Plasma Control Requirements for NCSX

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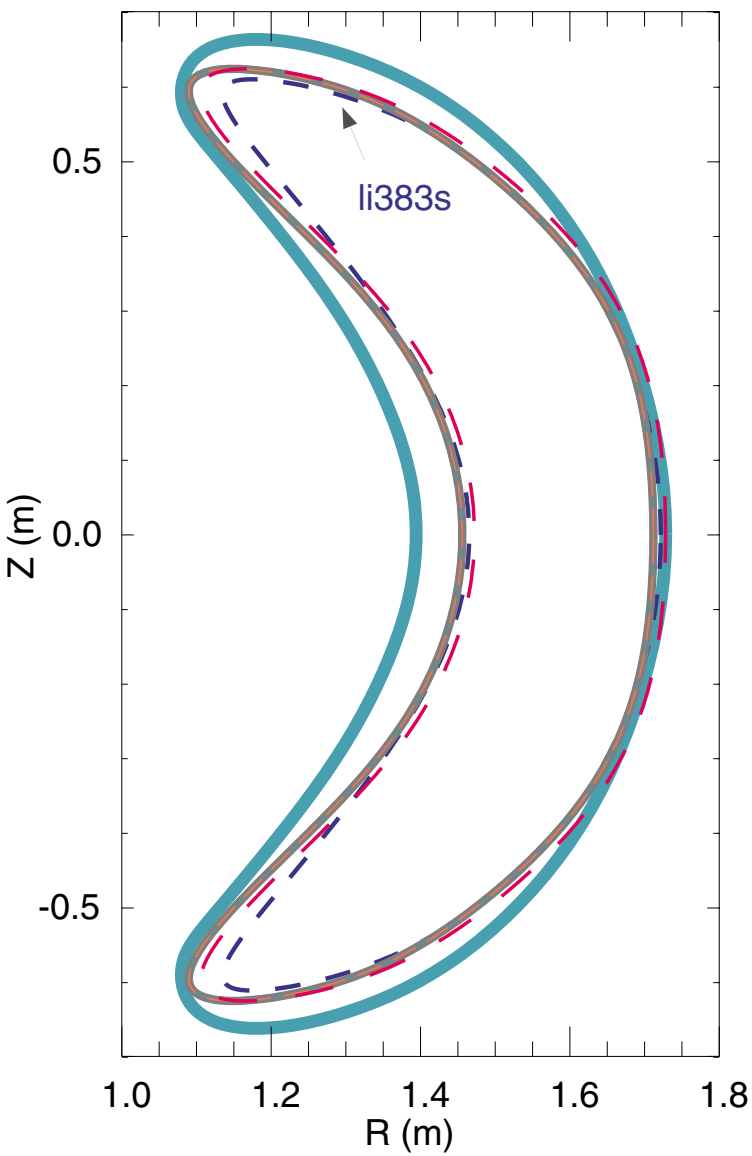
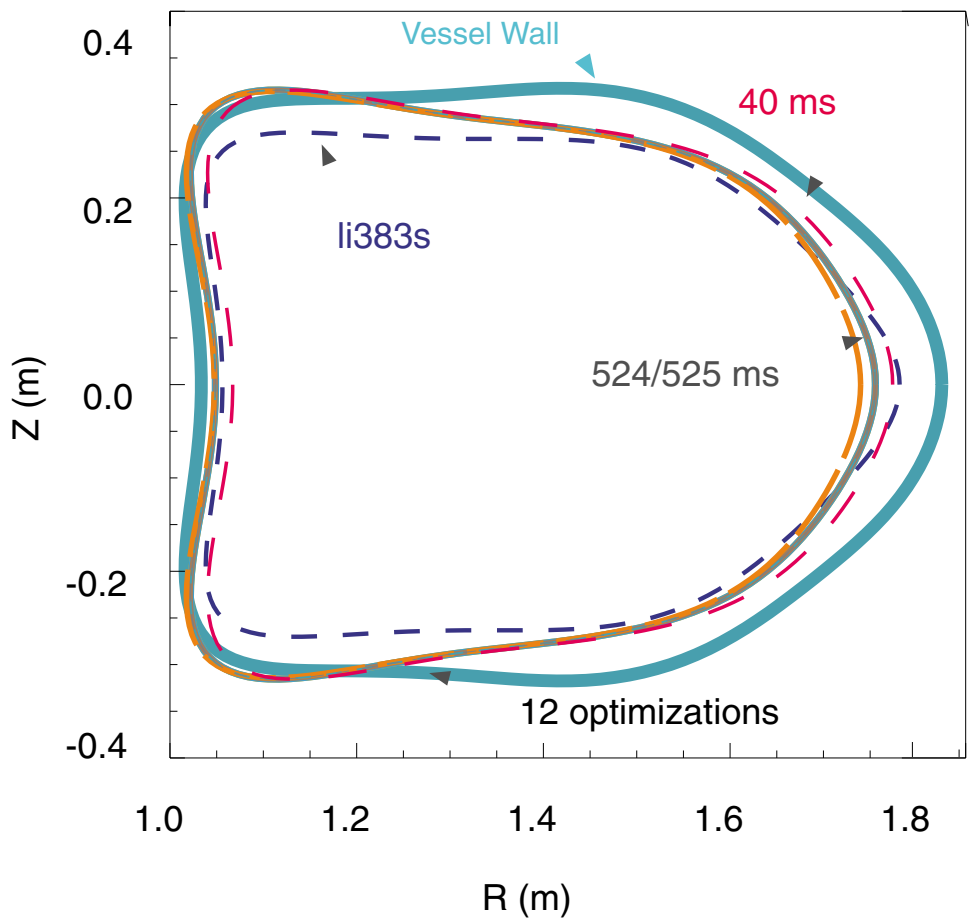
Plasma Control Is Needed for the NCSX Mission

