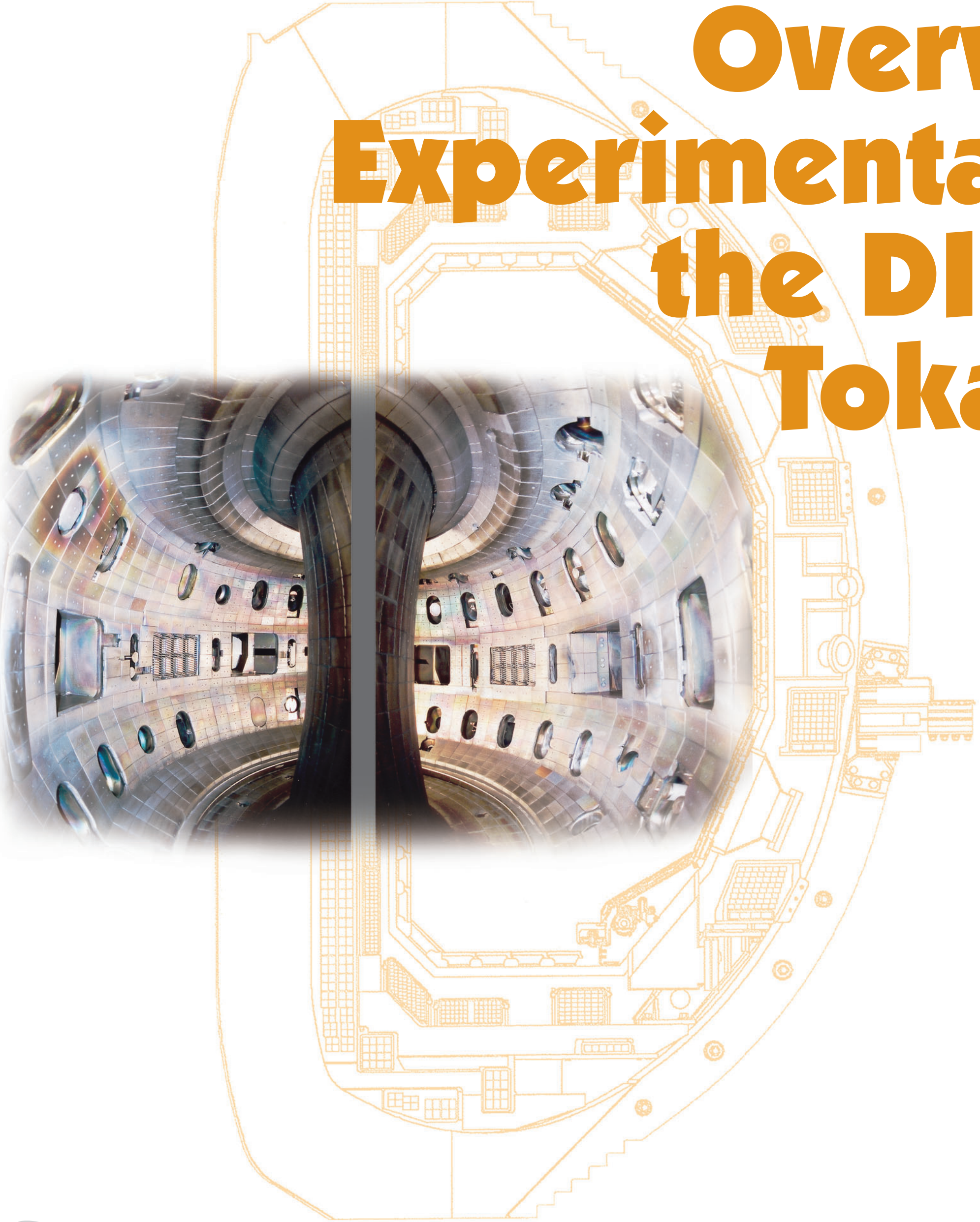


Overview of Recent Experimental Results from the DIII-D Advanced Tokamak Program

by
K.H. Burrell

for the DIII–D Team

Presented at
19th IAEA Fusion Energy Conference
Lyon, France

October 14–19, 2002

DIII–D
NATIONAL FUSION FACILITY
SAN DIEGO

255–02/KHB/wj

GENERAL ATOMICS

DIII–D ADVANCED TOKAMAK PROGRAM GOAL

- The DIII–D AT research program is developing the scientific basis for advanced operating modes in order to enhance the attractiveness of the tokamak as an energy producing system

- This requires optimizing for
  - High power density (high $\beta = 2\,\mu_o \langle p \rangle / B^2$)
  - High ignition margin (high energy confinement time τ_E)
  - Steady-state operation with low recirculating power (high bootstrap fraction)

- Key issues in optimization are
  - Active MHD stability control
  - Current profile control
  - Pressure profile control

DIII–D
NATIONAL FUSION FACILITY
SAN DIEGO

SUBSTANTIAL PROGRESS SINCE IAEA 2000

We have made progress since the last IAEA meeting in developing the building blocks needed for advanced operating modes:

- Substantially broadened the MHD stable operating space
  - Rotational stabilization of resistive wall modes yielding $\beta_N = \beta_N$ (ideal wall) $\cong 2\ \beta_N$ (no wall)
  - Increased β by 60% via stabilization of (3,2) neoclassical tearing mode with ECCD in sawtoothing plasmas
  - First stabilization of (2,1) neoclassical tearing mode using ECCD
- Developed plasma control tools
  - First integrated AT discharges with current profile control using ECCD
  - Pressure and density profile control with ECH and ECCD
- Demonstrated an improved, high q_{95} (>4) operating scenario for ITER
- Achieved solutions to key burning plasma issues
  - No ELM-produced, pulsed divertor heat load in QH–mode plasmas
  - Small heat and particle loads at inner divertor strike points in balanced double-null divertors
  - Disruption mitigation via massive gas puff

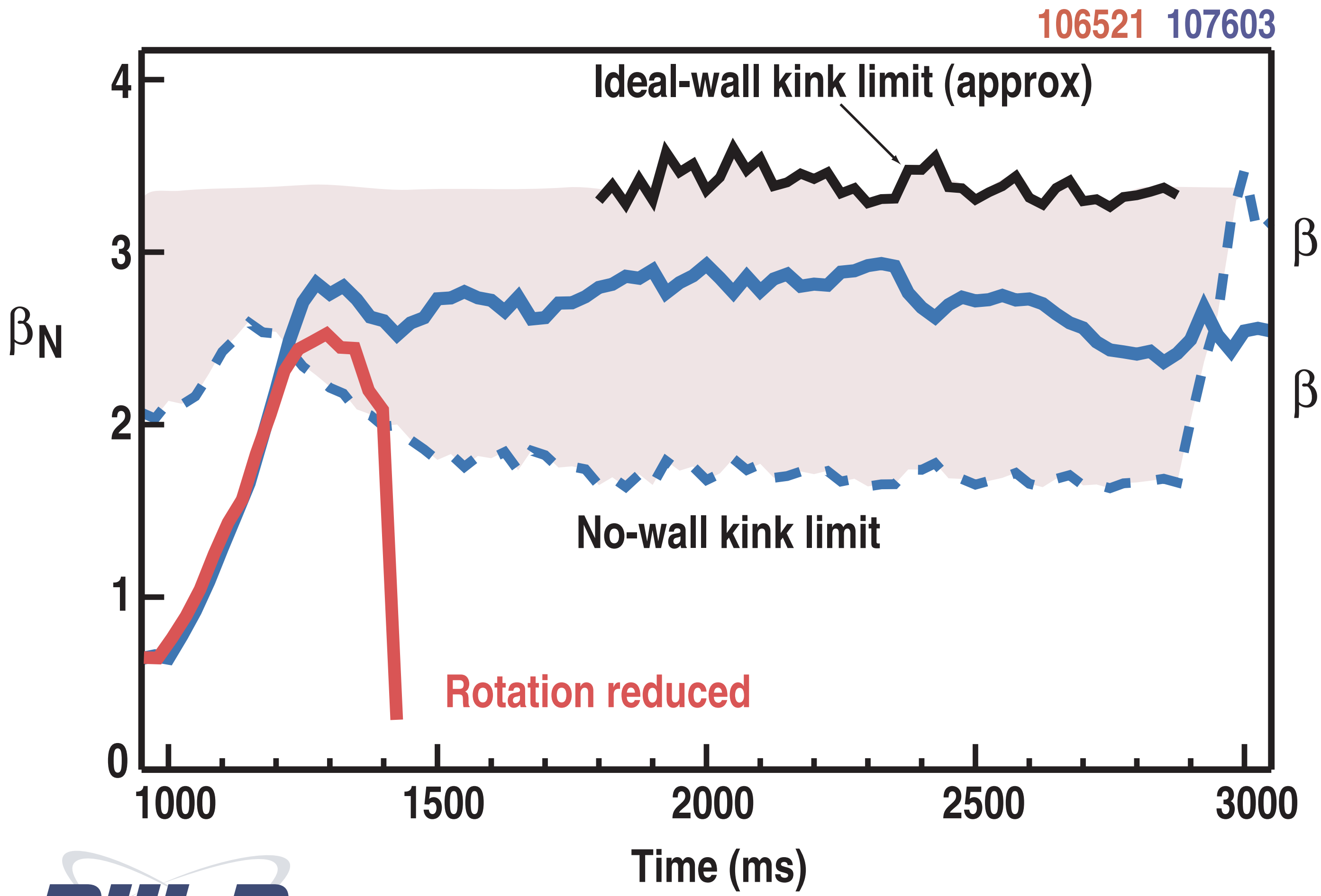
SUBSTANTIAL PROGRESS SINCE IAEA 2000

We have made progress since the last IAEA meeting in developing the building blocks needed for advanced operating modes:

- Substantially broadened the MHD stable operating space
  - Rotational stabilization of resistive wall modes yielding $\beta_N = \beta_N$ (ideal wall) ≅ 2 β_N (no wall)
  - Increased β by 60% via stabilization of (3,2) neoclassical tearing mode with ECCD in sawtoothing plasmas
  - First stabilization of (2,1) neoclassical tearing mode using ECCD
- Developed plasma control tools
  - First integrated AT discharges with current profile control using ECCD
  - Pressure and density profile control with ECH and ECCD
- Demonstrated an improved, high q_{95} (>4) operating scenario for ITER
- Achieved solutions to key burning plasma issues
  - No ELM-produced, pulsed divertor heat load in QH–mode plasmas
  - Small heat and particle loads at inner divertor strike points in balanced double-null divertors
  - Disruption mitigation via massive gas puff

MHD STABLE TOKAMAK OPERATING SPACE APPROXIMATELY DOUBLED BY SUPPRESSION OF EXTERNAL KINK INSTABILITY

- Key: sustainment of plasma rotation
  - Theoretically predicted (Bondeson and Ward, 1994)



$$\beta = \frac{\text{plasma pressure}}{\text{magnetic pressure}}$$

$$\beta_N = \frac{\beta}{I/aB_T}$$

tokamak stability parameter

DIII-D
NATIONAL FUSION FACILITY
SAN DIEGO

WALL STABILIZATION OF EXTERNAL KINK VIA PLASMA ROTATION BROADENS OPERATING SPACE

- Wall stabilization of the external kink is possible via stabilization of the resistive wall mode (RWM) by plasma rotation
  - Duration in previous experiments limited by the slowing of plasma rotation
- New Discovery: Rotation slowing at β above the no-wall limit is a consequence of "resonant field amplification" (RFA) [A. Boozer, Phys. Rev. Lett. 86 (2001)]
- New Discovery: Reduction of the non-axisymmetric (error) fields enables continued plasma rotation at β above the no-wall limit

⇒ Reduced error field

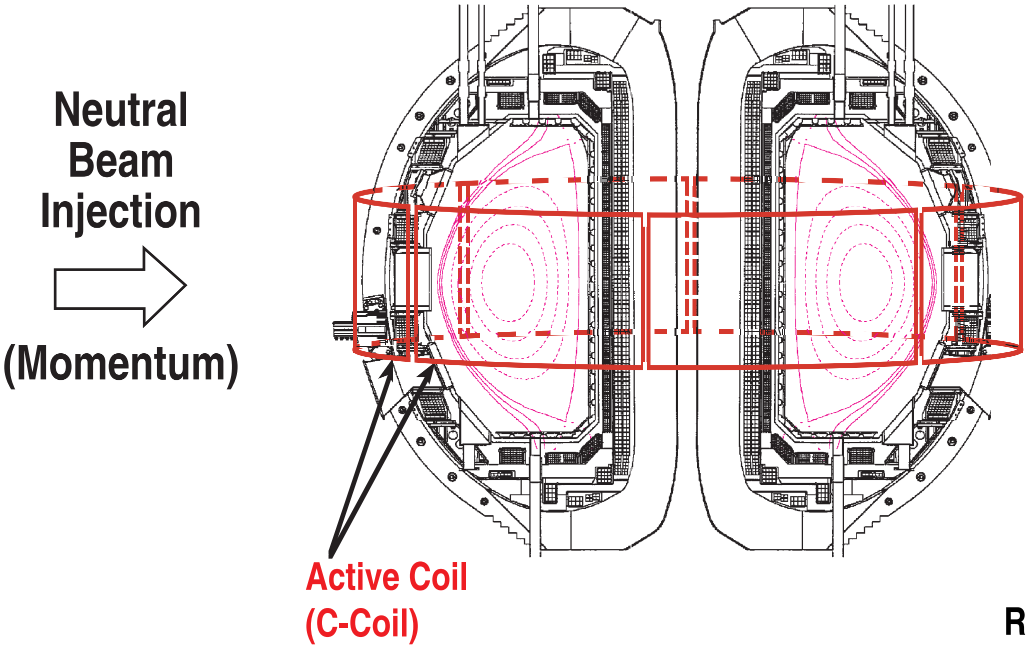
⇒ Sustained plasma rotation

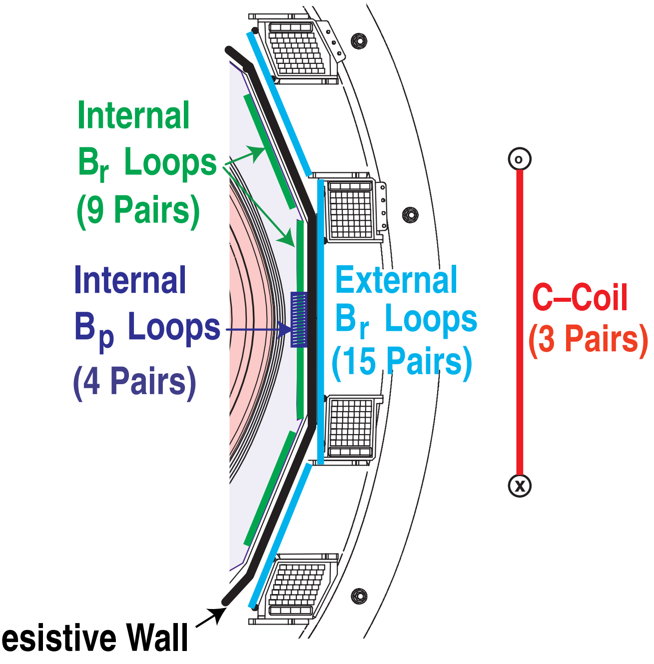
⇒ Stable operation well above the no-wall β limit (up to ideal-wall limit)

