FES Joule Milestone 2009
Third Quarter Report
June 30th, 2009

Annual Target: Conduct experiments on major fusion facilities to develop understanding of particle control and hydrogenic fuel retention in tokamaks. In FY09, FES will identify the fundamental processes governing particle balance by systematically investigating a combination of divertor geometries, particle exhaust capabilities, and wall materials. Alcator C-mod operates with high-Z metal walls, NSTX is pursuing the use of lithium surfaces in the divertor, and DIII-D continues operating with all graphite walls. Edge diagnostics measuring the heat and particle flux to walls and divertor surfaces, coupled with plasma profile data and material surface analysis, will provide input for validating simulation codes. The results achieved will be used to improve extrapolations to planned ITER operation.

Quarter 3 Milestone

Third Quarter - Experiments will have been carried out at multiple facilities. Experimental analysis and preliminary modeling will be in progress on results from multiple facilities. Make an initial evaluation of the results to date and adjust research plans as necessary.

Completion of 3rd Quarter Milestone

The 3rd quarter milestone has been completed by carrying out experiments on the facilities as well as analysis and modeling of some of the results. This quarter continued to see significant progress in carrying out dedicated, coordinated experiments at the facilities towards the 2009 Joule milestone. Research progress and activities are organized by facility below.

NSTX

The FY2009 retention milestone experiment was performed over 4 run days dedicated to ohmic and neutral beam heated plasmas both before- and with- lithium conditioning. In addition a sample probe designed to expose 4 samples to the plasma was commissioned and used to expose samples to these plasmas, retrieve them and analyze them the same evening. Further analysis was done off-site at Purdue University. Extensive calibrations of the pressure gauges and the neutral beam were done in support of this experiment. Preliminary results show high (>90%) prompt retention followed by outgassing that reduced the retention fraction to ~50%. The data analysis is ongoing

A new high accuracy baratron (MKS 690A) with calibration traceable to a NIST standard was installed on NSTX. This provides an absolute pressure measurement with a stated accuracy of 0.08% and without the complications of gauge factors for different gas species associated with ionization gauges. The baratron could be re-zeroed and read out remotely via an RS232 link. Calibrations of all the vessel ionization gauges, micro-ion gauges and residual gas analyzers were performed with this baratron as a standard by closing the turbomolecular and neutral beam torus interface valves and puffing varying amounts of gas into the vessel and comparing the pressure readings. The calibration was performed over the 1e-6 to 1e-3 torr range for both nitrogen and deuterium. The small effect of the TF field on the in-vessel microion gauges was measured separately.

For retention measurements of neutral beam heated plasmas it is necessary to take the neutral beam cold neutralizer gas into account. This was measured separately by operating the neutral beams into a closed vessel, with the ion accelerator voltage off and measuring the pressure rise. The neutral beam cryopumping speed was measured by injecting a fast gas pulse from the lower dome gas injectors and tracking the pumpout by the cryopanels (turbomolecular pumps closed). Because of the transition between viscous and molecular flow through the pump duct, the effective pumping speed is somewhat pressure dependent above 1e-4 torr.

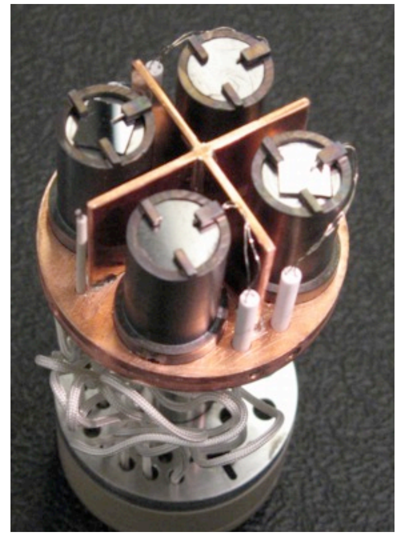

Fig. 1 Sample probe with ATJ graphite, Si and Pd samples

The vessel leak rate was measured before the start of plasma operations, however a small leak was plugged after the measurement, changing the leak rate and rendering the measurement moot. The leak rate was re-measured after a 10-day maintenance period when the outgassing rate should be negligible.

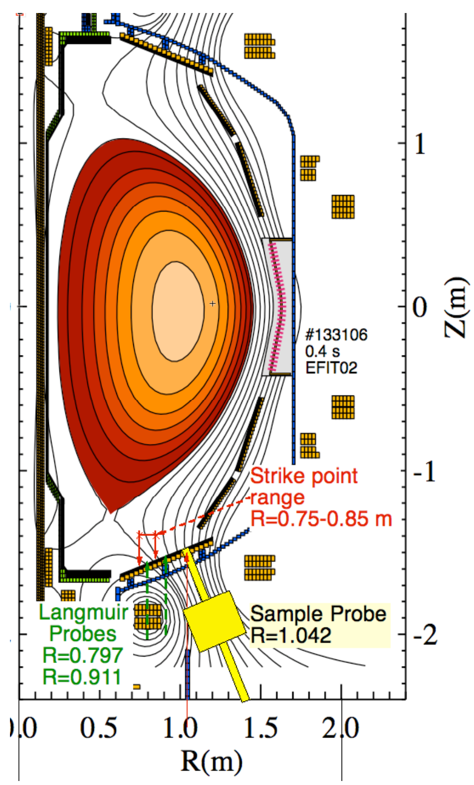

Fig. 2 Discharge magnetic geometry with sample probe location.

To address the milestone charge "*to develop understanding of particle control and hydrogenic fuel retention in tokamaks*" a sample probe was designed, constructed and commissioned in time for the retention experiments. This used a 4 ft. Thermionics linear translator to introduce four material samples into a tile gap in the NSTX outer divertor, expose them to dedicated plasmas and retrieve them for analysis later the same day without venting the vessel (Fig. 1). This rapid analysis of freshly exposed samples is important as lithium is chemically active and the chemical state of lithium coatings on plasma facing components can change quickly. The probe was designed to perform thermal desorption spectroscopy (TDS) in a chamber beneath the NSTX vessel promptly after sample exposure to several discharges. The design called for four samples on a 2" diameter probe with a secure disruption-proof mechanical attachment, thermal cooling, 16 thermocouple wire connections, two Langmuir probe connections and at least one heater connection. The design also required that the probe could be installed and removed by one hand reaching through an argon-filled glove bag and through 4" diameter port and without using any tools. The design met all the specifications and has been successfully used for all the retention run days so far.

