The Path to Magnetic Fusion Energy: Crossing the Next Frontier

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with contributions from

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FESAC is now looking at future opportunities for U.S. fusion research. This presentation proposes a mission for a major new U.S. facility, leading to a substantial new thrust in U.S. fusion research – crossing the next frontier towards practical fusion energy. We view this frontier to be at the plasma boundary: indeed with the demonstration of scientific break-even behind us, it is now time to address the logically first of the combined physics and technology challenges for fusion – the plasma-surface interface. Indeed one cannot step confidently on to the next challenges without resolving this one. The proposed mission of a new facility would be to *"To integrate a fusion-relevant plasma-material interface with sustained high-performance plasma operation."* If FESAC selects this mission area to be explored further, there will likely be many community meetings to focus the mission more specifically and to settle on the proper facility design to address the mission. The specific implementation we are presenting here should be viewed as an existence proof rather than a proposal.

The Orbach Challenge

"And so it would seem to me that the fusion community would want to build in something out there, so that when the ITER funding rolled off there was something else coming on, and so on."

Dr. Orbach spoke at PPPL in April of 2006, just before he gave a similar talk at FESAC. He challenged the U.S. fusion community to come up with an exciting and important proposal that would be competitive with other areas of DOE Science in attracting funding.

The Challenge

How can the U.S. be a leader in fusion, using increased domestic resources in parallel with ITER construction and operation?

It is reasonable to think of two phases:

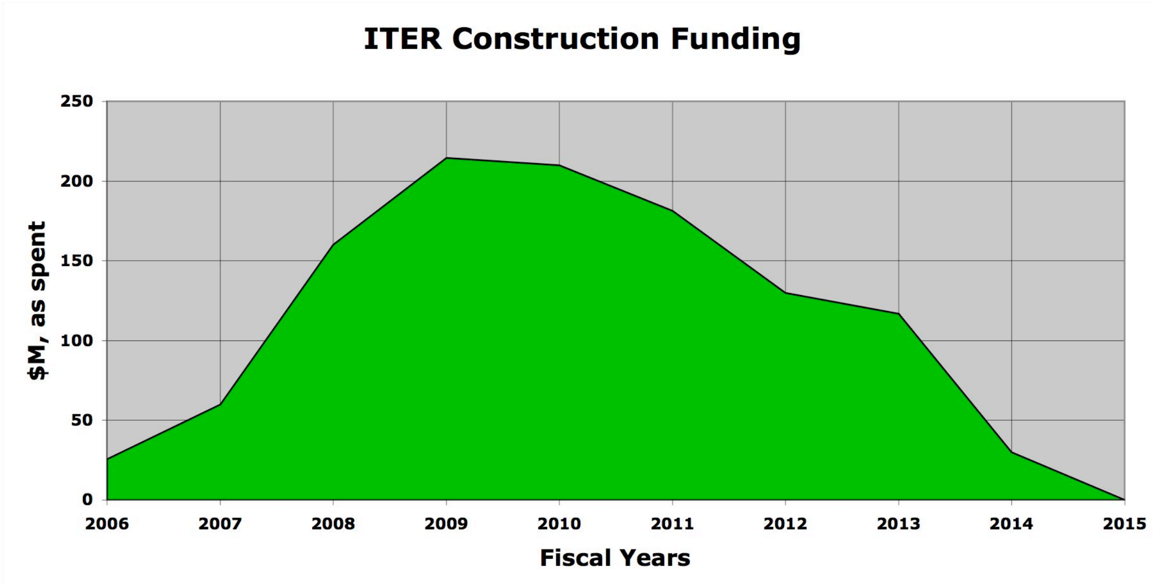
1. The ITER Construction "Roll Off" — **Focus of this talk**

2. The ITER High-Gain "Pay Off"

The underlying challenge is for the U.S. to take a unique leadership role again in fusion R&D, by using increased domestic resources in parallel with the latter phase of ITER construction and during ITER operations.

It seems reasonable to consider that we could target the "roll-off" from ITER construction funding starting in the ~2010-11 time frame for a significant initiative, and then consider that when ITER achieves high fusion gain, greater funds might be made available. This talk focuses on the nearer-term opportunity presented by the ITER construction "roll-off."

The ITER Roll-off Provides an Opportunity



The ITER construction roll-off provides an opportunity for the U.S. fusion community to propose enhancements to domestic research.

While the next Administration may not carry forward the "American Competitiveness Initiative" as it is defined today, it is likely that funding for ITER construction will peak and then roll off, and we should be prepared to take advantage of this opportunity. It is worth noting that the MFE portion of the 2003 FESAC Fusion Development Plan averages about $600M/year. The peak total fusion funding anticipated in 2010 of about $500M/year approaches the value needed to carry out that plan.

The U.S. will be a 13% player in the operation of ITER. The negotiated ITER agreement states that experiments will be selected based on scientific merit, taking into account the contributions of the participants.

Around the world ITER participants are building and operating major long-pulse confinement facilities to complement ITER, and to position themselves to take advantage of ITER for their next steps in fusion development. What should the U.S. do to take a unique leadership role in this context?

The U.S. can have a Unique Leadership Role by Integrating the Four NAS Plasma Science Areas

- **ITER will solidly address the issue of confinement and fusion gain.**
  - **Indeed this is ITER's signature scientific goal.**
  - **Extrapolation from Q = 10 to 25 should be possible based on scientific understanding and advanced simulation capability.**
- **The planned world program will solidly address stability.**
  - **Planned steady state superconducting tokamaks will solidly address this issue: ITER, EAST, JT-60SA, KSTAR, SST-1**
  - **Stellarators address stability and disruptions: LHD, NCSX, W7-X**
- **The planned world program will solidly address sustainment.**
  - **Planned steady state superconducting tokamaks will solidly address this issue: ITER, EAST, JT-60SA, KSTAR, SST-1**
  - **Stellarators run continuously without drive: LHD, NCSX, W7-X**
- **The planned world program does not address boundary physics.**
  - **The gap from ITER to Demo is a factor of four.**
  - **EAST, JT-60SA, KSTAR, LHD, SST-1, W7-X do not press this issue even as far as ITER.**

The NAS defined four key areas of fusion plasma science: confinement, stability, sustainment and boundary physics. Three of these four science areas are well covered by planned experiments world-wide, and it can be argued that adequate information should be available for Demo in these areas of fusion science. However the area of boundary physics and related plasma technologies is not being addressed at anywhere near the level required to make the step to Demo.

Measured Power Scrape-Off Width Does not Vary with Machine Size

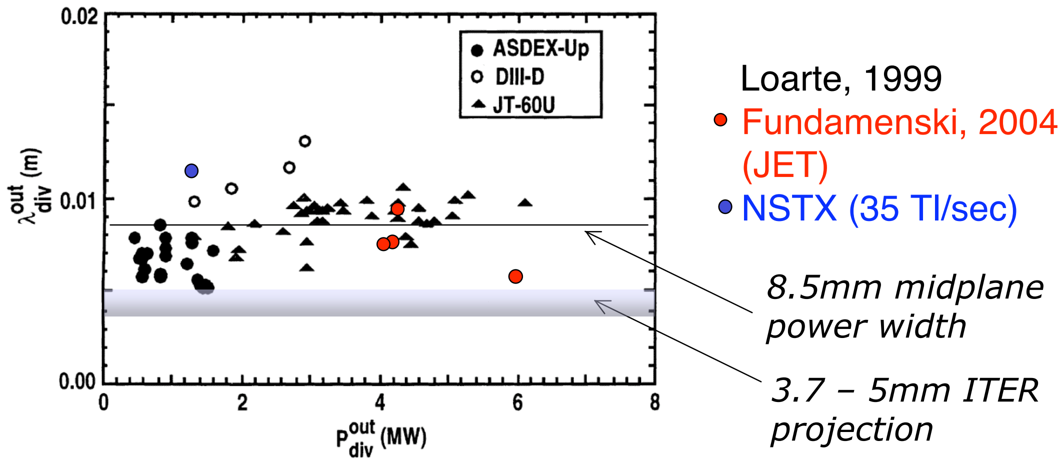


Fig. 5. Measured power deposition width versus divertor power for H-mode discharges without gas puff in the ITER power deposition database. (Mapped from strike point to outer mid-plane.)

The divertor heat-flux challenge ~ P_{in}/R
First wall heat-flux challenge ~ P_{in}/S

The power scrape-off layer width in tokamaks is not very well characterized by experiments or theory, but the data that have been collected and used for projecting to ITER (Loarte, Fundamenski) indicate that the power scrape-off layer width, when mapped to the outer mid-plane, is not strongly dependent on the size of a machine. The NSTX divertor data is likely somewhat broadened due to the significant gas puff rate. Due to evidence from JET that the power scrape-off width scales unfavorably with P_{out}, the power delivered to the outer divertor plate, the ITER Power and Particle Control document published in Nuclear Fusion (2007) projects 3.7 – 5mm for ITER. The challenge presented by the plasma to the divertor can be roughly characterized by the input power, $P_{in} = P_{aux} + P_{alpha}$, divided by the major radius, R[1]. One aspect of the scientific challenge in the divertor region is to find ways to radiate or spread this heat injected into the plasma before it reaches the divertor plate, without degrading plasma performance.

The heat flux challenge to the first wall can be characterized by the input power, P_{in}, divided by the plasma surface area, S. This represents a qualitatively different challenge, since the first wall stands in front of the main breeding blanket, and so cannot be nearly as massive as a divertor. The heat flux due to neutron bombardment, which peaks at the outer midplane, will add to this challenge.

[1]Kallenbach has analyzed multi-machine electron temperature profile data indicating that the edge/SOL electron temperature gradient scale length varies approximately linearly with machine size. However the power scrape-off width deduced from the simple 2-point model of electron heat flow, $\lambda_{div} = (2/7)\,\lambda_{Te}$, fails to fit experimental measurements of divertor plate heat flux by large factors and is even inconsistent with comparisons of λ_{Te} and λ_{div} in 2-D ITER divertor simulations. While this result is of considerable scientific interest, it does not bear directly on heat flux localization at the divertor plate.

Divertor Heat Flux is a Very Serious Issue for CTF/FDF and Demo

	CTF / FDF	Demo
P_{in} / R	45 MW/m	100 MW/m
2π * 8.5mm (conservative)	/ 0.053m	/ 0.053m
Double null, 15% up-dn asymmetry	x 0.575	x 0.575
Toroidal asymmetry	x 1.2	x 1.2
Flux expansion, including plate tilt	/ 10	/ 10
Peak heat flux with no radiation	60 MW/m^2 ~ 2 weeks	130 MW/m^2 months

One can estimate the peak heat flux following the method of Loarte et al. in the ITER Power and Particle Control document, Nuclear Fusion, 2007. Here we assume an outer midplane power scrape-off layer width of 8.5mm. (If we were to use an average of the published estimates for ITER, 3.7mm and 5mm, the peak heat flux would be increased by a factor of two.) We assume that double-null operation cannot be perfectly balanced, and take into account the effects of toroidal asymmetry due to the need to tilt and offset divertor segments in order to protect leading edges. The flux expansion, including a factor for plate tilt, is that quoted by Loarte et al. for ITER. The resulting heat flux, assuming that 100% of the input power ($P_{in} = P_{aux} + P_{alpha}$) is deposited on the divertor plates rather than radiated, is evidently very high. To handle this for the very long pulses anticipated in a Component Test Facility / Fusion Development Facility and Demo will require significant advances in both boundary plasma physics, to reduce the localized heat flux, and in power and particle handling technologies, integrated with high performance plasma operation.

