Status of FIRE

A Potential Next Step Option

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Next Step Options Program Advisory Committee Meeting
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http://fire.pppl.gov

Outline of FIRE Topics

Introduction

Objectives

Configuration and Parameters

Disruption Requirements

Diagnostics

Performance Projections

Pulse Evolution and Burn Control

Fast Alpha Effects

Long Pulse Capability (coil limits)

Advanced Tokamak (also Jardin)
Electrical Power Requirements

Cost Range

Timetable for Burning Plasmas

Critical Issues

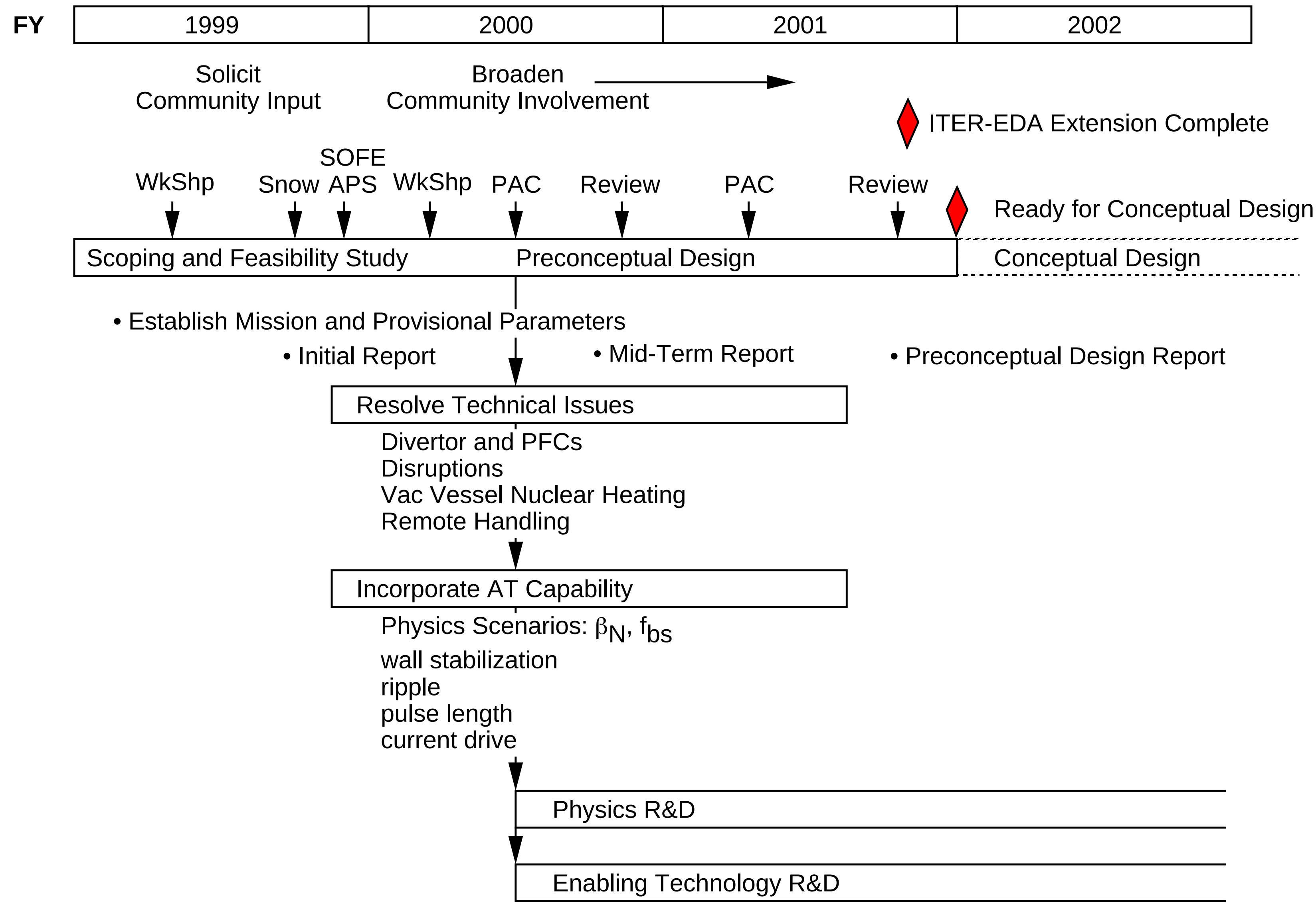
Conclusions

Contributors to the FIRE Design Study

FIRE is a design study for a major Next Step Option in magnetic fusion and is carried out through the Virtual Laboratory for Technology. FIRE has benefited from the prior design and R&D activities on BPX, TPX and ITER.

Advanced Energy Systems
Argonne National Laboratory
Bechtel Technology and Consulting
General Atomics Technology
Georgia Institute of Technology
Idaho National Engineering Laboratory
Lawrence Livermore National Laboratory
Massachusetts Institute of Technology
Oak Ridge National Laboratory
Princeton Plasma Physics Laboratory
Sandia National Laboratory
Stone and Webster
The Boeing Company
University of Illinois
University of Wisconsin

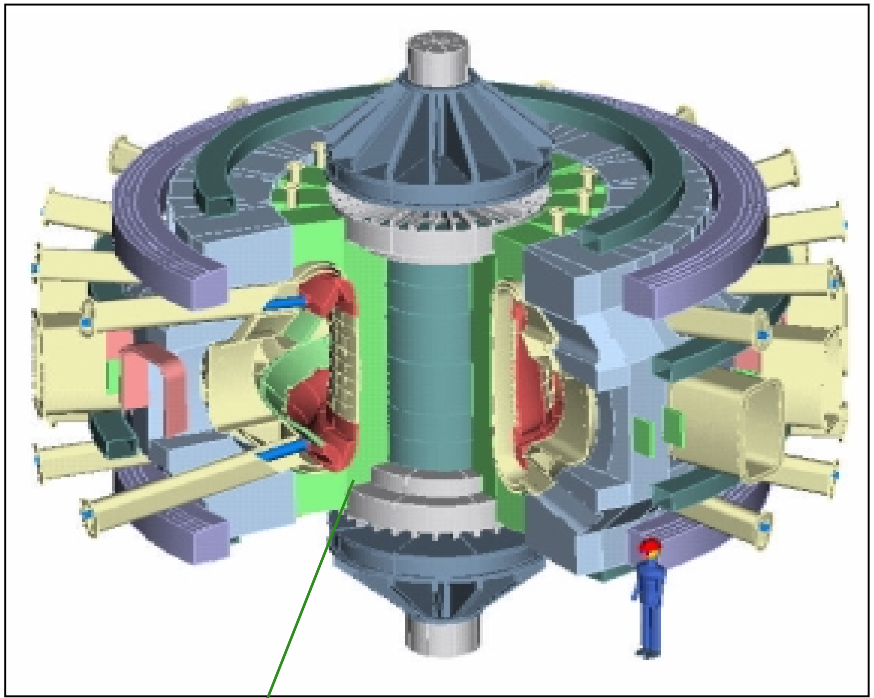
Basic Approach for an Advanced Tokamak Next Step (FIRE)



Fusion Science Objectives for a Major Next Step Experiment (e.g., FIRE)

- Explore and understand the physics of alpha-dominated fusion plasmas:
  - Energy and particle transport (extend confinement predictability)
  - Macroscopic stability (β-limit, wall stabilization, NTMs)
  - Wave-particle interactions (fast alpha driven effects)
  - Plasma boundary (density limit, power and particle flow)
  - Strong coupling of previous issues due to self-heating(self-organization?)
- Test techniques to control and optimize alpha-dominated plasmas.
- Sustain alpha-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 some advanced operating modes and configurations that have the potential to lead to attractive fusion applications.

Fusion Ignition Research Experiment
(FIRE)



LN BeCu ("HTS")

Design Goals

- R = 2.0 m, a = 0.525 m
- B = 10 T, (12T)*
- W_{mag} = 3.8 GJ, (5.5 GJ)*
- I_p = 6.5 MA, (7.7 MA)*
- $P_{alpha} > P_{aux}$, P_{fusion} ~ 220 MW
- Q ~ 10, τ_E ~ 0.55s
- Burn Time ~ 20s (12s)*
- Tokamak Cost ≤ $0.3B

  Base Project Cost ≤ $1B

* Higher Field Option

Attain, explore, understand and optimize alpha-dominated plasmas to provide knowledge for the design of attractive MFE systems.

A Robust and Flexible Design for FIRE has been Achieved

- Toroidal and poloidal coil structures are independent allowing operational flexibility
    - The toroidal field coils are wedged with static compression rings to increase capability to withstand overturning moments and to ease manufacturing.

- 16 coil TF system with large bore provides
    - Large access ports (1.3m high by 0.7m wide) for maintenance and diagnostics.
    - Low TF ripple (0.3% at plasma edge) provides flexibility for lower current AT modes without large alpha losses due to ripple.

- Double-null divertor configuration for H-mode and AT modes with helium pumping that is maintainable/replaceable/upgradeable remotely

- Double wall vacuum vessel with integral shielding (ITER-like) to reduce neutron dose to TF and PF coils, and machine structure.

- Cooling to LN2 allows full field (10T) flattop for 20s or 4T (TPX-like) flattop for 250s.

