

Getting Ready for the Technical Assessment at Snowmass



“Go ahead, make my FIRE”

Diversified International Portfolio for Magnetic Fusion and FIRE

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for the FIRE Study Team

Discussion at
Office of Fusion Energy Science, DOE
Germantown, MD

May 7, 2002

<http://fire.pppl.gov>

FIRE

Lighting the Way to Fusion



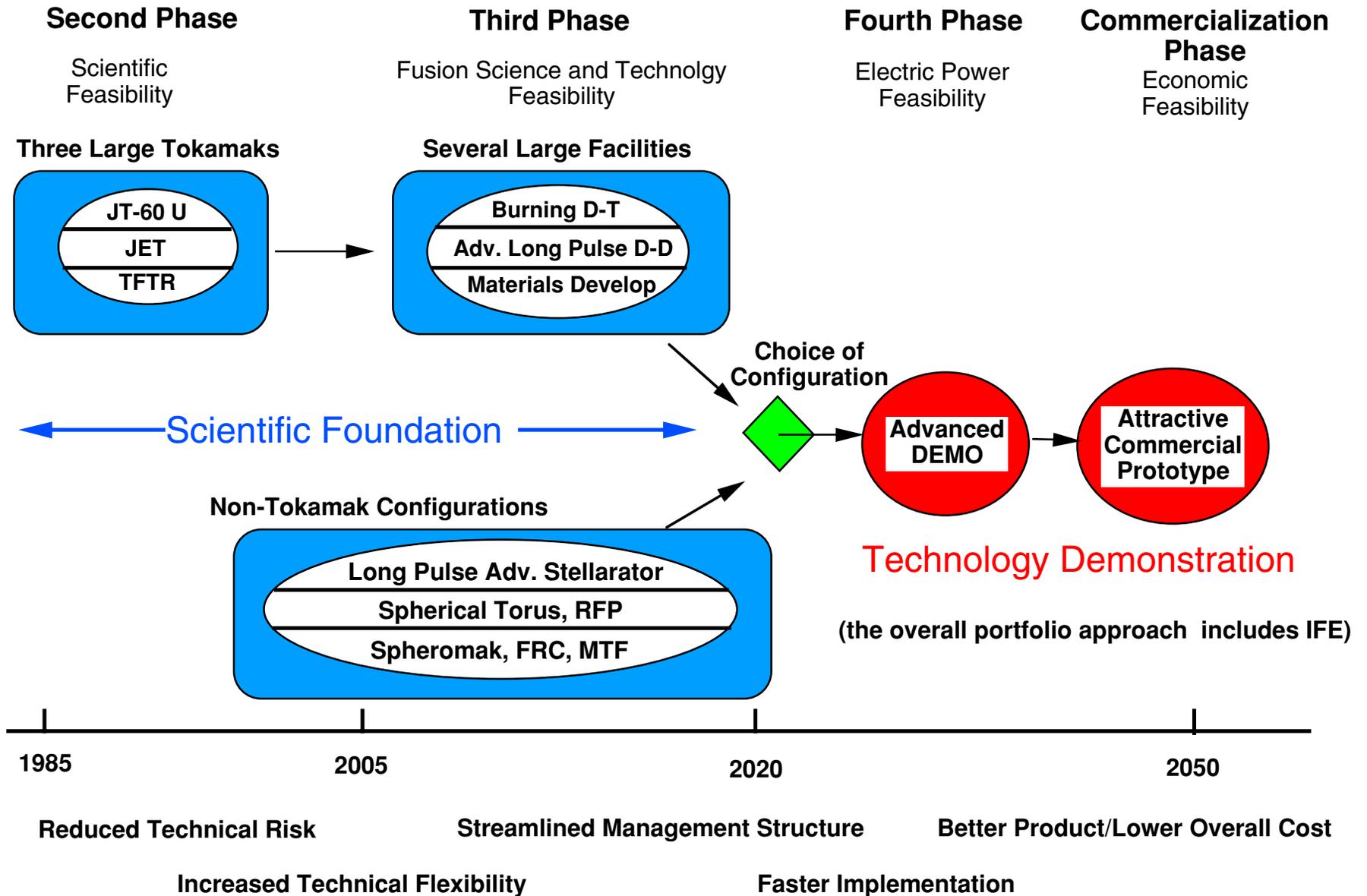
Critical Issues to be Addressed in the Next Stage of Fusion Research

- **Burning Plasma Physics**
 - strong nonlinear coupling inherent in a fusion dominated plasma
 - access, explore and understand fusion dominated plasmas
- **Advanced Toroidal Physics**
 - develop and test physics needed for an attractive MFE reactor
 - couple with burning plasma physics
- **Boundary Physics and Plasma Technology** (coupled with above)
 - high particle and heat flux
 - couple core and divertor
 - fusion plasma - tritium inventory and helium pumping
- **Neutron-Resistant Low-Activation Materials**
 - high fluence material testing facility using “point”neutron source

- high fluence component testing facility using volume neutron source -----

- Superconducting Coil Technology does not have to be coupled to physics experiments - only if needed for physics objectives

Diversified International Portfolio for Magnetic Fusion



Comparison of EU One Step to DEMO Power Plant(s) with ARIES

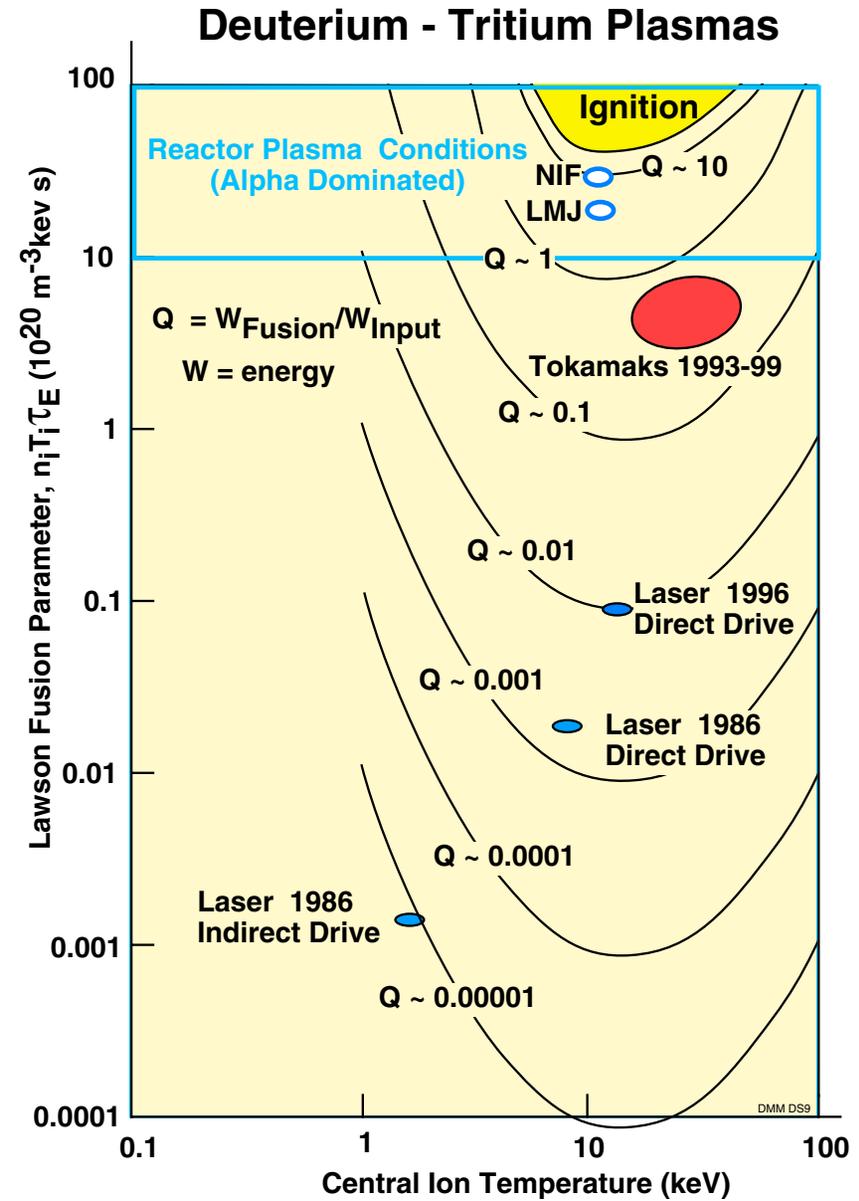
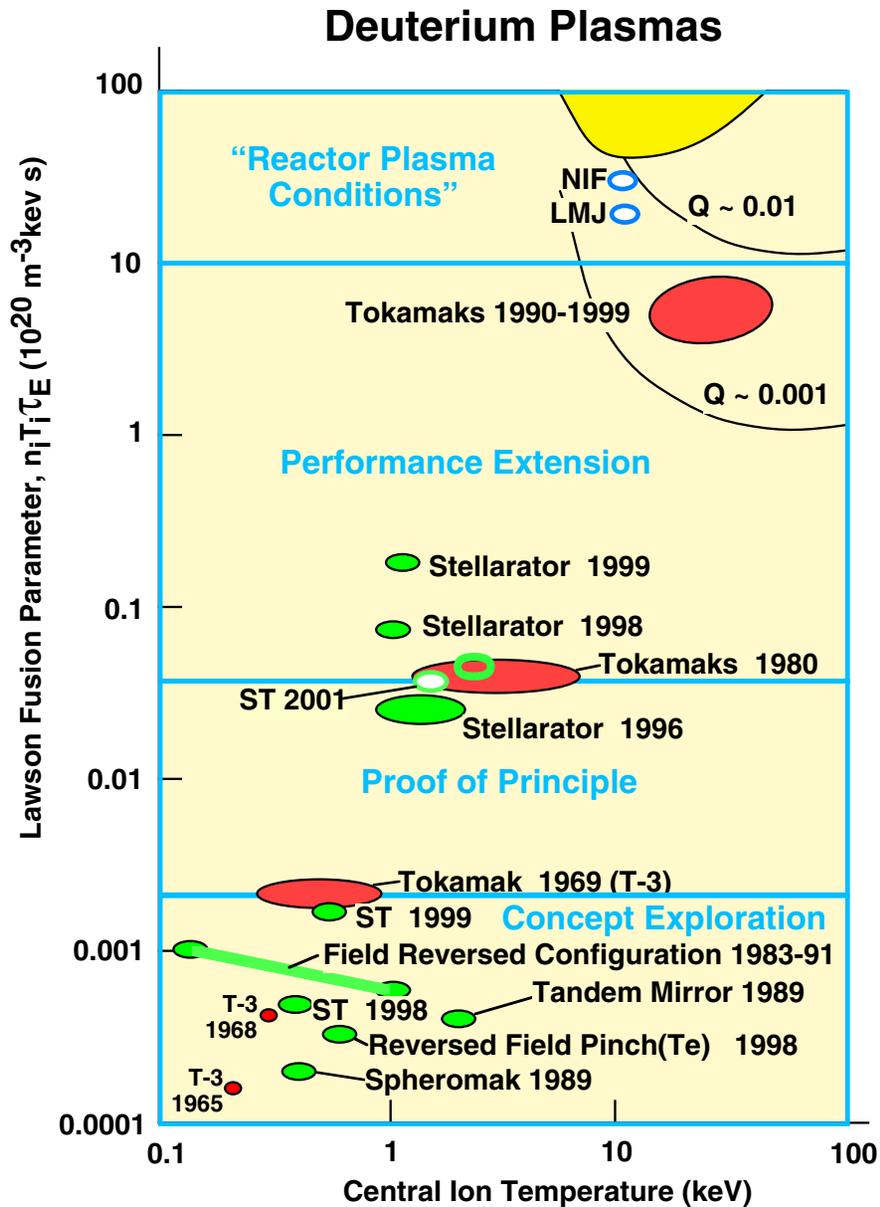
	A	B	C	D	ARIES-AT 1,000MW _e
R(m)	9.8	8.6	7.5	6.1	5.2
I(MA)	33.5	27.5	20.1	14.1	12.8
β_N	3.4	3.3	4	4.5	5.4
f _B (%)	36	36	69	76	92
H _H	1.2	1.2	1.3	1.2	1.4
q ₉₅	3	3	4.5	4.5	3.5
η_{th} (%)	31	42	44	59	59
Blanket energy gain	1.18	1.39	1.17	1.17	

