REVIEW OF TOKAMAK RESEARCH

by

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Presented at
28th European Physical Society Conference on Controlled Fusion and Plasma Physics
Madeira, Portugal

June 22, 2001
RESEARCH RESULTS FROM

Alcator C–mod  JET  TCV
ASDEX Upgrade  JFT–2M  TdeV
Compass–D  JT 60U  TEXTOR
DIII–D  MAST  TEXT
ET  NSTX  TFTR
FTU  PBX–M  TORE–SUPRA
HBT–EP  PLT  TRIAM–1M
HIT  START  T–10

SPECIAL THANKS FOR DIRECT CONTRIBUTIONS

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

MFE–Tokamak
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

\[ J(r) \propto T^{3/2} \]

\[ (J \propto T^{3/2}) \]

\[ (q = 4) \]
TOKAMAKS HAVE MADE EXCELLENT PROGRESS IN FUSION POWER
PLASMA EQUILIBRIUM THEORY
IS WELL UNDERSTOOD AND EXTENSIVELY USED

- Ampere's Law and the force balance equation \( \nabla \times \mathbf{B} = \mu_0 \mathbf{J} \) and \( \nabla \mathbf{P} = \mathbf{J} \times \mathbf{B} \) 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
SUCCESSFUL METHODS OF HEATING AND CURRENT DRIVE FOR STEADY-STATE HAVE BEEN DEVELOPED

<table>
<thead>
<tr>
<th>70's</th>
<th>80's</th>
<th>90's</th>
</tr>
</thead>
<tbody>
<tr>
<td>Explored heating methods</td>
<td>Multi-MW heating</td>
<td>Control of current profile</td>
</tr>
<tr>
<td>Wave coupling</td>
<td>Current drive</td>
<td>Control of MHD activity</td>
</tr>
<tr>
<td>Fast ion orbits</td>
<td>Heating to H-mode</td>
<td>High bootstrap fraction</td>
</tr>
<tr>
<td>NBI deposition</td>
<td>Global rotation</td>
<td>Full wave codes</td>
</tr>
<tr>
<td></td>
<td>Measured bootstrap current</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Ray tracing codes</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Fokker-Planck codes</td>
<td></td>
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</tbody>
</table>
THE PLASMA'S SELF-GENERATED BOOTSTRAP CURRENT IS THE BASIS FOR MODERN APPROACHES TO STEADY-STATE OPERATION

- An element of neoclassical transport theory
- $J_{bs} \propto$ local pressure gradient

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

- $H_{89} \sim 3.5$, $HH_{98y2} \sim 2.2$, $\beta_N \sim 2$, $\beta_P \sim 2.9$, $f_{BS} \sim 80\%$ 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

Neutral atoms $H^0, D^0, T^0$
NEUTRAL BEAM CURRENT DRIVE IN ACCORD WITH THEORY

Full current drive case in JT–60U (1.3 s)

- $I_p = 1.5$ MA
- $B_T = 3.7$ T
- $HH = 1.3–1.4$
- $\beta_N = 2.4–2.5$

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

$1.7$ MA

Efficiency

$\eta_{CD} = n_{ICDR}/P = f(Te)$

$\beta_N = 2.4–2.5$
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 \( V_\parallel \) 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

$\rho = 0.15$

$\rho = 0.5$

Second Harmonic Resonance

Steerable Antenna

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

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

$\rho = 0.5$

$\rho = 0.15$

MFE—Tokamak
Fully non-inductive discharges

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

\[ \text{IRn/P (}10^{20} \text{ A-M}^{-2}\text{-W}^{-1}) = 7.3 \times 10^{-3} \]

- Plasma current (kA)
- EC power (MW)
- Loop voltage (V)
- Current in Ohmic coil (kA)
- Line-averaged density (10^{19} m^{-3})
- Peak temperature (keV)
- Internal inductance
LOWER HYBRID HEATING AND CURRENT DRIVE (ω_{ci} < ω < ω_{ce})

Tore Supra

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, PROPAGATION, ABSORPTION AND MODE CONVERSION

\[ \omega = n\Omega_{ci} \]

**ICRF Antennas**

**Vacuum Vessel**

**Major Radius of Torus, R (m)**

**plasma**

**limiter**

**IBW**

**fast wave**

**Wave Propagation**

\( (ALCYON, PICES TORIC) \)

**Absorption Mode Conversion**

\( n_{||} = k_{\perp} \)

**Alcator C–Mod**

**Coupled power**

**Coupled**

**power**

**IBW Measured PCI Diagnostic**

**RF Frequency, MHz**

IBW

\[ 80.47 \]

\[ 80.50 \]

\[ 80.53 \]

