The Fusion Development Facility (FDF) Option for the Fusion Nuclear Science Facility (FNSF)

Fusion Nuclear Science and Technology Meeting

UCLA

August 18-20, 2009
The Fusion Development Facility Mission: Develop Fusion’s Energy Applications

• FDF will:
  – Close the fusion fuel cycle
  – Show electricity production from fusion.
  – Develop high temperature blankets for high efficiency electricity production.
  – Show hydrogen production from fusion.
  – Provide a materials irradiation facility to develop fusion materials.

• By using conservative Advanced Tokamak physics to run steady-state and produce 100-250 MW fusion power
  – Modest energy gain (Q<7)
  – Continuous operation for 30% of a year in 2 weeks periods
  – Test materials with high neutron fluence (3-6 MW-yr/m²)
  – Further develop all elements of Advanced Tokamak physics, qualifying them for an advanced performance DEMO

• With ITER and IFMIF, provide the basis for a fusion DEMO Power Plant
To Show Fusion Can Close its Fuel Cycle, FDF Will Demonstrate Efficient Net Tritium Production

- FDF will produce 0.4–1.3 kg of Tritium per year at its nominal duty factor of 0.3
- This amount should be sufficient for FDF and can build the T supply needed for DEMO
### The Main Nuclear Facing Structures in FDF Do Not See More Than 2 MW-yr/m²

<table>
<thead>
<tr>
<th>Component</th>
<th>Should Operate for</th>
<th>Planned Changeout every</th>
</tr>
</thead>
<tbody>
<tr>
<td>Toroidal Coil</td>
<td>Life of facility, 30 years, 6 MW-yr/m²</td>
<td>none</td>
</tr>
<tr>
<td>OH Solenoid</td>
<td>2 MW-yr/m² or about 2-5 years</td>
<td>With inner blanket</td>
</tr>
<tr>
<td>Inner Full Main Blanket</td>
<td>2 MW-yr/m² or about 2-5 years</td>
<td>3-5 years</td>
</tr>
<tr>
<td>Outer Full Main Blanket</td>
<td>2 MW-yr/m² or about 2-5 years</td>
<td>3-5 years</td>
</tr>
<tr>
<td>Divertor</td>
<td>0.5 MW-yr/m² or about 1 year</td>
<td>3-5 years</td>
</tr>
<tr>
<td>Launchers, Couplers</td>
<td>About 1 year</td>
<td>Upgrades as desired</td>
</tr>
<tr>
<td>Test blanket modules in ports</td>
<td>6 months</td>
<td>2 years</td>
</tr>
<tr>
<td>Materials Samples in ports</td>
<td>Removed at regular intervals</td>
<td>10 years integrated exposure</td>
</tr>
</tbody>
</table>
Port Sites Enable Nuclear and Materials Science.

- DIII-D size neutral beams
  - 3 Co 120 keV, rotation
  - 1 Counter, 80 keV for QH mode edge

- Off-axis current profile control
  - ECCD (170 GHz)
  - Lower Hybrid
  - NBCD

- Port blanket sites for fusion materials development

- Port blanket sites for fusion nuclear technology development
Teams of Universities, Labs, and Industry Will Work with the Site to Field Test Blanket Modules on FDF

- **Fusion electric blankets require**
  - High temperature (500-700 °C) heat extraction
  - Complex neutronics issues
  - Tritium breeding ratio > 1.0
  - Chemistry effects (hot, corrosive, neutrons)
  - Environmentally attractive materials
  - High reliability, (disruptions, off-normal events)

- **Fusion blanket development requires testing**
  - Solid breeders (3), Liquid breeders (2)
  - Various Coolants (2)
  - Advanced, Low Activation, Structural materials (2)

- **Desirable capabilities of a development facility**
  - 1–2 MW/m² 14 MeV neutron flux
  - 10 m² test area, relevant gradients (heat, neutrons)
  - Continuous on time of 1-2 weeks
  - Integrated testing with fluence 6 MW-yr/m²

- **FDF can deliver all the above testing requirements**
  - Test two blankets every two years
  - In ten years, test 10 blanket approaches
  
  **Produce 300 kW electricity from one port blanket**
FDF will Motivate the Needed, Large, Supporting Fusion Nuclear Science Program

On Test Specimens and Components,

- Materials compositions
- Activation and transmutation
- Materials properties (irradiated)
- Thermo-hydraulics
- Thermal expansion and stress
- Mechanical and EM stresses
- Tritium breeding and retention
- Solubility, diffusivity, permeation
- Liquid metals crossing magnetic fields
- Coolant properties
- Chemistry
- and more.....
Teams of Universities, Labs, and Industries Will Conduct a National Program of Materials Irradiation

- Provides up to 60 dpa of DT fusion neutron irradiation in controlled environment materials test ports for:
  - First wall chamber materials
  - Structural materials
  - Breeders
  - Neutron multipliers
  - Tritium permeation barriers
  - Composites
  - Electrical and thermal insulators