EU DEMO Blanket
tested on ITER
One Step to DEMO

~ARIES-ST
Blanket

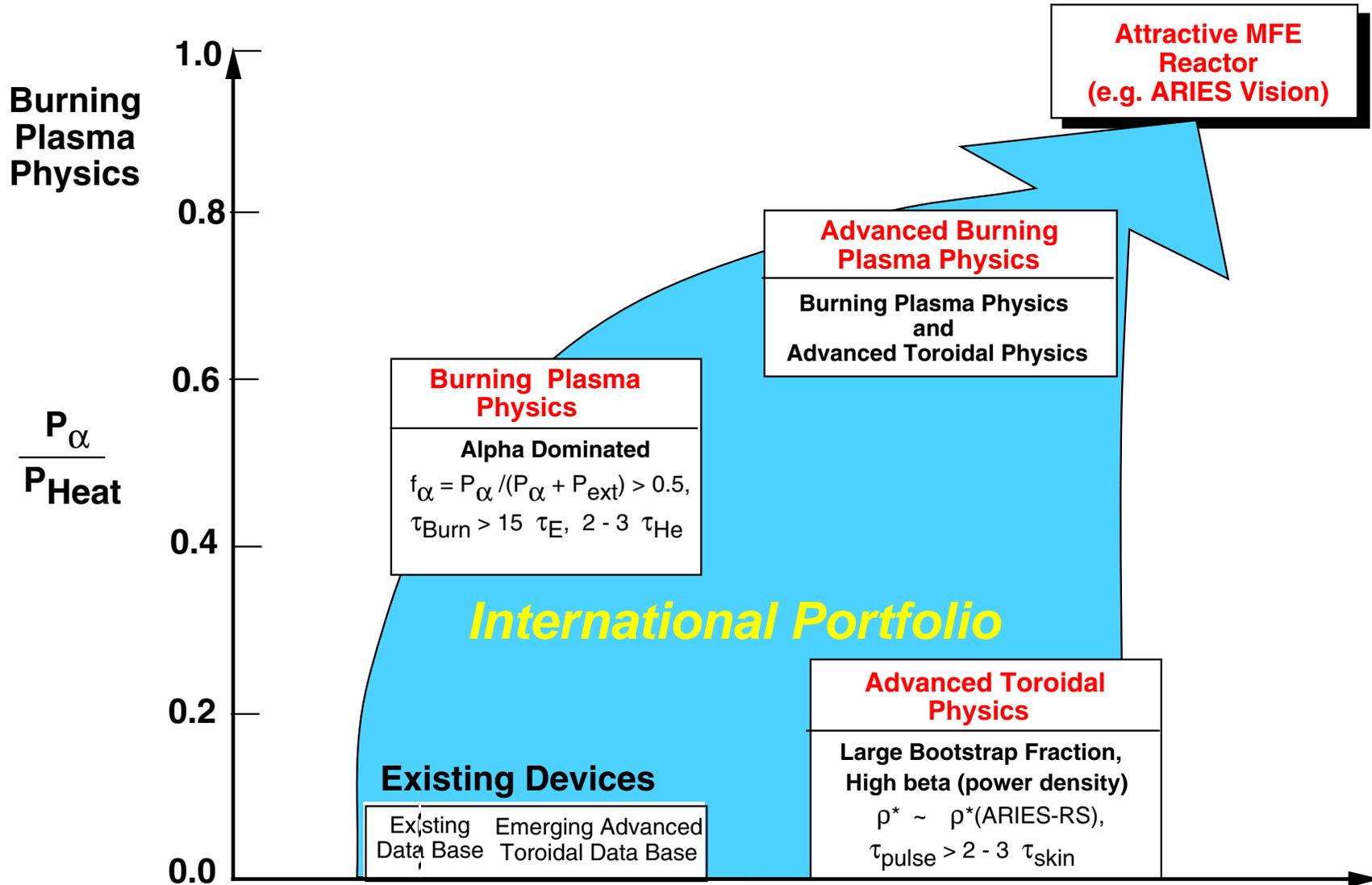
~ARIES-AT
Blanket

The Tokamak is Technically Ready for a High Gain Burning Exp't



The tokamak is sufficiently advanced to permit the design, construction and initiation of a next step burning plasma experiment within the next decade that could address the fusion plasma and self-heating issues for magnetic fusion.

FIRE's Goal is to Address the Critical Burning Plasma Science Issues for an Attractive MFE Reactor



Advanced Toroidal Physics (e.g., bootstrap fraction)

Attain a burning plasma with confidence using “today’s” physics, but allow the flexibility to explore tomorrow’s advanced physics.

Fusion Science Objectives for a Major Next Step Burning Plasma Experiment

Explore and understand the strong non-linear coupling that is fundamental to fusion-dominated plasma behavior (self-organization)

- Energy and particle transport (extend confinement predictability)
 - Macroscopic stability (β -limit, wall stabilization, NTMs)
 - Wave-particle interactions (fast alpha particle driven effects)
 - Plasma boundary (density limit, power and particle flow)
- Test/Develop techniques to control and optimize fusion-dominated plasmas.
 - Sustain fusion-dominated plasmas - high-power-density exhaust of plasma particles and energy, alpha ash exhaust, study effects of profile evolution due to alpha heating on macro stability, transport barriers and energetic particle modes.
 - Explore and understand various advanced operating modes and configurations in fusion-dominated plasmas to provide generic knowledge for fusion and non-fusion plasma science, and to provide a foundation for attractive fusion applications.

Advanced Burning Plasma Exp't Requirements

Burning Plasma Physics

$Q \geq 5$, ~ 10 as target, ignition not precluded

$f_\alpha = P_\alpha/P_{\text{heat}} \geq 50\%$, $\sim 66\%$ as target, up to 83% at $Q = 25$

TAE/EPM stable at nominal point, able to access unstable

Advanced Toroidal Physics

$f_{\text{bs}} = I_{\text{bs}}/I_p \geq 50\%$ up to 75%

$\beta_N \sim 2.5$, no wall ~ 3.6 , $n = 1$ wall stabilized

Quasi-stationary Burn Duration

Pressure profile evolution and burn control $> 10 \tau_E$

Alpha ash accumulation/pumping $> \text{several } \tau_{\text{He}}$

Plasma current profile evolution $1 \text{ to } 3 \tau_{\text{skin}}$

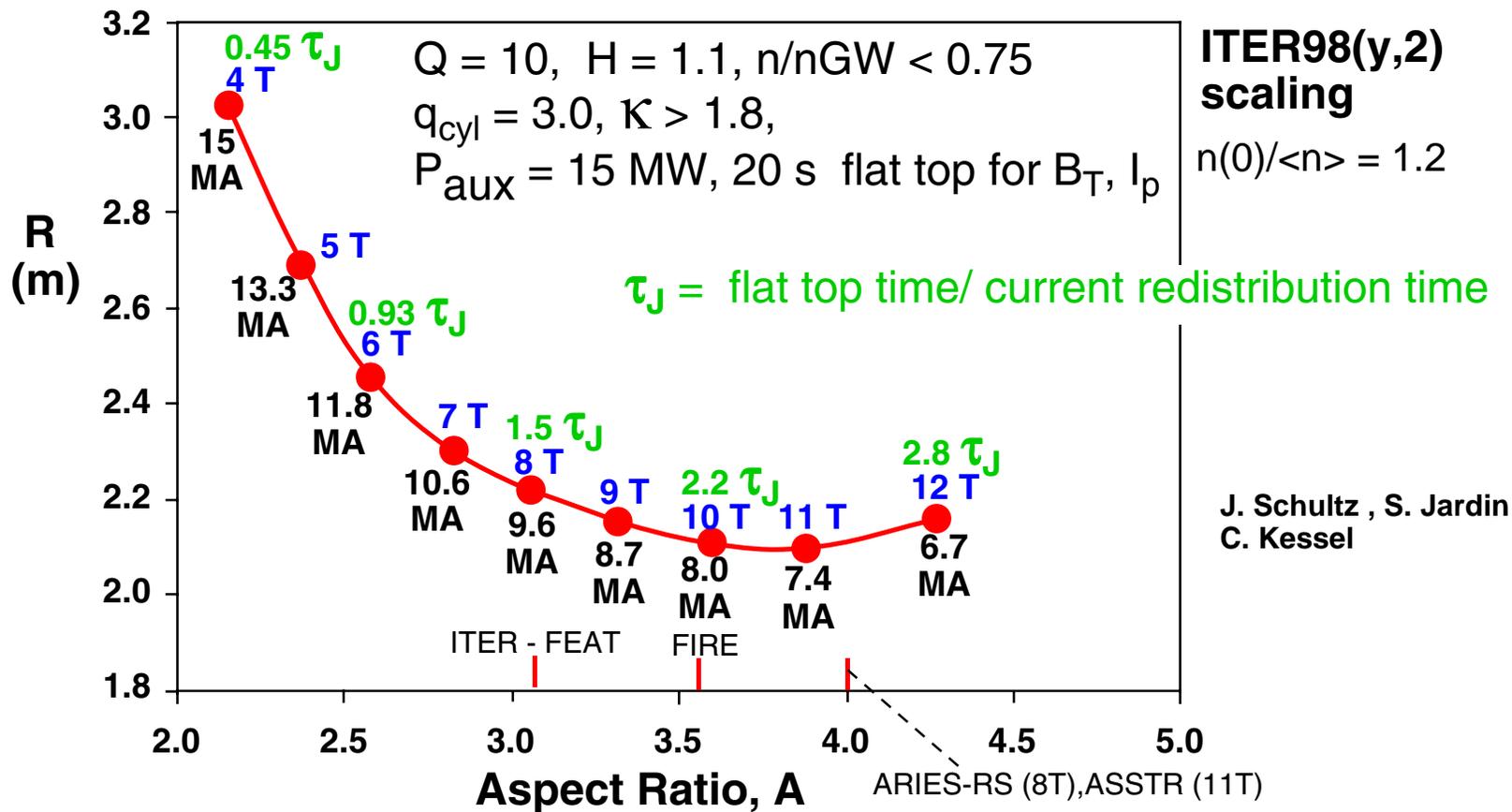
Divertor pumping and heat removal $\text{several } \tau_{\text{divertor}}$

FIRE has Adopted the Advanced Tokamak Features Identified by ARIES Studies

- High toroidal field
- Double null
- Strong shaping
 - $\kappa = 2.0, \delta = 0.7$
- Internal vertical position control coils
- Cu wall stabilizers for vertical and kink instabilities
- Very low ripple (0.3%)
- ICRF/FW on-axis CD
- LH off-axis CD
- LHCD stabilization of NTMs
- Tungsten divertor targets
- Feedback coil stabilization for Resistive Wall Modes (RWM)
- Burn times exceeding current diffusion times
- Pumped divertor/pellet fueling/impurity control to optimize plasma edge

Optimization of a Burning Plasma Experiment (H-Mode)

- Consider an inductively driven tokamak with copper alloy TF and PF coils precooled to LN temperature that warm up adiabatically during the pulse.
- Seek minimum R while varying A and space allocation for TF/PF coils for a specified plasma performance - Q and pulse length with physics and eng. limits.

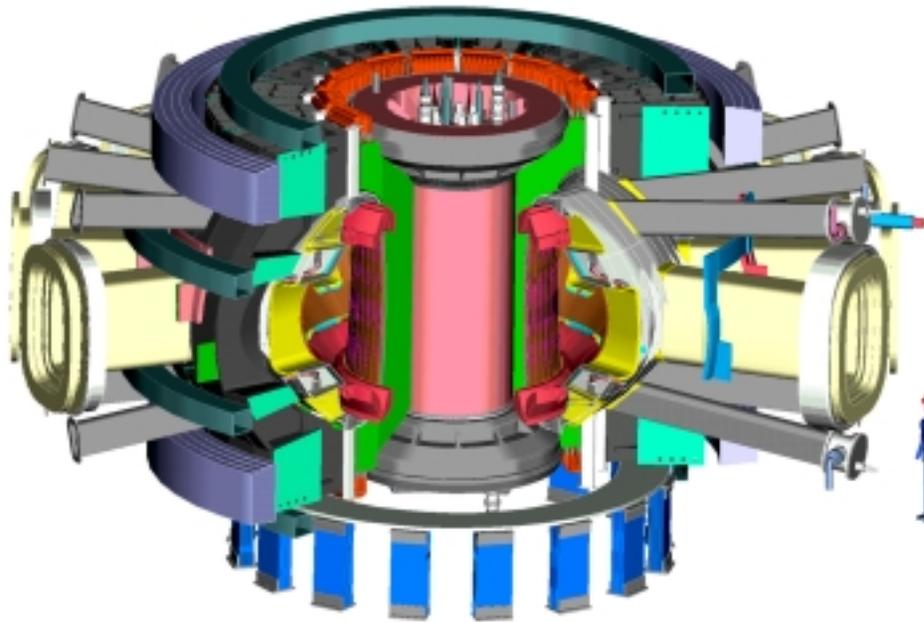


What is the optimum for advanced steady-state modes?

Fusion Ignition Research Experiment

(FIRE)

<http://fire.pppl.gov>



Design Features

- $R = 2.14 \text{ m}$, $a = 0.595 \text{ m}$
- $B = 10 \text{ T}$
- $W_{\text{mag}} = 5.2 \text{ GJ}$
- $I_p = 7.7 \text{ MA}$
- $P_{\text{aux}} \leq 20 \text{ MW}$
- $Q \approx 10$, $P_{\text{fusion}} \sim 150 \text{ MW}$
- Burn Time $\approx 20 \text{ s}$
- Tokamak Cost $\approx \$375\text{M}$ (FY99)
- Total Project Cost $\approx \$1.2\text{B}$ at Green Field site.

