Plasma Phys. Control. Fusion **43** (2001) 305–312 www.iop.org/Journals/pp PII: S0741-3335(01)16811-9

Initial results from coaxial helicity injection experiments in NSTX

R Raman[1], T R Jarboe[1], D Mueller[2], M J Schaffer[3], R Maqueda[4], B A Nelson[1], S Sabbagh[5], M Bell[2], R Ewig[1], E Fredrickson[2], D Gates[2], J Hosea[2], H Ji[2], R Kaita[2], S M Kaye[2], H Kugel[2], R Maingi[6], J Menard[2], M Ono[2], D Orvis[1], F Paoletti[5], S Paul[2], M Peng[6], C H Skinner[2], J B Wilgen[6], S Zweben[2] and the NSTX Research Team

[1] University of Washington, Seattle, WA 98195, USA
[2] Princeton Plasma Physics Laboratory, Princeton, NJ 08543, USA
[3] General Atomics, San Diego, CA 92186, USA
[4] Los Alamos National Laboratory, Los Alamos, NM 87545, USA
[5] Columbia University, New York, NY 10027, USA
[6] Oak Ridge National Laboratory, Oak Ridge, TN 37830, USA

E-mail: raman@aa.washington.edu

Received 5 September 2000

Abstract

Coaxial helicity injection has been investigated on the National Spherical Torus Experiment (NSTX). Initial experiments produced 130 kA of toroidal current without the use of the central solenoid. The corresponding injector current was 20 kA. Discharges with pulse lengths up to 130 ms have been produced.

1. Introduction

A spherical torus (ST) is a magnetic confinement concept that has the potential advantages of high beta, high bootstrap current fraction and increased stability. These favourable properties of the ST arise from its very small aspect ratio ($A \leqslant 1.5$). Experimental results from the Small Tight Aspect Ratio Tokamak (START) demonstrated some of the favourable high-beta properties with good confinement [1, 2]. This motivated the construction of two large STs, NSTX in the USA [3, 4] and the Mega Ampere Spherical Torus (MAST) in the UK [5]. To minimize the aspect ratio, elimination of the central solenoid is a consideration for future ST designs. This requires the demonstration of plasma creation and sustainment by non-inductive current drive methods. Coaxial helicity injection (CHI) is a promising candidate for generating a target plasma for non-inductive current drive methods and for driving edge current (inside the last-closed flux surface) during the sustained phase.

The first experiments on helicity injection current drive in a ST were conducted on the Current Drive Experiment-Upgrade (CDX-U) at the Princeton Plasma Physics Laboratory (PPPL) [6]. CHI was initially used for the generation of spheromak plasmas [7–9]. The possibility of using CHI in a ST was first proposed in the late 1980s [10]. The idea gained