- Materials compatibility tests in an integrated tokamak environment
  - Flow channel inserts for DCLL blanket option
  - Chamber components and diagnostics development
<table>
<thead>
<tr>
<th>Materials Which Could Be Irradiation Tested in FDF to 60 dpa</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>PFC surface materials:</strong></td>
</tr>
<tr>
<td>W, Mo...etc.</td>
</tr>
<tr>
<td>Joining materials: W to RAFM, W to ODFS, etc.</td>
</tr>
<tr>
<td>Engineered materials: BW, Low-Z+UFG refractory...etc.</td>
</tr>
<tr>
<td>Flow-through materials: e.g. C, B</td>
</tr>
<tr>
<td><strong>Structural materials:</strong></td>
</tr>
<tr>
<td>Different RAFM alloys: e.g. F82H, EUROFER...etc.</td>
</tr>
<tr>
<td><strong>Flow channel inserts for DCLL:</strong></td>
</tr>
<tr>
<td>SiC/SiC composite, SiC-foam</td>
</tr>
<tr>
<td><strong>MHD insulation for self-cooled options:</strong></td>
</tr>
<tr>
<td>Y$_2$O$_3$, Er$_2$O$_3$, AlN, sandwich layers...etc.</td>
</tr>
<tr>
<td><strong>Advanced structural materials:</strong></td>
</tr>
<tr>
<td>Different ODFS alloys, W-Re alloys, W-TiC alloys, Mo-alloys, SiC/SiC...etc</td>
</tr>
<tr>
<td><strong>Solid breeder materials:</strong></td>
</tr>
<tr>
<td>Li$_2$O, Li$_2$TiO$_3$, Li$_4$SiO$_4$</td>
</tr>
<tr>
<td><strong>Liquid breeder materials:</strong></td>
</tr>
<tr>
<td>Pb17Li, FLiBe...others.</td>
</tr>
<tr>
<td><strong>Neutron multipliers:</strong></td>
</tr>
<tr>
<td>Be, BeTi...etc.</td>
</tr>
<tr>
<td><strong>Tritium barriers:</strong></td>
</tr>
<tr>
<td>Al$_2$O$_3$...etc.</td>
</tr>
<tr>
<td><strong>Low flux tests:</strong></td>
</tr>
<tr>
<td>Superconducting materials for low and high temp. superconductors</td>
</tr>
<tr>
<td>Electrical and thermal insulator...etc.</td>
</tr>
<tr>
<td><strong>Shielding materials:</strong></td>
</tr>
<tr>
<td>SS, FS, W, B...etc.</td>
</tr>
<tr>
<td><strong>Structural joint materials:</strong></td>
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# The U.S. Fusion Nuclear Science Community Suggested an Aggressive Phased Research Plan

<table>
<thead>
<tr>
<th></th>
<th>1</th>
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<tbody>
<tr>
<td><strong>Fusion Power (MW)</strong></td>
<td>0</td>
<td>0</td>
<td>125</td>
<td>125</td>
<td>250</td>
<td>250</td>
<td>250</td>
<td>400</td>
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<tr>
<td><strong>P/A_WALL (MW/m²)</strong></td>
<td>1</td>
<td>1</td>
<td>2</td>
<td>1</td>
<td>1</td>
<td>2</td>
<td>2</td>
<td>3.2</td>
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<tr>
<td><strong>Pulse Length (Min)</strong></td>
<td>1</td>
<td>10</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
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<tr>
<td><strong>Duty Factor</strong></td>
<td>0.01</td>
<td>0.04</td>
<td>0.1</td>
<td>0.2</td>
<td>0.2</td>
<td>0.3</td>
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<tr>
<td><strong>T Burned/Year (kG)</strong></td>
<td>0.28</td>
<td>0.7</td>
<td>2.8</td>
<td>2.8</td>
<td>4.2</td>
<td>4.2</td>
<td>5</td>
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<tr>
<td><strong>Net Produced/Year (kG)</strong></td>
<td>-0.14</td>
<td>0.56</td>
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</table>
| **Main Blanket**     | He Cooled Solid Breeder Ferritic Steel | Dual Coolant Pb-Li Ferritic Steel | Best of TBM RAfs?
| **TBR**              | 0.8 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 |    |    |    |    |    |    |    |    |    |    |    |    |    |
| **Test Blankets**    | 1,2 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |
| **Accumulated Fluence (MW-yr/m²)**| 0.06| 1.2 | 3.7 | 7.6 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |

*General Atomics*
# FNSF Steering Group Formed Spring 09

<table>
<thead>
<tr>
<th>Name</th>
<th>Institute</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mohamed Abdou</td>
<td>UCLA</td>
</tr>
<tr>
<td>Vincent Chan</td>
<td>GA</td>
</tr>
<tr>
<td>Ray Fonck</td>
<td>U. Wisconsin</td>
</tr>
<tr>
<td>Dave Hill</td>
<td>LLNL</td>
</tr>
<tr>
<td>Rick Kurtz</td>
<td>PNNL</td>
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<tr>
<td>Dale Meade</td>
<td>Retired</td>
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<tr>
<td>Stan Milora</td>
<td>ORNL</td>
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<tr>
<td>Gerald Navratil</td>
<td>Columbia U.</td>
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<tr>
<td>Martin Peng</td>
<td>ORNL</td>
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<tr>
<td>Stewart Prager</td>
<td>PPPL</td>
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<tr>
<td>Ron Parker</td>
<td>MIT</td>
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<tr>
<td>James Rushton</td>
<td>ORNL</td>
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<tr>
<td>Ned Sauthoff</td>
<td>ITER</td>
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<td>Ron Stambaugh</td>
<td>GA</td>
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<tr>
<td>George Tynan</td>
<td>UCSD</td>
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<tr>
<td>Jim Van Dam</td>
<td>U. Texas</td>
</tr>
</tbody>
</table>
1 dpa - Objectives Realizable with less than 1 dpa lifetime of facility
(Stage I of FDF, duty factor 1-4% in a year. Longest operating duration a few hours)

Show significant fusion power can be produced.
ST version 50-100 MW. AT version 150—400 MW. This goal shared with ITER.

