The Physics of Spherical Torus Plasmas*

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Broad and important progress in plasma tests, theory, new experiments, and future visions of the Spherical Torus (ST, or Very Low Aspect Ratio Tokamaks) have recently emerged. These have substantially improved our understanding of the potential properties of the ST plasmas, since the preliminary calculation of the ST MHD equilibria more than a decade ago. Exciting data have been obtained from concept exploration level ST experiments of modest capabilities (with major radii up to 35 cm), making important scientific contributions to toroidal confinement in general. The results have helped approval and construction of new and/or more powerful ST experiments, and stimulated an increasing number of theoretical calculations of interest to magnetic fusion energy. Utilizing the broad knowledge base from the successful Tokamak and Advanced Tokamak research, a wide range of new ST physics features have been suggested. These properties of the ST plasma will be tested at the 1-MA level with major radius up to ~ 80 cm in the new proof of principle devices NSTX (U.S.), MAST (U.K.), and Globus-M (R.F.), which have just started full experimental operation. New concept exploration experiments, such as Pegasus (University of Wisconsin), HIT-II (University of Washington), and CDX-U (Princeton

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Plasma Physics Laboratory) and other experiments in Japan and Brazil, etc., present additional opportunities for important progress. This tutorial paper summarizes our understanding and projections of the physics of the ST plasmas, the investigation of which will hopefully bring new enthusiasm and advancements for fusion energy science research in the U.S. and the world.

I. Introduction

Confining high-pressure plasmas using modest magnetic field in small-size device has been a long-standing goal of magnetic fusion energy research. The Spherical Torus (ST, or Very Low Aspect Ratio Tokamak) concept was introduced[1] to contribute to this goal. In this paper, we summarize progress in understanding of the physics of the ST plasma during the past decade toward this goal. New experimental and theoretical results will be discussed to clarify the understanding derivable from these results, the opportunities for new research introduced, and the high potential indicated of the ST plasma. Since magnetic fusion energy research is simultaneously science and energy in a grand scale, the challenges in physics and technology R&D introduced by the ST will also be described.

Important advancements in plasma tests, theory, new experiments, and future visions of the ST have emerged during the past decade. These have substantially improved our understanding of the potential properties of the ST plasmas beyond the preliminary calculation of the ST MHD equilibria more than a decade ago.[1] Enticing data have been obtained from exploratory ST experiments of modest capabilities (with major radii up to 35 cm and plasma currents up to ~300 kA), and made important scientific contributions to toroidal confinement

physics in general. These include START (U.K.)[2], HIT-I and II (University of Washington),[3,4] CDX-U (PPPL),[5,6] HIST[7] and TS-3 (Japan),[8] etc.,

These results recently helped approval of improved and/or more powerful new ST experiments, such as the proof-of-principle level devices NSTX (U.S.),[9,10,11] MAST (U.K.),[12,13] and Globus-M (R.F.)[14] with major radii up to ~ 80 cm and plasma currents up to 1–2 MA. New devices at the concept exploration level, such as Pegasus (University of Wisconsin),[15] TST-2[16] and TS-4[17] in Japan, and the ETE in Brazil,[18] etc., present additional opportunities to test the scientific boundaries of the ST plasma. An increasing number of theoretical calculations of interest also emerged recently. Utilizing, and in some cases expanding beyond, the broad knowledge base of the successful research in the Advanced Tokamak (AT),[19,20] a widened range of ST physics properties have been suggested, and will be tested in these and other experiments.

This tutorial paper begins with a brief review of the motivation and description for the ST concept in Section II. The latest calculations of maximum β limits for large scale MHD instabilities follow in Section III; possible reduction in micro-instabilities and improvements in confinement in Section IV; noninductive current drive in Section V; and possible special plasma scrape-off layer conditions in Section VI. Section VII covers the physics issues of even higher plasma β and confinement for future ST devices, such as the next-step devices with major radii ~ 1.2 m and plasma currents ~10 MA and the future power plants of even larger sizes and currents. A discussion of how the ST plasmas may be related scientifically to the FRC and Spheromak plasmas is given in Section VIII. The paper closes with a discussion of the widened range of parameters thus introduced by the ST plasma and the likely scientific significance, the investigation of which will hopefully bring new advancement and enthusiasm for fusion energy sciences research in the U.S. and the world.

II. Motivation to Minimize the Aspect Ratio

Potential advantages for minimizing the tokamak aspect ratio ($A=R_0/a$, see Figure 1, R_0 and a being the major and minor radii, respectively) were first suggested by Sykes[21] and Troyon.[22] They showed, based on a large number of calculations of MHD instabilities of tokamak plasmas for A = 2.5–5, that the maximum stable average plasma beta ($\langle\beta\rangle_{max}$) scales as:

$$\langle\beta\rangle_{max} \approx 5\, C\, \kappa / A\, q_j \approx C\, I_p / a\, B_T. \qquad (1)$$

Here C is a constant (~3 %m·T/MA), κ (= b/a) is the elongation of the plasma cross section (see Figure 1), q_j is the safety factor at plasma edge, I_p is the plasma current, and Troyon defined B_T as the toroidal magnetic field at R_0, where $(R_0B_T) = (RB_\phi)$ is the toroidal field current function at the surface of the plasma.

Since this field in tokamak is usually within a few % of the average magnetic field in the plasma, the volume average beta $\langle\beta\rangle$ ($= 2\langle p\rangle/\mu_0\langle B^2\rangle$) is very close to the toroidal average beta β_T ($= 2\langle p\rangle/\mu_0 B_T^2$). Thus without introducing much error, the common usage of Eq. (1) in tokamak research has replaced $\langle\beta\rangle$ by β_T:

$$\beta_{Tmax} \approx 5\, \beta_N\, \kappa / A\, q_j \approx \beta_N\, I_p / a\, B_T, \qquad (2)$$

the second form of which has also been called the "Troyon scaling."

This alteration is a convenient simplification for a given plasma pressure. We follow the same practice throughout this paper, where β_{Tmax} is the maximum stable β_T.

As will be shown below that for a given plasma pressure in ST, β_{Tmax} (and β_N) can be substantially larger than $\langle\beta\rangle_{max}$ (and 3 %m·T/MA) because the poloidal component of B can be comparable or even larger than B_T over a large fraction of the plasma volume. Further, the large plasma poloidal current can produce large changes from the applied toroidal field within the plasma, introducing further adjustments to $\langle\beta\rangle$.

The first expression due to Sykes shows that β_{Tmax} increases with decreasing A and q_j and increasing κ. By reducing A alone by a factor of two would double β_{Tmax}, which would be a large and important improvement for magnetic confinement fusion. The second expression due to Troyon contains the "normalized current" (I_p/aB_T), which indicates the potential for the plasma to achieve high β_{Tmax}. More recently, a "shaping factor" S (= $q_jI_p/aB_T \approx 5\beta_N\kappa/A$) was observed[23] to represent the potential for the plasma to achieve high beta limit as well as good confinement. This suggests that high performance of the tokamak plasma also depends on β_N; increase in β_N itself increases the shaping factor of high performance tokamak plasmas, and hopefully also of ST plasmas.

It is therefore of interest naturally to investigate the extent of this potential increase in β_{Tmax} as A is reduced toward the limit of 1, which motivated the earlier study by Peng & Strickler. As will be shown Section III, the values for κ and β_N used in Eq. (2) would change substantially, and increase the value for β_{Tmax} beyond the indications of a constant β_N, despite large increases in q_j. That the value of q_j can increase as $A \rightarrow 1$ can be seen in Figure 2 below.

In Figure 2 are plotted the magnetic flux surfaces and representative field lines near the edge of Tokamak, ST, and Compact Toroid (CT) plasmas, based on MHD equilibrium

calculations. It is seen that as A is reduced from the Tokamak, the magnetic field line configuration is altered qualitatively. The outboard field line of the ST, because of the stronger poloidal field than the toroidal field, and comparable toroidal and poloidal circumferences, introduces only a fractional toroidal rotational transform, making only a modest contribution to q. The inboard field line, however, because of the very short toroidal circumference compared to the poloidal circumference, introduces many toroidal rotations within the inboard poloidal circumference. The net result is that q can be raised to ~10 in the ST plasma without relying on shaping of the plasma cross section or decreases in plasma current. Higher safety factor q is generally expected to improve stability of MHD modes in the plasma.

