Active Control of Plasma Boundaries Using Edge Currents

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Poster

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Active control of plasma boundaries using edge currents has been demonstrated in numerous experiments to produce edge conditions favorable to MHD stability, non-inductive current drive, fueling control, impurity and helium exhaust from the core plasma, reduced divertor heat loading, and access to enhanced performance regimes. There has been, however, no routine operational use of feedback stabilization to achieve and maintain favorable edge conditions for high performance plasmas. This work investigates the experimental facilities required to evaluate five innovative methods for edge stabilization employing halo currents, electrostatic biasing, segmented divertor biasing, current injection, and edge ergodization. These methods involve edge parameters and physics, and could be investigated in a tokamak using an electrically isolated and biasable passive shell and floating divertors, as electrodes for applying edge biasing. The stabilization of the plasma edge boundary is a neglected and next-step need.

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Driven Helical Plasma Surface Currents

• Recent experimental and theoretical work suggests that a mantle of negative edge current may have a stabilizing effect on long wavelength external kink modes.
- experiments: TFTR- J. Kesner *et al*., S. Sabbagh, *et al.*
- theory: J.J. Ramos, *et al*., L.E. Zakharov, *et al*., J. Kesner,*et al.*

• A current source located inside vessel on the plasma edge could generate perturbing magnetic fields on a faster time scale and with less power than external magnetic field coils.
  - fast time response and low power levels.
  - independent of vessel magnetic penetration time scales.

• ***Useful for Reactor External Magnetic Coil Design***

Sensitive helical plasma current stabilization experiments might conveniently *elicit or amplify mode responses and unforeseen phenomena* useful for optimizing reactor external coil design.

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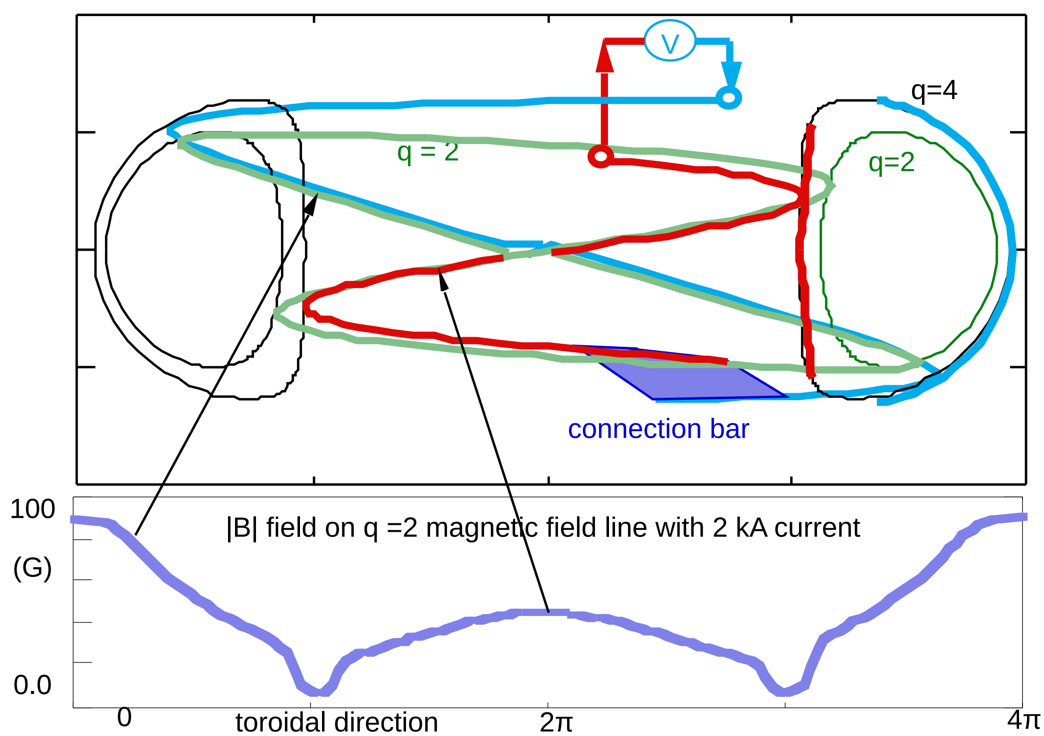
PLASMA HELICAL COIL
FOR
FEEDBACK STABILIZATION

• Helical current injected on q = 4 surface can produce a magnetic field on q = 2 surface for feedback control of shaped plasmas.

Simulation of helical plasma current on q=4 surface can produce a magnetic field on q=2 surface for mode control



• q=4 and q=2 field lines are well aligned except at corners

M. Okabayashi, PPPL

• Methods for helical current injection:
- Plasma electrodes
- Thermionic emitters
- Plasma electron sources

Active Control of Plasma Boundaries Using Edge Currents

• This work investigated the experimental facilities required to evaluate five innovative methods for edge control and feedback stabilization:

- Driven halo currents
- Electrostatic biasing
- Segmented divertor biasing
- Edge ergodization
- Current injection

• Two conceptual designs were investigated for an experimental facility with electrically isolated divertors and sufficient versatility to evaluate active mode stabilization of high power tokamak edge plasmas.

DRIVEN HALO CURRENTS*

• *ELECTRODES DRIVE A FORCE-FREE HELICAL CURRENT IN THE PLASMA HALO, CREATING A FIELD WHICH ACTS TO STABILIZE THE PLASMA.*

• A TSC simulation was performed. The results exhibited a weak dependence on halo width and vacuum region resistivity. Larger gain parameters and hotter, wider halo region were always found to be more effective (see figures).