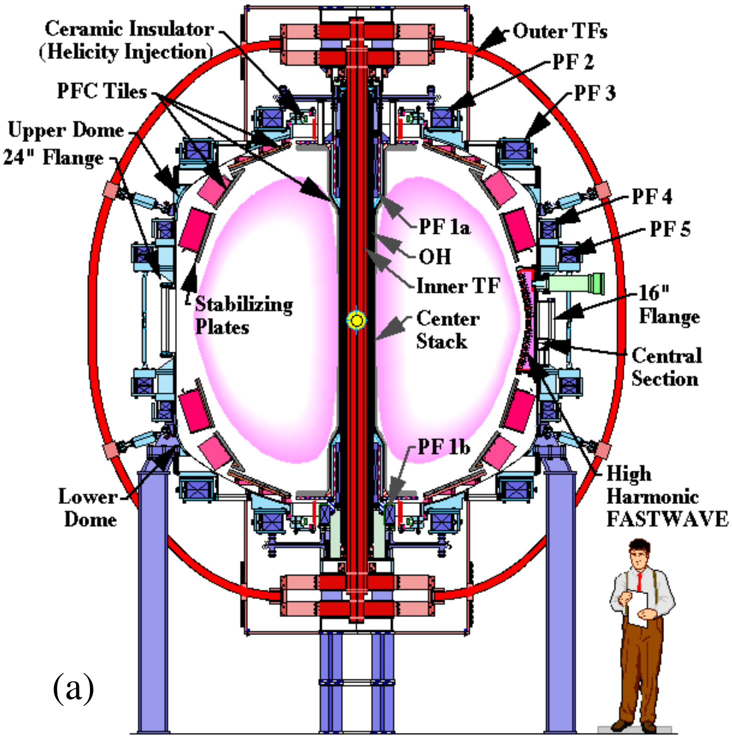

Figure 1. (a) The NSTX machine layout. The PF1B coil is a CHI specific coil used to produce the injector flux. (b) Exploded view of the NSTX lower divertor region showing the CHI components and a representative injector flux connecting the lower divertor plates.

support as a result of experiments conducted on the Proto-Helicity Injected Torus, and the Helicity Injected Torus-I (HIT-I) at the University of Washington [11]. These experiments used a thick conducting copper wall for equilibrium control of the CHI produced plasma configuration. These were followed by two other experiments, the Himeji Institute of Technology Spherical Torus (HIST) in Japan and the SPHEX device (operated in a ST mode) in the UK [12, 13]. These devices also employed passive wall stabilization for equilibrium control and confirmed that CHI could be used in the presence of an external toroidal field. Later, HIT was rebuilt as the HIT-II experiment, which extended CHI to a true ST device by employing 28 poloidal field coils for equilibrium control instead of from image currents induced on copper walls [14]. HIT-II produced substantial plasma currents (200 kA) using transformer action or using CHI [15]. An important goal for the NSTX programme is to conduct proof-of-principle experiments to validate this concept on a device that has a volume about 30 times that of HIT.

The CHI method drives current initially on open field lines, creating a current density profile that is hollow. Taylor relaxation [16] predicts a flattening of this current profile through a process of magnetic reconnection, leading to current being driven throughout the volume, including closed field lines. Current penetration to the interior is eventually needed for usefully coupling CHI to other current drive methods and to provide CHI produced sustainment current during the long-pulse non-inductive phase.

For a ST, the primary goal of a CHI system is to generate a suitable target plasma that can be ramped up in current to a high bootstrap current fraction regime using non-inductive current

