Abstract

The Pegasus Toroidal Experiment is an ultra-low aspect ratio ($A < 1.2$) spherical tokamak (ST) capable of operating in the high I_N regime ($I_N > 12$). Access to this regime requires a small center-post cross-section that consequently reduces the available inductive current drive from the central ohmic (OH) solenoid. Non-solenoidal plasma startup allows for more efficient use of the OH current drive and may possibly eliminate the need for a solenoid in future STs. Recent experiments on Pegasus use a single washer gun current source located near the outboard midplane to establish and sustain a tokamak-like plasma via DC helicity injection. A new gun head design permits high current (2 kA) injection with minimal impurity production and improved neutral fueling control. The washer gun and a biased anode are mounted at the same toroidal location, 20 cm below and above the mid-plane. The vacuum toroidal and vertical fields are chosen so the initial injected current follows a helical field line that connects the gun aperture to the anode. For a sufficiently large current density and small vertical field strength, the plasma relaxes into a tokamak-like configuration. With less than 2 kA of injected current, tokamak-like discharges with $I_P \approx 20$ kA are produced. Line-averaged densities near the Greenwald density limit of $1.0 \times 10^{19}\ m^{-3}$ indicate improved particle confinement. The formation of a current channel within the vacuum region separate from the gun injection region is verified using magnetic field measurements. Substantially longer current decay times (2 - 3 ms) indicate the buildup of stored energy. The length of the decay time is suitable for coupling to other current drive techniques. Discharges of 80 kA were obtained by applying < 10 mWb of OH flux to a 20 kA seed plasma. These results are compared to discharges initiated with two 1 kA washer guns mounted in the lower divertor region. Future experiments with multiple injectors are also described.

ST startup and current drive using plasma gun DC helicity injection on Pegasus

- Solenoid-free startup would extend the efficiency of inductive OH drive
  - Especially important for the low aspect ratio spherical tokamak (ST)

- Plasma gun point-source DC helicity injection on the Pegasus Toroidal Experiment
  - Plasma guns → low impurity, high J_{inj} source
  - Plasma guns mounted in lower divertor
  - Available current drive described using concepts of DC helicity injection and Taylor relaxation

- New midplane plasma gun system installed on Pegasus
  - Identify dependence on point-source location

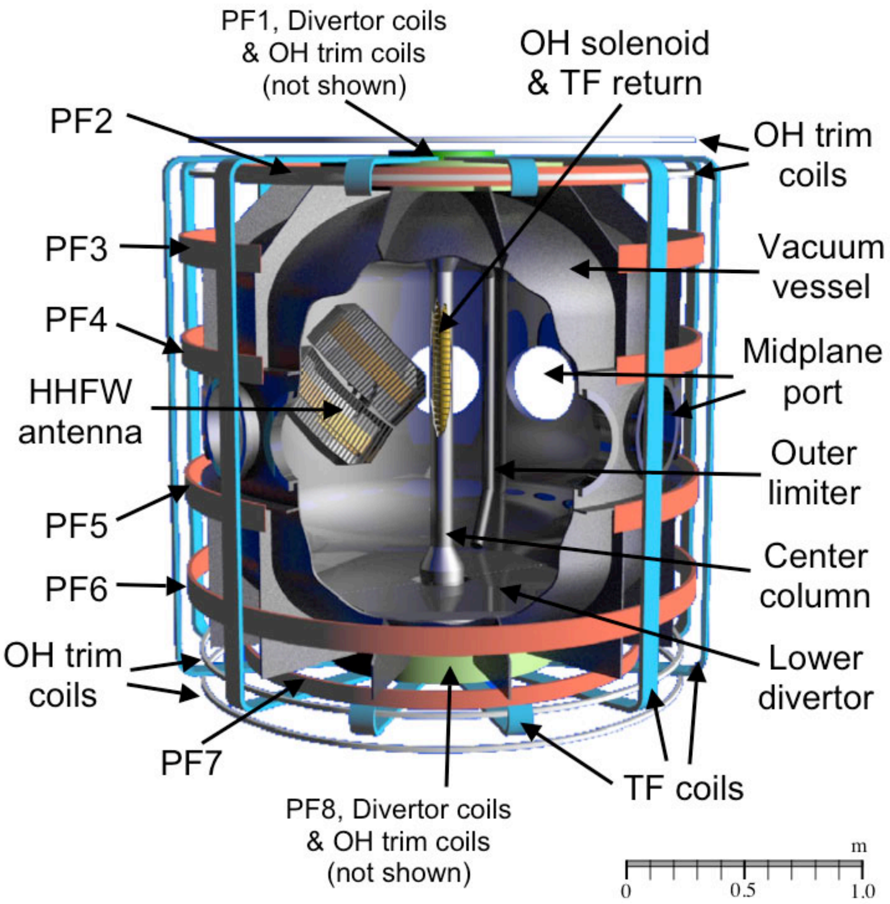
The Pegasus Toroidal Experiment is well suited for point-source DC helicity injection studies

- Ultra-low aspect ratio ($A < 1.3$)
  - Span large range of R_0/R_{inj}

- Flexible configuration
  - Independent PF & Div coils
  - Good port availability

Experimental Parameters

Parameter	Achieved	Goals
A	1.15-1.3	1.12-1.3
R (m)	0.2-0.45	0.2-0.45
I_p (MA)	≤ 0.18	≤ 0.30
I_N (MA/m-T)	6-12	6-20
RB_t (T-m)	≤ 0.06	≤ 0.1
κ	1.4–3.7	1.4–3.7
τ_{shot} (s)	≤ 0.02	≤ 0.05
β_t (%)	≤ 25	> 40
P_{HHFW} (MW)	0.2	1.0



Current drive in a tokamak is described using the concept of magnetic helicity

Total helicity in a tokamak geometry: $K = \int_V (\mathbf{A} + \mathbf{A}_{vac}) \cdot (\mathbf{B} - \mathbf{B}_{vac})\, \mathrm{d}^3\mathrm{x}$

$$\frac{dK}{dt} = -2\underline{\int_V \eta \mathbf{J} \cdot \mathbf{B}\, \mathrm{d}^3\mathrm{x}} - 2\underline{\frac{\partial \psi}{\partial t} \Psi} - 2\underline{\int_A \Phi \mathbf{B} \cdot \mathrm{d}\mathbf{s}}$$

- Resistive Helicity Dissipation
  - $\mathbf{E} = \eta \mathbf{J}$ → much slower than energy dissipation ($\eta \mathbf{J}^2$)
  - Turbulent relaxation processes dissipate energy and conserve helicity
- AC Helicity Injection: $\dot{K}_{AC} = -2\frac{\partial \psi}{\partial t}\Psi = 2V_{loop}\Psi$
- DC Helicity Injection: $\dot{K}_{DC} = -2\int_A \Phi \mathbf{B} \cdot \mathrm{d}\mathbf{s} = 2V_{inj}B_{\perp}A_{inj}$

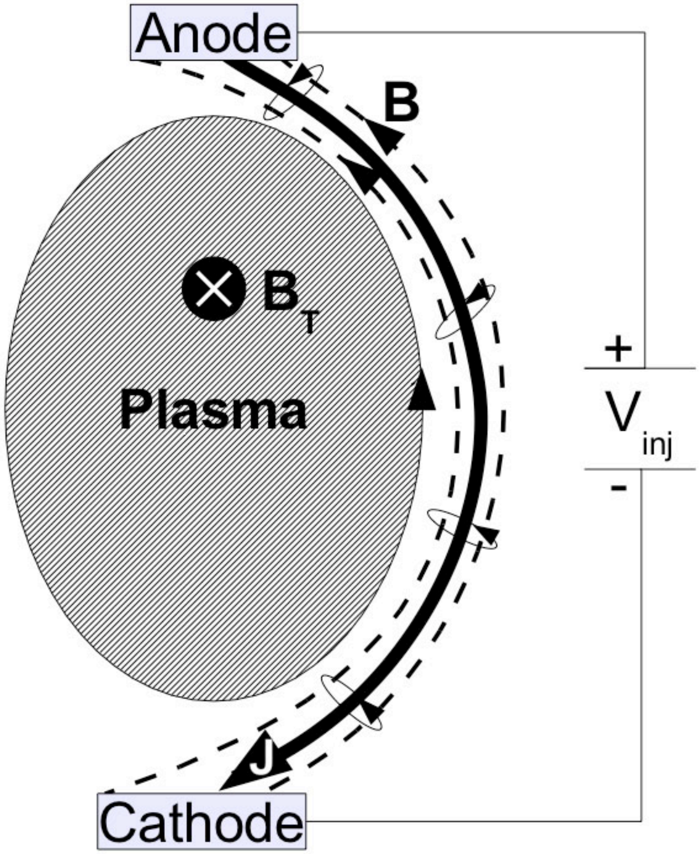
Electrostatic DC helicity injection has been demonstrated on a number of tokamak devices

