3.6 Integration and Control

3.6.1 ST Research Goals and Target Plasma Conditions

The IPPA and FESAC reports have established ambitious goals for NSTX, many of which involve the integration of several plasma conditions, each one of which represents a challenge itself. For example, the IPPA goal 3.2.1.6 is to "integrate high confinement and high beta" while the FESAC 5-year Objective #2.1 involves "...assessing high-beta stability, confinement, self-consistent high-bootstrap operation, and acceptable divertor heat flux, for pulse lengths much greater than energy confinement times".

Considerable progress has already been made in achieving high β and good energy confinement in NSTX. During the 2002 experiments, NSTX reached a normalized beta, which characterizes closeness to the MHD β-limit, $\beta_N \approx 6\%\cdot m\cdot T/MA$ in a discharge with a confinement time $\tau_E \approx 50ms$, corresponding to a confinement enhancement factor $H_{89P} \sim 2.5$ relative to the ITER-89P scaling expression, for a duration of ~400ms, *i.e.* ~ $8\tau_E$. These parameters were obtained in a discharge with $B_T = 0.5T$, $I_p = 0.8MA$ heated by 5MW of neutral beam injection. Figure 3.6.1 [1] compares the NSTX achievements in the figure of merit $\beta_N H_{89P}$ plotted against the pulse length normalized to the energy confinement time with the envelope of the advanced tokamak database.

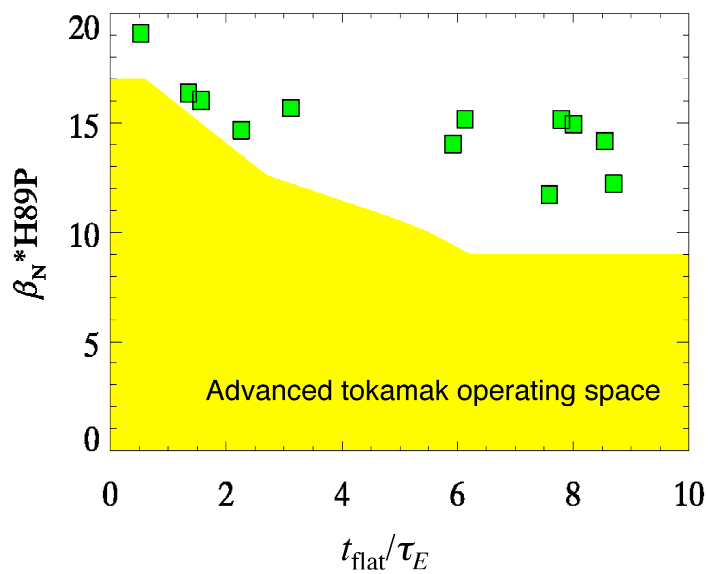

*Fig. 3.6.1 Values of the product $\beta_N H_{89P}$ for NSTX (green squares) plotted against the time for which the values are maintained normalized to the confinement time. The yellow shaded area represents the advanced tokamak database. For reference, β_N*H89P ~ 14.2 for ARIES-AT and β_N*H89P ~ 20.9 for ARIES-ST.*

Milestones for the years 2004 – 8 will extend the pulse length requirements to approach and exceed the current penetration time, which ranges from 250ms (measured) up to 600ms (predicted for discharges with strong RF heating) depending on the electron temperature achieved during auxiliary heating. Simultaneous optimizations of performance in additional areas then become necessary. Longer pulses will necessitate operating with a toroidal field $B_T \leq 0.5T$ to stay within allowable limits (See section 2.2 for a discussion of machine limits to pulse length). Furthermore, the inductive solenoid was operated close to its full rating. Longer pulses therefore add the requirement for non-inductive current drive, which must be efficient, since the power applied for current drive will count against the confinement time. With longer pulses and high input power, the total energy input and the consequent heating of the divertor surfaces become issues. Control of the divertor strike-points and of means for dispersing the heat flux through the scrape-off layer will be needed in such long-pulse discharges. The plasma control developments required for addressing these issues will be discussed in the section 3.6.3 – 3.6.6. The development of advanced plasma control capability in the forthcoming phase of NSTX research will provide tools for building on the foundation already established in the first three years of its operation.

3.6.2 Control System Status

The plasma in NSTX is presently controlled by a high-speed multi-processor computer system (SkyBolt II, currently with eight 333MHz G4 processors) [2, 3] which processes diagnostic measurements in real time to generate commands to the power supplies, the gas injection system and, in future, other actuators (see section 3.6.4) which can modify plasma behavior. The real-time algorithms are set up by a host computer (a UNIX workstation running Solaris that also acts as the Slot 0 VME controller) running the PCS (Plasma Control Software) [4] developed by General Atomics and adapted for NSTX under an ongoing collaboration. This software provides a convenient graphical interface for the operators to set up control parameters and reference waveforms, and also performs housekeeping functions such as archiving and retrieving control data using the MDS-Plus system. The PCS software allows different control strategies and algorithms to be applied during the various phases of a discharge, making it suitable for future experiments with non-inductive current drive and advanced plasma control.

