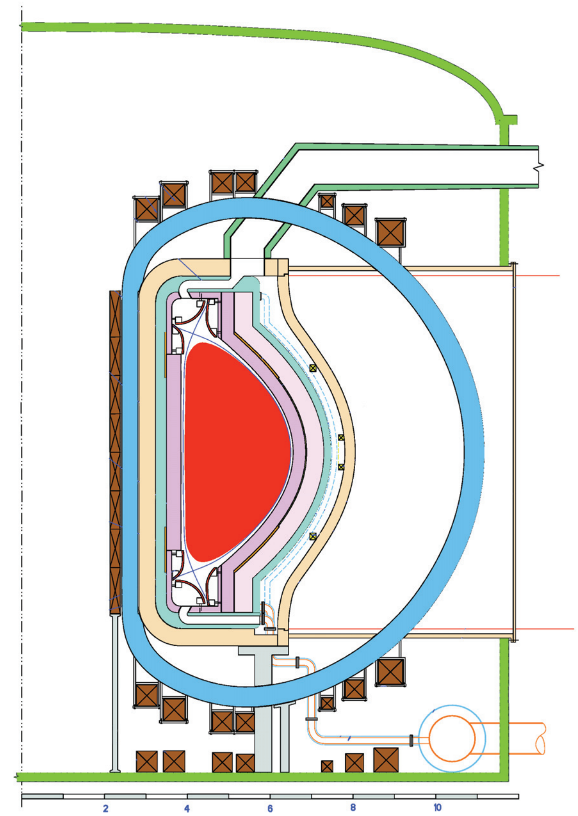
- **Nearly non-inductive operation sustained approaching the resistive time scale**

Current Drive
Transport
Stability

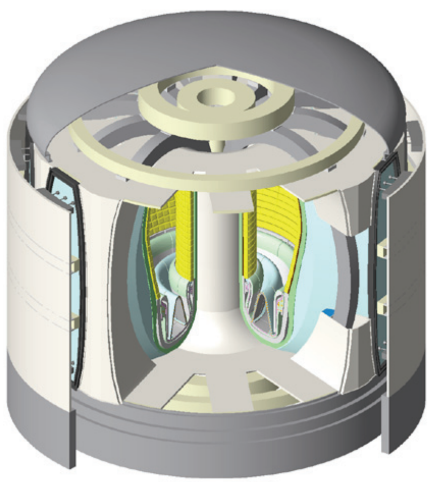
System Studies Project Present Understanding To Future Applications

- **Systems studies integrate "real-world" considerations into the optimization:**
  - Cost implications of design choices
  - Closed fuel cycle
  - Realistic wall loading and lifetime estimates

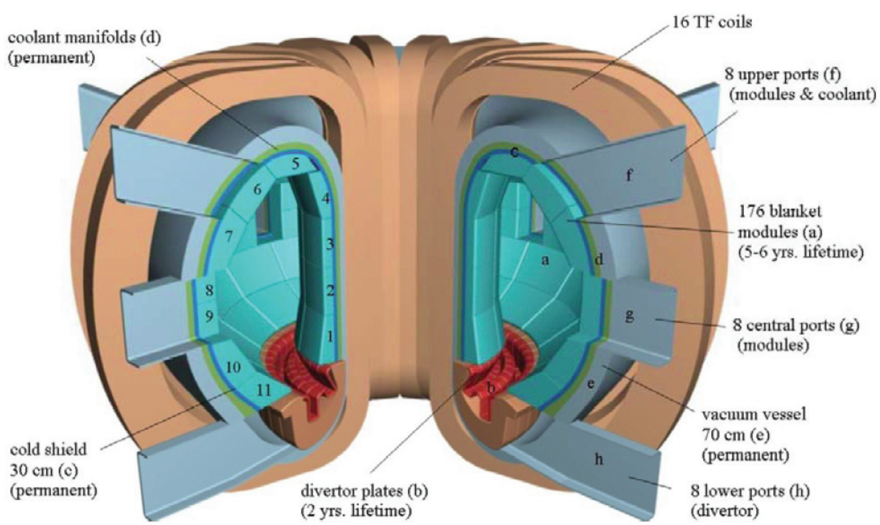
ARIES-AT (USA)



Slim CS (Japan)



PPCS Model C (EU)