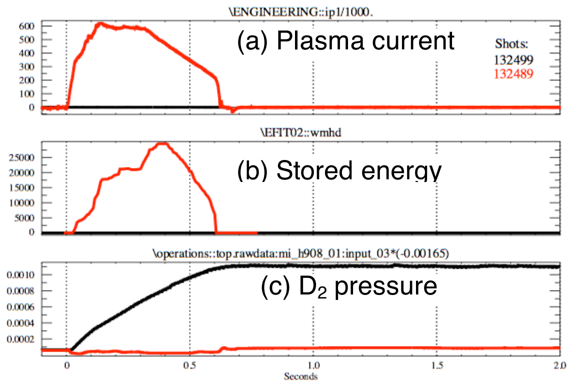

Fig. 3 (a) plasma current (b) stored energy (c) deuterium pressure for ohmic and gas-only pulses before Li conditioning. The prompt retention is 93%.

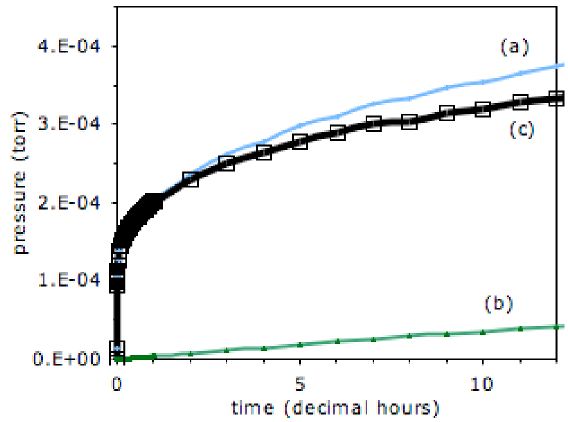

Fig. 4 Pressure rise over 12 hours (a) deuterium (b) nitrogen leak rate (c) difference .

Baseline discharges were developed with the outer strike point on the outer divertor, close to the sample probe (Figure 2). These discharges also were arranged to have a controlled ramp down of plasma current with plasma termination at very low stored energy. This was to avoid a tail-end disruption that could heat the tiles and influence the measured retention. IR camera measurements confirmed that there was no significant heating at discharge termination. Baseline discharges with ohmic heating only and also with neutral beam heating were used. The advantage of ohmic discharges is that all the torus interface valves to the turbo-molecular pumps and the neutral beam cryopumps could be closed, and the retention could be simply measured by comparing the vessel pressure rise to the rise of a gas-only pulse without a plasma (as was done previously at MIT). The neutral beam heated plasmas have significantly higher stored energy, but in this case the deuterium fueling and deuterium pumping by the beam box needs to be carefully tracked. To cross-calibrate the two situations, the retention of an ohmic plasma with and without the neutral beam torus interface valve open was compared and the difference found to be in the 5-10% range.

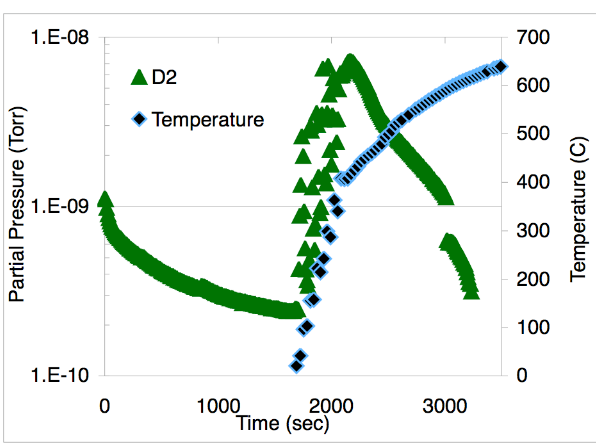

Fig. 5 Same-evening thermal desorption spectroscopy of ATJ graphite sample exposed to 6 NSTX neutral beam heated plasmas.

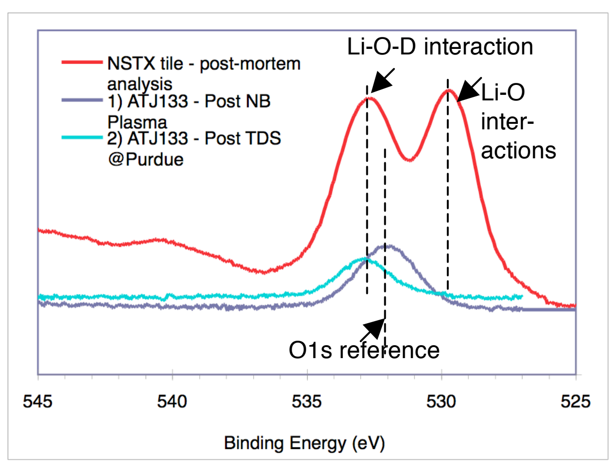

Fig. 6 Oxygen 1s spectrum of X-ray photo-electron spectroscopy (XPS) at Purdue Univ. of ATJ graphite sample exposed to 6 NSTX neutral beam heated lithium conditioned plasmas.

Experiments with ohmic (day one) and neutral beam discharges (day two) were performed before the application of any lithium conditioning. The experiments were then repeated on two additional days with a total of 1.7 and 3.3 g of lithium evaporated into the vessel on the ohmic and neutral beam days respectively. On each day two ATJ graphite samples, one Si and one Pd coated Si sample were exposed to 6-8 discharges and withdrawn from the vessel the same evening. Using the built-in heater one ATJ sample was heated to ~ 600 C and a residual gas analyser used to measure the thermal desorption spectrum (TDS). All the samples were then removed under argon and shipped to Purdue University for further TDS and x-ray photoelectron spectroscopy (XPS) analysis. The vessel pumping valves were closed before the last ohmic discharge and remained closed for 12-48 h to monitor the outgassing. For neutral beam discharges the neutral beam torus interface valve was closed immediately after the last shot of the day and the outgassing monitored for 12-48 h also.

Full data analysis will be performed in the fourth quarter. Preliminary analysis indicates that prompt retention following the discharge was 90% or higher, and the subsequent outgassing lowered the retention to the 50% range. X-ray photoelectron (XPS) data from Purdue University shows the presence of lithium-oxygen and lithium-deuterium functional groups elucidating on possible hydrogen retention mechanisms. XPS spectra from lithiated graphite samples will be compared to controls (without Li samples), Si and Pd witness samples. Four-point probe technique is used with Pd sample exposed to non-Li shots to quantify deuterium concentration on the PMI probe complementing D particle current measurements from nearby probes. Si witness samples are being analyzed to quantify the lithium layer thickness deposited and cross-calibrate against Li deposition on ATJ graphite samples. Additional work at Purdue University will include scanning electron microscopy (HR-SEM) to correlate surface morphology with measured retention from TDS, XPS and 4-point probe data. A sample of the data is in Figs. 3-6. Preliminary results from this experiment were presented at the 12th International Workshop on Plasma Facing Materials and Components for Fusion Applications in Juelich, 11-14 May 2009.