For control of the plasma equilibrium, the control computer receives digitized diagnostic data during the pulse directly from local digitizers (currently 64 channels) and from a high-speed fiber-optical link to remote digitizers in the NSTX Test Cell (currently 96 channels) using a data transfer method known as FPDP (Front Panel Data Port, which exists in both parallel [copper] and serial [fiber optic] formats). The serial FPDP data, which streams from diagnostics at different potentials (on NSTX the center stack diagnostics are isolated form ground and can float to >1kV to permit CHI operation), are combined by a multiplexing module called a FIMM [5] (FPDP Input Multiplexing Module), developed recently at PPPL and shown in Fig. 3.6.2. The FIMM will allow more commercial digitizer modules (32 channels each) to be added quite readily to accommodate future development. Each input signal is input to the control computer at 12-bit precision with a sampling rate of 5 kHz and a sample latency under 10 μs. From the control computer, command signals are sent to the NSTX power supplies on the Power Conversion (PC) Link, a system developed at the beginning of TFTR. The power supplies are phase controlled rectifiers operating in either 6- or 12- pulse configuration with a line frequency of about 65Hz provided by the motor-generator sets.

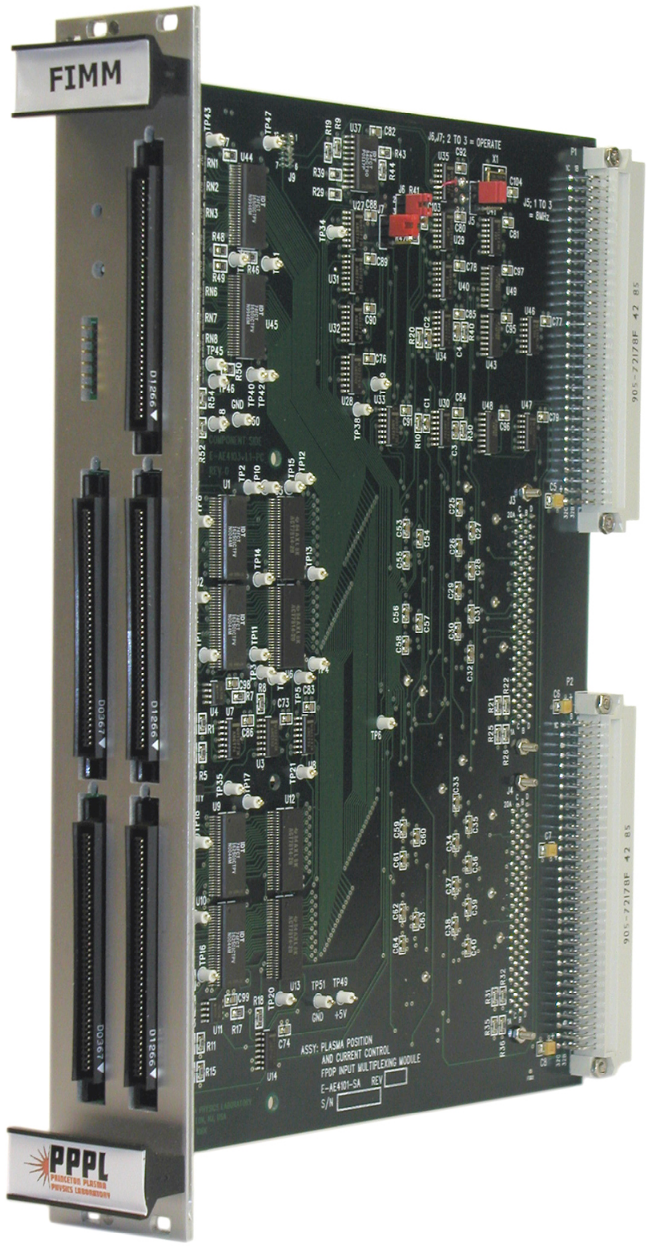

Figure 3.6.2 The FIMM, developed at PPPL, combines FPDP high-speed data streams from different source modules for input to the SkyBolt computer.

Lower data rates are required for controlling the gas injection system (GIS) since the response times of both the gas valves and the plasma itself are slower. Seven channels of diagnostic data are received and four channels output to valve drivers by the real-time computer by extending its VME bus to a second crate in the NTC. To provide reproducible flow rates, pulse-width modulation of the piezo-valves is employed with the duty-cycle automatically adjusted for the gas species and its upstream pressure. At present, only the prefill pressure prior to the discharge and the gas flow rates from the different valves during the discharge are programmable.

The implementation of gas control via the VME bus has increased the I/O overhead of the real-time computer system, thereby increasing the latency in communicating with the power supplies. This is a concern, particularly for the control of highly shaped plasmas. The effective frequency response of the PC-Link and power supplies is adequate for controlling the plasma current and the radial and vertical position, provided that the elongation is not too high and the internal inductance of the plasma is low: in experiments to date, κ up to ~2.5 has been controlled for $l_i \approx 0.6$. However, future experiments will demand greater capability for elongation control, so methods for improving the speed of the power supply control are being investigated in collaboration with General Atomics.

