Progress toward commissioning and plasma operation in NSTX-U *

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Abstract. The National Spherical Torus Experiment - Upgrade (NSTX-U) is the most powerful spherical torus facility being constructed at PPPL, Princeton USA. The NSTX upgrade construction project has entered its last phase, and preparation for plasma operations is now underway. The major mission of NSTX-U is to develop the physics basis for an ST-based Fusion Nuclear Science Facility (FNSF). 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). Enabled by key technology innovations, the upgrade will nearly double the toroidal magnetic field B_T, plasma current Ip, and NBI heating power compared to NSTX, and increase the TF flat top pulse length from 1 s to 6.5 s. The new center stack 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 Ip 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 $\beta_T \sim 25\%$ 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. NSTX-U is designed to attain the 100% non-inductive operation needed for a compact FNSF design. 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 favourable trends observed on NSTX holds at low collisionality, high fusion neutron fluences could be achievable in very compact ST devices. NSTX-U first plasma is planned for February 2015, at which time the transition to plasma operation will occur.

1. Introduction

A significant upgrade of the National Spherical Torus Experiment facility [1,2] has entered its last construction phase, 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) [3]. 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 also aims to determine the attractiveness of the compact ST for addressing key research needs on the path toward a fusion demonstration power plant (DEMO).

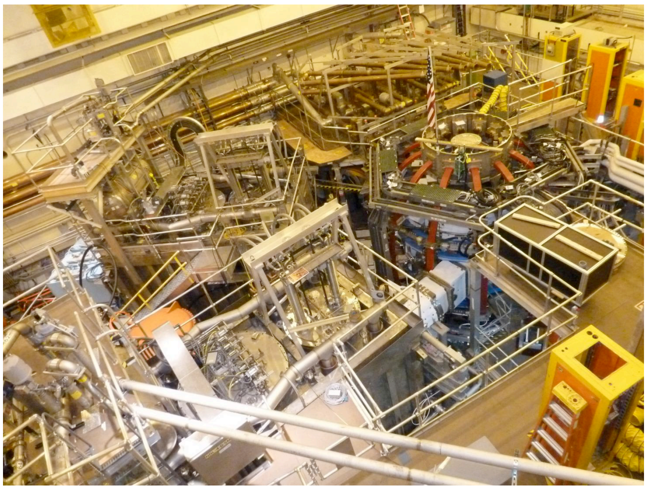

FIG. 1. Aerial view of the NSTX-U Test Cell. The newly installed 2nd NBI beam box can be seen in the foreground.

2. Four Major Mission Elements of NSTX-U

The NSTX-U facility is designed to address four major issues for tokamak/ST-based reactor systems [1,2]:

1. Demonstration of stability and control for steady-state high β plasmas. With the addition of the strongly tangential 2nd NBI beam system, NSTX-U is designed to attain 100% non-inductive operation at reactor relevant high $\beta_N \sim 5 - 6$. NSTX-U is also equipped with a set of six non-axisymmetric (3-D) control coils, which can be independently powered to actively control RWMs at high beta, control error fields, and apply resonant magnetic perturbations for plasma rotation and ELM control.

2. Plasma energy confinement physics at low plasma collisionality, especially electron energy transport. 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. The understanding of electron transport physics is especially critical for predominantly electron heated reactor plasmas, including ITER. If the favourable trends observed on NSTX hold at low collisionality, high fusion neutron fluences could be achieved in very compact ST devices.

3. Divertor solutions for mitigating high heat flux. The total auxiliary heating power of 20 MW provided by the NBI and HHFW systems will allow NSTX-U to uniquely produce reactor-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 will also 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.

4. Non-inductive start-up, ramp-up, and sustainment. With relatively small or no central OH solenoid capability expected in a compact lower aspect-ratio tokamak and/or ST reactors, NSTX-U is implementing a unique set of current drive tools. Coaxial helicity injection (CHI), together with ECH heating, will be used to create hot target plasmas with up to ~500 kA non-inductively. Target plasmas thus formed will be further heated with HHFW and NBI for current ramp-up and sustainment of ~ MA level plasma current, with ~ 70-80% bootstrap current fractions. If successful, it can also simplify the design of conventional reactors.

3. New Center-Stack and 2nd NBI Upgrade

NSTX-U will nearly double the toroidal magnetic field B_T, plasma current I_p, and NBI heating power compared to NSTX [4], 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 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 a doubling of the plasma confinement time, which would result in an order of magnitude increase in the fusion performance parameter nτT. With β_T ~ 25% at B_T = 1T, the absolute average plasma pressure in NSTX-U could become comparable to that of leading present-day tokamaks. A much more tangential 2nd NBI system, with 2-3 times higher current drive efficiency compared to the 1st NBI system, has been installed (Fig. 1).

