PROGRESS IN MFE SCIENCE — TOLKAMAK RESEARCH

by

R.D. Stambaugh

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### RESEARCH RESULTS FROM

<table>
<thead>
<tr>
<th>Alcator C–mod</th>
<th>JET</th>
<th>TCV</th>
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<tr>
<td>ASDEX Upgrade</td>
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<td>HIT</td>
<td>START</td>
<td>T–10</td>
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### SPECIAL THANKS FOR DIRECT CONTRIBUTIONS

<table>
<thead>
<tr>
<th>S. Allen</th>
<th>A. Hubbard</th>
<th>J. Lister</th>
<th>F. Romanelli</th>
<th>D. Whyte</th>
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<tr>
<td>S. Bernabei</td>
<td>I. Hutchinson</td>
<td>T. Luce</td>
<td>S. Seitz</td>
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<td>L. Berry</td>
<td>F. Jaeger</td>
<td>D. Meade</td>
<td>M. Shimada</td>
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<td>P. Bonoli</td>
<td>G. Janeschitz</td>
<td>J. Menard</td>
<td>G. Staebler</td>
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<td>K. Burrell</td>
<td>M. Kaufmann</td>
<td>G. Navratil</td>
<td>E. Strait</td>
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<td>M. Chu</td>
<td>S. Kaye</td>
<td>W. Nevins</td>
<td>A. Sykes</td>
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<td>E. Doyle</td>
<td>J. Kinsey</td>
<td>H. Ninomiya</td>
<td>E. Synakowski</td>
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<td>E. Frederickson</td>
<td>A. Kitsunezaki</td>
<td>W. Park</td>
<td>T. Taylor</td>
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<td>A. Garofalo</td>
<td>R. La Haye</td>
<td>R. Perkins</td>
<td>A. Turnbull</td>
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<td>C. Greenfield</td>
<td>L. Lao</td>
<td>R. Pinsker</td>
<td>R. Waltz</td>
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<td>M. Greenwald</td>
<td>G.S Lee</td>
<td>P. Politzer</td>
<td>M. Watkins</td>
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<td>R. Hawryluk</td>
<td>F. Leuterer</td>
<td>M. Porkolab</td>
<td>R. Weynants</td>
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<tr>
<td>J. Hosea</td>
<td>B. Lipschultz</td>
<td>R. Prater</td>
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MAIN POINTS

- We have learned a tremendous amount about magnetically confined plasmas
  - Measurements and theory
  - Calculations

- Exciting new directions are opening
  - Advanced Tokamak research

- We are technically ready for next steps
OUTLINE

• The tokamak equilibrium

• Heating and current drive

• Stability

• Confinement

• Power and particle control

• Burning plasma physics

• Next steps

• Conclusions
WHAT IS A TOKAMAK?

- An axisymmetric toroidal confinement configuration with a strong toroidal plasma current and an applied toroidal magnetic field strong enough to make the edge winding factor > 2

- Not part of the basic definition but certainly part of the opportunity for variation and innovation within the concept are:
  - Shape (elongation, triangularity)
  - Aspect ratio
  - Divertor or limiter boundary
  - Toroidal field strength
  - Current profile
  - Pressure profile
  - Rotation profile
  - Radial electric field profile
  - Wall stabilization

Advanced Tokamak

MFE-Tokamak
TOKAMAKS HAVE MADE EXCELLENT PROGRESS IN FUSION POWER

Fusion Power (MW)


YEAR

Ohmic
RF
NBI-D
NBI-DT
PLASMA EQUILIBRIUM THEORY IS WELL UNDERSTOOD AND EXTENSIVELY USED

- Ampere's Law and the force balance equation $\nabla \times \vec{B} = \mu_0 \vec{J}$ and $\nabla \vec{P} = \vec{J} \times \vec{B}$ lead to the Grad-Shafranov equation for the poloidal flux function.

Equilibrium codes solve this equation for the closed flux contours that give the tokamak its good confinement.

- Such codes are used extensively in
  - Experiment design, control of complex shapes is precise
  - On-line data analysis $W(t)$, $\beta(t)$, $\tau_E(t)$
  - Providing the geometry for transport analysis
PLASMA EQUILIBRIUM SHAPE CONTROL IS A HIGHLY DEVELOPED SCIENCE

DIII-D

TCV

PBX-M

JT-60

ASDEX UPGRADE

JT-60U

MFE—Tokamak
SUCCESSFUL METHODS OF HEATING AND CURRENT DRIVE FOR STEADY-STATE HAVE BEEN DEVELOPED

70's / 80's / 90's

Explored heating methods
- Wave coupling
- Fast ion orbits
- NBI deposition

- Multi-MW heating
- Current drive
- Heating to H-mode
- Global rotation
- Measured bootstrap current
- Ray tracing codes
- Fokker-Planck codes

Control of current profile
- Control of MHD activity
- High bootstrap fraction
- Full wave codes