<u>Alcator C-Mod</u>

The Alcator C-Mod Ideas Forum was held in April 2009. Two experiments are scheduled to be performed in FY2009 towards supporting the 2009 Joule Milestone.

1. The effect of fusion-reactor He fractions on D retention in Molybenum

2. Static vs. dynamic particle balance: Understanding the effects of confinement mode and divertor pumping

Experiment 2 in particular has grown out of the experiences and insights gained from using the static particle balance on DIII-D in 2009 (see below). By combining the two particle balance techniques on both C-Mod and DIII-D, better insights on the underlying physics of particle retentions will be gained. The experiment will compare dynamic particle balance in various stages of plasma confinement and heating levels.

Preliminary numerical modeling has been applied to deuterium retention and release in molybdenum based on C-Mod particle balance experiments [1,2]. The numerical model solves the time-dependent coupled particle continuity equations for D transport and trapping in refractory metals like W and Mo using accepted temperature dependent rate coefficients from literature (details can be found in [3] and [4]). At this stage, the purpose of the modeling is to ascertain whether the observations of D retention and release in C-Mod are consistent with the known interactions of H fuels in molybdenum (Mo). Unique among divertor tokamaks in the world, C-Mod plasma-facing components are only made of bulk refractory metals (99% Mo, <1% W). There are two principal observations of D fuel retention in Mo that we are particularly interested in modeling: 1) A constant rate of D fuel retention in the Mo divertor at ~ 1% of the incident ion fluence, and 2) Substantial release of retained D from the wall caused by disruptions.

Alcator C-Mod has carried out extensive experiments on global fuel retention using the static particle balance technique on cleaned Mo walls [1]. It is found that D fuel retention is constant from shot-to-shot, i.e. the wall continues to absorb D fuel particles and not release them. The rate of retention is ~ 1% of the total D ion fluence incident on the Mo divertor and main wall PFCs (Fig. 7). This behaviour was confirmed out to ~30 seconds of plasma exposure (about 15 shots) with no indication that the wall reservoir was saturating. A series of studies using different magnetic divertor geometries, Helium plasmas and the external applications of Boron films [1] strongly suggested that the D was being retained at the outer divertor of Alcator C-Mod; a location which is known to be in net erosion and have "bare" Mo surfaces (i.e. the Boronization films are quickly eroded from this region by plasmas). This is in stark contrast to carbon PFC tokamaks which find the long-term retention is dominated by codeposition in plasma-deposited at the inner divertor. codeposition in Boron was ruled out in C-Mod by doing experiments where all B films had been removed. Post-exposure analysis of C-Mod Mo tiles (Fig. 8) showed a volumetric concentration of D/Mo 1% down to the limit of the ion-beam analysis detection (~4

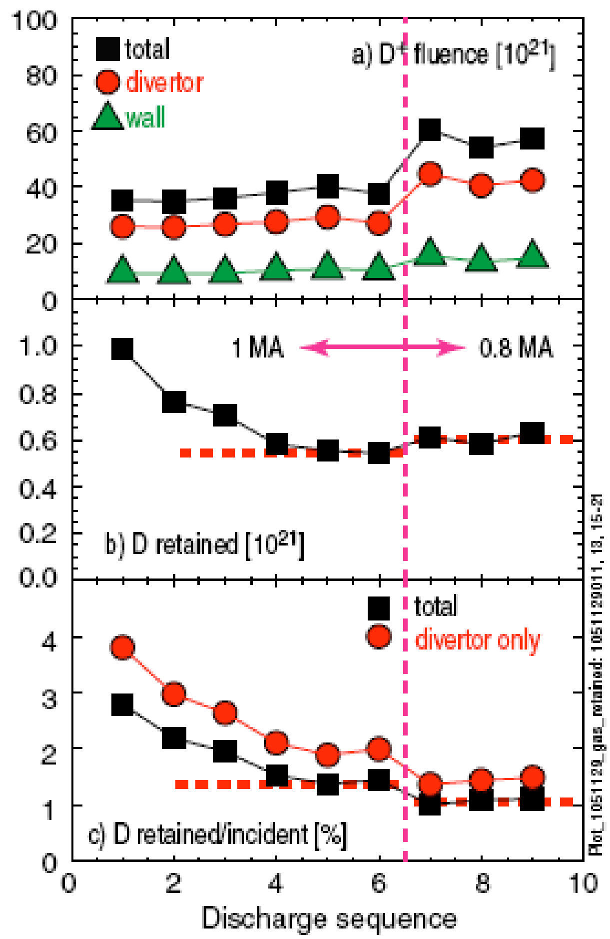

Fig. 7 Deuterium fuel retention on C-Mod in a sequence of repeated discharges. [1]

microns). Such long-term storage of D in Mo takes place in traps, i.e. sites of potential energy wells within the Mo from which the D cannot escape at thermal energies once it falls into the well, however the stated natural trap density in Mo is ~10^{-5} trap/Mo, much lower than seen in the C-Mod tiles. Experiments in DIONISOS [3] also showed that D/Mo ~1% could be obtained when the Mo was damaged by energetic ions from the plasma and/or high-energy beams. This suggested that the Mo was somehow being damaged in C-Mod under high particle flux bombardment from the plasma such as to create trap sites.

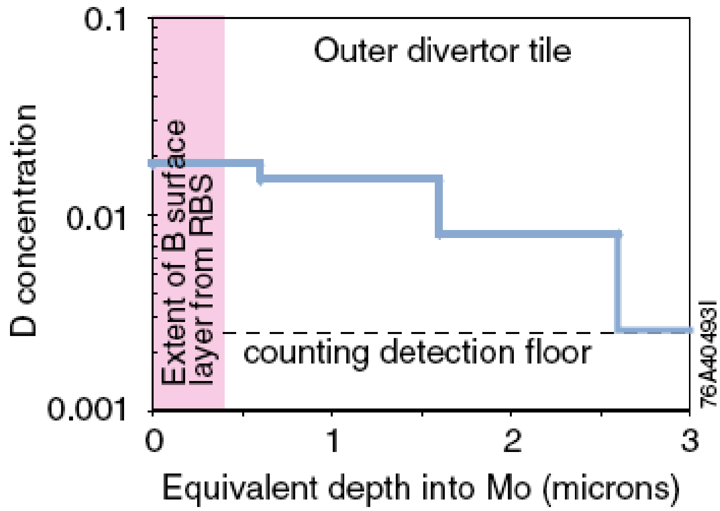

Fig. 8 Post-campaign Deuterium depth profile in Molybdenum tile using Nuclear Reaction Analysis. D concentration is normalized to Mo atom density [1].

