Extending the boundary heat flux width database to 1.3 Tesla poloidal magnetic field in the Alcator C-Mod tokamak

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Present experiments point to enormous power exhaust challenge in reactors

• "Eich" heat flux width scaling [1]:
  \[ \lambda_q \propto B_p^{-1} \]
  \[ q_{\parallel} \propto \frac{P_{SOL} B}{R} \]

• Unmitigated parallel heat fluxes in reactor \( \sim 10 \) GW/m\(^2\) in reactor-class devices [2]

• Uncertainty in projections

• Some simulations indicate much wider \( \lambda_q \) for ITER than empirical trends


Multi-machine ITPA database was limited in most important parameter: $B_p$

“The result is that practically only the poloidal magnetic field is identified to be statistically important.”[1]:

$$\lambda_q = (0.63 \pm 0.08) \times B_p^{-1.19 \pm 0.08}$$

- ITER 15-MA scenario at 50% higher $B_p$ than maximum in database
- C-Mod has been the only diverted tokamak operated at and above ITER-level $B_p$
- Major focus of C-Mod’s last campaign to characterize $\lambda_q$ at reactor-level $B_p$
- Present here new C-Mod measurements at reactor-relevant $B_p$, $\lambda_q$, and $q_{||}$

Overview

• New C-Mod $\lambda_q$ measurements in H-mode show inverse $B_p$ scaling continues to above ITER-level $B_p$

• Cross-confinement (L-, I-, and H-mode) $\lambda_q$ organizes with inverse square root of volume-average core plasma pressure
  • Linkage of physics setting core and boundary confinement
  • Challenge to 1st-principles and heuristic models

• Database challenges two assumptions of heuristic drift model

• Probe-based measurements improve resolution over IR analysis, revealing important limitations in assumptions and analytic equations used to fit data
Narrow $\lambda_q$ calls for improved diagnostics

- IR thermography only heat flux diagnostic used in multi-machine ITPA database
- IR on C-Mod has oblique view of outer divertor [1]
- Resolution estimated at $\sim 0.5$ mm (mapped to outer mid-plane) [2]
- Projected $\lambda_q$ at $B_p = 1.2$ T:
  \[
  \lambda_q \approx 0.63 \times 1.2^{-1.19} \approx 0.51 \text{ mm}
  \]
- Needed to develop a finer measurement tool...


C-Mod probes have sufficient resolution for high-$B_p$, narrow-$\lambda_q$ measurements

- Surface thermocouples expose refractory metal thermojunction to divertor plasma, directly measure surface heat flux [1]
  - Validated against calorimetry, IR thermography, and Langmuir probes
- Conservative resolution estimate $\sim 0.05$ mm (mapped to outer mid-plane)
  - 10x better than C-Mod IR system

Shot-integrated energy flux used to validate sensors for each shot in the database

- Integrate time evolution of surface heat flux from surface thermocouples and Langmuir probes [1]
- Compare to energy flux from tile thermocouples and calorimeters
- Identify and remove erroneous probes
- Plasma and total energy fluxes match: minimal photon and neutral energy fluxes
- Sheath heat flux assumptions appropriate
  - $T_i \approx T_e$
  - minimal net secondary electrons
  - account for net currents

Combined probe-based heat flux profile is a significant improvement over IR analysis

- Higher resolution
- Higher dynamic range:
  - Probes: ~10,000
  - IR: ~100

Note: different shots, both outer divertor
ITPA database used analytic method for obtaining $\lambda_q$ from measurements

- At divertor entrance: exponential profile truncated at last closed flux surface
  \[ q_{\parallel} = q_0 \exp \left( -\frac{\rho}{\lambda_q} \right) \rho > 0 \]

- Assume symmetric cross-field heat transport with Gaussian spreading, characteristic length $S$

- Convolve to get "Eich" profile [1]:
  \[ q(\rho) = \frac{q_0}{2} \exp \left( \left( \frac{S}{2\lambda_q} \right)^2 - \frac{\rho}{\lambda_q} \right) \operatorname{erfc} \left( \frac{S}{2\lambda_q} - \frac{\rho}{S} \right) + q_{BG} \]

- Includes uniform background heat flux $q_{BG}$

- Fit to measurements to extract $\lambda_q$

Standard fit insufficient to capture profile details revealed by probes

• Background term tries to fit far SOL
• Without background, the Gaussian spreading is a poor fit to private flux region

\[ q(\rho) = \frac{q_0}{2} \exp \left( \frac{(S)}{2\lambda_q} - \frac{\rho}{\lambda_q} \right) \text{erfc} \left( \frac{S}{2\lambda_q} - \frac{\rho}{S} \right) + q_{BG} \]
Improved “multi-\(\lambda\)” equation fits probe data over entire profile

- Including Gaussian convolution allows it to fit IR data as well, although almost never needed for probes
- IR analysis may be resolution limited
- Exponential private region seen in other tokamaks

\[
q_{||} (\rho) = \begin{cases} 
(q_0 - q_{pf}) e^{\rho/\lambda_{q, pn}} + q_{pf} e^{\rho/\lambda_{q, pf}}, & \rho < 0 \\
(q_0 - q_{cf}) e^{-\rho/\lambda_{q, cn}} + q_{cf} e^{-\rho/\lambda_{q, cf}}, & \rho \geq 0
\end{cases}
\]