- DC helicity injection is related to inductive current drive using:

$$V_{eff} \approx \frac{N_{inj} V_{inj} A_{inj} R_0}{A_p R_{inj}}$$

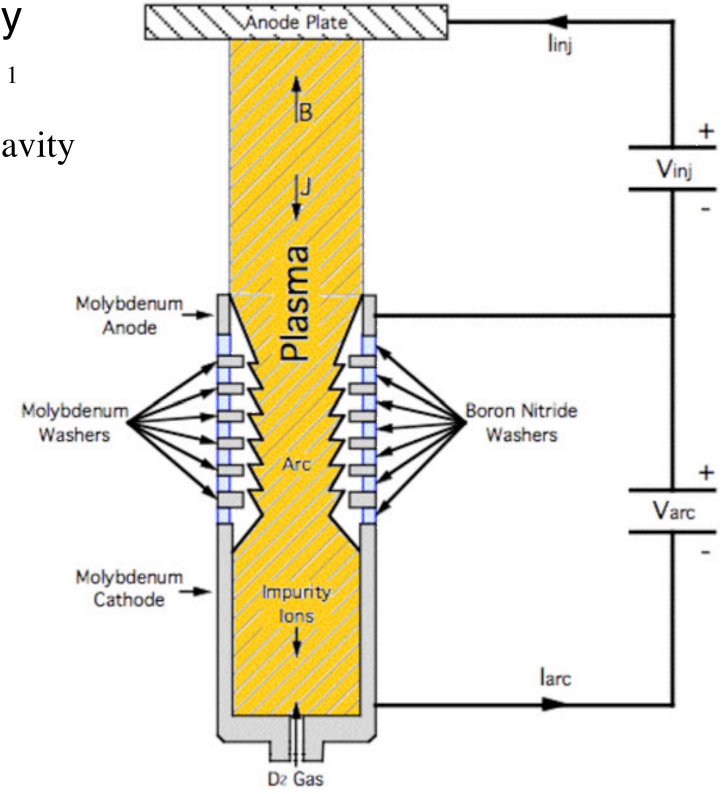
- V_{eff} increases with A_{inj}
  - A_{inj} maximized with CHI system
  - CHI demonstrated on HIT, HIT-II, NSTX
  - CHI not easily retrofitted onto Pegasus

- Point-source injection requires large V_{inj} as A_{inj} is reduced
  - Studied on CDX, CCT using emissive electrodes
    - Demonstrated tokamak-like plasma formation
    - Limited to low J_{inj} by impurity sputtering



Plasma guns provide a low impurity, point-source DC helicity injection scheme

- Arc discharge sustained in washer stack cavity
  - Washers stabilize arc while limiting surface contact [1]
  - Sputtered high-Z impurities mostly trapped in gun cavity
- Plasma column supports large J_{inj} without space charge limitations [2]
- Requires constant current arc and bias power supplies
  - I_{arc} = 2 kA using a pulse forming network
  - V_{arc} = 100 - 500 V
  - $I_{bias} < I_{arc}$ for impurity sputtering
  - I_{arc} < 2kA using current feedback control
  - V_{bias} < 800 V (before near-term upgrade)



[1] Den Hartog, D.J., Plasma Sources Sci. & Tech. **6** (1997)
[2] Fiksel, G, et. al., Plasma Sources Sci. & Tech. **5** (1996)

The maximum sustained I_p is determined by the balance between helicity injection and dissipation

- For $B_\phi >> B_\theta$: $\dot{K}_{diss} \approx 4\pi R_0 I_p B_{\phi 0}\langle\eta\rangle$
- Solving for $\dot{K}_{diss} = \dot{K}_{DC}$ gives

$$I_p = \left(\frac{N_{inj} V_{inj} A_{inj}}{R_{inj}}\right)\frac{1}{2\pi\langle\eta\rangle}$$

Maximum I_p related to plasma geometry and magnetic field strength through η term

- Use Spitzer resistivity as an approximation:

$$\eta \approx 5.2\times10^{-5}\, Z_{eff} \ln\Lambda / T_{eV}^{3/2}\ (\Omega\cdot\mathrm{m})$$

Maximum I_p related to T_{e0}

- Reworking the helicity balance expression gives:

$$\overline{T}_e \approx \left[10^6 \pi I_p Z_{eff} \ln\Lambda \left(\frac{R_{inj}}{N_{inj} V_{inj} A_{inj}}\right)\right]^{2/3}$$

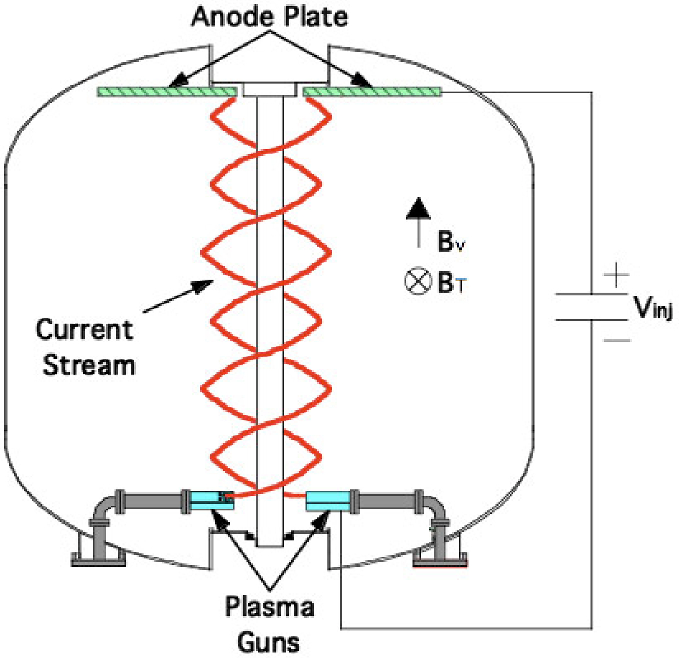
Assuming nearly flat temperature profile

Initial experiments on Pegasus mounted the plasma guns near the lower divertor

- Anode plate hung from upper divertor
- Crossed B_v and B_ϕ vacuum field
- Low current plasma follows helical field line connecting gun & anode
- Inboard injection
  - Maximize V_{eff} for given R_0, A_p and V_{inj}

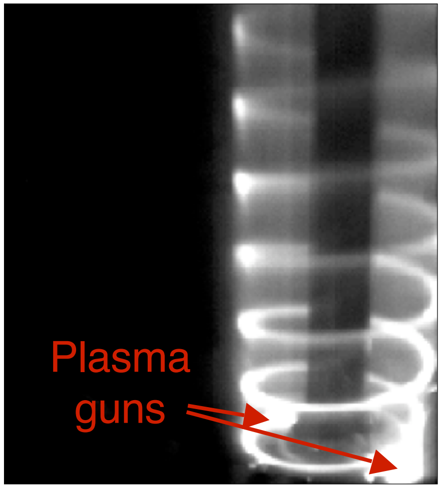
$$V_{eff} \propto R_0 / R_{inj}$$

  - Typical centerstack limited plasma startup
  - Capture sub-ms dynamics with fast camera
- Easily retrofitted into Pegasus