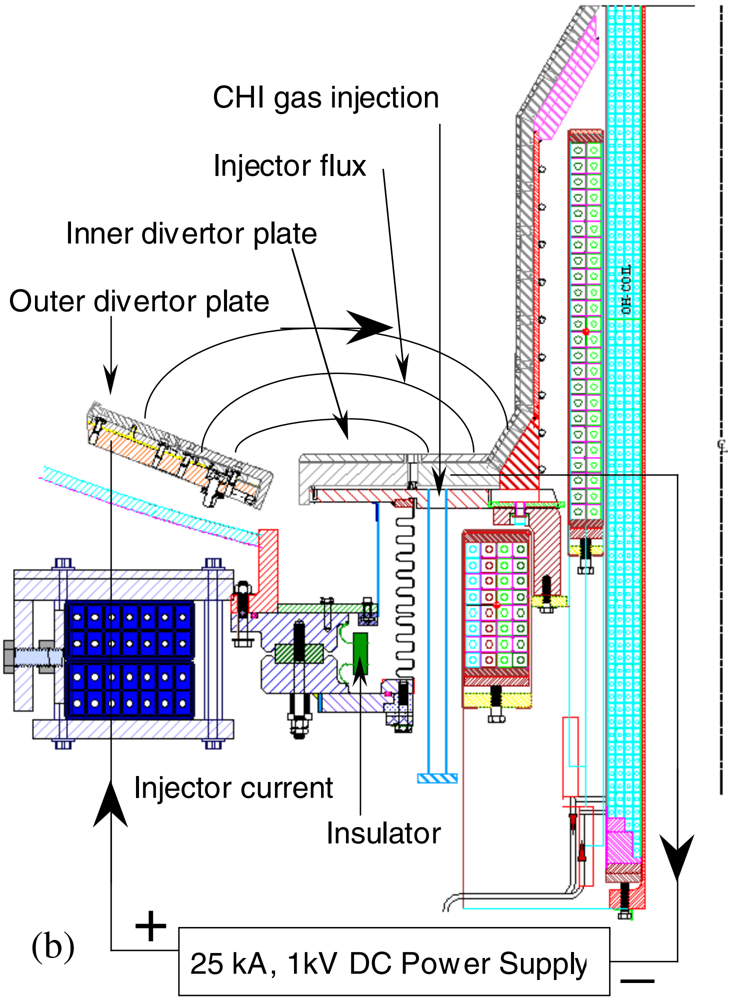

Figure 1. (Continued)

drive methods and without the use of an Ohmic coil. On NSTX, this requires the CHI system to produce target plasmas with a temperature of about 200 eV. This temperature is needed for coupling to the high harmonic fast wave (HHFW) current drive. A secondary goal is to extend the pulse duration of 1 MA Ohmic discharges by producing target plasma for Ohmic discharges.

This paper is organized as follows. Section 2 describes the CHI components on NSTX. In section 3 we describe the experimental results obtained thus far; the final section is a brief summary of the results.

2. Experiment description

The nominal NSTX machine parameters are major/minor radii of 0.85/0.65 m, elongation $\leqslant$2.2, a plasma volume of 14 m^3 and a machine volume of 30 m^3 [3]. As on the HIT device, the stainless-steel vacuum vessel of NSTX is fitted with toroidal ceramic breaks at the top and bottom so that the central column and the inner divertor plates (the inner vessel components) are insulated from the outer wall and the outer divertor plates. Four pairs of poloidal field (PF) coils placed symmetrically above and below the midplane outside the vacuum vessel are available for equilibrium control. The lower divertor region has an additional coil positioned below the inner divertor plate. This and the lowest PF coil allows one to set up the vacuum CHI injector flux in a manner such that it connects the lower inner and outer divertor plates. The coils allow for the generation of up to 500 mWb of injector flux. The corresponding maximum toroidal flux (TF) for a nominal 0.3 T discharge is 1.5 Wb.

