Research Thrust on Plasma Startup & Ramp-up to Enable Fusion Nuclear Science Research at Low-A

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1. Mission and Scope Summary

Developing and understanding techniques to create and grow tokamak plasmas without a central solenoid magnet to the point that they are acceptable targets for heating and current drive techniques will benefit any future magnetic confinement concept based on the tokamak or related concept. The need for such technique is especially pressing for the spherical tokamak, since it leaves little room for an Ohmic transformer or nuclear shielding for the central column.

Development of the research capability to perform fusion nuclear science research has been identified as the ITER-era goal of the ST program [1]. In order to develop such fusion nuclear research capability at low-A, a means of initiating and ramping up the plasma current to the multi-mega-Amp steady state level with minimal to no central induction is required. While NBI current drive is expected to provide some portion of the current ramp-up and, with bootstrap current, to maintain the steady state current, techniques must be developed to reliably produce startup plasmas with sufficient current for NBI fast particle confinement and sufficient density to give an acceptably short NBI particle ionization distance.

The mission of this research thrust is to develop an understanding of and demonstrate a plasma startup and growth scenario sufficient for the fusion nuclear science research program goals in the ITER era. The scope of this effort includes the R&D needed to develop an appropriate combination of DC helicity injection, RF heating, and bootstrap current overdrive to create a robust NBI target. Both DC helicity injection and EBW heating have been demonstrated to create substantial plasmas starting from zero plasma current, making complete elimination of the solenoid magnet from the ITER-era ST goal a realistic possibility.

Coaxial Helicity Injection (CHI) is a DC Helicity injection technique that has generated steady-state plasma currents up to 300 kA in HIT-II and 400 kA in NSTX as well as transient currents up to 100 kA in HIT-II and 160 kA in NSTX that were coupled to induction. Point-source DC helicity injection is a complementary startup technique that can initiate tokamak-like plasmas by driving DC current between a point plasma source and an electrode, both located on the outer edge of the plasma [2]. Driving DC current at the experimentally convenient edge, parallel to the equilibrium magnetic field, injects magnetic helicity, and hence drives current in the core plasma. This technique has proven successful in creating target plasmas with toroidal plasma current in excess of 100 kA in the PEGASUS Toroidal Experiment, which have been subsequently ramped up and sustained by other means, including inductive drive. The scientific challenge here is develop a sufficient understanding of Helicity Injection from both CHI and point-source HI studies to design an integrated approach for HI with attractive facility requirements for a reactor environment.

A technique using the electron Bernstein wave (EBW) has been demonstrated for non-inductive startup experiments on MAST.[3] The scenario relies on a mode conversion process, where microwaves launched in the ordinary mode (O-mode) from the low field side are converted to the extraordinary mode (X-mode) by reflection off a grooved polarizer on the center stack, which then convert to the EBW mode near the upper hybrid resonance layer. For a microwave power of 100 kW at 28 GHz and limited solenoid assist, plasma currents of up to 55 kA have been demonstrated, along with electron temperatures of 700 eV.

The target plasmas created by point-source DC helicity injection appear to be suitable targets for subsequent RF heating and current drive. These targets can be created directly in front of HHFW antennas located at the outboard midplane, with plasma characteristics that allow for > 50% single-pass absorption. Hence, HHFW is a good candidate for heating the initial target plasma, and enhancing the toroidal current through bootstrap overdrive. The target plasmas are also overdense and thus suitable for EBW heating and current drive. Coupling to the EBW via an O-X-B mode conversion process from an obliquely launched microwave antenna is a promising technique.

2. Closing Research Gaps for Startup & Ramp-up

The NBI target conditions must be identified in order to evaluate the suitability of the startup techniques. This requires validated modeling of NBI ramp-up and sustainment, including the effects of fast-ion modes on current drive efficiency. It is expected that the ST ITER-era goal will require startup plasma current on the order of a mega-Ampere. It is assumed that this startup plasma will then be ramped up non-inductively to 8-10 MA, using the same heating and current drive tools required for steady-state sustainment.

The highest currents achieved with DC helicity injection and EBW in existing devices are less than 0.2 MA and are expected to remain below 0.5 MA. This is a factor of 2 below the startup current expected for the ST ITER-era goal. Sufficient understanding of the toroidal field, plasma size, and plasma current scaling of the efficiency of the various start-up techniques combined with validated models of the start-up method physics and plasma equilibrium evolution will allow the required extrapolation.

Physical mechanisms consistent with the observed upper bounds on the plasma current achievable with point-source DC helicity injection are under study. Present research on Pegasus is focused on validating the parametric dependencies of these models in order to provide the needed predictive capability to extrapolate to the ITER-era goal. Modest upgrades to the Pegasus facility will enable a test of this technique up to ~0.3 MA. Experiments with point-source and coaxial helicity injection systems on Pegasus and NSTX will guide the development of this understanding.

