Fusion-Fission Hybrid Activities in ASIPP

Presented by Yican Wu

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Contents

1. Necessity
2. Fusion Driver Design
3. Blanket Design
4. R & D
5. Plan
Energy problem Currently in China

- China Population is ~1.3 billion, Average energy consumption per person is < 1/2 of the world level, < 1/10 of the developed country’s level.

- Fast development of economy at annual rate of 8~10 % has been kept for > 20 yeas (this year ~8%)

- China has been the 2nd largest energy producing and consumption country, and the 2nd largest CO$_2$ producer in the world
Energy problem future in China

- Population will be 1.5 billion at 2050, Conservatively predicted capacity of electricity will be 1200~1500 GWe
- China will be the 1st largest CO2 producer at 2025.

Serious shortage of energy resources ???
Serious pollution of environment ???

Renewable energy + Nuclear Energy
Fission power development and new problem

(Current Plan on Fission, China)

Policy: Develop nuclear power as fast as possible

- **2008:**
  - 9.1GW (~2% of total capacity, in operation)
  - 25.4GW under construction

- **2020:** 40GW (4% of total) → 70~100GW (new plan)
  - >3 new units to be constructed per year from now to 2020

- **~2050:** 240GW (20% of total)

Nuclear fuel supply?
Radioactive waste disposal?
Safety problem?
Fission power development and new problem
(Prediction on Future Fission, China)

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Ratio A</th>
<th>Ratio B</th>
<th>Nucl. Power</th>
<th>Capacity (Approximate Scale)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Low Level</td>
<td>10%</td>
<td>6%</td>
<td>120GW</td>
<td>Double in France</td>
</tr>
<tr>
<td>Mid. Level</td>
<td>20%</td>
<td>12%</td>
<td>240GW</td>
<td>Sum in US, France and RF</td>
</tr>
<tr>
<td>High Level</td>
<td>30%</td>
<td>18%</td>
<td>360GW</td>
<td>Sum all over the world</td>
</tr>
</tbody>
</table>

A: fraction of nucl. power in total electricity capacity
B: fraction of nucl. power in total primary energy capacity

Nuclear fuel supply ?
Radioactive waste disposal ?
Safety problem ?
Fusion status and its long road to go

• **Current:** EAST/HL2A, KSTAR, MAST, … (~2020)

• **Near Future:** ITER/IFMIF/CTF… (2020~2040)

• **Far Future:** fast/ultra-fast track to DEMO (???~2050?)

Fusion has a very good progress, but still needs hard work to economical utilization:

1. feasible to seek for near-term applications
2. necessary to find out near-term applications
Fusion-Driven Hybrid Multi-Functional Reactor

Fusion Core + Subcritical Fission Blanket

Functions:

- waste transmutation
- fuel breeding
- energy production
- material test
- other applications
Potential Advantages of Hybrids

- **Lower requirement on plasma-related parameters**
  (improved energy balance by fission blanket)
- **Rich neutrons to achieve multi-goals**
  (improved neutron balance by fusion neutrons)
- **Good passive and inherent safety performances**
  (subcritical)
- **Avoidance of nuclear proliferation**
  (large design margin because of subcritical features)
- **Benefit both fusion and fission**
  (fill in the gap, promote fusion, solve left problems by fission)
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FDS Series Fusion Reactors & Blankets
Conceptual Design for Plants

**FDS-I: Fusion-driven Subcritical System**
for early applications of fusion (multi-function)
e.g. waste transmutation, fuel breeding etc.

**FDS-II: Fusion Power Reactor**
for highly efficient electricity generation

**FDS-III: High Temperature Fusion Reactor**
for advanced applications, e.g. hydrogen production

**FDS-ST: Spherical Tokamak-based Reactor**
for exploiting and assessing innovative conceptual path
# Core Plasma Parameters for Plants

<table>
<thead>
<tr>
<th>Parameters</th>
<th>FDS-I</th>
<th>FDS-II</th>
<th>FDS-III</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fusion power (MW)</td>
<td>150</td>
<td>2500</td>
<td>2600</td>
</tr>
<tr>
<td>Major radius (m)</td>
<td>4</td>
<td>6</td>
<td>5.1</td>
</tr>
<tr>
<td>Minor radius (m)</td>
<td>1</td>
<td>2</td>
<td>1.7</td>
</tr>
<tr>
<td>Aspect ratio</td>
<td>4</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td>Plasma elongation</td>
<td>1.78</td>
<td>1.9</td>
<td>1.9</td>
</tr>
<tr>
<td>Triangularity</td>
<td>0.4</td>
<td>0.6</td>
<td>0.47</td>
</tr>
<tr>
<td>Toroidal magnetic field on axis (T)</td>
<td>6.1</td>
<td>5.93</td>
<td>8.0</td>
</tr>
<tr>
<td>Safety factor / q-95</td>
<td>3.5</td>
<td>5.0</td>
<td>8.03</td>
</tr>
<tr>
<td>Plasma current (MA)</td>
<td>6.3</td>
<td>15</td>
<td>16</td>
</tr>
<tr>
<td><strong>Avg. neutron wall load (MW/m²)</strong></td>
<td><strong>0.49</strong></td>
<td><strong>2.72</strong></td>
<td><strong>4</strong></td>
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<tr>
<td>Average surface heat load (MW/m²)</td>
<td>0.1</td>
<td>0.54</td>
<td>1.04</td>
</tr>
<tr>
<td>Fusion gain</td>
<td>3</td>
<td>31</td>
<td>32</td>
</tr>
<tr>
<td>Normalized $\beta^*_N$ (%)</td>
<td>3</td>
<td>5</td>
<td>4.8</td>
</tr>
</tbody>
</table>
Re-evaluate the performances of fusion-fission hybrid reactors