• The following scaling relation was derived for ITER-like geometry :

$$I_H^P = I_p \left(\frac{\mu_0 I_P}{R B_T} \right) \left(\frac{Z}{a} \right) \left(\frac{|n|}{\Delta\theta} \right) = \mathbf{0.03}\ I_p ,$$

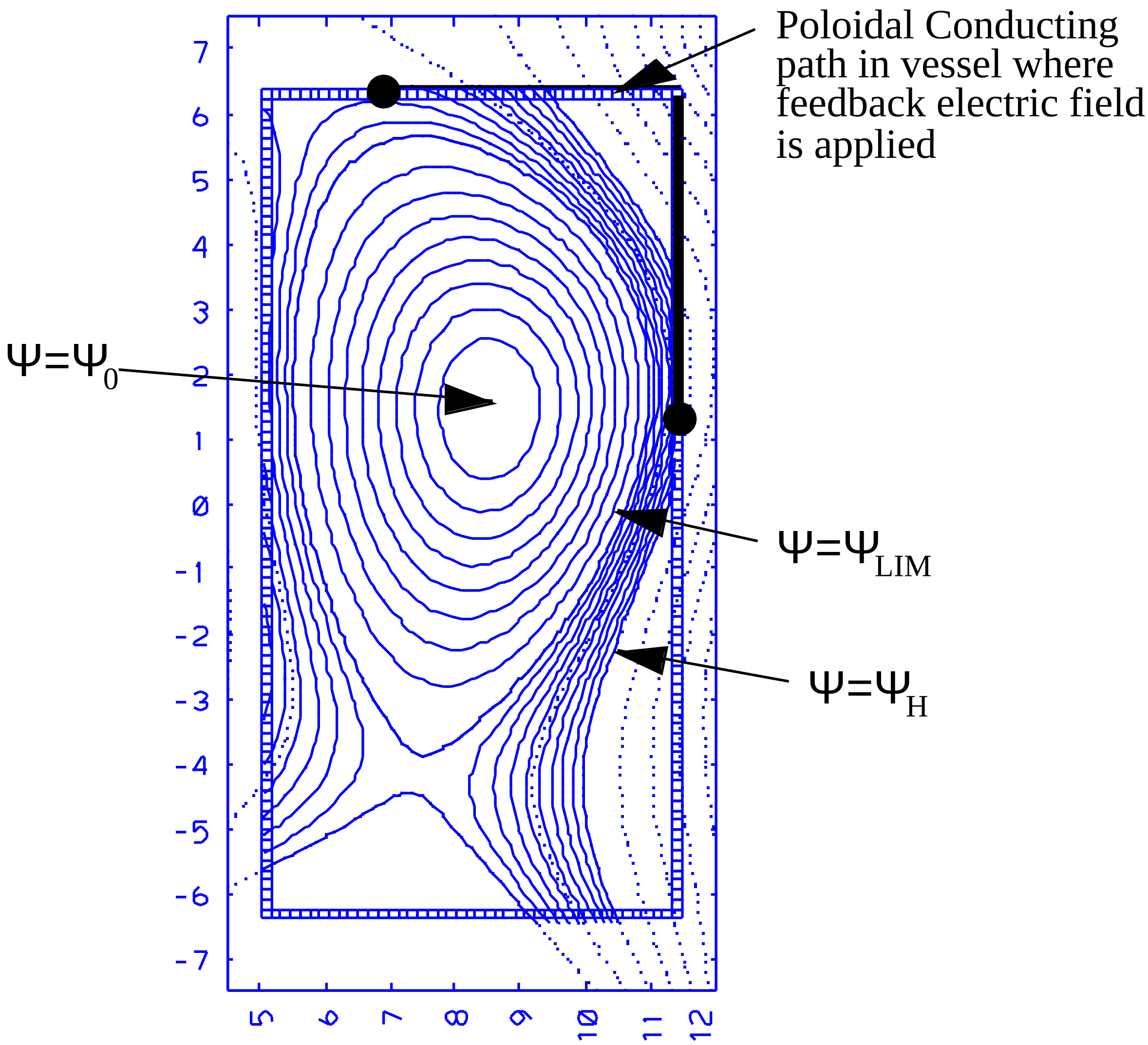
where *n* is the field index, Z is the vertical displacement, q is the poloidal angular extent in radians, and the other variables are in standard tokamak notation.

• This result is in agreement with maximum currents calculated in the MHD simulation; a maximum poloidal current in the vessel of about 1×10^4A for a PBX-M size machine, and about 6×10^5A for ITER.

• The method appears to be feasible for a wide range of plasma parameters, and would minimize control field interactions with vessel cryogenic structures, and thereby reduce recirculating power requirements for high power reactors.

*S. Jardin and J. Schmidt,"TSC Simulation of Feedback Stabilization of Axisymetric Modes in Tokamaks Using Driven Halo Currents", Nucl. Fus. in press.

TSC SIMULATION*



- Vessel volume divided into 3 regions:

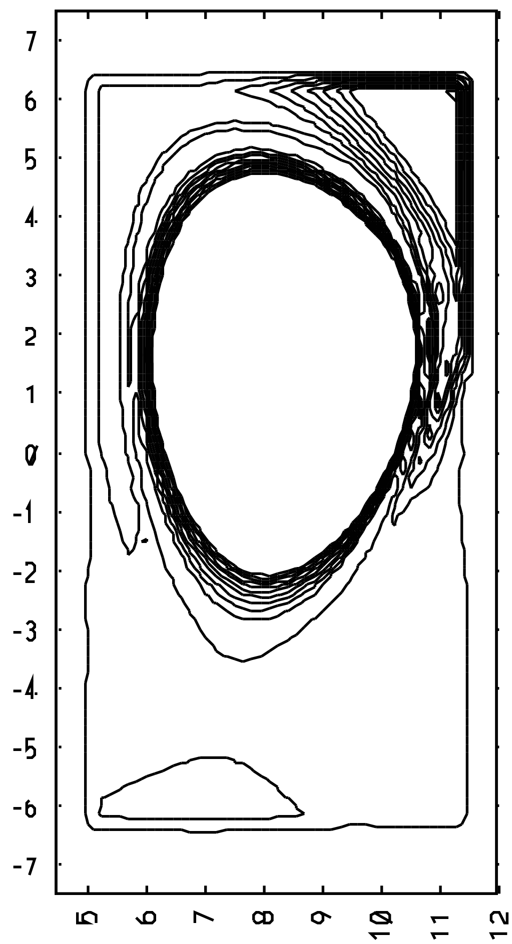
Plasma region: $\Psi_o < \Psi < \Psi_{lim}$

Halo region: $\Psi_H > \Psi > \Psi_{lim}$

Vacuum region: $\Psi_H > \Psi$

- The upper right corner of the vessel has a voltage difference proportional to plasma vertical displacement. Plasma vertically stabilized.