Planned World Long-Pulse Experiments Focus on Confinement, Stability and Sustainment, not Boundary.

Device	R (m)	a (m)	P_{in} (MW)	P_{in}/R (MW/m)	P_{in}/S (MW/m^2)	Pulse (sec)	I_p (MA)	Species	Comments
Planned Long-Pulse Experiments									
EAST	1.70	0.40	24	14	0.55	1000	1.0	H (D)	Upgrade capability
JT-60SA	3.01	1.14	41	14	0.21	100	3.0	D	JA-EU Collaboration
KSTAR	1.80	0.50	29	16	0.52	300	2.0	H (D)	Upgrade Capability
LHD	3.90	0.60	10	3	0.11	10,000	–	H (D)	Upgrade capability
SST-1	1.10	0.20	3	3	0.23	1000	0.2	H (D)	Initial heating
W7-X	5.50	0.53	10	2	0.09	1800	–	H	30MW for 10sec
ITER	6.20	2.00	150	24	0.21	400-3000	15.0	DT	Not for divertor testing
Component Test Facility Designs									
CTF (A=1.5)	1.20	0.80	58	48	0.64	~2 Weeks	12.3	DT	2 MW/m^2 neutron flux
FDF (A=3.5)	2.49	0.71	108	43	0.87	~2 Weeks	7.0	DT	2 MW/m^2 neutron flux
Demonstration Power Plant Designs									
ARIES-RS	5.52	1.38	514	93	1.23	Months	11.3	DT	US Advanced Tokamak
ARIES-AT	5.20	1.30	387	74	0.85	Months	12.8	DT	US Advanced Technology
ARIES-ST	3.20	2.00	624	195	0.99	Months	29.0	DT	US Spherical Torus
ARIES-CS	7.75	1.70	471	61	0.91	Months	3.2	DT	US Compact Stellarator
ITER-like	6.20	2.00	600	97	0.84	Months	15.0	DT	ITER @ higher power, Q
EU A	9.55	3.18	1246	130	0.74	Months	30.0	DT	EU "modest extrapolation"
EU B	8.60	2.87	990	115	0.73	Months	28.0	DT	EU
EU C	7.50	2.50	794	106	0.71	Months	20.1	DT	EU
EU D	6.10	2.03	577	95	0.78	Months	14.1	DT	EU Advanced
SlimCS	5.50	2.12	650	118	0.90	Months	16.7	DT	JA

The boundary is the most serious issue for ITER, CTF/FDF and Demo.

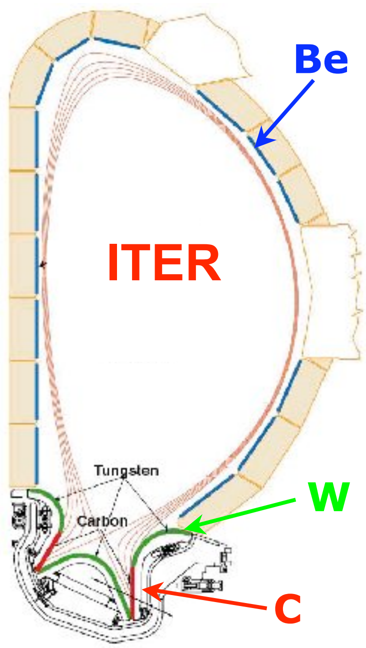
Since a fusion power plant about the same size as ITER must produce roughly five times as much fusion power as ITER, at Q ~25 rather than 10, it will challenge both its divertor and first wall ~4x more than ITER. This can be seen in the U.S., European and Japanese power plant designs. A Component Test Facility or Fusion Development Facility that delivers 2 MW/m^2 of neutrons to its blankets must handle about twice the divertor challenge of ITER, at dramatically greater pulse length and duty factor. For perspective, the FDF design point is equivalent to injecting 130 MW into JET, for 2-week pulses. The JET divertor plate currently reaches 14 MW/m^2 with injected power of 16 MW. Among the planned long-pulse experiments only JT-60SA is capable of extensive high-power operation in deuterium, critical for reactor-relevant plasma boundary characteristics: H-mode confinement and ELMs. Its divertor challenge is about half that of ITER and 7x less than that of a fusion power plant. We note that ITER is not designed for extensive diagnosis and frequent reconfiguration of its divertor and first wall.

The Plasma-Material Interface is a Critical Issue for ITER and Beyond

- **The currently planned approach is to start ITER with carbon divertor surfaces to avoid melting at ELMs and disruptions, but tritium retention with carbon is a very serious concern for ITER and a show-stopper for CTF/FDF and Demo.**
- **The currently planned approach is to start ITER with a Be first wall, to limit tritium retention and high-Z contamination. Be is not consistent with DEMO heat and particle loads, first wall temperature, and T/Be codeposition concerns.**
- **Carbon cannot work for CTF/FDF or Demo anyway, due to erosion and neutron damage. Tungsten presents its own challenges, including high-Z contamination and melting during ELMs and disruptions.**
- **There is no proven solution for CTF/FDF and Demo – scientific innovation is critical.**

ITER provides a test case for the seriousness of the plasma-material interface problem. The solutions chosen at present are highly contentious, with strong debate as to whether they will work even for ITER. Such approaches appropriate for a relatively short-pulse, low duty-factor device cannot be extrapolated to the much more demanding conditions of a Component Test Facility / Fusion Development Facility or Demo. New approaches must be developed and validated in long-pulse plasmas with a Demo-relevant plasma-material interface.

The First Wall and Divertor in JET are Being Replaced because of ITER Boundary Issues



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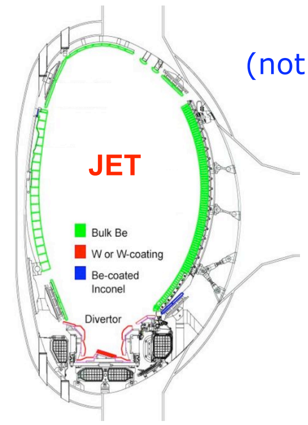


ITER's low temperature Be first wall does not extrapolate to CTF/FDF or Demo.

No proven solution is available for CTF/FDF and Demo.

JET's first wall and divertor are being rebuilt to provide data to inform ITER's design choices, but this data will come late and be very limited. We should not find ourselves in this position again when it comes to selecting a first wall and divertor approach for a Component Test Facility / Fusion Development Facility or Demo.

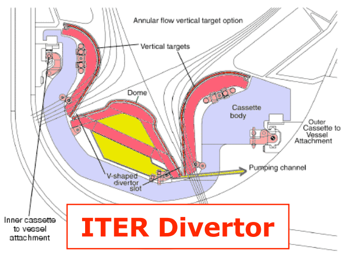
The Integrated Fusion Science Mission of NHTX

National High-power advanced Torus eXperiment

To integrate a fusion-relevant plasma-material interface with sustained high-performance plasma operation.

NHTX must have the PF and PFC flexibility & diagnostic access to study:

- **Multiple divertor geometries.**
- **Stellarator-like edge magnetic field.**
- **Magnetically expanded strike zone.**
- **Radiative edge zone.**
- **Multiple advanced solid materials.**
- **Liquid surfaces: lithium, gallium, tin.**
- **Multiple plasma heating technologies.**
- **Other innovations.**
- **INTEGRATED WITH A HIGH-PERFORMANCE PLASMA**



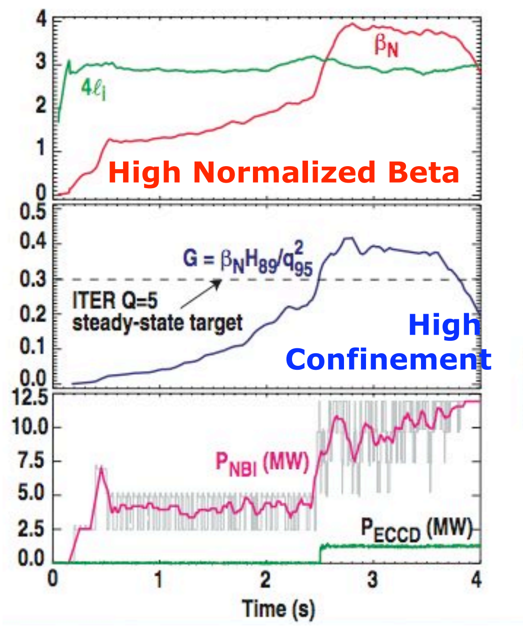
ITER Divertor

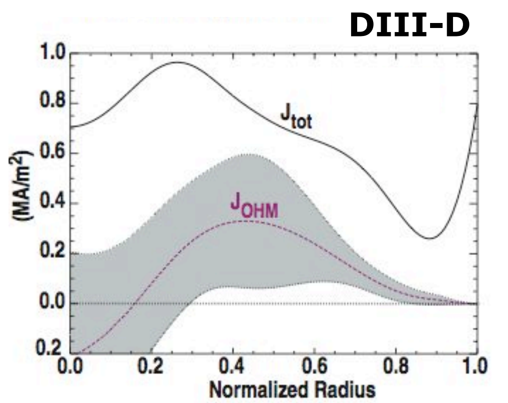
Such a device would:

Leapfrog the world **in integrated core and boundary science for later phases of ITER, for CTF/FDF, and for a Demo power plant – whether Tokamak, ST or Compact Stellarator.**

We are not committed to a particular machine name, nor acronym, but the idea of integrating a fusion-relevant plasma-material interface with a high-performance plasma is at the heart of the proposed mission. It is critical that a device with this mission should have excellent access for rapid change-out of components and for extensive and detailed diagnostic coverage of all aspects of placing-facing component engineering performance and of plasma-surface interactions, as well as a full set of advanced operating mode plasma measurements.

The Integrated Fusion Science Mission Requires Many High-Performance Plasma Features Simultaneously – in Long Pulse

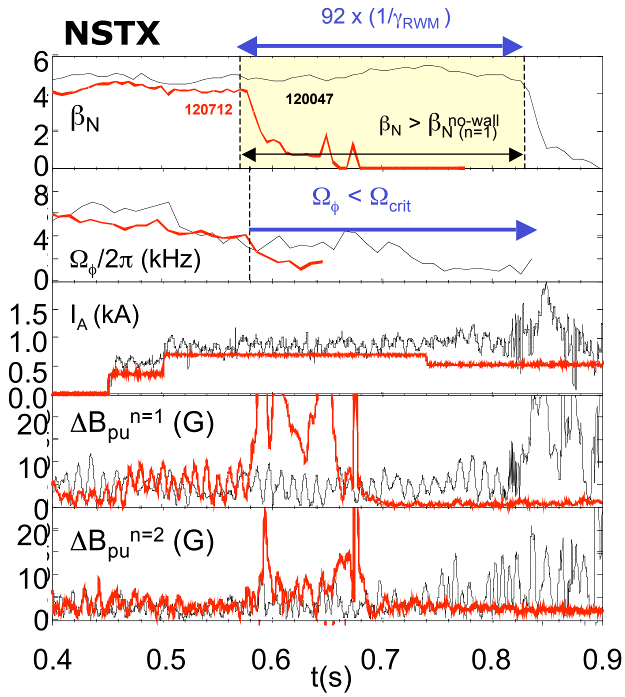




High Non-inductive Fraction

Current experiments are achieving the range of parameters that would be appropriate for the operation of NHTX. However these parameters need to be achieved for ~100x longer pulses. NHTX will also be equipped to push further in beta, confinement and bootstrap fraction, but a high level of plasma performance is not required to meet the basic mission of the device.

