# Steady State Physics

## Technical Subgroup of the Magnetic Fusion Concepts Working Group

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## Report of Magnetic Concepts / Steady State Subgroup

For electric-power generating stations based on magnetic fusion energy (MFE), it would be highly desirable to have a fusion core device that could operate continuously. Compared to a pulsed device such as the conventional tokamak, a steady-state core would simplify plant operation by providing a smooth and controllable fusion power output. It would avoid the need for energy storage in the shield to smooth the plant output and would promote reliability by minimizing the thermal cycling of plasma-facing components. Whether a pulsed or continuous system will prove to be the more attractive remains to be determined, but the clear benefits of continuous operation motivate development of steady-state magnetic confinement systems. In any case, the frequency of unscheduled terminations of the fusion core plasma must be reduced to an absolute minimum, and their effects mitigated, to avoid long unscheduled outages. The physics issues for maintaining continuous operation of a toroidal plasma and for avoiding discharge-terminating disruptions are closely related.

Developing a steady-state MFE system involves closely coupled issues of physics and technology. Superconducting magnet technology can facilitate physics studies of plasmas lasting minutes to hours. Wave-launching structures which continuously couple control power to the plasma and exhaust structures which continuously remove energy and particles from a high-power-density plasma must perform their functions under stringent conditions. While we focus mainly on the physics issues in this section, the related technology issues are also critical in the achievement of steady state MFE.

The physics goal for steady-state MFE is to understand the physics of a continuously-sustainable high-performance fusion plasma. Here, “continuously sustainable” means that the pulse length is not limited by plasma or plasma-wall interaction phenomena; it can be sustained well beyond the time scales for such phenomena, up to the engineering limits of the system. In addition, it means the plasma is reliable: the frequency of unscheduled terminations due to plasma phenomena is reduced to a level compatible with economical operation of the plant. By “high-performance” we mean that plasma has high fusion gain ($Q_{eng}$, the ratio of the fusion output power to the power required to operate the power plant), so as to minimize the amount of plant output power which must be recirculated back to the plant. To achieve this requires both good plasma energy confinement and a primary reliance on plasma sustainment methods that are efficient (e.g., self-generated plasma currents, superconducting coils) in preference to inefficient ones (e.g., non-inductive current drive).

### Key Issues for Steady-State Magnetic Fusion

Substantial progress toward the realization of a steady-state MFE system has been made through research on several different types of magnetic confinement experiments. In both the TRIAM tokamak and the ATF stellarator, discharges longer than 1 hour have been achieved.

These discharges were terminated by operator choice but, because of their low beta (<1%) and low power density, did not challenge physics or technology limits. In high-power, but still relatively low-beta (<2%), operation, discharges lasting up to ~100 s are realized before being terminated by impurity influx from the walls due to first-wall overheating or by uncontrolled density rise. The best example of this is the Tore Supra tokamak which, along with the LHD stellarator, focusses on the physics and technology of steady-state power and particle handling. Higher-beta (>3%) magnetic plasma configurations, of high potential interest for reactor economics, have been formed by transient techniques in tokamaks (e.g., DIII-D) and spherical tori (e.g., START), but so far have been maintained for no more than two seconds. The reason is that these devices do not yet have sufficient control capability to prevent the plasma from evolving from a stable to an unstable configuration, usually leading to disruptive termination.

We see that progress in steady-state MFE is currently limited by deficiencies in two areas: plasma control and power and particle handling. The two key steady-state MFE physics research issues for the next 5-10 years are, therefore, understanding how to control the plasma to achieve and sustain a high-performance plasma configuration and how to handle the power and particle exhaust in a manner compatible with a high-performance core plasma. These are serious issues which do not have easy solutions. A range of complementary approaches, making use of a variety of magnetic configurations, must be tested and understood to provide a high probability of successful resolution.

**Steady-State Plasma Control Approaches**

Control schemes for steady-state magnetically confined plasmas fall into three categories: 1) Equilibrium poloidal field and shape control, 2) Profile control, and 3) Magnetohydrodynamic (MHD) mode control. Within each category there are multiple candidates. Developing the needed understanding is best accomplished using the spectrum of magnetic configurations provided by the magnetic concepts portfolio.