Systems Studies Indicate the Performance Gaps from Present-Day Experiments

Present-day experiments need:

- Operation at higher β_N (will close gap in f_{BS} also)
- Push to higher density

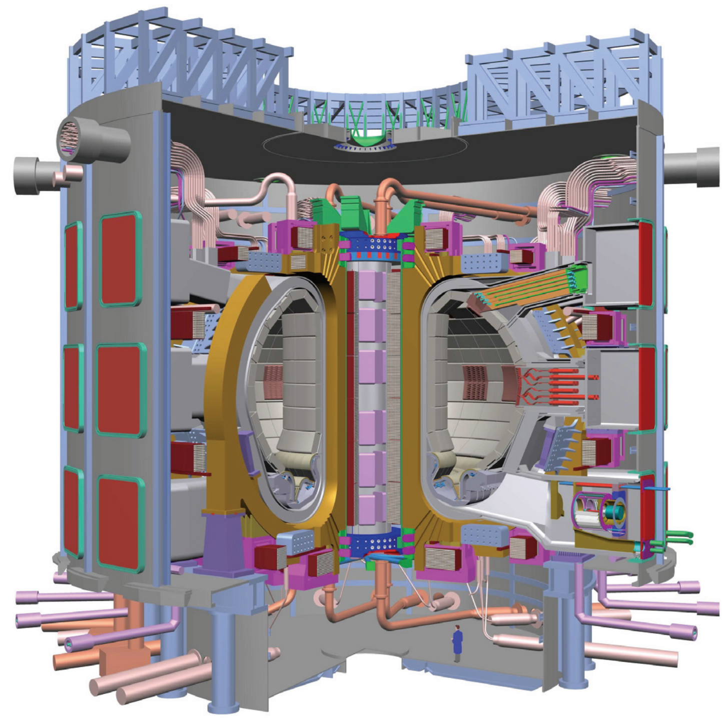
- Confinement improvement does not seem to be a significant need

	JT-60U $f_{NI} \approx 1$	DIII-D $f_{NI} \approx 1$	Slim-CS	EU PPCS	ARIES -AT
β_N	2.6	3.7	4.3	4.0-4.5	5.4
H (98y2)	1.0	1.5	1.3	1.2-1.3	1.7
f_{BS}	0.5	0.65	0.77	0.63-0.76	0.91
q_{95}	4.5	6.3	5.5	4.5	3.2
f_{DL}	0.5	0.5-0.6	0.98	1.5	0.9

Two Machines Have Been Commissioned To Fill The Gap To a Steady-State DEMO

- **ITER is a joint project of 7 global partners**

- **A primary physics objective is to achieve Q=5 in a steady-state scenario**



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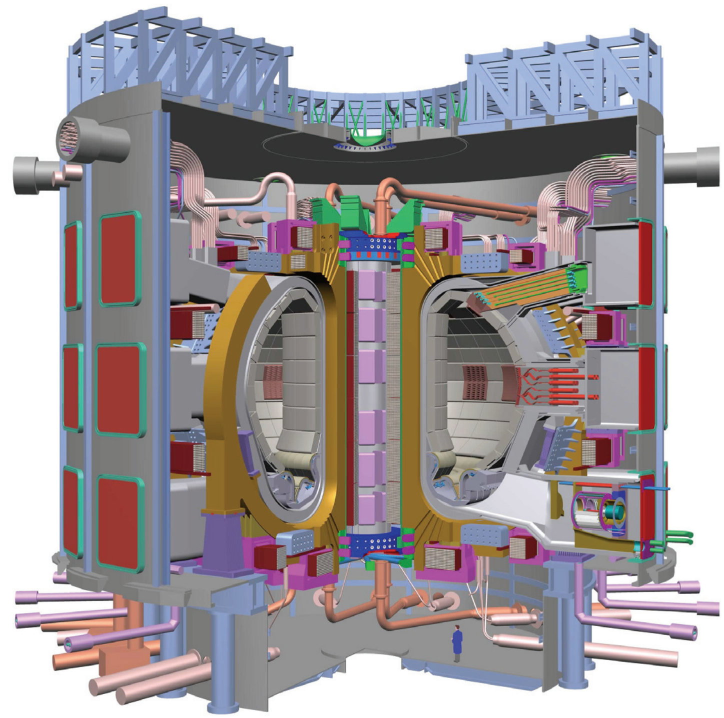
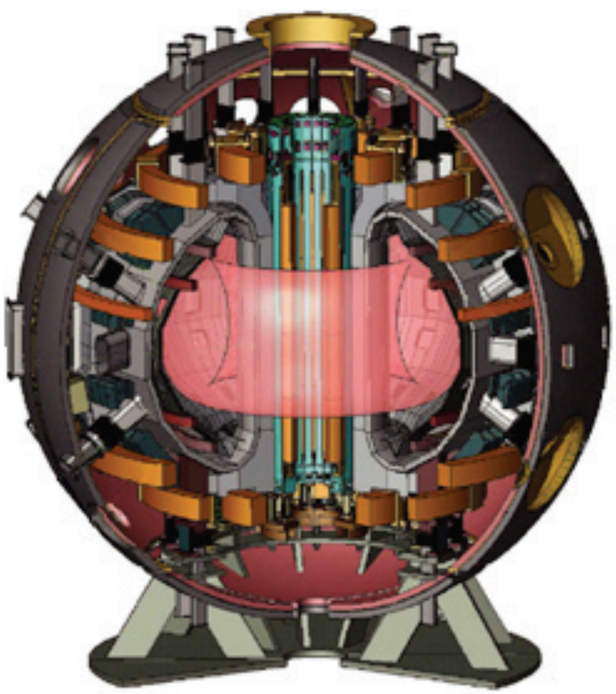