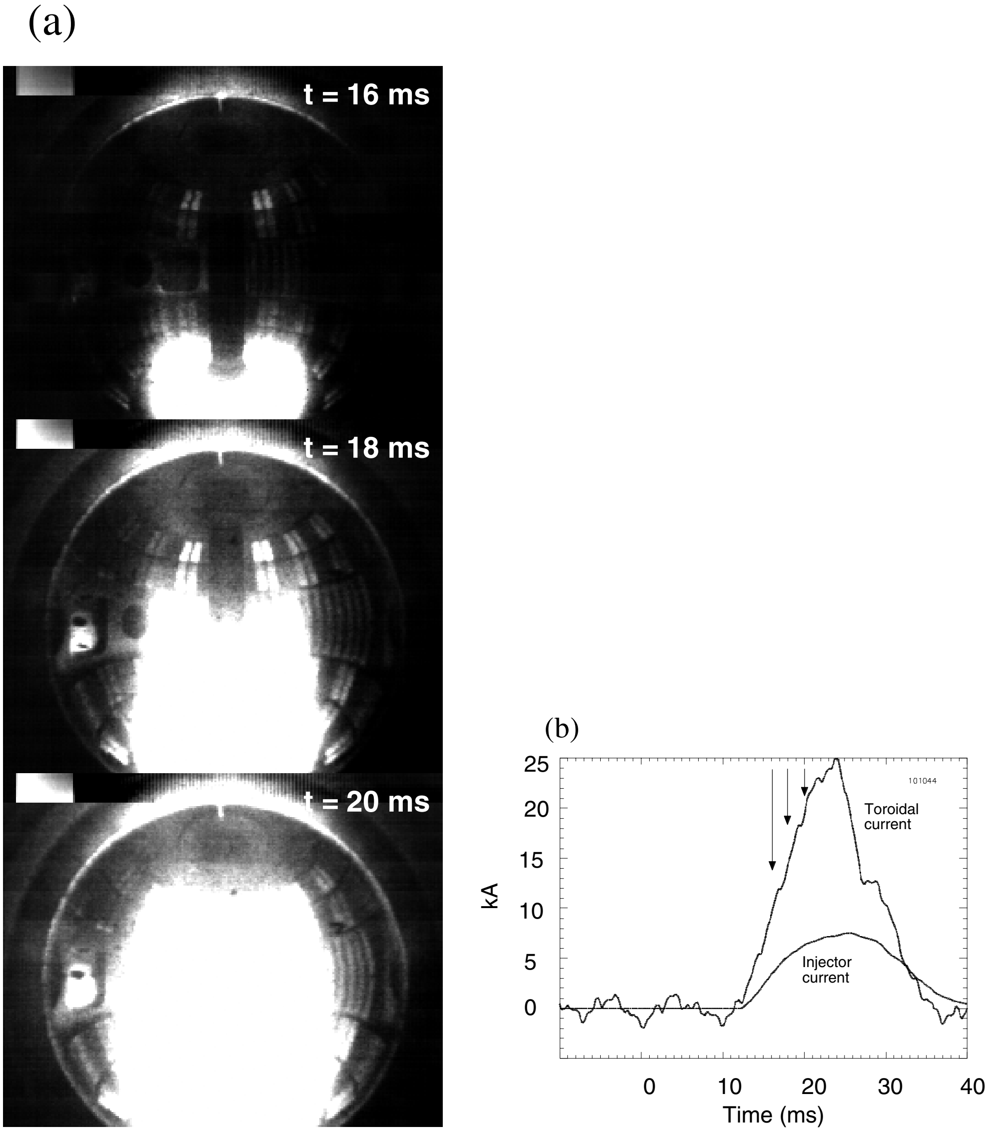

Figure 2. (a) Fast camera fish-eye-view images show that the CHI produced plasma starts in the lower divertor region at 16 ms, elongates to fill half the vessel at 18 ms and then nearly fills the vessel at $t = 20$ ms (SN 101044). (b) CHI injector current and CHI produced toroidal current. The arrows indicate image times.

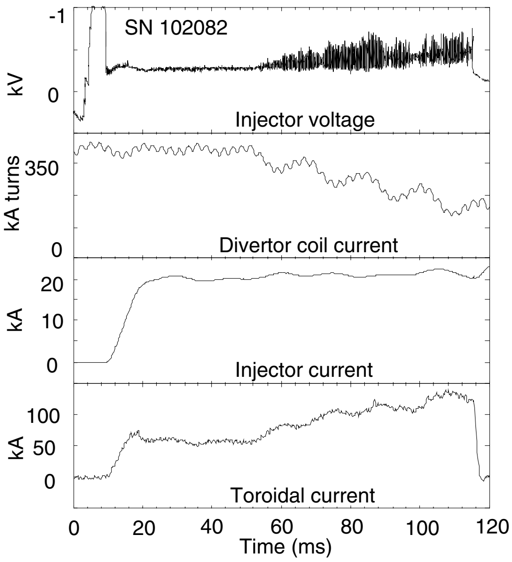

Figure 3. High-current CHI discharge on NSTX using preprogrammed coil currents for equilibrium control.