DIII-D
NATIONAL FUSION FACILITY
SAN DIEGO

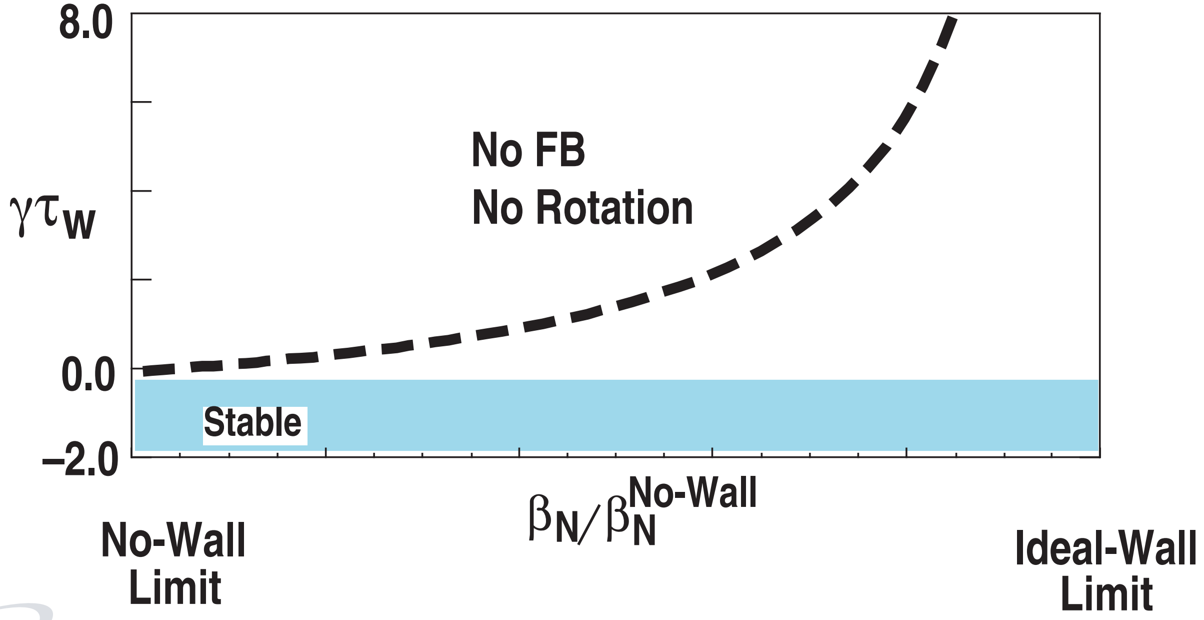
NON-AXISYMMETRIC “C-COIL” AND MAGNETIC FIELD SENSORS ARE USED FOR RWM AND RFA MINIMIZATION BY FEEDBACK CONTROL

- Six midplane coils (C–coil) connected in three pairs for n=1 control
- External and internal saddle loops measure δB_r
- Poloidal field probes measure δB_p with reduced coupling to the control coils





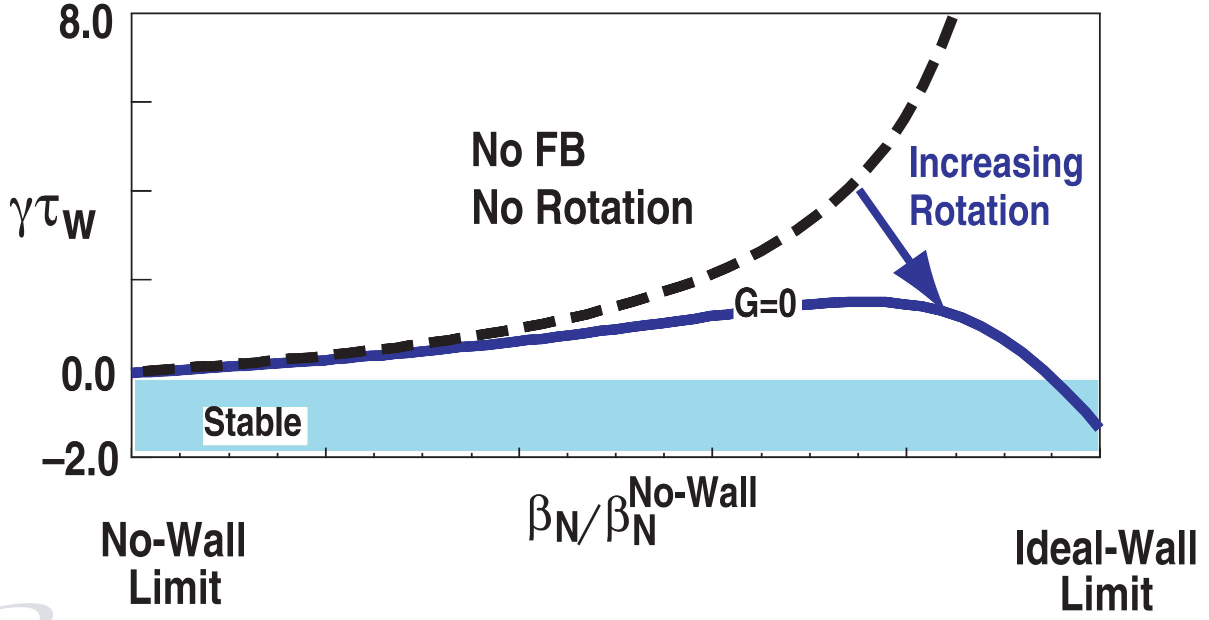
A MAGNETIC FEEDBACK SYSTEM COMBINED WITH ROTATIONAL STABILIZATION CAN PROVIDE A PATH TO IDEAL-WALL β_N LIMIT



A MAGNETIC FEEDBACK SYSTEM COMBINED WITH ROTATIONAL STABILIZATION CAN PROVIDE A PATH TO IDEAL-WALL β_N LIMIT

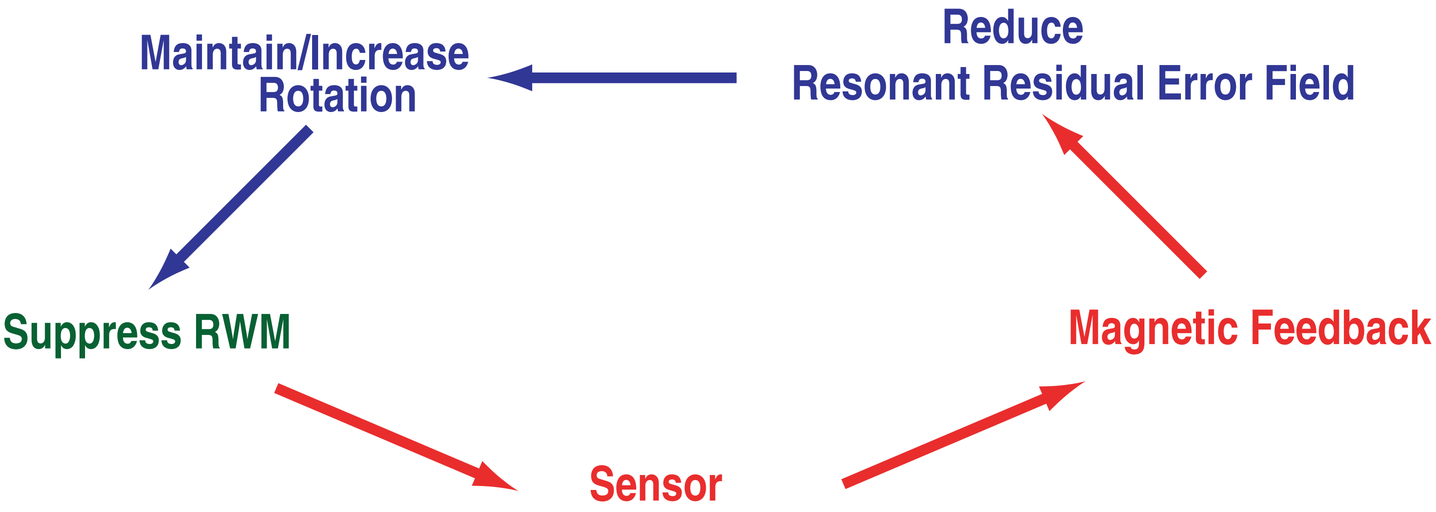
Maintain/Increase Rotation

Suppress RWM



255-02/KB/JY

A MAGNETIC FEEDBACK SYSTEM COMBINED WITH ROTATIONAL STABILIZATION CAN PROVIDE A PATH TO IDEAL-WALL β_N LIMIT



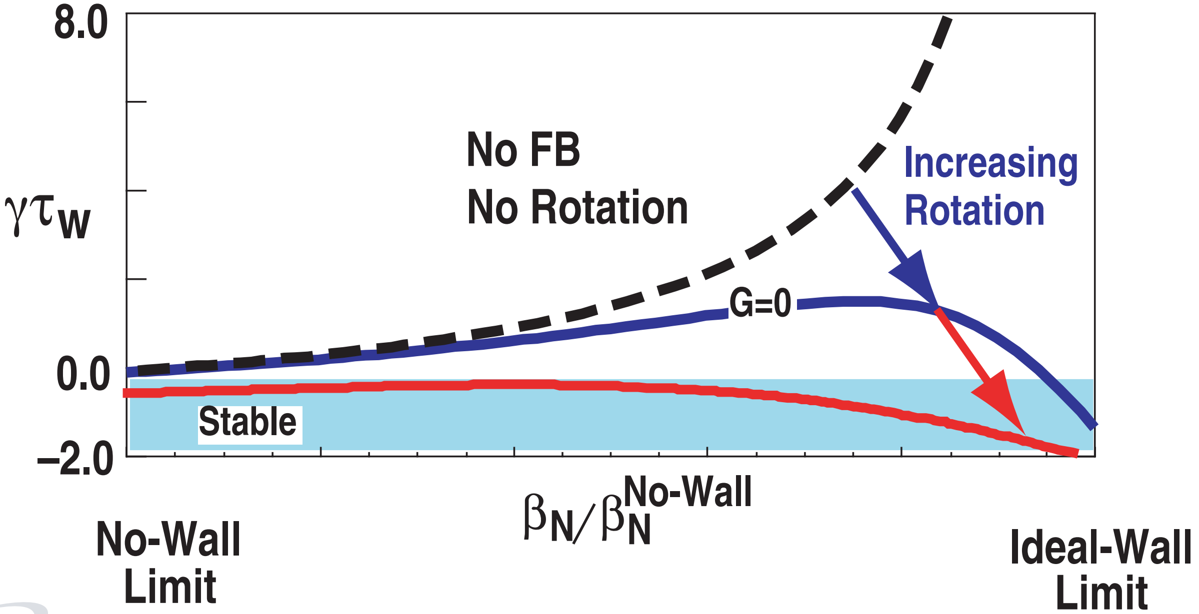
Reduce Resonant Residual Error Field

Maintain/Increase Rotation

Suppress RWM

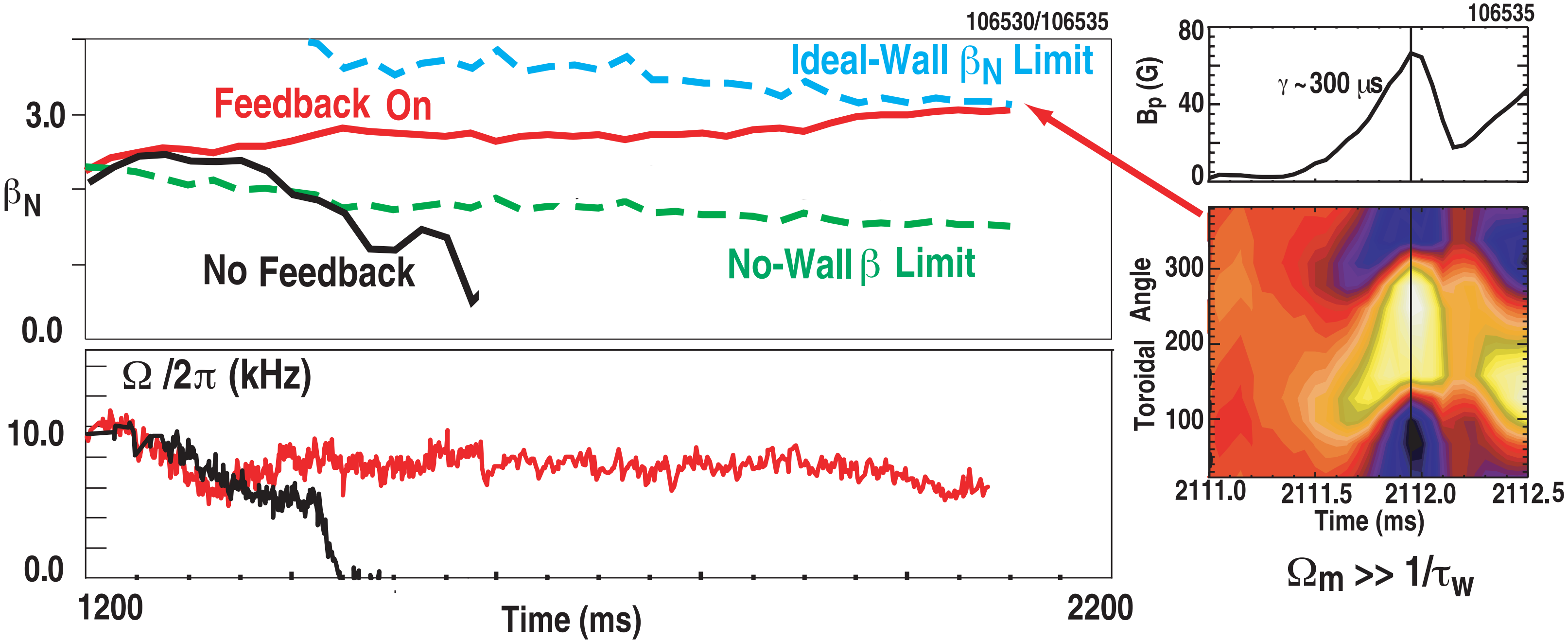
Sensor

Magnetic Feedback



FEEDBACK ALLOWS β_N TO APPROACH IDEAL WALL LIMIT

- $\beta_N \cong \beta_N$ (ideal wall) $\cong 2\ \beta_N$ (no-wall) (GATO-code)
- MHD at collapse grows on ideal-kink time scale