The projected increase in MHD stability and β_{Tmax} can be understood from this magnetic configuration. The ST combines short field line (due to high pitch angle relative to the horizontal plane) and bad curvature (where the field line "bulges" away from the plasma core) in the plasma outboard region, with long field line (due to low pitch angle) of good curvature (where the field line "bulges" toward the plasma core) in the plasma inboard region. In contrast the Tokamak and the CT have comparable lengths for field line of good or bad curvature. Increased dominance of good field line curvature leads to MHD stability at higher pressure gradient for a given magnetic field, i.e., higher plasma β.

III. Maximizing β and MHD Stability

The earlier calculations of free-boundary MHD equilibrium showed that the plasma naturally increases κ when A is reduced toward 1. This can be seen in Figure 3, κ increases naturally from 1.4 to 2 when A is reduced from 2.5 to 1.5, using only a vertical (roughly dipole)

magnetic field to control the plasma radial position. Further, to maintain $\kappa \sim 2$ for the same range in A (requiring shaping coils to produce quadrupole poloidal field), the toroidal and the poloidal magnet currents (in Ampere-turns), I_{TFC} and I_{PFC}, decrease by an order of magnitude to below I_p for fixed I_p and q_{edge} (~2.5) values. Reduced currents in magnets reduce the cost of magnetic fusion devices.

Since I_{TFC}/I_p scales roughly like $(I_p/aB_T)^{-1}$, the value for (I_p/aB_T) as high as 7 MA/m·T is calculated, leading to $\beta_{Tmax} \sim 20\%$ according to Eq. (2). Further, the value for S ($= q_j I_p/aB_T \approx 5\beta_N\kappa/A$) as high as ~ 20 MA/m·T is also indicated, which is substantially higher than the maximum achieved so far for the tokamak[23].

This potential of achieving high β_T at moderate q_{edge} was verified recently in the START experiment.[2,24,25] Figure 4 collects the β_T values observed in START as function of the normalized current I_p/aB_T, which shows a record of ~ 40%, corresponding to $\beta_N \sim 6$, using Eq. (2). The value of β_N therefore can exceed 3. It should be noted that in this case the volume average beta $\langle\beta\rangle$ is ~ 15%, indicating a strong increase of the average B over B_T (by ~70%) due to strong poloidal field toward plasma edge and enhanced toroidal field in the plasma core.

That the plasma also showed good energy confinement[26] can be seen in the camera image (Figure 5), which corresponds to $A \approx 1.35$, $I_p \sim 0.25$ MA, $q_{edge} \sim 3$, and $\kappa \sim 1.8$. The relatively sharp plasma edge, seen by the narrow thickness of H_α light, indicates a substantial edge density pedestal with a significant electron temperature (~ 50 eV) and H-mode confinement, consistent with more detailed analyses.[26] An energy confinement time in the range up to 5 ms is indicated for such plasmas, and compares favorably to tokamak H-mode scaling expressions. This is consistent with the high value for S (~ 18 MA/m·T).

That stable high β_T can be possible also at high q_{edge} (~10) can be seen in calculations carried out recently for NSTX.[10] Figure 6 shows that such a high q_{edge} value would limit β_{Tmax} to ~ 24% and β_N ~ 4.5, while producing a substantial bootstrap current fraction of f_{BS} ~ 41%, if the stabilizing effects of a nearby conductor wall is not assumed. However, assuming a full implementation of wall stabilization of low-n MHD instabilities,[27] β_{Tmax} can increase to ~ 40% with β_N ~ 8 and f_{BS} ~ 71%. Thus, higher β_N values at higher q_{edge} are theoretically possible if wall stabilization of MHD instabilities can be successful.

Larger bootstrap current fraction reduces the need for external noninductive current drive and the operating cost of future steady state ST devices. Near-term research on NSTX and MAST will test the physics basis for these calculations and improve understanding of MHD stability of ST plasmas.

Plasma disruptions have been observed more recently in START after nearby divertor coils are installed, which limited the vertical clearance for the plasma.[28] It is therefore of interest to study the potential impact of disruption on the ST device. A key consequence of disruption in Tokamak is the induced non-symmetric "halo" currents in the conductors in the vicinity of the plasma, and the large mechanical loads associated with such currents.[29,30] “Halo” currents resulting from large internal reconnection events (IRE) and forced vertical displacement events (VDE) were measured recently on CDX-U.[5] Figure 7 shows typical results of such currents, indicating that only a relative small halo current (< 5% of plasma current before the VDE) was induced.

The mechanical impact of plasma disruption also depends on the degree of non-symmetry in the distribution of the halo current in the inboard wall of the device. Since the inboard wall is relatively slender in configuration, the induced halo current tends to become symmetrized. This

is seen in recent calculations by Pomphrey et al. (1998)[31] and shown in Figure 8. By introducing non-symmetric halo current at about 1 m apart on the center post wall, a strong symmetrization of the wall current is seen as A is reduced from above 2 for tokamak to about 1.3 for NSTX. These results are encouraging for future ST devices of high plasma current, and will be studied in the present ST experiments.

Other very important issues of stability include the so-called "neoclassical tearing modes (NTM)," which are nonlinearly driven unstable by a deficit in bootstrap current within the tearing mode island.[32,33] The NTM has recently been observed in large nearly collisionless tokamak plasmas to limit plasma beta in case of significant pulse lengths.[34] In the case of ST plasmas, it was recently theorized[35,36] that the NTM could become less unstable due to the expected presence of large Pfirsch-Schlüter currents (or the Glasser term) at high beta.[37] More theoretical work and new experimental tests will be needed to understand the physics of NTM in ST and develop methods for stabilization, such via electron cyclotron wave and lower-hybrid wave heating and current drive.[33,38,39] Stabilization of NTM will permit β_{Tmax} to be approached and maintained for long durations.

IV. Reducing Microinstabilities and Increasing Flow Shear

Strong shearing rates in E×B plasma flow has been suggested[40,41] to stabilize micro-instabilities with comparable growth rates and improve tokamak plasma confinement when the internal transport barrier (ITB) is observed. It is therefore of interest to estimate the low-A trends of micro-instabilities and the E×B plasma flow.

A theoretical indication of the micro-instability trend was recently produced by Rewoldt et al.[42] As shown in figure 9, the instability growth rates for the collisionless trapped electron η_i modes and the kinetic ballooning modes on the 70% flux surface can become zero as the plasma A is reduced from 2.5 to 1.5, for plasmas betas up to the limit in the absence of wall stabilization. It is determined that the geodesic curvature of the magnetic configuration, averaged over the structure of the micro-instability, changes from destabilizing to stabilizing as A is reduced to the lower value. The magnitude of growth rates over the higher A values are of the order of 10^5 s^{-1}. More recent work by Kotschenreuther et al.[43] indicated such instabilities would remain at lowered growth rates for higher beta values that require a nearby conductor wall for stabilization.

Such instability growth rates are to be compared with the E×B flow shearing rates anticipated in high beta ST plasmas. An estimate was made by Synakowski[44] for a wall-stabilized high-beta NSTX plasma and shown in Figure 10. In the absence of momentum input, the plasma diamagnetic drift driven by the pressure gradient ∇p alone can be as high as ~ 10^6 s^{-1}. When a momentum is also introduced to achieve a Mach number of 0.5 in the plasma, the shearing rates can increase further to a factor 4 higher.

