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Recent Progress on Spherical Torus Research

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Recent Progress on Spherical Torus Research*

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Abstract: The spherical torus or spherical tokamak (ST) is a member of the tokamak family with its aspect ratio ($A = R_0/a$) reduced to $A \sim 1.5$, well below the normal tokamak operating range of $A \geq 2.5$. As the aspect ratio is reduced, the ideal tokamak beta β (ratio of plasma to magnetic pressure) stability limit increases rapidly, approximately as $\beta \sim 1/A$. The plasma current it can sustain for a given edge safety factor q_{95} also increases rapidly. Because of the above, as well as the natural elongation κ , which makes its plasma shape appear spherical, the ST configuration can yield exceptionally high tokamak performance in a compact geometry. Due to its compactness and high performance, the ST configuration has various near term applications, including a compact fusion neutron source with low tritium consumption, in addition to its longer term goal of attractive fusion energy power source. Since the start of the two mega-ampere class ST facilities in 2000, National Spherical Torus Experiment (NSTX) in the US and Mega Ampere Spherical Tokamak (MAST) in UK, active ST research has been conducted worldwide. More than sixteen ST research facilities operating during this period have achieved remarkable advances in all of fusion science areas, involving fundamental fusion energy science as well as innovation. These results suggest exciting future prospects for ST research both near term and longer term. The present paper reviews the scientific progress made by the worldwide ST research community during this new mega-ampere-ST era.

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I. Introduction

A. Overview - At the dawn of the 21st century, two magnetic fusion research laboratories on both sides of Atlantic started plasma operation nearly simultaneously on newly completed mega-ampere-class spherical tokamak (ST) facilities after multi-year design and construction. Since the National Spherical Torus Experiment (NSTX) [1, 2] at PPPL, USA, and Mega Ampere Spherical Tokamak (MAST) [3, 4] at Culham Laboratory, UK began operating in 2000, very active ST research has been conducted worldwide. In Fig. 1.1,

schematics of the MA-class ST facilities NSTX and MAST are shown. Both devices are similar in size and operating parameters, standing about three stories high, but with complementary engineering designs. More than sixteen ST research facilities

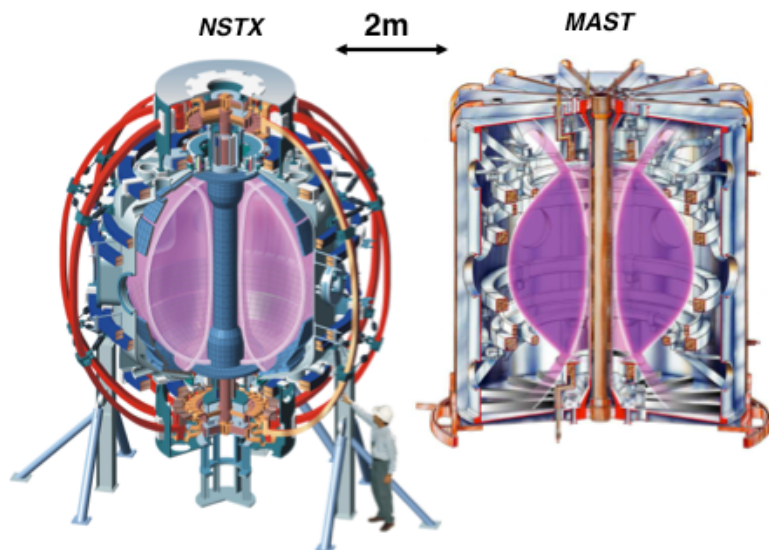


Fig. 1.1. MA-Class STs : NSTX and MAST devices.

operated during this period [1 - 19]. After over a decade of ST research at the MA level, there are a large number of significant ST results which are indeed ground breaking and unique to STs, while reaffirming many common physics features with conventional tokamaks. The present paper attempts to review these results, emphasizing those important experimental results which are unique to the STs.

As a member of the tokamak family [20], the ST conducts research that contributes to

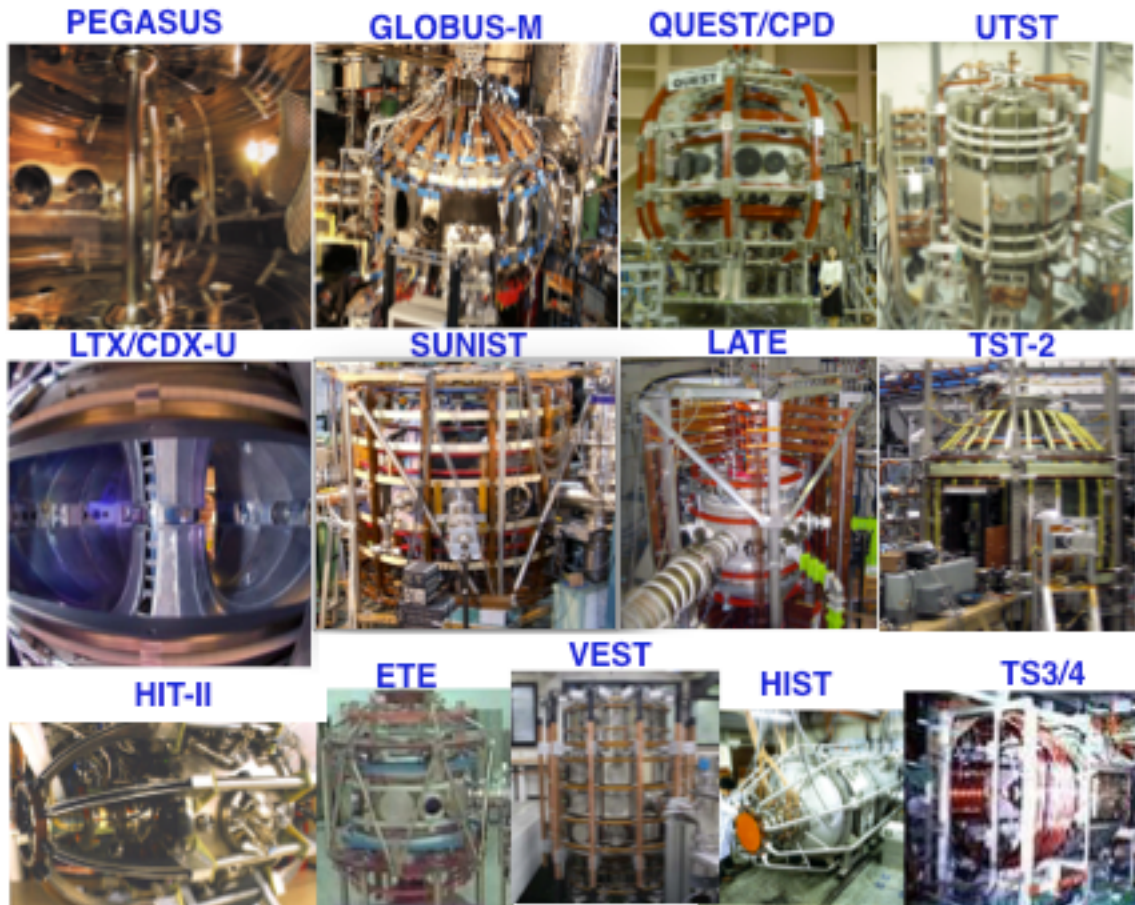


Figure 1.2. Medium to smaller ST facilities around the world.

advancing conventional tokamaks such as ITER [21, 22] by providing data that extends into a unique plasma and device parameter space. Since ITER represents a significant extrapolation from the present day tokamaks, ST research provides a different set of tokamak-related data for the improvement of predictive capabilities by providing leverage over a wide range of parameter space. The paper will attempt to motivate ST research in terms of contributions to the ST fusion energy development path, but the

relevance to conventional tokamak research and ITER will be also described. Whenever appropriate, the unique ST contributions to the advancement of toroidal plasma physics research in general will be pointed out.

In Sec. B, we describe the sixteen ST facilities operated worldwide since 2000. In Fig. 1.2, medium to smaller size ST devices around the world are shown. In Sec. C, the new 2 MA-class ST facilities NSTX-U [23] and MAST-U [24], presently under construction, are described. In Sec. II, the ST fusion development paths are described. In addition to ST fusion power plants, because of the compactness of the ST configuration, a number of possible near term applications including a high power heat flux facility, fusion neutron science facility (FNSF) or component test facility (CTF), and a ST pilot plant study are described. The ST configuration indeed appears to offer a very wide spectrum of possible neutron science facilities depending on the neutron source requirements and facility scale. One variant of an ST neutron facility, for example, projects a very compact ($R \sim 0.5$ m) neutron source working at a fusion power level of only a few MW. In Sections III to IX, progress in various topical scientific areas needed for the ST development path is described.

In Sec. III, we first describe macrostability research to understand and access high plasma β regimes, as this is one of the areas of unique ST contributions to fusion research. Here an important research emphasis is to access simultaneously the high β_T and high bootstrap current fraction f_{BS} regime for attractive high performance steady-state ST operations. As shown in Fig. 1.3, the present ST data base spans a large plasma beta space extending well beyond that obtained in tokamak research. An exciting new experimental

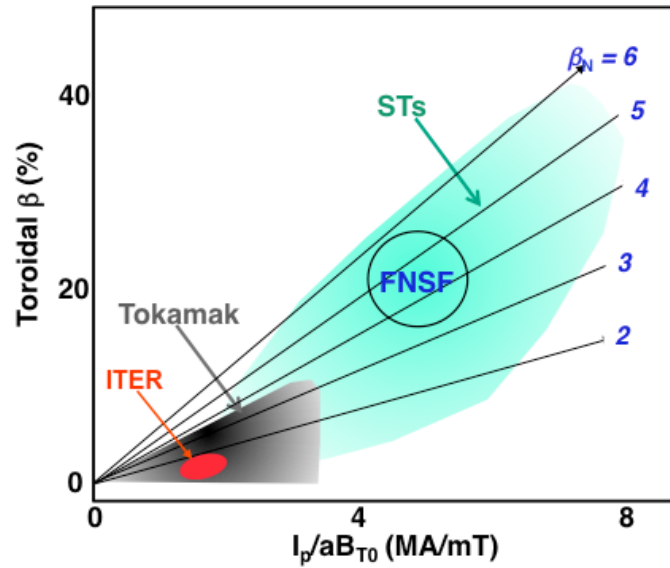


Figure 1.3. Experimentally achieved plasma beta values for STs ($R/a \leq 2.0$) and conventional tokamak ($R/a \geq 2.5$). The ITER and FNSF beta regimes are indicated.

observation which is unique to the ST research is the accessibility to high beta regime some degree of stability as shown in Fig. 1.4. The indication of stability at high beta particularly in the low I_i is evident in the observed plasma disruptivity plot as shown in Fig. 1.4(a) [25]. To gain more insight into this observed stability at high beta, in Fig. 1.4(b), the observed resistive wall mode (RWM) amplitude is plotted as a function of β_N/I_i [26]. Consistent with the previous observations in tokmaks, as the β_N/I_i increases, the plasma tends to be more unstable as indicated by the RWM amplitude increase with β_N/I_i . However, as shown

in the figure, after reaching a certain value of $\beta_N/I_i \sim 10$, the RWM amplitude actually decreases with higher β_N/I_i , an indication of enhanced plasma stability. This observation is attributed to a kinetic stabilization effect as described in more detail in Sec. III. This is an

encouraging result as a future fusion power plant is expected to run at as high values of β_N/I_i .

In Sec. IV, we describe the central solenoid-free ST start-up and ramp-up research due to its particular importance to the ST development path. No significant central solenoid is envisioned in most of the ST reactor schemes, as the compact design essentially precludes conventional OH coils. The elimination of the OH solenoid would be attractive for any tokamak reactor design as well. Naturally, this is an area of significant research emphasis within the world ST research community, as many ST facilities are dedicated to this research topic. There are a number of solenoid-free ST start-up concepts already demonstrated as shown in Fig. 1.5 [27 – 29]. The RF heating and current drive based concepts are perhaps the most widely pursued, utilizing electron cyclotron heating (ECH), electron Bernstein waves (EBW), and lower hybrid (LH) waves. The helicity

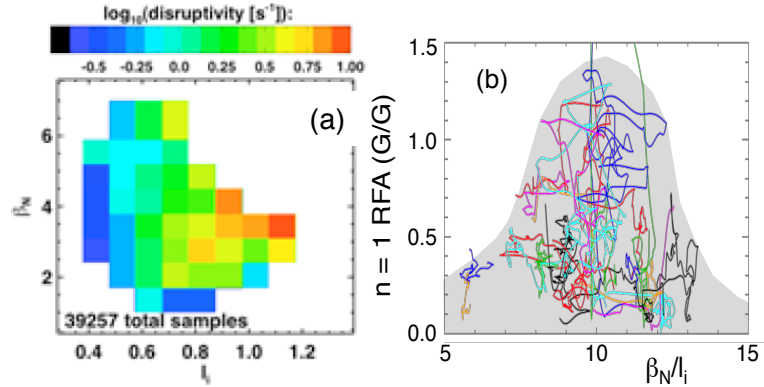


FIG. 1.4. ST MHD stability at high beta. (a) ST plasma disruptivity in β_N and I_i parameter space. (b) $n = 1$ resonant field amplification during high β discharges using active MHD spectroscopy, indicating improved stability at high β_N/I_i .

injection method uses electrical discharges to introduce plasma current. The merging-compression concept creates smaller current rings at the top and bottom, and then merges them at the mid-plane to form a larger current tokamak. However, non-inductive current ramp-up from the start-up plasma to the high current operating range is one research area requiring considerable development in the future.

In Sec. V, the plasma global confinement, local transport, and associated turbulence issues are addressed. High

energy confinement is essential to access compact high performance ST reactor regimes, and is generally desirable for fusion reactor economics. The observed ST H-mode

plasma ion transport is quite close to the neo-classical level in NSTX as shown in Fig. 1.6 [30]. This has been expected for the STs due to the high levels of ExB shear flow stabilization in the ST geometry. Another intriguing confinement

result is the improvement of global energy confinement time with reduced collisionality

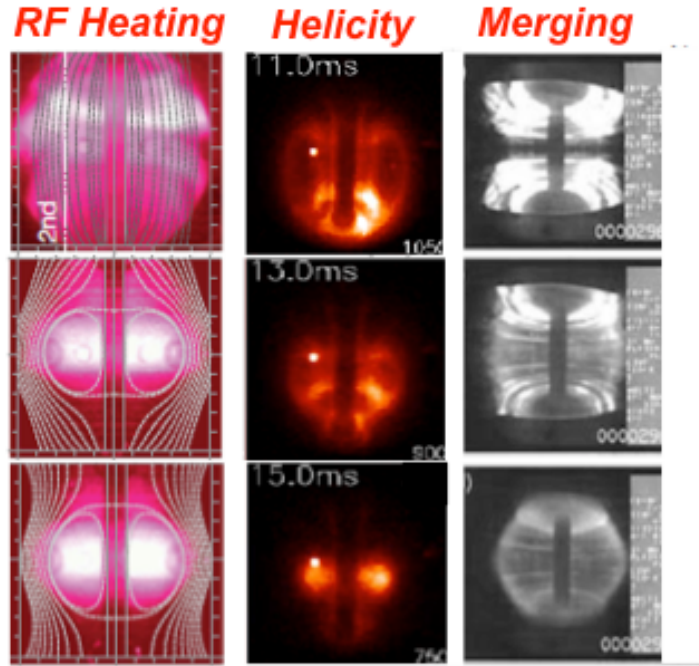


Fig. 1.5. Solenoid-free ST start-up concepts based on RF heating and current drive (LATE), helicity injection (NSTX), and merging (MAST) methods. Pictures are initial open field line phase at the top and the final closed flux ST formation phase at the bottom.

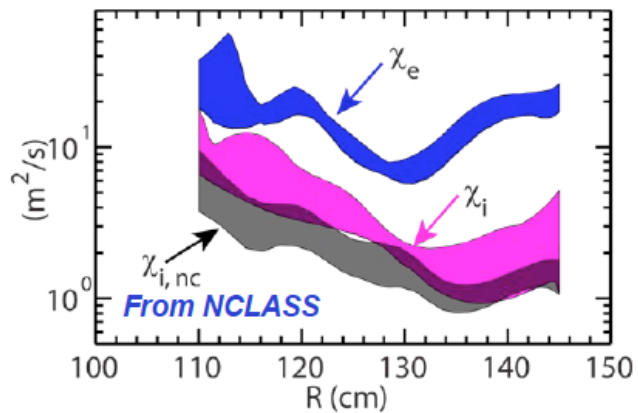


Fig. 1.6. Electron and ion thermal energy diffusivity profiles in NSTX H-mode plasmas. The ion neoclassical (NCLASS) value is shown .

have reached the high divertor peak heat flux of ITER ($\sim 10 \text{ MW/m}^2$), but NSTX-U/MAST-U can achieve DEMO/FNSF level heat flux ($\sim 40 \text{ MW/m}^2$). In terms of divertor heat mitigation, a snow-flake divertor configuration (shown in Fig. 1.8) was demonstrated [32]. This resulted in a significant divertor flux expansion of ~ 50 , peak heat flux reduction of $\sim \times 3$, and $\sim \times 10$ reduction with radiative snowflake divertor. Another innovative divertor configuration that lead to high flux expansion, the super-X configuration, is planned on MAST-U. Utilization of lithium as an alternative plasma facing material has been also actively pursued. In Sec. VIII, energetic particle physics research is described, with an emphasis on the alpha particle relevant physics where STs can explore a very wide parameter space. This includes the so-called super Alfvénic regimes, where the energetic particle velocity exceeds the local Alfvén velocity as illustrated in Fig. 1.9 [33]. The super Alfvénic condition is expected for fusion alphas in high performance burning plasmas as in ITER and ST/tokamak reactors.

In Sec. IX, radio frequency (RF) heating and current drive physics is covered since such systems are expected to be needed for plasma heating and current drive for attractive fusion reactors. While neutral beam injection is the method of choice for heating and

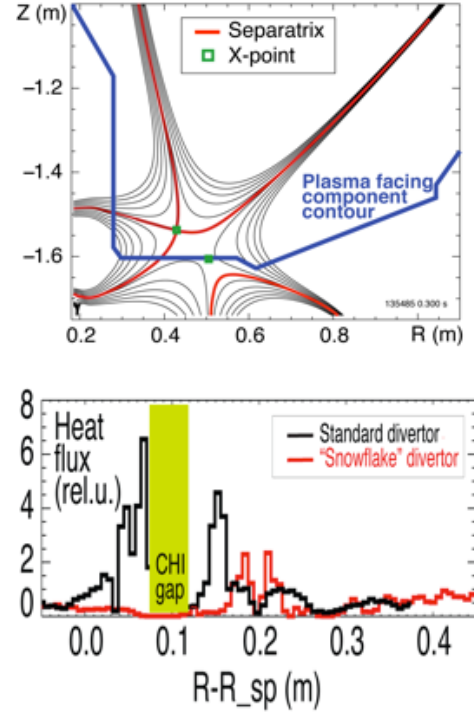


Fig. 1.8. Snow-flake divertor configuration (above) and measured divertor flux comparison with standard configuration.

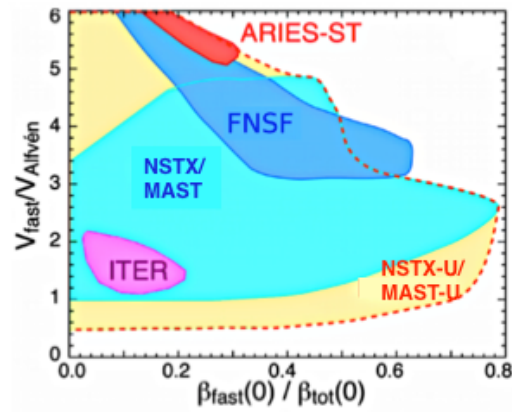


Fig. 1.9. Energetic particle parameter space with normalized velocity and beta values.

current drive for present day experiments due to its simpler physics, the RF systems may offer engineering / technology advantages in the fusion reactor environment. We will discuss high harmonic fast wave (HHFW) which directly heats electrons, compared to ICRF heating in tokamaks which tends to heat ions. Another

wave promising for ST is the electron Bernstein Wave (EBW), which can access the high beta high dielectric

plasmas of STs compared to the electron cyclotron heating (ECH) commonly used in tokamaks. We will then discuss the integrated scenario development in Sec. X, where the operating scenario optimization research is described. The integrated scenario synthesized all the knowledge learned from the individual science topical areas and applies it to formulate and test the best possible operating scenarios for future applications. Naturally, the main aim is to come up with scenarios for sustained high performance non-inductive operations as is the case for advanced tokamaks. For a tokamak system, it is highly important to increase the amount of bootstrap current as much as practical, since that would reduce the need for direct current drive and its associated recirculating power requirements and cost. In Fig. 1.10, an example of the rapid progress made in this area is shown, where the $n=1$ resistive wall mode instability which normally causes plasmas to terminate (particularly at low plasma rotation) was stabilized using passive and active feedback [34]. This resulted in the sustainment of a high β_N plasma even at low plasma rotation in NSTX. The Conclusions and Discussions are in Sec. XI. As supplemental information, in Appendix A, the basic properties of the ST/tokamak configuration is given along with the definitions of key ST/tokamak parameters. In Appendix B, a brief comment on ST concept development in the context of the tokamak research is given. In Appendix C, the ST's relationship with other confinement concept is briefly described. It should be noted that the present paper

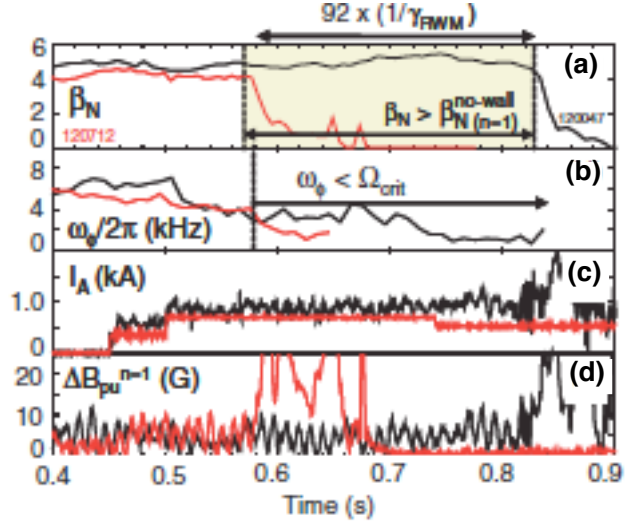


Fig. 1.10. RWM stabilization at low rotation. (a) β_N evolution, (b) rotation, (c) RWM/EF coil current, and (d) $n=1$ mode amplitude.

mainly utilizes material from peer-reviewed journals but wherever appropriate, major conference-related publications will be included. This particularly holds for those describing the most recent results, since ST research is very much a rapidly evolving area in the magnetic fusion. Since the present paper is an ST review article, the references are mainly given for ST research publications. We should note, however, that there is a strong synergy between on-going ST research and other configurations, particularly the very active tokamak research conducted worldwide. The relevant research on tokamaks and other configurations can generally be found in the references in the ST publications cited in this review paper. Also since there are so many ST papers, only the most recent publications on a given topic are referenced. The main purpose of this review paper is to give a comprehensive description of recent progress in ST research, rather than give a historical perspective on the research. For those readers interested in the historical development of the ST, relevant references are usually given within the publications cited in this paper.

B. Description of Present ST Facilities - There are more than sixteen ST experimental facilities in operation worldwide since 2000. Most of the on-going ST facilities were built from the previous ST / tokamak / spheromak facilities. In this section, we discuss the ST facilities operated during the period 2000 to the present (2013). This includes the Mega Ampere ST facilities depicted in Fig. 1.1, and a number of medium to smaller scale ST facilities shown in Fig. 1.2. Those ST facilities operated before 2000 including the START device are covered in the previous review/overview papers [35, 36]. It also should be noted that the most of the ST facilities are located at universities with a strong student education mission.

NSTX and MAST - Mega Ampere-Class STs: There are two mega-ampere class ST facilities, both of which started research operation in 2000, the National Spherical Torus Experiment (NSTX) in the U.S. [1, 2] and the Mega-Ampere Spherical Tokamak (MAST) in the U.K [3, 4]. Both NSTX and MAST are equipped with strong auxiliary heating and advanced diagnostics for comprehensive ST physics and integrated operational scenario development capabilities. In Fig. 1.1, schematics of the NSTX and

MAST devices and their key features are shown. The basic device parameters are defined in Appendix A. The NSTX and MAST device and plasma parameters are similar: the major radius, $R_0 = 80 - 85$ cm, the minor radius, $a = 55 - 65$ cm, the plasma elongation, $\kappa = 1.7 - 3.0$, the plasma triangularity, $\delta = 0.3 - 0.8$, the toroidal magnetic field, $B_{T0} \leq 5.5$ kG, the plasma current, $I_p \leq 1.5$ MA, the plasma volume, $V_p \leq 14$ m³, and the plasma energy, $E_p \leq 0.5$ MJ. While NSTX and MAST have similar device and plasma parameters, they have important complementary features. As can be seen in Fig. 1.11, NSTX has a near-spherical vacuum vessel with a set of passive stabilizing plates near the plasma to offer effective wall stabilization at high β . NSTX also has electrical insulation between the center-stack and the outer vacuum vessel to allow up to 2 kV of DC electrical bias voltage for co-axial helicity injection (CHI) plasma start-up.

In a somewhat contrasting design, as can be seen in Fig. 1.12, MAST has a large cylindrical vacuum vessel with internal poloidal field (PF) coil sets which provide flexibility for plasma shaping and divertor configuration. The internal PF coils, being

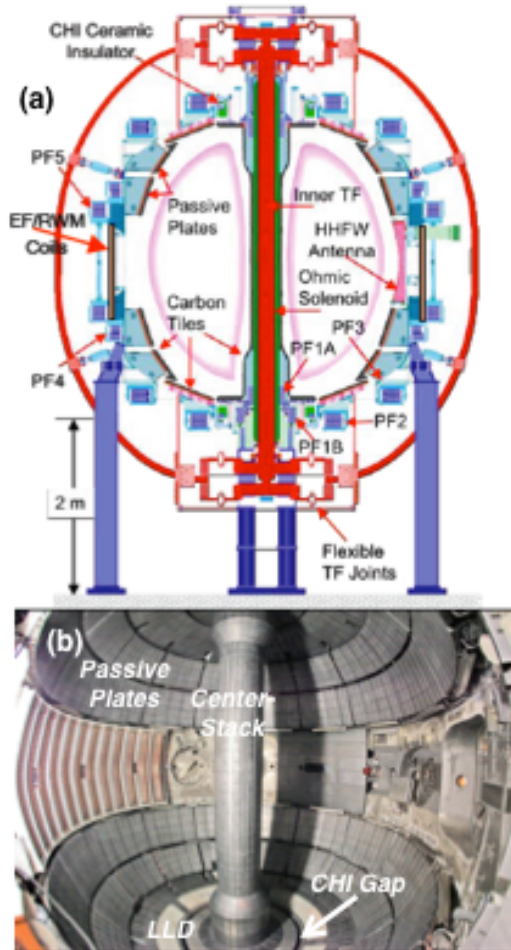


Fig. 1.11. (a) Schematic of NSTX device cross-section and (b) interior view.

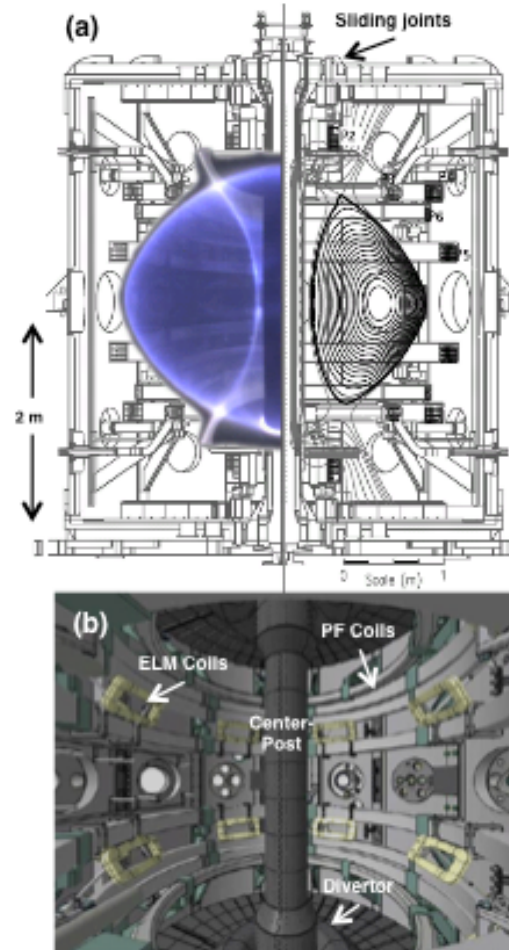


Fig. 1.12. (a) Schematic of MAST device cross-section and (b) interior view.

closer to the plasma, generally require less power for plasma control, and can also be used for PF coil-based plasma start up. MAST also has a large number of internal control (ELM) coils. Both devices have demountable TF coils, but MAST uses sliding joints and NSTX uses bolted flex joints. Both devices are heated by neutral beam injection (NBI), with $P_{\text{NBI}} \leq 7.4$ MW for NSTX and $P_{\text{NBI}} \leq 5.0$ MW for MAST. For radio frequency (RF) heating, NSTX has ~ 4 MW of high harmonic fast wave (HHFW) heating for electron heating and current drive, and MAST has ~ 0.2 MW of 28 GHz electron cyclotron heating for start-up and electron Bernstein wave (EBW) heating studies. Both MAST and NSTX are planning facility upgrades as described in Sec. C. Experimental research progress on NSTX and MAST is summarized in the overview Nuclear Fusion papers associated with the IAEA Fusion Energy Conferences [29, 37 - 49].

GLOBUS-M: GLOBUS-M in the Russian Federation is a medium-sized ST shown in Fig. 1.13, with an emphasis in its research program on RF and NBI auxiliary heating at relatively high magnetic field [5]. The Globus-M device and plasma parameters are $R_0 = 36$ cm, $a = 24$ cm, $\kappa = 1.5\text{-}2.0$, $\delta \leq 0.5$, $B_{T0} \leq 4$ kG, and $I_p \sim 0.2$ MA. Globus-M plasmas are heated by $P_{\text{NBI}} \leq 0.8$ MW and $P_{\text{ICRF}} \leq 0.3$ MW. Globus-M can operate at relatively low edge safety factor $q_a = 2.7 - 5$, compared to typical STs which operate at relatively large $q_a \geq 6 - 10$. Because of its compact size, high heat flux to the divertor plate can be achieved for plasma wall interaction studies. The facility has a unique fueling method of high speed, high density plasma jet injection. Globus-M is undergoing an upgrade (Globus-M2) on its magnets to operate at $B_{T0} \leq 1$ T with $I_p \sim 0.4$ MA [50]. Globus-M2 is scheduled to start plasma operation in 2015.

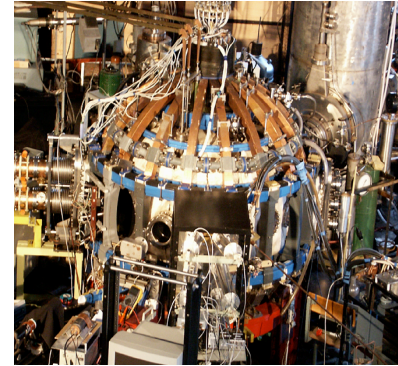


Fig. 1.13. Globus-M Device areal view

PEGASUS: PEGASUS at the University of Wisconsin is an ultra-low aspect-ratio medium size ST, and is presently focusing on plasma start-up and the MHD research [6, 51, 52]. The device has a highly-engineered Ohmic heating solenoid, with very high magnetic field ($B_{\text{sol}} \leq 15$) capability within the constraints of very tight space. As shown in Fig. 1.14, with a very slender

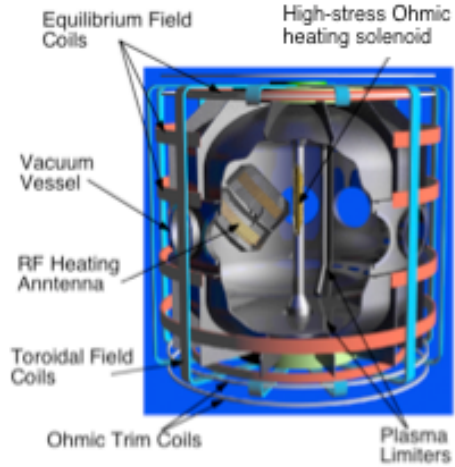


Fig. 1.14. Schematic of PEGASUS device.

center-post, the device can access an ultra-low $A \geq 1.13$ regime with ohmic current drive. The Pegasus device and plasma parameters are $R_0 \leq 40$

cm, $RB_T \leq 0.028$ Tm, and $I_p \leq 0.2$ MA. Because of the high ohmic resistivity, Pegasus has achieved up to $\beta_T \sim 20\%$ and also the H-mode by ohmic heating alone.

CDX-U/LTX: CDX-U/LTX is a medium-size ST which was initially dedicated to investigate innovative current drive research, but has been focusing on lithium technology in recent years. The Current Drive Experiment-Upgrade (CDX-U) device and plasma parameters were $R_0 = 34$ cm, $a = 22$ cm, $B_{T0} \leq 2.1$ kG, and $I_p \sim 0.1$ MA. The CDX-U was the first fusion device to test the effectiveness of large free surface liquid lithium as a plasma facing component [7]. The CDX-U facility was upgraded significantly in 2009 to become the Lithium Tokamak Experiment (LTX) [8]. The projected LTX device and plasma parameters are $R_0 = 40$ cm, $a = 26$ cm, $B_{T0} \leq 3.4$ kG, and $I_p \sim 0.4$ MA with > 100 msec current flat top. The device has an internal plasma wall conformed to the plasma. It consists of a conducting shell that can be heated up to 400°C continuously (550°C intermittently) to enable almost the entire plasma facing surface to be coated with liquid lithium as shown Fig. 1.15. The main objective of the LTX experiment is to investigate the

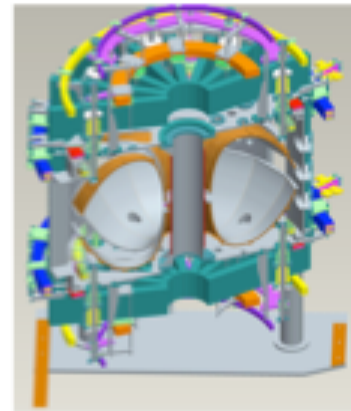


Fig. 1.15. Schematic of LTX device.

tokamak plasma performance enhancement under extremely low wall recycling.

HIT-II : The Helicity Injected Torus II (HIT-II) was a follow-on device to the Helicity Injected Torus (HIT) device [9] at the University of Washington. As shown in Fig. 1.16, HIT-II is a low-aspect-ratio tokamak with $R_0 = 0.3$ m, $a = 0.2$ m, and an on-axis toroidal field of up to 0.5 T. It is capable of both inductive (ohmic) and Coaxial Helicity Injection (CHI) current drive. It had demonstrated 0.2 MA of toroidal plasma current, using either CHI or induction separately. A unique feature of HIT-II is a feedback poloidal magnetic coil system, which can control loop voltage, boundary flux, and plasma equilibrium in real time, replacing the flux conserver in HIT. The HIT group completed HIT-II operation in 2005, and is presently investigating a new current drive concept based on continuous helicity injection using a spheromak (HIT-SI).

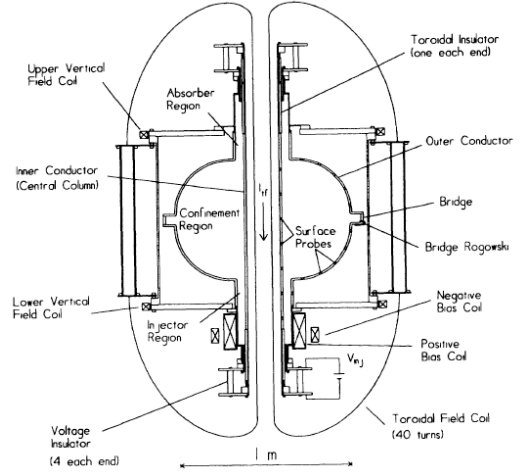


Fig. 1.16. Schematic of HIT-II device.

QUEST / CPD – The Q-shu University Experiment with Steady State Spherical Tokamak (QUEST) is the newest and largest spherical tokamak at Kyushu University in Japan. It came into operation in 2008, with the goal of achieving long-pulse non-inductive ST operations. The QUEST device parameters are $R_0 = 0.7$ m, $a = 0.48$ m, $A = 1.47$, and $B_T \sim 0.25$ T [10]. The facility has 2.45GHz (50 kW), 8.2 GHz (400 kW), and 28GHz (500 kW) ECH sources and an all metal wall with tungsten limiters. The device schematic is shown in Fig. 1.17. QUEST eventually aims

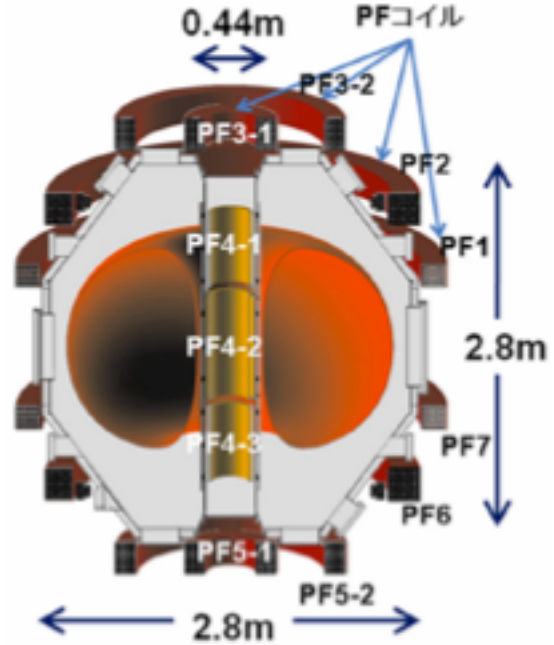


Fig. 1.17. Schematic of QUEST device.

to achieve about 100 kA of steady-state (~ 1000 sec). non-inductive operation with 1 MW of ECH/EBW power. In order to address physics and technology issues for steady-state ST operation, three research areas are being pursued. In the area of non-inductive current drive, EBW start-up and CD research is being carried out. Thus far, about 55 kA was driven with about 270 kW of 28 GHz ECH. In the area of heat and particle handling, detached divertor, closed divertor, hot wall, enhanced pumping and advanced fueling areas are being pursued. Finally, integrated control research including the core plasma, plasma wall interactions, and the wall itself will be carried. The QUEST device will be installing a hot wall as shown in Fig. 1.18, which can be heated to $300 - 500^\circ\text{C}$ to study issues related to particle retention in walls. Prior to QUEST operation, the Kyushu group built a smaller ST called CPD, which operated for 2006 – 2008, investigating non-inductive plasma start-up.

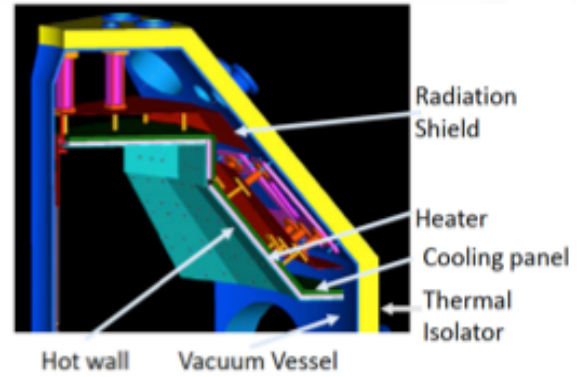


Fig. 1.18. Schematic of QUEST Hot Wall.

TST-2: TST-2 device is located at the University of Tokyo, Japan. The device and plasma parameters are $R_0 = 38$ cm, $a = 25$ cm, $B_{T0} \leq 3.0$ kG, and $I_p \sim 0.2$ MA [11]. The facility has radio frequency heating with high-harmonic fast wave (HHFW) power of up to 400 kW. A 100 kW ECH-based start up experiment was conducted in collaboration with the Kyushu University (QUEST) group. The TST-2 device interior view is shown in Fig. 1.19. A variety of start-up research has been conducted by the group [12]. The group also has conducted in-depth physics experiments including HHFW wave propagation, parametric instabilities, and electron Bernstein wave coupling physics. More recently, the group is

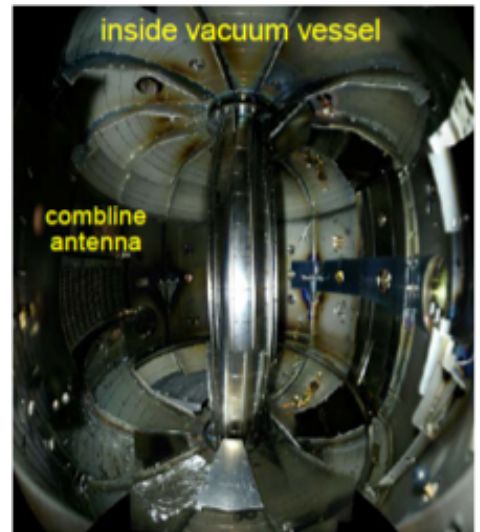


Fig. 1.19. Inside view of the TST-2 device

investigating plasma start-up utilizing lower hybrid wave current drive with a 200 MHz, 200 kW transmitter. I_p current ramp-up to 15 kA was achieved with various types of antennas including a “comblane” antenna and a dielectric-loaded waveguide array (grill) antenna.

LATE: The Low Aspect Ratio Torus Experiment (LATE) device at Kyoto University, Japan was built utilizing the former WT-3 tokamak facility [13]. The main objective of the LATE device is to demonstrate formation of spherical torus (ST) plasmas by electron cyclotron heating (ECH) alone (i.e., without a central solenoid) and establish its physics basis. The device cross section is shown in Fig. 1.20. It has a relatively large vacuum vessel about 1 m wide and 1 m height. The center post is only 11.4 cm wide, making it an ultra low aspect ratio device with $B_T \sim 1.15$ kG and $R_0 = 25$ cm. The facility has 2.45GHz (20 kW) and 5 GHz (200 kW) ECH sources. With the 5 GHz ECH system operated at ~ 190 kW, a start-up plasma with 20 kA was obtained with a very high current ramp-up rate of 260 kA/sec.

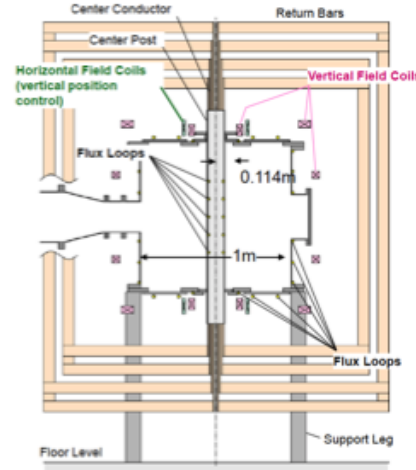


Fig. 1.20. The LATE device cross section schematic.

TS3//4: The TS-3 ST/CT merging device is a highly flexible basic physics facility with $R_0 = 0.18$ -0.22 m, $R_0/a=1.5$, and $B_T \sim 0.5$ kG at the University of Tokyo, Japan [14]. It produces two ST plasmas using electrodes for toroidal flux injection, and poloidal field coils for poloidal flux injection as shown in Fig. 1.21. This is followed by so-called co-helicity merging, and an ultra-high beta (up to 80%) ST plasma is formed. The plasma is transiently

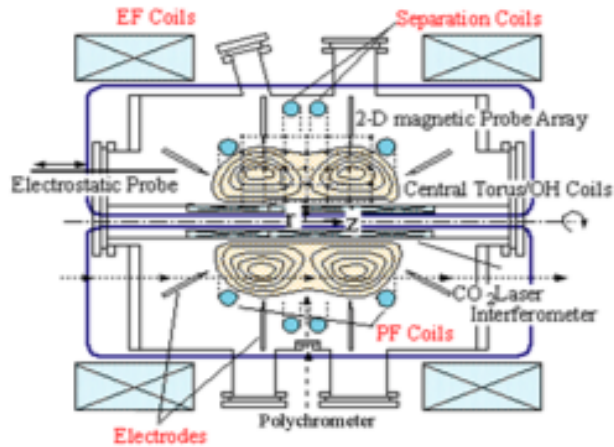


Fig. 1.21. Schematic of TS-3 device cross section

heated by the conversion of magnetic energy through reconnection during the merging process. Anti-helicity merging also produced an ST plasma, but at medium beta since there is little magnetic energy dissipation for this case. The merging process also helped create a start-up ST plasma as also demonstrated in MAST. The TS-3 and TS-4 devices contain a large number of internal magnetic sensors to make detailed magnetic measurements of magnetic reconnection and plasma formation. The TS-3 has recently started the NBI heating at the ~ 1 MW level to heat and sustain the formed plasma. TS-4 is a similar device to TS-3, but has larger dimensions.

UTST – The University of Tokyo Spherical Tokamak (UTST) is a new addition to the impressive collection of STs at the University of Tokyo, Japan. The device parameters are $R_0 = 0.4$ m (with a tall chamber height of 2.0 m), $R_0/a > 1.2$, $B_T = 1.8$ kG, and $I_p \sim 150$ kA [15].

The device is constructed to demonstrate the merging start-up scheme for a high-beta ST with ex-vessel PFC coils as shown

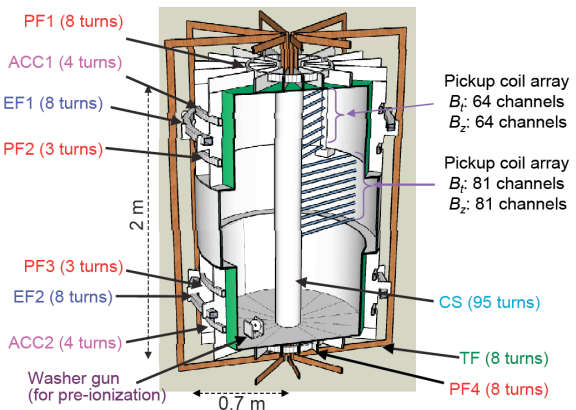


Fig. 1.22. Schematic of UTST.

in Fig. 1.22, in contrast with previous merging experiments conducted in TS-3/4 and MAST that all utilized in-vessel PF coils. The in-vessel PF coils are considered not desirable for reactor applications. To allow relatively rapid field penetration, the top and bottom sections of the UTST vacuum vessel are made of a thin stainless steel (1.5 mm) wall, supported by a 20 mm thick non-conducting material and 16 ribs. The device has washer guns in the top and bottom chamber to facilitate plasma start-up. The device contains a large number of internal magnetic sensors to make detailed magnetic measurements of magnetic reconnection and plasma formation as TS3/4. A 45 kV, 1 MW NBI system will be used for plasma heating and sustainment.

HIST – The Helicity Injected Spherical Torus (HIST) at Hyogo University, Japan is an ST facility which evolved from spheromak research. The device parameters are $R_0 = 0.3$

m, $a = 0.24\text{m}$, $R_0/a = 1.25$, $B_T = 1.0$ (3) kG, and $I_{TF} < 150$ kA [16]. The HIST device is shown schematically in Fig. 1.23. The main objectives of HIST is to use multi-pulse or repetitive transient CHI to achieve simultaneously quasi-steady plasma sustainment and good confinement. By pulsing CHI a second time, the pulse length was extended by a factor of ≥ 2 .

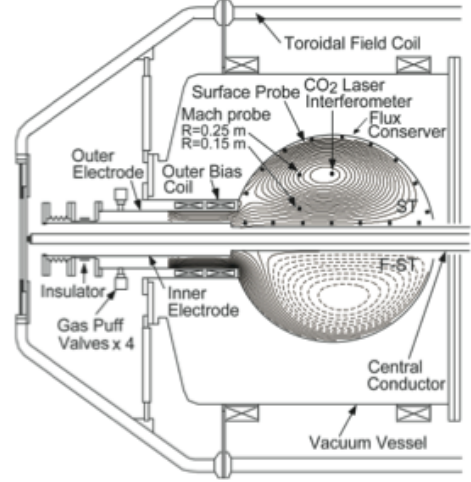


Fig. 1.23. Schematic of HIST device.