	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

3.1. New Center-Stack Fabrication

The new center-stack 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 connections to the outer TF legs [5]. 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 were vacuum pressure impregnated using the CTD resin system CTD 425, which is a 3 cyanate ester-epoxy blend with long service life and low viscosity to aid the impregnation process. The shear bond strength of the cured epoxy-glass composite is 40 MPa at 373 degrees K. After completing the TF bundle in 2013, the winding of the OH coil over the full TF bundle has commenced in 2014. After the OH winding was completed in June 2014, a VPI of the entire TF-OH center-bundle was successfully conducted as shown in Fig. 3. It should be noted that after VPI, it was found that the epoxy penetrated a material called Aquapour that was intended to maintain an air gap thickness of 0.1” between the OH and TF coils. This epoxy saturation of Aquapour made its removal by simply dissolving with water no longer possible. The impact of leaving the Aquapour in place on the NSTX-U operations was evaluated. The axial tension stress in the OH, generated due to the thermal growth of coils must be controlled by keeping the OH coil temperature at or above that of the TF. Based on the analyses of the NSTX-U plasma scenarios, it was concluded with a high degree of confidence that all of the NSTX-U physics objectives can be met with Aquapour in place. The presence of the Aquapour has some advantages of providing robust centering support for OH with respect to TF. The decision

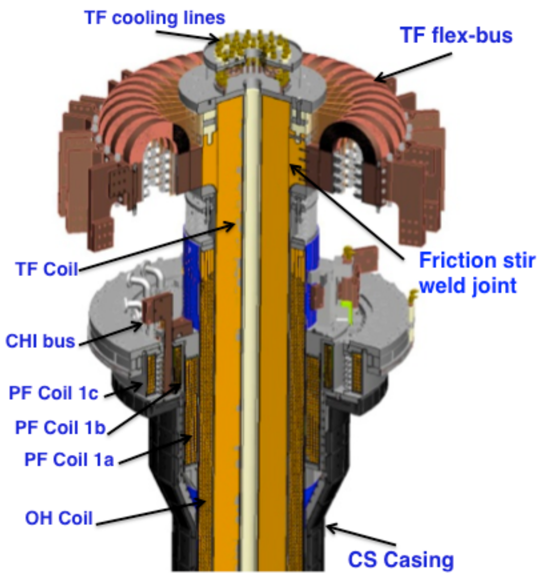

FIG. 2. A schematic of the new center-stack and the TF joint area..

therefore was made to leave the material in place. The TF and OH coils were electrically tested successfully to full test voltages of 4.5 kV and 13 kV, respectively. In parallel, the Center-Stack Casing was completed with the welding of the 700 inconel studs for mounting the carbon tiles to the walls. The graphite tiles (PFC) were mounted with surface diagnostics to the casing walls. The inner poloidal field (PF) coils were manufactured and are being mounted. The completed OH/TF bundle and CS casing will be transported to the NSTX-U for final assembly. Delivery of the completed CS Assembly is scheduled for the fall of 2014. The TF bundle is connected to the outer TF legs using a high strength Cu–Zr copper alloy flex joint. The flex joint has to accommodate the vertical cyclic movement due to the growth of the TF bundle (~ 1.7 cm), while resisting the jxB force of ~ 130 kA of current through the joint. The flex connection was fabricated from a solid U-shaped copper alloy piece by making 16 precision parallel cuts via electric discharge machining. The flex joint has successfully passed a ~ 300,000 cycle fatigue test.

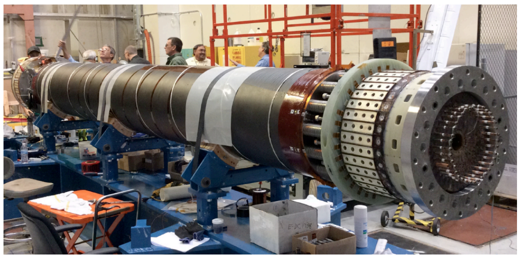

FIG. 3. Completed NSTX-U center TF-OH bundle.

