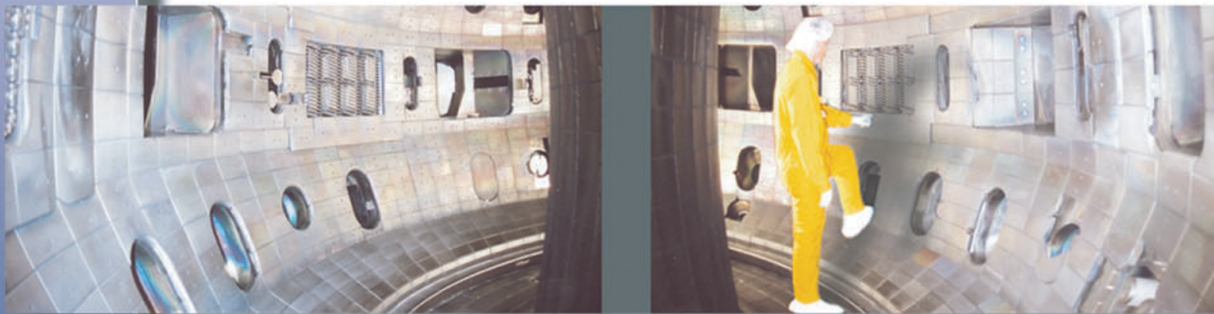
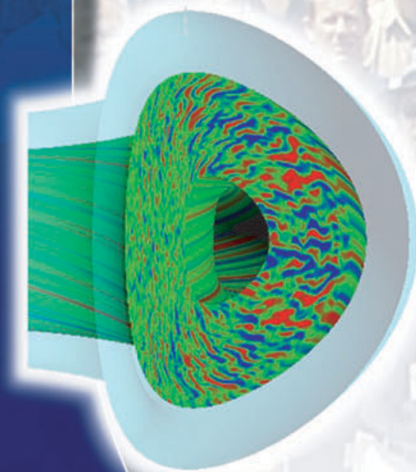




Development of Burning Plasma and Advanced Scenarios in the DIII–D Tokamak

by
T.C. Luce
for the DIII–D Team

Presented at
20th IAEA Fusion Energy Conference
Vilamoura, Portugal

November 1, 2004

DIII–D
NATIONAL FUSION FACILITY
SAN DIEGO

GENERAL ATOMICS

248-04/TCL//rs

THE DIII–D TEAM
CONSISTS OF >300 PROFESSIONALS FROM >70 INSTITUTIONS

US Labs

ANL (Argonne, IL)
INEL (Idaho Falls, ID)
LANL (Los Alamos, NM)
LLNL (Livermore, CA)
ORNL (Oak Ridge, TN)
PNL (Richland, WA)
PPPL (Princeton, NJ)
SNL (Sandia, NM)

US Industries

CompX (Del Mar, CA)
CPI (Palo Alto, CA)
Creare (Hanover, NH)
Digital Finetec (Ventura, CA)
FAR Tech (San Diego, CA)
HiTech Metallurgical (San Diego, CA)
IR&T (Santa Monica, CA)
Orincon (San Diego, CA)
SAIC (La Jolla, CA)
Surmet (Burlington, MA)
Thermacore (Lancaster, PA)
TSI Research (Solana Beach, CA)

US Universities

Alaska (Fairbanks, AK)
Auburn (Auburn, Alabama)
Cal Tech (Pasadena, CA)
Colorado School of Mines (Golden, CO)
Columbia (New York, NY)
Georgia Tech (Atlanta, GA)
Hampton (Hampton, VA)
Lehigh (Bethlehem, PA)
Maryland (College Park, MD)
MIT (Boston, MA)
Palomar (San Marcos, CA)
New York U. (New York, NY)
Texas (Austin, TX)
UCB (Berkeley, CA)
UCI (Irvine, CA)
UCLA (Los Angeles, CA)
UCSD (San Diego, CA)
U. New Mexico (Albuquerque, NM)
Washington (Seattle, WA)
Wisconsin (Madison, WI)

Russia

Ioffe (St. Petersburg, Russia)
Keldysh (Udmurtia, Moscow, Russia)
Kurchatov (Moscow, Russia)
Moscow State (Moscow, Russia)
Triniti (Troitsk, Russia)
Gycom (Nizhny Novgorod, Russia)

European Community

Cadarache (St. Paul-lez, Durance, France)
Consorzia RFX (Padua, Italy)
Culham (Culham, Oxfordshire, England)
Frascati (Frascati, Lazio, Italy)
FOM (Utrecht, The Netherlands)
IPP (Garching, Greifswald, Germany)
JET-EFDA (Oxfordshire, England)
KFA (Julich, Germany)
Kharkov IPT, (Ukraine)
Lausanne (Lausanne, Switzerland)
Chalmers U. (Goteberg, Sweden)
Helsinki U. (Helsinki, Finland)
U. Naples (Naples, Italy)
U. Strathclyde (Glasgow, Scotland)
U. Wales (Wales)

Japan

JAERI (Naka, Ibaraki-ken, Japan)
- **JT-60U**
- **JFT-2M**

Tsukuba University (Tsukuba, Japan)
NIFS (Toki, Gifu-ken, Japan)
- **LHD**

Other International

Australia National U. (Canberra, AU)
ASIPP (Hefei, China)
KAIST (Daegon, S. Korea)
KBSI (Daegon, S. Korea)
National U. (Taiwan)
Nat. Nucl. Ctr. (Kurchatov City, Kazakhstan)
SWIP (Chengdu, China)
U. Alberta (Alberta, Canada)
U. of Kiel (Kiel, Germany)
U. Toronto (Toronto, Canada)

DIII–D
NATIONAL FUSION FACILITY
SAN DIEGO

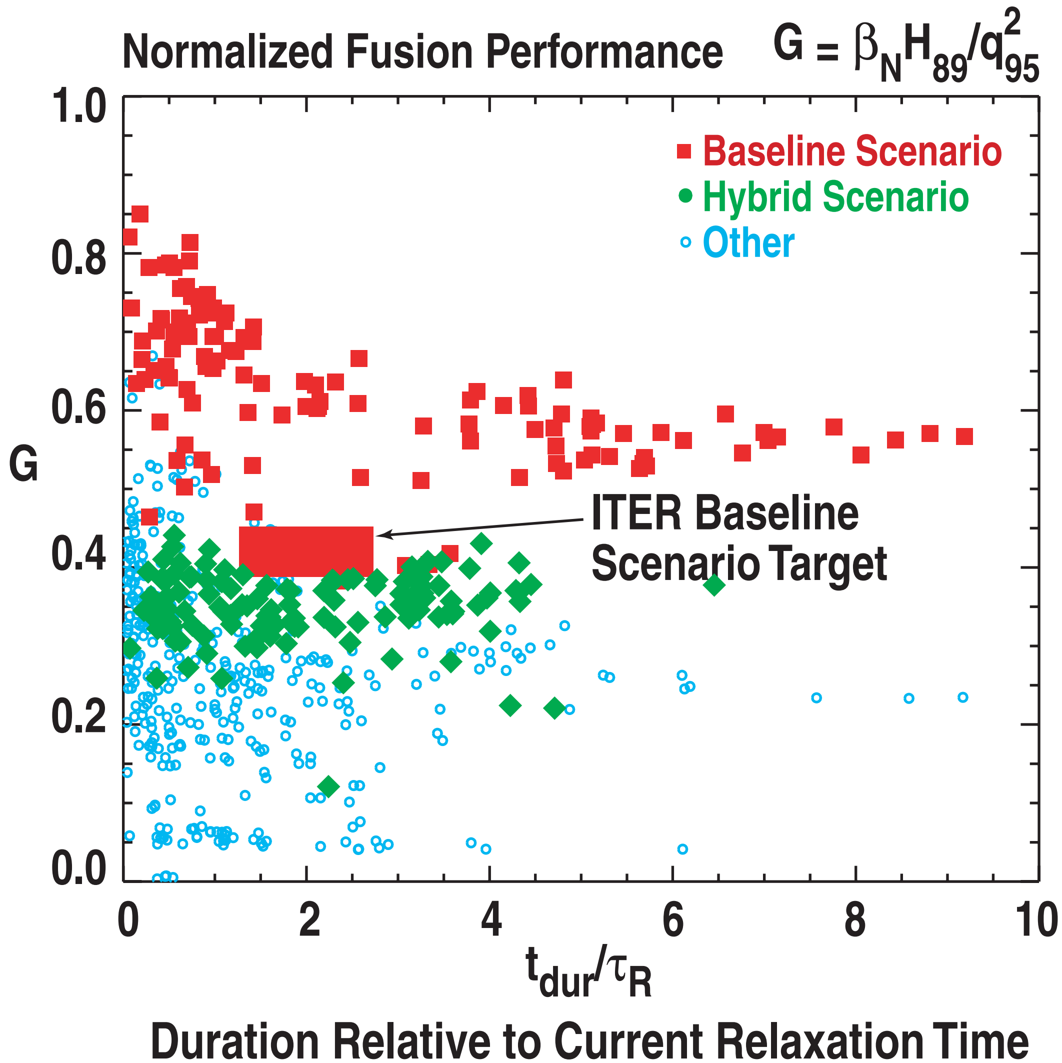
248-04/TLC/rs

DIII–D PROGRAM GOAL: TO ESTABLISH THE SCIENTIFIC BASIS FOR THE OPTIMIZATION OF THE TOKAMAK

Highlights presented here are organized around three themes:

- Strengthening the scientific basis for next-generation burning plasma experiments (ITER)
- Establishing the scientific basis for steady-state tokamak operation
- Investigating fundamental properties of tokamak plasma