MFE—Tokamak
THE PLASMA'S SELF-GENERATED BOOTSTRAP CURRENT IS THE BASIS FOR MODERN APPROACHES TO STEADY-STATE OPERATION

An element of neoclassical transport theory

\[ J_{bs} \propto \text{local pressure gradient} \]

(Kikuchi, PPCF 37 (1995))
A HIGH PERFORMANCE PLASMA WITH FULL NON–INDUCTIVE CURRENT DRIVE AND 80% BOOTSTRAP FRACTION IN JT–60U

- $H_{89} \approx 3.5$, $HH_{98y2} \approx 2.2$, $\beta_N \approx 2$, $\beta_p \approx 2.9$, $f_{BS} \approx 80\%$ for $6\tau_E$ with full non-inductive CD

- Current profile was largely determined by the bootstrap current, and was nearly stationary

JT 60 also 80% bootstrap fraction
NEUTRAL BEAM HEATING AND CURRENT DRIVE

- Workhorse for high temperature and $\beta$ studies
- Can drive current

Ion Sources $E_b$
- Positive ions $\leq 150$ keV
- Negative ions $\leq 1$ MeV

Drift orbit
Ionization event
Neutral atoms $H^0, D^0, T^0$

TFTR DATA

$E_b$

Major Radius (cm)  Time (s)
Full current drive case in JT–60U (1.3 s)

- \( I_p = 1.5 \text{ MA} \)
- \( B_T = 3.7 \text{ T} \)
- \( \beta_n = 2.4–2.5 \)

<table>
<thead>
<tr>
<th>Component</th>
<th>( I_{CD} )</th>
<th>( E )</th>
<th>( P )</th>
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<tbody>
<tr>
<td>NNBCD</td>
<td>0.6 MA</td>
<td>360 keV</td>
<td>4 MW</td>
</tr>
<tr>
<td>PNBCD</td>
<td>0.3 MA</td>
<td>85 keV</td>
<td>10–18 MW</td>
</tr>
<tr>
<td>Bootstrap</td>
<td>0.8 MA</td>
<td></td>
<td></td>
</tr>
<tr>
<td>ECH</td>
<td>1.7 MA</td>
<td></td>
<td>1.6 MW</td>
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</tbody>
</table>

Efficiency

\[ \eta_{CD} = \frac{n_{ICD} R}{P} = f(T_e) \]

\( \beta_n = 2.4–2.5 \)

\( I_{ICD} = 1.7 \text{ MA} \)

\( \text{Neutralizer} \)

\( \text{Ion source} \)

\( \text{NNBI} \)

\( \text{Co} \) dir.

\( \text{CTR} \) dir.

\( \text{Ion dump} \)

\( \text{MFE—Tokamak} \)
ELECTRON CYCLOTRON HEATING
AND CURRENT DRIVE ($\omega = n\omega_{ce}$)

- Waves propagate in vacuum, so antenna can be far from the plasma

- Inside the plasma the waves propagate up to a critical density (related to the plasma frequency) and are absorbed near the cyclotron resonance or its harmonics

- Damping of EC waves causes diffusion in $V_\perp$ direction. Collisional relaxation on ions generates current through generation of an asymmetric $V_{||}$ distribution

- Calculational tools include ray tracing codes (TORAY, GENRAY, BANDIT-3D) and Fokker-Planck codes (CQL3D, BANDIT-3D, Giruzzi, RELAX, Krivenski, Fukuyama)
MICROWAVE ELECTRON CYCLOTRON HEATING PROVIDES LOCALIZED CURRENT DRIVE

$J_{ECCD} = 35 \text{ kA}$

$J_{ECCD} = 92 \text{ kA}$

$\rho = 0.15$

$\rho = 0.5$

Second Harmonic Resonance

Steerable Antenna

MFE—Tokamak
Fully non-inductive discharges

210 kA sustained in steady state by 2.7 MW co-ECCD

IRn/P ($10^{20}$ A-M$^{-2}$W$^{-1}$) = 7.3 x $10^{-3}$
Lower Hybrid coupling requires $n_{\parallel} > 1$ (Brambilla, SWAN)

Phased array or waveguides

Ray tracing: the accessible waves cross the plasma and can undergo several reflections at the edge before being absorbed.

*Codes by: Cardinali, Bonoli, Ignat, Valeo, Harvey, Takase*

(Figures from Giruzzi)

Electrons heated by LH (PLT)

Damping of LH waves forms a parallel energetic electron tail in the distribution function via Electron Landau Damping. This asymmetry constitutes the non-inductive current (Fisch, Karney)
LHCD SUCCESSFUL IN MANY APPLICATIONS

- Plasma current initiated and ramped up by LHCD

- Plasma current maintained in steady state:
  - JET; 3 MA, 4 s
  - TRIAM-1M; 20 kA, 2 hr

- 2-minute-long discharge at $I_p = 0.8$ MA
- Injected energy = 290 MJ

- 2.0 Hours!
ICRF HEATING AND CURRENT DRIVE ($\omega = n\omega_{ci}$) INVOLVES WAVE EXCITATION, PROPOGATION, ABSORPTION AND MODE CONVERSION.