CHI is implemented on NSTX by driving a current the along the field lines that connect the inner and outer lower divertor plates. A 25 kA, 1 kV dc power supply is connected across the inner and outer vessel components, as shown in figure 1, to drive the injector current. The standard operating condition for CHI in NSTX uses the inner vessel and inner divertor plates as the cathode. The outer vessel is the anode. A dedicated gas injection system in the lower divertor region injects gas from four ports in the lower inner divertor plates toroidally separated from each other by 90°. For the initial CHI experiments, a fixed volume plenum was filled to a known pressure and quickly emptied into the divertor region by opening a fast valve. To increase the speed of the gas delivery system, a total of four plenums and, correspondingly, four fast valves are used, one for each of the four gas injection ports located in the cathode region. The initial fast puff transiently produces a high gas pressure in the lower divertor region. This facilitates gas breakdown when a voltage is applied to the lower divertor plates. An 18 GHz, 10 kW electron cyclotron heating pre-ionization system (ECH-PI) is used to produce a vertical electron cyclotron (EC) resonance layer near the centre stack (at a radius of 40 cm). The EC resonance layer intersects the inboard injector flux footprints, facilitating gas breakdown at the lower fill pressures.

The operational procedure involves first energizing the TF coils and the CHI injector coils to produce the desired flux conditions in the injector region. A voltage is now applied to the inner and outer divertor plates and a preprogrammed amount of gas is injected from the inner lower divertor plate ports. ECH-PI is turned on for a preprogrammed time duration. These conditions cause the gas in the lower divertor region to ionize and result in current flowing along magnetic field lines, connecting the lower divertor plates. The applied toroidal field causes the current in the plasma to develop a strong toroidal component, the beginning of the desired toroidal plasma current. If the injector current exceeds a threshold value, the resulting $\Delta B_{\mathrm{tor}}^2$, $(\boldsymbol{J}_{\mathrm{pol}} \times \boldsymbol{B}_{\mathrm{tor}})$, stress across the current layer exceeds the field line tension of the injector flux, causing the helicity and plasma in the lower divertor region to move into the main torus chamber. Once extended into the vessel, currents need to be driven in the PF coils for equilibrium position control.

3. Experimental results

The fact that this method works on a large ST is shown in figure 2. First, figure 2(a) shows the evolution of a CHI discharge as recorded by a fast framing camera [17]. These fish-eye-view images of the entire NSTX vessel show that at $t = 16$ ms, the discharge originates in the lower divertor region. As the injector current (shown in figure 2(b)) increases, the fluxes connecting the lower divertor plates are extended into the confinement chamber (at $t = 18$ ms). At even higher injector currents, the discharge nearly fills the entire chamber (at $t = 20$ ms). In this discharge a peak toroidal current (figure 2(b)) of 20 kA is obtained for an injector current of about 7 kA, resulting in a current multiplication of about three. The injector current is the current supplied by the 1 kV CHI power supply that flows through the plasma load. The toroidal current is that measured by the NSTX plasma current measurement system.

Figure 3 shows the applied injector voltage, divertor coil current, injector current and CHI produced toroidal current, for a high-current discharge. The divertor coil current is a measure of the poloidal flux (the injector flux) that connects the lower divertor plates (see figure 1). For the case where the divertor coil currents are in the same direction and linearly proportional to each other the injector flux is nearly linearly proportional to the current being driven in the CHI divertor coil. The applied voltage determines the amount of injector current that can be driven for this combination of injector flux and divertor gas pressure.

This discharge has two phases. For times less than 50 ms (phase 1), the applied voltage and the divertor coil currents are constant. This results in an injector current of 20 kA and a toroidal current of 50 kA (a current multiplication factor of about two). To increase the current multiplication factor, the divertor coil current is ramped down during phase 2 ($t > 50$ ms). This results in reduced injector flux and, consequently, the injector impedance increases. To compensate, the injector voltage is increased as shown in figure 3. The programmed voltage ramp maintains a constant injector current. Since the injector flux is decreasing with time, while the toroidal flux is constant, for reasons described in the next paragraph one expects the current multiplication factor to increase. Indeed, the toroidal current increased during the course of the discharge, resulting in a maximum of about 130 kA at $t = 100$ ms, a current multiplication factor of 6.5. The discharge for its entire duration was maintained in equilibrium using preprogrammed coil currents only.