Show avoidance and mitigation of off-normal events.
Can the tokamak be a steady-state device? Essential to get beyond Stage I.
Elements of Mission Scope for FNSF Achievable at Progressive dpa Levels.

3 dpa - Objectives realizable with less than 3 dpa lifetime of the facility or 1 dpa to any onemain blanket. Achievable with a low performance Stage II. Duty factor <5%. 2.5 accumulated weeks per year. One one-week long operating period per year.

**Opportunity to show high performance, steady-state, long duration, burning plasmas.**
Addresses the question are we a serious energy source? Key feasibility objective for the tokamak and for AT plasmas toward energy.

**Opportunity to extend operation to the kind of high performance plasmas needed in ARIES-AT DEMO**
Open ended toward AT modes for DEMO. Possible for ST if electron confinement substantially better than the ITER-H mode scaling.

**Test and develop auxiliary systems for true steady-state.**
A challenge for the technology and the engineering.

**Develop blanket PFC surfaces compatible with high performance, high duty factor plasmas.**
Addresses the PFC erosion and plasma purity issues in the nuclear environment.

**Demonstrate the production of high grade process heat from fusion.**
Done in main blankets and test blanket modules.

**Demonstrate fusion fuel self-sufficiency.**
What TBR is required? Main task for the main, full blankets.

**Show electricity production from fusion.**
Done as a demonstration on a test blanket module.

**Show hydrogen production from fusion.**
Done as a demonstration on a test blanket module.

**Begin to accumulate data on synergistic damage effects of strongly interacting materials and material interfaces.**
A unique contribution from FNSF, guiding materials selection for high-dpa materials testing program.
**Elements of Mission Scope for FNSF Achievable at Progressive dpa Levels.**

**10 – 20 dpa** - Objectives realizable with less than 10-20 dpa lifetime of the facility or 3-7 dpa to any one main blanket. Low end of fluence for the planned Stage II, Stage III, and Stage IV Main Blanket Programs. One 2-week long run per year. Duty factor 10% or 5 accumulated weeks per year.

Develop plasma measurements suitable for a fusion power plant.

*Perhaps the hardest challenge undertaken by FNSF.*

Develop blanket PFC surfaces compatible with high performance, high duty factor plasmas that handle alteration of physical properties by nuclear environment.

*Addresses the PFC properties issues in the nuclear environment.*

Develop blankets that produce tritium efficiently.

*From this is derived the two week continuous on-time requirement. Addresses also the PFC Tritium retention and permeation issues. Main mission for the main, full blankets.*

Develop high temperature blankets for high efficiency electricity production.

*Motivates the test blanket module program – a major research program in its own right with national coordination. Drives need for high neutron flux.*

Accumulate data on synergistic damage effects of strongly interacting materials and material interfaces.

*A unique contribution from FNSF, guiding materials selection for high-dpa materials testing program.*
**High Fluence** - Objectives which require reaching for fluence. Nominal target 30-60 dpa into Materials Irradiation Test Stations accumulated in 15 years or 20 dpa into any one main blanket over 5 years. Duty factor goal 30%. Three 2-week long runs per year. 15 accumulated weeks operation per year.

Show significant fusion power can be produced with a significant duty factor.

*Again, is this on a path to an energy source? Enables materials research below.*

Develop and perform irradiation tests on low activation, high strength, high temperature materials, material combinations, welded assemblies, etc. for long lived fusion blankets.

*Drives the need for fluence. Ports devoted to materials irradiation program that is large in scope and nationally coordinated*

Develop reliability data on fusion nuclear components for DEMO

*Tests of full assemblies in neutron environment with transmutation, activation, welds, etc.*
Goals and Objectives

Produce significant fusion power.

Make high performance, steady-state, burning plasmas.

Develop low activation, high strength, high temperature, radiation resistant materials for fusion.

Demonstrate the production of high grade process heat from fusion.

Demonstrate fusion fuel self-sufficiency.

Show fusion can produce electricity.

Show fusion can produce hydrogen.

Obtain first reliability, availability, and maintainability experience.
Program Resources

Fusion Nuclear Science Facility (FNSF) for all major objectives.

National fusion test blanket module program to extract heat and produce fuel (industry, labs, university teams), fielding and studying test units on FNSF and ITER.

Nationally coordinated fusion materials irradiation and development program using FNSF, an accelerator based neutron source, fission reactors, triple ion beam sources, linear plasma devices, materials test stands, and computational modeling of basic processes.

National program to develop plasma measurements suitable for a nuclear environment.

