MFE DEVELOPMENT PLAN–TOKAMAKS

R.D. Stambaugh
Presented at
General Atomics
to FESAC Development Path Panel

January 13 2003

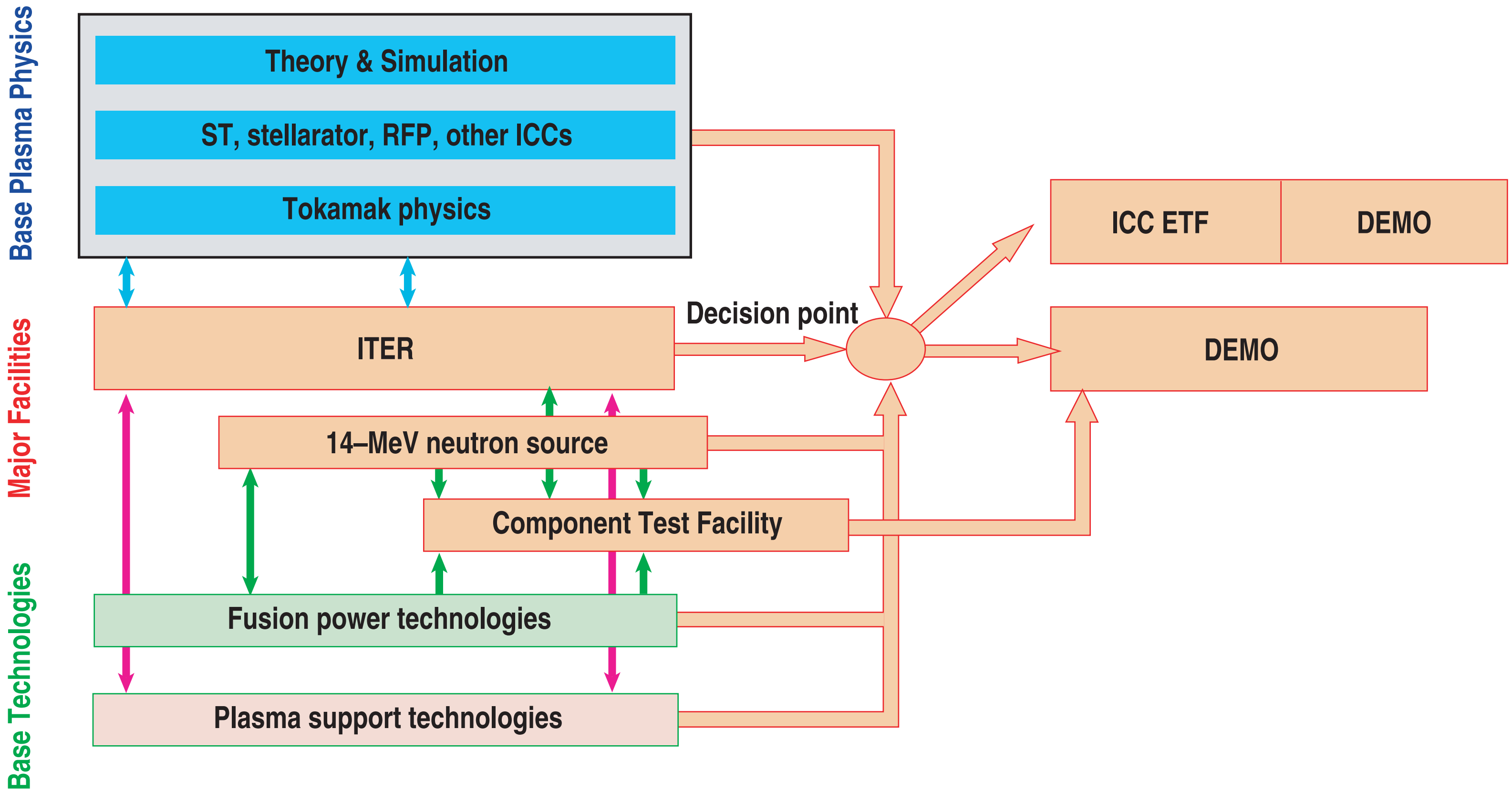
NATIONAL FUSION FACILITY
SAN DIEGO

004-03/RDS/JY

MFE DEVELOPMENT PLAN ELEMENTS-TOKAMAK

- Concept improvement
  - Advanced Tokamak physics, US machines - DIII–D, Alcator C–MOD, NSTX
- Burning plasma
  - ITER, FIRE, IGNITOR
- Steady-state (Long pulse research)
  - ITER, JT–60SC, KSTAR, LHD, HT–7U, SST, TORE-SUPRA, W7–X
- Materials and fusion energy technology development
  - IFMIF, Component Test Facility (CTF)

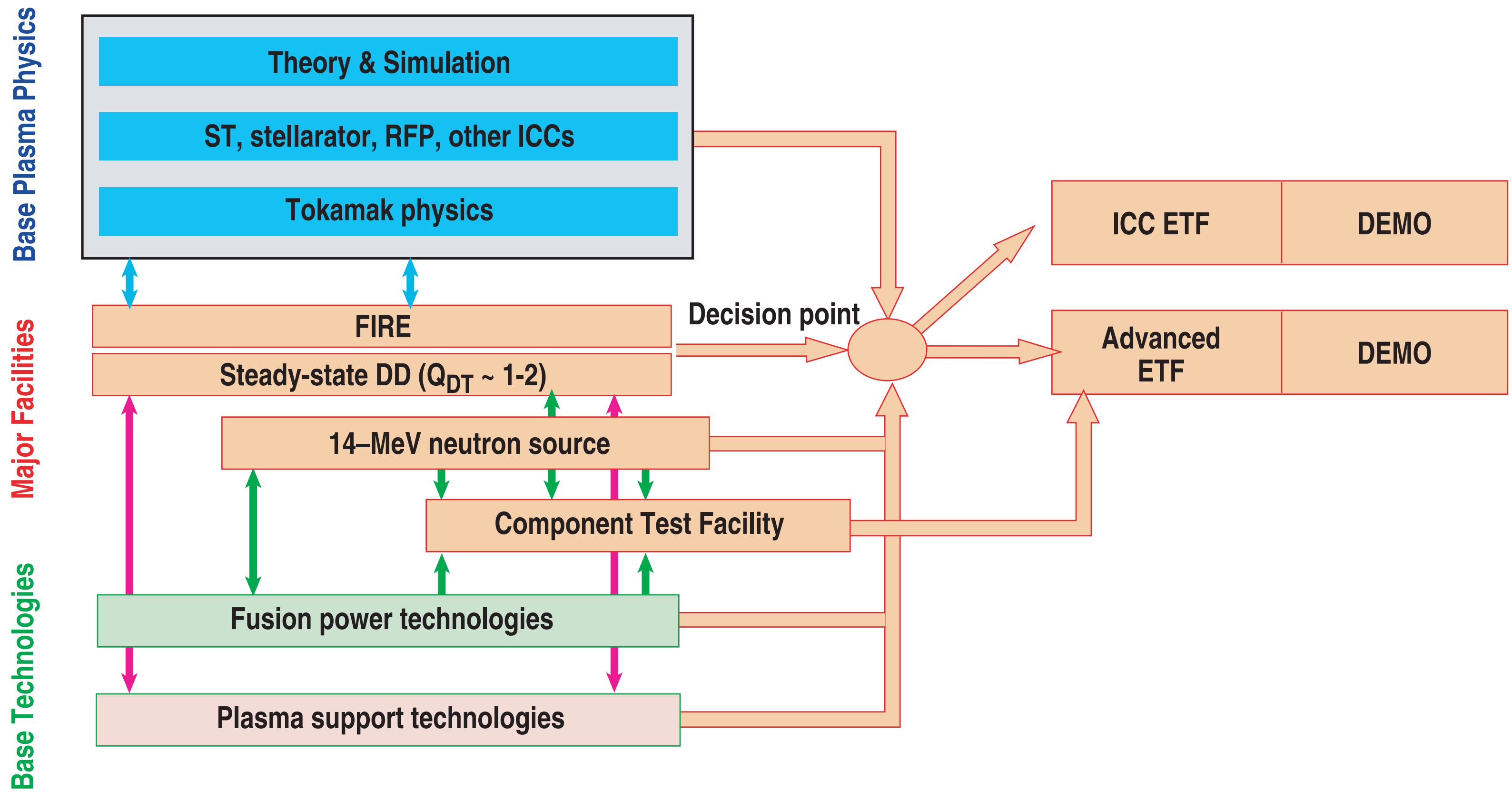
DEVELOPMENT PATH WITH ITER (FESAC)



DIII–D NATIONAL FUSION FACILITY SAN DIEGO

004-03/RDS/JY

DEVELOPMENT PATH WITH FIRE (FESAC)



DIII–D NATIONAL FUSION FACILITY SAN DIEGO

004-03/RDS/JY

U.S. BASE PROGRAM NEEDS INCREASED SUPPORT IN PARALLEL WITH ITER

- Current Tokamaks should run until ITER runs
- We must continue to learn and develop the scientific basis for fusion energy
- The advanced operating modes which are being developed will be the starting point for research in ITER
- The research and operating staff for ITER will be trained on current devices

IMPORTANT SCIENTIFIC CHALLENGES FOR NEXT DECADE

- Integration of AT building blocks into scenarios on which to base future machines
- Full exploration and exploitation of the Tokamak's AT potential
- Understanding the basic physics mechanisms of transport from turbulence
- Understanding the H–mode pedestal structure
- Understanding and controlling mass transport in the plasma boundary
- Developing radiative divertors compatible with steady-state AT operation

ADVANCED TOKAMAK PHYSICS IS CLOSE AT HAND

- **Building blocks nearly in place**
  - Wall stabilization looks like it will work
  - Neoclassical tearing mode stabilization with ECCD works
  - Current profile control demonstrations have started
  - Enhanced confinement states abound
  - ELM free regimes found (EDA in Alcator C–MOD, QH in DIII–D)
  - New era of plasma control starting
  - Disruption mitigation technique available

- **Basis for steady-state operation of ITER, CTF and DEMO at β_N = 4, H_{89} ~2.5-3.0 achievable in 4-6 years**
  - if major facilities are adequately support (+25% budget increase)
    - ★ more run time
    - ★ more plasma control tools
    - ★ adequate theory and computational support

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

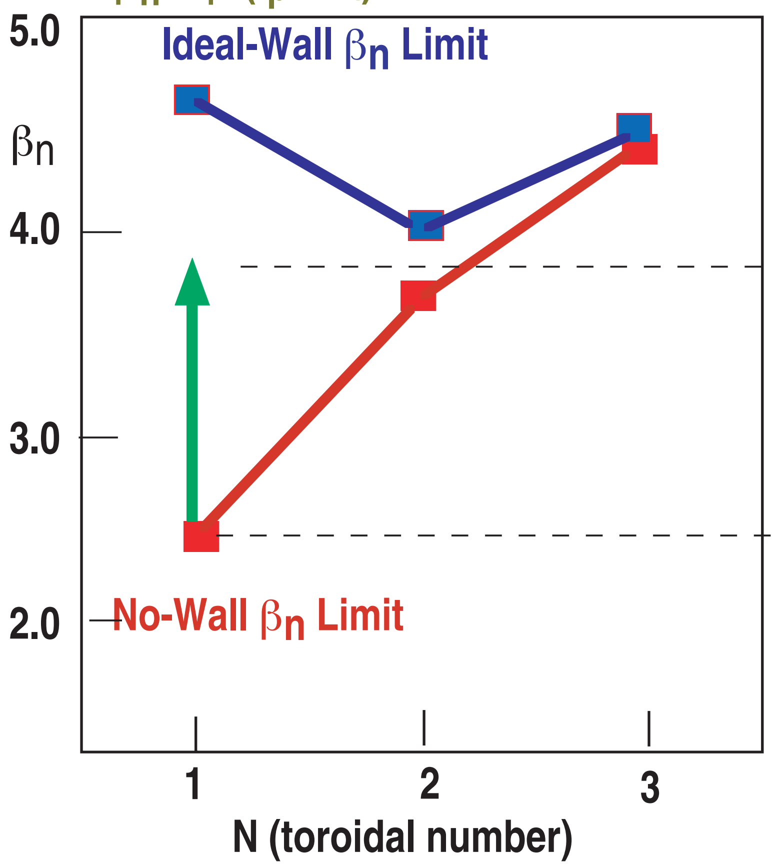
004-03/JY

SUCCESSFUL EXTERNAL KINK MODE SUPPRESSION: A PREREQUISITE FOR SELF-SUSTAINING FUSION PLASMA

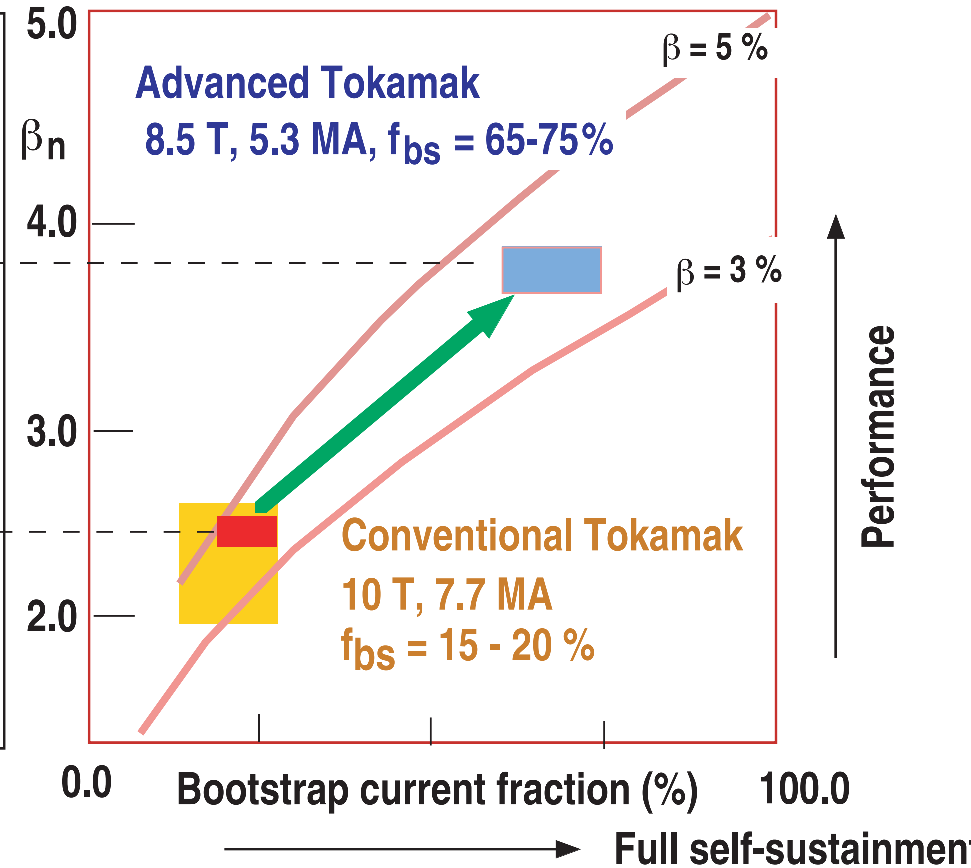
- High $\beta_n \rightarrow$ high bootstrap current fraction configuration
  - Example: Fusion Ignition Research Experiment (FIRE) device

