Plasma Response to Lithium-Coated Plasma-Facing Components in the National Spherical Torus Experiment

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Abstract

Experiments in the National Spherical Torus Experiment (NSTX) have shown beneficial effects on the performance of divertor plasmas as a result of applying lithium coatings on the graphite and carbon-fiber-composite plasma-facing components. These coatings have mostly been applied by a pair of lithium evaporators mounted at the top of the vacuum vessel which inject collimated streams of lithium vapor towards the lower divertor. In NBI-heated, deuterium H-mode plasmas run immediately after the application of lithium, performance modifications included decreases in the plasma density, particularly in the edge, and inductive flux consumption, and increases in the electron temperature and energy confinement time. Reductions in the number and amplitude of ELMs were observed, including complete ELM suppression for periods up to 1.2 s, apparently as a result of altering the stability of the edge. However, in the plasmas where ELMs were suppressed, there was a significant secular increase in the effective ion charge Z_{eff} and the radiated power as a result of increases in the carbon and medium-Z metallic impurities, although not of lithium itself which remained at a very low level in the plasma core, <0.1%. The impurity buildup could be inhibited by repetitively triggering ELMs with the application of brief pulses of an n = 3 radial field perturbation. The reduction in the edge density by lithium also inhibited parasitic losses through the scrape-off layer of ICRF power coupled to the plasma, enabling the waves to heat electrons in the core of H-mode plasmas produced by NBI for the first time. Lithium has also been introduced by injecting a stream of chemically stabilized, fine lithium powder directly into the scrape-off layer of NBI-heated plasmas. The lithium was ionized in the SOL and appeared to flow along the magnetic field to the divertor plates. This method of coating produced similar effects to the evaporated lithium but at much lower total amounts.

Keywords and subject classification

Tokamaks and spherical tokamaks, Plasma-wall interactions
PACS: 52.55.Fa, 52.40.Hf, 52.50.Gj, 52.50.Qt

1. Introduction

The National Spherical Torus Experiment (NSTX) produces plasmas with a toroidal aspect ratio as low as 1.3, typically with major radius 0.85 m and minor radius up to 0.65 m, in a single-null or symmetric double-null divertor configuration with an elongation of the plasma cross-section in the range 1.6 – 2.8. The tokamak operates at modest toroidal magnetic fields, typically 0.3 – 0.55 T at the nominal major radius, but with substantial toroidal plasma currents of 0.6 – 1.5 MA. The plasmas are heated by up to 7 MW of deuterium neutral beam injection (NBI) with beam acceleration voltages up to 95 keV, and up to 6 MW high-harmonic fast-wave (HHFW) power at a frequency of 30 MHz and with a controllable parallel wavelength [1,2]. As a result of its low aspect ratio and cross-section shaping, and the presence of closely fitting stabilizing plates above and below the midplane on the outboard side, plasmas in NSTX can achieve very high normalized beta, β_N ($= \beta_T \cdot a \cdot B_T / I_p$ where β_T is the toroidal beta, a the plasma minor radius, B_T the toroidal magnetic field and I_p the plasma current), up to 7 and routinely exceeding 5 (in units %·m·T/MA) [2,3].

The plasma facing surfaces of NSTX mostly consist of graphite tiles, with some carbon-fiber composite material tiles on the narrow center column and in the high-heat-flux regions of the divertor. At the midplane on the large major radius side of the plasma, the 12-element coupler for the HHFW power extends ±0.5 m vertically from the midplane and occupies approximately one quarter of the toroidal circumference. Its fixed Faraday shield, consisting of inconel bars coated with titanium carbide, is shaped to match the typical plasma cross-section with a gap of 0.06 – 0.12 m to the last closed flux surface. The vertical edges of the antenna structure are protected from plasma impingement with boron nitride shields. Around the remainder of the toroidal outer circumference in a band ±0.5 m about the midplane, the plasma faces the stainless-steel vacuum vessel across a radial gap typically 0.15 – 0.21 m.

Since 2005, NSTX has been investigating the potential of lithium coatings deposited on the carbon plasma-facing components (PFCs) to reduce the recycling of deuterium from them and to suppress oxygen impurities. Solid lithium has a strong chemical affinity for deuterium in atomic or ionic, but not molecular, forms and for oxygen (in water). Other tokamaks, including TFTR [4], T-11 [5] and FT-U [6] and the stellarator TJ-II [7] have observed significant operational benefits, in terms of improved confinement, density control and lower impurity levels, after introducing lithium onto their PFCs. The CDX-U device achieved very significant enhancements in the confinement time of ohmically heated plasmas when its plasma was limited on a liquid lithium surface contained in a full toroidal tray at the bottom of its vacuum vessel [8,9].