Equilibrium poloidal field and shape control. Toroidal magnetic systems use various combinations of external coils and internal plasma currents to provide the equilibrium poloidal magnetic field and the desired plasma cross section shape. In stellarators, helical fields produced by external coils maintain toroidal magnetic configurations with nested surfaces and rotational transform and can shape the plasma to achieve desired transport and stability properties. The coil shapes may be complex, but since they are superconducting the configuration is maintained with very little recycled power. The advanced tokamak and spherical torus rely, to the maximum extent possible, on the bootstrap current to maintain the configuration. The bootstrap current itself is driven by plasma pressure gradients, making it quite efficient, but typically some fraction of the total current must be driven externally to supplement the bootstrap current. Advanced tokamaks and spherical tori are sustained by combining the bootsrap current with external current drive from radiofrequency waves or neutral beams. However the efficiency of these current drive methods is low, so their use must be minimized to minimize recycled power requirements. Low-aspect-ratio (“compact”) stellarators use both helical fields and the bootstrap current to combine the best features of advanced tokamaks and stellarators.

Helicity injection is used to start up and possibly maintain spherical torus and spheromak plasmas by injecting linked poloidal and toroidal magnetic flux at the edge of the plasma using electrodes and relying on magnetic reconnection for flux penetration. Transport associated with the penetration processes and electrode erosion are concerns with this approach but its simplicity is an attraction. Induction is an established technique for start-up in tokamaks and provides experimental flexibility in spherical tori and some stellarators. Normally it cannot sustain a configuration because it is limited by the amount of flux that can be stored in the primary winding set. However, oscillating-field current drive (OFCD) has been proposed to overcome this limitation using phased alternating poloidal and toroidal fluxes, thereby injecting helicity, and is envisioned as a means of sustaining the reversed-field pinch.

<u>Profile control</u>. Beyond the basic requirements for sustainment of a magnetic configuration, many magnetic concepts are envisioned to require some form of profile control to maintain a configuration with optimum stability and transport properties. The advanced tokamak and spherical torus require current drive near the edge of the plasma to supplement the bootstrap current. Current profile control will be used in the reversed-field pinch to reduce transport. Current profile control is not envisioned to be needed for stellarator reactors but can provide valuable flexibility for planned stellarator physics experiments.

An emerging area of toroidal physics research is that of pressure, flow, and transport profile control. In a reactor the alpha self-heating profile and the pressure profile will be tightly linked to each other. However, it may possible to optimize the pressure profile and enhance confinement through control of transport barriers. Such approaches would benefit the entire family of toroidal configurations. Transient techniques, in which the current rise and heating turn-on are carefully staged during start-up, have been successful in tokamak experiments, but sustainable techniques need to be developed. One possible strategy is to use radio-frequency waves to tailor the pattern of plasma flows in such a way as to stabilize the transport-inducing turbulence. Ideas for accomplishing this need to be carefully examined, and the promising ones tested.

<u>MHD mode control</u>. High-performance magnetic plasma configurations can be subject to instabilities that, if not controlled, lead to unscheduled termination of the discharge. Examples are vertical instabilities, external kink modes, resistive wall modes, and neoclassical tearing modes. Passive and active stabilization approaches are used to control these pulse-length-limiting instabilities. Many tokamak and spherical torus designs include conducting walls close to the plasma to reduce the growth rate of the vertical and external kink instabilities by passive

stabilization. Reversed-field pinches and spheromaks also have a close-fitting stabilizing shell. While advantageous for plasma stability and performance, however, such structures in the high-radiation environment of a fusion plasma represent an engineering complication with adverse implications for cost and availability. A good understanding of the performance benefits is needed to assess the tradeoff.

Active feedback control of the vertical instability is well established for vertically elongated tokamaks and is also applicable to spherical tori. Steady-state tokamak and spherical torus configurations are also subject to resistive wall modes and neoclassical tearing modes; active control of the these modes is a relatively new research area. Active feedback can enhance performance, but at the cost of additional complexity (e.g., coils, external power systems and amplifiers) and recycled power. Stellarator configurations can be designed to be stable to these modes and so avoid passive and active stabilization systems. Tokamaks, spherical tori, reversed-field pinches, and spheromaks will likely include them.

