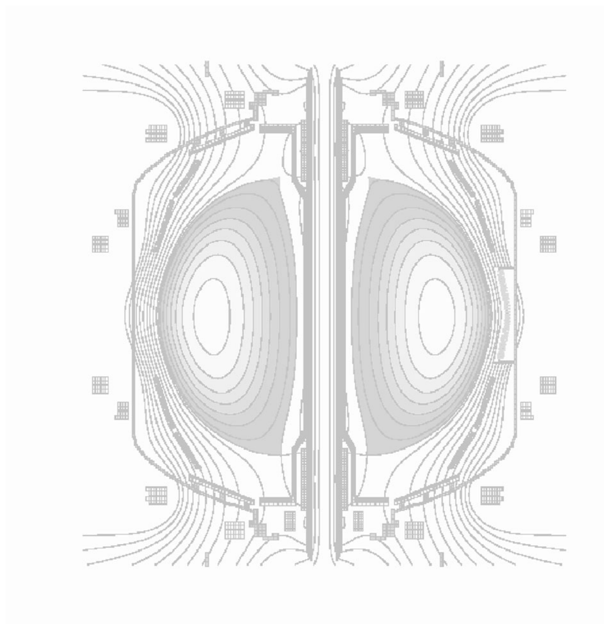

Chapter 6

NSTX and the Magnetic Fusion Energy Development Path

The planned research on NSTX will address the 5 and 10-year ST objectives set forth by the OFES Integrated Program Planning Activity (IPPA). NSTX research will thus advance the scientific basis for cost-effective Performance Extension and Fusion Energy Development (component testing) ST devices by determining the attractive ST regimes of high beta, strong toroidicity, and plasma shaping. This research will also contributes to advancing the broader science of Innovative Confinement Concepts (Spheromak, Compact Stellarator, Field Reversed Pinch, Reversed Field Configuration, etc.) and improving the next-step tokamak burning plasma experiment. The NSTX research program aims to contribute strongly to establishing the scientific and technological bases for an optimized DEMO.

6.1 ST Contributions in an Updated Plan for the Development of Fusion Energy

The Fusion Energy Science Advisory Committee (FESAC) of USDOE recently formulated an updated plan for the development of fusion energy and specifically for the deployment of a fusion demonstration power plant (DEMO) producing net electricity within approximately 35 years [1]. The Magnetic Fusion Energy (MFE) path of the plan identifies a set of major plasma-based facilities needed to address the scientific and technological challenges of Fusion Energy, in the program elements of configuration optimization, burning plasma, and component testing (see figure 6.1).

An important goal for NSTX research is therefore to determine if the ST can be an attractive candidate for a next-step Performance Extension (PE) experiment for configuration optimization [2], which in turn will provide the scientific database needed by an ST-based Component Test Facility (CTF) [3], and ultimately an ST-based DEMO. This goal is in accord with the IPPA MFES program Goal 2, which states that configuration optimization research should "Resolve outstanding scientific issues and establish reduced-cost paths to more attractive fusion energy systems by investigating a broad range of innovative magnetic confinement configurations." The ST is among the leading innovative concepts for configuration optimization, and so may be in a position to compete for one of the new MFE PE experiments. Detail of how the NSTX research planned for the next 5 years will address these needs will be discussed in Section 6.2.

Configuration Optimization is an important component of the updated FESAC plan for fusion energy sciences development. Innovative Confinement Concepts (ICC), such as the Spheromak, the Reversed Field Pinch (RFP), and Field Reversed Configuration (FRC), have the potential to make break-through progress to enable a highly attractive PE level experiments in the next decade. These configurations and the ST share many of the potentially attractive plasma parameters stemming from high β, and therefore share much of the research subject of interest. Detail of how NSTX research can contribute to advancing the physics basis of these ICC configurations in the next 5 year will also be provided in Section 6.2.

Burning plasma tests, an important component of the FESAC recommendations, are expected to be carried out in ITER [6.4] or FIRE [6.5]. The International Tokamak Physics Activity (ITPA) was organized recently to coordinate the development of additional database for the physics optimization of the ITER plasma. A secondary goal for NSTX research is therefore to extend the toroidal physics basis to very high beta and very strong toroidicity (very low aspect ratio), thereby to contribute to better physics understanding and thus more optimal utilization of the burning plasma experiment can be achieved. Detail of this planned contribution is discussed in Section 6.3. The relationship between NSTX, the future ST devices, the potential progress in research in the Spheromak, RFP, and FRC, and the tokamak burning plasma experiment(s) is depicted in Figure 6.2.

Successful implementation of the MFE development strategy will establish the scientific and technological bases for an optimized DEMO to produce net electricity in approximately 35 years. As depicted in Fig. 6.1, a new ST PE-level experiment for Configuration Optimization, assuming positive results from NSTX in the next 5 years, could be built with a goal of beginning research in parallel with Phase-I research on ITER. Assuming positive results from this PE experiment, an Energy Development level compact ST device can be built to begin component testing in parallel with Phase-II research on ITER. The spherical torus configuration may therefore be able to make critically important contributions to this strategic goal for MFE. A major programmatic goal of the present NSTX 5-year plan is to determine the attractiveness of the ST configuration to support a decision to proceed with a PE-level ST experiment.

