DEVELOPMENT OF A STEADY STATE FUSION CORE – THE ADVANCED TOKAMAK PATH

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GENERAL ATOMICS

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A: The Advanced Tokamak Concept and Research Progress

Executive Summary

A continuously operating fusion power plant requires a high performance highly non-inductive reactor core. The tokamak is the frontrunner concept to meet this challenge, being the basis for reactor designs amongst most of the ITER partners (SlimCS, Japan [Tobita 2009], EU-DEMO/FPP [Zohm 2010, Lee 2015], K-DEMO, Korea [Kim 2015, Kang 2016], ARIES-ACT [Kessel 2015], ARC [Sorbom 2015], ST pilot plant [Menard 2016]). The essence of the approach is to replace inductive current with a combination of bootstrap current, naturally arising from orbit effects at high pressure gradients, and auxiliary current drive such as from RF. These non-inductive currents are typically distributed broadly in the plasma (away from the center where inductive currents peak). This is a favorable property which improves stability, and thermal and fast ion confinement, allowing the plasma to be sustained at high pressure self-consistently. Other favorable configurations are also being explored based on peaked profiles and efficient core current drive.

However, validated projections of the plasma performance and necessary control approaches have yet to be established. This is fundamental to being able to set the parameters and required systems for a reactor. Critical issues are: (a) to understand turbulent and energetic particle transport, and how these interact with current, pressure and particle profiles to arrive at a self-consistent solution, (b) to project stable access to high β_N, control edge localized modes (ELMs) and, if needed, safely terminate the plasma, (c) to develop non-inductive plasma scenarios and required high efficiency reactor-compatible current drive tools, (d) to resolve compatibility with an advanced non-eroding divertor solution and relevant wall materials. Answers to these questions can determine the operating parameters of future devices, which so far have been aspirational in design; the techniques and physics basis to project how to obtain self-consistent, converged, high performance stable solutions are needed.

Our vision is to transform the DIII-D National Fusion Facility to address these critical questions – to discover the solutions in D-D plasmas, so that the approach for a D-T device can be realized. This requires integrated core-to-edge exploration, not only developing and understanding the physics solutions within each region, but also crucially the interactions between regions. In particular, a key constraint arises from divertor integration, where the high density of particles used to radiate heat must be contained in a way that is compatible with a high performance low collisionality core. This requires research and facility developments on multiple fronts:

(i) To characterize, discover and project improved how to improve turbulent transport and stability in burning plasmas by operating in the relevant regimes of $T_e \sim T_i$, low collisionality and low rotation, through increased torque-free and electron heating.

(ii) To resolve transients through new flexibility in 3D fields and plasma profiles, in order to understand how to achieve ELM suppression and maintain stability, and developing ‘inside-out’ plasma quenching tools to provide safe termination of fusion plasmas.

(iii) To determine the path to self-consistent fully non-inductive operation through high β_P and β_T plasmas with flexible current profiles, increased heating power and new current drive tools.

(iv) To resolve the development and physics of closed divertor configurations and reactor-relevant wall materials, through installation of new structures, new wall materials, and key testbed facilities.

(v) To reconcile the interactions of a dissipative dense divertor with a high performance low collisionality fusion core, by operating at parameters which access reactor-like physics regimes in both regions simultaneously.

This latter core-edge mission motivates a performance upgrade. Physics based simulations show that increased shaping and heating can simultaneously reach the low collisionalities of a high performance core, and high density for a dissipative divertor. Higher toroidal field raises plasma opacity (through higher current and density), thereby decreasing neutral and radiation penetration depths to capture the dynamic and underlying physics of this interaction, potentially reaching ITER-like absolute pedestal pressures and a thermal $Q_{D\text{-}T\text{-}equiv}$~1. Combined with new closed divertors, this will allow reactor-like integrated solutions to be studied – so called 'fusion equivalent regimes'. It also closes gaps on key parameters such as electron-ion coupling, fast ion fraction, bootstrap fraction, ρ^*, and parallel heat flux, reducing extrapolation and validating key physics to enable development of innovative approaches and accurate projection to reactors.

These developments will transform DIII-D to access the new physics regimes and science of future fusion reactors, representing a major re-development of the facility. Combined with state of the art diagnostics, advanced simulation, and its highly collaborative team, DIII-D will place the U.S. at the forefront of world-wide scientific capabilities, as a highly flexible tool to attain the scientific understanding needed to develop and project the ground-breaking solutions required for the reactor scale. Alongside an accompanying technology and engineering program, this will provide the knowledge and the confidence for the U.S. to take a decision on, and set the main parameters for, a U.S. steady state D-T reactor. It will also equip the U.S. to lead on ITER, and to bring back the benefits of ITER to the U.S., in order to provide the expertise to progress on this fusion energy path.

Fundamentally, fusion energy requires the resolution of scientific questions and techniques to resolve a path to commercial power. Critical questions can and should be resolved in D-D facilities, so that the U.S. program can proceed rapidly to D-T. DIII-D provides a unique tool meet this reactor challenge and pioneer a path to a low capital cost reactor, addressing the key challenges with unique flexibility and range, to develop fusion energy plasma solutions.

I. Motivation & Need – The Reactor Challenge

The achievement of a sustained burning plasma for fusion energy production represents one of the grand scientific and engineering challenges confronting the world today. A suitably performing and controlled plasma will be the culmination of an immense program of research being pursued across the globe. This work has made enormous progress in establishing the tokamak concept and developing a robust basis to project it to the reactor scale. This is embodied in the decision to proceed with the ITER device, a partnership between countries representing over half the world's population, due to commence operation within the next decade. ITER will establish dominantly self-heated fusing plasmas, proving that the physics works at the power plant scale, and providing crucial validation and insights into the development of fusion energy.

Alongside preparation for ITER, research focus is thus increasingly turning to sustainment of burning plasmas in continuously operating "steady state" conditions required for a cost-effective fusion power plant. A primer on the advanced tokamak approach to meet this challenge is provided in the attached appendix. A number of potential demonstration device concepts have emerged based on the aspect ratio ~3 tokamak (ARIES, EU-DEMO, K-DEMO, SlimCS, ARC, and of course ITER's mission 2). However, the technologies and approaches to enable these devices have not yet been resolved, motivating research initiatives to develop a viable path. Critical issues include projection of performance, avoidance of transients (disruptions, ELMs), safe quenching, a steady state divertor solution, suitable materials, and efficient current drive. Fundamentally, viable configurations must be developed together with confident projection through validated simulation in order to specify a steady state reactor. The relevant normalized pressure and plasma configuration have yet to be sustained fully non-inductively; a solution, and the necessary tools to sustain it, must be proven. Research on DIII-D seeks to confront these challenges.

Understanding plasma behavior is at the heart of this mission. This plays the key role in setting the required scale of the device, its performance, and the interaction with its containment and auxiliary systems. The research is challenging because of the exotic nature of the burning plasma state and the huge energy fluxes that flow through the device. These drive processes that define the performance and set limits to what can be achieved. The processes are complex and happen at a range of scales, from fine scale instabilities, through turbulent eddies, to macroscopic structures that can re-arrange the configuration entirely. Behavior depends on the specific mechanisms and channels involved; they thus require exploration in reactor-relevant physics regimes, with appropriate techniques to probe and measure their behavior. For instance:

- Steady state regimes require plasma configurations with different internal magnetic structure, high β and benefiting from strong shaping, altering fast ion resonances, α confinement and turbulent transport relative to the baseline regimes planned for ITER.
- Fusion α's heat electrons while fusion reactors operate at low collisionality and rotational shear, changing turbulence characteristics cf. present torque injecting beam heated devices.

