PROGRESS IN MFE SCIENCE — TOKAMAK RESEARCH

by

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Division of Plasma Physics Meeting
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MFE-Tokamak

RESEARCH RESULTS FROM

Alcator C–mod	JET	TCV
ASDEX Upgrade	JFT–2M	TdeV
Compass–D	JT 60U	TEXTOR
DIII–D	MAST	TEXT
ET	NSTX	TFTR
FTU	PBX–M	TORE–SUPRA
HBT–EP	PLT	TRIAM–1M
HIT	START	T–10

SPECIAL THANKS FOR DIRECT CONTRIBUTIONS

S. Allen
S. Bernabei
L. Berry
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E. Frederickson
A. Garofalo
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F. Leuterer
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T. Luce
D. Meade
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W. Park
R. Perkins
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P. Politzer
M. Porkolab
R. Prater
F. Romanelli
S. Seitz
M. Shimada
G. Staebler
E. Strait
A. Sykes
E. Synakowski
T. Taylor
A. Turnbull
R. Waltz
M. Watkins
R. Weynants
D. Whyte
S. Wolfe

MAIN POINTS

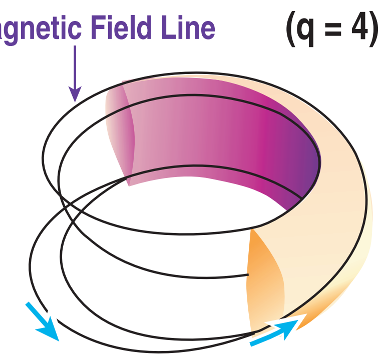
- We have learned a tremendous amount about magnetically confined plasmas
  - Measurements and theory
  - Calculations
- Exciting new directions are opening
  - Advanced Tokamak research
- We are technically ready for next steps

OUTLINE

- The tokamak equilibrium
- Heating and current drive
- Stability
- Confinement
- Power and particle control
- Burning plasma physics
- Next steps
- Conclusions

WHAT IS A TOKAMAK?

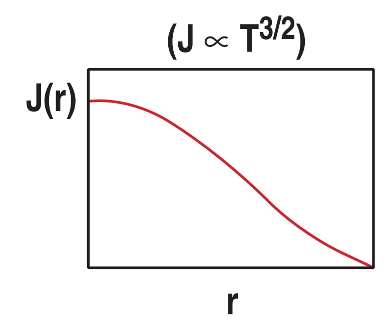
- An axisymmetric toroidal confinement configuration with a strong toroidal plasma current and an applied toroidal magnetic field strong enough to make the edge winding factor > 2

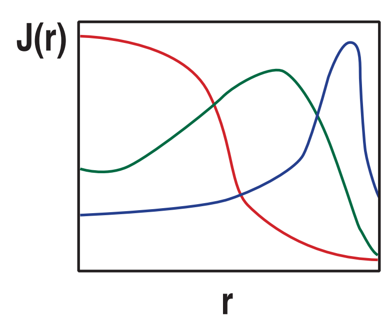


- Not part of the basic definition but certainly part of the opportunity for variation and innovation within the concept are:

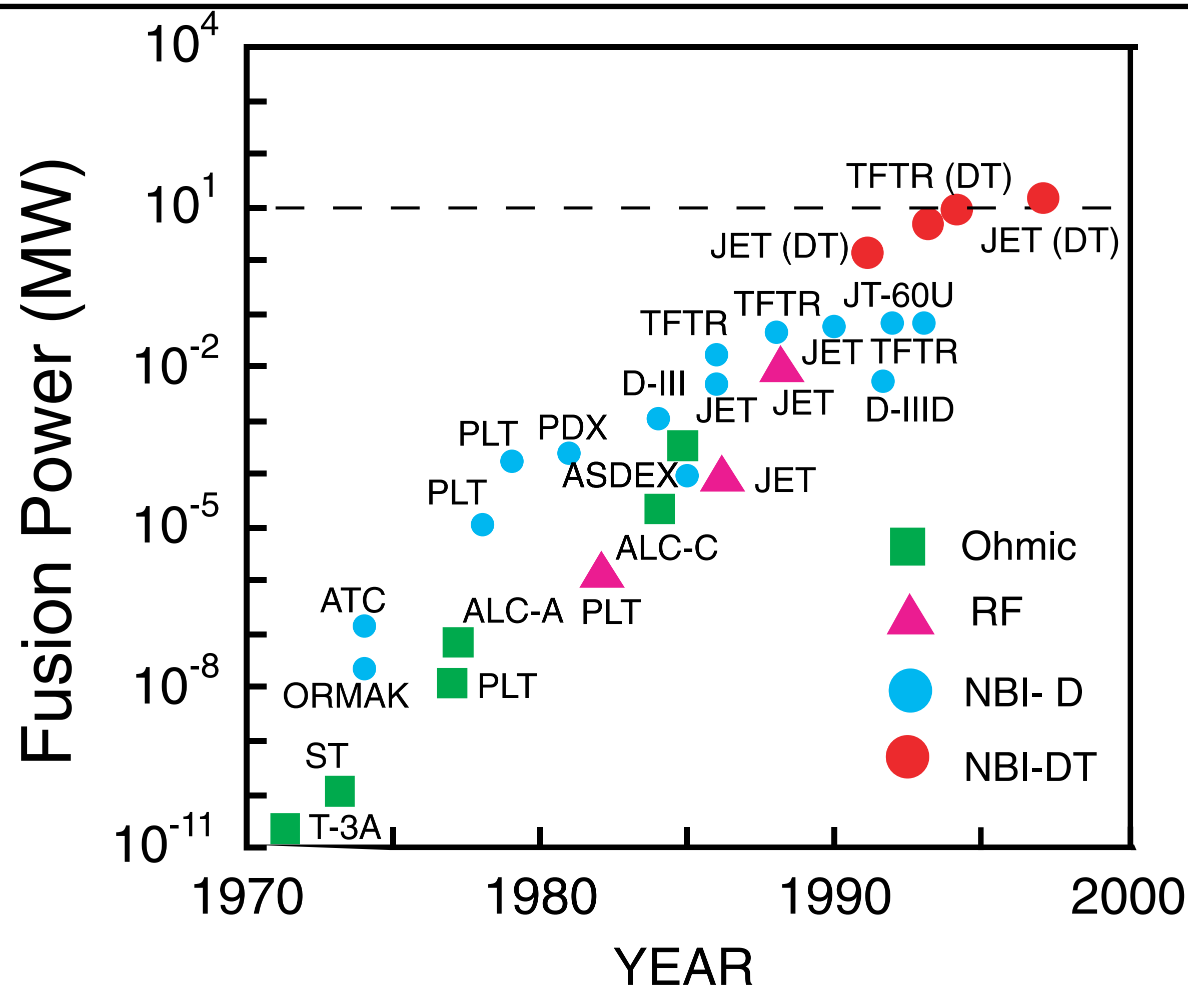
  - Shape (elongation, triangularity)
  - Aspect ratio
  - Divertor or limiter boundary
  - Toroidal field strength
  - Current profile
  - Pressure profile
  - Rotation profile
  - Radial electric field profile
  - Wall stabilization

  } Advanced Tokamak (covers: Current profile, Pressure profile, Rotation profile, Radial electric field profile, Wall stabilization)





TOKAMAKS HAVE MADE EXCELLENT PROGRESS IN FUSION POWER



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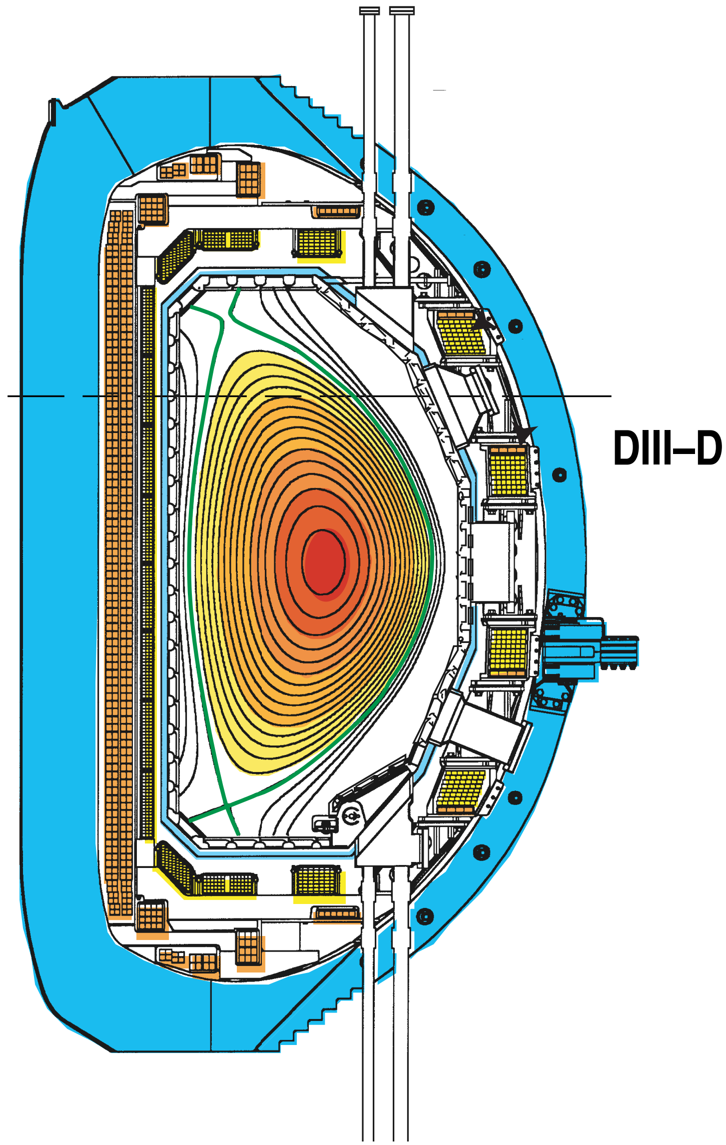
MFE-Tokamak

PLASMA EQUILIBRIUM THEORY IS WELL UNDERSTOOD AND EXTENSIVELY USED

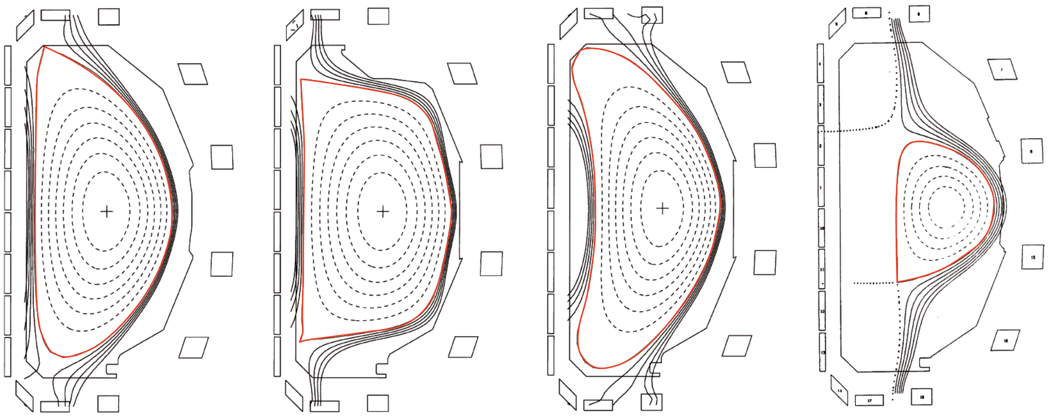
- Ampere's Law and the force balance equation $\vec{\nabla} \times \vec{B} = \mu_0 \vec{J}$ and $\vec{\nabla} P = \vec{J} \times \vec{B}$ lead to the Grad-Shafranov equation for the poloidal flux function.

  Equilibrium codes solve this equation for the closed flux contours that give the tokamak its good confinement.

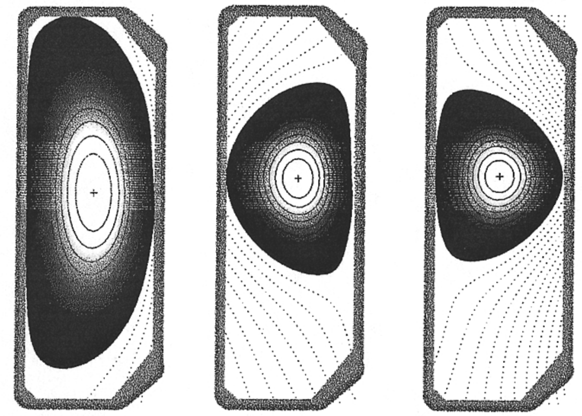
- Such codes are used extensively in
  - Experiment design, control of complex shapes is precise
  - On-line data analysis $W(t)$, $\beta(t)$, $\tau_E(t)$
  - Providing the geometry for transport analysis



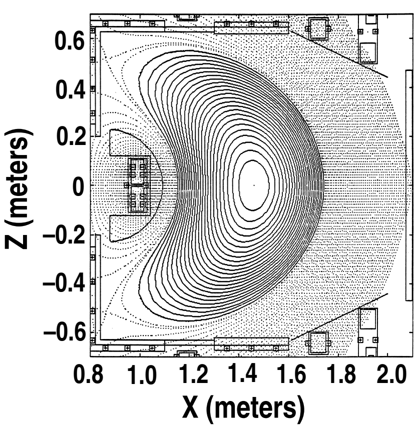
PLASMA EQUILIBRIUM SHAPE CONTROL IS A HIGHLY DEVELOPED SCIENCE



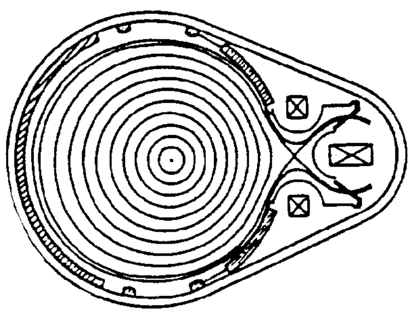
DIII–D



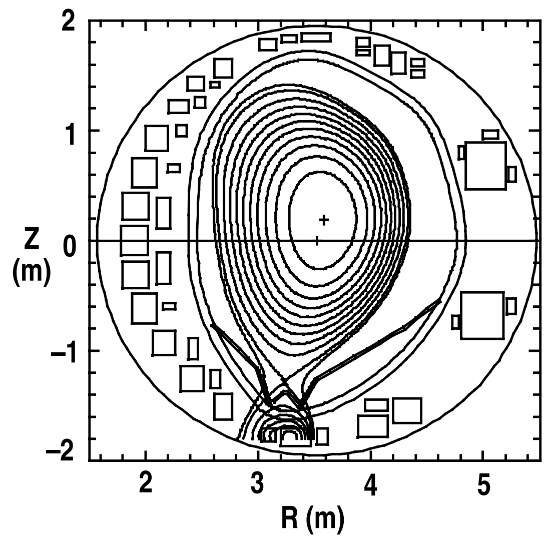
TCV



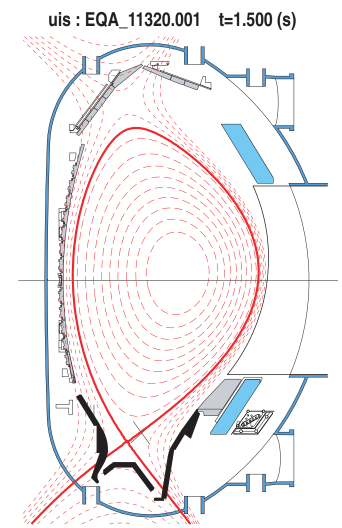
PBX–M



JT-60



JT–60U

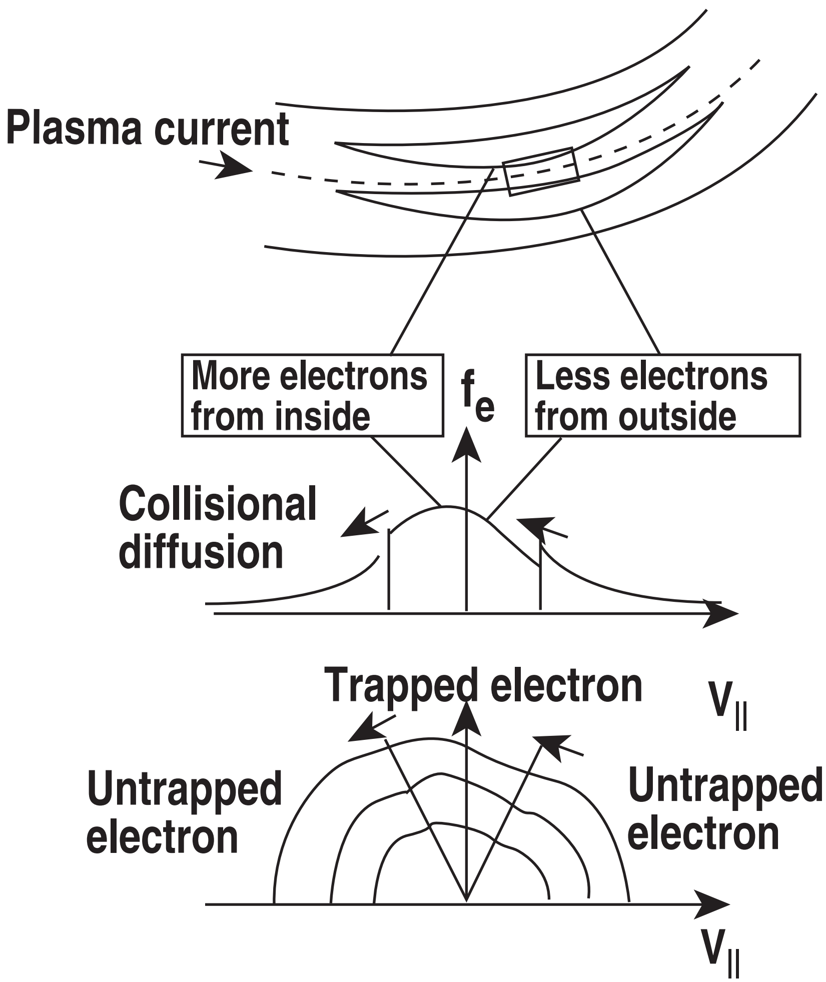


ASDEX UPGRADE

SUCCESSFUL METHODS OF HEATING AND CURRENT DRIVE FOR STEADY-STATE HAVE BEEN DEVELOPED

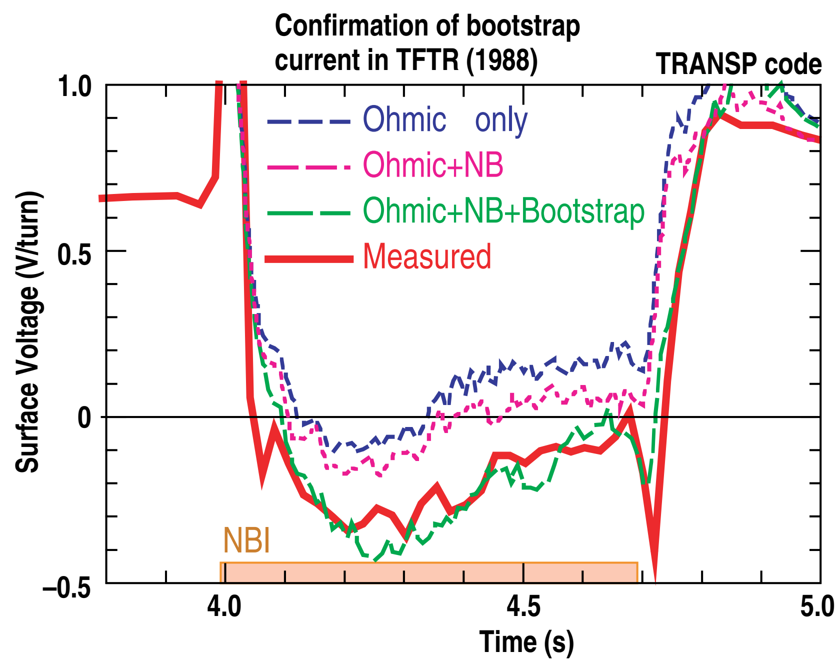
70's	80's	90's
Explored heating methods Wave coupling Fast ion orbits NBI deposition	Multi-MW heating Current drive Heating to H–mode Global rotation Measured bootstrap current Ray tracing codes Fokker-Planck codes	Control of current profile Control of MHD activity High bootstrap fraction Full wave codes

THE PLASMA'S SELF-GENERATED BOOTSTRAP CURRENT IS THE BASIS FOR MODERN APPROACHES TO STEADY-STATE OPERATION



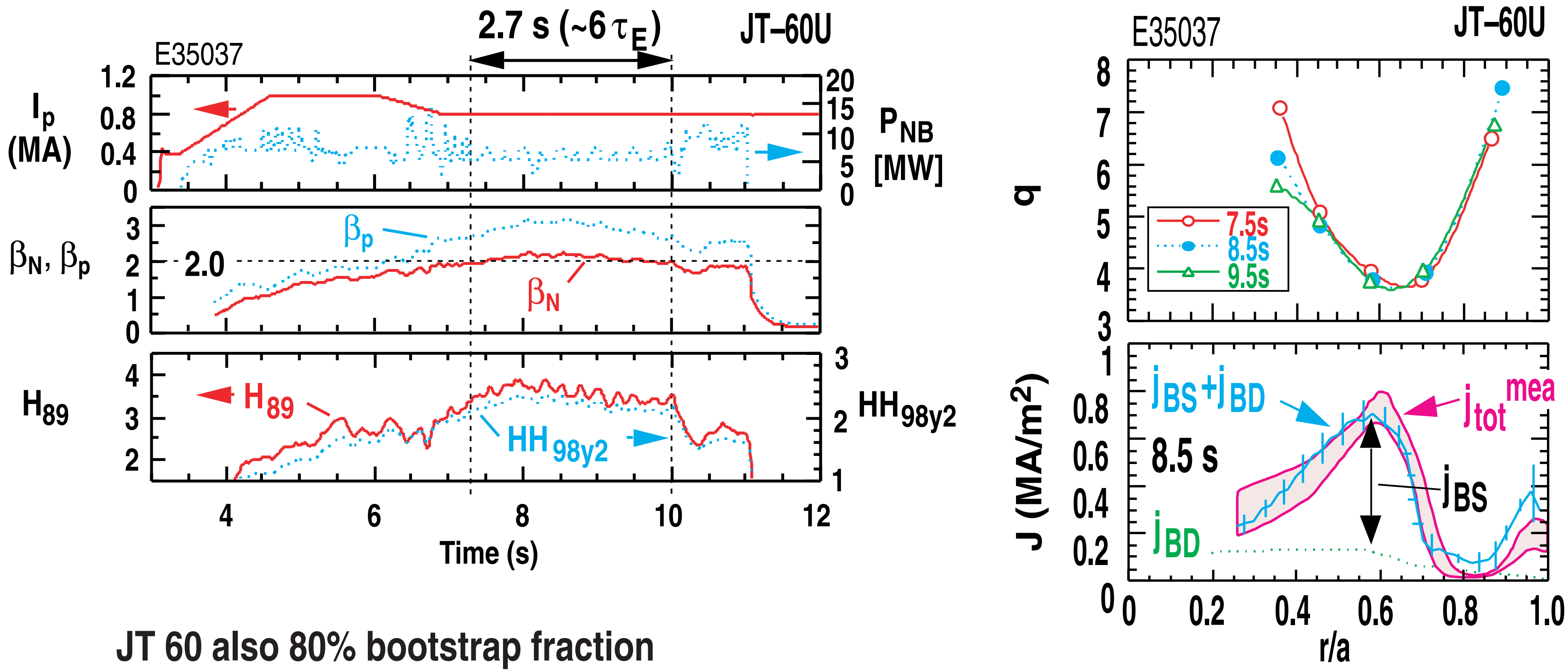
(Kikuchi, PPCF 37 (1995))

- An element of neoclassical transport theory
- $J_{bs} \propto$ local pressure gradient



A HIGH PERFORMANCE PLASMA WITH FULL NON–INDUCTIVE CURRENT DRIVE AND 80% BOOTSTRAP FRACTION IN JT–60U

- H_{89}~3.5, HH_{98y2}~2.2, β_N~2, β_p~2.9, f_{BS}~80% for $6\tau_E$ with full non-inductive CD
- Current profile was largely determined by the bootstrap current, and was nearly stationary



JT 60 also 80% bootstrap fraction

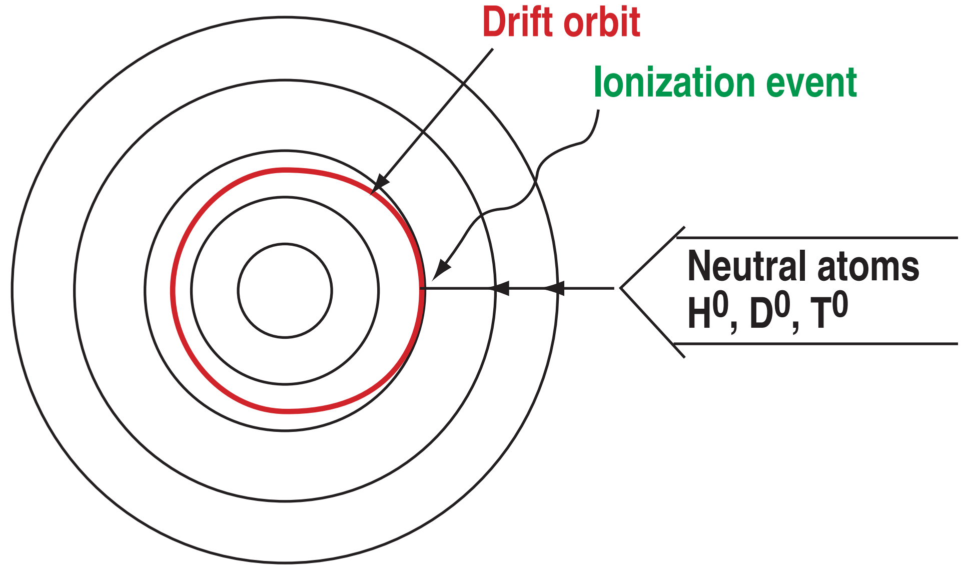
NEUTRAL BEAM HEATING AND CURRENT DRIVE

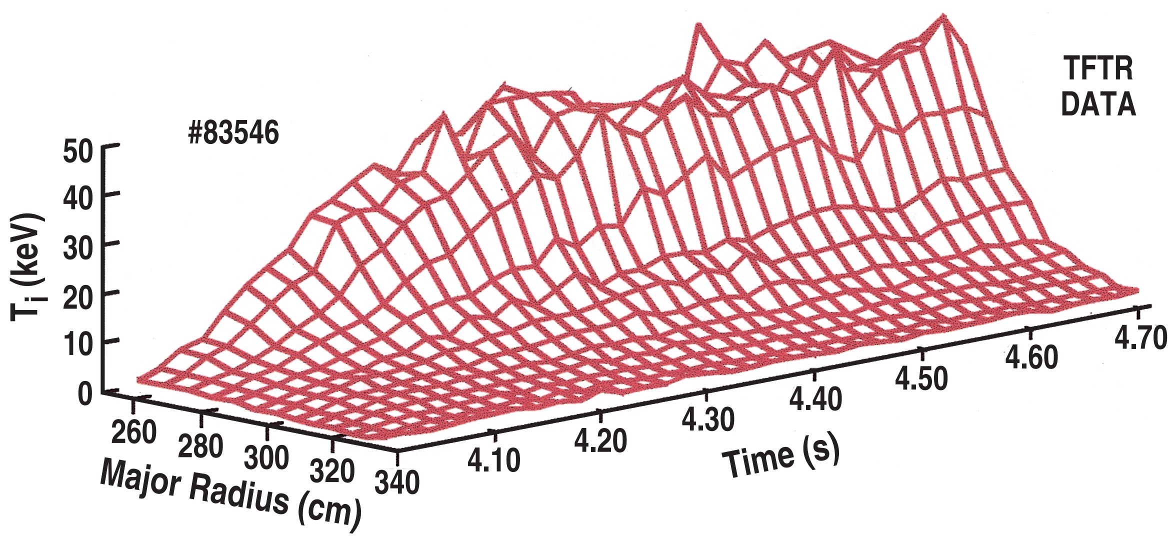
- Workhorse for high temperature and β studies
- Can drive current

