Spherical Torus Pathway to Fusion Power

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Abstract

Spherical Torus (ST) as an example of confinement concept innovation to enable a potentially attractive pathway to fusion power is discussed. Given the anticipated high performance in small size, the ST plasma could be used to stimulate innovation also in engineering, technology, and material combinations to provide a smarter, cheaper, faster pathway. This pathway could complement the mainline program based on the tokamak in making the desired progress in fusion energy sciences. The ST pathway could include a small VNS (Volume Neutron Source) with low fusion amplification (Q ~ 1–2) for Fusion Energy Development (*energy technology*) and a small Pilot Plant with high Q (~ 15–30) to practice Fusion Power Demonstration. Success in these steps also enhances the possibility for competitive non-electric applications of interest to society in time scales shorter than electric power generation. The scientific basis for these possibilities will be tested in the U.S. by the Proof of Principle experiment NSTX (National Spherical Torus Experiment) presently being built, and could be completed by a Proof of Performance and Optimization experiment such as a small DTST (Deuterium-Tritium Spherical Torus). Utilization of facilities and equipment already available in the U.S. would minimize the time and cost for these experiments and accelerate the approach to the stage of Fusion Energy Development.

I. Introduction

A recent review[1] by FESAC (Fusion Energy Science Advisory Committee) has led to a growing emphasis on concept innovation in the U.S. Fusion Energy Sciences Program (Fig. 1). Concept innovation is rooted in fusion and plasma science and aims to deliver confinement concepts of increased performance, lowered cost, and reduced time for development. Success in this direction will enable improved pathways to fusion power as well as possible competitive non-electric applications earlier than the electric power application of fusion energy.

In this paper we discuss the elements of a potentially attractive pathway utilizing the ST concept[2]. The example of ST serves to encourage continued innovation in fusion plasma science as well as in nearer-term, non-electric applications of fusion. Success in these endeavors should in turn enhance progress toward more innovative concepts and future fusion power.

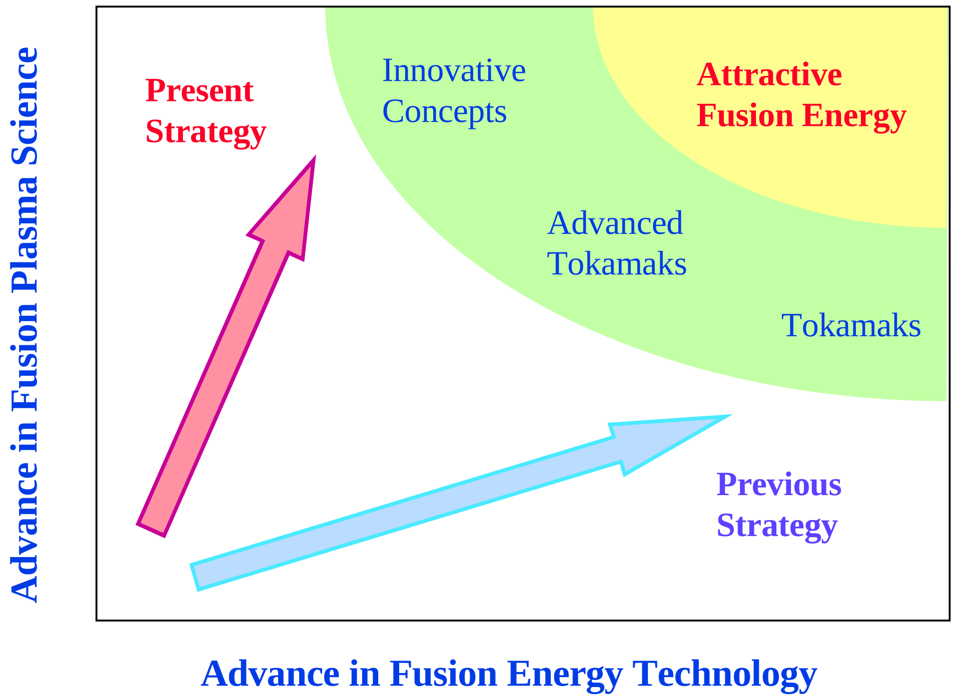


Figure 1. While advances in fusion plasma science as well as energy technology are required for future success of fusion power, present emphasis on the former aims to accelerate concept innovation and enable improved pathways to fusion power.

The small NSTX,[3] which is being built in the U.S., will join in 1999 the complementary MAST[4] (Mega Amp Spherical Tokamak) experiment in the U.K. to prove the scientific principles for the ST concept. A small DTST to prove the ST fusion performance could follow positive outcomes of NSTX and MAST. These steps would complete the fusion plasma science database needed to embark on a small VNS[5] to test and develop energy technology for fusion, which in turn could be followed by a small Pilot Plant[6] to test the practice of electric power production. In this paper we discuss these success-oriented possibilities enabled by an innovative concept such as the ST.

We will point out some of the challenges that have faced the development path based on the mainline tokamak ITER and the development of fusion energy in the U.S. We will recommend a paradigm shift for fusion development by taking advantage of small innovative concepts such as the ST. We will propose some design features of a simplified fusion device made possible by the ST, including the single-turn, normal conducting center leg[7] of the toroidal field coil (TFC). These design features help identify new challenges and opportunities, and suggest directions for further innovation.

We will clarify the database in fusion plasma science needed to enable the VNS and the nearer-term applications, in comparison with the more challenging data needs anticipated for attractive fusion power. There are two distinct levels of data needs that can be addressed sequentially during proofs of principle and performance: the first (more nearly conventional) and the second (advanced[8]) physics regimes of magnetohydrodynamic (MHD) stability. Operation of the VNStype plasma in the first physics regime would provide an adequate environment for testing energy technologies and material combinations needed by the Pilot Plant. Database for the second physics regime, combined with the newly developed fusion energy technologies and material combinations, would contribute to making attractive Pilot Plant and future fusion power. Advances in science and technology in this fashion would also introduce added opportunities for fusion applications other than electric power.

II. Present Challenges

The development of an attractive fusion confinement concept is expected to encompass the following stages:[1]

1) Concept Exploration,
2) Proof of Principle,
3) Proof of Performance and Optimization,
4) Fusion Energy Development (*Energy Technology*), and
5) Fusion Demonstration Power Plant (*DEMO*).

The tokamak concept began Stage 3 in the mid-1980s by the operation of TFTR[9] and JET[10] to prepare for experimentation of D-T-fueled plasmas (Fig. 2). These experiments successfully tested the production of fusion energy in the range of 10-20 MJ over 1-2 s in duration.[11,12] These devices, together with other large D-fueled tokamaks, have wrought great advances in fusion plasma science in general and for the tokamak concept[13] in particular.

These advances contributed to the worldwide interest in ITER (International Thermonuclear Experimental Reactor) and led to the ITER EDA[14] (Engineering Design Activity), which began in 1992. The objectives of ITER are to demonstrate controlled ignition and extended burn, steady-state operation, and technologies essential for a fusion reactor. ITER envisioned as the sole world fusion facility of the next generation for Stage 4, is to have a fusion core commensurate with a future tokamak power station. Ambitious performance are incorporated in the ITER design, such as five orders of magnitude increase in fusion energy produced per plasma pulse and more than 20-fold increase in plasma volume beyond JET and TFTR. The initial results of the EDA[15] suggest that the ITER project will entail very worthwhile but large requirements for the world fusion program.

This ambitious approach, however, has recently become a subject of wide discussion and debate in the U.S. A more recent review[16] by FESAC, while reaffirming the importance of the ITER objectives, identified some important technical issues and suggested upgrade strategies to provide increased confidence in achieving its objectives. The long time scale and the high cost required for developing fusion power and the limitations in the U.S. commitment to ITER have contributed to delaying the realization of ITER, which in turn would delay the progress toward fusion power. Further, fusion development has so far been limited to the Electric Utility as customer. Potential delays in ITER have contributed to weakening the support for fusion energy R&D by the U.S. scientific community and the utility industry.