TSC SIMULATION DEMONSTRATES DRIVEN HALO CURRENTS CAN STABILIZE ITER*



Wide Halo

• Poloidal current streamlines at a fixed time for halo feedback calculations with halo width $W_H = (\Psi_H - \Psi_{lim})/(\Psi_{lim} - \Psi_o) = 0.4$, and $T_H = 20$ eV, $a = 266$ cm, $E_{max} = 40$ V/m and $T_V = 0.1$ eV. Streamlines deep inside the plasma region are not shown.

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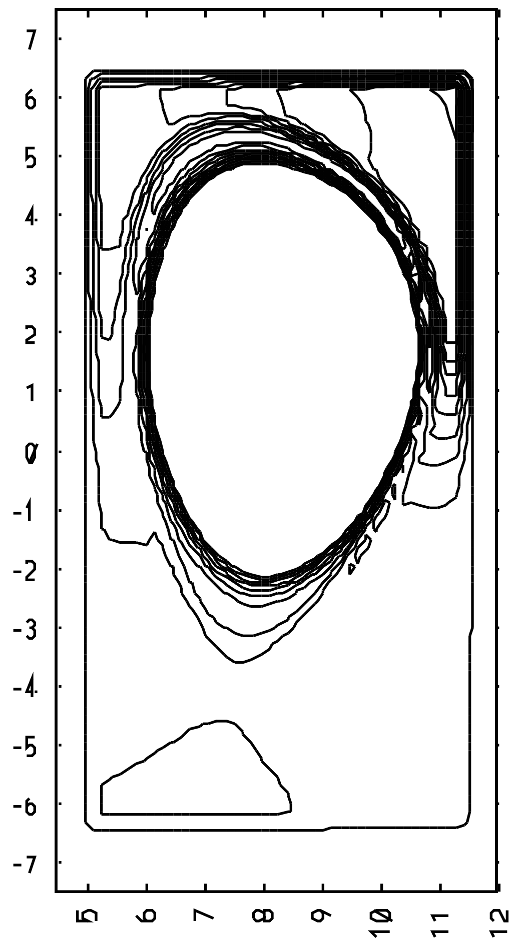
TSC SIMULATION DEMONSTRATES DRIVEN HALO CURRENTS CAN STABILIZE ITER*



Narrow Halo

• Poloidal current streamlines at a fixed time for halo feedback calculations with halo width $W_H = (\Psi_H - \Psi_{lim})/(\Psi_{lim} - \Psi_o) = 0.01$, and $T_H = 20$ eV, $a = 266$ cm, $E_{max} = 40$ V/m and $T_V = 0.1$ eV. Streamlines deep inside the plasma region are not shown.

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DIVERTOR AND PASSIVE PLATE BIASING*

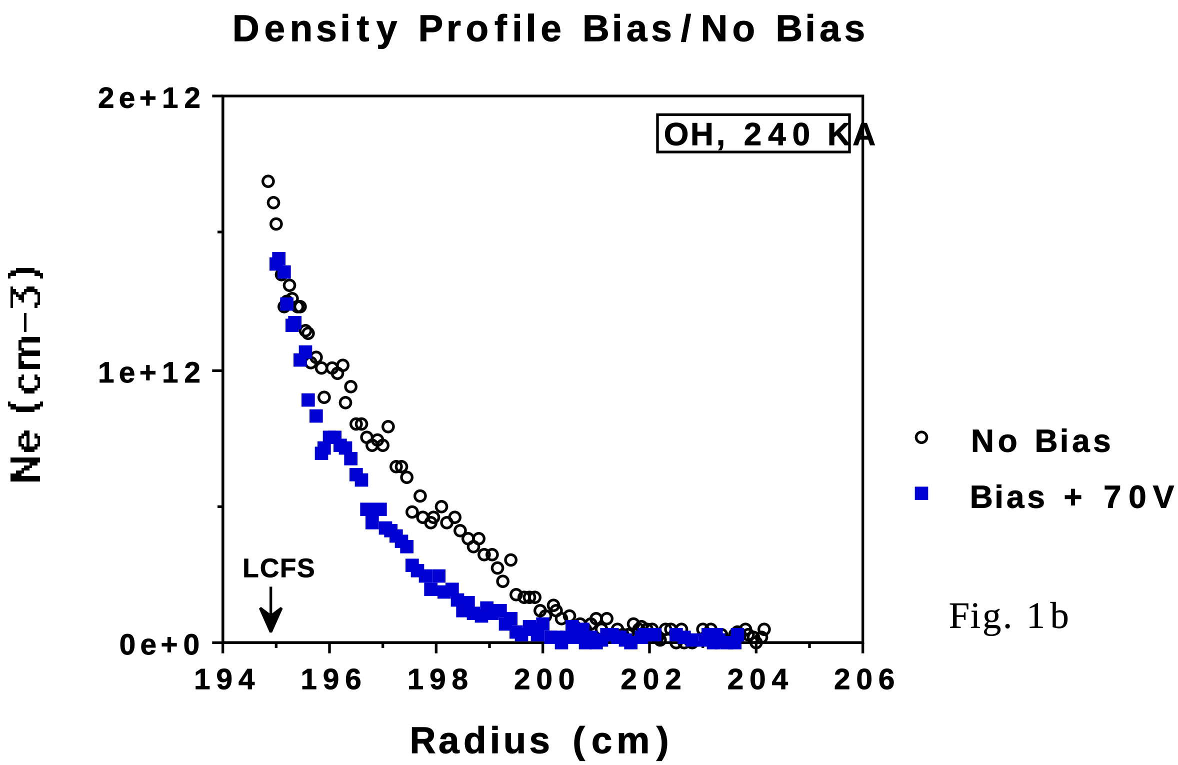

Fig. 1b

- Modification of PBX-M density profile with positive bias applied to outboard divertor strike points

- Biasing can adjust edge density in front of RF antenna

*L. Schmitz, UCLA and UCSD, *et al.*, in *Tokamak Plasma Biasing*, p 285, IAEA Tech. Com. Mt., Montreal, 1992, and in PPPL Report No. 3250, June 1997.