Ion Sources E_b

Positive ions $\leq$ 150 keV

Negative ions $\leq$ 1 MeV





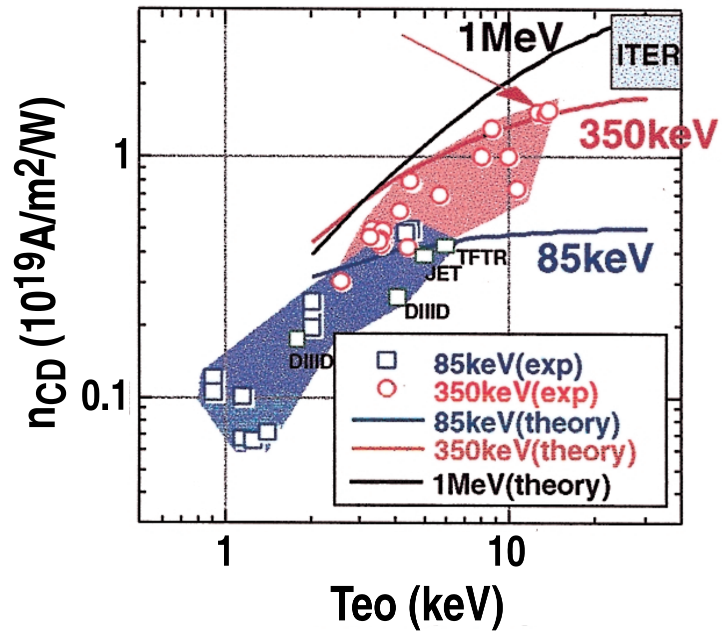
NEUTRAL BEAM CURRENT DRIVE IN ACCORD WITH THEORY

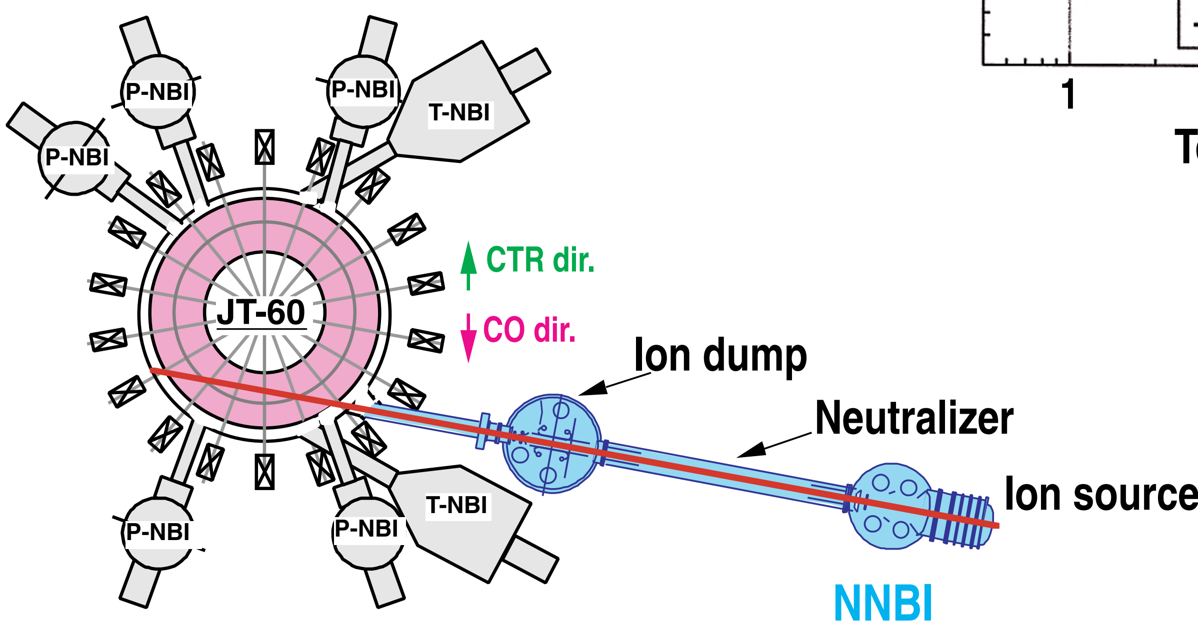
Full current drive case in JT–60U (1.3 s)

I_p = 1.5 MA B_T = 3.7 T
HH = 1.3–1.4 β_N = 2.4–2.5

	I_{CD}	E	P
NNBCD	0.6 MA	360 keV,	4 MW
PNBCD	0.3 MA	85 keV	10–18 MW
BOOTSTRAP	0.8 MA		
ECH	—	—	1.6 MW
	1.7 MA		

Efficiency
$\eta_{CD} = nI_{CD}R/P = f(Te)$





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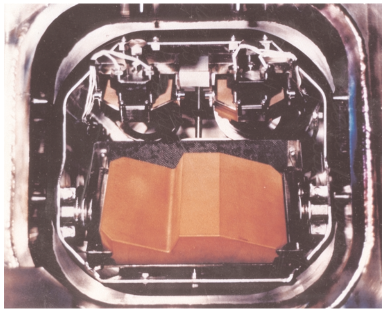
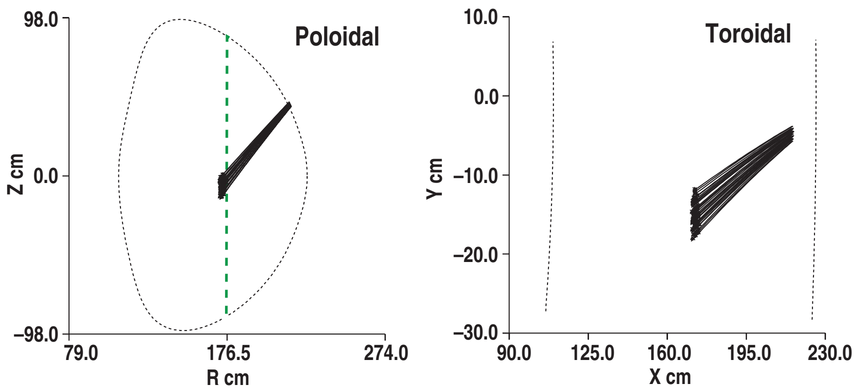
ELECTRON CYCLOTRON HEATING AND CURRENT DRIVE ($\omega = n\omega_{ce}$)

- Waves propagate in vacuum, so antenna can be far from the plasma

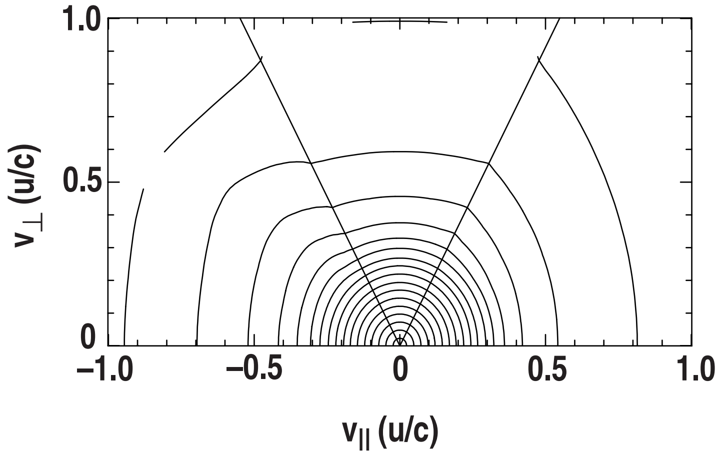


DIII–D

- Inside the plasma the waves propagate up to a critical density (related to the plasma frequency) and are absorbed near the cyclotron resonance or its harmonics

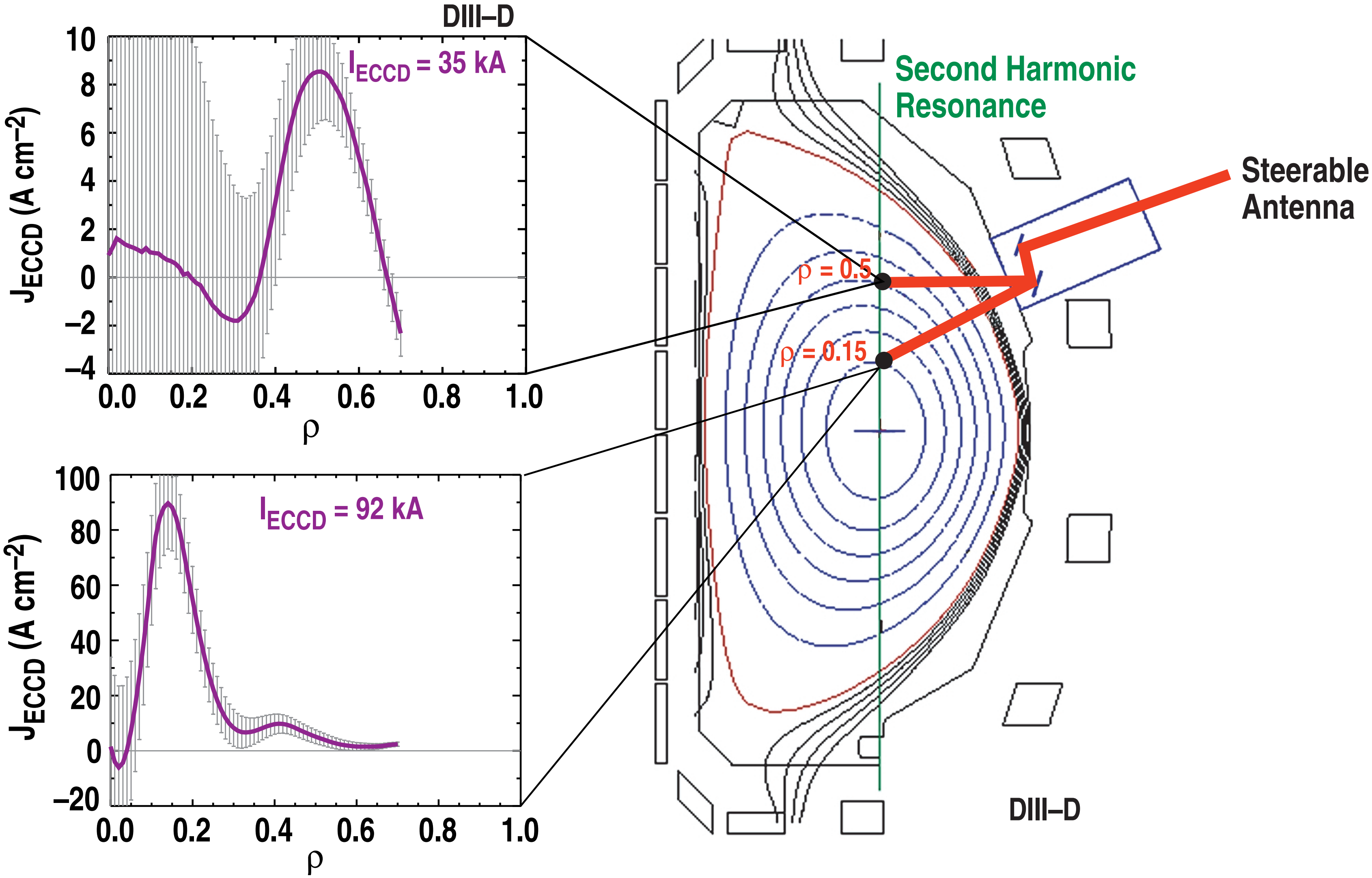


- Damping of EC waves causes diffusion in $V_\perp$ direction. Collisional relaxation on ions generates current through generation of an asymmetric $V_{||}$ distribution



- Calculational tools include ray tracing codes (TORAY, GENRAY, BANDIT-3D) and Fokker-Planck codes (CQL3D, BANDIT-3D, Giruzzi, RELAX, Krivenski, Fukuyama)

MICROWAVE ELECTRON CYCLOTRON HEATING PROVIDES LOCALIZED CURRENT DRIVE



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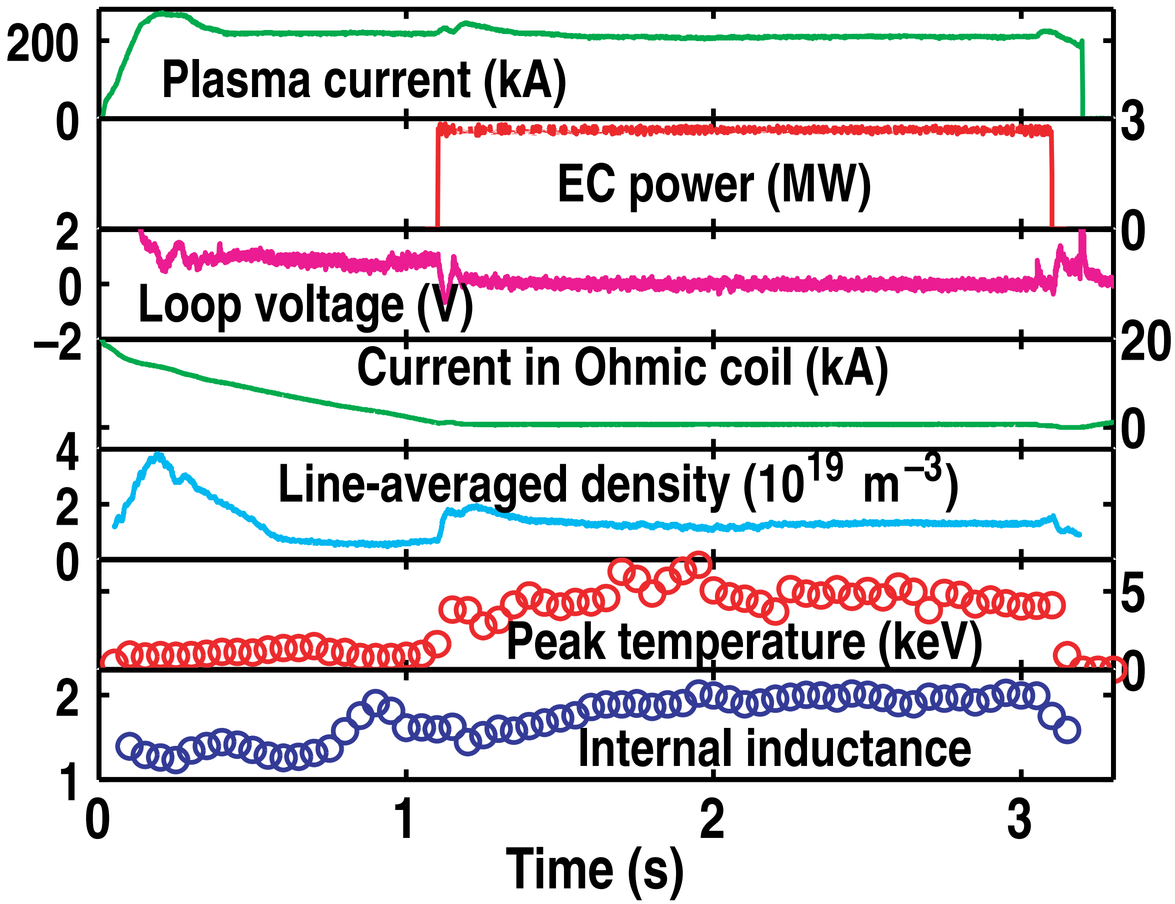
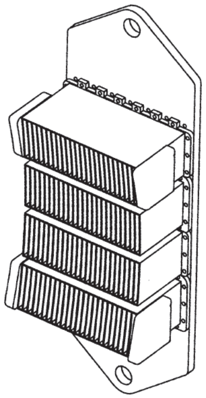
Fully non-inductive discharges

210 kA sustained in steady state by 2.7 MW co-ECCD



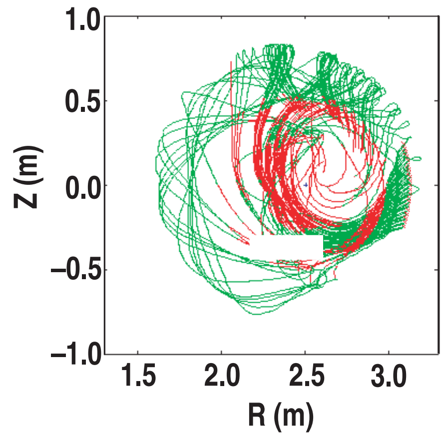
IRn/P (10^{20} A-M^{-2}-W^{-1}) =
7.3×10^{-3}

LOWER HYBRID HEATING AND CURRENT DRIVE ($\omega_{ci} < \omega < \omega_{ce}$)



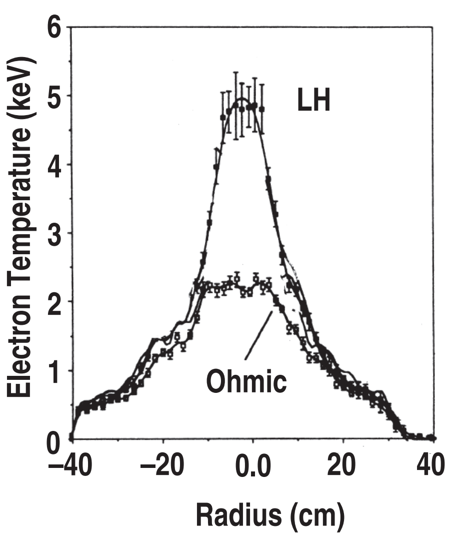
Lower Hybrid coupling requires $n_{||} > 1$
(Brambilla, SWAN)

Phased array or waveguides

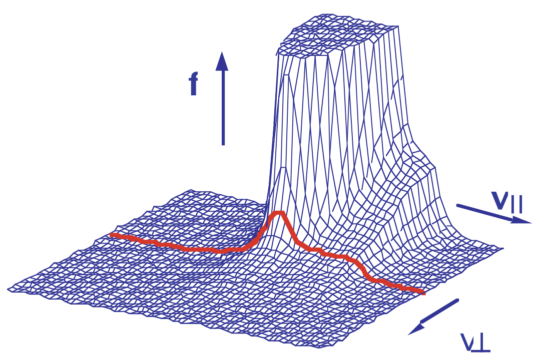


Ray tracing: the accessible waves cross the plasma and can undergo several reflections at the edge before being absorbed.

Codes by: Cardinali, Bonoli, Ignat, Valeo, Harvey, Takase
(Figures from Giruzzi)



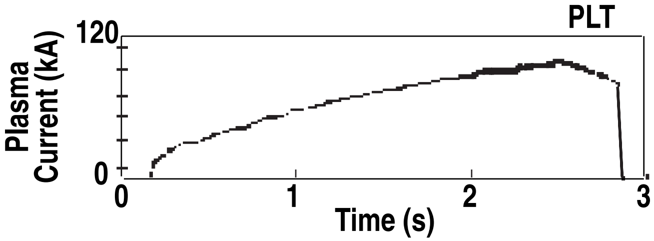
Electrons heated by LH (PLT)



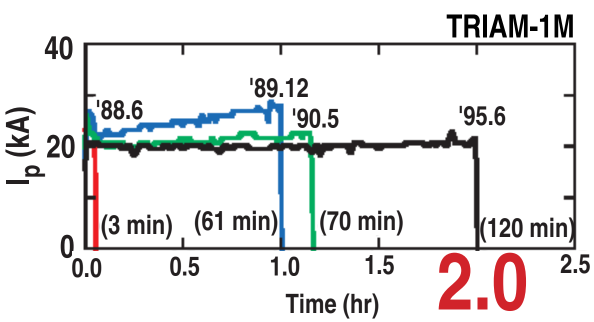
Damping of LH waves forms a parallel energetic electron tail in the distribution function via Electron Landau Damping. This asymmetry constitutes the non-inductive current (Fisch, Karney)

LHCD SUCCESSFUL IN MANY APPLICATIONS

- Plasma current initiated and ramped up by LHCD



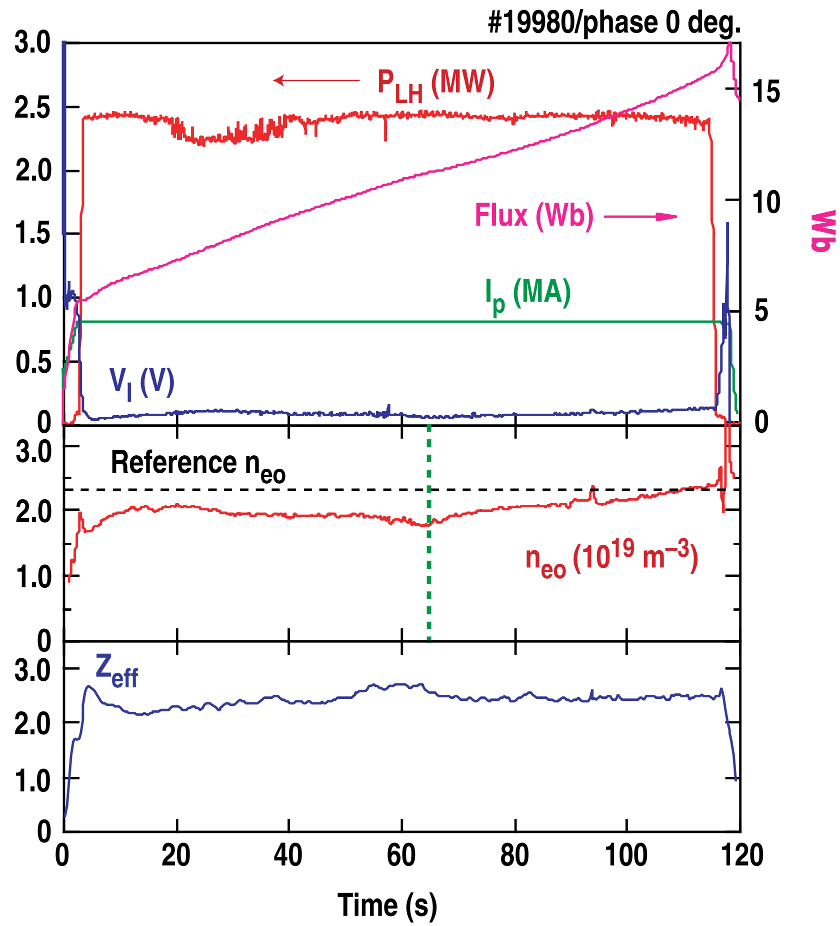
- Plasma current maintained in steady state:
  - JET; 3 MA, 4 s
  - TRIAM-1M; 20 kA, 2 hr



2.0 Hours!

Euratom cea   TORE SUPRA

- 2-minute-long discharge at I_p = 0.8 MA
- Injected energy = 290 MJ



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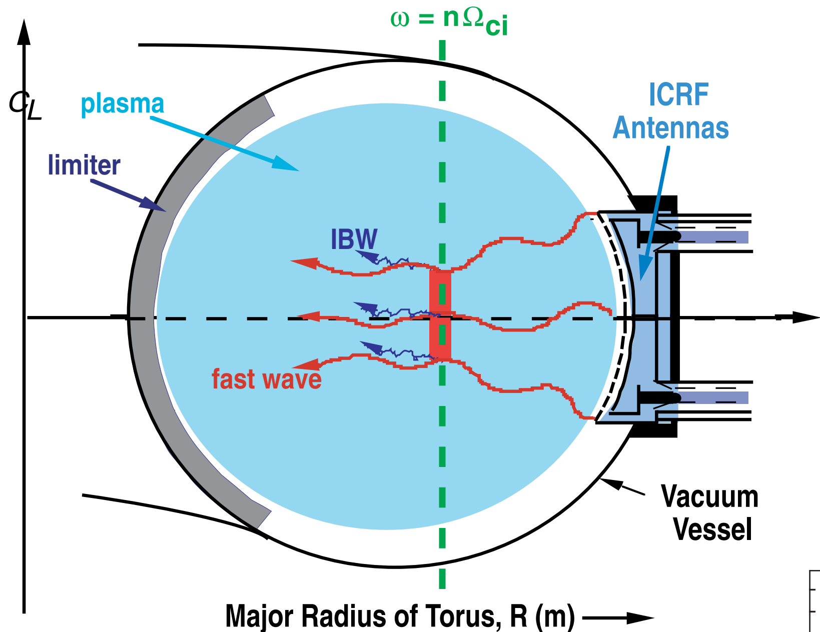
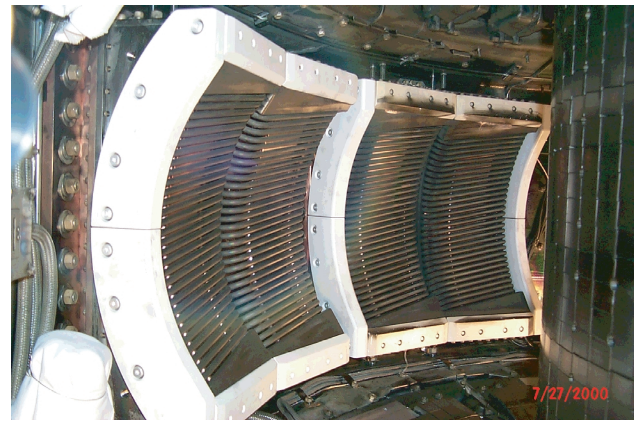
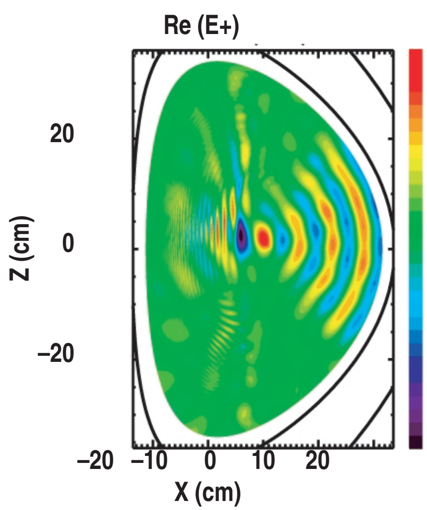
ICRF HEATING AND CURRENT DRIVE ($\omega = n\omega_{ci}$) INVOLVES WAVE EXCITATION, PROPOGATION, ABSORPTION AND MODE CONVERSION