It is therefore anticipated that the flow-shearing rate in a high beta ST could be much larger than the prevailing micro-instability growth rates. Opportunities therefore exist in NSTX and MAST to study the ITB properties under such conditions. A key challenge for the ST plasma investigation, similar to the tokamak plasma,[45] is the compatibility of plasma profiles determined by ITB with those required for stable β_{Tmax}. Simultaneous high beta and confinement could lead to high potential for achieving high fusion gain in small-size ST devices.[43]

V. Noninductive Current Drive

Noninductive current startup and maintenance will enable dispensing with the inboard solenoid magnets and dramatically reduce and size and complexity of future fusion energy producing ST devices. In case of noninductive startup, the HIT experiments[3,4,46,47] have demonstrated the production of ~200 kA plasma current for ~ 8 ms by helicity injection alone. As shown in Figure 11, coaxial helicity injection (CHI) converts dc voltage and current externally applied in the radial direction, to toroidal current within the vacuum chamber via reconnection of the magnetic field lines. The NSTX device is designed to investigate the CHI physics for plasma currents up to 0.5 MA, with improved plasma control and diagnostics. The very first test of CHI in NSTX[48] resulted in limited toroidal current (~ 25 kA) for ~ 40 ms.

For noninductive current maintenance, high harmonic fast wave (HHFW) was recently suggested by Ono et al.[49] to take advantage of the large dielectric constant and small magnetic field of the ST plasma. In this case the wave absorption rate, $k_{\perp i}$, is theorized to scale as:

$$k_{\perp i} \sim n_e / B^3 \sim \varepsilon / B, \qquad (3)$$

where $\varepsilon = (\omega_{pe}/\omega ce)^2 \sim 100$ for high beta ST plasmas. Strong absorption of HHFW by the NSTX plasma is therefore anticipated. More recently numerical calculations[50] using the RANT3D full-wave code were carried out for NSTX, and as shown in Figure 12. By assuming average density and temperature of 5×10^{13} cm^{-3} and ~ 1 keV, respectively, a complete absorption of 6-MW power in a key component ($n_\phi = 6$) of HHFW is estimated, leading to a driven current of ~ 0.5 MA. A full wave theory[51] was more recently used to calculate that strong absorption remains

even if all collisionless electron dissipation mechanisms and ion cyclotron damping over many harmonics are accounted for.

While such calculations are encouraging, it identified a strong need to control the launched toroidal mode spectrum of HHFW in NSTX. A 12-strap antenna system with 6 controllable phases are designed and installed in NSTX[11] for this purpose. Initial measurements on CDX-U[52] indicated high plasma loading at modest HHFW power and evidence of electron heating.

The very high dielectric constant of the plasma prevents the propagation of the Electron Cyclotron Wave (ECW) in the plasma. However, such plasmas provide an efficient medium for propagation and absorption of the Electron Bernstein Wave (EBW). Recently Ram & Bers[53] identified the prospect for efficiently converting outboard launched ECW to EBW in high beta plasmas, thereby avoiding the low density limit that constrains ECW to the condition of $\varepsilon = (\omega_{pe}/\omega_{ce})^2 \sim 1$. By taking advantage of the extreme close proximity (~ 1 mm) of the ECW cut-off layer and the EBW evanescent layer on the inside of the upper hybrid resonance, high conversion efficiencies are indicated for frequencies up to $\sim 8\omega_{ce}$. It is also theorized that by launching the ECW at an oblique angle to the outboard magnetic field, efficient current drive could also become possible.

Recent measurements[54] of the EBW emission from the core of the CDX-U plasma suggested that this wave conversion mechanism could be efficient.

VI. Scrape-Off Layer Configuration

High-performance plasmas in compact configurations introduce intense plasma and energy fluxes on the plasma facing components. Progress of the ST physics investigations therefore depends on dispersion of such fluxes over large wall areas.

Such an opportunity can be expected in inboard-limited NSTX plasmas, the scrape-off-layer (SOL) configuration of which is shown in Figure 13. It is seen that the outboard SOL is naturally divided into two regions: an inner region connected to the inboard limiter (A) with a thickness of ~ 2 cm at the outboard mid-plane, and an outer region connected to the divertor (B) without using divertor coils to form ×-points near the plasma edge. As shown in Figure 14, this SOL is unusual in its very large area expansion factor (= 10–30 for region A and = 5–10 for region B).

In addition, a large magnetic mirror ratio (~4) with a strong field-line curvature is also calculated for this SOL, which is subject to the instability of pressure-gradient-driven MHD interchange modes.[55] The requirement of marginal stability of this MHD mode leads to a lower bound for the SOL thickness, before less virulent non-ideal instabilities have a chance to become important in determining the cross-field transport. Such a MHD stability criterion can be written for the minimum SOL thickness in plasma pressure:

$$\lambda_{pmin} \sim 0.002\,(n_i\,T_i / R_0)\,(L_c / B_T)^2 \qquad (4)$$

Here n_i and T_i are the ion density (10^{20} m^{-3}) and temperature (keV) at plasma edge, L_c is the length (m) of the SOL field line between the plates, and f_c ~ 1 for double-null and ~ 2 for single-null and inboard-limited plasmas. It is of interest to note that because of large L_c (~ 16 m), small R_0 (= 0.85 m) and B_T (~ 0.3 T), λ_{pmin} can be ~ 5 cm on the mid-plane of inboard limited NSTX

plasmas with $n_i \sim 0.1\times10^{20}$ m^{-3} and $T_i \sim 0.1$ keV. The SOL "foot print" on the inboard limiter and divertor plates in this case could be as wide as 50 cm.

For double-null plasmas the lower bound λ_{pmin} is expected to be about 1/4 of the inboard limited plasma because of the reduced L_c. Further, the SOL area expansion factor in this case is expected to be much reduced because the magnetic mirror ratio along the SOL is much reduced from the inboard limited case. The resulting SOL properties are therefore anticipated to be closer to those for the larger aspect ratio tokamak.

The parametric dependence of this lower bound on SOL thickness can be tested in NSTX and MAST. NSTX recently produced the basic plasma shapes and SOL configurations (see Figure 15), which will be used in these investigations.

VII. Future Potential in β and Confinement

In future power-producing applications, it would be highly desirable to have self-sustaining ST plasmas of very high stable beta and good confinement. Very high beta ST plasmas have recently being identified by the ARIES Team of UCSD and other studies[56,57,58,59] providing a theoretical prospect for future attractive fusion power. An example of this is shown in Figure 16, having exceptionally high $\kappa = 3.4$, $\beta_T = 56\%$, $\beta_N = 8.2$, $\langle\beta\rangle = 42\%$, and $f_{BS} = 99\%$, and high q values (>4) throughout the plasma core. The very high $\langle\beta\rangle$ value is a result of strong diamagnetism of the plasma with high poloidal β, leading to a strong magnetic well near the magnetic axis.

It can be seen that such a plasma will require a full set of simultaneous optimization and control. These include vertical stabilization of ST plasmas with such a high κ; full stabilization

of low-n modes using a nearby conductor shell; stabilization of the neoclassical tearing modes at higher q surfaces where magnetic shear is high; the very hollow current profile of extremely low internal inductance that satisfies stability; fully aligned plasma current with the bootstrap current excepting 1% current near the magnetic axis; a pressure profile that produces the tailored bootstrap current profile; and a set of plasma particle and heat diffusion coefficients and fueling and heating profiles that maintain the desirable pressure profile.

To investigate the scientific feasibility of each of these conditions, and eventually to investigate all of these conditions simultaneously, will require $\kappa > 3$ capability, in addition to a large number of tools for plasma manipulation and measurement. The above requirements are similar in nature to what have been proposed and tested for the Advanced Tokamak research in recent years.[19,20]

Present-day ST experiments provide good opportunities for clarifying the ST confinement capabilities (see, Figure 17). Here a number of popular tokamak confinement scaling expressions are used to calculate the confinement times τ_E for the START, NSTX/MAST, and a next-step device with a plasma current in the range of 10–18 MA, assuming a β_T in the range of 20–25%. It is seen that roughly an order of magnitude uncertainty is encompassed by the tokamak scaling expressions when applied to the ST plasma. Using the START confinement times[26] as indications, and staying in the mid-range of the uncertainty, a τ_E in the range of 20–40 ms is suggested for NSTX/MAST. Confinement studies in present-day ST experiments will therefore help narrow this range of scaling expression for more reliable application to future ST plasmas.