**Wave Propagation (TORIC, PICES)**

**Alcator C–Mod**

**Coupler**

**Codes**

RANT3D

FELICE

**Absorption Mode Conversion**

AORSA

PICES

TORIC

METS

**IBW Measured PCI Diagnostic**

**MFE—Tokamak**
BASIC ICRF SCHEMES (MINORITY D AND $^3\text{He}$, $2\omega_C\nu_T$) FOR A DT REACTOR HAVE BEEN VERIFIED

- Mode conversion experiments in D – $^3\text{He}$ produced the highest electron heating efficiency in TFTR

- JET: 6 MW ICRF $\rightarrow$ 1.66 MW fusion power

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**Diagram:**

- Pulse No. 43015
- $P_{\text{ICRH}}$ = 6 MW
- $P_{\text{NBI}}$ (Diagnostic)
- $P_{\text{fus}}$ = 1.6 MW
- $Q_{\text{in}}$ = 0.25
- $Q = 0.25$
- $W_{\text{DIA}}$
- $H_{89}$
- $T_{\text{eo}}$ (ECE)
- $T_{\text{io}}$ (CXs)

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MFE—Tokamak
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HEATING AND CURRENT DRIVE CHALLENGES FOR THE NEXT DECADE

<table>
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<th>90s</th>
<th>2000 – 2010</th>
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<td><strong>Control of current profile</strong></td>
<td><strong>Current profile control</strong></td>
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<tr>
<td><strong>Control of MHD activity</strong></td>
<td><strong>Transport barrier control</strong></td>
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<tr>
<td><strong>High bootstrap fraction</strong></td>
<td><strong>Coupling of Fokker-Planck, transport, and stability codes</strong></td>
</tr>
<tr>
<td><strong>Full wave codes</strong></td>
<td><strong>Helicity injection</strong></td>
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<td></td>
<td><strong>Strong alpha heating</strong></td>
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MFE—Tokamak
MHD STABILITY PHYSICS MATURED IN THE 80's AND MOVED TO PROFILE OPTIMIZATION IN THE 90's

<table>
<thead>
<tr>
<th>70's</th>
<th>80's</th>
<th>90's</th>
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<tbody>
<tr>
<td>No heating power</td>
<td>NBI Power</td>
<td>β_T = 13%</td>
</tr>
<tr>
<td>Equilibrium codes</td>
<td>β_T = 5%-10%</td>
<td>Current profile measured</td>
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<tr>
<td>Tearing modes</td>
<td>β-limit scaling</td>
<td>Theory optimization of profiles</td>
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<tr>
<td>Sawteeth</td>
<td>Pressure profile measured</td>
<td>Profile variation and control in experiments</td>
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<td>Current limits</td>
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<td>Wall stabilization</td>
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<td></td>
<td>Ballooning codes</td>
<td>Halo currents</td>
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<tr>
<td></td>
<td>Shaping</td>
<td>Neoclassical tearing</td>
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<td>Second stable edge</td>
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<td></td>
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<td>Advanced Tokamak</td>
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### The Effects of Plasma Instabilities

The effects of plasma instabilities range from loss of the configuration to local transport.

<table>
<thead>
<tr>
<th>Spatial Scale of the Mode</th>
<th>Mode Description</th>
<th>Principal Consequence</th>
</tr>
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<tr>
<td>~a</td>
<td>Global kink modes</td>
<td>Disruptions</td>
</tr>
<tr>
<td>Ideal MHD (low $n$)</td>
<td></td>
<td>$\beta$ and $I_p$ limits</td>
</tr>
<tr>
<td>$\frac{1}{5}a$</td>
<td>Tearing modes</td>
<td>Macroscopic Transport</td>
</tr>
<tr>
<td>Resistive MHD</td>
<td></td>
<td>Profile Modification</td>
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<tr>
<td>Ideal Ballooning ($n \to \infty$)</td>
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<tr>
<td>$\frac{1}{10}a$</td>
<td>Edge Localized Modes</td>
<td>Periodic bursts at the edge</td>
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<tr>
<td>$\rho_i$</td>
<td>Ion Temperature Gradient Modes</td>
<td>Ion Transport</td>
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<td>Drift Waves</td>
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<tr>
<td>$\rho_e$</td>
<td>Electron Temperature Gradient Modes</td>
<td>Electron Transport</td>
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<td>Drift Waves</td>
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IDEAL MHD INSTABILITIES LIMIT THE MAXIMUM BETA

