U.S. Spherical Torus Fusion Energy Science Research
Input to the 1999 Fusion Summer Study

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The White Paper on Spherical Torus (ST, Spherical tokamak, or Very Low Aspect Ratio Tokamak) provides a summary of the promise, challenges, and opportunities in fusion energy science stemming from the unique magnetic configuration and potential of the ST plasma. This paper shows that, by taking advantage of the new opportunities and meeting the challenges, ST research

- Introduces new scientific opportunities in which to advance magnetic fusion energy science;
- Will produce in the next 4–5 years new database for these opportunities using Proof of Principle and Concept Exploration experiments;
- Offers the possibility of a reduced-cost Performance Extension and Optimization ST experiment of modest size within the next decade, with the possibility of D-T fusion burn at high Q;
- Offers the potential for a reduced-cost steady-state technology-intensive Fusion Energy Development ST device of modest-size within the subsequent decade; and
- Offers the potential for a reduced-cost ST Pilot Plant (a minimum DEMO) aimed at electricity and tritium self-sufficiency near the beginning of the third decade from today.

This White Paper is organized according to the major issues defined for the 1999 Fusion Summer Study [Snowmass WebPage, 1999], when a community-wide discussion of the comprehensive scope of fusion energy sciences will be carried out. A set of *recommendations* is advanced in response to these issues. This discussion of ST research is to be updated based on the results of the Summer Study, and streamlined into the White Paper of U.S. ST Fusion Energy Science Research. The White Paper will be updated on a continual basis over longer term.

To achieve these, the White Paper covers the following topics organized in sections:

1. Spherical Torus and Research Mission
2. Status of ST plasma physics development in experiments, theory, and future visions
3. New scientific opportunities of ST relative to the Tokamak, the Advanced Tokamak (AT), and the Compact Toroids (CTs) (addressing Plasma Science issues)
4. Concept Exploration and Proof of Principle Research (addressing Magnetic Fusion Concept issues)
5. Performance Extension, Optimization, and Burning Plasma Research (addressing Magnetic Fusion Concept Burning Plasma and Next Step Option issues)
6. Fusion Energy Development (addressing Magnetic Fusion Concept Integration, Technology, and Energy Development Path issues)
7. Visions for Pilot Plant, DEMO, and Power Plant (address Power Plant vision issues)
8. International Collaboration
9. Development Metrics and Roadmap
10. Non-electric Applications
11. Recommendations and Continued Innovation

The ST concept saw the first Concept Exploration data with $T_e > 300$ eV at the beginning of this decade [Sykes et al, 1992]. Proof of Principle experimentation will presently begin while Concept Exploration tests continue to push the envelope of the ST scientific domain. It is important to note that the great progress during the past decades in Tokamak and AT physics [Goldston, 1996; Taylor et al, 1997] has provided a solid scientific and technology basis to enable this rapid advancement of the ST concept. While the promise and challenges of ST to be summarized here are enticing, it is believed that progress in ST and other innovative confinement concepts would provide in time an enlarged basis for further concept innovation toward attractive fusion energy.

1. Spherical Torus and Research Mission

A comparison of the ST, the Tokamak, and the Compact Toroid (CT) magnetic field configurations near the plasma edge is shown in Figure 3-1 below. Whereas all configurations form axisymmetric closed toroidal magnetic surfaces to confine plasma, large differences are

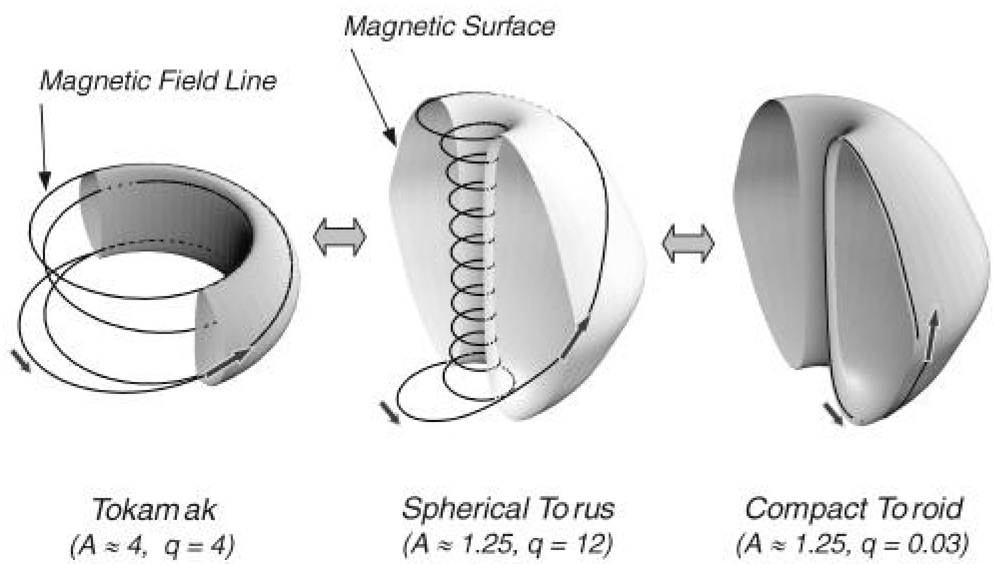

Figure 3-1. Magnetic field lines of Tokamak, Spherical Torus, and Compact Toroid plasmas on magnetic surfaces near the plasma edge. The scientific relationship among these confinement concepts can be defined by the similarities and contrasts between these configurations.

evident in the field line structure and aspect ratio. In the Tokamak the toroidal field is 5–10 times the poloidal field resulting in relatively low pitch angle (relative to the horizontal plane) of the field line throughout the plasma. In the AT the increased outward shift of the magnetic axis (due to increased poloidal β) tends to increase the outboard poloidal field and the pitch angle in fractions. In the ST the toroidal field is reduced to the poloidal field level or below in the plasma outboard region leading to a high pitch angle. In the CT the field line is nearly vertical in pitch due to the absence of toroidal field at the plasma edge.

It is seen that the ST combines short field line of bad curvature and high pitch angle toward the outboard plasma edge with long field line of good curvature and low pitch angle toward the inboard plasma edge. A consequence of dominating good field curvature is MHD stability at high plasma pressure in reduced magnetic field (i.e., high β). This advantage in β by reducing the aspect ratio in moderate to high-q toroidal configuration has been identified for some time in the tokamak research [Sykes et al, 1983; Troyon et al, 1984]. The aspiration of order-unity β without relying on an applied toroidal field has been an overarching goal of the CT research.

The mission of ST research in the U.S. is fully to support the policy goals of the U.S. Fusion Energy Sciences Program [FESAC, 1996], which are to

- Advance plasma science in pursuit of national science and technology goals.
- Develop fusion science, technology, and plasma confinement innovations as the central theme of the domestic program.
- Pursue fusion energy science and technology as a partner in the international effort.

Special ST plasma properties originate from the unique magnetic field line configuration just described. Based on the Tokamak and the AT models extended to very low aspect ratios, the ST is capable of operating in a much different scientific parameter domain (see Section 3). This parameter domain of high-temperature collisionless plasmas produced with modest heating power is characterized simultaneously by high β, an absolute "magnetic well", a plasma dielectric constant up to $\sim 10^2$, flow shearing rates up to and above 10^6/s, fully aligned ∇p-driven current profiles carrying significant (> 70%) plasma current, and supra-Alfvén particles. The investigation of the ST plasma therefore promises opportunities to advance plasma science in new and exciting ways, in support of the first policy goal.

The much enlarged scientific domain promised for the ST plasma introduces opportunities for innovation in experimentation, diagnostics, theory, and plasma enabling technologies (heating, fueling, current drive, etc.). Proof of Principle (NSTX [Ono, M. et al, 1998, Kaye et al., 1999], MAST [Cox et al, 1998]) and Concept Exploration (Pegasus [Fonck et al, 1998], HIT-II [Jarboe et al, 1999], CDX-U [Majeski, Kaita et al, 1998], Globus-M [Golant et al, 1997], etc.) research (see Section 4) will stimulate many such innovations. Success in these ST tests will provide the database needed for a practical Performance Extension device ($R_0 \leq 1.2$ m), which can be implemented in a cost effective fashion by utilizing existing fusion facility and equipment and with the possibility of D-T fusion burn at high Q. The very low aspect ratio, limiting the space in the hole through the middle of the ST plasma, also drives innovation in science (such as noninductive startup to full current), technology (such as Laser Forming [Crawford and Beamann, 1999] for building the monolithic normal conducting center leg for the toroidal field coil), and design (such as the fully demountable center stack assembly). The development of the ST confinement concept, with an emphasis on affordable steps toward practical fusion energy, will present new as well as challenging opportunities for innovation by all disciplines of the fusion program, in support of the second policy goal.

Steady-state ST devices of modest size ($R_0 \leq 1.2$m) and fusion power (≤ 70 MW) are envisioned for the technology-intensive Fusion Energy Development step, assuming success in achieving the first two policy goals. Such devices have relatively modest requirements compared to ITER, and will foster collaborative partnership as well as constructive competition in the world fusion community, in support of the third policy goal.

The policy goals of the U.S. Fusion Energy Sciences Program are therefore best served by investigating in-depth the scientific properties of the ST fusion plasma, and given positive results, building and utilizing reduced-cost ST devices to pursue the science and technology of the U.S. fusion program. Before addressing the envisioned ST research and future development, we present a brief summary of the present status of ST development.

2. Present Status of ST Development

Broad and important progress has recently been made in pioneering plasma tests, theory, new experiments, and envisioned ST device concepts for the future. The parameters of these experiments and device concepts are summarized in Table 2-1, and in the "data sheets" provided in Appendix 2-1. These have substantially improved and broadened our awareness of the possible properties of the ST plasma and its promise and challenge in the future, since the preliminary calculation of the ST MHD equilibrium more than a decade ago [Peng et al, 1986].