Note: different shots, both outer divertor
Assumption of Gaussian-like cross-field heat transport in the outer leg does not match data

- May be okay for high-dissipation regimes, but not where plasma transport dominates

- Asymmetric profile on either side of strike point reminiscent of upstream profiles [1]:
  - Large turbulent cross-field fluxes on low-field side
  - Minimal turbulent cross-field fluxes on high-field side

- Similar curvature drive and stabilization in the divertor leg?
  - If so, leg angle in poloidal plane would be important

- Asymmetric turbulent phenomena seen in fast camera imaging of private flux region [2,3]

- Private flux $\lambda_q$ scales roughly with boundary collisionality, no clear trend found yet

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C-Mod high-field, high-resolution $\lambda_q$ database with >300 shots

• Criteria: Good divertor heat flux profile from strike point sweep with low dissipation and relatively steady upstream conditions
  • $\sim$1/3$^{rd}$ EDA H-mode, all forward field ($B \times \nabla B$ drift to x-point), no ELMs
  • $\sim$1/3$^{rd}$ I-mode, all reverse field ($B \times \nabla B$ drift away from x-point), no ELMs
  • $\sim$1/3$^{rd}$ L-mode, half-and-half forward and reverse fields, no ELMs

• Wide range of engineering parameters
• Narrow range of shape due to need to keep strike point on sensors

<table>
<thead>
<tr>
<th>$B_T [T]$</th>
<th>$B_p [T]$</th>
<th>$\bar{n}_e [10^{20} / m^3]$</th>
<th>$P_{in} [MW]$</th>
<th>$\kappa$</th>
<th>$\delta$</th>
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<tbody>
<tr>
<td>2.7-8.0</td>
<td>0.43-1.3</td>
<td>0.44-5.2</td>
<td>0.52-5.5</td>
<td>1.5-1.8</td>
<td>0.48-0.61</td>
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C-Mod unmitigated heat flux conditions near reactor-level

- Heat flux widths from 0.5 mm to 2.5 mm
- Peak parallel heat fluxes from 0.1 GW/m$^2$ to 2 GW/m$^2$
  - Surface heat flux up to ~100 MW/m$^2$
New high-field C-Mod data extends empirical trend in H-mode to ITER-level $B_p$

- Better resolution, less scatter from probe measurements and improved analytic fitting equation

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New high-field C-Mod data extends empirical trend in H-mode to ITER-level $B_p$

- Better resolution, less scatter from probe measurements and improved analytic fitting equation
- $B_p$-scaling ITER: $\lambda_q = 0.51$ mm
- Minimum C-Mod database: $\lambda_q = 0.48$ mm
- Empirical poloidal field scaling stands in stark contrast to recent simulations [2,3]

Some have suggested $\rho^*$ to be cause of wider $\lambda_q$ predicted for ITER

- Simulated $\lambda_q$ for ITER $\sim$10 times wider than empirical scalings
- Difference sometimes attribute to transition to “new physics” at smaller $\rho^*$
- C-Mod $\rho_p^*$ at LCFS $\sim$10x larger than ITER
- JET $\rho_p^*$ at LCFS is $\sim$2x larger than ITER
  - Unlikely to see major change in $\rho_p^*$-based physics from JET to ITER
  - No change in $\sim$20x $\rho_p^*$ range in present experiments
- Need predictions from simulations that we can test in present experiments to have confidence in “new physics” for ITER

[1] ITPA Confinement Database DB4v5.
Poloidal field trend matters across confinement regimes

- **H-mode:**
  - Purely inverse scaling
  - Slightly wider than purely $B_p$ multi-machine scaling
  - Nearly identical to multi-machine scaling when including aspect ratio (Regression #15 [1])

- **L-mode:**
  - $\sim2x$ wider than H-mode
  - Nearly inverse scaling

- **I-mode:**
  - Scattered distribution bounded between L- and H-mode

Poloidal magnetic field also strongly influences core confinement

• H-mode (ITER DB): 
  \[ \tau_e \propto I_p^{0.93} \]

• L-mode (ITER DB): 
  \[ \tau_e \propto I_p^{0.96} \]

• I-mode (C-Mod DB): 
  \[ \tau_e \propto I_p^{0.69} \]
Volume-averaged core plasma pressure reduces scatter and unifies heat flux width across regimes

• Scaling with inverse square-root of core plasma pressure (equivalently stored energy at constant geometry)
  • Shots are non-hybrid, no ITBs, only ETBs

• L- and H-mode not surprising:
  • Stored energy dominated by plasma current
  • H-mode higher confinement and narrower $\lambda_q$ than L-mode; see also ASDEX-U and JET

• I-mode ties together intermediate stored energy and $\lambda_q$

• Simple rule-of-thumb: $\lambda_q [\text{mm}] \approx \frac{1}{\sqrt{\bar{\rho} [\text{atm}]}}$

