Accelerated compact fusion development and innovations leveraging spherical tokamaks

Executive Summary:
The integrated U.S. spherical tokamak (ST) research programs and facilities should be exploited and expanded to accelerate the development of compact tokamak fusion. ST research aims to further optimize the tokamak configuration by leveraging natural innovations in magnetic confinement fusion physics that occur as aspect ratio is reduced. Critical research plans across several devices focus on filling the most important gaps in magnetic fusion research. NSTX-U will be the most powerful ST device in the world program. It will enable new world-leading confinement, stability, control, and non-inductive plasma current sustainment experiments. Core and scrape-off layer (SOL) modifications in the LTX-β device with low recycling walls and beam heating will inform whether NSTX-U should transition to liquid lithium plasma facing components. Experiments on the Pegasus device will guide the efficacy of helicity injection for current formation at low aspect ratio and for extrapolating to larger STs. The unique ST parameter regime can be exploited to both improve fusion performance and to expand toroidal confinement and stability predictive capability allowing for optimizing the aspect ratio of compact next-step devices. This strategic thrust would accelerate fusion development, strongly support the full spectrum of U.S. fusion institutions including major laboratories and universities, and garner substantial international collaboration. These aspects of ST research were presented at community-led Magnetic Fusion Research Strategic Directions Workshops in Madison, WI (July 2017, stated as "Madison" below) and Austin, TX (December 2017, stated as "Austin" below), with material referenced in this document. Innovations highlighted at these meetings including the development of high temperature superconducting (HTS) technology for magnetic fusion with high field and current density and disruption-free, continuous tokamak plasma operation can be highly leveraged by U.S. ST research. A compact DT facility with a mission to retire risks to constructing DEMO should be considered as the next-step once the present generation of upgraded ST devices fills key research gaps.

ST Research aimed toward the community vision of net energy production
The tokamak is the leading toroidal magnetic fusion configuration with sufficiently high energy confinement to project to net energy gain. The dominant long-term vision for the U.S. magnetic fusion program stated in several summary presentations at Madison [1-3] and Austin [4,5] is the goal of producing net electricity in a compact tokamak device typically called a Pilot Plant. The desire for this particular vision was independently substantiated at Austin in the coordinated talks by Ryan Umstattd (ARPA-E Deputy Director for Commercialization) and M. Tillack (UCSD) that showed results of a survey of utility managers, venture capital and investment groups, government representatives, and others that identified the demonstration of net energy gain as the key achievement needed to commit private funds to fusion energy [6,7]. Steps toward this goal will naturally include burning plasma physics research. The Greenwald discussion group at Madison summarized the community stance that the U.S. should include a deuterium-tritium machine in its plans (referencing four talks at Madison [8-11]) and the theme continued strongly in Austin summaries [12,13]. The role of the machine as discussed ranged from a U.S. burning plasma experiment (BP), to a copper coil-based Fusion Nuclear Science Facility (FNSF), to the full vision of a net electricity-producing Pilot Plant.

An encouraging aspect of the Greenwald discussion group that carried forward to Austin was the positive group consensus on several aspects of the device and research that the U.S. should pursue. The device aspects and research are directly compatible with and highly leveraged by U.S. ST research. These included the development of high temperature superconductors (HTS) [14] to increase the magnetic field (which helps all tokamak designs to increase fusion power), compact device design further facilitated by the increased current density of HTS magnets, plasma operation with high or full non-inductive current drive (NICD) fraction, and continuous disruption free operation. The elements with high group agreement also included attention to plasma facing component materials and innovative power handling solutions such as a resilient first wall solution (e.g. liquid metal wall). Substantial research and development continues to determine optimizations that leverage the natural advantages of low aspect ratio to reduce the size and cost of tokamak configurations and thereby make them more attractive for electricity production [8,9,15]. Achieving these goals will support a U.S. strategic position to "make fusion work". Studies on compact tokamak systems for fusion development have shown that the fusion power gain depends on at least 8 key parameters related to plasma physics and fusion technology. A key geometric parameter for toroidal devices is the aspect ratio A = R / a = major radius / minor radius, and lower aspect ratio tokamaks configurations, A = 1.5-2.3, may offer several advantages for fusion development. Low-A (A < 2) tokamaks have a reduced surface area to volume ratio and could provide high neutron wall loading at small major radius (R = 1 – 2m) in an FNSF for developing fusion nuclear components in parallel with burning plasma experiments. Low-A tokamaks to date have also observed a strong increase in energy confinement with reduced collisionality, ν, that differs from conventional aspect ratio and may therefore enable high fusion gain in more compact devices. For net electricity production, recent studies have shown that the very high current density, J_{WP}, and field strength potentially achievable using HTS tape and cable technology may lead to superconducting magnet Pilot Plants and Power Plants with optimal A ≈ 2 (Figure 1) [16]. Lower-A designs would significantly reduce cryostat volume and TF magnet mass per unit fusion power, potentially increasing overall mass power density (Figure 2) [15].