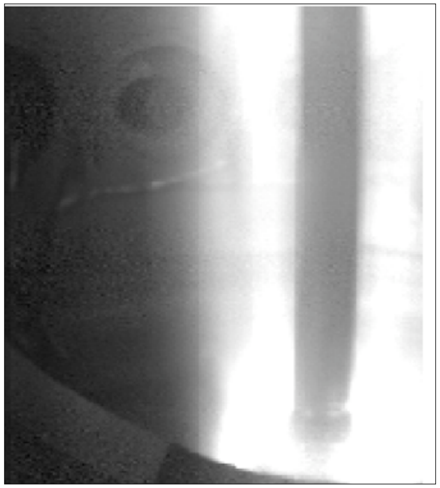


Zero current plasma filaments in vacuum magnetic field

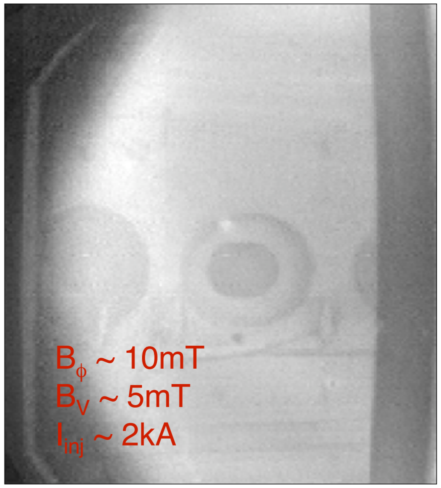
Magnetic topology relaxes into a tokamak-like configuration at sufficiently high I_{inj} and low B_v



Current filaments



“Current sheet”



Tokamak-like plasma

Small GI_{inj}/B_V ⟶ Large GI_{inj}/B_V

M = G

M = G

M > G

Toroidal current multiplication factor: $M = I_\phi / I_{inj}$

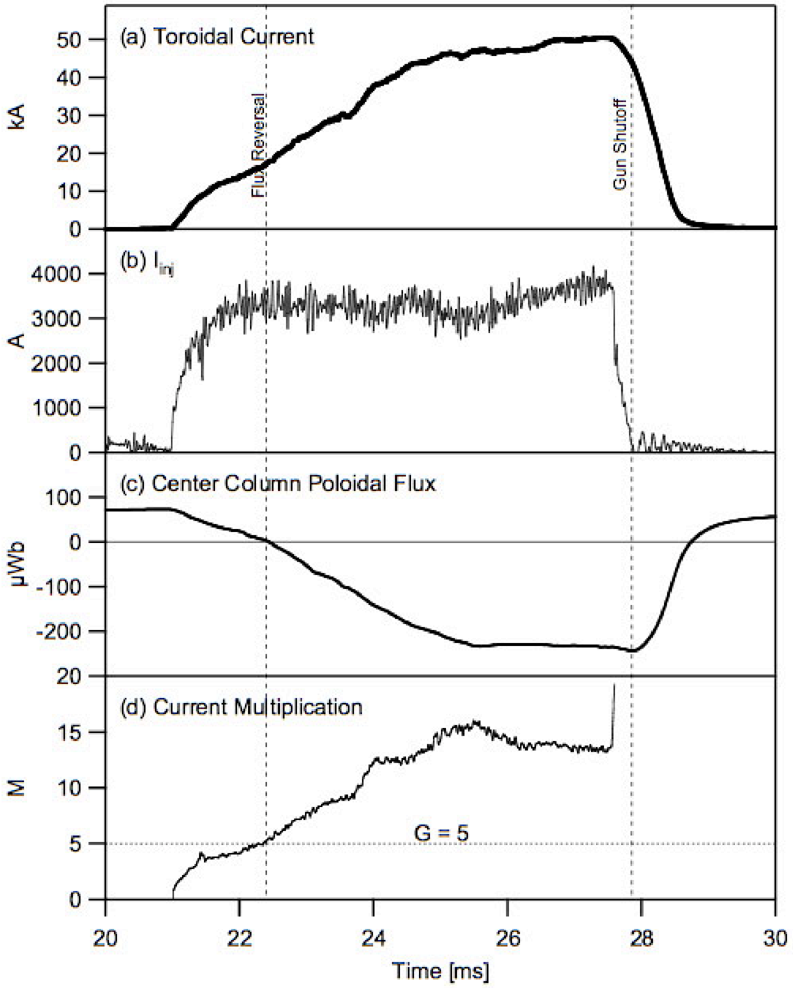
Geometric stacking factor: $G \approx \dfrac{\mu_0 I_{TF} \Delta z}{(2\pi R_{inj})^2 B_V}$

Gun - anode separation

The onset of flux amplification correlates with inboard poloidal magnetic flux reversal

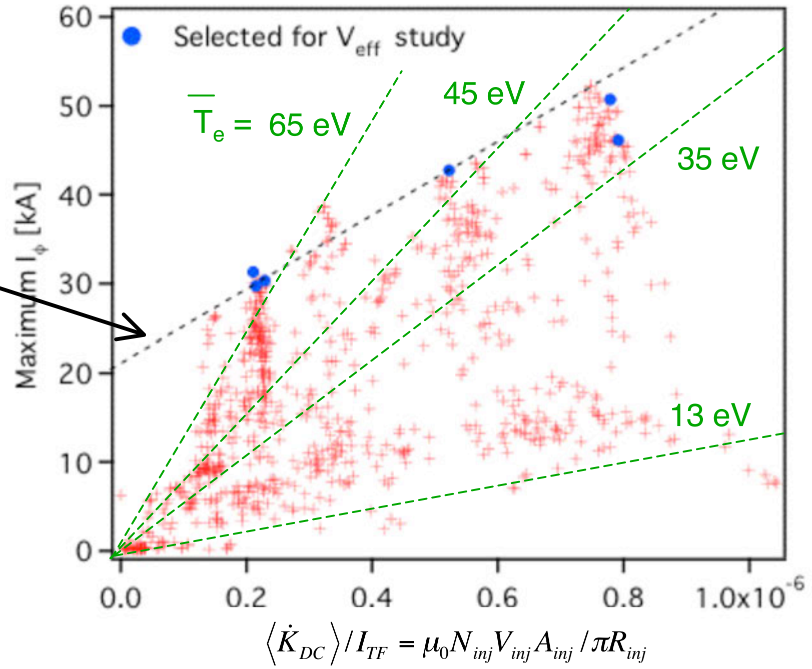
- $I_p > 50$ kA driven by $I_{inj} \leq 4$ kA
  - Static B fields (no inductive drive)
  - Two plasma guns
    - $A_{inj} = 3\ cm^2$, $R_{inj} = 16$ cm
- I_p persists after $I_{inj} = 0$
  - Suggests non-zero stored energy
- B_θ reversal observed at inboard midplane
  - Plasma is limiting on center column
  - Reversal correlates with $M > G = 5$

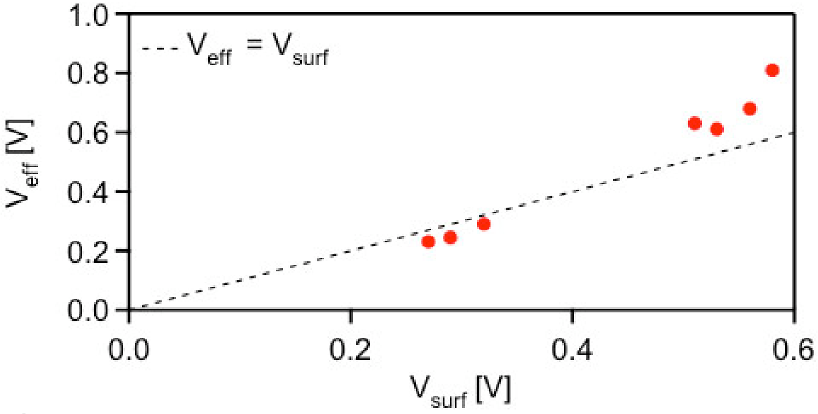
Shot 32606



Evidence that the maximum I_p is realized when the tokamak-like plasma achieves helicity balance

- Data set for static B fields
  - Each point represents one discharge
  - Includes one and two gun operations
  - Max I_p for given conditions achieved at max B_v that allowed flux reversal
- Steady-state helicity balance roughly approximates average T_e
  - Assume Spitzer η & $Z_{eff} = 2.5$
  - Calculated average T_e = 35 - 65 eV from approximate resistive helicity dissipation
- $V_{surf} \approx V_{eff}$ suggests plasmas achieve helicity equilibrium
  - V_{surf} estimated using a flux measurement at center column
  - G-S solver provides plasma geometry for V_{eff} calculation