Alcator C–Mod Coupler

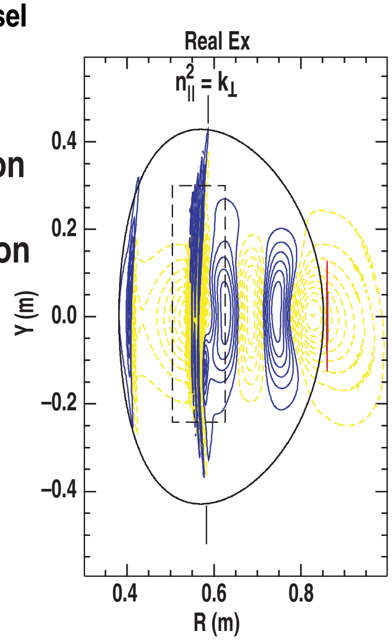
Codes
RANT3D
FELICE



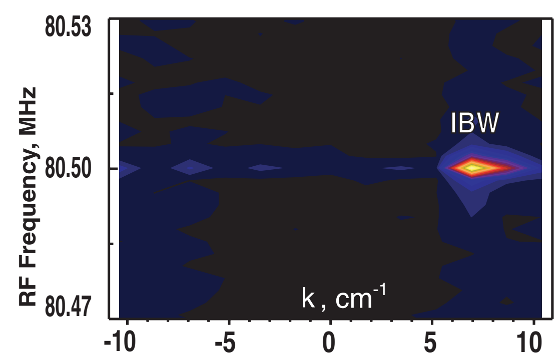
Wave Propagation
(TORIC, PICES)

Absorption
Mode
Conversion

AORSA
PICES
TORIC
METS



Alcator C–Mod



IBW Measured
PCI Diagnostic

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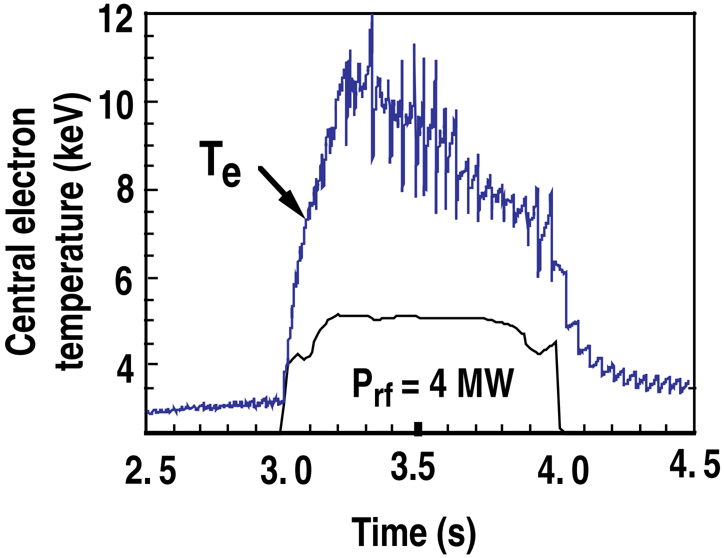
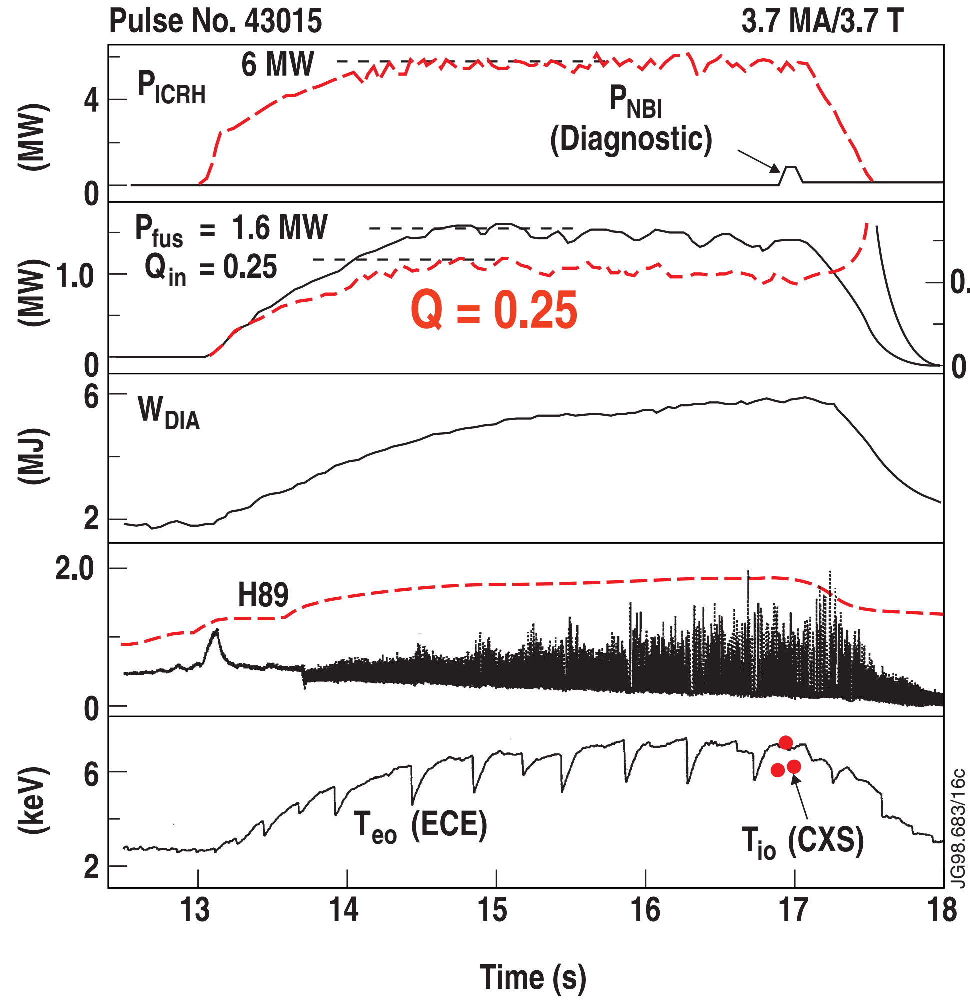
MFE—Tokamak

BASIC ICRF SCHEMES (MINORITY D AND ^{3}He, $2\omega_{CT}$) FOR A DT REACTOR HAVE BEEN VERIFIED

- Mode conversion experiments in D – ^{3}He produced the highest electron heating efficiency in TFTR



- JET: 6 MW ICRF → 1.66 MW fusion power



HEATING AND CURRENT DRIVE CHALLENGES FOR THE NEXT DECADE

90s	2000 – 2010
Control of current profile	Current profile control
Control of MHD activity	Transport barrier control
High bootstrap fraction	Coupling of Fokker-Planck, transport, and stability codes
Full wave codes	Helicity injection
	Strong alpha heating

MHD STABILITY PHYSICS MATURED IN THE 80's AND MOVED TO PROFILE OPTIMIZATION IN THE 90's

70's	80's	90's
No heating power	NBI Power	β_T = 13%
Equilibrium codes	β_T = 5%–10%	Current profile measured
Tearing modes	β-limit scaling	Theory optimization of profiles
Sawteeth	Pressure profile measured	Profile variation and control in experiments
Current limits	Kink codes	Wall stabilization
	Ballooning codes	Halo currents
	Shaping	Neoclassical tearing
		Second stable edge
		Advanced Tokamak

THE EFFECTS OF PLASMA INSTABILITIES RANGE FROM LOSS OF THE CONFIGURATION TO LOCAL TRANSPORT

Spatial Scale of the Mode	Mode Description	Principal Consequence
$\sim a$	Global kink modes Ideal MHD (low n)	Disruptions β and I_p limits
$\sim \frac{1}{5} a$	Tearing modes Resistive MHD Ideal Ballooning ($n \to \infty$)	Macroscopic Transport Profile Modification
$\sim \frac{1}{10} a$	Edge Localized Modes	Periodic bursts at the edge
ρ_i	Ion Temperature Gradient Modes Drift Waves	Ion Transport
ρ_e	Electron Temperature Gradient Modes Drift Waves	Electron Transport

IDEAL MHD INSTABILITIES LIMIT THE MAXIMUM BETA

Change in potential energy for a small displacement ξ:

$$\delta W = \frac{1}{2}\int dr^3 \left\{ \frac{|\delta B|^2}{\mu_o} + \frac{B^2}{\mu_o}\left|\nabla \bullet \xi_\perp + 2\xi_\perp \bullet \kappa\right|^2 + \gamma p \left|\nabla \bullet \xi\right|^2 - J_{||}(\xi_\perp \times b) \bullet \delta B - 2(\xi_\perp \bullet \nabla p)\,(\kappa \bullet \xi_\perp) \right\}$$

- field line bending
- magnetic field compression
- fluid compression

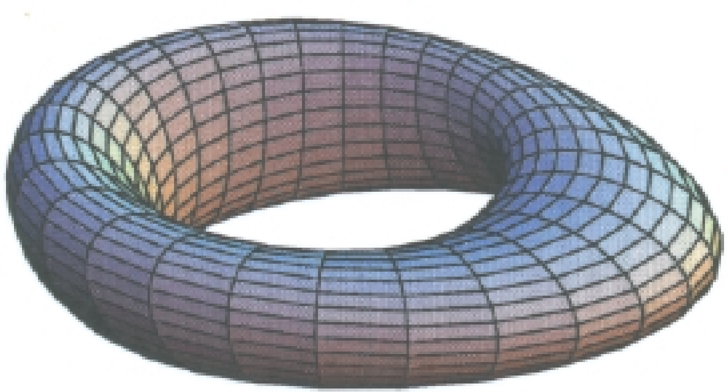
STABILIZING

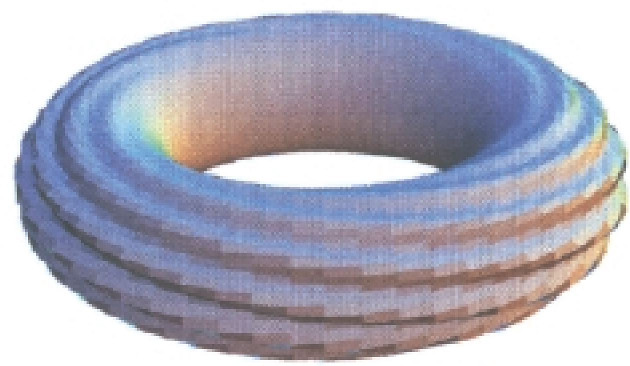
- parallel current
- pressure gradient

DESTABILIZING

Kink Mode: low n, global

Ballooning Mode: High n, localized in bad curvature region





Pressure-driven Kink (Kink-ballooning) Mode

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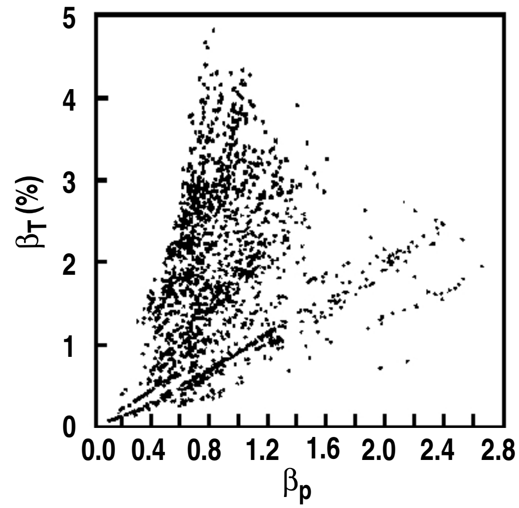
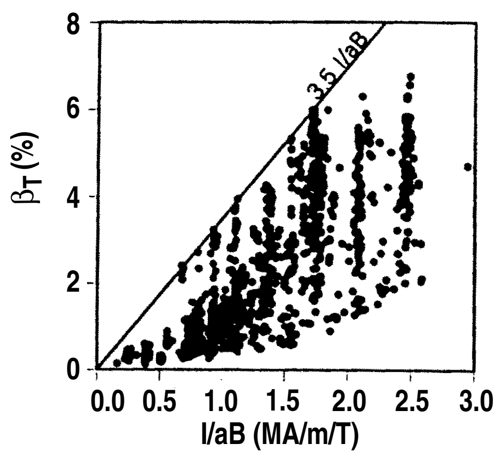
(J. Freidberg, Ideal MHD; G. Bateman, MHD Instabilities; others)

BETA LIMIT SCALINGS WERE DERIVED THAT FIT WELL EXPERIMENTAL RESULTS

Early work (Doublet III – 1984)





DIII–D Data (1988)

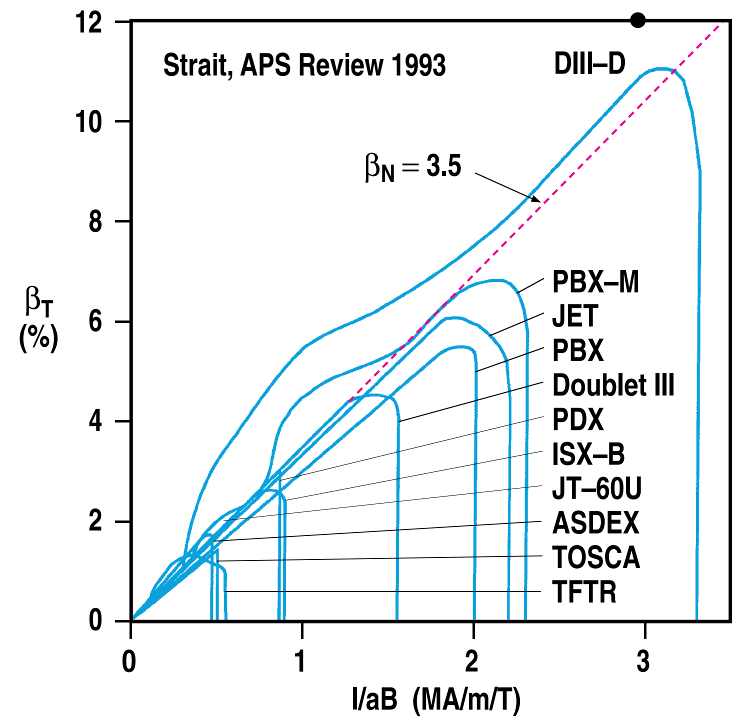
⇒

Theory calculations (1982–1984), Troyon scaling

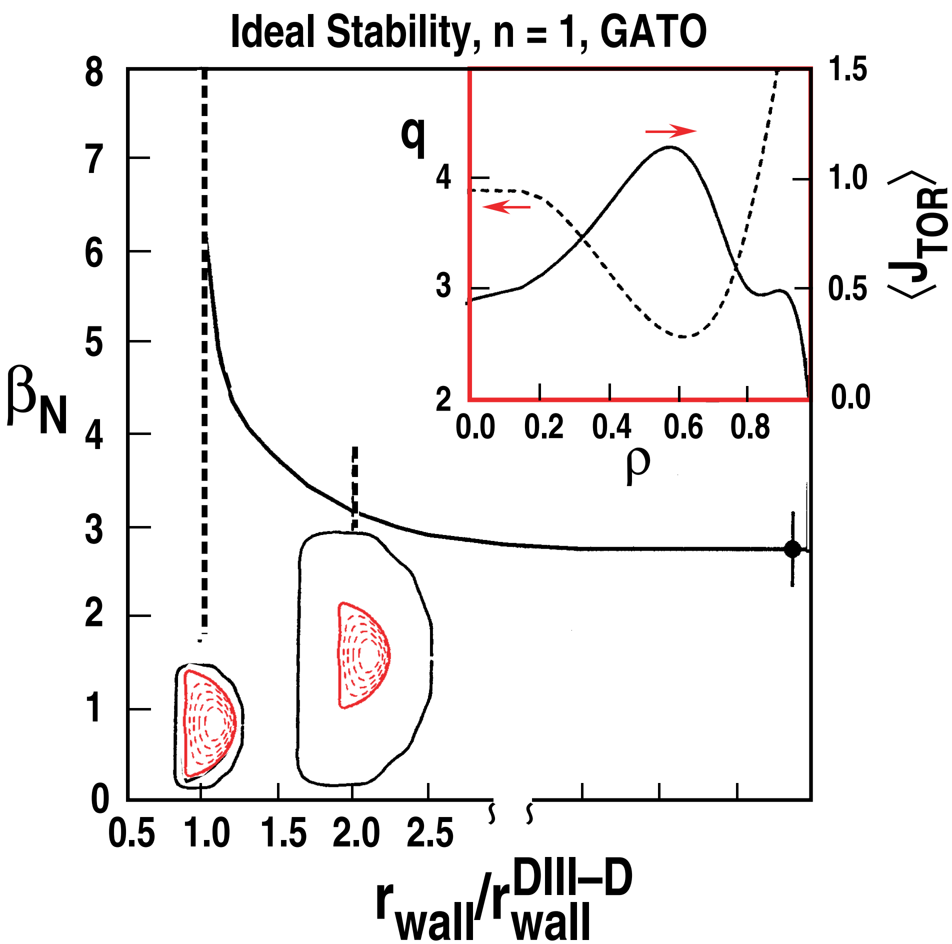
$$\beta_T\,(\%) \le 2.8\,\frac{I\,(\mathrm{MA})}{a(\mathrm{m})\,B_T\,(\mathrm{T})}$$, Define $\beta_N = \beta_T/(I/aB)$

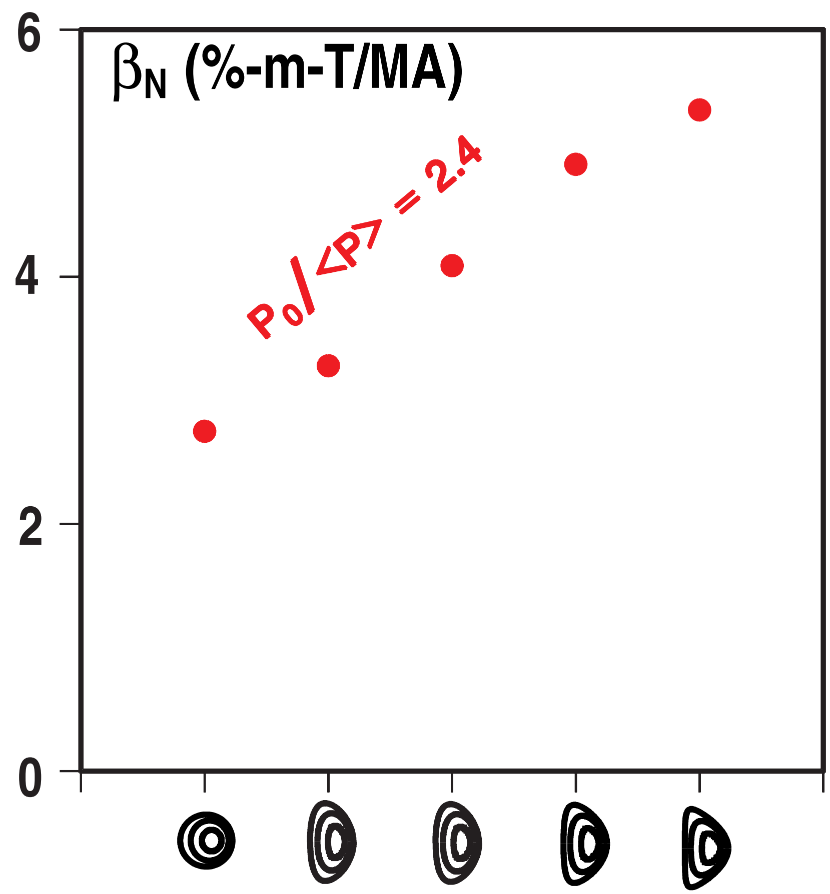
$$\beta_p\,\beta_T = 25\left(\frac{1+\kappa^2}{2}\right)\left(\frac{\beta_N}{100}\right)^2$$

- Fusion power $\beta_T^2 B^4$
- Bootstrap fraction $c\varepsilon^{1/2}\beta_p$

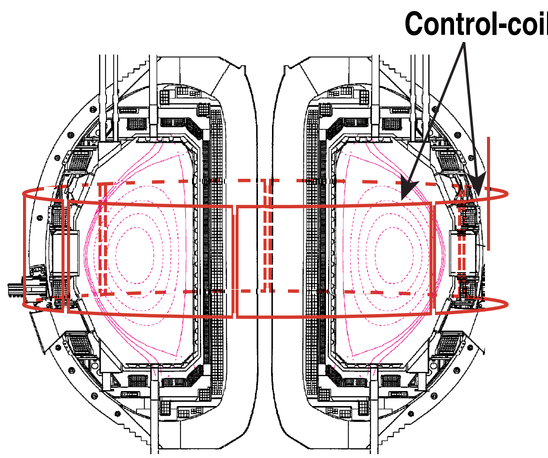


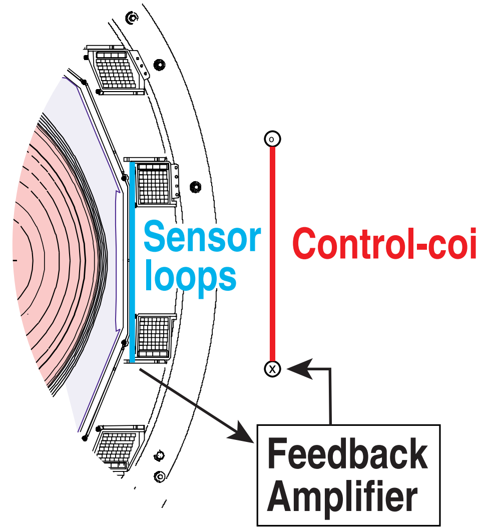
WALL STABILIZATION, PLASMA SHAPING, AND OPTIMAL PRESSURE AND CURRENT PROFILE MAY DOUBLE THE STABLE OPERATING SPACE OF THE TOKAMAK

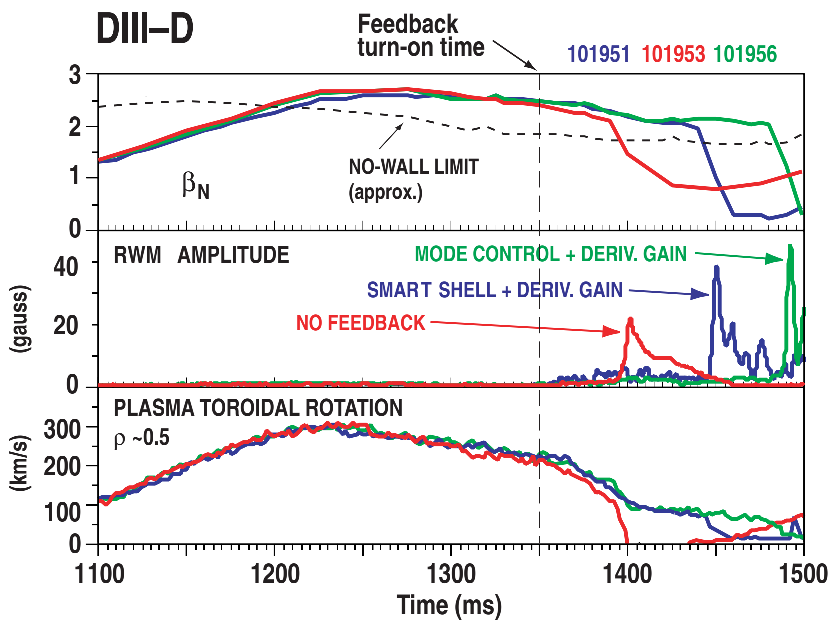


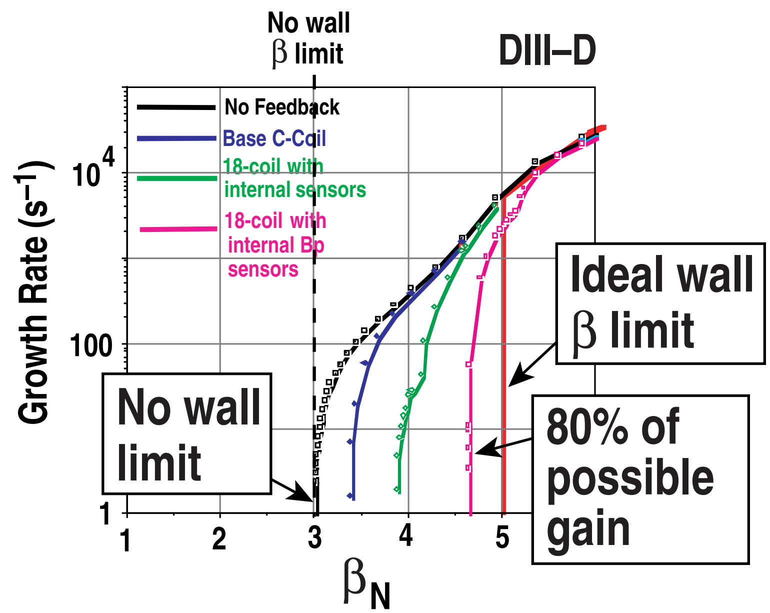


IDEAL KINK MODE GROWTH IS SLOWED BY A RESISTIVE WALL AND RESPONDS TO FEEDBACK STABILIZATION



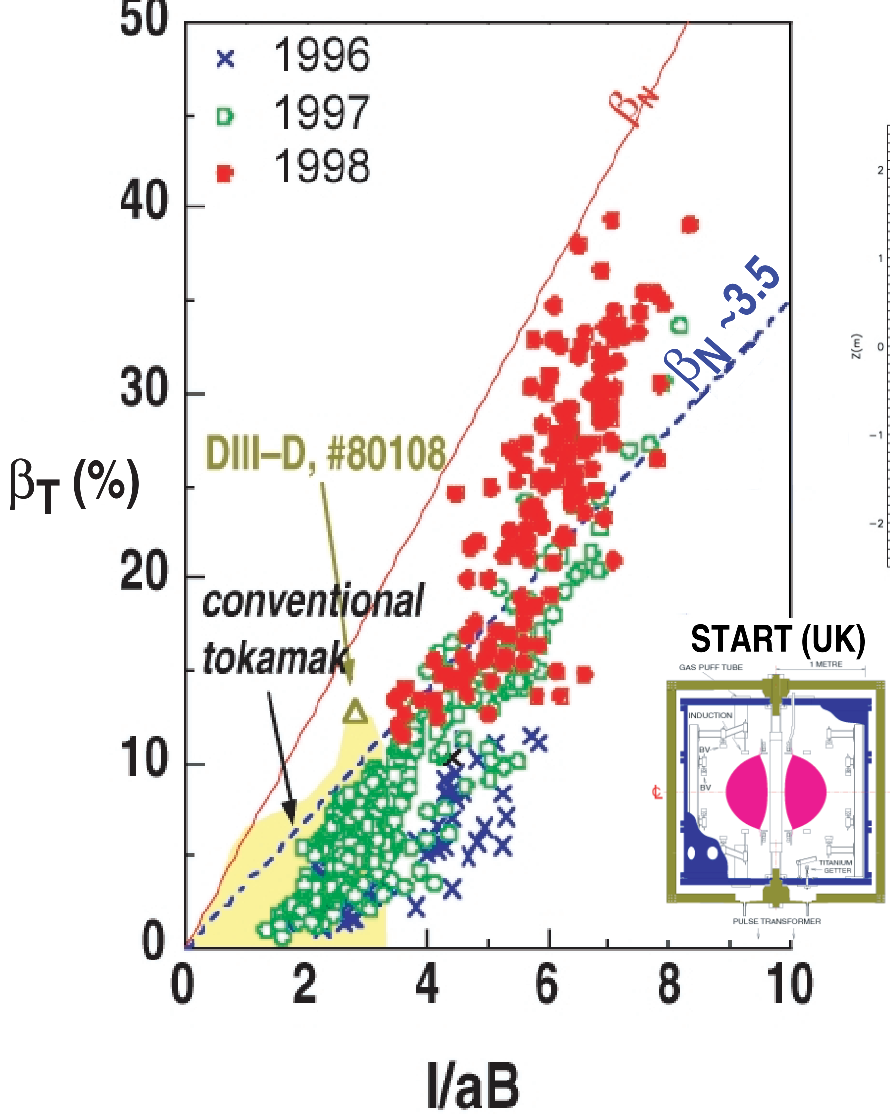




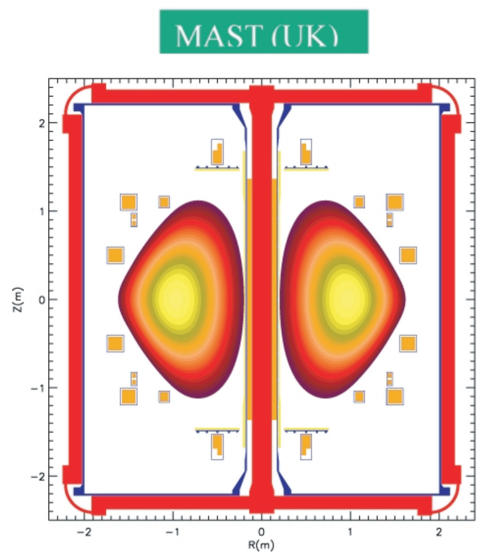


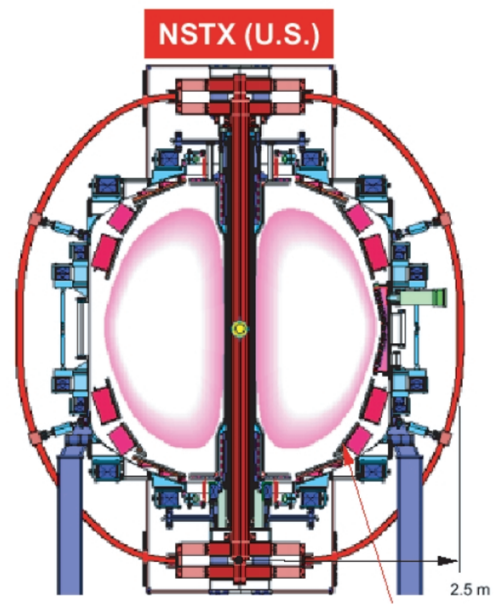
LOW ASPECT RATIO RAISES β_N and β_T

Record β on START (achieved through NB Heating)