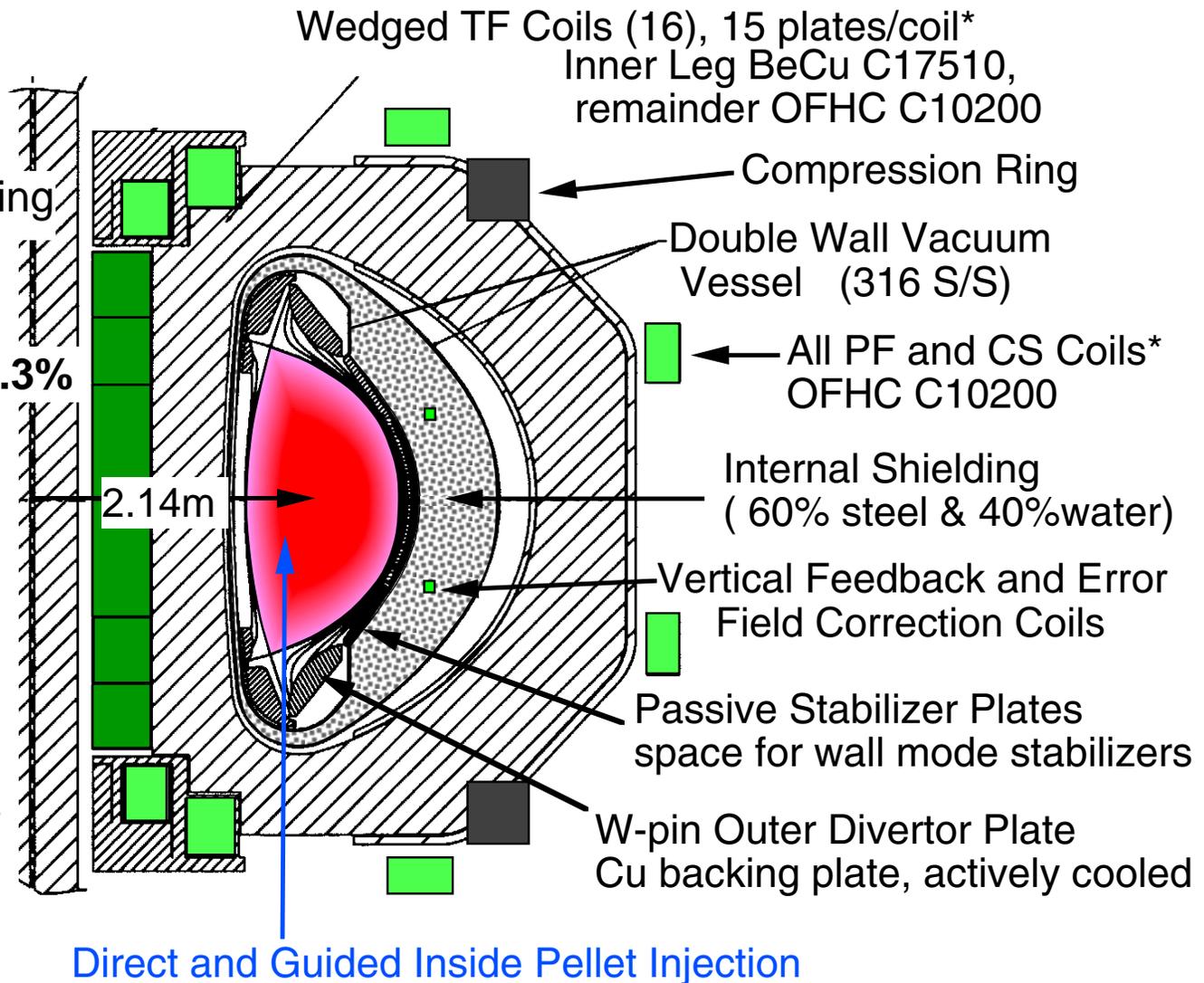
Mission: Attain, explore, understand and optimize magnetically-confined fusion-dominated plasmas.

CIT + TPX = FIRE leading to ARIES

High-Field Copper-Alloy Coils have Advantages for BP Expt's

AT Features

- DN divertor pumping
- strong shaping
- very low ripple < 0.3%
- internal coils
- space for wall stabilizers
- inside pellet injection
- large access ports



*Coil systems cooled to 77 °K prior to pulse, rising to 373 °K by end of pulse.

Basic Parameters and Features of FIRE

R, major radius	2.14 m
a, minor radius	0.595 m
κ_X, κ_{95}	2.0, 1.77
δ_X, δ_{95}	0.7, 0.55(AT) - 0.4(OH)
q ₉₅ , safety factor at 95% flux surface	>3
B _t , toroidal magnetic field	10 T with 16 coils, 0.3% ripple @ Outer MP
Toroidal magnet energy	5.8 GJ
I _p , plasma current	7.7 MA
Magnetic field flat top, burn time	28 s at 10 T in dd, 20s @ P _{dt} ~ 150 MW)
Pulse repetition time	~3hr @ full field and full pulse length
ICRF heating power, maximum	20 MW, 100MHz for 2Ω _T , 4 mid-plane ports
Neutral beam heating	Upgrade for edge rotation, CD - 120 keV PNBI?
Lower Hybrid Current Drive	Upgrade for AT-CD phase, ~20 MW, 5.6 GHz
Plasma fueling	Pellet injection (≥2.5km/s vertical launch inside mag axis, guided slower speed pellets)
First wall materials	Be tiles, no carbon
First wall cooling	Conduction cooled to water cooled Cu plates
Divertor configuration	Double null, fixed X point, detached mode
Divertor plate	W rods on Cu backing plate (ITER R&D)
Divertor plate cooling	Inner plate-conduction, outer plate/baffle- water
Fusion Power/ Fusion Power Density	150 - 200 MW, ~6 -8 MW m ⁻³ in plasma
Neutron wall loading	~ 2.3 MW m ⁻²
Lifetime Fusion Production	5 TJ (BPX had 6.5 TJ)
Total pulses at full field/power	3,000 (same as BPX), 30,000 at 2/3 B _t and I _p
Tritium site inventory	Goal < 30 g, Category 3, Low Hazard Nuclear Facility

Guidelines for Estimating Plasma Performance

Confinement (Elmy H-mode) - ITER98(y,2) based on today's data base

$$\tau_E = 0.144 I^{0.93} R^{1.39} a^{0.58} n_{20}^{0.41} B^{0.15} A_i^{0.19} \kappa^{0.78} P_{\text{heat}}^{-0.69} H(y,2)$$

Density Limit - Based on today's tokamak data base

$$n_{20} \leq 0.8 n_{\text{GW}} = 0.8 I_p / \pi a^2,$$

Beta Limit - theory and tokamak data base

$$\beta \leq \beta_N(I_p/aB), \quad \beta_N < 2.5 \text{ conventional}, \quad \beta_N \sim 4 \text{ advanced}$$

H-Mode Power Threshold - Based on today's tokamak data base

$$P_{\text{th}} \geq (2.84/A_i) n_{20}^{0.58} B^{0.82} Ra^{0.81}, \quad \text{same as ITER-FEAT}$$

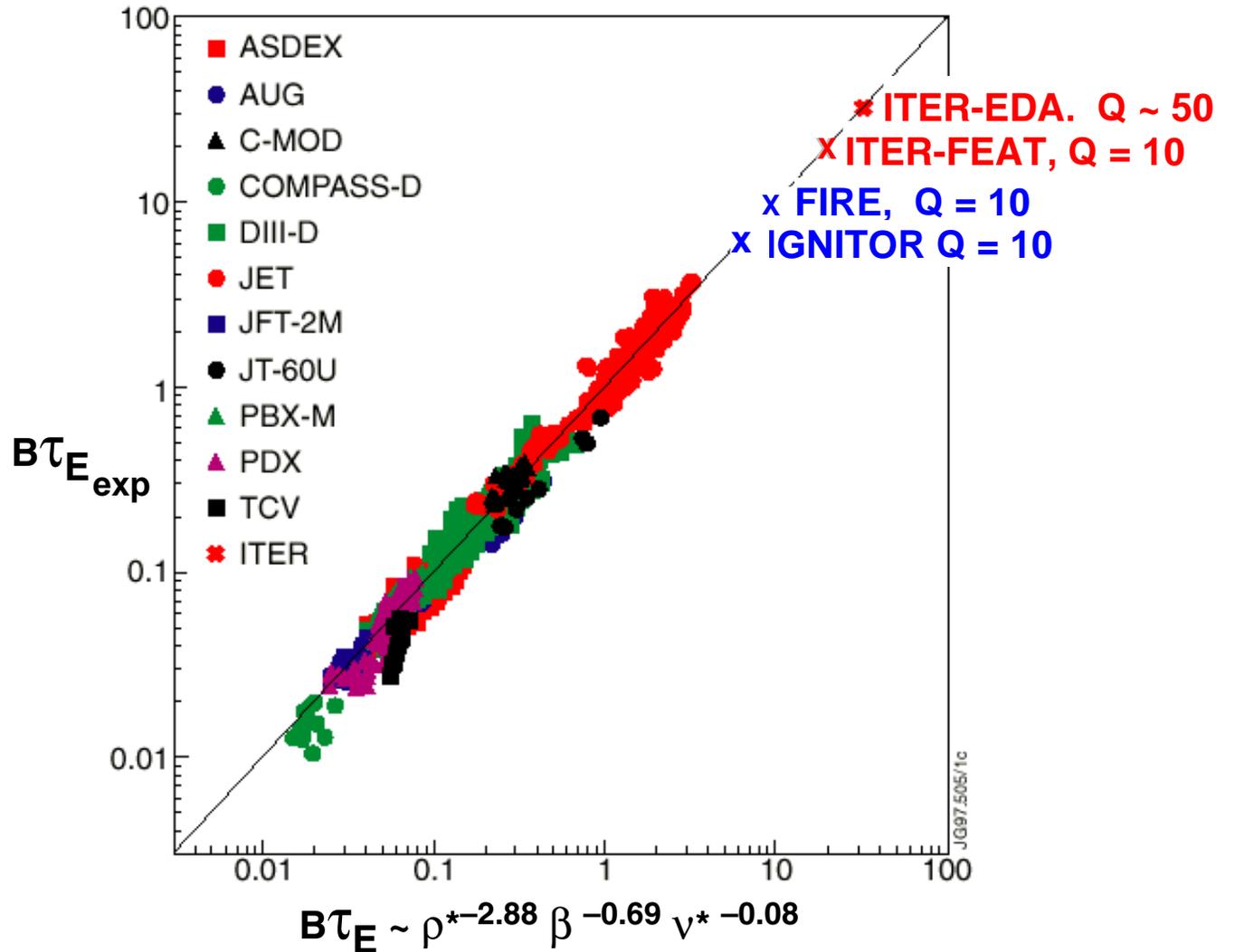
Helium Ash Confinement $\tau_{\text{He}} = 5 \tau_E$, impurities = 3% Be, 0% W

Understanding is mainly empirical. Better understanding is needed from existing experiments with improved simulations, and a benchmark in alpha-dominated fusion plasmas is needed to confirm and extend the science basis.

FIRE is a Modest Extrapolation in Plasma Confinement

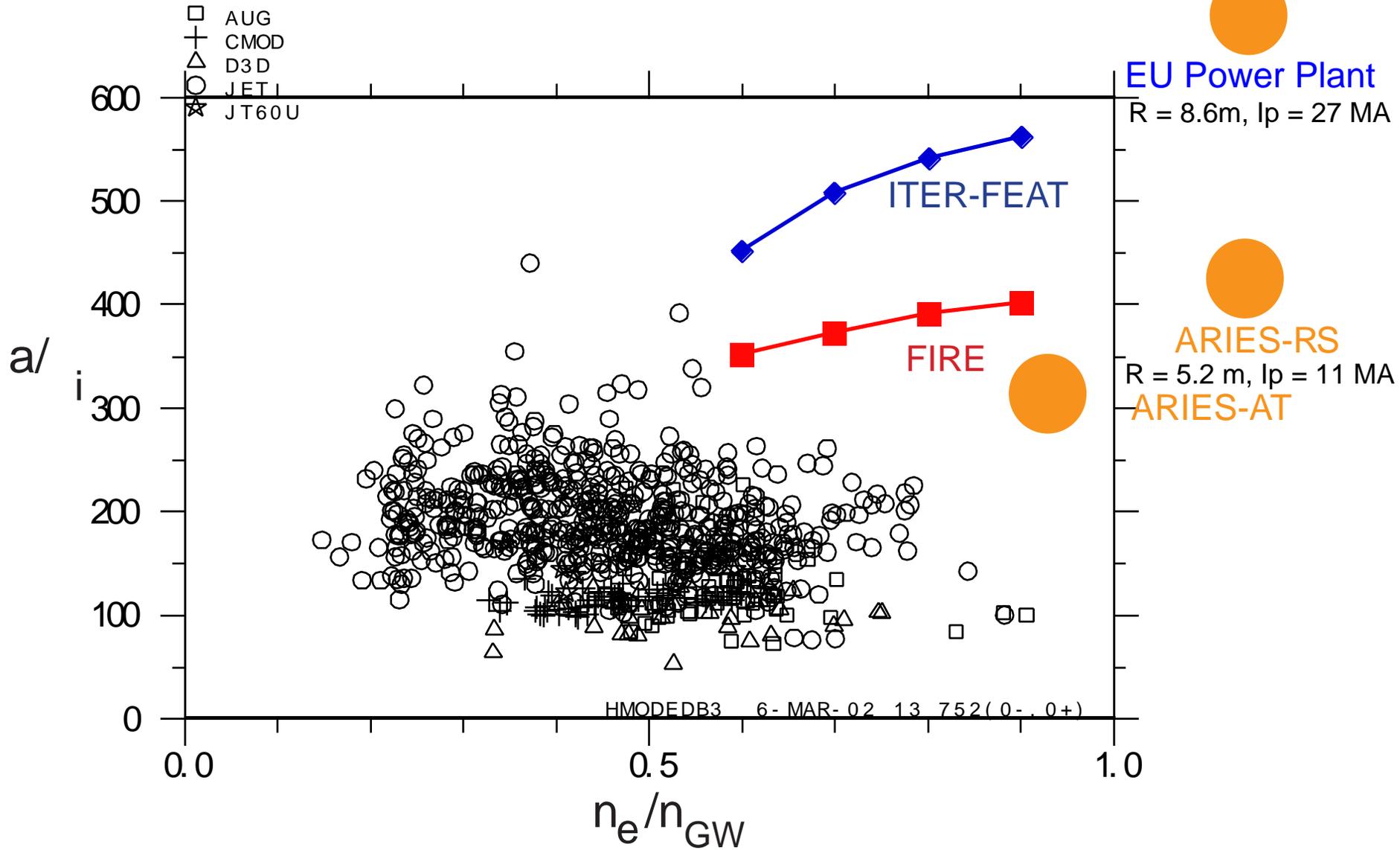
Dimensionless Parameters
$\omega_c \tau = B \tau$
$\rho^* = \rho/a$
$v^* = v_c/v_b$
β