DIVERTOR AND PASSIVE PLATE BIASING*

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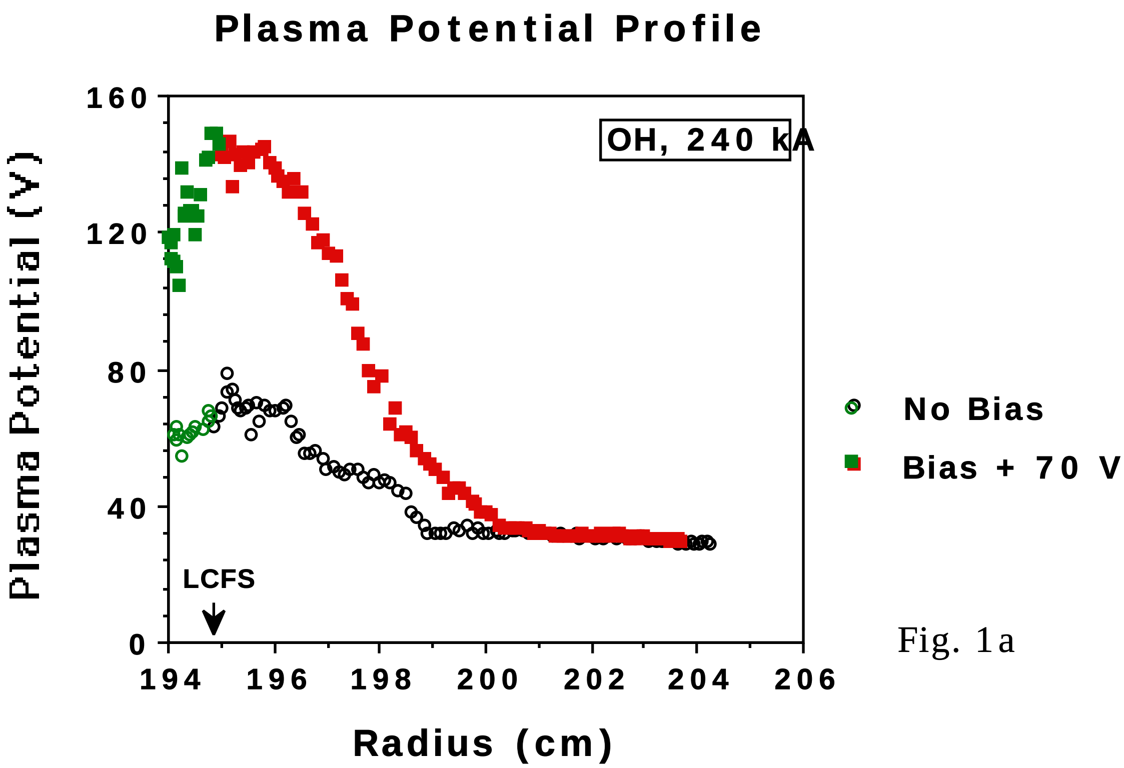

Fig. 1a

• Modification of PBX-M plasma potential with positive bias applied to outboard divertor strike points

• Biasing controls plasma edge potential.

DIVERTOR AND PASSIVE PLATE BIASING*

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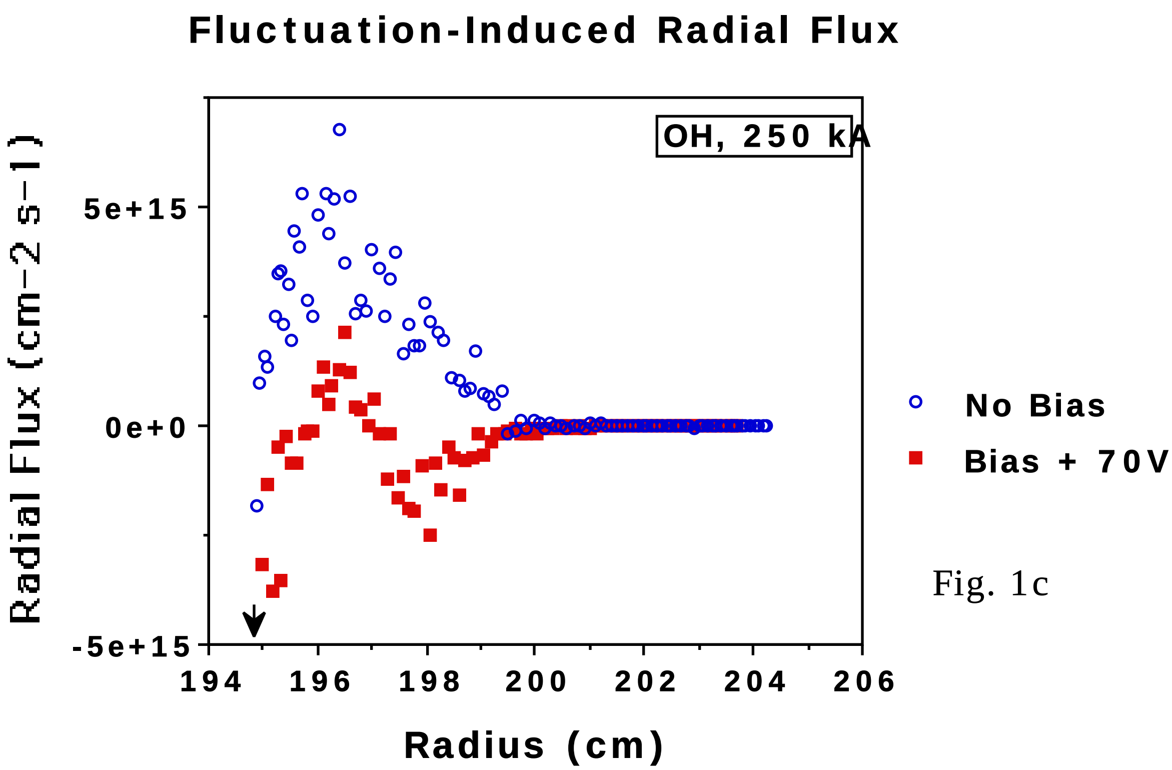

Fig. 1c

• Modification of PBX-M fluctuation-driven radial particle flux with positive bias applied to outboard divertor strike points.

