Overview of Recent Experimental Results from the DIII-D Advanced Tokamak Program

by

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for the DIII-D Team

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GREAT ATOMICS
DIII–D ADVANCED TOKAMAK PROGRAM GOAL

- The DIII–D AT research program is developing the scientific basis for advanced operating modes in order to enhance the attractiveness of the tokamak as an energy producing system.

- This requires optimizing for:
  - High power density (high $\beta = 2 \mu_0 \langle p \rangle / B^2$)
  - High ignition margin (high energy confinement time $\tau_E$)
  - Steady-state operation with low recirculating power (high bootstrap fraction)

- Key issues in optimization are:
  - Active MHD stability control
  - Current profile control
  - Pressure profile control
We have made progress since the last IAEA meeting in developing the building blocks needed for advanced operating modes:

- Substantially broadened the MHD stable operating space
  - Rotational stabilization of resistive wall modes yielding
    \[ \beta_N = \beta_N \text{ (ideal wall)} = 2 \beta_N \text{ (no wall)} \]
  - Increased \( \beta \) by 60% via stabilization of (3,2) neoclassical tearing mode with ECCD in sawtoothing plasmas
  - First stabilization of (2,1) neoclassical tearing mode using ECCD

- Developed plasma control tools
  - First integrated AT discharges with current profile control using ECCD
  - Pressure and density profile control with ECH and ECCD

- Demonstrated an improved, high \( q_{95} (>4) \) operating scenario for ITER

- Achieved solutions to key burning plasma issues
  - No ELM-produced, pulsed divertor heat load in QH–mode plasmas
  - Small heat and particle loads at inner divertor strike points in balanced double-null divertors
  - Disruption mitigation via massive gas puff
SUBSTANTIAL PROGRESS SINCE IAEA 2000

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MHD STABLE TOKAMAK OPERATING SPACE APPROXIMATELY DOUBLED BY SUPPRESSION OF EXTERNAL KINK INSTABILITY

Key: sustainment of plasma rotation

Theoretically predicted (Bondeson and Ward, 1994)

\[ \beta = \frac{\text{plasma pressure}}{\text{magnetic pressure}} \]

\[ \beta_N = \frac{\beta}{l/aB_T} \]

tokamak stability parameter

Ideal-wall kink limit (approx)

No-wall kink limit

Rotation reduced

\( \beta_N \)

106521 107603

Time (ms)
WALL STABILIZATION OF EXTERNAL KINK VIA PLASMA ROTATION BROADENS OPERATING SPACE

- Wall stabilization of the external kink is possible via stabilization of the resistive wall mode (RWM) by plasma rotation
  - Duration in previous experiments limited by the slowing of plasma rotation

- New Discovery: Rotation slowing at \( \beta \) above the no-wall limit is a consequence of “resonant field amplification” (RFA) [A. Boozer, Phys. Rev. Lett. 86 (2001)]

- New Discovery: Reduction of the non-axisymmetric (error) fields enables continued plasma rotation at \( \beta \) above the no-wall limit

\[ \Rightarrow \text{Reduced error field} \]
\[ \Rightarrow \text{Sustained plasma rotation} \]
\[ \Rightarrow \text{Stable operation well above the no-wall } \beta \text{ limit (up to ideal-wall limit)} \]
NON-AXISYMMETRIC “C-COIL” AND MAGNETIC FIELD SENSORS ARE USED FOR RWM AND RFA MINIMIZATION BY FEEDBACK CONTROL

- Six midplane coils (C–coil) connected in three pairs for n=1 control
- External and internal saddle loops measure $\delta B_r$
- Poloidal field probes measure $\delta B_p$ with reduced coupling to the control coils
A MAGNETIC FEEDBACK SYSTEM COMBINED WITH ROTATIONAL STABILIZATION CAN PROVIDE A PATH TO IDEAL-WALL $\beta_N$ LIMIT
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Maintain/Increase Rotation

Suppress RWM

No FB
No Rotation

$G=0$

Increasing Rotation

No-Wall Limit

Ideal-Wall Limit

Stable
A MAGNETIC FEEDBACK SYSTEM COMBINED WITH ROTATIONAL STABILIZATION CAN PROVIDE A PATH TO IDEAL-WALL $\beta_N$ LIMIT

