FDS Team

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A Preliminary Consideration on China MFE Development Concerning Blanket Aspects

Presented by Yican WU

Contributed by FDS Team

First Workshop on MFE Development Strategy in China

Beijing, Jan.5-6, 2012

Assumptions

- Next step depends on several later steps

  (roadmap to final goal)

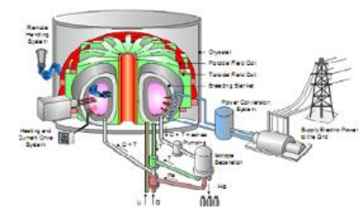
- Next steps depends on previous steps

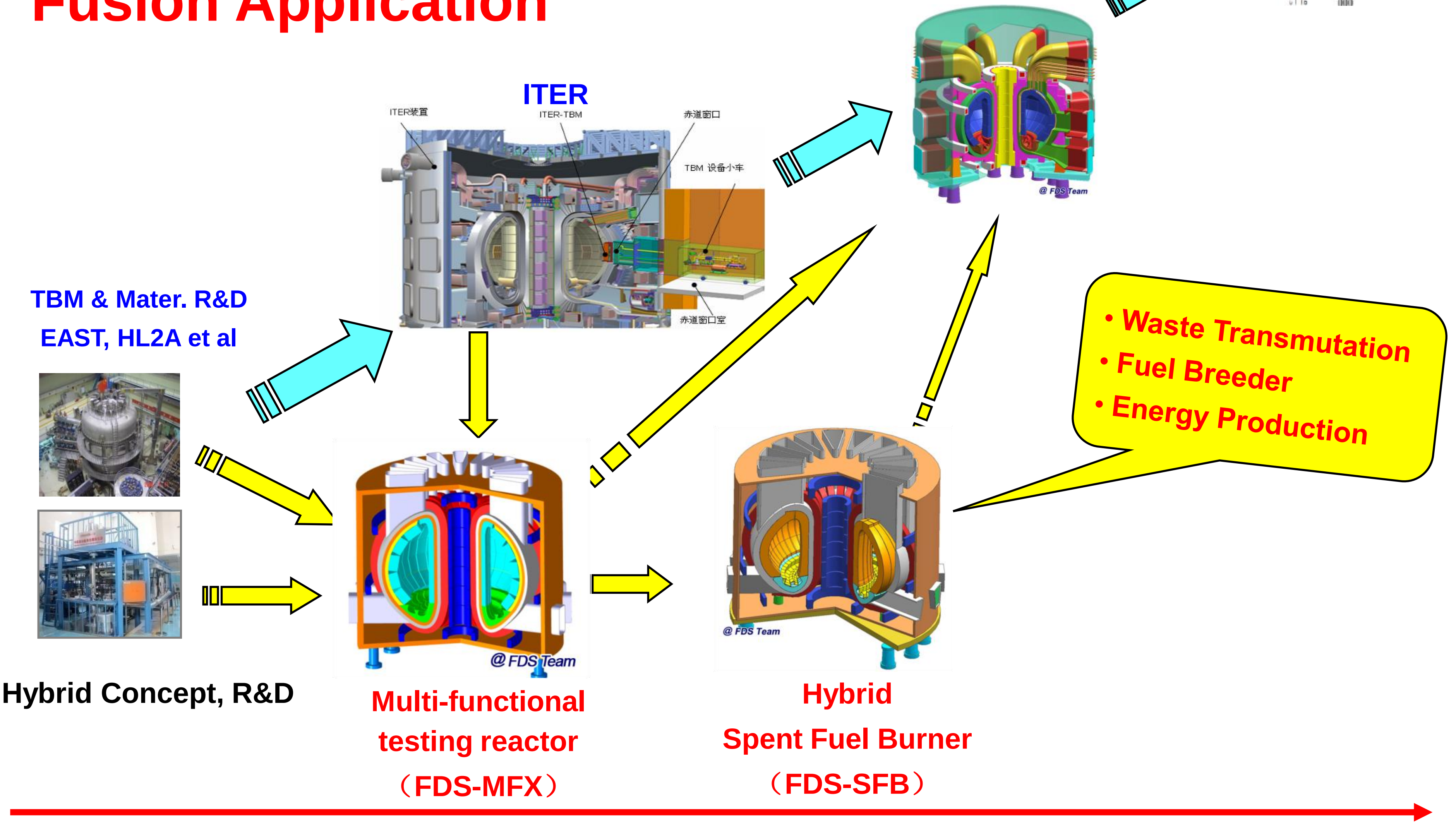
  (existing basis)