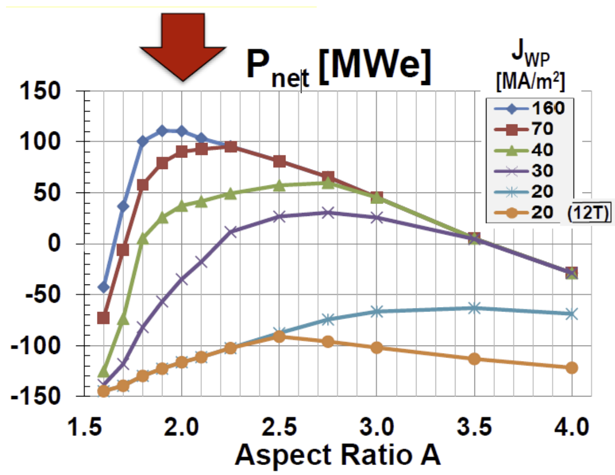

Figure 1: Net fusion power vs. device aspect ratio analysis showing that high current density, high T superconducting (HTS) cable motivates lower-A tokamak pilot plant designs (shown at Madison and Austin meetings in [8,16])

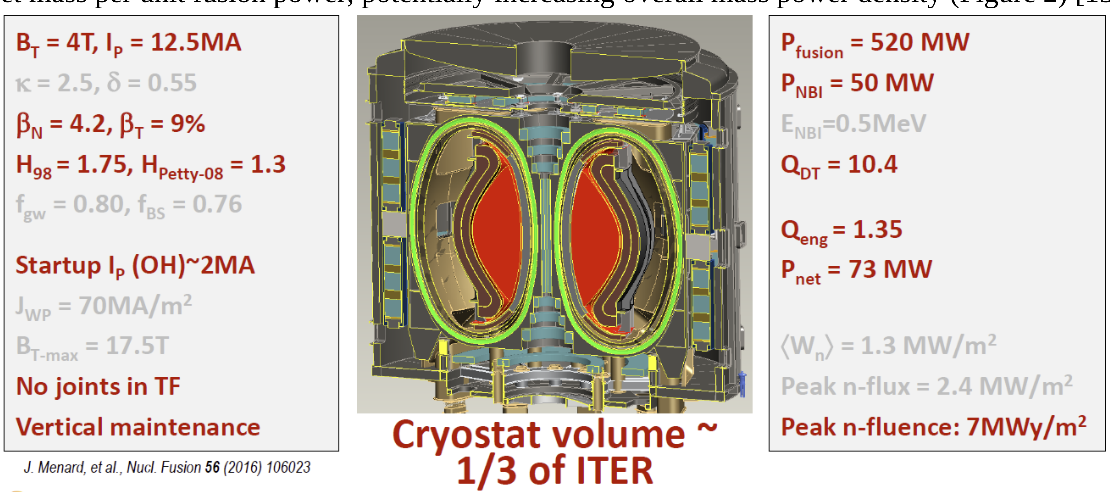

Figure 2: Compact HTS Pilot Plant designs with net electricity production and tritium self-sufficiency based on ST physics understanding are envisioned to be significantly smaller (1/3 the volume) than conventional aspect ratio burning plasma tokamaks (from the Madison meeting) [8].