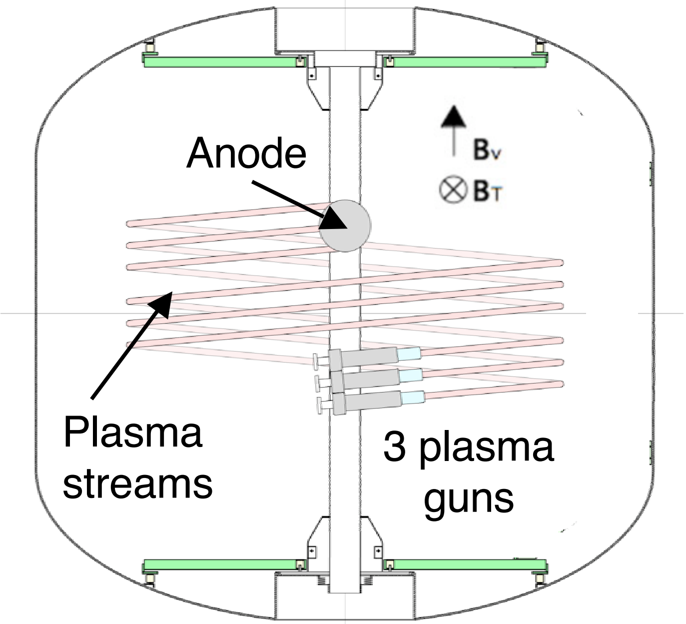
Outboard plasma gun system designed to explore point-source injection at the other geometric extreme

- High-field side injection
  - For fixed I_{inj}, A_{inj} & A_p:

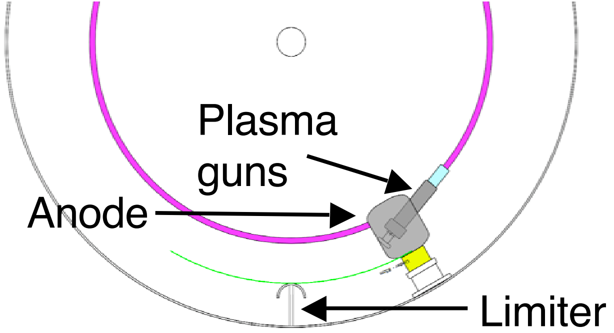
$$V_{eff} \propto Z_{inj} R_0 / R_{inj}$$

  - Sacrifice favorable R_0/R_{inj} scaling
  - The effect on Z_{inj} is unknown
    - Study dependence with injector geometry, plasma and injector parameters, etc.
- Outboard limited plasma
  - More dynamic evolution of EF required to maintain outboard position
  - Gain PF induction current drive
- Installation is straightforward
  - Outboard side of vessel is the most accessible

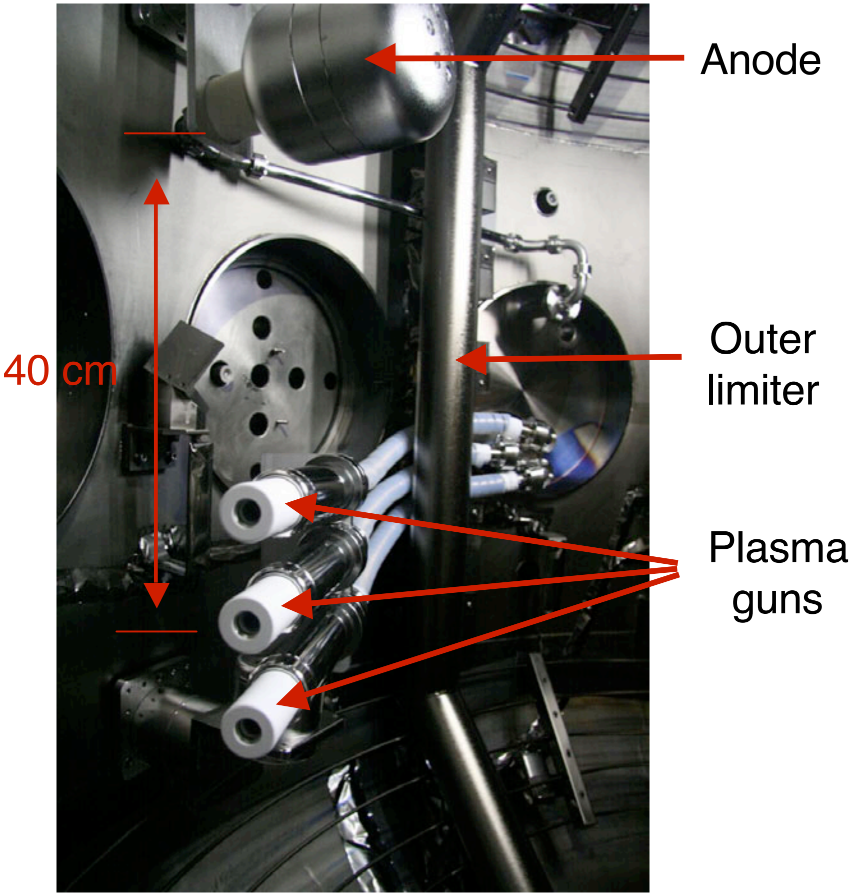
Side view

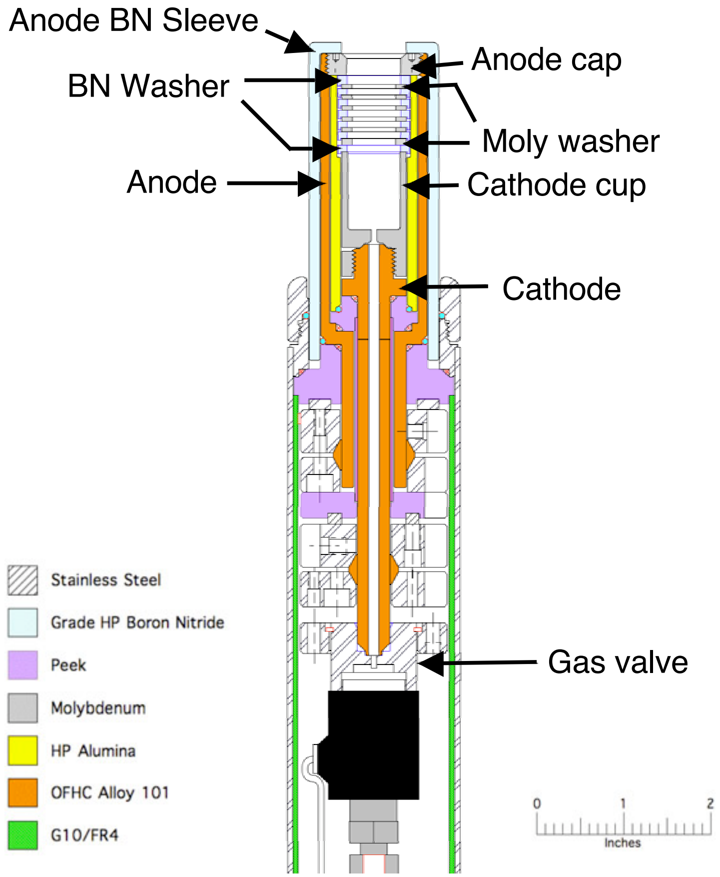


Top view (rotated)



Three plasma guns were recently mounted near the outboard midplane on Pegasus





A first-order model is being developed to estimate the maximum sustained I_p with outboard injection

- Simplest operation scenario → static B field, single gun

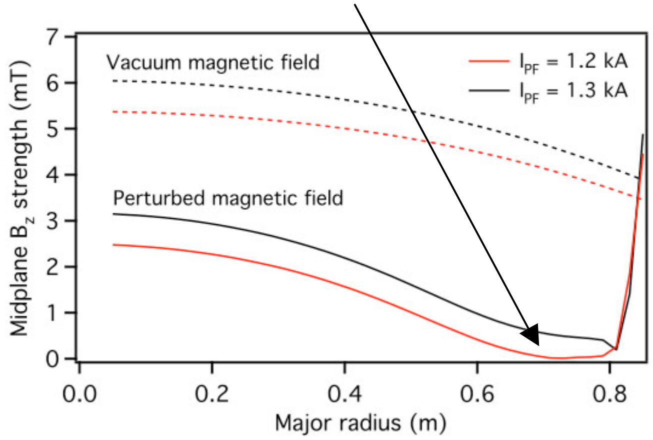
- Use 2-D field model to calculate maximum B_v
  - Predicts magnetic field regimes that allow for field reversal
  - B_v required for radial force balance in static field scenario