**Real Ex**

**AORSA**

**PICES**

**METS**

**Tore Supra Coupler**

**Codes**

**RANT3D**

**FELICE**
BASIC ICRF SCHEMES (MINORITY D AND $^3$He, $2\omega\text{CT}$) FOR A DT REACTOR HAVE BEEN VERIFIED

- Mode conversion experiments in $D - ^3He$ produced the highest electron heating efficiency in TFTR

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

![Diagram showing the evolution of electron temperature ($T_e$) and fusion power ($P_{fus}$) over time]
HEATING AND CURRENT DRIVE CHALLENGES FOR THE NEXT DECADE

90s

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

2000 – 2010

Current profile control
Transport barrier control
Coupling of Fokker-Planck, transport, and stability codes
Helicity injection
Strong alpha heating
MHD STABILITY PHYSICS MATUR ED IN THE 80's AND MOV ED TO PROFILE OPTIMIZATION IN THE 90's

70's        /        80's        /        90's

No heating power
Equilibrium codes
Tearing modes
Sawteeth
Current limits

NBI Power
$\beta_T = 5\%-10\%$
$\beta$-limit scaling
Pressure profile measured
Kink codes
Ballooning codes
Shaping

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

MFE—Tokamak
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>
</thead>
<tbody>
<tr>
<td>∼a</td>
<td>Global kink modes</td>
<td>Disruptions</td>
</tr>
<tr>
<td></td>
<td>Ideal MHD (low n)</td>
<td>β and Ip limits</td>
</tr>
<tr>
<td>∼1/5 a</td>
<td>Tearing modes</td>
<td>Macroscopic Transport</td>
</tr>
<tr>
<td></td>
<td>Resistive MHD</td>
<td>Profile Modification</td>
</tr>
<tr>
<td></td>
<td>Ideal Ballooning (n → ∞)</td>
<td></td>
</tr>
<tr>
<td>∼1/10 a</td>
<td>Edge Localized Modes</td>
<td>Periodic bursts at the edge</td>
</tr>
<tr>
<td>ρ_i</td>
<td>Ion Temperature Gradient Modes</td>
<td>Ion Transport</td>
</tr>
<tr>
<td></td>
<td>Drift Waves</td>
<td></td>
</tr>
<tr>
<td>ρ_e</td>
<td>Electron Temperature Gradient Modes</td>
<td>Electron Transport</td>
</tr>
<tr>
<td></td>
<td>Drift Waves</td>
<td></td>
</tr>
</tbody>
</table>
IDEAL MHD INSTABILITIES LIMIT THE MAXIMUM BETA

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

$$\delta W = \frac{1}{2} \int dr^3 \left\{ \frac{\delta B^2}{\mu_0} + \frac{B^2}{\mu_0} \, \nabla \cdot \xi + 2 \xi \cdot \kappa^2 + \gamma p \nabla \cdot \xi + \frac{1}{2} (\xi \cdot \nabla p) (\kappa \cdot \xi) \right\}$$

field line bending
magnetic field compression
fluid compression
parallel current
pressure gradient

STABILIZING

Kink Mode: low n, global

Pressure-driven Kink (Kink-ballooning) Mode

DESTABILIZING

Ballooning Mode: High n, localized in bad curvature region

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

Theory calculations (1982–1984), Troyon & Sykes

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

Define \( \beta_N = \frac{\beta_T}{I/aB} \)

2.8 = Troyon-kink
4.4 = Sykes-balloon

Fusion power \( \beta_T B^4 \)

Bootstrap fraction \( c \epsilon^{1/2} \beta_p \)

"MFE–Tokamak"
WALL STABILIZATION, PLASMA SHAPING, AND OPTIMAL PRESSURE AND CURRENT PROFILE MAY DOUBLE THE STABLE OPERATING SPACE OF THE TOKAMAK

Ideal Stability, \( n = 1 \), GATO

\[ \beta_N (\% \cdot m \cdot T/MA) \]

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

\[ P_0 / \langle P \rangle = 2.4 \]

MFE—Tokamak
IDEAL KINK MODE GROWTH IS SLOWED BY A RESISTIVE WALL AND RESPONDS TO FEEDBACK STABILIZATION.
LOW ASPECT RATIO RAISES $\beta_N$ and $\beta_T$

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

New MA Spherical Tori

$\beta_N \sim 3.5$

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

![Graph showing unstable region with $\dot{w} > 0$ and stable region with $\dot{w} < 0$]
STABILIZATION OF NTMs BY ECCD

ASDEX–Upgrade

Similar results from JT–60U
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
EDGE LOCALIZED MODES (ELMS) ARE NOW UNDERSTOOD TO BE INTERMEDIATE n KINKS.