Change in potential energy for a small displacement $\xi$:

$$\delta W = \frac{1}{2} \int dr^3 \left\{ \frac{\delta B^2}{\mu_0} + \frac{B^2}{\mu_0} \nabla \cdot \xi_{\perp} + 2 \xi_{\perp} \cdot \kappa^2 \right. + \gamma p \left| \nabla \cdot \xi \right|^2 - J_{||}(\xi_{\perp} \times \mathbf{b}) \cdot \delta \mathbf{B} - 2(\xi_{\perp} \cdot \nabla p) (\kappa \cdot \xi_{\perp}) \right\}$$

**STABILIZING**

**DESTABILIZING**

Kink Mode: low $n$, global

Ballooning Mode: High $n$, localized in bad curvature region

Pressure-driven Kink (Kink-ballooning) Mode

(J. Freidberg, Ideal MHD; G. Bateman, MHD Instabilities; others)
BETA LIMIT SCALINGS WERE DERIVED THAT FIT WELL EXPERIMENTAL RESULTS

Early work (Doublet III – 1984)

\[ \beta_T (%) \leq 2.8 \frac{I}{a(m)} B_T (T) \]

Theory calculations (1982–1984), Troyon scaling

\[ \beta_T (%) \leq 2.8 \frac{I}{a(m)} B_T (T) \]

Define \( \beta_N = \beta_T/I_aB \)

\[ \beta_p \beta_T = 25 \left( \frac{1 + \kappa^2}{2} \right) \left( \frac{\beta_N}{100} \right)^2 \]

Fusion power \( \beta_T^2 B^4 \)

Bootstrap fraction \( \epsilon_{1/2} \) \( \beta_p \)

Strait, APS Review 1993

DIII–D Data (1988)

\[ \beta_N = 3.5 \]
WALL STABILIZATION, PLASMA SHAPING, AND OPTIMAL PRESSURE AND CURRENT PROFILE MAY DOUBLE THE STABLE OPERATING SPACE OF THE TOKAMAK

Ideal Stability, $n = 1$, GATO

\[ \beta_N (\% \text{-m-T/MA}) \]

\[ \langle J_{\text{TOR}} \rangle \]

\[ \frac{P_0}{\langle P \rangle} = 2.4 \]

$MFE$—Tokamak
IDEAL KINK MODE GROWTH IS SLOWED BY A RESISTIVE WALL AND Responds TO FEEDBACK STABILIZATION

Control-coil

Sensor loops

Feedback Amplifier

DIII–D

Feedback turn-on time

101951 101953 101956

βN

NO-WALL LIMIT
(approx.)

RWM AMPLITUDE

MODE CONTROL + DERIV. GAIN

SMART SHELL + DERIV. GAIN

NO FEEDBACK

PLASMA TOROIDAL ROTATION

ρ ~0.5

Time (ms)

1100 1200 1300 1400 1500

Growth Rate (s⁻¹)

100 200 300 400 500 600 700 800 900 1000

No wall limit

β limit

80% of possible gain

MFE—Tokamak
LOW ASPECT RATIO RAISES $\beta_N$ and $\beta_T$

Record $\beta$ on START (achieved through NB Heating)

New MA Spherical Tori

$\beta_T$ (%) vs $\beta_N$ (UK)

- 1996
- 1997
- 1998

DIII-D, #80108

conventional tokamak

START (UK)

NSTX (U.S.)

$MFE$—$Tokamak$
TEARING MODES

Classical

- Finite resistivity
- Current can diffuse and form clumps — magnetic islands — on rational q flux surfaces
- Driven by \( \nabla J \)
- Growth time 10s of milliseconds

Neoclassical

- \( \nabla P = 0 \) in island removes equilibrium bootstrap current
  - Helical current perturbation amplifies seed island
- Providing auxiliary current drive predicted to stabilize NTM

\[ \beta_p, \text{ min} \]

- Unstable, \( w > 0 \)
  - \( \frac{I_f}{I_p} = 0.02 \)
  - \( I_f/I_p = 0 \)

\[ \text{Width} \]

\[ \beta_p \]
STABILIZATION OF NTMs BY ECCD

ASDEX–Upgrade

ECCD in DIII–D

Similar results from JT–60U

MFE—Tokamak
PRECISE CONTROL NEAR THE $\beta$–LIMIT IS THE KEY TO AVOIDING DISRUPTIONS

1. Need to operate close to stability limits
   - Good control
   - Knowledge of limits
   High performance DIII–D discharge regulated 5% below 2/1 tearing limit for 35 $\tau_E$ (6.3 seconds)

2. Mitigation of disruption consequences massive gas puff or pellets
   - No runaway electrons
   - Reduced halo currents and forces on structural components
   - Reduced heat pulses to the divertor surfaces

$\beta_N$ 104266 $\beta_N$ 104276 DIII–D

Normalized Beta $\beta_N = 2.7$

$m/n = 2/1$

Time (ms)
EDGE LOCALIZED MODES (ELMS) ARE NOW UNDERSTOOD TO BE INTERMEDIATE n KINKS
BOTH ALCATOR C–MOD AND DIII–D HAVE FOUND ELM–FREE REGIMES WITHOUT DENSITY OR IMPURITY ACCUMULATION
Advanced Tokamak stability theory points to states with very broad pressure profiles and hollow current profiles and nearly 100% bootstrap current as perhaps the ultimate potential of the Tokamak.