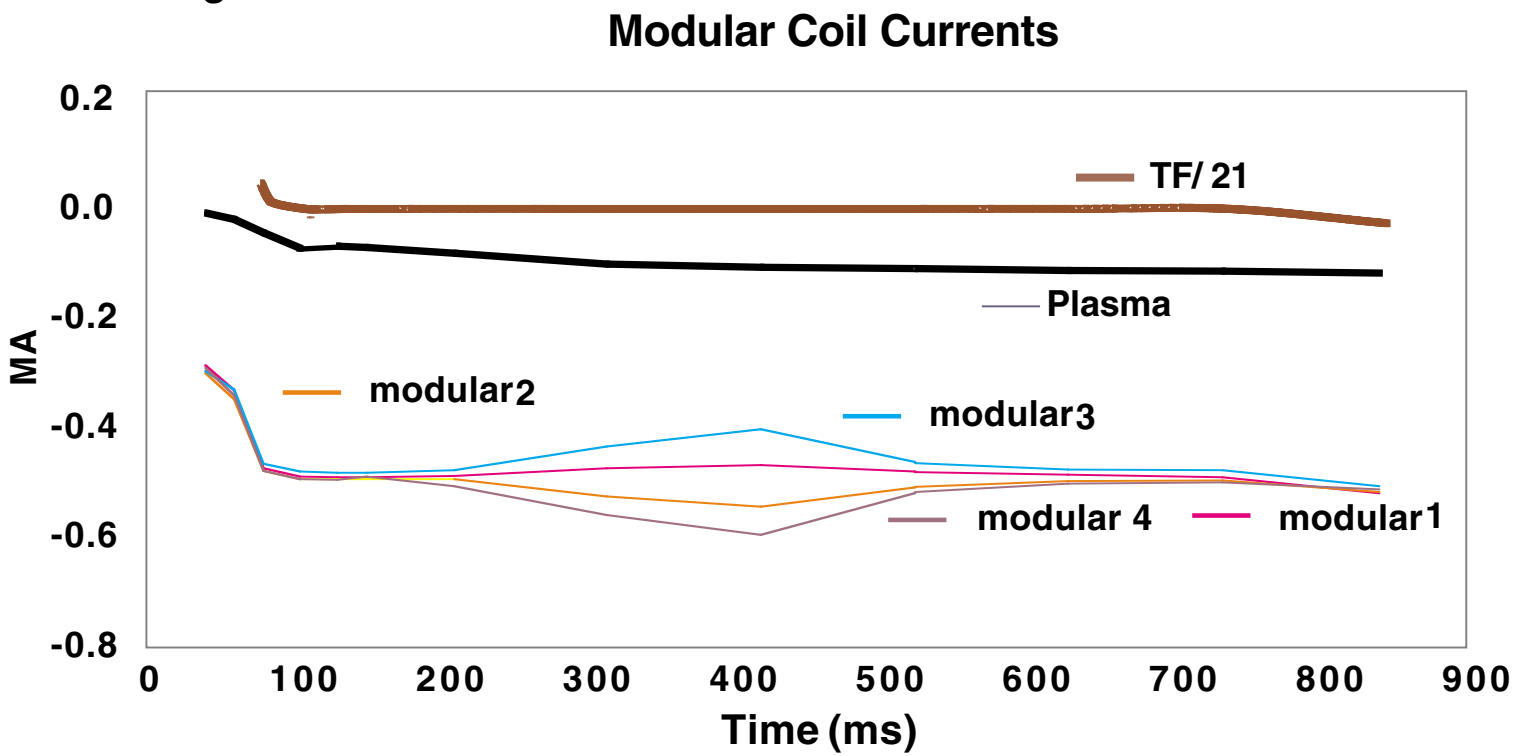
In developing a stable discharge evolution for the R=1.7m li383 configuration), the procedure used was to use the optimizer to directly address the plasma shape. This shape was known to produce good results with the coil set used at that time. After the fact, we evaluated the physics targets and found them to be in the acceptable range.