The FIRE Engineering Report and 16 FIRE papers presented at the IEEE Symposium on Fusion Engineering are available on the web at http://fire.pppl.gov

Basic Parameters and Features of FIRE Reference Baseline

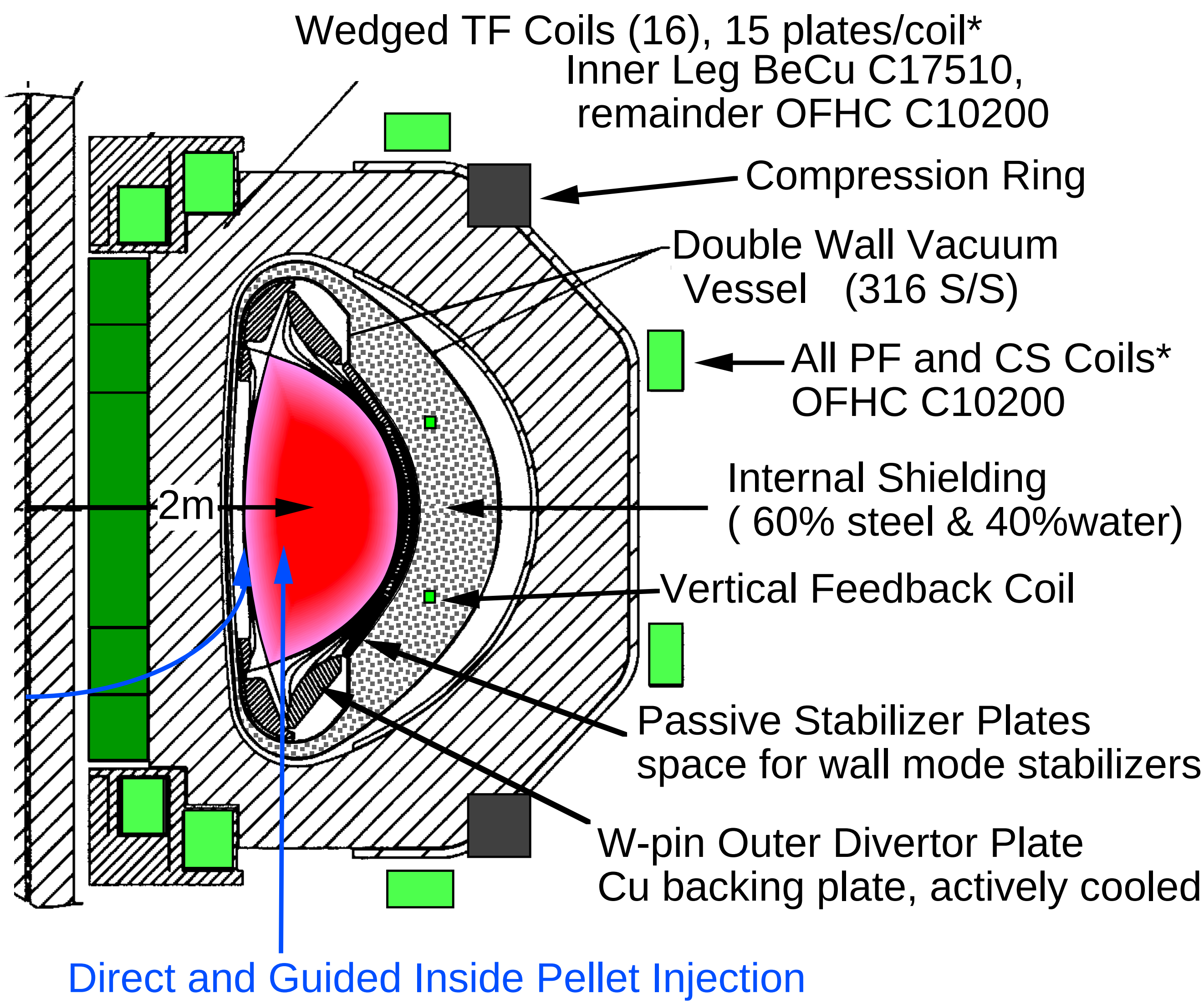
R, major radius	2.0 m
a, minor radius	0.525 m
$\kappa 95$, elongation at 95% flux surface	~1.8
$\delta 95$, triangularity at 95% flux surface	~0.4
q95, safety factor at 95% flux surface	>3
Bt, toroidal magnetic field	10 T with 16 coils, < 0.5% ripple @ Outer MP
Toroidal magnet energy	3.7 GJ
Ip, plasma current	~6.5 MA (7.7 MA at 12 T)
Magnetic field flat top, burn time	21 s at 10 T, P_{fusion} ~ 200 MW)
Pulse repetition time	2 hr @ full field
ICRF heating power, maximum	30 MW, 100MHz for $2\Omega_T$, 4 mid-plane ports
Neutral beam heating	None, may have diagnostic neutral beam
Lower Hybrid Current Drive	None in baseline, upgrade for AT phase
Plasma fueling	Pellet injection (≥2.5km/s vertical launch inside mag axis, possible guided slower speed pellets)
First wall materials	Be tiles, no carbon
First wall cooling	Inertial between pulses
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-inertial, outer plate active - water
Fusion Power/ Fusion Power Density	~200 MW, ~10 MW m-3 in plasma
Neutron wall loading	~ 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 Bt and Ip
Tritium site inventory	Goal < 30 g, Category 3, Low Hazard Nuclear Facility

Upgrade to B = 12T and Ip = 7.7MA with a 12 second flat top has been identified.

FIRE Incorporates Advanced Tokamak Innovations

AT Features

- DN divertor
- strong shaping
- very low ripple
- 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.

FIRE Disruption and Disruption-Related Design Basis Recommendations

Parameter	Value (Range)	Comment
Frequency	10% (10-30%) per pulse	30% for plasma development ≤ 10% for mature (repetitive) operation
Number (3,000 full performance attempts)	300 (900)	300 at full W_{th} and W_{mag}, balance at ≤ 0.5 W_{th} and full W_{mag}
Thermal energy	33 MJ	For typical 200 MW plasma
Thermal quench duration	0.2 (0.1–0.5) ms	Single or multi-step thermal quench
Fraction of W_{th} to divertor	80–100%	By conduction to targets, up to 2:1 toroidal asymmetry
Fraction of W_{th} to FW (baffle)	≤ 30%	By radiation (to FW) or conduction (to baffle)
In-divertor partition (inside/outside)	2:1 – 1:2	For SN plasmas. Significant uncertainty. No data for DN plasmas
Poloidal localization in divertor	3-x normal SOL; (1-x to 10-x)	Incident energy, with up to 2:1 toroidal asymmetry. Plasma shielding and re-radiation will likely redistribute in-divertor energy
Magnetic energy	35 (?) MJ	For 6.5 MA, total out to VV
Current quench duration	6 (2-600) ms	Duration ≥30 ms: more-severe VDE and halo current
Maximum current decay rate	3 MA/ms	May occur only during fastest part of current quench; typical maximum rate ~1 MA/ms
Fraction of W_{mag} to FW, by radiation	80–100%	By radiation, with poloidal peaking factor ~ 2
Fraction of W_{mag} to FW, by localized conduction	0-20%	From VDE: depends on VDE evolution and in-vessel halo current. Hot-plasma VDEs may also deposit ~0.2-1.0 W_{th} on localized portion(s) of FW. Toroidal alignment critical

Table Continues

FIRE Disruption and Disruption-Related Design Basis Recommendations (cont'd)

Parameter	Value (Range)	Comment
VDE frequency	TBD (??? 1% of pulses, or 10% of disruptions???)	Very uncertain. May be able to maintain vertical position control after thermal quench. But margin/noise sensitivity uncertain. Control failure yields VDE or loss of after-thermal-quench control
Halo current fraction $I_{h,max}/I_{p0}$	0.4 (0.01-0.50)	Highest value may apply (depends on passive stabilizer configuration)
Toroidal peaking factor	2 ($1.2 \leq$ TPF ≤ 4)	TPF up to 2 yields 'sinϕ' distribution; TPF > 2 yields 'localized filament'
$(I_{h,max}/I_{p0})$*TPF	≤ 0.50 (typical maximum)	Data bound is ≤ 0.75 (see text)
Runaway electron current (following disruption or fast shutdown)	50% I_p (0-50%)	Highly uncertain. $I_{RA} > 1$ MA requires ≥ 1 A seed source. Not expected in thermal plasma, but pellet shutdown may seed avalanche. MHD fluctuations may offset part or all of avalanche growth.
Runaway energy	~15 MeV	Limited by knock-on avalanche
Localization of runaway deposition	≤ 1 m^2	Poloidal localization to a ~0.1-m (poloidal) section of the FW or divertor target expected; toroidal localization depends on pfc and wall alignment to toroidal field

- **Basis: ITER EDA /EG and *ITER Physics Basis, Chapter 3***

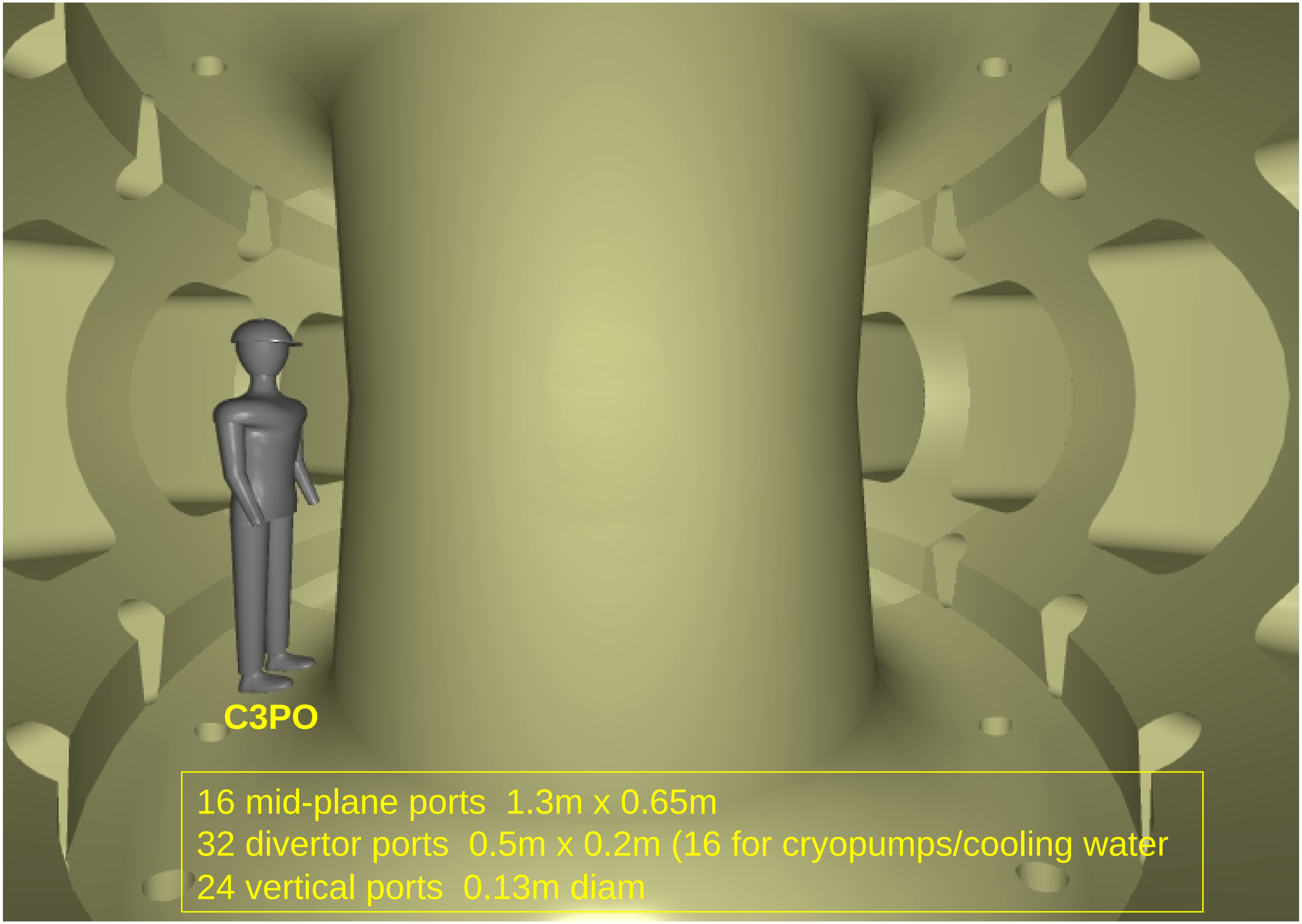
- **Lacks for FIRE: thermal quench data, DN data**

Summary of Disruption Issues

- Disruption, halo current and runaway electron avalanche (RAe) characteristics have been specified (based on ITER Physics Basis; VV and in-VV response TBD
- Thermal quench data (SN) quality is poor; DN data is lacking (R&D for C-Mod and DIII-D)
- Divertor plasma shielding and radiative energy redistribution is critical issue
- Halo current magnitude and VV force estimated: need TSC and toroidal asymmetry model (3-D plasma) for details; passive stabilizer role and asymmetry needs further physics R&D (ASDEX-U)
- Possibility of after thermal quench VDE stabilization TBD
- Outcome of RAe uncertain (seed and MHD levels); potential for serious in-vessel damage

Much work to be done in this area.

