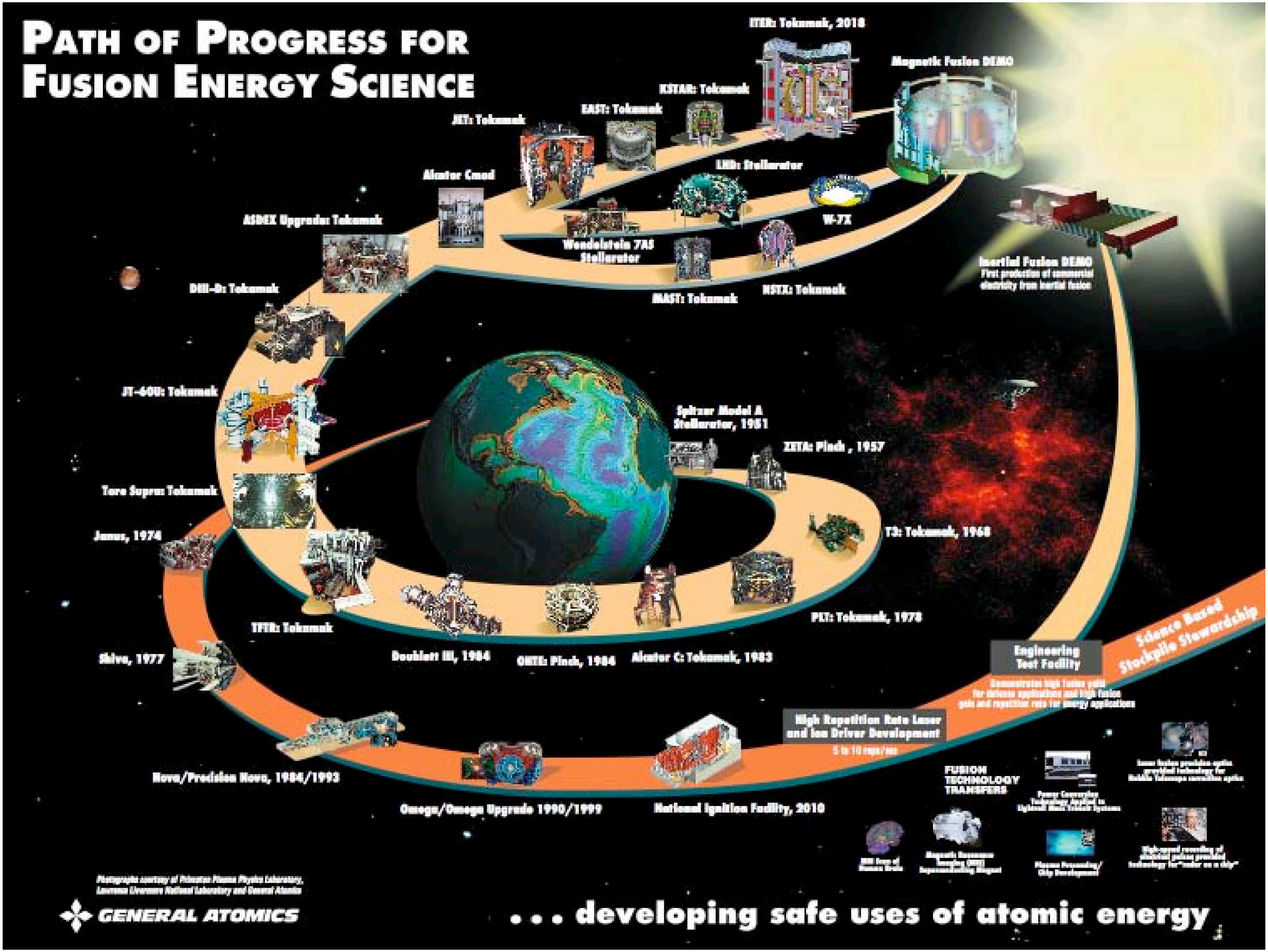
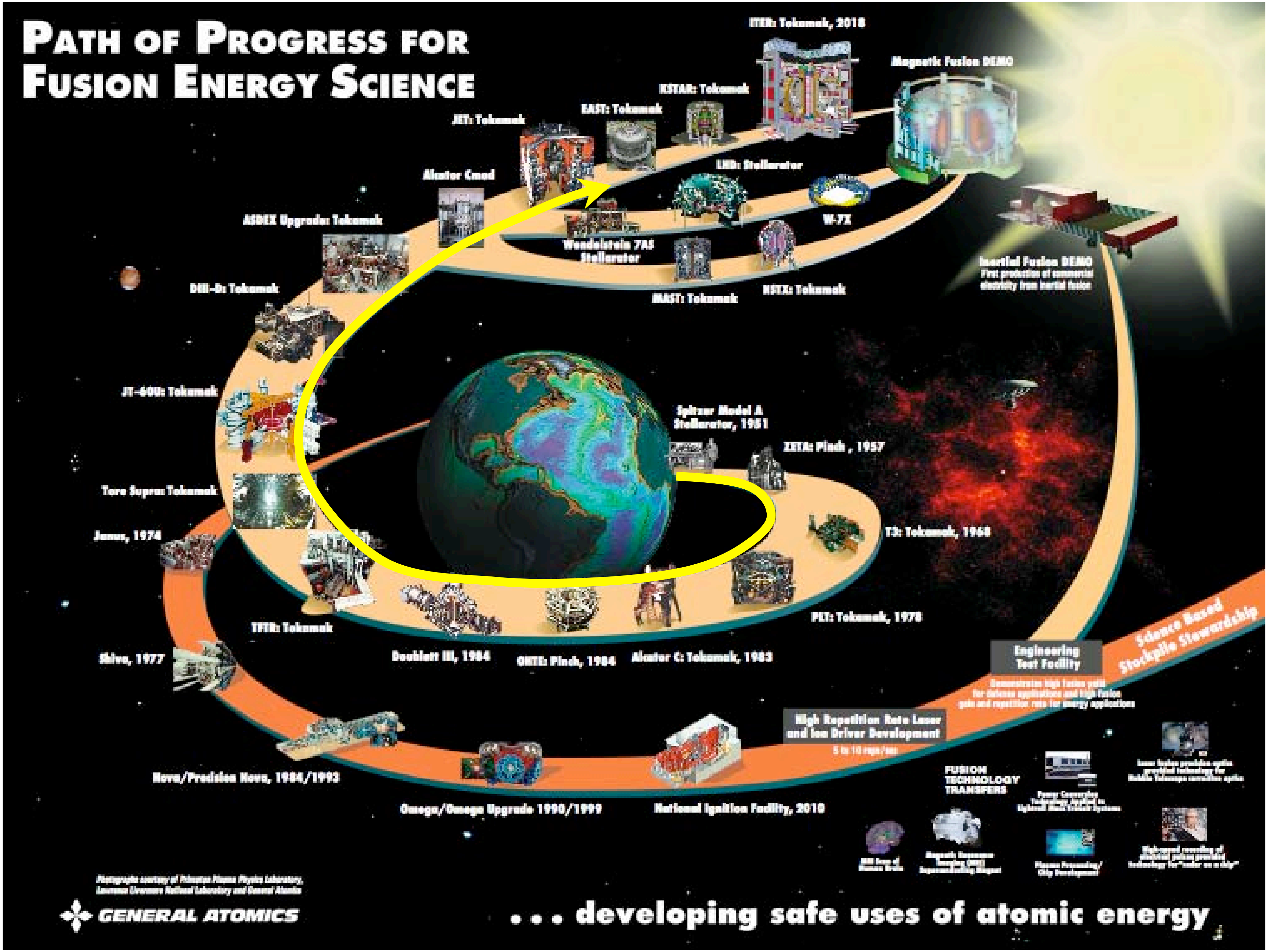
50 Years of Fusion Research

Dale Meade
Fusion Innovation Research and Energy®
Princeton, NJ

PPPL Colloquium
April 1, 2009
Princeton, NJ 08543



PATH OF PROGRESS FOR FUSION ENERGY SCIENCE
ITER: Tokamak, 2018
Magnetic Fusion DEMO
KSTAR: Tokamak
EAST: Tokamak
JET: Tokamak
LHD: Stellarator
Alcator Cmod
W-7X
ASDEX Upgrade: Tokamak
Wendelstein 7AS Stellarator
MAST: Tokamak
NSTX: Tokamak
Inertial Fusion DEMO
First production of commercial electricity from inertial fusion
DIII-D: Tokamak
JT-60U: Tokamak
Spitzer Model A Stellarator, 1951
ZETA: Pinch , 1957
Tore Supra: Tokamak
T3: Tokamak, 1968
Janus, 1974
PLT: Tokamak, 1978
TFTR: Tokamak
Shiva, 1977
Doublett III, 1984
OHTE: Pinch, 1984
Alcator C: Tokamak, 1983
Engineering Test Facility
Science Based Stockpile Stewardship
Demonstrate high fusion yield for defense applications and high fusion gain and repetition rate for energy applications
High Repetition Rate Laser and Ion Driver Development
5 to 10 reps/sec
Nova/Precision Nova, 1984/1993
Omega/Omega Upgrade 1990/1999
National Ignition Facility, 2010
FUSION TECHNOLOGY TRANSFERS
Photographs courtesy of Princeton Plasma Physics Laboratory, Lawrence Livermore National Laboratory and General Atomics
GENERAL ATOMICS
. . . developing safe uses of atomic energy



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Fusion Prior to Geneva 1958

- A period of rapid progress in science and technology
  - N-weapons, N-submarine, Fission energy, Sputnik, transistor, ...
- Controlled Thermonuclear Fusion had great potential
  - Much optimism in the early 1950s with expectation for a quick solution
  - Political support and pressure for quick results (but budgets were low, $56M for 1951-1958)
  - Many very "innovative" approaches were put forward
  - Early fusion reactors - Tamm/Sakharov, Spitzer
- Reality began to set in by the mid 1950s
  - Collective effects - MHD instability (1954)
  - Bohm diffusion was ubiquitous
  - Meager plasma physics understanding led to trial and error approaches
  - A multitude of experiments were tried and ended up far from fusion conditions
  - Magnetic Fusion research in the U.S. declassified in 1958

Requirements for Development of Fusion

- General issues understood very early

- Reactor plasma conditions ($n\tau_E \approx 3\mathrm{x}10^{20} \mathrm{m}^{-3}\mathrm{s}$, $T_i \sim 20$ keV, $Q \geq 25$)
  - confinement (turbulence), plasma heating

- Neutron Wall Loading ~ 4 MWm^{-2} (for economic attractiveness)
  - material damage ~ 40 dpa/yr with low radioactive waste
  - tritium breeding (TBR > 1) to complete the fuel cycle

- Fusion Power Densities (~ 5 MWm^{-3}, —> p ~ 10 atm)

  $\beta = \langle p \rangle / B_c^2$, MHD stability and coil engineering

- Plasma Wall Interaction -

  ~ 1 MW m^{-2} thermal load on wall

  low impurity levels, low tritium retention (< 0.5 kg-T)

- High-duty cycle, essentially steady-state

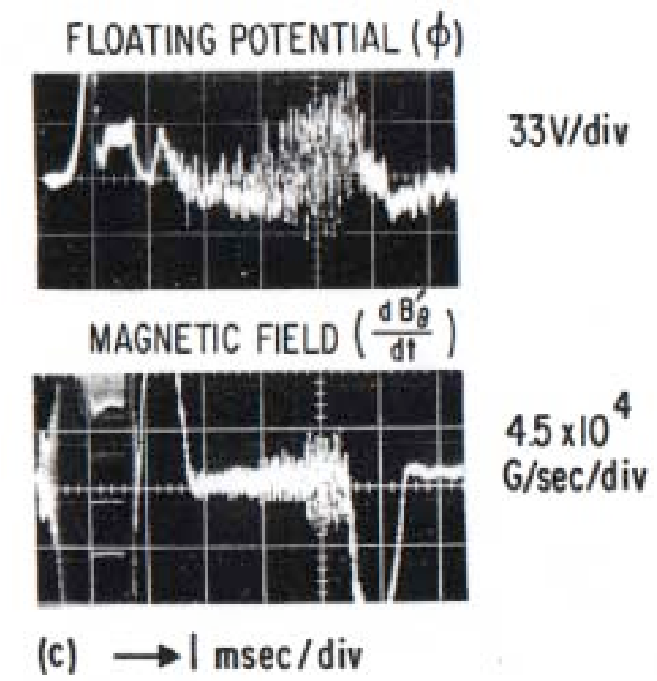
Fusion Plasma Physics, a New Scientific Discipline, was born in the 1960s

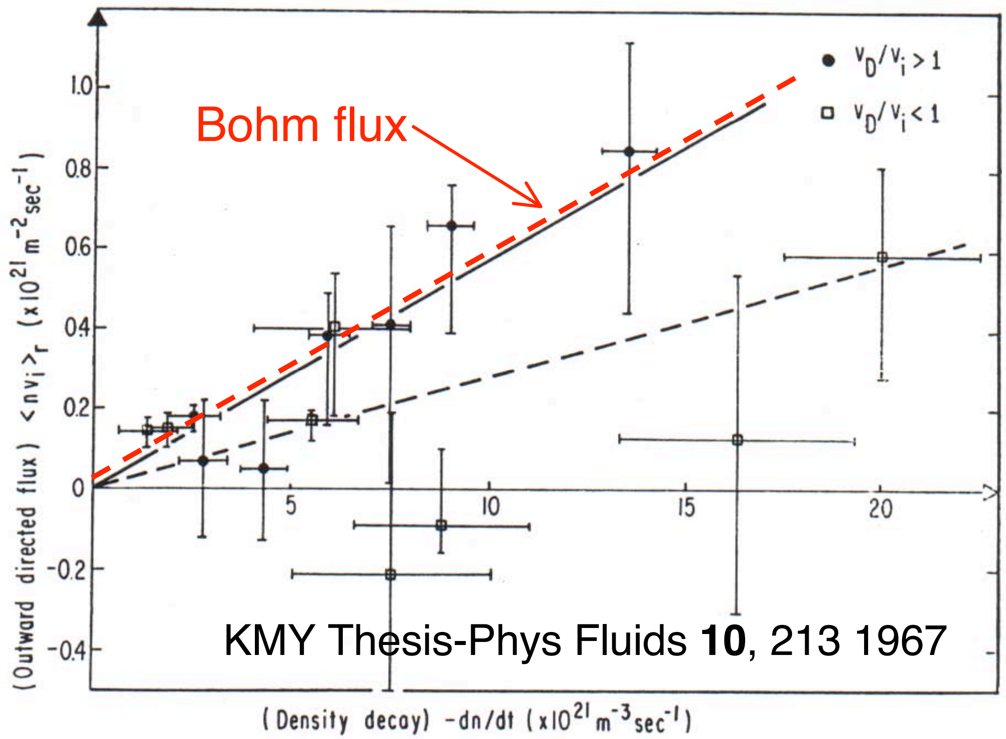
- Theory of Fusion Plasmas
  - Energy Principle developed in mid-50s became a powerful tool for assessing macro-stability of various configurations
  - Resistive macro-instabilities
  - Linear stability analyses for idealized geometries revealed a plethora of microinstabilities with the potential to cause anomalous diffusion Trieste School
  - Neoclassical diffusion developed by Sagdeev and Galeev
  - Wave propagation became basis for RF heating

- Experimental Progress (some examples)
  - Most confinement results were were dominated by instabilities and ~ Bohm diffusion
  - Stabilization of interchange instability by Min|B| in mirror - Ioffe
  - Stabilization of interchange in a torus by Min<B> in multipoles - Kerst/Ohkawa
  - Quiescent period in Zeta due to strong magnetic shear in self-organized state
  - Confinement gradually increased from 1 τ_B to 5-10 τ_B for low temp plasmas
  - Landau Damping demonstrated

The Early 1960s - The Depths of Despair for Toroidal CS

- The first stellarator experiments in the late ’50s were plagued with instabilities. Stellarators were limited by fluctuations causing “pump out, Bohm Diffusion or anomalous diffusion.”
- Model C was built to reduce complications of impurities (divertor) and wall neutrals (a = 5 cm). Experiments in 1961-66 confirmed Bohm diffusion.





Stabilization of MHD Interchange by Geometry (minimum |B|) in a Mirror Machine

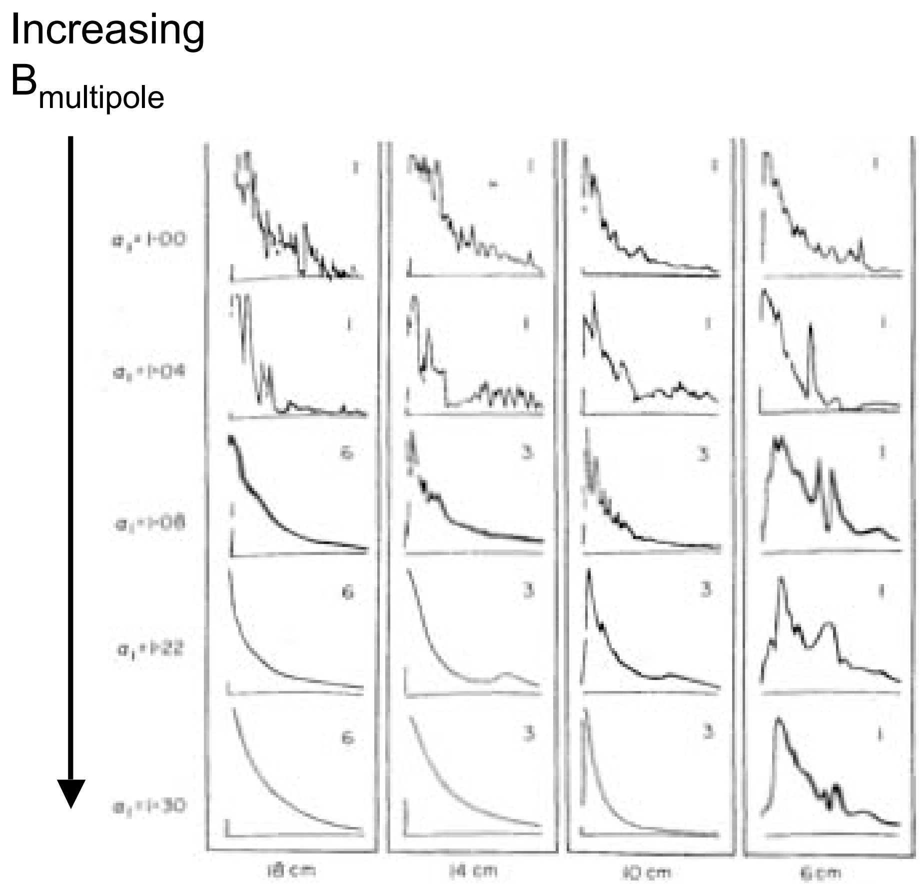
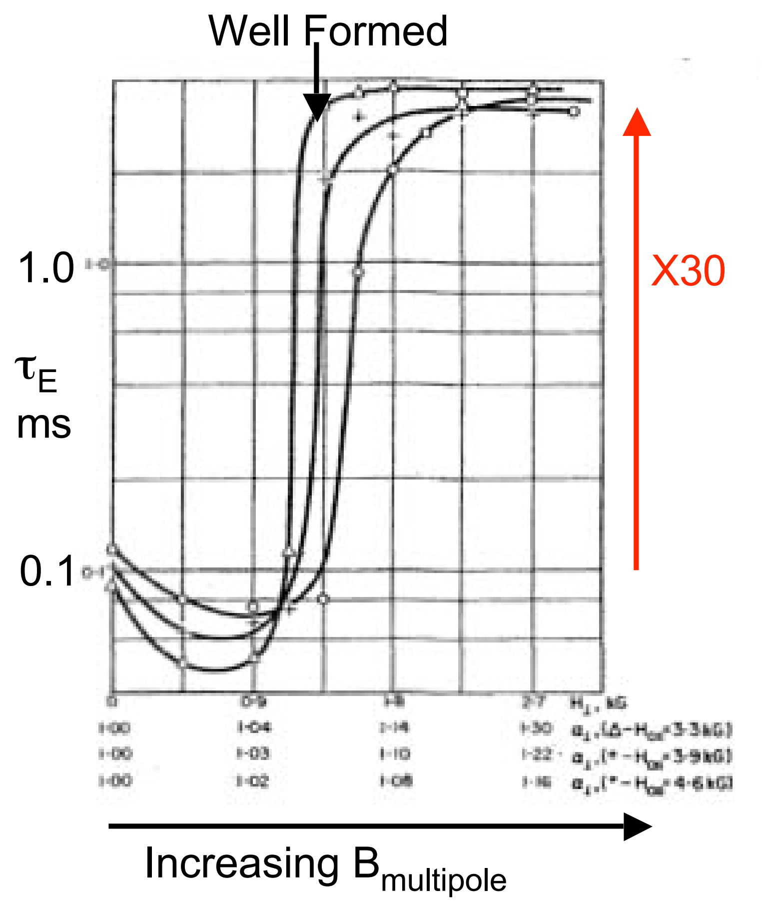


FIG. 8.—Oscillograms of ion current entering a Langmuir probe placed at various distances from the axis of the trap. (The figures in the right-hand corners of each oscillogram are the sweep lengths in milliseconds.)