The 2002 experiments demonstrated the benefits of gas fueling from the high field side, *i.e.* injecting gas on the center stack, for obtaining H-mode transitions reliably. The real-time computer does not yet control the flow from the center-stack gas feed. The prototype injector used until now is opened fully before the start of the plasma and it empties the contents of its plenum during the discharge. For this injector, there is a time delay of about 0.2s between opening the injection valve and the appearance of gas in the vessel because the gas must flow through a long (~2m), narrow (3mm diameter) pipe from the valve to the feed point at the mid-plane. The flow characteristics of this injector are essentially determined by the initial pressure of the gas in the plenum. During the 2002 outage, an additional high-field-side injector was installed using a shorter, larger-diameter pipe to convey the gas to the upper shoulder of the center stack, rather than the mid-plane. It is expected that this injector will provide greater controllability for the gas which may permit it to be incorporated in the real-time control system in the same way as the low-field-side injectors.

Until recently, the algorithms for plasma equilibrium control on NSTX were fairly rudimentary [6], providing feedback control only of the plasma current, the vertical position of the current centroid, and the outboard radial gap, nominally at the midplane. Shape control was provided by programming the poloidal field (PF) coil currents in advance. The outboard gap was determined by a flux projection technique using the poloidal magnetic flux and field measured behind the passive stabilizing plates inside the vacuum vessel.

In the last week of the FY'02 run (June 2002), limited control of the equilibrium during the current flattop was demonstrated using an isoflux control algorithm based on a real-time solution of the Grad-Shafranov equation for toroidal equilibrium with the rtEFIT [7] code developed by General Atomics. This solution is a best fit to the set of about 130 measurements of coil currents and magnetic fluxes and fields currently digitized in real-time. Figure 3.6.3 compares an example of the plasma boundary computed by rtEFIT during one of the demonstration shots with that calculated by the standard EFIT code in off-line analysis.

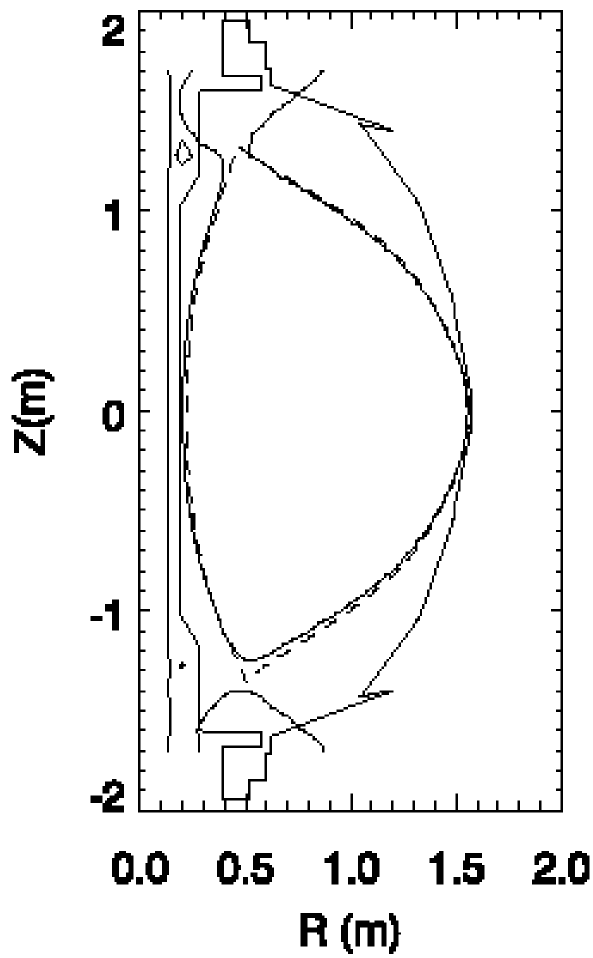

Figure 3.6.3 Plasma boundary calculated by rt-EFIT (solid) and between-shots EFIT (dashed) for shot 110068 at 290 ms.

During the FY'03 run, the capabilities for equilibrium control based on rtEFIT were expanded by implementing more control variables (additional gaps). The entire complement of NSTX poloidal field coils were controlled using the rtEFIT/isoflux control algorithm. The early startup phase of each discharge was still controlled by a flux-projection technique, but that the time window available for rt-EFIT control was expanded from the start to the end of the plasma current flat-top. Work during the FY'04 run will focus on expanding the utility of the rtEFIT/isoflux control algorithm until it is the default for plasma control on NSTX.

3.6.3 Control System Upgrades and Real-time Diagnostic Data Processing

The inclusion of profile data in the rtEFIT reconstruction will greatly facilitate model based plasma profile control. Plans and possibilities for expanding plasma control by including real-time diagnostic data, with emphasis on data relevant for profile control, in the real-time physics analysis are discussed below. These will require additional input channels from sources in different locations and in different potential classes. The present SkyBolt II computer already has considerable capability for real-time processing of diagnostic data and the number/speed of the processors can be easily expanded further if needed.

2004:*Control of the plasma-antenna gap to maintain the RF loading* – During high-harmonic fast-wave (HHFW) heating, varying plasma conditions in the boundary affect the loading presented by the plasma, which can limit the RF power and cause trips on the generators due to high reflected power. Previously, it was demonstrated on TFTR that the antenna loading for ICRF power in the more traditional regimes of fundamental minority and second-harmonic majority heating, could be maintained by real-time feedback control of the gap between the plasma boundary (the last closed flux surface) and the antenna grill. It is planned to implement a similar feedback system in NSTX. Measurements of the RF voltages and currents in the antenna elements will be digitized and transmitted in real time to the control computer where the effective loading resistance will be computed and compared with a reference value. The difference signal will be used to modulate the requested value for the outer gap between the plasma boundary and the antenna during the time of the RF heating: if the loading is too small, the gap will be reduced, and *vice versa*. This capability will be introduced when NSTX returns to operation in FY'04.

