Fusion-Fission Hybrid Activities in ASIPP

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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)

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

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**

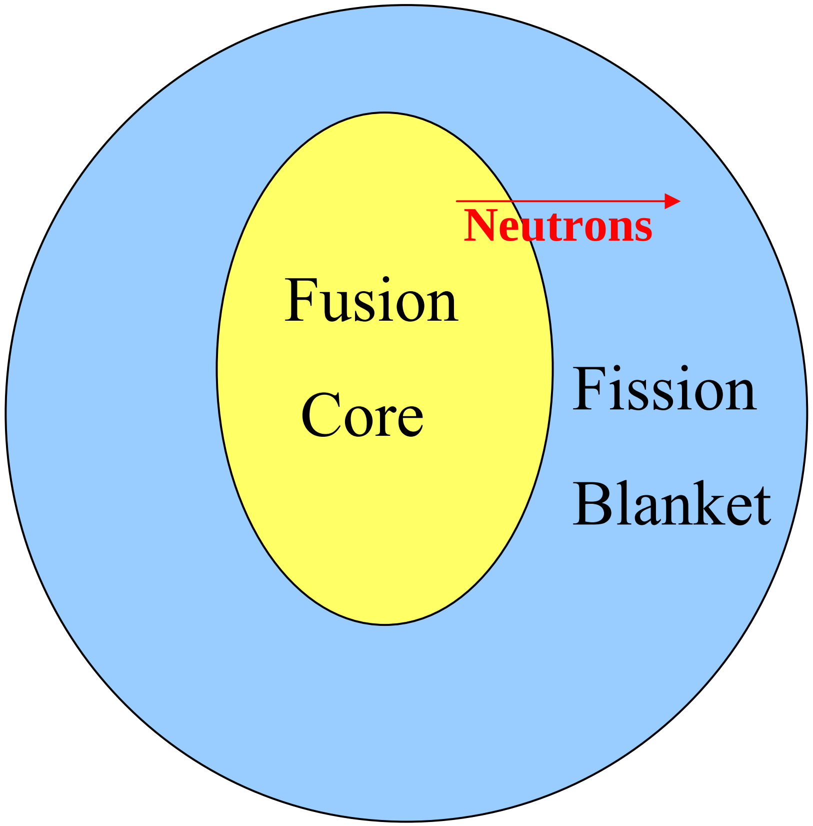
(???~2050?)

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)

Contents

Y. WU et.al, J. of Fusion Engineering and Design, 2006, 81(23-24): 2713-2718

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**

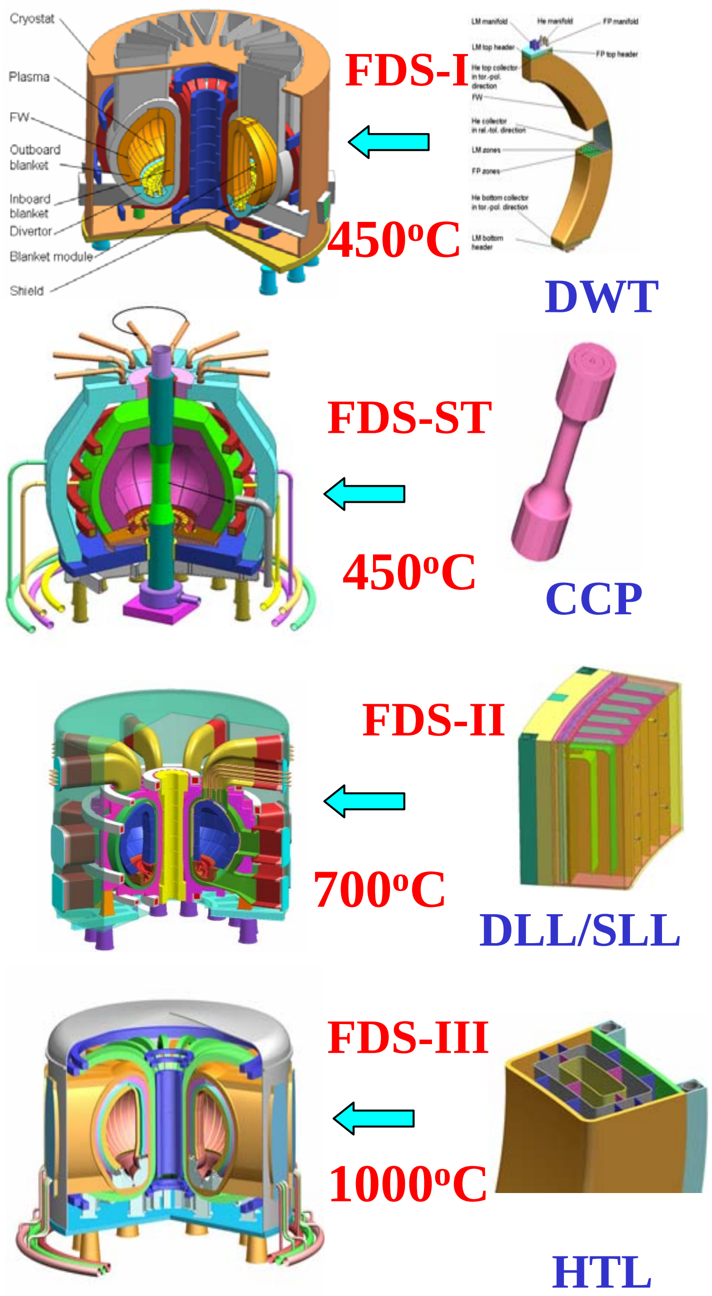
  for highly efficient electricity generation

- **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

Parameters	FDS-I	FDS-II	FDS-III
Fusion power (MW)	**150**	**2500**	**2600**
Major radius (m)	4	6	5.1
Minor radius (m)	1	2	1.7
Aspect ratio	4	3	3
Plasma elongation	1.78	1.9	1.9
Triangularity	0.4	0.6	0.47
Toroidal magnetic field on axis (T)	6.1	5.93	8.0
Safety factor / q-95	3.5	5.0	8.03
Plasma current (MA)	6.3	15	16
Avg. neutron wall load (MW/m^2)	**0.49**	**2.72**	**4**
Average surface heat load (MW/m^2)	0.1	0.54	1.04
Fusion gain	3	31	32
Normalized beta β_N (%)	3	5	4.8

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

Plasma Core Parameters for Next Facility

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

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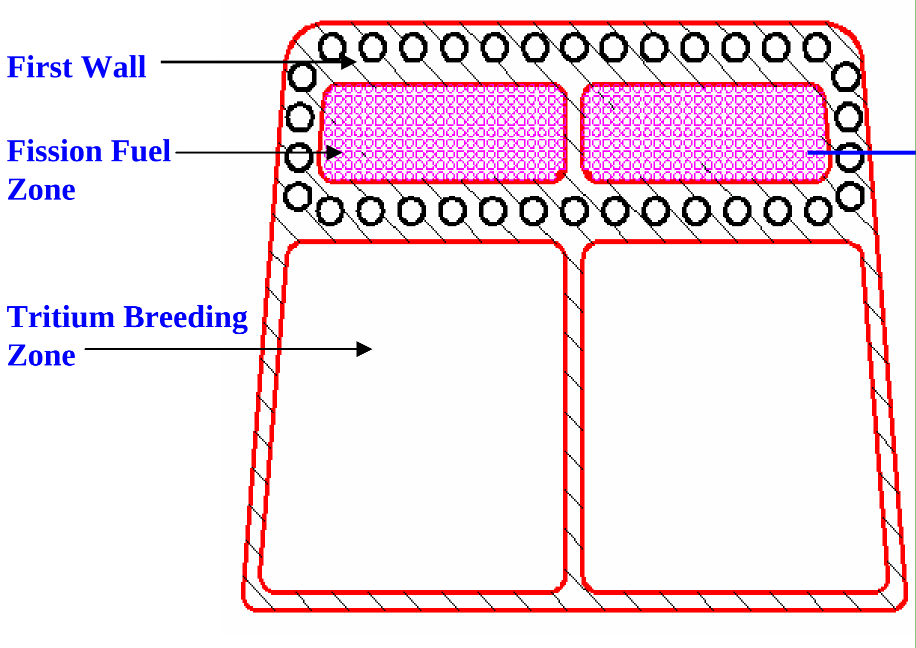
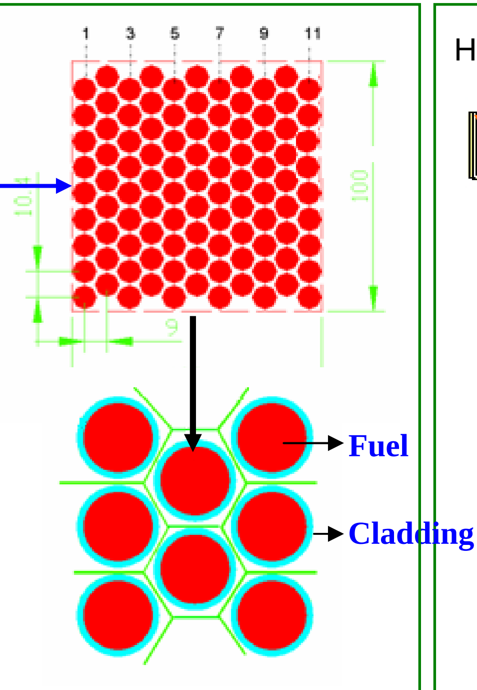
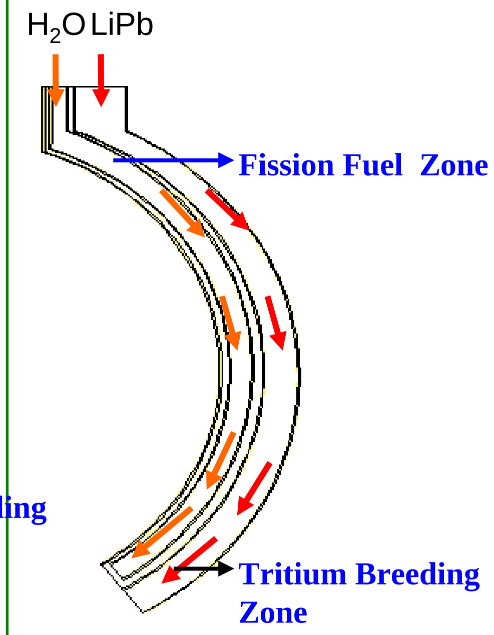
Water-cooled Blanket Concept