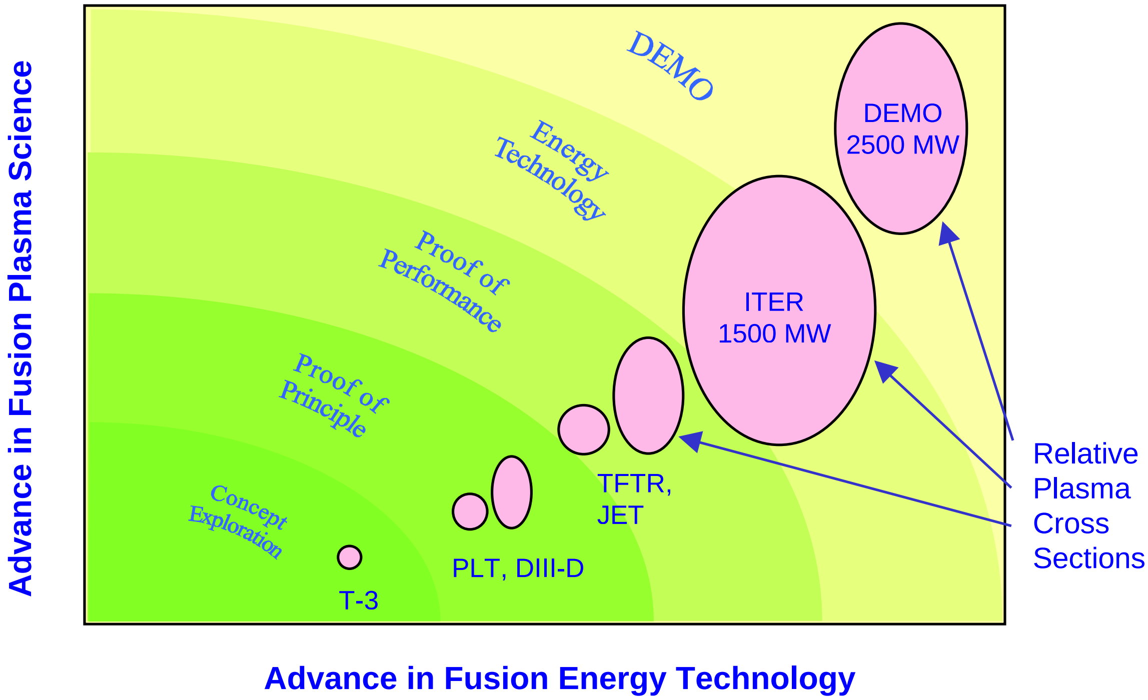


Figure 2. Examples of tokamak experiments leading up to ITER, such as the pioneering T-3[17] (Russia), the PLT[18] and DIII-D[19] (modified from D-III), the TFTR and JET (EC). Great advances in fusion science and energy technology are targeted in this pathway to demonstrate fusion power by a tokamak DEMO.

III. Paradigm Based on Innovative Concept

These challenges accentuated the need to innovate for more attractive magnetic fusion concepts and to examine potentially competitive short-term applications of fusion in diverse markets. Innovations in broad science and technology areas of interest to fusion should be pursued to increase the chances for fusion to become more relevant to the needs of society. A cost-effective strategy needs to be found to develop nearer-term applications and broaden the customer base for fusion, along the pathway to future fusion power. These ideas are consistent with the current fusion strategy articulated by FESAC.[1]

The ST concept provides a useful example in this paradigm (Fig. 3). The concept has benefited from successes in recent exploration tests (START,[20] CDX-U,[21] HIT,[22] TST-3,[23] etc.). More recently in START, an average toroidal beta (plasma pressure over the applied toroidal magnetic field pressure at the major radius) above 30% was measured,[24] when a neutral beam power around 0.5 MW was injected briefly into a small, well-confined ST plasma carrying 0.25 MA in plasma current. A high-speed video image of the plasma is shown in Fig. 4, indicating the "spherical" nature of the ST plasma.

The ST concept became ready in 1995 to venture into the Proof-of-Principle stage (Stage 2). Broadly based decisions soon followed to build NSTX and MAST. These ST devices are

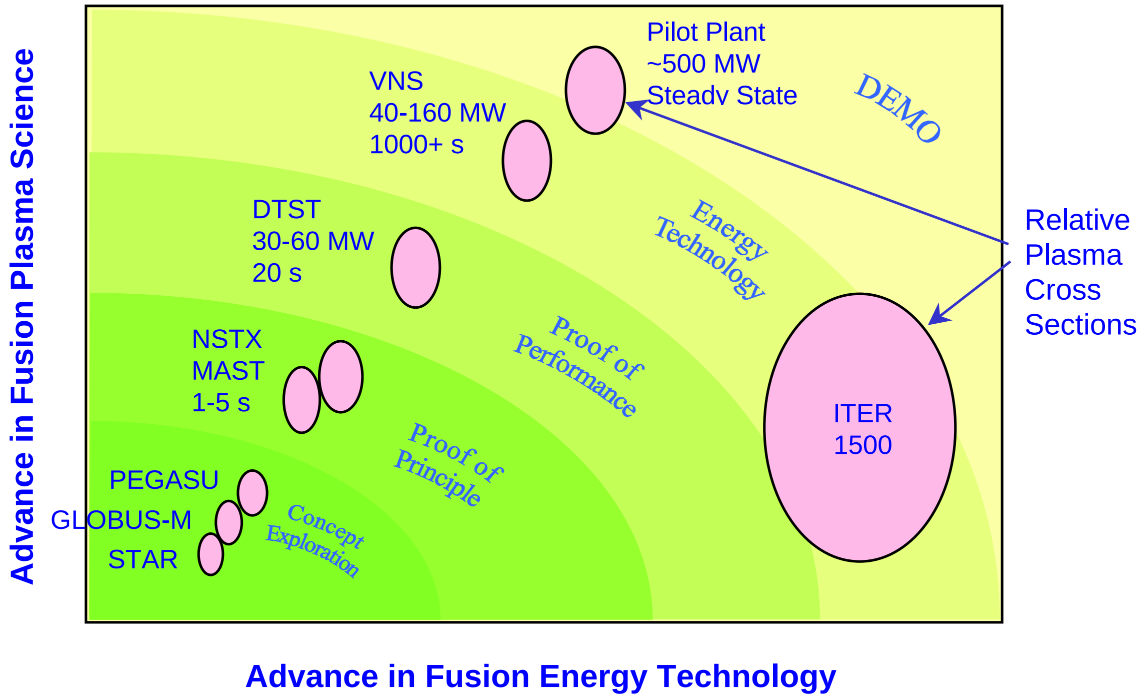


Figure 3. A pathway to fusion power utilizing the ST concept. Similar plasma dimensions are presently projected for the ST devices for Proof of Principle, Proof of Performance, Proof of Energy Technology, and the initial efforts for Power Demonstration.