VEST – The Versatile Experiment Spherical Torus (VEST) is the newest ST located in Seoul National University, Korea, and it came on line in 2012 [17]. The device parameters are $R_0 = 0.4$ m (with a tall chamber height of 2.4 m), $R_0/a > 1.3$, $B_T = 1.0$ (3) kG, and $I_p \sim 30$ (100) kA as shown in Fig. 1.24. The main objectives of VEST is to conduct basic research on a compact, high- β ST with an elongated chamber with a partial ohmic solenoid to study innovative partial solenoid start-up, divertor physics, etc. Similar to UTST, the VEST device has separate upper and lower chambers where smaller ST plasmas, formed by the partial ohmic solenoid, can then be merged to form a larger higher current ST plasma in the main chamber. The OH and PF coils are located outside of the vacuum chamber, which has a relatively thick vacuum chamber wall (3.4 mm inboard and 6 mm outboard walls). The device utilizes ECH pre-ionization to assist the plasma start-up.

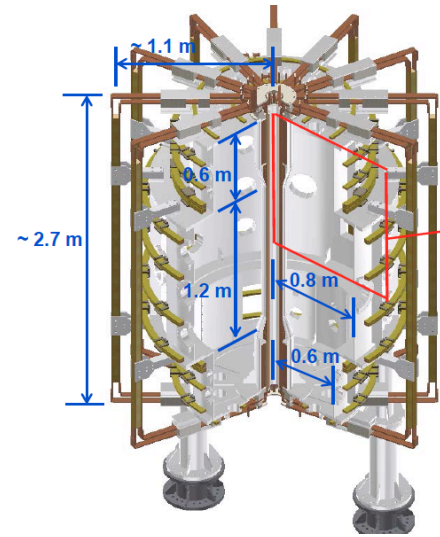


Fig. 1.24. Schematic of VEST device.

SUNIST – The Sino United Spherical Tokamak (SUNIST) is an ST device located in Tsinghua University, China. The device parameters are $R_0 = 0.3$ m with $R_0/a > 1.3$, $B_T = 1.5$ kG, and $I_p \sim 50$ kA as shown in Fig. 1.25 [18]. The main objectives of SUNIST research is to investigate the properties of low aspect ratio toroidal plasmas and non-

inductive plasma startup and current drive using both electron cyclotron and Alfvén waves. It is one of the few fusion research facilities located at a university in China, with a strong student training emphasis.

Other international STs - There are also other spherical tokamaks. The C-TOKASTAR device at Nagoya University, Japan is a low-aspect-ratio tokamak-helical hybrid device [53]. In Brazil, ETE is presently exploring wall eddy current effects [19]. In Italy, a new type of ST, PROTO-SPHERA, is being pursued [54]. This concept is composed of an ST with closed flux surfaces and a force-free screw pinch with open flux surfaces, and driven by electrodes to eliminate the in-board toroidal magnet entirely.

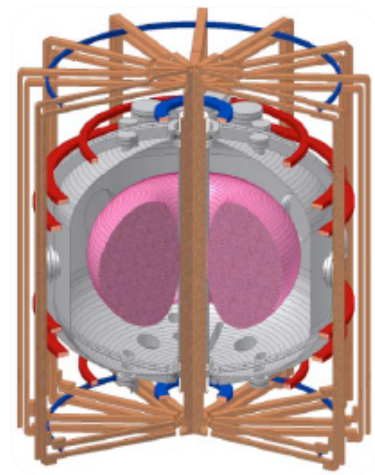


Fig. 1.25. Schematic of SUNIST device.

C. Description of NSTX-U and MAST-U – After over a decade of operations, both NSTX and MAST facilities are presently undergoing significant upgrades. Both devices are planning to double the toroidal magnetic field, plasma current, and heating power, and extend the plasma pulse length from ~ 1 sec to ~ 5 sec. The anticipated plasma performance enhancement is a quadrupling of the plasma stored energy and near doubling of the plasma confinement

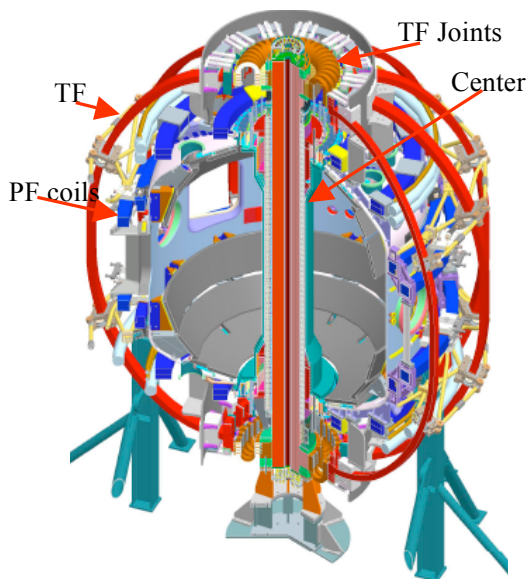


Fig. 1.26. Schematic of NSTX Upgrade.

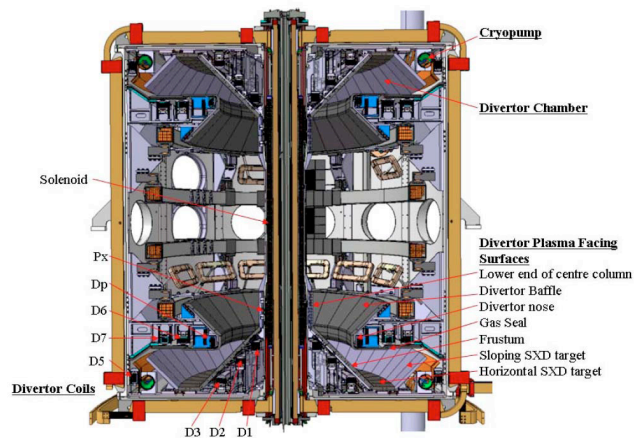


Fig. 1 Cross-section of MAST Upgrade.

Fig. 1.27. Schematic of MAST Upgrade.

time, which would result in an order of magnitude increase in the well-known fusion performance parameter of $n\tau T$. Even though the device sizes remain relatively compact, with ~ 1 T toroidal magnetic fields and $\sim 25\%$ toroidal beta values, the absolute plasma pressure expected in the upgrade STs could be comparable to that of the much larger present day tokamaks. This would assure that the exciting contributions of the ST community will continue to be at the forefront of the world fusion program. Schematics of NSTX-U and MAST-U are shown in Figs. 1.26 and 1.27.

As can be seen in the figures, both NSTX-U and MAST-U retain the basic configuration of NSTX and MAST, as much of each facility is utilized including the vacuum vessel, PF coils, and outer TF coils. Both upgraded facilities continue to provide complementary physics and technology capabilities. For example, the 2nd NSTX-U neutral beam achieves off-axis CD by strong tangential NBI injection on the mid-plane. The MAST-U beams, on the other hand, utilize vertically off-set NBI to achieve off-axis CD. The divertor PMI heat load solutions utilizing large divertor flux expansion to be pursued for NSTX-U are the snow-flake configuration and liquid lithium, while on MAST-U, the super-x configuration will be pursued. The mission elements of the upgrades are in three areas: 1. Provide the necessary data base for ST-FNSF or ST-CTF, 2. Develop demo-relevant divertor PMI solutions, and 3. Advance toroidal plasma physics to develop better predictive capabilities to support ITER. A brief description of each of the upgrades is given below.

NSTX-Upgrade - An upgrade to the NSTX facility (NSTX-U) was proposed to provide timely input for the FNSF construction decision, develop new solutions for the plasma-material interface, and better support ITER [23]. The main elements of the NSTX Upgrade Project are a new and more powerful center-

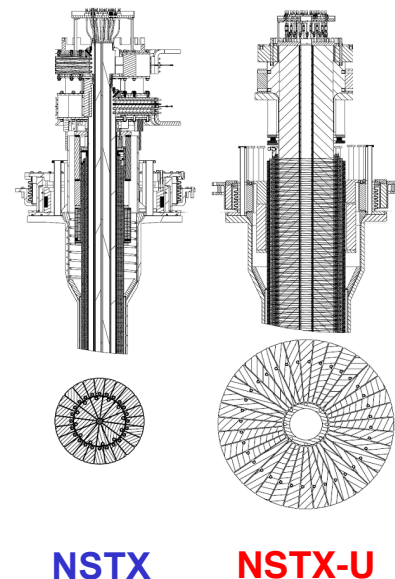


Fig. 1.28. NSTX and NSTX-U center stack schematics and the TF coil cross sections.

stack [55] and a tangentially-aimed 2nd Neutral Beam Injection (NBI) system. NSTX-U will double the toroidal field from ~ 0.5 T to 1 T, the plasma current from ~ 1 MA to 2 MA, the NBI heating and current drive power from ~ 7 MW to 14 MW at 90 kV, and it will greatly increase the peak field plasma pulse length from 1 sec to 7 sec. The larger cross section inner TF coil enables the doubling of the field and seven-fold increase in the pulse length as shown in Fig. 1.28. The NSTX-U and NSTX device parameters are given in Table 1. The tangential injection angles of the 2nd NBI, as shown in Fig.

	R_0 (m)	A_{min}	κ	I_p (MA)	B_T (T)	t_{TF} (s)	NBI (MW)	f_{NI}, f_{BS} (%) @1MA
NSTX	0.854	1.28	2.8	1	0.55	1	7	<70, <60
NSTX-U	0.934	1.5	2.8	2	1	6.5	14	$\leq 100, \leq 80$

1.29, enables much higher ($\sim 2x$)

Table 1. NSTX and NSTX U Parameters

plasma current drive efficiency and current profile control needed for fully non-inductive plasma operation. The vacuum vessel and associated magnetic field coil support structures are being enhanced in order to handle the anticipated 4x greater electromagnetic forces. The NSTX-U Project is now in the final construction phase, to be completed in 2014, and NSTX-U research operations will start in 2015.

From the start of operation in FY 2015, NSTX-U will have additionally 6 MW of High-Harmonic Fast Waves (HHFW) for heating and current drive. The total 20 MW provided by the NBI and HHFW systems will allow NSTX-U to uniquely produce FNSF/DEMO-relevant high divertor heat fluxes. Innovative divertor heat and particle solutions, such as the up-down symmetric, high flux expansion snowflake divertor configuration, will be used to control these heat fluxes. NSTX-U is also equipped with a set of six non-axisymmetric (3-D) control coils, which can be independently controlled to enhance plasma stability at high beta, control error fields, and apply resonant magnetic perturbations for ELM control. NSTX-U will continue to explore the use of lithium PFC coating techniques

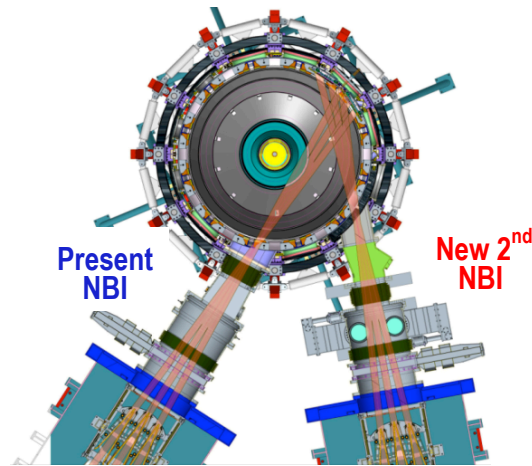


Fig. 1.29. Top view of the NSTX-U with the 2nd tangential NBI.

for enhanced plasma performance and divertor power and particle handling. A lithium granular injector will be implemented for ELM pacing at high injection rate to control impurities and to reduce peak ELM heat flux. Coaxial helicity injection (CHI) will be used to create plasmas with up to 400 kA non-inductively, and develop the solenoid-free tokamak/ST design needed for FNSF. If successful, it can also simplify the design of conventional reactors.

MAST-Upgrade – An upgrade to MAST (MAST-U) has been planned for a similar time frame as NSTX-U [56]. The main motivation is to make substantial progress on the most critical issues for the ST-Component Test Facility (ST-CTF), and to contribute strongly to aspects of ITER and DEMO-relevant physics that can be best studied in a spherical tokamak.

	R_0 (m)	A_{min}	κ	I_p (MA)	I_{TF} (MA)	t_{TF} (s)	NBI (MW)	f_{NBI}, f_{BS} (%)@1MA
MAST	0.85	1.32	2.5	1.5	2.0	1	3.8	<30, <30
MAST-U*	0.85	1.32	2.5	2	3.2	6.5	7.5	$\leq 100, \leq 40$

Table 2. MAST and MAST-U parameters. *Phase 1 of upgrade only.

The table for MAST and MAST-U device parameters is shown in Table. 2. There are two programmatic emphases for the MAST-U design. The first goal is to test a novel “Super-X” divertor configuration (SXD) to reduce divertor peak heat loads, and to access predominantly non-inductively driven discharges with stationary plasma currents ~ 1 MA for longer than 2-3 current diffusion time. The MAST-U construction period is planned for 2013 – 2015, and its research operation is planned to start in 2016.

A cross-section of MAST-U is shown in Fig. 1.30. Both the MAST top and bottom divertors will be upgraded from the existing open configuration without density control to a closed, pumped divertor (cryogenic pumps

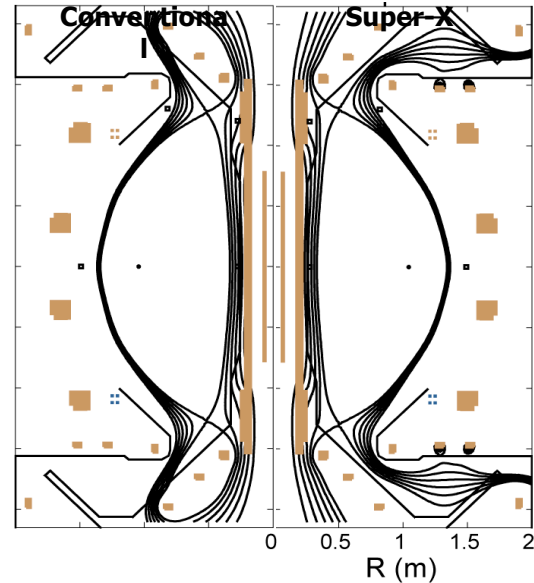


Fig. 1.30. The MAST-U divertor configuration with conventional and Super-X configurations.

based on supercritical helium) in both divertors to provide density control and access to low densities. Using eight new coils, each divertor is designed to allow the outer divertor leg to be operated either in a conventional X-point configuration or the SXD configuration as shown in Fig. 1.30. In the SXD configuration, access to the detached regime on the outer leg is aided by a magnetic configuration that increases the connection length, in a part of the scrape-off-layer that is well separated from the confined plasma region. This is achieved by creating a region of very low poloidal field in the baffled region of the divertor, together with stretching the outer leg to a large radius ($R \approx 1.5$ m compared to the X-point major radius ~ 0.8 m). Gas puffing of both fuel and impurities can be used in this separated ‘Super-X’ region to further cool the divertor leg and aid detachment at given core parameters.

The demonstration of predominantly non-inductively driven scenarios in MAST-U will be achieved using an upgraded neutral beam configuration. The first stage will have three neutral beam sources, each injecting up to 2.5 MW, with two of the sources installed in elevated positions ($Z=0.65$ m) for off-axis current drive as shown in Fig. 1.31. Two of the beam sources will be installed in a new Double Beam Box.

Apart from the pumped divertor and the off-axis neutral beam injection, several other upgrades to MAST subsystems are planned to allow pulse lengths of several seconds at high $I_p \geq 1$ MA. These enhancements include a new center column with three new shaping coils and 70% more flux capability, as well as a new long-pulse toroidal field power supply allowing 50% higher B_{TF} for more than 2s ($I_{rod} \leq 3.2$ MA) compared to MAST .

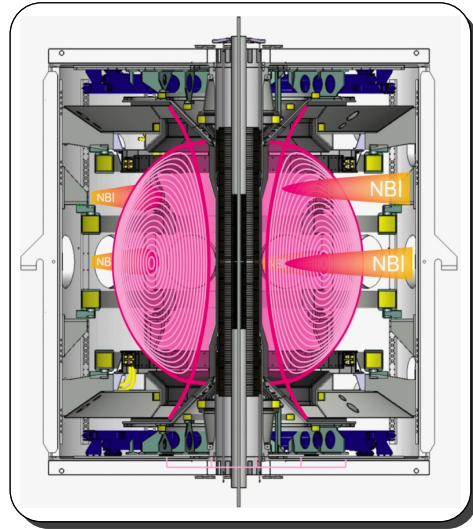


Fig. 1.31. The double beam box NBI configuration for off-axis current drive in MAST-U.

II. ST Fusion Development Path

A. Introduction. - While an ST is essentially a tokamak, the ST configuration assumes a spherical shape as the aspect-ratio $A \equiv R/a$ is reduced, due to its natural elongation as shown in Fig. 2.1 [57, 58]. Earlier advanced ST configurations reactor studies with high bootstrap current fractions have motivated the ST fusion development path [59 - 62]. As the aspect ratio is reduced, there are a number of

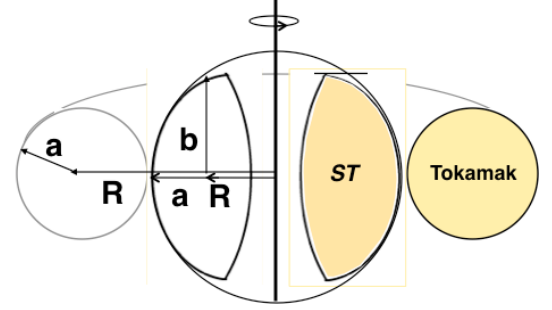


Figure 2.1. Cross-sections of basic spherical tokamak ($R/a = 1.5$) and conventional tokamak ($R/a = 3$).

important changes in the device characteristics. Perhaps the most significant is the toroidal coil current I_{TF} compared to the plasma current I_p . Their relationship can be approximated by,

$$I_{TF}/I_p \sim (2 A^2 q^*) / (1 + \kappa^2) \quad (1)$$

where $\kappa \equiv b/a$ is the plasma elongation and $q^* \equiv I_p B_{T0} / 2 \pi R_0 B_p$ is the cylindrical safety factor, where q^* should be ≥ 2 for stable tokamak/ST operations [21, 63]. For a given q^* , one can see from Eq. (1) that I_{TF}/I_p decreases rapidly as A is reduced and κ increased. For example, for an ST with $\kappa = 3$, $A = 1.6$, and $q^* = 3$, the ratio $I_{TF}/I_p \sim 1.53$ as I_{TF} becomes comparable to I_p . On the other hand, for a conventional tokamak with $\kappa = 1.5$, $A = 3$, and $q^* = 3$, the ratio $I_{TF}/I_p \sim 16$ and is an order of magnitude higher than that of STs. In other words, STs can sustain almost as much plasma current as the toroidal coil current, while a conventional tokamak can support only about 10%. This property of $I_{TF} \sim I_p$ makes the TF coil current in the ST relatively modest, and utilization of copper-based TF coils [64]

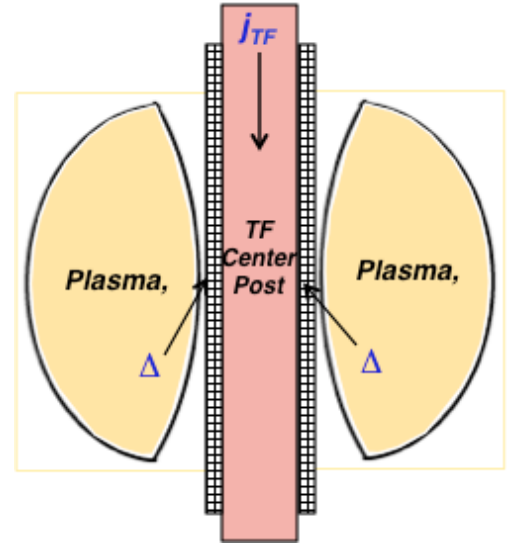


Figure 2-2. Copper based ST cross section with toroidal magnet and shielding gap Δ defined as gap between inner plasma edge and in-board magnet radius R_{TF} , where $B_T = B_T(\text{Max})$.

becomes practical as shown in Fig. 2.2. The ability to sustain high I_p also enables ST high performance as both the plasma beta β_T and confinement goes up with I_p . The relatively light-weight copper TF center-post can be

replaced, along with the divertor, PFC, blanket components, during periodic maintenance periods. Vertical replacement concepts of the center post have been developed for STs as described below. While the copper TF coil system increases recirculating power because of the copper coil ohmic loss, a copper coil-based power plant design study shows that the recirculating power fraction can be significant but manageable. The 1 GW-e ARIES-ST power plant design [65], for example, has the TF-coil ohmic loss (329 MW) to require about ~ 1 GW more total thermal power output as described in Sec. II. C.

Another important parameter is β_T , since the thermal fusion power output scales as β_T^2 . The inherently high β_T operation for STs can be seen in the following relation derived from Eq. 1,

$$\beta_T \equiv \beta_N I_p / (aB_{T0}) = \beta_N (5 I_p / I_{TF}) A \sim \beta_N 5 (1 + \kappa^2) / (2 A q^*) \quad (2)$$

where the normalized β (or β_N) is considered to be relatively constant, though there is a tendency for β_N to increase by a factor of two for an ST compared to a conventional tokamak. As can be seen in Eq. (2), for a given β_N and β_T , $I_p / (aB_{T0})$ increases strongly with higher κ and reduced A and q^* . An ST tends to be able to access relatively high κ because of the natural elongation, which makes STs more vertically stable for a given κ . In Fig. 2.3, calculated ideal MHD stability limits for β_T , κ , and β_N for fully self-sustained wall-stabilized equilibria as a function of aspect ratio are plotted [66, 67]. The

calculations show that β_T increases rapidly as A decreases. The rapid β_T increase is due to the combined effect of increasing κ and β_N as A is decreased by Eq. (2). The fusion power plant operating regime which requires near fully self-sustained plasma is usually intended to be designed below this ideal stability limit with a safety margin $\sim 10\%$.

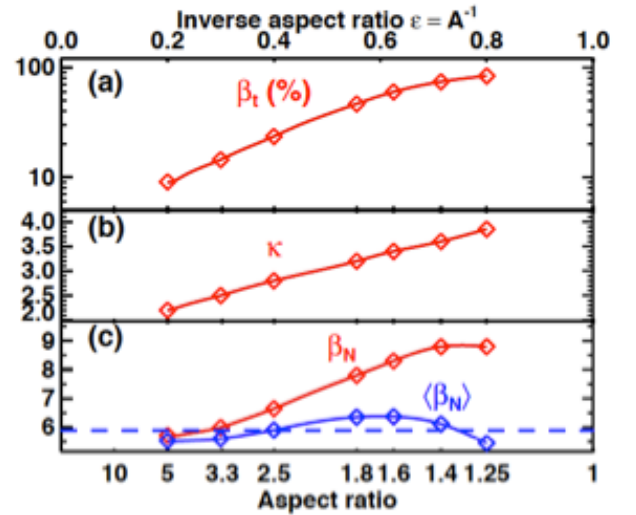


Fig.2.3. (a) Toroidal beta, (b) plasma elongation, and (c) normalized beta values versus aspect ratio for wall-stabilized fully self-sustained equilibria.

Another important quantity for an ST and tokamak power plant is the bootstrap current fraction f_{BS} . The bootstrap current is the neoclassical pressure driven current driven by the toroidal momentum transfer between trapped and passing particles, and has been confirmed in tokamak experiments [68, 69]. Ideally, one would like to have a large bootstrap current fraction, maintained by the pressure gradient from the fusion alpha heating power, for economical tokamak / ST based power plant operation. The f_{BS} can be expressed as

$$f_{BS} \equiv I_{BS} / I_p = C_{BS} \beta_p / A^{0.5} = (C_{BS}/20) A^{0.5} q^* \beta_N \propto A^{-0.5} (1+\kappa^2) \beta_N^2 / \beta_T \quad (3)$$

where C_{BS} is a profile-dependent constant, typically ~ 0.6 , and β_p is the poloidal beta. Since β_N rises by 50% as A is reduced from 3 to 1.5, $\beta_N A^{0.5}$ tends to be relatively constant. As there is a tendency for an ST power plant to run with higher values of q^* , the ST may be a somewhat more favorable for achieving the high f_{BS} required for economical power plant operation. The plasma profiles are shown in Fig. 2.4 for $A = 1.6$ and 3.3 [66]. The profiles are quite similar for both cases except for the q -profiles. The $A = 1.6$ case maintains nearly monotonic q -profile throughout all radii

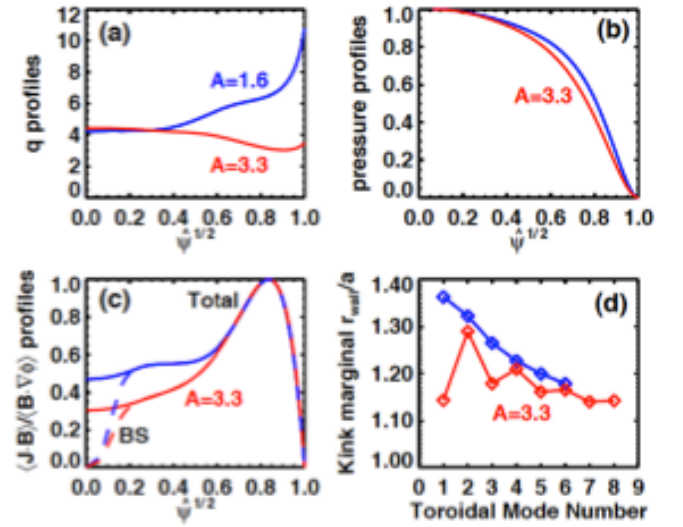


Fig. 2.4 (a) q (safety factor) profiles, (b) normalized pressure profiles, (c) normalized current density profiles, and (d) kink marginally-stable wall position divided by plasma minor radius for the $A=1.6$ and $A=3.3$ equilibria shown in Figure 2.3.

whereas the $A = 3.3$ case q -profile is relatively flat with reversed shear profile as envisioned for the Advanced Tokamak (AT). While the reversed shear may have the advantage of improved confinement, it may be more prone to MHD instabilities. This q -profile difference between the ST and AT may become an important consideration for choosing a future DEMO/Power Plant design option. To optimize the aspect-ratio choice for a tokamak DEMO/power plant is another important mission for on-going ST research. It should be pointed out that since the bootstrap current profile is determined by the plasma pressure profile, and fusion DEMO/Power Plant is mainly heated by the fusion α -particles, the plasma pressure profile is determined mainly

by the plasma energy transport. It is therefore highly desirable to develop a tool to control the plasma energy transport for a high bootstrap current fraction DEMO/Power Plant [70].

A copper-based ST appears to be particularly attractive in the near term for a compact fusion neutron science facility (FNSF), where net electrical power generation is not its primary mission [71, 72]. A simplified schematic is shown in Fig. 2.2, where a single turn (solid copper) TF post generates the required toroidal field. As shown in the figure, some space (Δ) between the in-board TF magnet and the plasma is inevitably needed. At minimum, one would need an appropriate tile material to protect the copper TF coil from the plasma in that vicinity, where Δ maybe only a few cm. For simplicity, we assume a uniform current density j_{TF} within the TF center post with radius R_{TF} . Then the total TF center rod current I_{TF} goes up as R_{TF}^2 or $I_{TF} = \pi R_{TF}^2 j_{TF}$. But since $R_{TF} \propto R$ for a fixed A , I_{TF} scales as R^2 . Since the device toroidal field $B_{TF} \propto I_{TF}/R$, $B_{TF} \propto R$ or B_{TF} goes up linearly with device major radius R for a fixed A . There are two ways to generate fusion neutrons, thermal fusion and beam-target fusion reactions. The thermal fusion power P_{fusion} generated for a given device is proportional to $V \times (nT)^2$, or $P_{fusion} \propto V \times B_T^4 \beta_T^2$ where V is the plasma volume, nT is the plasma pressure, and $\beta_T \propto nT / B_T^2$. Since $V \propto R^3$, we obtain

$$P_{fusion} \sim V \times (nT)^2 \propto V \times B_T^4 \beta_T^2 \propto R^3 R^4 \beta_T^2 \propto R^7 \beta_T^2 \quad (3).$$

As can be seen from Eq. 3, P_{fusion} has a particularly strong size scaling. The neutron flux $F_{neutron}$ on the first wall, which is relevant for blanket development (one of the main purposes of the FNSF), also has a strong dependence on the device size as $F_{neutron} \propto P_{fusion} / R^2 \propto R^5 \beta_T^2$. For this reason, the thermal fusion power output rises dramatically as the size of a copper-based ST increases. Accordingly, there is a minimum size for the FNSF facility required for significant thermal neutron production where the ITER-like plasma pressure is required. A smaller FNSF, however, can produce fusion neutrons via NBI (neutral beam injection) driven beam-target reactions in a similar manner to the TFTR/JET DT experiments [73, 74]. For effective beam target reactions, it is only necessary to produce target plasmas with sufficiently high plasma parameters, i.e., comparable to TFTR and JET. For efficient beam-ion fusion neutron production, electron temperatures must be sufficiently high to reduce the beam-ion slowing

down rate compared to the beam-ion fusion reaction rate. To achieve such target plasma parameters, one may need plasma pressure (nT) or $B_T^2 \beta_T$ values comparable to those achieved for TFTR/JET. The $B_{T0}^2 \beta_T$ product (in units of $T^2 \%$) is ~ 30 for TFTR/JET-level plasma pressures, ~ 70 for ITER, and $\sim 120 - 200$ for DEMO. It is instructive to examine what level of plasma pressure certain-sized STs can attain. In Fig. 2.5, $B_{T0}^2 \beta_T$ is plotted as a function of R_0 (as a measure of the achievable plasma pressure) for possible FNSF parameters: $A = 1.6$, $q^* = 3$, $\kappa = 3$, and a relatively modest beta of $\beta_N = 3$ at the two gap distances of $\Delta = 0$ and 10 cm. In this FNSF regime, neutral beams are assumed to provide both plasma heating and the required current drive, where the bootstrap current fraction is $\sim 50 \%$. In the in-board TF center rod, we assume $j_{TF} = 2 \text{ kA/cm}^2$, which is a typical value if steady-state water cooling is assumed. For $\Delta = 0$, TFTR-JET-like $B_{T0}^2 \beta_T \sim 30$ may be achieved with an ultra-compact ST device with $R_0 \sim 50$ cm, provided that

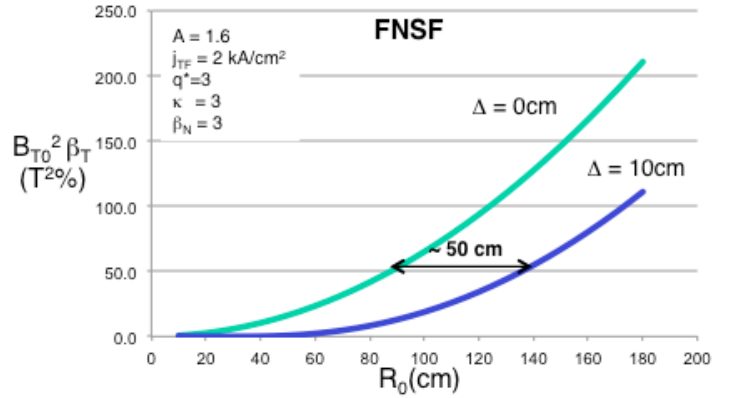


Fig. 2.5. Plasma pressure dependence with the gap distance as a function of the device major radius R_0 for FNSF-like parameters;

the plasma energy confinement is sufficiently high to attain $\beta_N = 3$. This would correspond to an H-factor 1.5 (i.e., 50% better than typical H-mode confinement). One can obtain more typical FNSF values of $B_{T0}^2 \beta_T \sim 50 - 100$ for a device of $R_0 \sim 80 - 120$ cm. A pilot plant (DEMO-like-plasma) value of $B_{T0}^2 \beta_T \sim 200$ may be achievable with $R_0 \sim 150 - 200$ cm. While this is a very encouraging result, it should be also noted that the effect of Δ cannot be ignored. For a low-aspect-ratio geometry, Δ is a particularly a high-leverage quantity. In Fig. 2.5, one can see that the impact of such a gap is quite significant. A relatively modest gap distance of $\Delta = 10$ cm would increase the device size R_0 by 50 cm (which is five times Δ) to produce plasmas with equivalent $B_{T0}^2 \beta_T$. So for a very compact FNSF, it is essential to keep the gap distance to a minimum, perhaps to $\sim 2-3$ cm, or one must develop an engineering design to accommodate higher $j_{TF} > 2 \text{ kA/m}^2$. In Fig. 2.6, a similar plot is shown for ST DEMOs, one for the ARIES-like copper design and another for a superconducting coil ST design. The copper ST DEMO

operating design point is $B_{T0}^2 \beta_T \sim 200$, to produce the ~ 3 GW of fusion power needed to provide the 1 GW-electric to accommodate the TF ohmic recirculating power loss. For a copper design, we assumed an ARIES-like $A = 1.6$, $q^* = 3$, and a relatively conservative $j_{TF} = 1$ kA. For plasma parameters, a relatively advanced $\kappa = 3.3$ and $\beta_N = 6.5$ are assumed to ensure high bootstrap current fraction f_{BS} at high current. The $\Delta = 20$ cm is assumed to provide sufficient shielding for the center-post, and a $B_{max} = 7$ T which is again quite modest. For Fig. 2.6, the operating point of $R_0 \sim 3.2$ m is used. For a superconducting magnet ST (SC-ST) design, a conservative $j_{TF} = 3$ kA was assumed. The SC-ST DEMO operating design point is $R_0 \sim 4.4$ m with $B_{T0}^2 \beta_T \sim 120$, since only 2.2 GW of fusion power is needed for 1 GW-electric due to the lower recirculating power. An assumed $B_{max} = 12$ T is also an acceptable value from the SC magnet technology point of view. The shielding distance Δ of 100 cm is assumed. Because of the relatively small solid angle of the relatively slender center post, a sufficient tritium breeding ratio (TBR) can be achieved

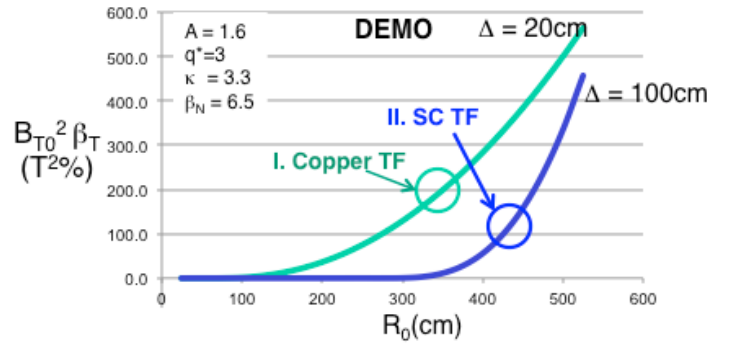


Fig. 2.6. Operating points depicted by circles for ST DEMOs. I. ARIES-like copper TF reactor with $B_{max} = 7$ T at operating point of $R_0 = 3.2$ m. II. Superconducting ST with shielding distance of $D = 1$ m and $B_{max} = 12$ T at operating point of $R_0 = 4.4$ m.

without needing a tritium blanket for the inboard side. The operating point for the SC-ST is $R_0 \sim 4.4$ m, which is only 1.1 m or $\sim \Delta$ larger than the copper version, due to the larger j_{TF} allowable for the SC coil. The SC-ST size could be reduced if the SC magnet technology can be improved further to allow higher j_{TF} and B_{max} . The target device and plasma dimensionless design parameters for various future devices are shown in Table III.

In Sec. B, the fusion neutron science facilities (FNSF) are discussed. While there are many versions of FNSF, we consider here three representative ST-FNSF designs as shown in Table III. FNSF-I represents the ORNL-led design [75], FNSF-II the PPPL-led design [76], and FNSF-III the Culham-led design [77]. The neutron wall loading $W_L = 1$ MW/m² is assumed for the FNSF design point. It should be noted that it is possible to obtain higher neutron wall loading for a given FNSF device. For example, by increasing B_T by $\sim 20\%$, one can double the thermal fusion

<i>Parameters</i>	FNSF-I	FNSF-II	FNSF-III	Pilot	ARIES-ST
$R_0(m)$	1.3	1.0	0.85	2.2	3.2
A	1.7	1.7	1.55	1.7	1.6
$B_T(T)$	2.9	3.0	2.5	2.4	2.1
$I_p(MA)$	6.7	7.5	6.5	18.0	29.0
κ	3.1	3.0	2.4	3.3	3.4
δ	0.5	0.5	0.4	0.5	0.64
q^*	4.1	3.5	2.3	3.0	2.9
q_{95}	10	7.3	6.6	7.3	9.0
β_N	3.3	4.6	3.5	5.3	7.4
$\beta_T(\%)$	10.1	19.5	16.6	31	50
f_{BS}	0.45	0.8	0.4	0.86	0.96
H	≤ 1.25	1.2	1.3	1.34	1.47
N_G	0.4	0.8	0.59	0.70	0.6
$P_{Fusion}(MW)$	76	60	35	645	2980
$W_L(MWm^{-2})$	1	1	1	2.0	4.1

Table III. Device and plasma parameters of future ST facilities.

power, since the thermal fusion power scales approximately as B_T^4 for given β_T . As shown in Table III, the ST-FNSF facilities are designed with relatively modest values of β_N and β_T which have been already achieved in present-day STs, at least transiently. The required confinement H factor of 1.2 – 1.3 is also close to the range observed in present-day STs as discussed in Sec. V. The size variation and therefore the potential facility cost variation among three design points are quite large, even though they are producing same W_L . A larger facility such as FNSF-I tends to have more conservative physics assumptions. On the other hand, q^* tends to be proportionally lower (more MHD unstable and more prone to disruptions) for a smaller FNSF, which results in similar values of I_p for comparable plasma/fusion performance. As noted above, the I_p value tends to define the fusion performance of a device with similar A. Another point to note is that the FNSF-II design point is motivated by the NSTX experimental database, which has somewhat higher β_N and f_{BS} but also higher N_G compared to other FNSF designs. A pilot plant study has been led by PPPL [78], and it has similar plasma parameters as the FNSF-II since the NSTX database motivated the choice of its operating parameters. Compared to FNSF, the pilot plant is a much larger device ($\sim \times 2$) with a larger fusion power output ($\sim \times 10$), and attempts to achieve

an engineering $Q = 1$ with the possibility of net electricity generation [80]. As shown in Table III, the pilot plant operates with plasma parameters closer to the ST power plant study (ARIES-ST) in terms of κ , q^* , β_N , β_T , and f_{BS} [65]. The ARIES-ST operates with a very high β_N and f_{BS} to minimize the recirculating power needed for non-inductive current drive for power generation.

B. Fusion Neutron Science Facilities - Fusion tritium breeding blanket module development represents a major technological challenge for magnetic fusion research, regardless of magnetic

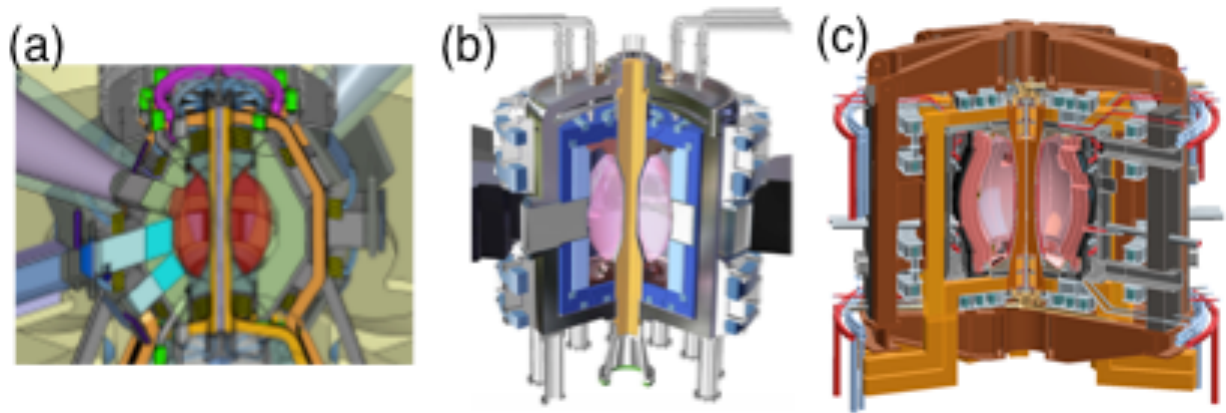


Fig. 2.7. Three representative FNSF design studies. (a) Culham compact design with $R_0 = 85\text{cm}$, (b) ORNL medium size design with $R_0 = 130\text{ cm}$. (c) PPPL pilot design with $R_0 = 220\text{ cm}$.

configuration. An international study on a high-volume plasma-based neutron source (VNS) for fusion blanket development was conducted to determine the requirements for a neutron source for fusion nuclear technology development [71]. This international team considered the technical issues and the required testing facilities for fusion blanket and first wall systems, to meet the goal of demonstrating blanket availability in DEMO $>50\%$. The study also noted that other possible test stands, including non-neutron test stands, fission reactors (low energy neutrons), and accelerator-based point neutron sources, would not satisfy the testing requirements, and concluded that “if the cost of such a VNS device can be drastically below that anticipated for ITER, the former can become a highly attractive developmental step in fusion energy research.” Due to its potentially low cost, the ST-based compact VNS has attracted numerous studies since the VNS assessment study. We should note here that the volumetric fusion neutron source test facility has been called by many names including VNS (Volume Neutron Source) in earlier times, CTF (Component Test Facility) mainly used in the UK and RF

to refer to a device with a similar mission to VNS, and more recently, FNSF (Fusion Neutron Science Facility) as a general term for a fusion neutron science facility encompassing the mission of the VNS / CTF. In this paper, for simplicity, we shall call all of these volumetric neutron facilities as the FNSF. While ST-FNSF studies have been carried out since 1990's [72, 79, 80], we shall mention three newer ST-FNSF conceptual design studies performed since 2000, as the experimental data from NSTX and MAST became available. The size of the FNSF devices also vary greatly from $R_0 \sim 50 \text{ cm}$ for an ultra-compact low level (few MW) neutron source design to $R_0 \sim 2 \text{ m}$ for a pilot plant design which attempts to test all the technical aspects of DEMO, including net power generation and tritium self-sufficiency. The ST-concept therefore offers a broad range of FNSF possibilities, depending on the cost and its mission. Three representative examples we mention here (also see Table III) are the newest design studies led by the Culham [77], ORNL [75], and PPPL [76, 78] groups. In Fig. 2.7, schematics of the three facility designs are shown.

The Culham-led effort is shown in Fig. 2.8(a), and represents the most compact design with $R_0 \sim 85 \text{ cm}$, $A = 1.55$, $\kappa = 2.4$ at a I_{TF} of 10.5 MA and $I_p = 6.5 \text{ MA}$ at a $q^*=2.3$ [77]. Relatively modest values of β_N of 3.5 and β_T of 17% are assumed. About 50% of the plasma current is driven by NBI, and another 50% by the bootstrap current. An H-mode confinement factor (ITER 98-pby2) of 1.3 is assumed to produce 35 MW of total fusion power with 44 MW of auxiliary heating, yielding 1 MW/m^2 neutron wall loading. The Culham FNSF center-rod cross section is shown in Fig. 2.8. In addition to the water-cooled TF center rod, it includes the 2.5 cm thick inboard shield, and a small start-up solenoid. The average rod current density is $\sim 7 \text{ kA/cm}^2$ for this design. An operating point of $B_{T0}^2 \beta_T \sim 100$ is expected to result in significant thermal fusion neutron production along with those produced by beam-target reactions. A vertical core replacement concept has also been developed. As a variation of the Culham design,

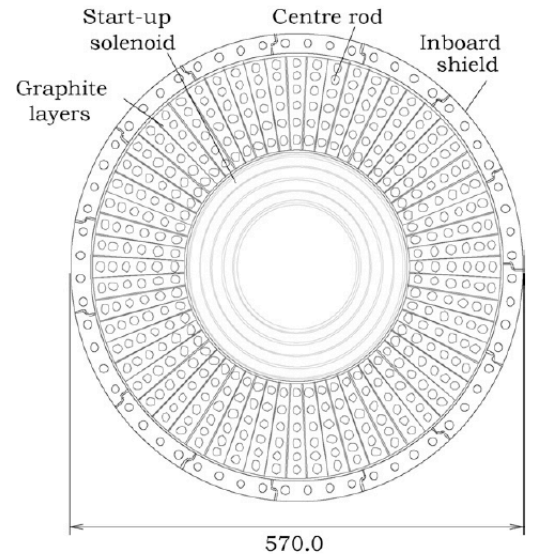


Fig. 2.8. Culham design of center rod cross section at mid-plane.

there are even more compact FNSF device designs with $R_0 \leq 50$ cm, where neutron generation is based mainly on beam-target reactions at $B_{T0}^2 \beta_T \leq 20$ [81].

The ORNL-led FNSF design is shown in Fig. 2.7 (b), based on a medium-sized ST design with $R_0 = 130$ cm, and $A = 1.7$ with $B_{T0} = 2.9 - 3.6$ T. As with other FNSF designs, about 50% of the current is driven by NBI and another 50% by the bootstrap current [75]. The operating beta is a relatively conservative $\beta_N \leq 3.5$, operating at $\sim 20\%$ below the no-wall beta stability limit. The vertical remote handling concept is shown in Fig. 2.9. An H-mode confinement factor of ≥ 1.25 is assumed. There are two operating points for blanket R&D. At $B_{T0} = 2.9$ T, $I_p = 6.7$ MA, $\beta_N =$

3.3, and β_T of 10%, 76 MW of total fusion power is produced with 44 MW of auxiliary heating, yielding 1 MW /m² of neutron wall loading. For a higher neutron wall loading of 2 MW /m², operating at $B_{T0} = 3.6$ T, $I_p = 8.4$ MA, $\beta_N = 3.5$, and β_T of 11% produces 152 MW of total fusion power with 60 MW of auxiliary heating. The average rod

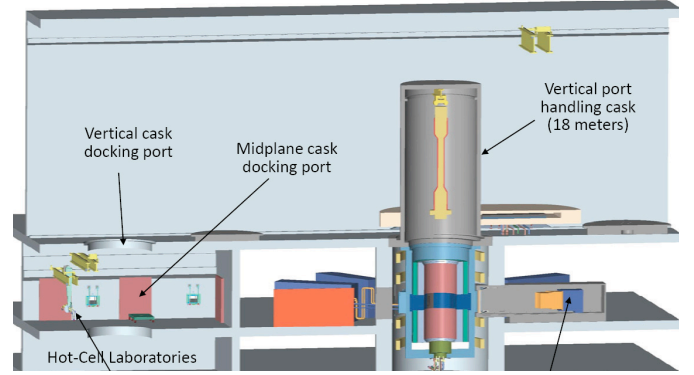


Fig. 2.9. ORNL vertical remote handling concept

current density for this design is more manageable at ≤ 4 kA/cm² from the engineering viewpoint, or about the half of the Culham design. A study for optimizing the cooling channels was also conducted to minimize the temperature in the centerpost [82]. For this design, the operating plasma pressure of $B_{T0}^2 \beta_T \sim 80 - 140$ is expected. This is similar to that of the Culham design, but with considerably higher fusion power due to the much larger $\sim 3x$ plasma volume.

The PPPL led FNSF/pilot plant design is shown in Fig. 2.7 (c) and it addresses the ST pilot plant mission with a relatively large device having $R_0 \sim 220$ cm and $A = 1.7$ at $B_{T0} = 2.4$ T [78]. This device is smaller than a DEMO but runs essentially at the DEMO-level plasma performance. Its aim is to test all of the essential DEMO physics and technology functions, including net power generation and tritium self-sufficiency. Accordingly, the physics parameters are DEMO-like with $\kappa \sim 3.3$, $q^* \sim 3$, $\beta_N \sim 6$, $\beta_T \sim 30 - 40\%$, and $f_{BS} \sim 90\%$. An H-mode confinement factor of 1.35 is assumed, which is a similar value to other FNSF designs. While it is more expensive than other

FNSF designs, it can greatly reduce the technology and physics risks for designing an ST-DEMO. The total fusion power of 600 – 1000 MW (which could be used to run the plant) can be generated with a heating power of 50 – 60 MW, producing a DEMO-like neutron wall loading of 2 - 3 MW /m². The TF center-post has a 10 cm thick shield to reduce radiation damage and nuclear heating. The average center-post current density is quite conservative at ≤ 2 kA/cm². For this design, $B_{T0}^2 \beta_T \sim 200$, which is a similar plasma pressure to that in a typical DEMO design. A vertical disassembly procedure has been developed, as shown in Fig. 2.10.

