Research of Particular Importance to Fusion for NSTX-U Coupled with Theory

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Issues that block the path to fusion power plants define the most important research areas.

1. Tokamak shutdown without disruptions.
2. Experimental simulation of power plant conditions.
3. Re-formation of magnetic surfaces after a disruption.
4. Divertor freedom using 3D fields.
5. T_i behavior with electron heating.

1. Tokamak Shutdown without Disruptions

Remarkably little is written on the shutdown of ITER. The key paper is [1]: *Controlled shutdown of ITER discharges* by de Vries et al, Nucl. Fusion **58**, 026019 (2018).

- The current profile is difficult control [2] but takes a long time to evolve ~ 15 minutes in ITER. When it evolves in a dangerous way, plasma termination is the obvious strategy.

- Termination requires the removal of the poloidal flux in the plasma chamber, $\sim$ 60 V·s in ITER

- Reference [1] claims:

  "In ITER, a fully controlled current ramp-down from $I_p = 15$ MA to below $I_p = 1$ MA can be achieved in 60 s."

  No details are given. Reference [1] discusses axisymmetric simulations using Corsica and DINA in which the shutdown time is 210 s.

Some Issues with a Fast Shutdown [2]

a. Removal of poloidal flux, $\sim$ 60 V·sec in ITER chamber.

- Number of seconds determined by the loop voltage:

  (i) around the magnetic axis $V_\ell = 2\pi R_a \eta j_a$, takes about 15 min at 10 keV.

  (ii) the loop voltage at the wall, requires pulling poloidal flux into transformer.

- Pulling flux into the transformer drives a negative current near edge. *Can produce instabilities, but these are not included in a 2D Corsica or DINA simulation.*

- Internal inductance ℓ_i (central peakedness of the current) cannot be allowed to become too large. *A large ℓ_i makes axisymmetric position control difficult and is associated with disruptions.*

Additional Issues with a Fast Shutdown

b. Plasma particles must be removed from the chamber as I_p drops to satisfy Greenwald limit, $n < n_G \propto I_p/\pi a^2$.

c. Transition from an H to an L mode is sudden and can complicate position control. *In a burning plasma, DT power drops as* $(nT)^2$ *and in ITER cannot be replaced by auxiliary heating.*

d. Interaction of a plasma that pre-disruption had strong axisymmetric shaping can be fast or unstable [3–5].

Subtle due to complicated wall (tiles, blanket modules, etc) [6].

e. Location of power flow to wall must be controlled.

Careful control not required when $I_p \lesssim 3$ MA.

2. Experimental Simulation of Power Plant Conditions

- Time for $\vec{B}$ to penetrate chamber walls is more than an order of magnitude longer in ITER and power plants than in existing machines.
- Loop voltage, vertical / shaping fields, and heating power can be altered in existing machines to control plasma displacement, plasma current profile, and H to L transition in a way that will not be possible in burning plasmas.
- Existing machines could simulate the effects of (a) highly conducting walls by adjusting coil currents and (b) the loss of DT power by adjusting the external power input.

3. Re-formation of surfaces after a disruption

- Disruptions break the magnetic surfaces, which flattens the current $j_{||}/B = const.$, and quenches the thermal energy.

- A non-axisymmetric ideal evolution tends to make spatial separation between magnetic surfaces vary exponentially [7, 8]. Where surfaces are close η/μ_0 can interdiffuse field lines from different surfaces. Surface breakup takes an ideal evolution time multiplied by the logarithm of the magnetic Reynolds number, $R_m \equiv \mu_0 a v/\eta \sim 10^7$ for an ITER disruption.

- Disruptions are so fast $\vec{B}\cdot\hat{n}$ remains axisymmetric on walls. Without a drive for kinks, $\vec{\nabla}(j_{||}/B) = 0$ and $\vec{\nabla}p = 0$, an axisymmetric equilibrium is favored.

- Magnetic surfaces tend to re-form but physics and math very different from breakup. Ideal flows are unimportant. Timescale presumably proportional to η.

Effect of Surface Re-formation on Runaways [9–12]

Runaway requires confinement and $T_e \lesssim 500$ eV.

Runaways increase by a factor of ten (a hundred with impurities) *per MA drop in plasma current.*

- **In a non-nuclear device,** hot-tail electrons are the seed.

Runaway must occur in ≈ 20 ms for a hot-tail avalanche.

- **In a nuclear device,** tritium decay and gamma rays from the walls give an adequate steady-state seed.

- Runaway danger very different between outside-in versus inside-out re-formation of magnetic surfaces.

- Inside-out reformation places runaways on magnetic surfaces—localized deposition on the walls is difficult.

Energy in runaways is $\lesssim 10\%$ of original plasma thermal enengy.

- Outside-in allows extremely localized deposition.

Runaways in a chaotic core are confined by an annulus of magnetic surfaces.