**ARIES—AT**

- $A = 3.3$
- $\kappa = 2.5$
- $\delta = 0.6$
- $\beta = 14\%$
- $\beta_N = 6$

**ARIES—ST**

- $A = 1.6$
- $\kappa = 3.6$
- $\delta = 0.64$
- $\beta = 56\%$
- $\beta_N = 8.2$

(J. Menard, S. Jardin, J. Manickam)
### STABILITY CHALLENGES FOR THE NEXT DECADE

#### 90's

- $\beta_T = 13\%$
- Current profile measured
- Theory optimization of profiles
- Profile variation and control in experiments
- Wall stabilization
- Halo currents
- Neoclassical tearing
- Second stable edge
- Advanced Tokamak

#### 2000–2010

- Wall stabilized $\beta$-limit
- Bootstrap fraction $\rightarrow 100\%$
- Pressure and current profile control
- Very hollow J(r)
- Broad pressure profiles
- Optimum edge stability
- Feedback stabilization or avoidance of neoclassical tearing
- Disruption mitigation
- 3–D MHD, understand disruptions away from $\beta$-limit
THE 90's HAVE SEEN EXCITING ADVANCES IN CONFINEMENT SCIENCE

70s / 80s / 90s

Global $\tau_E$
Variable results
Linear theory scaling

Reproducible results
(Empirical scaling)
1-D Transport codes
1-D Profile measurements
H-mode edge barrier

Wind tunnel scaling
3-D non-linear turbulence simulations
Comprehensive theory based models
Turbulence measured
Consensus on ion transport
Internal transport barriers
Neoclassical ion transport attained
$E\times B$ shear stabilization

MFE—Tokamak
In the 80’s consistent scaling behavior was seen across many tokamaks implying

- A common underlying transport physics was discoverable
- Multi-machine confinement scaling relations could be constructed, e.g.

\[ \tau_{E, \text{th, ELMy}} = 0.85 \tau_{E, \text{th, ELM-free}} = 0.031 I_p^{1.06} B^{0.32} P^{-0.67} M^{0.41} R^{1.79} n_e^{0.17} \varepsilon^{-0.11} \kappa^{-0.6} \]

- Dimensionless wind tunnel scaling is providing a more fundamental physics basis
STRATEGY TO CALCULATE TRANSPORT

- Theory-based 3D nonlinear simulations being used to benchmark theoretical transport models which are then compared to experiment

- Linear gyrokinetic codes describe local ballooning mode instabilities
  - Long wavelength — ion temperature gradient (ITG) and trapped electron driven
  - Short wavelength — electron temperature gradient (ETG) driven

- Nonlinear flux tube and approximate gyrofluid codes
  - $\rho_i/a \rightarrow 0$
  - Only local ballooning

- Nonlinear codes spanning several hundred gyroradii
  - Finite $\rho_i/a$
  - More time consuming

- ITG/trapped electron flux tube simulations have been used to benchmark gyrofluid local transport code models with comprehensive physics

- International profile data base after 1995 allows systematic and comparative statistical tests of transport code models

Reasonable Agreement
With Experiment

$\text{RMS err} = 26.4\%$
DIII-D, JET, TFTR
L&H-modes

Experimental $W_{th}$ (MJ)
GLF Predicted $W_{th}$ (MJ)

Kinsey, General Atomics
ITER profile database+DIII-D

MFE—Tokamak
RECENT EXCITEMENT
TRANSPORT BARRIERS FORMED BY SHEARED E×B FLOW

Basic Idea: Sheared E×B flow compresses turbulent eddies in the radial direction

$T_e, T_i, n_e$ versus $r/a$

$\chi_e, \chi_i$ versus $r/a$

$\langle \tilde{n}_r \rangle$ versus $\Gamma_r = \tilde{n}_r$
SHEARED E×B FLOW SUPPRESSION OF TURBULENCE UNDERLIES BOTH EDGE AND CORE TRANSPORT BARRIERS

\[ E_r = (Z_i e n_i)^{-1} \nabla P_i - v_{\theta i} B_\phi + v_{\phi i} B_\theta, \]

The \( E \times B \) shearing rate \( \omega_{E \times B} = \left| \frac{(RB_\theta)^2}{B} \frac{\delta}{\delta \psi} \left( \frac{E_r}{RB_\theta} \right) \right| \)

