REPORT OF THE
FESAC
TOROIDAL ALTERNATES PANEL

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CHAPTER 1

EXECUTIVE SUMMARY

1. EXECUTIVE SUMMARY

As the world fusion program is moving into the ITER era and the National Ignition Facility at LLNL prepares for operation starting in 2009, the U.S. Fusion Energy Sciences Program is developing a comprehensive strategic plan for its research portfolio. This portfolio includes a variety of fusion concepts ranging from the main-line tokamak and advanced tokamak to stellarators and other toroidal systems, to mirror and dipole configurations, to inertial fusion energy concepts.

As part of an overall strategic planning process, The Department of Energy Under Secretary for Science asked FESAC to "critically evaluate the status of, and scientific opportunities for, major alternate magnetic confinement configurations." Specifically, FESAC was asked to:

- "identify and justify a long-term objective for each concept as a goal for the ITER era, and to
  - critically evaluate the goal chosen for each concept, and its merits for fusion development;
  - identify and prioritize scientific and technical questions that need to be answered to achieve the specified goal;
  - assess available means to address these questions; and
  - identify research gaps and how they may be addressed through existing or new facilities, theory and modeling/computation."

In addition, the Under Secretary asked that FESAC

- "identify and prioritize the unique toroidal fusion science and technology issues that an alternate concept can address, independent of its potential as a fusion energy concept."

This report seeks to answer these charges as the product of a six-month study by a seventeen member FESAC subpanel consisting of experts from the U.S. fusion program.

Over the years a wide variety of magnetic configurations have been studied as attractive paths to fusion energy. At present the tokamak is the leading contender, largely because of its superior plasma physics performance, which has resulted in an intensive development program. For historic reasons other configurations of interest have been denoted as "alternate concepts;" that is, they are alternates to the tokamak. However, it is worth noting that all the concepts considered here embody forms of magnetic confinement in toroidal geometry and have significant areas of common physics.

The charge to FESAC focused on four specific toroidal alternate concepts: the Stellarator, the Spherical Torus (ST), the Reversed Field Pinch (RFP), and the Compact Torus (CT) which consists of the spheromak and the Field-Reversed Configuration (FRC). Detailed information can be found in the Overview (Chapter 2) and chapters on individual concepts (Chapters 4-7).

Scientists have made considerable research progress on each of these four concepts over the past decade, utilizing new diagnostics, upgraded facilities, and improved simulation codes to gain scientific insight. As a result, new opportunities will exist during the ITER-era for taking new steps to improve understanding and achieve plasma parameters closer to fusion conditions for these devices.

In developing this report, the FESAC panel sought input from the fusion science community in written form and in open meetings. the panel worked with experts in the community to identify and understand the ITER-era goals (i.e., goals for the next 15-20 years) and the associated scientific and technical issues that need to be addressed to reach those goals. This was a highly interactive process involving considerable technical detail. In parallel, the panel developed a methodology for evaluating these goals and prioritizing the issues on a concept-by-concept basis by balancing overall scientific merit against risk and weighing importance, urgency, and general applicability of results. This evaluation and prioritization methodology largely followed that developed by the 2007 FESAC panel that examined Priorities, Gaps, and Opportunities for a Long-Range Strategic Plan for Magnetic Fusion Energy.

The panel focused its efforts on concept-by concept evaluations rather than direct comparisons between the four alternates, as this provides a unified and easily readable structure for the report. Clearly some of the concepts are much closer to reaching fusion conditions, some have stronger theoretical support for their expectations, while others face more obvious and compelling challenges than others. However, in all cases clearly defined research needs have been identified which can provide a reasonable basis for planning future research investments.

The complete summary of the panel findings is contained in Chapter 3. Here we highlight five general findings for the toroidal alternates, followed by identification and evaluation of the ITER era goals for the four concepts examined in this report.

We highlight the ITER-era goals here because they are the primary driver for the prioritized list of scientific and technical issues (more fully explained in Chapters 4-7 of this report). Detailed consideration of these issues, in turn, formed the basis for evaluating the ITER era goal using the agreed upon methodology, as detailed in Appendix A. So, following the general findings below, we briefly state the ITER-era goal for each concept, highlight the potential advantages, and provide a synopsis of the goal evaluation carried out more fully in Chapter 3.

1.1. GENERAL FINDINGS REGARDING TOROIDAL ALTERNATES RESEARCH

1. The overall quality of the science in toroidal alternates research is excellent, with broad benefit to the U.S. fusion program and to general plasma sciences including applications to other disciplines. The work is strongly focused on developing scientific understanding as the path forward to achieving ITER-era goals.
2. Alternate Concepts research provides significant benefit to the broader U.S. fusion and plasma science program by effectively recruiting and training bright young people to be our nation's next generation fusion scientists.
3. Predictive simulation plays a central and increasingly visible role in toroidal alternates research, in many cases pushing the state-of-the-art computational capability.
4. Alternate concept research requires similar tools to other parts of the fusion program, but it has uniquely urgent needs in two areas: (1) theory and simulation, which are particularly challenged by complex 3D resistive MHD physics, kinetic effects, and anomalous transport seen in these experiments; and (2) diagnostic capability, which is especially vital for developing physics understanding of the less mature concepts. These areas deserve priority emphasis and support to strengthen scientific contributions and solidify projections to next step experiments.
5. Promise for Fusion Energy. Some of the four concepts we have considered are much more highly developed than others, yet all of them require further development and investigation before any definitive assessment of their fusion energy capabilities is possible.

1.2. CONCEPT SPECIFIC FINDINGS RELATED TO ITER-ERA GOALS

Here in brief summary we present the ITER-era goal for each concept, very briefly mention the potential advantages for the concept, and provide a short synopsis of the panel's evaluation of the scientific benefits and risks related to the goal; we were not asked for, nor do we make recommendations that such goals should be adopted by the U.S. Fusion Energy Sciences Program. Further explanation may be found in Chapter 3 – Findings, where we also list the high priority issues for each concept.

Stellarator

ITER-Era Goal. *Develop and validate the scientific understanding necessary to assess the feasibility of a burning plasma experiment based on the quasi-symmetric (QS) stellarator.*

- Potential Advantage for Fusion. The stellarator, through the use of external coils to produce the full set of confining magnetic fields, offers the potential for a robust, steady-state fusion device without need for external current drive. Quasi-symmetric configurations may lead to improved confinement and more compact designs.
- Evaluation Synopsis. This ITER-era goal addresses critical scientific and technical issues for quasi-symmetric stellarator configurations. Achieving the goal will advance the knowledge of steady-state confinement, but requires significant extrapolation in plasma parameters to demonstrate the benefits of the quasi-symmetry, as well as a design strategy that addresses both robust flux surfaces and manufacturing constraints.

Spherical Torus

ITER-Era Goal. *Establish the ST knowledge base to be ready to construct a low aspect-ratio fusion component testing facility that provides high heat flux, neutron flux, and duty factor needed to inform the design of a demonstration fusion power plant.*

- Potential Advantage for Fusion. The ST is a low-aspect-ratio tokamak that minimizes the center-post volume and nuclear shielding requirements while at the same time allowing operation at higher normalized plasma pressure. Together, these offer the potential for an attractive test facility for developing fusion components.
- Evaluation Synopsis. The ITER-era goal is clear, well motivated, and tied tightly to the overall fusion energy roadmap. Achieving this goal will advance knowledge of low-aspect ratio tokamak confinement, but entails significant extrapolation in non-inductive current drive, electron transport, power handling, and magnet technology.

Reversed Field Pinch

ITER-Era Goal. *Establish the basis for a burning plasma experiment by developing an attractive self-consistent integrated scenario: favorable confinement in a sustained high beta plasma with resistive wall stabilization.*

- Potential Advantage for Fusion. The distinctive feature of the RFP that motivates interest as a fusion concept is that the externally applied magnetic field is relatively small, leading to simpler magnets. Current flowing within the plasma self-consistently generates the confining magnetic field, potentially yielding more compact devices.

- Evaluation Synopsis. The ITER-era goal is clear and addresses critical scientific and technical issues for the RFP approach. Achieving this ambitious goal would establish the possibility for a low-external field approach to magnetic fusion. Significant challenges in establishing current sustainment with good confinement will need to be overcome to realize this goal.

Compact Torus (Spheromak and FRC)

ITER-Era Goal. *To demonstrate that a compact toroid with simply connected vessel can achieve stable, sustained or long-pulse plasmas at kilovolt temperatures, with favorable confinement scaling to proceed to a pre-burning CT plasma experiment.*

- Potential Advantage for Fusion. The CT concepts seek use internal plasma currents to confine fusion plasmas in a compact, simply connected chamber to potentially reduce unit cost significantly while increasing maintainability of fusion power systems.
- Evaluation Synopsis. The ITER era goal for the CT is clear and aims for critical progress toward fusion energy with self-organized plasmas; achieving this goal would advance and validate magnetic confinement in a simply-connected chamber with no external toroidal field. However, the goal is highly ambitious, requiring a large extrapolation in stability, confinement, and sustainment, and there is limited theoretical or experimental basis for prediction.

CHAPTER 2

INTRODUCTION

2.1. OVERVIEW

The mission of the U.S. Fusion Energy Sciences Program is to advance plasma science, fusion science, and fusion technology — the knowledge base needed for an economically and environmentally attractive fusion energy source. Achieving this goal has and will continue to require a high level of dedicated research and innovation to overcome the simultaneous challenges of plasma physics and fusion engineering.

Over the years a wide variety of magnetic configurations have been suggested as attractive paths to fusion electricity. At present the tokamak is the leading contender, largely because of its superior plasma physics performance, which has resulted in an intensive development program. For historic reasons the other magnetic configurations of interest are denoted as "alternate concepts" where alternate implies non-tokamak. The alternates include several toroidal concepts, as well as other non-toroidal concepts such as linear magnetic mirrors; however the main effort is focused on the toroidal configurations. Thus, rather than "tokamak and alternates" a more accurate description of the main components of the US Fusion Program would be "Toroidal Magnetic Concepts" indicating that there is essentially a continuous spectrum of toroidal configurations aimed at fusion energy.

There are technical challenges for all configurations, but the tokamak, by virtue of its performance and promise, is the vehicle for the world's first MFE burning plasma experiment, ITER. Even so, it is not as yet clear which configuration will ultimately lead to the most attractive fusion reactor. In terms of comparisons each configuration is distinguished by a different choice in trade-offs between plasma physics and fusion engineering with a quantum gain in one usually being offset by a quantum loss in the other.

Therefore, the US and World fusion programs have maintained a broad approach in toroidal magnetic configurations with each class of experiments progressing toward the ultimate goal of producing electricity along two parallel paths: (1) by means of its own focused research progress, and (2) by expanding the general knowledge base of plasma physics and fusion engineering. It should be emphasized that each of the alternates discussed in this report offers a potential opportunity for an attractive reactor but also faces significant scientific and technological issues. Addressing these issues can draw on the progress made in tokamak science and technology; conversely, as the alternates operate in quite different parameter regimes they will broaden the science and technology in ways which may help improve the tokamak.

In February 2008, as part of an overall strategic planning process for the Fusion Energy Sciences Program, The Department of Energy Under Secretary for Science asked FESAC to "critically evaluate the status of, and scientific opportunities for, major alternate magnetic confinement configurations." This report seeks to address the charge by providing an informed evaluation of the four specific toroidal magnetic configurations as requested: the Stellarator, Spherical Torus, Reversed Field Pinch, and Compact Torus.

This Introduction is meant to provide the overall context for the more detailed concept evaluations that follow, by giving a brief overview of the salient features of each concept. We start with a brief discussion of common issues for fusion energy, which must be addressed by any concept. These issues provide motivation for exploring the various concepts. We include the tokamak both for completeness and because it provides the basis by which many of the advantages and challenges for concepts are often compared. The Introduction concludes with a discussion of overall progress towards the fusion goal.

2.2. COMMON ISSUES FOR FUSION ENERGY

There are a well known set of common issues related to fusion energy development that any concept for magnetic confinement fusion energy must address. These have been laid out in various ways in numerous reports and include:

- Plasma confinement, transport, and overall energy balance
- Configuration sustainment (e.g., field generation, current drive)
- Operating limits (e.g., absolute plasma pressure for given coil limits, fuel-density limits)
- Thermal loads and operating lifetime of plasma-facing components (e.g. divertors)
- Plasma exhaust, particle and impurity control, overall tritium fuel cycle
- Wall neutron loading (thermal loading and neutron damage to materials)
- Safety, reliability, maintainability in a nuclear environment, environmental impact

Various magnetic fusion energy concepts offer potential reactor advantages in one or more of these areas. However, in the quest to reach fusion conditions in the laboratory, fusion research since the inception of the program, has emphasized the first three issues, looking at the other issues only as needed to optimize plasma performance. Ultimately, fusion research must produce an integrated solution to all these issues, as enumerated in the 2007 FESAC report on Priorities, Gaps, and Opportunities for a Long-Range Strategic Plan for Magnetic Fusion Energy (the Greenwald report).

There are two well-known metrics for fusion plasma performance – the parameters $\langle n\rangle\tau_E$ and $\langle T_i\rangle$. Here n is the density, T_i is the ion temperature and τ_E is the energy confinement time characterizing thermal conduction transport losses (i.e. it is the natural plasma cooling down time when all heating sources are removed). Also, here and elsewhere $\langle\ \rangle$ denotes volume average over the plasma. For a fully ignited fusion reactor a simple power balance in the plasma, equating alpha particle heating to thermal conduction loss, leads to a minimum requirement on the value of $\langle nT\rangle\tau_E = \langle p\rangle\tau_E$ of ≈ 8 atm-sec at $\langle T_i\rangle \approx 15$ keV, where $p = n(T_i + T_e)$ is the total plasma pressure. In the range from 10 to 20 keV, the fusion power density varies as $P_{fusion} \propto p^2 \approx n^2T^2$. One generally aims to maximize the fusion power density to lower the size and cost of fusion reactors, subject to other constraints as above. This requires high temperatures and good energy confinement.

There are fundamental limits to the plasma pressure which can be achieved with magnetic confinement fusion that are related to magnetohydrodynamic (MHD) stability. The limit depends upon the magnetic geometry, plasma collisionality, and internal profiles of density, temperature, flow/rotation, current, and proximity of conducting walls. It is usually expressed in terms of $\beta = p/P_B$, the ratio of the plasma pressure to *total* magnetic field pressure, $P_B = B^2/2\mu_0$ sometimes expressed as a percentage. In magnetically confined plasmas, $0 < \beta < 1$.

Both high β and high confining magnetic field in the plasma are needed to reach fusion conditions and produce high fusion power density, since $nT\tau_E = p\tau_E = \beta B^2\tau_E$, $P_{fusion} \propto \langle\beta^2 B^4\rangle$, and $\tau_E \propto B$. This motivates research to increase β and operate with the highest confining field possible. When the magnetic field is provided by external magnets, the field at the magnet (B_m) is often a limiting factor due to engineering constraints, so it is advantageous to have configurations with a large ratio of B to B_m. In a different strategy, some magnetic fusion concepts largely rely on "self-organized" internal plasma currents to produce the confining field, thereby eliminating complex external coils almost completely. For these concepts, finding efficient means to drive the plasma current is a major challenge.

The dependence on internal "self-organized" plasma current to produce some portion of the magnetic field generally reduces the stability or controllability of the configuration since the current is not constrained by fixed conductors, but is free to evolve with the plasma motion, which often gives rise to instabilities. Thus, across concepts there is generally a tradeoff between the degree of high β self-organization and controllability of configuration stabilities. Typically, higher β is achieved transiently with less external controls while more stable plasmas are achieved steadily at lower β. Toroidal alternate concepts, together with the tokamak, cover a full spectrum of this trade-off. Each of these concepts takes a particular trade-off combination, accompanied by a unique set of scientific and technical challenges toward a realistic fusion reactor scenario as considered in this report. We now provide a brief overview of these configurations.

2.3. THE TOKAMAK

The US tokamak program includes the world-class DIII-D and Alcator C-Mod experimental facilities at General Atomics and MIT respectively, as well as some smaller, university scale devices. Equally important, the program includes the US R&D effort for ITER, which is planned to be the dominant activity for at least the next two decades. The overall tokamak program includes experiment, theory/computation, diagnostic development, and enabling technology.

The tokamak has risen to its position of prominence because of its superior plasma physics performance in terms of macroscopic MHD equilibrium and stability, heating, and transport. This performance is largely attributable to several basic features of the tokamak configuration as illustrated in Fig. 2-1. These features are (1) an axisymmetric toroidal magnetic geometry, (2) a large externally applied toroidal field and (3) an internally induced toroidal current produced by a central solenoid, usually referred to as the Ohmic heating (OH) transformer. The plasma current provides a substantial amount of resistive heating to the plasma to raise its temperature into the keV range. It also produces a poloidal magnetic field that, in combination with the toroidal field, results in helically twisted field lines needed for plasma confinement. The applied toroidal field provides magnetohydrodynamic (MHD) stability at high plasma current and pressure.

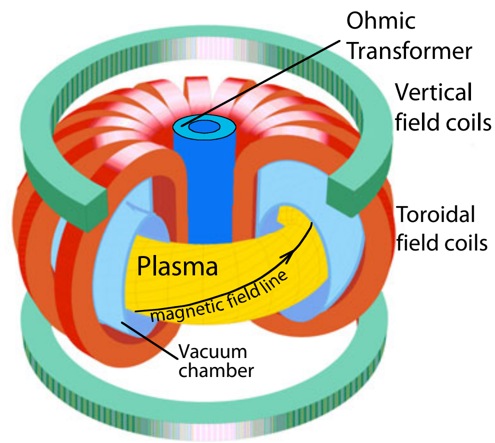


Fig. 2-1. Basic elements of the tokamak.

While a considerable effort has been expended on optimizing the plasma physics performance of tokamaks, the effort to optimize the power plant vision has received less attention, primarily because this is still a future, although steadily approaching goal. This reactor optimization involves the interplay between engineering and plasma physics. Issues to be addressed include steady-state current drive without the OH transformer, first-wall materials, and sudden disruptions of the plasma current, which can cause mechanical damage to the tokamak, and the need for a large, well shielded, high field superconducting magnet system. Assuming resolution of these issues, the ARIES team has developed a tokamak reactor design based on advanced tokamak (AT) operation. The end result of this design is a power plant with a competitive cost of electricity (CoE). The ARIES design, however, requires several substantial improvements in both plasma physics and fusion technology.

2.4. TOROIDAL ALTERNATE CONCEPTS

Toroidal Alternate Concepts considered here are a series of toroidal configurations whose mission-oriented goal is to provide alternate approaches to a fusion reactor. Each approach seeks to provide an attractive reactor vision by addressing one or more of the fusion issues outlined above, though usually at the expense of introducing new issues. We now very briefly discuss the stellarator, the spherical torus (ST), the reversed field pinch (RFP), and the compact torus (CT including the spheromak and field-reversed configuration or FRC).

2.4.1. The Stellarator

The stellarator is a 3-D toroidal-helical configuration with nearly all magnetic fields (both main toroidal field and poloidal field) produced by external coils, as in Fig. 2-2. Because the magnetic field and geometry of the plasma are largely determined by the external coils, stellarators can be optimized to operate with little or no plasma current and thus need no OH transformer or external current drive. Stellarators come in a variety of configurations depending on the details of the 3D field structure. Stellarators have confinement and β comparable to, but slightly lower than that of similar size tokamaks. They have the advantage of being an inherently steady state device although requiring a large toroidal magnetic field. Also, because there is either zero or only a small amount of induced toroidal current one expects the disruption problem to be almost, if not completely, eliminated.

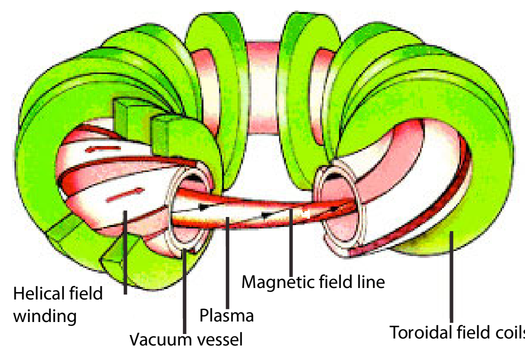


Fig. 2-2. Conventional stellarator with toroidal coils and helical windings.

The US vision for the stellarator seeks to optimize the magnetic configuration in a particularly innovative way by making use of the recent idea of “quasi-symmetry”. If successful, the principle of quasi-symmetry has the potential to yield improved confinement and lead to a more compact, economical reactor than that currently envisioned by the European and Japanese stellarator programs. Even so, new problems arise that are not present in the tokamak. Perhaps the most important one is that the superconducting coils needed to produce the quasi-symmetric toroidal-helical fields are substantially more complicated and difficult to build than the TF system for a tokamak. This is the trade-off that needs to be evaluated in future research. In view of the potential gains to be made it is therefore disheartening, although understandable, that the NCSX project has been canceled.

2.4.2. The Spherical Torus

The spherical torus (ST) is a very low aspect ratio ($A = R/a$, where a is the minor radius and R the major radius) tokamak that minimizes the center-post volume and nuclear shielding requirements [see Fig. 2-3]. Although the toroidal field in the center of the plasma is small by tokamak standards, one still needs a substantial TF magnet because the maximum field at the inner edge of the TF coil is relatively large as a consequence of the small aspect ratio. In fact it is worth noting that the need for a high toroidal field along the inside edge of the TF coil is a technologically limiting design feature for the stellarator, ST, and the tokamak.

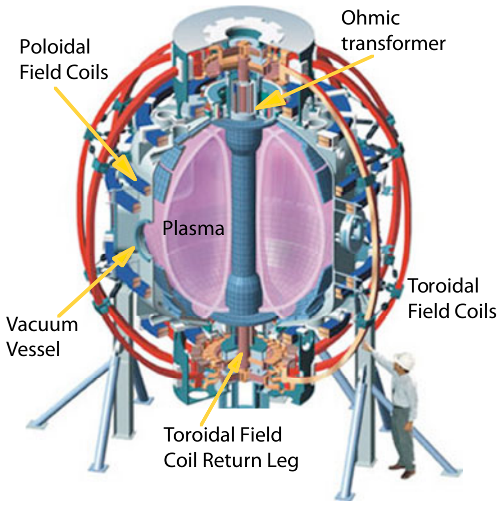


Fig. 2-3. Spherical Torus (NSTX experiment).

In terms of the physics, the low aspect ratio allows a higher plasma pressure relative to the magnetic pressure (i.e., high β) and thus makes a more efficient use of magnetic energy. Even so, in absolute units the engineering constraints at the inner leg of the TF-coil limit the ultimate achievable plasma pressure to somewhat lower values than in an equivalent standard tokamak. Overall, the price for increasing β in the ST is (1) a more difficult, even if smaller, TF magnet system, and (2) reduced volt-seconds in the OH transformer which is required for start-up and ramp-up of the plasma current.

One aim of the ST community is a compact Component Test Facility (CTF), whose path to success is achieved by eliminating the magnet shielding on the inner leg of the device through the use of demountable copper coils. The assumption is that low aspect ratio will lead to a smaller device overall, with a sufficiently large surface accessible for component testing. Thus the ST may represent an economical approach to a CTF. The Panel would like to note that a CTF mission, while very important to reach the fusion energy goal is not an official goal of the US Fusion program at this time. Furthermore, even within the context of a CTF mission, the Panel notes that a detailed comparative analysis has yet to be carried out which demonstrates that the spherical tokamak is the most desirable concept for this mission. These are tasks for the future.

2.4.3. The Reversed-Field Pinch

The reversed field pinch (RFP) attempts to improve the vision for magnetic fusion energy by greatly reducing the requirements on the Toroidal Field magnet system. This is accomplished by driving large toroidal currents within the plasma to produce the confining field, as shown in Fig. 2-4. Note that the TF coil is required to produce only a small initial confining field, which reverses as the toroidal current is ramped up to its maximum value. The RFP may achieve β greater than that of a conventional tokamak with a corresponding substantially reduced field on the coils. If successful this could lead to a somewhat smaller, technologically simpler, more economical reactor. The trade-off is that reducing the externally applied toroidal magnetic field leads to a reduction in plasma performance – increased MHD instabilities, shorter confinement times, and more difficult steady state plasma sustainment primarily due to increased twist in the field lines.

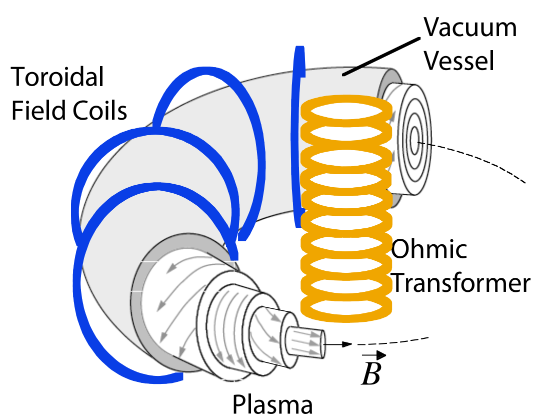


Fig. 2-4. The reversed-field pinch configuration.

Over the last decade major improvements in the performance of the reversed field pinch have been achieved. Most recently, multiple, simultaneous, MHD instabilities have been successfully stabilized by a sophisticated feedback system on RFP experiments in Sweden and Italy. With respect to transport losses, for base operation of an RFP these are in general high compared to a comparably sized tokamak. However, current profile control developed by the Wisconsin RFP group has improved performance to near tokamak levels by stabilizing various MHD instabilities related to the internal magnetic field generation, although this improvement has so far been attained only transiently.

Equally important, the RFP, like the tokamak, must be able to achieve steady-state operation in order to produce an attractive reactor from an engineering point of view. Further, current drive requirements facing the RFP are inherently more difficult than in a tokamak because (1) the natural transport driven toroidal current (i.e. the bootstrap current) is small in an RFP, and (2) the required current density is large in the RFP. The best option for steady-state current sustainment in the RFP may be Oscillating Field Current Drive (OFCD), where AC inductive loop voltages maintain a DC current via magnetic self- organization. The compatibility of this approach with the simultaneous requirement of good confinement is a key issue.

2.4.4. The Compact Torus

The compact torus approach actually consists of two configurations — the spheromak and the field reversed configuration (FRC) as illustrated in Fig. 2-5(a) and 2-5(b). Both share the property of having no toroidal field coils and no OH transformer whatsoever. As a result, the coils and vacuum vessel are cylindrical, though the plasma is still toroidal, potentially easing fabrication, assembly and maintenance. This advantage, combined with the elimination of the large toroidal magnet required in a tokamak, provides both CT concepts with a vision for a smaller, technologically simpler, economical reactor.

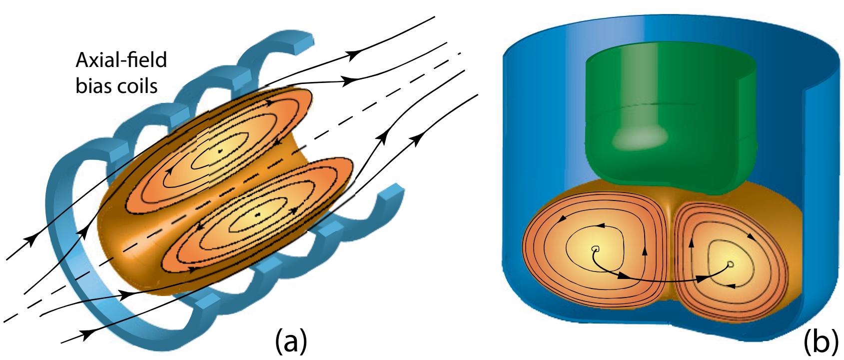


Fig. 2-5. (a) Field-reversed configuration, (b) spheromak configuration

Neither concept has yet achieved keV temperatures with acceptable transport losses so it is difficult to predict their future performance with respect to tokamaks. Furthermore, because there is only a small, turbulently induced toroidal field in the spheromak, or a zero toroidal field in the FRC one would expect both of these concepts to have a more difficult time matching the plasma physics performance of tokamaks with respect to MHD equilibrium and stability, transport, and current sustainment. Formation and current sustainment need major advances for the spheromak to succeed as a reactor. MHD stability and current sustainment in larger, reactor scale FRCs are comparably difficult. The largest FRC, the LSX was terminated by OFES in 1991, while LLNL recently canceled the SSPX, the largest spheromak in the US. Each of these actions has slowed the potential progress of the spheromak or FRC to rapidly advance as a serious reactor contender.

2.5. ALTERNATE CONCEPTS RESEARCH IN RELATION TO THE BROADER FUSION PROGRAM

The alternate concepts covered in this report represent about 25% of the non-ITER research within U.S. Fusion Energy Sciences program according the FY2009 Presidential budget request to Congress. In general, there is a strong connection between the alternates and the remainder of the fusion program maintained across all levels, from individual scientific collaborations to service on advisory and review panels, participation in topical conferences and workshops, and participation in the U.S. Burning Plasma Organization and ITER. Indeed, many tokamak scientists started their research career with thesis work on alternate concept experiments based at universities. Small-scale experiments at universities provide a unique hands-on experience and a strong emphasis on developing increased understanding of the behavior of high temperature plasmas that will be of use to the ultimate fusion energy mission. The basic fusion science plus student educational training missions of university research programs should not be underestimated.

2.6. STATUS OF ALTERNATE CONCEPTS RELATED TO THE FUSION ENERGY MISSION

In the last section of the overview an attempt is made to quantify the current status of the technical progress of each alternate concept related to the fusion energy mission. Two assessments are presented.

The first quantitative assessment, related to reaching fusion plasma conditions (the Lawson criterion) is illustrated in Fig. 2-6. Shown here is the classic plot of $\langle n \rangle \tau_E$ vs. $\langle T_i \rangle$. The data illustrated represent the "best" repeatable performance to date on one or several types of discharges for each alternate concept. Also shown are the data for several tokamaks and the projected value for ITER. The solid curve in the upper right hand corner is a plot of $\langle n \rangle \tau_E$ vs. $\langle T_i \rangle$ corresponding to the ignition boundary. Gaining access to the region above the curve is the basic goal of fusion research. The dashed curve slightly below the ignition boundary corresponds to the substantially reduced requirements for a Component Test Facility. The main purpose of this figure is to illustrate how far each concept has progressed towards its final goal, and nothing more. It would be unfair and incorrect to use this data to compare concepts with each other or the tokamak since the amount of resources applied to each differ enormously.

The second quantitative assessment attempts to normalize performance to resource allocation, thereby allowing a comparison, however crude, between the various alternate concepts and the tokamak. Ideally one would like to evaluate the ratio of performance to cost. As expected it is very difficult to accurately calculate the funds expended on any given concept since many experiments have had add-on funding during their lifetime, have been able to take advantage of site credits, have vastly different levels of expensive diagnostic equipment, etc. The approach taken

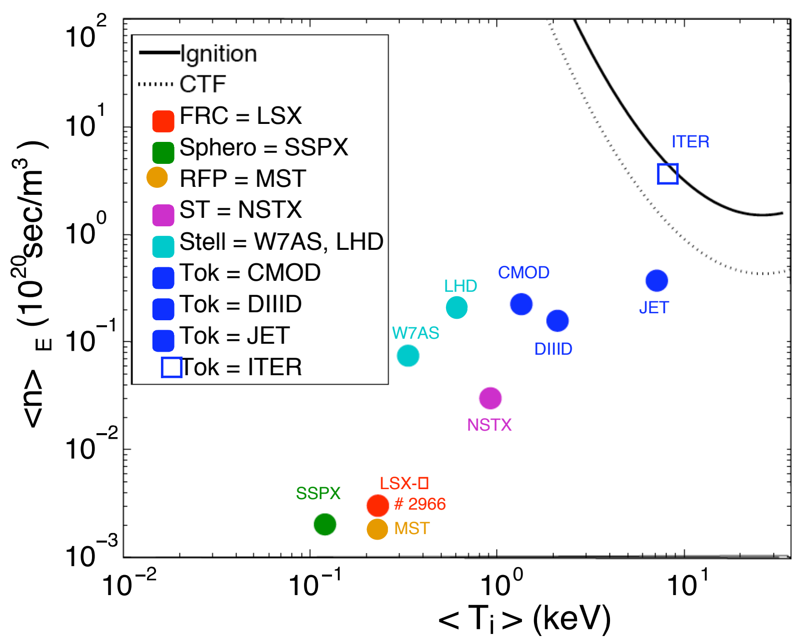


Fig. 2-6. Achievements of the various concepts relative to the goal of reaching fusion conditions.

here is based on Sheffield's analysis [Nucl. Fusion **25**, 1733 (1985)]. Sheffield has shown that β/χ is a good measure of physics performance reasonably independent of size or resources. Here β is the previously defined normalized pressure and χ is the thermal diffusivity, both basic properties of the plasma not directly related to size. In a fairly direct way β measures MHD stability and χ measures plasma energy transport. For a fusion reactor a combination of high β and small χ is needed for ignition, with some ability to trade-off one versus the other. The ratio β/χ is useful since it simultaneously takes both properties into consideration.

A useful macroscopic version of the metric to measure relative performance is

$$\frac{\bar{\beta}}{\bar{\chi}} = \left(\frac{\langle p \rangle}{\langle B^2 \rangle / 2\mu_0}\right)\left(\frac{\tau_E}{a_{eff}^2}\right) = \frac{2\pi^2 R_0}{W_{mag}} \langle p \rangle \tau_E \quad .$$

Here $a_{eff} \equiv (V_p/2\pi^2 R_0)^{1/2}$ is the effective minor radius of the plasma, R_0 is the major radius of the plasma, V_p is the plasma volume, and $W_{mag} = \langle B^2/2\mu_0 \rangle$ is the stored magnetic energy. Also, note that the quantity $\langle B^2/2\mu_0 \rangle$ includes both the toroidal and poloidal magnetic energies.

Figure 2-7 is a plot illustrating $1/\bar{\chi}$ vs. $\bar{\beta}$ for the various concepts, as determined from common analysis of published data points. There is a large amount of information contained in this plot and it is worthwhile to examine it carefully. First, the dotted curves are curves of constant $\bar{\beta}/\bar{\chi}$. In an approximate sense, experimental operation at any point along a given curve corresponds to similar performance in terms of achieving plasma ignition. Different points along the same curve represent different trade-offs between $\bar{\beta}$ and $\bar{\chi}$, each point, however, producing the same value of $\langle p \rangle \tau_E$ (in the same machine at the same temperature). The more desirable regime of operation corresponds to higher curves. The shaded band represents the approximate regime corresponding to fusion reactors.

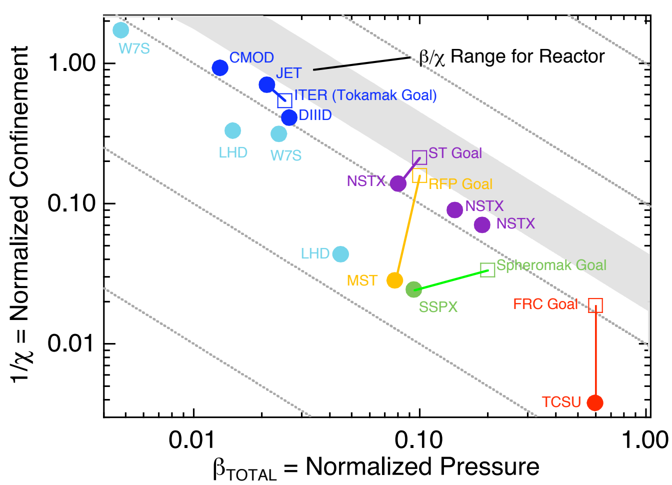


Fig. 2-7. Normalized performance of the various concepts in terms of β/χ as above.

Next, note that the various concepts are color coded. The deep blue points correspond to tokamaks and serve as a reference. The alternates are denoted as follows: spherical tokamak = violet, stellarator = light blue, RFP = orange, spheromak = green, and FRC = red. There are also different shape markers for each concept. The solid circular points represent the actual experimental data from the "best" discharges discussed in the $\langle n \rangle \tau_E$ vs. T_i diagram. (Published data from the LSX FRC experiment [Hoffman, Nucl. Fusion **33**, 27-38, (1993)] was not included because the discharges were essentially transient, with the pulse length comparable to the confinement time.) The open square markers designate the ITER era goal for each concept as discussed in the main body of the report. The curves connecting the points represent the path that each concept must follow if it is to achieve its ITER-era fusion goal.

What can be concluded from this figure? As expected, the tokamak has achieved the best relative performance to date. Interestingly, tokamaks both large and small, including ITER, all operate on the same high value of $\bar{\beta}/\bar{\chi}$. The path to the ITER-era goal shows that only a very small extrapolation in MHD and transport is required.