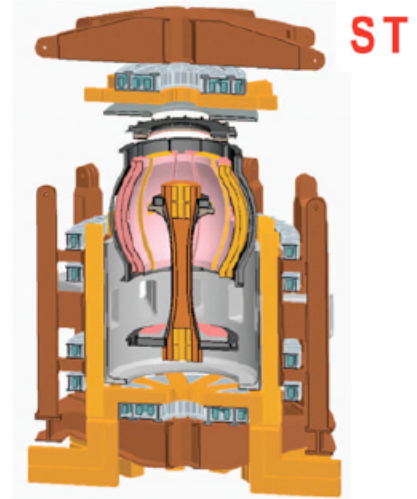


Fig. 2.10. PPPL ST pilot plant vertical maintenance scheme.

C. Other Possible Near Term ST Facilities – In addition to the ST-FNSF facilities described in Sec. B, there are a variety of possible so-called “next-step” ST facilities utilizing the compactness of high-performance ST plasmas. In the early part of the decade, a conceptual design activity was carried out for a 10 MA-class ST device [83, 84]. It was to be a follow-on device to NSTX to qualify the ST plasma performance needed for ST-FNSF devices and thus minimize the physics risks. Those physics gaps for the ST-FNSF are now expected to be filled with NSTX-U and MAST-U as described in Sec. I.D. Here we present two examples of utilizing compact ST configurations, i.e., a plasma-material development facility and a fusion-fission transmutation facility.

Plasma material interface experiments – There is a general consensus in the community that the plasma material interface (PMI) is one of the greatest challenges in a fusion power plant. For example, if one were to expect a 2-3 GW fusion power plant of the size of ITER, the expected heat flux would be four to six times that of ITER for a much longer divertor component operation time. Similar PMI challenges are expected for an ST-based FNSF and DEMO/Power Plant. Satisfactory solutions are not available today, and clearly innovative solutions are needed due to the magnitude of the PMI challenge as discussed in Sec. VII. The FNSF facility, particularly in the shakedown phase, can be used to develop those PMI solutions. But since

FNSF will be considered to be a nuclear facility, its construction approval may only come after satisfactory PMI solutions are available. The NSTX-U and MAST-U devices will be investigating innovative divertor/PMI solutions with DEMO/FNSF level divertor heat fluxes, but with limited plasma pulse lengths and duty factors. To address those PMI issues in an integrated facility, and taking advantage of high power density of an ST configuration, the National High Power Advanced Torus Experiment (NHTX) was proposed [85]. The basic device parameters are $R = 1m$, $P_{in} = 50 MW$, $a = 0.55m$, $\kappa = 2.7 - 3.0$, $B_T = 2T$, and $I_p = 3 - 4 MA$, with the goal of yielding very high power flux levels of $P/S = 1.1 MW/m^2$ and $P/R = 50 MW/m$ for extended pulse durations. It uses steady-state, water-cooled, demountable copper coils for maximum flexibility and accessibility. The NHTX facility can accomplish its mission with relatively modest confinement and beta parameters, and without the complexity of a nuclear environment as the facility is mostly operated with deuterium.

Fusion-Fission Transmutation System - Another ST application area is the fusion-fission transmutation system (FFTS), which has been examined recently with a replaceable compact ST fusion-core as shown in Fig. 2-11 [86]. The basic aim of this facility is to use the high-energy spectrum of fusion neutrons from the ST-core to irradiate the spent fission-fuel-based blanket modules. This is believed to greatly enhance net-energy production, while eventually reducing the highly toxic long-lived actinide level in the spent fission fuel. The parameters of the proposed ST core of the Texas FFTS device are $R_0 = 135 cm$, $A = 1.8$, $\kappa = 3.0$, and $B_{T0} = 2.9 T$ [86]. With $I_p = 10 - 14 MA$, $\beta_N = 3.3$, and β_T of 15 - 18 %, 50 MW of auxiliary heating, there would be 0.9 MW /m² neutron wall loading. The FFTS neutron core is therefore similar to the 1 MW/m² version of the ORNL FNSF in terms of device size and plasma parameters. The device utilizes the super-x divertor configuration (to be tested on MAST-U) to handle the intense steady-state divertor heat flux.

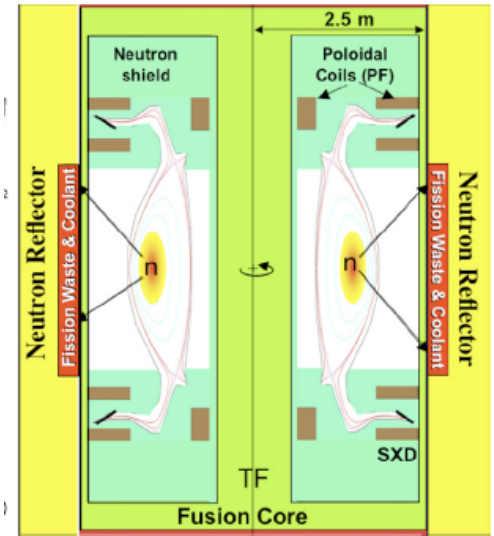


Fig. 2.11. 2D schematic of a ST-based fusion-fission hybrid with super-x divertor.

D. ST Power Plants – In this section, we briefly describe the ST-based power plant studies. There are two main versions of the ST-power plant designs. One is based on the copper TF magnets. As shown in Fig. 2.6, copper magnets have the advantage of allowing more compact size reactors but requires larger recirculating power due to the resistive loss associated with copper [65, 87]. The copper TF center-post must also be refurbished periodically. The other class is the super conducting (SC) TF design being pursued in Japan [88] and Korea [89]. The size of the power plant is larger because of the thicker in-board neutron shield ($\Delta \sim 100$ cm) as shown in Fig. 2.6. The advantage of the SC-ST is that because of the neutron shield, the TF magnet is expected to last for the plant life, and it can also accommodate a modest OH solenoid for start-up. Due to the low-aspect-ratio geometry (the center-stack or the central post profile is relatively slender), all of the ST power plant designs achieve tritium self-sufficiency without requiring an inboard breeding blanket.

Copper-based ST Power Plant - Perhaps the most comprehensive study to date is the ARIES-ST study, which examined a 1000 MW electric fusion ST power plant conceptual design with $A = 1.6$, $R_0 = 3.2$ m, $\kappa = 3.4$, and $\delta = 0.74$ [65]. The normal conductor toroidal field coil magnet for an on-axis toroidal field of 2.1 T results in 329 MW of joule losses. The normal TF magnet design for this operational level is technically conservative, with $j_{TF} \sim 1$ kA/m² and low $B_{TF-MAX} \sim 7.4$ T. The PF coils are assumed to be superconducting to minimize the re-circulating power. The assumed ARIES-ST physics parameter range is quite aggressive however, with assumed limits to be 90% of the ideal MHD limits, namely, $\beta_T = 50\%$, $\beta_N = 7.4$, $\beta_P = 1.7$, $I_p = 28$ MA, and $f_{BS} = 0.96$. The fusion power of 2980 MW and the total thermal power of 3370 MW, with average neutron wall loading of 4.1 MW/m², are rather high. The re-circulating power fraction of 0.34 is significant due to the TF-coil ohmic losses, assuming a net plant efficiency of 0.3. The power core uses an advanced ‘dual-cooled’ breeding blanket, with a flowing PbLi breeder material and He-

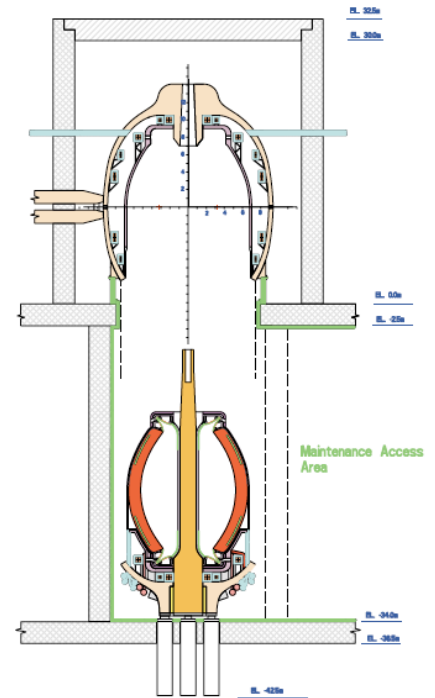


Fig. 2.12. Vertical maintenance scheme of ARIES-ST power core.

cooled ferritic steel structures that can achieve a thermal conversion efficiency of $\sim 45\%$. The inner TF copper center-post is protected by a 20 cm-thick metallic inboard shield that can reduce the center-post radiation damages. It can meet the class C waste disposal criteria, and operate at a sufficiently high temperature for nuclear heating to be recoverable as useful heat. A low voltage and high current TF power supply (8.5 V and 290 MW) for the single turn TF will be placed in the area just outside of the radiation shield to minimize Joule losses outside the TF coils. The estimated cost of electricity is 78.6 mill per kW electric hour, and similar to other fusion power plant designs. This is due to the relatively simple normal conductor magnetic engineering design, with very low TF current density in the center post. The possible advantages of the ARIES-ST power plant design is the ease of maintenance through vertical access as depicted in Fig. 2-12. This is quite similar in philosophy to that of the ST- FNSFs described in Sec. II-B. By lowering the entire plasma core, including blanket modules and plasma facing components, one can envision replacing the entire spent core with a new (or refurbished) one with a relatively short reactor down time. The removed spent core is to be refurbished during the period of power plant operation between core maintenance, which could be separated by a few years. In addition to the ARIES-ST study conducted in the US, there was an independent integrated modeling study at UK of important plasma physics issues related to the design of a steady-state spherical tokamak (ST) fusion power plant, similar to the ARIES-ST in plasma and device parameters [87].

Superconductor magnet-based ST power plant - The SC-ST design has the major advantage of potentially very lower recirculating power, which would reduce the fusion power output requirement by nearly 1 GW for a 1 GW-e power plant. Also, the SC magnet system could last for the entire plant life with small re-circulating power; however, PFC and blanket module maintenance may require more complex horizontal access between the TF magnets, as with conventional aspect-ratio tokamak reactors. The SC-ST TF coils may provide larger horizontal access space because of the overall low toroidal field coil current, making the TF magnet outer leg less bulky. Because of the larger size and lower recirculating power, the plasma pressure for the SC-ST maybe significantly lower, i.e., $B_{T0}^2 \beta_T \sim 120$ instead of ~ 200 . This could substantially reduce plasma performance requirements. The lower neutron flux and divertor heat loads also reduces the design requirements of the PFCs and blanket components. Because of the

neutron shield, a central ohmic solenoid can be incorporated into the design to facilitate plasma start-up. The key challenge is to come up with the most compact SC-ST design that is realistically achievable. The technology of SC magnets has been advancing rapidly, including high temperature SC's which may allow higher magnetic fields, and coil current densities. Another important consideration is how to minimize the inboard neutron shield thickness, while providing a sufficient TBR (tritium breeding ratio). Here we introduce two SC-ST studies conducted in Japan and Korea.

Japanese SC-ST power plant studies - The superconductor (SC) magnet tokamak DEMO study in Japan has looked into the lower aspect ratio design VECTOR, with $A = 2.3$, $R = 3.2$ m, and a “Slim CS” with $A = 2.6$, $R = 5.5$ m to reduce the reactor size. More recently, an SC-ST power plant design called JUST was proposed, where $A = 1.8$, $R_0 = 4.5$ m, $B_{TF} = 2.36$ T, $\kappa = 2.5$, and $\delta = 0.35$. A JUST device schematic is shown in Fig. 2.13 [88]. The SC-TF is Ni_3Sn , and $B_{max} = 12.8$ T with $j_{TF} = 2800$ A/cm², which is well within present-day Ni_3Sn technology. The neutron shield is 80 cm thick for this design. The plasma operating parameters are $\beta_T = 22\%$, $\beta_N = 7.2$, $I_p = 18$ MA, and $f_{BS} \sim 1$ which is considerably less stringent than ARIES-ST with much lower I_p and β_T . The total fusion power is 2.2 GW for net 1 GW electric because of low recirculating power. The plasma pressure $B_{T0}^2 \beta_T \sim 122$, which is low compared to the ~ 200 required for ARIES-ST. This is because of the 2x larger volume and 1.5x lower fusion power output. The design has a central ohmic solenoid coil, wound over the TF magnet, which can bring the plasma current to the ~ 4 MA level and alleviate much of the start-up problem. Further current ramp up is to be done by ECCD and bootstrap current overdrive. To alleviate the divertor heat load problem, the JUST design incorporates an innovative liquid lithium divertor.

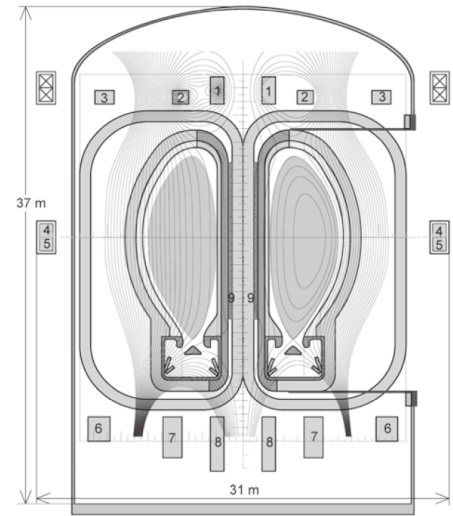


Fig. 2.13. Schematic of JUST device with liquid lithium divertor.

Korean SC-ST reactor studies - With availability of high-field high-temperature superconductor magnets in the recent years, a compact, 1000 MWe-class SC-based ST power plant design with

minimum re-circulating power was examined with a self-consistent system analysis code for $A = 1.5 - 2.0$ [89]. The high-temperature superconducting magnets with high critical current density opens up the possibility of a fusion power plant with compact size and small re-circulating power, simultaneously with low aspect ratio. A central ohmic start-up solenoid is envisioned. A model of the radial build is shown in Fig. 2.14. As in all ST based systems, the tritium blanket is not placed in the inboard region. The use of WC (tungsten carbide) appears to be effective, since in addition to its good neutron shielding properties, it can reflect neutrons toward outboard tritium breeding blanket. This can help maximize the tritium breeding ratio (TBR), so tritium self-sufficiency therefore appears to be possible with only an outboard blanket.

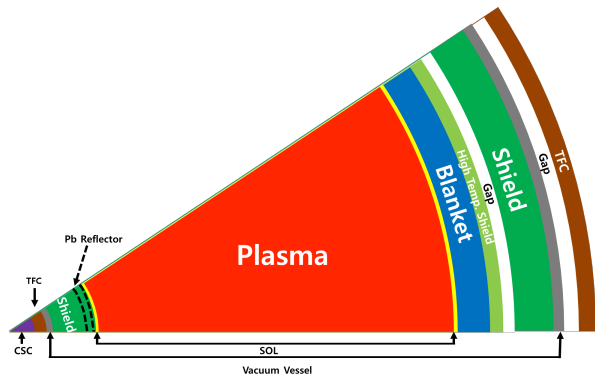


Fig. 2.14. Radial build of a compact superconducting coil ST reactor.

III. Macrostability

A. Introduction and Motivation – The physics of macrostability has been a success story for the long history of tokamaks and STs. The ideal MHD model has been used successfully to predict plasma equilibrium and its stability properties for all types of tokamaks, large and small [21, 22]. Over the years, MHD stability codes have been improved to take into account various physical processes to support experiments. For example, the experimental quest for high beta regimes in NSTX has been guided by ideal and resistive MHD theory, and more recently, by theoretical kinetic MHD models using, for example, the MISK code [90]. As noted in Sec. II, high β_T is needed for high fusion reactivity, and high β_N is needed for the high bootstrap current fraction required for non-inductive plasma sustainment. The highest β_T regime in the STs has been accessed with high values of I_p/aB_{T0} as shown in Fig 1.3 . The trend observed in tokamak experiments are also shown in the figure. The slope is the $\beta_N \equiv \beta_T aB_{T0} / I_p$. The START device obtained high $\beta_T \leq 40\%$, which is consistent with theoretical expectations [91, 92]. NSTX was also able to achieve the high β_T regime with $\beta_T \sim 35 - 39\%$ at $I_p \geq 1$ MA [93]. The ultra-low-aspect-ratio device PEGASUS has obtained the relatively high $\beta_T \sim 25\%$ with Ohmic heating alone [6]. Ultra-high β_T was also obtained in a transient manner in the smaller TS-3 experiment [14], where merging ST plasmas inducing reconnection-based anomalous ion heating that resulted in high β_T (see Sec. IV.D). However, these very high β_T results were obtained by going to high I_p/aB_{T0} (i.e., lower β_N) which has only a small bootstrap current fraction. While it is important to demonstrate high β_T for high fusion power density for a given magnetic field, as discussed in Sec. II, it is critical to access high bootstrap current fraction f_{BS} for the fully non-inductive steady-state reactor operation needed for FNSF and a DEMO Power Plant. To achieve simultaneously high β_T and f_{BS} requires access to high β_N and high κ as shown by Eq. (3), i.e., $f_{BS} \propto A^{-0.5} (1+\kappa^2) \beta_N^2 / \beta_T$. Therefore, the quest to achieve stable high β_N and κ has become a central topic for ST macrostability research in recent

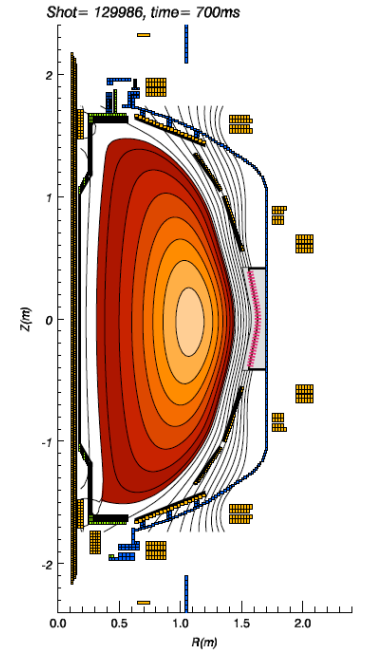


Figure 3.1. Reconstruction of a typical high $\kappa \sim 2.8$, high $\beta_n \sim 1.8$ equilibrium.

years, which is similar to the goal for the advanced tokamak reactor concept [94]. One of the readily controllable plasma parameters is κ . Most future ST devices are designed with relatively high $\kappa \geq 2.5$ to generate significant bootstrap current. Figure 3.1 shows the equilibrium cross-section for such a discharge in NSTX, with $\kappa \sim 2.7$ at $\beta_N \sim 5.5$ maintained for $0.5 \text{ s} \sim 2\tau_{\text{CR}}$ [95]. These discharges achieved high non-inductive current fractions, i.e., $f_{\text{NI}} \sim 65\%$ and $f_{\text{BS}} \sim 50\%$. Because of low ohmic flux consumption ($V_{\text{loop}} \sim 0.1 - 0.2 \text{ V}$), the discharge duration of the high elongation discharges is now determined by the heating limits of the TF coils on NSTX. For β_T and β_N optimization, detailed macrostability or magneto-hydrodynamic (MHD) calculations were performed. They showed that with optimized plasma pressure and current profiles, it is indeed possible to produced reactor relevant plasmas with high β_T and high β_N values simultaneously. In order to reach high β_N , there are a number of obstacles, such as resistive wall modes (RWMs), as β passes through the so-called no-wall beta limit and approaches the so-called with-wall or ideal MHD beta limit as described in Sec. B. Neoclassical tearing modes (NTMs), which are considered to be a highly important research topic for tokamaks, turned out to be not as serious for STs due to its higher beta and higher q operations as discussed in Sec. C. In addition, the importance of error fields was recognized for MHD stability, particularly at high beta as described in Sec. D. ST MHD research also contributes strongly to the plasma disruption physics research discussed in Sec. E.

B. High β_N Regime – Because of its relevance to attractive reactors, recent ST MHD research has been focusing on access to the reactor-relevant high β_N regimes. As described in Sec. II, the high β_N regime is necessary for a steady-state tokamak reactor design with simultaneously high β_T and f_{BS} . One of the highly significant ST research accomplishments is the attainment of high β_N regimes near the

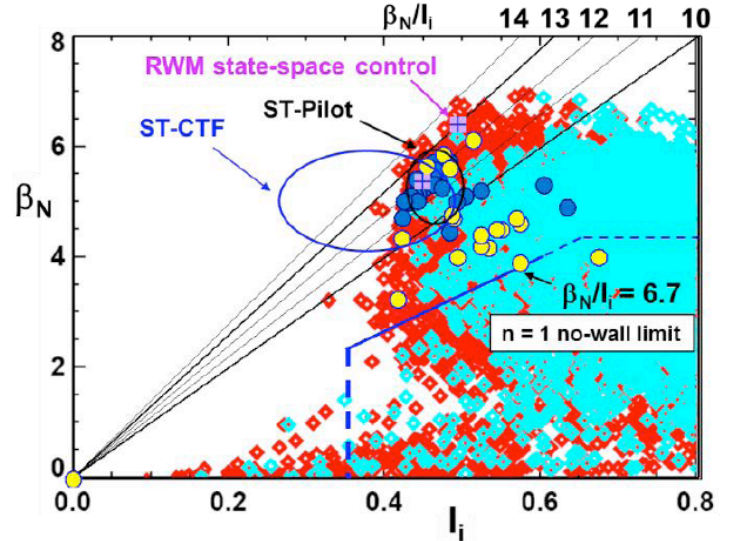


Fig. 3.2. High β_N , low I_i operational space in NSTX. Red/cyan points indicate plasmas with/without $n=1$ active RWM control. Blue circles indicate stable long pulse plasmas with active RWM control; yellow indicates disruptions.

ideal wall limit, through RWM stabilization and control as described in this section. In Ref. 59, it was shown theoretically that the high β_N regime for STs is indeed accessible through plasma shape and plasma profile control. Another insight for higher β_N can be seen in Fig. 3.2, where the achieved β_N is plotted as a function of plasma internal inductance l_i [48, 96]. The l_i is a measure of plasma current profile where the broader plasma current profile, the lower the value of l_i . Lower values l_i for $\beta_N > 1-2$ are not accessible since the plasma is generally more kink unstable. Yet, for high performance reactor operation, it is generally desirable to access high values of β_N/l_i . As one can see from Fig. 3.2, the NSTX database extends to $\beta_N \sim 7$ and $\beta_N/l_i \sim 13.5$, well above the normal expected tokamak beta limits of $\beta_N \sim 4$ and $\beta_N/l_i \sim 6$. The NSTX database now intercepts the higher l_i portion of the planned operational ranges for ST-FNSF and ST pilot plants. It should also be noted that the achieved β_N now exceeds the computed β_N no-wall limit (shown by the blue line) or β_{N-NW} by up to a factor of two. At lower $l_i \sim 0.38$, the plasma can reach the current-driven ideal kink stability limit, so that passive or active kink and resistive wall mode (RWM) stabilization is critical as discussed in the following sections. As mentioned in Sec. I, the plasma is not necessarily most unstable at highest beta values in NSTX [25]. Instead, disruptions occurred less frequently at high values of β_N/l_i , which is also supported by active MHD spectroscopy diagnostic measurements [26]. They show that resonant field amplification (RFA) (a measure of plasma going kink unstable) decreases at higher values of β_N/l_i as shown in Fig. 1.4. This favorable stability property at high values of β_N/l_i bodes well for the prospects for a ST-FNSF and ST-power plant.

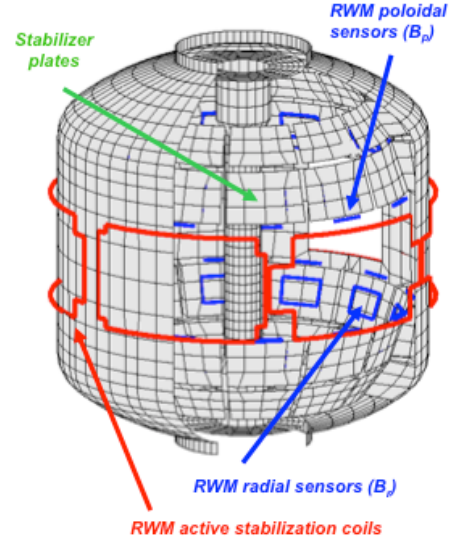


Fig. 3.3. Diagram of NSTX showing internal B_r and B_p sensors, passive stabilizing plates and ex-vessel 3-D control coils.

C. Resistive Wall Modes (RWM) – As the plasma beta increases, ideal MHD instabilities such as internal pressure and current driven kink/ballooning instabilities set the β limit. A so-called ‘no-wall’ limit, β_{N-NW} , occurs in devices without a passive stabilizing conducting structure. The MAST device has a conducting structure, i.e., a vacuum vessel, relatively far away (see Fig.

1.12) so the limiting β_N is likely to be closer to the β_{N-NW} limit. The NSTX device specifically installed a relatively thick closely-fitted copper passive stabilizing wall, and six ex-vessel control coils and related magnetic B_p and B_r sensors, as shown in Fig. 3.3 [97, 98]. They were used to investigate the RWM stabilization in order to exceed the β_{N-NW} limit and approach the so-called ‘with-wall’ or ideal limit, β_{N-WW} . The global kink/ballooning MHD instabilities can be stabilized with the presence of a nearby conducting wall. In order for wall stability to be effective, the mode structure must allow coupling of the perturbed field to the conducting wall. The ST magnetic field geometry results in the MHD instability mode structures significantly different from those in advanced tokamaks. In Fig.

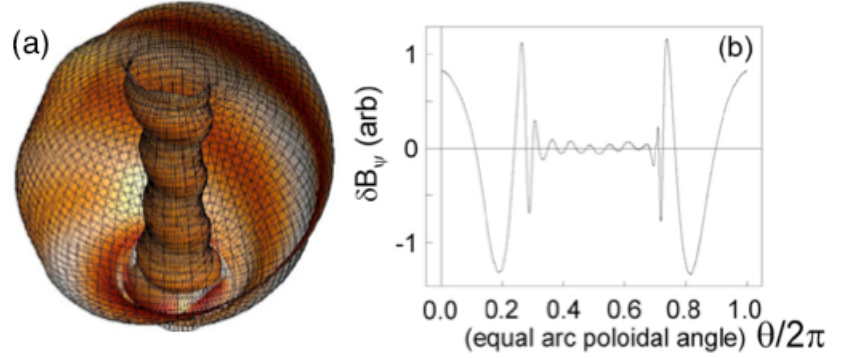


Fig. 3.4. NSTX RWM Structure. (a) DCON computer normal perturbed field for unstable RWM. (b) Theoretical computed poloidal variation of the $n=1$ RWM field at the plasma boundary as a function of poloidal angle.

3.4(a), the DCON computed perturbed magnetic field normal to the outer most flux surface is shown [96]. The computation uses an EFIT experimental equilibrium reconstruction, and the illustration includes the sum of the $n = 1-3$ components scaled to the measured RWM sensor amplitudes and relative phases. The perturbed field amplitude shown has been scaled up by a factor of 10 to clarify the mode shape. Based on the measured field amplitudes for $n = 1-3$, an estimate of the real space displacement of the mode is about 3 cm at the outboard mid-plane. This is consistent with estimates of the mode amplitude based on fast visible light camera images, and electron temperature measurements from past experiments obtained by shifting relative timing of the two independent 30 Hz Nd:YAG lasers used by the Thomson scattering diagnostic [99]. Figure 3.4 (b) shows the poloidal variation of the mode field perturbation at the plasma edge. Here, the poloidal angle θ is 0 at the outboard mid-plane and π at the inboard mid-plane. The mode has a strong ballooning character, being significantly larger on the outboard portion of the plasma.

RWM stabilization with plasma rotation – As the plasma approaches and exceeds the so-called no-wall beta limit, resistive wall modes (RWMs) (typically $n=1, 2$, and 3) can be excited

as described above. The predominant mode is $n=1$, which is a kink-like mode with a particularly fast growth rate. The presence of a nearby conducting wall slows the mode growth. Earlier studies indicate that the mode growth can be suppressed if the plasma rotates sufficiently fast to make the nearby wall look like an ideal (superconductor-like) conductor for the growing modes. Plasma rotation frequencies normalized to the Alfvén frequency (ω_ϕ/ω_A) of a few per cent relative to the mode rotation frequency can passively stabilize the RWM in theory [100]. Rotational RWM wall stabilization has been extensively studied on tokamaks and STs with the NBI injection, which imparts significant momentum to spin the plasma. Early RWM stabilization models, which relied solely

on plasma collisionality as a stabilizing energy dissipation mechanism was quite successful in explaining much of the RWM stabilization physics with plasma rotation. As shown in Fig. 3.5, RWM stabilization with a rotation frequency larger than $\Omega_{crit} \equiv \omega_A/(4q^2)$ is clearly demonstrated [96]. While this simple rotational RWM stabilization had many successes, it had a significant

shortcoming. On NSTX, some RWMs are more unstable with higher rotation speed, contradicting the model as shown in Fig. 3.6 [101]. As shown in the figure, the RWM wall stabilization is not a simple function of the rotation speed. The RWM stabilization models which solely relied on plasma collisionality as a stabilizing energy dissipation mechanism predicts reduced stability at reduced collisionality, a concern

for the future reduced collisionality plasmas. A breakthrough in the understanding of RWM physics was made through the inclusion of kinetic effects [102]. This kinetic model described below was successful in resolving the puzzling stability behavior with rotation observed in

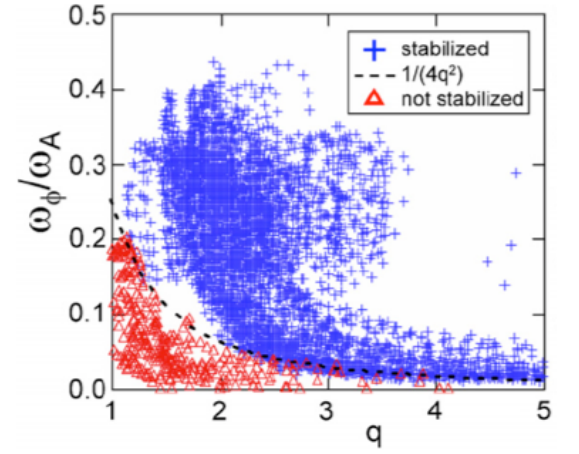


Figure 3.5. Observed kink/RWM stability versus local ω_ϕ/ω_A , parametrized by local q value. Ω_{crit} is well defined by the Bondeson–Chu expression $\omega_A/(4q^2)$.

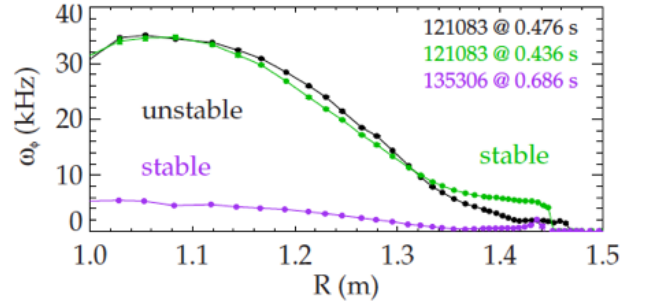


Fig. 3.6. Toroidal plasma rotation profiles for NSTX, showing a high rotation stable case, an intermediate rotation unstable case, and a low

NSTX. But it also has important implications for next-step devices operating at reduced collisionality such as ITER/FNSF/DEMO.

Kinetic effects in RWM stabilization with plasma rotation – The NSTX RWM research effort

has established a new understanding of RWM stability by making quantitative correlations between experiments reaching the mode marginal stability point and kinetic RWM stabilization theory, which takes into account various plasma resonance frequencies [90, 103]. While the effects could be complex, some clarity has emerged on the role of resonance and off-resonance with collisionality as shown in Fig. 3.6 [104]. Figure 3.7 shows

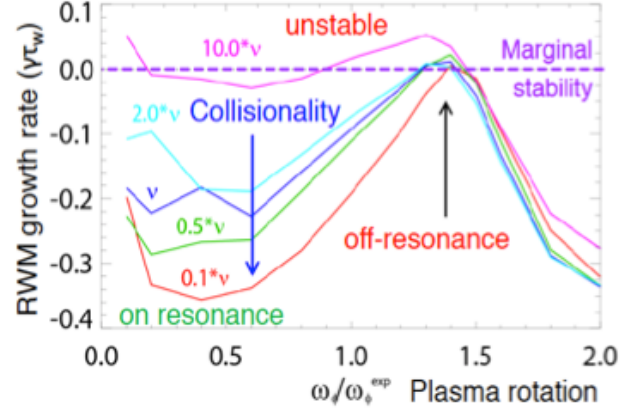


Fig. 3.7. MISK computed kinetic RWM $n = 1$ stability vs. plasma rotation at various levels of scaled collisionality.

normalized growth rate vs. scaled experimental rotation for a given NSTX discharge. One can clearly see that a stabilizing resonance with the ion precession drift frequency occurs at low rotation ($0.2 < \omega_\phi/\omega_\phi^{\text{exp}} < 0.7$) and with bounce and transit frequencies at higher rotation

($\omega_\phi/\omega_\phi^{\text{exp}} > 1.6$), while in between these extremes there exists a narrow range of off-resonance rotation profiles with more unstable marginal stability ($1.3 < \omega_\phi/\omega_\phi^{\text{exp}} < 1.5$). In general, with other parameters equal, the reduced collisionality expected in future devices can enhance the RWM stability of on-resonance plasmas, while leaving the reduced stability of off-resonance plasmas roughly unchanged. This favorable collisionality behavior is in contrast to the predictions of previous fluid models [100],

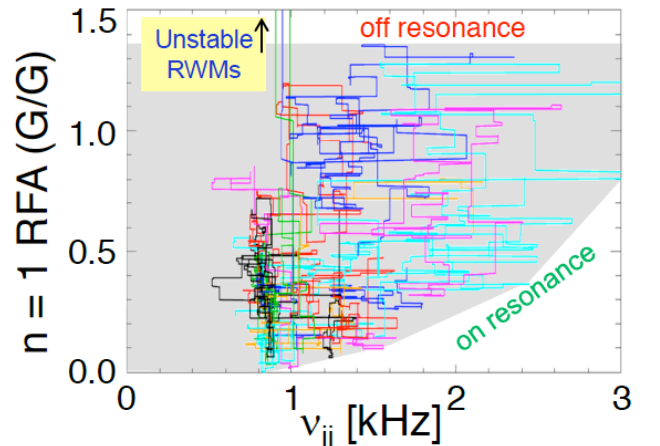


FIG. 3.8. $n = 1$ RFA amplitude vs. ν_{ij} , showing a relatively large change at low RFA (“on resonance”) vs. almost no change at high RFA.

where decreased collisionality was thought to be universally destabilizing. In conjunction with kinetic resonances, collisions play a different role that can appear contradictory when compared to simpler models. This is because reduced collisions lower the collisional dissipation that is important when plasma rotational resonances are not present, but also decrease the damping of the competing resonant kinetic stabilizing effects, allowing them to be more powerful. Experiments which use $n = 1$ resonant field amplification to measure RWM stability [26] indicate the expected gradient in RWM stability for plasmas with high $5.5 < \beta_N/l_i < 13.5$ (as most are above the $n = 1$ ideal no-wall stability limit). Figure 3.8 shows the trajectory of the RFA amplitude over the entire discharge for 20 shots, over which the ion collisionality, n_{ii} , is varied by a factor of 3.75. The theoretically-expected gradients in kinetic RWM stability are generally reproduced by the upper/lower boundaries of the $n = 1$ RFA amplitude. At high $n = 1$ RFA amplitude (the upper boundary), the plasma is off-resonance, and there is almost no change in RWM stability (indicated by the $n = 1$ RFA amplitude) vs. v_{ii} . At low $n = 1$ RFA amplitude (the lower boundary), the plasma has greater stabilization by kinetic resonances, and there is a clear increase in RWM stability (i.e., a decrease in $n = 1$ RFA amplitude) as v_{ii} is decreased.

Active RWM stabilization at low plasma rotation – While the physics understanding of RWM passive stabilization has made impressive progress in the recent years, there is significant benefit to having active stabilization of RWMs, since they could still become unstable due to various kinetic effect as shown in the previous sections. For example, RWM active stabilization can be used when ω_ϕ passes through the unstable band of the off-resonance region, or for extremely low plasma rotation operation expected for ITER and DEMO. Active stabilization is typically realized by a feedback control loop consisting of magnetic sensors capable of detecting low frequency (i.e., resistive wall time scale $\sim 1/\tau_W$)

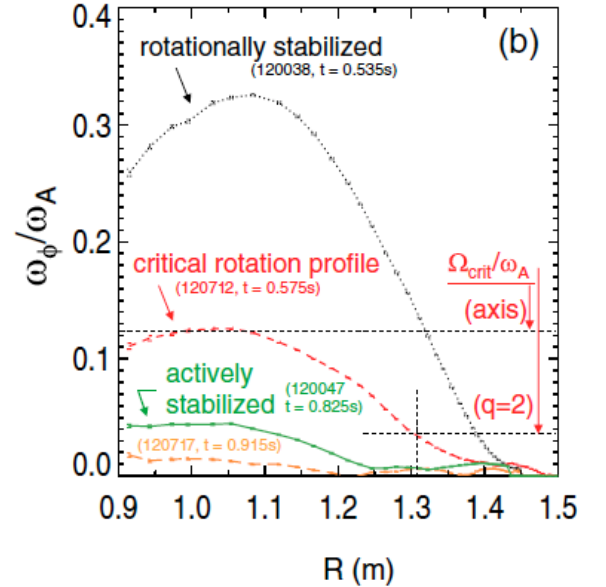


Fig. 3.9. Actively stabilized plasma equilibrium and rotation profiles. Shown are ω_ϕ/ω_A vs. R for plasmas that are rotationally stabilized, at critical rotation profile, and actively stabilized below Ω_{crit} .

modes, a set of control coils to provide magnetic fields in response to the detected modes (typically to try to cancel the dominant $n=1$ magnetic perturbation), and a control algorithm that determines the form of the response. On NSTX, active feedback control has been incorporated into routine high beta operations and has been steadily improved. The RWM sensor signal from poloidal and radial field RWM sensor arrays (48 coils in total shown in Fig. 3.3) were used to achieve feedback stabilization [96]. An example of RWM feedback stabilization of a low-rotation high-beta plasma is shown in Fig. 1.10 [34]. Both discharges are otherwise similar with constant $P_b = 6.3$ MW. The plasma without active stabilization has $\beta_N = 4.1$ as the rotation frequency $\omega_r/2\pi$ drops to below 4 kHz. At this time, RWM passive stabilization becomes insufficient, and the $n = 1$ RWM becomes unstable and β_N collapses. In contrast, the plasma with active stabilization does not suffer an unstable RWM, and continues to increase in β_N up to 5.6 and β_T up to 19.4% as ω_r continues to decrease to $\omega_r/\Omega_{\text{crit}} = 0.2$ near $q = 2$. The RWM is actively stabilized above $\beta_N^{\text{no-wall}}$ and below Ω_{crit} for significantly long durations exceeding $90/\gamma_{\text{RWM}}$ and seven τ_E , limited only by the toroidal field coil pulse length. The corresponding rotation profiles are shown in Fig. 3.9. Combined $n = 1$ RWM feedback control and β_N feedback control were used to generate more reliable high pulse-averaged β_N with low levels of fluctuation at various levels of plasma rotation. These active control system improvements have led to high success rates in stabilizing low l_i plasmas with near record ratios of β_N/l_i between 12 – 13.

More recently, an advanced RWM state space controller was implemented. It uses a reduced order model of the 3D conducting structure of NSTX (e.g., discrete passive conducting plates and a vacuum vessel including the large NBI port), and $n = 1$ ideal plasma instability

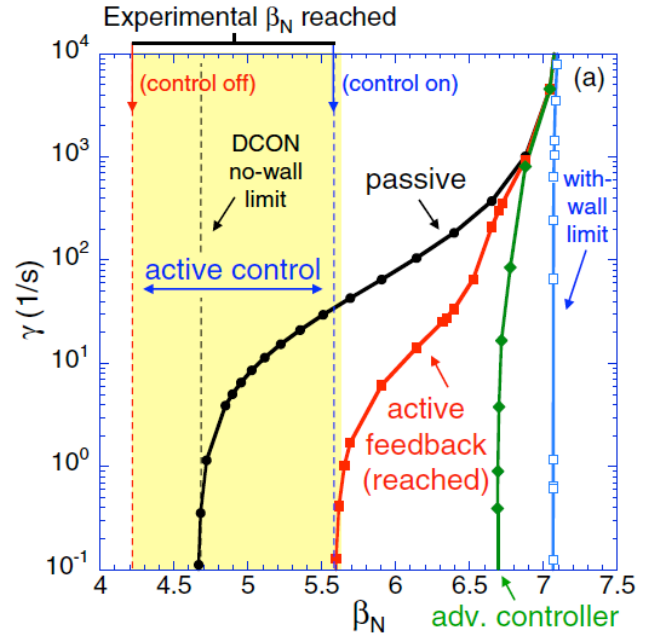


Figure 3.10. Experimental β_N reached with $n = 1$ active control (shaded region) compared with theoretical feedback control performance using proportional gain and with an advanced state-space controller.

eigenfunctions computed by the DCON MHD stability code, with variable phase to allow the controller to track mode rotation. Of particular interest is the predicted active control improvement for high beta operation with conducting structures (e.g., vacuum vessel in NSTX [105] and blanket in ITER [106]), using the advanced RWM state space controller. This ability is especially important for burning plasma devices, as it will be necessary to move such coils farther away from the plasma than in present experiments, and install shields to protect such coils from neutron damage. Implementation of this system on NSTX and supporting studies have shown the advantage of the state derivative feedback approach [105]. The theoretical performance of the controller for plasmas with an insignificant level of passive stabilization is shown in Fig. 3.10. Using the plasma eigenfunction and wall response appropriate for an equilibrium near the no wall limit, the plasma can be controlled up to $\beta_N \sim 5.6$. Using input for an equilibrium near the with-wall β_N limit β_{NWW} , the plasma can be controlled up to $\beta_N \sim 6.7$ close to $\beta_{NWW} \sim 7.1$. In NSTX, plasmas limited only by coil heating constraints have exceeded $\beta_N = 6.4$, $\beta_N/I_i \sim 13$ using RWM space controller as indicated in Fig. 3.2 [26]. The advanced RWM state space controller is also predicted to significantly improve ITER performance [106]. Overall, the active n=1 RWM stabilization (plus other improvements) appears to be quite effective in maintaining high beta operation near the ideal MHD stability limit.

D. Neoclassical Tearing Modes - The Neoclassical Tearing Modes (NTMs) are magnetic islands at the resonant surfaces $q = m/n$, destabilized by a helically-perturbed bootstrap current [107]. Within the magnetic island, the plasma pressure tends to equilibrate, and the reduction of the pressure gradient reduces the bootstrap current within the island. The resulting magnetic perturbation term proportional to $A^{-1/2} \beta_p$ (see Eq. 3) tends to cause the magnetic islands to further grow (i.e., become unstable) for positive values of magnetic shear length $L_q = q / (dq/dr)$, which applies to normal ST/tokamak operations. The NTMs are therefore usually stable for reversed magnetic shear and also for most stellarators. Since NTMs were observed at relatively low plasma beta in tokamaks and NTMs with growing magnetic islands degrade plasma confinement in present day tokamaks and possibly in ITER [108], exploring these modes and techniques for their suppression have been a high priority in large aspect ratio tokamaks. However, NTM research in STs has been relatively limited. Early NTM observations in MAST [109] and NSTX have been reported [110]. A computational study for low-aspect ratio

equilibria showed stabilizing effects due to shaping / magnetic shear and curvature effects [111, 112]. The scarcity of NTM studies in STs is perhaps due to the fact that NTMs have thus far not played a significant role in determining high beta stability and confinement in ST experiments. This lack of NTM activity in STs may be due to the generally more stable characteristics of their plasmas, namely, strong shaping, high beta, and higher q operation, all of which tend to reduce the NTM drive as expected theoretically and computationally. NTMs are stabilized by the so-called classical stability index Δ' (which is negative), and by good average magnetic field curvature which scales as $(q^2-1) \beta$, but destabilized by the helically perturbed bootstrap current term. A simplified NTM equation can be written as,

$$(\tau_R/r^2) dw/dt \sim \Delta' - 6.35 (q^2-1) (L_q^2/rL_p) \beta / w + (L_q/L_{pe}) (A^{-1/2} \beta_p) / w \quad (4)$$

where $L_p = -p / (dp/dr)$ is the pressure scale length, and w is the characteristic island width [107]. The second curvature term in Eq. (4) is usually not significant for conventional tokamaks, but can be important for STs with generally higher operating q and β . The third helically perturbed bootstrap current term in Eq.(4) is relatively insensitive to aspect ratio. Recently, a comparison experiment between NSTX and DIII-D indeed showed the magnetic curvature stabilizing term to be important for NSTX parameters [113]. Overall, NTMs in ST plasmas therefore appear to be more stable than in conventional tokamaks. Another important feature is the projected high f_{BS} ST reactor q profile [shown, for example, in Fig 2.4 (a)] that is monotonic and relatively high $q \geq 4$. The absence of low q resonances eliminates many of dominant resonance surfaces (i.e., magnetic islands) that are responsible for NTM drive. So, while we cannot completely rule out NTM-related issues for STs, the results thus far are quite encouraging in that NTMs may not be a serious obstruction for future ST facility operations. The longer-pulse, lower-collisionality discharges in NSTX-U and MAST-U will offer a good experimental test bed for the future NTM studies.

E. Effects of 3D Fields – The effect of 3-D fields has been an active area of research for both tokamaks and STs. In this section, the effects of non-axisymmetric fields on the ST plasma performance are discussed. The 3-D resonant magnetic perturbation on ELM control will be discussed in Sec. VI.D. A dramatic change in NSTX plasma performance was observed when

machine improvements were made in 2002 as shown in Fig. 3.11 [114]. As seen in Fig. 3.11, the achieved peak β_T dramatically increased from the 20% range to the 35% range and β_N from the 4 range to 6, after poloidal field coil realignment and the installation of a bakeout system which generally improved vacuum condition. Prior to the machine improvements, the plasma beta limit was set primarily by confinement degradations associated with low n ($n=1, 2$) tearing mode activity due to the poloidal coil misalignment. It should be noted that the $n=1$ error field reduction was also accompanied by high temperature bakeout during this period, so the operational limit improvements were attributed to both error field and vacuum condition. Additional error field compensation for the intrinsic $n=3$ fields generated by imperfections in the PF coils, and dynamic compensation for $n=1$ error fields arising from coil movement, resulted in further plasma performance improvements [115]. The dynamic error fields generated during plasma operation were detected by the error field sensors, and compensated by using the 3-D control coils shown in Fig. 3.3.

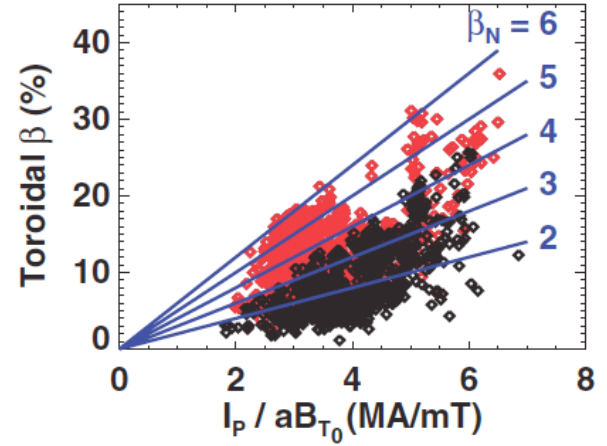


Fig. 3.11. Database plot of β_t vs. $I_p/(aB_{T0})$, showing that the data are bound by a line with slope $\beta_N \sim 6.0-6.5$ in NSTX. Discharges are divided into experimental data in 2002 (red) vs earlier results (black).