Development of plasma facing components which handle enormous heat, plasma, and neutron fluxes while supporting high plasma performance and fusion fuel self-sufficiency - using FNSF, research tokamaks, linear plasma devices, and test stands.

Research on complete tritium processing loops for net tritium production using FNSF and other test systems.

Research tokamak programs to develop the high performance plasma operation modes.

Development of auxiliary systems to support true steady-state operation.

Development of jointed magnets, both copper and superconducting, for an effective maintenance scheme.
Reliability, Availability, Maintainability, and Inspectability Have Dominated the FDF Design Concept

- It is the main reason the machine is based on a **jointed copper coil**
  - To enable planned changes in the entire blanket structure
  - So the entire machine can be taken apart readily and any component remotely serviced.
  - So the nuclear components can be accessible by simple crane lift.

- The machine must be **reliable** to achieve continuous two week operation and the **availability** to achieve a duty factor 0.3 on a year.

- It must be **maintainable** because it is a research environment and problems must be repairable and the blankets changeable.

- **Inspection** of the components is an integral part of the research; we need to find out what is happening to all these components.

- The FDF Program will be a test bed for learning how to engineer reliable first wall/blanket structures and to gain first information on **reliability growth**.
FDF Will Be a Necessary Progress Step to DEMO in Dealing With Off-Normal Events

• **Goals**
  – Operate for $10^7$ seconds per year, duty factor 0.3.
  – Run two weeks straight without disruption
  – Only one unmitigated disruption per year

• **Disruption Strategy**
  – Real-time stability calculations in the control loop.
  – Active instability avoidance and suppression - RWMs, NTMs
  – Control system good enough to initiate soft shutdowns and limit firing the disruption mitigation system more than 20 times per year.
  – Disruption mitigation system 99% reliable.

• **ELM Suppression**
  – Resonant magnetic perturbation coils
  – QH-mode
FDF Will Be a Useful Intermediate Step in Managing Disruptions Between ITER and DEMO

Typical requirements for disruption avoidance

<table>
<thead>
<tr>
<th>Device</th>
<th>ITER</th>
<th>FDF</th>
<th>DEMO</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pulse length (s)</td>
<td>400</td>
<td>$1\times10^6$</td>
<td>$3\times10^7$</td>
</tr>
<tr>
<td>Number of pulses per year</td>
<td>1000</td>
<td>10</td>
<td>1</td>
</tr>
<tr>
<td>Fast shutdowns per year</td>
<td>100</td>
<td>20</td>
<td>5</td>
</tr>
<tr>
<td>Time between fast shutdowns (s)</td>
<td>$4\times10^3$</td>
<td>$5\times10^5$</td>
<td>$6\times10^6$</td>
</tr>
<tr>
<td>Unmitigated disruptions per year</td>
<td>5</td>
<td>1</td>
<td>0.3</td>
</tr>
</tbody>
</table>
Internal Components and Sources Must Be Developed to Operate in High Neutron and Particle Fluence

- Biggest step FDF makes is in on time, $10^7$ seconds per year, and hence neutron and particle fluence.

- RF launchers and antennas

- True steady-state gyrotons and klystrons.

- Neutral beam pumping and damage from neutron streaming

- Control coils behind the blankets
The Critical Issues of Plasma Wall Interactions and Plasma Facing Components Will Be Squarely Addressed in FDF

- **Hot wall operation**, above 400°C at least, will be an entirely new regime.
- **Erosion**
  - Biggest step FDF makes is in on time, $10^7$ seconds per year, and hence neutron and particle fluence. (100 times ITER).
  - Divertor gross erosion estimates range from mm (W) to cms (C) per year.
  - Tons of material per year will erode and redeposit. Properties?
  - Solutions? Tungsten? Detached divertor? Others?
- **Tritium Retention**
  - Cannot be allowed to prevent TBR > 1
  - Even for carbon, codep estimates (very temperature dependent) < 0.5 kg retained in two weeks, then quickly removed by oxygen baking
- **Heat Flux Handling**
  - Axisymmetric maintenance scheme allows precision alignment of surfaces to hide edges and allow maximal use of flux expansion, progressing through the “x-divertor, snowflake divertor, super x-divertor.”
  - High density helps promote radiation and containment of neutrals in the divertor, helping separate core and divertor plasmas (super-x?)
- **Other issues**
  - Fast plasma shutdowns, first wall heat fluxes in fault conditions, EM forces
FDF is viewed as a direct follow-on of DIII-D (50% larger) and Alcator Cmod, using their construction features:

- Plate constructed copper TF Coil which enables...
- TF Coil joint for complete disassembly and maintenance
- OH Coil wound on the TF Coil to maximize Volt-seconds
- High elongation, high triangularity double null plasma shape for high gain, steady-state
- Red blanket produces net Tritium
The Baseline Maintenance Scheme is Toroidally Continuous Blanket Structures

Remove
- Upper sections of TF
- Divertor coil
- Top of vacuum vessel

Access to blanket structure obtained
- Blanket segments removed as toroidally continuous rings

Benefits
- Blankets strong for EM loads
- Toroidal alignment assured

Difficulties
- Provision of services (coolants) to blanket rings near the midplane through blankets above
Two Options Being Considered for TF Coil Joint: C-mod Type Sliding and Sawtooth Joint (Rebut)