Energy confinement during DC helicity injection on Pegasus is consistent with L-mode tokamak confinement scaling, which indicates that cross-field transport dominates over parallel transport. At higher plasma current, the increase in electron temperature could lead to an increase in parallel conductivity and saturation of the sustainable current. This effect could play a major role in determining the maximum plasma current achievable with this technique. The research on Pegasus should identify if and when the onset of significant parallel transport occurs.

Pegasus uses a washer-stack arc discharge to provide a low-impurity plasma source cathode for DC helicity injection. Further research on the current source and accompanying technology is required to develop the required understanding and a system capable of the injected current and pulse length for the ITER-era goal. This research would best be performed on a dedicated test or diagnostic stand integrated into the Pegasus facility.

Point-source HI on the outboard plasma side naturally includes induction from the PF coil systems to aid plasma formation and growth, and the PF induction can account for up to half of the effective loop volts to drive current growth. Beyond this intrinsic PF induction, the target plasmas have been ramped up using Ohmic induction, but it remains to be assessed if these targets can be ramped up using RF or NBI. Pegasus has the capability to test coupling to up to 1 MW of HHFW with some re-commissioning of existing facilities, and could test coupling up to 2 MW of HHFW with expansion of the existing facility, including a second antenna. Test of EBW heating and CD on Pegasus would require a new research effort, in collaboration with interested groups.

While initial startup experiments using EBW are encouraging, a full startup scenario to produce target plasmas suitable for NBI has not yet been demonstrated. Methods for understanding and controlling the current and density evolution, as well as plasma current scaling as a function of launched power, need to be determined. Switching from an EBW based startup phase to a heating and/or current drive phase has yet to been shown. Experiments on current devices, such as MAST or Pegasus, are needed to further the science of this promising technical approach.

Experimental studies on several devices have investigated using induction from PF coils located outside the vacuum vessel, combined with auxiliary heating and current drive to generate startup currents. JT60U has been able to produce 100kA with ECH and outer PF coils alone. NSTX has also produced small transient plasmas utilizing only outside coils. However, experiments on major devices to date have not generated a completely solenoid-free startup scenario to a steady state configuration.

DIII-D is uniquely qualified in this area with high degree of control over the PF coils and substantial auxiliary heating and current drive capabilities using the EC and neutral beam systems. A new magnetic scenario under development for testing in upcoming DIII-D experiments utilizes current ramps in the divertor coils and vertical field coils to produce a first order null in the outboard limiter region. Loop voltages of-order 5V are predicted and should be sufficient for ECH assisted breakdown and current ramp up. The plasma will naturally expand inward and increase in minor radius from the outboard limiter. ECCD and early neutral beam injection will be used to reduce the flux consumption during this phase of startup. Final configuration is expected to be a fully non-inductive H-mode plasma driven by neutral beams to the maximum current of about 600-700 kA possible with the present DIII-D system.

If no combination of DC helicity injection and RF heating and current drive is capable of creating the target for NBI ramp-up to full current, a small solenoid made with mineral insulated conductor could be installed in the ST ITER-era goal. This technology is being evaluated independently in a separate white paper [4].

3. Elements of the Research Thrust

The following research elements should be performed as part of this thrust using the indicated resources:

1) Point-source DC helicity injection – Pegasus with upgrades & plasma gun development capability
   i) determine parametric scaling of sustainable current limit
   ii) develop physical understanding of point-source helicity injection for CD
   iii) develop hardware capable of delivering the power and helicity injection needed for handoff to RF/NBI
   iv) demonstrate startup to $I_p \geq 0.3$ MA
   v) integrate HI concepts from point-source and CHI studies to define an attractive HI injection concept for scaling to larger-scale experiments
2) RF startup, heating and current drive – Pegasus with upgrades, MAST
   i) assess feasibility of HHFW CD in startup/steady-state ramp-up phase
   ii) demonstrate EBW growth to $I_p \geq 0.3$ MA
   iii) determine EBW mode coupling requirements for startup, ramp-up and steady-state
3) Integrated startup experiments – Pegasus with upgrades, MAST, NSTX, DIII-D
   i) test RF ramp-up of DC helicity injection or CHI target
   ii) couple helicity injection and/or RF target to NBI for ramp-up/sustainment
   iii) test RF/NBI ramp-up and sustainment of target with outer PF induction
4) Predictive modeling – NIMROD & SWIM
   i) develop models for integrated startup and ramp-up
   ii) validate models against experimental data

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[1] "Report of the FESAC Toroidal Alternates Panel", Nov. (2008).
[2]"High-current tokamak startup using point-source DC helicity injection," Battaglia, D.J., Bongard, M.W., Fonck, R.J., Redd, A.J., and Sontag, A.C., 2008 submitted for publication.

[3] V. Shevchenko, et al., “28 GHz Start-up System on MAST”, Proceedings of the 34th Annual meeting of the EPS Conference on Plasma Physics, Warsaw, Poland, 2007.
[4] M. Cole, “Research Thrust on Magnet Technology to Enable Fusion Nuclear Science Testing”