2. Lithium Coating Techniques and Their Effects in NSTX

In the first NSTX experiments with lithium, solid lithium pellets with mass 2 – 5 mg were injected into sequences of ohmically-heated helium discharges [10]. The lithium became ionized in the scrape-off layer and was transported along field lines and deposited on the divertor targets, a technique similar to that developed in TFTR in the early 1990s [11]. In 2006, an initial experiment was conducted with a lithium evaporator, dubbed “LITER”, mounted at the top of the vacuum vessel in a gap between tiles of the upper, outer divertor plate. This evaporator directed an expanding plume of lithium vapor downwards from an electrically heated, re-entrant stainless-steel oven to deposit a film of lithium on the room-temperature lower divertor surface and the surrounding regions of the PFCs. In 2007, the evaporator was enlarged and re-aimed to provide a higher deposition rate on the divertor region [12], although with only a single evaporator, roughly one third of the divertor was shadowed from the lithium vapor stream by the center column and remained uncoated. In

2008, NSTX was equipped with a pair of LITERs, each mounted above the upper divertor and injecting through gaps in the tiles separated by about 150° toroidally [13,14].

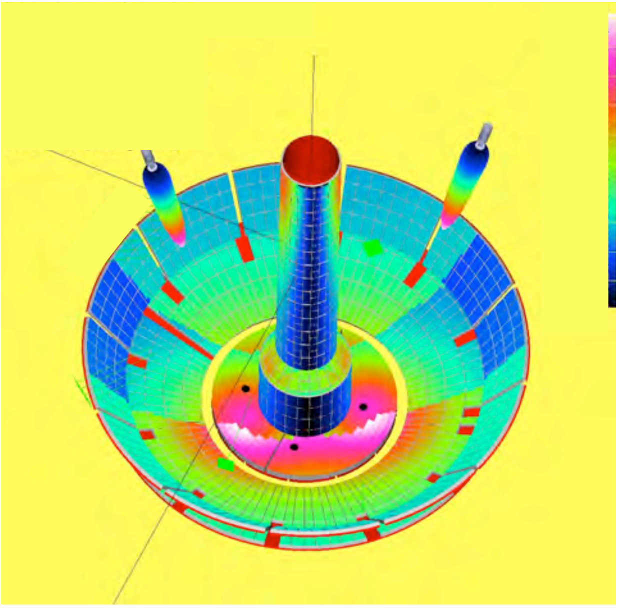

Figure 1. Calculated pattern of lithium deposition on the PFCs in the lower half of NSTX by two evaporators (LITERs). The color scale on the upper right is linear in deposited film thickness ranging from zero (black) to 30 monolayers (~8 nm) per minute (white) on the PFCs for a total evaporation rate of 20 mg/min.

The lithium evaporation rate from each LITER is determined by its temperature. The deposition pattern from a LITER was measured in a laboratory vacuum chamber by measuring the lithium deposited on a quartz crystal micro-balance which was scanned across the vapor plume. The angular distribution from the oven was roughly Gaussian, essentially independent of the evaporation rate, with a 1/e angular width of the lithium vapor stream of about 20°. Figure 1 shows the calculated deposition pattern of lithium on the PFCs in the lower part of the vacuum vessel: with the two LITERs, the lithium coats the whole of the lower divertor in NSTX. It should be noted that although the front face of the HHFW coupler receives direct lithium coating, some PFCs receive little lithium coating and there are areas of the vacuum vessel wall facing the plasma that are shadowed from the evaporators.

During experiments with lithium coating in NSTX, evaporation rates up to about 80 mg/min but typically 20 mg/min from each LITER have been employed for periods 5 – 15 min, depositing from 20 mg to over 1 g and typically around 200 mg of lithium between discharges. The evaporators have considerable thermal inertia and reducing their evaporation rate requires several minutes, so they were equipped with shutters to interrupt the lithium vapor streams during both tokamak discharges and periods of glow-discharge cleaning applied between tokamak discharges. It had been found in 2007 that evaporation into the glow discharge produced excessive deposition of lithium near the evaporator exit ducts due to immediate ionization of the lithium. Since 2008, in periods when the two LITERs have been used, boronization [15], which used to be applied routinely after about 2 weeks of operation, has been discontinued. Also, when the LITERs are in use, helium glow-discharge cleaning (HeGDC) between tokamak discharges has been eliminated, allowing a larger number of discharges to be run each day. HeGDC is still usually applied for a period of 30 min before starting a day of operation. The LITER reservoirs have been refilled several times during operational periods by withdrawing the assemblies into air locks. During the 10 weeks of operation with the evaporators in 2008, a total of 180 g of lithium was evaporated of which it is estimated that about 110 g was deposited on the PFCs. The majority of tokamak discharges in NSTX this year (2009) have been run with lithium coating applied beforehand.