How to define a next step – CFETR ?

A preliminary consideration
On MFR development roadmap

Roadmap Proposal for Fusion Application





What we have done
for MFR concept development

@ FDS Team

FDS Series Fusion & Subcritical Reactors Conceptual Design for DEMO/Plants

Pure Fusion Reactor:

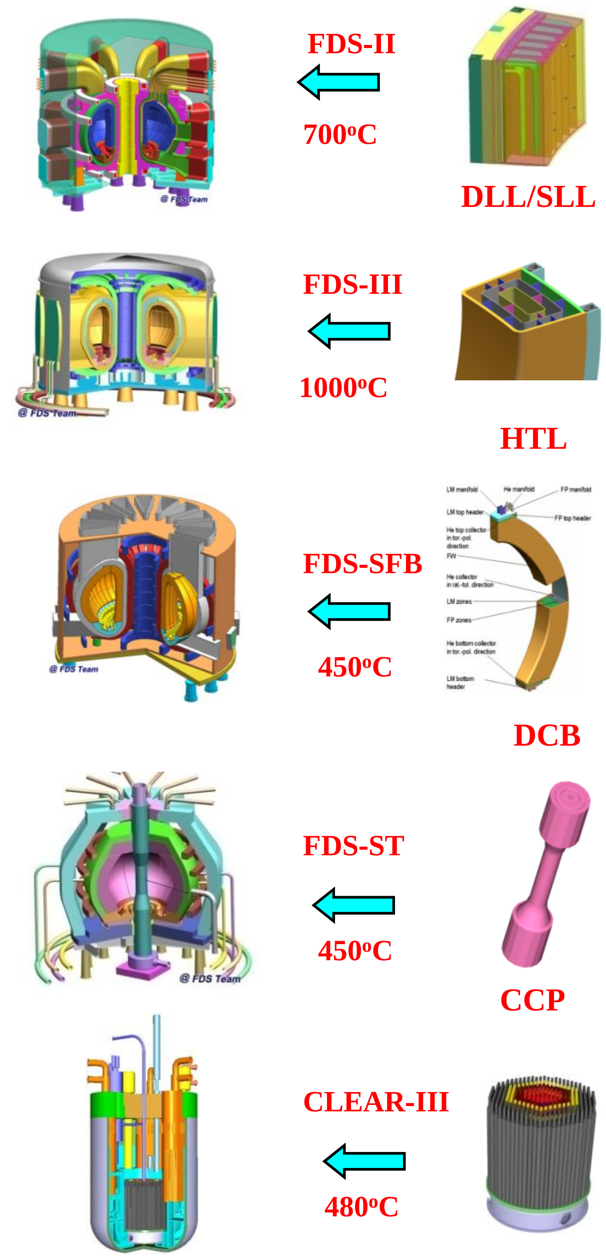
- **FDS-II: Fusion Power Reactor**
  for highly efficient electricity generation
- **FDS-III: High Temperature Fusion Reactor**
  for advanced applications, e.g. hydrogen production

Hybrid Reactor:

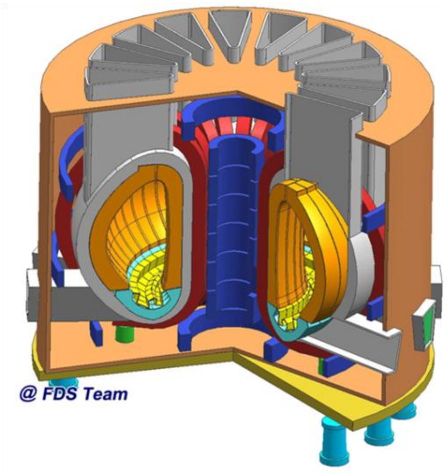
- **FDS-SFB: Fusion Driven Subcritical Reactor**
  for spent fuel burning
  (energy production, fuel breeding, waste transmutation)
- **FDS-ST: Spherical Tokamak-based Reactor**
  for exploiting and assessing innovative conceptual path
- **CLEAR-III: Accelerator Driven Subcritical System**
  (nuclear waste transmutation)

Research Reactor :

- **FDS-MFX: Fusion Driven Multi-Function Expermental Reactor**
  to test and verify the technology&engineering of FDS-SFB
- **CLEAR-I: Accelerator Driven Subcritical Experimental Reactor** to test and verify the technology&engineering of CLEAR-III

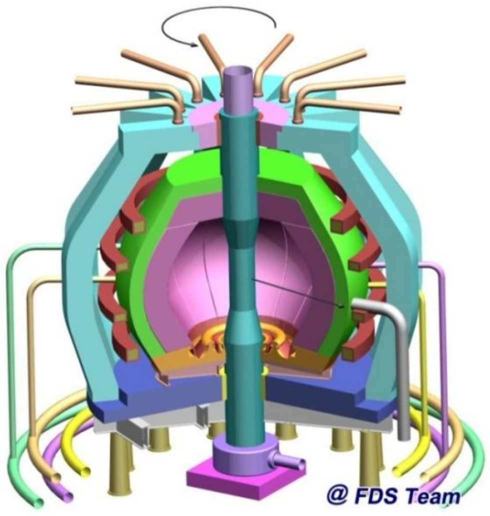


Options for FDS Fusion Drivers



- **Regular Tokamak – FDS-SFB**
  - Fusion power: 100~200MW
  - Power Gain ~3
  - Neutron wall loading ~0.5MW/m2
- **Spherical Tokamak – FDS-ST**
  - Fusion power: 100~200 MW
  - Power Gain ~5
  - Neutron wall loading ~1 MW/m2
- **Magnetic Mirror – FDS-GDT**

  Fusion power: ~50MW



Plasma Parameters of FDS Series Tokamak Reactors

Parameters	FDS-SFB	FDS-II	FDS-III	FDS-ST	EAST*	ITER**
Fusion power (MW)	150	2500	2600	100	0.08	500
Major radius(m)	4	6	5.1	1.4	1.95	6.2
Minor radius(m)	1	2	1.7	1.0	0.46	2
Aspect ratio	4	3	3	1.4	4.2	3.1
Plasma elongation	1.78	1.9	1.7	2.5	1.8	1.70
Triangularity	0.4	0.6	0.47	0.45	0.45	0.33
Plasma current (MA)	6.3	15	16	9.2	1.5	15
Toroidal field on axis (T)	6.1	5.9	8	2.5	4.0	5.3
Safety factor /q_95	3.5	5.0	4.8	5.5	/	3
Auxiliary power /Padd(MW)	50	80	80	19	/	73
Energy multiplication /Q	3	31	32	5	/	≥10
neutron wall load(MW/m2)	0.5	2.72	5	1.0	E-4~E-3	0.57
Average surface heat load (MW/m2)	0.1	0.54	1	0.2	0.2	0.27