Data for plasma currents up to 310 kA, temperatures up to several hundred eV, and densities in the range of 10^{14} cm^{-3} have been obtained from several pioneering, Concept Exploration ST experiments of limited capabilities. The Small Tight Aspect Ratio Tokamak (START, U.K.), among many results, achieved under neutral beam injection average toroidal betas β_T up to 40% (with q ~ 3 at plasma edge), and with Ohmically heating alone β_T up to 24%, in well-confined H-mode-like ST plasmas [Sykes et al, 1997; Gates et al, 1998; Gryaznevich et al, 1998; Sykes et al, 1998; Morris et al, 1999; Conway et al, 1999]. It also showed vertically stable plasma elongations as high as 3.5 given adequately hollow plasma current profiles [Sykes et al, 1994]. The Helicity-Injected Tokamak-I and II (HIT-I and II, University of Washington) [Jarboe et al, 1997; 1999] and Himeji Institute Spherical Torus (HIST, Japan) [Nagata et al, 1998] achieved plasma currents up to 250 kA by applying radial electric field near the single-null region of the plasma (i.e., Coaxial Helicity Injection, CHI). The Current Drive Experiment-Upgrade (CDX-U, PPPL) [Ono et al, 1996] and START measured negligible halo currents during (sometimes forced) disruptions. The Tokyo Spheromak-3 (TS-3, Japan) [Morita et al, 1997] showed that a small external toroidal field with coil currents as little as 10% of the plasma current can prevent the plasma tilt instability, converting a Spheromak into a ST of aspect ratios (A) as low as ~1.1. There it has also shown the possibility of a Field Reversed Configuration (FRC)-like ST equilibrium with β_T approaching 50% [Ono, Y. et al, 1998]. These and other interesting properties have confirmed the equilibrium and stability features projected in early theories of the ST plasma. Important contributions to toroidal confinement physics [Morris et al, 1999] have been made despite of the limited parameter ranges of these experiments (see Table 2-1).

These results have helped approval of new and/or more powerful ST experiments such as NSTX [Ono, M. et al, 1998], MAST [Cox et al, 1998], Pegasus [Fonck et al, 1998], Globus-M [Golant et al, 1998], etc., and have stimulated an increasing list of theoretical calculations of interest to magnetic fusion energy over the longer term. Thanks to the broad knowledge base and analysis tools produced in (and modified from) the successful Tokamak and AT research [Goldston et al, 1994; Taylor et al, 1997], a range of interesting ST physics features have been

*Table 2-1. Nominal Design Parameters and Key Measurements (in **Bold**) of the World ST Devices (June 1999)*
(average T_i in keV, average density in $10^{20}\ m^{-3}$, β_T for average toroidal beta, P_{AUX} for auxiliary heating power)

Device	Stage	Country	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?	$T_i(0)$	$\langle n \rangle$	τ_E(s)	β_T(%)
Retired														
HIT-I	CE	US	**0.3**	**0.2**	**1.85**	**0.01**	**0.5**	**0.25**	**CHI**	Y		**0.6**		
START	CE	UK	**0.32**	**0.25**	**≤3.0**	**0.05**	**0.3**	**0.31**	**1.0**	Y	**0.35**	**1**	**0.005**	**40**
TST-M	CE	Japan	**0.35**	**0.25**	**1.5**	0.01	**0.3**	0.1	**Ohmic**	N		**0.1**		
Operating														
TS-3	CE	Japan	**0.2**	**0.19**	**0.8–2**	**0.001**	**0–0.2**	**0.08**	**Ohmic**	N	**0.05**	**1**		**50**
HIT-II	CE	US	**0.3**	**0.2**	**1.85**	**0.035**	**0.5**	**0.2**	**CHI, Ohmic**	Y				
HIST	CE	Japan	**0.3**	**0.24**	**2.0**	**0.005**	**0.2**	**0.15**	**Ohmic**	N	**0.03**	**0.5**		**10**
CDX-U	CE	US	**0.34**	**0.22**	**1.6**	0.05	**0.2**	0.2	**0.3**	Y	**0.1**	**0.5**	**0.001**	**5**
Under Construction														
ETE	CE	Brazil	0.3	0.2	1.8	0.02	0.6	0.4	Ohmic	N				
TST-2	CE	Japan	0.37	0.23	1.5	0.1	0.4	0.2	Ohmic	N				
TS-4	CE	Japan	0.5	0.45	0.8–3	0.005	0–0.5	0.3	Ohmic	N				
Commissioned or Achieved First Plasma														
Pegasus	CE	US	0.45	0.4	2–4	0.04	0.1	0.3	2	Y				
Globus-M	PoP	RF	0.36	0.24	2.2	<1	0.65	0.5	5	Y				
MAST	PoP	UK	≤0.85	≤0.65	2.5	5	0.5	2	6.5	Y				
NSTX	PoP	US	0.85	0.68	2	5	0.3	1	11	Y				
Design Concept Examples for Future Possibilities														
DTST	PE-BP	US	1.17	0.83	3	50	1.7	10	40	Y	~15	~1		23–34
CTF	FED	UK	0.8	0.53	2.3	s.s.	3.3	13.4	69	Y				
VNS	FED	US	1.1	0.78	3	s.s.	2.0	10	50	N	~15	~1		~23
FDF	PP	US	1.12	0.7	3	s.s.	2.9	13	25	Y	~20	~4	~0.67	<54%
ST Power	Elec.	UK	3.9	2.8	3	s.s.	1.9	37	100	Y	~25	~2		≥50
ARIES-ST	Elec.	US	3.2	2.0	3.4	s.s.	3.0	28	120	Y	~25	~2		≥50

(CE = Concept Exploration, PoP = Proof of Principle, PE = Performance Extension, BP = Burning Plasma, FED = Fusion Energy Development, PP = Pilot Plant, Elec. = Power Plant) identified. To name a few, MHD stable equilibria of β_T up to 60% ($\langle\beta\rangle$ up to 40%, β_N up to 9) with fully aligned ∇p-driven current up to ~100% for A = 1.1-1.6 have been calculated by researchers from the U.S. [Menard et al, 1997; Miller et al, 1997], U.K. [Hender et al, 1996], and R.F. [Medvedev, 1998; 1999]. Avoidance of unstable Neoclassical Tearing Modes (NTM) due to the large Pfirsch-Schlüter current in such equilibria (represented by the "Galsser" term [Glasser, 1988]) is suggested by researchers at University of Wisconsin [Kruger et al, 1998; Hegna et al, 1998]. High β at modest magnetic field leads to large ∇p-driven flow shearing rates of the order of 10^6 s^{-1}, first noted by General Atomic (GA) researchers [Stambaugh et al, 1996] to far exceed the rates needed so far in Tokamaks to suppress ion turbulence that leads to the "anomalous" plasma loss. Calculations at PPPL [Rewoldt et al, 1996] have indicated the possibility of reduced or eliminated electrostatic and electromagnetic microinstabilities in high-β ST plasmas due to the increased good curvature region, and raised the possibility, along more recent work [Kotschenreuther, Dorland et al, 1998] of attaining ion neocalssical confinement in such plasmas.

These possibilities have energized broadly based visions for the future development of the ST confinement concept. A small pulsed Performance Extension test of low cost could be envisioned at the 10-MA level to be capable of testing D-T burning plasma physics for limited durations and at moderate fusion power (~80 MW) [Peng et al, 1998], for high Q particularly if neoclassical ion confinement could be approached. A successful outcome at this level would provide the physics database for the steady-state technology-intensive Fusion Energy Development tests at high neutron wall loading ($\geq$ 1 MW/m^2) and neutron fluence ($\geq$ 0.3 MW-a/m^2) per year. The design concept for such a compact Volume Neutron Source (VNS) has been analyzed in U.S. [Peng et al, 1997; Cheng et al, 1998] and in U.K. [Hender et al, 1999a] as a Component Test Facility (CTF). Analysis at GA has suggested that such steps, if already located at a nuclear site, could be upgraded to a Fusion Development Facility [Stambaugh et al, 1997] producing ~5 or more times the fusion power to test the production of electricity under conditions prototypical of a power plant (such as tritium self-sufficiency). The National ARIES Team led by the University of California at San Diego has recently developed an attractive design concept for the ST power plant [Najmabadi et al, 1998] and has identified scientific issues of high leverage for near-term investigation [Jardin et al, 1999; Menard et al, 1999]. Similarly viable concepts have also been advanced by researchers in U.K [Robinson et al, 1998; Hender et al, 1999b]. Noninductive startup of full plasma current to eliminate the inboard solenoid [Peng et al, 1998] and wide dispersion of intense plasma exhaust fluxes [Peng et al, 1995] have been identified also to be crucial in permitting powerful ST fusion devices of small size.

Most of the projected physics properties for such ST plasmas will be tested at the 1-MA level in the new Proof-of-Principle experiments with R_0 ~ 0.8 m: the NSTX in U.S. [Ono, M. et al, 1998] and the MAST in U.K. [Cox et al, 1998], scheduled to begin research during the second half of 1999. New Concept Exploration tests are also scheduled or proposed. These include the Pegasus (University of Wisconsin) [Fonck et al, 1998] to test extreme low A ($\rightarrow$ 1.1) and the scientific connections to the CTs, the Globus-M (Ioffe Institute, R.F.) [Golant et al, 1997] to test innovative Lower-Hybrid Wave heating and current drive, HIT-II [Jarboe et al, 1998] to push the boundaries of CHI, CDX-U [Majeski et al, 1999] to test ST plasma-liquid-Lithium interactions and tilt-stabilized FRC and Field Reversed Mirror (FRM) plasmas by small external toroidal field, as well as several upgraded small ST experiments [Shiraiwa et al, 1999; Narushima et al, 1999] in Japan.

It is important to note that, as can be seen in Table 2-1, the encouraging data are so far limited to very small, pulsed experiments. Extensive experimental verification of these attractive estimates needs to be carried out on higher performance STs in order to have confidence in the projections for these yet larger ST devices mentioned above. However, it is clear that new opportunities for scientific advancement are clearly indicated that will lead along a pathway to ultimately producing fusion energy in the ST configuration. The scientific relationship between these projections and the present understanding of the Tokamak, the AT, and the CT plasmas are summarized in the next section.

3. Scientific Opportunities of ST Plasma

This section focuses on the special scientific opportunities introduced by the ST plasma by addressing the following questions defined by the Plasma Science Working Group of the 1999 Fusion Summer Study [Snowmass WebPage, 1999]:

– How the investigation of the ST plasma properties could address the outstanding scientific issues most critical to the cost-effective development of practical fusion power? What unique scientific features of the ST plasma are particularly suited for such investigation?
– What would facilitate these investigations?
– What new and existing programs and facilities could enable effective resolution of these issues during the next decade?