An important physics result associated with 3-D fields is the production of the neoclassical toroidal viscosity (NTV) [116]. The NTV is caused by the interaction of the plasma with 3D fields in a tokamak. The NTV magnetic braking, for example, can be a useful tool to control the toroidal rotation for MHD mode control. Comparison of the measured dissipation of plasma angular momentum caused by the externally applied $n=3$ non-resonant fields to the theoretical

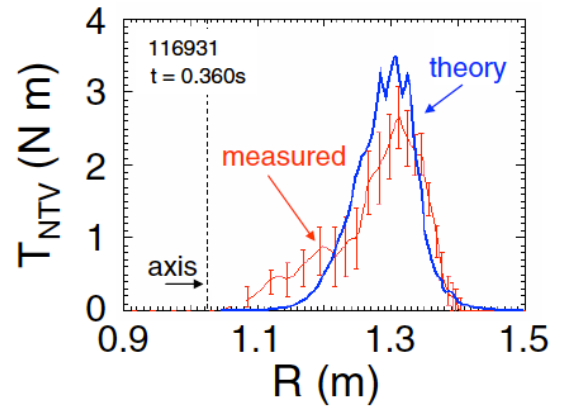


Fig. 3.12. Comparison of measured $d(I\Omega_e)/dt$ profile to theoretical integrated NTV torque for an $n = 3$ applied field configuration.

NTV torque profile is shown in Fig. 3.12 [117]. This good agreement was obtained when the effect of toroidally-trapped particles is included. The NTV with an applied $n=2$ non-resonant field was also investigated, and showed broader rotation damping profile than the $n=3$ case [118]. This is consistent with theoretical expectations. The observed increased plasma rotation damping rate with Ti during non-resonant magnetic braking is found follow theoretical prediction of $\tau_{NTV} \sim T_i^{5/2}$. To achieve a more precise picture of the NTV physics, including effects such as non-ambipolar transport, a new particle orbit code is being developed [119].

Another important advance in understanding the role of error fields was made with the new ideal perturbed equilibrium code (IPEC) [120], which can compute the 3-D perturbed tokamak plasma equilibrium with high resolution. IPEC can explain the successful cancellation of error fields by the relatively simple 3-D control coils in NSTX and DIII-D. This theoretical approach has also been applied to the understanding of NTV physics by including bounce and precession resonances and the use of the 3-D perturbed magnetic field lines [121]. Another example of the value of IPEC can be seen in Fig. 3.13, where the locking threshold values of the external error field currents (applied by the 3-D control coils) versus density for ohmic and NBI-heated high β plasma are plotted as a

function of the plasma density [122]. As shown by Fig. (a), the usual linear density correlation can be seen for ohmic discharges, but it is no longer valid for high β plasmas when the external measures are used. By using the total resonant fields calculated with IPEC instead of external currents or external fields, the linear correlation is regained for both ohmic and NBI-heated plasmas as shown in Fig. (b). This resolution comes from plasma amplifications by a factor of 2–3 in high density and high β plasmas.

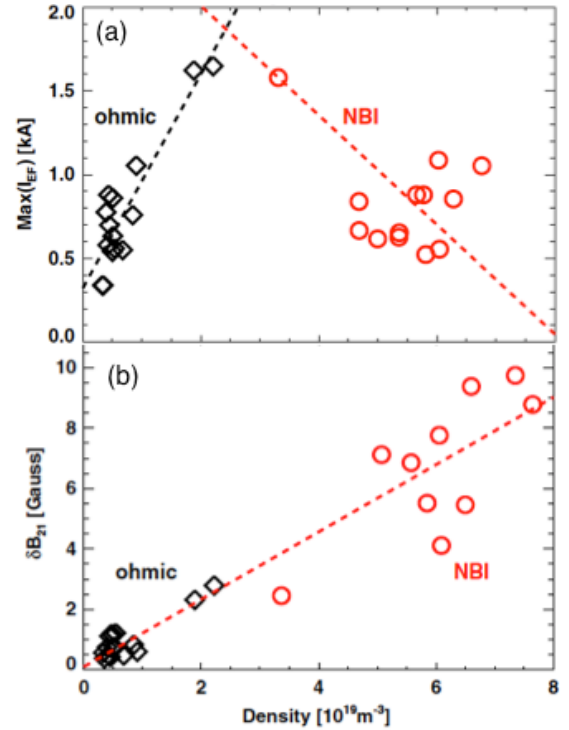


Figure 3.13. (a) Summary of locking threshold measurements in external currents versus density for NBI-heated high β cases (red circles) as well as ohmic plasma cases (black diamonds). (b) Revision of (a) using total resonant fields at the $q = 2/1$ surface utilizing IPEC.

F. Disruptions - Major disruptions have been identified as the most serious issue for the tokamak reactor approach. While not all of the major disruptions are triggered by MHD-related events, the vast majority of them are. In particular, the non-axisymmetric halo currents generated in the in-board (high field) side by a disruption are problematic for tokamak reactors due to the very large $j \times B$ force on PFCs and the first wall.

For example, if a 2 MA halo-current (20% of total plasma current of 10 MA) is generated in the inner wall at a 10 T magnetic field, there will be about 2,000 tons/m of dynamic radial force imparted on the current carrying vacuum chamber wall and PFC components. If the force is non-axisymmetric, the resulting large localized stress could cause unacceptable reactor chamber wall failure. The large electric fields generated by disruptions can accelerate electrons to relativistic energies, and these electrons can also cause damage

to the device PFCs [123]. A large database of disruption rate and disruptivity statistics spanning 2006 - 2010 of NSTX operation has been analyzed [25]. Figure. 3.14 shows disruptivity as a function of β_N and q^* , pressure peaking factor $F_p \equiv p(0)/\langle p \rangle$, plasma shaping factor $S \equiv q_{95}I_p/aBT$, and l_i . Strikingly, and consistent with the dedicated stability experiments described above, no clear increase is found in disruptivity at increased β_N and $l_i < 0.8$. Significant increases in disruptivity are found for $q^* < 2.4$, at low plasma shaping, and at high values of F_p , and l_i , each of which are generally expected. Increased S , and decreased F_p typically beneficial for stability, are also shown to yield reduced disruptivity in this analysis. The disruptivity trend is quite encouraging, as future ST FNSF and Power Plant operating designs trend toward higher β_N , increased S , and decreased F_p and l_i .

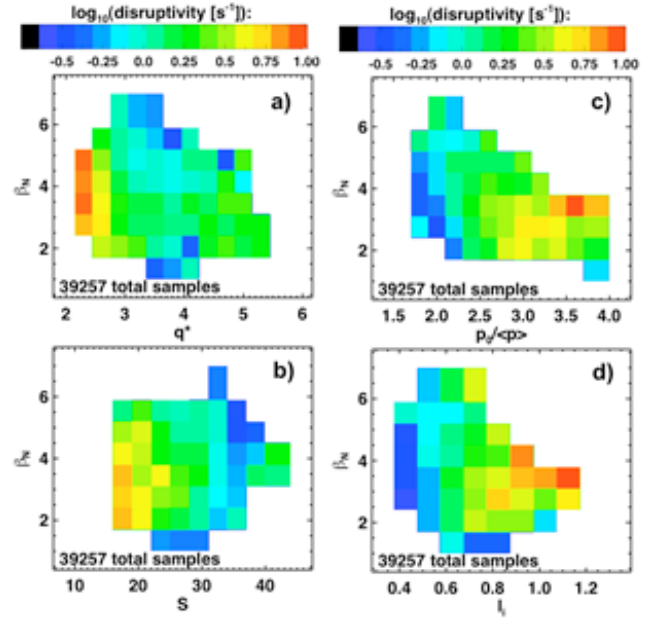


Fig. 3.14. Disruptivity as a function of β_N and a) q^* , b) shape factor, c) pressure peaking, and d) l_i .

An extensive NSTX database study has been conducted to determine the detectability of disruptions based on multiple-input criteria [124]. Quantitative evaluation of the levels of measured input, including low frequency $n = 1$ RWM amplitude, neutron emission compared to a computations from a rapidly-evaluated slowing-down model, ohmic current drive power, and plasma vertical motion (all able to be evaluated in real-time) are determined to maximize disruption detectability, while minimizing false positives. A false positive occurs when the disruption alarm is declared more than 0.3 s in advance which is essentially a false warning. Results illustrate that no single diagnostic dominates the detection algorithm; a combination of signals is required. In total, 17 threshold tests are computed and a weighted sum is evaluated every 2 ms for ~ 1700 disruptive discharges tested. A flag noting that a disruption is imminent is set when the single weighted sum is sufficiently large. This approach has shown high success. Figure 3.15 illustrates a histogram of the warning times that this approach finds (with a reset time of 25 ms and eight points required for a positive flag). A total of 98% of the disruptions are flagged with at least 10 ms warning, with $\sim 6\%$ false positives. The majority of false positives are due to near-disruptive events. The number of missed warnings found is largely due to locked modes and RWMs. This multiple-input criteria approach is an innovative promising approach for maximizing the reliability of disruption detection for tokamaks and STs. Such a disruption avoidance system is planned for use in the upcoming operation of NSTX-U.

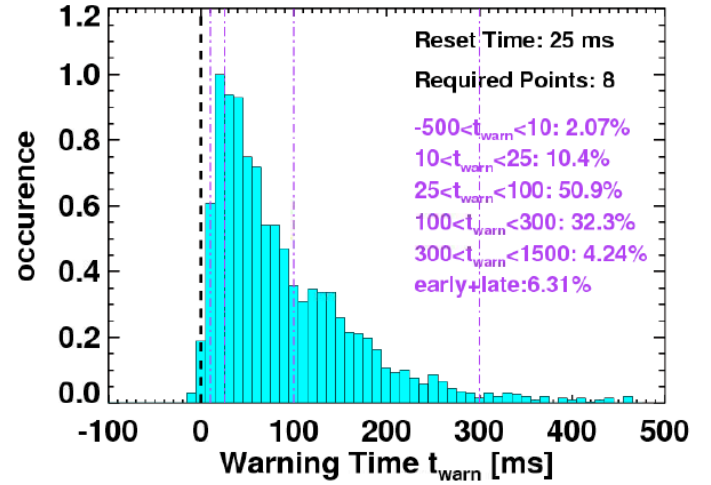


Fig. 3.15. Histogram of warning times computed for 1700 disruption discharges.

IV. Solenoid-free Start-up

A. Introduction and Motivation: Because of the use of a minimally-shielded single turn copper TF center-post without a central solenoid for future compact ST devices such as FNSF, it is critically important to develop a reactor compatible solenoid-free plasma start-up and current ramp-up technique. A multi-turn ohmic solenoid with electrical insulating material without sufficient shielding, which is not possible for compact STs, is considered to be unacceptable for the fusion neutron environment. An inboard ohmic solenoid also occupies the very valuable central region of the device, and installing it would increase the inboard major radius of the plasma, thus proportionally increasing the overall ST device size. As noted in Sec. II, the inboard gap distance Δ is a particularly high-leverage parameter for compact ST-FNSF performance. Solenoid-free start-up is therefore one of the most widely pursued research topics within the world ST community. It should be also noted that the ST configuration has relatively low external and internal inductances, owing to its spherical configuration with relatively small major radius, which significantly reduces the poloidal flux required to achieve full plasma current. If an acceptable start-up and current ramp-up technique can be developed, it should also lead to a significant improvement in the attractiveness of the tokamak reactor design. There are essentially four classes of solenoid-free start-up concepts which were pursued for STs. The first one is to utilize plasma waves such as electron cyclotron and electron Bernstein waves for heating and current drive, which will be described in Sec. B. The second approach is "helicity injection" or HI, which is essentially a direct injection of toroidal current via electrodes or electron guns (Sec. C). The third approach is to use merging-compression to create ST plasmas. This method has a possibility of heating ions rapidly due to the magnetic energy dissipated during the reconnection process, achieving very high beta ST plasmas as demonstrated in the TS-3 experiments (Sec. D). Finally, there are concepts developed for inductive flux-based start-up that includes the ohmic solenoid, but in a way to make it more reactor compatible (Sec. E). It should be noted that there is an informative recent review article on tokamak / ST start-up including solenoid free start-up [125].

B. Plasma wave-based start-up: Electron cyclotron heating (ECH) is a commonly used technique to create an initial plasma discharge, since ECH at the electron cyclotron fundamental

frequency ($\omega \sim \Omega_e$) could heat electrons essentially from zero temperature. At higher density, the electron cyclotron wave is expected to convert into the electron Bernstein wave (EBW) above the so-called electron cyclotron wave cut-off density. This condition is readily satisfied for ST plasmas due to their high plasma dielectric ($\omega_{pe} \gg \omega_{ce}$) properties. The EBW physics for current profile control in particular is discussed in Sec. IX. During the start-up, due to the low plasma density, the ECH condition is usually satisfied. The ECH start-up was previously investigated on CDX-U, where a toroidal current ~ 1 kA was generated with an ECH power of ~ 8 kW. In CDX-U, initially in an open field line configuration, toroidal plasma currents were generated by ECH owing to pressure driven currents such as the one associated with electron precessional drifts [126]. As the current is increased, the poloidal field created by the generated current becomes significantly larger than the vacuum field, producing closed flux surfaces to form an ST or tokamak configuration with an x-point in the in-board side. After flux surface closure, bootstrap currents can be generated within the closed flux surfaces to maintain the plasma current [127]. The ECH/EBW start-up experiments have been performed in a number of ST devices in recent years: LATE [27, 128], TST-2 [129, 130], CPD [131], QUEST [132], and MAST [133]. An illustrative ECH/EBW start-up and ramp-up experiment was performed on the LATE device, which is shown in Fig. 1.20 [27]. The plasma current and applied vertical field are plotted in Fig. 5.1(a). When a 5 GHz microwave pulse is turned on, plasma breakdown takes place immediately along the external helical field lines, near the 2nd EC resonance layer at $R \sim 27$ cm as shown in Fig. 5.1(b) [27]. Then the plasma current spontaneously jumps up to 7 kA under the steady vertical field B_v and forms an initial closed field structure [Fig. 5.1(c)]. This current jump has been also observed in TST-2, CPD, and QUEST. This is the first stage denoted by (I) in Fig. 5.1(a). In the next stage (II), I_p ramps up with the ramping up of vertical magnetic fields B_v . At the final stage (III), even when the B_v ramp-up rate is set to be very low, I_p still ramps up at the same rate of ~ 260 kA/s as in the preceding stage (II). The I_p ramp occurs against the reverse

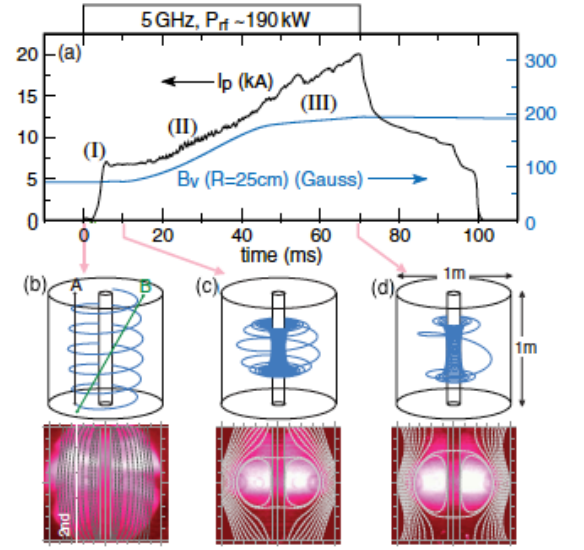


FIG. 5.1. A 20 kA current ramp-up discharge in LATE. (a) plasma current and B_v ; (b), (c), and (d) field lines and plasma images at the breakdown, initial closed field formation, and final low aspect ratio torus formation, respectively.

loop voltage V_L which appears in this stage. The I_p finally reaches 20 kA at the end of microwave pulse, producing a low aspect ratio torus or an ST plasma [Fig. 5.1(d)]. It should be noted that the applied rf power is constant throughout the discharge, yet the plasma current ramps up continuously from 0 to 20 kA. The final line-averaged plasma density of $\sim 4 \times 10^{11} \text{ cm}^{-3}$ is higher than the plasma cutoff density of $3 \times 10^{11} \text{ cm}^{-3}$, suggesting that EBW is playing an important role in the experiment. The presence of ~ 100 keV electrons is detected, and from the measured $\beta_p \sim 1.5$, the energetic electron stored energy component is estimated to be 30 times larger than that of the thermal contribution. The rapid current ramp-up rate observed is attributed to the rf-driven quasilinear diffusion of fast electrons quite possibly by EBW, which is an electrostatic wave with much stronger electric fields compared to the electromagnetic EC waves. More recently, the ECH start-up technique was extended to QUEST (see Fig. 1.17), where a plasma start-up current of up to 65 kA with $P_{rf} \leq 350$ kW and a plasma duration of up to 30 minutes at ~ 20 kA were obtained [134, 135]. The QUEST was able to handle the steady-state rf heat load with actively cooled tungsten limiters and PFCs. On the MAST device (see Fig. 1.12), ECH-assisted plasma current start-up has recently achieved a record $I_p \sim 72$ kA with only 75 kW of gyrotron source power, and without the use of solenoid flux [136, 137]. This is an intriguing and promising result for the future application of this technique for STs, as it may be able to have favorable current generation efficiency similar to that of lower hybrid current drive (LHCD) [138]. This increases with $\sim P_{rf}$ rather than $\sim P_{rf}^{0.5}$ as expected for the pressure-driven current approach. It is noted that the LHCD start-up is being investigated in TST-2 and generated a plasma current of 15 kA in low-density start-up plasmas [139].

C. DC-Helicity Injection based start-up - A DC-helicity injection (HI) based current start-up technique utilizes the magnetic poloidal flux injected into a torus by an injection of DC toroidal electric currents [140]. This method has been previously successfully used to create spheromaks [141]. A schematic of the coaxial helicity injection (CHI) set-up is shown in Fig. 5.2. The two sections (inboard and outboard) of the vacuum chamber serve as electrically-isolated forming electrodes. When bias voltage is applied, an electrical discharge is formed between the two electrodes. Due to the dominance of the toroidal field B_T during start-up, since the applied poloidal field B_p is very small compared to B_T (i.e., $B_T \gg B_p$), the electrical discharge currents thus generated flow predominantly in the toroidal direction since the electrons tend to flow along the magnetic field lines. The toroidal current therefore multiplies by a so-called “stacking factor,” which is essentially the number of toroidal transits around the torus before exiting.

Typically, injected electrons can spiral around the torus several times before exiting, so one can obtain total toroidal current several times the injected electric current. There is a net force of $j_{pol} \times B_T$ on the injected current, which tends to push the current ring away from the injector region toward the main plasma as depicted in Fig. 5.2. The injected current ring being pulled away from the injector can be disconnected from the electrodes via magnetic reconnection. When the poloidal field generated by the injected toroidal current starts to dominate over the applied poloidal (vertical) field, a closed-flux (tokamak-like) configuration can be formed. Once the closed flux surface is formed, the

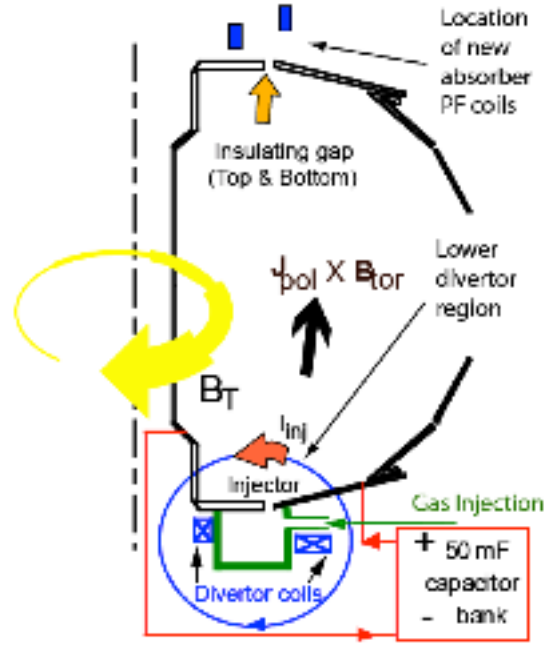


Fig. 5.2. CHI schematic drawing of the NSTX including the location of the insulating gaps between the divertor plates.

physics changes dramatically and the so-called helicity injection (HI) current drive concept is introduced to explain this current drive mechanism. HI current drive is based on the conservation principle that magnetic field energy associated with the current tends to decay faster than the magnetic helicity K [141]. K represents a degree of linkage of toroidal magnetic flux with poloidal magnetic flux. It is given by $K = 2 \int \phi d\psi$, where ϕ is the toroidal flux inside a flux surface and ψ is the poloidal flux, defined to be zero at the wall. In general the rate of change of helicity is given by $dK/dt = 2 \int_{vacuum} \mathbf{E} \cdot \mathbf{B} dV - 2 \int_{plasma} \mathbf{E} \cdot \mathbf{B} dV$, where the first integral is calculated over the plasma volume V in the absence of plasma current but with the same magnetic field and flux boundary conditions as applied to the second integral which is performed in the presence of the actual plasma. The first integral can be thought of as the injection term and is equal to $2 V_{inj} \psi_{inj}$ for CHI, where V_{inj} is the voltage between the coaxial electrodes and ψ_{inj} is the flux that penetrates both electrodes. The second integral can be thought of as the resistive dissipation of helicity and is often called K/τ_K . For transformer current drive the helicity injection rate, for example, can be given as $2V_{loop}\dot{\phi}_{tor}$.

The HI method was first tested in a tokamak/ST geometry in the CDX/CDX-U experiment, using a localized emissive cathode generating up to ~ 10 kA level toroidal current [142, 143]. The HI technique was further improved with the dedicated devices HIT and HIT-II, utilizing co-axial (or axi-symmetric) electrodes which extended the achievable plasma current to the $\sim 100 - 150$ kA range [144, 145]. A schematic of the HIT-II device

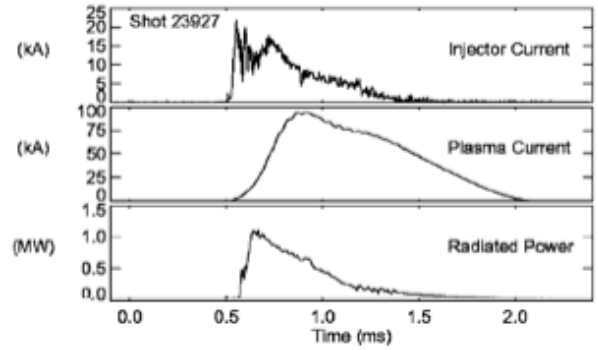


Fig. 5.3. CHI only discharge in HIT-II: Injector current, CHI-produced plasma current, and radiated power signal from a wide-angle bolometer.

is shown in Fig. 1.16. A rotating $n=1$ distortion observed in the experiment is believed to facilitate current relaxation (and the reconnection process) or radial helicity transport. This technique, termed the coaxial helicity injection (CHI), was then transferred to the larger NSTX device. A toroidal current of 240 kA was generated flowing along open field lines with a the current multiplication factor of 10 [146]. This type of DC current drive appears to be suitable to drive edge currents in steady-state, but it appears to suffer from impurity generation. On HIT-II, a new type of CHI termed “transient CHI” was developed [147, 148], where, with the injector flux foot prints sufficiently close together, the injector current is rapidly reduced to zero. This causes the oppositely directed field lines in the injector region to reconnect causing closed flux surfaces. In Fig. 5.3, transient CHI discharge traces in HIT-II are shown. One can see that the plasma current persists much longer than the injector current. In this way, closed flux surfaces were obtained as a plasma current of 40 kA decayed resistively. As can be seen in

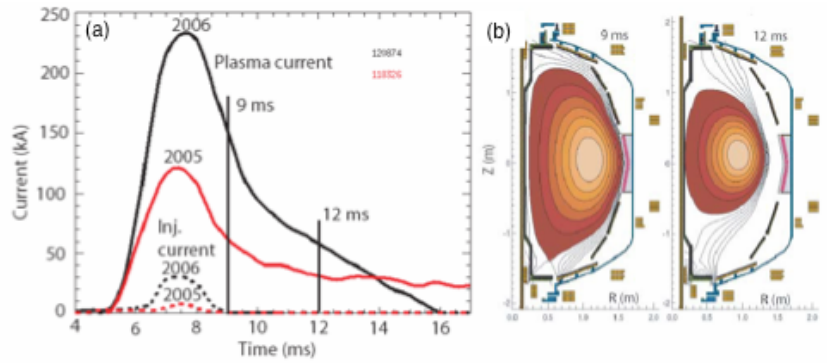


Fig. 5.4. (a) Discharge evolution of 160 kA closed flux current produced by CHI alone in NSTX. Discharges in 2006 operated at higher toroidal field and injector flux. (b) Equilibrium reconstructions show the shape evolution of the CHI produced plasma in response to decaying current. EFIT analysis is possible when no injector current is present.

the figure, the radiative power becomes small once the injector current is turned off. This transient CHI was then tested on NSTX, which resulted in ~ 160 kA of total plasma current with closed flux surfaces as shown in Fig. 5.4 [149]. The plasma current amplification of as much as 100 was observed. As can be seen in Fig. 5.4 (a), the plasma current persists well after the injection current is completely turned off, and the plasma current decays resistively, indicative of a closed flux configuration consistent with EFIT reconstruction results shown in Fig. 5.4 (b). This technique also appears to be very energetically efficient, as it can produce a plasma current of ~ 10 A/J of capacitor bank energy. This technique is also attractive due to its minimal impurity generation from the very short current injection period. It was also shown that application of ohmic induction to a CHI-initiated discharge resulted in 1 MA H-mode plasmas with significantly reduced ohmic flux consumption [150, 151]. A CHI performance projection based on the HIT-II and NSTX results shows the possibility of driving as much as 0.5 - 1 MA of plasma current in NSTX-U and 2-3 MA for a FNSF-like device. It should be noted that the electrical insulation for CHI does not require high resistance, since the only requirement is its resistance to be well above the relatively small CHI discharge resistance. This gives a choice of CHI insulating material to minimize possible neutron damage. Also, the CHI insulator maybe placed behind a neutron shield to further reduce neutron damage.

Another significant development in HI-based start-up is the electron gun injection experiment on the PEGASUS device (see Fig. 1.14) [6]. While similar to the earlier CDX/CDX-U experiment in terms of injection configuration, the experiment utilizes electron guns or cold cathode instead of emissive cathodes [52]. The PEGASUS experiment was able to produce up to 160 kA of toroidal plasma current using this method [152]. The current multiplication factor can be quite high i.e., ~ 30 . Due to

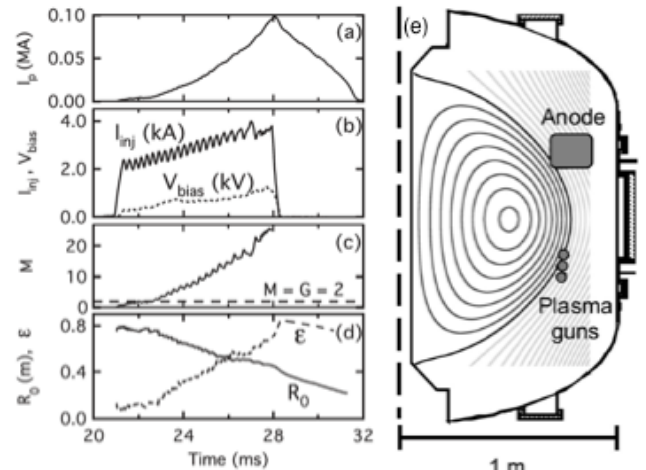


Fig. 5.5. Localized plasma gun start-up in PEGASUS (a) – (d) Time wave forms for plasma current, bias voltage and total injected current, current multiplication, and plasma major radius and inverse aspect ratio, respectively. (e) An experimental set up schematic of plasma gun start-up. Magnetic equilibrium reconstruction of poloidal flux surfaces at $t = 28$ ms.

the magnetic relaxation process, high ion temperatures from impurity Doppler spectroscopy were also observed. The PEGASUS gun approach is attractive for the portability of the plasma gun. For example, a plasma gun for NSTX-U is being developed to achieve 0.5 – 1.0 MA level of start-up current. For the FNSF, one can envision a scenario where the plasma gun is retracted to a safe location once the start-up is complete to avoid neutron damage.

D. Merging Start-Up - Pioneered in START, MAST has demonstrated a novel merging compression technique to form ST plasmas by only utilizing the poloidal field (PF) coils placed inside the vacuum vessel [29]. A schematic of the MAST cross section is shown in Fig. 1.12. By rapidly ramping down the current in the top and bottom PF coils, breakdown occurs around the PF coils, forming plasma rings with considerable toroidal currents. By programming the current waveforms, the plasma rings thus created can be “pinched” off from the PF coils, and merged

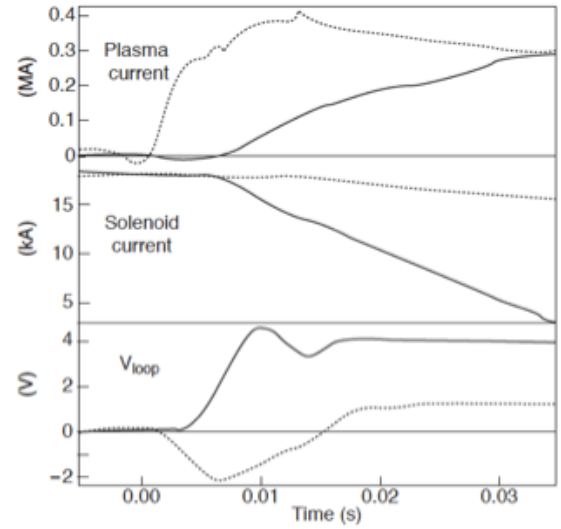


Fig. 5.6. Direct induction (solid curves) and merging-compression (dotted curves) schemes for MAST discharges.

into a single ST plasma which can then be compressed radially to further increase the plasma current. The visible fast camera picture of the plasma evolution is shown in Fig. 1.5. The merging-compression scheme is compared with the more conventional direct induction scheme in Fig. 5.6 [29]. In direct induction (solid curve), breakdown occurs in a poloidal field null, and the solenoid current is ramped down to give an initial loop voltage of 4 V, which produced a plasma current ramp of 9 MA/s. In a merging-compression discharge (dotted lines), the process produces 400 kA of plasma current very rapidly before the solenoid ramp begins; the low ramp rate associated $V_{loop} \sim 1$ V is sufficient to maintain the plasma current. By this method MAST was able to produce the plasma current of ~ 450 kA routinely without an ohmic solenoid. Because of the reconnection process, a high ion temperature was also observed [153].

In the smaller TS-3 device, a merging technique was pioneered to obtain various plasma configurations including high beta and medium beta STs. Instead of induction, electrodes were used to form the initial double spherical tokamak discharges as shown in Fig. 1.21 [14]. The merging reconnection process can channel some of the poloidal magnetic energy into ion thermal energy as observed to the HI experiments. As shown in Fig. 5.7, the TS-3 ST plasma achieves ultra high beta with unity β_T and $\beta_N \sim 17$, maintained stably over 200 μ s. The ultra-high beta ST plasma had the maximum pressure gradient and magnetic shear at the edge, which may have helped its MHD stability. Another variant of the merging technique is a repetitive injection of ST plasmas into the main plasma in the HIST device as shown in Fig. 1.23 [154].

While the merging-compression method with internal coils works well on present devices, whether this method can be extended to ST reactors is a challenge. There, the internally placed PF coils are in contact with fusion plasmas and it maybe technically difficult to provide adequate shielding for the coils. To overcome this difficulty, the UTST device is constructed with all external PF coils as shown in Fig. 1.22 [15, 155]. The recently-completed VEST device (Fig. 1.24) also uses similar merging-compression techniques with all external PF coils [17, 156].

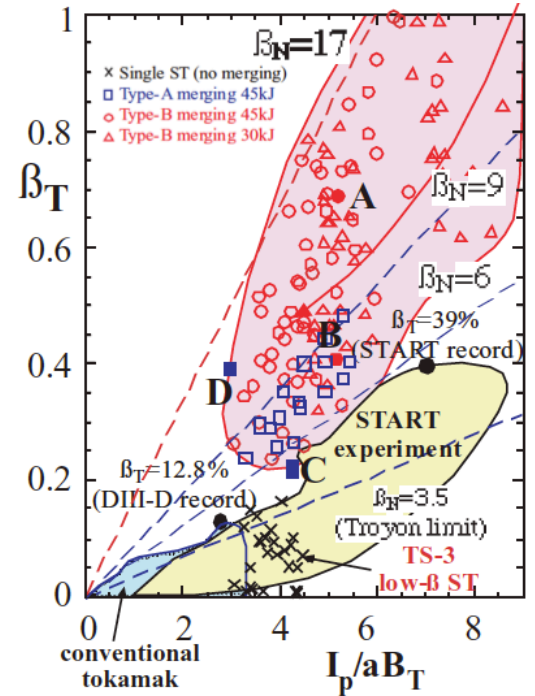


Fig. 5.7. Toroidal betas β_T of single low-beta STs (no merging) and high-beta STs produced by mergings as a function of I/aB_T in TS-3.

E. Inductive drive based start-up - A number of scenarios utilizing magnetic induction that includes a conventional central ohmic solenoid have been also investigated by the ST community. One variation of inductive plasma start-up uses vertical field coils with currents initially in the negative direction. This provides ohmic induction as the coil currents are swung down to zero. For this scheme to work, it is important to have sufficient pre-ionization due to the inherent “error” field in the applied vertical field, and the plasma is radially unstable. Using 100 kW of ECH power, 10 kA of plasma current was produced solely by the simple outer PF coil

vertical field swing in TST-2 [12]. This technique was extended to an experiment on JT-60U, resulting in a non-inductively driven 600 kA high β_p plasma [157]. A refinement to the PF coil-only start-up is to create a multi-pole null region in the outer part of the plasma to reduce the error field, while retaining some of the poloidal flux needed for subsequent current ramp-up [158]. Recently, an all-metal iron-core transformer compatible with high neutron flux, that incorporates high-resistivity inserts, to break up eddy current patterns in the copper toroidal field coil design, has been proposed [159]. It should be also noted that for superconducting (SC) ST power plant designs, as with those based on tokamaks, neutron shielding is already available for the SC magnets. Because of this, an ohmic heating solenoid with a sufficient start-up capability can be incorporated into the design as mentioned in Sec. II.C. [88, 89]. Overall, it appears that there is indeed a sufficient number of options available for plasma start-up for future STs and tokamak reactors. Perhaps the greater future challenge is current ramp-up from the start-up plasma to full current operation. Progress in this area of research has been limited thus far on STs, since the research focus has been on start-up issues. One notable success for non-inductive ramp-up is JT-60U, where the plasma current was successfully ramped up non-inductively from ~ 100 kA at initial start-up formation to ~ 600 kA in a high β_p discharge [157]. Current ramp-up studies are therefore a key research topic on NSTX-U and MAST-U. The QUEST research goal is to achieve long-pulse 100 kA discharges non-inductively. That said, start-up research is still quite important in that higher start-up current would reduce the requirements for non-inductive current ramp-up, and lessen the difficult challenges of performing it efficiently.

V. Transport and Turbulence.

A. Introduction and Motivation – A fundamental understanding of plasma transport physics is critical for designing future magnetic fusion devices. It is often quoted that the fusion energy gain factor Q scales as H^{5-7} where H is the confinement improvement factor compared to the standard H-mode. This would then imply that one would need to know the H -factor to within 10% in order to have sufficient confidence of predicting Q within a factor of two. As the operating parameters of future devices (including ITER) represent a considerable extrapolation from present day experiments, accurate confinement predictions are a key element of present-day fusion energy research. Indeed, one of the key uncertainties of MA-class ST plasma performance has been the confinement quality of higher current, lower collisionality ST fusion plasmas. For example, ST plasmas, due to their large mirror ratio ($B_{Tmax}/B_{Tmini} \sim A+1/A-1 = 2$ for $A = 3$, and $= 5$ for $A = 1.5$), can have a significant trapped particle population compared to tokamaks. How that can influence the anomalous transport is a natural question for ST plasmas as they enter the collisionless regime.

Interestingly, as mentioned in the introductory section, observed MA-class ST plasma confinement actually showed an improving trend with reduced collisionality, i.e., $\tau_E \sim \nu^{*-1}$ as shown in Fig. 1.7. This is a somewhat surprising but very encouraging result, as the operating parameters of future devices are at

least an order of magnitude lower in collisionality compared to present day devices. It is generally agreed that plasma transport by so-called classical or neoclassical mechanisms such as collisional transport is relatively well understood. Therefore, the main focus of present-day transport research is on the less well understood plasma transport by plasma microturbulence, often termed “anomalous” transport. The generally high-beta nature of ST plasmas could host plasma turbulence with increasing electromagnetic character. This can potentially lead to greater anomalous transport, particularly for electrons. As illustrated in Fig. 5.1, the modes which are believed to be responsible for plasma transport include ion scale microturbulence (i.e., low- k_θ or $k_\theta \rho_s \sim 1$ where $\rho_s = C_s/\Omega i$). Examples are the ion temperature

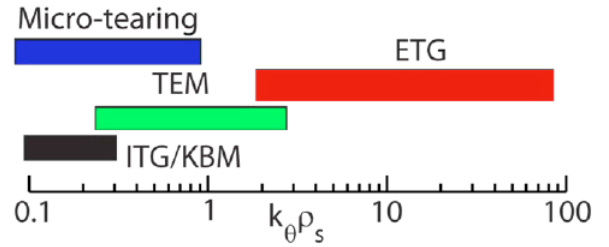


Fig. 5.1. Illustration of wave number range for various microturbulence as labeled in NSTX parameter range.

gradient (ITG) mode [160], trapped electron mode (TEM) [161], kinetic ballooning mode (KBM) [162], and micro-tearing (MT) mode [163]. Electron scale microturbulence (i.e., high- k_θ or $k_\theta \rho_s \sim 10$) includes electron temperature gradient (ETG) drift waves [164]. In addition, there are energetic-particle-driven global and compressional Alfvén eigenmodes (GAE and CAE), which can also affect power deposition and electron energy transport in the plasma core region [165]. In terms of drive and stabilization mechanisms, electron beta β_e and collisionality $\nu^{e/i}$ are important parameters as illustrated in Fig. 5.2 [166]. All of the ion scale ballooning-type drift waves (e.g., ITG, TEM, and KBM) tend to be stabilized by increasing $\nu^{e/i}$, while the ETG mode is insensitive to $\nu^{e/i}$ and the MT mode is often more unstable at higher $\nu^{e/i}$. The ETG and ITG/TEM modes are excited at low beta, and KBM and MT tend to become more important in high beta plasmas. So, it is vital to understand the fundamental transport physics

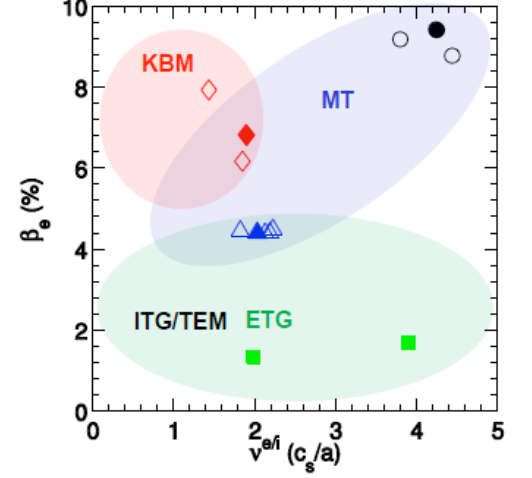


Fig. 5.2. Local values of β_e and $\nu^{e/i}$ ($r/a = 0.6 - 0.7$) for various H-mode discharges. The colored regions illustrate where various microinstabilities are generally predicted to occur.

mechanisms based on micro-turbulence to gain sufficient confidence in predicting future ST performance, as the future reactor facilities including ITER represent significant capital investment. In addition to the benefits of predictive capability, if some tools to control plasma transport can be developed through better understanding of microturbulence driven transport, they can be utilized to achieve more optimum pressure and current profiles for reactor performance optimization [70].

Global scalings are discussed in Sec. B. However, it is crucial to understand fundamental transport mechanisms for each of the plasma quantities, including ion thermal energy, electron thermal energy, energetic particles (e.g., α -particles), toroidal momentum, and thermal particles including impurities. The ion transport physics is discussed in Sec. C. The electron transport physics is particularly important for burning plasma regimes, since fusion alphas predominantly heat electrons as discussed in Sec. D. The momentum transport is discussed in Sec. E. Finally, thermal particle transport physics, including impurity transport, is discussed in Sec. F. Energetic

particle transport is discussed in the energetic particle section in Sec. VIII.

B. Global Confinement Scaling - Global confinement scaling has been actively pursued as a tool to quickly quantify the dependence of confinement on various plasma and device dimensional and non-dimensional parameters, and assess the plasma performance of present-day devices as well as predict the performance of future devices including ITER. Perhaps the most well-known confinement scaling is the ITER98_{phy,2} H-mode scaling [21], which is expressed in engineering variables as

$$\tau_{E,th} = 0.0562 I_p^{0.93} B_T^{0.15} R_0^{1.97} \kappa^{0.78} n_e^{0.41} \mu^{0.19} P_L^{-0.69} A^{-0.58} \quad (5)$$

There is a small adjustment for the average ion mass μ , giving a slight advantage for higher mass tritium-operation compared to that of hydrogen and/or deuterium. Here, the absorbed power P_L is defined as the total heating power into thermal plasma component, taking into account possible heating efficiency minus the change in the plasma stored energy dW_{th}/dt . The H-mode confinement scaling was derived by using the tokamak data base, so it is naturally of interest to see if this scaling actually applies to STs with much lower A and higher κ . For comparing the present-day tokamak experimental data base to ITER, the main extrapolating parameters are I_p , R_0 , and P_L . For STs, the main extrapolation to future ST devices such as FNSF from NSTX/MAST experiments would be based on I_p , R_0 , B_T , and P_L . If we were to compare ST and tokamak power plants with a given fusion power output, one might expect quite similar μ , n_e , and P_L . However, an ST power plant compared to that based on a tokamak may have $\sim 2 \times I_p$ and κ , but only $\sim 1/2 \times B_T$, R_0 , and A . Then, the $\tau_{E,Thermal}$ for the ST and tokamak plants as predicted by Eq. 5 would be essentially the same. The higher κ is desirable for the ST not only from the MHD and non-inductive operation point of view, as discussed in Sec. III, but also from the confinement scaling point of view if the tokamak scaling such as the one described by Eq. 5 holds true. The desirability of higher κ tends to drive the design point for future STs toward $\kappa \sim 2.5 - 3.3$.

The B_T and I_p confinement dependence observed in boronized + He GDC conditioned NSTX plasmas is shown in Fig. 5.3 [167]. The observed dependence $\tau_{E,th} \propto I_p^{-0.4} B_T^{-0.9}$ is clearly different from the tokamak scaling of $\tau_{E,th} \propto I_p^{0.93} B_T^{0.15}$. The MAST experiments also found a stronger than linear B_T dependence and weaker than linear I_p dependence consistent with that of NSTX and in contrast with the conventional tokamak scaling [168]. For constant q (i.e., fixed $B_T : I_p$ ratio) or $B_T \propto I_p$, the NSTX scaling then gives $\tau_{E,th} \propto I_p^{-1.3}$, which is somewhat stronger than the tokamak scaling of $\tau_{E,th} \propto I_p^{-1.1}$. This scaling trend in NSTX can be explained if the global confinement trend is dominated by electron energy transport, as the ion energy transport is close to neoclassical as discussed in Sec. B. The recent experiments on NSTX with evaporated lithium wall conditioning, as described in more detail in Sec. VII, showed a strong I_p dependence however, and weak B_T dependence similar to that of the tokamak scaling in Eq. 5. The difference in the scaling with lithium and non-lithium PFCs presents a challenge for this type of scaling approach. It is found that collisionality unifies the lithium and non-lithium confinement results, as discussed below.

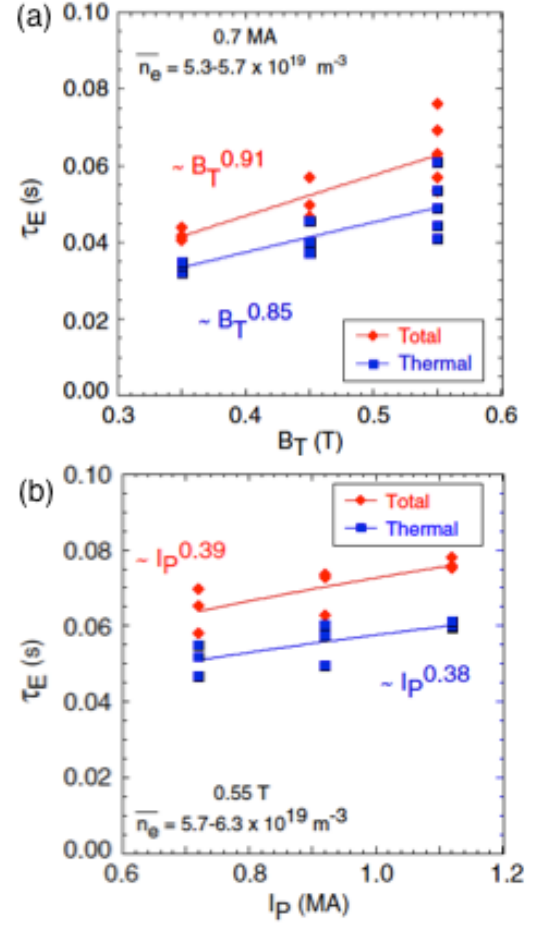


Figure 5.3. Total (red) and thermal (blue) energy confinement times vs. (a) B_T at constant I_p , n_e , and P_L , and (b) I_p at constant B_T , n_e , and P_L .

Another way of expressing the ITER98_{y,2} H-mode confinement scaling is to use the dimensionless quantities or “physics variables” [21].

$$\tau_{E,th} \propto \tau_B \rho_*^{-0.70} \beta^{0.90} \nu_*^{-0.01} M^{0.96} q_*^{-3.0} A^{-0.73} \kappa^{2.3} \quad (6)$$

where τ_B is the Bohm time $= a^2 B / T$, ρ^* is the normalized toroidal Larmor radius ρ_L/a , q^* is cylindrical safety factor, and ν^* is the normalized collisionality. In this physics variable-based scaling, the extrapolation from present-day tokamaks to future devices would be mainly in ρ^* and ν^* . Since the ν^* dependence is so weak, the most important physics parameter for extrapolations is considered to be ρ^* for tokamaks. For STs on the other hand, there is relatively small variation in ρ^* , but the ν^* variation is quite large. Clearly if the ν^* dependence is indeed so weak as suggested by Eq. (6), then the confinement scaling for

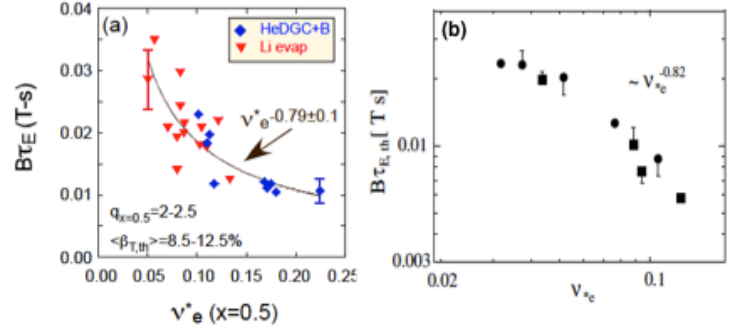


Figure 5.4. (a) Normalized confinement time as a function of collisionality at mid-radius in NSTX. Blue points are from discharges that used HeGDC+B wall conditioning, while red points are from discharges that used Li. (b) Collisionality scan of thermal energy confinement time in MAST.

