Finding Promising Alternatives in Magnetic Fusion Research (II)

T. K. Fowler

Princeton, June 12, 2007

(based on a similar talk at the GA 50 Year Celebration, May 16, 2007)

Today I want to discuss what I believe is the basis for embarking on an advanced toroidal confinement effort beyond improvements in tokamaks and stellarators now under way. I believe it is imperative to do so.

My concern is that fusion research is missing the long-foreseen Global Warming window of opportunity. I submit that this has happened in part because, due to its large size and cost, it has taken 20 years to launch ITER, giving a time horizon for commercial fusion energy beyond the attention of policy makers.

It is not too late to change this. But only if there exists a faster route of development than that now envisaged for the tokamak.

How did the present situation arise, and how could we change it? While scientists do not control their fate, their scientific decisions do influence it. I suspect we are where we are because of early fears derived from current-driven instability in pinch experiments. Yet, with RFP and now spheromaks, a quiet revolution has occurred that should make us question whether the $q > 1$ regime of the tokamak is the only viable path.

The price for $q > 1$ is a strong toroidal field, and it is this that has dictated the pace and scale of tokamak progress. Added to this has been the alluring simplicity of inductive current drive, which also drives up the size of ITER. Advanced tokamaks seek to dispense with inductive drive, by auxiliary current drive. Here I suggest that profile control by auxiliary current drive can also be used to stabilize the $q < 1$ regime, as has already been done in MST, and a similar success in stabilizing spheromaks would allow us to dispense with toroidal coils altogether. If successful, this could lead to very small devices and a faster development path.

Fortunately, we do not need to speculate about this, for the need or lack of need for toroidal coils is a matter of tearing modes, and the non-linear development of tearing modes can now be calculated by resistive MHD codes, such as NIMROD already calibrated to MST for RFP’s and SSPX for spheromaks.

What I suggest, then, is a concerted effort to apply computer simulation to assess the $q < 1$ versus $q > 1$ regimes, with profile control as the centerpiece. The success of profile control on tokamaks and RFP’s gives us hope, as does evidence for stable states in spheromaks. And computational success could be followed by relatively small experiments to confirm the results, in part reusing current drive equipment already available.

FINDING PROMISING ALTERNATIVES IN MAGNETIC FUSION RESEARCH (II)

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1. Everyone should love ITER.

2. But fusion may be missing the "global warming" window of opportunity.

3. Can ITER alone put fusion back in the race?

4. Improvements to the tokamak are already under way: Is there anything else out there?

5. With profile control to stabilize the current, the $q < 1$ regime may yield smaller toroidal devices, hence a faster development path parallel to ITER.

6. Over-promising should not be repeated -- but we ignore the energy goal at our peril.

HOW SMALLER?

Consider the spheromak...

1. No toroidal coil; gun startup (no ohmic coil):

SSPX -- $a = 0.25$m, $T = 0.5$ KeV.

2. Smaller plasma volume means less beam power to build up by NBI current drive:

$dI/dt = C_1(P/nV) - I/\tau_\Omega$; $\tau_\Omega = 5a^2T^{3/2}$

3. If sufficiently stable, T rises as I grows:

$dT/dt = C_2(P/nV) - T/\tau_E$

Buildup if $P > 100\, n_{20}^{1.3}\, a^{1.7}$ (MW) -- L mode

SSPX size: 6 MW$\rightarrow$ I = 10MA, T > 5 KeV

4. Buildup and sustainment probably requires profile control (PC) to stabilize tearing modes:

a. MST: PC demonstrated and simulated

b. Stable states demonstrated on SSPX

τ_E (core)$\approx$ 5 - 10 ms, as good as L mode

HOW FASTER?

1. Toroidal coil was introduced to stabilize tearing modes: do we need it?

2. NIMROD as "Numerical Torus"

a. Resistive MHD code NIMROD already calibrated to experiments (MST, SSPX)

b. Code can be used to compare $q > 1$ versus $q < 1$, optimize toroidal design

c. Can be used to aid design of experiments to minimize steps to demonstrate ignition

d. This would suggest additional calibration experiments on existing or intermediate facilities.

3. Reuse of NBI and RF at major tokamak laboratories could expedite construction, reduce costs.

WHY NOW?

1. ITER construction offers a decade of opportunity ...

a. Restoring the "dual path" of the 1970's-80's strengthens the program:

"Forging ahead with what we know while looking for something better."

b. Improved tokamaks and stellarators already follow the dual path tradition.

2. Innovation has been an important U.S. contribution to magnetic fusion research:

a. Profile control has become a major tool of innovation in magnetic fusion research.

b. Profile control could be the cornerstone of a re-invigorated dual path effort.

c. With profile control, the $q < 1$ regime (spheromak, RFP) could now be incorporated in an advanced toroidal confinement program, with heavy emphasis on resistive MHD computer simulation.

WHAT PAYOFF?

1. Improved prospects for fusion reactors, faster development path, would enhance the value of ITER.

2. New solutions to problems already revealed by ITER design would speed up development:

Example: Mirror-like divertor in spheromaks (which could be explored via the "Numerical Torus").

3. The $q < 1$ regime could reduce the cost of electricity produced by fusion. Tradeoffs:

a. Reduced or zero toroidal field can increase allowed B, reduce reactor size (ignition $a \propto 1/B$).

b. Smaller size trades against possibly higher injection power: for $Q_E = \eta_{TH} (P_{FUS}/P)$ ...

$\$/\mathrm{Kwe} \approx (1 - 1/Q_E)^{-1}\{C_3(a/P_{WALL}) + (C_4/Q_E)\}$

Spheromak: $Q_E \geq 4$, recirc. $\leq 33\%$ ("sagging λ", alpha channeling may increase Q_E)

Papers available at: fowler@nuc.Berkeley.edu

APPENDIX

Thoughts on a Development Plan for Stabilized Sheromaks

1. NIMROD study of auxiliary current drive and profile control:

- **-- Stability may require flat $\lambda = \mu_o j/B$ out to a critical radius r_C**
- **-- Concentrate current drive near r_C?**

2. Possible sequence of experiments:

	a	P	E	I	T	t
SSPX	0.25	1.5	25	0.5	0.5	0.01
POP	0.25	6	80	10	>5	2
Ignition	0.75	40	80	50	10	100*

*or high current gun

Assumptions:

1. Polodial coils needed at POP stage, test on upgrade of SSPX

2. L mode (like SSPX) during buildup

3. ITER-like scaling of $n\tau$ in ignited state

Figure 4
Poincaré Plot showing 3D flux surfaces