Schematic view of outboard blanket



Fuel pin

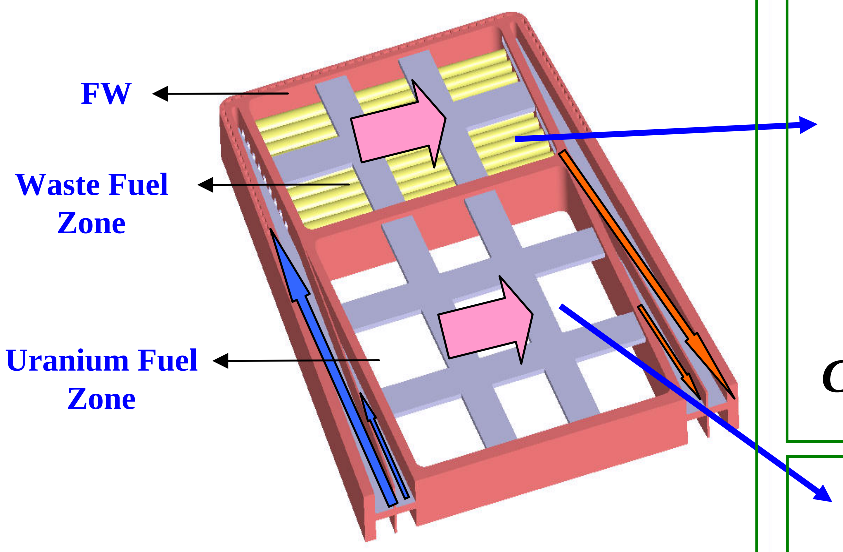


Coolant flow scheme ∨

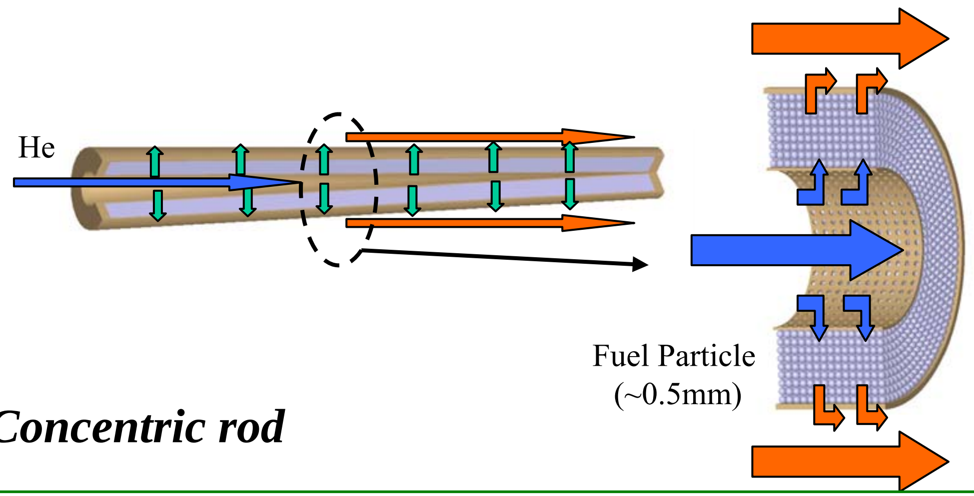
Fission Fuel Zone
Coolant: Water
Flow scheme: Poloidally
Fuel style: Fuel Pin
PWR technology

Tritium Breeding Zone
Coolant: LiPb
Flow scheme: Poloidally
LiPb self-cooled

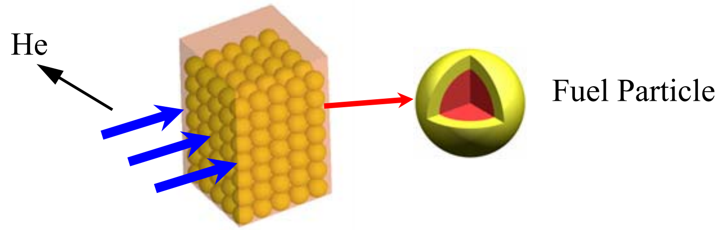
Helium-cooled Blanket Concept



Schematic view of outboard blanket



Concentric rod

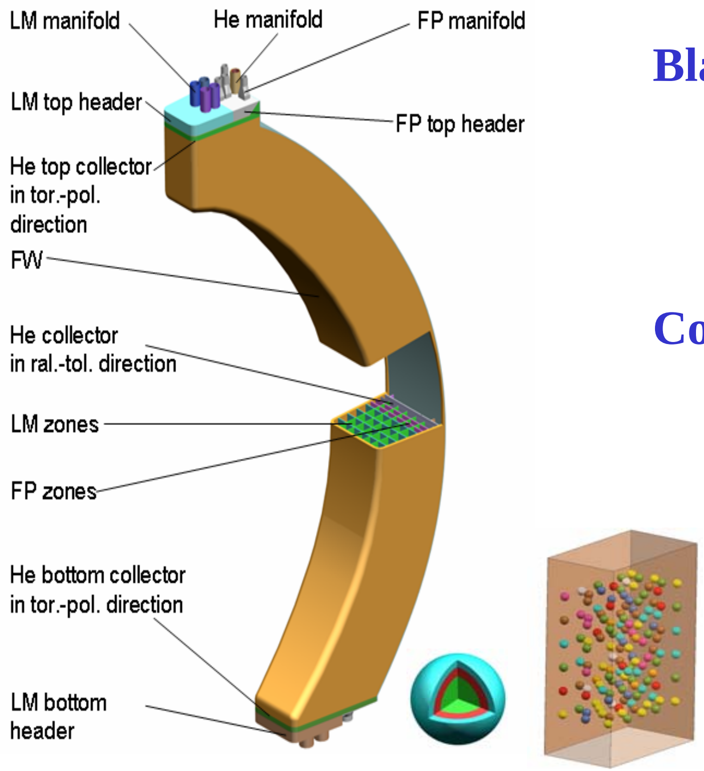


Pebble bed

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

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.

Y.C. WU et.al, Fusion Eng. Design 81(2006)1305-1311

Initial Characteristics

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

Neutron source energy			D-T neutron 14MeV
Neutron source intensity			1.7781E+19 n/s 5.3343E+19 n/s 1.7781E+20 n/s
Fusion power			50 MW 150 MW 500 MW
FDS-EM FDS-FB FDS-WT	**Water-cooled**	**Fuel type (in Fuel zone)**	**PuO2, MAO2, UO2 (rod, PWR-fuel-like)**
	Helium-cooled	**Fuel type (in Fuel zone)**	**PuO2, MAO2, UO2 (particle, HTGR-fuel-TRISO-like)**
Tritium breeder			LiPb
Coolant			Water Helium gas He-LiPb dual coolants (FDS-I)

FDS-EM Design Constraints and Objectives

Items	Constraints and Objectives
K_{eff}	≤ 0.95 (safety margin limit)
Pd_{max}(MW/m3)	≤ 100 (cooling capability limit)
TBR	≥ 1.05 (tritium sustainability requirement)
Energy Multiplication (M)	Reasonable Power Output
	~90 for P_{fu}=50MW
	~30 for P_{fu}=150MW
	~9 for P_{fu}=500MW

FDS-FB Design Constraints and Objectives

Items	**Constraints and Objectives**
Keff	≤ 0.95 (safety margin limit)
$Pd_{max}(MW/m^3)$	≤ 100 (cooling capability limit)
TBR	≥ 1.05 (tritium sustainability requirement)
Breeding Fissile Pu (BSR)	Water-cooled/He-cooled
	maximizing breeding