[Hahm and Burrell, Phys. Plasmas 2, 1648]
EQUILIBRIUM SCALE SHEARED $E \times B$ FLOWS CAN QUENCH ITG TRANSPORT IF THE SHEARING RATE EXCEEDS THE MAXIMUM LINEAR GROWTH RATE OF THE TURBULENCE

- ITG simulation of local annulus $160 \rho_S$ wide [R.E. Waltz, et al., Phys. Plasmas 1, 2229 (1994)]
- Application of $E \times B$ shear $\omega_{E \times B} \sim \gamma_{\text{max}}$ breaks up eddies and considerably reduces transport

**Graphical Data**

- **Graph:**
  - Plot of $\chi_\gamma / \chi_\gamma(0)$ versus $\omega_{E \times B} / \gamma_{\text{max}}$ for different values of $s^\gamma$.
  - Lines for $s^\gamma=0.5$, $s^\gamma=1.0$, and $s^\gamma=1.5$.
  - $a/L_T = 3$

**Image:**

- No $E \times B$ flow in the Tokamak.
Recent advance: Small scale sheared poloidal flows can shear apart radial eddies, reducing their radial step size and the transport by an order of magnitude.
ION-NEOCLASSICAL TRANSPORT WITHOUT TURBULENCE, ACROSS ENTIRE PLASMA RADIUS

- Color contour map of fluctuation intensity as function of time from FIR scattering data
  - Higher frequencies correspond to core, low to edge

- Total ion thermal diffusivity at time of peak performance
  - $H = 4.5 \quad W = 4.2$ MJ
  - $\beta = 6.7\% \quad \beta_N = 4.0$

- Neoclassical $NCS H$-mode edge Experiment 87977

\[
\chi_{i}^{\text{tot}} = \frac{Q_i}{n_i} \nabla T_i
\]
CONFINEMENT CHALLENGES
FOR THE NEXT DECADE

90s / 2000 – 2010

Theory based calculations of transport barrier formation
Control ITB radius and gradient
Understand electron transport
electron turbulence diagnostics
First-principles diffusion coefficients
Momentum and particle transport
Nonlinear turbulence simulations
with both electrons an ions
Complete dimensionless scaling
Access conditions for H–mode
Edge pedestal structure

Wind tunnel scaling
3–D non-linear turbulence simulations
Comprehensive theory based models
Turbulence measured
Concensus on ion transport
Internal transport barriers
Neoclassical ion transport attained
E×B shear stabilization

MFE—Tokamak
THE SCIENCE OF POWER AND PARTICLE EXHAUST LEAPED FORWARD IN THE 90's

70s

Limiters
Impure, radiating core plasmas

80s

Divertors
Clean plasmas
Low core radiation

90s

He ash and fuel exhaust
Radiative divertor plasmas
Recombination
2–D measurements
2–D fluid codes
THE JET DIVERTOR IS TYPICAL OF TOKAMAKS TODAY

Axisymmetric lower single null with graphite tiles to handle high heat flux
THE PHYSICS ELEMENTS THAT ARE DOMINANT IN THE DIVERTOR PROBLEM ARE NOW INCORPORATED IN 2-D CODES

- Strong parallel transport
  Fluid drifts
  Actual flux surface geometry
- Non-equilibrium radiation rates
  2-D flow patterns
- Neutral recycling
  Recombination
  Detailed divertor structures
- Erosion of surfaces
  Ablation during intense heat pulses
AN EXAMPLE OF EXCELLENT AGREEMENT BETWEEN B2-E IRENE CALCULATED AND MEASURED RADIATION DISTRIBUTIONS

ASDEX–UPGRADE

Bolometer line integral \([\text{MW/m}^2]\)

Distance Separatrix-Bolometer [m]

Radiation losses [MW/m]

Excellent agreement
RECOMBINING DIVERTOR PLASMAS DISCOVERED

Scaling of Lyman Series Line Intensities
Shows When the Upper Levels of the Lines
Are Populated by Recombination

Alcator C–Mod
- $T_e \sim 1$ eV at divertor plate (probes)
- $T_e 0.4-0.6$ eV in divertor plasma (spect.)

Low Electron Temperature

Modeling

Data

UEDGE

Alcator C–Mod

Experimental Spectra

$Z(m)$

$R (m)$

$\text{Log} [T_e (eV)]$

$\text{UEDGE}$

$MFE—Tokamak$
DIVERTOR DETACHMENT IN ALCATOR C-MOD

Growth of the recombination region

0.5 sec
0.6 sec
0.7 sec
0.8 sec

Ip(MA)
\bar{n}_e (10^{20} \text{ m}^{-3})
\frac{n_e T_e}{\text{Pa}}
G_{\text{plate}} (\text{m}^{-2}\text{s}^{-1})

Time (sec)

MFE—Tokamak
DIVERTOR DETACHMENT IN ALCATOR C-MOD

Growth of the recombination region

0.5 sec

0.6 sec

0.7 sec

0.8 sec

Ip(MA)

n_e (10^{20} m^{-3})

n_eT_e (Pa)

G_{plate} (m^{-2}s^{-1})

Time (sec)

kW/m^{3}

5.2

4.6

3.9

3.3

2.6

2.0

1.3

0.7

Divertor Detachment

upstream

div.

