Progress towards stable Steady State Operation on NSTX

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Plasma shaping has been demonstrated to be an important determining factor for MHD stability (including ELM stability). Recent improvements to the shaping capability on the NSTX device have been followed by a substantial improvement in long pulse performance. In particular, the poloidal field coils which are used for creating high triangularity double null plasmas was reduced in height so as to be made compatible with simultaneous high κ ~2.5, and high δ ~ 0.8 operation. As predicted by equilibrium and stability calculations, this modification has led to the widening of the NSTX operating space. Integrated scenario simulations [1] made using the TSC code [2] indicate that a 100% non-inductive operational scenario is attainable on NSTX as a result of the plasma shaping enhancements. In addition, advanced shape control using rtEFIT/isoflux control has improved the reliability of access to these long pulse regimes.

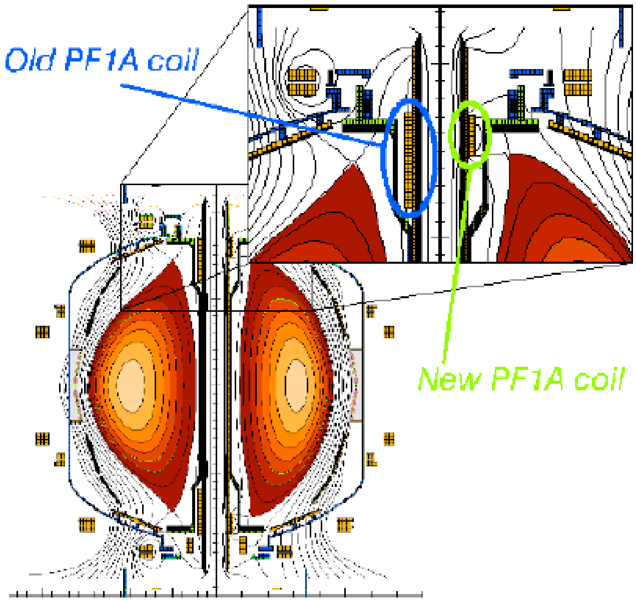

Figure 1: Plasma equilibrium with shaping factor $S= q_{95}(I_p/aB_t))$ ~ 37 [MA/m•Tesla] (right hand-side) compared with a highly shaped plasma from 2004 (left). Shown in the blow-up is the PF1A coil X-point coil before and after modification

Operation of NSTX with the stated poloidal field coil modifications has lead to the achievement of a record shape factor ($S= q_{95}(I_p/aB_t)$) of ~37 [MA/m•Tesla]. An example of a highly shaped equilibrium is shown in Figure 1, which also shows the modified poloidal field coil, juxtaposed with a typical double null high κ plasma from 2004. Achieving high shape factor is an important result for future ST burning plasma experiments as exemplified by studies for future ST reactor concepts, such as ARIES-ST [3], as well as neutron producing devices such as the Component Test Facility (CTF) [4], which rely on achieving even higher shape factors in order to achieve steady-state operation while maintaining MHD stability.

Plasmas with high shape factor have been sustained for pulse lengths which correspond to τ_{pulse} = 1.6s ~ 50τ_E ~ 5τ_{CR}, where τ_{CR} is the current relaxation time and τ_E is the energy confinement time. These plasmas had a pulse average value of β_t ~ 15% at a plasma current of I_p = 0.7MA and a vacuum toroidal field Bt = 4.5kGauss. At higher plasma current of I_p =1MA, pulses have been maintained with pulse average β_t ~ 20% for pulse lengths of τ_{pulse} = 1s ~ 25τ_E ~ 2τ_{CR}. Shown in Figure 2 is a plot of the value of the figure of merit $\beta_N H89$ plotted vs. τ_{flat}/τ_E, where τ_{flat} is the current flattop time, for data from 2005. The longest discharges are terminated by engineering constraints on coil currents, not by MHD modes. However, n=1 MHD modes do reduce performance late in the discharge.

Such long pulses are not possible on NSTX without substantial bootstrap and neutral beam driven currents since, at low aspect ratio, the available solenoid flux is limited. The non-inductive current fraction in the longest pulse discharges has reached ~65%, with ~55% pressure driven current and ~10% neutral beam driven current. The constituent components of the plasma current have been analyzed using the TRANSP code and data from the recently commissioned NSTX MSE system. NSTX is the only ST in the world that has MSE. An interesting feature of these discharges is the observation that the central value of the safety factor *q(0)* remains elevated for several current diffusion times. This observation may be related to the "hybrid" mode observed on several tokamaks, and that is being proposed as an alternate operational scenario for ITER.

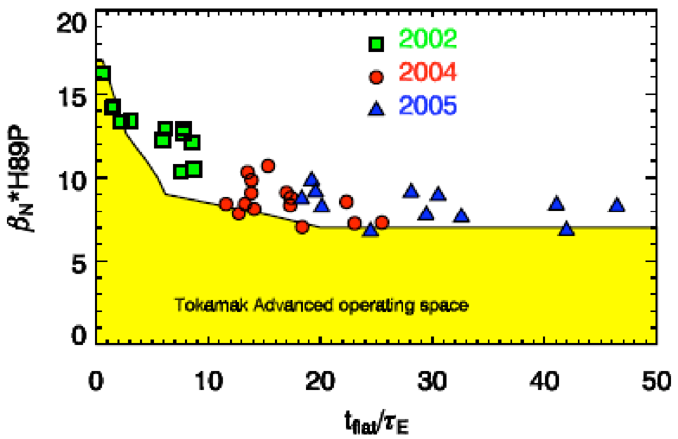

Figure 2: *A plot of the figure of merit β_NH89 plotted versus pulse length normalized to the energy confinement time. Color indicates year the data was taken*

An important factor in increasing pulse length in the past on NSTX and other devices has been the utilization of the "early H-mode" scenario [5]. This scenario has been further optimized during the 2005, reducing initial flux consumption. The early H-mode scenarios had previously been limited by large ELMs, which reduced confinement and consumed flux. The long pulse regimes obtained more recently have exhibited a substantial reduction in the frequency and size of ELMs. The reduction in ELM magnitude and frequency has improved energy confinement time.

NSTX operates with peak divertor heat fluxes which are in the same range as those expected for the ITER device, i.e. with P_{heat_max} ~ 10MW/m^2. High triangularity, high elongation plasmas on NSTX have been demonstrated to have reduced peak heat flux to the divertor plates to <3 MW/m^2. This is consistent with the enhanced flux expansion that is associated with the strong shaping.

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