2004 – 2006: *Meeting the control requirements for resistive wall modes* - An important element of the NSTX program over the next several years is identifying the control needs and implementing an active feedback system for the control of resistive wall modes (RWMs), instabilities which would be stabilized by a perfectly conducting wall close to the plasma but which can grow on a resistive timescale if the wall has finite conductivity and the rate of plasma rotation is insufficient. NSTX has already operated with normalized-β above the "no-wall" limit and the growth of the RWM in this condition has been inferred from the development of kink-link perturbations of the plasma column and the rapid slowing of the plasma rotation driven by the neutral beams.

It has been demonstrated, on DIII-D in particular, that real-time control of both extrinsic error fields arising, for example, from coil misalignments, and the fields generated by the growing instabilities themselves, can facilitate continued plasma rotation and suppress instability growth. On NSTX, a set of detectors for non-axisymmetric poloidal field perturbations was installed during FY'02. Although the requirements for real-time processing of data from these detectors to permit stabilizing RWMs is not yet known, it is likely that the required response time will be considerably shorter than that needed for axisymmetric position control. However, this should still be within the capabilities of the present data acquisition system and control computer. Most of the diagnostic signals for RWM control will be generated in the diagnostic racks close to the existing digitizers in the NTC for the real-time system. Additional digitizers will be needed, and, possibly, an additional VME crate, controller and a pair of fiber optic link modules. The actuators for RWM stabilization are discussed in Sec. 3.6.4 below.

2003 – 2005: Control system development for coaxial helicity injection (CHI) - Coaxial helicity injection (CHI) presents several challenges for real-time control. At present, the CHI discharge is established by preprogramming the coil currents and the injector voltage without feedback control over the discharge evolution. In the experiments from FY'02 and earlier, most CHI discharges ended in an "absorber arc", that is a localized breakdown across the insulating gap at the opposite end of the vacuum vessel from the injecting gap. These absorber arcs are believed to have been caused by an inadequate length of insulator in the absorber and the development, as the CHI discharge evolved, of an unfavorable magnetic field distribution which allowed a relatively short connection length along field lines between components at different potentials. In the 2002 opening, a new absorber insulator was installed which increases the separation of the components at different potentials. The new insulator reduced the frequency of absorber arcs in FY'03. In addition, two PF coils were added to provide additional capability to null the poloidal field in the absorber region. Power supplies are now being built to energize these control coils.

The first task in implementing control for CHI is developing the appropriate control strategy for the primary CHI discharge to maximize the toroidal plasma current and to provide conditions conducive to the formation of closed magnetic flux surfaces while avoiding absorber arcs. This is needed to provide reproducible conditions for diagnosing the discharge to gain an understanding of the basic process and subsequently for coupling CHI to other methods of current sustainment. The initial implementation will be based on flux projection from external loops and sensors with a progression later to rt-EFIT control.

Because the CHI plasma is not axisymmetric, data from several magnetic sensors distributed toroidally will be needed. Analyzing the plasma configuration with rt-EFIT during CHI will require a substantial modification of the code to include current on the open field lines between the injector electrodes. The control requirements for CHI initiation and feedback control will be attempted in FY'04. Also in FY'04, an assessment will be made of the need for absorber field nulling, with implementation of feedback control for this in FY'05, if it is required.

Once reliable CHI initiation has been established, a method for making the transition to inductive and/or RF-driven current sustainment will be implemented. The first tests of inductive sustainment will be made in FY'04 using a simple (non-feedback) control algorithm already developed. Feedback control of the transition and subsequent discharge evolution will also be introduced in FY'04, using a plasma current control plus flux projection technique similar to that used in ohmic discharges. The capability for handing over to sustainment by HHFW current drive will be introduced in FY'05.

<u>2005</u>: *Inclusion of real-time pressure measurements in the control system analysis* - The raw data from the photodetectors of the multi-point Thomson scattering (MPTS) measurement will be processed in real time to provide the temperature, density and, hence, the pressure profiles of the electrons, albeit with somewhat less accuracy than the full offline analysis. The electron pressure profile shape alone has been used as an additional constraint in offline analysis with the EFIT code and shown to improve the quality and accuracy of the equilibrium. A similar technique can be employed to improve the faithfulness of the rt-EFIT analysis and will be useful for predicting HHFW power deposition. With the present two lasers, the MPTS system can provide full profile data at 60Hz, which is comparable to the time-slice analysis rate for rt-EFIT on a single processor. To provide this data in real time from the MPTS instrumentation room, up to 4 32-channel digitizers (for the full 20 spatial channels), a VME crate and controller and 2 fiber optic link modules, in addition to the FIMM will be required. Alternatively a data multiplexing scheme could be implemented, requiring fewer input data channels while increasing data latency. This upgrade will be undertaken in FY'05.

