PULSE LENGTH EXTENSION AND HIGH PERFORMANCE ON THE NATIONAL SPHERICAL TORUS EXPERIMENT (NSTX)

D. GATES[1], M. BELL[1], R. BELL[1], D. DARROW[1], S. JARDIN[1], S. KAYE[1], B. LEBLANC[1], R. MAINGI[2], J. MENARD[1], D. MUELLER[1], M. ONO[1], F. PAOLETTI[3], M. PENG[2], S. SABBAGH[3], E. SYNAKOWSKI[1]

[1] Princeton Plasma Physics Laboratory, Princeton, NJ 08543
[2] Oak Ridge National Laboratory, Oak Ridge, TN
[3] Columbia University, New York, NY 10027

Abstract

One of the primary issues facing of the Spherical Torus (ST) concept is the limited space for a solenoid in the narrow central core of the torus. This geometric feature limits the ohmically driven pulse length attainable for an ST. Experiments were carried out on the National Spherical Torus Experiment [Ono, M., et al., Nucl. Fus. **40** (2000) 557] (NSTX) which had the express purpose of extending the plasma pulse length for full current (1MA) discharges. Two externally controlled parameters were varied; the time of neutral beam injection and the value of the toroidal field. Neutral beam heating was applied during the current ramp. These experiments succeeded in extending the achievable pulse length from ~300ms during ohmic discharges at 3kGauss to >500ms with neutral beam injection and a toroidal field of 4.5kGauss. More importantly, the plasma current flattop at 1MA was extended to 250ms, whereas 3kGauss ohmic discharges only reach 1MA transiently. The primary effect of both raising the toroidal field and neutral beam injection into the current ramp was to elevate the central safety factor, q. This was beneficial because the onset of large sawteeth, caused by the appearance on a $q=1$ surface in the discharge, often will trigger a loss of plasma confinement on NSTX. This experiment also broadened the operating space of spherical tori for several other important plasma parameters which include: record stored energy 165kJ, record H factor = 1.4 (relative to ITER98pby2 [ITER Team, Nucl. Fus. **39** (1999) 2175]), record T_e =1.6keV, and record T_i = 2.1keV. Of particular importance is the elevated confinement, which bodes well for the future of the spherical torus concept.

1. Motivation

The spherical torus magnetic confinement concept is a promising alternative device for containing thermonuclear plasmas [1]. The spherical torus has received much attention following the experimental demonstration of high $\beta(= 2\mu_0\langle P\rangle/B^2)$ [2] with good confinement [3] on the pioneering START [4] device. These results have given rise to two new experiments NSTX [5] and MAST [6] (Mega-Amp Spherical Torus), one in the US and one in the UK (respectively). One of the primary challenge facing this concept is the limited volt-second capability of the slender center-stack. In fact most (conceptual) ST reactor designs eliminate the solenoid altogether and rely instead on complete non-inductive current drive. In almost all such designs, bootstrap current is assumed to provide nearly all the non-inductive current drive.

In addition to the bootstrap current, NSTX has three main tools capable of driving non-inductive current: high harmonic fast wave heating/current drive (HHFW), (which drives current by preferentially imparting parallel wave momentum to electrons), coaxial helicity injection (CHI), which involves the direct injection of current from electrodes inside the device, and neutral beam injection. Experiments are proceeding investigating the utility of all these tools. This paper will describe the efforts to date using neutral beam injection during the current ramp on NSTX to extend the achievable plasma pulse duration.

2. Experimental Method

As a first step towards achieving the goal of steady state operation, a series of experiments was carried out with the expressed purpose of extending the achievable pulse length on NSTX. These experiments employ neutral beam injection during the current ramp as a tool for pulse length extension. Early neutral beam injection can potentially change ohmic flux consumption via three mechanisms:

1) electron heating which lowers the plasma resistivity
2) neutral beam current drive
3) bootstrap current

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4) modification of MHD behavior

The toroidal field was also systematically varied during these experiments. The motivation for this aspect of the experiment is that the plasma pulse length in earlier neutral beam injection experiments [7, 8] was determined by the appearance of a *q=1* (where q = surface in the plasma. The onset of 1/1 MHD activity was correlated with reconnection events that led to a loss of confinement. By raising the toroidal field, the appearance of a *q=1* surface can be delayed. Early neutral beam injection can also delay the appearance of the *q=1* surface by raising the electron temperature, thus raising the electrical conductivity and thereby slowing the penetration of current to plasma core.

