

# Summary S/1-2 Experiments

**S** Stability,

**W**ave-plasma interactions, **C**urrent drive,  
**H**eating, **E**nergetic particles,  
**P**lasma-material interactions, **D**ivertors,  
**L**imiters, **SOL**



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**322-6 Oroshicho, Toki 509-5292, Japan**  
**National Institutes of Natural Sciences**

# General Topics

## 1. ITER oriented

(1) ELM mitigation (**S**)

(2) Divertor armor strategy (**D**)  
PMI, Tungsten

(3) Current drive and heating (**W**)

(4) Disruption mitigation (**S**)

## 2. ITER/DEMO oriented

(1) Steady state (**W,S**)

(2) High  $\beta$  (**S**)

NTM, RWM

(3) PMI & SOL/DIV physics (**D**)

(4) TAE (**W, S**)

(5) Heating (**W**)

EBW etc.

Primarily ITER oriented

High Priority Technical Issues Identified at STAC-2

**1. Vertical Stability**

**2. Shape Control / Poloidal Field Coils**

**3. Flux Swing in Ohmic Operation and CS**

**4. ELM Control**

**5. Remote Handling**

**6. Blanket Manifold Remote Handling**

**7. Divertor Armour Strategy**

**8. Capacity of 17 MA Discharge**

**9. Cold Coil Test**

**10. Vacuum Vessel / Blanket Loading Condition Test**

**11. Blanket Modules Strategy**

**12. Hot Cell Design**

**13. Heating Current Drive Strategy, Diagnostics and Research Plan**

# World Wide Fusion Research by Tokamaks, Helicals, STs, RFPs etc.

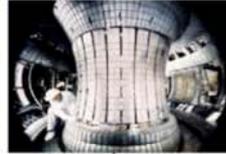
**MAST**  
EURATOM/UKAEA,  
UK



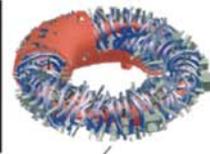
**JET**  
European Fusion Development  
Agreement, EU



**ASDEX-Upgrade**  
Max Planck Institute for Plasma  
Physics, Germany



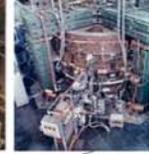
**W7-X**  
Max Planck Institute for Plasma  
Physics, Germany



**Globus-M**  
Ioffe Institute,  
Russia



**T-10**



**LHD**  
National Institute for Fusion  
Science, Japan



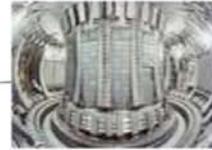
**JT-60U**  
Japan Atomic Energy Agency,  
Japan



**JT-60SA**  
Japan Atomic Energy Agency,  
Japan



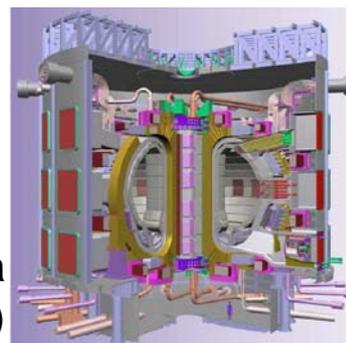
**Alcator C-mod**  
Massachusetts Institute of Technology,  
USA



**NSTX**  
Princeton University,  
USA



**DIII-D**  
General Atomics,  
USA



**TEXTOR**  
Forschungszentrum Jülich,  
Germany



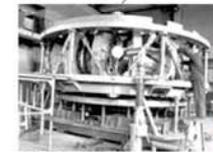
**TCV**  
Plasma Physics Research Centre,  
Switzerland



**TJ-II**  
CIEMAT, Spain



**Tore Supra**  
CEA, France



**URAGAN**



**RFX**  
CNR, Italy



**FTU**  
ENEA, Italy



**H-1NF Heliac**  
Australian National University,  
Australia



**HL-2A**  
Southwestern Institute of Physics,  
China



**EAST**  
Institute of Plasma Physics,  
China



**HT-7**  
Institute of Plasma Physics,  
China



**KSTAR**  
National Fusion Research  
Center, Korea



SST-1, IPR, India

Superconducting tokamaks / helical systems

Tore Supra(1988), LHD(1998), EAST(2006), KSTAR(2008)  
(ITER, JT-60SA, SST1, W7-X)

Burning Plasma  
ITER (2018)

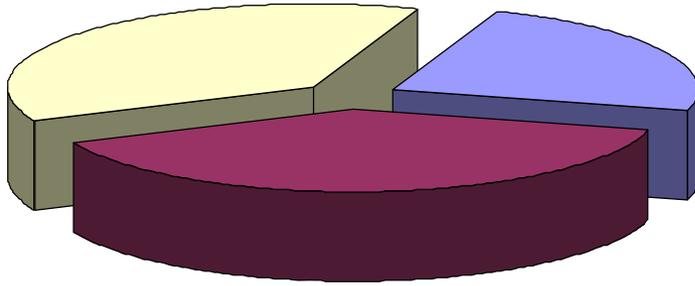
# Statistics

## Comparison of Topics by Category

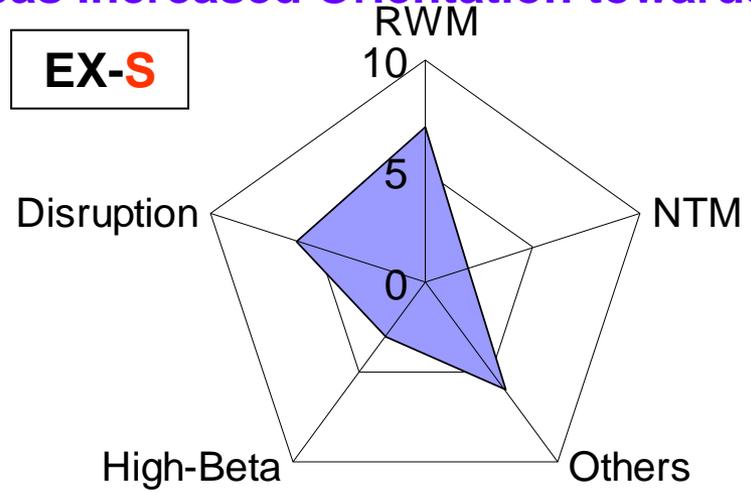
Typical Tendency of Research in these Areas Increased Orientation towards ITER

EX-D (38)

EX-S (24)

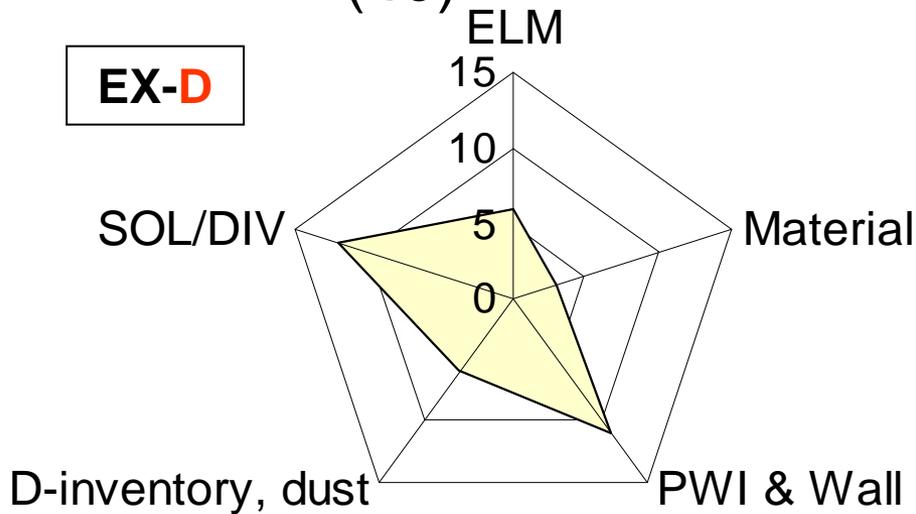


EX-S

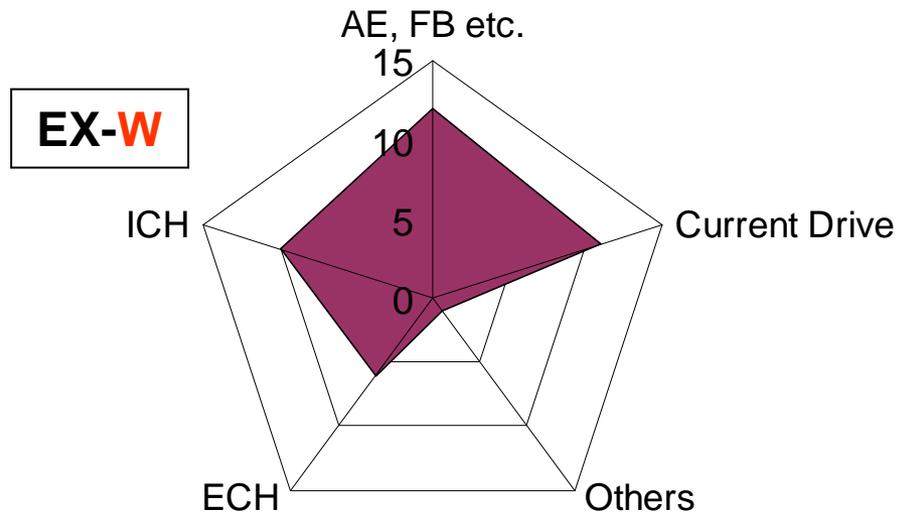


EX-W (40)

EX-D



EX-W

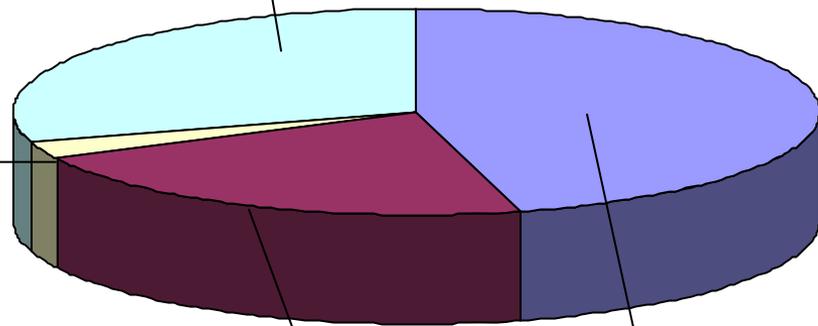


# Comparison by Device Concept

EX-W

RFP

Helical



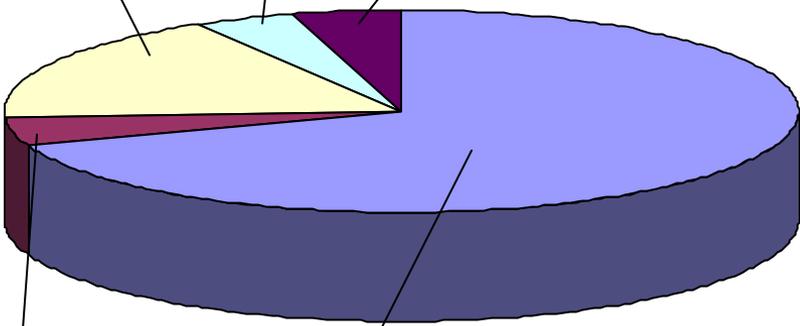
EX-S

EX-D

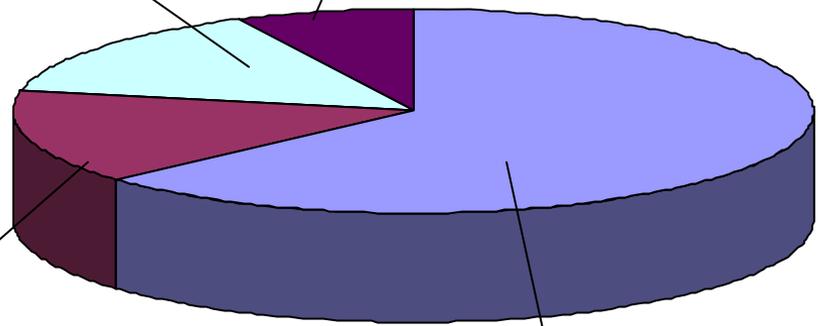
ST

Tokamak

RFP Helical Others



Helical Others



ST

Tokamak

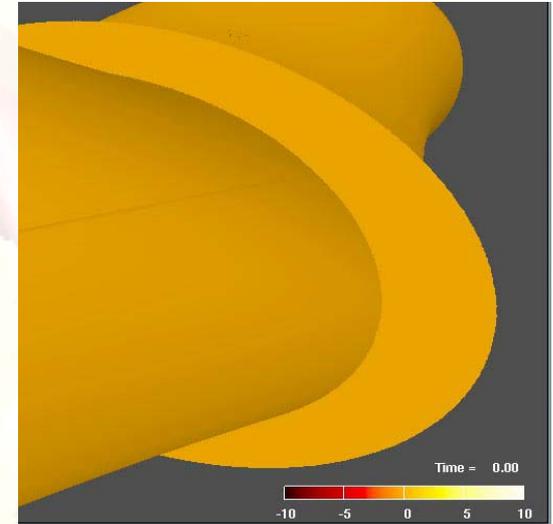
ST

Tokamak

# INES

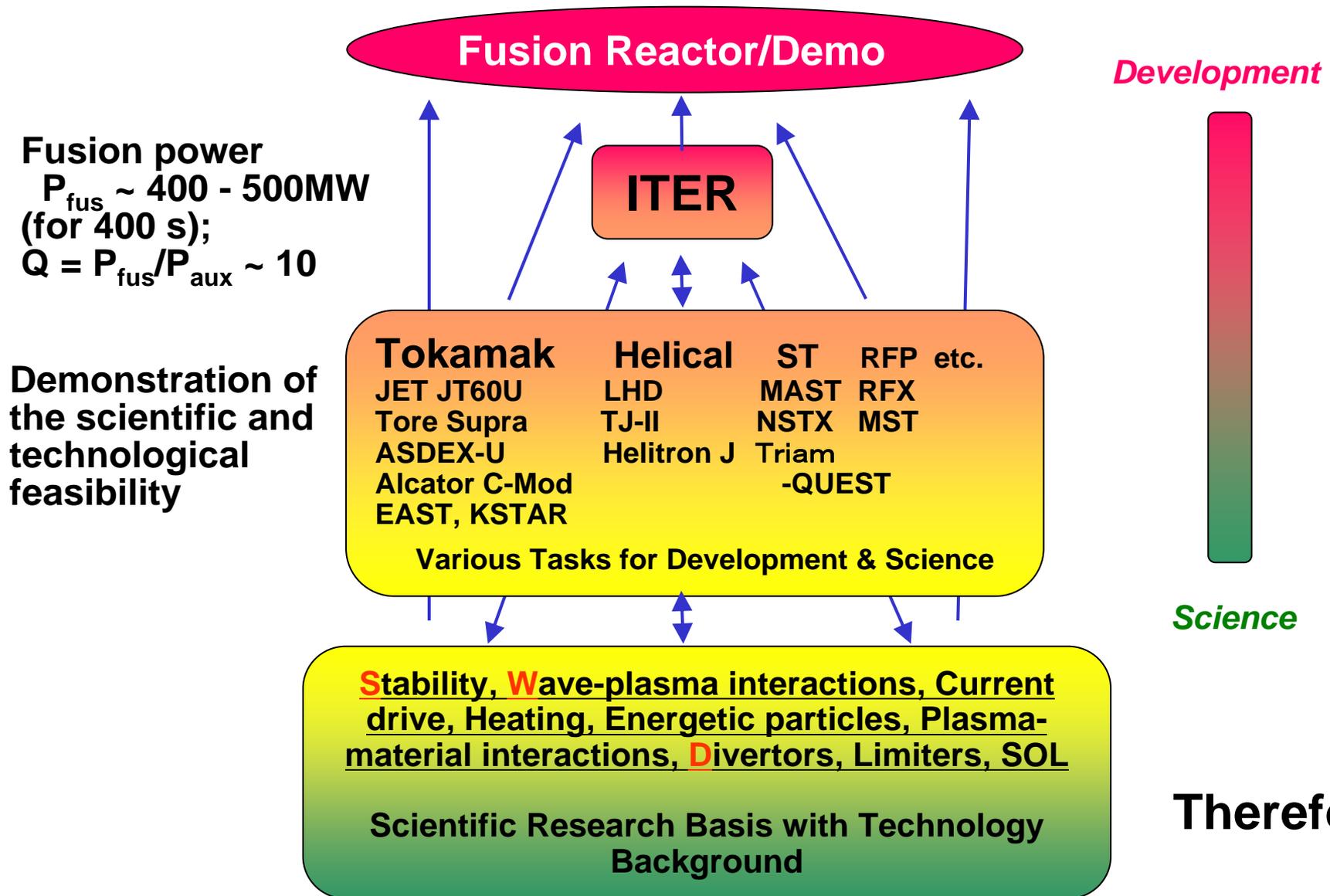
## What is a plasma?