MFE—Tokamak
EXHAUST OF FUEL AND HELIUM ASH DEMONSTRATED

- Plasma density regulated constant by gas fueling and divertor pumping

- Divertor pumping prevents accumulation of helium ash (injected by neutral beams)

- Pumpout rate of helium adequate for fusion reactor
CODES TO CALCULATE THE EROSION OF DIVERTOR SURFACES ARE BEING TESTED AGAINST EXPERIMENTAL DATA

- Erosion during normal operation
  - REDEP matches DIII–D data for carbon
  - Treats physical and chemical sputtering and 2-D material transport
  - Codes: REDEP, WBC, ERO, DIVIMP, MCI, IMPMC

- Erosion during ablative heat pulses
  - 2-D codes treat vaporization melting vapor shield formation, radiation transport
  - Tested against plasma gas experiments
  - Codes: WURZ, LANGYEL, HASSANEIN

Disruption Erosion in the Divertor

Calculation by Wuerz for 1.5 m long divertor slot (ITER ~ 1 m)

OSP Erosion and Modeling (DIII–D)

Erosion rate (cm / exp-year)

Gross

Net

$Y_{\text{eff}}$

Experiment

Disruption Erosion in the Divertor

power density profile of hot plasma

Calculation by Wuerz for 1.5 m long divertor slot (ITER ~ 1 m)
POWER AND PARTICLE EXHAUST CHALLENGES FOR THE NEXT DECADE

90s

veys

Helium ash and fuel exhaust
Radiative divertor plasmas
Recombination
2-D measurements
2-D fluid codes

2000 – 2010

Optimal plasma edge shape
2-D SOL/divertor flows
Helium and fuel exhaust in AT regimes
Use of copious core radiation
Understanding erosion and redeposition (T inventory)
Modeling and mitigating disruption erosion

MFETokamak
<table>
<thead>
<tr>
<th>Area</th>
<th>Status</th>
<th>Advanced Tokamak Challenge</th>
<th>Promise</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heating</td>
<td>Understood, technology developed</td>
<td>Pressure profile control, alpha heating</td>
<td>Burning plasmas</td>
</tr>
<tr>
<td>Current drive</td>
<td>Physics understood</td>
<td>High bootstrap fraction, local profile control</td>
<td>Steady-state bootstrap fraction (\rightarrow 100%)</td>
</tr>
<tr>
<td>Stability</td>
<td>Operating space understood, predictable</td>
<td>Wall stabilization</td>
<td>Double the stable operating space</td>
</tr>
<tr>
<td>Confinement</td>
<td>Closing in on ability to calculate</td>
<td>Transport barrier control</td>
<td>Near neoclassical ion confinement</td>
</tr>
<tr>
<td>Power and particle control</td>
<td>Major physics elements calculable</td>
<td>Low density divertors compatible with current drive</td>
<td>Steady-state with low surface erosion</td>
</tr>
</tbody>
</table>
WE ARE READY TO TAKE UP BURNING PLASMA AND STEADY-STATE ISSUES

Burning Plasma Issues

- DT plasma properties
- Alpha confinement
- Alpha ash exhaust
- Remote maintenance
- Alpha driven instabilities
- Self-heated profiles
- High gain burn control

Steady State Issues

- High bootstrap fractions (AT)
- Steady-state magnets
- Steady-state current drive
- Tritium inventory
- Hour long pulses
- Resolve disruption issue
- Blanket development
- Low activation materials
- Tritium breeding
- Month long operation
- First electric output
**ALPHA HEATING OBSERVED**

**TFTR**

D–T, $P_{fus} \approx 5$ MW
(6 Plasmas)

D Only
(17 Plasmas)

**JET**

**Transonic Heating Observed**

$P_{\alpha}$ (MW)

$T_e(\beta)$ (keV)

$T_e(D) - T_e(\beta)$ (keV)

Major Radius (m)

$W_{\text{Dia}}$ (MJ)

$W_{\text{th}}$ (MJ)

$\tau_{E, \text{th}}$ (s)

$n/(n_T + n_D)$

MFE—Tokamak
CLASSICAL ALPHA CONFINEMENT VERIFIED (TFTR)

First orbit loss (3% at 2.5 MA)

Radial transport

Slowing down spectrum

Double Charge Exchange Technique
He^{++} + Li^{+} \rightarrow He^{0} + Li^{3+}

MFE—Tokamak
219-00/rs
THEORETICALLY PREDICTED ALFVÉN EIGENMODES WERE OBSERVED

AE Modes excited in JET by ICRH minority ions
- TAE 200 kHz
- EAE 400 kHz

Substantial fast ion losses in TFTR from TAE modes driven by neutral beam or ICRF tail ions