- IOFFE IAEA Salzburg 1961, J Nuc Energy Pt C 7, p 501 1965

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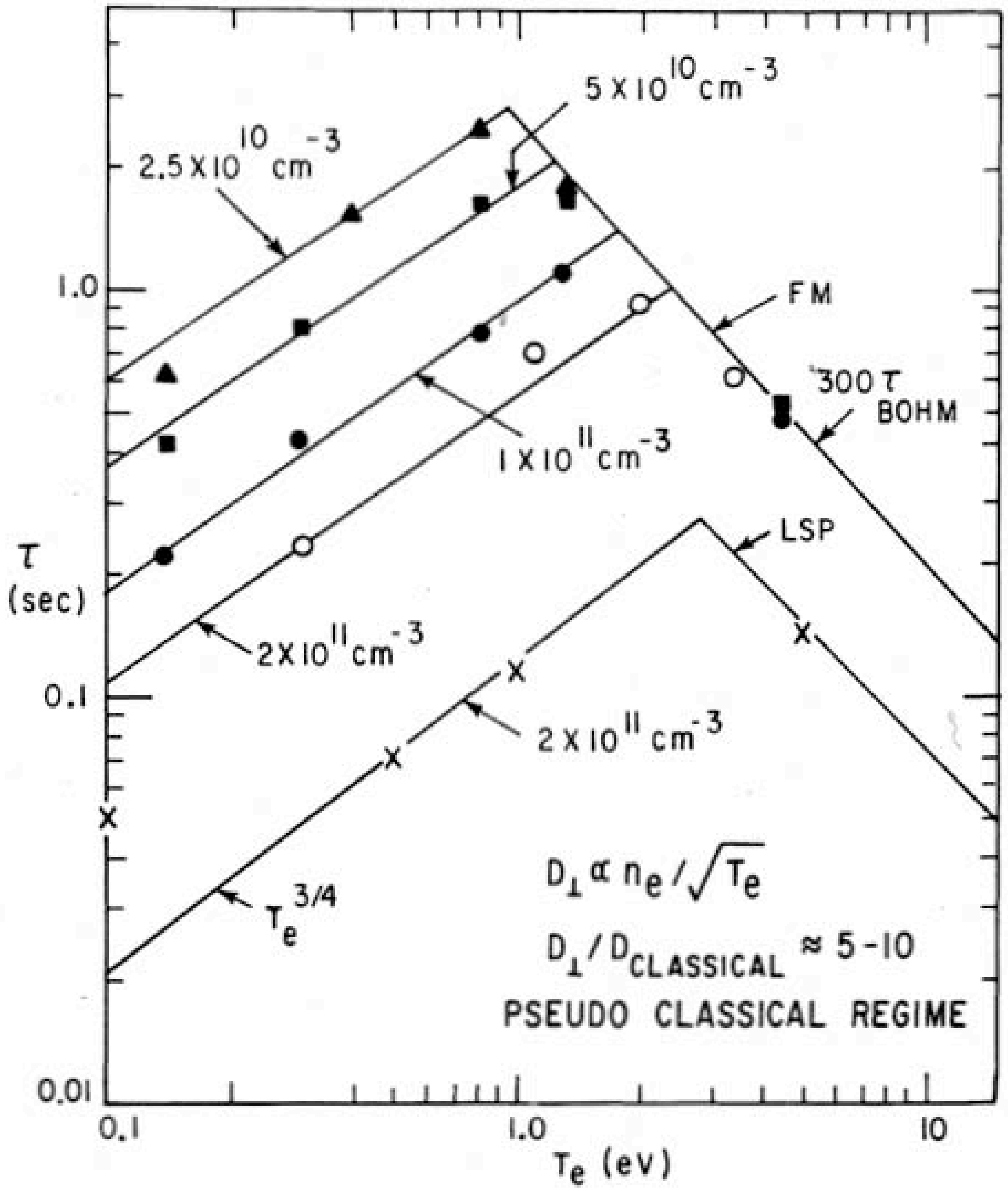
Enhanced Confinement in Low Temperature Plasmas



FM-1

LSP

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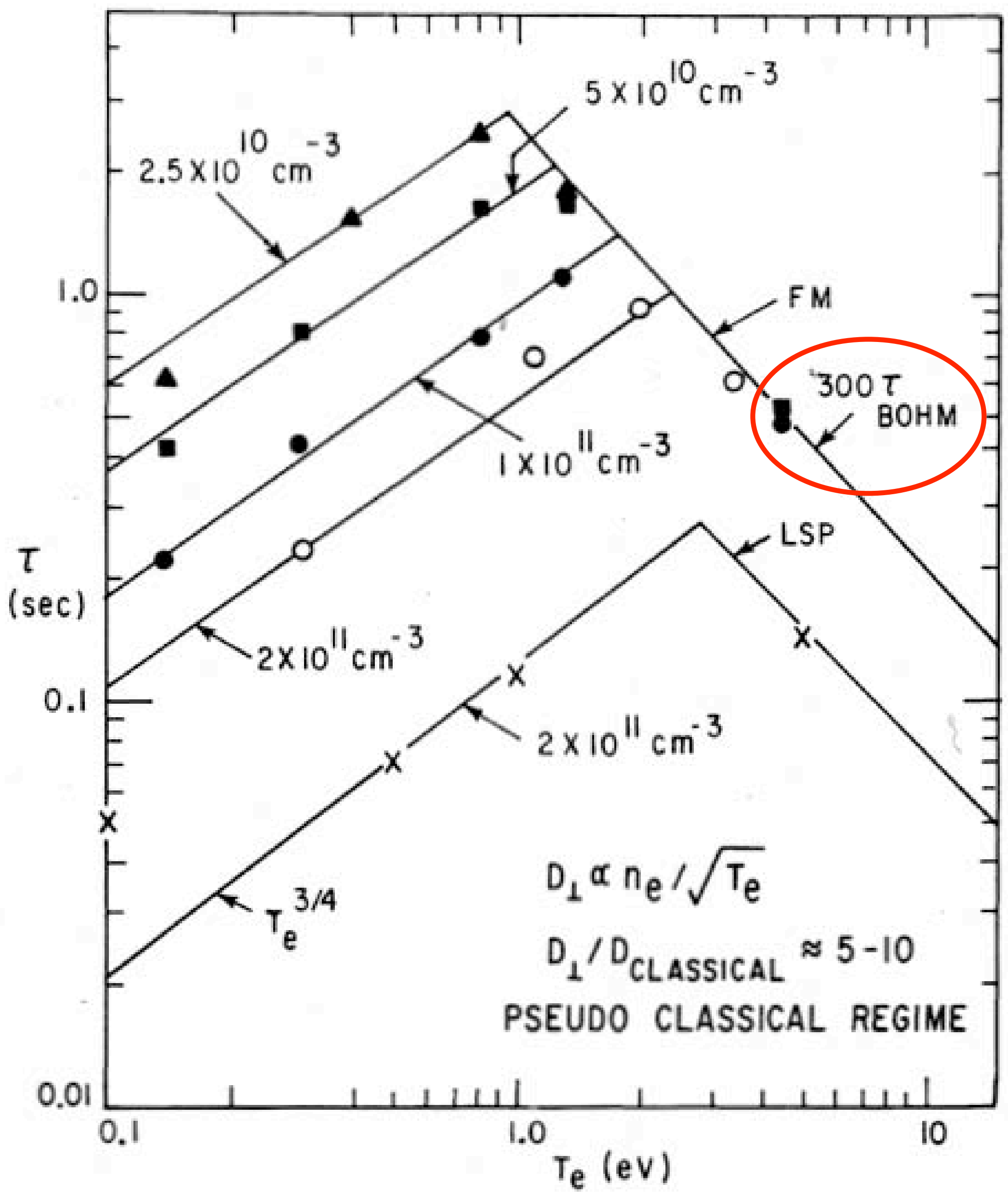
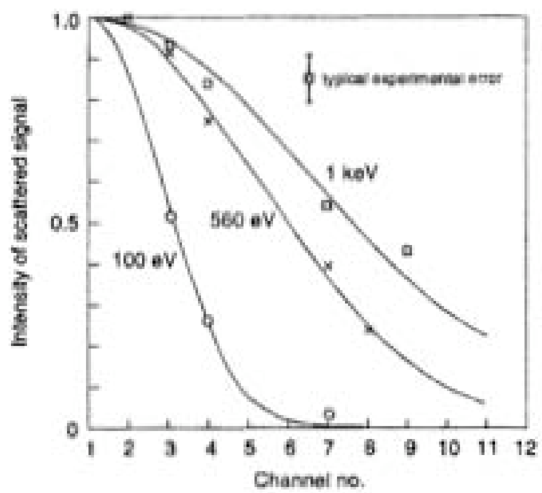


FM-1

LSP

1968-69 T-3 Breaks Bohm, Tokamaks Proliferate

- Hints of a major advance at IAEA Novosibirsk 1968, but skeptics abound
- Thomson Scattering (Peacock/Robinson) Dubna 1969 confirms $T_e \approx$ 1 keV

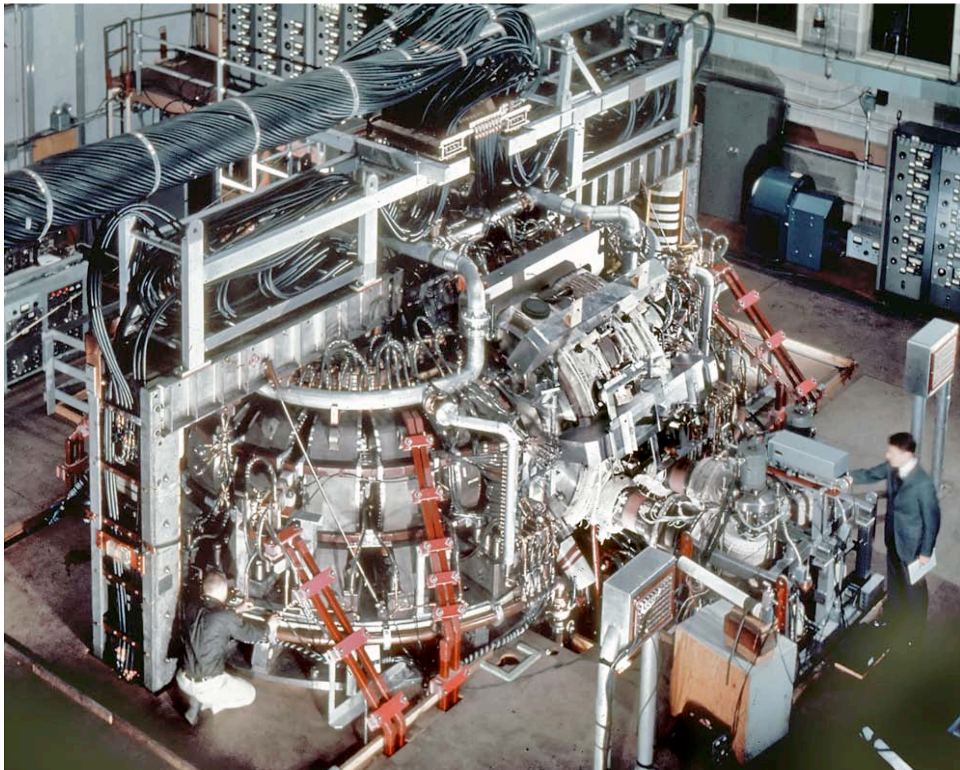


From Braams and Stott

- Energy confinement ≈ 30 τ_B - Bohm barrier broken for a hot plasma
- Skeptics converted to advocates overnight, the phone lines from Dubna to Princeton were busy with instructions to modify Model C.

40 years ago

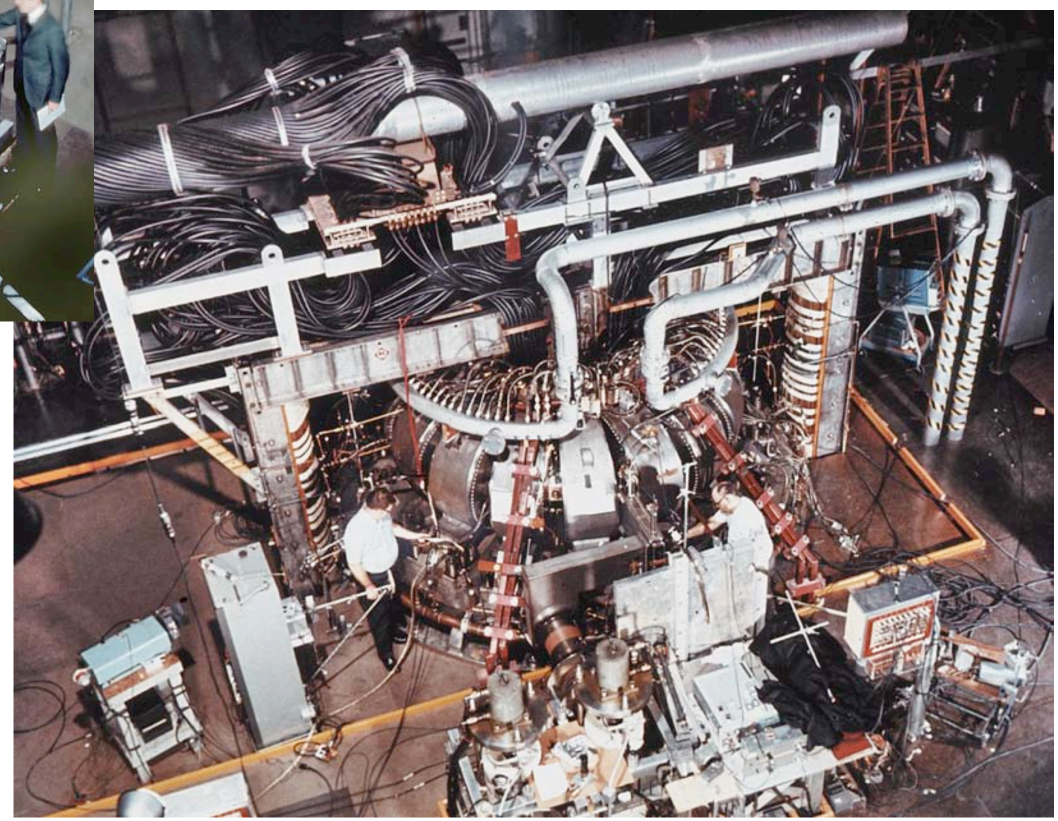
Model C Stellarator Converted to Tokamak in 6 months



Model C Stellarator
1969

T-3 results are quickly reproduced and extended.

Symmetric Tokamak (ST)
1970



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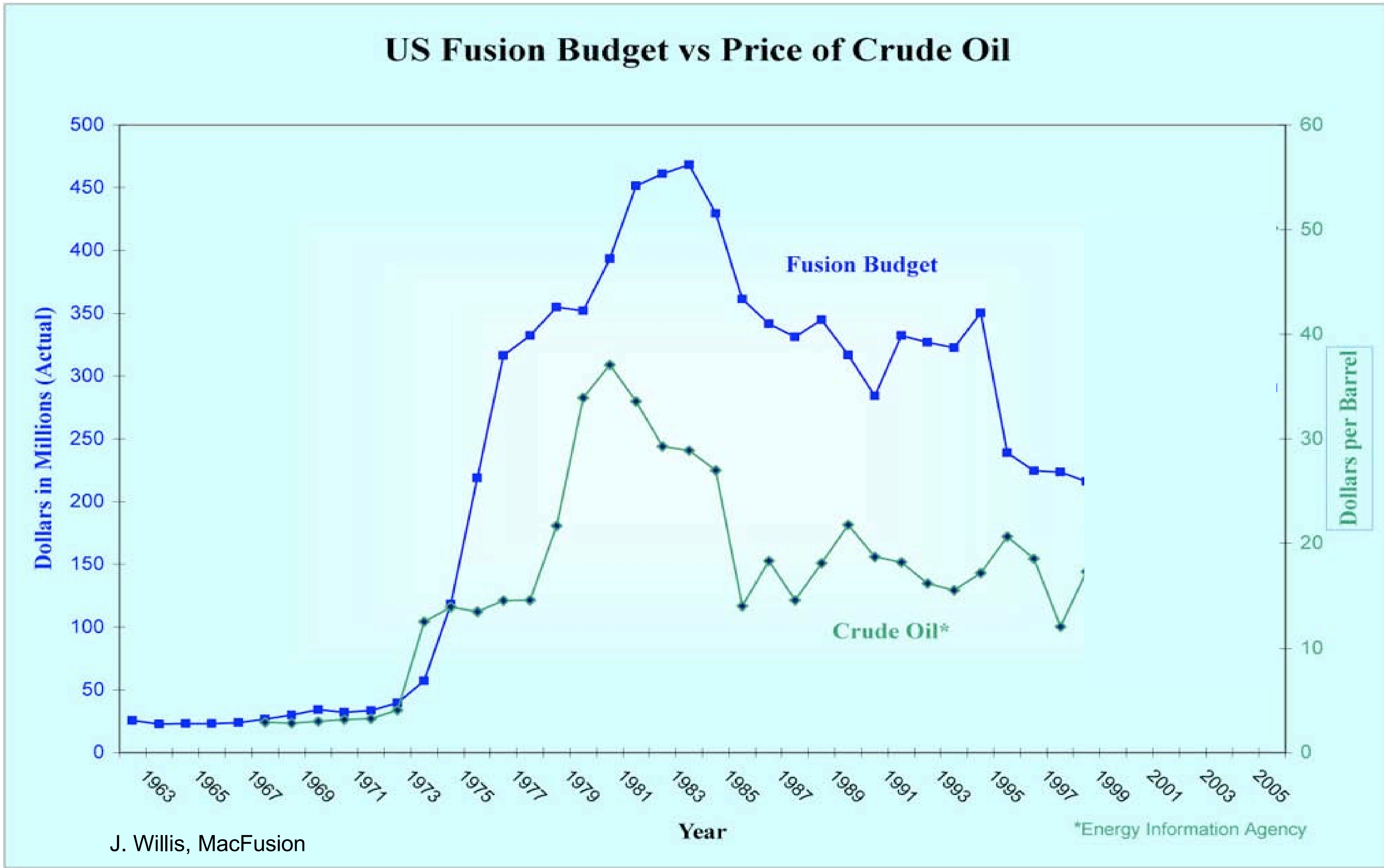
- Energy confinement $\approx 30\ \tau_B$ - Bohm barrier broken for a hot plasma
- Skeptics converted to advocates overnight, Model C Stellarator converted to Symmetric Tokamak (ST) in 6 months, T-3 results are quickly reproduced.
- During the 1970's ~ many medium size (I_p < 1 MA) tokamaks (TFR, JFT-2a, Alcator A, Alcator C, ORMAK, ATC, PLT, DITE, DIII, PDX, ASDEX, ... were built with the objectives of :
  - Confinement scaling with size, I_p, n, T,.......
  - Auxiliary heating (compression, ICRF, NBI, ECRH, LH)
  - Current Drive (LH, NBI, ...)
  - Impurity control (limiters, divertors)

40 years ago

Fusion was Prepared for a Major Next step when Opportunity Knocked (1973 Oil Embargo)

- Amid calls for increased energy R&D, Fusion budgets rise sharply
  - US Fusion budget increased a factor of 15 in 10 yrs.



Fusion was Prepared for a Major Next step when Opportunity Knocked (1973 Oil Embargo)

- Amid calls for increased energy R&D, Fusion budgets rise sharply
  - US Fusion budget increased a factor of 15 in 10 yrs.

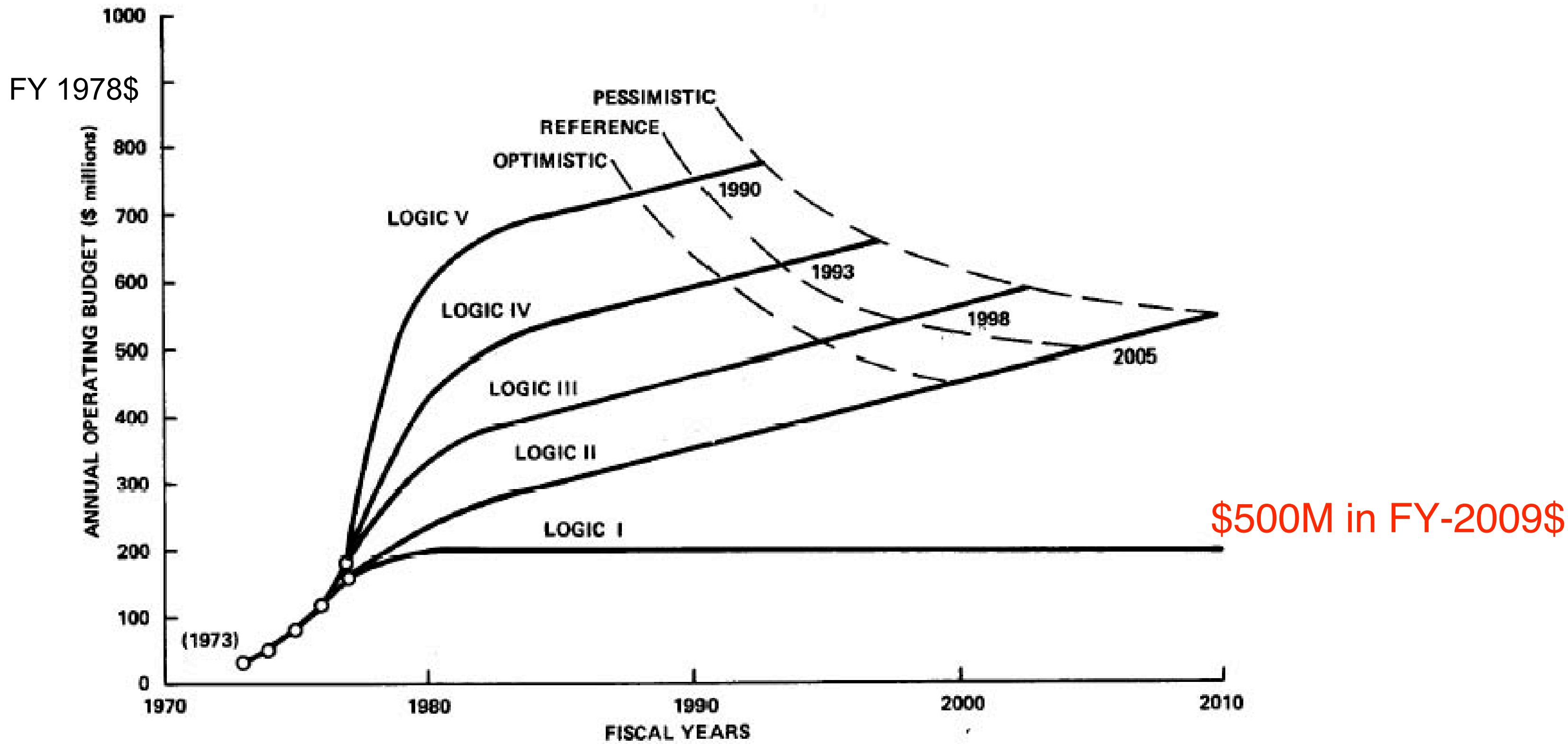
- Four Large Tokamaks approved for construction less than a decade after T-3
  - TFTR conservative physics/strong aux heating const began 1976
  - JET shaped plasma - const began 1977
  - JT-60 poloidal divertor- const began 1978
  - T-15 Superconducting TF (NbSn) const began 1979

These were very large steps, taken before all the R&D was completed.