Advantages of ST design driving research

The ST is an optimization of the tokamak device that leverages natural innovations in magnetic confinement fusion physics. With respect to toroidal plasma physics contributions, low-A tokamaks naturally access higher plasma beta (up to order unity recently demonstrated in the Pegasus device [17]) and higher fast-ion and toroidal flow velocities normalized to the Alfven speed. This unique physics regime provides access to unique operational regimes to test confinement, stability, power handling, and current sustainment theory, enabling critical tests of a wide range of tokamak physics models for improved predictive capability. Perhaps the most obvious natural advantage of the ST is the compact geometry of the system, headlined in the prior section. Since the toroidal magnetic field B scales as 1/R in toroidal devices, a key associated advantage of low-A is the high degree of field utilization in the device (as noted in Austin WG-SA3 "innovations" talk [16]), as the ST plasma uniquely operates in regions of lower R. Additionally, lower A yields higher field line curvature of the ST and is a key physical reason for increased ST plasma global MHD stability limits. Examining the potential benefit of ST design for thermonuclear D-T fusion power density, $P_{fusion} \sim (nT)^2 \sim B_0^4 \cdot \beta^2$ in terms of relevant stability parameters as $P_{fusion} \sim B_0^4 \cdot \varepsilon^2(\beta_N^2/q_*^2)(1+\kappa^2)^2$, where the normalized beta stability parameter $\beta_N=\beta/(I/aB_0)$, safety factor, $q_* \sim B_0 a\varepsilon(1+\kappa^2)/I_p$, κ is the plasma elongation, and $\varepsilon = 1/A$, we see that the natural high shaping of the ST (κ up to 3 in the NSTX device vs. $\kappa < 2$ in conventional tokamaks) and the very high β_N accessed in the ST compared to high values in advanced tokamaks at more conventional aspect ratio (Figure 3) [18] strongly increase P_{fusion} for a given q*. Another significant element to tokamak optimization is the natural ability of the plasma to itself drive the required toroidal plasma current. This so-called "bootstrap current" is a neoclassical effect, with the fraction of the bootstrap current having the scaling $f_{BS} \sim \varepsilon^{1/2}\beta_p \sim (\beta_N/l_i)\, q_*/((1+\kappa)\varepsilon^{1/2})$, where β_p is poloidal beta and l_i is the plasma internal inductance. High f_{BS} values near or at 100% are highly desired in that such operation reduces the requirements on other forms of added current drive that are typically power inefficient. The term β_N/l_i is important for two reasons. First, it represents an important global plasma MHD stability parameter (for kink/ballooning and resistive wall modes). As shown in Figure 3, STs have produced plasmas at β_N/l_i up to 14, which is significantly larger than values $\beta_N/l_i \sim 4$ produced and considered high in advanced tokamaks at more conventional aspect ratio. Second, the plasma self-driven bootstrap current tends to naturally broaden the current profile (lowering l_i) significantly as f_{BS} increases. So, high f_{BS} values are most easily produced at high β_N/l_i. The f_{BS} scaling with β_N/l_i, the natural tendency of high f_{BS} to reduce l_i, and the key element of greater ST stability allowing very high β_N/l_i are three effects that synergistically combine in the ST to produce plasma with high stability and high self-driven current fraction required for any long-pulse/stationary tokamak. The NSTX-U device uniquely leverages this synergy through stabilizing plates and specially-aimed high NBI power (12 MW) allowing 100% NICD.