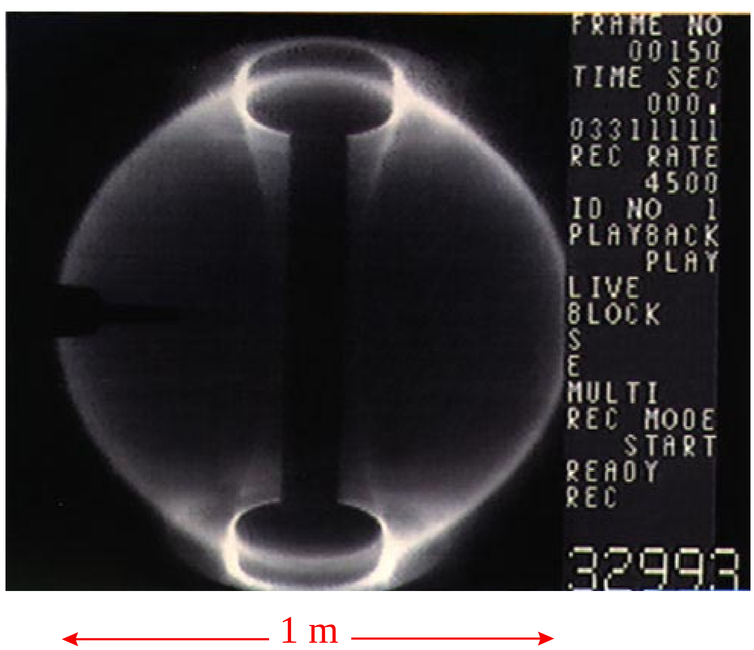


Figure 4. A high-speed video image of the small START plasma under NBI heating (*courtesy of START team*), indicating good plasma confinement and stability at high average toroidal beta (≥30%).

expected to have modest plasma major radii (~0.8 m) and produce sizable plasma currents (1–2-MA level) for up to several seconds. Smaller new experiments, such as Pegasus[25] (University of Wisconsin, U.S.), GLOBUS-M[26] (Ioffe Institute, Russia), and CDX-U[21] are also being built or upgraded to explore the scientific boundaries of the ST plasma parameter space.

Successful outcome from the proof-of-principle experiments would in turn encourage consideration of a small D-T-fueled ST device to prove the fusion performance for the ST concept (Stage 3). An example of such a DTST is estimated to require plasma major radii around 1.1 m and plasma currents around 10 MA, and produce fusion powers in the range of 30–60 MW for up to 20 s in time. This DTST would fit the capabilities available at fusion research facilities in the U.S. Present-day technologies for plasma heating, current drive, and power and particle handling, etc., successfully utilized to enable experimentation in TFTR,[11] JET,[12] and JT-60U,[27] are expected to be adequate for the small DTST.

A successful DTST would provide the fusion performance database adequate for entering Stage 4 of the development. A ST-based VNS for this purpose has been estimated[28] to require plasma major radii and currents similar to a DTST, plus the large challenge of steady state operation and high neutron fluence. Initially a VNS would deliver 0.5–1.0 MW/m^2 in fusion neutron wall loading, similar to the ITER design. The plasma fusion performance for this initial phase of VNS is expected to be in the first physics regime characterized by average toroidal betas around 25%, large external noninductively driven currents (~40% in fraction), and confinement adequate for a modest fusion amplification Q (~1–2).

New technology and material combination database developed with this VNS performance would encompass all the fusion core components, including test blanket modules, nuclear shields, first wall and tiles, divertors and plates, and external drive systems. A ST VNS would in addition provide database for the normal-conducting center leg of the TFC under large irradiation. The new data[29] would enable the development of these components for handling increased neutron wall loading and heat flux in the VNS. Technology and material combination database toward 5 MW/m^2 would be desirable for a Pilot Plant[30] to practice electric power production and begin Stage 5 of fusion development. The merits of building a small Component Test Facility (CTF) following MAST and NSTX was also suggested recently by Robinson.[31]

The VNS plasma can produce this high wall loading[29] if it could operate successfully in the second (advanced[8]) physics regime. The principles and the performance for this more attractive physics regime can be tested using the NSTX and the DTST. It is anticipated that this advanced regime will require extensive physics optimization, since it calls for simultaneous achievement in steady state of several highly desirable plasma properties. These include superb plasma confinement (approaching the irreducible neoclassical level); very high betas (~50%) relying on wall stabilization and mode control; well-aligned self-driven plasma currents (≥90% in current fractions); efficient external drive of the remaining differences in plasma current; and dispersion (by an order of magnitude) of power and particle fluxes over broad areas of the plasma facing components. Profiles for plasma density, temperature, current, diffusivities, fueling, heating, and driven current throughout the plasma core, the edge and the scrape-off layer (SOL) must be controlled or optimized to achieve this highly desirable state of plasma condition.

Given the advanced physics regime, a VNS upgraded[29] to produce and handle 100s of MW in fusion power and several MW/m^2 in wall loading could begin tests of electric power production in the style of a Pilot Plant. However, reliable operation of a Pilot Plant would also require extensive efforts utilizing the VNS to test and optimize[5] all energy technology components for long life times and reliability. Reliable operation in the optimized advanced-regime plasmas using high-performance energy technology components will therefore require extensive database from NSTX, DTST and VNS. The Pilot Plant utilizing these advanced databases is estimated to be similar to the DTST and VNS in plasma size and current.

The plasma sizes and performances for Proof of Principle, Proof of Performance, Fusion Energy Development (*Energy Technology*), and a pilot Fusion Demonstration Power Plant (*DEMO*) in a pathway that utilizes the ST are depicted in Fig. 3. Such a pathway aims to advance in concert fusion plasma science (from conventional to advanced physics regime) and energy technology (from moderate to high neutron wall loading) for Stages 2–4. This pathway should remain robust to the outcome of the upcoming decisions beyond the ITER EDA. As a complement to ITER, a ST VNS would test and develop at neutron fluence beyond the ITER design energy technologies needed for the DEMO. ITER and VNS combined would then establish the scientific and technological base for the DEMO. On the other hand, the VNS alone would test and develop energy technologies for electrical power production. NSTX, DTST and VNS combined would then establish the scientific and technological base for a Pilot Plant.

Devices that enable a cost-effective pathway to fusion power should characterize the new paradigm for fusion development. The potential small sizes and high performances of the ST devices, compared to the mainline tokamak ITER, highlight this important benefit of fusion concept innovation. In the next section we discuss the envisioned design configuration for the ST fusion device in support of such a pathway.

IV. Small Simplified Configuration

Fusion device designs with a small and simplified configuration will be indispensable in building a cost-effective pathway to fusion power. A design concept for a ST VNS, currently under a SBIR (Small Business Innovative Research) phase-2 study,[32] provides an example. Figure 5 depicts this design for the VNS containing a DTST-size ST plasma. It features modular designs and

1) A single-turn, water-cooled, normal-conducting TFC center leg with little or no shielding.[7]
2) Sliding-joints to permit a demountable center leg that carries little external loads,
3) Vacuum vessel combined with the TFC return legs to simplify the mechanical load path,
4) Removable covers for the vacuum vessel to permit vertical remote maintenance for all activated components: the divertors, the blankets, the first wall, some shields, and the center leg,
5) Radial access for blanket test modules, plasma drive systems, and in-vessel maintenance, and
6) Hands-on maintenance after shut down for all components external to shield boundary.