- Once a tokamak-like plasma has formed . . .
  - Estimate plasma shape vs R_0
  - Determine $I_{p,max}$ that satisfies these requirements

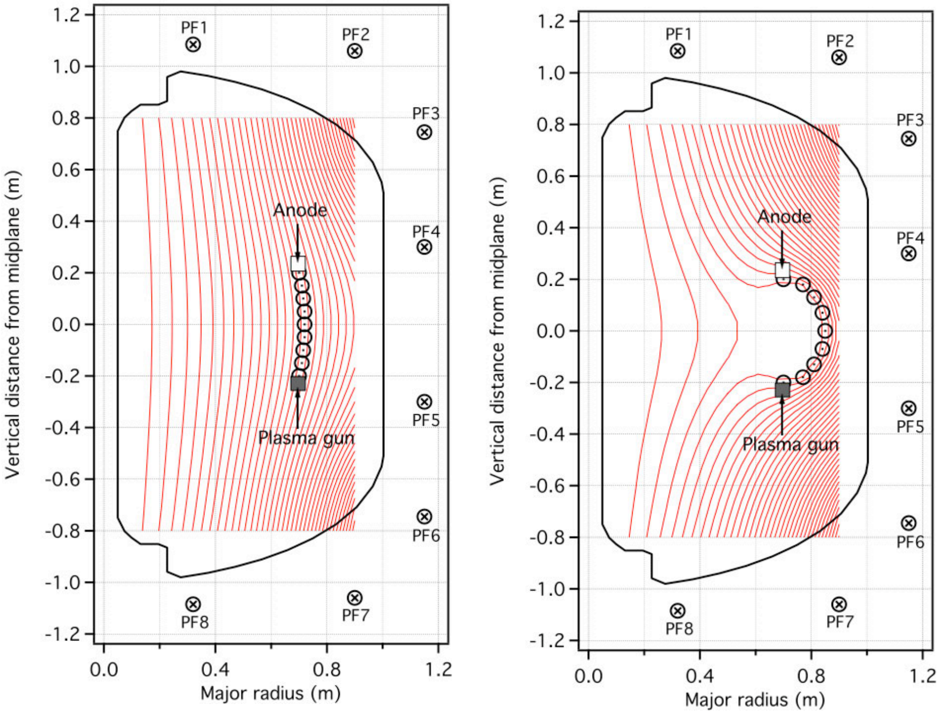
    (1) Force balance
    (2) Tokamak stability
    (3) Helicity balance
    (4) Taylor relaxation requirements

A 2-D current filament code is used to determine maximum B_v that leads to inboard field reversal

- Experimental observation: M > G correlates with inboard B field reversal
- Geometric windup = 2
  - Treat the discrete filaments as a toroidally averaged current sheet tied to a flux surface
- Max B_v that allows field reversal when $I_{PF} \sim 1.2$kA
  - $B_v \sim 0$ at inboard edge



Force-free plasma filaments perturb the vacuum magnetic field

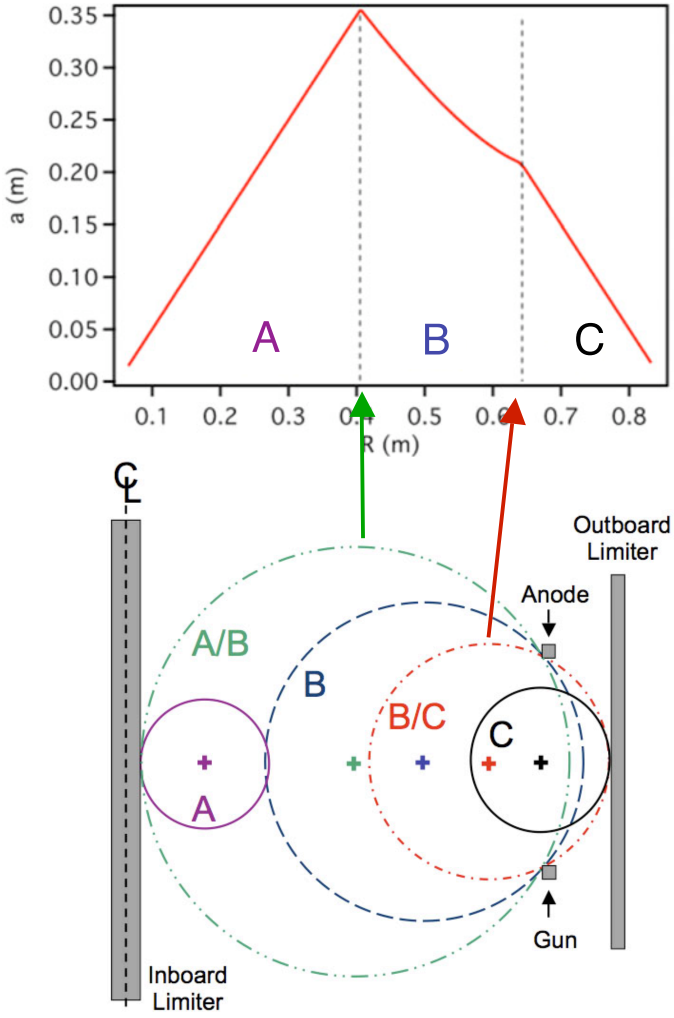


I_{inj} = 0 A

I_{inj} = 2 kA

I_{TF} = 300 kA, I_{PF} = 1.2 kA (PF1-3, 6-8)

The simple model assumes an outboard limited, large aspect ratio, circular cross-section plasma



Plasma limiting surface: (A) Center column

(B) Gun / anode (C) Outer limiter

Large-A radial force equilibrium

$$B_v = -\frac{\mu_0 I_p}{4\pi R_0}\left[\ln\left(\frac{8R_0}{a}\right) + \Lambda - \frac{1}{2}\right]$$ ①

Fixing B_v at the maximum value from the field reversal model gives $I_p(R_0)$

Assuming $\Lambda = \beta_p + \ell_i/2 - 1 \approx -1$ gives most optimistic I_p

Large-A cylindrical edge q

$$q_a \approx a^2 I_{TF} / R^2 I_p$$ ②

Considered for edge kink stability

Estimate of self-consistent T_{e0} calculation provides maximum I_p from helicity balance

③

Find a self-consistent solution for [1]

$$V_{eff} I_p = W_k / \tau_e + dW/dt$$

Assume $dW/dt \approx 0$ at $I_{p,max}$

τ_e approximated using empirical energy confinement relations:

ITER97L : Tokamak L-mode [2]

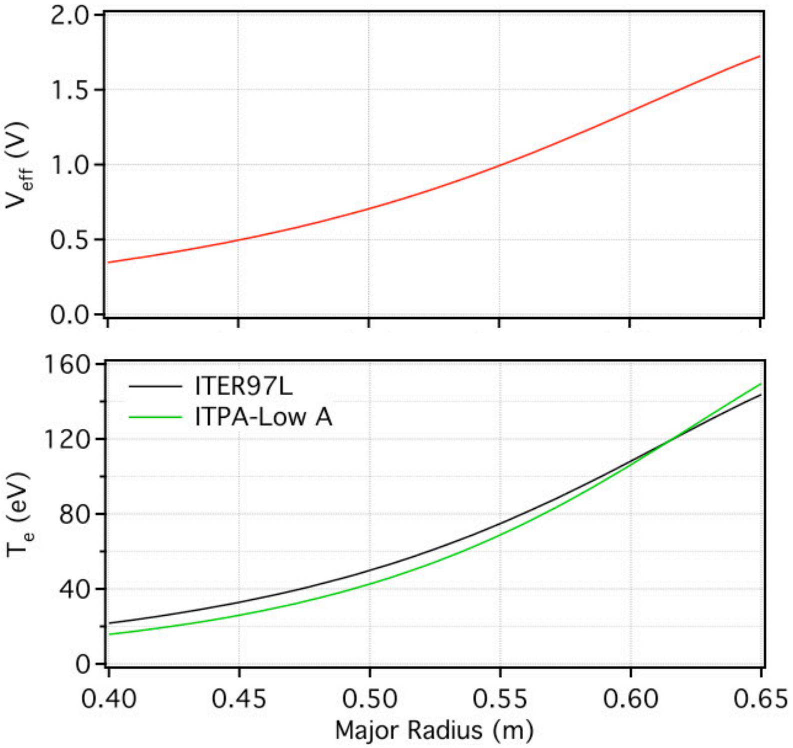
ITPA - low A: ST & tokamak H-mode [3]

