Review of ITER Physics Issues and Possible Approaches to their Solution

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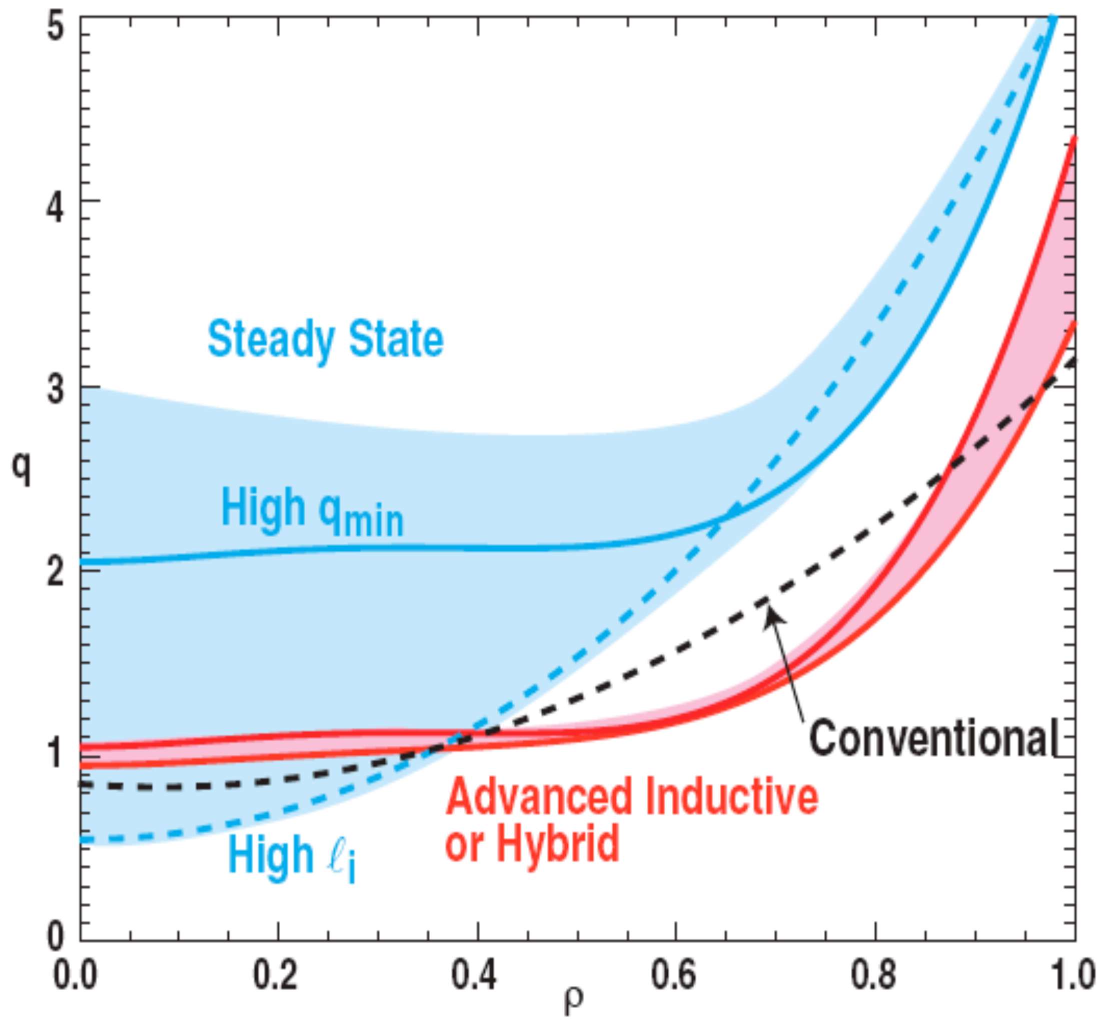
GENERAL ATOMICS

New Operating Modes are Becoming Available to ITER

- **ITER Baseline- Conventional Sawtoothing, Elming H-mode**
  - Conservative route to 500 MW, 400 seconds
  - Some issues remain
- **Hybrid, Advanced Inductive**
  - Extended pulse length
  - Same fusion power
  - New baseline??
- **Advanced, Steady-state**
  - High gain
  - True steady-state
  - Points to DEMO

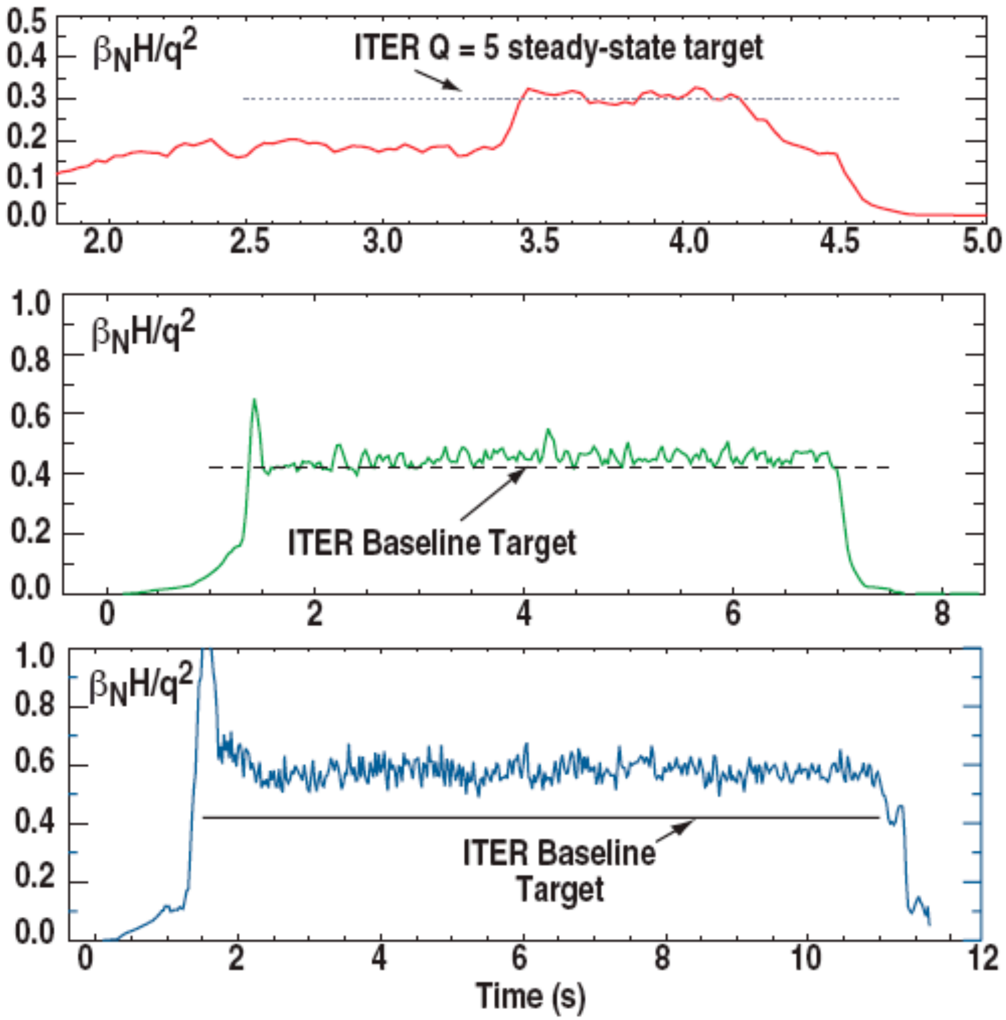


GENERAL ATOMICS

ITER Must Show the Way to Advanced Operation of a Fusion DEMO Power System

Parameter / Feature	Sawtoothing ELMing H-mode	Hybrid, Advanced Inductive	Advanced, Steady-State
Stability, β_N	~ 2, < 2.8	2.5 - 3.5	3.0-4.5 (SN), 3.5-5.5 (DN)
Confinement, H_{98Y2}	1.0	1.3-1.6	1.3-1.6
Non-inductive fraction	20-30 %	50-70 %	100 %
Bootstrap fraction	20-30 %	40-60 %	50-75 %
n/n_G	0.8	0.5-0.7	0.5-0.6
q_0	<1.0	>1.0 - 1.8	2-4 Flat to Reversed Shear
q_{MIN}	<1.0	>1.0 - 1.8	>2.0
q_{EDGE}	3	3.5-5	4-5
Sawteeth	Present, Sought	Absent, 3/2 mode?	Absent, High q_{MIN}
NTM 3/2	Present, Sawtooth Triggered	Present, Helpful?	Absent, High q_{MIN}
2/1	Present, Sawtooth Triggered	Present	Absent, High q_{MIN}
Startup	Outer Limiter, Slowly Growing Plasma, Divert and Heat after Flattop	Inner Wall, Divertor and Large Size as Early as Possible, Heat on Rampup	Inner Wall, Divertor and Large Size as Early as Possible, Heat on Rampup