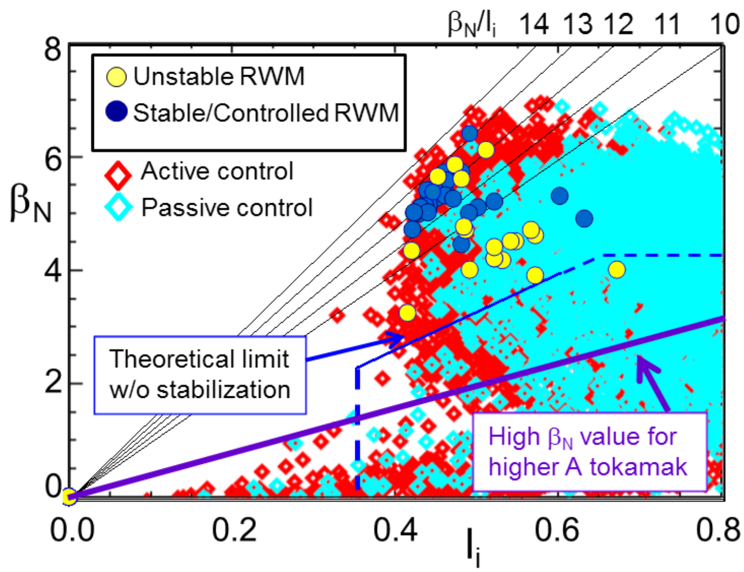

Figure 3: Very high stability parameters β_N and β_N/l_i have been produced and maintained in the high auxiliary powered NSTX device. [18]

Opportunities, benefits, and innovations in ST research

The key opportunity highlighted in this whitepaper is the exploitation of the integrated U.S. ST program to point to near-term fusion testing and developing more attractive fusion concepts. The U.S. has made a substantial investment in the ST configuration including: major upgrades to the NSTX facility; world-leading studies of increased confinement by low-recycling walls in the Lithium Tokamak eXperiment beta (LTX-β); and developing innovative means of plasma current generation in the Pegasus toroidal facility. The U.S. should realize and exploit the game-changing potential of NSTX-U in establishing the potential increase in energy confinement [9] and macroscopic stability expected by theory at low plasma collisionality [19], and by further validating the observed positive and unintuitive stability enhancement

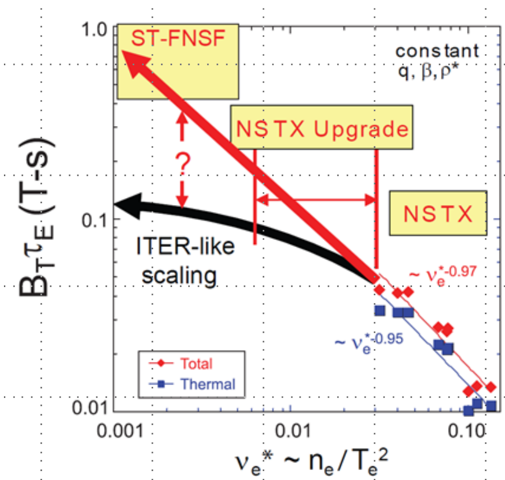

Figure 4: Increase in τ_E at lower ν^*_e in ST plasmas and projection to next-step devices. [9]

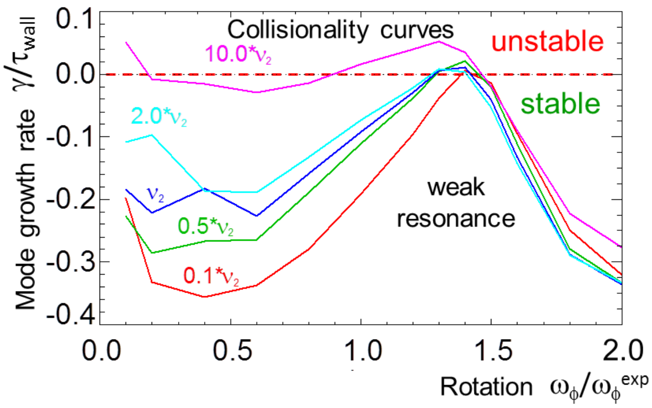

Figure 5: Projected macrostability increase at reduced collisionality and preferred rotation in the ST. [19]

at high β [18]. The advanced heating, current drive, and control capabilities of NSTX-U should be utilized to access stable and controlled self-consistent, fully non-inductive scenarios at high β with no transformer action [20]. The critical gap in understanding regarding plasma energy confinement and macrostability in the ST extrapolated to reduced collisionality is illustrated in Figures 4 and 5. In the range of normalized electron plasma ν examined to date in the MAST and NSTX devices, the energy confinement time, τ_E, scales nearly inversely with ν^*_e. (Figure 4) with no dependence on plasma β, while the ITER98y2 formula scales as $\nu^{*\,-0.1}_e\,\beta^{-0.9}$. Also, when rotation is set at preferred levels with experimental profile shapes, reduced ν has a strong stabilizing effect on global MHD modes by kinetic effects (broad resonances between the rotation and the ion precession drift (Figure 5)). These positive ST confinement/stability trends need to be proven at the highest B and plasma pressure (NSTX-U). [19]