3.2. NSTX-U Device Structural Enhancements

In order to handle the anticipated 4x greater electromagnetic forces for NSTX-U, the vacuum vessel and associated magnetic field coil support structures must be enhanced accordingly. The umbrella structure reinforcements, and PF 2/3 support upgrade hardware and PF 4/5 support upgrade hardware enhancements were completed. Two new outer TF legs were fabricated and installed to replace the ones with cooling water leak and electrical insulation issues. The outer TF leg support upgrades were also fabricated and installed. The TF-VV clevises to better support outer TF legs were welded onto the vessel. The new umbrella legs were installed on the machine. The vacuum vessel leg attachment connections were modified to clear the clevises. In addition, since the plasma disruption forces are also expected to increase by a factor of 4, the internal passive plates were reinforced by replacing the stainless steel attachment hardware with Inconel versions. The support structure enhancement activities are now complete.

3.3. Digital Coil Protection System

To protect NSTX-U from unintended operational conditions 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 [6]. 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. In the initial instance of DCPS, the algorithms will test approximately 125 force and stress calculations against limit values, using both two models for the plasma shape and two models for potential post-disruption currents; this results in 500 total force/stress calculations in addition to 14 thermal limit calculations. The update rate of each type will be 200 μs for both the force-based and the thermal-based signals. Redundant current measurements for each coil and plasma current will be provided as inputs. This type of sophisticated coil protection system if fully demonstrated could be utilized for safe operations of future fusion devices including ITER.

3.4. Neutral Beam Injection System Upgrade

The 2nd NBI upgrade scope is to add a complete, functional second beam-line (BL) to NSTX-U at aiming tangency radii of 110, 120, and 130 cm compared to 50, 60, 70 cm for the present 1st NBI. A schematic of the present and new 2nd NBI systems are shown in Fig. 4. This task largely utilizes the existing TFTR NBI infrastructure. The 2nd NBI tasks include the TFTR NBI BL tritium decontamination, refurbishments, sources, relocation, services, power and controls, and NSTX-U Test Cell (NTC) arrangements. In addition, there are vacuum vessel (VV) modifications, the NBI and Torus Vacuum Pumping System ducts, and NBI armour. The 2nd NBI BL (BL2) refurbishment and relocation have been completed and the jobs closed. The BL box and lid were moved into the NTC and reassembled. The installation of the support structure and alignment of the BL has been completed. The refurbished 90 inch flange, ion dump, calorimeter, and bending magnet were installed on the BL. The source platform has been fully decontaminated and installed. Relocation also included moving three High Voltage Enclosures (HVEs) from the TTC Basement into the NTC. The High Voltage Transmission Lines were refurbished and relocated to the NTC where they were installed on the HVE and Source connections. The three NB Ion Sources were installed on the BL2 platform and vacuum connections completed. The BL2 and its sources are ready for vacuum pump down. With the successful completion of the above tasks, preoperational testing of BL systems and BL2 power and controls has started to properly commission subsystems and confirm readiness for operations. This testing has begun in earnest in 2014 and flows into start up operations based on project schedules. At present the cryogenics start-up is scheduled for October 2014. Due to safety precautions, the BL2 pump down is scheduled after installation of the VV CS and closure of the VV at the end of November. The beam preoperational test procedure low voltage conditioning is scheduled to follow in early December. Pending appropriate approvals, the 2nd NBI upgrade project is scheduled to complete in January, 2015.

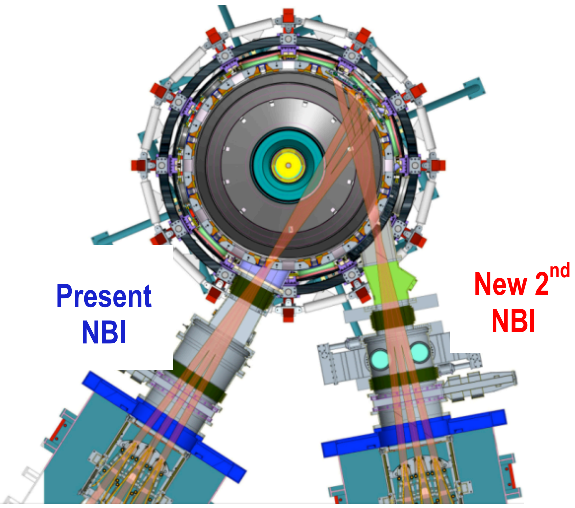

FIG 4. Top view of the NSTX-U with the 2nd tangential NBI.

4. Additional Science Facility Enhancements

4.1 HHFW for Current Ramp-up

A 6 MW High-Harmonic Fast Waves (HHFW) is also being prepared for electron heating and current ramp-up [7]. While the HHFW system is basically unchanged, the antenna feed-thru conductors were modified to be able to handle the higher disruption loads (~ x4) in NSTX-U.