- Instabilities, and the interactions of 3D fields or local current drive to control them, are influenced at a fundamental level by rotation, current profile, collisionality and β_N.
- A reactor requires a dense collisional divertor solution, which must be reconciled with a high pressure H-mode pedestal, radiative mantle and impurity dynamics.
- The interaction with and choice of surrounding materials and auxiliary systems such as current drive must be resolved, including the back reaction on and control of the core.

The DIII-D program has conducted first-of-a-kind, self-consistent physics-based predictive 1.5D simulations to identify the key parameters and techniques that lever the development of a low capital cost, compact advanced tokamak pilot plant. This represents a crucial stage before a larger, potentially lower cost of electricity (but higher capital cost) power plant is built. The studies exploit the FASTRAN suite and constituent TGLF, EPED and current drive models [Park 2017a], developed and validated in the DIII-D research program [Holcomb 2014, Park 2017b], to predict converged, self-consistent fully non-inductive net-electric fusion steady states. This provides key insights over the usual 'systems code' approach where desired parameters are simply asserted. The studies show (Fig. 1) that a combination of high β and high density is required to ensure sufficient fusion power and reduce auxiliary current drive requirements. Efficiency of systems to provide residual current drive is key – a somewhat ambitious $\eta_{th}=\eta_{CD}=0.4$ is set as a target here. Higher field is also helpful, acting to raise pedestal and core pressure; while 6T solutions, accessible with conventional superconductors were found at 4m radius, 7T provides margin to relax assumptions on current drive efficiency, density or safety factor. With these benefits, a modest scale device can be constructed, with tolerable neutron and divertor heat loads. Similar considerations arise for larger scale devices that seek to reduce cost and scale, such as ARIES-ACT1 [Kessel 2015]. The challenge is to determine if the required plasma and systems performance can be achieved.

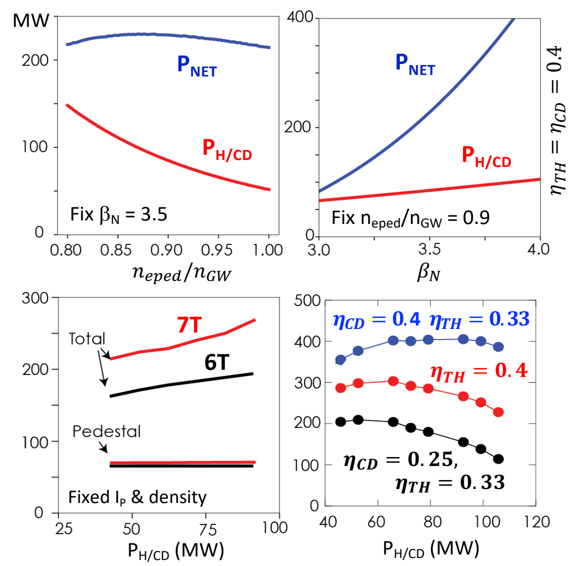

Fig. 1: Simulations of fully non-inductive plasmas in a compact net-electric advanced tokamak pilot plant with 4m radius, 7T, and $\eta_{th}=\eta_{CD}=0.4$. Auxiliary heating and current drive is adjusted to ensure each point is fully non-inductive.

These considerations identify the critical research capabilities and directions necessary:

1. **Fully non-inductive steady state regimes:** It is important to be able to explore strongly shaped regimes with a range of current and pressure profiles, varying β_N, density, q_{95} and other parameters to test predicted configurations such as those in Fig. 1 and identify how to optimize and access them, achieve self-consistent heating and current solutions, and test compatibility with divertor solutions and the wall. Critical further issues include pedestal performance and density limit. More efficient reactor-compatible **current drive tools** must also be developed.

2. **Extension to burning plasma relevant conditions:** These regimes must also be developed to low rotation, low core collisionality, and T_e~T_i with coupled ions and electrons, in order to access and assess relevant turbulent transport and stability physics processes. Reduced fast ion fraction regimes (with lower neutral beam fast ion) are also an important aspect to assess.
3. **Control of instabilities and ELMs** must be developed both through configuration optimization and with dedicated tools, including safe quenching when needed. n=3 and n=4 3D fields are likely necessary to control ELMs; it is thus vital to develop understanding of how to optimize 3D spectra to achieve control without driving locked modes.
4. **Core-edge integration** represents a fundamental tension, where divertor solutions favor high density while core solutions and current drive require low collisionality. A key to this is closure – both physical and magnetic – to isolate the regions as far as possible. Research must explore (i) the interaction between the core and closed divertors in relevant physics regimes, (ii) common governing parameters such as shape, and (iii) techniques to improve compatibility in each region, such as the recently discovered super-H mode pedestal, or the super X, snowflake or small angle slot closed divertor. Radiative mantle techniques and control of impurities must also be developed. *(Development of advanced divertor concepts and studies of materials interactions are crucial, but the subject of a separate white paper).*

Addressing these issues will establish the basis for confident projection to future fusion reactors, in order to determine their design, required parameters and auxiliary systems. Progress will also be highly levering to U.S. participation in ITER, where its operating scenarios face many of the same issues; expertise gained here would enable U.S. leadership on ITER, enhance its chances of success, and help translate lessons from ITER to the U.S. fusion energy path.

II. Program to Develop Steady State Reactor Solutions

New research capabilities are needed to resolve the path to a steady state fusion reactor. Many devices around the world exploit co-injected neutral beams to reach high performance, heating the ions and driving favorable rotation, unlike in a fusion reactor. The push to operate with reactor relevant wall materials, though important to study, has forced facilities to high collisionality, as they use gas puffing to drive ELMs to flush impurities from the core; on present scale devices, this forces the pedestal to the reactor-irrelevant ballooning limited part of the operational space, and the core to elevated collisionality [Maggi 2015]. Flexibility to access high β_N and vary current and pressure profiles is also very limited, while reactor compatible current drive tools (which must be more efficient than present technologies, and solve antenna loading and PFC issues) have yet to be developed. The world's facilities are well suited to explore improved divertor concepts with the highly flexible MAST-U and beam-upgraded TCV facilities coming on line this year, and tungsten divertors on JET, ASDEX Upgrade and WEST (heated). However, integration of closed divertor approaches with high performance fully non-inductive cores remains elusive.

Nevertheless, there are exciting research tools both in existence and planned in the near term that can address key elements of this challenge. In the U.S., the key facility is the DIII-D tokamak, which has pioneered many of foundational elements of the AT approach (see appendix for examples). This facility is now being redeveloped to access reactor relevant AT physics regimes; we describe the elements of this below. The essence of the approach is to provide access to, and flexibility in, relevant physics regimes. Starting this year (2018), major improvements in current drive tools, electron heating, 3D fields and divertors will be implemented on DIII-D. The resulting capabilities will complement those elsewhere around the world:

- The superconducting EAST and KSTAR facilities can operate high β_P regimes for long pulses, but are limited in absolute performance or β_T. The model here has been for DIII-D to use its flexibility to scope out these high β_P regimes and understand the physics basis, prior to long pulse testing abroad. Key opportunities include assessment of long pulse control technologies and long time scale wall evolution [FESAC 2012 international report].
- JT-60SA will be a key facility to test projections to larger scale. It begins operation in lower single null as an ITER-satellite. However, high power AT operation comes 6-8 years later at reduced field and current (actually lower than DIII-D). A double null 'advanced' divertor follows in a more speculative extended research phase after that [JT-60SA research plan, March 2016].
- JET and ASDEX Upgrade AT capabilities have been hampered by the installation of metal walls preventing access to reactor-like low ν^* regimes. But there are indications of moderately advanced AT regime access on ASDEX Upgrade with strong core electron heating to overcome impurity accumulation [Stober2016] and so study metal wall .