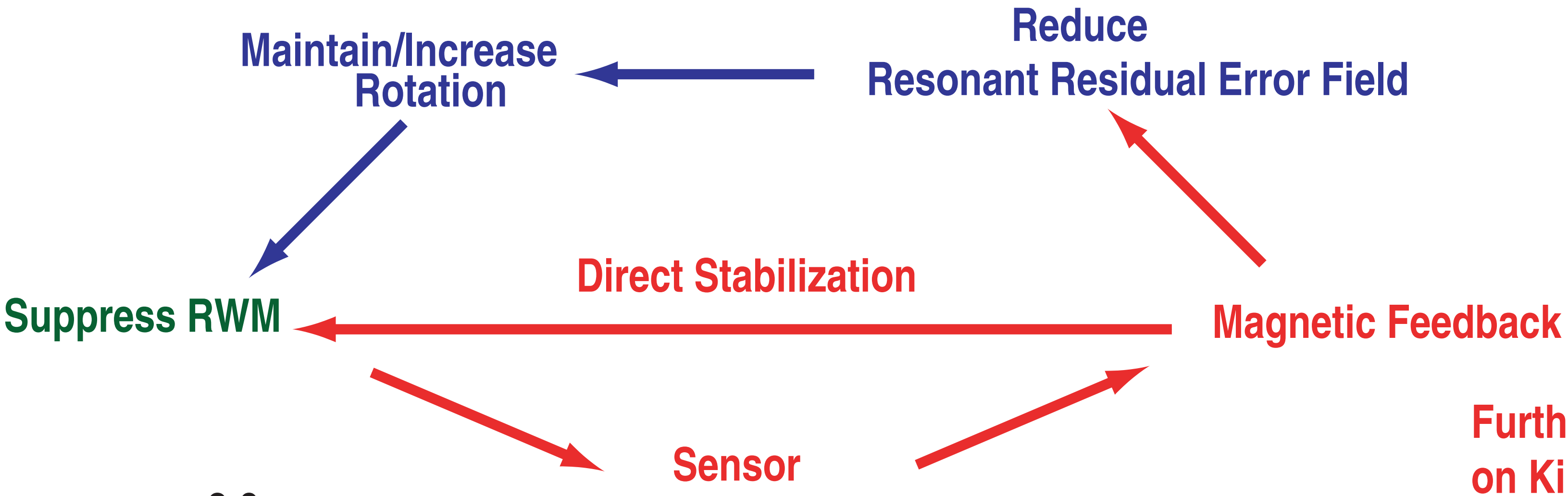


$\Omega_m >> 1/\tau_W$

- Rotational stabilization also possible with preprogramed C–coil current
  - Detailed feedback control of C–coil not necessary for rotational stabilization

DIII–D
NATIONAL FUSION FACILITY
SAN DIEGO

A MAGNETIC FEEDBACK SYSTEM COMBINED WITH ROTATIONAL STABILIZATION CAN PROVIDE A PATH TO IDEAL-WALL β_N LIMIT



Reduce Resonant Residual Error Field

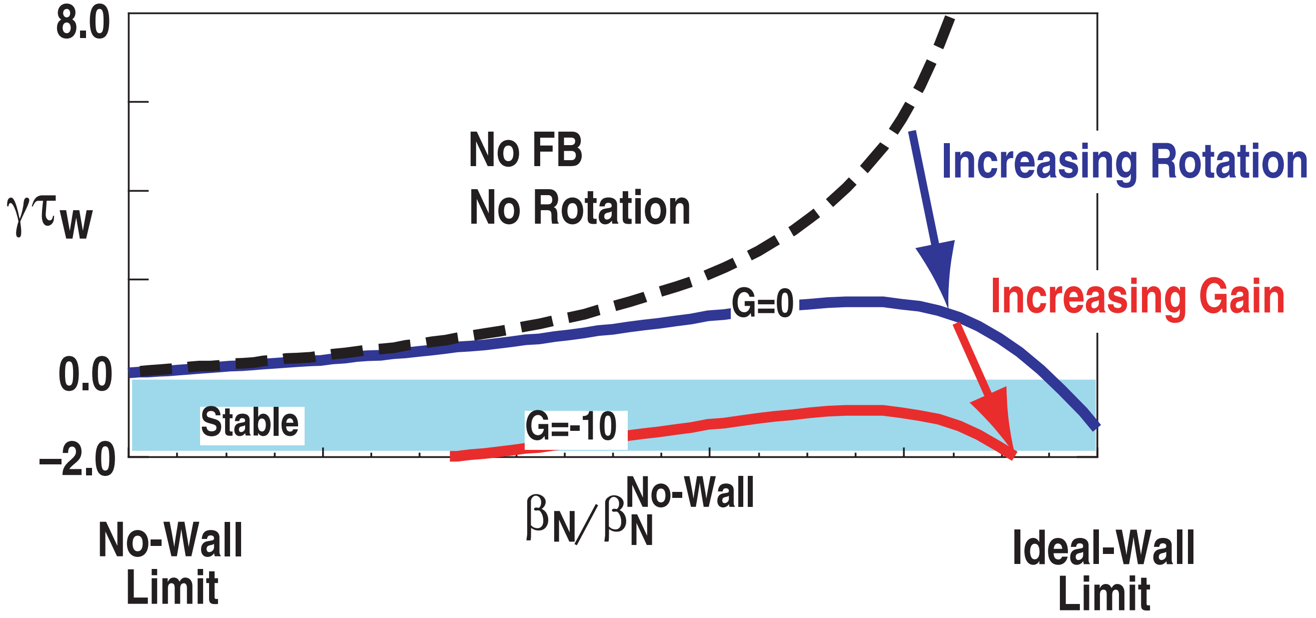
Maintain/Increase Rotation

Suppress RWM

Direct Stabilization

Magnetic Feedback

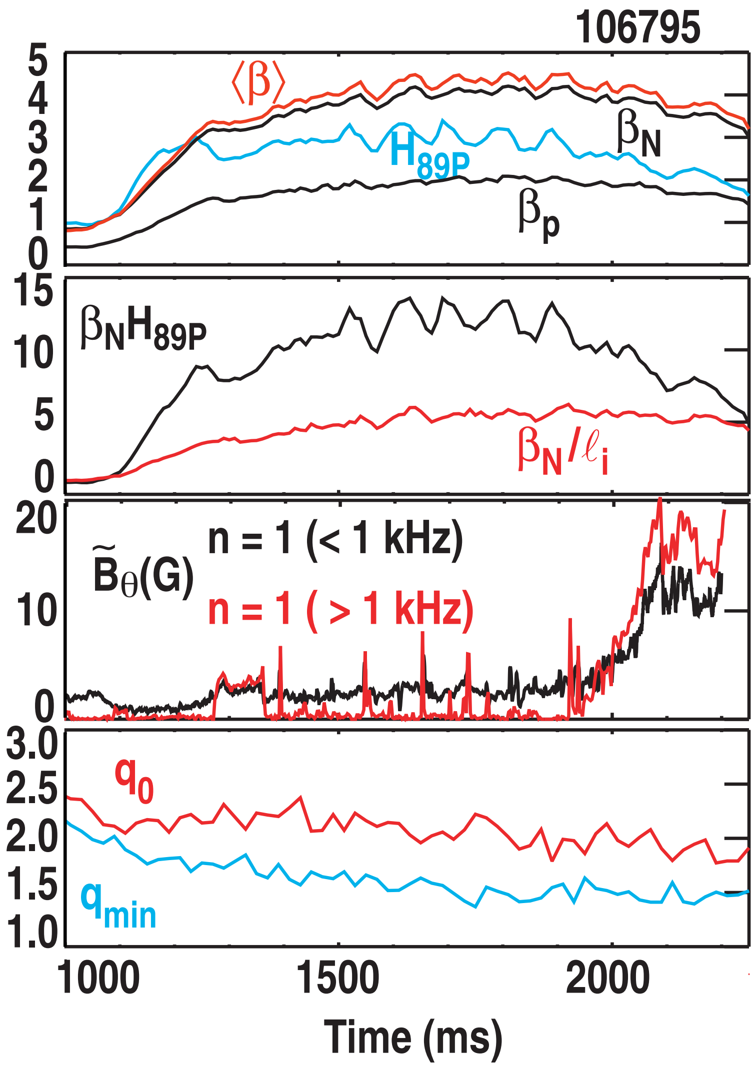
Sensor



Further Details on Kink and RWM Stabilization in E.J. Strait et al., EX/S2-1

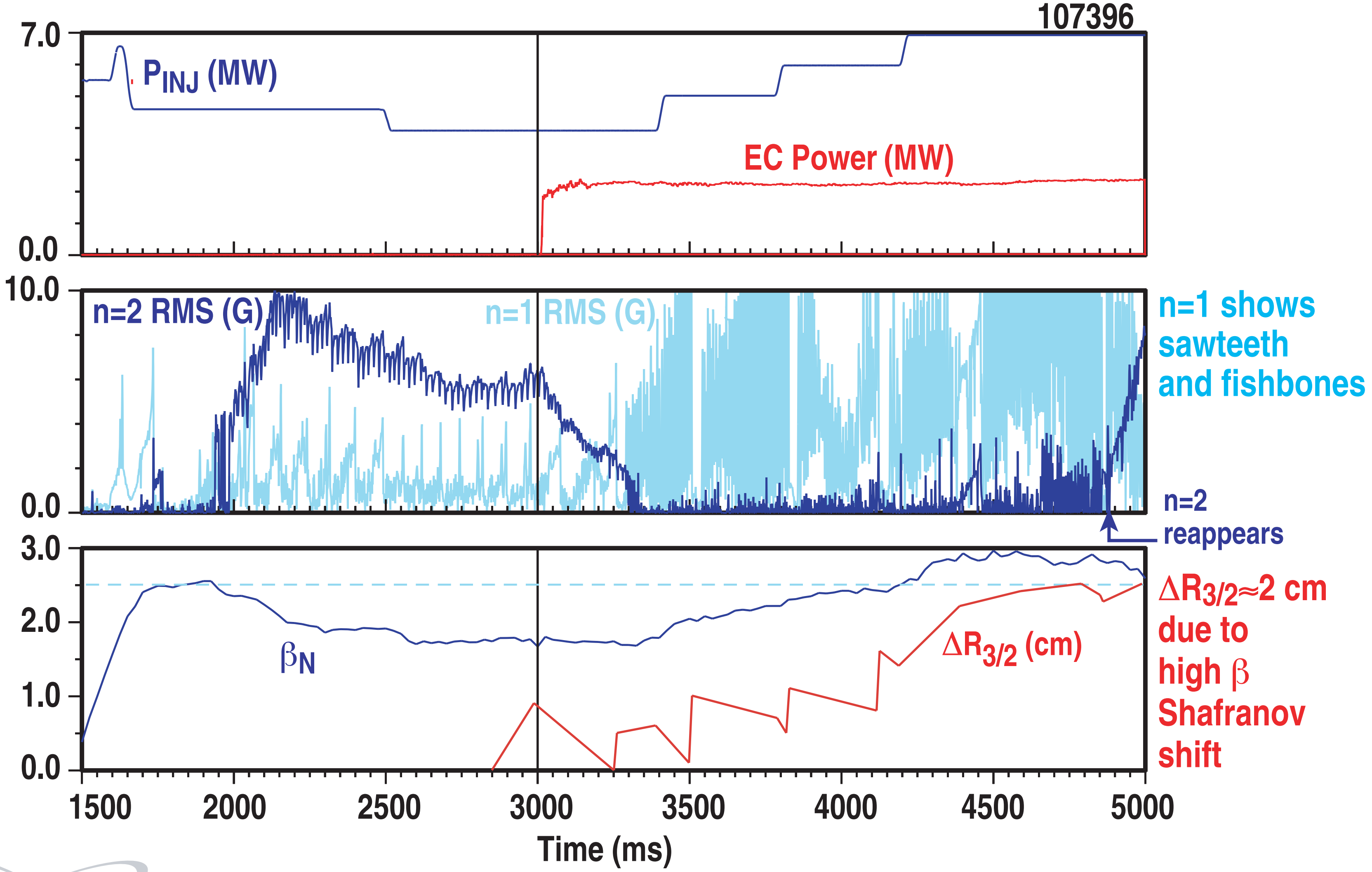
RWM STABILIZATION BY ROTATION ALLOWS HIGH β_N H_{89} OPERATION IN ADVANCED TOKAMAK PLASMAS

- $\beta_N H_{89} \geq 10$ for 680 ms ($4\tau_E$)
- $\beta = 4.2\%$, $\beta\tau_E = 0.66\%$ s, $\beta_p = 2$
- $\beta_N = 1.5\ \beta_N$ (no-wall)
- Bootstrap fraction 65%
- Total non inductive fraction 85%
- Duration limited by drop in q_{min} leading to onset of 2/1 neoclassical tearing mode (NTM)
  - Motivates work on current drive and NTM stabilization



β_N RAISED 60% AFTER ECCD SUPPRESSION OF m/n = 3/2 NTM

- Location of ECCD optimized in real time to minimize NTM amplitude
  - Location held fixed when amplitude is zero
  - Mode reappears as q = 3/2 moves radially by 2 cm off ECCD location

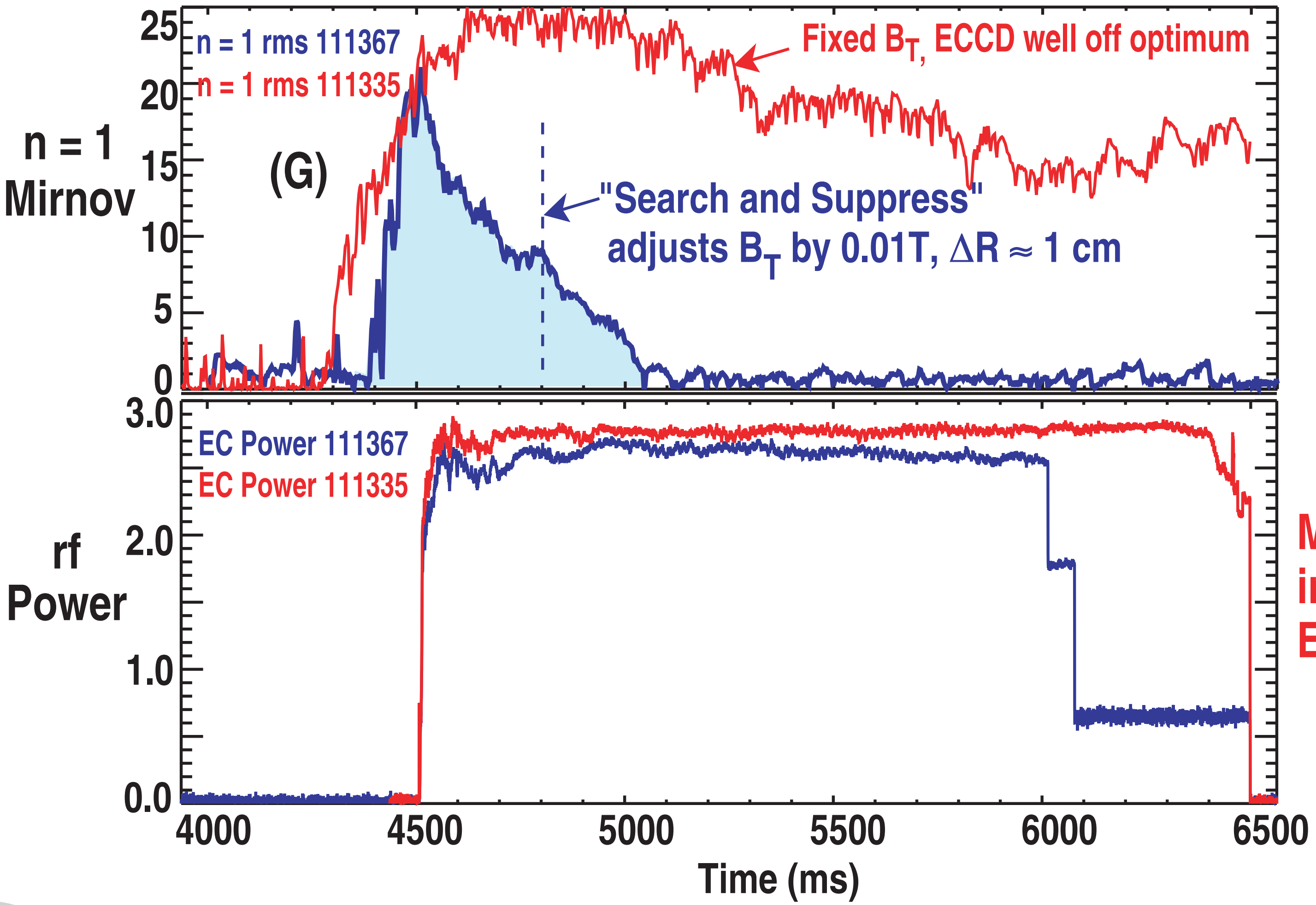


DIII-D NATIONAL FUSION FACILITY SAN DIEGO

255–02/KHB/wj

DEMONSTRATED COMPLETE SUPPRESSION OF THE m/n = 2/1 TEARING MODE BY RADIALLY LOCALIZED ECCD

- β_N is feedback controlled to temporarily rise to excite the mode
- Location of ECCD optimized (#111367) by toroidal field PCS "Search and Suppress"