Similarity Parameter
$B R^{5/4}$



Kadomtsev, 1975

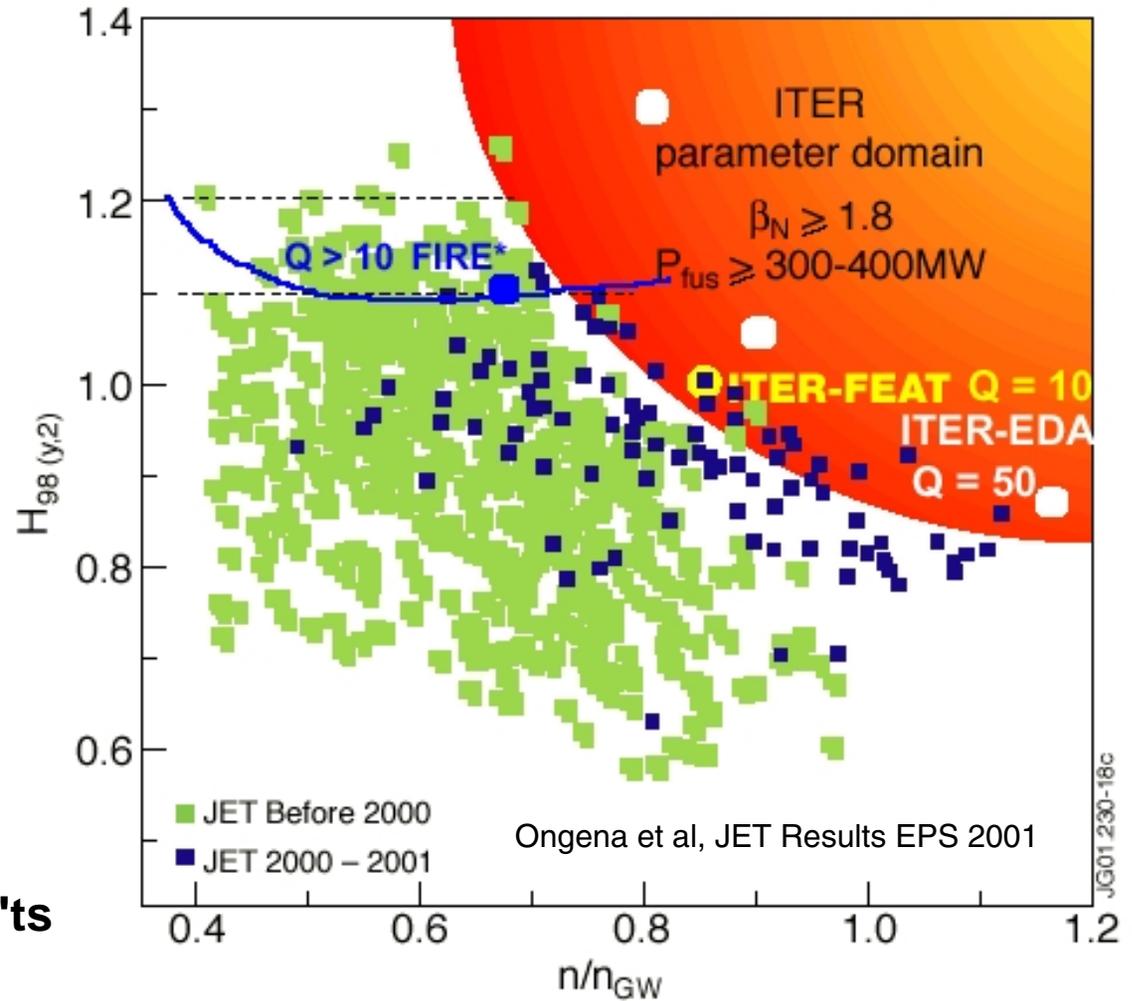
FIRE would Extend the Transport Understanding Toward ARIES



**FIRE and ITER-FEAT calculated for $Q = 10$,
 a/ρ_i evaluated at plasma $\sim 0.5a$**

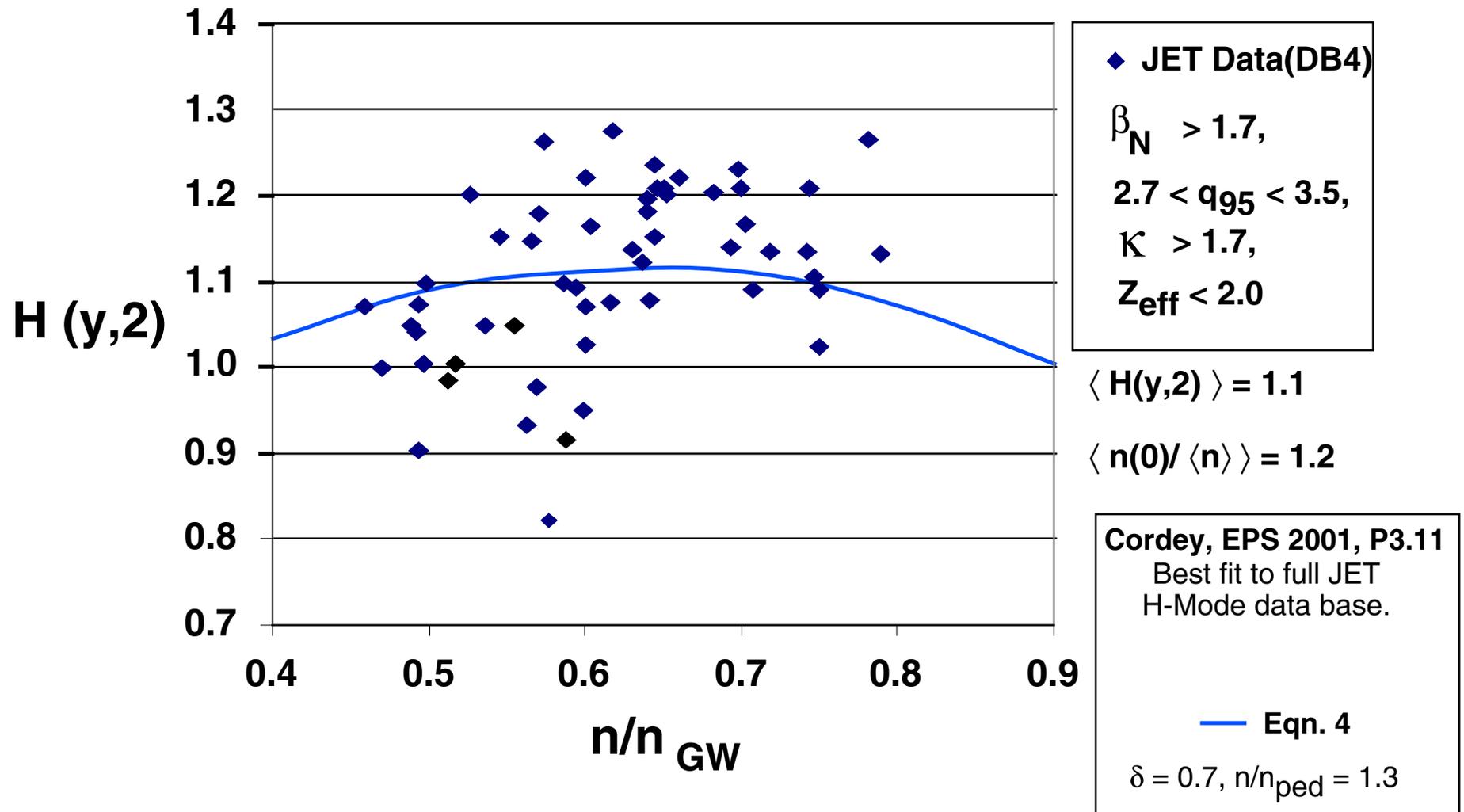
FIRE's Operating Density and Triangularity are Near the Optimum for the Elmy H-Mode

- The optimum density for the H-Mode is $n/n_{GW} \approx 0.6 - 0.7$
- H-mode confinement increases with δ
 - $\delta \approx 0.7$ FIRE
 - $\delta \approx 0.5$ ITER-FEAT
- Elm size is reduced for $\delta > 0.5$
- Z_{eff} decreases with density (Mathews/ITER scaling)
- DN versus SN ? C- Mod Exp'ts



Cordey et al, H = function (δ , n/n_{GW} , $n(0)/\langle n \rangle$) EPS 2001

JET H-Mode Data Selected for FIRE-like Parameters



Increasing Triangularity Enhances H-Mode Confinement

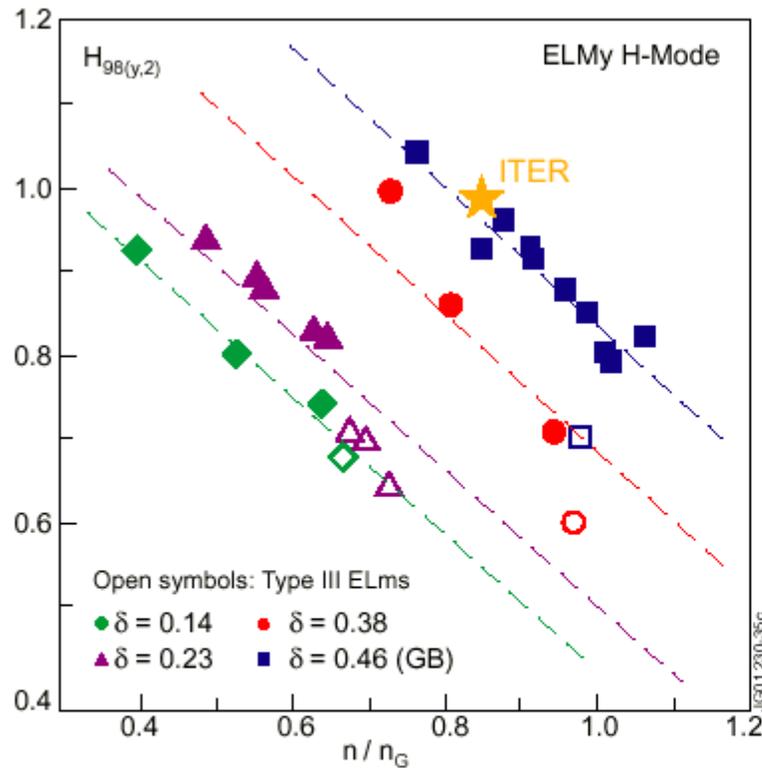


Figure 2.2-2 Confinement Enhancement Factor Relative to the ITERH-98(y,2) Scaling as a Function of n/n_G in JET⁵

- Trade-off between triangularity and heating power: lower δ discharges need higher P_{in}/P_{L-H}