New MA Spherical Tori





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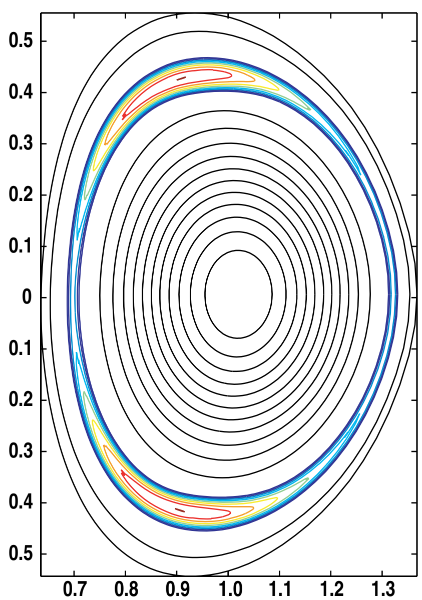
TEARING MODES

Classical

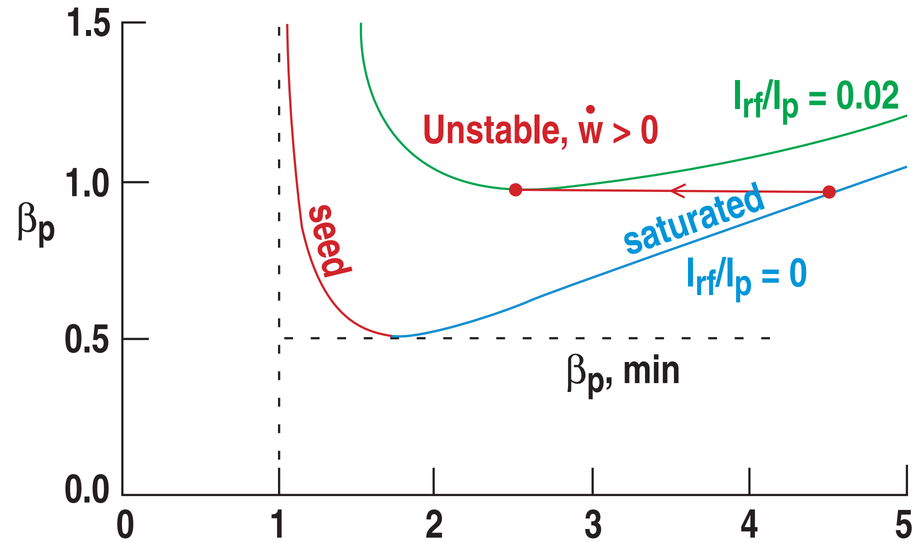
- Finite resistivity
- Current can diffuse and form clumps — magnetic islands — on rational q flux surfaces
- Driven by ∇J
- Growth time 10s of milliseconds

Neoclassical

- $\nabla P=0$ in island removes equilibrium bootstrap current
  - Helical current perturbation amplifies seed island
- Providing auxiliary current drive predicted to stabilize NTM

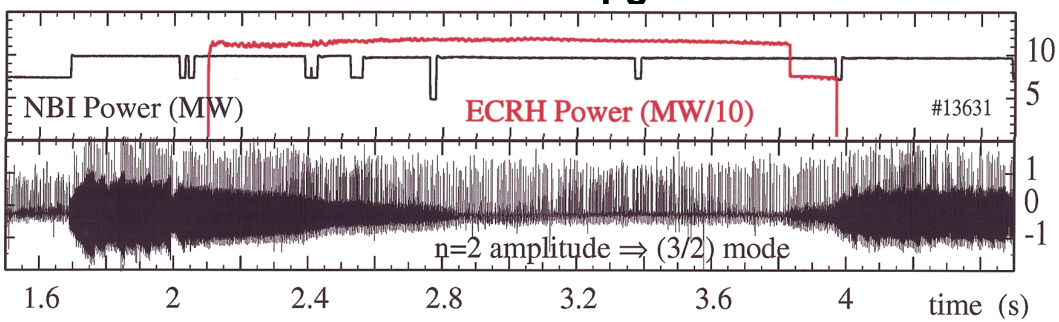


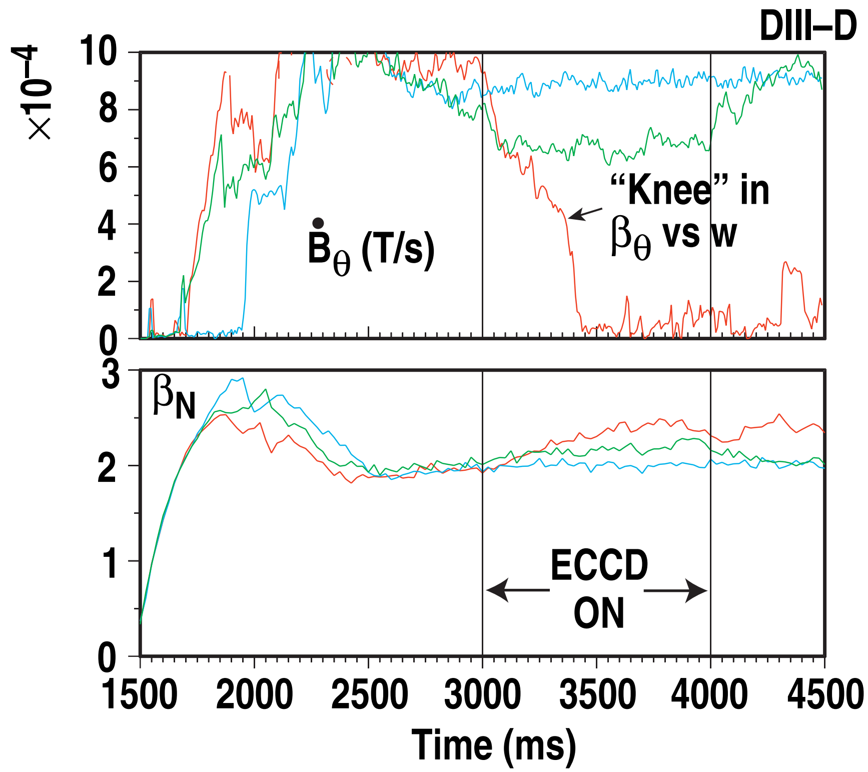
PEST–III

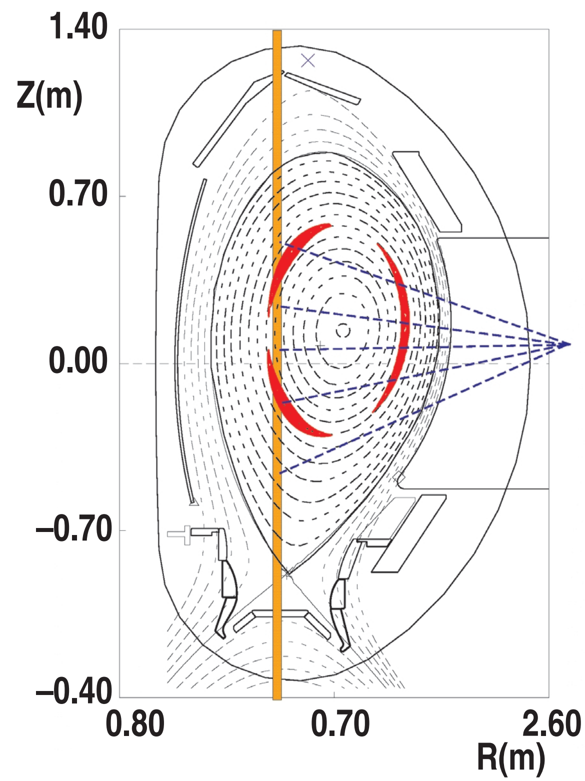


STABILIZATION OF NTMs BY ECCD

ASDEX–Upgrade



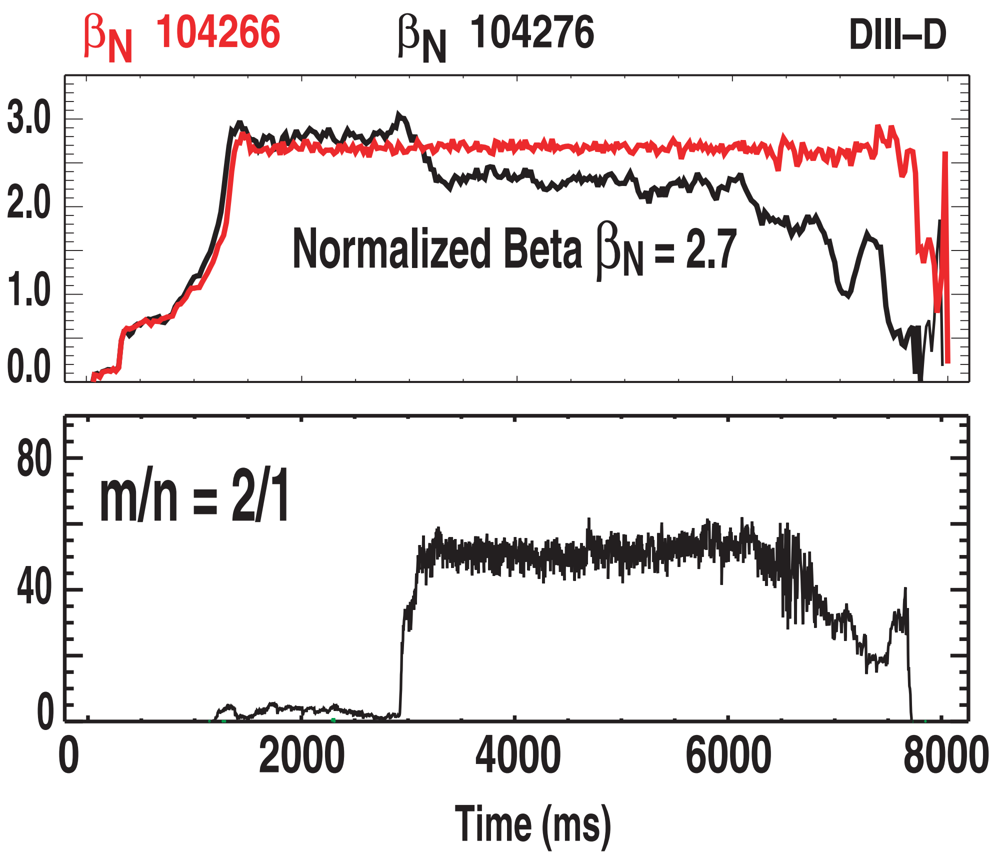




ECCD in DIII–D

Similar results from JT–60U

PRECISE CONTROL NEAR THE β–LIMIT IS THE KEY TO AVOIDING DISRUPTIONS

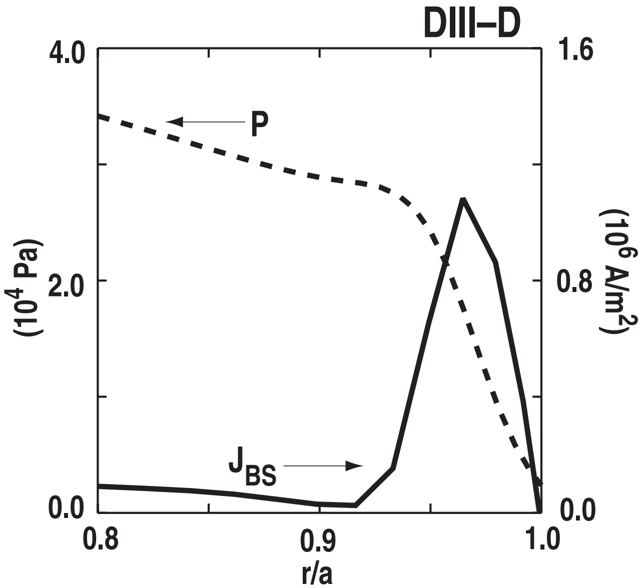


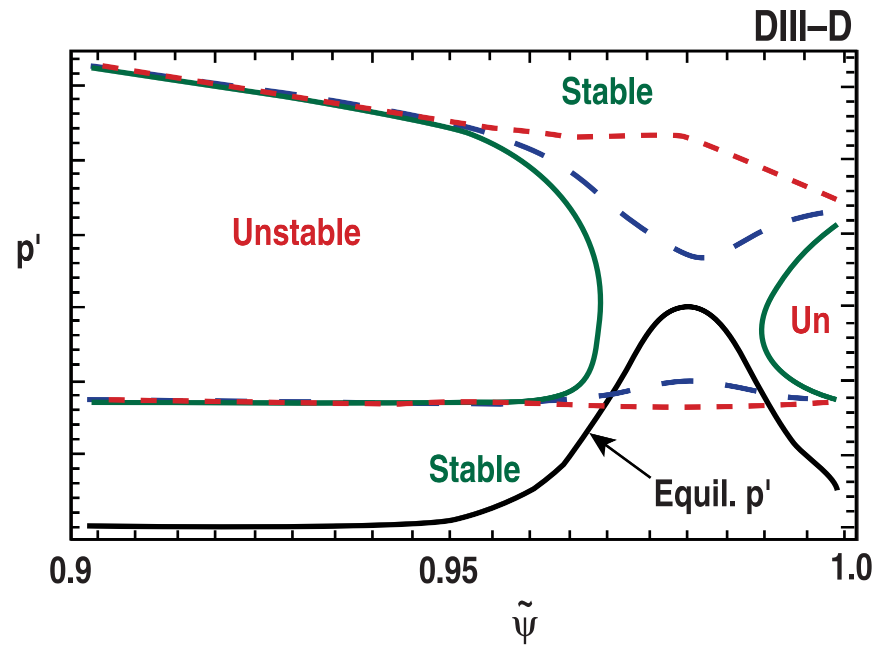
1. Need to operate close to stability limits
   - Good control
   - Knowledge of limits

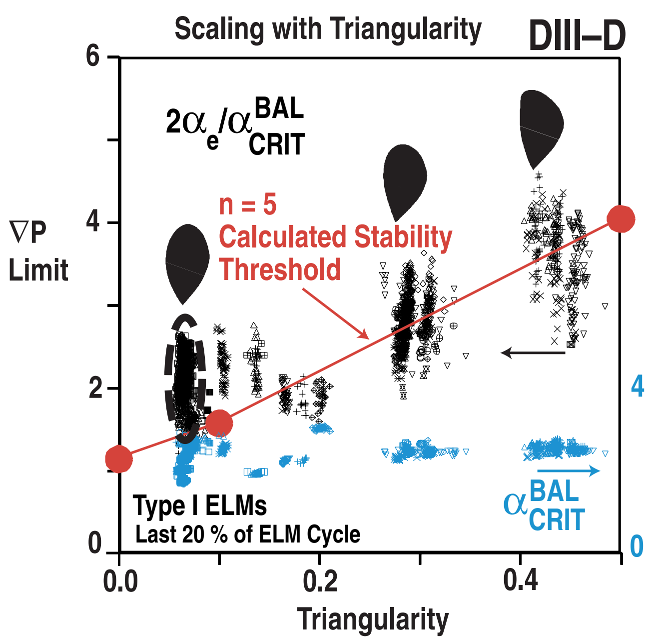
   High performance DIII–D discharge regulated 5% below 2/1 tearing limit for 35 τ_E (6.3 seconds)

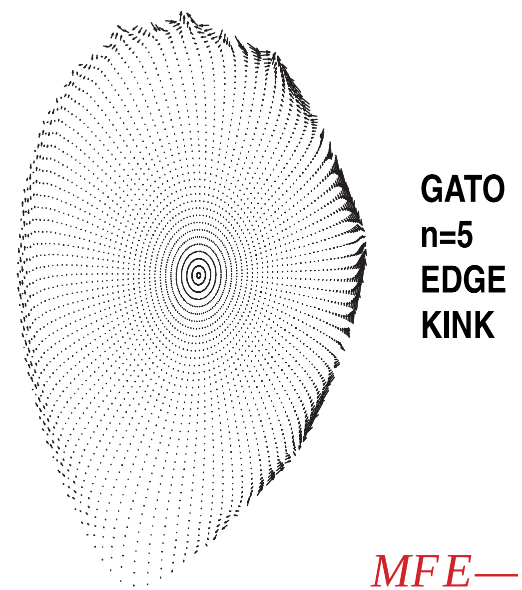
2. Mitigation of disruption consequences massive gas puff or pellets
   - No runaway electrons
   - Reduced halo currents and forces on structional components
   - Reduced heat pulses to the divertor surfaces

EDGE LOCALIZED MODES (ELMS) ARE NOW UNDERSTOOD TO BE INTERMEDIATE n KINKS









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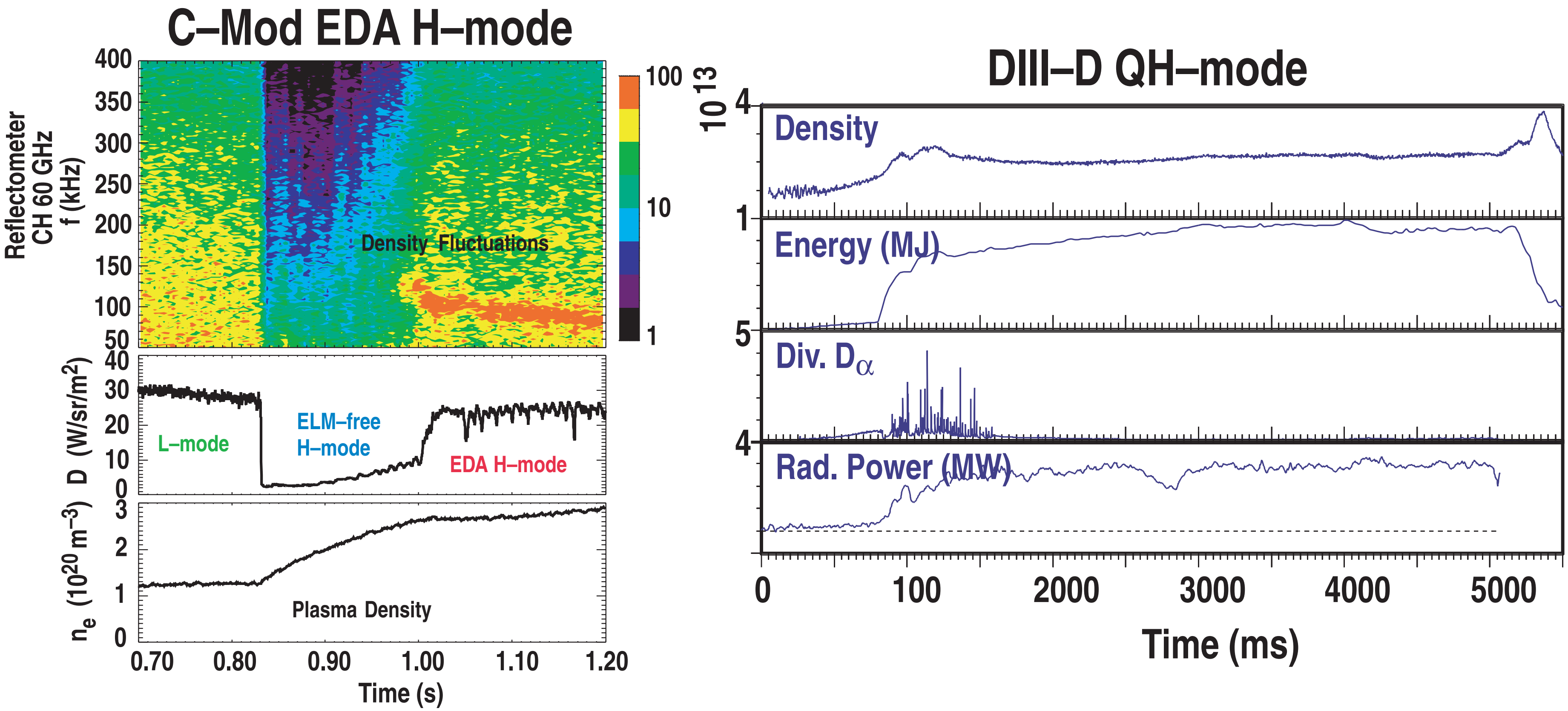
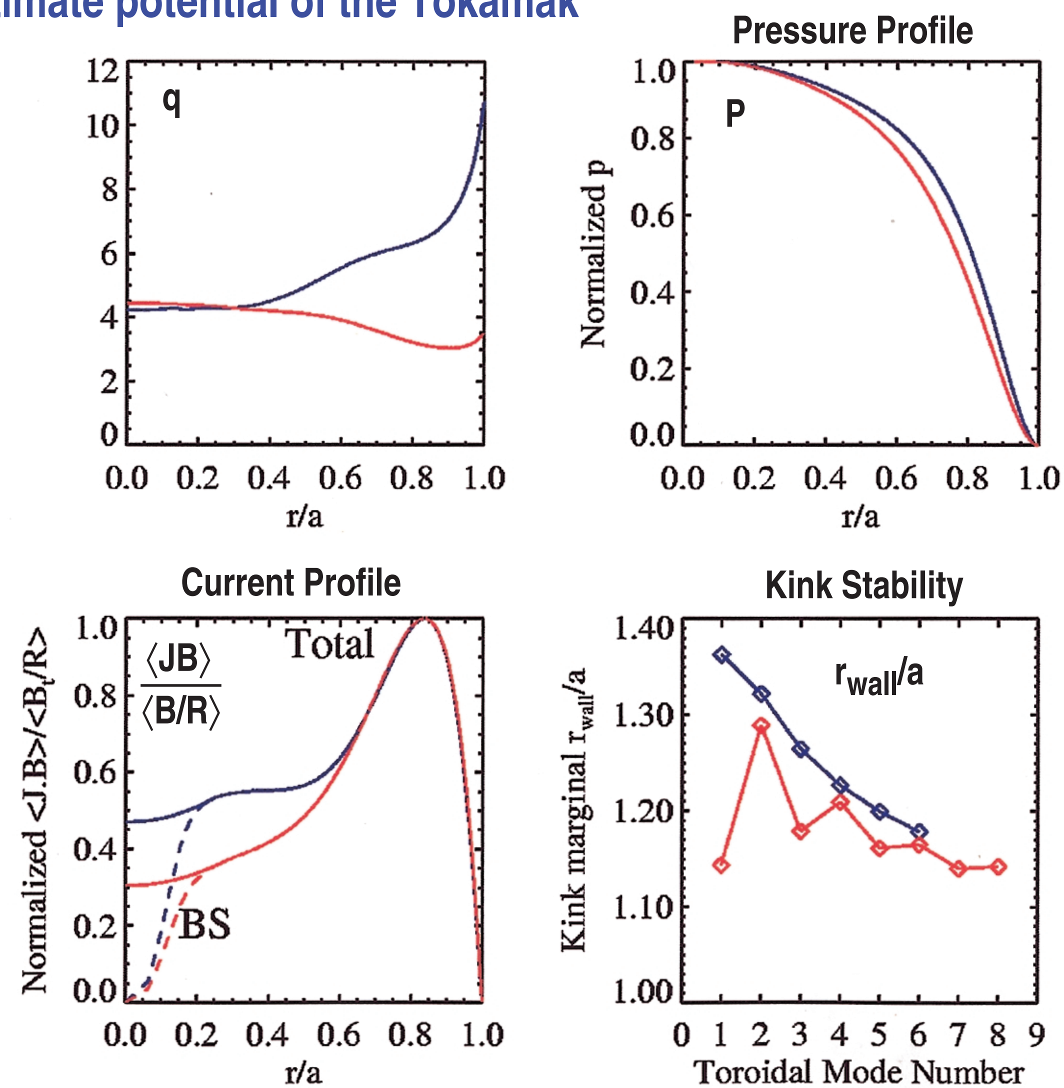
BOTH ALCATOR C–MOD AND DIII–D HAVE FOUND ELM–FREE REGIMES WITHOUT DENSITY OR IMPURITY ACCUMULATION



THE FUTURE

- Advanced Tokamak stability theory points to states with very broad pressure profiles and hollow current profiles and nearly 100% bootstrap current as perhaps the ultimate potential of the Tokamak



(J. Menard, S. Jardin, J. Manickam)

ARIES—AT

A=3.3

κ=2.5

δ=0.6

β=14%

β_N=6

ARIES—ST

A=1.6

κ=3.6

δ=0.64

β=56%

β_N=8.2

STABILITY CHALLENGES FOR THE NEXT DECADE

90's	2000–2010
β_T = 13%	Wall stabilized β-limit
Current profile measured	Bootstrap fraction $\rightarrow$ 100%
Theory optimization of profiles	Pressure and current profile control
Profile variation and control in experiments	Very hollow J(r)
Wall stabilization	Broad pressure profiles
Halo currents	Optimum edge stability
Neoclassical tearing	Feedback stabilization or avoidance of neoclassical tearing
Second stable edge	Disruption mitigation
Advanced Tokamak	3–D MHD, understand disruptions away from β-limit