* : Phase-III; ** : D-T phase

Y. Wu, Fusion Engineering and Design, 81(2006) 2713-2718.

FDS-GDT: GDT-based FNS Conceptual Design

	Version A	Version B	Version C
Mirror to mirror length (m)	**7**	**16**	**100**
Magnetic field at mid-plane (T)	0.6	1	0.5
Magnetic field in mirror(T)	15	25	25
Target plasma density (m^{-3})	1.3×10^{20}	1.5×10^{20}	1.3×10^{20}
Radius at the mid-plane (m)	0.12	0.16	0.08
Electron temperature (keV)	**0.5**	**0.8**	**3.7**
Mean energy of fast ions(keV)	14	30	60
Maximal plasma β	0.36	0.3	0.6
Neutral beam power (MW)	**8**	**35**	**120**
Neutral beam energy (keV)	30	70	120
Injection angle of neutral beam	30°	30°	15°
D-T fusion power (MW)	**0.25**	**3.04**	**50**
Neutron wall load(MW/m^2)	**0.11**	**0.90**	**0.5**
Neutron flux of plasma edge(MW/m^2)	**0.53**	**2.42**	**10**
Length of testing/blanket zone(m)	0.5×2	1×2	15.5×2

Options for FDS-SFB Fission Blankets

- **FDS-SFB/WCB**

Water-Cooled Blanket

With pin fuel

- **FDS-SFB/HCB**

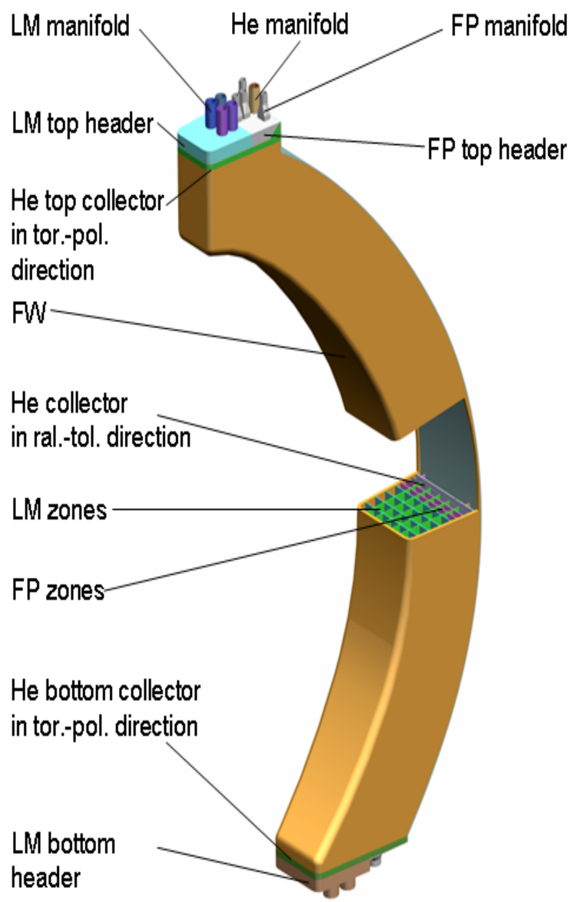
He-Cooled Blanket

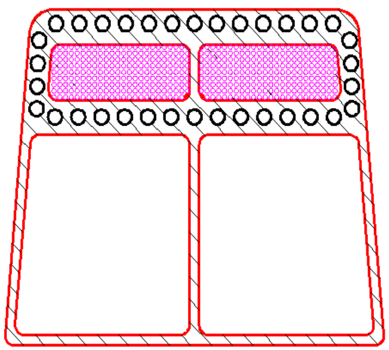
With plate-type fuel

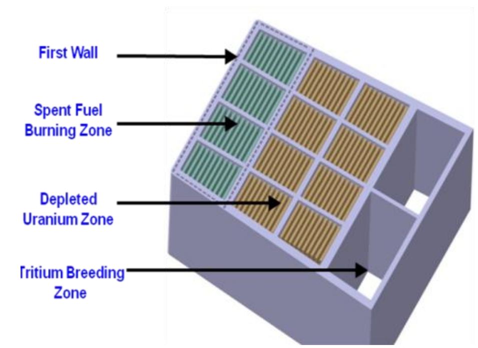
- **FDS-SFB/DCB**

He/LiPb Dual-Cooled Blanket

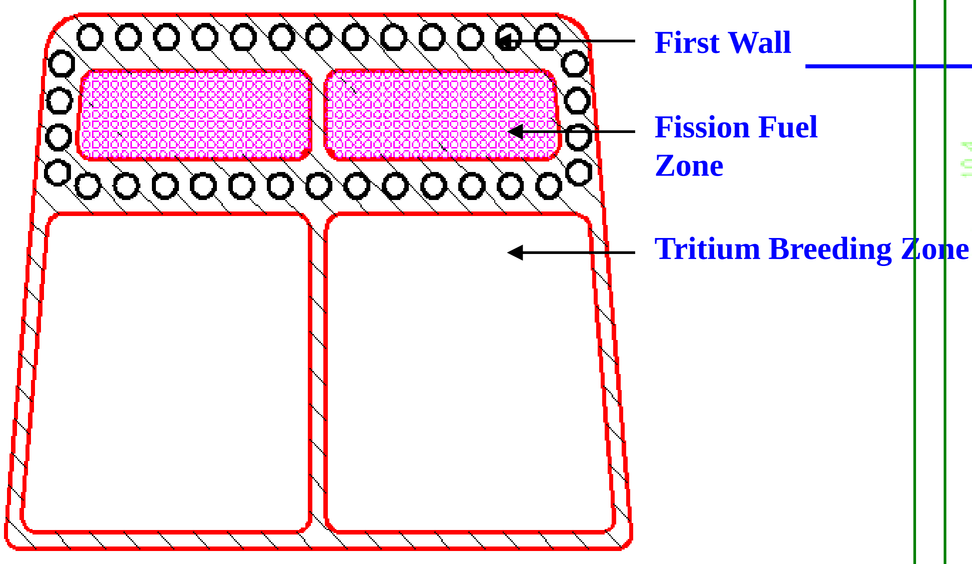
With TRISO-like particle fuel



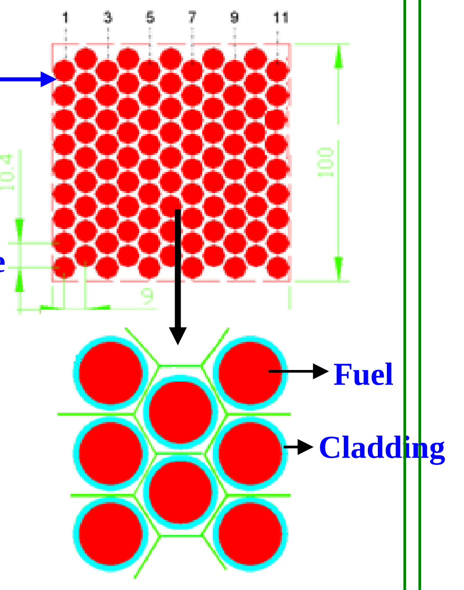




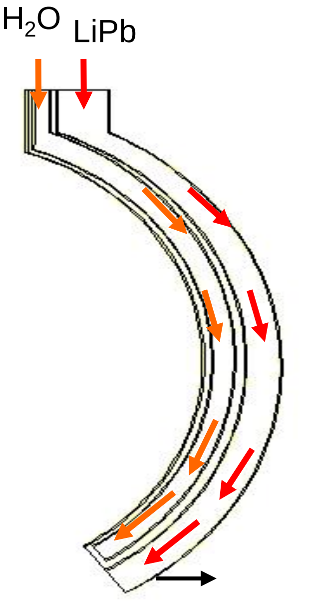
Water-Cooled Blanket(WCB)