**Steady-State Power and Particle Handling Approaches**

Power and particle exhaust depends on: 1) the geometry of the magnetic configuration at the plasma edge and its interface with target structures and pumps, and 2) the operating mode and materials. Understanding of power and particle handling for magnetic fusion can best be acquired by studying a range of magnetic configurations. Throughout this section we identify some power and particle handling issues important for steady-state operation. The topic is discussed in greater depth in the Boundary Control section of the Magnetic Concepts Working Group report.

<u>Geometry</u>. The edge magnetic geometry is strongly dependent on the magnetic configuration. In recent years the axisymmetric poloidal divertor has been studied extensively in tokamaks; this geometry is also an option for the spherical torus. Stellarator divertor configurations include helical and island divertors. The expanded-boundary limiter (or “natural” divertor) is an option for the spherical torus and limiters remain an interesting alternative for tokamaks.

<u>Operating Mode and Materials</u>. Effectiveness in reducing peak heat fluxes onto target surfaces, in exhausting particles, and in maintaining compatibility with the plasma core depends on materials choices and careful control of edge conditions. Plasma impurities can be used to radiate power in the divertor region (“radiative divertor”) or the plasma periphery (“radiating mantle”) in order to distribute heat fluxes over larger surface areas. Carbon, because of its low atomic number, has been a popular choice as a target material in experiments in recent years, but concerns about tritium retention have prompted a growing interest in other materials, both solid (tungsten, beryllium) and liquid (lithium, flibe). External pumps are ultimately needed for steady-state particle exhaust, but wall pumping can also play an important role, depending on the

materials used. Internal pumping structures closely coupled to the divertor are cost-effective as research tools to study particle control issues. Sustainable fueling techniques include gas injection and pellet injection and both have been successfully demonstrated on tokamaks and other concepts.

**Opportunities to Resolve Steady-State Magnetic Fusion Issues**

Steady state issues have been considered and are being investigated for all the various configurations in the magnetic concepts portfolio.

Stellarator. Large experiments based on high-aspect-ratio, currentless stellarator concepts are being pursued in Japan and Europe. In the U.S., the focus is on hybrid configurations that combine the bootstrap current with external helical fields to obtain stellarator benefits (low recycled power, freedom from disruptions) at tokamak-like performance and aspect ratios. Complementary approaches are being studied: the advanced-tokamak-like quasi-axisymmetric (QA) approach with substantial bootstrap current and the advanced-stellarator-like quasi-omnigeneous (QO) approach with minimal bootstrap current. Among the key issues for planned U.S. experiments are testing the use of external helical fields for disruption suppression in such configurations, beta limits, the reduction of neoclassical and energetic-orbit losses through magnetic configuration design optimization, and the reduction of anomalous losses through techniques such as flow-shear control. For stellarators generally, steady-state power and particle handling is now a key issue with the advent of large powerful machines capable of multi-megawatt very-long-pulse (30 minute) operation. Research on LHD, W7-AS, and (later) W7-X will aim to determine which approaches are most effective. Island and helical divertor configurations will be tested.

The new LHD facility in Japan and the W7-AS in Germany provide near-term opportunities for the U.S. to collaborate on high-power long-pulse divertor experiments aimed at resolving steady-state power and particle handling issues for stellarators. Configuration design work for lower aspect-ratio QA and QO experiments is developing the knowledge base for these innovative approaches. The proposed experiments based on these approaches (NCSX and QOS respectively) will provide the opportunity to investigate stability and transport physics issues and the compatibility with divertor solutions being developed internationally.

Tokamak. The aim of advanced tokamak research is to integrate high core performance (enhanced confinement, high beta) and extend the pulse length of tokamak operation. This is now the main research thrust for the U.S. tokamaks (DIII-D and Alcator C-Mod) and a major topic for many foreign tokamak programs.