3. Results

3.1. Beam Injection

The target plasma chosen for the early neutral beam extension experiment was a center stack limited

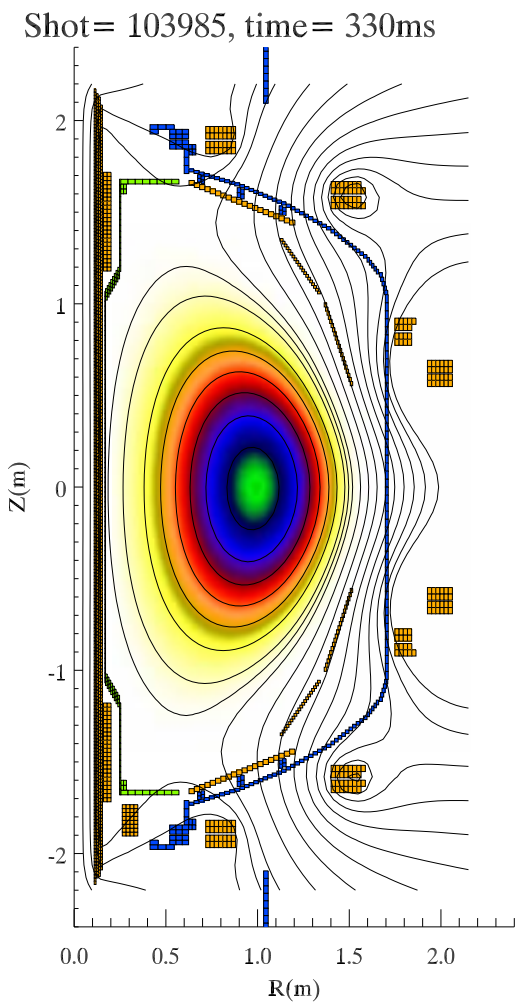

Figure 1 *Equilibrium cross-section for a typical discharge in the pulse-length extension experiment*

discharge with moderate elongation ($\kappa \sim 1.8$), and with a ~10cm gap between the outboard midplane and the nearest solid surface. The flux surfaces from a typical EFIT [9, 10] equilibrium reconstruction is shown in Figure 1. The toroidal field at the machine geometric axis was 3kG. The plasma was grown to full radius within 100ms, with the plasma current programmed to reach maximum current at 200ms. These parameters were chosen to simplify interpretation of the results. In particular, $\kappa \sim 1.8$ was chosen because it is the natural elongation, guaranteeing vertical stability regardless of the internal inductance of the plasma. The outer gap was chosen as a compromise between maximizing the plasma volume and a desire to avoid plasma-wall interactions. The growing plasma start up scenario was chosen based on calculations of beam orbit confinement which indicate that at low current increasing the plasma wall separation reduces the beam ion loss fraction. Lost ions, in this context, refers to those ions that impact solid surfaces, not just those that leave the last closed flux surface. The effect of losses due to charge exchange were not included in the calculations. This operational scenario is believed to allow for better fast ion confinement during early neutral beam injection, which broadens the operating range for the experiment.

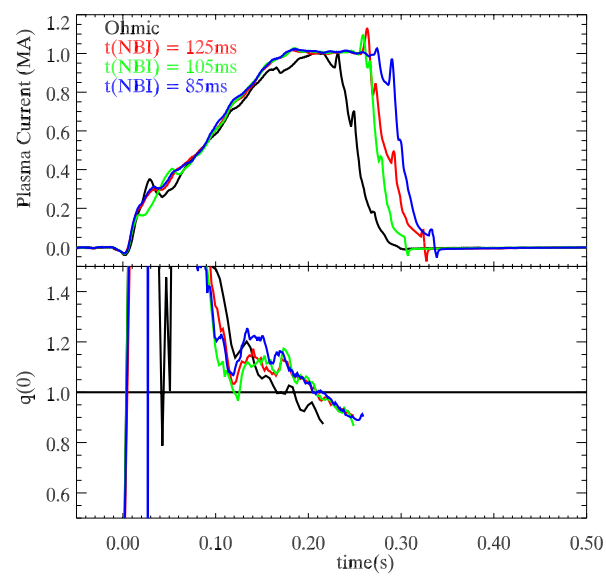

Figure 2 *Plasma current traces of a series of discharges for which the beam injection time was varied.*