More information in R.J. La Haye et al. EX/S1-3

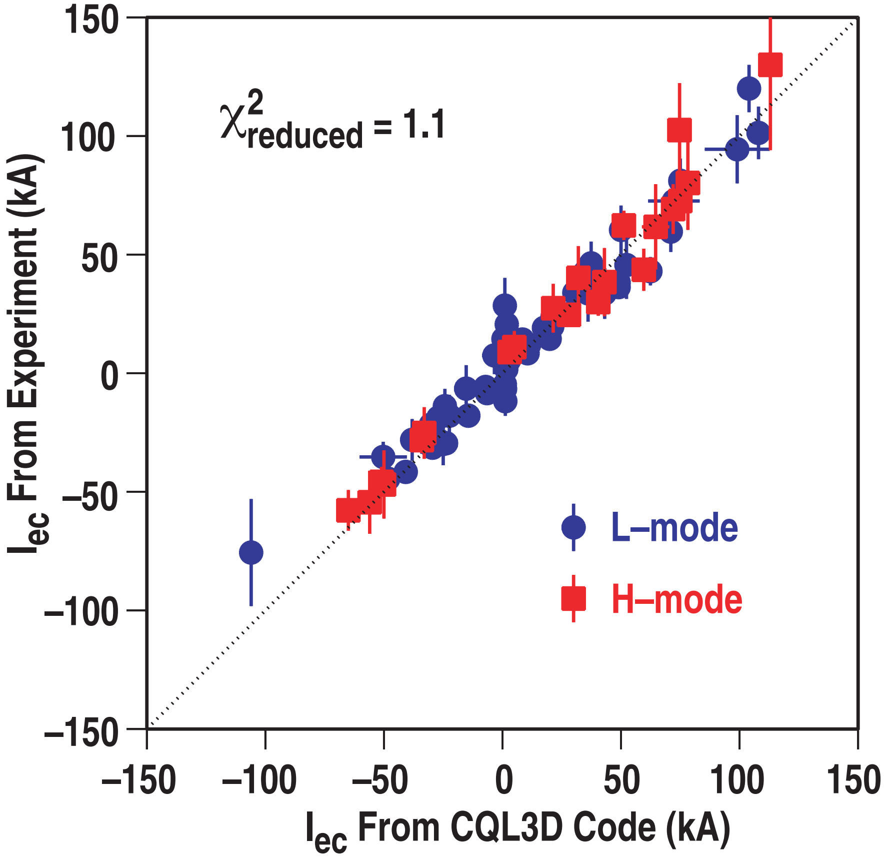
SUBSTANTIAL PROGRESS SINCE IAEA 2000

We have made progress since the last IAEA meeting in developing the building blocks needed for advanced operating modes:

- Substantially broadened the MHD stable operating space
  - Rotational stabilization of resistive wall modes yielding $\beta_N = \beta_N$ (ideal wall) $\cong 2\ \beta_N$ (no wall)
  - Increased β by 60% via stabilization of (3,2) neoclassical tearing mode with ECCD in sawtoothing plasmas
  - First stabilization of (2,1) neoclassical tearing mode using ECCD
- Developed plasma control tools
  - First integrated AT discharges with current profile control using ECCD
  - Pressure and density profile control with ECH and ECCD
- Demonstrated an improved, high q_{95} (>4) operating scenario for ITER
- Achieved solutions to key burning plasma issues
  - No ELM-produced, pulsed divertor heat load in QH–mode plasmas
  - Small heat and particle loads at inner divertor strike points in balanced double-null divertors
  - Disruption mitigation via massive gas puff

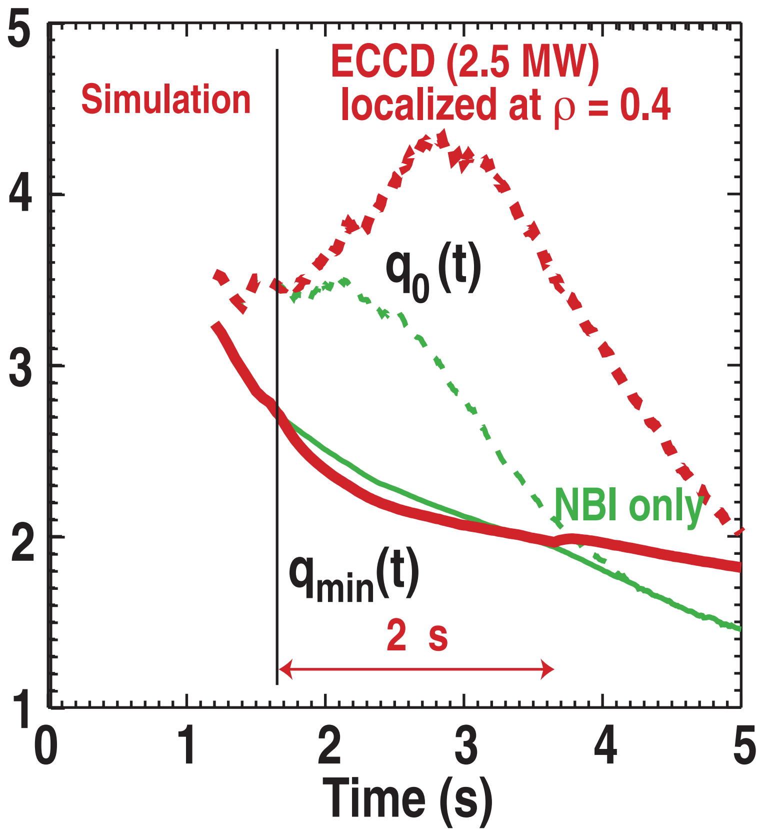
VALIDATED ECCD THEORY ALLOWS USE OF DETAILED COMPUTER MODELS TO DEVELOP EXPERIMENTS

- Excellent agreement of ECCD theory and experiment



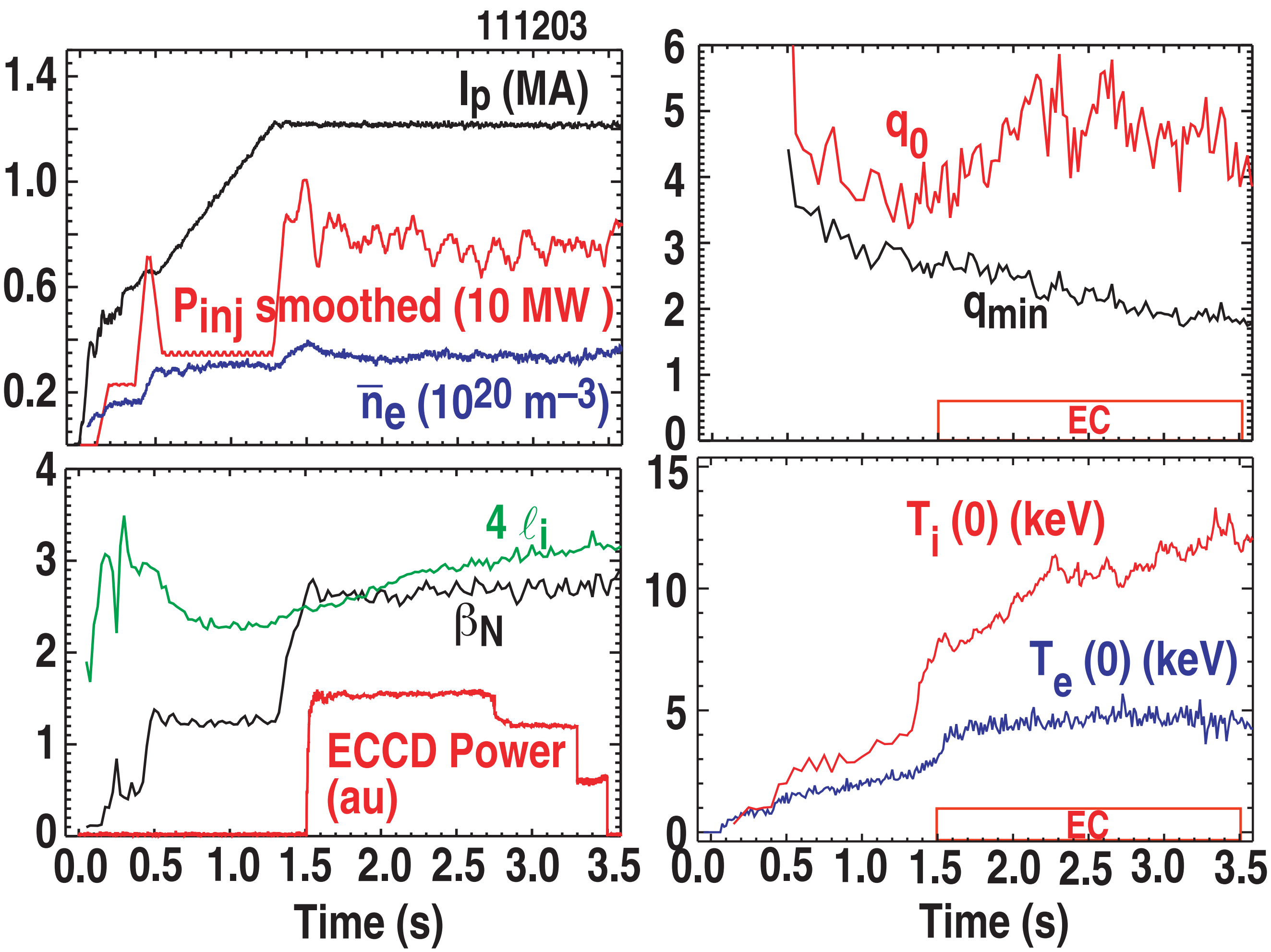
More information in C.C. Petty, et al., EX/W-4

- Prediction of enhanced negative central shear in AT plasma with ECCD at ρ = 0.4



ECCD PRODUCES CURRENT PROFILE MODIFICATION IN ADVANCED TOKAMAK PLASMA

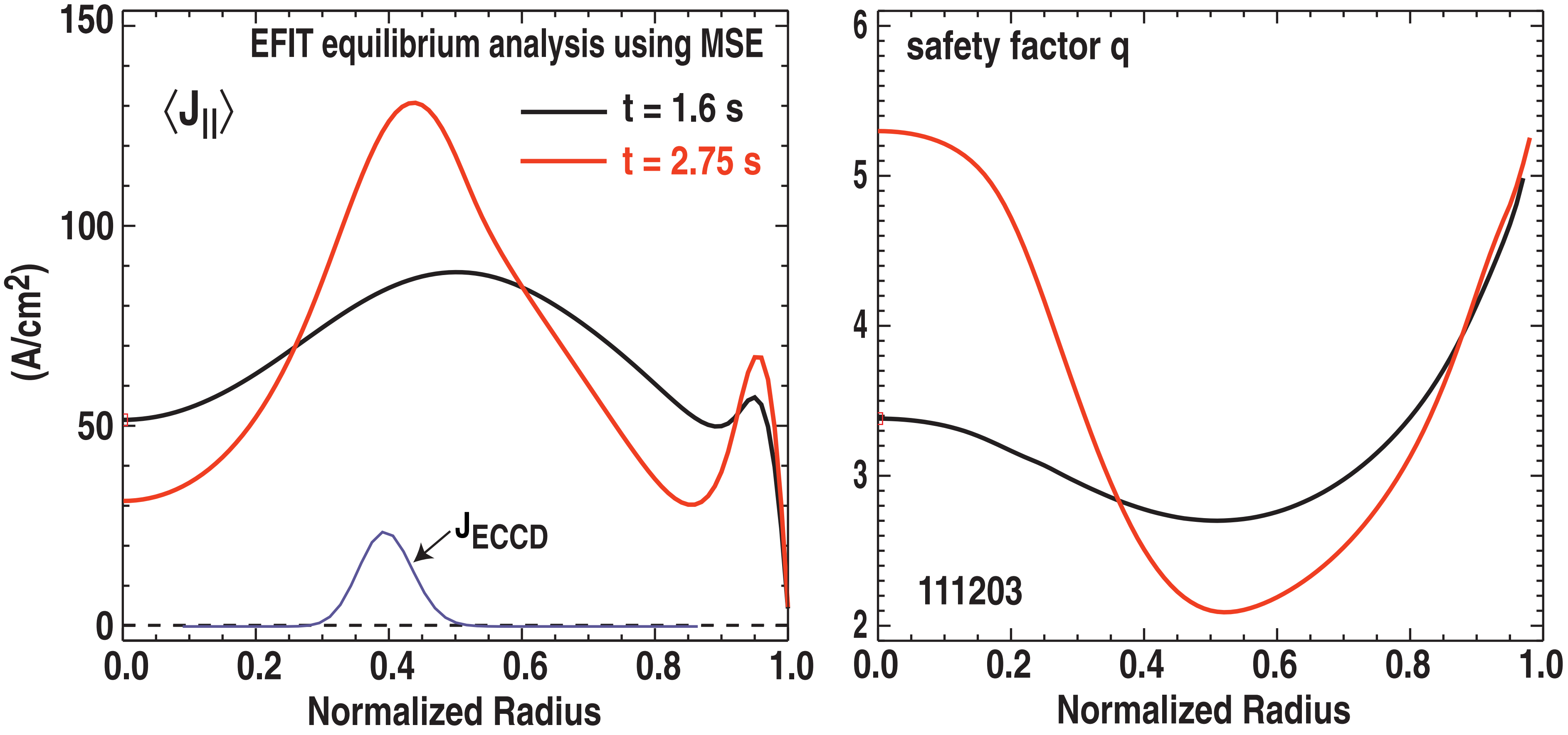
- $\beta_N H_{89} \gtrsim 7$ for full 2.0 s ECCD pulse
- β_N at or slightly above β_N (no-wall)
- Total non-inductive current fraction ≳90%
- q profile modified during high β, AT phase of shot