Note: triangularity is determined at the separatrix

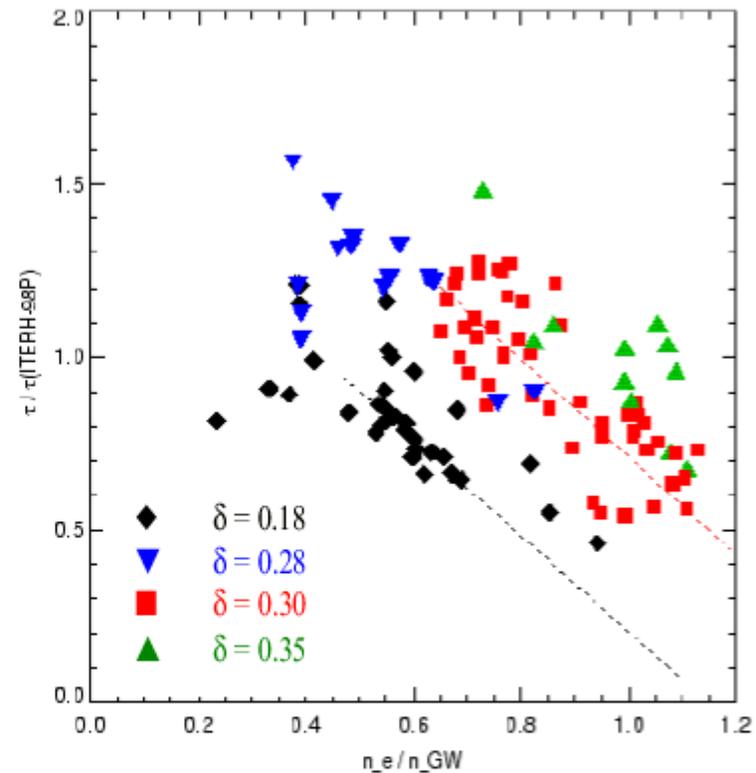
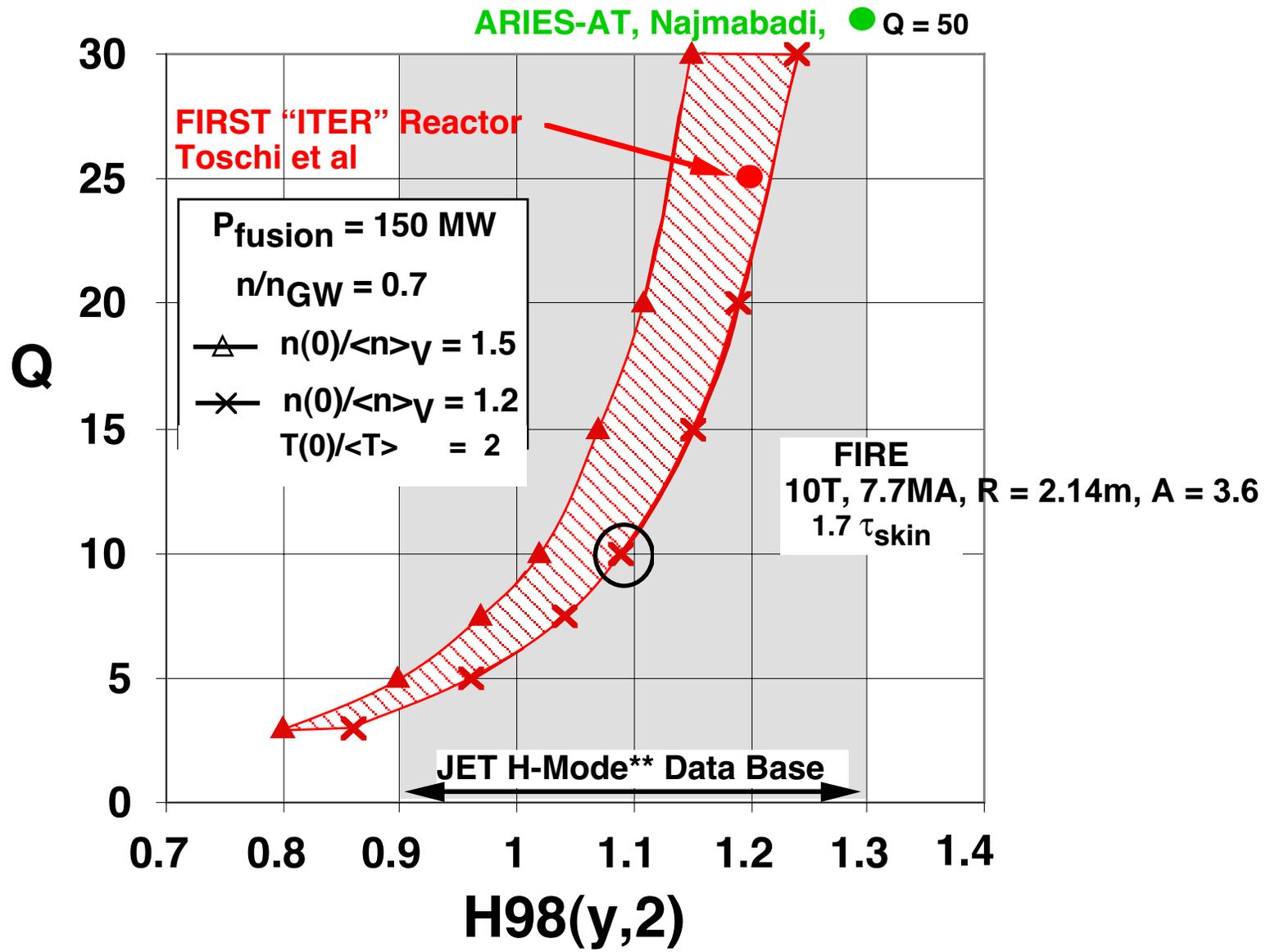
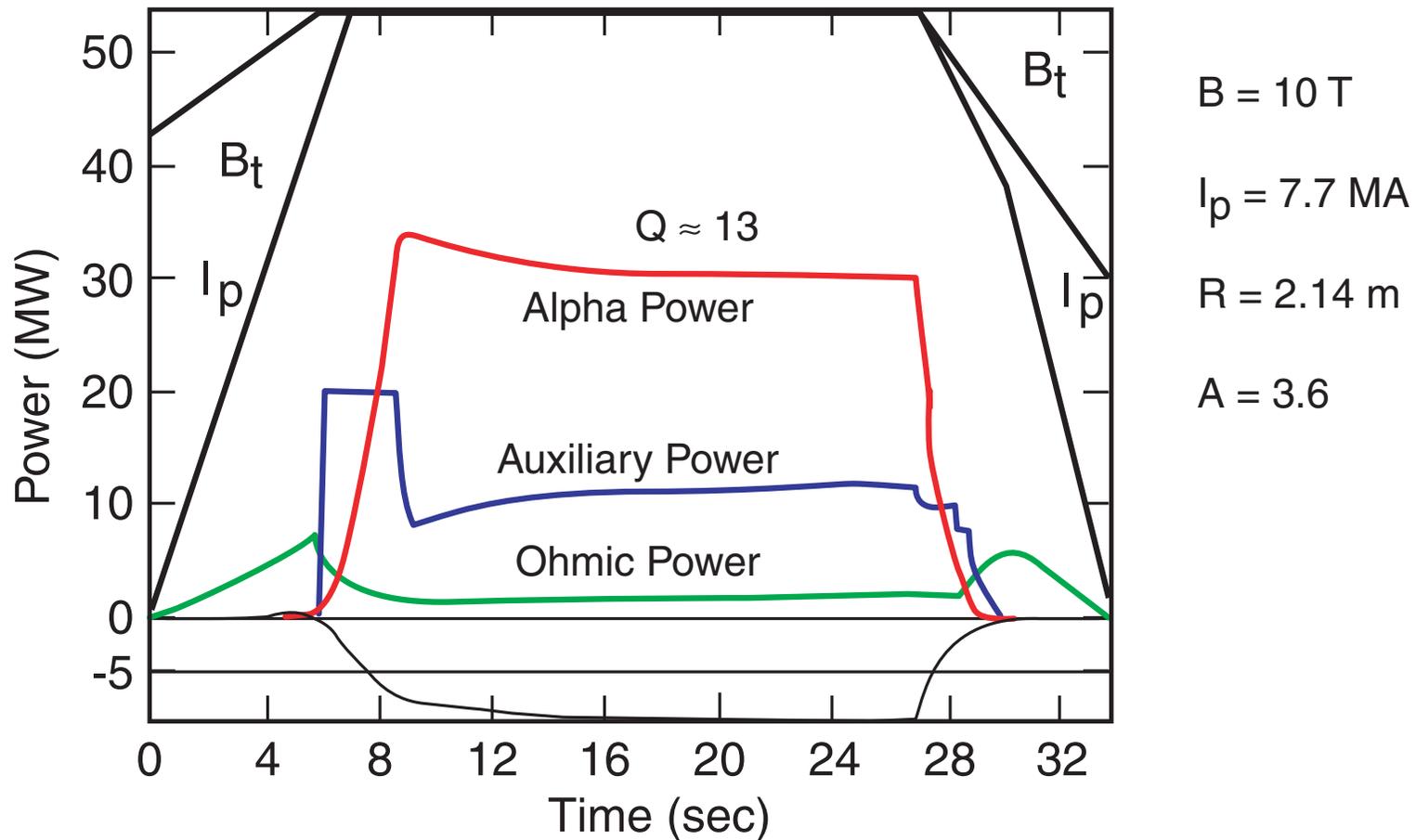


Figure 2.2-3 Confinement Enhancement Factor Relative to the ITERH-98P(y) Scaling as a Function of n/n_G in ASDEX Upgrade⁶

Projections to FIRE Compared to Envisioned Reactors



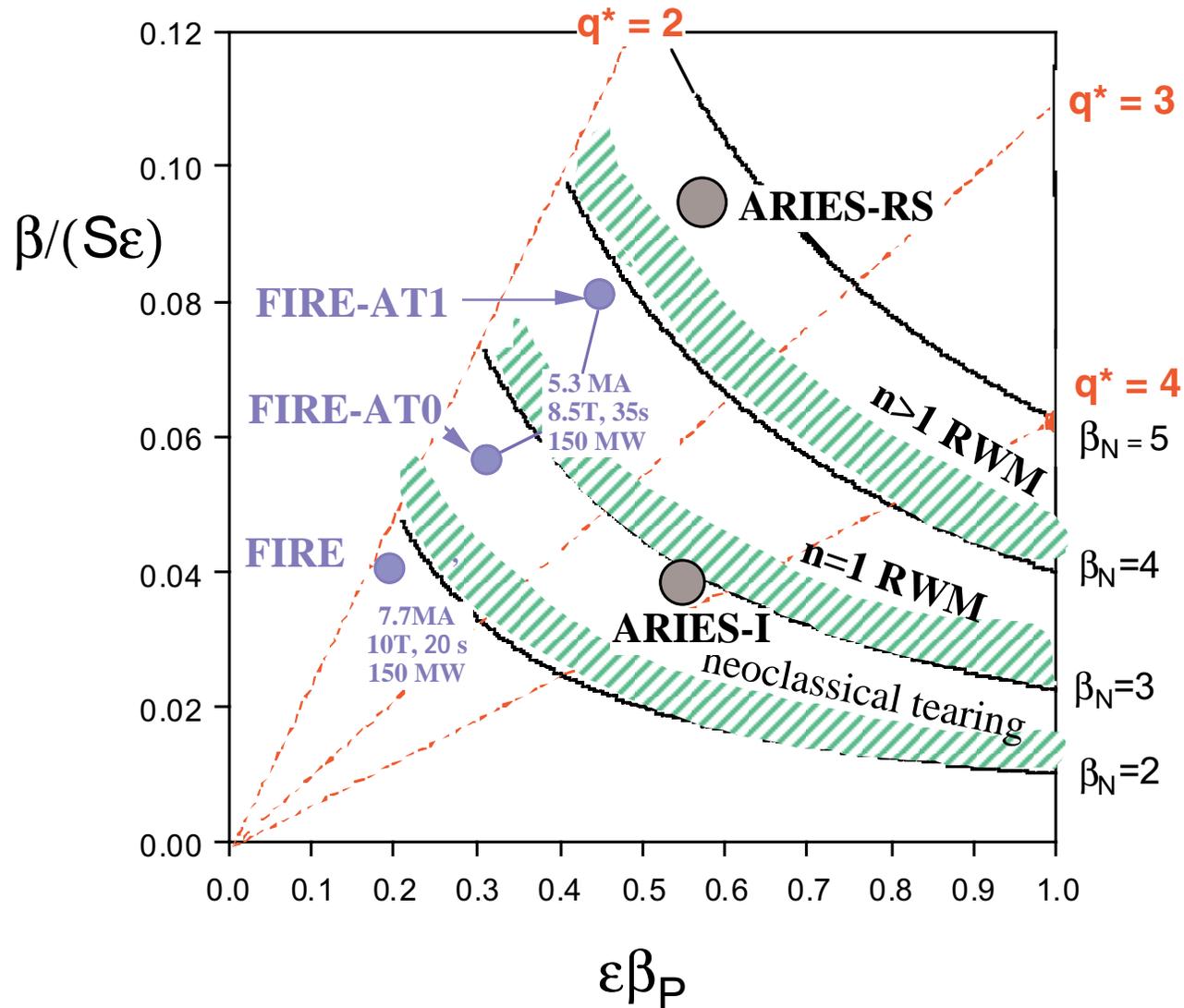
Simulation of Burning Plasma in FIRE



- ITER98(y, 2) with $H(y, 2) = 1.1$, $n(0)/\langle n \rangle = 1.2$, and $n/n_{GW} = 0.67$
- Burn Time $\approx 20 \text{ s} \approx 21\tau_E \approx 4\tau_{He} \approx 2\tau_{CR}$

