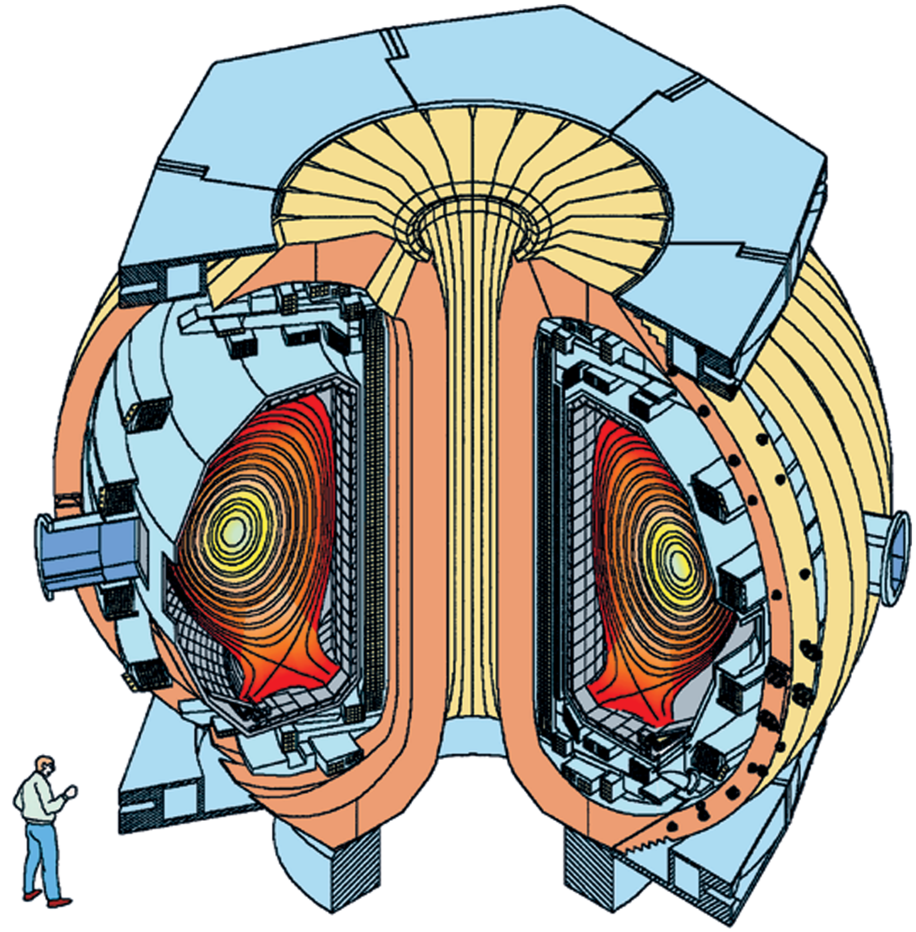
Control Issues Related to Startup of Tokamaks

by
Gary Jackson, General Atomics



Presented at
ITER International Summer School
Austin, Texas

June 1, 2010

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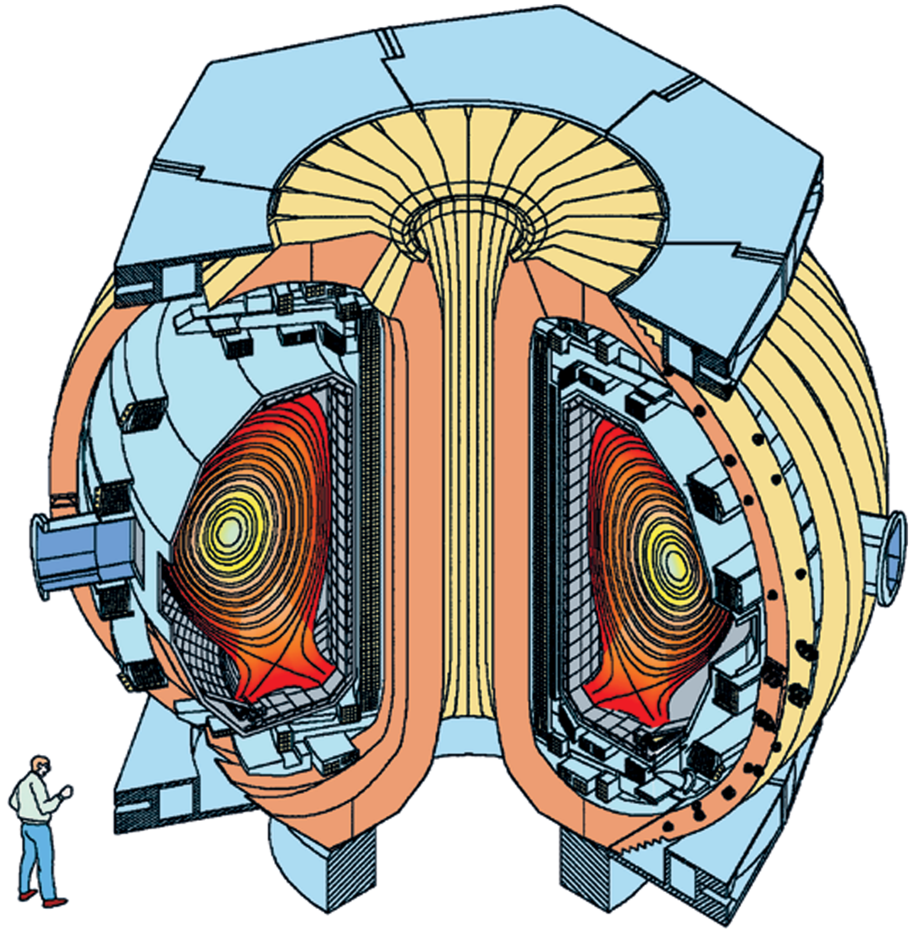


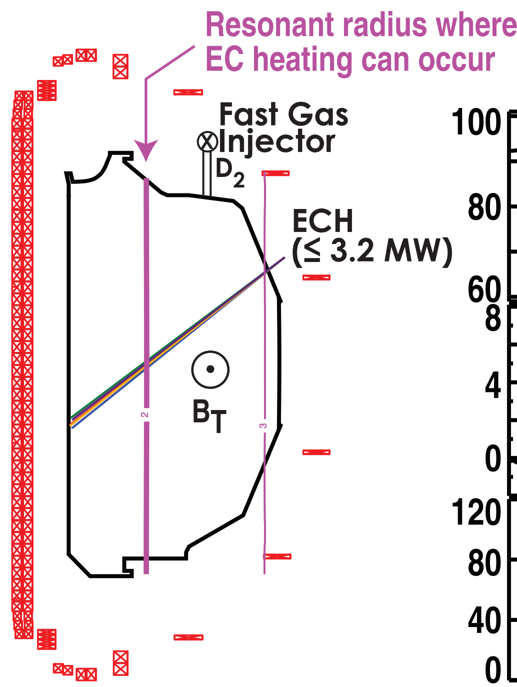
OUTLINE

- How does a tokamak start up?
  - 5 phases from collisionless heating => weakly ionized => fully ionized plasma with shaping control
  - Use the DIII-D tokamak as an example
- Simple example of feedback control: Plasma current (I_p)
- Radial and Vertical Position feedback control during initial I_p ramp
- Isoflux shape control
- "Issues" for startup in the DIII-D tokamak
- ITER startup challenges
- Simulating ITER startup in DIII-D
  - Improving the ITER baseline startup scenario
  - Internal inductance feedback to maintain adequate vertical stability
- What we've learned (with applications for ITER)

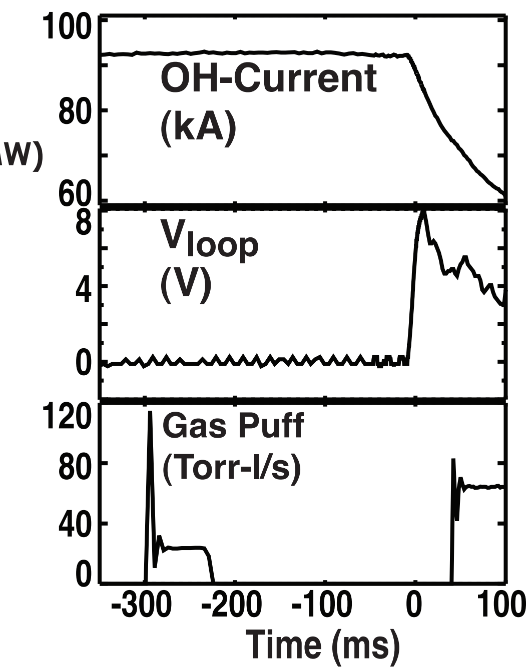
How Does a Tokamak Start Up?

- **Needs a source of electrons**
  - Cosmic rays (timing is too unreliable)
  - Photoionization (not routinely used)
  - Local electron source (ionization gauge in a port is sufficient)
  - RF: electron cyclotron (EC) initiation planned for ITER
  - Other sources (helicity injection, etc.)
- **Like an automobile, a tokamak needs spark and gas to start**
  - Spark: Either Ohmic heating (OH) or RF (typically ECH)
  - Gas: Hydrogen or deuterium "prefill" with fast valve
- **.... But a tokamak also needs toroidal field for confinement**

- **Ohmic Heating (OH) Coil a.k.a. "E-Coil"**
- **Transformer (48:1) produces toroidal loop voltage, V_{loop}**
- **$B_{Toroidal}$ = 2.1 T (not shown)**



DIII-D



Typical Ohmic Start-up Sequence to Obtain An Upper Single Null Discharge

Time	Event
	charge OH solenoid (E-coil)
–300 ms	Begin deuterium prefill
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–79 ms	Start driving outer field shaping coils to targets (Breakdown is earlier with EC asist)
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–3 – +8 ms	Time range for breakdown and I_p formation
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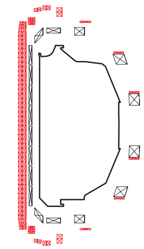
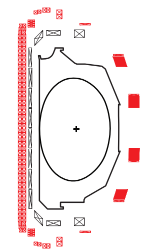
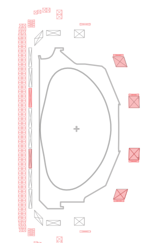
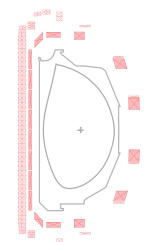
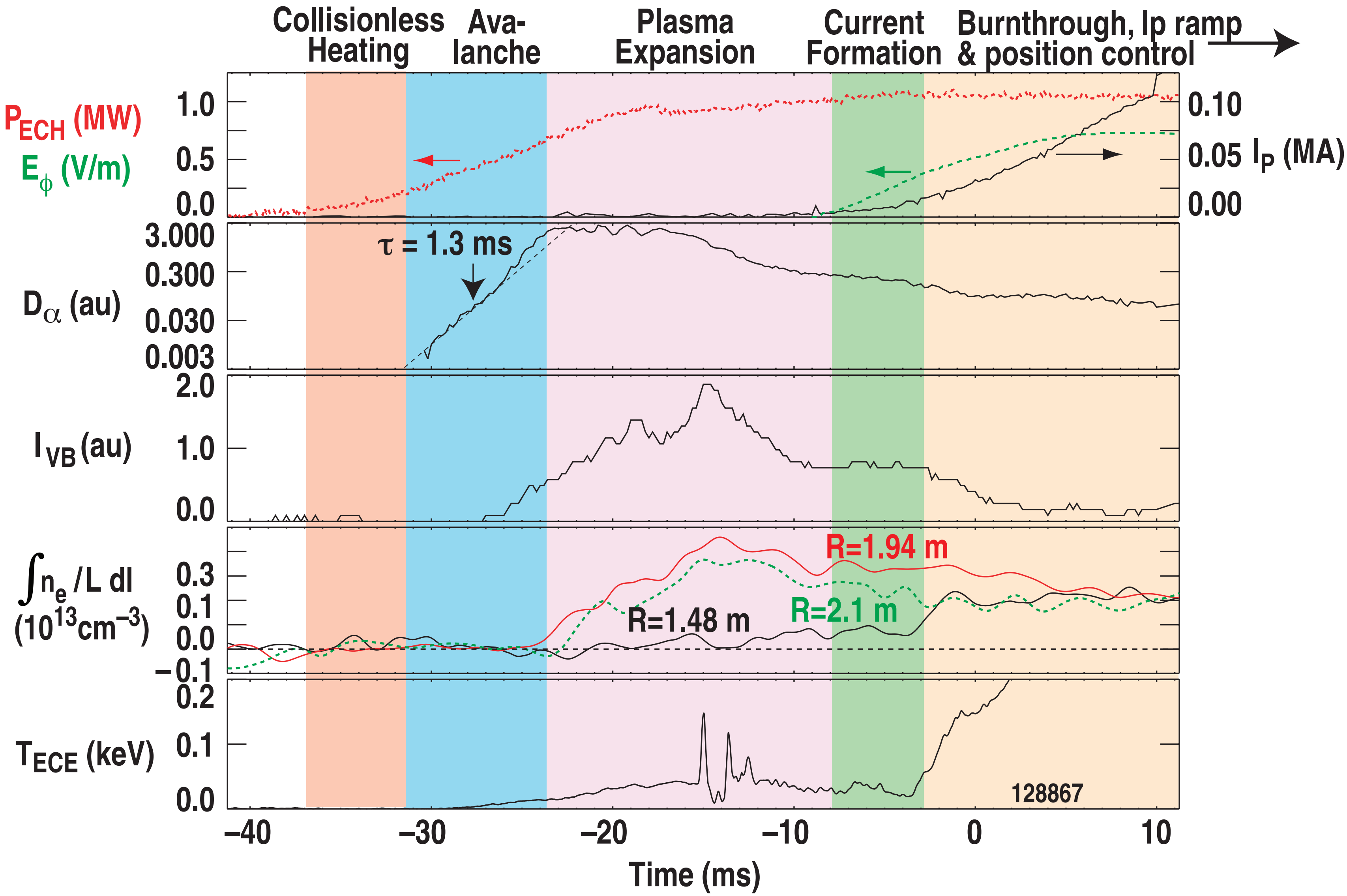
200 ms	Initiate density feedback
280 ms (typ)	Divert
1000–1600 ms	Current flattop