Model with Cover Weldment and Sliding

Model with Rebut Toothed Engagement of Horizontal and Vertical TF Legs, Compression Ring Holds it Together
The Baseline Operating Modes of FDF are more Conservative than ARIES-AT

<table>
<thead>
<tr>
<th>ITEM</th>
<th>FDF (0-D)</th>
<th>ARIES-AT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Aspect Ratio</td>
<td>3.5</td>
<td>4.0</td>
</tr>
<tr>
<td>Elongation $\kappa$</td>
<td>2.31</td>
<td>2.21</td>
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<tr>
<td>Fraction of $\kappa$ Limit*</td>
<td>0.95</td>
<td>0.95</td>
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<tr>
<td>Triangularity $\delta$</td>
<td>0.7</td>
<td>0.78</td>
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<tr>
<td>Fraction of the Beta Limit+</td>
<td>67%</td>
<td>104%</td>
</tr>
<tr>
<td>$\beta_N$</td>
<td>3.69</td>
<td>5.4</td>
</tr>
<tr>
<td>$C_{bs} = f_{bs}/(\beta_p/A^{1/2})$</td>
<td>0.6 (&gt; 0.8 ONETWO)</td>
<td>0.784</td>
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<tr>
<td>Peak Midplane Neutron Wall Loading (MW/m$^2$)</td>
<td>2.0</td>
<td>4.8</td>
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<tr>
<td>Neutron Wall Loading Peaking Factor</td>
<td>1.3</td>
<td>1.45</td>
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<tr>
<td>$Z_{eff}$</td>
<td>2.10</td>
<td>1.99</td>
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</table>
Issues to be Addressed in Rebaselining FDF

- Inner and Outer Gaps (+5-10 cm)
- Inboard blanket/shield (+10 cm?)
- OH coil - ceramic, concrete insulator?
- Outboard blanket/shield (+30 cm?), PF shielding
- Vertical stability with PF further away, smaller PF?
- Realistic divertor geometry adds machine height
- More shielding/breeding in divertor region, associated pumping ducts
- Increase Cbs for more bootstrap per $\beta_p$?
- Increase fraction of beta limit?
- Less conservative impurity assumptions?
- More ECH and less (no?) NBI? Smaller ports?
- Compromise pedestal to get LHCD into $\rho \sim 0.7$?
- Is sector maintenance scheme a real possibility?
## Changes Made in Rebaselining FDF