A series of discharges were carried out wherein the neutral beam injection time was varied. Discharges with only ohmic heating typically suffered a reconnection event that terminated the I_p ramp at ~800kA. The cause of the reconnection event in the ohmic discharges is not

yet apparent. The neutral beam injection was therefore always initiated prior to the reconnection event. The effect of the neutral beam injection was dramatic in that even for the case with injection starting just before the time that the reconnection would have occurred during the ohmic discharge, the plasma current easily reached 1MA with a 50ms flattop. The results of the moving the beam earlier were less dramatic, however. Shown in Figure 2 are the plasma current traces for the above described series of shots. As is apparent in the plot, the maximum pulse length achieved did not depend strongly on the time of the beam injection. However, it is true that the longest shot in this series was the one with the earliest beam injection.

MHD instabilities determined the pulse length of each of the discharges in the scan. The reason that the neutral beam heating extended the pulse length is probably due to the modifications in the evolution of the q-profile associated with the slower current diffusion time. This effect can be seen in the second frame in Figure 2, which shows the evolution of the central q as determined by EFIT (without the benefit of internal magnetic field pitch angle measurements). Clearly the appearance of the $q=1$ surface in the ohmic discharge precedes that of the shots with neutral beam injection.

3.2. Toroidal field scan

The discharge that had the longest pulse length in the beam timing scan was used as a base case for the toroidal field scan. Four different values of toroidal field

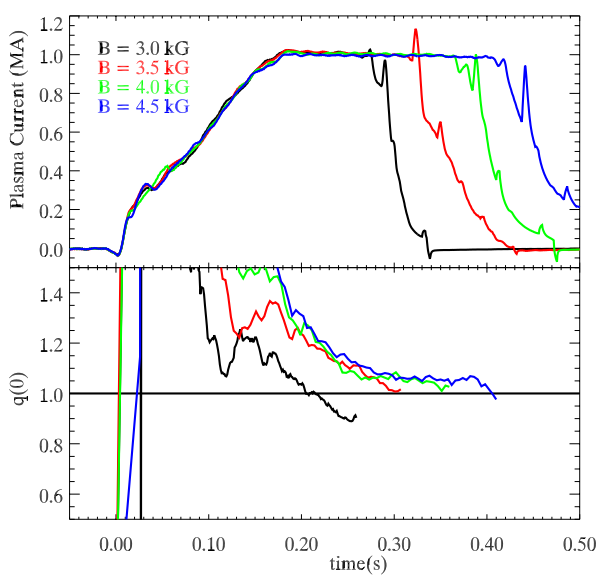

Figure 3 *Plasma current traces of a series of discharges for which the toroidal field (at R=86cm) was varied between 3.0kGauss and 4.5kGauss in steps of 500G. The increase in toroidal field extended the achievable pulse length dramatically.*

were investigated ranging from 3kG to 4.5kG in 500G steps. The plasma current traces for the longest lived shots at each of the different toroidal field values is shown in Figure 3. As can be seen from the figure, the effect of varying the toroidal field on the plasma pulse length is substantial. The maximum flattop length achieved during 3kG operation was 100ms. At 4.5kg, the maximum current flattop time increased to 250ms. Each 500G increment in toroidal field added roughly 50ms of current flattop.

The q at the magnetic axis is also plotted versus time for each of the discharges in the toroidal field scan Figure 3. The calculated central q should be taken only as an indication of the trend in central q, rather than an actual measure of this quantity, given the lack of q-profile data. The trend is clear that as the toroidal field is raised the time of appearance of a $q=1$ surface is delayed. The delay in the appearance of a $q=1$ surface is also confirmed by observations of the onset of $m=1$, $n=1$ mode activity made using an array of soft x-ray detectors. The correlation of the onset of sawteeth with the end of the high performance phase of the discharge will be discussed in a future paper.

