SCIENCE AND TECHNOLOGY OF THE 10-MA SPHERICAL TORI*

MARTIN PENG
Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA
(on assignment at Princeton Plasma Physics Laboratory)

WAYNE REIERSEN, STAN KAYE, STEVE JARDIN, JON MENARD, DAVE GATES,
JOHN ROBINSON, FRED DAHLGREN, LARRY GRISHAM, DICK MAJESKI,
DAVE MIKKELSEN, MASAYUKI ONO, JOHN SCHMIDT, RANDY WILSON,
ROBERT WOOLLEY
Princeton Plasma Physics Laboratory, Princeton, NJ 08543, USA

EDWARD CHENG, RALPH CERBONE
TSI Research, Inc., 225 Stevens Avenue, Suite #203, Solana Beach, CA 92075, USA

DENNIS STRICKLER, JOHN GALAMBOS
Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA

IGOR SVIATOSLAVSKY
University of Wisconsin, Madison, WI 53706, USA

KERCHUNG SHAING
University of Texas at Austin, Austin, TX, USA

XUEREN WANG
University of California, San Diego, La Jolla, CA 920-93-0417, USA

Abstract

The scientific parameters and the technology issues for a modest-size Spherical Torus (ST) at 10 MA in plasma current are discussed. This class of devices include a D-T-capable ST experiments (DTST, R_0 = 1.2 m) for Proof of Performance for limited pulse lengths and neutron fluences, and a steady-state volume neutron source (VNS, R_0 = 1.1 m) for testing Fusion Energy Components to high neutron fluences. The scientific issues of interest to the DTST include noninductive ramp up of plasma current in a limited time scale (~40 s), confinement needed for high-Q burn, behavior of energetic particles, physics and techniques to handle intense plasma exhaust, and the possibility of high performance plasma regimes free of disruptions or large disruption impact. Also of interest to the VNS would be steady state operation using large external current drive possibly at a modest Q (~ 1–2) achieving significant neutron wall loading (~1 MW/m^2) and a configuration relatively amenable for remote maintenance. A much longer time scale would be permitted for noninductive current ramp up. The center leg of the TF coils, possibly multi-turn for DTST and necessarily single-turn for VNS without significant nuclear shielding, is a technical and material issue of unique importance to the ST. Positive-ion Neutral Beam Injection (NBI) and HHFW (~80 MHz) heating and current drive systems already available to date are likely adequate for the DTST following pulse length extension to ~50 s. For the high densities needed for enhancing the neutron wall loading (to ~ a few MW/m^2) in a VNS, a negative-ion NBI system may become desired. Given an adequate physics database, the remaining enabling technologies needed by the VNS appear largely similar in nature to those of the ITER EDA design.

1. INTRODUCTION

The Spherical Torus (ST) configuration has recently emerged as an example of confinement concept innovation that could enable attractive steps [1] in the development of fusion energy. The scientific potential for the ST has been indicated by recent encouraging results from START [2], CDX-U [3], HIT-II [4], etc. The scientific principles for the D-fueled ST will soon be tested by NSTX (National Spherical Torus Experiment [5]) in the U.S. and MAST (Mega-Amp Spherical Tokamak [6]) in the U.K. at the level of 1–2 MA in plasma current.

More recently, interest has grown in the U.S. in the possibility of near-term ST fusion burn devices at the level of 10 MA in plasma current. The missions for these devices would be to test burning plasma performance in a D-T-fueled ST (i.e., DTST) and to develop fusion energy technologies in a steady state ST-based Volume Neutron Source (VNS [7]), both of modest size. This paper reports the results of analysis of the key science and technology issues for these devices.

2. REPRESENTATIVE PARAMETERS

The parameters for the 10-MA ST devices have been estimated using a ST version of the SUPERCODE [8]. The results are given below, in comparison with NSTX and MAST. The two columns for each concept indicate major anticipated modes of operation.