High Performance NHTX Plasmas will Support Development of Advanced Regimes for Demo



Sabbagh, et al., PRL 97 (2006) 045004.

Access, flexibility and pulse count allow Demo studies:

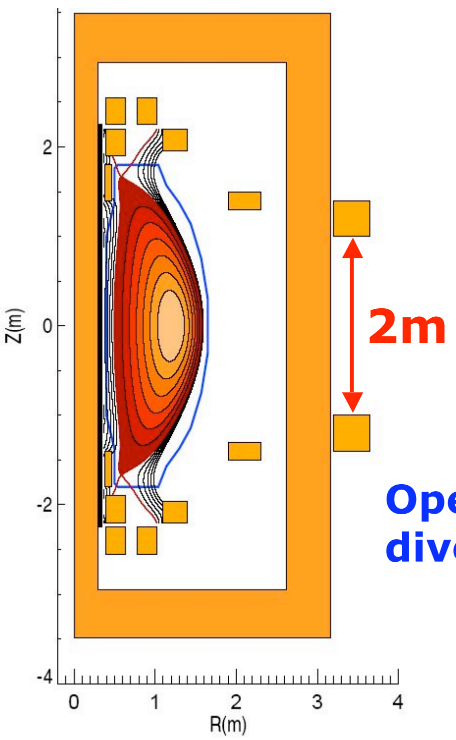
- High beta
  - e.g., RWM
- Confinement
  - e.g., turbulence
- Sustainment
  - e.g., current drive
- ELM & disruption avoidance

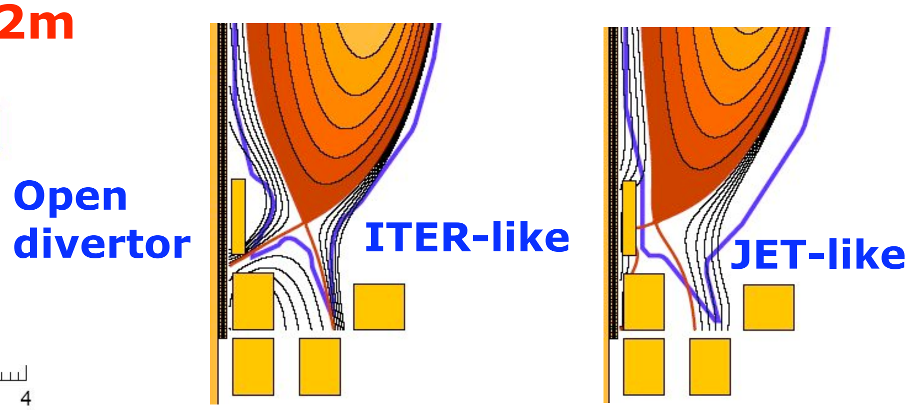
Demonstrate reliable disruption prediction, avoidance and mitigation to meet Demo spec of < 1 serious disruption per year.

NHTX will provide a test-bed to demonstrate long-pulse operation at the highest plasma parameters, and hopefully to develop techniques to predict and control disruptions with adequate reliability for a Component Test Facility / Fusion Development Facility or Demo. Note that the disruptions on NHTX will not be as destructive as those on Demo, or even ITER, if measured in terms of stored energy / surface area. On the one hand this implies that it is not necessary for NHTX to eliminate disruptions in order to address its mission. On the other, NHTX (and ITER) operation will likely need to be supplemented with additional test stand studies to extrapolate to Demo-level disruptions, if disruption-free operation cannot be assured.

Access and Divertor Flexibility are Critical for the NHTX Mission

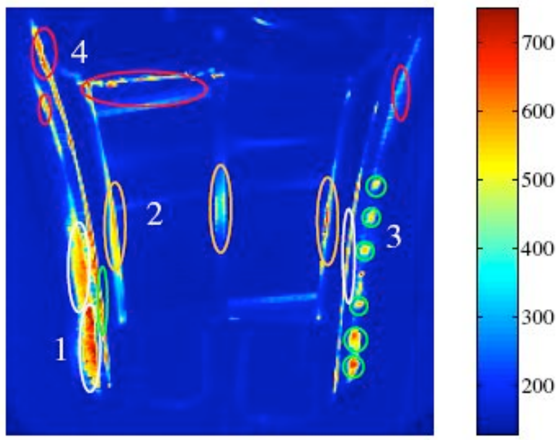
- Excellent access for heating, diagnostics & maintenance.
- Highly flexible divertor configuration.
- Demountable Cu coils for major component change-out.





A relatively low-aspect-ratio design provides excellent radial and tangential access for extensive plasma diagnostics. It also allows very large mid-plane ports to change out components, and vertical access using the demountable coils, to change out major systems. A segmented divertor coil system combined with a two-layer vertical field system provides excellent flexibility to control the plasma shape, with special focus on the divertor region. As divertor designs are further developed, divertor heating, cooling and pumping will be integrated with the PF design.

Diagnostic and Heating System Access are Critical for the NHTX Mission



Tore Supra, France
ICRF antenna

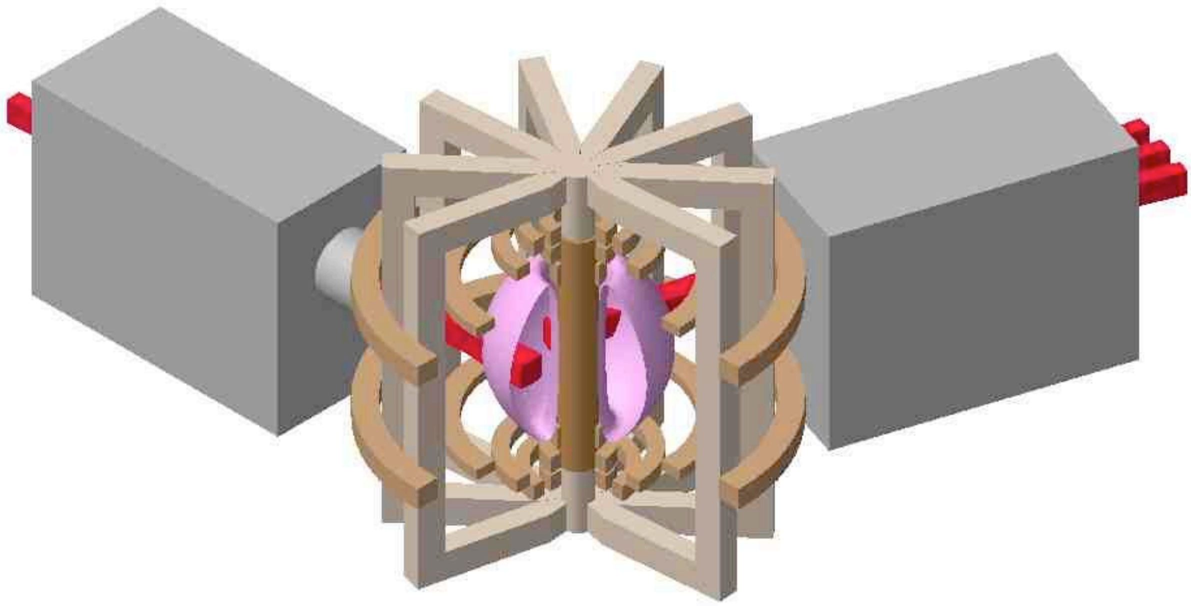
Figure 10. IR image of antenna Q1 on shot TS33748 at $t = 63.7$ s. Unit is °C. Superimposed on the image, a selection of zones on the front faces, classified according to their sensitivity to different sources of additional power are: zone 1 (white): mainly sensitive to the total power, zone 2 (orange): mixed total ICRF power and private ICRF power, zone 3 (green): sentitive to LH power only and zone 4 (red): predominantly private ICRF power.

- **Extensive view in toroidal and poloidal angle of all plasma-material interactions.**
- **Extensive in-situ surface analysis capabilities.**
- **Extensive PFC engineering performance measurements.**
- **A full set of advanced confinement, stability and sustainment plasma diagnostics.**

Modern tokamak devices have extensive plasma diagnostic systems. In NHTX it will be necessary, *in addition*, to measure the performance of steady-state plasma facing components and to diagnose plasma-material interactions at essentially all locations around the torus. This will require extensive IR measurements and the application of lab surface measurement techniques, with new approaches to allow wide area coverage, and both real-time and post-shot in-situ analysis.

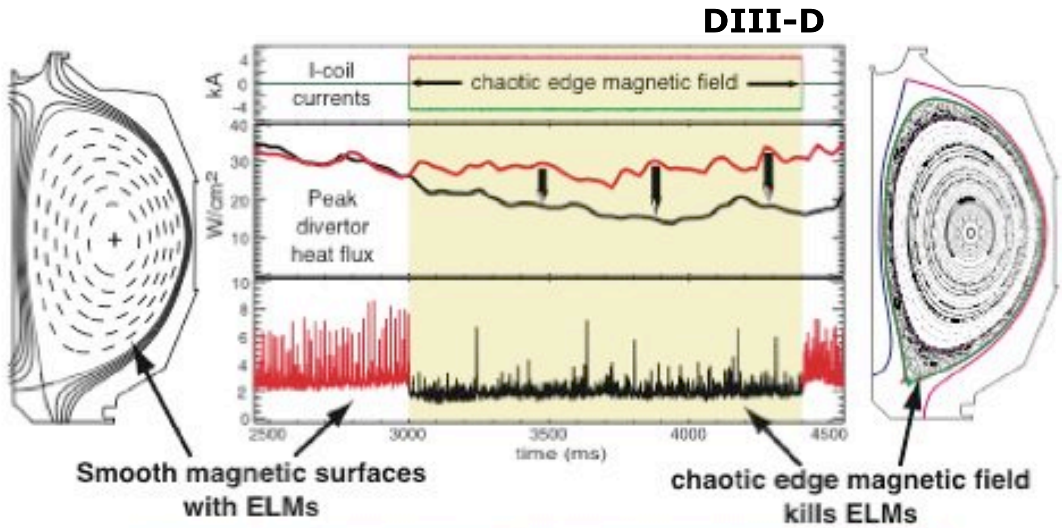
The diagnostic access in ITER is not adequate for this purpose. CTF/FDF with 3x higher neutron flux, 20x higher neutron fluence and the requirement at an early date to fill the mid-plane ports with blanket test modules, will have a great deal less diagnostic access even than ITER.

NHTX Coil Set Allows Excellent Access to the Plasma



A relatively low-aspect-ratio design provides excellent radial and tangential access for extensive PFC engineering performance monitoring, surface analysis, and plasma physics diagnosis. It also allows very large mid-plane ports to change out components, and vertical access using the demountable water-cooled copper coils, to change out major systems. A segmented divertor coil system combined with a two-layer vertical field system provides excellent flexibility to control the plasma shape, with special focus on the divertor region.