Lithium coating of the divertor and surrounding PFCs by the single evaporator in 2007 produced several changes in the behavior of standard H-mode, NBI-heated plasmas in a lower single-null divertor configuration, including the expected reduction in oxygen impurities and deuterium recycling. The density in the early phase of the discharge was reduced with lithium but, since this increased the likelihood of developing locked MHD modes during the current ramp, it was necessary to increase the gas fueling rate to maintain about the same density as in discharges without lithium. There was a reduction in the number edge-localized modes (ELMs) by progressive increases in the number and duration of ELM-free periods and a significant improvement in the energy confinement time with lithium [12,16]. With full lithium coverage of the lower divertor, the beneficial effects of lithium were more pronounced [13,14], but so too were some deleterious effects. Figure 2 shows a comparison of the time evolution of several representative plasma parameters for two discharges run under the same external conditions (current, field, heating power, plasma geometry) immediately before the first introduction of lithium and after 260 mg of lithium had been evaporated since the preceding discharge. The lithium coating produces an improvement in energy confinement, evident in the increases in both the total plasma stored energy and that in the electrons. The level of deuterium Balmer-alpha emission from a chord viewing the lower divertor is significantly reduced, despite increasing the gas fueling rate to maintain the plasma density. There is an almost complete suppression of ELMs in the discharge with lithium, although the plasma remains in the H-mode [16]. The H-mode power threshold was reduced when lithium was applied. Although not shown here, the line emission from low-ionization states of oxygen and carbon measured by instruments viewing the lower divertor were also lower following the introduction of lithium. However, with lithium, there is also a significant increase in the total power radiated from the plasma and in the average Z_{eff} deduced from a central-chord measurement of visible bremsstrahlung. Interestingly, the lithium coating appears to last for only one or two discharges, after which its effects are noticeably reduced. However, lithium applied at the end of one day after the last discharge has been run, does produce benefits for the first discharge on the subsequent day, suggesting that it is plasma bombardment and not interaction of the lithium surface by residual gas in the vacuum that causes the passivation.

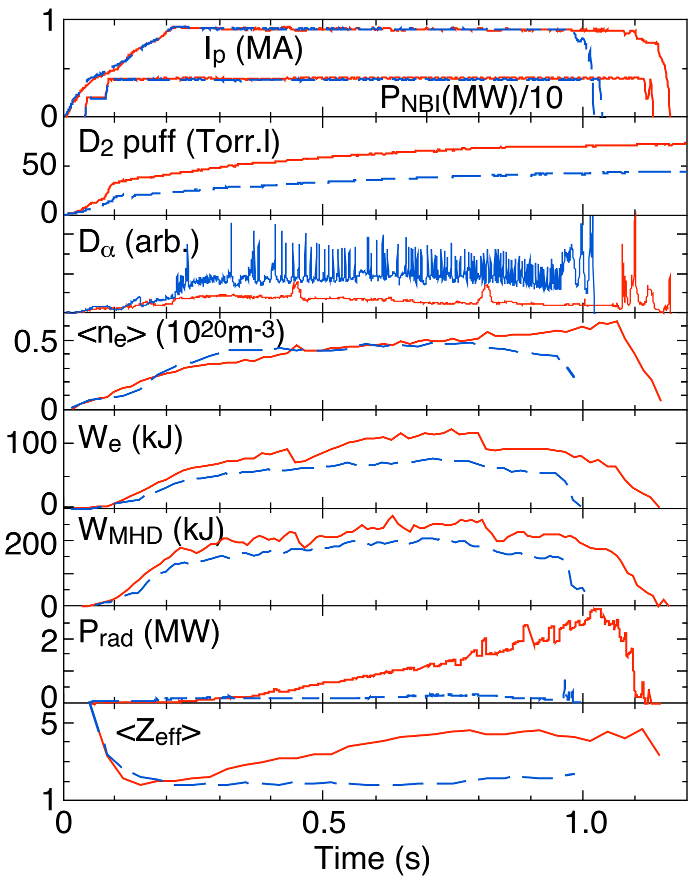

Figure 2. Time evolution of representative plasma parameters for similar discharges without (blue dashed) and with (red solid) 260mg of lithium applied by the two lithium evaporators.