Near-Term ST Devices	**NSTX/MAST**		**DTST**		**VNS**	
Mission: ***To test or develop***	*Physics Principle*		*Physics Performance*		*Energy Technology*	
Mode of operation	*First Regime*	*Adv. Regime*	*Sustained Current*	*Transient Current*	*First Regime*	*Adv. Regime*
Major radius (m)	~0.80		~1.2		~1.1	
Aspect ratio	≥1.25		1.4		1.4	
Toroidal field (T) at major radius	0.3–0.6		1.7		2.1	
Plasma current (MA)	1–2		10	18	~10	
Edge safety factor	10–5		10	5	9	
Plasma cross section elongation	2–2.5		3		3	
Normal beta (%Tm/MA)	5	8	7	3.3	4	7.6
Average toroidal beta (%)	25	45	50	40	25	45
Bootstrap current fraction (%)	50	80	80	25	50	90
Plasma drive power (MW)	6–11		40	10	40	70
NBI energy (keV)	70–80		110		110	400
Fusion power (MW)	–		80	200	66	260
H (ITERH-PB98y,1) [9]	1–2		2.5	1.2	1	2
Plasma flattop (burn) time (s)	5–1		~20	~10	~1000	
Neutron wall load (MW/m^2)	–		1.1	2.6	1.0	4.0
Neutron fluence/year (MW/m^2)	–		~0.003		~0.3	~1.2

We find that the D-T-fueled ST plasmas at the 10-MA level would be characterized by modest major radius (~1.2 m) and moderate toroidal field (~1.7 T). This device size would allow only a modest central solenoid (~1-Wb capability) to provide a target plasma of ~ 1 MA in plasma current for subsequent ramp up to full current via noninductive techniques. As a result the time needed for this ramp up, tentatively assumed to be ~30 s, would largely determine the total device pulse length (~50 s). The DTST would require an NBI energy up to 110 kV, a flattop burn time ~20 s, and a minimal neutron fluence per year (~0.003 MW-yr/m^2) to prove fusion plasma performance at the level of Q~2 and fusion power ~80 MW. A high toroidal beta (~50%) and a significant enhancement of confinement (H~2.5) relative to ITERH-PB98 [9] would be required for this purpose. However, this case would have a nearly noninductively sustained plasma current for the duration of burn (~20 s), suggesting that the plasma would be approaching the "advanced physics" regime.

For DTST operation at ~18 MA, a transient burn for ~10 s, which is long compared to the energy confinement time but short compared to the resistive decay of the plasma current, would be assumed. However, a longer time scale for ramp up to full current should be anticipated, suggesting a significant increase in pulse length and current drive capabilities from the nearly sustained mode of operation at 10 MA. Our estimate suggests that Q~10 could be obtained with a modest confinement enhancement (H ~ 1.2), as is already measured in START [2]. Because of the lowered safety factor and bootstrap current, the plasma would be in the First Stability regime at 40% average toroidal beta.

For the technology-intensive, steady state VNS, the initial operation could rely on the burning plasma data from the DTST and provide ~1.0 MW/m^2 in neutron wall loading, ~ 70 MW in fusion power, and ~0.3 MW-yr/m^2 in neutron fluence per year. NBI energies of ~110 kV could be adequate for accessing the First Stability regime characterized by modest beta (~25%) and moderate bootstrap current fraction (~50%). A relatively conservative confinement enhancement of H~1 is assumed for this case. Assuming the success of the "advanced physics" regime (β_N~ 8, H ~ 2), the performance for the VNS plasma could be improved by about a factor of 4 in fusion power, if negative-ion NBI system could be utilized to access the high density (about twice the previous case).

III. KEY SCIENTIFIC PARAMETERS OF INTEREST

Key scientific principles to be tested by NSTX/MAST for the DTST are identified. These include

1) Noninductive current formation and ramp-up to eliminate the solenoid, via CHI and RF-only techniques [10], taking advantage of bootstrap current overdrive [11];
2) Plasma heating and current drive via HHFW [12] and NBI [13] for steady-state operation;
3) High plasma beta [14] with well-aligned bootstrap current to permit high fusion power density and ease current drive;
4) Confinement in the presence of transport barriers [15] and improved neoclassical ion transport [16]; and
5) Limited power and particle flux densities at the limiters and divertors via SOL with large mirror ratio and flux expansion [17].