Stellarator-like Edge Needs to Be Investigated in Long Pulse

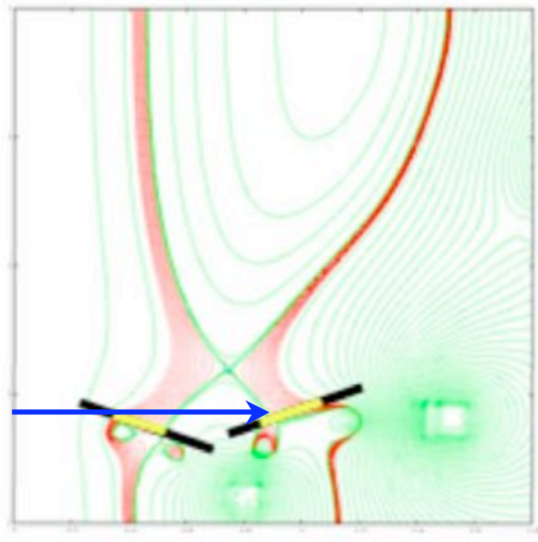


Results from DIII-D, JET, NSTX suggest that magnetic islands, ergodicity can reduce Edge Localized Modes.

How do we optimize this configuration for very long pulses?

Islands and ergodic magnetic fields can suppress Edge Localized Modes, and also split up the divertor strike point, reducing the peak heat flux. The implications of a configuration of this kind for where all of the heat goes will need to be studied in long pulses, as will approaches to pumping such a configuration. The NHTX design for this concept should be configured to provide maximum information on divertor optimization for compact stellarators as well as for tokamaks.

Expanded Magnetic Flux can Spread Heat Load, Enhance SOL Radiation

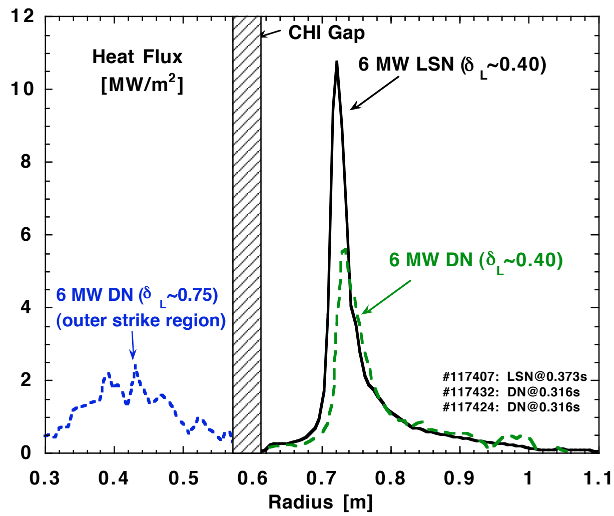


In addition, compressed flux and reduced R_x can increase heat loads for testing purposes.

The U. Texas group has been exploring the idea of spreading divertor heat flux using a poloidal magnetic field system which causes the poloidal flux to expand as it approaches the divertor plate. This can be accomplished with the NHTX segmented divertor coils. The greater scrape-off-layer volume in such an "X-Divertor" should permit enhanced radiated power in addition to the geometrical effect of expanding the flux tubes carrying power.

The flexible NHTX coil set also permits compressing the poloidal flux and reducing the major radius of the strike point to increase the peak heat load.

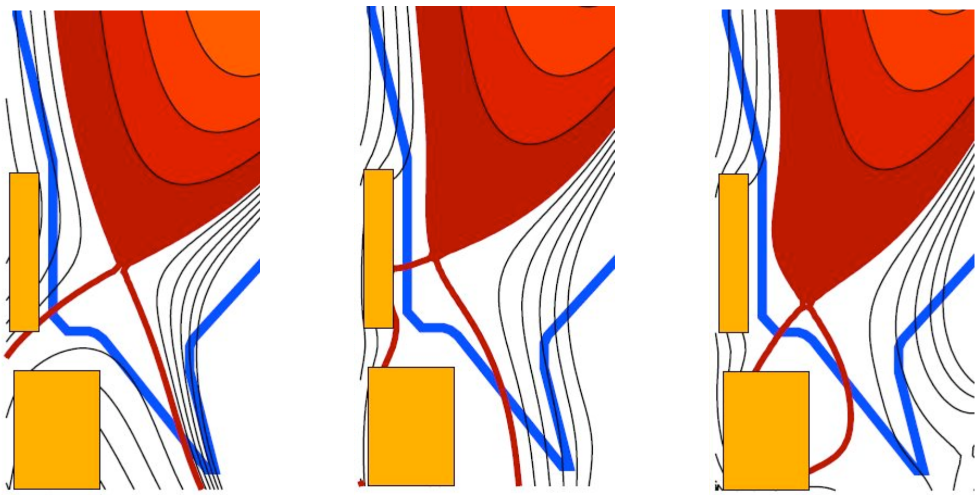
NSTX Has Shown Dramatic Impact of Flux Expansion



- **Low-A has dramatic SN/DN difference.**
- **Low-A gets strong effect from varying R_x.**
- **Flux expansion has a dramatic effect.**
- **What are the limits to this approach in long pulse?**

NSTX has shown that peak divertor heat flux can be manipulated in a number of ways. At low A there is a very strong difference between single null and double-null operation. Single-null plasmas in NSTX reach ITER's peak allowable steady divertor heat flux. Higher-triangularity plasmas have much lower strike point major radius, R_x, for higher P/R_x. The figure above shows that flux expansion at the outer strike point in a highly triangular plasma dramatically reduces the peak heat flux. It will be necessary to understand the limits to this approach in long pulse, such as the efficiency of pumping a flux-expanded divertor strike point, and the requirements for careful divertor tile alignment.

NHTX is Very Flexible with Respect to Flux Expansion



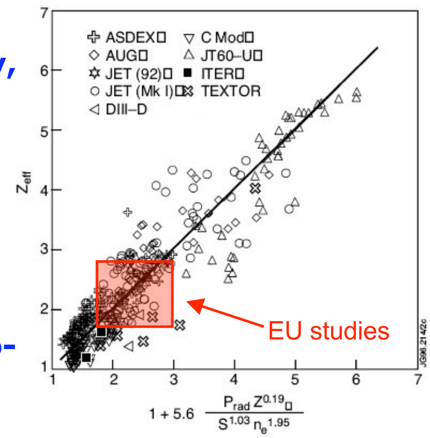
x 7.5 x 23 x 40

Heat flux expansion from midplane

This figure shows a wide range of flux expansion that can be achieved in NHTX, in this case using a tilted divertor plate in JET-like geometry.

High P_{in} / P_{LH} is Needed to Test Radiative Solution

- **Can fusion plasmas operate at high performance, and without thermal instability, with up to 90% radiated power to reduce divertor heat flux?**
- **Physics test requires input power exceeding H-mode threshold power by a very large factor, up to x10, if most radiated power comes from the plasma core.**
- **NHTX has unique capability to test the Demo-relevant physics in this area:**



P_{in}/P_{LH}@0.85*n_g

– **JT-60SA**	**4.9**
– **NHTX**	**12**
– **ITER**	**3.6**
– **ARIES-AT**	**11**

EU-B:
Z_{eff} = 2.7
n/n_g = 1.2
H_H = 1.2
R_0 = 8.6m
I_p = 28MA

Recent European reactor designs have leaned heavily on protecting the divertor through very high radiated power fraction, using moderate-Z impurities. The high Z_{eff} that is required in the EU power-plant designs according to computations, consistent with the very rough experimental scaling shown in the figure, results in significant fuel dilution and very strong demands on the density operating point and size / plasma current. (Some EU designs move to lower Z_{eff} but increase n/n_g to 1.5 to achieve adequate radiated power.) This reflects the seriousness which the EU group attributes to the divertor heat load problem.

To test this important approach experimentally and compare (and combine) it with others, it is necessary to sustain good H-mode confinement with very high radiated power fraction. Thus the available heating power must exceed the H-mode threshold power by a large fraction. The H-mode threshold scaling proposed by the ITPA indicates that NHTX is well suited to examine this physics.

Other Approaches Worthy of Consideration for Testing

- **Sweeping divertor strike point.**
  - **Likely impossible in ITER or other conventional S/C tokamaks due to heating in cold structure.**
  - **Possible in ARIES-ST, maybe ARIES-AT with high-temperature S/C.**
  - **Issue of pumping with moving strike point.**
- **Sharing heat load with center post.**
  - **Likely impossible at conventional aspect ratio, due to need to breed tritium at center post.**
  - **Possible in ST, but will need very careful shaping of plasma and first wall.**
- **Enhancing SOL turbulence.**
  - **Are there means to spread heat load by artificially enhancing SOL turbulence, without reducing pedestal?**
  - **Interesting results from MAST.**
- **In-situ surface treatment**
  - **Coating replacement in real time.**
  - **Resurfacing on a ~monthly basis.**

There are a wide range of proposed approaches to reducing divertor heat and particle loads or dealing with the resulting erosion. Community discussion will be required to determined which ideas should be pursued in NHTX. These examples clearly illustrate the need for access and flexibility in NHTX.

Tritium Retention is a Critical Issue for ITER and Beyond

- **Carbon erosion and redeposition can result in up to a 50:50 mix of C and DT in surface films.**
- **In TFTR and JET 40 – 50% of all injected tritium remained in the vessel. New clean-up techniques are likely needed for ITER, if carbon is used.**
- **Safety concerns could limit the operational period between clean up to hours – potentially acceptable for ITER, but not for CTF/FDF or Demo.**
- **Tungsten or flowing lithium may be means to reduce tritium retention to acceptable levels.**

⇒ **NHTX mission requires long pulses, up to 1000C walls, replaceable plasma facing materials and components and tritium operational capability.**

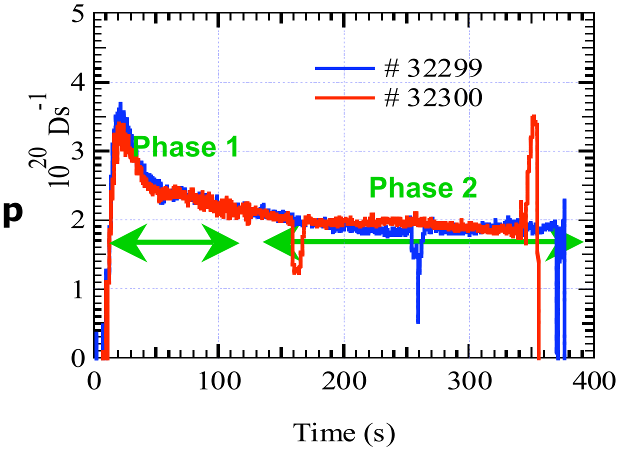
The solutions for tritium retention being pursued for ITER are focused on clean-up techniques that can be employed on nights, weekends or off-weeks. Since ITER operates with a modest pulse length, removal of tritium between pulses or weeks of operation may be acceptable to keep the in-vessel tritium inventory below 330g, as required for safety considerations. Such solutions are not acceptable for CTF/FDF or Demo, which must operate continuously for weeks or months, and so would accumulate an unacceptable level of tritium in a single shot.

Because hydrogen isotope diffusivity and solubility in materials is very sensitive to temperature, it is critical that NHTX be able to operate at Demo-relevant high wall temperatures. The likely wall temperature will depend on the material being tested, but for example tungsten walls can operate and should be tested at very high temperatures.

Since NHTX will operate extensively in DD, it will be necessary to use at least trace levels of tritium to test approaches to limiting tritium accumulation. While experimentation in DD will be very valuable, it will be difficult to assure that a large reservoir of D is not present in the machine after years of DD operation, even if the deuterium pump-out rate equals the deuterium fueling rate in a favored configuration. Trace (*e.g.,* 1%) tritium experiments would circumvent this problem by allowing complete accounting.

