Abstract—A number of major hardware items have been installed and/or modified on the Alcator C-Mod tokamak for the most recent run campaign, and additional near-term upgrades will be installed shortly. Topics that are covered in this paper include the changeover to all-metal plasma-facing components, the titanium lower hybrid waveguide grill, the gas jet disruption mitigation system, and the results these made to plasma operation. The design of a new cryopump, which will be installed soon, is also presented.

I. ALL-METAL TILES AND BORONIZATION

Prior to the 2005 run campaign, all the boron nitride (BN) protection tiles on the ICRF antennas and other in-vessel structures were replaced with molybdenum tiles (Figs. 1,2), leaving the machine with entirely all-metal plasma-facing components, as it was prior to 2000. There were a number of reasons for removing the BN protection tiles. Plasma energy confinement during the years with BN had not been as good as in the early years of C-Mod operation with molybdenum antenna tiles. In fact, the best plasma confinement had been obtained in 1996 with all moly tiles and thin coatings of boron deposited by a particular boronization process. It is conjectured that introduction of the BN tiles, together with many boronizations, may have resulted in too large a buildup of boron layers. In addition the hydrogen fraction (in primarily deuterium plasmas) was also higher, and this can negatively impact the H-minority ICRF absorption, which is the principal auxiliary heating scheme in C-Mod. The increased hydrogen fraction could have come from the BN material and/or thick boron layers, which are somewhat porous. The BN tiles were also prone to fracturing, caused by mechanical shaking of the machine during disruptions. Finally, all-metal walls are relevant to the ITER/reactor issue of tritium retention. In addition to changing the antenna tiles, all other in-vessel surfaces, including areas hidden from plasma line-of-sight, were cleaned of accumulated boron coatings, either by wipe down in-situ or by removal and ultrasonic cleaning where possible.

A significant portion of the 2005 run campaign was devoted to studying the performance of the ICRF antenna system with molybdenum protection tiles, and extensive experimentation with boronization methods to see if the best energy confinement performance from C-Mod’s early years could be recovered. In fact, during initial operation prior to first boronization, all three ICRF antennas were conditioned in less than 3 weeks of operation and all performed similarly. Over 4 MW of total net power became reliably available. As shown in Fig. 3,
no deleterious moly impurity injections have been observed during high power heating. Furthermore, the hydrogen-to-deuterium ratio, as well as $Z_{eff}$ change insignificantly with RF power.

A. Boronization

Once the ICRF antennas were conditioned and operating well at full power, several different ‘boronization’ methods were tried. The usual technique is to run ECDC (electron cyclotron discharge cleaning) plasmas in a mixture of deuterated diborane (B$_2$D$_6$) and helium, either lasting for hours overnight between runs, or for minutes between discharges. Overnight boronization typically deposits a quantity of boron equivalent to a uniform layer $O(100)$ nm thick. Other boronization methods that were tested included glow discharge with the B$_2$D$_6$/He mixture, and sprinkling boron grains into the plasma. The ECDC method turned out to be clearly superior.

Bolometric measurements show that ECDC boronization reduces the total radiated power loss from the plasma compared to the unboronized state, as shown in Fig. 4. Spectroscopic observations indicate that this is primarily due to a reduction of molybdenum impurity radiation. Since the radiated power is a significant loss channel, its reduction due to boronization results in substantially higher thermal energy in the plasma ($W_{\text{tot}}$), as seen in the 2nd trace in the figure, as well as an increased DD fusion neutron rate, even though the RF power is only marginally higher. The first boronization of the campaign generally increased the plasma energy by about a factor of 2, as seen in Fig. 5. After the second boronization of the campaign, several of the highest performance shots to date on C-Mod were achieved, including one with a new C-Mod record thermal energy, 250 kJ, and a volume-averaged pressure of $\langle p \rangle = 1.80$ atm, which is a world record for tokamaks. We therefore conclude that with proper boronization, an all metal-wall machine can achieve excellent wall conditions and plasma energy confinement.

II. TITANIUM LOWER HYBRID WAVEGUIDE GRILL

The lower hybrid (LH) system on Alcator C-Mod[1] will be used to drive plasma current non-inductively. The output end of the system consists of 96 individual waveguides, permitting flexible phasing of the LHRF waves for better control of driven current position and profile. The external waveguide assembly is coupled through a vacuum interface and into the plasma with a 96-channel launcher shown in Fig. 6. The launcher grill, which was installed for the most recent campaign, was made of titanium. This choice of material was based on its thermal expansion characteristics, which reasonably matches that of the ceramic windows that are brazed into each waveguide and serve as the vacuum interface. As can be seen in the figure, during the initial low-power tests of the LH system, glowing hot particles were ejected from a number of the individual waveguide mouths. Plasma spectroscopy identified these as titanium. Initially this was thought to indicate normal RF conditioning of the new grill.