The full analysis to obtain the ion temperature for the data from charge-exchange recombination spectroscopy (CHERS) during neutral beam injection is too complex to be applied in real time. These data are recorded with a 2-D CCD array with a relatively slow readout, and the conditions in NSTX necessitate a complicated removal of background emission from the edge region of the plasma and an elaborate fitting procedure. However, as experience is gained with the new CHERS system, which was installed during the summer 2002 outage, it may be possible to devise a faster readout system for a subset of the data and an approximate method for analysis of CHERS data adequate for real-time control.

If real-time analysis of the ion and electron profiles becomes available, then an approximate calculation of the contribution of fast ions to the total pressure could also be performed. This would then enable rt-EFIT analysis with full kinetic profiles for plasmas heated by NBI. Furthermore, if the capabilities of the SkyBolt computer were expanded by adding or upgrading processors, it would be possible also to calculate the deposition of the High-Harmonic Fast Wave (HHFW) power during RF heating and current drive and subsequently use this information for profile control. At present, such calculations rely on separate ray-tracing of the launched waves and kinetic modeling of the wave absorption. Several advances in the speed and accuracy in the calculation of wave-plasma interactions have occurred in the last year under the auspices of the SciDAC initiative. From numerical analysis and modeling of HHFW-heated plasmas, it should be possible to develop approximate methods which are suitable for real-time control.

<u>2006 – 2007</u>: *Inclusion of real-time current profile and magnetic field data* - Data from both the Motional Stark Effect (MSE) polarimeter and Faraday rotation measurements from the far-infrared interferometer/ polarimeter (FIReTIP) will help to constrain the toroidal current profile in the rt-EFIT analysis. The first two channels of the MSE system, measuring the collisionally induced fluorescence (CIF) from the heating neutral beams, were installed during the summer 2002 outage and will be progressively commissioned starting in the FY'04 run. The remaining 8 channels of the CIF system will be added in later in the year. When this system has been calibrated and its performance fully characterized, its data will be incorporated into the real-time analysis beginning in FY'06. The data requirements for this are modest and the calculations are relatively straightforward although a considerable amount of calibration data will be involved and provisions must be made to keep this data current. The equipment required for gathering this data from the MSE room is similar to the requirements for the MPTS data, *viz.* high-speed digitizers, a VME crate and controller and fiber optic link modules. Faraday rotation measurements from the FIReTIP polarimeter will also be included in the rt-EFIT analysis, although since the total Faraday rotation on a measurement sightline is the path integral of the product of the local magnetic field and the density, a double inversion of the data is required to obtain the local magnetic field.

In FY'05, a new MSE diagnostic based on laser-induced fluorescence (LIF) will be installed. This diagnostic can provide a measurement of |B|, the magnitude of the local magnetic field, as a function of time and space throughout the plasma. These data can be incorporated into the rt-EFIT analysis to constrain the total plasma pressure profile, including the fast ion component. This would represent a significant extension of capability for real-time control, with obvious application to future ignition devices where there will be populations of energetic alpha particles. This would also enable real-time calculation of MHD stability and transport coefficients.

<u>2006 – 2008</u>: *Real-time calculation of MHD stability limits* – With the inclusion of data from both the kinetic and magnetic profile diagnostics in the rt-EFIT analysis, the calculated pressure and q-profiles could, in turn, be used to calculate the proximity of the equilibrium to stability boundaries, thereby permitting feedback on the heating power, or other means, to avoid instabilities. The first implementation of such a stability analysis might involve simple parametrizations of the MHD pressure limits in terms of normalized-β, the internal inductance l_i and the pressure profile peaking factor $p(0)/\langle p \rangle$ as has been quite successful in characterizing the limits in NSTX so far. Further development to assess the stability of specific classes of modes, such as energetic particle instabilities and edge instabilities could be added as theoretical analysis and tools are developed.

3.6.4 Provision of Tools and Actuators

<u>2004</u>: *Improving the speed of power supplies for vertical position control* - Experiments will be conducted in FY'04 to increase the plasma cross-section elongation to improve MHD stability. To control plasmas with elongation approaching $\kappa = 3$ and further away from the passive stabilizer plates than present equilibria, faster response than is available from the existing phase-controlled rectifiers and their communication link will be needed. The control requirements for these fast equilibrium control supplies, the existing PF power supplies and the RWM control supplies (discussed below) will be assessed together so that all power supplies can use common hardware and communication software. One possibility for providing greater speed is the fast-switching power supply being built in a collaboration with the University of Washington to control the current in the two new CHI control coils installed during the 2002 outage to null the poloidal field in the vicinity of the absorber insulator. These power supplies can provide fast regulation of a DC current provided by one of the existing phase controlled rectifiers.

For controlling the vertical instability at the highest plasma elongations, a reconnection of the toroidal and top-bottom links between the passive stabilizers or even a reconfiguration of the plates to match the plasma boundary more closely may be required. Such a reconfiguration would be undertaken in FY'06 if experiments indicate a continuing favorable trend of performance with increasing elongation of the plasma cross-section. The reconfiguration of the PF1A coils, discussed in Ch. 2, to allow simultaneous high triangularity and elongation has no control impact, at least as currently envisioned.