The discussion is organized in the following four topical areas:

1) MHD Equilibrium and Stability
2) Turbulence and Transport
3) Wave and Particle Interactions
4) Plasma Boundaries and Interfaces

As indicated in Section 2, simultaneous ST physics parameters have been calculated for $\kappa \sim 2$ (NSTX) and $\kappa \sim 3$ (in future ST device concepts). These are summarized in Table 3-1 in comparison with the simultaneous parameters so-far achieved by the Tokamak (A ~ 3), and the simultaneous parameters so-far projected for AT (A ~ 3) and the FRC (A ~ 1.25, $\kappa \sim 2$). It is seen that, in the four scientific topics identified by the Plasma Science Working Group, the ST parameters often bridge the Tokamak and the AT on one side and the FRC on the other. The investigation and understanding of the ST plasma properties can therefore benefit from as well as contribute to these confinement concepts [Morris, 1999]. A more detailed summary of the interesting ST physics properties can be found elsewhere [Peng et al, 1999].

1) MHD Equilibrium and Stability

Stable high β of order unity has been a long-standing goal for MFE research, driven by the critical need for low-cost magnets of simplified configuration. The investigation and understanding of the following ST equilibrium and stability properties are crucial to this goal (see Table 3-1):

– Plasmas with high vertical elongation κ (up to 3 or higher), very low internal inductance ℓ_i,, and broad pressure profiles.
– Reduced plasma disruptivity.
– MHD equilibrium properties in the presence of strong (near Alfvén speed) plasma flows.
– Properties of ideal MHD modes for equilibrium with the aforementioned properties without relying on wall-stabilization (the "no-wall" regime).
– Properties of ideal MHD modes with wall stabilization.
– Properties of the Resistive Wall Modes (RWM).
– Properties of the Neoclassical Tearing Modes (NTM) in the presence of ST specific large Pfirsch-Schlüter currents.
– Effects of strong plasma flows on the above modes [Wahlberg & Bondeson, 1999; Taylor, 1999].

DRAFT *Table 3-1. Simultaneous Parameters of Interest to Science of Tokamak (A~3), AT (A~3), NSTX (κ~2), ST (κ~3), and FRC Plasmas of Similar Plasma Current (~1 MA) and Minor Radius, Based on Best Achievements or Theoretical Projections to Date*

Science Topic	Parameters of Interest	Tokamak (no-wall)	AT (with-wall)	NSTX (no-wall)	NSTX (with-wall)	ST (κ~3+wall)	FRC
MHD Equilibrium and Stability	– Natural elongation κ_N without vertical control	~1.3	~1.3	~2	~2.5	~3	TBD
	– Edge q for high β and ∇p-driven current fraction	~3	~4	~10	~10	~10	~0
	– q_0; ℓ_i (internal inductance)	~1; ~1	~2–3; ~0.6	~2; ~0.4	~3; ~0.2	~4; ~0.1	~0; ~0.1
	– Plasma sound speed v_S in plasma core	$\ll v_A$	$\ll v_A$	$\le v_A$	$\sim v_A$	$\ge v_A$	$\gg v_A$
	– Stable $\langle\beta\rangle/\beta_T$[a]	≤0.1/0.1	≤0.05/0.05	≤0.10/0.25	≤0.25/0.4	≤0.4/0.6	~1/»1[b]
	– Flow speed for stabilizing wall mode (~$0.1v_A$)	~0	$\le v_S$	$\ll v_S$	$\ll v_S$	$\ll v_S$	N/A[b]
	– Max. Pfirsch-Schlüter / plasma current density	≤0.1	≤0.2	≤0.5	≤1	≤1	~0
	– Aligned ∇p-driven current fraction	≤0.2	≤0.8	≤0.4	≤0.7	~1	~1
Turbulence and Transport	– Particle trapping fraction near edge	≤0.6	≤0.6	≤0.9	≤0.9	≤0.9	~0
	– Average normalized ion gyroradius (ρ_i/a)	0.05[c]–0.005	0.05[c]–0.005	0.3[c]–0.03	0.3[c]–0.03	0.3[c]–0.03	0.2–0.06[d]
	– Magnetic well: $(\lvert B\rvert_{edgemin}-\lvert B\rvert_{min}) / \lvert B\rvert_{edgemin}$	~0	~0	≤0.10	≤0.3	≤0.4	≤1
	– ∇p-driven plasma flow shearing rates ($\gamma_{E\times B}$), s^{-1}	$\sim 10^5$	$\sim 2\times10^5$	$\sim 5\times10^5$	$\sim 10^6$	$\sim 2\times10^6$	$\sim 10^7$[e]
Wave and Particle Interactions	– Magnetic helicity/I_p (~$1.6\ell_i\kappa a^2 I_{TF}$) [μHAWb]	~10	~10	~1	~0.5	~0.3	~0
	– Plasma poloidal flux/I_p (~$\mu_0\ell_i R_0$) [μH]	~2	~1	~0.4	~0.2	~0.1	~0.01
	– Plasma dielectric constant $\sim(\omega_{pe}/\omega_{ce})^2$	~1	~1	~40	~60	$\sim 10^2$	$\sim 10^3$–∞[f]
	– Energetic particle (neutral beam, NB) speed	$v_{NB}<v_A$	$v_{NB}<v_A$	$v_{NB}>v_A$	$v_{NB}>v_A$	$v_{NB}>v_A$	$v_{NB}\gg v_A$[f]
	– Core energetic "banana" width / minor radius	~0.1	~0.2	~0.4	~0.5	~0.6	N/A
Inboard-Limited Plasma Boundaries and Interfaces	– Inboard-limited SOL magnetic mirror ratio	≤2	≤2	≤4	≤4	≤5	~0.3[g]
	– Normalized radius of field line curvature (R_c/a)	~4	~4	~2.3	~2.3	~2.4	~2.3
	– Inboard-limited SOL geometric expansion	~2–4	~2–4	≥10	≥10	≥15	→∞
	– Normalized connection length ($L_c/4\pi R_0$)	~3	~4	~2	~2	~1.5	~1
	– Inboard/outboard plasma surface ratio	~1/2	~1/2	~1/4	~1/4	~1/4	0

[a] Average beta $\langle\beta\rangle$ = volume-average pressure over volume-average magnetic field energy density, whereas average toroidal beta β_T = volume-average pressure over applied toroidal magnetic field energy density at plasma major radius. For tokamaks $\langle\beta\rangle \approx \beta_T$, for ST $\langle\beta\rangle \le 0.5\beta_T$ when $\beta_p < 1$ and $\langle\beta\rangle \ge 0.5\beta_T$ when $\beta_p > 1$, whereas for FRC without toroidal field β_T is undefined.

[b] It is presently proposed that there is no MHD stability β limits for FRC, within its high equilibrium limit near unity. In the event that a small toroidal field exists in the FRC plasma, β_T can be much greater than 1.
[c] Deuterium at 80 kV from NBI introduces the upper value; the lower value assumes an ion temperature ~keV's.
[d] In FRC, $\rho_i/a = 2/S^*$, $S^* = r_s/(c/\omega_{pi})$, $S^* \sim 10$–30, where ρ_i is defined by the externally applied (vertical) field of FRC.
[e] The theoretical possibility of sheared flow comparable to the Alfvén speed to change FRC equilibrium and stabilize tilt instability should be examined theoretically. The theoretical possibility for flow shearing to suppress turbulence and reduce transport in FRC should be considered.
[f] The absence of toroidal field in FRC leads to a core region with negligible or zero magnetic field, and hence $v_{NB} \gg v_A$ and $(\omega_p/\omega_{ce})^2 \rightarrow \infty$.
[g] Magnetic field nearly vanishes near the $\times$-point on the axis of symmetry, leading to a reversed magnetic mirror pushing particles toward the $\times$-point. There is no inboard-limited plasma for FRC in the absence of a central conductor.

2) Turbulence and Transport

Good confinement, such as the neoclassical confinement in the absence of transport-enhancing turbulence, has been a long-standing goal for MFE research, driven by the critical need for achieving ignited fusion burn in small-size plasmas. The investigation and understanding of the following ST turbulence and transport properties are crucial to this goal (see Table 3-1):

- Properties and the thickness of the ion thermal transport barrier near the plasma edge in the presence of higher particle trapping fraction, large average magnetic shear, and large (ρ_i/a).
- Properties and possible stabilization of the electrostatic and electromagnetic micro-turbulence due to increased average good curvature.
- Properties and possible suppression of plasma micro-turbulence by very large ∇p-driven and/or externally driven plasma flow shearing rates.
- Properties and thickness of core ion and electron thermal transport barriers in high q_0, very small ℓ_i, and large (ρ_i/a) plasmas.
- Properties and possible reduction in neoclassical ion transport in the ST plasma.

3) Wave and Particle Interactions

Wave and particle interactions are central to the science of plasma heating (including fusion α heating) and noninductive current drive, which are key to MFE plasma current operation and burning plasma physics. Because of the inductive limitations, non-inductive start-up and current sustainment are critical to the concept. The investigation and understanding of the following wave and particle interactions properties are crucial to this success.

- Properties of MHD tearing instabilities and magnetic reconnection associated with the production of large plasma current via CHI.
- Launching and propagation properties of the High Harmonic Fast Wave (HHFW) [Ono, 1996] and the Electron Bernstein Wave (EBW) [Abhay et al, 1998] in ST plasmas, which are characterized by large dielectric constant, magnetic field shear, and steep edge plasma gradients.
- Interactions between the electrons with HHFW, such as via Landau damping which could drive plasma current, and with EBW, such as via stochastic heating near the "accumulation point" where the perpendicular wavelength becomes small. The effects of these mechanisms on plasma heating and current drive processes.
- Interactions between the HHFW and energetic ions with high temperature thermal and NBI fast ions..
- Properties of instabilities driven by supra-Alfvén energetic particles produced by NBI and/or HHFW and fusion α particles [Gorelenkov et al, 1999; Akers et al, 1999].
- Properties of enlarged "banana" orbits of these energetic ions in the ST plasma core of low ℓ_i.