Advanced Tokamak Modes Offer Performance Extension in ITER



- **Steady-state scenarios**

  $Q_{ITER} \sim 5$, t_{dur} (phys) $= \infty$,

  $I_p^{(ITER)} \approx 9$ MA

- **Hybrid scenario**

  $Q_{ITER} \sim 10$, $t_{dur} > 1$ hr,

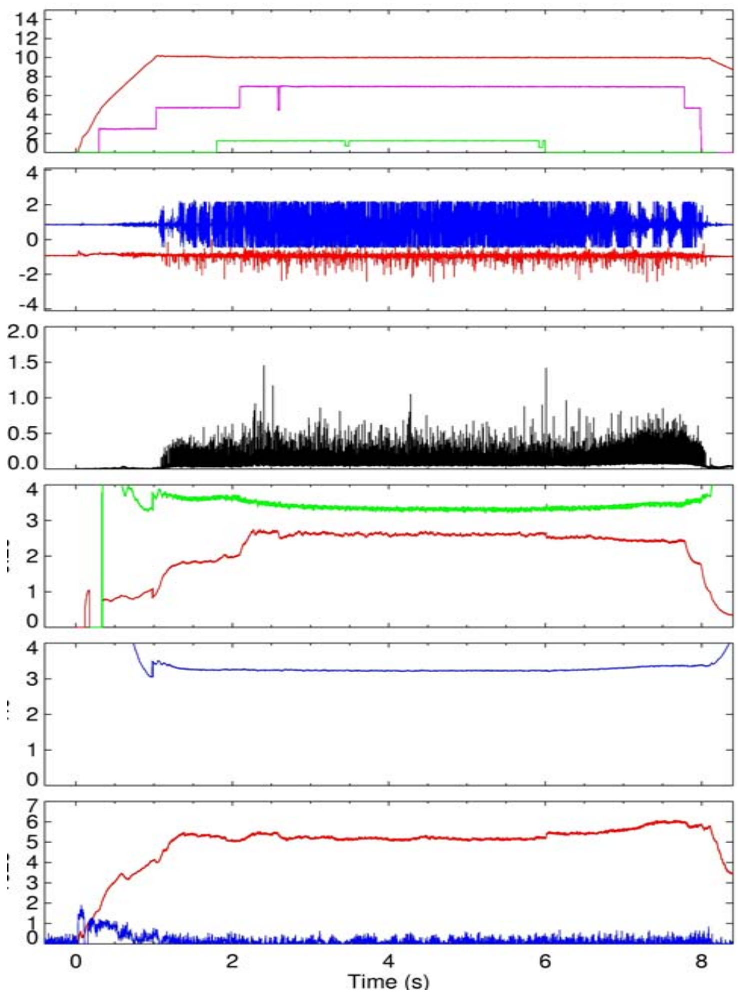
  $I_p^{(ITER)} \approx 11$ MA

- **Advanced H-mode**

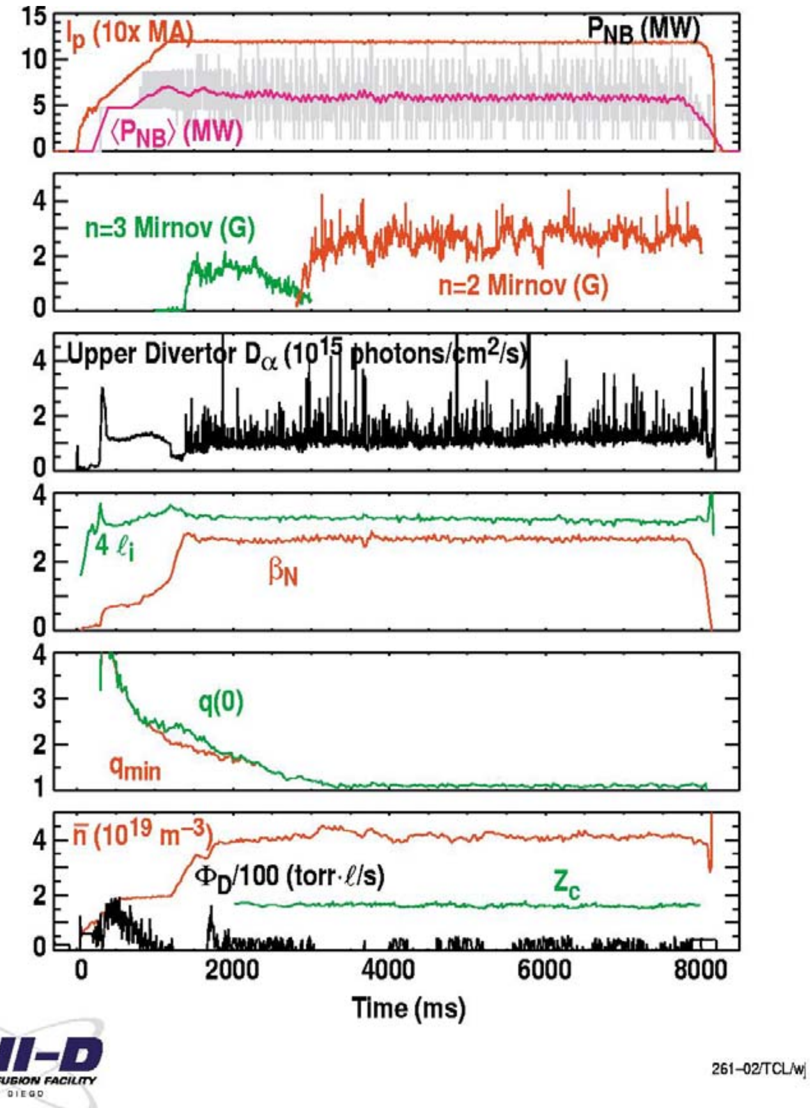
  $Q_{ITER} \sim 40$, $t_{dur} > 30$ min,

  $I_p^{(ITER)} \approx 14$ MA

Hybrid scenario in ASDEX Upgrade and DIII-D



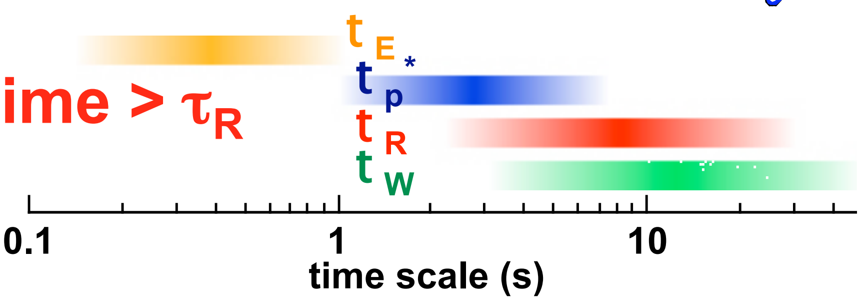
ASDEX Upgrade



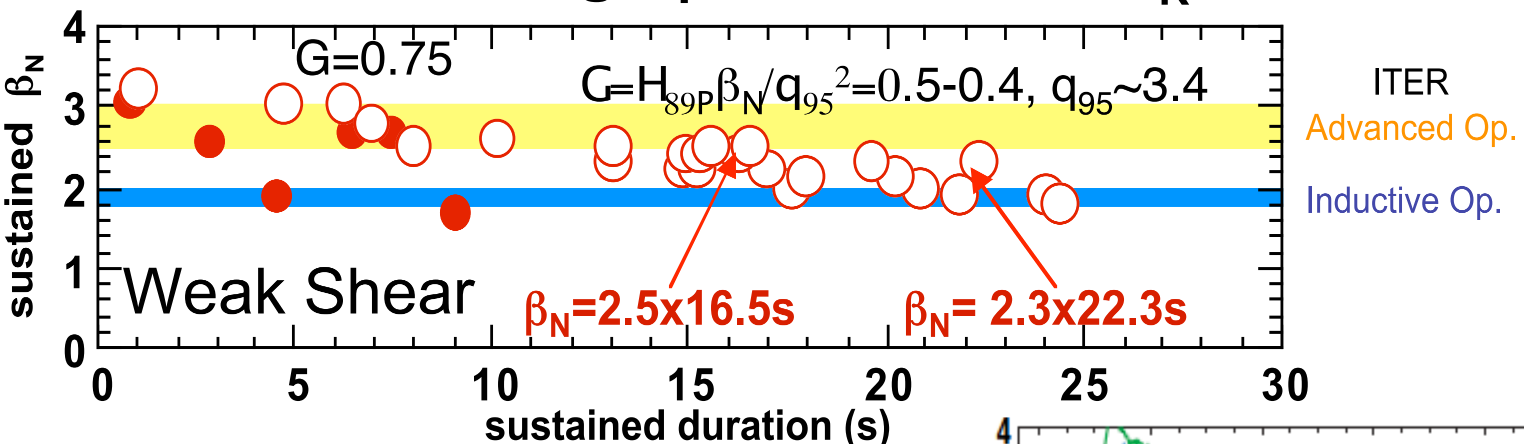
1.1 Long Pulse Operation: high β & G sustained>> τ_R

High β & AT (self regulating) regime > τ_R

Particle control > t_w



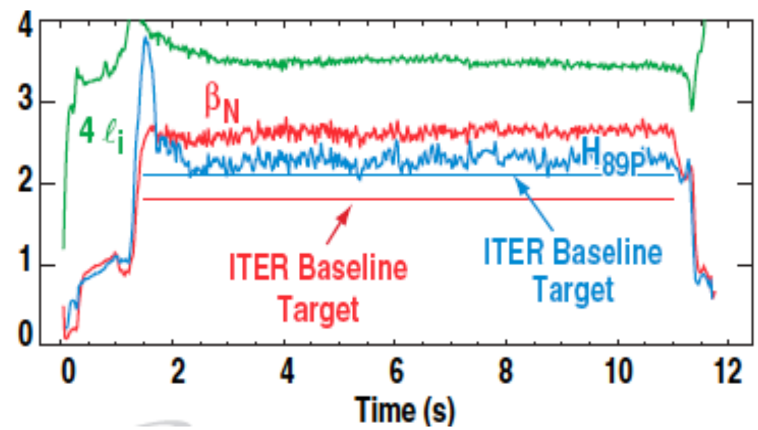
*JT-60U: extended high-β duration =13τ_R



*DIII-D: 9.5s ITER baseline scenar

~ 9τ_R, <β>=4%, G~0.55

*JET: 20s reversed shear



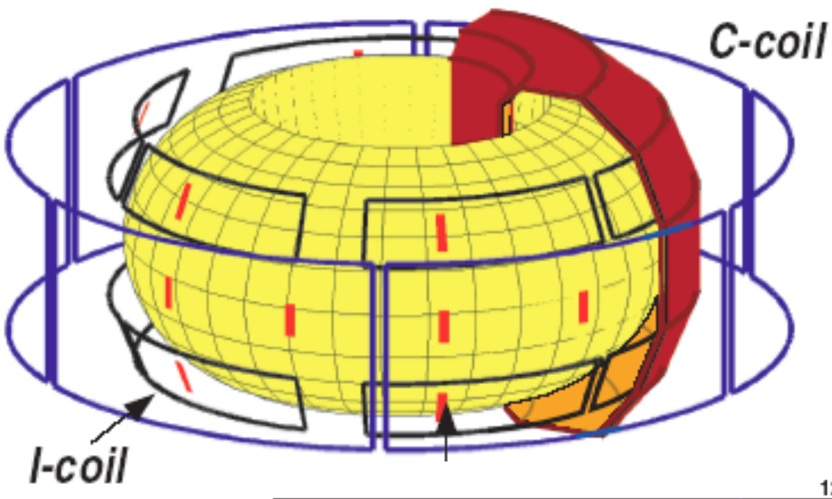
Effective Resistive Wall Mode Stabilization Is Important to Fusion Energy Using the Tokamak

- **Increasing β_N is the only way to simultaneously increase**
  - Fusion Power
  - Bootstrap Fraction

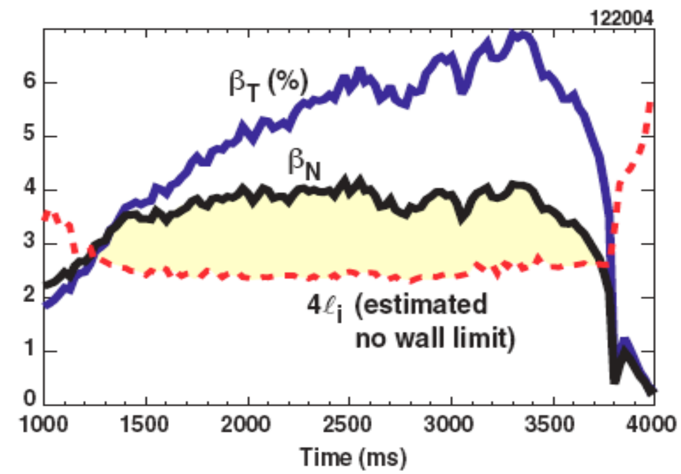
$$\beta_p \, \beta_T = 25 \left(\frac{1+\kappa^2}{2}\right)\left(\frac{\beta_N}{100}\right)^2$$

Fusion power $\beta_T^2 B^4$

Bootstrap fraction $c\varepsilon^{1/2}\beta_p$



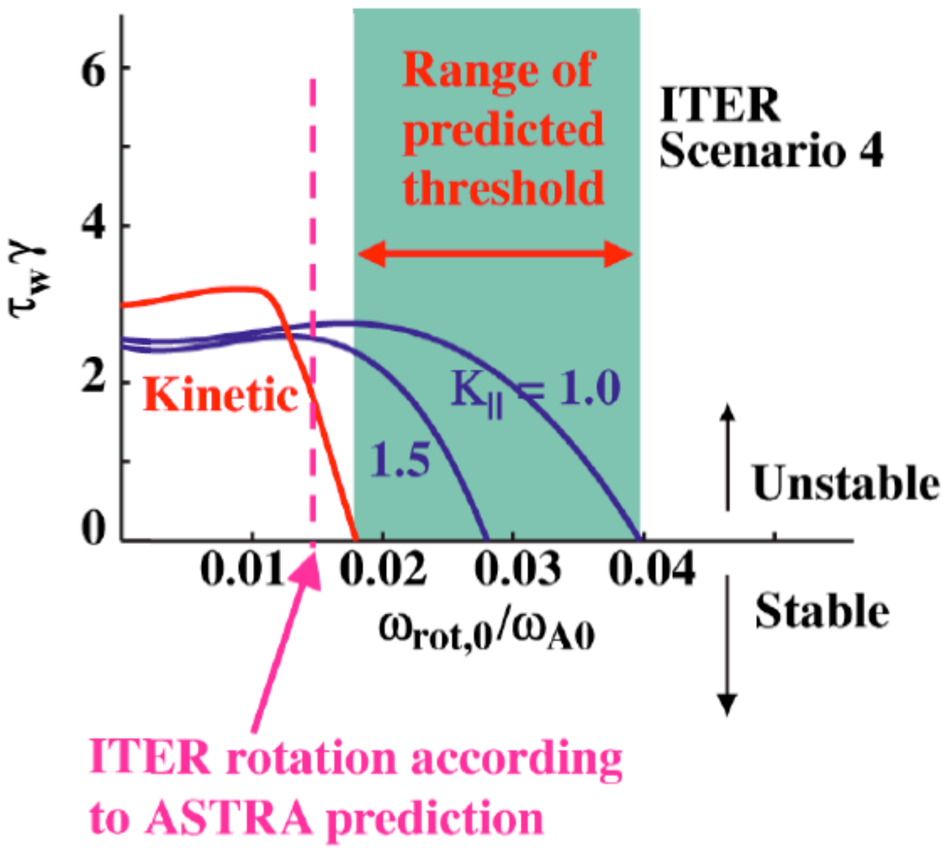
- **RWM stabilization to increase β_N requires**
  - either sufficient plasma rotation and a set of 10-50 Hz coils to counteract RFA (already proven)
  - Or a set of kHz bandwidth 2/1 helicity coils for direct feedback (work in progress)



[Garofalo APS 2005 invited]