NATIONAL FUSION FACILITY
SAN DIEGO

255–02/KHB/wj

ECCD PEAKS CURRENT DENSITY AT RESONANCE LOCATION AND PRODUCES STRONGER NEGATIVE MAGNETIC SHEAR



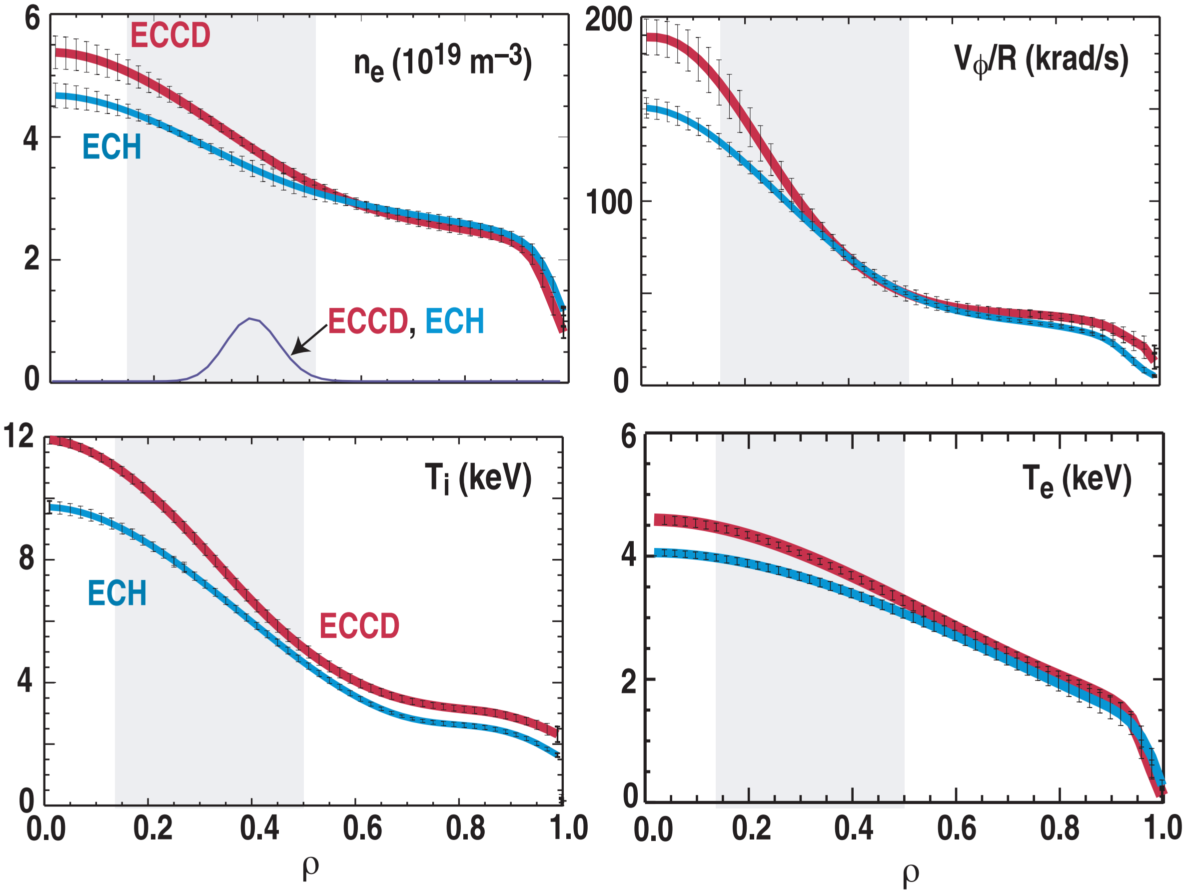
- Clear evidence of q–profile modification also seen in quiescent double barrier (QDB) plasmas (E.J. Doyle, et al. EX/C3-2)

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

268–02/MRW/wj

ECCD CAN TRIGGER FORMATION OF CORE TRANSPORT BARRIERS IN ADVANCED TOKAMAK DISCHARGES

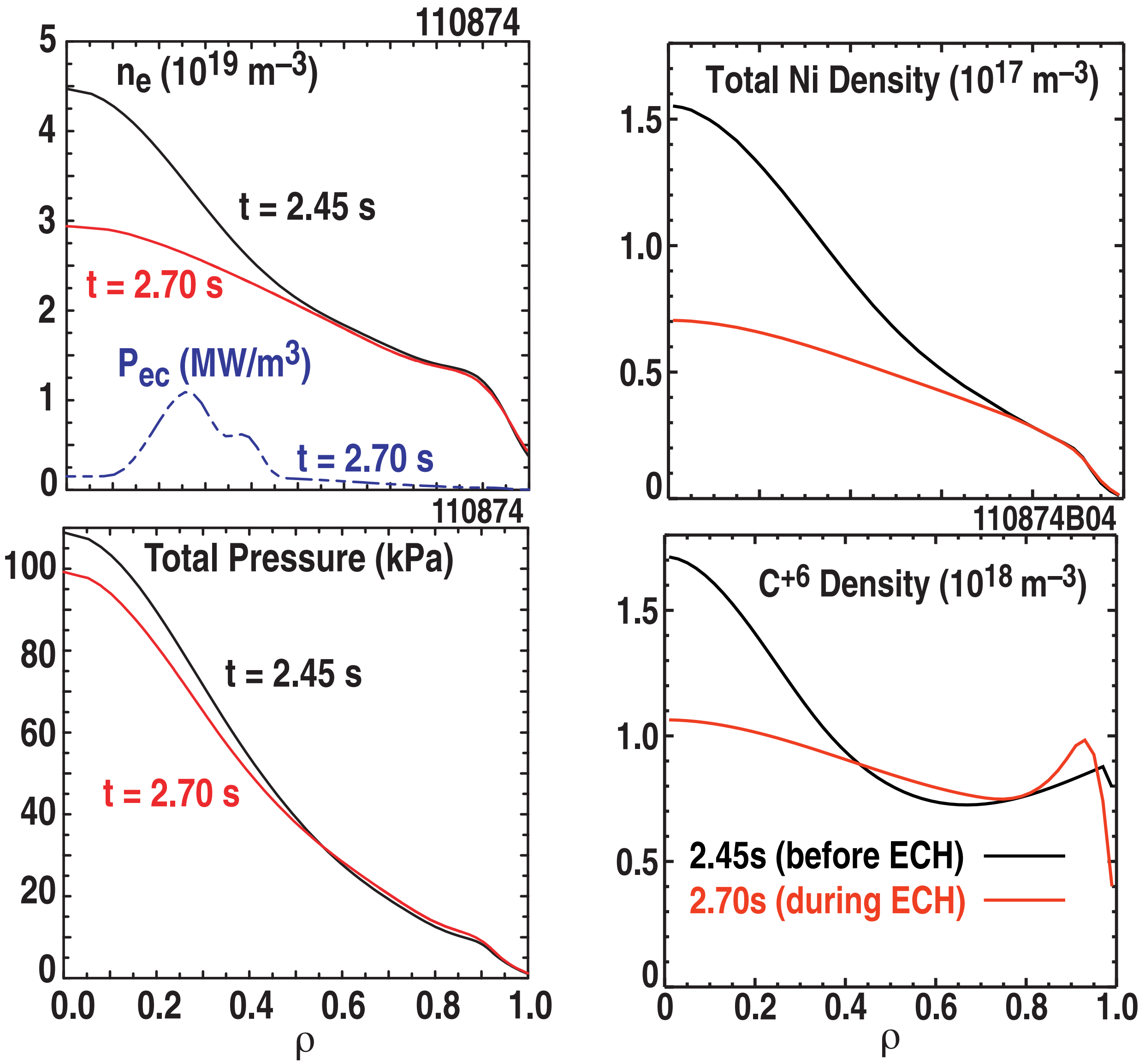
- Core barriers seen in all four transport channels with ECCD
  - No barriers in ECH case with no current drive
- Gyrokinetic stability code analysis shows E×B shear and Shafranov shift stabilization are both important
- More information in M.R. Wade et al. EX/P3–16



255-02/KB/JY

DENSITY AND IMPURITY PROFILES MODIFIED WITH ECH AND ECCD IN QUIESCENT DOUBLE BARRIER PLASMAS

- EC power applied near $\rho = 0.2$ in plasma with core transport barrier already formed
- Density peaking reduced, leading to much reduced central impurity densities and factor 1.3 reduction in Z_{eff}
- More information in E.J. Doyle, et al., EX/C3-2



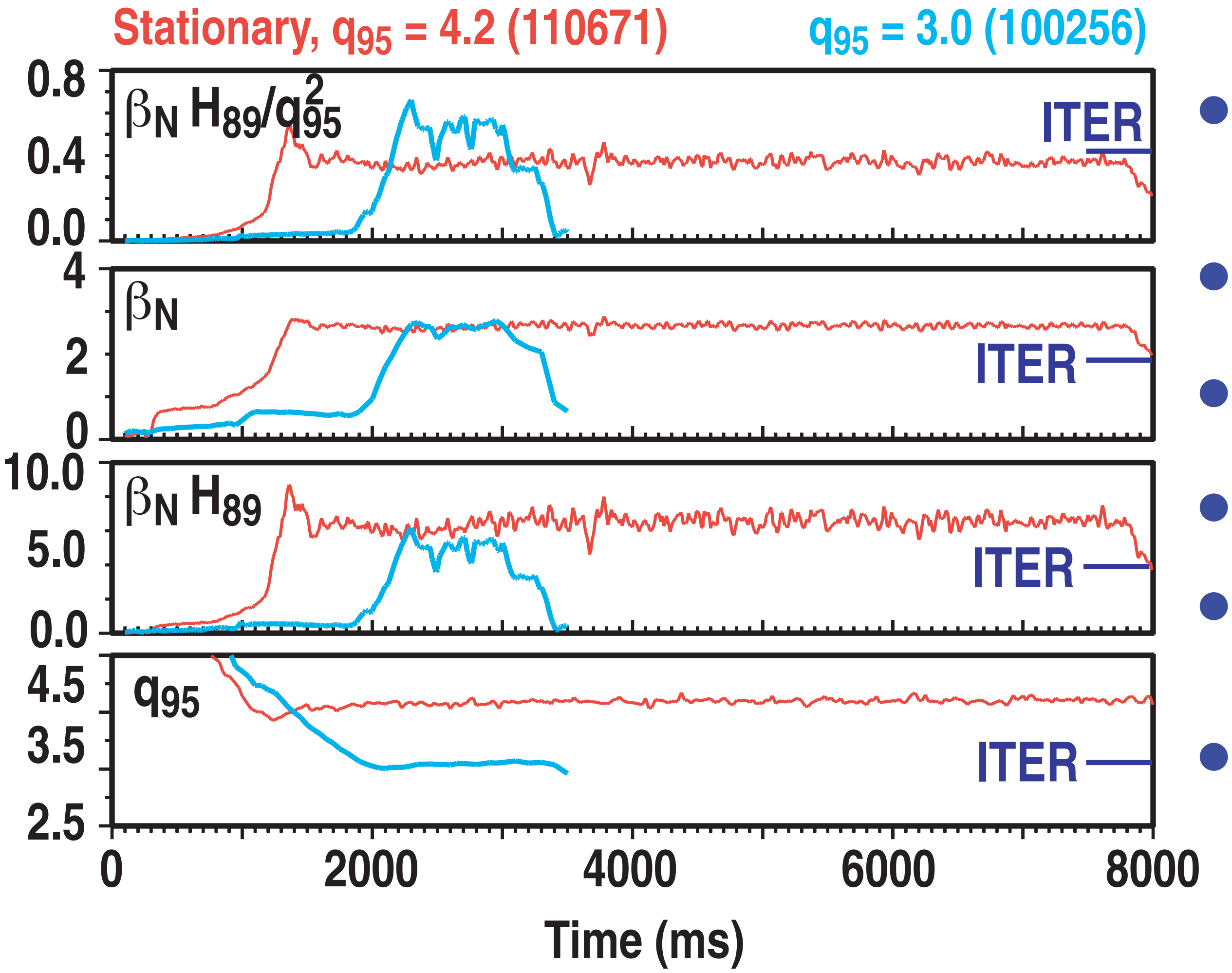
255–02/KHB/wj

SUBSTANTIAL PROGRESS SINCE IAEA 2000

We have made progress since the last IAEA meeting in developing the building blocks needed for advanced operating modes:

- Substantially broadened the MHD stable operating space
  - Rotational stabilization of resistive wall modes yielding $\beta_N = \beta_N$ (ideal wall) ≅ 2 β_N (no wall)
  - Increased β by 60% via stabilization of (3,2) neoclassical tearing mode with ECCD in sawtoothing plasmas
  - First stabilization of (2,1) neoclassical tearing mode using ECCD
- Developed plasma control tools
  - First integrated AT discharges with current profile control using ECCD
  - Pressure and density profile control with ECH and ECCD
- Demonstrated an improved, high q_{95} (>4) operating scenario for ITER
- Achieved solutions to key burning plasma issues
  - No ELM-produced, pulsed divertor heat load in QH–mode plasmas
  - Small heat and particle loads at inner divertor strike points in balanced double-null divertors
  - Disruption mitigation via massive gas puff

STATIONARY PLASMAS WITH $\beta_N H/q_{95}^2 \simeq$ ITER DESIGN VALUE AND $q_{95} > 4$ HAVE BEEN DEMONSTRATED ON DIII–D



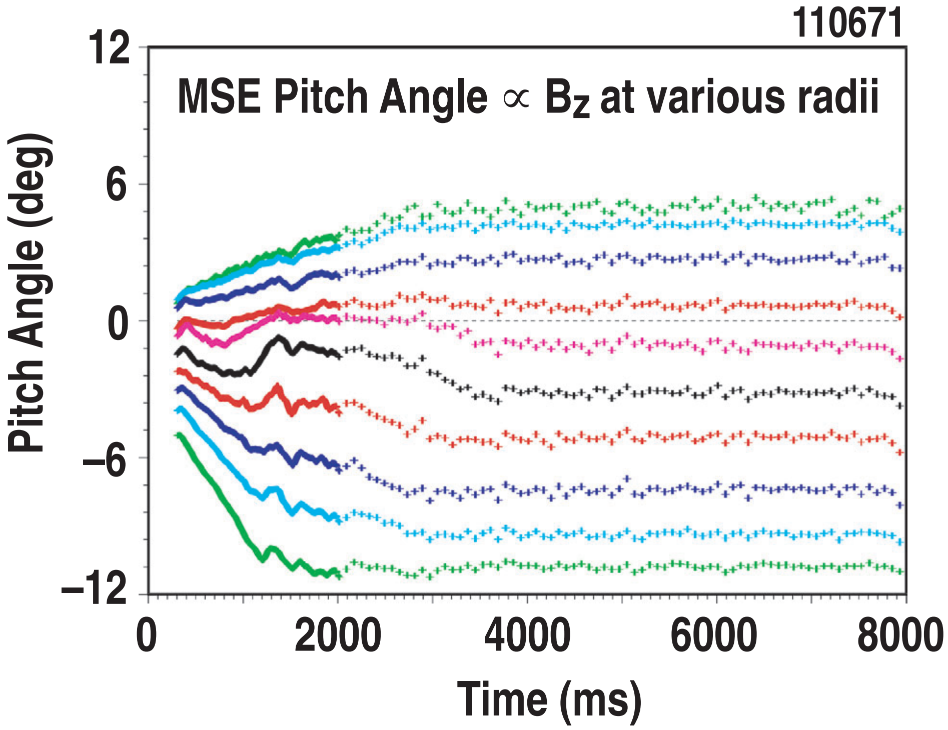
- Fusion gain proportional to $\beta_N H_{89}/q_{95}^2$
- $q(0) > 1$
- 3/2 NTM prevents sawteeth
- $\beta_N \simeq \beta_N^{\text{no-wall}}$
- Non-inductive current fraction $\simeq$ 50%
- Candidate for extended pulse length, hybrid scenario for ITER