DIII-D

\[ \frac{2\alpha_e}{\alpha_{CRIT}} \]

n = 5

Calculated Stability Threshold

\[ \nabla P \]

Limit

Type I ELMs

Last 20% of ELM Cycle

GATO

n=5

EDGE KINK

MFE—Tokamak
BOTH ALCATOR C–MOD AND DIII–D HAVE FOUND ELM–FREE REGIMES WITHOUT DENSITY OR IMPURITY ACCUMULATION

C–Mod EDA H–mode

DIII–D QH–mode

Reflectometer

CH 60 GHz

f (kHz)

Time (s)

Reflectometer

Density

Energy (MJ)

Div. Dα

Rad. Power (MW)

Time (ms)

Plasma Density

Density Fluctuations

0.70 0.80 0.90 1.00 1.10 1.20

0 1 2 3 4

0 1 2 3 4

0 100 200 300 400

0 100 200 300 400

0 100 200 300 400

0 100 200 300 400
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

<table>
<thead>
<tr>
<th>90's</th>
<th>2000–2010</th>
</tr>
</thead>
<tbody>
<tr>
<td>( \beta_T = 13% )</td>
<td>Wall stabilized ( \beta )-limit</td>
</tr>
<tr>
<td>Current profile measured</td>
<td>Bootstrap fraction ( \rightarrow 100% )</td>
</tr>
<tr>
<td>Theory optimization of profiles</td>
<td>Pressure and current profile control</td>
</tr>
<tr>
<td>Profile variation and control in experiments</td>
<td>Very hollow ( J(r) )</td>
</tr>
<tr>
<td>Wall stabilization</td>
<td>Broad pressure profiles</td>
</tr>
<tr>
<td>Halo currents</td>
<td>Optimum edge stability</td>
</tr>
<tr>
<td>Neoclassical tearing</td>
<td>Feedback stabilization or avoidance of neoclassical tearing</td>
</tr>
<tr>
<td>Second stable edge</td>
<td>Disruption mitigation</td>
</tr>
<tr>
<td>Advanced Tokamak</td>
<td>3–D MHD, understand disruptions away from ( \beta )-limit</td>
</tr>
</tbody>
</table>

MFE—Tokamak
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
Concensus on ion transport
Internal transport barriers
Neoclassical ion transport attained
$E\times B$ shear stabilization

MFE—Tokamak
TOKAMAK CONFINEMENT PROVED (EMPIRICALLY) PREDICTABLE

- 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.

  \[ \tau_{E, \text{th}, \text{ELMy}} = 0.85 \tau_{E, \text{th}, \text{ELM-free}} \]

  \[ = 0.031 I_p^{1.06} B^{0.32} \]

  \[ p^{-0.67} M^{0.41} R^{1.79} n_e^{0.17} \varepsilon^{-0.11} \kappa^{-0.6} \]

- 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
  - $\rho_i/a \to 0$
  - 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

![Graph showing Reasonable Agreement With Experiment](image)
RECENT EXCITEMENT
TRANSPORT BARRIERS FORMED BY SHEARED E×B FLOW

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

Contour plots for $\Gamma_r = n\tilde{v}_r$
SHEARED $E \times B$ 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

![Graph and image showing turbulence suppression](image)

\[ \frac{\chi_i}{\chi_i(0)} \quad \omega_{E \times B} / \gamma_{\text{max}} \]

\[ a/L_T = 3 \]

No $E \times B$ flow
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
  - $H = 4.5$  $W = 4.2$ MJ
  - $\beta = 6.7\%$  $\beta_N = 4.0$

Neutron Rate (x10$^{16}$/s)

$\chi_i^{\text{tot}} = Q_i/n_i \nabla T_i$

219–00/RDS/wj
CONFINEMENT CHALLENGES FOR THE NEXT DECADE

90s / 2000 – 2010

- 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
- $E \times B$ shear stabilization

- 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
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

MFE—Tokamak
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

Excellent agreement
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**

**UEDGE**

**Experimental Spectra**

**Recombining Divertor Plasmas Discovered**
DIVERTOR DETACHMENT IN ALCATOR C-MOD

Growth of the recombination region

Divertor Detachment

\[ I_p (MA) \]
\[ n_e (10^{20} \text{ m}^{-3}) \]
\[ n_e T_e (\text{Pa}) \]
\[ G_{\text{plate}} (\text{m}^{-2}\text{s}^{-1}) \]