- Maintain/Increase Rotation
- Reduce Resonant Residual Error Field
- Magnetic Feedback
- Sensor
- Suppress RWM

Graph showing the relationship between $\gamma \tau_W$ and $\beta_N/\beta_N^{\text{No-Wall}}$. The graph indicates that increasing rotation helps to suppress RWM and reduce resonant residual error field, leading to stability within the no-wall limit and potentially reaching the ideal-wall limit.
FEEDBACK ALLOWS $\beta_N$ TO APPROACH IDEAL WALL LIMIT

- $\beta_N \equiv \beta_N$ (ideal wall) $\equiv 2 \beta_N$ (no-wall) (GATO-code)
- MHD at collapse grows on ideal-kink time scale

Rotational stabilization also possible with preprogrammed C–coil current

- Detailed feedback control of C–coil not necessary for rotational stabilization
A MAGNETIC FEEDBACK SYSTEM COMBINED WITH ROTATIONAL STABILIZATION CAN PROVIDE A PATH TO IDEAL-WALL $\beta_N$ LIMIT
RWM STABILIZATION BY ROTATION ALLOWS HIGH $\beta_N$ H$_{89}$ OPERATION IN ADVANCED TOKAMAK PLASMAS

- $\beta_N$ H$_{89} \geq 10$ for 680 ms ($4\tau_E$)
- $\beta = 4.2\%$, $\beta \tau_E = 0.66\%$ s, $\beta_p = 2$
- $\beta_N = 1.5 \beta_N$ (no-wall)
- Bootstrap fraction 65%
- Total non inductive fraction 85%
- Duration limited by drop in $q_{\text{min}}$ leading to onset of 2/1 neoclassical tearing mode (NTM)
  - Motivates work on current drive and NTM stabilization
\[ \beta_N \text{ RAISED 60\% AFTER ECCD SUPPRESSION OF } m/n = 3/2 \text{ NTM} \]

- Location of ECCD optimized in real time to minimize NTM amplitude
  - Location held fixed when amplitude is zero
  - Mode reappears as \( q = 3/2 \) moves radially by 2 cm off ECCD location

\[ \Delta R_{3/2} \approx 2 \text{ cm due to high } \beta \text{ Shafranov shift} \]

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DEMONSTRATED COMPLETE SUPPRESSION OF THE $m/n = 2/1$ TEARING MODE BY RADIAILLY LOCALIZED ECCD

- $\beta_N$ is feedback controlled to temporarily rise to excite the mode
- Location of ECCD optimized (#111367) by toroidal field PCS "Search and Suppress"

More information in R.J. La Haye et al. EX/S1-3
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VALIDATED ECCD THEORY ALLOWS USE OF DETAILED COMPUTER MODELS TO DEVELOP EXPERIMENTS

- Excellent agreement of ECCD theory and experiment
- Prediction of enhanced negative central shear in AT plasma with ECCD at $\rho = 0.4$

More information in C.C. Petty, et al., EX/W-4

Graph showing $\chi^2_{\text{reduced}} = 1.1$ and agreement of ECCD theory and experiment with experimental data from CQL3D Code.

Graph showing prediction of enhanced negative central shear in AT plasma with ECCD localized at $\rho = 0.4$. Simulation data compared with experimental data ($q_0(t)$ and $q_{\text{min}}(t)$) and NBI only case.
ECCD PRODUCES CURRENT PROFILE MODIFICATION IN ADVANCED TOKAMAK PLASMA

- $\beta_N H_{89} \geq 7$ for full 2.0 s ECCD pulse
- $\beta_N$ at or slightly above $\beta_N$ (no-wall)
- Total non-inductive current fraction $\geq 90\%$
- $q$ profile modified during high $\beta$, AT phase of shot

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**Graphs:**

- $I_p$ (MA)
- $P_{\text{inj}}$ smoothed (10 MW)
- $\bar{n}_e$ ($10^{20} \text{ m}^{-3}$)
- $q_0$, $q_{\text{min}}$
- $T_i(0)$ (keV)
- $T_e(0)$ (keV)