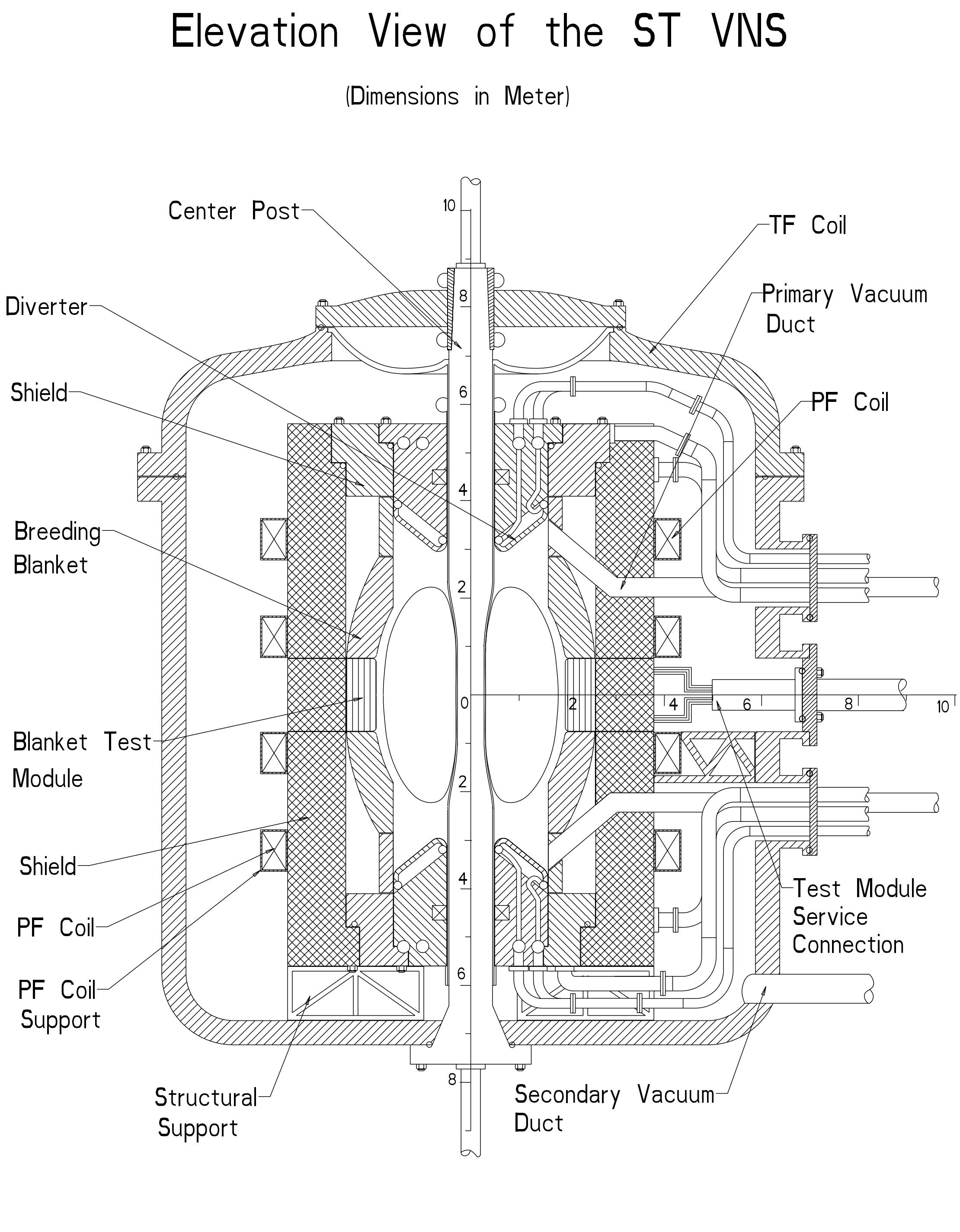


Figure 5. Elevation view depicting the design concept for a ST VNS fusion device. A single turn center leg of the TFC with little or no nuclear shielding enables the various attractive features.

This unusual approach for the TFC will require the initiation and ramp-up of the plasma current without reliance on induction, permitting the removal of the solenoid inboard of the plasma torus. As a result the space required for the inboard hole of the torus is minimized, which in turn minimizes the aspect ratio and size for the ST plasma. This leads to small size for in-vessel components and test modules while maintaining high wall loading, and minimizes the total volume of material subjected to activation. This TFC approach and the small ST plasma combine to permit the simplified torus configuration that makes remote maintenance more straightforward. The modest plasma minor radius (≤0.8 m) and magnetic fields (~2–3 T) further permit the use of present-day neutral beam injection (NBI) and rf technologies for plasma heating and current drive. These factors further help reduce the number and scale of new technologies needed, including the superconducting TFCs being replaced by the normal conducting option.

These potential advantages, however, are subjected to the lack of experience and data for the TFC center leg of normal conductor alloys. Recent analysis[33] suggested that a GlidCop center leg could perform adequately for the purpose of VNS. Database obtained from the initial tests in VNS would open opportunities for improvements in the material, design, and fabrication of the center leg, for possible utilization in a VNS of increased performance and a Pilot Plant.

V. Nearer-Term Applications

Non-electric applications of fusion neutrons and heat have been considered in the recent years,[34,35] utilizing large, powerful, and relatively costly fusion device concepts based on the tokamak. These applications can become of increased interest if fusion neutrons and heat could be produced at lowered costs in small units, as is our hope for the ST concept.

For example (Fig. 6), the fusion neutrons produced for a limited duration in a DTST-like fusion source could be utilized for studying neutron science and radiography.[36] The more abundant neutrons produced by the VNS in steady state could be used for producing radioisotopes[34] for medical and other applications. A test module in a VNS (Fig.5) could be dedicated to this purpose utilizing the 14-MeV neutrons. If a ST-VNS-like fusion source can reliably produce up to 1 MW/m^2 in neutron wall loading, appropriate high-gain blanket concepts, driven by moderate fusion neutrons to a multiplication of 30–50, could be conceived[37] for transmuting and burning long-life fissile Pu or heavy actinide wastes produced by fission. The potential advantages of smaller unit size, thermal power and nuclear inventory plus higher neutron fluxes compared to a tokamak-driven system[38] should bring added interest to this topic, especially where these heavy actinides could become available. Further, high neutron wall loading produced in a Pilot Plant could be useful for breeding tritium, in unit sizes much smaller than a tokamak equivalent.[35] In a longer term, if high efficiency for converting heat to electricity could be achieved in a Pilot Plant, production of hydrogen as a clean chemical fuel[39] and conversion of hazardous chemicals into safer forms may deserve some attention.

The up-coming Proof of Principle experiments NSTX and MAST will test the new scientific principles of the ST fusion source, which might enable these potential applications. Initial tests for some of these applications could be carried out in a VNS in modules of modest dimensions.

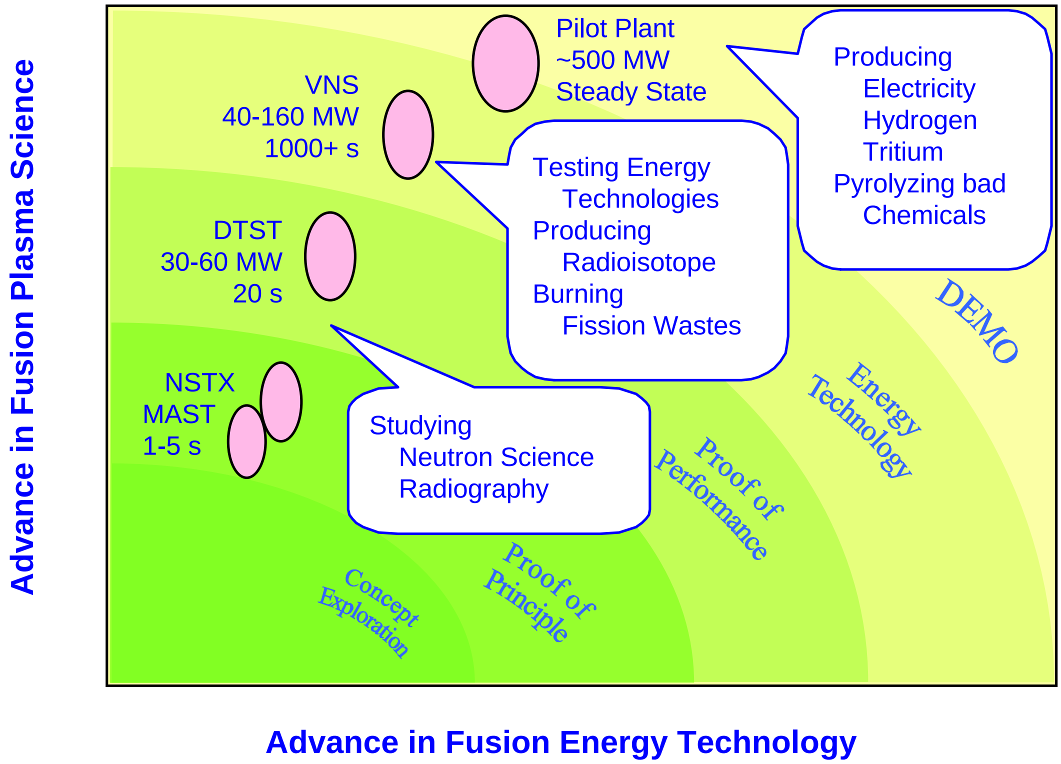


Figure 6. Possible nearer-term applications of fusion neutrons produced along the pathway to fusion power.