*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

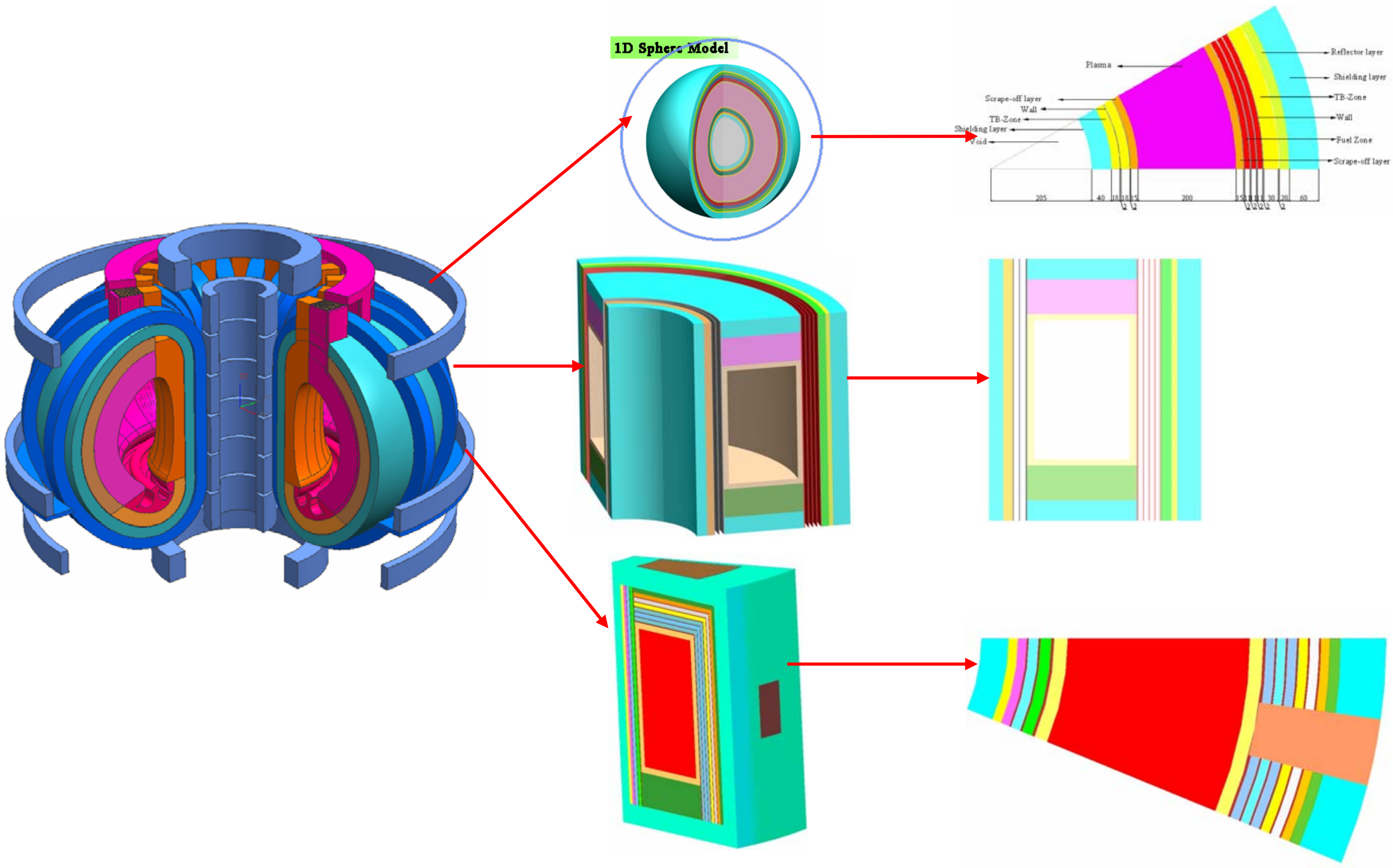
Items	Design Constraints and Objectives	
K_{eff}	≤ 0.95	
TBR (Tritium Breeding Ratio)	≥ 1.05	
Pd_{max} (MW/m^3)(Zone-averaged)	≤ 100	
Fuel Inventory*	minimization while keep balance of LLMA/Pu	
Transmutation Fraction ** /TSR ***	**LLMA**	**maximizing transmutation**
	Pu	**maximizing transmutation**

— *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 yea

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**

Calculation and Analysis

FDS-I

- Neutronics
- Thermalhydraulics
- Themo-mechanics
- Safty analysis (static & Transient)
- Economics

Results on Hybrids: FDS-EM /FB /WT

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 ~100, the max. fissile fuel breeding ratio BSR can be ~10, the max. waste transmutation ratio TSR can be ~15, depending on specific designs

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

4. Further optimization of design scenarios/parameters, detailed engineering analysis are underway

FDS-I Safety Analysis
Plant States and Selection of Reference Transients

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)**

FDS-I Safty Analysis

Conclusions

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 supercriticalilty under any conditions if possible.

References

1. Conceptual Design of the Fusion-driven Subcritical System FDS-I, Y. Wu, S. Zheng, X. Zhu, et al. Fusion Engineering and Design, 2006, 81：1305-1311.
2. Design and Safety Analyses on the Fusion-Fission Hybrid System (FDS-I), Y. Wu, FDS Team, 3th RCM of CRP on “Studies of Innovative Reactor Technology Options for Effective Incineration of Radioactive Waste”, Chennai, India, 15-19 Jan., 2007
3. The Fusion-driven Hybrid System and Its Material Selection, Y.C. Wu, J.P. Qian, J.N. Yu, Journal of Nuclear Materials, 2002, 307-311:1629-1636.

Contents

R&D Activities – Fusion Core

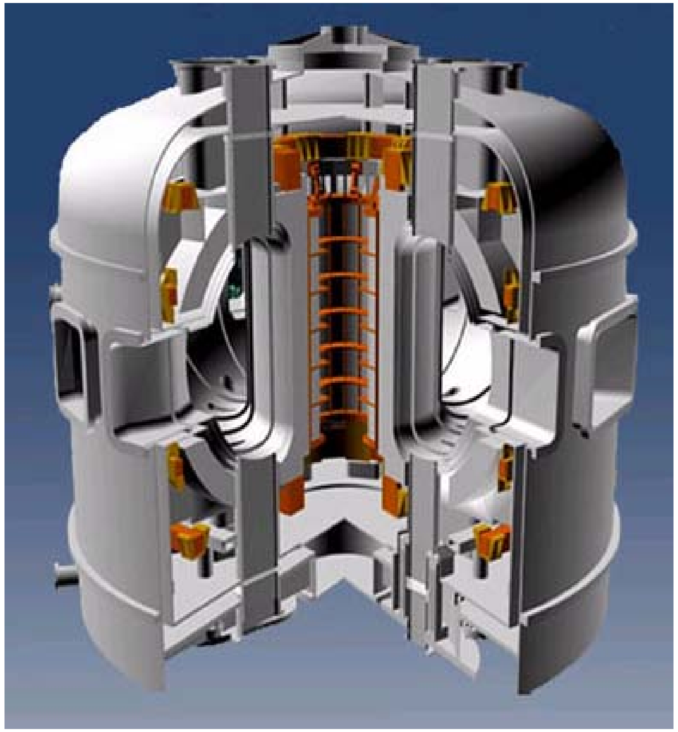
1. EAST superconducting tokamak experiment
2. Blanket materials and TBM development
3. Design and analysis tools development

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



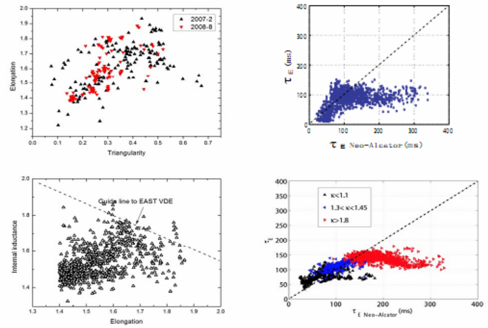
Main Parameters

B_T	3.5~ 4.0 T
R_0	1.7 m
a/b	0.4/0.8 m
Δ_{max}	~ 2
I_P	1.~1.5 MA
H&CD	10 ~ 15 MW
Diverter	Double & single Null
Expected $nT\tau$	$\sim 10^{19\sim20}$ m^{-3} s kev

Long or steady-state operation

- **Project approved in 1998**
- **Construction began in 2000**
- **First Plasma in 2006**

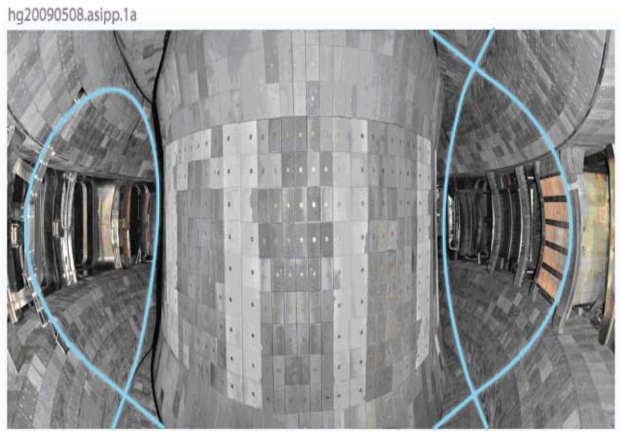
EAST Divertor Configuration Discharge



In 2008

In 2009

Plasmas: Ip~0.6MA, Bt~3T, P_{LHCD}~1.2MW, Td ~63s, fully non-inductive at 250kA
Shaping: kappa ~ 1.9, delta ~ 0.65, limiter, DN, SN
Control: RZIp feedback + shaping programming ;iso-flux