Assume peaked n_e and T_e profiles

$$n_e(r) = n_{e0}\left(1-(r/a)^2\right)^{\alpha_n} \quad T_e(r) = T_{e0}\left(1-(r/a)^2\right)^{\alpha_T}$$

with Greenwald density scaling

$$n_{GR}(10^{20}\,\mathrm{m}^{-3}) = I_p(\mathrm{MA})/\pi a(\mathrm{m})^2$$



V_{inj}	800V	$\overline{n}_e / n_{GR}$	1.0
N_{inj}	1	Z_{eff}	2.5
A_{inj}	3 cm^2	T_i/T_e	0.5
R_{inj}	70 cm	α_N, α_T	1.0

[1] McCool, S.C. et. al., Bull. Am. Phys. Soc. **39**, 1994

[2] Kaye, S.M. et. al., Nuc. Fusion **37**, no. 9, 1997

[3] Takizuka, T. et. al., Plasma Phys. Con. Fusion **46**, no. 5A, 2004

Taylor relaxation criteria also limits the total sustainable I_p for a given plasma geometry

④

Considering force-free equilibrium:

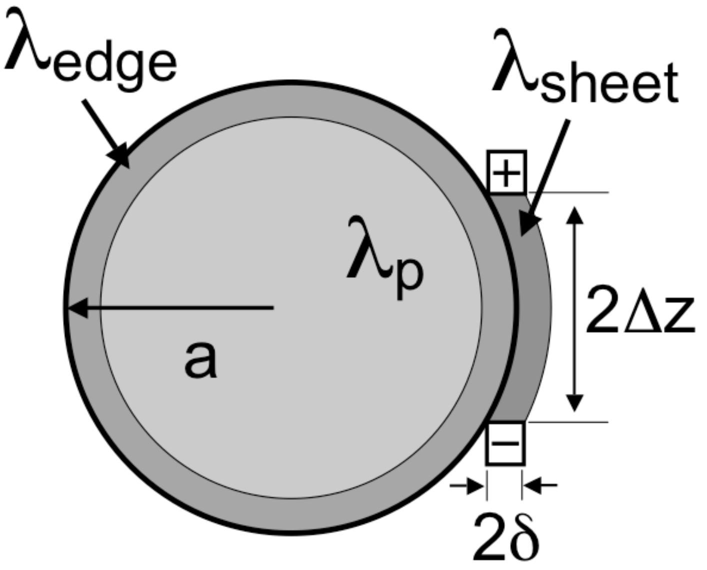
$$\nabla \times B = \mu_0 J = \lambda B$$

Current penetration via Taylor relaxation requires:

$$\bar{\lambda}_{sheet} > \bar{\lambda}_{plasma}$$

$$\bar{\lambda}_{sheet} = \frac{\mu_0 J_{\phi,sheet}}{B_{\phi,sheet}} \approx N_{inj} \frac{I_{inj}}{I_p} \frac{a}{\bar{R}_L \delta} \quad \text{(using } G \approx q_a\text{)}$$

$$\bar{\lambda}_{plasma} = \frac{\mu_0 J_{\phi,plasma}}{B_{\phi,plasma}} \approx \frac{2R_0 I_p}{a^2 I_{TF}} \quad \text{(for large A and circular cross-section)}$$



Averaging $\bar{\lambda}_{sheet}$ over the plasma surface area gives[1] :

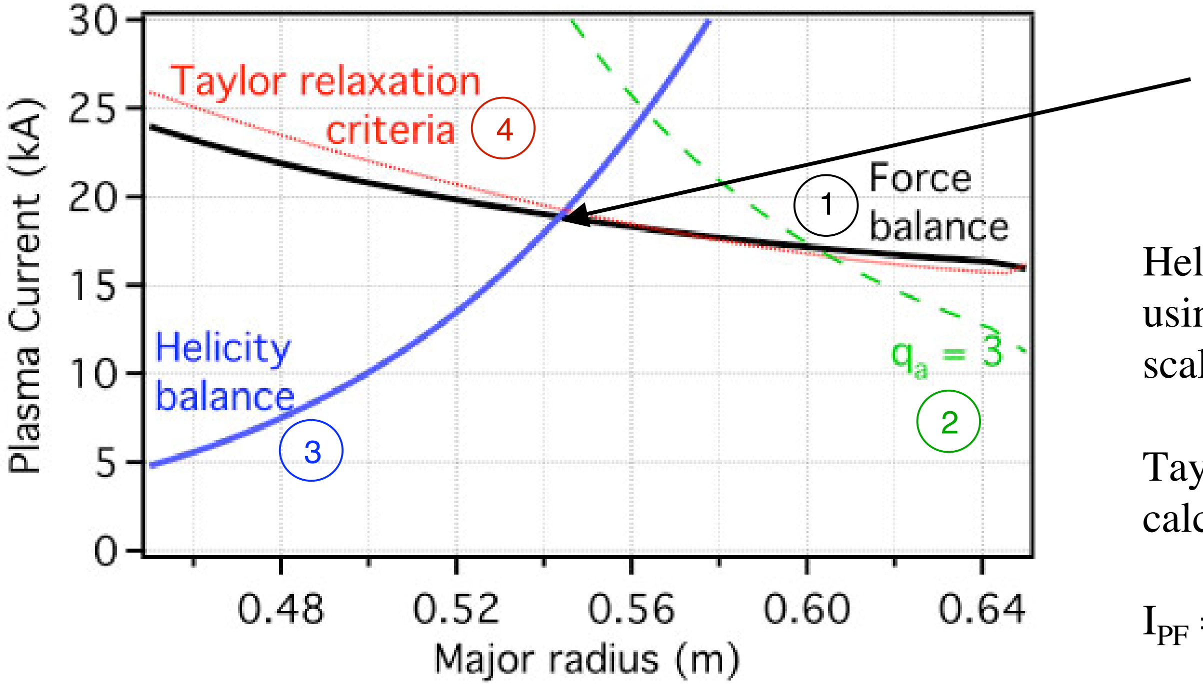
$$\frac{L}{2\pi a} \bar{\lambda}_{sheet} > \bar{\lambda}_{plasma}$$

which leads to

$$I_p < \varepsilon I_{TF} \left[\frac{N_{inj} I_{inj}}{I_{TF}} \frac{R_0}{\bar{R}_L} \frac{L}{2\pi\delta} \right]^{1/2}$$

[1] Holcomb, C.T. et. al., Phys. Plasmas **13**, 2006

Maximum sustained plasma current achieved when plasma simultaneously satisfies four criteria



$I_{p,max}$ ~ 19 kA at R_0 ~ 55 cm

Helicity balance calculated using ITER97L confinement scaling

Taylor relaxation limit calculated using $I_{inj} = 2$ kA

$I_{PF} = 1.2$ kA, $I_{TF} = 300$ kA

- $I_{p,max}$ realized when plasma is in force and helicity balance
  - Determined when black and blue lines cross
- Operation point satisfies tokamak stability and Taylor relaxation criteria
  - $I_{p,max}$ point is below red curve
  - $q_a > 3$ at $I_{p,max}$ point

Flux amplification is observed with outboard midplane plasma gun DC helicity injection