The design concepts for a next step spherical torus, NSST, and an ST-based CTF are presently been explored, utilizing the latest physics and engineering results from ST research [2, 3] and the latest concepts for ST-based Volume Neutron Source (VNS) [6] and Power Plant [7, 8]. These efforts are part of preparation for the achievement of a 10-year objective of the IPPA MFE program Goal 3: "Assess potential of Spherical Torus as a basis for burning plasma studies and/or fusion-nuclear component testing." In the case of NSST, a substantial solenoid may be included to provide flexibility in operating conditions for a broad spectrum of physics investigations. Since only modest neutron fluence may be anticipated in NSST, multi-turn insulated TF coils are permitted. In the case of CTF, the requirements [9] of high fusion neutron fluence (up to 6 MW-yr/m^2 over >10 m^2 test area for the life of device) and availability (ultimately 30%) dictate that attractive ST-reactor-relevant features are mostly incorporated. These features include a single-turn center leg for the TF coils, elimination of the inboard solenoid magnet, and straight-line remote access for all fusion core components such as power blankets, divertors, and the TF center leg to minimize time for component replacement. Equally important will be to achieve the highest possible tritium breeding fraction as part of the testing program, to minimize the net consumption of tritium as the technology for tritium breeding is developed and tested. These features require emphasis in the physics basis for solenoid-free operation in NSTX research and the subsequent extension to NSST research.

The initial assessments show designs for these next-step ST experiments that are compact and as a result potentially cost-effective. Table 6.1 provides the key device parameters together with parameters for NSTX and a future first-of-a-kind ST DEMO concept, which is updated from the ST power plant concepts [7, 8] by requiring a net electrical power of 1000 MW based on forward-looking power conversion technologies. The NSST and CTF physics parameters are chosen to be sufficiently similar to enable a strong emphasis on the technological mission of component testing in CTF, based on a successful physics test on NSST. Whereas the CTF and DEMO assume solenoid-free operation, NSST as in the case of NSTX is expected to have the capability to test solenoid-free operation.

Table 6.1 Initial concept parameters for NSST and CTF, in contrast with NSTX and possible future ST DEMO

	NSTX	***NSST***	***CTF***	***DEMO***
Major radius, R_0 (m)	0.85	1.5	~1.2	~3.4
Aspect ratio, A	≥1.3	1.6	~1.5	~1.4
Nominal elongation, κ	2	2.5	~3	~3.2
Max triangularity, δ	0.8	0.6	~0.4	~0.5
Toroidal field at R_0, B_T (T)	0.3 – 0.6	1.1 – 2.6	~2.2	~1.8
Plasma current, I_p (MA)	~1	5 – 10	~11	~30
Nominal fusion power (MW)	–	0 – 100	~70 – 300	~3000
Solenoid induction flux (Wb)	0 – 0.6	0 – 15	0	0
Pulse length (s)	5 – 1	50 – 5	Steady state	Steady state

6.2 NSTX Contributions to NSST and CTF Concept Definition and Optimization, and to MFE Innovative Confinement Concept Portfolio

Since the application of auxiliary heating in FY2001 on NSTX and MAST, ST research has progressed rapidly to the level where cost-effective next ST steps (NSST and CTF) have become strong possibilities. The research planned for NSTX will be in a crucial position to shape the key physics features of these next-step experiments by addressing essential issues and to enable realistic extrapolation from the NSTX plasmas to these experiments. The key physics parameters of interest (figures of merit) to this research have been identified and are provided in Table 6.2.

The relative increments in these figures of merit from NSTX to NSST to CTF and to DEMO have been chosen to be most significant between NSTX and NSST. This ensures that new physics beyond NSTX will mostly be challenged fully for the first time in NSST research. Data and successful physics modeling at these steps are therefore expected to bring high confidence to the plasma conditions to be achieved in CTF and DEMO, as only more modest extension in physics beyond NSST would be required. This approach has been cohosen because the dominating goal to CTF must be technology testing, not physics research. The following provides a discussion of the relatively unique importance of the NSTX research in light of these parameter increments, organized according to the "Implementation Approaches" already identified by IPPA to address the 5-year ST objective of the IPPA Goal 2 for the MFES Program.

6.2.1 Solenoid-free initiation, ramp-up, and sustainment of plasma current – The understanding of solenoid-free initiation, ramp-up, and sustainment from NSTX is very important in enabling cost effective higher performance ST devices, as in the case of future Advanced Tokamak reactors. NSTX research in this topical area aims to establish the physics principles for MA-level plasmas as basis to design ST devices of larger size and current. As indicated in Table 6.2, the internal poloidal flux content ($\propto \ell_i R_0 I_p$) and the stored poloidal field energy ($\propto \mu_0 \ell_i R_0 I_p^2$) are the key figures of merit for this topic. These values in NSTX are expected to be about 0.5 Wb and 0.26 MJ, respectively, under high beta and confinement conditions. Large progress in these figures are anticipated for NSST for solenoid-free operation, by factors of 4 and 20, respectively, whereas further progress to CTF would be only require factors of 2 and 4, respectively. Final progress toward DEMO would require similar factors of 3 and 4, respectively. A key factor in these progressively reduced ratios of increase is the estimated reductions in ℓ_i, which result directly from the anticipated increases in plasma elongation, beta, and bootstrap fraction, as indicated in the preceding tables.