A hybrid reactor for energy production: FDS-EM
A hybrid reactor for fuel breeding: FDS-FB
A hybrid reactor for waste transmutation: FDS-WT

based on available or very limitedly extrapolated fusion and fission technologies

To define a Hybrid for next step

Y.Wu et al, Presented at the 3rd IAEA Technical Meeting on “First Generation of Fusion Power Plants - Design and Technology”
13 – 15 July 2009, IAEA HQ, Vienna, Austria
## Plasma Core Parameters for Next Facility

<table>
<thead>
<tr>
<th>Parameters</th>
<th>ITER</th>
<th>EAST</th>
<th>FDS-I</th>
<th>FDS-EM/FB/WT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fusion power (MW)</td>
<td>500</td>
<td>-</td>
<td>150</td>
<td>50</td>
</tr>
<tr>
<td>Major radius (m)</td>
<td>6.2</td>
<td>1.95</td>
<td>4</td>
<td>4</td>
</tr>
<tr>
<td>Minor radius (m)</td>
<td>2</td>
<td>0.46</td>
<td>1</td>
<td>1</td>
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<tr>
<td>Aspect ratio</td>
<td>3.1</td>
<td>4.2</td>
<td>4</td>
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<tr>
<td>Plasma elongation</td>
<td>1.85</td>
<td>1.8</td>
<td>1.78</td>
<td>1.7</td>
</tr>
<tr>
<td>Triangularity</td>
<td>0.33</td>
<td>0.45</td>
<td>0.4</td>
<td>0.45</td>
</tr>
<tr>
<td>Toroidal magnetic field on axis (T)</td>
<td>5.3</td>
<td>3.4-4.0</td>
<td>6.1</td>
<td>5.1</td>
</tr>
<tr>
<td>Safety factor / q-95</td>
<td>3</td>
<td>-</td>
<td>3.5</td>
<td>2.03</td>
</tr>
<tr>
<td>Plasma current (MA)</td>
<td>15</td>
<td>1.5</td>
<td>6.3</td>
<td>6.1</td>
</tr>
<tr>
<td>Average neutron wall load (MW/m²)</td>
<td>0.57</td>
<td>-</td>
<td>0.49</td>
<td>0.17</td>
</tr>
<tr>
<td>Average surface heat load (MW/m²)</td>
<td>0.27</td>
<td>0.1-0.2</td>
<td>0.1</td>
<td>0.1</td>
</tr>
<tr>
<td>Fusion gain</td>
<td>&gt;10</td>
<td>3</td>
<td>3</td>
<td>0.95</td>
</tr>
<tr>
<td>Normalized beta, ( \beta_N ) (%)</td>
<td>2.5</td>
<td>-</td>
<td>3</td>
<td>3</td>
</tr>
</tbody>
</table>
Contents

1. Necessity
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**Water-cooled Blanket Concept**

**Fission Fuel Zone**
- Coolant: Water
- Flow scheme: Poloidally
- Fuel style: Fuel Pin
- PWR technology

**Tritium Breeding Zone**
- Coolant: LiPb
- Flow scheme: Poloidally
- LiPb self-cooled
Helium-cooled Blanket Concept

Waste Fuel Zone
Coolant: He
Flow scheme: Radially-Poloidally-Radially
Fuel style: Concentric rod

Uranium Fuel Zone
Coolant: He
Flow scheme: Radially-Poloidally-Radially
Fuel style: Pebble bed

Schematic view of outboard blanket

Concentric rod

Pebble bed
He/LiPb Dual-cooled Blanket Concept

Blanket design --- high energy multiplication
Emphasis on circulating particle or pebble bed fuel configurations considering geometry complexity of tokamak, frequency of fuel discharge and reload

Concept options:
DWT-CPL: the He&LiPb DWT blanket with Carbide heavy nuclide Particle fuel in circulating Liquid LiPb coolant.
DWT-OPG :Oxide heavy nuclide pebble bed fuel in circulating helium-Gas
DWT-NPG: Nitride heavy nuclide Particle fuel in circulating He-Gas.

**DWT-CPL:** The AC appears in the form of the TRISO(TRi-ISOtropic)-like carbide particles coated with SiC suspending in the LiPb slurry. The circulating fuel form has the advantages of good compatibility with complex geometry, easy control of fuel cycle and fast response to emergency fuel removal etc.