- **JT-60SA is a joint project of Japan and the EU**
- **The goal is to extend the physics basis of single-null and double-null approaches to steady-state scenarios**

Developing Accurate Methods for Projecting Existing Scenarios to Future Tokamaks Is Essential

- **Prediction of the pressure profile is the key uncertainty in the projections**

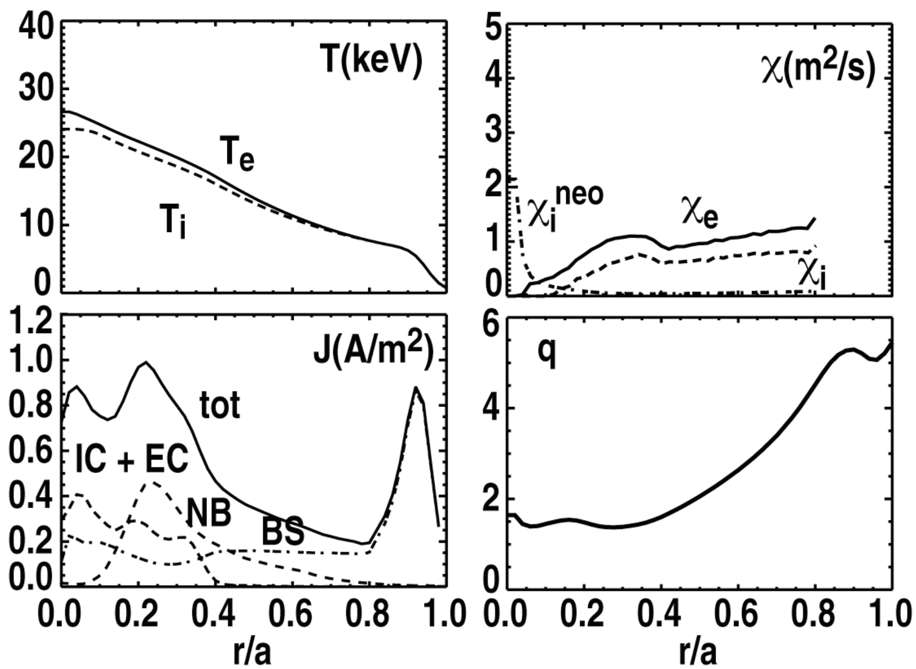
- **0-D projections**
  - Unlikely that conventional database analysis can be done (small number of tokamaks, diverse scenarios)
  - Dimensionless parameter scaling accounts for profile effects, but assumes the sources are similar
  - Essential to imitate the characteristics of self-heating (dominant electron heating without fueling or torque input)

- **1-D projections**
  - Combine theory-based transport models and empirical information
  - Essential that transport models have precise accounting for dependence on current profile

Theory-Based Models Can Be Applied to Find Self-Consistent Solutions for Future Tokamaks