The performance of alternates which use external coils to produce a large toroidal field (i.e., the stellarator and the ST) closely approaches the tokamak. The spherical torus has a somewhat higher $\bar{\chi}$, although one should recall that for the CTF application, higher transport may be acceptable because the goal is neutron production and not electricity. The stellarator has a slightly lower $\bar{\beta}/\bar{\chi}$. What is hidden for the stellarator is the advantage that they are inherently steady state, disruption-free devices and the disadvantage that a given size machine is usually more expensive to build than a comparable tokamak. Still, it is encouraging to see that stellarator performance is approaching that of a tokamak. Learning how to build less expensive stellarators remains a critical goal for the future. Overall, both the ST and stellarator programs must learn to maintain good confinement while operating at slightly higher values of β to reach their ITER-era goals.

In normalized units, the performance of those alternates concepts that rely on plasma currents to generate the confining field (i.e., the RFP and CT) has improved considerably but is still substantially lower than that of tokamaks. Eliminating the large external toroidal field does indeed have a major impact on performance. The conclusion is that these alternates have some very difficult scientific challenges to overcome. In particular, significant improvements in confinement will be needed to reach their ITER-era goals. The physics is more difficult but the reactor vision is more attractive.

2.7. THE STRUCTURE OF THE REPORT

Building on this overview, the main content of the report (Chapters 4-7) examines in detail each of the alternate concepts described above. These chapters each begin with a brief overview of the fundamental operating principles, key advantages, and current research status for the concept. Then the future goals, critical issues, and facilities and gaps are set forth in a uniform, self-consistent manner. The major findings, including the goal evaluations, are collected in Chapter 3 following this Introduction. Chapter 8 provides an overview of the panel process and methodology for evaluation and prioritization. Our goal in this process is to provide DOE and the fusion community with the information needed to develop a strategic plan for investing in toroidal alternate concepts research as part of the US Fusion Program.

CHAPTER 3

FINDINGS

3.1. GENERAL FINDINGS

The charge to the Panel asked for specific information on each of four lines of research on toroidal magnetic confinement concepts: the Spherical Torus, the Stellarator, the Reversed-field Pinch, and Compact Tori (encompassing both field-reversed configurations and spheromaks). The charge did not ask for a relative comparison between concepts, though it is inevitable that such comparisons can and will likely be made from a set of similar individual evaluations. Therefore, apart from the first five general findings, the remainder of the findings are organized concept by concept. More complete information related to the goals, issues, facilities, gaps, and opportunities can be found in the individual chapters on each concept.

- **Finding 1.** *The overall quality of the science in toroidal alternates research is excellent, with broad benefit to the U.S. fusion program and to general plasma sciences including applications to other disciplines. The work is strongly focused on developing scientific understanding as the path forward to achieving ITER-era goals.*

  This review shows that progress on toroidal alternate concepts is closely tied to improved understanding of key physics issues, obtained from a close coupling of theory and experiment where possible. Alternates research has provided better understanding of magnetic reconnection processes, 3D field effects, flow damping, electron and ion transport, pressure limits, and plasma-wall interactions, among other areas, all of which are applicable not only to the alternates, but to the tokamak program and other related fields as well. These benefits are evident by collaborative activities and an excellent record of publications.

- **Finding 2.** *Alternate Concepts research provides significant benefit to the broader U.S. fusion and plasma science programs by effectively recruiting and training bright young people to be our nation's next generation fusion scientists.*

  Research on toroidal alternates has broad support from universities, through on-campus experiments and collaborations with larger facilities. The smaller scale of the projects provides a "hands-on" environment for recruiting new students into fusion science research, and the science is interesting. In addition, the connection to the development of fusion energy is a strong motivator for young people seeking to make a difference to the future, and the link to larger-scale fusion efforts supported by OFES often leads to future employment opportunities.

- **Finding 3.** *Predictive simulation plays a central and increasingly visible role in toroidal alternates research, in many cases pushing the state-of-the-art computational capability.*

  From 3D effects to resistive MHD, research on alternate concepts involves complex plasma physics that challenges understanding. The challenge is often compounded by a sparse data set from limited measurements on a small number of devices, thus providing strong motivation for use of numerical simulation to formulate a coherent picture of the behavior of these plasmas. Improved codes and visualization tools are making this possible to a greater degree now than before, and such codes are being used on a more routine basis for interpreting results and designing new experiments. However, the computational challenges remain significant, both in terms of the physics in the codes and the demands for processor time.

- **Finding 4.** *Alternative concept research has uniquely urgent needs in two areas: (1) theory and simulation, which are particularly challenged by complex 3D resistive MHD physics, kinetic effects, and anomalous transport seen in these experiments; and (2) diagnostic capability, which is especially vital for the less mature concepts. These areas deserve priority* *emphasis and support, within the alternates program, to strengthen scientific contributions and solidify projections to next step experiments.*

- **Finding 5.** *Promise for Fusion Energy: Some of the four concepts we have considered are much more highly developed than others, yet all of them require further development and investigation before any definitive assessment of their fusion energy capabilities is possible.*

  All four concepts face challenges that are common to any approach for magnetic fusion energy. These challenges have been enumerated by the FESAC Strategic Planning Panel in their 2007 report on *"Priorities, Gaps, and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy."* Ultimately, a concept's overall potential for fusion energy will be determined by the economic attractiveness of an integrated solution to these issues. The scientific and technical issues identified in our report correspond to the minimum set required to achieve the stated ITER-era goal for each concept, and not the full set to be resolved for fusion energy development.

3.2 CONCEPT-SPECIFIC FINDINGS

The panel worked with the scientific community to identify a long-term objective for each concept as a goal for the ITER-era (the next 15-20 years) and develop a list of scientific and technical questions or issues that need to be addressed to achieve the specified goal. Subsequently, the panel evaluated the goals and prioritized the issues on a concept-by-concept basis using the methodology outlined in Appendix A. These goal statements and descriptions, along with the possible initiatives listed in each concept chapter are not meant to imply endorsement of a specific course of action on one concept as compared to others. For clarity in relating goals, issues, means, and gaps, we proceed one-by one through the concepts as Findings 6-9 here and have selected only the highest priority issues (Tier 1) – a complete listing of the issues with more comprehensive description of each, appears in the following concept chapters.

3.2.2 Finding 6: Stellarator

- **6a. ITER-Era Goal for the U.S. Stellarator Program.**

  Develop and validate the scientific understanding necessary to assess the feasibility of a burning plasma experiment based on the quasi-symmetric (QS) stellarator.

  This goal focuses on conducting the scientific and engineering research required to write a physics-basis document, similar to the ITER Physics Basis documents published in Nuclear Fusion, which would be necessary to begin construction of a burning plasma experiment based on a quasi-symmetric stellarator. It is anticipated that the LHD, W7-X, and ITER experiments would provide significant understanding relevant to the QS stellarator concept, reducing, but not eliminating, the need for intermediate scale experiments.

  Importance and Relevance. This goal represents a logical next step to fusion energy for the stellarator. The design space for stellarators is quite extensive, which motivates the validation of sufficient predictive capability to confidently choose among alternatives and extrapolate to burning plasma conditions without the construction of a full sequence of scale models for a wide variety of stellarator types. Therefore, the research required to achieve this goal addresses critical scientific and technical issues for the stellarator and is certain to advance the knowledge of plasma physics as applied to fusion-relevant conditions. It is likely to contribute significantly to the optimization of tokamaks and perhaps to other confinement concepts.

Technical Risk. There is significant risk involved in addressing this goal in the ITER era. There is little doubt that a stellarator configuration can confine plasma at the parameters necessary for fusion burn. However, the advantage of the stellarator to do this in stationary conditions without continuous external drive comes at the cost of a non-axisymmetric magnetic configuration, leading to poorer fast and thermal particle and energy confinement. The concept of quasi-symmetry, with a strong theoretical and emerging experimental basis, seeks to minimize this cost. The element of utmost importance for all stellarators, whether quasi-symmetric or of another type, is the validation of a design principle for magnetic coils which encompasses both manufacturing realities (including cost) and provision of a robust magnetic configuration for all phases of operation of the stellarator. The absence of such a principle at this time implies a significant element of risk to the timely completion of the goal. The use of quasi-symmetry as a design principle for improving confinement also carries some risk. There is no database of experimental data on the response of anomalous (non-collisional) transport in any of the possible symmetries at this time. Therefore, the degree to which the configuration must be symmetric will likely not be demonstrated experimentally before construction of a next-step device to validate the concept. This point is also somewhat tied to the issue of coil complexity and accuracy. The international stellarator program is very strong in the area of steady-state operation, but some experimental basis will be needed for assuring that the results obtained in stellarators without quasi-symmetry can be transferred to those with quasi-symmetry. Progress toward the stellarator goal can be made in the near term with concerted theoretical and modeling efforts, with validation by experiments as available. However, achievement of this goal in the ITER era, even for a single quasi-symmetric configuration, will require resources at or beyond the level of the present U.S. tokamak program.

To have a realistic chance of achieving the goal, a choice will need to be made among the various symmetries possible for stellarators. Therefore, it may be that a definitive statement about the best approach to stellarator optimization or a definitive statement ruling out stellarators as a viable approach to fusion energy will not be possible in the ITER era. [It is worth noting that the shutdown of NCSX construction during the panel's deliberations has had a severe impact on the U.S. stellarator program. That community has not had time to come to consensus on a new roadmap for QS stellarator development.]

- **6b. High Priority Issues for the Stellarator (Tier 1 Issues).**

1. **Simpler Coil Systems.** *Can we find ways to reduce the fabrication risk and cost of optimized high performance stellarator devices?* Complexity of coil shape, along with the fabrication and assembly tolerances needed to meet the physics requirements, are major cost drivers. Improved understanding of 3D field effects and use of error-correction coils to relax constraints may lead to simpler, cheaper coil systems.
2. **High Performance Integration.** Can improvements observed in smaller experiments be carried over to a high performance level device and what are its required attributes?

There is a need to demonstrate that benefits of quasi-symmetric configurations can be realized in high-beta, low collisionality plasmas with good overall energy confinement.

3. **Predictive Capability.** Can a predictive capability for quasi-symmetric systems be developed by building upon the work in the tokamak program coupled with a smaller experimental database? The reduced experimental database for stellarators, especially for quasi-symmetric configurations, places a greater burden on validated predictive models in the design process for a burning plasma experiment.
4. **Power Handling.** *Can divertor solutions be found for a 3D quasi-symmetric stellarator system compatible with high performance operation?* The complex 3D boundary of the stellarator challenges divertor design to achieve low peak heat flux with adequate particle control and ash exhaust. Higher operating density in the stellarator may be helpful.

- **6c. Scientific Benefit of Stellarator Research.** Addressing the high priority issues related to the stellarator ITER-era goal will improve understanding of toroidal fusion scientific and technological issues across a broad front, independent of whether the stellarator ultimately remains an attractive concept for fusion energy. Four areas of research are most visible:

— 3D Field Effects. The effect of magnetic asymmetry on confinement and stability is an issue with broad importance; for example, resistive wall modes in tokamaks and RFPs break axisymmetry and thus depend on three-dimensional confinement physics. Non-asymmetric fields have been used to control edge-localized modes (ELMs) in tokamaks and are planned for ITER: stellarator codes are now being used to aid design of new ELM control coils in tokamaks. 3D field errors appear to be key factor determining stability limits due to close coupling with plasma rotation.

— Transport. How magnetic symmetry alters transport due to electrostatic fluctuations and, especially, electron heat transport are key questions for fusion, but currently poorly understood for all concepts. Study of the role of flow-shear in stabilizing electrostatic turbulence and transport in QS stellarators will very likely improve understanding of similar transport in tokamaks. Energetic particle transport, an important component of quasi-symmetric stellarator research, also has direct application to other concepts through model validation activities.

— Power Handling and Particle Control. The unique 3D geometry of the stellarator edge plasma provides an important testing platform for understanding the boundary region of fusion plasmas. Research on parallel and perpendicular energy and particle transport, coupled with validation of edge-plasma simulation codes, will drive improvements to measurement and modeling capability that have broad applicability to the tokamak and other toroidal confinement concepts.

— Disruptions. An additional set of benefits to fusion science will come from tests of disruption control by means of externally imposed rotational transform. In particular the QS stellarator research program should reveal how much plasma current, and of what kind, can be allowed while maintaining the observed stellarator resistance to disruption. Such research might suggest methods for disruption control in other devices.

3.2.3 Finding 7: Spherical Torus

- **7a. ITER-Era Goal for the U.S. Spherical Torus Program.**

  Establish the ST knowledge base to be ready to construct a low aspect-ratio fusion component testing facility that provides high heat flux, neutron flux, and duty factor needed to inform the design of a demonstration fusion power plant.

  The ST goal largely aims to provide the groundwork for extending fusion development beyond the ITER mission rather than seeking primarily to achieve high-gain burning plasma conditions in the ST configuration. In addition to improved understanding of key ST physics issues, design of a reliable test facility for developing fusion components will also require a broad knowledge base in fusion technology.

  Importance and Relevance. The ITER-era goal is clear, well focused and tied tightly to the fusion energy roadmap, providing the fusion program with a valuable option for a component testing facility (CTF). The need for a nuclear testing facility has been well established through a number of planning exercises and is on the critical path to Demo, though such a mission has not been adopted officially by the U.S. Fusion Program. Since resistive losses in the non-superconducting toroidal field coils will be quite large, the ST may not be suitable for an electricity producing reactor, thus the CTF might be the primary long-range mission for the low-aspect ratio tokamak. It will be challenging to establish when the ITER-era goal has been met, since it depends on the accumulation of a sufficient knowledge base for confidence in taking the step to CTF. One might calibrate the effort required for that level of confidence by comparison with the ITER physics basis documentation. Thus, the ST experimental program would need to be tied to vigorous modeling and validation process. While general contributions to fusion science would be about average for programs of this scale, ST research can provide a substantial benefit to the tokamak program by advancing the science of tokamak confinement and its dependence on aspect ratio, broadening the available parameter space and allowing for wider testing of models.

  Technical Risk. The goal requires significant extrapolation in plasma performance and the level of knowledge required. In some areas there is a sound technical basis for extrapolation but in many others the science is incomplete or untested. Examples include solenoid-free plasma current ramp-up, electron transport and plasma-wall interactions. Achieving the ST goal is likely to require very significant resources thus the FES strategic planning process will need to consider the following critical decision points:

  1. After sufficient research on STs and Tokamaks, the optimal aspect ratio for CTF should be assessed.
  2. Well in advance of a decision to proceed with a CTF, the program roadmap would need to expand basic research in materials and engineering sciences to accompany development of the ST physics basis.

  In following this roadmap, schedule risks could be incurred if all necessary components of this research are not pursued.

- **7b. High Priority Issues for the Spherical Torus (Tier 1 Issues)**

  1. **Start-Up and Ramp-Up.** Is it possible to start-up and ramp-up the plasma current to multi-MA levels using non-inductive current drive with minimal or no central solenoid? The tight aspect ratio of the ST leaves little room for an ohmic transformer. Non-inductive ramp-up by either NBCD or RFCD may be challenging due to a variety of effects at low field and current.

  2. **First-Wall Heat Flux.** What strategies can be employed for handling normal and off-normal heat flux consistent with core and scrape-off-layer operating conditions? The relative small major radius of the divertor in an ST increases the peak heat flux, and the compatibility of radiative solutions with requirements for non-inductive current drive are unknown. While alternative divertor geometries have been proposed for the ST, experimental validation is required.

  3. **Electron Transport.** *What governs electron transport at low-aspect ratio and low collisionality? Is it adequate to meet the goal?* ST experiments find that energy transport is dominated by electron transport, which is considerably higher than that observed in conventional tokamaks. It will be essential to determine the underlying physics and scaling to be confident of achieving the ITER-era goal.

  4. **Magnets.** *Can we develop reliable center-post magnets and current feeds to operate reliably under substantial fluence of fusion neutrons?* Stresses on the center-post magnets in an ST are expected to be up to an order of magnitude higher than present experiments and a high-current single-turn demountable design remains to be tested. Engineering design studies and tests will be required to demonstrate necessary reliability.

- **7c. Scientific Benefit of Spherical Torus Research.** Addressing the high priority issues related to the Spherical Torus goal will provide a number of benefits to fusion energy science and technology. Important areas of research with broad benefit include:

  — Macroscopic Stability. The research program entailed in reaching this goal would extend the knowledge of tokamak plasma stability to higher β, higher elongation, and lower aspect ratio. For example, reaching local β of order unity with fusion level temperatures and densities, would contribute critical knowledge to other alternative concepts with potentials for similar β values, such as the RFP and CT. Also, high-beta discharges at low-aspect ratio bring different mode couplings into play, which complements active mode control research (e.g., RWM, NTM, and ELM control) on standard-aspect ratio tokamaks.

  — Transport. Low aspect ratio highlights the importance of neoclassical transport effects for both thermal and energetic particles. The transport in both ions and electrons scales differently in the ST than in standard-aspect-ratio tokamaks. Transport studies and comparison with gyrokinetic codes can provide important tests to theory that have broad implications to many confinement concepts. Experiments show that many unstable modes can be excited by energetic particles such as NBI and, ion future devices alpha particles, thus providing a useful platform for studying fast-ion transport.

  — Boundary Physics. Aiming to design an ST-based CTF strongly motivates boundary research because of the expected high heat flux at the divertor targets resulting from the small relative major radius of the standard ST divertor. Understanding scrape-off-layer transport and demonstrating the compatibility of radiative dissipation and other techniques to reduce peak heat loads and surface erosion has broad application to other toroidal magnetic confinement concepts.

— Non-Inductive Current Drive. All toroidal alternates require an efficient means to form and sustain the magnetic configuration; the challenge is large in the ST due to low magnetic field and limited space for an ohmic transformer. DC helicity injection, neutral beam, and RF current drive are being investigated for startup, rampup, and sustainment in the ST and in the RFP. Development of these technologies can benefit both other alternates and standard-aspect-ratio tokamaks.

- **7d. Additional Comment on the Relationship Between the Spherical Torus and the Tokamak.** The panel spent some time seeking to understand the differences between the spherical torus and the higher-aspect-ratio tokamak as two separate toroidal confinement concepts with distinct physics issues. Fundamentally, the ST is a low-aspect-ratio tokamak. Clearly, low aspect ratio and low toroidal field on axis change plasma stability and operating limits and are likely to affect transport processes, as evidenced by increased electron transport in the ST. However, a growing international database and improved theoretical understanding point to the great commonality of underlying physics between the spherical torus and the tokamak, thus blurring the distinction between these devices. Planned upgrades to the two largest ST facilities, NSTX and MAST, will increase their aspect ratio, bringing them closer to the operating space of other tokamaks.

3.2.4 Finding 8: Reversed Field Pinch

- **8a. The ITER-Era Goal for the RFP Program.**

  Establish the basis for a burning plasma experiment by developing an attractive self-consistent integrated scenario: favorable confinement in a sustained high beta plasma with resistive wall stabilization.

  This goal aims for the eventual demonstration of a burning plasma reversed field pinch that would provide the basis for possible construction of an RFP Demo fusion reactor. Because magnetic self-organization plays a fundamental role in the operation and performance of the RFP, achieving this goal also informs fusion science independent of its potential as a fusion energy concept.

— Importance and Relevance. This goal is the critical milestone for the RFP approach to fusion energy. Achieving the goal would establish the possibility of a low-external-field approach to magnetic fusion with reduced technological demands on magnets and the potential for high-power density designs limited only by wall loading. Techniques and physics understanding of current sustainment developed in the RFP have potential transfers to other configurations such as the spheromak. The RFP has been used as a test bed for increased understanding of magnetic self-organization and dynamo effects, which are of value to other configurations, and to other fields of science such as astrophysics.

— Technical Risk. There is a significant risk in achieving the RFP ITER Era goal. Comparison of results from present well-diagnosed Proof of Principle experiments and nonlinear resistive MHD modeling provide some confidence that the understanding exists to expect success in attaining the ITER Era goal. Many elements of the theoretical understanding are sound, but others aspects (confinement scaling during current sustainment) are incomplete. Risk could be mitigated by a step-wise approach involving research on current experiments (with Lundquist number, $Lu \sim 10^6 - 10^7$), proceeding to an advanced Proof of Principle experiment ($Lu \sim 10^7 - 10^8$) and finally to a Performance Extension experiment ($Lu \sim 10^8 - 10^9$) as results warranted. The resources needed to achieve the first stage with an advanced Proof of Principle facility are expected to require a moderate increase in the US RFP budget. The second stage Performance Extension facility will require a large increase in the existing RFP annual budget.

- **8b. High Priority Issues for the Reversed-Field Pinch (Tier 1 Issues).**

  1. **Confinement and Transport.** *What governs transport when magnetic fluctuations are reduced and how does energy confinement depend upon Lundquist number?* Current profile control reduces magnetic fluctuations and energy transport. The transport mechanism that limits confinement in this regime and the confinement scaling are not known. Both Lundquist number (τ_R / τ_A) and normalized gyroradius, $\rho*$, differ by more than an order of magnitude between present experiments and burning plasma conditions.
  2. **Current Sustainment.** *Can Oscillating Field Current Drive sustain the RFP configuration with high efficiency as compared to long-pulse induction?* A steady-state RFP will require external current drive since neoclassical bootstrap current is small due to low toroidal field. RF and Neutral Beam current drive efficiency are also expected to be low. Oscillating Field Current Drive is proposed, but the theoretical efficiency needs to be verified at high plasma temperature.
  3. **Integration.** *Is good confinement compatible with current sustainment at high Lundquist number?* Current drive often creates tearing-unstable current profiles in the RFP that reduce confinement. The detrimental effect of inductive current drive (e.g., Oscillating Field Current Drive) on transport may be reduced at higher magnetic field and temperature (i.e., Lundquist number), but this too must be verified by experiment.

- **8c. Scientific Benefit of Reversed Field Pinch Research.** Aiming to achieve the ITER-era goal for the RFP will contribute significantly to fusion energy sciences and broader scientific disciplines. The unique role of magnetic turbulence and the plasma dynamo in the RFP, along with its low safety factor ($q < 1$), bring out new aspects to many branches of fusion sciences, testing fundamental understanding.

  — Transport. Current profile control in the RFP provides a powerful tool to study transport in regimes controlled by either magnetic or electrostatic turbulence. Understanding transport and its scaling in such plasmas, for which magnetic shear and the gyroradius are relatively large, extends and tests the knowledge base acquired from tokamaks (at high externally applied field). Studies of current transport by the plasma dynamo can be useful to understanding current profile evolution in other toroidal configurations where internal MHD modes may be present.

  — MHD Stability and Beta Limits. The RFP is susceptible to multiple resistive wall instabilities even at zero beta. Thus, RFP research has and will continue to develop feedback techniques for multiple mode stabilization directly applicable to other configurations. On the other hand, all RFP experiments operate at high beta, and recently beta values have been achieved that exceed theoretical MHD stability limits for localized interchange and global tearing modes. Work to determine the beta limit and compare against theory will be valuable for other high-beta concepts.

— Magnetic Self-Organization. Spontaneous reversal of the toroidal field in the RFP represents a clear case of magnetic self-organization, similar to the formation of the spheromak and field-reversed compact torus (FRC). Thus, studies of reconnection, dynamo alteration of the current density profile, momentum transport, reconnection heating of ions, transport from magnetic stochasticity, and magnetic helicity transport in the RFP are particularly relevant to these other configurations. These effects are strongly related to each other, so that understanding the individual phenomena sums to a general understanding of magnetic self-organization that is also observed in the tokamak under certain conditions.

— Astrophysics. Through magnetic self-organization, RFP physics has strong links to related phenomena in space and astrophysical plasmas. Through funded collaborations with plasma astrophysicists as part of the NSF/DOE Physics Frontier Center for Magnetic Self-Organization in Laboratory and Astrophysical Plasmas (CMSO), RFP researchers have been playing a central role in applying understanding gained in the laboratory to astrophysics (as well as applying physics learned through astrophysical studies to the RFP).

3.2.5 Finding 9: Compact Torus

- **9a. The ITER-Era Goal for the CT is.**

To demonstrate that a compact toroid (CT) with simply connected vessel can achieve stable, sustained or long-pulse plasmas at kilovolt temperatures, with favorable confinement scaling to proceed to a pre-burning CT plasma experiment.

The CT goal aims to move from present concept exploration experiments to an experiment of sufficient scale or plasma parameters to provide a solid scientific basis (performance and scaling) for continuing CT research with the eventual goal of fusion energy. The CT program involves two partially related, but still significantly different concepts – the spheromak and the field reversed configuration (FRC).

— Importance and Relevance. The ITER era goal for the CT is clear and on a direct path towards fusion energy. Achieving this goal would significantly change the outlook for the CT as it would demonstrate major progress in MHD stability, confinement, and current drive for these configurations, well beyond levels achieved in previous experiments. Reaching the goal likely would not have much impact on the other concepts. Unique CT problems involve formation and sustainment. Initial formation is not a major problem for most of the other fusion concepts and there is limited overlap in sustainment methods. In terms of improving the overall knowledge of fusion science, the fact that CT's and the RFP operate with a low toroidal field (in contrast to tokamaks and stellarators), suggests that they will indeed make valuable contributions in this area, provided that there are adequate diagnostics.

— Technical Risk. The ITER-era goal is ambitious for either of the CT's — the spheromak or the FRC. A large extrapolation in scientific parameters is needed for either of the CT's to reach their goal. Due to limited resources for these concept exploration experiments, relatively little data is available and the data that is available is far from the parameters characterizing the goal. There are also substantial gaps in theory, due mainly to the complexity of the hybrid-resistive MHD models governing the behavior of these plasmas. There is yet to be found a convincing theory or simulation that shows that an FRC can be MHD stable at large s, or can achieve much smaller values of χ at modest s. Similarly, for the spheromak there is no convincing theory or numerical simulation that demonstrates how to simultaneously achieve good sustainment with good transport.

Although it would require a major increase in CT funding to significantly advance CT research, this would have only a modest impact on the overall fusion budget. A combination of improved diagnostics, theory and simulation would be needed to show how either concept can solve its difficult physics problems before making a large new step.

- **9b. High Priority Issues for the Compact Torus (Tier 1 and 2 Issues).**

FRC Issues

1. **Stability.** *Is global stability possible at large-s ($a/\langle\rho_i\rangle$) in low collisionality FRCs?* Advancing the FRC concept rests heavily on the stability issue with respect to low-n MHD modes, which are predicted to be ideal MHD-interchange unstable in an FRC. Present experiments have shown stability with low s (s < 5), thought to be due to kinetic stabilization, but the ITER-era goal requires substantially larger s (~10).
2. **Transport.** *What governs energy transport and can it be reduced at high temperature?* Little is known about confinement in FRCs other than what is implied by the flux and particle lifetime measurements. Reducing transport is key to attaining the ITER-era goals, in terms of both confinement and sustainment.
3. **Sustainment.** *Is energy-efficient sustainment possible at large-s and is it compatible with good confinement?* CTs have a unique requirement for creating and maintaining the poloidal flux without an inductive transformer. Rotating Magnetic Field sustainment is the only demonstrated method at present, but its efficiency is limited by anomalous resistivity. Ways to reduce the resistivity need to be developed.

Spheromak Issues

1. **Sustainment.** *Can efficient time-averaged current drive be maintained simultaneously with good confinement?* Experiment, theory and resistive MHD simulations find that current drive by electrostatic helicity injection opens magnetic surfaces through magnetic fluctuations, resulting in rapid energy losses. Simultaneous current drive and good confinement will thus require a new approach to helicity injection or the use of other current-drive technologies.
2. **Formation.** *Can formation and buildup techniques be developed to achieve fusion relevant magnetic fields?* Electrostatic helicity injection has successfully generated MA spheromaks in SSPX, but significant energy is required and the amplification of the bias poloidal flux has been in the range of 2-6. Both these need an order of magnitude improvement for burning-plasma experiments.

3. **Transport.** *What mechanisms govern transport and confinement in low collisionality spheromak plasmas?* Energy confinement experiments in slowly-decaying plasmas yield thermal conductivities in the tokamak L-mode range, although there is a very limited data base. The electron transport mechanism has not been studied, although it is found that the best confinement occurs when the q-profile does not cross low-order resonances.

- **9c. Scientific Benefit of Compact Torus Research.** The unique features of the strongly self-organized Field-Reversed and Spheromak configurations often challenge application of results to other toroidal configurations such as the tokamak or stellarator. In other cases, such as for the FRC and Magnetized Target Fusion, or Coaxial Helicity Injection for ST startup and CT fueling of tokamaks, the connection to CT research is more immediate. Areas where CT research benefits other areas of fusion science include:

— Reconnection Physics and Resistive MHD. Formation and buildup of both the spheromak and the FRC are governed by magnetic reconnection and other resistive MHD physics. Events resulting from these are ubiquitous in both laboratory and space/astrophysical plasmas. Both concepts can therefore play an important role in validating 3D resistive MHD codes, which are now being applied to problems such as disruption mitigation and ELM control in tokamaks. Reconnection studies on the spheromak complements similar work on the RFP, and both devices have contributed to the NSF/DOE Center for Magnetic Self-Organization in Laboratory and Astrophysical plasmas.

— MHD Stability. Overall MHD stability of the FRC is not well understood but is thought to be largely due to finite-gyroradius effects, especially at low-s. At higher s, a minority population of fast ions or plasma flow may stabilize low-n modes. The FRC thus provides a test bed for extending MHD analysis into this new regime.

— Helicity Injection and Transport. The spheromak relies on DC helicity injection to build toroidal current, as does the RFP. Both concepts seek to sustain their discharges with minimal impact on confinement through current profile control to adjust helicity transport by magnetic fluctuations. More recently, DC helicity injection has been used for ST startup in HIT-II and NSTX. These experiments are using common simulation tools such as the NIMROD code to analyze results and understand implications for next step experiments and there is close connection between the spheromak and ST communities on this subject.

CHAPTER 4

THE STELLARATOR

4.1. CONCEPT DESCRIPTION

The stellarator is a fully three-dimensional toroidal magnetic confinement configuration that has demonstrated steady progress towards the fusion energy goal over the past five decades. Stellarators (and their closely related variants the torsatron and the heliotron) have much in common with tokamaks, including the plasma parameters and confinement that have been attained.

The distinguishing feature of the stellarator is its use of external coils to produce both the toroidal and poloidal confining magnetic fields (Fig. 4-1). This feature leads to three key benefits. Because the magnetic field and geometry of the plasma are largely determined by the external coils, stellarators can be optimized to operate with little or no plasma current and thus need no OH transformer or external current drive. This makes them intrinsically compatible with steady-state operation and minimizes a major source of magnetohydrodynamic (MHD) instabilities, which may mitigate the occurrence or impact of "off-normal" events such as ELMs or disruptions. Typically, disruptions are not observed in stellarators, even in experiments with significant plasma current or near operating limits. These results encourage the expectation that disruptions will not occur in stellarator burning plasmas. Due to these characteristics, stellarators may offer a more direct path to a robust, steady-state fusion device.

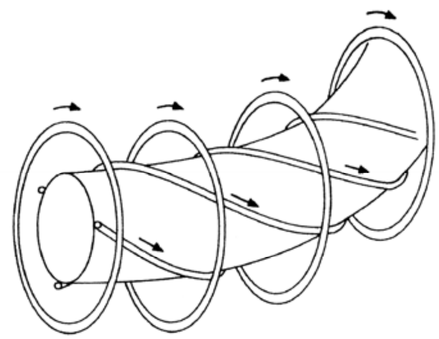


Fig. 4-1. Basic coil configuration for a simple stellarator showing toroidal field coils and helical windings to provide rotational transform.

The dependence on external coils offers clear benefits to the stellarator as a fusion reactor, but it also introduces a number of technical and scientific challenges. While the magnetic geometry of most toroidal confinement systems is ideally symmetric about the main toroidal axis, the magnetic field of the stellarator necessarily breaks this axial symmetry. Thus, one critical aspect of stellarator research is the physics of 3-D plasma confinement under high performance conditions. The 3-D properties of stellarator magnetic geometry have particular implications for confinement of alpha particles, impurity accumulation in the plasma, as well as complications in the design and operation of divertors.

To address this issue, the US stellarator program has focused on the use of quasi-symmetry: the careful tailoring of the magnetic field strength to produce approximate symmetry with respect to an appropriate angular coordinate (Fig. 4-2). The key advantage of quasi-symmetric optimization is that it is predicted to provide good neoclassical particle (especially of alpha particles) and energy confinement, while reducing viscous flow damping to levels comparable to that in tokamaks.

A second critical aspect of the 3-D stellarator magnetic field is the requirement for non-planar magnetic field coils, or coil arrays (Fig. 4-3). The design and construction of magnetic field systems for stellarators – particularly, the use of modular coil systems – has often tested the limits of both manufacturing technologies and metrology. A compelling objective is therefore to find magnetic field coil systems and supporting structures that can simplify the construction of stellarator devices.

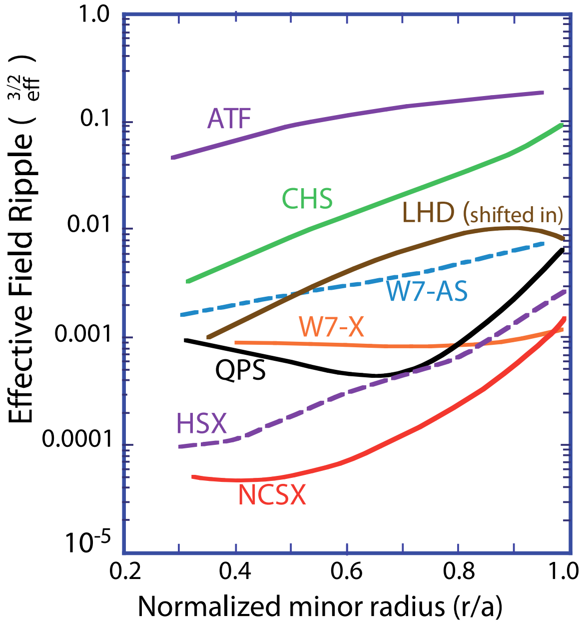


Fig. 4-2: Effective field ripple vs. plasma minor radius for a number of stellarators showing lower ripple for quasi-symmetric stellarators such as QPS, HSX, and NCSX.

It is noted that as this report was being written, the decision was made to terminate the National Compact Stellarator Experiment (NCSX). The NCSX project demonstrated that compact quasi-axisymmetric stellarators can be designed with attractive physics characteristics and that appropriate coils can be fabricated. The ARIES-CS study investigated the use of quasi-symmetric configurations for fusion energy production and found that the configuration was attractive and led to systems similar in size and other characteristics to advanced tokamaks. The US stellarator program remains committed to the development of the quasi-symmetric stellarator approach.

The existing U.S. stellarator program will continue to make important contributions to stellarator development in both theory and experiment. The code V3FIT, developed for the construction of three-dimensional equilibria, has application in both stellarator and tokamak research. The Helically Symmetric Experiment (HSX) and Compact Toroidal Hybrid (CTH) devices are both Concept Exploration devices that seek to extend the understanding of the stellarator fusion concept. HSX is exploring the physics of quasi-symmetry while CTH is exploring disruption occurrence, 3-D equilibrium reconstruction, and the correction of error fields.

In the world-wide stellarator program, significant advances are being made towards the fusion energy goal. The leading research activities are in Japan and Germany. The Large Helical Device (LHD) in Japan, a superconducting torsatron/heliotron with a divertor that performs a broad range of integrated research. The Wendelstein 7-X (W7-X) stellarator under construction in Germany is a superconducting, optimized, modular stellarator with a divertor. The international stellarator community also uses several smaller stellarators. However, none of the experiments outside the US are investigating quasi-symmetric stellarators.

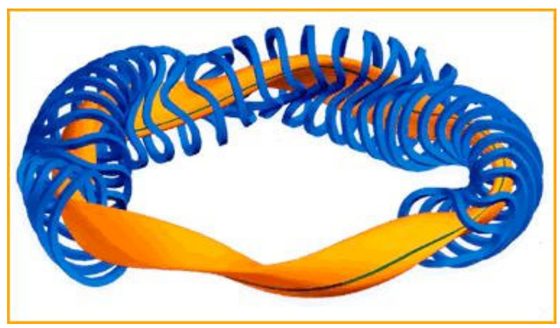


Fig. 4-3. W7-AS modular non-planar coils and plasma.

This broad portfolio of significant domestic and international research activity supports the central long-range mission of stellarator research:

> ***To achieve sufficient scientific understanding and plasma conditions to justify designing a fusion reactor based on a fully steady-state, passively stable stellarator.***

4.2. GOAL FOR THE ITER-ERA

Develop and validate the scientific understanding necessary to assess the feasibility of a burning plasma experiment based on the quasi-symmetric (QS) stellarator.