4) Plasma Boundaries and Interfaces

Physical mechanisms of the plasma boundaries and the interfaces with the plasma facing components (PFC) are central to the science of power, particle, and edge control (including fueling), which are critical to the success of future fusion power sources. The ST plasma has the added challenge of intense plasma energy and particle fluxes associated due to its compact nature. The investigation and understanding of the properties of Plasma Boundary and Interfaces of the inboard-limited, Naturally Diverted (ND) ST plasma are crucial to the success in meeting this challenge (see Table 3-1):

- Properties of the MHD, fluid, and kinetic instabilities and fluctuations in the SOL plasma, in the presence of increased magnetic mirror ratio (~4 or more), stronger field line curvature ($R_c \sim 2a$), and large magnetic geometric expansion (≥ 10).
- Properties of the plasma cross-field and parallel transport processes in such a geometry.
- The distribution of power flux between the inboard and outboard SOL, Q_{in}/Q_{out}.
- Effect of SOL fluctuations on transport and the formation of the plasma edge transport barrier.

It should be noted that the ST SOL for the DN and SN divertor configurations are expected to have the magnetic mirror ratio and the geometric expansion factor comparable to those for the Tokamak (see Table 2-1), while having a further reduced connection length compared to the ND configuration of the ST plasma. The knowledge from the Tokamak and the AT SOL research is expected to be applicable to these ST configurations. Further, the Topical Group of Plasma Boundary and Interfaces advances four issue areas:

1) Atomic Physics and Radiation Transport
2) Plasma Neutral Interactions
3) Plasma Material Interactions
4) Turbulence and Instabilities

It is seen that the ST plasma SOL may be different concerning Turbulence and Instabilities, but similar to the Tokamak and the AT in the first three issue areas.

Recommendations

It is seen that a large number of special physics features are identified here for the ST plasma that are critical to the cost-effective development of practical fusion power. There remain many other physics features that are common to the Tokamak, the AT, and the ST. The key physics issues related to advancing the ST concept for MFE will be discussed in Sections 4 and 5. Substantial commonality as well as important differences between the ST and the other configurations have increased our confidence in the ST physics parameters shown in Table 3-1. However, the new ranges of physics parameters could also uncover so-far unexpected plasma mechanisms and behavior, leading to new understanding and driving further innovation in research. *Thorough investigation of the new ST physics features will therefore offer new opportunities for advancing the science of toroidal fusion plasmas in ways so far unavailable to other magnetic confinement concepts. Existing and new ST experiments should therefore be empowered to take full advantage of this new scientific opportunity.*

An optimistic example for possible new advancement can be seen in the MHD equilibrium limit of $\langle\beta\rangle \sim 1$ for the FRC plasma (see Table 3-1), which serves to inspire innovation further to increase the β limit for the ST. A ST plasma with $\langle\beta\rangle \to 1$ could be similar in MHD equilibrium to a FRC-like plasma tilt-stabilized by an externally applied toroidal field while still excluding most toroidal flux from the plasma core. *Concept Exploration experiments, such as TS-3 [Ono, Y. et al, 1998], Pegasus [Fonck et al, 1999], CDX-U [Majeski et al, 1999], and TS-4 [Ono, Y. et al, 1999], to study the transition between the Compact Toroids and the ST plasmas are therefore of particular interest for potential further improvement beyond the presently perceived ST concept.*

For Questions 2) and 3) of the Plasma Science Working Group, *it is recommended that a national ST research effort be coordinated to enhance broad scientific participation in present and future ST experimental user facilities*. This participation can include introduction and utilization of advanced diagnostics, theory, advanced computation, remote collaboration with effective data access, as well as investigations of possible interest to the basic science of magnetized collisionless plasmas. *In the next 5-10 years, the Proof-of-Principle experiment NSTX, and the Concept Exploration experiments Pegasus, HIT-II, and CDX-U in the U.S. should be fully utilized to resolve the new issues of high scientific leverage identified here*. Given positive physics outcome in the next several years, a moderate-cost Performance Extension experiment could be scientifically ready to test the ST fusion plasma science at conditions appropriate for D-T fusion burn (~10 keV in temperature, ~10^{20} m^{-3} in density).

The key issues and opportunities for developing the ST magnetic fusion concept at the Concept Exploration and Proof of Principle levels will be discussed in Section 4; those focused on the Performance Extension, Optimization, and Burning Plasma level will be discussed in Section 5.

4. Concept Exploration and Proof of Principle Research

The Magnetic Fusion Concept Working Group (WG) of the 1999 Fusion Summer Study has suggested that each confinement concept identify

- The key issues and challenges and
- The opportunities for understanding, predicting, and addressing these issues in the next 5–10 years.

The discussion is organized in the following topical areas (not in the original order):

1) MHD Equilibrium and Stability
2) Turbulence and Transport
3) Plasma Boundary and Particle Control
4) Burning Plasma
5) Steady State
6) Concept Integration and Performance Measures

As mentioned previously, important ST issues can derive from the special ST physics features as well as the features that are common with the Tokamak and the AT. Though the issues derived from the common features should also be tested in the near term, it is reasonable to expect that these issues show manageable uncertainty and can rely on methods already used successfully in the Tokamak and the AT. Those issues stemming from the special ST physics features will engender new uncertainty as well as potential, and should therefore receive emphasis. These issues are highlighted by *****.

The key opportunities for addressing these issues in the next 5–10 years are provided by a number of Concept Exploration and the Proof of Principle experiments capable of testing ST plasmas up to keV's in temperature and 10^{20} m^{-3} in density. Special near-term needs in diagnostics and theory to take advantage of the unique ST physics potential are also identified here.

1) MHD Equilibrium and Stability

The research goal in this area is to develop understanding of ideal and non-ideal MHD stability limits, including disruption avoidance and mitigation, in ST plasmas potentially reaching order-unity central β. The ideal MHD stability β limit can be cast into two levels of performance and complexity:

- The "no-wall" regime that has stable plasmas without relying on a nearby conducting wall. Such plasmas are expected to retain substantial flexibility in variations of plasma profiles while providing a ∇p-driven current at a substantial fraction of the total plasma current. Typical projected values for β_T and ∇p-driven current fraction in the ST plasma are ~25% and ~40%, respectively, for a κ~2 and q~10.

- The "wall-stabilized" (or the so-called "second-stability") regime requires a nearby conducting wall for stabilization as well as optimized plasma profiles to provide a nearly fully aligned ∇p-driven current, in a fashion similar to the thesis of the AT plasma [Goldston et al, 1994; Taylor et al, 1997]. For κ~2 and q~10, typical projected values for

β_T and ∇p-driven current fraction are ~40% and ~70%, respectively. For κ~3, typical projected limits for β_T and ∇p-driven current fraction become ~55% and ~99%, respectively, which have been assumed to set the desired MHD limits for future economic Power Plants [Najmabadi et al, 1998; Robinson et al, 1998; Hender et al, 1999].

Success in achieving this research goal will depend on the investigation and resolution of the following special ST as well as the common physics issues:

1a) MHD equilibrium dependencies: The range of achievable MHD equilibria as functions of the edge safety factor q, safety factor q_0 on axis, inductance ℓ_i, current and pressure profiles, plasma β, and ∇p-driven current fraction for varying plasma shaping (elongation κ, triangularity δ, aspect ratio A, etc.).

1b) *Dynamic MHD equilibrium: Effects of high plasma flows on MHD equilibrium and stability.

1c) ***Ideal MHD β limits of the "no-wall" regime**: Robustness of MHD stability to deviations in the plasma pressure and current profiles, and the degree to which this sensitivity differs from that of a conventional R/a tokamak

1d) ***Neoclassical Tearing Modes**: Reduction of island size or full stabilization of the Neoclassical Tearing Modes (NTM) due to the substantial Pfirsch-Schlüter currents (represented by the "Glasser" term) in the ST plasma [Kruger et al, 1998; Hegna et al, 1998].

1e) *Resistive Wall Modes: For plasmas entering the "wall-stabilized" regime, the achievable stabilization of Resistive Wall Mode (RWM) by substantial plasma flow [Garofalo et al, 1998], and/or by active mode control [Navratil et al, 1998]. Avoidance of locked modes is a closely related issue. The data will form a key basis for achieving and maintaining the maximum plasma β theoretically allowed in the "wall-stabilized" regime.

1f) *Edge Localized Modes: Special role of ST geometry (field line pitch, curvature, etc.) and kinetic effects on ELMs.

1g) *Disruption avoidance and mitigation: Reduced disruptivity due to high-q operation and enhanced vertical stability.

The diagnostics needed for investigating MHD Equilibrium and Stability properties of the ST plasma, utilizing the Concept Exploration and Proof of Principle experiments (see, Section 2), are relatively well known in the Tokamak and the AT research [see, diagnostics summary]. Modifications or improvements are proposed or currently under way to deal with the anticipated special conditions of high β, steep gradients, and low magnetic field with strong spatial variations. Examples include the new Motional Stark Effect (MSE) diagnostic systems [Levinton et al, 1998] for core magnetic fields, rf reflectometry [Peebles et al, 1998; Wilgen et al, 1996?] for plasma core and edge density profiles, and the Electron Bernstein Wave (EBW) emission at supercritical densities [Efthimion et al, 1998] for core density and temperature profiles.

New theoretical modeling and analysis will be required to calculate properly the dynamic MHD equilibrium where the plasma flow speed is comparable to the Alfvén speed v_A (noting that for high beta plasmas the Alfvén speed approaches the ion acoustic speed) as well as the MHD instabilities in such an equilibrium. This topic offers opportunities for important theoretical and computational advancement that may be crucial to the successful development of very high β MFE configurations including the ST and the FRC. The effects of the large plasma flow in the dynamic equilibrium on other modes (NTM, RWM, ELM) should also be modeled for comparison with data.

2) Turbulence and Transport

The research goal in this area is to develop understanding of energy and particle transport processes and establish physics-based predictive models to extrapolate to the Performance Extension experiment with confidence. Approaching the neoclassical ion confinement is expected to be a key requirement for approaching high-Q fusion burn in a ST plasma at the 10-MA level.

Significant progress has been achieved in understanding Turbulence and Transport in MFE plasmas, particularly the Tokamak and the AT, during the past 5 years. These advances are directly relevant to identifying and solving physics issues crucial to achieving this research goal for the ST plasma. The selection of the issues below are driven by the special physics features of the ST plasma: large particle trapping fractions, large (ρ_i/a), strong magnetic well, large flow shear, and the large "banana" widths for the energetic ions in the plasma core (see Table 3-1):

2a) *Stabilization of micro-instabilities: Increase in the expected orbit-averaged good curvature and enhanced flow sharing rates in the ST plasma may suppress micro-turbulence. This process should be investigated to establish a physics basis for projection to enable neoclassical-level confinement in the ST Performance Extension experiments.