The production and maintenance of ITB in tokamak plasmas have allowed the plasma transport locally to approach the neoclassical limit. Since the full neoclassical confinement

time[60] is substantially above the scaling indications, investigation of ITB in ST plasmas will help improve our understanding of plasma transport when local neoclassical confinement is approached. More recently theory[61] has suggested that the ion neoclassical orbits can be "squeezed" by large gradients in electric field (sheared plasma flow), potentially leading to reductions in neoclassical transport. At present with a lack of concrete confinement database, the scaling projections suggest that a plasma current in the wide range of 10–18 MA would be needed to have a reasonable capability to test conditions for substantial Q (~1–10) in D-T plasmas, or equivalent Q in D-D plasmas. Results from NSTX/MAST in the near future will hopefully narrow the ranges in plasma current and confinement required for such tests.

VIII. Relationship Between ST and CT

As indicated in Figure 2, the outboard magnetic configuration of the ST begins to approach that of the Spheromak (assuming no axial current) and the FRC plasmas. As seen in Figure 3, this approach would be particularly close if the aspect ratio A is reduced to the extreme lower practical limit of 1.05–1.1, when I_{TFC}/I_p could approach 0.1 (see, Figure 3). Capability to explore this limit in A has been incorporated in Pegasus[15] and TS-3,[62,8] for example.

In the case of TS-3, it is found that a value of $I_{TFC}/I_p \sim 0.2$ is required to stabilize the tilt instability of Spheromak, which corresponds to a q_{edge} of ~ 3 at the extreme low A limit. By merging two spheromaks with opposing toroidal field, FRC with relatively hot ions (~10^2 eV) have been produced in TS-3.[63] When a toroidal field is also applied at the same time, a high-beta FRC-like ST configuration with strongly depleted internal toroidal flux was formed and maintained for a short time scale. An average $\langle\beta\rangle > 60\%$ was also observed initially. Although

such plasmas are likely dominated by collisions, radiation, and cold electrons (~10 eV), observation of such an equilibrium is consistent with the prospect that the extreme low A provides a possible way of bridging the ST and CT plasmas.

A comparison of FRC and ST plasmas was also carried out recently in the Rotamak-ST device.[64] There it was seen that the ST configuration provides some increase in stability and confinement over the Rotamak-FRC plasma. A ST near this A limit appears to be amenable to the Rotamak current drive technique, and should therefore be tested at parameters above the radiation barrier.

Coaxial helicity injection (CHI) represents another example where the CT physics is transferred and somewhat improved when a toroidal field is applied to a otherwise spheromak-like plasma. A significant change due to the presence of toroidal field, however, in the reconnection mechanism during CHI has been identified recently.[65] Investigation of the scientific connections between the ST and CT plasmas in common experiments would therefore contribute to exploring the lower boundaries in A of the ST plasmas, and derive new understanding of toroidal plasma.

IX. ST Parameter Space and Development Steps

The discussions in this paper have made it clear that the ST is likely to extend the plasma parameter space beyond the tokamak and the CT plasmas, in directions of importance to future fusion energy applications and in ways that could alter our evolving projections of the future of magnetic fusion energy. The particular ST science promise and fusion energy future can be summarized in Table 1 below:

It is seen that the high leverage of eliminating the inboard solenoid for the ST configuration is associated with new challenges. These include a full reliance on the physics of noninductive startup, a single turn center leg of the toroidal field coil to avoid insulators exposed to high neutron dose, and an increased recirculating power for a future power plant due to the resistive dissipation of the center leg.

However, ST plasmas appear also to have the prospect of introducing potentially new science and promising new ranges of parameters not available so far in the Tokamak and CT systems. Some of these possibilities are listed in Table 2 below:

It is seen that these scientific features would be new to magnetic fusion energy research if high-temperature collisionless ST plasmas can become a reality. The possible physics consequences, if borne out, could potentially change our perception of future fusion power based on magnetic confinement. For example, the need to maintain large ρ_i* for confidence in extrapolations to larger size ST devices may lead to the choice of smaller A and B_T, rather moving toward higher A, larger B_T and much reduced ρ_i*.

Assuming that the physics discussed in this paper were substantially borne out by experimental tests, relatively affordable steps would become available for the development of fusion energy science. As depicted in Figures 17 and 18, a next-step experiment beyond the NSTX/MAST could have a major radius of about 1.2 m and a plasma current in the range of 10–18 MA. This device would focus on making two orders of magnitude improvements in plasma fusion parameter $Tn\tau_E$ and the equivalent fusion amplification in limited pulse lengths. Given success in physics at this level, a steady state embodiment of a similar-size device would then provide reduced-cost opportunities for the technology intensive R&D aimed at increasing capabilities to handle copious neutron wall loading and heat fluxes at the level of future power

producing systems. ST experimentation in the next several years should therefore aim to provide a preliminary determination on many of these potentially attractive properties.

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Table 1. Relationship between the ST science promise and future magnetic fusion energy

Science Promise	⇔	**Energy Future**
High plasma pressure, low magnetic field	↔	Lower device and magnet costs
Suppressed turbulence, improved confinement	↔	Small unite size
Dispersed plasma exhaust	↔	Reliable first wall
Self-sustaining plasma current	↔	Reduced operating cost
Startup without solenoid	↔	Simplified, compact configuration
Design Feature	⇔	**Challenge**
Startup without solenoid	↔	Noninductive startup physics
	↔	Single-turn center conductor
	↔	Recirculating power, conductor lifetime

Table 2. Possible new physics regimes introduced by the ST plasma

New Scientific Features	**⇒**	**Possible Physics Consequences**
Strong magnetic well (~30%)	→	Guiding center orbit compression
Near-omnigenous orbits		Reduced neoclassical transport
Sound speed ~ Alfvén speed	→	Mass effect in dynamic MHD equilibrium
		Mass effect on stability and turbulence
Fast ion >> Alfvén speeds	→	New class of toroidal Alfvén Eigenmodes
		New class of wave-fast-particle instabilities
Large $\rho_i^* \sim 0.03 - 0.01$	→	Effects on H-mode pedestal thickness
Larger $\rho_{fast\text{-}i}^* \sim 0.3 - 0.1$		Effects on neoclassical transport
Very low ℓ_i (~0.1–0.2) high β	→	Enhanced neoclassical loss in core
		Reduced neoclassical loss at edge
		Minimized poloidal flux and helicity content

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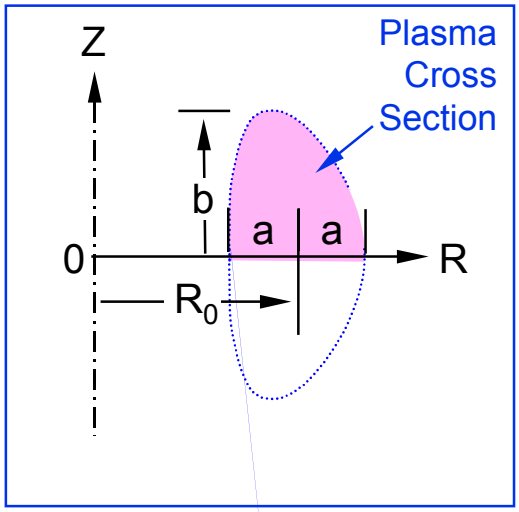


