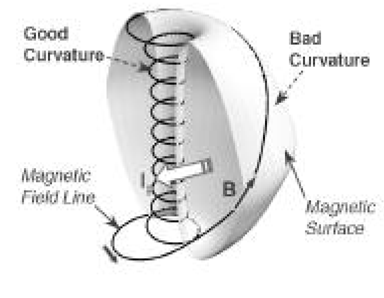
M-6. SPHERICAL TORUS

Description



ST plasma magnetic configuration with A ~ 1.25, elongation κ = 2, and edge safety factor q = 12.

- The aspect ratio ($A=R_0/a$) of the ST plasma approaches unity (1.1–1.6 typically) compared to A=2.5–5.0 so far for Tokamak and Advanced Tokamak (AT). As a result the ST plasma resembles a sphere with a modest hole through its center, uses modest applied toroidal field (TF), and has large I_p/aB and I_p/I_{tfc} values.
- Its magnetic surfaces combine short field line of bad curvature and high pitch angle (relative to the horizontal plane) 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 line curvature is magnetohydrodynamic (MHD) stability at high plasma pressure, giving the potential for order-unity average toroidal and central plasma betas ($\beta=2\mu_0 p/B^2$). High β and the magnetic configuration combine to widen the parameter domain for magnetic fusion plasmas, the investigation of which will strengthen the scientific basis for attractive energy (MFE) and other applications.
- The wider domain promises high-performance fusion plasmas possessing large trapped particle fraction (up to 90% near edge), Pfirsch-Schlüter current (~I_p), and toward the outboard, a magnetic well (~30%) with nearly omnigenous particle trajectories, dielectric constant ($\sim\omega_{pe}^2/\omega_{ce}^2 \gg 1$), normalized gyroradius ($\rho^* = \rho/a \sim 0.03$–$0.01$), supra-Alfvén fast ions ($v_{fast} > v_A$), gradient-driven flow shearing rate ($\sim 10^6 s^{-1}$), magnetic mirror ratio (~4) and flux tube expansion (≥10) in naturally diverted (ND) outboard scrape-off layer (SOL) of inboard limited plasmas.
- The ST configuration also requires certain features in engineering and technology to maximize the potential ST benefits. These include single-turn demountable, normal-conducting toroidal field coil (TFC) center leg and vertical replacement and assembly of fusion core components in toroidally symmetric sections, anticipated to simplify remote maintenance.

Status

- The concept exploration experiments, such as the Small Tight Aspect Ratio Tokamak (START) in the United Kingdom, the Helicity Injected Tokamak (HIT) at the University of Washington, and the Current Drive Experiment-Upgrade (CDX-U) at Princeton Plasma Physics Laboratory (PPPL) have recently produced very encouraging though preliminary results despite modest sizes ($R_0 \leq$ 35 cm). These include average toroidal betas up to 40%, central local betas ~100%, H-mode confinement, a large range in operating density ($\leq 2\times10^{20} m^{-3}$) and q (1–30), a small and nearly symmetric halo current (~5% of I_p) during plasma disruptions, and helicity injection startup of I_p (up to 200 kA).
- Several new or upgraded ST experiments in United States will begin experimentation in 1999 or sooner. These include the National Spherical Torus Experiment (NSTX) at PPPL with $R_0 \sim 0.8$ m, $I_p \sim 1$–2 MA; and smaller experiments Pegasus (University of Wisconsin), HIT-II (University of Washington), CDX-U (PPPL), to explore ST physics boundaries in extreme low A, noninductive startup, and radio frequency (rf).
- NSTX is a national Proof of Principle research facility at PPPL. Construction is expected to be completed in April 1999.

Current Research and Development (R&D)

R&D Goals and Challenges

- The NSTX R&D goals are to investigate noninductive startup and maintenance of plasma current to eliminate the central induction solenoid; efficient plasma heating and current drive; turbulence suppression to improve confinement; stability at high toroidal average β (up to 45%) with large well-aligned pressure-gradient-driven (thermoelectric) current (up to 80–90% of I_p); and dispersion of plasma exhausts over large wall and tile areas. The investigation is to span the wide scientific domain identified above and will challenge the breadth of expertise of the fusion community.
- The NSTX with other ST experiments aim to establish in 4-5 years database for a Performance Extension ST device, which in turn aims to produce database for an Energy Development ST device (see Metrics).

Related R&D Activities

- Foreign ST experiments, such as the Mega-Amp Spherical Tokamak (MAST) in the United Kingdom; Globus-M (Russian Federation), TS-3&4, TST-M, HIST (Japan), and ETE (Brazil) are complementary to the domestic ST experiments in goals and capabilities (e.g. MAST has poloidal field coils inside a large chamber and no stabilizing shell near the plasma).
- Advanced Computational Simulation is needed to account for the strongly toroidal ST geometry and physics features.
- Enabling Technologies is of particular importance to ST due to the anticipated high power densities in its small size.
- Power plant conceptual studies identify innovative approaches for near-term R&D to maximize the ST advantage.

Recent Successes

- ST experimental results are very encouraging; several new ST experiments are approaching completion of (see Status).
- Under recent Small Business Innovative Research (SBIR) funding, feasible design concepts for ST-based volume neutron source (VNS) and ST-driven transmutation power plants to burn fission actinides, requiring only ST plasmas of modest-Q in the first-stability regime.
- Feasible ST reactor plasma operation scenarios, assuming the advanced physics regime and simplified maintenance concepts, were developed by the Advanced Reactor Innovation and Evaluation Studies (ARIES) group, United States, and the United Kingdom Atomic Energy Authority (UKAEA) Culham Fusion, United Kingdom.