- **Example is an ITER steady-state scenario based on DIII-D data**

- **Simulations solve for fluid variables self-consistently in time**
  - Temperature and current evolution can be predicted, but electron density and toroidal rotation generally are not predicted
  - Helium ash from fusion is treated self-consistently
  - Boundary conditions taken from empirical scaling

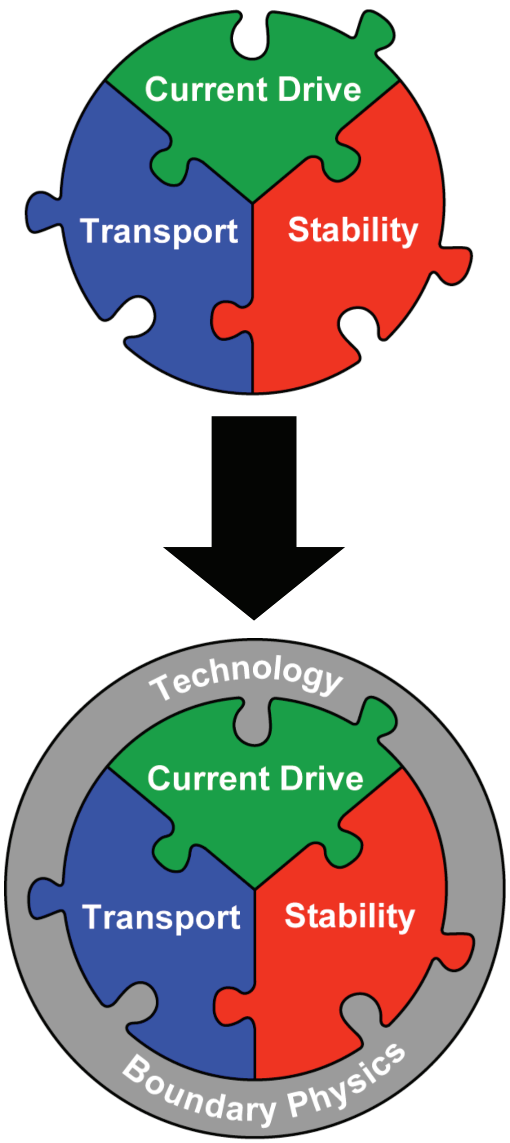


ITER Day-1 heating mix:
P_{NBI} = 33 MW, P_{EC} = 20 MW, P_{IC} = 20 MW

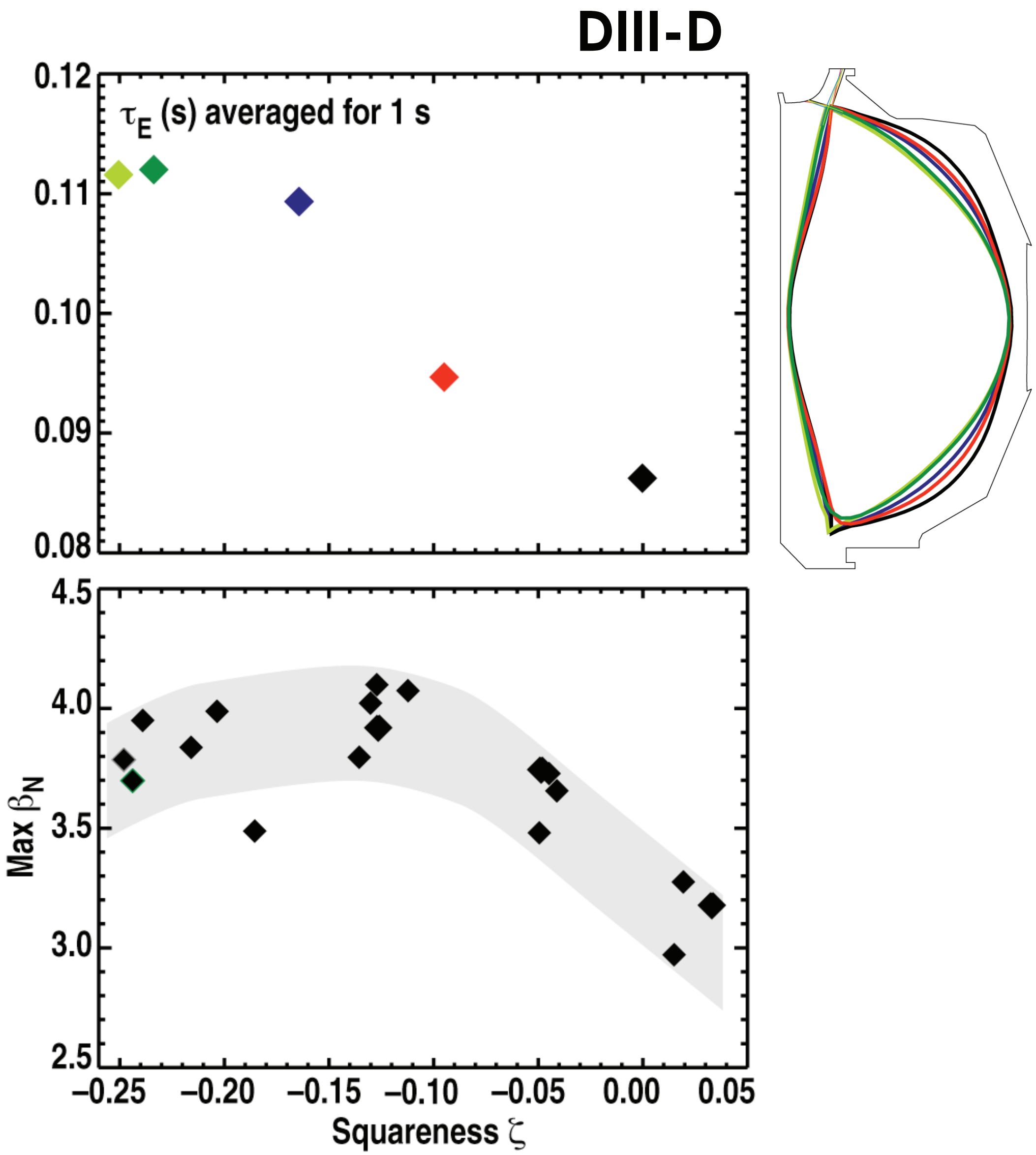
β_N = 2.9
H = 1.4
f_{BS} = 0.57
q_{95} = 5.0
f_G = 0.91
Q = 4.6

System Studies Emphasize Integration of the Core Scenario Into the Power Plant Environment

- **A very significant conclusion from both experiments and modeling is that scenarios exist that are stationary**