Figure 1. Cross section of toroidal plasma

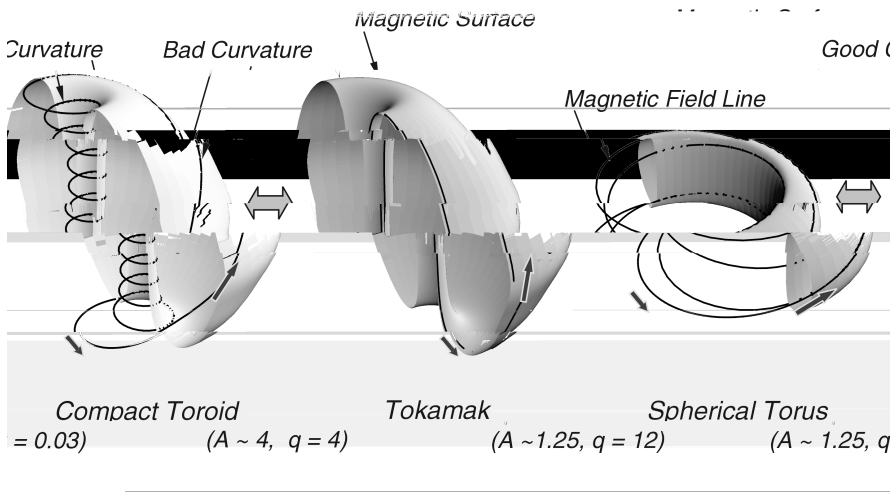


Figure 2. **The magnetic configuration of the Spherical Torus plasma as related to the Tokamak and Compact Toroid plasmas**

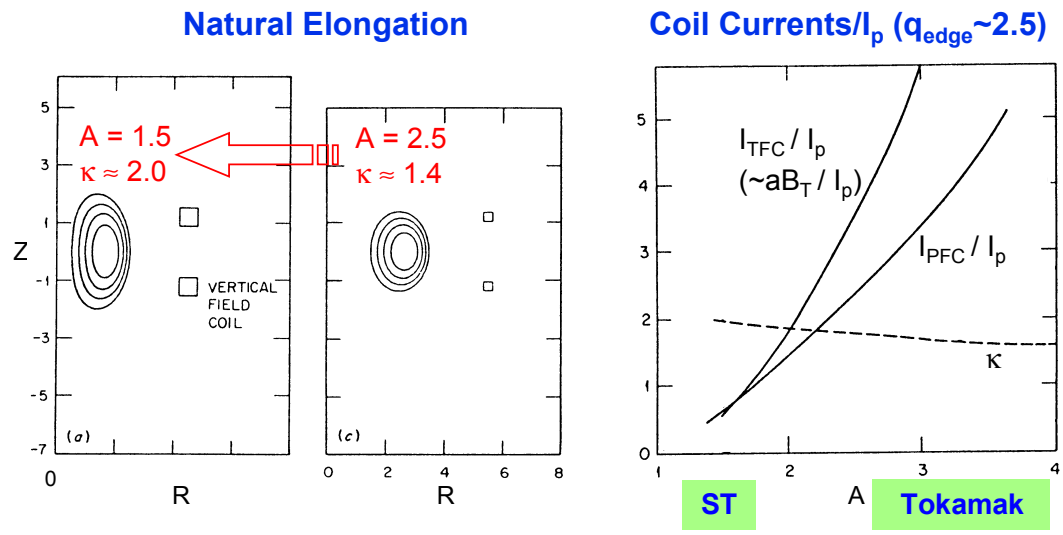


Figure 3. Results from free-boundary MHD equilibria for different aspect ratios. The left-hand figure compares the plasma cross section as A is reduced from 2.5 to 1.5. The right-hand figure shows the coil currents required to maintain κ near 2, as a function of A.

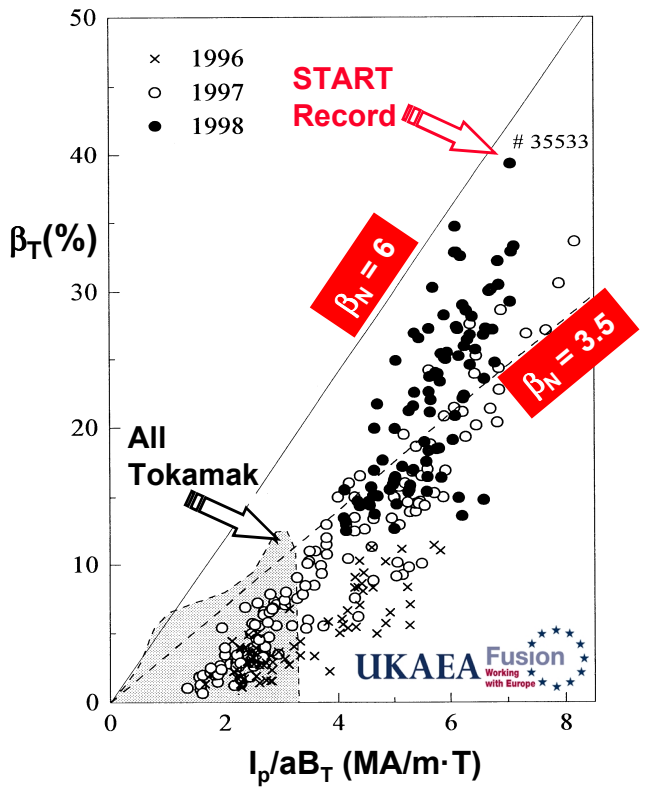


Figure 4. Average toroidal beta β_T measured in START during 1996-1998 as a function of I_p/aB_T.

(Courtesy of A. Sykes & START Team, U.K.)

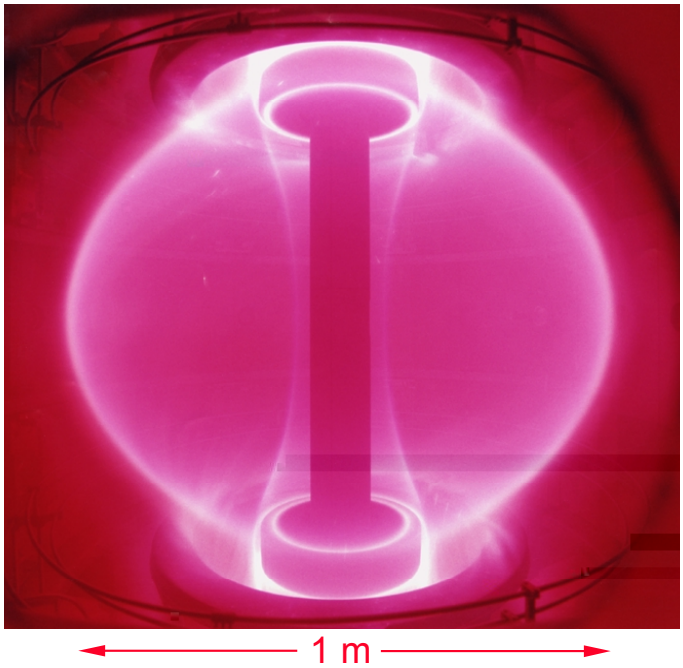
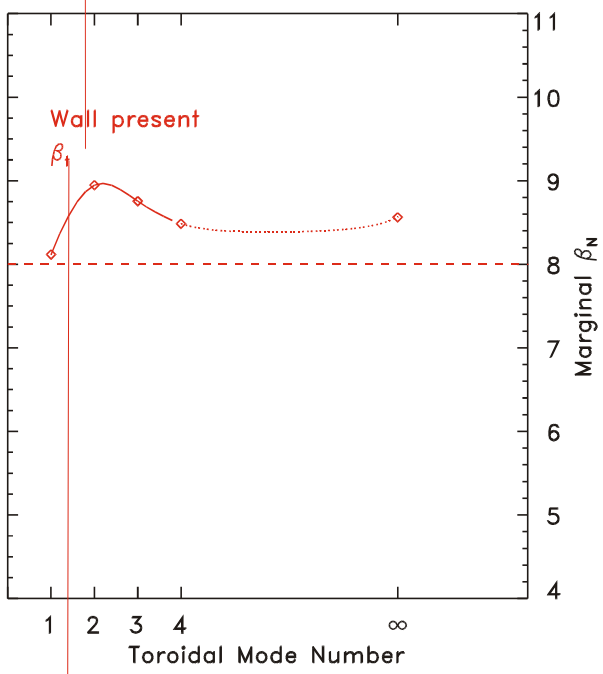


Figure 5. Fast video picture of plasma in START at high β, with $I_p \sim 250$ kA, $q_{95} \sim 3$, and $\kappa \sim 1.8$.



$_{edge}$ ~ 10 are assumed.

