The new heated outer divertor and boundary research capabilities for Alcator C-Mod

Motivation
Design goals
Physics goals
Status and plans

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Paper BO7-00002 2012 APS-DPP meeting, Providence, RI

October 29, 2012
We are building a new outer divertor for C-Mod to enhance our contributions to divertor/SOL research

- Toroidally continuous and highly aligned to meet the heat flux challenge
  - No leading edges in the high heat flux / high flux expansion region
- Solid tungsten tiles heatable to 600°C to advance fusion material science
  - Tungsten plasma-surface interactions studies at reactor temperatures
The confluence of a new outer divertor with new diagnostics will enable C-Mod to address reactor and ITER issues

- **Toroidally continuous and highly aligned to meet the heat flux challenge**
  - No leading edges in the high heat flux \ high flux expansion region
- **Solid tungsten tiles heatable to 600° C to advance fusion material science**
  - Tungsten plasma-surface interactions studies at reactor temperatures

- Has successfully undergone preliminary design review
- Current activities are
  - Continue prototyping components
  - Continue electromagnetic analysis
  - Continue thermal analysis
  - Diagnostic design
- Manufacture to start 3/2013
- Operational at end of FY14
Tritium fuel retention in reactors is still a serious concern

Retention of tritium in reactors has to be very low to avoid T site limitations

- $D_{\text{retained}}/D_{\text{incident}} < 10^{-6}$

Lab studies at fluxes $10^2 - 10^4 \times$ lower than in C-Mod or reactor

High-Z Retention of Implanted D

![Graph showing retention as a function of temperature](image-url)

Active control of the divertor temperature allows research on fuel retention in tungsten at reactor temperatures

The C-Mod divertor temperature will be varied to assess predicted D retention in a reactor-temperature tokamak environment

- Allows assessment of retention under reactor-level fluxes/conditions – not accessible through laboratory studies

Potentially access a new operational regime

- Overall recycling will be modified, potentially affecting core performance, e.g.,
  - ‘Super shots’ in TFTR due to low recycling
  - Boronization -> improved confinement

- Will be combined with new in-situ, between discharge, measurement of D in surface

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High-Z Retention of Implanted D

Temperature [K]

<table>
<thead>
<tr>
<th>Temperature [K]</th>
<th>D retained/D incident</th>
</tr>
</thead>
<tbody>
<tr>
<td>600</td>
<td>$10^{-2}$</td>
</tr>
<tr>
<td>400</td>
<td>$10^{-3}$</td>
</tr>
<tr>
<td>500</td>
<td>$10^{-4}$</td>
</tr>
<tr>
<td>800</td>
<td>$10^{-6}$</td>
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</tbody>
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C-Mod divertor characteristics are optimal for studying divertor & SOL physics with ITER/reactor conditions

**Plasma characteristics**

- Density, temperature, parallel heat flux ($q_\parallel$), radiation emissivity, recombination rate match that of ITER
  - Other tokamak divertor densities and heat fluxes ~ 10x lower
- Mean free paths for n-n collisions, radiation absorption approach that of ITER

**The C-Mod divertor is the only plasma in the world that can be used to benchmark ITER divertor simulations before ITER operates**
Our new ion-beam diagnostic (AGNOSTIC) is a quantum step forward for surface studies and is being developed at C-Mod

• Currently-available surface analysis to understand local retention, surface erosion and material migration relies primarily on post-campaign analysis of tiles, studied ex-vessel
  ■ Information – a year delayed after removal
  ■ Surface condition averaged over a campaign
• Only current alternative are small sample holders
  ■ S³ (C-Mod), DiMES (DIII-D), ASDEX-Upgrade

• New 1MeV D ion diagnostic developed at MIT
  ■ 5 minutes/analysis location, between plasmas
  ■ Measurements of D⁰, B, W and Mo areal density
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  ■ 5 minutes/analysis location, between plasmas
  ■ Measurements of D⁰, B, W and Mo areal density
  ■ Can interrogate multiple points poloidally and toroidally by varying toroidal/poloidal fields

The combination of AGNOSTIC with the new tungsten divertor brings a unique opportunity to the US and worldwide.
AGNOSTIC has its first results at the end of the recent run campaign.