Long Pulses are Needed to Study Tritium Retention Issue

Puff - Pump



Tore Supra, France
Carbon PFCs

General Features of Retention:

- **Phase 1: Decreasing retention rate**
  - **~5 sec (JET) to 100 sec (Tore Supra)**
- **Phase 2: Constant retention rate**
  - **N_{wall}/N_{inj} ~ 50 - 80%**

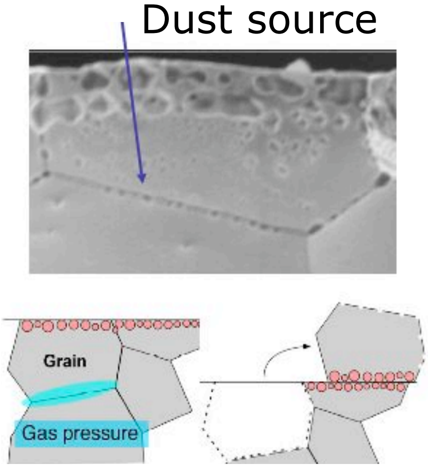
⇒ NHTX pulse length should be > 200 sec

The Tore Supra group has observed two phases of deuterium retention. The gas retained in the chamber during the first phase, at a rate above that of the steady second phase retention, is pumped out between shots. However the gas retained at the constant retention rate through the whole pulse is retained in the chamber. JET is reported to have a shorter first phase, while it is known that a significant number of ~5 second pulses are required on DIII-D to establish equilibrium between gas puff rate, plasma density, wall recycling and divertor pumping.

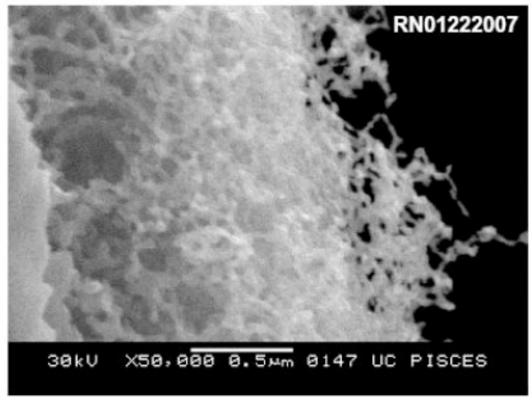
NHTX should be designed to minimize start-up and shut-down effects on gas retention. Pulse length capability in the range of 200 - 1000 seconds seems advisable, but should be the topic of further discussions.

Tungsten may be a Good Plasma Facing Material, but...

Dust source



He-induced foam



Nagoya University

UCSD

Melting at ELMs & disruptions are potential show stoppers.
At high power and fluence, dust and foam are concerns. T retention?
Tungsten radiates very strongly from the plasma core.
Must be tested at Demo conditions, including wall temperature.

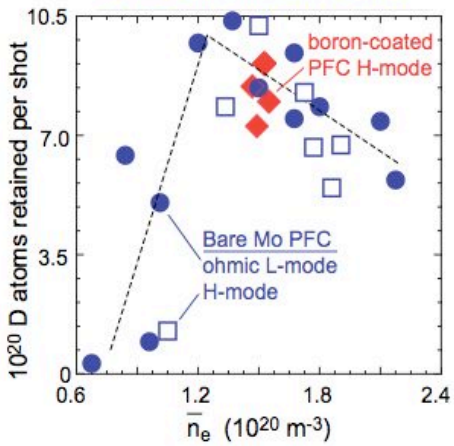
Some materials specialists feel that the only approach possible to reduce both tritium retention and erosion due to sputtering is to work with tungsten surfaces. However melting at ELMs and disruptions is a very major concern. Lab experiments in Japan and at UCSD indicate, furthermore, that dust production and surface bubble or foam formation are concerns under normal operation. The potential impact of these on tritium retention is not understood. NHTX will need to expose samples of tungsten which have been irradiated with neutrons, for example in IFMIF, since dislocations caused by neutron bombardment may function as trapping sites for tritium in the bulk of the material.

Tungsten self-sputtering is also a concern, and if moderate-Z impurities are used to enhance edge and SOL radiation, some authors believe that there may be potential for significant sputtering due to these impurities.

Since tungsten radiates very strongly from the plasma core, so even modest levels of plasma contamination are unacceptable, it is critical to test tungsten plasma facing components at Demo-relevant power and particle fluxes, and because tritium permeation and other materials properties are sensitive to material temperature, these tests must be undertaken at Demo-relevant wall temperature.

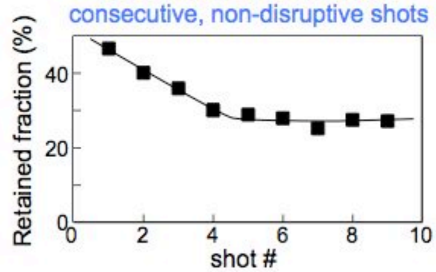
C-Mod Shows Strong D Retention with All-Molybdenum Walls

Retention in C-mod is not caused by co-deposition with boron

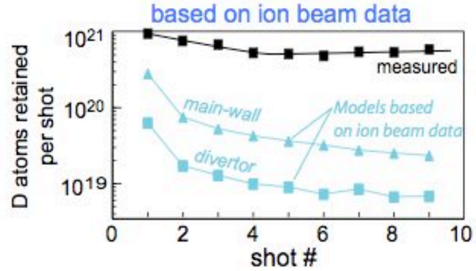


- Similar retention rates observed with all-metal versus boronized PFCs: 20-40% of fuelled gas, ~0.5% of incident ion flux.

No evidence of saturation in consecutive, non-disruptive shots



Retention inconsistent with models based on ion beam data



ASDEX-U has D retention with W walls, C divertor.

Experiments on C-Mod indicate that refractory metals such as molybdenum can retain very significant amounts of tritium. Lab tests, however, predicted 20 - 100x lower retention. ASDEX-U is seeing large D retention with tungsten walls and a carbon divertor.

Laboratory tests are critical to help provide fundamental understanding of the physics of plasma-surface interaction, and will need to be enhanced if the U.S. chooses to take on the NHTX mission. These tests span a range from ion beams to continuous plasma sources to plasma guns to simulate ELMs and disruptions. It is evident from this example, however, that lab tests alone will not be adequate to allow confidence in the ability of a CTF/FDF or Demo to operate continuously for weeks or months with a fusion-relevant plasma-material interface, sustained high-performance plasma operation and low tritium retention, as required to accomplish their missions.

Liquid Lithium is Attractive as a Plasma-Facing Material

FTU, Italy
Capillary Porous System (CPS)

- **Successful tests in TFTR, T-11, FTU, CDX-U, NSTX**
- **Reduces recycling, improves confinement.**
- **E-beam test to 25 MW/m^2 continuous operation.**
- **Plasma gun test to 15 MJ/m^2 off-normal load.**
- **Direct route to tritium removal, no dust.**

Experimentally lithium has been shown to reduce oxygen content and hydrogen recycling, and improve plasma confinement.

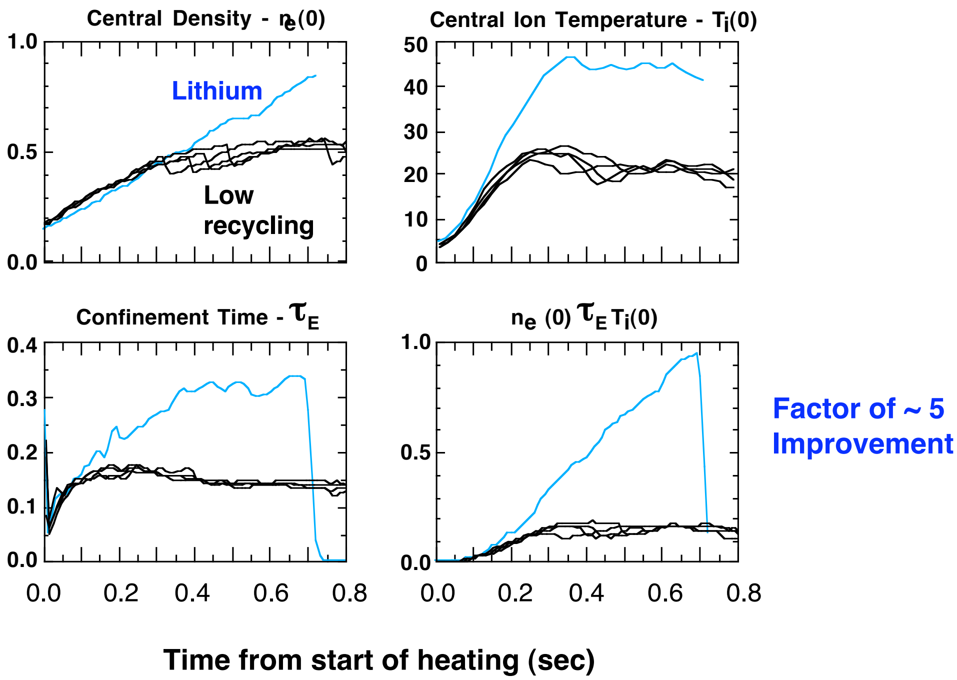
If lithium is flowed at a moderate rate through the fusion chamber, it should provide a direct mechanism to remove dissolved tritium, without having to contend with strong MHD forces.

Porous capillary systems are a promising way to present a film of liquid lithium to a fusion plasma. A Russian team has demonstrated the ability of such films to handle very high heat fluxes for long periods, and also to handle extremely high sudden off-normal heat loads, such as might be expected in disruptions or large ELMs.

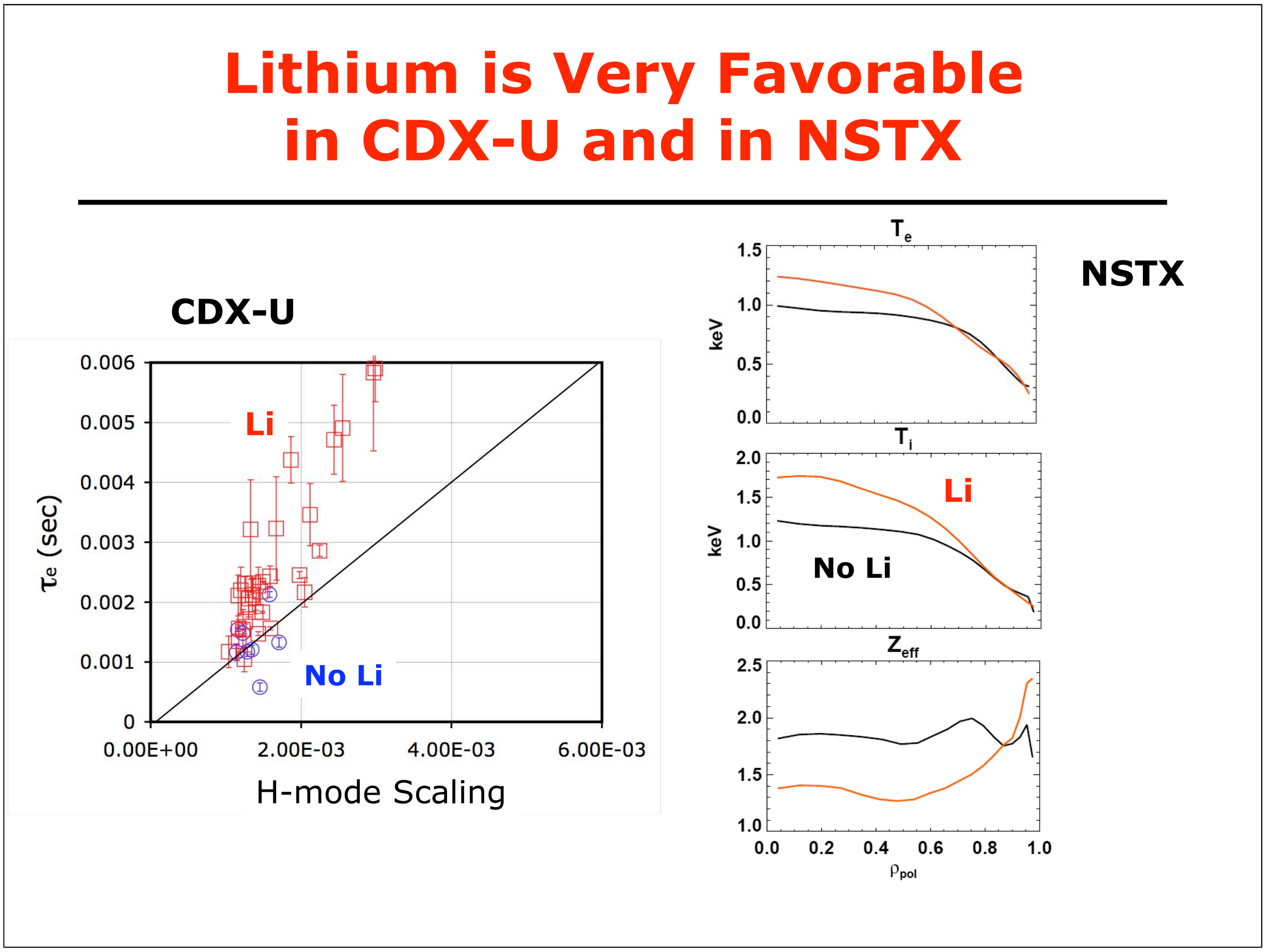
In normal operation lithium must be held well below 600C, in order to reduce the rate of evaporation. Tin and Gallium are also potentially attractive liquid metals for high power flux application, which could be tested in NHTX.