# Initial Characteristics

(Hybrids: FDS-EM /-FB /-WT)

<table>
<thead>
<tr>
<th>Neutron source energy</th>
<th>D-T neutron 14MeV</th>
</tr>
</thead>
<tbody>
<tr>
<td>Neutron source intensity</td>
<td>1.7781E+19 n/s</td>
</tr>
<tr>
<td></td>
<td>5.3343E+19 n/s</td>
</tr>
<tr>
<td></td>
<td>1.7781E+20 n/s</td>
</tr>
<tr>
<td>Fusion power</td>
<td>50 MW</td>
</tr>
<tr>
<td></td>
<td>150 MW</td>
</tr>
<tr>
<td></td>
<td>500 MW</td>
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</table>

<table>
<thead>
<tr>
<th>Fusion power</th>
<th>Water-cooled</th>
<th>Fuel type (in Fuel zone)</th>
<th>PuO2, MAO2, UO2 (rod, PWR-fuel-like)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fusion power</td>
<td>He-LiPb dual coolants (FDS-I)</td>
<td>PuO2, MAO2, UO2 (particle, HTGR-fuel-TRISO-like)</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Tritium breeder</th>
<th>LiPb</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>Coolant</th>
<th>Water Helium gas He-LiPb dual coolants (FDS-I)</th>
</tr>
</thead>
</table>
### FDS-EM Design Constraints and Objectives

<table>
<thead>
<tr>
<th>Items</th>
<th>Constraints and Objectives</th>
</tr>
</thead>
<tbody>
<tr>
<td>$K_{\text{eff}}$</td>
<td>$\leq 0.95$ (safety margin limit)</td>
</tr>
<tr>
<td>$P_{d_{\text{max}}}$ (MW/m3)</td>
<td>$\leq 100$ (cooling capability limit)</td>
</tr>
<tr>
<td>TBR</td>
<td>$\geq 1.05$ (tritium sustainability requirement)</td>
</tr>
<tr>
<td>Energy Multiplication ($M$)</td>
<td>Reasonable Power Output</td>
</tr>
<tr>
<td></td>
<td>$\sim 90$ for $P_{fu}=50$MW</td>
</tr>
<tr>
<td></td>
<td>$\sim 30$ for $P_{fu}=150$MW</td>
</tr>
<tr>
<td></td>
<td>$\sim 9$ for $P_{fu}=500$MW</td>
</tr>
</tbody>
</table>
**FDS-FB Design Constraints and Objectives**

<table>
<thead>
<tr>
<th>Items</th>
<th>Constraints and Objectives</th>
</tr>
</thead>
<tbody>
<tr>
<td>Keff</td>
<td>( \leq 0.95 ) (safety margin limit)</td>
</tr>
<tr>
<td>( P_{d_{\text{max}}}(\text{MW/m}^3) )</td>
<td>( \leq 100 ) (cooling capability limit)</td>
</tr>
<tr>
<td>TBR</td>
<td>( \geq 1.05 ) (tritium sustainability requirement)</td>
</tr>
<tr>
<td>Breeding Fissile Pu (BSR)</td>
<td>Water-cooled/He-cooled</td>
</tr>
<tr>
<td></td>
<td>maximizing breeding</td>
</tr>
</tbody>
</table>

*BSR: Ratio of the fissile Pu mass bred by FDS-FB to the fissile Pu mass depleted by a referred PWR per year.*
# FDS-WT Design Constraints and Objectives

<table>
<thead>
<tr>
<th>Items</th>
<th>Design Constraints and Objectives</th>
</tr>
</thead>
<tbody>
<tr>
<td>$K_{\text{eff}}$</td>
<td>$\leq 0.95$</td>
</tr>
<tr>
<td>TBR (Tritium Breeding Ratio)</td>
<td>$\geq 1.05$</td>
</tr>
<tr>
<td>$P_{d_{\text{max}}}(\text{MW/m}^3)(\text{Zone-averaged})$</td>
<td>$\leq 100$</td>
</tr>
<tr>
<td>Fuel Inventory*</td>
<td>minimization while keep balance of LLMA/Pu</td>
</tr>
<tr>
<td>Transmutation Fraction(^{<strong>})/TSR(^{</strong>*})</td>
<td>** LLMA maximizing transmutation</td>
</tr>
<tr>
<td></td>
<td>** Pu maximizing transmutation</td>
</tr>
</tbody>
</table>

---

*from 3000MW\(_{th}\) PWR with fuel burned to 33 GW.D/T after 10 years decay, annual production of a referred typical PWR (e.g., LLMA:35kg; Pu:288kg; LLFP: 42kg)

— Transmutation fraction\(^{**}\): Percent of the waste mass transmuted by FDS-WT to the waste mass loaded into the FDS-WT per year

— TSR\(^{***}\): Ratio of the waste mass transmuted by FDS-WT to the waste mass produced by a referred PWR per year
Models for 1D / 2D / 3D Analyses
Objective Parameters’ Definitions

- **M: Blanket Energy Multiplication**
  
  Ratio of fission power produced by FDS-EM to the source neutron power (80% of fusion power in the deuterium-tritium fusion fuel cycle)

- **BSR: Breeding Support Ratio**
  
  Ratio of the fissile Pu mass bred by FDS-FB to the fissile Pu mass depleted (~400kg) by a referred PWR per year

- **TSR: Transmutation Support Ratio**
  
  Ratio of the waste mass transmuted by FDS-WT to the waste mass produced (Pu: 288kg; LLMA: 34.7kg; $^{137}$Cs:10kg; $^{129}$I: 5.96kg; $^{99}$Tc: 25.69 kg) by a referred PWR per year
Calculation and Analysis

FDS-EM & -FB & -WT

- Neutronics
- Thermalhydraulics
- Themo-mechanics

Y.Wu et al, Presented at the 3rd IAEA Technical Meeting on “First Generation of Fusion Power Plants - Design and Technology”
13 – 15 July 2009, IAEA HQ, Vienna, Austria
Calculation and Analysis

**FDS-I**

- Neutronics
- Thermalhydraulics
- Themo-mechanics
- Safety analysis (static & Transient)
- Economics
1. Three types of hybrid concepts i.e. EB, FB and WT are conceptually designed and re-evaluated based on available or very limitedly extrapolated fusion (i.e. a fusion power of 50~500MW) and fission technologies (i.e. Water-cooled PWR or He-cooled HTGR technologies).