Collisionless
Avalanche
Expansion
Ip Formation
Burnthrough, Feedback, Ip ramp

Tokamak Startup Can be Divided into Five Phases
— EC Delineates These Phases More Clearly than OH Alone



Jackson, et al., Fus. Sci & Technol.,**57**(2010)27

Collisionless Heating

- **Ohmic Heating: A few (order of 1 cm^{-3}) electrons are heated**
  - Many transits around the tokamak as electrons follow field lines and gain energy (neglecting drifts and collisions).
  - Typically for an Ohmically heated (OH) scenario, $L_{wall} \geq 2000$ m

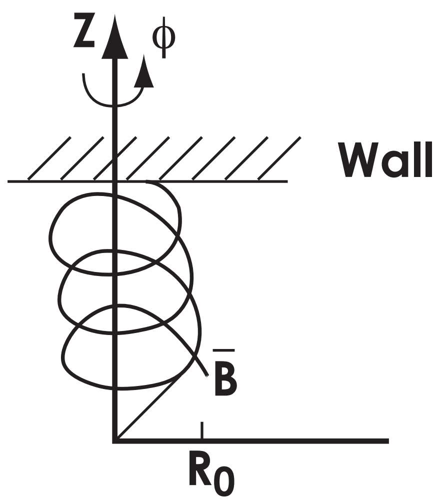
- **Example (for DIII-D):**
  - $2\pi R_0 = 11$ m
  - $V_{loop} = 4$ V (at time of breakdown)
  - $\mathcal{E}_{electron} \approx 2000/11 \times 4 = 730$ eV (before striking the wall)
    - Well above threshold for ionizing collisions (≈ 20 eV for hydrogen and deuterium)

- **ECH (or other types of RF) can also heat electrons**
  - Heating efficiency can be low at low T_e, so hundreds of kW are usually required
  - Heating is confined to a small region where beam intersects EC resonance
    - Electrons then drift to wall

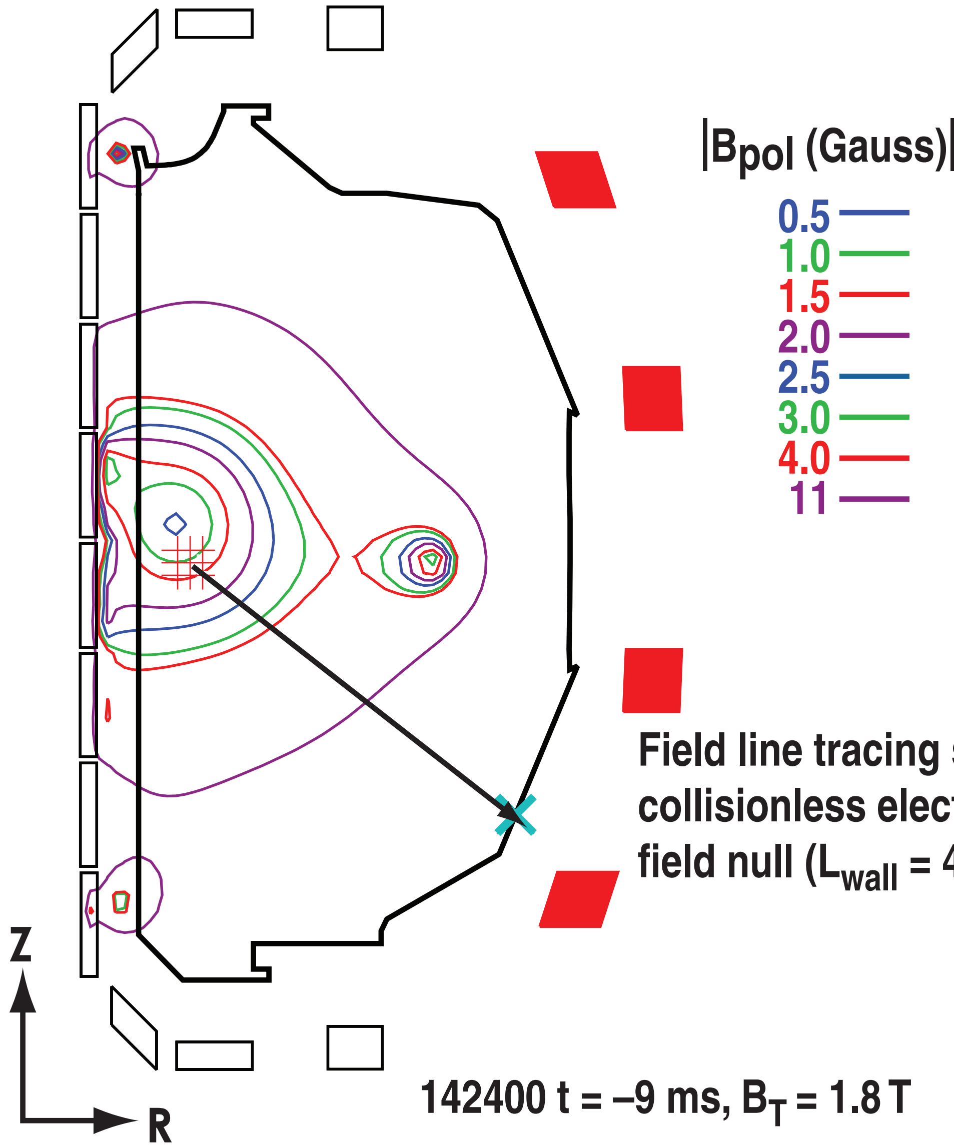
Plasma control during the earliest phases ($I_p \leq 40$ kA) is accomplished with feedforward control

- Coil currents to 4 outer field shaping coils are calculated based on current in the OH transformer and desired position of the poloidal magnetic field null
  - Field null is necessary so electrons have a long path for heating and ionizing collisions before striking the wall
  - Measurements taken at t=-80 ms
  - Commands to coils calculated using multipole expansion of desired poloidal fields (Lazarus, et al., Nuc. Fus. 38(1998)1083)
  - Current ratio between coils ensures the proper field index, $R/B_Z \times dB_Z/dR|_{R=R0}$, for positional stability as plasma current forms

Feed Forward Programming Provides a Region of Small B_{pol}, Allowing Long Connection Length to the Wall

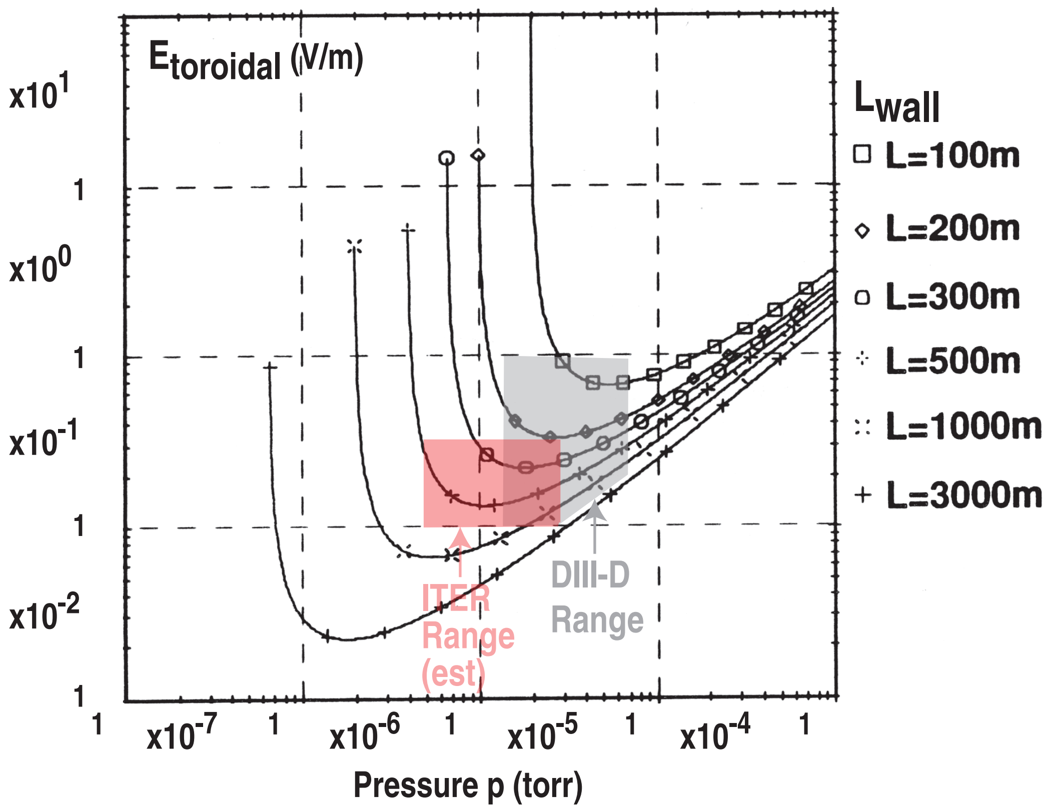


- Connection length, L_{wall} is determined by a magnetic field line spiraling to the wall
- $\mathcal{E}_{electron} = \int_{L_{wall}} q\,\bar{E} \cdot d\hat{\ell}$

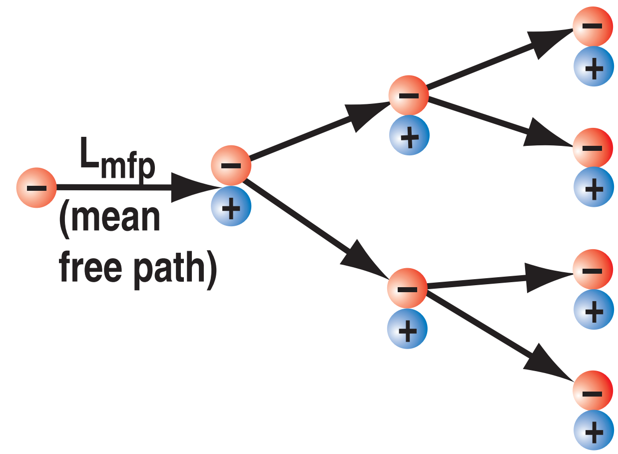


Field line tracing shows strike point for collisionless electrons originating near field null (L_{wall} = 4.3 km)

With Neutral Gas, Ionizing Collisions Occur, Producing an Avalanche



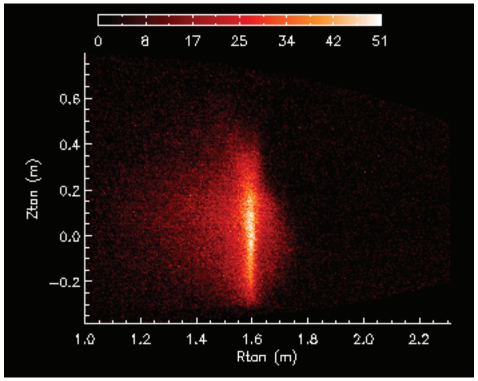
(Generally, $L_{mfp} << L_{wall}$)



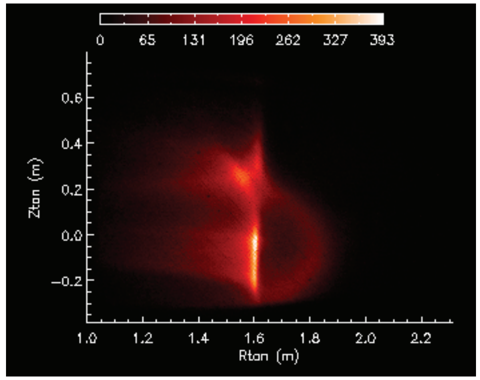
- DIII-D has a wide operating space for breakdown
- A successful avalanche phase does not ensure start-up
- ITER parameter space is more limited due to lower inductive electric field, 0.3 V/m

Lloyd et al., Nucl. Fus 31(1991) 2031

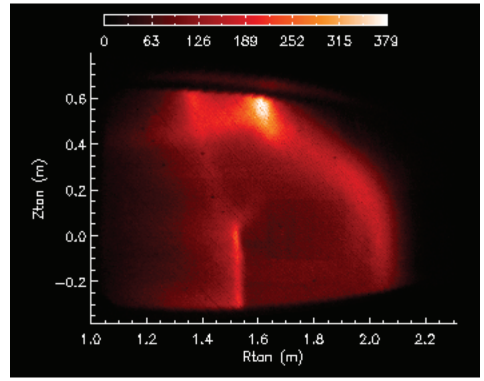
Plasma Formation and Evolution is Observed by a Fast Camera, Viewing C^{III} Emission



t= -12.7 ms, Ip =1.8 kA, V_L= 0V

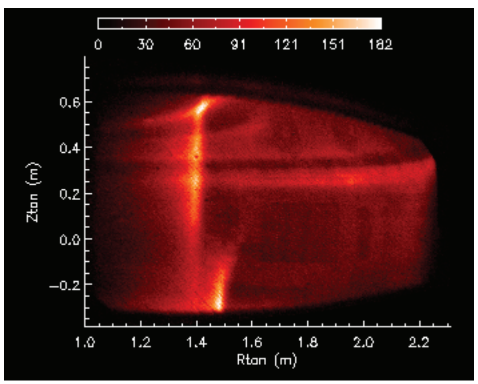


-9,3 ms, 1.9 kA, 0V

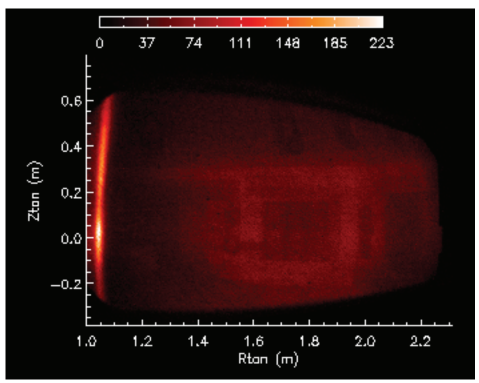


-4.3 ms, 5.6 kA, 0.6V

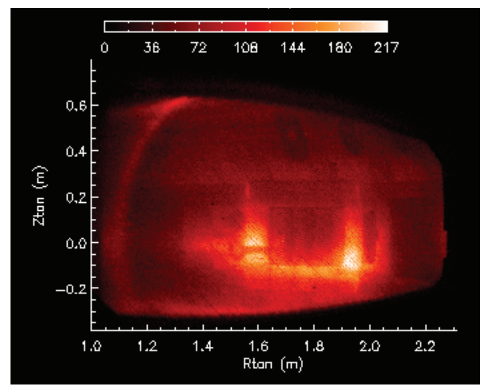
R_{IW}(midplane) = 1.02 m
R_{X2} = 1.64 m
R_{OW}(midplane)=2.36m
B_ϕ=1.9T, V_L=3V, $B_{V,pgm}$=-30G
$C^{III}_{ionization}$ = 48 eV
$C^{III}_{burnthrough} \approx$ 16-24 eV
135899