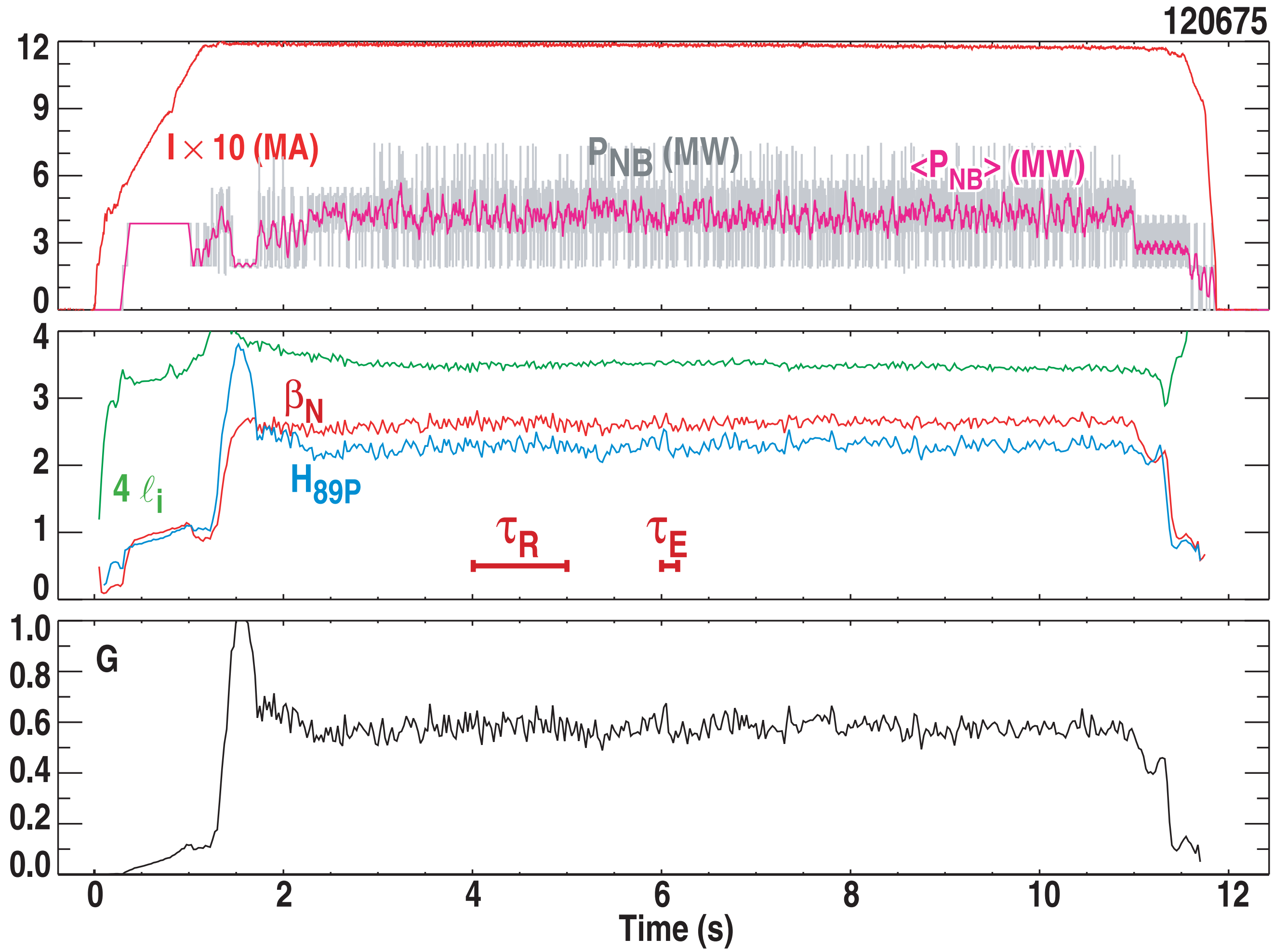
DIII–D EXPERIMENTS INCREASE CONFIDENCE IN REACHING THE ITER PERFORMANCE TARGETS



- Stationary discharges with:
  - — Performance >40% higher than the ITER baseline scenario
  - — Performance equal to the ITER performance target obtained at 30% lower equivalent current

248-04/TCL/rs

STATIONARY HIGH PERFORMANCE ACHIEVED UNDER CONDITIONS SIMILAR TO THE ITER BASELINE SCENARIO



Sawteeth

Type I ELMs

q_{95}=3.2

<β>=4.0%

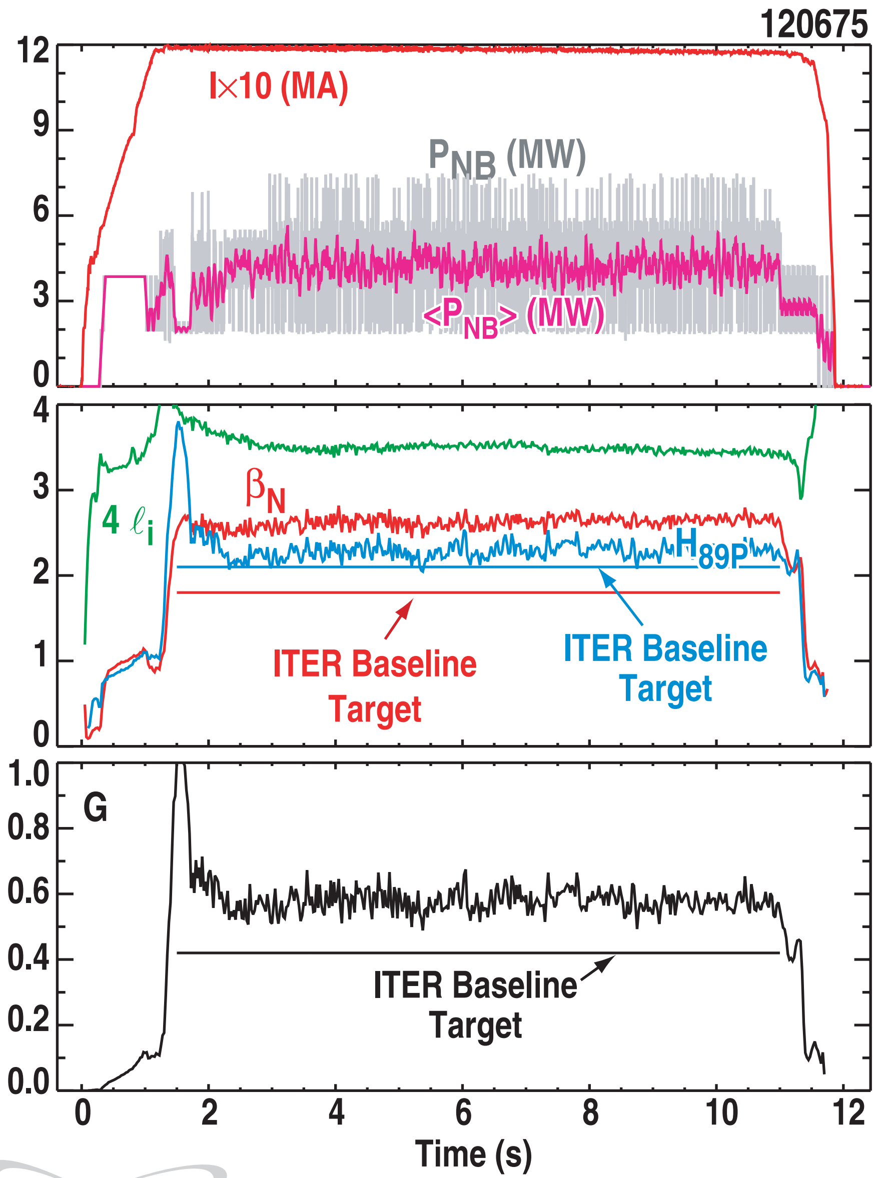
t_{dur}=9.5 s

t_{dur}/τ_R=9.2

t_{dur}/τ_E=55

- Key advance is operation at higher pressure (β_N = 2.8), due to initiation at a relatively benign resistive instability (m=3/n=2 tearing mode) before the onset of sawteeth

STATIONARY HIGH PERFORMANCE ACHIEVED UNDER CONDITIONS SIMILAR TO THE ITER BASELINE SCENARIO



Projection to ITER

$\beta_N = 2.8$ $q_{95} = 3.2$ $n/n_G = 0.85$

$B = 5.3$ T $I = 13.9$ MA

	H	P_{fus} (MW)	P_{aux} (MW)	Q_{fus}
ITER89P	2.4	780	60	12.9
IPB98y2	1.47	740	18.5	39
DS03	1.25 (1.63)*	700	0	∞

*DIII–D Value

Flattop time = 2300 s > 30 min

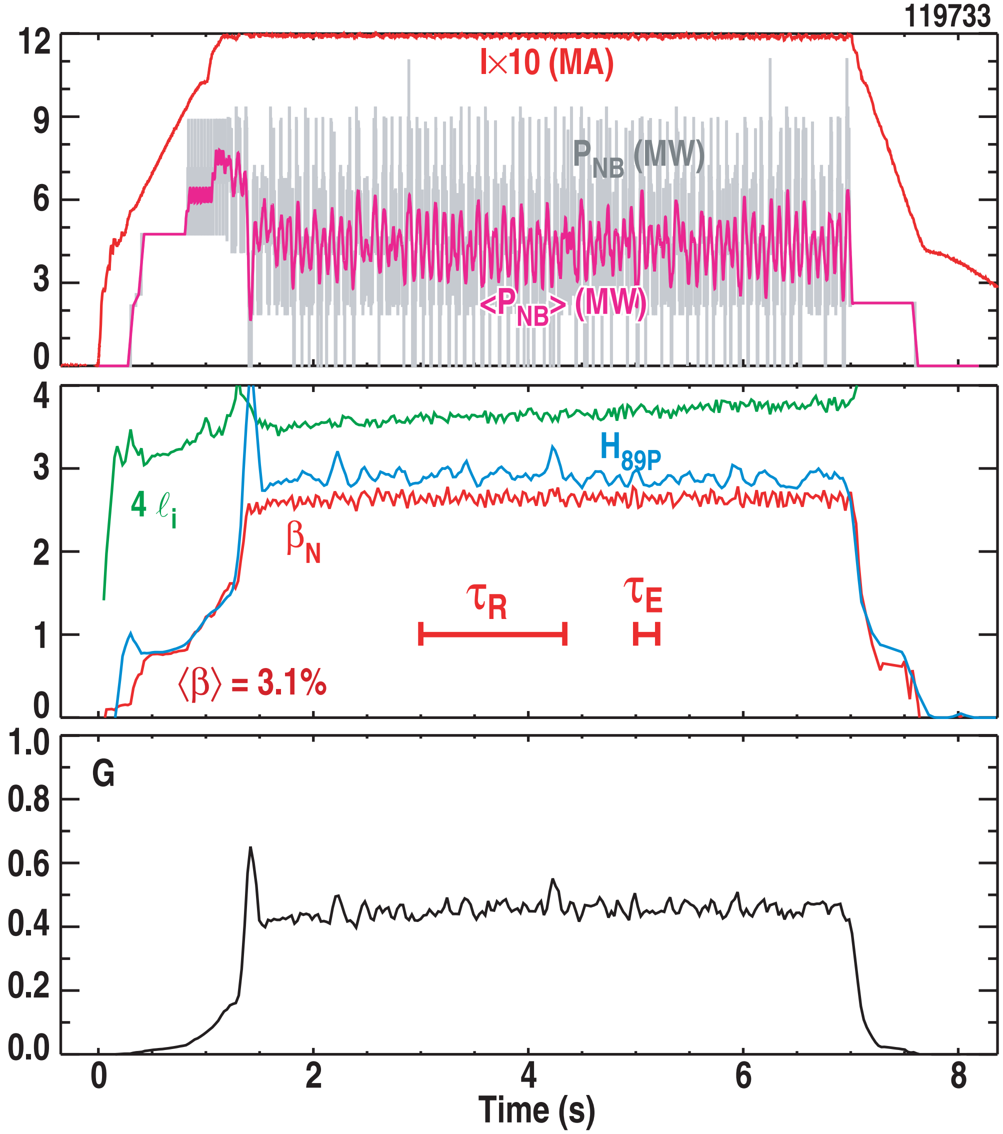
DIII–D NATIONAL FUSION FACILITY SAN DIEGO