The science of solenoid-free operation on NSTX is relevant to the startup and sustainment research on the Spheromak and the RFP. Current initiation via CHI alone on HIT-II and NSTX with an applied toroidal field has been shown to require long timescales (up to 300 ms so far) compared to the similar startup techniques used on the Steady State Physics Experiment (SSPX) Spheromak in the absence of an applied

Table 6.2 Key physics parameters, estimated for NSST and CTF in comparison with those for NSTX and a future DEMO, as measures of physics extrapolation from NSTX to future ST devices

	NSTX	***NSST***	***CTF***	***DEMO***
Solenoid-free startup				
Internal inductance, ℓ_i	0.5	0.25 – 0.5	~0.25	~0.13
Internal poloidal flux $\propto \ell_i R_0 I_p$ (m-MA)	0.43	1.9 – 7.5	~3.6	~10
Poloidal field energy $\propto \ell_i R_0 I_p^2$ (m-MA^2)	0.43	9.4 – 75	~43	~150
Stable high β's				
Nominal Greenwald density, n_G	~0.5	~0.5	~0.5	~0.6
Beta normal, β_N	≤8	8 – 4	4 – 8	~8
Average toroidal beta, β_T	0.2 – 0.4	0.4 – 0.2	0.2 – 0.4	~0.5
Beta gradient, β_T' (/m)	0.25 – 0.5	0.26 – 0.13	0.13 – 0.26	~0.06
Alligned bootstrap current fraction, f_{BS}	0.7	0.8 – 0.2	0.5 – 0.8	~0.9
Resonant field error / B_T (%)	~ 0.1	~ 0.03	~ 0.01	< 0.01
Rational q values in plasma	≥ 1 – 2	≥ 1 – 3	> 2 – 3	> 3
Reduced turbulence & improved confinement				
Average temperature (keV)	~1	4 – 8	~10	~20
Average collisionality, ν*	0.2	0.03	~0.02	~0.02
Thermal ion minor radius, a/ρ_i	40	80 – 120	~100	~150
Ion confinement neoclassical factor, H_{Neoc}	~1	~1	~1	~1
Electron confinement H-mode factor, H_{98e}	~0.7	~1	~1	~1
Alfvén Mach number, $M_A = V_{Plasma}/V_{Alfvén}$	0.3	~0.3	~0.3	~0.1
Flow shearing rate (10^5/s)	1 – 10	1 – 10	1 – 10	0.3 – 3
Effective heating and sustainment				
$\omega_{pe}^2/\omega_{ce}^2$	50	50 – 20	~20	~25
Beam ion minor radius, a/ρ_{Beam}	5	15 – 22	~22	~34
Fusion α minor radius, a/ρ_α	–	N/A – 6	~6	~14
$V_{Beam}/V_{Alfvén}$	4	1	~1	~1
$V_\alpha/V_{Alfvén}$	–	N/A – 4	~4.5	~5
Dispersed plasma fluxes				
P/R (MW/m), $f_{rad} = 0.5$	8	13 – 20	20 – 40	87
Integrated attractive operations				
τ_{pulse}/τ_{skin}	~3	~10 – 0.5	→∞	→∞

toroidal field. The magnetic reconnection process anticipated for closure of the poloidal flux surface also appears to require more time on NSTX than on SSPX. Progress to be made on NSTX in modeling and understanding of this plasma initiation process will therefore be most beneficial to these concepts. On the other hand, the science of effective RF (such as EBW) heating of the electrons during current initiation will be of interest to both the ST and the RFP configurations. In the case of NSTX, an outboard field null created by only outboard poloidal field coils will benefit from this heating during inductive initiation of the plasma current using only these coils. Planned research on NSTX and MST in this area will therefore be mutually beneficial.

6.2.2 Stability limits – The plasma betas ($\beta_N \sim 4 – 8$, $\beta_N \sim 0.2 – 0.4$) relative to the MHD limits and plasma densities normalized to the Greenwald limit ($n_G \sim 0.5$) are expected to remain in the same range for these ST steps. This points to the unique relative importance of NSTX research in stability limits, the shaping-dependence of which is believed to be relatively well understood, but needs confirmation for the special features of ST physics, such as strong rotation, significant finite Larmor radius effects, high q_0, etc. The large Alfvén Mach number ($M_A = V_{Plasma}/V_{Alfvén}$ up to 0.3) in NSTX has already exhibited measurable effects on plasma equilibrium and stability. NSTX data and modeling on this topic must therefore be advanced fully to enable reliable projections to NSST and CTF. The physics that dictates the projections of M_A to larger ST devices is therefore a critical and open topic of NSTX research. On the other hand, the stability limit and the ELM bursts of the H-mode pedestal plasma in ST is expected to depend strongly on the plasma shaping (κ and δ) at the pedestal, as suggested by recent tokamak investigations. NSTX research will have the first opportunities to investigate the effects of strong toroidicity, in addition to flexible strong shaping of the plasma, on the pedestal stability limit, for direct applications to NSST and CTF. Further, the collisionality (ν^*) at the pedestal is expected to affect its stability by restraining the local bootstrap current. The effects of strong reduction in ν^* due to substantial decreases in bounce time of outboard trapped ions resulting from strong toroidicity will also be encountered in NSTX research.

As in the case of the Advanced Tokamak, the ST plasma is expected to require active feedback control of the Resistive Wall Mode (RWM) to reach the very high beta regime (the so-call “wall-stablized” regime) with the very high well-aligned bootstrap current fractions (~90% or higher) required for economic fusion power generation. Extensive research on this topic has been continuing on DIII-D, demonstrating strong interactions of the RWM with the static field errors, plasma rotation, active feedback of mode-resonant fields, and possibly plasma shaping. Initial results on NSTX have been encouraging in having entered the "wall-stabilized" regime for durations much greater than the resistive wall time by a combination of static error field reduction (to ~ 0.1%) and very large plasma rotation (M_A up to 0.3 due to tangential NBI). Assuming that the external field errors can be adequately eliminated in future devices (assuming to ~ 0.01% in CTF and < 0.01% in DEMO), the effects of active feedback control, plasma rotation, and plasma shaping (including strong toroidicity in ST) on RWM is therefore a key outstanding issues for NSTX and NSST research. Plasma rotation is expected to depend on plasma turbulence, transport, and momentum input, and needs to be addressed as part of turbulence and transport research (see the next subsection).