2. The neutonics analyses showed the max. energy multiplication $M$ can be $\sim 100$, the max. fissile fuel breeding ratio $BSR$ can be $\sim 10$, the max. waste transmutation ratio $TSR$ can be $\sim 15$, depending on specific designs.

3. Preliminary thermalhydraulics/thermo-mechanics analyses have been carried out to assess the feasibility, and the results showed those designs can be conceptually achievable.

4. Further optimization of design scenarios/parameters, detailed engineering analysis are underway.
1. Operational states
   Normal operation
     Startup/Shutdown of the Reactor
   Anticipated operational occurrences (AOOs)
     Protected Plasma OverPower (PPOP)
     Unprotected /protected Transient OverPower (UTOP)

2. Accident conditions
   Within design basis accident (DBA)
     Unprotected Plasma OverPower (UPOP)
     Protected Loss of Flow Accident (LOFA)
     Protected Loss of Coolant Accident (LOCA)
     Protected Loss Of Heat Sink (LOHS)
   Severe accidents
     Unprotected Loss of Flow Accident (ULOFA)
     Unprotected Loss of Coolant Accident (ULOCA)
     Unprotected Loss Of Heat Sink (ULOHS)
     Collapse Accident (CA)
Conclusions

- The reactivity temperature coefficient is negative due to the fuel inventory decreased in the blanket while the coolant expanding.
- There is no severe accident occurred under any protected accident and UTOP and UPOP.
- For the ULOFA and ULOHS, the structure melting might cause the CA, but the supercriticality could be avoided if the number of collapsed blankets is not more than 3.
- A very reliable Emergency Fusion Power Shutdown System (EFPSS) is necessary.
- Design needs to be optimised to avoid supercriticality under any conditions if possible.

References

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R&D Activities – Fusion Core

1. EAST superconducting tokamak experiment
2. Blanket materials and TBM development
3. Design and analysis tools development
ASIPP
EAST
Experimental Advanced Superconducting Tokamak

Main Missions:
- To Investigate plasma physics of advanced steady-state operation modes
- To Establish technology basis of full superconducting tokamaks for future reactors

Main Parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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</thead>
<tbody>
<tr>
<td>$B_T$</td>
<td>3.5~4.0 T</td>
</tr>
<tr>
<td>$R_0$</td>
<td>1.7 m</td>
</tr>
<tr>
<td>$a/b$</td>
<td>0.4/0.8 m</td>
</tr>
<tr>
<td>$\Delta_{\text{max}}$</td>
<td>~ 2</td>
</tr>
<tr>
<td>$I_P$</td>
<td>1.~1.5 MA</td>
</tr>
<tr>
<td>H&amp;CD</td>
<td>10 ~ 15 MW</td>
</tr>
<tr>
<td>Diverter</td>
<td>Double &amp; single Null</td>
</tr>
<tr>
<td>Expected nT $\tau$</td>
<td>$\sim 10^{19-20}$ m$^3$ s kev</td>
</tr>
</tbody>
</table>

Long or steady-state operation

- Project approved in 1998
- Construction began in 2000
- First Plasma in 2006
EAST Divertor Configuration Discharge

Up to 60 sec Long pulse discharge with LHCD
# EAST Operation Plan

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>Ip (MA)</td>
<td>0.3-0.5</td>
<td>0.5-1</td>
<td>1-1.5</td>
</tr>
<tr>
<td>R (m)</td>
<td>1.85</td>
<td>1.85</td>
<td>1.85</td>
</tr>
<tr>
<td>a (m)</td>
<td>0.45</td>
<td>0.45</td>
<td>0.45</td>
</tr>
<tr>
<td>K</td>
<td>1.2-1.5</td>
<td>1.2-1.9</td>
<td>1.5-1.9</td>
</tr>
<tr>
<td>D</td>
<td>0.2-0.3</td>
<td>0.3-0.5</td>
<td>0.3-0.6</td>
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<tr>
<td>ICRF (MW)</td>
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<td>4/30-110MHz</td>
<td>4/30-110MHz</td>
</tr>
<tr>
<td>CW</td>
<td>4/20-70MHz</td>
<td>4/20-70MHz</td>
<td></td>
</tr>
<tr>
<td>LHCD (MW)</td>
<td>2/(2.45GHz)</td>
<td>4/2.45GHz</td>
<td>4/2.45GHz</td>
</tr>
<tr>
<td>10-1000s</td>
<td>4/4.6GHz</td>
<td>4/3.7GHz</td>
<td></td>
</tr>
<tr>
<td>NBI (MW) 10-100s</td>
<td>4/40-80keV</td>
<td>8/40-80keV</td>
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<tr>
<td>ECRH (MW) CW</td>
<td>4/140GHz</td>
<td>4/140GHz</td>
<td></td>
</tr>
<tr>
<td>t (s)</td>
<td>5-20s</td>
<td>5-400</td>
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<td>Diagnostics configuration</td>
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<td>PFC</td>
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<td></td>
</tr>
<tr>
<td>Internal coils</td>
<td></td>
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</tr>
<tr>
<td>TBM</td>
<td></td>
<td></td>
<td>EAST-TBM</td>
</tr>
</tbody>
</table>
**EAST's Possible Contributions to ITER**

- **EAST Team leading >70% China ITER Procuerment Package** although EAST is smaller than ITER, but both have similar technology basis and similar magnetic configuration.