Budget

- For NSTX in FY 99, PPPL receives \$5.5M to complete the Project, \$3.5M to prepare NBI, \$0.9M for laser scattering system, \$7.9M to operate the national facility; scientists from PPPL and 13 collaborating institutions receive \$5.4M and \$2.8M, respectively, to begin research. Collaborators' research funding for FY 00 will hopefully grow toward \$5M.

Anticipated Contributions Relative to Metrics

Metrics

R&D targets and opportunities estimated for the ST-based VNS (R_0~1.1 m) and power plant (R_0~3 m):

Energy Metrics	VNS	Power	Physics Metrics	VNS	Power
TF center leg material	DS-Cu[a]	DS-Cu[a]	Full noninductive I_p ramp up	Yes	Yes
Q-engineering	~0	~4–6	Thermoelectric current fraction	~0.5	~0.9
Accessibility	Ample[b]	Ample[b]	Max β limit (%) / required β (%)	50/25	75/60
Maintenance approach	Eased[c]	Eased[c]	Required H_f (ITER98-H)	~1	~2
Environmental and safety	*d*	*d*	Plasma power / wall area (MW/m^2)	~1	~1.6
Development path			Power & particle handling	ND	DN[h]
Modularity	*e*	*e*	Plasma / applied magnetic energy	~0.1	~0.1
Commonality of physics	*f*	*f*	B at R_0 / B at TF coil	~0.25	~0.25
Unit capital cost ($B)	~1	~3	Uncontrolled shutdown freq.	→0	→0
Cost of electricity (mills/kW-hr)	–	~70	Neutron wall loading (MW/m^2)	~1	~5
Near term non-electric applications	*g*	–			

[a] Dispersion strengthened Copper, which is estimated to be adequate for the VNS and could be improved for reactor.

[b] The modest TF permits ample space between the TF return legs for access to the fusion plasma and core.

[c] Demountable center leg for the TF coil and compact fusion core enable vertical access to axisymmetric core components.

[d] Environment and safety issues, such as due to the juxtaposition of actively cooled normal, current-carrying conductors and high performance power handling and conversion components, need to be investigated and resolved in ST VNS.

[e] ST offers a modular low-cost path beyond NSTX: DTST for Performance Extension (R_0~1.1 m), VNS for Energy Development (R_0~1.1 m, Q_{eng}=0), Pilot Plant (R_0~1.4 m, Q_{eng}~1), and power demonstration plant (R_0~3 m, Q_{eng}~4–6).

[f] The ST benefits from and contributes to MFE R&D by enlarging its scientific domain (see Description). The ECW propagation and conversion through ST plasma outboard resembles rf wave experimentation of the earth ionosphere.

[g] Success of ST VNS could enable viable nearer-term non-electricity applications such as source for neutron science, transmutation of nuclear waste, and production of tritium and isotopes for fusion, industrial, and medical uses. The fusion physics and technology metrics for such applications are significantly less than those required for fusion power.

[h] Double-Null (DN) divertors may be needed to ensure advanced physics regime operation assumed in a ST power plant.

Near Term ~ 5 years

- NSTX and MAST research programs plan to verify the physics metrics for the first-stability regime for DTST and VNS, which do not require active profile and mode control. The programs further plans to clarify the physics metrics for the advanced physics regime for power plant, which requires detailed control and optimization of ST plasma properties.
- Design studies for the DTST and the ST VNS can be carried out, incorporating technologies already available or developed for the International Thermonuclear Experimental Reactor (ITER) Engineering Design Activity (EDA). Existing U. S. R&D facilities can be used to reduce costs for DTST and VNS.
- To improve Q-engineering for ST reactors, compact toroid (CT)-like approaches should be explored to minimize the applied toroidal field via edge current drive (e.g. CHI and Rotamak current drive) while maintaining the desirable ST plasma properties.

Mid Term ~ 20 years

- A DTST would verify the physics metrics for VNS and reactor in fusion relevant regimes at very high average plasma pressures ($\propto\beta B^2 \geq 100\% T^2$). Early verification of the advanced physics regime would enable integrated testing of ignited burn and energy technologies, which is the mission of an engineering test reactor (ETR), in a single VNS-size device.
- A ST VNS would be built to extend DTST results to steady state and begin R&D towards establishing the energy metrics, including TFC conductors to resist neutron damage and activation, and fusion blanket and core components.
- Equally important would be progress in theory and computation, fusion technologies and materials, environmental and safety techniques, and advanced systems design to ensure readiness to embark on an ETR.

Long Term > 20 years

- A ST ETR (or Pilot Plant) would be built to demonstrate all metrics for fusion power, including scientific and technological feasibility, energy technology qualification, and safety and environmental soundness.
- The ST configuration would offer for DEMO a range of unit powers, from the small Pilot Plant (P_{DT}~300 MW, Q_{eng}~1) for zero net electricity, to the full-size power plant (P_{DT}~3000 MW, Q_{eng}~4–6) for 1000 MW in net electricity.

Proponents' and Critics' Claims

- Proponents claim that the ST will provide an innovative, low-cost development path to an attractive fusion power source, with relatively simple technology and device maintenance. Disruptions may be less of a problem in the ST relative to the tokamak due to a) lower magnetic field, b) the absence of a delicate zero-shear point in the plasma, and c) high q operation. Neoclassical tearing modes may be stabilized by Pfirsch-Schluter effects, and only very modest rotation is required for shear-flow stabilization of turbulence and for MHD wall-mode stabilization.
- Critics claim that the modest engineering Q of a GWe ST, largely due to the need to drive current in the copper center column, will push it to larger unit size. This may not be attractive in the energy market of the future. The ST could also suffer from disruptions, and may have nearly the same size and complexity as a tokamak.