Outside-In Re-formation of Magnetic Surfaces

- Favored when resistivity is high near the plasma edge. *Mitigated if currents induced in walls produce strong 3D fields.*
- Runaways fill a chaotic core, confined by an annulus.
- Annulus can be punctured *by being pushed into the wall, a plasma kink striking the wall, or by a resistive instability.*
- The annulus breaks by a pair of magnetic flux tubes—one in and one out—carrying increasing flux extending between the reservoir and the wall. Called a turnstile. Runaways move only one way along $\vec{B}$, so only one of the flux tubes is important.
- The quicker the turnstile opens compared to the runaway transit time, the broader the spreading on the wall [13].
- Two ways to avoid localization: *(1) fast breaking of annulus and (2) inside-out reformation of surfaces.*

Experiments on Localization of Runaway Losses

- Damage from extreme localization of runaway losses is seen in many experiments, but not all.
- In highly unstable JET (PRL **126**, 175001 (2021)) and DIII-D (NF **61**, 116058 (2021)) plasmas, runaway spreading was sufficient to avoid problems.
- The fusion relevance of tokamaks requires the extreme damage of runaways be avoided.
- This defines the importance of determining why runaway loss is sometimes concentrated and sometimes not.
- Outside-in versus inside-out surface re-formation after disruptions is a critical issue.

4. Divertor Freedom using 3D Fields [14]

- As shown by J-K Park on KSTAR [15], the near separatrix region in a tokamak can be made chaotic while preserving the quasi-axisymmetry in the plasma core.
- Critical divertor issues:

Ratio of pump opening area to plasma surface area, f_d

Ratio of plasma density in divertor to main plasma n_p.

Divertor temperature $T_d \sim 100$ eV from $C_s(T_d) \approx \frac{n_p}{n_d}\frac{L_c}{\tau_p}$.

L_c connection length from stagnation point to divertor.

τ_p particle confinement time of main plasma.

Pump needs neutrals $T_n \lesssim 1$ eV and neutral density $\sim 10 n_d$

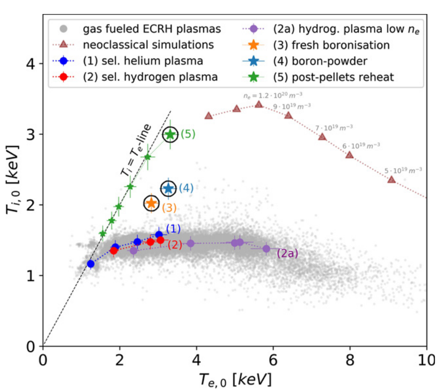
$$\text{Compression ratio, } \frac{\text{neutral density near the pump}}{\text{average neutral density in chamber}} \sim \frac{1}{f_d} \tag{1}$$

- Non-axisymmetric divertors have far more freedom than axisymmetric (thickness, connection length, etc.), which allows stable detachment. Computational studies tested by experiments are required for tokamaks and stellarators. Hamiltonian methods allow fast determinations of what is possible and what is controllable [14].

5. Ion Temperature with Electron Heating [16]

Ion-electron equilibration: $\frac{\frac{3}{2}T_i}{\tau_{Ei}} = \frac{T_e - T_i}{\tau_{eq}}$ or $\boldsymbol{\frac{T_i}{T_e} = \frac{\tau_{Ei}}{\tau_{Ei} + \frac{3}{2}\tau_{eq}}}$.

- When ion heating is by equilibration with directly heated electrons (*as in DT fusion*), ion temperature has a maximum where ion energy confinement time $\boldsymbol{\tau_{Ei} \approx \tau_{eq}(T_e)}$.
- When $\boldsymbol{T_e \gtrsim 30}$ keV, then $\boldsymbol{nT_i\tau_{eq} > n\tau_E T_i}$ for fusion.
- Neutral beam heating is deceptive on confinement needed for a DT burn. Need a clamp on $\boldsymbol{T_e}$.



W7-X results, Beurskens, Bozhenkov, Ford, et al., Nucl. Fusion **61**, 116072 (2021).

Critical Path to Fusion Defines Research of Highest Importance

Tokamak-based fusion requires retirement of the risks that derive from

1. The lack of robust axisymmetric position definition.
2. Limitations of actuators to control current and pressure profiles.
3. Sensitivity to plasma collapse (disruptions) of these profiles.
4. Unreliability of predictions on the required timescale for plasma shutdown.

The abstract of the paper *Plasma steering to avoid disruptions in ITER and tokamak power plants* [2] summarized these issues:

"*Steering tokamak plasmas is commonly viewed as a way to avoid disruptions and runaway electrons. Plasma steering sounds as safe as driving to work but will be shown to more closely resemble driving at high speed through a dense fog on an icy road.*"

NSTX-U Research Critical to the Feasibility of Tokamak-Based Fusion

Some topics that could be addressed

1. Tokamak shutdown without disruptions.
2. Experimental simulation of power plant conditions.
3. Re-formation of magnetic surfaces after a disruption.
4. Divertor freedom using 3D fields.
5. T_i behavior with electron heating.

These also provide a focus for theory and computations.

References

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