2004: Control of the NBI heating power - It is planned to implement control of the heating power to maintain β close to but below the evolving stability limit. This depends on full implementation of rt-EFIT analysis with appropriate diagnostic inputs. With only three NBI sources available, pulse-width modulation of the sources will be required to achieve fine control. In addition to providing a suitable real-time data link from the control computer to the NBI controllers, the algorithms for choosing the modulated source(s) and the characteristics of the modulation (time on *vs.* off) must be developed. This control for the NBI power will be progressively enhanced as capabilities for real-time analysis of the stability limits are developed.

2005 – 2007: Feedback stabilization of MHD Instabilities - For the stabilization of MHD modes, several new tools may be required. In principle, complete control of the radial profiles of pressure and toroidal current would allow optimization of ideal MHD stability, but providing these capabilities would probably not be energetically favorable in the face of plasma transport processes, so more efficient means are required for specific classes of instability.

As expected from theoretical studies, the outboard passive plates provide significant stabilization of kink-like modes if β is sufficiently high to cause the magnetic perturbations to become large on the outboard side. The experiments from FY'02 indicated that the reduction in the error fields arising from the careful realignment of the PF5 coils has helped to suppress the RWM [8], although it has not been eliminated. To control the RWM growth at even higher β, active control of the error fields, which can be amplified by the plasma itself, will be needed. A set of coils mounted outside the vacuum vessel to provide radial field corrections varying in space and time will be installed in FY'04. Initially, these coils will be used to null the apparent time-varying error fields responsible for the rotation damping and subsequent RWM growth. The corrections fields to be applied will be inferred from measurements in ensembles of discharges and averaged over a suitable moving time window. For this control strategy, the frequency response of the power supplies is modest. However, for eventual real-time feedback control of the RWM, the required frequency response will exceed that of the existing phase controlled rectifiers operating at standard line frequencies and additional fast power supplies will be required. The RWM coils will be upgraded to a set inside the vacuum vessel if the external coils are found to be inadequate for dynamic mode stabilization.

Neoclassical tearing modes (NTMs) have been identified in some NSTX plasmas when β_p exceeded a value around 0.4 and *q(0)* (= q_{min}), as determined by EFIT from the external magnetic data, was below 1.5. An example of the island structure for such a mode in an NSTX plasma is shown in Fig. 3.6.4. More recently, plasmas with $q_{min} > 2$ have reached β_P up to 1.2 for extended periods (>0.2s) without the growth of NTMs. This suggests that control of the q-profile to maintain $q_{min} > 2$ may be sufficient to avoid NTMs. However, the optimum q-profile for stability to ideal pressure-driven instabilities has not yet been determined. It is possible that NTMs will become a problem in future high-β plasmas in NSTX if, for example, it is deemed advantageous to run with lower q_{min}. In tokamaks with normal aspect ratio, NTMs have been controlled by current drive applied at the location of the NTM island to replace the local bootstrap current, the absence of which is responsible for the island growth and which is reduced by the flattening of the pressure gradient in the island. Current drive by Electron Bernstein waves (EBW) is envisaged as the main control tool for NTMs in NSTX. A method to control the toroidal and/or poloidal angles at which the waves are launched will need to be developed in order to optimize the coupling and power deposition for the necessary localization of the driven current. A 1 MW, 15 GHz system is proposed for FY'06 with an upgrade to higher power in the following year.

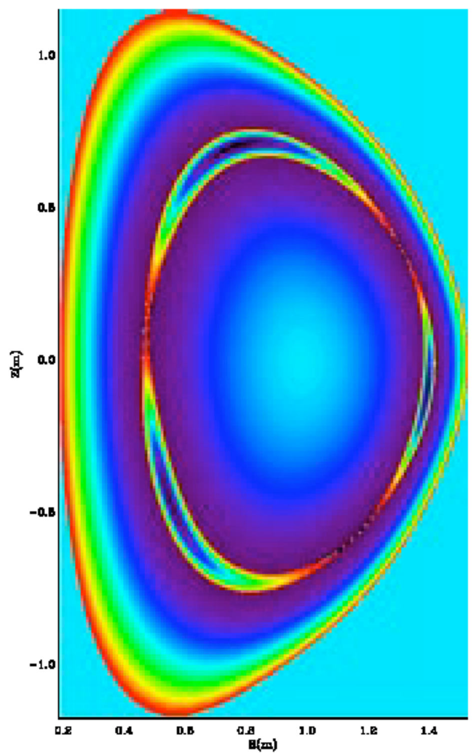

Figure 3.6.4 Simulated island structure for a m/n = 3/2 neoclassical tearing mode for NSTX shot 103698 at 192ms. The island width is 10% of the minor radius.

<u>2004 – 2007</u>: *Control of plasma fueling and exhaust* - As in standard tokamaks, fueling and density control are major issues for the ST. Already, the limitations of gas fueling from the edge and the effects of

strong recycling at the first wall are impacting NSTX operation. In L-mode plasmas the fueling efficiency is low for gas injected on the low-field side (LFS) and access to the H-mode has proved difficult with this method. Access to the H-mode has been improved by fueling with gas injected from the center column, *i.e.* on the high field side (HFS). However, this gas is introduced from an external plenum through a narrow, long pipe. Once the injector valve is opened, there is essentially no control over the flow rate which reaches a maximum determined by the fill pressure and then decays slowly, providing continuous, uncontrollable fueling and contributing to the observed monotonic density rise during the H-mode. This limits the range of conditions available for experiments and prevents the assessment of the confinement characteristics in steady state. The H-mode density profile can also develop a significant peak near the boundary where the transport appears to be reduced but the particle source from the fueling and recycling is strong. Hence, a primary concern for NSTX is the development of tools which are effective for controlling the flux of neutrals to the plasma edge, particularly in NBI heated discharges.