• Biasing can control edge fluctuations

• Biasing lowered H-mode threshold 25%

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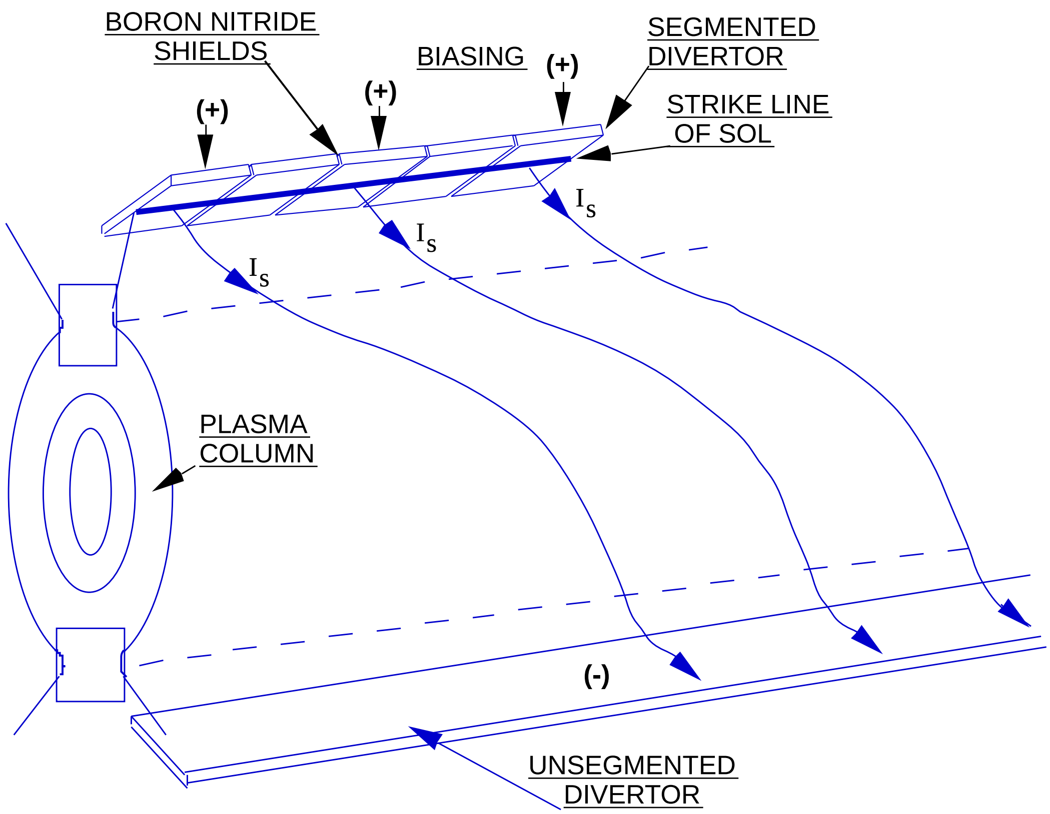
SEGMENTED DIVERTOR BIASING*



• Toroidally segmented upper outer divertor for feedback control of scrape-off layer currents. Bias is applied to different segments by the feedback stabilization control system.

• The resulting injected helical flux pattern will match closely the dominant eigenmode of the external kink and be phased to provide a restoring force on the growing instability.

* R. Goldston, "Toroidally segmented divertor biasing and current injection", *Contrl. Fus. and Plas. Phys*., in press.

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SEGMENTED DIVERTOR BIASING*

- ***BIAS CURRENT***

A preliminary estimate of the current required for an instability of poloidal mode number m (~nq) requiring a response capability of dB_q/B_q is:

$$dI/I_p \sim dB_q/(mB_q) \sim 1\%$$

For a PBX-M size machine @ I_p ~ 750 kA, this corresponds to a "stripe" of current of about 7.5 kA. For ITER @ I_p~21 MA, this corresponds to about 210 kA per stripe.

- ***BIASING ELECTRODES***

This current density can be provided by simple biasing of divertor electrodes, if plasma conditions at electrode allow:

$$n_e/T_{ev} \sim 2 \times 10^{20}$$

This can be achieved, with a density of 6 x10^{19} at the divertor plate and a temperature of 10 eV. A similar estimate for ITER gives a requirement on n_e/T_{ev} of 5x10^{20}, which is in the range of what is expected.

- ***BIAS VOLTAGE***

For a PBX-M size machine with an SOL T_{ev} ~30 eV, the voltage drop estimated using the parallel Spitzer resistivity in the SOL for Z_{eff} ~ 2, gives a very reasonable of value of 45 V for the voltage resistive drop, and depending on the SOL electron temperature, similar or even lower for ITER.

SEGMENTED DIVERTOR BIASING*

- ***POWER DISSIPATION***

  RMS resistive power dissipation in the SOL for 2 “stripes” would be ~350 kW for a PBX-M size machine, and ~10 MW for ITER. A similar power dissipation would be expected in the sheath for the case of simple plate biasing.

- ***INDUCTANCE***

  The inductance of a current stripe from the stored poloidal field energy for a PBX-M size machine gives $L \sim 1.3 \times 10^{-6}$ H. For ITER ~2x higher. For a reactive power $\leq$ 2x the resistive power, the frequency range limited to ~ 4 kHz in a PBX-M size machine, and ~50 Hz in ITER - appropriate for stabilization of *resistive-wall modes* and *tearing modes* in these devices.

- ***EXTERNAL KINKS***

  The resulting helical flux injection pattern is almost ideally shaped for control of external kinks since it will match nearly exactly the dominant eigenmode of the kink.

- ***TEARING MODE***

  • The resultant polidal mode structure will not match the instability precisely. Inclusion of higher n and/or m components may constrain the island motion.