THE 90's HAVE SEEN EXCITING ADVANCES IN CONFINEMENT SCIENCE

70s

- Global τ_E
- Variable results
- Linear theory scaling

80s

- Reproducible results (Empirical scaling)
- 1–D Transport codes
- 1–D Profile measurements
- H–mode edge barrier

90s

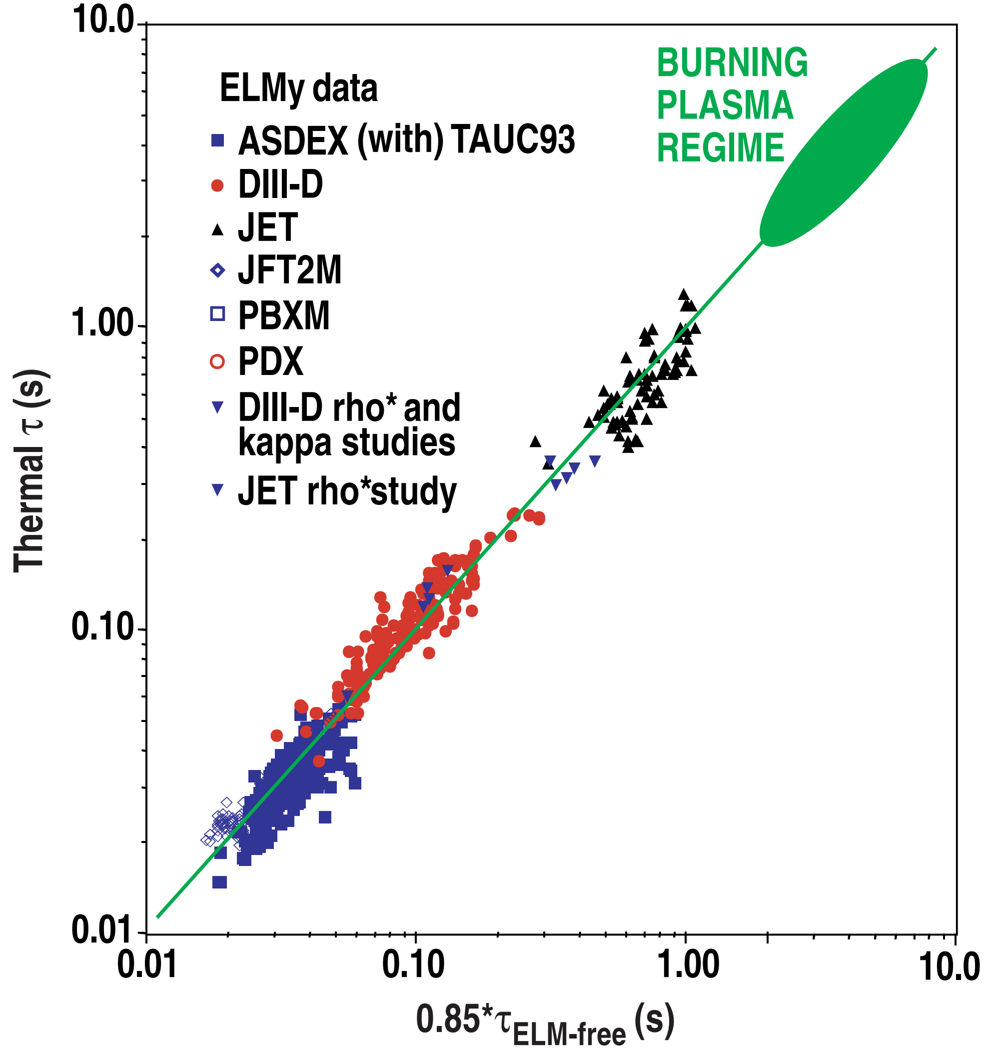
- Wind tunnel scaling
- 3–D non-linear turbulence simulations
- Comprehensive theory based models
- Turbulence measured
- Concensus on ion transport
- Internal transport barriers
- Neoclassical ion transport attained
- E×B shear stabilization

TOKAMAK CONFINEMENT PROVED (EMPIRICALLY) PREDICTABLE

- In the 80's consistent scaling behavior was seen across many tokamaks implying
  - A common underlying transport physics was discoverable
  - Multi-machine confinement scaling relations could be constructed, e.g.

$$\tau_{E, th, ELMy} = 0.85\ \tau_{E, th, ELM\text{-}free}$$
$$= 0.031\ I_p^{1.06}\ B^{0.32}$$
$$P^{-0.67}\ M^{0.41}\ R^{1.79}\ n_e^{0.17}\ \varepsilon^{-0.11}\ \kappa^{-0.6}$$

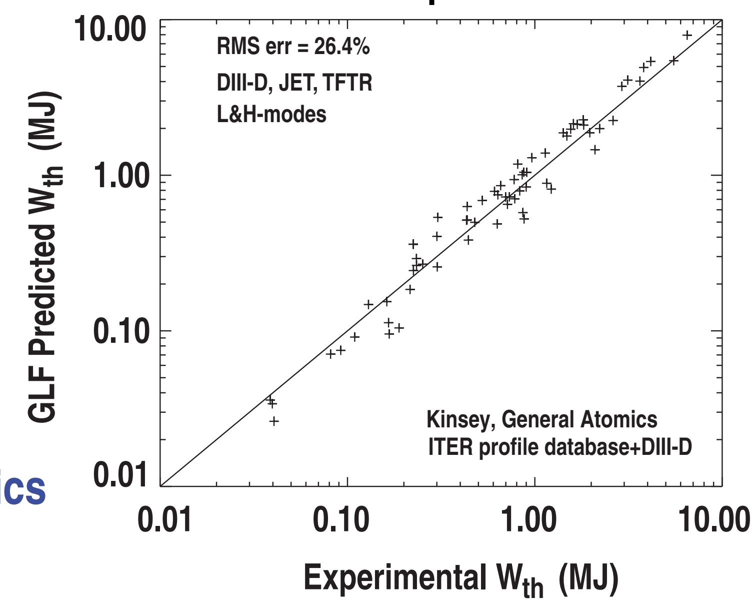
- Dimensionless wind tunnel scaling is providing a more fundamental physics basis



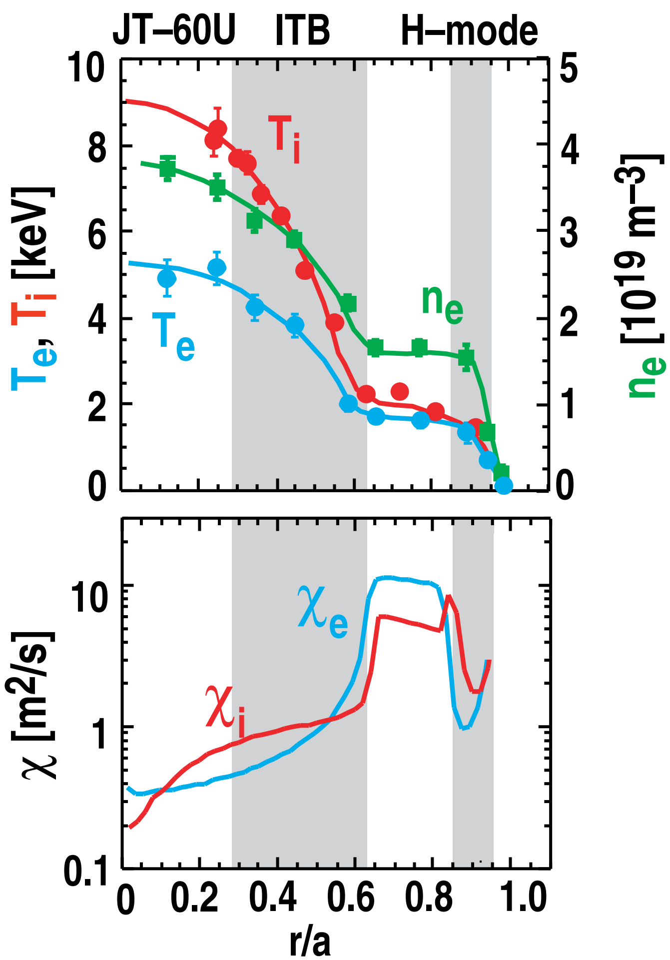
STRATEGY TO CALCULATE TRANSPORT

- Theory-based 3D nonlinear simulations being used to benchmark theoretical transport models which are then compared to experiment
- Linear gyrokinetic codes describe local ballooning mode instabilities
  - Long wavelength — ion temperature gradient (ITG) and trapped electron driven
  - Short wavelength — electron temperature gradient (ETG) driven
- Nonlinear flux tube and approximate gyrofluid codes
  - $\rho_i/a \rightarrow 0$
  - Only local ballooning
- Nonlinear codes spanning several hundred gyroradii
  - Finite ρ_i/a
  - More time consuming
- ITG/trapped electron flux tube simulations have been used to benchmark gyrofluid local transport code models with comprehensive physics
- International profile data base after 1995 allows systematic and comparative statistical tests of transport code models

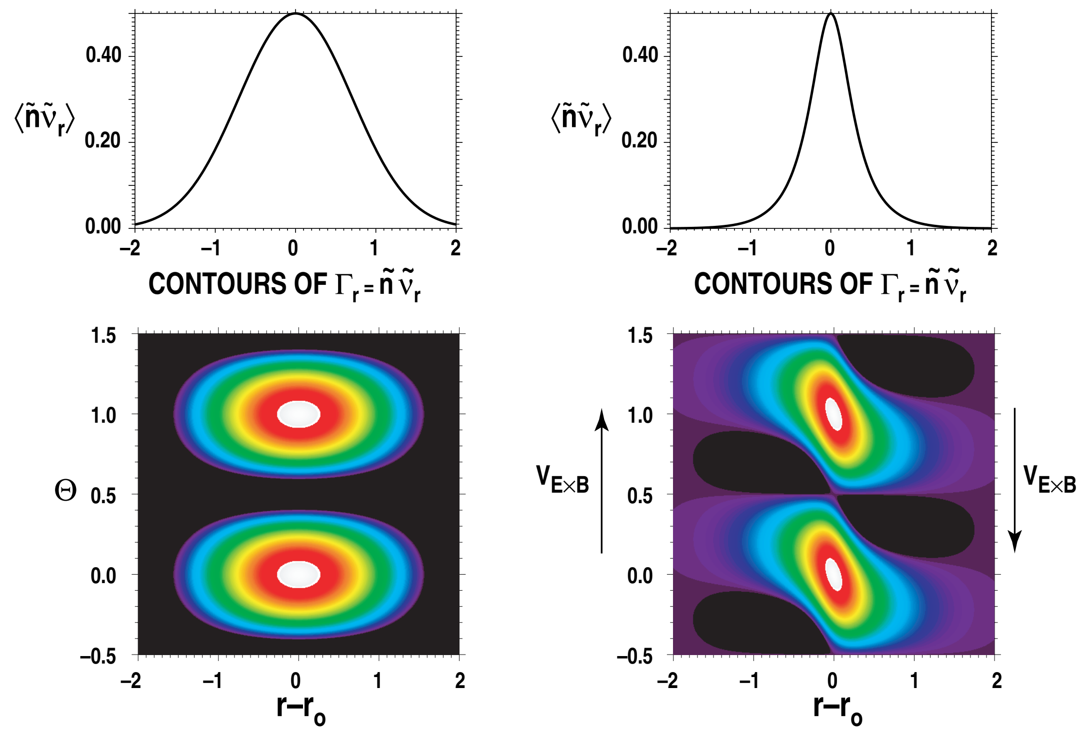
Reasonable Agreement With Experiment



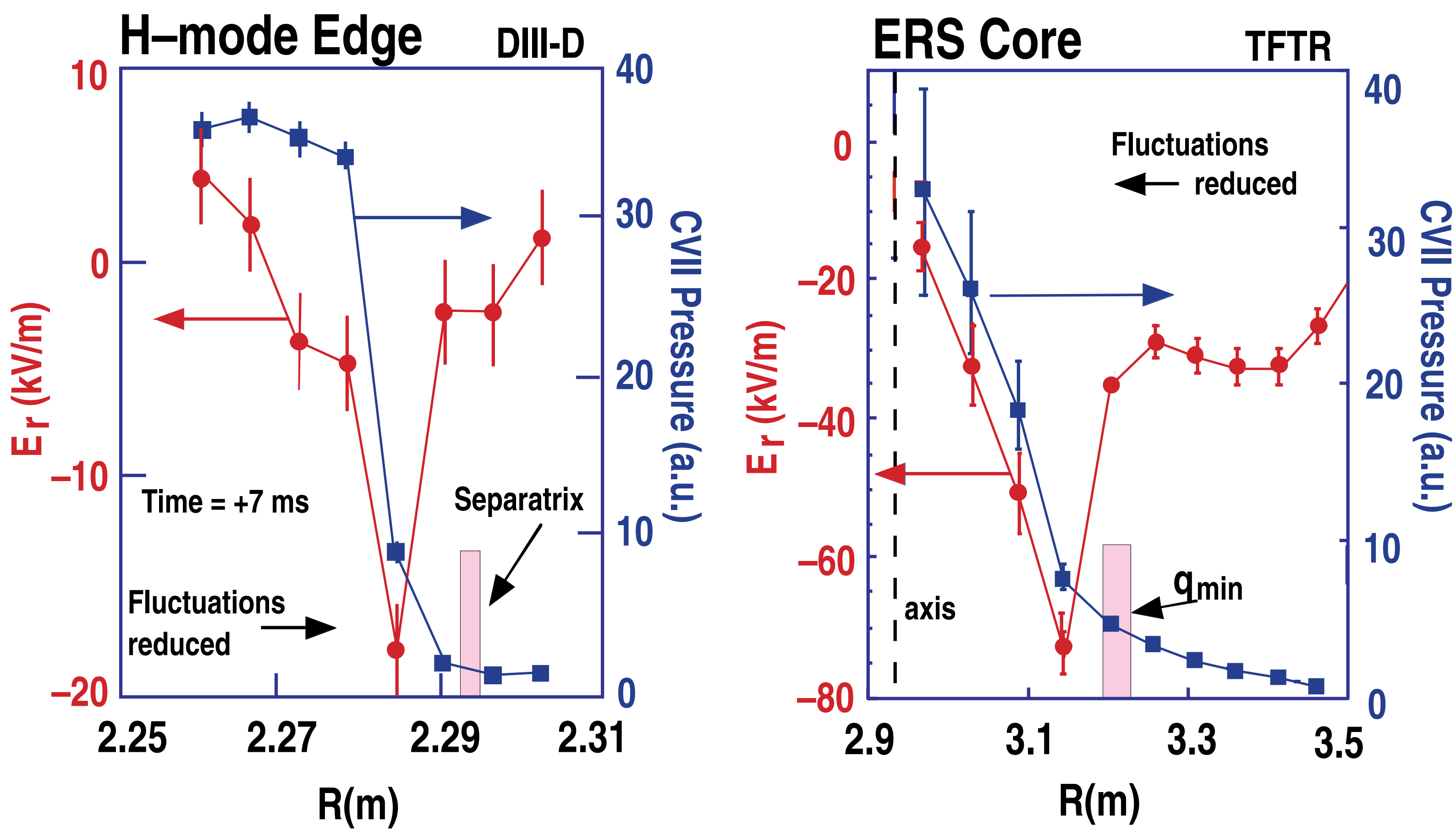
RECENT EXCITEMENT
TRANSPORT BARRIERS FORMED BY SHEARED E×B FLOW



Basic Idea: Sheared E×B flow compresses turbulent eddies in the radial direction



SHEARED E×B FLOW SUPPRESSION OF TURBULENCE UNDERLIES BOTH EDGE AND CORE TRANSPORT BARRIERS

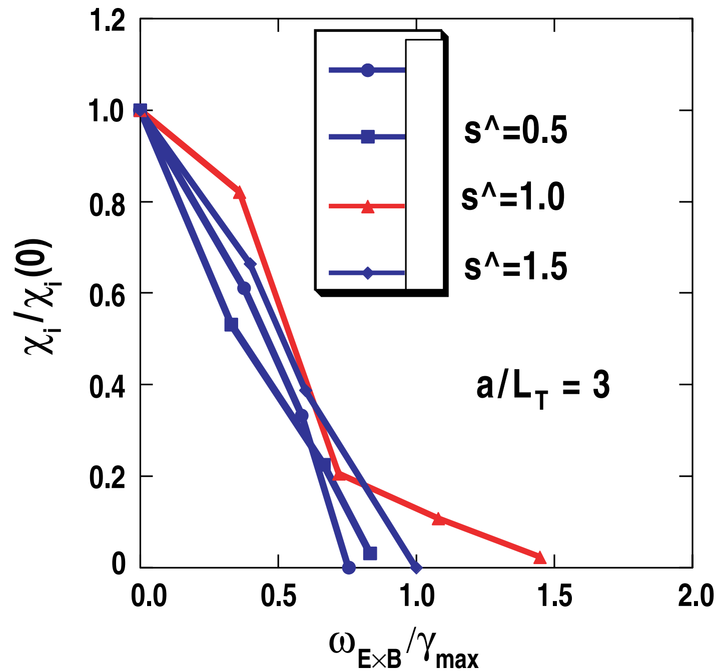


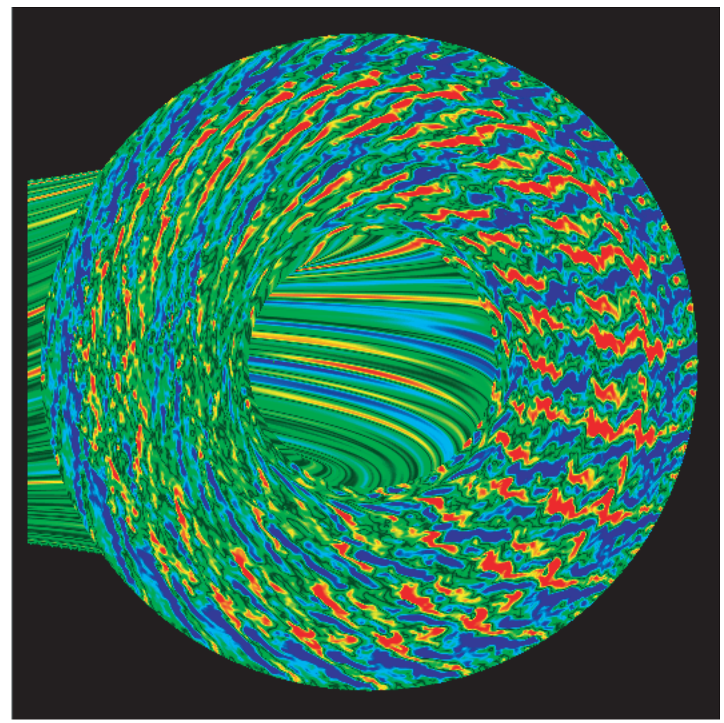
$E_r = (Z_i e n_i)^{-1} \nabla P_i - v_{\theta i} B_\phi + v_{\phi i} B_\theta$, The E×B shearing rate $\omega_{E\times B} = \left| \frac{(RB_\theta)^2}{B} \frac{\delta}{\delta_\psi} \left(\frac{E_r}{RB_\theta} \right) \right|$

[Hahm and Burrell, Phys. Plasmas 2, 1648]

EQUILIBRIUM SCALE SHEARED E×B FLOWS CAN QUENCH ITG TRANSPORT IF THE SHEARING RATE EXCEEDS THE MAXIMUM LINEAR GROWTH RATE OF THE TURBULENCE

- ITG simulation of local annulus 160 ρ_s wide [R.E. Waltz, et al., Phys. Plasmas 1, 2229 (1994)]
- Application of E×B shear $\omega_{E\times B} \sim \gamma_{max}$ breaks up eddies and considerably reduces transport





No E×B flow

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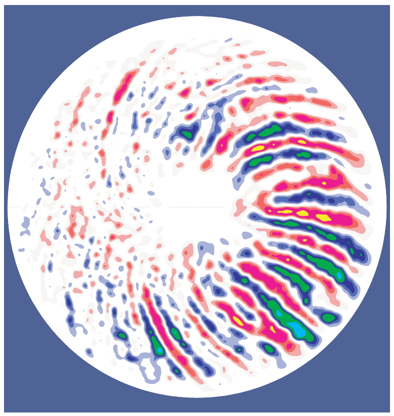
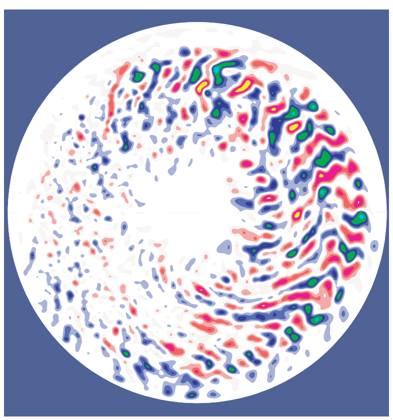
PLASMA TURBULENCE SIMULATION CODES USE FULL TOROIDAL GEOMETRY TO CALCULATE TRANSPORT RATES

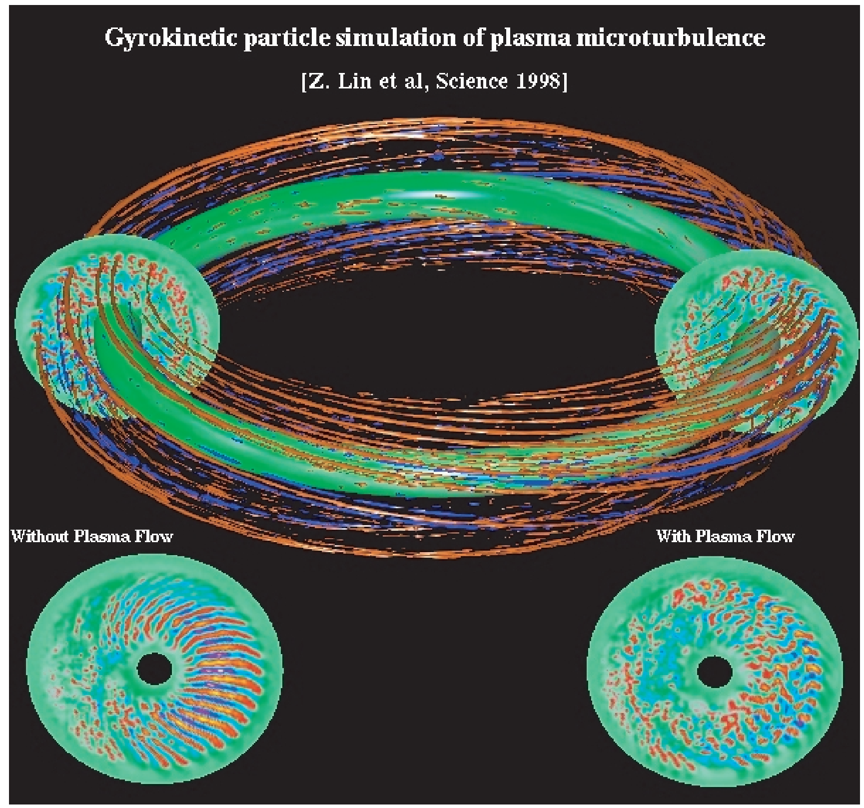
- **Recent advance: Small scale sheared poloidal flows can shear apart radial eddies, reducing their radial step size and the transport by an order of magnitude**

Without sheared flows



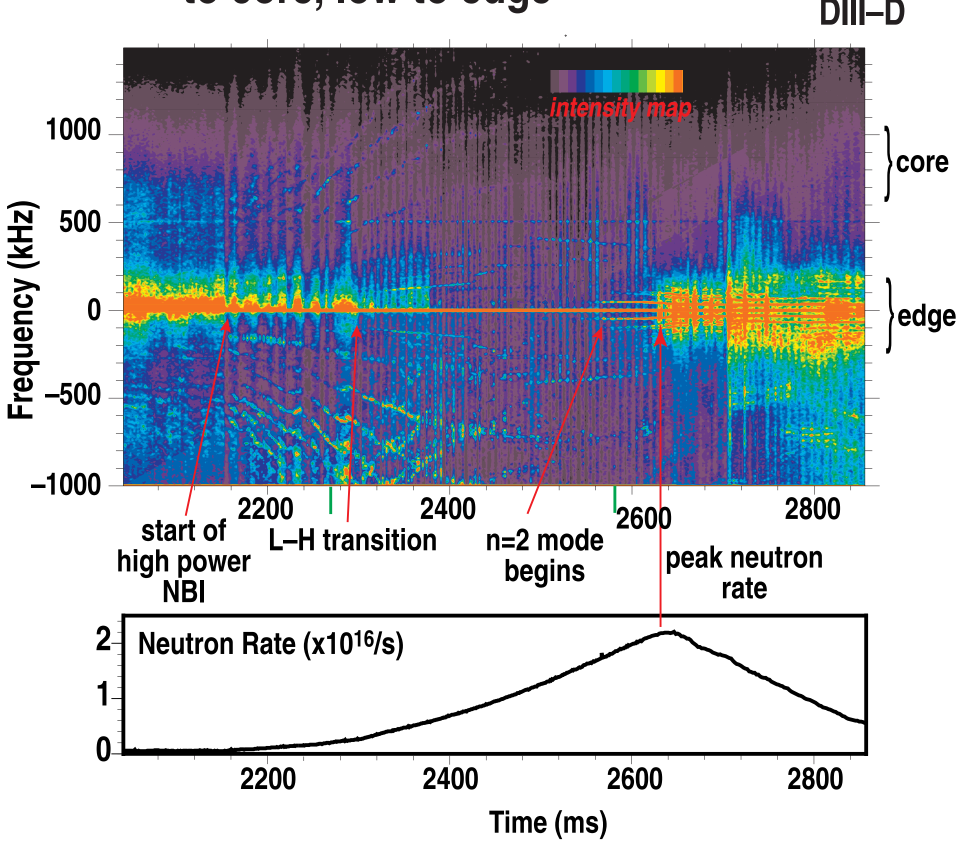
With sheared flows



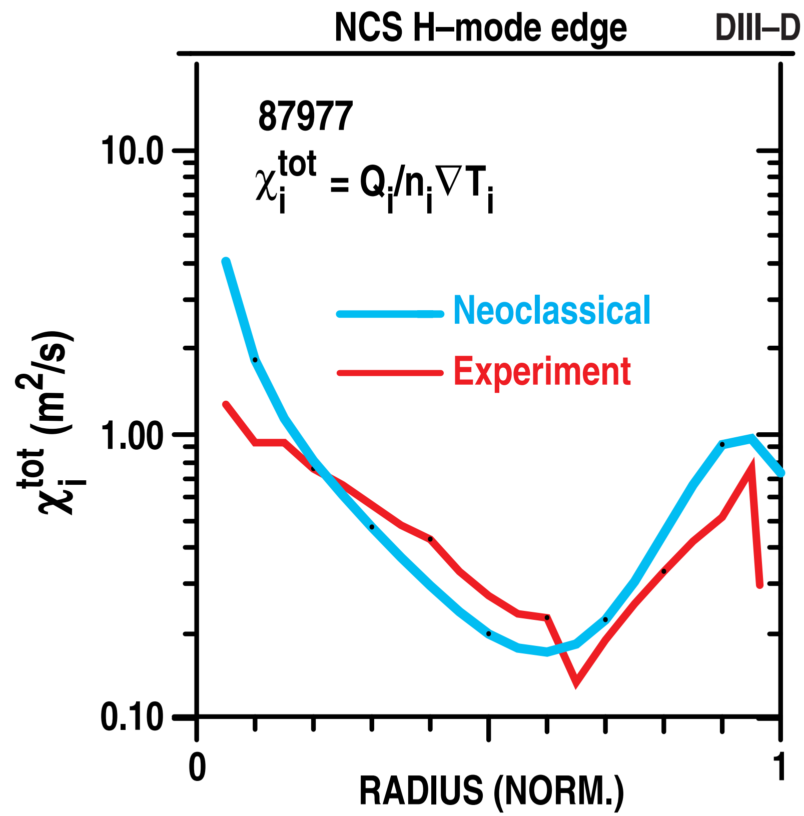


ION-NEOCLASSICAL TRANSPORT WITHOUT TURBULENCE, ACROSS ENTIRE PLASMA RADIUS

- Color contour map of fluctuation intensity as function of time from FIR scattering data
  - Higher frequencies correspond to core, low to edge