In this paper we focus on the DIII-D strategy to meet the steady state reactor challenge, identifying distinctive actions that are required to resolve how to establish a fully non-inductive high performance core with suitable PFC compatibility. We also discuss collaborative elements, including the spherical tokamak, which can provide important physics tests on the aspect ratio, R/a~3 path (which we discuss), but must pursue further steps for an ST path (which we leave to other white papers). Divertor optimization and materials issues are left to other papers, though integration between core and divertor/wall requirements remains a key challenge covered here.

1. Fully Non-Inductive Steady State Regimes

A foundational element of the steady state approach is to demonstrate that self-consistent fully non-inductive solutions can be sustained at required performance levels, and to understand what the performance limits and required control tools are. DIII-D is being upgraded with large rises in flexibly deposited current drive and increases in available heating power to access and study the necessary configurations to make these determinations, also pioneering new current drive technologies to proof out the tools that will be required in a steady state fusion reactor.

Starting in 2018, DIII-D neutral beam systems will be re-oriented to double off-axis current drive power. Two of the 8 beams will be toroidally steerable, allowing all power to be injected in the plasma current direction to assess high β_N scenarios, and enabling rotation variation at lower current drive levels. Beam energy rises will increase current drive and electron heating. This will be augmented by increases in electron cyclotron current drive power, which can be used to finely tune the current profile for performance and stability control thanks to its precise deposition control. To this end, a new 1.5MW gyrotron is presently being commissioned and three further units ordered, with more planned.

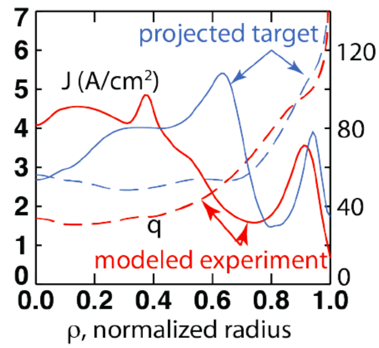

Fig. 2 Modeled current (solid, left axis) and q (dash, right axis) profiles for present (red) and projected (blue) plasmas.

This will broaden current profiles and remove rational q flux surfaces (Fig. 2) to access projected-stable configurations. Simulations predict transport and stability limits will rise to reach β_N~5 fully non-inductively (Table 1), comparable to ARIES-AT parameters, with flexibility to explore the roles of current, pressure and fast ion profile, with modeled solutions ranging from peaked current so-called 'high l_i' scenarios, to the broad current 'high q_{min}' shown here. This will enable tests of (i) kinetic damping and resistive MHD close to ideal MHD β limits, (ii) probing and control of energetic particle driven instabilities, (iii) assessment of the role of current profile in high β electromagnetically driven turbulence, and (iv) compatibility of advanced profiles with fully non-inductive sustainment and the above performance limiting physics.

DIII-D will also assess the physics of three promising new current drive technologies, which simulations indicate could lead to much greater efficiency in future reactors, also addressing coupling and antenna issues – potential game changers in required scale and performance of the device. These are top launch ECCD, ultra-high harmonic fast wave (Helicon), and high-field side lower hybrid current drive (HFS LHCD). Helicon has, in fact, already demonstrated good coupling in low power tests on DIII-D (appendix, Fig A-8) and is proceeding to a 1 MW installation in 2019, alongside proof of principle top launch ECCD tests. HFS LHCD is planned soon after. Projections indicate these could substantially improve current drive and further broaden profiles on DIII-D (Fig. 3) to further study

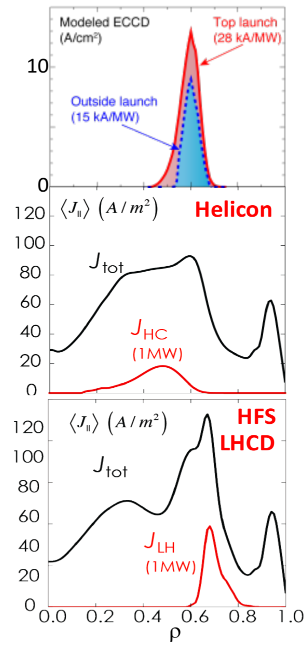

Fig 3: New current drive tools improve efficiency and increase capability.

Table 1: Existing best performance plasma (inductive) compared to FASTRAN/IPS simulations of DIII-D operating points with upgrades. q_{95}=5-6. f_{NI}≡total noninductive current / I_P

Case	On-axis NBI (MW)	Off-axis NBI (MW)	ECH (MW)	Transport limited β_N	Ideal MHD limited β_N
Shot 147634 q_{min}~1.5	7.5	3.3	3.5	3.5 (f_{NI}=0.75)	3.7 (with wall limit‡)
Predicted q_{min}>2	9.5	10.7	9	5.1 (f_{NI}=1.0)	4.9 (with wall limit)
Predicted q_{min}~1 "high *li*"	7	13*	9	4	4.1 (no-wall limit)

‡*This is the limit predicted by FASTRAN with the ~6 s, partially inductive discharge taken to t=∞.*

**With unfavorable B_T direction for off-axis current drive. Off-axis NBI broadens pressure only in this case.*

transport, stability and energetic particle physics, also raising ideal MHD β_N limits to values approaching 6, potentially fostering more robust margins for ideal and resistive MHD stability.

This work will establish the potential for fully non-inductive high β_N scenarios with stationary current and pressure distributions that are consistent with current drive sources, and macroscopic and Alfvénic stability. It would be complemented by studies on NSTX-U or MAST-U, where the low fields cause beam ions to be super-Alfvénic (like fusion α's) enabling important further model validation. NSTX-U could also explore current drive physics with high harmonic fast wave, and further validate models of kinetic effects in wall mode physics, as well as general stability maintenance. The planned superconducting facilities in Asia will help validate long pulse control and wall evolution issues, with JT-60SA providing important extrapolation to larger scale.

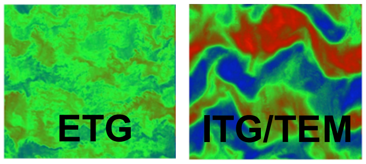

Fig 4: GYRO predictions of turbulence at high (left) and low (right) rotation.

2. Burning Plasma Relevant Conditions

The critical further step in developing a steady state core is to project regimes to burning plasma conditions. Rotation, collisionality, T_e/T_i and energetic particle content all play crucial roles in determining the structure, magnitude and channels of turbulent transport, modifying fluctuations from fine scale instabilities to large structures (Fig. 4), and altering pinch and diffusive effects. These parameters are also critical determinants of ideal MHD β_N limits through kinetic resonances with plasma rotation and energetic particle interactions [Reimerdes 2011]. Collisionality and rotation can further play important roles in pedestal stability and height.