Up to 60 sec Long pulse discharge with LHCD

EAST Operation Plan

	Phase I (2007-08)	Phase II (2009-13)	phase III (2014-20)
Ip(MA)	0.3-0.5	0.5-1	1-1.5
R(m)	1.85	1.85	1.85
a(m)	0.45	0.45	0.45
K	1.2-1.5	1.2-1.9	1.5-1.9
D	0.2-0.3	0.3-0.5	0.3-0.6
ICRF(MW)	1.5/30-110MHz	4/30-110MHz	4/30-110MHz
CW		4/20-70MHz	4/20-70MHz
LHCD(MW)	2/(2.45GHz)	4/2.45GHz	4/2.45GHz
10-1000s		4/4.6GHz	4/4.6GHz
			4/3.7GHz
NBI(MW)10-100s		4/40-80keV	8/40-80keV
ECRH(MW)CW		4/140GHz	4/140GHz
t (s)	5-20s	5-400	5-400
Diagnostics	20	30—40	40
configuration	SN,DN,Cryo-pump	SN,DN,Cryo-pump	SN,DN,Cryo-pump
PFC	C (cooling)	C/W (cooling)	W(cooling)
internal coils	VFB coil	VFB coil	VFB+I coils

TBM

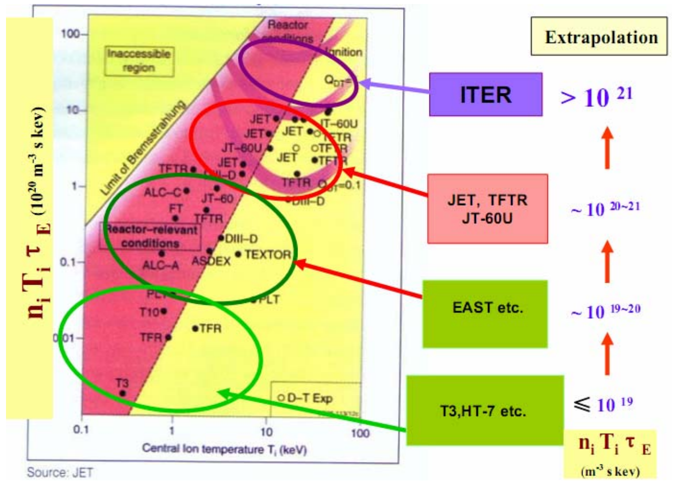
FS insert test

EAST-TBM

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 plateform 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	ITER	EAST
Total fusion power	500 ~ 700 MW	(~10^{16} D-D Neutrons S^{-1})
Inductive pulse time	≥ 400 s (Q ≥10)	~ 10 s
No-inductive pulse time	1000~3000s (Q ~5)	~ 1000 s
Expected n T τ	~$10^{21\sim22}$ m^{-3} s kev	~$10^{19\sim20}$ m^{-3} s kev
B_T (6.2 m)	5.3 T	3.5 - 4.0 T (1.7m)
R_0	6.2 m	1.7 m (1.85m)
a	2.0 m	0.4 m (0.45m)
κ_{95}	1.70 / 1.85	1.8 / 2.0
δ_{95}	0.33 / 0.49	0.30 / 0.60
Ip	15 (17) MA	1.0 (1.5) MA
Divertor Configuration	Single Null	Single & Double Null
Auxiliary Heating / CD Power	73 – 110 MW	4 – 20 MW



R&D Activities - Blanket

1. EAST superconducting tokamak experiment
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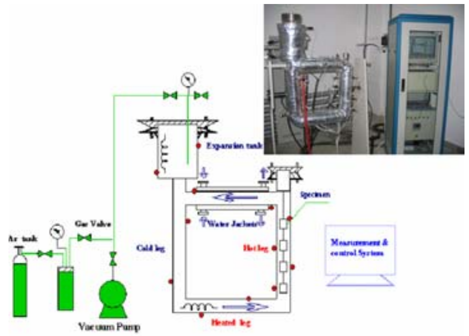
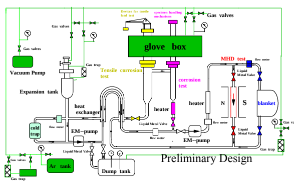
Development &Test Strategy of TBMs

Stage I: Out-of-pile Test
(1/3 size)



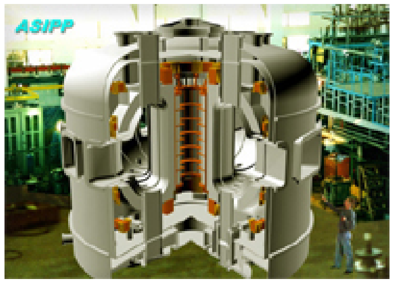
Thermal convection loop
Forced convection loop

Forced Convection LiPb Experimental Loop

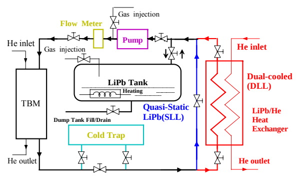


- R&D on materials (RAFM, Coating and FCI) and fabrication technology
- Diagnostic and measurement
- Out-of-pile test of 1/3 mockup etc.
- MHD

Stage II: Test in EAST
(1/2 size)

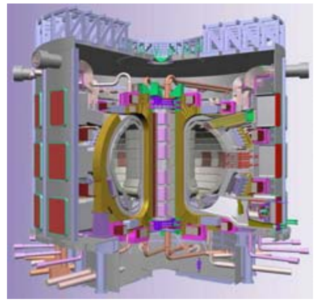


LiPb/He system for TBM in EAST

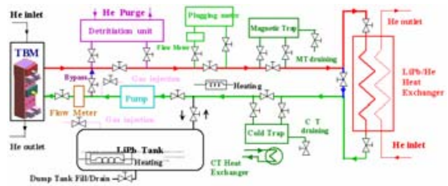


- 1/2 mockup test in EAST
- EM and thermo-mechanics, partially neutronics performances
- Influence on plasma
- MHD

Stage III: Test in ITER
(full size)



LiPb/He system for TBM in ITER



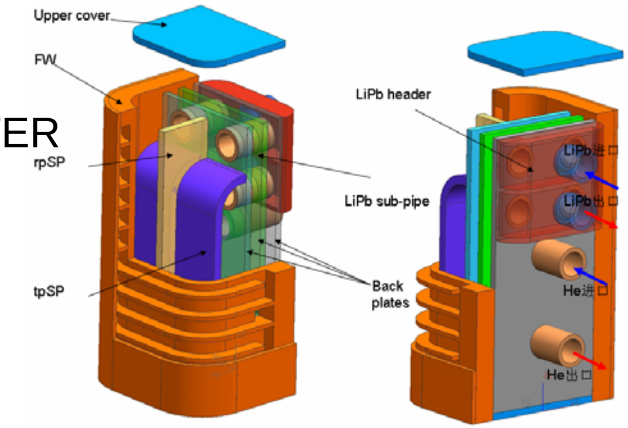
- To confirm results of EM/Thermo-mechanics test in EAST,
- To test neutronics, tritium and integration performances in ITER

Material R&D and Out-of-pile Mockup Test

(1/3-Size TBM)

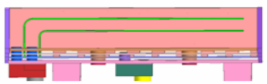
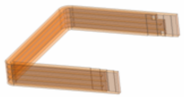
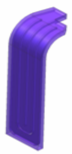
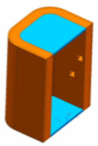
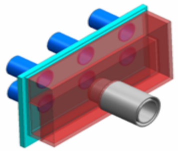
Objectives:

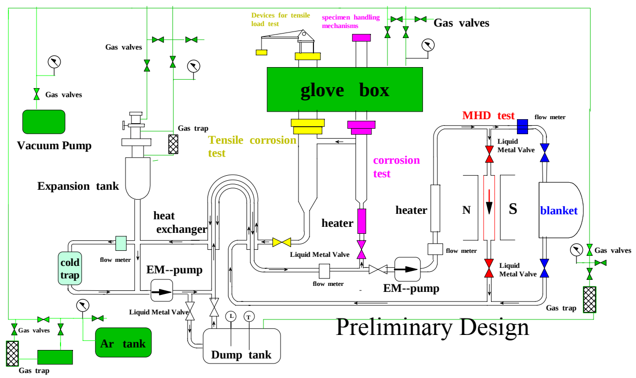
- Validation of the fabrication route and techniques
- Validation of mechanical performances
- Assessment of reliability and safety with regard to ITER standards.



Test Items:

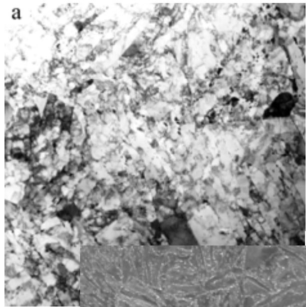
- Leak and pressure test.
- MHD and heat removal from FW.
- Mock-up connected to LiPb loop
- Hydrogen control and extraction to simulate tritium extraction
- Irradiation performance in fission reactors.