These characteristics are expected to expand the scientific domain of toroidal plasma parameters and may dominate aspects of the physics of ST plasmas, as summarized in the table below:

Physics	**Parameters of Interest**	**10-MA ST**
Confinement	- Inverse aspect ratio	0.7-0.8
	- Edge safety factor	~10
	- Magnetic well depth	~0.3
	- Average normalized ion gyroradius (ρ_i*)	~0.015
	- Diamagnetic flow shear rate (ω_D*)	~10^6
Stability	- Stable average toroidal beta (β_t)	~0.4
	- Ideal wall stabilization ($\beta_{\text{N-with wall}}/\beta_{\text{N-without wall}}$)	~2
	- Flow speed for stabilizing wall mode (~0.1 v_A)	» c_s
	- Neoclassical tearing mode ($\beta_{\text{NCT}}/\beta_{\text{N-with wall}}$)	~1
H-Mode	- Edge T_i pedestal (large edge heat flux, ρ_i, magnetic shear)	large
	- L-H transition power threshold ($\propto$ nB at edge)	low
	- H-mode for inner-wall limited operation	likely
Edge-Divertor:	(Outboard SOL for inboard limited "natural divertor" plasma)	
	- SOL inboard-to-outboard magnetic mirror ratio	~4
	- SOL flux tube expansion	≥10
	- Connection length compared to double-null plasma	~2 times
Noninductive Ramp up	- Plasma helicity per I_p (~1.6 $\ell_i \kappa a^2 I_{TF}$, H-Wb)	9
	- Plasma poloidal flux per I_p (~$\mu_0 \ell_i$ R, H)	0.4×10^{-6}
	- MHD stable bootstrap current overdrive at high q	yes
Current Maintenance	- NBI and α: guiding center orbit containment	~100 kV D^+
	- Energetic particle driven instabilities	$v_\alpha \gg v_{NB} \geq v_A$
	- RF: plasma dielectric constant ~$(\omega_p/\omega_{ce})^2$	~10
	- Alignment of stable pressure-driven current for q ≥ 10	~100 %

Although the ST plasmas are similar to the standard aspect ratio tokamaks in many aspects, the range of physics parameters indicated above can lead to interesting new or different plasma behaviors of importance. NSTX, MAST, and other upcoming ST experiments will shed light on the significance of these differences. Possible examples are discussed below.

Key issues of fusion plasma performance to be tested in the DTST for the ST-based VNS should stem primarily from the presence of significant heating by the fusion alpha-particles for operations at Q~2. In these ST plasmas the Alfvén speed is expected to be significantly below the energetic alpha and NBI ion speeds in the outboard region, leading to a new regime for possible Alfven mode instabilities. For high safety factors q~10 at edge and ≥2 at the axis, an increased vulnerability to orbit losses enhanced by magnetic ripples is expected. Orbit compression due to strong magnetic well [14] and sheared flow [15] may reduce such orbit losses. The interaction of the energetic alpha particles with the HHFW heating and current drive is expected to be an important new issue of interest. Issues relating to dominating alpha heating would become important if Q~10 or higher could be reached, assuming strong transport barrier [18] in the "advanced physics" regime.

Our analysis also indicates that the enabling technologies in plasma heating, current drive, fueling, plasma-surface interaction, and power and particle removal required by DTST and VNS (at ~1 MW/m^2 in neutron wall loading) are generally available at present in fusion research. These include NBI at energies up to 120 keV, ICRF at frequencies up to 80 MHz, and dispersed heat fluxes (≤10 MW/m^2) for the inboard-limited plasmas with naturally diverted outboard SOL. Energy technology issues unique to the compact VNS stem primarily from the single-turn water-cooled copper center leg [19] of the toroidal field coil. This center leg is expected to endure intense neutron bombardment, radiation hardening, significant activation [20], and entail special safety issues of copper disposal and related to possible liquid-metal coolant of the test blankets. At ~1 MW/m^2 in neutron wall loading, designs for fusion core components of the ITER EDA could be adopted.

The database for the "advanced regime" physics will be tested in NSTX/MAST and could be tested in DTST; that for the energy technologies at high neutron wall loads (~4 MW/m^2) could be developed in VNS, assuming the "advanced regime" physics. Successful outcome of these efforts could eventually justify the economic viability for small Pilot Plants [21] and attractive Power Plants [22] in the future.

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