Given this background the numerical model is used to examine how the Mo surface in the C-Mod divertor would respond to typical incident ion flux so as to match the experimental results. The most important process is diffusion; the plasma Deuterons are implanted at a shallow depth (~10 nm) and then diffuse both back to the surface and into the bulk where they can find vacant traps and therefore be retained. The model uses laboratory measured temperature dependent diffusion coefficients. Also important is the assumed trap (empty) density and their distribution in the material. Based on the tile analysis we set the maximum trap density at 1% volumetric concentration to Mo. We show two simulation examples.

The first example (Fig. 9) assumes that the traps are being produced at the surface by the D bombardment itself, at a rate of 1% traps per incident ion (there is a possibility the damage is caused by low-Z impurity ions in the plasma). The traps are therefore only produced at the surface and are then assigned a diffusion coefficient equal to that of the solute D. It is well known from laboratory experiments that the traps have finite mobility in the Mo [3] but this is certainly an over-simplification. An average temperature of 600 K is assigned for the outer divertor of RF heated discharges, and the measured typical ion flux density ~10^{23} ions/s/m^2 is used. The simulation shows that the D and trap diffusion is sufficiently fast that up to ~30s the retained D is linearly increasing at a rate of 1% of incident ion fluence (of course this is somewhat “forced” by our imposition of 1% trap production efficiency). While this linear retention with fluence is valid to the limits of the C-Mod experiments (~30 s), the model

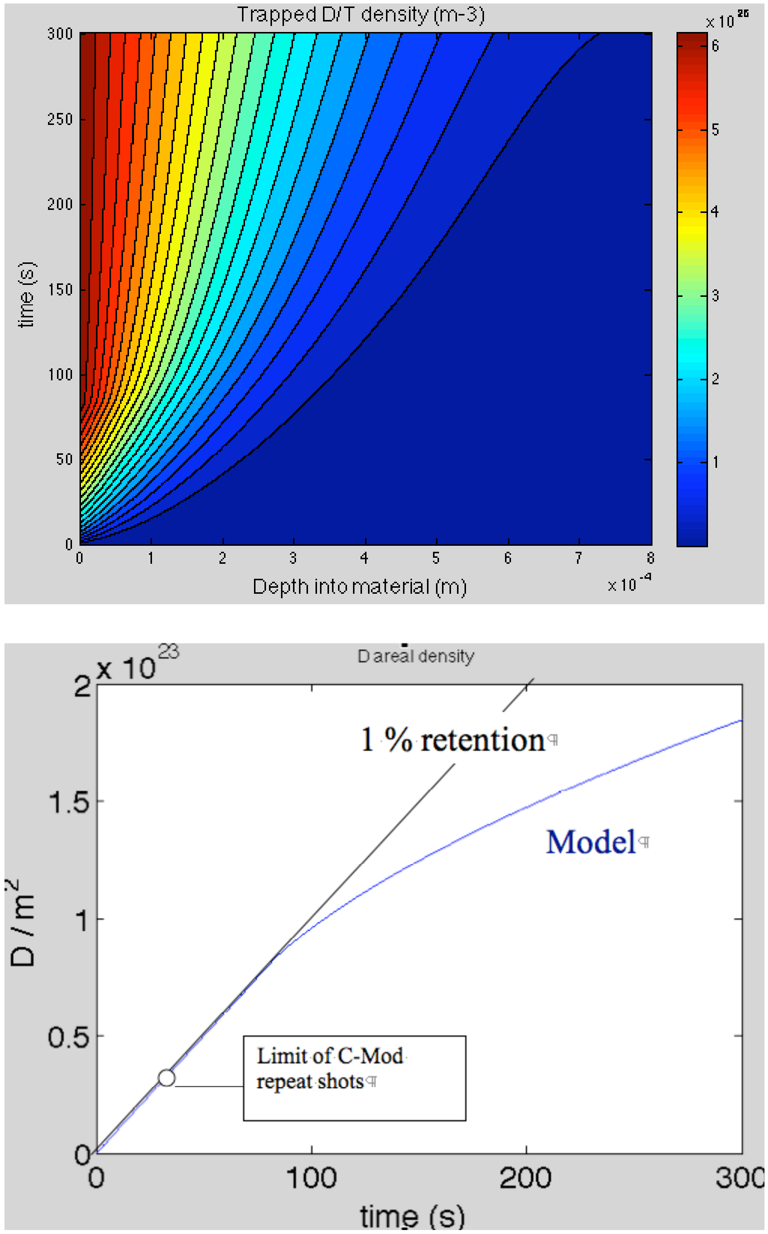

Fig. 9 Numerical simulation of D retention in Mo at divertor with surface trap production and mobile traps. (Top) Contour of trapped D versus time and depth into Mo (Bottom) Integrated D fluence retained.

suggests the retention should begin to slowly saturate starting at ~100s; at this time the diffusion starts to limit the D access to deeper locations and since 1% is the maximum allowed concentration this is required to store more D. This suggests the retention would be limited at this temperature for very long pulses, which is positive for devices like ITER. At the same time a concern is that the D is stored so deeply in the Mo; ~50 microns even for the short timescales of C-Mod experiments, raising a concern that recovery of this D will be difficult.

The second simulation example (Fig. 10) shows that if the trap concentration is 1%, but immobile, then the retention should appear to follow permeation and increase with square root of time on the C-Mod shot timescales ~30 s. This does not apparently occur in the experiments, highlighting the importance of trap mobility in this process.

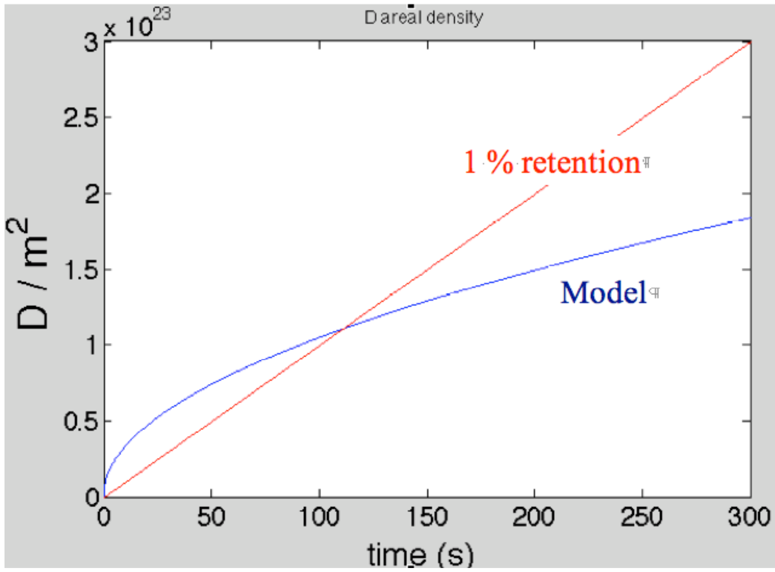

Fig. 10 Numerical simulation of D retention in Mo at divertor with immobile traps at 1% concentration.