External kink β_n limit

- $\beta_n = \beta/(I_p/aB_t)$



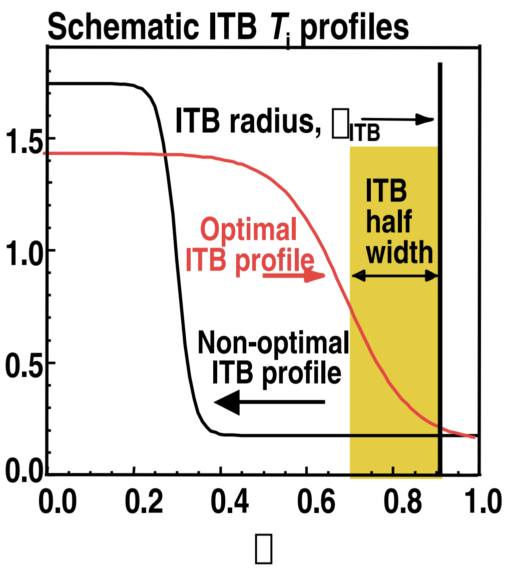
FIRE Performance

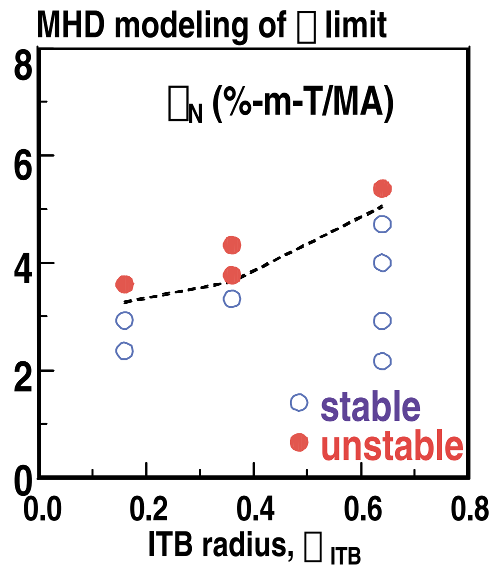


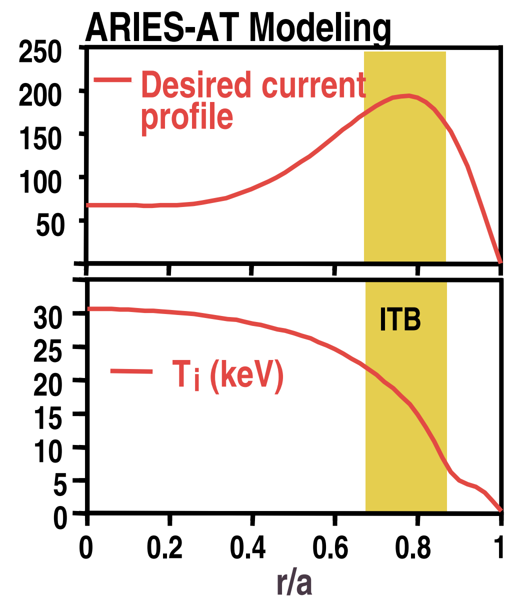
DIII-D NATIONAL FUSION FACILITY SAN DIEGO

176-02/jy

INTERNAL TRANSPORT BARRIER CONTROL IS ESSENTIAL





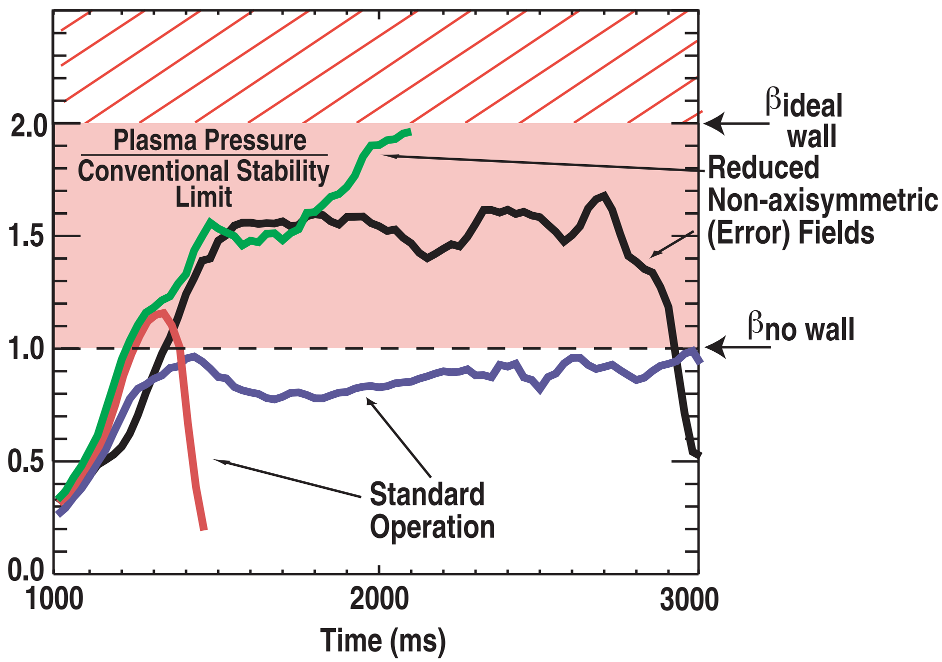


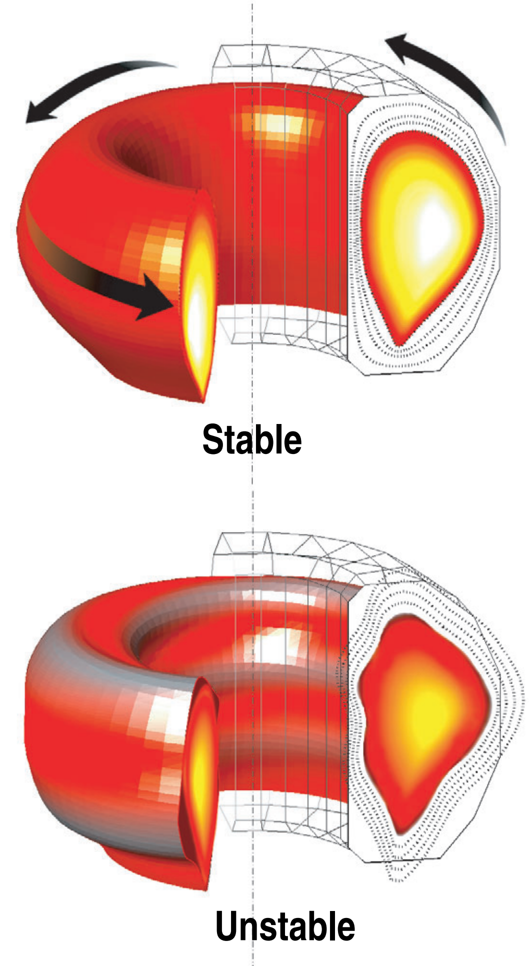
- Fusion performance: Need to maximize volume inside barrier.
- MHD stability: Beta limit maximized with barrier location and width.
- Bootstrap current: Better aligned with larger barrier position.
- *Large barrier radius and large barrier width both highly desirable.*

DIII-D NATIONAL FUSION FACILITY SAN DIEGO

WALL STABILIZATION LOOKS LIKE IT WILL WORK MAJOR BREAKTHROUGH IN 2001

- Spinning plasma improves prospects for fusion energy



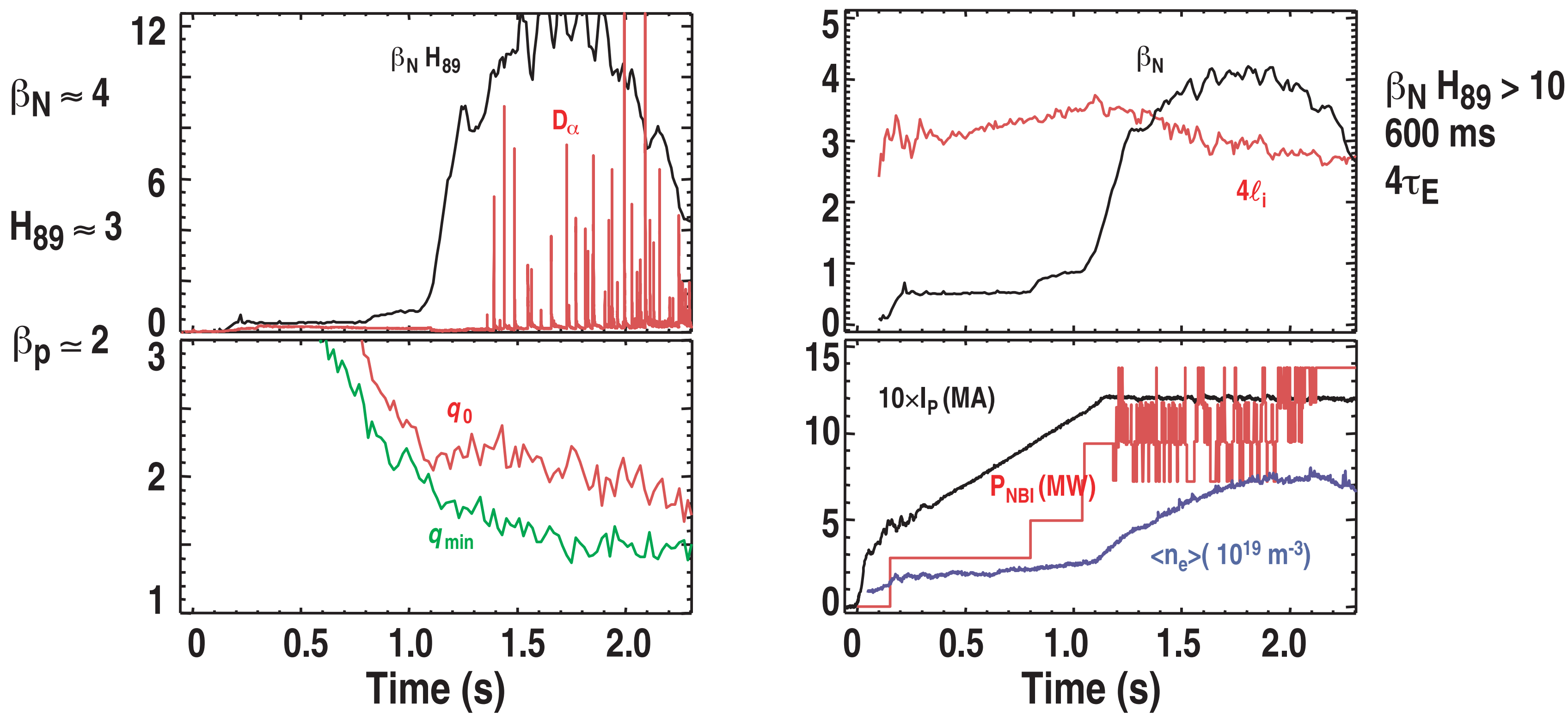


- $\beta_N > \beta_N^{\text{no wall}}$
- $\beta_N \sim \beta_N^{\text{ideal wall}}$

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

004-03/RDS/JY

RESISTIVE WALL MODE MITIGATION ALREADY ALLOWS OPERATION ABOVE NO-WALL LIMIT AT HIGH β_N



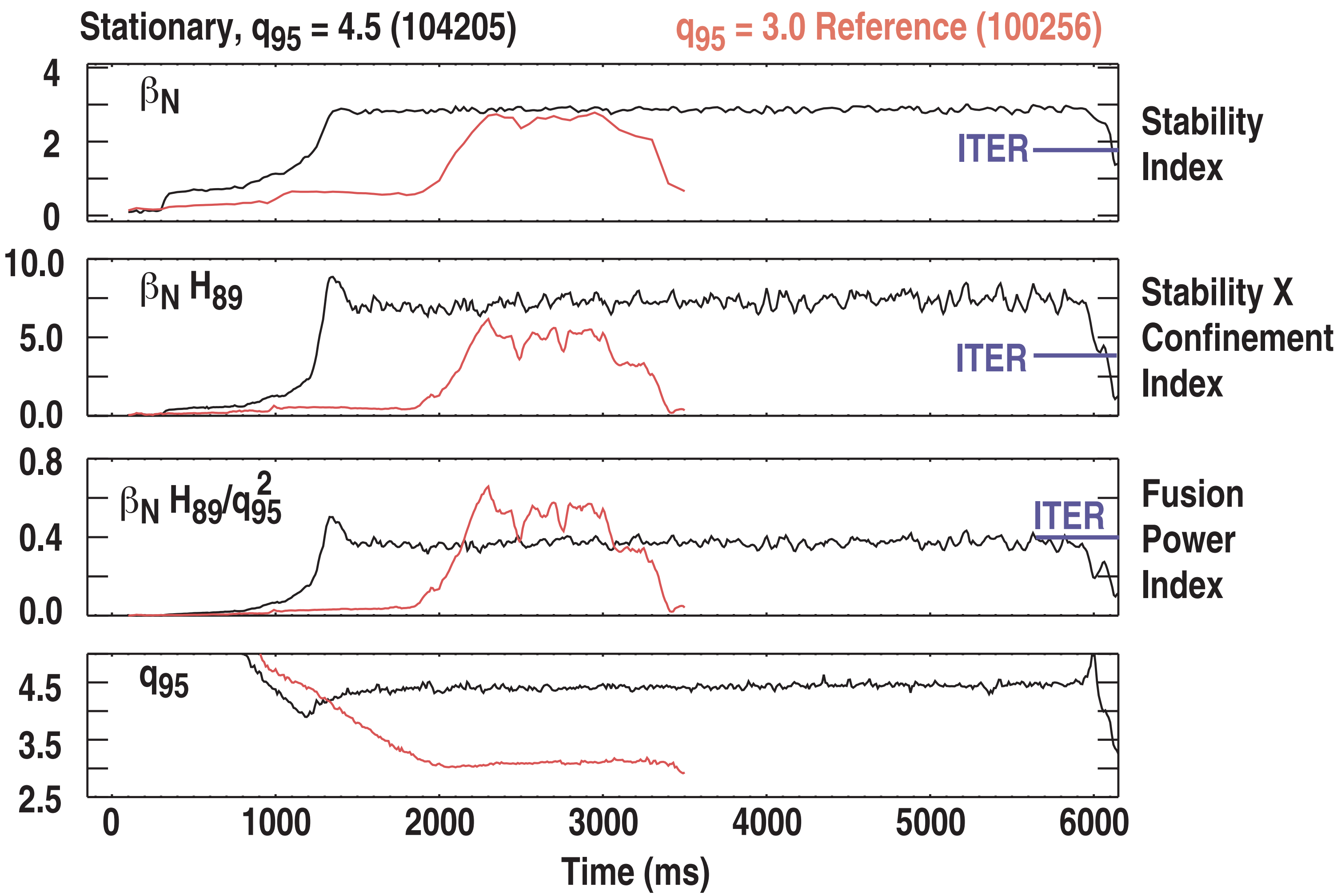
- Achieved through rotational stabilization of resistive wall mode
- Technique now in routine use during high beta AT experiments
- Duration and β limited by tearing mode as *q* profile evolves