- Fourth State of the Matter
- State-of-the-Art Complexity
- High Energy Density
- Warm Dense Matter
- Non-linear Phenomena
- Far Equilibrium State
- Self-Organization

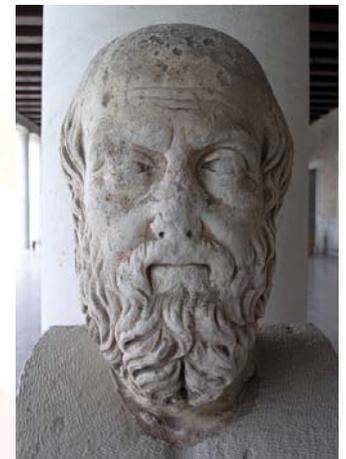


**To understand these, 50 years were necessary  
We had to wait for the development of Plasma Physics  
(Complexity, Ilya Prigogine 1977)**

# Required: Integration of Science and Technology Structure with Well Defined Multi-Layers Renormalization among them



# Herodotus



The Greek historian Herodotus wrote the following about the specialization of physicians in a developing country at that time -

**“The art of medicine among them is distributed thus: each physician is a physician of one disease and of no more; and the whole country is full of physicians, for some profess themselves to be physicians of the eyes, others of the head, others of the teeth, others of the affections of the stomach, and others of the more obscure ailments”<sup>1</sup>**

**Therefore there were no physicians who could treat the whole body**

*Herodotus* was the ‘Father of History’ from the 5<sup>th</sup> century B.C.

<sup>1</sup>History of Herodotus, Book II: Euterpe 84. Translated by G. C. Macaulay

# Progressivism in Fusion Research

Two approaches are necessary

Reductionism

Holism (Hierarchy, Re-normalization)

René Descartes professed both!

(Discours de la methode)



- 1) Sustainable progressivism is possible ?
- 2) We must not look back ?
- 3) How is topology in progressivism ?
- 4) Science can support progressivism?

**Scientific basis is needed!**

## 1. ITER oriented

(1) ELM mitigation (**S**)

(2) Divertor armor strategy (**D**)

PMI, Tungsten

(3) Current drive and heating (**W**)

(4) Disruption mitigation (**S**)

Session IT/1:ITER-1, R.J.Hawryluk, C.G.Lowry, P.R.Thomas

**S:** Stability,

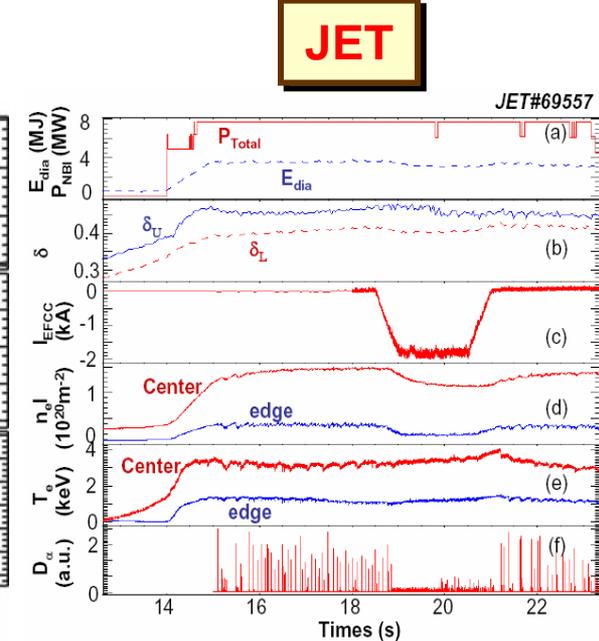
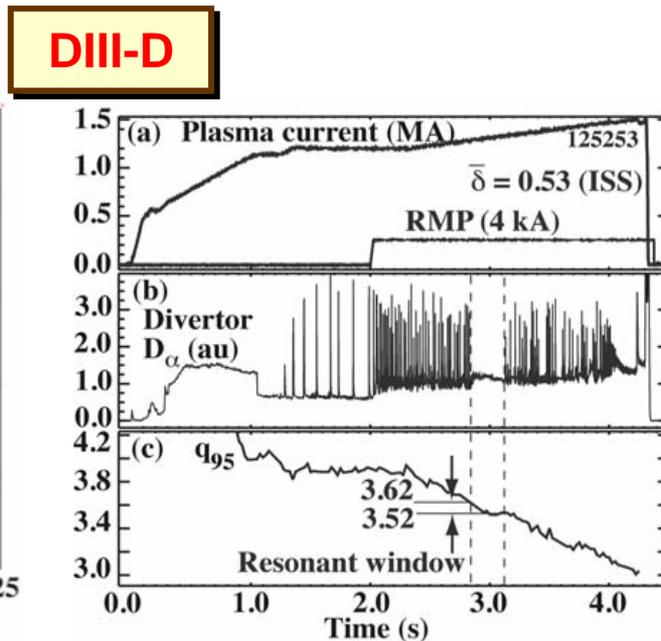
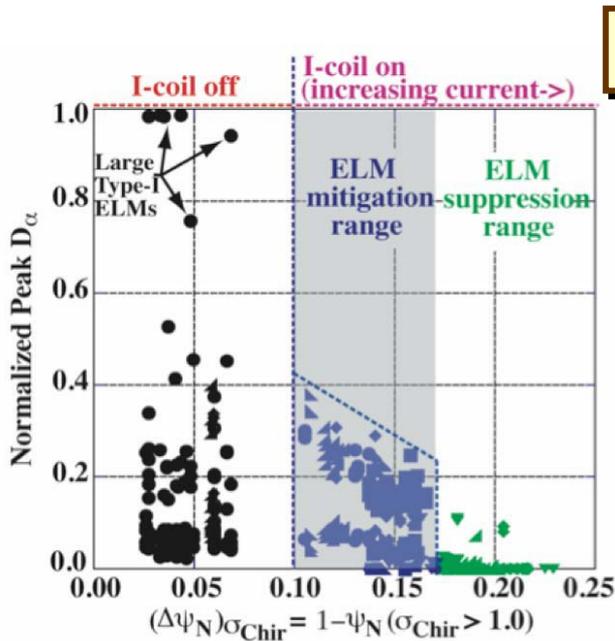
**W:** Wave-plasma interactions, Current drive, Heating, Energetic particles

**D:** Plasma-material interactions, Divertors, Limiters, SOL

# (1) ELM Control (I)

(S)

- ELM mitigation/suppression technique with edge ergodization is established by active control coils
- Significant results of inter-machine-experiments (**JET, JT-60U, DIII-D and AUG**)
  - T. E. Evans, et al., "Operating Characteristics in *DIII-D* ELM-Suppressed RMP H-modes with *ITER* Similar Shapes" EX/4-1
    - Clear relation between ELM size and island overlap parameter
      - available to scale *DIII-D* perturbation fields to *ITER* perturbation coil design
      - q95 ELM suppression window is identified
  - Y. Liang, et al., "Active Control of Type-I Edge Localized Modes with  $n=1$  and 2 fields on *JET*" EX/4-2
    - Reduction of ELM size & decrease in overall  $n_e$ 
      - compensation available with gas puff



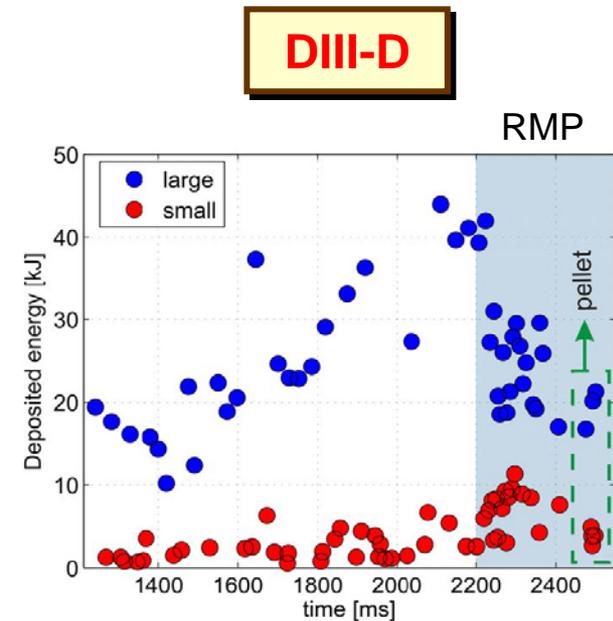
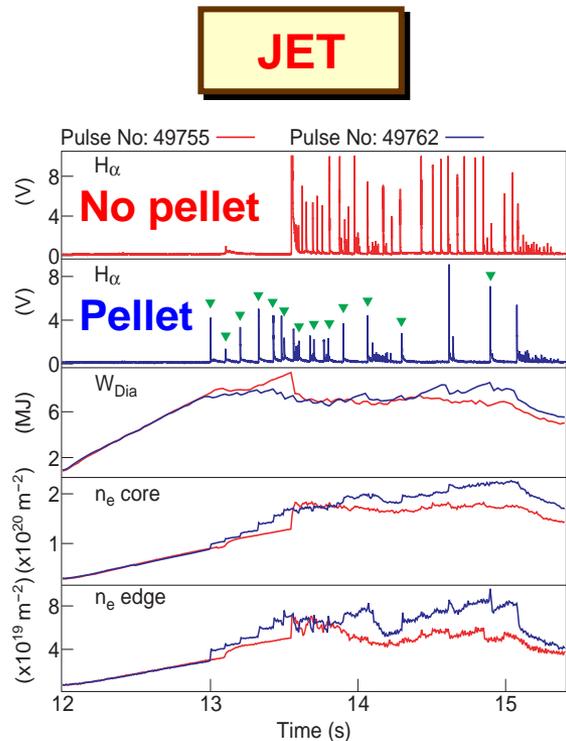
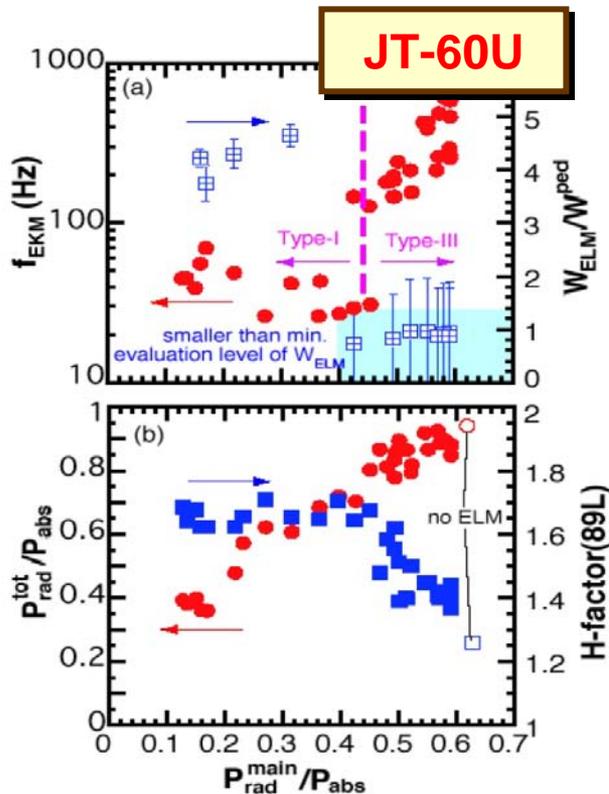
# (1) ELM Control (II)

- ELM mitigation/suppression by impurity seeding is available (JT-60U)
- ELM pacing and mitigation by pellets has been demonstrated (JET)
- Divertor heat loads are certainly reduced by ELM control RMP (DIII-D)

**N. Asakura**, "Investigation of impurity seeding and radiation control for long-pulse and high-density H-mode plasmas in JT-60U" EX/4-4Ra Ar injection → reduction of ELM frequency

**P.T.Lang**, "Investigating Pellet Physics for ELM Pacing and Particle Fuelling in ITER" EX/4-5 Pellet pacing in JET very likely to work at ITER

**M. W. Jakubowski**, "Divertor heat loads in RMP ELM controlled H-mode plasmas on DIII-D" EX/P6-2 Resonant Magnetic Perturbation → reduction of ELM energy



# (2) Divertor armor strategy (I)

## - H,D retention in plasma facing components -

- Hydrogen and deuterium retention has been made clear by using gas balance analysis during discharges and post mortem analysis to provide the reliable prediction of the tritium inventory in ITER (**Tore Supra**)