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**Figure:**

- Time (s)
- ECCD Power (au)
- EC

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**Additional Information:****

- DIII-D National Fusion Facility
- SAN DIEGO
- 255–02/KHB/wj
ECCD PEAKS CURRENT DENSITY AT RESONANCE LOCATION AND PRODUCES STRONGER NEGATIVE MAGNETIC SHEAR

EFIT equilibrium analysis using MSE

\[ \langle J_\parallel \rangle \]

- Clear evidence of q–profile modification also seen in quiescent double barrier (QDB) plasmas (E.J. Doyle, et al. EX/C3-2)
ECCD CAN TRIGGER FORMATION OF CORE TRANSPORT BARRIERS IN ADVANCED TOKAMAK DISCHARGES

- Core barriers seen in all four transport channels with ECCD
  - No barriers in ECH case with no current drive
- Gyrokinetic stability code analysis shows $E \times B$ shear and Shafranov shift stabilization are both important
- More information in M.R. Wade et al. EX/P3–16
DENSITY AND IMPURITY PROFILES MODIFIED WITH ECH AND ECCD IN QUIESCENT DOUBLE BARRIER PLASMAS

- EC power applied near $\rho = 0.2$ in plasma with core transport barrier already formed
- Density peaking reduced, leading to much reduced central impurity densities and factor 1.3 reduction in $Z_{eff}$
- More information in E.J. Doyle, et al., EX/C3-2

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**Graphs:**

- Ne (10$^{19}$ m$^{-3}$)
- Total Pressure (kPa)
- Total Ni Density (10$^{17}$ m$^{-3}$)
- C$^+6$ Density (10$^{18}$ m$^{-3}$)
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Stationary plasmas with $\beta_N H_{89}/\gamma_{95}^2 \approx$ ITER design value and $\gamma_{95} > 4$ have been demonstrated on DIII-D.

- Fusion gain proportional to $\beta_N H_{89}/\gamma_{95}^2$
- $q(0) > 1$
- $3/2$ NTM prevents sawteeth
- $\beta_N \approx \beta_{N_{\text{no-wall}}}$
- Non-inductive current fraction $\approx 50$
- Candidate for extended pulse length, hybrid scenario for ITER
CURRENT PROFILE IS FULLY RELAXED AND WALL PARTICLE INVENTORY IS EQUILIBRATED AFTER 3.0 s

- $\tau_{\text{dur}} \approx 36 \tau_E \approx 2 \tau_{\text{CR}}$
- Wall not important in particle balance

MSE Pitch Angle $\propto B_z$ at various radii

Wall Inventory ($T \cdot \ell$)
$N_{\text{wall}} (T \cdot \ell/s)$

$n_c/n_e (\rho = 0.12)$
$n_c/n_e (\rho = 0.67)$

More information in T.C. Luce et al. EX/P3–13
ENERGY TRANSPORT IN LONG PULSE DISCHARGE COMPARABLE TO THAT OBTAINED IN LOW $q_{95}$ REFERENCE SHOT

- $\chi_{\text{eff}}$ substantially lower than that expected by $q$ scaling of transport
  - Global confinement scaling: $\chi_{\text{eff}} \propto q^{1.4}$
  - Nondimensional transport studies: $\chi_{\text{eff}} \propto q^2$

![Graph showing thermal diffusivity vs. normalized radius for different $q_{95}$ values.](image)
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![Graph showing $\chi_{\text{eff}}$ vs. normalized radius for different $q$ values](graph.png)
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- GLF23 drift-wave mode simulation give good agreement with measured profiles
- Model contains ITG, TEM, and ETG with effects of \( E \times B \)
- More information in J.E. Kinsey et al TH/P1-9

![Graph showing thermal diffusivity and temperature profiles](image-url)
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QUIESCENT H–MODE RUNS ELM–FREE FOR LONG PULSES WITH CONSTANT DENSITY AND RADIATED POWER