+4.0ms, 25kA, 2.6V



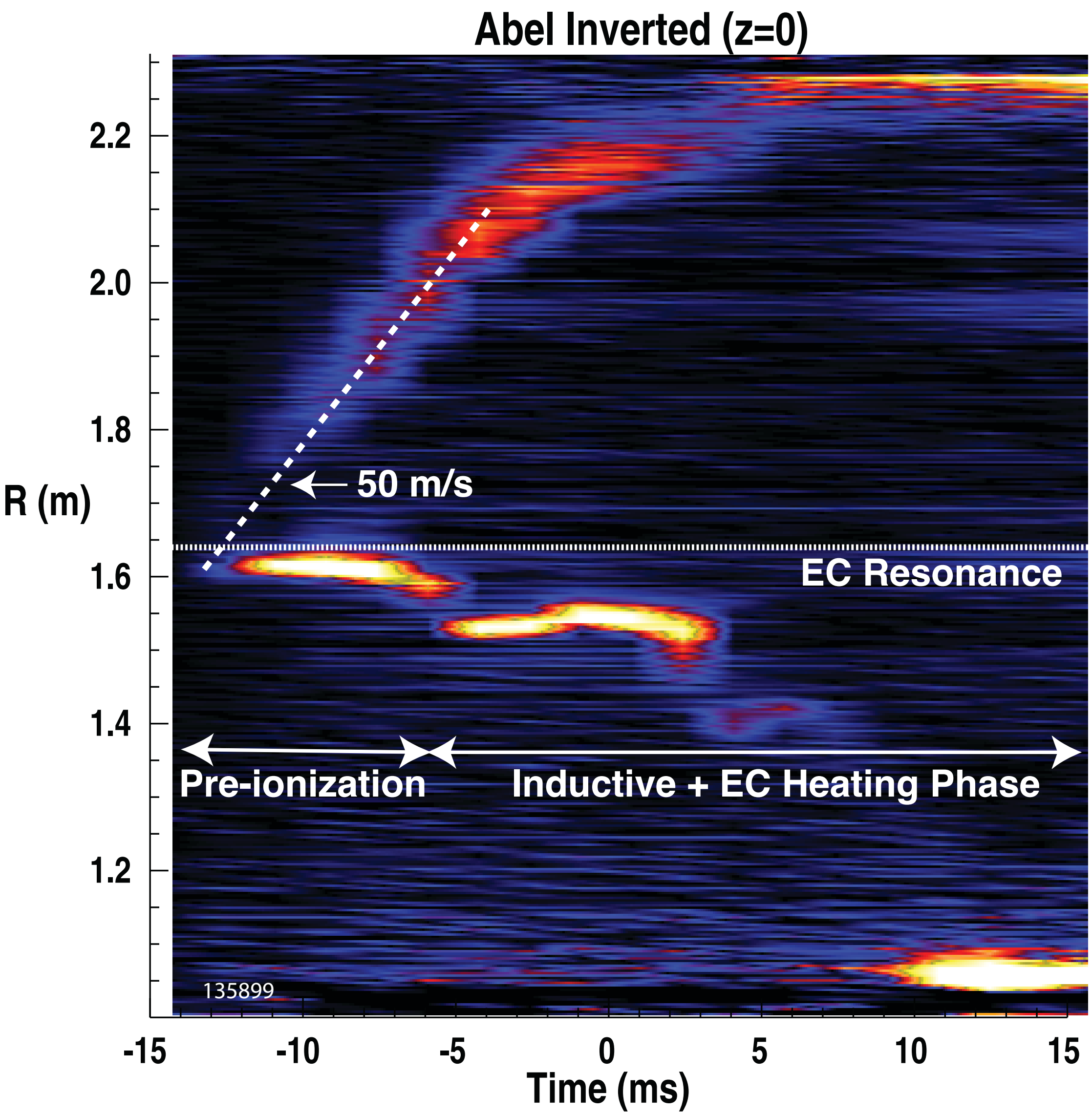
t= 12 ms, Ip 61kA, V_L= 3.0V



+39 ms, 98kA, 3.0V

J.H .Yu

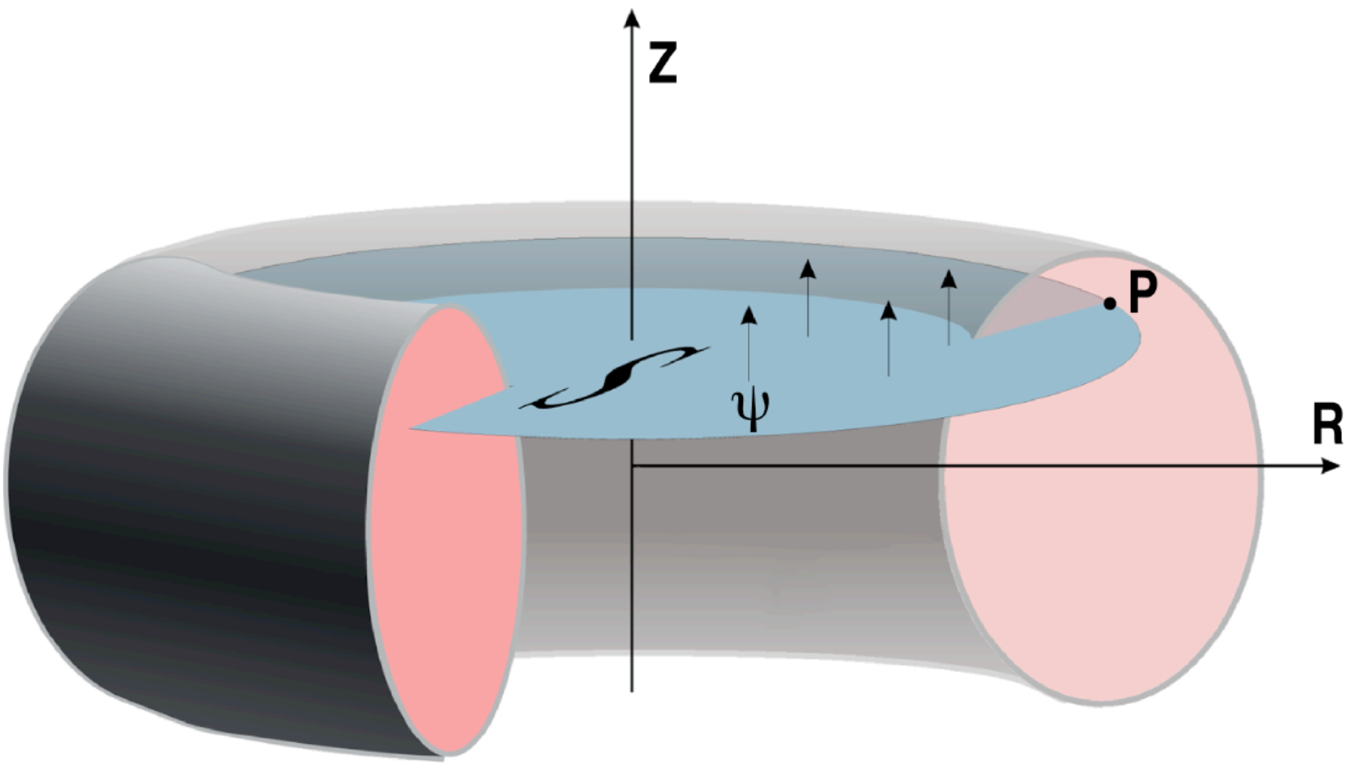
ECH Allows Breakdown to Initiate Near the Vessel Center and Initially Expand Outward



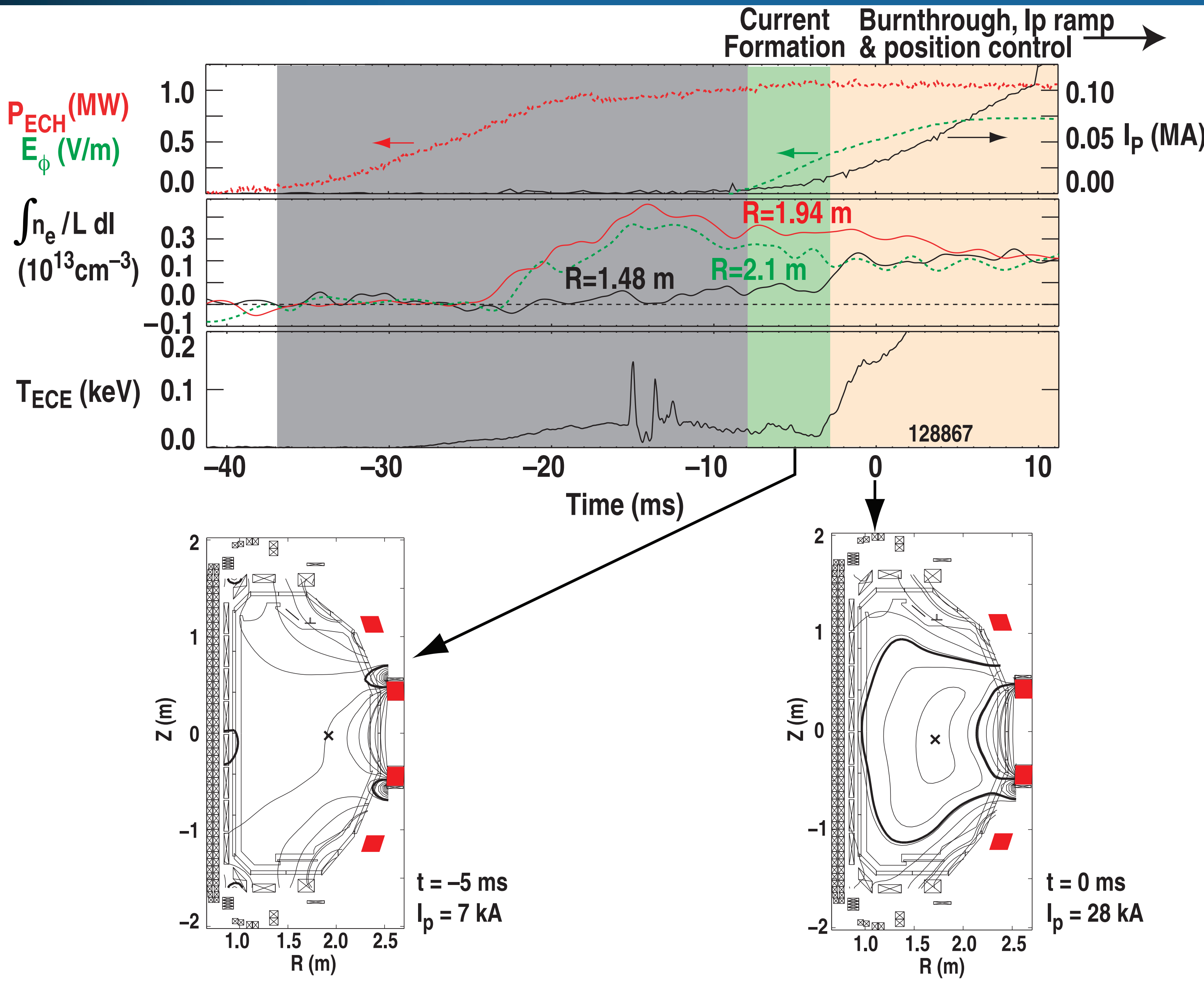
- Breakdown initiates near the EC resonance
- Abel inversion shows initial plasma expansion (due to *ExB*) at nearly constant velocity
- E due to charge separation from grad(B) and curvature drifts
- $v_{expansion} \approx 50$ m/s (P_{EC} = 1 MW)
  $v_{expansion} = \bar{E}x\bar{B}/|B|^2$
  => E_Z = 95 V/m
- During the Ohmic heating phase, plasma expands inwards in discrete steps

Definition of Poloidal Flux

- Assume axisymmetric plasma (same at all poloidal angles f)
- P = point in the poloidal cross-section; y(P) = magnetic flux through surface S bounded by ring through P
- Defines the *poloidal flux function* y(R,Z)