VI. Proof of Principle Experiment NSTX

The Proof of Principle experiments NSTX and MAST are expected to begin experimentation in 1999 and 1998, respectively. The mission of NSTX is to investigate, with broad participation by fusion laboratories and universities, the scientific principles of

1) Noninductive startup, current sustainment and profile control,
2) Confinement and transport,
3) Pressure limits and self-driven currents,
4) Scrape-off layers and divertors, and
5) Stability and disruption resilience.

These principles are to be investigated in the physics regimes of relevance to the near-term applications such as the small fusion sources for the VNS and future applications such as the Pilot Plant and the Power Plant. These regimes are characterized by simultaneous

1) High confinement (near the neoclassical limit), average toroidal betas (25 – 45%), and self-driven current fractions (50 – 90%);
2) Efficient external noninductive sustainment of current increments; and
3) Highly dispersed plasma energy and particle fluxed on the plasma facing components

under steady-state conditions with fully "relaxed" profiles. As pointed out in Section 4, noninductive initiation and ramp-up of the plasma current must be investigated to determine if the solenoid can be eliminated to minimize the size of the ST fusion device.

That the ST plasma promises these attractive properties can be surmised from its magnetic configuration. Figure 7 compares the magnetic configuration of ST with those of the tokamak and the spheromak[40] having an identical plasma cross-section. It is seen that the magnetic field line on the inboard of the ST plasma is largely toroidal in direction (around the axis of rotational symmetry of the torus), similar to that of the tokamak plasma. However, the magnetic field line on the outboard of the ST plasma is largely poloidal in direction (around the cross section of plasma), similar to that of the spheromak plasma. As a result, the length of the field line on the stable (inboard) side of the plasma for the ST is longer than that on the unstable (outboard) side, while the reverse is true for the other cases. This enables the ST to maintain higher plasma pressure stably than the other configurations.

Many interesting plasma properties are expected of the ST plasma as a result. For example, the ST has a much-reduced TFC current relative to the plasma current[2] compared to the tokamak. This reduces the toroidal field to the level of poloidal field and helps form a "magnetic well" when the plasma is heated to a moderate poloidal beta ~1 (average plasma pressure divided by the pressure of the average poloidal field near plasma edge). Highly elongated plasmas can be formed in the ST using simple poloidal field coil systems and are more stable against vertical

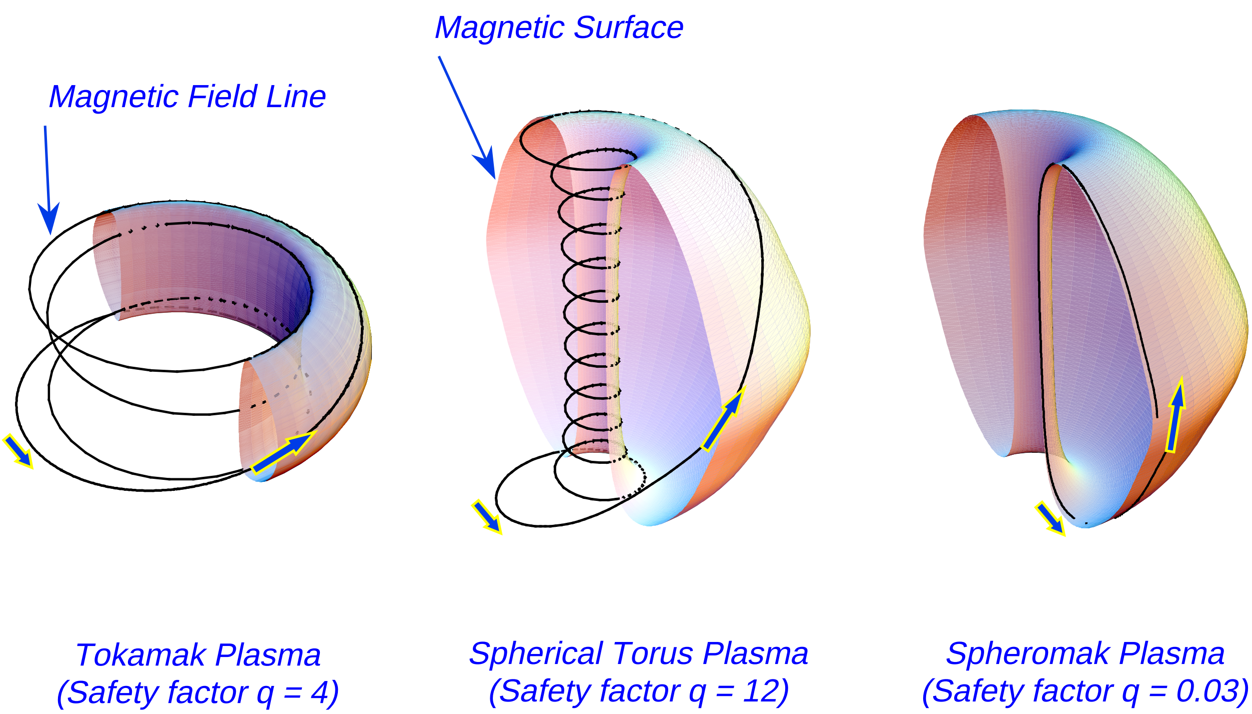


Figure 7. Magnetic surfaces and field lines near the edge of the tokamak, the spherical torus, and the spheromak plasmas.

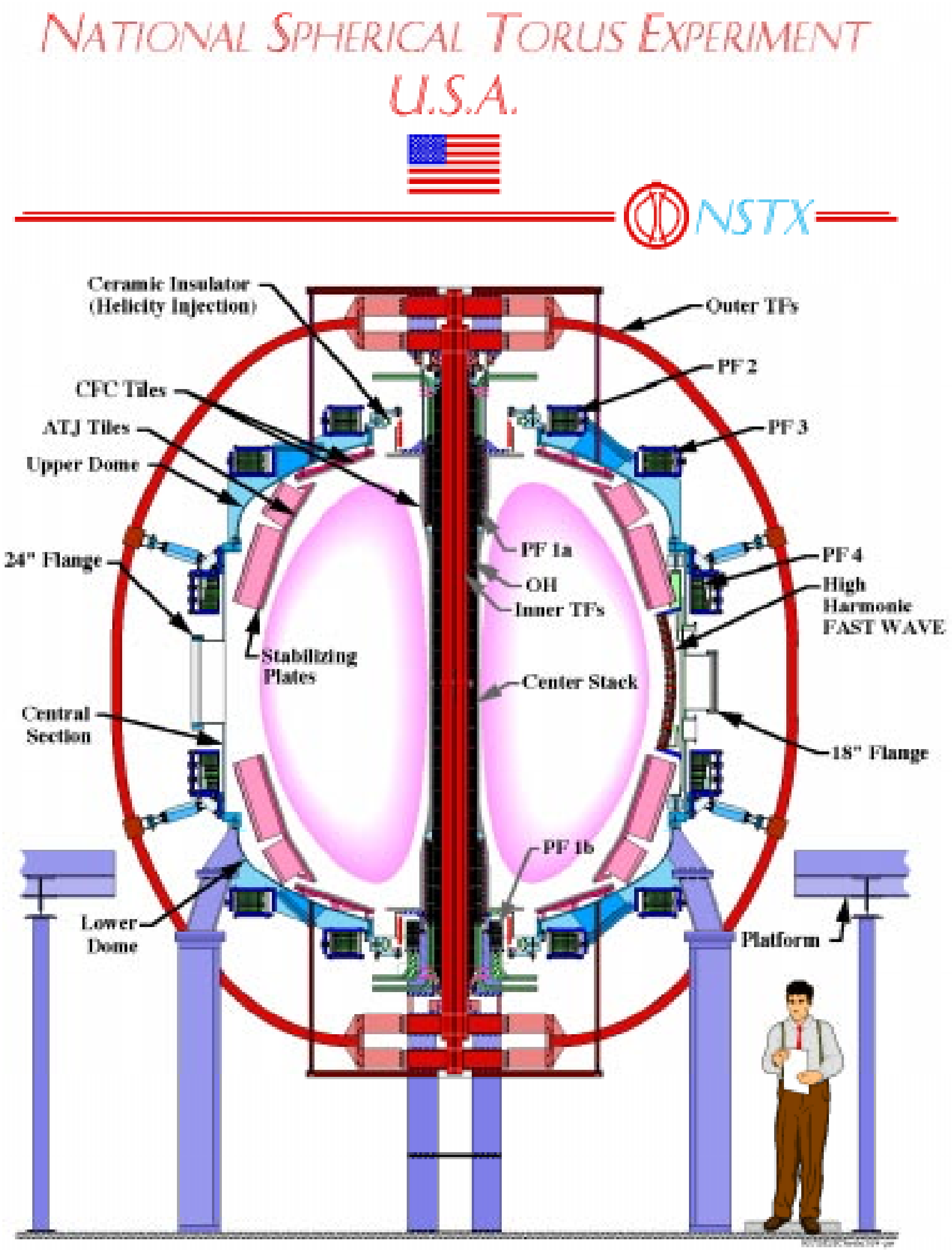
displacements than the tokamak plasmas with a similar elongation. High plasma elongations also contribute to increasing plasma current, confinement and stability.