<table>
<thead>
<tr>
<th>Issue</th>
<th>Change Made</th>
</tr>
</thead>
<tbody>
<tr>
<td>No inner and outer gaps, plasma to wall</td>
<td>Inner gap $7\quad \lambda_q$ Loarte (5 cm)</td>
</tr>
<tr>
<td></td>
<td>Outer gap $11\quad \lambda_q$ Loarte (8 cm)</td>
</tr>
<tr>
<td>50 cm inboard blanket/shield requires OH solenoid made with ceramic</td>
<td>Add 10 cm inboard shielding (Sawan) to enable organic insulators in OH and TF</td>
</tr>
<tr>
<td>insulators</td>
<td>OH coil – ceramic insulators</td>
</tr>
<tr>
<td>Outboard 50 cm blanket/shield inadequate</td>
<td>80 cm outboard blanket/shield</td>
</tr>
<tr>
<td></td>
<td>Impact on vertical stability from PF coils further away to be assessed</td>
</tr>
<tr>
<td>Outer PF Coil Currents</td>
<td>Corrected (lowered) for real equilibria and moving them further out (increased)</td>
</tr>
<tr>
<td>Need realistic divertor geometry</td>
<td>Highly angled divertor plate and pumping duct included – increased machine height</td>
</tr>
<tr>
<td>Inadequate divertor coil shielding (25 cm)</td>
<td>50 cm divertor blanket/shield increased machine height.</td>
</tr>
<tr>
<td>Midplane port shielding of PF coils</td>
<td>Coils moved apart. Shielding added.</td>
</tr>
<tr>
<td>ONETWO gives more bootstrap fraction than spreadsheet, $f_{bs} = C_{bs}$ $\beta_p$ / $A^{1/2}$</td>
<td>Increase $C_{bs}$ from 0.6 to 0.75. Also increase $f_{bs}$ from 60% to 75%</td>
</tr>
<tr>
<td>Pressure peaking factor 2 not optimal</td>
<td>Pressure peaking 2.25 as per stability optimization (Garofalo). Increase density peaking to 1.5</td>
</tr>
<tr>
<td>Helium fraction 10% fixed assumption</td>
<td>Helium fraction set 5%, the estimate from alpha production rate and conservative $T_{He^+}/T_E = 10$.</td>
</tr>
<tr>
<td>Current Drive efficiencies too approximate</td>
<td>Calibrate current drive efficiencies to ONETWO calculations</td>
</tr>
<tr>
<td>Fraction of beta limit 67%</td>
<td>No change</td>
</tr>
<tr>
<td>$H_{\text{ITER98y2}} = 1.6$ (conservative)</td>
<td>No change</td>
</tr>
<tr>
<td></td>
<td>Pre 8/09</td>
</tr>
<tr>
<td>------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>A aspect ratio</td>
<td>3.5</td>
</tr>
<tr>
<td>a plasma minor radius</td>
<td>m 0.71</td>
</tr>
<tr>
<td>Ro plasma major radius</td>
<td>m 2.49</td>
</tr>
<tr>
<td>k plasma elongation</td>
<td>2.31</td>
</tr>
<tr>
<td>DeltaRb Radial Build Inner Blanket</td>
<td>m 0.5</td>
</tr>
<tr>
<td>Jc centerpost current density</td>
<td>MA/m² 16.7</td>
</tr>
<tr>
<td>Bo field on axis</td>
<td>T 6.01</td>
</tr>
<tr>
<td>TF Stress Stress in TF Coil</td>
<td>MPA 276</td>
</tr>
<tr>
<td>OH Stress Stress on OH Coil</td>
<td>MPA 228</td>
</tr>
<tr>
<td>Pf fusion power</td>
<td>MW 245.94</td>
</tr>
<tr>
<td>Pinternal power to run plant</td>
<td>MW 546.38</td>
</tr>
<tr>
<td>Qplasma Pfusion/Paux</td>
<td>4.17</td>
</tr>
<tr>
<td>Pn/Awall Neutron Power at Blanket</td>
<td>MW/m² 2.00</td>
</tr>
<tr>
<td>BetaT toroidal beta</td>
<td>0.058</td>
</tr>
<tr>
<td>BetaN normalized beta</td>
<td>mT/MA 3.69</td>
</tr>
<tr>
<td>fbs bootstrap fraction</td>
<td>0.60</td>
</tr>
<tr>
<td>Pcd current drive power</td>
<td>MW 58.95</td>
</tr>
<tr>
<td>Ip plasma current</td>
<td>MA 6.71</td>
</tr>
<tr>
<td>q safety factor</td>
<td>5.00</td>
</tr>
<tr>
<td>Ti(0) Ion Temperature</td>
<td>keV 19.08</td>
</tr>
<tr>
<td>Te(0) Electron Temperature</td>
<td>keV 19.08</td>
</tr>
<tr>
<td>n(0) Electron Density</td>
<td>E20/m³ 3.01</td>
</tr>
<tr>
<td>nbar/nGR Ratio to Greenwald Limit</td>
<td>0.57</td>
</tr>
<tr>
<td>Zeff</td>
<td>2.10</td>
</tr>
<tr>
<td>W Stored Energy in Plasma</td>
<td>MJ 69.71</td>
</tr>
<tr>
<td>Paux Total Auxiliary Power</td>
<td>MW 58.95</td>
</tr>
<tr>
<td>TauE TauE</td>
<td>sec 0.64</td>
</tr>
<tr>
<td>H H factor over 89P L-mode</td>
<td>2.92</td>
</tr>
<tr>
<td>HITER98Y2 H factor over ELMY H</td>
<td>1.60</td>
</tr>
<tr>
<td>Hpetty H factor over ELMY H</td>
<td>1.22</td>
</tr>
</tbody>
</table>
FDF Configuration and Dimensions Post 8/09

- TF CENTER POST
- TF1 1.42 x 0.308
- COOLING MANIFOLD
- DIVERTOR PUMPING DUCT
- OUTER BLANKET SHIELD 0.8 THK
- SHIELDING
- PF2 0.020 x 0.020
- PF3 0.065 x 0.063
- INNER BLANKET 0.6 THK
- R 2.7
- 5.17
- 6.30
- α = 0.77
FDF Made Substantial Input into the ReNeW Process

- Prepared and submitted 17 white papers for the Workshops on Harnessing Fusion Power and Taming the Plasma Material Interface (UCLA) and the Workshops on Understanding Burning Plasmas and Creating Predictable High Performance Steady-State Plasmas (GA).
- Participated in some white papers submitted by ORNL.
- At the UCLA workshops we made 8 oral presentations. We made additional presentations at the meetings at GA.
- Participated in the various ReNeW panel deliberations
- Participated in the ReNeW Workshop at Bethesda, June 8-12, 2009.
Theme I

“Research Thrusts made Possible by a Fusion Development Facility,” (Exec. Summary), R.D. Stambaugh, V.S. Chan, A.M. Garofalo, J.P. Smith, and C.P.C. Wong


“Active Realtime Control Issues and Role of a Fusion Development Facility,” D.A. Humphreys


“Tearing Mode Avoidance and Stabilization,” R.J. La Haye

“Creating a Self-heated Plasma,” V.S. Chan

Theme II

“Research Thrusts made Possible by a Fusion Development Facility,” (Exec. Summary), R.D. Stambaugh, V.S. Chan, A.M. Garofalo, J.P. Smith, and C.P.C. Wong


“Active Realtime Control Issues and Role of a Fusion Development Facility,” D.A. Humphreys

“Validated Theory and Predictive Modeling,” V.S. Chan

“Integration of High-performance, Steady-state Burning Plasmas,” A.M. Garofalo


“A Fusion Development Facility to Test Heating and Current Drive Systems for DEMO,” R. Prater and V.S. Chan
Theme III

“Research Thrusts made Possible by a Fusion Development Facility,” (Exec. Summary), R.D. Stambaugh, V.S. Chan, A.M. Garofalo, J.P. Smith, and C.P.C. Wong


“Taming the Plasma Material Interface RF Antennas, Launching Structures, and Other Internal Components,” R.W. Callis