For the FY’03 run a new HFS injector was installed with a shorter, larger diameter pipe which introduces the gas from the center column at the “shoulder” above the midplane, rather than at the midplane itself. Some control of the gas flow rate in time is possible with this injector. However, in the brief FY’03 run, it appeared that H-mode access was less reliable than for the midplane HFS injector.

A prototype system is being developed to inject gas supersonically from a shaped nozzle near the midplane on the LFS. Supersonic gas injection through the plasma edge has been shown to be beneficial in other tokamaks. If these benefits are confirmed in NSTX, the injector will developed into a permanent system in FY’04 and its control will be incorporated into the real-time control system.

A deuterium pellet injector is planned for installation on NSTX in FY’05. Success was obtained in TFTR in maximizing the fueling rate from multiple pellets by triggering the injection of subsequent pellets when the electron temperature had reheated after the preceding pellet. Such control would be possible in NSTX once the electron temperature data is available in real-time in the control computer. In standard aspect-ratio tokamaks, pellets injected on the high field side have shown the expected improvements in penetration compared to low-field side injection. This effect, which is related to the ∇B drift, could be expected to be more significant in the ST and might permit use of smaller pellets. The capability for HFS pellet injection will be introduced in FY’06

For controlling the density profile, modifying the wall material and providing active pumping in the divertor will be studied in NSTX. Both these techniques have proved successful in other tokamaks. For example, coating the carbon surfaces with lithium in TFTR produced dramatic changes in the recycling and improvements in confinement and fusion performance. In FY'03, an injector for low-velocity lithium or boron pellets will be installed on NSTX. If pellets produce beneficial effects on recycling, density profile control or plasma performance, a lithium evaporator, currently being developed on CDX-U, will be installed on NSTX in FY'05, or possibly FY'04 if results from CDX-U are sufficiently encouraging. Control of the plasma boundary relative to the actively pumping surfaces will clearly be required for optimal plasma exhaust.

A design study has been conducted for a cryo-pump to be installed in the divertors of NSTX. A decision on proceeding with this will be made at the end of FY'04 with availability expected in FY'06. Such a system may require a reconfiguration of the secondary passive plates and would have many implications for plasma control and integration, since the pumping rate will be highly dependent on the plasma configuration in the x-point region and through the pump throat, which in turn has consequences for plasma shaping and thus stability. Maintaining adequate flexibility for plasma shape control will be a essential component of the system design.

Another possible method for plasma exhaust is a liquid lithium surface module. Experiments to assess the potential of this method are being conducted on CDX-U and the PISCES facility. The major problem with this approach is making the system compatible with machine operations, in particular with disruptions, CHI, and HHFW operations. In addition to control of the plasma interaction with such a module during normal operation, control strategies would have to be developed for off-normal events, to prevent disruption of the lithium containment, and possibly its flow, in its support structure. Assuming a successful outcome of the enabling research, such a module would be installed in FY'08.

Power handling in the divertor is a critical issue for the ST. Experiments are being planned for the FY'04 run to characterize the edge plasma and the power and particle flows to the divertor. If this research indicates that power fluxes will become unacceptable in long pulse operation, one possibility for ameliorating the divertor power flux is to increase the edge radiation, either by changing the density to enhance the radiation from intrinsic impurities or by injecting a suitable recycling (noble) gas. The issue will be localizing the radiation to the divertor while avoiding both radiation from and dilution of the core plasma. Additional gas injectors in the divertor and their integration into the overall plasma control strategy will be needed.

If high power fluxes to the divertor cannot be avoided by other means, changing the divertor tiles to a carbon-fiber composite or another advanced material and sweeping of the strike points will be needed for long pulse operation in NSTX. Feedback control on the local surface temperature on the tiles is a possibility. The present NSTX coil set allows the outer-strike point to be swept ~ 30cm but the plasma triangularity changes accordingly from 0.3 to 0.8.

3.6.5 Flowchart of Planned Developments

The phased development of control capability and the integration of those elements to meet the goals of the NSTX research program are presented as a flow chart in Fig. 3.6.5.