Duration limited by neutral beam pulse length

- $I_p$ (MA)
- $D_\alpha$ (a.u.)
- $T_i^{ped}$ (keV)
- $T_e^{ped}$ (keV)
- $\bar{n}_e$ ($10^{19} \text{ m}^{-3}$)
- $n_e^{ped}$ ($10^{19} \text{ m}^{-3}$)
- $P_{inj}$ (MW)
- $P_{rad}$ (MW)
QUIESCENT H–MODE HAS BEEN SEEN OVER A RANGE OF PARAMETERS

- Requires neutral beam injection counter to $I_p$ direction plus divertor cryopumping
- QH–mode seen to date for
  \[ 3.4 \leq q_{95} \leq 5.8 \]
  \[ 1.0 \leq I_p \text{ (MA)} \leq 2.0 \]
  \[ 1.8 \leq B_T \text{ (T)} \leq 2.1 \]
  \[ 1.0 \leq n_e^{\text{ped}} \text{ (10^{19} m^{-3})} \leq 6.5 \]
- Low field example at
  \[ B_T = 0.95 \text{ T} \]
  \[ I_p = 0.67 \text{ MA} \]
  \[ n_e^{\text{ped}} = 1.1 \times 10^{19} \text{ m}^{-3} \]
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  $1.8 \leq B_T$ (T) $\leq 2.1$
  $1.0 \leq n_e^{\text{ped}} (10^{19} \text{ m}^{-3}) \leq 6.5$
- Low field example at $B_T = 0.95$ T, $I_p = 0.67$ MA and $n_e^{\text{ped}} = 1.1 \times 10^{19}$ m$^{-3}$
- QH–mode recently seen in ASDEX–U

![High Density QH–Mode](image)
THE INBOARD DIVERTOR PARTICLE AND HEAT FLUXES ARE RELATIVELY LOW IN SYMMETRIC DOUBLE-NULL PLASMA

- ELM activity at the inboard target(s) is significantly reduced in DN → ELMs are generated on the outboard side (consistent with ELITE analysis)
- Strong variation in the time-averaged particle flux ratios
  e.g., $\frac{\Gamma_{in}}{\Gamma_{out}} \approx 0.2$ (DN) $\approx 1$ (SN)
- Strong variation in the time-averaged heat flux ratios
  e.g., $\frac{q_{in}}{q_{out}} \approx 0.05\text{--}0.15$ (DN) $\approx 0.3\text{--}0.5$ (SN)
- Results simplify divertor design in future DN tokamaks

Modeling (UEDGE) indicates that particle drifts in the divertor play important roles in interpreting these results.
NEON/ARGON GAS JET IMPURITY INJECTION INTO A STABLE PLASMA RESULTS IN A RAPID, CLEAN PLASMA TERMINATION

- 70 bar gas jet propagates through plasma without significant MHD activity
- High radiated power from neon collapses central $T_e$ and $\beta$
- Ten-fold increase in density
- Fast and clean current quench
  - No sign of non-thermal $e^-$ owing to high gas density
- Plasma remains well centered in vessel
IONIZATION/ENERGY BALANCE MODEL (KPRAD) MATCHES KEY FEATURES OF GAS JET MITIGATION EXPERIMENTS:

INITIAL BURNTHROUGH $\rightarrow P_{\text{rad}}$ $\rightarrow T_e$ COLLAPSE $\rightarrow n_e$ CLAMPED

Model result:
Neon gas jet into DIII–D
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Neon gas jet into DIII–D

Extrapolation to ITER is promising
(see D.G. Whyte et al. EX/S2-4)
REAL-TIME DISRUPTION DETECTION IS BEING USED TO TRIGGER GAS JET FOR VERTICAL DISRUPTION MITIGATION

- Gas jet triggered when plasma control system detects vertical plasma shift

**Graphs:**
- **Horizontal Axis:** Time after removal of vertical stabilization (ms)
- **Vertical Axis:**
  - Plasma Vertical Position (cm)
  - Vertical Instability Trigger from Control System to Neon Gas Jet
  - Plasma Current (MA)

**Legend:**
- **Equilibrium Position**
- **Trigger levels**
- **no mitigation**

**Graph 2:**
- **Y-axis:** Peak poloidal halo current (MA)
- **X-axis:** Toroidal peaking factor
- **Legend:** no mitigation