These features in turn lead to additional plasma properties of interest to future applications. These properties are important issues for experimental verification on NSTX, and include:

1) Minimized plasma magnetic fluxes and helicity (which is roughly the product of the toroidal and poloidal magnetic fluxes) for a given plasma current, that eases noninductive startup of the plasma current via CHI[41] (coaxial helicity injection) or ECH (electron cyclotron heating) driven bootstrap current;[42]
2) Very large self-driven plasma current (up to 90% in fraction) that minimizes the need for external current drive for sustainment and also ensures stability at high average beta (up to 50%).[43] About half of these levels would be adequate for a viable VNS of small radii, whereas the full levels could enable attractive Pilot Plant and Power Plant in the future;
3) Strong magnetic well in the plasma core (~30% has been calculated[43] so far) and reversed local magnetic shear at the outboard plasma core, that tend to reduce the microturbulence[44] responsible for large enhanced loss of energy from the plasma core;
4) Strong sheared flow[45] driven by large pressure gradient resulting from high beta in the plasma, that tends to inhibit[46] the remaining microturbulence and potentially enable confinement at the neoclassical level;
5) Large sheared flow and magnetic well, which could also improve the neoclassical confinement[47] in the absence of microturbulence;
6) High plasma dielectric constant at high temperatures, that permits new branches of radio frequency waves such as the High Harmonic Fast Wave (HHFW)[48] to heat plasma and drive plasma current;
7) Plasmas having SOL with high mirror ratios (up to ~4) and strong curvature, potentially leading to dispersed heat and particle fluxes on the divertor plates and reduced heat fluxes on the inboard tiles;[49] and
8) Ion speeds due to NBI or high ion temperature in excess of the Alfven speed in the outboard low field region of the plasma.

Tests of these intriguing properties and their consequences will be carried out during the initial years of NSTX experimentation. The results will help determine if the ST concept would be ready to begin Stage 3 for Proof of Performance, and if so, provide initial database for use by the design of a possible DTST.

The design and fabrication of NSTX (Fig. 8) began in October 1996 as a joint project by PPPL, ORNL, University of Washington, and Columbia University. The major parameters designed for NSTX are also provided. The NSTX will utilize much equipment and facility already available at PPPL (Fig. 9) and elsewhere. These include the ICRF (ion cyclotron radio frequency) system to deliver 6 MW for 5 s in HHFW, ECH (electron cyclotron heating) systems, NBI systems to deliver 5 MW for 5 s as a future addition, power supplies, diagnostics systems, and the NSTX Test Cell (previously the TFTR Hot Cell). The first plasma in NSTX is planned for April 1999 prior to experimental operation beginning in May 1999.



Baseline Parameter

Major radius
85 cm

Minor radius
68 cm

Plasma current
1 MA

Toroidal field
0.3 T

RF heating and current drive
6 MW

Flat-top time
5 s

Figure 8. Artist's rendition of the NSTX design, characterized by a slender demountable center stack containing the TFC inner legs and a full solenoid, modest TFC return legs, fully bakable vessel covered with graphite tiles, simplified poloidal field coil system, and large access ports.

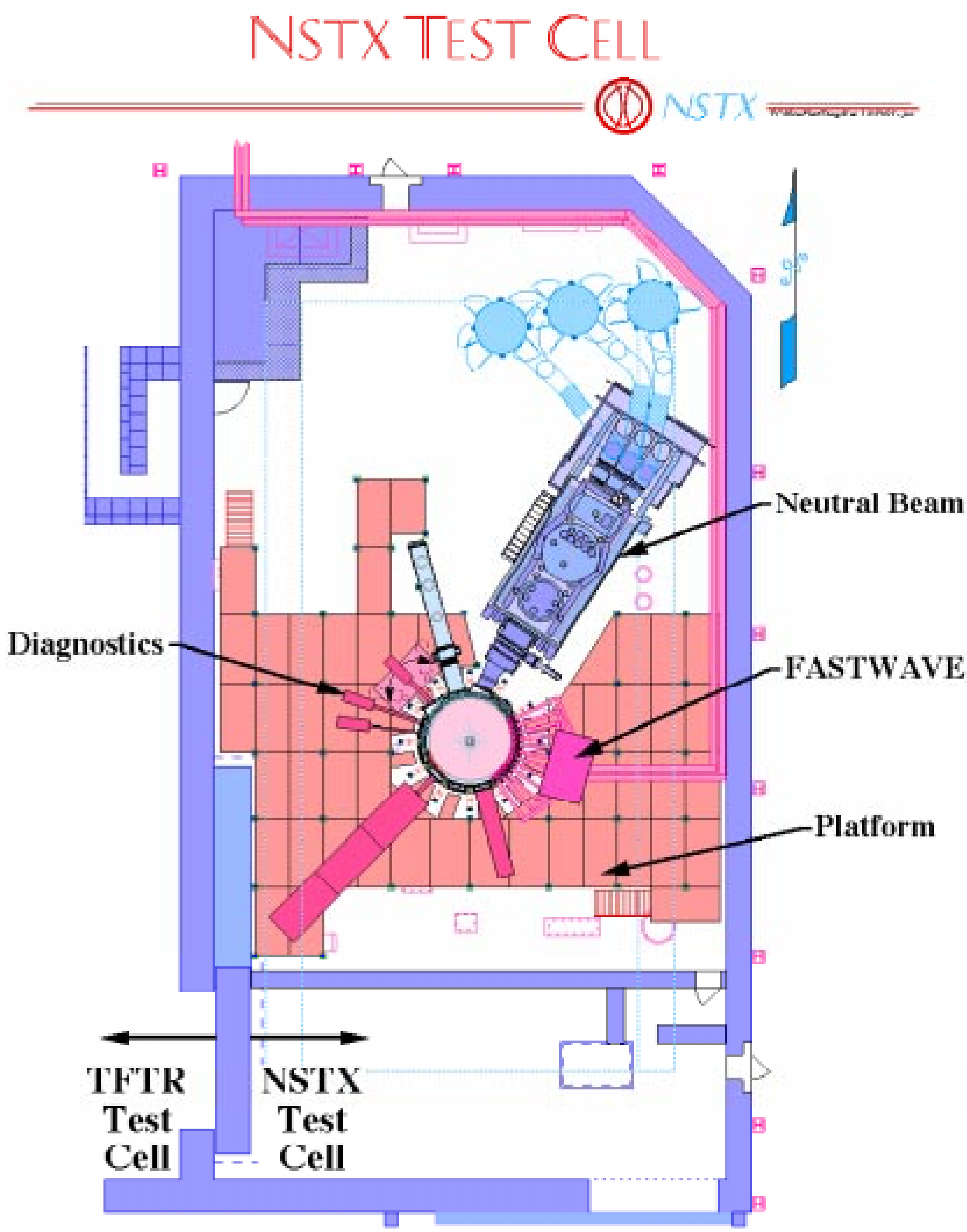


Figure 9. Layout planned for locating NSTX in the Test Cell (previously the Hot Cell) at PPPL utilizing existing ICRF fast wave and diagnostics systems. Also indicated is a possible arrangement for a TFTR NBI system as a future addition.