He flow distribution in Back Manifold 	Heat transfer/extraction in First Wall - Verification of transf. coef. - Instrumented full size FW channels 	Heat transfer/extraction in rpSP -Verification of transf. coef. - Instrumented full size rpSP channels 
Heat transfer/extraction in tpSP -Verification of transf. coef. - Instrumented 1:5 size FW channels 	Test of a FW mock-up - Fabrication qualification - Cycling/endurance tests - 1:5 TBM size 	Simulating LiPb flow distribution in LiPb header 

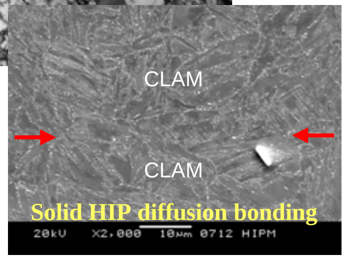


Development of CLAM Steel

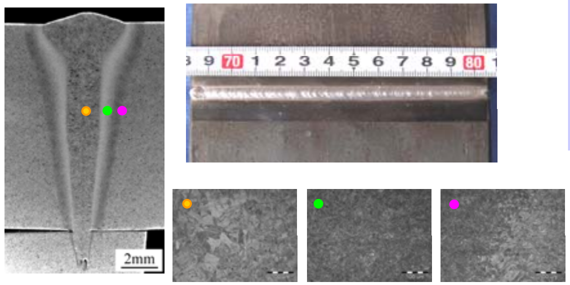
- ***Compositions Design***
- ***Fabrication techniques***
  - *Heat of several hundred kg*
- ***Joining techniques***
  - *Solid HIP diffusion bonding*
  - *Tungsten Inert Gas welding*
  - *Electron Beam welding*
- ***Coating technique***
- ***Corrosion test***
- ***Irradiation Test***
  - *Neutron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



TEM



Solid HIP diffusion bonding



Electron beam welding of CLAM

first wall of Mockup

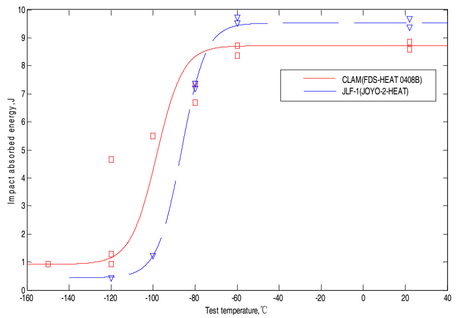
Mockup

Main Compositions

Element	CLAM	EUROFER97	F82H	JLF-1	9Cr-2WVTa
Fe	Bal.	Bal.	Bal.	Bal.	Bal.
Cr	9.0	8.8	7.7	8.9	8.9
W	1.5	1.1	1.9	1.9	2.0
Mn	0.45	0.37	0.16	0.48	0.44
V	0.20	0.19	0.15	0.19	0.23
Ta	0.15	0.068	0.023	0.084	0.06
C	0.10	0.10	0.09	0.10	0.11
Si	</0.10	<0.05	0.10	0.24	0.21
Y	</0.2				

Tensile properties of CLAM and other RAFMs at RT

Steel	Heat treatment(℃)	σ_b(MPa)	$\sigma_{0.2}$(MPa)	δ_5(%)
CLAM(HEAT 0204) (2mm plate)	980℃/30min+760℃ /90min	652	470	26.6
CLAM(HEAT 0408A) (ϕ12 bar)	1040℃/30min+ 760℃/90min	652	501	28.8
CLAM(HEAT 0408A) (ϕ12 bar)	980℃/30min+760℃ /90min	669	514	24.8
CLAM(HEAT 0603A) (12mm plate)	980℃/30min+760℃/9 0min	699.8	560.8	-
Eurofer 97 (14mm plate)	980℃/30min+760℃ /90min	652	537	20.8
F82H (15mm plate)	1040℃/40min +750℃/60min	669	548	21.7



HEAT0408B and JLF-1(JOYO-2-HEAT

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℃ (in operation)
  - DRAGON-II : Inconel, 700℃ (in operation)
  - DRAGON-III: SiC_f/SiC, 700-1000℃ (in fabrication)
- **Forced convection loop (MHD)**
  - DRAGON-IV: 480-700℃ (in fabrication)
- **Auxiliary system for TBM in EAST**
- **Auxiliary system for TBM in ITER**

DRAGON-I

✓ **Design Objectives:**

- Thermal convection loop (500℃)
- 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℃
- Flow rate : ~ 0.08m/s

DRAGON-II

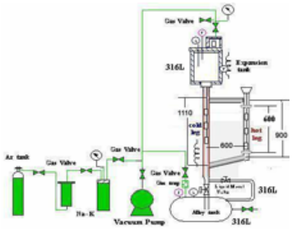
✓ **Design Objectives:**

- High temperature (700℃) thermal convection loop
  - hot leg: 700℃
  - cold leg: 480-640℃
- Obtain corrosion results for the TBM-DLL concept

✓ **Key issues:**

- Material selection (700℃)
- Fabrication technology
- Smelting of Large quantity LiPb

DRAGON-III



Quartz Loop Base

SiC_f/SiC Loop

✓ **Design Objectives:**

- High temperature (700-1000℃) thermal convection loop
- Pursue the manufracture of SiC_f/SiC materials for fusion
- Compatibility of SiCf/SiC with LiPb

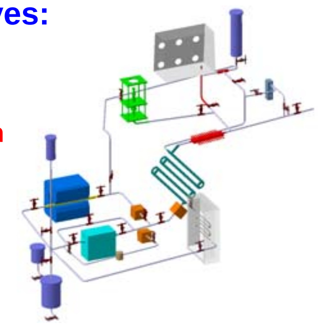
✓ **Key issues:**

- Fabrication of SiC_f/SiC Loop
- Connection technology
- Heating method of high temperature (up to 1000℃)

DRAGON-IV

✓ **Design Objectives:**

- MHD Experiment (>2T, 1m/s);
- Gerenral corrosion in flowing LiPb: -800℃;
- Stress corrosion: max~50kN.



✓ **Key issues:**

- Structureal Materials
- LiPb Purification
- Loop construction and measurement technology

Fabrication of SiC_f/SiC Composite

Requirements:

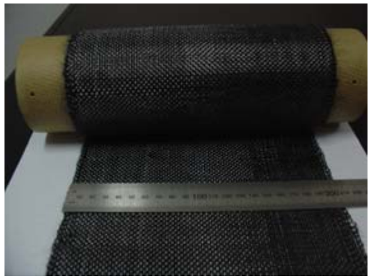
- Low / high thermal conductivity
- Low electrical conductivity
- Good compatibility with LiPb under elevated temp.
- Stable under neutron irradiation

Key issues:

- Fabrication of SiC_f/SiC pipe
- Fabrication of FCI
- Bonding technology of SiC_f/SiC composites

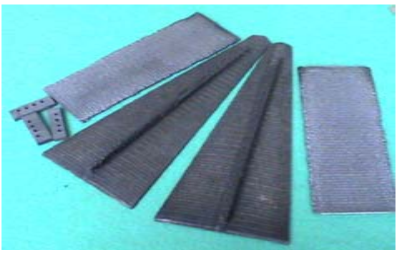
SiC_f/SiC composites

SiC fiber



SiC fiber felt

SiC fiber cloths



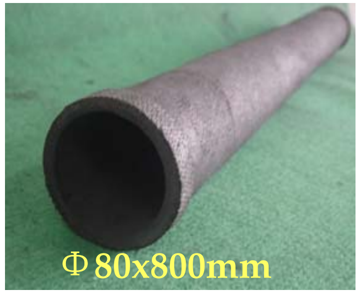
Continuous SiC fiber reinforced ceramic matrix composites

Strength of Continues SiC fiber reach 2.8-3.0GPa

Loop Technology

Fiber 3D braid preform

Connection of metal and SiC composite



SiC_f/SiC pipe

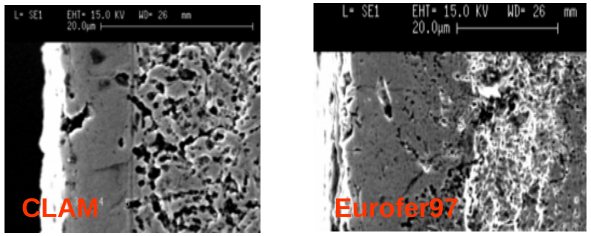
SiC_f/SiC composites were fabricated by CVI + PIP + CVD.

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

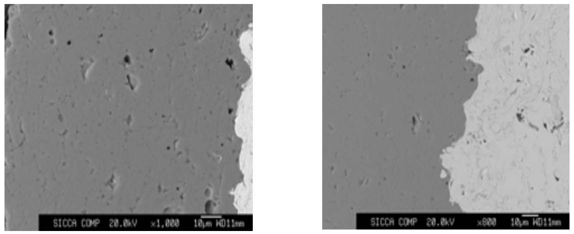
SiC_f/SiC Loop

Coatings and Corrosion Experiments

Coating fabrication



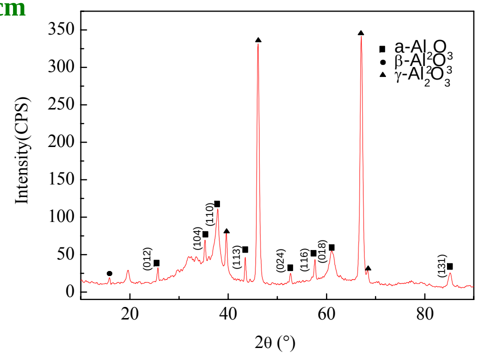
CVD Coating



SEM image of Al_2O_3 coatings section by APS

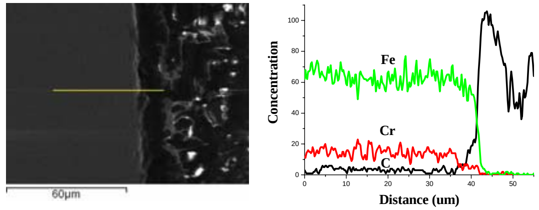
- **The properties of the Al_2O_3 coatings on CLAM:**
  - **Bond Strength: UTS~31.7MPa**
  - **Specific resistance: SR>1010Ω·cm**
  - **Roughness: Ra~4μm**
  - **Microhardness: ~HV951 (200g)**
  - **Density Porosity: DP ~7.2%**