STs would be relatively simple. Another problem with the tokamak scaling suggested in Eq. (6) is the very strong beta degradation ($\beta^{0.90}$), which would be rather unfavorable for future higher beta reactor operation and generally higher beta STs. However, the confinement data from NSTX and MAST suggest a very different parameter scaling, showing improved confinement with reduced collisionality and relatively weak beta dependence, which is quite encouraging for future ST facilities. In Fig. 5.4, the collisionality dependence is shown for (a) NSTX [31, 167] and (b) MAST [169], where the confinement trend indicates improvement with reduced ν^* in both devices, i.e., $B\tau_{E,th} \propto \nu^{*-0.79}$ for NSTX and $B\tau_{E,th} \propto \nu^{*-0.82}$ for MAST. For NSTX, those discharges that used Li evaporation wall conditioning are in red, and those that used helium glow discharge cleaning (HeGDC) plus boronization conditioning are in blue [31]. It is seen in the plot that the Li evaporation discharges generally have lower collisionality and higher confinement, extending the range of collisionality to lower values. Interestingly, this collisionality scaling appears to unify the confinement trends of these discharges, despite the different parametric dependences on engineering variables as noted above. The scaling exhibits a strongly favorable trend of increasing normalized confinement with decreasing collisionality, which bodes well for future lower collisionality STs. There has been a re-examination of the

tokamak scaling utilizing improved regression techniques with the addition of NSTX and MAST data. This has yielded more collisionality dependence and much less beta degradation compared to the original ITER scaling (Eq. 6), which is now expressed as $B\tau_{E,th} \propto \rho_*^{-2.6} \beta^{0.08} \nu_*^{-0.2}$ [22, 170]. Here, the collisionality and beta scalings are closer to that observed in NSTX and MAST. Also, in a review paper on the dimensional confinement analysis [171], it is pointed out that there is a significant $\sim x3$ variation in the ν_* trend among tokamak devices, which perhaps makes the correlation with ν_* less clear. The paper also points out a significant variation in the observed beta degradation even among tokamak experiments. It is encouraging to note that the beta degradation is weak for those devices reaching the highest values of β_N , including STs. This is particularly interesting, since the beta degradation is generally thought to depend on increased electromagnetic effects on turbulence as beta is increased. This makes the weakening of beta degradation at high beta counter-intuitive. Clearly, the device-to-device variations on dimensional confinement behavior suggests an acute need for more fundamental understanding of transport mechanisms if we are to develop a reliable predictive capability for future facilities. Indeed, extending the confinement data toward lower collisionality is one of the important missions for NSTX-U and MAST-U.

C. Ion Energy Transport - Due to stronger shaping and relatively low magnetic field, the ST configuration can create strong ExB sheared flow rates which can exceed that of the growth rate of ion transport relevant turbulence such as ITGs and TEMs. The resulting ExB shear stabilization of ITGs/TEMs makes the ion transport in STs near the neoclassical level for most of the plasma cross section [172]. Indeed, in NSTX and MAST H-mode discharges, the observed ion transport (from TRANSP analysis) is very close to the neoclassical level for much of the plasma profile, as shown in Fig. 1.6, while the electron energy transport remain anomalous. While close to neoclassical ion transport has been observed in

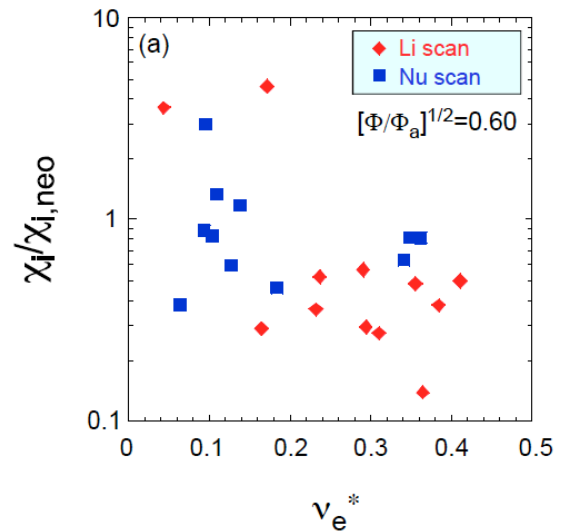


Figure 5.5. Ion thermal diffusivity, χ_i normalized to the neoclassical ion thermal diffusivity as determined by NCLASS as a function of ν_e^* at $r/a = 0.6$. Values from both the collisionality (Nu) scan (blue) and Li scans (red) are shown.

tokamaks locally, this global close to neoclassical ion transport behavior appears to be a characteristic of ST H-mode plasmas. This near neoclassical ion transport behavior is consistent with the observed weaker I_p dependence of the global confinement scaling in STs, as mentioned in Sec. B. Due to the good ion energy confinement, the NBI heated H-mode plasmas in NSTX and MAST generally have a higher ion temperature compared to the electron temperature, even though a higher fraction of the NBI power ($\sim 60 - 70 \%$) is estimated to flow into electrons. Also, the ion energy transport trend appeared to change as the collisionality is reduced in NSTX, as shown in Fig. 5.5, where the observed χ_i at mid-radius of $r/a = 0.6$ is plotted as a function of ν_e^* [31]. As can be seen in the figure, the ion diffusivity χ_i is trending up well above the neoclassical values (i.e., $\chi_{i,neo}$) at lower collisionality regime. While the absolute level of χ_i may have a factor of two range of uncertainty, there is a clear trend of increasing χ_i from $\chi_i \sim \chi_{i,neo}$ to perhaps becoming four to five times more anomalous at lower collisionality. This is opposite the trend in electron transport, where χ_e is decreasing with reduced collisionality as discussed in Sections B. and D. While χ_e still dominates over χ_i in the present experiments, the trend indicates that it is essential to understand the behavior of both χ_e and χ_i in NSTX-U and MAST-U as much lower collisionality regimes are explored. Ion-scale turbulence research has made significant progress in recent years due to the implementation of the beam emission spectroscopy diagnostic (BES) on NSTX and MAST [173, 174]. One may note that because of the multiple possible ion transport relevant turbulence modes with a similar range of wavelength and frequency (i.e., ITG, TEM, and KBM), it is often difficult to clearly identify one particular mode responsible for the ion energy transport. In the core region of the MAST plasma, highly non-linear saturated “critically-balanced” turbulence behavior was observed [175]. Recently the ion-scale density turbulence measured by BES was compared with a non-linear, global, gyro-kinetic simulations [176]. The results indicate that while there are some degree of agreement seen in the mid-radius region, there is a significant shortfall, for example, in predicted fluctuation level and ion heat flux in the peripheral region. In the steep gradient region of the NSTX H-mode edge pedestal during ELM-free, MHD-quiescent periods, turbulence scaling consistent with TEM, KBM, or MT, but not with ITG, was observed [177]. To further understand the ion transport physics, ion-scale turbulence diagnostics (e.g., BES) along with electron gyro-radius (ETG) scale high-k turbulence diagnostic will be implemented on NSTX-U and MAST-U.

D. Electron Energy Transport – Electron energy transport is the most actively investigated transport physics topic in NSTX, since it is presently the dominant energy transport mechanism and the transport is clearly anomalous. The availability of the innovative high-k microwave tangential scattering diagnostic enabled, for the first time, measurement of radially resolved electron gyro-scale turbulence such as ETGs [178]. The physics of electron transport is likely to be also relevant in fusion reactors including ITER, as the plasma heating sources, such as negative ion neutral beams, ECH heating, and fusion-produced 3.5 MeV α -particles, all primarily heat electrons in the core. As expected, the electron transport physics is quite complex, and presently three main electron transport mechanisms have been reported for NSTX plasmas. The electrons, due to their very high mobility, can cause anomalous heat transport under a variety of conditions. The recent experiments on NSTX with evaporated lithium (Li) wall conditioning showed a clear indication of electron temperature profile broadening suggesting improved electron energy confinement in the edge region [179, 180]. A controlled lithium evaporation experiments showed that the main confinement improvement is in the electron energy channel [181, 182]. The electron diffusivity at $r/a = 0.7$ showed continued reduction as a function of the Li deposition amount prior to the discharge up to the maximum Lu deposition ~ 100 mg. The edge particle recycling was reduced from $\sim 98\%$ to $\sim 90\%$. The reduced recycling resulted in the reduced edge density which together with generally higher electron temperature reduced the edge collisionality. The enhanced confinement with the Li deposition reducing edge collisionality is consistent with the inverse collisional scaling observed in NSTX [31]. The role of ETGs for electron energy transport is now emerging. In the collisional regime, the micro-tearing (MT) mode appears to be causing rapid electron transport through generation of stochastic magnetic fields. In strongly neutral beam heated plasmas, the excitation of Global Alfvén Eigenmodes (GAEs) with a large fluctuating magnetic field component (like MTs) were observed to cause strong electron energy transport in the core region of the plasma. This may be again due to the generation of stochastic magnetic fields, producing a clamping of the central electron temperature even with a three-fold increase in NBI heating power. We shall now briefly describe the status of our understanding of these three electron energy transport mechanisms.

Electron energy transport associated with ETGs - The ETGs are the electron gyro-radius (ρ_e) scale turbulence known to cause electron transport in the presence of an electron temperature gradient [161]. The first definitive indications of ETGs was observed by taking advantage of the relatively large electron gyro-radius ρ_e scale length in NSTX due to high beta (i.e., high electron temperature per given magnetic field), using a radially-resolved tangential high-k scattering diagnostic on NSTX [178]. The system measures electron gyroscale fluctuations with $k_{\perp} \rho_e \leq 0.6$ and $k_{\perp} \leq 30 \text{ cm}^{-1}$, with the radial and wave number resolutions of $\Delta R \sim \pm 2 \text{ cm}$ and $\Delta k \sim 1 \text{ cm}^{-1}$, respectively. In NSTX L-mode plasmas, high-k fluctuation measurements revealed electron gyroscale fluctuations consistent with ETG turbulence. The high-k data also showed enhanced fluctuations when the electron temperature gradient exceeded the ETG linear critical gradient defined by,

$$(R/L_{Te})_{\text{crit}} = (1 + Z_{\text{eff}} T_e/T_i) (1.3 + 1.9 s/q) (1 - 1.5 \epsilon) \quad (7)$$

where Z_{eff} is the ionic effective charge (~ 2.5), q is the magnetic safety factor, $s = (r/q) \times (dq/dr)$ is the magnetic shear, and $\epsilon = r/R$ is the inverse aspect ratio [183]. The ETGs appear to play an effective role in regulating the electron temperature gradient near the ETG critical temperature gradient. In NSTX, HHFW core electron heating causes a very strong increase in the detected density fluctuations in the range of $k_{\perp} \rho_e = 0.2 - 0.4$ at $R = 1.2 \text{ m}$. In Fig. 5.6, two electron temperature profiles and corresponding measured spectral density of fluctuations are shown [184]. They indicate that the ETG scale fluctuations increase greatly when the temperature gradient exceeds the predicted ETG critical temperature gradients. This type of relatively clean ETG fluctuation measurement can be attributed to the lack of other possible modes in this frequency and wavelength range, which is not the case for the ion-

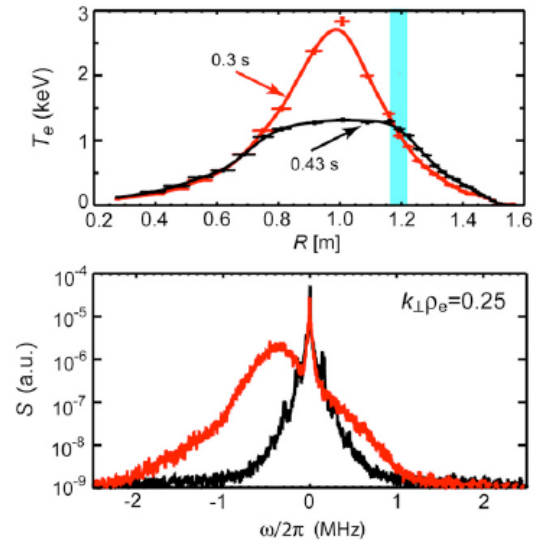


Figure 5.6. Temperature profiles (top) and spectral density of fluctuations (bottom) at 0.3 s (red) and 0.43 s (black). The blue stripe indicates the location of measurement where L_{Te} is 15 cm and 50 cm, respectively. Negative frequencies (bottom) correspond to Doppler shifted frequency spectrum due to wave propagation in the electron diamagnetic direction.

scale turbulence as discussed in the previous section. It was also observed that electron gyro-scale fluctuations in NSTX H-mode plasmas with large toroidal rotation and, correspondingly, large $E \times B$ flow shear rates, increased when the electron temperature gradient is marginally stable with respect to the ETG linear critical gradient. The fluctuation amplitudes decreased when the $E \times B$ shearing rate exceeded the ETG linear growth rate [185]. The observations indicate that $E \times B$ flow shear can therefore be an effective suppression mechanism, even for ETG turbulence near marginal stability. Another example of ETG physics came from the reversed magnetic shear regime, where the ETGs are predicted to be stabilized and improved energy confinement has been observed on NSTX [186, 187]. In Fig. 5.7, the electron temperature profiles (a) and associated q-profiles (b) are shown under condition of same HHFW

electron heating power [188]. The variable here is the magnetic shear, where the reversed shear (RS) configuration represented by the red curves. One can see that a very steep electron temperature gradient (i.e., an electron temperature

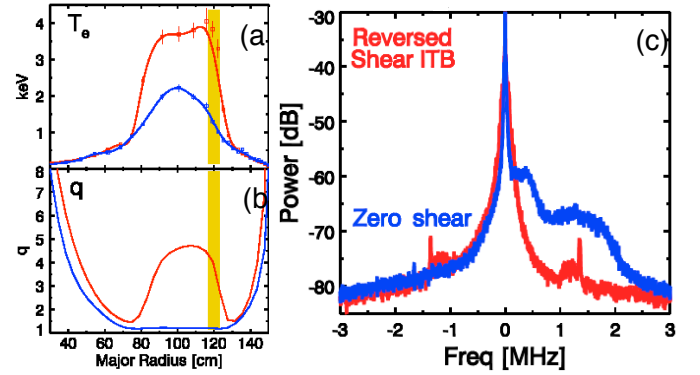


Fig. 5.7. Comparison between a case with an e-ITB and strongly negative magnetic shear (red) vs. a zero reversed shear case (blue). (a) Electron temperature profiles. (b) q-profiles. Shaded region indicates the high-k measurement region. (c) High-k microwave scattering fluctuation power spectra.

internal transform barrier) is developed at $R = 1.2$ m, where the high-k fluctuation is being monitored at the RS region as indicted by the yellow strip. In the discharge with normal magnetic shear, the

electron temperature gradient relaxes to more typical profiles as represented by the blue curves. A comparison of the measured high-k fluctuations is shown in Fig. 5.7 (c) for those two cases. One can see that for the RS case, the ETG fluctuation is essentially stabilized and the same HHFW electron heating power significantly increases the central electron temperature. The bursts of ETGs appear to regulate the electron temperature gradient and maintain high core electron temperatures during the RS phase. The mechanism for the bursting of the ETGs is not fully understood, but the estimated enhanced transport appears to be sufficient to regulate the temperature gradient even though the bursts occur only about 1.6 % of the time [188]. A nonlinear ETG simulations of the NSTX RS plasma also showed ETG stabilization and corresponding reduction in the electron thermal diffusivity [189]. Another example is reflected in improved energy confinement associated with density gradients [190]. This has been

observed for a long time in fusion research, as for example, in the peaked density profile of the TFTR super-shots [73]. The density gradient term alone, if large enough, could determine the ETG critical temperature gradient. This physics was investigated after the ELM event where the density gradient term a/L_{ne} in the pedestal region increased by a factor of five but a/L_{Te} remained constant (as did q , etc.) This resulted in ETG fluctuation reduction and a factor of two reduction in the transport coefficient in the density gradient region. All of these ETG experimental results appear to point to the basic validity of the ETG stability theory and its significant role in electron energy transport in ST plasmas. It should be also noted that while the ETGs were not directly observed in MAST, a nonlinear gyrokinetic flux-tube simulation showed that the ETG mode may be producing experimentally significant electron energy transport in MAST-like plasmas [191].

Electron transport due to micro-tearing modes – Micro-tearing modes (MTs) are small scale tearing modes with large toroidal (n) and poloidal (m) mode numbers, driven resistively unstable and generating stochastic fields near the tokamak

rational surfaces, $q=m/n$. Anomalous electron transport occurs when neighboring tearing mode-induced magnetic islands overlap. The approximate radial distance between two neighboring rational surfaces [m/n and $(m+1)/n$] is $\Delta r = 1 / (nq')$. With many toroidal modes present, the minimum distance between adjacent resonant surfaces is $\delta r = 1 / (n^2 q')$ [192]. A recent breakthrough in understanding anomalous electron thermal transport behavior with collisionality is the identification of the electromagnetic effects in causing electron thermal transport in the outer half of a set of NSTX high beta H-mode plasmas. This was achieved with state-of-art nonlinear

gyrokinetic simulations as shown in Fig. 5.8 [193]. The microtearing mode-driven electron thermal transport is plotted as a function of normalized electron collision frequency. As can be seen in the figure, there is good qualitative agreement with the experimentally-observed values, including the observed reduced electron thermal diffusivity with reduced collisionality. The

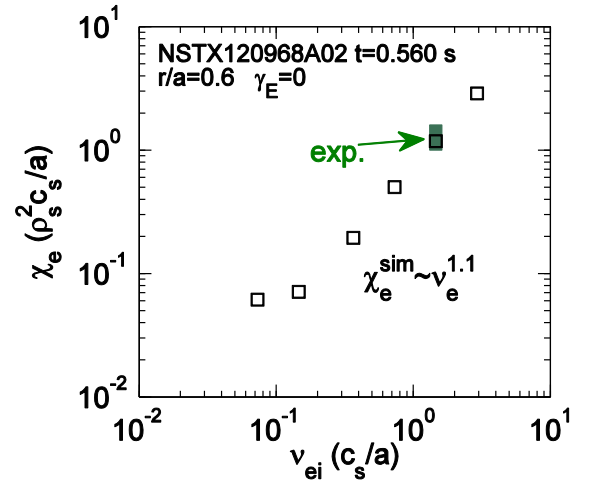


Fig. 5.8. Normalized electron thermal diffusivity vs. normalized electron collision frequency. The shaded square shows the experimental values with uncertainties. All calculations are based on an NSTX H-mode plasma using the GYRO code.

micro-tearing modes therefore may be avoided in future STs with higher toroidal magnetic field and much less collisional plasmas. It should be noted that there is, however, no direct measurement of micro-tearing modes at the present time. This would be a goal of future investigations on NSTX-U and MAST-U.

Anomalous electron thermal energy transport by GAEs – With intense neutral beam injection, it was observed that core electron temperature, remain relatively constant (or even exhibit some reduction). This even holds when the NBI power is tripled in NSTX as shown in Fig. 5.9 (a) [165]. The radial electron temperature profile instead broadens with the NBI power, indicating greatly enhanced electron thermal diffusivity in the core with the applied NBI power as shown in Fig. 5.9 (b). The behavior is observed to correlate with the increased excitation of global Alfvén eigenmodes (GAEs) (see Sec. VIII). Recent measurements by high- k scattering and beam emission spectroscopy diagnostics showed an absence of

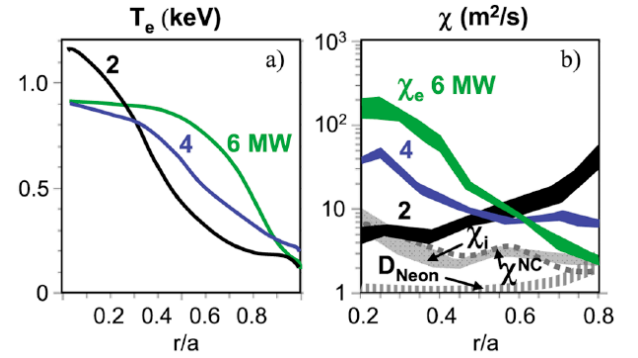


Fig. 5.9. (a) T_e profiles with $P_{NBI} = 2, 4$, and 6 MW NSTX H-modes. (b) TRANSP computed χ_e profiles in the same plasmas. Also shown are the χ_i and NCLASS ion thermal diffusivity for the 6 MW case and the measured neon diffusivity.

any significant level of plasma turbulence, precluding normal turbulent transport in the region of active GAEs. Overlapping GAE modes can resonantly couple to the bulk thermal electrons and induce enhanced stochastic diffusion. The modeling work on GAE-driven electron transport appears to confirm this electron transport mechanism in the core of the NBI-heated plasmas. The proposed effect is potentially important for future STs as well as for any burning plasma where the fusion alphas could drive the GAEs.

E. Toroidal Momentum Transport – Plasma toroidal momentum transport is an active area of transport physics research for tokamaks and STs in recent years, due to its potential importance for future devices including ITER for macrostability and transport physics [21, 22]. The angular momentum balance equation is given by

$$m R \partial (n V P) / \partial t = \eta - (1/r) \partial (r \Gamma_\phi) / \partial r - (1/r) \partial (m r R V_\phi \Gamma_p) / \partial r \quad (8)$$

where m is the ion mass, n is the ion density, η is the local torque density, Γ_p is the (radial) ion particle flux (which gives rise to a typically small convection of momentum), and Γ_ϕ is the (radial) angular momentum flux which can be written as $\Gamma_\phi = -m n R \chi_\phi \partial (V_\phi) / \partial r + m n R V^{Pinch} V_\phi$, including a radial momentum pinch term, V^{Pinch} . As can be given by Eq. 8, for a given local toroidal torque density η (both intrinsic and externally applied for example by NBI), the toroidal rotational profile is determined by the toroidal momentum transport or flux. As described in the RWM section (III.C), the plasma rotational profile is crucial for RWM stability in high beta operation for STs. The $E \times B$ shear produced by plasma rotation shear is

believed to be partly responsible for the neoclassical level ion transport generally observed in NSTX and MAST as described in Sec. C. The anomalous momentum transport is believed to be driven by ion-scale micro-turbulence such as the ITG, TEM, and KBM, which can also drive anomalous ion energy transport. In Fig. 5.10, the measured ion thermal and momentum diffusivities in an NSTX H-mode plasma are plotted along with the corresponding neoclassical values computed by GTC-NEO [194]. As can be seen from the figure, unlike the relatively large neoclassical ion thermal diffusivities, the neoclassical momentum diffusivity is essentially zero. This presence of an anomalous effective momentum diffusivity may indicate the presence of residual ion scale turbulence not apparent in the ion energy transport due to the large neoclassical heat transport. In a perturbative experiment, it was also possible to deduce the radial momentum pinch term as well as the momentum diffusivity. A large inward V^{Pinch} of up to 40 m/s is observed between $0.6 < \rho < 0.8$ [195]. It is also noted that if the inward pinch is assumed to be zero, then the inferred effective momentum diffusivity is reduced by more than a factor of two. The momentum pinch drive may be caused by low-k turbulence, which theoretical models [196, 197] then predict a $V^{Pinch} \propto \chi_\phi / R$, which appears to be consistent with the experimentally-

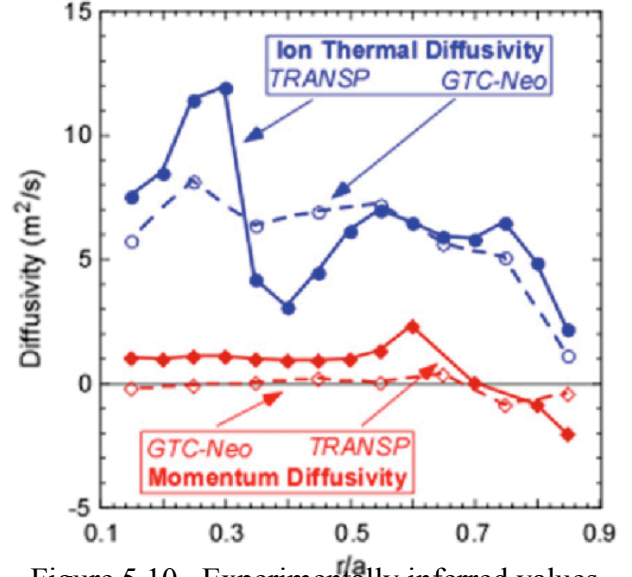


Figure 5.10. Experimentally inferred values of χ_i and χ_ϕ compared with the neoclassical values computed by GTC-NEO for an NSTX H-mode plasma.

observed trend. To further understand momentum transport, intrinsic plasma rotation (without any toroidal momentum injection) has been studied in tokamaks [198]. Intrinsic plasma rotation was also investigated in MAST where the gas fueling location was found to be important for H-mode access [199]. Recently, in NSTX ohmic plasmas, the intrinsic rotation generated in the edge was found to be well correlated with change in the ion temperature gradient [200]. This observation is consistent with the theory of residual stress. Overall, the momentum transport physics, while far from well understood, appears to have made a remarkable progress in recent years for both STs and tokamaks. Low-k turbulence diagnostics such as BES are being utilized to better understand turbulence-driven momentum transport physics.

F. Particle and Impurity Transport -

Particle transport physics is of importance because of fueling, impurity dilution, and accumulation issues. For example, if the particle confinement is too good, impurities, including fusion alpha particles, can accumulate in the plasma core. This could lead to not only fuel dilution, but also eventual

radiative collapse of the main plasma as observed in long-pulse ELM-free H-mode plasmas. Nevertheless, particle transport physics is not as well investigated as energy transport physics, perhaps due to difficulties in measuring ion density profiles in any given experiment. Particle transport physics also requires tracking various ionization states, particularly for higher z impurity species. Also, from neoclassical theory, collisions among ion species could modify the neoclassical ion transport. Particle transport is therefore a complex coupled problem comprising all particle species, including electrons and ions. While particle neoclassical transport theory and modeling tools are well developed, anomalous particle transport mechanisms are not well understood either experimentally or theoretically. Impurity transport has been investigated in tokamaks and STs using controlled impurity injection techniques such as impurity gas and pellet injection. In CDX-U, impurity transport physics was investigated using VUV spectroscopic techniques. The measured core impurity diffusivity and inward pinch velocity are consistent with neoclassical analysis [201]. In NSTX, impurity transport physics was investigated using the tangential (scintillator based) multi-energy soft x-ray (SXR) array, which has identical groups of overlapping sightlines that view the same plasma volume through filters with different cutoff energies [202 - 204]. Earlier neon gas puff experiments in the NSTX L-mode showed close to neoclassical transport behavior, confirming the earlier CDX-U results [201]. The impurity transport in the H-mode also showed diffusivity levels consistent with neoclassical predictions [205]. The study also found that the plasma rotation can enhance diffusive and convective coefficients for heavy and not fully stripped impurities. For neon gas puff experiments in the H-mode, the time histories of the neon emissivity profiles after the injection were modeled using the one dimensional (radial) and time-dependent Multiple Ionization Stage Transport (MIST)

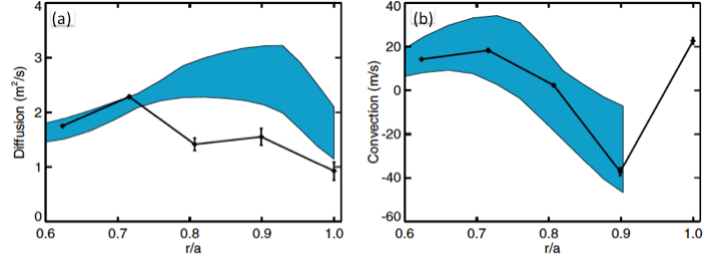


Figure 5.11. (a) The resulting diffusion profile (b) and convection profile for the $I_p = 1.1$ MA, $B_T = 0.55$ T NSTX H-mode case. The shaded region represents the results from NCLASS neoclassical transport calculations.

code. The resulting neon diffusivity and convection velocity profiles are shown in Fig. 5.11 as labeled [195]. There, the diffusivity is large (several m^2/s) at the outer radius ($r/a > 0.8$) while the core diffusivity is small ($\leq 1 \text{ m}^2/\text{s}$). In the figure, the corresponding neoclassical values are calculated by NCLASS [196] and plotted, showing generally good agreement with the experimentally-observed values for the core region. Also, the effect of plasma rotation on impurity transport has been studied, which could explain an observed deviation from NCLASS without invoking anomalous transport. The comparison has been done for other ion species, and particle transport behavior in both L-mode and H-mode plasmas appears to be consistent with the NCLASS values in the NSTX core region. The edge region is believed to be anomalous, but actual quantitative measurements of particle transport is quite difficult because of the influence of edge recycling.

VI. H-mode physics in STs

A. Introduction and Motivation - The so-called “high” confinement mode or H-mode has been a topic of intense tokamak research since its discovery in the early 1980’s [206]. In tokamaks, the H-mode has been of interest particularly for its good (high) confinement and high pedestal pressure. The ITER design goal of fusion gain of $Q = 10$ is based on H-mode access with a sufficient confinement H-factor of (i.e., $H \sim 1$) and pedestal height [21, 22]. As described in Sec. V, the Q depends very sensitively on H and it tends to increase with square of the pedestal pressure. H-mode access is also quite vital for compact ST reactors. A typical compact ST-based FNSF as discussed in Sec. II assumes relatively high confinement factors of up to $H \sim 1.2$ -1.3. Importantly, the broader current and pressure profiles of H-modes are highly desirable for high bootstrap current fraction non-inductive operation as discussed in Sec. III. While H-mode research has a long history, many of the important physics insights are just starting to emerge in the recent years, owing to the newly available diagnostics and analysis tools. Since the H-mode “barrier” and resulting “pedestal” occur near the plasma edge region (where the plasma aspect ratio is the lowest), the relevant H-mode physics in STs takes place in a truly low-aspect-ratio regime (compared, for example, to the core transport physics which take place in the higher aspect-ratio region). The ST H-mode pedestal also generally occurs at higher plasma beta values compared to tokamaks. With accentuated pedestal parameters, H-mode studies in STs can therefore contribute substantially to on-going worldwide H-mode research. Since global MHD stability and confinement properties of H-mode plasmas have been already discussed in Sections III and V, respectively, we will devote the present section to the H-mode-specific physics issues, pointing out some unique features for STs. In Sec. B, the H-mode transition and power

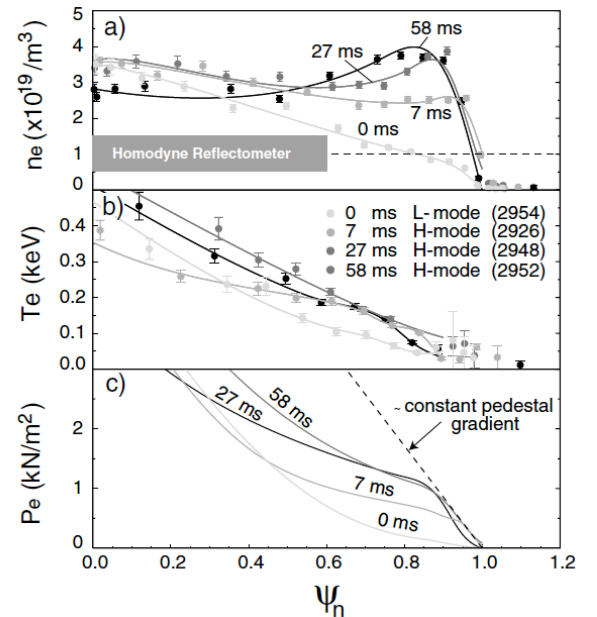


Fig. 6.1. Evolution of the thermal electron radial profile vs normalized poloidal flux: (a) density, (b) temperature, and (c) pressure.

threshold research for STs is discussed. In Sec. C, the H-mode pedestal stability and ELM physics is described, and in Sec. D, the ELM control tool development is summarized.

B. H-modes transition and power threshold – For STs, H-modes were first observed in START [206] and subsequently studied in NSTX and MAST, as its access was facilitated by auxiliary heating power and fine plasma divertor control [199, 207 - 225]. Recently, the H-mode was also observed in PEGASUS ohmic plasmas [226]. Overall, an H-mode in STs looks and behaves qualitatively similar to that in tokamaks. This observation reinforces the robustness of fundamental H-mode physics for tokamaks and STs. In Fig. 6.1, typical H-mode plasma profiles measured in MAST are shown [209]. A strong density barrier is particularly evident near the edge, giving the well-known box-like profile ($t \geq 7$ ms) compared to the centrally peaked L-mode plasma prior to the H-mode transition ($t = 0$ ms). Density “ears” form at the plasma periphery due to impurity fueling, evolving over a time scale of ~ 20 ms. The change in T_e (Fig. 6.1 b) is less pronounced. The pedestal pressure gradient (Fig. 6.1 c) remains nearly constant throughout H-mode. These H-mode features are quite similar to those observed in NSTX and tokamaks in general.

One of the important questions that needs to be answered is the H-mode power threshold for future STs and tokamaks including ITER. Since auxiliary heating power is quite expensive, it is highly desirable to achieve H-modes with minimum auxiliary heating power. The H-mode threshold power in STs is observed to vary greatly (by as much as an order of magnitude) with the vacuum/wall conditions and the plasma configurations. Once the vacuum and walls are well conditioned, the H-mode becomes accessible even with relatively modest sub-MW auxiliary heating power. With aggressive “boronization” and/or “lithiumization” as

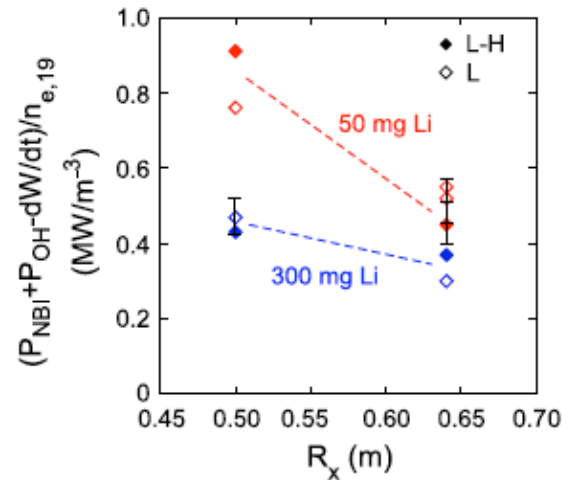


Figure 6.2. Density-normalized loss power as a function of X-point radius for two different lithium evaporation rates. The solid symbols denote discharges that have transitioned into the H-mode at that loss power, while the open symbols denote those that remain in the L-mode.

described in Sec. VII, the H-mode power threshold becomes quite low, enabling the H-mode access even in ohmic plasmas [199, 207, 227]. An in-board gas puff also reduces the H-mode power threshold. The benefit of in-board gas puff for H-mode access appears to be significant for STs compared to tokamaks [199, 207, 221]. The in-board gas puff has been therefore routinely used in MAST and NSTX, and it enabled PEGASUS to achieve H-mode in ohmic plasmas [226]. With this inherent power threshold variability in mind, controlled H-mode threshold experiments were conducted on NSTX and MAST in well-conditioned plasmas [214, 215]. Since the non-activation phase of ITER is likely to use either hydrogen or helium plasmas, the H-mode power thresholds were compared for deuterium and helium plasmas. The results indicate that the L-H power threshold is approximately 20-40% greater in helium than in deuterium in NSTX [214] and about 40% in MAST [215]. These observations are similar to those in tokamaks except the power threshold is still a few times that of the scaling based on tokamaks [228]. Wall conditioning with lithium reduced the L-H power threshold in NSTX significantly, by as much as $\sim 60\%$ as shown in Fig. 6.2 [214]. Also in NSTX, the power threshold increases with increasing I_p and decreases with increasing R position of

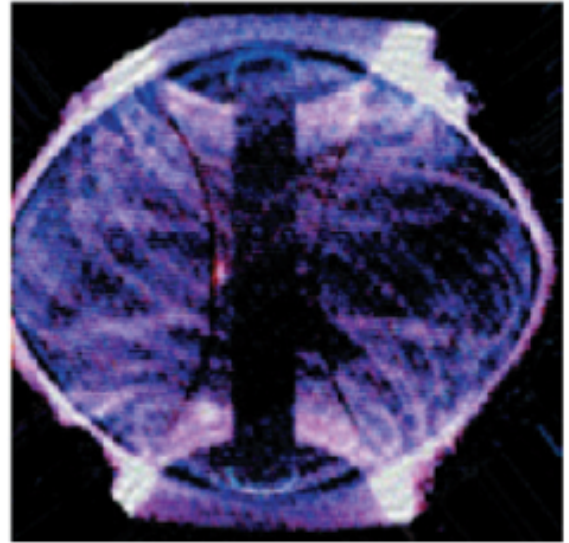


Fig. 6.3. High-speed video images of MAST plasmas with an ELM event, showing evidence of a filamentary structure.

the x-point location as shown in Fig. 6.2 [214, 217]. Both trends are consistent with the increased threshold with toroidal magnetic field at the x-point for a given q . This result is consistent with the prediction that the kinetic neoclassical transport, including ion orbit loss, sets

the edge radial electric field and the ExB shear available to sustain the H-mode transport barrier. In MAST, the power threshold was investigated as a function of x-point height and κ . Both NSTX and MAST saw an increase in the H-mode power threshold with application of resonant magnetic perturbations (RMPs) [214, 215, 220]. This observation is similar to those from tokamaks and is somewhat expected since RMPs tend to degrade edge confinement (or change edge rotational shear.) There are some differences between MAST and NSTX, where in MAST, there is no obvious correlation with E_r shear profiles [215], but in NSTX, some dependence on the E_r shear was observed [214]. Also in MAST, there is no evidence for a critical T_e needed to access H-mode, but in NSTX, a critical T_e could explain the reduced threshold power with lithium which increased the edge T_e . Another notable observation is a strong minimum of the L-H power threshold at $drsep \sim 0$ (balanced double null configuration) [213, 218]. The effect observed in STs is much stronger than that observed at higher R/a. Overall, while there has been very good progress in obtaining an extensive H-mode experimental data base for STs and tokamaks, the fundamental understanding needed to explain the observed H-mode power threshold and develop an associated predictive capability is still lacking today.

C. H-mode pedestal and edge localized modes (ELMs) – Another active area of research involves H-mode pedestal and Edge Localized Modes (ELMs) physics. The ELMs are periodically-

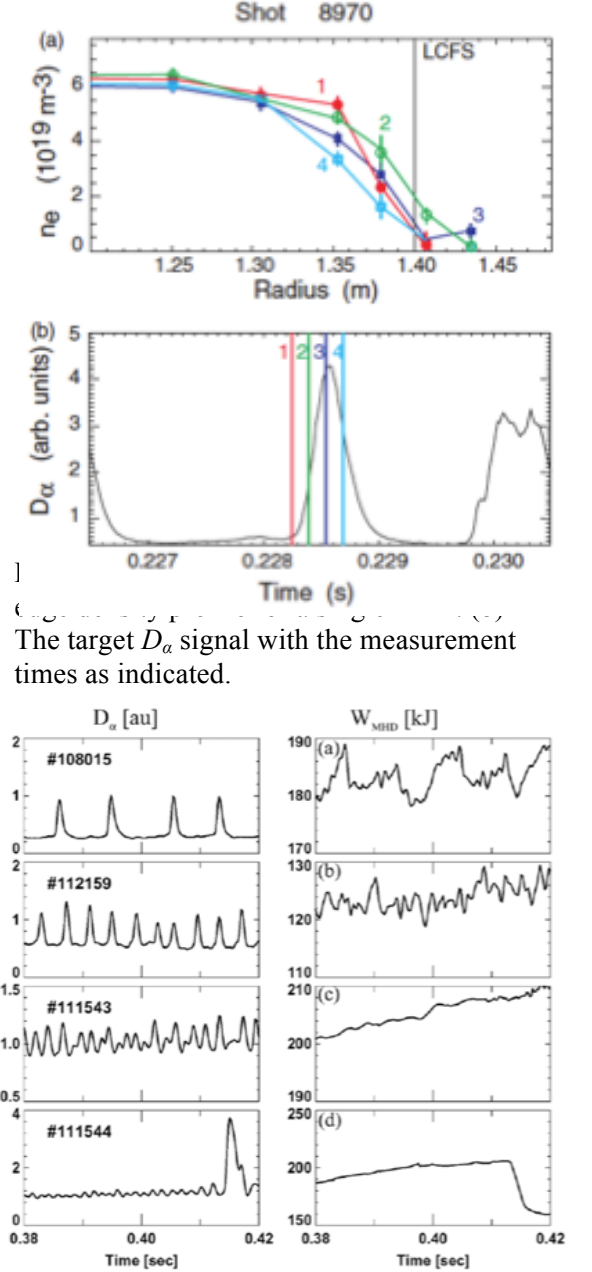


Fig. 6.5. Examples of different ELM types in NSTX: (a) large, Type I in DN configuration, (b) medium Type III in configurations close to DN, (c) small, Type V in LSN configuration and (d) mixed Type I/Type V in LSN configuration.

bursting modes occurring in the edge region of H-mode plasmas. They act as a regulating mechanism for the H-mode pedestal, and ELMs also facilitate the rapid expulsion of plasma both energy and particles from the pedestal region to outside the last closed flux surface (LCFS). The ST configuration readily allows global views of ELM activity as shown in Fig. 6.3, where the high-speed video images of MAST plasmas at the start of the ELM cycle show a global filament structure of $n \leq 10$ [224, 225]. In order to enhance the ELM image, the background plasma light from the preceding non-ELM frame has been subtracted. The time evolution of the edge density profile during ELM was captured by timing the four Nd-Yag lasers of the Thomson scattering diagnostic in MAST as shown in Fig. 6.4 (a). The laser

timing is indicated in Fig. 6.4 (b): (1) The solid red circles and lines indicate $\sim 50 \mu\text{s}$ before the ELM. (2) The open green circles and lines are $\sim 100 \mu\text{s}$ after the start of the ELM. (3) The solid blue squares and lines are $\sim 250 \mu\text{s}$ after the start of the ELM, and (4) the open cyan squares and lines are $\sim 400 \mu\text{s}$ after the start of the ELM. As shown in Fig. 6.4, the ELM event rapidly expels the plasma density from the pedestal.

There are a number of different types of ELMs observed in STs as shown in Fig. 6.5 [222]. The type I and III ELMs are regularly observed in tokamaks as well as in STs as shown in Fig. 6.5 (a) and (b), respectively. As shown in Fig. 6.5, the large Type I ELMs can produce 5-15% range stored energy reduction while the intermediate Type III can produce a 1-5% reduction. The Type V ELMs as shown in Fig. 6.5 (c) and (d) appear to be unique to NSTX [222, 223]. The Type V

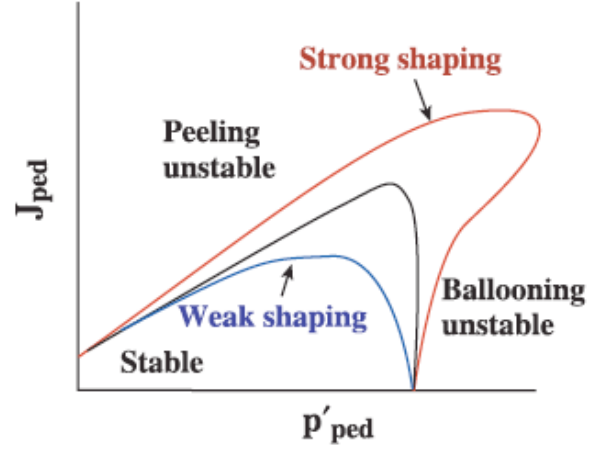


Fig. 6.6. Schematic diagram of the peeling–ballooning stability limit for different shaped discharges as a function of edge current and pressure gradient.

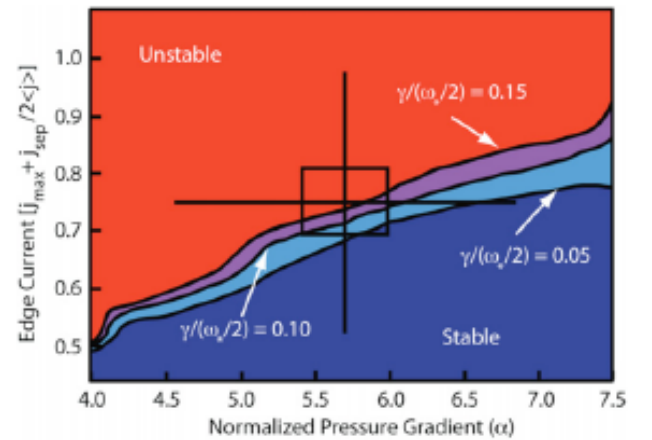


Fig. 6.7. Peeling–ballooning stability diagram as calculated by the ELITE code for NSTX H-mode during the type-I ELM phase.

ELMs which occur at higher edge collisionality, are observed by several diagnostics but have no measurable impact on the stored energy. A high-speed camera image of the Type V ELMs shows that the global filament structure (usually one or two compared to multiple for Type I or III), is very narrow (~ 6 cm) and only lasts for a small fraction of the time of the larger ELM events. If the Type V ELMs can be extrapolated to future devices, it could be an attractive operating regime for next step fusion experiments including ITER.

The large Type I ELMs could introduce a high heat flux (predicted to be as high as 1 GW/m^2 for ITER-class plasmas) onto the divertor plates. This is due to the collapse of the H-mode pedestal, and could seriously damage the plasma-facing components in future large STs and tokamaks. Therefore, the stakes are extremely high for fully understanding H-mode

pedestal physics, and developing satisfactory predictive capabilities to assure safe and acceptable H-mode operation. The H-mode pedestal stability and ELM excitation have been successfully described by the peeling-ballooning MHD stability model for tokamaks (included in the ELITE code) [229, 230]. According to this model, the H-mode pedestal stability is mainly determined by the current-driven kink-peeling modes and the pressure-driven ballooning modes as shown in Fig. 6.6. As also shown in the figure, the plasma shaping could expand the stable boundary. For STs, the strong shape-factor is therefore expected to play an important role in the pedestal stability. The peeling-ballooning stability diagram for the Type I ELM in NSTX is shown in Fig. 6.7 [231], and for a similar analysis performed for the MAST Type I ELM in Fig. 6.8 [232]. For NSTX, the ELMing discharges lie close to the peeling mode stability boundary, while for MAST, the experimental point appears to lie close to the tip of the peeling-ballooning

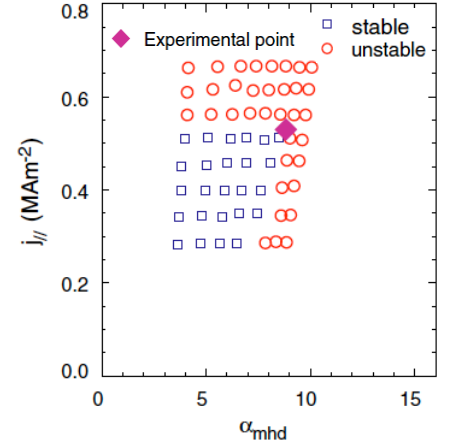


Fig.6.8. Stability diagram plots from ELITE showing edge current density versus normalized pressure gradient for Type I ELM in MAST.

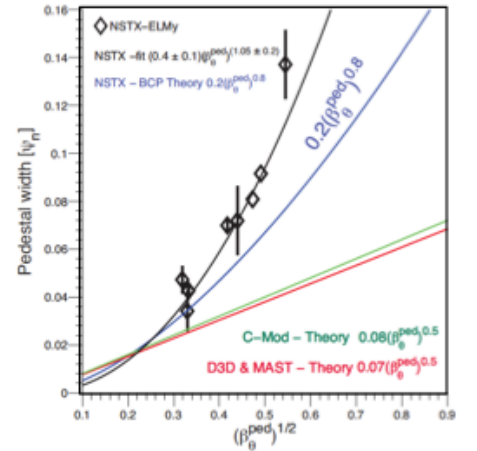


Fig. 6.9. The pedestal width (in ψ_n) scaling with $(\beta_q^{\text{ped}})^{1/2}$. This width scaling effectively provides a relation between the width and the height of the pedestal. The best fit for NSTX width is $(\beta_q^{\text{ped}})^{1.05}$. A theoretical model applied to NSTX data using BCP shows a $(\beta_q^{\text{ped}})^{0.8}$ scaling.

stability boundary. We also note that the PEGASUS H-mode also appears to be inside the peeling mode stability boundary, perhaps due to the strong shape characteristic of the ultra-low aspect ratio geometry [233]. Due to generally higher ballooning limits for STs, the first access to the kink/peeling branch is expected. The H-mode pedestal creates a strong pressure gradient which induces significant edge bootstrap currents, so the edge current and pressure are coupled. An observation which may illustrate the difference between STs and tokamaks is shown in Fig. 6.9, where the pedestal width is plotted as a function

of the pedestal poloidal β [234]. The NSTX pedestal width follows a $\Delta = c(\beta_{\theta}^{\text{ped}})^1$ dependence, which is quite different from the $\Delta = c'(\beta_{\theta}^{\text{ped}})^{0.5}$ scaling for MAST [232] and DIII-D [235]. The ~ 1.0 exponent scaling observed for NSTX has been shown to be consistent with preliminary analyses using the ballooning critical technique (BCP) [236], which has yielded a 0.8 exponent scaling of the pedestal width as shown in Fig. 6.9. It should be noted that the BCP model shows a 0.5 exponent for standard aspect ratio tokamaks. This example shows the good progress being made in understanding of the pedestal physics.