Plasma Current	0.3 MA	=>	3MA to 7MA
Plasma Volume	1 m^3	=>	35 m^3 to 100 m^3
Auxiliary Heating	0.1 MW	=>	20 MW to 40 MW

1976 US Plan for Fusion

Fusion Power by Magnetic Fusion Program Plan July 1976 ERDA – 76/110/1

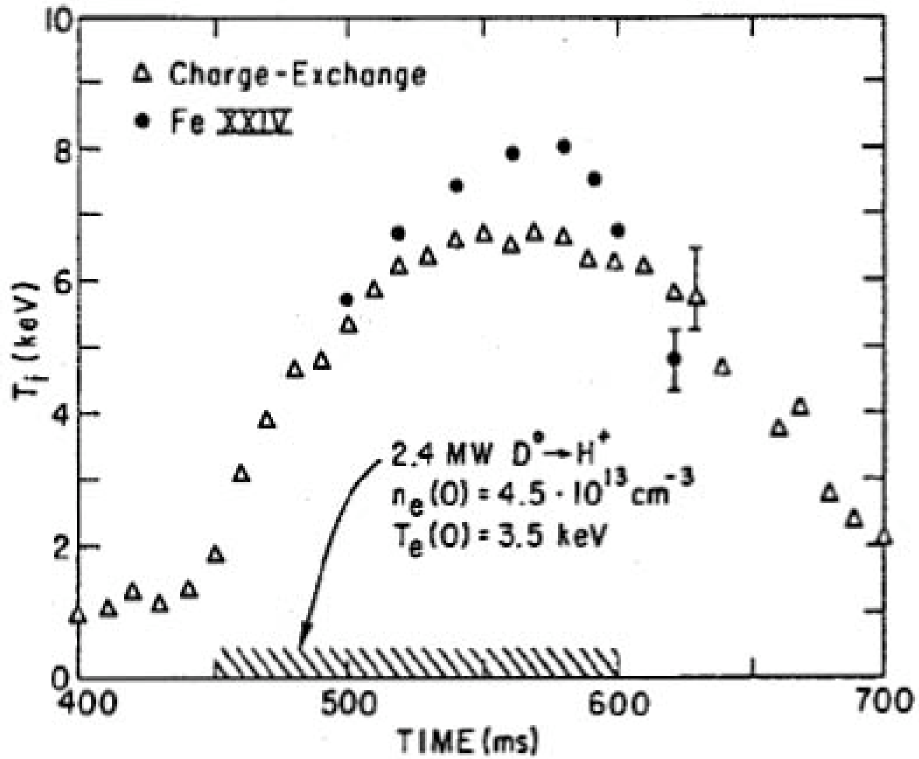


FUSION R&D PROGRAM OPERATING BUDGET AND LOCI OF DEMO OPERATING DATES FOR LOGIC I THRU V

- Logic IV became the basis for the MFE Act of 1980.

Optimism about Confinement Increased in the late 1970s

- Trapped Ion instabilities were predicted in the early 1970s to be a threat to the achievement high T_i in tokamak geometries.

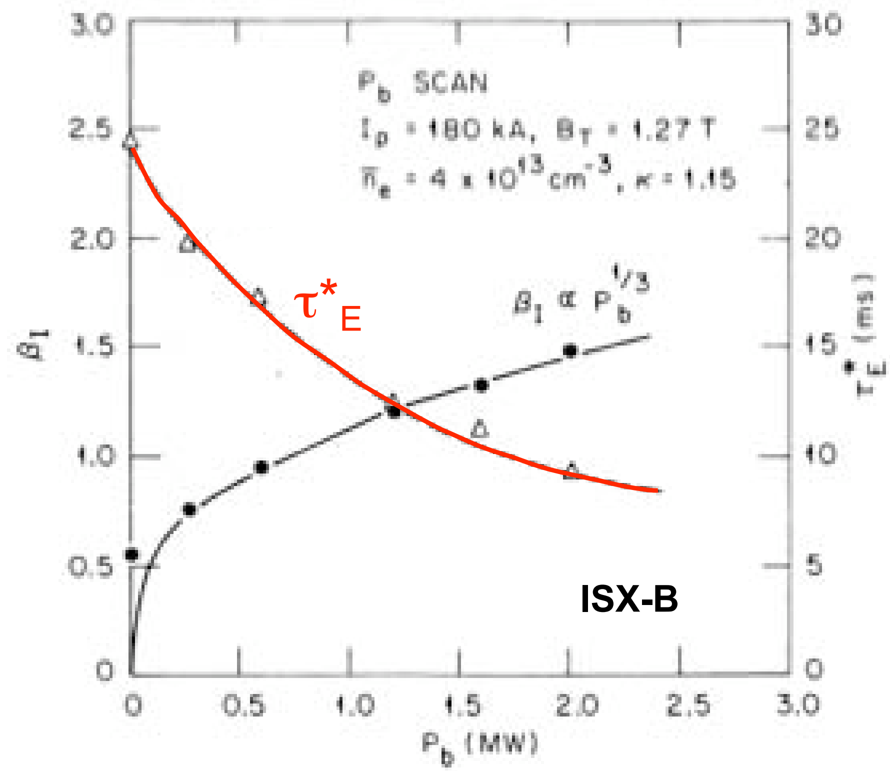


30 years ago

- In 1978, T_i ~ 5.8 keV was achieved in a collisionless plasma reducing concerns about Trapped Ion instabilities. T_i was increased to 7 keV in 1980.

- In ~ 1979 Alcator A with only ohmic heating achieved $n\tau_E \approx 1.5 \times 10^{19}$ m^{-3} s, consistent with optimistic scaling $\tau_E \sim na^2$.

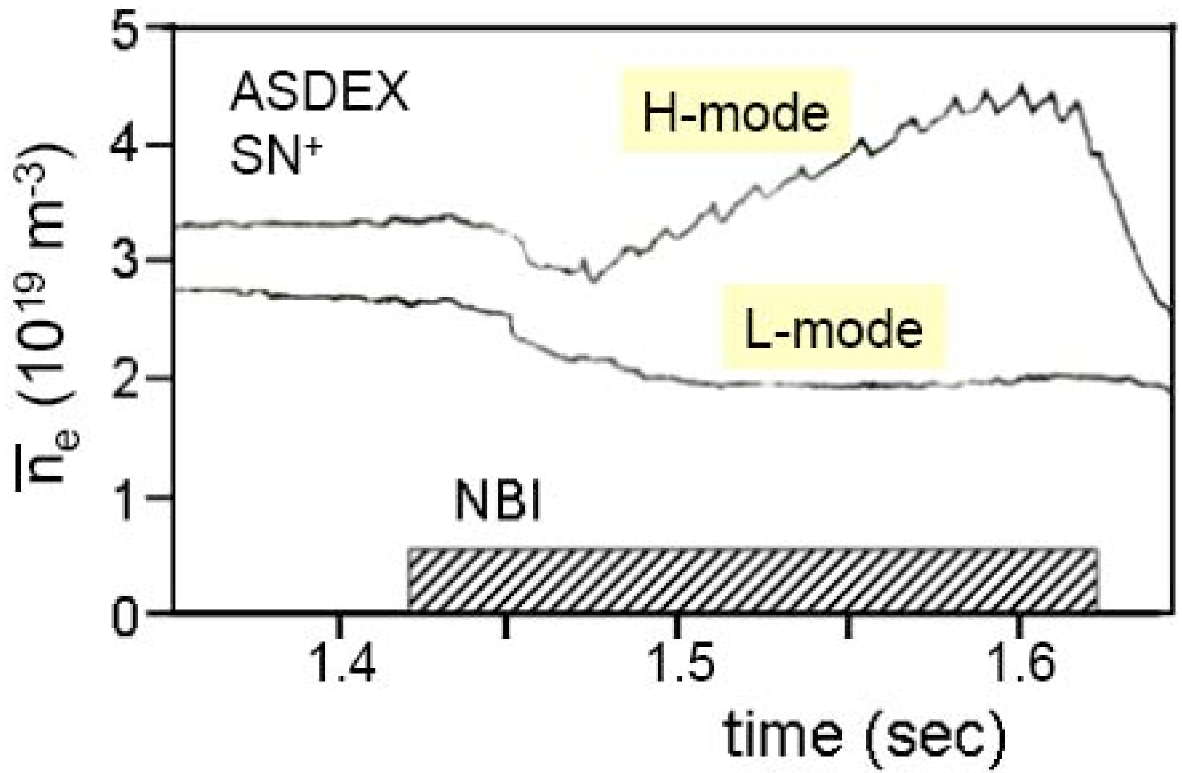
Auxiliary Heating Reveals New Trends 1981



- Auxiliary heating allowed controlled experiments to reveal the scaling of the global global confinement time.

- Confinement degradation observed as heating power was increased - Low mode scaling would threaten objectives of the large tokamaks, and tokamak based reactors.

H-Mode Discovered on ASDEX- 1982



P_{NBI} = 2.6 MW H in D
B_o = 2.2 T
I_p = 320 kA
Configuration: **SN+**

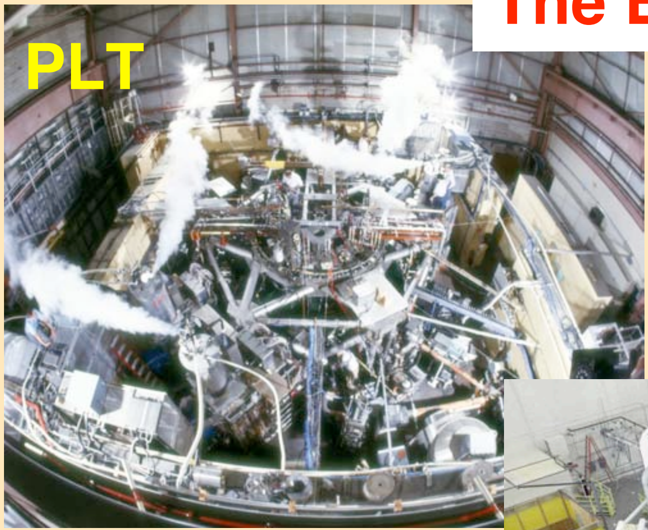
Two branches:

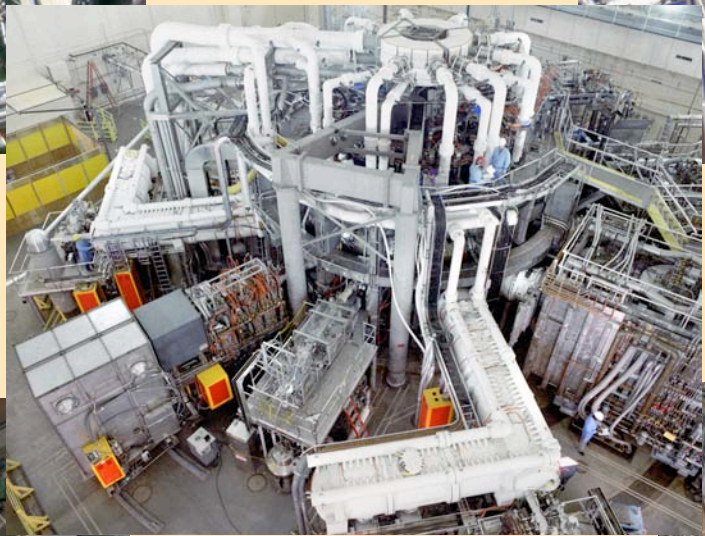
Type „a“: L-mode
Type „b“: H-mode

- Facilitated new insights and understanding of transport, and
- Provided the baseline operating mode for ITER

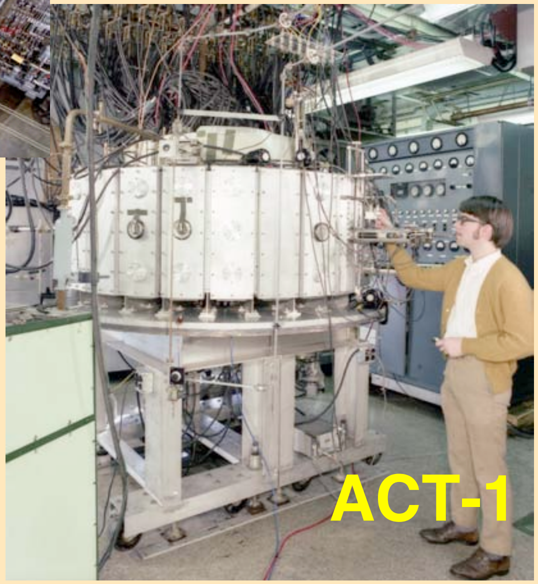
F. Wagner, IPP

The Early 1980s at PPPL





TFTR



Tokamak Optimization

- By the early 80s
  - It was clear tokamak performance would need to be improved, if the tokamak were to lead to an attractive fusion power source.
  - The benefits of cross-section shaping for increased confinement and beta were demonstrated and understood in Doublet IIA and Doublet III.
  - The β limit formulation by Troyon and Sykes provided a design guide for β.
  - Empirical scaling formulations (e.g., Goldston scaling) provided guidance for τ_E
  - An understanding of divertors emerged from JFT-2a, PDX, ASDEX, DIII, DITE.
- A second generation of flexible optimized tokamaks:
  DIII-D, AUG, JT-60U, PBX, Alcator C-Mod were built in the late 1980s to extend and develop the scientific basis for tokamaks.

Large Tokamaks Extend Plasma Parameters

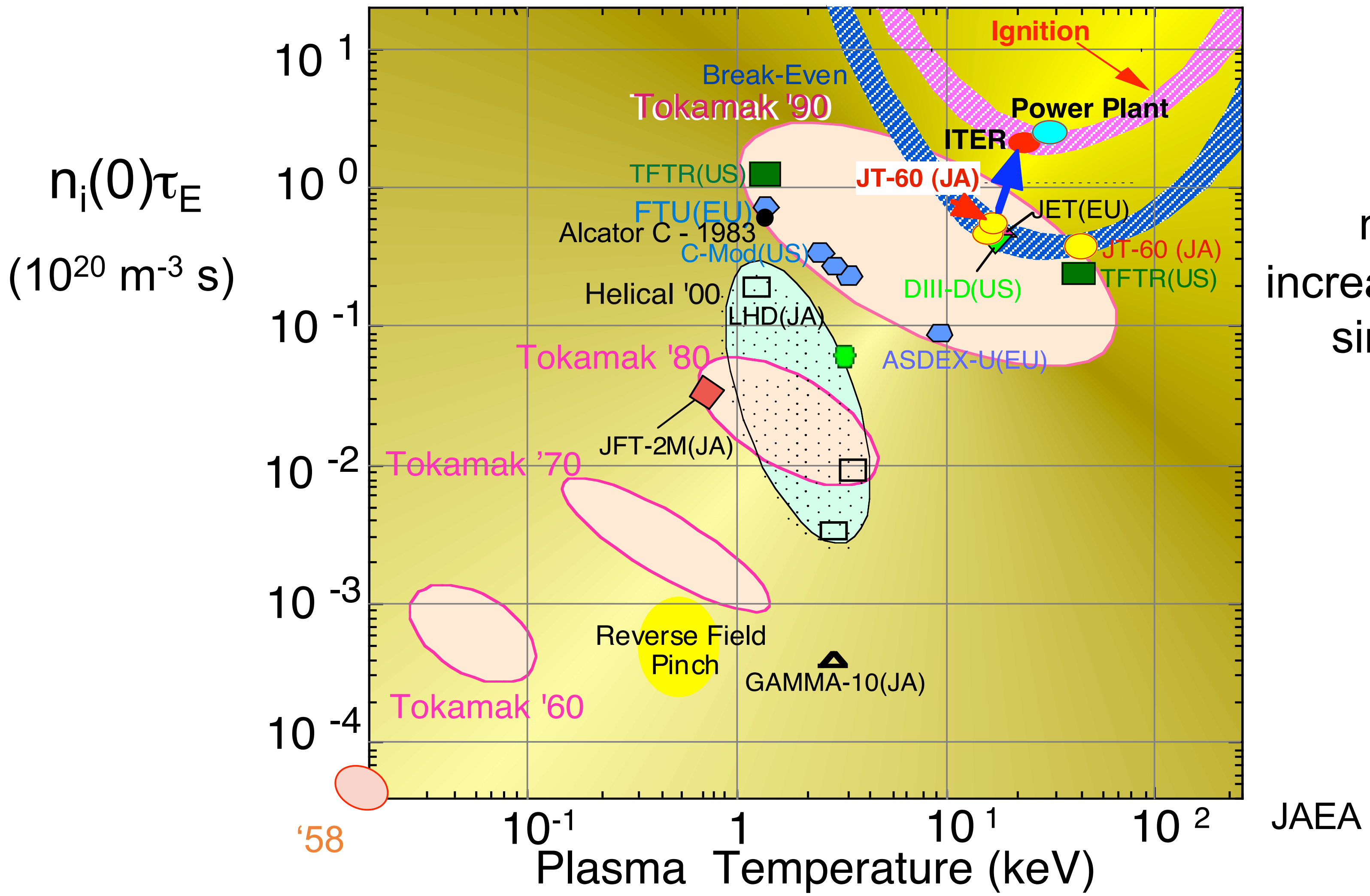
- After about 6 years of construction TFTR, JET and JT-60 began operation 1982-84.

- By the mid 80s, after 4 years of operation the plasma parameter range had been significantly extended
  - T_i~ 20 keV and $n_e(0)\tau_E$ ~ 1.5×10^{19} m^{-3} s with neutral beam injection
  - $n_e(0)\tau_E$ ~ 1.5×10^{20} m^{-3} s and T_i~ 1.5 keV with pellet injection
  - H-Mode extended to large tokamaks, new improved performance regimes discovered.
  - Bootstrap current and current drive extended to MA levels
  - Divertor extended to large scale

- Complex Technology demonstrated at large scale

- Enabling Technology - Neutral beams, pellet injection, PFCs

Fusion Temperatures Attained,
Fusion Confinement One Step Away



$n_i(0)\tau_E T_i$ increased by ~10^7 since 1958

Significant Fusion Power (>10MW) Produced 1990s

- **1991 JET 90/10-DT, 2 MJ/pulse, Q ~ 0.15, 2 pulses**

- **1993-97 TFTR 50/50-DT, 7.5MJ/pulse, 11 MW, Q ~ 0.3, 1000 D-T pulses**,
  - Alpha heating observed, Alpha driven TAEs - alpha diagnostics
  - ICRF heating scenarios for D-T
  - 1 MCi (100 g) of T throughput, tritium retention
  - 3 years of operation with DT, and then decommissioned.

- **Advanced Tokamak Mode Employed for High Performance**
  - Improved ion confinement TFTR, DIII-D, $Q_{DTequiv}$ ~ 0.3 in DIII-D 1995
  - $n\tau_E T$ record => $Q_{DTequiv}$ in JT-60U DD using AT mode 1996
  - Bootstrap and current drive extended

- **1997 JET 50/50-DT 22MJ/pulse, 16 MW, Q ~ 0.65, ~100 D-T pulses**
  - Alpha heating extended, ICRF DT Scenarios extended,
  - DT pulse length extended
  - Near ITER scale D-T processing plant
  - Remote handling

TFTR Fusion Achievements (1)

Official Objectives (1976):

1. Study plasma physics of large tokamaks,
2. Gain experience with reactor scale engineering,
3. Demonstrate D-T fusion energy production (1 to 10 MJ per pulse),

- First magnetic fusion experiment to achieve power plant fuel temperature of 200 million °C (20 kev) (1986), and ultimately a record 510 million °C (45kev) (1995)

- Record Lawson $n\tau_E \approx 1.5\mathrm{x}10^{20}$ m^{-3} s at 1.5 keV using pellet injection in deuterium plasmas

- First magnetic fusion experiment to use fusion power plant fuel mixture of 50% deuterium (D) and 50% tritium (T) (1993)

- First magnetic fusion experiment to produce fusion power exceeding 10 million watts (1994)

- Fusion Power Gain ~ 0.3, and factor of 10^6 greater than achieved when TFTR design started. the public goal was breakeven or Q ~ 1.

- Record peak plasma pressure of 6 atm in a D-T plasma, higher than that expected in ITER. (1994)

- Record fusion power densities of 0.3 million watts per cubic meter comparable to the 0.5 MWm^{-3} expected in ITER (1994)

TFTR Physics Achievements (2)

- First direct measurements of long wavelength turbulence that formed the basis for an improved theoretical modeling of turbulence (1990)
- First observation of the "bootstrap current" in a tokamak, a self generated current that is the key feature of steady-state tokamak fusion power plants (1986)
- First experimental observation of the "enhanced reversed shear" confinment mode that is now a key feature of steady-state tokamak power plant concepts. (1994)
- First demonstration of radio frequency heating of a D-T plasma using second harmonic tritium resonance, 1994.
- First demonstration of mode conversion heating in a D-T plasma, 1995.
- First observation of neoclassical tearing modes that limit output of high performance fusion plasmas (1995)
- First unambiguous measurements of self-heating by alpha particles in a DT fusion plasma (1995)
- First measurements of instabilities excited by fusion alpha particles (1996).

TFTR Engineering Achievements (3)

- First to identify and quantify the retention of tritium in graphite plasma facing components as a major issue for fusion power plants using trace tritium (1989) and 50/50 DT (1995)

- First closed cycle processing of tritium on a fusion experiment using 50% deuterium (D) and 50% tritium (T) (Feb 1997)

- About 1 million curies of tritium was handled safely over a three year period during over 1,000 DT experiments.

- TFTR engineering systems operated above the original design ratings for toroidal field and neutral beam power.

- The total cost of the TFTR program (design, construction, operation and decommissioning was $1.65 B, or 22% of the the US fusion program from 1975 to 1997.

The Next Step Burning Plasmas

- 1980 - Fusion Engineering Device (FED), SC or Cu coils, 200 MW, 200s as part of MFE Act to be competed after expenditure of $1.6B

- 1984 - Tokamak Fusion Core experiment (TFCX), SC coils, 200 MW, ss estimated cost $1.7B - cancelled too expensive

- 1986 - Compact Ignition Tokamak (CIT), LN Cu coils - 400 MW, 5 s, $0.7B

- 1989 - CIT was in FY89 budget with PACE funding for design, but was withdrawn by DOE (Hunter) when ignition could not be guaranteed.

Compact Ignition Tokamak (1985-1989)

CIT PARAMETERS AND OPERATIONAL LIMITS

	TOKAMAK	POWER SUPPLIES
MAJOR RADIUS	2.1 M	
MINOR RADIUS	0.65 M	
ASPECT RATIO	3.25	
ELONGATION (95% SURFACE)	2.0	
FIELD ON AXIS	10 T*	7 T
CURRENT @ $q = 3.1$	11 MA	7.7 MA
NEUTRON WALL LOADING @ 0.8 BETA LIMIT	6.0 MW/M^2	
TF FLAT-TOP TIME	5 SEC	
OHMIC HEATING	54 VOLT-SEC.	
ENERGY/PEAK POWER	11.9 GJ/1300 MW	6.2 GJ/600 MW

*A LIMITED NUMBER OF 11 T DISCHARGES IS ALSO AVAILABLE.

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Based on today's understanding, CIT would have "ignited" with Q = 35 using a conservative H98(y,2) = 0.92 !!!!