The stellarator ITER-era goal emphasizes understanding — knowledge — rather than specific facilities or performance. The targeted knowledge is that necessary to inform the design of a high-Q burning plasma experiment based on quasi-symmetric stellarator confinement. Such an experiment would provide a crucial step in the longer range (post ITER-era) assessment of the ultimate value of the stellarator as a steady-state, reliable fusion reactor.

The high level of scientific understanding required to assess the feasibility of a stellarator burning-plasma experiment will entail a substantial research program in fusion science and technology. The core research effort will be performed in collaboration with the large international stellarator program, involving both existing and new stellarator and tokamak facilities. Central elements of this research program include:

- Theory and modeling of confinement and stability;
- Experimental validation;
- Engineering studies focusing on simplified coil structures and divertor design;
- Systematic exploitation of scientific advances in the neighboring program of tokamak research, including ITER.

Quasi-symmetry is an essential element of the goal, not only because of US theoretical and experiment strength in this area, but also because theory predicts several key advantages for quasi-symmetry. The improved particle orbits resulting from quasi-symmetry should enhance alpha-particle confinement and reduce thermal transport. Furthermore it appears that quasi-symmetry may allow

- Simultaneous optimization of confinement and stability;
- Impurity control;
- Lower flow damping, for shielding error fields and reducing anomalous transport;
- Lower aspect ratio.

4.3. SCIENTIFIC AND TECHNICAL QUESTIONS

The following scientific and technical issues and questions must be resolved in order to achieve the ITER-era goal for Stellarator research. These have been prioritized into three tiers as described in Chapter 3.

Tier 1:

- **Simpler Coil Systems.** Although in principle simpler to operate than tokamaks, stellarators are more complex to build. C*an we find ways to reduce the fabrication risk and cost of optimized high performance stellarator devices?* Significant cost drivers are the fabrication and assembly tolerances specified for the non-planar coils, and the complexity of the coil shape. The use of trim and error-correcting coils to relax design and assembly constraints on the main coils, and the appropriate allocation of the tolerance budget, needs further investigation. Coil shapes are driven by the physics and engineering optimization targets. Recent experiments and theory, such as investigations of island shielding and MHD stability, suggest the possibility that some design constraints can be relaxed. There is also a need to investigate different coil topologies, with the objective of designs that are easier to fabricate and assemble. Magnetic materials might be used to reduce the complexity of the main field coils. More broadly, engineering needs to play a prominent role in the physics design.

- **Integrated High Performance of Quasi-Symmetric Optimized Stellarators.** The stellarator program needs to demonstrate the ability of quasi-symmetric configurations to confine high-beta, low collisionality, plasmas with comparable ion and electron temperatures. *Can this be accomplished without current drive or active stabilization, and without danger of disruptions?* To satisfy the ITER-era goal of determining the feasibility of a burning plasma experiment we need to understand the scaling of confinement for a high performance quasi-symmetric stellarator.
- **Confinement Predictability.** Development of a validated predictive capability is necessary for extrapolation to a fusion energy system for any concept. With a smaller database than tokamaks, the stellarator must improve in this respect in order to achieve its ITER-era goal. Quasi-symmetry provides a strong connection to full symmetry. *To what degree can the tokamak database be used to develop predictive understanding of quasi-symmetric stellarator performance?*
- **Divertors.** A key element for any fusion device is an effective divertor. Control of neutral recycling and impurities at high power levels, is critical. Divertor design is more complicated in a 3D system, due to the helical shape of the outer flux surfaces. *What is the optimal divertor design in quasi-symmetric 3D system?* The higher density limits in stellarators may help to reduce the edge plasma temperature and increase edge radiative cooling. Stellarator design flexibility allows the edge magnetic topology to be designed. Flux expansion might be used to reduce divertor power loading.

Tier 2:

- **Operational Limits.** Stellarator experiments demonstrate that the operating limits for plasma density and beta are set by different nonlinear responses in stellarators than in tokamaks. In particular, neither density nor beta is typically limited by MHD instabilities or disruptions in stellarators. *What sets the operational limits on density and beta in a quasi-symmetric stellarator?* There is a need to determine mechanisms limiting confinement at high beta, and to develop and validate predictive models for future design.
- **Impurity and Fusion Ash Accumulation.** Stellarator reactors are expected to operate in regimes that may lead to neoclassical impurity accumulation. Impurity accumulation has been observed experimentally in stellarators in some regimes, especially with neutral beam injection. However, high confinement regimes also exist where the core remains impurity-free and the plasma is measured to expel impurities, such as in the high-density H-mode observed in W-7AS. Quasi-symmetry can, theoretically, reduce the 3D effect sufficiently to allow temperature screening to dominate, expelling impurities as in tokamaks. *What determines impurity accumulation in stellarators and what are the design implications?*
- **Anomalous Transport Reduction.** In optimized stellarators, neoclassical transport is reduced and turbulence-driven transport (so called “anomalous” transport) becomes dominant, as found in tokamaks. The turbulent transport mechanisms are predicted to be substantially the same in both tokamaks and stellarators, but have been more extensively investigated in tokamaks. *Can*

quasi-symmetry and optimized 3D shaping reduce turbulent transport and increase confinement? In quasi-symmetric stellarators flow damping has been observed to be reduced in the symmetric direction, which may allow stronger flow-shear and stronger reduction of anomalous transport, as in tokamaks. Low ripple quasi-symmetric stellarators may have lower turbulence and transport levels than other stellarators. In addition, stellarators can be shaped to reduce the calculated turbulence growth-rate, such as by choice of q-value and magnetic shear. However neither the optimum turbulent transport level in quasi-symmetric stellarators nor the maximum acceptable residual magnetic ripple are fully understood.

Tier 3:

- **Energetic Particle Instabilities.** Alfvenic instabilities and energetic-particle modes (EPMs) are frequently observed in high-performance tokamak regimes, and have also been studied in a number of stellarators. *Can sufficient understanding of these instabilities and transport be developed in quasi-symmetric stellarator configurations so that their effects can be included in optimization of the configuration and operating point?* These instabilities can produce fast-ion transport across the flux surfaces and loss from the plasma. Projected stellarator burning plasmas optimize at high density, reducing the fast-alpha thermalization time and the drive for Alfvenic instabilities and EPMs. The design of the operating point for a burning plasmas ensure stability to alpha-driven modes.
- **Disruptions.** Disruptions in low-current stellarators are generally not observed, even when operated at betas above the ideal stability limits. Disruptions also were not observed in stellarators with large ohmic currents provided a rotational transform of ~0.15 was produced by the coils. However, some quasi-symmetric stellarators, like NCSX, would have significant bootstrap current at high beta. An issue for quasi-symmetric devices is: *How much transform can be usefully generated by the bootstrap current before disruption-like effects begin to appear?* For quasi-axisymmetric configurations, this will determine the minimum level of three-dimensional shaping that must be added to an axisymmetric tokamak to stabilize disruptions.
- **ELM-Free High Performance.** There is a need to determine the significance of the ELMs observed in stellarators. Stellarators can operate in high-performance, high confinement discharges both with and without ELMs. *What are the conditions necessary for ELM suppression in stellarators and how do ELMs impact the ITER-era goal?* In LHD, the ELMs are observed to be due to instabilities on edge resonant surfaces with $q \leq 1$. The significance of the observed ELMs has not been assessed. Tokamak experiments have shown ELM suppression through ergodization of the edge — a strategy that might be especially straightforward to implement in stellarator geometry.
- **Profile Sensitivity of Operational Limits.** Stellarators can operate with a wide variety of temperature and density profiles without requiring detailed control. *Can quasi-symmetric stellarators operate in high performance regimes without detailed profile control?* Sensitivity of operating limits (density and beta) to profiles has not been systematically studied, but could arise through non-linear dependencies of the plasma equilibrium and transport on the profiles. In a burning plasma, this could be affected by the alpha heating profile and the particle fueling profile. There is a need to test high performance operating regimes for sensitivity of the operating limits to the heating and fueling techniques.

- **Superconducting Stellarator Coils.** High T_c superconductors (HTS) could have a large potential impact on the projection of stellarators to burning plasmas. *Can this technology be applied to advantage for stellarator coils?* Their less constrained radius of curvature can accommodate bends in modular coils, they might be easier to manufacture, and their higher current density permits smaller plasma/coil separation. Some of these benefits have been recognized in the previous discussion of the simpler coil systems. Commercial development is predicted to reduce costs by the ITER-era. It is possible that bulk HTS magnetic elements could be used to provide stellarator shaping without the need for three-dimensional coils.

4.4. FACILITIES AND GAPS

4.4.1. Stellarator Facilities and Analysis Capabilities

Stellarator research has been ongoing worldwide for the last 50 years, with major developments in the last decade. Stellarators are currently being investigated at the performance extension scale in the Japanese and European fusion programs. These programs have (or are constructing) large superconducting long pulse experiments (LHD and W7-X). Neither of these programs is investigating quasi-symmetry or compact configurations. The US has extensive collaborations with both the Japanese and European stellarator programs. With the cancellation of the NCSX project, the US fusion program has two concept exploration experiments, HSX and CTH.

4.4.1.1. Stellarator Experiments Operating or Under Construction

- **HSX (University of Wisconsin, US).** Quasi-helically symmetric concept exploration experiment (R=1 m, a=0.12 m, B=1 T, P=0.3 MW ECH); focus on improvements in neoclassical and anomalous transport with quasi-symmetry in hot electron (2.5 keV) collisionless plasmas. Presently HSX is the only device worldwide investigating quasi-symmetry.
- **CTH (Auburn University, US).** Low aspect ratio concept-exploration torsatron with ohmic heating (R=0.75 m, a=0.28 m, $B \leq 0.5$ T); focus on disruption avoidance, 3D equilibrium reconstruction and trimming of error fields.
- **CNT (Columbia University, US)**. Small stellarator with ultra-simple planar coils; focus on non-neutral plasmas and basic plasma science.
- **LHD (NIFS, Japan).** Operating high-performance, large superconducting torsatron/heliotron with divertor; investigating a broad range of integrated research. R=3.9 m, a=0.6 m, B=3 T, P=22 MW NBI, 3 MW ICRF, 1 MW ECH. Has achieved volume averaged beta of 5%, T_i = 5 keV, and operation for 54 minutes.
- **Heliotron-J (Kyoto University, Japan).** Moderate sized helical axis Heliotron; focus on effects of configuration variation on confinement.
- **TU-Heliac (Tohoku University, Japan).** Small heliac; focuses on L-H transitions in driven rotating plasmas.
- **W7-X (IPP Greifswald, Germany).** High performance, large superconducting quasi-isodynamically optimized stellarator with divertor (R=5.5 m, a=0.5m, B=3T, P=10 MW ECH); under construction – estimate for completion 2014.

- **WEGA (IPP Greifswald, Germany).** Older small stellarator; investigating production of overdense plasmas with EBW heating.
- **H-1NF (Australian National University, Australia).** Flexible heliac; focus on RF heating and configurational studies.
- **TJ-II (CIEMAT, Spain).** Proof-of-principle scale flexible heliac.
- **L-2 (GPI, Russia).** Older high shear stellarator. Infrequent operation.

It should be noted that the W7-AS modular stellarator (Germany) and CHS torsatron (Japan), though shut down in recent years, are still producing analyses of results with high impact on today's stellarator research program.

A broad collection of stellarator design, analysis and modeling tools have been developed for 3D configurations. The US has played a prominent role in this development. With a focus on 2D quasi-symmetry there is also an opportunity to broaden models available by drawing upon some of the extensive body of work undertaken for the tokamak program, where appropriate.

4.4.1.2. Examples of 3D Modeling Tools Available for Design and Analysis

- VMEC, PIES, HINT, SIESTA, BETA: 3-D equilibrium codes
- STELLOPT: Multivariate optimized design of stellarators
- V3FIT: 3-D reconstruction
- Terpsichore & CAS3D: ideal stability
- GS2 and GENE: flux tube gyrokinetic codes.
- ORBIT: Monte Carlo code.
- FULL linear microstability code (electrostatic modes).
- M3D nonlinear MHD code
- EMC3/IRENE 3-D edge modeling code.
- COBRA and ANACONDA ballooning stability codes.
- STELLGAP and AE3D: Alfvén continuum gap structure and discrete Alfvén eigenmodes for general 3-D equilibria
- PENTA: Neoclassical moments method analysis for 3-D configurations; includes plasma flow
- DKES: Stellarator neoclassical transport coefficient matrix
- DELTA5D, GNET: Full-f and delta-f Monte Carlo analysis for thermal and energetic particle transport in 3-D
- BOOTSJ: Asymptotic, low collisionality bootstrap current

4.4.2. Stellarator Available Means, Gaps, and Initiatives

Tier 1:

- **Simpler Coil Systems.**
  — Available Means to Address the Issue. Stellarators have been constructed successfully using helical coils (LHD and other heliotrons and torsatrons), modular coils (IMS, W7-AS, and HSX), and 3D arrays of planar coils (CNT, TJ-II, H-1). To date, fully optimized

configurations have only been designed using modular coils, which appear to face stiffer engineering challenges than helical coils. Designs have been investigated using saddle coils combined with toroidal field coils, but have not been constructed. The US has unique capabilities for such studies, building upon the STELLOPT code suite developed for the design of NCSX, QPS, and ARIES-CS. This code uniquely provides the capability to optimize both the engineering characteristics of coils and the physics properties of the generated magnetic equilibrium. Since the completion of the engineering design of NCSX, activities in this area have ceased.

— Gap. New ideas need to be investigated and developed for simplifying the construction of 3D magnetic configurations. These include ways to simplify the physics design constraints, the coil engineering, and the fabrication process.

— New Programs/Initiatives Needed to Close the Gap. To make progress, the key aspects of engineering simplification must be understood and appropriate design metrics developed, including simplification of blankets and maintenance in a future fusion energy system. A systematic exploration of the engineering advantages for different coil topologies needs to be conducted in combination with assessing the impact of improved physics understanding on the optimization of the plasma shape. Several ideas have been proposed recently to simplify the engineering of stellarator configurations, including the use of magnetic materials and high temperature superconductors to provide shaping, and the systematic use of trim coils to reduce fabrication and assembly tolerances. Verifying the potential advantages of such approaches will ultimately require construction of a new stellarator facility.

- **Integrated High Performance of Quasi-Symmetric Optimized Stellarators.**

— Available Means to Address the Issue. LHD and W7-X are PE-class experiments, exploring the full spectrum of confinement physics and integrated performance in superconducting long-pulse configurations. LHD is operating and has achieved the highest performance stellarator plasmas, as seen in the table below. W7-X has been optimized for high beta with low neoclassical transport, good fast-ion orbit confinement using non-symmetric quasi-isodynamic principles. However, it has an aspect ratio A=11 and projects to reactors much larger than tokamak systems and ITER. HSX is the only stellarator investigating quasi-symmetry, and should continue investigating its fundamental properties.

— Gap. There is a world-wide gap in the ability to study the integrated plasma performance and confinement for any type of quasi-symmetry. In addition, there is a gap in the studying of reduced aspect ratio (A < 7) stellarators optimized for good orbit confinement, good thermal confinement, and high beta.

— New Programs/Initiatives Needed to Close the Gap. To achieve the ITER-era Goal, it will be necessary to have an intermediate class quasi-symmetric stellarator in order to provide an integrated evaluation of the quasi-symmetric approach to confining a high performance plasma.

- **Confinement Predictability.**

— Available Means to Address the Issue. Codes have been developed to examine individual effects, typically in the linear regime, such as linear ideal MHD stability, and linear micro-stability. Most of these models assume that the flux surfaces are simply connected, closed and nested. Many codes are available for calculating neoclassical transport and orbit confinement in the full 3D magnetic field. Several experiments have comprehensive diagnostic sets and can provide data for detailed comparison with predictive models, including LHD and TJ-II, and data is available from the earlier experiments CHS and W7-AS, however none of these experiments are fully optimized for good orbit confinement. HSX is studying electron transport at low beta in a quasi-symmetric configuration. W7-X will provide data at high beta in a quasi-isodynamic configuration. Due to the close relationship between the confinement physics of tokamaks and stellarators (particularly quasi-symmetric stellarators), the stellarator models can build upon tokamak models and understanding. In addition, stellarator models will be directly engaged and informed with the tokamak experiments using 3D magnetic perturbations and shaping. There are significant efforts starting in Japan and Europe on integrated modeling of stellarators, with opportunities for the US to collaborate building upon its tokamak and stellarator modeling expertise.

— Gap. Predictive models for stellarator equilibrium, stability, and confinement are more challenging due to the 3D geometry and the potential presence of magnetic islands and stochastic regions. To address this issue, non-linear models need to be developed, for both turbulent transport and MHD stability, and these need to be integrated together to understand the kinetic modifications of the 3D MHD equilibrium and stability. Finally, there is a world-wide gap in the ability to provide experimental data to validate models of equilibrium, confinement, and stability in quasi-symmetric configurations with significant beta and $T_i \sim T_e$.

— New Programs/Initiatives Needed to Close the Gap. An integrated modeling program for 3D magnetic configurations should be established, to develop appropriate non-linear models for both stellarators and other 3D configurations, such as perturbed tokamaks. This program must include experimental validation using data from existing experiments and new quasi-symmetric experiments assessing confinement at high beta and performance.

- **Divertors.**

— Available Means to Address the Issue. W7-AS and LHD have successfully studied the use of an edge island chain coupled to a divertor target structure, demonstrating impurity control and enhanced confinement (relative to ISS95 scaling). An island divertor will also be used on W7-X. LHD can also use a continuous helical divertor with a helical separatrix surrounding a stochastic flux region. Due to their long-pulse capabilities, both LHD and W7-X have very active divertor programs, and W7-X plans to study tungsten coated divertor tiles and PFCs. The IPP/Greifswald group has developed the EMC3/IRINI code to model 3D stellarator divertors. This code has been applied to W7-AS, LHD, and the DED on TEXTOR, giving reasonable agreement with experimental measurements. This code is being used to predict the W7-X divertor performance and adjust its design.

— Gap. The gap in this area is to adapt and investigate 3D divertors for a quasi-symmetric configuration, for lower aspect ratio stellarator designs, and at higher power flux levels.

— New Programs/Initiatives Needed to Close the Gap. Divertor designs for quasi-symmetric configurations must be developed and included in the configuration optimization process. The designs should build upon the internationally developed expertise and models, as well as new modeling capabilities. The designs must be validated on quasi-symmetric experiments, demonstrating the compatibility of a divertor design and high integrated performance.

Tier 2:

- **Operational Limits.**
  - — Available Means to Address the Issue. The beta limit has been most extensively explored on W7-AS, LHD, and earlier experiments, and LHD will continue studies to understand its beta limit. W7-X has been designed to maintain good flux surfaces to beta = 5% at high aspect ratio. When it operates, it will provide a test whether this improves the beta limit and whether this allows high confinement at high beta. The US has pioneered the development of design and analysis tools to optimize stellarator configurations for good flux surfaces and MHD stability at high beta, and developed lower aspect ratio quasi-symmetric designs with these characteristics (e.g. Aries-CS, NCSX). US collaborations with LHD and W7X should be strengthened to make use of our analysis tools, including equilibrium modeling and reconstruction, and improve the understanding of 3D confinement limits.
  - — Gap. No experiments exist or are planned to test whether quasi-symmetry can produce high confinement at high beta. In addition, no experimental facilities are available to test whether the beta limit is affected by the presence of bootstrap current, as will exist in quasi-symmetric configurations.
  - — New Programs/Initiatives Needed to Close the Gap. To close this gap and achieve the ITER-era Goal, it will be necessary to have an intermediate class quasi-symmetric stellarator to evaluate its operating limits at high performance.
- **Impurity and Fusion Ash Accumulation.**
  - — Available Means to Address the Issue. LHD and W7-AS have studied impurity accumulation extensively, and W7-X will continue these studies in a quasi-isodynamic configuration. These studies will provide data to validate models of impurity transport and divertor efficiency.
  - — Gap. The gap on this issue is the lack of any experiment that can test impurity transport physics for quasi-symmetric configurations, especially at low ion collisionality, where the electric field effects are predicted to be strongest.
  - — New Programs/Initiatives Needed to Close the Gap. An experimental test is needed of whether the residual ripple can be low enough in a quasi-symmetric configuration to allow temperature-screening or anomalous transport to sweep the impurities out of the plasma.
- **Anomalous Transport Reduction.**
  - — Available Means to Address the Issue. Non-linear turbulence simulation models are being developed for stellarators, in some cases based upon models originally developed and studied for tokamaks. LHD and W7-X will continue to study turbulent transport and flow-shear suppression of turbulence in non-quasi-symmetric configurations, which have large flow damping. HSX is studying flow-shear suppression of TEM turbulence in a quasi-symmetric configuration and has observed internal transport barriers. Comparisons between measured anomalous transport in stellarators and model predictions should be extended to test and validate the models.

— Gap. There is no experiment available to study the effect of quasi-symmetry on ion driven turbulence and on ion thermal transport in high beta plasmas, Whether quasi-symmetry increases the zonal flow amplitude or effectiveness in turbulence suppression under these conditions is an open question; HSX will examine flows in a hot-electron plasma. In addition, there is no experiment that can test whether 3D plasma shaping of a quasi-symmetric configuration can be used to further reduce turbulent transport, e.g. through the magnetic shear and curvature.

— New Programs/Initiatives Needed to Close the Gap. An intermediate class quasi-symmetric stellarator, capable of studying $T_i \sim T_e$ high beta plasmas is needed to test and assess our understanding of turbulent transport in quasi-symmetric configurations, and the ability access improved confinement.

Tier 3:

- **Energetic Particle Instability.**

  — Available Means to Address the Issue. The occurrence and mode structure of Alfvenic and other fast-ion instabilities has been studied on CHS, W-7AS, and LHD. HSX and H-1 are studying Alfvenic instabilities with energetic electrons. Data will be provided by W7-X when it operates. The CAS3D, STELLGAP and other codes have been used to calculate the linear mode structure and spectrum in three-dimensional equilibria, and found reasonable agreement with experiments.

  — Gap. No non-linear models exist for 3D geometry of the coupling of Alfvenic and fast ion instabilities to the fast-ion population, the instability threshold, and the self-consistent fast ion transport. When candidate models are available, they need to be validated against well documented experiments with varying fast-ion beta and significant thermal-ion beta, to test kinetic effects.

  — New Programs/Initiatives Needed to Close the Gap. Non-linear models of Alfvenic and fast ion instabilities and their coupling to fast ion confinement need to be developed for 3D magnetic configurations. These models then need to be validated against experimental data.

- **Disruptions.**

  — Available Means to Address the Issue. Disruptions are not generally observed in stellarators, even when approaching the observed maximum density or pressure. Studies of disruptions in early stellarators with large Ohmic currents found that iota=0.5 ($q = 2$) disruptions were suppressed when at least 15% of the magnetic rotational transform was generated by the external coils. W7-AS was able to produce disruptions at edge $q = 2$ due to a classical tearing instability reconnecting the entire plasma column, using strong Ohmic current ramps to produce unstable current profiles. CTH is studying the conditions for disruption with Ohmic plasma current, and will compare with theoretical stability calculations. LHD could search for disruptions at high beta with moderate neutral-beam driven current.

  — Gap. In the world program, there are no experiments planned or available to study whether disruptions might occur with significant bootstrap current, such as produced at high beta in quasi-symmetric configurations.

— New Programs/Initiatives Needed to Close the Gap. A high beta, quasi-symmetric stellarator with sufficient pulse length is needed to test the impact of significant bootstrap current on the disruptivity of stellarators. This experiment should assess the fraction of the magnetic transform that can be generated via the bootstrap current without risk of disruptions.

- **ELM-Free High Performance.**

— Available Means to Address the Issue. ELMs have been observed in some experiments on CHS, W7-AS, and LHD and studied. The projected impact of ELMs, if present, on a stellarator burning plasma should be estimated, based on these experiments. In addition, the data should be analyzed to attempt to identify why ELMs are absent in some regimes, and what the ELM mechanisms may be. These studies should also work with the RMP ELM-stabilization experiments on tokamaks to develop a common understanding.

— Gap. Impact of ELMs on stellarator performance is not understood.

— New Programs/Initiatives Needed to Close the Gap. Experiments should continue on LHD, W7-X, and other experiments to test ELM stabilization techniques and document significance. Stellarator design criteria should be developed to eliminate ELMs or make them inconsequential.

- **Profile Sensitivity of Operational Limits.**

— Available Means to Address the Issue. Experiments on LHD and W7-X (when operating) could be used to test the sensitivity of the density and beta limits to plasma profiles. LHD experiments have demonstrated that the maximum operating density can be increased using pellet fueling of the core, as also observed on tokamaks. W7-AS and CHS archived data could be analyzed to see if such a dependence was explored fortuitously. HSX can assess the sensitivity of the density limit in a quasi-symmetric configuration.

— Gap. There is no experiment that can test in a quasi-symmetric configuration the sensitivity of the operating limits on the plasma profiles and heating profile. Since quasi-symmetric configurations have some bootstrap current, this may couple the plasma profiles to the equilibrium, modifying operating limits.

— New Programs/Initiatives Needed to Close the Gap. A quasi-symmetric stellarator experiment is needed that can access high beta and good confinement to evaluate the sensitivity of the operating limits to the plasma and heating profiles.

- **Superconducting Magnets.**

— Available Means to Address the Issue. The advantages of high temperature superconducting coils for fusion applications were identified in the Priorities, Gaps and Opportunities panel report. Magnet design groups have conducted engineering studies as part of the NCSX and ARIES-CS design processes.

— Gap. No stellarators are presently using high temperature superconductors nor are any planning to at this time, despite the predicted benefits.

— New Programs/Initiatives Needed to Close the Gap. A program is needed to investigate how high-temperature superconductors can be applied to three-dimensional configurations, including analysis of design constraints, fabrication techniques, and appropriate R&D.

Novel methods to simplify stellarator design using HTS components should be explored and developed.

4.4.3. New Programs and Initiatives Needed to Close Gaps and Accomplish Goal

There is a need to understand how to reduce cost and complexity of stellarator construction, addressing the gap in the highest priority issue. Many of the needed tools are in existence or could be made available with modest investment. Engineering criteria need to be incorporated in the physics design process, including edge/diverter structures, component fabricability, assembly, and maintainability with appropriate tolerance levels. Technical lessons learned from the recent construction projects need to be factored into this analysis.

Some of the issues identified as needing resolution for the ITER-era goal can be resolved on the PE-class experiments in Europe and Japan. Increased collaboration within and between the stellarator and tokamak worldwide communities, is needed to bridge understanding between quasi-symmetric stellarators and tokamaks. Increased utilization/upgrades of existing experiments should be pursued as appropriate.

It is clear that to achieve the ITER-era goal a quasi-symmetric experiment of sufficient scale needs to be undertaken within this timeframe to demonstrate, in an integrated fashion, that the benefits of quasi-symmetry seen at the CE level can be extended to high performance, high beta plasmas. In particular, this device should be designed to address the gaps identified in the following issues

- Integrated high performance of quasi-symmetric optimized stellarators
- Design and assembly of simplified coil systems
- Predictive capability
- Divertors
- Operational limits and their profile sensitivity
- Impurity and fusion ash accumulation
- Anomalous transport reduction
- Energetic particle instabilities
- Disruptions

The form of such a device will depend on outcomes of the above mentioned studies.

Stellarator Concept Operating Parameters

Parameter	Present Values W7AS	Present Values LHD	Reactor Target (ARIES-CS)
Confining Field[(a)] (T)	2.5	2.64	5.7
Plasma current[(b)] (MA)	0	~ 0	4[(b)]
Edge magnetic rotational transform iota(a)	0.35	1.5	0.65
Flattop duration Δt (sec)	~0.2	~0.3	∞
External sustainment/current drive type	—	—	—
External sustainment/current drive power[(c)] (MW)	—	—	—
Current drive efficiency (η)	—	—	—
Major Radius[(d)] (m)	2	3.75	7.76
Minor Radius[(d)] (m)	0.186	~0.6	1.7
Elongation[(d)] (k)		1	1.8
Average density $\langle n_e \rangle$ (m^{-3})	1.1×10^{20}	2.1×10^{20}	4×10^{20}
Central T_e ; $\langle T_e \rangle$ (keV)	0.7	0.85	11.8; 6.6
Central T_i ; $\langle T_i \rangle$ (keV)	0.7	0.85	11.8; 6.6
Average beta (%)	0.5	1.5	6.4
Energy confinement time (s)	0.06	0.11	1.19
Fusion power density $B\tau_E$ (T-s)	0.15	0.29	6.8
Normalized confinement time H_{ISS95}[(e)]			2
$\rho_* = \rho_D / a$	1.2e-3	2.3e-3	1.2e-3
$S_a = a/ r_a$			51
Collisionality (ν_*)[(f)]	19	3	0.15
Normalized pulse length (τ/τ_r)[(g)]	12	0.9	∞
Normalized pulse length ($\tau/\tau_{Ti=Te}$)[(g)]	62	92	∞
Estimated Fusion Power (MW)	—	—	2440
Estimated neutron wall loading (MW/m^2)	—	—	2.6
Exhaust plasma exhaust power (MW/m^2)			2[(h)]

[(a)]peak on axis
[(b)]bootstrap current only
[(c)]power to plasma needed to maintain configuration, magnetic field, or plasma current
[(d)]mean values
[(e)]Normalized to the ISS-95 confinement scaling. U. Stroth et al., Nucl. Fusion 36 (1996) 1063.
[(f)]Evaluated at r/a ~ 0.7
[(g)]τ_r ($\tau_{Ti=Te}$) is time scale for configuration redistribution (temperature equilibration)
[(h)]Average power on divertor plates. The plate design was not fully optimized and the peak heat load is predicted to be in the range between ~5 and ~18 MW/m^2

CHAPTER 5

THE SPHERICAL TORUS

5.1. CONCEPT DESCRIPTION

The Spherical Torus is a tokamak with very low aspect ratio (A = 1.1 – 2). While the magnetic topology remains the same, changes in the geometry have important influences on the device's properties. At low aspect ratio, the ratio of the toroidal field at the inner edge relative to the outer edge becomes much larger which also modifies the ratio of the poloidal field, B_p, to the toroidal field, B_T, (see Fig. 5-1). This results in a shorter field line near the outer edge with a longer and stronger field line near the inner edge. The dramatic change in the ratio of field components increases the number of toroidal transits of the field line for each poloidal transit, increasing the safety factor q and allowing larger plasma current for a given toroidal field and plasma cross section.

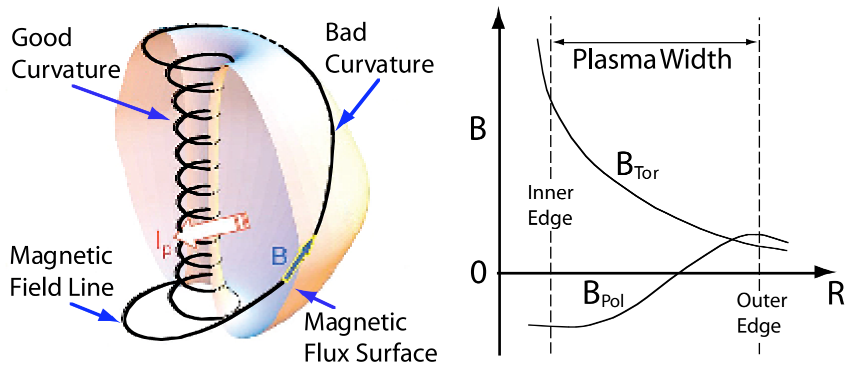


Fig. 5-1. The Spherical Tokamak magnetic configuration. The poloidal field, B_p, is comparable to the toroidal field, B_T, near the outboard plasma edge, and much smaller near the inboard plasma edge.

The short field line near the outer edge, is curved toward the plasma (bad curvature), while the opposite is the case for the long field line near the inner edge. The result is improved average curvature and greater stability for pressure driven modes compared to tokamaks at higher aspect ratios. This allows operation at higher normalized plasma pressure, $\langle\beta\rangle = 2\mu_0\langle p\rangle/\langle B^2\rangle$, a prediction that has been born out by experiments which reach $\langle\beta\rangle$ = 20% and local β_0 at the plasma center above 100%. By extending the Tokamak MHD stability calculations to the ST, stable $\langle\beta\rangle$ as high as 40% has been calculated. The higher β provides increased stability margins within which to design fusion energy systems with reliable plasma operation.

Like the tokamak, the ST has been shown to have reasonably good energy confinement, though the dominant processes seem to be somewhat different. ST confinement is determined by transport through the electron channel, while the ions tend to be more important in standard aspect ratio tokamaks. The strong magnetic shaping of the ST plasma edge and scrape-off layer introduces opportunities to mitigate the otherwise high plasma heat flux on the plasma facing components. The reduced magnetic field and plasma size also reduces the plasma inductance and lowers the external induction required for start-up and ramp-up operation. Combining these properties introduces new potential for reduced fusion power, tritium inventory, and capital cost of a fusion energy system. The lowered plasma B in compact designs encourage the use of demountable copper coils that lead to improved access in a fusion nuclear environment, full modularization of fusion core components, extensive remote handling of these components, and increased maintainability and availability. One application for which the ST may be particularly well suited is a relatively compact volume neutron source used to provide an integrated environment for testing the fusion nuclear science and technology needed for a demonstration power plant. Typically this application has been called a Component Test Facility (CTF) and is an integral part of the U.S. fusion development roadmap.

The ST also faces special challenges and tradeoffs. The limited inboard space limits the available inductive capabilities to assist plasma start-up and ramp-up to full current. The absence of inboard nuclear shielding puts severe requirements on the design of inboard magnets and limits its lifetime under fusion nuclear conditions, necessitating demountable magnets. Relatively compact size leads to increased heat flux on plasma facing components. As local β approaches unity, $v_{thermal}$ approaches v_{Alfven} and plasma frequency exceeds the electron cyclotron frequency, uncertain deviations from the Tokamak NBI and RF heating and current drive physics are introduced. It is therefore important to understand and utilize the ST advantages for potential mitigation of these challenges.

The progress so far and the present status support a long-term mission of the ST R&D, which is:

To develop a compact, high beta, burning plasma capability for fusion energy.

5.2. SPHERICAL TOKAMAK GOAL FOR THE ITER ERA

These potential advantages encourage the use of an ST for a component testing facility (CTF) to address many of the remaining gaps in the knowledge base described in the FESAC report on “Priorities, Gaps, and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy.” Several of the gaps identified by FESAC – in particular “taming the plasma material interface” and “harnessing fusion power” – are not addressed by existing and planned programs, including ITER. These gaps could be closed with a component testing program using D-T fueled plasmas with durations exceeding those planned for ITER, increased progressively to study the physical properties with the longest time constants of interest to these components. The total fusion neutron flux and fluence required for component performance verification are anticipated to be 1 MW/m^2 and up to 1 MW-yr/m^2, respectively, [5-1], substantially larger than those planned for ITER. The ST goal for the ITER era, as recommended by the ST and the broader fusion R&D communities, is therefore:

Establish the ST knowledge base to be ready to construct a low aspect-ratio fusion component testing facility that provides high heat flux, neutron flux, and duty factor needed to inform the design of a demonstration fusion power plant.

Note that the ST ITER-era goal encompasses the attainment of scientific understanding and not the construction of a CTF-like device. However, to focus that research, the ST ITER-era goal uses as a target, an available conceptual design for a ST-based CTF aimed at Demo technology testing and demonstration. This concept utilizes a relatively small device size (R_0 ~ 1.2 m) and includes the following set of design assumptions: The plasma $\langle\beta\rangle$ would be near or below the “no-wall” stability limit, with a bootstrap current fraction of ~50%. The plasma would be heated by NBI, and NBI current drive would provide nearly all of the remaining non-inductive current drive. NBI would also provide sufficient plasma rotation to reduce ion thermal diffusion and lead to ion temperatures substantially in excess of the electron temperature. Normalized plasma size ($1/\rho^*$) would be within factor of 2 of the present-day MA-level experiments. Extended divertor channels would be designed to reduce the peak plasma heat fluxes to acceptable levels. A degree of fusion self-heating (Q ~ 1-2) would be adequate with limited fusion power and tritium inventory. Extensive modularized components permitting high maintainability would be utilized to ensure a high duty factor.

Since it may also be possible to bridge the same gaps using a Tokamak device, achieving the ST goal will call for a close collaboration with the Tokamak program in order to arrive at an aspect-ratio optimized design. A broad parallel effort in materials and engineering science will also be required to develop Demo-relevant test components, carry out the tests in this ST goal facility, investigate and understand the underlying physical properties of interest, and improve the test components for renewed testing, eventually to obtain the knowledge base needed to design and build Demo-capable components.

[5-1] M. Abdou, Fusion Engin. Design **27** (1995) 111-153.