D. ELM control – An important research area for ELMs is their control. If one can reliably stabilize

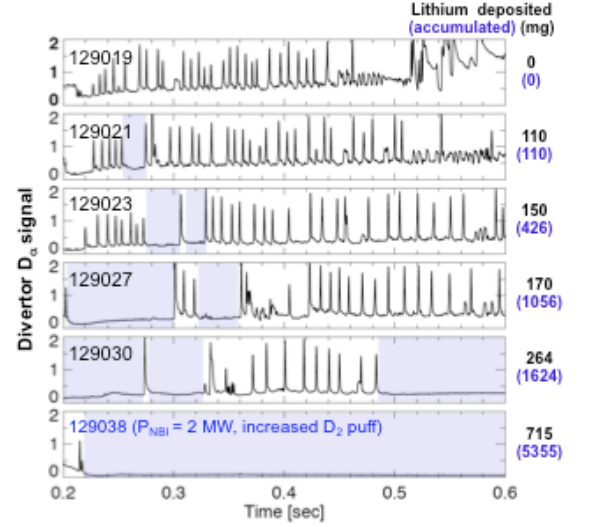


Fig. 6.10. Temporal edge D-alpha signal for various lithium deposition rate. The regularly occurring spikes represents the Edge Localized Modes (ELMs).

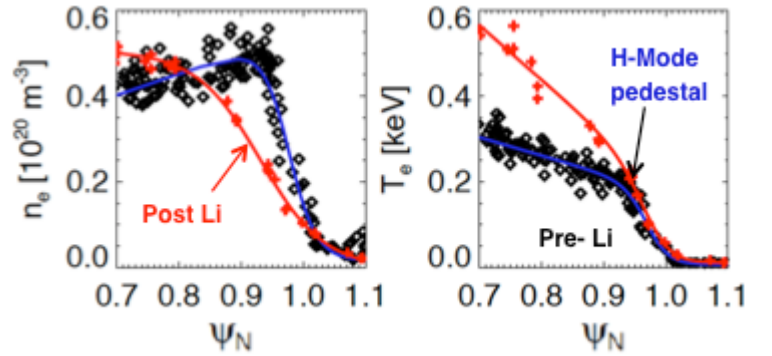


Fig. 6.11. Profiles for n_e and T_e for pre- and post lithium discharges (black and red lines, respectively).

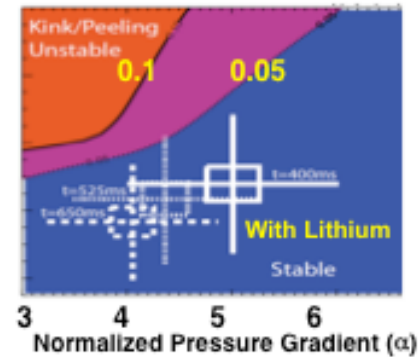


Fig. 6.12. Stability boundary (blue to orange color transition) from ELITE code with fixed boundary kinetic EFITs for post-lithium discharge.

ELMs or reduce their virulence, there will be a huge benefit for future tokamaks and STs including ITER. This is therefore a very active on-going research topic. Recently, ELMs were stabilized by the stochastic plasma boundary produced by a set of resonant magnetic perturbation (RMP) fields ($n=3$) on a tokamak [237]. The H mode transport barrier and core confinement were unaffected, despite a threefold drop in the toroidal rotation. Subsequent experiments on NSTX and MAST with similar RMP fields, however, resulted in varied effects on ELMs, even though a sufficiently stochastic boundary should have been created in the ST experiments. On MAST, RMP fields changed the ELM frequencies [238].

On NSTX, the application of RMP fields did not stabilize ELMs, but actually triggered ELMs in lithium-induced ELM-free plasmas [239-242]. Quite unexpectedly, lithium application in NSTX has resulted in a complete suppression of ELMs as shown in Fig. 6.10 [243, 244]. Plasmas previously exhibiting robust Type 1 ELMs gradually transformed into discharges with intermittent ELMs, and finally into continuously-evolving ELM-free discharges as the lithium evaporation rate is increased. The main change in the edge pedestal region is the reduction of density in the plasma scrape-off layer, and increased electron temperature in the pedestal region of the plasma as shown in Fig. 6.11. Overall, however, the pedestal plasma pressure is much more enhanced for the lithium-stabilized ELM case. The increase in the pedestal pressure that appeared to result in ELM suppression seems contradictory at first. It should be pointed out however that while the H-mode pedestal pressure height is enhanced, the pedestal pressure gradient is not, i.e., the pedestal width increased dramatically. A detailed edge pedestal stability analysis using the ELITE code showed that the ELM-ing plasma started out to be near the kink/peeling mode instability boundary (as shown in Fig. 6.7). It then moved well into the stable region as shown in Fig. 6.12, because of the inward shift of the pressure and edge current peak into the reduced magnetic shear region [243, 245].

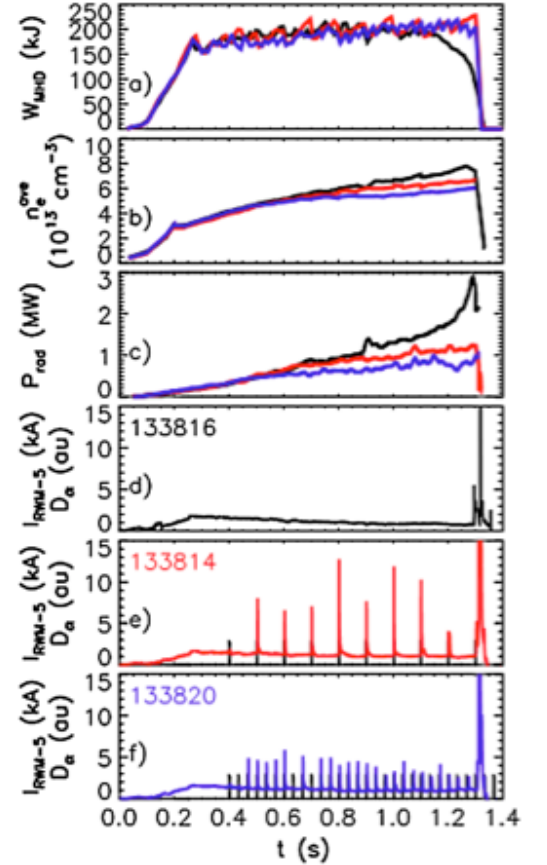


Fig. 6.13. Comparison of discharges with lithium conditioning only (black line) and with combined lithium and 3D field-induced ELMs (red and blue or gray lines): (a) stored energy, (b) n_e^{ave} , (c) P_{rad} , and (d)–(f) $I_{\text{RWM-5}}$ and D_α emission.

On NSTX, ELM-free discharges exhibited impurity and radiation buildup, often leading to plasma disruptions. The application of a $n=3$ RMP field on NSTX was able to trigger the ELMs with relatively high reliability, as shown in Fig. 6.13 [239-241]. As shown in Fig. 6.13, by actively triggering the ELMs, the impurity accumulation was reduced sufficiently to prevent radiative collapse, but without degrading the plasma confinement in high performance discharges. Increasing the ELM triggering rate also reduced impurity levels and the ELM-triggered stored energy loss. The ELM triggering mechanism is being investigated with a 3-D MHD modeling [242]. It should be also noted that this ELM triggering is also known as the ELM pacing, and other techniques such as the repetitive lithium granular injection and vertical jogs have been successfully developed [246, 247]. These ELM stabilization and de-stabilization results show the complex nature of the H-mode ELM stability, and highlights the need for further research both in tokamaks and STs.

VII. Boundary Physics

A. Motivation - Boundary physics is arguably the most challenging area of research for developing an attractive magnetic fusion reactor design.

Fusion reactors such as FNSF and Demo are envisioned to operate steady-state at much higher heat fluxes (2 – 3 x) than in present-day operating divertors, including conditions expected in ITER. The smaller major radius and higher power density of STs can produce very high divertor heat fluxes, so that measures of divertor heat load such as “P/R” can be correspondingly high. The peak divertor heat flux for presently-operating STs can regularly exceed 10 MW/m^2 [248] and transiently reach

much larger values during ELMs and disruptions to approach values expected in ITER. The ST magnetic field pitch can be quite steep, with the pitch angles $\sim 45^\circ$

(or $B_T \sim B_P$) at the outboard mid-plane. This is nearly an order of magnitude larger than that of typical tokamaks as shown in Fig 7.1 [249]. This sharp field pitch and small major radius makes the divertor connection length, i. e., the distance along the field line between the outer mid-plane and the divertor plate, to be relatively short. The heat from the core plasma is expected to flow out in the outer mid-plane region due to the unfavorable field curvature, and the heat is readily exhausted along the field lines into the divertor region. Typically $\sim 50\%$ of the plasma heat is expected to reach the divertor plate, while the rest can be dissipated through various mechanisms including radiative losses. The short connection length of STs tends to make the peak heat flux high, since there may not be sufficient time for radial heat spreading via turbulence and other mechanisms. On the other hand, due to its relatively low toroidal field and smaller divertor major radius (i.e., strong toroidicity), cross-field “anomalous” transport can be greatly accentuated in STs. The large mirror ratio along the divertor field line could also affect the parallel heat transport [250]. Another potential positive feature for the ST divertor configurations is the flexibility of the ST geometry, which can greatly increase the divertor flux expansion. For example, the snow-flake, x-, and super-x configurations can produce a divertor

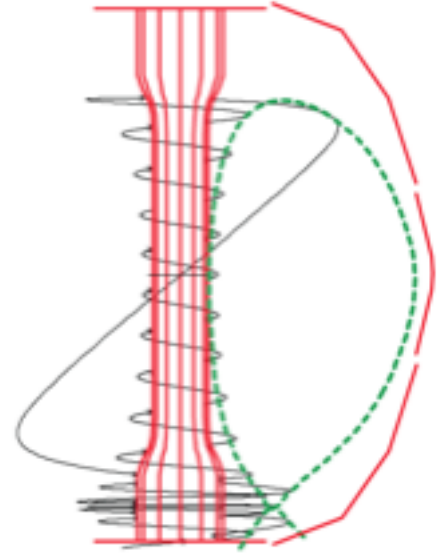


Fig. 7. 1 . Visualization of the field line on the flux surface $y_N = 1.005$ in the NSTX standard divertor.

flux expansion of up to ~ 50 as described below. Indeed, these low aspect ratio features of STs enable highly-versatile studies of boundary/divertor physics and the exploration of attractive divertor solutions. In Sec. VII-B, divertor heat load and divertor configurations are described. We should note that the region we consider in this section is outside of the last closed flux surface. The H-mode and ELM effects were discussed in Sec. VI. In Sec VII-C, the edge turbulence and transport behavior is summarized. In Sec. VII-D, the plasma facing component (PFC) and divertor boundary research using lithium is described.

B. Divertor heat loads and their mitigation

Inboard/outboard divertor heat load asymmetry - The ST configuration, with small major radius and relatively short connection length, can produce very high divertor heat fluxes over 10 MW/m^2 even in present-day devices. On the MAST device, a strong asymmetry of power sharing between the inboard and outboard divertors was found. The shallow field line pitch for the inboard divertor in contrast to the strong outboard field line pitch can be seen in Fig. 7.1. The power flows predominantly (i. e., $P_{\text{out}}/P_{\text{in}} \sim 50$) to the outboard side, which is also the case during ELMs in that $P_{\text{out}}/P_{\text{in}} \sim 20$ [251]. Since it would be indeed technically challenging to handle the high heat flux for the inboard divertor due to the limited area and access, this very large power partition asymmetry toward outer divertor is quite favorable for the ST concept. Similar very low heat fluxes for the inner divertor was observed in NSTX plasmas [252, 253]. The inherently low heat flux enables the inner divertor to “detach,” which further reduces the heat flux. Because of the very low power flux for the in-board divertor in STs, the main emphasis for diverter heat flux mitigation research has been focused on the outboard divertor for STs.

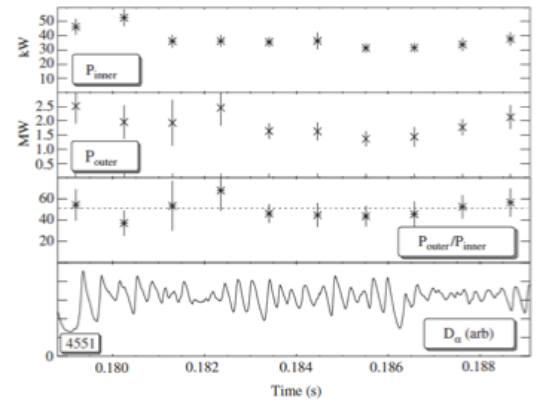


Fig. 7.2. Total power to the outboard and inboard targets (summing contributions from the upper and lower targets in each case) for an L-mode phase of shot 4551, with all strike points fully attached. More than 98% of power arriving at the targets is detected at the outboard side.

Heat flux mitigation by flux expansion - In NSTX, utilizing the high power neutral beam injection (NBI) system, a single null discharge with a moderate triangularity of 0.4 was observed to have a peak outer divertor heat flux during the H-mode that increased with NBI power, such that where the peak divertor heat flux, $q_{div, peak}$, reached as high as $\sim 10 \text{ MW/m}^2$ for $P_{NBI} \sim 5 \text{ MW}$ [248, 254]. This level of observed $q_{div, peak}$ on NSTX is similar to that expected for ITER. To develop a method to reduce $q_{div, peak}$, a number of experiments were performed on NSTX. One of the most reliable ways to reduce $q_{div, peak}$ is by the so-called the divertor flux expansion, f_{exp} . If the magnetic field flux from the mid-plane region can expand to a larger surface area, the peak heat flux is usually reduced accordingly. It is therefore instructive to look at the $q_{div, peak}$ on f_{exp} . In Fig. 7.3, the effects of the magnetic

flux expansion on the peak divertor heat flux $q_{div, peak}$ is plotted as a function of flux expansion f_{exp} in highly shaped ($d \sim 0.8$, $k \sim 2.2 - 2.4$), lower single null H-mode discharges with $I_p = 1.0 - 1.2 \text{ MA}$ and $P_{NBI} = 6 \text{ MW}$ in NSTX [254]. The $q_{div, peak}$ is reduced from 8 MW/m^2 to 2 MW/m^2 by increasing f_{exp} from 10 to 40. The ideas for divertor heat flux mitigation through expanding the divertor flux include the snow-flake divertor (SFD) [255] and the X-divertor [256, 257], which are based on additional field null points in close to the usual single null configuration of a conventional divertor. The multiple field nulls make the field null size larger, and cause greater field null expansion in the null region. This type of multiple-null configuration was tested on NSTX, and shown to be quite effective in reducing the divertor heat flux. This configuration is illustrated in Fig. 7.4. (a), and can result in a very large

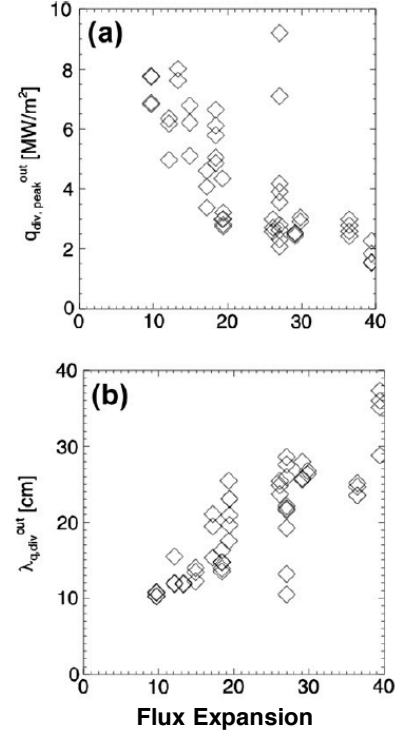


Fig. 7.3. Effect of f_{exp} as measured at the outer strike point in NSTX. (a) Reduction in $q_{div, peak}$ as f_{exp} is increased. (b) Broadening of the heat flux profile, $\lambda_{q, div}$ as f_{exp} is increased.

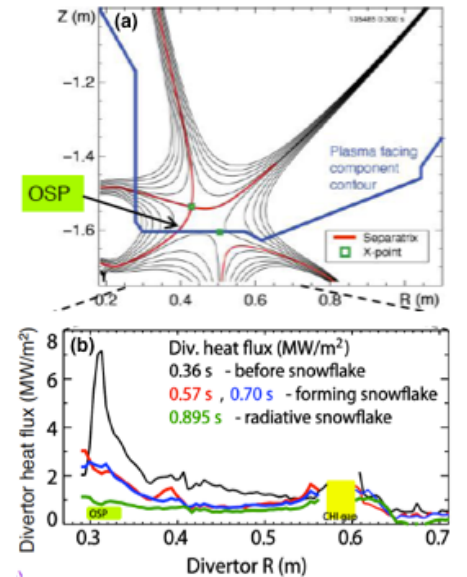


Fig. 7.4. (a) Poloidal flux contours for the asymmetric SFD. (b) Heat flux profile during the SFD discharge as labeled.

effective divertor flux expansion of ~ 50 . As shown in Fig. 7.4 (b), the SFD configuration resulted in a significant reduction of a factor of ~ 3 in the divertor heat flux [249]. It should be noted that the SFD configuration has a possibility of multiple heat flux channels [255]. The SFD configuration in NSTX can also be referenced to the x-divertor configuration. In terms of future possibilities, an innovative concept being considered is the super-X divertor (SXD) configuration [258]. By extending the outer divertor flux line to larger major radius, a large flux expansion is achieved that brings down the plasma temperature and creates a partially detached divertor condition. Another advantage of the SXD is that it significantly increases the divertor connection length, while increasing the divertor volume by expanding it into lower toroidal field region. The MAST device is upgrading its divertor to the SXD configuration as shown in Fig. 7.5. [259, 260]

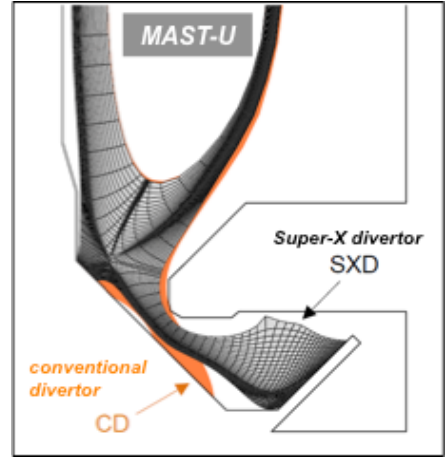


Fig. 7.5. Super-x divertor configuration compared to the conventional divertor planned in MAST-U.

Radiative cooling and other approaches to heat flux mitigation - In addition to the divertor flux expansion, there are other ways to reduce the divertor heat flux. It can be reduced in principle by a factor of two by going to a double null configuration, utilizing the upper and lower divertors for power exhaust. Further progress in reducing the peak flux in NSTX was demonstrated through the partially detached divertor (PDD) regime, resulting in a peak heat flux reduction of up to 60% [261, 262]. Deuterium gas puffing into the divertor area reduces the electron temperature to about a few eV in front of the divertor plate, which is sufficient to facilitate radiative cooling. The radiative cooling effect can be seen in the SFD configuration in Fig. 7.4. (b), where radiative cooling can cause partial divertor detachment and further reduces the heat flux by about a factor of three [249]. The combination of the SFD and radiative cooling can reduce the heat flux by an order of magnitude. It maybe noted that the SFD configuration can increase the volume of the field null region and significantly increase the divertor connection length which tend to enhance the radiative cooling [255]. While radiative cooling and partial divertor detachment together form a powerful tool for reducing the divertor heat flux, a potential weakness of this approach is that it is highly non-linear. A heat pulse caused by an ELM, for

example, can often reduce the effectiveness of radiative cooling. The resulting rise in electron temperature can result in reduced radiation, unless some seed ions of heavier impurities such as argon are introduced. The heavier impurity seeding could result in plasma performance degradation and core impurity accumulation as observed in tokamak experiments.

Heat flux mitigation by lithium – A potentially important observation for divertor heat flux mitigation is the divertor heat flux reduction accompanying the Li coating of divertor surfaces in NSTX [263]. As shown in Fig 7.6, the measurements showed a $\sim 50\%$ reduction in peak heat load on the divertor strike point surfaces with only a modest amount of Li (~ 300 mg) evaporation prior to the discharge compared to 150 mg evaporation. It is estimated that $< 10\%$ of the evaporated Li is deposited over the LLD surfaces. The heat flux reduction is accompanied by an increase in the localized radiation measured by bolometers from the region above the inner and outer strike points. Motivated by this observation, a liquid lithium (LL) based radiative divertor concept termed the RLLD (radiative liquid lithium divertor), has been proposed [264]. The evaporated or injected Li is readily ionized by the plasma due to its low ionization energy, and the poor Li particle confinement near the divertor plate enables ionized Li ions to radiate strongly, resulting in a significant reduction in the divertor heat flux. This radiative process has the desired effect of spreading the localized divertor heat load to the rest of the divertor chamber wall surfaces, facilitating divertor heat removal while maintaining low recycling for improved plasma performance. The modeling results indicated that the Li radiation can be quite strong and explains the NSTX Li results. The same model predicts that only a small amount of Li ($\sim$ a few moles/sec) is needed to significantly reduce the divertor peak heat flux for typical (1 GW-electric) fusion reactor parameters.

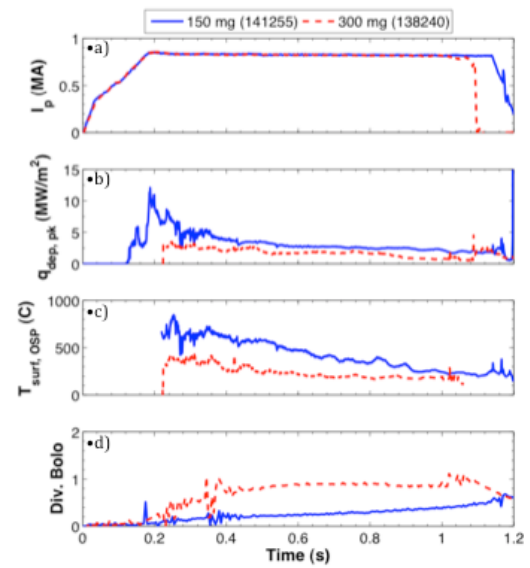


Fig. 7.6. Divertor surface temperature and corresponding heat flux as a function of Li evaporation for otherwise similar NBI heated H-mode discharges. Pre-discharge Li evaporation of 150 mg shown as blue solid lines and 300 mg as red dashed lines.

Effects of 3-D fields - With the utilization of 3-D fields for various purposes (see Sec. III), the effects of 3-D fields on the divertor heat flux have been also investigated in NSTX [265] and MAST [266]. The main effect of the 3-D fields is to cause toroidal and poloidal variations in the divertor heat flux. This effectively increases the peak heat flux locally, since the heat flux can no longer be toroidally uniform. In Fig. 7.7, the divertor heat flux is shown in red for a detached divertor plasma with light gas puffing [265]. As the 3-D ($n=3$) fields is applied, one can see multiple heat flux peaks appear in blue at the locations where the increased heat flux peaks are apparent. At this point, the divertor is back to the attached regime as evident from the increased heat flux. Finally, the heat flux increase associated with ELMs, as indicated by the green trace, can cause reattachment.

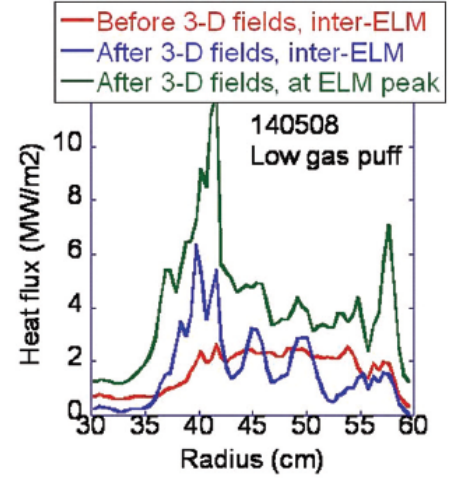


Fig. 7.7. Measured heat flux profiles with and without the 3-D field application, as labeled, for the detached divertor plasma with lower gas puffing.

C. Boundary Turbulence & Transport – As the plasma heat comes out of the core plasma (presumably predominantly from the mid-plane region where there is unfavorable curvature), an important research topic is how the heat (or energy) and particles diffuse out radially (and poloidally) while flowing toward the divertor region (i. e., outside the last closed flux surface) via parallel transport. The radial (turbulent) heat transport could largely determine the peak divertor plate heat flux, which then determines the divertor heat handling design requirements for future devices including ITER. It is therefore of critical importance that a predictive (theoretical) capability is developed to be able to quantify the radial transport. This area of research is actively pursued for tokamaks and STs for its obvious importance. For example, there is a multi-device study that reveals a notable trend in the divertor heat flux.

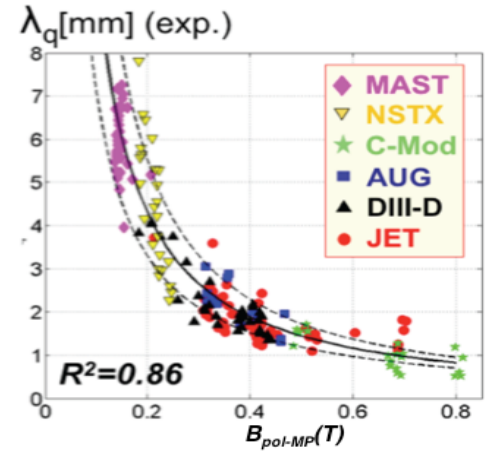


Fig. 7.8. Power fall-off length λ_q at the mid-plane versus the poloidal magnetic field at the outer mid-plane.

This is shown in Fig. 7.8, where the power fall-off length l_q at the mid-plane is plotted as a function of the poloidal magnetic field at the outer mid-plane for various tokamak/ST devices [267]. This is an unfavorable trend, as the $l_q \sim B_{pol}^{-1}$ dependence would predict even higher peak fluxes for the higher B_{pol} (and I_p) operation expected for future devices including ITER and FNSF. As can be seen from the figure, the NSTX and MAST contributions to this database are represented in the lower B_{pol} part of parameter space. This trend can be tested in NSTX-U and MAST-U with the doubling of B_{pol} from NSTX and MAST. This observed trend is consistent in absolute magnitude with the predictions of a recently formulated heuristic drift-based model, assuming non-turbulent particle transport coupled with anomalous electron thermal transport [268].

Edge Turbulence Study – In order to develop a physics-based understanding of the radial transport of heat and particles, there have been extensive edge turbulence transport studies in many fusion devices utilizing primarily visible fast cameras [269], gas-puff-imaging (GPI) measurements [270], edge probes [271, 272], beam emission spectroscopy (BES) [173, 174] together with theory and modeling [273 – 276]. The region of interest is outside of the last-closed-flux-surface (LCFS), and it is in the open field line region. It is therefore not so surprising that the observed edge turbulence in STs is similar to the edge turbulence in large and small tokamaks, stellarators, and RFPs [277]. This is apparently due to the generic drift wave and nonlinear “blob” or “filamentation”

formation mechanisms in all (open field line) toroidal plasmas [273, 274]. The advantage of the ST geometry is illustrated in Fig. 7.9 where the fast camera images of edge turbulence can be obtained for the entire plasma [278, 279]. The full view turbulence images of (a) inter-ELM, (b) L-mode, and (c) ELM phases from the same discharge are shown, and (d) the corresponding intensity traces normalized to the peak ELM amplitude are shown for the three cases. The

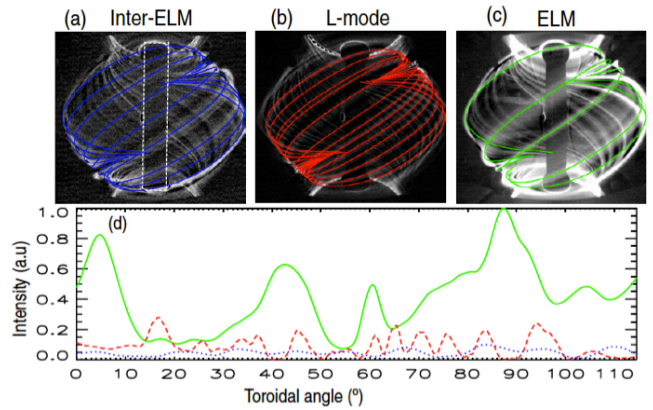


Figure 7.9. Full view camera images of edge filaments for (a) inter-ELM, (b) L-mode and (c) ELM periods within the same discharge (shot #15586), and (d) the intensity traces are superposed normalized to the peak ELM intensity showing the contrast in measured intensities across the three phases.

number of filaments per fixed toroidal angle $\Delta\phi$, which is indicative of the quasi-toroidal mode number, is the largest for the L-mode phase, followed by the inter-ELM, and lastly the ELM phase. The edge turbulence appears predominantly to assume filament-like structures. The inter-ELM and L-mode filaments are observed to move radially with a constant speed, consistent with the $E \times B$ motion. However, the ELM filaments can often accelerate presumably driven by the free energy generated by the H-mode pedestal collapse, as discussed in Sec. VI.

A GPI diagnostic was developed to obtain more detailed local turbulence information and to develop new further insights into the physics of scrape-off-layer (SOL) transport [280, 281]. A puff of neutral gas is injected toward the core from a port in the outboard wall of the machine.

Collisions with plasma electrons stimulate atomic emission from the neutrals and the radiation is captured on a fast camera. Its fast camera views along a local magnetic field line, where the 2-D (radial and poloidal) time evolution of plasma edge fluctuation images are enhanced by a localized D_2 gas puff. Except for the modest amount of gas introduced in the region to illuminate the edge turbulence, the GPI is considered to be non-perturbing diagnostic; in contrast, edge probes could be perturbing. The

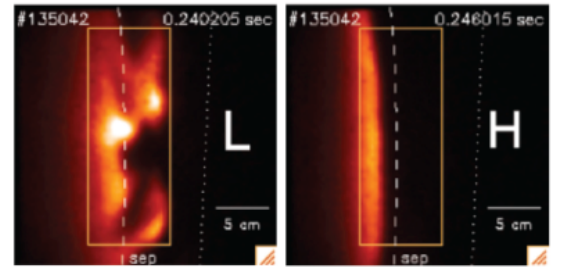


Fig. 7.10. Typical GPI images of the light emission in the NSTX L-mode and H-mode. Also shown is the best estimate for the separatrix location (dashed line) and the shadow of the rf antenna/limiter location (dotted line.)

resulting images (an exposure time of 3 ms / frame at 64x64 pixel resolution) are shown in Fig. 7.10 for an L-mode (left) and H-mode (right), where highly non-linear regions of strong localized light emission, known as ‘blobs’ (which are equivalent to the filaments on MAST) are seen in the L-mode frame [281]. The blobs appear in nearly all discharges (even in an H-mode but with much less frequency and intensity), typically having cm-like radial scale lengths and moving both poloidally and radially in NSTX. The blobs are ubiquitous hotter and denser “plasmoids,” as they are expelled from the inner hotter and denser birth zone, i. e., near the separatrix (last closed flux surface) where $\sim \text{Max} [-\nabla \ln p]$, to the outer colder scrape-off layer. The blobs are highly elongated along the field line, i. e., are filamentary in structure (as shown in Fig. 7.8), owing to the rapid parallel heat conduction. The blobs appear to regulate the radial power transport in the separatrix region by the frequency of their occurrence rather than their magnitude. A cross machine comparison was also made between NSTX and Alcator C-Mod

[282]. A similar study has been done on the MAST device [283]. Very good progress has been made through the comparison of various models (both electrostatic and electromagnetic) with experiment to understand the physics of blobs [275, 276, 284]. The 2D scrape-off layer turbulence (SOLT) code simulates turbulence driven by magnetic curvature and drift-wave effects in a 2D plane normal to the magnetic field $\mathbf{B}$. The simulation domain is the outer mid-plane of the tokamak, encompassing both the edge and SOL regions. The model includes the effects of wave-phase directionality (drift waves and background flows), curvature drive, radial transport (turbulent Reynolds stress and blobs), sheared flows, and dissipation (sheath loss and friction). A GPI frame showing a typical blob from NSTX shot no. 112825 and one from a SOLT simulation are shown in Fig. 7.11 [284] Note that the simulation blob is smooth and poloidally elongated, whereas the experimental blob is smaller and more circular in shape. Factor-of-two agreement between the simulated and experimental number of blobs and size distributions were obtained for the best-case simulations. Blob characteristics such as the blob size and radial velocity have thus far been explained relatively well with the electrostatic-based model. The role of the blobs on energy and particle transport and on the divertor heat and particle flux is under active investigation.

D. Plasma facing material and lithium research

- The low aspect ratio and high power densities in STs provide unique opportunities and challenges for plasma-facing components (PFCs). While large P/R values make special demands on PFCs for high-performance ST plasmas, ST experiments initially used PFC

materials that were common to conventional tokamaks. As mentioned earlier, the first ST in the United States was CDX-U. As with all tokamaks, impurity influx from PFCs limited plasmas performance. The CDX-U vacuum vessel was primarily stainless steel, and the main impurities were oxygen and carbon. Titanium gettering was used to reduce their content in CDX-U plasmas [201].

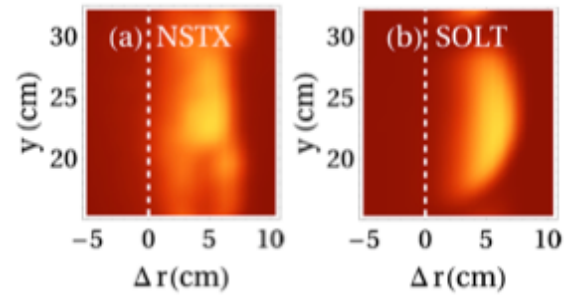


Fig. 7.11. (a) Actual NSTX, shot no. 112825 and (b) synthetic SOLT GPI intensity images. The magnetic separatrix is at $r = 0$ in the NSTX shot. Intensities are normalized by their respective global maxima over the frame.

Larger STs, like most conventional tokamaks, have used graphite as the primary PFC material, and share common conditioning approaches. In NSTX, for example, the vacuum vessel walls were subject before plasma operations to prolonged bakeout at 350°C to remove water from the PFCs, and boronization to control oxygen [285]. Helium glow discharge cleaning (GDC) has also been performed between discharges to remove deuterium in the PFCs, following the prescription for improved plasma performance in earlier tokamaks like TFTR [286]. The TFTR experiments also included the earliest demonstration of lithium coatings as an effective PFC conditioning technique, and the highest plasma stored energy was achieved with its application [287].

The effect of lithium PFC coatings on ST plasmas was investigated extensively on NSTX. The compact geometry of the ST makes it amenable not only to imaging plasmas in the main vacuum chamber, but also in the divertor region. Such imaging is important because of the toroidal and poloidal asymmetries in the divertor heat flux that occur naturally and are introduced with 3-D fields as discussed earlier. To this end, two fast visible cameras were absolutely calibrated and installed on NSTX-U, with views that covered the entire lower divertor. A coherent fiber optic bundle coupled the light from the viewports to the cameras, and narrow bandpass filters for carbon and lithium lines and the CD band in a filter wheel between the focusing and collimating lenses allowed spatially-resolved measurements of the impurity emission [288]. Divertor surface temperatures were obtained with a two-color IR camera. The image was split and passed through filter with bandpasses of 4-6 μm and 7-10 μm . This removed any uncertainties introduced by assumptions about the surface emissivity. The IR camera has good sensitivity over a temperature range from 100 to 700C, and can acquire images with a time resolution of less than a millisecond [289].

The first apparatus used to introduce lithium into NSTX were the lithium pellet injector (LPI) and lithium evaporator (LITERs) [179]. Lithium was injected into NSTX discharges with the LPI, and the LITERs consisted of lithium-filled ovens with apertures that directed lithium vapor toward the lower NSTX divertor region. Both techniques resulted in the enhanced absorption of deuterium by the NSTX walls. In particular, the improved confinement using the LITERs was the first demonstration of the efficacy of the approach for divertor H-mode plasmas as well as limiter discharges as in TFTR, where the value of lithium PFC coatings was initially shown. The

empirically observed effectiveness of lithium-coated graphite to absorb deuterium is unexpected from stoichiometry of lithium and carbon. It predicts the formation of lithium carbide, which is unable to bind additional deuterium. Quantum-classical atomistic simulations have shown, however, that in the presence of oxygen, the deuterium can be bound in a lithium-carbon-oxygen-deuterium system. This has been supported by laboratory experiments that investigated the effect of oxygen on deuterium uptake in lithium-coated graphite samples [290].

Salient characteristics of the H-mode plasmas achieved with lithium PFC coatings, as described earlier, include enhanced pedestal temperatures and stabilization of ELMs. While the latter is attractive as it reduces heat loads on PFCs, it has also been associated with the accumulation of carbon in the core plasma. Among the unexpected results from experiments with lithium evaporation on NSTX PFC surfaces is the amount of lithium relative to carbon in the plasma core [291]. Analysis of impurity transport indicated that ELM suppression and changes in neo-classical transport can lead to the observed carbon accumulation, and that high core carbon concentrations enhance neoclassical lithium particle diffusivities [292]. This mechanism alone is not sufficient for low core lithium concentrations, and further research into a more complete explanation will be conducted in NSTX-U.

The core carbon accumulation limits discharge duration, so a means by which the carbon can be reduced through controlled ELM destabilization (“ELM pacing”) is desirable. A technique for ELM pacing is the injection of lithium granules (“micropellets”) at a controlled frequency into the discharge. An oscillating piezoelectric disk causes submillimeter-sized granules to fall through a hole in its center, and they are subsequently propelled into the plasma by a set of rotating blades. The system was originally designed to another approach for lithium wall conditioning in NSTX, but its potential for ELM pacing as well was demonstrated on EAST [246].

Solid PFC coatings that result from techniques like lithium evaporation are difficult to extrapolate to long-pulse applications. The availability of lithium only on the PFC surface limits the amount of deuterium it can bind, and any solid PFC is subject to long-term damage

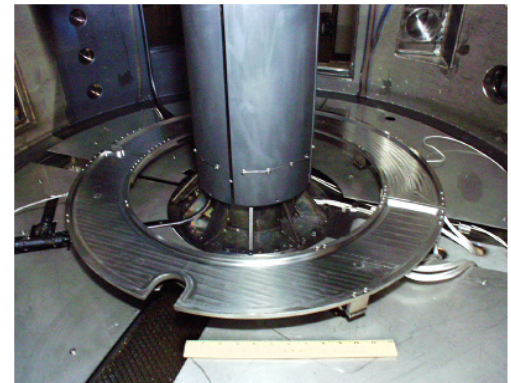


Fig. 7.12. Liquid lithium divertor tray inside CDX-U vacuum vessel. One-foot (30.48 cm) ruler in foreground provides scale. Heat shield with titanium carbide coating protects center stack. White cables emerging from tray provide power to heatersaaaaaaa.

in a fusion reactor environment [293]. The alternative is to investigate the feasibility of liquid metal PFCs, and the ST has features that make it a convenient test-bed. The first studies of large area, free-surface liquid lithium PFC were conducted on CDX-U (Fig. 7.12). Its vacuum chamber was a large cylinder, with a center stack occupying a relative small volume. Its low toroidal field (TF) compared to conventional tokamaks of similar plasma size (e. g., CDX-U has the same 22 cm minor radius as Alcator C-Mod) permits the TF coils to have a relatively small cross section. The large space between them provides good access to the vacuum chamber, which allowed a fully-toroidal liquid lithium limiter to be easily inserted and assembled within it. The result was a 2000 cm^2 free surface of liquid lithium that formed a limiter for CDX-U plasmas. When CDX-U PFCs were also evaporatively coated with lithium, the resulting reuction in recycling led to global energy confinement that exceeded scaling predictions for Ohmic ST plasmas by a factor of six [7, 294]. The next step beyond the toroidal liquid lithium limiter was to create a low-recycling lithium PFC that fully encloses the plasma. This has been achieved in the Lithium Tokamak Experiment (LTX), which was constructed by replacing the original lithium limiter with a conducting shell inside the CDX-U vacuum vessel. The shell consists of 1 cm-thick copper with an explosively-bonded 1.5 mm stainless steel liner as the PFC. The stainless steel keeps the copper from reacting with the lithium, and the copper maintains a uniform PFC temperature up to and above the point where the lithium liquefies [295, 296].

The same basic design is found in the Liquid Lithium Divertor (LLD) for NSTX (Fig. 7.13). The goal of the LLD was to extend the capability of the divertor to bind lithium beyond the amount that a solid surface would provide. Unlike the LTX shell, however, a 0.25 mm stainless steel liner was braised to 2.2 cm copper [297]. The heat loads to the NSTX divertor were considerably higher compared to LTX PFCs, so the thickness of the liner was greatly reduced to insure that the thermal response was determined by the copper. The stainless

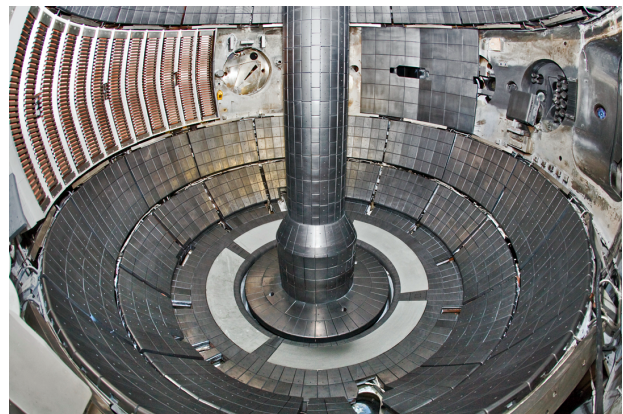


Fig. 7.13. Interior of NSTX vacuum vessel. LLD appears as light-colored sections surrounding the center stack. Darker areas are graphite tiles constitute bulk of PFCs. RF antennas, ports for diagnostics, and neutral beam armor are main features in NSTX midplane.

steel surface was plasma-sprayed with molybdenum (Mo) create a 0.17 mm-thick layer with 45% porosity. A plasma-sprayed Mo coating was also applied to a second LTX shell, to be installed after experiments with the original uncoated shell were completed. It was decided to use a Mo coating on the LLD from the start, as sputtering is more of a concern in the NSTX divertor. The porous Mo surface would also help restrain the lithium against potentially large electromagnetic forces that could expel the liquid lithium [298].

Experiments were performed in NSTX that included placing a strike point of lower-single-null plasmas on the LLD. The LLD was loaded with lithium by evaporation from the LITERs, and the ability of its surface to retain the lithium was demonstrated by the absence of any spectroscopic evidence for Mo in NSTX discharges [299]. Examination of the LLD surface after plasma operations did not reveal any damage, and this was consistent with test stand exposures of an LLD sample to heat loads of 1 to 2 MW/m² for up to three seconds with a diagnostic neutral beam [300]. The LLD performance also demonstrated the efficacy of Mo as a substrate for lithium coatings. Future plans for NSTX-U PFCs include the phased replacement of graphite tiles with Mo tiles, and studies where lithium-coated Mo samples are exposed to discharges that simulate NSTX-U divertor plasmas are in progress [301].

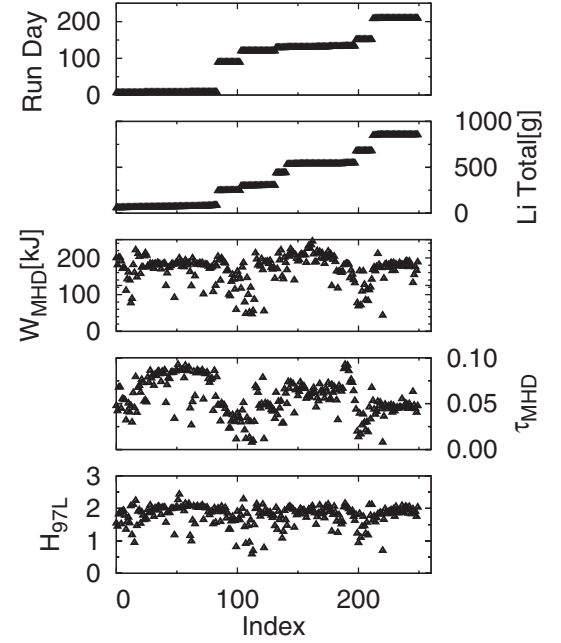


Fig. 7.14. Selected entries from the LLD experimental database. Run day is the number of days since the beginning of the 2010 experimental campaign, Li Total is the integrated amount of lithium evaporated into the vessel, W_{MHD} is the plasma stored energy, τ_{MHD} is the energy confinement time, and H_{97L} is the H-factor compared to the ITER 97L global confinement scaling. The values of W_{MHD} , τ_{MHD} , and H_{97L} are averages taken between

Any effects specifically attributable to the LLD on NSTX plasma performance were less clear. For example, the fueling required to achieve and maintain stable discharges was comparable to what was needed when graphite PFCs were coated with solid lithium [297, 302]. There is also no significant change in the confinement time over the run year over during which NSTX plasmas were exposed to the LLD, as shown in Fig. 7.14 [298]. The suspicion is that the LLD

surface is contaminated by compounds like lithium hydroxide that are formed when its static lithium coating interacts with the partial pressure of water in the NSTX vacuum vessel [303]. Lithium hydroxide, for example, liquefies at a temperature over twice as high as that for lithium. The contaminated LLD thus could have remained solid even if the lithium beneath it melted, and this was observed during test stand exposures of LLD samples [300]. The lithium PFC coating with the LITERs will continue on NSTX-U, and various additional lithium capabilities such as upward flash lithium evaporators and a lithium granule injector will be also implemented. For a longer term, the NSTX-U PFC will evolve toward increasing coverage with high-Z PFCs. A liquid lithium loop system is also being developed for NSTX-U.

A possible solution is to have a flowing liquid lithium divertor, and a special ST has been dedicated to its development. The Kazakhstan Tokamak for Material testing (KTM) has a vacuum vessel with a volume of 12.3 cubic meters, and it has a divertor consisting of mounted plates on a rotary table that can also be moved vertically. This provides KTM with the unique capability of replacing the plates without venting the vacuum vessel, and the compact divertor geometry the ST provides makes the concept feasible. The first lithium divertor module on KTM is a plate with a capillary porous surface, fed by liquid lithium feeding volume behind it [304]. The idea of individual test divertor modules prototypes the approach other devices will take in developing flowing liquid lithium divertor concepts.

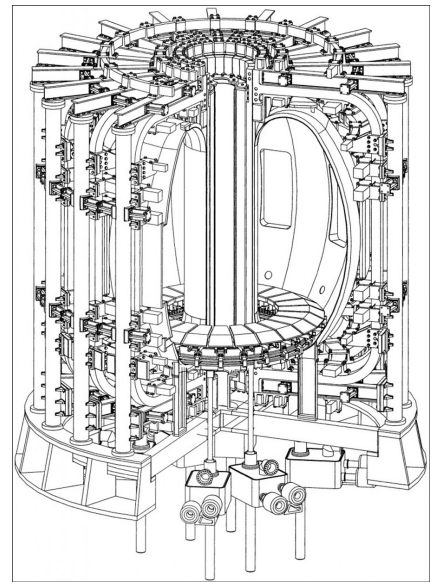


Fig. 7.15. Schematic of KTM. The divertor is made up of individual divertor plates that can be removed without venting the vacuum vessel.