  - No active profile control required on the pressure evolution time scale
- **Physics issues that arise from integration with the power plant environment**
  - Design of appropriate coil and divertor locations
  - Compatibility of divertor solutions with core scenario
  - Availability limitations because of disruptions



Optimizing the Boundary Shape May Be Critical for Realizing Steady-State Scenarios



- **Plasma shape strongly affects pressure limits and confinement**
  - Variation in "squareness" not included in standard scalings

- **Important to consider for design of divertor and poloidal field coil locations**

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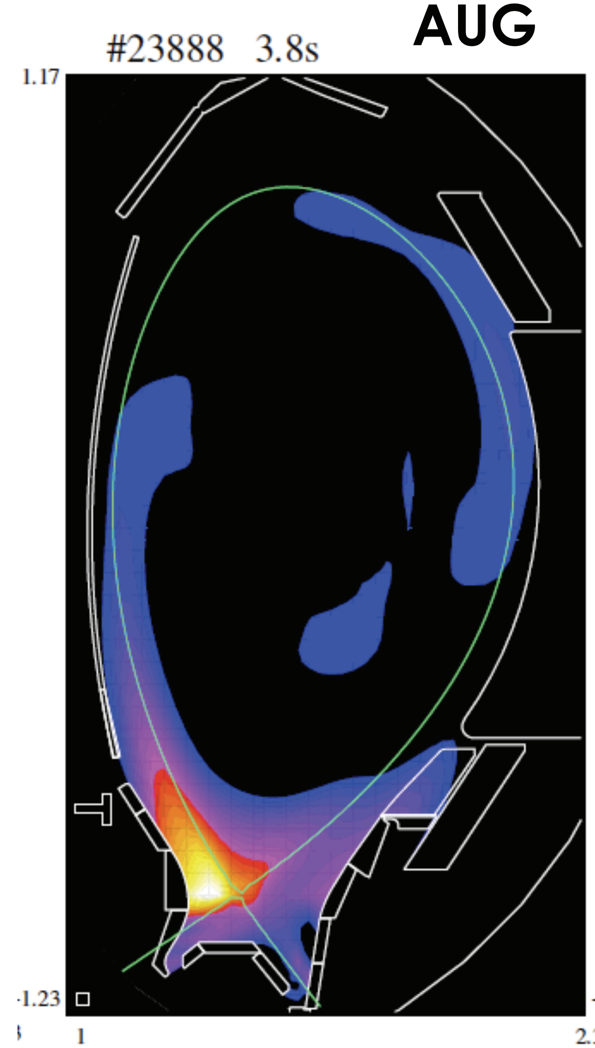
Compatibility of Methods To Handle Steady-State Heat Flux With Advanced Scenarios Is Being Tested