FIRE would have Access for Diagnostics and Heating



Provisional List of Diagnostics (1)

- Magnetic Measurements
  - Rogowski Coils, Flux/voltage loops, Discrete Br, Bz coils, Saddle coils, Diamagnetic loops, Halo current sensors, Hall effect sensors
- Current Density Profiles
  - Motional Stark effect with DNB, Infrared polarimetry
- Electron Density and Temperature
  - Thomson Scattering, ECE Heterodyne Radiometer, FIR interferometer, Multichannel Interferometer, ECE Michelson interferometer, ECE Grating Polychromator, Millimeter-wave Reflectometer
- Ion Temperature
  - Charge Exchange Spectroscopy with DNB, X-Ray Crystal Spectrometer, Charge Exchange Neutral Analyzer (edge)
- Visible and Total Radiation
  - Visible Survey Spectrometer, Visible Filterscopes, Visible Bremsstrahlung Array, Bolometer Arrays, Plasma TV and Infrared TV
- Ultra Violet and X-Ray Radiation
  - UV Survey Spectrometer, Hard X-ray detectors, Soft x-ray Spectrometer, X-ray pulse height analysis

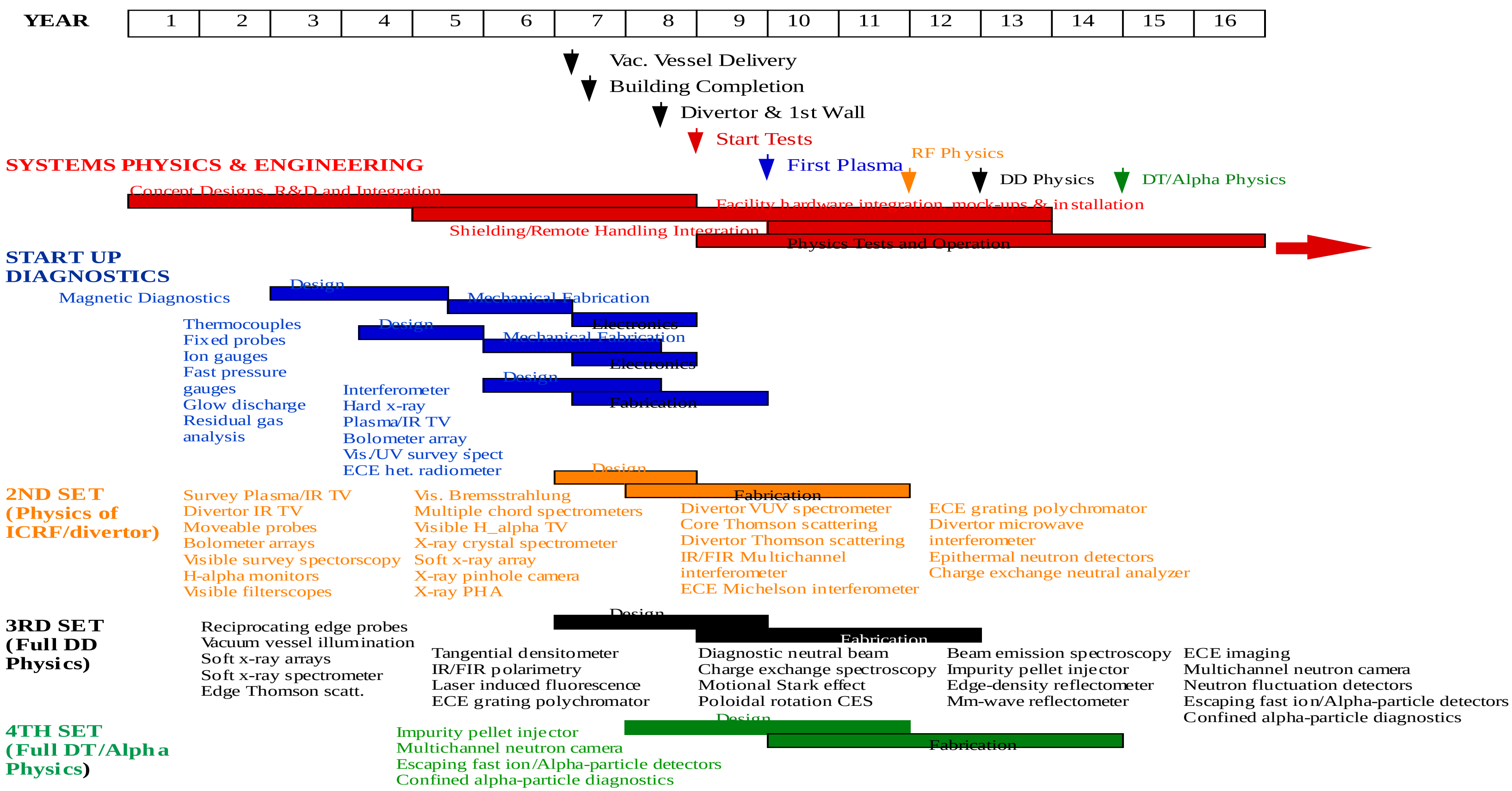
Provisional List of Diagnostics (2)

- MHD and Fluctuations
  - Mirnov Coils, Locked-mode coils, Soft x-ray array, Beam emission spectroscopy, Millimeter wave reflectometer, Collective scattering
- Particle Measurements and Diagnostic Neutral Beam
  - Epithermal Neutron detectors, Multichannel Neutron Collimator, Neutron Fluctuation detectors, Diagnostic Neutral Beam
- Charged Fusion Products
  - Escaping Alpha Particle detectors, IR TV (shared with total radiation), Collective Scattering (CO2?), α-CXRS, Knock-on neutron detectors
- Divertor Diagnostics
  - Divertor IR TV, Visible Hα TV, UV Spectrometer, Divertor Bolometer Arrays, Multichord visible spectrometer, Divertor Hα monitors, ASDEX-type Neutral Pressure Gauges, Divertor Thomson Scattering, Penning Spectroscopy, Divertor reflectometer
- Plasma Edge and Vacuum Diagnostics
  - Thermocouples, Fixed Edge Probes, Fast Movable Edge Probes, Torus Ion Gauges, Residual Gas Analyzers, Glow Discharge Probes, Vacuum Vessel Illumination

R&D Concerns

- What are impacts of high-field, highly shaped, high-$n_{e,}$ high radiation, RF-only on diagnostics selection and development?
  - Reliability of magnetic diagnostics?
  - Lifetime of plasma-facing mirrors, other optical elements?
  - ECE overlap?
  - Interferometry refraction/wavelength?
  - Functionality of x-ray systems?
  - CXRS and MSE techniques; capability for diagnostic neutral beam(s)?
  - Inside-launch reflectometry?
  - Confined alpha-particles?

FIRE: Diagnostics Schedule



FIRE DIAGNOSTICS SCHEDULE: REVISION 0 1 SEPTEMBER 1999

K. M. Young 5/2/00

Recent Innovations have Markedly Improved the Technical Basis for a Compact High Field Tokamak Burning Plasma Exp't.

Tokamak experiments (1989-1999) have developed enhanced confinement modes that scale (e.g.,ITER-98H) 1.3 times higher than the 1989 CIT design assumption.

Alcator C-Mod - the prototype for Compact High Field tokamaks has shown:

- Confinement in excess of 1.4 times the 1989 design guidelines for CIT and ~1.15 times the recent ITER-98H design guidelines.
- Successful ICRF heating at high density in shaped diverted plasmas.
- Successful detached divertor operation at high power density.

VDEs and halo currents have made internal hardware design more difficult.

D-T experiments on TFTR and JET have shown:

- Tritium can be handled safely in a laboratory fusion experiment!!!
- D-T plasmas behaved roughly as predicted with slight improvements in confinement in plasmas with weak alpha-heating.

Engineering Innovations to increase capability and reduce cost

- Improved coil and plasma facing component materials, improved 3-D engineering computer models and design analysis, advanced manufacturing.

Guidelines for Estimating Plasma Performance

Confinement (Elmy H-mode) - Based on today's tokamak data base

$$\tau_E = 0.094\, I^{0.97}\, R^{1.7}\, a^{0.23}\, n_{20}^{0.41}\, B^{0.08} A_i^{0.2}\, \kappa^{0.67}\, P_{heat}^{-0.63}$$

Density Limit - Base on today's tokamak data base

$n_{20} \leq 0.75\, n_{GW} = 0.75\, I_p/\pi a^2$, H98 $\approx$ 1 up to 0.75 n_{GW} (JET, 1998)

Beta Limit - theory and tokamak data base

$\beta \leq \beta_N(I_p/aB)$, β_N ~2.5 conventional, β_N ~ 4 advanced

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

Pth $\geq$ (0.9/Ai) $n^{0.75}$ B R^2, nominal L to H, with H to L being ~ half when well below the density limit.