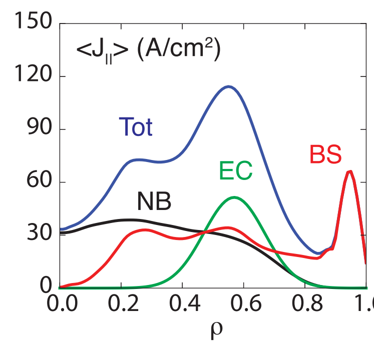

Fig 5: Advanced current profiles predicted with balanced neutral beam torque

To address these issues DIII-D plans progressive increases in electron and torque-free heating. A second pair of beams will be made toroidally steerable to enable full-power balanced torque operation. Combining with the planned upgrade to 9MW ECCD, which increases electron heating and current drive, simulations predict this will enable fully non-inductive plasmas with advanced tokamak profiles (Fig. 5 with q_{min}>1.5) and burning plasma relevant parameters (T_e/T_i≈1, low ν* and rotational shear) at β_N~4 and q_{95}=6. Additional helicon or HFS LHCD improves on this further, replacing lost neutral beam current drive with further off-axis currents to study the influence of advanced tokamak profiles on transport and stability up to β_N~5 in torque free H-modes (Fig. 6). Separately, this enables operation at higher density to study coupled electron-ion turbulence at T_e~T_i and zero torque.

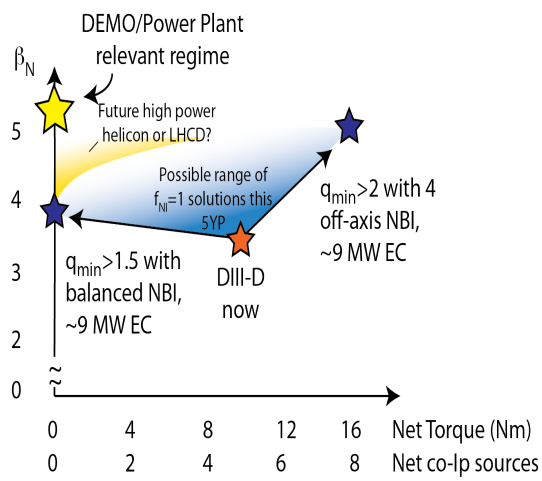

Fig 6: Predicted (blue) and target (yellow) operational space range with DIII-D heating and current drive upgrades.

This will enable development of validated models to predict burning plasma performance, with assessments of how these parameters impact critical gradients for drift wave induced turbulent transport in all channels. Research will evaluate how scenarios can be adapted to loss of ExB shear stabilization, and how optimization of the magnetic shear profile may be used to compensate. ECH will also enable precise perturbative tests of turbulence and pedestal behavior. Kinetic MHD stabilization physics will be advanced by varying the influence of fast ions using variable beam geometry, voltage, and by increasing $\beta_{thermal}/\beta_{fast}$ using ECH. Collaboration with other facilities will validate and extend the underlying physics; for example, the spherical tokamak provides an important testing ground for transport with strong electron heating, while JT-60SA can explore ρ* scaling of transport and stability.

3. Control of Instabilities and ELMs

Sustained operation of tokamak fusion plasmas requires control and mitigation of deleterious transient events. Plasma instabilities, including edge-localized modes (ELMs) and core or global instabilities that lead to disruptions, could prevent reactors achieving their mission through damage to the facility or de-rated operation to avoid potential damage.

A critical element is the use of 3-D fields to control instabilities such as ELMs, where significant progress has been made on DIII-D. However, present capabilities have limited harmonic flexibility to toroidal mode numbers of n=1 or 2, whereas the optimal fields for ELM and rotation control have n=3 or 4. Simulations developed in the DIII-D program predict that at these higher n it will be possible to vary the plasma response to independently control interactions in different parts of the plasma, thus controlling rotation profile, locked modes and ELMs independently. These simulations also predict that much more efficient coil sets for 3-D control are possible. A projected 12 coil midplane array (Fig 7) will test this physics, providing the first detailed spectral optimization studies for ELM and rotation control with n=3 or 4 fields. This will help understand how to best use these coils to develop validated optimized 3-D configurations for steady state fusion reactors, as well as to meet the mission in ITER, which this coil set closely emulates.

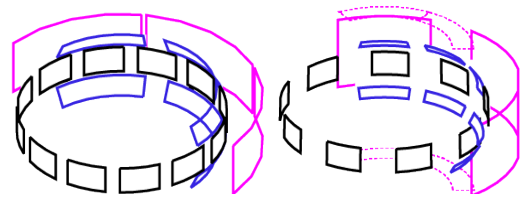

Fig. 7. Non-axisymmetric coil configurations planned for DIII-D (left) and ITER (right)

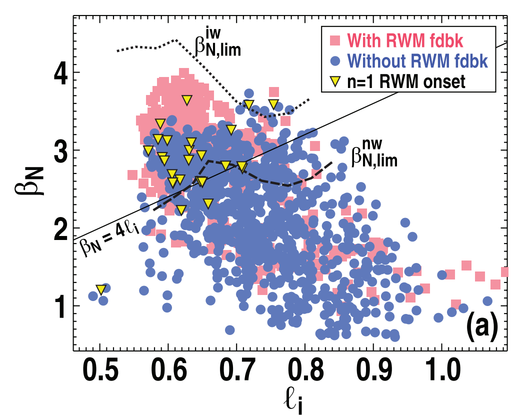

Fig. 8. Passive stabilization (blue) & active feedback (pink) extend the stable range in β_N beyond the no-wall stability limit toward the ideal-wall limit (dotted) in DIII-D.

It is also necessary to resolve the physics and develop control of global plasma stability at high β_N, where kinetic stabilization mediates a dissipative interaction of the pressure driven kink with the resistive wall. Toroidally steerable, variable-voltage neutral beams will vary ion velocity distribution and rotation to explore the resonant interaction of this mode with orbital frequencies of trapped ions behind this kinetic damping effect. The enhanced 3-D coil set with upgraded 3-D diagnostics will probe plasma response to measure dissipation at *n* up to 6, and develop advanced active control techniques at β_N approaching the ideal-wall limit (Fig. 8 [Hanson 2017]) to project a path to high β_N stable operation.

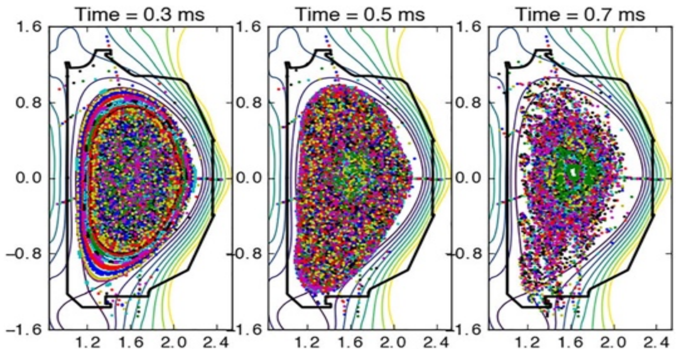

Fig. 9. Poincare plots of magnetic field lines after core-localized deposition of argon [NIMROD code, Izzo2017]. Core surfaces are stochastized immediately while outer surfaces are retained until later.

Should reactor systems fail, safe means to quench a fusion plasma are needed. Present techniques inject particles at the edge, limiting assimilation and allowing dangerous runaway electron beams to grow. DIII-D plans to develop 'inside-out' disruption mitigation using low-Z shell pellets filled with dust to deposit particles to the core. Modeling (Fig. 9) indicates this will dramatically improve all aspects of the disruption: 100% impurity assimilation assures high radiation fraction for rapid thermal quench; decay of the plasma current occurs more slowly (reducing induced forces) due to a still-warm plasma edge, while stochasticity generated in the core dissipates incipient runaway electrons. Particle and radiative diagnostics will be deployed to validate models of the quench and runaway dissipation, to develop predictive understanding.