In work leading to the PVR, the optimization was done on explicit physics targets, Kink & ballooning symmetry and quasi-symmetry. After the fact we look at the plasma shape, shown below in the symmetry planes. Twelve cases are identical in shape. The one at 40 ms would not be a problem. The 524/525 ms shapes are a quirk of the optimization due to a minor ballooning unstable region.


li383s
0.5
0.0
-0.5
Z (m)
1.0
1.2
1.4
1.6
1.8
R (m)

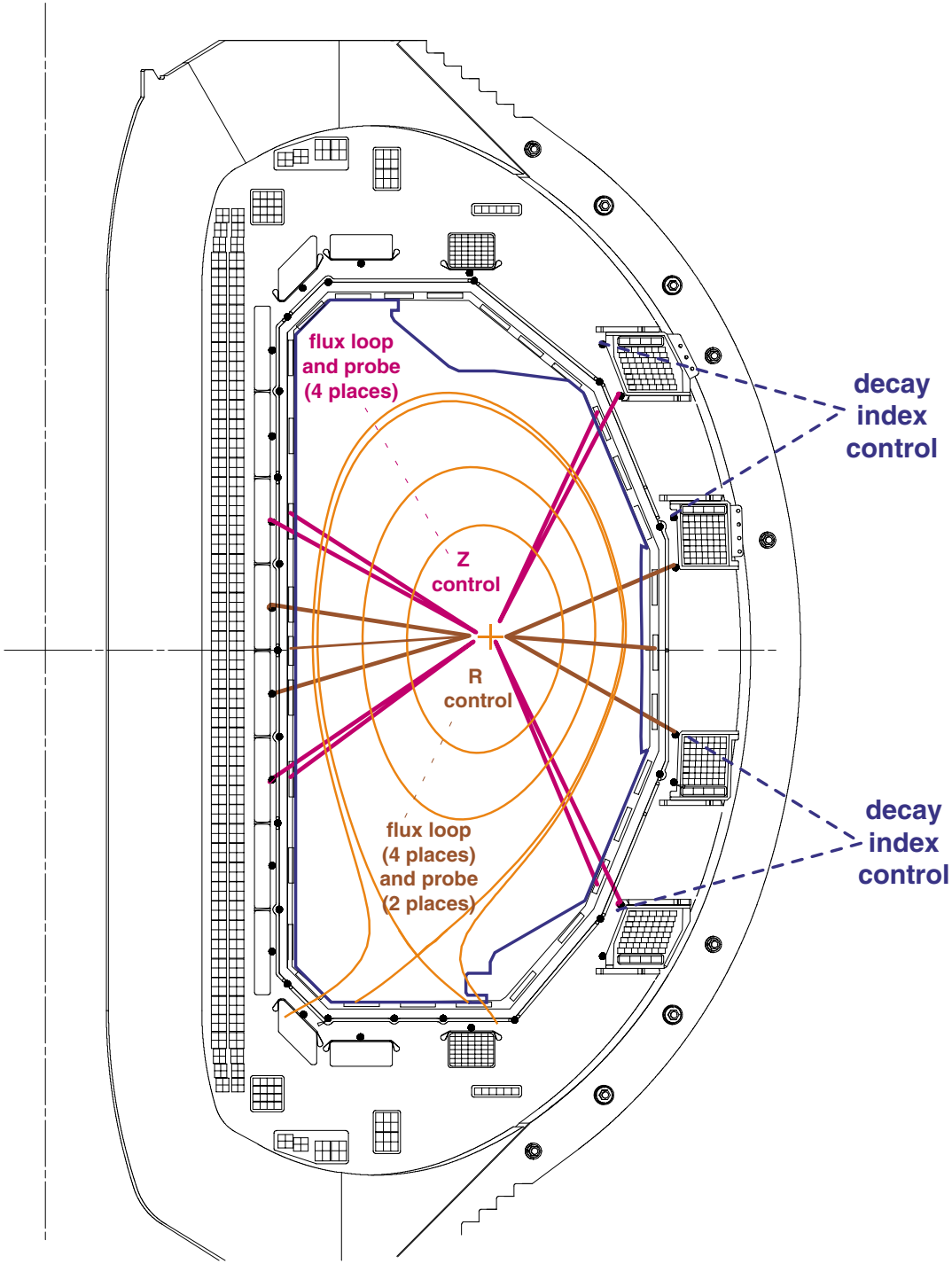

Vessel Wall
40 ms
li383s
524/525 ms
12 optimizations
0.4
0.2
0.0
-0.2
-0.4
Z (m)
1.0
1.2
1.4
1.6
1.8
R (m)