ITPA Joint Experiment

Wade – Wed. PM Talk

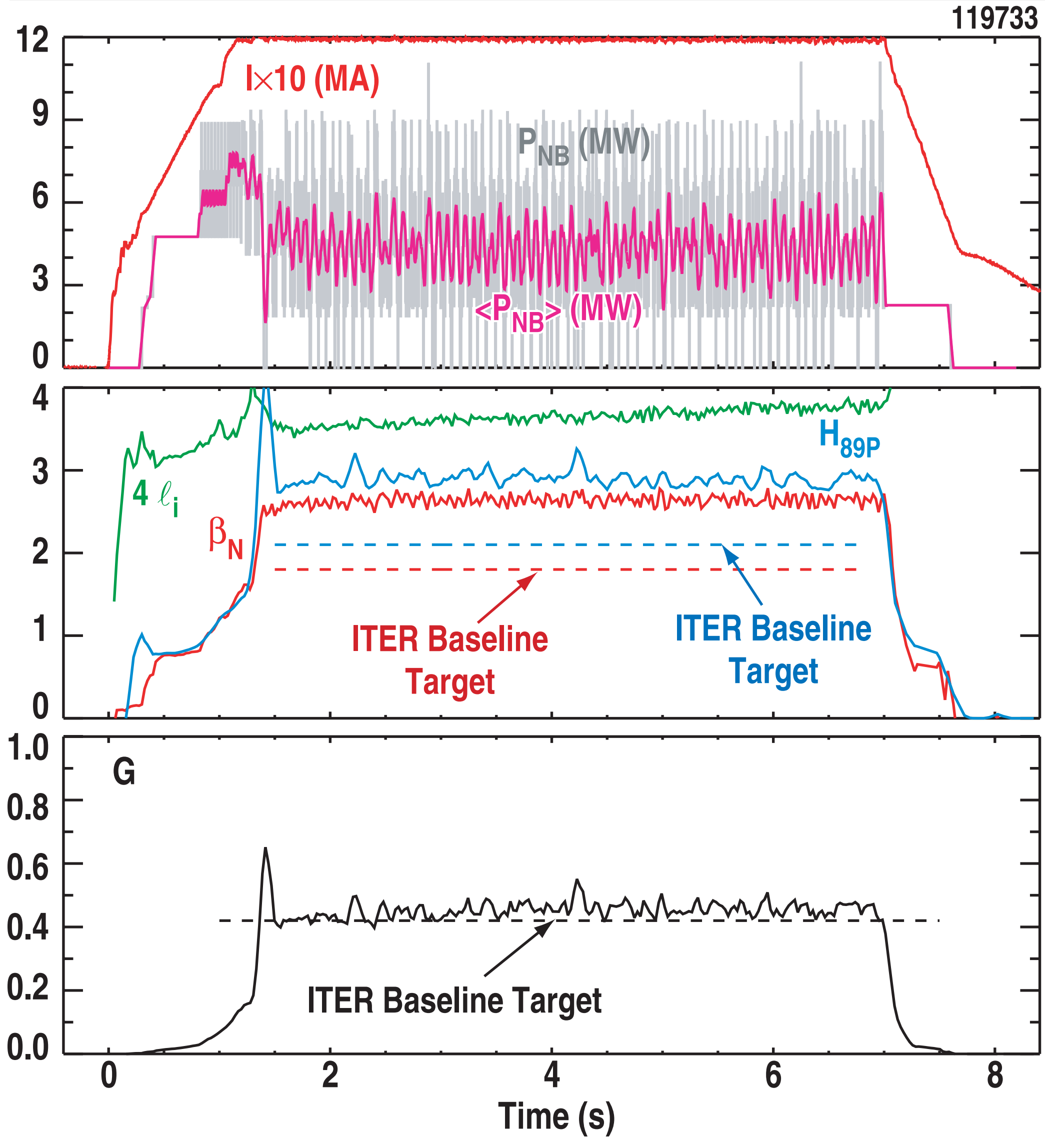
248-04/TCL/rs

CANDIDATE HYBRID SCENARIO DISCHARGES REACH ITER PERFORMANCE TARGET AT REDUCED EQUIVALENT CURRENT



- Hybrid scenario goal is maximum fusion energy (or neutron fluence) per inductive pulse
- DIII–D approach is reduced current and higher normalized pressure, up to no-wall pressure limit (~4 ℓ_i)
- Key advance is again the initiation of a resistive mode (m=3/n=2), which prevents the onset of sawteeth and allows high normalized pressure ($\beta_N \leq 3.2$)

CANDIDATE HYBRID SCENARIO DISCHARGES REACH ITER PERFORMANCE TARGET AT REDUCED EQUIVALENT CURRENT



Projection to ITER

$\beta_N = 2.7$ $q_{95}=4.4$ $n/n_G = 0.85$

$B = 5.3$ T $I = 10.8$ MA

	H	P_{fus} (MW)	P_{aux} (MW)	Q_{fus}
ITER89P	2.75	440	49	9.0
IPB98y2	1.59	440	49	9.0
DS03	1.78 (1.81)*	370	0	∞

*DIII–D Actual Value

Flattop Time = 3900s >1 hour

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

ITPA Joint Experiment

Wade – Wed. PM Talk

248-04/TCL/rs

DIMENSIONLESS SCALING EXPERIMENTS SHOW ENERGY CONFINEMENT IS INDEPENDENT OF BETA

- β scalings for direct experiments and database analysis are very different

IPB98y2

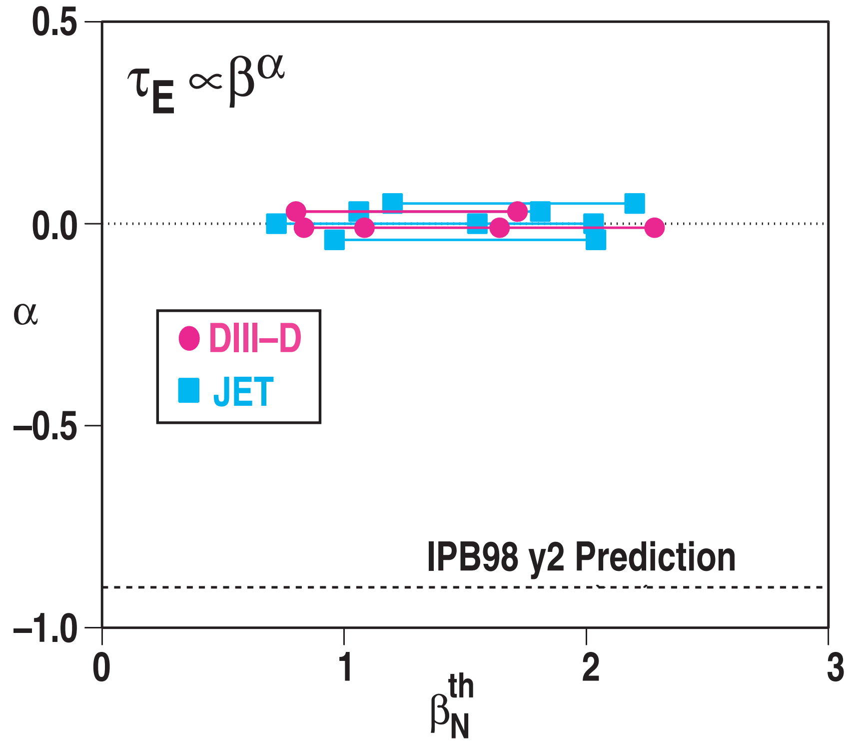
$$\tau_E = 0.0562\ I^{0.93}\ B^{0.15}\ n^{0.41}\ P^{-0.69}\ R^{1.97}\ A^{-0.58}\ \kappa^{0.78}\ M^{0.19}$$

$$B\tau_E \propto \rho_*^{-2.70}\ \beta^{-0.90}\ \nu_*^{-0.01}\ q^{-3.0}$$

DS03

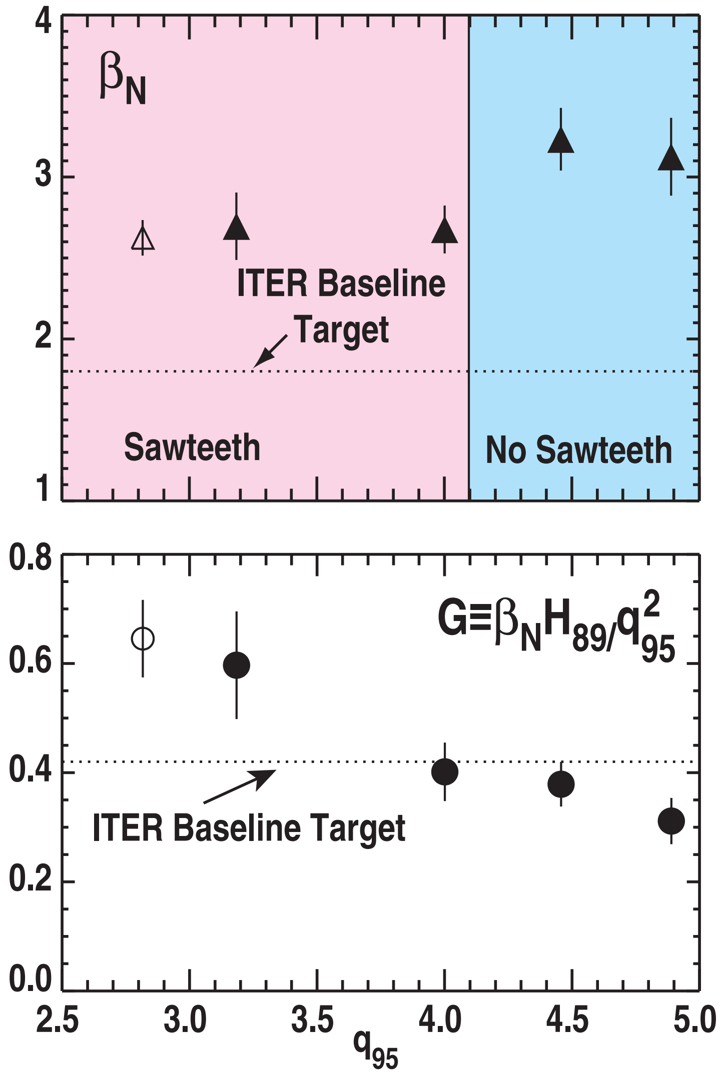
$$\tau_E = 0.028\ I^{0.83}\ B^{0.07}\ n^{0.49}\ P^{-0.55}\ R^{2.11}\ A^{-0.3}\ \kappa^{0.75}\ M^{0.14}$$