Theme IV

“Research Thrusts made Possible by a Fusion Development Facility,” (Exec. Summary), R.D. Stambaugh, V.S. Chan, A.M. Garofalo, J.P. Smith, and C.P.C. Wong


“Closing the Fusion Fuel Cycle,” C.P.C. Wong, R.D. Stambaugh, and M. Sawan

“Making Electricity and Hydrogen – the FDF Port Test Blanket Module Program,” R.D. Stambaugh, C.P.C. Wong, and V.S. Chan

Fusion Nuclear Science Thrust to establish fusion fuel self-sufficiency and develop systems to extract fusion power reliably, efficiently, safely and with minimal environmental impact:

- Create models and perform small-scale validation experiments for components and sub-scale systems needed for efficient (i.e., high temperature) and reliable power handling and extraction (non-nuclear experiments and facilities).
- Create, model and test techniques, components, and facilities needed to reliably, efficiently, and safely fuel the fusion plasma. Including tritium recovery from the plasma chamber and breeding materials.
- Develop and test autonomous remote maintenance approaches, techniques and equipment capable of operating in a fusion nuclear environment.
- Understand failure modes and conduct reliability growth program.
- Utilize ITER burning plasma to perform integrated experiments on first wall/breeding blanket systems in a fusion environment (i.e., TBM).
- Address and resolve key FNS research needs in a fully integrated fusion nuclear science facility (FNSF):
  - Define research requirements and mission, evaluate alternatives through first stages of design and R&D, and select best Demo-relevant option(s).
  - Design, build and operate FNSF to mature science and technology base in preparation for Demo.
Suggested Top Level Research Thrusts to Come Out of ReNeW

- Develop fusion’s energy applications.
- Close the fusion fuel cycle.
- Demonstrate the production of high grade process heat from fusion.
- Show electricity production from fusion.
- Develop high temperature blankets for high efficiency electricity production.
- Show hydrogen production from fusion.
- Show significant fusion power can be produced with a significant duty factor.
- Develop and perform irradiation tests on low activation, high strength, high temperature materials for long lived fusion blankets.
Suggested Top Level Research Thrusts Made Possible By a Fusion Development Facility.

- Show high performance, steady-state, burning plasmas operating for weeks.
- Develop plasma measurements suitable for a fusion power plant.
- Show control of high performance plasmas for weeks.
- Show significant fusion power can be produced with a significant duty factor.
- Show avoidance and mitigation of off-normal events.
- Develop auxiliary systems for true steady-state.
- Develop fusion’s energy applications.
- Demonstrate fusion fuel self-sufficiency.
- Demonstrate the production of high grade process heat from fusion.
- Show electricity production from fusion.
- Develop high temperature blankets for high efficiency electricity production.
- Show hydrogen production from fusion.
- Develop and perform irradiation tests on low activation, high strength, high temperature materials for long lived fusion blankets.
The Critical Issues of Plasma Wall Interactions and Plasma Facing Components Will Be Squarely Addressed in FDF

- **Hot wall operation**, above 400°C at least, will be an entirely new regime.
- **Erosion**
  - Biggest step FDF makes is in on time, $10^7$ seconds per year, and hence neutron and particle fluence. (100 times ITER).
  - Divertor gross erosion estimates range from mm (W) to cms (C) per year.
  - Tons of material per year will erode and redeposit. Properties? Solutions? Tungsten? Detached divertor? Others?
- **Tritium Retention**
  - Cannot be allowed to prevent TBR $> 1$
  - Even for carbon, codep estimates (very temperature dependent) $< 0.5$ kg retained in two weeks, then quickly removed by oxygen baking
- **Heat Flux Handling**
  - Axisymmetric maintenance scheme allows precision alignment of surfaces to hide edges and allow maximal use of flux expansion, progressing through the “x-divertor, snowflake divertor, super x-divertor.”
  - High density helps promote radiation and containment of neutrals in the divertor, helping separate core and divertor plasmas (super-x?)
- **Other issues**
  - Fast plasma shutdowns, first wall heat fluxes in fault conditions, EM forces
Emerging Double Null Divertor Concept in FDF

- Structures impede the mobility of neutrals away from the divertor target area and ExB flows that couple the outer and inner divertors.

- Up/down symmetric design, allowing pumping from outboard side.

- Tilted divertor plate and pumping access.

- Precision Toroidal alignment enabled by vertical maintenance scheme allows maximal use of flux expansion, even to the SX divertor.