- Total ion thermal diffusivity at time of peak performance
  - H = 4.5 W = 4.2 MJ
    β = 6.7% β_N = 4.0



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CONFINEMENT CHALLENGES FOR THE NEXT DECADE

90s

- Wind tunnel scaling
- 3–D non-linear turbulence simulations
- Comprehensive theory based models
- Turbulence measured
- Concensus on ion transport
- Internal transport barriers
- Neoclassical ion transport attained
- E×B shear stabilization

2000 – 2010

- Theory based calculations of transport barrier formation
- Control ITB radius and gradient
- Understand electron transport electron turbulence diagnostics
- First-principles diffusion coefficients
- Momentum and particle transport
- Nonlinear turbulence simulations with both electrons an ions
- Complete dimensionless scaling
- Access conditions for H–mode
- Edge pedestal structure

THE SCIENCE OF POWER AND PARTICLE EXHAUST LEAPED FORWARD IN THE 90's

70s

- Limiters
- Impure, radiating core plasmas

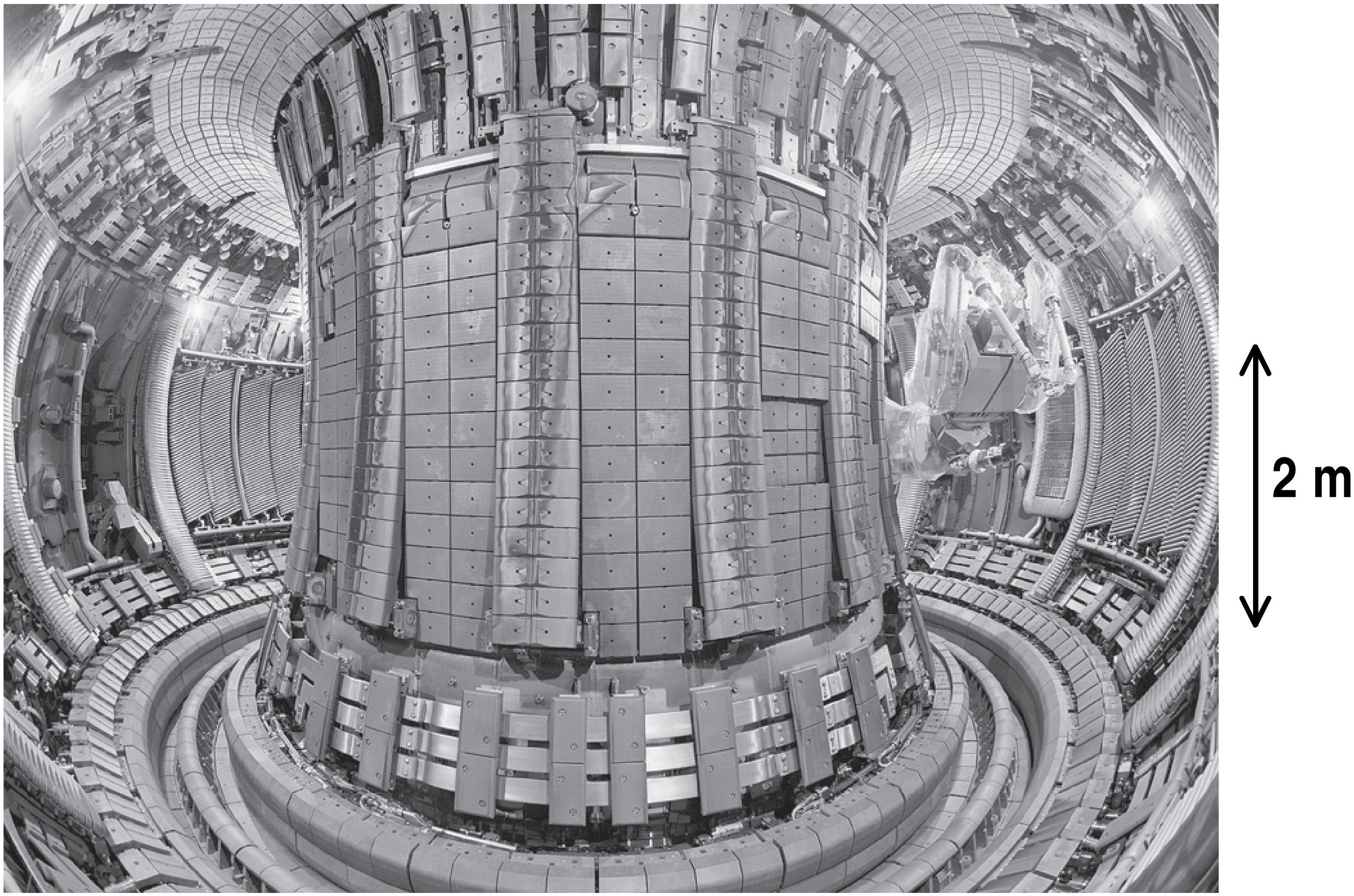
80s

- Divertors
- Clean plasmas
- Low core radiation

90s

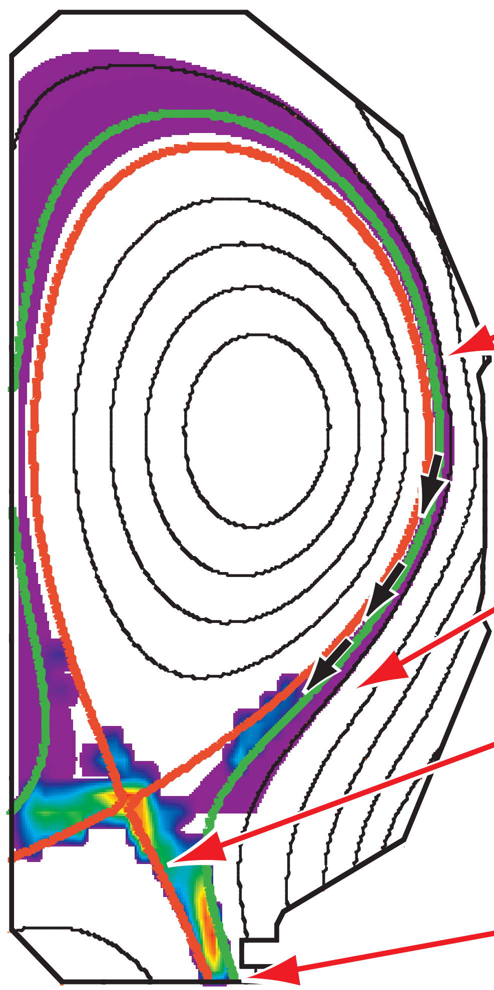
- He ash and fuel exhaust
- Radiative divertor plasmas
- Recombination
- 2–D measurements
- 2–D fluid codes

THE JET DIVERTOR IS TYPICAL OF TOKAMAKS TODAY



Axisymmetric lower single null with graphite tiles to handle high heat flux

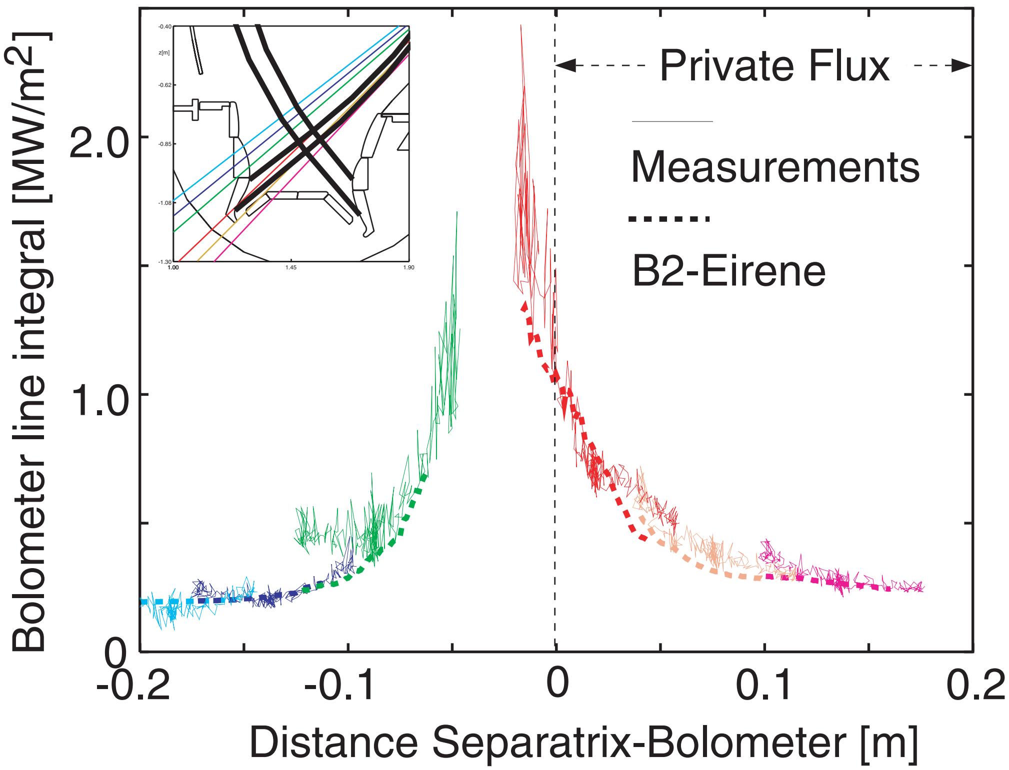
THE PHYSICS ELEMENTS THAT ARE DOMINANT IN THE DIVERTOR PROBLEM ARE NOW INCORPORATED IN 2-D CODES

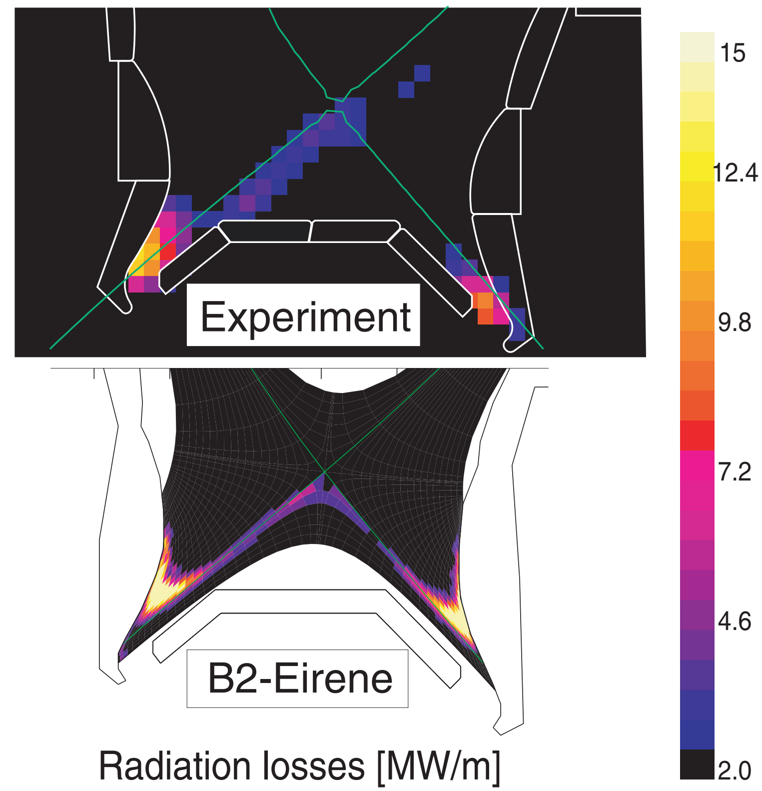


- Strong parallel transport
  Fluid drifts
  Actual flux surface geometry
- Non-equilibrium radiation rates
  2-D flow patterns
- Neutral recycling
  Recombination
  Detailed divertor structures
- Erosion of surfaces
  Ablation during intense heat pulses

AN EXAMPLE OF EXCELLENT AGREEMENT BETWEEN B2-E IRENE CALCULATED AND MEASURED RADIATION DISTRIBUTIONS

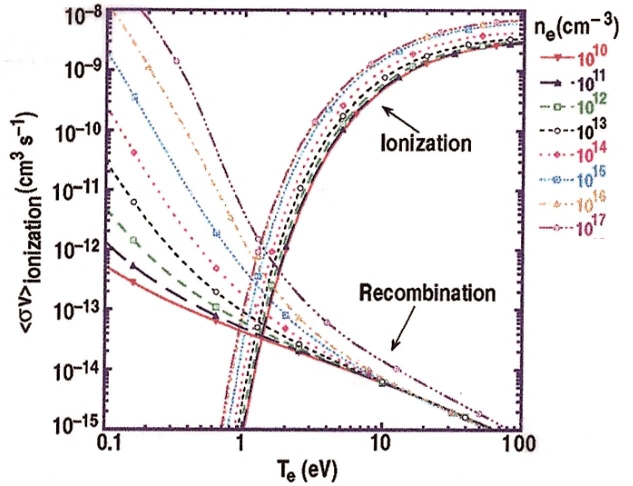
ASDEX–UPGRADE





Excellent agreement

RECOMBINING DIVERTOR PLASMAS DISCOVERED

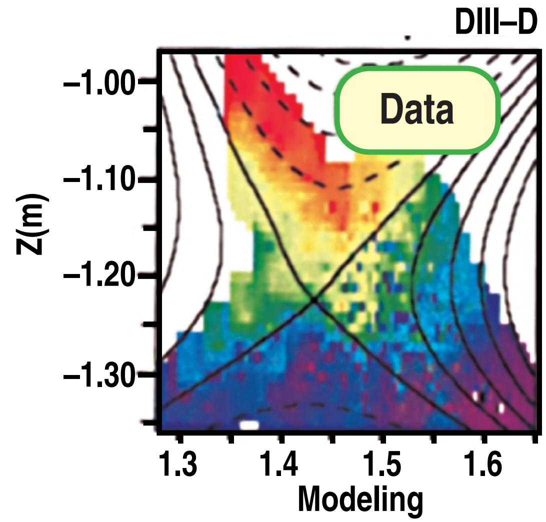


Scaling of Lyman Series Line Intensities Shows When the Upper Levels of the Lines Are Populated by **Recombination**

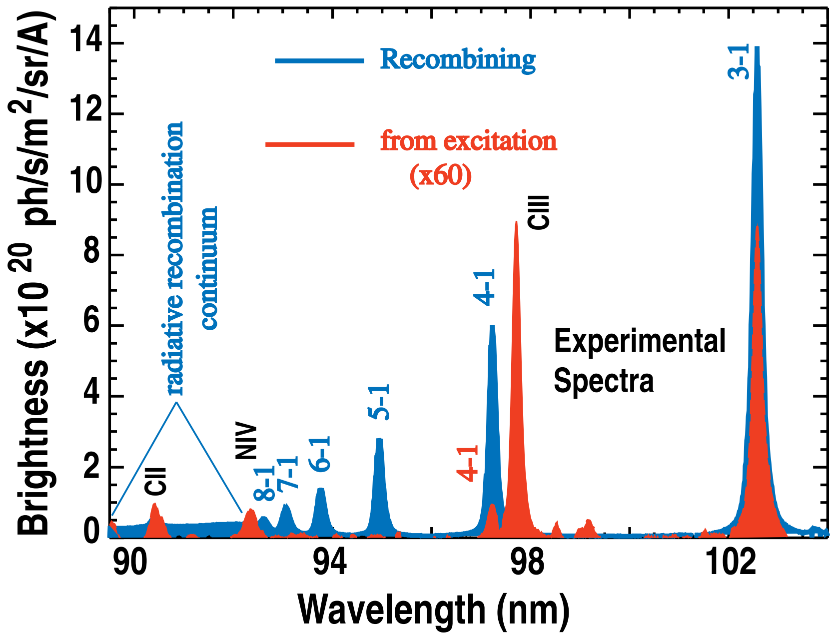
Alcator C–Mod

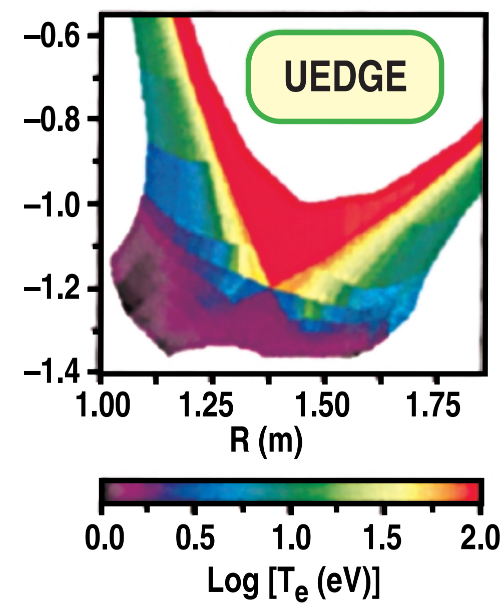
- T_e ~ 1 eV at divertor plate (probes)
- T_e 0.4-0.6 eV in divertor plasma (spect.)

Low Electron Temperature



Alcator C–Mod





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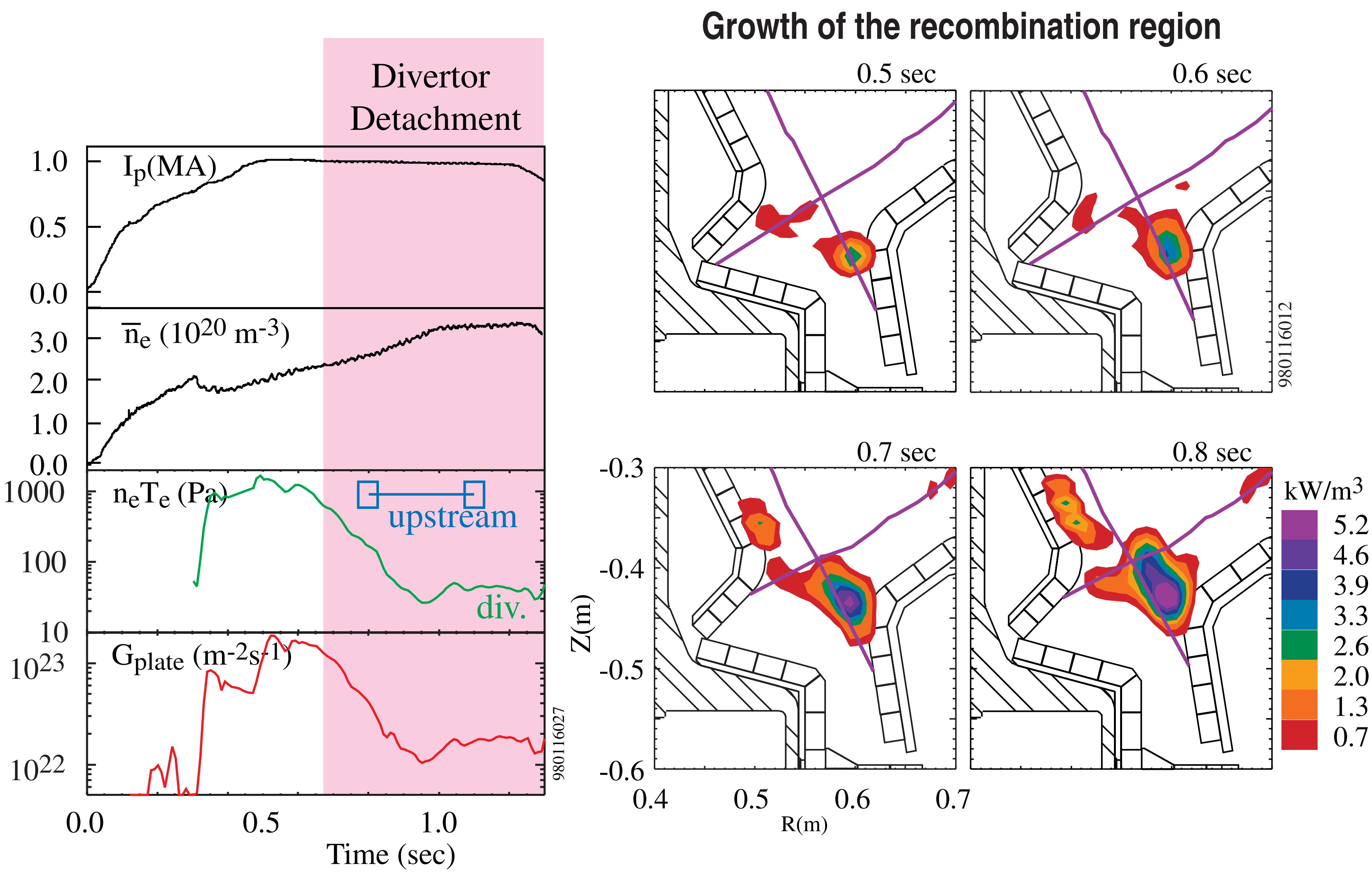
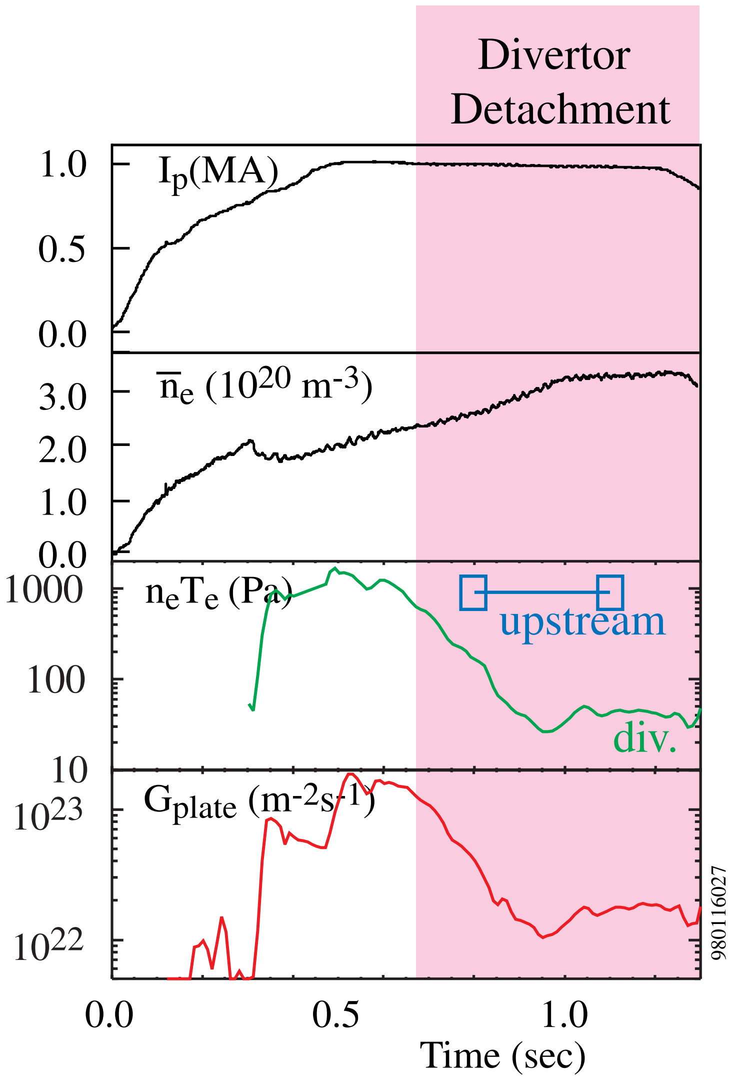
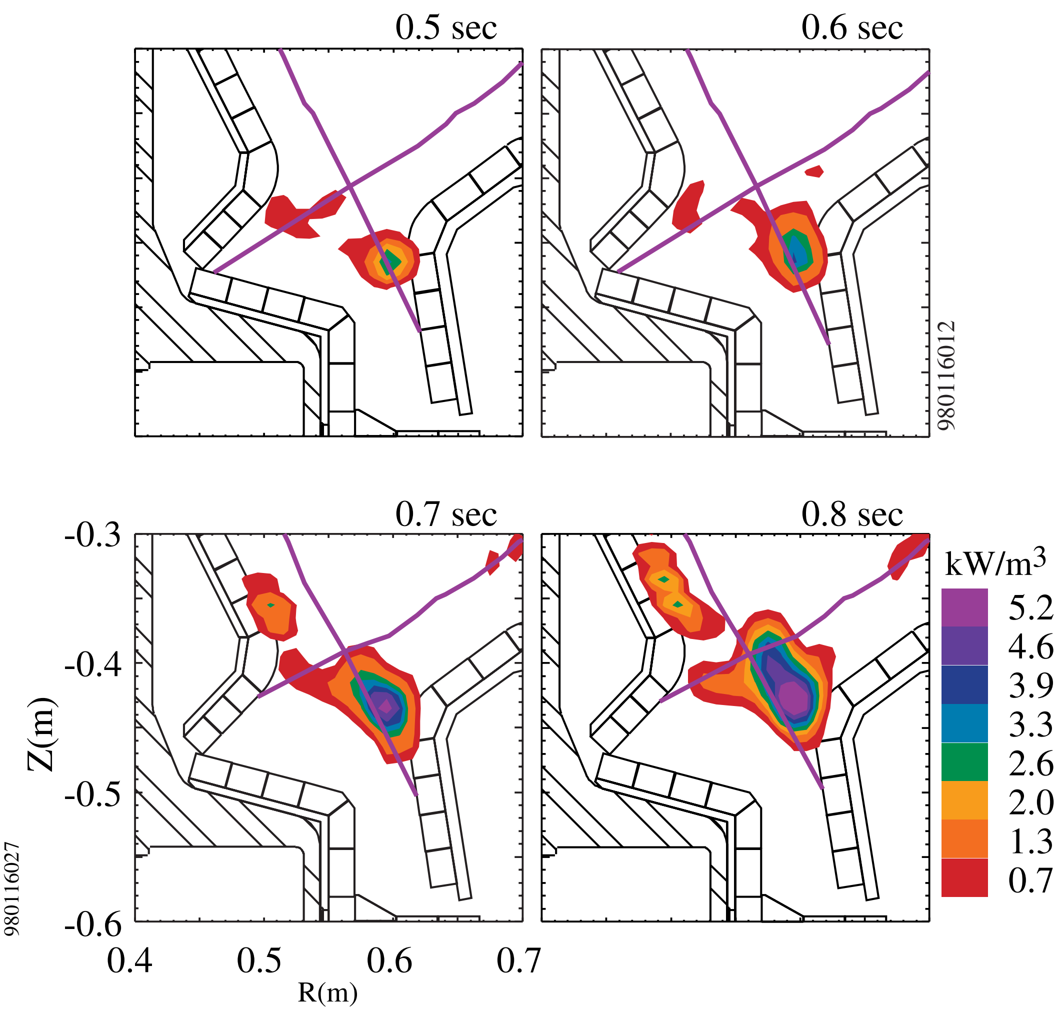
DIVERTOR DETACHMENT IN ALCATOR C-MOD