To handle those disruption loads, compliant connectors were designed, tested, and installed between the feed-throughs and antenna straps for the NSTX-U operations. In order to increase the power from the existing 12-strap HHFW antenna, the rf voltage stand-off was tested on an rf test stand with the new compliant feeds. The test demonstrated rf voltage stand-off of 46 kV, which is about twice the value required. The HHFW antenna system with the compliant feeds and improved back plate grounding were installed in NSTX-U, as shown in Fig. 5. The tests also showed rf-induced arc-prone areas behind the back-plate. For improved rf diagnostics, rf probes and tile sensors were also installed.

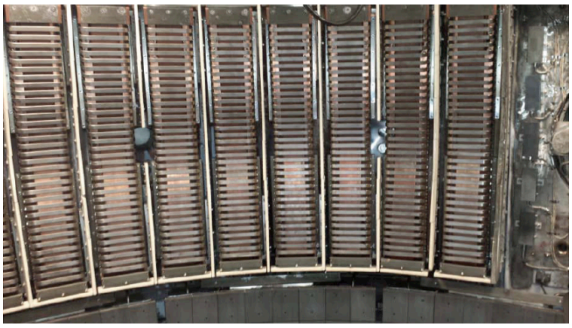

FIG. 5. Enhanced HHFW antenna installed in NSTX-U with compliant feeds and improved back-plate grounding.

4.2 Tools for High-beta / High Bootstrap Current Fraction Operations

To support high beta NSTX-U operation, a number of macro-stability tools are being prepared. Mid-plane RWM control coils, and equilibria require re-computation of n = 1 active RWM control performance using proportional gain, and RWM state space control. The upgrade also adds new capability, such as independent control of the 6 RWM coils with 6 switching power amplifier (SPA) sources. This new capability, combined with the upgrade of the RWM state space controller will also allow simultaneous n = 1 and n = 2 active control, along with n = 3 dynamic error field correction. Finally, the active control performance of the proposed off-mid-plane non-axisymmetric control coils (NCC) also needs to be evaluated. A significant increase in controllable β_N is expected with the RWM state space control in NSTX-U, as was found for NSTX.

Disruption Mitigation Systems - A key issue for ITER, and the tokamak/ST line of fusion devices in general, is the avoidance and mitigation of disruptions. Most of the disruptions are expected to be mitigated by massive gas injection (MGI). In support of the planned MGI Experiments on NSTX-U, the University of Washington has successfully built and tested an electromagnetic MGI valve for installation on NSTX-U. The operating principle of the valve is similar to the valve that is being planned for ITER. The lower divertor valve and the mid-plane MGI valves will be installed on NSTX-U early in 2015. The valve is at present undergoing off-line tests at the Univ. of Washington. After these tests are completed later this year, three such valves will be built for installation on NSTX-U.

4.3 Non-Inductive Start-Up with CHI

For non-inductive start-up, the coaxial helicity injector (CHI) is prepared to support plasma start-up current of well over the 400 kA required to couple a CHI started discharge to non-inductive current ramp-up [8]. The baseline PFCs for the initial NSTX-U operation are graphite tiles. Because of the increased plasma heat loads due to the increased NBI heating power and pulse duration, it was decided to enhance the protection of the "CHI Gap". The CHI Gap is the region between the NSTX-U inner and outer vacuum vessels, above or below the CHI insulators. The graphite tiles on both the inner and outer divertors will be extended downwards, coming in close contact to the PF-1C coil casing and stainless steel outer vessel flanges and shielding these components from plasma contact. This narrower and deeper CHI gap will protect the vessel and PF-1C coil from excessive heat flux and protect the plasma from metal contamination, while continuing to provide the capability for CHI operations. The final tile installation is planned in the fall after the installation of the new center-stack.

4.4 Novel Power Exhaust Solutions

In NSTX-U, the projected peak divertor heat fluxes can reach 20-40 MW/m^2 with $I_p \leq 2$ MA, $P_{NBI} \leq 12$ MW, with pulse length up to 5 sec. NSTX-U will explore novel solutions to boundary physics power exhaust challenge by testing so-called "snowflake" (SF) divertor configuration, and liquid metal plasma facing components to mitigate erosion and melting problems.

Snowflake Divertors - In NSTX-U, two up-down symmetric sets of four divertor coils will be used to test snowflake (SF) divertors [9] The modeling projections for the NSTX-U SF divertor geometry are favourable, showing large reductions in divertor T_e and T_i, as well as peak divertor heat fluxes due to the geometric and radiation effects, both with 4% of carbon impurity levels and with neon or argon seeding.