(Courtesy of M. Ono and CDX-U Team)

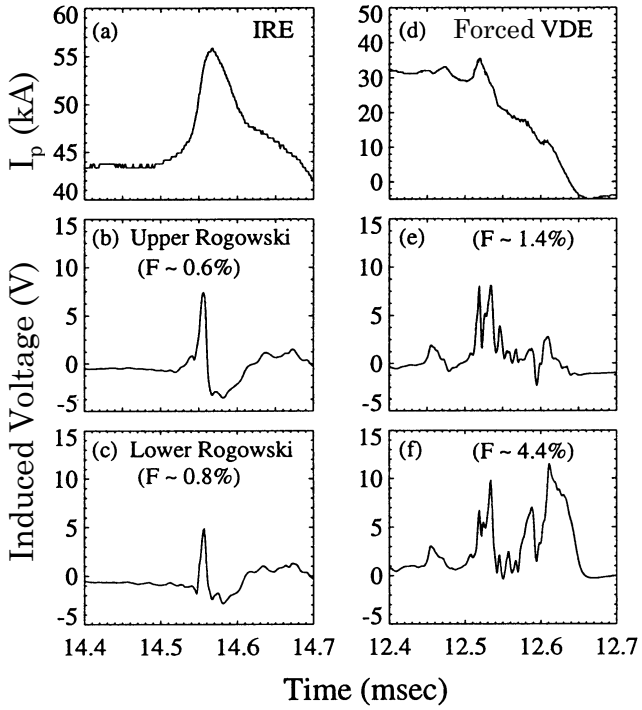


Figure 7. Plasma currents during and the voltages on the Rogowski coils near the upper and lower ends of the center post in CDX-U induced by an Internal Reconnection Event (IRE) and a forced Vertical Displacement Event (VDE).

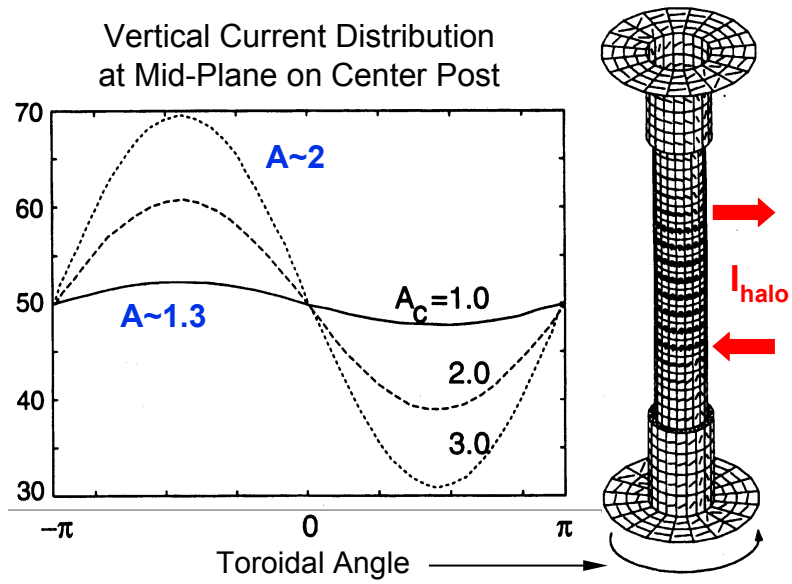


Figure 8. Distribution of current at the mid-plane of the center post for different aspect ratio plasmas, when a non-symmetric (1 + sinusoidal) "halo" current is introduced at locations 1 m apart.

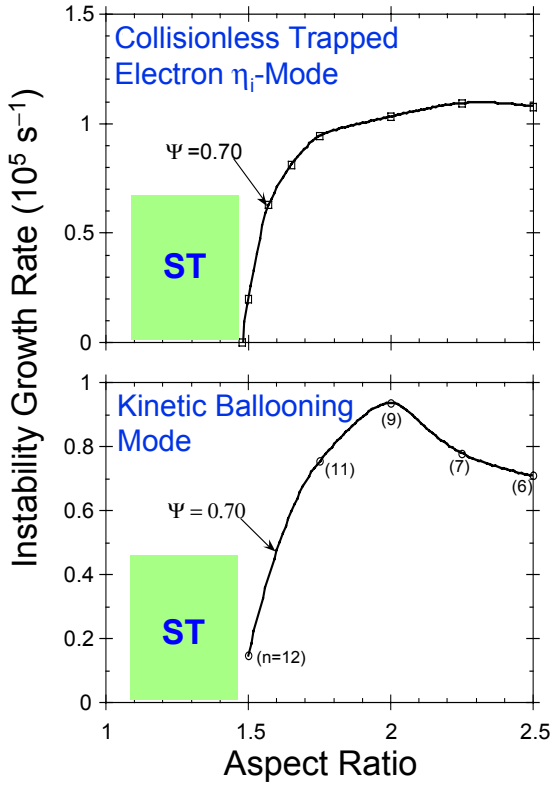


Figure 9. Growth rates of the collisionless trapped electron η_i-mode and the kinetic ballooning mode on the 70% flux surface in plasmas of different aspect ratios and near the first-stability regime limits.

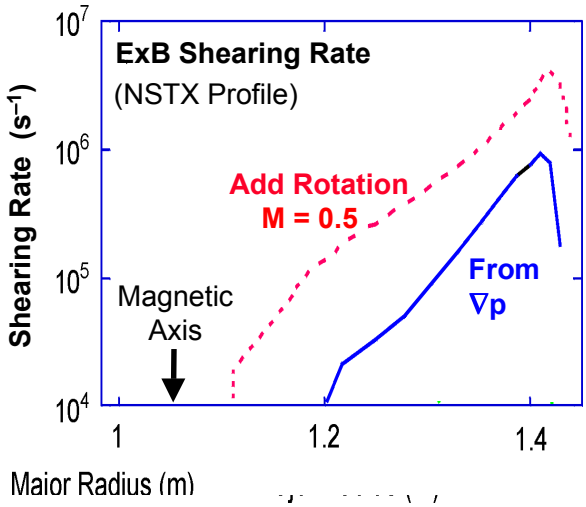
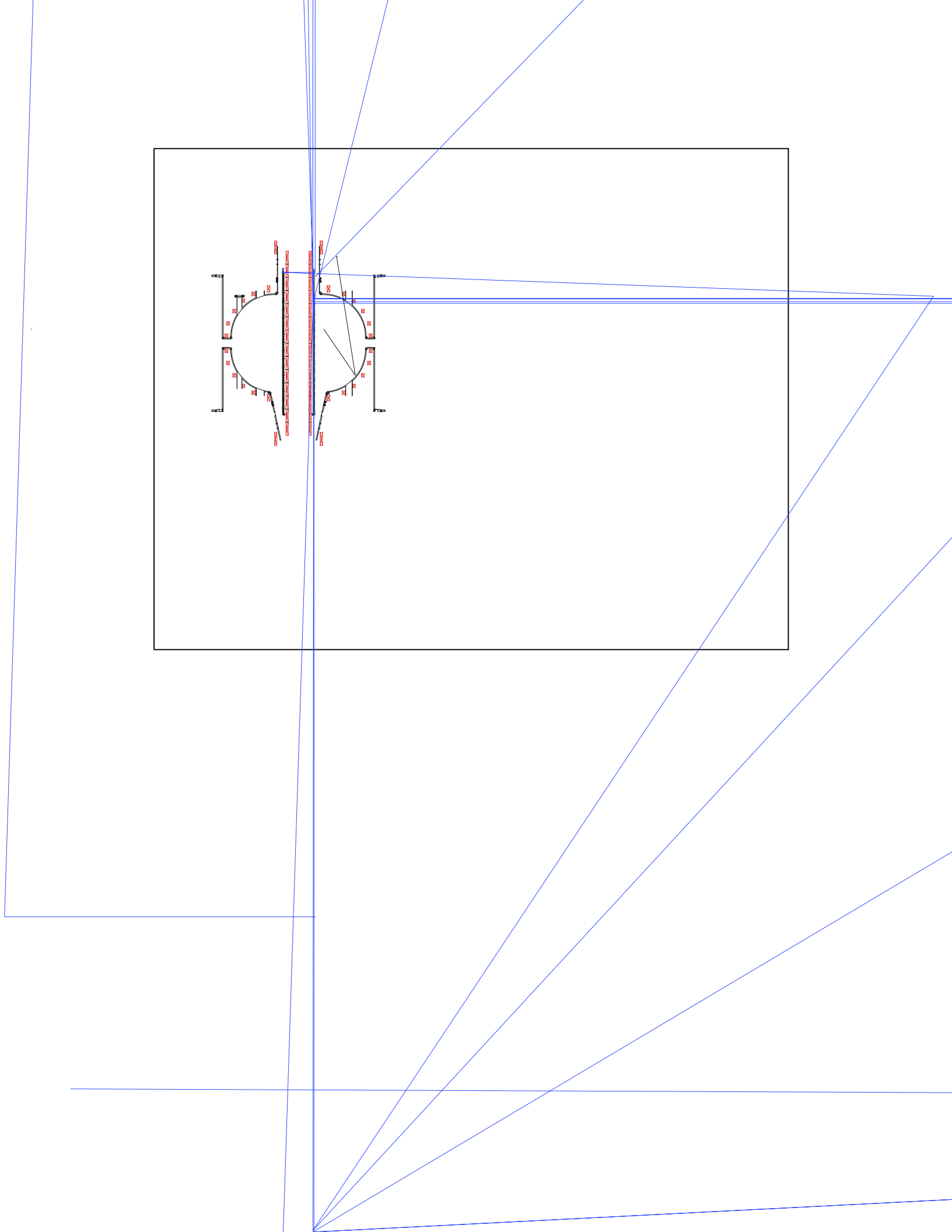