DIVERTOR DETACHMENT IN ALCATOR C-MOD



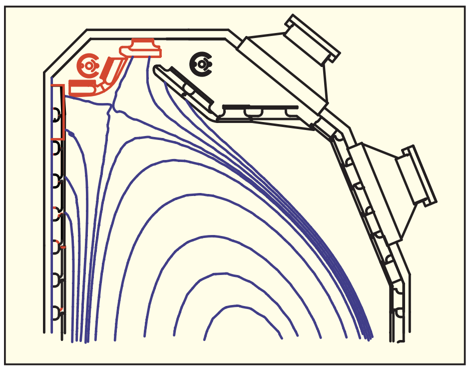
Growth of the recombination region

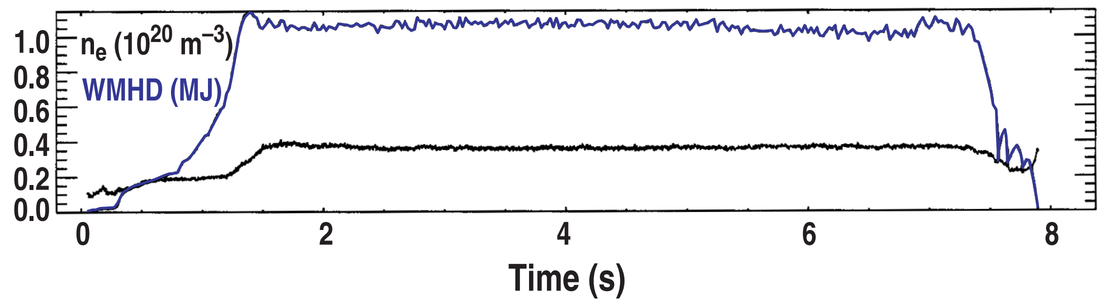


EXHAUST OF FUEL AND HELIUM ASH DEMONSTRATED

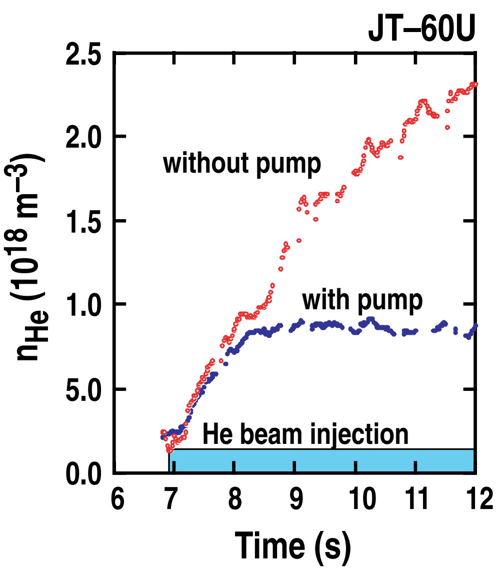
- Plasma density regulated constant by gas fueling and divertor pumping

DIII–D Divertor 2000

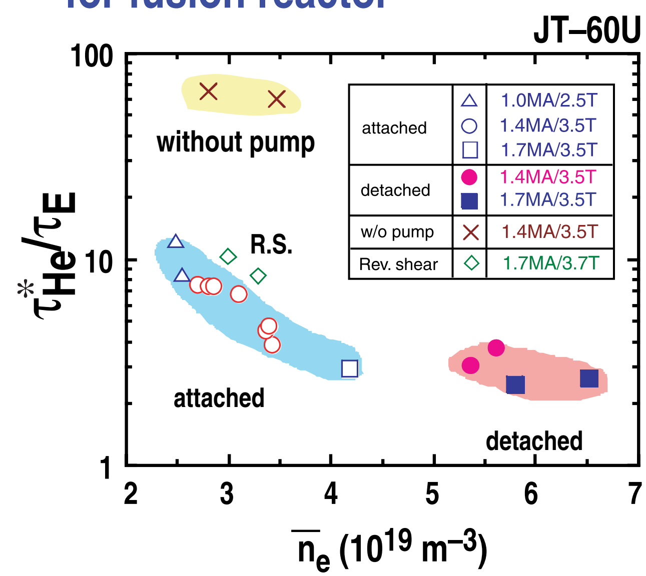




- Divertor pumping prevents accumulation of helium ash (injected by neutral beams)



- Pumpout rate of helium adequate for fusion reactor



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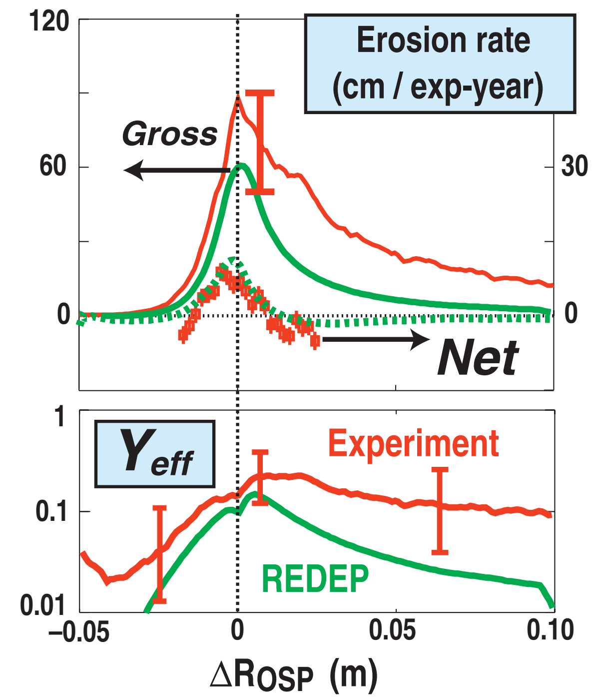
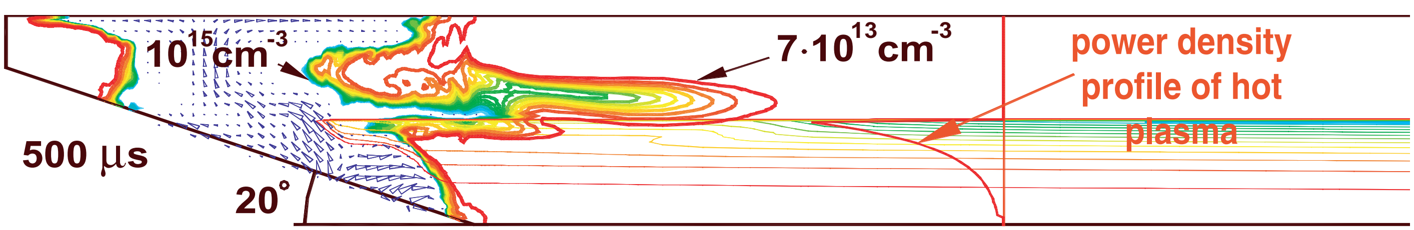
CODES TO CALCULATE THE EROSION OF DIVERTOR SURFACES ARE BEING TESTED AGAINST EXPERIMENTAL DATA

- **Erosion during normal operation**
  - REDEP matches DIII–D data for carbon
  - Treats physical and chemical sputtering and 2-D material transport
  - Codes: REDEP, WBC, ERO, DIVIMP, MCI, IMPMC
- **Erosion during ablative heat pulses**
  - 2-D codes treat vaporization melting vapor shield formation, radiation transport
  - Tested against plasma gas experiments
  - Codes: WURZ, LANGYEL, HASSANEIN

OSP Erosion and Modeling (DIII–D)



Disruption Erosion in the Divertor



Calculation by Wuerz for 1.5 m long divertor slot (ITER ~ 1 m)

POWER AND PARTICLE EXHAUST CHALLENGES FOR THE NEXT DECADE

90s	2000 – 2010
Helium ash and fuel exhaust	Optimal plasma edge shape
Radiative divertor plasmas	2–D SOL/divertor flows
Recombination	Helium and fuel exhaust in AT regimes
2–D measurements	Use of copious core radiation
2–D fluid codes	Understanding erosion and redeposition (T inventory)
	Modeling and mitigating disruption erosion

SCIENTIFIC BASIS — DEEP, EXTENSIVE, FULL OF PROMISE

Area	Status	Advanced Tokamak Challenge	Promise
Heating	Understood, technology developed	Pressure profile control, alpha heating	Burning plasmas
Current drive	Physics understood	High bootstrap fraction, local profile control	Steady-state bootstrap fraction $\rightarrow$ 100%
Stability	Operating space understood, predictable	Wall stabilization	Double the stable operating space
Confinement	Closing in on ability to calculate	Transport barrier control	Near neoclassical ion confinement
Power and particle control	Major physics elements calculable	Low density divertors compatible with current drive	Steady-state with low surface erosion

WE ARE READY TO TAKE UP BURNING PLASMA AND STEADY-STATE ISSUES

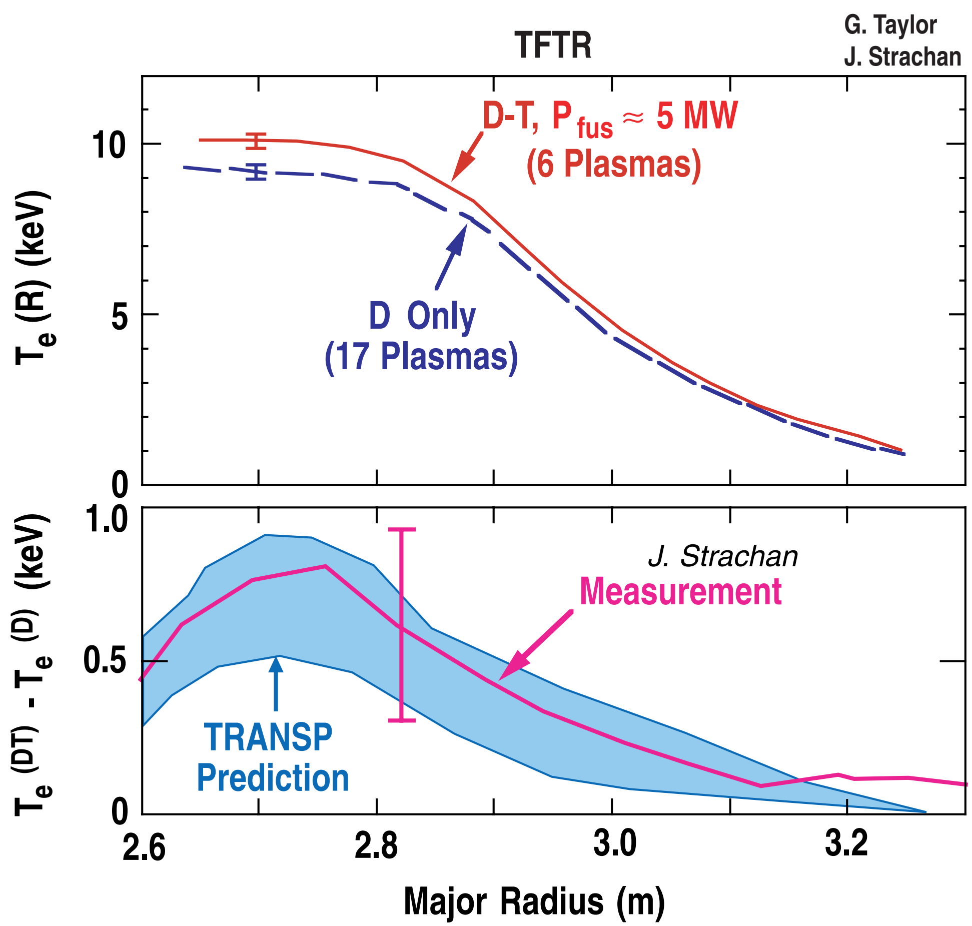
Burning Plasma Issues (More Gain ↓)

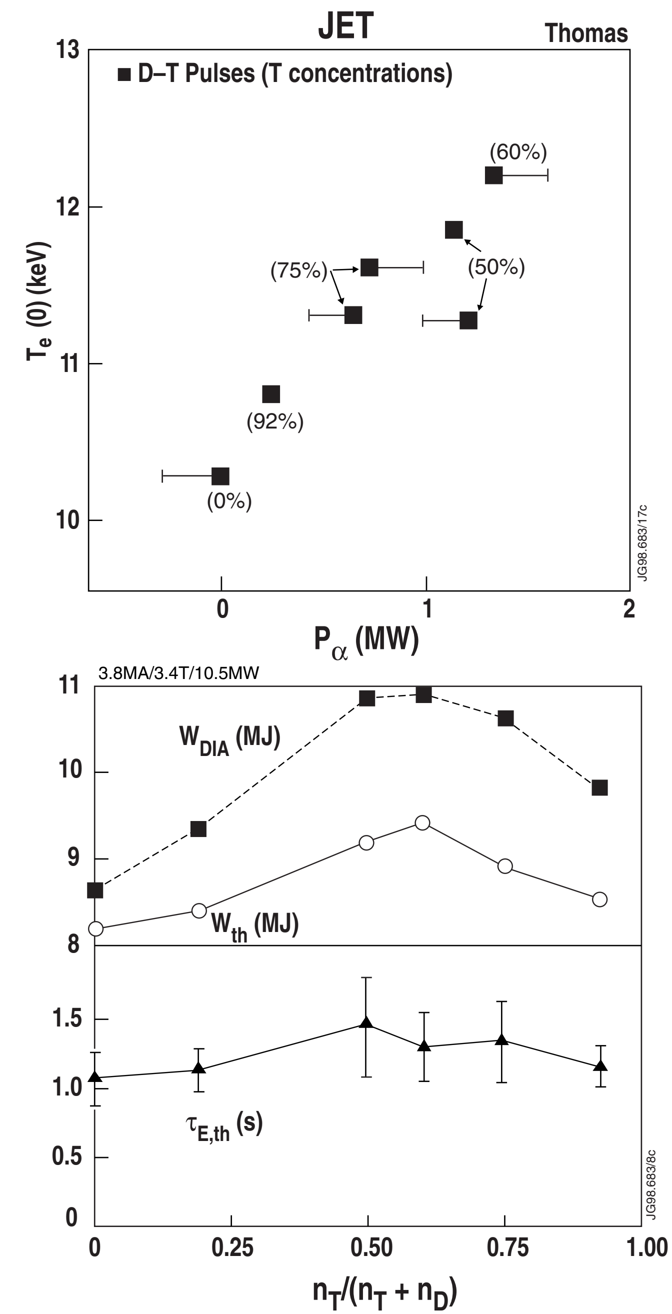
- DT plasma properties
- Alpha confinement
- Alpha ash exhaust
- Remote maintenance
- Alpha driven instabilities
- Self-heated profiles
- High gain burn control

Steady State Issues (Fluence)

- High bootstrap fractions (AT)
- Steady-state magnets
- Steady-state current drive
- Tritium inventory
- Hour long pulses
- Resolve disruption issue
- Blanket development
- Low activation materials
- Tritium breeding
- Month long operation
- First electric output

ALPHA HEATING OBSERVED





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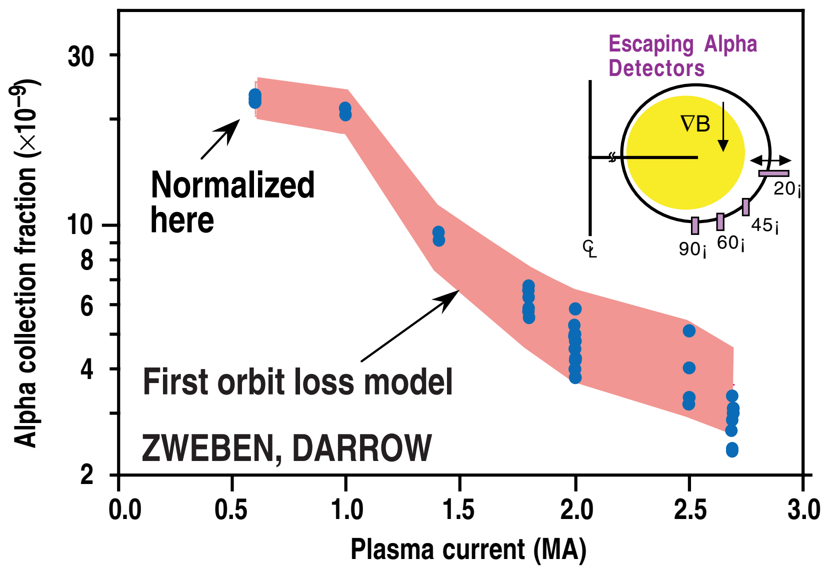
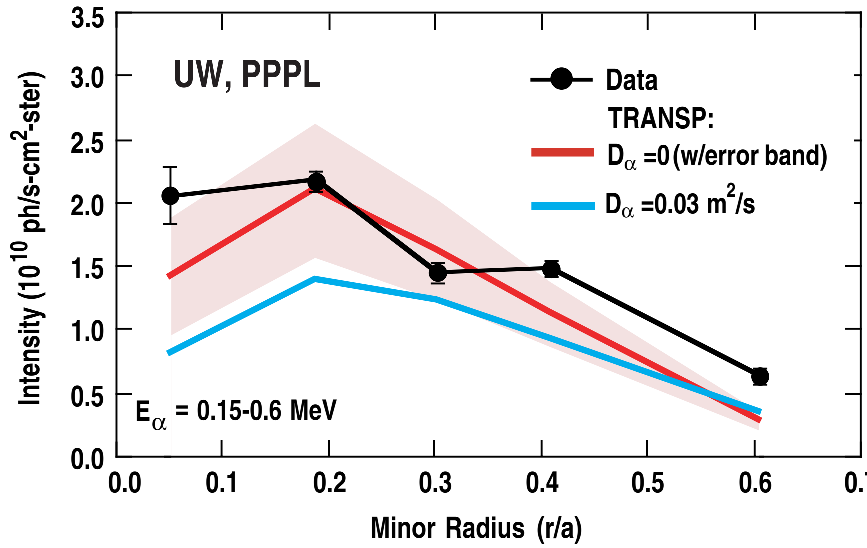
CLASSICAL ALPHA CONFINEMENT VERIFIED (TFTR)

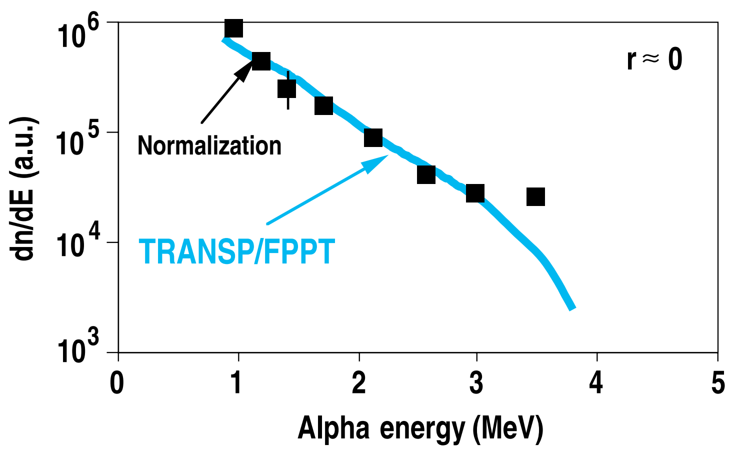
First orbit loss (3% at 2.5 MA)



Radial transport

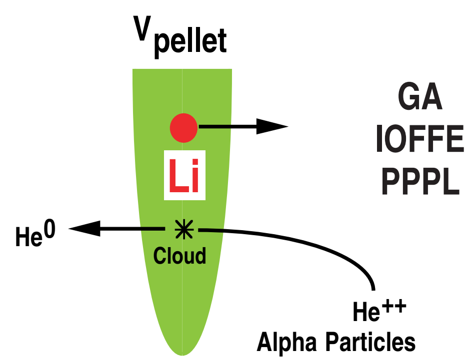


Slowing down spectrum



Double Charge Exchange Technique

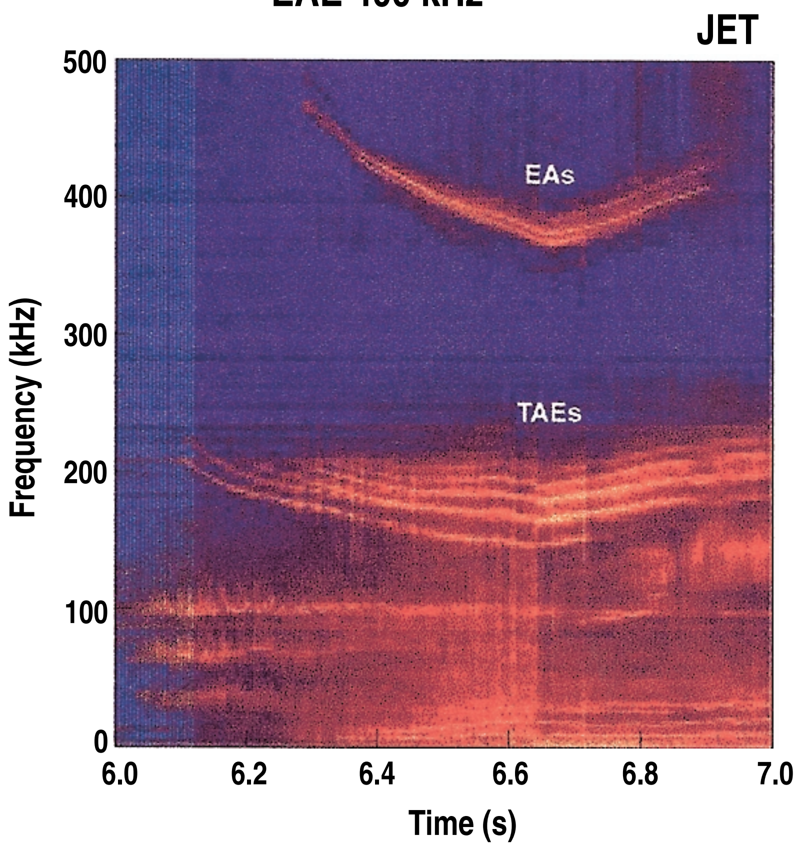
$He^{++} + Li^{+} \Rightarrow He^{0} + Li^{3+}$



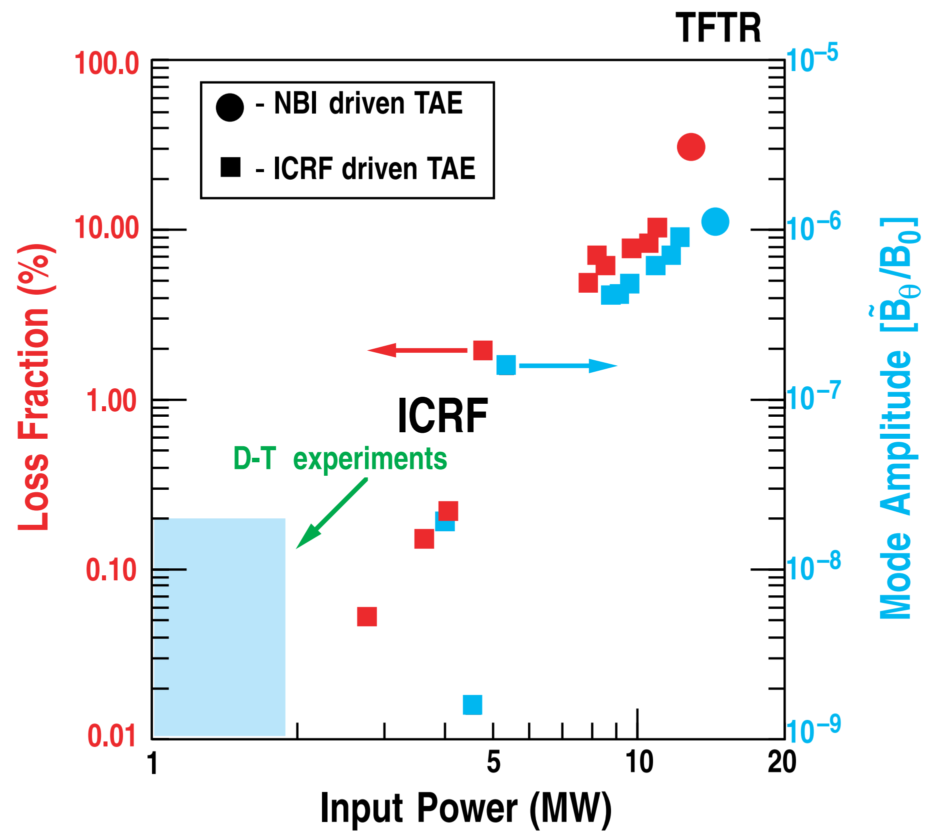
THEORETICALLY PREDICTED ALFVÉN EIGENMODES WERE OBSERVED

AE Modes excited in JET by ICRH minority ions

TAE 200 kHz
EAE 400 kHz

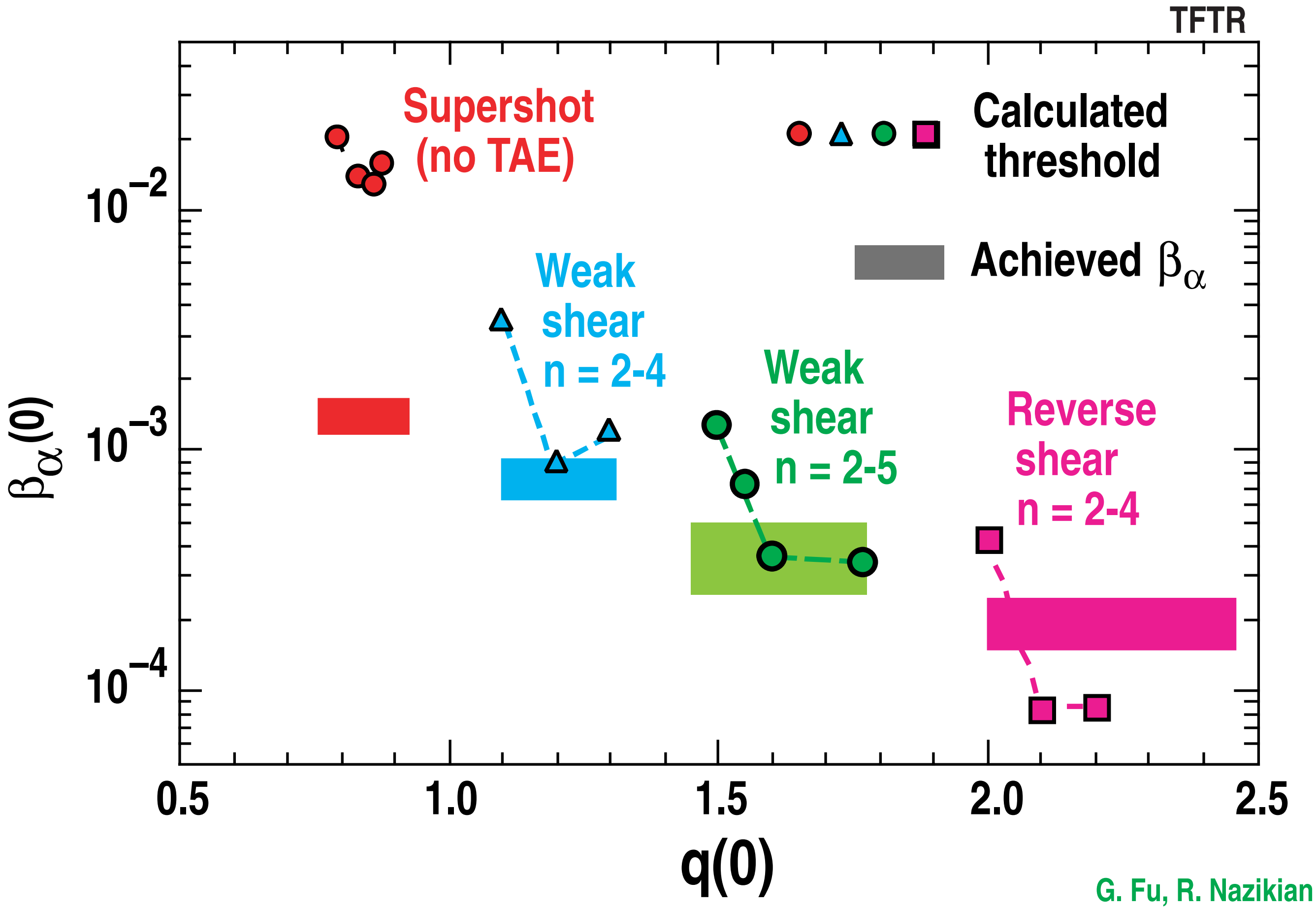


Substantial fast ion losses in TFTR from TAE modes driven by neutral beam or ICRF tail ions