5.3. SPHERICAL TOKAMAK ISSUES

The following are a set of critical issues which must be addressed and resolved for the ST ITER-era goal.

Tier 1:

- **Start-Up and Ramp-Up.** Start-up and ramp-up to full current with minimal or no central solenoid

  The tight aspect ratio of an ST leaves little room for an Ohmic transformer or nuclear shielding for the central column. (It would be desirable if an ST could be operated without a central solenoid.) Research will be required to determine the minimal volt-seconds required to establish a tokamak discharge in such a device. Approaches might include some combination of coaxial helicity injection (CHI), plasma gun start-up, use of the outer poloidal field coil set to generate a toroidal electric field or current overdrive using bootstrap and some form of RF current drive. Alternate approaches which attempt to provide required volt-seconds by means of an iron core transformer or mineral-insulated solenoid also need to be explored. Once established, the plasma current must be ramped up, again with minimal or no volt-seconds, to the multi-mega-Amp level. A critical regime will be the gap between start-up and a level of current (fast-particle confinement) and density (ionization distance) where neutral beam injection (NBI) can be effectively employed. Sufficient thermal confinement is required to provide a target consistent with high efficiency current drive and high bootstrap fraction. There has been progress using some of these methods in the tokamak program. Validated models including the effects of fast-ion modes on current drive efficiency are needed. It is assumed that techniques for ramping the plasma current to its final value will also be sufficient for maintaining steady state.

  This issue is critical and will require a major extrapolation from current results. An ST-based CTF is estimated to require ~1 MA start-up plasma current, while present ST experiments have only generated ~0.16 MA without solenoid action. Further, an ST-based CTF is estimated to require 8-10 MA flat-top plasma current. This plasma current must be ramped up non-inductively from an assumed starting current of ~1 MA, and it is assumed this will be achieved utilizing the same heating and current drive tools used for plasma sustainment.

- **Plasma-Material Interface.** Strategies for handling normal and off-normal heat flux consistent with core and scrape off layer operating conditions

  Handling high normal and off-normal heat and particle fluxes will be a critical for any fusion energy system. For some simple geometric reasons, the issue can be more challenging in an ST. The small major radius increases the surface averaged loading and the larger pitch of the field lines on the outboard side of the device leads to shorter connection lengths making it more difficult to spread the heat load sufficiently. Proposed operation at low normalized density could reduce turbulence in the plasma scrape-off layer, potentially reducing the footprint for power loss. Particle control is an issue since pumping is required at relatively low normalized density. There is also a set of issues associated with handling the high power and particle fluxes for very long pulses (up to10^6 s) and high duty cycle in a nuclear environment – which will not be addressed in ITER or other planned devices. Testing will be required on likely lifetime limiting effects such as erosion, fatigue and radiation-induced changes in materials properties like embrittlement or thermal conductivity.

Resolution of this issue is needed and will require better scientific understanding and innovative approaches. This is estimated to require operation of a device with major-radius-normalized divertor heat flux at least a factor of two higher and for pulse durations 3 orders of magnitude longer than can be accessed in existing or planned experiments (including ITER).

- **Electron Energy Transport.** Adequacy and predictability of electron energy confinement at low aspect-ratio and low collisionality

  ST experiments find that thermal confinement is dominated by electron transport. Electron temperature is a critical parameter for several aspects of ST operation envisioned for its nuclear testing goal, including requirements for efficient NBI current drive. Measured values of electron thermal diffusivity are typically an order of magnitude higher than in conventional aspect ratio tokamaks. While an area of active research, relatively little is known about the actual mechanisms which transport heat through the electron channel in any magnetically confined plasma. (For example, proposed mechanisms operate over the full set of available scale lengths from the electron gyro-radius to the device size.) It will be critical to understand the physics and/or scaling of this confinement channel in order to predict the electron temperature achievable in future devices which will operate at higher magnetic field and much lower collisionality.

  Characterization and scaling of electron transport will need to be extended over a significant new range. The volume-average electron temperature projected to be needed is approximately an order of magnitude higher (and the electron collisionality up to 2 orders of magnitude lower) than achieved in present ST experiments. Conceptual design for the available ST-CTF example is only approximately 30% larger than present STs, but are projected to have 4 times higher toroidal field and 5-10 times higher plasma current. Achieving the ITER-era goal will likely require substantially improved understanding of anomalous electron transport.

- **Magnets.** Reliable center post magnets and current feeds to handle substantial fluence of neutrons

  Although the ST operates with a relatively low average toroidal field, the low aspect ratio geometry and operation without a nuclear shield for the central column, create a number of unique challenges. The ratio of average field to peak field is relatively higher making the mechanical design of the central column difficult even in the current generation of ST experiments. The proposed ST goal will require operation at maximum fields 2-3 times above the levels in current machines, leading to magnetic stresses about an order of magnitude higher. Materials strain will depend on design details. Material properties, including mechanical strength, toughness and electrical and thermal conductivity, will be affected by the high neutron fluence and must be predicted and controlled. To maintain low aspect ratio, the central column of an ST based nuclear testing experiment could have only minimal shielding, making regular replacement of the central column a key requirement. Conceptual design studies incorporate high-current, demountable joints to ease fabrication and maintenance. Insulator lifetime, particularly for a central solenoid if one is required, is another critical aspect of this issue.

Requiring maximum toroidal magnetic fields that are 2-3 times higher than in present ST devices, this issue represents a major extrapolation in achieved parameters. The normally conducting magnets that produce this field must operate continuously in a high neutron flux environment with demountable joints to facilitate maintenance and access to device internal components. Inductive start-up current drive using an iron core transformer or a small solenoid using mineral insulated conductor (MIC) has not been tested on any device. This issue must be resolved for the ST ITER-era goal.

Tier 2:

- **Integration.** Demonstrated integrated high-performance scenarios

  An ST based CTF is projected to require simultaneous achievement of plasmas with a degree of self-heating (Q ~ 1-2); with no inductively driven current (~50% bootstrap current); with profiles consistent with macroscopic stability and adequate confinement; vanishingly low disruptivity and high, steady-state heat flux and particle control at low normalized density. Substantial research will be required to identify workable scenarios, control tools and measurement techniques.

  While some steps toward integrated operation have been taken, full resolution of this issue is essential for the ITER-era goal. A lengthy development process should be anticipated. Data from the tokamak program will be useful in resolving some of the physics questions.

- **Disruptions.** Disruption avoidance and mitigation for reliable continuous operation

  The ST goal incorporates very long pulses (up to 10^6 s) and high availability, introducing stringent requirements on disruption avoidance and mitigation far beyond the current plans for ITER or any other long-pulse tokamaks. The goal assumes plasma conditions within known operating and stability limits to reduce the probability of disruption but requires continuous operation of sensors and actuators. The science underlying disruption frequency under nearly-steady conditions below known operating limits will need further research.

  The ST goal requires a large extrapolation in achieved performance: disruption avoidance for durations ~3 orders of magnitude longer than present or planned devices. This issue must be resolved for the ITER-era goal.

- **RF Heating and Current Drive.** Efficient RF heating and/or current drive at the mega-Amp level in over-dense plasmas

  RF requirements for the ST goal have an unusual status. Conceptual designs rely on NBI alone for heating and current drive. However, if NBI is not fully sufficient for all requirements including heating and current drive during ramp-up or profile control during any phase of operation, well tested RF scenarios may not be available. This is another result of the aspect ratio which leads to operation with the electron cyclotron frequency well below the plasma frequency; an "over-dense" condition that prevents common RF heating and current drive schemes from working. Some testing has begun on alternate schemes including electron-Bernstein wave and high-harmonic fast-wave ICRF.

Resolution of this issue may be critical if NBI drive is not sufficient. If so, it will require innovative RF schemes which are largely undeveloped.

- **3D Fields.** Control of error fields, ELMs and RWM using far-away 3 dimensional control coils

  Current tokamak experiments often require non-axisymmetric trim coils to correct error fields or to control edge localized modes (ELM) or resistive wall modes (RWM). In non-nuclear devices, these coils can be placed close to the plasma within a range where higher order moments of the applied field can have a substantial effect. However, in a fully nuclear device entailed by the ST goal, it is likely that coils will need to be placed farther away from the plasma, relative to the scale size of the machine, reducing their efficiency. Research must be carried out to determine the requirements for these 3 dimensional control coils and their compatibility with a nuclear experiment. Note that limitations on coil positions will be more stringent for this application than in ITER.

  It will be important to resolve this issue. The current state of knowledge is evolving rapidly, but is relatively immature. Data from the tokamak program will be useful in resolving some of the physics questions.

- **Ion Scale Transport.** Predictive understanding of ExB shear and suppression of ion scale turbulence and transport (energy, momentum, particle)

  In current ST experiments, ion thermal transport tends to be close to neoclassical (collisional) transport levels. This is believed to be the result of large sheared flows (ExB shear), driven by NBI induced rotation, that suppress ion-gyroradius scale turbulence. Reliable extrapolation to the ST goal plasma conditions requires better understanding of momentum transport in order to predict the rotation profiles that will exist. Moreover, while neoclassical transport is sufficiently large to exhaust power in the ion channel for current experiments, future devices that operate at higher field and current may begin to see significant anomalous ion transport. The integrated goal also requires predictive understanding of particle transport, which is also believed to be caused by ion-gyroradius scale turbulence

  Achieving the goal requires some level of predictive capability for all of the transport channels. While an active area of research, a great deal is still unknown, especially with regards to particle and momentum transport. Data from the tokamak program will likely be useful in resolving some of these physics questions, though differences and similarities must be carefully assessed.

- **Fast Particle Instabilities.** Impact of super-Alfvénic ion driven instabilities on NBI heating, current drive, and torque at low aspect-ratio

  The ST goal involves plasmas with large populations of fast ions from fusion reaction products and neutral beam systems. Moreover, because it would operate at low $\nu*$ and high β, with $v_{thermal}$ close to v_{Alfven}, an ST goal device is projected to have a relatively high level of sub-Alfvénic NBI ions and super-Alfvénic fusion alpha particles compared to a standard aspect ratio tokamak. These populations are capable of driving various macroscopic instabilities which can cause rapid loss of the fast particles, reducing heating and current drive efficiency. Research into the generation, effects and control of fast-ion modes will be required.

  Adequate confinement of fast ions from NBI and fusion reactions must be guaranteed. This is an active area of tokamak (and ST) research but ST conditions represent a substantial extrapolation in the ratio of fast-particle velocity to Alfven velocity.

Tier 3:

- **NTMs.** Avoidance of neoclassical tearing mode in rotating plasmas near the no-wall beta limits

  Neoclassical tearing modes are the result of a macroscopic instability which occurs on low-order rational q surfaces at high normalized pressure (β) and low collisionality. Research is needed because the ST cannot use techniques for mode suppression developed in standard aspect ratio tokamaks. These methods use electron cyclotron current drive which is not accessible in an over-dense ST. The ST may able to avoid this instability by running with a current profiles that removes low order rational q values from the plasma ($q_{\min} > 2\text{-}3$), in combination with lower β relative to the no-wall limit and at smaller normalized device size ($1/\rho^*$).

  This issue must be resolved, but will benefit greatly from research carried out on tokamaks. It will require only a moderate extrapolation from current state of the art, but may require innovative or untested approaches.

- **Continuous NBI Systems.** Continuous operation of positive or negative-ion energetic neutral beam systems

  Neutral beam systems have been designed and operated for long pulses relative to thermal time constants for their internal components. However, an ST-based CTF requires a dramatic extrapolation in operating time, bringing lifetime issues into sharp focus. Alternate approaches for beam neutralization which do not present impractically large gas loads to the fuel system must be found. Since these requirements go well beyond those for ITER or other planned long-pulse experiments, this issue will likely need to be addressed within an enabling technology R&D activity associated with a component testing program.

  The pulse length required for an ST goal facility will be uniquely long. For example, over 20 years of operation on existing machines, individual sources have accumulated 10^5 sec of high-power operation with perhaps $5\text{x}10^5$ sec of use on the filaments. The NBI systems of an ST-based CTF will require on the order of 10^7 sec operation without significant maintenance.

5.4. SPHERICAL TOKAMAK AVAILABLE MEANS, GAPS, AND INITIATIVES

5.4.1. Overview of Available ST Research Facilities

Spherical tokamak research is a world-wide endeavor with 22 ST experiments now operational in eight countries (see Fig. 5-2). NSTX (U.S.) and MAST (U.K.) are the largest and most capable facilities in the world ST program. These devices have plasma major radii ~ 0.8-1 m, plasma currents in the mega-ampere range, toroidal magnetic field strengths 0.5-0.6 Tesla, neutral beam injection heating 5-7 MW, and additional RF heating up to 6 MW. NSTX and MAST have extensive growing sets of modern diagnostics for plasma profiles, fast-ions, turbulence, magnetic fields, and various photon emissions. The NSTX and MAST facilities are complimentary in several ways – NSTX utilizes a close-fitting vessel and conformal conducting plates to help suppress vertical and kink instabilities, while MAST utilizes a much larger vessel with a far conducting wall and internal poloidal field coils. While the NSTX configuration provides improved plasma stabilization, the MAST geometry offers increased flexibility and access for diagnostics and internal components including the divertors. The different configurations are used to test a complementary range of startup approaches. Both programs have proposed significant upgrades.

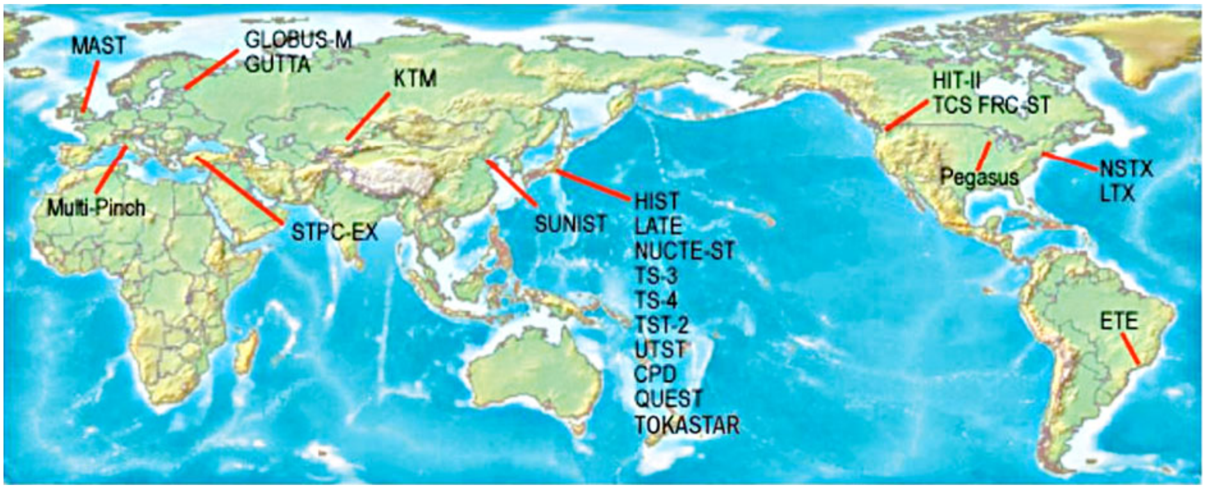


Fig. 5-2 Location of ST programs.

In the U.S., the Pegasus experiment is testing, in the limit of A approaching 1.1, non-solenoidal ST start-up using the plasma guns, investigating wave physics in over-dense plasma conditions, and assessing the possibility of super extended divertor studies. The LTX experiment is investigating the potential scientific and engineering advantages of liquid lithium-coated plasma facing components to reduce particle recycling, thereby to increase plasma energy confinement.

The ST program in Japan has focused on innovative very high beta and very long pulse research. HIST, TS-3, TS-4, and UTST are testing and utilizing reconnection physics in various startup configurations to explore and understand the science of toroidal betas approaching unity. TST-2 and LATE are investigating plasma current initiation using ramped poloidal fields and a range of EC, EBW, LH, and HHFW heating and current drive, coupled with significant measurements, modeling and simulation. These have provided input to the new medium sized QUEST experiment, which just started research operation to investigate very long-pulse ST plasma-wall interface physics in more than one hour plasma operation utilizing EBW and NBI heating and current drive, and a current up to 500 kA as the final goal.

5.4.2. Available Means, Gaps, and Initiatives

The following are a set of available means, gaps, and new initiatives to address critical issues which must be resolved for the ST ITER-era goal.

Tier 1:

- **Start-Up and Ramp-Up.** Start-up and ramp-up to full current with minimal or no central solenoid
  - Available Means to Address the Start-Up Issue. Coaxial Helicity Injection (CHI) start-up will be tested to ~0.2-0.3 MA on NSTX, and NSTX will also investigate bootstrap current overdrive to levels approaching 0.3-0.4 MA using HHFW heating power up to 4 MW. On MAST, up to 1 MW of EBW heating and current drive will be tested to achieve ~0.5 MA. ECH and EBW start-up to 0.1-0.2 MA will also be studied on TST-2, LATE, and the new long-pulse ST experiment QUEST (to be operational in 2008). On Pegasus, plasma gun

start-up will be tested to 0.1-0.3 MA, and this technique is also proposed to be tested on NSTX to assess the size and field scaling of gun start-up.

— Gap. The level of start-up current expected to be achieved in available facilities (~0.5 MA) is estimated to be a factor of 2 below estimated requirements (~1 MA).

— New Programs/Initiatives Needed to Close the Gap. Increased/upgraded toroidal field on NSTX to ~1 T could increase CHI start-up currents to ~0.5 MA and aid HHFW heating efficiency of start-up plasmas. Assuming successful demonstration and understanding of plasma start-up to ~0.5 MA on NSTX and MAST, reliable extrapolation to the ~1 MA could be achieved with sufficient understanding of the toroidal field, plasma size, and plasma current scaling of the efficiency of the various start-up techniques combined with validated models of the start-up method physics and plasma equilibrium evolution.

— Available Means to Address the Ramp-Up Issue. On NSTX, upgraded HHFW heating power will investigate bootstrap current overdrive ramp-up to ~0.5 MA. On MAST, EBW heating and current drive will be tested to achieve ~0.5 MA ramp-up current.

— Gap. The level of ramp-up current expected to be achieved in available facilities (~0.5 MA) with wave heating techniques is estimated to be an order of magnitude below a possible ST goal requirement (~8-10 MA). Plasma current ramp-up is envisioned to be achieved using neutral beam current drive possibly supplemented by RF heating. Increased current drive efficiency through higher electron temperature and reduced normalized density is predicted to be required to achieve plasma current ramp-up and sustainment using neutral beam injection. Neutral beam ramp-up is not possible to test in present ST devices for several reasons: (1) injection is too perpendicular and results in high bad-orbit loss fractions at low current, (2) insufficient absorption at low plasma density, and (3) insufficient NBI power.

— New Programs/Initiatives Needed to Close the Gap. Major upgrades of NSTX/MAST for higher toroidal field (up to 1 T/0.8 T), higher beam power (up to 15 MW/12.5 MW) assisted by increased RF power (up to 6 MW HHFW/2 MW EBW), longer pulse-length (up to 5 s/2.5 s), and improved density control are projected to enable the first tests of NBI current ramp-up — from start-up currents of ~0.5 MA to peak ramp-up currents of ~1 MA. Experimental determination of the achievable ramp-up current as a function of field and power, combined with time-dependent modeling accounting for projected confinement and stability are needed to begin to project the requirements for ramp-up to the full operating current. Successful demonstration of NBI current ramp-up utilizing the NSTX/MAST upgrades and improved modeling capabilities could reduce the ramp-up gap. An ST facility with higher toroidal field and NBI power that could achieve higher ramp-up current (~3-5 MA) may be needed to reliably extrapolate to the possible plasma currents of an ST-based CTF.

- **Plasma-Material Interface.** Strategies for handling normal and off-normal heat flux consistent with core and scrape off layer operating conditions

— Available Means to Address Issue. NSTX and MAST will assess heat flux width physics and scaling for the ST. NSTX will test the compatibility of a liquid lithium divertor for particle control to access up to a factor of 2 reduced density target plasmas with ITER-level high heat flux up to ~10 MW/m^2 for short pulses (~1 s). On MAST, the impact of a long divertor channel and divertor biasing will be tested. LTX will test liquid lithium wall/limiter solutions in a very low recycling edge for short pulses (< 0.1 s), and QUEST will test long-pulse (~100s of seconds) ST divertors at low divertor heat flux. Existing/upgraded higher aspect ratio tokamaks (DIII-D, C-Mod, JET) will assess high heat flux within a factor of 2 of ITER for short durations (2-10 s), and KSTAR and JT-60SA at full heating power will eventually access similar heat fluxes for longer durations (100s of seconds). These experiments, combined with enhanced predictive capability for the SOL and divertor heat flux can narrow the divertor performance and understanding gap.

— Gap. Heat-flux mitigation strategies applied so far for present and planned experiments (including ITER) likely do not extrapolate to an ST-based CTF without using much extended divertor channels. This is due to the possible, similarly high outboard parallel heat flux, lower normalized divertor electron density (associated with the expected reduced-density operating scenario), and reduced field-line length leading to higher plasma temperature at the divertor. Plasma facing components with increasingly longer pulse durations beyond the ITER design needed by an ST-based CTF have not been tested at all.

— New Programs/Initiatives Needed to Close the Gap. NSTX/MAST upgrades of field, current, and power could enable access to high unmitigated heat fluxes (~20 MW/m^2) for short durations ~2-3 s. This level is within a factor of 2 of an unmitigated ST-based CTF level if the CTF design example is assumed. NSTX with upgrades could increase divertor flux expansion (20 $\rightarrow$ 60) using an X-divertor, and radiative divertors could be assessed for heat-flux mitigation at higher heat flux and lower divertor density. Upgrades to MAST could test the compatibility of high heat flux mitigation and low divertor density operation using cry-pumps. Additional heat flux mitigation methods (such as the Super X-divertor) to reduce the peak divertor heat flux by up to a factor of ~5 could be tested in upgraded or new facilities, and if successful, would significantly narrow or remove the short-pulse heat-flux mitigation gap provided the divertor heat-flux width remains as presently assumed.

In this case, toroidal and linear facilities to test, in D-D operation, mitigation of such divertor heat fluxes for very long pulse durations (up to ITER, ~3000 s) will narrow the gap. Such facilities could also test the impact of high heat and particle fluxes on neutron irradiated samples (from HFIR in the nearer term and IFMIF in the longer term for example) and assess erosion, fatigue, embrittlement, and thermal conductivity changes. Such testing could inform the choice of materials for an ST-based CTF. The initial operation of the device itself in D-D would generate substantial data on even longer pulse-lengths progressively up to and beyond 3000 s. Testing and improvements of the plasma material interface science with increasing pulse durations in D-T beyond the ITER design durations can only be carried out in an ST-based CTF.

- **Electron Energy Transport.** Adequacy and predictability of electron energy confinement at low aspect-ratio and low collisionality

— Available Means to Address Issue. NSTX will test a liquid lithium divertor for density control, reduced collisionality, and possibly increased electron temperature. Reduced collisionality combined with existing high-k and planned low-k turbulence diagnostics will improve understanding of electron transport in the ST. The LTX experiment will test a liquid lithium wall to achieve a very low recycling regime and assess the associated impact on electron energy confinement. NSTX has tested and measured electron turbulence and energy transport in high beta and found a favorable combined dependence on increased magnetic field and plasma current, and a strong improvement in the presence of reversed magnetic shear. Measurements and interpretations of electron energy transport in tokamak plasmas with ~10 keV electrons will inform the research on ST.

— Gap. The volume-average electron temperature projected to be needed in an ST-based CTF is approximately an order of magnitude higher than is achievable in present ST experiments. The understanding of electron transport is insufficient to reliably extrapolate to high electron temperature needed for high beam current drive efficiency. The high plasma beta of the ST may increase the role of electromagnetic effects on electron transport, implying that ST-specific research on electron transport is likely needed for developing the ST knowledge base for the ITER-era goal.

— New Programs/Initiatives Needed to Close the Gap. Upgrades to NSTX/MAST to double the nominal field, current, and heating power combined with improved density control could provide up to an order of magnitude lower electron collisionality and double the electron temperatures at similar plasma beta and gyro-radius normalized to plasma size. These major upgrades would provide significant additional data to determine the parametric scaling of electron transport in field, current and magnetic shear, and combined with improved turbulence diagnostics would provide insight into the modes responsible for electron (and ion) transport. However, predictive capability for electron confinement sufficient to extrapolate to possible ST-based CTF conditions could remain a significant challenge, dependent on how favorable the parametric scaling results turn out to be. An ST facility capable of operating with plasma size, toroidal field, plasma current, electron density, and heating power closer to possible ST-based CTF parameters could assess with high confidence if electron confinement is sufficient to support long-pulse ST operation if necessary, and if successful, could close the electron transport gap.

- **Magnets.** Reliable center post magnets and current feeds to handle substantial fluence of neutrons

— Available Means to Address Issue. The design and operational experience of the magnet systems of existing ST devices can be utilized to begin to quantify the allowable magnet current density and stress that can be sustained for long pulse durations. We note that the integrated operational time of present ST devices corresponds to ~5-10% of the pulse-length of a single maximum ST-based CTF shot near the end of the testing program.

— Gap. No existing ST device has produced toroidal field at the plasma geometric center above 0.55 T and at the magnet surface above 3.9 T, whereas the ST-CTF example for an ST-based CTF indicated that 2.2 T and 8 T would be required, respectively. No existing ST device utilizes a single-turn TF coil; only START used it and was shut down in 2000. For start-up, iron-core transformers and mineral-insulated conductor solenoids have not been tested on any device.

— New Programs/Initiatives Needed to Close the Gap. The proposed center-stack upgrades of NSTX and MAST would double the toroidal field from ~0.5 T to 1 T, which is within a factor of 2 of the possible field of an ST-based CTF. Successful design and implementation of higher toroidal field (TF) in upgraded NSTX/MAST devices would significantly narrow the gap in coil design for higher toroidal field in the ST. However, an ST-based CTF would very likely utilize a single-turn central TF coil, whereas present and upgraded ST facilities will have multi-turn TF coils. Thus, additional R&D will be needed to develop the single-turn TF coil and the high current joints connecting the central TF rod to the outer TF conductor, and the joints connecting the TF coil to power supplies. ST facility upgrades or new experiments at the Concept Exploration (CE) level incorporating a single-turn demountable central TF rod, an iron-core transformer, and/or an MIC solenoid to elucidate engineering issues associated with these magnet concepts could inform the ST-based CTF design in a cost-effective manner. Assessments of the impact of neutron irradiation on magnet mechanical strength, electrical and thermal conductivity of the conductor, and impact on MIC insulation and solenoid structural integrity have been performed using fission spectrum neutrons, and results should be incorporated into the ST-based CTF magnet design. Successful field-coil major upgrades of existing devices, combined with design and prototyping of goal-relevant TF and solenoid magnets accounting for irradiation effects could close the magnet gap.

Tier 2:

- **Integration.** Demonstrated integrated high-performance scenarios
  - — Available Means to Address Issue. Present ST experiments have sustained (for several current redistribution times) the toroidal beta (~15-20%) and bootstrap current fraction (~50%) projected to be needed for the ST-based CTF operational scenario. Active pumping using a liquid lithium divertor will be tested in the near-term in NSTX as a means to achieve density control in H-mode discharges to reduce the plasma collisionality and increase the NBI current-drive efficiency. Self-heating by alphas at moderate Q ~ 0.25-0.6 has already been demonstrated and characterized in TFTR and JET, and existing simulation tools and integrated modeling should be sufficient to extrapolate to Q~1-2 conditions. Integrated high-performance plasma scenarios developed for long-pulse tokamaks and ITER would also strongly complement ST integration research.
  - — Gap. Favorable operating scenarios for the ST are projected to have low normalized density fraction (~20-30%), utilize up to 50% beam-driven current fraction, and operate with a Hot-Ion H-mode confinement enhancement factors up to 50% above the ITER H-mode scaling. Present ST experiments may have difficulty achieving normalized density fractions below 50%, have sustained only 10-15% beam-driven fraction (due to high density operation and non-optimal beam injection geometry), and have sustained confinement enhancements 10% above ITER scaling. Different, potentially more favorable, energy transport mechanisms for the electrons and ions have been indicated in present-day ST experiments — accentuating the importance of understanding energy transport in the ST for projecting to high integrated performance.
  - — New Programs/Initiatives Needed to Close the Gap. Major upgrades of present ST experiments (MAST/NSTX) to higher field (to increase electron temperature and NBI current-drive efficiency) combined with more tangential and off-axis injection of neutral beams (for increased NBI current-drive efficiency and elevated q) could provide access to increased NBI current fraction and sustained fully non-inductive plasma conditions relevant stability properties. Additional pumping capabilities/techniques may be needed to access the low normalized plasma density assumed. Additional means to increase plasma thermal confinement may be needed if reduced collisionality does not result in sufficient confinement enhancement. Increased liquid lithium wall coverage is a possibility if LTX shows favorable confinement results. Improved predictive capability for confinement –

especially electron confinement – is needed to reliably extrapolate integrated performance. Integration of the above plasma characteristics with high divertor heat flux and acceptably low tritium retention with the ST goal-relevant first-wall conditions (metal walls, high wall temperature) must be demonstrated for sufficiently long pulses. Such integration could be tested in a high heat-flux long-pulse facility capable of operating with the ST goal-relevant integrated plasma conditions, and/or during the first operational phase of an ST-based CTF.

- **Disruptions.** Disruption avoidance and mitigation for reliable continuous operation
  - Available Means to Address Issue. A determination and parameterization of the operational limits — and a determination of the distance below those operational limits that provides disruption avoidance — is needed. Present ST devices have and will continue to provide data on operational limits. Present and future long-pulse tokamaks will also contribute significantly to the characterization and avoidance of disruptions to aid design and operation of an ST-based CTF, and disruption mitigation techniques developed for tokamaks/ITER would likely be applicable.
  - Gap. Present ST experiments have not achieved the integrated plasma performance conditions of expected to be needed for an ST-based CTF, and these performance conditions could influence the nature and frequency of disruptions. Disruption avoidance for durations progressively up to ~5-6 orders of magnitude longer than present ST plasma pulse durations and progressively up to ~3 orders of magnitude longer than present or planned long-pulse devices (including ITER) will be required.
  - New Programs/Initiatives Needed to Close the Gap. Upgraded NSTX and MAST facilities can determine the operational limits and disruption-free operating scenarios for short pulse durations with integrated plasma conditions representative of an ST-based CTF. Development and tests of ST-goal-relevant sensors, actuators, and control techniques for very long pulses could be developed in ST and/or tokamak long-pulse facilities operating with ST-relevant integrated plasma conditions. Demonstrated disruption-free long-pulse operation and improved predictive capability for disruption avoidance may be sufficient to extrapolate to ST goal pulse durations. Demonstrated disruption avoidance for ST goal pulse-durations and integrated plasma conditions may only be possible in an ST-based CTF itself.
- **RF Heating and Current Drive.** Efficient RF heating and/or current drive at the mega-Amp level in over-dense plasmas
  - Available Means to Address Issue. NSTX has demonstrated efficient and reliable HHFW heating at intermediate power levels with L-mode edge conditions for short durations (~1 s), and antenna upgrades are planned to extend sustained and efficient HHFW heating to H-mode plasma conditions. Research on EBW coupling, heating, and current drive physics is planned to be performed in MAST (and to a lesser extent NSTX). ICRF and ECH/ECCD launcher designs and operational experience from present and planned long-pulse tokamaks and ITER are relevant to the ST HHFW/EBW systems.
  - Gap. The presently achieved EBW transmission efficiency in H-mode is ~ 50-60%. Higher coupling efficiency will help minimize edge heating and losses and to maximize the net EBW heating and current drive efficiency in possible high-power EBW applications to an ST-based CTF. High power and high power-density HHFW and EBW launchers capable of

surviving high heat and neutron flux for durations longer than the ITER design will be needed if it is needed by an ST-based CTF.

— New Programs/Initiatives Needed to Close the Gap. Major upgrades to NSTX/MAST to increase HHFW/EBW heating power and plasma pulse duration could test the role of edge effects on wave coupling and antenna and first wall damage for short pulses. This includes the effects of surface waves, sheaths, and parametric decay instabilities for the HHFW, and edge density fluctuation and ponderomotive density expulsion effects for the EBW. A long-pulse high-heat-flux ST/tokamak facility utilizing HHFW and/or EBW could test the survivability of ST- specific launchers and nearby first wall components for durations and heat fluxes far beyond what is achievable in present ST devices and up to the ITER-level durations and heat fluxes – thereby narrowing the gap. Tests of the physical properties that limit the performance of HHFW and EBW launcher systems beyond the ITER level (pulse length, heat flux, nuclear environment) likely need to be carried out in an ST-based CTF.

- **3D Fields.** Control of error fields, ELMs and RWM using far-away 3 dimensional control coils

— Available Means to Address Issue. Present tokamak and ST facilities, and future long-pulse tokamaks and ITER are addressing (and will continue to address) the physics requirements and actuators necessary for simultaneous control of error fields, ELMs, and RWMs.

— Gap. A first-principles predictive capability for ELM avoidance and mitigation in tokamaks and STs using 3D resonant magnetic perturbations (RMPs) does not yet exist and is needed for an ST-based CTF design (and for ITER and Demo). Similarly, a predictive capability for the rotational stabilization of the RWM has not been fully developed.

— New Programs/Initiatives Needed to Close the Gap. Additional theory, modeling, and upgraded RMP coil systems should be implemented on STs and tokamaks (as needed) to develop a predictive capability needed for RMP ELM mitigation. Far away coils should be implemented and tested on an ST device with ST goal-relevant integrated plasma conditions to demonstrate integrated control of error fields, ELMs and RWMs extrapolable to the needed conditions. Upgrades of NSTX to higher field, current, and temperature would enable access to reduced ion collisionality and provide unique ST data to elucidate the underlying mechanisms for rotational stabilization of the RWM and for toroidal flow damping by 3D fields.

- **Ion Scale Transport.** Predictive understanding of ExB shear and suppression of ion scale turbulence and transport (energy, momentum, particle)

— Available Means to Address Issue. Significant progress has been made in understanding shear-flow suppression of ion-gyro-scale turbulence in tokamaks, and STs are presently implementing the turbulence diagnostics necessary to study this physics in the ST. Present STs and tokamaks are beginning to study the underlying mechanisms for anomalous momentum transport.

— Gap. There is inadequate predictive capability for momentum transport to assess whether appropriate rotation can be achieved for the required ion thermal confinement, error field suppression, and rotational stabilization of the RWM. There is insufficient knowledge of the underlying causes and scalings of particle transport to determine the fueling and pumping requirements to achieve low normalized density.

— New Programs/Initiatives Needed to Close the Gap. Major upgrades of present ST experiments (MAST/NSTX) to higher field, current, NBI torque, and varied torque deposition combined with turbulence diagnostics and gyrokinetic simulations can be utilized to develop a predictive capability for momentum transport in the ST. These same capabilities combined with effective particle pumping to reduce and control the plasma density can be utilized to develop a predictive capability for particle transport in the ST.

- **Fast Particle Instabilities.** Impact of super-Alfvénic ion driven instabilities on NBI heating, current drive, and torque at low aspect-ratio

  — Available Means to Address Issue. Present ST experiments have identified the instabilities driven by super-Alfvénic fast-ions from neutral beam injection, and these experiments are beginning to investigate the impact of multi-mode interactions on fast-ion redistribution and loss. For example, Toroidal Alfvén Eigenmode (TAE) avalanches have been observed to cause significant redistribution/loss of NBI fast ions (up to 15% neutron rate decrements) for plasma conditions relevant to an ST-based CTF.

  — Gap. There is presently insufficient predictive capability for the transport of fast-ions by fast-ion-driven instabilities to determine with high reliability the heating, current drive, and torque input by the proposed dominant heating method - neutral beam injection.

  — New Programs/Initiatives Needed to Close the Gap. Off mid-plane tangential beam injection calculated for a possible ST-based CTF for more efficient current-drive may be destabilizing to Toroidal Alfvén Eigenmodes (TAE), and the projected higher normalized fast-ion pressure fraction (due to low density operation) may lead to the excitation of TAE avalanches resulting in fast-ion loss. Upgrades of present ST experiments (MAST/NSTX) to higher field and current, more tangential and off-axis injection of neutral beams, and reduced normalized density could produce fast-ion distribution function characteristics similar to those expected in an ST-based CTF. These experimental conditions combined with additional fast-ion diagnostics and advanced non-linear modeling could be utilized to develop a predictive capability for the transport of both NBI fast-ions and alpha particles by fast-ion instabilities - thereby closing the gap.