A key issue at present is profile control, since current profile relaxation and pressure profile evolution are limiting the pulse length of high-performance discharges. For advanced tokamaks, the plasma control challenge is to maintain an optimum configuration in the presence of large bootstrap currents. Because the bootstrap current profile depends sensitively on the configuration, the problem is a highly non-linear one. Complementary current profile control approaches will be tested in the U.S.: electron cyclotron on DIII-D and lower hybrid on Alcator C-Mod. Pressure profile control that is both precise and sustainable is in a more primitive state. Ion Bernstein waves (IBW) are believed to have good potential for local turbulence and transport control; so far several exploratory experiments have been performed (PBX-M, DIII-D, TFTR) with mixed results. The use of mode-converted IBW will be tested on Alcator C-Mod; direct IBW launch mechanisms have been studied on the FTU tokamak in Italy; and the folded waveguide launch technology of interest for IBW will be tested on LHD. The issue of pressure profile control via local transport control is highlighted as one needing greater emphasis and focus. The existing data need to be fully understood to guide future research, a range of ideas needs to be examined, scientific and technological innovations must be encouraged, and promising approaches must be tested.

Feedback control of neoclassical tearing modes (through localized heating or control of the safety factor profile) and external kink / resistive wall modes (through conducting structures and either plasma rotation or external coil feedback control) are key issues for the steady state tokamak with high performance. These are being investigated experimentally on DIII-D. Disruption avoidance/mitigation has been demonstrated on several tokamaks by characterizing plasma limits and operating well away from them. However, feedback control to avoid disruptions is an immature area and requires research in identifying both precursors and effective responses.

In the area of power and particle handling, pumped divertor operation in tokamaks has demonstrated a level of global particle control sufficient for reactors. However, this has not been demonstrated at the level of plasma performance that would lead to significant simultaneous heating in the divertor, prototypical of burning plasmas or reactors. Fueling methods need to be extended to regimes with both high performance and internal transport barriers. Examples are gas injection into the scrape-off layer (SOL) but not intended to fuel the core, gas injection in the divertor for power handling (detached operation), and impurity introduction for plasma core, edge (radiating mantle), and SOL radiation of power. The technological issues are tightly coupled here since high heat flux materials would be required in the divertor and continuous cooling of plasma-facing components and divertors is necessary for steady state operation. Although several issues associated with fueling, pumping, and impurity control, as well as high flux divertor and core/edge/SOL radiation can be examined in DIII-D and C-MOD, very long pulses (greater than 100 s) are required to demonstrate confidence in these methods and supporting technologies. Internationally, these steady state issues are being examined on Tore Supra (where technological issues have been identified after very long pulses), and in KSTAR (after 2004). Possible future devices where power and particle handling could be addressed are

ITER-RC and FIRE, both of which have burning plasmas and long-pulse operating scenarios and therefore require capability approaching that of a reactor.

Spherical Torus. Steady-state issues for the spherical torus (ST) are considered primarily in the context of the volume neutron source application. Such a device requires continuous operation but, compared to power plants, only modest core performance ($Q \approx 1$) that would relax the requirements on beta, fraction of bootstrap current, and conducting walls. Many of the steady-state issues are closely related to those of the tokamak and indeed will benefit from tokamak research, but there are important differences as well. Because its low aspect ratio restricts the space available for primary windings, the ST cannot rely on induction for startup so a key control issue is to determine what non-inductive current-drive methods are most effective in ramping up to steady-state conditions. Candidates are coaxial helicity injection (CHI), neutral beam injection (NBI), high-harmonic fast waves (HHFW), and electron cyclotron and electron Bernstein waves (ECW, EBW), all of which will be studied on the NSTX experiment. The key issues for these approaches are insulator feasibility for CHI, supra-Alfvénic instabilities for NBI, wave physics and energetic-ion absorption for the novel HHFW regime, and wave penetration in overdense plasmas for ECW and EBW. As with the tokamak, profile evolution to unstable configurations is expected to be a pulse-length limiting mechanism in the ST if not controlled. The promising ST profile control candidates are NBI and HHFW. Electron Bernstein wave (EBW) heating is also being explored as a means of providing localized power deposition. These profile control approaches will all be studied in NSTX. In contrast to advanced tokamaks, which must operate close to stability limits, the ST can obtain high-beta performance without wall stabilization at high values of the safety factor ($q_\psi$=10), where disruptions and internal reconnection events may not occur. If not, the disruption avoidance strategy may be simpler for the ST than for the tokamak.