Another stability issue relates to the Neoclassical Tearing Modes (NTM), which is driven by loss of the local bootstrap current in a magnetic island of finite size in plasmas with high beta and bootstrap current fractions, particularly at q surfaces of low rational numbers in the plasma core (3/2, 2/1, 5/2, most notably). Such modes have been observed and stabilized in tokamaks via localized current drive using the ECCD. For over-dense plasmas such as ST, EBWCD is expected to be required for this stabilization when low q values (from 1 to 2.5) exist in the plasma, which defines one of the critically important roles of EBW research in the 5-year plan. Another key figure of merit for stabilization NTM in ST devices is the achievement of $q_{min} > 2.5$ operations. EBWCD at large (r/a) values (from 0.5 to 0.8) will therefore be a critically important capability to ensure the achievement of this condition. Such a requirement is consistent with the desired operation at high beta and bootstrap current fractions projected for CTF and DEMO. Stabilization and avoidance of NTM should therefore be fully tested in NSTX and NSST to establish the needed confidence for CTF and DEMO.

NTM of finite sizes can be triggered by other instabilities (sawteeth, ELM's, or fast ion driven bursts of Alfvén modes) as observed in high temperature tokamak plasmas. Another research of interest to NSTX and NSST is therefore the possible avoidance of these trigger mechanisms, by operating the ST plasmas with high q0 and q_{min} (> 2.5) and away from large ELM instabilities. Another important issue relates to the standard tearing modes at rational-q surfaces that can be driven unstable as β approaches the ideal stability limits. This research will be encountered in NSTX as the plasma β is pushed toward the "wall-stabilized" limits, as discussed earlier in this subsection.

Energetic ions, such as those introduced by neutral beam injection heating and by fusion burn (3.5-MeV α's), can also drive MHD instabilities in the Alfvén and ion cyclotron frequency ranges. A key figure of merit in the properties of these instabilities, which are of high importance in the burning plasma physics studies, is the ratio of the fast ion velocity to the plasma Alfvén velocity ($V_{Fast}/V_{Alfvén}$). Because of the relatively modest magnetic field and very high beta expected of the future ST devices, $V_\alpha/V_{Alfvén}$ is expected to be large (~4-5) in the plasma core, which is similar to $V_{Beam}/V_{Alfvén}$ in NSTX. New phenomena related to these instabilities (such as the Compressional Alfvén Eigenmodes in broad spectrum bands at a fraction of the ion cyclotron frequency) have already been observed in NSTX. Investigation of the fast ion driven instabilities in NSTX, and later verification of the same in NSST, will establish a solid scientific basis for controlling and avoiding such instabilities in CTF and DEMO burning plasmas.

Maintaining macroscopic stability at very high β is a topic of common interest to the Spheromak, the RFP, the FRC, and the ST. In the paramagnetic plasma regime ($\beta_p < 1$), the investigations of tearing modes in Spheromak and RFP plasmas are of key importance, and are also of high interest to NSTX research. On the other hand, NSTX research has recently produced plasmas that entered the diamagnetic regime ($\beta_p > 1$), the limiting case of which is the FRC plasma where the toroidal field is almost completely excluded from the core. The stability properties of a strong magnetic well (~30% from the outboard plasma edge in NSTX as projected for plasmas with $\beta_T \sim 40\%$ and $\beta_N \sim 8$) is therefore one of the key research topics of shared interest with the FRC.

6.2.3 Turbulence and energy and momentum transport – The key parameters of interest to this broad topical area are expected to include ν^*, a/ρ_i, β_T, q, q′, and the flow shearing rate. The greatest increments in the former two of these parameters (with factors of 5, 2 – 3, respectively) are estimated to occur between NSTX and NSST. In addition, an in-depth understanding of plasma viscosity and momentum transport, to determine the plasma flow shearing rate, acquires a critical importance in view of the large uncertainties in this topic at present, and its leverage on plasma turbulence and stability limits. Recent measurements and modeling of the NSTX plasma suggested the strong possibility that the Ion Temperature Gradient (ITG) driven modes are stabilized by a combination of large flow shearing rates and large β' obtained in NSTX under tangential neutral beam heating. NSTX results have shown neoclassical ion confinement behavior or better over a broad radial zone, as long as MHD quiescence is maintained. It is therefore highly attractive to extend this confinement condition of neoclassical ions to NSST and CTF, based on understanding of the physics related to the above identified physics parameters.

However, the electron energy confinement times in NSTX have so far been relatively poor under sustained H-mode conditions where the ions behave near the neoclassical limit. The anomalously large electron energy loss mechanisms remain an important confinement topic in toroidal research. NSTX, owing to its access to very high beta (order unity local betas having already been obtained) and likely strong suppression of ion electrostatic turbulence, provides an extended range of conditions (in the parameters identified above) in which to investigate the remaining electron and electromagnetic turbulence and transport mechanisms. A critical question for NSTX to resolve will be whether the electron energy confinement in the ST would have a scaling close to the tokamak H-mode scaling, while maintaining approximately neoclassical ion energy confinement. Such confinement behaviors would lead to highly attractive burning plasma conditions in NSST, CTF and DEMO, as indicated in Table 6.1.