At $I_p \approx 20$ kA, Closed Flux Surfaces form and Confinement Increases



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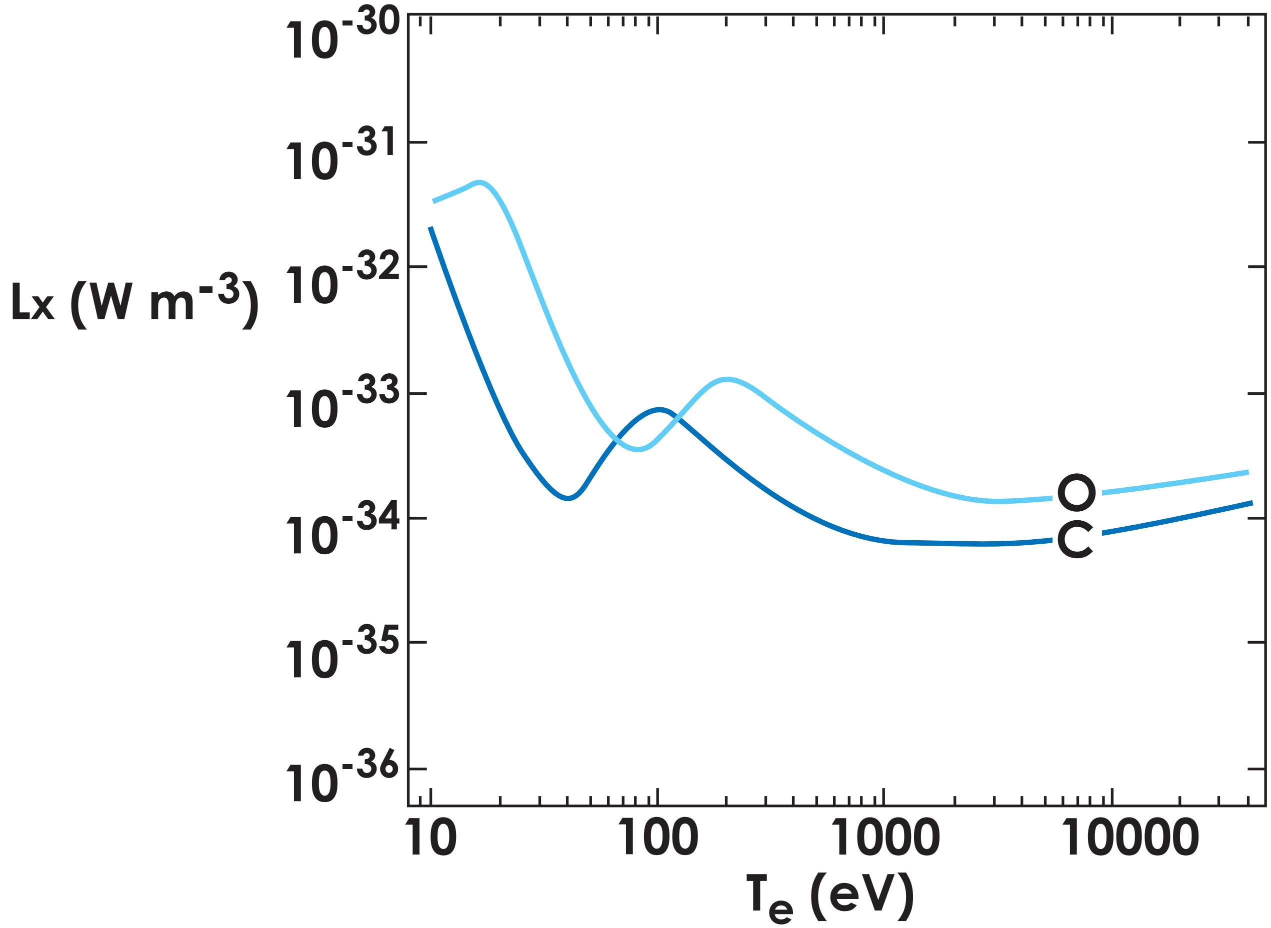
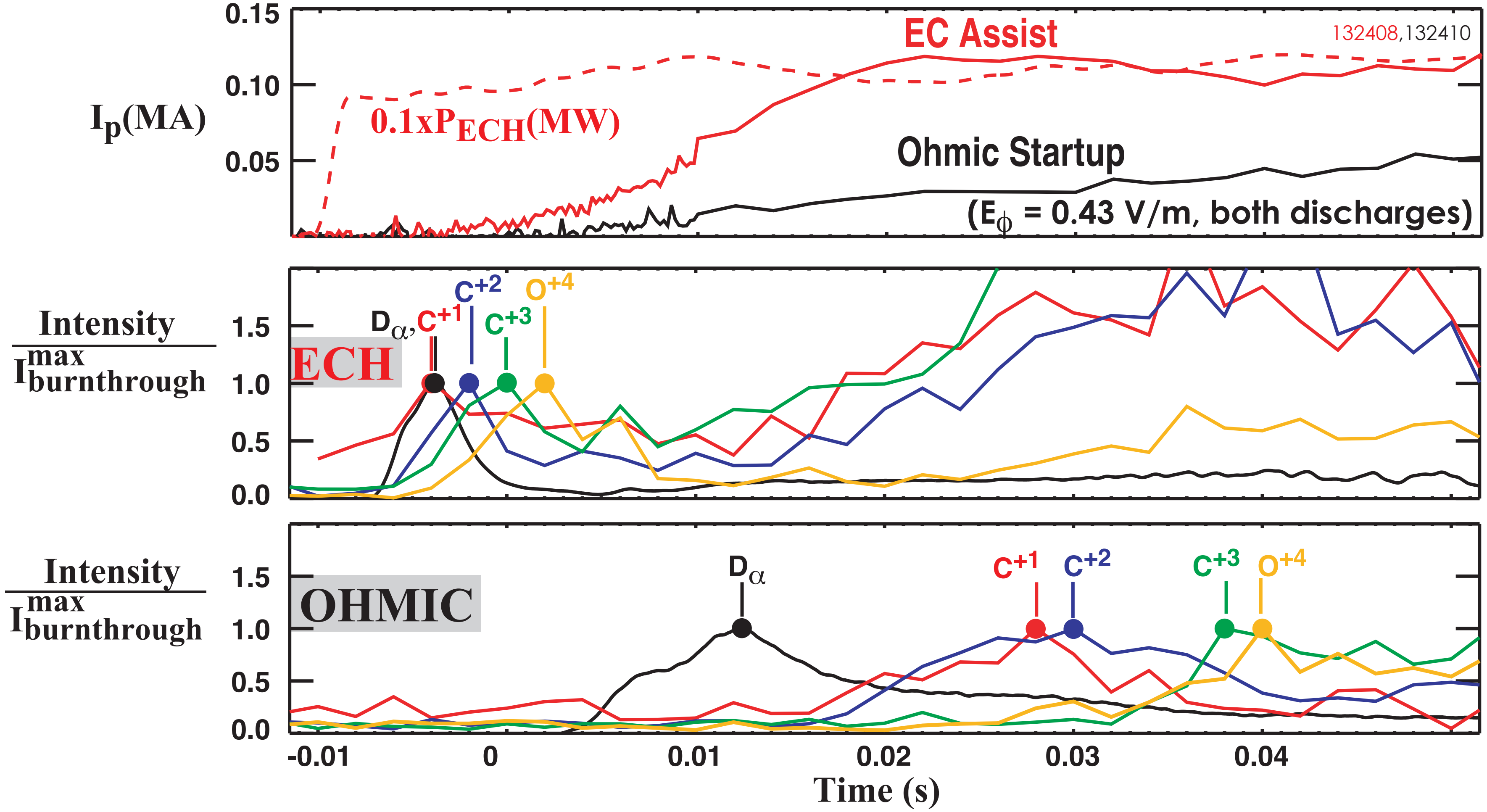
"Burnthrough" in Tokamaks Occurs when Radiation from Carbon and Oxygen Significantly Decreases with Increasing T_e

- Successful burnthrough requires $P_{in} = P_{RF} + P_{EC} > P_{rad}$
- Radiation function $L(T_e)$, resulting power density: $P_{rad} = L*n_e*n_Z$



BREAKDOWN AND BURNTHROUGH ARE MORE PROMPT AND MORE REPRODUCIBLE WITH EC ASSIST

- **Example: charge states D_α, C^{+1}, C^{+2}, C^{+3}, O^{+4} indicate increasing ionization potential**



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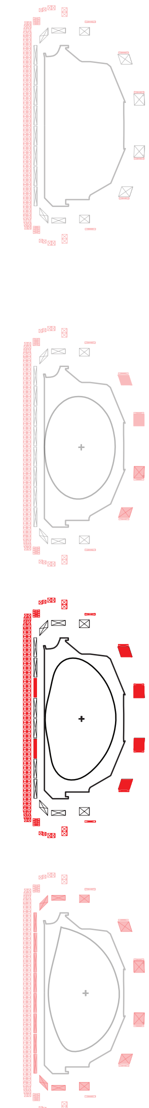
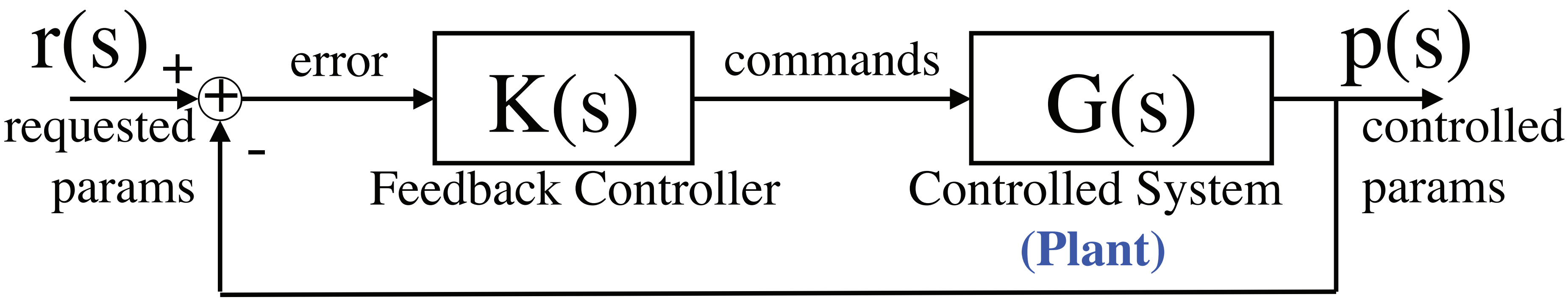
Phases: Collisionless, Avalanche, Expansion, Ip Formation, Burnthrough, Feedback, Ip ramp



Objectives of Control – Tracking and Regulation

- **Derivation of Closed-Loop Transfer Function:**



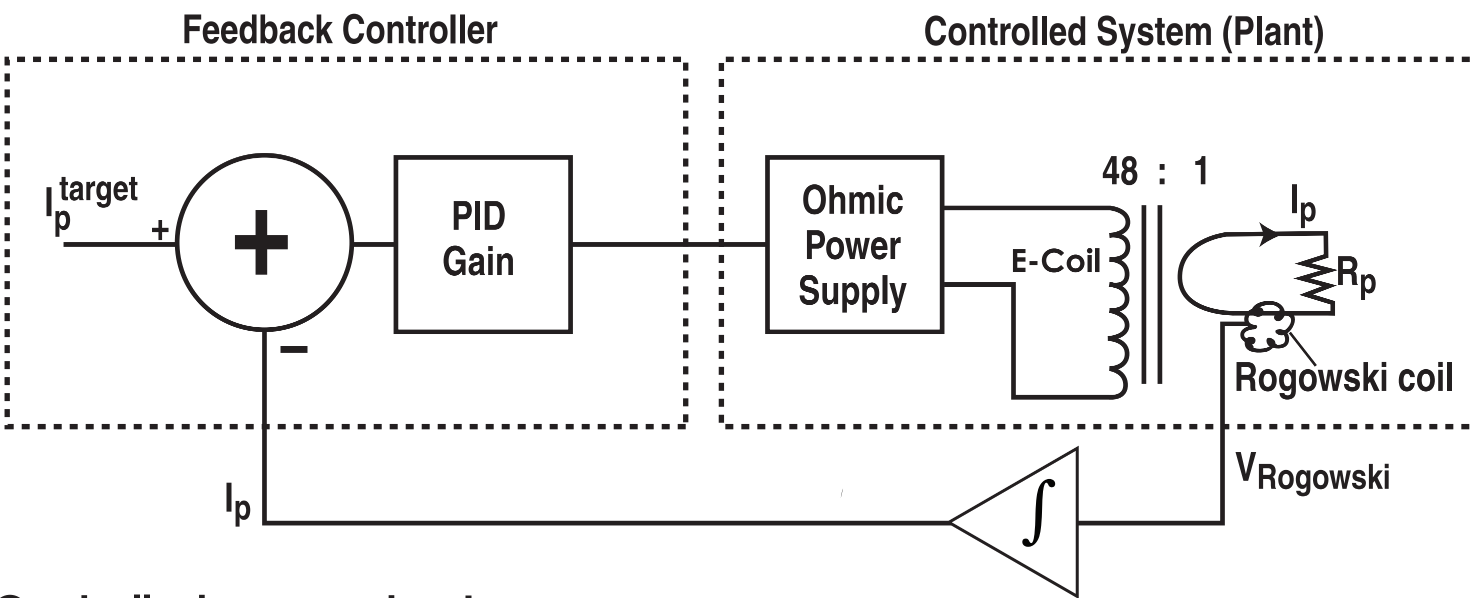
$$p(s) = G(s)K(s)(r(s) - p(s))$$

$$(I + G(s)K(s))\,p(s) = G(s)K(s)r(s)$$

$$\Rightarrow \quad \frac{p(s)}{r(s)} = \frac{G(s)K(s)}{(I + G(s)K(s))}$$

Mike Walker, May 31, 2010

Plasma Current Feedback is the First Step in Feedback Control of a Tokamak Discharge