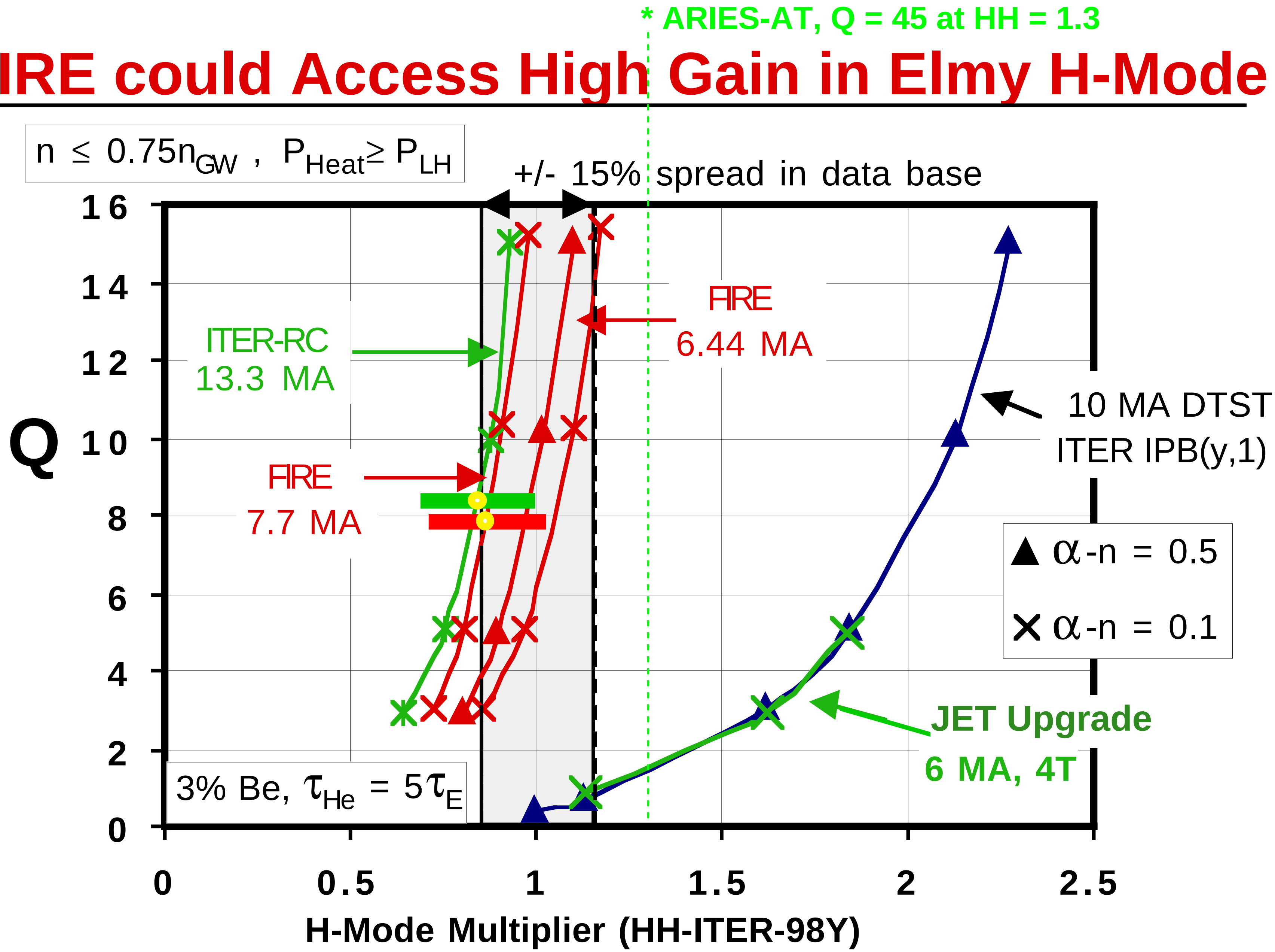
Helium Ash Confinement $\tau_{He} = 5\, \tau_E$, impurities = 3% Be

Understanding is mainly empirical. Better understanding is needed from existing experiments with improved simulations, and a benchmark in alpha-dominated fusion plasmas is needed for the design of an Fusion Energy Demonstration project.

Nominal FIRE Plasma Parameters from 0-D Simulations

R, plasma major radius, m	2.0
A, plasma minor radius, m	0.525
R/a , aspect ratio	3.8
κ_95, plasma elongation at 95% flux	1.77
δ_95, plasma triangularity at 95% flux	0.4
q_95	3.02
B_t, toroidal magnetic field, T	10
I_p, plasma current, MA	6.44
l_i(3), internal plasma inductance	0.8
Fraction of bootstrap current	0.25
Ion Mass, 50/50 D/T	2.5
<ne>, 10^20 /m^3, volume average	4.5
α_n, density profile peaking = 1 + α_n	0.5
<n>l/Greenwald Density Limit, ≤ 0.75	**0.70**
<T>n, density averaged temperature, keV	8.2
T(0), central temperature, keV	13.1
α_T, temperature profile peaking = 1 + α_T	1
Impurities, Be:high Z, %	3 : 0
Alpha ash accumulation, n_α/n_e, %	2.6
Zeff	1.41
ν*, collisionality at q = 1.5	0.043
P_ext , MW	22
P_fusion, MW	223
P_heat , MW	56.5
tau_p*(He)/tau_E	5.00
tau_E, energy confinement time s	0.57
ITER98H-multiplier, ≤1	**1.04**
ITER89P - Multiplier	2.41
$n_d(0)T(0)\tau_E$, 10^20 m^-3keVs	41.69
Q_DT	10.16
IA, MA	24.5
Plasma current redistribution time, s	13.9
Pheat/P(L->H), ≥ 1	**1.149**
W_p, plasma thermal energy, MJ	32.18
β_total, thermal plasma + alphas, %	3.11
β_N, ≤ 2.5	**2.54**
Core Plasma Pressure, atmospheres	~ 20

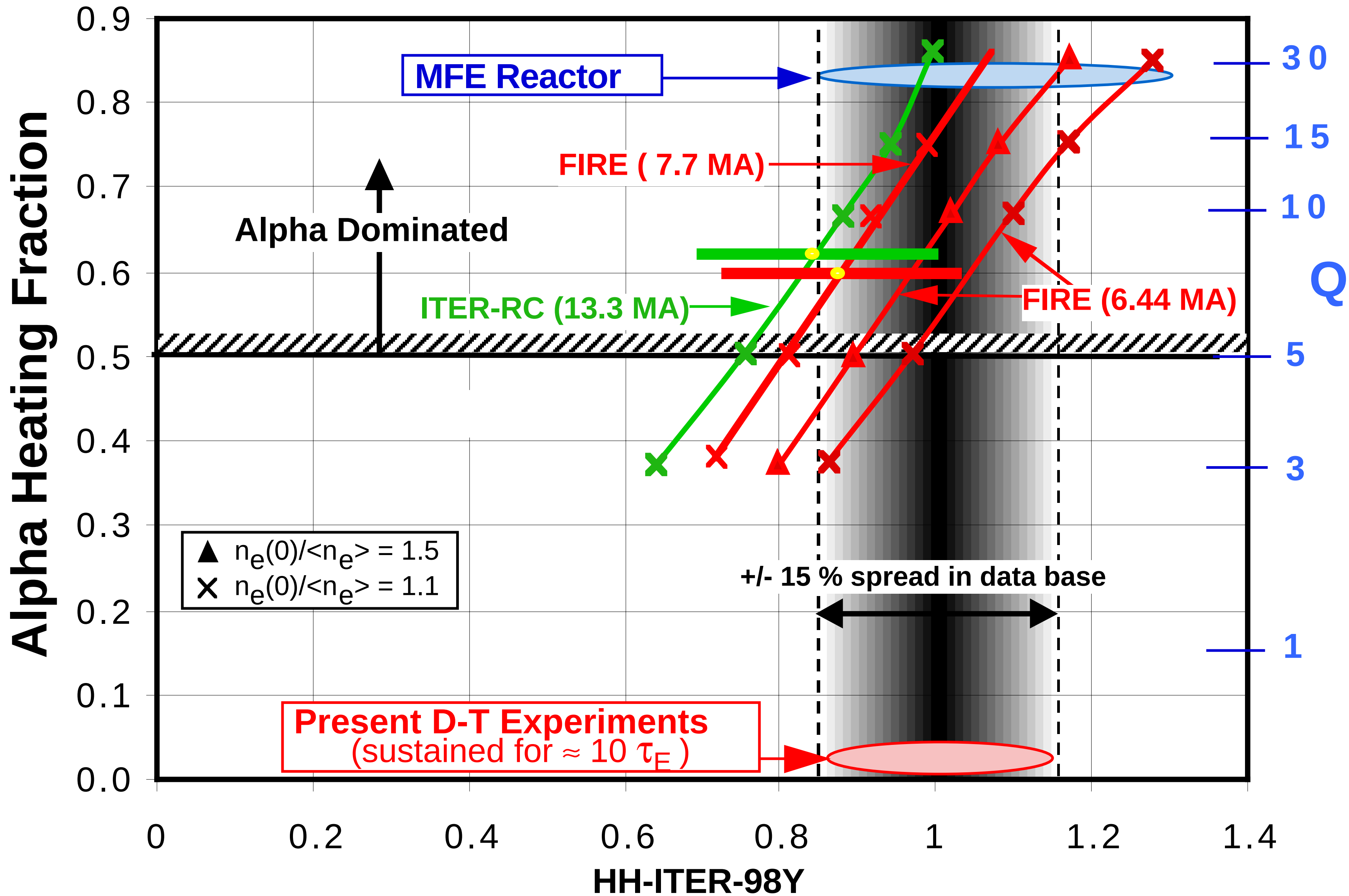
FIRE could Access High Gain in Elmy H-Mode



The baseline FIRE (6.44 MA) can access the alpha-dominated regime (Q > 5) for HH = 1.