Figure 4 shows the line average temperature for the shots in the toroidal field scan. While there are variations in the temperature with toroidal field, there is no systematic trend. From this we conclude that the primary cause of the pulse length extension is not due to the reduction in resistive flux consumption

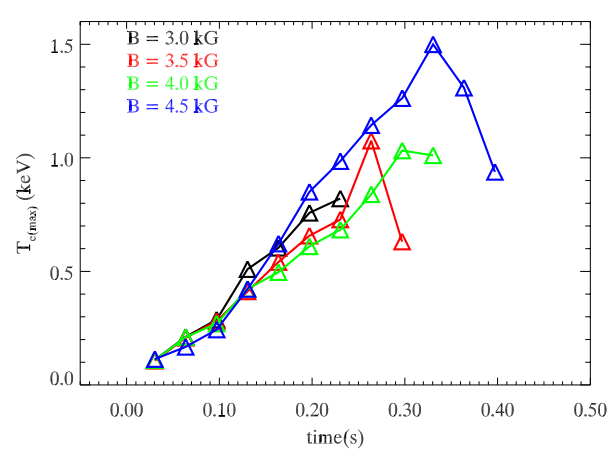

Figure 4 *Line average temperature as measured by multi-point Thomson scattering for each of the discharges in* Figure 3.

associated with an increase in electron temperature. Instead, the primary effect of raising the toroidal field is to raise the safety factor, q, to higher values and thereby delay the onset of MHD activity which begins with

appearance of a *q=1* surface in the interior of the plasma. Shown in Figure 5 is a detail view of the onset of the m=1 mode that is correlated with the end of the discharge. The radius of the q=1 surface in these discharges is often a large fraction of the minor radius of the plasma .

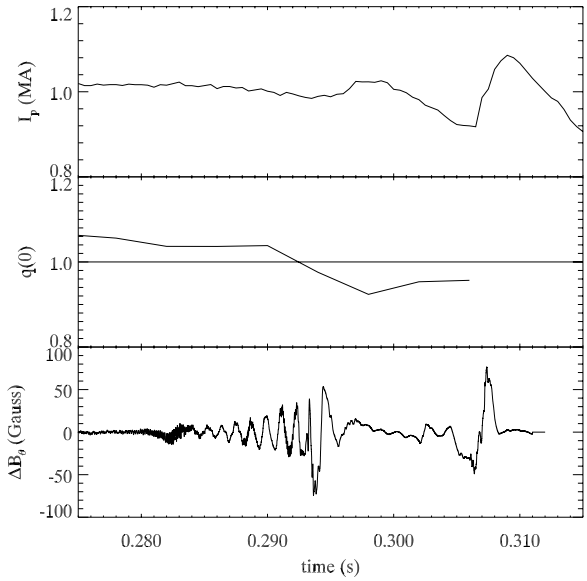

Figure 5 *a) plasma current, b) the q(0) as determined by the EFIT equilibrium code, c) the perturbed poloidal magnetic field as measured by a mirnov coil mounted on the vacuum vessel wall. All from a 4.0kGauss shot similar to the one shown in Figure 3.*

An interesting feature observed in several discharges with early beam heating is the sudden increase in the electron temperature that is observed at t = 250ms for the 4.0kGauss shot. The cause of this spontaneous increase in temperature is not known at this time, but this is under active investigation.

3.3. Plasma parameters

As a result of the higher than expected confinement times that were observed on NSTX during this

Parameter	*Value*
$T_e(0)$	1.6keV
$T_i(0)$	2.0keV
n_e	$5.9\times10^{19}m^{-3}$
Z_{eff}	1.5
$n_e/n_{Greenwald}$	0.6
W	165kJ
β_N	4.5
$H98(pby2)$	1.4
β_t	22.7%

Table 1 *Maximum values for plasma parameters achieved during the pulse extension campaign*

experimental campaign, the pulse length extension experiments have achieved many other parameters of note. Table 1 shows a list of the maximum value of each plasma parameter achieved during this series of discharges. Also shown are derived quantities of interest that have been achieved such as confinement improvement that has been achieved relative to various scaling laws, and beta relative to various theoretically predicted beta-limits.