4. Core-Edge Integration with a Closed Detached Divertor & Relevant Materials

A fundamental issue for a steady state reactor is to find a solution that simultaneously delivers high core performance and has compatibility with the divertor and wall. Fusion reactor cores will operate at low collisionality, ν^*, due to their high field and current [$\nu^* \sim n_e / (I_P^2 B_T^2 \text{shaping}^2) \sim n_e^3 / P^2$], while divertor protection requires a high absolute density dissipative divertor with a high degree of radiation to spread heat and reduce particle energies; a state known as 'detachment'. However, such dissipative techniques and wall interactions can lead to influxes that adversely affect the core and pedestal performance. Conversely, access to high power low collisionality cores can lead to divertor and wall fluxes that are particularly challenging.

Part of the solution is to alleviate this tension by improving behavior in each region. For instance, closed divertors, such as the promising 'small angle slot' (SAS) configuration [Guo 2017], facilitates detachment at lower upstream density, with neutral dynamics optimized to reduce particle energies at all radii (Fig. 10). Similarly, a super-H mode pedestal raises pressure (noting $\nu^* \sim n_e^3 / P^2$) to achieve a high density, low ν^* solution (appendix Fig. A-9). Studies will exploit power upgrades and increased shaping flexibility to explore these optimizations in each region and interactions between them. Here a 'performance upgrade' (discussed next page) is important to fully explore the relevant physics. Nevertheless, installation of closed configurations in both

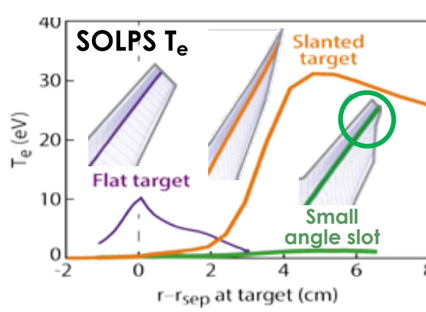

Fig 10: Closed divertor with optimized structure facilitates detachment.

the main upper and lower pumped divertors will enable investigation of the high-closure detached approach at high power and particle flux at present power levels, and its interaction with the plasma core. A critical aspect will be to understand how this is mediated through the pedestal. The interaction mechanisms will be explored through new profile, turbulence and neutrals diagnostics, exploiting relevant low ν^* access. Helicon or HFS LHCD are projected to enable high density fully non-inductive β_N~5 plasmas to assess core-radiative divertor solutions in the present DIII-D configuration (Fig. 11).

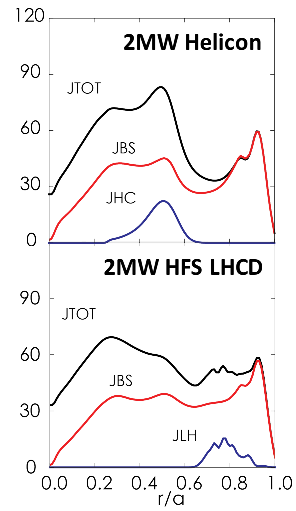

Fig 11: High density fully non-inductive plasmas projected at n_e/n_{GW}~0.9.

Compatibility and interaction of impurities arising from the wall and radiative mantle techniques with core performance will also be assessed, as well as the sourcing and transport of high-Z materials. Here perturbative impurity transport studies will be facilitated by a new laser blow off system and a hot tile test facility, as well as changes to wall and divertor materials. Siliconization and a subsequent SiC main wall will provide a low carbon environment to assess impurity and radiative divertor dynamics, as well as a potentially interesting candidate reactor wall material.

5. Fusion Equivalent Regimes (FER) with a DIII-D Performance Upgrade

The above program will validate physics models and develop technical solutions for phenomena from the core to the edge at reactor relevant parameters for each, developing a valuable projective physics understanding. The final step is to develop integrated solutions in *fusion equivalent regimes*. This means exploring the solutions in actual reactor-like conditions, such as coupled electron-ion turbulence, super-Alfvénic ion distributions, or high opacity plasma edge. This is important in order to understand the highly non-linear interactions between different phenomena, and to reduce extrapolation; we would be demonstrating integrated reactor solutions at reactor relevant integration parameters. To address this, a performance upgrade is proposed.

The most critical aspect is to reconcile the core and the edge: from the basic scaling of $\nu^* \sim n_e^3 / P^2$, divertor and core cannot simultaneously operate at reactor relevant parameters (n_e and ν^*) unless they are also at reactor-relevant absolute pressure, P. (The precise values needed depend on the mapping of pedestal to divertor density, which depend on progress in pedestal and divertor research missions through techniques like super-H mode and closed divertor operation). Core and edge strongly interact, with, for instance, leakage of neutrals and impurities from closed divertor solutions impacting pedestal, an interaction that is itself altered by increasing opacity as reactor-like densities are approached. As reactor densities are approached, divertor, scrape-off layer (SOL) and pedestal become increasingly opaque, and pedestal profiles become more strongly

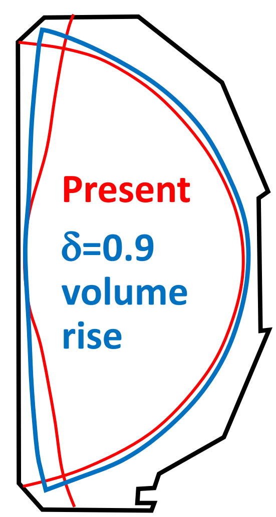

Fig 12: Triangularities up to 0.9 and a volume rise are possible on DIII-D.

dependent on transport and pinch effects. To assess this and develop integrated solutions requires both regions to be in the relevant density regimes, with the divertor challenged by relevant heat flux, characterized by $q_{||} \sim P_{SOL}B_{pol} / R$ when considering detachment.

Significant progress can be made by simply exploiting the existing vessel space and augmenting other systems. With removal of the upper inner cryopump, a rise in triangularity to 0.9 is possible, potentially raising pedestal stability and height, plasma volume and current carrying capacity (Fig 12). The resulting elevated pedestal pressures made possible by this shape and current rise would start to decouple ν^* and density to explore high performance cores with high power dissipative divertor configurations, and begin to understand the changes in pedestal dynamic with increasing opacity. The decoupling of ν^* from Greenwald density fraction would also help to resolve the physics and critical parameters for density limit – a key enabling parameter for reactor performance and divertor dissipation (see section I). With DIII-D plasmas already having transiently accessed $Q_{DTequiv}$~0.5, $Q_{DTequiv}$~1 is conceivable, an important demonstration, albeit with significant beam ion fusion. Accompanying the shaping rise with appropriate heating and current drive choices leads to a dramatic rise in performance in steady state conditions (Fig. 13), with a tripling of pedestal height and stored energy. ITER-like ν^*_{ped} are obtained at double the present density accessible, while fast ion fractions are greatly reduced and T_e>T_i. This is already a significant step toward 'fusion equivalent' performance, with physics tests of solutions becoming possible with the new range of parameters accessed.

