National Spherical Torus Experiment (NSTX) and Planned Research

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The U.S. fusion energy sciences program began in 1996 to increase emphasis on confinement concept innovation. The NSTX [1,2] is being built at PPPL as a national fusion science research facility in response to this emphasis.

NSTX is to test fusion science principles of the Spherical Torus (ST) plasmas, which include:

1) High plasma pressure in low magnetic field for high fusion power density,
2) Good energy confinement is a small-size plasma,
3) Nearly fully self-driven (bootstrap) plasma current,
4) Dispersed heat and particle fluxes, and
5) Plasma startup without complicated inboard solenoid magnet.

These properties of the ST plasma, if verified, would lead to possible future fusion devices of high fusion performance, small size, feasible power handling, and improved economy.

The design of NSTX is depicted in Fig.1. The device is designed to study plasmas with major radius up to 85 cm, minor radius up to 68 cm, elongation up to 2, with flexibility in forming double-null, single-null, and inboard limited plasmas. The nominal operation calls for a toroidal field of 0.3 T for 5 s at the major radius, and a plasma current at 1 MA with q ~ 10 at edge. It features a compact center stack containing the inner legs of the toroidal field coils, a full size solenoid capable of delivering 0.6 Wb induction, inboard vacuum vessel, and composite carbon tiles. The center stack can be replaced without disturbing the main device, diagnostics, and auxiliary systems. The vessel will be covered fully with graphite tiles and can be baked to 350^{o}C. Other wall conditioning techniques are also planned.

The NSTX facility extensively utilizes the equipment at PPPL and other research institutions in collaboration. These include 6-MW High Harmonic Fast Wave (HHFW) power at ~30 MHz for 5 s, which will be the primary heating and current drive system following the first plasma planned for April 1999, and small ECH systems to assist breakdown for initiation. A plethora of diagnostics from TFTR and collaborators are planned. A NBI system from TFTR capable of delivering 5 MW at 80 keV for 5 s, and more powerful ECH systems are also planned for installation in 2000.

The baseline plan for diagnostics systems are laid out in Fig. 2 and include:

1) Rogowski coils to measure total plasma and halo currents
2) Flux loops and Mirnov coils to measure poloidal flux and fluctuations
3) Visible TV camera to observe plasma shape evolution
4) IR camera to observe heat loading on tiles
5) Multi-channel bolomer to measure radiated power profile
6) 170-MHz microwave interferometer to measure line-integrated density
7) Survey spectrometer (SPRED) to measure plasma impurities
8) Soft X-ray imaging system to measure plasma instabilities and fluctuations
9) H_α line detectors to measure edge hydrogen recycling
10) CHERS and high-throughput CHERS background array to measure ion temperature and toroidal rotation
11) X-ray pulse height analysis to measure core electron temperature
12) Neutral particle analyzer to measure core ion temperature and fast ions
13) Visible spectrometer to measure edge and divertor density and temperature
14) Ultra-soft X-ray array to measure startup and impurity behavior
15) Langmuir probes and thermocouples to measure divertor plasma parameters

The NSTX research plan is envisioned, for purpose of preparation, to have three phases: Startup and Ohmic Heating; "First Stability Regime;" and "Advanced Physics Regime." Figure 3 depicts the approximate time scale, research experimental run-weeks, major research tools, and goals.

The first stability regime is expected to be characterized by average toroidal betas up to 25% for q ~ 10, significant bootstrap current fraction ~ 50%, and confinement at par with those indicated by the tokamak L-mode scaling expressions (such as the ITER power law 89P). Detailed control of plasma profiles is not anticipated to be necessary in this regime.

The advanced physics regimes is expected to be characterized by average toroidal betas up to 40-50% for q ~ 10-15, nearly full bootstrap current fraction ~ 80-90% well aligned with the total current profile, and confinement approaching the level of neoclassical ions. Detailed control of all plasma, heating and current drive profiles is anticipated to be necessary to reach this regime.

The investigation is expected to contribute much to the understanding of ST plasmas in particular and toroidal plasmas in general. The considerations include:

- Noninductive plasma formation via Coaxial Helicity Injection (CHI) and ECH, taking advantage of the small inductance and helicity content in the ST plasma. Results from HIT-II and HIST, presented at this conference, have been encouraging.
- Heating and current drive via ECH, HHFW, and NBI, taking advantage of the strong local magnetic shear, very high dielectric constant, supra-Alfven energetic and thermal ions, the potential for large fully aligned bootstrap current, and the small volume of the ST plasma. Results from START and CDX-U, presented at this conference, have been encouraging.
- Ultra high stability and operation limits, taking advantage of naturally large stable elongations (up to 4 observed), strongly stabilizing magnetic field line structure, modest Alfven speed compared to sound speed, large B/R, minimal external inductance, and inboard rigidity of poloidal flux. Results of 40% average toroidal beta from START, presented at this conference, have been encouraging.
- Reduced microturbulences and enhanced transport barrier, taking advantage of precession reversal at high beta, very large sheared flow at high beta and momentum input, and compressed neoclassical orbits due to large pressure gradient and sheared flow. Tokamak results from DIII-D, JET, and JT60-U, etc., presented at this conference, have been encouraging.
- Edge and SOL, divertor and limiter, taking advantage of the large magnetic mirror ratio, outboard field line curvature, and flux expansion in the inboard-limited, naturally diverted SOL.
- Integrated and sustained operation, taking advantage of the above ST plasma features and the stable monotonic q profiles in the presence of *very* hollow current profiles.