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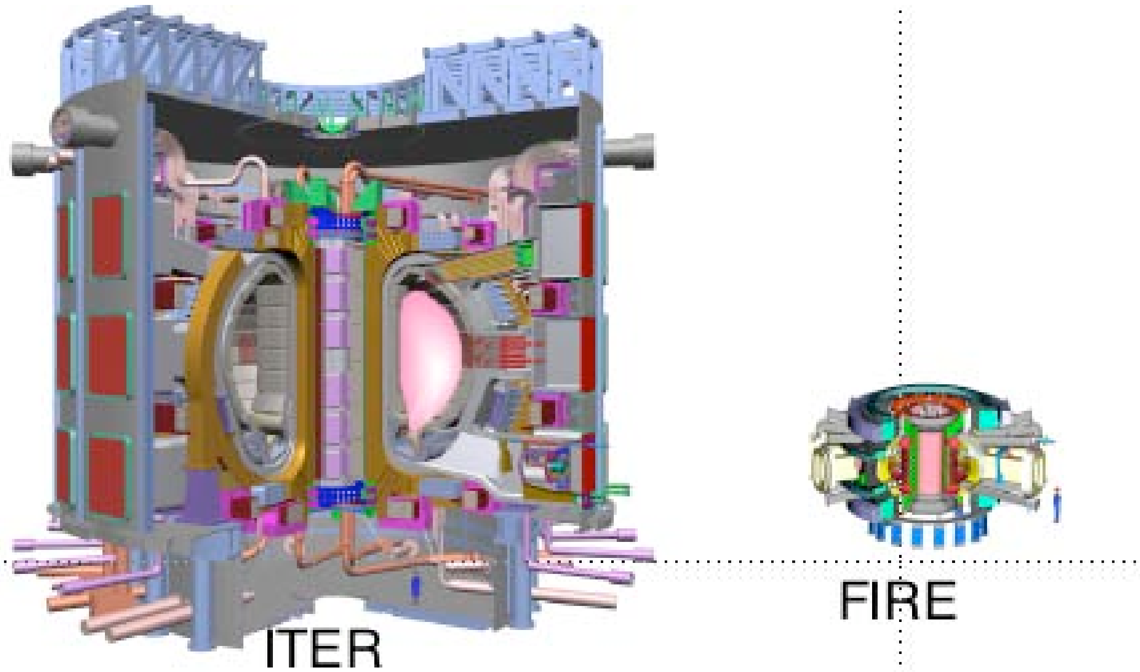
- 1990 - BPX a larger CIT with less ambitious goals and higher cost was put forward - cancelled in Sept 1991(SEAB, Townes Panel) on to TPX
- 1992 - ITER - US joins ITER as one of four partners, has lead design center
- 1997 - US leaves ITER after completion of Engineering Design Activity
- 1998 - US initiates study of advanced CIT called FIRE

Burning Plasmas in the 21st Century



Snowmass 2002 - assessment of ITER , FIRE and Ignitor

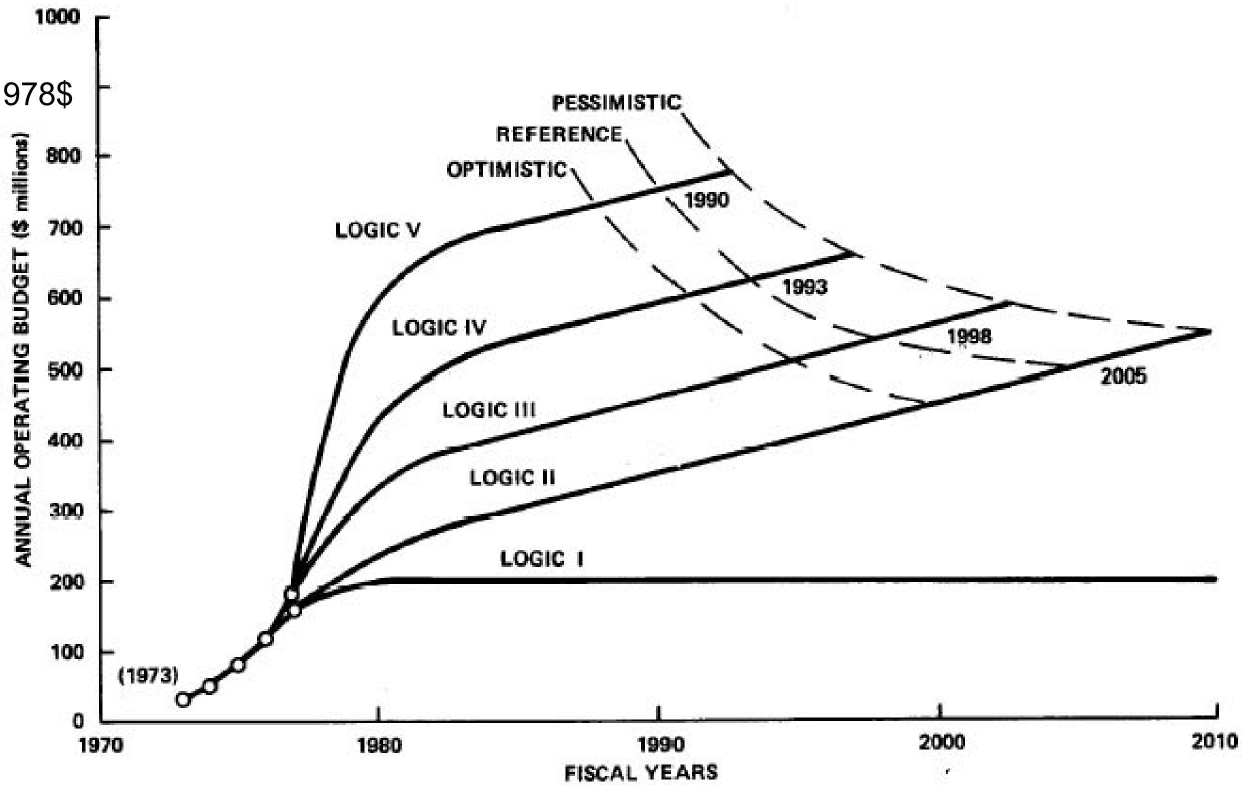
- ITER/FIRE - similar H-Mode and AT physics capability
- ITER design complete with >80 procurement packages ready to go to industry
- ITER total cost estimate = $5B, FIRE total cost estimate = $1.2B

US cost (10%) = $0.5B US Cost (100%) = $1.2B

Why Couldn't US MFE Take the Next Step?

Fusion Power by Magnetic Fusion Program Plan   July 1976   ERDA – 76/110/1

FY 1978$



FUSION R&D PROGRAM OPERATING BUDGET AND LOCI OF DEMO OPERATING DATES FOR LOGIC I THRU V

- Logic IV became the basis for the MFE Act of 1980.
- The US Fusion Program evolved on to Logic I - we never get there.

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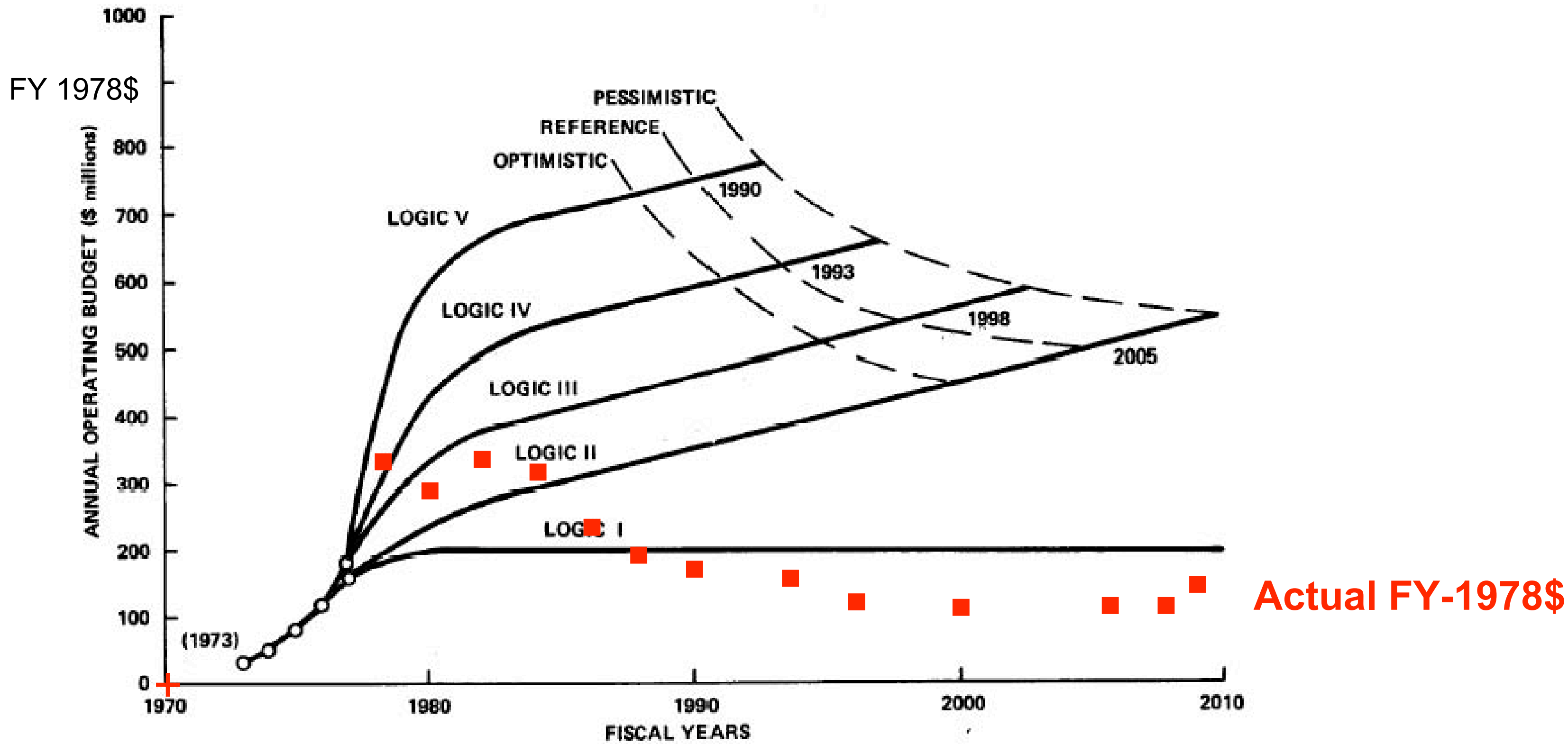


FUSION R&D PROGRAM OPERATING BUDGET AND LOCI OF DEMO OPERATING DATES FOR LOGIC I THRU V

- Logic IV became the basis for the MFE Act of 1980.
- The US Fusion Program evolved on to Logic I - we never get there.

The Next Challenge -
Sustainment of Fusion Plasma Conditions

- **Steady-state operation is a highly desirable** characteristic for a magnetic fusion power plant. This requires:
  - Sustained magnetic configuration
    - The **stellarator** (helical) configuration is inherently steady-state, or
    - **Advanced tokamak** with high bootstrap current fraction and moderate external current drive is also a possible steady-state solution.
  - Effective removal of plasma exhaust and nuclear heat
    - Power density and distribution of removed power
    - Effect of self conditioned PFC on plasma behavior
- Helical/Stellarator Resurgence
  - Confinement, beta approaching tokamak
  - Opportunities for configuration optimization
- Long Pulse Superconducting tokamaks - T-7, T-15, Tore Supra, TRIAM, EAST, KSTAR, SST-1, JT-60SA

Realizing The Advanced Tokamak

- Plasma cross-section shaping to enhance plasma current, power production
  - 1968 Ohkawa (Plasma Current Multipole), 1973 T-9 Finger Ring,

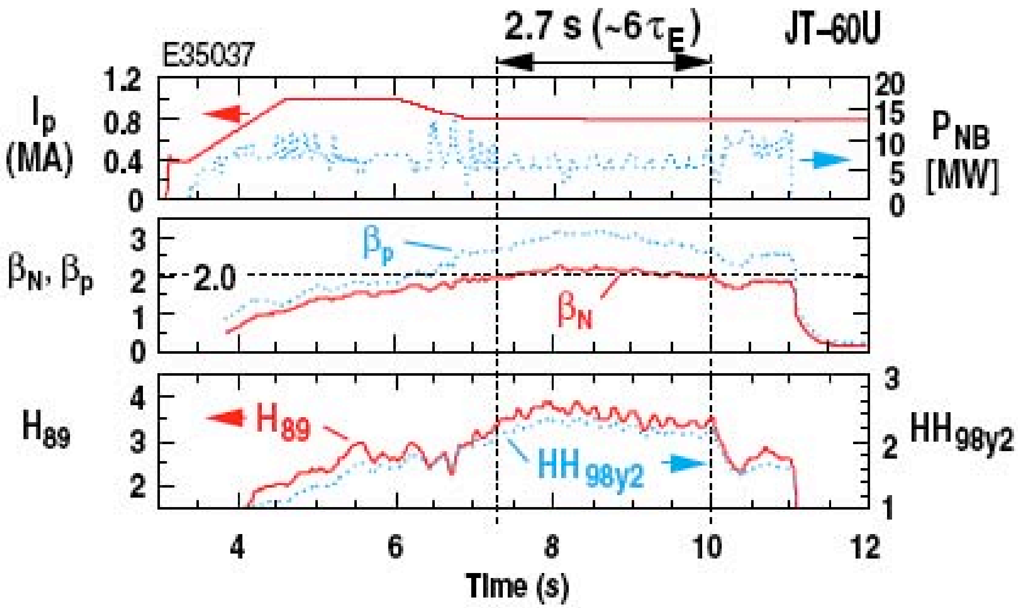
Tihiro Ohkawa with toroidal multipole at GA 1966

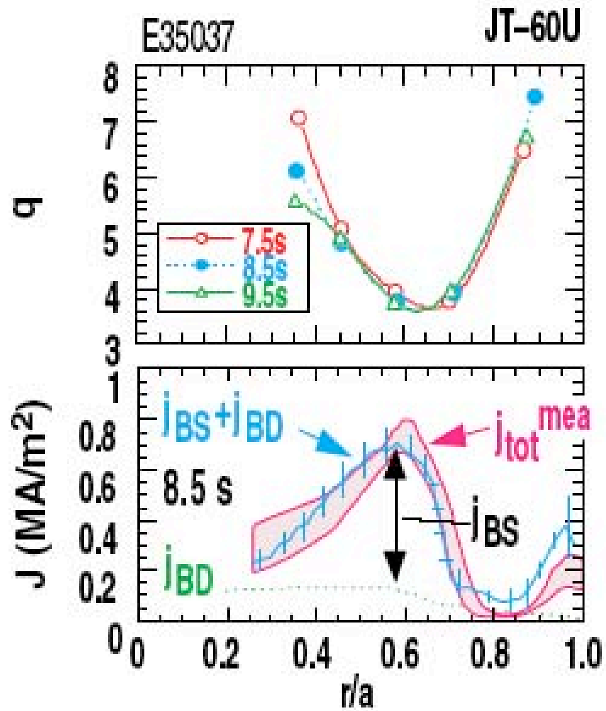
Realizing The Advanced Tokamak

- Plasma cross-section shaping to enhance plasma current, power production
  - 1968 Ohkawa (Plasma Current Multipole), 1973 T-9 Finger Ring,
  - 1990s Spherical (low aspect ratio) Tokamaks
- Bootstrap Current (self generated current)
  - Predicted 1971 - Bickerton
  - First observation 1983 in a mulitpole exp't - Zarnstorff/Prager
  - Observed in 1986 in tokamak -TFTR - Zarnstorff
- Beta limit physics "understood" for tokamak
  - $\beta = \beta_N (I_p/aB)$ where $\beta = \langle p \rangle / \langle B^2 \rangle$, 1983, Troyon, Sykes
  - NTM Stabilization by ECRH ASDEX Upgrade, DIII-D or Reversed Shear
  - Resistive Wall Stabilization DIII-D ~2005
- Confinement enhancement by stabilizing ITG using Reversed Shear
- Reversed shear with a hollow current profile provides the above:
  - PEP modes on JET 1988
  - ERS modes on TFTR 1994
  - NCS modes on DIII-D 1994
  - RS modes on JT-60U 1995 - record $nT\tau$
  - But all were transient

A HIGH PERFORMANCE PLASMA WITH FULL NON–INDUCTIVE CURRENT DRIVE AND 80% BOOTSTRAP FRACTION IN JT–60U

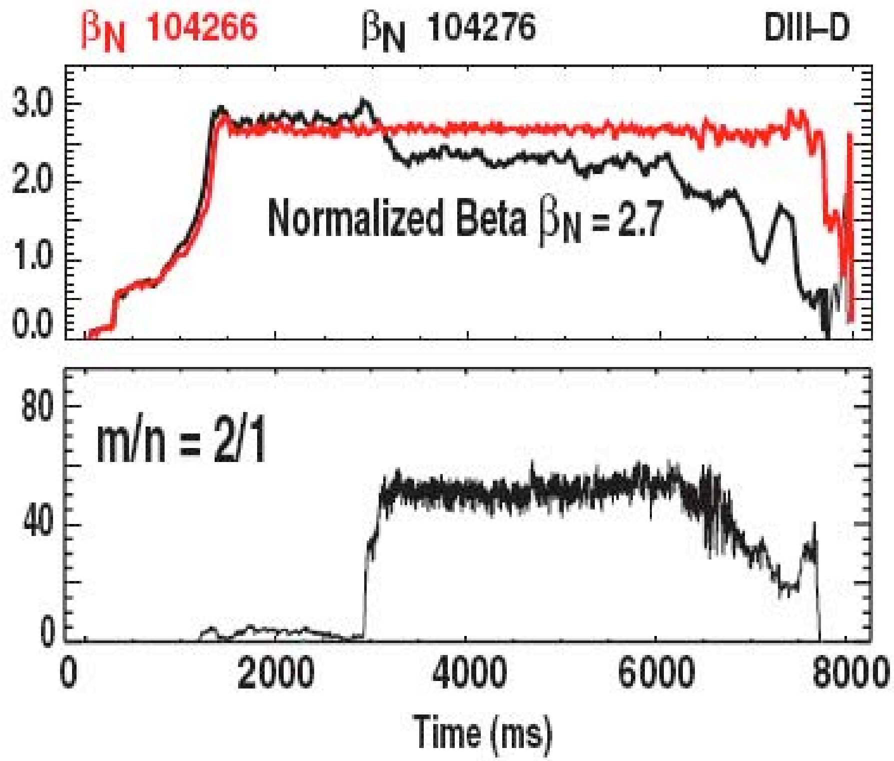
- H_{89}~3.5, HH_{98y2}~2.2, β_N~2, β_p~2.9, f_{BS}~80% for $6\tau_E$ with full non-inductive CD
- Current profile was largely determined by the bootstrap current, and was nearly stationary





JT 60 also 80% bootstrap fraction

PRECISE CONTROL NEAR THE β–LIMIT IS THE KEY TO AVOIDING DISRUPTIONS

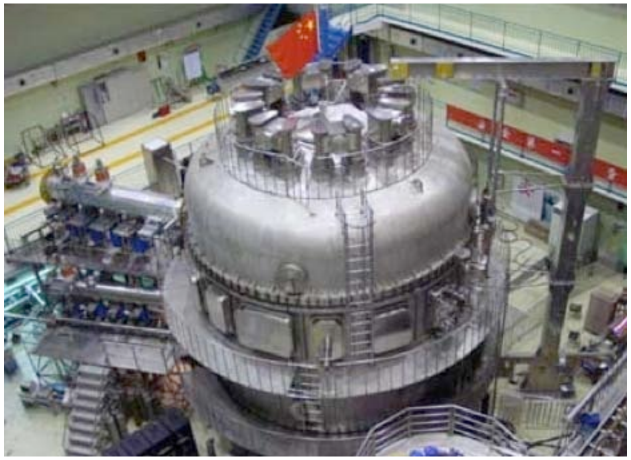


1. Need to operate close to stability limits
   - Good control
   - Knowledge of limits

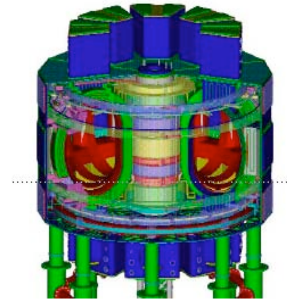
   High performance DIII–D discharge regulated 5% below 2/1 tearing limit for 35 τ_E (6.3 seconds)

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

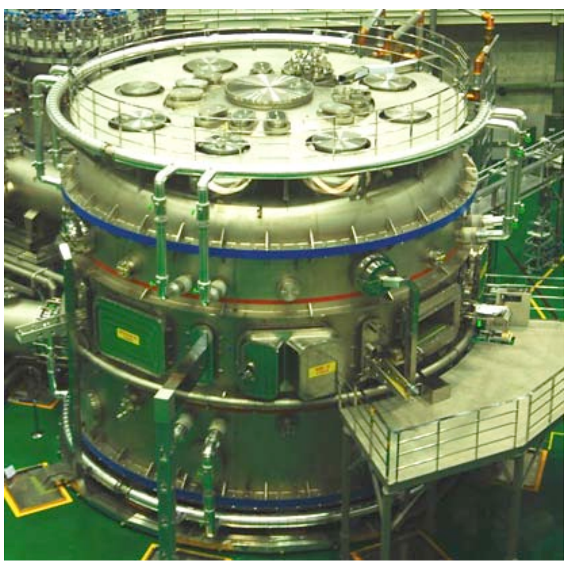
Four New Superconducting Tokamaks will Address Steady-State Advanced Tokamak Issues in Non-Burning Plasmas