Expected to be the only new ST experiments capable of MA in plasma current in the near future, NSTX and MAST should carry out research encompassing all innovative scientific issues of the ST plasma just described. Table 1 summarizes these research topics for the initial years of experimentation on NSTX, and points to the NBI as an important tool in this effort to advance innovation in fusion plasma science.

VII. Proof of Performance Experiment and Beyond

Assuming positive physics outcome from NSTX and MAST, the parameters for a DTST, a VNS, and a Pilot Plant can be estimated and are provided in Table 2. It is seen that these devices are expected to have similar major radii around 1.1 m. Parameters for the NSTX are also listed for comparison.

The mission of the DTST would be to prove at low neutron fluence the fusion plasma performance for the VNS and the Pilot Plant. This DTST can fully utilize facilities and expertise already available in the U.S. Take the TFTR Test Cell facility as an example; the existing motor

Table 1. Topics of interest to research on NSTX for fiscal years 1999-2001

NSTX Working Group Topics • Subtopics	**FY 1999**	**FY 2000**	**FY 2001**
1) **Slow (MHD) Mechanisms for Current Formation and Sustainment**			
• Inductive mechanisms (w & w/o electron-cyclotron preionization)	√		
• Plasma operational space and scenarios	√	√	
• Coaxial helicity injection, CHI	√	√	√
• Noninductive RF techniques		√	√
2) **Fast Mechanisms for Heating and Current Drive**			
• High Harmonic Fast Wave heating and current drive physics	√	√*	√*
• NBI heating and current drive physics		√*	√*
• Large and well-aligned self-driven current physics			√*
3) **Magnetics and Stability Limits**			
• Beta limiting processes	√	√*	√*
• Fast-ion driven instabilities (e.g., Alfven modes)		√*	√*
• Control of plasma and unstable modes		√*	√*
4) **Plasma Transport and Fluctuations**			
• Global confinement	√	√*	√*
• Local transport		√*	√*
• Microinstabilities and turbulence		√*	√*
• Turbulence suppression and transport barrier formation			√*
5) **Divertor, Scrape-Off Layer, Power and Particle Handling**			
• Vessel and tile conditioning	√	√	
• SOL properties of diverted and inboard wall limited plasmas	√	√	√*
• Maintenance of edge transport barriers		√	√*
• Effects of large mirror ratios (e.g., velocity-space instabilities)		√	√*

* NBI and NBI-based diagnostics assumed

Table 2. Progression of plasma parameters and performance requirements projected for NSTX, DTST, VNS, and Pilot Plant along the ST pathway to fusion power

Parameters and Features	**NSTX**	**DTST**	**VNS**	**Pilot**
Major radius (m)	0.85	1.1	1.07	~1.1
Applied toroidal field (T) at major radius	0.3–0.6	1.9	2.1	~2.6
Plasma current (MA)	1–2	10	10	~15
Plasma cross section elongation	2–3	~3	~3	~3
Toroidal beta (%)	25–45	24–40	24–37	~50
Bootstrap current fraction (%)	50–90	50–90	~60	~90
Plasma drive power (MW)	6–11	18–33	24–57	~25
NBI energy (keV)	80–110	120	120–400	400
Fusion power (MW)	–	33–60	40–160	~500
Plasma flat-top pulse (s)	5–1	20	1000+	s.s.
Neutron wall load (MW/m^2)	–	0.5–1.0	0.5–2.0	~8
Neutron fluence per year ($MW\text{-}yr/m^2$)	–	0.003	0.2–0.6	~4.0
Tritium breeding ratio (%)	–	0	≤70	≥100

generator capability of 4 GJ would be adequate for powering all the magnets and auxiliary plasma drive systems for a plasma burn time of more than 20 s. The total available auxiliary drive power for the plasma amounts to more than 40 MW in ICRF and NBI, though extension of the NBI pulse to ~20 s will be required. The dose limitation for the PPPL site is estimated to be 2×10^{22} fusion neutrons per year, permitting up to about 70 full D-T burn pulses at 40 MW in fusion power each year. Because of the small size, the cost of building a DTST is expected to be comparable to the value of the already-installed facility and equipment.

The VNS and the Pilot Plant would have much higher pulse length and neutron fluence required for proving energy technology and practicing power production, respectively. NBI energies in the 80–120-kV range are expected to be adequate for experimentation on NSTX and DTST with plasma densities up to 10^{14} cm^{-3}. To reach fusion powers in the range of 160–500 MW in the VNS and the Pilot Plant, however, would require plasma densities of a few times 10^{14} cm^{-3}. These densities would suggest the use of NBI at higher energies (~400 kV), such as those been utilized in JT60-U.[50] Successful outcome in HHFW heating and current drive at high temperatures and densities could complement or relieve the use of NBI.

The challenging fusion energy development and demonstration missions for these latter devices are expected to lead to large requirements and cost in engineering, material, shielding, technology, remote maintenance, tritium, and facility. For Example, the weights estimated for the NSTX and the DTST devices are about 100 tons and 1000 tons, respectively, the latter largely due to significant but limited neutron shielding requirements. Shields would dominate the weight for the VNS device (Fig. 5), pushing the device weight beyond 5000 tons. Steady state operation would also entail large increases in the cost of the fusion device and facility.

Further advances in ST fusion science and energy technology beyond the VNS would be required for the Pilot Plant to ensure reliability and long life. Proper design concept for a Pilot Plant could be determined effectively following the positive outcome of the VNS efforts.[5] Because of the similar plasma sizes, the VNS plasma and component performances can in principle be upgraded in time to approach those for the Pilot Plant.[6,30] Partial tests of electricity generation in small scale in a VNS should reduce the risk of subsequent electricity generation in large scale in the Pilot Plant.

Figure 10 indicates this relationship among the ST devices for Concept Exploration, NSTX and MAST for Proof of Principle, DTST for Proof of Performance, VNS for Fusion Energy Development (*Energy Technology*), and Pilot Plant to begin the effort of Fusion Demonstration Power Plant (*Power Demonstration*), along an attractive pathway to fusion power. Low-cost upgrades of NSTX to 5 MA in plasma current (for 10 s in pulse length) and to 30 s in pulse length (at 2 MA in plasma current) could be considered for testing in Deuterium the ST physics principles at plasma pressures similar to those of today's largest tokamaks.