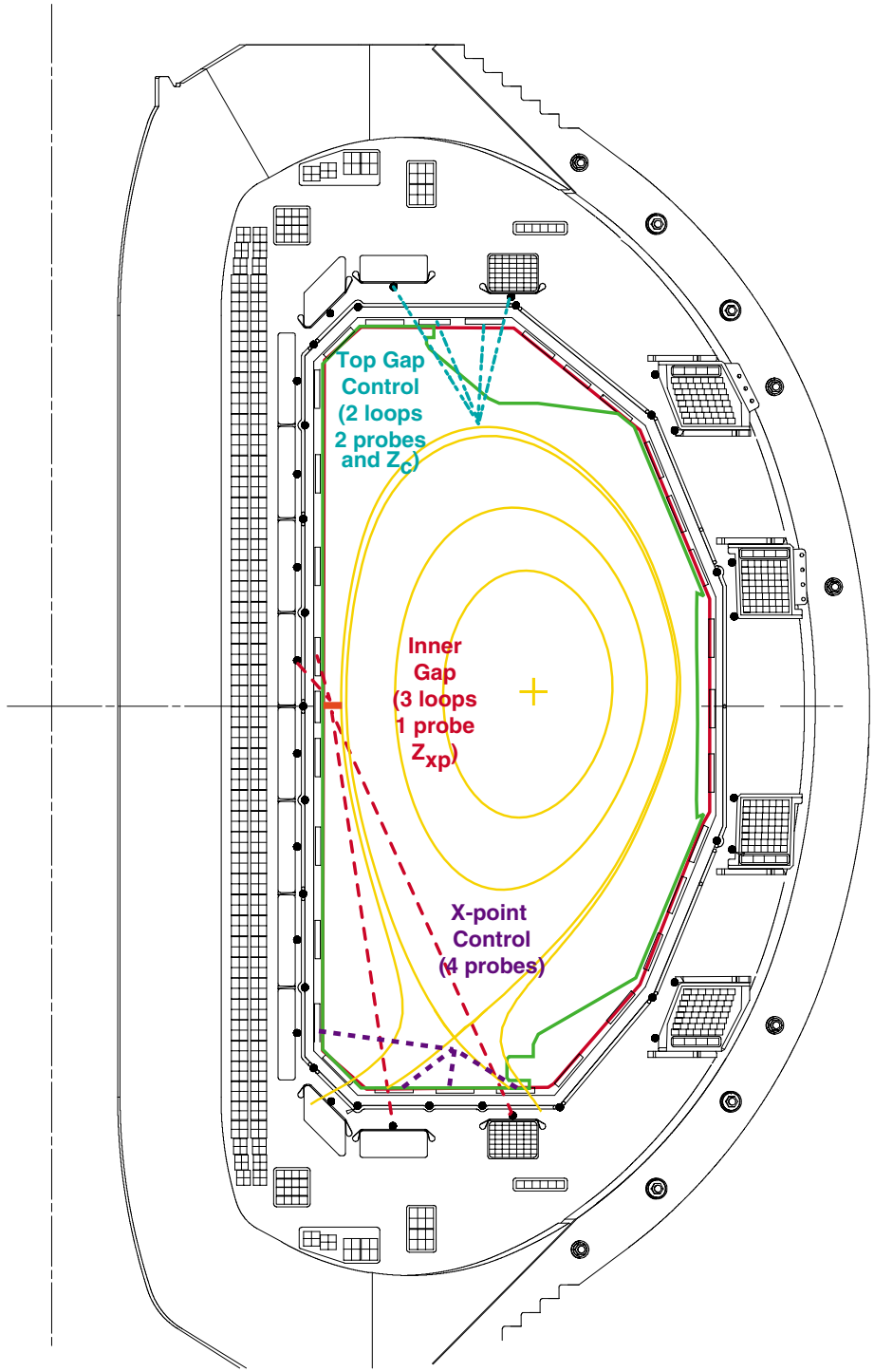
The changes in modular coil currents are modest.