- AE Modes absent in highest fusion power cases

OBSERVED α-DRIVEN TAES CONSISTENT WITH FULL LINEAR THEORY

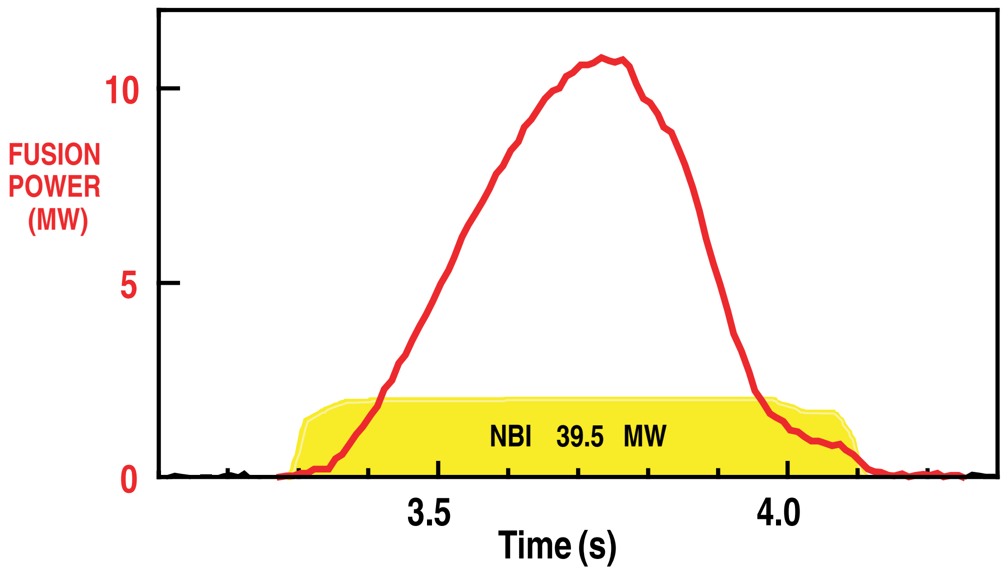


- Calculations with NOVA-K code
- Weak shear and high q(0) are destabilizing
- Weak or reverse shear plasmas in a reactor may be unstable to high-n TAEs

COPIOUS FUSION POWER HAS BEEN PRODUCED

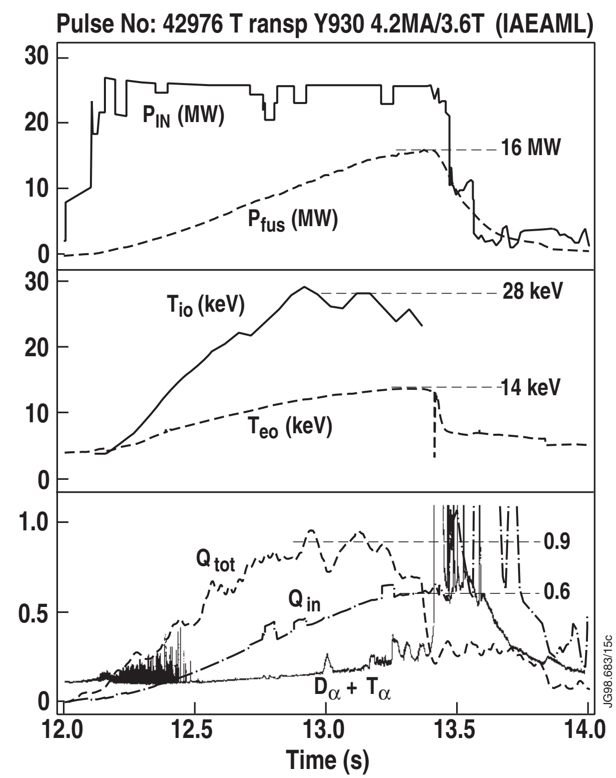
TFTR D-T Campaign

- 10.7 MW
- $P_{FUSION}/P_{HEAT} = 0.27$
- 1.55 GJ fusion energy



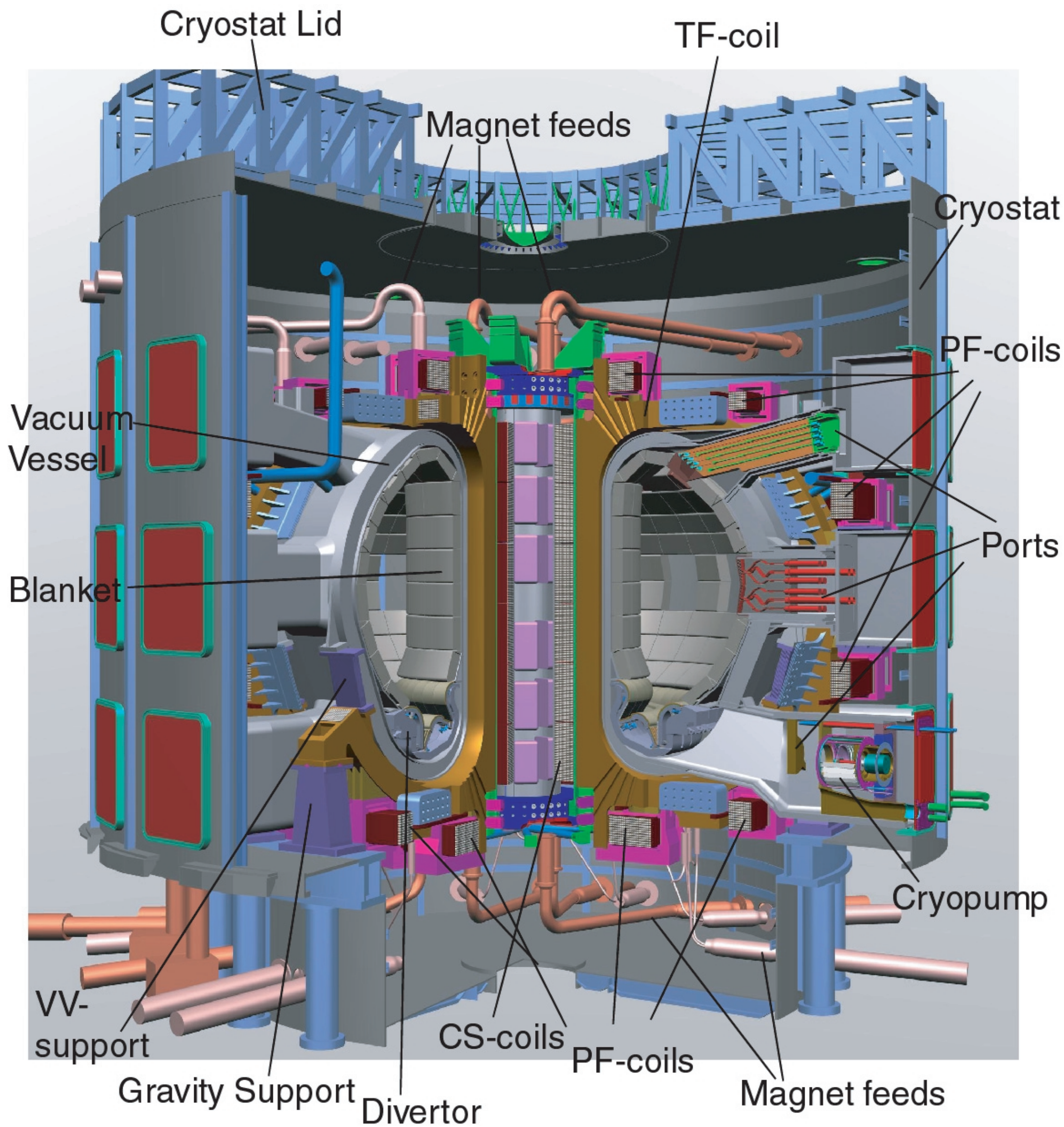
JET D-T Campaign

- 16 MW
- $P_{FUSION}/P_{HEAT} = 0.6$
- 0.68 GJ fusion energy



THE ITER-FEAT MACHINE

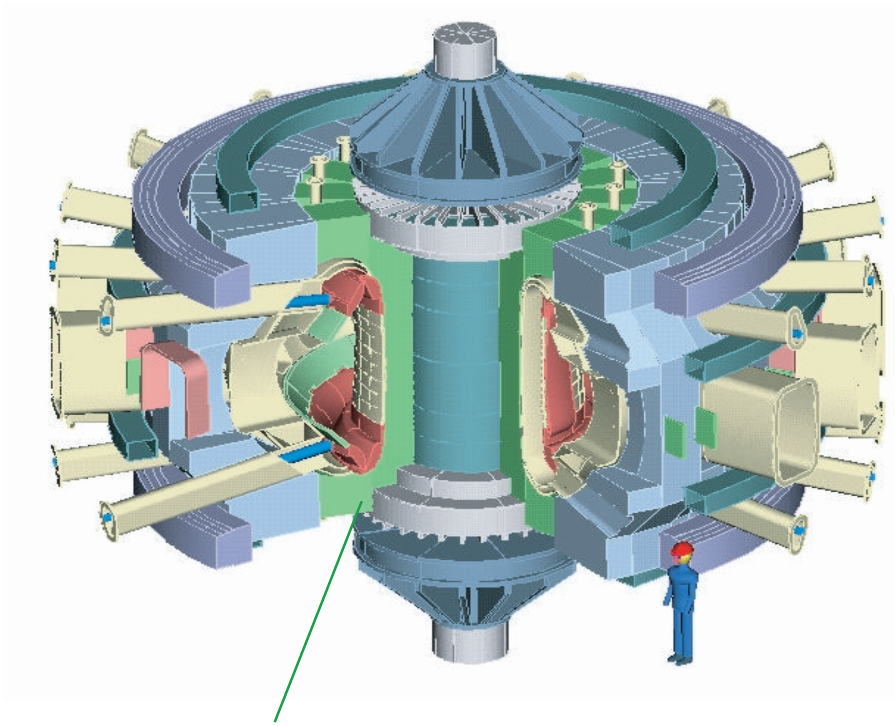
- **Cut through cryostat, TF and PF coils, Vacuum Vessel, Blanket and Divertor**



Fusion Ignition Research Experiment

(FIRE)

http: / / fire. pppl. gov



Design Goals

- $R = 2.0$ m, $a = 0.525$ m
- $B = 10$ T, (12T) *
- $W_{mag} = 3.8$ GJ, (5.5T) *
- $I_p = 6.5$ MA, (7.7 MA) *
- $P_{alpha} > P_{aux}$, $P_{fusion} < 200$ MW
- Burn Time ≈ 18.5 s (≈ 12 s) *
- Tokamak Cost ≤ $ 0.3B
  Base Project Cost ≤ $ 1B

* Higher Field Mode

Attain, explore, understand and optimize fusion- dominated plasmas that will provide knowledge for attractive MFE systems

DIII-D NATIONAL FUSION FACILITY SAN DIEGO

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A Proposal of JT-60 Modification

JAERI

¥ To conduct researches on steady state operation of tokamaks

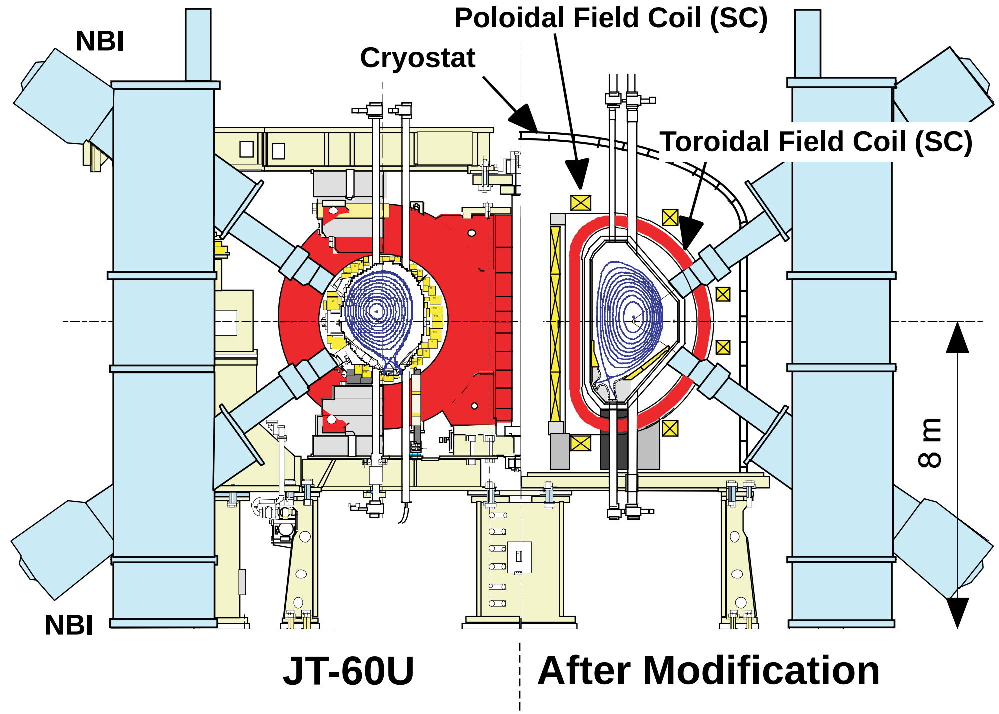
¥ To contribute to the ITER operation

¥ Under discussions at the Fusion Council

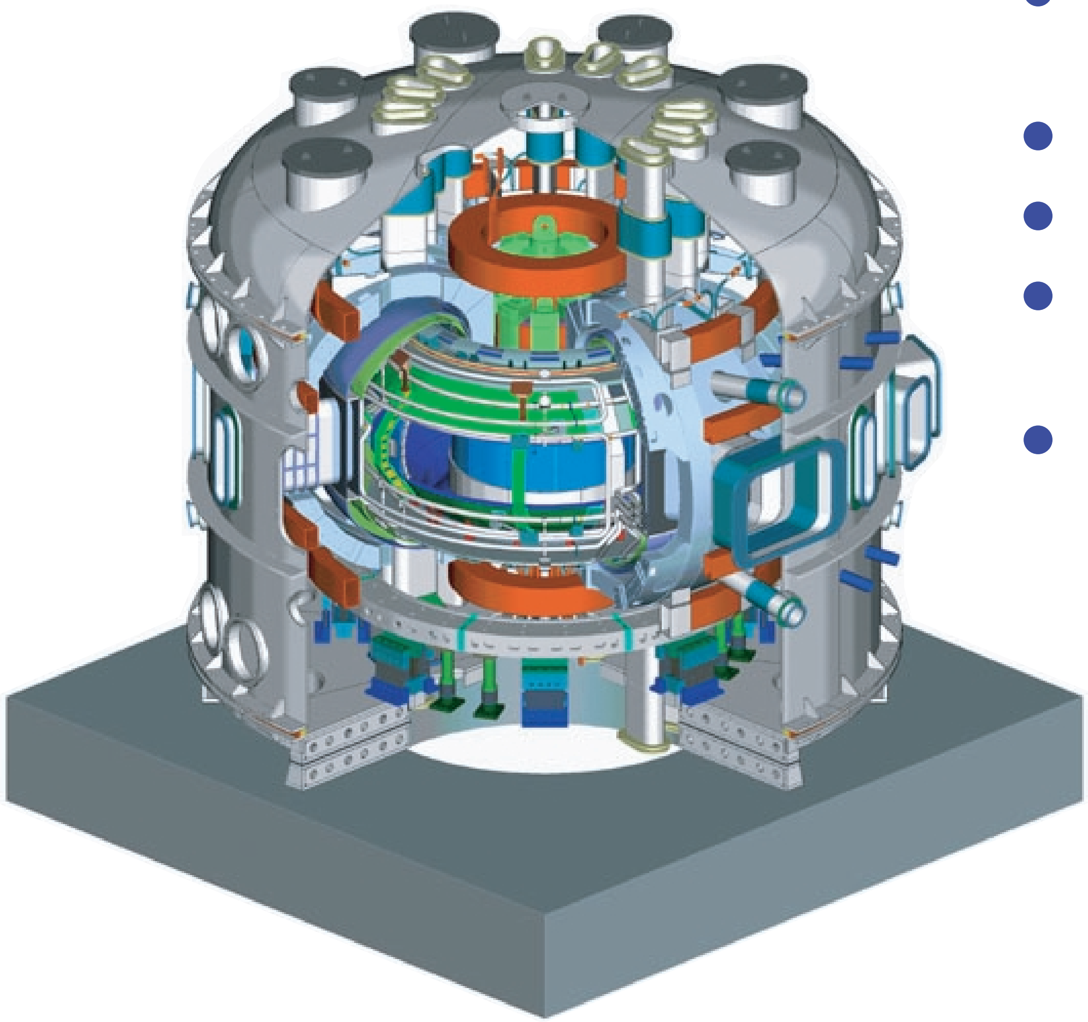
Main Parameter

Parameter	JT-60U	JT-60 (After Modification)	Compact ITER Pulse	Compact ITER Steady-state
Pulse Length	15 s	100 s	400 s	Steady
Maximum Input Power	40 MW (10 s)	40 MW (10 s) ≥10MW (100 s)	73 MW	73MW
Plasma Current I_p	3-5 MA	4 MA	15 MA	7.8 MA
Toroidal Field B_t	4 T (at 3.4 m)	3.8 T (at 2.8 m)	5.3 T	4.98 T
Major Radius R_p	3.4 m	2.8 -3 m (2.8 m*)	6.2 m	6.6 m
Minor Radius a_p	0.9 m	0.7-0.9 m (0.85 m*)	2.0 m	1.6 m
Elongation κ_{95}	1.8 (δ_{95}=0.06)	≤1.9 (1.7*)	1.7	2.0
Triangularity δ_{95}	0.4 (κ_{95}=1.33)	≤0.45 (0.35*)	0.35	0.35
Working Gas	DD	DD	DT	DT

* Nominal Design Value



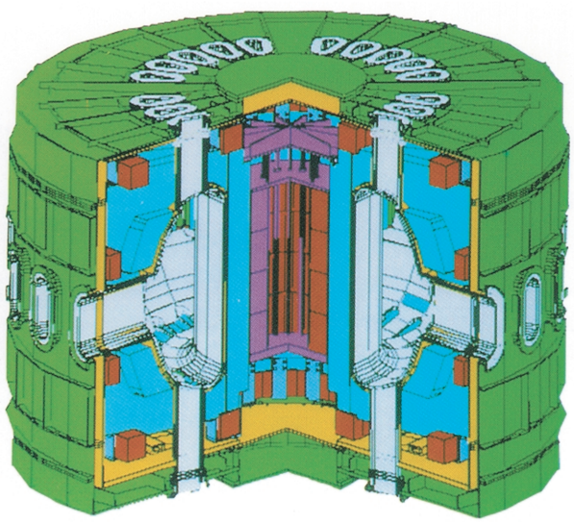
EXTENDING THE ADVANCED TOKAMAK: KSTAR



- 20–300 s pulse length (S/C technology)
- B = 3.5 T, I = 2 MA
- R = 1.8 m, a = 0.5 m
- Double-null divertor, $\kappa = 2$, $\delta = 0.8$
- 16-27 MW profile control: (neutral beam, ion cyclotron, lower hybrid)

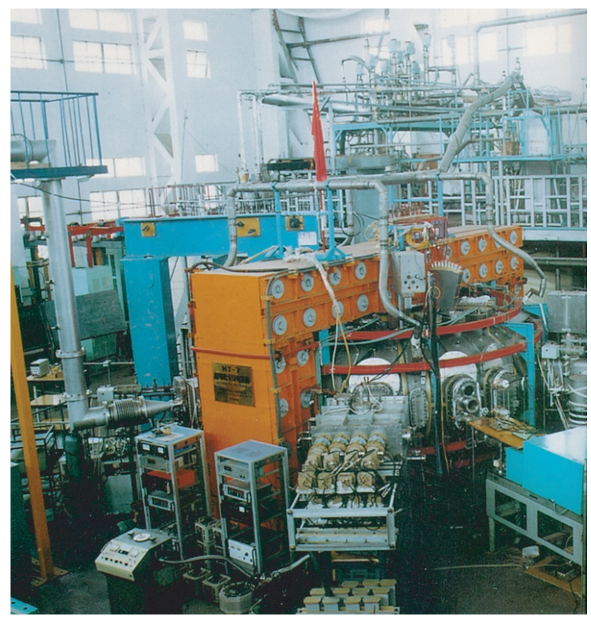
HT–7U ADVANCED TOKAMAK – HAFEI CHINA
INSTITUTE OF PLASMA PHYSICS ACADEMIA SINICA

HT–7U



ASIPP

HT–7

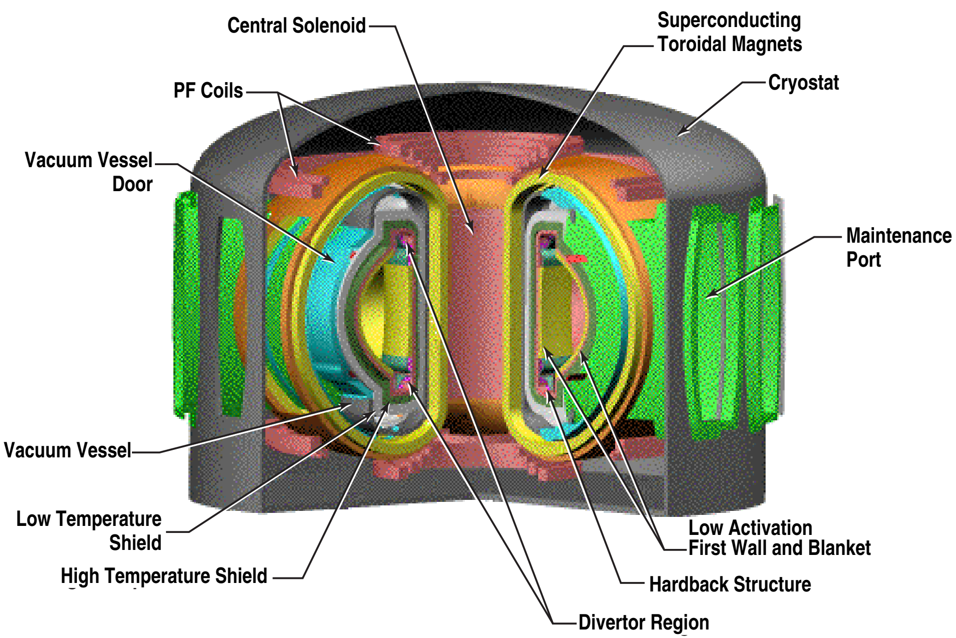


Construction: Approved
Conpletion: mid 2003

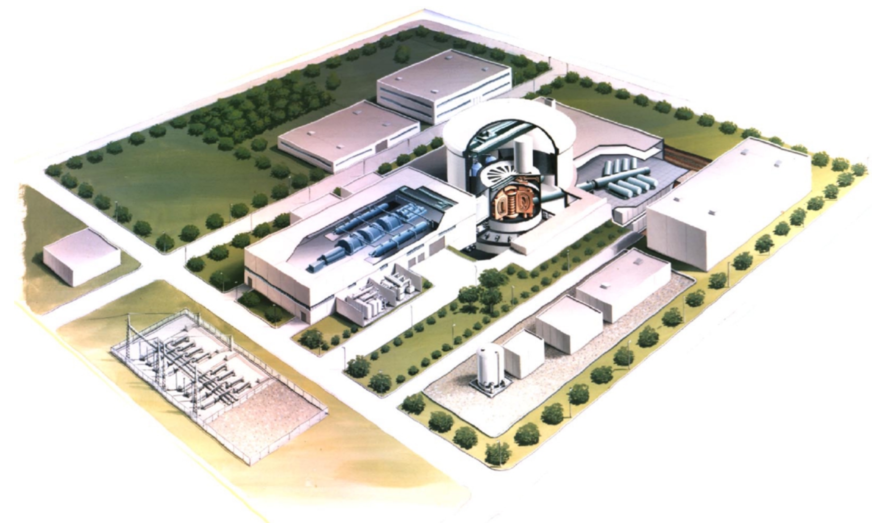
R/a = 1.7/0.4 m
B = 3.5 T
I = 1 MA
κ = 1.6–2.0
δ = 0.4–0.8

THE ADVANCED TOKAMAK LEADS TO AN ATTRACTIVE FUSION POWER PLANT

- **The U.S. ARIES — RS system study**



- **The Japanese SSTR system study**



- **Attractive features**
  - Competitive cost-of-electricity
  - Steady-state operation
  - Maintainability
  - Low-level waste
  - Public and worker safety

	Conventional	AT
Size, major radius (m)	8	5
COE ¢/kWhr	~13	~7
Power cycle	Pulsed	Steady state

SUMMARY

- Research in the tokamak has greatly advanced fusion energy science
- Tokamak research has shown fusion energy is feasible in the laboratory
- The tokamak is scientifically and technically ready to proceed to burning plasma and/or steady-state next steps
- Advanced Tokamak research seeks to find the ultimate potential of the tokamak as a magnetic confinement configuration
  - Anticipated results point to practical and attractive fusion energy