COLLABORATION WITHIN IEA LARGE TOKAMAK FRAMEWORK (1)

- Two promising advanced scenarios have been identified for further investigation
  - one compatible with hybrid operation (long inductive pulse at high fusion yield)
  - and one compatible with steady state (fully non-inductive) operation

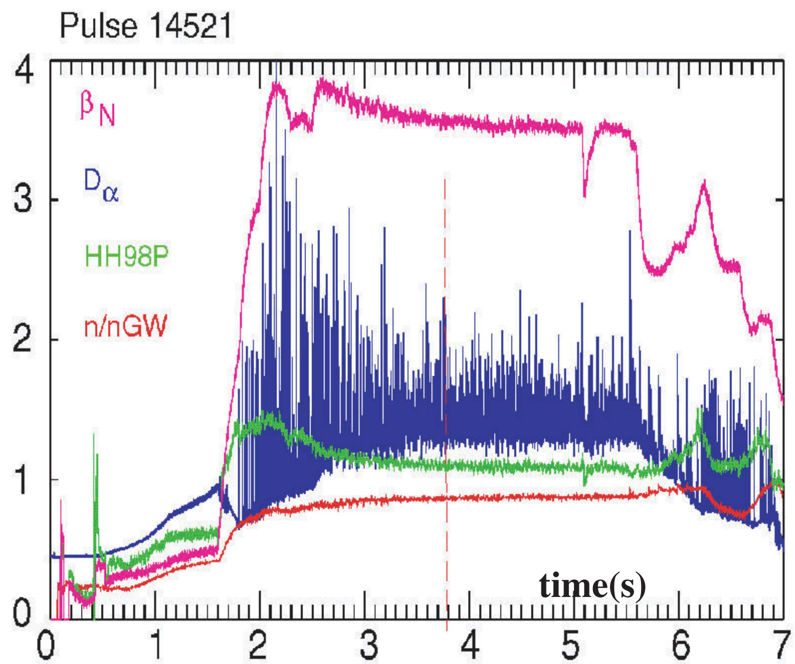
- Hybrid scenario is defined by a flat q profile with q_{min} >~1. The four largest divertor machines (JT-60U, JET, AUG, DIII–D) have operated at high normalized β(>2.5) under these conditions for long pulses (>20 τ_E)
  - The high priority is to map the existence domain for this scenario for each machine and identify across the domain the mechanism by which the current is prevented from peaking (fishbones, tearing mode, etc.). Variations of performance (e.g., $\beta\tau_{th}$) with q_{95} and density at fixed field and shape are of high interest. Variations with shape (performance and ELM behavior) and T_e/T_i are also of interest

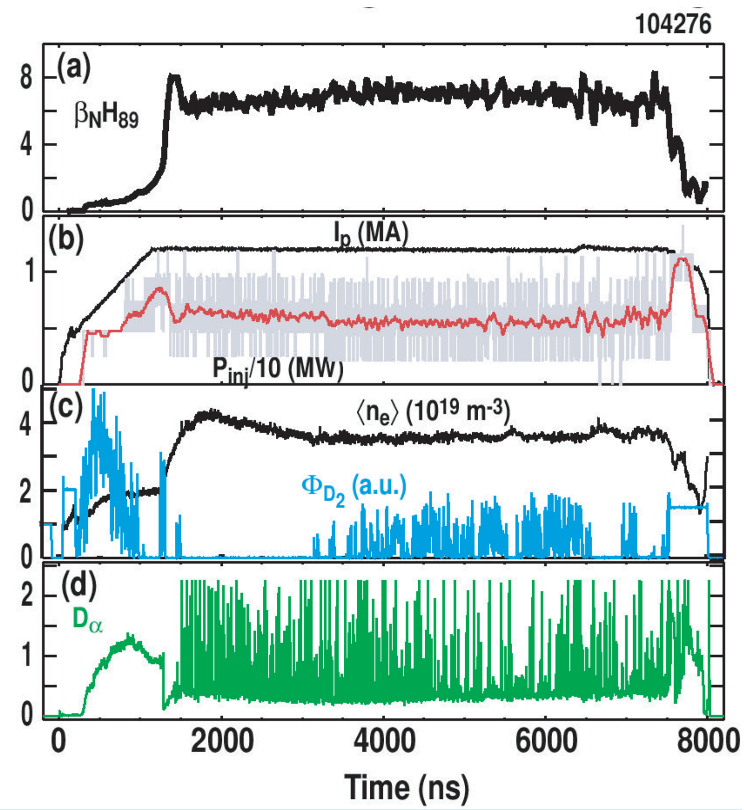
STATIONARY PLASMAS THAT WOULD ENABLE ITER TO RUN 4000 SECONDS AT 500 MW FUSION POWER HAVE BEEN DEMONSTRATED ON DIII–D



324-02/JY

SHALLOW SHEAR REVERSAL AT q_0 AROUND 1:
HIGH FUSION YIELD HYBRID ITER SCENARIOS





In AUG with NBI +off-axis NBCD high β_N at q_{95}=3.6, H_{98}=1.3, n/n_G ~1, I_{BS} /I_P~0.6 and type II ELMS

DIIID: High performance sustained for 35 τ_E (t_{relax}=1.8s)

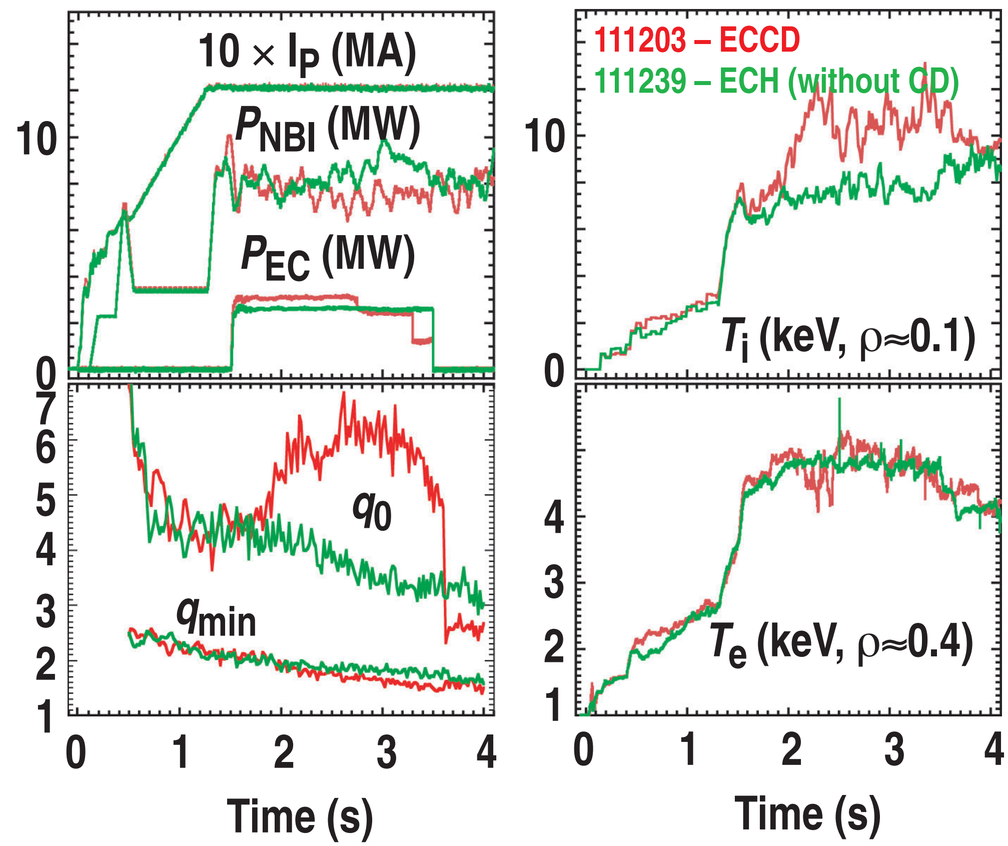
In both machines: no sawteeth. Question:
How to maintain q_0 very close to 1:Are fishbones(AUG) or mild tearing modes (DIIID) accceptable in a BPX?

RECENT DIII–D EXPERIMENTS HAVE DEMONTRATED THE ABILITY TO CONTROL THE CURRENT PROFILE IN HIGH PERFORMANCE DISCHARGES USING OFF-AXIS ECCD

High Bootstrap Fraction AT

I_p	1.2 MA	
ECCD	0.13 MA	10%
NBCD		30%
Bootstrap		53%
OHMIC		7%
Non-Inductive		93%

B_T	1.85 T
EC	2.5 MW
NB	8 MW
B_T	3.1%
β_N	2.8
H	2.5
$\beta_N H$	7

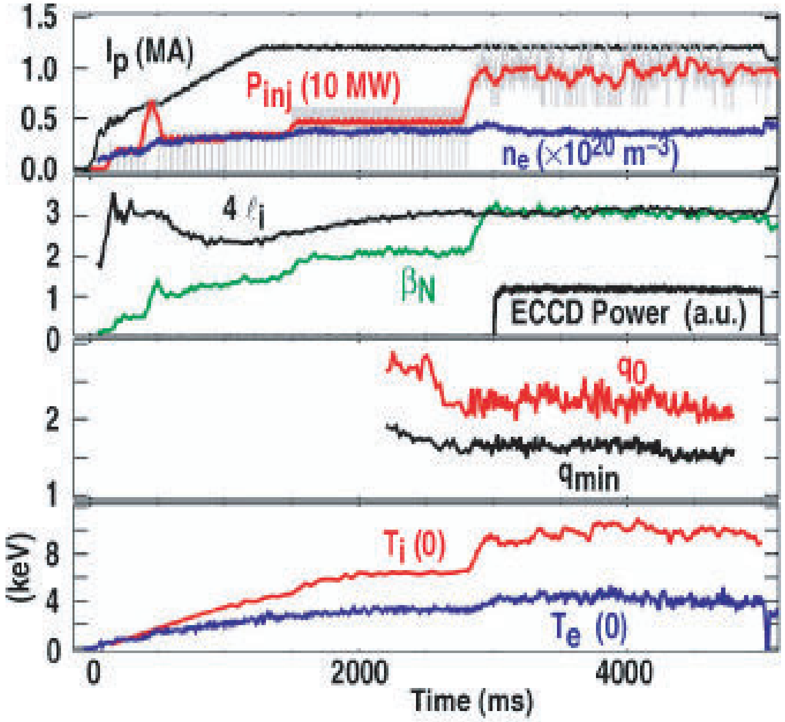


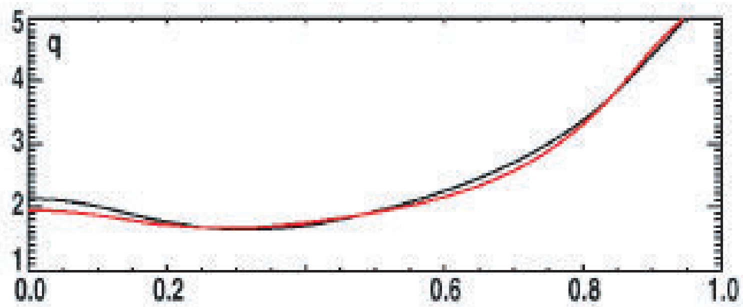
DIII–D NATIONAL FUSION FACILITY SAN DIEGO