Figure 3 shows a comparison of the electron density and temperature and ion temperature profiles close to the time of peak plasma stored energy for the two discharges compared in Figure 2. The broadening of the temperature profiles after lithium coating is striking. The

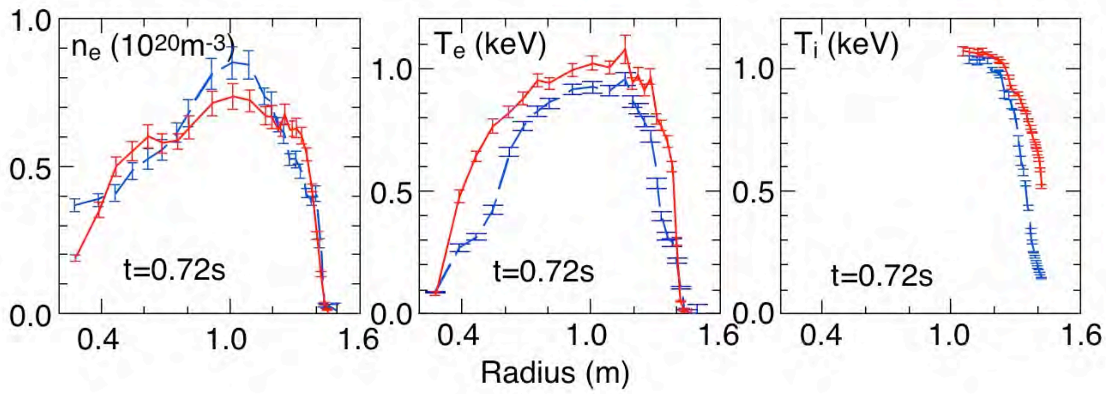

Figure 3. Profiles of the electron density and electron and ion temperature measured close to the time of peak stored energy for the discharges in Fig. 2.

reduction in the edge electron density apparent in the Thomson scattering data was confirmed by a frequency-swept microwave reflectometer probing the outer edge. Figure 4 compares the total plasma stored energy, determined by equilibrium analysis with EFIT, and the total electron stored energy obtained by volume integration of the Thomson scattering data, for an ensemble of shots in similar conditions without and with >100 mg of lithium applied. There is a 20% improvement in overall confinement and a 44% improvement in electron confinement with lithium. The broadening in the electron temperature profile gives rise to a lower loop voltage and a broader current density profile, evidenced by a lower internal inductance, l_i, during the current flattop, as seen in Figure 5; both these attributes are beneficial for extending the pulse length of NSTX discharges. As seen in Figure 6, the improvement in global electron confinement with increasing lithium deposition is reflected in a reduction in the electron thermal diffusivity calculated by TRANSP using the measured profiles of the

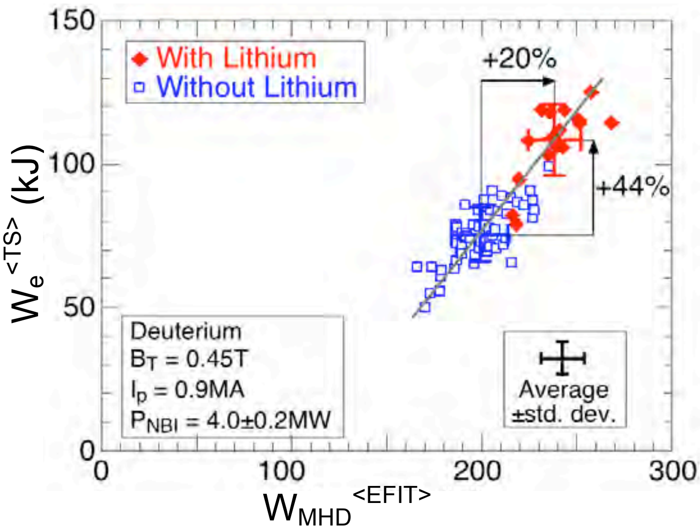

Figure 4. Total stored energy from EFIT analysis and electron stored energy from volume integration of Thomson scattering measurements of n_e, T_e for similar discharges with and without lithium coating of the lower divertor

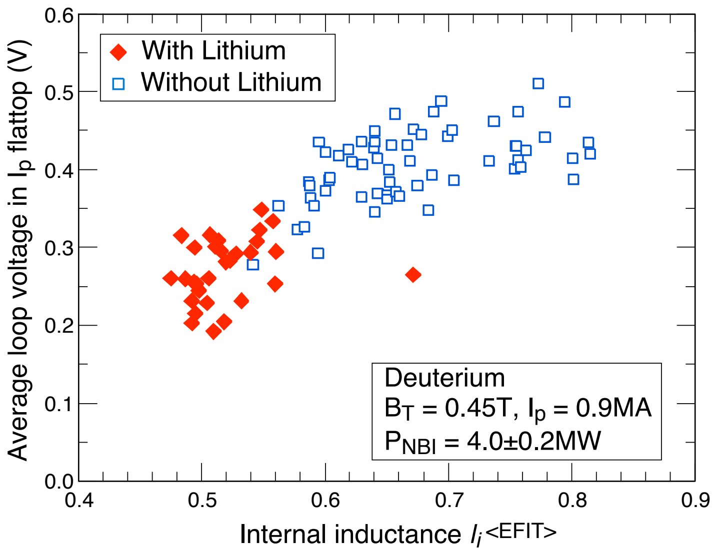

Figure 5. Changes of the loop voltage averaged through the plasma current flat-top and the internal inductance l_i from EFIT analysis for standard NBI-heated H-mode plasmas with and without lithium coating applied.