When operating in the absence of substantial global MHD instabilities, NSTX plasmas heated by intense tangential neutral beam injection (NBI) have exhibited large toroidal plasma flows (up to 250 km/s or $M_A = V_{plasma}/V_{Alfvén} \sim 0.3$) and flow shearing rates (up to a large fraction of 10^6 /s). These conditions are currently believed to be required for the excellent ion confinement observed so far in the sustained H-mode plasmas on NSTX. Transport estimates have suggested that the momentum confinement times may be larger than the neoclassical ion energy confinements under such conditions. It is therefore assumed that large tangential NBI powers would be applied to NSST and CTF, to ensure similarly favorable ion and momentum confinement conditions. However, the requirement of high-Q operations in a future ST DEMO will diminish the momentum input from NBI relative to the large anticipated increases in the plasma moment of inertia. Recent tokamak experiments large RF heating without a large momentum input has shown that a toroidal rotation can still exist, with velocities roughly in the order of the diamagnetic flow. For a future ST DEMO with order unity plasma beta, the diamagnetic velocities can be about one order of magnitude below the plasma Alfvén velocities. In view of this possibility, the issues of momentum transport and the minimum plasma rotation and flow shear required for achieving high confinement and high beta plasma are therefore of high importance to the long-term viability of fusion power based on the ST configuration.

The present dominance of electron turbulence and energy loss from the plasma core is shared by plasmas in SSPX, MST, and NSTX. These experiments are expected also to share an increased role of electromagnetic fluctuations (due to increased plasma β and compressibility) in the electron turbulence. The research and diagnostic tools (such as via mm μwave scattering of high-k fluctuations) planned for NSTX in FY04 can therefore be utilized in the electron energy confinement studies on SSPX and MST.

6.2.4 Energetic ion confinement and driven MHD instabilities – The key parameters of interest in this topical area are expected to be the minor radius normalized by the gyroradius of the fast ions (a/ρ_{fast}), and the ratio of fast ion speed over the plasma Alfvén velocity ($V_{Beam}/V_{Alfvén}$). Because of the relatively modest field and substantial Greenwald densities achieved in the ST plasmas, 80-kV beam deuterium ions move at strongly supra-Alfvén velocities ($V_{Beam}/V_{Alfvén} \sim 4$). Broad spectra of Compressional Alfvén Eigenmodes (CAE's) near the ion cyclotron frequencies, and other Alfvén instabilities at lower frequencies, are often observed on NSTX to influence the confinement of the fast ions, and have been theoretically attributed to the presence of these supra-Alfvénic fast ions. The NSTX plasmas already provide unusually small a/ρ_{fast}, as indicated in Table 6.2. To increase confidence in extrapolation from the NSTX results, it is valuable that these parameters for neutral beam ions in NSTX (5 and 4, respectively) are similar to those for fusion α particles in NSST (inductive pulsed) and CTF (non-inductive sustained). Data and physics modeling for the fast ion effects in NSTX are thus applicable to the same effects in NSST and CTF. The effect of different anisotropy in the fast ion velocity distribution between beam ions and fusion α's is a remaining topic of importance in NSTX research, to ensure reliable extrapolations to NSST.

Fusion α particle velocities in possible future D-T experiments based on the Spheromak, the RFP, and the FRC, because of further reductions in magnetic field, are expected to be even more strongly supra-Alfvénic than those in future ST D-T experiments. NSTX studies in this topic will therefore be of value to neutral beam heating experiments on the SSPX and the MST, as well to possible future burning plasma tests in these innovative configurations.

6.2.5 RF heating and current drive in over-dense plasmas – A key parameter that determines the nature of HHFW and EBW interactions with the ST plasma is the dielectric constant $\varepsilon = \omega_{pe}^2/\omega_{ce}^2$, which are in the high range of 50 – 20 for all ST devices provided in Table 6.2. NSTX research to investigate the effects of this range of ε on HHFW- and EBW-plasma interactions will provide critical data for application to RF heating and current drive in NSST and CTF. A strong source in EBW power can further be applied to investigate RF-only initiation of the plasma current in conjunction with a vertical field provided by the outboard poloidal field coils (to heat and initiate an inboard plasma at small radius via bootstrap current), and to ramp up the plasma current by adding RF current drive following initiation. This is therefore a unique opportunity for ST research at the MA level, relative to the growing success in current initiation in tokamaks using combinations of ECCD and LHCD. A strong database in this area will enable critical optimization and cost reduction in the NSST and CTF design concepts, at the multiple-MA to 10-MA level.

Plasmas in SSPX, MST, FRC, and Pegasus are expected to be even more over-dense ($\varepsilon = 100$ or higher). MST and Pegasus already plan to conduct EBW heating and current drive tests in the near future. Research on this topic on NSTX will therefore provide valuable information to this research of common interest on these ICC experiments.

6.2.6 Power and particle handling – As indicated in Table 6.2, the magnitude for the heat flux scale factor, P/R, is estimated to increase in steps of about factor of 2 from NSTX to NSST to CTF and finally to an ST DEMO. The large values of P/R for future ST devices are anticipated to be a substantive challenge, as in the case of all compact fusion devices of high performance. NSTX investigation and physics modeling in this topical area will be extended to definitive studies of the effects of the strongly increased magnetic mirror ratio of the SOL (due to the strong toroidicity) in the case of diverted and inboard limited plasmas with varied mirror ratios. Technology innovations in plasma facing materials, such as liquid lithium surfaces, may be required to develop optimal solutions in combination with the physics solutions. Progress in this area will ultimately determine the practical sizes and fusion neutron wall fluxes achievable in future ST devices, and indeed determine the economic attractiveness of fusion power based on the ST configuration. Present ST research using large heating power and varied plasma edge configurations will provide the critical information needed to take the step to the next level of ST plasmas suggested for NSST, and in turn for CTF. The progress to be made in this topic on NSTX is expected to have a large impact on the future prospects of all ICC configurations of compact size and potentially high performance, including the Spheromak, the RFP, and the RFC.