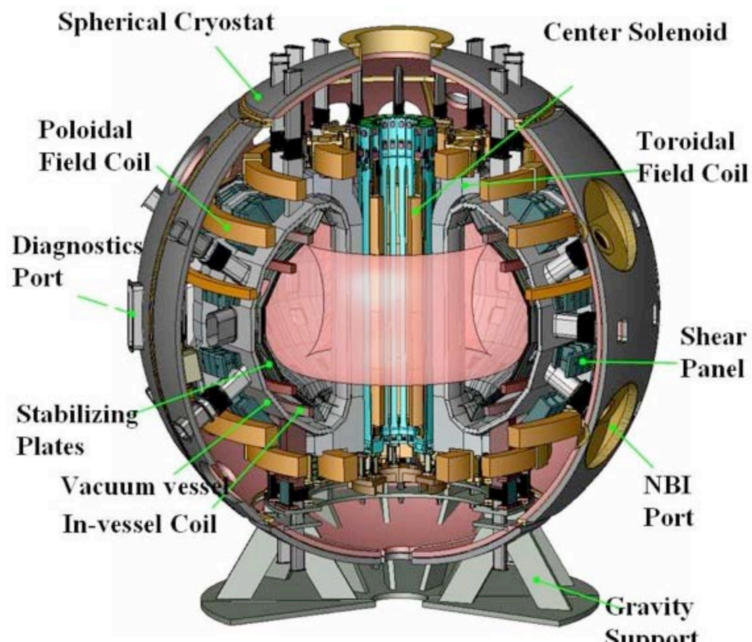
EAST: R = 1.7m, 2MA, 2006



SST-1: R =1.1m, 0.22MA, 2008

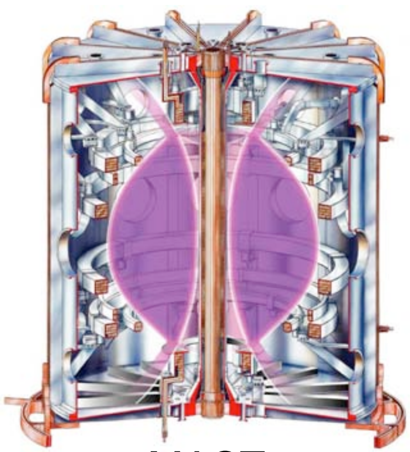


KSTAR: R = 1.8m, 2MA, 2008

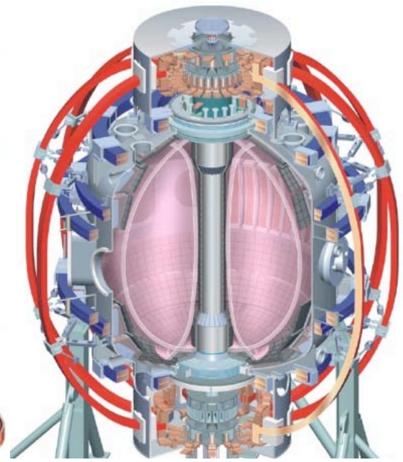


JT-60SA: R = 3m, 5.5 MA, 2014

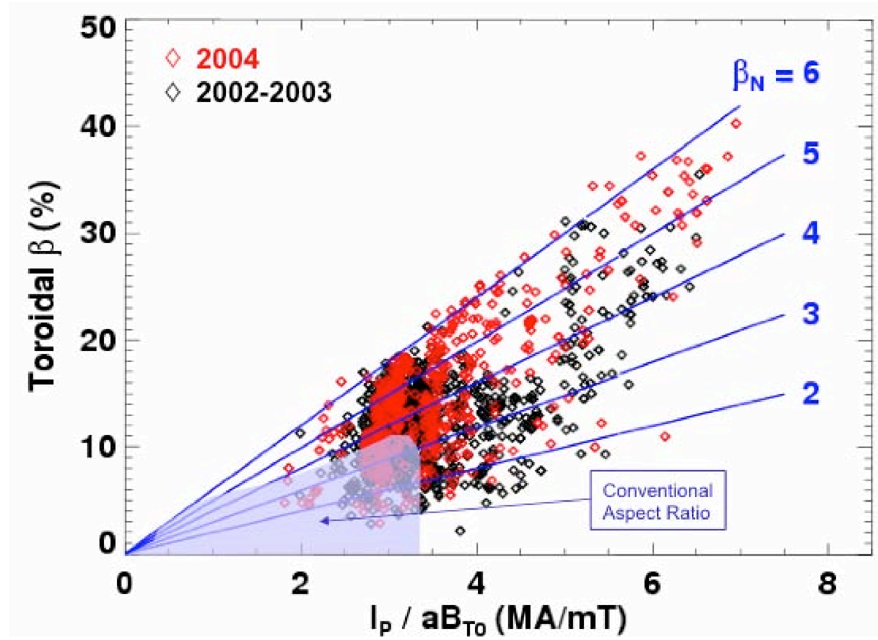
Optimizing the 2-D Geometry of a Tokamak



MAST



NSTX



- Higher β-limits at lower aspect ratio recognized in mid 1960s
- $\beta_t \approx 40\%$ achieved in START 1991-96 and NSTX 2004
- What is the optimum aspect ratio for overall system performance?
- Very Low aspect ratio may allow a Cu TF coil engineering solution in a D-T environment

The Stellarator/Helical (3-D) Systems

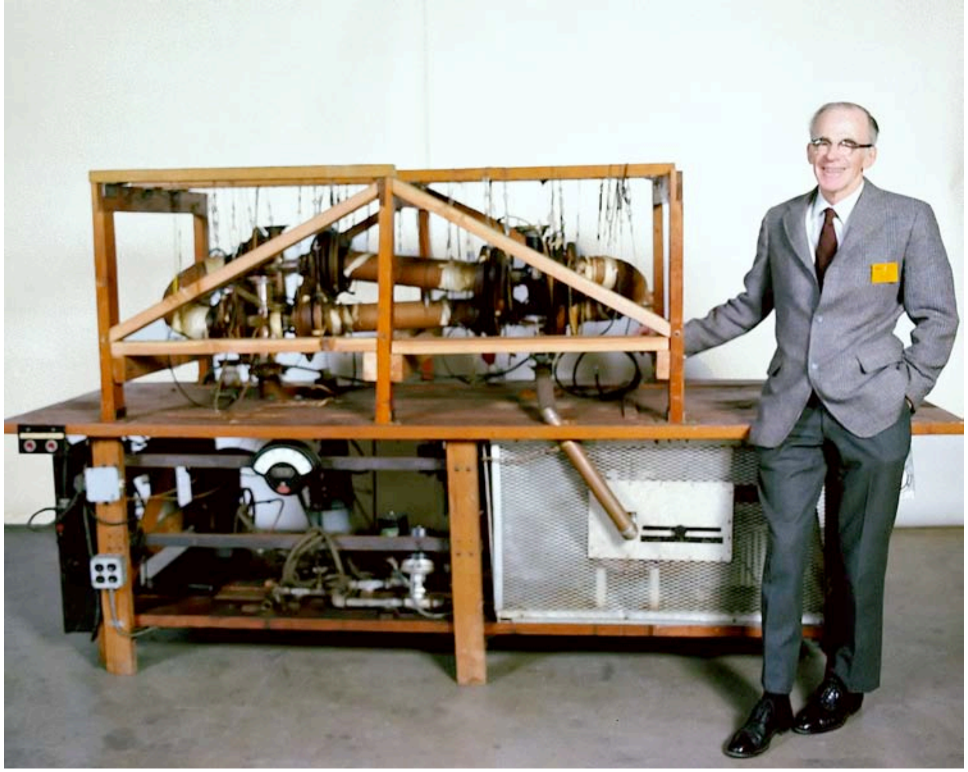


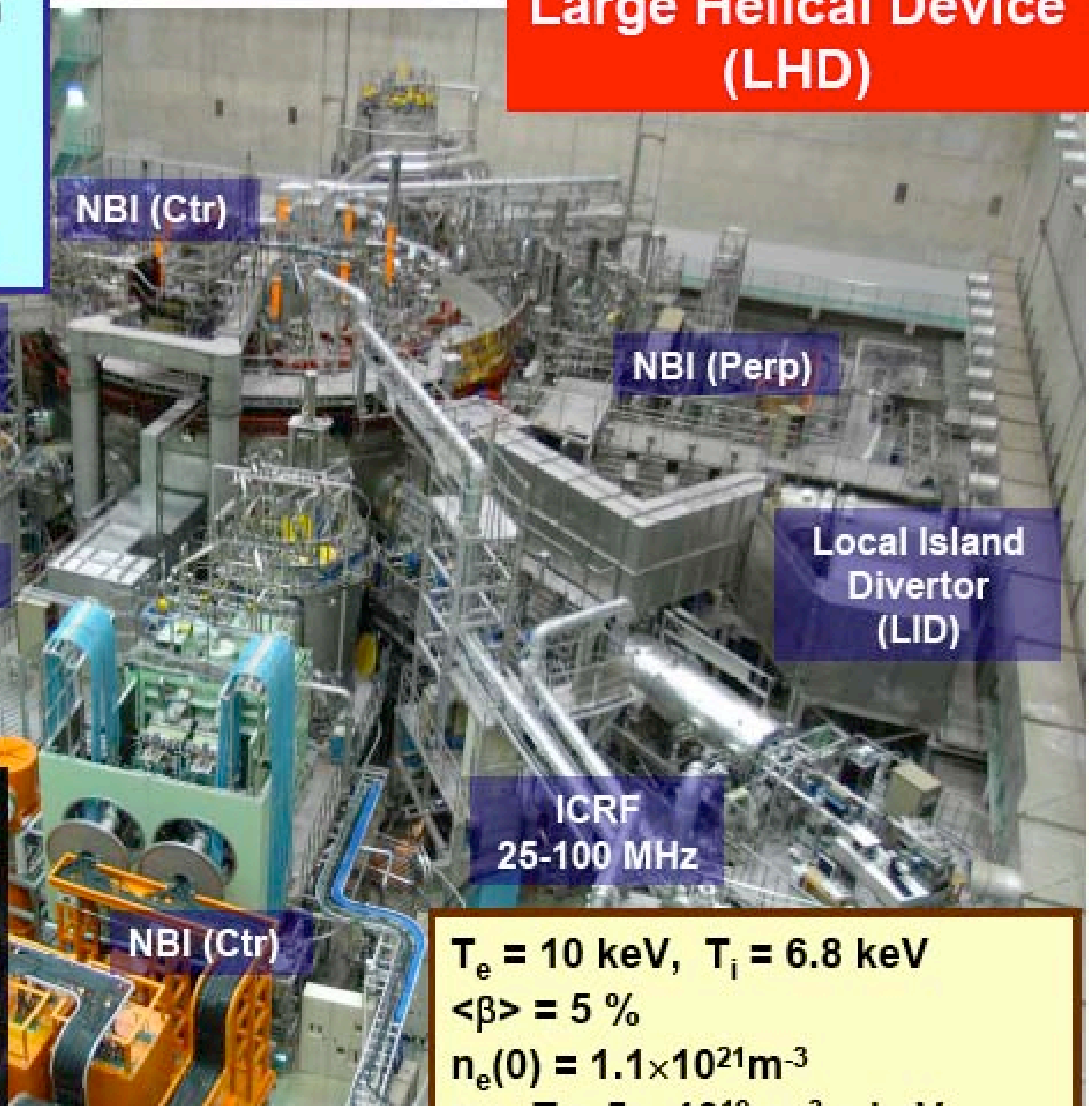
Figure 8 Stellarator (Model A ~ 1954) and Spitzer (1993)

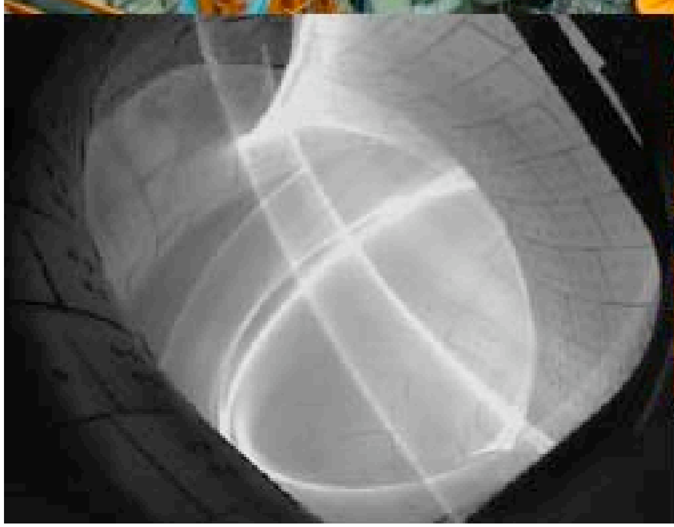
• The stellarator as first proposed by Spitzer May 1951 was a thermonuclear power generator based on a linear cylinder with uniform magnetic field. A toroidal stellarator based on a Figure 8 was described later.

• PPPL Model C - converted to tokamak in 1969, and the main stellarator effort was carried forward by IPP and Japan Univ's/NIFS through the 70s and 80s.

Large Helical Device (LHD)

External diameter	13.5 m
Plasma major radius	3.9 m
Plasma minor radius	0.6 m
Plasma volume	30 m^3
Magnetic field	3 T
Total weight	1500 t





T_e = 10 keV, T_i = 6.8 keV

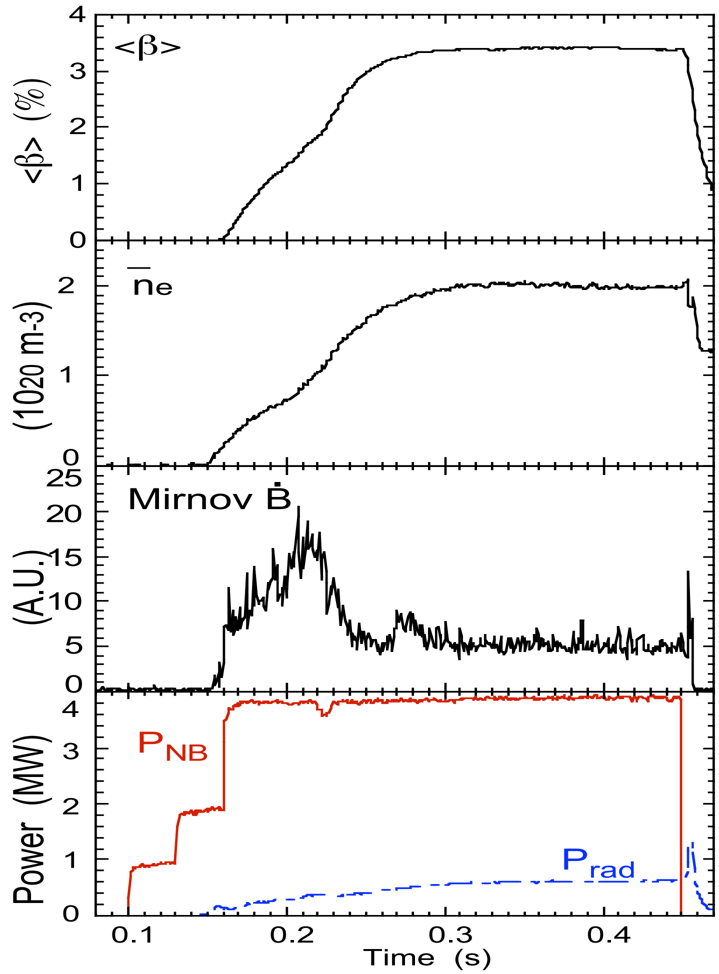
$\langle\beta\rangle$ = 5 %

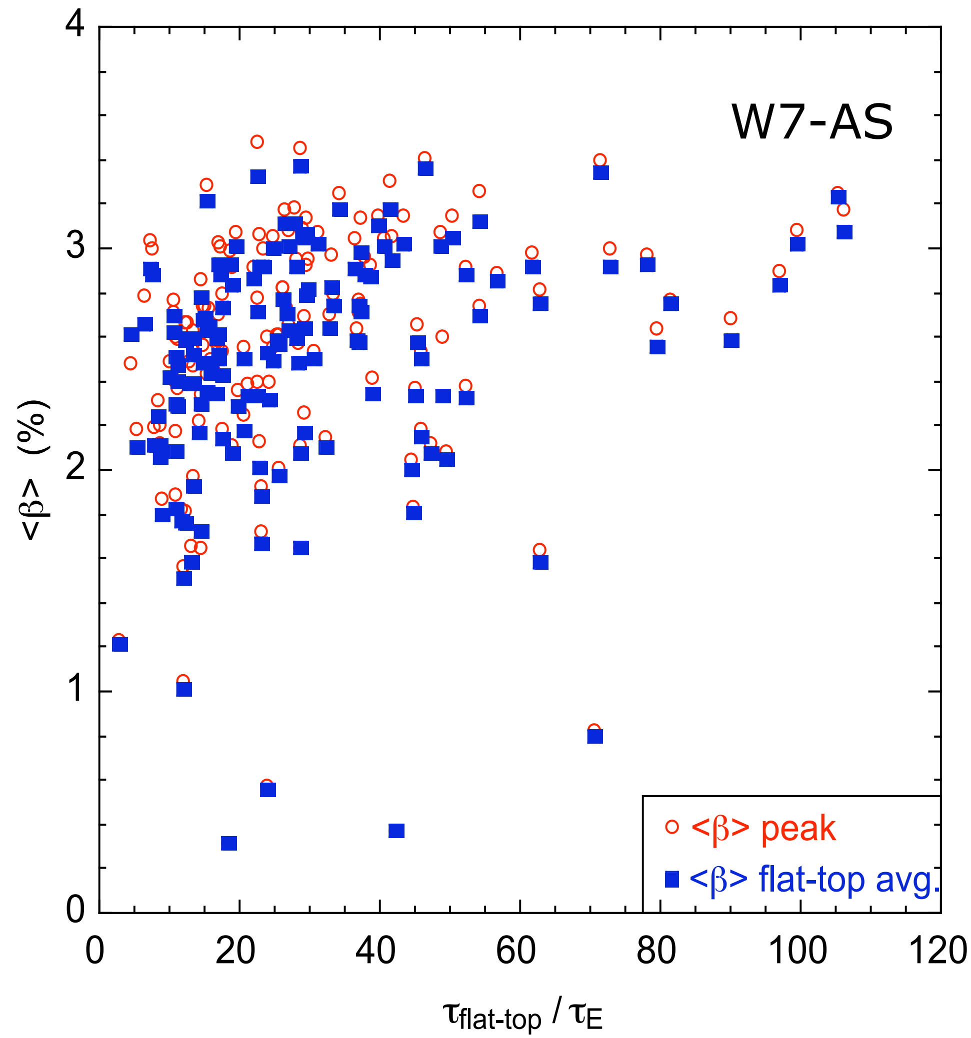
$n_e(0) = 1.1\times10^{21} m^{-3}$

$n\, \tau_E\, T = 5 \times 10^{19}\ m^{-3}$ s keV

Long pulse : 0.6 MW for 1 hour

Sustained Hi β in Partially Optimized Stellarator W7-AS

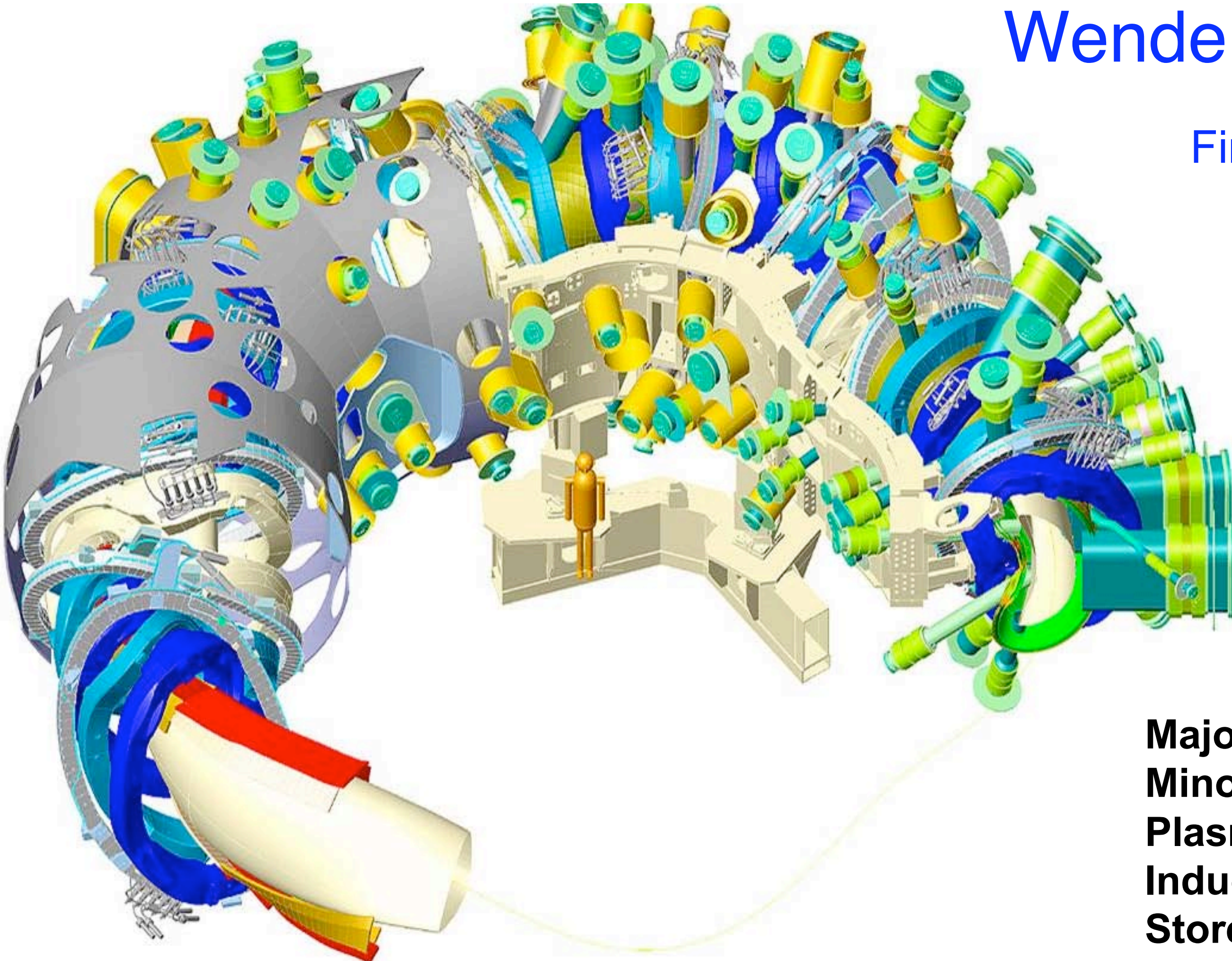




- W7-AS was the first stellarator device based on modular non-planar magnetic field coils
- demonstrated commonality with tokamak physics like access to H-mode confinement regime

An Optimized Stellarator is Under Construction

Max-Planck-Institut für Plasmaphysik



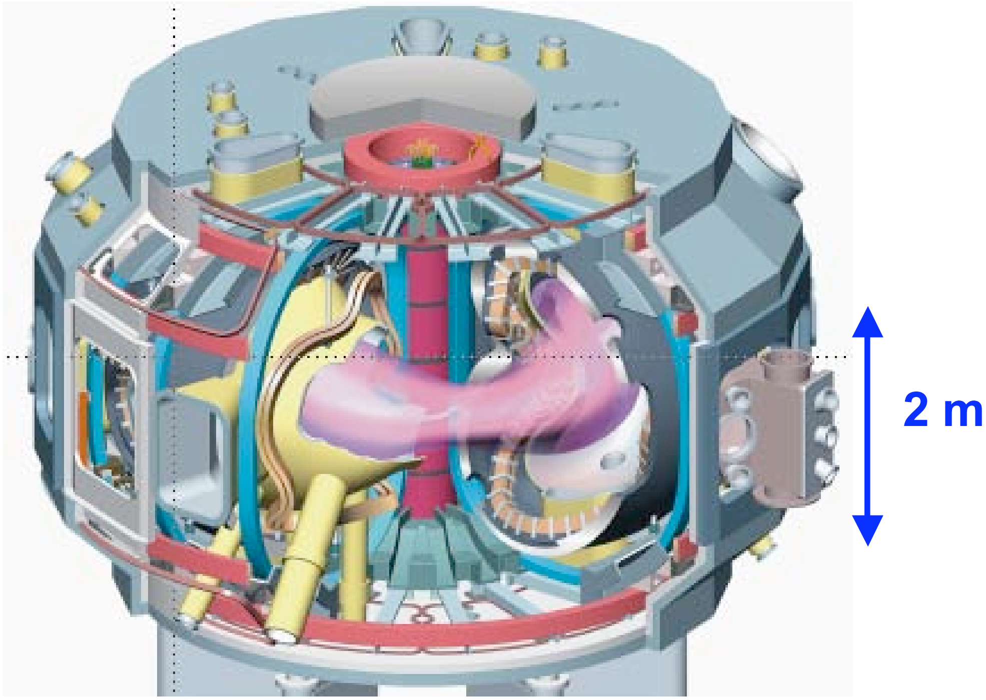
Wendelstein 7-X

First Plasma 2014

Major radius:	**5.5 m**
Minor radius:	**0.53 m**
Plasma volume	**30 m^3**
Induction on axis:	**3T**
Stored energy:	**600 MJ**
Machine mass:	**725 t**
Pulse length:	**30 min**
Aux Heating	**20-40 MW**

W-7X is based on W-7AS, and is optimized to reduce bootstrap plasma currents, fast particle loss, neoclassical transport, with good flux surfaces , MHD stability and feasible coils.