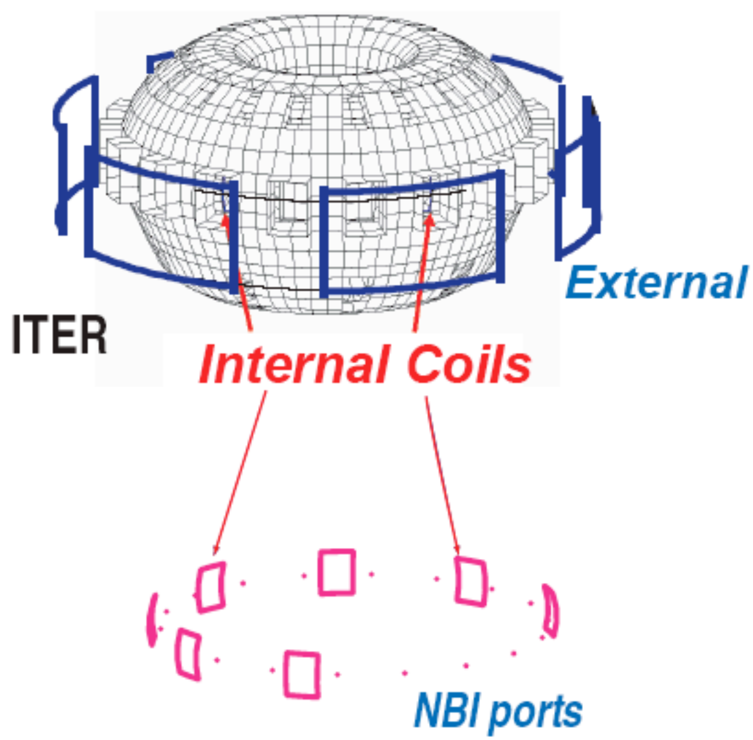
Increased Rotation in ITER May Be Necessary for Rotational Stabilization of the RWM

- **Factor two range of theory predictions of rotation theshold**
- **Rotation in ITER predicted by NBI torque balancing diffusive momentum loss**
  - χ_ϕ like χ_ι assumed
  - Plasmas with no momentum input rotate
- **Double the momentum input**
  - For NBI, $M/P = (2m/E)^{1/2}$
  - Third NBI Beamline at 250 keV
- **Rotation threshold may be lower than thought**
  - Garofalo, EX/7-1Ra

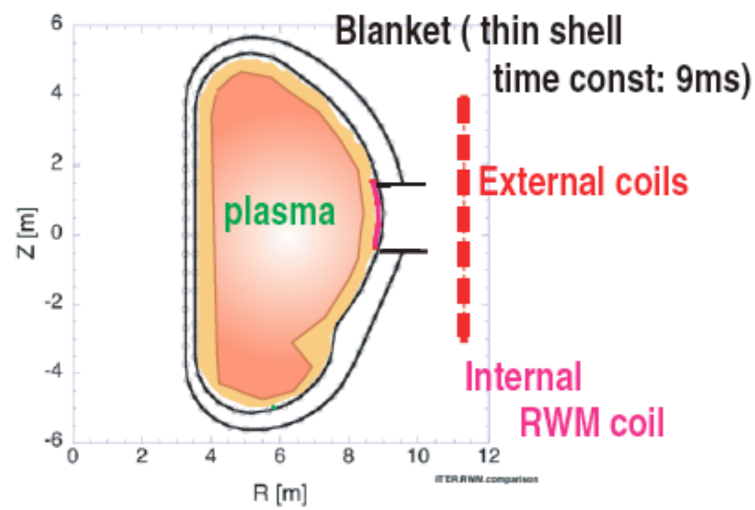


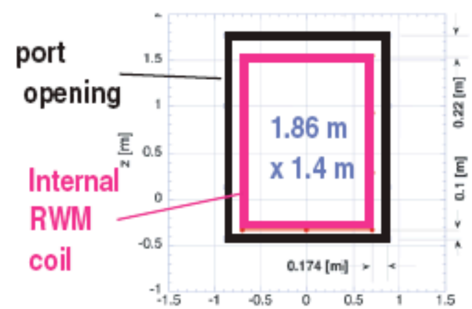
Y. Liu, et. al., Nucl. Fusion 44, 232 (2004)
A.R. Polevoi, FEC 2002 paper CT/P-08.

Schematic Layout of Possible RWM Coils on ITER



- RWM coils are located at every 40 degrees except NBI ports (toroidally asymmetric 7 coils)



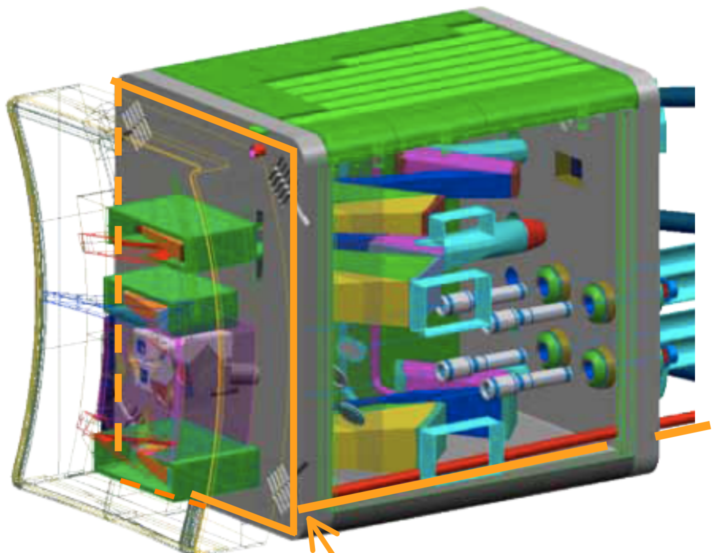


- The study analyses the effect of a single turn coil in the 10 cm gap between the blanket shield module (BSM) and the port extension of a mid-plane port plug.

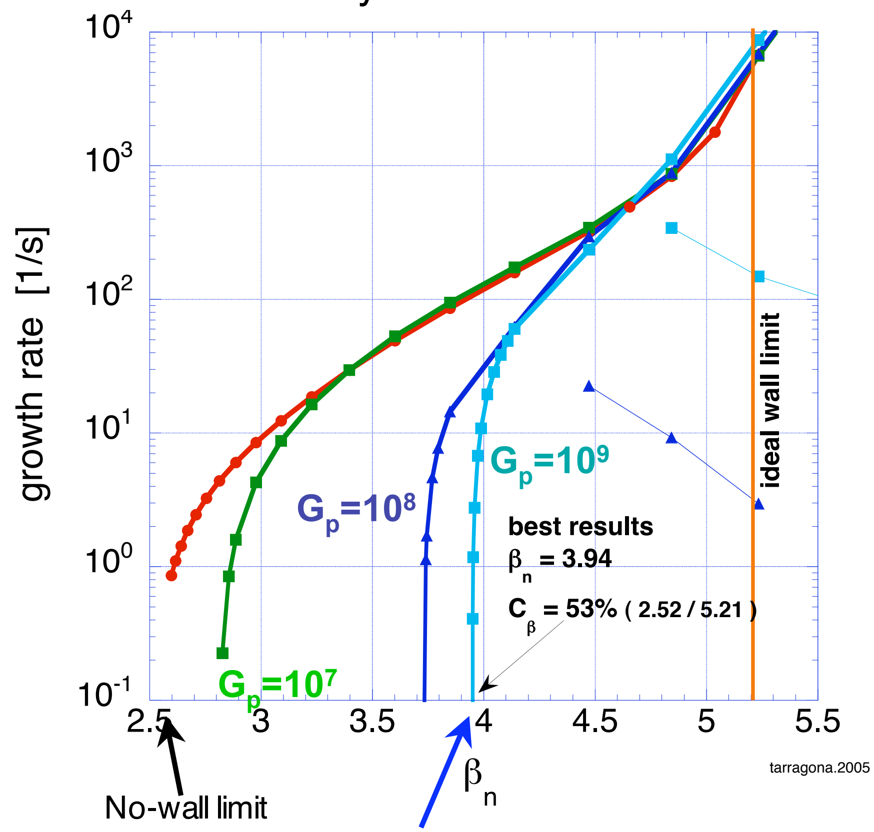
Columbia University

Applying Internal RWM Feedback Coils to the Port Plugs in ITER Increases β-limit for n = 1 from β_N = 2.5 to ~ 4

RWM Coil Concept for ITER



VALEN Analysis with Blanket Module



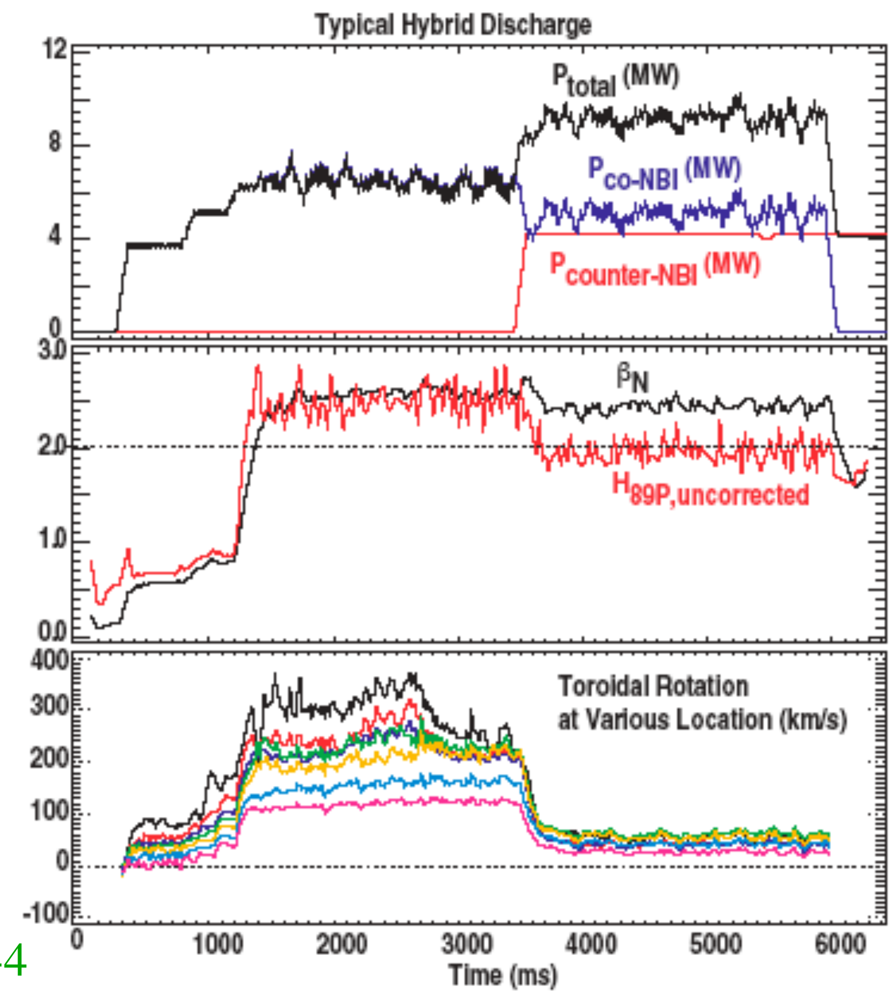
- Baseline RWM coils located outside TF coils
- **Internal RWM coils would be located inside the vacuum vessel behind shield module but inside the vacuum vessel on the removable port plugs.**

- **7 RWM Coils mounted behind Port Plug Blanket Module with simple proportional gain G_p feedback control loop**
- **Advanced Feedback stabilizes β_n > 4**