• Or, in SI units: $\lambda_q [\text{m}] \approx \sqrt{\frac{0.1 [\text{N}]}{\sqrt{\bar{\rho} [\text{Pa}]}}}$
Volume-averaged core plasma pressure reduces scatter and unifies heat flux width across regimes

- Can have the same core pressure and divertor heat flux width over a wide range of poloidal magnetic fields
- Direct link between physics setting core and boundary cross-field physics?
- Marginal stability from core, through pedestal, and to divertor?
First analysis of $\overline{p}$-$\lambda_q$ size scaling: combine ITPA databases

• Examine individual machines at high and low $B_p$ values
• Take the maximum extent of heat flux widths from ITPA $\lambda_q$ database $\lambda_q$-$B_p$ plot (Fig. 3) [1]
• Sort ITPA H-mode confinement database by $B_p$, take mean volume-averaged plasma pressure $\pm 2\sigma$ [2]

Initial analysis suggest similar scaling across machines in H-mode

- For a given machine and $B_p$, heights of ellipses are $\lambda_q$-ranges and widths are $\tilde{p}$-ranges.
- Good agreement with C-Mod data, especially higher-field data.
- May need correction for low-aspect ratio.
- Projections remarkably close between scalings and remarkably narrow.

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<tbody>
<tr>
<td>$B_p^{-1}$ scaling</td>
<td>0.51</td>
<td>0.49</td>
<td>0.39</td>
<td>0.26</td>
</tr>
<tr>
<td>$\tilde{p}^{-1/2}$ scaling</td>
<td>0.51</td>
<td>0.42</td>
<td>0.39</td>
<td>0.28</td>
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Hints of confinement–heat flux width link in previous C-Mod H-mode results

• Heat flux width decreases with increased stored energy [1]
  • Over much smaller range, trend less clear

• Links to results elsewhere
  • Increased confinement and decreased heat flux width in NSTX with Li [2,3]
  • ASDEX-U (Happel talk yesterday)

• Connection between core and boundary confinement identified in C-Mod 20 years ago [4]

New C-Mod data also extends HD model to new parameter space, lower $\lambda_q$

- Heuristic Drift model assumptions [1]:
  - Cross-field particle transport dominated by classical drifts in H-mode
  - Parallel Pfirsch–Schlüter flows balanced by near sonic-parallel losses to divertor
  - Anomalous cross-field electron thermal diffusion fills in SOL plasma channel
  - Spitzer–Härm parallel heat loss
  - Small volumetric particle sources in SOL

- Fit multi-machine database well
- New C-Mod data follows HD model

I-mode challenges HD model particle transport assumption

- I-mode
  - L-mode-like particle confinement
  - H-mode-like energy confinement
- HD model assumes:
  - Turbulent particle losses dominate L-mode
  - Particle drift losses dominate H-mode

\begin{align*}
\text{all } B_p &= 1.2 \, T
\end{align*}
I-mode challenges HD model particle transport assumption

- I-mode
  - L-mode-like particle confinement
  - H-mode-like energy confinement
- HD model assumes:
  - Turbulent particle losses dominate L-mode
  - Particle drift losses dominate H-mode
- But, I-mode follows HD model scaling as well as H-mode; L-mode close in some cases
- Particle channel does not set $\lambda_q$ as assumed in HD model
  - $\lambda_q$ related to upstream temperature profile, depending on collisionality [1] (also Happel talk yesterday)

Neutral pressure measurements challenge HD model particle recycling assumption

• Expect increased neutral particle sources to broaden SOL particle width [1]
• Measured $\lambda_q$ should then be wider than $\lambda_{HD}$ prediction with increased neutral particle recycling
• Yet, experimental trend is weak or in the opposite direction over $\sim 10^3$ change in neutral pressure
• HD model gets the right answer, but for the wrong reason?

Conclusion 1: Benchmarked, probe-based sensors have improved resolution and dynamic range over IR analysis

- An improved analytic 4-exponential divertor heat flux profile (near/far private/common regions)
  - Results in improved common SOL heat flux fitting, less scatter

- **Poor applicability of uniform Gaussian spreading parameter S**, cross-field transport is asymmetric under low dissipation

- Private flux region $\lambda_q$ is narrower than common, possibly linked to curvature driven/stabilized cross-field pressure gradients

- Private flux $\lambda_q$ correlated with edge collisionality, no clear trend yet found

![Graph showing heat flux profile with parameters and data points](image)
Conclusion 2: Alignment of $\lambda_q$ across confinement regimes with inverse square root of core plasma pressure

- Connection of cross-field transport setting core confinement and divertor heat flux width?
- Initial look reveals similar trend across machines in H-mode, requires further exploration
- Simple rule-of-thumb:
  $$\lambda_q \text{ [mm]} = \frac{1}{\sqrt{\bar{\rho}} \text{ [atm]}}$$
Conclusion 3: H-mode inverse $B_p$ scaling continues to above ITER-level $B_p$

- Empirical trend challenges models and heuristic assumptions

- Future work:
  - Examine private flux width scaling in detail
  - Connect upstream profiles across regimes, search for common/different physics
  - Multi-machine, multi-confinement mode examination of pressure-based $\lambda_q$ scaling