6.2.7 Integrated high performance operation for pulse lengths far beyond the plasma skin time – It is critically important for NSTX to test plasma properties for durations much larger than (i.e., 2 – 3 times) the plasma skin time. Such plasmas need to have simultaneously high beta, good confinement, high fractions of non-inductive currents (bootstrap current combined with non-inductively driven currents from neutral beams and rf power), progressively reduced reliance on inductive current drive, and successful power and particle handling. Measurement and physics modeling of such plasmas in NSTX will be incorporated into simulation codes to identify viable techniques, critical challenges, and plasma operation scenarios in extending the duration of high-performance ST plasmas. Successful operating scenarios can then be extended to even larger multiples of the skin time (to about 10 or higher in NSST) at multiple-MA level. These results will provide the foundation required for extending the techniques to steady state operation in CTF.

In view of the preceding discussion, NSTX research at the Proof of Principle level on NSTX will therefore likely meet the largest ST physics challenges in establishing the physics database for extrapolation to future ST experimental devices, such as the NSST. Successful investigation in NSTX of the key physics issues identified here, and subsequent verification of these issues in NSST, would establish a solid scientific basis for making relatively moderate extensions to CTF and DEMO. The key physics issues for a cost-effective development of magnetic fusion energy using the ST configuration are identified here and incorporated into the NSTX 5-year plan.

6.3 NSTX Contributions to Better Physics Understanding and Optimal Utilization of Burning Plasma Experiment

The International Tokamak Physics Activity (ITPA) was established in 2001 to coordinate and enhance the development of the physics basis for tokamak burning plasmas. This activity continues the tokamak physics R&D activities that were carried out on an international level for many years to broaden the physics basis in support of the ITER design, and bring substantial benefits to general tokamak research and all fusion programs worldwide. The activity aims to provide validated experimental data, analyzed experimental results, and theoretical models and simulation results to advance the understanding of toroidal fusion plasma physics. Such understanding will provide a solid basis to study fusion performance of and identify and resolve issues for the tokamak burning plasmas, such as ITER. The ITPA represents the most comprehensive attack on the key physics issues anticipated of the tokamak burning plasma. A set of ITPA high priority issues and research topics have been identified recently to focus the ITPA efforts.

NSTX research has recently succeeded in extending the fusion plasma physics to regimes of substantially higher β and stronger toroidicity. NSTX data and analysis can therefore provide valuable information in resolving some of the high priority issues identified by the ITPA, particularly those involving substantial uncertainties relating to β and aspect ratio dependence. The following provides a preliminary discussion of such contributions from NSTX. More topics of interest to the ITPA are likely to be identified as further progress is made in NSTX research.

6.3.1 β-scaling of confinement in ELMy H-modes – A recent β-scan at fixed a/ρ_i and ν^* on DIII-D has shown that confinement can be independent of β. This is not consistent with the ITER confinement scaling for the ELMy H-mode plasmas, which contains a strong inverse β dependence ($\tau_E \propto 1/\beta$). This uncertainty in confinement scaling is expected to have a large effect on DEMO concepts that assume higher β. A β-scan can be carried out on NSTX at $B_T = 0.6$ T and at a/ρ_i and ν^* values close to those of similarly shaped DIII-D plasmas at the same field. This will in effect extend the range of β values by about a factor of 2 – 3 and help resolve this important issue for the ITER confinement scaling expressions, reducing the uncertainties in making confinement projections to a future tokamak DEMO.

6.3.2 β-limits with internal transport barrier (ITB) operation – The achievable β-limit has been a key issue in developing steady state operation scenarios for ITER plasmas with a strong ITB and the associated pressure profiles. It has been experimentally shown that the β-limit varies strongly with the types of ITB and the q-profile. This dependence needs to be resolved for reliable predictions of the sustained plasma operation needed in the second (technology testing) phase of ITER. NSTX has recently achieved pulse lengths beyond the skin time by a combination of H-mode and q-profile variations, achieving β_T values as high as 20%. Extended ITB-like zones (to ~30% of the minor radius) have been measured in ion temperature and plasma flow shear profiles in these plasmas. The correlation of these ITB-like zones with the q profiles will be measured and studied in FY04. The pressure profile can in principle be altered over a range in such plasmas by variations in the edge fueling rates and the sources and power levels of the injected neutral beams. The dependence of the achievable β_T on the variations of pressure- and q-profiles can be measured and analyzed to help increase the confidence of the tokamak beta limits to be applied to the ITER burning plasmas.

6.3.3 Impact of ELMs on the pedestal and SOL, and effect of aspect ratio – The scaling for the pedestal width is an urgent issue for predicting the ITER H-mode conditions. Recent studies of ELMs in tokamaks show strong dependence on plasma triangularity and overall magnetic configuration of the plasma edge-SOL region. Theoretical studies indicate that edge peeling modes may be critical in determining ELM stability. It is therefore anticipated that the ELM behavior and limits will show substantial dependence in the aspect ratio and plasma shapaing. Data from NSTX H-mode plasmas concerning the impact of ELMs on the pedestal and SOL will therefore contribute to the fundamental understanding of ELM physics and to resolving this uncertainty in pedestal width and help develop techniques to improve the ITER plasma conditions in ELMy H-mode.