- **EAST is an important pre-test platform for technologies and physics to ITER at least before ITER D-T plasma operation.**

- **EAST will make an important contribution to DEMO development** if it can achieve long pulse or SS operation with elongated divertor configuration and high performance plasma.

### Parameters

<table>
<thead>
<tr>
<th>Parameters</th>
<th>ITER</th>
<th>EAST</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total fusion power</td>
<td>500 – 700 MW</td>
<td>(10^{16}) D-D Neutrons S(^{-1})</td>
</tr>
<tr>
<td>Inductive pulse time</td>
<td>≥ 400 s (Q ≥ 10)</td>
<td>~ 10 s</td>
</tr>
<tr>
<td>No-inductive pulse time</td>
<td>1000–3000s (Q ~ 5)</td>
<td>~ 1000 s</td>
</tr>
<tr>
<td>Expected n T (\tau)</td>
<td>(10^{12–22}) m(^3) s kev</td>
<td>(10^{16–28}) m(^3) s kev</td>
</tr>
<tr>
<td>(B_1) (6.2 m)</td>
<td>5.3 T</td>
<td>3.5 - 4.0 T (1.7m)</td>
</tr>
<tr>
<td>(R_{\theta})</td>
<td>6.2 m</td>
<td>1.7 m (1.85m)</td>
</tr>
<tr>
<td>(a)</td>
<td>2.0 m</td>
<td>0.4 m (0.45m)</td>
</tr>
<tr>
<td>(\kappa_{st})</td>
<td>1.70 / 1.85</td>
<td>1.8 / 2.0</td>
</tr>
<tr>
<td>(\delta_{st})</td>
<td>0.33 / 0.49</td>
<td>0.30 / 0.60</td>
</tr>
<tr>
<td>(I_p)</td>
<td>15 (17) MA</td>
<td>1.0 (1.5) MA</td>
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<tr>
<td>Divertor Configuration</td>
<td>Single Null</td>
<td>Single &amp; Double Null</td>
</tr>
<tr>
<td>Auxiliary Heating / CD Power</td>
<td>73 – 110 MW</td>
<td>4 – 20 MW</td>
</tr>
</tbody>
</table>
R&D Activities - Blanket

1. EAST superconducting tokamak experiment
2. Blanket materials and TBM development
3. Design and analysis tools development
ASIPP

Development & Test Strategy of TBM

Stage I: Out-of-pile Test (1/3 size)
- Thermal convection loop
- Forced convection loop

- R&D on materials (RAFM, Coating and FCI) and fabrication technology
- Diagnostic and measurement
- Out-of-pile test of 1/3 mockup etc.
- MHD

Stage II: Test in EAST (1/2 size)
- LiPb/He system for TBM in EAST

- 1/2 mockup test in EAST
- EM and thermo-mechanics, partially neutronics performances
- Influence on plasma
- MHD

Stage III: Test in ITER (full size)
- LiPb/He system for TBM in ITER

- To confirm results of EM/Thermo-mechanics test in EAST,
- To test neutronics, tritium and integration performances in ITER
Material R&D and Out-of-pile Mockup Test

(1/3-Size TBM)

Objectives:
- Validation of the fabrication route and techniques
- Validation of mechanical performances
- Assessment of reliability and safety with regard to ITER standards.

Test Items:
- Leak and pressure test.
- MHD and heat removal from FW.
- Mock-up connected to LiPb loop
- Hydrogen control and extraction to simulate tritium extraction
- Irradiation performance in fission reactors.
Development of CLAM Steel

- **Compositions Design**
- **Fabrication techniques**
  - Heat of several hundred kg
- **Joining techniques**
  - Solid HIP diffusion bonding
  - Tungsten Inert Gas welding
  - Electron Beam welding
- **Coating technique**
- **Corrosion test**
- **Irradiation Test**
  - Neutron Irradiation
  - Electron Irradiation
  - Plasma Irradiation
  - Ion Irradiation

<table>
<thead>
<tr>
<th>Element</th>
<th>CLAM</th>
<th>EUROFER97</th>
<th>F82H</th>
<th>JLF-1</th>
<th>9C-2WVTa</th>
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</thead>
<tbody>
<tr>
<td>Cr</td>
<td>9.0</td>
<td>8.8</td>
<td>7.7</td>
<td>8.9</td>
<td>8.9</td>
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<tr>
<td>W</td>
<td>1.5</td>
<td>1.1</td>
<td>1.9</td>
<td>1.9</td>
<td>2.0</td>
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<td>Mo</td>
<td>0.45</td>
<td>0.37</td>
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<td>Cu</td>
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<td>0.068</td>
<td>0.023</td>
<td>0.064</td>
<td>0.06</td>
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<td>C</td>
<td>0.10</td>
<td>0.10</td>
<td>0.09</td>
<td>0.10</td>
<td>0.11</td>
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<tr>
<td>Si</td>
<td>&lt;0.10</td>
<td>&lt;0.05</td>
<td>0.10</td>
<td>0.24</td>
<td>0.21</td>
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<tr>
<td>Y</td>
<td>&lt;0.2</td>
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</tbody>
</table>