Controlled parameter: I_p

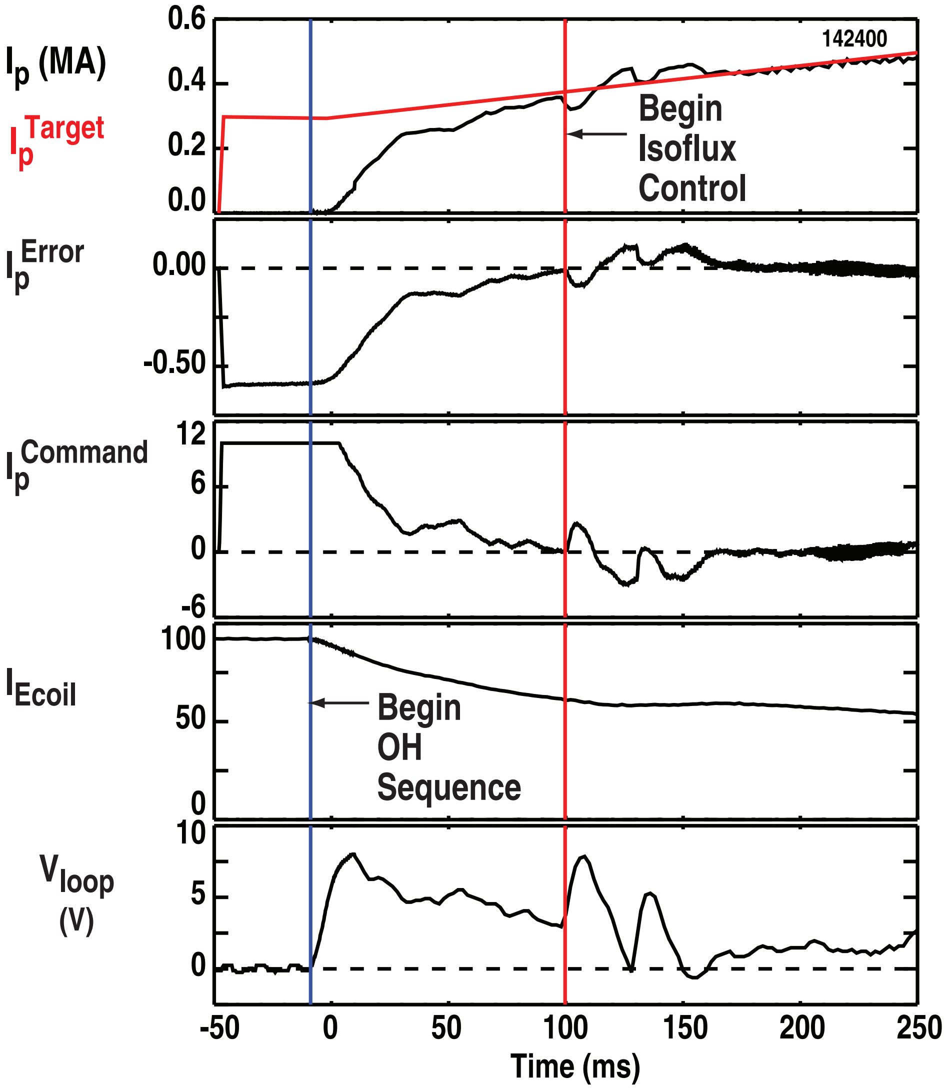
Measured with a Rogowski coil, cross section, A and N turns/m

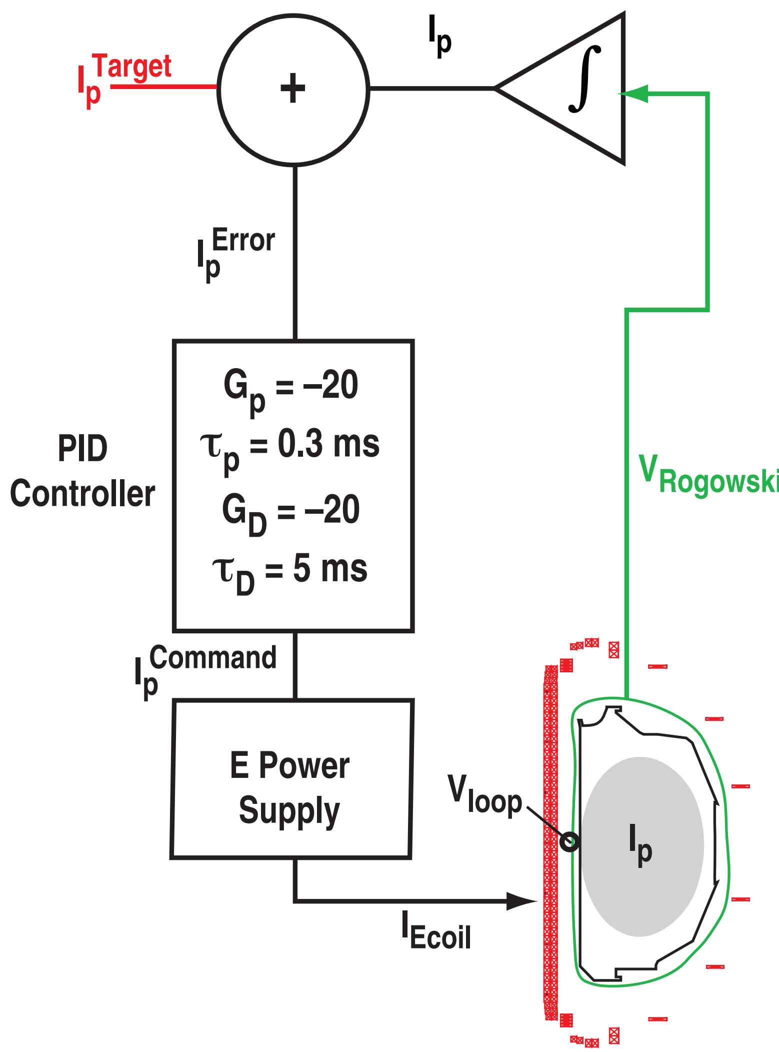
Amphere's Law (coil encircles the plasma)

$$V_{Rogowski} = NA \int \dot{B} \cdot \hat{\ell}\, d\ell = NA\, \mu_o \dot{I}_p$$

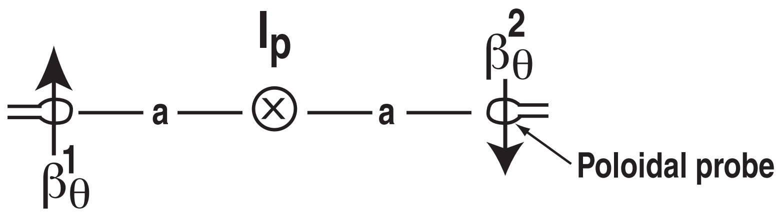
$$I_p = \frac{1}{\mu_o NA} \int V_{Rogowski}\, dt$$

Plasma Current Feedback is the First Step in Control of Tokamak Discharge (Con't.)





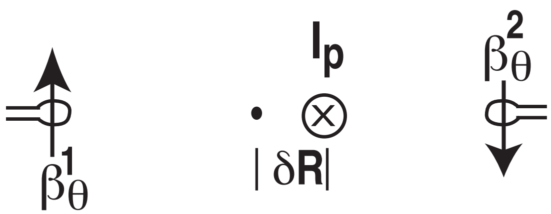
Radial Position Control — Basic Concepts



For a cylinder, probe signals are balanced

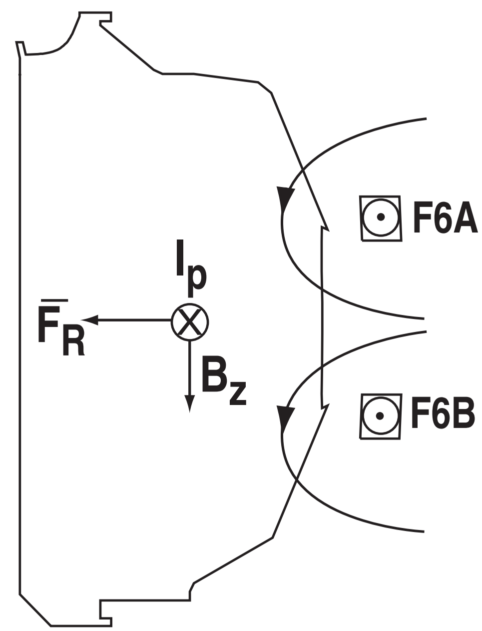
$$\Delta\beta_\theta = \beta_\theta^1 - \beta_\theta^2 = 0$$

Now, consider a displacement, δR



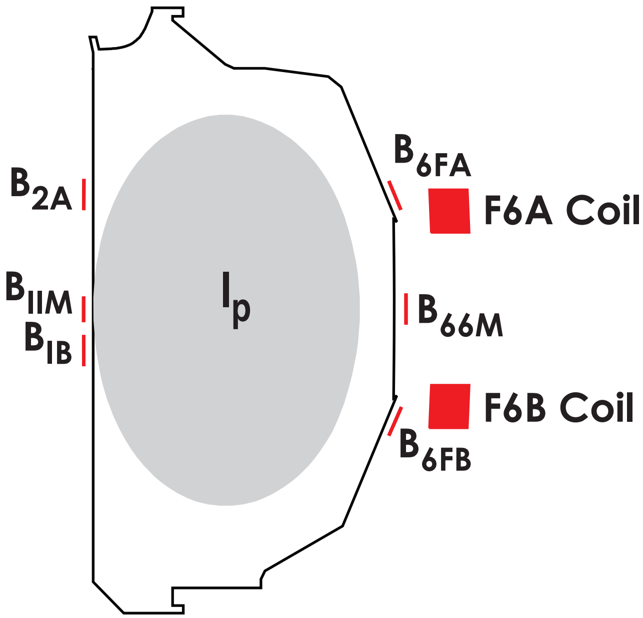
$$\Delta\beta_\theta = -\frac{\delta R}{\pi a^2}\mu_o I_p$$

- Outer Coils can provide a vertical field, producing a padial restoring force



Radial Position Control

Radial Position Control During the Early Phase (15 to 100 ms) Uses Multiple Probes



Radial Position Control

$$DFR_1 = fct(B_{11M}\,,\ B_{2A}\,,\ B_{1B})$$

$$DFR_2 = fct(B_{6FA}\,,\ B_{F6B}\,,\ B_{66M})$$

$$\delta_R \propto DFR = c_1 \text{ x } DFR_1 \text{ x} R_p^{target} - c_2 \text{ x } DFR_2$$

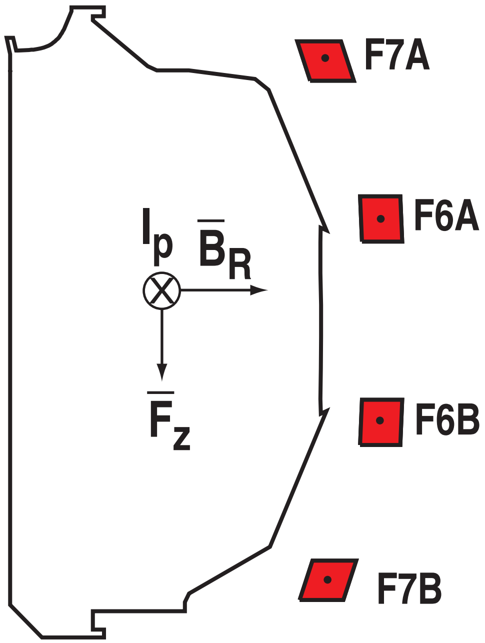
- Initial position feedback error calculated from “M” matrix and δ_R estimate

$$\overline{F}_{error} = \overline{\overline{M}}\ \overline{DFR}$$

where $\overline{DFR}$ is a column error vector calculated from sensor inputs and target waveforms
$\overline{\overline{M}}$ is a gain matrix $\overline{F}_{error}$ is error signal for each F-coil

Vertical Position Control

- Similar analysis using different sensors can be used to calculate a vertical displacement, δz
- δz error signals for 4 outer Field-shaping coils (F-coils) are calculated by the Plasma Control System (PCS)
- δR and δz are feedback controlled simultaneously using linear super-position



Vertical Position Control

Radial and Vertcical Control Matrix for Initial Plasma Current Ramp (10 to 100 ms)

F_{error} — Radial Control: DFR; Vertical Control: F6AMOD, F6BMOD, F7AMOD, F7BMOD

$$\overline{F}_{error} = \overline{\overline{M}}\,\overline{E}$$

E

	DFR	F6AMOD	F6BMOD	F7AMOD	F7BMOD
F1A	0.	0.	0.	0.	0.
F2A	0.	0.	0.	0.	0.
F3A	0.	0.	0.	0.	0.
F4A	0.	0.	0.	0.	0.
F5A	0.	0.	0.	0.	0.
F6A	-100.00	100.00	0.	0.	0.
F7A	0.	0.	0.	100.00	0.
F8A	0.	0.	0.	0.	0.
F9A	0.	0.	0.	0.	0.
F1B	0.	0.	0.	0.	0.
F2B	0.	0.	0.	0.	0.
F3B	0.	0.	0.	0.	0.
F4B	0.	0.	0.	0.	0.
F5B	0.	0.	0.	0.	0.
F6B	-100.00	0.	100.00	0.	0.
F7B	0.	0.	0.	0.	100.00
F8B	0.	0.	0.	0.	0.
F9B	0.	0.	0.	0.	0.