2b) *Quality of transport barriers: Edge and core transport barrier formation is key to improved confinement in MFE plasmas including the ST. It is necessary to improve the quality of the transport barriers that modifies the pressure profile and improve confinement without reducing MHD stability. The dependence of the quality of the transport barriers on increased (ρ_i/a), q, and I_p, and much decreased ℓ_i in the ST plasma should therefore be investigated. The property of transport barriers to ion energy, electron energy, and particle transport is an important topic of investigation.

2c) *Neoclassical transport model: Measurement and verification of the neoclassical transport in the ST plasma is a key part of the effort to improve confinement.

2d) *Hydrodynamic turbulence: Large plasma flows in order-unity β ST plasmas may introduce a class of hydrodynamic micro-turbulence not accessible to low β configurations, and set new limitations to confinement improvement.

The profile, flow, and fluctuations diagnostics needed for investigating Turbulence and Transport properties of the ST plasma, utilizing the Concept Exploration and Proof of Principle experiments (see, Section 2), are relatively well known in the Tokamak and the AT research [see, diagnostics summary]. Modifications or improvements are proposed or currently under way to deal with the anticipated special conditions of high β, steep gradients, and low magnetic field with strong spatial variations. Examples include the new 2-D rf reflectometry array [Luhmann et al, 1998] potentially capable of providing 3-D images of micro-fluctuations in the plasma core, and continued development of EBW emission diagnostics [Efthimion, P. et al, 1998] for measuring core profiles as well as fluctuations, etc.

New theoretical modeling and analysis will be required to calculate properly the electrostatic and the electromagnetic micro-instabilities in the presence of possible high flow speeds. When the plasma flow speed becomes comparable to the Alfvén speed v_A and the sound speed v_s, the hydrodynamic forces become important in the dynamics of the micro-instabilities. This topic offers opportunities for important new theoretical and computational advancement that may be crucial to the successful development of very high β and high confinement MFE configurations including the ST. Theory and modeling of electron-based micro-instabilities and turbulence, and their possible stabilization and suppression, is of very high importance in this pursuit.

3) Plasma Boundary and Particle Control

The Plasma Boundary and Particle Control Topical Group of the Magnetic Concept WG identified the Limiters, the Poloidal Divertors, the Ergodic Divertor, and the Radiative Boundary (Poloidal Boundary) as the key boundary control methods. These are applicable to the ST plasma, which has an additional requirement of strong Divertor and Vessel Biasing (> kV) for the CHI. The Group also identified the following Plasma Boundary Control goals for discussion:

A. Heat Removal
B. Impurity and Density Control
C. He Pumping
D. First Wall Lifetime
E. Edge/Core Transport Coupling

Achieving these goals in the ST Proof of Principle Experiments will depend on the magnitude of plasma energy and particle fluxes on the PFC, which in turn depend on the plasma exhaust power and the thickness and geometry of the SOL, mitigated by radiation. The exhaust power depends on the plasma energy content (e.g., β limit) and confinement (e.g., possible turbulence suppression). The research outcome in plasma β and confinement will therefore determine the level of challenges we will encounter in achieving these goals.

Far-reaching progress has been achieved in understanding Plasma Boundary and Particle Control in MFE plasmas, particularly in the Tokamak and the AT, during the past 5 years. The techniques, materials, and models already developed so far can be readily applied to the ST plasma investigations. Given plasma β and confinement, the success in achieving these goals will depend on the investigation and resolution of the following issues:

3a) *****SOL plasma instabilities: Study the effect that the unique features of the ST boundary (mirror ratio and curvature, etc.) have on influencing the instabilities that may cause cross field transport in the SOL.

3b) *Cross-field and parallel transport in SOL: This cross field SOL transport, coupled with parallel transport in the SOL, influence the SOL footprint and thus the heat and particle flux upstream.

3c) *Wall and PFC conditioning and treatment: To handle the substantial dc voltage and current, and the expected high power and particle fluxes during CHI-non-inductive and high power auxiliary heating and current drive.

3d) Radiatively cooled boundary and SOL plasma: This technique and its impact on the core plasma should be investigated for the ST plasma, particularly for the DN and SN divertor configurations where relatively intense power and particle fluxes on the divertor plates are expected.

3e) Detached plasma SOL: The effects of differing diverted plasma configurations and power fluxes on the ability to form detached plasmas, and the impact of such plasmas on the core properties should be investigated.

3f) *Integrated modeling of SOL plasma: Great advances in integrated modeling and computation of the tokamak SOL has been achieved during the past 5 years [Porter et al, 1998; Hsu et al, 1999]. These modeling and computation tools includes effects of turbulence-induced cross-field transport, and should be adjusted to include the special features of the ST SOL, and utilized to analyze the SOL data for verification, to establish predicative capability for future more powerful ST experiments.

3g) Interactions of SOL, edge, and core plasma conditions: The properties of the plasma SOL, edge, and core are known to affect the core plasma of the Tokamak and the AT. This relationship in the case of the ST plasma for the differing diverted plasma configurations should be investigated to help identify the most effective means for controlling the plasma boundary and interfaces for improved core performance. Examples include the relationship between the turbulence in the SOL of the ND plasma and turbulence suppression related to the H-mode edge pedestal, and the possibly effects of the magnetic well, which is spatially close to the outboard SOL, on efficiency of fueling via outboard pellet injection and gas puffing.

3h) Tritium retention and removal: PFC and wall materials free of significant retention of tritium should be tested in ST devices to determine compatibility with the above-mentioned requirements, for use in future D-T-fueled ST devices.

The plasma edge and SOL diagnostics needed for investigating the Plasma Boundary properties of the ST plasma, utilizing the Concept Exploration and Proof of Principle experiments (see, Section 2), are relatively well known in the Tokamak, the AT, and the Stellarator research [see, diagnostics summary]. Examples include the edge and divertor laser Thomson scattering measurements of electron temperature and density profiles. Modifications or improvements are proposed or currently under way to deal with the special conditions of large geometric expansion, low magnetic field, and large mirror ratios. Examples include the new 2-D Ultra-Soft-X-Ray tomography [Stutman et al, 1998] to measure small-scale fast fluctuations in the plasma edge and SOL, sophisticated Fast Reciprocating Probes [Boedo et al, 1998] to measure profiles, and testing special materials suitable for survivable and "clean" CHI electrode [Ulrickson et al, 1999], ultra-fast infrared camera to detect hot spots on the PFC, B-dot probes to measure the detail of the CHI current drive mechanisms, special instrumented material coupons on the PFC to record the plasma-material interaction, etc.

4) Burning Plasma

The capability of a ST device to test burning plasma physics depends on success enhancing MHD stability, plasma confinement and power and particle (and impurity) control. The issues relating to plasma current drive are included in 5. Steady State below. The burning plasma physics issues associated with the fusion α-particles for a ST plasma at ~10 MA in current will be discussed in Section 5. Here we focus on the research issues directly relevant to burning plasma physics that can be tested and benchmarked in the Proof of Principle experiments at ~1 MA using NBI heating at ~80 kV and HHFW. The 80 keV ions in NSTX simulate the α-particles in the performance extension ST in terms of banana width, Larmor radius, and V/V_A.

The ST Proof of Principle experiments plan to investigate the following issues of direct relevance to burning plasma physics:

4a) Energetic ion orbit confinement: In the presence of large magnetic wells and low TF.
4b) ***High-frequency Alfvén modes and Associated Particle Loss**: Study the broad spectrum of TAE modes excited by super-Alfvenic energetic ions. The fast ion loss is expected to be mitigated by the presence of a magnetic well and large $B_{\theta a}$.
4c) Beam ion anisotropy: The anisotropic distribution of beam ions in the velocity space may lead to phenomena not available for isotropic distributions such as of the fusion α. The effect of perpendicular heating of beam ions by HHFW on the dynamics of energetic particle heating should therefore be investigated to develop an understanding of the anisotropy effects.
4d) ***Heating profile compatibility with high β and turbulence suppression**: High performance ST plasmas can be maintained for several energy confinement times if the energetic ion heating profiles can lead to a pressure profile that is compatible with high β and suppressed turbulence.

Issues that can only be studied in a burning plasma experiment include the effects of heating source being proportional to $\sim n^2T^2$ of the plasma, which is not directly controlled externally, and the need to control burn through control of n, T, and the transport coefficients. These will be discussed in the next section.

The diagnostics needed for investigating the Burning Plasma relevant physics, in addition to those already discussed, include the energetic ion distribution measurements via neutral particle analyzers and fusion products probe, TAE and other high frequency fluctuations measurements via magnetic pickup and reflectometry, and the wall locations of the fast ion losses.

5) Steady State

The Steady State Topical Group raised the following questions:

– What are the key physics and technology issues for achieving and sustaining a steady-state operation for the ST plasma? What role do self-generated currents play in the ST concept and can they be controlled in steady state?
– What are the advantage and disadvantages of steady-state operation compared repetitively pulsed operation of the ST plasma?
– What are the relevant time scales for important physical phenomena for the ST plasma and how do existing facility pulse lengths relate to these time scales?

– What mechanisms are responsible for limiting pulse length? Which of the key steady-state physics and technology issues can be tested on existing devices? Which issues require a new long-pulse facility?

Steady state issues for the driven, burning ST plasma in a Fusion Energy Development device such as a Volume Neutron Source (VNS) will be discussed in Section 6. Here we focus on the physics of noninductive startup and steady state operation that can be tested in Concept Exploration and Proof of Principle experiments. The full potential of the ST plasma depends on success in achieving noninductive startup and steady state operation.

5a) *Noninductive startup: Present and future ST devices must rely on noninductive methods for initiating and ramping up the plasma current to the full level because of the V-sec limitations of the skinny center post. HIT-II and NSTX plan to test and develop the CHI technique for this purpose. NSTX will also explroe other rf methods such as HHFW and ECH/EBW. NSTX is designed to test noninductive startup over a pulse length up to 5 s.