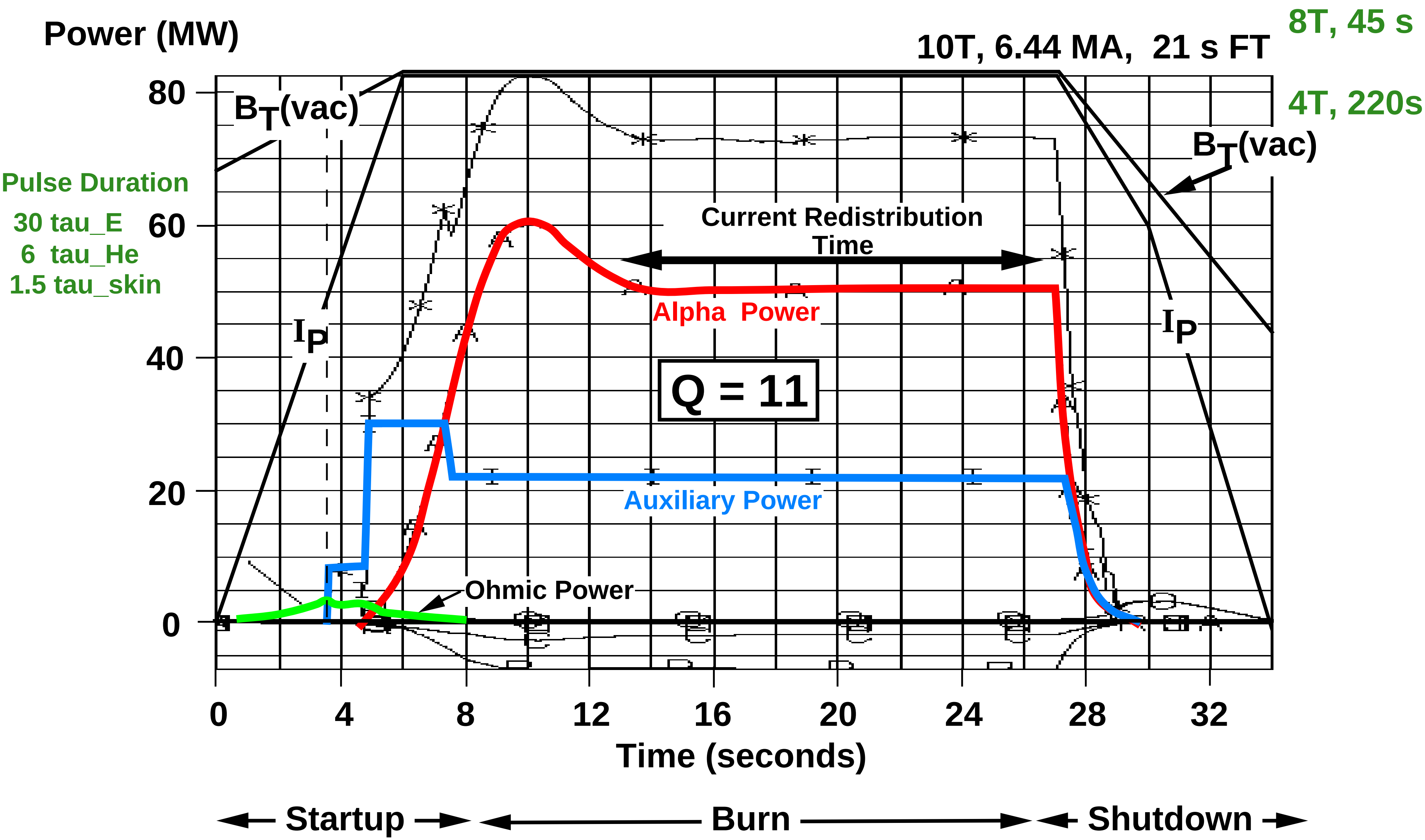
The Energy Mission is vulnerable to uncertainties in confinement.

FIRE could Access Alpha-Dominated Plasmas in H-Mode



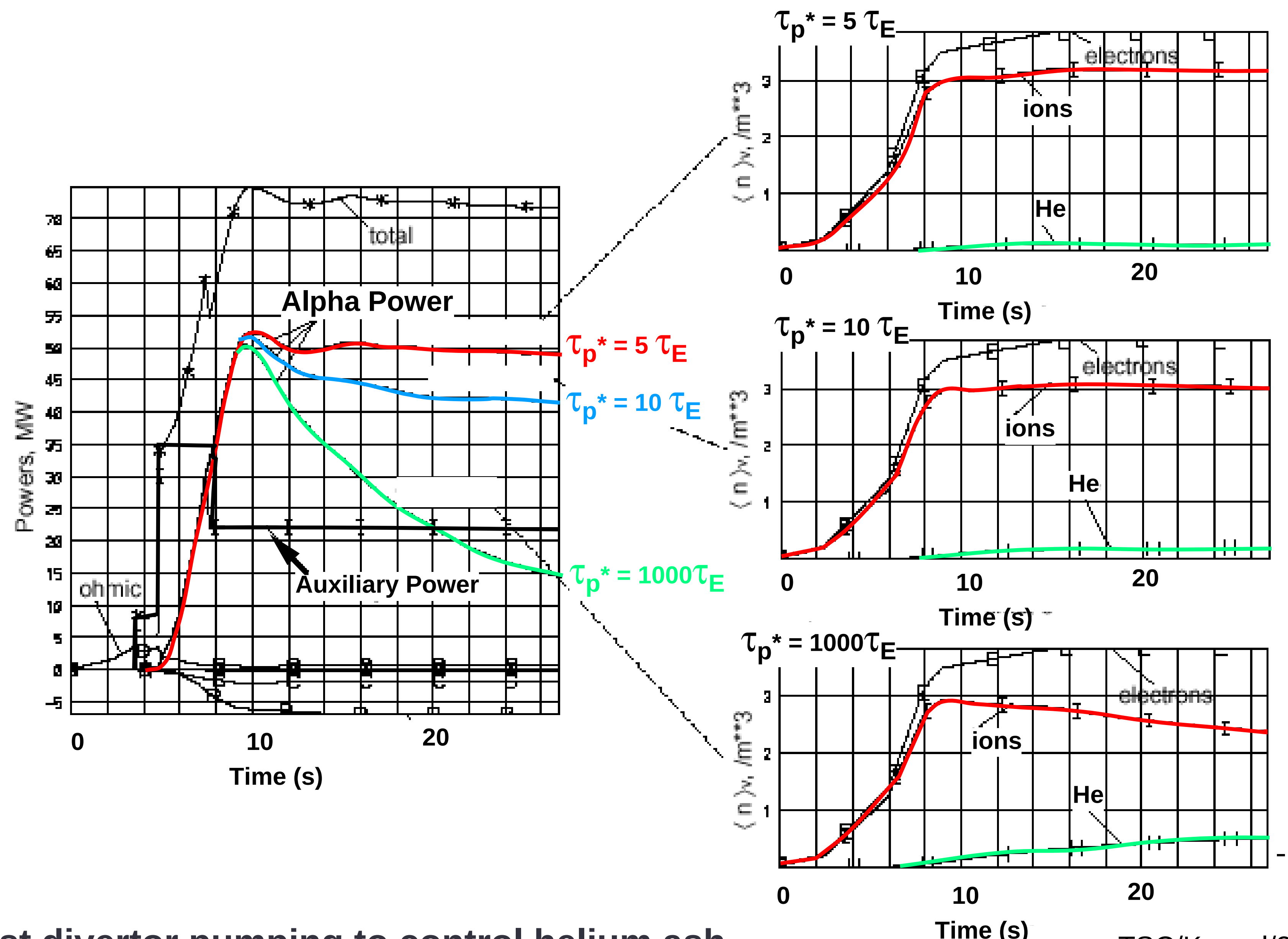
The Science Mission is robust to uncertainties in confinement.

1 1/2 -D Simulation* of Burn Control in FIRE



* The Tokamak Simulation Code (TSC) is one of several plasma simulation codes. **Click here http://w3.pppl.gov/topdac/**

Helium Ash Accumulation could be Explored on FIRE



Adjust divertor pumping to control helium ash

TSC/Kessel/21-q.ps

Single Particle Alpha Loss in AT Regimes

- **Summary of recent work (White)**
  - » Analysis of alpha loss using guiding center (ORBIT) code with collisions
  - » FIRE with $q(0) \approx 3$ has 6% prompt loss, 12 % loss at 50 ms($\approx \tau_s$)
  - » Loss concentrated at midplane

- **Action Items**
  - » Calculate power density of prompt loss alphas on first wall using ORBIT and/or LORENTZ code
  - » Need to benchmark loss predictions to experiment
    - Ripple experiments on JET (δ16 -> δ32), JFT-2M

Collective Instabilities in FIRE

- **Non-perturbative Instabilities in FIRE for positive and reverse magnetic shear (Gorelenkov)**

  » **Non-perturbative Alfvén eigenmodes relevant to FIRE**

  » **HIgh-N STability analysis applied to q(0)<1 and q(0)>1 reference plasmas**

  » **q(0)<1 plasmas are unstable to low-n RTAEs**
    - **internal redistribution possible**

  » **q(0)>1 plasmas are always unstable to low-n RTAEs**
    - **modes strongest near q-min (as seen on TFTR)**
    - **internal redistribution possible**

Key issue is whether modes will be strong enough to significantly enhance loss

theory and experimental activites planned, see Physics Workshop Summary

Non-linear TAE Physics and Resonance Overlap

- For high-n modes, need to assess role of resonance overlap in burning plasma
- Action Items
  - » Determine if TFTR experiments are a good example of resonance overlap
    - ORBIT analysis needed with multiple modes (White)
    - compare to Fokker-Planck-MHD simulations (Todo)
    - extrapolate to burning plasma

FIRE could Access High-Gain Advanced Tokamak Regimes for Long Durations

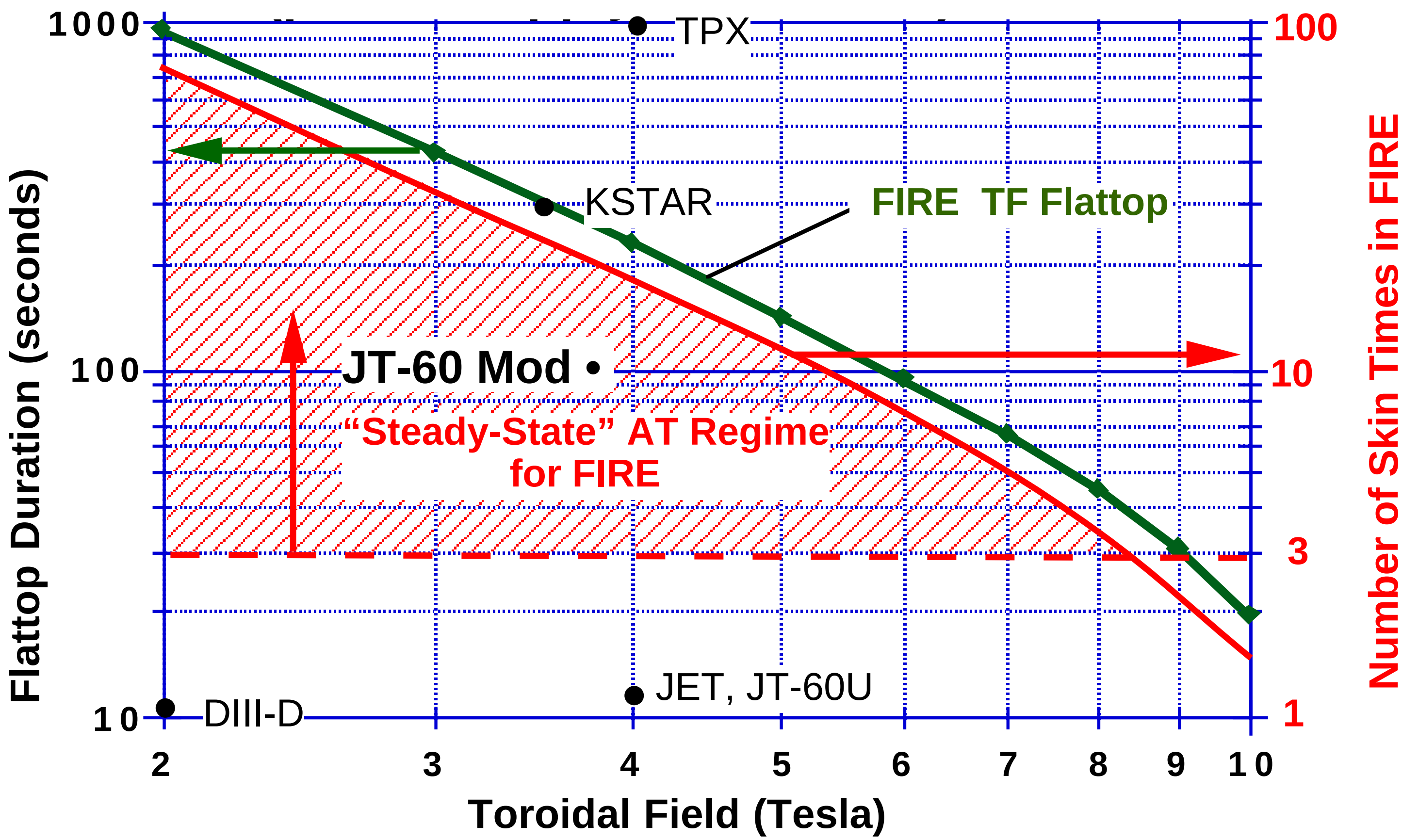
- The coupling of advanced tokamak modes with strongly burning plasmas is a generic issue for all advanced “toroidal” systems. The VLT PAC, Snowmass Burning Plasma and Energy Subgroup B recommended that a burning plasma experiment should have AT capability.