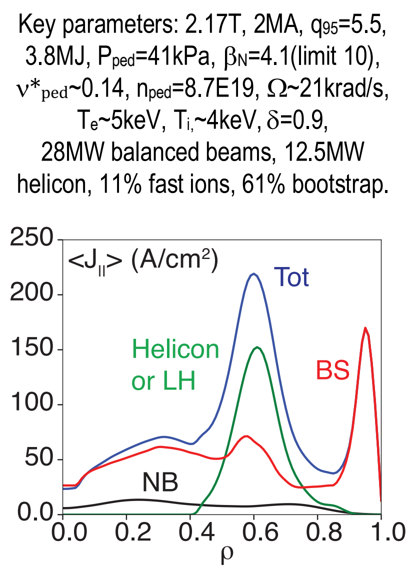

Fig. 13: Projected plasmas with shape and power upgrade.

A toroidal field rise enables the facility to go further in opacity and the core-edge mission (Fig. 14), achieving ITER-like levels in the ν^*_{ped} – density trade off at 2.5T, and projected to reach ITER-like absolute pedestal pressure at 4T. The 2.5T point could enable a highly significant step in achieved physics parameters and regimes, as discussed here, and is achievable with augmentation of the existing TF set. 4T represents a much more major infrastructure change, requiring careful technical assessment, with specific parameters somewhat dependent on the need arising after conducting research at the 2.5T level. Nevertheless, in a reactor, the pedestal is more opaque to neutrals, and its structure becomes predominantly determined by transport and pinch processes (which depend on collisionality, ν^*_{ped}) rather than by neutral deposition. Penetration depths for neutral ionization, Δ_{CX}, scale

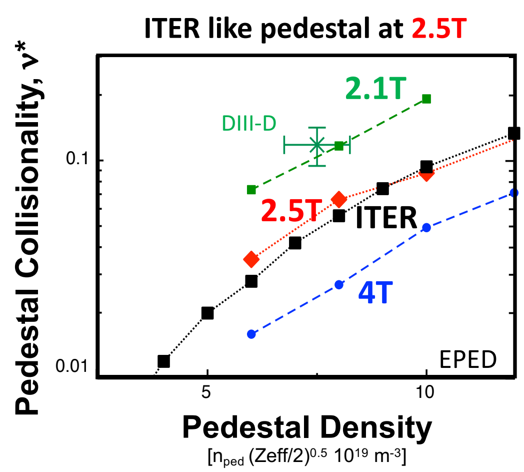

Fig. 14: EPED predicted pedestal space with field upgrades to DIII-D.

predominantly with density (Δ_{CX} = 1.91E17 $T_{ped}^{0.425}$ / n_{ped} [Loarte, private communication]). At present fields, 2.1T, in DIII-D, Δ_{CX} is comparable to pedestal width for reactor-relevant ν^*_{ped} < 0.2, and significant influence of neutrals is observed on pedestal structure. But Δ_{CX} reaches half to a third of this at 3 - 4T, thus enabling DIII-D to capture the reactor dynamic.

A closed divertor concept (Fig. 15) will be combined with this performance upgrade, with structure optimized to promote detachment at all radii, in order to develop and study detachment in the resulting high heat flux scenarios. These capabilities will enable development of compatible pedestal and divertor configurations and associated physics investigation at reactor relevant parameters for both regions simultaneously, thus correctly capturing the interactions between them to develop an integrated solution. In particular, the upgrade would increase parallel heat flux to stress test divertor solutions at reactor relevant levels, as compatibility with the core is developed. Parallel heat flux is projected to rise a factor of 3 from 0.8 GW/m^2 in steady state now, to 2.9 GW/m^2 at 2.5T and an ITER-like 5.2 GW/m^2 at 4T.

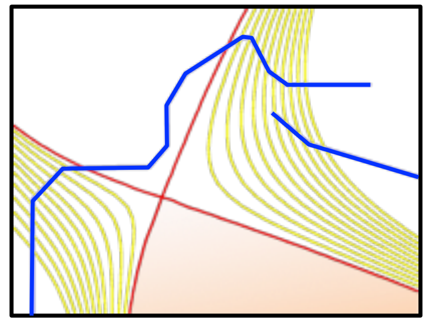

Fig. 15: A closed divertor concept would be combined with the performance upgrade.

The resulting plasmas would be higher in absolute density, reducing energetic particle populations to more reactor-relevant levels, and increasing thermal fractions (and lowering ν^*) to raise bootstrap fraction and enable efficient reactor regimes. The hotter temperatures would also raise auxiliary current drive efficiency. These plasmas would have strong electron-ion coupling to capture and test models of reactor-like turbulence. Additional required heating power, provided in part by neutral beams and helicon or HFS LHCD, or by ECH if the toroidal field is raised. This could be augmented by negative ion neutral beams, which injects α-particle-relevant super-Alfvénic beams to test the α physics in steady state configurations (relevant magnetic shear and β). Except for neutral beams, these techniques heat the electrons without torque, and can drive current to ensure studies are at burning plasma relevant steady state parameters. Plasmas are projected to reach a thermal $Q_{D\text{-}T\text{-}equiv}$~1; in short, DIII-D would be 'doing fusion' without actually doing the fusion part itself – *a fusion equivalent regime*.

Given the target of reactor like pedestal collisionality and divertor density, it makes sense to phase these upgrades, as progress improving the pedestal:divertor density metric through highly shaped advanced pedestals, closed divertors and increased heating power, can be used to set the target in field and current for the upgrade to fully reactor relevant core-edge parameters. This final set of developments, though not small, would bring about a powerful U.S. ability to finalize and demonstrate solutions for a successor steady state D-T facility, which as discussed in the opening sections, could then provide a one-step solution to fusion energy, with net electric power and nuclear science and breeding missions combined, to provide confidence for the private sector to take over the mission and deliver reliable competitive magnetic fusion energy.

III. Conclusions

Research to understand the requirements and scale of a future fully non-inductive burning steady state device is vital to the design of that device. The U.S. program is a world leader in this physics, with unique scientific expertise, and experimental, simulation and diagnostic tools. The DIII-D tokamak is at the fore-front of this effort, providing a highly flexible and an outstandingly diagnosed national user facility. Plans are already underway to reconfigure DIII-D for this exciting mission with a major upgrade commencing within months. The facility has the potential to confront physics challenges at reactor relevant parameters to enable development and confident projection of solutions for future fusion reactors. The present upgrade plan will address the critical physics of steady state operation from the core to the edge at relevant parameters individually. A more significant 'performance upgrade' would enable the development of integrated demonstration solutions and physics investigations that close the extrapolation gap on fusion plasmas, to provide the confidence to move directly to a net electric pilot plant nuclear science D-T fusion reactor (with accompanying technology and engineering research).

Both steps will position the U.S. as a world leader in the critical elements of fusion science and technology for reactors, and a strong collaborative partner in the world-wide fusion endeavor. These developments will also enable vital U.S. preparation for, and engagement with ITER, as set out in other white papers. Overall, the capabilities discussed in this proposal will enable DIII-D to make vital and needed contributions to the U.S. path to fusion energy, resolving critical research and even some technology questions, and developing the confident projection capability necessary to decide on, and specify future fusion reactors.

Acknowledgement

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Appendix A: The Advanced Tokamak Concept and Research Progress

The fundamental goal of advanced tokamak research is to develop a plasma that projects to high fusion gain with steady-state operation. In an advanced tokamak discharge, the inductively driven current density profile of a conventional discharge, which peaks at the axis, is replaced with a broadly distributed current density profile arising from the bootstrap effect combined with localized, externally-driven current density to tailor the profiles. The objective is self-consistent pressure and current density profiles that maximize confinement and stability in order to allow operation at normalized pressure (β_N) significantly above what is achievable in a conventional tokamak, with a minimum of external power input. (An alternative approach deploys peaked current profiles that raise no-wall ideal MHD stability and confinement to reach high β_N).