5b) *Noninductive current maintenance: Because of the high dielectric constant in STs, HHWF is uniquely suited to heat electrons, drive current and provide for current profile control. This technique has been tested in CDX-U and will be explored in NSTX and Pegasus. Another technique suited for high β plasmas is the Rotating Magnetic Field, which may be tested in CDX-U, and will be tested for the FRC. CHI as a method for maintaining edge current will also be investigated. NSTX is designed to test this condition for a pulse length of 5 s, if the proper current profile can be formed rapidly such as via ohmic induction, HHFW, NBIm and/or CHI. A pulse length of 10 s may be incorporated into the upgraded center stack, because of the anticipated increase in plasma temperature.

5c) *Nearly fully aligned ∇p-driven current: This condition can in principle be achieved in the high-β "wall-stabilized" regime with optimized pressure and current profiles regime. This investigation is of crucial importance to the feasibility of the ST Pilot Plant, DEMO, and Power Plant. A pulse length ~10 s may be incorporated into the upgraded center stack, because of the anticipated increase in plasma temperature and the magnetic flux diffusion time.

Special diagnostics will be needed for CHI and HHFW experimentation. These are similar to those identified for Plasma Boundary and Particle Control.

6) Concept Integration and Performance Measures

This topical group asks what understanding and capability must be developed to establish the basis for a useful ST fusion energy system? The group further suggests a number of key issues to address relating to the development of the ST concept. The responses to these questions are arranged as follows:

– This subsection will address the questions appropriate for the level of Proof of Principle:
  - the highest priority new physics developments in physics integration that can be tested in the next few years.

- Section 5 will address the questions appropriate for the level of Performance Extension and Burning Plasma Physics for limited pulse lengths:
  - Burning plasma physics for long pulse compatible with high-β stability in the "no-wall" regime and ITER H-mode level confinement.
  - burning plasma physics for limited pulse compatible with high-β stability in the "no-wall" regime and strong ITB to approach neoclassical confinement conditions;
  - issues critical to motivation and justification of readiness to be resolved by the Proof of Principle experiments;
  - requirements in plasma technology; and
  - opportunities to minimize the cost of this step, etc.
- Section 6 will address the questions appropriate for the level of Fusion Energy Development:
  - steady-state burning plasma compatible with high-β stability, nearly neoclassical confinement, partial noninductive current drive, and high-intensity plasma boundary conditions;
  - issues critical to motivation and justification of readiness to be resolved by the Performance Extension experiment;
  - small-size ST based VNS to minimize development cost;
  - requirements in fusion technology, material, and environmental safety;
  - requirements for a nuclear-grade fusion site; and
  - appropriate U.S. role in an international VNS facility, etc.
- Section 7 will address the questions appropriate for the level of Pilot Plant, DEMO, and future Power Plant:
  - optimized steady state burning plasma with mutually compatible very high-β stability with wall-stabilization, good confinement, nearly full noninductive current drive, and high intensity plasma boundary conditions;
  - issues critical to motivation and justification of readiness to be resolved by the Fusion Energy Development tests;
  - small-size ST based Pilot Plant to minimize cost for demonstrating tritium self-sufficiency and electricity production;
  - requirements for a utility energy site; and
  - appropriate U.S. role in an international Pilot Plant facility or larger, etc.
- Sections 8, 9, and 10 will address the questions relating to International collaboration, summarize the ST roadmap and development metrics; and highlight non-electric applications.

The highest priority ST physics developments in physics integration at the Proof of Principle stage, for making progress toward the Performance Extension stage and beyond, can be cast into the following advancing levels of time scale and complexity:

6a) <u>*The compatibility between MHD stability at high β_T in the "no-wall" regime and H-mode confinement for several energy confinement times</u>. The key physics elements include:

- The formation of plasma current and q profiles at low β that is approximately consistent with high β following auxiliary heating.
- Heating of electrons to “freeze” the q profile, such as by HHFW and NBI.

– Achievement of H-mode plasma (~ITER98H) with the pressure profile compatible with MHD stability at high β via HHFW and NBI heating.
– The maintenance of high β in the "no-wall" regime via control in heating power and profile, taking advantage of the significant tolerance to plasma current and pressure profiles variations without violating MHD stability.
– Noninductive current drive (via HHFW, NBI, EBW, CHI) coupled with substantial ∇p-driven current fraction to maintain current magnitude for several τ_E's.
– Limited-pulse operation of plasma boundary and particle control.

This integrated plasma condition is envisioned as likely appropriate for testing the achievement of low-Q (~1) fusion burn in a ST Performance Extension (PE) experiment with ~10 MA plasma current. Plasma divertor configurations (DN, SN, and ND) should be tested for best compatibility with this condition.

6b) *The compatibility between the above condition with near-neoclassical ion confinement resulting from substantial suppression of micro-turbulence, for several energy confinement times. Additional key physics elements are:
– The formation of ion transport barrier and growth of the barrier thickness where steep pressure gradient is compatible with MHD stability at high β, such as by time and power-controlled HHFW and NBI.
– Adjustments in q profile (i.e., ℓ_i), density profile, and I_p to obtain consistency among the heating and fueling profiles, the neoclassical transport coefficients, and the pressure profile which tends to be broad for stability. The sizes of the thermal and energetic ion gyroradii and "banana" widths relative to the plasma minor radius in the plasma core are expected to be important variables in achieving this condition.
– Inductive capability limited to current flat-top maintenance for several τ_E's.

This integrated plasma condition is envisioned as likely appropriate for testing high-Q (~10) fusion burn in a ST Performance Extension (PE) experiment with ~10 MA plasma current. Plasma divertor configurations (DN, SN, and ND) should be tested for best compatibility with this condition.

Internal electron transport barrier has been observed in several tokamaks [Hatae et al, 1998] usually co-located with the ion transport barrier in the plasma core. This would lead to more centrally peaked pressure profiles, and a modest reduction in the "no-wall" MHD stability β limit [Paoletti et al, 1999] in comparison with the broad pressure profile commonly calculated so far. However, the reduction in β is possibly more than compensated by the large improvement in confinement beyond the ion-only neoclassical regime. Such a condition is expected to be equally important as a basis for high-Q experiment at ~10 MA plasma current.

6c) *The compatibility between MHD stability in the "no-wall" regime with H-mode confinement and steady-state current drive, for more than the magnetic flux diffusion time. The key physics elements of this issue include the above-mentioned conditions, as well as:
– The production of substantial ∇p-driven current fraction (such as ~40% in the case of NSTX operating at $\kappa \sim 2$ and $q_{egde} \sim 10$).

– Noninductive current drive of the remaining plasma current via NBI, HHFW, CHI, and possibly EBW in a compatible current profile.
– Steady state operation of plasma boundary and particle control.

This integrated plasma condition is envisioned as likely appropriate as the physics basis for achieving driven low-Q fusion burn in a ST Fusion Energy Development (FED) experiment such as the VNS with ~10 MA plasma current. Plasma divertor configurations (DN, SN, and ND) should be tested for best compatibility with this condition.

6d) *The compatibility between MHD stability at very high β in the "wall-stabilized" regime with near-neoclassical ion and electron confinement, steady-state nearly self-sustained plasma current (>70%), for more than the magnetic flux diffusion time. The new physics elements include:

– Stabilization of the Resistive Wall Modes and Edge Localized Modes.
– The production and maintenance of large ∇p-driven current fraction (such as ~70% in the case of NSTX operating at $\kappa \sim 2$ and $q_{egde} \sim 10$).
– Noninductive current drive of the remaining plasma current via NBI, HHFW, CHI, and possibly EBW in a compatible current profile.
– Long-pulse operation of plasma boundary and particle control.

This integrated plasma condition is envisioned as likely appropriate as a physics basis approaching that needed by the future Pilot Plant, DEMO, and Power Plant. Plasma divertor configurations (DN, SN, and ND) should be tested for best compatibility with this condition.

6e) *The compatibility between MHD stability at very high β_T (~55% in the case of NSTX with an upgraded center stack to permit $\kappa \sim 3$ and relying on wall-stabilization), near-neoclassical ion and electron confinement resulting, steady-state fully self-sustained plasma current (~90%), for more than the magnetic flux diffusion time.

This integrated plasma condition is envisioned as likely appropriate as a physics basis for the future Pilot Plant (PP), DEMO, and Power Plant (PP) based on the ST concept.

The table below depicts this progression of the ST concept integration at the Proof of Principle level:

Level of ST Physics Integration	**6a**	**6b**	**6c**	**6d**	**6e**
MHD stability β_T limit (%)	25	20–25	25	40	55
Stabilization	no-wall	no-wall	no-wall	with-wall	with-wall
Elongation κ	2	2	2	2	3
Neoclassical confinement	H-mode	Ion (+elec)	H-mode	ion	ion
∇p-driven current fraction (%)	40	small	40	70	90
Noninductive driven current (%)	60	small	60	30	10
Inductive current maintenance (%)	0	high	0	0	0
Plasma flat-top duration	$5\tau_E$	$5\tau_E$	$>\tau_{skin}$	$>\tau_{skin}$	$>\tau_{skin}$
Relevant to	PE	BP	FED	PP-PP	PP-PP

Recommendations

By accounting for the broad research topics of the Tokamak and the AT as well as the anticipated special ST physics parameters, a comprehensive ST research program at the CE and PoP levels are defined here. The juxtaposition of the common as well as special ST physics issues makes evident the unique opportunity introduced by the ST research in magnetic fusion energy science:

The ST Proof-of-Principle research is ready to benefit from the full spectrum of fusion expertise and advance into attractive new physics territories in direct support of the first two policy goals of the U.S. Fusion Energy Sciences Program.

To take advantage of this opportunity, it is recommended that:

1) *Steps be taken to enable national and international expert participation of the NSTX research program in all the key physics issues identified here.*
2) *The FY-2000 NSTX Research Forum, likely to be scheduled for December 1999, be expanded to invite the national and international Tokamak and ST researchers to discuss and recommend the key research to address these issues.*
3) *This ST Research Forum emphasizes research issues addressing ST physics integration for neoclassical confinement and high* β *in the "no-wall" regime for several energy confinement times.*

5. Performance Extension (PE), Optimization, and Burning Plasma Research

The Burning Plasma Physics Subgroup of the Magnetic Fusion Concepts WG raised a number of issues for the ST concept to address: the physics issues unique to burning plasmas and the capability of a candidate ST experiment to address these issues; the goals for this experiment; required physics performance; expected research contribution; and approximate cost and schedule. Physics elements of importance to the burning plasmas in confinement, MHD, energetic particles, and edge physics and power dispersal, and particle control are to be addressed. Also to be addressed particularly for the ST plasma is the physics database needed to support the start of engineering design.