Schematic view of outboard blanket



Fuel pin

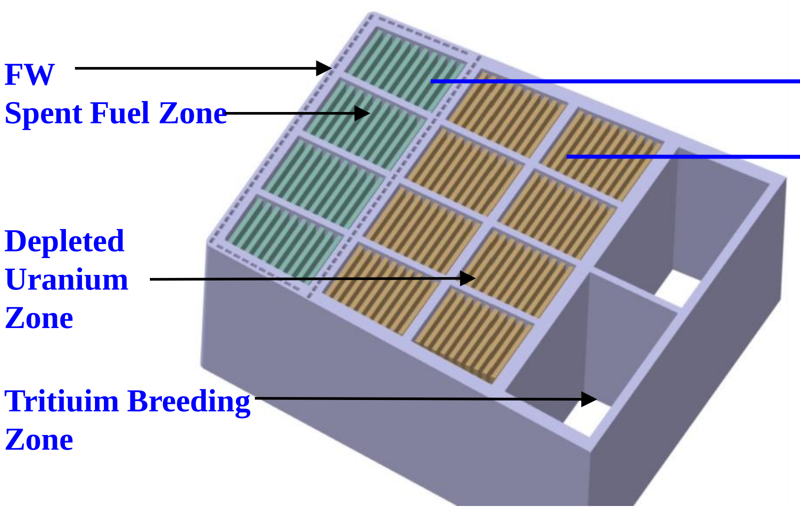


Coolant flow scheme

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(HCB)



Schematic view of outboard blanket

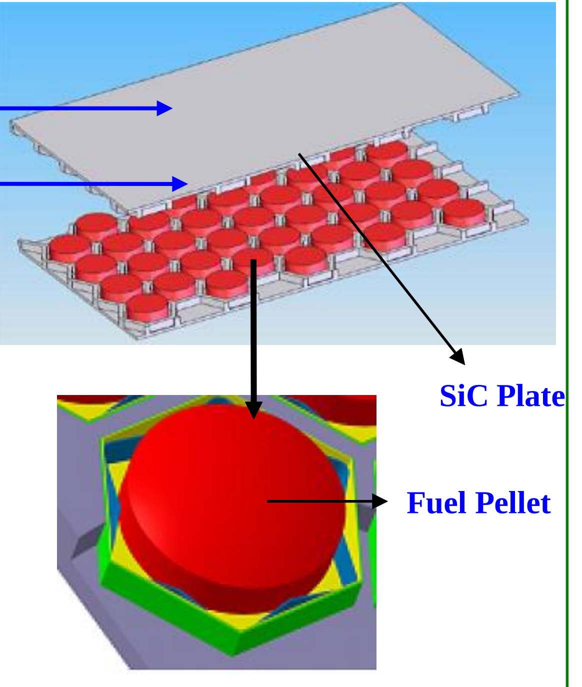
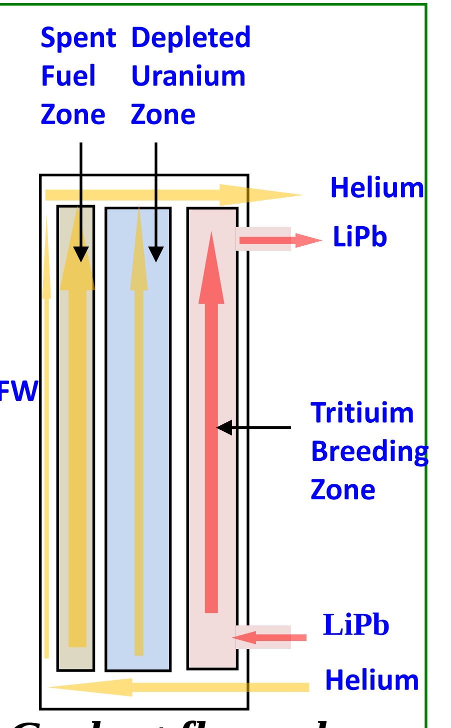