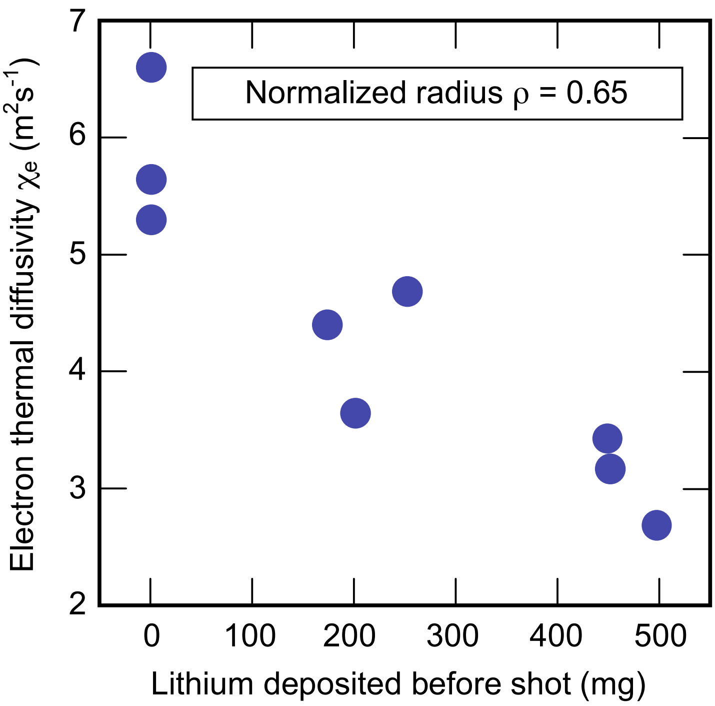

Figure 6. Variation of the electron thermal diffusivity calculated by TRANSP in the outer region of the plasma for a group of similar discharges as the amount of lithium coating was increased.

temperatures and density and assuming classical thermalization of the neutral beam heating.

In addition to using the LITERs, another method for introducing lithium into NSTX was explored at the end of the 2008 experiments. Fine lithium powder (~40μm particle diameter) was introduced during a discharge into the plasma scrape-off layer on the outboard side. The powder, which is a commercial product, is stabilized against rapid oxidation in air by a thin coating of lithium carbonate (less than 0.1% Li_2CO_3 by mass relative to lithium). The powder was loaded in a hopper inside a small vacuum chamber connected to a port at the top of the main vacuum vessel. The bottom of the hopper was a disc of piezo-electric material with a small hole in the center. By resonantly oscillating the disc at ~2.2 kHz, a stream of lithium powder was produced which fell through a guide tube, emerging through a gap in the PFCs. Powder flow rates of 5 – 40 mg/s were well tolerated throughout standard NSTX plasmas with NBI heating. This is in contrast to the lithium pellets originally used in NSTX which tended to terminate the ohmically heated plasmas when they penetrated through the scrape off layer to the plasma core. During the powder injection, toroidally localized plumes of LiI and LiII radiation were seen in fast camera images extending along field lines from the powder impingement region on the scrape-off to the upper and lower divertors. Most interestingly, small amounts of lithium powder, ~10 mg, introduced into a discharge beginning at plasma initiation, produced effects on the plasma, in terms of total and electron confinement and the suppression of ELMs, comparable to the effects of the several hundred milligrams of lithium typically deposited by the LITERs but distributed widely on the lower PFCs. This suggests that it is mainly lithium on the divertor plates in contact with the immediate scrape-off layer that is effective, although it should be noted that the experiments with lithium powder injection took place after more than 100 g of lithium had already been deposited on the PFCs by evaporation. With the powder injection technique, most of the confinement benefits of lithium were realized with less increase in both radiated power and Z_{eff} than occurred with the LITERs.