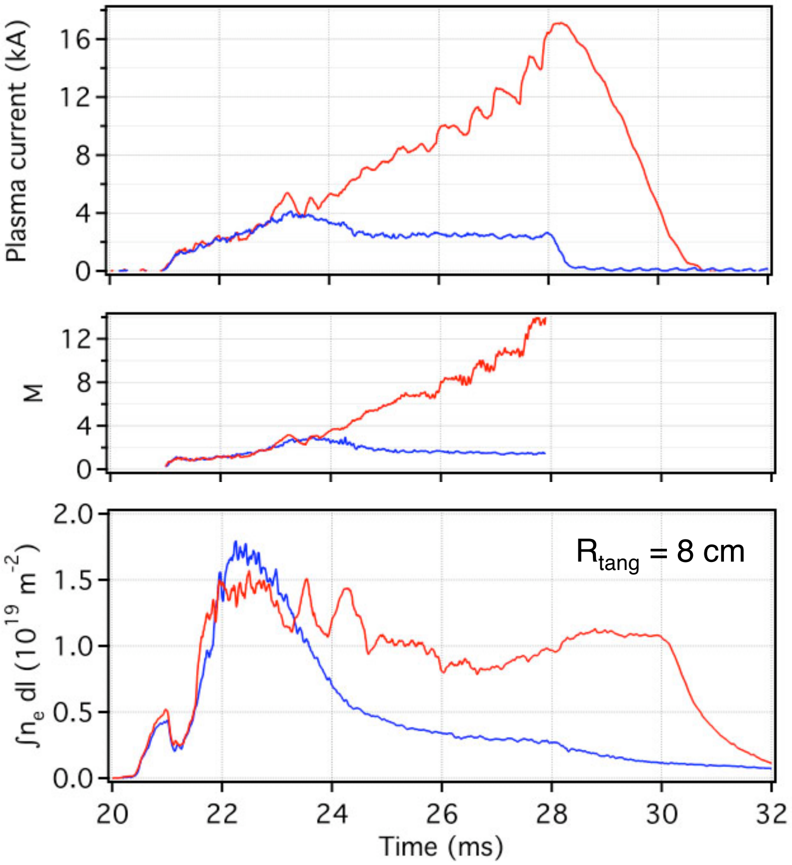
- Initial results suggest agreement with crude model
  - I_{PF} field reversal threshold ~ 1.2 kA → agrees with 2-D filament model
  - $I_{p,max}$ ~ 17 kA → correlates with helicity and force balance limit
  - R_0 ~ 55 cm at $I_{p,max}$ determined from midplane Mirnov measurements

- Evidence of flux amplification
  - G ~ 2, M > 12 achieved
  - Long I_p decay after injector shutoff
  - $\int n_e\,dl$ suggests improved particle confinement with flux amplification
  - Line averaged density near Greenwald density limit

Single gun, static B field discharges

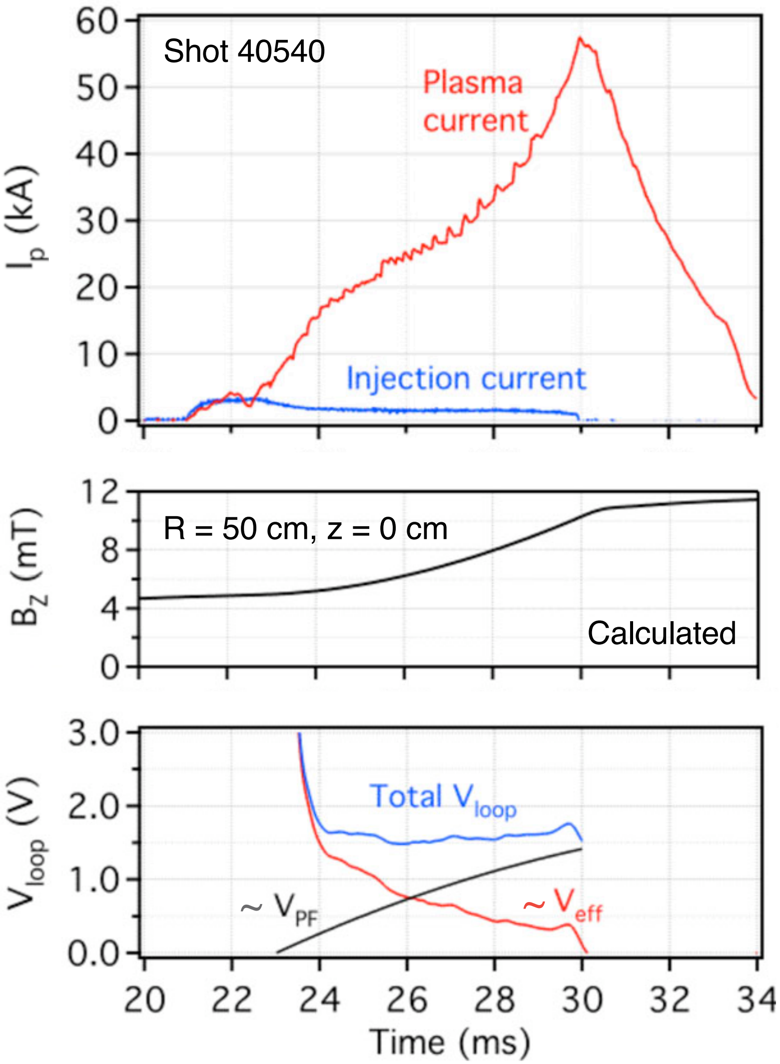
Shot 39762: I_{TF} = 300 kA, I_{PF} = 1.3 kA

Shot 39761: I_{TF} = 300 kA, I_{PF} = 1.2 kA

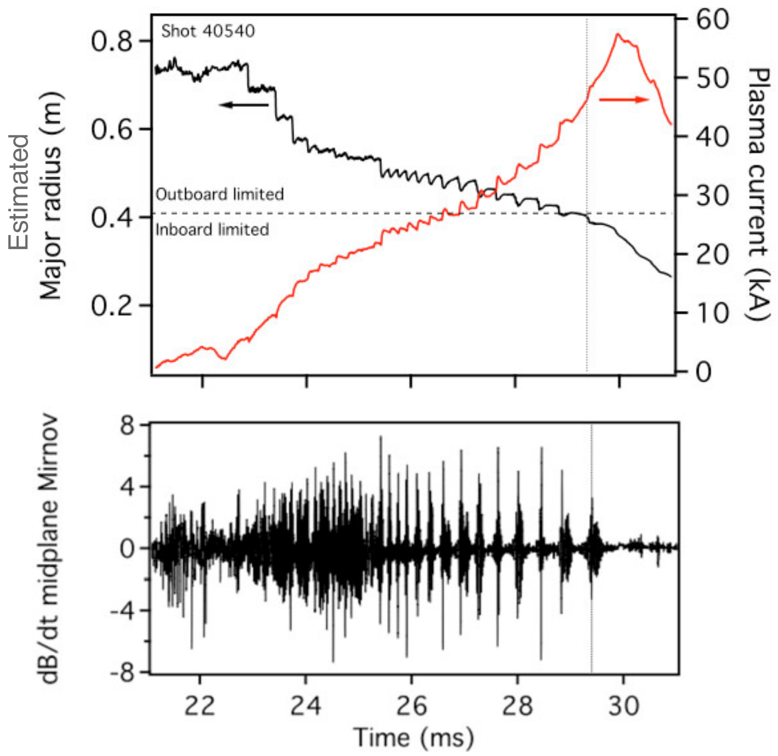


Poloidal field magnetic induction coupled to tokamak-like plasma during DC helicity injection

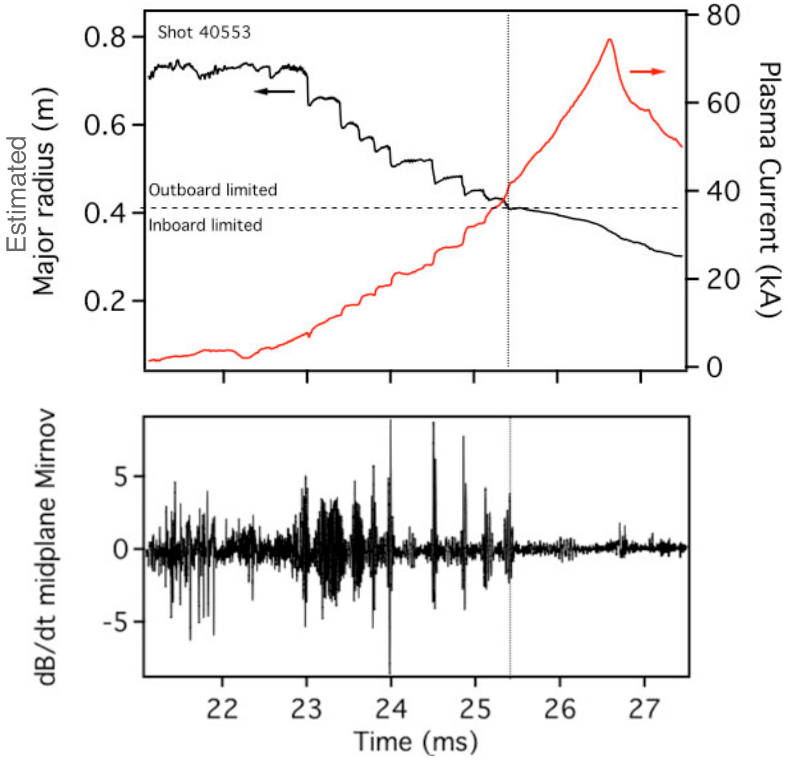
- AC helicity injection successfully coupled to DC helicity injection
  - Low B_v required for field reversal
  - Larger B_v required after field reversal to maintain radial force balance with larger I_p
- Demonstration of B_V field ramp during gun injection
  - Two plasma guns in operation
  - Ramp begins after field reversal
  - Induction provides additional current drive
- $V_{total} \approx 1.5$ V
  - V_{eff} calculated using R_0 and plasma shape estimation from magnetic measurements
  - V_{PF} calculated using 2-D vacuum field model that includes wall effects