- **Steady-state scenarios optimize at higher q_{95} and lower density ➔ higher stress on the divertor**
- **Impurity seeding in the divertor is proposed to distribute the heat flux by radiation**
  - Initial experiments are mostly in conventional tokamak scenarios
- **Note that the radiation is occurring at the inner divertor leg, while the dominant heat flux is at the outer leg**

AUG

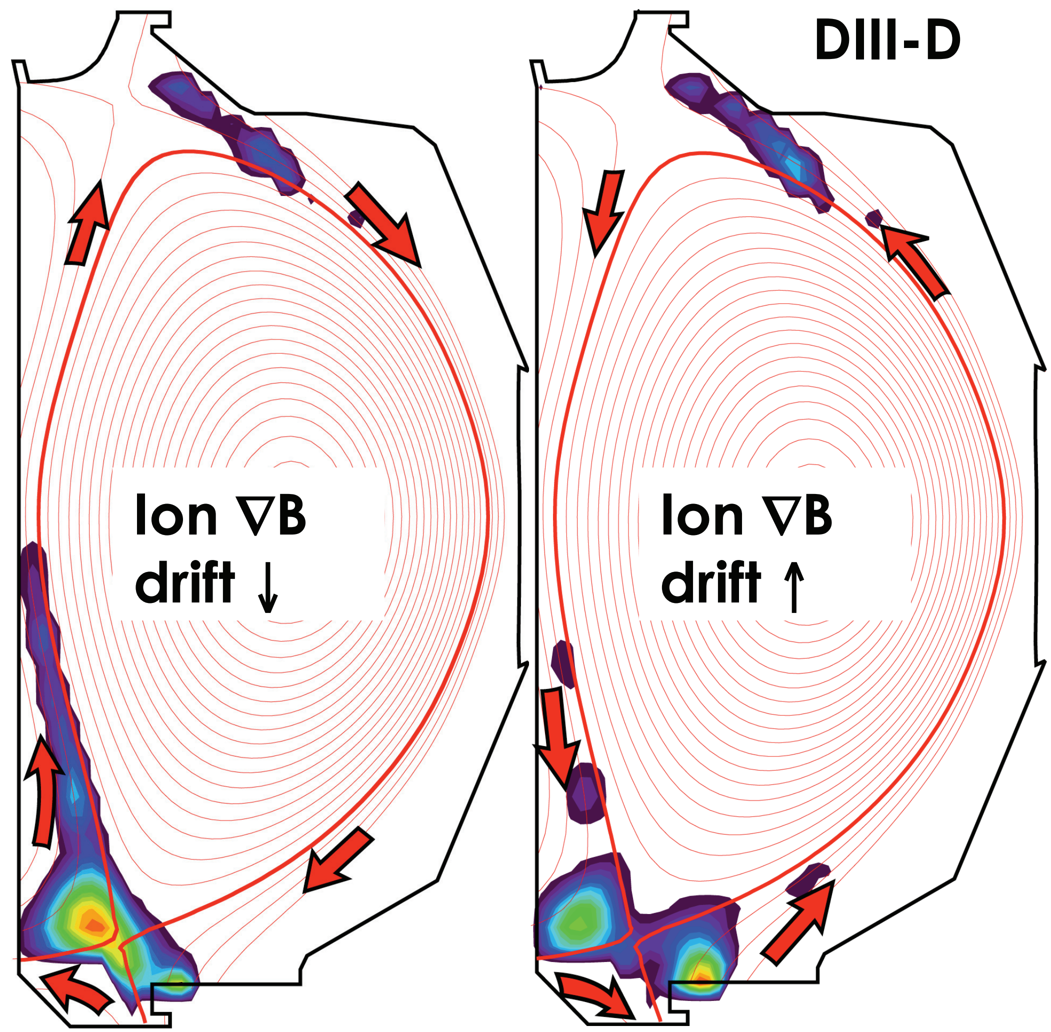


Direction of the ExB Drifts in the Divertor and SOL Strongly Affects the Radiation Pattern