Unfinished Business: Quasi-Axisymmetric Stellarators (NSCX)



- Intriguing opportunities for steady-state disruption-free operation
- 3-D with need for high precision leads to hardware complexity and higher costs
- How symmetric does quasi-symmetric have to be?
- What is the future in this area?

An International Team is Forged to Develop a New Energy Source

Gorbachev and Reagan

- Agreed to "cooperation on fusion research" November 21, 1985 Geneva
- The IAEA provides the framework for International Collaboration
- By Dec 2005, EU,JA, RF, KO,CN, IN and US had signed ITER agreement

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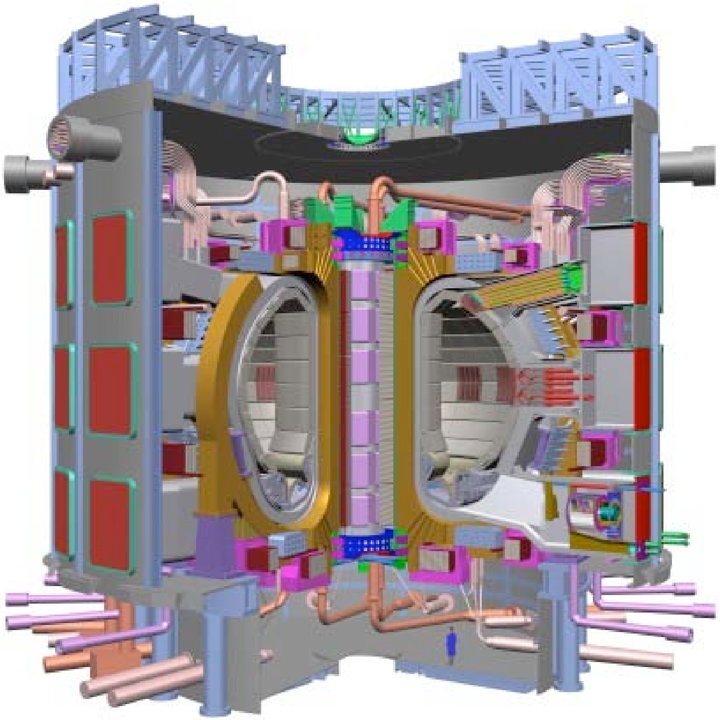
ITER is Now Underway



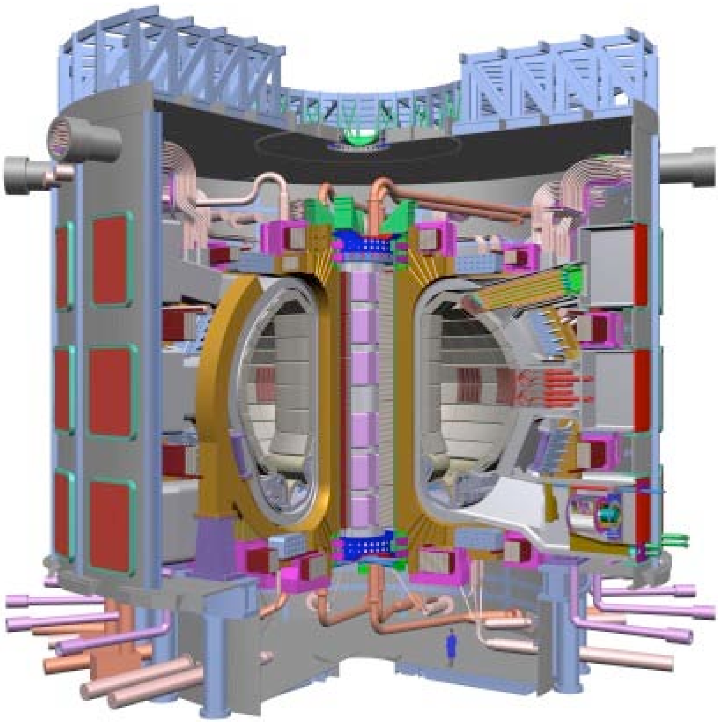
Reactor scale

ITER Site Under Construction

First Plasma planned for 2018

First DT operation planned for ~2022

ITER is Now Underway



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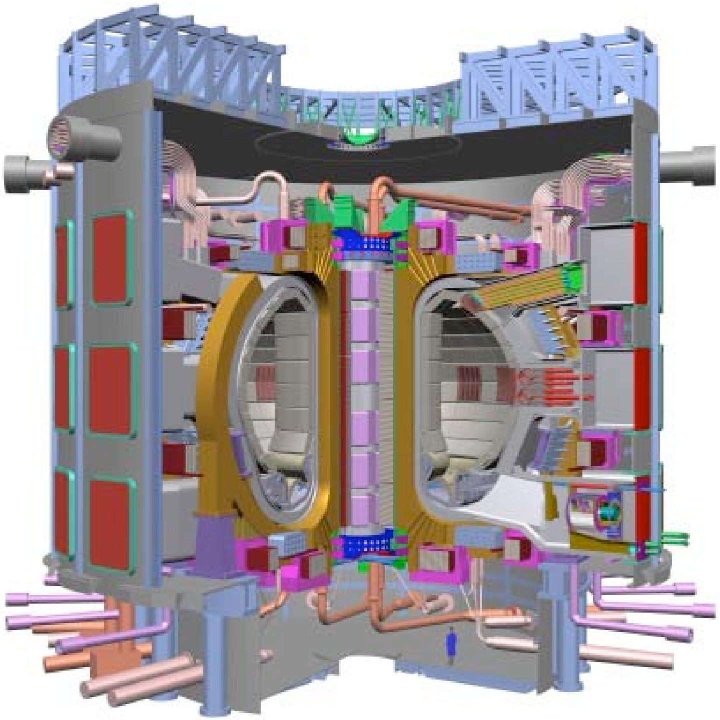
ITER is Now Underway



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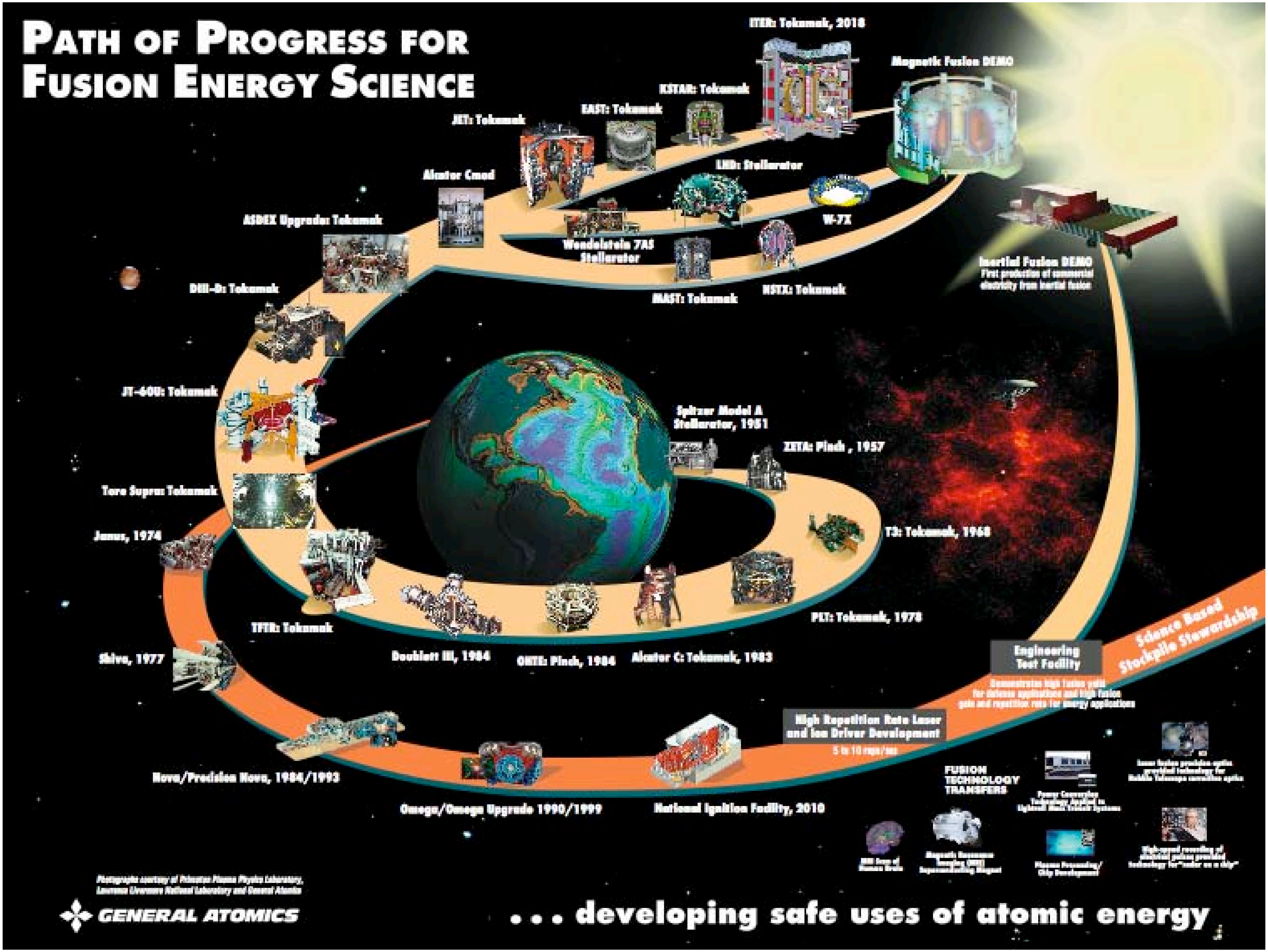
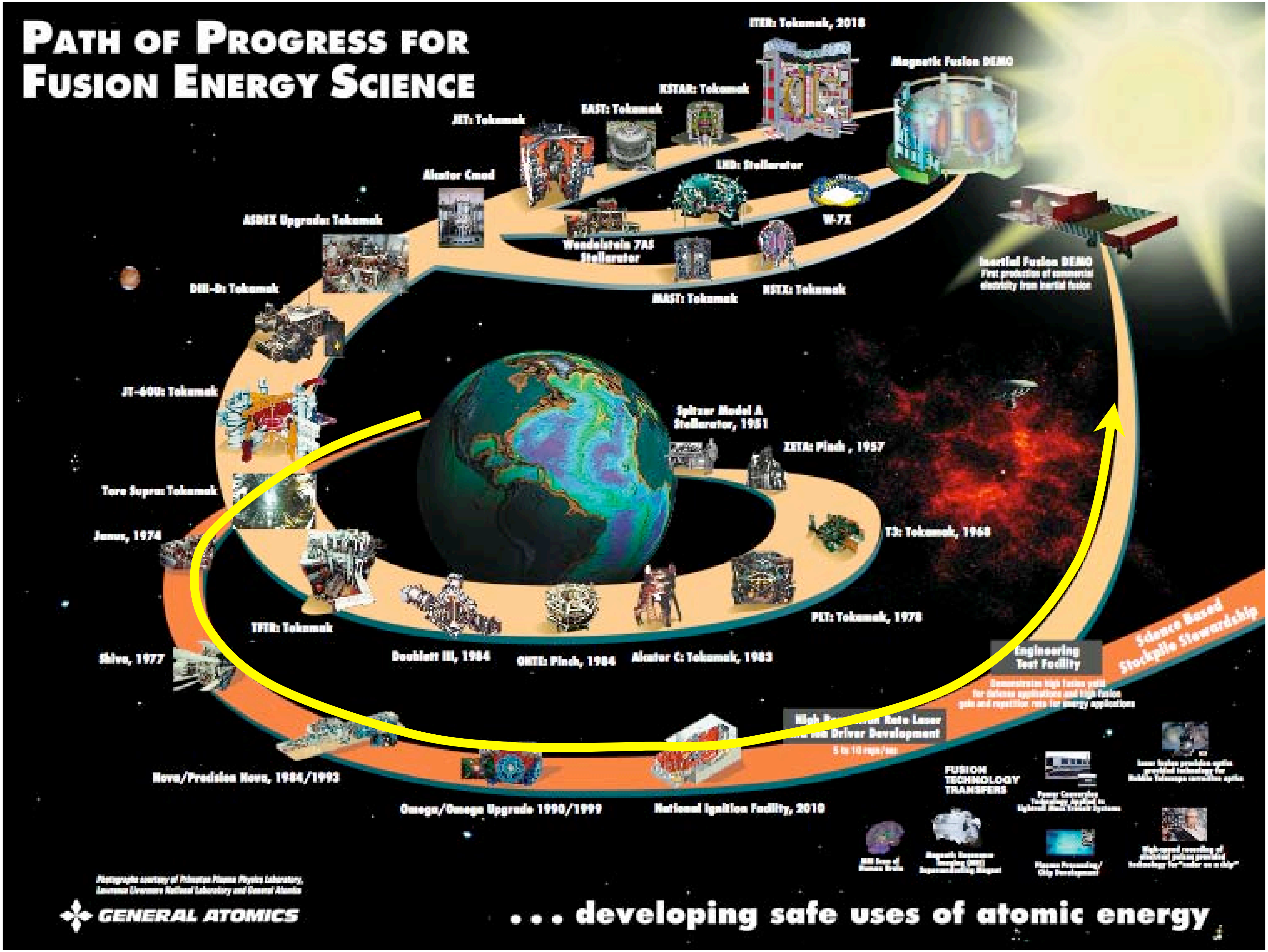


PATH OF PROGRESS FOR FUSION ENERGY SCIENCE
ITER: Tokamak, 2018
Magnetic Fusion DEMO
KSTAR: Tokamak
EAST: Tokamak
JET: Tokamak
Alcator Cmod
LHD: Stellarator
W-7X
ASDEX Upgrade: Tokamak
Wendelstein 7AS Stellarator
MAST: Tokamak
NSTX: Tokamak
Inertial Fusion DEMO
First production of commercial electricity from inertial fusion
DIII-D: Tokamak
JT-60U: Tokamak
Spitzer Model A Stellarator, 1951
ZETA: Pinch, 1957
Tore Supra: Tokamak
T3: Tokamak, 1968
Janus, 1974
TFTR: Tokamak
PLT: Tokamak, 1978
Shiva, 1977
Doublett III, 1984
OHTE: Pinch, 1984
Alcator C: Tokamak, 1983
Engineering Test Facility
Demonstrate high fusion yield for defense applications and high fusion gain and repetition rate for energy applications
Science Based Stockpile Stewardship
High Repetition Rate Laser and Ion Driver Development
5 to 10 reps/sec
Nova/Precision Nova, 1984/1993
Omega/Omega Upgrade 1990/1999
National Ignition Facility, 2010
FUSION TECHNOLOGY TRANSFERS
Plasma Processing/ Chip Development
Photographs courtesy of Princeton Plasma Physics Laboratory, Lawrence Livermore National Laboratory and General Atomics
GENERAL ATOMICS
. . . developing safe uses of atomic energy



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Omega/Omega Upgrade 1990/1999
National Ignition Facility, 2010
FUSION TECHNOLOGY TRANSFERS
GENERAL ATOMICS
. . . developing safe uses of atomic energy

Inertial Confinement Fusion, Early Days

- Radiation compression of DT to produce fusion energy demonstrated in the early 50s in Greenhouse George Cylinder test (and others).
- Invention of the laser in early 60s offered the possibility of a programmable repetitive driver for micro targets. Research continued on intense particle beam drivers in USSR and US.
- Idealized calculations in late 60s suggested 1kJ needed to achieve breakeven using micro targets and direct drive.
- 1972- Nature article by Nuckolls et al with computer modeling of laser driven compression *Nature* Vol. 239, 1972, pp. 129
- Laser driven experiments at LLNL and elsewhere from mid 70s to mid 80s (Nova), revealed importance of plasma instabilities and driver uniformity, raising required driver energy to MJ range.

Construction of NIF/LMJ - ICF Burning Plasmas

- Classified Centurion-Halite nuclear tests in ~1986 are reported to have validated compression modeling
- Many aspects of US ICF declassified in Nov 1994, allowing target designs to be discussed.
- Omega Project achieves gain of 0.01 using direct drive of a DT capsule in 1996.
- Fast Ignitor concept (1995) offers possibility of reduced driver energies
- There has been dramatic progress in driver intensity and pellet fabrication in the past 40 years, and many challenges remain.
- Multiple paths in drivers (Glass, KrF, Z-pinch) are being pursued.

NIF Enabled by Rapid Advance in Laser Technology

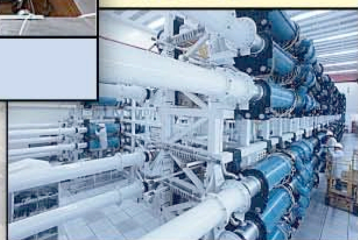
- Glass laser energy has increased 10^6
- Fusion energy will need:
  increased efficiency
  increased repetition rate

Janus

1972

100 J
1.05 µm

Shiva



1978

10 kJ
1.05 µm

Nova

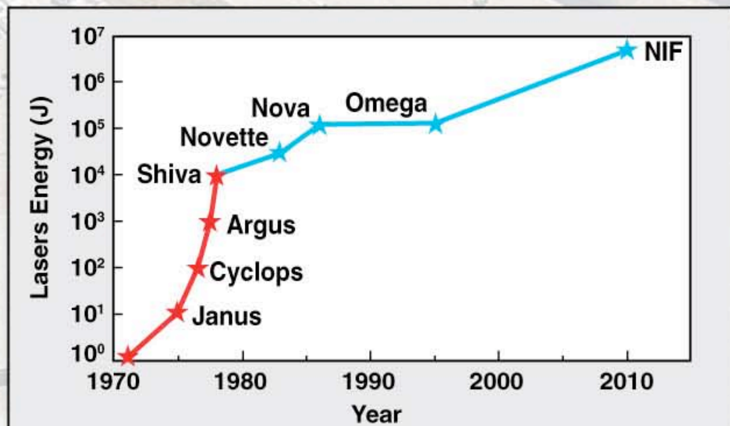
1984

30 kJ
351 nm

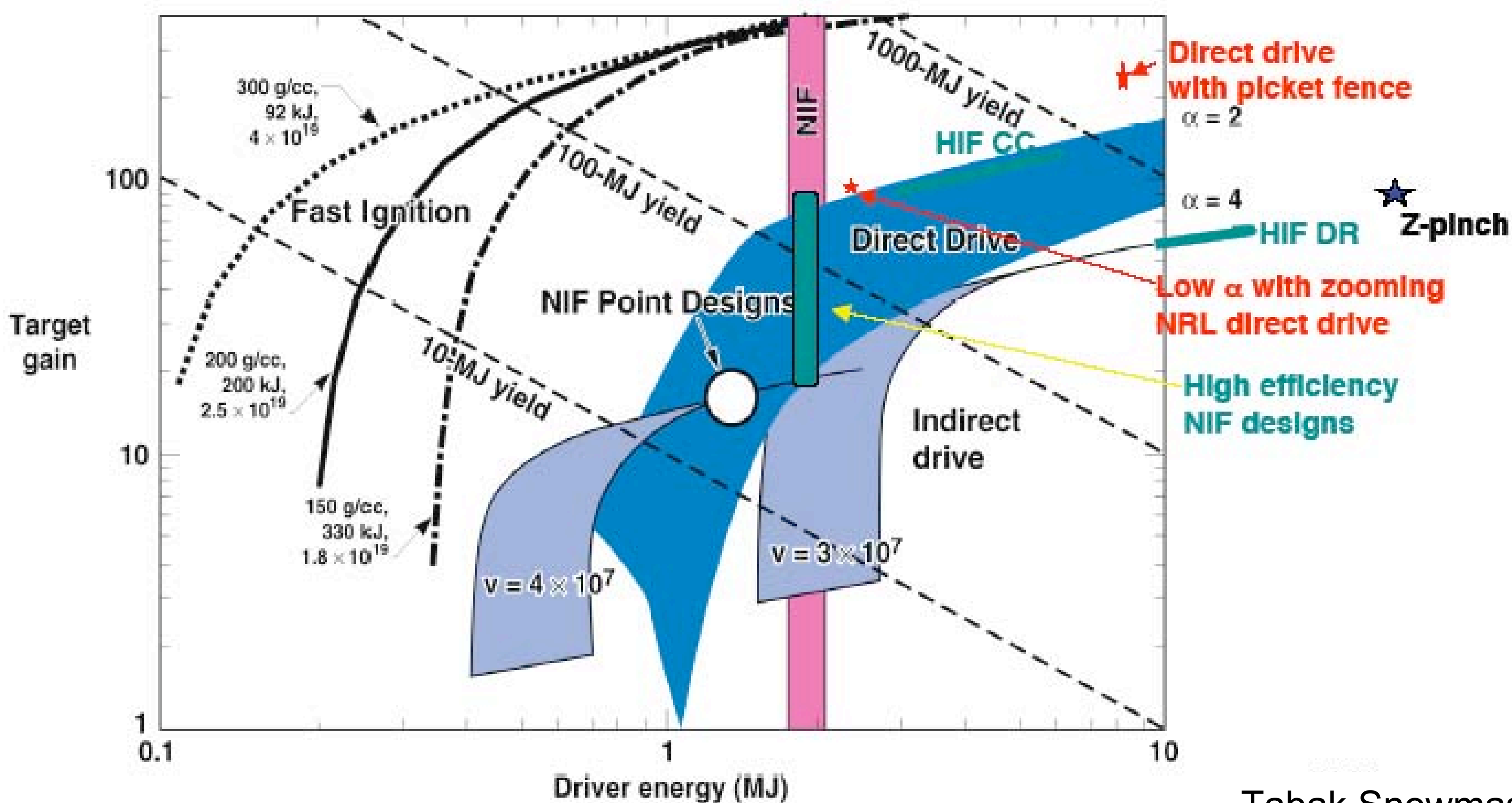
NIF

2009

1.8 MJ
351 nm



Target Designs with Varying Degrees of Risk Provide Adequate Gain for all Driver Concepts