As part of the calibration procedure for the Thomson scattering (TS) diagnostic on Alcator C-Mod, the vacuum vessel is filled to nearly an atmosphere of room-temperature deuterium gas (for Raman scattering). This was attempted three different times during the first part of the run campaign, but failed each time because of dust in the machine. The dust got progressively worse with each TS calibration attempt. After the 3rd attempt, the dust contamination was so bad that it precluded normal full-length plasma operation. The short plasmas that were obtained were plagued by titanium injections, as seen in Fig. 7. In addition, these plasmas also had uncontrolled density increases. This is understandable, given that Ti is a well-known hydrogen getter. The Ti dust apparently soaked up D$_2$ after each shot and then released it when heated up during the next shot. At some point the coupler vacuum seals started leaking and the LH grill had to be removed from the machine (which was accomplished in just a few days, since it did not require a manned access into the vacuum chamber). The titanium coupler showed massive deterioration.

By weighing the grill it was determined that more than a
kilogram of titanium had been eaten away and turned to dust by chemical reaction with deuterium gas. The erosion was bad enough that many of the individual waveguides had openings to their adjacent neighbors. Much of the dust settled in piles in the waveguide mouths along the lower rows, as clearly seen in Fig. 8. But about 300 g of Ti is unaccounted for on the grill structure and presumably settled all over the interior of the tokamak. If each Ti atom can getter one D₂ molecule (as TiD₂), this represents a sink of $1.7 \times 10^4$ Pa-m$^3$ of D₂ molecules, which is equivalent to the total inventory of gas pulsed into $\sim 5500$ C-Mod discharges. In addition to the problems with impurity injections and uncontrolled density fueling, the large quantity of titanium dust rendered the nearest ICRF antenna unusable because it wouldn’t hold high voltage, and it also caused gate valves on the nearby gas jet system (described in the next section) to leak.

Nevertheless, once the LH grill was removed from the tokamak, nearly normal plasma operation was recovered after roughly a week or two of additional running, although occasional Ti injections still continued to occur. The Ti experience implies that unless there is an enormous source of dust, it doesn’t affect plasma performance very much. The nearest ICRF antenna, however, remained inoperable. We have recently stopped operation to allow a short manned access in order to vacuum out the dust and disassemble and clean out the affected ICRF antenna.

Despite the obvious damage from exposing the titanium...
coupler to deuterium gas in the tokamak, we have been unable to reproduce the problem in the lab. Ti test samples from the same stock as the LH grill, prepared in the same way as the grill material, do not produce dust when exposed to an atmosphere of D₂ at the relevant temperature. The problem apparently involves something more subtle, presumably having to do with in-vessel surface conditioning with ECDC, or glow discharge cleaning, or perhaps boronization. Further investigation is ongoing. In any case, we STRONGLY recommend that ITER not use titanium inside of its first wall.

III. Gas Jet Disruption Mitigation System

Tokamak plasmas are subject to disruptions, which are rapid, undesirable terminations of the plasma discharge. Disruptions can suddenly release large amounts of energy and generate large electromagnetic forces that can damage the tokamak. Disruptions are a major concern for high-field, high-energy density devices such as Alcator C-Mod, ITER, and future tokamak reactors. Reliable mitigation of these effects using benign techniques would be a key improvement in tokamak operation.

High-velocity noble gas jet injection on the DIII-D tokamak[2] has shown promise in reducing the deleterious effects of disruptions. However, a number of questions remain concerning the effectiveness of gas jet disruption mitigation on high-pressure, high-energy density plasmas characteristic of Alcator C-Mod and ITER. In particular, the ability of the gas jet to penetrate into C-Mod plasmas and to radiate energy quickly enough needs to be tested. These experiments are now being carried out on the Alcator C-Mod tokamak, which has ITER-relevant plasma pressures, energy densities, and magnetic fields.

The engineering design of the gas jet system benefited greatly from the prior experience of DIII-D. The gas jet outlet nozzle was placed extremely close to the plasma edge (2-3 cm), in the shadow of a protective limiter to prevent damage, as seen in Fig. 9. The external gas jet hardware was designed to reduce extraneous volumes, such as gate valve bodies, expansion chambers, and excessive tubing, in order to maximize the amount of gas fired into the plasma, without compromising overpressure relief safety mechanisms. The heart of the system is a 300 mℓ high pressure plenum and fast valve provided by ORNL. The plenum can be filled remotely with 70 bar of noble gas (He, Ne, Ar, Kr) or deuterium. Most of the valves, pumps, and pressure gauges of the gas jet system are under PLC control (Fig. 10), although the fast valve is fired by camac signal under MDSplus control during the plasma discharge.

Fig. 9. The gas jet nozzle is only 2-3 cm from the plasma edge.

Fig. 10. The gas jet system is monitored and controlled by PLC.

Initial experiments using several different noble gases have confirmed that gas jet mitigation is quite benign as far as subsequent discharges are concerned. Measurements of halo currents (a major contributor to electromagnetic forces) show that they can be cut in half, as illustrated by the colored points in Fig 11. High-Z noble gases, such as argon (Z = 18) and krypton (Z = 36), are found to be better at reducing halo currents compared to low-Z helium. Another goal of the gas jet experiments is to determine whether the high instantaneous thermal loads on divertor surfaces is reduced by the rapid conversion of plasma energy into benign light by the injected gas jet atoms. Data from infrared camera images of the divertor...
surfaces (Fig. 12) is being analyzed to answer this question. In addition, a battery of detailed temperature, bolometry, and magnetic fluctuation measurements and calculations are revealing the mechanisms by which gas jets penetrate into high performance fusion plasmas.