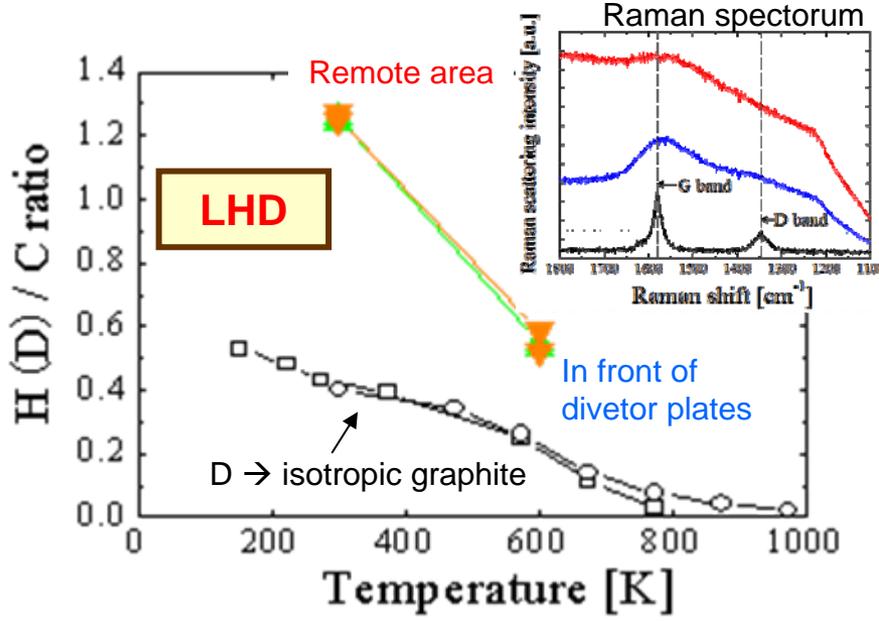
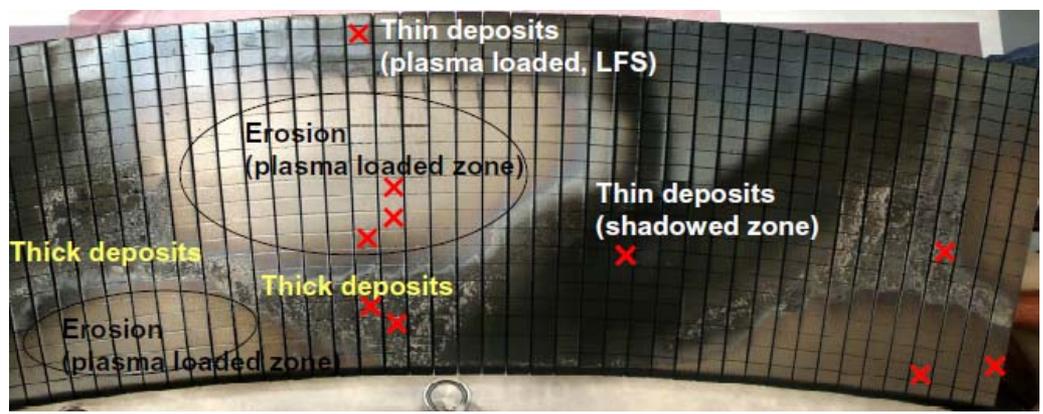
**E. Tsitron** "Deuterium Inventory in **Tore Supra** : reconciling particle balance and post mortem analysis" EX9-1

- Post mortem analysis using Thermal Desorption Spectroscopy and Nuclear Reaction Analysis reveals that D inventory is mainly due to co-deposition (90%), in particular due to gap deposits
- 50% of the inventory deduced from gas balance has been found through post mortem analysis

**T.Hino** "Hydrogen Concentration of Co-deposited Carbon Films Produced in the Vicinity of Local Island Divertor in **LHD**" EXP4-08

- Carbon deposition layer with the polymeric a-C:H structure was formed at areas both near by and far from the divertor plate. Very high H/C is observed at the remote area

**Tore Supra**



## (2) Divertor armor strategy (II)

### - Success of W wall operation -

(D)

- Scaling laws for tritium retention in co-deposits have been developed based on systematic laboratory studies. The reduced D inventory has been confirmed in AUG with full-tungsten plasma facing components

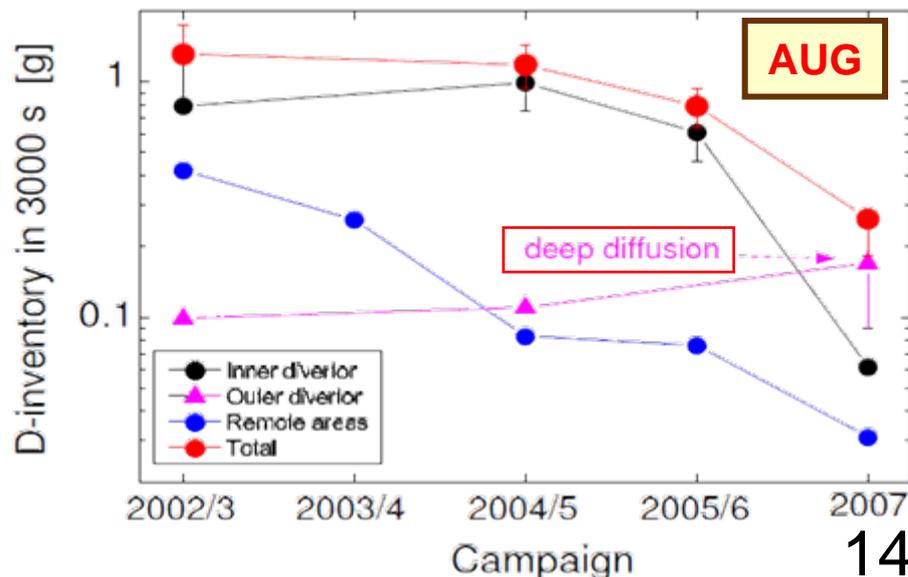
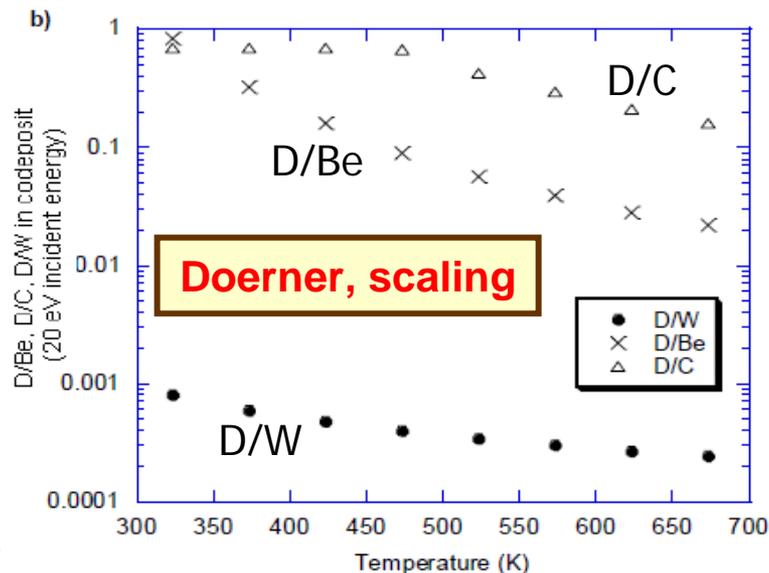
**R.P.Doerner:** "Issues Associated with Codeposition of Deuterium with ITER Materials" EX/P4-04

- Scaling laws for tritium retention in co-deposits of each of the three likely ITER materials (C, Be and W) have been developed based on systematic laboratory studies. The temperature and incident particle energy dependence of the inventory have been indicated

**A.Kallenbach:** "Non-Boronized Operation of ASDEX Upgrade with Full-Tungsten Plasma Facing Components" EX/9-2

**V.Rohde:** "Dynamic and Static Deuterium Inventory in ASDEX Upgrade with Tungsten First Wall" EX/P4-03

- D deposition was reduced to be  $\sim 1/10$  in full-W PFCs in AUG.
- Deep implantation of D inside the VPS-W layer causes the relatively high D deposition at the outer divertor



# (2) Divertor armor strategy (III)

(D)

## - W behavior -

- W PFCs have been examined successfully in **ASDEX Upgrade** (full W PFCs) and **JT-60U** (partial W divertor plate). In both devices, W accumulation was observed under certain conditions
- Further experiments, e.g. effect of large transient heat flux posed by ELMs, are recognized necessary to foreseeing ITER situation

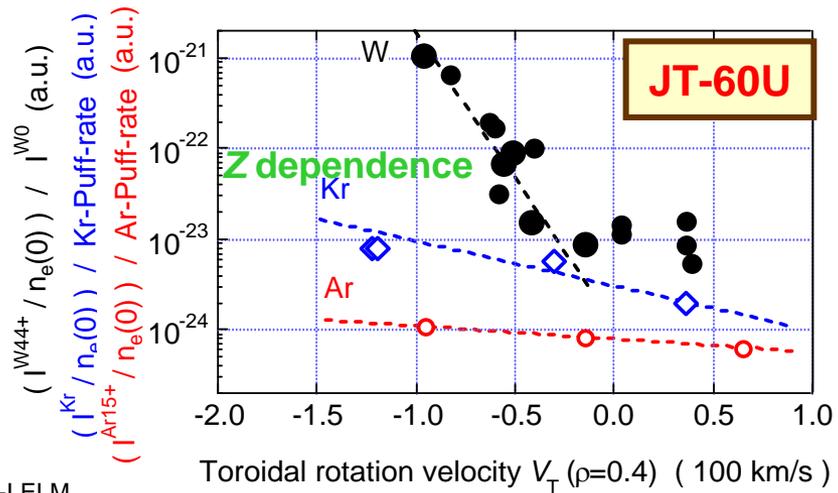
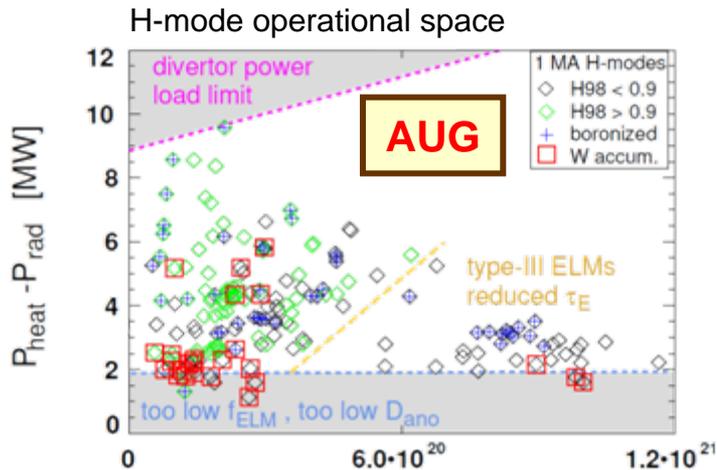
**A.Kallenbach:** “Non-Boronized Operation of **ASDEX Upgrade** with Full-Tungsten Plasma Facing Components” EX/9-2

- Central W accumulation → in combination with density peaking
- Most important control parameters → the central heating power and ELM frequency (gas puff)
- W accumulation around the pedestal top → large neo-classical inward pinch at ETB region

**T.Nakano:** “Impurity accumulation in the main plasma and radiation processes in the divertor plasma of **JT-60U**” EX/P4-25

**Y. Ueda:** “Localized Tungsten Deposition in Divertor Region in **JT-60U**” EX/P4-16

- Z dependence of the accumulation was observed in the core accumulation
- The toroidal plasma rotation velocity, and thus the radial electric field plays an important role
- Deposition profiles of tungsten indicate that inward drift in the divertor region plays a significant role in W transport in JT-60U



## (2) Divertor armor strategy (IV) - Dust characterization -

- For one of the safety issues, dust characterization is very important
- Statistical and three dimensional measurements of dust behavior and structure are necessary for projections of dust production and accumulation rates in next step device

**D.L.Rudakov:** “Dust Studies in *DIII-D* and *TEXTOR*” EX/9-3

- Dust size distribution and radial density profile were measured by using Mie scattering light

**C. Castaldo:** “Detection of Dust Particles in *FTU*” EX/P4-05

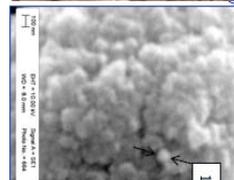
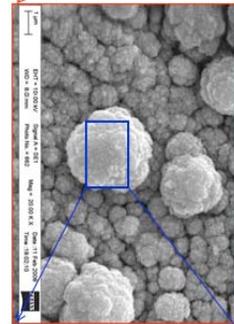
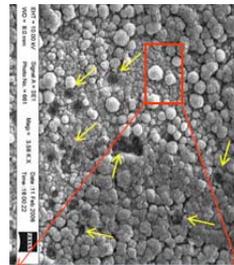
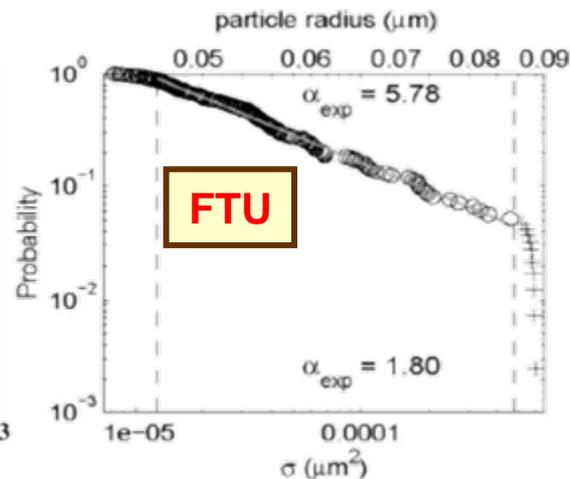
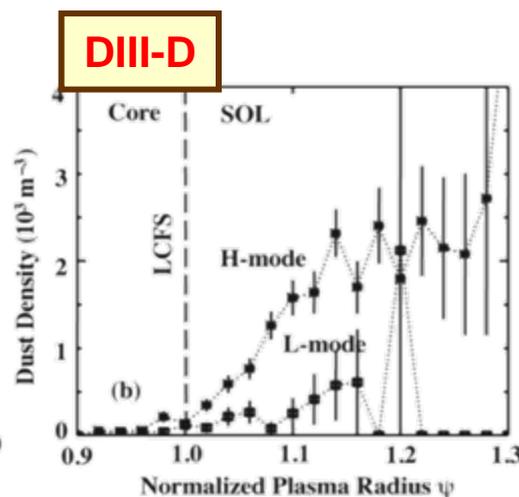
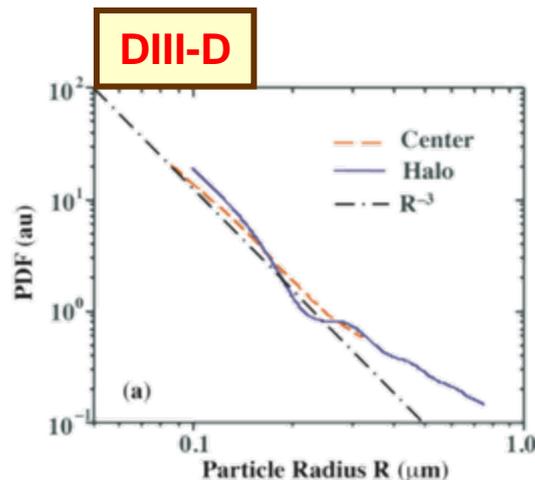
- Dust size distribution after disruptions was measured by using Rayleigh scattering light

**N.Ashikawa:** “Characterization of heated dust particles using infrared and dynamic images I n *LHD*” EX/P4-07

- Dust movement was observed by using a fast framing IR camera

**L.N. Khimchenko:** “Study of Dust Morphology, Composition and Surface Growth under I TER-relevant Energy Load in Plasma Gun *QSPA-facility*” EX/P4-13

- Fractal structure (cauliflower) was observed on the W and CFC surface after the type-I ELM like plasma irradiation.