$$B\tau_E \propto \rho_*^{-3}\ \beta^{0}\ \nu_*^{-0.14}\ q^{-1.7}$$

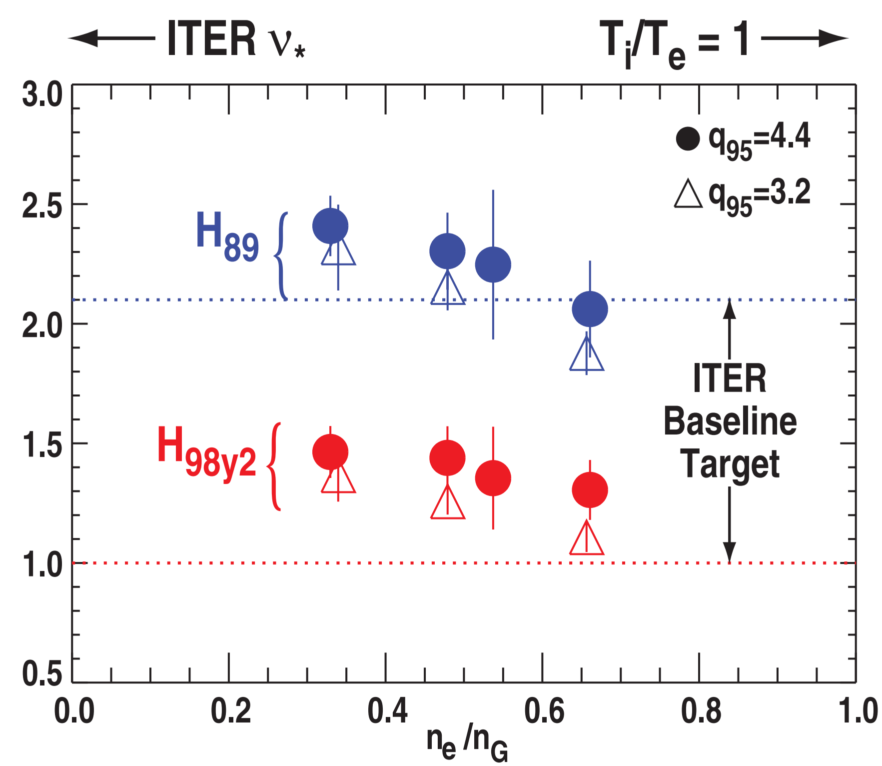


- Absence of β dependence is consistent with drift-type turbulence

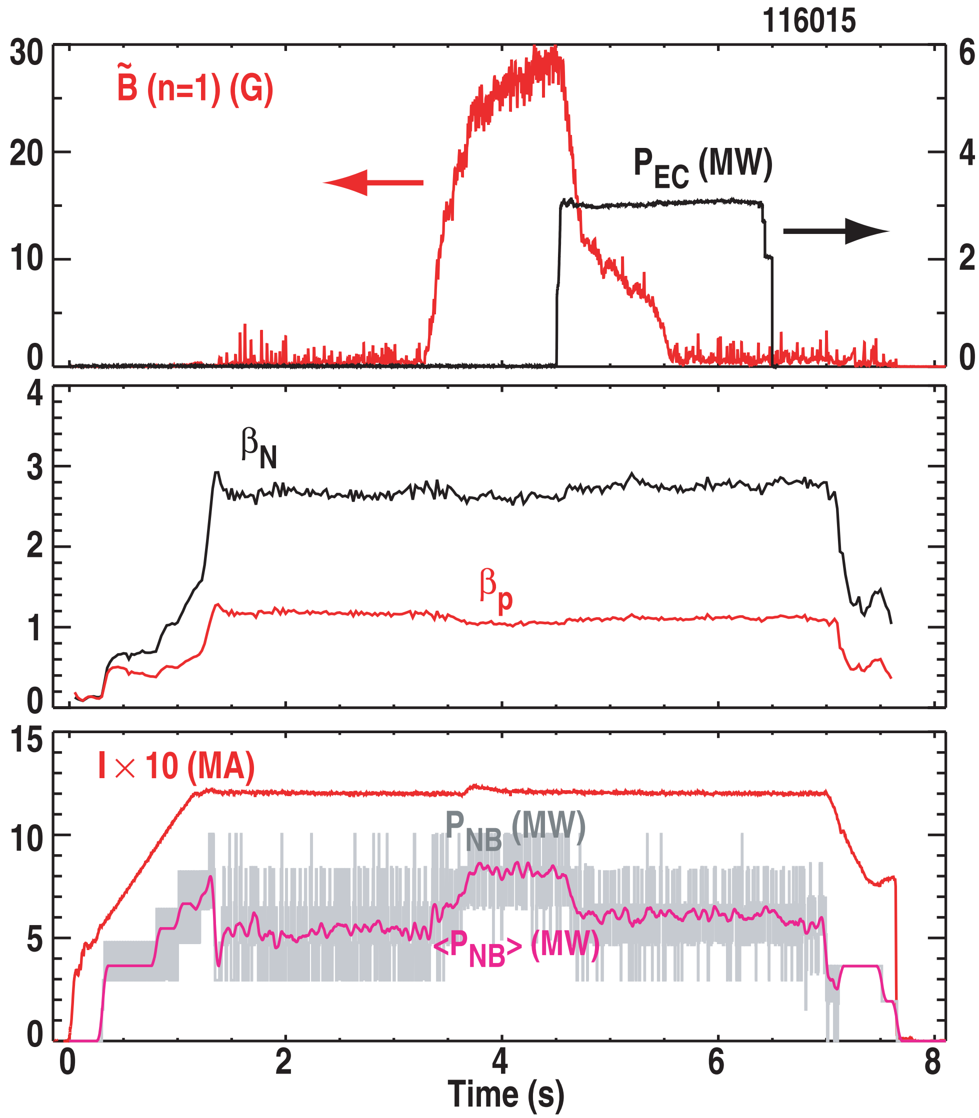
STATIONARY HIGH PERFORMANCE DISCHARGES HAVE BEEN OBTAINED OVER A WIDE RANGE OF PARAMETERS



- Pressure limit higher without sawteeth
- Fusion performance maximizes at low q_{95}
- Confinement drops slightly with increasing density



ROBUST STABILIZATION OF m=2/n=1 TEARING MODE OBTAINED WITH ELECTRON CYCLOTRON CURRENT DRIVE



- This instability sets the operational β limit in the ITER baseline scenario. Locking of this mode often leads to disruptions $\Rightarrow$ stabilization would avoid more extreme mitigation measures

- Stabilization is obtained at high pressure ($\beta_N = 2.8$, $\beta_p = 1.1$)

- Stabilization is optimized automatically by active feedback system to place the ECCD at the location of the instability

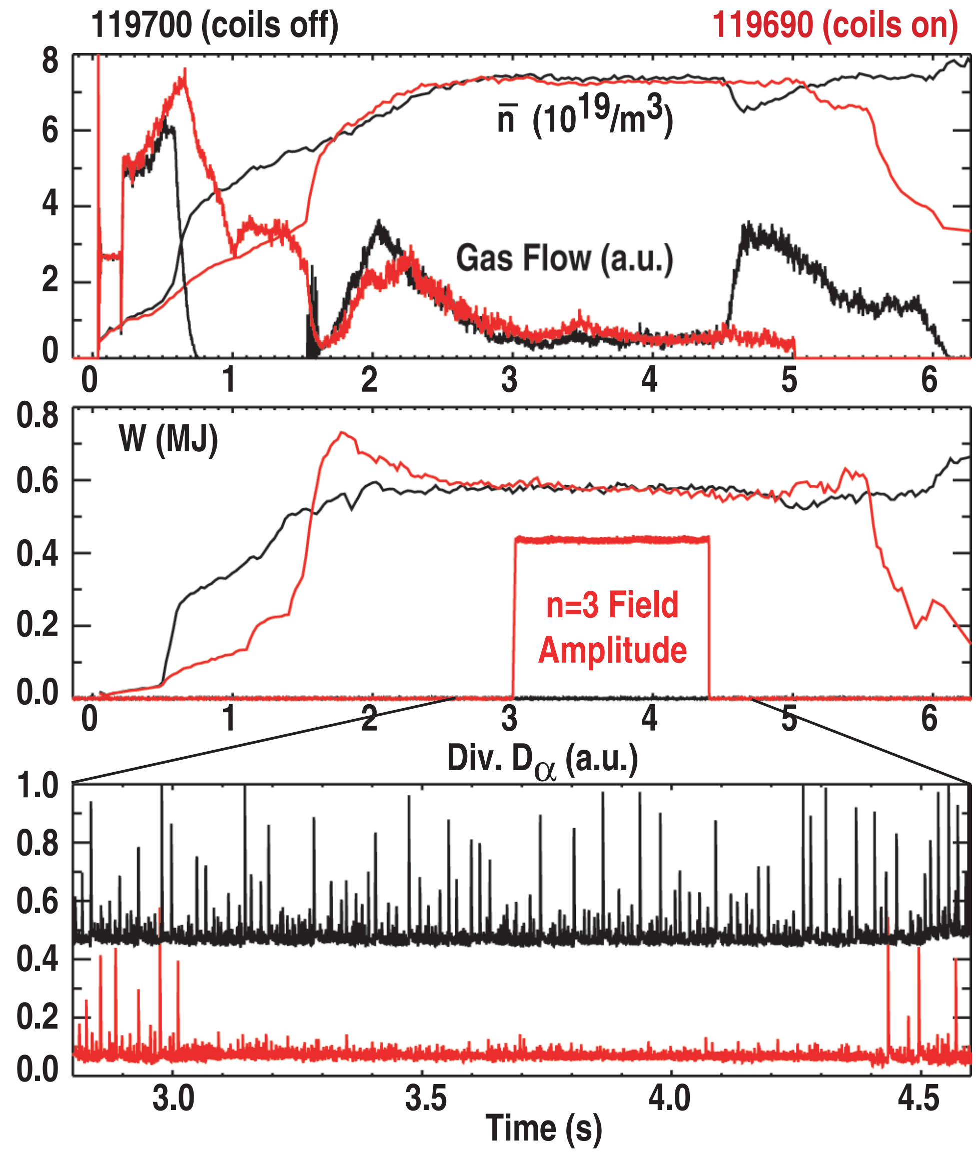
DIII-D NATIONAL FUSION FACILITY SAN DIEGO