* R. Goldston, "Toroidally segmented divertor biasing and current injection", *Contrl. Fus. and Plas. Phys*., in press.

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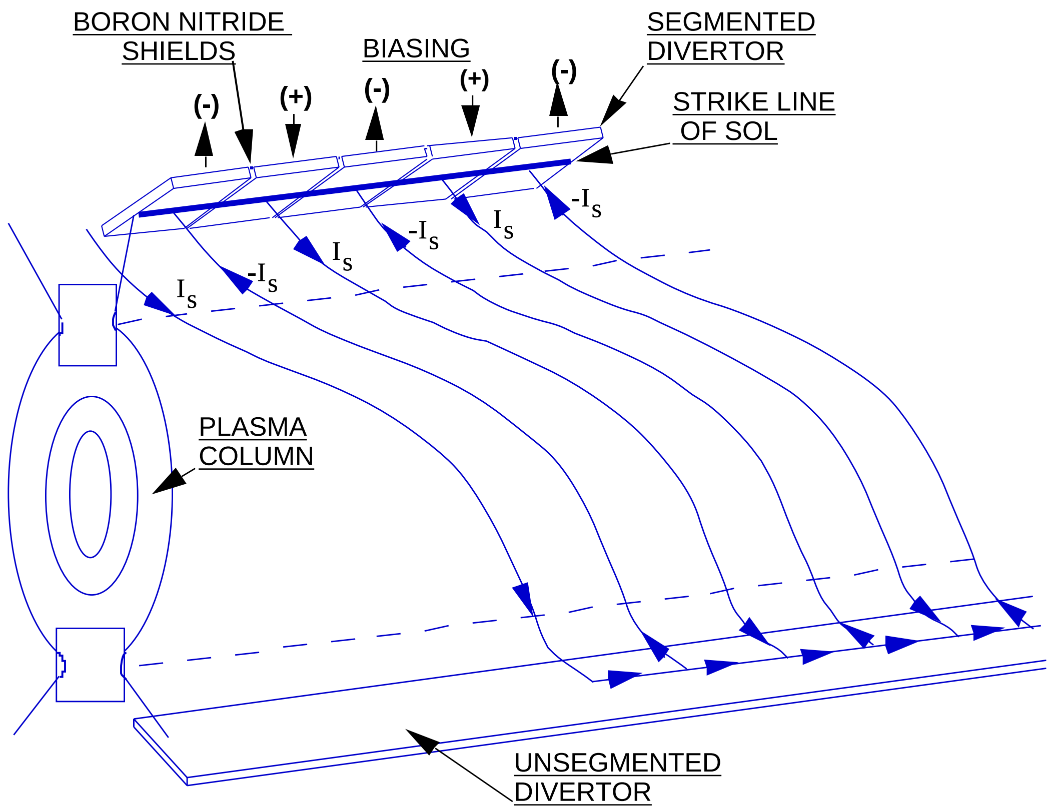
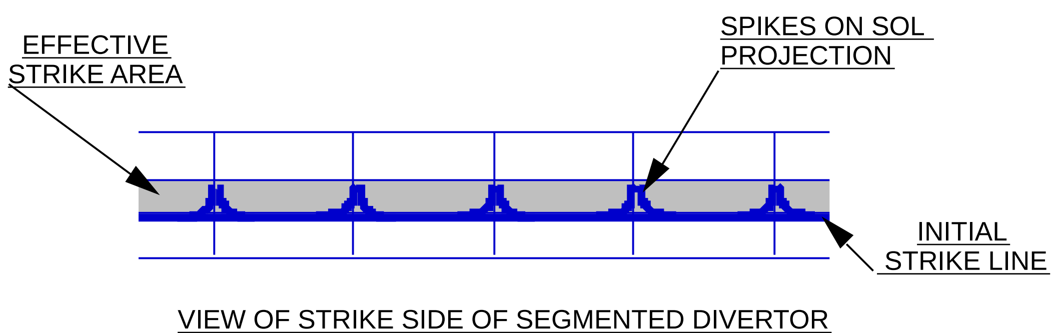
EDGE ERGODIZATION*



(a)

• Upper outer divertor toroidally segmented into separate electrodes with bias of opposite polarity applied to each segment.



(b)

• View of strike side of toroidally segmented divertor for twisting the magnetic field lines in the outer scrape-off layer so as to broaden the effective deposition area of the incident outer strike points.

* L. Zakharov, *et al.*, in PPPL Report PPPL-3250, June 1997.

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EDGE ERGODIZATION*

- SPATIALLY ALTERNATING BIAS

  The spatially alternating bias of one divertor plate without segmentation of the opposite plate drives current along the field lines in the SOL and causes an alternate twisting of the magnetic field lines in the flux tubes thrusting into the respective biased segment - causes spikes in strike-line perpendicular to the initial axisymmetric strike-line.

- ENERGY DEPOSITION

  The energy deposition will be averaged over a larger area, the width is determined by the length of the spikes (l_n). This averaging makes l_n to be the characteristic width of the energy deposition onto the divertor plate, resulting in substantial reduction of peak power density if the spike lengths exceed the initial width of the SOL. The length l_n of the spikes on the divertor plate may be estimated by neglecting the curvature of the geometry of the SOL

$$l_n = \frac{0.8 \, I_s}{l_\varphi \, B \sin \alpha} L$$

I_S is the current through each segment, B is the main magnetic field, α is the angle between the magnetic field line and the divertor plate in the plane of the magnetic surface, and L is the length of the magnetic field line between the middle of the plasma and the strike line on the divertor plate.

* L. Zakharov, *et al*., in PPPL Report PPPL-3250, June 1997.

EDGE ERGODIZATION*

- In terms of total current $I_{tot} = \Sigma|I_s|$ through the divertor plate, l_n may be expressed as

$$l_n = \frac{0.8\ I_{tot}\ q_{sol}}{I_\varphi\ B\ \sin\ \alpha}\ L\ , \qquad q_{sol} \equiv \frac{L}{2\pi R}$$

- For ITER, $B = 5$ T, $q_{sol} = 2$, $\sin\alpha$ @ 0.5 and $I_{tot} = 0.2$ MA corresponding to the available ion current from the plasma deposit, the resulting l_n is 13 cm, which is an order of magnitude larger than the initial value of ~ 1 cm.

