SPHERICAL TOKAMAK PHYSICS RESEARCH IN PREPARATION FOR THE OPERATION OF NSTX-U

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Spherical Tokamaks (STs) are an attractive path toward fusion pilot plants (FPPs) and both the public and private sectors are designing FPPs based on such a configuration. As the National Spherical Tokamak eXperiment-Upgrade (NSTX-U) is getting closer to restarting operations, NSTX-U research is focused on closing the scientific and technological gaps for the FPP by reinforcing the advantages of STs, like high performance and stability, while addressing the challenges of high heat fluxes by developing power and particle handling methods.

The understanding of reversed magnetic shear (RMS) operation for improved stability has been advanced. The nonlinear MHD evolution of some NSTX discharges that had non-monotonic safety-factor profiles and peaked pressure profiles at early times was examined using the MHD code M3D-C1. For sufficient pressure, these discharges become linearly unstable to multiple pressure driven modes which flatten the pressure profile and can lead to a nearly monotonic q-profile, in qualitative agreement with experimental results (Fig 1) [1]. The 1.5D equilibrium and transport solver TRANSP was used to develop operating scenarios with fast plasma current ramps and early beam injection into large plasma volumes that should allow the generation of RMS in NSTX-U [2]. Machine learning algorithms have been developed to both identify patterns in the NSTX and NSTX-U discharges that can lead to long-lasting improved confinement with reversed magnetic shear and to help identify precursors of RMS-destroying MHD events [3].

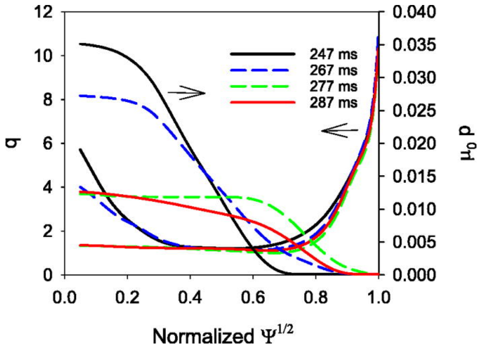

Fig 1: Example of experimental RMS that evolves toward a flattened q-profile [3]

The impact of low and high-frequency instabilities on non-inductive current drive and energetic particle (EP) confinement has been studied. Magnetic-island-induced 3D electric field was shown to drive an electron current. While for conventional aspect ratio tokamaks this may result in a significant global reduction of the electron bootstrap current, the island-induced current loss in STs was shown to be local to the island region, and it was found to be minimal in the reactor-relevant high-β_p regime [4]. Nonlinear simulations of EP driven global Alfvén eigenmodes (GAEs) showed that while the resonant particles driving the instability experience negligible guiding center orbit displacement, GAEs can induce EP losses up to 0.5% of total particle inventory on relatively short time scales ~0.1ms, suggesting that multiple GAE bursts over ~10ms can lead to noticeable EP losses [5].

Performance improvements continued also through edge and core transport related studies. A width-height scaling based on a linear kinetic-ballooning-mode (KBM) threshold model that explained the experimental scaling observed in NSTX was developed [6]. The effect of neutral interactions on parallel transport in the NSTX scrape-off layer was also investigated in a continuum gyrokinetic code that has been coupled to a continuum kinetic model of neutral transport [7]. The role of the parallel magnetic fields ($\delta B_{\parallel}$) on plasma turbulence and transport was investigated using the fully electromagnetic gyrokinetic code CGYRO, revealing that they increase both predicted growth rates and quasilinear fluxes. Moreover, TGYRO predictive modeling indicates that the inclusion of $\delta B_{\parallel}$ significantly improves the accuracy of temperature profile predictions in NSTX high-β plasmas [8]. Gyrokinetic simulations were also performed to study electron temperature gradient stability and thermal transport and results were compared with the theory-based transport multi-mode model (MMM) reduced model code [9].

Tools for improved equilibrium reconstruction and control have been developed. A neural-network surrogate model for the MMM, able to recreate the results of MMM while significantly reducing the computation time, was developed, enabling future applications in real-time estimation and control [10]. Pressure profiles from TRANSP have been used to improve accuracy of EFIT equilibrium reconstructions [11]. A data-driven approach applicable to real-time plasma control systems was developed based on analysis of data from NSTX and other tokamaks and implemented in the Disruption Event Characterization and Forecasting (DECAF) code to improve vertical controllability boundary determination in plasma operational space [12].

FPPs in general, and compact size FPPs in particular, will have extremely high heat fluxes to the plasma facing components (PFC). Using liquid lithium as PFC is a possible solution to withstand them. Its application in NSTX-U has been extensively studied. SOLPS-ITER, the coupling of a multi-fluid plasma transport code with the 3D kinetic Monte Carlo neutral transport code EIRENE, has been used to explore the effect of several design parameters of a lithium vapor cave detached divertor design on lithium concentration at the last closed flux surface, indicating significant flexibility in implementing this concept experimentally for NSTX-U [13]. SOLPS-ITER has been augmented with a liquid metal boundary condition algorithm, allowing direct two-way coupling of the plasma analysis with the two-dimensional analytical slab flow model which includes heat convection in the liquid metal plasma facing component (Fig 2) [14]. This new capability was deployed to show that using the Capillary Porous System with Flowing liquid metal (CPSF) as an evaporator creates an efficient vapor box divertor design. Liquid Lithium (LL) on the plasma-facing surface creates a feedback effect that could self-regulate the amount of lithium evaporated. The resulting temperature distributions indicate the possibility of creating a lithium vapor shield divertor for NSTX-U within the operational window of reduced activation ferritic/martensitic (RAFM) steel using CPSF [15].

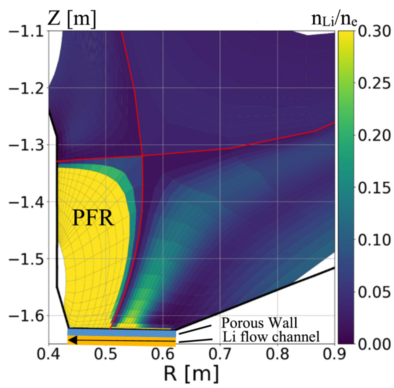

Fig 2: SOLPS analysis results after the coupling with the LL flow model [14].

Divertor heat flux control has also been studied using an X-point radiator regime in discharges with a snowflake divertor [16], as well as the development of new tools for monitoring the heat flux, like the open-source 2D inverse heat flux analysis code HYPERION [17].

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