M

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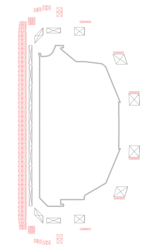
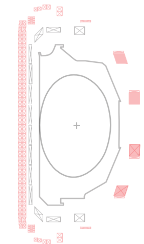
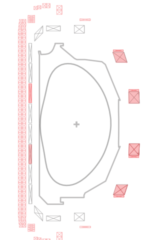
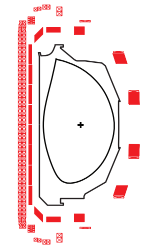


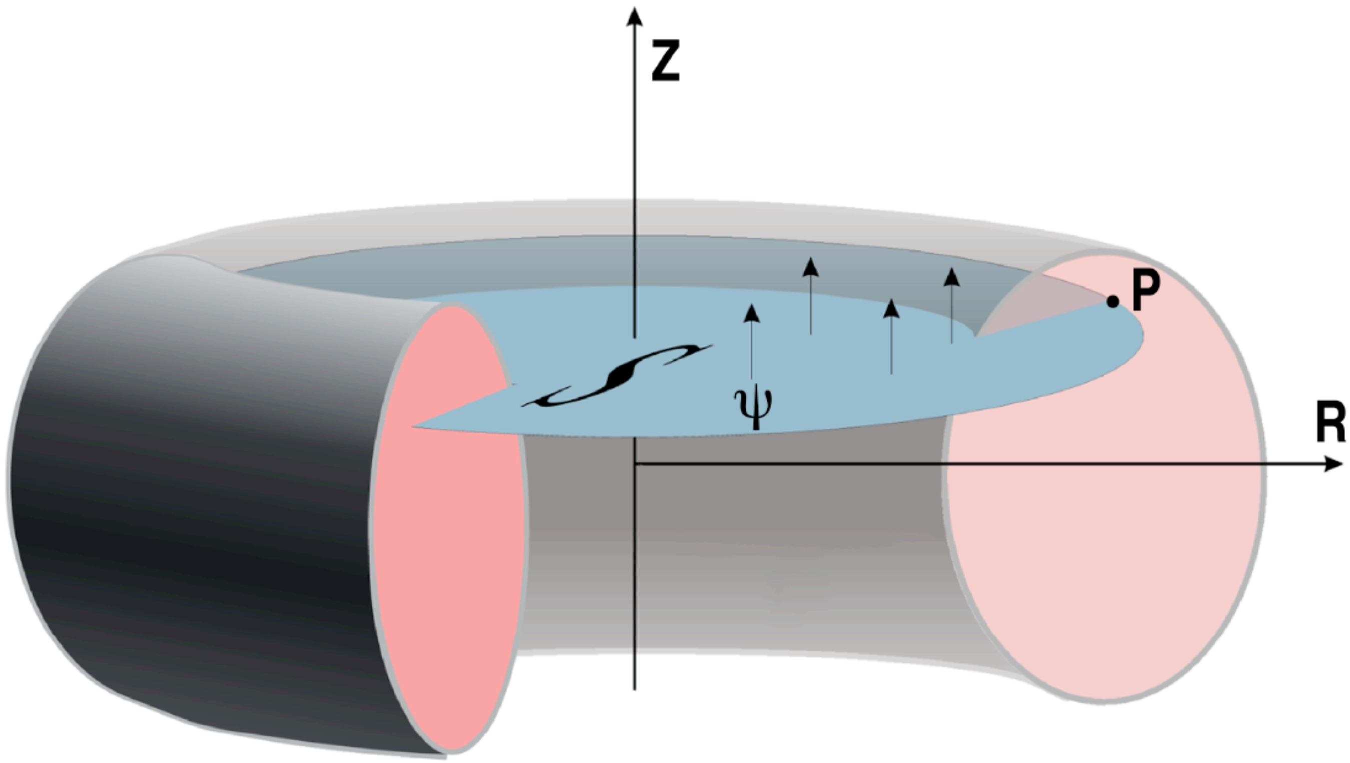






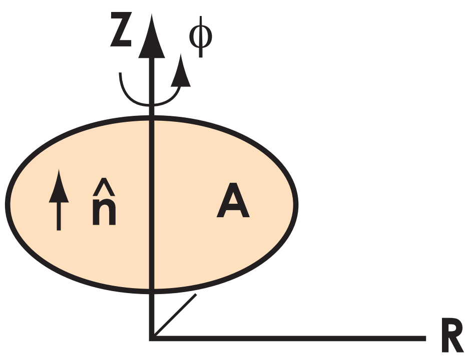
Definition of Poloidal Flux

- **Assume axisymmetric plasma (same at all poloidal angles f)**
- **P = point in the poloidal cross-section; y(P) = magnetic flux through surface S bounded by ring through P**
- **Defines the *poloidal flux function* y(R,Z)**



ISOFLUX Shape Control

- **Plasma shape in DIII-D is precisely controlled using an "Isoflux" formulation calculated in real time**
- **Plasma flux poloidal flux is defined as**



$$\overline{B}(R,z) = \sum_j \overline{B}_j \bullet \hat{n}(\overline{R} - \overline{R}', \overline{z} - \overline{z}') \quad = fct(F\text{-}coil,\ E\text{-}coil,\ or\ j_{plasma}\ at\ R_j',\ Z_j')$$

$$\Psi \equiv \int dA \overline{B} \bullet \hat{n}$$

$$\psi_{DIII-D} \equiv -\frac{1}{2\pi}\Psi = \quad -\int 2\pi R\, B_z(R,z)\, dR\ \ (for\ A\ in\ Z\ plane)$$

- **For a diverted configuration, the flux value at the X-point, ψ_{Xpt} is determined**
- **The plasma shape is defined by the last closed flux surface (LCFS) where,**
  $\psi_{LCFS} = \psi_{Xpt}$

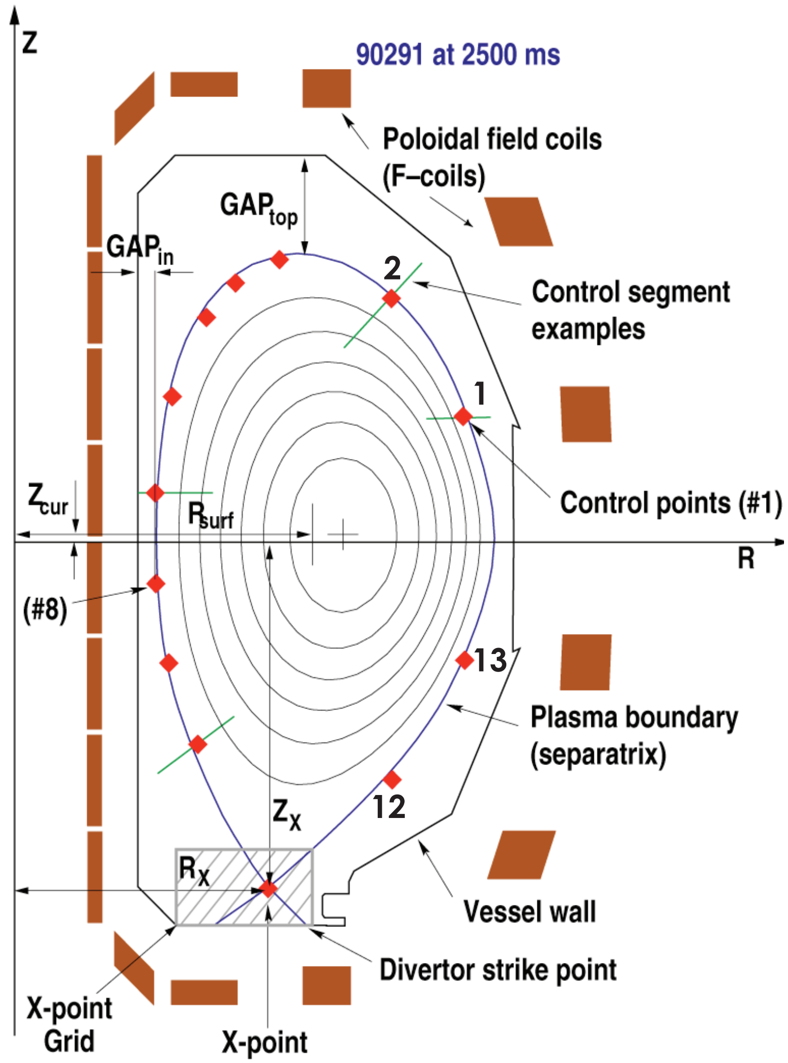
Ferron, et al. Nuc. Fus. 38(1998)1055

ISOFLUX Shape Control (Cont.)

- Target plasma surface defined by control points
- X-point is located by real time equilibrium calculation
- Flux at X point is determined
- Relative flux at other control points is determined
- Plasma Control System (PCS) regulates PF coil voltages to correct flux errors
- Use standard PID or MIMO control
- Fast inner loop stabilizes vertical instability

$$\psi_{error,n} = \psi_{seg,n} - \psi_{Xpt,n}$$

$$\overline{F}_{error} = \overline{\overline{M}}\,\overline{\psi}_{error}$$



M Matrix for Single Null Divertor

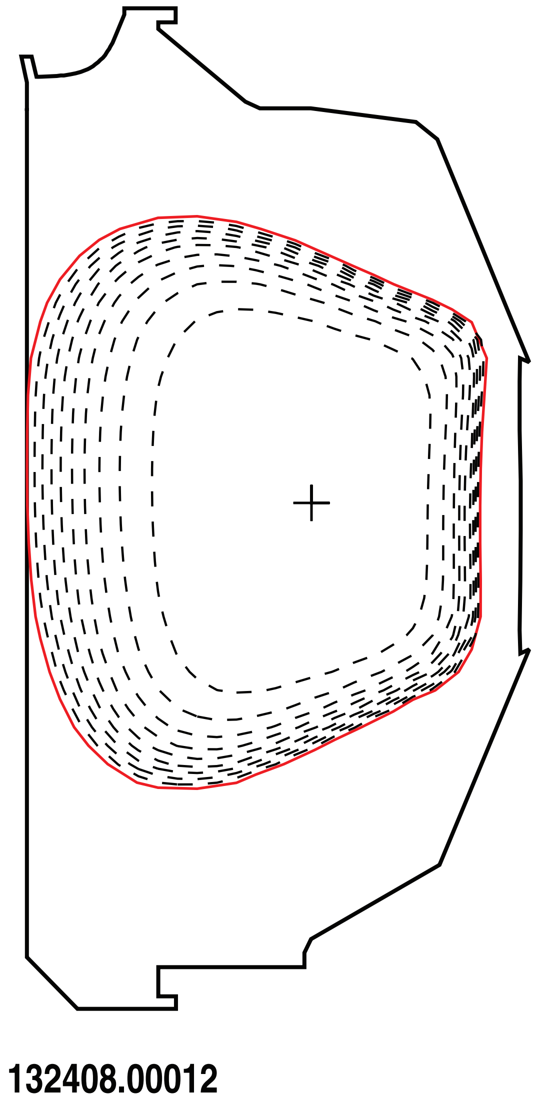
F_{error}

	SEG01	SEG02	SEG03	SEG12	SEG13	SEG14	RX1	ZX1
F1A	0.00000	0.	0.	0.	0.	0.	0.	0.
F2A	0.	0.	0.	0.	0.	0.	0.	0.
F3A	0.	0.	0.	0.	0.	0.	0.	0.
F4A	0.	0.	0.	0.	0.	0.	0.	0.
F5A	0.	0.	0.	0.	0.	0.	0.	0.
F6A	600.00	200.00	0.	0.	0.	0.	0.	0.
F7A	0.	0.	400.00	0.	0.	0.	0.	0.
F8A	0.	0.	0.	0.	0.	0.	0.	0.
F9A	0.	0.	0.	0.	0.	-50.00	0.	0.
F1B	0.	0.	0.	0.	0.	0.	0.	0.
F2B	0.	0.	0.	0.	0.	0.	0.	0.
F3B	0.	0.	0.	0.	0.	0.	0.	0.
F4B	0.	0.	0.	0.	0.	0.	250.00	-100.00
F5B	0.	0.	0.	0.	0.	0.	250.00	-100.00
F6B	600.00	0.	0.	0.	200.00	0.	0.	0.
F7B	0.	0.	0.	500.00	0.	0.	0.	0.
F8B	0.	0.	0.	0.	0.	0.	-100.00	-250.00
F9B	0.	0.	0.	0.	0.	0.	-250.00	-250.00

ψ_{error}

M_{matrix}

Efit Movie (12 ms to 520 ms during Ip Ramp)



Control during plasma startup: What can go wrong?

- **Hardware Failures (e.g. power supplies)**
  - Fault Identification and Communication System (FICS) is essential to isolating problems expeditiously in a complex environment
- **"Plasma issues"**
  - Impurities preclude burnthrough (poor wall conditioning, vacuum leaks, etc.)
  - Low density produces locked modes and disruptions
  - High density leads to poor performance (no H-mode, density limit disruptions)
- **Shaping problems**
  - Handoff between phases sometimes causes oscillations
  - Accuracy of shape control can be an issue in some experiments
- **Operator and programming errors**
  - Electronic startup checklist has dramatically reduced errors
- **Computer Faults (21 real-time processors can produce faults)**
- **Sensors (primarily magnetics)**
  - Crucial for successful operation (76 magnetic probes & 40 psi loops)
  - Sensors must be "exquisitely" calibrated. Daily calibration procedure improves reliability, but sensors can fail during shot
  - Realtime EFIT can sometimes indicate a failed sensor (distorted shape)