With a uniquely large range of fast particle velocity compared to the Alfvén velocity, and fast particle beta compared total plasma beta, STs with strong neutral beam injection (NBI) heating are particularly

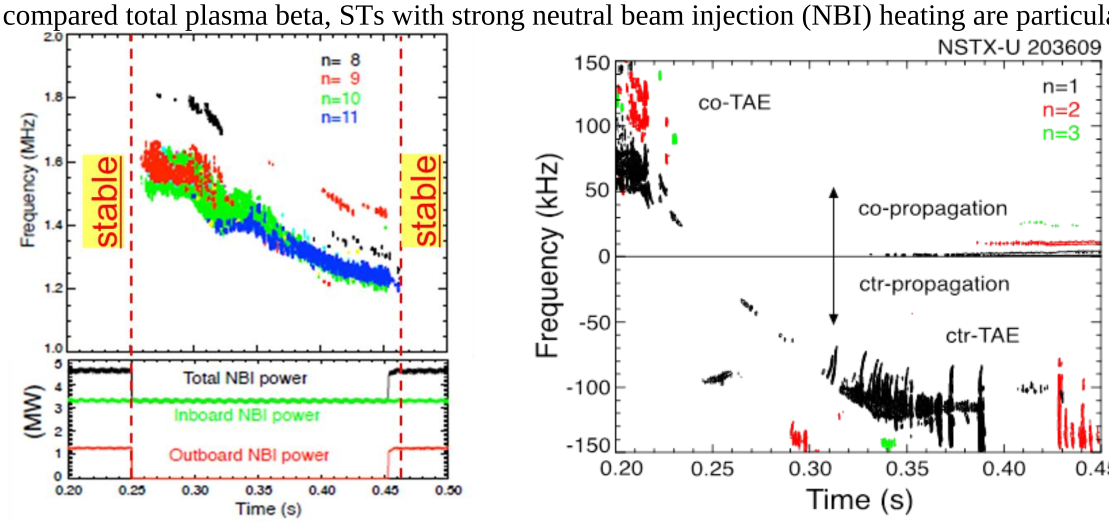

Figure 6: (left) Complete stabilization of GAE modes in NSTX-U when applying more tangential NBI, (right) discovery of counter-propagating TAE modes during off-axis NBI, qualitatively expected by theory.

well-suited for studying and controlling instabilities driven by energetic particle populations with application to understanding and predicting Alfvénic turbulence in burning plasmas including ITER. Early results from the NSTX-U device illustrate two important examples. Global Alfvén eigenmodes were stabilized when more tangentially-aimed NBI heating was applied, in agreement with the HYM code [21], and counter-propagating toroidal Alfvén eigenmodes were found when hollow fast-ion profiles were generated using off-axis NBI, as qualitatively expected from theory (Figure 6). [22] The result emphasizes how phase space variations of the EP population can significantly alter stability.

To realize the full potential of low-A tokamak systems, further innovations to increase energy confinement and non-solenoidal current formation would be advantageous and may be required. Results from LTX-β combining low recycling with strong beam heating and high beta will inform whether NSTX-U should transition to liquid metal plasma facing components, and in particular with liquid lithium plasma facing components (PFCs) to access very high confinement states in a high β, low collisionality plasma. Low recycling lithium walls have been shown to increase energy confinement since TFTR [23]. Lithium PFCs produce a strong increase in energy confinement in NSTX [24,25] through the production of wider and higher pressure profile pedestals and in LTX [26] through flatter and higher T profiles (Figure 7). [8] The production of completely flat temperature profiles with lithium PFCs has been demonstrated in LTX [27], and is expected to stabilize thermal gradient-driven modes (ETG, ITG) (Figure 7).