Columbia University

There Are Other Benefits to Increased Plasma Rotation

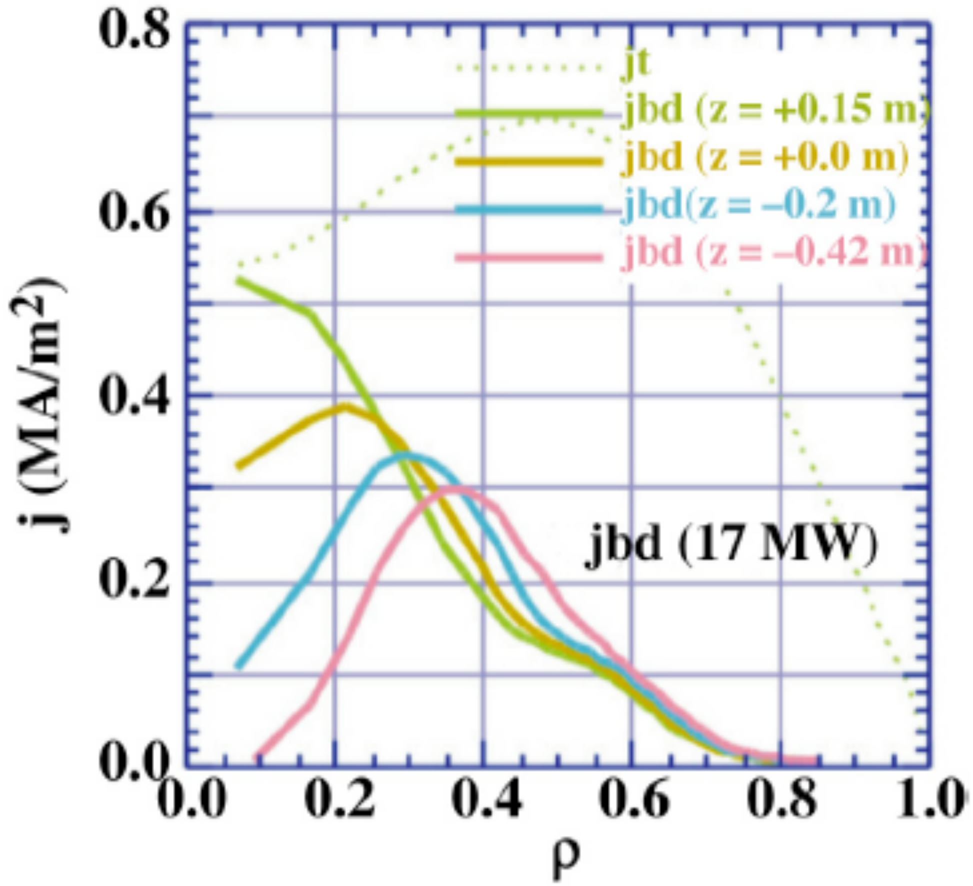
- **Rotation improves confinement**
  - Sheared ExB flow effect well known
- **Increased resilience to locked modes**
- **Lower β_N threshold for NTMs with no rotation**



Wade, OV/1-4

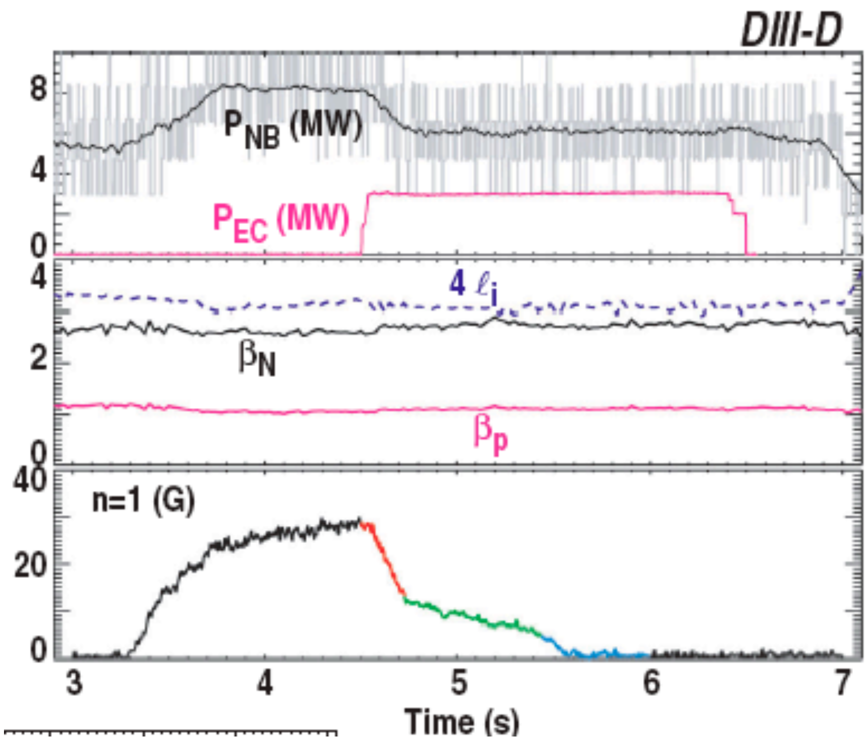
Auxiliary System Upgrades Should Focus on More Current Drive (off-axis)

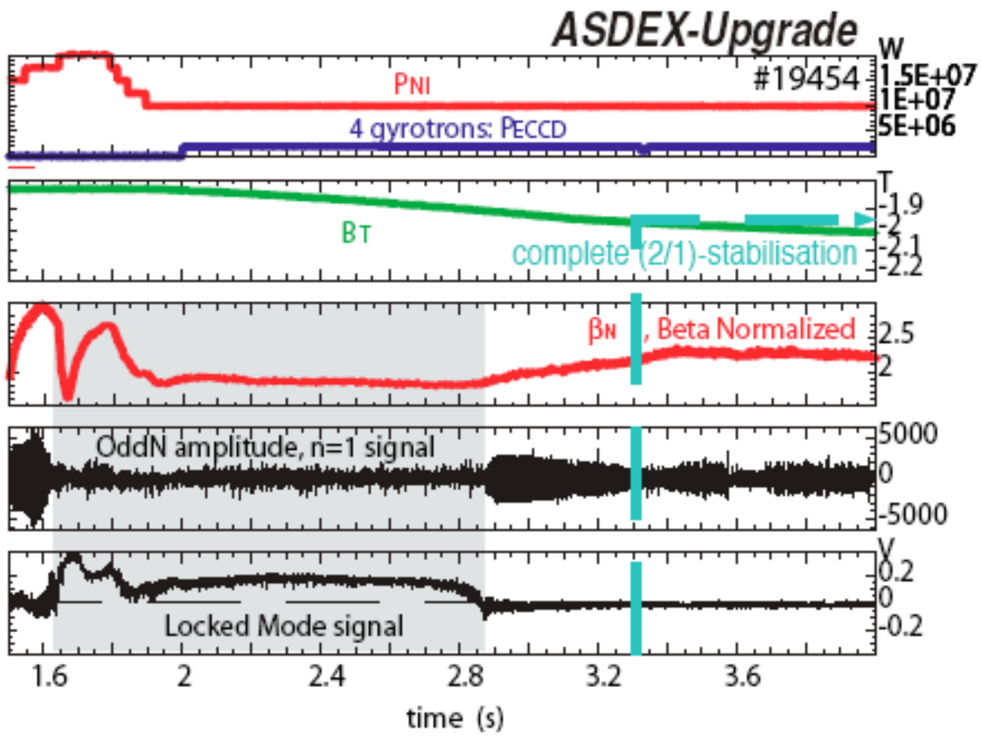
- **NBI - more rotation**
  - More β_N
  - More bootstrap current
  - Can NBI produce off-axis current drive?
- **More ECCD**
- **Add LHCD**
  - Need to show advanced modes, including H-mode, are compatible with an LH launcher closely coupled to the outer midplane plasma



NTM Stabilization by Continuous ECCD inside the Islands Has Been Established

- **Suppression of the 2/1 NTM in DIII-D and ASDEX Upgrade works**
  - Preemptive suppression in DIII-D
- **Similar results from JT-60U**
- **Modulated ECCD to limit the island size showing some promise in ASDEX Upgrade**

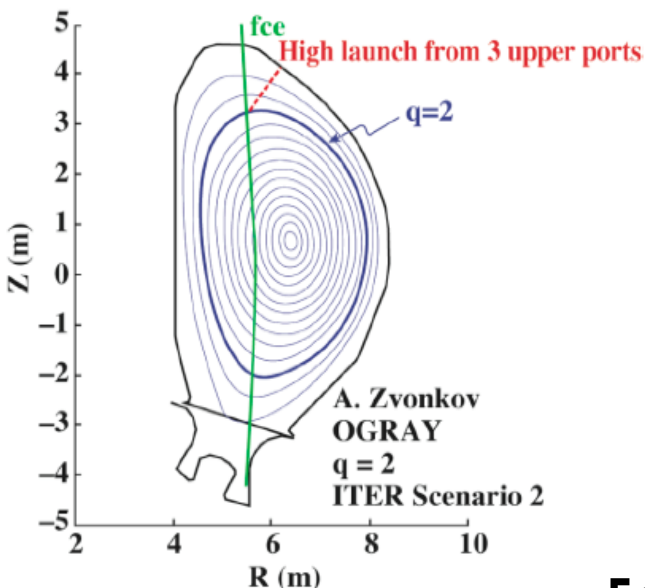


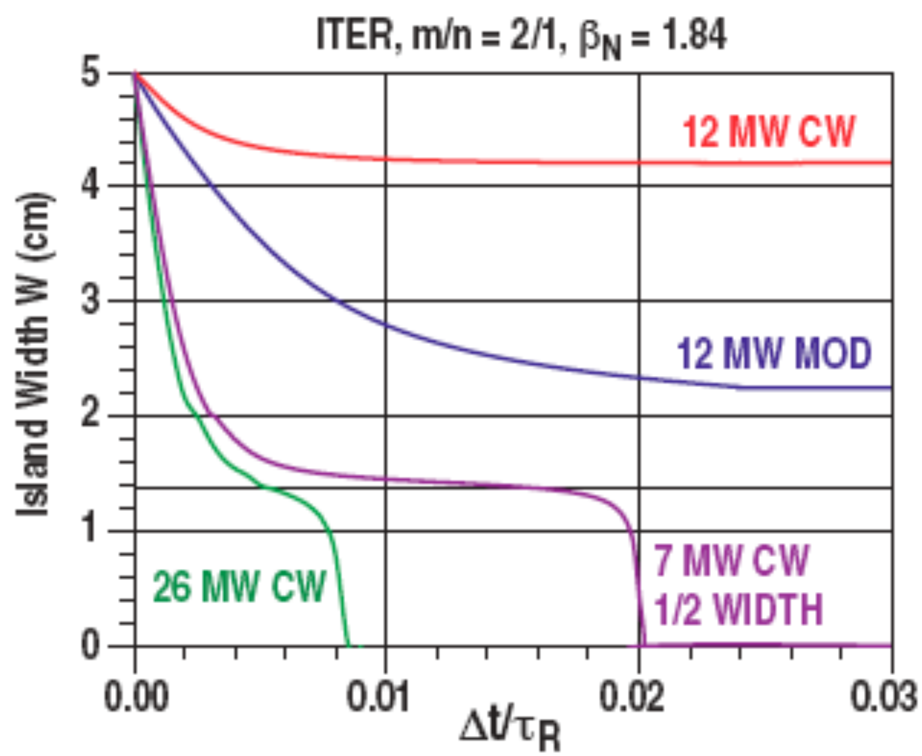


Excess EC Power Requirements Using Remote Steering Launcher May Be Solved by Front Steering

- **Far off midplane EC ports**
  - Broaden deposition
- **Remote steering launcher**
  - Inadequate focussing

R. J. LaHaye, et. al, Nucl. Fusion **46**, 451 (2006)



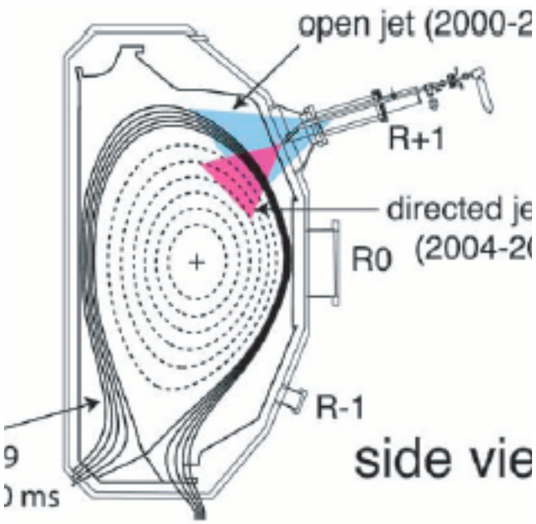


R.J. LaHaye, et. al., EX/P8-12
G. Saibene, et. al., IT/P2-14
M. A. Henderson, et. al., IT/P2-15

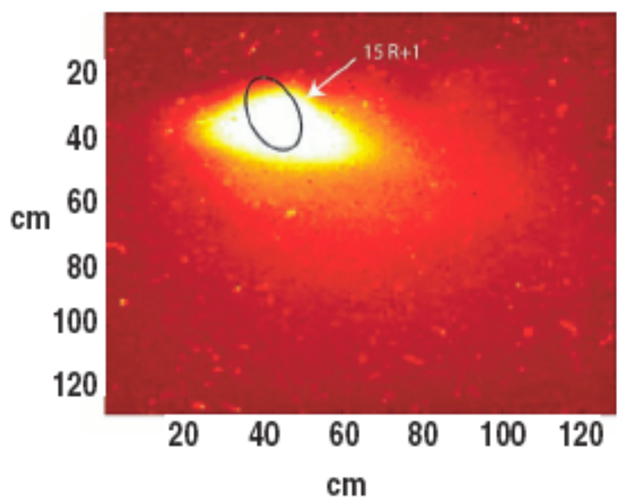
- **Front Steering Launcher**
  - Much better focussing
  - 3 MW modulated adequate for 2/1 mode
  - Must handle heat and force loads, neutrons