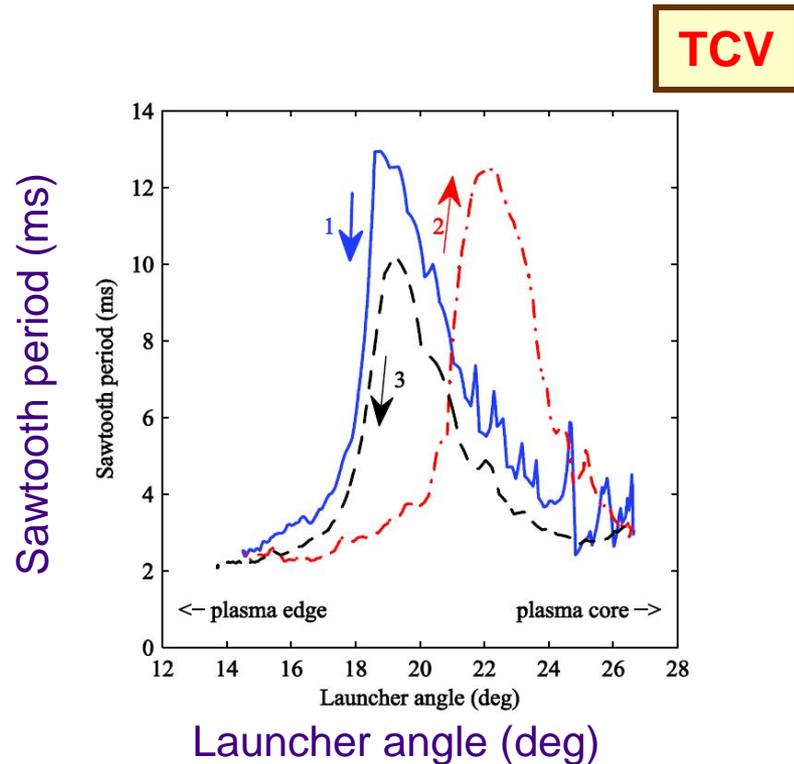
**QSPA**

100 nm

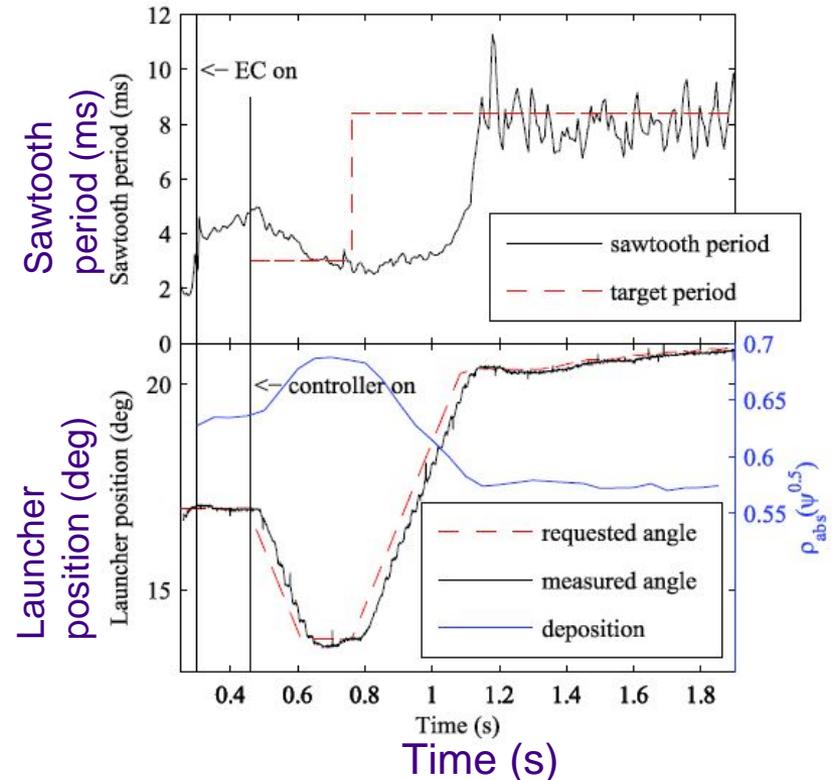
### (3) Current drive and heating (I)

- Control of the sawtooth instability has been successfully demonstrated using real time control of the EC launcher injection angle to modify the current profile around the  $q=1$  surface

**J.I.Paley:** "Real time control of plasmas and ECRH systems on **TCV**" EX/P6-16



Dependence of sawtooth period on EC launcher injection angle was investigated



The sawtooth control was demonstrated using the EC launcher injection

## (3) Current drive and heating (II)

- **Lower hybrid current drive**
  - **J.R.Wilson**: “Lower hybrid heating and current drive on the **Alcator C-Mod** tokamak” EX/P6-21
  - **M.Goniche**: “Effect of gas injection during LHCD experiments in ITER- Relevant coupling conditions” EX/P6-22, **JET**
  - **J.P.Gunn**: “Suprathermal electron beams and large sheath potentials generated by RF-antennas in scrape-off layer of **Tore Supra**” EX/P6-32
- **Fast wave**
  - **R.I.Pinsker**: “Experimental study of fast wave absorption mechanisms in **DIII-D** in the presence of energetic ions” EX/P6-24
  - **C.K.Pillips**: “High harmonics fast wave heating and CD on **NSTX** suggest that fast wave propagates away from the antenna with low magnetic field” EX/P6-25
- **ICRF**
  - **S.J.Wukitch**: “Ion Cyclotron antenna impurity production and real time matching in **Alcator C-Mod**” EX/P6-23
  - **V.Bobkov**: “ICRF antenna operation with Full W-wall in **ASDEX Upgrade**” EX/P6-31
  - **J.Ongena**: “Heating optimization studies at **JET** in support of ITER” EX/P6-33

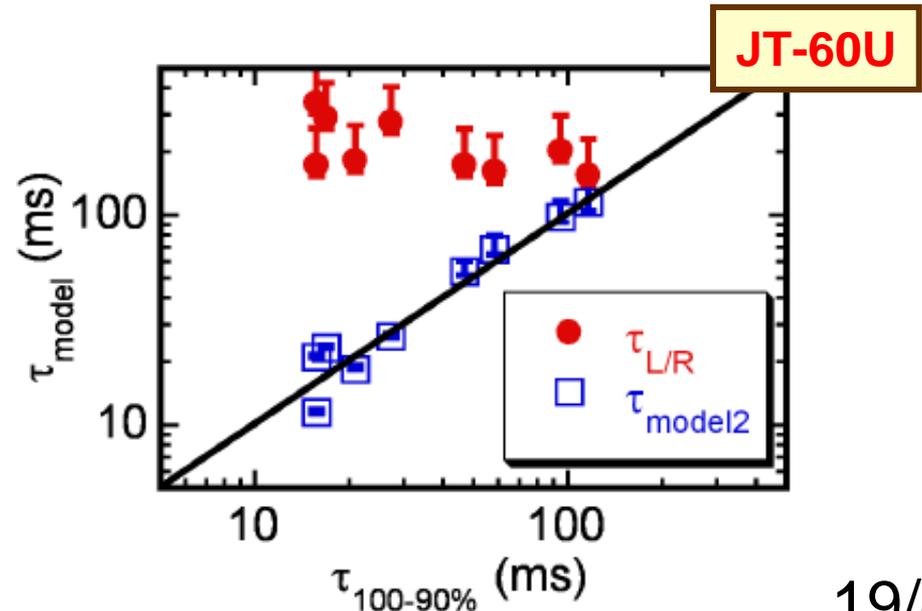
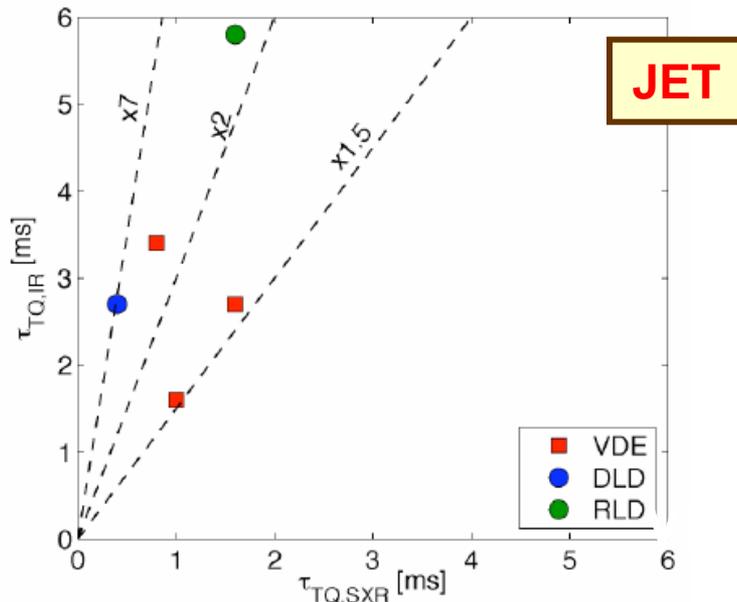
# (4) Disruption and the effect on PFC

(S)

- Characteristics of thermal and current quenches have been investigated in tokamaks
- During **thermal quench** on JET, energy deposition timescale is identified to be 2-7 times longer than that of the plasma thermal energy collapse
- Measurements in the early phase of the **current quench** in JT-60U suggest that the L/R model used for scaling law should be revised with a time dependent inductance

G.Arnoux, "Heat Loads on Plasma Facing Components During Disruptions on JET", EX/7-2 Ra

M.Okamoto, "Study of Current Decay Time during Disruption in JT-60U Tokamak", EX/7-2 Rc



## (4) Disruption Mitigation (I)

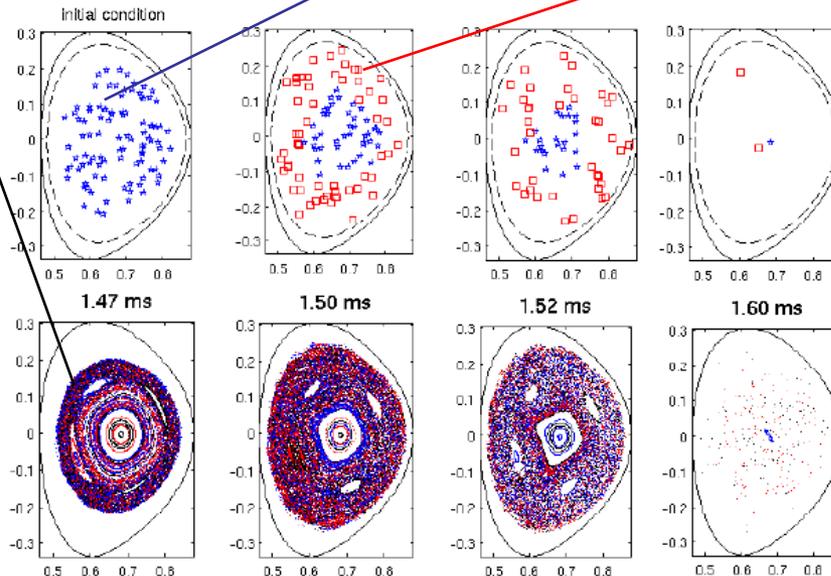
- **Massive gas injections (MGI)** are useful to avoid the generation of runaways in addition to the force on structures due to eddy and halo currents, and high surface power loading

**E.S. Marmor**, “Overview of the **Alcator C-Mod** research program” OV/4-4

magnetic flux surfaces

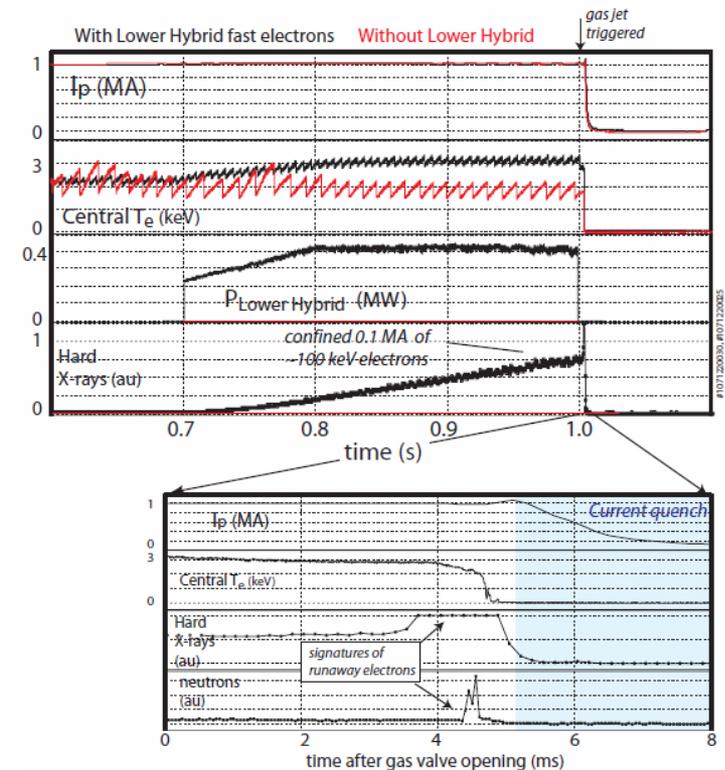
Supra thermal electrons

runaway electrons



Growth of strong MHD mode leads to the stochasitization of magnetic field lines

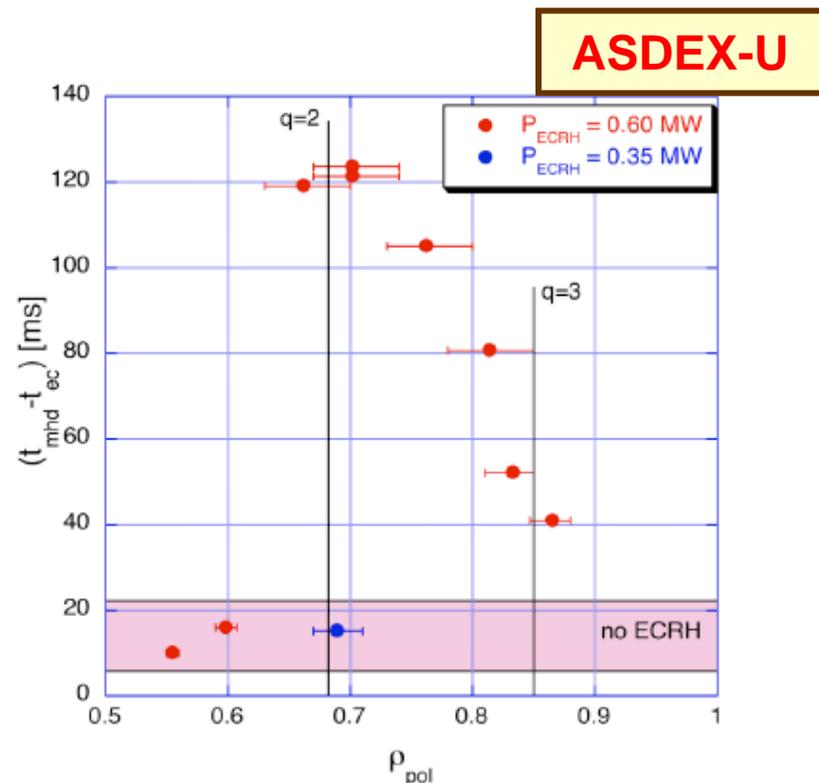
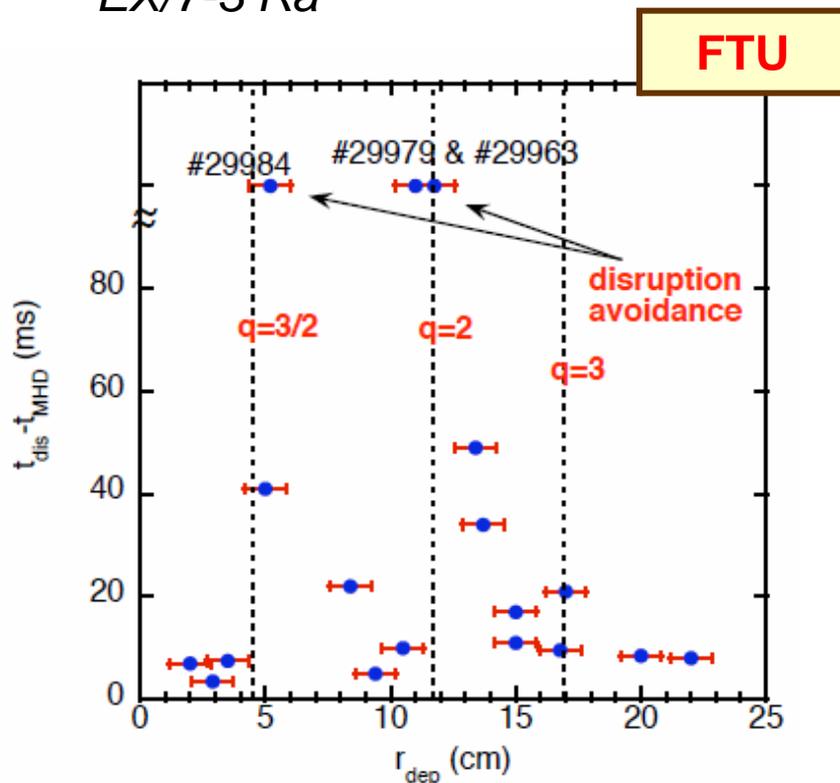
**Alcator C-MOD**