These parameters indicate that the performance of the spherical torus concept at the 1MA level compares well to standard aspect ratio tokamak performance. It should also be pointed out that these results were achieved in the first NSTX physics campaign to have substantial neutral beam heating power. Of particular interest is the energy confinement improvement over scaling law

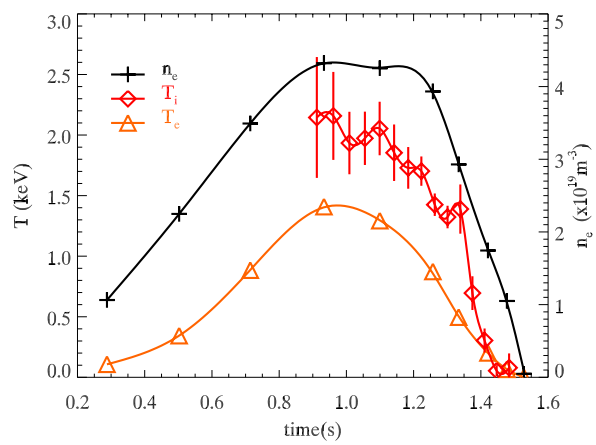

Figure 6 *Electron and ion temperature profiles for shot 104001.*

predictions, keeping in mind that all of the data presented in this paper is from plasmas which were inboard limited, and that were not in H-mode.

The line averaged densities of the discharges presented were in the range from 0.4 – 0.6 x $n_{Greeenwald}$, where $n_{Greenwald} = I_p/\pi a^2$ [11]. The density was not explicitly varied in these discharges, however wall and beam fueling caused the density to rise throughout each discharge. The measured electron temperature profiles for shot 104001 at time are shown in Figure 6. This represents the highest electron temperature achieved to date on NSTX. The highest central ion temperature, $T_i(0) \sim 2keV$, measured to date on NSTX was measured during the long pulse campaign.

The highest toroidal β, $\beta_t = 22.7\%$, achieved to date on NSTX was also achieved during this experimental campaign. It was achieved with a toroidal field of 3.0kG and a heating power of 4.5MW. It should be noted that

the equilibrium reconstructions that are being used to estimate β_t do not include the measured kinetic profiles as constraints, due to the recent availability of these measurements on NSTX, which is a new facility. Equilibria which are self consistent with measured profiles will be developed as soon as the kinetic profiles are independently validated. The highest stored energy achieved to date on NSTX of 165kJ was obtained during a long pulse discharge with a toroidal field of 4.5kG. This result was not simultaneous with the highest β_t, which was achieved at lower toroidal field.

4. Analysis

4.1. Resistive flux consumption

In order to determine more precisely the results of the early beam injection experiments, an analysis of the flux consumption during the discharge was performed

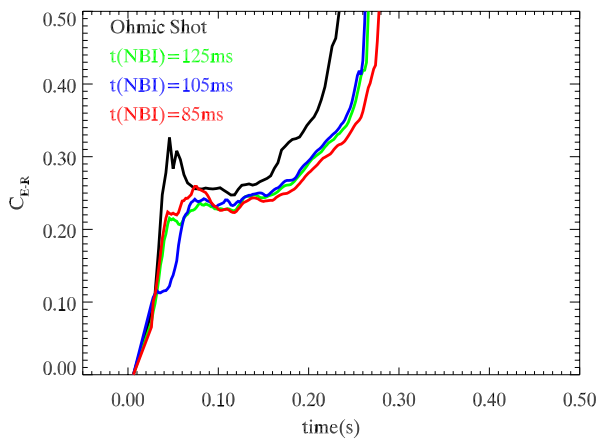

Figure 7 *Resistive Ejima-Wesley coefficient as a function of time for the shots shown in Figure 2*

according to the method described in [12], which is based on the methods of Ejima and Wesley [13]. Shown in Figure 7 is the resistive Ejima-Wesley coefficient, which is a measure of the resistive flux consumption, for each of the discharges shown in Figure 2.

Flux consumption analysis for the toroidal field scan is shown Figure 8. The results from this series discharges are quite clear in that the flux consumption is not greatly reduced as the TF is raised. It should be noted that the injected neutral beam heating power is lower in the lower toroidal field shots than it was in the shots with higher toroidal field. This was necessary to avoid the onset of MHD modes that appeared to limit the achievable stored energy before the shot was over. In fact in the higher toroidal field discharges, it is likely that a lower beam power than was available from a single source would have been sufficient to reach the limit on stored energy by the end of the discharge.