6.3.4 Aspect ratio comparison of Neoclassical Tearing Modes (NTM) – Recent comparison experiments for the NTM properties between MAST and DIII-D with similar plasma shape and current helped confirm important aspects of the NTM theory in the low aspect ratio regime. However, detailed analysis indicates that large increases in plasma resistivity need to be assumed in order to match the quantitative behavior between theory and measurement. Further, NTM behavior is theoretically predicted to also depend on bootstrap current, β_p, and the local ion drift velocities, which are expected to have divergent variations in aspect ratio dependence. It is suggested that β ramp-up and ramp-down operation following sawtooth-seeded NTM occurs would reveal important dependence of NTM on the seeding process as well as on the above mentioned plasma parameters. High performance NSTX and MAST plasmas have been observed to possess larger ρ_i/a than similar scale tokamaks by about a factor of 5, higher q_{edge} by about a factor of 2, and a potential wider variations in the core q profiles. Using the same plasma shape and current, it is possible for NSTX to provide NTM measurements at a very different range of values in plasma neoclassical resistivity, bootstrap current fraction, ρ_i/a, β_p/A, q profiles, and the ion drift velocities. These tests will therefore provide important new opportunities to test the underlying theories of NTM and help resolve the remaining uncertainties in making NTM predictions for the ITER sustained operation at high β_p/A values (approaching 1).

By making detailed comparison with otherwise similar tokamak plasmas, NSTX research can not only make substantial contributions to improving the tokamak burning plasma experiment, but also receive substantial benefits in advancing the understanding of the ST plasmas. A substantial number of the NSTX research team members have begun an active participation in all the ITPA topical group activities.

6.4 Discussion

With the ability to produce high performance plasmas over a substantially widened range in important physics parameters (see Table 6.2), NSTX research during the next 5-years plans to make critically valuable contributions to the progress toward practical fusion energy recommended by the recent FESAC panel on development path. In this chapter, we have shown how the planned research on NSTX will aim to produce the database needed to enable a largest extrapolation of key physics parameters to the next PE-level ST experiment, NSST (with $R_0 \sim 1.5$ m). Required extrapolations beyond NSST to an ST based CTF ($R_0 \sim 1.2$ m), and in turn to an ST based DEMO ($R_0 \sim 3.4$ m), are estimated to be much more modest in magnitude as a result.

Because NSTX share similar values of many of the key physics parameters (see Table 6.2) with the ICC experiments, SSPX, MST and FRC, the physics to be developed in the NSTX program will therefore make important timely contributions to the development of these ICC configurations toward a future PE level experiment, and in some cases, to the possible development of a ICC based burning plasma experiment.

Because NSTX can extend high-q toroidal plasma β and toroidicity to substantially higher values, comparison experiments with tokamak plasmas of similar conditions can also contribute to better physics understanding, so more optimal utilization of the burning plasma experiment ITER can be achieved.

These planned and potential contributions make the present proposed 5-year research plan on NSTX a highly attractive investment in fusion energy science.

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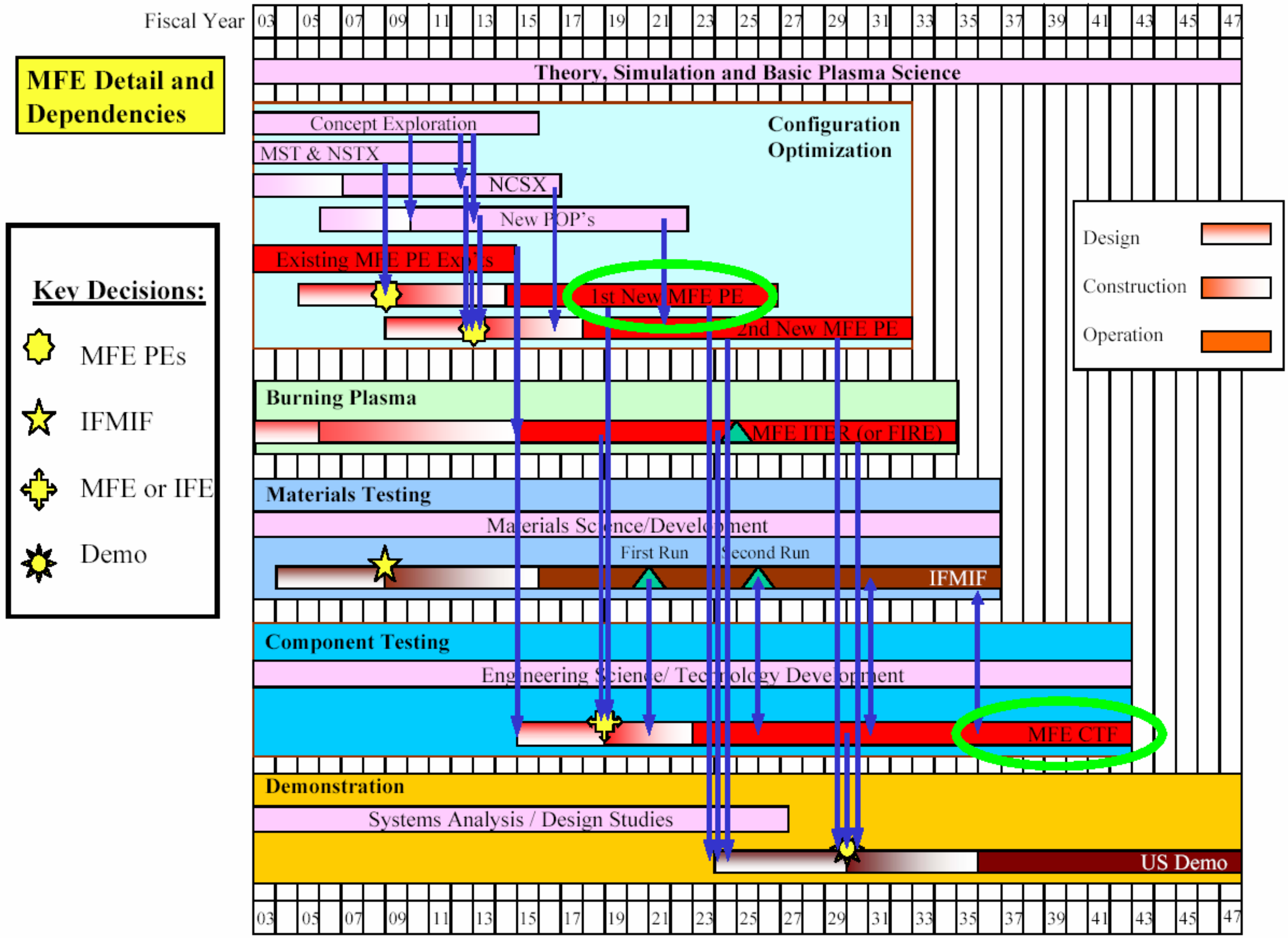