## (4) Disruption Mitigation (II)

- Local heating by ECRH on  $q = 2$  ( $3/2$ ) surface enables the delay or avoidance of the density limit disruptions (**FTU, AUG**)
- Power threshold of the disruption suppression has been found in **AUG**

*B. Esposito*, "Disruption Control on **FTU** and **ASDEX Upgrade** with ECRH",  
EX/7-3 Ra



## 2. ITER/DEMO oriented

(1) Steady state (**W,S**)

(2) High  $\beta$  (**S**)

NTM, RWM, etc.

(3) PMI & SOL/DIV physics (**D**)

(4) TAE (**W, S**)

(5) Heating (**W**)

EBW, HHFW

**S:** Stability,

**W:** Wave-plasma interactions, Current drive, Heating, Energetic particles

**D:** Plasma-material interactions, Divertors, Limiters, SOL

# (1) Steady State Operation (I)

(W, S)

## - Aiming at realization of non-inductive steady state operation

**A. Fasoli**, “Overview of Physics Research on the **TCV** Tokamak” OV/1-1

- Full Bootstrap Current operation was realized with electron ITB

**N. Oyama**, “Overview of **JT-60U** Results toward Establishment of Advanced Tokamak Operation” OV/1-3

- Long-pulse high- $\beta_N$  and high confinement without NTM

**E.J. Strait**, “**DIII-D** Research in Support of ITER “ OV/1-4

- Fully non-inductive current sustainment with  $\beta_N \sim 3.7$

- High bootstrap fraction of  $> 90\%$  with  $\beta_N \sim 5$

**G. Giruzzi**, “Investigation of Steady-State Tokamak Issues by Long Pulse Experiments on **Tore Supra**” OV/3-3

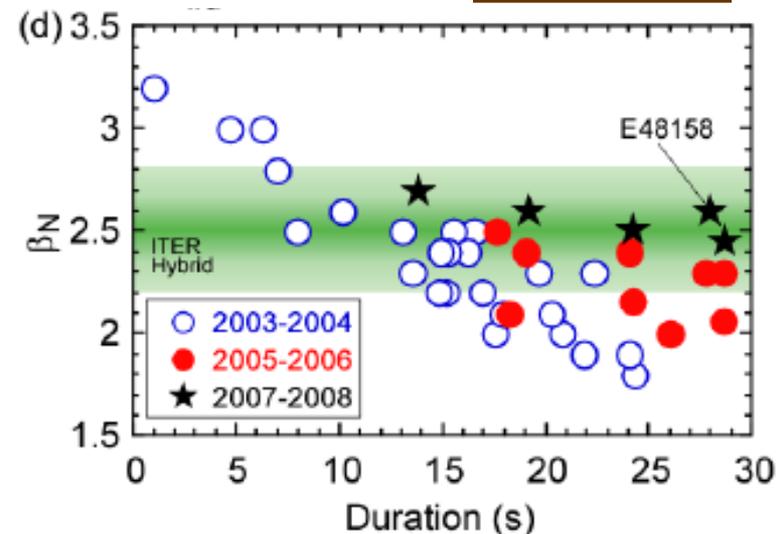
- High-power of 12MW operation

**Baonian Wan**, “Recent experiments in the **EAST** and **HT-7** Superconducting Tokamaks” OV/3-4

- 400s long pulse

**J.S. Bak**, “Overview of Recent Commissioning Results of **KSTAR**”

**JT-60U**

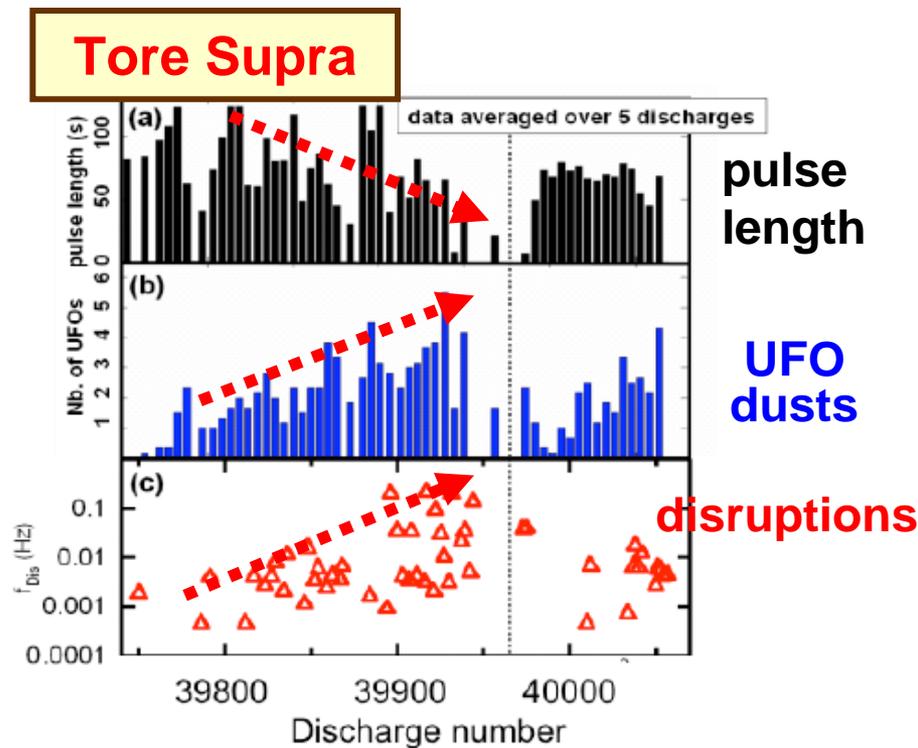
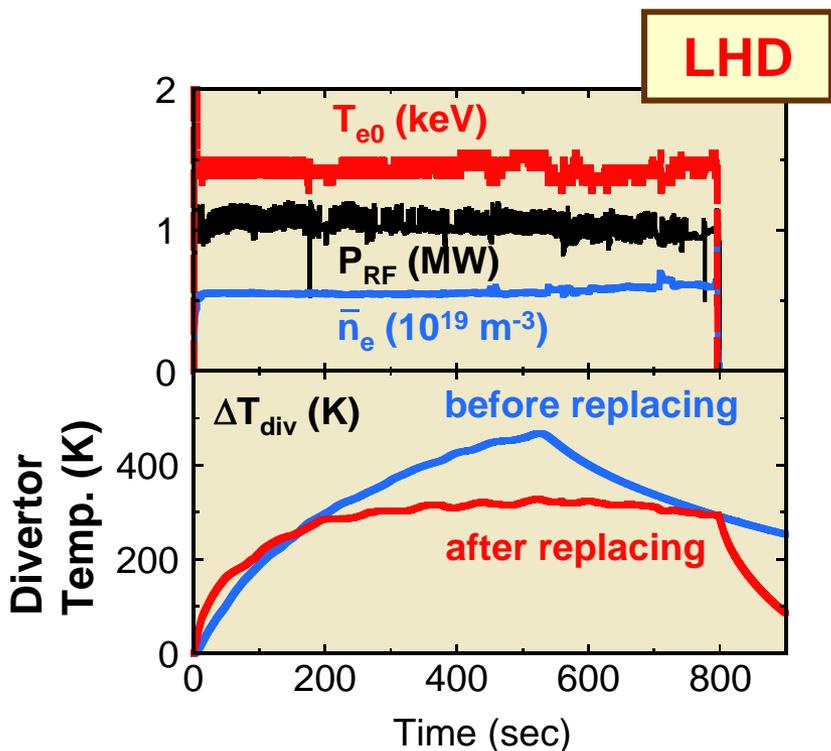


# (1) Steady State Experiment in SC Devices (II)

- Heat load on divertor plate and PFC is the common issue in magnetic confinement systems (LHD, Tore Supra)
- Net current is not required in helical devices

R.Kumazawa, "Progress towards RF Heated Steady-State Plasma Operations on LHD by Employing ICRF Heating Methods and Improved Divertor Plates" EX/P6-29

A.Ekedahi, "Operational Limits during High Power Long Pulses in Tore Supra" EX/P4-3



B.Wan "Recent Experiments in the EAST and HT-7 Superconducting Tokamaks" OV/3-4

J.S.Bak "Overview of Recent Commissioning Results of KSTAR" FT/1-1

# (2) Resistive Wall Mode

- Discharge with over  $\beta_N^{no-wall}$  has been established by exceeding critical plasma rotation
- The development of the control scenario leads to higher- $\beta_N$  regime !  
(DIII-D, JT-60U, NSTX, RFX-mod)

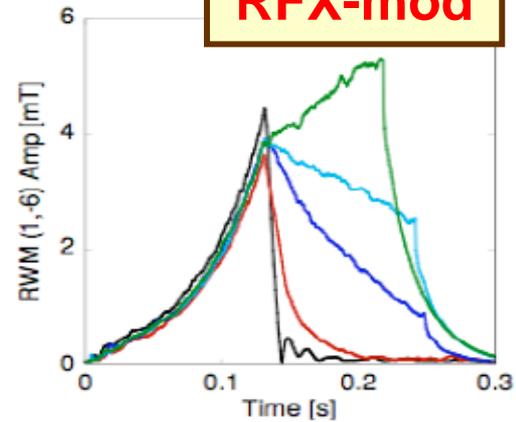
J.R. Drake, "Reversed-Field Pinch Contributions to Resistive Wall Mode Physics and Control" EX/P9-7

M. Okabayashi, "Comprehensive Control of Resistive Wall Modes in DIII-D Advanced Tokamak Plasmas" EX/P9-5

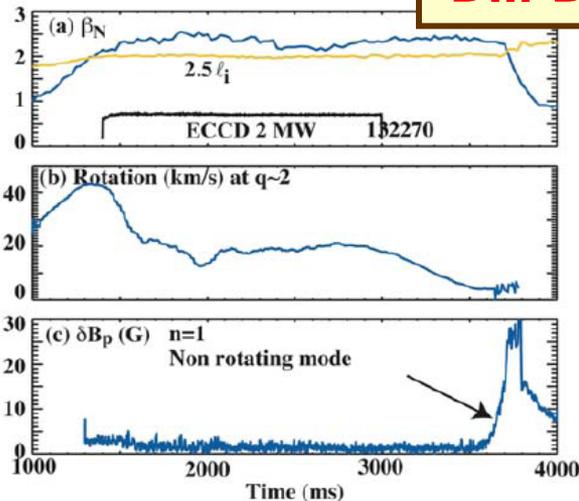
G. Matsunaga, "Dynamics and Stability of Resistive Wall Mode in the JT-60U High-beta Plasmas" EX/5-2

S.A. Sabbagh, "Advances in Global MHD Mode Stabilization Research on NSTX" EX/5-1