Energy conservation and helicity balance require that the ratio λ_{ST} of the obtained toroidal current to the toroidal flux of the discharge in the ST ($\lambda_{\mathrm{ST}} = \mu_0 I_{\mathrm{toroidal}}/\Psi_{\mathrm{toroidal}}$) is less than the ratio of the injector current to the poloidal flux ($\lambda_{\mathrm{inj}} = \mu_0 I_{\mathrm{inj}}/\Psi_{\mathrm{inj}}$) where Ψ_{inj} is measured between the toroidal breaks between the inner and outer electrodes using the poloidal flux loop signals. Here the term 'ST' refers to the device and not to the type of plasma present in the machine. During steady-state current drive, the efficiency for the current drive is given as $\varepsilon = \lambda_{\mathrm{ST}}/\lambda_{\mathrm{inj}}$ [16, 18, 11]. Here the efficiency is defined as the injector flux utilization efficiency. The maximum current multiplication is predicted to be the ratio of the ST flux over the injector flux. From the above, $(I_{\mathrm{toroidal}}/I_{\mathrm{inj}}) = (\lambda_{\mathrm{ST}}/\lambda_{\mathrm{inj}})(\Psi_{\mathrm{toroidal}}/\Psi_{\mathrm{inj}})$, so current multiplication also depends on the λ ratio. A uniform λ cannot be assumed. However, λ_{ST} is not expected to exceed λ_{inj}. For the shot shown in figure 3, figure 4 shows the ratio of the toroidal flux, calculated assuming that the plasma fills the entire region between the electrodes, to the injector poloidal flux multiplied by the injector current and compared to the measured toroidal current. This analysis shows that within uncertainties the amplification factor is nearly as large as possible (at $t = 110$ ms); the measured current approaches 80% of the theoretical maximum current, showing that CHI is effective at driving toroidal current on large devices. The ratio is low during the early part of the discharge as only a small portion of the maximum possible toroidal flux (1.5 Wb) links the injector flux. While the CHI has successfully driven a toroidal current in NSTX, it is not yet known whether this current is flowing on closed field lines. Although a definite conclusion will probably have to await the implementation of diagnostics for the direct measurement of the spatial profile of the toroidal current, there is some indirect evidence for the reconnection process occurring during the CHI.

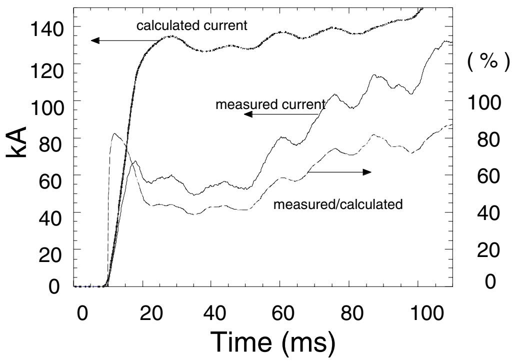

Figure 4. Measured and calculated ($I_{\mathrm{toroidal}} = (\Psi_{\mathrm{toroidal}}/\Psi_{\mathrm{inj}})I_{\mathrm{inj}}$) toroidal currents shows that the measured current approaches 80% of the maximum possible current.

Figure 3 shows an increase in the voltage fluctuations during phase 2 of the discharge. The fluctuations have a coherent mode at about 9 kHz. The analysis of the Mirnov coil signals shows a similar global mode during phase 2. A toroidal Mirnov coil array on the centre column shows that the magnetic signature of the discharge is toroidally symmetric with a coherent $n = 0$ mode, similar to that seen in the HIST experiment [12]. There is no toroidal Mirnov array on the NSTX outer vessel to discern the toroidal mode number near the outer shell. On the HIT experiments, $n = 1$ coherent mode activity, located on the outer shell probes only, is seen during high-current discharges [11, 14]; $n = 0$ activity is also seen by the outer Mirnov coils in some discharges. On NSTX, coherent fluctuations were also seen by a fast photodetector that sampled a chord passing through the centre of the plasma. The fact that this mode activity is seen by both magnetic and optical diagnostics is an encouraging result. On HIT experiments, a $n = 1$ mode seen on the outer vessel Mirnov coils is deemed necessary for the generation of closed flux surfaces. An outer vessel Mirnov array is being installed on NSTX.