Massive Gas Injection Mitigates All Consequences of Disruptions

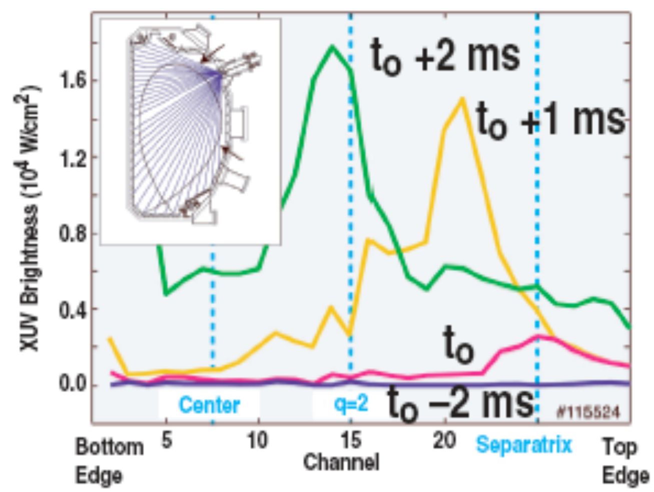
- **Essential Feature - Get Rosenbluth density of electrons (bound or free) into the plasma in a millisecond**
  - Substantially reduces forces from halo currents
  - Substantially reduces heat pulse to divertor
  - Quenches runaway electron production



- Fast framing camera shows neutral gas does not penetrate



- Fast bolometer measures impurity ion propagation

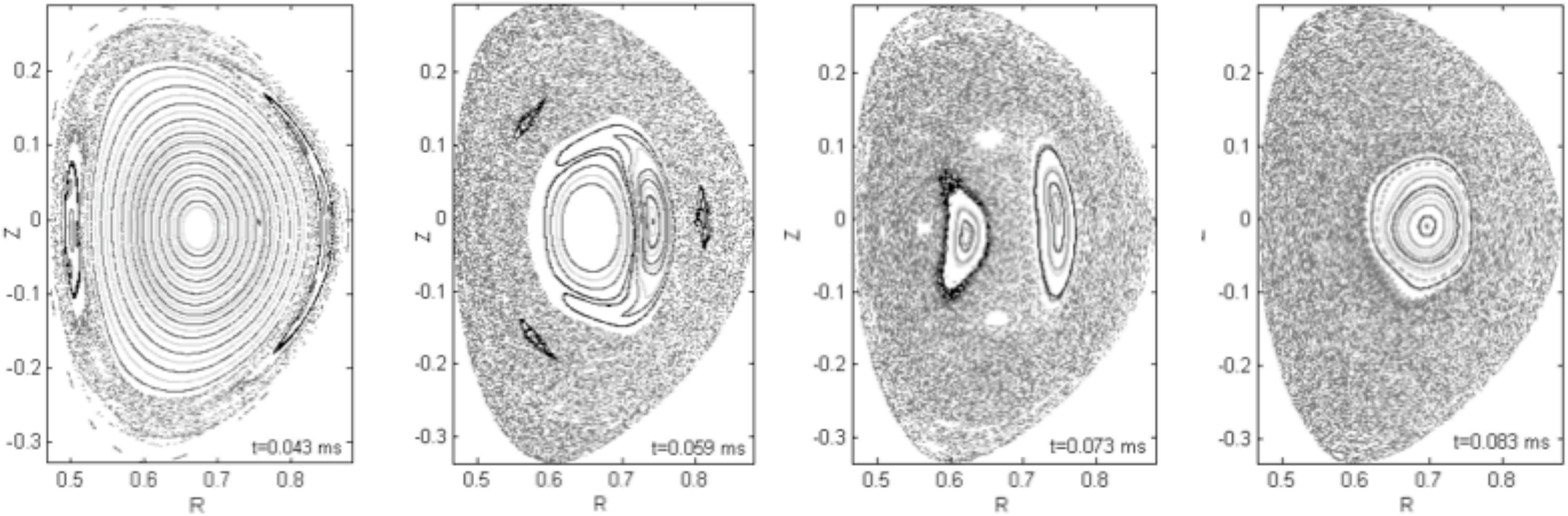


DIII-D NATIONAL FUSION FACILITY

MHD Mixing May Get Impurities in to the q=2 Surface Fast Enough

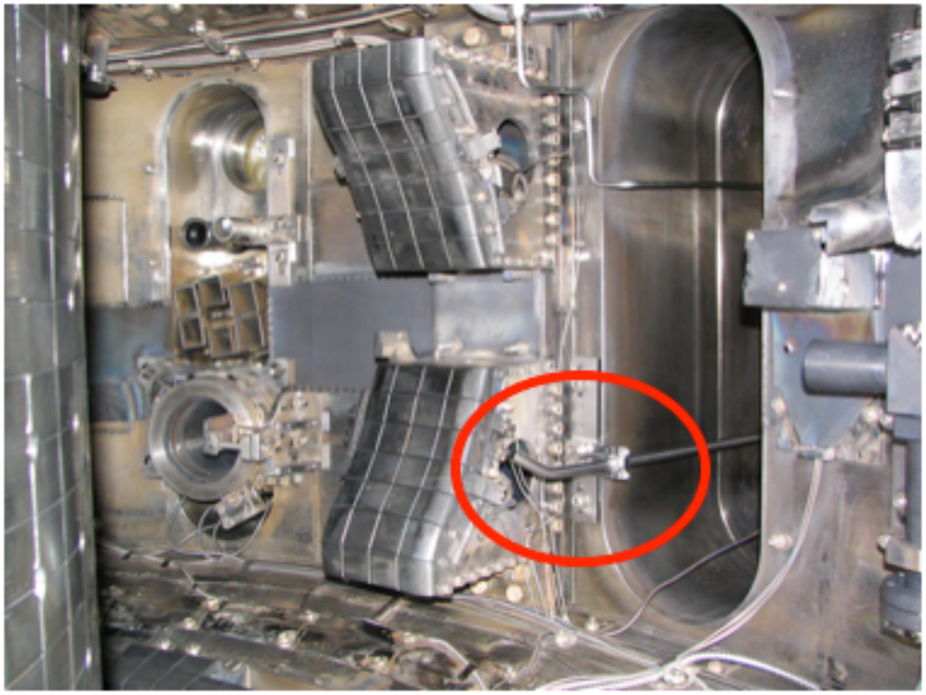
- **Nimrod code modeling of Alcator Cmod**
  - But how to project this process to ITER?

2/1, 1/1 modes cause large ergodic region: core cools, impurities mix in



Massive Gas Injection Mitigation of Disruptions Should Be Installed on ITER

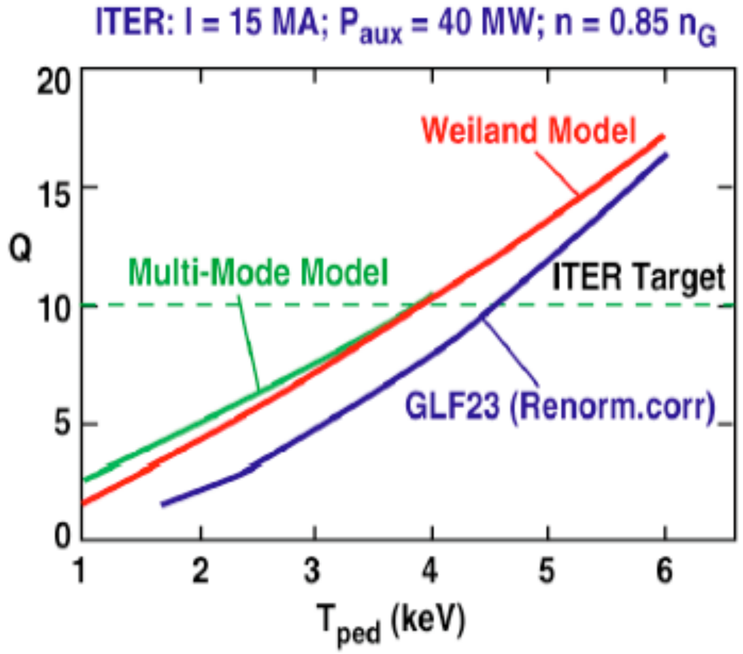
- **Simple, inexpensive system**
- **Probably will work**
- **R&D on liquid jet as backup**



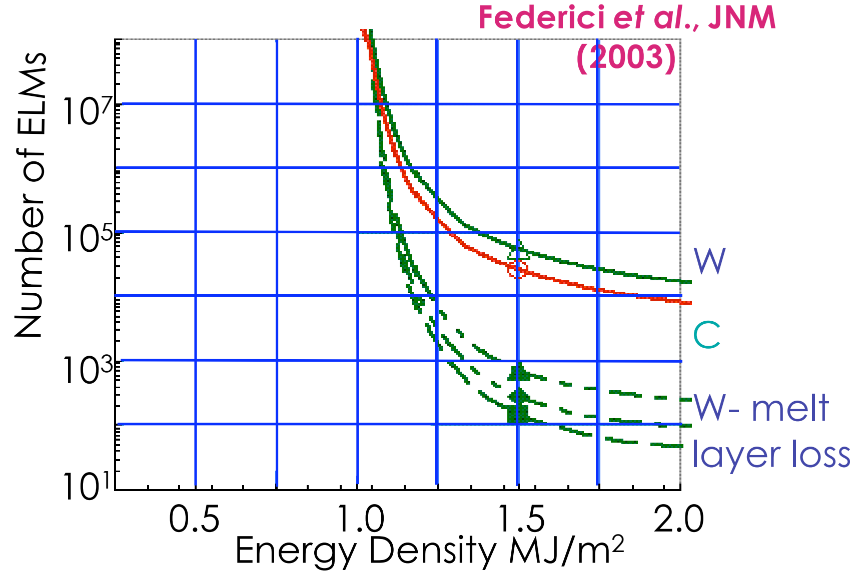
Injects 0.5-1.0×10^{23} atoms in a few ms at 70 bar (plasma inventory is 1.5-3.0×10^{20} D^{+}, e^{-})

Alcator C-Mod

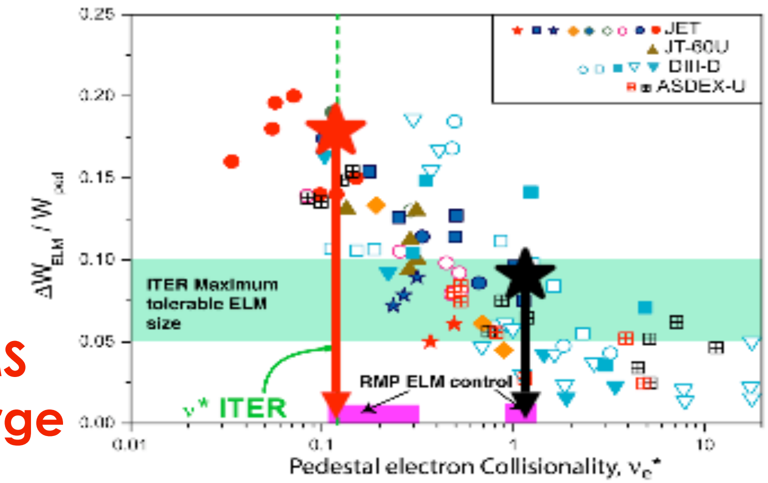
ELMs Must Be Reduced a Factor Two or Eliminated in ITER