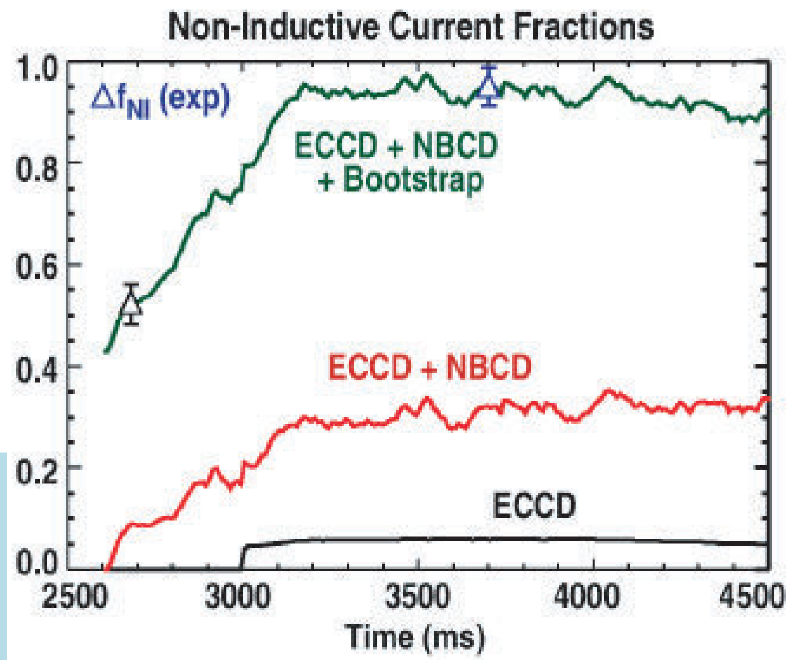
235-02/jy

RECENT RESULTS FROM DIII–D: HIGH BOOTSTRAP AT q_{min} ~1.5

β_N=3.1, β=3.3%,H_{89}=2.4



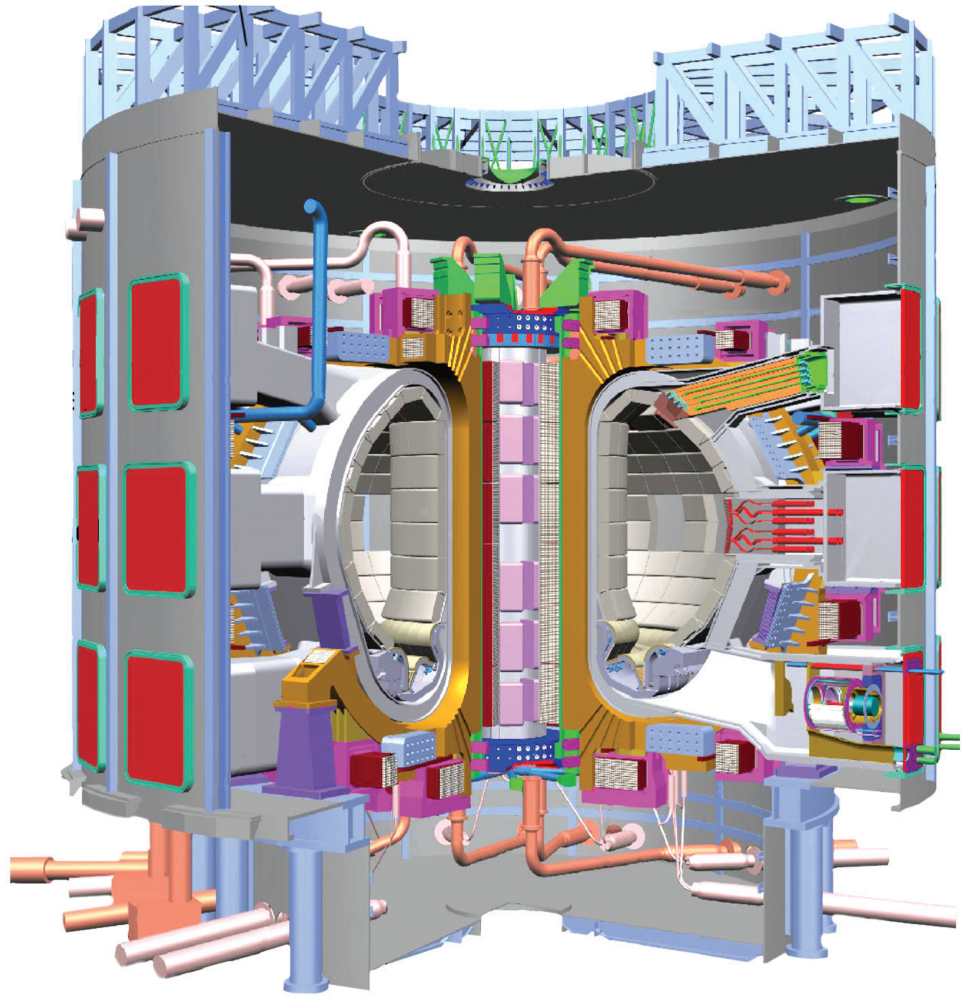




➢With off-axis ECCD, up to 95% NI current drive at the no-wall beta limit with β_N=3 and H_{89}=2.5

ITER MOVING FORWARD

- Four site proposals
- U.S. community recommends rejoining negotiations
- Others (China) seeking to join as full partners



GENERAL ATOMICS

Q= 10 reference scenario(s): milestone

Parameter	400 MW	560 MW	260 MW
R/a (m/m)	6.2/2.0	←	←
κ_{95}/δ_{95}	1.7/0.33	←	←
B_T (T)	5.3	←	←
I_P (MA)	15.0	←	←
q_{95}	3	←	←
$\langle n_e \rangle$ $(10^{20}m^{-3})$	1.01	1.18	0.83
$\langle n_e \rangle / n_G$	0.85	1.0	0.7
$\langle T_e \rangle$ (keV)	8.8	9.0	8.7
$\langle T_i \rangle$ (keV)	8.0	8.2	7.9
P_{FUS} (MW)	400	560	260
$P_{NB} + P_{RF}$ (MW)	33 + 7	33 + 23	17 + 9
Q	10	←	←
P_{RAD} (MW)	47	71	30
$P_{LOSS}/P_{L\text{-}H}$	1.8 (87/48)	2.4 (124/53)	1.3 (55/42)
β_N	1.8	2.1	1.4
β_P	0.65	0.77	0.52
li (3)	0.84	0.84	0.85
τ_E (s)	3.7	3.1	4.7
$H_{H98(y,2)}$	1.0	←	←
τ_{He}^{*}/τ_E	5.0	←	←
$f_{He,axis/ave}$ (%)	4.3/3.2	4.1/3.1	4.1/3.1
$f_{Be,\ axis}$ (%)	2.0	←	←
$f_{Ar,\ axis}^{*1}$ (%)	0.12	0.16	0.10
$Z_{eff,\ ave}$	1.66	1.77	1.60
V_{loop} (mV)	75	75	82

conservative requirements

ITER

steady state („advanced“) scenarios:

- development needed
- spectrum of scenarios
- scenarios illustrative

	Scenario 4		Scenario 6	Scenario 7	
	WNS	WNS	SNS	WPS	Low-Q
R/a (m)	6.35/1.85	6.35/1.85	6.35/1.85	6.35/1.85	6.35/1.85
B_T (T)	5.18	5.18	5.18	5.18	5.18
I_p (MA)	9.0	9.5	9.0	9.0	11.0
κ_{95}/δ_{95}	1.85/0.40	1.87/0.44	1.86/0.41	1.86/0.41	1.84/0.43
$<n_e>$ ($10^{19}m^{-3}$)	6.7	7.1	6.5	6.7	5.7
n/n_G	0.82	0.81	0.78	0.82	0.57
$<T_i>$ (keV)	12.5	11.6	12.1	12.5	9.3
$<T_e>$ (keV)	12.3	12.6	13.3	12.1	12.1
β_T (%)	2.77	2.67	2.76	2.75	2.2
β_N	2.95	2.69	2.93	2.92	1.9
β_p	1.49	1.25	1.48	1.47	0.77
P_{fus} (MW)	356	338	340	352	174
$P_{RF} + P_{NB}$ (MW)	29 + 30 *1	35 + 28 *1	40 + 20 *2	29 + 28 *3	36 + 50
$Q = P_{fus}/P_{add}$	6.0	5.36	5.7	6.2	2.0
W_{th} (MJ)	287	292	287	284	212
P_{loss}/P_{L-H}	2.59	2.74	2.63	2.6	3.0
τ_E (s)	3.1	2.92	3.13	3.07	2.15
f_{He} (%)	4.1	4.0	4.0	4.0	3.0
f_{Be} (%)	2	2.0	2	2	2
f_{Ar} (%)	0.26	0.16	0.2	0.23	0.19
Z_{eff}	2.07	1.87	1.89	1.99	1.86
P_{rad} (MW)	37.6	30.6	36.2	34.6	22
P_{loss} (MW)	92.5	100.0	91.6	92.7	99
l_i (3)	0.72	0.43	0.6	0.69	0.58
I_{CD}/I_p (%)	51.9	49.7	53.7	50.2	73.6
I_{bs}/I_p (%)	48.1	50.3	46.3	49.8	26.4
I_{OH}/I_p (%)	0	0	0	0	0
$q_{95}/q_o/q_{min}$	5.3/3.5/2.2	5.0/3.8/2.7	5.4/5.9/2.3	5.3/ 2.7/2.1	4.1/ 1.5/1.3
$H_{H98(y,2)}$	1.57	1.46	1.61	1.56	1.0
τ_{He}^*/τ_E	5.0	5.0	5.0	5.0	5.0

WITH ADEQUATE RESOURCES, FUSION PROGRESS CAN EVOLVE RAPIDLY

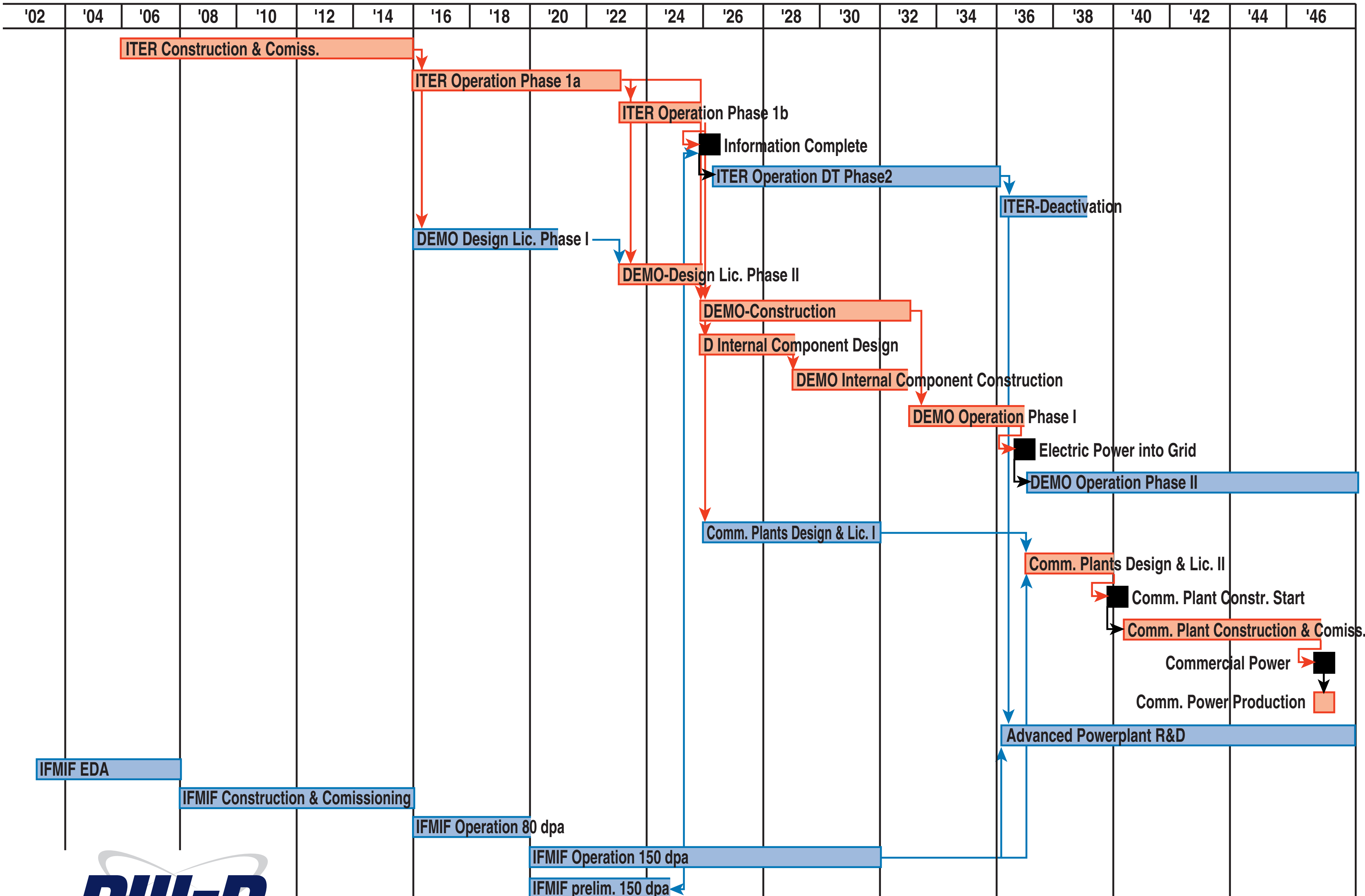
1. Advanced Tokamak, steady-state basis will be available before ITER operates

2. First phase of ITER will focus on advanced, long pulse modes, not the conventional OH driven operation

3. Work in ITER and parallel actual long pulse work in other superconducting machines will establish steady-state operation by the end of ITER phase 1a

4. The plasma physics will be in hand for a steady-state, high performance demo and for possible use of ITER as a CTF

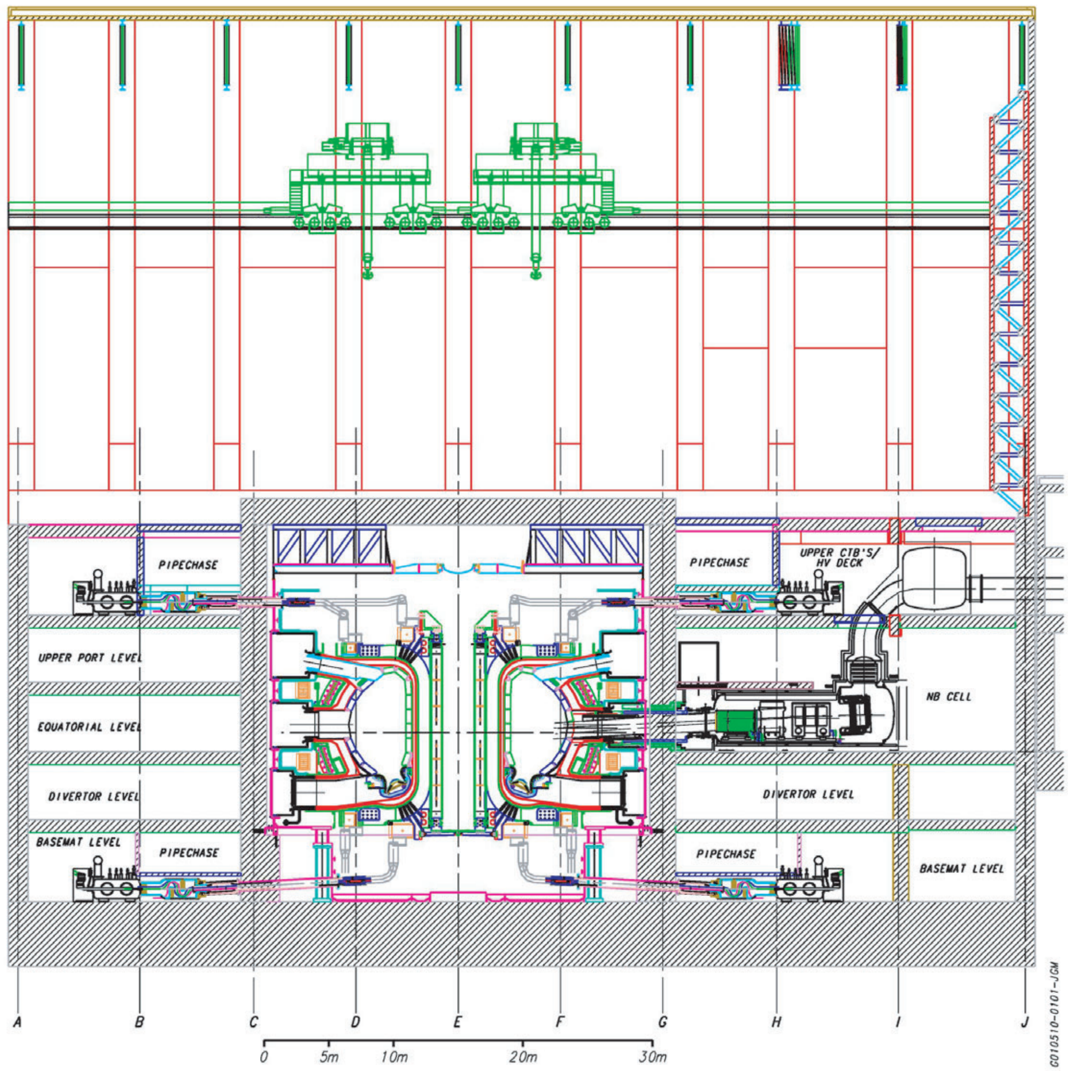
ROADMAP TO FUSION POWER: POSSIBILITIES FOR ACCELERATION (K. LACKNER)