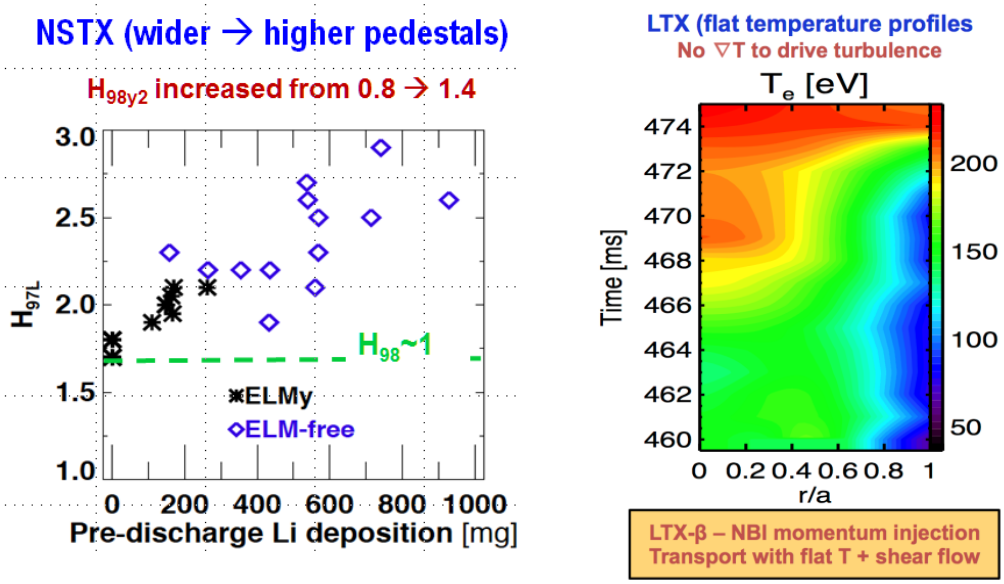

Figure 7: (left) Energy confinement enhancement in NSTX due to lithium PFCs. (right) Flat T profiles produced in LTX, favorable for stabilizing thermal gradient-driven modes.

Sheared flow generated by NBI may reduce remaining turbulence mechanisms. LTX-β will investigate the effect of flat temperature profiles and strong shear on anomalous transport in an ST, and extend the results to NSTX-U. LTX achieved very low collisionality ($\nu^*_{e,i} < 0.1$ over most of the plasma volume, and approaching 0.01 in the edge). [28] LTX-β will thus contribute to the study of the stability effects of low collisionality in NSTX-U. Low collisionality SOL conditions will be studied in detail in LTX-β modifying the SOL power deposition profile with unique benefits for the ST. Trapped particle effects in STs with collisionless SOL plasmas can reduce the total power flow to the divertor by 80 – 90%, with most of the SOL power radially transported to, and broadly distributed over, the wall rather than the divertor. SOL broadening due to the large poloidal gyroradius with very high edge temperatures further reduces peak divertor power by an expected order of magnitude in a reactor. These significant SOL changes produced by lithium walls are expected to be most apparent in the ST. [29]

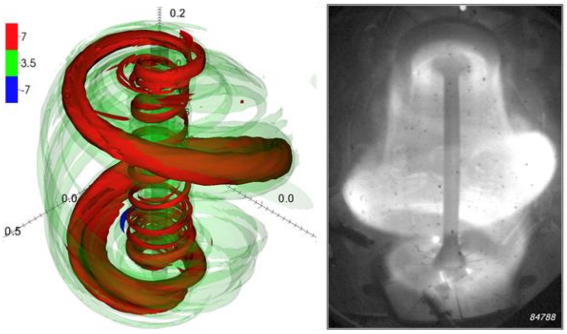

Figure 8: (left) NIMROD (MHD code) simulation of divertor LHI in Pegasus compared to (right) fast camera image of experimental diveror LHI in Pegasus.