Liquid metal divertors and/or first wall components need to be tested in a short-pulse divertor machine at high power flux before applying them to NHTX.

Lithium had Dramatic Impact on Confinement in TFTR

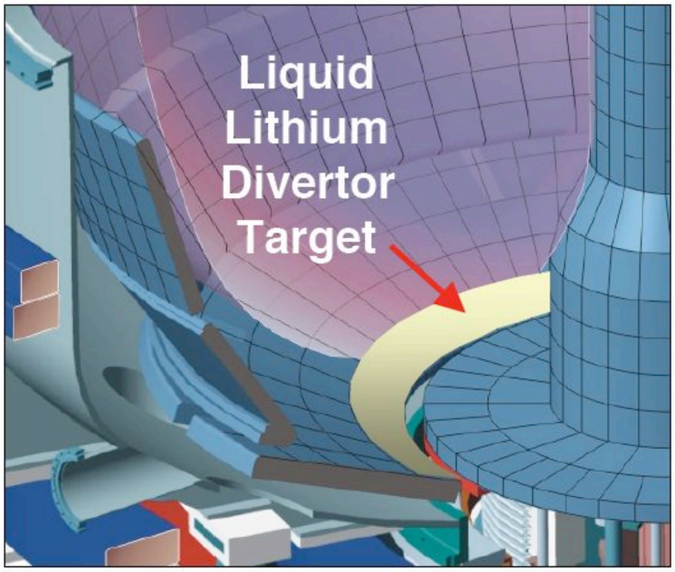


The impact of lithium wall coating on TFTR plasma performance was dramatic. Core confinement was substantially improved; central temperature and density increased as lithium further reduced recycling compared with well-conditioned, already low-recycling walls.



CDX-U has seen dramatic pumping with liquid lithium, and achieved approximately 2x H-mode confinement in Ohmically heating plasmas with lithium coating, as measured with magnetic diagnostics. NSTX has also seen significant improvement in performance with lithium evaporation in the divertor region.

Liquid Lithium Divertor Target will be Tested on NSTX



- **Lithium has been effective on limiter tokamaks.**
- **NSTX is the first set of tests with a divertor.**
- **Can very low recycling dramatically change fusion?**

NSTX will test a large-area liquid lithium divertor in FY09. This will provide the first test of power and particle handling by liquid lithium in a divertor device.

L. Zakharov has predicted that the very low recycling that can be achieved with a liquid lithium divertor surface will lead to flat temperature profiles and peaked density profiles, resulting in dramatically reduced turbulence levels and dramatically improved confinement. This idea will first be tested in the Lithium Tokamak Experiment, with a small plasma limited on liquid-lithium coated walls. NSTX will test this concept in a higher-power divertor plasma.

Low Aspect Ratio is Attractive for the NHTX Mission

- **Low R, copper coils are attractive for high P_{in}/R**
  - **Cost for new long-pulse heating/current drive ~$10/Watt.**
  - **At P_{in}/R = 50MW/m, ΔR = +1m costs $500M, just in power.**
  - **Low R and good access are difficult in a S/C device.**
- **A potential size target for NHTX is:**
  - **R ~ 1m for P_{in}/R ~ 50MW/m with affordable heating systems.**
  - **a ≥ 0.5m for diagnostic and maintenance access, flexibility in beam-driven current profile, P_{in}/S within reactor range.**
  - **⇒ R/a ≤ 2. Complements other facilities worldwide, supports cost-effective low-A Component Test Facility, Demo-ST.**
- **Preliminary studies show a favorable design point, with demountable water-cooled copper magnets.**
  - **Use existing PPPL MG sets = 4500 MJ**
  - **Use available site line power = 400 MW**
  - **Upgrade existing PPPL heating systems = 40 MW**
  - **Upgrade 680 MVA power-conversion systems to 1000 sec**
  - **TFTR Test Cell well shielded for deuterium, DT operation**

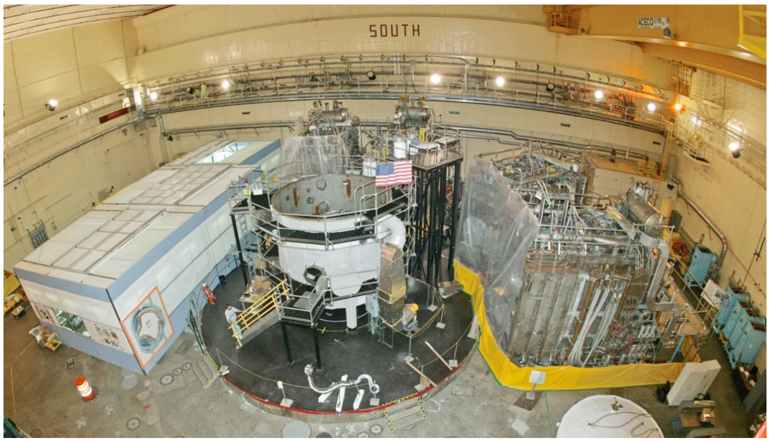
To achieve high P_{in}/R at a reasonable cost requires the choice of a relatively small R. If the community selects a goal of P_{in}/R = 50 MW/m, then each additional increment of 1m in R increases the cost of steady-state heating systems by ~$500M, not including the increased cost of the fusion device and its magnet power systems. It is difficult to reduce R in a superconducting device because of the needed thermal insulation, and access is limited by the requirement for a low-temperature cyrostat.

If we choose R ~ 1m for affordable heating systems, it is desirable nonetheless to keep a > 0.5m for access and heating flexibility. This leads to the choice of a moderately low aspect ratio device, which would complement the other steady-state tokamak devices world-wide, with A = 2.6 - 5.5.

Preliminary systems code studies support this choice as well to minimize the required magnet power. A favorable design point is found using steady-state water-cooled copper coils, with R = 1m, a = 0.55m, B = 2T, I_p = 3.5 MA. This is consistent with placement in the existing TFTR facility, which would considerably reduce costs and accelerate the implementation of such a device.

Much more community discussion, of course, will be required to settle on the parameters and siting of such a device.

TFTR Test Cell is Well Suited for the NHTX Mission



Thick shield walls
32 MW NBI
8 MW ICRF

680MVA@200sec power conversion
4500 MJ storage
400 MVA line power

The TFTR test cell was planned for use for the TPX experiment, and provides considerable "site credits" for a long-pulse device with extensive high-power DD and limited DT operation. Four neutral beam lines and a high-power ICRF system are available to be upgraded cost-effectively to long pulse, as was designed for TPX. The replacement cost for the TFTR facility is about $700M.

NHTX Heating and Current Drive

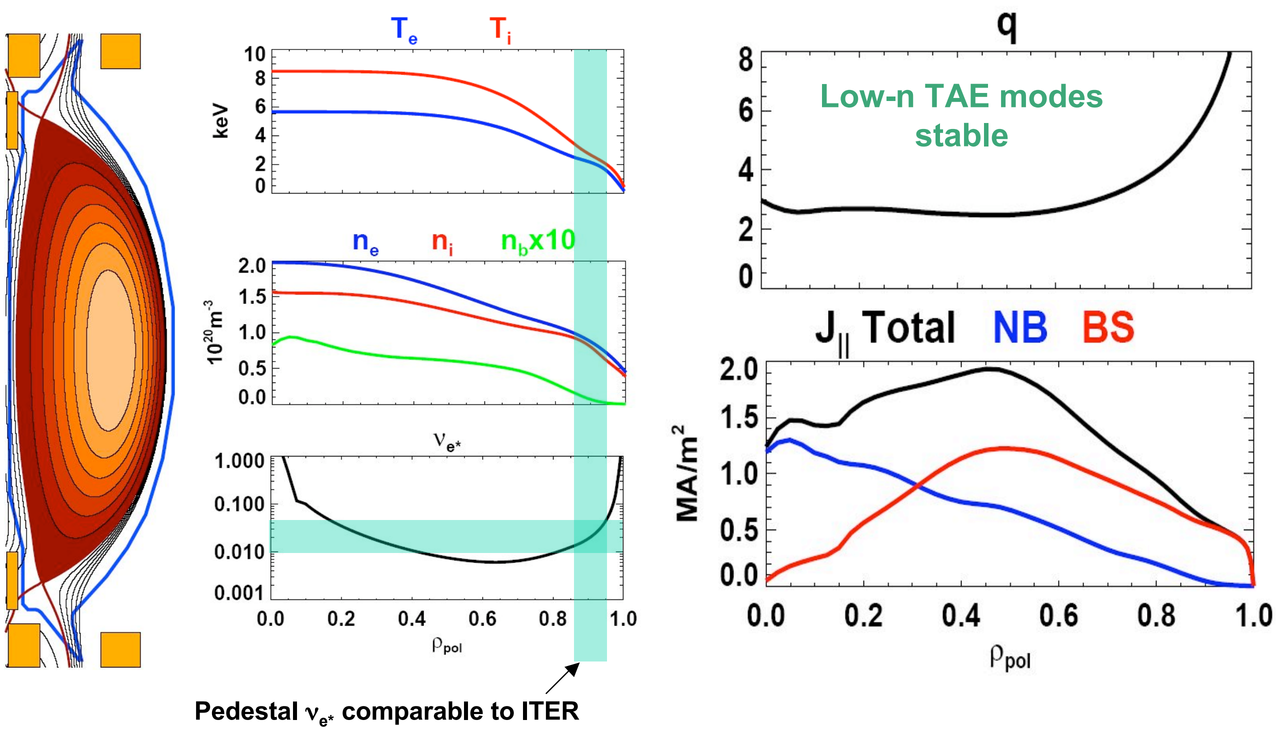
- **NHTX should be equipped initially with 50 MW**
  - **32 MW, 110 kV D_0 NBI, steerable off axis.**
  - **18 MW RF – type to be determined.**
- **For confidence in 3 – 4 MA steady-state operation, desirable to be able to drive 1–1.5 MA with beams + RF at moderate beta. (R_0 = 1m.)**
  - **Tangential NBI provides ~ 1MA, with flexible current profile control.**
  - **Lower CD is needed in advanced modes.**
- **Results from NSTX, DIII-D, C-Mod will be critical to selection of RF system(s).**
  - **EBWCD: High efficiency, remote coupling.**
  - **Inside-launch 120 GHz 2nd harmonic ECCD: lower efficiency, more complex access.**
  - **LHCD: Current ramp studies, intimate coupling.**
  - **ICRF: Cost-effective electron heating, intimate coupling.**

The heating and current drive systems have not been fully decided for NHTX, but 18 MW of RF is allocated in addition to the 32MW of NBI that will be available.