DIII–D
NATIONAL FUSION FACILITY
SAN DIEGO

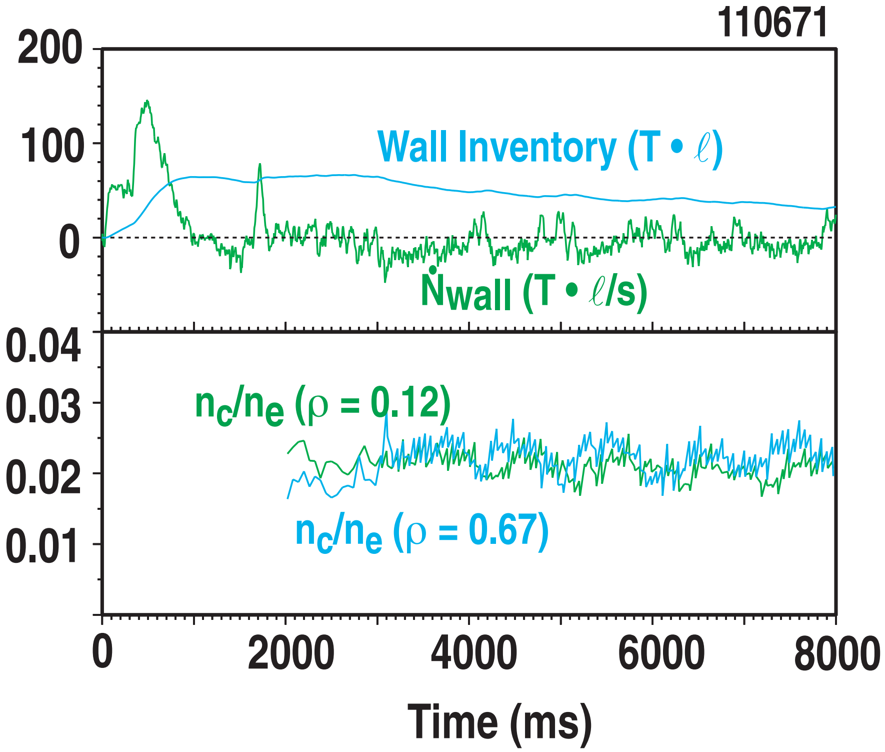
255-02/KB/JY

CURRENT PROFILE IS FULLY RELAXED AND WALL PARTICLE INVENTORY IS EQUILIBRATED AFTER 3.0 s

- $\tau_{dur} \simeq 36\ \tau_E \simeq 2\ \tau_{CR}$



- Wall not important in particle balance



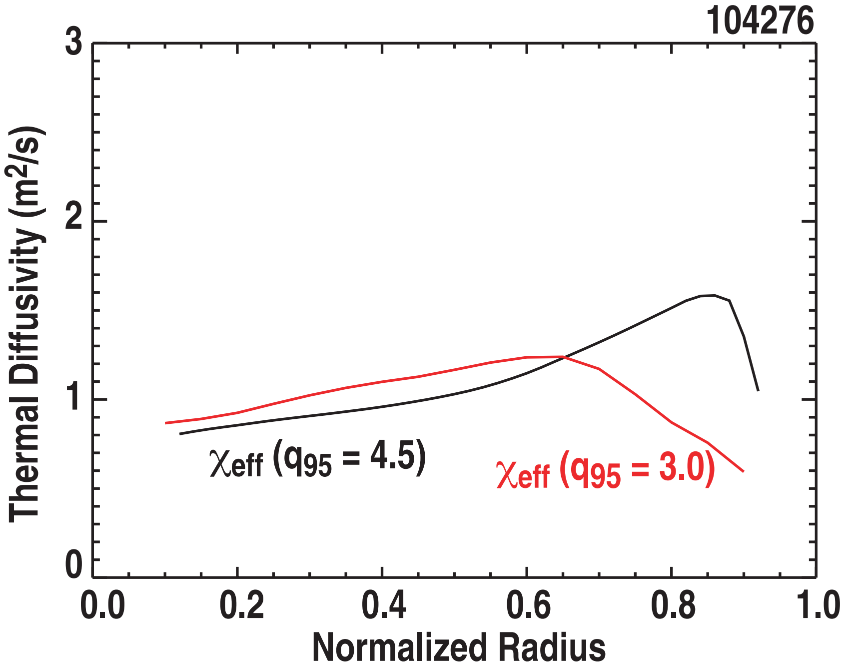
More information in T.C. Luce et al. EX/P3–13

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

255-02/KB/JY

ENERGY TRANSPORT IN LONG PULSE DISCHARGE COMPARABLE TO THAT OBTAINED IN LOW q_{95} REFERENCE SHOT

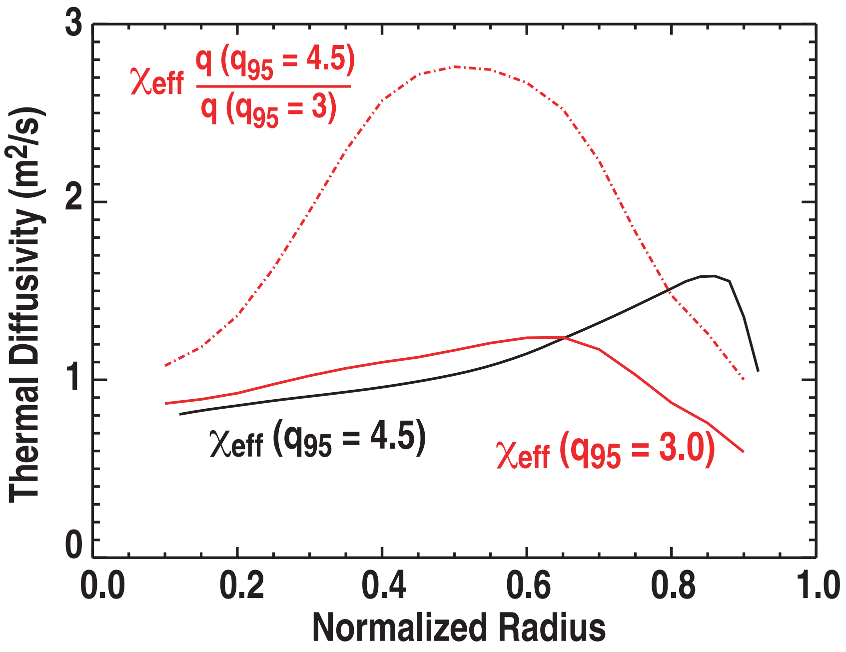
- χ_{eff} substantially lower than that expected by q scaling of transport
  - Global confinement scaling: $\chi_{eff} \propto q^{1.4}$
  - Nondimensional transport studies: $\chi_{eff} \propto q^2$



DIII-D NATIONAL FUSION FACILITY SAN DIEGO

ENERGY TRANSPORT IN LONG PULSE DISCHARGE COMPARABLE TO THAT OBTAINED IN LOW q_{95} REFERENCE SHOT

- χ_{eff} substantially lower than that expected by q scaling of transport
  - Global confinement scaling: $\chi_{eff} \propto q^{1.4}$
  - Nondimensional transport studies: $\chi_{eff} \propto q^2$
- Shown: $\chi_{eff} \propto q$



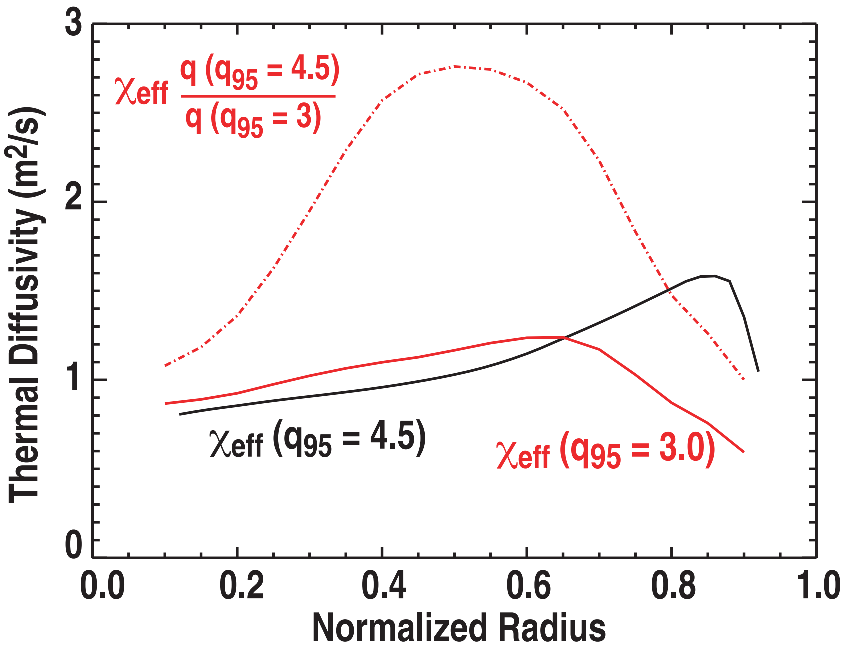
DIII-D NATIONAL FUSION FACILITY SAN DIEGO

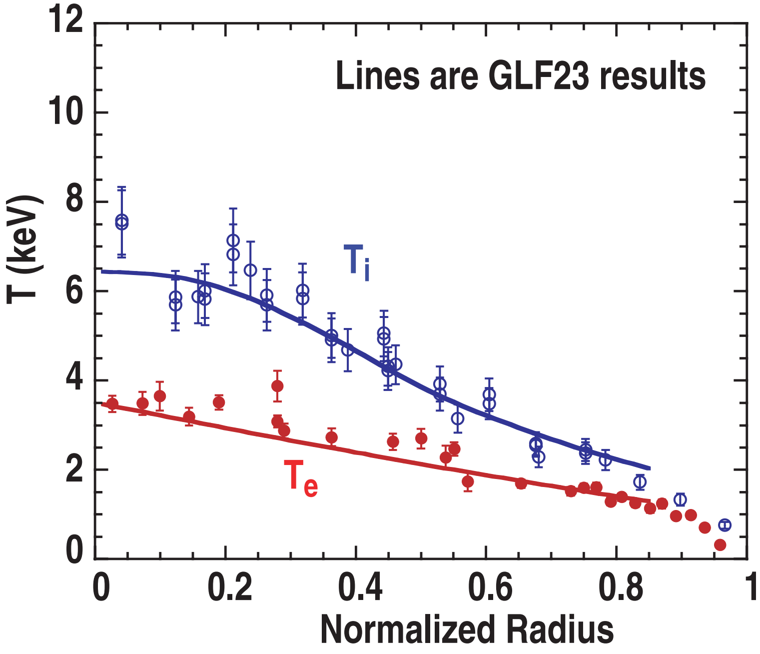
255-02/KB/JY

ENERGY TRANSPORT IN LONG PULSE DISCHARGE COMPARABLE TO THAT OBTAINED IN LOW q_{95} REFERENCE SHOT

- χ_{eff} substantially lower than that expected by q scaling of transport
  - Global confinement scaling: $\chi_{eff} \propto q^{1.4}$
  - Nondimensional transport studies: $\chi_{eff} \propto q^2$
- Shown: $\chi_{eff} \propto q$

- GLF23 drift-wave mode simulation give good agreement with measured profiles
- Model contains ITG, TEM, and ETG with effects of E×B
- More information in J.E. Kinsey et al TH/P1-9





255-02/KB/JY

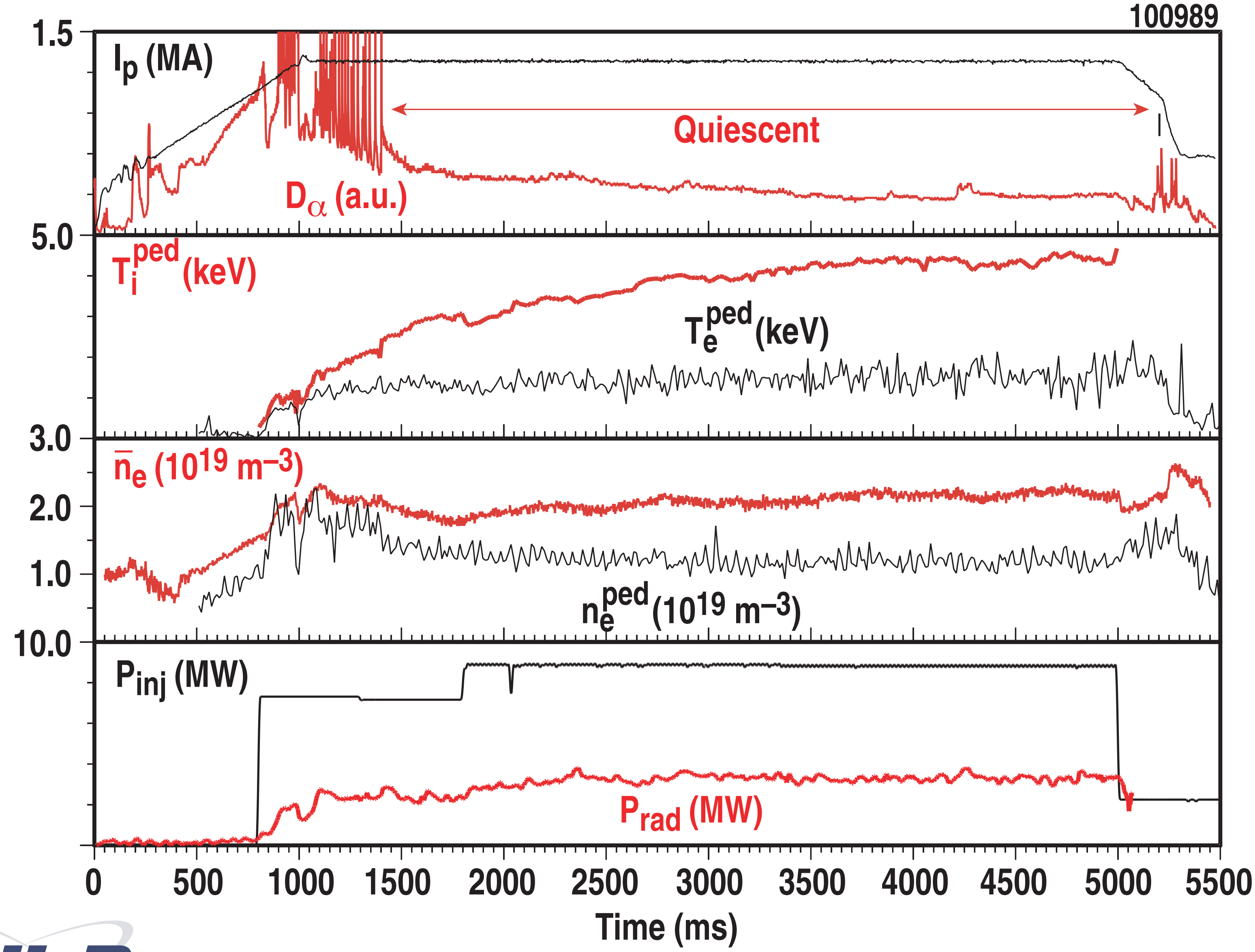
SUBSTANTIAL PROGRESS SINCE IAEA 2000

We have made progress since the last IAEA meeting in developing the building blocks needed for advanced operating modes:

- Substantially broadened the MHD stable operating space
  - Rotational stabilization of resistive wall modes yielding $\beta_N = \beta_N$ (ideal wall) $\cong 2\ \beta_N$ (no wall)
  - Increased β by 60% via stabilization of (3,2) neoclassical tearing mode with ECCD in sawtoothing plasmas
  - First stabilization of (2,1) neoclassical tearing mode using ECCD
- Developed plasma control tools
  - First integrated AT discharges with current profile control using ECCD
  - Pressure and density profile control with ECH and ECCD
- Demonstrated an improved, high q_{95} (>4) operating scenario for ITER
- Achieved solutions to key burning plasma issues
  - No ELM-produced, pulsed divertor heat load in QH–mode plasmas
  - Small heat and particle loads at inner divertor strike points in balanced double-null divertors
  - Disruption mitigation via massive gas puff