Tier 3:

- **NTMs.** Avoidance of neoclassical tearing mode in rotating plasmas near the no-wall beta limits

  — Available Means to Address Issue. Present ST experiments are developing a predictive capability for low-order (3/2, 2/1) NTM stability by assessing the impact of low aspect ratio, flow and flow shear, and large normalized gyro-radius on NTM stability. Extensive research on NTM stability and active NTM suppression has been carried out in tokamaks – in particular for ITER – and this research provides the foundation for understanding NTMs in the ST.

  — Gap. The proposed operating scenario aims to avoid deleterious 2/1 NTMs by operating with $q_{\min} > 2$. However, elevated $q_{\min}$ could also adversely impact thermal confinement, global MHD stability, and Alfvén eigenmode stability. Present ST experiments are partially inductively driven and are unable to sustain integrated scenarios with $q_{\min} > 2$ to test avoidance of low-n NTMs and assess the impact of elevated q on other aspects of plasma performance.

— New Programs/Initiatives Needed to Close the Gap. Upgrades of present ST experiments (MAST/NSTX) to higher field combined with more tangential and off-axis injection of neutral beams could provide access to goal-relevant plasma scenarios with elevated q which are free of lowest-order rational surfaces. However, if achieving the integrated plasma performance goal requires operation at low central q and is prone to NTM instabilities, then efficient, localized, and controllable EBW current drive would need to be developed for NTM suppression.

- **Continuous NBI Systems.** Continuous operation of positive or negative-ion energetic neutral beam systems

— Available Means to Address Issue. For positive NBI, long pulse operation (100s of seconds) in the ~100 keV energy range is expected to be demonstrated by the KSTAR beam program using the conceptual design of the 1000 s TPX beamline. For negative-ion NBI (NNBI), the JT-60SA program intends to develop a 500 keV, 10 MW, 30 s beam. ITER is developing NNBI beamlines with a planned capability of 16.5 MW at 1 MeV for 1000 s.

— Gap. An ST-based CTF will require effectively continuous (~10^6 s) beam operation. However, present beams require cryopumps and must be frequently regenerated (every few ~10^3 s) to avoid exceeding the explosive limit of hydrogen. Multiple cryopumps to enable cyclic regeneration of some cryopumps while others continue to pump will require large space around the end of the NBI system. Further, there may be difficulty in achieving the needed NNBI power density at the reduced beam voltage (200-250 kV) that may apply to an ST-based CTF.

— New Programs/Initiatives Needed to Close the Gap. A dedicated research program and facility to develop beam neutralization for very long pulses is needed. Options include extending/improving cryopumping methods developed for ITER or developing potentially more efficient techniques — for example lithium jet neutralization. For NNBI, the development of techniques to increase beam current density may also be required to minimize the port size for beam injection to maximize the volume available for component testing.

Summary Tables

Table 5-1 provided below summarizes the means to close the ST ITER-era goal gaps, and Table 5-2 provides a list of key parameters for the ST concept for reference.

Table 5-1

		Means to close gap		
	ST Issue	**Improved predictive capability**	**Major upgrades of existing ST facilities**	**New initiatives and facilities**
Tier 1	**Start-up and ramp-up:** Start-up and ramp-up to full current with minimal or no central solenoid	**Start-up**: Validated non-linear 3D MHD simulations of plasma current start-up in ST (for CHI and plasma guns)	**Start-up:** Double B_T = 0.5T to 1T to test CHI and gun start-up scaling with toroidal field	
			Ramp-up: Double B_T = 0.5T to 1T, double P_{NBI}, add more tangential NBI, 5x longer pulse - all to demonstrate NBI ramp-up to ~1MA in ST plasma	ST device with ramp-up capabilities comparable to ST goal: B_T ~ 2T, P_{NBI} ~ 30MW, E_{NBI} ~ 100-250keV, with sufficient pulse-length to demonstrate current ramp-up approaching ST goal values
	Plasma-Material Interface: Strategies for handling normal and off-normal heat flux consistent with core and scrape off layer operating conditions	Understanding of SOL/divertor heat flux width and scaling	Double P_{aux}, 5x longer pulse, cryopumps and/or long-pulse lithium divertor for reduced density, test heat-flux mitigation strategies at low density and high heat flux	Long-pulse device with ST-goal level divertor/first-wall heat fluxes and other relevant divertor parameters (density, geometry) to demonstrate ST-level long-pulse heat-flux mitigation
	Electron energy transport: Adequacy and predictability of electron energy confinement at low aspect-ratio and low collisionality	Low, intermediate, and high-k electrostatic and electromagnetic turbulence diagnostics, validated non-linear gyro-kinetic simulations of electron transport	Double B_T = 0.5T to 1T, I_P = 1 to 2 MA, double P_{AUX} to assess electron confinement scaling with B_T, higher T_e, lower collisionality	ST device with (short pulse) plasma conditions approaching ST goal: B_T ~ 2T, I_P at least ~ 4MA, P_{NBI} ~ 20-30MW, and low normalized density to test electron confinement under ST-goal conditions
	Magnets: Reliable center post magnets and current feeds to handle substantial fluence of neutrons		Double B_T = 0.5T to 1T in present ST devices to inform engineering design of ST-goal B_T = 2T toroidal magnetic field	Concept Exploration (CE) level engineering + physics device(s) to prototype single-turn demountable TF, iron-core transformer, and mineral insulated solenoid
Tier 2	**Integration:** Demonstrated integrated high-performance scenarios	Time-dependent free-boundary modelling of plasma evolution using reduced models for core and edge transport and stability	Double B_T, access low normalized density plasmas, add more tangential and off-axis NBI, and double P_{NBI} to access fully-non-inductive integrated scenarios needed for ST goal	ST device that can demonstrate compatibility of a goal-relevant sustained integrated plasma scenarios with mitigation/control of goal-level heat and particle fluxes
	Disruptions: Disruption avoidance and mitigation for reliable continuous operation	Development of real-time stability limit/margin calculations and sensors, actuators, and algorithms to avoid limits.	Upgrade present devices to access and sustain fully-non-inductive integrated scenarios relevant to ST goal, and assess disruption thresholds/margins for short pulse (5-10s)	Very long pulse facility capable of demonstrating disruption avoidance while operating with integrated plasma scenarios relevant to ST goal - including goal-relevant EF, ELM, and RWM control

Table 5-1 (Cont.)

		Means to close gap		
	ST Issue	**Improved predictive capability**	**Major upgrades of existing ST facilities**	**New initiatives and facilities**
Tier 2 (continued)	**RF Heating and Current Drive:** Efficient RF heating and/or current drive at the mega-Amp level in over-dense plasmas	Improved diagnosis and modelling of RF edge effects for high-power HHFW and EBW - for example validated full-wave combined coupling+heating codes	Develop improved wave launchers for resilience to edge effects for high-power HHFW and EBW	Long-pulse, high-heat-flux device capable of testing RF launcher technology relevant to ST HHFW and EBW launcher systems, and relevant to ST-goal edge plasma conditions
	3D Fields: Control of error fields, ELMs and RWM using far-away 3 dimensional control coils	First-principles understanding of RMP suppression of ELMs, rotation threshold for RWM stabilization, and plasma flow damping by 3D fields	Upgrade ST devices to access reduced collisionality plasmas relevant to ST-goal ELM and RWM stability and flow-damping by 3D fields. Test near and far-away coils for EF/ELM/RWM control.	
	Ion Scale Transport: Predictive understanding of ExB shear and suppression of ion scale turbulence and transport (energy, momentum, particle)	Low-k electrostatic and electromagnetic turbulence diagnostics, validated non-linear gyro-kinetic simulations of ion transport	Double field, current, and power to decrease ion neoclassical transport and assess role of ion turbulence in ion energy, momentum, and particle transport. Vary torque profile with more tangential NBI. Implement pumping for particle transport research.	
	Fast Particle Instabilities: Impact of super-Alfvénic ion driven instabilities on NBI heating, current drive, and torque at low aspect-ratio	Validated non-linear 3D hybrid kinetic MHD simulations of fast-ion transport for ST-goal-relevant fast-ion instabilities	Double field and current, control density, and add more tangential and off-axis NBI to access fast-ion distributions function characteristic directly relevant to the ST-goal target scenarios	
Tier 3	**NTMs**: Avoidance of neoclassical tearing mode in rotating plasmas near the no-wall beta limits		Double field and NBI heating/current-drive power, control density, and add more tangential and off-axis NBI to sustain integrated scenario with $q_{min} > 2$ to test NTM avoidance and impact on projected fusion performance relative to ST goal	
	Continuous NBI systems: Continuous operation of positive or negative-ion energetic neutral beam systems			Dedicated research program to develop beam neutralization for ST-goal pulse lengths - options include extending/improving cryo-pumping methods developed for ITER and/or lithium jet neutralization. Also develop methods to increase NNBI power density for E_{NBI} ~250keV.

Table 5-2
ST Concept Key Parameters[a]

Parameter	Present ST Value (All Values Simultaneous)			ITER-Era Goal		Reactor Target
	High τ_E L-mode (116960)	Mid-I_P H-mode (129269)	High-I_P H-mode (116293)	0.1 MW/m^2	1.0 MW/m^2	4.1 MW/m^2
Confining Field (T)[b]	0.47	0.44	0.44	1.13	2.18	2.1
Plasma current (MA)[c]	1.0	0.92	1.4	3.4	8.2	29
Pulse length Δt (s) and $\Delta t/\tau_E$	0.38, 2-4	0.8, 10-15	0.45, 5-10	$\leq 1.2\times10^6$	$\leq 1.2\times10^6$	$\sim10^7$
External sustainment/current drive type	NB	NB	NB	NB+RF	NB+RF	NB
Sustainment/current drive power (MW)[d]	2	3.8	7.5	15	31	28
Current drive efficiency (10^{20} A/Wm^2)	0.02	0.014	~0.008	0.050	0.17	0.21
Major Radius (m)[e]	0.81	0.85	0.85	1.2	1.2	3.2
Minor Radius (m)[e]	0.63	0.59	0.58	0.8	0.8	2
Elongation (κ)	1.9	2.4	2.5	3.07	3.07	3.4
Volume Average n_e ($10^{20}m^{-3}$)	0.18	0.48	0.65	0.44	1.05	1.6
Volume Average T_e (keV)	0.87	0.80	0.72	3.1	6.8	16.5
Volume Average T_i (keV)	1.1	0.93	0.97	5.4	10.3	16.5
β_T / β_N (% /)	13 / 3.8	18 / 5.0	27 / 5.0	14.4 / 3.8	18.0 / 3.8	50 / 7.4
Global energy confinement time τ_E (s)	0.090	0.063	0.050	0.24[f]	0.42[f]	2.1
Fusion power density (MW/m^3) / $B\tau_E$ (T-s)	– / 0.042	– / 0.028	– / 0.022	0.16 / 0.27	1.6 / 0.92	3.4 / 4.4
Core electron transport (χ_e m^2/s)	3-5	5-12	~10-15	1.43[g]	0.82[g]	~1.2[g]
Core ion transport (χ_i m^2/s)	2-4	5-12	~5-10	0.19[g]	0.06[g]	~1.2[g]
$\rho_* = \rho_D /a$	0.023	0.024	0.025	0.017	0.012	0.006
$S_\alpha=L/\rho_\alpha$, (L = minor radius)	5.1[h]	4.5[h]	4.4[h]	2.3	4.5	9.0
Collisionality (ν_*) (electrons)	0.01	0.08	~0.14	~0.005	~0.001	~0.001
Norm. pulse length (τ/τ_r)[i]	0.3-0.5	1-2	0.7-1.3			
Fusion Power (MW)				7.5	75	2980
Neutron wall loading (MW/m^2)				0.1	1.0	4.1
Exhaust power flux (MW/m^2)		2-4 (10)[j]		10-2[k]	40-8[k]	6-10[l]

[a]Table values based upon known or estimated values from present experiments, possible ITER-era targets based on extrapolation from present experiments, and estimated reactor conditions from previous reactor studies.
[b]At Major Radius.
[c]Bootstrap + driven current.
[d]Power to plasma needed to maintain configuration, magnetic field, or plasma current.
[e]Mean values if not axisymmetric.
[f]Estimated by $\tau_{Ee} = 0.7\tau_{ITER98H}$; $\tau_{Ei} = 0.44\tau_{neo,i}$; $(W_e+W_i)/\tau_E = W_e/\tau_{Ee} + W_i/\tau_{Ei}$.
[g]$\chi \sim a^2/4\tau_E$.
[h]Deuterium beam ion at 80 kV.
[j]Maximum NSTX values provided in parentheses.
[i]τ_r is relevant time-scale for plasma current profile redistribution - D.R. Mikkelsen - Phys. Fluids B 1, 333 (1989).
[k]Range spanned for conventional and super-X divertor (SXD) configurations.
[l]Design value using a range of mitigation approaches.

CHAPTER 6

THE REVERSED FIELD PINCH

6.1. THE RFP CONFIGURATION

The distinctive feature of the reversed field pinch (RFP) that motivates its interest as a fusion energy system is that the externally applied magnetic field is relatively small. Current flowing within the plasma generates most of the confining magnetic field. Like the tokamak, the RFP configuration is toroidally symmetric with a helical magnetic field as shown in Fig. 6-1. However, in a tokamak, the toroidal field is externally applied and typically ten times the magnitude of the poloidal field. The magnetic field lines are therefore more tightly twisted in the RFP, measured by the safety factor, $q = rB_T / RB_P$. The tokamak operates with $q > 1$ (lower helical twist), while the RFP operates with $q < 1$ (higher helical twist). The higher twist yields increased magnetic shear, an important stabilizing influence on the plasma. This difference in magnetic topology makes the RFP a unique research platform to advance toroidal fusion energy sciences, exploring regions of toroidal configuration space complementary to the tokamak and other concepts.

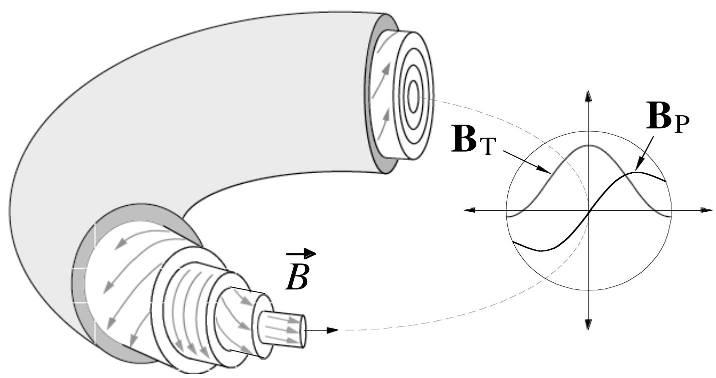


Fig. 6-1. The RFP magnetic configuration. The poloidal field inside the plasma, B_p, is comparable to the toroidal field, B_T, which is reversed compared to the small external toroidal field provided by the TF magnets.

Because the magnetic field is more concentrated within the plasma, the RFP offers a number of potential benefits for fusion application, including high engineering beta, the use of normal (rather than superconducting) magnets, high mass-power-density, efficient assembly, and possibly free choice of aspect ratio. The physics beta value, which measures the plasma pressure, p, with respect to the confining magnetic field pressure, $B^2/2\mu_0$, is automatically high since the toroidal field, B_T, is small. In present experiments the beta value measured using the average magnetic pressure within the plasma has been increased up to $\langle\beta\rangle = 2\mu_0\langle p\rangle/\langle B^2\rangle = 12\%$, a large value. A better indicator for efficient use of magnetic field for fusion energy application is the engineering beta, in which the magnetic pressure is measured at the magnet surface. For configurations with high safety factor the maximum field strength at the magnet is of order twice the field inside the plasma, whereas in the RFP the field at the magnet is less than inside the plasma. The engineering beta in an RFP fusion reactor might be as much as $\sim 4\langle\beta\rangle$. A large engineering beta implies superconducting magnets are not necessary, since the field strength is relatively low at the magnets. High beta also offers the potential for a reactor system with high ratio of fusion power to the reactor system mass (high mass-power density), an indicator of favorable economics and potential for a compact design that facilitates system assembly, maintenance and reliability. To date, the physics of the RFP does not depend strongly on aspect ratio. (There might be some benefit to low aspect ratio if a smaller number of unstable tearing modes exists, but this has not been explored in detail.) Hence the choice of aspect ratio can be made on engineering grounds.

These potential advantages have been validated through the comprehensive TITAN power plant system study, completed around 1990. However, the study was predicated on a set of physics and technology assumptions that are yet to be verified. A relatively small yet vital worldwide RFP program is making good progress addressing critical physics issues (and to some extent the technology issues), but the RFP is substantially less developed than the tokamak. One of the main challenges for the RFP is attaining reactor-relevant energy confinement. The limiting transport mechanisms need to be identified, and the scaling of confinement with plasma parameters needs to be determined. Another major challenge is the development of efficient sustainment of the plasma current, which will likely need to be provided using external sources (the self-generated neoclassical plasma current in the RFP is weak at fusion-relevant beta values). The plasma-boundary interface has not received much attention in the RFP. Although some aspects of the boundary are generic, the RFP's magnetic geometry may require unique solutions to particle and heat control. Also, to enable the potential of any compact reactor system, significant engineering and material science advances are required to manage the implied large heat and neutron loads on the wall. These and the other primary scientific and technical issues are discussed in Section 6.3.

The RFP provides a significant opportunity to advance fusion and plasma science more generally. The tokamak provides a fairly broad range of operation for safety factor $q > 1$, but the approximate lower bound on q is a strong restriction. Generally, the RFP (with $q < 1$) provides new information since it extends our understanding to low field strength, testing the understanding derived at high field. Some examples of important scientific contributions from the RFP research include: (1) electrostatic transport with weak externally applied magnetic field and large magnetic shear (extending the knowledge base from $q > 1$configurations); (2) active control of multiple resistive wall instabilities; (3) beta limits; (4) magnetic self-organization and its control: and (5) linking fusion energy science to astrophysics.

The research mission for the RFP as a fusion energy system is to

RFP mission: Develop the scientific and technical basis for a fusion power source that uses a small externally applied magnetic field

The sections below identify the RFP goal for the ITER era, issues that must be resolved, and the research gaps that exist in resolving these issues.

6.2. RFP GOAL FOR THE ITER ERA

The panel has identified the goal for RFP research in the ITER era as

RFP goal: Establish the basis for a burning plasma experiment by developing an attractive self-consistent integrated scenario: favorable confinement in a sustained high beta plasma with resistive wall stabilization.

This goal derives from recent research that has demonstrated high beta RFP plasmas with improved confinement in transient conditions. The next step for the ITER era is to maintain improved confinement at high Lundquist number using current drive methods that extrapolate to either steady-state or long-pulse high-gain fusion scenarios. [In the RFP, the Lundquist number, $S = \tau_R / \tau_A$, is related to the plasma current, temperature, and density, $S \sim I_p T_e^{3/2} / \sqrt{n}$. It is the primary dimensionless scaling parameter for plasma models based on resistive magneto-hydrodynamics (MHD).] Developing current sustainment consistent with improved confinement is the highest priority challenge to achieve a self-consistent integrated scenario. Identification of important transport mechanisms and confinement scaling will be a major science objective, both in scenarios involving magnetic self-organization and in scenarios utilizing current profile control. Active stabilization of resistive wall modes is expected to be essential for the pulse durations necessary for this research. Research toward these goals will be coordinated with the international RFP program.

Scientific Contributions:

Achieving the above goals will contribute significantly to fusion energy sciences and broader scientific disciplines. Below we elaborate a few such examples:

- **Electrostatic Transport at Weak Field:** RFP plasmas in which confinement is improved by suppressing magnetic fluctuations represent a new physics regime dominated by electrostatic fluctuations. Understanding electrostatic transport in such plasmas, for which magnetic shear and the gyroradius are relatively large, would extend and test the knowledge base acquired from tokamaks (at high externally applied field). The scaling of confinement in these plasmas is a new area of study.
- **Resistive Wall Instabilities:** The RFP is susceptible to multiple resistive wall instabilities even at zero beta. Thus, RFP research has and will continue to develop feedback techniques for multiple mode stabilization directly applicable to other configurations.
- **Beta Limits:** All RFP experiments operate at high beta, and recently beta values have been achieved that exceed theoretical MHD stability limits for localized interchange and global tearing modes. The RFP is an excellent test bed for understanding the behavior of high beta plasmas, including those that exceed MHD stability limits (local or global).
- **Magnetic Self-Organization:** In its standard regime (without confinement improvement via current profile control) the RFP exhibits a set of phenomena that are associated with magnetic self-organization. Particularly relevant to many configurations are RFP studies of reconnection, dynamo alteration of the current density profile, momentum transport, reconnection heating of ions, plasma transport from magnetic stochasticity, and magnetic helicity transport. These effects are strongly related to each other, so that understanding the individual phenomena sums to a general understanding of magnetic self-organization as occurs in the tokamak under certain conditions, in the spheromak generally and in the RFP. The RFP program is arguably unique in this endeavor.
- **Linking Fusion Energy Science to Astrophysics.** Through magnetic self-organization, RFP physics has strong links to related phenomena in astrophysical plasmas. Through funded collaborations with plasma astrophysicists, RFP researchers have been applying understanding gained in the laboratory to astrophysics (as well as applying physics learned through astrophysical studies to the RFP).

6.3. SCIENTIFIC AND TECHNICAL ISSUES FOR THE RFP

The following scientific and technical issues have been identified for the RFP. These issues have been prioritized in three tiers through the process described in Chapter 3.

Tier 1:

- **(1) Identify Transport Mechanisms and Establish Confinement Scaling.** Stochastic magnetic turbulence associated with global MHD tearing instability is the main transport mechanism in RFP plasmas without current density profile control. This leads to relatively poor confinement. The theoretical MHD understanding of tearing and magnetic self-organization is advanced (though incomplete), and a variety of experimental observations are consistent with stochastic transport expectations. This motivated application of current density profile control to reduce tearing instabilities, which experimentally yields a (transient) ten-fold improved confinement, within a factor of 2-3 of confinement for a tokamak of the same size, current, and heating power. Non-stochastic transport is likely important in this new confinement regime.

The scaling of the magnetic turbulence with Lundquist number, $S = \tau_R / \tau_A$, is especially important for scenarios involving magnetic self organization, e.g., use of Oscillating Field Current Drive (OFCD). A large gap in Lundquist number exists between present day experiments and burning plasma conditions. Also, two-fluid physics effects are increasingly important at low collisionality, requiring theoretical extension beyond MHD. A possible self-organized branch is the single-helicity state, with one dominant tearing mode and implied small stochasticity. Recent experiments suggest that a natural tendency toward the single-helicity state might occur at high S. Active stimulation of the single-helicity state might also be possible via magnetic control at the plasma boundary.

In RFP plasmas with current density profile control and reduced tearing modes, greatly improved energetic electron confinement indicates stochastic magnetic transport becomes less dominant. This is largely an unexplored confinement regime for the RFP, both experimentally and theoretically. For example, the theory for drift wave turbulence should be extended to the RFP, where magnetic configuration differences such as low safety factor and reversed-shear will affect the prediction for turbulence and transport. The physics associated with ρ^* scaling and a connection to empirical scaling in high safety factor configurations might be expected if electrostatic turbulence is dominant in this new confinement regime.

This issue is critical to resolve. The transport mechanism that limits confinement in the RFP is not known, and confinement scaling needs to be determined. The Lundquist number in present experiments, $S = 10^{5\text{-}7}$, is significantly less than $S = 10^{10}$ expected for burning plasma conditions. Likewise, the normalized gyroradius, ρ^*, will be ten-fold smaller at burning plasma conditions. Improved understanding of stochastic transport and electrostatic transport in low-q plasmas will have broad impact on fusion science.

- **(2) Current Sustainment.** It is expected that the plasma current in the RFP must be driven by an external source. The RFP's weak toroidal magnetic field implies that the parallel neoclassical bootstrap current tends to be small, and 100% non-inductive current drive by RF or neutral beams is generally considered impractical because the current drive efficiency is low. A remote possibility might be large bootstrap current at ultra high beta, but this challenges theoretical stability limits.

  Inductive current drive is very efficient and could permit RFP reactor scenarios with high fusion gain. There are two possibilities: long pulse toroidal induction and steady-state OFCD. There are no known physics issues with toroidal induction with respect to current drive. For example, the parallel electrical conductivity of RFP plasmas appears neoclassical (the Spitzer value decreased roughly two-fold by the trapped electron correction).

  OFCD is inductive but unconventional. Purely AC toroidal and poloidal loop voltages are applied in quadrature, and plasma relaxation is expected to maintain the current density profile near marginal tearing stability. Since no DC magnetizing flux accumulates, the current drive is steady-state. The physics basis for OFCD has been established in nonlinear resistive MHD computational studies, and experimental tests have demonstrated 10% current drive (the remainder by background toroidal induction).However, there is uncertainty that OFCD will work as expected at parameters beyond those presently accessible in MHD computation, and full sustainment must be demonstrated experimentally.

Although OFCD would provide steady-state DC current, an unavoidable AC current modulation accompanies the AC loop voltages simply from inductive response. This modulation is a concern if too large, since dynamic changes in the magnetic equilibrium during the course of the OFCD cycle might permit instability otherwise not present, e.g., additional resonant tearing modes. The AC modulation scales with plasma resistance, or inverse to the Lundquist number.

This issue is critical to resolve. Optimization of the RFP is likely to be strongly affected by the differences between steady-state or pulsed operation. The OFCD concept is novel, and although the fractional current drive by OFCD obtained to date is encouraging, full sustainment is yet to be demonstrated. An extension of Lundquist number from $S=10^{5-6}$ in present experiments to $S>10^{7-8}$ is expected necessary to avoid exceeding equilibrium extremes of established RFP operation.

- **(3) Integration of Current Sustainment and Improved Confinement.** Any fusion concept must have a self-consistent or integrated set of solutions for all issues. In the RFP, the first order integration requirement is efficient current sustainment with good confinement. To illustrate the challenge, toroidal induction, although very efficient, tends to create a peaked current density profile which is MHD tearing unstable. This causes the magnetic field to become stochastic, thereby degrading confinement. Current density profile control for tearing stability improves confinement, but the current drive techniques must ultimately be efficient enough to project a high fusion gain system.

  A possible strategy is to separate in time the building of the plasma current from the time when fusion burn occurs. An example is a hybrid inductive scenario using OFCD to build the current and a self-similar current ramp-down to maintain good confinement, a particular prescription for inductive current profile control. Confinement during the OFCD phase need only be good enough for efficient current drive. This is a type of pulsed scenario, but the plasma current never goes to zero, and the gap in fusion power generation could be order seconds instead of minutes.

  The issues for current density profile control are current drive efficiency, localization, and mixture of current drive types. Inductive current profile control is efficient but inherently not steady-state nor localized. Non-inductive current drive could provide localized and stationary control, but the current drive efficiency is generically lower. In addition to issues such as RF accessibility, determining the amount of non-inductive current that must be provided for tearing stability is crucial.

  Since OFCD depends on plasma relaxation, which might require too large magnetic fluctuations, a key issue is to determine the Lundquist number scaling of the magnetic turbulence and/or establish access to the single-helicity state during OFCD operation. This is analogous to S-scaling for long pulse toroidal induction without current profile control. It is not clear if current profile control can be coupled with OFCD, since that might negate the relaxation process on which OFCD is based.

This issue is critical to resolve. The viability of the RFP will be greatly enhanced if good confinement can be combined with current sustainment methods that extrapolate to fusion-relevant scenarios. Some techniques that are important to this issue, such as self-similar ramp-down and RF current profile control, are only beginning to be tested.

Tier 2:

- **(4) Plasma Boundary Interactions.** The majority of RFP experiments to date have been circular cross-section limited plasmas. Sufficient wall protection has been provided to handle wall heat loads that can reach ~2 MW/m^2 locally, but particle and impurity control have not received much attention. Boronization and similar coating techniques have been used successfully but only occasionally. The lack of advanced boundary control has not severely hampered research to date, but the RFP's state of development might be similar to the tokamak era where attention to boundary control had an enormous influence on plasma behavior.

  Since the poloidal field is dominant in the RFP, a poloidal divertor may not be the best option. Other options for particle/impurity control that have been discussed include a toroidal (bundle) divertor, pumped limiters, or a liquid wall. The later two would be less perturbing to axisymmetry. For reference, the TITAN system study employed a toroidal divertor and assumed very high fraction radiated power from the core and edge.

  Since the best means to control the boundary in the RFP is unknown, the issue is to evaluate and test options. The large database on boundary control in tokamak research will be a valuable resource, but the techniques (e.g., poloidal divertor) may not be directly transferable. This is clearly an opportunity for research, but the existing RFP facilities are limited in ability to test options.

  This issue is important to resolve. Although some aspects of the plasma-boundary interface are generic, it is likely that particular solutions suited for the RFP will be required. Given past experience in other magnetic configurations, control of the boundary (both particle and heat flux) will likely have large impact on RFP performance, but it may not be essential to make good progress on the Tier 1 issues.

- **(5) Energetic Particle Effects.** Short-pulse neutral beam injection into RFP plasmas has demonstrated good energetic ion confinement, even in the presence of magnetic stochasticity. The single-particle confinement of alpha particles with similar normalized gyroradius should therefore not be an issue. However, energetic ions with speed comparable to the Alfven speed can potentially cause instabilities often called Alfven eigenmodes, as observed in other magnetic configurations. In the RFP, these instabilities are little studied both theoretically and experimentally. Toroidicity-induced eigenmodes should be possible, although configuration details like smaller safety factor and magnetic shear are specific to the RFP. Existing theoretical tools should be straightforward to adapt for the RFP configuration to predict and describe possible energetic particle modes.

  Experimentally there is little evidence for Alfven eigenmodes. Such instabilities require a larger fraction of energetic ions, attained experimentally by neutral beam injection and eventually by fusion products in burning RFP experiments. If such modes occur in a reactor, they could cause alphas to be lost at high energy, or they couple to the thermal distribution and degrade confinement. Theoretical and experimental assessment of energetic ion effects is therefore an essential step towards a successful RFP burning plasma experiment.

This issue is important to resolve. As for other magnetic configurations, the RFP is likely to be susceptible to energetic particle driven instability. Minimal research has been performed in this area, both theoretical and experimental. Neutral beam injection which creates fast ions satisfying $V_{ion}/V_A > 1$ could provide the capability to begin experimental investigation.

- **(6) Determine Beta-Limiting Mechanisms.** Although the poloidal beta value ~20% required for an attractive RFP reactor has already been achieved (simultaneous with improved confinement), understanding beta-limiting mechanisms is needed to reliably project performance in a burning plasma. Theoretically, at poloidal beta ~20% pressure-drive for tearing and interchange are predicted to become important mechanisms. Experimentally, auxiliary heating is needed to adjust beta independent of other parameters like energy confinement. Also, optimizations of the plasma cross-sectional shape, pressure and current profiles, and aspect ratio have not been completely assessed, even theoretically. At very high beta, if achievable, the bootstrap current becomes sizable. This could help reduce the requirement on efficient external current drive.

  This issue is important to resolve. Since the mechanism(s) that limit beta are not identified, it is not certain the demonstrated high beta value can be maintained as the RFP progresses toward burning plasma conditions. Existing experiments are able to access the theoretical thresholds for resistive MHD pressure-driven instability, but not the ideal MHD limit of beta ~ 50%.

Tier 3:

- **(7) Self-Consistent Reactor Scenarios.** In addition to the integration of current sustainment and improved confinement [Issue (1)-(3)], the solutions to other issues such as RWM stabilization and boundary control must be mutually compatible. So far active control of RWM instability does not exhibit strong coupling to confinement, for example, but this is a recent achievement and issues could arise as the pulse length and sustained good confinement periods are extended. Boundary control is not well developed, and cross-issue linkages could occur, for example non-axisymmetry effects on plasma stability if a toroidal divertor is employed, or impact on confinement if a highly radiating core and mantle are required to achieve a compact configuration.

  This issue is important but less urgent. Beyond integrated good confinement and efficient sustainment (a Tier 1 issue), more fundamental and targeted research is needed in other areas critical to an overall self-consistent integrated scenario.

- **(8) Optimization of RWM Control for a Fusion Environment.** It has long been established through MHD theory and computation that multiple current-driven ideal instabilities arise in the RFP in the absence of a surrounding conducting shell. In addition, computation reveals that the resistive tearing modes grow without bound in the absence of a shell (whereas with a shell they reach a saturated level). Experiments have validated these expectations. With a resistive shell, it is observed that all the expected modes grow on the time scale of the shell resistive diffusion time. Eventually, the instabilities terminate the plasma (for finite applied loop voltage), also in agreement with computation.

  However, it has now been demonstrated in experiment with a resistive shell that all of these instabilities (of order 10-20 modes) can be suppressed (with the tearing modes held to their conducting wall value) through active feedback using saddle coils covering the shell surface. Thus, in a physics sense, this long-standing RFP problem has been largely solved. Nevertheless attention to RWM control and near term optimization cannot be neglected, since it is likely that the pulse lengths required to fully address other issues will exceed the pulse length provided by passive thick-wall stabilization.

In the long term it is necessary to develop a scenario that is most compatible with the engineering of an attractive reactor. This requires determination of the allowed flexibility in the feedback system, such as the required proximity of the coils to the plasma surface, the number of coils, and the extent of surface coverage with coils.

This issue is important but less urgent. While ongoing optimization of resistive wall mode control will be essential for experimental work at pulse lengths necessary to achieve the ITER era goal, full optimization of RWM control suited for a reactor environment is lower priority.

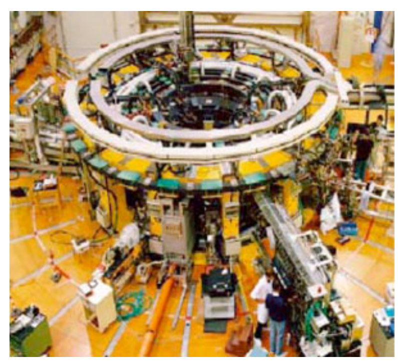
6.4. RESEARCH FACILITY, GAPS AND INITIATIVES TO ACHIEVE THE RFP GOAL

6.4.1. Available Facilities and Tools

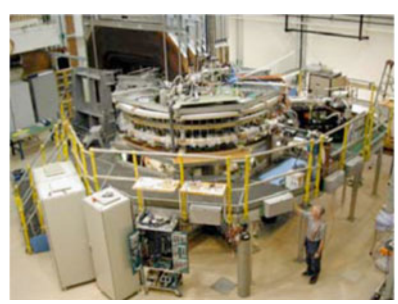
- **Experimental Facilities.** There are four complementary RFP experiments operating in the world. These facilities are described below in terms of how they are being used to address key RFP issues. Their existing experimental capability is summarized.

  — The MST Facility (UW-Madison). The MST facility is the centerpiece of the US proof-of-principle RFP program. It is physically large in the RFP context (R/a = 1.5/0.5 m), but has medium plasma current (0.5 MA) and pulse length capacity (< 0.1s). The MST program focuses on confinement and beta studies, through current profile control and auxiliary heating. It is also testing the OFCD concept for steady-state current sustainment, but at a fractional current drive level (partial OFCD superposed on regular toroidal induction).

  — The RFX-mod Facility (Italy). The world's highest power RFP facility is RFX-mod (R/a = 2.0/0.46 m), with a designed plasma current capability of 2 MA and pulse length to date of~0.5 s. The plasma is surrounded by a metal shell with a vertical field penetration time of 50 ms. The centerpiece of a recent facility upgrade is a 192 RWM coil system which fully covers the 2D toroidal surface for active and broadband MHD control. This program extends optimization of the quasi-single-helicity configuration.



  — The Extrap-T2R Facility (Sweden). The Extrap-T2R (R/a = 1.24/0.18 m, 0.3 MA) is a moderate size RFP which, like RFX-mod, employs a full coverage active feedback coil system totaling 128 coils and independent power supplies. Active control of all resistive wall modes has been demonstrated for 0.1 s, which corresponds to 15 wall-times. Identifying the minimum required number of control coils for RWM control, and improved understanding of mode-locking to the wall are main research themes.



— The RELAX Facility (Japan). A new, low aspect ratio RFP has recently begun operation at the Kyoto Institute of Technology. This is a much smaller scale facility (R/a = 0.51/0.25 m, I < 100 kA, pulse length < 10 ms) than those above. At low aspect ratio, the safety factor is increased, and the spatial separation of major resonant surfaces is therefore increased, possibly impacting plasma relaxation behavior. A recent finding from RELAX is that the quasi-single-helicity state is achieved at a much lower plasma current than at the normal aspect ratios.

- **Available Tools for Modeling and System Studies.** Extensive nonlinear resistive MHD computation has been performed to describe the multimode magnetic fluctuations in the RFP, and such codes (e.g., DEBS) are in active use. Nonlinear two-fluid studies are underway, particularly using the NIMROD code. PIC simulations of reconnection related to the RFP (in simplified geometry) are underway using a code written for astrophysical plasmas. Gyrokinetic simulation to examine electrostatic turbulence has begun using codes developed for tokamaks. Fokker-Planck studies have been performed using the CQL-3D code. At present, a modest, targeted power plant system study (essentially an update to the TITAN study) is beginning.