Figure 10. The E×B plasma flow shearing rates on the outboard mid-plane in high-β NSTX plasma.



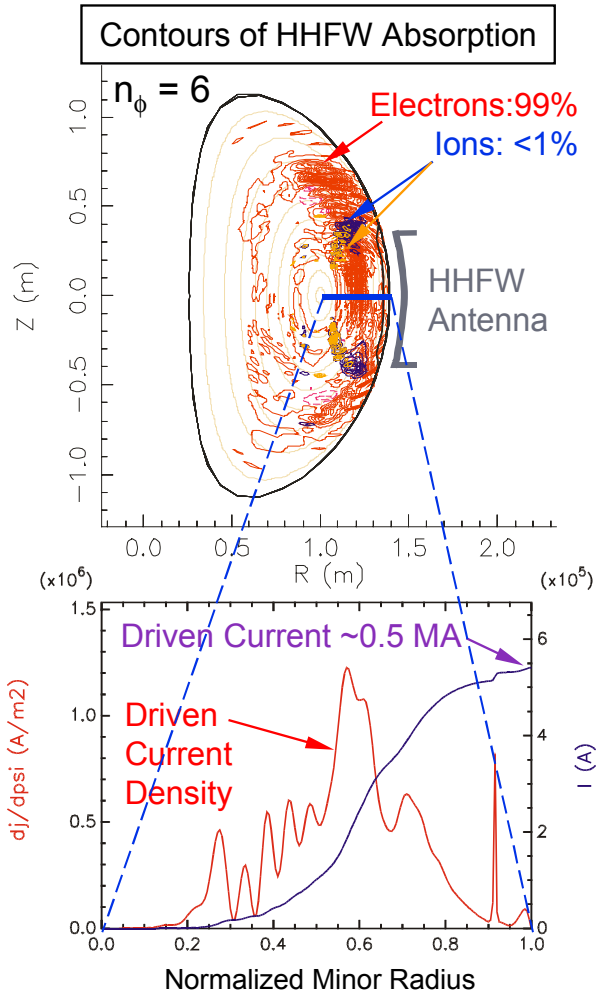
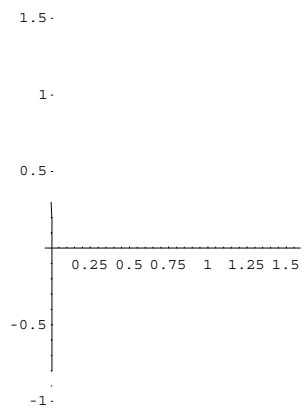
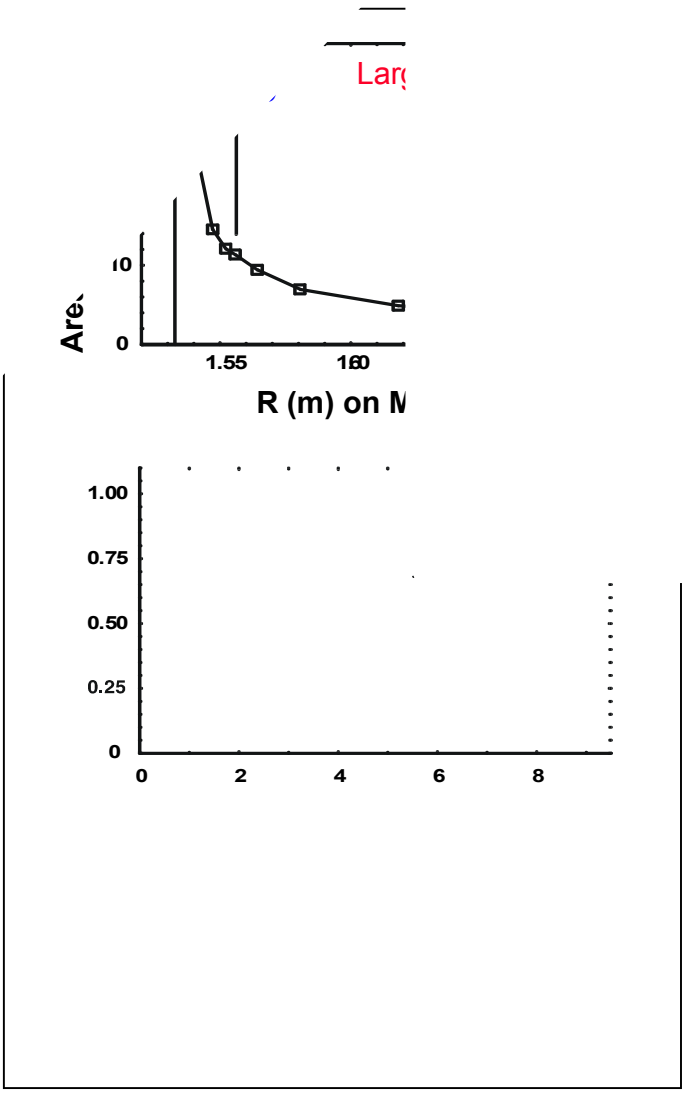


Figure 12. Contours of constant rate of absorption by electrons and ions of High-Harmonic Fast Wave with a toroidal mode number $n_\phi = 6$. The resulting driven current density and the integrated driven current is plotted as function of the normalized minor radius.