The conclusion of simulating the retention is that it is physically possible for the Mo to retain Deuterium as seen in the C-Mod experiments, mainly by allowing access of the Mo to deeper trap sites through diffusion. However there are many unanswered questions as to the production and mobility of the traps, which seems to be the most important processes occurring in setting the retention level.

We now move to the second overall observation from C-Mod: disruptions release large amounts of the retained hydrogenic (H or D) fuel. Disruptions appear to play a major role in limiting global retention in C-Mod. An examination of particle balance over a span of 300 shots [1] (Fig. 11) shows that the net wall uptake is limited to ~zero solely due to the natural occurrence of disruptions. On average, disruptions at full current (~15% of all discharges) lead to release of 6-7x the amount of D retained in a single non-disruptive shots (all discharges in the previous discussion on retention were non-disrupting). Rapid heating of the Mo is the underlying reason for the D release. In order to thermally release the D from trap sites (typical energy ~1.4 eV) the temperature must be elevated substantially above room temperature, while at the same time the

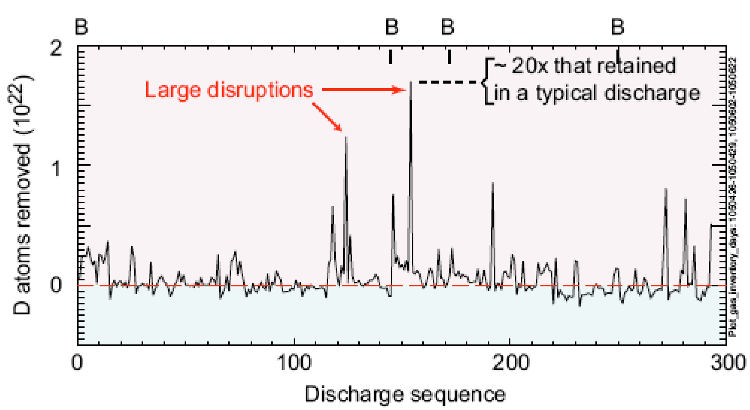

Fig. 11 Shot-to-shot particle balance in C-Mod. B marks times of boronizations [1].

D diffusivity rapidly increases, allowing the liberated D to "escape" to the surface and be released as D_2 molecules. These critical rates (de-trapping, diffusion, recombination) are governed by the Arenhhius relationship and are therefore exponentially sensitive to temperature. This suggests that a minimum amount of energy density is required in the short timescales of disruptions (1 ms) to sufficiently heat the Mo, or else minimal D will be recovered. This expectation is supported by C-Mod experiments (Fig. 12) where the stored core energy/temperature of planned disruptions was slowly increased. When the core T exceeded ~ 2 keV (stored energy 40 kJ) the release of H from the walls increases exponentially.

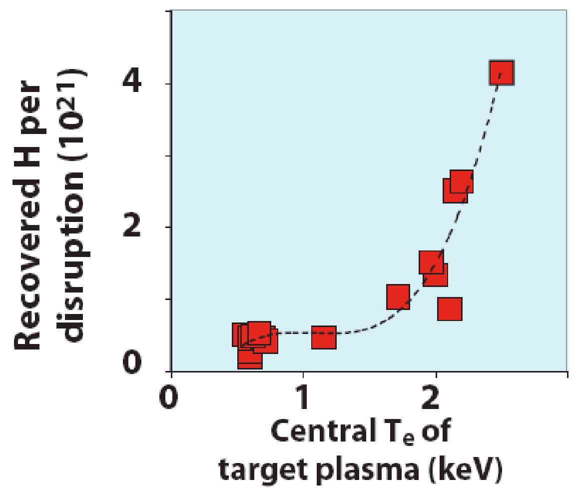

Fig. 12 Hydrogenic fuel recovery on C-Mod following intentional disruptions with varying target plasma Te (at constant density). [2]

The numerical simulation is used to model the release of D during disruptions (Fig. 13). The D is assumed to be retained at 1% D/Mo up to 50 microns depth as suggested by the retention modeling. A trap energy of 1.4 eV is assumed based on laboratory experiments. Taking typical plasma stored energy (thermal + magnetic) in C-Mod suggests a range of temperatures above 1000 K could be obtained for the divertor tiles in a 1 ms heat pulse due to a disruption. Fig. 13 shows that temperatures above 1000 K are sufficient to strongly deplete the stored D, even on a 1 ms timescale, although the depletion is strongly dependent on temperature assumptions. This simulation result is then qualitatively consistent with the observation that rapid heating during a disruption is sufficient to recover the D retained over 10's of seconds of regular C-Mod discharges. This also suggests controlled disruptive heating can be a useful tool to limit retention.

In summary the preliminary numerical simulations have shown consistency with the experimental C-Mod results on D retention and release, although many details need to be resolved. In particular the mechanism of trap formation and mobility in the deuterium need to be understood much better.