Figure 6.1: Illustrative programs and major facilities that comprise the updated fusion development plan formulated by the FESAC Panel on Development Path. NSTX research can make potentially critical contributions to the MFE Configuration Optimization as a possible new PE-level experiment, followed by a compact Energy Development level ST device for Component Testing (circled in the figure).

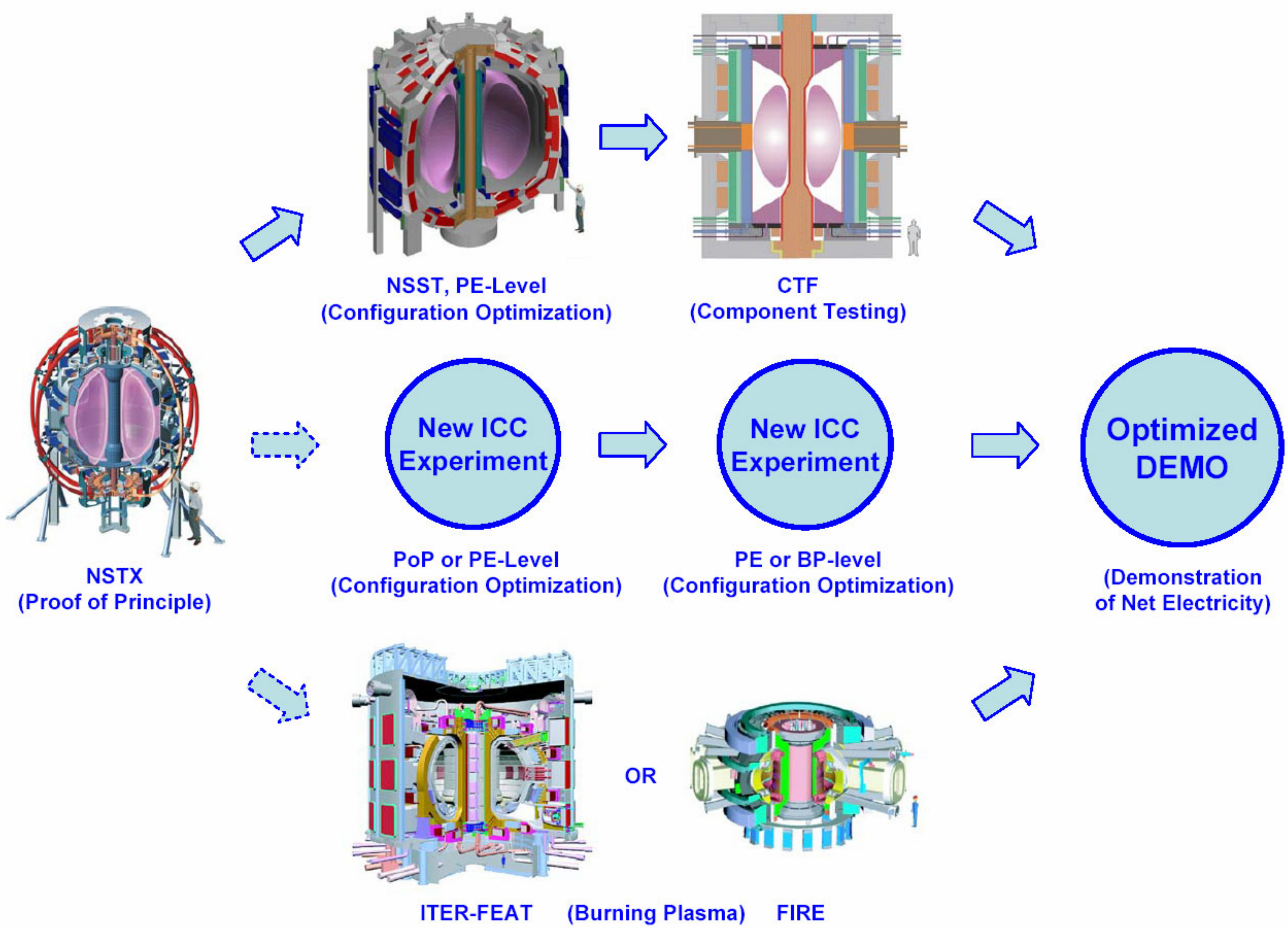

Figure 6.2: NSTX research will provide physics database crucial to the optimal designs of NSST and CTF; contribute to advancing the development of Innovative Confinement Concepts Spheromak, Reversed Field Pinch, and Field Reversed Configuration; and contribute to better physics understanding and optimal utilization of burning plasma experiment ITER or FIRE.