Scrape-Off Layer Geometry of Inboard Limited ST Plasma




Larg
Are
10
0
1.55
160
R (m) on M
1.00
0.75
0.50
0.25
0
0
2
4
6
8

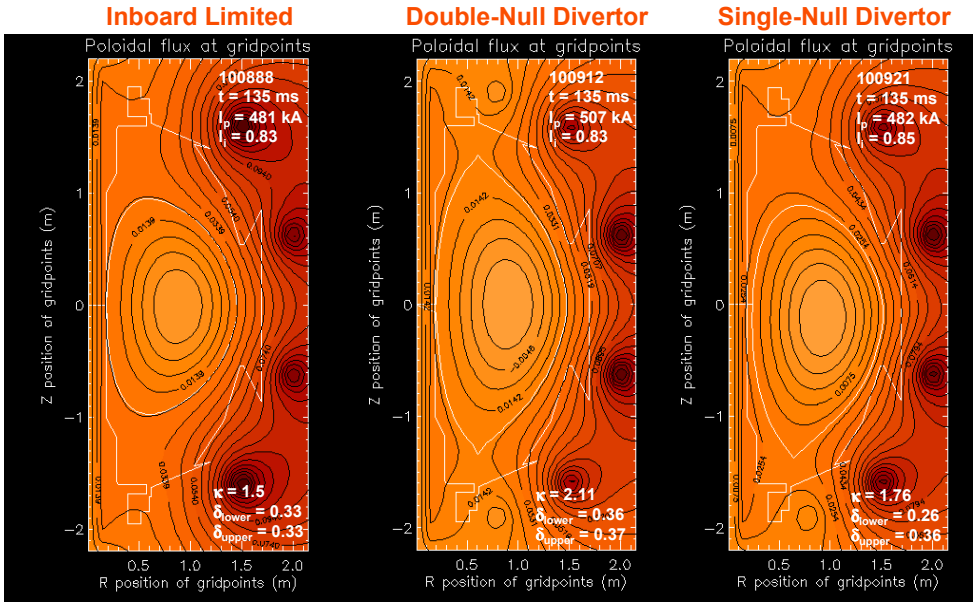


Figure 15. MHD equilibria for inboard limited, double-null diverted, and single-null diverted plasmas produced in NSTX, modeled with the EFIT code using magnetic measurements around the plasma.

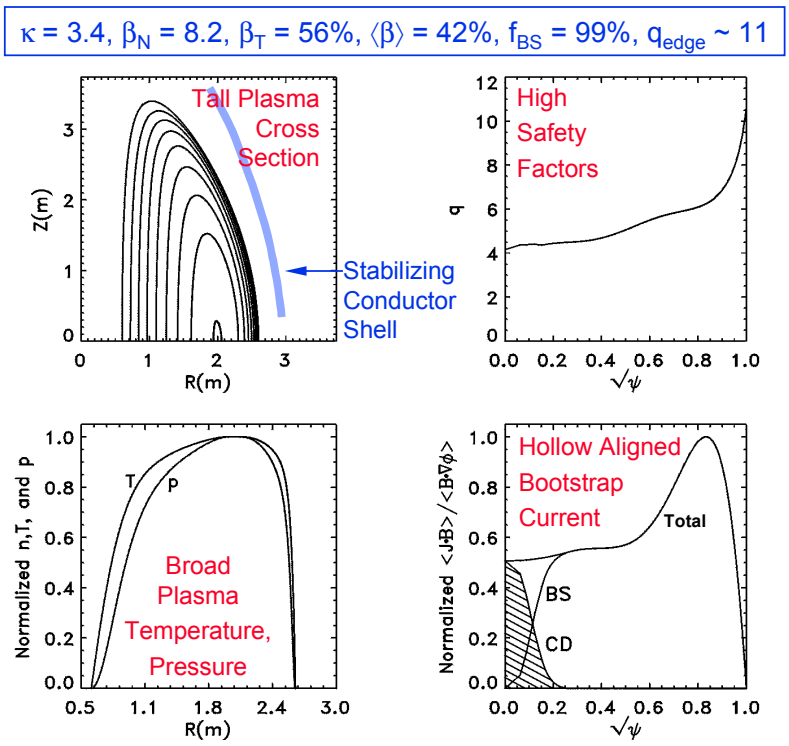


Figure 16. Theoretical predictions of very high β MHD stable ST plasma characterized by high κ, β_N, β_T, ⟨β⟩, q, and 99% self-sustaining plasma current. A nearby conductor shell is needed for stability of low-n kink modes.

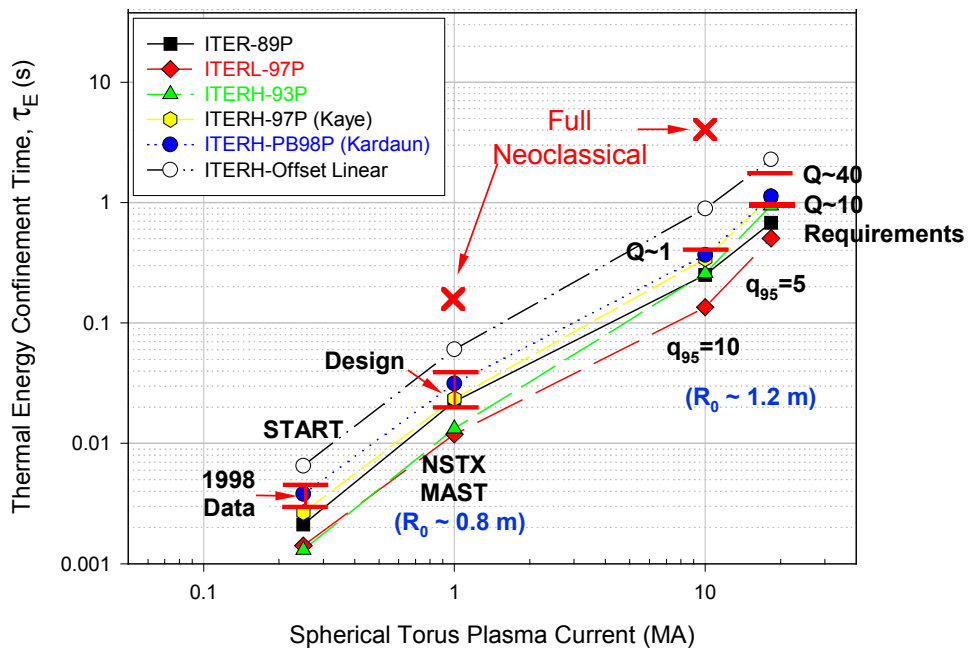


Figure 17. Thermal energy confinement times (τ_E) estimated for the START (R_0 = 0.3 m, I_p = 0.25 MA), NSTX/MAST ($R_0 \approx$ 0.8 m, I_p = 1 MA), and next-step ST ($R_0 \approx$ 1.2 m, I_p = 10–18 MA) plasmas according to the popular tokamak scaling expressions. Also indicated are the confinement times measured in START, used in the design of NSTX/MAST, required to achieve significant fusion energy amplification (Q) in the next-step ST, and assuming the full neoclassical (electron and ion) values.

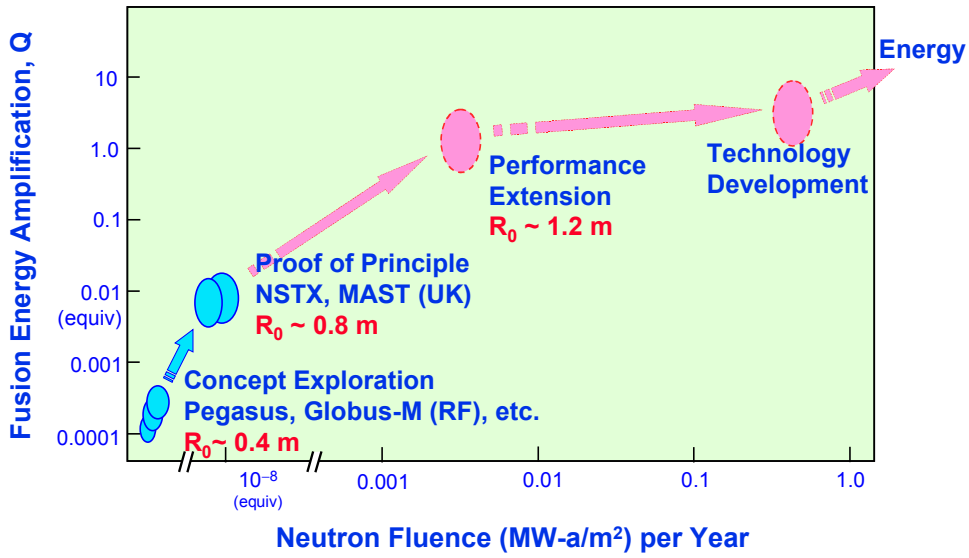


Figure 18. Possible next-step devices beyond the present ST experiments: A limited-pulse "performance extension" experiment to test the physics of fusion grade plasmas equivalent to achieving substantial fusion amplification (Q), and a subsequent steady state "fusion energy" device of similar size to develop technology.