- Data obtained at multiple spatial points both between discharges and between run days
  - Analysis underway
The combination of C-Mod divertor/SOL characteristics and diagnostic enhancements will allow access to new studies

- **Materials issues for ITER and reactors**
  - Fuel retention as a function of material temperature (already covered)
  - Tungsten gross and net erosion rates

- **Understanding SOL-divertor physics under ITER and reactor-like conditions**
  - Divertor/SOL code benchmarking (already covered)
  - e-folding width of parallel heat flow, $\lambda q$
  - Impurity seeding

- **Compatibility of high-Z with RF heating and current drive**

C-Mod presents a unique opportunity to study these issues under reactor-like conditions of parallel heat flux and material temperature
The physics of tungsten erosion and prompt re-deposition must be studied under reactor-relevant conditions

- Reactor surface erosion requires net erosion to be extremely low (~1mm erosion/year)
  - Net $\Gamma_W/\Gamma_D \sim 10^{-6}$
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![Diagram of tungsten erosion and prompt re-deposition](image)
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  - Data averaged over campaign so that the fluxes are lower than reactor levels

W. Wampler et al, J. of Nucl. Mat. 266-269 (1999) 217
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- Measurements indicate that prompt re-deposition is not as high as expected
  - Data averaged over campaign so that the fluxes are lower than reactor levels
- Simulations do not agree with experiment
  - $\Rightarrow$ Need dedicated discharges at reactor conditions (ne, Te, sputtering rate IN C-MOD), along with in-situ AGNOSTIC measurement to make progress

Heat flux footprint width on the divertor is a central physics question that improved measurements will address.

Multi-machine database for low divertor recycling, H-mode conditions appears to produce a robust, yet puzzling result:

*Heat flux power channel width is independent of machine size – dependent on $B_{pol}$*

$\lambda_q \sim 1 \text{ mm in ITER!}$
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The above is at odds with a previous ITPA study\(^1\) implying \(\lambda_q \sim 5 \text{ mm.}\)

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\(^1\)B. Lipschutz et al, Nucl. Fusion 47 (2007) 1189
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Another analysis, based on pedestal and divertor physics\(^2\), gives \( \lambda_q \sim 20 \text{ mm} \)

- C-Mod is the only tokamak that has the ITER Bpol, \( \lambda_q \) of 1mm, AND parallel heat flux, \( q|| \), thus making an important contribution to understanding the physics

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\(^1\)B. Lipschutz et al, Nucl. Fusion 47 (2007) 1189

\(^2\)D. Whyte et al., Paper P1-037, 20\textsuperscript{th} Int’l Plasma Surface Interactions meeting, Juelich, 5-2012.
ITER or a tokamak Demo likely requires some radiative dissipation of power before it reaches the first-wall

- Current C-Mod results show significant $Z_{\text{eff}}$ rise (>1) for the ITER $q_{\|}$ level without reaching divertor detachment
  - May be impossible to find an operational regime where seeding is compatible
- Can we control the nonlinear detachment process to control the detachment location?
- Which measurements can be used in a reactor for feedback control of detachment?
- More research is needed for ITER and beyond

Demonstration of impurity seeding control of divertor heat loads still lacking in addressing reactor questions
The C-Mod boundary program is poised to make important contributions to ITER and reactors

- The combination of C-Mod characteristics, diagnostics and upgrades position it to make unique contributions

  - **Materials issues for ITER and reactors**
    - Tungsten gross and net erosion rates
    - Fuel retention as a function of material temperature
  
  - **Understanding SOL-divertor physics under ITER and reactor-like conditions**
    - e-folding width of parallel heat flow, $\lambda q$
    - Code benchmarking for predicting ITER and reactor operation
    - The compatibility of impurity seeding with clean core plasmas

  - **Compatibility of high-Z with RF heating and current drive**
END of presentation
The new outer divertor design is well underway

- Full design has undergone preliminary design review
  - Divertor and new dome area – required for limiting the IR heating of the inner divertor
- Primary activities are
  - Prototyping components
  - Electromagnetic analysis
  - Thermal analysis
  - Langmuir probe design
Two operational modes planned with the new divertor

1) Long-pulse operation (4 seconds of flattop with up to 8MW of LHCD & ICRF)
   • Outer divertor **NOT** actively heated
     ■ 1a – ‘High core radiation’ => ~80% of input power to walls
     ■ 1b – ‘Low core radiation’ => 30% of input power to walls, remainder to divertor
     ■ Outer divertor temperature ratchets up to an elevated temperature dependent on the power flow to it during plasma discharges (temperature determined by equilibrium of radiative losses between shots to plasma energy deposited during discharges)

2) Short-pulse operation (1 second flattop) with actively heated divertor
   • Outer divertor **actively heated up to as high as 600ºC**

• In all cases the steady state vessel temperature would be higher than currently (40ºC)