QUIESCENT H–MODE RUNS ELM–FREE FOR LONG PULSES WITH CONSTANT DENSITY AND RADIATED POWER

- Duration limited by neutral beam pulse length



DIII-D
NATIONAL FUSION FACILITY
SAN DIEGO

255-02/KB/JY

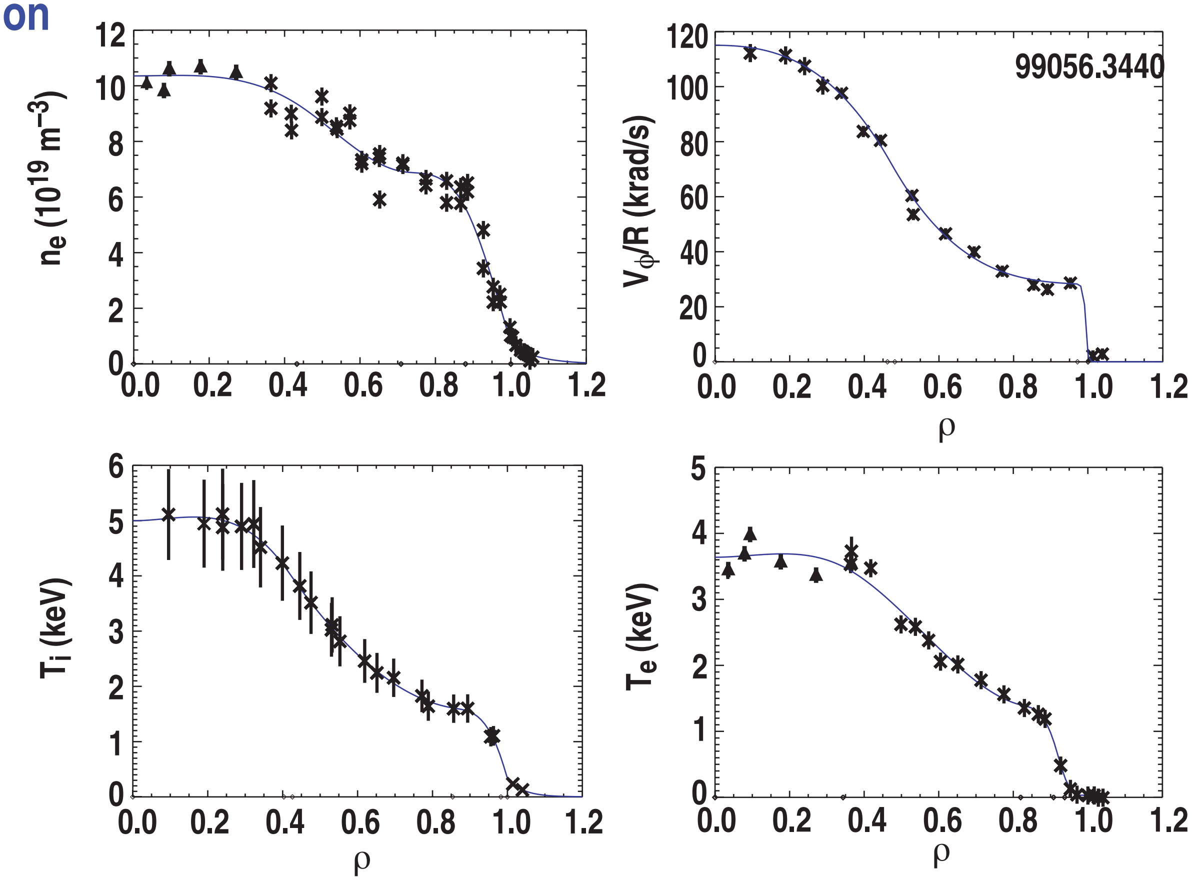
QUIESCENT H–MODE HAS BEEN SEEN OVER A RANGE OF PARAMETERS

- Requires neutral beam injection counter to I_p direction plus divertor cryopumping

- QH–mode seen to date for
  - $3.4 \leq q_{95} \leq 5.8$
  - $1.0 \leq I_p$ (MA) ≤ 2.0
  - $1.8 \leq B_T$ (T) ≤ 2.1
  - $1.0 \leq n_e^{ped}$ (10^{19} m^{-3}) ≤ 6.5

- Low field example at $B_T = 0.95$ T, $I_p = 0.67$ MA and $n_e^{ped} = 1.1{\times}10^{19}$ m^{-3}

High Density QH–Mode



QUIESCENT H–MODE HAS BEEN SEEN OVER A RANGE OF PARAMETERS

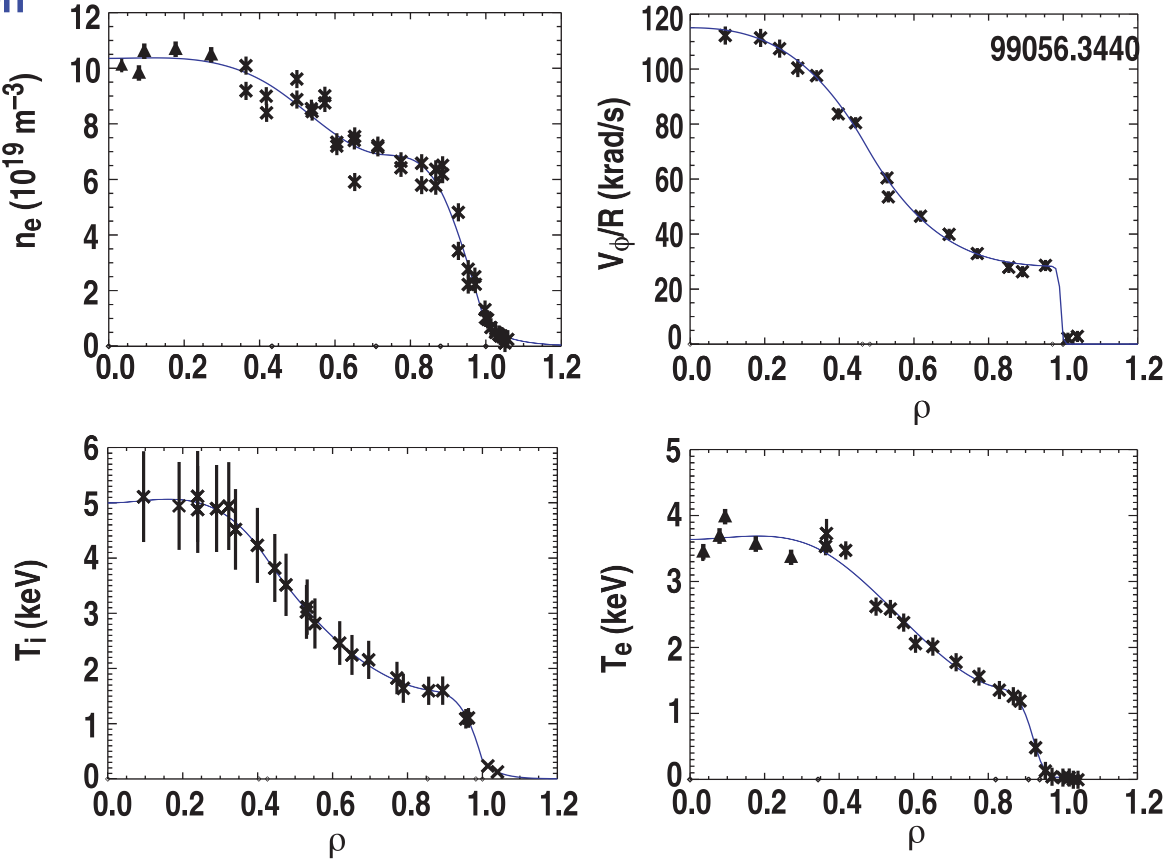
- Requires neutral beam injection counter to I_p direction plus divertor cryopumping

- QH–mode seen to date for
  - $3.4 \le q_{95} \le 5.8$
  - $1.0 \le I_p$ (MA) ≤ 2.0
  - $1.8 \le B_T$ (T) ≤ 2.1
  - $1.0 \le n_e^{ped}$ (10^{19} m^{-3}) ≤ 6.5

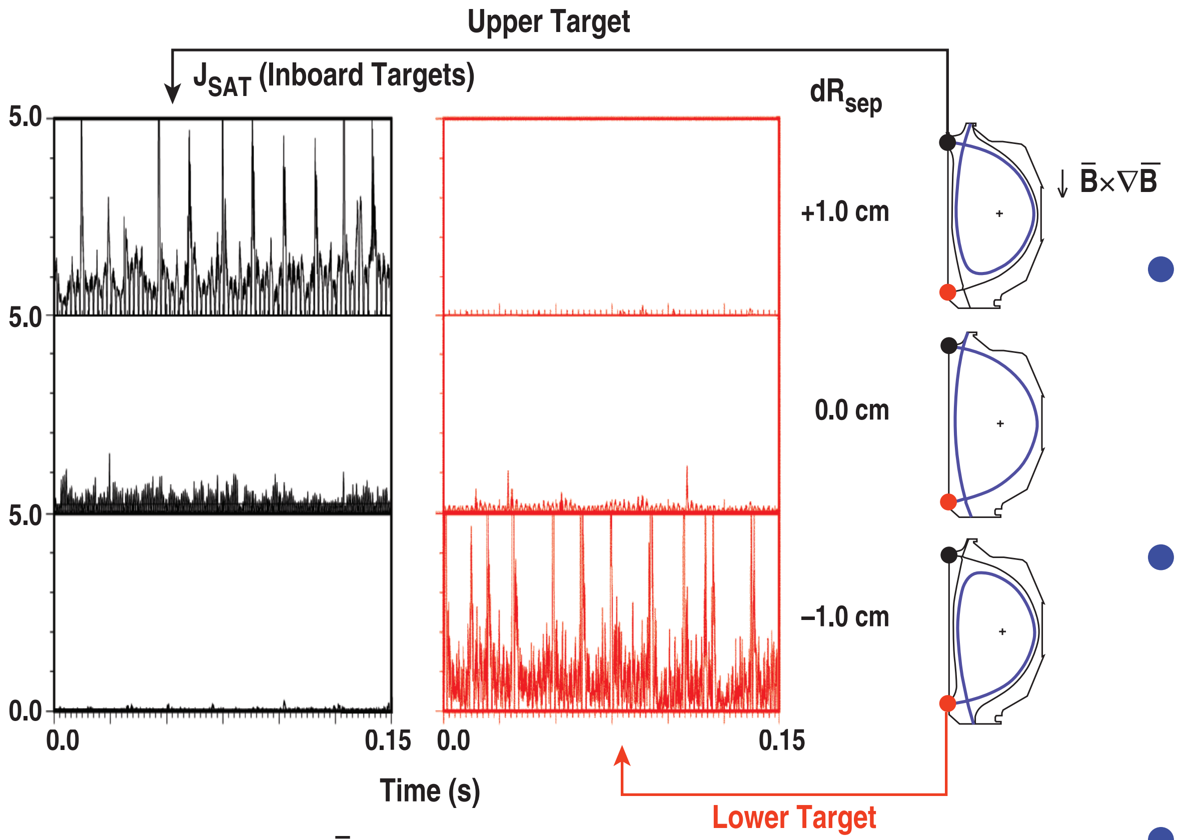
- Low field example at $B_T = 0.95$ T, $I_p = 0.67$ MA and $n_e^{ped} = 1.1{\times}10^{19}$ m^{-3}

- QH–mode recently seen in ASDEX–U

High Density QH–Mode



THE INBOARD DIVERTOR PARTICLE AND HEAT FLUXES ARE RELATIVELY LOW IN SYMMETRIC DOUBLE-NULL PLASMA



$$\frac{\bar{n}_e}{n_{gw}} = 0.45 - 0.65$$

- ELM activity at the inboard target(s) is significantly reduced in DN
  →ELMs are generated on the outboard side (consistent with ELITE analysis)
- Strong variation in the time – averaged particle flux ratios

  e.g., $\frac{\Gamma_{in}}{\Gamma_{out}} \approx 0.2$ (DN)

  ≈ 1 (SN)
- Strong variation in the time – averaged heat flux ratios

  e.g., $\frac{q_{in}}{q_{out}} \approx 0.05\text{-}0.15$ (DN)

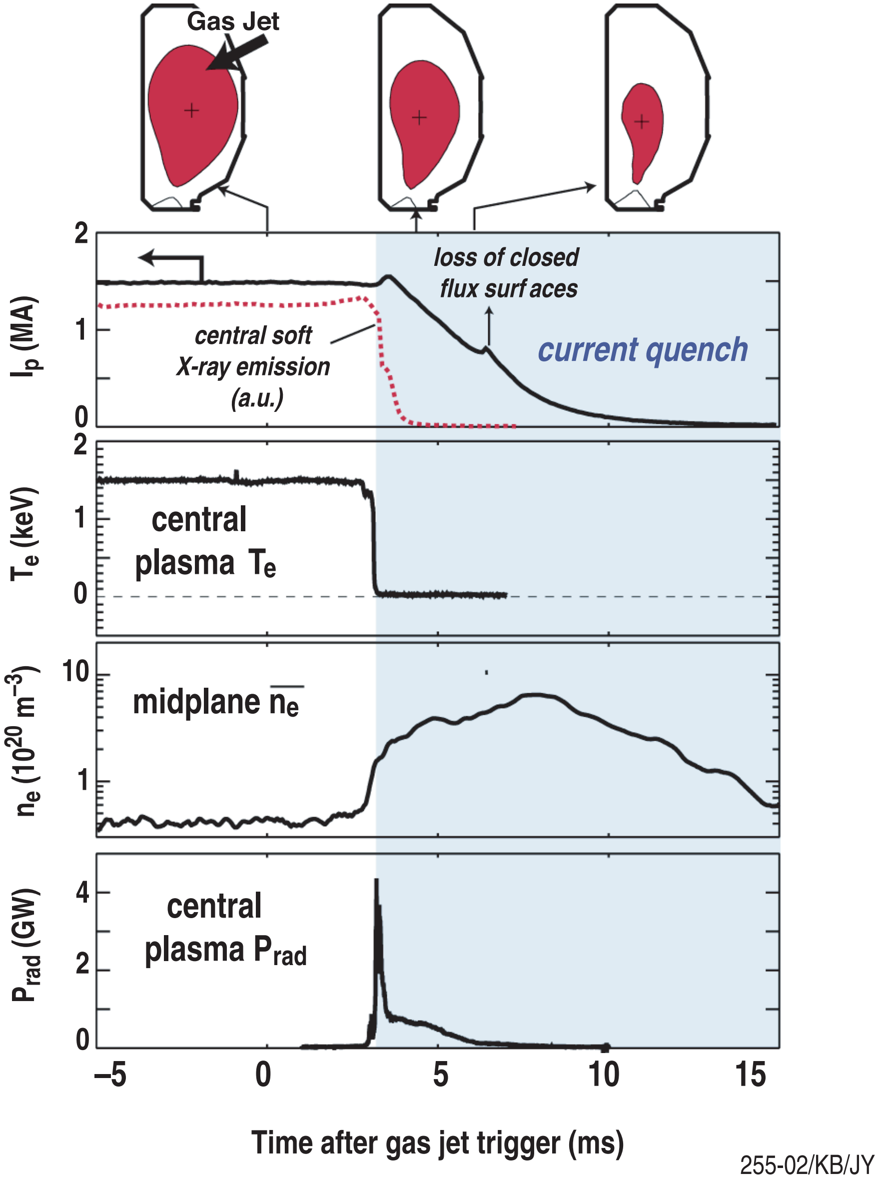
  $\approx 0.3\text{-}0.5$ (SN)
- Results simplify divertor design in future DN tokamaks