**Notes:**
- Divertor $J \times B$ forces from VDE reduced using gas jet mitigation
- Rapid + centered current quench

**SAN DIEGO DIII-D NATIONAL FUSION FACILITY**
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## ADDITIONAL PRESENTATIONS CONTAINING DIII–D RESULTS

<table>
<thead>
<tr>
<th>Topic</th>
<th>Author</th>
<th>Paper</th>
</tr>
</thead>
<tbody>
<tr>
<td>Feedback stabilization of NTMs with ECCD</td>
<td>R.J. La Haye</td>
<td>EX/S1-3</td>
</tr>
<tr>
<td>Stabilization of resistive wall modes</td>
<td>E.J. Strait</td>
<td>EX/S2-1</td>
</tr>
<tr>
<td>Modeling the stabilization of RWMs</td>
<td>M.S. Chu</td>
<td>TH/P3-10</td>
</tr>
<tr>
<td>Disruption mitigation by high pressure gas injection</td>
<td>D.G. Whyte</td>
<td>EX/S2-4</td>
</tr>
<tr>
<td>Sustaining steady-state AT discharges</td>
<td>M.R. Wade</td>
<td>EX/P3-16</td>
</tr>
<tr>
<td>Electron cyclotron current drive</td>
<td>C.C. Petty</td>
<td>EX/W-4</td>
</tr>
<tr>
<td>Electron cyclotron technology for plasma control</td>
<td>R.W. Callis</td>
<td>CT-7Rc</td>
</tr>
<tr>
<td>Scaling and modeling of high bootstrap tokamaks</td>
<td>F.W. Perkins</td>
<td>EX/P3-18</td>
</tr>
<tr>
<td>Internal transport barrier physics in QDB discharges</td>
<td>E.J. Doyle</td>
<td>EX/C3-2</td>
</tr>
<tr>
<td>Turbulence stabilization by equilibrium and zonal flows</td>
<td>G.R. McKee</td>
<td>EX/C4-1Ra</td>
</tr>
<tr>
<td>Comparison of simulations with turbulence measurements</td>
<td>T.L. Rhodes</td>
<td>EX/C4-1Rb</td>
</tr>
<tr>
<td>Comprehensive gyrokinetic simulations</td>
<td>R.E. Waltz</td>
<td>TH/P1-20</td>
</tr>
<tr>
<td>Alternate ITER baseline scenario</td>
<td>T.C. Luce</td>
<td>EX/P3-13</td>
</tr>
<tr>
<td>DIII–D-like AT scenario for ITER</td>
<td>L.L. Lao</td>
<td>EX/P3-12</td>
</tr>
<tr>
<td>Transport modeling for burning plasma experiments</td>
<td>J.E. Kinsey</td>
<td>TH/P1-09</td>
</tr>
<tr>
<td>ELM stability, peeling-ballooning mode</td>
<td>P.B. Snyder</td>
<td>TH/3-1</td>
</tr>
<tr>
<td>H−mode pedestal width and neutral penetration</td>
<td>R.J. Groebner</td>
<td>EX/C2-3</td>
</tr>
<tr>
<td>Acceptable ELM Regimes for Burning Plasmas</td>
<td>A.W. Leonard</td>
<td>EX/P3-06</td>
</tr>
<tr>
<td>Turbulence in the SOL of C−Mod, DIII–D, and NSTX</td>
<td>J.L. Terry</td>
<td>EX/P5-10</td>
</tr>
<tr>
<td>Blobs and cross-field transport in the tokamak edge</td>
<td>S.I. Krasheninnikov</td>
<td>TH/4-1</td>
</tr>
<tr>
<td>The effects of drifts on the boundary plasma</td>
<td>G.D. Porter</td>
<td>EX/P3-07</td>
</tr>
<tr>
<td>H−mode pedestal and extrapolation to ITER (ITPA)</td>
<td>T.H. Osborne</td>
<td>CT–3</td>
</tr>
<tr>
<td>Transport and ITB physics (ITPA)</td>
<td>P. Gohil</td>
<td>CT/P–05</td>
</tr>
</tbody>
</table>