• EDGE FLOWS

- *Plasma diffusing radially outward encountering an ergodic region at the edge, flows along the ergodized flux lines to the wall rather than along flux lines to the divertors.*

• IMPURITY FLOWS

- *Counter flowing impurity influxes from the wall are impeded by the pressure gradient of the outward flowing plasma. Residual inward flowing impurities reaching the ergodized edge layer are swept toward the divertors by the edge parallel flow.*

* from L. E. Zahkarov *et al.*, in PPPL-3250, June 1997.

EDGE CURRENT INJECTION

• The use of thermionic current injection has been demonstrated for driving steady state currents in the edge/SOL region of a tokamak plasma [1].

• The possible efficacy of using injected edge current for the stabilization of pressure driven kink modes has been proposed [2].

• A negative current mantle could be generated in the edge/SOL of a tokamak plasma either by means of:

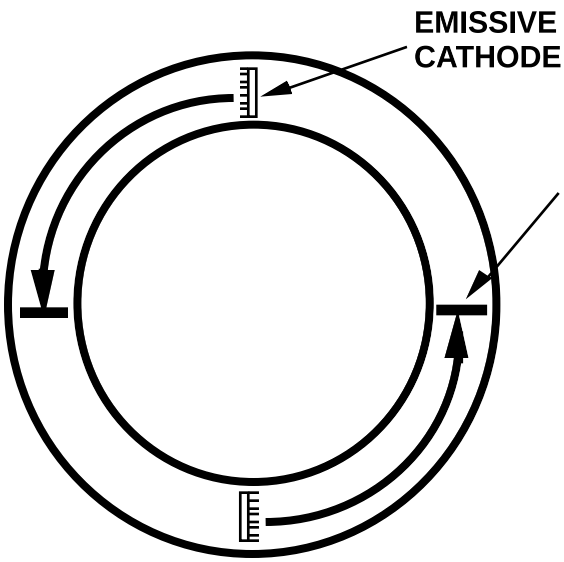
- hot cathode current injectors located near the SOL plasma
- electron current injection from small plasma sources

[1] M. Ono, in *New Ideas in Tokamak Confinement Research, Trends in Physics,* M. Rosenbluth, ed. , 410, 1994, AIP Press.
[2] J. Kesner, J. J. Ramos, S. C. Luckhardt, *Nucl. Fus.* **34**,(6), 795 (1994).

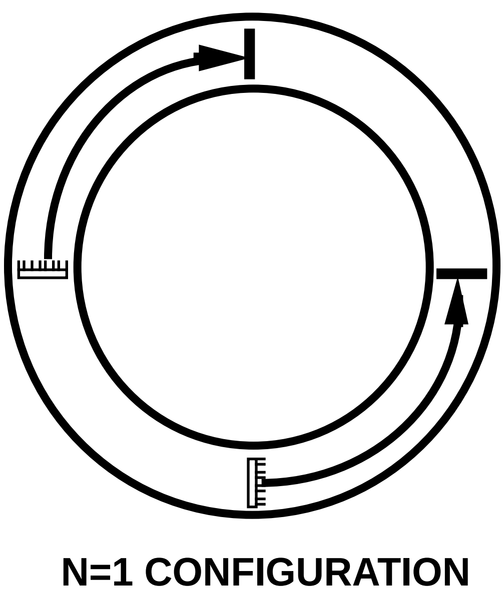
EDGE CURRENT INJECTION
*with thermionic emitters**

• n=0, n=1 Active Edge Current Control with Emissive Cathodes

ACTIVE EDGE CURRENT DRIVE
(TOP VIEW)