Results from Pegasus support the viability of ST operation with no central solenoid. Significant currents exceeding 0.15 MA have been driven using local helicity injection (LHI). The result motivated a proposal to upgrade the device with the goal to develop and validate the LHI concept and produce system designs to generate up to 1 MA startup current for NSTX-U. [9] Pegasus experiments and divertor LHI simulations using the NIMROD code (Figure 8) will guide the optimization and understanding of helicity injection for current formation and growth at low-A, improve our fundamental understanding of magnetic reconnection, and ultimately be applied to generate completely solenoid-free current formation, ramp-up, and sustainment in larger devices.

High performance ST plasmas provide key data that leverages the kinetic MHD stabilization theory at high beta and low aspect ratio to best validate the underlying physics. Present validation established a new paradigm explaining how kinetic effects can stabilize macroscopic MHD modes. [30] High beta ST data and analysis supplies critical components of a disruption forecasting approach identified as an innovation in the Austin WG-SA3 "innovations" talk. [31] This included a reduced disruption forecasting model whose components specially leveraged the high beta ST data in validating the stabilization physics theory in a regime uniquely tested by the ST. Predicted instability statistics from this forecasting model are produced by comparison to the underlying ST database, addressing disruption avoidance - one of two highest priority Department of Energy Office of Science research elements (Figure 9). Possible advantages of ST design (relatively low B field for a given β_N) for disruption amelioration should also be investigated, such as the potential for runaway electron (RE) mitigation through pitch angle scattering by instabilities (e.g. whistler modes) whose growth rate scale as 1/B [32].

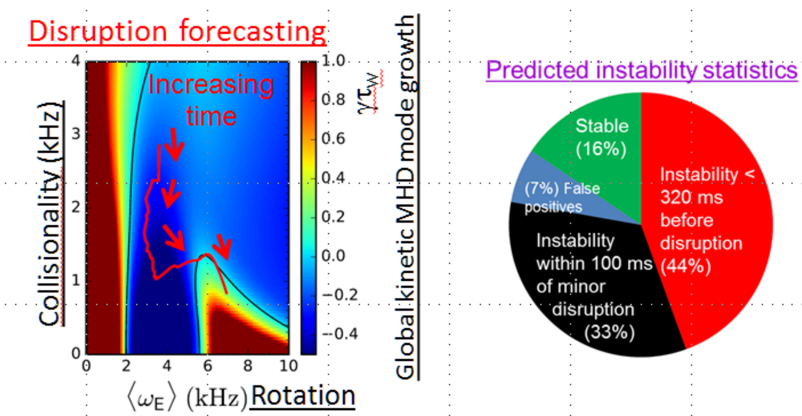

Figure 9: Disruption forecasting model vs. ν and plasma rotation uniquely leveraging high beta ST data to implement the underlying stabilization physics theory and compare to experiment [30].

With success of these activities, a major new U.S. ST experiment at optimized aspect ratio could employ HTS, full non-inductive current drive (including plasma start-up and ramp-up), continuous disruption-free operation, and a resilient PMI solution (e.g. liquid metals), to demonstrate a substantially more compact, attractive tokamak fusion system. Research and development of all of these areas received consensus-level agreement in the Madison workshop [4] that carried over to the Austin workshop.

Programmatic Context

The U.S. is presently a world-leader in low-A/ST physics research in several areas including high-β MHD stability and control, turbulent transport, fast-ion-driven instabilities, non-inductive plasma start-up, sustainment, plasma disruption prediction and avoidance, and advanced liquid metal plasma facing components. With sufficient support, the U.S. will maintain and expand leadership in these key research areas. U.K. venture capital is funding HTS-magnet-based ST development. This is a positive development, and should spur the U.S. to accelerate HTS magnet development for low-A configurations in order to maintain leadership.