The ultimate choice of the RF systems should be made both on the basis of current drive needs for NHTX and also needs to demonstrate operation of launchers in a Demo-like environment.

Note that the overall philosophy is that NHTX should not need to achieve the challenging target of Demo-like bootstrap fraction (or beta-poloidal) in order to address the issues of power handling and tritium retention. However NHTX should provide a test-bed where such high performance, high-bootstrap operation can be developed as well.

3 MA is Achievable with Only NBI and Bootstrap



Studies indicate that neutral beam current drive is very effective in NHTX, and that variations in the beam orientation can provide flexible control over the driven current profile, including significant on-axis current drive, similar to what can be obtained with inductive drive. In effect these regimes can be considered to be equivalent to the “hybrid mode” regimes obtained on various devices, including NSTX, but with the inductive drive replaced with neutral beam current drive. Since this regime is far from the with-wall beta limit, and it should have strong rotation and rotational shear, it is not predicted to be close to stability limits.

The addition of 18 MW of RF power will provide additional control and flexibility.

National High-power advanced Torus eXperiment can Address the Integrated Fusion Science Mission

Device	R (m)	a (m)	P_{in} (MW)	P_{in}/R (MW/m)	P_{in}/S (MW/m^2)	Pulse (sec)	I_p (MA)	Species	Comments
Planned Long-Pulse Experiments									
EAST	1.70	0.40	24	14	0.55	1000	1.0	H (D)	Upgrade capability
JT-60SA	3.01	1.14	41	14	0.21	100	3.0	D	JA-EU Collaboration
KSTAR	1.80	0.50	29	16	0.52	300	2.0	H (D)	Upgrade Capability
LHD	3.90	0.60	10	3	0.11	10,000	–	H	Upgrade capability
SST-1	1.10	0.20	3	3	0.23	1000	0.2	H (D)	Initial heating
W7-X	5.50	0.53	10	2	0.09	1800	–	H	30MW for 10sec
NHTX	**1.00**	**0.55**	**50**	**50***	**1.13**	**1000**	**3.5**	**D (DT)**	**Initial heating**
ITER	6.20	2.00	150	24	0.21	400-3000	15.0	DT	Not for divertor testing
Component Test Facility Designs									
CTF (A=1.5)	1.20	0.80	58	48	0.64	~2 Weeks	12.3	DT	2 MW/m^2 neutron flux
FDF (A=3.5)	2.49	0.71	108	43	0.87	~2 Weeks	7.0	DT	2 MW/m^2 neutron flux
Demonstration Power Plant Designs									
ARIES-RS	5.52	1.38	514	93	1.23	Months	11.3	DT	US Advanced Tokamak
ARIES-AT	5.20	1.30	387	74	0.85	Months	12.8	DT	US Advanced Technology
ARIES-ST	3.20	2.00	624	195	0.99	Months	29.0	DT	US Spherical Torus
ARIES-CS	7.75	1.70	471	61	0.91	Months	3.2	DT	US Compact Stellarator
ITER-like	6.20	2.00	600	97	0.84	Months	15.0	DT	ITER @ higher power, Q
EU A	9.55	3.18	1246	130	0.74	Months	30.0	DT	EU "modest extrapolation"
EU B	8.60	2.87	990	115	0.73	Months	28.0	DT	EU
EU C	7.50	2.50	794	106	0.71	Months	20.1	DT	EU
EU D	6.10	2.03	577	95	0.78	Months	14.1	DT	EU Advanced
SlimCS	5.50	2.12	650	118	0.90	Months	16.7	DT	JA

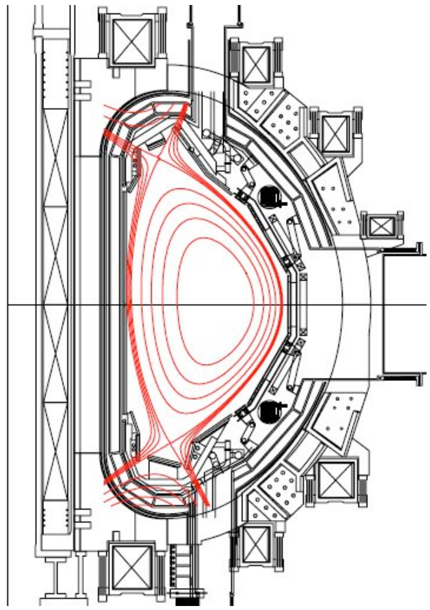
NHTX leapfrogs the field in the key area for CTF/FDF & Demo success.

* Flux compression, low R_x/R, SND, additional power allow higher heat flux.

The NHTX strawman "existence proof" device exceeds ITER considerably in P_{in}/R, and is in the reactor range of P_{in}/S. ITER is not designed for extensive divertor and first wall diagnostics, it does not have flexibility in its divertor configuration, and it does not have a high-temperature wall, so it is not well suited to the NHTX mission. (Note that a significant increase in the A of NHTX and so reduction in minor radius and elongation would result in P_{in}/S beyond reactor values.) CTF/FDF is mission-constrained like ITER, in that the basic machine design must provide confidence in producing 2 MW/m^2 of neutron flux, resulting in reduced access. The much higher neutron flux and fluence even than ITER, as well as the need to fill the midplane ports with blanket test modules, makes extensive diagnostic coverage impractical. The CTF/FDF requirement for high duty factor nuclear operation is also inconsistent with the scientific and technological flexibility required for the NHTX mission.

Using flux compression, reducing the major radius of the divertor strike point, operating in single-null and adding power if necessary should permit NHTX to provide the information about the plasma-surface interface needed for a CTF/FDF and Demo.

JT-60SA is an Excellent Device to Study Advanced Tokamak Plasma Physics

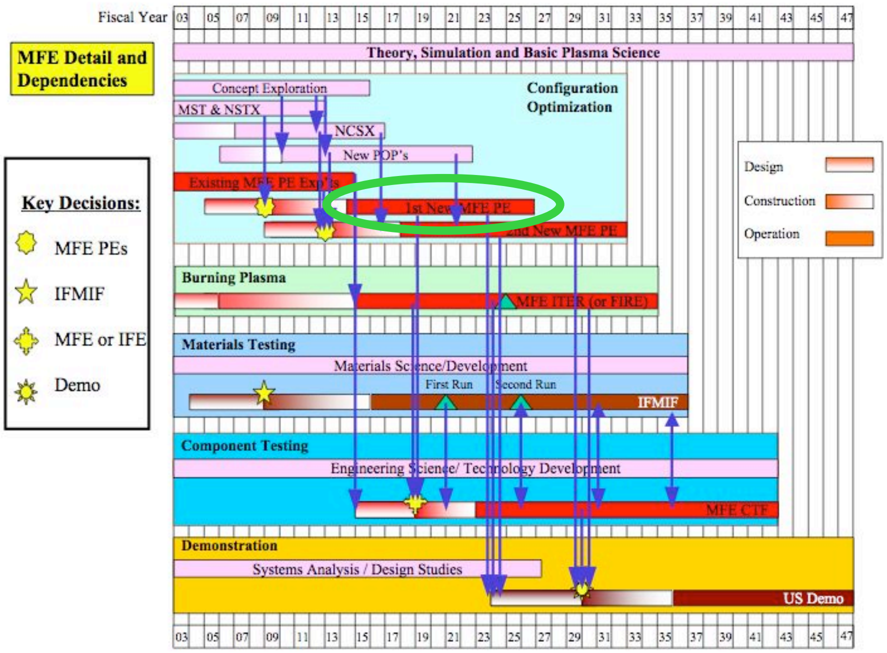


- Strong shaping
- Flexible heating and current drive
- DD capability
- Complementary $A = 2.6 - 3.1$

- Limited access, divertor flexibility
- Low T_{wall}
- Low P/R, P/S
- 100 sec @ high power
- No DT

JT-60SA, a joint JA – EU undertaking, is a superb device for studying advanced tokamak physics with pulses long compared with plasma processes, and the U.S. should plan to collaborate strongly on it, as we should hope Japanese and European scientists would want to collaborate on NHTX, with its focus on integrating a fusion-relevant plasma-surface interface with sustained high-performance plasma operation. JT-60SA does not incorporate, however, the key features that are required for the NHTX mission.

The FESAC Fusion Development Plan Calls for a New TFTR/JET-Class U.S. Facility



NHTX and IFMIF support CTF and Demo.
CTF construction is triggered by ITER success.

The FESAC development path plan indicated that a "First New MFE Performance Extension" device should provide information for the construction of CTF. IFMIF should also support the construction of CTF. This programmatic logic indicates that NHTX should validate the key plasma-material interaction physics and technology for CTF while IFMIF provides the data to determine what materials should be used to construct its blankets, and what properties should be assumed for those materials when irradiated.

Programmatically, it was assumed that the decision to construct such an expensive device as a CTF (or FDF) would amount to a decision between MFE and IFE, and good results from ITER would be required to support the decision to go forward with a multi-billion-dollar MFE CTF rather than a programmatically equivalent device for IFE, which would be similarly expensive.

NHTX Resolves Critical Issues for CTF/FDF and ITER

- **Low-A CTF is Very Attractive**
  - **Reduced device cost, tritium burn rate, plant power.**
  - **Access for blankets, tangential neutral beam current drive.**
  - **NHTX provides confinement data at low A.**
- **NHTX can Test Start Up / CD Approaches at scale**
  - **Need to select amongst: EBW/ECCD, Lower Hybrid, Neutral Beam Drive; Iron Core, CHI, PF-only.**
  - **Can avoid inner blanket or shield in CTF/FDF w/o OH.**
  - **A big payoff at any aspect ratio.**
  - **Non-inductive ramp-up is a key feature of ARIES-AT, SlimCS.**
- **Long Pulse, High P_{in}/R with Low Tritium Retention**
  - **Required for successful CTF/FDF design.**
  - **Supports decision for ITER second-phase PFC's.**

NHTX can resolve critical issues for a CTF/FDF and ITER. A low-A CTF is attractive since the device cost is reduced, the tritium burn rate is reduced (and fraction of neutrons intercepted by the inner-wall shield is reduced), and the overall plant power is reduced. NHTX would provide data at low A to support the decision on A for a CTF.

NHTX, while it would be provided with a transformer for startup, could test non-inductive or low-inductive (equivalent to using a small iron core) startup schemes. This would allow the elimination of the OH solenoid in a CTF/FDF, which reduces cost very considerably. Non-inductive ramp-up is assumed in the ARIES-AT and SlimCS designs, and requires development.