- FIRE, with strong plasma shaping, flexible double null poloidal divertor, low TF ripple, dual inside launch pellet injectors, and space reserved for the addition of current drive (LHCD) and/or a smart conducting wall, has the capabilities needed to investigate advanced tokamak regimes in a high gain burning plasma.

- The LN inertially cooled TF coil has a pulse length capability ~250 s at 4T for DD plasmas. This long pulse - AT capability rivals that of any existing divertor tokamak or any under construction. **The coils are not the limit.**

- Recent AT regimes on DIII-D (Shot 98977) sustained for ~ 16 τ_E serve as demonstration discharges for initial AT experiments on FIRE. Need to develop self-consistent scenarios with profile control on FIRE with durations ~ 3 τ_{skin} .

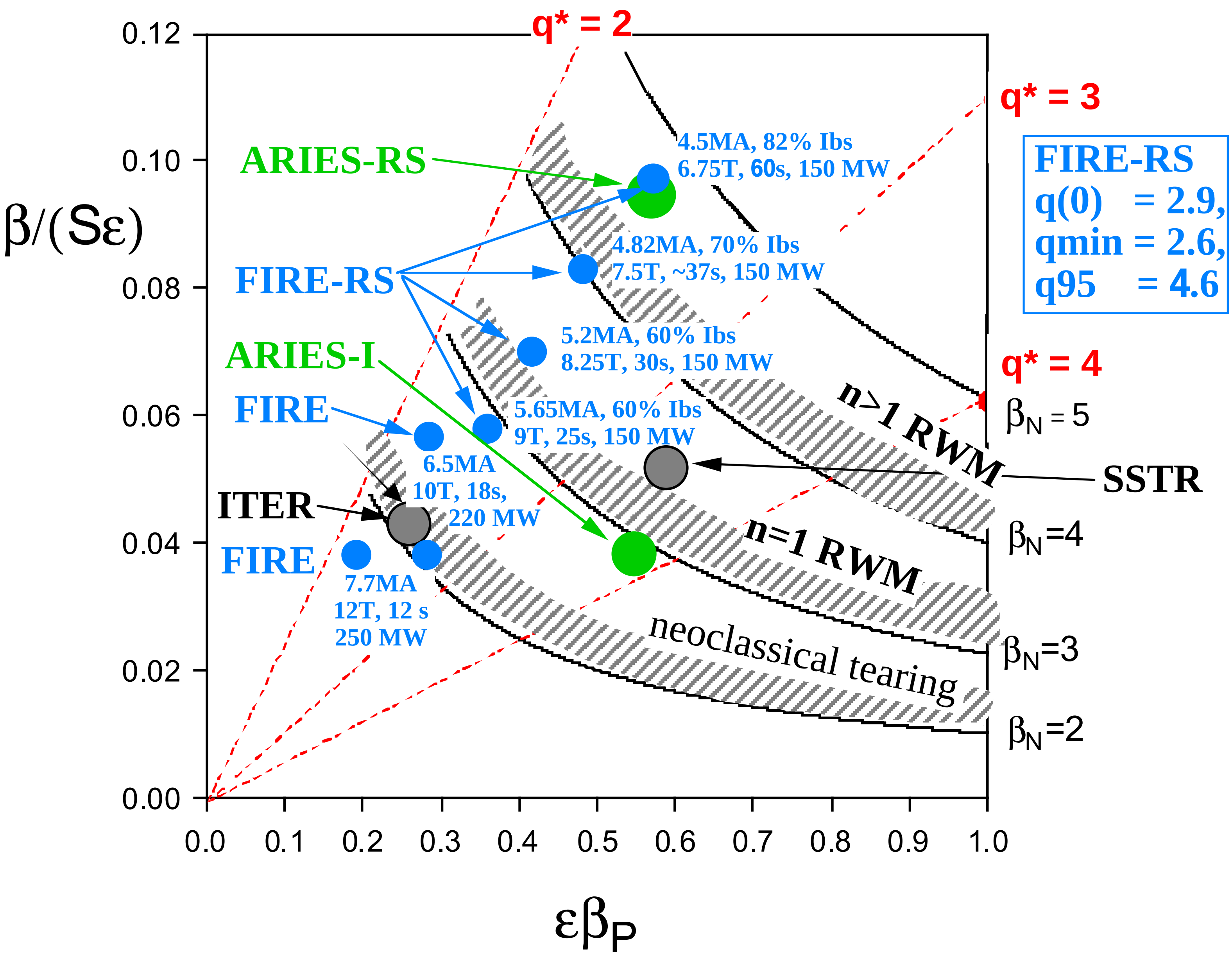
FIRE could Access "Long Pulse" Advanced Tokamak Mode Studies at Reduced Toroidal Field.



Note: FIRE is ≈ the same physical size as TPX and KSTAR.
At Q = 10 parameters, typical skin time in FIRE is 13 s and is 200 s in ITER-FEAT .

The combination of JET-U, JT-60 Mod, KSTAR and FIRE could cover the range fromsteady-state non-burning advanced-tokamak modes to "quasi-equilibrium" burning plasmas in advanced tokamak modes.

FIRE can Access MHD Regimes of Interest from Today's Data Base to those Envisioned for ARIES-RS



FIRE can Test Advanced Regimes of Relevance to ARIES-AT

Confinement Required to access this regime

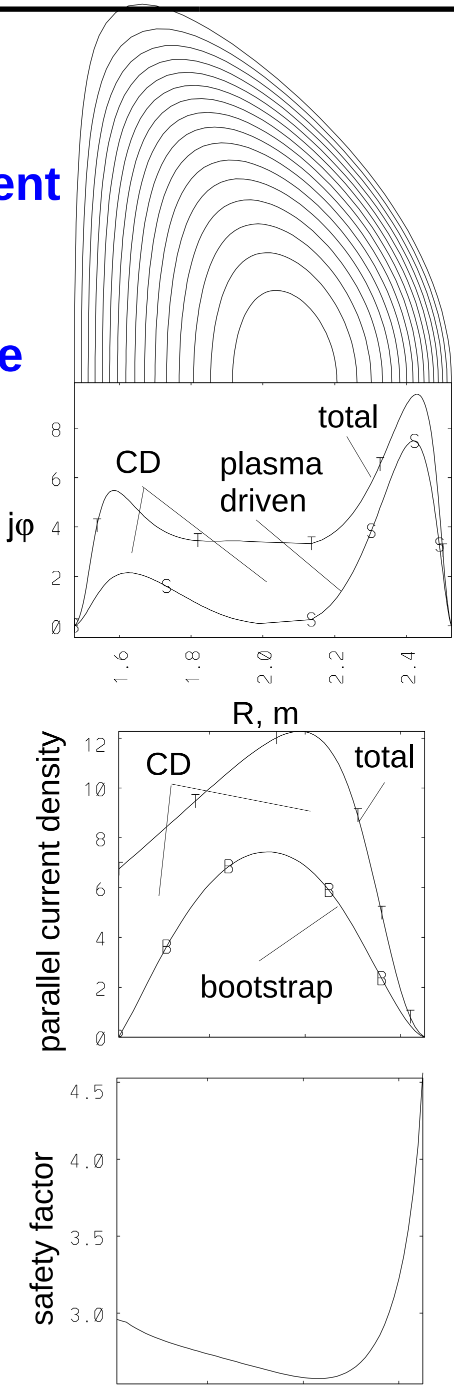
Q = 10, HH = 1.2

or

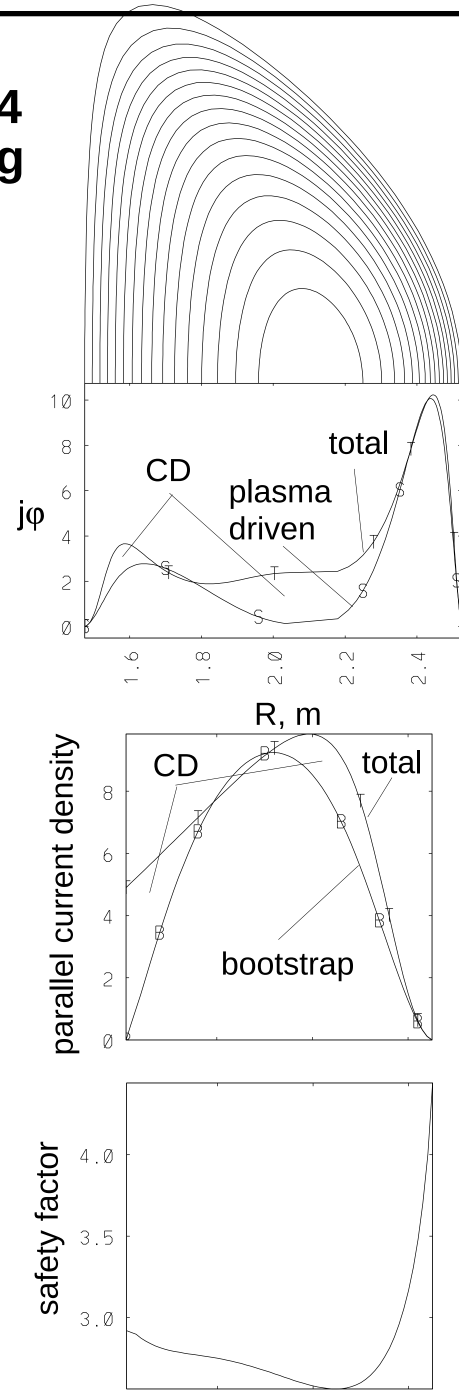
Q = 5, HH = 1.06

Duration

~2 tau_skin



Case 1 Modest AT		Case 4 Strong AT
30	Flat top(s)	60
5.65	Ip(MA)	4.50
9.00	B_T(T)	6.75
2.90	q_o	2.90
2.60	q_{min}	2.60
1.31	β_p	2.11
2.60	β_N	4.50
3.10	β(%)	5.70
0.42	li	0.39
0.50	f_{bs}	0.82
165	P_{fus}(MW)	170
29.4	W_{th}(MJ)	30.1
0.65	n_e/n_{Gr}	0.81
2.40	α-loss(%)	9.40