AE Modes absent in highest fusion power cases

- NBI driven TAE
- ICRF driven TAE

MFE—Tokamak
OBSERVED \( \alpha \)-DRIVEN TAES CONSISTENT WITH FULL LINEAR THEORY

- Calculations with NOVA-K code
- Weak shear and high \( q(0) \) are destabilizing
- Weak or reverse shear plasmas in a reactor may be unstable to high-n TAEs

G. Fu, R. Nazikian
COPIOUS FUSION POWER HAS BEEN PRODUCED

**TFTR D-T Campaign**
- 10.7 MW
- $P_{\text{FUSION}}/P_{\text{HEAT}} = 0.27$
- 1.55 GJ fusion energy

**JET D-T Campaign**
- 16 MW
- $P_{\text{FUSION}}/P_{\text{HEAT}} = 0.6$
- 0.68 GJ fusion energy

---

**Pulse No: 42976 T ransp Y930 4.2MA/3.6T (IAEAML)**

- $P_{\text{IN}}$ (MW)
- $P_{\text{fus}}$ (MW)
- $T_{\text{io}}$ (keV)
- $T_{\text{eo}}$ (keV)

**FUSION POWER (MW)**
- 0
- 5
- 10

**Time (s)**
- 3.5
- 4.0

**MFE—Tokamak**
THE ITER-FEAT MACHINE

- Cut through cryostat, TF and PF coils, Vacuum Vessel, Blanket and Divertor
Fusion Ignition Research Experiment (FIRE)

Design Goals
- R = 2.0 m, a = 0.525 m
- B = 10 T, (12T) *
- $W_{mag} = 3.8$ GJ, (5.5T) *
- $I_p = 6.5$ MA, (7.7 MA) *
- $P_{\text{alpha}} > P_{\text{aux}}$, $P_{\text{fusion}} < 200$ MW
- Burn Time $\approx 18.5$ s ($\approx 12$ s) *
- Tokamak Cost $\leq$ $0.3$B
  Base Project Cost $\leq$ $1$B

* Higher Field Mode

Attain, explore, understand and optimize fusion-dominated plasmas that will provide knowledge for attractive MFE systems


A Proposal of JT-60 Modification

- To conduct researches on steady state operation of tokamaks
- To contribute to the ITER operation
- Under discussions at the Fusion Council

**Main Parameter**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>JT-60UA after modification</td>
<td></td>
</tr>
<tr>
<td>Cryostat</td>
<td></td>
</tr>
<tr>
<td>Poloidal Field Coil (SC)</td>
<td></td>
</tr>
<tr>
<td>Toroidal Field Coil (SC)</td>
<td></td>
</tr>
<tr>
<td>NBI</td>
<td></td>
</tr>
</tbody>
</table>

- To contribute to the ITER operation
- Under discussions at the Fusion Council
EXTENDING THE ADVANCED TOKAMAK: KSTAR

- 20–300 s pulse length (S/C technology)
- $B = 3.5 \text{ T}$, $I = 2 \text{ MA}$
- $R = 1.8 \text{ m}$, $a = 0.5 \text{ m}$
- Double-null divertor, $\kappa = 2$, $\delta = 0.8$
- 16-27 MW profile control: (neutral beam, ion cyclotron, lower hybrid)
HT–7U ADVANCED TOKAMAK – HAFeI CHINA
INSTITUTE OF PLASMA PHYSICS ACADEMIA SINICA

Construction: Approved
Completion: mid 2003

R/a = 1.7/0.4 m
B = 3.5 T
I = 1 MA
κ = 1.6–2.0
δ = 0.4–0.8
THE ADVANCED TOKAMAK LEADS TO AN ATTRACTIVE FUSION POWER PLANT

- The U.S. ARIES — RS system study
- The Japanese SSTR system study

Attractive features
- Competitive cost-of-electricity
- Steady-state operation
- Maintainability
- Low-level waste
- Public and worker safety

<table>
<thead>
<tr>
<th></th>
<th>Conventional</th>
<th>AT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Size, major radius (m)</td>
<td>8</td>
<td>5</td>
</tr>
<tr>
<td>COE $c/kWhr$</td>
<td>~13</td>
<td>~7</td>
</tr>
<tr>
<td>Power cycle</td>
<td>Pulsed</td>
<td>Steady state</td>
</tr>
</tbody>
</table>
SUMMARY

- Research in the tokamak has greatly advanced fusion energy science

- Tokamak research has shown fusion energy is feasible in the laboratory

- The tokamak is scientifically and technically ready to proceed to burning plasma and/or steady-state next steps

- Advanced Tokamak research seeks to find the ultimate potential of the tokamak as a magnetic confinement configuration
  - Anticipated results point to practical and attractive fusion energy