Confirmation of the first stability regime physics would enable viable steady state volume neutron source (VNS) [3] designs that are modest in size and fusion power while delivering high neutron wall loading. However, the advanced physics regime would be required for the ST plasma to introduce a viable possibility for future economic fusion power sources.

An U.S. national research team involving numerous fusion research institutions is being formed to begin research on NSTX in May 1999.

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References

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[2] Martin Peng, "Spherical Torus Pathway to Fusion Power," to appear in May issue of J. Fusion Energy.
[3] E. T. Cheng et al., "Study of a Spherical Torus based Volumetric Neutron Source," to be published in Fusion Engineering and Design, 1997.

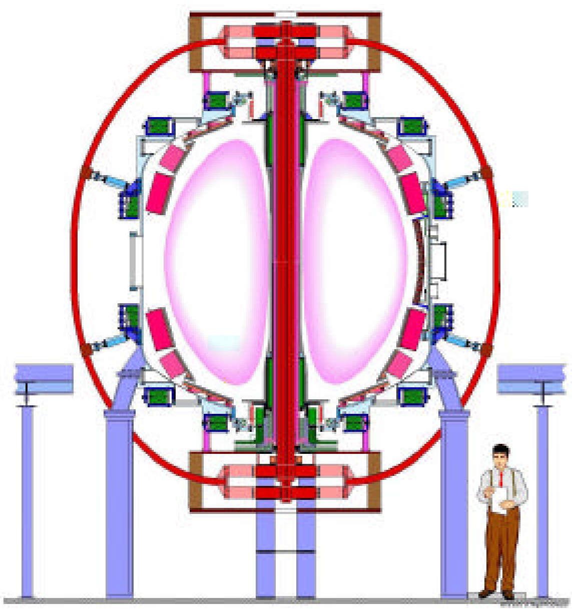

Figure 1. Elevation view of NSTX

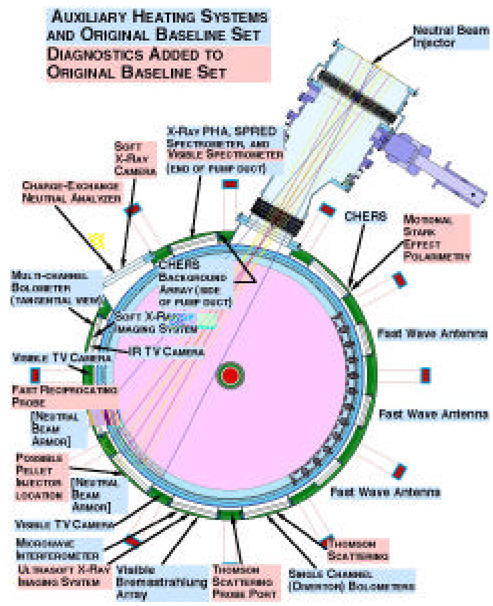

Figure 2. NSTX baseline diagnostics plan

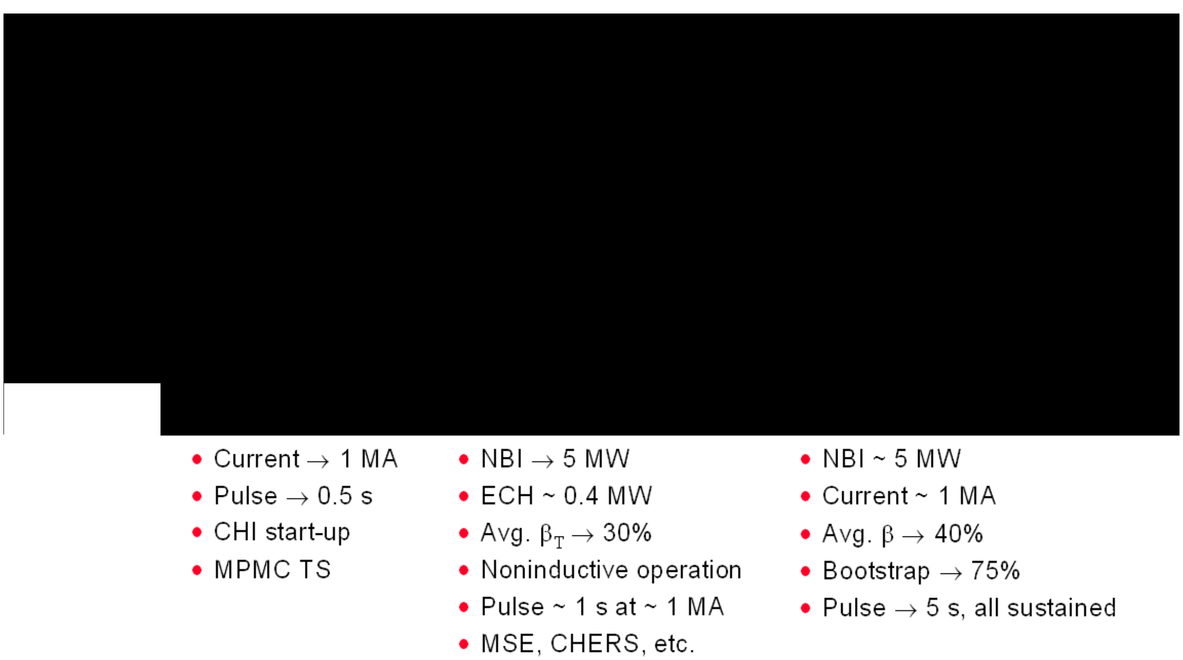

Figure 3. An envisioned NSTX research plan