Most fundamentally, however, techniques to operate in long pulse at high P_{in}/R with low tritium retention provide confidence in making the large investment in a CTF/ FDF. Some of the techniques developed using NHTX should also be able to be tested in the later phase of ITER.

Close Linkage of NHTX and IFMIF with CTF/FDF is Necessary

CTF/FDF Construction Decision

- **FESAC Development Path Approach**
  - NHTX provides PFC technology & divertor configuration
  - NHTX provides information on steady-state performance, *vs.* A
  - IFMIF provides materials choices and their properties for blankets
  - ITER provides confirmation of fusion performance at long pulse
- **Earlier Construction / Higher Risk Approach**
  - Choice of PFC technology, divertor configuration, aspect ratio, steady-state mode based on lab tests, short-pulse cold-wall experiments, EAST, KSTAR
  - Blanket materials chosen based on fission-spectrum tests
  - Tritium retention and high duty factor, high P_{in}/R operation are major risks

CTF/FDF Operation Phase

- Access and flexibility are limited. P_{in}/R is low pre-DT.
- Blanket testing requires 30% duty factor, 2 week pulses, 10–20m² of midplane test area ⇒ 8 pulses/year, very limited diagnostic access.
- NHTX provides innovations in PFC technology, steady-state operation.
- IFMIF provides innovations in materials.
- CTF/FDF implements innovations in change-outs as testing proceeds.

Close linkage with NHTX and IFMIF is required for success.

Close linkage of NHTX and IFMIF to CTF/FDF is necessary.

For the CTF/FDF construction decision the *FESAC Development Path approach* has both NHTX and IFMIF providing key technical information. Success on ITER provides the impetus to proceed aggressively with magnetic fusion energy and begin construction of CTF/FDF. Alternatively one could choose a *higher risk approach*, basing the design choices for CTF/FDF largely on lab tests and short pulse experiments. It would seem critical in this case also to leverage as much as possible on long-pulse experiments in EAST and KSTAR.

In either case, close linkages of NHTX and IFMIF to CTF/FDF during operation are critical. If CTF/FDF is designed with a moderate level of flexibility, it can incorporate innovations in power and tritium handling from NHTX and in materials from IFMIF. It can also import improved advanced operating modes. Given the limited access for diagnostics in CTF/FDF, the design for 3x higher neutron flux and 20x higher neutron fluence than ITER, and the need to proceed to component testing as soon as significant DT operation is possible, it is not practical to perform the NHTX mission on CTF/FDF.

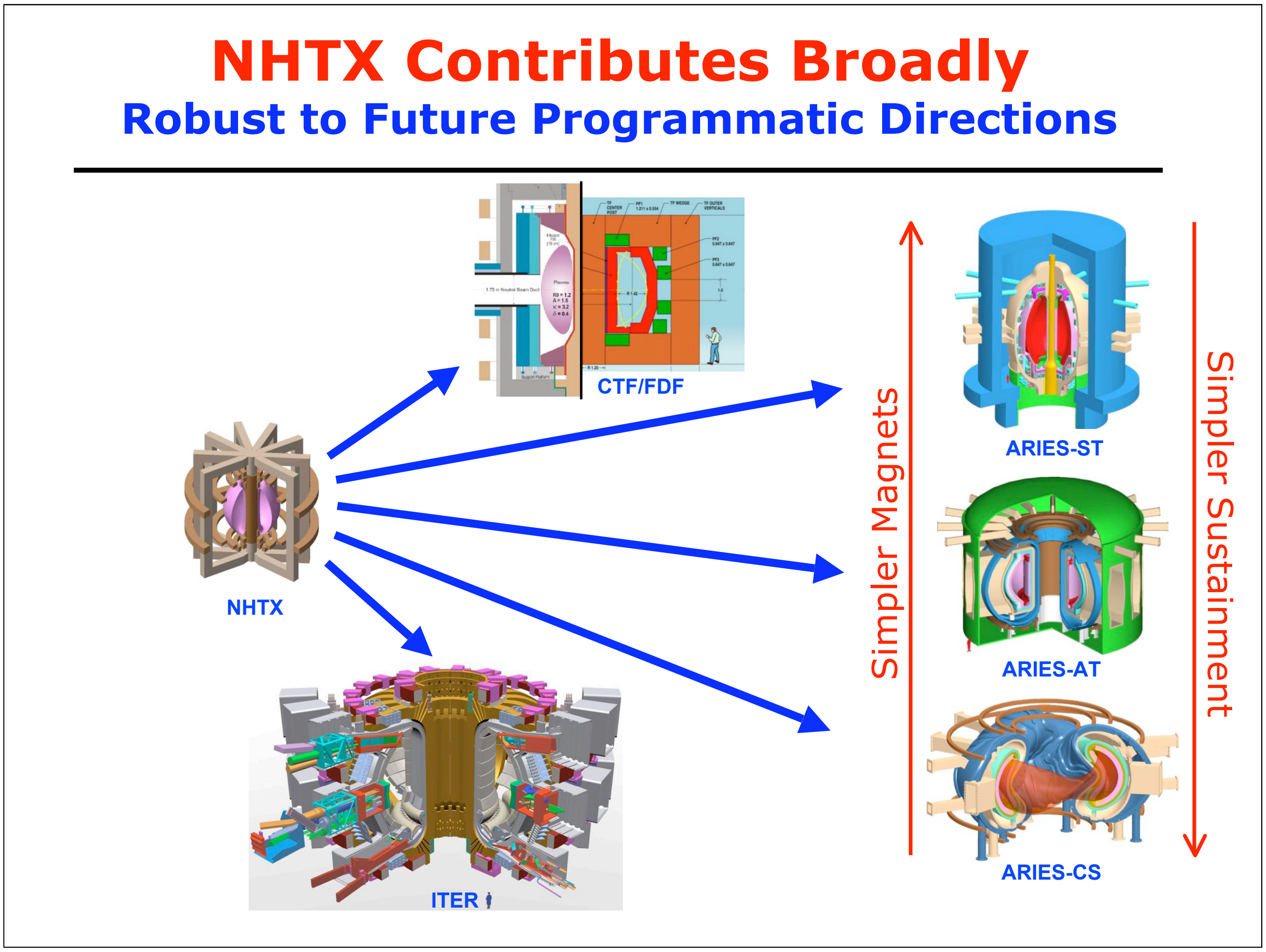
It is worth noting that close linkages between NHTX and IFMIF will also be very valuable. NHTX will be able to test the effects of neutron irradiation on tritium retention, and should be able to take advantage of IFMIF tests of irradiation effects on materials and bonding techniques, to provide the most realistic surface studies for CTF/FDF and Demo.

NHTX Resolves Issues Specific for an ST Demo

- **Coupled with ITER results, theory and computation, NHTX provides needed confinement data at low A**
- **Operated at high beta / bootstrap, NHTX demonstrates sustained ST-Demo operating point**
- **DT operation provides α-particle stability tests**
- **NHTX tests start-up / CD approaches at scale**
- **Long pulse, high P_{in}/R with low tritium retention consistent with high performance plasma operation**
  - ***Required for any Demo design: Tokamak, ST or Compact Stellarator***

If a low aspect ratio is selected for NHTX, it would provide key information in each key area of physics: confinement, sustainment, energetic particles, start-up and current drive, and boundary physics, opening up the range of aspect ratio choice allowable for Demo. In particular, short pulse operation at 50:50 DT will be available in NHTX to test alpha particle stability and loss.

Most fundamentally, of course, the information on long pulse, high performance, high P_{in}/R, low tritium retention operation available from NHTX will be required for any Demo. ARIES-ST has a particularly high value of P_{in}/R, and NHTX can test ideas such as power-sharing to the center column and swept divertor strike point that are specially relevant to the ST Demo, with its non-breeding center column and normal-conducting coils.



The decision to construct a device with the mission of NHTX is robust against other, later programmatic decisions. NHTX supports a CTF/FDF decision, but if it is decided not to go forward with CTF/FDF, planning on a robust second phase of ITER and significant blanket testing in the first phase of Demo, NHTX would provide the information needed on long-pulse, high P_{in}/R, low tritium retention operation for iterated divertor and/or first-wall components in ITER and for Demo.

Furthermore, since the NHTX mission is largely generic with respect to plasma configuration, it will provide key information for an ST, Advanced Tokamak or Compact Stellarator demo. Since we do not know now how difficult steady-state operation will be in a driven device (ST or AT), nor do we know the real cost of superconducting magnets of varying complexity (high-T superconductors in AT and complex shapes in CS), we are not now in a position to make this choice.

NHTX is Part of a U.S. Program Aimed at the Integrated Fusion Science Mission

- **Materials and Technology**
  - **Develop and test new plasma-facing materials.**
  - **Develop and test new plasma-facing technologies.**
  - **Develop plasma technologies (*e.g.,* RF launchers, diagnostics) for long-pulse, high heat flux.**
- **Experiment**
  - **Enhance focus on innovative boundary solutions.**
  - **Advance integrated high performance physics on C-MOD, DIII-D, NCSX, NSTX**
  - **Collaborate on superconducting facilities abroad.**
- **Theory and Computation**
  - **Focus SciDAC's / FSP on Demo-relevant plasma boundary solutions. Expand this research.**
  - **Design new plasma-facing materials.**
  - **Advance the theory of high-performance operating modes.**

If the U.S. chooses to address the NHTX mission, this would bring with it a strengthening of focus in this area.

The area of plasma technologies, including plasma-facing materials and components as well as RF launchers, and the test stands to develop them, should be much more strongly emphasized to address this mission with confidence.

As ITER was downsized from P_{fus} = 1500 MW to 500 MW, this issue has lost programmatic emphasis, but of course the challenge remains, and the step from ITER to Demo is now greater than before. This area should have increased emphasis in ongoing toroidal experiments.

Theory should be focused more strongly on the edge and divertor to support this initiative, and new materials (*e.g.*, tungsten alloys less liable to dust formation) should be developed computationally and tested experimentally

NHTX Must be Designed, Constructed and Operated fully Nationally – A New Paradigm

- **The Integrated Fusion Science Mission requires skills in all areas:**
  - **Confinement**
  - **Stability**
  - **Sustainment**
  - **Boundary**
- ***These skills are only available on a fully national basis.***
- **Design, Construction and Operation must be undertaken through a fully national collaboration, with fully national leadership and fully national participation.**

It is clear that a major initiative such as this needs to be further developed through national discussion, and then designed, constructed and operated fully nationally. This would need to go beyond the model employed on TPX in order to assure complete community involvement and investment in the program.

The U.S. is Positioned to Lead the World Across the Next Critical Frontier for Fusion

- **Major long-pulse confinement experiments will operate in parallel with ITER in China, Europe, India, Japan, and South Korea. The ideas behind the tokamak experiments are based on the 1993 U.S. TPX proposal… *but the science has moved on*.**

- **There is an opportunity for an exciting U.S. experiment that would cross the next frontier in fusion by developing plasma-boundary science at the level needed for CTF/FDF and Demo.**

- **NHTX would open up the aspect-ratio choice for Demo to include low A. Its value is robust to future program direction.**

- **Such a device would provide a key driver for U.S. experiment, technology, theory, and computation – and workforce development – leveraging our participation in ITER.**

The NHTX mission defines a very exciting national initiative which could be started soon, and would put the U.S. in a leadership position in the race to fusion.