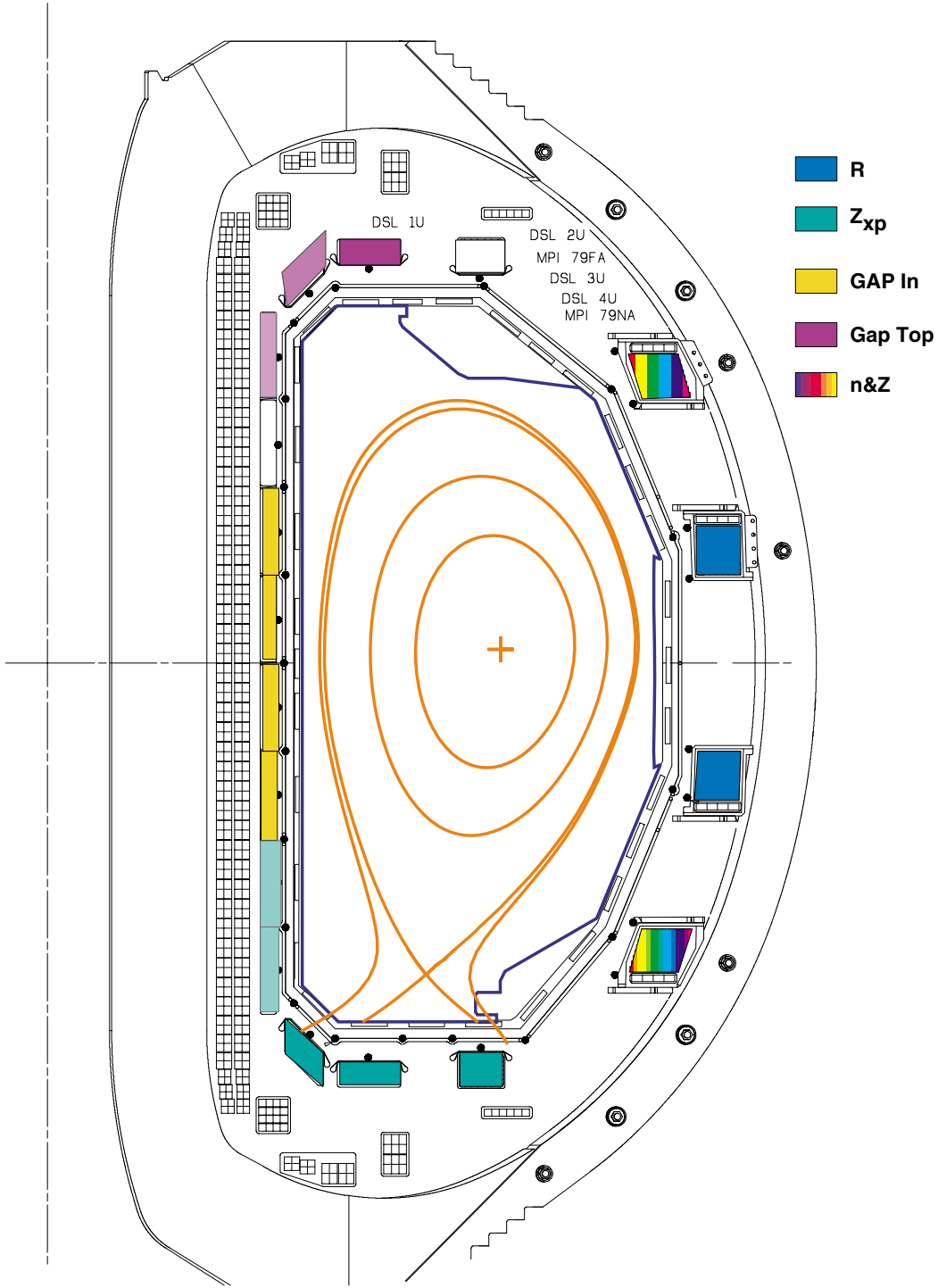


This is an excellent result. While it does point to a need for active control of the helical field, all that is needed to maintain a stable, quasi-axisymmetric discharge is to control the plasma boundary shape.
To give some perspective on what is involved I want to describe a control system for a D3D single null plasma. I want to go through this and then discuss what more will be needed for the stellarator. All signals are normalized to plasma current.

There are 6 explicit controls. R and Z of the current centroid, Z_{xp} the height of the x-point, the inner gap, the top gap, and the quadrupole field. These are not an orthogonal set.
The signals are combined by a simple regression analysis; eg
$I_{div}=a_1 \bullet B_{4b}+a_2 \bullet B_{8b}+a_3 \bullet B_{89b}+a_4 \bullet B_{9b}$
and the a_i's are determined by fitting to many equilibria. Here, the B's are the probe signals in volts. It is common (and necessary) to begin with imagined equilibria and simulated B's and ψ's.


flux loop
and probe
(4 places)
Z
control
R
control
flux loop
(4 places)
and probe
(2 places)
decay
index
control
decay
index
control


Top Gap
Control
(2 loops
2 probes
and Z_c)
Inner
Gap
(3 loops
1 probe
Z_{xp})
X-point
Control
(4 probes)


DSL 1U
DSL 2U
MPI 79FA
DSL 3U
DSL 4U
MPI 79NA
R
Z_{xp}
GAP In
Gap Top
n&Z

The control system described here was used on DIII-D for many years and works pretty well. In recent times it has been replaced by a real-time equilibrium solvewhich updates the gap values every 15 ms.

The most notable deficiencies are these:
The a_i's are really dependent on plasma parameters.
The control signals are sent to particular coils - other coils in the group are sent control signals which are flux ratios to the primary coils. Again this is less than perfect when plasma parameters change.