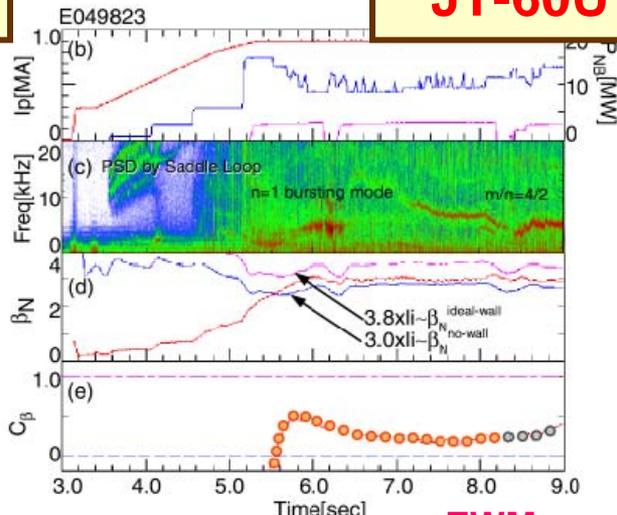
RFX-mod



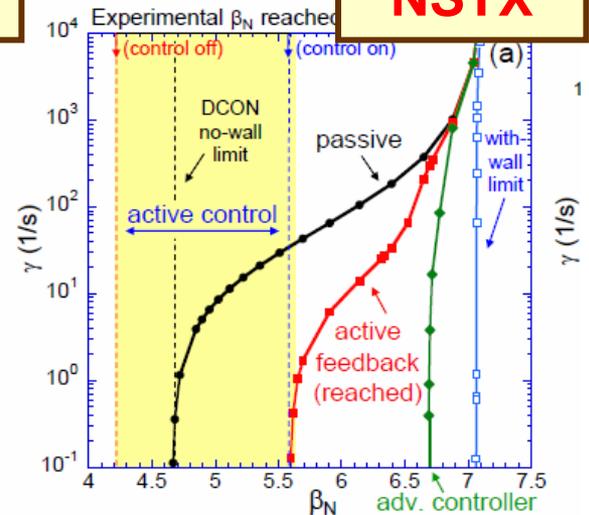
DIII-D



JT-60U



NSTX



# (2) Neoclassical Tearing Mode

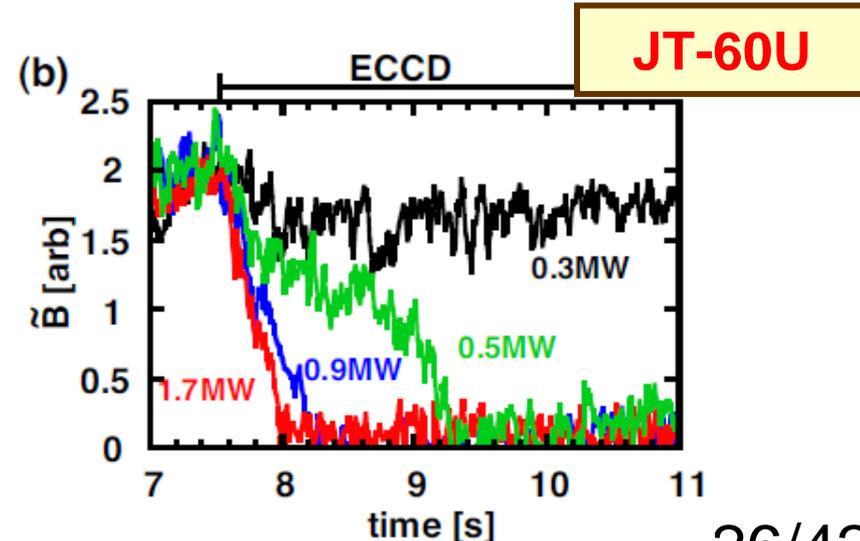
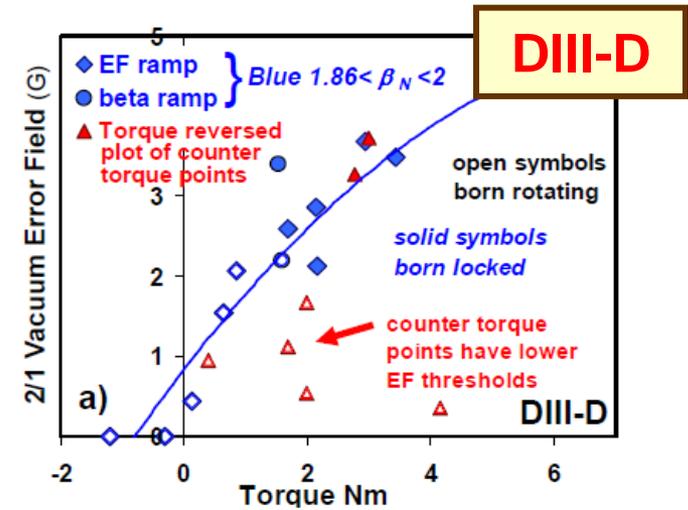
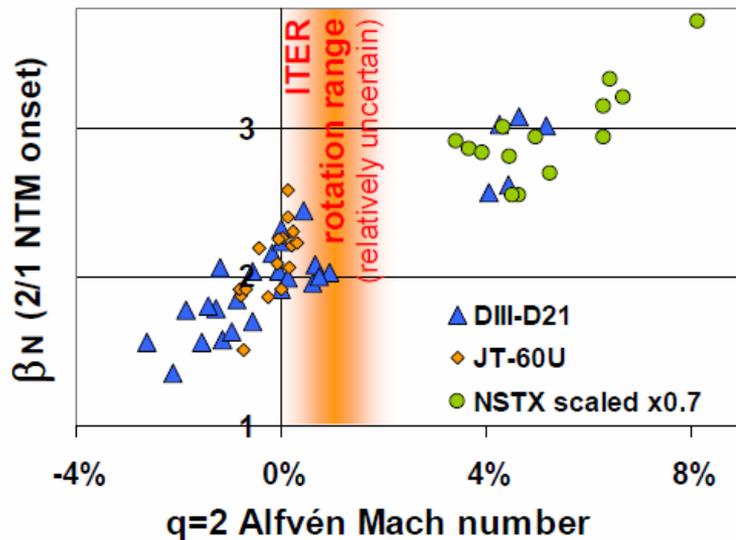
- Thresholds of **rotation (rotating shear)**, **error field** and  $\rho^*$  to NTM have been found through multi-machine extrapolations

R. J. Buttery, "Multimachine Extrapolation of Neoclassical Tearing Mode Physics to ITER", IT/P6-8

A. Isayama, "Neoclassical Tearing Mode Control with ECCD and Magnetic Island Evolution in JT-60U", EX/5-4,

- The minimum EC power for NTM suppression has been evaluated quantitatively in JT-60U

**DIII-D, JT-60U, NSTX**



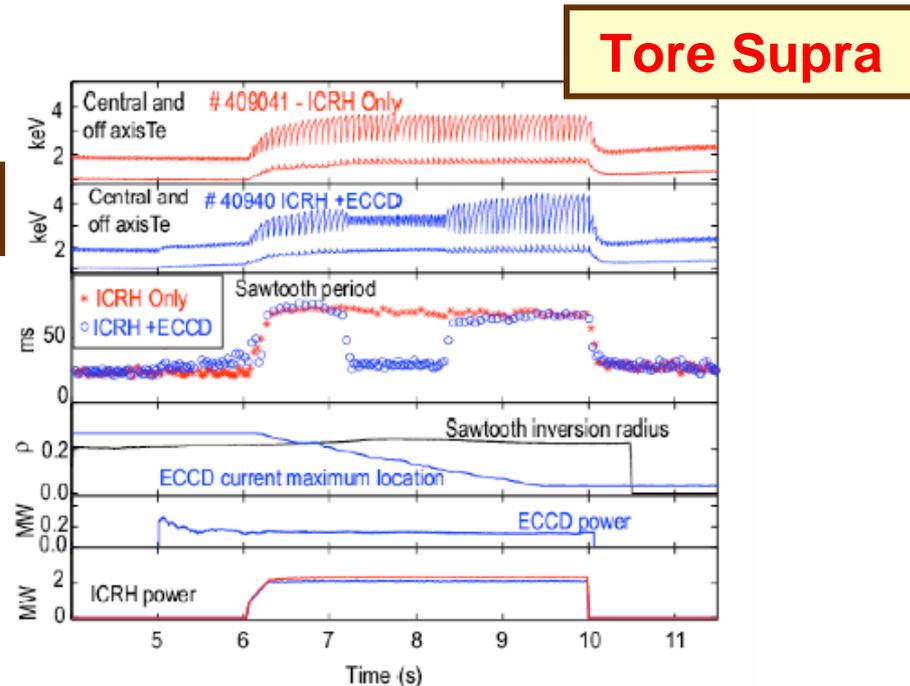
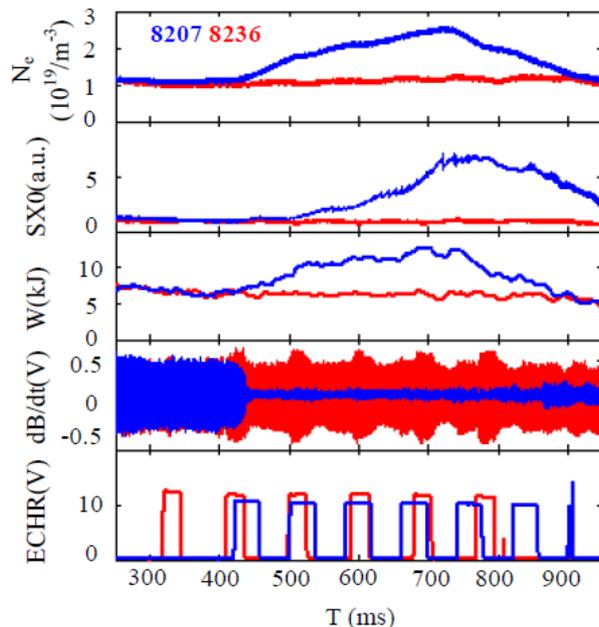
## (2) MHD Mode Control

- The control of Sawtooth Oscillation by ECCD/ECRH has been demonstrated (HL-2A, Tore Supra)
- Optimization of deposited position of EC power is very important for mode suppression

J.I. Paley, “Real Time Control of Plasmas and ECRH Systems on TCV”, EX/P6-16

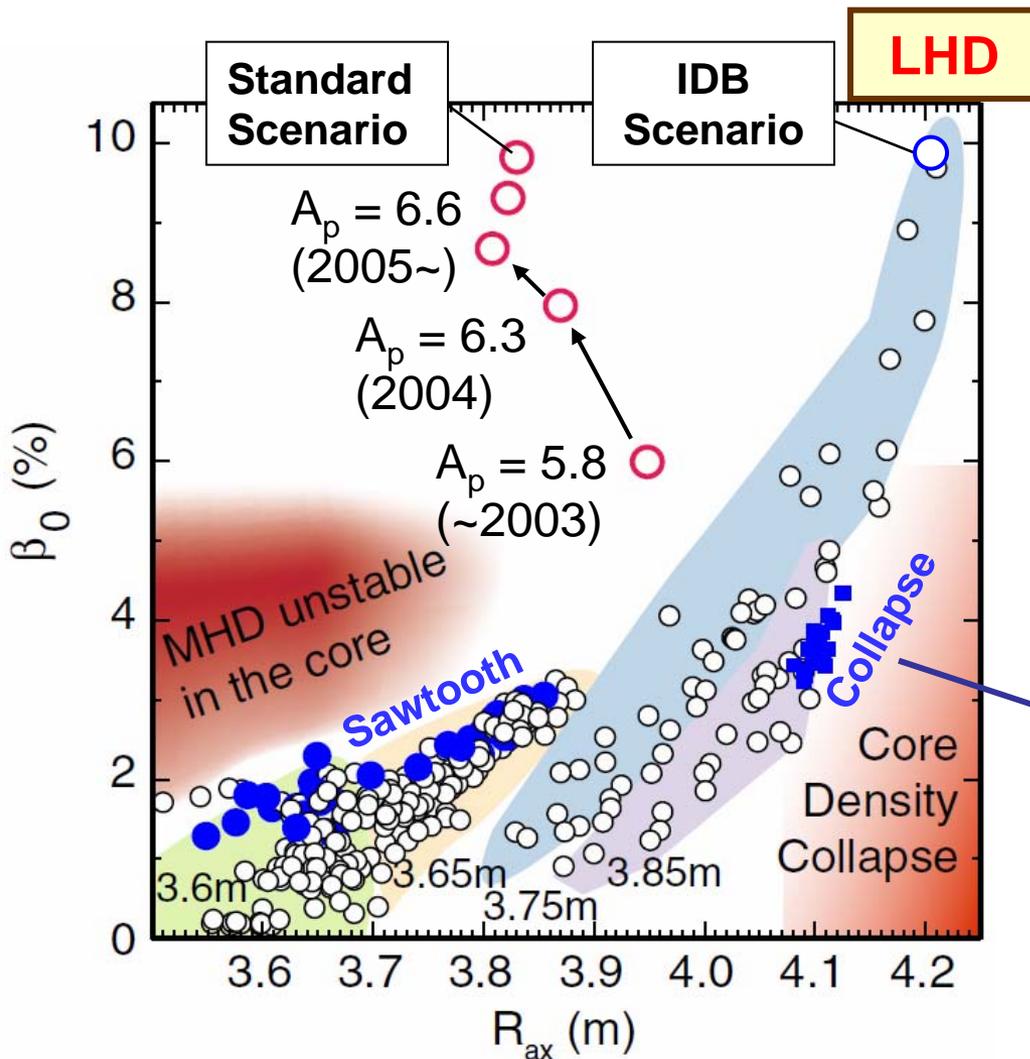
Yi Liu, “Study on Stabilization of Tearing mode with ECRH and its resultant transport properties on HL-2A tokamak”, EX/P9-2

G. Giruzzi, “Investigation of Steady-State Tokamak Issues by Long Pulse Experiments on Tore Supra”, OV/3-3



# (2) Access to high-beta regime in Helical System

S. Ohdachi, "Two approaches to the reactor-relevant high-beta plasmas with profile control in the Large Helical Device" EX/8-1Rb



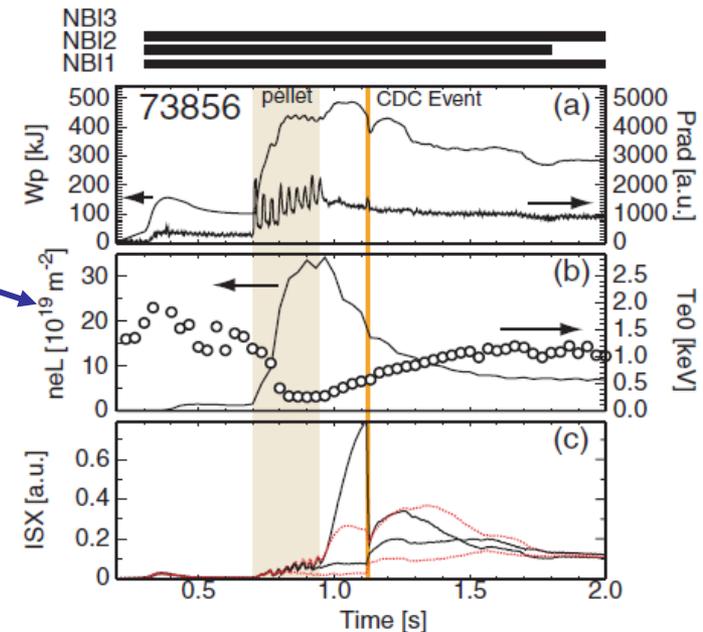
Two approaches to high  $\beta_0$  regime due to the avoidance of MHD instability

**Standard Scenario (broad profile)**

→  $\langle \beta \rangle = 5\%$ ,  $\beta_0 \sim 10\%$

**IDB Scenario (peaked profile)**

→  $\langle \beta \rangle = 2\%$ ,  $\beta_0 \sim 10\%$



# (3) PWI and SOL/Divertor physics (I)

(D)

## - Li coated and Liquid Li PFCs -

- Lithium coated PFCs (evaporation and Liquid Li limiter) reduce the recycling and impurity, and modify the profiles of plasma parameters. As the results, improved confinement, extended periods of MHD quiescence, and so on are achieved
- The modification of the profiles seems to be different between devices
- Liquid Li divertor is being installed on NSTX for the further examination