The bootstrap current arises naturally from orbit effects in the presence of high pressure gradients [Galeev 1968], consistent with the requirement for high absolute pressure in a high fusion gain power plant. For efficient steady-state operation the bootstrap current must provide a large fraction of the total plasma current and this fraction scales with β_P so that achievement of steady-state is easier at relatively low I_P, and thus relatively low fusion performance. Thus high absolute pressure is essential meet the fully non-inductive goal with sufficient fusion performance for an efficient reactor. In present medium B_T devices, this leads to steady-state solutions at high β_N. Increasing B_T through utilization of high temperature superconductors (if viable for a fusion reactor) could reduce the required β_N, but the configuration will still be an optimization in β_P, β_T, density and many other parameters. Achievement of an understanding of the physics of a high bootstrap current fraction, steady-state discharge in which the absolute plasma pressure is maximized, and validation of this understanding in experimental, stationary operation of this type of discharge is a key research challenge for the advanced tokamak program.

The advanced tokamak concept benefits from a natural synergy between non-inductive current distributions and plasma properties. Non-inductive currents are typically distributed broadly in the plasma (away from the core where inductive currents peak). This is a favorable property which improves stability, and thermal and fast ion confinement, allowing the plasma to

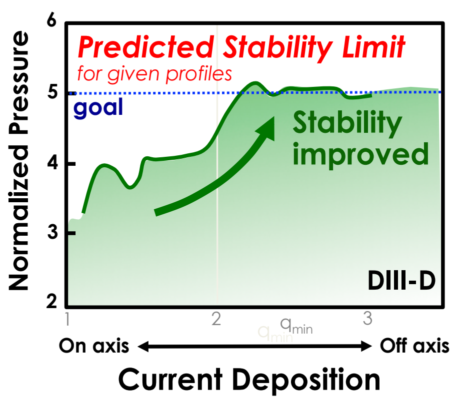

Fig. A-1: High β_N stability improves with off-axis currents.

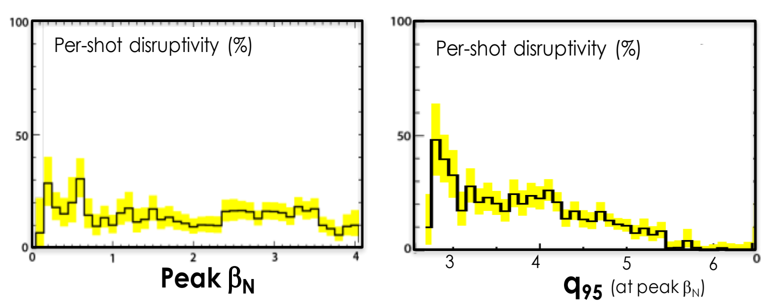

Fig A-2: Disruptivity independent of β_N but falls to zero at q_{95}~6.

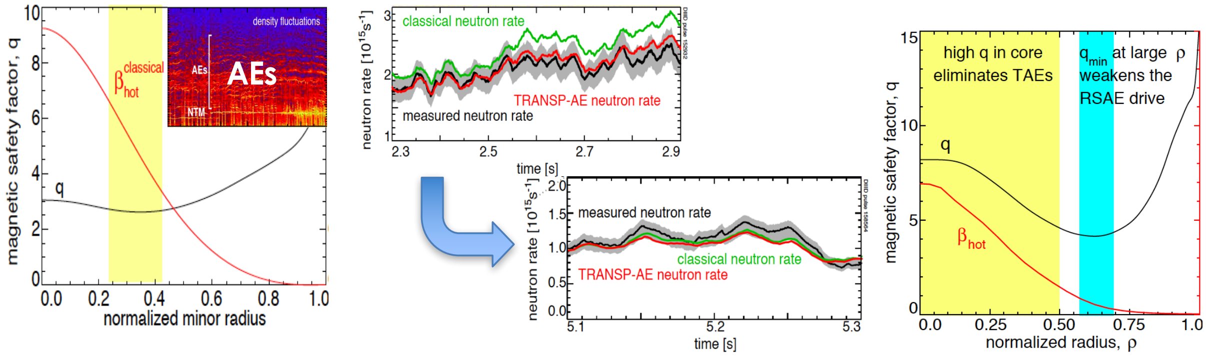

Fig A-3: Presently accessible current profiles are subject to fast ion redistribution (left) but Ohmic discharges demonstrate classical confinement of fast ions with broader current profiles.

operate at high pressure. Broad current profiles displace the destabilizing current gradient outwards leading to the eigenfunctions for the least stable ideal MHD modes also moving outwards and interacting more strongly with the conducting vacuum vessel wall. This is a dissipative interaction and has the effect of raising the attainable pressure in the device (Fig A-1). The broader profile also raises the central safety factor of the device, decreasing field pitch to remove the lowest order tearing modes. While higher order tearing modes can still occur, these are generally found close to ideal MHD limits and can be regarded as an extension of the ideal MHD properties, as set out in Brennan [Brennan 2003]. The resulting configuration is far more resilient to disruptions than inductive scenarios such as the ITER baseline, as the plasmas operate with higher safety factor, where tearing modes are encountered ahead of ideal MHD, and bleed out energy rather than cause a disruption (Fig A-2).

Broader profiles can also lead to improved stability to energetic-particle-driven modes, and thus to reduced energetic particle transport. With relatively peaked current profiles the weak shear region, where reverse shear Alfvén eigenmodes can be destabilized, aligns with a region of strong fast ion pressure gradient that provides the drive for the instability. As the current profile is broadened, this weak shear region is displaced further out, where the fast ion pressure gradient is reduced, which may improve stability (Fig. A-3). The reversal in magnetic shear is also highly stabilizing to turbulence, leading to reduced thermal transport and improved energy confinement

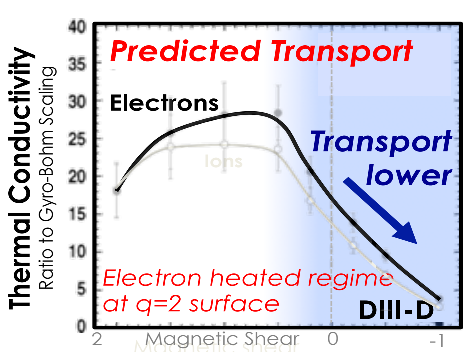

Fig A-4: GYRO simulations project reduced turbulence with broad current profiles.

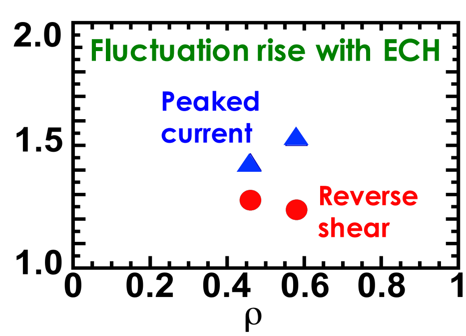

Fig A-5: DIII-D demonstrates favorable low turbulence regimes better maintained with reverse shear broad current profiles.

(Fig A-4). Indeed, recent progress on DIII-D has shown how reversed shear has helped maintain good energy confinement when electron heating dominates [Yoshida, 2017, Fig A-5]. These behaviors create a virtuous circle in which improved performance helps sustain the current and pressure profiles that in turn maintain the favorable stability and transport properties.