To examine the effect of injector impedance at lower pressure, a scan was conducted in which the injector current was measured as the vessel pressure was reduced from 16 to 4 mTorr with the same flux and applied voltage. It was found that there is little change in injector current as the pressure is reduced. Experiments were also successful in generating discharges at a vessel pressure of 1 mTorr. Since pressures of about 4 mTorr are compatible with high recycling divertor operation, these results indicate that on NSTX, from a density standpoint, it should be possible to couple CHI discharges to Ohmic discharges. Compared to similar measurements on HIT-II, these results indicate a more favourable density scaling of the injector impedance that very likely results from the longer field line length on NSTX.

4. Summary and discussion

In summary, the initial CHI experiments on NSTX have successfully generated 130 kA of toroidal current using about 20 kA of injector current. Stable discharges lasting for 0.13 s have been produced using preprogrammed coil currents and at vessel neutral densities compatible with high recycling divertor operation. Experiments to date have shown that CHI engineering systems can be successfully applied to a large ST for the production of substantial toroidal currents as predicted by previous work on smaller experiments. However, there is insufficient evidence at this time to confirm the generation of closed flux surfaces in these discharges. Future experiments aimed at studying closed flux generation will involve producing CHI discharges at higher toroidal currents that will be diagnosed using multipoint Thomson scattering temperature and density profiles. The region of closed flux will be reconstructed using an equilibrium fitting code adapted to allow current flow on open field lines and the private flux region.

Acknowledgments

We would like to thank the NSTX Engineering and Operations group for their extensive effort in the installation and commissioning of the CHI hardware on NSTX and for machine operation. Pioneering work on smaller experiments that motivated this large-scale experiment on NSTX were first conducted by Thomas R Jarboe at the University of Washington and by Masa Ono at the Princeton Plasma Physics Laboratory. This work is supported by US DOE contract numbers DE-AC02-76CH03073, DE-AC05-00R22725, DE-AC03-99ER54463, DE-FG02-99ER54524, DE-FG03-99ER54519 and W-7405-ENG-36

References

[1] Sykes A *et al* 2000 *Phys. Rev. Lett.* **84** 495
[2] Gates D *et al* 1998 *Phys. Plasmas* **5** 1775
[3] Ono M *et al* 2000 *Nucl. Fusion* **40** 557
[4] Kaye S *et al* 1999 *Fusion Technol.* **36** 16
[5] Darke A C 1995 *Fusion Eng.* **2** 1456
[6] Ono M *et al* 1980 *Phys. Rev. Lett.* **44** 393
[7] Turner W C *et al* 1981 *J. Appl. Phys.* **52** 175
[8] Jarboe T R *et al* 1980 Production of field-reversed configurations with a magnetized coaxial plasma gun *Proc. Int. Symp. on Phys. of Open Ended Fusion Systems (Tsukuba 1980)*
[9] Armstrong W T *et al* 1980 *Plasma Physics Controlled Nuclear Fusion Research* vol 1 (Vienna: IAEA) p 481
[10] Jarboe T R 1989 *Fusion Tech.* **15** 7
[11] Nelson B A *et al* 1995 *Phys. Plasmas* **2** 2337
[12] Nagata M *et al* 1998 *17th IAEA Fusion Energy Conf. (Yokohama, 1998)* IAEA-CN 69/EXP4/10
[13] Browning P K *et al* 1992 *Phys. Rev. Lett.* **68** 1722
[14] Jarboe T R *et al* 1998 *Phys. Plasmas* **5** 1807
[15] Jarboe T R *et al* 1998 *17th IAEA Fusion Energy Conf. (Yokohama, 1998)* IAEA-CN 69/PDP/02
[16] Taylor J B 1986 *Rev. Mod. Phys.* **28** 243
[17] Maqueda R J and Wurden G A 1999 *Nucl. Fusion* **39** 629
[18] Barnes C W *et al* 1986 *Phys. Fluids* **29** 3415