- **Drifts toward the outer strike point give better heat flux reduction and help avoid detachment of the inner strike point**

- **Not clear that ExB drifts will still dominate in power plant divertor conditions**

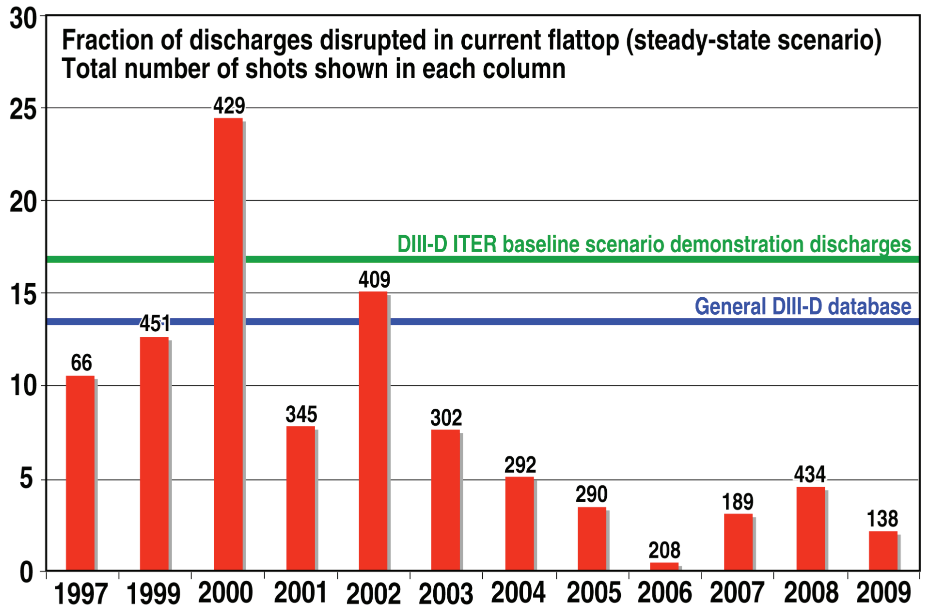
- **There are many potential consequences**
  - Not possible to have the drift direction the same in double-null
  - Change in shape and field direction will affect core confinement and stability



Red arrows indicate direction of E x B flow

Disruption Occurrence Data From DIII-D Indicate Steady-State Scenarios Are More Robust

- **ARIES-AT design goal: 1 unplanned outage per year**
  - This is a hardware and software reliability requirement--stationary plasmas are stable!
- **Occurrence of disruptions in DIII-D steady-state scenario is now much lower than the general database**
  - General database: 13.7% (>3500 discharges)
  - ITER baseline demonstration: 16.7% (186 discharges)
  - Steady-state scenario (1997-2002): 15.2% (1700 discharges)
  - Steady-state scenario (2003-2009): 4.2% (1853 discharges)



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- **Steady-state scenarios are significantly less likely to disrupt**
  - Data indicate the relative response of scenarios to faults and disturbances
  - Magnetic energy is also reduced by 2x

Physics Milestones for Near-Term Steady-State Tokamak Research

Three critical physics issues can be addressed in present-day tokamaks with modest upgrades:

- **Performance with dominant electron heating without core fueling or applied torque**
  - Will approach the dimensionless parameters relevant to the power plant core
- **Demonstration of stationary performance without active profile control**
  - Necessary condition for feasibility and attractiveness of next step
  - Definition of access conditions and development of feedback control methods to obtain them
- **Test of compatibility of heat and particle flux handing solutions with high performance core scenarios**
  - Most difficult step from the physics perspective

Conclusions

- **The quest for steady-state tokamak operation is a rich scientific endeavor**
  - Theory and experiment working in concert have yielded greater understanding of magnetized plasmas, which in turn is used to design and optimize attractive steady-state solutions
- **The tremendous progress in understanding and optimizing steady-state scenarios gives us hope that steady-state tokamak operation can be realized for fusion energy production**