**Main Compositions**

- **Tensile properties of CLAM and other RAFMs at RT**

<table>
<thead>
<tr>
<th>Steel</th>
<th>Heat treatment(℃)</th>
<th>σ_b(MPa)</th>
<th>δ_b(%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CLAM(HEAT 0204) (2mm plate)</td>
<td>900℃/30min+760℃/90min</td>
<td>652</td>
<td>470</td>
</tr>
<tr>
<td>CLAM(HEAT 0408A) (Ø 12 bar)</td>
<td>1040℃/30min+760℃/90min</td>
<td>652</td>
<td>501</td>
</tr>
<tr>
<td>CLAM(HEAT 0408A) (Ø 12 bar)</td>
<td>980℃/30min+760℃/90min</td>
<td>669</td>
<td>514</td>
</tr>
<tr>
<td>CLAM(HEAT 0603A) (12mm plate)</td>
<td>980℃/30min+760℃/90min</td>
<td>669.8</td>
<td>560.8</td>
</tr>
<tr>
<td>EUROFER 97 (14mm plate)</td>
<td>980℃/30min+760℃/90min</td>
<td>652</td>
<td>537</td>
</tr>
<tr>
<td>F82H (15mm plate)</td>
<td>1040℃/40min+750℃/40min</td>
<td>669</td>
<td>548</td>
</tr>
</tbody>
</table>

**DBTT of CLAM:** about -102 ℃, 16 ℃ lower than JLF-1.
## Development of LiPb Loop Technology

### Operation/Fabrication/Design:

- **Thermal convection loop (compatibility)**
  - **DRAGON-I**: 316SS, 500°C (in operation)
  - **DRAGON-II**: Inconel, 700°C (in operation)
  - **DRAGON-III**: SiCf/SiC, 700-1000°C (in fabrication)

- **Forced convection loop (MHD)**
  - **DRAGON-IV**: 480-700°C (in fabrication)

- **Auxiliary system for TBM in EAST**
- **Auxiliary system for TBM in ITER**

### DRAGON-I

- **Design Objectives:**
  - Thermal convection loop (500°C)
  - Compatibility of CLAM and 316L steel with LiPb.

- **Main Parameters:**
  - Loop Size: 0.5m × 0.5m
  - Structural Material: SS316L
  - Inner-diameter: 22mm
  - LiPb Inventory: 1 liter
  - Temperature: 420 ~ 480°C
  - Flow rate: ~ 0.08m/s

### DRAGON-II

- **Design Objectives:**
  - High temperature (700°C) thermal convection loop
  - Pursue the manufacture of SiCf/SiC materials for fusion
  - Compatibility of SiCf/SiC with LiPb

- **Key issues:**
  - Fabrication of SiCf/SiC Loop
  - Connection technology
  - Heating method of high temperature (up to 1000°C)

### DRAGON-III

- **Design Objectives:**
  - High temperature (700-1000°C) thermal convection loop
  - Pursue the manufacture of SiCf/SiC materials for fusion
  - Compatibility of SiCf/SiC with LiPb

- **Key issues:**
  - Fabrication of SiCf/SiC Loop
  - Connection technology
  - Heating method of high temperature (up to 1000°C)

### DRAGON-IV

- **Design Objectives:**
  - MHD Experiment (>2T, 1m/s);
  - General corrosion in flowing LiPb: -800°C;
  - Stress corrosion: max~50kN.

- **Key issues:**
  - Structural Materials
  - LiPb Purification
  - Loop construction and measurement technology

### DRAGON-IV

- **Key issues:**
  - Material selection (700°C)
  - Fabrication technology
  - Smelting of Large quantity LiPb
Fabrication of SiC_f/SiC Composite

Requirements:
- Low / high thermal conductivity
- Low electrical conductivity
- Good compatibility with LiPb under elevated temp.
- Stable under neutron irradiation

Key issues:
- Fabrication of SiC_f/SiC pipe
- Fabrication of FCI
- Bonding technology of SiC_f/SiC composites

SiC_f/SiC composites were fabricated by CVI + PIP + CVD.