There are 17 ST experiments operational or under construction world-wide. Substantial cooperation between ST programs in the U.S., U.K. and Japan has made major contributions to ST research. Further, U.S. ST researchers are in high demand for collaborations on conventional and ST facilities worldwide. Key examples include collaboration between NSTX/NSTX-U, DIII-D, and KSTAR (high A = 3.5) to test first principles physics versus magnetic field geometry (aspect ratio effects). Also, collaboration opportunities exist between NSTX-U and the MAST-U ST to leverage their complementary capabilities including investigation of power exhaust strategies compared to the world-leading advanced/super-X configuration on MAST-U, and to test MHD mode stabilization physics in high beta plasmas with (NSTX-U) and without (MAST-U) a copper stabilizing wall. Additionally, operational overlap between these devices enables impactful scientific studies and encourages collaboration and friendly competition. Past DOE committees have stated that all such collaborative efforts by U.S. ST scientists need to tie back to research on a major U.S. program for the U.S. to benefit from the collaboration resources spent.

Universities, national labs, and industry have played an essential role in the integrated U.S. ST program by leading major research program elements, providing innovative diagnostics, and in research operations. This connection has fostered the critical inclusion of students and young researchers for educational purposes and to help assure the infusion of new ideas into the program. Strategic changes to ST research program management could further empower collaborating universities and national labs in the management process of the flagship and satellite ST facilities. This is a necessary component of mutual support and motivation amongst researchers. Advanced HTS magnet development, flowing liquid metal systems, innovative actuators for current formation, and other enabling capabilities could be strongly supported by industry if additional resources were available.

Possible 15 Year U.S. research agenda

The immediate research program for U.S. low-A research has the following priorities: (i) demonstrate and understand plasma energy confinement scaling, stability, and sustainment properties at low collisionality needed to develop and validate the integrated core physics basis (NSTX-U), (ii) continue to investigate plasma exhaust and particle control solutions to identify self-consistent core-edge solutions, through radiative divertor configuration studies (domestically at NSTX-U, and through international collaboration at MAST-U) and resilient first-wall (liquid lithium), low-recycling PFC research (LTX-β, NSTX-U), (iii) continue to develop solenoid-free start-up methods including localized helicity injection at higher field (Pegasus-E), coaxial helicity injection (international collaboration on QUEST), and study non-inductive current ramp-up via NBI and High-Harmonic Fast Wave current drive (including overdriven current scenarios) for current ramp-up (NSTX-U).

Assuming favorable core performance and sustainment results from the immediate near-term, there should be increased emphasis on (i) demonstrating sustained, high-pressure operation without carbon PFCs, requiring a transition to high-Z PFCs and initiation of a liquid metal program, (ii) implementing optimal solenoid-free startup methods developed on Pegasus-E, QUEST, ST-40 and others to demonstrate they can be integrated with optimized current ramp and flat-top scenarios. In addition to the elements above, leveraging positive results from high-temperature superconducting toroidal field coil R&D with high field and current density could be utilized in a high-field next-step-ST that would accelerate the path toward a demonstration fusion power plant (DEMO).

Research Directions Beyond 15 years

The results of near-to-medium term low-A research will ideally improve the low-A physics basis that, when coupled to the complementary physics basis at higher-A developed from DIII-D and other international tokamaks, will allow further optimization of future devices. Assuming success of the near-term research agenda, such a long-term research agenda would logically be integrated into a national initiative for a new facility. A compact DT facility with a mission to retire risks to constructing DEMO should be considered as the next-step once the present generation of upgraded ST devices fills key research gaps.

Challenges regarding the ST approach

Several gaps exist in understanding regarding the confinement and stability performance of the ST as the plasma collisionality is reduced in future devices. Such gaps are stated above and simply reinforce the need for research in such areas to verify theory thereby generating the required understanding. The resolution of these gaps may lead to further positive conclusions for the ST approach, and in some cases a positive result is required for the ST to remain a viable approach as a neutron source or an energy producing device. There is clearly a limit to how small the aspect ratio can be made, including the mechanical strength of the toroidal field magnets, the desire for shielding at low major radius, and the possible need for even a small central solenoid to allow some level of transient current control for plasma start-up or to navigate transient plasma events. Operation at a reduced magnetic field causes potential issues with RF wave accessibility, requiring mode conversion or alternate means of auxiliary heating and current drive. Also, with the advantages of compact design emerge the challenges of tight space for components including exhaust/divertor systems, control actuators, breeding blanket and shielding [33]. Solutions to such challenges will require research in the corresponding physics and technology areas, materials, and whole-device engineering design.

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