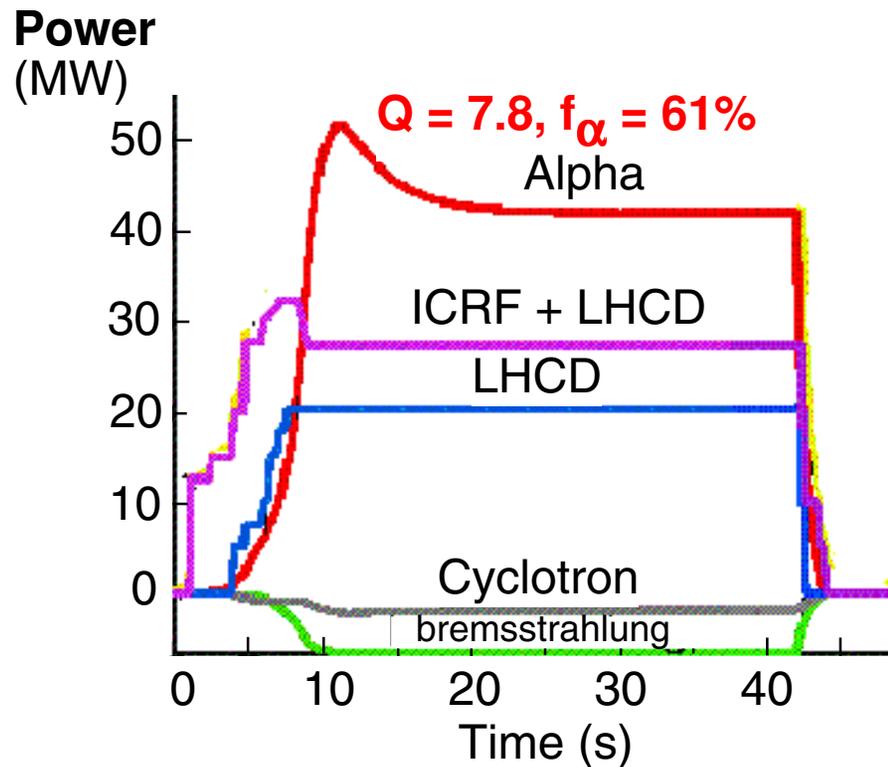
$$Q = P_{\text{fusion}} / (P_{\text{aux}} + P_{\text{oh}})$$

FIRE would Test a Sequence of AT Modes



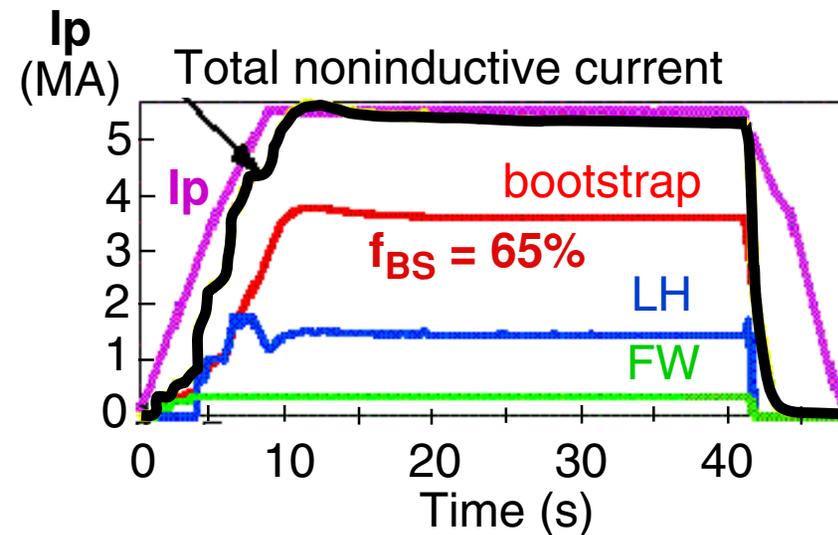
Advanced Burning Plasma Physics could be Explored in FIRE

Self-Heating Dominant

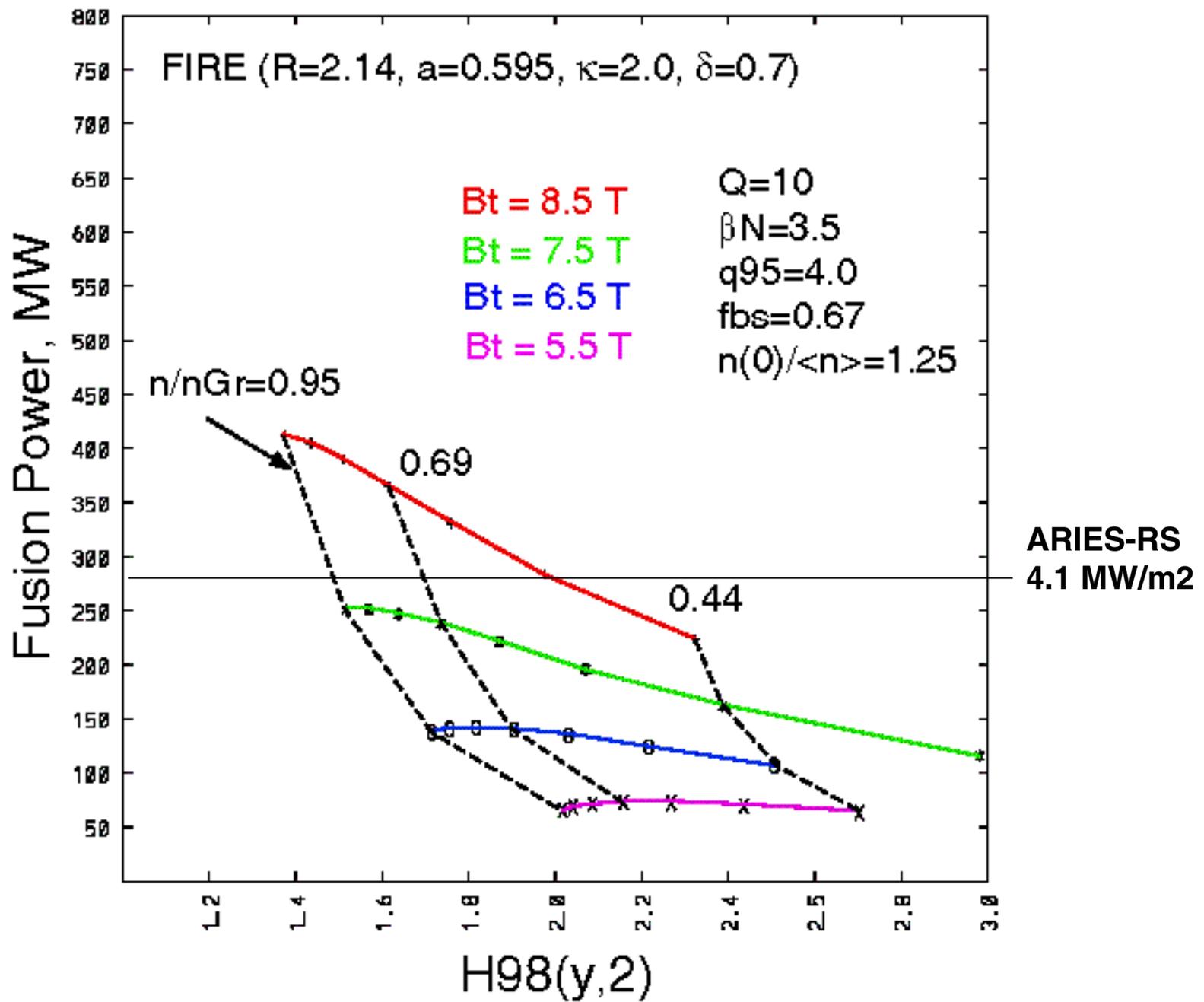


Self-Current Drive Dominant

Fully Non-Inductive for $> 1 \tau_{CR}$



Tokamak simulation code results for $H(y, 2) = 1.6$, $\beta_N = 3.5$, would require RW mode stabilization. $q(0) = 2.9$, $q_{min} = 2.2$ @ $r/a = 0.8$, 8.5 T, 5.5 MA



Edge Physics and PFC Technology: Critical Issue for Fusion

Plasma Power and particle Handling under relevant conditions
Normal Operation / Off Normal events

Tritium Inventory Control
must maintain low T inventory in the vessel \Rightarrow all metal PFCs

Efficient particle Fueling
pellet injection needed for deep and tritium efficient fueling

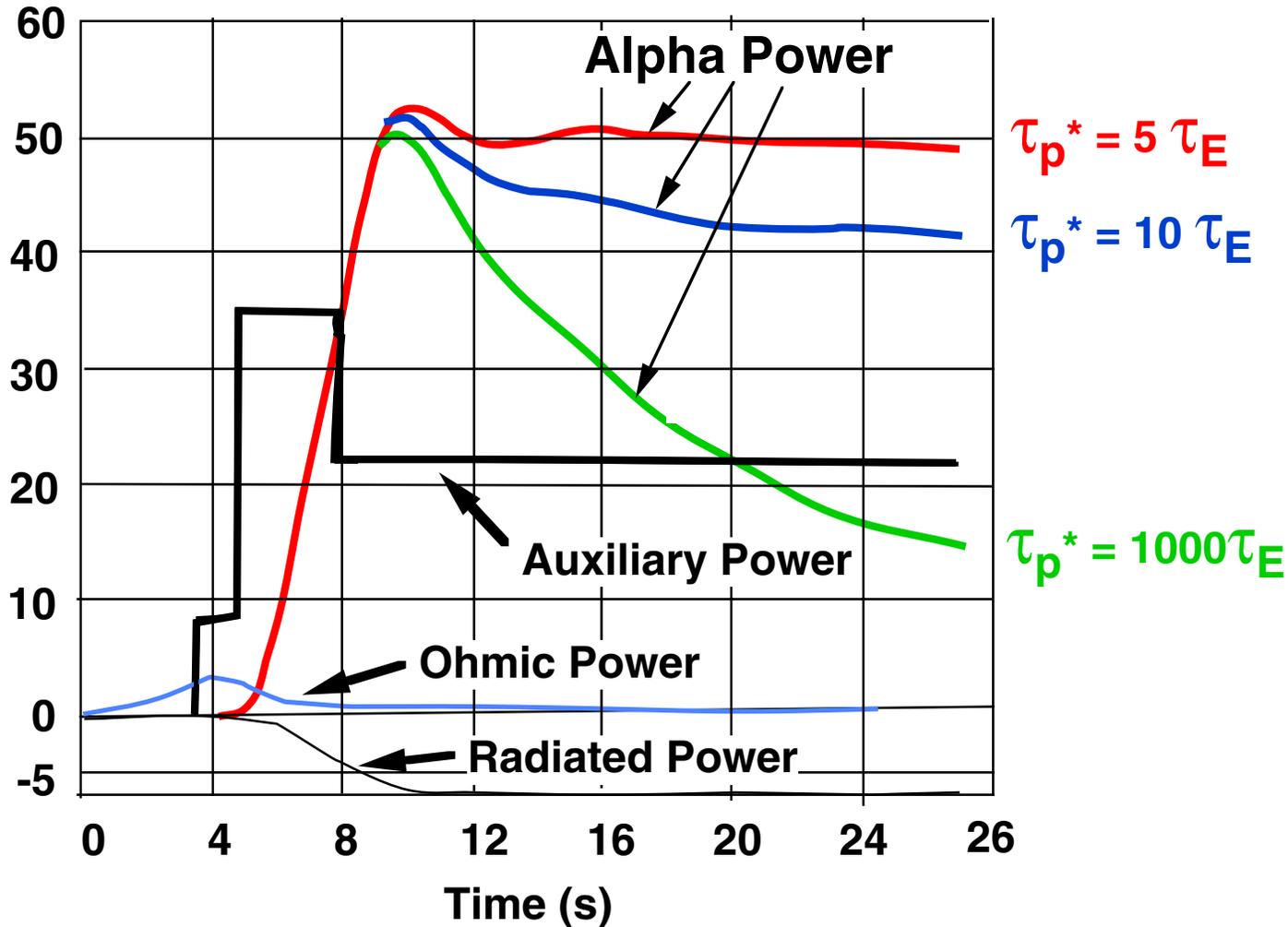
Helium Ash Removal
need close coupled He pumping

Non-linear Coupling with Core plasma Performance
nearly every advancement in confinement can be traced to the edge
Edge Pedestal models first introduced in \sim 1992 first step in understanding
Core plasma (low n_{edge}) and divertor (high n_{edge}) requirements conflict

Solutions to these issues would be a major output from a next step experiment.