CVI—Chemical Vapor Infiltration
PIP—Polymer Infiltration and Pyrolysis
CVD—Chemical Vapor Infiltration

Fiber 3D braid preform
Connection of metal and SiC composite
SiC_f/SiC pipe

SiC_f/SiC Loop
Coatings and Corrosion Experiments

Coating fabrication

- CVD Coating
- CLAM Eurofer97

Flowing Experiment at 480 ℃

- EDX line scan of CLAM (before cleaned)
- A relative low concentration of Fe,Cr contrast to the matrix

- EDX line scan of CLAM (before cleaned)
- Similar concentration profile of Fe,Cr in the corrosion layer with 6000hrs

SEM image of Al₂O₃ coatings section by APS

- The properties of the Al₂O₃ coatings on CLAM:
  - Bond Strength: UTS~31.7MPa
  - Specific resistance: SR>1010 Ω·cm
  - Roughness: Ra~4 μ m
  - Microhardness: ~HV951 (200g)
  - Density Porosity: DP ~7.2%

- The phases of coatings on CLAM by APS are γ-Al₂O₃ and α-Al₂O₃

<table>
<thead>
<tr>
<th>Specimens</th>
<th>No</th>
<th>UTS (MPa)</th>
<th>A(%)</th>
</tr>
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<tbody>
<tr>
<td>Original</td>
<td>1#</td>
<td>450</td>
<td>19.5</td>
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<td></td>
<td>2#</td>
<td>440</td>
<td>20.5</td>
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<td>Corrosion</td>
<td>3#</td>
<td>433</td>
<td>17.0</td>
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<td>4#</td>
<td>430</td>
<td>15.0</td>
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</table>
Test in EAST

(1/2 Size-reduced TBMs)

Objectives:
- Preliminary validation of design codes and data
- Checking of feasibility & availability of auxiliary system
- FM Influence on Plasma

EAST-TBM Test in EAST:
- ElectroMagnetic performance
  (MHD pressure drop, influence on plasma)
- Thermo-mechanics/Thermofluid dynamics performances
- Partially neutronics performance (DD neutrons), Diagnostic instruments

<table>
<thead>
<tr>
<th>Device</th>
<th>EAST</th>
<th>ITER</th>
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<tbody>
<tr>
<td>Phase</td>
<td></td>
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<tr>
<td>R (m)</td>
<td>1.95</td>
<td>6.2</td>
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<td>A (m)</td>
<td>0.46</td>
<td>2</td>
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<td>Bt (T)</td>
<td>3.5-4.0</td>
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<td>Neutron rate (n/s)</td>
<td>$10^{15}$-$10^{17}$</td>
<td>$1.77\times10^{20}$</td>
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<tr>
<td>Avg.HF(MW/m²)</td>
<td>0.1-0.2</td>
<td>0.11</td>
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<tr>
<td>Port Size</td>
<td>0.97m x 0.53m</td>
<td>2.2m x 1.7m</td>
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<tr>
<td>Pulse (sec)</td>
<td>~1000</td>
<td>100-200</td>
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Test in ITER
(Full-Size TBM: solid and liquid TBM)

ITER Operation (Day 1) 2016

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<th>Year</th>
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</table>

In EAST:
- D-plasma(1000sec)
- FM-Influence on plasma
- Thermo-mechanics/Thermodynamics performances
- EM & MHD Test
- Corrosion and Coating Test
- Auxiliary system

EAST-TBM

EM-TBM

TT-TBM

IN-TBM

Mat. R&D

Input-of-tokomak

Integrated Test:
- On-line tritium extraction
- Synergic nuclear effects
- MHD integrated effects
- Reliability and Safety

Neutronics Test:
- Neutronics performance
- N-fluence & gamma spectra
- Nuclear heat
- T-breeding
- Code & Data

ThermoDynamics & Tritium Behavior Test:
- Tritium recovery
- FCI performance
- Thermodfluid MHD
- Code & Data

EectroMagnetic-Test:
- To confirm results of EM/Thermo-mechanics test in EAST
- Auxiliary system

In EAST:
- D-plasma(1000sec, 10^{15-17}\,\text{n/s})
- Thermo-mechanics/Thermodfluid dynamics performances
- FCI & MHD Test
- Partially neutronics
- Diagnostic instruments

High DT

Low DT

In EAST:
- D-plasma(1000sec, 10^{15-17}\,\text{n/s})
- Thermo-mechanics/Thermodfluid dynamics performances
- EM & MHD Test
- Corrosion and Coating Test
- Auxiliary system
R&D Activities - Design

1. EAST superconducting tokamak experiment
2. Blanket materials and TBM development
3. Design and analysis tools development
Design and Analysis Tools Development

1. Multi-functions neutronics calculation code system: VisualBUS
2. Multi-physics (neutronics, thermalhydraulics, MHD) coupling simulation codes: NTC/MTC
3. System Engineering (safety/economy) analysis codes: RiskA, SYSCODE
4. Fusion virtual assembly system: FVAS
5. Material database and component reliability database management system: FUMDS, RiskBase

- 50~100 man-years each program

Key Tools for Fusion/Fission Reactor Design & Analysis
VisualBUS
Integrated Multi-functional Neutronics Analysis System

**Basic Functions - Calculation (BUS)**
- Particle Transport: SN/MC/MC-SN (1D/2D/3D)
- Isotope Depletion: Bateman/Runge-Kutta Methods
- Material Activation / BHP
- Radiation Damage
- Radiation Protection

**Auxiliary Functions – Interface (Visual)**
- MCAM/SNAM/RCAM:
  CAD-based Automatic modeling for MC/SN/MC-SN
- SVIP: Visualization of mixing models-physical fields
- MOO: Parameters optimization using GA/ANN/SA algorithm
- NTC: Neutronics -Thermalhydraulics Coupling
- MTC: Magnetics -Thermalhydraulics Coupling

**Data Libraries**
- HENDL/MG/CG/MC (Hybrid Evaluated Nuclear Data Lib)
- FENDL 1.0/2.0/2.1
- Others
MCAM Test and Application