6.4.2. Available Means, Research Gaps and Needed New Capabilities

- **(1) Transport Mechanisms and Confinement Scaling.** Understanding transport and confinement scaling in the RFP requires an extension of physics parameter space, most importantly in Lundquist number and normalized gyroradius. In the limit where stochastic magnetic transport is dominant, the scaling of the magnetic turbulence with Lundquist number is crucial. One of the key questions here is whether or not the magnetic self-organization spontaneously transitions from a multi-helical state (many tearing modes) to a single-helicity state (one dominant tearing mode). It may also be possible to stimulate the single-helicity state by external control, for example using a non-axisymmetric magnetic boundary condition. In the limit where stochastic magnetic transport is minimized, evidence suggests electrostatic transport may be most important. This limit has begun to be accessed using current profile control, but it would also occur in RFP plasmas undergoing magnetic self-organization if the magnetic turbulence becomes small at high Lundquist number. If electrostatic transport is dominant, it is reasonable to expect that the main controlling parameters will be the normalized gyroradius and collisionality, as observed in high safety factor plasmas in which electrostatic turbulence is the main cause of transport. Nonlinear resistive MHD computation is the main theoretical tool for understanding magnetic self-organization and its dependence on Lundquist number.

  — Available Means. Presently MST operates to I_p=0.5 MA and RFX operates to I_p=1.5 MA, with 2.0 MA expected in the near future. The achievable range of Lundquist number is S=10^{5-7}. The normalized gyroradius is $\rho^* \approx 0.02$ in present plasmas where the magnetic turbulence is reduced via current profile control and electrostatic turbulence may play an increasingly important role in transport.

— Gap. The achievable range in Lundquist number in present facilities is $S=10^{5-7}$compared to the ITER era target of $S=10^{8-9}$. The normalized gyroradius for the ITER era target is also 3 times smaller than for present plasmas with reduced magnetic turbulence. There also exists a large gap in computation and theory: single-fluid computation in cylindrical geometry has been used to study RFP dynamics with Lundquist number $S \leq 10^6$, compared to the RFP goal values of $S=10^{7-8}$ where two-fluid physics is considered to be important. Gyrokinetic calculations are only now beginning to be applied to the RFP geometry.

— Needed New Capabilities. Extending the Lundquist number range and decreasing the normalized gyroradius to their ITER era target values is required, but this could be achieved in two steps to reduce cost and technical risk: first using an advanced proof-of-principle facility with sufficient plasma current, control tools, and pulse length to attain $S=10^{7-8}$, then later employing a performance-extension facility to attain $S=10^{8-9}$. To improve understanding and to guide experiments, single-fluid computation should be extended to higher Lundquist numbers, including two-fluid physics at a proper stage. Gyrokinetic computation in RFP configuration is important to predict and understand residual electrostatic transport in plasmas with reduced magnetic turbulence.

- **(2) Current Sustainment.** The demonstration of Oscillating Field Current Drive (OFCD) requires sufficiently high Lundquist number and pulse duration. A primary concern is that the magnetic equilibrium modulation associated with the required AC loop voltages must be small enough to avoid unacceptably large dynamic variation of the magnetic equilibrium during the course of an OFCD cycle. Present OFCD experiments operate with $S=10^{5-6}$, while values of $S=10^{7-8}$ are required to maintain the oscillation in the safety factor within the range of established RFP operation. Exceeding the standard range of safety factor could introduce effects such as additional resonant tearing modes that would not normally be present. It is also necessary that the plasma (and OFCD) duration be at least comparable to the plasma's L/R current relaxation time, so that the full effect of the current drive is established. Note that the L/R time also increases with Lundquist number. Nonlinear resistive MHD or two-fluid computation is a major research tool to address this issue.

— Available Means. MST has power supplies specially built to provide the large AC loop voltages required for OFCD experiments, but its modest $S=10^{5-6}$ and relatively short pulse duration (<0.1s) will likely limit the current fraction driven by OFCD to ~20%. The power supplies for RFX are less capable than MST for producing large AC loop voltages, but the pulse length for RFX is much longer than for MST. Single-fluid computation is available to study OFCD at moderate Lundquist numbers in cylindrical geometry.

— Gap. The Lundquist number will need to be $S=10^{7-8}$ to maintain small enough AC modulation for a 100% OFCD demonstration, and the pulse length will need to be comparable or greater than the L/R time for saturated current drive. A gap in the capability for nonlinear resistive MHD computation also exists in reaching values of $S=10^{7-8}$ where two-fluid physics is considered to be important.

— Needed New Capabilities. Upgrades to the existing RFP facilities would be useful to demonstrate larger fraction current drive by OFCD. This permits assessment of critical physics issues and validation of theoretical models with gradual investment. However, an advanced proof-of-principle facility that provides access to high Lundquist number and long pulse is required to demonstrate full OFCD sustainment. As noted above in the context of understanding transport, the combined computational capability with higher S and two-fluid physics is required to understand physics and guide experiment.

- **(3) Integration of Current Sustainment and Good Confinement.** Current sustainment for the RFP could be steady-state using OFCD, quasi-steady-state using a hybrid combination of OFCD and inductive current profile control, or conventional pulsed induction. Achieving good confinement in at least one of these cases is essential, built on the success of resolving Issues (1) and (2) described above. Numerical simulations can provide much-needed guidance in the experiment.
  - — Available Means. Optimization of inductive current profile control is ongoing. The assessment of confinement using steady induction at Lundquist number $S=10^{5\text{-}7}$ is also ongoing, with indications that a single-helicity state may be the natural self-organization process at high S. MST is able to begin assessment of confinement with OFCD current drive fraction up to 20% (presently 10%). Non-inductive RF current profile control tools are being assessed, based on lower hybrid and electron Bernstein waves, each presently at ~ 200 kW level (too small for effective current profile control). Neutral beam injection might be able to provide some additional current drive. Numerical investigations can be performed by single-fluid codes at moderate Lundquist numbers.
  - — Gap. The scaling of confinement with OFCD current sustainment cannot be fully addressed until 100% current drive fraction is demonstrated. Also, demonstration of the nearly steady-state hybrid using OFCD and inductive current profile control based on a self-similar ramp-down requires 100% OFCD current drive. Numerical simulations currently do not reach the Lundquist numbers specified for the RFP goal, nor employ the necessary two-fluid physics. Gyrokinetic simulations are not available to study compatibilities of the electrostatically driven transport with current sustainment.
  - — Needed New Capabilities. An advanced proof-of-principle facility that provides access to high Lundquist number and long pulse is required for assessment of confinement with full OFCD sustainment. Such a facility should also be adept in current profile control techniques, for example self-similar ramp-down and RF techniques if their efficacy and efficiency are sufficiently good. Numerical capabilities for single-fluid and two-fluid modeling at large S and gyrokinetic simulations are required.
- **(4) Plasma Boundary Interactions.** This is a very much underdeveloped area for the RFP research. Although some knowledge is transferable from research in other concepts, significant parts of this area are unique for RFP.
  - — Available Means. The existing RFP facilities are not well equipped to investigate advanced control of the plasma boundary. They employ basic limiter structures to protect the wall from plasma interaction, and particle control is maintained by fueling adjustments and limiter conditioning, occasionally using a thin boron coating (in situ application).
  - — Gap. The existing facilities are also not easily modified to investigate magnetic divertor geometries, although components for other strategies like pumped limiter or liquid walls might be testable.
  - — Needed New Capabilities. New devices or major upgrades will be required to adequately address the plasma-boundary issue. This research could begin in a smaller device at the concept exploration level. The ohmic heating for smaller RFP plasmas is substantial due to large plasma resistance, helpful to efficiently attain heat flux levels relevant to future

devices. It will be vital to develop boundary control solutions that are consistent with requirements for confinement, sustainment, and resistive wall mode control, and therefore significant boundary control capability should be considered for any new device. Scenarios with high radiation fraction will be beneficial, for which compatibility with confinement requirements is particularly important.

- **(5) Energetic Particle Effects.** This is another largely unexplored area for the RFP research. Although some knowledge, either experimental or theoretical, is transferable from research in other concepts, significant parts of this area are unique for RFP.
  - — Available Means. The planned installation of powerful neutral beams in MST and RFX-mod should be able to provide the first test beds for the effects of energetic particles in the RFP. The beam is injected tangentially in MST while mostly perpendicularly in the RFX-mod, providing complementary geometries for the interactions between energetic ions and background thermal plasmas and Alfvénic modes. Existing codes in tokamaks, such as NOVA and ORBIT, might be suitable for this research after some adjustments.
  - — Gap. However, these studies will be limited to cases where the ion speeds are comparable to Alfven speeds and energetic ion populations are small. The adjustments for tokamak codes may be nontrivial depending on the code design and detailed physics outcomes.
  - — Needed New Capabilities. Installation of neutral beams at larger energies and currents are required to study the effects of energetic particles with speeds much faster than the Alfven speed with significant populations. Theoretical and numerical tools, such as NOVA and ORBIT, require adaptation for the RFP configuration.
- **(6) Determining Beta Limiting Mechanisms.** Understanding beta limiting mechanisms is important to reliably project performance in a burning plasma. Control of the plasma pressure profile may become important to avoid instability and maintain the high beta important to achieve the ITER-era goals. Close interplays between experiments and theory/simulation based on 3D nonlinear computation with finite pressure are crucial in addressing this issue.
  - — Available Means. There is a reasonable likelihood that present experiments have the capability to determine the beta limit at moderate Lundquist numbers. This requires increasing beta from its present value. Two complementary techniques are underway: pellet injection and neutral beam injection. Both techniques have limitations in present experiments. Pellet injection into improved confinement plasmas increases density and beta. But it is not known whether this will continue to the beta limit. Neutral beam injection soon to be available in current experiments (~1MW for 20-30 ms) will enhance beta through the added pressure of the fast ions. However, the plasma parameters (energy confinement time, fast ion slowing down time, and plasma duration) make fast ion thermalization difficult, most likely limited to higher density operation provided by pellet injection. Single-fluid codes with finite pressure are available and already in use at moderate Lundquist numbers.
  - — Gap. One of the expected pressure-limiting instabilities is resistive MHD modes that depend on S, and extension of beta limit studies to high Lundquist number is not available at present facilities. Higher Lundquist number may also permit larger confinement times, which generally makes access to high beta easier. Computation using single-fluid codes at larger Lundquist numbers and two-fluid physics is currently not available.

— Needed New Capabilities. Testing beta limits at larger S available at an advanced proof-of-principle facility and a subsequent performance-extension facility is essential in achieving the RFP goal. Extended capability for computational tools at larger S and with two-fluid physics is also required.

- **(7) Self-Consistent Reactor Scenarios.** Beyond improved confinement and current sustainment, a self-consistent scenario must also integrate RWM control and plasma boundary control to achieve high performance RFP discharges and establish the knowledge base for a burning plasma experiment. System code studies combined with 3D computation with realistic boundary conditions likely plan an important role in addressing this issue.

  — Available Means. No experimental facilities are available to address this issue. The existing TITAN system code could provide some insights with probably limited capabilities.

  — Gap. Current facilities are very limited in their ability to investigate self-consistent plasma scenarios. For example, RFX-mod can examine resistive wall instabilities in depth, but lacks strong capability for current profile control, beta enhancement, and oscillating field current drive. MST can apply current profile control and possibly enhance beta, but cannot include reactor- relevant resistive wall mode stabilization (MST operates with a conducting shell) and is limited in OFCD. No facility exists to test plasma boundary control using divertor concepts.

  — Needed New Capabilities. The development of self-consistent scenarios would require a facility that includes all the aforementioned capabilities, probably at the performance-extension level. Further development of system code in coordination of 3D nonlinear fluid computations, which incorporate toroidal divertor, are required to develop, predict and optimize self-consistent reactor scenarios.

- **(8) Optimization of RWM Control for a Fusion Environment.** As an important but less urgent issue, RWM control needs to be optimized to be compatible with the engineering of an attractive reactor.

  — Available Means. The RFX-mod and Extrap T2R facilities have sufficient flexibility for extensive study of feedback stabilization of resistive wall instabilities. The feedback coil network can be configured to test various combinations of wall coverage – each of feedback coils are independently controlled.

  — Gap. Next steps beyond existing capability include identifying the required proximity of coils to the plasma surface and studying the effects of plasma rotation, for example by tangential neutral beam injection.

  — Needed New Capabilities. Testing feedback coil proximity and plasma rotation might be possible through extensive modification of existing facilities, or perhaps better suited to a new, smaller facility optimized for this purpose. Ultimate integrated tests will need to be demonstrated in a performance-extension level facility at a later stage.

6.4.3. Required New Facilities and Tools

- **Experimental Facilities.** The resolution of the issues and gaps described above translate into two categories of experimental facilities, properly staged in time, in order to achieve the ITER-era goals for the RFP. The first includes facilities that address a subset of issues that require a broader parameter space than presently available, or in a unique setup unavailable in the existing facilities. The scales of these facilities can be either on the proof-of-principle (but beyond MST and RFP-mod capabilities) or concept-exploration levels. The combined physics parameters from these facilities should be sufficiently advanced to be able to establish the basis for the second category of facility - one that establishes an integrated scenario at advanced parameters that will form the basis for a burning plasma experiment. Thus, the second category of facility should be a performance extension experiment.

From the above section, we observe that resolving issues of confinement, current sustainment and their integration [Issues (1)-(3)] require facility capabilities beyond those presently available. These topics require expansion in physics parameter space that is accomplished by higher current and longer duration than in present experiments exploring improved confinement and OFCD. Two dimensionless parameters can be identified as important for confinement. The Lundquist number is expected to determine, in part, magnetic fluctuations. The normalized gyroradius is expected, in part, to determine electrostatic fluctuations (although this is not yet a well-established expectation for the RFP). A significant excursion in these two dimensionless parameters is desirable. Oscillating field current drive also requires an expansion in Lundquist number so that the voltage swing is acceptably small. The issues on beta limits [Issue (6)] and energetic particle effects [Issue (5)] can be studied at extended parameter space, built on the knowledge from studies on the existing facility.

An example of a facility (at the proof-of-principle level) that satisfies all the above criteria is an RFP with similar size as MST but current increased to 1.5 MA and plasma duration increased to about 0.2 s. For illustration of confinement expectations, consider three scaling scenarios. The confinement time would increase by a factor of 3 if the scaling is as for a tokamak dominated by electrostatic fluctuations (perhaps the most promising regime for the RFP), by a factor of 5 if the scaling is for the historical constant beta RFP that corresponds to a strong dependence of magnetic fluctuations on Lundquist number, and almost not at all if the magnetic fluctuations depend weakly on Lundquist number as indicated in limited scaling studies of the standard RFP (with multiple tearing instabilities). The Lundquist number would increase about ten-fold to 10^7 for standard RFP operation, sufficient for a strong test of OFCD with acceptably small oscillations. The normalized gyroradius would decrease about three-fold in improved confinement plasmas, suitable for examining scaling of electrostatic transport. Although the plasma current of the facility is similar to that of RFX-mod, it would have added capability in inductive current profile control, OFCD, neutral beam injection, and RF wave injection.

A separate facility is likely needed to explore methods to control the plasma-wall interaction [Issue (4)], possibly at the concept-exploration level. For example, an RFP with a toroidal magnetic divertor or with a liquid lithium boundary (pending results elsewhere) would open up a new area of RFP research. Tests of such concepts could then be extended at the relatively modest scale (in plasma current) of MST.

The experiment to demonstrate an integrated scenario [Issue (7)] should operate with improved confinement, resistive wall mode feedback stabilization [Issue (8)], either oscillating field current drive or an alternative pulsed scenario, and appropriate control of the plasma-wall interaction. The physics parameters should be such that the extrapolation to a burning plasma experiment can be done with confidence. The specification of such parameters requires results from the above first category facilities, but would be at the level of a performance extension experiment (e.g., temperature > 5 keV, plasma current > 4 MA).

- **Tools for Modeling and System Studies.** The required new computational tools include 3D nonlinear codes based on two-fluid and gyrokinetic formula. Some of these codes can be an extension of the existing codes from tokamak community, and some may require addition of unique RFP geometry such as toroidal divertor. The TITAN system code may require modernization to include a few new operational scenarios.

Table 6-1 summarizes the needed facilities and computational tools. Table 6-2 provides a list of key parameters for the RFP concept for reference.

Table 6-1
Facilities and Computational Tools

Issues	Existing Facilities		Needed Facilities			Existing Theory, Comutation, Modeling	Needed Theory, Computation, Modeling
	MST	RFX	Advanced Control PoP	CE's	PE (ITER-era Goal)		
Identify transport mechanisms and establish confinement scaling	currennt profile control, S-scaling at Ip<0.5 MA	S-scaling, single-helicity state, at Ip<2 MA	current profile control, S-scaling, single-helicity state, at Ip<2 MA	single-helicity state at low aspect	establish base for burning-plasma experiment	3D nonlinear MHD, test-particle stochastic theory	3D nonlinear 2-fluid, gyrokinetics ($S=10^8$)
Current sustainment	partial OFCD, pulsed	pulsed	OFCD, pulsed, and hybrid-inductive	optimize bootstrap current at low aspect ratio	OFCD, pulsed, or hybrid-inductive	3D nonlinear MHD ($S=10^6$)	3D nonlinear 2-fluid ($S=10^8$)
Integration of current sustainment and improved confinement	extend current profile control duration (inductive, rf)	S-scaling, single-helicity state (pulsed)	long-pulse profile control, S-scaling for OFCD/pulsed	limited capability	integrated high performance, long pulse	combine above	combine above
Plasma boundary interactions	limiter	limiter	begin with limiter, explore divertor and other concepts at a later stage	test options: toroidal divertor, pump limiter, liquid wall, etc	integrated control	relevant capability from larger fusion community	adapt for RFP configuration
Energetic particle effects	tangential beam injection, substantial fast ion beta, $v_i/v_A \geq 1$	perpendicular beam injection, $v_i/v_A \leq 1$	beam injection with $v_i/V_A >> 1$	limited capability	fully assess energetic particle effects	Relevant codes, e.g. NOVA, ORBIT, etc. exist	adapt for RFP configuration
Determine beta-limiting mechanisms	possibly identify at Ip<0.5 MA	possible with further improved confinement	identify at Ip<2 MA	limited capability	verify and control at high performance	3D nonlinear MHD with finite pressure	3D nonlinear 2-fluid
Self-consistent reactor scenarios	limited capability	limited capability	limited capability	limited capability	substantial capabilities	TITAN system study codes	3D nonlinear model in realistic geometry (e.g., toroidal divertor)
Optimization of RWM control for a fusion environment	not accessible	RWM physics, partial engineering optimization	as needed, to decouple from other issues	investigate geometric constraints for reactors	integrated control	linear and 3D nonlinear MHD	none known

Table 6-2
RFP Key Parameters[(a)]

Parameter	Present Value MST/RFX-mod[(a)]	ITER-Era Goal[(b)]	Reactor Target[(c)]
Field at magnetic axis (T)	0.6 / 1.8	2.5	9-14
Average field at plasma surface (T)	0.21 / 0.65	1	4-6
Maximum field at a coil (T)	$\leq$ 0.21 / 0.65	< 1	~(a/r_{coil}) 4-6
Plasma current (MA)	0.5 / 1.5	4	26-18
Pulse length Δt (sec) and $\Delta t/\tau_E$	0.1&10 / 0.5&250	5 & 17	>>1
External sustainment/current drive type	Induction	Induction	Induction
Ext. sustainment/current drive power (MW)	2-4 / 12	9	15-30[(d)]
Current drive efficiency (A/W, Spitzer)	0.2 / 0.13	0.4	1.7-0.6
Major radius, R_0 (m)	1.5 / 2.0	2.4	4.9-3.9
Minor Radius, a (m)	0.51 / 0.46	0.8	1.4-0.6
Elongation, κ	1	1 (optimum?)[(e)]	1 (optimum?)[(e)]
Average electron density, $\langle n_e \rangle$ (10^{20} m^{-3})	0.1-0.3 / 0.3	0.4	~3-9
Peak electron T_e (keV)	2-0.7 / 0.9	5	$\langle T_e \rangle$ ~10
Peak ion T_i (keV)	1.3-0.6 / 0.6[(f)]	5	$T_i \approx T_e$
Average plasma pressure (MPa)	$2.1\text{-}3.4\times10^{-3}$ / n.a.	0.1	1.5-4
Average beta, $2\mu_0\langle p\rangle/\langle B_t^2+B_p^2\rangle$ (%)	5-8 (12)[(g)] / n.a.	10	10.5
Energy confinement time (s)	12×10^{-3} / 3×10^{-3}	0.3	1.1-0.2
Fusion power density (MW/m^3)	< 10^{-6}	< 3×10^{-3}	12-83
Core electron transport[(h)] (χ_e m^2/s)	~ 5 / ~ 18	0.5	0.3
Core ion transport[(h)] (χ_i m^2/s)	~ 5 / ~ 18	0.5	χ_e /4
$B(a)\tau_E$ (T-s)	2.5×10^{-3} / 2.0×10^{-3}	0.3	1.2-1.8
$\rho^* = \rho_D / a$	20×10^{-3} / 6×10^{-3}	7×10^{-3}	$2\text{-}3 \times 10^{-3}$
$S_\alpha = a / \rho_\alpha$	1.1 / 3.1	7.3	35-23
Normalized collisionality, $\nu_i^* \sim \nu_{ii} 2\pi a/v_{th}$	4.4×10^{-3} / 17×10^{-3}	0.6×10^{-3}	$2\text{-}3 \times 10^{-3}$
Normalized pulse length $(\tau/\tau_{L/R})$[(i)]	0.1 / <1	0.3	>>1
Normalized pulse length $(\tau/\tau_{Ti=Te})$	0.5-5 / 5	16	>>1
Estimated Fusion Power (MW)	~ 10^{-6}	< 0.1	2300
Estimated neutron wall loading (MW/m^2)	Negligible	< 1×10^{-3}	6-18
Average heat load on wall (MW/m^2)	0.1 / 0.3	0.1	4.5-7.5[(j)]

[(a)]Present value data are for the best performing plasmas from MST and RFX-Mod (Italy) near their respective maximum toroidal plasma currents. Where a range of numbers is quoted for MST, this refers to lower-density and higher-density (with pellet injection) results.
[(b)]The numbers in the ITER-era column represent only one possible set of fusion parameter goals. The numbers (e.g., current drive power, electron temperature, and energy confinement time) are roughly consistent with one another.
[(c)]Two reactor parameter sets are provided. The second numbers in the listed ranges are the assumed parameters for the TITAN study. The first numbers in the ranges are for reduced wall load at lower aspect ratio, derived by scaling the TITAN parameters at fixed beta, temperature, and fusion power generation. These are not optimized; rather they illustrate the changes for a larger plasma with reduced wall load.
[(d)]The values listed are for the sustained burn period. The ohmic dissipation is expected to be substantially larger during startup while T_e is lower, e.g., $(4\text{ keV}/10\text{ keV})^{-1.5}=4$.
[(e)]An optimum RFP cross sectional shape has not been identified, theoretically or experimentally.
[(f)]RFX-Mod central ion temperature is an estimate. This applies to all subsequent RFX-Mod parameters that depend on T_i.
[(g)]Maximum 12% average beta is achieved in MST at lower $I_p \approx 0.2$ MA.
[(h)]The values listed are $a^2/4\tau_E$. Ion confinement is not yet well determined due anomalous heating effects. For best performance plasmas in MST, global energy confinement for ions appears to be at least as good as for electrons.
[(i)]The current relaxation time is order the toroidal current L/R time ~ $\tau_R/10$, where $\tau_R=\mu_0 a^2/\langle\eta\rangle$.
[(j)]The TITAN study employed a toroidal divertor for particle control, but only a very small fraction of the heat was collected at the divertor. Most of the thermal power was radiated uniformly over the entire first wall surface, by doping the plasma with heavy noble gas. About 70% of the radiated power was assumed from the core plasma. The average heat load at the first wall was 4.6 MW/m^2, and the peak value in the divertor was 7.5 MW/m^2. The P_{wn}=6 MW/m^2 case would presumably have 1/3 lower heat loads, but the larger issue is that boundary control in the RFP is not well developed.

CHAPTER 7

THE COMPACT TORUS

7.1. CONCEPT DESCRIPTION

A compact torus (CT) fusion concept is characterized by a toroidal magnetic geometry that can be formed in a simply-connected vacuum vessel – a sphere or cylinder without the central post used in tokamaks. Simply-connected implies that there is no external toroidal field (TF) coil or ohmic (OH) transformer; it is the absence of these two components that leads to relatively compact devices. In terms of a fusion power application CT's have the advantage of allowing high power-density reactors, attractive because of their smaller size and lower cost.

At present two different CT configurations are under investigation in the US fusion program: (1) the field reversed configuration (FRC) and (2) the spheromak. Although they share the same compact geometric features, the basic physics of these two concepts is quite different.

The bulk currents of the FRC are diamagnetic, leading to poloidal magnetic fields and zero or small self-generated toroidal fields. The FRC is thus predominately a diamagnetic entity. The attractions of such a configuration are its very high beta (approaching unity), and simple geometry, with a natural, linear divertor outside the separatrix. In short-pulse experiments macroscopic equilibrium is provided by image currents flowing in the flux conserver surrounding the plasma. In future, longer-pulse experiments, super-conducting flux conserver coils or equilibrium vertical field coils will be required. A diagram of an FRC is given in Fig. 7-1.

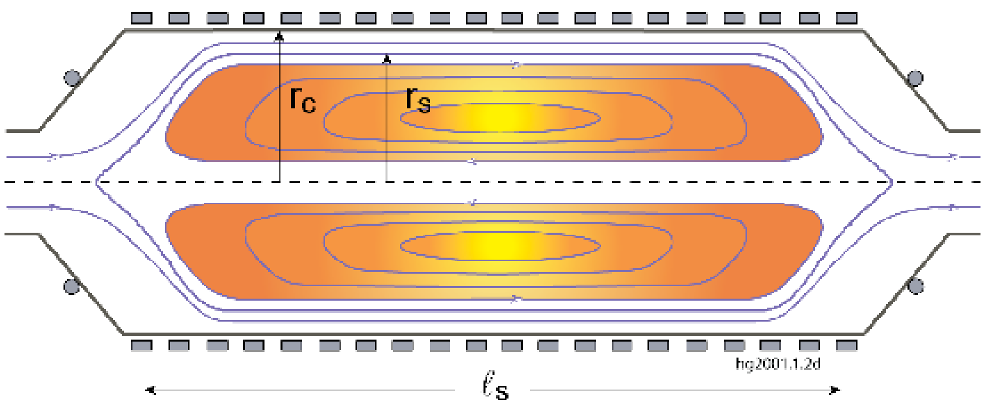


Fig. 7-1. Cross section of an FRC plasma showing external field coils, flux surfaces, and field directions. Location of magnetic separatrix, rs, is indicated.

Initially, field reversed configurations were formed using theta-pinch technology, with typical total temperatures (T_i+T_e) ~ 2 keV and T_e ~ 0.6 keV. However, the poloidal flux was limited to 10s of mWb due to the size and impractical voltages required with theta-pinch technology. Significant efforts have been made in the last decade to form FRCs by merging spheromaks with oppositely directed helicities and by driving current with rotating magnetic fields (RMF). The merging spheromak process is pulsed in nature and it is unclear what the flux limitation is for this technology. In contrast, steady-state FRCs formed and sustained by RMF have achieved, for the first time in CTs, a total temperature over 300 eV, with MW level power inputs, but with poloidal fluxes of only 3-4 mWb, limited by highly anomalous resistivity.

The most critical issue for the FRC is stability at high values of *s*, the ratio of plasma minor radius to the average internal ion Larmor radius. Pulsed FRCs have been made with *s* up to ~ 5; and steady-state FRCs have *s* values of 1-2. However reactors will require *s* ~ 30, and equilibria at such high *s* are calculated to be unstable to ideal MHD modes. Plasma flows and/or some large-orbit ions may improve stability. Tangential neutral beam injection (TNBI) has been proposed as an FRC current and flux sustainment technology, and the associated large-orbit ions could provide stability at large s. Recent calculations indicate complete non-linear stability with s values up to 6 when high energy ions are introduced.

The spheromak has currents that flow both parallel and perpendicular to the magnetic field, with the parallel component dominating. The net result is a non-zero but modest safety factor, $q < 1$. Since by definition a spheromak has no TF coil, the toroidal field is generated purely by plasma currents, although in principle some of the toroidal field could be supplied by a coil. In steady-state spheromak experiments an aspect ratio $R/a < 2$ has been achieved; $R/a \sim 1$ is predicted for fusion-plasma conditions. A cross section of the SSPX spheromak experiment with plasma and reconstructed magnetic flux surfaces is shown in Fig. 7-2.

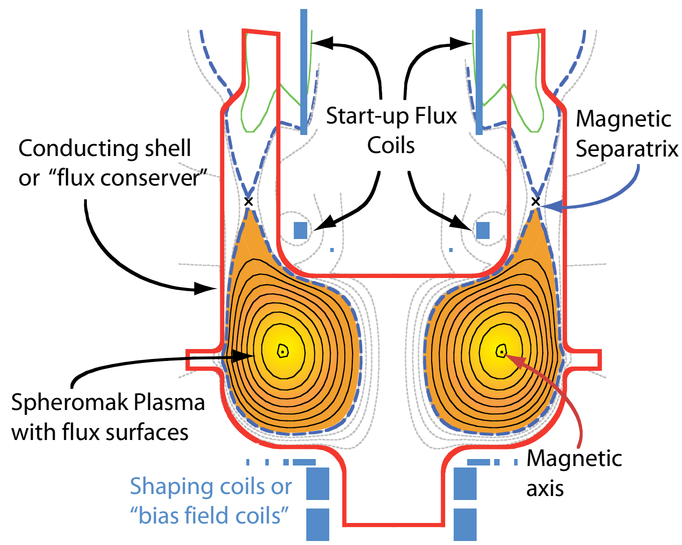


Fig. 7-2. Cross section of a spheromak plasma (SSPX) inside the solid copper flux conserver showing contours of constant magnetic flux.

In the spheromak configuration, linked toroidal and poloidal magnetic fluxes (termed helicity) are injected into a vessel with electrically-conducting walls. A magnetic dynamo rearranges the fluxes at fixed helicity into the toroidal, spheromak geometry. In experiments, magnetic fields in excess of 1 Tesla have been generated, with toroidal currents in excess of 1 MA. Good energy confinement has also been demonstrated: experiments have attained core electron thermal conductivities in the range of 2-10 m^2/s (deduced from ohmic power balance), and T_e = 0.5 keV peak value, although not simultaneously with the current drive. The challenge is thus to achieve either efficient current drive and good confinement simultaneously or to devise a reactor-compatible scenario with separate current-drive and confinement phases.

Simultaneously achieving current drive and good energy confinement is, in fact, a fundamental issue for spheromak research. The transport of helicity into the core plasma by a magnetic dynamo, which drives the plasma current, requires breaking the magnetic surfaces. This allows rapid thermal conduction across the mean magnetic field that forms the spheromak. Although at high electron temperature the associated magnetic fluctuations can be small, experiments and simulations to date find that the thermal losses are large enough to clamp the electron temperature at low values. A related issue arises in RFPs, and developing a deeper understanding of this physics offers an important scientific opportunity, coupling laboratory plasma physics to magnetized plasmas throughout the universe.

The attractive technology features in a CT associated with the absence of a TF coil and OH transformer are thus counterbalanced by inherently more difficult plasma physics due to the absence of large toroidal fields. Also, for both the FRC and spheromak the absence of an OH transformer makes it more difficult to initiate and sustain long pulse discharges. These major areas of CT research have shown, and continue to show, substantial progress.

It is worth noting that the FRC and spheromak occupy unique and quite distinct regions of fusion configuration space. Understanding the physics of the CTs extends plasma and fusion science to parameter regimes that cannot be accessed in other confinement concepts, thereby advancing our science by testing it in novel ways. The FRC allows access to a broad spectrum from $q = 0$ (corresponding to zero toroidal field) to a large q, possibly beyond $q=1$, similar to the spherical torus (ST) but without the center column. The spheromak on the other hand can be viewed roughly as the tight aspect ratio limit of a reversed field pinch (RFP). Both have relatively small toroidal fields, implying a small to modest safety factor: $q \sim 0.5$-1 for a spheromak while $q \leq 0.2$ for an RFP, which typically has an aspect ratio $R/a \sim 4$. The spheromak can also be viewed as the low toroidal field limit of the tight aspect ratio spherical torus (ST). The ST has a TF coil and an OH transformer, thus generating a higher safety factor $q \sim 3$. A standard tokamak has both a larger safety factor ($q \sim 3$) and a larger aspect ratio ($R/a \sim 3$).

Research on compact tori supports the mission:

Develop a compact magnetic fusion reactor without toroidal field coils or a central solenoid.

In the detailed assessment below the goals and issues for the FRC and spheromak are treated separately. Despite their similar vessel geometries the basic physics of these two devices is sufficiently different that separate treatments are warranted.

7.2. GOALS FOR ITER ERA

The ITER-era goal for the CT is:

To demonstrate that a CT with simply connected vessel can achieve stable, sustained or long pulsed plasmas at kilovolt temperatures, with favorable confinement scaling to proceed to a pre-burning CT plasma experiment.

The CT program currently consists of experiments at the concept exploration level, involving both the spheromak and the FRC. Although CT funding has traditionally been a small part of the alternate concepts program, both CT configurations have made substantial plasma physics progress in recent years. Still, further progress on the experimental and theoretical/modeling fronts is required for either of these concepts to warrant a Proof-of-Principle experiment, to be followed by a pre-burning experiment during the ITER-era. Below is a more detailed description of the specific ITER-era goals for both the spheromak and the FRC.

7.2.1. FRC

The primary goals for the FRC are to demonstrate simultaneously global stability and low transport, for parameters substantially closer to burning plasma conditions. Stability has always been the principal concern about FRCs, The experimentally observed robust stability of small FRCs is usually attributed to their kinetic nature, as represented by the number s of internal ion gyro-radii between the field null and the separatrix. Other effects related to high energy ion components and strong, sheared flow have been calculated to enhance FRC stability, and there is experimental evidence that FRCs can be a form of high-β Minimum Energy State (MES). There have been experimental demonstrations of good, quiescent stability in non-sustained FRCs with *s* values up to 4 in both oblate and prolate FRCs, but much higher *s* values will be needed to provide adequate confinement with tokamak level transport coefficients in moderate density (n_e = 10^{20} - 10^{21} m^{-3}) steady-state FRCs. Steady-state FRCs have been produced and sustained using Rotating Magnetic Fields (RMF) and achieved the total temperature well over 200 eV at density of 10^{19} m^{-3} at low input power. However, these FRCs presently have relatively low *s* values of 1-2 and poloidal fluxes of 3-4 mWb, due to high anomalous resistivity. One goal of the RMF effort is to form a good target for neutral beam injection, which can sustain the high-beta configuration while contributing to stability with large-orbit ions.

7.2.2. Spheromak

For the spheromak the primary goal is to develop either an efficient means for current sustainment compatible with acceptable transport, or a pulsed-plasma scenario with separate current drive and confinement phases. The spheromak has made significant progress towards its ITER-era goal by achieving MA plasma currents, magnetic fields of order 1 Tesla, density ~ 10^{20} m^{-3} and electron temperatures greater than 0.5 keV. Numerical simulations capture many features of the experiments. However, it has not yet been possible to achieve good energy confinement simultaneously with current drive via helicity injection. The efficiency of field buildup (i.e., plasma formation) is another important issue that needs to be substantially improved, especially for the pulsed plasma scenario. Lastly, experiments and numerical simulations demonstrate that internal profiles of current and pressure are critical to the achievement of good stability and confinement. Learning how to control profiles measured by improved experimental diagnostics, combined with numerical studies, is another important ITER-era issue likely to be necessary for fusion quality, steady state or long pulsed CT plasmas.

7.3. SCIENTIFIC AND TECHNICAL ISSUES

7.3.1. FRC Issues

Tier 1:

- **Stability at Large *s*:** Achieve global stability at large *s* in low collisionality FRCs

  The most important issue in meeting the FRC ITER-era goal is to extend the presently observed FRC stability at relatively low *s* (< 4) to fusion-grade plasma conditions. The required *s* values depend strongly on confinement quality. A diffusivity value of ~0.5 m^2/sec and *s* ~ 30 is a reasonable reactor vision, while at $D_\perp$ ~ 0.01 m^2/s, an *s* value of 8 would be feasible. Pulsed scenarios at low *s* may provide another option if transport is low enough.