IV. FUTURE PLANS: CRYOPUMP

One of Alcator C-Mod’s long-range research thrusts is to study advanced plasma regimes, with the goal of driving a high fraction of plasma current non-inductively for long pulses. Lower hybrid current drive is an important component of this, and requires relatively low density for good efficiency. Even though C-Mod has an all-metal wall and in-vessel hardware, which retain much less inventory of fueling gas than carbon tiles, it is still likely that a high-speed pumping system will be required in the vessel to permit low-density operation for long pulses. A liquid helium (LHe) cryopump has been designed and is currently being fabricated, with installation scheduled in FY2006. It is designed to achieve an overall system pumping speed of \( \geq 10,000 \, \ell/s \) (10 m\(^3\)/s). The cryopump will be installed in the upper part of the vacuum vessel. The neutral gas pressure in the upper vessel region for magnetic equilibria of interest (i.e. lower x-point, as shown in Fig 13) has been measured to be \( \geq 0.4 \, \text{Pa} \). Thus the pumping throughput will be \( \geq 4 \, \text{Pa}\cdot\text{m}^3/\text{s} \), which is a particle removal rate of \( \geq 1 \times 10^{21} \, \text{D}_2 \) molecules per second. This is about an order of magnitude higher than the fueling due to outgassing of the wall surface inventory. Fig 13 shows the plasma cross-section, the lower divertor structure, and the cryopump in the upper divertor region. Placement in the upper chamber, which currently does not have any specialized divertor hardware in it, greatly simplifies the engineering design, since it allows the cryogenic tubing to be toroidally continuous, with only one inlet and one outlet. This would not be possible in the lower divertor region because of existing divertor support structure. However, in order to shield the cryopump from direct contact with plasma flowing up the scrape-off layer (SOL), new ceiling tile plates, with slots for gas access, have to be installed as well. There is also a room-temperature shield to cutoff the line-of-sight view between the plasma and the pump to prevent radiative heat transfer from the hot plasma.

Although the C-Mod vacuum vessel is up-down symmetric, the specialized divertor structure is only in the lower part of the vessel, and most high-performance plasma operation is with lower single-null equilibria, since the divertor hardware is built to take the required heat loads. This would seem to imply that, from a physics standpoint, the cryopump should be placed in the lower divertor region. However, the standard lower single-null equilibrium still has a ‘virtual’ or ‘secondary’ x-point above the top of the plasma, and an associated scrape-off layer flux. It is empirically found on C-Mod that the scrape-off decay length of the thermal power is much shorter than the decay length of the density, as shown in Fig. 14. Since the secondary separatrix during lower null operation is typically \( > 5 \, \text{mm} \) outside of the primary separatrix (referenced to the flux positions at the plasma midplane), the thermal power arriving at the upper divertor is relatively low, while there is still substantial density outflow. So the upper divertor is nearly as good a location for gas pumping as the lower, with the added advantage that it doesn’t have to handle the thermal load for the usual lower-null configuration.

The cryopump is designed to pump neutral gas molecules, not plasma ions. Plasma ions and electrons streaming along the SOL field lines up toward the cryopump will be stopped by ceiling tile plates, as mentioned previously. At this interaction
surface the charge plasma particles recombine into neutral atoms and molecules. There are toroidally discontinuous slots in the ceiling plates, shown in Fig. 15, through which some fraction of the neutral atoms and molecules pass into the upper chamber. Further collisions with material surfaces reduces the gas molecule energies to about room temperature. It turns out that the design of the ceiling plates and tiles will actually be able to handle the thermal loads that arise during operation with a single upper-null plasma configuration, so that option is indeed available for physics experiments. Also, current sensors embedded in the ceiling plate structure will provide information on disruption halo currents.

The cryopump itself consists of a toroidal pipe (shown in purple in Fig. 15) through which LHe flows once around the torus. (The LHe inlet and outlet are at the same toroidal location). The outside of this pipe is the active pumping surface. It is surrounded by a concentric radiation shield (pale yellow) which is cooled to 77 K by LN\(_2\) flowing through a number of small pipes (blue). This shield also mechanically supports the LHe tube. The shield has a toroidally-continuous gap structure that allows gas molecules to enter the pump, but prevents line-of-sight views between the LHe pipe and the surrounding room-temperature vessel and hardware. The entire cryopump is supported in the upper vessel by a number of thin wire harnesses (not shown) in order to cut down on thermal conduction.

Detailed engineering stress analyses have been carried out with ANSYS[3] to design and ensure survivability of all pump components to \(\vec{J} \times \vec{B}\) disruption electromagnetic loads (arising from toroidal eddy currents as well as halo currents), thermal stresses during cryogenic cooldown to 4.2 K, and sudden quench of the cryogenic fluids (which would happen if the vessel vacuum system was compromised by a large leak).

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**REFERENCES**