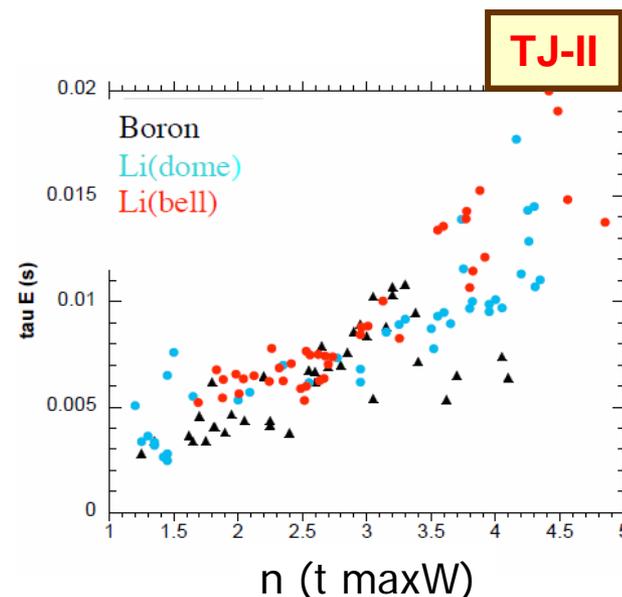
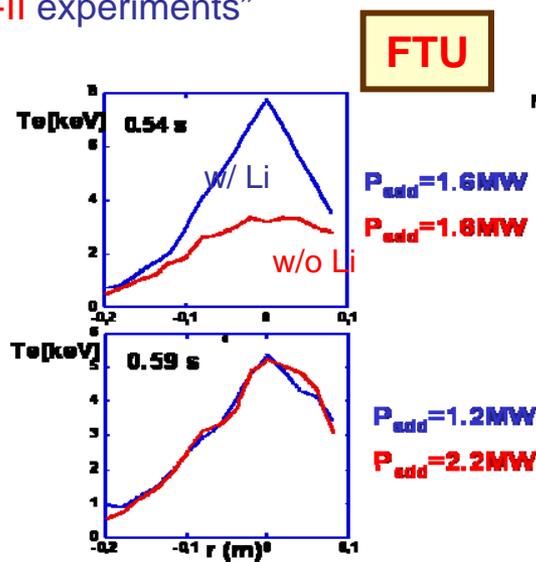
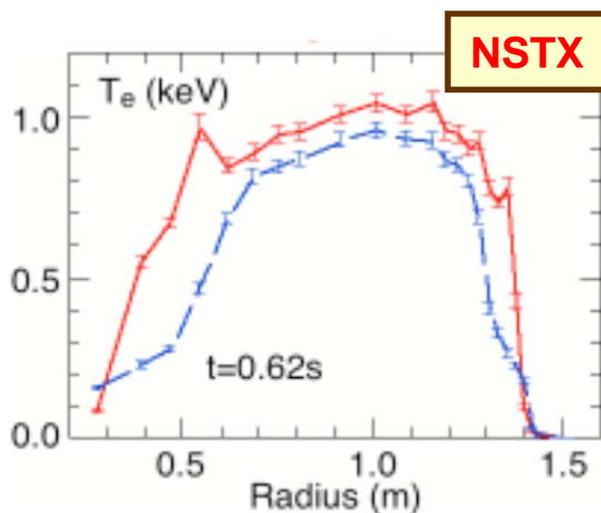
G:Mazzitelli: EX/P4-06 "Status and Perspective of the Liquid Material Experiments in FTU and ISTTOK"

R.Kaita: EX/P4-09 "Plasma Performance Improvement with Lithium-Coated Plasma-Facing Components in NSTX"

H. Zushi: EX/P4-12 "Active Particle Control Experiments and Critical Flux Discriminating between the Wall Retention and Release in the CPD Spherical Tokamak"

V.A.Vershkov: EX/P4-14 "Experiments with Lithium Gettering of the T-10 Tokamak"

J.Sánchez: OV/4-5 "Overview of TJ-II experiments"



Li effects on temperature profiles

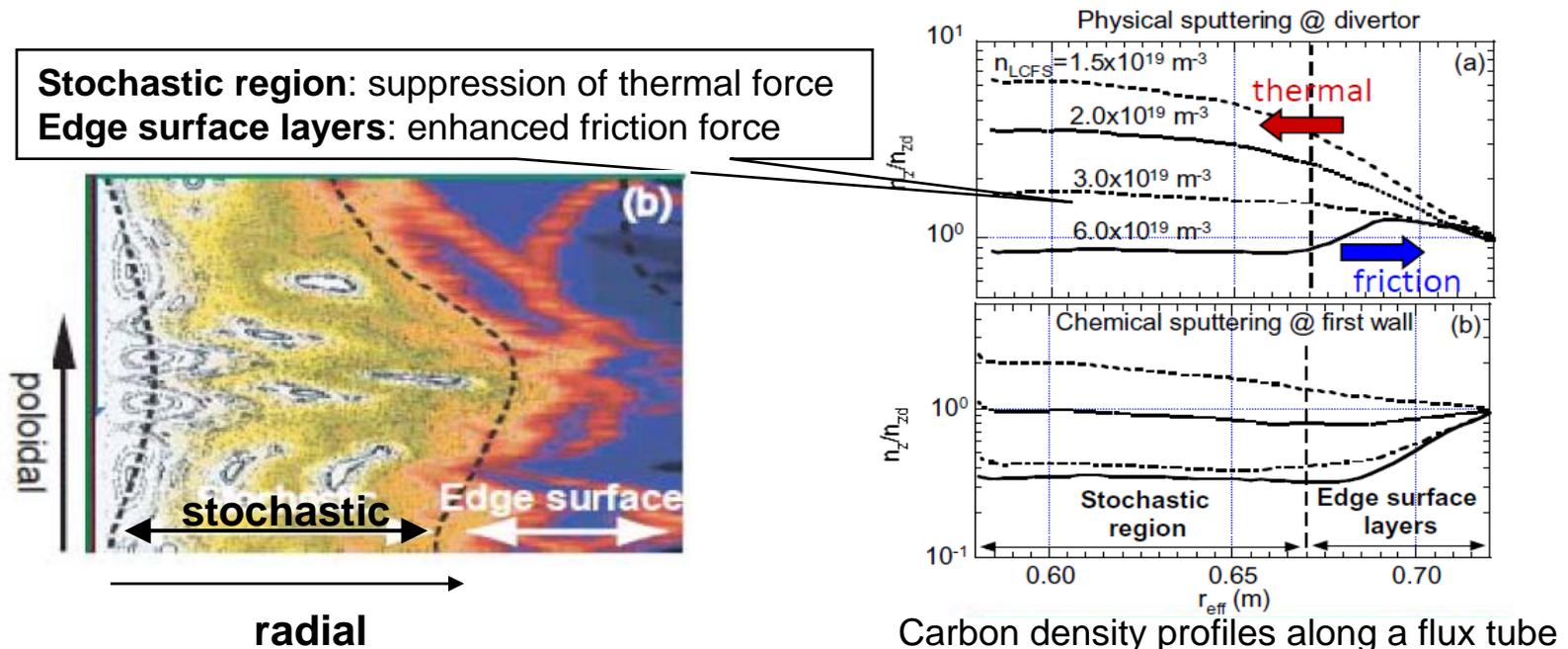
# (3) PWI and SOL/Divertor physics (II)

## - SOL/Divertor physics -

(D)

- Impurity transport in the ergodic layer has been identified in LHD with broad stochastic divertor region
- Using experimental data and 3 dimensional transport code, EMC3-EIRENE, impurity transport in the ergodic layer was investigated
- The remnant islands in the stochastic layer have an impurity screening potential when the perpendicular energy transport dominates over the parallel one at high SOL densities

M.Kobayashi: "Study on impurity screening in edge ergodic layer of LHD" EX/9-4

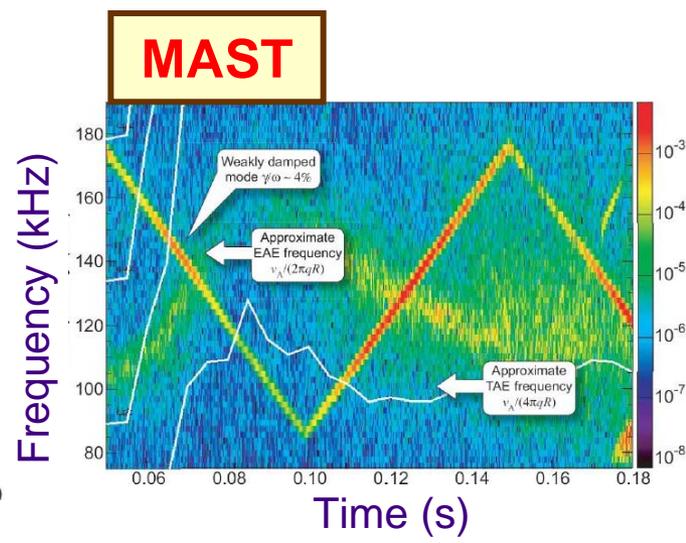
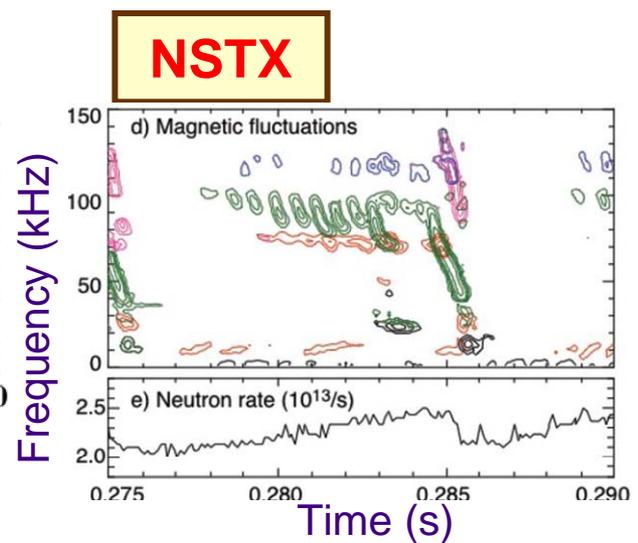
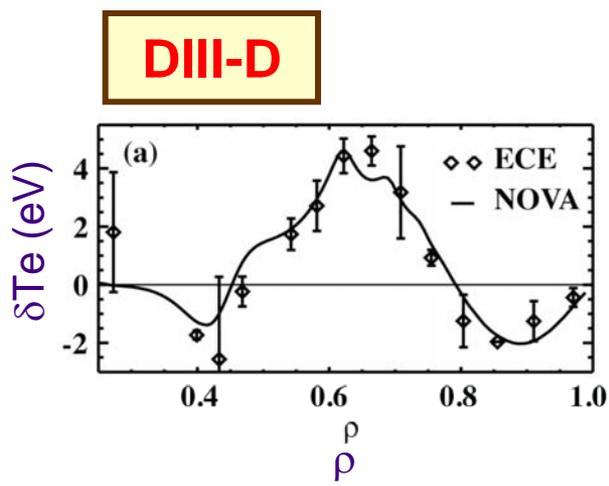


Carbon density profiles along a flux tube

Connection length of field lines profile

# (4) Alfvén eigenmodes and the avalanches

- Good agreements were found in
  - 1) Spatial profile of Alfvén eigenmode between measurements in **DIII-D** and MHD calculations using NOVA code
  - 2) Fast ion loss phenomenon between measurements in **NSTX** and calculation using ORBIT code
- Alfvén eigenmode spectra with mode damping rate have been measured using the active excitation systems in **Alcator C-Mod** and **MAST**



**M.A.Van Zeeland** : “Alfvénic instabilities and fast ion transport in the **DIII-D** tokamak” EX/6-2

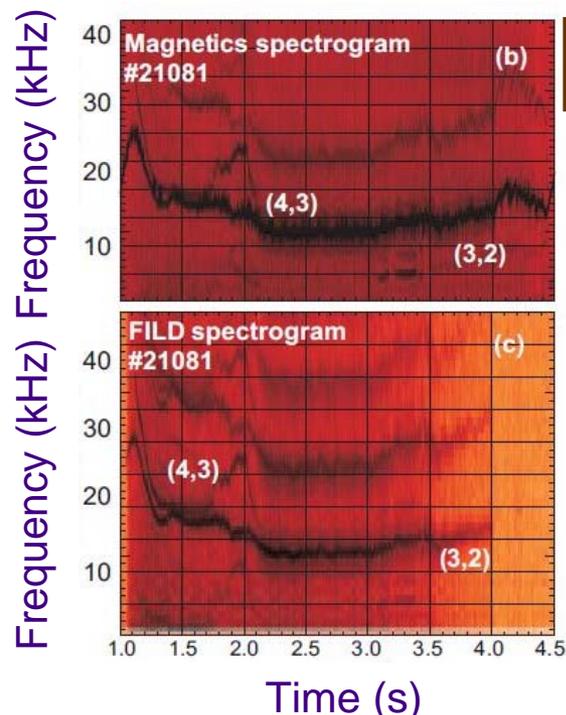
**E.D. Fredrickson** : “Toroidal Alfvén eigenmode avalanches in **NSTX**” EX/6-3

**J.A.Snipes**: “Characterization of stable and unstable Alfvén eigenmodes in **Alcator C-Mod**” EX/P8-6

**S.D.Pinches**: “Fast particle instabilities in **MAST**” EX/P8-7

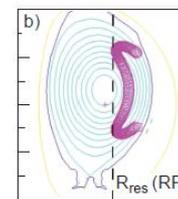
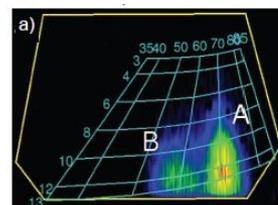
# (4) Progress in understanding of energetic particle physics

- It was demonstrated that the frequency analysis of fast ion loss detector signals in **AUG** enables the identification of the instability that induces fast-ion loss
- The lost ions were identified with trapped alpha particles in **JET**. A linear dependence of the loss intensity on Alfvén eigenmode amplitude has been directly derived from the experiments



**AUG**

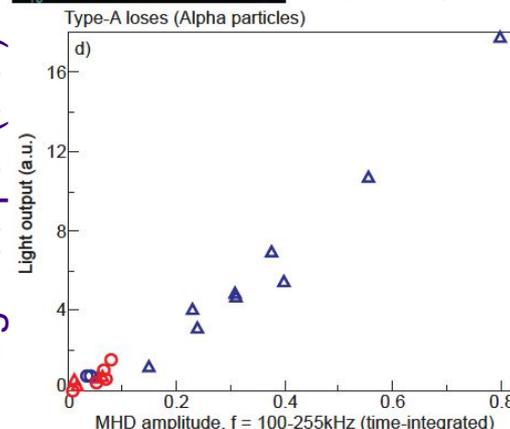
**M. Garcia-Munoz :**  
“MHD induced fast-ion losses on **ASDEX Upgrade**” EX/6-1



**JET**

**V. Kiptily :**  
“Recent progress in fast-ion physics on **JET**” EX/P8-8

Light output (a.u.)