The phases of coatings on CLAM by APS are γ-Al_2O_3 and α-Al_2O_3



Flowing Experiment at 480℃

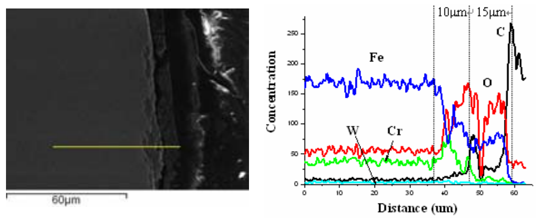
6000 hrs



EDX line scan of CLAM (before cleaned)

A relative low concentration of Fe,Cr contrast to the matrix

8000 hrs



EDX line scan of CLAM (before cleaned)

Similar concentration profile of Fe,Cr in the corrosion layer with 6000hrs

Tensile tests after expose 3000hrs

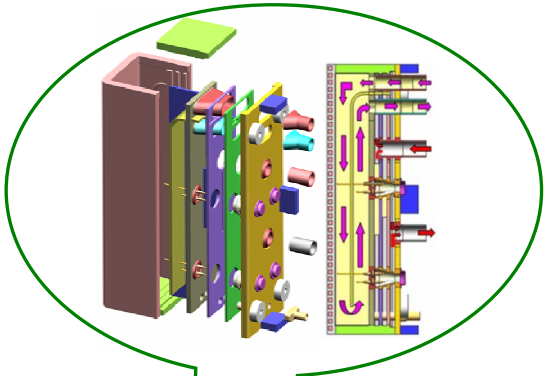
Specimens	No	UTS (MPa)	A(%)
Original	1#	450	19.5
	2#	440	20.5
Corrosion	3#	433	17.0
	4#	430	15.0

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

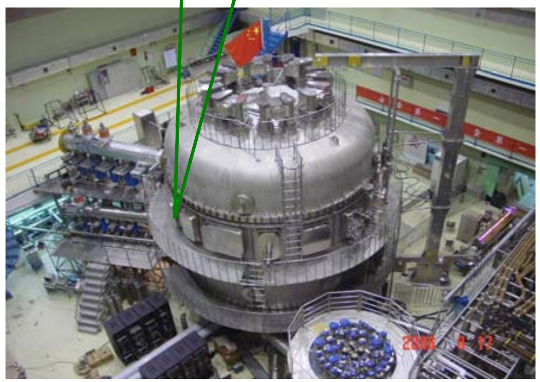


EAST-TBM Test in EAST:

- **ElectroMagnetic performance**

  (MHD pressure drop, influence on plasma)
- **Thermo-mechanics/Thermofluid dynamics performances**
- **Partially neutronics performance** (DD neutrons), **Diagnostic instruments**

Device	EAST	ITER		
Phase	DD	HH	DD	DT
R (m)	1.95			6.2
A (m)	0.46			2
Bt (T)	3.5-4.0			5.3
Neutron rate (n/s)	10^{15}~10^{17}			$1.77\text{x}10^{20}$
Avg.HF(MW/m^2)	0.1~0.2	0.11		0.27
Port Size	0.97m x 0.53m	2.2 m x 1.7m		
Pulse (sec)	~1000	100-200		400



Test in ITER

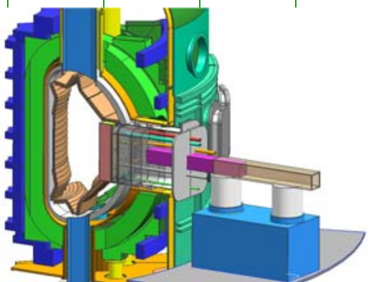
(Full-Size TBMs: solid and liquid TBMs)

≤2011	2012	2013	2014	2015	ITER Operation (Day 1) 2016	2017	2018	2019	2020	2021	2022	2023	2024	2025	≥2026
	DD	DD	DD	DD	HH	HH	HH	DD	Low DT	Low DT	Low DT	High DT	High DT	High DT	

Mat.R&D
In out-of-tokomak

EAST-TBM

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



EAST-TBM

In EAST:
D-plasma(1000sec, $10^{15\sim17}$n/s)
Thermo-mechanics/Thermofluid dynamics performances
FCI & MHD Test
Partially neutronics
Diagnostic instruments

EM-TBM

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

NT-TBM

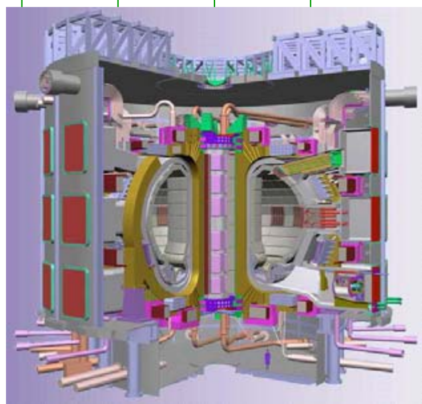
Neutronics Test:
Neutronics performance
N-fluence&gamma spectra
Nuclear heat
T-breeding
Code &Data

TT-TBM

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

IN-TBM

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



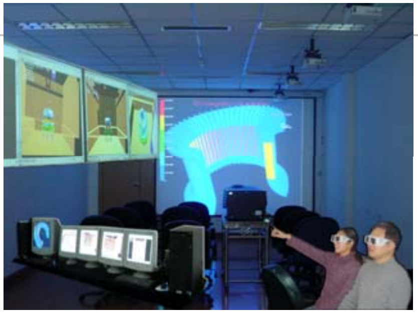
R&D Activities - Design

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 (safty/economy) analysis codes: **RiskA, SYSCODE**
4. Fusion virtual assembly system: **FVAS**
5. material database and component reliability database management system: **FUMDS, RiskBase**

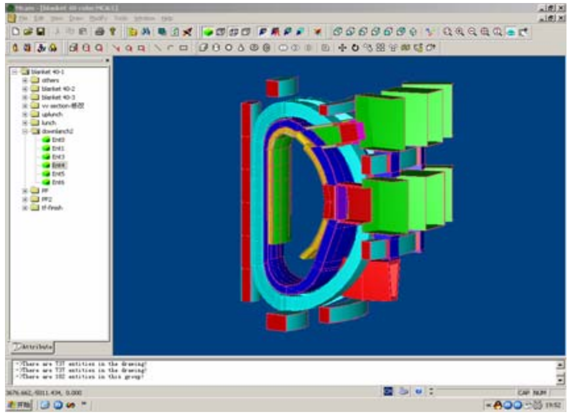
- 50~100 man-years each program



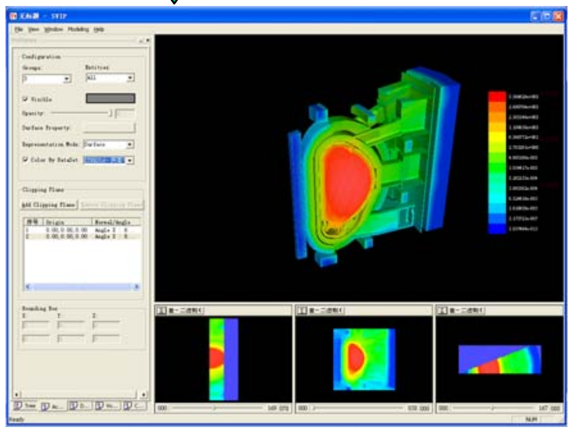
Key Tools for Fusion/Fission Reactor Design & Analysis

VisualBUS

Integrated Multi-functional Neutronics Analysis System

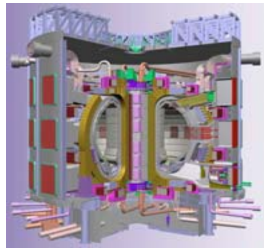


Automatic Neutronics Simulation

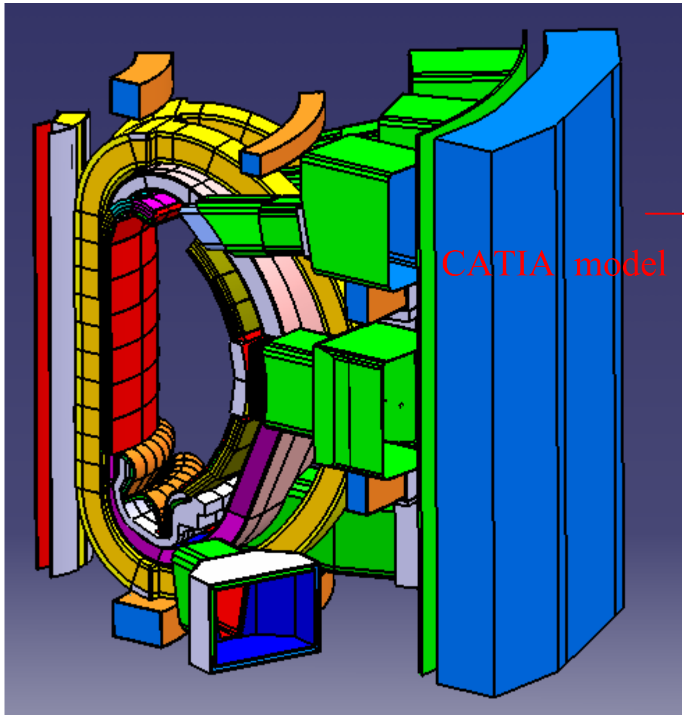


- **Basic Functions - Calculation (BUS)**
  - Particle Transport: SN/MC/MC-SN (1D/2D/3D)
  - Isotope Depletion: Bateman/Runge-Kutta Methods
  - Material Activation / BHP
  - Radiation Damage
  - Radiation Protection
- **Auxiliary Functions – Interface (Visual)**
  - **MCAM/SNAM/RCAM**:
    CAD-based 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