TRANSP analysis of these discharges indicates that the bootstrap current represents between 15 and 20% of the total driven current. Also, the neutral beam driven component of the of the non-inductive current is calculated to be ~7%. Whereas these numbers are not ignorable, they support the conclusion that the primary effect of the neutral beam injection is to delay the onset of a *q=1* surface.

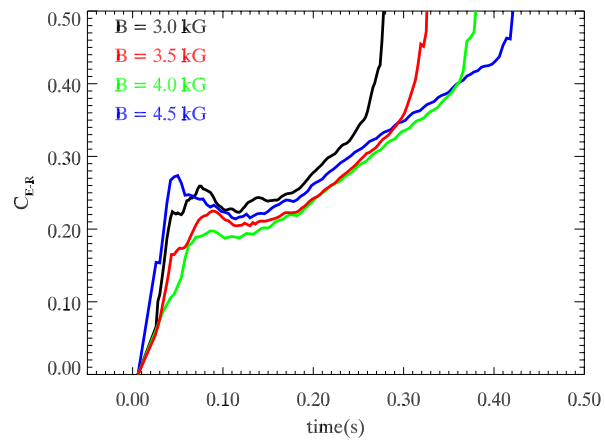

Figure 8 *Resistive Ejima-Wesley coefficient as a function of time for the shots shown in* Figure 3

4.2. TSC

The Tokamak Simulation Code (TSC) was used during the design phase of NSTX [14] to determine the number of volt-seconds required to reach the specified 1MA plasma current requirement. Since TSC was developed for use on conventional aspect ratio tokamaks there was some uncertainty in using the code to make this determination, primarily due to potential differences in plasma energy and particle transport that might arise as a result of the much lower aspect ratio on NSTX. The coefficients of the standard TSC model [15], which is based on the Coppi-Tang transport model, [16,17] during the design phase were adjusted to match the level of transport predicted by the Lackner-Gottardi scaling law [18] in accordance with the transport coefficients observed on the START experiment [3].

TSC analysis was performed on the shots presented above in order to determine if the assumptions made during the design phase of NSTX were accurate and, if not, to quantify an gross differences between the design assumptions and the measured plasma performance. The procedure followed to match the TSC simulations to the NSTX data was: 1) Assume a standard TSC transport model which is based on , 2) Use the measured PF/OH/TF coil currents as inputs, 3) Preprogram the density as a function of space and time to match the Thomson scattering density profile, 4) Assume 2%