## Energetic-electron driven fishbone

**R.Sabot:** “Observation of fast particles modes in **Tore-Supra**” EX/P8-9

**W.Chen:** “Destabilization of internal kink mode by energetic electrons on the **HL-2A** tokamak” EX/P8-11

**R. Cesarío:** “Fishbone-like internal kink instability driven by supra-thermal electrons on **FTU** generated by lower hybrid radiofrequency power” EX/P8-12

# (4) Energetic-particle driven instabilities in helical plasmas and the impact on zonal flows (W, S)

- Reversed shear Alfvén eigemodes and geodesic acoustic modes are excited simultaneously in reversed-shear plasmas of LHD

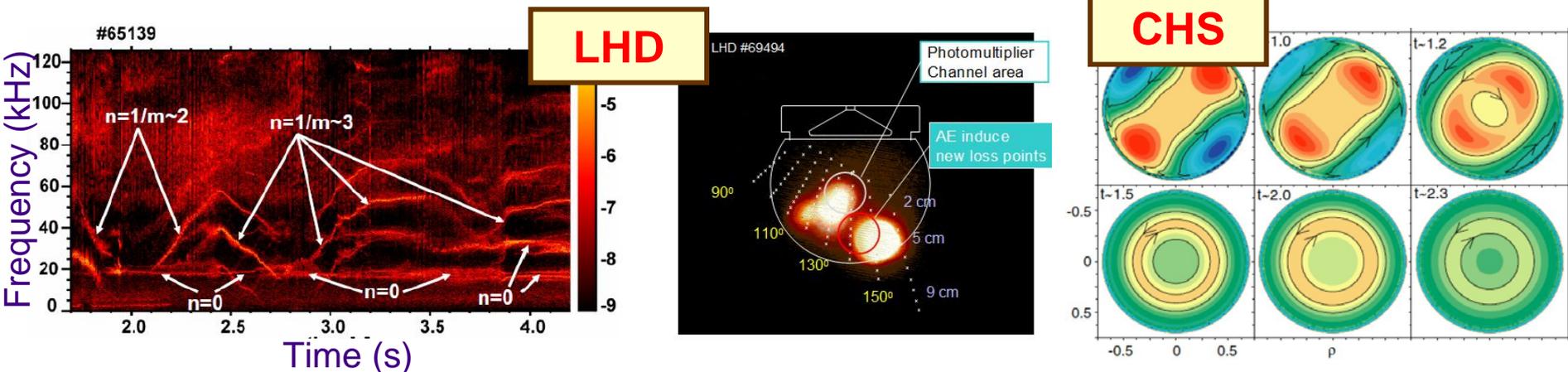
**K.Toi:** “Alfvén eigemodes and geodesic acoustic modes driven by energetic ions in an LHD plasma with non-monotonic rotational transform profile” EX/P8-4

- The Alfvén eigemode that induced fast-ion loss was identified with the lost-ion orbit trace from the lost ion probe into the LHD plasma

**M.Nishiura:** “Fast-ion transport during repetitive burst phenomena of toroidal Alfvén eigemodes in the Large Helical Device” EX/P8-5

- A new kind of zonal flow, which is generated from the fast-ion orbit loss current induced by CHS fishbone mode, was detected using the twin heavy ion beam probes

**A.Fujisawa:** “Oscillatory zonal flows driven by interaction between energetic ions and fishbone-like instability in CHS” EX/P8-1



**Y. Todo:** “Simulation study of interaction between energetic ions and Alfvén eigemodes in LHD” TH/P3-9CCV

# (5) Electron Bernstein wave heating and high harmonic fast wave heating

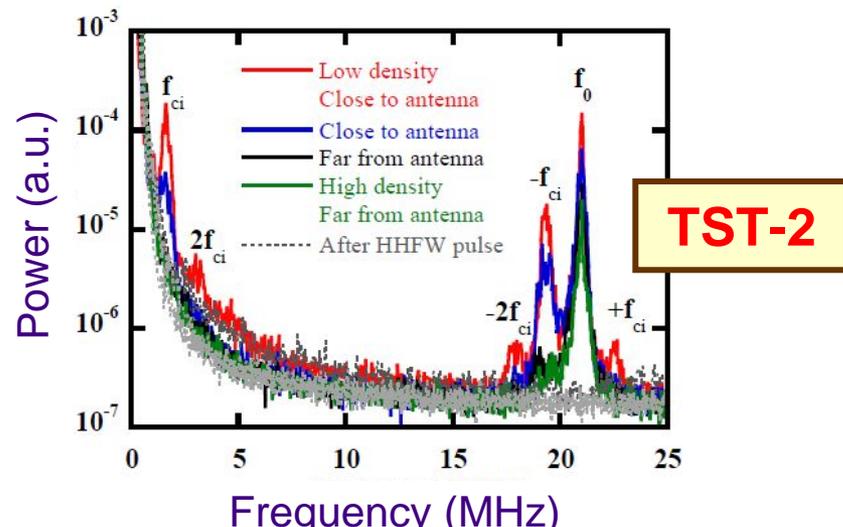
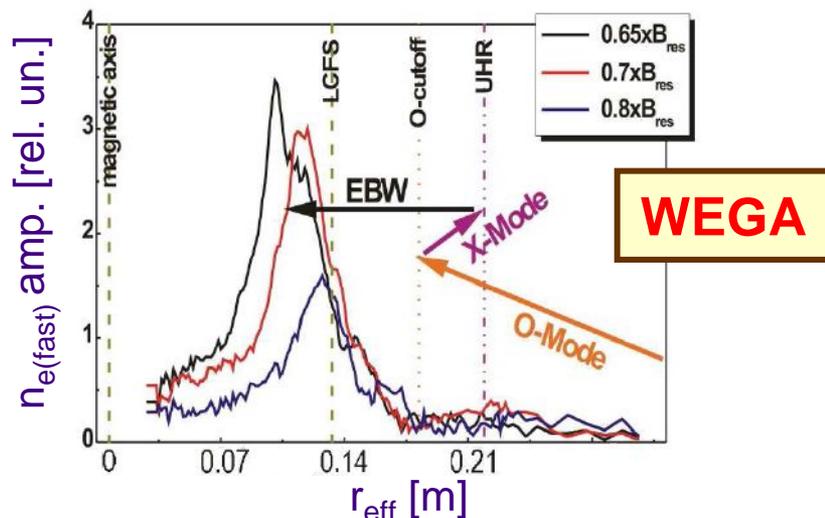
(W)

- The power deposition profiles of the electron Bernstein wave heating measured in the **WEGA** stellarator were consistent with the ray-tracing calculation

**H.Laqua**: “Fundamental investigation of electron Bernstein wave heating and current drive at the **WEGA** stellarator” EX/P6-18

- The parametric decay instability during high harmonic fast wave heating in **TST-2** spherical tokamak is weak when the plasma density is high and the plasma is far away from the antenna

**Y.Takase**: “Parametric decay instability during high harmonic fast wave heating experiments on the **TST-2** spherical tokamak” EX/P6-27



**EBW:** **H.Igami** (EX/P6-13 **LHD**), **S.J.Diem** (EX/P6-17 **NSTX**), **J.A.Goetz** (EX/P6-19 **MST**)  
**HHFW:** **H. Kasahara** (EX/P6-30 **LHD**)

## (5) ICRF heating and plasma profile control

(W)

- **ICRF heating**
  - **H. Okada:** “Velocity distribution of fast ions generated by ICRF heating in **Heliotron J**” EX/P6-28
  - **R. Kumazawa:** “Progress towards RF heated steady-state plasma operation on **LHD** by employing ICRF heating methods and improved divertor plates” EX/P6-29
- **Plasma profile control and current drive**
  - **S. Kubo:** “Profile control by local ECRH in **LHD**” EX/P6-14
  - **K. Nagasaki:** “Effect of magnetic field ripple on ECCD in **Heliotron J**” EX/P6-15
  - **Y. Nakamura:** “Time evolution of the bootstrap current profile in **LHD** plasmas” EX/P6-20
  - **M. Turnyanskiy:** “Current profile control studies on **MAST**” EX/P6-26

## (5) Non-inductive plasma startup of ST and tokamak

(W)

- Startup with EC
  - **H.Tanaka**: “Non-solenoidal formation of spherical torus by ECH/ECCD in **LATE**” EX/P6-8
  - **A.Ejiri**: “Non-inductive plasma current start-up by EC and RF power in the **TST-2** spherical tokamak” EX/P6-6
  - **T.Yoshinaga**: “Physics study of EC-excited current generation via current jump in the **Compact Plasma-wall-interaction experimental Device**” EX/P6-9
  - **J.Bucalossi**: “Electron cyclotron resonance heating assisted plasma start-up in the **Tore Supra**” EX/P6-12
- Startup with helicity injection
  - **M.Nagata**: “Experimental and computational studies of MHD relaxation generated by coaxial helicity injection in the **HIST** spherical torus plasmas” EX/P6-7
  - **R.Raman**: “Solenoid-free plasma start-up in **NSTX** using transient CHI” EX/P6-10
  - **A.C.Sontag**: “Current profile modification influence on MHD and non-solenoidal plasma startup in the **Pegasus Toroidal Experiment**” EX/P6-11

# Technical Summary (I)

A large progress has been achieved after the last IAEA FEC in Chengdu

## 1. ITER oriented

### (1) ELM mitigation (DII-D, JET, JT-60U, AUG)

An increased level of confidence for achieving robust ELM-controlled-H-modes in ITER was obtained

### (2) Divertor armor strategy (Tore Supra, LHD, AUG, JT-60U, JET, FTU)

Much less retention can be expected using tungsten, and examinations of compatibility between tungsten PFCs and plasma operation have begun

### (3) Current drive and heating (TCV, C-Mod, JET, Tore Supra, DIII-D, NSTX, AUG)

Control of sawtooth instability using the EC launcher injection angle has been demonstrated

### (4) Disruption mitigation (JET, JT-60U, C-Mod, FTU, AUG)

Disruption mitigation due to massive gas injection and ECH has been demonstrated, whereas advanced measurements lead to understanding of disruption physics

- Contribution from small devices, Tokamak, Helical, ST, etc.

**M.Gyaznevich**, OV/P1-1, "Results of joint experiments and other IAEA activities on research using small tokamaks"

# Technical Summary (II)

## 2. ITER/DEMO oriented

### (1) Steady state

Fully non-inductive current/current-less operation and ITER/DEMO relevant scenario has been demonstrated (Tore Supra, LHD, JT-60U, DIII-D, TCV, EAST, KSTAR, HT-7)

### (2) high $\beta$

Active control methods of RWM, NTM and other instabilities have been developed, based on understandings of their physical mechanisms (DII-D, JT-60U, NSTX, LHD, RFX-mod, etc.)

### (3) PWI & SOL/DIV physics

Improvements of plasma performances using Lithium coating and suppression of impurity contamination in an ergodic layer has been obtained (NSTX, FTU, TJ-II, T-10, LHD, etc.)

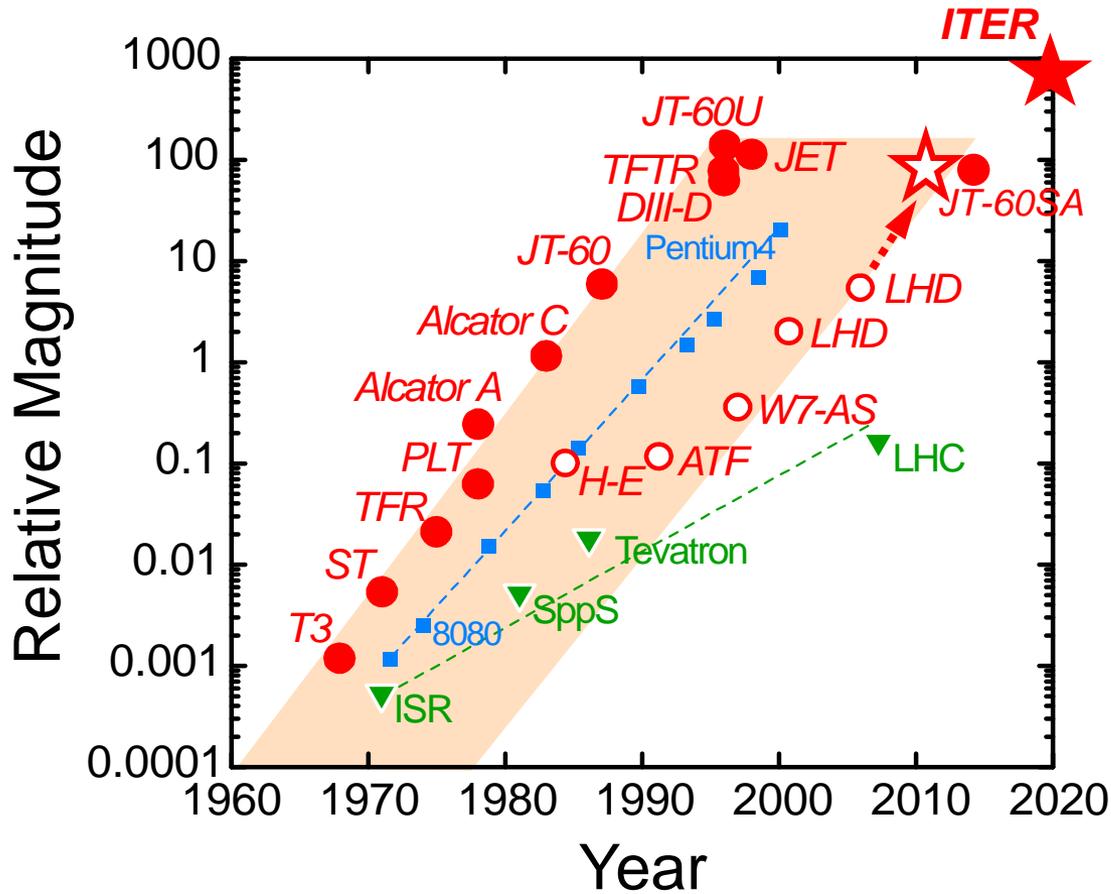
### (4) TAE

Oscillating zonal flow driven by interaction between energetic particles and MHD modes should be emphasized for burning state plasmas (DIII-D, NSTX, MAST, C-Mod, LHD, CHS, JET, Tore Supra, HL-2A, FTU, etc.)

### (5) Heating

Electron Bernstein and higher harmonic fast waves have made good progress (WEGA, TST-2, LHD, NSTX, MST, etc.)

Up to now, fusion research has progressed as rapidly as other areas of big science and high-technology, i.e. computers and high energy physics



**Fusion:**  
Triple product  $nT\tau_E$  doubled every 2 years

**Moore's law :**  
Transistor number doubles every 2 years

**Accelerators :**  
Energy doubles every 3 years

This is the development by 1 Million times in 50 years from a table top device to big science

Since the First Geneva IAEA Conference,  
50 years are long or short?



- + The Lawson parameter  $n\tau T$  has doubled every two years.
- + This means 1000 times per 20 years or more
- + The previous slide showed that  $n\tau T$  has developed 1M times during a half century
- + This is a great development comparable to Moore's Law in super computer development

50 years have been necessary to develop plasma physics and fusion research for the peaceful usage of fusion energy

We have done our best, and we only need to maintain our effort for a much shorter period to insure that the goal is achieved

# Environmental destruction is proceeding severely Nowadays, an 80% reduction of CO<sub>2</sub> exhaust is necessary

We feel  
Our Progress  
Relief and  
Satisfaction



A pit dwelling in Japan 5000 years ago



The Arctic sea

Ancestors' home  
Descendents?

# Conclusion: Roles and Functions of Fusion Research

**This 22<sup>nd</sup> IAEA FEC will be recorded as a landmark conference addressing the environment problem of the Earth**

- Achieving long-term integration of physics and engineering necessary for energy development
- Promoting the development of research that follows the critical path
- Securing the basic sciences and supporting technologies necessary for fusion research
- Continually disseminating scientific results and leading the development of advanced science and technology in the field of nuclear fusion
- Steadily training necessary human resources

**The Dream is Alive  
Now Fusion Energy is an Achievable Goal!**