ITER presents unique challenges for startup and control

ITER CHALLENGE

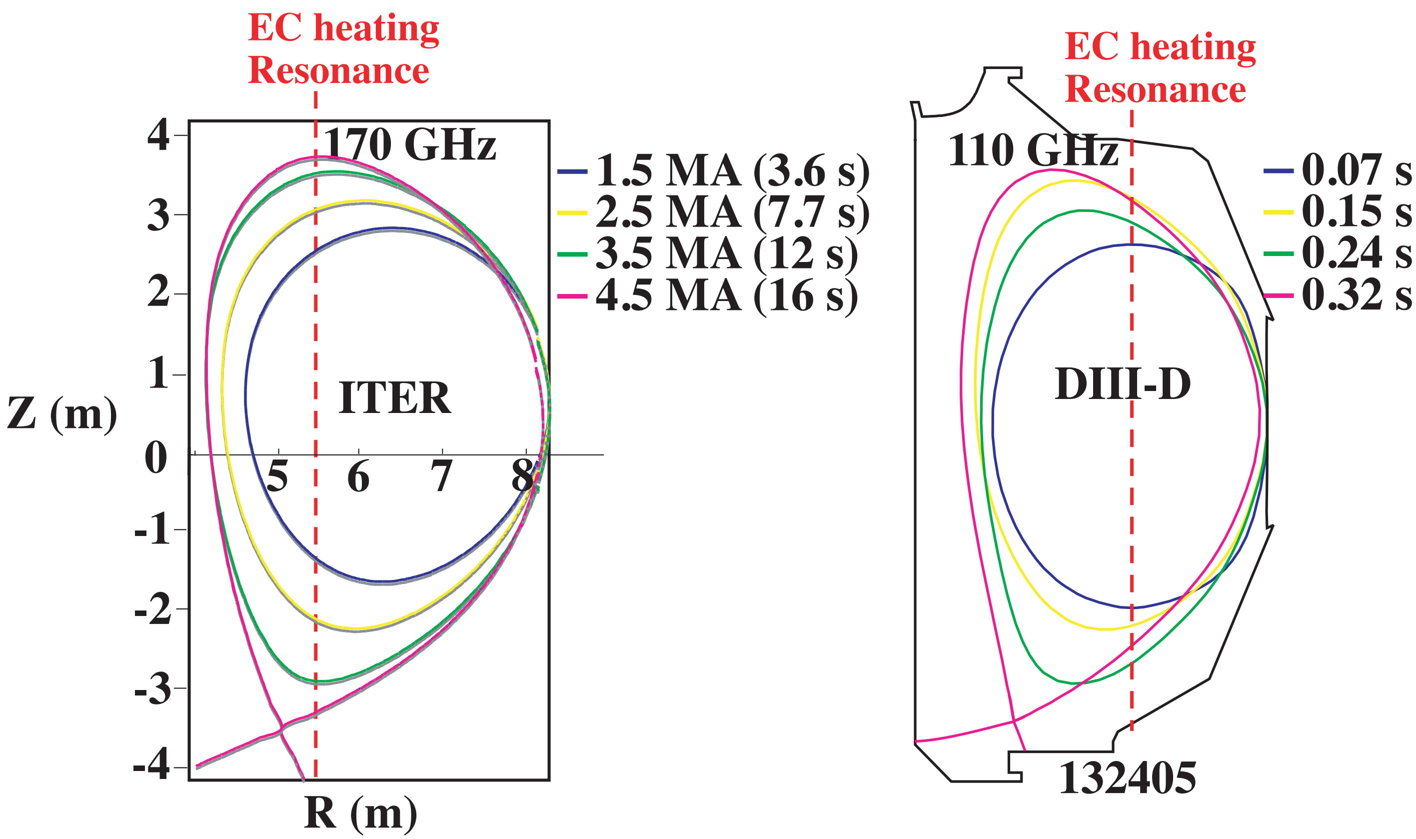
- Low inductive electric field and large vessel currents for startup
- Limited Ohmic heating (OH) power for burnthrough phase
- Power supplies limit range of current density profiles
- Minimize flux consumption
- Control heat flux to sensitive areas
- Discharges must operate well within stability limits
- Poloidal Coil set combines functions of OH and shape control

DIII-D EXPERIMENTS AND MODELING HAVE INVESTIGATED THESE CHALLENGES

- Time scaled by resistive diffusion time (≈50:1)
- Size scaled by machine dimensions of ITER & DIII-D (3.6:1)
- Normalized parameters (I_p/aB, l_i, β_N, and shape) are similar
- Modeling of the ITER coil system and benchmarking experiments is in progress

A NEW, LARGER VOLUME, STARTUP SCENARIO WAS DEVELOPED, DIVERTING EARLIER IN TIME TO MINIMIZE LIMITER HEATING

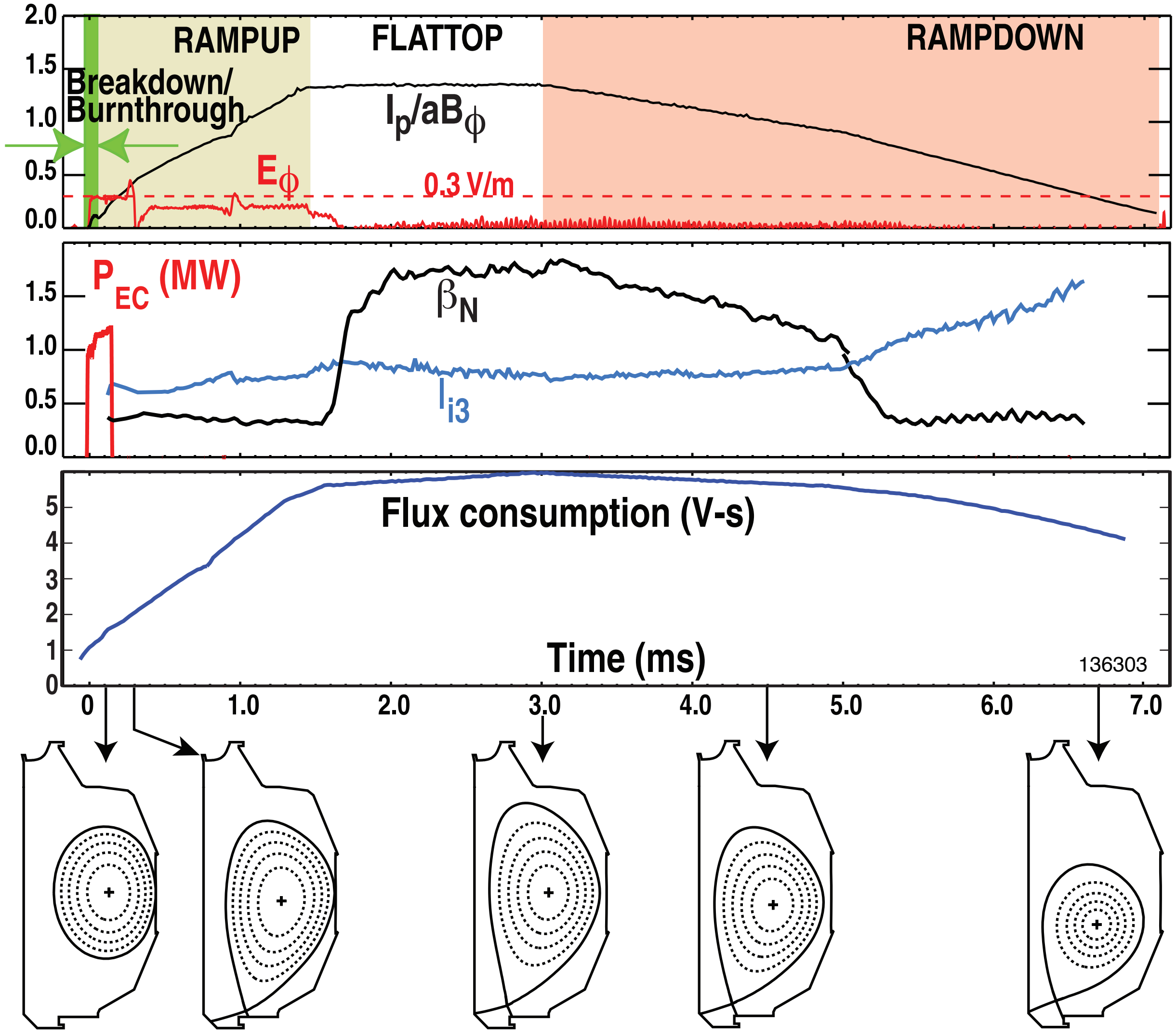
"IMPROVED" ITER STARTUP SCENARIO



- The EC resonance location is inside the plasma volume for effective power deposition during burnthrough in both devices

- Temporal evolution of the large-bore ITER startup scenario

DIII-D Has Experimentally Simulated All Phases of the ITER Scenario in a Single Discharge



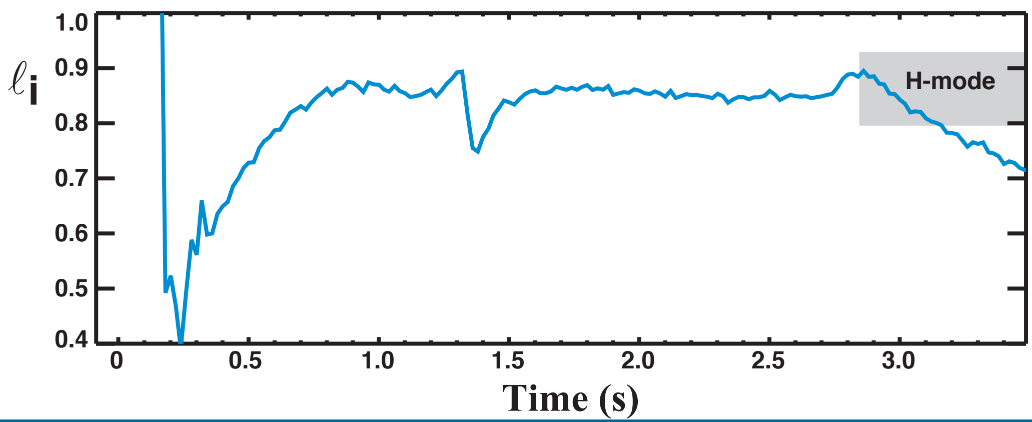
- EC assist allowed robust rampup for $E_\phi \geq 0.21$ V/m
- ITER Baseline H-mode (scenario 2) achieved after OH rampup
- ECH produced reliable breakdown and burnthrough of low Z impurities
- No additional flux consumption during rampdown
- Strike points held fixed during aperture reduction

TO REMAIN WITHIN THE ITER DESIGN RANGE, ℓ_i CAN BE CONTROLLED DURING RAMP-UP BY VARYING dI_p/dt

- Internal inductance, ℓ_i , is calculated from the plasma current density profile
  - Higher ℓ_i generally operates closer to the vertical stability limit -- NOT GOOD!
- ITER Poloidal Field (PF) Coil constraints place limitations on ℓ_i during rampup
  - ℓ_i too high and vertically unstable
- Specific ℓ_i profile may be required (within PF constraints) to achieve desired fusion burn scenarios
- First ITER startup scenario in DIII-D experimental simulations produced high ℓ_i sometimes leading to vertical instabilities
- Feedback control of the current ramp can produce desired ℓ_i target
  - Ohmic power supply is the actuator
  - Plasma Control System (PCS) calculates ℓ_i realtime (rtEFIT)
  - PCS computes an error signal and varies dI_p/dt by controlling power supply voltage

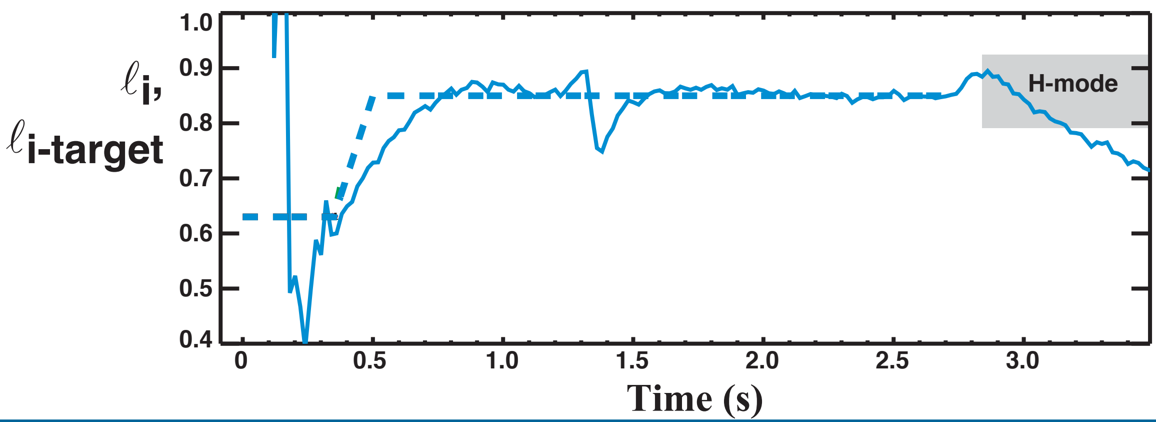
To Remain within the ITER Design Range, l_i can be Controlled by Varying the I_p Ramp Rate

- ITER Poloidal Field (PF) Coil constraints place limitations on ℓ_i
  - Specific ℓ_i may be required (within PF constraints) for advanced inductive scenarios
- Feedback control of I_p can produce desired ℓ_i target
  - Plasma Control System (PCS) calculates ℓ_i realtime (rtEFIT)



To Remain within the ITER Design Range, l_i can be Controlled by Varying the I_p Ramp Rate

- ITER Poloidal Field (PF) Coil constraints place limitations on ℓ_i
  - Specific ℓ_i may be required (within PF constraints) for advanced inductive scenarios
- Feedback control of I_p can produce desired ℓ_i target
  - Plasma Control System (PCS) calculates ℓ_i realtime (rtEFIT)
  - $\ell_i(3)$ compared to target and PCS computes an error signal



To Remain within the ITER Design Range, l_i can be Controlled by Varying the I_p Ramp Rate