DIII-D NATIONAL FUSION FACILITY SAN DIEGO

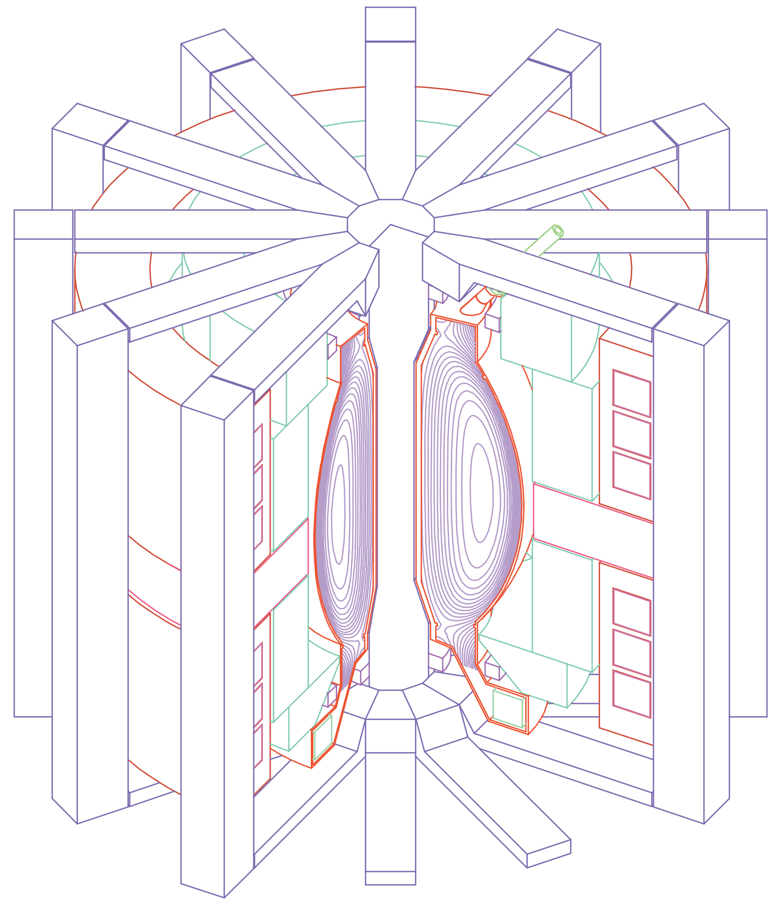
004-03/RDS/JY

ITER RADIAL ACCESS-SECTOR MAINTENANCE SCHEME



MISSIONS AND DESIRES OF A COMPONENT TEST FACILITY

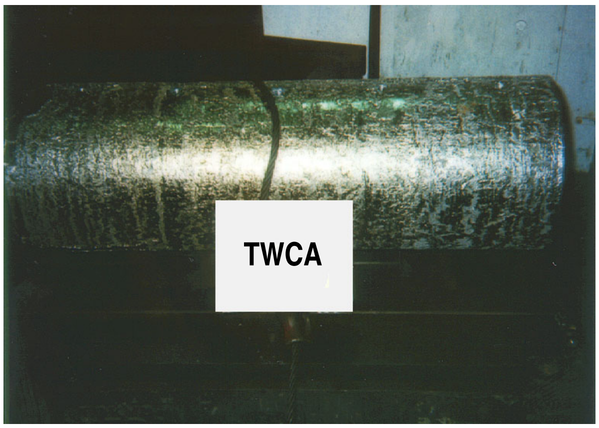
- Small, low cost
- Neutron wall loading 2 MW/m^2
- Steady-state/high duty factor (80%)
- Fusion energy technology development
- Tritium breeding development
- Maintainable
- Flexible to blanket changeouts



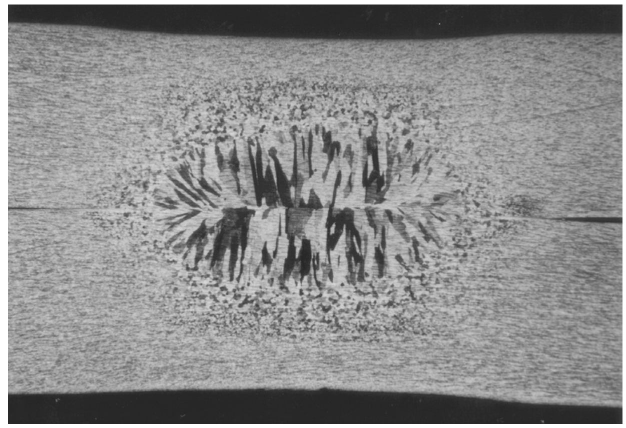
GENERAL ATOMICS

VANADIUM ALLOY MANUFACTURING TECHNOLOGY DEVELOPMENT

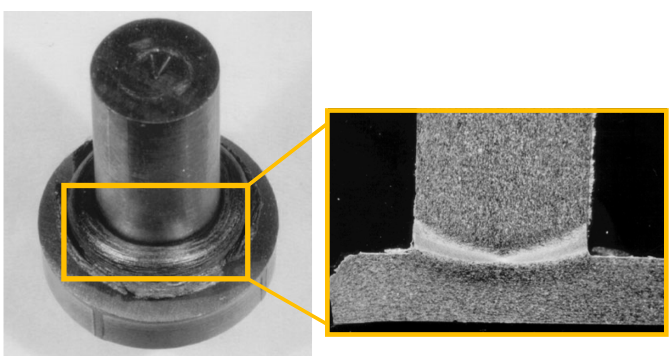
- Collaboration with GA, ANL, ORNL, PNL



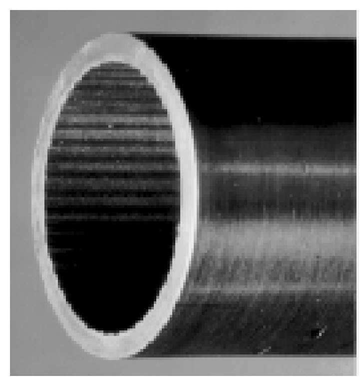
800 kg Ingot of Vanadium for Radiative Divertor



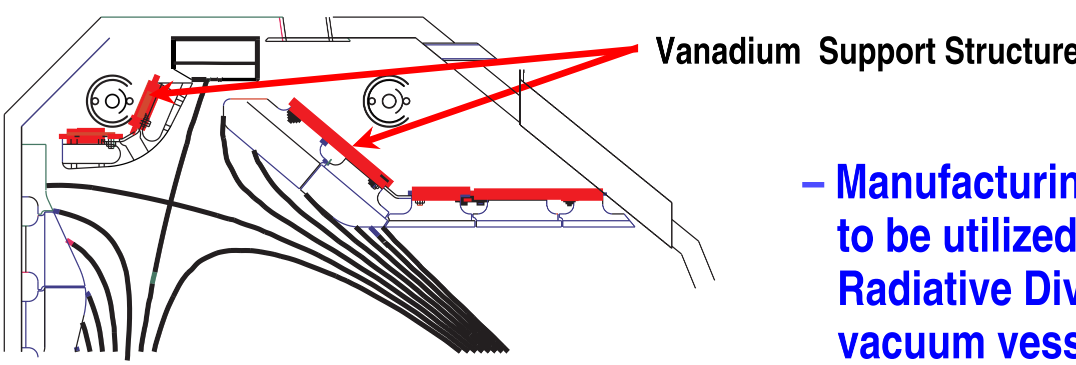
Resistance Weld Joining .150” Thick Vanadium Alloy Sheets



Inertial Weld of Vanadium Alloy Stud to Vanadium Alloy Plate



Section of High Heat Flux Module Made from Vanadium Alloy Tubing Developed by GA



Radiative Divertor for DIII–D

– Manufacturing technologies to be utilized in the DIII–D Radiative Divertor . . . future vacuum vessels

GENERAL ATOMICS

THERE ARE MANY BLANKET CONCEPTS TO TEST AND DEVELOP

Worldwide Blanket Options for DEMO*

Breeder	Coolant	Structural Material
Solid breeders Li_2O, Li_4SiO_4, Li_2ZrO_3, Li_2TiO_3	Helium or H_2O	Ferritic steel, vanadium alloy, SiC composites
Self-cooled liquid-metal breeders Lithium, LiPb	Lithium, LiPb	Ferritic steel, vanadium alloy with electric insulator, SiC composites with LiPb only
Separately cooled liquid-metal breeders Lithium LiPb	Helium Helium or H_2O	Ferritic steel, vanadium alloy Ferritic steel, vanadium alloy, SiC composites

*Almost all concepts use beryllium as the neutron multiplier

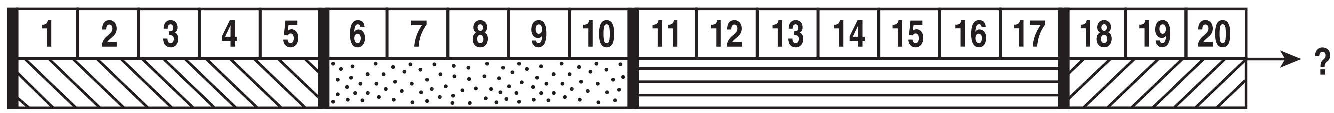
- Need a CTF that can accomodate breakdowns in test blanket modules–accomodate three (?) iterations per blanket type

Results of an international study on a high volume plasma based neutron source for fusion blanket development, M. Abdou et. al., Fusion Technology 29 (Jan. 1996) 1.

DIII-D NATIONAL FUSION FACILITY SAN DIEGO

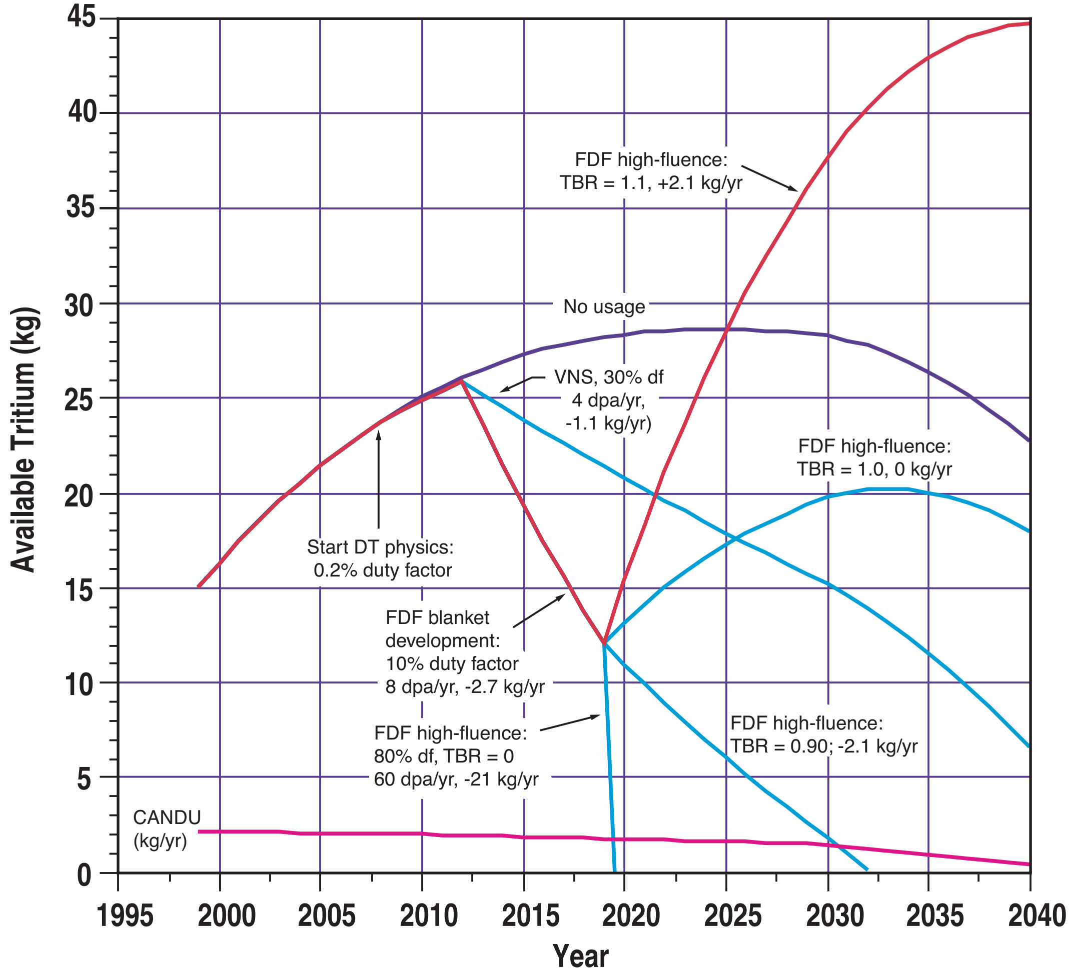
COMPONENT TEST FACILITY

- Steady progress through sequenced objectives



PHASES	D-D PHYSICS	DT PHYSICS	BLANKET DEVELOPMENT	TRITIUM SELF SUFFICIENCY
FUSION POWER (MW)	~0	10 → 300	400	500
ELECTRIC POWER (MW)	–200	–400	–400	–400
DUTY CYCLE	0.4%	0.4%	0.4% → 10%	100%
	60 SECOND PULSES	60 SEC — 5 MIN.	~ WEEK LONG RUNS	STEADY-STATE
NEUTRON WALL LOAD (MW/m^2)		2	3	3-4
TRITIUM PER YEAR				
BURNED (kg)	~0	0.1	2.7	27
PRODUCED [NET] (kg)	0	0	3.1 [0.4]	31 [4]