Confinement Required to access this regime

Q = 10, HH = 1.56

or

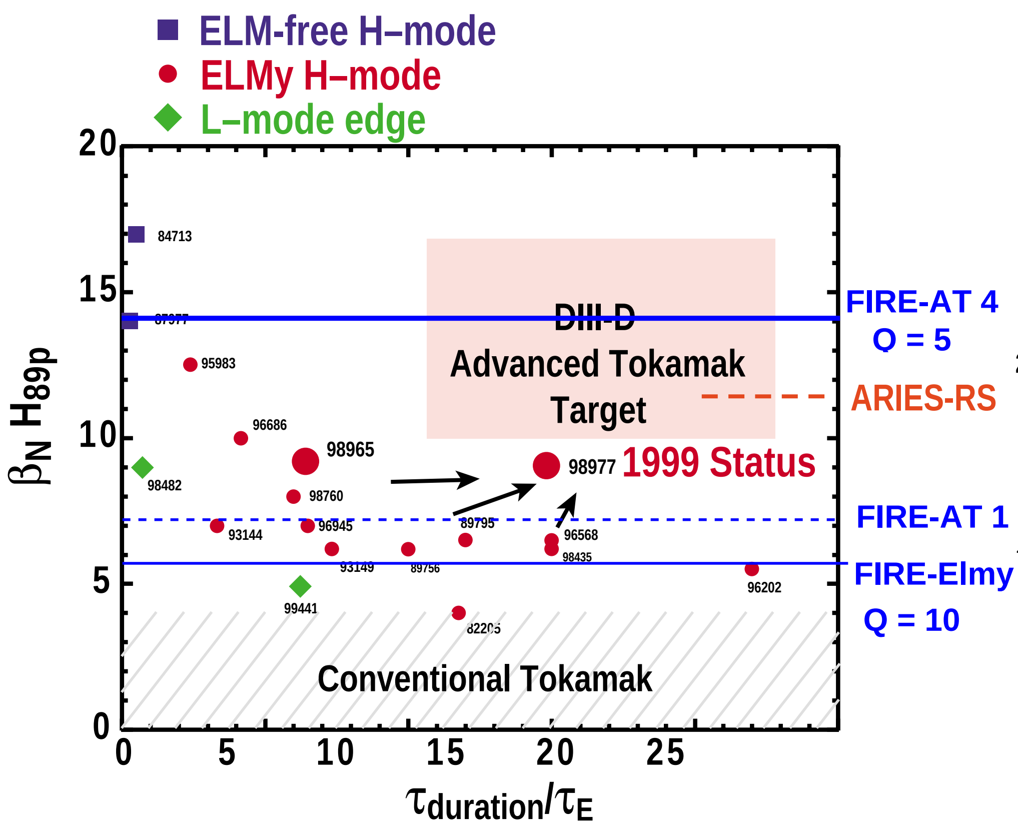
Q = 5, HH = 1.36

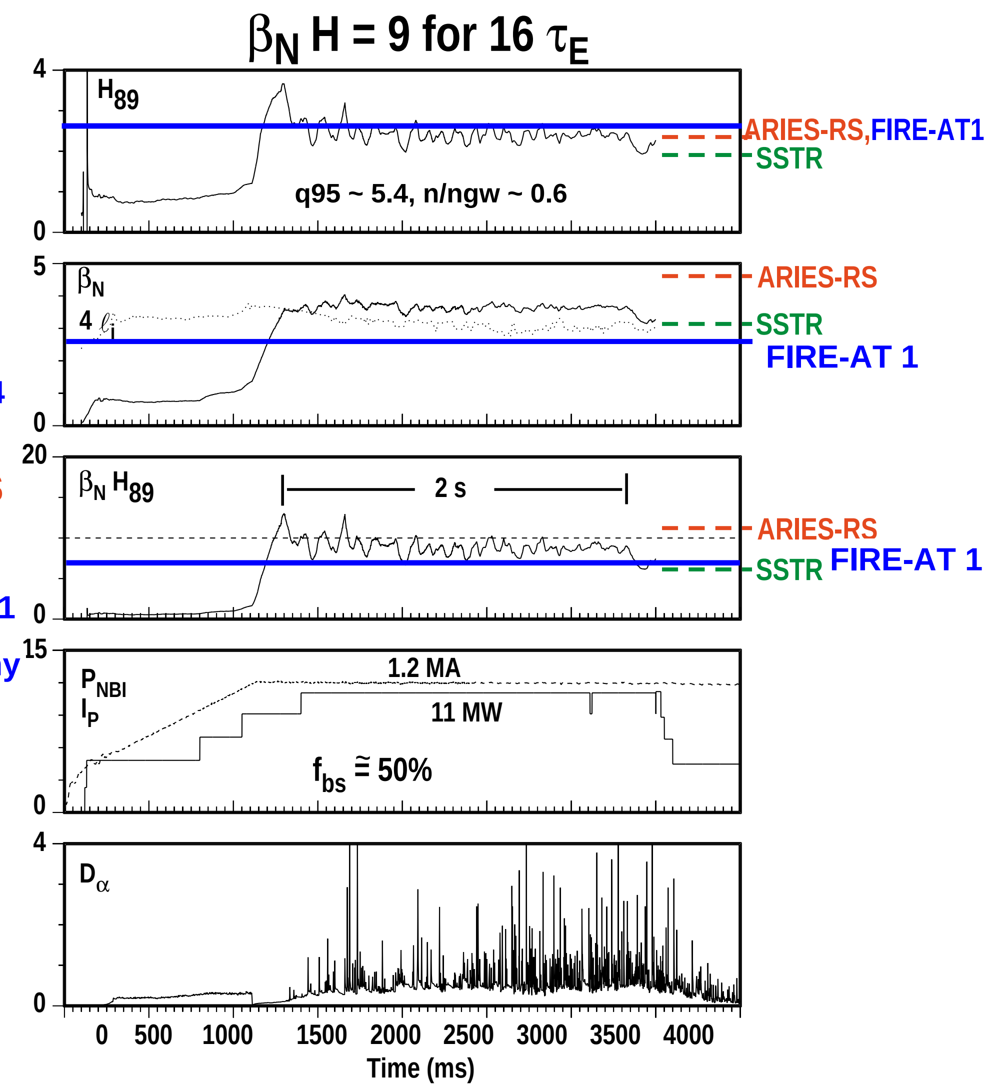
Duration

~ 4 tau_skin

The transport calculations assumed 150 MW of fusion power and n(0)/<n> = 1.5.

Long-Pulse Advanced Tokamak Performance Achieved in DIII-D Leads to Interesting High-Gain Advanced Burning Plasma Experiments





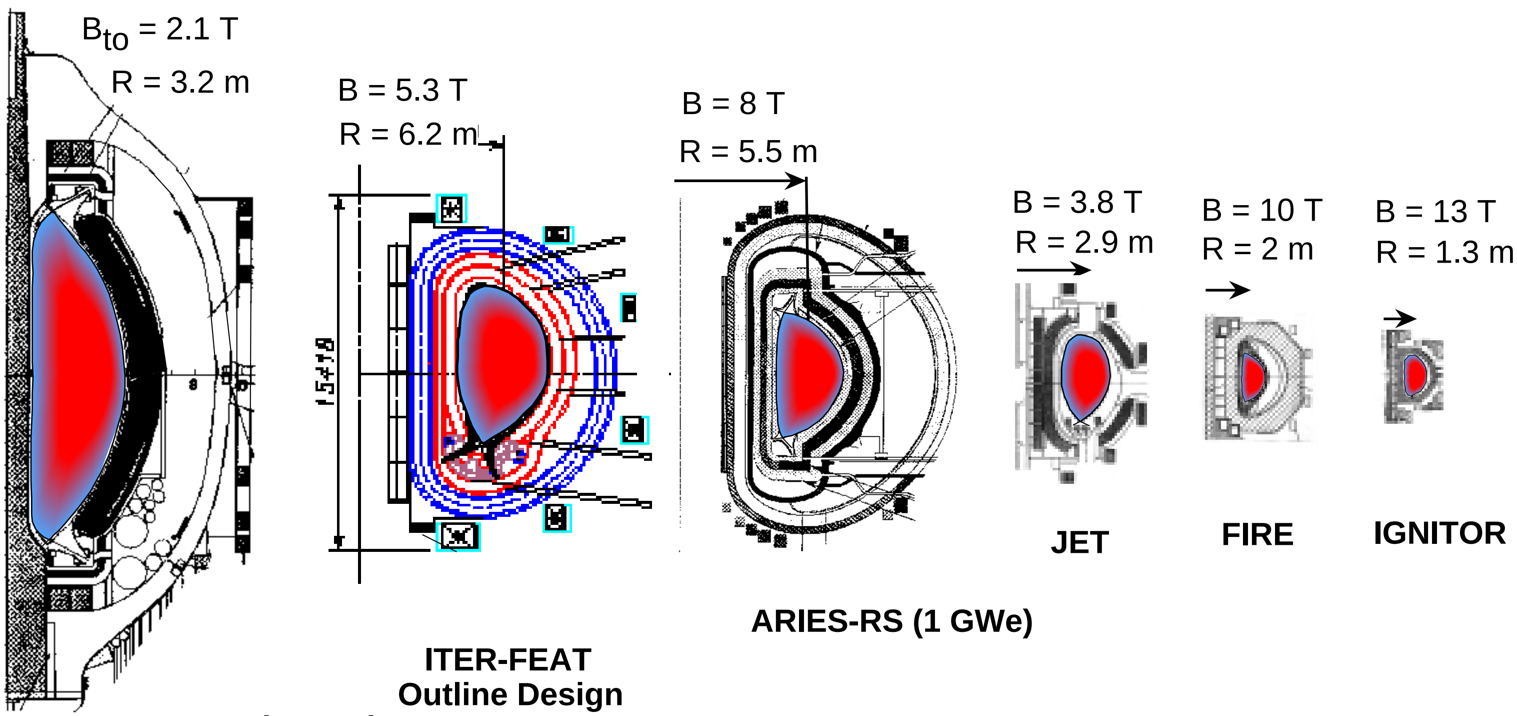
FIRE-Elmy is conventional Elmy H-Mode
FIRE-AT 1 is modest AT with 50% fbs and β_N = 2.6
FIRE-AT 4 is strong AT with 82% fbs and β_N = 4.5

DIII–D NATIONAL FUSION FACILITY SAN DIEGO

DIII-D shot 98977 is close to a Demonstration Discharge for FIRE-AT 1
FIRE-AT 1 requires q95 = 4.5, n/ngw = 0.65, β_N H89 = 7.1, and produces fbs = 50% and Q = 10 (Pfusion =150 MW, Pin = 15 MW). This mode would be useful for quasi-steady experiments ~ 2 skin times.