Modeling (UEDGE) indicates that particle drifts in the divertor play important roles in interpreting these results

DIII-D
NATIONAL FUSION FACILITY
SAN DIEGO

NEON/ARGON GAS JET IMPURITY INJECTION INTO A STABLE PLASMA RESULTS IN A RAPID, CLEAN PLASMA TERMINATION

- 70 bar gas jet propagates through plasma without significant MHD activity
- High radiated power from neon collapses central T_e and β
- Ten-fold increase in density
- Fast and clean current quench
  - No sign of non-thermal e$^-$ owing to high gas density
- Plasma remains well centered in vessel

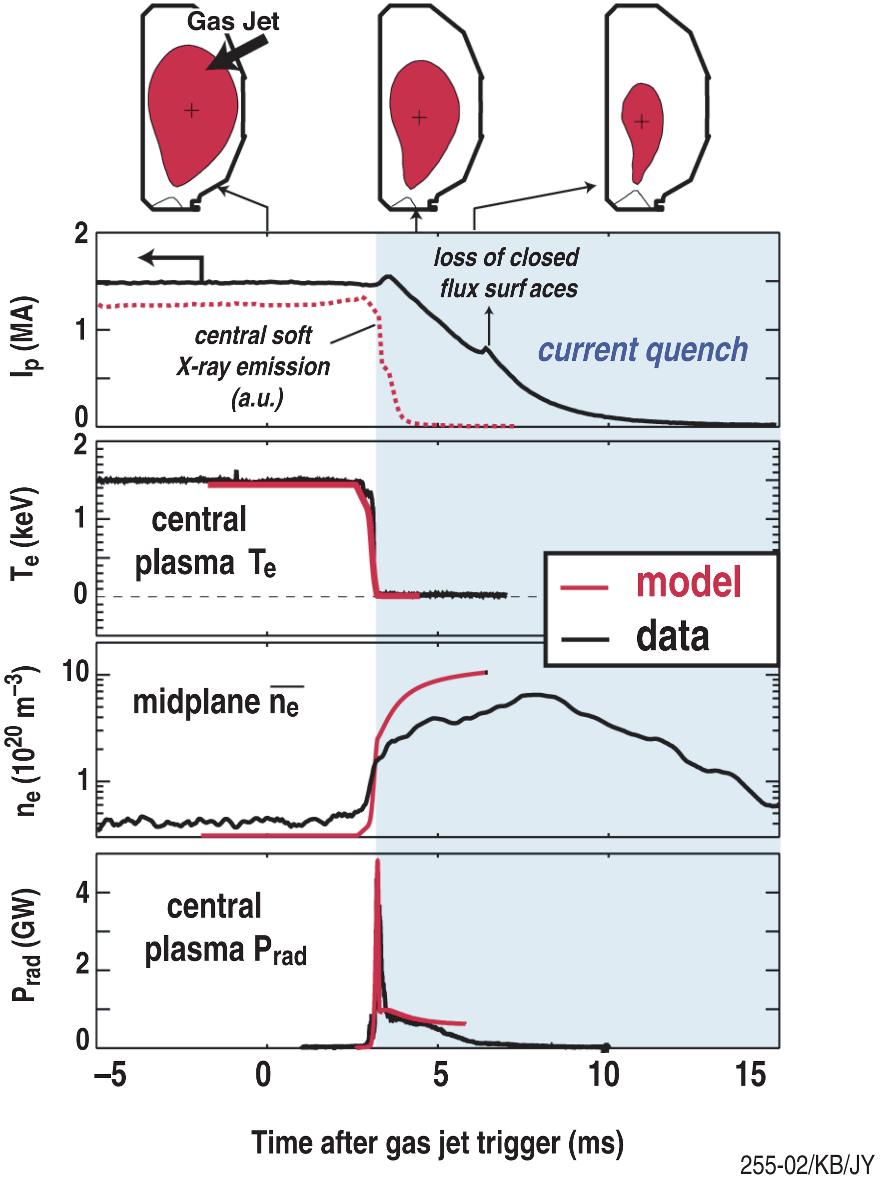


255-02/KB/JY

IONIZATION/ENERGY BALANCE MODEL (KPRAD) MATCHES KEY FEATURES OF GAS JET MITIGATION EXPERIMENTS:

INITIAL BURNTHROUGH → P_{rad} → T_e COLLAPSE → n_e CLAMPED

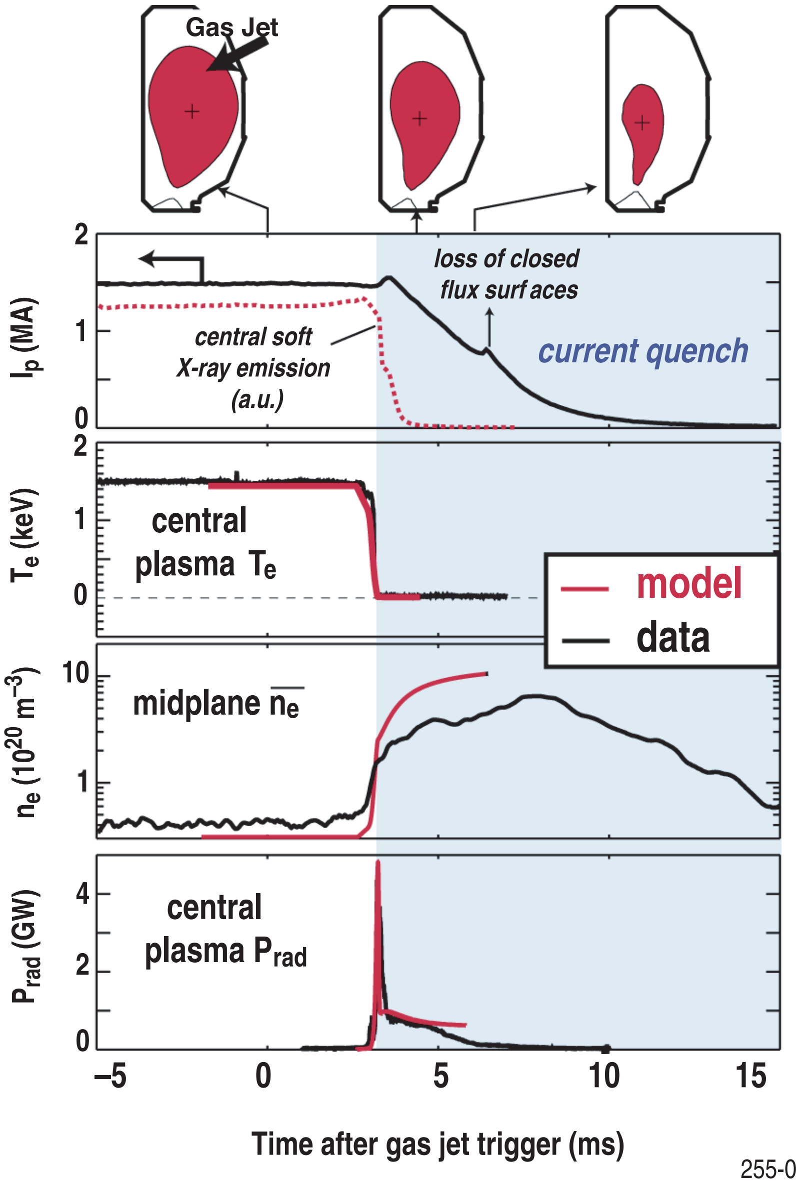
Model result:
Neon gas jet into DIII–D



255-02/KB/JY

IONIZATION/ENERGY BALANCE MODEL (KPRAD) MATCHES KEY FEATURES OF GAS JET MITIGATION EXPERIMENTS:

INITIAL BURNTHROUGH → P_{rad} → T_e COLLAPSE → n_e CLAMPED



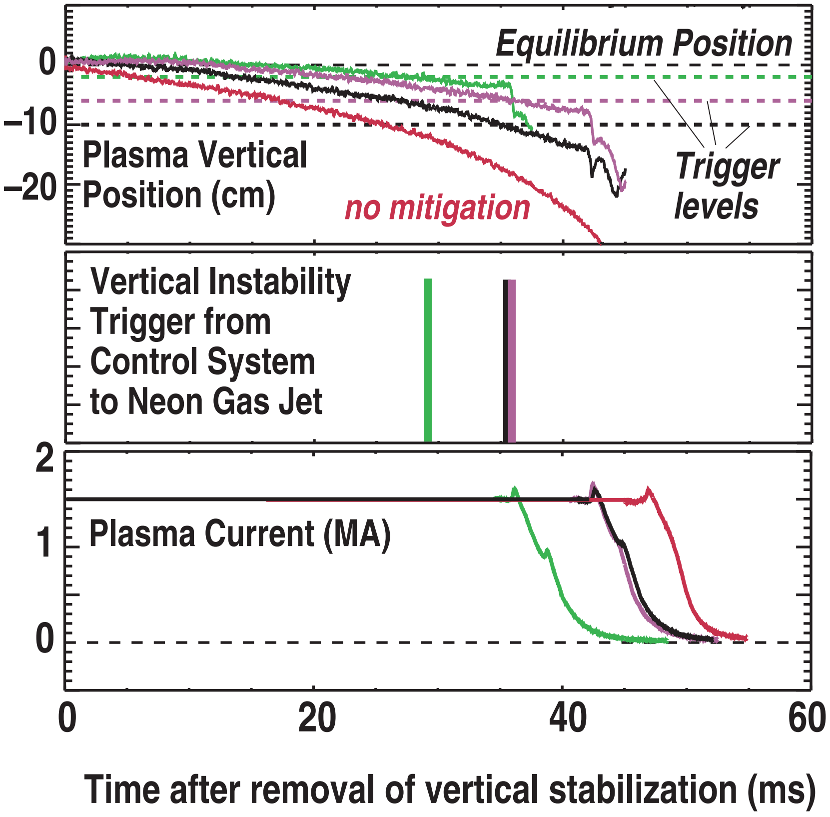
Model result:
Neon gas jet into DIII–D

Extrapolation to ITER is promising (see D.G. Whyte et al. EX/S2-4)

255-02/KB/JY

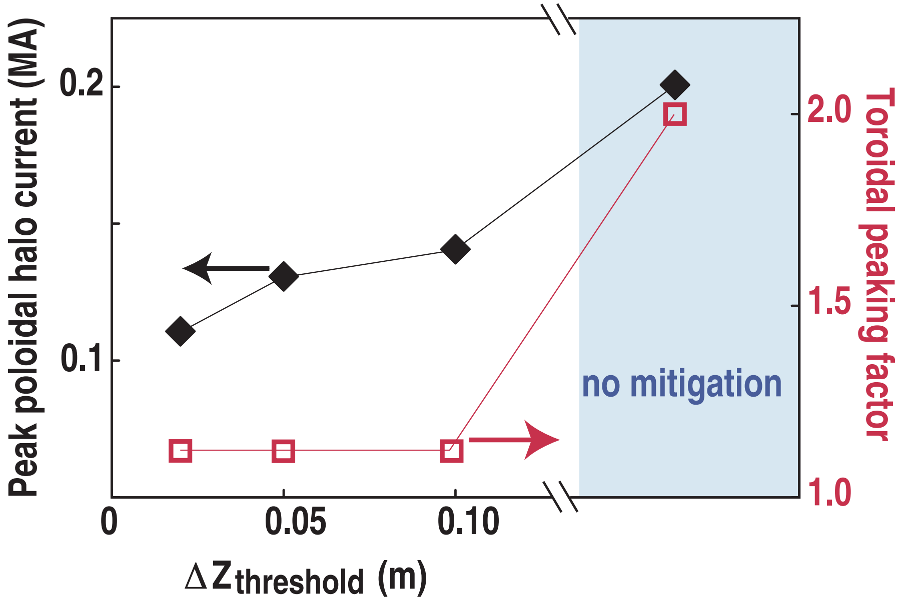
REAL-TIME DISRUPTION DETECTION IS BEING USED TO TRIGGER GAS JET FOR VERTICAL DISRUPTION MITIGATION

- Gas jet triggered when plasma control system detects vertical plasma shift



Divertor J×B forces from VDE reduced using gas jet mitigation

Rapid + centered current quench



255-02/KB/JY

SUBSTANTIAL PROGRESS SINCE IAEA 2000

We have made progress since the last IAEA meeting in developing the building blocks needed for advanced operating modes:

- Substantially broadened the MHD stable operating space
  - Rotational stabilization of resistive wall modes yielding $\beta_N = \beta_N$ (ideal wall) $\cong 2\ \beta_N$ (no wall)
  - Increased β by 60% via stabilization of (3,2) neoclassical tearing mode with ECCD in sawtoothing plasmas
  - First stabilization of (2,1) neoclassical tearing mode using ECCD
- Developed plasma control tools
  - First integrated AT discharges with current profile control using ECCD
  - Pressure and density profile control with ECH and ECCD
- Demonstrated an improved, high q_{95} (>4) operating scenario for ITER
- Achieved solutions to key burning plasma issues
  - No ELM-produced, pulsed divertor heat load in QH–mode plasmas
  - Small heat and particle loads at inner divertor strike points in balanced double-null divertors
  - Disruption mitigation via massive gas puff

ADDITIONAL PRESENTATIONS CONTAINING DIII–D RESULTS

Topic	Author	Paper
Feedback stabilization of NTMs with ECCD	R.J. La Haye	EX/S1-3
Stabilization of resistive wall modes	E.J. Strait	EX/S2-1
Modeling the stabilization of RWMs	M.S. Chu	TH/P3-10
Disruption mitigation by high pressure gas injection	D.G. Whyte	EX/S2-4
Sustaining steady-state AT discharges	M.R. Wade	EX/P3-16
Electron cyclotron current drive	C.C. Petty	EX/W-4
Electron cyclotron technology for plasma control	R.W. Callis	CT-7Rc
Scaling and modeling of high bootstrap tokamaks	F.W. Perkins	EX/P3-18
Internal transport barrier physics in QDB dischanges	E.J. Doyle	EX/C3-2
Turbulence stabilization by equilibrium and zonal flows	G.R. McKee	EX/C4-1Ra
Comparison of simulations with turbulence measurements	T.L. Rhodes	EX/C4-1Rb
Comprehensive gyrokinetic simulations	R.E. Waltz	TH/P1-20
Alternate ITER baseline scenario	T.C. Luce	EX/P3-13
DIII–D-like AT scenario for ITER	L.L. Lao	EX/P3-12
Transport modeling for burning plasma experiments	J.E. Kinsey	TH/P1-09
ELM stability, peeling-ballooning mode	P.B. Snyder	TH/3-1
H–mode pedestal width and neutral penetration	R.J. Groebner	EX/C2-3
Acceptable ELM Regimes for Burning Plasmas	A.W. Leonard	EX/P3-06
Turbulence in the SOL of C–Mod, DIII–D, and NSTX	J.L. Terry	EX/P5-10
Blobs and cross-field transport in the tokamak edge	S.I. Krasheninnikov	TH/4-1
The effects of drifts on the boundary plasma	G.D. Porter	EX/P3-07
H–mode pedestal and extrapolation to ITER (ITPA)	T.H. Osborne	CT–3
Transport and ITB physics (ITPA)	P. Gohil	CT/P–05