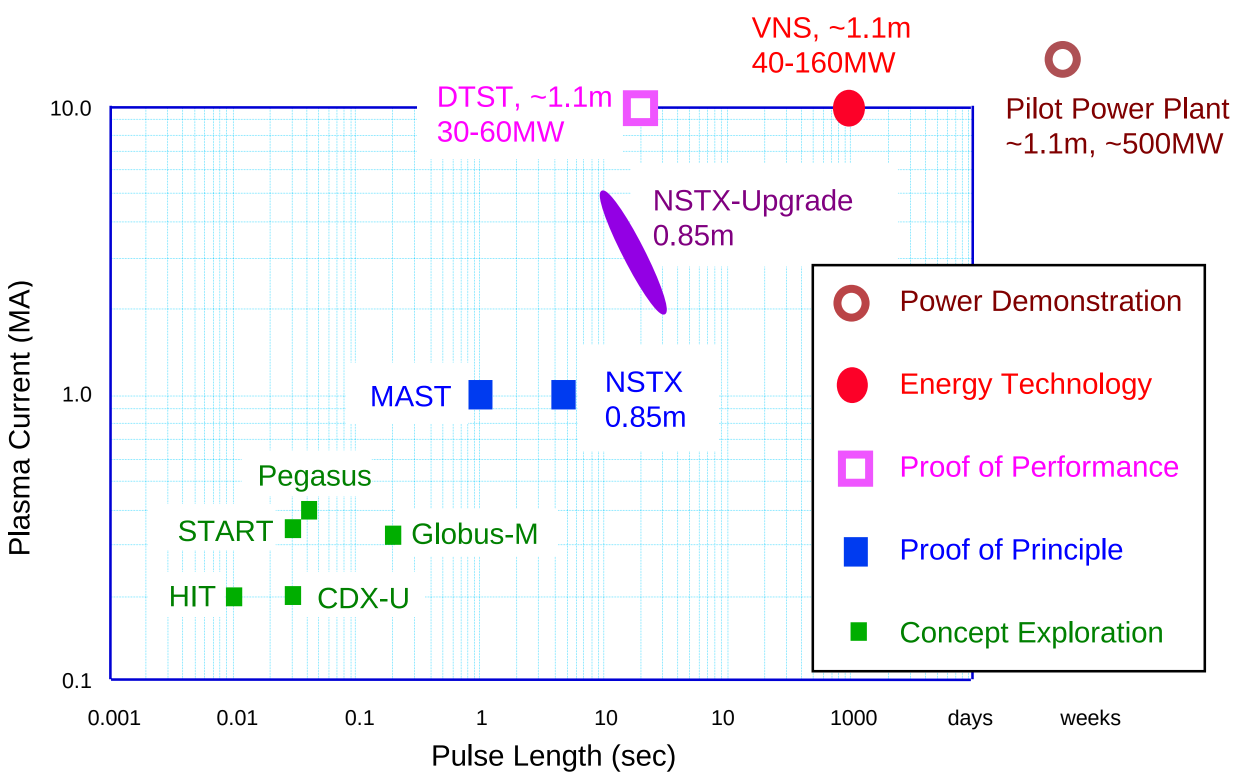


Figure 10. Plasma currents and pulse lengths of spherical torus experiments in advancing stages of development toward fusion power. Major radii and D-T burn powers are indicated for DTST, VNS, and Pilot Plant. NSTX and MAST will begin to prove the scientific principles of this pathway.

VIII. New Opportunities and Challenges

The ST concept and the potentially attractive ST pathway to fusion power serve as examples for the new opportunities that may derive from concept innovation. This, together with the tokamak and other emerging innovative concepts, should add credence to a renewed fusion energy sciences program in the U.S. However, the new opportunities also introduce some foreseeable challenges, particularly in the process of selecting among the new concepts those appropriate for testing at progressive stages of development. The following provides some ideas for discussion.

Innovation is largely unpredictable in inception and outcome. While we encourage innovation, a working understanding of what could best promote new concepts for exploration will be very helpful. Among the necessary considerations for this purpose should be scientific novelty (so far unexplored or not adequately explored) and potential interest to fusion in the near term as well as the long term.

The progress of an innovative concept toward D-fueled Proof of Principle test will most likely require more than interesting experimental results from the exploratory tests. The test results must also be cast in terms of the present understanding of fusion and the related scientific research, and be estimated to remain scientifically viable for some future applications. The new understanding should provide a basis adequate for the design of the Proof of Principle experiment. A nationally based effort should be formed to garner the broad expertise and equipment needed for peer review and cost-effective implementation.

The progress toward Proof of Performance and Optimization most likely will entail requirements in addition to positive outcomes in the scientific principles. A clear pathway to possible near-term and future fusion applications should drive the decision for D-T-fueled tests. The interests of the world fusion community must also be taken into account. Highly cost-effective sites in the world, including those in the U.S., should be considered for this purpose. If the outcome for a concept is very positive at the level of Proof of Principle, utilization of the “advanced” fuel cycles,[51] such as D-^{3}He, and D-D, should also be evaluated.

A possible VNS and an ensuing Pilot Plant, including nearer-term applications of the copious fusion neutrons and heat so produced, would most likely require a “nuclear” site fully equipped to handle high activation, manage tritium, and comply with safety and environmental rules. An international partnership will most likely be required to bring such an effort into reality. The experience of the ITER process would be extremely valuable for this purpose, independently of the status of the ITER effort at the time.

Fusion power based on the scientific and technological database accumulated along a cost-effective pathway and utilizing attractive innovations could then maximize the chance for success of fusion power in the future.

In this respect, the ST concept has been fortunate so far in obtaining very encouraging results in Concept Exploration tests using START, CDX-U, HIT, TS-3, etc., as well as theoretical calculations. With safety factors q similar to and likely higher than those in successful tokamaks, the new ST data can draw from and contribute to the extensive knowledge of fusion energy sciences already acquired in the world fusion community. As a result there is significant confidence in the present estimates of the potential contributions by the ST concept through the stage of Fusion Energy Development. World-class Proof of Principle facilities NSTX and MAST are under preparation for utilization by the fusion community beginning in 1998-1999. However, there is not yet adequate assurance at present that the results from these and other new tests will justify experimentation at the Proof of Performance level. The viability and decision for a DTST should therefore await the scientific outcome of NSTX and MAST.

Further, there is at present no assurance that satisfactory answers can soon be found for the engineering and technology needed by the Pilot Plant and the Power Plant in general. The issues for the ST concept in particular include high heat flux on the plasma facing components, and high neutron flux on blankets, the TFC center leg, and the plasma facing components. As a result there is at present a range of assumptions and opinion[52] on the concepts for the future ST Power Plants. Varying projections, however, have served to identify issues of high potential leverage and interest in the development of the ST concept. Innovations in these and other areas are therefore encouraged and should build on the continued progress in ST, tokamak, other innovative concepts, and enabling technologies and materials (such as for high conductivity plus strong resistance to radiation damage and activation). The ability to utilize the small VNS for Fusion Energy Development may become crucial in an effort to find attractive solutions for these and other issues.

Our discussion indicates that advances in these technology areas in the nearer term should in turn enable tests in fusion plasma science at progressive levels, particularly in compact configurations of high power densities such as the ST. Nearer-term, non-electric-power applications of the ST fusion source may demand less in these issues and realize interim benefits of fusion energy, while contributing to the ultimate realization of fusion power.

The ST, with the possibility of small fusion sources at lowered costs and in simplified configuration, should therefore contribute greatly in overcoming some of these challenges in the nearer term. The NSTX, with a major radius of only 0.85 m, will begin experimentation in 1999. The example of DTST, with a major radii of only ~1.1 m, would well fit the available fusion facilities and expertise to require a moderate cost. The scientific database so established could lay the foundation for launching via a small VNS the fusion energy sciences development into Stage 4 in the not too distant future.

The base performance of this ST VNS plasma would require only the first (conventional) physics regime, leaving large margins from the limits anticipated for the advanced physics regime. This should enhance the reliability of the initial VNS plasma operation, while the Proof of Principle and Proof of Performance ST experiments explore the reality of the advanced regime. It is therefore hopeful that these experiments working in concert could, for the nearer term, bring about significant advances in fusion energy sciences. Continued innovation, such as in new physics concepts of high performance that also reduces or eliminates the center leg of the TFC, and in new conductor materials that resist neutron damage and activation, could in a longer term continue this advancement toward practical fusion power.

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