The compact size of the ST presents a special challenge for power and particle handling. The key issue at this time is to determine which of the two possible geometries is more promising, the tokamak-like X-point divertor or the ST natural divertor. The latter takes advantage of the high flux expansion factor (~10) that is possible by allowing the plasma to contact the center post at a reduced level of inboard plasma heat flux. Both geometries will be tested in NSTX.

Reversed-Field Pinch. Achieving steady-state operation in the reversed-field pinch (RFP) is a concept-specific issue because the primary sustainment methods envisioned for the other concepts do not apply. Bootstrap currents will be low because of the low poloidal beta in the RFP, and radiofrequency current drive efficiencies are too low to be economically practical. The most promising technique for the RFP is oscillating-field current drive (OFCD), although there are concerns about its compatibility with RFP transport reduction goals. Exploratory experiments on ZT-40M several years ago showed promising results. The next step experimentally is to extend performance to significant fractions of the total current to investigate issues of efficiency and penetration, as well as the viability of this approach. Such experiments are planned to be carried out on the MST facility as part of the RFP proof-of-principle program. Since the physics of OFCD is similar to that of CHI, collaborations with the ST and spheromak programs would be beneficial. In addition there is a need and an immediate opportunity to strengthen the theoretical foundations of OFCD. Available tools for this include the M3D and NIMROD codes.

Spheromak. For the spheromak, which also has little bootstrap current to exploit, the most promising sustainment approach is coaxial helicity injection (CHI). Erosion of CHI electrodes is thus a concern and their compatibility with good plasma performance is a key issue. For this reason, edge physics is a high-priority focus for spheromak research in the coming years, including experiments on the SSPX facility and modeling using the UEDGE code. Other key issues being studied are CHI efficiency, which is potentially very high, and compatibility with good confinement and profiles optimized for high beta. As an alternative to CHI, electrodeless helicity injection methods are being explored. Control of tilt and shift modes are necessary for sustainment of the spheromak configuration, and feedback control of the resistive-wall version of these modes is likely to be required. Power and particle handling issues for the spheromak, except for those associated with CHI, are the generic ones for magnetic fusion, although differing in detail (e.g. a relatively narrow scrape-off layer but large flux expansion in the divertor). An issue for the spheromak program in the next few years is to determine whether continuous or pulsed operation is the more promising direction. Repetitive pulsing would introduce challenges (e.g., thermal and mechanical cycling of the first wall or use of a liquid wall) but would also relax some constraints associated with continuous operation.

Field-reversed configuration and other emerging concepts. A number of emerging concepts were considered, but only briefly, in this study. They include the field-reversed configuration (FRC), the flow pinch, ion rings, electrostatic confinement, levitated dipole, and centrifugal confinement. For some of these, steady state operation is a natural consequence of how they are formed. Most are at such an early stage of development that steady state issues are not central at this time, though extending beyond very short pulses is. Sustainment possibilities considered for the FRC include neutral beams, compact-toroid injection, and rotating fields (the rotomak concept, which is being tested experimentally at the University of Washington). For the levitated dipole, the issue is continuous particle fueling. The physics of the anticipated convective cell is envisioned as central in transporting fuel injected at the outside (by gas puff or pellets) to the reactive high-field region on the inside. This will be studied on LDX.

**Summary and Key Opportunities**

Achieving steady-state operation is a critical objective for the goal of developing an optimum magnetic confinement system for fusion. The two over-arching issues are those which are currently limiting progress toward a steady-state system:

- Plasma control, to achieve and sustain a high-performance plasma configuration, and
- Power and particle handling, compatible with a high-performance plasma configuration.

Some progress toward resolving these issues has been made, but it is clear that their full resolution will be extremely challenging. Complementary approaches, pursued in parallel, are needed to have a reasonable chance of succeeding in a timely manner. Fortunately, the existence of a portfolio of magnetic configurations provides such an opportunity, namely the opportunity to simultaneously pursue a broad range of scientific issues important for steady state operation.