FDF equilibrium

359-06/RDS/rs
A Closed Geometry to Best Take Advantage of Radiative Dissipation with Tilted Divertor

- Closure to impede mobility of neutrals away from divertor and facilitate pumping
  - UEDGE modeling showed bad synergy between tilted plate and impurity radiation in open divertor
- Up/down symmetric design, allowing pumping from both private flux region and outboard side
  - Optimal design may not be exactly up/down symmetric because of effect of ExB drifts
- Tilted divertor plate
  - Angle between magnetic field line and divertor tile surface ~1°
  - Divertor must present a perfectly toroidally flat surface to field lines
Peak Divertor Heat Flux < 10 MW/m$^2$ Estimated Using Multi-machine, Multi-parameter Scaling

- **Estimate of exponential power flux width:**
  - Integral power flux width: $\lambda_q$ (Loarte 1999)
  - Exponential heat flux width $\lambda_{\text{exp}} = 0.5 \times \lambda_q$

- **Estimate of peak heat flux at the divertor target**
  (assuming zero SOL and divertor radiation):

\[
Q_{\text{out}}^{\text{peak}} = \frac{(P_{\text{heat}} - P_{\text{rad}})F_{\text{OD}} \sin \theta}{N_D 2\pi R_D F_X \lambda_{\text{exp}}}
\]

- Radiated power in the core $P_{\text{rad}} = 54\% P_{\text{heat}}$
- Number of divertors $N_D = 2$ for double-null operation
- Fraction of power to outer divertor $F_{\text{OD}} = 0.8$
- Poloidal tilt angle between field lines and divertor plate $\theta = 10^\circ$
- Major radius at divertor strike point, $R_D = 2.3$ m
- Flux expansion at divertor strike point, $F_X = 4$

**FDF 2 MW/m$^2$ case**
(baseline)

$P_{\text{heat}} = 108$ MW, $B_T = 6$ T, $q_{95} = 5$

$\lambda_{\text{exp}} = 7.2$ mm

\[
Q_{\text{out}}^{\text{peak}} = 8.3$ MW/m$^2$
SOLPS Modeling Shows Strong Reduction of Peak Heat Flux with Increasing Gas Puff

- **Low gas-puff case** is conduction-limited case to compare heat flux width to empirical scalings
  - Gas-puff raised just until point where there is some parallel $T_e$ gradient
  - $\lambda_{\exp,OMP} \sim 3$ mm
  - $T_{e,OSP} \sim 150$ eV, $P_{rad}=16$ MW

- **Medium gas puff case**
  - $\lambda_{\exp,OMP} \sim 3$ mm
  - $T_{e,OSP} \sim 12$ eV, $P_{rad}=33$ MW

- **High gas puff case (nearly detached)**
  - $\lambda_{\exp,OMP} \sim 10$ mm
  - $T_{e,OSP} \sim 2$ eV, $P_{rad}=39$ MW
## A Number of Factors Can Alter the Peak Heat Flux From Simple Field Line Following Estimates

<table>
<thead>
<tr>
<th>Factor</th>
<th>Increase/Decrease</th>
</tr>
</thead>
<tbody>
<tr>
<td>Increased Core Plasma Radiation</td>
<td>Down 1.3x</td>
</tr>
<tr>
<td>SOL and Divertor Radiation</td>
<td>Down 4-5x</td>
</tr>
<tr>
<td>Increasing Flux Expansion at the Target Plate, “X” Divertor</td>
<td>Down 2x</td>
</tr>
<tr>
<td>Snowflake Type Divertor</td>
<td>Down 1.6x</td>
</tr>
<tr>
<td>Super X Divertor</td>
<td>Down at least 2x</td>
</tr>
<tr>
<td>Slow Rotation of the Distorted Heat Flux Pattern Produced by the RMP Coils to Suppress ELMs</td>
<td>Down 1.5x</td>
</tr>
<tr>
<td>Less Plate Tilt Angle, perhaps 20 deg</td>
<td>Up 2x</td>
</tr>
<tr>
<td>Shorter Midplane Heat Flux Falloff Length by Factor 2</td>
<td>Up 2x</td>
</tr>
<tr>
<td>Excursions in Control from DN to SN</td>
<td>Up 1.6x</td>
</tr>
</tbody>
</table>

### Maximum cumulative potential increase of order a factor 6
### Maximum cumulative potential reduction of order a factor 18
- Considering Snowflake, X, and SX divertor options as forms of flux expansion
DIII-D Shows Radiative Dissipation and Stochastic Edge Viable Approaches to Reducing Peak Heat Flux

- Divertor radiation preferentially enhanced using puff and pump technique
  - P_{rad}/P_{NBI} \sim 60\% \text{ with } Z_{eff} \sim 2.0
    - Argon and D2 injection
  - \beta_N=2.6, H_89=2.0, G=0.4 maintained
    - Experiments thus far focused on puff and pump studies, not fusion performance

- Resonant Magnetic Perturbation splits strike points
- Angular rotation of the RMP will result in a time averaged broadening of the OSP footprint
Precision Alignment Enables Full Use of Divertor Plate Tilting and Flux Expansion

- Divertor can be built in a factory setting as a single precision toroidal ring with precision placement of tiles on it
- Slight fish-scaling of the tiles may be done to hide edges
- Problem of heat flux handling becomes axisymmetric
- Because of precision alignment, infrared thermography pictures of the divertor in DIII-D show near-total axisymmetry
DIII-D Will Carry Out Tests of Various Advanced Divertor Approaches for Increasing Flux Expansion

X-divertor coils being evaluated for DIII-D


DIII-D Snowflake Divertor experiment proposed for FY09
M.V. Umansky, et al., LLNL

DIII-D Super-X Divertor experiment proposed for FY10
P. Valanju et al., IFS, Univ. of Texas

DIII-D plasma with 2nd x-point set 5 cm below primary x-point

$I/I_{\text{max}} = 92\%$

75% 53%

SXD obtained for FDF with just one extra PF coil

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