- ITER Poloidal Field (PF) Coil constraints place limitations on ℓ_i
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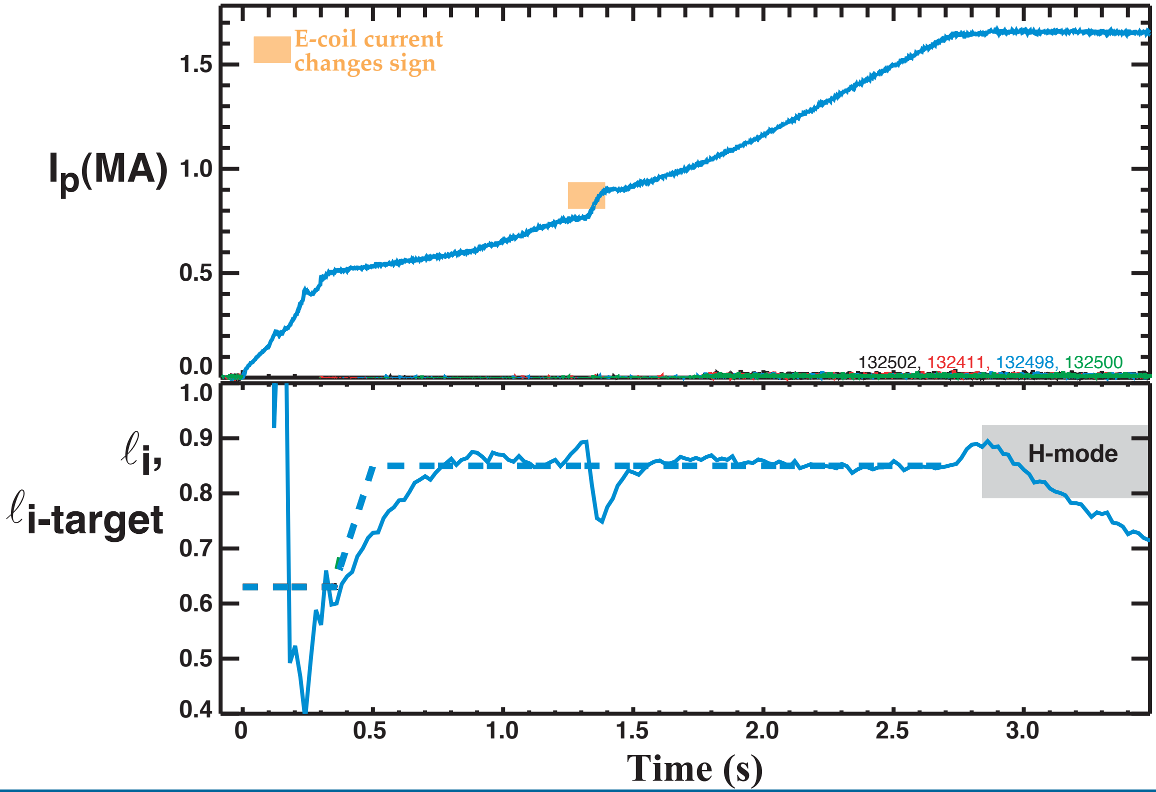
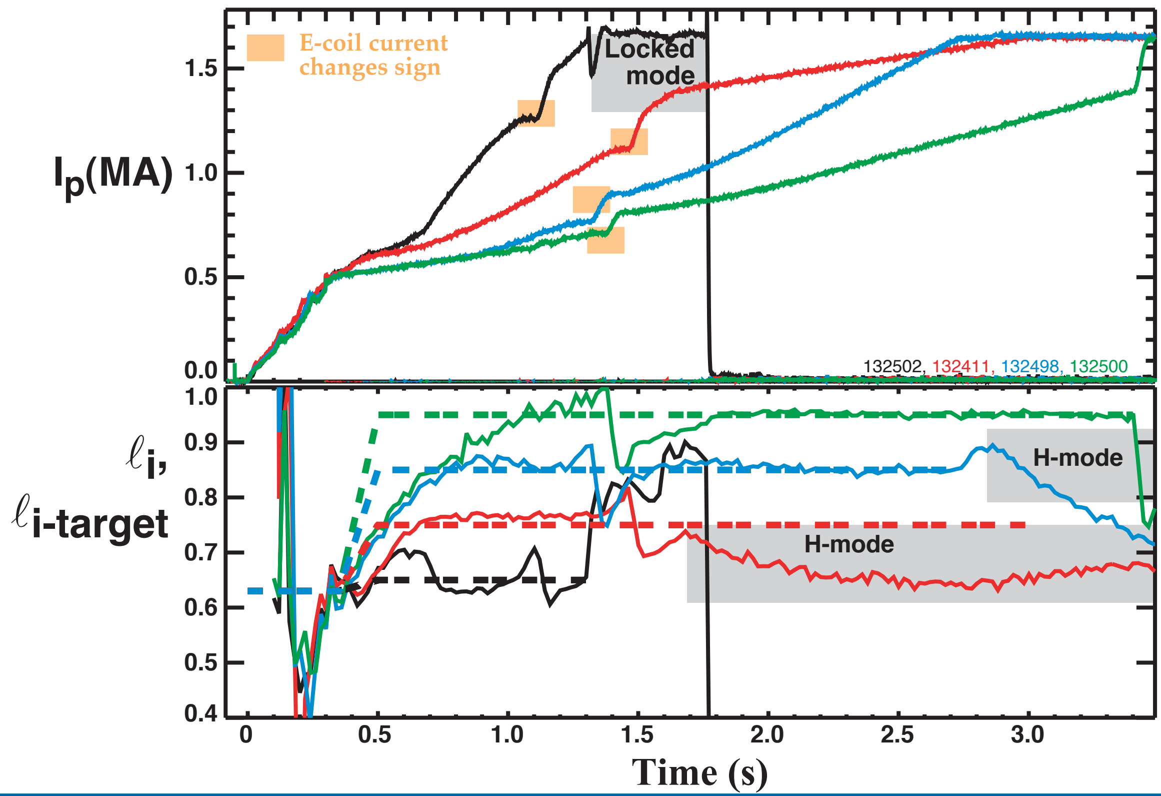
  - $\ell_i(3)$ compared to target and PCS computes an error signal
  - I_p controlled with Ohmic power supply as the actuator



To Remain within the ITER Design Range, l_i can be Controlled by varying the I_p Ramp Rate

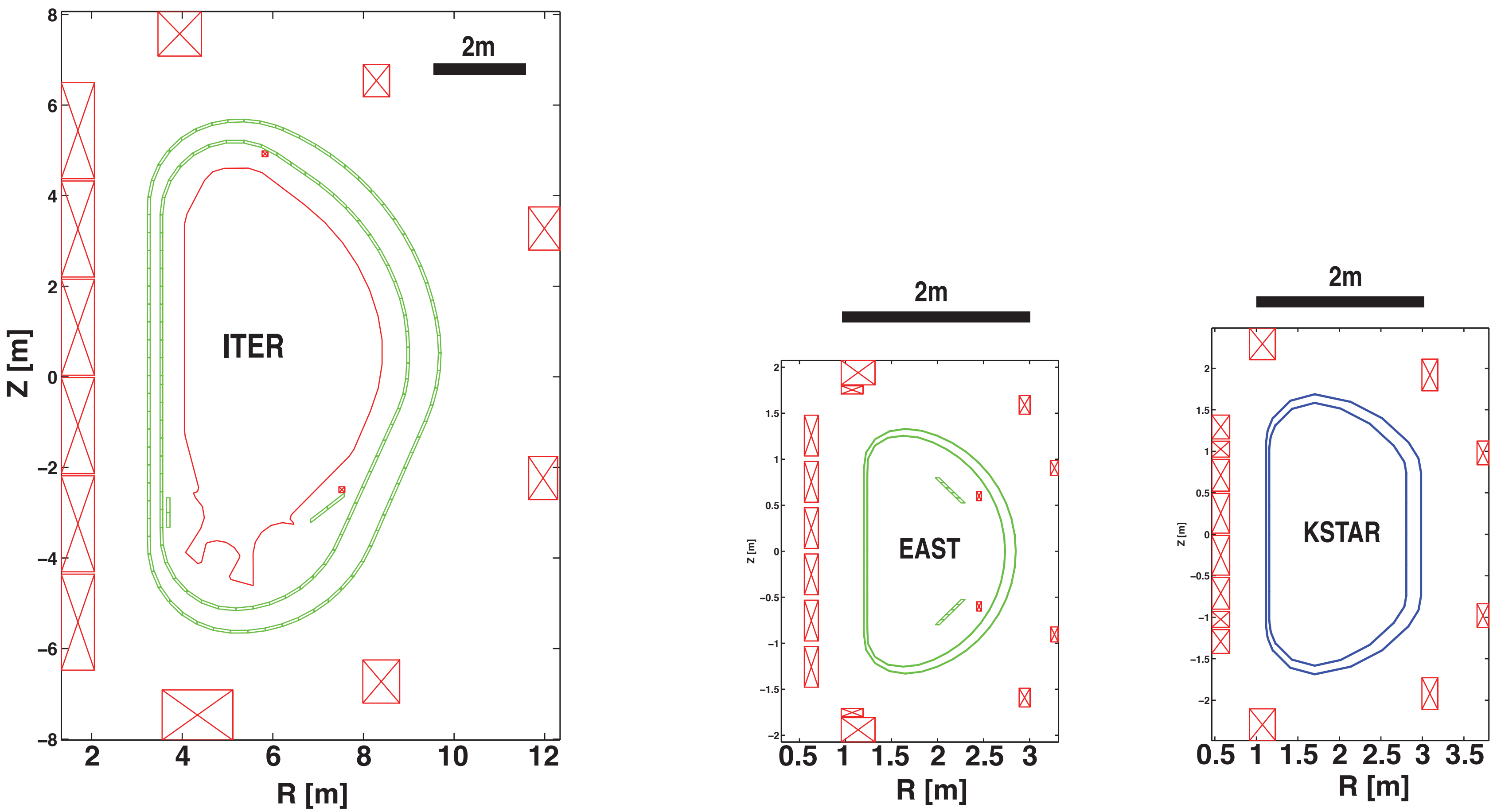
- ITER Poloidal Field (PF) Coil constraints place limitations on ℓ_i
  - Specific ℓ_i may be required (within PF constraints) for advanced inductive scenarios
- Feedback control of I_p can produce desired ℓ_i target
  - Plasma Control System (PCS) calculates ℓ_i realtime (rtEFIT)
  - $\ell_i(3)$ compared to target and PCS computes an error signal
  - I_p controlled with Ohmic power supply as the actuator



- Feedback control achieved over ITER design range of internal inductance

ITER and recently commissioned tokamaks (EAST, China, and KSTAR, Korea) have combined OH and field shaping functions

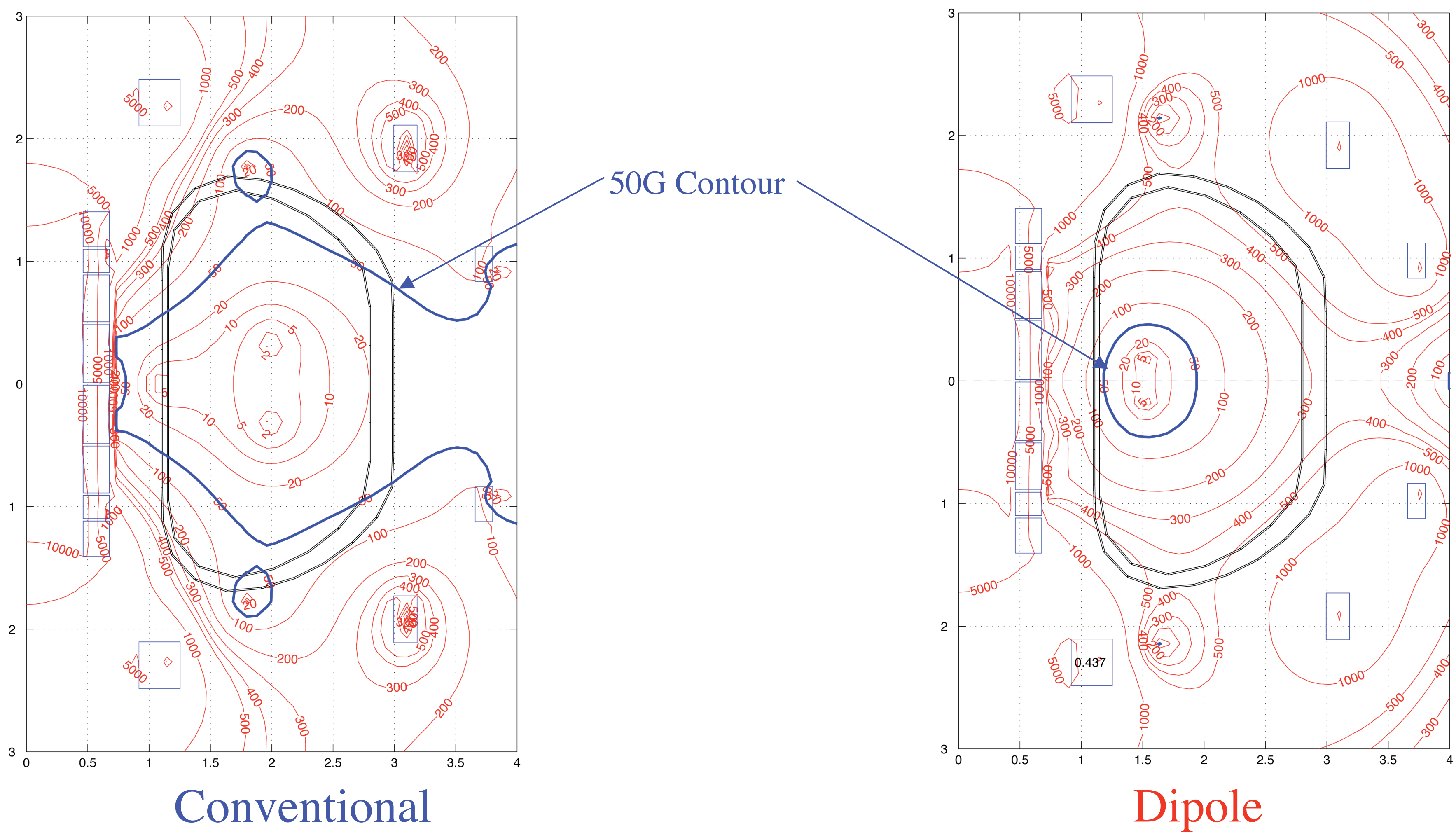
- **Fewer coils than DIII-D (18 F-coils + OH) make control more challenging**
- **EAST and KSTAR have less available flux for Ip ramp**
- **All 3 tokamaks have lower inductive voltage, hence less OH power**
  - **RF assisted breakdown and burnthrough becomes more important for reliable and reproducible startup**



Trade-offs are necessary to optimize startup performance, especially with a more limited coil set

- **Example: KSTAR "dipole" configuration more desirable during the initial commissioning phase**

  ... But the field null region is reduced making breakdown and burnthrough more difficult



Lessons learned (with applications for ITER) from 32 years of operation in DIII-D (previously Doublet III)

- Each tokamak shares many similarities for control, but also has unique challenges
  - ITER will be no exception
- Startup of a discharge is complex, and has implications for the flattop phase (burning plasma in ITER)
- A robust and well controlled startup is necessary to allow access to flattop experiments
- Reliable operation requires well calibrated (and well maintained) sensors
- Simulation of the PCS and a detailed plasma model will be required for ITER to obtain required reliability and performance
  - But expect surprises-- modeling can't solve every problem
  - Tools must be useable by many users, not just 'experts'
- Complete archiving of data will be required (post mortem analysis, etc.)