Although DIII-D has not yet been equipped with the tools to explore the full potential and range of advanced tokamak solutions (for which a research plan is elucidated in this note), significant progress has been made in validating key aspects of the physics basis and developing fully non-inductive scenarios.

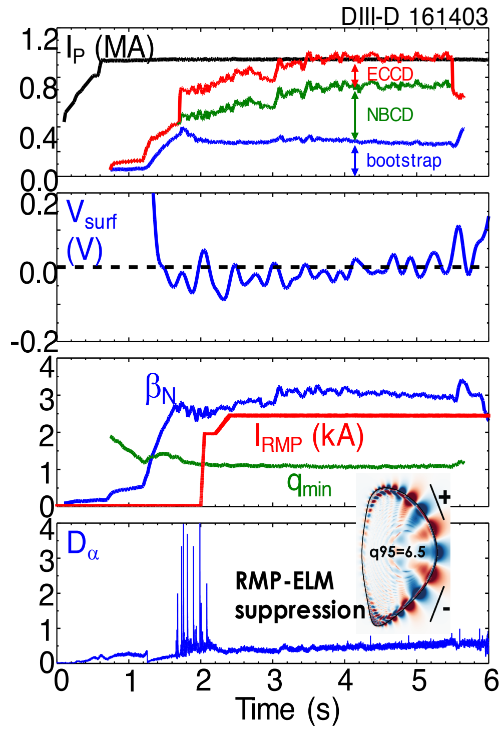

Fig A-6: Fully non-inductive steady state discharge with RMP ELM suppression.

- Fully non-inductive discharges have been sustained on DIII-D (Fig. A-6) with well aligned current profiles in single null configurations that show promise for ITER with projected Q>5 [Petty 2017].
- RMP-ELM suppression has been shown to be robust and more easily achieved at high β due to the increased 3D field plasma response (also Fig. A-6) [Petty 2017].
- Stable operation above the β-limit predicted by ideal MHD theory has been demonstrated [Strait 1995], with important validations demonstrated of theoretically predicted kinetic stabilizing effects that enable this operation [Reimerdes 2011].
- The mechanism of fast ion transport has been identified as stochastization from overlapping Alfvénic modes leading to a critical gradient behavior. The potential to alleviate this through current profile modification and electron heating has also been established (Fig A-7) [Collins 2016].

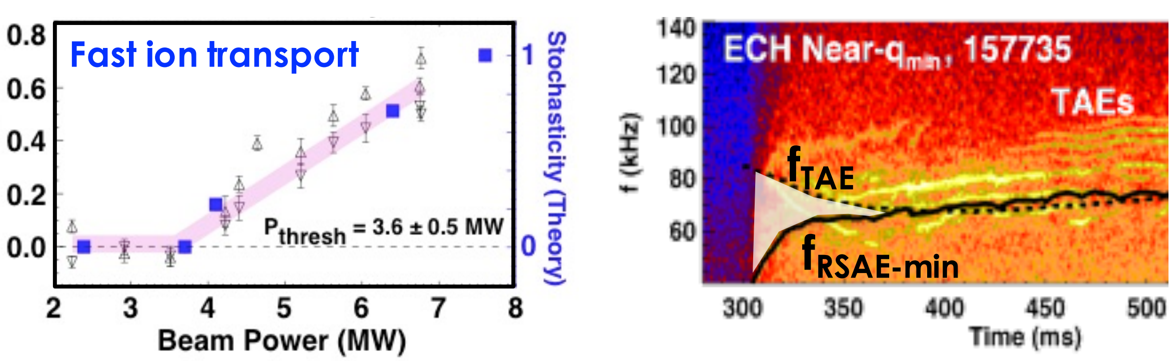

Fig A-7: Fast ion transport exhibits a criticla gradient behavior (left). Electron heating and reduce the windon for Alfvenic instability.

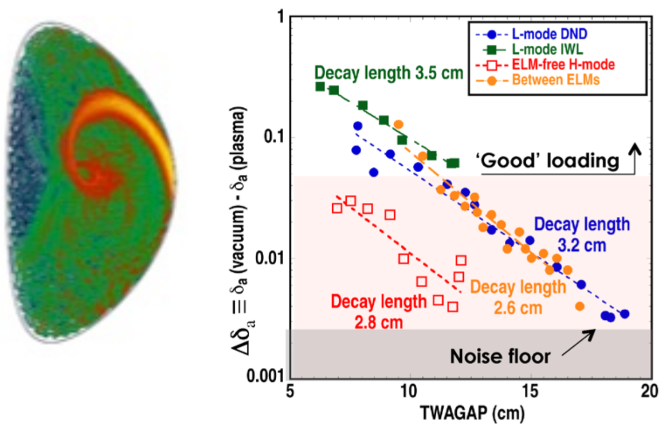

Fig A-8: A low power helicon antenna has demonstrated good coupling in high performance plasmas.

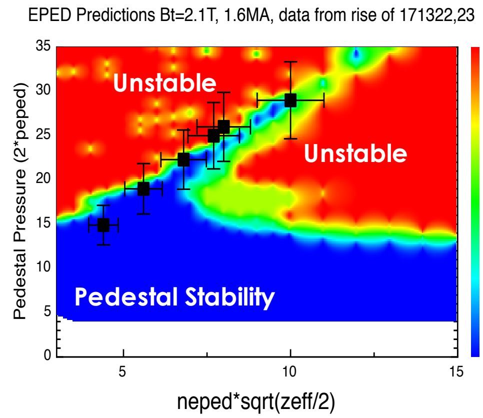

Fig A-9: Prediction (color) and experimental access (black points) to enhanced high density pedestals.

- Radiative divertor techniques have been demonstrated compatible with a high-power core, and are found to lead to *improved* energy confinement through modifications to the H-mode pedestal in high-performance near-double null plasmas [Petrie 2017].
- A new, potentially more efficient actuator for off-axis current drive has demonstrated efficient coupling at low power (Fig A-8) – helicon ultrahigh harmonic fast wave [Pinsker 2016]. High power installation for testing at the MW level is planned in 2018.
- High density "super H mode" pedestals [Solomon 2014] have been sustained in ELMing discharges, leading to record pedestal pressure in DIII-D, with H_{98}~2.5 and absolute densities and temperatures similar to ITER (Fig A-9). This high density approach may be highly levering to a high bootstrap, dissipative divertor solution.

Nevertheless, DIII-D's present tool set has not provided access to configurations with sufficient performance or the reactor relevance necessary to validate the physics and demonstrate the potential for fusion power plants. The fully non-inductive regimes studied thus far have been at β_N and q_{95} below the values necessary to demonstrate the physics of fully noninductive operation in discharges with the high absolute plasma pressure required for a power plant. Capability to further broaden the current density profile, to raise ideal MHD β_N limits and eliminate fast ion transport is required. On the transport side, progress has been made at high β_P and q_{95} with internal transport barriers [Ding 2017], but translation to higher fusion performance in future reactors is also projected to require broader current profiles. Overall heating power needs to be increased in order to access the required range of β_N. Dominant electron heating with zero torque input will enable the study of non-inductive regimes with reactor relevant rotation and T_e/T_i – key parameters governing turbulence and stability. Finally, compatibility between the high-performance core and divertor solutions has yet to be demonstrated. These are all elements that will be dealt with through significant upgrades starting this year, as set out in this paper.