Lithium Application Tools - With encouraging results in NSTX [10], a number of lithium application tools are being prepared for NSTX-U. Two downward-oriented lithium evaporators (LITERs) will be re-installed. An electron beam heated upper aiming evaporator to cover the upper divertor region is being developed. The electron beam enables Li to be promptly evaporated for providing fresh lithium coatings. A lithium granular injector (LGI) for ELM pacing which was successfully demonstrated on EAST, is being prepared for NSTX-U. The NSTX-U LGI system is capable of injecting horizontally-redirected spherical lithium granules (0.6 mm) at speeds approaching 100 m/s. It is anticipated that much higher pacing frequencies can eventually be achieved using the basic injector technology. Granule feeding rates (pacing frequencies) of 500 Hz have been achieved in the laboratory tests.

5. NSTX-U Plasma Operation Start-Up Planning

An operational plan toward full NSTX-U operational capability is being developed. A draft plan is shown in Table 2, based on assessment of physics needs for the first year of operations. The 1st year goal is to operate NSTX-U with the electromagnetic forces (I_pB_T) at halfway between NSTX and NSTX-U limits and 50% of the NSTX-U design-point of heating of the TF coil. This still allows NSTX to operate at $B_T \sim 0.8$ T, $I_p \sim 1.6$ MA, and a maximum flat-top duration of 3.5 s in the first year, which is far beyond the achieved NSTX parameters.

The device will be inspected and refurbished as needed at the end of the each operating year. For the second year, the toroidal magnetic field will be increased to its full field value of 1 T. but with the heating of the TF coil kept to 75% of the design-point. This will allow 3 sec discharges at full field and current. The same limits should allow the full 5 sec discharges at $B_T \sim 0.8$ T and $I_p \sim 1.6$ MA. The device will be brought to full operational capability in the third year of NSTX-U operation.

	NSTX (Max.)	FY 2015 NSTX-U Operations	FY 2016 NSTX-U Operations	FY 2017 NSTX-U Operations	Ultimate Goal
I_p [MA]	1.2	~1.6	2.0	2.0	2.0
B_T [T]	0.55	~0.8	1.0	1.0	1.0
Allowed TF I^2t [MA^2s]	7.3	80	120	160	160

TABLE 2: NSTX-U Device Ramp-up Plan.

6. Discussions and Conclusions

The NSTX upgrade construction project has entered its last phase, and preparation for plasma operations is now underway. The major mission of NSTX-U is to develop the physics basis for a compact ST-based Fusion Nuclear Science Facility (FNSF). 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). The new center-stack 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 Ip of up to 2 mega-Amp (MA) for 5 sec compared to the 1 MA for 1 sec in NSTX. A much more tangential 2nd NBI system, with 2-3 times higher current drive efficiency compared to the 1st NBI system, has been installed. NSTX-U is designed to attain the 100% non-inductive operation needed for a compact FNSF design. NSTX-U first plasma is planned for February 2015, at which time the transition to plasma operation will occur.

***Acknowledgments:** This work was supported by DoE Contract No. DE-AC02-09CH11466.

References:

[1] MENARD, M.K., GERHARDT, S.P., BELL M., et al., *Nucl. Fusion* **52,** 083015 (2012).
[2] GERHARDT, S.P, ANDRE R., MENARD J.E., *Nucl. Fusion* **52,** 083020 (2012).
[3] PENG, Y-K.M., CANIK, J.M., DIEM, S. J., et al., *Fusion Sci. Technol.* **60,** 441 (2011).
[4] ONO, M., KAYE, S. M., PENG, Y.-K. M., , et al., *Nuclear Fusion* **40**, 557 (2000).
[5] DUDEK, L., et al., *Fusion Engineering and Design* **87**, 1515 (2012).
[6] ERICKSON, K.G., et al., *IEEE Transactions on plasma science* **42**, 1811 (2014).
[7] ELLIS, R., et al., *AIP Conference Proceedings, 20th Topical Conference on Radiofrequency Power in Plasmas,* **1580**, 350 (2014).
[8] RAMAN, R., et al., *IEEE Transactions on plasma science* **42**, 2154 (2014).
[9] SOUKHANOVSKII, V.A., et al., 2012 *IAEA Fusion Energy Conference* EX/P5-21.
[10] ONO, M., et al., *Fusion Engineering and Design* **87**, 1770 (2012).