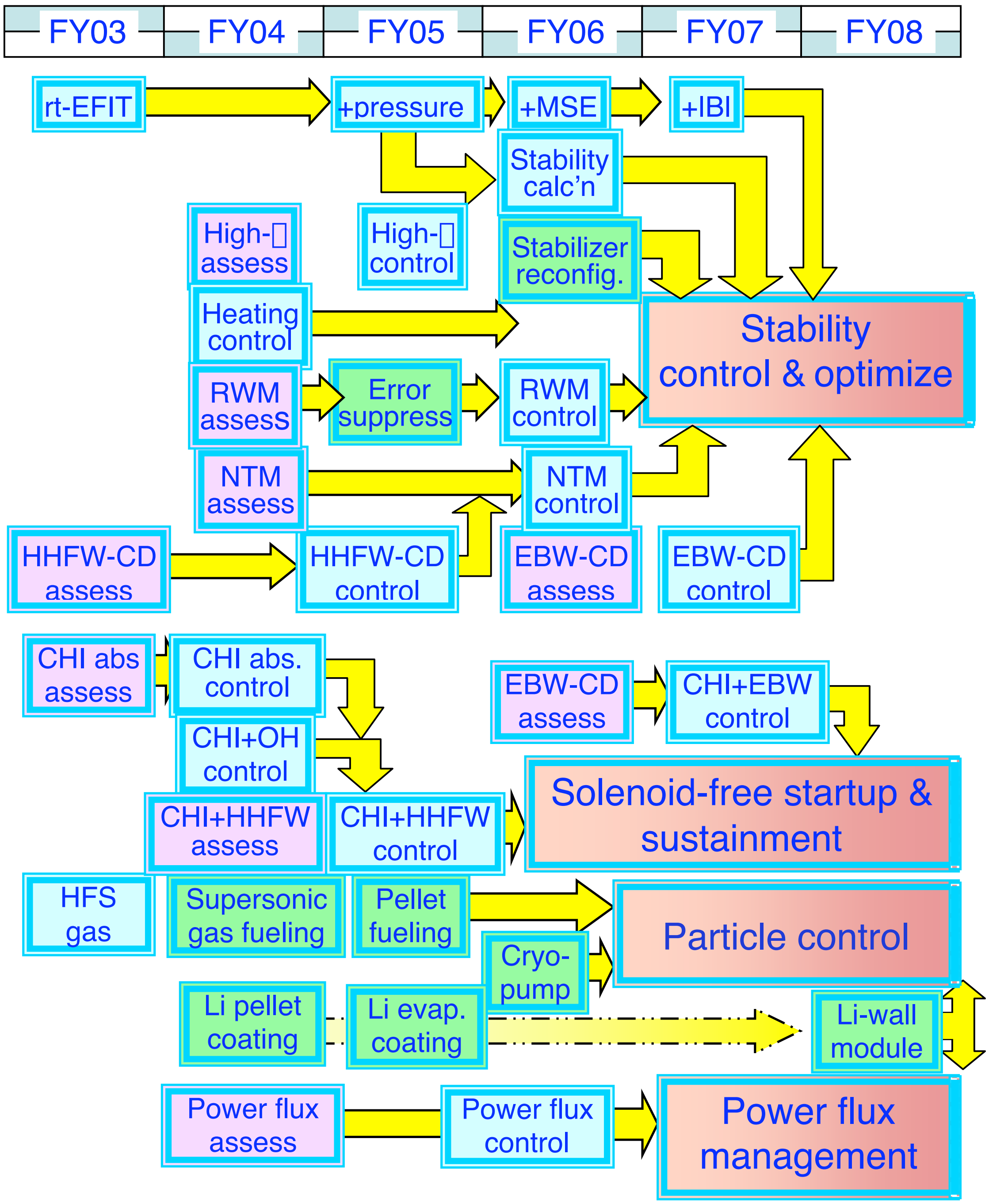

Fig. 3.6.5 Flowchart of developments in plasma control and their integration to achieve the research goals of NSTX. A light purple background indicates an experimental assessment in support of development while a green background is a major hardware element. The elements with a blue background represent the implementation of control for each capability or tool.

References for Sec. 3.6

[1] "*Long Pulse, High β, and bootstrap sustained scenarios on NSTX*" D. A. Gates and the NSTX Team, Phys. Plasmas **10** 1659 (2003)

[2] "*Control System development plan for NSTX*" D. A. Gates, *et al*., IEEE - Trans. Nucl. Science, **47,** 222 (2000)

[3] "*Upgrade for the NSTX control computer*" D. Mueller, *et al*., ., IEEE - Trans. Nucl. Science **47,** 219 (2000)

[4] "*A Structured Architecture for Advanced Plasma Control Experiments*" B. G. Penaflor, *et al*., Proc. of the 19th Symposium on Fusion Technology, Lisbon, Portugal, Vol. **1** (1996) p. 965

[5] "*NSTX Real Time Plasma Control Data Acquisition Hardware*", R. Marsala and H. Schneider, to be published in Proc. of 22nd Symposium on Fusion Technology (SOFT) Helsinki, Finland (2002)

[6] "*Initial Operation of NSTX with Plasma Control*", D. A. Gates, *et al*., in the Proceedings of the European Conference on Plasma Physics and Controlled Fusion (Budapest), PPPL Report No. 3453

[7] "*Real Time Equilibrium Reconstruction for Control of the Discharge in the DIII-D Tokamak*", J. R. Ferron, *et al*., Proc. of the 24th Euro. Conf. on Contr. Fusion and Plasma Phys., Berchtesgaden, Germany 1997, Vol. 21A, Part III (European Physical Society, 1998) p. 1125

[8] "*β-Limiting MHD instabilities in improved-performance NSTX spherical torus plasmas*" J. E. Menard, *et al*., Nucl. Fusion **43** (2003) 330