Helium Ash Removal Techniques Required for a Reactor can be Studied on FIRE

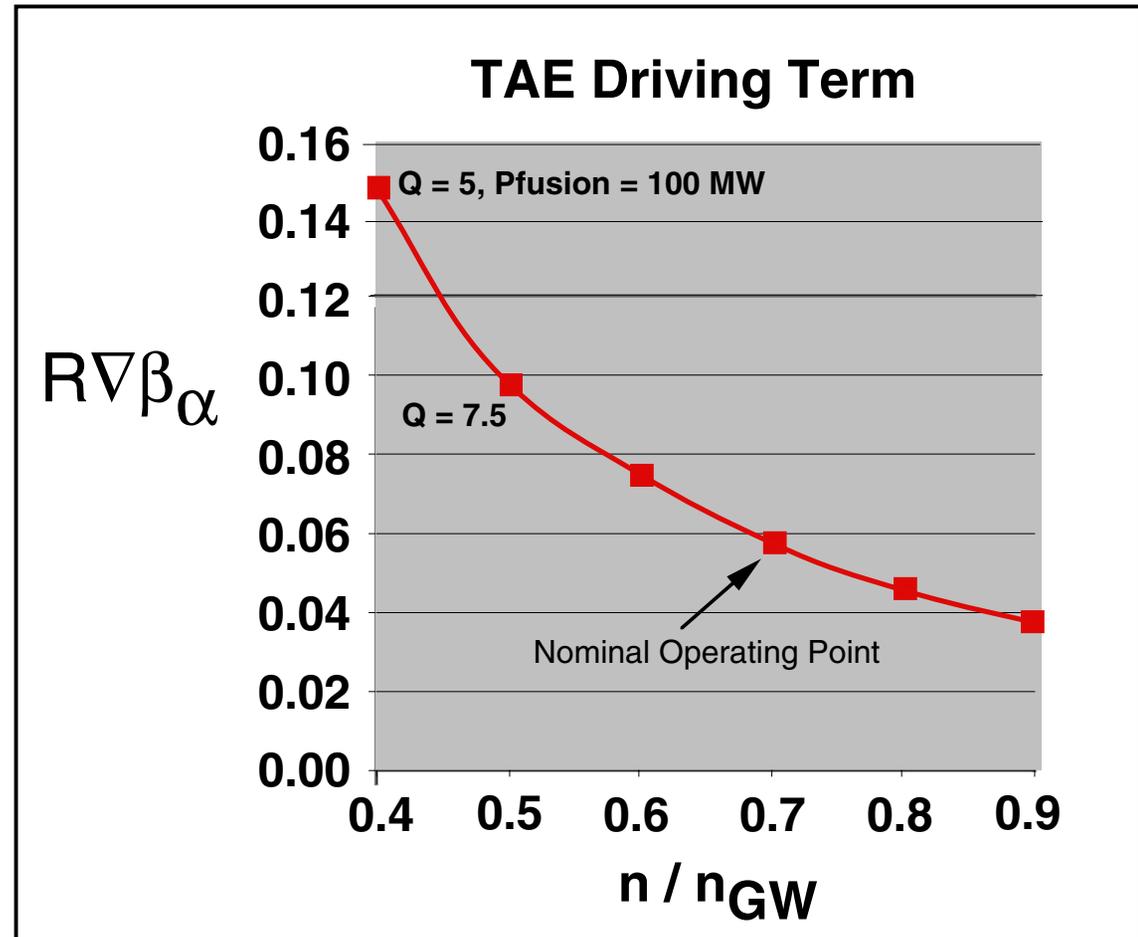
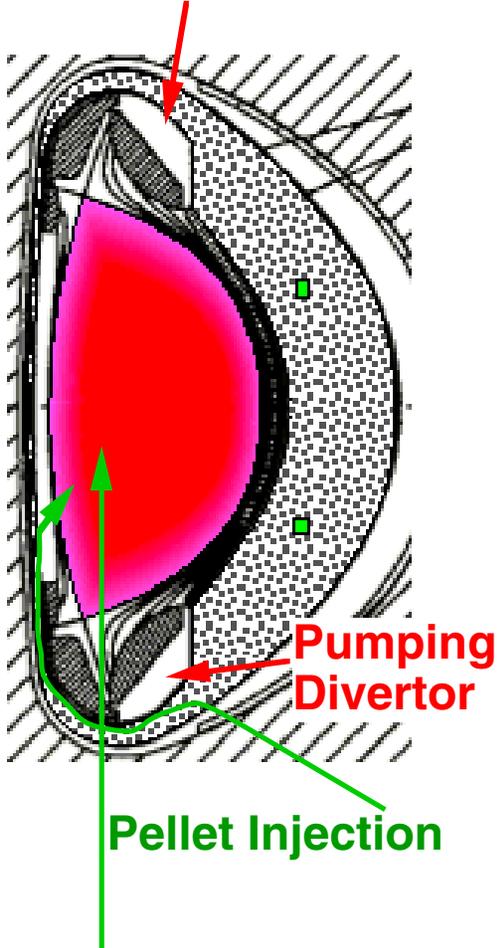
Power, MW



Fusion power can not be sustained without helium ash punping.

Energetic Particle Drive can be Varied in FIRE Using Divertor Pumping and Pellet Injection

Pumping Divertor



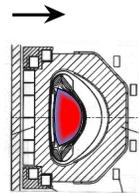
FIRE: $H(y,2) = 1.1, \alpha_n = 0.2, \alpha_T = 1.75,$
 $Q = 10, P_{fusion} = 150 \text{ MW}$ except where noted

FIRE would Test the High Power Density In-Vessel Technologies Needed for ARIES-RS

P_{fusion}
= ~ 150 MW

Volume
= 27 m³

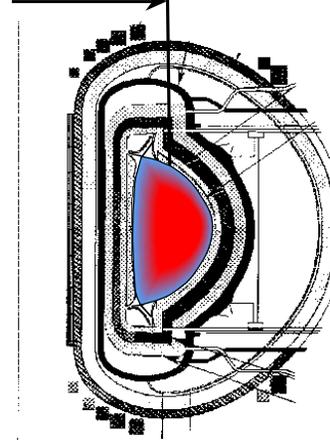
B = 10 T
R = 2.14 m



FIRE

~ 3X

B = 8 T
R = 5.5 m



ARIES-RS The "Goal"

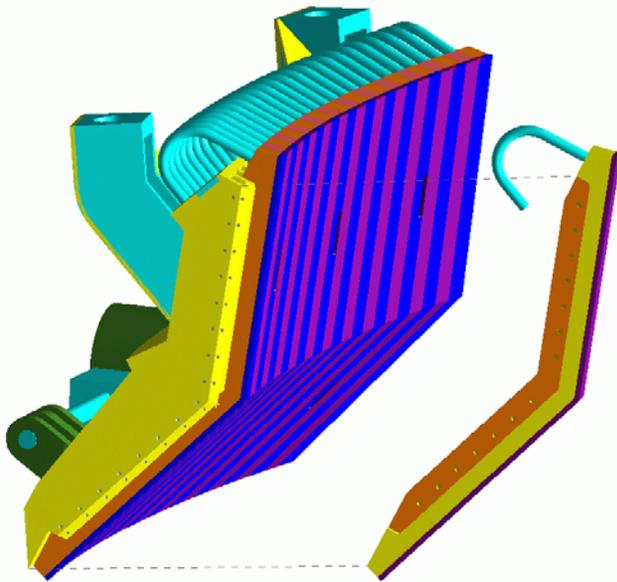
P_{fusion}
= 2170 MW

Volume
= 350 m³

	JET	FIRE	ARIES-RS
Fusion Power Density (MW/m³)	0.2	5.5	6
Neutron Wall Loading (MW/m²)	0.2	2.3	4
Divertor Challenge (P_{heat}/NR)	~5	~10	~35
Power Density on Div Plate (MW/m²)	3	~15-19 → 6	~5
Burn Duration (s)	4	20	steady

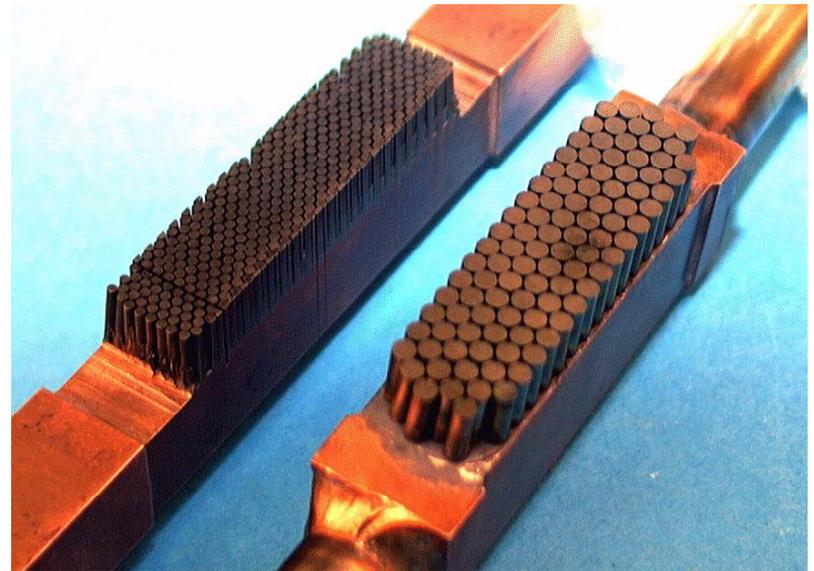
Divertor Module Components for FIRE

Sandia



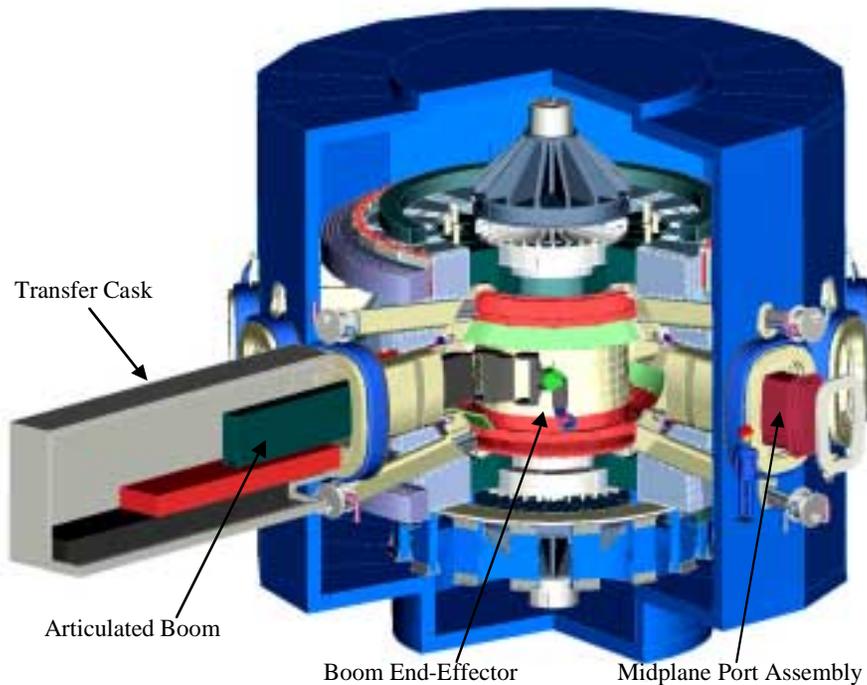
**Finger Plate for
Outer Divertor Module**

**Two W Brush Armor Configurations
Tested at 25 MW/m²**



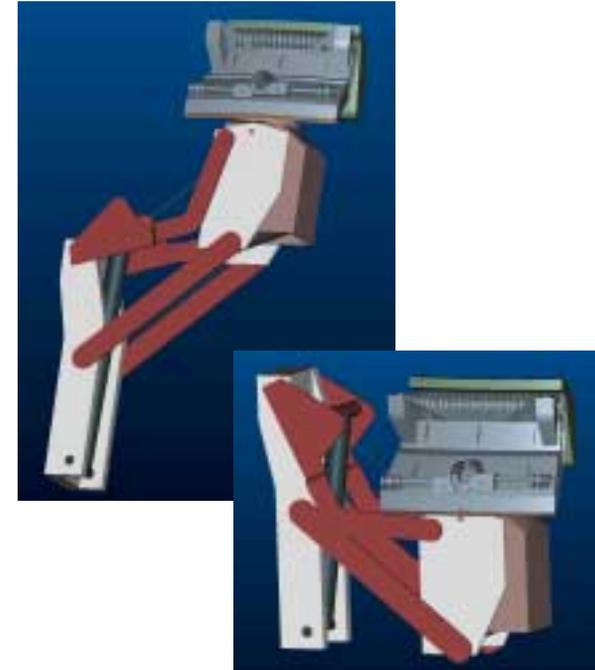
Carbon targets used in most experiments today are not compatible with tritium inventory requirements of fusion reactors.

FIRE In-Vessel Remote Handling System



In-vessel transporter

- Articulated boom deployed from sealed cask
- Complete in-vessel coverage from 4 midplane ports
- Fitted with different end-effector depending on component to be handled
- First wall module end-effector shown



Divertor end-effector

- High capacity (module wt. ~ 800 kg)
- Four positioning degrees of freedom
- Positioning accuracy of millimeters required

Cost Background for FIRE

- Three tokamaks physically larger but with lower field energy than FIRE have been built.

Water Cooled Coils	B(T)	R(m)	Coil Energy (GJ)	Const. Cost
TFTR (1983), US	5.2	2.5	1.5	\$498M
JET (1984), Europe	3.4	2.96	1.4	~\$600M
JT-60 (1984), Japan	4.4	3.2	2.9	~\$1000M
FIRE*, US	10	2.14	5.0	(~ \$1000M)

* FIRE would have liquid nitrogen cooled coils.

Cost estimates from previous design studies with similar technology.