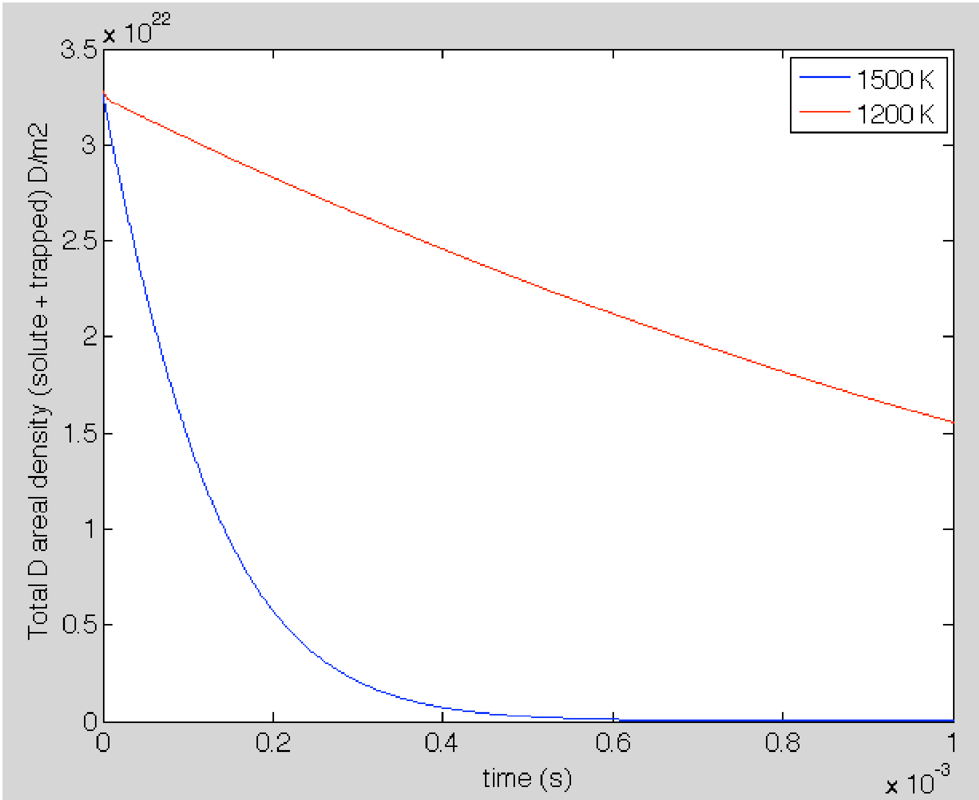

Fig. 13 Simulation of disruption-heating induced reduction of retained D in Mo for two different assumptions of surface temperature.

[1] B. Lipschultz, D.G. Whyte, J. Irby, B. LaBombard, G. Wright *Nuclear Fusion* **49** (2009) 045009.

[2] D.G. Whyte, *et al*. Proc. 21st IAEA Fusion Energy Conference, Chengdu, China, October 2006, EX/P4-29.

[3] G.M.Wright Ph.D. Thesis U. Wisconsin 2007.

[4] D. G. Whyte, J. Nucl. Mater. **390-391** (2009) 911.

DIII-D: *Synopsis* Scientists from DIII-D and C-Mod conducted particle balance experiments in DIII-D to measure wall retention with carbon PFCs to compare with similar experiments in C-Mod and NSTX. Data were obtained from the ohmic, L-mode, and H-mode phases in discharges heated by neutral beam injection (NBI) and in discharges heated by EC. The DIII-D results show that *dynamic* (DIII-D) and *static* (C-Mod) particle balance techniques are in quantitative agreement. The wall retention rate was greatest during ramp-up and L-mode. Nearly zero wall retention rates in graphite during both EC and NBI heated H-modes were measured.

Detailed Report

The centerpiece of the work on the 2009 FES joint research milestone this quarter was the planning and execution of a joint particle balance experiment between the C-Mod and DIII-D Boundary groups. In particular, Dennis Whyte and Bruce Lipschultz from MIT played key roles. The goal of this experiment was to compare two techniques of particle balance and thereby compare particle transport in the DIII-D machine with carbon walls and the C-MOD tokamak with all-metal walls. C-Mod has pioneered a "static" particle balance technique where plasma discharges are carried out with a closed vessel and no pumping. Careful accounting of the pressure rise and removal of particles is used to determine the average particle balance on a shot by shot basis. This technique was used for the first time last year in L-mode discharges, and the analysis was presented in the 2nd quarter milestone report. DIII-D has pioneered a "dynamic" particle balance technique where the sinks and sources in the particle balance equation are calculated as a function of time from the plasma density, particle inputs, and particle exhaust. DIII-D has three helium cryopumps that can provide pumping on the order of 20,000-30,000 T L/s each. Auxiliary heating with either ECH (no particle input) or neutral beams (with a particle input) can be compared. There was some doubt by the staffs of both C-Mod and DIII-D that these two techniques yielded similar results, and so a major goal of the DIII-D experiment was to use both measurement techniques in an ELMing H-mode plasma heating with ECH. It is anticipated that a second experiment will be carried out on C-Mod later in the FY.

The experimental plan was developed through a series of videoconferences held on several dates, including April 16, April 28, May 5, and May 21. A detailed plan was developed which necessitated extensive interaction with the DIII-D operations group as the tokamak would be operated in a new way for the static particle balance measurements. The concept was to close off all the vessel cryopumps and neutral beams and create an ECH H-mode. This was felt to be a simpler configuration, as it did not involve the particle inputs (both energetic and room temperature) of the neutral beams. The cryopumps would be used to exhaust particles for several discharges, and then the cryopumps would be warmed up so that the pressure rise times the vessel volume could be used to calculate the particles that were not retained in the walls. This required the technical staff at DIII-D to develop new techniques to rapidly warm the helium cryopumps by introducing helium and then nitrogen gas into the pumps to regenerate them. Qualification of this technique required a design review, modification of the cryopump system, and tests before plasma operations.

Another facet of the experiment that emerged from the discussion between the two experimental groups was a bake of the tokamak at the end of the day (with no pumping) after plasma operations to estimate the total number of particles that were retained in the wall. DIII-D routinely bakes to ~350ºC for a weekend with pumping, but this plan envisioned a bake after plasma operations without pumping, and then cool down with experiments on the next day (with a clean wall). This also required the DIII-D operations group to develop a new baking and cool-down technique. In short, this experiment definitely stretched the operational flexibility of DIII-D, and required careful planning and execution by the operations group.

This experiment was originally scheduled for mid-April, but was postponed to mid-June so that both operational and personnel constraints could be addressed. Dennis Whyte traveled to DIII-D for the experiment, and stayed in San Diego for a week because the experiment was delayed because of an issue with the TF coil (which turned out to be minor and was easily fixed).

Over a two-day period in June, a sequence of very successful particle balance experiments was completed. As the C-Mod tokamak was starting operations, it was more effective to use a full-time video link between MIT and the DIII-D control room. This worked very well, and the MIT staff played a key role in the successful execution of the experiments. First, as shown in Fig. 14, the both static and dynamic particle balance measurements were completed in ECH H-mode plasmas. Before the four shot plasma sequence, the cryopumps were regenerated by heating followed by a cool-down, which took about 20 minutes. The neutral beam Torus Isolation Valves