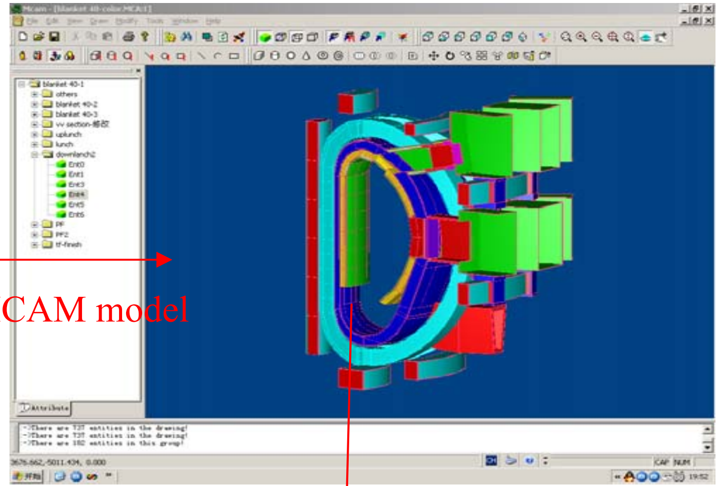
MCAM Test and Application



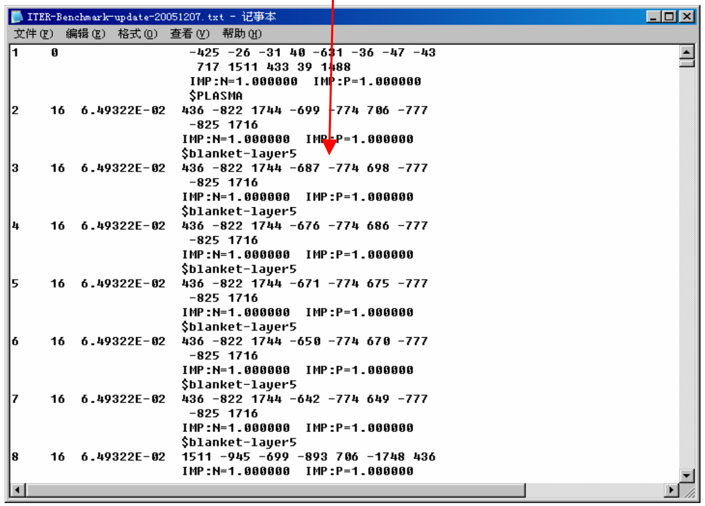
- ITER Benchmark Model Conversion



CATIA model

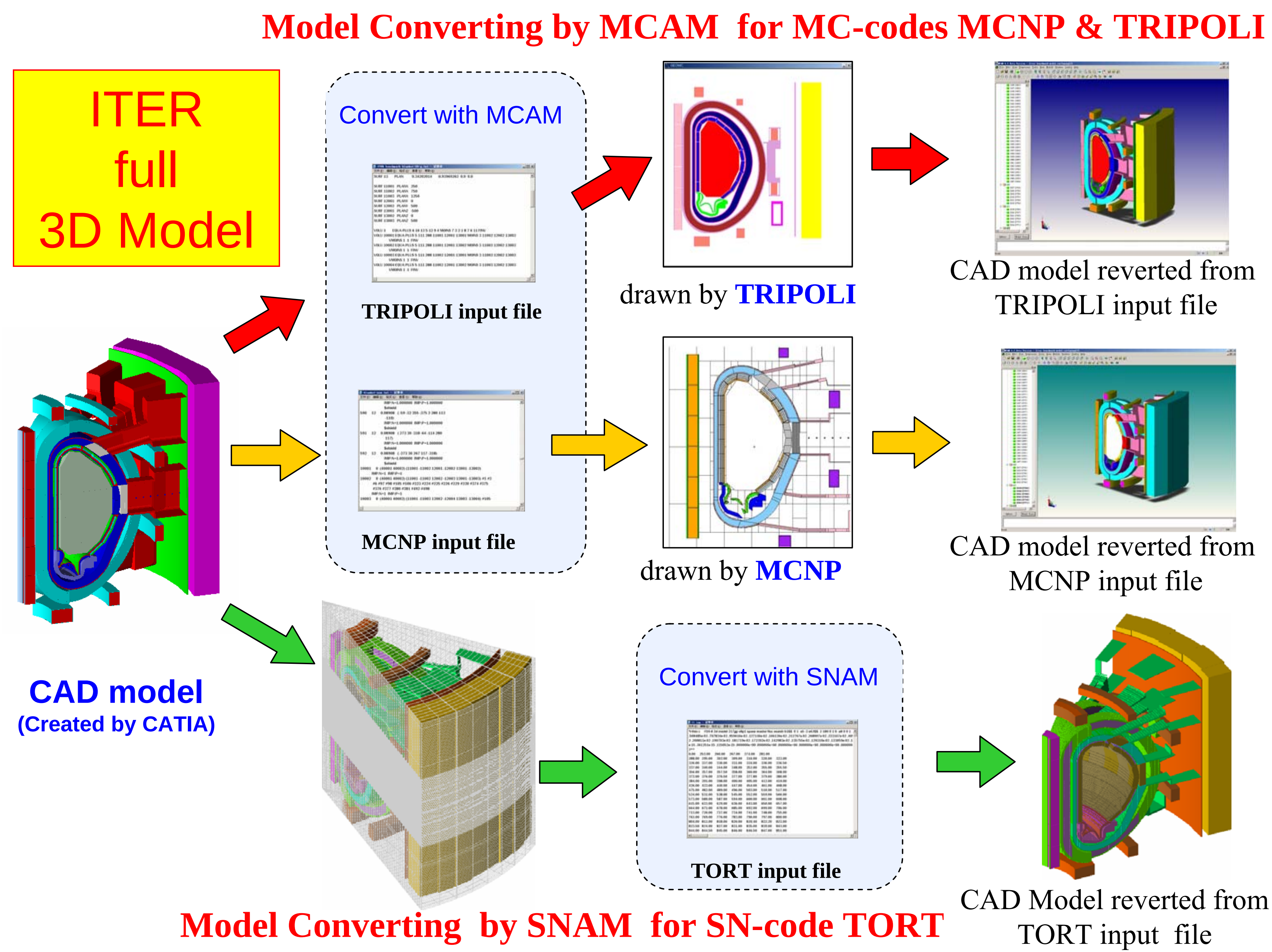


MCAM model




```
1     0                 -425 -26 -31 40 -631 -36 -47 -43
                         717 1511 433 39 1488
                        IMP:N=1.000000  IMP:P=1.000000
                        $PLASMA
2     16  6.49322E-02  436 -822 1744 -699 -774 706 -777
                        -825 1716
                       IMP:N=1.000000  IMP:P=1.000000
                       $blanket-layer5
3     16  6.49322E-02  436 -822 1744 -687 -774 698 -777
                        -825 1716
                       IMP:N=1.000000  IMP:P=1.000000
                       $blanket-layer5
4     16  6.49322E-02  436 -822 1744 -676 -774 686 -777
                        -825 1716
                       IMP:N=1.000000  IMP:P=1.000000
                       $blanket-layer5
5     16  6.49322E-02  436 -822 1744 -671 -774 675 -777
                        -825 1716
                       IMP:N=1.000000  IMP:P=1.000000
                       $blanket-layer5
6     16  6.49322E-02  436 -822 1744 -650 -774 670 -777
                        -825 1716
                       IMP:N=1.000000  IMP:P=1.000000
                       $blanket-layer5
7     16  6.49322E-02  436 -822 1744 -642 -774 649 -777
                        -825 1716
                       IMP:N=1.000000  IMP:P=1.000000
                       $blanket-layer5
8     16  6.49322E-02  1511 -945 -699 -893 706 -1748 436
                       IMP:N=1.000000  IMP:P=1.000000
```

Successfully converted the ITER benchmark model into MCNP input and obtained correct nuclear results