In this study we identified a range of promising approaches for steady-state plasma control and for steady-state power and particle handling. We consider these two classes of issues (control and power and particle handling) to be of equal importance, but we recognized that the power and particle handling issues were being addressed with a greater degree of expertise by the Boundary Control working group. For this reason only, we gave greater emphasis to plasma control issues in our assessment of opportunities. We examined the spectrum of magnetic concepts and identified opportunities in all of them for making progress in resolving steady-state issues. These were reported above. Among them, some key opportunities stood out as being particularly important, and in concluding we would like to highlight these.

1. Current profile control to prevent evolution to unstable configurations.

There is a near-term opportunity, in a complementary pair of programs using U.S. tokamaks, to resolve the issue of maintaining a stable advanced-tokamak plasma configuration by current profile control. Electron cyclotron current drive will be tested in combination with neutral beam-driven and bootstrap currents on DIII-D over the next three years. The planned complementary program will test lower hybrid current drive in combination with ion cyclotron heating and bootstrap currents on Alcator C-Mod from 2002 through 2008. In the longer term, the NSTX (from 2001) will extend advanced-tokamak current profile techniques to the spherical torus concept and the new Korean KSTAR superconducting tokamak (from 2004) will extend the pulse length of advanced-tokamak discharges to 20-300 s.

2. Helical fields and 3D shaping for disruption suppression.

The stellarator proof-of-principle program proposed by U.S. stellarator researchers will, in the next decade, test the use of externally-generated helical fields to avoid disruptions in high-beta plasma configurations with tokamak-like aspect-ratios. Complementary design approaches have been developed and it is planned to test both. The quasi-axisymmetric (QA) approach, which has bootstrap current levels and physics properties similar to the advanced tokamak, will be tested in the NCSX proof-of-principle experiment. The quasi-omnigeneous (QO) approach, which minimizes the bootstrap current and is similar to optimized stellarators, will be tested in the QOS concept-exploration-level experiment. Either of these approaches, if successful, would provide a basis for developing a steady-state plasma configuration with low recycled power and without disruptions.

3. MHD mode control for steady-state, high-beta scenarios.

A complementary set of programs planned for the next five years provides a good opportunity to test the elements of a strategy for controlling dangerous MHD instabilities using the tokamak, spherical torus, and reversed-field pinch. Feedback control of the neoclassical tearing mode with electron cyclotron waves, and of the resistive wall mode with coils will be investigated in the DIII-D tokamak over the next three years. The NSTX spherical torus will operate for its first 2-3 years without its conducting wall to provide a baseline for assessing the performance gains once the conducting wall is installed thereafter. The physics understanding will be relevant to tokamaks as well as the ST. The RFP proof-of-principle program includes an opportunity for a concept-exploration-level study of kink mode stabilization with a resistive shell. By carrying out these complementary programs over the next five years, we will realize significant gains in our understanding of the passive and active control of MHD modes that can prematurely terminate high-beta magnetic configurations.

4. Innovative current drive for startup and sustainment.

Several innovative approaches to magnetic configuration startup and sustainment will be studied in the next five years. In the NSTX spherical torus, coaxial helicity injection (CHI), high-harmonic fast waves, neutral beams, and bootstrap current are available. Experiments to determine the best startup and sustainment scenarios using these elements have high priority. The new SSPX spheromak will add to the understanding of CHI and the associated edge physics and transport implications. Oscillating-field current drive will receive its most significant test to date in experiments on the MST reversed-field pinch device. In addition there is a need and an opportunity now for theoretical research to strengthen the physics foundations for this approach. Finally rotating magnetic field sustainment (the rotomak concept) will be undergoing an important test in the University of Washington field-reversed configuration (FRC) facility.

5. Local turbulence and transport suppression for pressure and bootstrap profile control.

We highlight this as an area in need of attention, where the currently planned research is inadequate in comparison with its importance. Local transport control is the only possibility for controlling pressure and bootstrap current profiles in self-sustained magnetic plasma configurations. There has been substantial progress in understanding how to improve confinement by forming transport barriers using transient methods, but sustainable techniques are needed. Flow-shear control by radiofrequency waves is the most likely tool, and experiments with mode-converted ion Bernstein waves (IBW) are being carried out on Alcator C-Mod. However, direct-launch IBW also needs an in-depth investigation, and more new ideas are needed. Scientific and technological innovations in this area should be encouraged and initiatives to test promising ideas should be undertaken.