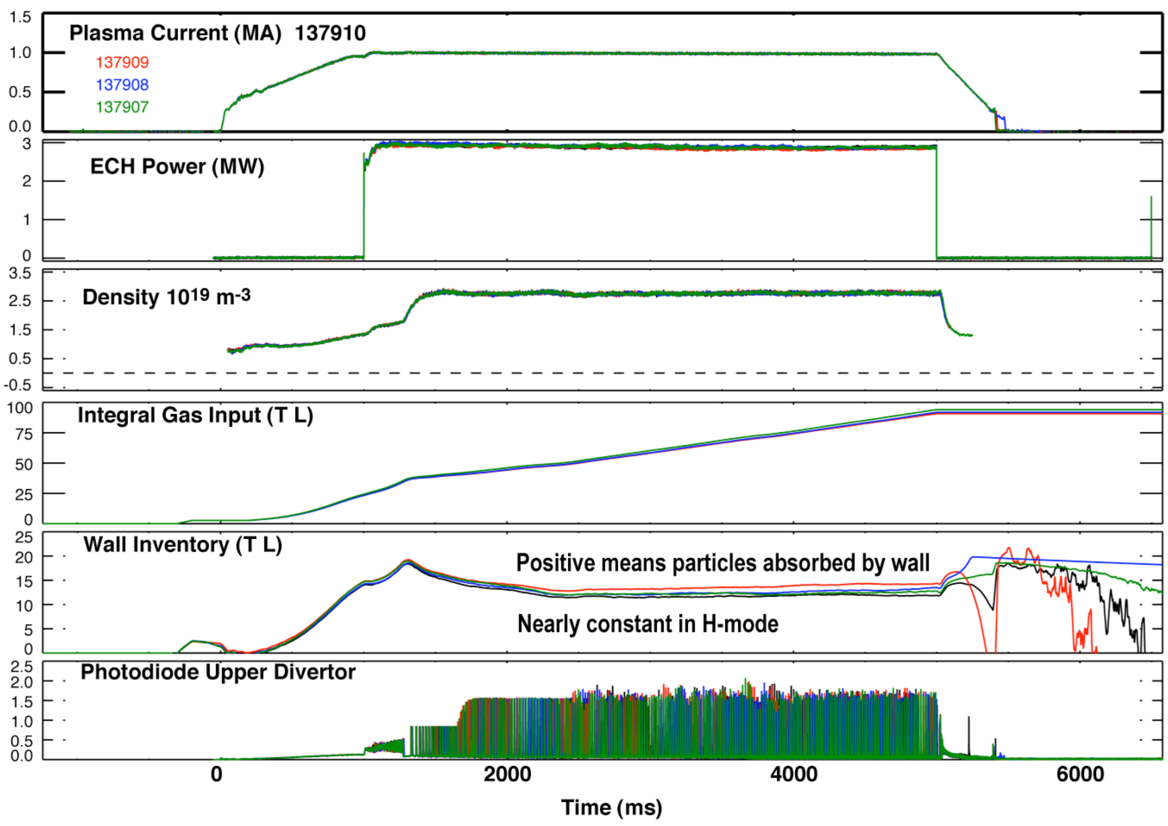

Fig. 14. The plasma current, electron density, gas fueling rate, and ECH power is shown for a series of four reproducible ECH H-mode DIII-D shots. The results from the dynamic particle balance for the wall inventory are shown; the sign convention on the wall inventory is that a positive value means that wall is removing particles from the plasma. Note that the wall inventory increases during the startup and L-mode period of the plasma, but during H-mode, the wall flux is very small.

were closed, as were the main turbomolecular pumps so the only pumping was due to the cryopumps. Four repeatable plasma shots were carried out, and the dynamic particle balance was calculated for these shots. After this sequence, the cryopumps were regenerated and the resulting pressure rise times the vessel volume was used to calculate the particle exhaust by the static technique. The total exhaust from the dynamic measurements were then calculated.

As the experiments were carried out at the end of the quarter, all analysis should be considered preliminary. However, these initial results are very encouraging in that the exhaust by the two techniques was very comparable: 75 T L for the static exhaust is compared with 75 TL for the same four ECH ELMing H-mode shots with the dynamic technique. Other shot series with different densities showed 139 TL by the static as compared with 135 TL with the dynamic, and 273 TL compared with 299 TL. One general trend was found in all of the H-mode discharges, which is shown in Fig. 14, is that the *wall retention* is quite large during L-mode and ohmic ramp-up with the large gas puff, but then drops to nearly zero in the H-mode phase of the shot

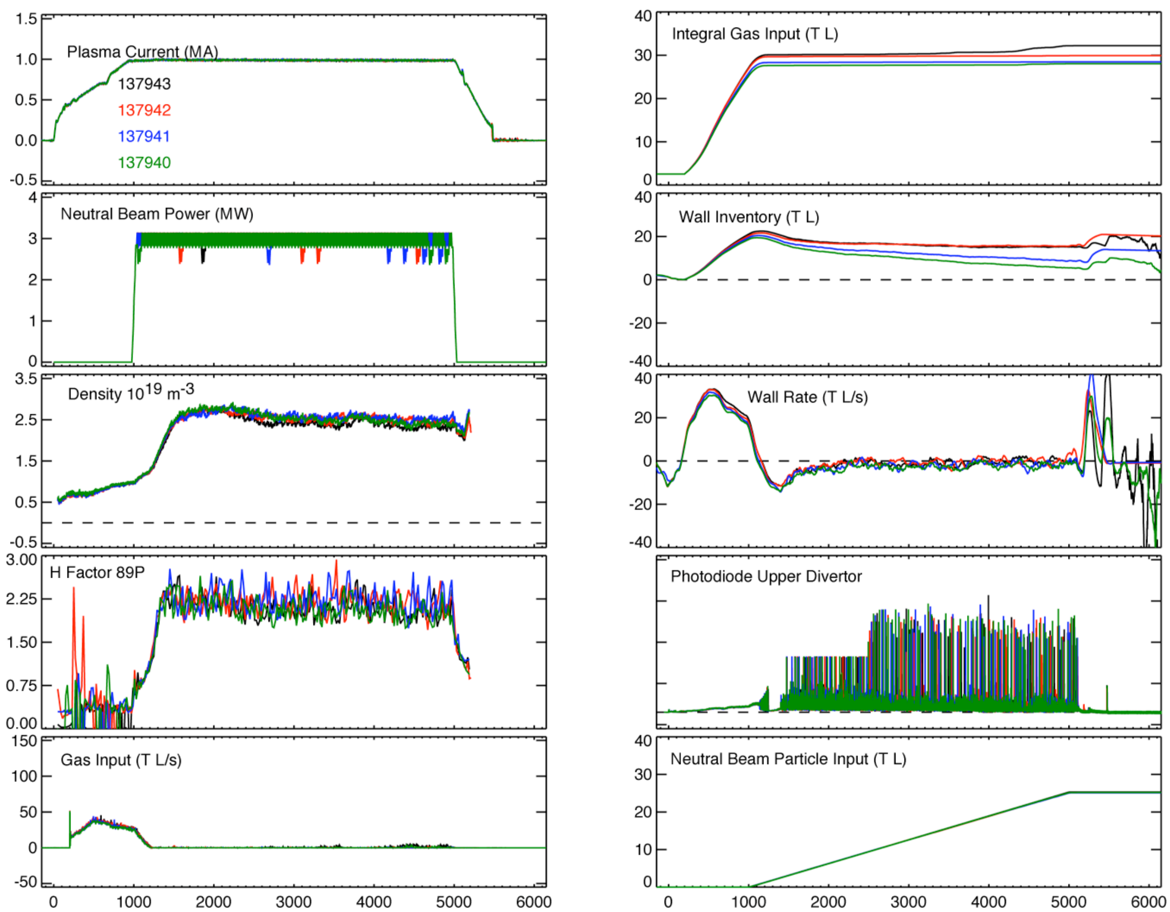

Fig. 15. The plasma current, neutral beam power, electron density, H-factor, gas input, integral of the gas input, wall inventory, wall rate, photodiode signals, and neutral beam particle input from the dynamic particle balance are shown for a series of four reproducible neutral beam heated H-mode DIII-D shots. The sign convention on the wall inventory is that positive means that the wall is a pump. Note that just like the ECH case, the wall flux is quite large in the L-mode period of the plasma, but during the H-mode, the wall flux is very close to zero.