• ITER Benchmark Model Conversion

Successfully converted the ITER benchmark model into MCNP input and obtained correct nuclear results.
Model Converting by MCAM for MC-codes MCNP & TRIPOLI

ITER full 3D Model

CAD model reverted from
TRIPOLI input file
drawn by TRIPOLI

CAD model reverted from
MCNP input file
drawn by MCNP

Model Converting by SNAM for SN-code TORT

CAD Model reverted from
TORT input file

CAD model (Created by CATIA)
Coupling MCNP-FISPACT for Activation & Burnup Calculation

- Automatically couple transport calculation and burnup calculation
  - Flux distribution: transport calculation with MCNP
  - Isotope transmutation and decay: burnup calculation with FISPACT

- Automatically couple 2-step dose rate calculation
  - Neutron flux: transport calculation with MCNP
  - Gamma source: calculation with FISPACT
  - Dose rate: decay photon transport calculation with MCNP
**Nuclear Data Library: HENDL**

Hybrid Evaluated Nuclear Data Library for Fusion, Fission and Hybrid Application

- From **various international evaluated neutron nuclear data libraries**, such as FENDL, ENDF/B, JENDL, JEF and BROND
- Support **various kinds of data format**, such as ACE(MC), MATXS(MG), AMPX, ANISNB, CARD.

**Multi-functional Data Library**

- transport.lib
- burnup.lib
- activation.lib
- irradiation.lib
- dose-factors.lib

**Various Kinds of Physics Effects**

- resonance self-shielding
- doppler
- thermal upscatter
- irradiation damage
Transients coupling calculation
- Multi-group SN quasi-static neutron transport equation
- Multi-velocity-field, multiphase, Eulerian, fluid-dynamics model

Multi-group data base (resonance self-shielding and temperature)

DBA (design basic accident) and severe accident analysis

Thermal reactor / fast reactor / subcritical reactor transient safety analysis
**RiskA: An Integrated Reliability/Probabilistic Safety Analysis Program**

- FMEA (Failure Mode Effects Analysis)
- FTA (Fault Tree Analysis)
- ETA (Event Tree Analyses)
- Importance Analysis
- Uncertainty & Sensitivity Analyses
- Reliability Optimization
- Reliability Data Management

**SYSCODE: A Fusion/hybrid System Analysis Program for Parameters Optimization and Economics Analysis**

- Cost-benefit calculation for fusion and fusion-fission hybrid systems
- Parameters optimization with multi-objectives and constraints by using the GA/SA algorithms
- Sensitivity/ Uncertainty analyses of parameters by using Monte Carlo or other methods.
- System analyses by integrating Physics model, engineering model and financial models

Y. Wu et.al, Chinese J. Nucl. Sci. & Eng., Vo.27, No.3 (2007)
FVAS: Fusion Virtual Assembly System

- **Planning & validation of assembling procedures**
  Manipulating virtual tool, assembling virtual part by interaction; Supporting record of assembly process, aiding analysis of assembly plan

- **Training simulator**
  Improving the skill of trainer, Roaming in virtual real environment
Fusion Virtual System

- Parameter-variable Design
- Automatic Modeling/Calculation/Visualization
- Virtual Assembly and Operation

ASIPP

Neutronics analysis

Thermo & Safety analysis

System Analysis/Virtual Simulation
Contents

1. Necessity
2. Fusion Driver Design
3. Blanket Design
4. R & D
5. Plan
History of Hybrids in China

I. Fusion-Fission Hybrid Project supported by MOST
   (National High Tech Program, 1986-2000)
   • Focused on fuel breeding, detailed concept design
   • led jointly by ASIPP & SWIP

II. Fusion-based Transmutation Research Projects by CAS etc.
   (Fundamental Research Program, 2000-present)
   • Focused on advanced concepts on transmutation
   • led by ASIPP
Ongoing Plan

Concept optimization

- Re-evaluation of various concepts
- DEMO design optimization

Next facility definition and design

Supporting R&D

- EAST heating systems and physics experiments
- Blanket materials and test loops
- Fission fuel cycle technology
- Codes & Data Libs development
Time Schedule

- 2010: Definition of DEMO/Exp. facility
- 2010-2015: Design and R&D for Hybrid
- 2015-2025: Hybrid
- 2025-2035: DEMO-PROTO
- 2035-2045: commercial plants
Roadmap of Fusion to DEMO

- Waste Transmutation
- Fuel Breeder
- Energy Multiplication
- Materials Test

~15 yr earlier

Now~2015  2015~2025  2030~2040  2050
1. **A practical way to fusion DEMO** has been proposed based on an intermediate step of fusion-fission hybrid for waste transmutation/fuel breeding/energy production etc., considering the energy status in China.

2. **EAST** can be served as an important basis and pre-test platform of full superconducting tokamak for ITER/DEMO, can be easily extrapolated to a tokamak for hybrids.

3. **TBM concepts development and related R&D** have been performed, and proposed to be tested in EAST/ITER.

4. **A series of plants/DEMO concepts and design software** have been developed, especially a re-evaluation of hybrids has shown the feasibility and attractiveness of hybrids.

5. Further work is needed to draw a final conclusion/decision on next facility with wide collaboration.
The End

Thanks !

www.fds.org.cn