This section will also address the issues raised by the Concept Integration and Performance Measures Topical Group of the Magnetic Concept WG for the PE experiment, identified in Section 4:

- burning plasma physics for limited pulse compatible with high-β stability in the "no-wall" regime and improved confinement toward the neoclassical transport;
- issues critical to motivation and justification of readiness to be resolved by the Proof of Principle experiments;
- requirements in plasma technology; and
- opportunities to minimize the cost of this step, etc.

Here we use the example of a small-size DTST (Deuterium-Tritium-capable ST) device ($R_0 \leq 1.2$ m) designed to achieve ~10 MA in plasma current (see Appendix 2-1 and Table 5-1 below), which is at present a candidate for a PE level experiment [Peng et al, 1998]. The preceding discussion on the ST fusion plasma science (see Section 3) and magnetic fusion concept physics (see Section 4) have identified the issues and the potential for this PE experiment to achieve low Q (~1) burn with the ITER H-mode-level confinement, as well as the high-Q (~10) burn with extensive internal transport barrier. Variations to this example are readily available, with R_0 ranging from ~ 0.8 m [Kotschenreuther et al, 1998] for identifying the prospect for ignition, to ~1.1 m in the case of a nearly self-sustained VNS, which approaches Power-Plant-like conditions (see Appendix 2-1). The discussion here applies equally well to these examples.

Included in Table 5.1 are the representative parameters projected for the PoP experiments NSTX and MAST, the DTST, and the Fusion Energy Development (FED) device concept VNS (see Appendix 2-1). Two levels of operation are indicated. The lower level is defined by the "no-wall" regime, which has potential to achieve low-Q (~1) current driven operation and the ITER H-mode confinement, as well as high-Q (~10) current driven operation if strong ITB can be achieved to approach neoclassical ion confinement. The higher level is defined by "wall-stabilized" regime with potential not only to achieve high Q (~10 or higher) but also to achieve self-sustained plasma current for steady-state operation. These two levels of physics performance are good examples to define the range of issues discussed here, and correspond to the ST Concept Integration Issues 6a and 6b defined in Section 4 for the PoP experiments. Also included for DTST is the steady state test in D-D of the VNS driven mode in S-T. This corresponds to the ST Concept Integration Issue 6c.

The design concept for DTST is briefed in Appendix 2-1, which includes a thin solenoid providing ~1 Wb flux capability in single-swing, and a multi-turn demountable jointed center stack similar to the NSTX design.

Table 5-1. Representative parameters for the PoP experiments NSTX and MAST in "no-wall" and "wall-stabilized" regimes; the PE device DTST in D-D long-pulse, low-Q pulsed, and high-Q pulsed Burning-Plasma modes; and the FED device in the Q~1 driven VNS mode and the high-Q sustained Power-Plant (PP) mode

Near-Term ST Devices	NSTX/MAST		DTST			VNS	PP-like
Development Stage	*Proof of Principle*		*Performance Extension*			*Fusion Energy Development*	
Mode of operation	*No-Wall*	*Wall-Stabilized*	*D-D H~1*	*Q~1 H~1*	*Q~10 H~3*	*Q~1 Driven*	*Q>10 Sustained*
Major radius (m)	0.85		~1.2			~1.1	
Aspect ratio	≥1.25		1.4			1.4	
Toroidal field (T) at major radius	0.3–0.6		0.9	1.7	1.7	2.1	
Plasma current (MA)	1–2		~5	~10	~10	~10	
Edge safety factor	10–5		~10			~10	
Plasma cross section elongation	2–2.5		3			3	
Normal beta β_N (%Tm/MA)	5	8	3.5	3.5	5	4	8
Average toroidal beta β_T (%)	25	40	23	23	34	27	50
Average density (10^{20} m^{-3})	0.5	0.5	0.5	1	0.7	1.1	1.2
Average temperature (keV)	1	1.6	3	7	15	9	16
∇p-driven current fraction (%)	40	70	50	50	75	50	90
Plasma drive power (MW)	6–11		25	35	4	40	25
NBI energy (keV)	70–80		110			110	400
Fusion power (MW)	–		–	36	40	66	260
H (ITERH98H) [Kardaun, 1998]	1–2		1	1.2	3	1	3
Plasma flattop (or burn) time (s)	1–5		~100	~10	~10	~1000	
Neutron wall load (MW/m^2)	–		0.6			1.0	4.0
Neutron fluence/year (MW/m^2)	–		~0.003			~0.3	~1.2

Goals of DTST

Two levels of goals are of interest: PE and Burning Plasma Physics. The PE program aims to explore the ST physics at or near the parameters appropriate for D-T fusion burn to generate sufficient confidence that the parameters needed by the FED device can be achieved and a affordable development path can be attempted. Since the ST FED plasma is anticipated to require only Q~1, so should the PE program aim to test, albeit for limited pulse length in D-T or for long pulse without fusion power in D-D. This dual testing capability is needed in this case to establish the required physics basis for decision to invest in the technology-intensive FED device.

The Burning Plasma Physics program aims to investigate and demonstrate the ST fusion burn plasma physics at high-Q burn (~10 or greater), a condition that can lead to an economic ST Power Plant. The finite pulse length for the PE device at full fusion performance limits the operation to several energy confinement times; integration of MHD, transport, startup, heating, and edge control physics sustained only over this time scale would be required. Integration of these with self-sustained current profiles beyond the magnetic flux diffusion time scale will be required for the FED device, operated in a Power-Plant-like mode, if supported by the outcome of physics tests in DTST (see Table 5-1 and Section 6).

Database Needed to Support Start of PE Device Engineering Design

To support the design of the PE experiment for driven Q~1 tests, the key scientific issues for which database and understanding are needed from the Proof of Principle tests (NSTX, MAST) are identified below according to Table 5-1 and the discussions is Section 4. The database aims to enable reliable predictions for the PE plasma at ~10 MA level.

1) Noninductive startup and ramp-up of large plasma current (such as to ≥0.5 MA) at high edge q (~10) (see Issue 5a, Section 4) using a combination of CHI, HHFW, NBI, and EBW (such as up to 11-MW total power in 5 s time scale).
2) Plasma heating to the "no-wall" β_T limit (such as ~25%) using a combination of NBI, HHFW, and EBW (see Issue 1d, Section 4) with the current profiles consistent with low ℓ_i (such as ~ 0.2–0.3).
3) Confinement depencencies in ρ* and ν* (in line with such as the ITERH98 scaling [Kardaun, 1998]), with pressure profile consistent with the "no-wall" stability limit. The ratios of ρ* and ν* between the PoP and the PE plasmas are ~1.5 and ~10, respectively, assuming the parameters given in Table 5-1.
4) Plasma properties affecting maintenance of the current and pressure profiles required for good stability and confinement for several energy confinement times (such as ~1–2 s), and subsequently for a longer time scale up to 5 s, via combination of substantial ∇p-driven current fraction (such as ~40%) and noninductive current drive (such as ~60%) using CHI, NBI, HHFW, and EBW.
5) Properties of NTM and ELMs to identify plasma conditions for their avoidance (Issues 1e, 1f, 1g, Section 4).
6) Properties of NBI ion orbit confinement (with heating by HHFW) and the beam ion driven TAE and "fishbone" modes to identify plasma heating and current drive conditions for minimized impact. The NBI ion gyroradius and "banana" width as fractions of minor radius in the PoP experiments are similar to those for the fusion α particles in the 10-MA ST (Issues 4e, 6a, Section 4).
7) Properties of the boundary and SOL plasmas and techniques of power, particle, and impurity control in various divertor configurations (ND, DN, and SN) to identify effective approaches for the PE plasma operation with likely high power density and longer pulse. The ND configuration may lead to low heat fluxes (~1 MW/m^2) at the inboard limiter and the top and bottom divertor plates (Table 3-1, Section 3; Issues 3a, 3b, and 5d, Section 4), possibly at some expense to core confinement and β limit. The DN and SN configurations may lead to high heat fluxes (~10 MW/m^2) at the outboard divertor plates, possibly mitigated by gains in confinement and β limit.

To support the design of the Burning Plasma Physics experiment at high Q (~10), additional scientific issues for which database and understanding are needed from the Proof of Principle tests (NSTX, MAST) are identified below:

8) Plasma properties related to micro-turbulence suppression and internal transport barrier formation for confinement times beyond the ITER98H scaling toward the neoclassical ion confinement (Issues 2a, 2b, 2c, 2d, 2f, Section 4), *as well as* the properties of neoclassical ion transport in such ST plasmas. An equivalent H-factor relative to ITER98H of ~2 would reduce the total heating power to around ~6 MW while the plasma remains within the β limits of the "no-wall" regime. Neoclassical ion confinement with anomalous electron confinement (ITER98H) in NSTX would scale to H~3 in the 10-MA ST for high Q capability.
9) Plasma properties affecting the maintenance of the current and pressure profiles required for the "no-wall" stable β and confinement H up to ~2 for several energy confinement times (such as ~2–3 s), sustained by solenoid induction and combination of CHI, NBI, HHFW, and EBW. The effects of relatively large "banana" width of the NBI ions in the plasma core on the maintenance of the high-β and high-confinement profiles are of high interest, for direct application to fusion α heating at the next stage.
10) In this case, the divertor heat fluxes are reduced from the "driven" VNS-mode described previously, and opens up the opportunity for DN and SN operation in NSTX. This in turn would increase the probability for further improved confinement, a condition of high interest to high-Q capability at the next stage.

A summary of the confinement time projections from START to NSTX and to the 10-MA ST is provided in Figure 5-1 below. It is seen that the in general the tokamak scaling expressions are not expected to apply to the ST plasma, though use of the ITER98H scaling expression is consistent with the data from START. Further, the neoclassical transport model in ST plasmas is an important issue to be investigated at the Proof of Principle level (see Sections 3 and 4 discussion on Turbulence and Transport). It is therefore of high importance to quantify the approach toward neoclassical ion transport with sufficiency accuracy for reliable application to the 10-MA ST.