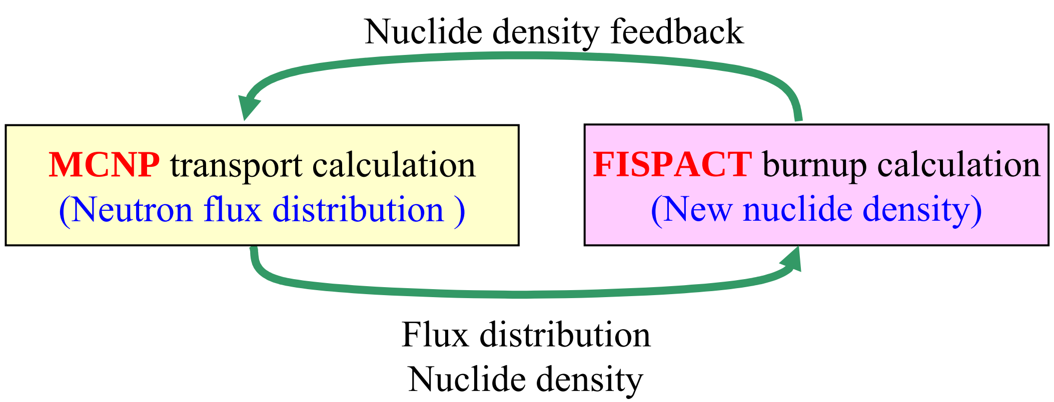


Model Converting by MCAM for MC-codes MCNP & TRIPOLI
ITER full 3D Model
Convert with MCAM
TRIPOLI input file
MCNP input file
drawn by TRIPOLI
drawn by MCNP
CAD model reverted from TRIPOLI input file
CAD model reverted from MCNP input file
CAD model
(Created by CATIA)
Convert with SNAM
TORT input file
CAD Model reverted from TORT input file
Model Converting by SNAM for SN-code TORT

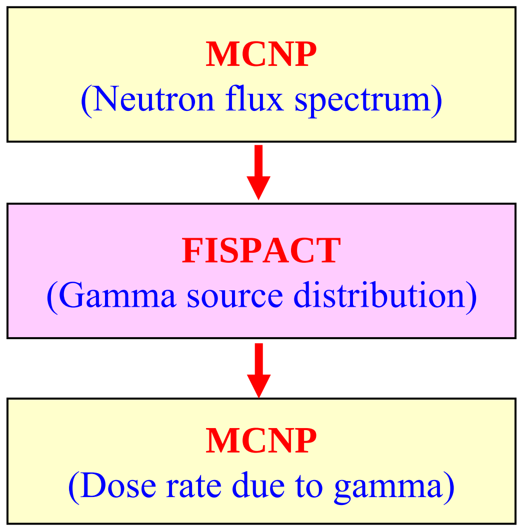
Coupling MCNP-FISPACT for Activation & Burnup Calculation

- **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

→ Coupling transport and burnup calculation



→ 2-Step dose rate calculation



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

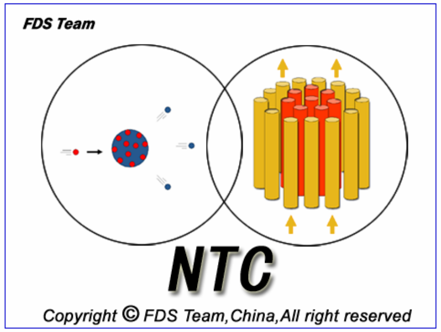
- **transport.lib**
- **burnup.lib**
- **activation.lib**
- **irradiation.lib**
- **dose-factors.lib**

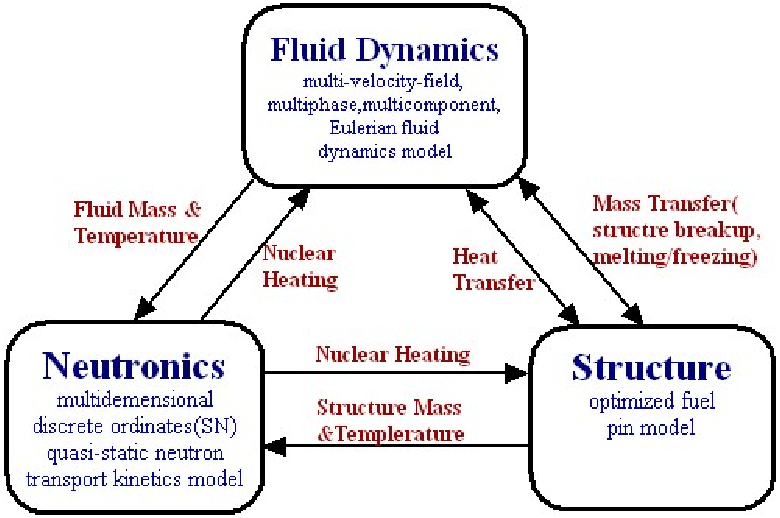
Various Kinds of Physics Effects

- **resonance self-shielding**
- **doppler**
- **thermal upscatter**
- **irradiation damage**

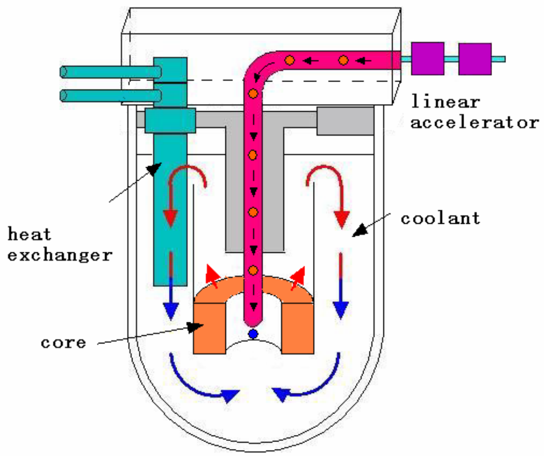
NTC: Neutronics-Thermohydraulics Coupling Code

- **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**

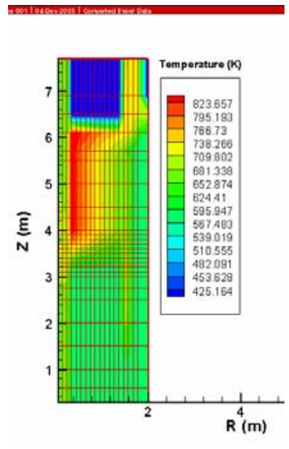




NTC overall code structure



ADS system



Temperature distribution

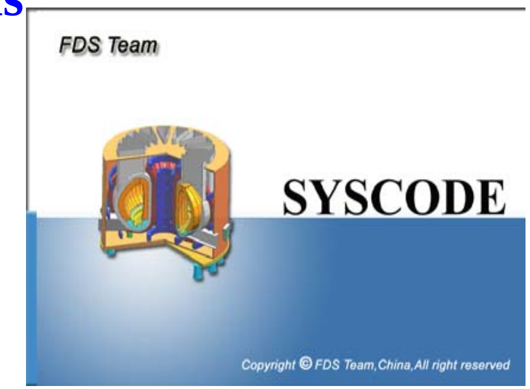
System Analysis Programs

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)

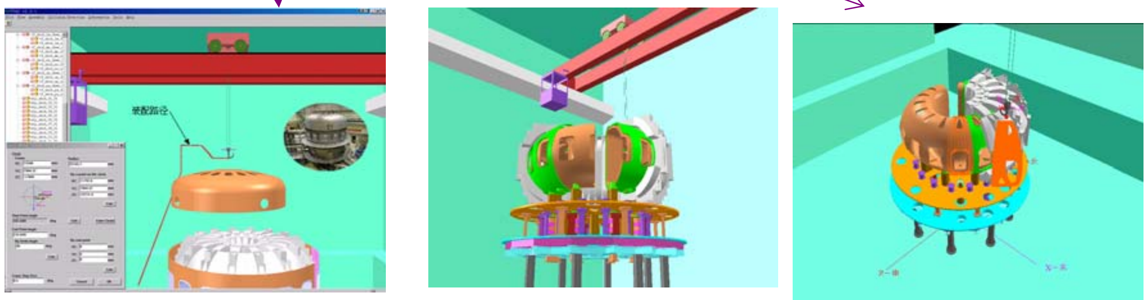
FVAS: Fusion Virtual Assembly System

- **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



EAST virtual assembly

Fusion Virtual System

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

System Analysis/Virtual Simulation

Contents

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 Transmution Research Pojects by CAS etc.

(Fundmental 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

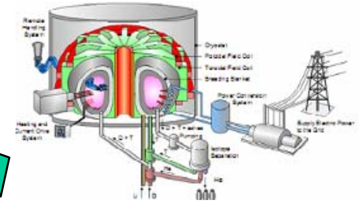
Supporting R&D

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

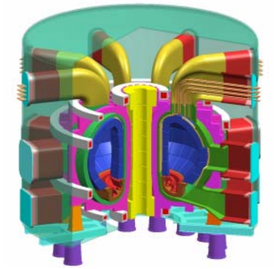
Time Schedule

- **-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**

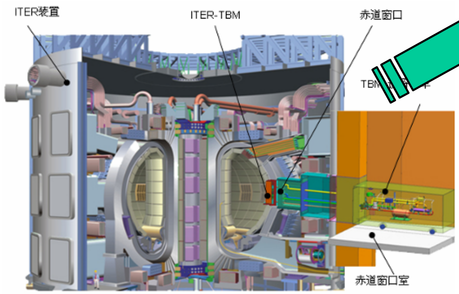
Roadmap of Fusion to DEMO



FPP



DEMO



ITER

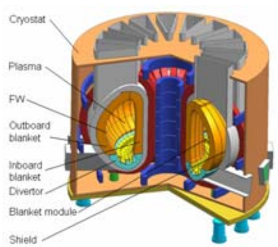
TBM & Materials R&D

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

~15 yr earlier

EAST

Concept & Eng. design



Hybrid

Now~2015

2015~2025

2030~2040

2050

Summary

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, escpecially 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 !

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