  Advancing the FRC concept rests heavily on the stability issue with respect to low-*n* MHD modes. Radial modes are predicted to be MHD-interchange unstable in an FRC. The *n*=2 rotating interchange mode, driven by the centrifugal force due to plasma rotation, is the only observed instability in quiescent FRCs. In conventional θ-pinch FRCs it is usually stabilized by applying external multipole fields. Calculations have demonstrated that RMF also provides stabilization. This mode is thus successfully controlled in modern experiments. The principal remaining instabilities are axial modes. The n=1 tilt has been considered the most dangerous according to MHD calculations. In contrast to MHD predictions, the tilt is rarely seen.

In general, FRCs have proven to be extremely rugged, surviving dynamic formation, translation, and capture events in θ-pinch experiments and spheromak-merging collisions. There have been experimental demonstrations of quiescent plasmas in non-sustained FRCs with s values up to 4 in both oblate and prolate FRCs. At s ~ 1, FRCs have been sustained for times in excess of 10^4 τ_A. (Here τ_A, the Alfven time ~ 1/tilt-mode growth rate.) The observed stability is primarily attributed to kinetic effects as associated with the relatively small value of s. High energy, axis encircling ions have been calculated to improve FRC stability, and their generation and study is an important part of the FRC development. Recent calculations for a non-rotating FRC, normally unstable even in an oblate flux conserver, have exhibited complete non-linear stability with s up to 6, when high energy ions were introduced.

The observed FRC stability may also result from non-kinetic effects, primarily strong flow with flow shear, and possibly magnetic shear arising from small but finite toroidal fields. Though the toroidal field is much lower than the poloidal field, the safety factor, q, can be greater than 1, particularly at high elongation; stabilizing magnetic shear may result. Local shear is also predicted to exist in FRCs driven by odd-parity RMFs. Furthermore, there is experimental evidence that FRCs can be formed in high-Minimum Energy States, which may promote stability and confinement.

Tier 2:

- **Transport:** Reduce transport in keV FRC plasmas and establish transport scaling

  Transport in the FRC is highly anomalous. The target transport coefficients are reactor-design dependent and far smaller than the smallest achieved in an FRC, $D_\perp$ ~ 5m^2/s. In fully diamagnetic plasmas, such as FRCs, particle and flux loss rates are related, which may also be related to energy transport. The science of transport in FRCs is in an early and exciting stage. Little is known about confinement in FRCs other than what was implied by the flux and particle lifetime measurements. A physics-based scaling is one of the most important FRC needs. Major challenges are to apply sufficient standard diagnostic techniques to measure transport in FRCs, with the same intensity as done early in the tokamak program, the 1970s, and to develop first principles models of plasma transport suitable for kinetic and MHD-like FRCs.

  For FRCs with bulk diamagnetic toroidal plasma currents, reducing transport is key to attaining the ITER-era goals, in terms of both confinement and current drive. The key to anomalous cross-field resistivity is thought to be the drift parameter ratio, $\gamma_d = v_{de}/v_{ti}$, where v_{de} is the electron drift velocity and v_{ti} is the ion thermal velocity. This parameter appears in all theoretical calculations, with $D_\perp \propto \gamma_d^2 D_B$, where D_B is the Bohm diffusivity. Since γ_d scales as $1/(n^{1/2}r_s)$ with r_s being the separatrix radius (independent of temperature for a high β FRC), this scaling would lead to near classical transport rates at reactor dimensions.

- **Current Drive and Sustainment:** Achieve efficient current drive and sustainment of keV FRCs with good confinement

  CTs have a unique requirement for creating and maintaining the poloidal flux without an inductive transformer. The experimental goal for current drive during the ITER era is to demonstrate a high efficiency method that can build-up and sustain magnetic flux to the level that scales to a burning-plasma experiment. There are currently three candidate current drive and sustainment concepts for the FRC: Rotating Magnetic Fields (RMF), Neutral Beam Injection (NBI), and charged fusion products in burning plasmas. (Flux sustainment is a more descriptive terminology for FRCs than current drive since diamagnetic current arises solely from pressure gradients.) The only means currently demonstrated for sustaining a FRC (without any internal flux-adding coils) is RMF. The original, even-parity RMF method of current drive was predicted to generate open field lines, hence degrading confinement. Odd-parity RMF is predicted to maintain field-line closure, essential for good confinement. First experiments from TCS (UW) and PFRC (PPPL) are promising but not yet conclusive. Fluid theory for even-parity RMF is well developed and successfully demonstrated. Theory for odd-parity RMF current drive needs further development, in part because the unique azimuthal electric field (E_ϕ) at the O-point null necessitates a kinetic treatment. E_ϕ should form a beam-like high-energy distributions of axis-encircling orbits near the null, aiding stability and current drive.

RMF current-drive efficiency is presently limited by highly anomalous resistivity and the associated transport coefficient, $D_\perp$. To reach the ITER-era current drive goal, $D_\perp$ will have to be substantially reduced. It is important to extend experiments to lower values of γ_d to reduce transport rates if the drift wave driven transport scaling observed in the previous low-*s* FRC experiments can be realized. In addition, tangential NBI has been proposed as a flux sustainment technology for both prolate and oblate FRCs. Tangential NBI should contribute strongly to the current drive goal and provide useful sources of particles and energy near the field null. For it to be effective in forming trapped high-energy ion rings within the FRC separatrix, the target FRCs must contain sufficiently high poloidal fluxes. In FRC reactors, fusion products spiral primarily in the ion diamagnetic direction, and could provide appreciable current drive. Self-consistent effects need to be examined.

For most FRC reactor concepts, the scientific value of current-drive research is very high. The pulsed high density (PHD) concept is the exception. With repetitive regeneration of the short-pulse fusion plasma, the PHD requires far less in the way of current sustainment and may offer an alternative path to fusion.

Tier 3:

- **Fast Particles:** Understand effects of fast particles on current drive, stability and confinement

  Fast particles are expected to play an important role in the stabilization of large-*s*, MHD-like FRCs. Calculations show that high-energy, axis-encircling ions may improve FRC stability. Recent calculations for a non-rotating FRC normally unstable even in an oblate flux conserver have exhibited complete non-linear stability when high energy ions were introduced. Tangential NBI has been proposed as a flux sustainment technology for both prolate and oblate FRCs.

  However, it is unknown what fraction of high energy ions, whether from NBI, RMF, or charged fusion reaction products would be required to provide FRC stability. Beam-driven instabilities in FRCs, such as the TAE modes seen in tokamaks, have not been studied. Experimental facilities leading to a burning plasma should have the ability to provide high energy ions. In addition, more attention must be paid to modeling FRC stability, e.g., increasing the capabilities of codes to properly handle RMF_o and tangential NBI.

- **Heating:** Demonstrate efficient heating methods (RMF, NBI, and compression) to increase the temperature to near burning-plasma values.

  The methods proposed for plasma heating are tangential NBI, RMF and compression. Questions are whether high heating powers can be applied and absorbed while improving confinement and stability and whether heating produces a good diamagnetic current profile. Neutral-beam heating physics has been studied extensively on tokamaks; drag and pitch angle scattering are well understood. The likelihood of heating both electrons and ions is high. The physics of compressional heating, as proposed for pulsed high-density experiments, is also well understood, though again stability and transport behavior must cooperate to achieve the target temperatures. The RMF current drive method is well understood, but the additional heating noted beyond simple Ohmic has only begun to be calculated with numerical codes. Calculations indicate that RMF, particularly odd-parity RMF, can preferentially heat different ion species, which could be important for advanced fuel schemes.

7.3.2. Spheromak Issues

Tier 1:

- **Sustainment and Confinement:** Achieve efficient time-averaged current drive simultaneously with good confinement.

  A stationary-state spheromak requires simultaneous current drive and good confinement. To date, experiment, theory and resistive MHD simulations find that current drive by electrostatic helicity injection opens magnetic surfaces through magnetic fluctuations, resulting in rapid energy losses and low T_e. Good confinement, with electron thermal conductivities in the spheromak core comparable to tokamak L-mode values, has been obtained in slowly decaying plasmas with no current drive inside the separatrix. Simultaneous current drive and good confinement will thus require a new approach to helicity injection or the use of other current-drive technologies.

  Experiments of the refluxing concept, in which the plasma is rebuilt periodically, were explored in SSPX. The confinement phase was found to be relatively quiescent. Experiments also found that the electron temperature rapidly recovered following the rebuilding. This option to sustainment and confinement in a potential reactor will require considerable additional experiments and simulations to evaluate its potential.

  In both continuous and pulsed/reflux operation, it is important to maintain internal MHD stability during the confinement phase. This was done in SSPX by maintaining current outside the separatrix, but long-pulse, detailed magnetic fluctuation limits will likely require internal profile control. Initial experiments, however, can address buildup and confinement without full profile control, which could be explored as a follow-on effort. Similarly, density and particle control may be needed for long-pulse or steady-state efficient current drive and good confinement.

- **Formation:** Develop an efficient formation technique to achieve fusion-relevant spheromak magnetic fields.

  Electrostatic helicity injection has successfully generated MA spheromaks in SSPX, and CTX obtained 3 MA in a small flux conserver. Significant energy is required as both the magnetic field and plasma kinetic energy are supplied by the injection. In experiments to date, this process has operated at 5% to 20% efficiency, and the amplification of the bias poloidal has been in the range of 2-6. Both these need improvement for fusion-scale experiments; for example, flux amplification needs to be ~ 50 to reduce the edge Ohmic losses relative to those in the spheromak sufficiently for a reactor.

Resistive MHD simulations agree very well with the experimental flux amplification in SSPX up to a limit where the experiment saturates for reasons which are uncertain. Other spheromak experiments have observed a similar saturation, although possibly not at the same values. There have been no whole-device simulations of these experiments.

Any formation techniques studied should scale to a reactor and/or be designed to generate an improved understanding of the physics.

Tier 2:

- **Transport:** Determine underlying transport mechanisms and confinement scaling in low-collisionality spheromak plasmas.

  Energy confinement experiments in slowly-decaying plasmas yield thermal conductivities in the L-mode range, although there is a very limited data base. Power-balance calculations with 0.1 keV < $T_{e,peak}$ < 0.5 keV and densities ~ 10^{20} m^{-3} yield electron thermal conductivities ~1 m^2/s - 10 m^2/s in the core of the spheromak. Because the experiments were at the concept exploration level, there were no measurements of ion temperatures or the ion thermal conductivity. The electron transport mechanism has not been studied, although it is found that the best confinement occurs when the *q*-profile does not cross low-order resonances. Resistive MHD simulations support this result: in slowly decaying plasmas closed magnetic surfaces form and the electron temperature rises to a few hundred eV, consistent with an assumed thermal conductivity. If the *q*-profile evolves to cross low-order surfaces, islands form; when they overlap they result in stochastic magnetic field lines and high thermal losses throughout much of the spheromak.

  A better understanding of thermal loss mechanisms is essential to achieving the ITER-era goal for CTs. In addition, this understanding will broaden our scientific knowledge by extending confinement studies to the safety factor range ~0.2 < q < 1 which lies between the tokamak and the RFP.

- **Beta Limits:** Understand beta-limiting mechanisms.

  Experiments in SSPX find that there is a limiting electron beta, ~ 5% (peak), although the highest temperature shots exceeded this by up to a factor of 2. The ion contribution is not known, but could contribute up to a similar factor. It is unknown whether the achieved betas were limited by Ohmic power balance or the onset of pressure-driven modes. A transient peak electron beta ~ 20% was observed in CTX, resulting in a pressure-driven mode. There are no computational studies of beta limits, although simulations are consistent with experiment.

  Understanding this limit, including the effects of spheromak shaping, may allow it to be increased significantly. Such a result would allow significant improvement of the spheromak reactor vision.

- **Particle Balance and Density Control:** Understand particle balance and control of plasma density and impurities.

Experiments have typically operated with an initial gas pulse, with the density evolving to a limit apparently determined by recycling from the flux conserver walls and the helicity injector. There is some evidence that in clean discharges the density evolves to a constant fraction of the current density, similar to the Greenwald limit in tokamaks and perhaps associated with stability related to electron-streaming. Impurities have been successfully controlled by coating the copper flux conserver and gun with a refractory metal (tungsten) layer, discharge cleaning, and gettering using titanium. These techniques for controlling density and impurities and of refueling are unlikely to be adequate for long-pulse spheromaks.

Particle and density control is a complex scientific issue, involving plasma-wall interactions, penetration of the plasma by neutral particles, density pinching, plasma flows, etc. Achievement of the ITER-era goals will likely require extension beyond techniques necessary for the Tier 1 experiments.

Tier 3:

- **Fast Particles:** Evaluate the effect of fast particles on current drive, stability, and confinement

  Fast particles from NBI, RF, and fusion will have effects on a broad range of physics issues. Their effects will increase in importance as spheromak physics evolves beyond the fundamental issues of current drive and confinement. Experiments beyond the ITER era will require a good scientific understanding of alpha-particle science in spheromaks, and may draw on science developed in ITER or other burning plasmas.

- **Resistive Wall Modes:** Demonstrate resistive wall mode control

  The n=1, m=1 tilt and shift modes become resistive wall modes in long pulse plasmas. The basic physics of these modes and their control are well understood from tokamak and RFP research, but no spheromak experiment has a pulse length long enough to study their onset and development in the spheromak. Developing techniques in the spheromak can draw on science developed in experimental results from other magnetic fusion concepts.

- **Technology:** Develop the technology for long pulse operation

  Spheromak specific technology will often not be addressed by other components of fusion program: for example, handling power loading on the flux conserver and electrodes for long pulse electrostatic helicity injection will likely need development. It is, however, premature to scope this issue in depth.

7.4. FACILITIES AND GAPS

7.4.1. FRC Facilities and Gaps

The US is presently the world leader in FRC research. Table 7-1 shows a list of the various facilities available to study FRC issues. These experiments have been very useful in demonstrating the possibilities of FRC stability and have spurred much theoretical interest, but they are all too small to address the basic issues of stability, confinement, and current drive under conditions relevant to a burning plasma. On the modeling front, several 2-D codes have been used to model FRC formation and RMF current drive. 3-D hybrid codes (e.g. HYM at PPPL) have been used to study FRC stability, including the effects of beams and RMF. NIMROD is being refined and benchmarked at the Plasma Science and Innovation Center (U. Washington, U. Wisconsin, Utah State) for a similar set of tasks.

The US FRC research programs have gaps in both machine capabilities and theoretical / modeling tools for addressing the scientific and technical issues to achieve the ITER-era goals.

Table 7-1
US FRC Research Facilities and Principal Efforts

Facility	Principal Effort
TCSU (U. Washington)	Steady-state current drive and sustainment of FRCs using rotating magnetic fields
PFRC (PPPL)	Heating of FRCs by odd-parity RMF
PV Rotomak (Prairie View A&M)	Heating of FRCs by magnetic reconnection of two FRCs driven by RMF
SSX (Swarthmore College)	Formation and stability of CTs (spheromaks and FRCs) by merging spheromaks
MRX (PPPL)	Formation and stability of CTs by merging spheromaks
Colorado FRC (U. Colorado)	Turbulence, flow, and transport in FRCs by merging spheromaks (under construction)
FRXL (LANL)	Liner compression of FRCs
PHD (U. Washington)	Adiabatic compression of FRCs

Tier 1:

- **Stability at Large *s*:** Achieve global stability at large *s* in low collisionality FRCs

  This issue needs to be addressed on both theoretical/modeling and experimental fronts. While significant progress has been made in the understanding of experimentally observed stability in low-*s* regimes, there is still no sound physics basis for FRC macroscopic stability in the large *s*, MHD-like regimes. More efforts must be made to understand whether the stability achieved in the present low *s* FRCs can be extended to large *s*, taking into account two-fluid effects, non-MHD effects, primarily from fast particles produced by TNBI or fusion products) and plasma rotation and flow shear.

  On the experimental front, near reactor *s* values cannot be accessed in the FRC devices presently operating. To address this issue, large increases will be needed in the flux and *s*, requiring increases in machine size, magnetic-field strength, heating power, with improved experimental diagnostics. Neutral beams can contribute strongly to this goal by providing additional stability, as well as current drive.

Tier 2:

- **Transport:** Reduce transport at keV FRC plasmas and establish transport scaling, and
- **Current Drive and Sustainment:** Achieve efficient current drive and sustainment of keV FRCs with good confinement

  The two issues in Tier 2 are closely coupled together. Since the FRC is a diamagnetic entity with current flowing predominantly across the poloidal fields, both confinement quality and current drive inefficiency are limited by poor cross-field transport in present sustainment experiments.

FRC Transport is poorly understood. This can be addressed, to some extent, in the present machines by improving diagnostic techniques, coupled with modeling, to identify transport mechanisms, and establish transport scaling. The first three experiments listed in Table 7-1, i.e., TCSU, PFRC and PV Rotomak, involve using RMF for the formation and current sustainment of FRCs, but the current drive efficiency is low due to high resistivity or transport, while SSX (s~5), MRX, and Colorado FRC utilize merging spheromak techniques to study all CTs (including both spheromak and FRC), including basic issues of FRC stability and transport, but limited to relatively low $s \leq 5$. Thus, the key is reducing the present high anomalous resistivity.

Since FRC transport appears to have a strong dependence on the drift parameter γ_d, a straightforward way to reduce the transport rates and increase current drive efficiency is to increase machine size to reduce γ_d. In addition, this would concomitantly allow tests of stability issues, as *s* increases with machine size at a given temperature. A major innovation would be the inclusion of tangential NBI, which not only supplements RMF current drive but may contribute to stabilization and confinement.

Tier 3:

- **Fast Particles:** Understand effects of fast particles on current drive, stability and confinement

  There are presently no means to address the fast particle issue in Tier 3. In order to trap fast particles, e.g., from tangential NBI, FRCs must have sufficient poloidal fluxes and lifetimes. None of the existing FRC experiments listed in Table 7-1 are ready for tangential NBI, and only very limited efforts have been made on the theoretical/modeling front in the understanding of beam effects.

  To address this issue requires upgrading the present facilities to add NBI or using the new facility if available.

- **Heating:** Demonstrate efficient heating methods (RMF, NBI, and compression) to increase the temperature to near burning-plasma values.

  Several heating methods are being explored in the various small FRC experiments including PFRC (RMF_o), PHD (adiabatic compression), and FRXL (liner compression). In particular, NBI and odd-parity RMF_o have been proposed for steady-state heating of FRCs. As noted above, beam-heating experiments will require a larger, higher field FRC target than currently available in the existing experiments listed in Table 7-1, while for RMF_o, RMF penetration at smaller values of B_ω / B_e must be achieved to get good ion heating at acceptable small circulating power, and thus a larger, higher field FRC is needed.

  In summary, to achieve the ITER-era goals for the FRC will require a larger machine with higher fields than in present devices to get large *s* and enough flux for internal confinement, and significantly improved numerical predictive capabilities. Progress toward the ITER-era goal cannot be made without this.

		Means to Close Gap		
	FRC Issue	**Improved Diagnostics**	**Predictive Capability**	**Upgraded or New Facilities**
Tier 1	Stability at large *s*	Make detailed magnetic and profile measurements to understand and control instabilities	Explore possible physics mechanisms for stability at large *s*	Need to upgrade the present facilities to access large-*s* regimes *If successful, one or more follow-on PoP-level experiments will be needed to meet the ITER-era goal*
Tier 2	Transport	Make detailed measurements of fluctuations and confinement properties to identify transport mechanisms	Explore transport mechanisms and establish a physics based transport scaling	Use the present facilities, or upgraded/new facilities if available
	Current drive and sustainment	Make detailed profile measurements to evaluate current drive efficiency and effects on confinement quality	Assess various current drive techniques using RMF, NBI, or by charged fusion reaction products.	Need to upgrade the present facilities to add neutral beams or use new facilities if available
Tier 3	Fast particles	Need to develop fast particle related diagnostics	Evaluate effects of fast particles on current drive, stability and confinement	Need to upgrade the present facilities to add neutral beams or use new facilities if available
	Heating	Make detailed profile measurements to determine heating efficiency of various techniques	Evaluate various heating techniques	Need to upgrade the present facilities or use new facilities, if available, for beam heating experiments

7.4.2. Spheromak Facilities and Gaps

- **Existing Facilities and Simulation Capabilities.** There are several existing, small experiments that explore spheromak and spheromak-related physics at a Concept Exploration (CE) level. These facilities and their primary thrusts are listed below; the primary issues addressed (at a CE level) are identified:
  - **HIT-SI.** Experimental program investigates a concept to inductively drive current in a bow-tie spheromak plasma. It uses two non-axisymmetric injectors to inject helicity at a constant rate with odd symmetry. *Addresses*: Sustainment and confinement, Formation, Beta limits, Particle balance and density.
  - **SSX.** Experimental program investigates the merging of counter-helicity spheromaks to form FRCs or spheromaks. The program investigates magnetic reconnection during the merging, and stability of the resulting plasma configuration. *Addresses*: Formation.
  - **SSPX** (shutdown, data analysis ongoing) T_e~0.5 keV, B_{tor}> 1Tesla, I_{plasma} ~ 1MA, n_e ~ 1×10^{20} m^{-3}, achieved good (but transient) core confinement approaching tokamak L-mode. Achieved reasonable internal current profile control to avoid low-order mode rational surfaces by programming the initial flux distribution and discharge current. *Addresses* Sustainment and confinement, Formation, Beta limits.

— **Caltech** Experiments. Investigate the physics of spheromak formation by using a magnetized planar coaxial helicity source. The main issues being studied are topological evolution, helicity and mass injection, flows and stagnation, kink instabilities, flux amplification, relaxation and reconnection, and the generation of energetic particles. *Addresses*: Formation, Particle balance and density.

— **MRX.** Flexible experimental platform for inductive spheromak/FRC formation and merging of formed plasmas. Utilized for study of fundamental physics of magnetic reconnection and magnetic self-organization phenomena. *Addresses*: Formation, Particle balance and density.

— **PBX** (new). Multi-pulsed startup experiment. Aims to show that by repetitive injection of plasma from a coaxial source, the energy density of a spheromak can be increased in a step-wise manner to achieve high current amplification. *Addresses*: Formation.

— **LANL – DRX** (internal funding, new) The Driven Relaxation Experiment is a new experiment designed to explore power coupling efficiency (and possible resonances in that coupling) to maximize flux and current amplification while preserving stability. Key features include ability to vary aspect ratio (flux conserver length: diameter) and reaching very high gun lambda (gun current / bias poloidal flux). *Addresses*: Sustainment and confinement, Formation.

With the shutdown of SSPX there is no facility with the volume, vacuum and wall cleanliness, and power to study spheromak issues at high magnetic fields, density, and temperature. The small CE experiments excel at exploring physics and studying innovative ideas but cannot test them at plasma parameters on the road path to fusion-energy experiments.

Numerical simulations, largely using the single-fluid, resistive MHD approximation have proved effective in predicting spheromak properties. These results have been benchmarked against data from the SSPX experiment, with excellent agreement in spheromak formation (up to an experimental saturation level) resulting from helicity injection. Many features are successfully predicted at the quantitative or semiquantitative level. Specific activities include:

- The leading modeling tool, NIMROD. Spheromak modeling is continuing at UW-Madison, UW, LLNL, Woodruff Scientific, Tech X, PSI center. The NIMROD team (http://nimrodteam.org) provides code development for a wide range of magnetized plasma applications.
- The "PSI Center" activities which include developing codes, validating them against experiments and visualization for ICC program. The goal is to develop codes that can accurately predict the behavior of experiments before they are built.
- Analytic theory. This is done at LANL, LLNL, Woodruff Scientific, UW.

7.4.3. Spheromak Research Gaps

- **Sustainment and Confinement.** With the closing of the SSPX experiment at LLNL, there is no present facility capable of conducting high-power spheromak experiments in an high-quality vacuum environment. Although many ideas can be tested in existing (but small) facilities, sustainment and confinement will require capabilities and a size at least equivalent to SSPX. One possible path forward is to initiate focused, Tier 1 experiments on current drive in a device at the concept exploration level which is large enough to allow confinement times of several ms.

Ongoing work at the concept-exploration level includes an alternate, inductive means of helicity injection; formation by induction; and formation by multipulsed merging of spheromaks. In addition, the MHD simulation code is being upgraded to include two fluid and finite Larmor radius effects which will better model the physics in reconnection layers. Whether any of these approaches can drive current while suppressing magnetic fluctuations sufficiently to allow good confinement is presently unclear.

Spheromak current drive using neutral beams or rf, perhaps following formation using helicity injection, offers a possibility for sustaining current with good confinement. Although there are no experiments using beams or rf, preliminary calculations on the use of beams are encouraging. More detailed modeling of these alternates would provide a basis for determining whether experiments are likely to be efficient and effective enough to be successful.

Bootstrap currents are thought to be small in existing experiments, but the development of theory that applies to spheromaks is needed to understand how large such currents can be in fusion-quality plasmas.

Finally, there have been encouraging experiments of the pulsed-reflux scenario that separates the current drive phase from the confinement phase. In this approach, helicity injection is used to rebuild the current periodically. Experiments have demonstrated good confinement in the slowly-decaying plasma following pulsed buildup; in a reactor this would be the burn phase. Simulations and experiments can be used to evaluate this approach. Ultimate success will require improving both the efficiency of helicity injection and the magnitude of current and flux amplification.

— <u>Facility Requirements</u>. A new experiment of at least the size and capabilities of SSPX is required. Upgrades to existing experiments will be unable to meet this need. The design of a new facility should be approved on the basis of well thought out, peer-reviewed physics with a reasonable likelihood of both new science and progress towards a reactor vision; it is likely to have significant differences from the SSPX design which was based on our understanding of a decade ago.

  It is important to recognize that the experiment described above is not of sufficient size and capability to achieve the ITER-era goal. Assuming success in resolving the Tier I and 2 physics issues (at least in part), one or more Proof-of-Principle experiments will be needed to address the physics of the spheromak in the multi-keV regime. This might be possible by an upgrade, but it is unlikely to be sufficiently large to obtain the necessary physics results.

— <u>Diagnostic Requirements</u>. The SSPX experiment had a moderately large suite of plasma diagnostics. There were, however, many parameters, important to its mission, that could not be measured. An example was a means of measuring the ion temperature in high density and temperature spheromak plasmas, e.g. by charge-exchange recombination. It is important to support a new experiment at a sufficient resource level that it can obtain the data needed for its mission. Support needs to include sufficient funding for the scientific personnel who operate the diagnostics and interpret the resulting data.

— <u>Computational Development Requirements</u>. Computational capabilities which can support whole-device simulations are available at the NERSC and in many laboratories in the US. Continued upgrading of these facilities and of the numerics in NIMROD and other codes will yield significant payoffs by including more physics and allowing faster computations. Also, a significant effort is underway to include 2-fluid effects in NIMROD; this will allow modeling of detailed reconnection physics and may help improve much of the semi-quantitative agreement with experiment. This effort should be strongly supported.

The ability to use computational simulations can also play a significant role in the design of future experiments and should be utilized as much as possible. Opportunities include optimization of the geometry for spheromak formation and exploration of current profile control techniques.

Computational support for studying current drive by neutral beams or rf will require development if either is to be used in an experiment.

Facility gaps also exist to address the other spheromak issues:

- **Formation.** Small experiments are presently exploring multi-pulse buildup, merging of inductively-formed and helicity-injected spheromaks, and buildup using possible resonances in coupling between the drive and the spheromak. There has been no systematic exploration of the role of the flux conserver and gun geometries, among other factors.

  Resolving the Formation issue will require the same or similar facilities and computational development as Sustainment and confinement. The role of the flux conserver and helicity injector geometries can be explored using resistive MHD simulations. Such calculations are much less expensive than experiments and thus can be used to guide the design options for increasing the amplification of the applied magnetic flux.

  The existing small CE experiments and resistive MHD simulations can explore new ideas, but a larger experiment and improved simulations will be required to fully develop the necessary capabilities.

- **Transport.** A well diagnosed spheromak experiment is needed to study energy transport in spheromak plasmas as there are presently none capable of evaluating confinement and transport mechanisms. One option will be to extend the use of a facility developed to study current drive, although upgrades to that facility will likely be needed, e.g. by adding necessary diagnostics. Alternatively, a new experiment can be designed to focus on confinement.

- **Beta Limits.** No present experiments are capable of addressing beta limits or the physics underlying these limits. Experiments on beta limits and the mechanisms leading to them will require adding auxiliary power to ohmic heating from current drive to separate the measured behavior from the effects of ohmic scaling. Neutral beam injection heating, for example, can provide a tool to explore these limits and the associated physics. Plasma shaping may also be important.

  Present theoretical results on spheromak beta limits were generated in the 1980s. Modern simulation capabilities should be applied to a study of this issue. Among other conclusions it will be important to know whether the design of the transport facility would need significant modifications to study beta limits.

- **Particle Balance and Density Control.** This issue can probably be addressed in the facilities developed for current drive and confinement.

Tier III issues: Fast particles, Resistive wall modes, and Technology will probably require dedicated PoP-level facilities or significant upgrades to the facilities used for Tier I and II. Detailed scoping of the requirements is premature until the Tier I and II issues are addressed.

		Means to Close Gap		
	Spheromak Issue	**Small CT Exp.**	**Simulations**	**New Facilities**
Tier 1	Sustainment and confinement (S&C)	Explore new ideas and physics	Explore new helicity injection techniques Develop Reflux scenario	A large CE facility to: • Develop new helicity injection technique • Develop Reflux sustainment • Explore alternate current drive ideas *If successful, one or more follow-on PoP-level experiments will be needed to meet the ITER-era goal*
	Formation	Explore new ideas and physics	Explore new ideas for formation and optimize by varying geometry and other parameters	A large CE facility (possibly the same as for S&C) to test and explore improvements in efficiency and effectiveness
Tier 2	Transport			Use the S&E facility and/or a new facility to study the physics of transport in spheromaks
	Beta limits	Explore limits	Evaluate beta limits and effects of geometry an other parameters	Use the S&E facility and/or a new facility to determine beta limits
	Particle balance and density control	Explore physics and techniques		Evaluate physics and develop capabilities in a facility constructed for other spheromak physics developments
Tier 3	Fast particles			Study fast particle physics in a PoP-level facility
	Resistive wall modes		Evaluate application of techniques developed in tokamaks and RFP	Develop resistive wall mode science and control techniques in a PoP-level facility
	Technology			Prepare for pre-burning and burning plasma experiments

Key Parameters for Steady–State FRCs[a]

Parameter	Present Value[b]	ITER-Era Goal	Reactor Target
Confining Field (T)[c]	0.03	0.12 – 0.5	1.8
Plasma current (MA)[d]	0.1	1 – 4	5 – 25
Pulse length Δt (sec) and $\Delta t/\tau_E$	0.01, 75	0.1, 20	∞
External sustainment/current drive type	RMF	RMF, TNBI	RMF, TNBI, fusion product
External sustainment/current drive power (MW)[e]	2	5	10 – 50
Current drive efficiency (η)	0.01	0.25 – 1	0.5
Major Radius (m)[f]	0.25	0.65	1.5
Minor Radius (m)[f]	0.10	0.27	0.62
Elongation (κ)	4	4	0.8 – 4
Central density n_e (10^{20} m^{-3})	0.1	0.3 – 1.2	4
$\langle T_e \rangle$ (keV)	0.2	0.65 – 2.5	10
$\langle T_i \rangle$ (keV)	0.05	0.65 – 2.5	10
Average beta (%)	0.6	0.6	0.6
Energy confinement time (s)[g]	1.5×10^{-4}	5×10^{-3} – 2.5×10^{-2}	1
Fusion power density $B\tau_E$ (T-s)	5×10^{-6}	10^{-3} – 2.5×10^{-2}	3.6
Core electron transport (χ_e m^2/s)[g]	20	1 – 5	0.5
Core ion transport (χ_i m^2/s)[g]	?	1 – 5	0.5
$\rho_* = <\rho_D>/a$	1	0. 1	0.03
$S_\alpha = a/\rho_\alpha$ (4 Mες)			
Collisionality (ν_*) ($=a/\lambda_{mfp}$)	0.1	10^{-3}	10^{-4}
Normalized pulse length (τ/τ_r)[h]			
Normalized pulse length ($\tau/\tau_{Ti=Te}$)[h]	75	20	∞
Estimated Fusion Power (MW)			200 – 1000
Estimated wall loading (MW/m^2)			10
Estimated plasma exhaust power (MW/m^2)			1

[a]The ITER era goals contain near-term and far-term values, depending on the plasma resistivity. The reactor target contains values for an oblate and a prolate CT. If the wall loading is too high the radius can be made larger and the magnetic field and density lower.
[b]Indicate if not simultaneous.
[c]Peak on axis.
[d]Ohmic or driven or diamagnetic.
[e]Power to plasma needed to maintain configuration, magnetic field, or plasma current.
[f]Mean values if not axisymmetric.
[g]Measured or estimated from power balance, size, beta, or ne, Te, and Ti.
[h]τ_r ($\tau_{Ti=Te}$) is relevant time scale for configuration redistribution (temperature equilibration).

Key Parameters for Pulsed FRCs[a]

Parameter	Present Value[b]	Reactor Target
Confining Field (T)[c]	0.5 – 10	20
Plasma current (MA)[d]		
Pulse length Δt (sec) and $\Delta t/\tau_E$	4×10^{-4}, 1	4×10^{-4}, 1
External sustainment/current drive type	Not Needed	Not Needed
External sustainment/current drive power (MW)[e]		
Current drive efficiency (η)	0.01	0.25 – 1
Major Radius – r_s (m)[f]	0.02 – 0.2	0.05
Minor Radius (m)[f]		
Elongation (κ)	3 – 10	15 – 20
Central density n_e (10^{20} m^{-3})	1×10^{21} – 2×10^{23}	1×10^{23}
Central T_e (keV)	0.5	2
Central T_i (keV)	6	5 – 10
Average beta	0.76 – 0.98	0.87
Energy confinement time (s)[g]	4×10^{-4}	5×10^{-4} – 1.5×10^{-3}
Fusion power density $B\tau_E$ (T-s)	2×10^{-4}	10^{-3} – 2×10^{-3}
Global particle transport (m^2/s)[g]	2	0.2
Fusion energy (MJ)	NA	1.5 – 5 MJ/pulse
Neutron wall loading (MW/m^2)	NA	1 – 10
Estimated plasma exhaust power (MW/m^2)	NA	1 – 10

[a]The compiled data listed in the table are obtained from various past and present experiments, and do not represent simultaneous measurements.
[b]Indicate if not simultaneous.
[c]Peak on axis.
[d]Ohmic or driven or diamagnetic.
[e]Power to plasma needed to maintain configuration, magnetic field, or plasma current.
[f]Mean values if not axisymmetric.
[g]Measured or estimated from power balance, size, beta, or ne, Te, and Ti.

Spheromak Key Parameters

Parameter	Present Value[a]	ITER-Era Goal	Reactor Target
Confining Field (T)[b]	1.1	2.5	5 (wall value)
Plasma current (MA)[c]	1	20	47
Pulse length Δt (sec) and $\Delta t/\tau_E$	0.01, 10	SS, QSS[d]	SS, QSS, Pulsed
External sustainment/current drive type	CHI[e]	CHI, SIHI[e], other	?
External sustainment/current drive power (MW)[f]	P_{edge}=50 P_{ohm}=5	100	30 - 60
Current drive efficiency (η)	0.1	0.2	0.6
Major Radius (m)[g]	0.32	1.3	1.5
Minor Radius (m)[g]	0.18	1	1.5
Elongation (κ)	1.2	1.2	1.2
Central density n_e or $\langle n_e \rangle$ (10^{20} m^{-3})	1-2	2	2.3
Central T_e or $\langle T_e \rangle$ (keV)	0.5	5	20
Central T_i or $\langle T_i \rangle$ (keV)	?	5	20
Central beta (% and β_N)	10, β_N=4	20	20
Energy confinement time (s)[h]	0.001 (P_{ohm}) 0.0001 (P_{edge})	0.04	0.43
Fusion power density $B\tau_E$ (T-s)	0.001	.1	2
Core electron transport (χ_e m^2/s)[h]	< 10	10	5
Core ion transport (χ_i m^2/s)[h]	?	10	5
$\rho_* = \rho_D/a$	0.02 (T_i=T_e)	$4\text{x}10^{-3}$	$4\text{x}10^{-3}$
$S_\alpha = a/\rho_\alpha$ (4 M$\varepsilon\varsigma$)	1	6	20
Collisionality (ν_*) ($=a/\lambda_{mfp}$)	10^{-2}	10^{-3}	10^{-4}
Normalized pulse length (τ/τ_r)[i]	0.01	–	–
Normalized pulse length ($\tau/\tau_{Ti=Te}$)[i]	2	–	–
Estimated Fusion Power (MW)	0	0	3400
Estimated wall loading (MW/m^2)	0	0	20
Estimated plasma exhaust power (MW/m^2)	40	5	5

[a]Indicate if not simultaneous.
[b]Peak on axis.
[c]Ohmic or driven or diamagnetic.
[d]SS = Stationary State, QSS = Quasi-Stationary State (e.g. "Refluxed").
[e]CHI = Coaxial Helicity Injection, SIHI = Steady Inductive Helicity Injection.
[f]Power to plasma needed to maintain configuration, magnetic field, or plasma current.
[g]Mean values if not axisymmetric.
[h]Measured or estimated from power balance, size, beta, or ne, Te, and Ti.
[i]τ_r ($\tau_{Ti=Te}$) is relevant time scale for configuration redistribution (temperature equilibration).

