Progress toward commissioning and plasma operation in NSTX-U *

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The National Spherical Torus Experiment - Upgrade (NSTX-U) project has entered the last phase of construction, and preparation for plasma operations is now underway. A recent aerial view of the NSTX-U test cell is shown in Fig. 1, with the newly installed 2nd NBI beam box. The major mission of NSTX-U is to develop the physics basis for an ST-based Fusion Nuclear Science Facility (FNSF) [1]. The ST-based FNSF has the promise of achieving the high neutron fluence needed for reactor component testing with relatively modest tritium consumption. At the same time, the unique operating regimes of NSTX-U can contribute to several important issues in the physics of burning plasmas to optimize the performance of ITER. The NSTX-U program further aims to determine the attractiveness of the compact ST for addressing key research needs on the path toward a fusion demonstration power plant (DEMO).

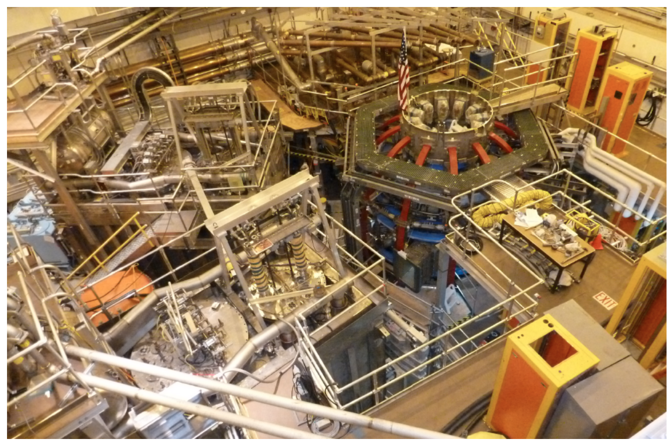

Fig. 1. Aerial view of the NSTX-U Test Cell. The newly installed 2nd NBI beam box can be seen in the foreground.

NSTX-U will nearly double the toroidal magnetic field B_T, plasma current I_p, and NBI heating power compared to NSTX, and increase the TF flat top pulse length from 1 s to 6.5 s. The NSTX-U and NSTX device parameters are given in Table 1. The new center stack, shown in Fig. 2, will provide B_T = 1 Tesla (T) at a major radius of R_0 = 0.93 m compared to 0.55 T at R_0 = 0.85 m in NSTX, and will enable a plasma current I_p of up to 2 mega-Amp (MA) for 5 sec compared to the 1 MA for 1 sec in NSTX. The anticipated plasma performance enhancement is a quadrupling of the plasma stored energy and at least doubling of the plasma confinement time, which would result in an order of magnitude increase in the fusion performance parameter $n\tau T$. With β_T ~ 25% at B_T = 1T, the absolute average plasma pressure in NSTX-U could become comparable to that of present-day tokamaks. A much more tangential 2nd NBI system, with 2-3 times higher current drive efficiency compared to the 1st NBI system, is installed (Fig. 1). NSTX-U is designed to attain the 100% non-inductive operation needed for a compact FNSF design [1, 2]. With higher fields and heating powers, the NSTX-U plasma collisionality will be reduced by a factor of 3-6 to help explore the trend in transport towards the low collisionality FNSF regime. If the favorable trends observed on NSTX holds at low collisionality, high fusion neutron fluences could be achievable in very compact ST devices.

	R_0 (m)	A_{min}	κ	I_p (MA)	B_T (T)	t_{TF} (s)	NBI (MW)	f_{NI}, f_{BS} (%) @1MA
NSTX	0.854	1.28	2.8	1	0.55	1	7	<70, <60
NSTX-U	0.934	1.5	2.8	2	1	6.5	14	≤100, ≤80