<u>Required DTST Physics Performance and Expected Research Contribution to the FED Stage</u>

The physics performance to be accomplished by the DTST can be defined by the database needs of the FED device. Assuming successful physics outcome of the PoP experiments, these larger devices are expected to have similar plasma sizes and currents (see Table 5-1), and hence similar plasma physics parameters. The DTST answers all fusion plasma physics issues so that the FED device can enter into technology-intensive R&D with confidence.

To verify the physics for a driven VNS at low Q (~1), the DTST must achieve:

1) Noninductive startup and ramp-up of plasma to high currents (such as ~10 MA) at high edge q ~10 using a combination of CHI, HHFW, NBI, and EBW for up to 40-MW total power. At present a total time of ~40 s, which is comparable to the plasma skin time with several keV in temperature, is assumed for this noninductive current ramp-up. The total device operation time at full B_T and drive power, including the noninductively maintained plasma flat-top of ~10 s, is assumed to be ~50 s.

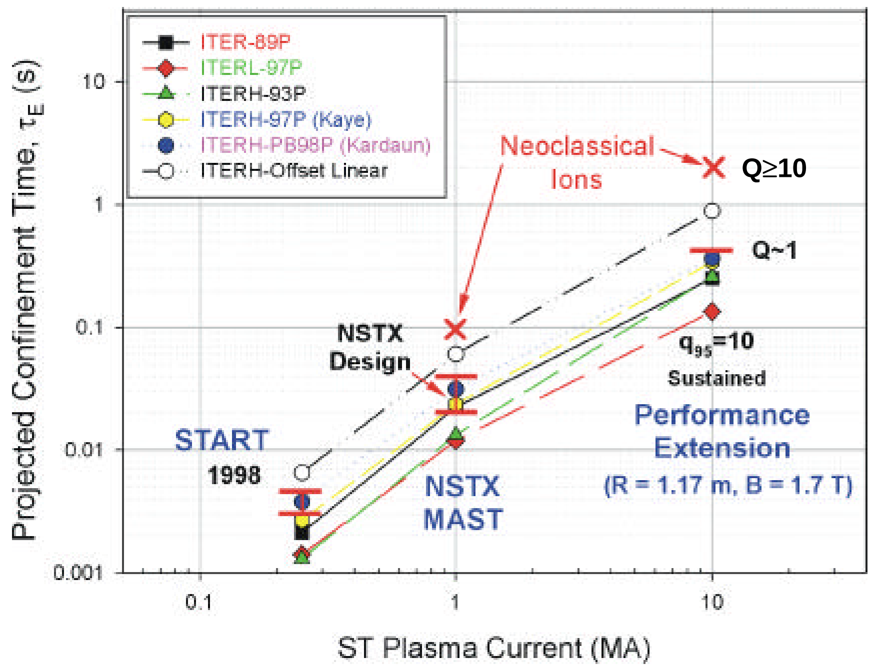

Figure 5-1. Projections of energy confinement times for NSTX and a ST plasma at 10 MA for Performance Extension testing according to Tokamak scaling expressions and separately assuming neoclassical ions. START data is included for comparison.

2) Heating of plasma to somewhat below the "no-wall" limit β_T ~23% using a combination of NBI, HHFW, and EBW with hollow current profile and low ℓ_i ~ 0.2–0.3.
3) Achieving plasma confinement according to the ITERH98 scaling with an H-factor of ~1.2, and a pressure profile consistent with the "no-wall" stability limit.
4) Maintenance of the current and pressure profiles required for good stability and confinement in D-T at full plasma current for several energy confinement times (up to ~10 s), via combination of substantial ∇p-driven current (~50%) and externally driven current (~50%) using CHI, NBI, HHFW, and EBW, while producing ~40 MW fusion power at Q~1.
5) Maintenance of the current and pressure profiles required for good stability and confinement in D-D at half plasma current (~5 MA) and half field (~0.9 T) for more than the plasma skin time (up to ~100 s), via combination of substantial ∇p-driven current (~50%) and externally driven current (~50%) using CHI, NBI, HHFW, and EBW.
6) Avoidance of substantial impact of the NTM and ELMs.
7) Adequate confinement of fusion α particles and tolerable levels of the TAE and "fishbone" modes.
8) Adequate plasma boundary, SOL, power and particle control using various divertor configurations. The ND configuration would lead to low heat fluxes (~2 MW/m^2) at the inboard limiter and the top and bottom divertor plates. The DN and SN configurations would lead to high heat fluxes (~20 MW/m^2) at the outboard divertor plates.

To test Burning Plasma Physics at high Q (~10), the DTST must further achieve:

9) An increase in β_T to 34% and the substantial ∇p-driven current fraction to about 75% in the "no-wall" regime for $\kappa = 3$ plasma.
10) A substantial approach to the neoclassical ion confinement via suppression of micro-turbulence. The equivalent H-factor for ITERH98 scaling would be about 3, reducing the auxiliary heating power to ~4 MW while producing ~40 MW fusion power.
11) Inductive maintenance of the full plasma current for several energy confinement times (~10 s) by swinging from zero to full solenoid current, while maintaining consistency between the dominant α heating profile – accounting for the relatively large "banana" orbits, the neoclassical ion transport, and the pressure profile.

Similarly to the PoP experiments, much reduced power and particle fluxes are obtained in this case from the driven operation at 10 MA because the total heating power is reduced from ~48 MW to ~12 MW. This opens up the opportunity for DN and SN operation, which in turn would increase the probability for improved confinement. The ability to handle power and particle fluxes at the divertor and limiter plates in Performance Extension experiments, either via the ND configuration [Peng et al, 1997] or via improved confinement to reduce the total heating power, will provide crucial database for the steady-state operation required in a Fusion Energy Development device such as the VNS.

Scale of Cost and Schedule, Plasma Technology, and Cost Savings

Cost and schedule information for the national PE facility should be estimated by carrying out a design scoping study. In the absence of such, a scale of the cost for the 10-MA PE can be surmised only very crudely from a comparison with the NSTX cost data. The major NSTX cost items include the magnets (~$6M for NSTX including supporting structure), the vessel and the plasma facing components (~$7M for NSTX including supporting structure), the auxiliary heating and current drive systems (~$9M for HHFW, NBI, and some EBW), and the facility modifications (~$8M for NSTX). Shielding "igloo" for personnel safety and the full diagnostics systems at the TFTR level must be added for the larger device.

The total "volume" of the 10-MA PE device increases to a factor of roughly $f_V \sim (R_{PE}/R_{PoP})^3 \times (\kappa_{PE}/\kappa_{PoP}) \sim (1.2/0.85)^3 \times (3/2) \sim 4.2$. The magnetic field stored energy scales as $f_M \sim f_V \times (B_{PE}/B_{PoP})^2 \sim 4.2 \times (1.7/0.3)^2 \sim 130$. The cost of the magnet systems scale roughly $C_M \sim f_M^{0.5} \sim 11$, leading to a scaled cost of ~$70M.

Total device vessel surface area scales as $f_S \sim (R_{PE}/R_{PoP})^2 \times (\kappa_{PE}/\kappa_{PoP}) \sim (1.2/0.85)^2 \times (3/2) \sim 3$. For steady state (~50 s or more) operation with active room-temperature heat removal, the plasma facing components is assumed to be ~ 3 time more costly per area. The cost for the vessel and plasma facing component scales roughly as $C_S \sim 3f_S \sim 9$, leading to a scaled cost of ~ $60M. Substantial improvements in plasma facing component and fueling technologies beyond the level of present-day PE tokamaks will be needed in concert with the PE program.

The auxiliary heating and current drive systems on NSTX are refurbished from existing equipment. A TFTR NBI system is being refurbished for pulsed operation at ~$1.2/W and 80-kV for NSTX. The cost for upgrading the TFTR NBI system to ~50 s or more (nearly steady state) operation at ~110 kV is estimated to require ~$2.5/W [Grisham, 1995]. For a total NBI power of 32 MW in four beam systems, a scaled cost of ~$80M is obtained. An existing ICRF systems is refurbished for pulsed operation at ~$0.5/W in NSTX. Upgrading to 50-s (nearly steady state) operation at 8 MW is estimated to require ~$1.5/W [Wilson, 1999], leading to a scaled cost of ~$12M. Four-MW commercially off-the-shelf EBW systems at ~50 GHz are assumed to cost ~$8M. The total heating and current drive systems cost therefore scales to ~$100M, which includes collaboration with the U.S. plasma technology experts.

The balance of facility modifications will encompass the entirety of the power supply and control systems, the heating and cooling systems, the tritium management systems, the environmental systems, etc. The NSTX-driven modification is limited to about 20% of the total system. This scales the facility modifications cost to ~$40M, assuming that the present tritium systems are reused.

A basic set of modern diagnostics systems at the level of TFTR is assumed to be ~$20M, assuming collaboration with the U.S. diagnostics expertise.

Shielding (such as the “igloo” on TFTR) for personnel protection is needed to allow access to the test cell immediately following high-power D-D or any significant D-T operation. This shielding needs to cover the vessel as well as components with major access such as the NBI systems (a dominating requirement). Since the “igloo” shielding on TFTR costed about $30M, a shielding cost of about ~$20M is assumed for the present device, which has a significantly smaller vessel and assumes the same number of NBI systems.

Pending an adequate design scoping study, the scaled cost of a ST facility utilizing the PPPL facility for Performance Extension experimentation is ~$310M, scaled from the NSTX experience. The replacement value of the existing equipment and facility is estimated to be over $300M in 1993 dollars [TPX design, 1992]. Broad collaboration with the U.S. expertise in fusion plasma science and technology will ensure cost-effective application of the resources.

International partners for the national DTST program should be sought to benefit from the broad scientific expertise worldwide and possibly accrue substantial cost savings. Siting DTST in the U.S. will ensure U.S. a leadership role in the ST fusion energy science research.

Following completion of the conceptual design that produces detailed cost and schedule information (requiring ~10% of the Total Project Cost) and approval of the construction project, the national PE project can be assumed to take 3.5 year to complete if adequate funding is provided.

Recommendations

To guide the NSTX Research Program in the near term, it is recommended that a *nationally based physics scoping study be carried out to quantify the range of the required ST physics performance at 10-MA current for PE as well as Burning Plasma Physics research.* This will enable in the next 4-5 years the U.S. fusion program to take advantage of the new opportunities offered by ST in the consideration of the Next Step Options.

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