(the inventory is nearly constant). While the mechanisms for this need careful analysis, it is encouraging that there is a condition where the *wall retention flux* is very small in H-mode, and if this can be maintained, the wall loading during the H-mode would be minimized. Just as important, the validity of the dynamic particle balance was demonstrated; this is much easier to carry out on a regular basis as it does not require special pumping setups on the tokamak.

After the C-Mod and DIII-D teams developed some confidence with the two techniques in a case with ECH H-mode, it was decided to try to extend the technique to neutral beam heated discharges. This has the complication that the neutral beams have their own cryopumps that can exhaust particles, and the beams also introduce particles in addition to heating. A similar technique to the ECH discharges was developed, except that the neutral beam valves were open for a short period during the plasma discharge, thereby minimizing the pumping. As shown in Fig. 15, after the five-hour bake (after the first day of the experiment) we were able to achieve reproducible ELMing H-modes with neutral beam injection; four shots in a sequence are shown.

Note that this has the same general behavior, that the *wall retention flux* (middle panel on the right) is high in the L-mode startup period of the discharge, but nearly zero during the H-mode part of the shot. This is also very encouraging, indicating that the *wall retention flux* is small during the H-mode period with neutral beam heating. In these cases, the particle exhaust by the static technique was 160 TL, compared with 159 for the sum of four shots of the dynamic analysis.

We compare the dynamic particle balance for similar discharges with ECH and neutral beam heated H-mode in Fig. 16. The duty cycle of the neutral beams was adjusted to yield nearly the same power as the ECH heated case. Note that there was not sufficient time to optimize the H-mode in these discharges and this will be the subject of future research. The *wall retention rate* in the L-mode part of the discharge is very similar and large, while the *wall retention flux* during the H-mode portion of the shot is nearly zero.

Between the ECH and neutral beam run days, a high temperature (~300°C) bake was carried out. The pressure was carefully monitored so that the total particle exhaust from the cryopumps and the machine could be determined. Initial estimates show that nearly 1000 TL was removed from the walls during the overnight bake. This quantity may reflect both particles retained during these experiments and particles retained during prior experiments since the last high temperature bake.

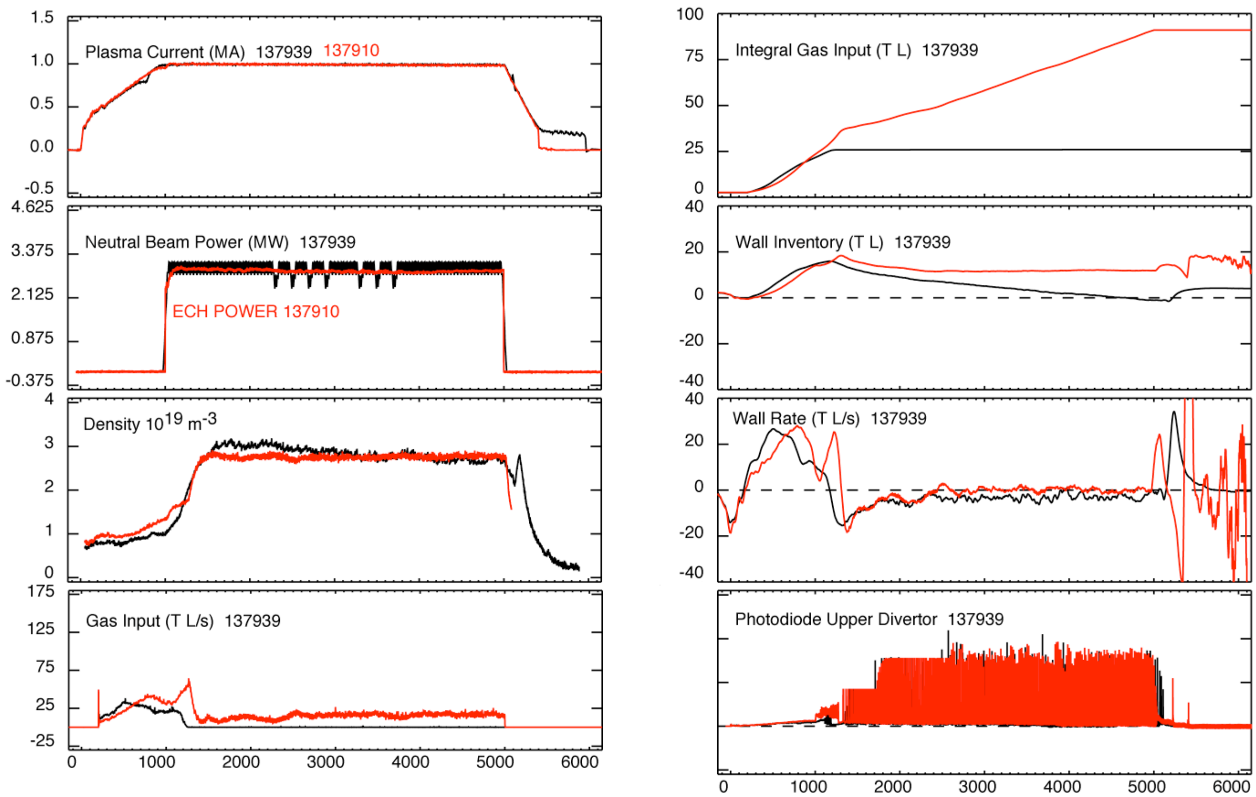

Fig. 16. The plasma current, neutral beam or ECH power, electron density, gas input, integral of the gas input; wall inventory and wall rate from the dynamic particle balance; along with the photodiode signal are shown for a series of four reproducible neutral beam heated H-mode DIII-D shots. The sign convention on the wall inventory is that positive means that the wall is a pump. Note that in both cases, the wall flux is quite large in the L-mode period of the plasma, but during the H-mode, the wall flux is very close to zero.