APPENDIX A

METHODOLOGY FOR EVALUATION AND PRIORITIZATION

In this appendix we provide an overview of how the panel addressed the charge to FESAC from Dr. Orbach. Where relevant, we describe how the necessary information was obtained from appropriate experts in the fusion community.

A.1. ADDRESSING THE CHARGE TO THE PANEL

The charge from Dr. Ray Orbach to FESAC asked for a critical evaluation of "the status of, and scientific opportunities for, major alternate magnetic confinement configurations." As noted in the Chapter 2 overview, the major alternate magnetic configurations subject to this review are the stellarator, spherical tokamak (torus), reversed field pinch (RFP), and compact tori (the spheromak and field-reversed configurations-the FRC). These four concepts rely on toroidal magnetic field configurations with closed magnetic flux surfaces for energy and particle confinement to reach fusion conditions.

The charge specifically requested that FESAC:

A. "Identify and justify a long-term objective for each concept as a goal for the ITER-era" that is "seen to have promise for fusion energy," and then

1. "critically evaluate the goal chosen for each concept, and its merits for fusion development;
2. identify and prioritize scientific and technical questions that need to be answered to achieve the specified goal;
3. assess available means to address these questions; and
4. identify research gaps and how they may be addressed through existing or new facilities, theory and modeling/computation."

B. "Identify and prioritize the unique toroidal fusion science and technology issues that an alternate concept can address, independent of its potential as a fusion energy concept."

The panel understands the ITER-era to span the next 15-20 years (2008 – 2028), during which ITER will be built, commissioned, and conduct experiments aimed at demonstrating fusion burn with energy gain Q>10. We anticipate success for ITER meeting its mission goals during this time, positioning world fusion programs to critically evaluate next steps in the development of fusion energy. The mission and goal statements for the alternate concepts covered by this report reflect this optimism.

A key challenge for this panel was to produce expert, but relatively unbiased information and evaluation for each of the concepts. This was accomplished by recruiting panel members from the US fusion science community eight scientific experts for the concepts (two advocates for each) and nine "at-large" members who might be familiar with one or more concepts, but were not currently working on any of them. The panel was subsequently organized into four concept working groups consisting of two experts/advocates and two at-large members. During internal panel deliberations, the at-large members also acted as a separate working group for initial goal assessment and prioritization.

In addition to the concept experts on the panel, we sought expert advice from the larger fusion community via a very interactive process. Early on, the panel invited concept advocacy groups to provide their answers to the questions contained in the FESAC charge in the form of short reports (~15 pages). In parallel, the larger US fusion research community (e.g., University Fusion Association, US Burning Plasma Organization, Fusion Power Associates, Laboratory Program Leaders, attendees at the annual ICC meeting, and PIs for major university experiments) was invited to submit brief written comment via the Panel's website. The panel subsequently discussed the written input received and responded with a set of questions for each concept. Concept experts then were invited to reply to the questions with presentations at an open Panel meeting in Dallas. Time was also provided for general input from the larger fusion community.

Early in its deliberations, the Toroidal Alternates Panel, in consultation with OFES and the chair of FESAC, decided that it would be premature to conclude that one or more of these concepts had no promise or potential for fusion energy and should be dropped from further consideration. Clearly some concepts are much closer to reaching fusion conditions than others, and some have much stronger theoretical or experimental support than others. The challenges facing some concepts are more evident and compelling than those faced by others. However, there are significant gaps in understanding (experimental and theoretical) for all the concepts that that rule out completing a definitive technical assessment of their ultimate capability towards fusion energy. We believe that our evaluation of the ITER-era goal and related discussion of the prioritized list of technical issues for each concept will provide a solid basis for planning future research investments.

A.2. IDENTIFICATION AND EVALUATION OF ITER-ERA GOALS

The charge letter asked the Panel to identify and critically evaluate ITER-era goals, which are two normally separate tasks that sometimes can be in conflict. To minimize this potential conflict, at the beginning of its deliberations the Panel asked those in the fusion research community working on each concept to put forward their goal for the ITER-era. After initial evaluation of these goals, the Panel provided feedback to the community suggesting modifications based on the panel's assessment. The concept experts on the Panel then worked with the community experts to revise the ITER-era goals. These revised goals were discussed extensively in light of detailed consideration of the physics issues during Panel meetings. Ultimately, the Panel used the following set of questions as the basis for understanding and critically evaluating the ITER-era goals for the concepts. Panel discussions were informed by scoring individual input on a scale of 1 (no) to 5 (yes) for each question and the language of the report reflects the consensus on these questions.

1. **<u>Overall mission</u>: How does the stated ITER-era goal for this concept contribute to the US fusion energy science research in terms of:**

   a. an eventual demonstration of a burning plasma?

   b. fusion development beyond ITER?

   c. fusion science independent of its potential as a fusion energy concept?

   The goals presented for each concept encompass a mix of these elements. Understanding this mix was the starting point for the goal assessment.

2. **<u>Importance and relevance</u>: Does the goal address critical scientific and technical issues to advance this concept and fusion science?**

   To better make this assessment, we identified four specific questions that, taken together, provide an overall assessment of important and relevance of the goal.

   a. Is the goal clear (i.e., Will we know when or if we achieve it)?

   b. Will reaching the goal significantly change the outlook for the concept (i.e., address the major issues for the concept)?

   c. Will reaching the goal resolve major issues for other concepts?

d. Will achieving the scientific goals for this concept significantly advance our knowledge of fusion plasma science?

3. **Technical risk: Are the goals reasonably achievable based upon the current state of knowledge for this concept?**

   The following questions seek to determine if the goal for each concept is overly ambitious, reasonable, or simply very modest, within the context of the overall US fusion program, tokamak research, and the full spectrum of existing alternate concepts research. The panel addressed these questions after identifying and prioritizing the scientific and technical questions as described in Section A.3 below.

   a. What degree of extrapolation in parameters or technical capability does the goal represent?

   b. Is there a sound scientific basis (theory and/or experiment) to anticipate success?

   c. Are the resource requirements significant, modest, or minimal? (i.e., large enough for program-wide impact, doubling of effort, possible with relatively constant effort)

A.3. PRIORITIZATION OF SCIENTIFIC AND TECHNICAL QUESTIONS

The ITER-era goal for each concept is the primary driver for the prioritized list of scientific and technical issues laid out in Chapters 4-7 of this report. As in the case of the ITER-era goals, the Panel sought input from concept experts to identify and prioritize the issues. This too was an interactive process between the Panel and the US research community and forms the technical basis for the report. Many of these issues have largely been identified in earlier reports, such as the 1999 FESAC report, Opportunities in the Fusion Energy Sciences Program, and the 2005 FESAC report, Scientific Challenges, Opportunities and Priorities for the U.S. Fusion Energy Sciences Program. We should note that in the concept chapters, we have used the term “issue” instead of “question” because in many cases this more clearly identifies the specific research goal that must be achieved for the concept to reach its ITER-era goal.

The issue prioritization process adopted by the Panel was adapted from that used by the 2007 FESAC Greenwald panel to produce their report entitled Priorities, Gaps and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy. The prioritization of the issues was based on three general considerations; namely,

- Importance: How important is it to resolve the issue in order to achieve the ITER-era goal?
- Urgency: How important is it to resolve the issue now rather than later?
- Generality: Will resolving the issue have broad impact on the US fusion program?

Prioritization into one of three Tiers was based on a set of ranking criteria also adapted from the Greenwald Panel process, with Tier 1 issues being highest priority and Tier 3 lowest. The ranking criteria, shown below, provided a means for uniform assessments for the various concepts.

Tier 1 Issue Criteria

1. Issue is critical for reaching the agreed upon goal.
2. Resolution of this issue requires major extrapolation from current state of knowledge.
3. Progress on this issue is essential before other research areas can be adequately addressed.
4. Scaling is untested and/or physics uncertain.
5. Issue contributes in an important way to the viability of the concept as a fusion energy source.
6. Progress would have the broadest impact on fusion and plasma science.

Tier 2 Issue Criteria

1. Issue is important for reaching the goal and/or for the viability of the concept as a fusion energy source.
2. Resolution of this issue requires major extrapolation from current state of knowledge.
3. Only limited scaling data and physics basis exist.
4. Progress on this issue would be helpful for research on other configurations.
5. Progress would have a moderate impact on fusion science.

Tier 3 Issue Criteria

1. Reaching the goal will require moderate extrapolation from current state of knowledge.
2. Some scaling data and/or a partially validated physics basis are available.
3. Present status does not hinder progress on other issues.
4. Information for resolving this issue may come from other parts of the FES program.
5. Progress would have a narrow impact on fusion science.

The full panel considered the complete list of issues for each concept using these ranking criteria and the prioritization was revisited several times over the time that the panel was active to confirm the findings. We did not aim to have equal numbers of Tier 1 issues across the concepts, nor did we seek a precise prioritization of the issues within each Tier, though the highest priority issues generally appear first in the text in each chapter. We also did not try to normalize the prioritization between concepts; that is to say, a given Tier 1 issue for one concept may be more significant than a Tier 1 issue for another concept, depending on the maturity of the concept development.

APPENDIX B

PANEL CHARGE LETTER

Under Secretary for Science

Washington, DC 20585

February 20, 2008

Professor Stewart C. Prager
Chair, Fusion Energy Sciences Advisory Committee
Department of Physics
University of Wisconsin
1150 University Avenue
Madison, Wisconsin 53706

Dear Professor Prager:

The October 2007 FESAC report, entitled "Priorities, Gaps, and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy," identified the scientific gaps and opportunities that lie ahead in fusion science research for developing the knowledge base for fusion energy after success on ITER. While many of the issues raised are generic in nature, that report focused on issues arising from development of scientific understanding concentrating mainly on the tokamak confinement concept. In addition to the tokamak, alternate magnetic confinement concepts are being studied, at varying levels of effort, in the Fusion Energy Sciences program.

The last detailed discussion of alternate concepts is contained in the Alternative Concepts Sub-panel Report in the July 1996 FESAC Report. It was noted that there are two reasons for research in alternate confinement configurations. First, such investigations can advance fusion energy science to produce knowledge and discoveries not possible through the study of one configuration only. Second, an alternate concept might itself evolve into a fusion energy system.

To continue the planning process for a robust, integrated fusion science program in the ITER era, it is important to critically evaluate the status of, and scientific opportunities for, major alternate magnetic confinement configurations. The concepts that have attracted the most attention, and are the focus of this charge, are the stellarator, spherical torus, reversed field pinch, and compact tori (spheromak and field-reversed configuration). This charge is a follow-on to the 2007 FESAC report mentioned above, and is expected to follow a similar methodology where appropriate. However, the scope of this charge covers both reasons for alternate concept research.

For those concepts that are seen to have promise for fusion energy, please identify and justify a long-term objective for each concept as a goal for the ITER era. Each goal should, at a minimum, be eventual demonstration of a burning plasma, or a rationale for gaining relevant burning plasma information from ITER experiments, so that the concept could credibly contribute to fusion development beyond ITER. With that in mind, I ask that FESAC: 1) critically evaluate the goal chosen for each concept, and its merits for fusion development; 2) identify and prioritize scientific and technical questions that need to be answered to achieve the specified goal; 3) assess available means to address these questions; and 4) identify research gaps and how they may be addressed through existing or new facilities, theory and modeling/computation.

It is also important to elucidate the merit of an alternate configuration even if it does not extrapolate to a fusion energy concept itself. I thus ask that FESAC identify and prioritize the unique toroidal fusion science and technology issues that an alternate concept can address, independent of its potential as a fusion energy concept. These specific issues should improve our basic understanding of toroidal confinement and/or synergistically improve potential fusion energy confinement systems through integrated program-wide science campaigns.

In my earlier charge to FESAC, dated February 7, 2007, I noted that a second charge would likely be sent to FESAC to help in developing a long-term strategic plan. I now anticipate a series of charges to FESAC to better address each major component of the program. The emphasis here should be on scientific issues and opportunities, while reserving more detailed considerations of specific activities and initiatives to the later discussions. Your reports from these charges will be integrated with additional studies within the fusion research community to build a resource-loaded long-term plan for the Office's programs.

Please respond to this charge by October 1, 2008.

Sincerely,

Raymond L. Orbach

Raymond L. Orbach

APPENDIX C

PANEL MEMBERS AND WORKING GROUPS

FESAC Toroidal Alternates Panel Members

David Anderson	University of Wisconsin	dtanders@wisc.edu
Jeff Freidberg	MIT	jpfreid@mit.edu
Martin Greenwald	MIT	g@psfc.mit.edu
Houyang Guo	TriAlpha Energy	hguo@trialphaenergy.com
Rich Hazeltine (VC)	U. Texas	rdh@physics.utexas.edu
Dave Hill (Chair)	LLNL	hilldn@fusion.gat.com
Bick Hooper	LLNL	hooper1@llnl.gov
Hantao Ji	PPPL	hji@pppl.gov
Tim Luce	General Atomics	luce@fusion.gat.com
Dale Meade	FIRE	dmeade@pppl.gov
Jon Menard	PPPL	jmenard@pppl.gov
Martin Peng	ORNL	pengym@ornl.gov
John Sarff	U. Wisconsin	jssarff@facstaff.wisc.edu
John Sheffield	ISSE @ U. Tennessee	jsheffi1@utk.edu
Xianzhu Tang	LANL	xtang@lanl.gov
Ed Thomas	Auburn U.	etjr@physics.auburn.edu
Mike Zarnstorff	PPPL	zarnstor@pppl.gov

Working Group Structure

Panel Members	ST	Stellarator	RFP	CT	At Large
Dave Hill (C)					X
David Anderson		E(L)			
Jeff Freidberg				AL	X
Martin Greenwald	AL				X
Houyang Guo				E(L)	
Richard Hazeltine (VC)	Th	Th			X
Bick Hooper				E	
Hantao Ji			E(L)		
Tim Luce		AL			
Dale Meade			AL		X
Jon Menard	E				
Martin Peng	E(L)				
John Sarff			E		
John Sheffield	AL				X
Xianzhu Tang			Th	Th	X
Ed Thomas		AL			X
Mike Zarnstorff		E			
Totals	5	5	4	4	9

(E) expert; (AL) at-large member; (L) working group leader; (Th) Theory support

APPENDIX D

TOROIDAL ALTERNATES PANEL MEETING DATES AND LOCATIONS

Bi-weekly teleconferences with the full panel were scheduled from April 10 through August 18, 2008

Weekly teleconferences with the full panel were scheduled from August 18 through November 11, 2008.

Three face-to-face meetings took place with the full panel:

1. June 30 — July 2, 2008

   Location: Dallas Wynham DFW North

   Subjects/purpose: prioritization and evaluation methodology, structure of report, concept goals, receive community input

2. August 5–7, 2008

   Location: Gaithersburg Hilton, MD

   Subjects/purpose: goal evaluation, issue prioritization, report structure, introductory chapter, working group breakout meetings

3. October 1–2, 2008

   Location: Holiday Inn Town Lake, Austin, TX

   Subjects/purpose: review of issues, discussion of facilities, gaps, findings and recommendations, plan for report editing, short-form summary of report findings

APPENDIX E

WRITTEN INPUT REQUEST LETTERS

April 22, 2008

Dear Colleague,

The newly formed FESAC panel, charged to assess toroidal alternate concepts for magnetic fusion, is beginning its work. The panel has been asked to "critically evaluate the status of, and scientific opportunities for, major alternate magnetic confinement configurations," which are the stellarator, the spherical torus, the reversed field pinch, and compact tori (spheromak and field-reversed configurations).

Specifically, the panel has been asked to "identify and justify a long-term objective for each concept as a goal for the ITER era," i.e., the next 15-20 years. Further, the panel has been asked to

1 "critically evaluate the goal chosen for each concept, and its merits for fusion development;

2 identify and prioritize scientific and technical questions that need to be answered to achieve the specified goal;

3 assess available means to address these questions; and

4 identify research gaps and how they may be addressed through existing or new facilities, theory and modeling/computation."

Finally, the charge asks us to "identify and prioritize the unique toroidal fusion science and technology issues that an alternate concept can address, independent of its potential as a fusion energy concept." A copy of the complete charge letter is attached.

We are asking for your help as we gather information to address Dr. Orbach's charge to FESAC by providing us a succinct report (10-15 pages) in response to the key elements of the charge as outlined above. *We are seeking one report per concept at this time, so we encourage the groups working on related concepts to work together on this.* The CT groups may want to consider separating spheromaks and FRCs. The scope of the charge includes theory, computation, and any necessary supporting technology. If it seems best to produce several shorter documents on the concept in order to provide input sooner, that would fine (i.e., one document to define goals and one to define discuss other elements of the charge). We hope to use this input to inform our panel discussions over the next couple of months. We would like to receive these by the end of May 2008 at the latest.

In addition to this request for written input from each of the concepts, we are planning on making a more general call to the broader fusion community for 2-page web-based submissions and will be inviting representatives from each concept to meet with the full panel, most likely in early July. Overall, we expect this to be an iterative working process between the panel and concept advocates; representative experts from each of the four concepts on the panel can help facilitate these interactions. Therefore, it's not necessary to perfect your input before submitting it.

We are now studying existing information on the four concepts contained in prior FESAC reports (these can be found on the panel's web site (http://fusion.gat.com/tap) as we determine next steps in our work. You may want to review these as you prepare your input, in particular, the two-page summaries of concepts from the 1999 FESAC panel report, which gives a 5 year and 20 year view for each concept.

Please feel free to contact others on the panel or me directly if you have questions.

David N. Hill (Panel Chair)

Distribution list:

ST: M. Ono, R. Raman, R. Maingi, S. Sabbaugh, C. Hegna, R. Majeski
Stellarator: S. Knowlton, D. Anderson, M. Zarnstorff, J. Harris
RFP: S. Prager, B. Chapman, D. Den Hartog, G. Fiksel, J. Goetz
CT: A. Hoffman, T. Jarboe, H. McLean, G. Wurden, M. Brown

cc:

PPPL: R. Goldston, R. Hawryluk
ORNL: S. Milora, D. Hillis
LANL: G. Wurden
UFA: S. Knowlton
USBPO: J. VanDam

Attachment: Charge letter to FESAC from Dr. Ray Orbach

May 6, 2008

Dear Colleague,

The FESAC Toroidal Alternates Panel has been asked to “critically evaluate the status of, and scientific opportunities for, major alternate magnetic confinement configurations,” which are the stellarator, the spherical torus, the reversed field pinch, and compact tori (spheromak and field-reversed configurations).

Specifically, the panel has been asked to “identify and justify a long-term objective for each concept as a goal for the ITER era,” i.e., the next 15-20 years. Further, the panel has been asked to

1. “Critically evaluate the goal chosen for each concept, and its merits for fusion development;
2. Identify and prioritize scientific and technical questions that need to be answered to achieve the specified goal;
3. Assess available means to address these questions; and
4. Identify research gaps and how they may be addressed through existing or new facilities, theory and modeling/computation.”

Finally, the charge asks us to “identify and prioritize the unique toroidal fusion science and technology issues that an alternate concept can address, independent of its potential as a fusion energy concept.” The scope of the charge includes theory, computation, and any necessary supporting technology.

We have already requested that groups working on these concepts provide the panel with succinct reports (one 10-15 page report per concept) in response to the key elements of the charge as outlined above. (The CT groups we given the opportunity to separate spheromaks and FRC concepts.) We expect to receive these papers by the end of May.

We also want to provide an opportunity for scientists in the broader fusion community to advise us, so we have created a new link on the Panel's website: http://fusion.gat.com/tap/ . Please keep your comments as brief as possible. Avoid lengthy introductory remarks: the essential benefits proposed by concept proponents are well known. We are seeking the technical information on these concepts necessary to make the critical evaluation and prioritization called for in the charge.

We will make this information, as well as the 10-15 pg concept reports, available to the community. It would be best to post pdf or plain text files, or links to such, including a brief file descriptor. For the open forum, there will be a short delay between submission and posting to allow for spam filtering. If your input is related to a specific concept, please select the appropriate area for posting.

Dave Hill (panel chair)

Toroidal Alternates Panel Meeting Agenda

Wyndham Hotel North, Dallas Fort-Worth Airport

June 30 – July 30, 2008

- **Monday, June 30**
  - 8:30–10:00 Panel prepatory discussion on ST and stellarator
    Break
  - 10:00–12:00 ST Goals, Issues, Gaps (Public meeting)
  - 12:00–1:30 Lunch
  - 1:30–3:30 Stellarator Goals, Issues, Gaps (Public meeting)
    Break
  - 3:30–4:30 Public comment session (by prior request)
  - 4:30–6:00 Panel discussion of presentations, goal evaluation methodology

- **Tuesday, July 1**
  - 8:30–10:00 Panel preparatory discussion on CT and RFP
    Break
  - 10:00–12:00 CT Goals, Issues, Gaps (Public meeting)
  - 12:00–1:30 Lunch
  - 1:30–3:30 RFP Goals, Issues, Gaps (Public meeting)
    Break
  - 3:30–4:30 Public comment session (by prior request)
  - 4:30–6:00 Panel discussion of presentations, issue prioritization methodology

- **Wednesday, July 2**
  - 8:30–11:30 Working Group Discussions
  - 11:30–12:15 ST and Stellarator Working Group reports (20 min each)
  - 12:15–1:15 Lunch
  - 1:15–2:00 CT and RFP Working Group reports (20 min each)
  - 2:00–3:00 Discussion and cross cutting issues and report content

APPENDIX F

LIST OF COMMUNITY INPUT DOCUMENTS AVAILABLE ON THE TAP WEBSITE AT HTTP://FUSION.GAT.COM/TAP/COMMUNITY

General Subject Submissions

Classic 1995 Abdou paper on VNS by Hill, David (LLNL)
a standard reference by Mohammed Abdou on neutron fluence for testing
- File: 1995-Abdou VNS.pdf

Requirements for component testing by Hill, David (LLNL)
2 vg from a presentation by Mohammed Abdou on requirements for a CTF
- File: Two VGs on Testing Abdou.pdf

Development Path by Goldston, Rob (PPPL)
Development path spending profile
- File: Dev Path for FESAC TAP.pdf

Updated Concept Parameter Tables by Hill, David (LLNL)
some corrections to the concept parameter tables: MS Word document
- File: 20080627 concept key parameters.doc

PDF file version of concept parameter tables by Hill, David (LLNL)
Latest version of concept parameter tables in pdf format
- File: 20080627 concept key parameters.pdf

ICC2008 Overview of TAP process by Hill, David (LLNL)
slides from my talk at the 2008 ICC Workshop presenting overview of the Toroidal Alternates Panel
- File: 20080625 ICC-TAP-overview-v4.pdf

Concept Parameters Table-updated by Hill, David (LLNL)
fixed some typos and notation
- File: 20080617 concept key parameters.pdf

Program goals by Majeski, Dick (PPPL)
Revision of a suggested goal for the alternates program
- File: Suggested alternates goal-rev.pdf

Overall programmatic direction by Majeski, Dick (PPPL)
Suggested goal for the toroidal alternates program
- File: Suggested-alternates-goal.pdf

Clarification letter on community input by Hill, David (LLNL)
the attached pdf file may help clarify the Panel's request for community input based on recent Panel discussion
- File: input clarification letter.pdf

ST-Related Submissions

Response to FESAC questions 3a and 2 by Majeski, Dick (PPPL)
Presentation at the 30 June FESAC TAP meeting with response to question 3a. Response to question 2 is also appended (but was not presented).
- File: FESAC-TAP-3a-6=30=1000.pdf

FESAC_TAP_STQ3b/neutral beam revised by Grisham, Larry (PPPL)
the earlier neutral beam technology file inadvertently left out discussion of power density, which this includes
- File: FESAC_TAP_STQ3b-wport.pdf

Movie of vertical assembly of ST CTF by Rasmussen, Dave (ORNL)
vertical assembly of ST CTF
- File: CTF Vertical May 31, 2007.avi

Movie of ST CTF by Rasmussen, David (ORNL)
movie in .avi format of ST CTF overview
- File: ctf overview may 16, 2007.avi

Movie of ST CTF by Rasmussen, David (ORNL)
movie in .avi format of ST CTF overview
- File: ctf overview may 16, 2007.avi

Super X divertor for ST development by Kotschenreuther, Mike (U. Texas)
Super-X divertor for ST development
- File: SuperX divertor FESACJune08.4.pdf

Answers to ST question 1 by Rasmussen, Dave (ORNL)
Dave's talk on behalf of Brad Nelson addressing panel question 1
- File: ST Q1 Presentation-v6.pdf

2-pager for FESAC TAP Q1 by Rasmussen, David (Oak Ridge National Laboratory)
What are the essential features of the device that would fulfill the ST goal
- File: FESAC_TAP_Q1_v4c.pdf

Q5 by La Haye, Robert (GA)
Two pager for Questions 5
- File: Q5.doc

Q3b by Tsai, Jim (ORNL)
NBI Technology Issues for ST Goal
- File: FESAC_TAP_STQ3b-r3.doc

Q3b by Tsai, Jim (ORNL)
NBI Technology Issues for ST Goal
- File: FESAC_TAP_STQ3b-r3.pdf

Response to question 2 by Sontag, Aaron (UW - Madison)
Response to Question 2) from FESAC Panel Feedback to the ST Community
- File: Question 2_v3.pdf

Updated ST Community Input Document by Ono, Masayuki (PPPL)
Updated ST Community Input Document with factual corrections 6/27/08
- File: ST_Community Inputs to TAP_6_27_08.pdf

ST Question response by Ono, Masayuki (PPPL)
A two page response to the ST question 4a
- File: FESAC_TAP_Q4a_response.pdf

Response to Question 4c) by Ying, Alice (UCLA)
Reply requested by the FESAC TAP to Question 4c on ST Fusion nuclear operations and tritium
- File: FESAC_Inputs Q4c.pdf

Response to Question 4b) by Sabbagh, Steven (Columbia University)
Reply requested by the FESAC TAP to Question 4b on ST Macrostability
- File: FESAC_TAP_Q4b_response_v2.pdf

Community response to question 6 by Sontag, Aaron (UW - Madison)
Response to Question 6) from FESAC Panel Feedback to the ST Community
- File: Question 6_v3.doc

Response to question 3a by Majeski, Dick (PPPL)
Response to the question "How will the ST program address startup/sustainment, transport, boundary physics?
- File: Question_3a_response.pdf

Plasma start-up by Shevchenko, Vladimir (EURATOM/UKAEA, Culham Science Centre)
New results of EBW-assisted plasma start-up in MAST
- File: EC15_paper_VS.pdf

Gaps and Facilities by Goldston, Rob (PPPL)
A Perspective on Gaps and Facilities for the Spherical Torus
- File: Gaps and Facilities, Spherical Torus.pdf

ST CTF by Sontag, Aaron (University of Wisconsin - Madison)
2005 paper by M. Peng on ST-CTF conceptual design
- File: Peng-CTF05-ppcf5_12B_S20.pdf

Suggested elements of FESAC Panel question 3a by Peng, Menard, Martin, Jon (ORNL, PPPL)
More details for Question 3a from the FESAC Panel are provided
- File: Elements of Question 3a-Jun1608.pdf

ST power plant by Buttery, Richard (UKAEA)
Wilson 2004 NF paper on an ST power plant
- File: Wilson ST PP.pdf

ST CTF by Buttery, Richard (UKAEA)
Wilson IAEA 2004 paper on ST CTF design
- File: FT_3-1Ra.pdf

FESAC Panel feedback questions by Peng, Martin (ORNL)
The FESAC Toroidal Alternates Panel seek answers from the ST community on a list of additional questions regarding the ST priorities, gaps, and opportunities.
- File: Toroidal_Alternates_Panel_feedback_on_ST_Jun1308.pdf

ST community input by Ono, Masa (PPPL)
Community input document for the ST
- File: ST Inputs to FESAC Alt Panel-6-5-08.pdf

Stellarator-Related Submissions

Constructability, Plan for Simplification by Neilson, Hutch (PPPL)
Stellarator community will conduct in-depth engineering and physics studies to simplify and improve constructability.
- File: Constructability_Stellarators_3.pdf

Stellarator plans by Knowlton, Stephen (Auburn University)
Short white paper on topics of near-term focus in US stellarator research
- File: Stellarator_WP_TAP_08_04_bb.doc

Mission and Goals by Anderson, David (UW-Madison)
pdf version of Mission and Goals 7_11_08
- File: Stellarator_Mission_&_Goals.pdf

Mission and Goals by Anderson, David (UW-Madison)
TAP version of mission and goals 7_11_08
- File: Stellarator_Mission_&_Goals.doc

Backup material for stellarator answers by Knowlton, Steve (Auburn University)
backup slides for answers to TAP questions to stellarator communit
- File: TAP_Stell_backup answers.pdf

Stellarator answers to panel questions by Harris, Jeff (ORNL)
answers from stellarator community
- File: stellarator_tap_15.pdf

Response to panel's questions by Knowlton, Stephen (Auburn University)
Written response to questions submitted by TAP to stellarator community
- File: Stellarator_Questions v.3.pdf

Stellarator Assembly Experience by Knowlton, Stephen (Auburn University)
Document on W7-X construction experience submitted in response to specific question from TAP.
- File: Experience gained during W7X.pdf

Stellarator Constructability by Neilson, Hutch (PPPL)
Status of NCSX coils and assemblies, problems encountered, significance for future stellarator
- File: Stellarator Constructability and NCSX.pdf

Quasi-Axisymmetry and ITER by Boozer, Allen (Columbia University)
A one-page discussion of the importance of quasi-axisymmetric shaping to the achievement of the programmatic goal of ITER.
- File: QA-ITER(6-08).pdf

Numerical Simulation Initiative for Stellarators by Reiman, Allan (Princeton Plasma Physics Lab)
A major initiative to develop a predictive numerical simulation capability for stellarators, in combination with a strong experimental collaboration program, would allow the US to play a prominent role in the international stellarator program, and would help the US fusion program to derive benefit from the large international stellarator facilities.
- File: Code Development Program for Stellarators 6_19.doc

Gaps and Facilities by Goldston, Rob (PPPL)
A Perspective on Gaps and Facilities for the Compact Stellarator
- File: Gaps and Facilities, Compact Stellarator.pdf

Stellarator Questions by Anderson, David (University of Wisconsin-Madison)
Questions the panel would like addressed at the Dallas Meeting
- File: Stellarator_Questions.doc

Stellarator community input by Knowlton, Stephen (Auburn University)
15 page input document from the stellarator community
- File: Stellarator_TAP_final.pdf

Role of Non-axisymmetric shaping for DEMO by Boozer, Allen (Columbia University)
Draft of the paper that accompanies the 2008 EPS invited paper on Stellarators and the path from ITER to DEMO
- File: Boozer_EPS(2008).pdf

RFP-Related Submissions

DFW presentation on the RFP by Prager, Stewart (U. Wisconsin)
Presentation at DFW meeting responding to questions from panel: Stewart Prager and Daniel Den Hartog
- File: RFP-response-prager-denHartog.pdf

RFP table of parameters by Chapman, Brett (UW-Madison)
Table of present, ITER-era, and reactor-target RFP parameters (from RFP community)
- File: RFP parameter table.pdf

Questions to RFP White Paper by Ji, Hantao (PPPL)
questions by TAP
- File: Qs-RFP-final.doc

Community Input by Prager, Stewart (U. Wisconsin)
Community Input Document on RFP for the FESAC Panel
- File: RFP FESAC.pdf

CT-Related Submissions

Hoffman's answers to Panel questions by Hill, David (LLNL)
June 20th file from Alan Hoffman with pre-DFW answers to FRC questions
- File: FRC answers-Hoffman-June20.pdf

CT mission and goals by Guo, Houyang (University of Washington)
CT mission and goals suggested by TAP based on the CT community input
- File: CT_Mission_And_Goals.pdf

Spheromak presentation at DFW meeting by Jarboe, Tom (U. Washington)
Presentation to panel at DFW meeting: answering questions on the spheromak
- File: UPload Jarboe talk.pdf

FRC July 1 TAP presentation by Hoffman, Alan (University of Washington)
Slide show presented to TAP 7/1/08
- File: FESAC [Compatibility Mode].pdf

5 minute slot by Woodruff, Simon (Woodruff Scientific, LLC)
Highlighting some of the differences between Jarboe's answers to panel questions and those of Woodruff, Sovinec and McLean
- File: Woodruff-5-minutes.pdf

Pulsed High Density FRC Fusion Presentation by Slough, John (University of Washington)
Viewgraphs presented to FESAC panel at DFW meeting
- File: Viewgraphs for FESAC panel on PHD.pdf

Updated FRC design aid by Cohen, Sam (PPPL)
updated FRC design aid
- File: FRC-Design-Aid-COHEN.pdf

Slides from talk by Jarboe, Tom (University of Washington)
Slides from my talk July 1

FRC designs by Cohen, Samuel (PPPL)
A diagram showing present and proposed FRCs in the r-B plane, with s and nu indicated*
- File: FRC_Design_Aid_Diagram.pdf

Design aid for FRC by Cohen, Sam (PPPL)
Design aid for FRCs
- File: 5minuteFESAC Cohen.ppt

Spheromak panel question answers by Woodruff, Simon (Woodruff Scientific, LLC)
Answers to panel questions from Woodruff, Sovinec and McLean
- File: FESAC-Q&A-SW-CS-HSM.pdf

Answers to questions on spheromak CT by Jarboe, Thomas (University of Washington)
Answers to the question for the spheromak CT
- File: Spheromak questions with answers.pdf

FRC Questions and Issues by Guo, Houyang (University of Washington)
Request for community input at DFW meeting
- File: FRC_Questions.doc

Rotamak FRC by Majeski, Dick (PPPL)
A brief discussion of the potential importance of propagating waves effects in rotamaks
- File: Rotamaks and waves.pdf

Spheromak questions and issues by Hooper, Bick (LLNL)
Request for community input for Dallas meeting
- File: Spheromak questions and issues.pdf

SPIRIT Concept and CT Research on MRX by Yamada, Masaaki (PPPL, Princeton University)
Description of experimental CT research on MRX
- File: SPIRIT_2 pager_pdf.pdf

Quasi-steady FRC Fusion Reactor by Slough, John (University of Washington)
A unique approach to achieving a burning plasma in ITER era that employs the "conventional" regime of a high density FRC where the FRC has already demonstrated the confinement scaling and stability required for fusion gain at confining fields of 15 T. To achieve the fusion regime, the FRC is repetitively formed, kinetically heated, and compressed into a high field SC solenoid. This process can be accomplished with high efficiency providing an ideal volumetric neutron source for component testing and tritium breeding.
- File: Quasisteady FRC Fusion Reactor.pdf

Oblate CTs by Schaffer, Michel (GA)
Oblate FRCs formed by counter-helicity spheromak merging; high magnetic flux CTs
- File: Oblate CTs.pdf

FRC 15 page community input-Hoffman by Hoffman, Alan (U. Washington)
15 page community input document on the FRC
- File: CT15pager-allW03-Hoffman.pdf

FRC 15-pager, community input by Cohen, Samuel (PPPL)
This report is in response to Dave Hill's request for a 10-15 page document describing the status of FRC research and gaps in the present research programs.
- File: FRC_FESAC_15pagerCohen.pdf

Oblate CTs by Schaffer, Michel (General Atomics)
Oblate FRCs formed by counter-helicity spheromak merging; high magnetic flux CTs
- File: Oblate CTs.doc

FRC 15-pager by Hoffman, Alan (University of Washington)
Concept description, goals, issues, and gaps for the FRC
- File: CT15pager-allW03.doc

Spheromak Community Input by McLean, Harry (Lawrence Livermore National Lab)
Spheromak Community 15 Pager Input
- File: spheromak_15-pager_v21_final.pdf

FRC 2-pager by Cohen, Samuel (PPPL)
This 2 page document describes what research gaps could be filled in a small FRC device, using RMF only for heating, stabilization and current drive.
- File: FESAC-2pager_Cohen-FRC.pdf