264–99

Potential Next Step Burning Plasma Experiments and Demonstrations in MFE



Cost Drivers	ARIES-ST	ITER-FEAT	ARIES-RS	JET	FIRE	IGNITOR
Plasma Volume (m^3)	810	837	350	95	18	11
Plasma Surface (m^2)	580	678	440	150	60	36
Plasma Current (MA)	28	15	11	4	6.5	12
Magnet Energy (GJ)	29	50	85	2	5	5
Fusion Power (MW)	3000	500	2200	16	200	100
Burn Time (s), inductive	steady	300	steady*	1	20	5

* assumes non-inductive current drive

FIRE Power Requirements for BeCu or CuTF Coils

	10T (20s flattop)		12T (12s flattop)	
BeCu	Peak Power (MW)	Peak Energy (GJ)	Peak Power (MW)	Peak Energy (GJ)
TF	490	11.5	815	11.5
PF	250	2.2	360	3.7
RF	60	1	60	0.6
$\sum$	800	14.7	1235	15.8
Grid	550 (TF&RF)	12.5	600 (TFbase)	10.9
MG	250 (PF)	2.2	635 (TFsupp&PF&RF)	4.9

	10T (45s flattop)		12T (25s flattop)	
Cu	Peak Power (MW)	Peak Energy (GJ)	Peak Power (MW)	Peak Energy (GJ)
TF	267	12.6	345	13.2
PF	250	5	360	4.6
RF	60	2.3	60	1.3
$\sum$	577	19.9	765	19.1
Grid	577 (All Systems)	19.9	404 (TF&RF)	14.5
MG	0	0	360 (PF)	4.6

Preliminary FIRE Cost Estimate (FY99 US$M)

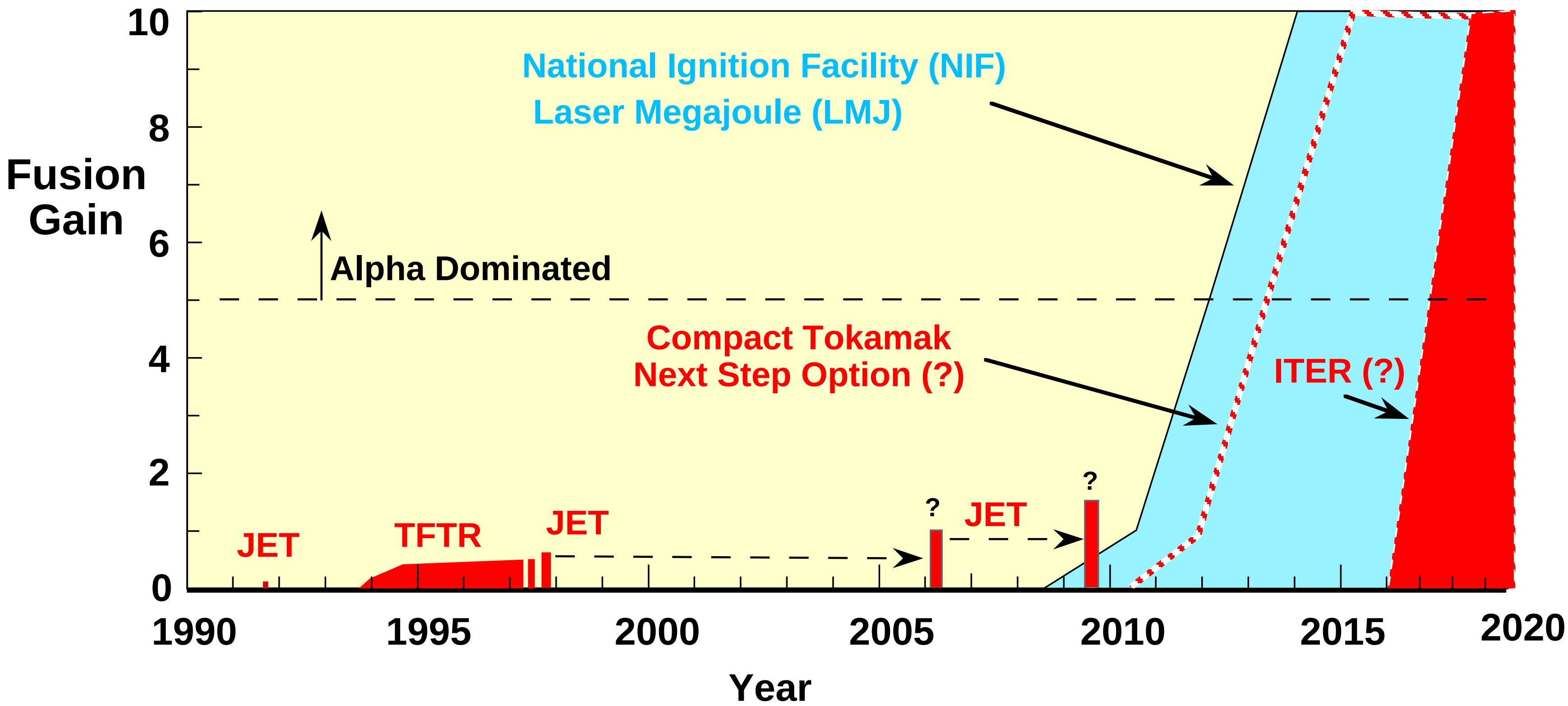
	Estimated Cost		Contingency		Total with Contingency
1.0 Tokamak Core	210.2		66.0		276.2
1.1 Plasma Facing Components		44.8		13.5	
1.2 Vacuum Vessel/In-Vessel Structures		34.6		10.9	
1.3 TF Magnets /Structure		103.7		34.8	
1.4 PF Magnets/Structure		13.0		2.6	
1.5 Cryostat		1.8		0.5	
1.6 Support Structure		12.3		3.7	
2.0 Auxiliary Systems	147.5		46.1		193.6
2.1 Gas and Pellet Injection		7.1		1.4	
2.2 Vacuum Pumping System		13.0		2.0	
2.3 Fuel Recovery/Processing(Rough Estimate)		20.0		10.0	
2.4 ICRF Heating		107.4		32.7	
3.0 Diagnostics (Startup)	18.4		12.2		30.6
4.0 Power Systems	149.4		37.4		186.8
5.0 Instrumentation and Controls	18.9		2.5		21.4
6.0 Site and Facilities	172.2		40.8		213.0
7.0 Machine Assembly and Remote Maintenance	70.7		18.0		88.7
8.0 Project Support and Oversight	107.6		16.2		123.8
9.0 Preparation for Operations/Spares	16.2		2.4		18.6
Preconceptual Cost Estimate (FY99 US$M)	**911.1**		**241.6**		**1152.7**

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

This estimate is work in progress and will be finalized in August 2000.

June 1, 2000

Timetable for Burning Plasma Experiments



- Even with ITER, the magnetic fusion program will be unable to address the alpha-dominated burning plasma issues for ≥ 15 years.
- Compact High-Field Tokamak Burning Plasma Experiment(s) would be a natural extension of the ongoing "advanced" tokamak program and could begin alpha-dominated experiments by ~ 10 years.
- More than one high gain burning plasma facility is needed in the world program.
- The information "exists now" to make a technical assessment, and decision on a magnetic fusion burning plasma experiment(s) for the next decade.

Critical Issues for FIRE and Magnetic Fusion

The critical physics and engineering issues for FIRE are the same as those for fusion, the goal of FIRE is to help resolve these issues for magnetic fusion. The issues and questions listed below need to be addressed in the near future.

- Physics
  - confinement - H-mode threshold, edge pedestal, enhanced H-mode, AT-modes
  - stability - NTMs, RWM, disruptions: conducting wall? feedback coils? VDE(DN)?
  - heating and current drive - ICRF is baseline: NBI & LHCD as upgrades?
  - boundary - detached divertor operation, impurity levels, confinement
  - self-heating - fast alpha physics and profile effects of alpha heating

  Development of self-consistent self-heated AT modes with external controls

- Engineering
  - divertor and first wall power handling (normal operation and disruptions)
  - divertor, first wall and vacuum vessel for long pulse AT modes
  - evaluate low inventory tritium handling scenarios, higher fluence TF insulator
  - complete many engineering details identified in FIRE Engineering Report
  - evaluate potential sites for Next Step MFE experiment
  - complete cost estimate for baseline, identify areas for cost reduction

Major Conclusions of the FIRE Design Study

- Exploration, understanding and optimization of alpha-dominated (high-gain) burning plasmas are critical issues for all approaches to fusion.
- The tokamak is a cost-effective vehicle to investigate alpha-dominated fusion plasma physics and its coupling to advanced toroidal physics for MFE. The tokamak is technically ready for a next step to explore fusion plasma physics.
- The FIRE compact high field tokamak can address the important alpha-dominated plasma issues, many of the long pulse advanced tokamak issues and begin the integration of alpha-dominated plasmas with advanced toroidal physics in a $1B class facility.
- The FIRE design point has been chosen to be a “stepping stone” between the physics accessible with present experiments and the physics required for the ARIES vision of magnetic fusion energy.
- A plan is being developed for an Advanced Tokamak Next Step that will address physics, engineering and cost issues in FY 2000-1 with the goal of being ready to begin a Conceptual Design in 2002.

http://fire.pppl.gov