3. Discussion

Possible mechanisms for the suppression of ELMs in H-mode plasmas by the lithium coating are being evaluated. Figure 7 shows that in discharges with ELMs suppressed by lithium, the peak in the pressure gradient in the edge was both reduced in magnitude and shifted to smaller normalized poloidal flux Ψ_N, compared to the discharges without lithium during the last 20% of their natural ELM period. This analysis took data from ensembles of shots with and without lithium to assemble detailed profiles of the electron and ion temperatures and densities mapped to the normalized poloidal flux Ψ_N in the plasma edge. Subsequent analysis of the MHD stability of these profiles has revealed that the edge plasma is close to the stability boundary for peeling-ballooning modes for the discharges without lithium which exhibit ELMs. However, in the discharges with lithium, the changes in both the edge profile and the resulting equilibrium move the stability boundary away from the experimental values [17]. Another possibility is that reducing recycling with lithium creates a high current density at the separatrix, stabilizing MHD instabilities at rational surfaces inside the separatrix [18].

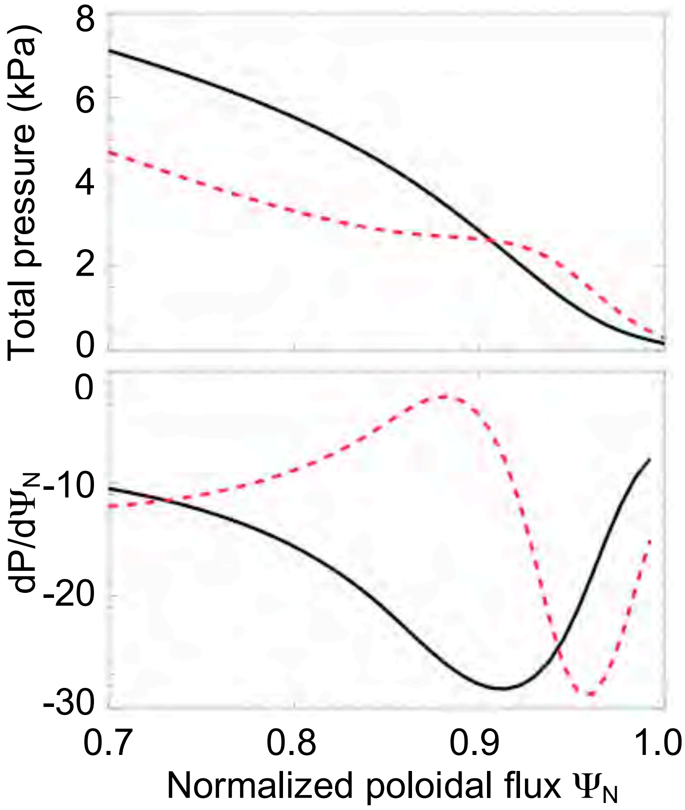

Figure 7. Total plasma pressure and pressure gradient as functions of normalized poloidal flux Ψ_N for two shots without (red dashed) and with (black solid) lithium. The mapped profiles are from equilibrium analysis constrained by magnetic data and an ensemble of measured temperature and density profiles. The data for the shot without lithium are for times within the last 20% of the ELM period.

The source of the increased radiation in the discharges with lithium appears to be metallic impurities, notably iron, while the increase in the Z_{eff} is more attributable to an increase in the carbon impurity content [19]. Figure 8 shows a comparison of the radial profiles of the concentrations of both carbon and lithium ions in the plasma measured by charge-exchange recombination spectroscopy in a matched set of discharges before the introduction of lithium and with lithium coating of ~180 mg applied beforehand. The concentration of lithium is surprisingly low but the carbon concentration, which usually exhibits a hollow profile in H-mode discharges in NSTX, is actually higher with lithium coating by about a factor 2. Carbon can contribute to a rise in average Z_{eff} up to 4 in the discharges with lithium. Although the concentration of carbon is high, it becomes fully stripped and contributes little to the radiated power except near the plasma edge. The low concentration of lithium compared to carbon suggest that any lithium that is sputtered or evaporated from the surfaces of the PFCs becomes rapidly ionized in the scrape-off layer and is transported back out of the