CTF WILL NEED TO BE AT LEAST TRITIUM SELF-SUFFICIENT AT HIGH DUTY FACTOR

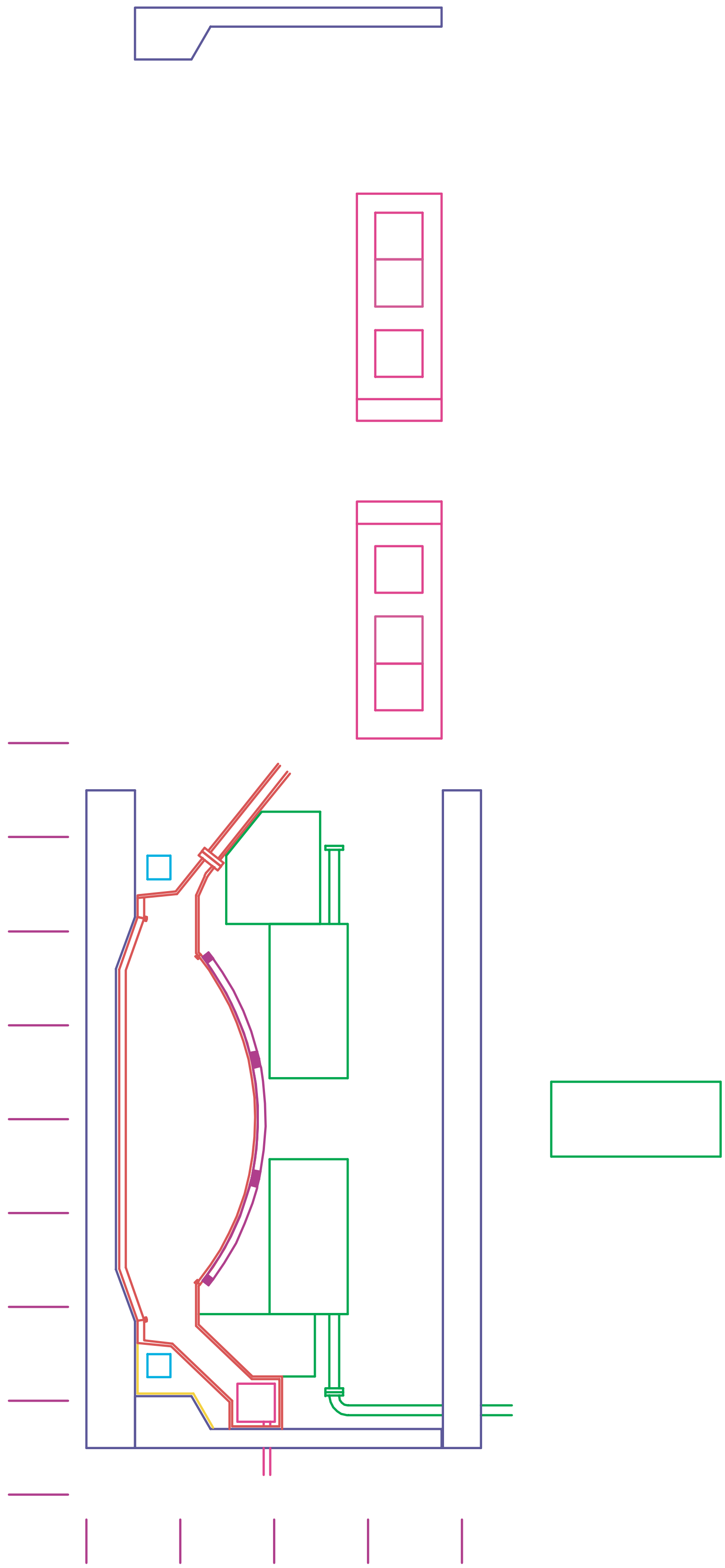


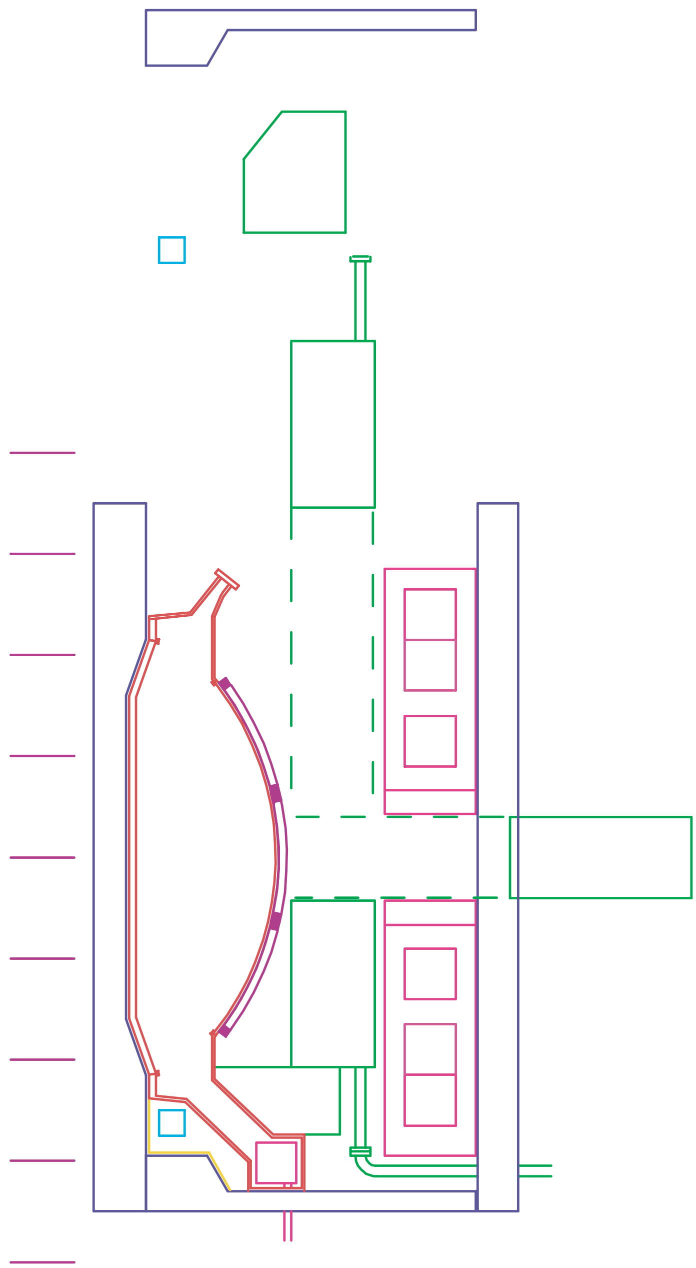
004-03/RDS/JY

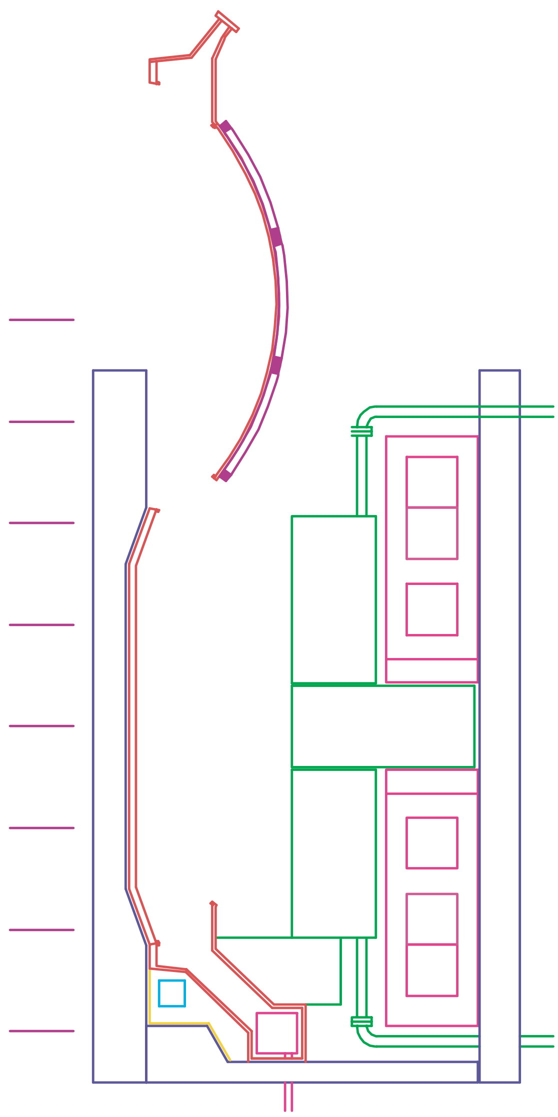
GENERAL ATOMICS

SOME MAJOR DESIGN CONSIDERATIONS

- Maintainability and flexibility
  - Sector versus vertical maintenance
- Two consistent packages
  - Shielded (0.5 m) multi-turn insulated TF coil and OH coil
    - ★ Longer component life
    - ★ Better impedance match of TF coil to power supply (0.5 MA, ~300V)
  - Minimally shield single turn TF with no OH
    - ★ Requires unusual 20-60 MA, 10V power supply
- Aspect ratio choice
    - ★ Appears a relatively unconstrained choice from physics

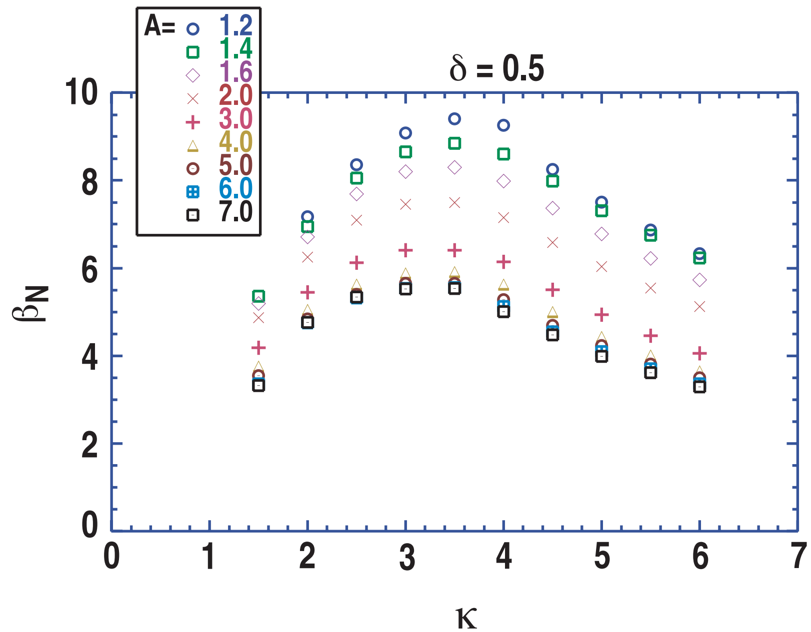




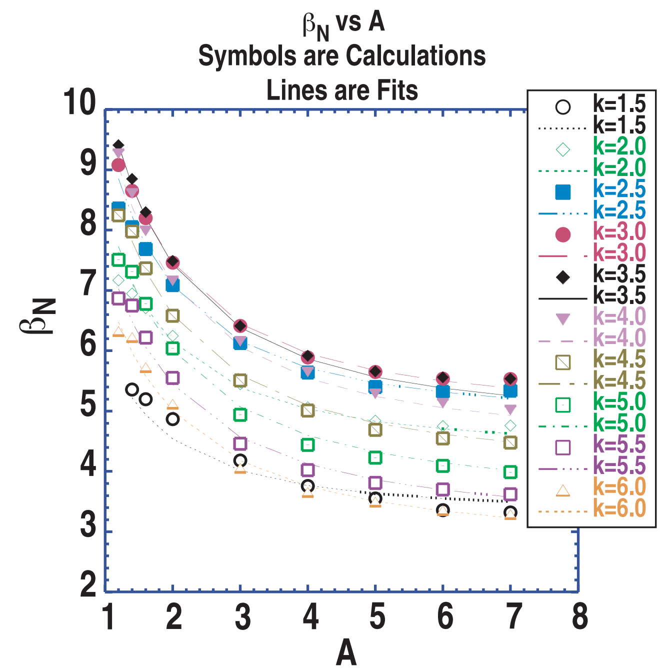


SYSTEMATIC STUDY OF β_N-LIMIT VS A AND κ FOR f_{bs} = 0.99

β_N vs K



β_N vs A



- Optimum plasma states for next step tokamaks, Y.R. Lin-Liu and R.D. Stambaugh, GA-A23980 (6/02) submitted to Nuclear Fusion