Petty – Thurs. PM Talk

248-04/TCL/rs

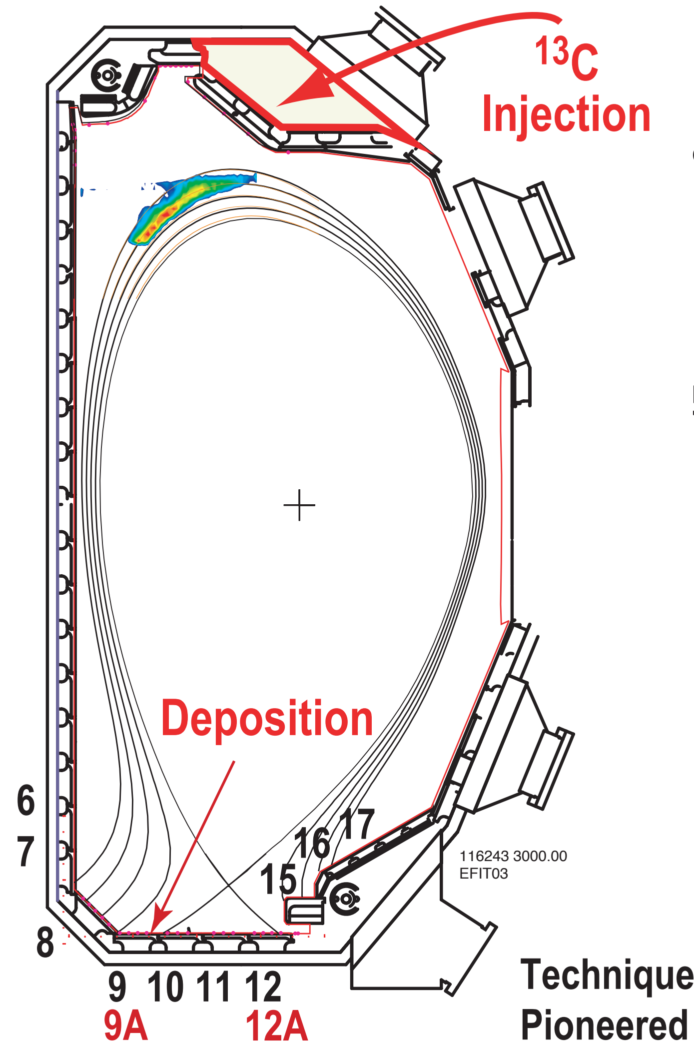
STOCHASTIC EDGE ELIMINATES LARGE TYPE-I ELMS

- Non-axisymmetric fields (n=3) have been used to eliminate ELMs
  - Edge instability changes character
  - Power flow is still to the divertor
  - Technique applied in ITER shape
- Impulsive heat flux reduced by at least a factor of 3
- Confinement and edge pedestal height unchanged

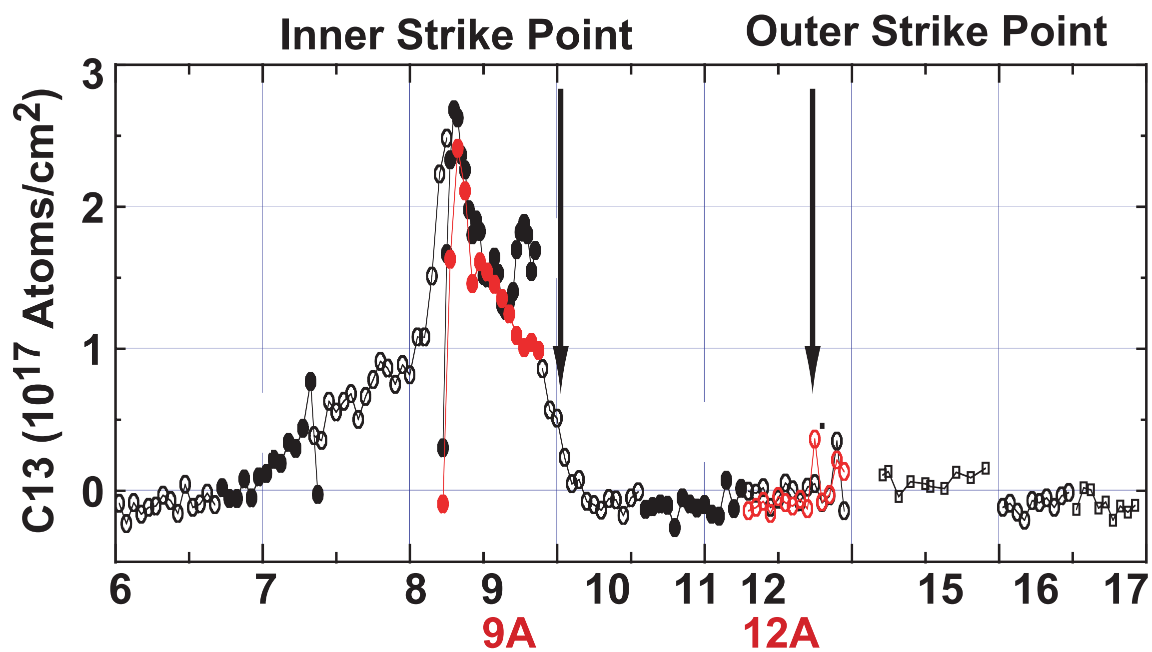


Evans – Tues. PM Talk

CARBON MIGRATION USING ^{13}C AS A TRACER INDICATES TRITIUM CO-DEPOSITION MAY BE LOCALIZED AT THE INNER DIVERTOR



Technique Pioneered by JET



- Measured deposition highly localized at inner divertor; toroidally symmetric
- Modeling of deposition, carbon plume, and core carbon inventory requires flow velocity M~0.4 at the separatrix in the direction of the inner divertor

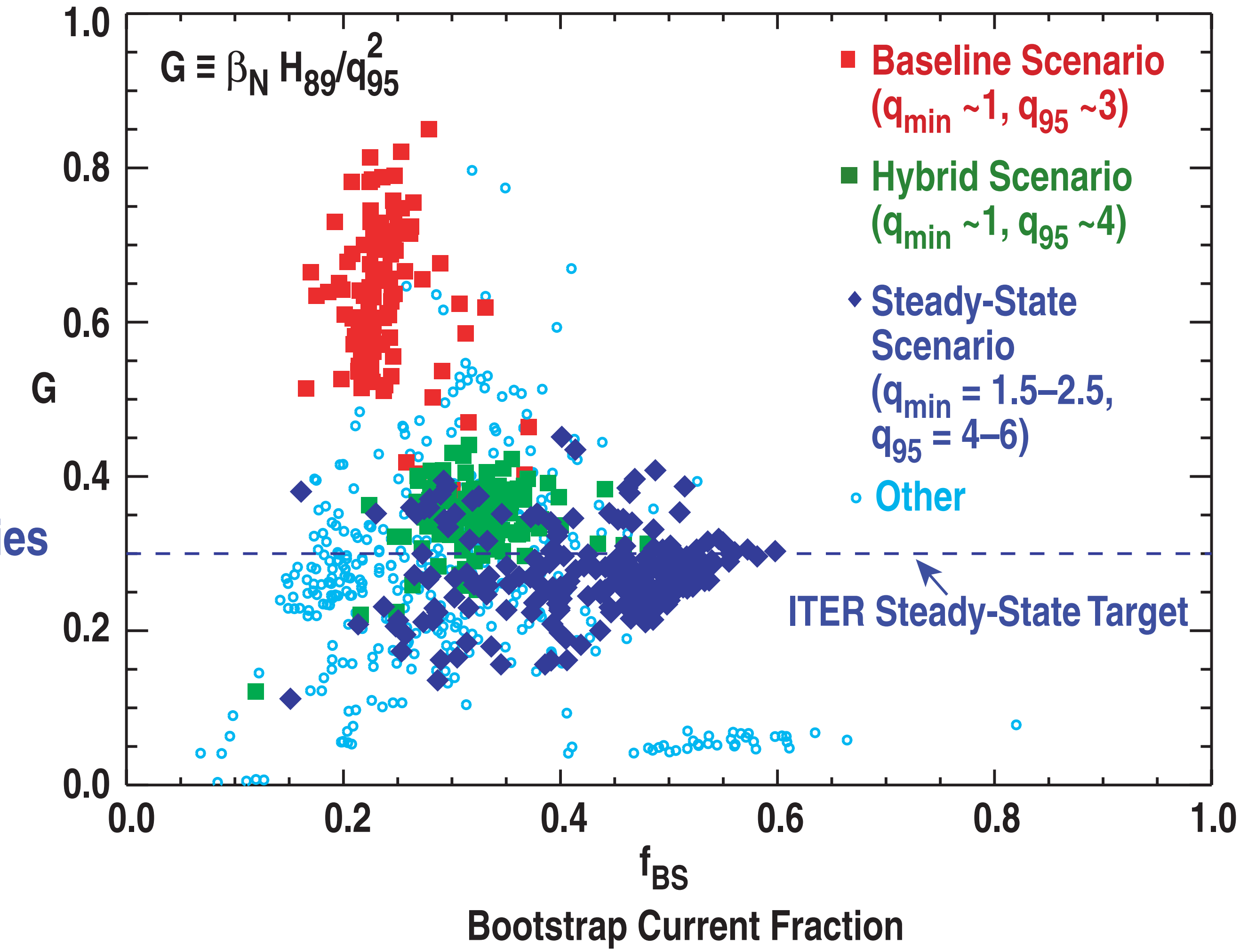
DIII-D NATIONAL FUSION FACILITY SAN DIEGO

248-04/TCL/rs

STEADY-STATE TOKAMAK OPERATION REQUIRES A COMPROMISE BETWEEN FUSION PERFORMANCE AND BOOTSTRAP CURRENT

- Full non-inductive discharges in present tokamaks that project to high fusion gain are a necessary first step