The first could be fixed by using neural networks to make $a_i=a_i(\beta_p,l_i)$ or a real time equilibrium solution as is now done on DIII-D. The second has no ready solution.
This is likely to be a bigger issue in a stellarator.

The Stellarator

It will be necessary to control helical field (HF) to preserve shape in more than one plane. The simultaneous requirements of stability and quasi-symmetry suggest 2 planes.

If the poloidal coils (PF) are affecting the plasma shape, possible more planes will need to be invoked to achieve adequate control.

It will be necessary to have enough redundancy to verify periodicity.

Next Steps

There are 3 broad categories to be addressed: equilibrium, choice of diagnostics, and finding control algorithms,

Equilibrium:

It is reasonable to start with the tokamak approach.

To begin, it is necessary to generate a large set of free-boundary equilibria with the final coil set. This can be done now, except that it is necessary to calculate the response of a large set of magnetic diagnostics.

In EFIT this is handled with a Greens function formalism where the unit current responses for the individual coils and the computational grid elements ("toroidal plasma current") are stored as tables. For a stellarator this would need to be generalized to include poloidal plasma current. VMEC can do the equilibrium.

A proviso, we should probably try some cases that break stellarator symmetry - so that we can at least recognize it. I don't think this is tested in VMEC.

Magnetic Diagnostics:

The whole notion of response functions for diagnostics needs to be developed so the Green's tables can be built based on generic diagnostics characterized by type, location and size can be built. Then the solution needs to be put in the R,ϕ,Z plane for the magnetic response calculations.

The choice of magnetic diagnostics is not obvious. For the tokamak, I think flux loops are better than localized measurements. So, I think saddle coils will be attractive for the stellarator. A planar poloidal flux loop will see some combination of poloidal and toroidal flux - it is unclear to me whether this is good or bad.

Control Algorithms:

The first requirement for control is the generation of simulated signals for the postulated magnetic diagnostic set. Than signals and boundary shapes are generated for a plausible range of discharge. It is not yet known what set of magnetics will yield the boundary in adequate detail. To study this control problem one needs a code to generate diagnostic responses to equilibria. We hope to begin on this code in the near future.

We would hope to use flux projection ($\psi_{surf}=\psi_{coil} + l\cdot\nabla\psi$) as a first approach. It is not clear our intuition will allow flux projection as we need to maintain shape in more than one toroidal plane and the coil currents are not related to shape in a simple way.
If this proves too difficult then principal component analysis will relate the boundary shape to the sensors. (If this fails, the postulated magnetic diagnostic set was not a good choice.) A refinement is to generate algorithms with coefficients which are functions of β, I_P, etc. A next level of sophistication would be to use neural networks, e.g., $\mathbf{y}_{shape}=\mathbf{U}\cdot tanh(\mathbf{V}\cdot\mathbf{x}_{magnetic})$ as compared to a linear model this allows the dominant differences among the elements of $\mathbf{x}_{magnetic}$ to change as the values $\mathbf{x}_{magnetic}$ change.

Control Algorithms (cont'd):

Additionally there is a gain matrix relating the errors to the voltages applied to the coils, $\mathbf{E}=\mathbf{G}_{pid}\ (\mathbf{y}_{shape}-\mathbf{y}_{demand})$. In DIII-D this matrix is sparse. Making it full has been a low-level effort for several years - it is still sparse. Although the actual algorithms are generated by regression analysis, one can usually guess which sensors to use for a particular function in the tokamak and which coils to tie to the actual control function.
Thus the sparseness of the gain matrix is only an inconvenience.

In a stellarator with modular coils, the separation into particular coils being domininant is not likely. For example, if one wanted some R_0, quasi-symmetry, and stability it is unlikely that some particular subset of coils will provide each and not have a significant affect on the other features, i.e., $\mathbf{G}_{pid}$ will not be sparse.
This implies that a more formal approach such as PCA will be needed to find suitable control algorithms. Recall that I began with the assertion that plasma shape control is all that is needed and the solutions were found with the coil set (0907). This means that suitable control is possible.