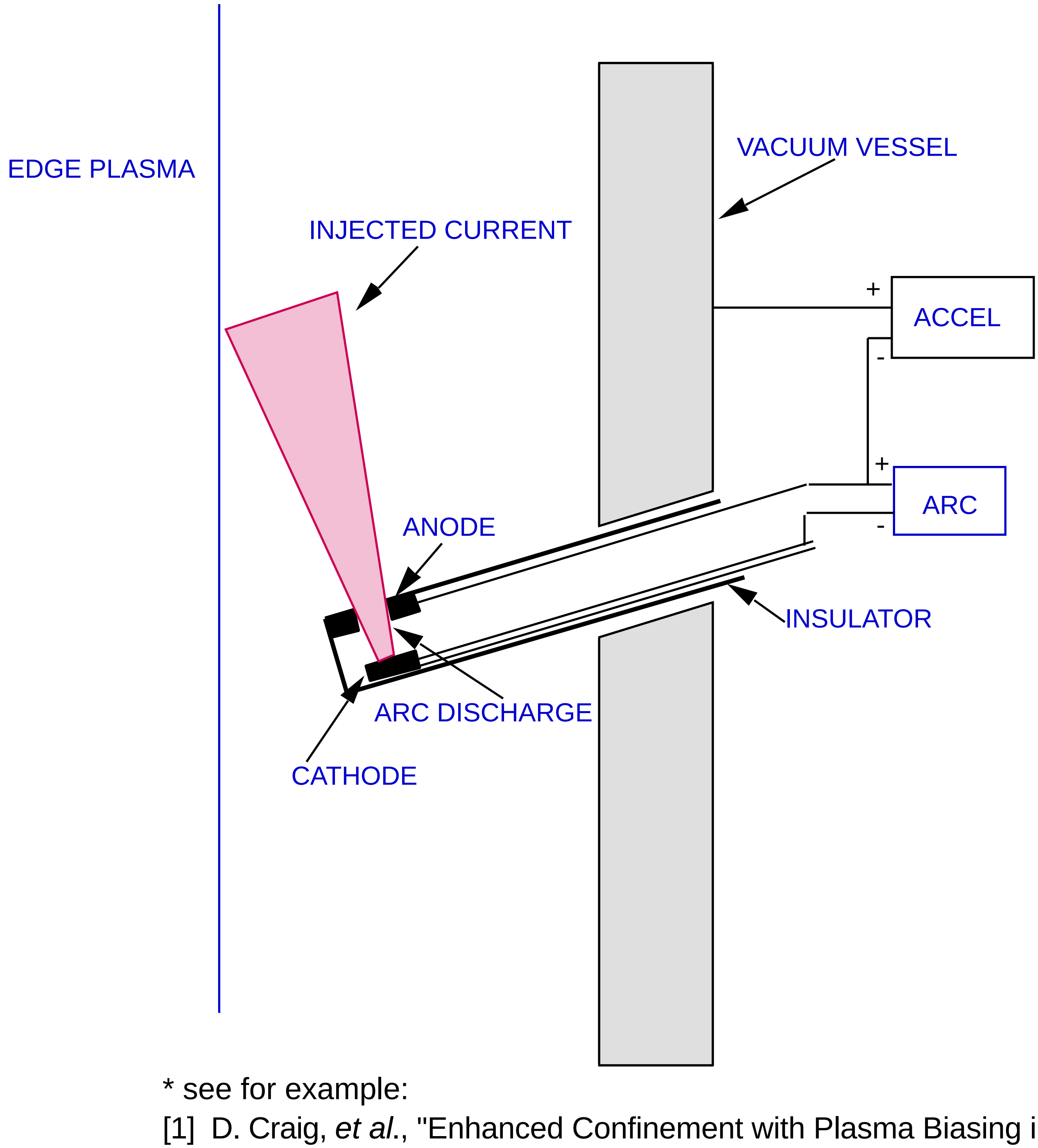
N=0 CONFIGURATION



N=1 CONFIGURATION

* S.C. Luckhardt and J. Kesner, "Edge Current Drive and Advanced Tokamak Experiments", to be published.

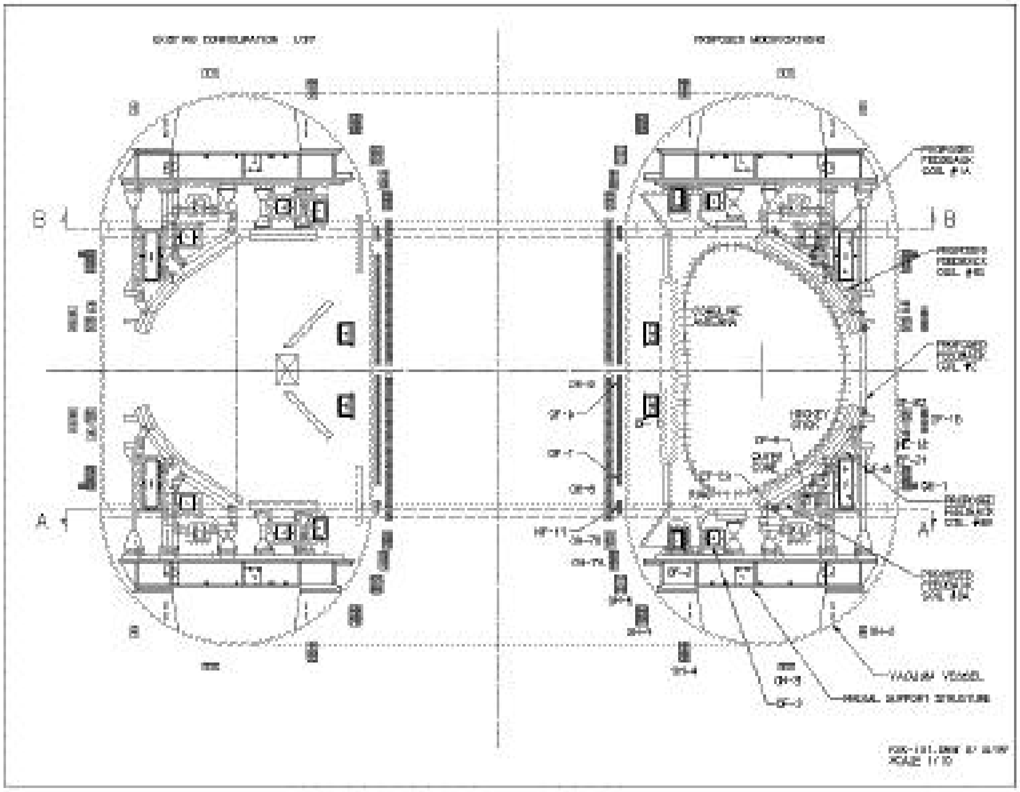
EDGE CURRENT INJECTION FOR FEEDBACK STABILIZATION
*with Plasma Electron Emitters**



* see for example:

[1] D. Craig, *et al.*, "Enhanced Confinement with Plasma Biasing in the MST Reversed Field Pinch", Phys. Rev. Lett., 79(10), 1865 (1997).

[2] G. Fiksel, *et al.*, Plasma Sources Sci. Technol., 5, 78 (1996).



(A) Present Configuration   (B) Alternate Configuration

(A) Partial schematic of present PBX-M facility showing the midplane pusher coil for highly-indented plasmas and the biasable, electrically isolated, poloidally and toroidally segmented, passive stabilizer plates. Electrical buses to each of the 5 passive plate elements (not shown) allow the application of bias voltages for edge control experiments.

(B) An alternate configuration of the PBX-M facility for performing the edge feedback stabilization experiments on double and single null "D" shaped plasmas.

SUMMARY AND CONCLUSIONS

• The five methods for plasma edge stabilization investigated in this work involve different responses from the underlying edge physics and may reveal unforeseen phenomena.

• Two conceptual designs were investigated for an experimental facility to evaluate the proposed methods for plasma edge stabilization requiring *electrically isolated divertors and isolated large area limiters or passive stabilizer plates*.

• It was found that the proposed edge stabilization methods can be implemented in concert with several available core stabilization techniques using either the available PBX-M facility for highly indented plasmas, or a feasible reconfiguration for double and single null "D" shaped plasmas.

• *The stabilization of the plasma edge boundary is a neglected and next-step need that can be addressed in the near term.*

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