- **Ablation threshold for heat pulses, 1.0 MJ/m²**



- **High Q requires high pedestal for stiff core transport model**

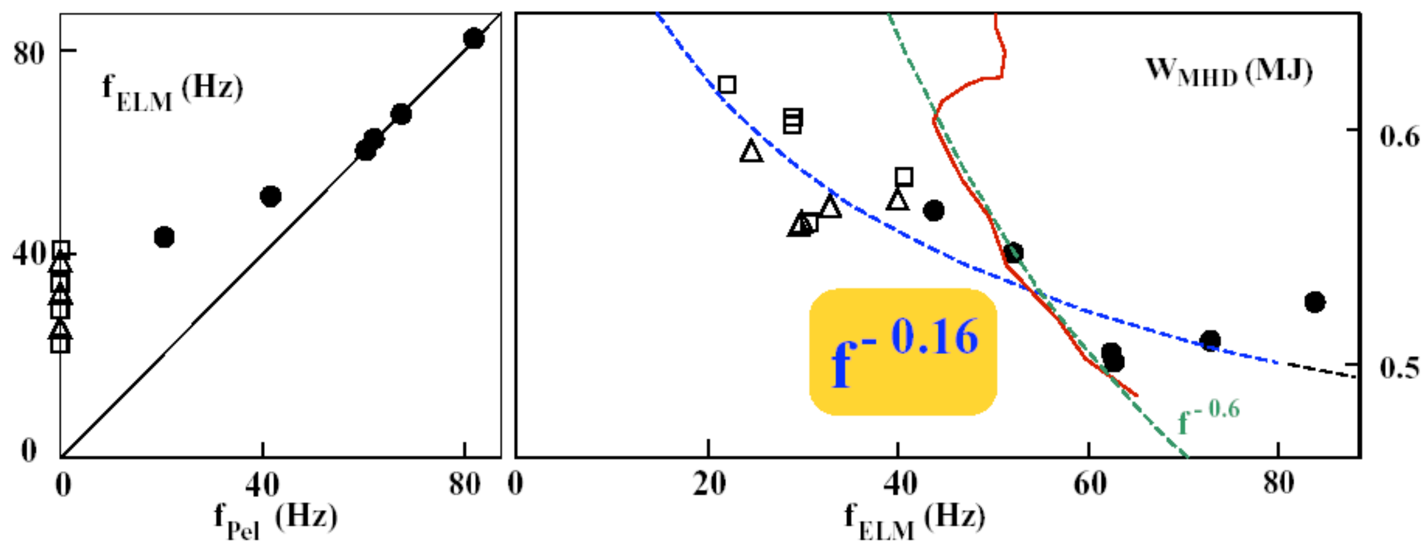


- **At ITER's collisionality, ELMS will be a factor two too large**

ELM CONTROL BY PELLET PACE MAKING

Replace linearly unstable peeling/ballooning mode by local trigger perturbation

OV/1-5 AUG Guenter



EX/2-6 AUG Lang

Pace making

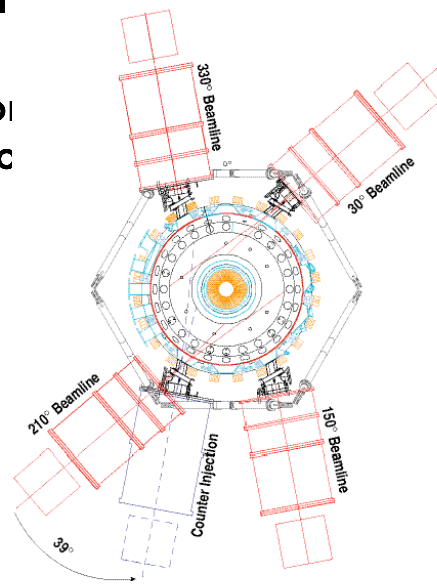
Performance

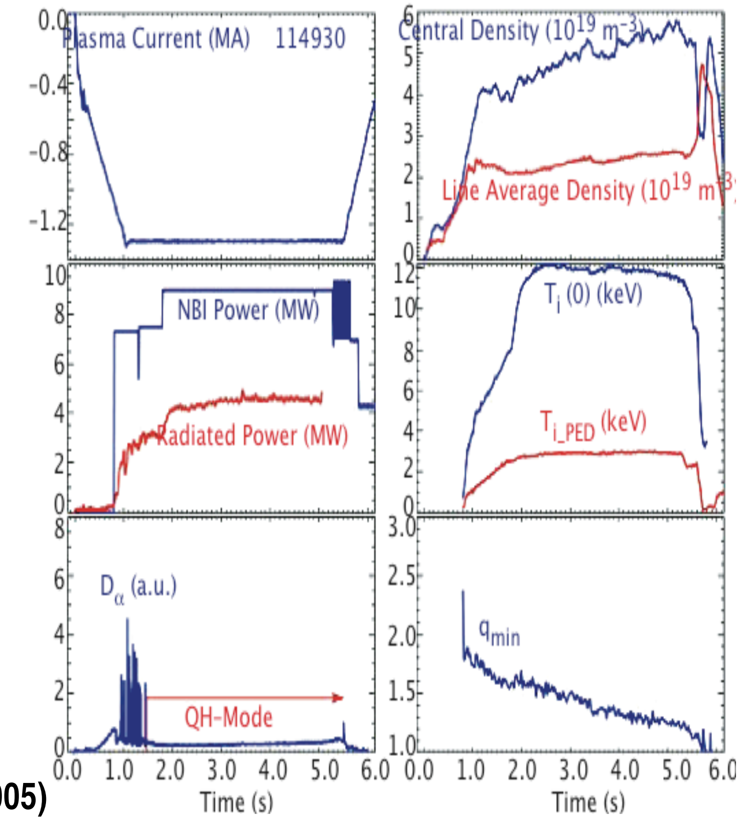
- only minor confinement degradation with increased ELM frequency compared to, e.g., gas puffing (pedestal temperature reduced!)
- energy loss per ELM for pellet triggered ELMs as for “natural” ELMs
- successful ELM control also by small wobbling

ELM Free QH–mode Discovered on DIII–D With Counter Neutral Beam Injection

- **Operating space includes pedestal β_T and ν^* comparable to ITER and:**
  - Upper and lower single null and double null shapes
  - With ion $B \times \nabla B$ drift away fro and toward the active X-po
  - High and low triangularity $0.15 < \delta_{avg} < 0.8$
  - Range of elongation $1.64 < \kappa < 2.1$
  - Range of safety factor $3.1 < q_{95} < 5.8$
  - Low to moderate n_e^{ped} $.07 < n_e^{ped}/n_{GW} < 0.48$
  - No clear power threshold: typically $P_{inj} > 5$ MW of counter beam injection required

SHOT	ν_i^*	β_{TPED}(%)	$H_{ITER-89P}$	Z_{EFF}	ν_e^*
115099	0.03	0.82	1.8	2.6	0.2
118820	0.07	0.97	1.9	2.0	0.2
99056	0.14	0.94	1.8	2.4	0.3
ITER	0.08	0.91	1.8	1.7	0.1



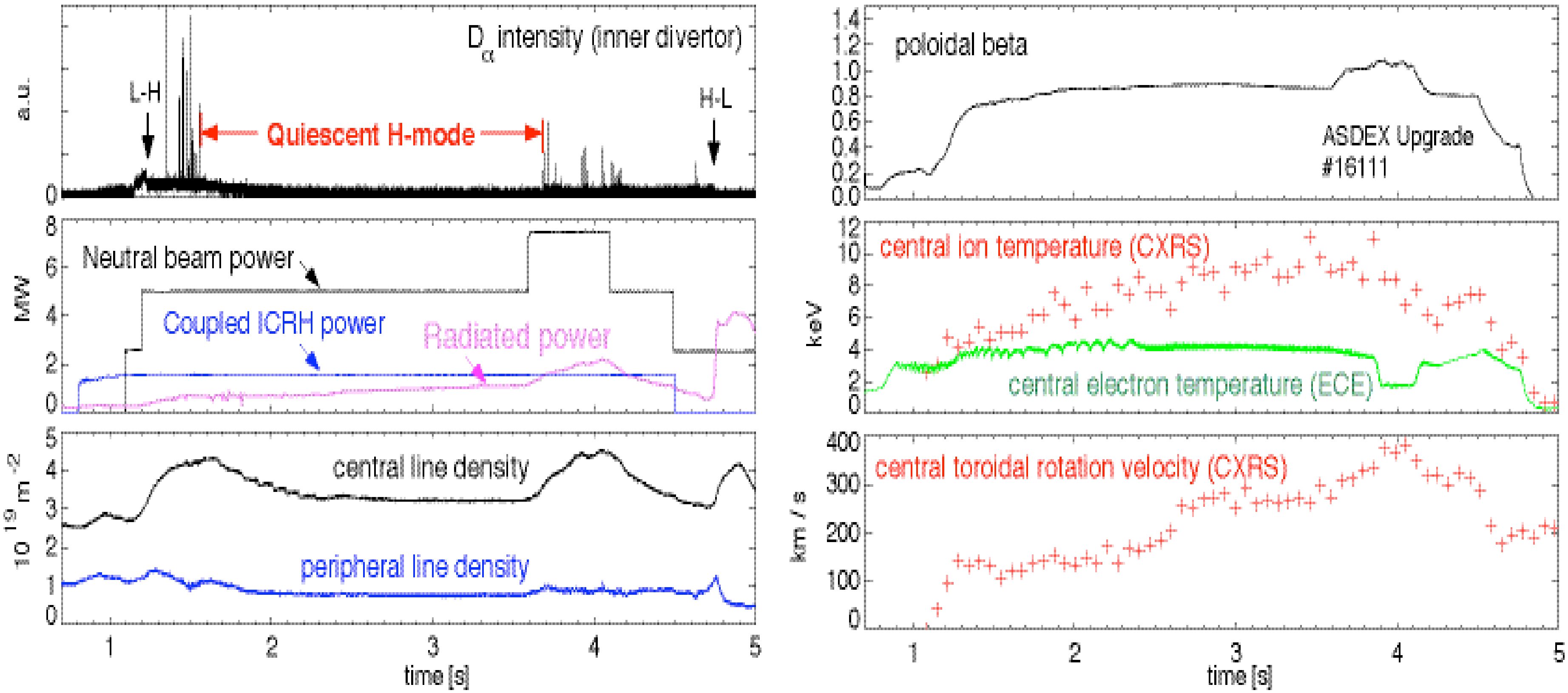


Greenfield, PRL, 86, 4544 (2001)
Burrell, Phys. Plas., 12, 056121 (2005)