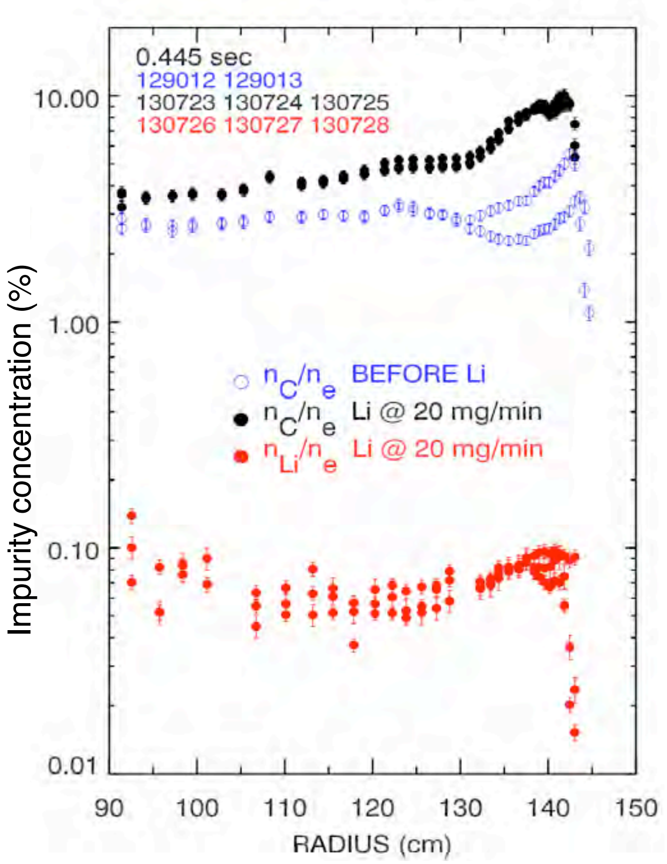

Figure 8. Radial profiles of the lithium (red) and carbon (black, blue) for sets of similar NBI-heated, H-mode discharges. The discharges in blue were made before any lithium was applied. The discharges in black were made with ~180 mg of lithium applied since the preceding discharge and at the end of the 2008 experiments when over 100g of lithium had been applied in total.

plasma so that very little crosses separatrix and enters the main discharge. The increased carbon level is thought to be caused by the suppression of the ELMs, although the possibility of an increase in the source of carbon due to changes in the plasma edge and scrape-off layer cannot yet be ruled out.

The source of the metallic impurities causing the rise in radiated power, particularly from the plasma center, has not yet been unambiguously determined. As previously described, there are metal surfaces facing the plasma on the outboard midplane which could be exposed to bombardment by energetic beam-injected ions on unconfined orbits. Because of the low magnetic field on the large major radius side in an ST plasma, mirror-trapped energetic ions can make large excursions outside the plasma boundary. The flux of energetic ions to the plasma-facing surfaces would be expected to increase with the lithium coating applied because measurements with a spectroscopic array and neutral pressure gauges at the vessel wall show a decrease in the neutral deuterium density in the outboard midplane region. Increasing both the gap between the plasma and the outer wall by 0.1m from its standard value, and changing the plasma current from 0.7 to 1.2 MA to decrease the flux of energetic ions, did reduce the ratio of the radiated power to the total electron content of the plasma, suggesting that the metallic content of the plasma was reduced by a factor up to a factor 2. However, this reduction was less pronounced than would have been expected if the flux of energetic ions to the outer metal areas was the sole source of the metal influx.

Concerns over the significant increase in radiated power and Z_{eff} which accompanied lithium coating and its suppression of normal ELMs, prompted an investigation whether applying non-axisymmetric error fields with external coils could be used to trigger ELMs to flush impurities [20]. NSTX is equipped with three pairs of nearly rectangular error-field correction (EFC) coils mounted on the mid-plane outside the vacuum vessel to produce spatially and time varying radial magnetic field perturbations. The diametrically opposite coil pairs are powered by three power amplifiers, which can drive currents up to 3 kA at frequencies up to several kHz to produce field perturbations with dominant toroidal harmonics either n = 1 or n = 3. As shown in Figure 9, this proved successful. In two similar shots, both with lithium applied beforehand, ELMs are suppressed in a shot with no EFC current applied, but reappear when the EFC coils are energized with a train of 11 ms pulses at 40 Hz to produce an n = 3 field perturbation. The ELMs are synchronized to the applied field, although it is evident that the triggering is not infallible. Both the radiated power and the Z_{eff} were lower in the discharge with the triggered ELMs, although neither returned to the lowest levels seen without lithium.