DIII-D NATIONAL FUSION FACILITY SAN DIEGO

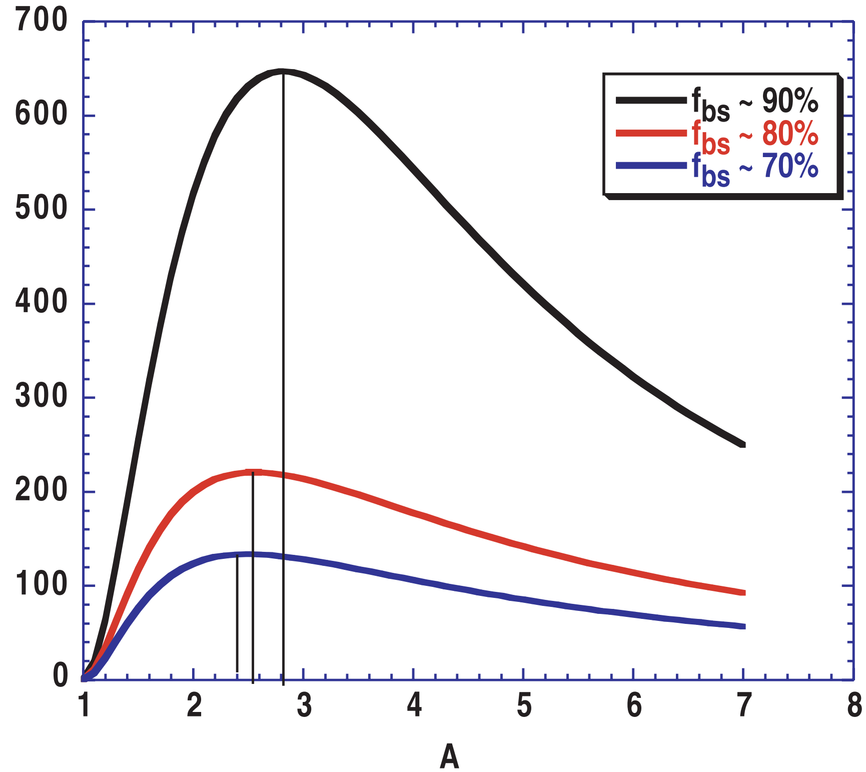
004-03/RDS/JY

USING LIN-LIU'S β-LIMIT DATA, ONE Q INDEX OPTIMIZES FOR A = 2.4-2.8

$$Q = \frac{P_F}{P_{CD}} = \frac{\gamma_{CD} P_F}{nIR(1 - f_{bs})} = \frac{\gamma_{CD}\,\beta_N^2\,\kappa\,B_c^2\left(1 - \frac{1}{A}\right)^2 Ra^2}{f_{GR}\,R(1 - f_{bs})}$$

- $n = f_{GR} \dfrac{I}{\pi a^2}$
- B_C = field at centerpost (fixed maximum from stress)
- $f_{bs} = c_{bs}\,\beta_p / \sqrt{A} = \dfrac{c_{bs}}{20}\sqrt{A}\;\; q_{cyl}\;\; \beta_N$
- Express $\beta_N(A)$ as $\beta_{NO}A^{-\alpha}$ and κ as $\kappa_O A^{-\phi}$
- Optimize the function $$\frac{A^{-2\alpha}\left(1 - \frac{1}{A}\right)^2 A^{-\phi}}{\left(1 - \frac{c_{bs}}{20}\; q_{cyl}\;\; \beta_{N0}\; A^{1/2-\alpha}\right)}$$

for α = 1/2; ϕ = 1/2 A_{max} = 2.3
α = 1; ϕ = 1/2 A_{max} = 1.5



GENERAL ATOMICS

004-02/RDS/JY

CONFINEMENT QUALITY IS A CHALLENGE TO FIND A SMALL MACHINE

- Scalings
  - $ITER_{89P}$, $\tau_{89p} = 0.048\, I_p^{0.85}\, R_0^{1.2}\, a^{0.3}\, n^{0.1}\, B_o^{0.2}\, (m_{D-T}\, \kappa/P_{HEAT})^{0.5}$
  - $ITER_{98(y,2)}$, $\tau_{98} = 0.0562\, I_p^{0.93}\, R_0^{1.97}\, A^{-0.58}\, \kappa^{0.78}\, B_o^{0.15}\, n_{19}^{0.41}\, m_{DT}^{0.19}\, P_L^{-0.69}$
  - τ_{GB} (Petty) $= 0.028\, I_p^{0.83}\, B_T^{0.07}\, n_{19}^{0.49}\, R_0^{2.11}\, A^{-0.3}\, \kappa^{0.75}\, m^{0.14}\, P_L^{-0.55}$
    - ★ $\tau_E \propto B_T^{-1}\, \rho_*^{-3}\, \beta^{0}\, \nu_*^{-0.15}\, q^{-1.7}$

Scalings Reward	CTF Wants
Large Size	Small Size
High Current	Lowish Current
Low q	Medium q
Low Bootstrap Fraction	High Bootstrap Fraction
High Plasma Density	Low Plasma Density
Inductive Operation	Steady-State
Low Power Density	High Power Density

C. Petty, et. al., "Feasibility study of a compact ignition Tokamak based upon gyrobohm scaling physics" GA-A23590, Fusion Technology (2001)

A SMALL CTF MACHINE–OPERATING MODES

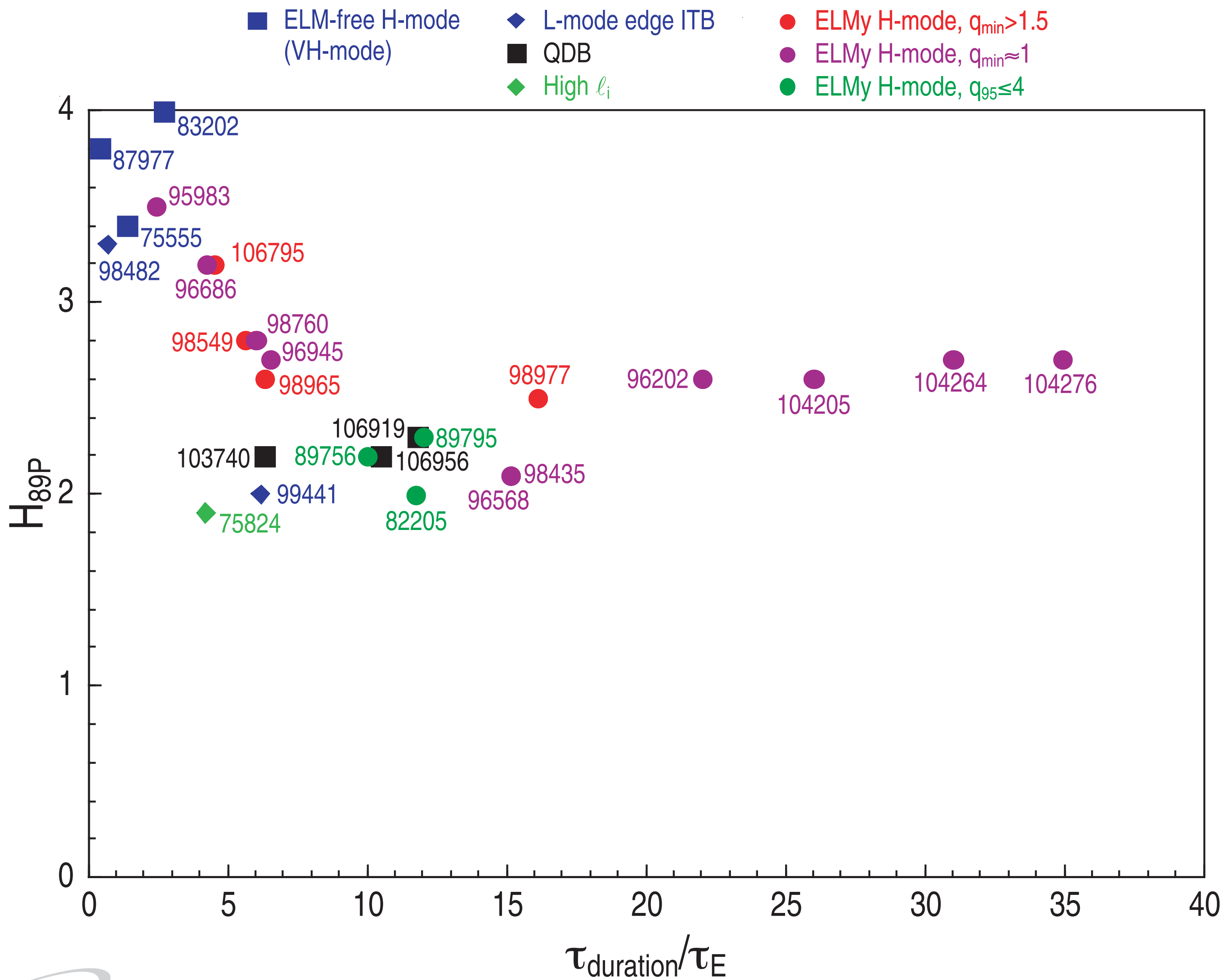
A = 2.5, 65% OF β–LIMIT, WALL LOAD 2 MW/m^2

			Standard Physics	Bootstrap 0.5	Bootstrap 0.7	Bootstrap 0.9
A	aspect ratio		2.50	2.50	2.50	2.50
k	plasma elongation		2.51	2.51	2.51	2.51
Ro	plasma major radius	m	2.17	2.17	2.17	2.17
a	plasma minor radius	m	0.87	0.87	0.87	0.87
Rsol	solenoid radius	m	0.40	0.40	0.40	0.40
Rtf	toroidal coil radius	m	0.80	0.80	0.80	0.80
Tshield	Thickness of blanket/shield	m	0 .50	0.50	0.50	0.50
Jc	centerpost current density	MA/m2	30.10	28.50	34.00	38.10
Bo	field on axis	T	4.19	3.97	4.73	5.30
Bc	field at conductor	T	11.35	10.74	12.82	14.36
Pf	fusion power	MW	312.96	320.58	331.30	316.02
Pn/Awall	Neutron Power at Blanket	MW/m2	1.96	2.01	2.08	1.98
Pinternal	power to run plant	MW	537.69	399.34	471.82	550.36
Qplasma	Pfusion/Paux		3.02	5.49	6.63	6.32
BetaN	normalized beta	mT/MA	3.00	4.12	4.12	4.12
fbs	bootstrap fraction		0.30	0.50	0.70	0.90
BetaT	toroidal beta		0.13	0.14	0.10	0.08
Ip	plasma current	MA	15.13	11.79	10.05	8.76
framp	induct ramp frac		0.08	0.1	0.12	0.14
Pcd	current drive power	MW	103.58	58.35	30.33	8.61
Paux	total auxiliary power	MW	103.58	58.35	50.00	50.00
Pheat	Total Heating Power	MW	166.17	122.47	116.26	113.2
Ti(0)	Ion Temperature	keV	20.00	20.00	20.00	20.00
Te(0)	Electron Temperature	keV	20.00	20.00	20.00	20.00
n(0)	Electron Density	E20/m3	2.26	2.28	2.32	2.27
nbar/nGR	Ratio to Greenwald Limit		0.28	0.37	0.44	0.49
Zeff			2.40	2.40	2.40	2.40
W	Stored Energy in Plasma	MJ	101.96	103.19	104.90	102.45
TauE	TauE	sec	0.61	0.84	0.90	0.91
H	H factor over 89P L-mode		1.99	2.99	3.46	3.77
HITER98Y2	H factor over ELMY H		1.21	1.77	2.08	2.30
Hpetty	GyroBohm, No Beta dep		0.99	1.43	1.68	1.86