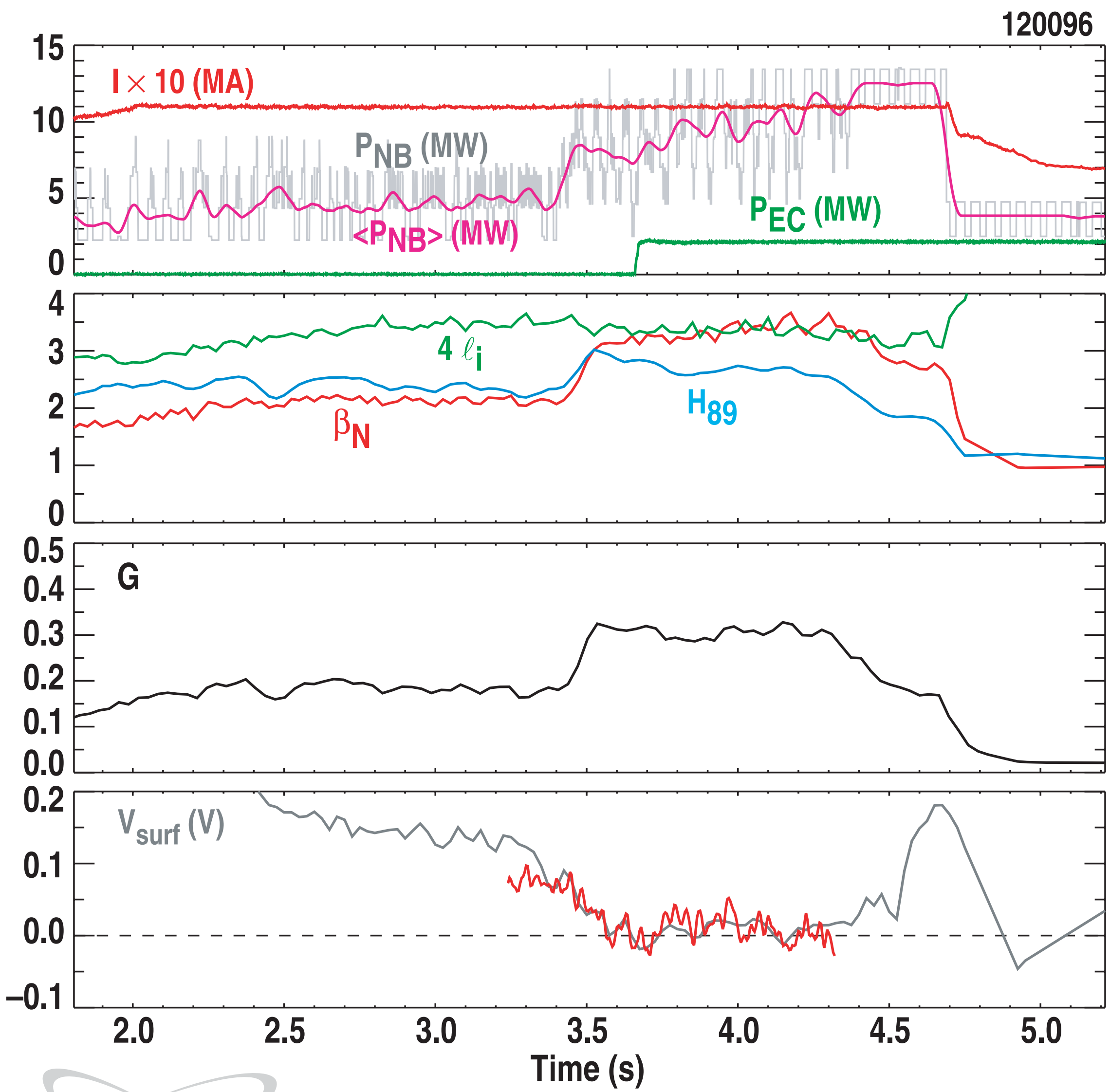
- Control of discharge parameters and instabilities will be an essential component of a steady-state tokamak



DIII-D NATIONAL FUSION FACILITY SAN DIEGO

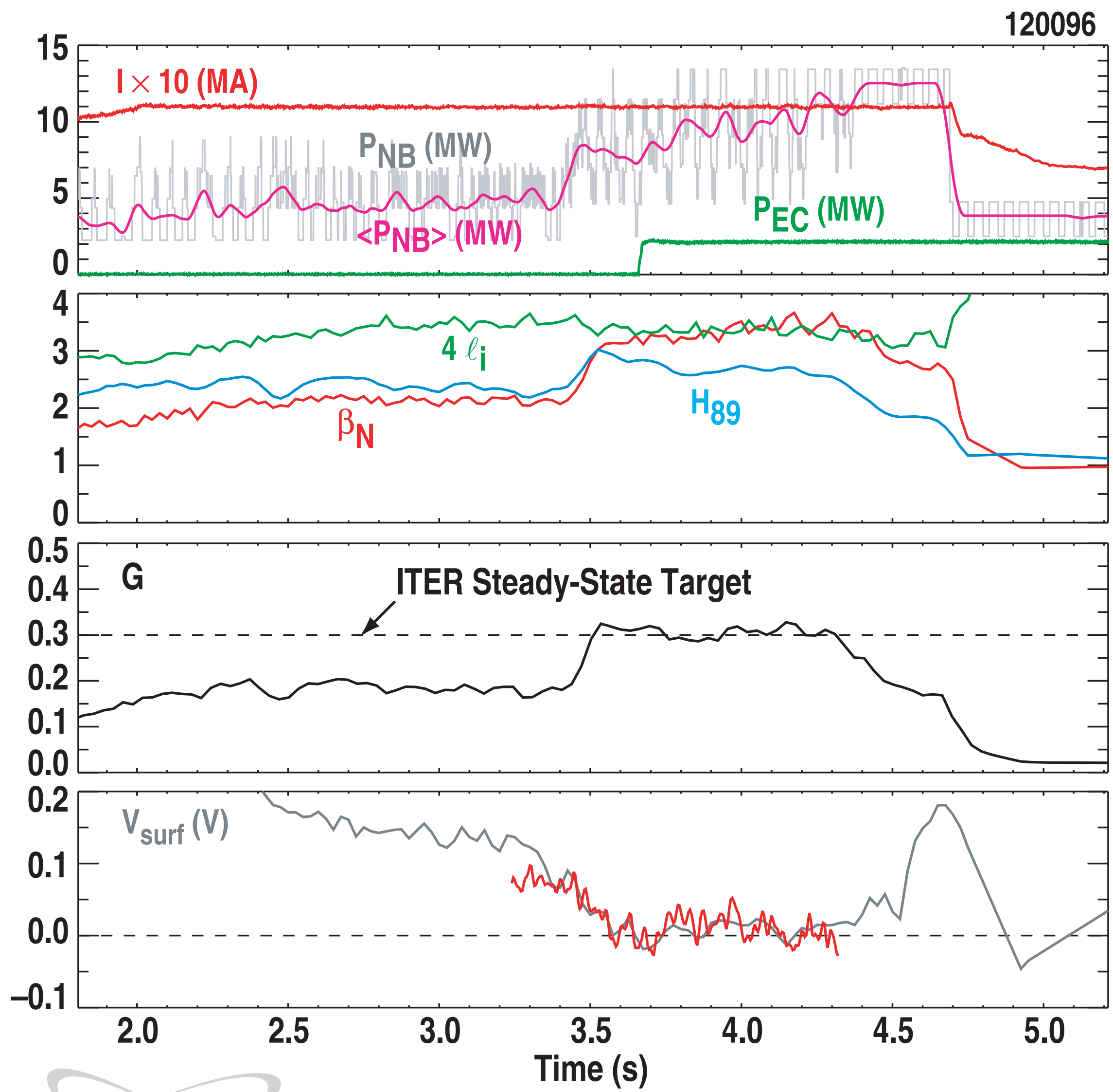
248-04/TCL/rs

FULL NON-INDUCTIVE DEMONSTRATION DISCHARGE HAS BEEN OBTAINED



- Pressure at or above the no-wall pressure limit ($\beta_N \geq 4\ \ell_i$) for high fusion power
- Elevated q_{min} (>1.5) for enhanced bootstrap current (f_{BS} ~0.6)
- Reduced current (q_{95} ~5) to minimize non-inductive current requirements

FULL NON-INDUCTIVE DEMONSTRATION DISCHARGE HAS BEEN OBTAINED



$B = 5.3$ T	$\beta_N = 2.8$
$I = 9.3$ MA	$q_{95} = 5.4$
$\bar{n} = 0.63\times10^{20}$ m^{-3}	$H_{DS03} = 1.28$
$P_{NB} = 33$ MW	$I_{NB} = 2.3$ MA
$P_{EC} = 20$ MW	$I_{EC} = 0.5$ MA
$P_{IC} = 20$ MW	$I_{IC} = 0$ MA
$P_{fus} = 340$ MW	$I_{BS} = 6.7$ MA
Type I ELMs	$Q_{fus} = 4.7$

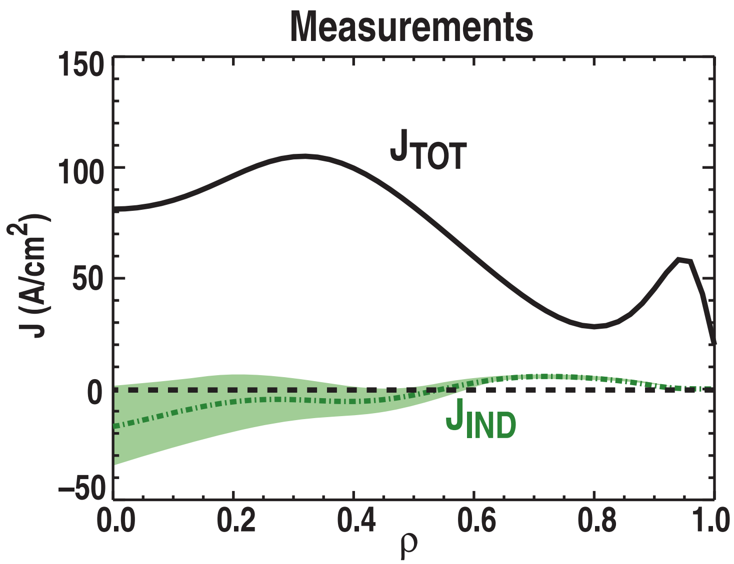
DIII-D NATIONAL FUSION FACILITY SAN DIEGO