DIII-D NATIONAL FUSION FACILITY

QH-MODE IN ASDEX UPGRADE

EX/1-4 AUG Suttrop

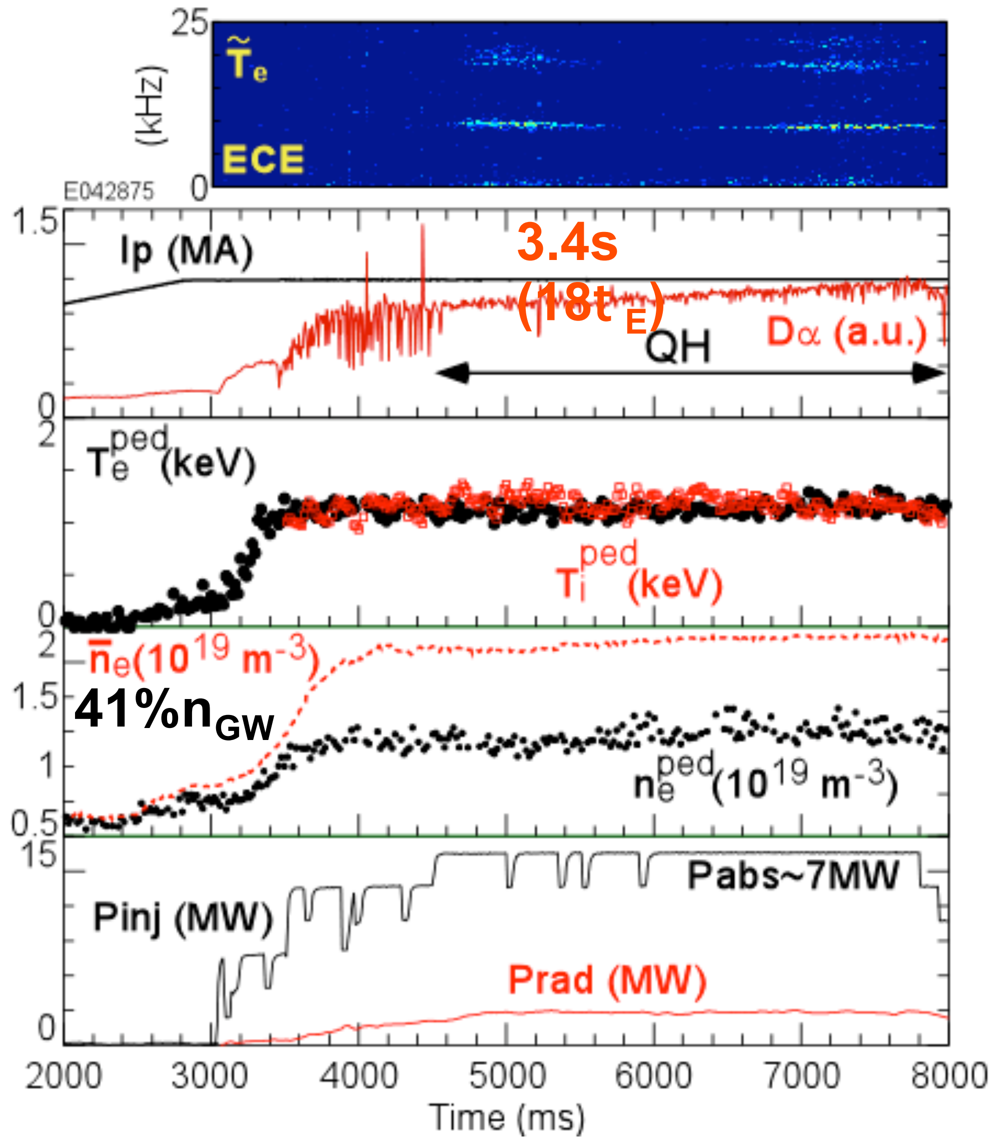


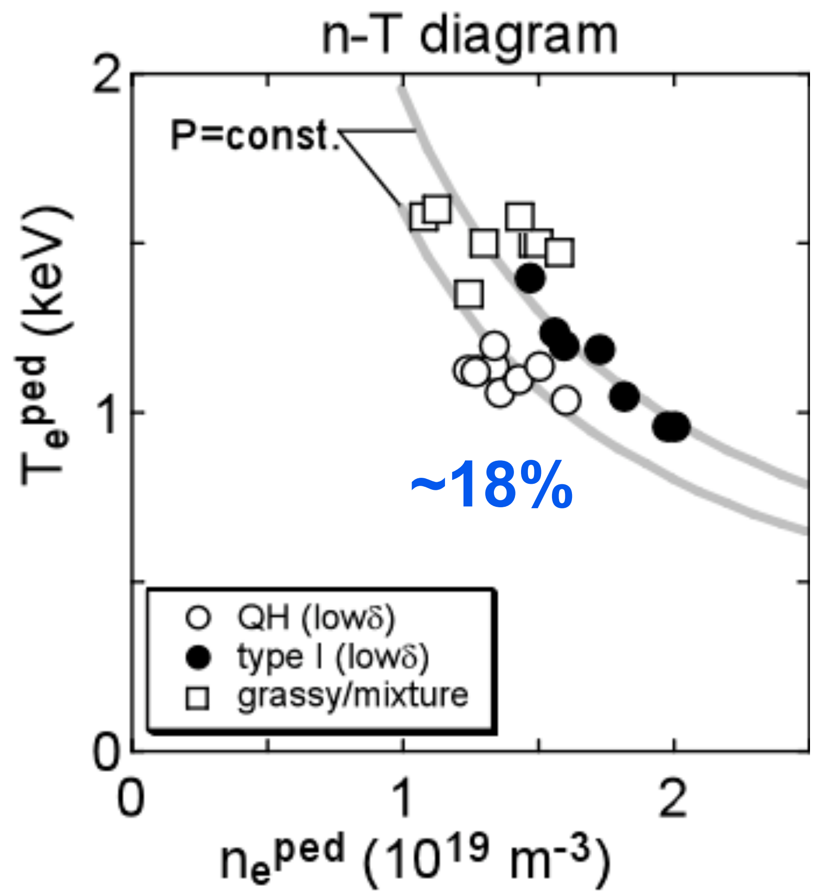
- Large E_R in the barrier, 2 × normal H-mode
- Energetic particle effects near the barrier
- EHO/HFO necessary features

QH-MODE IN JT-60U

EX/2-1 JT-60U Oyama

- **Pedestal parameters almost constant during QH phase**





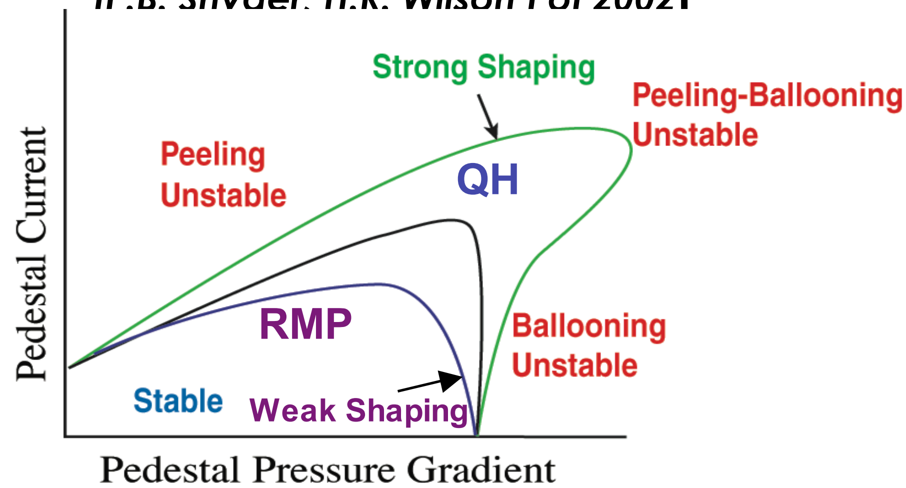
T_i^{ped} also smaller in QH phase

How to Implement QH-mode on ITER Requires ~ 2 More Years Research

- **If the physics is ion orbit loss charging the edge plasma negative (counter beam)**
  - add an 80 keV, 5 MW counter beam on ITER
  - Swing DNB to counter
- **If the physics is in the edge rotation,**
  - may be difficult to implement in ITER
- **DIII-D and JT-60U are now both equipped with the co/counter NBI to determine the physics**
  - JT-60U adds ripple loss effect

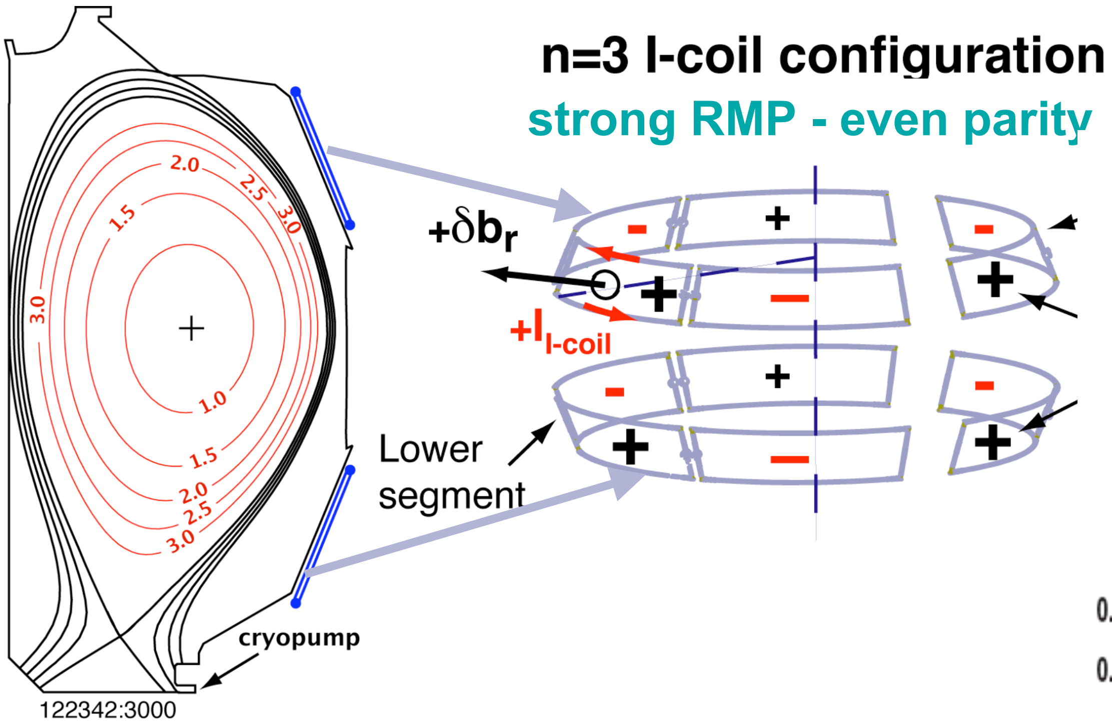
Schematic P-B Stability Diagram

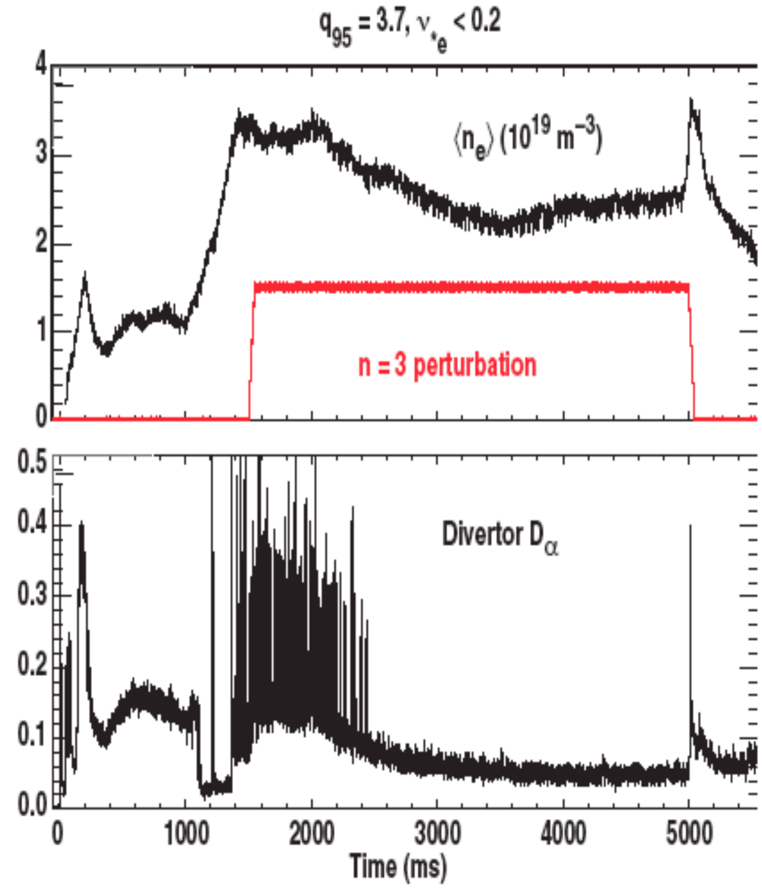
[P.B. Snyder, H.R. Wilson PoP2002]



Both QH-mode and RMP ELM Suppression consistent with Peeling/ballooning theory

Edge Localized Resonant Magnetic Perturbations Eliminate ELMs at ITER's Pedestal Collisionality





- **Flexible control of poloidal (m) spectrum with n=3, even and odd parity, toroidal phasing**
- **Pumping used to reduce ν^*_e initially in weakly shaped (low δ) plasmas and recently in high δ ITER similar shapes**

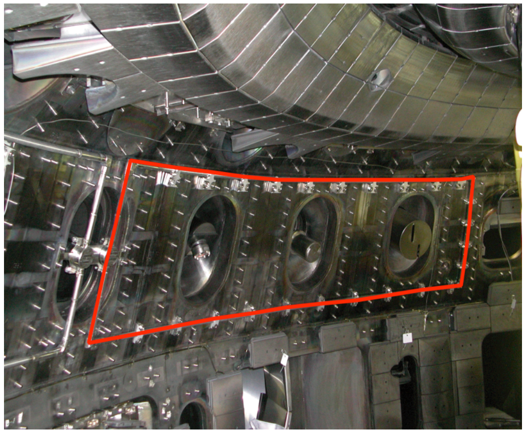
DIII-D
NATIONAL FUSION FACILITY

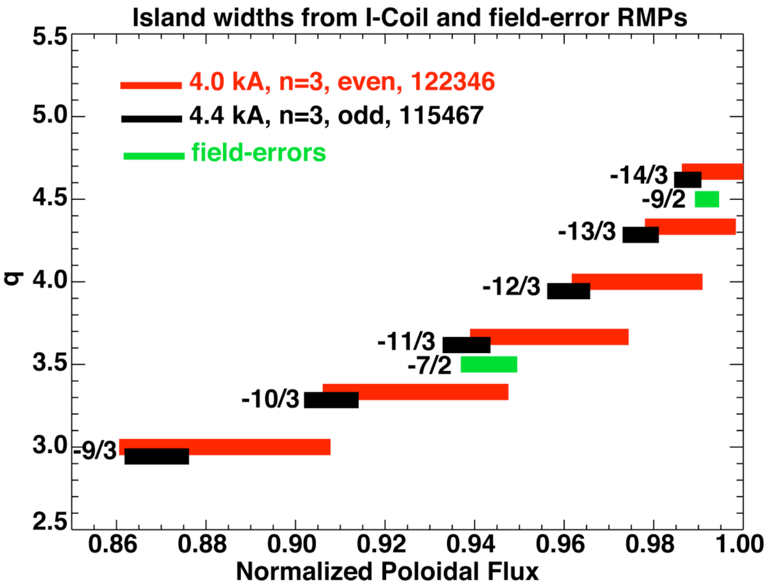
Application on ITER Desirable, but Hard to Find Space for the Required Coils