The tokamak-like plasma can “detach” from the gun during rapid PF ramps



Slower PF ramp
Plasma detaches at 29.4 ms



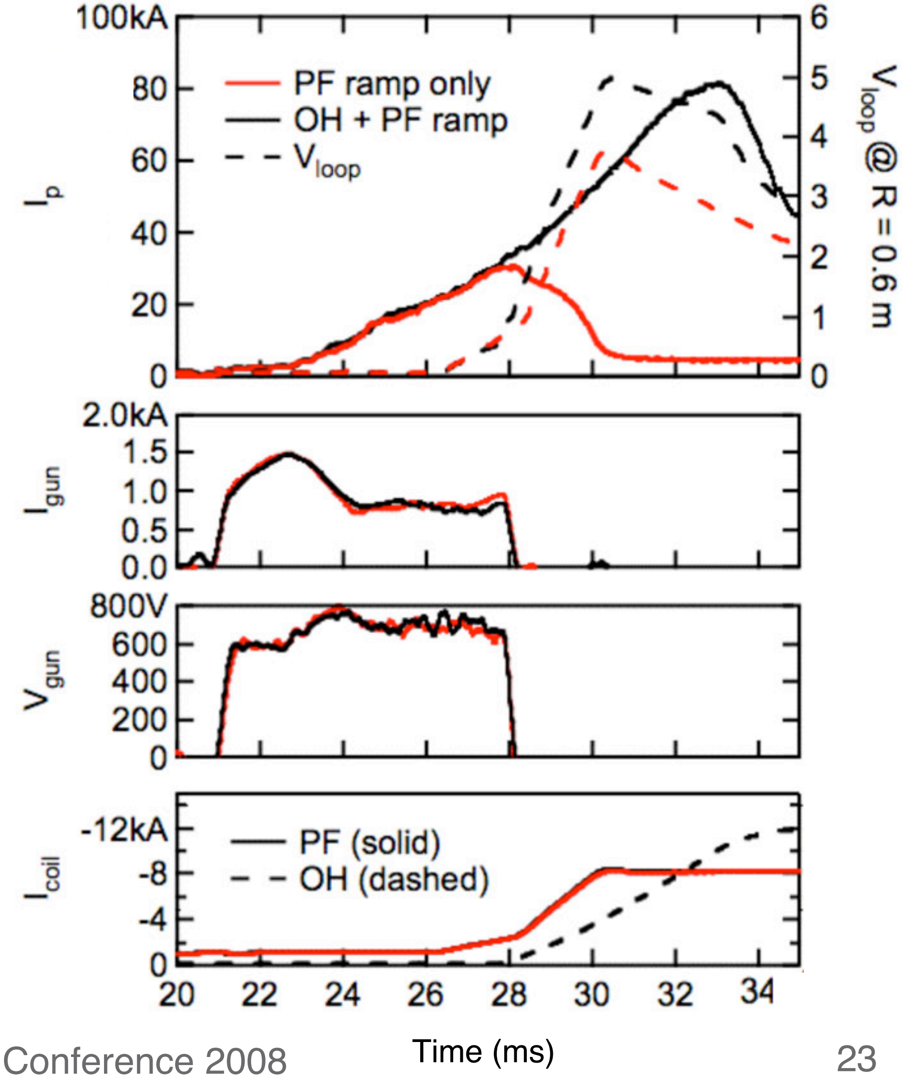
Faster PF ramp
Plasma detaches at 25.3 ms

After detachment, current drive is purely inductive and MHD activity is reduced.

Handoff from point-source DC helicity injection to OH induction has been demonstrated

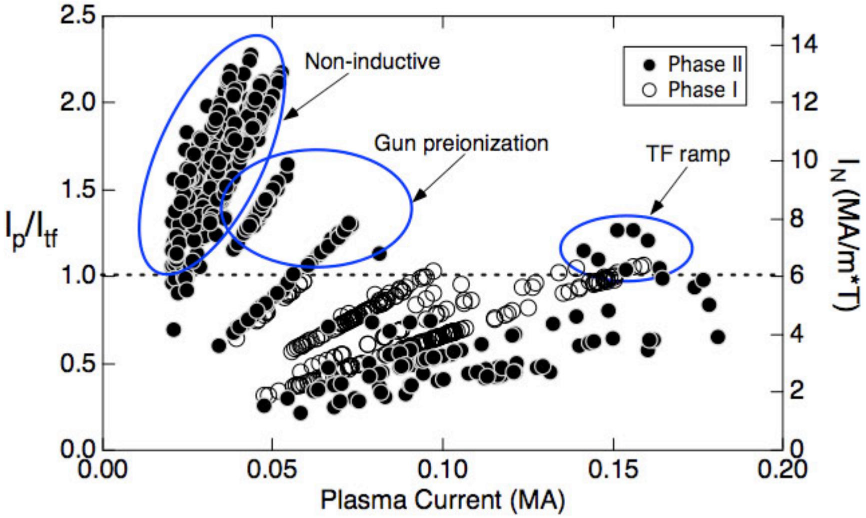
- Data shown for single outboard plasma gun system
  - < 10 mWb of OH flux
  - Prototype system → OH handoff experiments with new 3 gun system are planned

- Demonstrated using both point-source geometries
  - OH drive is applied after DC helicity injection is terminated
  - Outboard limited plasmas have longer L/R decay → easier to “catch” plasma after injector termination



Non-inductive startup provides a path to high β_t operations in the $I_N > 12$ regime on Pegasus

- Point-source edge current drive provides tool for modifying the current profile
- Access to $I_N > 12$ achieved using non-inductive startup
- No evidence of β stability limits at high I_N
  - Discharges have been limited by available current drive
- Non-inductive startup with hand-off to OH drive will extend the operational space
- Hand-off to HHFW in future



$$\beta_t = \beta_N I_N \qquad I_N = I_P / aB_\phi$$

$$\beta_t = \frac{2\mu_0 \langle p \rangle}{B_{\phi 0}^2}$$

Summary

- ST startup and current drive via point-source DC helicity injection was demonstrated on the Pegasus Toroidal Experiment
  - Poloidal flux amplification observed and correlated with inboard field reversal
  - Maximum I_p described by force and helicity balance
  - Observed increase in L/R decay and modest plasma heating

- A new midplane gun array provides a test of the geometric dependence of point-source DC helicity injection
  - A model is being developed based on early plasma gun work
  - Initial single gun, static B field results consistent with simple max I_p model
  - Evidence such as flux amplification and increased particle confinement suggest a tokamak-like plasma is formed in the vacuum region using outboard injection

- Magnetic induction compatible with DC helicity injection
  - PF induction provides current drive and maintains radial force balance with larger I_p plasma
  - Fast PF ramps can cause the tokamak-like plasma to detach from the gun
  - Handoff to OH induction also demonstrated

Future work

- Near term experimental plans
  - Planned hardware upgrades: Larger bias voltage supply, 20% higher TF fields
  - > 100 kA target plasmas using 3 guns + PF induction
  - Handoff > 100 kA target plasmas to OH induction

- Modeling and analysis
  - Develop and test more sophisticated model of I_p limits
  - Plasma geometry calculation using current filament code or G-S solver
  - PF induction/compression modeling using TSC

- Long term goals
  - Develop and test new plasma gun head designs to optimize current injection while maintaining low impurity injection
  - Gain a deeper understanding of the initial plasma relaxation into a tokamak-like configuration using internal probe measurements and simulations
  - Develop a complete predictive model for optimizing a plasma gun point-source system in an arbitrary tokamak geometry
  - Demonstrate high-power tokamak startup and handoff to HHFW current drive