VIII. Energetic Particles

A. Introduction and Motivation - The fusion alpha heating in reacting plasmas makes energetic particle (EP) physics an integral part of magnetic fusion research. Since fusion plasmas are in the regime where the 3.5 MeV alpha particle velocity usually exceeds the Alfvén velocity (V_A) (i.e., the super-Alfvénic regime) in magnetically confinement fusion reactors including ITER, the physics of the super-Alfvénic regime is particularly of interest [21, 22]. The EPs generated during MeV-class neutral beam injection (NBI) and ICRF heating and current drive can also excite Alfvénic modes, which can influence their heating and current drive efficiency. The Alfvénic mode excitation could become even more significant in ITER with auxiliary heating such as NBI and ICRF in addition to the alpha heating [305, 306]. Super-Alfvénic fusion alpha particles are particularly energetically favorable for exciting a number of Alfvénic modes, which tend to occur when fast ion velocity (V_{Fast}) is resonant with V_A , i.e., $V_{Fast} \sim V_A$. Those excited Alfvénic modes can cause "anomalous" particle slowing-down as well as radial transport. In some cases, the Alfvénic mode excitation can lead to a rapid loss of EPs directly to the plasma edge and the first wall. This EP transport and losses could reduce the effectiveness of alpha heating (which is a primary means of maintaining the burning plasma condition), but also could cause serious damage to the plasma facing components. The Alfvénic mode excitation could also cause deterioration of electron energy confinement, and thus the fusion plasma performance (as discussed in Sec. V). Utilizing high-energy NBI as the source of fast particles, the NSTX/MAST experiments can access a wide range of Alfvén Mach number $M_A \equiv V_{fast}/V_A$ and normalized fast ion pressure β_{fast}/β_{tot} , overlapping and extending beyond tokamaks and ITER as shown in Fig. 1.9. NSTX-U and MAST-U will encompass an even broader parameter space, and will more closely approach those expected for future devices such as ITER and future STs (Fig. 1.9.) The capability of spanning a much broader range of parameters for EP physics studies than conventional tokamaks provides an important opportunity for advancement of extrapolations to burning plasma regimes. In this section, we will focus on ST EP research which is unique or complementary to that on tokamaks. We will therefore focus on the EP modes discovered in STs, and the energetic ion transport/loss mechanisms which are driven by the excited EP modes. The EP research on STs began with the mode identification as described in Sec. B. This has expanded to the investigation of EP transport and resulting losses as described in Sec. C. More

recently, due to improved EP-related diagnostics and theoretical modeling tools, some successes in reproducing the observed EP transport and losses by the EP induced modes were achieved, and they will be discussed in Sec. D. The EP research is now entering the era of theory/modeling validation of simulation based on non-linear mode overlapping and the resulting EP transport, to improve the predictive capability needed for ITER and future STs.

B. Energetic-Particle-Driven Modes – The most commonly observed and investigated EP driven phenomena are Alfvén waves. The Alfvén wave phase velocity is approximately $V_A \equiv c \Omega_i / \omega_{pi} = B / (4\pi n_i m_i)^{1/2} \propto B / n_i^{1/2}$. The Alfvén wave excitation by EP tends to occur for $V_{Fast} \sim V_A$. The toroidal Alfvén eigenmode (TAE) is a naturally occurring Alfvén wave in tokamak geometry, where the effective wavelength is given by $\lambda_{||} \sim 4 \pi q R_0$ which is the nominal length of the lowest eigenmode in the flux tube. Therefore,

$\omega_{TAE} = k_{||} V_A = V_A / (2qR_0)$ is a nominal TAE frequency. The typical TAE frequency range is 50 – 200 kHz in MAST and NSTX. In STs, a variety of TAE-type modes have been easily excited, as shown in Fig. 8.1 [307]. There are TAEs which have stationary mode frequencies [Fig. 8.1 (a)] and downward (b) and upward (c) chirping modes.

Some modes are simultaneously upward and downward chirping, and are often called “angel fish” or “hole-clump” modes [308]. A theoretical analysis for the hole-clump

mode has been performed, and has been shown to be caused by a distortion (s) in the energetic ion velocity distribution [309]. There is a class of Alfvén eigenmodes (AEs) which exist near the minimum q (q-min) in reversed-shear (rs) discharges termed the rs-AE. In an evolving discharge, q-min tends to decrease in a current relaxation time scale so the rs-AE tends to rise in frequency and is also called the Alfvén cascading (AC) mode. Perhaps the most notable contribution of STs for understanding TAEs is in the effects of plasma beta values on TAEs.

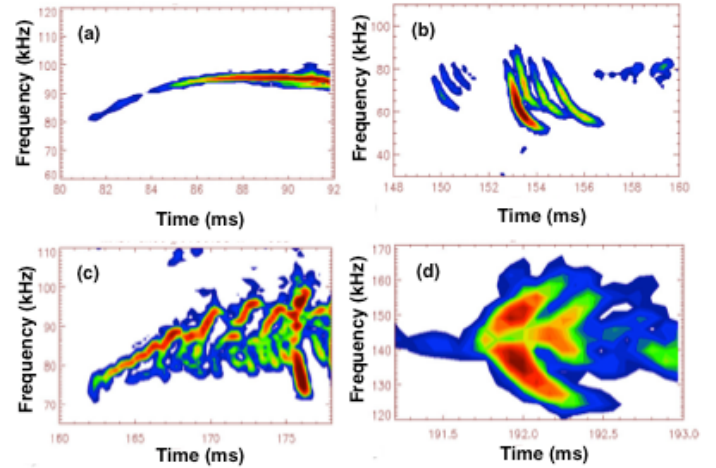


Figure 8.1. Examples of TAEs observed in NB heated discharges on MAST (a) long-lasting TAE mode with quasi-stationary frequency at the center of the TAE gap; (b) chirping-down modes; (c) chirping-up modes; (d) hole-clump mode with starting frequency at the TAE gap center.

The wave dispersion relation for the rs-AE or AC is essentially that of a shear Alfvén wave, including plasma pressure or acoustic term as given by

$$\omega_{AC}^2 = [k_{||}^2 V_A^2 + (1 + 7T_i/4T_e) 2Cs^2/R_0^2], \quad k_{||} = (m - nq_{min})/q_{min}R \quad (9)$$

where m and n are the poloidal and toroidal mode numbers, respectively [309]. In this equation, the coupling to the geodesic acoustic mode (GAM) or β -induced Alfvén eigenmode (BAE) is included. This is related to the fact that

as the plasma β approaches ~ 1 , V_A approaches C_s which facilitates coupling. The acoustic term can become dominant at high β , and when $m \sim nq_{min}$ making $k_{||} \sim 0$. It should be noted that the TAE frequency detected by a magnetic probe has a Doppler shifted frequency component for a rotating plasma which has to be taken into account. In MAST, a stabilizing trend in the TAE-type modes was observed with increasing beta, where the TAEs were stabilized for the beta above $\sim 15\%$ for the NBI beam energy of 45 kV [310]. Figure 8.2 presents statistics on the

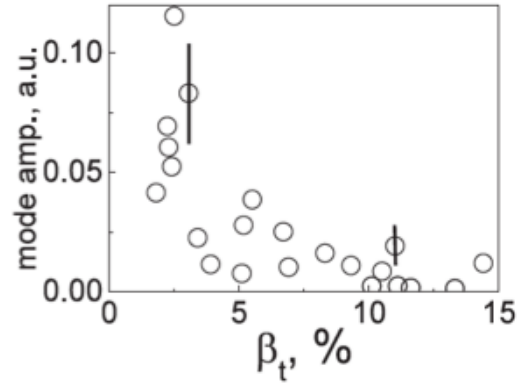


Fig. 8.2. Dependence on β of the maximum amplitude in a single burst of chirping modes, which start in the TAE-gap, in NBI discharges on MAST.

maximum amplitudes of chirping modes plotted as a function of β in several similar NBI discharges on MAST. A clear decrease in the amplitude with the β -value is seen, with the mode amplitudes close to zero at $\beta = 15\%$. Such a decrease in the chirping mode amplitude may be associated with a significant increase in the thermal ion Landau damping, as V_A slows down toward V_{Ti} with increasing β .

However, on NSTX with higher 90 kV energy beams, the Alfvén Mach number and therefore the drive can be considerably higher than in the MAST case. The higher beta regime in NSTX was accessed with higher density, which slows the Alfvén velocity and therefore increases the Alfvén Mach number further, resulting in higher drive. In NSTX, as the β is increased, the cascading behavior transitions to a pure TAE mode with flat frequency at the TAE gap frequency. This is shown in the early phase of the discharge in Fig. 8.3. As the β is increased further (in this case with time), the TAEs evolve toward bursting-type TAE modes. This is also shown in Fig. 8.3 where the plasma is just above the threshold beta of disappearance of the AC [311]. Here $\beta_e \sim$

7% (with a total $\beta \sim 20\%$) at $R = 1.2$ m. The flat mode frequency is an indication of the acoustic term becoming dominant in Eq. 9. The $n=3$ (green) and $n=4$ (blue) TAEs appear and disappear as, presumably, the $q_{\min}$ passes through and between low order rationals. In a higher $\beta > 20\%$ NSTX plasma, a new class of global MHD eigenmodes termed beta-induced Alfvén-acoustic eigenmodes (BAAEs) has been observed [312]. The modes with $n=1, 2$, and 3 arise in the gaps in the low frequency Alfvén-acoustic continuum below the GAM frequency. These modes also exhibit a bursting or avalanche behavior, with similar levels of neutron drops and EP loss as in the case of the TAE avalanche. This will be discussed more in the next section related to the EP loss. Another low frequency EP mode is the so-called fishbone instability which is an $n=1$ precessional drift resonant mode discovered in PDX [313], and since then observed in virtually all NBI driven tokamaks. Fishbones are also seen in NSTX and MAST, but they have not been studied as a high priority since their

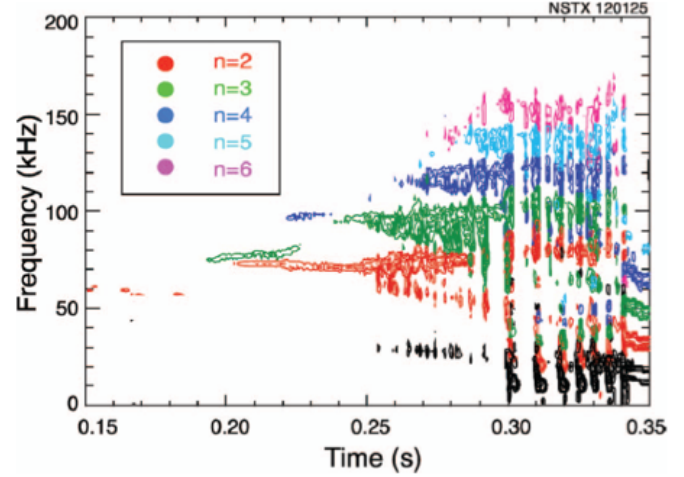


Fig. 8.3. Color Spectrogram of Mirnov coil from discharge at the threshold beta below which Alfvén cascades are seen. Frequency sweeping is largely absent.

physics is relatively well understood. Some fishbones observed in NSTX have higher n numbers (i.e., $n = 2, 3$) and they are conjectured to be driven by trapped-ion resonances [314].

Two new types of high frequency EP modes first discovered in STs are Compressional Alfvén Eigenmodes (CAEs) and Global Alfvénic Eigenmodes (GAEs), as shown in Fig. 8.4 [315]. As can be seen from the figure, the CAEs [316 - 318] and GAEs [319] exist at higher frequency ranges, well above the TAEs, rsAEs, and other modes thus far discussed. The CAEs and GAEs appear in a broad spectrum of nearly equally spaced peaks in the frequency range from $\sim 0.2 \Omega_i$ to $\sim 1.2 \Omega_i$ (or 500 kHz to a few MHz) in NSTX [316]. The CAEs are compressional Alfvén waves which can exist above and below the ion cyclotron frequency, and GAEs represent the shear Alfvén wave roots that exist below the ion cyclotron frequency ($f_{ci} \sim 3$ MHz for this case). The CAE dispersion relation is $\omega_{CAE} = k V_A$, where k is mainly defined by the radial wave number. The GAEs are well described by Eq. 9, and can occur at high frequency compared to

TAEs because of negative higher n values (e.g., $n = -4$ and $m = 2$). The beam-induced GAEs are believed to be responsible for the possible rapid electron energy loss in the core region of the NSTX NBI heated plasma as discussed in Sec. V.C [165]. As can be seen from Fig. 8.4, a variety of EP modes can be excited simultaneously and their behavior is highly time-varying, particularly during the initial current ramp-up phase where the plasma parameters are rapidly changing. In addition to the magnetic pick-up loops, there are various techniques such as microwave reflectometry [320] and beam emission spectroscopy (BES) [173, 174] that can be used to measure the EP

mode amplitudes. The observed EP modes were compared with theoretical models and show generally good agreement with linear theory. A quantitative comparison of the measured TAE mode structure with theory has been successfully made with the NOVA-K Toroidal Alfvén Eigenmode code [321] for STs [315, 322] and tokamaks [21-22].

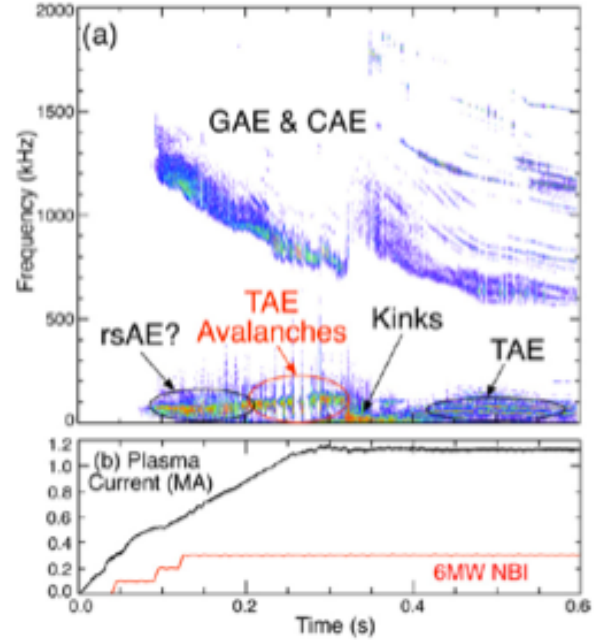


Fig. 8.4. (a) Spectrogram of EP induced magnetic fluctuations in NSTX, (b) plasma current and neutral beam power evolution.

C. Energetic particle transport/losses - EP transport and losses have been investigated actively on STs and tokamaks due to their importance for ITER and future burning plasma devices. Here we shall focus on plasma mode-driven transport but not include classical and neoclassical processes (such as ripple transport), since they were actively investigated in tokamaks and relatively well understood [21, 22]. For STs, because of the strong EP drive as shown in Fig. 1.9 (i.e., super-Alfvénic conditions and high normalized fast ion β), significant EP transport processes by various EP modes have been observed. Utilizing EP and current profile diagnostics, significant EP radial transport and prompt losses have been measured on NSTX. The prompt losses were observed particularly when multiple over-lapping modes or "avalanches" were simultaneously excited. The TAE (Toroidal Alfvén Eigenmode) avalanche is the most commonly observed avalanche process, involving a simultaneous excitation of multiple TAE

modes in NSTX [315, 323]. In Fig. 8.5 (a), a TAE avalanche is shown where the magnetic pickup coil frequency spectrogram with toroidal mode numbers 1 through 4 appear together in an avalanche on an expanded time scale [323]. A total neutron rate drop of $\sim 13\%$ is shown in (b). Since there is no significant change in the bulk plasma parameters, the neutron drop is interpreted as an outward transport of fast ions due to the avalanche. The total detected fast particle loss flux by a scintillator fast ion loss probe (s-FLIP) [324], located at the outer mid-plane edge, is shown as a function of time in Fig. 8.5 (c) [323]. As can be seen, the fast ion loss comes in a rapid short burst coinciding with the avalanche. It should be noted that the neutron drop is usually observed during a TAE avalanche but the fast ion loss is not always observed, indicating that fast ion transport during an avalanche does not necessarily lead to fast ion loss. In Fig. 8.6, the brightest of the s-FLIP frames with a superimposed interpretation grid for the gyro-radius and pitch angle of the lost particles is shown. The relatively wide bright spot size indicates that the loss is relatively broad in pitch angle but within a narrow range of gyro-radius (~ 17 cm) which corresponds to ions near the injected NBI energy of 90 keV. This observation therefore shows that the TAE avalanche can produce highly focused lost ions at maximum energy in a very narrow time frame. This very rapid and focused

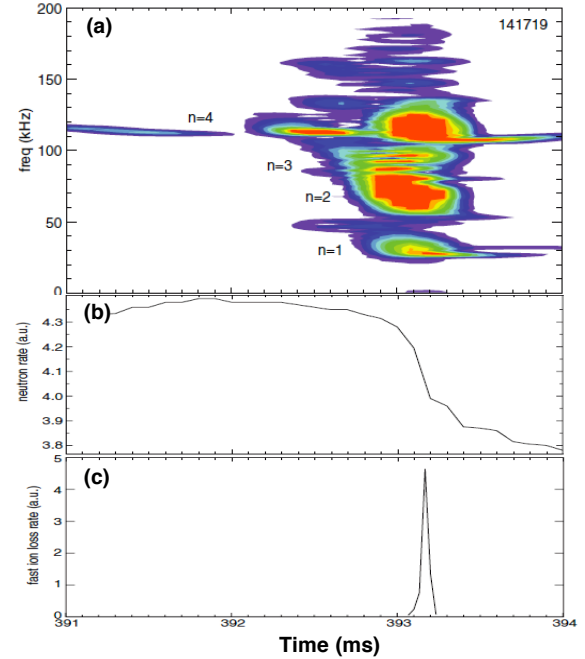


Fig. 8.5. (a) Magnetic pickup coil frequency spectrogram during a TAE avalanche on an expanded time scale, (b) neutron rate, and (c) total fast ion loss rate measured by the sFLIP

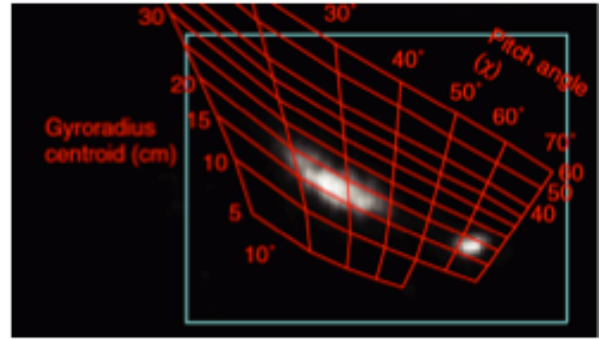


Fig. 8.6. sFLIP detector camera frame at time of peak loss with interpretive grid defining the gyro-radius centroid and pitch angle of the lost particles. Based upon the EFIT magnetic field at the detector location, 90 keV D ions should have a gyro-radius of 17 cm.

fast ion loss could cause damage to PFCs in ITER and future reactors. Another fast ion diagnostic, the Fast Ion D- α (FIDA) system [327] also measured a level of core fast ion density decrease similar to that deduced from the neutron rate drop as shown in Fig. 8.7 [328]. The fast ion transport due to the TAE avalanche and its effect on the NBI driven current has also been investigated [329]. In this study, an upper bound on the fast-ion diffusivity of $\sim 0.5 - 1 \text{ m}^2/\text{s}^{-1}$ is found. This level of fast ion diffusivity can significantly broaden the NBI driven

current profile, which can be beneficial from the plasma stability point of view but can also lead to a loss of current drive efficiency. Another type of avalanche process was found with lower frequency modes consistent with the BAAE avalanche with $n = 1, 2$, and 3 [328]. The total neutron rate drop is $\sim 13\%$ as with the TAE avalanche, and the rapid fast ion particle loss with a wide pitch angle range but relatively close to the injected energy of 90 keV is again quite similar to the TAE avalanche. Another avalanche process is observed for the higher frequency GAE, where the GAE avalanche of $n = -7$ to -10 can trigger the TAE avalanche of $n=1$ through 6 [329]. While not an EP mode, low frequency ($f = 5 - 8 \text{ kHz}$) continuous $n=1$ MHD modes can also transport EPs and significantly modify the NBI current drive profiles [330]. One modeling calculation shows that the fast ion diffusivity of $20 \text{ m}^2/\text{s}^{-1}$ within the region of mode activity of $r/a < 0.45$ is required to match the measured neutron drop. It should be also noted that an application of $n = 3$ fields has altered the dynamics of high-frequency bursting energetic-ion-driven Alfvén modes [331]. Calculations indicate that the 3D perturbation affects the orbits of fast ions that resonate with the bursting modes. The fast ion transport effects on the NBI current drive will be discussed more in Section X.

D. Theory / modeling validation – It is clearly vital to develop a predictive capability for EP mode excitation, particularly the avalanches and their consequences in terms of EP transport and

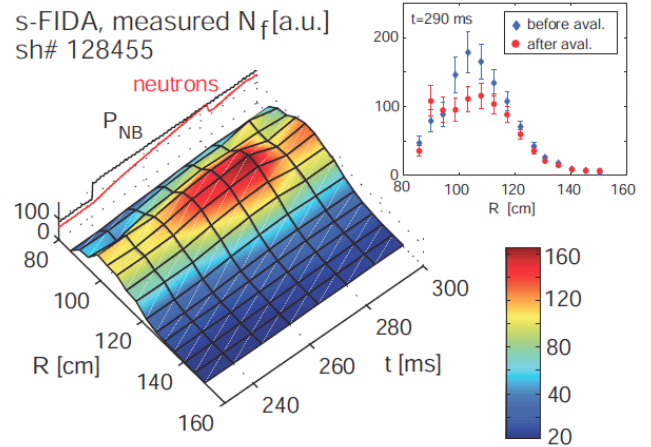


Fig. 8.7. Color Measured evolution of the fast-ion profile in NSTX by FIDA. The decrease after $t=280 \text{ ms}$, following a TAE avalanche, is detailed in the inset.

losses. A major advance in recent years is in improved diagnostics (as described above) and simulation/modeling capabilities for explaining their observations [315, 322, 323]. The simulations usually involve multiple steps. First, the mode amplitude profiles are needed. Once good mode identification is made, the NOVA-K code is used to provide the corresponding TAE structures. The NOVA-K generated eigenmodes are then scaled to an overall amplitude that produced the best fit to the displacements measured, for example, by the microwave reflectometer system. This is done for the entire plasma volume, since the measured profiles are available at only specific and limited locations. Then the selected eigenmodes are used in the ORBIT code [332] to simulate the fast-ion loss. In Fig. 8.8, the calculated net neutron rate drop in ORBIT simulations (red squares), the neutron drop due to lost beam ions (blue circles), and the neutron drop in the confined beam ion population due to fast ion redistribution and loss of energy to the TAE are shown [315]. It should be noted that the neutron drop from redistribution and energy loss occurs well before the fast ions are lost from the plasma. There is an apparent threshold for energy loss in the fast ion population at a normalized mode amplitude of ≈ 0.4 , and the threshold for fast ion loss onset occurs at a normalized mode amplitude of ≈ 1 . Therefore it appears that there is good agreement between the measured neutron rate drop and that predicted at the measured mode amplitude. With the simplifying assumptions used in the modeling, it is surprising that the agreement is as good as it is. The use of the ideal eigenmodes, with unphysical interactions with the continuum, the use of the unperturbed fast ion distributions in the presence of multiple Alfvénic instabilities, the use of a guiding center code in a situation with large Larmor radii, together with the general uncertainties in equilibrium reconstruction, could all potentially contribute to large uncertainties in the simulations. The NOVA-K is a linear mode modeling code, so a non-linear model is needed to fully describe the highly non-linear nature of the TAE avalanches. In the experimental area, continued improvement in measurements of the

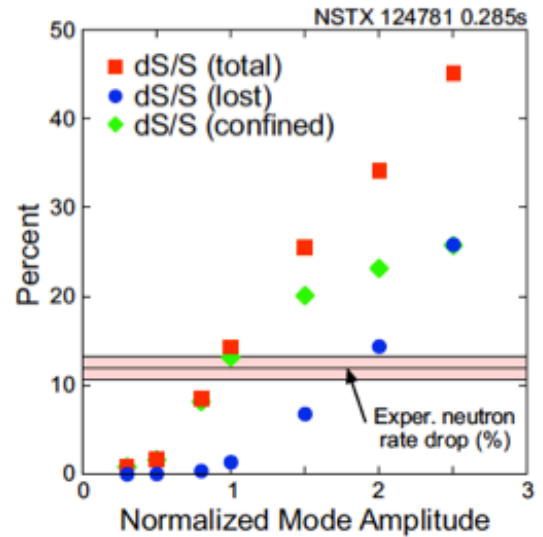


Fig. 8.8. Simulated neutron rate drop due to TAE avalanche (red), neutron rate drop resulting from lost beam ions (blue) and neutron rate drop in confined beam ion population from energy loss (green).

EP population and excited modes are needed. Future improvements planned for NSTX-U and MAST-U thus include upgrades to various diagnostics such as microwave reflectometers, BES, FIDA, s-FLIP, and ss-NPAs.

IX. Plasma waves for heating and current drive in over dense ST plasmas

A. Introduction and Motivation - Efficient heating and current drive are essential for the success of the ST/tokamak fusion reactor concept. In present day ST experiments, neutral beam injection (NBI) is a reliable and established method for heating and current drive. At the same time, it is highly desirable to develop radio-frequency (RF) wave based heating and current drive since the NBI-based approach tends to be technically challenging for reactor applications in terms of beam penetration and compatibility with the fusion reactor neutron environment. For developing attractive RF heating and current drive methods, ST plasmas present a special challenge due to being in the so-called "over-dense" regime [i.e., $(\omega_{pe}/\omega_{ce})^2 \gg 1$ where ω_{pe} and ω_{ce} are the electron plasma and electron cyclotron frequencies, respectively.] For a typical tokamak plasma, $(\omega_{pe}/\omega_{ce})^2 \sim 1 - 2$ in the plasma core while for STs, it can be $(\omega_{pe}/\omega_{ce})^2 \sim 10 - 20$ due to a factor of $\sim 3 - 4$ lower magnetic field. The over-dense condition makes some of the conventional radio-frequency wave heating and current drive concepts such as Electron Cyclotron wave Heating (ECH) and Lower Hybrid wave Current Drive (LHCD) not practical due to the severe wave accessibility limits imposed by the over-dense condition [333]. It should be noted that they can still be viable as a plasma start-up technique due to much lower start-up densities as discussed in Sec. IV. For this reason, RF heating and current drive research in STs has thus far focused on the High Harmonic Fast Wave (HHFW) and Electron Bernstein Wave (EBW) approaches as described in Sec. B and C.

B. High harmonic fast wave for electron heating and current drive - The High Harmonic Fast Wave (HHFW) heating and current drive was proposed for its promise to heat electrons in high beta and over-dense conditions for STs [334].

The HHFW is in the compressional Alfvén wave branch (i.e., same branch as the conventional ion cyclotron range of frequency or ICRF heating in tokamaks) but at higher ion cyclotron harmonic frequencies (~ 10 times greater). The wave perpendicular phase velocity is approximately the Alfvén speed or $\omega/k_{\perp} \sim V_A \equiv c \Omega_i / \omega_{pi} =$

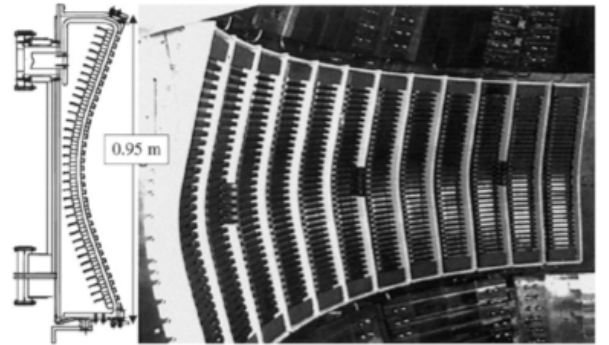


Fig. 9.1. NSTX HHFW antenna array.

$B/(4\pi n_i m_i)^{1/2} \propto B/n_i^{1/2}$. It can be also shown that the Alfvén speed slows down as the ion plasma beta is increased, approaching $\omega/k_{\perp} \sim V_A \sim V_{Ti}$ at the unity ion beta. The HHFW starts to take on the whistler-like characteristics, and electron damping via magnetic pumping as well as the usual electron Landau damping becomes significant [314]. For NSTX plasma parameters, the electron damping was predicted to be strong enough to attain single-pass absorption at high β , with some possibility for localized off-axis heating and current drive.

Initial HHFW physics experiments were performed in the Current Drive Experiment- Upgrade (CDX-U) with a rotatable dual strap antenna system, establishing basic HHFW wave excitation as a function of the antenna tilt angle with respect to the magnetic field line pitch, and determining the wave propagation characteristics and associated electron heating [335]. The HHFW physics was also investigated in TST-2 initially using a six-element combline antenna at lower power (1 kW) where high efficient wave lanching qualitatively consistent with modeling was observed [10]. A two-strap antenna was later employed on TST-2 at higher injected power of ~ 240 kW where parametric decay instabilities (PDI) were investigated [336]. The HHFW high power heating experiment on NSTX has been conducted utilizing a 12 strap antenna system powered by six 30 MHz high power rf transmitters (~ 1 MW each) with real time phase control

[337]. The NSTX 12 strap antenna array is shown in Fig. 9.1. Efficient electron heating by HHFW has been confirmed on NSTX [338, 339]. As we can see in Fig. 9.2, application of ~ 3 MW of HHFW power resulted in the production of plasmas from an initial Ohmic temperature of a few hundred eV to a central electron temperature $T_{e0} \sim 6.2$ keV in helium and $T_{e0} \sim 5.2$ keV in deuterium at central electron density of $\sim 1.5 \times 10^{13} \text{ cm}^{-3}$ [338]. These temperatures are particularly impressive for NSTX, which operated at a maximum axial toroidal field of only 0.55 T. The electron temperature is typically very peaked during RF heating but in deuterium, it is broader with a steeper T_e gradient, giving the appearance of internal transport barrier (ITB) formation. While we shall use the word “ITB” for the steep T_e gradient region, it is not fully

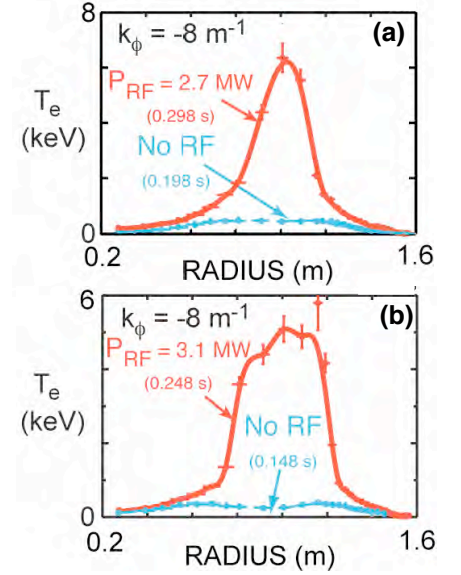


Fig. 9.2. (a) $T_e(R)$ immediately prior to HHFW heating $t = 0.198$ s (dashed line) and during 2.7 MW of $k_\phi = -8 \text{ m}^{-1}$ heating $t = 0.298$ s (solid line) of a helium plasma. (b) $T_e(R)$ immediately prior to rf heating $t = 0.148$ s (dashed line) and during 3.1 MW of $k_\phi = -8 \text{ m}^{-1}$ heating $t = 0.248$ s (solid line) of a deuterium plasma.

resolved at the present time that the appearance of the ITB-like steep temperature gradient is due to the formation of a transport barrier or the improved core heating efficiency due to the higher electron temperature. It is instructive to look at the time evolution of HHFW as shown in Fig. 9.3 [339]. An overlay of the time evolution of T_{e0} and the central ion temperature T_{i0} from the x-ray crystal spectrometer [340] is shown. We see that T_{e0} gradually increases to a peak of ~ 4 keV over ~ 200 ms, which is much longer than the electron energy confinement time of ~ 30 ms; T_{i0} also increases, with some delay, to 2 keV before dropping to near 1 keV. An MHD reconnection event occurs at 0.22 s. As shown in the figure, the central temperature rise is very gradual considering the constant applied RF power. Assuming constant heating efficiency, TRANSP analysis for this plasma shows that the core electron heat diffusivity goes down by a factor of 10. The TRANSP calculations also suggest that the q profile is weakly reversed, as a result of the bootstrap current generated in the core region which could be causing the core electron confinement improvement. It could also be noted that the HHFW heating efficiency may be improving in time as the electron temperature increases. The increasing HHFW core heating efficiency might be the result of improving core HHFW absorption, which further increases the core temperature.

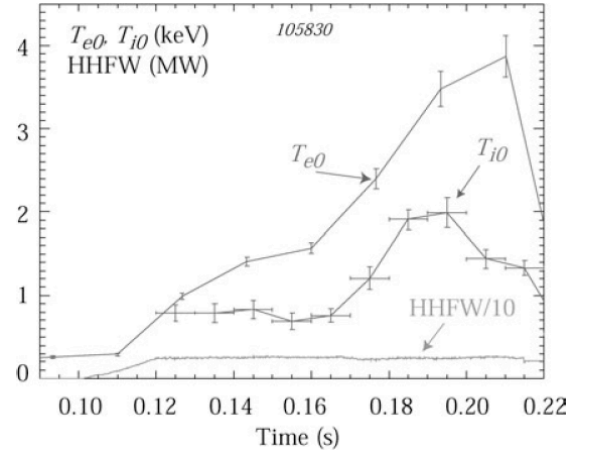


Fig. 9.3. Central temperatures T_{i0} and T_{e0} as a function of time.

In initial HHFW current drive experiments with the antennas phased to launch a toroidally directed spectrum, differences in loop voltage have been observed that are consistent with plasma current being driven by the RF wave. The magnitude of current driven, inferred from the magnetic measurements, is $\sim 50 - 100$ kA and roughly consistent with theoretical estimates [341 - 344]. One of the motivations for HHFW current drive is current ramp-up from a lower start-up plasma. While the maximum arc-free HHFW P_{RF} was limited to ~ 1.4 MW for this experiment, it was nevertheless possible to generate a plasma current $I_p = 300$ kA in a $B_{T0} = 0.55$ T, deuterium H-mode plasma. The loop voltage was close to zero as shown in Fig. 9.4, and there was a calculated $f_{NI} = 0.7 - 1$ [344]. The H-mode starts around $t = 0.22$ s, and an “ITB” at $r/a \sim 0.4$ is formed at $t = 0.38$ s. This positive result was made possible by better outer gap control during the H-mode phase, so that the H-mode was sustained for ~ 250 ms. Assuming full HHFW efficiency, 100 – 120 kA of FWCD is calculated to be generated during the flat top of the RF

heating pulse. The calculated bootstrap current oscillates between 100 and 230 kA. These large fluctuations in bootstrap current result from plasma pressure profile changes near the end of the RF power ramp-up (at 0.26 s), at the formation of the ITB (at 0.3 s), and inside the ITB (at 0.38 s) as indicated in Fig. 9.4. Three-quarters of the non-inductive current was estimated to be generated inside the ITB. This suggests that the HHFW is a promising tool for current ramp-up. A strong modeling effort utilizing advanced computational tools is also being pursued [345]. It should be noted that the HHFW core electron heating capability was used as a means of generating highly peaked electron temperature profiles to study the role of electron temperature gradient (ETG) modes on electron transport (Sec. V), and the H-mode power threshold with low toroidal plasma rotation (Sec. VI).

The HHFW acceleration of fast ions was also studied since HHFW goes through a number of ion cyclotron harmonic resonances as it propagates toward the plasma core. With NSTX parameters, the HHFW absorption by the bulk ions is estimated to be negligible, but there are some ion cyclotron harmonic interactions expected with the NBI fast ions [334]. In NSTX, there appears to be no direct indication of bulk ion absorption; however, there are some indication of NBI fast ion acceleration by HHFW and an associated neutron increase [346]. For NSTX-U, since the toroidal magnetic field would double while the HHFW frequency is fixed at 30 MHz, the ion cyclotron harmonics will be halved to ~ 5 . A theoretical model calculation for the NSTX-U parameters indicates that some bulk thermal ion cyclotron harmonic heating may be possible for the first time [345].

While HHFW can be effective in core electron heating, there has been significant variability in the HHFW heating efficiency in the NSTX experiments. Helium plasmas usually provided more

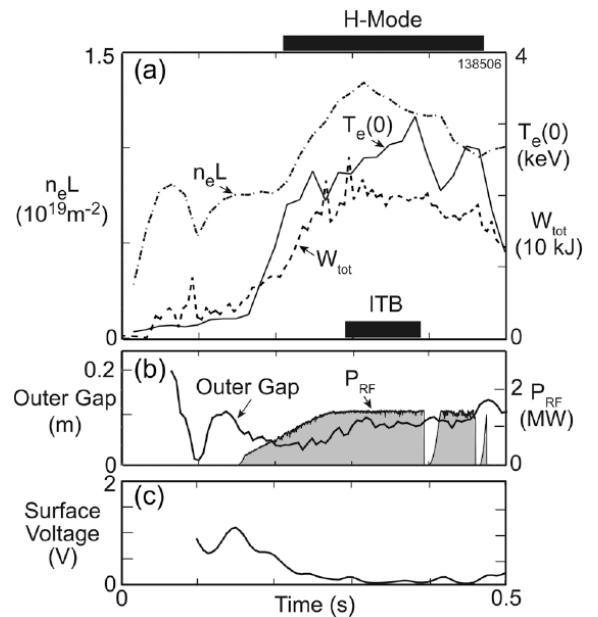


Fig. 9.4. Time evolution of an $I_p = 300$ kA HHFW-generated H-mode plasma. (a) Line integrated density ($n_e L$), central electron temperature ($T_e(0)$) and total plasma stored energy (W_{tot}). (b) Outer gap between the last closed flux surface and the front of the HHFW antenna on the mid-plane and RF power. (c) The measured loop voltage.

reliable HHFW heating compared to deuterium plasmas. The heating efficiency improved strongly with the toroidal magnetic field, nearly doubling when the magnetic field increased from $B_{TF} = 4.5$ kG to 5.5 kG. Lithium coatings of plasma facing components (PFCs), which reduced the edge density and collisionality, also helped to improve the HHFW heating efficiency, particularly in deuterium plasmas. The heating efficiency dropped off sharply with higher wave phase velocity. To understand the sources of HHFW inefficiency, various possible mechanisms were investigated. One of the most significant effects affecting the HHFW heating efficiency was found to be the edge density profile [347]. The heating efficiency was shown to depend strongly on the “onset” density at which the HHFW starts to propagate as shown in Fig.

9.5. A sequence of RF pulses was applied with $k_{\parallel} = 14 \text{ m}^{-1}$ and -8 m^{-1} , respectively. As can be seen in Fig. 9.5(a), the first pulse has poor heating efficiency compared to the subsequent pulses, particularly for the longer wavelength case where $k_{\parallel} = -8 \text{ m}^{-1}$. One important variable was the edge density as shown in Fig. 9.5(b) where the density must be lower than the so-called “onset” density. The onset density is defined as the density at which the HHFW waves start to propagate, i.e.,

$$n_e (10^9 \text{ cm}^{-3}) = (5.0Z/\mu)(n+1)B_T^2(n_{\parallel}^2 - 1) \quad (10)$$

where Z is the ion charge, μ is the ion mass, n is the ion cyclotron harmonic number, $n_{\parallel} = k_{\parallel} c / \omega$ is the launched wave number, and B_T is the local magnetic field. If the edge density is above the onset density and the HHFW wave is propagating, a significant loss of RF power could occur before reaching the main plasma through various wave dissipation processes, including collisional damping as well as by means of parametric decay instabilities. Since the onset density is lower for longer wavelength, the onset condition becomes more challenging to avoid for lower $n_{\parallel}$. This explains

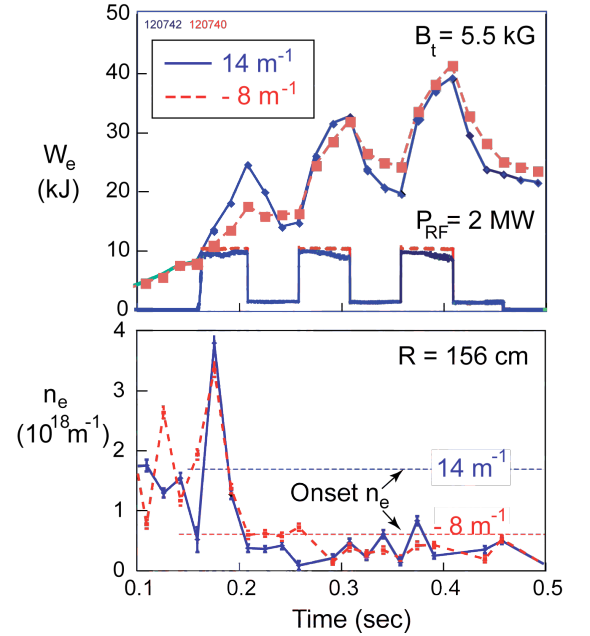


Fig. 9.5. (a) Electron stored energy evolution with modulated HHFW power of 2 MW with $k_{\parallel} = 14 \text{ m}^{-1}$ and -8 m^{-1} , respectively. (b) Edge electron density (2 cm in front of Faraday shield) vs. time. The onset density for perpendicular propagation is indicated by the horizontal dashed lines for $k_{\parallel} = 14 \text{ m}^{-1}$ and -8 m^{-1} as marked.

why the HHFW heating efficiency tends to be lower for lower $n_{||}$, and improves with lithium application to PFCs, which reduces the edge plasma density. It also explains why a higher toroidal field and $n_{||}$ tend to lead to better HHFW heating efficiency, since the onset condition is easier to avoid. The HHFW edge power loss mechanisms have been investigated in NSTX. A significant fraction of the HHFW power can be lost to the scrape-off layer (SOL) and deposited in bright spirals on the divertor [348]. While the underlying mechanism has not yet been identified, this process maybe correlated with the onset density observed above. The edge RF power deposition decreases substantially when the onset density is pushed a few centimeters away from the antenna, and this correlates with improved core heating efficiency. This result indicates that edge density control is quite important for efficient HHFW heating and current drive, and this edge interaction maybe also relevant for ITER ICRF [349]. The parasitic effects are likely to be reduced with higher magnetic field for NSTX-U [345]. Another possible mechanism investigated is the PDI, i.e., a three wave coupling process where the launched HHFW wave, for example, can decay into an ion Bernstein wave (IBW) and an ion quasi-mode (IQM) with matching wave frequencies and wave numbers [350]. This can often lead to inefficient RF heating, since the wave energy can be channeled to other waves which can damp heavily at the plasma edge. In TST-2, the PDI processes were investigated using probes and reflectometer, where the decay waves were found to be consistent with HHFW or IBW for the lower side band and IQM for the lower frequency mode [336]. On NSTX, such an instability was indeed observed to cause strong ion heating at the plasma edge [351]. This parametric decay appears to occur under all conditions when HHFW RF power is applied and it is not directly correlated with the RF antenna phasing $n_{||}$ and magnetic field, both of which do strongly affect the HHFW heating efficiency.

C. Electron Bernstein waves for localized current drive

In ST or tokamak reactors, non-inductive plasma current profile control is an important research topic, as the achievable beta limit is sensitively dependent on the plasma current profile. For this reason, a technique for driving the plasma current at a desired location (often in the mid-to-outer region $r/a > 0.5$) even at a few % level is highly desirable. The conventional Fisch-Boozer current drive becomes inefficient for $r/a > 0.5$ due to trapped particles. On the other hand, the

so-called Ohkawa current can be driven efficiently, even in the outer region, by moving electrons from barely passing into trapped orbits via electron cyclotron harmonic heating. Indeed, numerical modeling of electron Bernstein wave current drive (EBWCD) for a $\beta = 40\%$ ST plasma predicts efficient, off-axis Ohkawa EBWCD. The calculated normalized current drive efficiency increases with r/a , and is a factor of 2 higher at $r/a = 0.7$ than has been obtained with electron cyclotron current drive near the axis of large aspect ratio tokamaks [352].

The Electron Bernstein Wave (EBW) is an electrostatic short wavelength wave, which exists due to finite electron gyro radius (i.e., finite electron temperature) effects. It can propagate in the over-dense conditions of STs, and due to its electrostatic nature, can be strongly absorbed locally near the electron cyclotron layer and the locations of its harmonics. It is therefore well suited to heat and drive plasma currents when good localization is desired, for example, for the current profile control needed for high performance ST operations. This conclusion is based on EBW modeling calculations for localized current drive [353]. However, since the EBW is a finite electron temperature plasma mode, it must be coupled from an external launcher which is typically designed for conventional electromagnetic electron cyclotron extraordinary X mode and/or ordinary O mode launching. The X mode can couple to EBWs in the vicinity of the upper hybrid resonance (UHR) (i.e., the B-X process), and the O mode can couple first to the slow X mode, which subsequently mode converts to EBWs near the UHR (i.e., B-X-O process). An informative review of EBW for heating and plasma diagnostic for fusion devices can be found in Ref. 354.

EBW Emission Experiments – There have been two classes of EBW experiments: one focuses on the natural EBW emissions (EBWE) coming out of the hot plasma interior and the other actually injecting EBW power from a launcher. In the hot over dense electron core of ST plasmas, the EBWs are naturally excited by

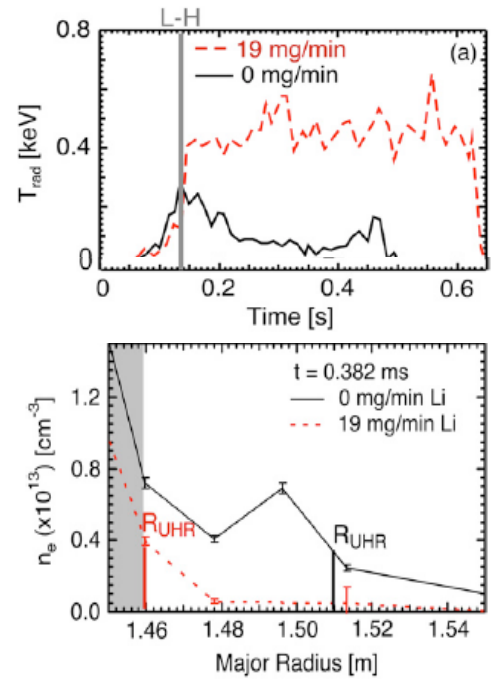


Fig. 9.6 (a) The EBW coupling efficiency in H-mode with and without lithium in the scrape-off-layer. (b) The Thomson scattering n_e profiles with and without lithium in the scrape-off-layer for H-mode.

the optically thick plasmas near the electron cyclotron harmonic layers, and this easily satisfies the EBW blackbody condition. This makes the EBW an attractive electron temperature diagnostic technique for STs [353]. The EBW mode conversion analysis predicts that the EBW mode conversion efficiency is reversible (i.e., the mode conversion efficiency is the same for EBW to X/O modes and X/O modes to EBW) [355]. Therefore, by measuring the EBW emission or EBWE efficiency (i.e., the ratio of the measured EBWE T_e with another independent T_e measurement), one can obtain the mode conversion efficiency for EBWE as well as the EBW coupling from an external launcher via the B-X or B-X-O process. The first B-X EBWE was measured on CDX-U by comparing the EBWE T_e with Thomson scattering T_e measurements, where near 100% conversion efficiency occurred with ~ 50 eV electron temperature ST plasmas [356]. Since efficient B-X mode conversion requires a rather steep density gradient near the mode conversion layer (i.e., around the UHR), a movable limiter was used to control the density gradient for optimum coupling. The B-X process, which requires a very steep density gradient at the mode conversion layer for optimum coupling, may not be practical for larger ST plasmas. On NSTX, EBWE with a B-X-O process was pursued. The first series of EBWE experiment on NSTX showed an efficient EBW coupling ($\sim 0.8 \pm 0.2$) in L-mode plasmas, in good agreement with the numerical EBW modeling prediction of 0.65 [357]. However, the coupling was rather poor ($<10\%$) in the initial EBW coupling experiments into H-modes. Simulations show that indeed 80% of the EBW energy could be dissipated by collisions in the edge plasma. However, application of lithium as a PFC coating greatly improved the EBWE coupling to $\sim 60 - 70\%$ even in H-mode plasmas, as shown in Fig. 9.6 (a) [358]. The lithium PFC coating reduced the scrape-off-layer density as shown in Fig. 9.6 (b). This moved the conversion layer into the higher temperature pedestal region, which in turn reduced the collisional damping of EBW. This improved coupling was explained largely due to the reduced collisional damping from 70% to only 15% after lithium PFC conditioning. As in the case of HHFW, edge conditioning has proved to be highly important for efficient EBW coupling. Another application of the EBWE is innovative measurements of plasma parameters such as magnetic field pitch and plasma rotation at the plasma edge being developed on MAST [359, 360].