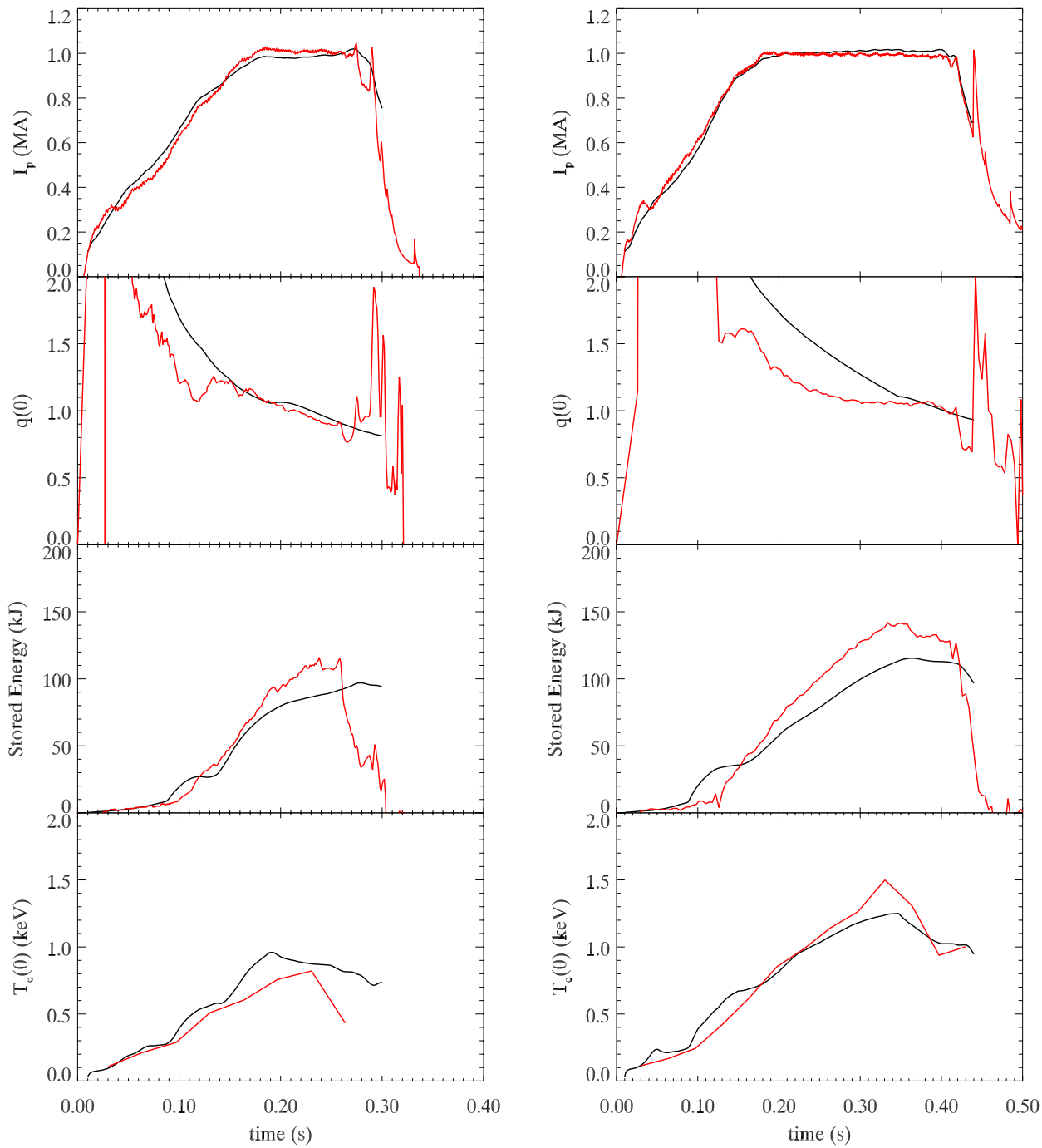

Figure 9 *Comparison between TSC simulations (black) and 2 shots form the toroidal field scan, a) 103701 (3.0kGauss) shot and b) 103985 (4.5kGauss). The stored energy and q(0) in red are from EFIT reconstructions.*

Carbon impurity concentration, 5) Vary the assumed transport coefficient and impurity concentration to improve the match between the observed and calculated T_e and I_p, 6) repeat, until match is reasonable. Shown in Figure 9 are the results of the final TSC run compared to the measured/reconstructed values (from EFIT reconstructions) from the experiment. As is apparent, the match is quite reasonable.

A summary comparison between the values of the transport coefficients used to fit the data and those used in predicting NSTX performance indicates that ion thermal diffusivity is in good agreement between the

two simulations, but that the electron confinement used to predict NSTX performance was optimistic by a factor of 2 over that used to fit NSTX data. Interestingly this did not substantially affect the predicted pulse lengths, since the bulk of the flux in the transformer goes towards ramping up the stored flux in the plasma, rather than to resistive consumption (as shown in reference [12]).

5. Summary

A successful series of experiments were carried out to extend the pulse length of NSTX discharges to values sufficiently long to carry out MHD studies. In particular, plasma pulses with 1MA current were sustained for 0.5 seconds, with flat-top times of 0.25 seconds. The pulses were extended by two methods: early neutral beam injection and raising the toroidal field, delaying the onset of a q=1 surface. During the course of these experiments, the available operating range of many other plasma parameters was extended.

These results are extremely encouraging for the spherical torus concept in that, if the very high confinement time relative to standard aspect ratio tokamak scaling laws were to continue as size and field are scaled up, one can imagine using this device as the basis for a low cost burning plasma experiment. In order to imagine the ST as a reactor further pulse length extension experiments are required. The bootstrap current fraction in these discharges is estimated to be ~15-20%. This value is far below that required for a steady state ST. Additional experiments to extend the pulse further using bootstrap current will be carried out. Nonetheless, these experiment are encouraging since they have pushed the ST concept into a previously unattained regime of high β and low collisionality, while maintaining good confinement.

6. Acknowledgements

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