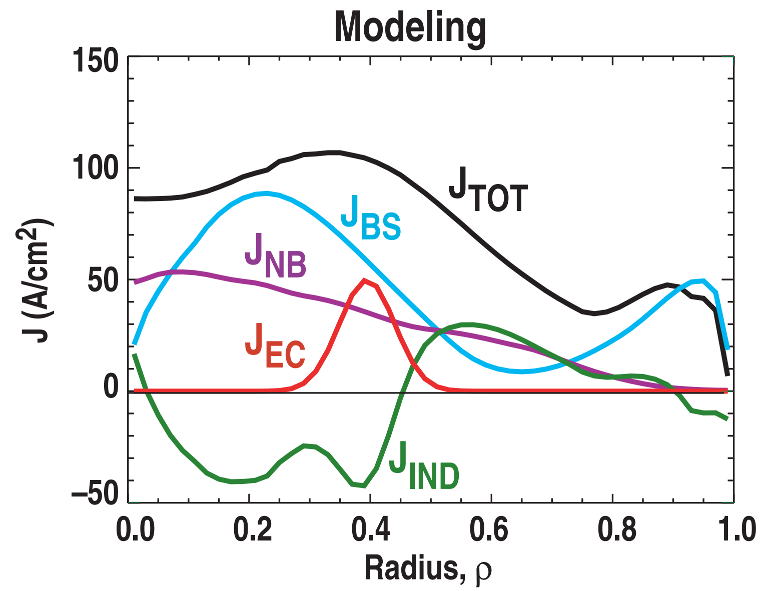
ITPA Joint Experiment

Murakami – Tues. AM Talk

248-04/TCL/rs

MEASUREMENTS AND MODELING INDICATE FULL NON-INDUCTIVE CURRENT SUSTAINMENT WITH GOOD ALIGNMENT

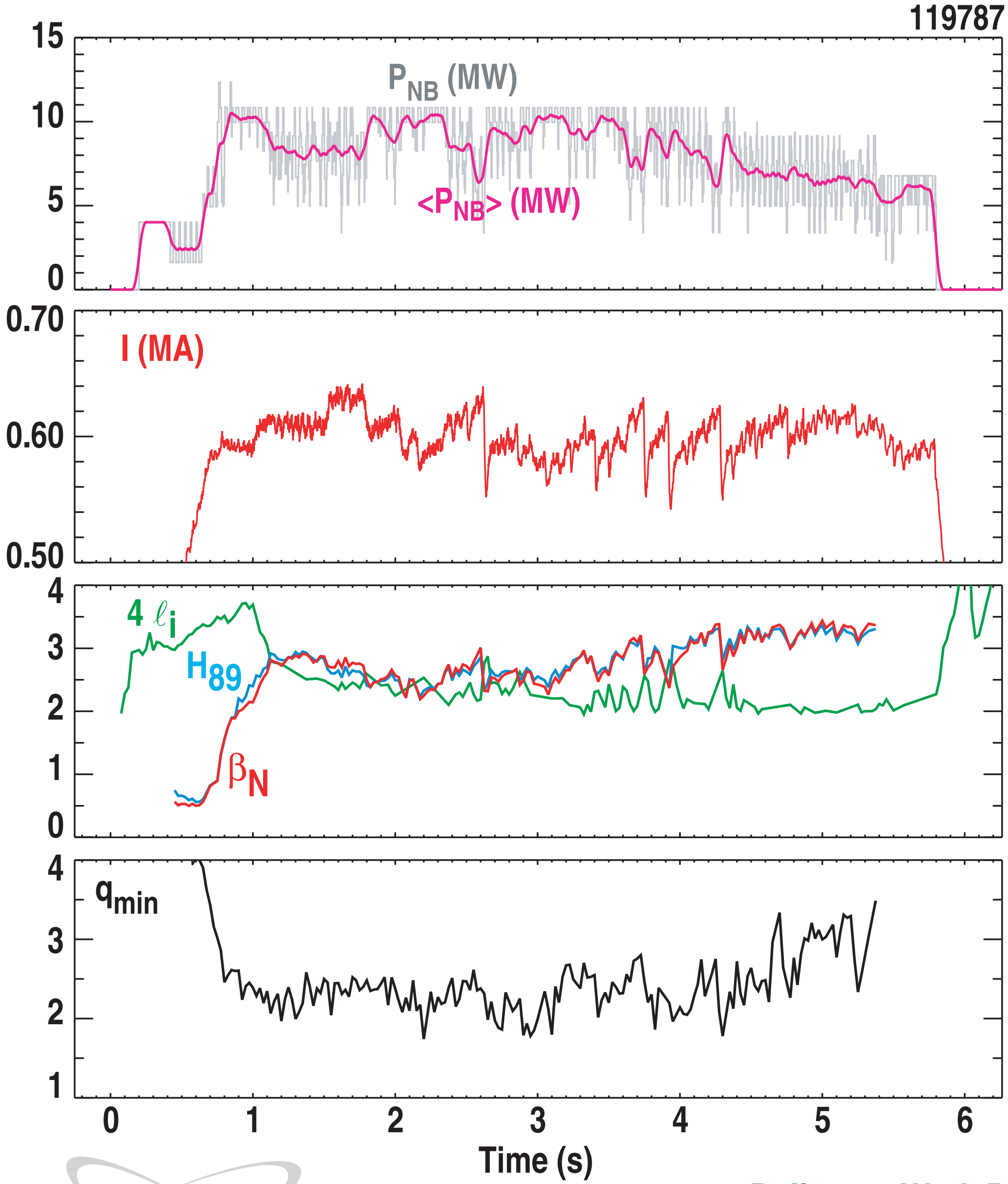




- Surface voltage from equilibrium reconstructions shows no net inductive flux (but note $\Re = 0.15\ \mu\Omega \Rightarrow 15$ mV = 100 kA)
- Internal electric field determined from equilibrium reconstructions shows little spatial structure ⇒ non-inductive sources well aligned to total current
- Toroidal electric field inferred directly from MSE data confirms equilibrium analysis

- Modeling gives nearly full non-inductive current:
  - Bootstrap 59%
  - Neutral Beam 31%
  - Electron cyclotron 8%
  - Inductive 2%

TRANSFORMERLESS OPERATION SHOWS CONTROL OF HIGH BOOTSTRAP FRACTION PLASMAS WILL BE CHALLENGING



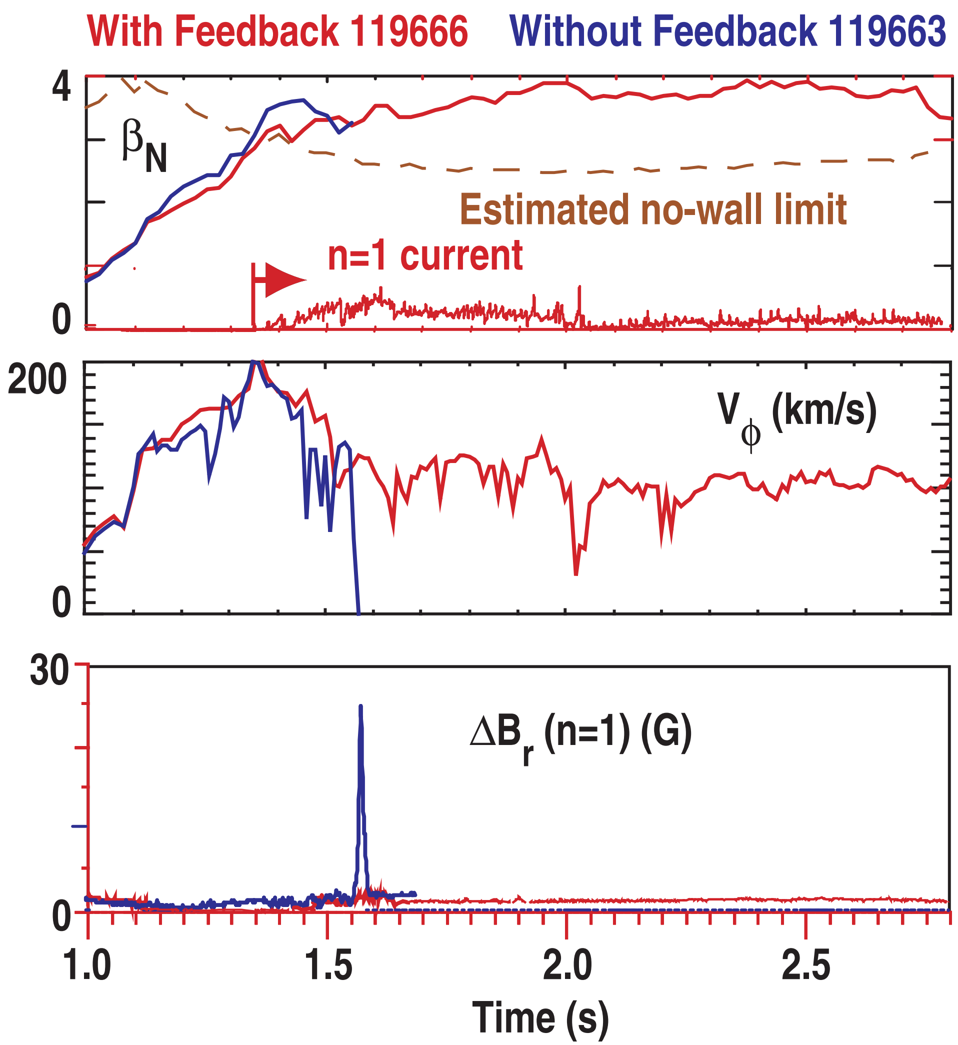
- The desired steady-state operating point may not be a stationary solution to the coupled fluid equations. If not, active control is required.
- Inductive control of the plasma current may be desirable ⇒ non-inductive overdrive will be required.
- At high safety factor (q_{95}~10) and high qmin (~3), the bootstrap current fraction is >80%.

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

Politzer – Wed. PM Poster

248-04/TCL/rs

ACTIVE RESISTIVE WALL MODE STABILIZATION ALLOWS EXTENDED OPERATION ABOVE THE NO-WALL PRESSURE LIMIT



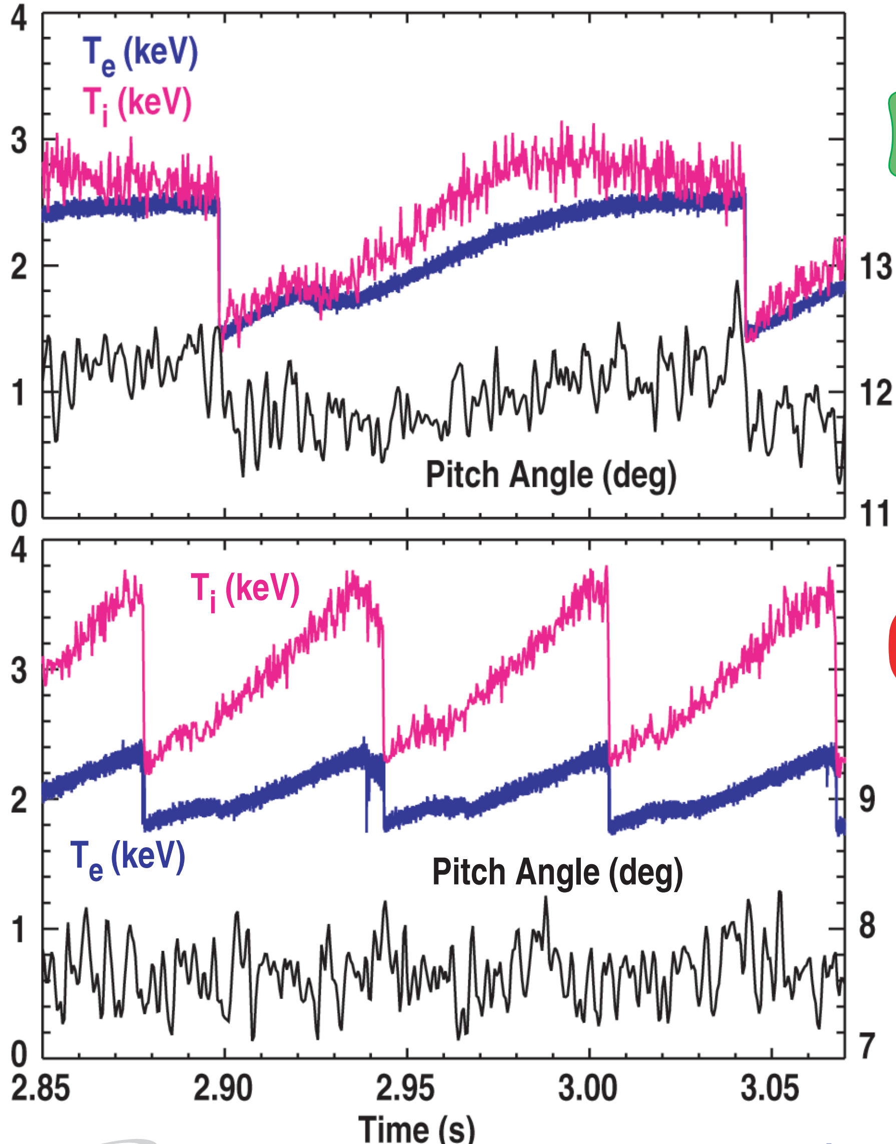
- Rotation is effective in stabilizing the RWM up to the ideal-wall limit in many cases
- Rotation is ineffective in the case shown, perhaps due to a reduction in the number of rational surfaces in the plasma interior (lower q_{95})
- Active feedback using the 12 internal coils in n=1 symmetry allows operation above the no-wall limit for ~200 growth times