Table 1. NSTX and NSTX U Parameters

Several noteworthy technological innovations were introduced for the new center-stack fabrication. The new TF center bundle shown in Fig. 2 utilizes 36 wedge shaped conductors, with friction stir welded stubs at both the lower and upper ends that bolt to flex joints that attach to the outer TF legs. Each conductor includes a cooling passage with a copper tube soldered into a groove on one side of the wedge with a specially developed eutectic (94Sn/4Ag) solder paste formulated with non-ionic flux. All of the NSTX upgrade coils will be vacuum pressure impregnated using the CTD resin system CTD425, which is a cyanate ester epoxy blend with low viscosity to aid the impregnation process and high shear bond strength of 40 MPa at 100 °C. The OH coil is being wound in place on a 2.5 mm thick cylinder of moldable Aquapour mandrel material cast over the cured TF bundle. The Aquapour material is designed to wash away after coil curing, to leave a thin but uniform air gap between the TF bundle and OH coil for thermal expansion and motion between the two coils. The TF bundle will then be supported externally at the top and bottom of the Center Stack to center it within the OH coil. The TF bundle is connected to the outer TF legs using a high strength Full-Hard C15100 Cu–Zr copper alloy flex joint. The flex joint has to accommodate the vertical cyclic movement due to the growth of the TF bundle (~1cm) during operation, while resisting the JxB forces due to 130 kA (at full field) flowing through each joint crossing the TF and PF fields surrounding it. The flex connection was fabricated from a solid U-shaped copper alloy piece by making 15 precision parallel cuts via electric discharge machining. The flex joint has successfully passed a ~ 300k cycle load test. The doubling of magnetic fields and plasma currents increases electromagnetic forces by a factor of four. Accordingly, the vacuum vessel and the support structures were enhanced. To protect NSTX-U from unintended failures due to the power supplies delivering current combinations and consequential forces or stresses beyond the design-basis, a digital coil protection system (DCPS) is being implemented. The DCPS is designed to prevent accidental (either human or equipment failure) overload beyond the design conditions of the structure which the power supply system could generate, even while each individual power supply is operating within its allowable current range. Approximately 100 force-based and 14 thermal algorithms are anticipated with a 200 μs update rate. Redundant current measurements for each coil and plasma current will be provided as inputs. In addition, a unique TF turn-to-turn fault detector will be implemented.

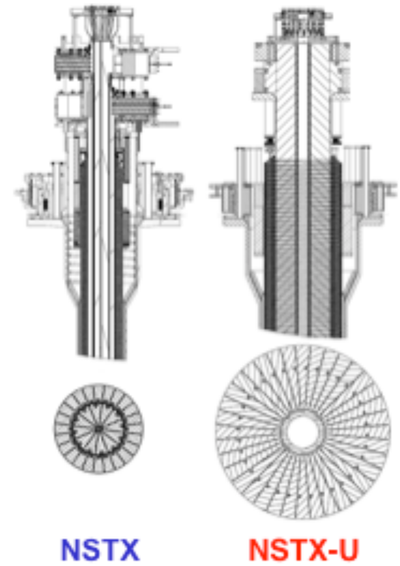

Fig. 2. NSTX and NSTX-U center stack schematics and the TF coil cross sections.

In addition, NSTX-U will have a 6 MW High-Harmonic Fast Waves (HHFW) system for heating and current drive. The HHFW antennas are being modified with compliant feeds for ~ four times larger disruption loads and improved voltage stand-off. The total auxiliary heating power of 20 MW provided by the NBI and HHFW systems will allow NSTX-U to uniquely produce FNSF/DEMO-relevant high divertor heat fluxes of ~ 40 MW/m^2. Innovative divertor heat and particle solutions, such as the up-down symmetric, high flux expansion snowflake divertor configuration, will be used to control these heat fluxes. NSTX-U is also equipped with a set of six non-axisymmetric (3-D) control coils, which can be independently controlled to actively control RWMs at high beta, control error fields, and apply resonant magnetic perturbations for plasma rotation and ELM control. NSTX-U will continue to explore the use of lithium PFC coating techniques for enhanced plasma performance, and divertor power and particle handling. A lithium granular injector will be implemented for ELM pacing at high injection rate to control impurities and reduce the peak ELM heat flux. Coaxial helicity injection (CHI) will be used to create plasmas with up to ~500 kA non-inductively, and develop the solenoid-free tokamak/ST design basis needed for FNSF. If successful, it can also simplify the design of conventional reactors. NSTX-U first plasma is planned for November 2014, at which time the transition to plasma operation will occur.

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[1] J.E. Menard, S. Gerhardt, M. Bell et al., *Nucl. Fusion* **52,** 083015 (2012).
[2] S. P. Gerhardt, R. Andre, and J.E. Menard, *Nucl. Fusion* **52,** 083020 (2012).
[3] L. Dudek, J. Chrzanowski, P. Heitzenroeder, et al., *Fusion Engineering and Design* **87**, 1515 (2012).