Aspect Ratio 2.5, 65% of the Beta Limit, Neutron Wall Loading 2 MW/m2

004-03/RDS/JY

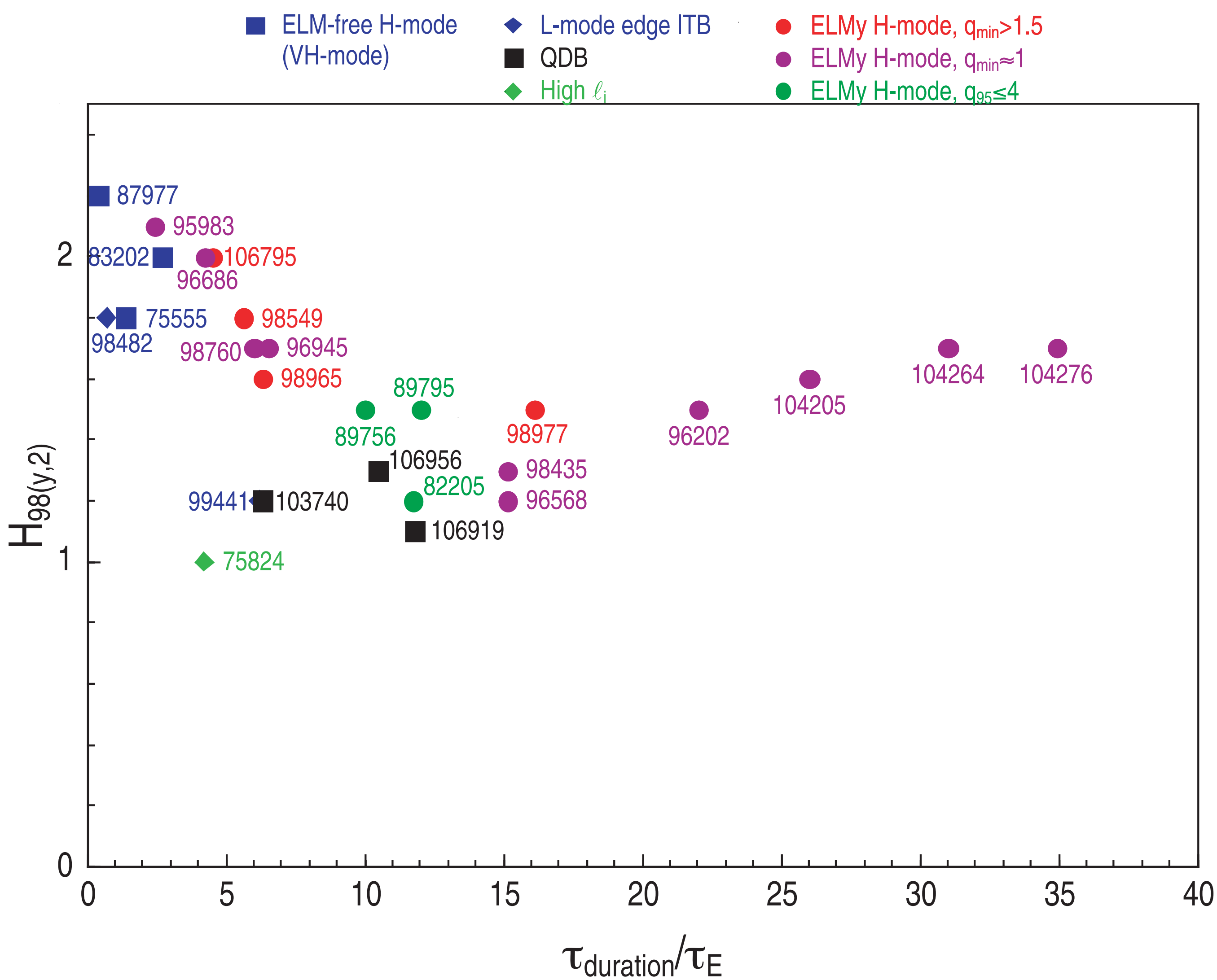
DIII–D CONFINEMENT RESULTS AGAINST ITER$_{89P}$ L–MODE SCALING



DIII–D NATIONAL FUSION FACILITY SAN DIEGO

004-03/RDS/JY

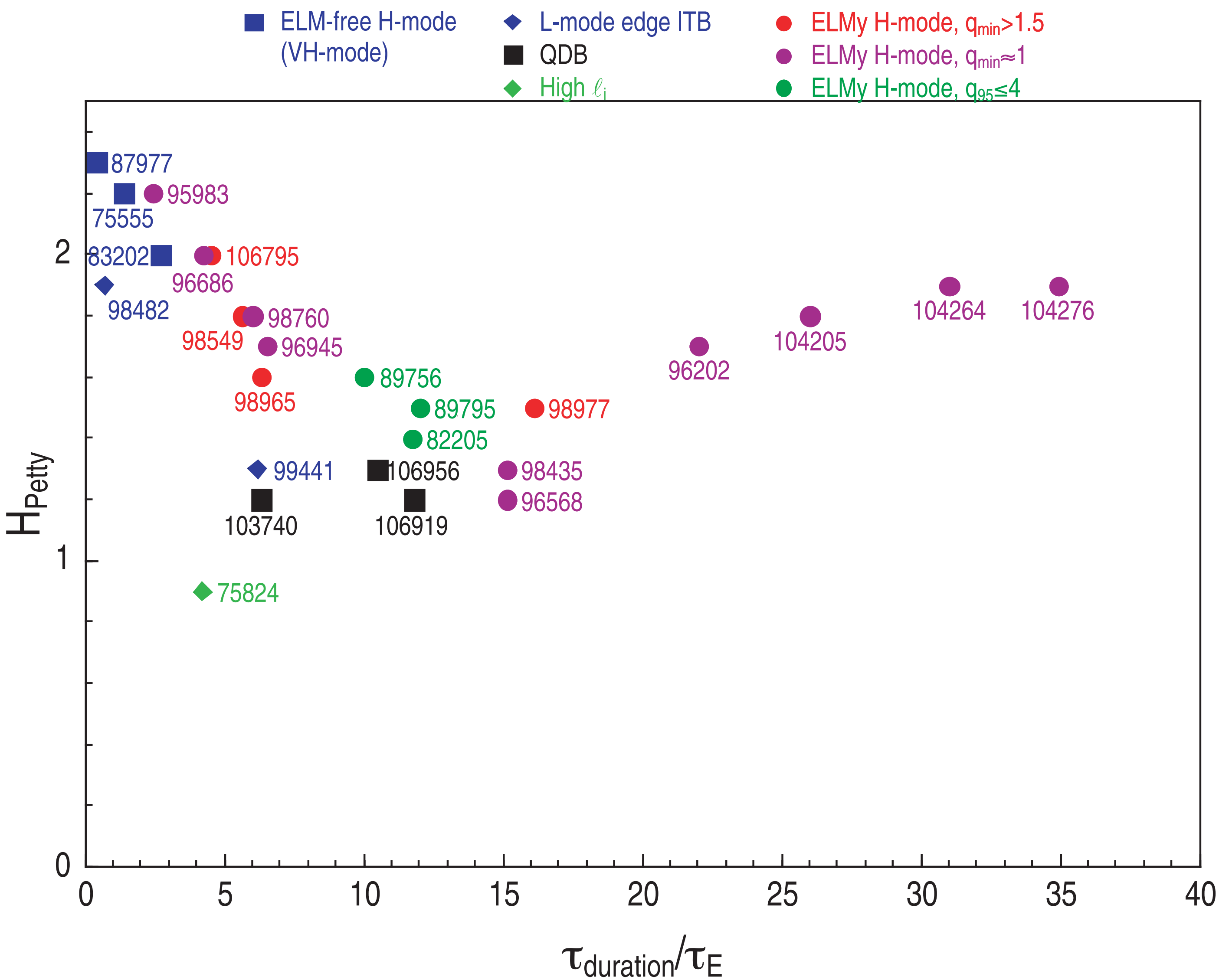
DIII–D CONFINEMENT RESULTS AGAINST ITER ELMy H–MODE SCALING



DIII–D
NATIONAL FUSION FACILITY
SAN DIEGO

004-03/RDS/JY

DIII–D CONFINEMENT RESULTS AGAINST PETTY'S GYROBOHM SCALING



DIII–D NATIONAL FUSION FACILITY SAN DIEGO

004-03/RDS/JY

COST ESTIMATES FOR CTF

- **Construction**
  - DIII–D ($0.6B) × $(2.17/1.67)^2$ = $1.0B
  - TPX (~$0.75B in 1995) × $(1.04)^7$ = $1.0B
  - Spreadsheet with ITER component costs = $1.2B
- **Blanket development program**
  - $25-$50M per year
- **Operating costs**
  - $100M/year for staff, non-electricity costs
  - Electricity costs a key concern ($44M/100MW @ 5¢/KWH)
    - ★ 400MW @ 10% duty factor ⇒ $18M/year
    - ★ 400MW @ 100% duty factor ⇒ $176M/year

D–D					D–T					Blanket Dev							Steady-State		
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20 →

		D–D	D–T	Blanket Dev	Steady-State
Construction	$1.X B				
Blankets		0	Shielding	$25-50M/yr	$25-50M/yr
Operation (50MW base load)		$123M/yr	$123M/yr	$140M/yr	$276M/yr
Total	($1-2B)	$123M/yr	$130M/yr	$165-190M/yr	$301-326M/yr

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

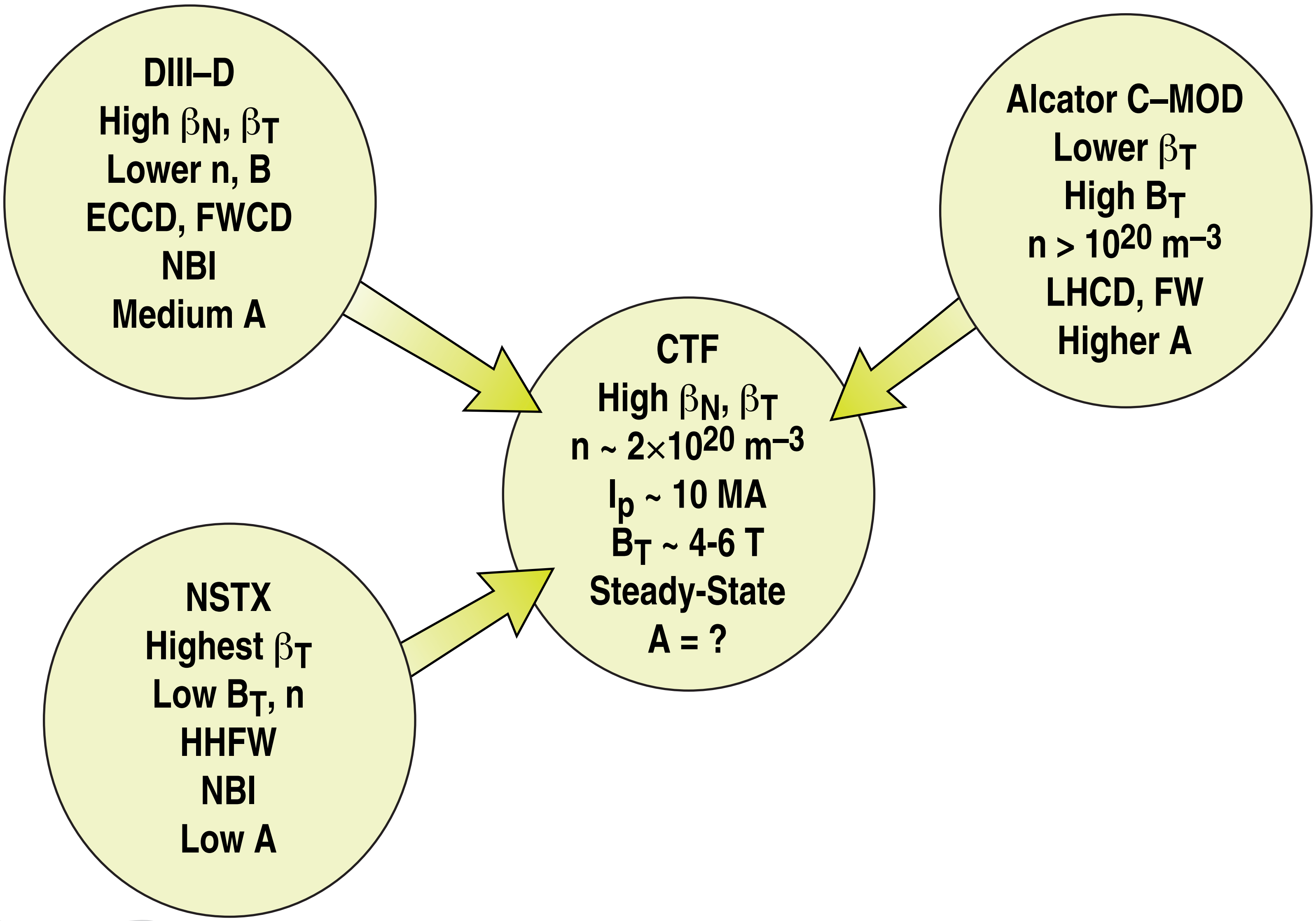
004-03/RDS/JY

A CTF DESIGN STUDY IS NEEDED

Key issue is electric cost

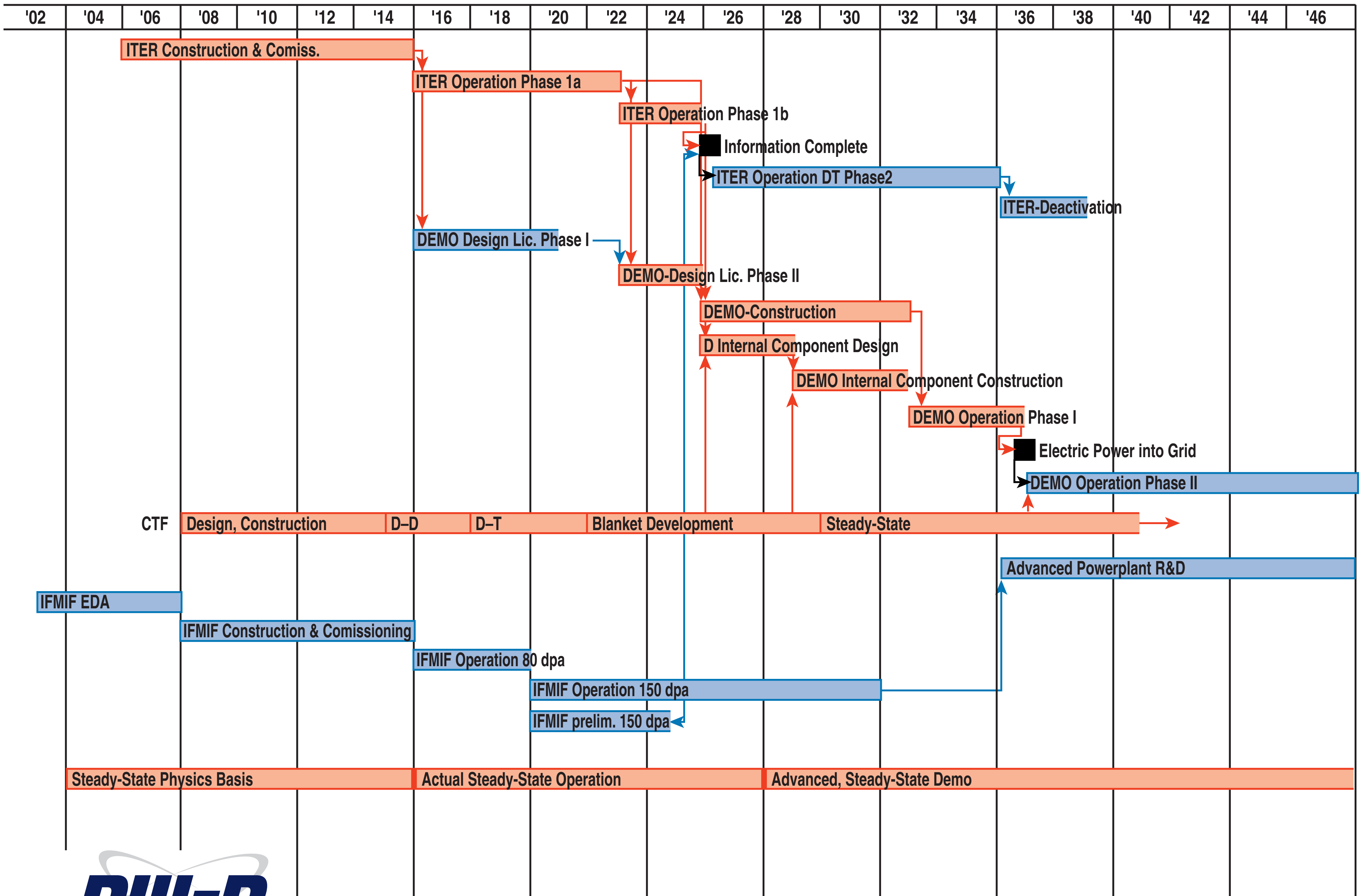
- Mission – do we need 100% duty factor or will 10% do ? – (Abdou's “Fusion Break-in” stage)
- Lower neutron flux requirement – 1 MW/m^2
  — Power exhaust (P/R ~60, twice ITER)
- Use more advanced physics (β_N > 4)
- Get rid of OH solenoid and minimize shielding of the single turn centerpost
- Make it large enough to make its own electricity in the steady-state phase

THE U.S. HAS A GOOD SET OF MACHINES TO TRIANGULATE ON A CTF



DIII–D NATIONAL FUSION FACILITY SAN DIEGO

ROADMAP TO FUSION POWER: POSSIBILITIES FOR ACCELERATION (R.S.)



DIII–D NATIONAL FUSION FACILITY SAN DIEGO

004-03/RDS/JY

WITH ADEQUATE RESOURCES, FUSION PROGRESS CAN EVOLVE RAPIDLY

1. Advanced Tokamak, steady-state basis will be available before ITER operates

2. First phase of ITER will focus on advanced, long pulse modes, not the conventional OH driven operation

3. Work in ITER and parallel actual long pulse work in other superconducting machines will establish steady-state operation by the end of ITER phase 1a

4. The plasma physics will be in hand for a steady-state, high performance demo and for possible use of ITER as a CTF

5. A CTF design study is needed