Time (sec)

0.5 sec
0.6 sec
0.7 sec
0.8 sec

0.0 0.5 1.0
0.4 0.6 0.8
-0.6 -0.5 -0.4
-0.3
-0.4
-0.5
-0.6

kW/\text{m}^3
5.2
4.6
3.9
3.3
2.6
2.0
1.3
0.7

MFE—Tokamak
GOOD CONFINEMENT AT THE DENSITY LIMIT REALIZED

TEXTOR RI–MODE

TECTOR # 75679

I_p = 350 kA  B_T = 2.25 T

ELM-free H-Mode Quality

Greenwald Density Limit

P_ICRH: 1.3 MW

P Ni: 1.0 MW

Ne-VIII intensity (a.u.) (γ = 60 %)

P_rad/P_tot

n / n GW

D_2 (10^{23} s^{-1})

160 x \tau_E

\bar{n}_e (10^{20} m^{-3})

JET

Pulse No: 52014 2.5 MA/2.7 T, P_{NB} = 14 MW

β_N ~ 2

H_{98(y,2)} ~ 1

P_{NBI} ~ 14 MW

\text{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

DIII–D Divertor 2000

JT–60U

MFE—Tokamak
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

Experiment

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

90s / 2000 – 2010

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

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
<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

Alpha Issues

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

More Gain

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 Graph](Image)

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

**D Only (17 Plasmas)**

**JET Graph**

- **Measurement**
- **TRANSNP Prediction**

**TFTR**

- **J. Strachan**
- **G. Taylor**
- **J. Strachan**

**JET**

- **D–T Pulses (T concentrations)**
- **(60%)**
- **(75%)**
- **(50%)**
- **(92%)**
- **(0%)**

**TFTR**

- **α Heating Observed**

**JET**

- **3.8MA/3.4T/10.5MW**

<table>
<thead>
<tr>
<th>$W_{\text{DIA}}$ (MJ)</th>
<th>$W_{\text{th}}$ (MJ)</th>
<th>$\tau_{E,\text{th}}$ (s)</th>
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<tr>
<td>11</td>
<td>10</td>
<td>1.5</td>
</tr>
<tr>
<td>9</td>
<td>9</td>
<td>1.5</td>
</tr>
<tr>
<td>8</td>
<td>8</td>
<td>1.5</td>
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</table>

**MFE—Tokamak**
CLASSICAL ALPHA CONFINEMENT VERIFIED (TFTR)

First orbit loss (3% at 2.5 MA)

Radial transport

Data

\( D_{\alpha} = 0 \) (w/error band)

\( D_{\alpha} = 0.03 \, m^2/s \)

Intensity \((10^{10} \, \text{ph/s-cm}^2\text{-ster})\)

\( E_{\alpha} = 0.15-0.6 \, \text{MeV} \)

\( r/a \approx 0 \)

Double Charge Exchange Technique

\( \text{He}^{++} + \text{Li}^{+} \Rightarrow \text{He}^{0} + \text{Li}^{3+} \)

Slowing down spectrum

Normalization

\( dv/dE \) (a.u.)

\( r = 0 \)

\( \text{TRANSFP/FPPT} \)

\( \text{Ga} \)

\( \text{IOFFE} \)

\( \text{PPPL} \)

MFE—Tokamak

219-00/rs

ZWEBEN, DARROW

\( r \approx 0 \)

E_{\alpha} = 0.15-0.6 \, \text{MeV} \)

Normalized here

First orbit loss model
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

- NBI driven TAE
- ICRF driven TAE

AE Modes absent in highest fusion power cases
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

TFTR

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
- \( \frac{P_{\text{FUSION}}}{P_{\text{HEAT}}} = 0.27 \)
- 1.55 GJ fusion energy

**JET D-T Campaign**
- 16 MW
- \( \frac{P_{\text{FUSION}}}{P_{\text{HEAT}}} = 0.6 \)
- 0.68 GJ fusion energy
Cut through cryostat, TF and PF coils, Vacuum Vessel, Blanket and Divertor
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

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Main Parameter

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value 1</th>
<th>Value 2</th>
<th>Value 3</th>
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MFE—Tokamak

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219-00/jy
EXTENDING THE ADVANCED Tokamak: KSTAR

- 20–300 s pulse length (S/C technology)
- $B = 3.5 \, T$, $I = 2 \, MA$
- $R = 1.8 \, m$, $a = 0.5 \, m$
- 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

ASIPP

HT–7

MFE—Tokamak
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>
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<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>

MFE–Tokamaks
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