EBW Heating and Current Drive Experiments – The application of EBW for heating and current drive has been previously demonstrated on tokamaks and stellarators. In Sec. IV-B, we

discussed the role of EBW in ST start-up experiments, where EBW played a crucial role in continuing to provide heating and current drive power even as the plasma entered the over dense regime, where the normal ECH process ceases to be effective. For EBW heating and current drive of the main high density ST plasmas, the experimental data are presently limited to the TST-2 and MAST device. In TST-2, the EBW heating via X-B MC process (a reverse of the CDX-U B-X MC EBWE experiment) using a 200 kW 8.2 GHz source available at Kyushu University [361]. With sufficient density gradient, the heating efficiency of up to 50% was observed. In MAST, the applied EBW power of $P_{rf} \sim 250$ kW at 60 GHz was injected into the plasma core via O-X-B mode conversion, resulting in an energy rise of ~ 7 kJ or about 10% of the plasma stored energy increase [362]. Because of the relatively high RF frequency, the heating layer for this case is $5-6 \Omega_e$ which may be too high for efficient heating. Another scenario, achieved by compressing a high-density ohmic H-mode plasma in major radius, resulted in a relatively low harmonic of $2-3 \Omega_e$ becoming accessible and led to a core electron temperature rise of ~ 100 eV. The estimated mode-conversion efficiency was relatively high ($\geq 50\%$) in the experiment. Both NSTX-U and MAST-U plan to implement MW-class ECH/EBW heating and current drive systems for start-up and further EBW heating and CD research.

X. Integrated Scenarios

A. Motivation - In order to establish a sufficient database for designing future STs facilities, it is essential to perform experiments in present ST facilities to test integrated scenarios that can be scaled to match the performance of future ST devices [363, 364]. The key goal of integrated scenario development is to test properly-scaled device and plasma parameters that match as many key dimensionless parameters as possible for future STs under conditions required for steady-state non-inductive operations. The main key dimensionless device and plasma parameters include A (aspect-ratio), κ (elongation), δ (triangularity), ρ^* (rho-star or gyro-radius normalized by plasma minor radius), ν^* (plasma collisionality), q^* (cylindrical safety factor), q_{95} (edge safety factor), β_N (normalized beta), β_T (toroidal beta), β_P (poloidal beta), f_{BS} (bootstrap current fraction), HH (H-mode confinement enhancement factor), and N_G (Greenwald density fraction). The target device and plasma dimensionless design parameters for various future devices are shown in Table III in Sec. II.

One of the key device/plasma control parameters is the plasma elongation κ , since it is an externally-controllable parameter which potentially has a major impact on the plasma performance, particularly for non-inductive operation as discussed in Sec. II and III. The plasma elongation achieved in NSTX and target range for future devices are illustrated in Fig. 10.1 [327]. As can be seen in the figure, NSTX was able to scan a relatively wide range of A . In the inset, the $\kappa = 3$ plasma equilibrium achieve in NSTX is shown. For NSTX and NSTX-U, the plasma height is limited by the vacuum vessel to be about 3 m. Therefore, the plasma height limitation tends to limit the achievable

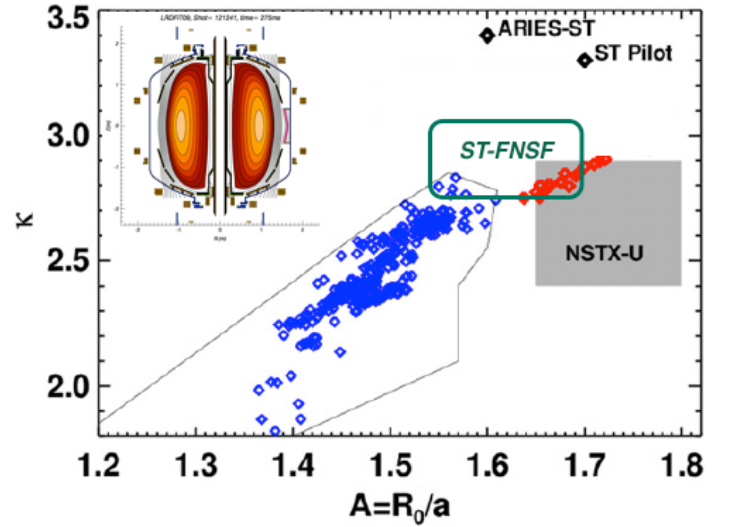


Fig. 10.1. Plot of elongation versus aspect ratio for NSTX discharges. The black line encloses the historical NSTX operating space for discharges with flat-top duration exceeding 0.5 s, and specific high-performance discharges are indicated in blue. The points collected in the high aspect-ratio experiment are indicated in red. The approximate future device operating space is as labeled. Inset shows $\kappa = 3$ NSTX equilibrium.

κ to ~ 3.0 . For FNSF, $\kappa \sim 3.0$ might be satisfactory but for high f_{bs} devices such as pilot and ARIES-ST, higher κ values of 3.3 – 3.4 is desired. As noted in Sec. II, one can relax physics parameters such as κ , q^* , and β_T toward a more FNSF-like range if the copper TF magnet is replaced by a superconducting magnet, due to greatly reduced recirculating power. The superconducting device size increases, however, to $R_0 \sim 4.5$ m due to the additional magnet shielding needed, while larger values of β_N and f_{BS} are still required to minimize the necessary non-inductive current drive.

Clearly, there are still a number of physics issues one needs to address. At the present time, experimental research has been mostly carried out in individual topical areas to develop the necessary tools as discussed in the previous sections. The tools developed in those topical areas then must work together in an integrated manner before we have sufficient confidence to design and build future devices that can operate in steady-state reliably with the projected plasma performance. Once those tools become available, their integration can take place relatively quickly in devices like NSTX-U and MAST-U.

While tokamaks have been working for some years on integrated scenario development aimed for the ITER operations, for FNSF, integration scenario development has just begun. A question commonly asked is whether or not it is possible to achieve reliable steady-state (i.e., non-inductive) plasma operation with all the projected plasma parameters as indicated in Table III. In Sec 10.B, we shall discuss the importance of plasma optimization for plasma performance particularly toward achieving non-inductive operations. There are many aspect of plasma optimization, and we shall attempt to cover only the main ones. In Sec. 10.C, we will summarize the progress being made toward non-inductive operation and the prospects for full non-inductive operation in the upgraded ST facilities.

B. Plasma Optimization – Plasma optimization is an important part of integrated scenario development, and is made challenging since there are many parameters to optimize. While a particular set of plasma parameters is chosen for a design point, there maybe other parameters which are easier to achieve while satisfying the same facility operational goals. For example, as noted in Table III, there are three FNSF design points with relatively large device and plasma parameter variations, but are still aiming for the same goal of steady-state 1 MW/m^2 neutron wall loading. Also, certain plasma parameters are coupled so one cannot arbitrarily choose those

parameters independently. If the confinement is not sufficient, the target plasma parameters may not be reached with the available heating power. In Fig. 10.2, the achieved β_N and figure of merit for $f_{bs} < 0.5\epsilon^{0.5}\beta_p$, chosen from the entire NSTX discharge data base, are plotted as a function of the achieved pulse duration [364]. It should be noted for NSTX, because of the limited inductive volt-second availability (~ 0.6 V-sec), a typical ohmic discharge would only last for ≤ 0.5 sec. The larger the non-inductive current fraction, the lower the loop voltage becomes. This enables the discharge to extend well beyond ~ 0.5 sec possible by available inductive drive as illustrated in Fig. 10.2(b). The benefit of the shape factor $S \equiv q_{95}I_p/aB_T$ is also apparent in achieving both high β_N and f_{NI} , as shown in Fig. 10.2. We should also note that in Sec. III.F, we discussed major disruptions, and in Fig. 3.12, depicted the disruptivity dependence

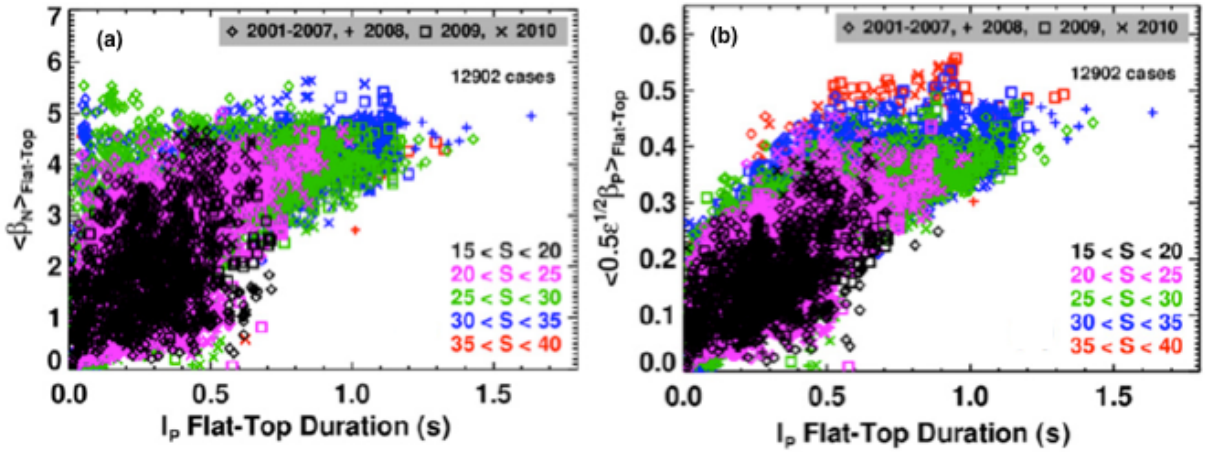


Fig. 10.2. Flat-top average of global performance parameters plotted against the I_p flat-top duration. The colors represent different values of the shape parameter. The symbols are indicative of the year when the discharge was taken, as indicated in the legend. The quantities shown are (a) β_N and (b) $0.5\epsilon^{1/2}\beta_p$.

of β_N , and q^* , pressure peaking factor $F_p \equiv p(0)/\langle p \rangle$, S , and internal inductance factor l_i for NSTX [25]. The disruptivity trend we saw in Fig. 3.12 is actually encouraging, as future ST FNSF and power plant designs trend toward higher β_N and S and lower F_p and l_i , in the direction of reduced disruptivity. Disruptivity diagrams such as Fig. 3.12 are quite informative in determining not only if the target operational plasma parameters are realistically achievable, but also a safe path to reach the target parameters without suffering a disruption. Indeed, the observed disruptivity tends to be reduced for high β_N in the $q^* = 3 - 4$ range. This is where most of the future device design points lie as shown in Table III. We now comment briefly on the plasma performance dependence on some of the key plasma and device parameters:

Aspect-ratio Scaling - The aspect ratio A is important as it distinguishes STs from conventional tokamaks. Most of the ST devices operating today are indeed operating in the projected ST future device range of $A \sim 1.5 - 1.7$. In the present-day experimental data base, plasmas with a wide aspect ratio range of $A \sim 1.2 - 1.7$ have been investigated; for example, Pegasus, TS-3-4, UTST, VEST, and HIST are in the ultra-low- A range of ≤ 1.4 , and other ST devices, i.e., NSTX, MAST, GLOBUS-M, and QUEST, etc., are in the mid-range of $A \sim 1.4 - 1.7$. The aspect-ratio scaling of MHD stability has been investigated based on modeling and experimental [66, 67]. Thus far, while there are some observable trends as described in Sec. III and V, there have been no dramatic qualitative changes observed in the MHD stability and confinement properties with A for ST regimes. This observation does give some flexibility for choosing an optimum aspect ratio in the range of $A \leq 1.7$, depending on the design objectives of future STs.

Plasma equilibrium and shape control - Plasma equilibrium and control are the fundamental part of to integrated scenario development. A comprehensive description of plasma equilibrium reconstruction has been provided in [365] and real time reconstruction developed for plasma control in tokamaks [366] applied to STs is described in Ref. 367. For beta control, the real time control system is also applied to the NBI heating system [368]. While a high plasma shaping factor is quite important as noted in Fig. 10.2, κ is probably the most important in terms of plasma performance. As described in previous chapters, higher κ helps achieve higher I_p , β , and f_{bs} , all of which are highly important for fusion reactor performance. Here we show an example of the plasma elongation research conducted in NSTX [248]. As shown in Fig. 10.3, the achieved “sustained beta” parameter $\epsilon^{1/2}\beta_p\beta_T$ (i.e., a figure of merit of non-inductive operation $\propto f_{bs}\beta_T$) is plotted as a function of the elongation factor

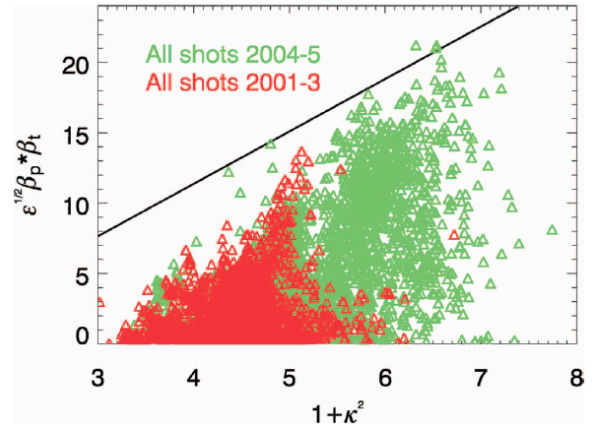


Fig. 10.3. The quantity $\epsilon^{1/2}\beta_p\beta_T$ averaged over the plasma current flat-top is plotted against $1 + \kappa^2$, also averaged over the current flat-top. The straight line in the figure is meant to guide the eye.

$1 + \kappa^2$. The red points cover a period of NSTX operation where the plasma control system is not adequate to control high elongation discharges. They show mainly the $\kappa \leq 2$ regime, which is considered to be “naturally” stable for STs and thus not needing fast feedback control as

described in Sec. III. The green points were obtained once an adequate (i.e., sufficiently fast) plasma control system was implemented in NSTX to control higher elongation ($\kappa \geq 2.5$) discharges. The higher elongation resulted in a higher sustained beta parameter as shown in Fig. 10.3 [248]. This example illustrates the importance of plasma elongation, and adequate control capability to achieve the desired plasma parameters in a stable fashion. That is the reason for the choice of relatively high κ values (≥ 3) for all of the future ST devices shown Table III. Utilizing the updated plasma control system, stable control of high elongation discharges $\kappa \sim 3$ was demonstrated as shown in the inset in Fig. 10.1 [95]. While κ confinement scaling is not yet established for STs, a favorable confinement trend has been observed for a higher A tokamak [369]. The triangularity δ is considered to be also important for MHD stability, pedestal, and ELM behavior, and divertor performance. High-triangularity high performance discharges with $\delta \sim 0.8$ have been demonstrated on NSTX. However, given the limited number of in-board PF coils, high $\delta \sim 0.8$ plasmas may not be accessible, and it is thus unclear what would be the optimum δ for future STs. This is reflected in the relatively modest range of $\delta \leq 0.64$ chosen for ST-based fusion systems as shown in Table III (Sec. II). There is an additional shape factor called “squareness,” but its precise benefit is not well established at the present time, due mainly to the lack of experimental data. The squareness was also scanned in a set of recent experiments in NSTX [370] but no strong plasma performance dependence has been thus far found.

ρ^* and ν^* scaling – The ρ^* (ratio of ion gyro-radius to plasma minor radius) is considered to be an important parameter for tokamaks for NTMs and possibly energetic particle instabilities. Since ρ^* is expected to decrease by an order of magnitude for future machines such as ITER compared to present day tokamak experiments, ρ^* scaling is an important research topic for tokamaks [171]. For STs, the ρ^* is expected to remain relatively unchanged from present-day STs to future STs including the ST Demo, perhaps varying by a factor of two. While the ρ^* variation maybe modest for STs, the confinement projection can be significantly influenced, for example, by whether ρ^* scaling is Bohm-like or gyro-Bohm-like for STs. This area of research still remains to be an important future research topic for STs. On the other hand, the ν^* (*collisionality*) for the present-day STs are one to two orders of magnitude higher than what is anticipated in future STs. The ν^* can, for example, influence ELM behavior, transport physics, bootstrap current generation, and boundary physics as described in the previous sections. On

NSTX and MAST, the plasma confinement (mainly in the electron channel) shows a strong dependence of ν^{*-1} as discussed in Sec. V. If this trend holds true, then future STs maybe able to achieve even better confinement due to much lower ν^* . The ν^* scaling is therefore viewed as a particularly important dimensionless parameter for STs, and thus is one of the high-priority research areas for NSTX-U and MAST-U.

Plasma beta scaling – As discussed in Sec. III, the plasma betas are crucial parameters for fusion power production (β_T) as well as plasma maintenance with high bootstrap current fraction (β_N). The present-day experiments achieved beta parameters closer to that needed for the ST-FNSFs (i.e., $\beta_T \leq 20\%$ and $\beta_N \leq 5.0$) as shown in Fig. 10.2 and 10.3. It is also quite fortunate that the high β_N regime exhibits relatively stable MHD behavior and good confinement up to the highest $\beta_N \sim 6$ achieved thus far. The achievement of ST-FNSF-like beta regimes with a significant off-axis bootstrap current fraction would likely require advanced MHD control, such as an active RWM control system. This is because of the relatively low internal inductance expected in these plasmas, which makes them prone to the $n=1$ kink/RWM instability as described in Sec. III. The bootstrap current fraction is at the 50-60 % level in present high-performance H-mode ST plasmas, which is sufficient for the ST-FNSF but not at the ultra-high $\sim 90 - 95$ % level envisioned for ST power plants.

Current and plasma profile control – Current and plasma profile control is one of the desired features for future STs particularly for the ST Power Plant to achieve the near-perfect current and pressure profile alignment needed for the high β_N and f_{BS} regime. This is also the case for advanced tokamak operational scenarios. If one were to use NBI as a non-inductive current drive tool, for example, as envisioned in the FNSF, then off-axis NBI current drive is a promising tool for current profile control in future STs. Off-axis NBI was investigated on MAST by shifting the magnetic axis of the plasma far off the mid-plane ($Z_{mag} = 0.35\text{m}$) as shown in Fig. 10.4 [371]. The MAST results indicates that broadening the fast ion deposition profile by off-axis

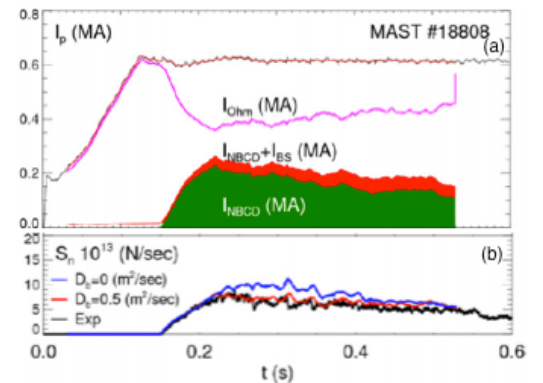


Fig. 10. 4. (a) TRANSP simulated distribution of the total plasma current, I_p , between Ohmic, bootstrap and neutral beam driven components for $D_b = 0.5\text{m}^2\text{s}^{-1}$. (b) Experimentally observed neutron yield is compared with TRANSP simulations for $D_b = 0$ and $0.5\text{m}^2\text{s}^{-1}$.

neutral beam (NB) injection helps to avoid harmful plasma instabilities, and significantly extends the operational window of MAST. Long pulse (>0.65 s) H-mode plasmas were achieved with plasma durations limited only by present machine and NBI engineering limits as shown in Fig. 10.4 (a). As shown in the figure, about 1/3 of the plasma current is driven by NBICD with a small incremental fraction by j_{bs} . In order to match the experimentally observed neutron rate and stored energy, however, a low level of anomalous fast ion diffusion ($D_b \sim 0.5 - 1.0 \text{ m}^2 \text{ s}^{-1}$) is required. This anomalous diffusion with incremental reduction in current drive efficiency is attributable to $n=1$ fish-bone-like MHD activity. The introduction of the fast ion diffusion broadens the neutral beam current drive profile and degrades the relative contribution of NB-driven current from $\sim 40\%$ to $\sim 30\%$. This so-called anomalous fast ion diffusion was also observed previously on NSTX due to a rotating $n=1$ MHD kink/tearing mode [330]. As shown in Fig. 10.5, the NBI current drive radial diffusion was deduced from the observed neutron rate as shown in Fig. 10.5(a). The observed calculated neutron rates agree well until the on-set of the $n=1$ mode around 1.05 sec. Then a significant deviation develops suggesting anomalous fast particle diffusion/confinement. Two examples of

diffusion profiles (high central diffusivity in red and lower broader diffusivity in black) are shown. As shown in Fig. 10.5(b), the NBICD profile (green) is dramatically broadened by diffusion. The broadened NBICD profile tends to make the plasma more MHD stable, which could enable stationary plasma operation. However, the large core $n=1$ mode reduced the plasma rotation, and degraded confinement and performance significantly. The NBI ion transport due to TAE was described in Sec. VIII.C, and this type of current diffusion can be relatively benign in terms of its effects on plasma performance [326, 327]. The NBI on-axis/off-axis current drive will therefore require consideration of possible instability-driven (anomalous) radial transport, which could broaden the driven current

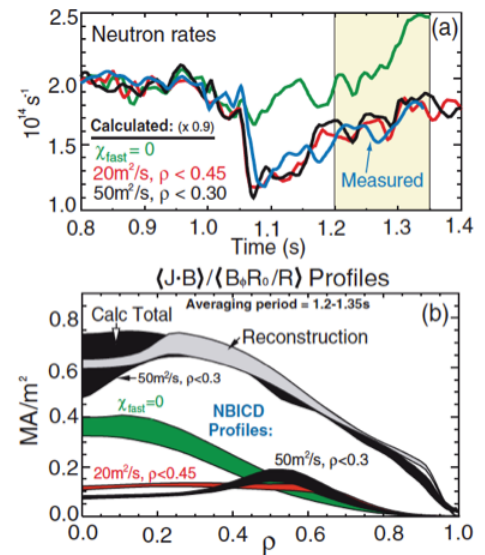


Fig. 10.5. (a) Measured (blue line) and calculated neutron rates (x 0.9) and (b) comparison of reconstructed (gray) and calculated (black) total parallel current density profile for $t = 1.2-1.35$ s for the best-fit fast-ion diffusivity model. NBICD profiles for the diffusivities of (a) are also shown.

with the benefit of making the plasma more MHD-stable but could also reduce the current drive efficiency. The use of EBW off-axis current drive for current profile control has also received some attention as described in Sec. IX.C. Off-axis current drive physics will be further investigated in NSTX-U and MAST-U, where significant off-axis current drive capabilities will be implemented [24, 25, 372]. As for plasma profile control, the profiles are largely determined by the confinement properties of the plasma. The H-mode produces much broader plasma pressure profiles, which enabled NSTX and MAST to explore high beta / performance regimes. The plasma rotation profiles are also determined by the momentum transport, but the application of a 3-D field enables some control of the rotation speed and profile. Because of the need for the real time plasma control, diagnostic systems are also moving toward real time capabilities for NSTX-U and MAST-U.

C. Prospects for steady-state high performance discharges – All of the future ST devices listed in Table III are designed to run essentially steady-state. Since it is highly challenging and costly to actually run an experimental research facility truly steady-state, the experimental goal for present-day STs is to reach a condition which is termed “physics steady-state.” Here we are not addressing the engineering or technology steady-state conditions, such as wall-particle interaction time scales which could be hours or even longer. The longer time scale non-inductive ST operation is the goal of the QUEST device [134]. The physics steady-state conditions would be the time for the plasma to be maintained for much longer than a certain physics driven time constant such as the energy, particle, and momentum confinement time, and the current diffusion time τ_{CD} for current profile stationary conditions. Since the τ_{CD} is usually the longest physics time constant, the physics steady-state can be effectively defined as a

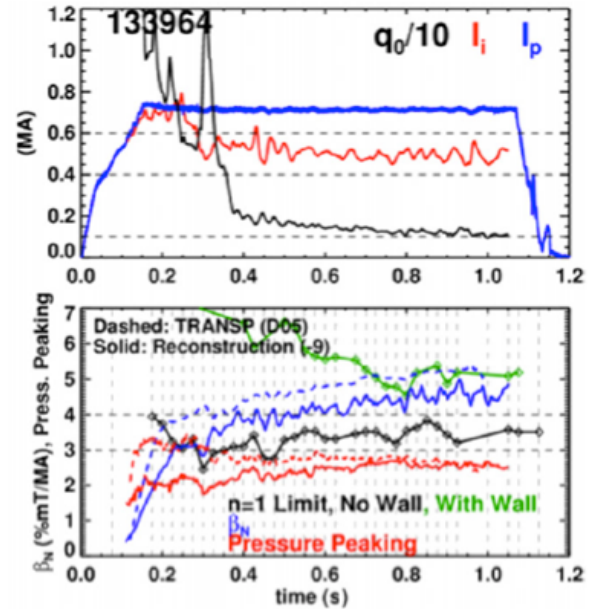


Fig. 10.6. A high- β_p (0.7MA and 0.48 T) scenario NSTX discharge with lower- l_i and higher pressure peaking. (a) The plasma current, internal inductance and central safety factor as labeled. (b) The normalized β , pressure peaking factor and no- and with-wall stability limits.

plasma discharge time duration $2-3 \tau_{CD}$. The long pulse experiments that have been performed on NSTX were mainly possible due to discharges with relatively high bootstrap current fractions of $\sim 50 - 60\%$. The discharges had a non-inductive current fraction of $\sim 70\%$, with very limited available OH flux of only ~ 0.6 V-S. An example of such a discharge is shown in Fig. 10.6, where the pulse length is sufficiently long to reach some degree of stationary conditions as evident by the nearly flat q_0 and I_i evolution as shown in Fig. 10.6(a) [364]. The $\beta_N \sim 5$ is well above the no-wall limit, and even the with-wall (or ideal) limit is approached, with f_{BS} well over 50% and f_{NI} reaching $\sim 70\%$. We note that these parameters reflect the conditions for FNSF.

As we prepare for fully non-inductive operation on NSTX-U and MAST-U, it is natural to ask how the present NSTX results could extrapolate to NSTX-U. With near doubling of B_T and lower collisionality, one could expect significantly higher T_e and T_i . In Fig. 10.7 (a), the discharge shown in Fig. 10.6 is indicated as the leftmost point at a temperature multiplier TM of one [342, 349]. The points for $TM > 1$ were obtained using TRANSP to increase TM and iterating until the current profile reached a stationary state while the input power, field, current, and the boundary shape were held fixed. With increased TM, the f_{NI} rises and crosses 1 to reach a

fully non-inductive state at $TM \sim 1.4$ (or a 40% higher temperature). The required confinement enhancement was 1.5 and $\beta_N \sim 6.5$. The q_0 is raised from 1 to 1.45, and the mid-radius ν^* is reduced by $\sim 50\%$ from 0.17 to 0.09. The current and q profiles are shown for the end points ($TM = 1$ and 1.4) in Fig. 10.8(b). On both NSTX-U and MAST-U, the available B_T enhancement should be nearly a factor of 2, with considerably more off-axis beam power for current profile control. Generally higher temperature and reduced collisionality increases both bootstrap current and NBICD. If the confinement enhancement is not sufficient, one could introduce additional NBI power, which also increases the current through NBICD. It is also quite possible that the

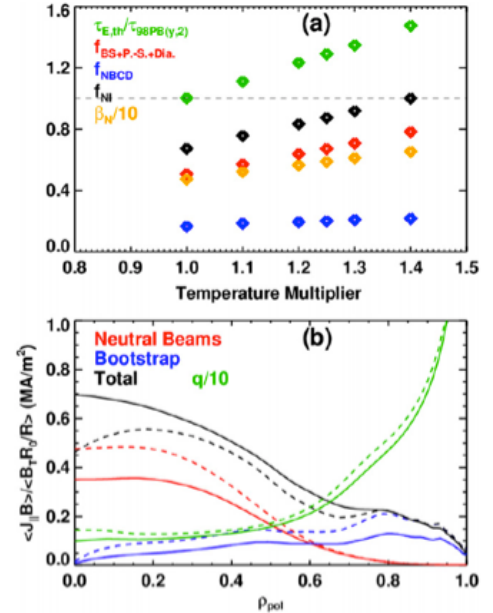


Fig. 10.7. (a) Dependence of the confinement time, pressure-driven current fraction, beam current drive fraction, total non-inductive fraction and β_N on the temperature multiplier in the TRANSP simulations, and (b) profiles for the base configuration (solid) and fully non-inductive configuration (dashed).

MHD (both low frequency $n=1$ and higher frequency TAE avalanche) instabilities could affect the NBICD performance through anomalous particle diffusions, as noted above and also in Sec. VIII.C. Some moderate fast ion diffusion $\sim 1\text{m}^2\text{s}^{-1}$ can actually be beneficial for MHD stability by reducing the pressure peaking and raising the q_{min} with somewhat elevated confinement requirement [372]. While there are a number of challenges, the prospects for achieving the FNSF-relevant fully non-inductive regime still appears to be quite bright for the upcoming experiments on NSTX-U and MAST-U.

XI. Discussion and Conclusions

The spherical tokamak (ST) is a class of tokamak confinement devices that has numerous features that make it unique for fundamental studies in plasma physics and plasma-surface interactions, and attractive as a fusion reactor concept. The increase in the ideal tokamak β and sustainable current with low aspect ratio, together with natural plasma elongation, enable high performance plasmas to be achieved in a compact geometry. These features are being exploited in more than sixteen ST research facilities around the world, including two megampere-scale facilities, NSTX in the United States and MAST in the United Kingdom. Both NSTX and MAST are undergoing upgrades that include higher neutral beam powers, increased pulse lengths, and field coil modifications to allow greater flexibility in magnetic topology.

The role of the ST in fusion energy development has two main features. The first is involves a Fusion Neutron Science Facility (FNSF). The requirement of a fusion tritium breeding blanket module is a major technological challenge for magnetic fusion research, and the compact nature of the ST offers an economical approach to the high-volume plasma-based neutron source (VNS) required for its development. The second is the development of fusion power plant, and two approaches have been pursued. The copper-based ST power plant has been the focus of the ARIES-ST study in the US. It has the advantage of allowing a more compact side reactor, but the higher resistivity of the copper means larger recirculating power. The superconducting toroidal field (TF) magnet approach pursued in Japan and Korea results in a larger power plant because of the neutron shielding requirements, but it also means that the TF magnet is expected to last for the lifetime of the plant. The clear advantage of using either copper-based or superconducting magnets has yet to be shown, so further exploration of each approach to ST reactor design is needed.

The achievement of high beta values has been a salient characteristic of STs, and it has been successfully understood in terms ideal and resistive MHD models, and more recently, kinetic MHD theory. Resistive wall modes (RWMs) have been an obstacle for

achieving high β_N , particularly as β approaches the ideal MHD beta limit. This has been overcome in ST experiments with techniques that have including conducting structures that conform to the plasma boundary, active RWM stabilization coils, and plasma rotation. For stability against neoclassical tearing modes (NTMs), the stronger shaping, higher beta, and higher q that are characteristic of STs provide them with an inherent advantage over tokamaks with larger aspect ratios. As in all tokamaks, non-axisymmetric effects are important for STs. Performance improvements have been achieved in NSTX, for example, by correcting for poloidal field coil asymmetries and applying 3-D control coils. Disruptions are a major issue for tokamaks, and the consequences of non-axisymmetric halo currents induced in the inboard (“high-field”) when they occur are a particular concern. Techniques for detecting the onset of disruptions have been tested using an extensive database of NSTX plasmas, and will be the basis of a future disruption avoidance system.

Tokamaks have historically relied on an ohmic solenoid in the center of the device for starting up the discharge and ramping up the plasma current. Such a solenoid, however, increases the dimensions of ST reactors due to the space required for coil itself and the neutron shielding that must be supplied. This makes solenoid-free start-up among the most active areas of ST research. The spherical configuration and the small major radius of the ST, however, results in relatively low external and internal inductances, which reduces the poloidal flux needed to achieve full plasma current. Approaches presently under investigation include RF heating and current drive, helicity injection, and merging-compression. Startup with magnetic induction is also under consideration, as there are ST power plant designs that can accommodate shielding for ohmic solenoids within the requirements of protecting superconducting magnetics. Because of their importance, the development of start-up techniques will continue to be a key element of future ST research.

As with magnetic fusion devices in general, a fundamental understanding of plasma transport physics is critical for the proper interpretation of present-day ST results and extrapolating them to future STs. A particular area of interest is the confinement quality

of fusion plasmas in the higher current and lower collisionality regime for STs. Compared to tokamaks, STs have large trapped particle populations because of their large mirror ratios, and it is important to know how this affects anomalous transport in low-collisionality plasmas. Global scaling trends indicate that confinement improves with decreasing collisionality, but the underlying mechanism for anomalous transport is not as well understood as the basis for classical or neoclassical transport. This has motivated a focus on anomalous transport in ST research, especially since high-beta ST plasmas could have electromagnetic turbulence that can significantly affect electron confinement. For ions, the transport in ST H-mode plasmas is at neoclassical levels in part due the ExB shear stabilization of ion temperature gradient and trapped electron modes. The high shaping and low magnetic field in the STs can result in strong ExB sheared flow rates that exceed that of the growth rate of these modes. The primary cause for electron transport, in contrast, is not as clear. Improved modeling of driving mechanisms like electron temperature gradient instabilities are being developed, however, as are innovative diagnostics for measuring their characteristics in STs.

Operation in the “high” confinement mode or H-mode have been the primary means of obtaining high performance plasmas in present day tokamaks, and the success of ITER is predicated on H-mode achievement for high fusion gain. Access to the H-mode is also critical for ST reactors, where the broader current and pressure profiles of H-mode plasmas are advantageous for non-inductive operation with high bootstrap current fractions. As with large aspect ratio tokamaks, there is an incentive to minimize the heating power threshold needed to achieve H-modes. Among the salient results in NSTX, for example, was the significant reduction in the threshold power after lithium wall conditioning. While experiments in MAST do not seem to suggest a critical electron temperature (T_e) needed to access the H-mode, the increase in edge T_e observed after the application of lithium could explain the NSTX results. A ubiquitous characteristic of H-modes is the presence of Edge Localized Modes (ELMs). Because of their compact nature, plasmas in STs are amenable to visualization of the global structure of ELMs with fast camera imaging. For ELM control, coils that introduce resonant magnetic perturbations have changed the frequency of ELMs on MAST. They have also been used

to destabilize ELMs after ELM-free discharges were established with lithium conditioning in NSTX. The detailed physics of the H-mode power threshold and ELM dependence on pedestal parameters is still not well understood. The ST will continue to provide a unique configuration for addressing such questions, as the pedestal dependence of the H-mode means that it can be studied in a region that is most sensitive to the effects of low aspect ratio.

The compact configuration of STs allows very high divertor heat loads, which makes them well suited to address the challenges they pose for future reactors. With peak divertor heat fluxes in present STs exceeding 10 MW/m^2 , and more than 100 MW/m^2 transiently during ELMs and disruptions, they approach values anticipated in ITER. The flexibility of the ST geometry, however, also makes it amenable to test methods for mitigating the effects of high heat loads. The snowflake divertor configuration, for example, can create a divertor flux expansion of up to ~ 50 . Complementing the need for heat load mitigation techniques, an understanding is also required of the physics underlying transport of heat and particles in the plasma edge. Novel diagnostics for the study of edge turbulence transport have been developed, such as gas puff imaging (GPI). Measurements with GPI have shown that as with turbulence in the open field line region of other magnetic confinement devices, STs share the same generic drift wave and nonlinear “blob” or “filamentation” formation mechanisms. What is unique, however, is that the compact ST geometry allows fast camera images of edge turbulence to be obtained for the entire plasma. While features of blobs like their size and radial velocity are generally consistent with electrostatic turbulence models, their role in energy and particle transport and divertor heat and particle flux requires further investigation.

High heat fluxes in STs also provide a strong incentive to develop PFCs for handling them. As in conventional tokamaks, carbon was the first PFC material to be used in high power STs. Lithium coatings were initially demonstrated as an effective means of conditioning carbon PFCs in TFTR, and lithium evaporation has been routinely used to obtain high performance discharges in NSTX. Carbon PFCs and solid PFCs in general, however, have serious limitations for reactor applications. This motivated the exploration

of liquid lithium PFCs, and a large area liquid lithium limiter was successfully implemented in CDX-U. This was followed by the installation of a liquid lithium divertor (LLD) in NSTX. The compact ST geometry allowed each concept to be investigated with a relatively small PFC area compared to conventional tokamaks. The KTM takes further advantage of this ST feature by having a design that allows the replacement of divertor modules without venting the vacuum vessel. The KTM program is presently focused on flowing liquid lithium divertors, and their development is expected to continue as an active area of ST research in the future.

The study of energetic particle (EP) physics is a key element of magnetic fusion research, due to the importance of alpha heating in fusion reactor plasmas. An understanding of the super-Alfvénic regime is of special interest, since the alpha particle velocity typically exceeds the Alfvén velocity in ITER and future magnetic fusion reactors. Super-Alfvénic fusion alpha particles and EPs driven with auxiliary heating can excite Alfvénic modes, which can cause an "anomalous" slowing down of particles as well as radial transport. The consequences of such effects include a reduction of alpha heating effectiveness, and damage to PFCs due to the rapid loss of EPs. The ST has the capability of spanning a much broader range of parameters relevant to EP physics studies compared to conventional tokamaks, and is thus particularly attractive as a basis for extrapolations to burning plasma regimes. Compressional Alfvén Eigenmodes (CAEs) and Global Alfvénic Eigenmodes (GAEs) constitute a new class of high frequency modes discovered in STs. Significant EP radial transport and prompt losses have also been observed in STs in conjunction with Alfvénic modes. Prompt losses occurred with the simultaneous excitation of multiple over-lapping modes, or "avalanches," particularly associated with the TAE (Toroidal Alfvén Eigenmode). Substantial progress has been made in simulating such phenomena, but non-linear models still need development for the predictive capability required for ITER and future STs. Improvements in diagnostics are also planned to better characterize Alfvénic modes and EP populations in STs.

Neutral beam injection (NBI) has been shown to be an effective means for heating and current drive in ST plasmas. However, limitations in beam penetration and the need to

shield large complex structures close to the plasma chamber constrain the practicality of NBI for fusion reactors. Radiofrequency (RF) waves are an alternative for heating and current drive. The ST presents a special challenge compared to conventional tokamaks, because its plasmas operate in the overdense regime. The lower ST field means that the ratio of the electron plasma frequency to the electron cyclotron frequency is in the range of 10 to 20. This ratio is an order of magnitude lower in conventional tokamaks, where RF techniques like Electron Cyclotron Heating (ECH) and the Lower Hybrid Current Drive (LHCD) have been successfully demonstrated. In STs, on the other hand, the wave accessibility limits imposed by the overdense condition have led to a focus on the High Harmonic Fast Wave (HHFW) and Electron Bernstein Wave (EBW) as alternative RF approaches. A 6MW HHFW system has been used for plasma heating and current drive in NSTX, and EBW capabilities in the MW range are planned for MAST-U and NSTX-U. While RF power sources can be readily isolated from the plasma chamber, much work remains to be done in developing launching structures compatible with the high heat and particle fluxes that future ST PFCs must handle.

The success of future STs depends on the understanding gained from integrated scenario development in current STs. This involves operating them in ways that match as many key dimensionless parameters as possible. The actual steady-state conditions expected in a reactor are not achievable in present STs. However, they are able to operate with plasma durations that are long compared to the time scales of key physics parameters, including the energy, particle, and momentum confinement times, and the current diffusion time required to achieve steady-state current profiles. In NSTX, for example, plasmas were achieved with β_N above the no-wall limit and high bootstrap current values that are characteristic of FNSF conditions. These parameter ranges can be further explored when the NSTX-U and MAST-U devices become operational.

The ST builds on the success of the tokamak approach to achieve high-performance plasmas. At the same time, STs are able to reach parameters that attractive for FNSF and fusion power plants, but are not as readily accessible at larger aspect ratio. Significant

challenges remain, however, and they are being addressed through the extensive efforts of the international ST community.

APPENDIX A: Basic characteristics of tokamaks

The basic tokamak configuration is depicted in Fig. A-1. A detailed description of tokamaks is provided, for example, in Wesson's book "Tokamaks" [20]. A comparison of tokamak and ST plasmas is described in Sec. II and depicted in Fig. 2.1. The essential ingredient of a tokamak configuration is the toroidal magnetic field $B_{TF} \propto I/R$, where R_0 is the major radius. This is generated by toroidal field coils, usually consisting of a number of coils placed uniformly around the torus. Their purpose is to generate a nearly axi-symmetric toroidal field, with minimal toroidal field ripple between the coils. The toroidally-flowing plasma current I_p creates the poloidal field B_{PP} (Fig. A-1). Additional poloidal fields B_{PF} , generated by a number of externally-placed circular poloidal field coils around the plasma with their current in the toroidal direction, maintain the desired plasma equilibrium. The B_{PP} , together with B_{TF} and B_{PF} , generate "nested" toroidal flux surfaces for confining a stable toroidal plasma. The so-called equilibrium poloidal field coils shown in Fig. A.1 have their coil current direction opposing the plasma current. This applies essentially a vertical field B_{PF} to exert a radially-inward force $\propto I_p \times B_{PF}$, to hold the outwardly expanding plasma current ring. There are other poloidal field coils (not shown) for plasma shape control and divertor x-point creation. In general, a tokamak facility typically has a central ohmic heating solenoid (not shown) to induce plasma current by induction. The tokamak configuration is ideally an axi-symmetric system, although the study of 3-D fields is an active research topic as described in Sec. III. The world-wide effort in tokamak research has led to the International Thermonuclear Experimental Reactor (ITER). The ITER device is presently under construction, and is intended to demonstrate burning fusion plasmas. The ITER Physics Requirement Documents [21, 22] also provide a good summary of the status of tokamak research until the time of its publication (1999 and updated in 2007.)

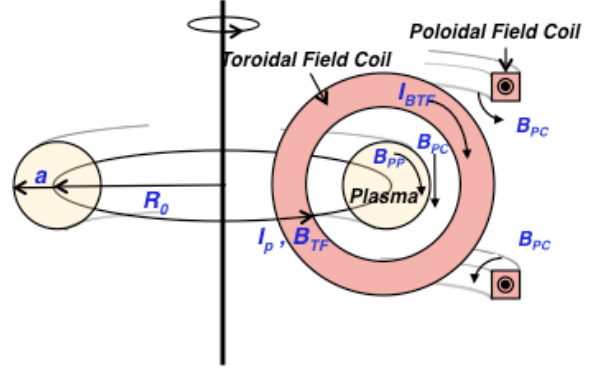


Figure A.1. Basic characteristic of a simple circular tokamak

The definitions for often used parameters are as follows:

a = plasma horizontal minor radius at the mid-plane

b = plasma vertical minor radius

$\kappa \equiv b/a$ = plasma elongation

R_0 = plasma major radius

$A \equiv R_0/a$ = plasma aspect ratio

I_p = plasma toroidal current

$L_p \equiv$ effective plasma minor circumference $\sim 2 \pi a [(1 + \kappa^2)/2]^{1/2}$

$B_p \equiv$ poloidal magnetic field $\equiv \mu_0 I_p / L_p$

$q^* \equiv$ cylindrical kink (MHD) safety factor $\equiv I_p B_{T0} / 2 \pi R_0 B_p$

$B_{TF} \equiv$ toroidal magnetic field $\propto 1/R$

$\beta_{TF} \equiv$ toroidal beta $\equiv 2 \mu_0 \langle p \rangle / B_{T0}^2$

$\beta_N \equiv$ normalized beta $\equiv 40 \pi \beta_T a B_{T0} / \mu_0 I_p$

Appendix B: Brief background for STs and tokamaks

The tokamak configuration is the most developed concept within magnetic confinement fusion research with a history of over 50 years [20 - 23]. A brief description of tokamaks and the definitions of their key parameters are given in Appendix A. With the confirmation of a high plasma electron temperature of ~ 1 keV in the 1960's in Russia, tokamak research expanded rapidly worldwide. Soon after the demonstration of good energy confinement in the tokamak configuration, interest in high plasma beta (β) naturally developed for economical tokamak reactor designs. The β is the ratio of plasma pressure to the applied toroidal magnetic pressure, and the thermo-nuclear fusion power produced roughly scales as β^2 for a given toroidal magnetic field. Ideal

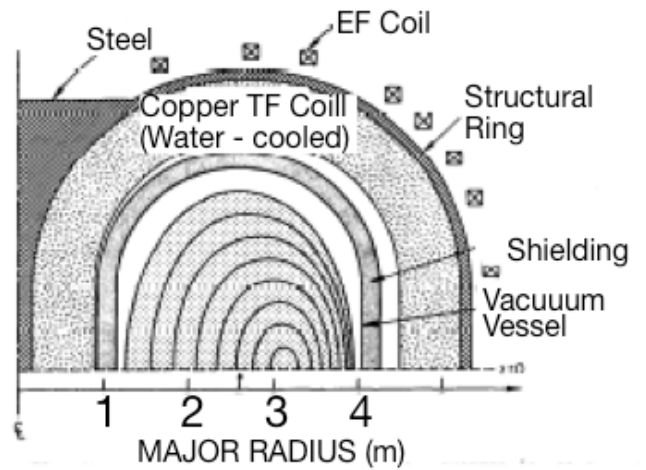


Fig. B-1. Elevation view of illustration design for SMARTOR (Small-Aspect-Ratio Torus). EF coil positions are schematic only. $B_{TF} = 11$ T at $R =$

tokamak MHD calculations were carried out in the 1970's, when numerical tokamak stability codes were developed for the first time. It was recognized quite early (i.e., well before the availability of experimental data) that high- n ballooning and lower- n kink modes were the main beta-limiting ideal MHD instabilities for tokamaks [373 - 375]. In Ref. 374, for example, it was predicted that the tokamak β limit increases as the inverse of the tokamak aspect ratio $A \equiv R_0/a$, where R_0 is the tokamak plasma major radius and a is the minor radius, (i.e., $\beta \propto 1/A$). Therefore one may note that because of its favorable beta limit, the potential attractiveness of the low-aspect-ratio regime was appreciated fairly early in the history of tokamaks. This led to a number of proposals to build low aspect ratio tokamak facilities. In 1977, for example, an idea was put forth for a small aspect-ratio tokamak ("SMARTOR"), shown in Fig. B-1, to demonstrate thermonuclear ignition in a compact tokamak [376]. In the 1980's, the tokamak experimental database and theoretical calculations were used to formulate the well-known Tryon scaling [377], as described by Eq. (1) in Sec. I.

Appendix C : Relationship of STs with other magnetic confinement concepts

While the ST is a tokamak concept, ST research has significant overlap with other alternate magnetic confinement concepts, and has been influenced by them. Prior to the start of ST experiments, the spheromak concept was actively investigated as a possible attractive reactor concept. This is because its fusion blanket module does not link the magnetic field coils, and thus greatly simplifies reactor engineering. The main issue for spheromaks, as it turned out, is magnetic stability and confinement. A spheromak plasma, for example, is found to be tilt-unstable. One of the ways to stabilize this tilt instability is to insert a toroidal field (B_{TF}) coil rod through the center of the plasma. The application of a B_{TF} of sufficient magnitude could make a spheromak into a spherical torus plasma. While this insertion of a TF rod has been discussed by the spheromak community, it was not implemented in major spheromak experiments [140, 378]. This is perhaps because such a linked coil addition would eliminate the spheromak's reactor advantage. Smaller spheromak facilities, including TS-3/4 and HIST, are now capable of operating as STs with

the installation of a TF coil rod. The Coaxial Helicity Injection (CHI) plasma start-up concept, developed on HIT / HIT-II and implemented on NSTX, is a direct application of an approach developed through spheromak research. (See Sec. IV.C) The merging-compression start-up being developed on MAST was also utilized for spheromak/ST start-up in TS-3/4. (See Sec. IV.D) The PEGASUS facility, with an ultra-low-aspect-ratio configuration, is exploring the boundary between STs and spheromaks. An RFP (MST) has demonstrated the dramatic benefit of current profile control on MHD stability and plasma confinement [379]. Current profile control is also an important long term goal of STs and advanced tokamaks. Another RFP (RFX) has demonstrated the feedback stabilization of MHD modes, which has some commonality with RWM feedback stabilization research in STs and tokamaks [380]. The copper TF magnet approach for STs is also similar to other high-beta based magnetic confinement concepts, such as RFPs [379, 380] and FRCs [381]. This is because the power loss to the magnetic field coils can be made manageably small, due to the relatively modest requirements from the point of view of the reactor recirculating power. A major perceived advantage of STs over other high beta systems at the present time is the existence of possible steady-state operating scenarios, with high bootstrap current fractions and good tokamak confinement.

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