- **High order mode spectrum required**
  - n =3
  - m = 9-14
  - Localize perturbation to the edge

- **Coils can be inside or outside vacuum vessel (but close to plasma)**
  - May be hard to find space on ITER

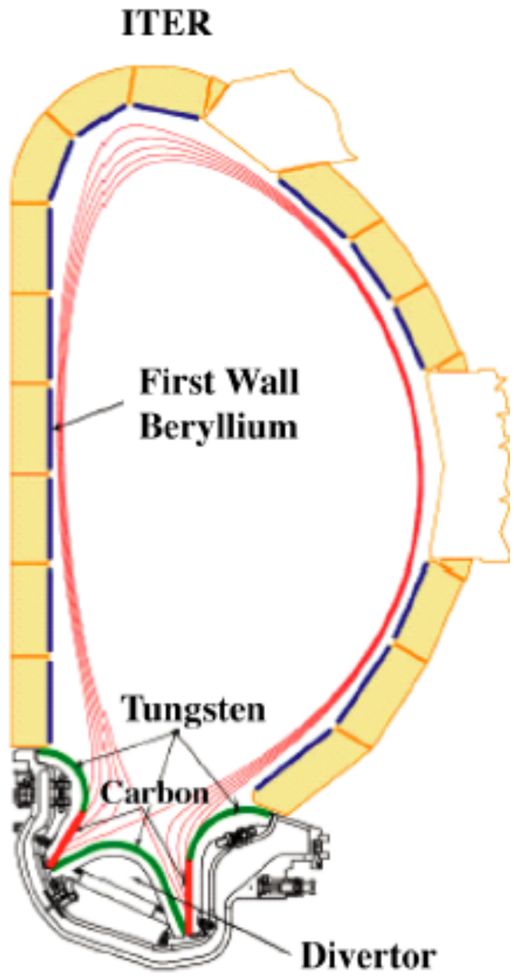
DIII-D internal coils before covering with graphite tiles





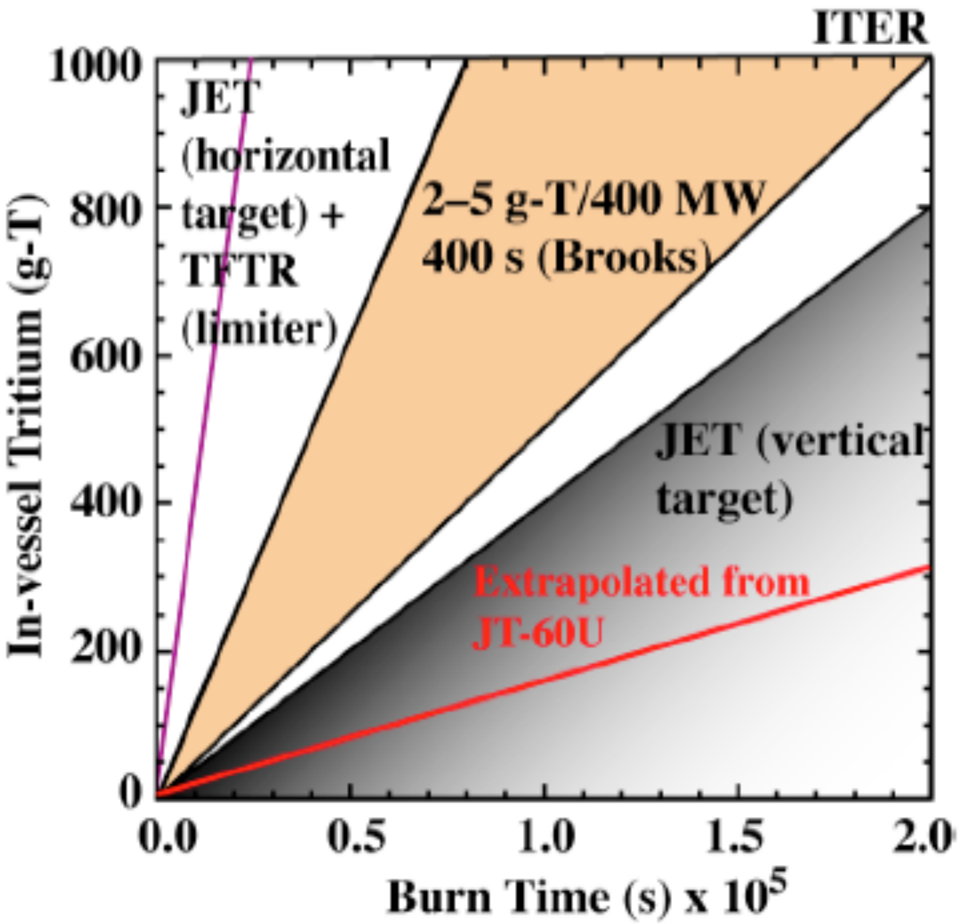
Present ITER Plan Uses a Mixture of Plasma Facing Materials

- **Carbon at Divertor Strike Points**
  - Can handle ELM, disruption heat pulses
  - Tokamak high performance basis on carbon
  - Tritium codeposition problem
- **Be main chamber first wall**
  - Plasma most vulnerable to main chamber wall materials, low Z most tolerable.
  - May melt in a mitigated disruption
- **Tungsten in lower heat flux divertor zones**
  - Low erosion in reactor
  - Melting, cracking
  - Plasma has very low tolerance for W
- **Mixed materials complicates situation**
  - Little known about mixed materials T retentior
  - Harder to understand what is going on
  - Coatings needed on metals (IT/1-4)but not carbon (OV/1-4)



Predictions of T Codeposition Range Widely 25 - 500 Full Power and Pulse Length Discharges

- **Recent JT-60U results encouraging**
  - Hot wall (300^{0}C) operatio
  - Excellent tile alignment
  - H/C deposition ratio 0.04
- **DIII-D hot samples and mirrors**
  - Little to no deposition (200^{0}C), even in gaps
- **Oxygen baking to remove codeposits must be developed**



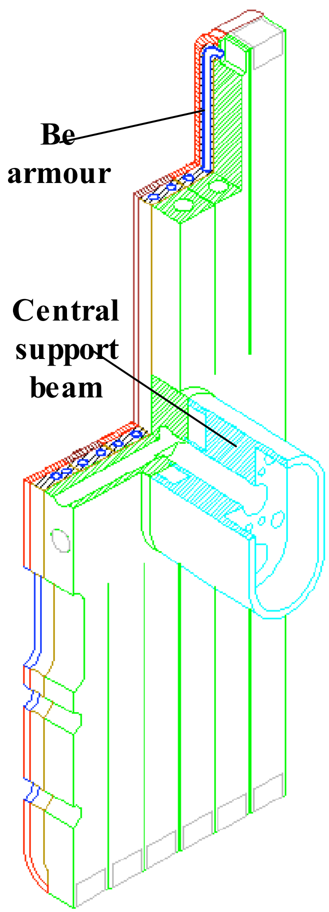
ITER's Research Plan Should Envision Two Major Changeouts of Plasma Facing Materials

- **Materials best for early objectives may not be best for DEMO transition**
  - Achieve 500 MW, Q=10, 400 seconds
  - Run very long pulse, advanced steady-state capable plasmas for fluence and handoff to DEMO

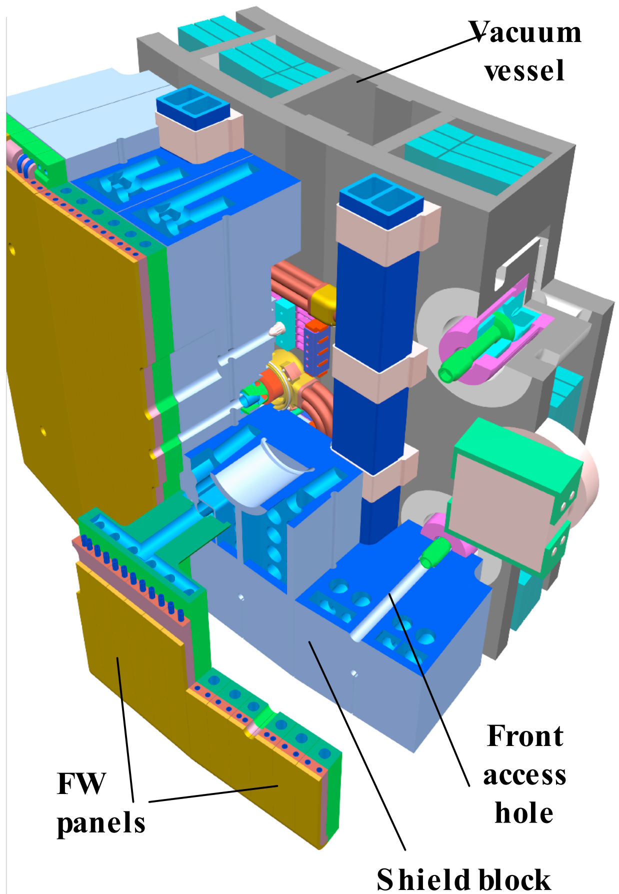
- **The world has the required complement of tokamaks to perform the required research, but time is needed**
  - DIII-D and JT-60U - all carbon and hot walls (JT-60U)
  - JET plans Be and W - all metal interior
  - ASDEX Upgrade - all W interior in 2007
  - Alcator Cmod - All Mo interior
  - EAST and KSTAR are coming
  - Side laboratory materials studies are vital

- **ITER's current first wall design provides for changeout of plasma facing materials**
  - But remote maintenance equipment should speed up the process from two years to less than one year

ITER FW/Shield Blanket Design



FW panel (back-side view)



Blanket module

Approximately 1800 FW panels for 440 blanket modules.
45 different types of blanket module.

Table 1 ITER FW/Blanket Main Parameters

Parameter	Value
Number of modules	440
First wall surface area	680 m^2
Total Weight of modules	1,610 t
Weight limit for module	4.5 t/module
Typical blanket module dimension (Inboard equator)	1415x1005x450 mm
Nominal gap between modules in the toroidal direction	14 mm Straight Inboard 20 mm Top and Outboard
Nominal gap between modules in the poloidal direction	10mm Straight Inboard 20 mm Top and Outboard
FW cooling tube inner diameter	10 mm
Approximate area of blanket side and back walls (for safety analyses)	1956 m^2 (~ 3 time area of FW)

ITER Will Make Many Unique Research Contributions

- **ITER will make the dominant contribution to alpha physics**
  - Adequacy of alpha diagnostics is a concern

- **Only in ITER can we learn about transport in the reactor regime of low collisionality and gyro-radius/system size**
  - Adequacy of turbulence diagnostics is a concern

- **Only in ITER can we learn the physics of the edge, SOL, and divertor at high absolute densities and simultaneous low collisionality**

Some Issues Applicable to Any of the ITER Scenarios Should Be Addressed in Current Experiments

- **Confinement scaling with beta should be resolved**
  - ITER98Y2 has $\beta^{-0.9}$, discourages advanced high beta scenarios
  - Dimensionless parameter experiments see $\beta^{0.0}$
- **L-H transition physics needs renewed fundamental work**
  - Perhaps informed by core transport barrier discoveries
  - Promising model - Equilibrium sheared ExB flow puts the plasma close to transition, then a zonal flow surge puts it over the top
- **Innovative approaches to central fueling needed**
  - Even inside launch pellets only penetrate one-third the way in owing to pellet speed limitations in curved guide tubes.

(See Perkins, IC/P7-15)

ITER Will Be the Landmark Experiment in Fusion, With Great Upside Potential and Technical Reach

- **Current research must support and the design must take into account:**
  - Ability to change plasma facing materials and the choices appropriate to ITER's sequenced mission objectives
  - Diagnostics for alpha particles and turbulence

- **ITER Design needs to take into account recent research advances in:**
  - Resistive Wall Mode (RWM)stabilization
  - Neoclassical Tearing Mode (NTM)stabilization
  - Edge Localized Mode (ELM) stabilization
  - Disruption mitigation
  - Advanced performance modes with special startup and off-axis current drive requirements

- **ITER research program will make unique contributions to:**
  - Alpha physics
  - Confinement in the reactor regime of dimensionless parameters
  - Divertor physics at high absolute density and low collisionality
  - Study of the complex feedback loops involving the current profile, bootstrap current, alpha heating, transport barriers, instabilities, etc. in high performance, burning, steady-state plasmas