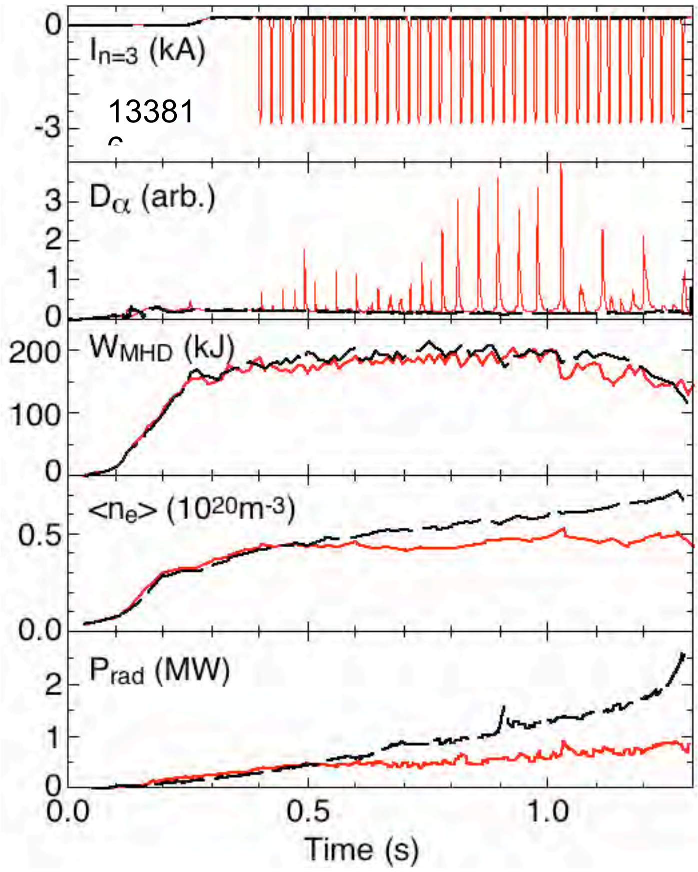

Figure 9. Comparison of similar discharges without (black) and with (red) the application of a train of pulses in the error-field correction coils to create an n = 3 magnetic field perturbation. The pulses trigger repeated ELMs which have been suppressed by the lithium coating. The ELMs have little effect on the stored energy, but produce a reduction in the volume average density and a significant reduction in the radiated power.

Lithium coating has also contributed to improving the heating of electrons in the core plasma by HHFW RF power in NSTX [21]. After extensive lithium evaporation, 3.1 MW HHFW heating launched with a parallel wavenumber $k_{||} = -14$ m^{-1} produced central electron temperatures up to 5 keV in deuterium plasmas at a toroidal magnetic field of $B_T = 0.55$ T. When the RF phase between successive elements of the antenna was varied to change the wavelength of the launched waves, the heating efficiency was similar for $k_{||} = -7$ m^{-1} and significant heating was even observed for the fastest waves with $k_{||} = -3.5$ m^{-1} which was not the case before lithium was applied. These results are consistent with the hypothesis that the location of the surface in front of the antenna where the electron density reaches the threshold for perpendicular propagation of the waves, $n_{crit} \propto B \times k_{||}^2/\omega$ (where B is the magnetic field, $k_{||}$ the parallel wavenumber and ω the frequency) is critical to achieving good coupling of the HHFW power to the core of the plasma [22]. Localized heating of the divertor surface in regions connected to the RF antenna along field lines during HHFW heating has been observed when the density ~2 cm in front of the antenna exceeded the threshold density. With lithium coating, the surface at which the density reaches the threshold is moved far enough away from the antenna that parasitic losses of the RF power to surface modes are avoided.

4. Summary and Plans

The plasma facing components in NSTX are now routinely coated with a layer of solid lithium applied by evaporation. Coating of the divertor surface has also been accomplished by introducing a stream of fine lithium powder into the scrape-off layer during normal plasma operation. The routine use of lithium has supplanted the periodic boronizations previously employed to reduce oxygen impurities. The lithium has provided several benefits to operation, including improved discharge reproducibility, lower inductive flux consumption which is important for NSTX, and better energy confinement, notably of the electrons. The reduced recycling of deuterium by the lithium coated PFCs has necessitated increasing the gas fueling rates to avoid the occurrence of locked modes, particularly in the current ramp-up phase of the discharge. Although the lithium content of the plasmas remains very low, the suppression of ELMs in H-mode discharges that accompanies lithium coating has caused an increase in both the carbon and higher-Z metal content of the plasma, affecting the Z_{eff} and radiated power. The application of pulsed non-axisymmetric magnetic fields with a dominant toroidal harmonic $n = 3$ by external coils has been used to trigger repeated ELMs in discharges and to regain control over the radiated power and density rise that otherwise detract from the significant confinement benefits that lithium coating produces.

In the next year, four metal plates forming a toroidal annulus with their plasma-facing surfaces coated with a porous molybdenum layer will be installed in the outer lower divertor plate of NSTX where they will intercept field lines from the outer scrape off layer of the plasma [23]. Lithium will be evaporated onto the plates which will then be heated above its melting point. The thin liquid film is expected to provide a much larger capacity for absorbing and retaining hydrogen isotopes than a solid film because of the much higher mobility of hydrogen in the liquid. Modeling shows that this "liquid-lithium divertor" should provide significant edge pumping in NSTX, comparable to what could be provided by a cryo-pump but with less restriction on allowable plasma position and shapes.

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