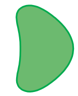
Okabayashi – Wed. AM Talk

PROGRESS IN UNDERSTANDING TOKAMAK PHENOMENA IS BEING MADE IN MANY AREAS

- Magnetic geometry changes the character of the sawtooth instability
- Development of diagnostics and theory to understand the H–mode pedestal
- Electron heat transport exhibits no threshold behavior
- Plasma fueling is dominated by neutrals originating in the divertor
- Plasma rotation is strongly changed by ECH (no external torque applied)
- Gas jets mitigate disruptions safely despite a short penetration length

SAWTOOTH INSTABILITY CHANGES CHARACTER WITH CROSS-SECTION SHAPE

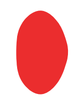




Indented plasmas:

— Electron and ion temperatures remain close

— Significant evolution of the central current density



Oval plasmas:

— Ion temperature significantly above electron temperature despite roughly equal heating

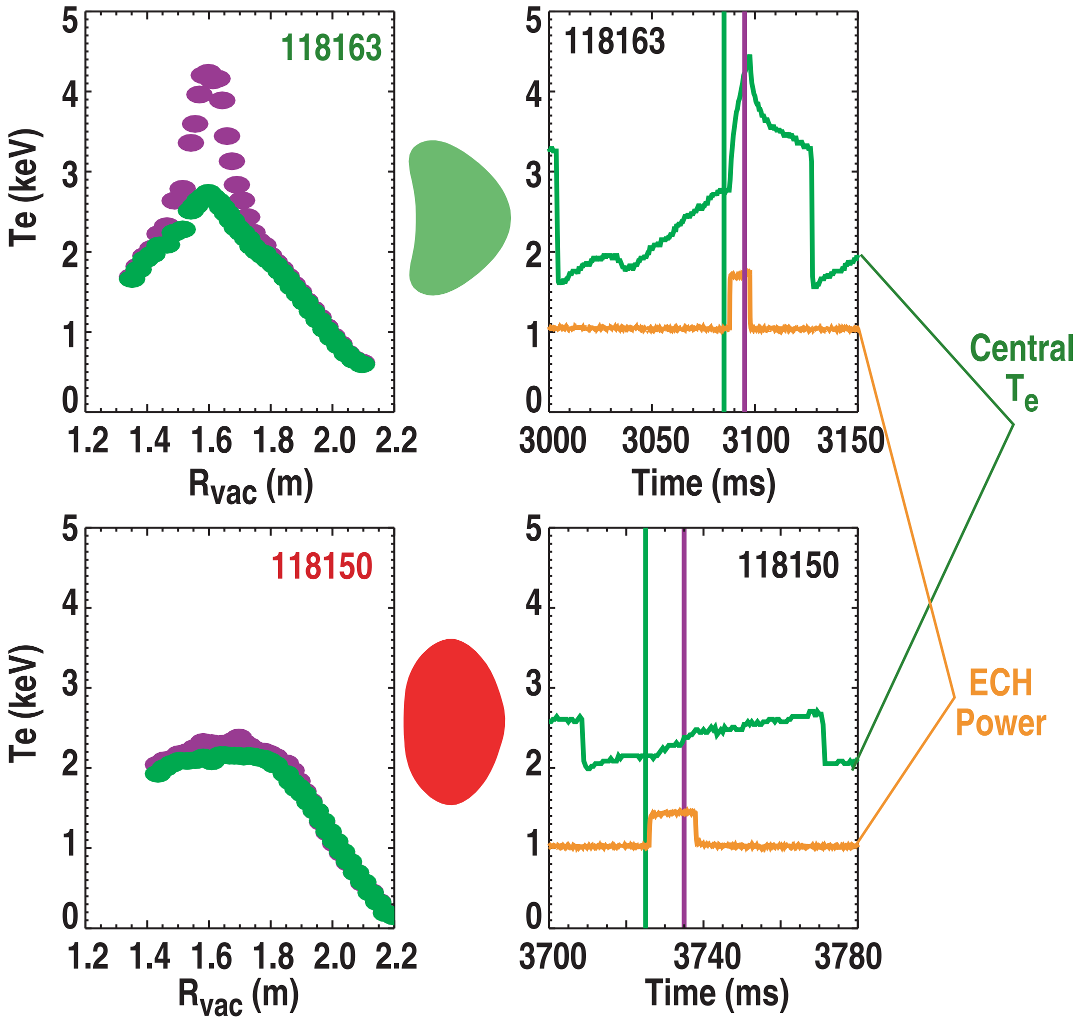
— No significant evolution of the central current density

Lazarus – Fri. AM Poster

DIII-D NATIONAL FUSION FACILITY SAN DIEGO

248-04/TCL/rs

PLASMAS THAT VIOLATE THE MERCIER CRITERION DO NOT SUPPORT AN ELECTRON PRESSURE GRADIENT



Indented Plasmas:

- Mercier limit occurs at q<1
- Local electron heating results in strongly increased gradient

Oval Plasmas:

- Mercier limit occurs at q>1
- Local electron heating results in almost no change in gradient

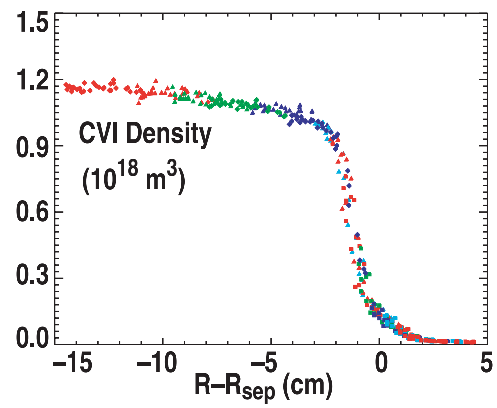
Lazarus – Fri. AM Poster

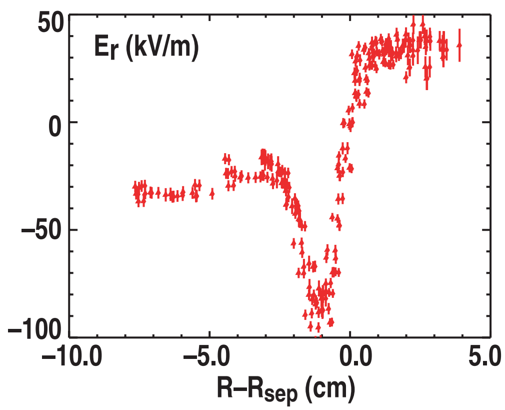
DIII-D NATIONAL FUSION FACILITY SAN DIEGO

248-04/TCL/rs

PREDICTIVE UNDERSTANDING OF THE EDGE PEDESTAL REQUIRES CONTINUING IMROVEMENTS IN MEASUREMENT AND THEORY

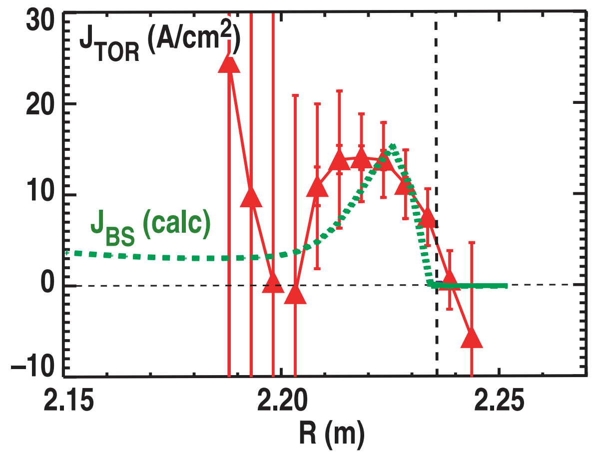
Pedestal Pressure





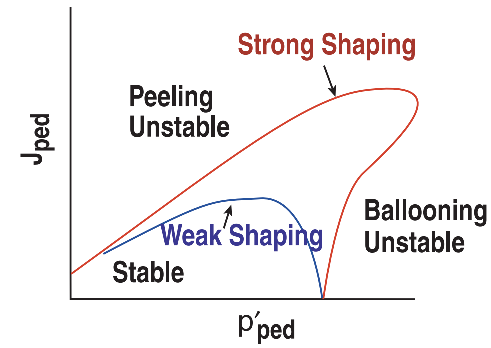
- Need continuous electron pressure measurement

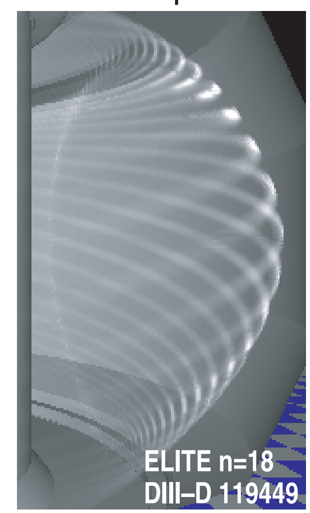
Pedestal Current



- Need improved time resolution

Stability Theory





- Need nonlinear (time-dependent) theory

CONCLUSIONS

- Performance projections for ITER using the present design basis are conservative. DIII–D results provide confidence in reaching the ITER performance targets and optimism that they can be exceeded. Technical challenges are being addressed actively.

- A steady-state tokamak scenario has been demonstrated in DIII–D that projects to modest fusion gain in ITER. The ability to control such plasmas near the performance boundaries is the next task.

- The DIII–D facility is providing unprecedented views of tokamak phenomena, enabling the development and validation of theories and models of plasma behavior.

DIII–D AND GENERAL ATOMICS PRESENTATIONS

	Tuesday	Wednesday	Thursday	Friday	Saturday
AM Talks	Murakami – Steady-state Scenario	Okabayashi – RWM Control	Nazikian – Energetic Particles Fredrickson – MHD Instabilities	Waltz – Gyrokinetic Simulations	Hollmann – Disruptions
AM Posters		Fenstermacher – Pedestal Physics	West – QH–Mode Kinsey – Transport Modeling Parks– Pellets	Petrie – Divertor Geometry and Pumping Groth – Plasma Fueling Rudakov – SOL Transport Lazarus – Sawteeth deGrassie – Rotation	
PM Talks	Evans – ELMs (Suppression, etc.)	Wade – Hybrid Scenarios	Petty – Tearing Modes		
PM Posters		Ferron – β Limits in Steady-State Scenarios Politzer – High Bootstrap Discharges	Solomon – Poloidal Rotation	DeBoo – Critical Gradients Rhodes –Turbulence Measurements	

Burning Plasma   Steady-State   Tokamak Physics   Other

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

248-04/TCL/rs