Tabak Snowmass

FI Expt's -
Omega, FIREX,
HIPER

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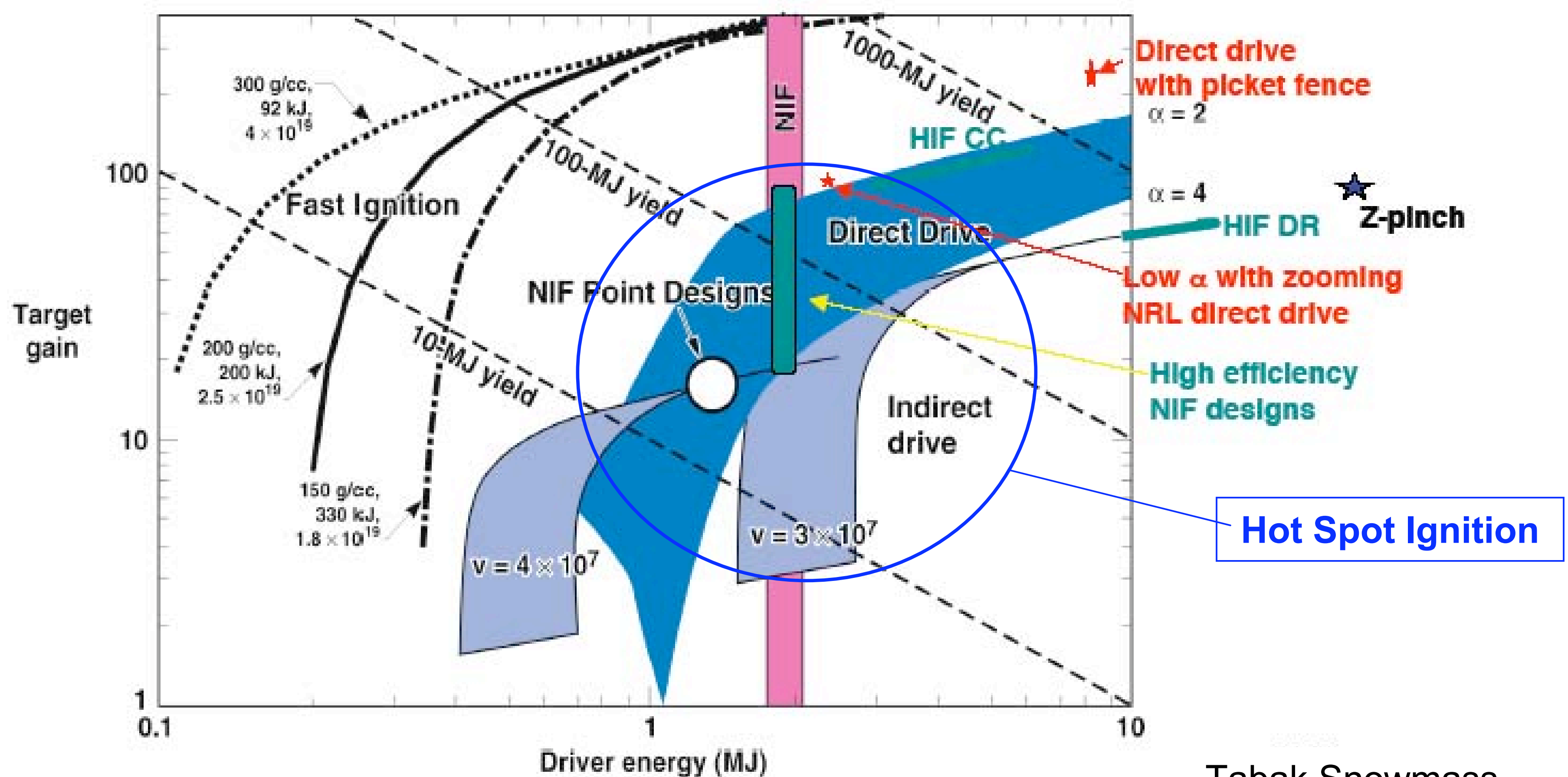

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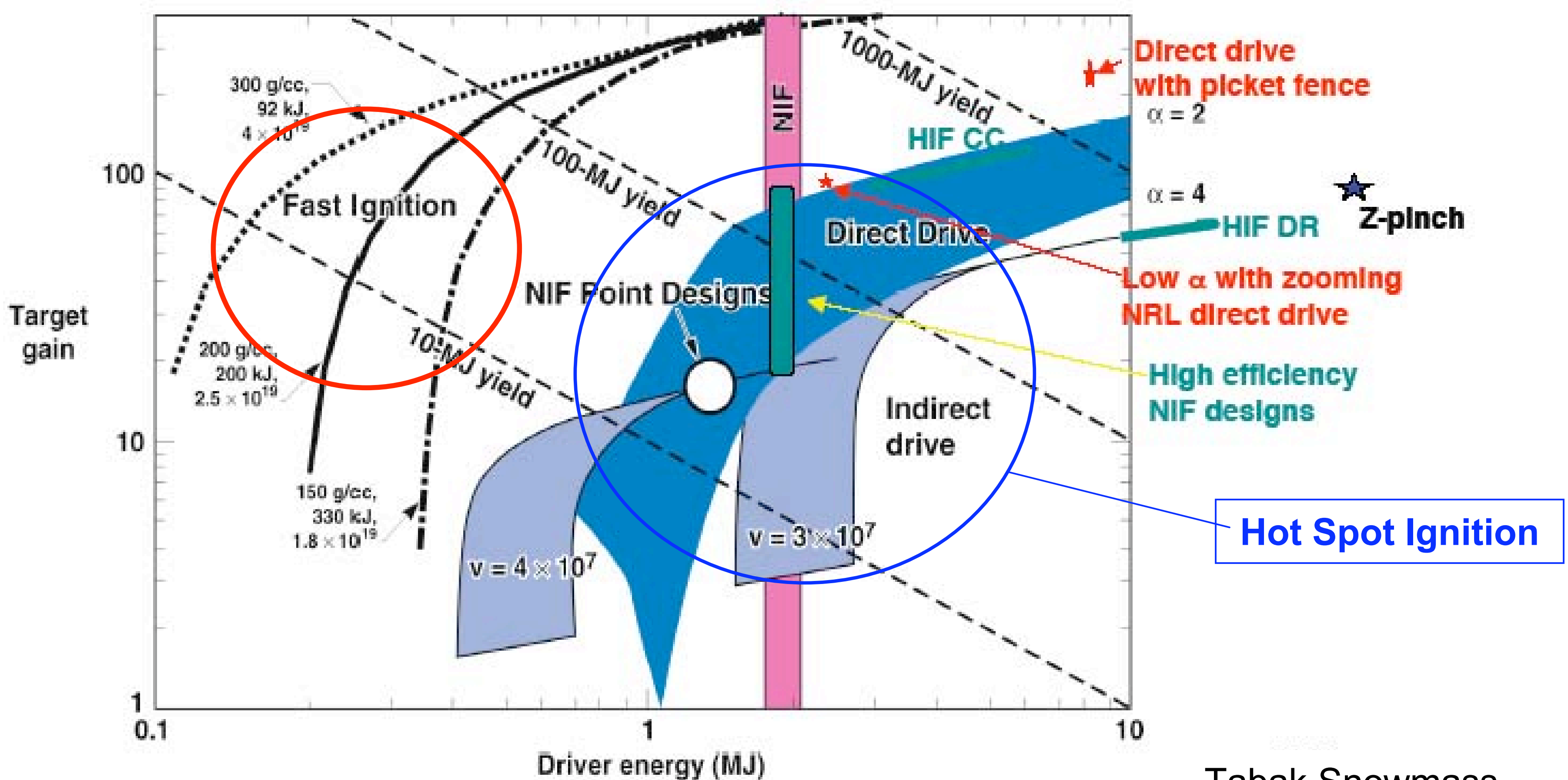
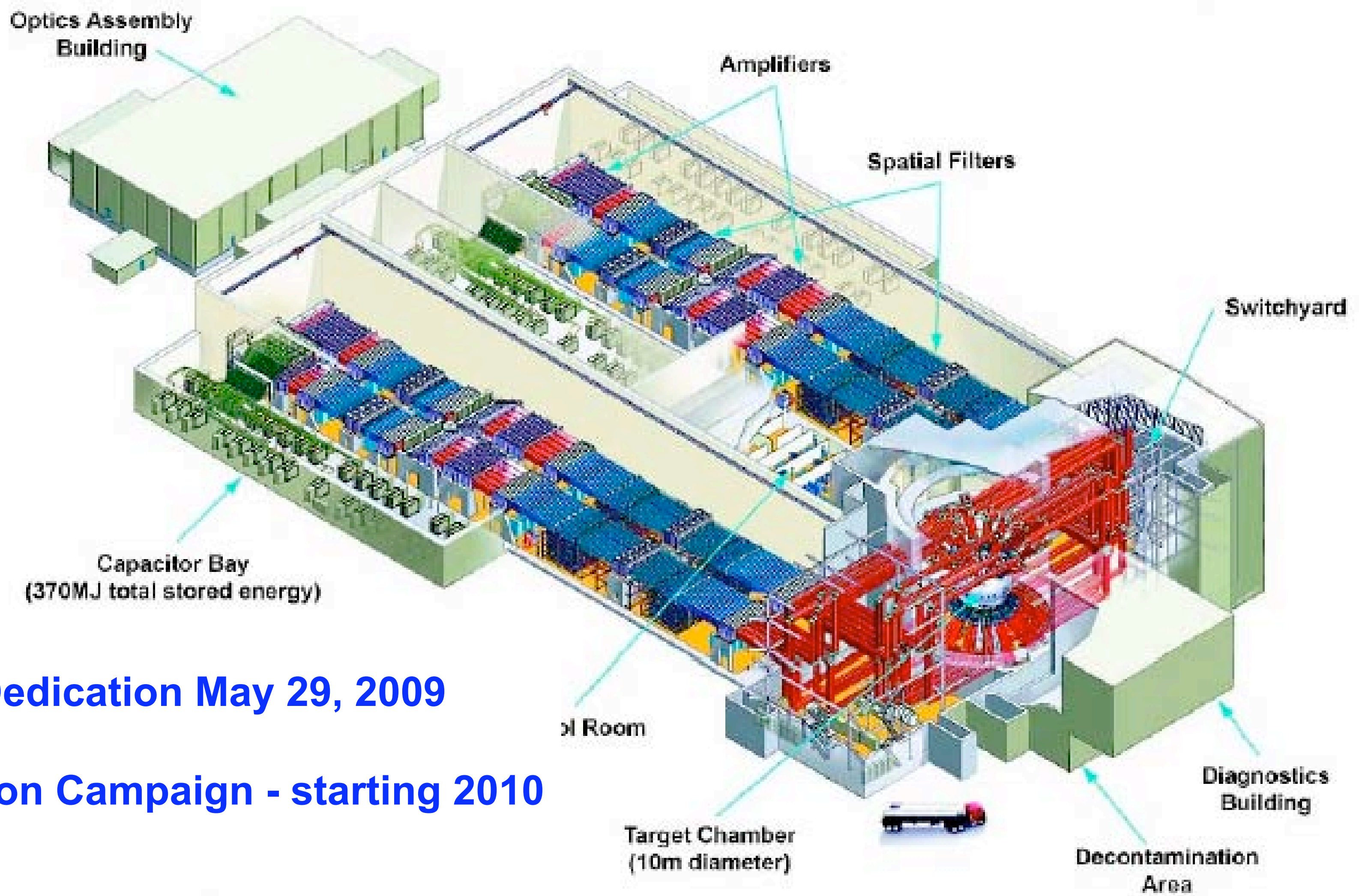


Tabak Snowmass

FI Expt's - Omega, FIREX, HIPER

The National Ignition Facility (NIF), a nominally 1.8MJ/500TW blue laser being built at Livermore, meets the requirements for ignition



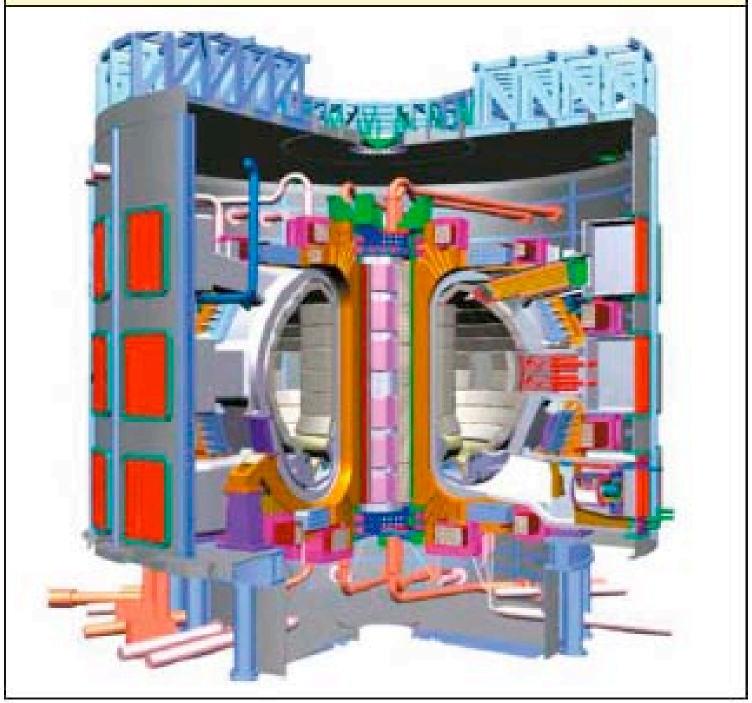
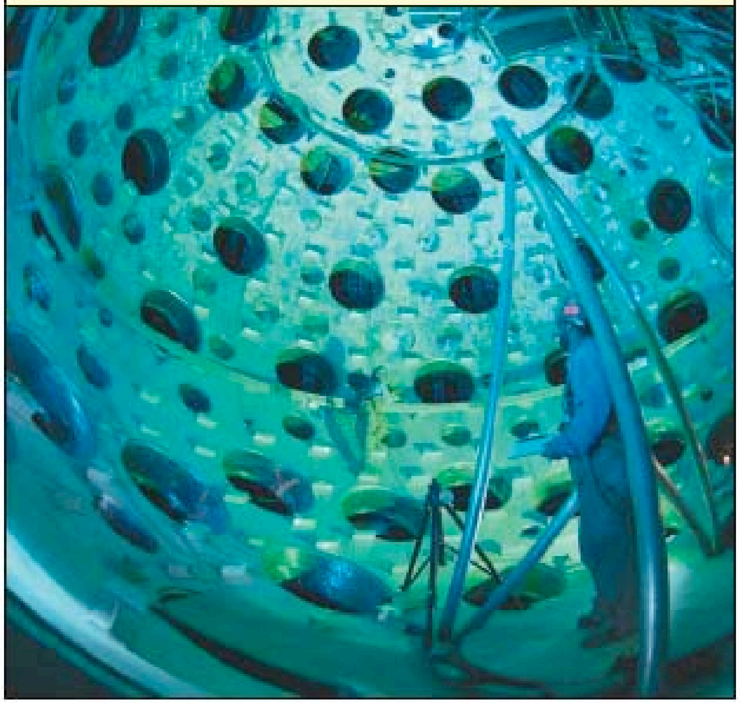
NIF Dedication May 29, 2009

Ignition Campaign - starting 2010

Some Overall Highlights

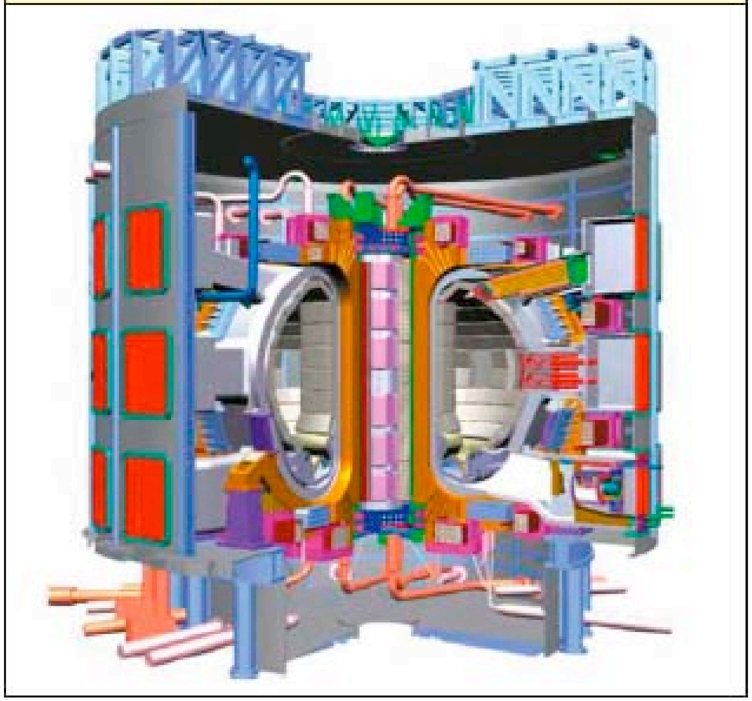
- A strong scientific basis has been established for fusion.
- Diagnostics and Plasma Technology (Aux heating, CD, pellet inj) enabled progress.
- Computer Simulations are becoming more realistic and integral to analysis and prediction.
- Several promising paths to fusion, each working on optimization and sustainment.
- Temperatures needed for fusion achieved - in many facilities.
- Confinement needed for fusion is being approached - one step away.
- Complex fusion systems have been operated at large scale.
- Fusion systems using fusion fuel (DT) operated safely.
- Fusion could move much faster if required resources were applied.
- Now on the threshold of energy producing plasmas in both magnetic and inertial fusion.

Facilities to Produce Fusion Energy are under Construction

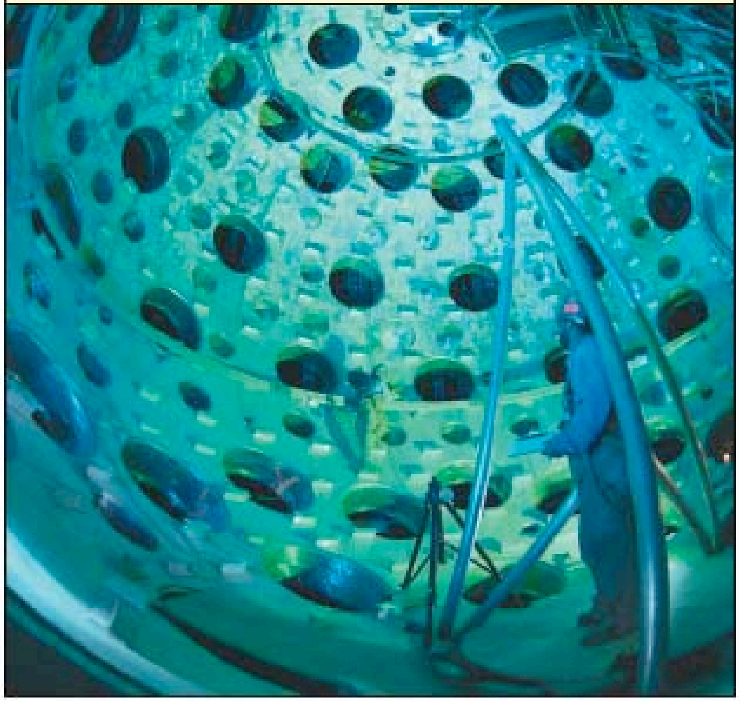
Magnetic Fusion Energy	Inertial Fusion Energy
	