Liquid N, Cu coils	B(T)	R(m)	Coil Energy (GJ)	Const. Cost
CIT (1989),	11	2.14	5	\$680M (FY-89)
BPX (1991)	9.1	2.59	8.4	\$1,500M (FY-92)
BPX-AT(1992)	10	2.0	4.2	\$642M (FY-92)
FIRE Goal	10	2.14	5.0	(~\$1,000M FY-99)
PCAST (120s)	7	5.0	40	~\$5,815M (FY-95)

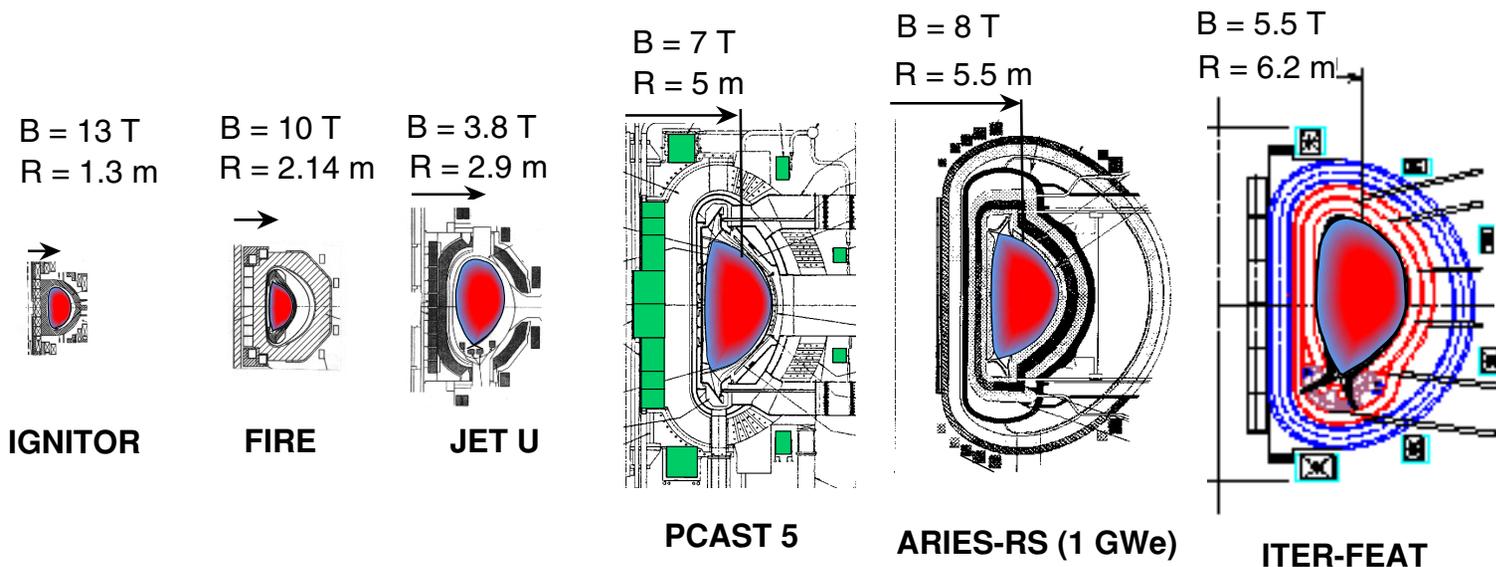
Preliminary FIRE Cost Estimate (FY99 US\$M)

	Estimated Cost	Contingency	Total with Contingency
1.0 Tokamak Core	266.3	78.5	343.8
1.1 Plasma Facing Components	71.9	19.2	
1.2 Vacuum Vessel/In-Vessel Structures	35.4	11.6	
1.3 TF Magnets /Structure	117.9	38.0	
1.4 PF Magnets/Structure	29.2	7.2	
1.5 Cryostat	1.9	0.6	
1.6 Support Structure	9.0	1.8	
2.0 Auxiliary Systems	135.6	42.5	178.1
2.1 Gas and Pellet Injection	7.1	1.4	
2.2 Vacuum Pumping System	9.6	3.4	
2.3 Fuel Recovery/Processing	7.0	1.0	
2.4 ICRF Heating	111.9	36.6	
3.0 Diagnostics (Startup)	22.0	4.9	26.9
4.0 Power Systems	177.3	42.0	219.3
5.0 Instrumentation and Controls	18.9	2.5	21.4
6.0 Site and Facilities	151.4	33.8	185.2
7.0 Machine Assembly and Remote Maintenance	77.0	18.0	95.0
8.0 Project Support and Oversight	88.8	13.3	102.2
9.0 Preparation for Operations/Spares	16.2	2.4	18.6
Preconceptual Cost Estimate (FY99 US\$M)	953.6	237.8	1190.4

Assumes a Green Field Site with **No** site credits or significant equipment reuse.

June 5, 2001

Comparison of Burning Plasma Device Parameters



Cost Drivers	IGNITOR	FIRE	JET U	PCAST	ARIES-RS	ITER-FEAT
Plasma Volume (m ³)	11	27	108	390	350	828
Plasma Surface (m ²)	36	67	160	420	420	610
Plasma Current (MA)	12	7.7	6	15	11.3	15
Magnet Energy (GJ)	5	5	1.6	40	85	50
Fusion Power (MW)	100	150	30	400	2170	400
Burn Duration (s), inductive	~1	20	10	120	steady	400
τ Burn/ τ CR		~2	0.6	1	steady	2
Cost Estimate (\$B-2000\$)		1.2	~0.6	6.7	10.6*	4.6?

* first , \$5.3 B for 10th of a kind

AR RS/ITERs/PCAST/FIRE/IGN

FIRE Issues and Needs

- Most are the same as for ITER-FEAT!
- Differences arise due to:
 - Double null divertor - higher δ , shorter path to divertor, neutral stability point no asymmetric alpha ripple loss region, ($\delta B/B = 0.3\%$)
 - Lower density relative to n_{GW} , higher density relative to NBI, RF, neutrals
 - All metal PFCs, esp. W divertor targets, • No neutral beam heating
- Specific Interests (requests)
 - Core Confinement (H-Mode and close relatives)
 - Understand requirements for enhanced H-modes at $n/n_{GW} \approx 0.6 - 0.7$
 - Compare SN \Rightarrow DN or nearly DN ; maybe more than triangularity
 - Extend global studies/analysis $H = H(\delta, n/n_{GW}, n(0)/\langle n \rangle)$
 - H-mode power threshold for DN, hysteresis, $H = f(P - P_{th})$
 - Pedestal height/width as SN \Rightarrow DN; elms as SN \Rightarrow DN
 - Rotation as SN \Rightarrow DN
 - Expand H-Mode data base for ICRF only plasmas
 - Demonstration discharges and similarity studies
 - Density Profile Peaking - expectations/requirements?

FIRE Issues and Needs (p.2)

- Internal Transport Barriers (AT Modes)
 - Access to ATs with: RF heated, $q_{95} \sim 3.5 - 4$, $T_i/T_e \approx 1$,
 - density peaking needed for efficient LHCD
 - $n = 1$ stabilization by feedback
- SOL and Divertor - Impurities
 - Justification for using $n_z \downarrow$ as $n_e \uparrow$?
 - ASDEX Upgrade and C-Mod Hi Z impurity in core and “tritium” retention
 - Consistency of partially detached divertor with good τ_E and He removal
 - Models and improved designs for extending lifetime (Elms/disruptions)
- Plasma Termination and Halo Currents
 - Does DN neutral zone reduce force or frequency of disruptions?
 - Develop early warning, mitigation and recovery techniques
- Finite- β effects
 - stabilization of NTMs using LHCD (Δ' modification)
 - elms for enhanced confinement modes
 - TAE, EPM studies in DD with beams and RF
- Diagnostic development - high priority needs to added in a future meeting

More Work Needed to Define Plasma Control Possibilities

- Density (core, edge)
 - pellet fueling/divertor pumping
 - density relative to nGW, fast alpha
- ITBs
 - ICRH ala C-Mod
 - control timing and strength of ITBs
- Current Profile Control
 - ramping, Lower Hybrid Current Drive
- Rotation Control
 - edge NBI injection being looked at
 - What are the rotation requirements?
- RWM Stabilization
 - feedback coils in port plugs near plasma
- Disruption
 - pellets, jets, neural net control systems

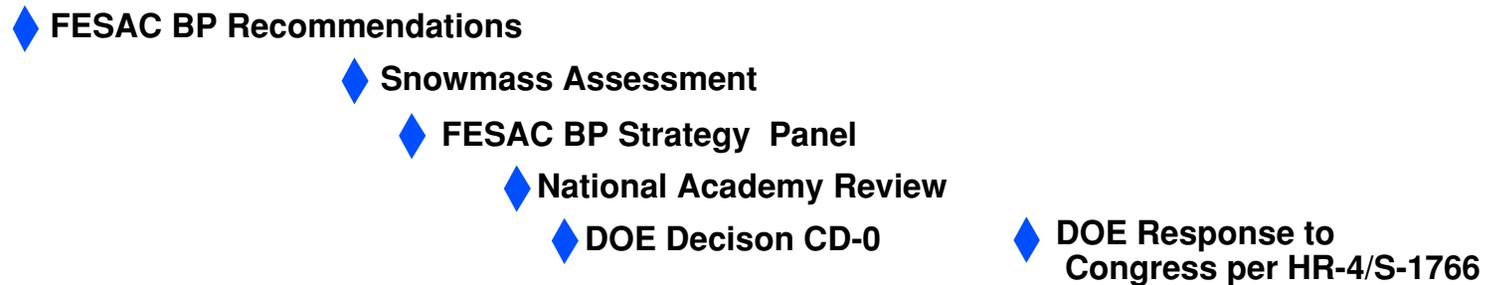
Timetable for a Major Next Step in Magnetic Fusion

USFY 2001	USFY 2002	USFY 2003	USFY 2004	USFY 2005
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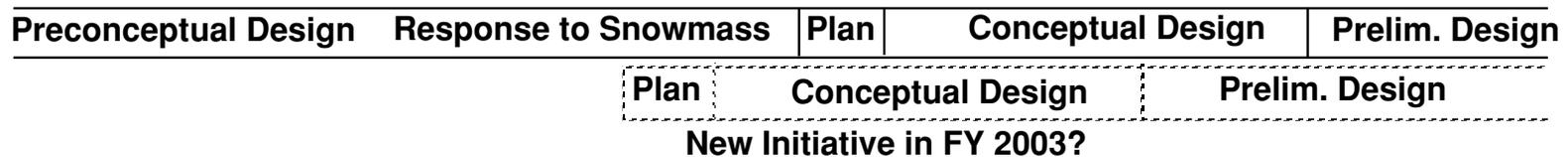
ITER Activities and Decisions



US Activities and Decisions



U.S. Burning Plasma Design Activity - FIRE



Summary

- A Window of Opportunity may be opening for U.S. Energy R&D. We should be ready. The Diversified International Portfolio has advantages for addressing the science and technology issues of fusion.
- FIRE is being designed to :
 - address the important burning plasma issues, performance ~ ITER
 - investigate the strong non-linear coupling between BP and AT,
 - stimulate the development of reactor relevant PFC technology, and
 - provide generic BP science and possibly BP infrastructure for non-tokamak BP experiments in the U. S.
- Some areas that need additional work to realize this potential include:
 - Apply recent enhanced confinement and advanced modes to FIRE
 - Understand conditions for enhanced confinement regimes-triangularity
 - Compare DN relative to SN - confinement, stability, divertor, etc
 - Complete disruption analysis, develop better disruption control/mitigation.
- If a positive decision is made in this year, FIRE is ready to begin Conceptual Design in FY2003 with target of first plasmas ~ 2010.

<http://fire.pppl.gov>

European Fusion Program Status

- **Final Negotiations for 6th Framework Program.** Program composition with reduced budget is under intense debate. JET's future is threatened.

- **Response to Airaghi Report Recommendations due May 2002**

3. To proceed with the 'Next Step' in the international collaboration perspective of the New-ITER, the European Union should within the next 2 years:

- **Conclude negotiations on the legal and organisational structure of the future venture**
- **Actively seek a European site for the New-ITER, since this is the best option from a European viewpoint.**
- **Conduct a thorough review of the financial issues, including the different financial costs and benefits of siting it in Europe, Canada or Japan, and establish the extent to which Japan would support the construction of New-ITER outside Japan.**
- **Examine in detail the recent interesting expression of interest received from the Canadian Consortium.**

4. In the same 2-year period, due to the uncertainty over the outcome of the international negotiations, Europe should study an alternative to New-ITER, which would be suitable to be pursued by Europe alone. For example, a copper magnet machine which would still achieve the required objective of demonstrating a burning plasma under reactor conditions even if this would delay the integration of the superconducting technologies.

Japan Fusion Program Status

- JAFY 02 underway with significantly reduced fusion budgets (- 50%) for JAERI Fusion and LHD (-30%). ITER activities funded at \approx \$3.5M.
- ITER Decision still on hold, site offer > 6 months behind schedule.
- Significant amount of work has been done on JT-60 SC.
 - ISFNT paper by Matsukawa
 - SOFE paper by IshidaDecision on hold pending ITER decision.
- Next stage of Large Helical Device being planned.
- Reactor Studies have goals similar to US ARIES Goals
cost competitiveness of fusion important – advanced physics and technology

Portfolio Elements

A Strategy for the US

- Given:
 - the growing support for fusion and burning plasmas within the US
 - FESAC recommendations on BP
 - HR4 recommendations on fusion and BP
 - Positive statements from the Administration
 - uncertainty of ITER
 - Reduced fusion budgets in EU and JA
 - Lack of site proposals by either EU or JA
- The US needs a Technically Based Roadmap for Magnetic Fusion
 - Snowmass, FESAC, and NRC reviews will provide a basis
 - must address the fundamental issue of:
 - One Large Integrated Project versus Diversified International Portfolio
- Near Term Actions
 - develop a Roadmap for the US Magnetic Fusion Program
 - do our homework on possible ITER roles **prior to joining ITER negotiations**
 - continue to develop a design for a US based burning plasma experiment as per HR4 as viable option until the parties commit to ITER construction.