ITER	NIF

Facilities to Produce Fusion Energy are under Construction

Magnetic Fusion Energy	Inertial Fusion Energy



ITER

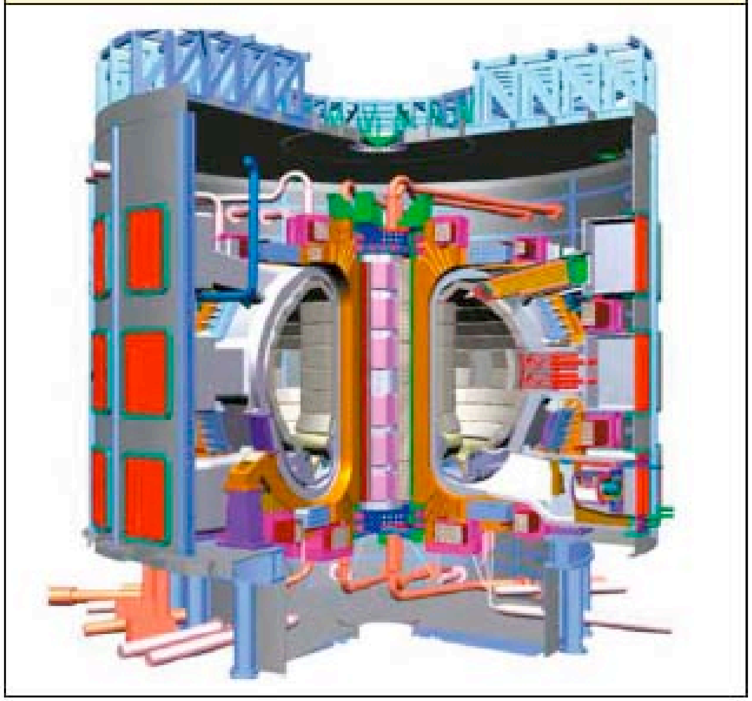


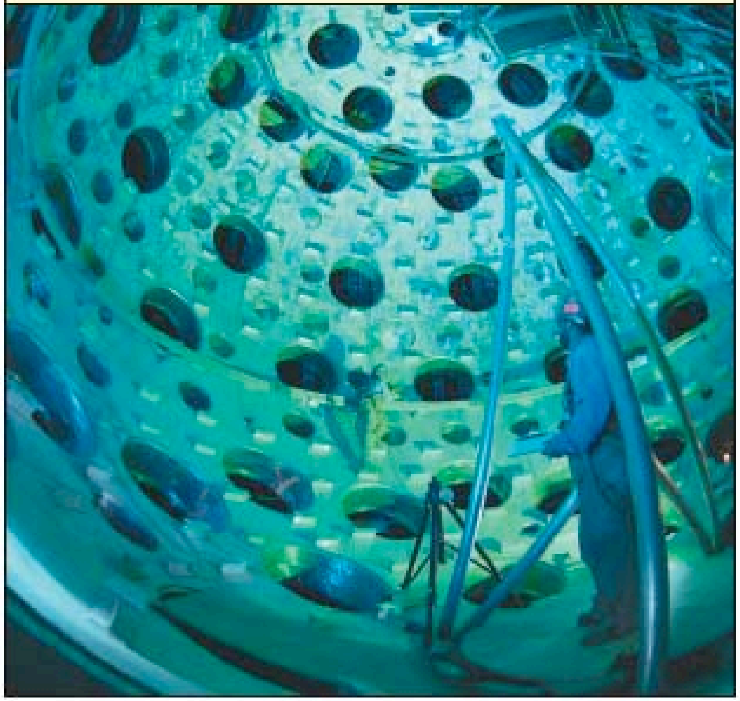
NIF

First D-T	~2010
Fusion Gain, Q	10 - 20
Fusion Energy/pulse	40 MJ

Facilities to Produce Fusion Energy are under Construction

Magnetic Fusion Energy	Inertial Fusion Energy





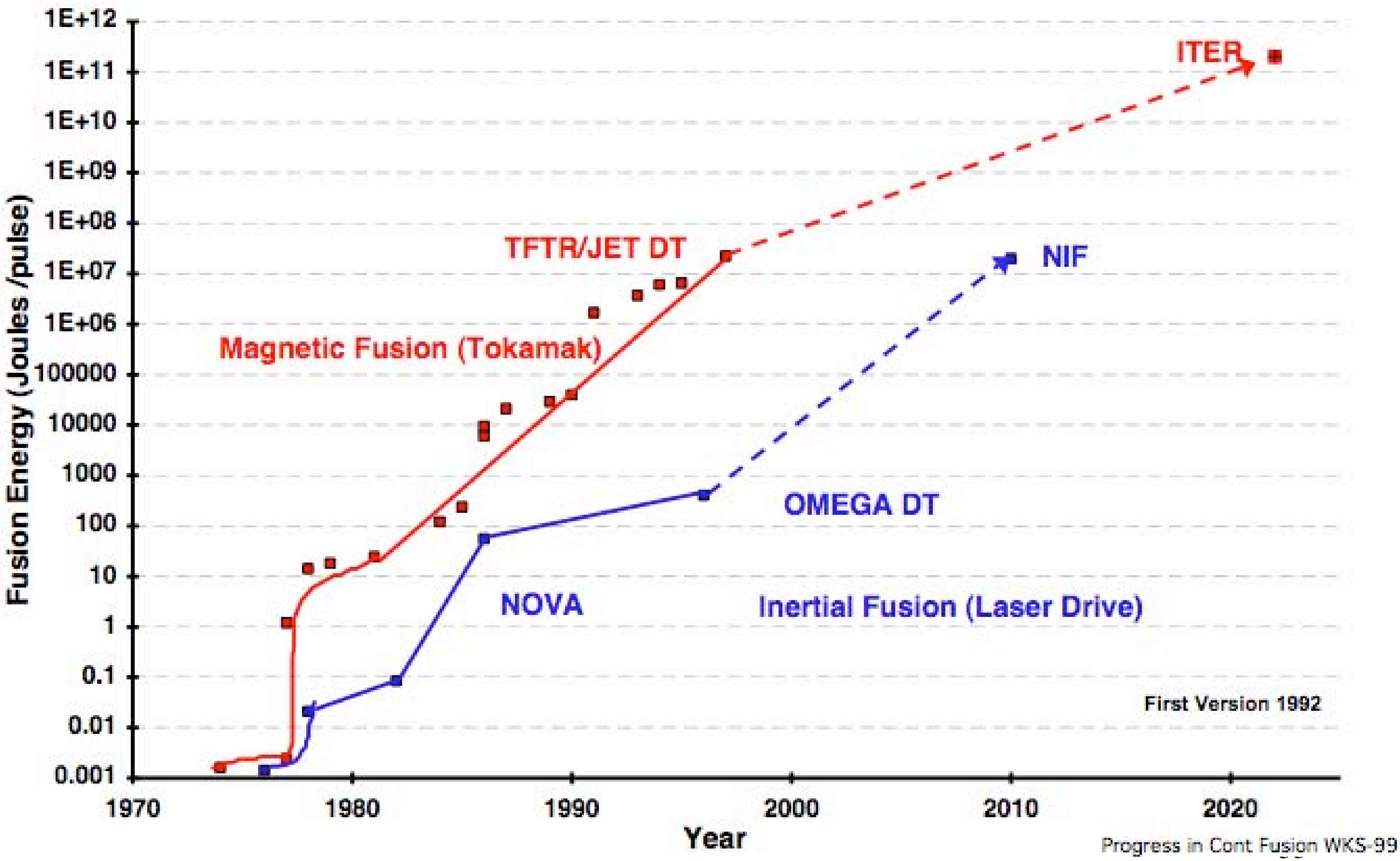
ITER

First D-T	~2022
Fusion Gain, Q	10
Fusion Energy/pulse	200,000 MJ

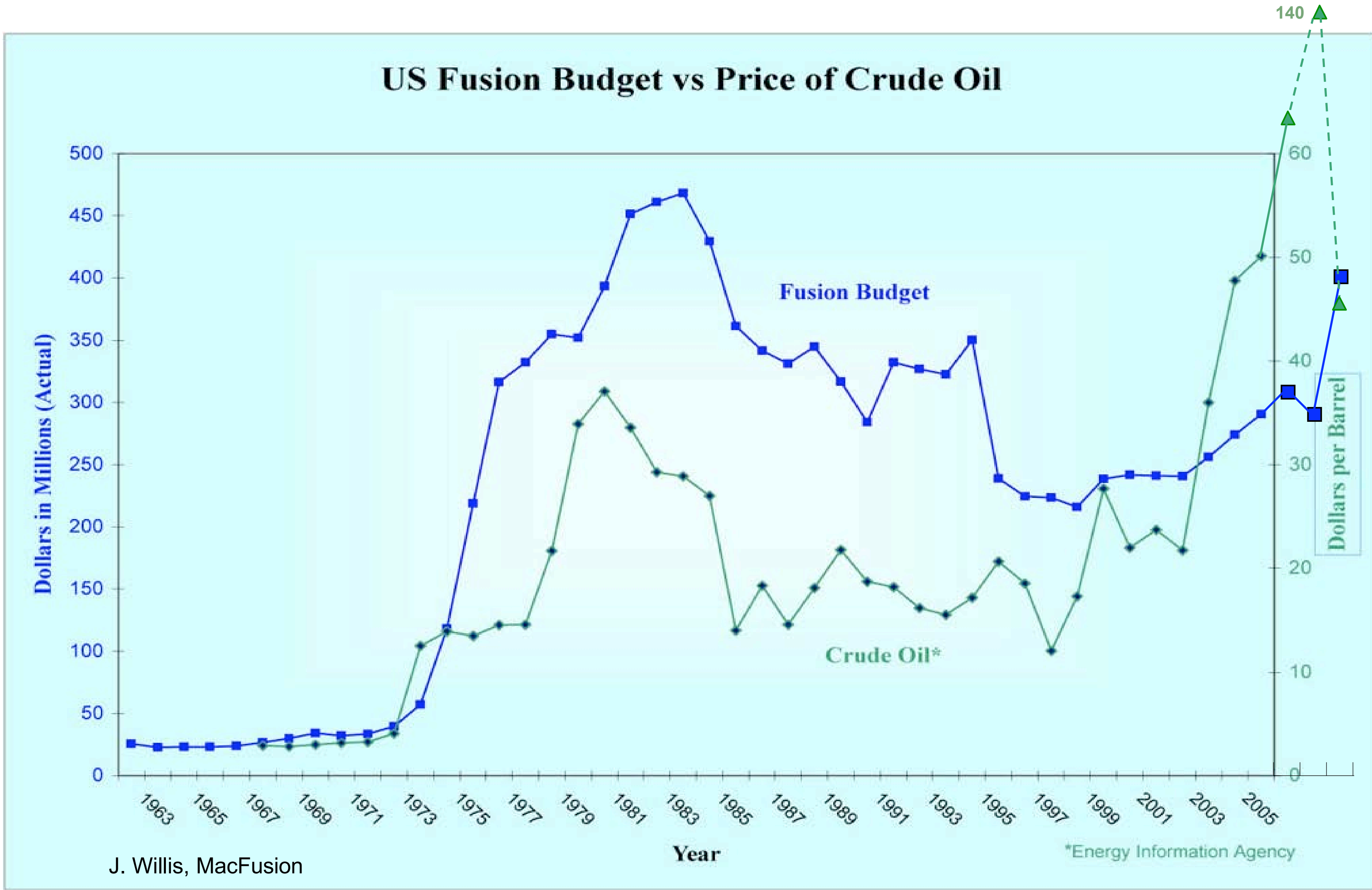
NIF

First D-T	~2010
Fusion Gain, Q	10 - 20
Fusion Energy/pulse	40 MJ

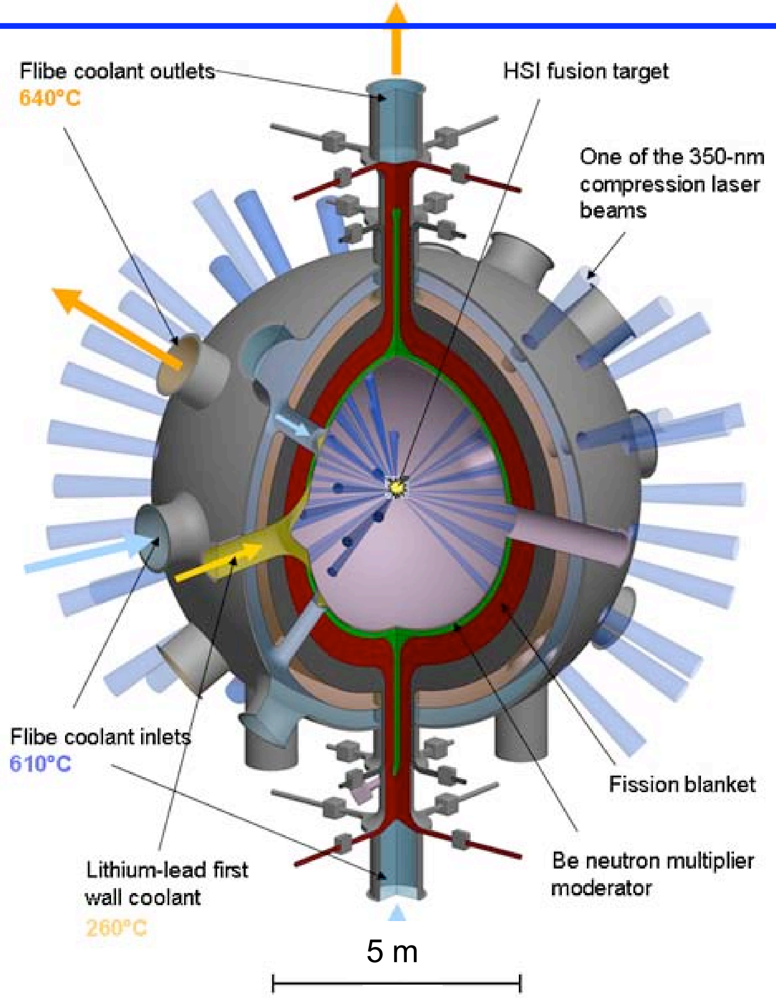
NIF and ITER will Extend Progress in Fusion Energy



Is Fusion Prepared for a Major Next Step if Opportunity Knocks Again?

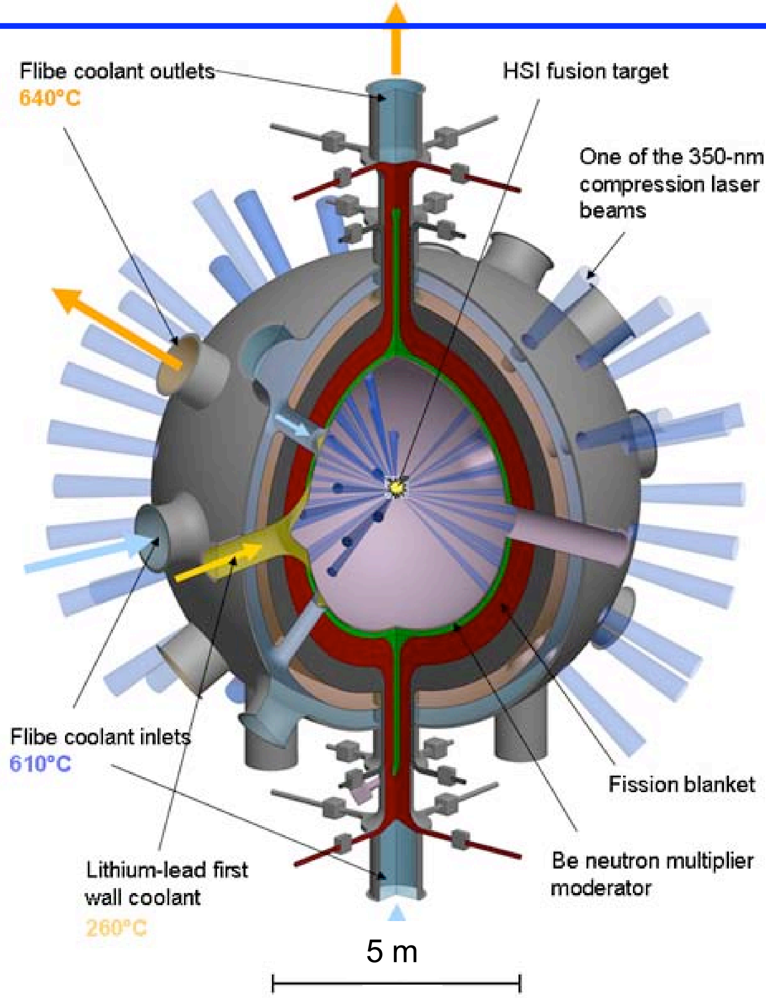


Fusion in 1 Decade



LIFE

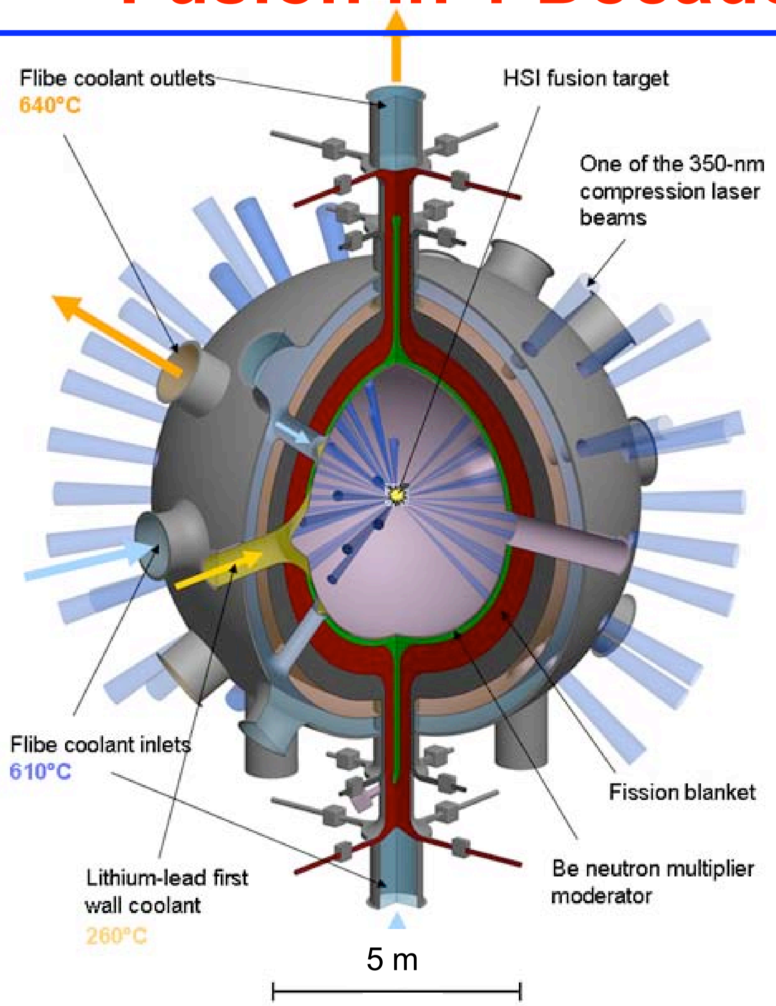
Fusion in 1 Decade

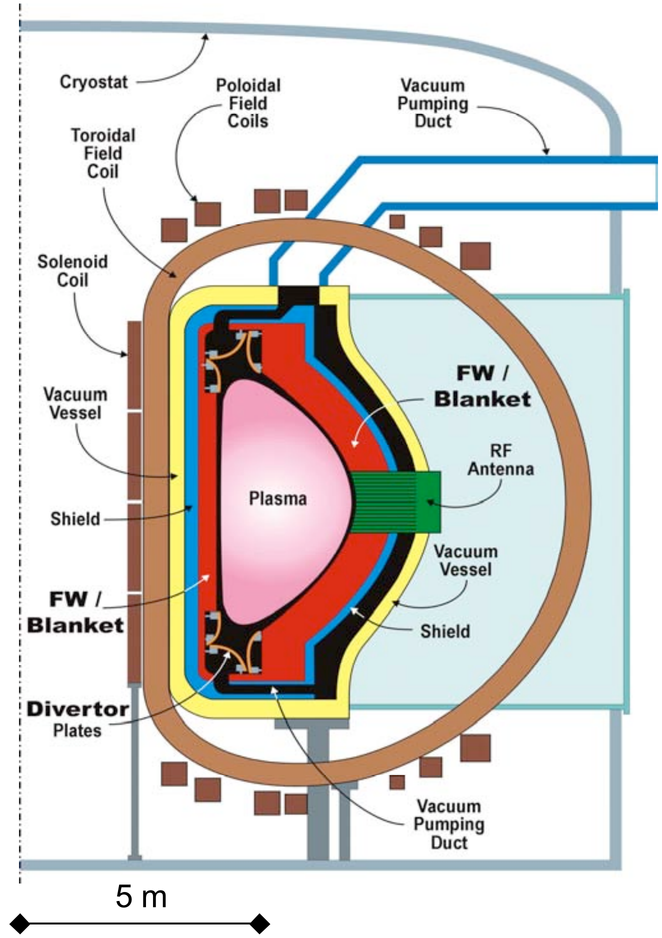


LIFE

First D-T	~2020
Fusion Gain, Q	25
Fusion Power	400 MW
Fission + Fusion Power	2,500 MW

Fusion in 1 Decade   Fusion in 3 Decades





LIFE		DEMO
First D-T	~2020	
Fusion Gain, Q	25	
Fusion Power	400 MW	
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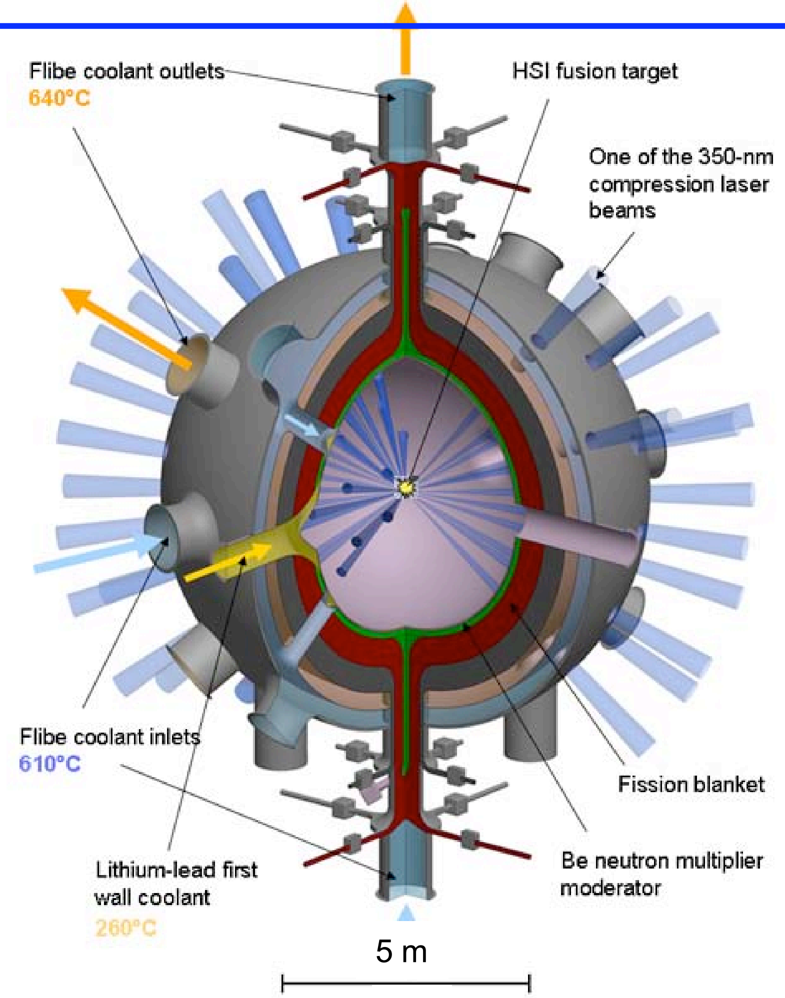
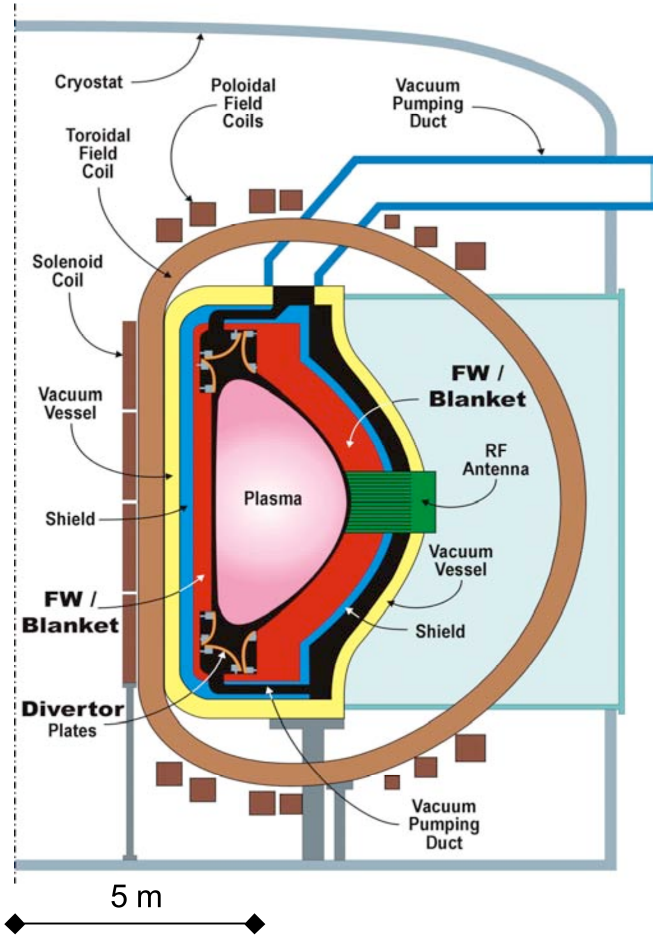
Fusion in 1 Decade | Fusion in 3 Decades





LIFE

First D-T	~2020
Fusion Gain, Q	25
Fusion Power	400 MW
Fission + Fusion Power	2,500 MW

DEMO

First D-T Power	~2025+15
Fusion Gain, Q	20 - 45
Fusion Power	2,500 MW

Concluding Thoughts

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- Unfortunately, magnetic fusion has missed some critical opportunities that have delayed the program eg. 20 years ago with CIT.
- US MFE has drifted into a “treading water” phase, and is adding more small steps instead of taking a bold step forward.
- The MFE community needs a compelling vision to make a major step forward within a decade and establish the credibility of magnetic fusion. The clock is ticking and so is NIF.

Even Uncle Sam is getting impatient!

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I want you to get on with fusion,

Even Uncle Sam is getting impatient!

I want you to get on with fusion,

and get it done in my lifetime.

Acknowledgements

E. Frieman, I. Bernstein, K. Fowler, J. Sheffield, R. Stambaugh, M. Kikuchi, O. Motojima, D.Campbell, M. Watkins, F. Wagner, J. Callen, J. Willis, S. Dean, S. Milora, R. Goldston, E. Marmar