Plate-type Fuel



Coolant flow scheme

Spent Fuel Zone
Coolant： Helium
Flow scheme： Poloidally
Fule style： Plate-type fuel

Depleted Uranium Zone
Coolant： Helium
Flow scheme： Poloidally
Fule style： Plate-type fuel

Tritium Breeding Zone
Coolant： LiPb
Flow scheme： Poloidally

Helium/LiPb Dual-Cooled Blanket(DCB)

Structural Mat.: RAFM

Tritium Breeder: LiPb

Fuel Type: Carbide

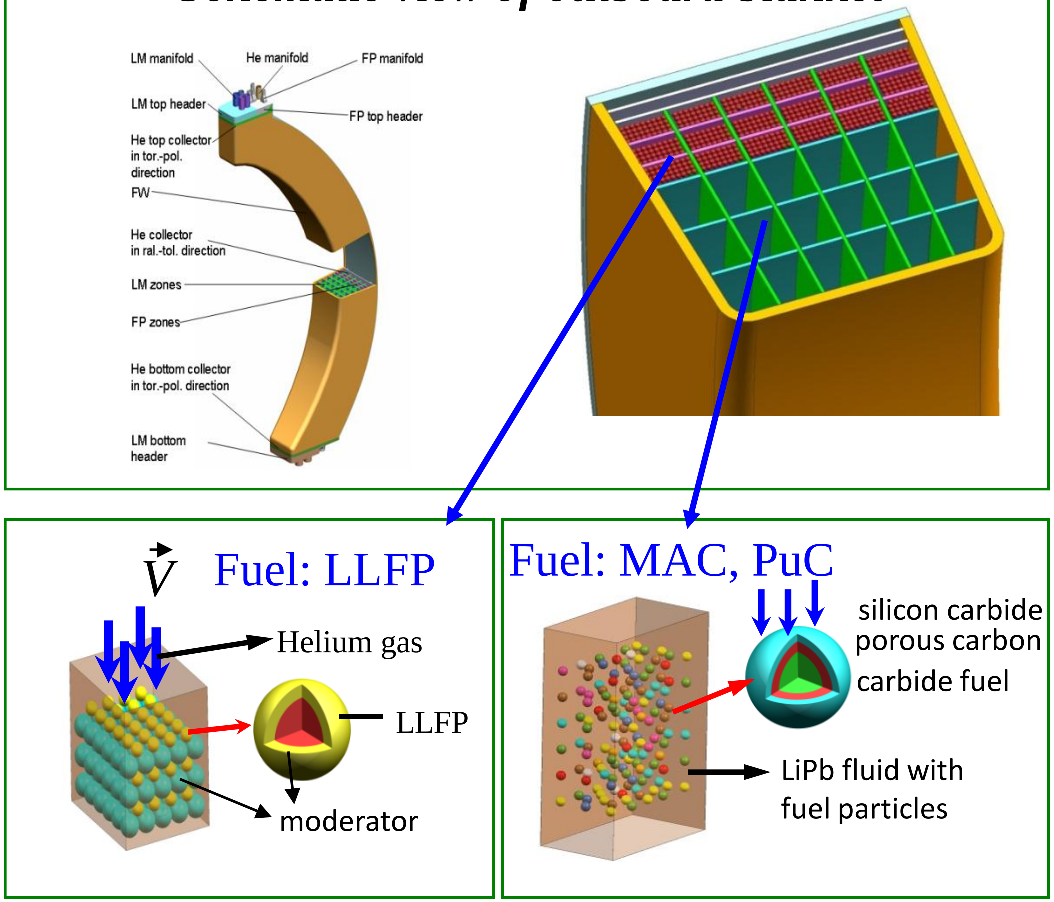
Actinide Zone:

- Coolant: LiPb
- Flow scheme: Poloidally
- Fuel style: Pebble bed

Fission Production Zone:

- Coolant: LiPb
- Flow scheme: Poloidally
- Fuel style: Coated Particle

Schematic view of outboard blanket



Objective Parameters' Definitions

- **M: Blanket Energy Multiplication Factor**

  Ratio of fission power produced by FDS-SFB to the source neutron power (80% of fusion power in D-T fusion）

- **BSR: Breeding Support Ratio**

  Ratio of the fissile Pu mass bred by FDS-SFB to the fissile Pu mass depleted (~400kg) by a referred PWR(1GWe) per year

- **TSR: Transmutation Support Ratio**

  Ratio of the transuranium (TRU) mass transmuted by FDS-SFB to the transuranium mass produced by a referred PWR(1GWe) per year

FDS-SFB Design Summary

1. FDS-SFB concept are designed based on available or very limitedly extrapolated fusion (i.e. a fusion power of 50~150MW) and fission technologies.
2. Three types of blanket concepts with various types of coolants (Water, Helium, Helium/LiPb) and fission fuels have been developed.
3. The neutronics analyses showed

   the max. energy multiplication M can be ~130,

   the max. fissile fuel breeding ratio BSR can be 5~10,

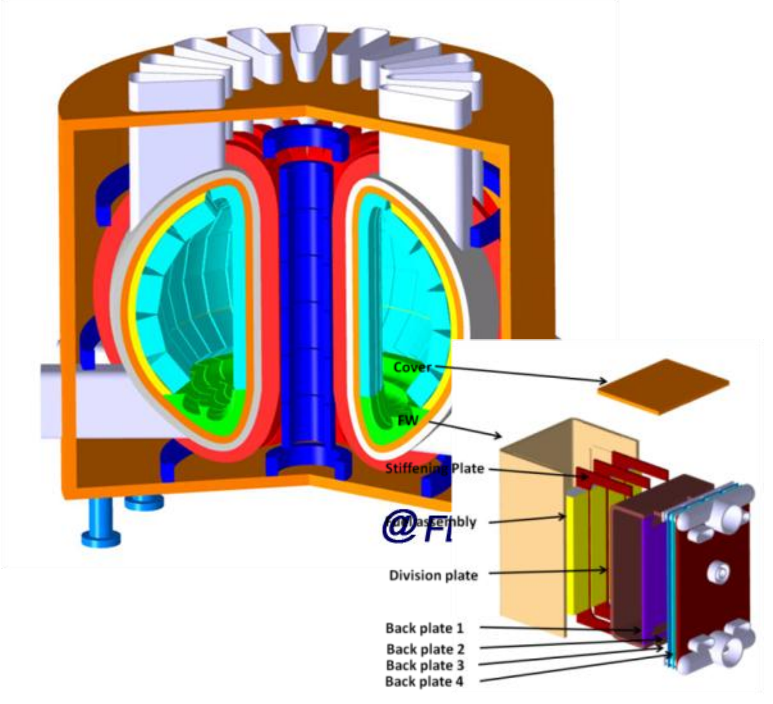
   the max. waste transmutation ratio TSR_{TRU} can be 5~10,

   depending on specific designs.
4. The thermohydraulics/thermomechanics analyses preliminarily showed the concept is feasible and achievable

FDS-MFX: Multi-Functional eXperimental Reactor

Multi-Types-of-Blankets

Multi-Testing-Phases



■ **1st Phase** (~3 years)**:**

- **Tritium breeding blanket**

■ **2nd Phase** (2 years)**:**

- **Natural uranium blanket** (~2 years)**:**

For hybrid reactor principle validation

■ **3th Phase** (3 years)**:**

- **Enriched uranium blanket** (3 years)**:**

－High enriched U fuel is adopted in several modules (64% enriched U), to achieve high power density and high neutron flux test

－Natural uranium is used as fuel in other modules

■ **4th Phase** (≥5 years)**:**

- **Spent fuel blanket**

- Spent fuel is adopted in the blanket

Blanket Concept

■Outboard blanket

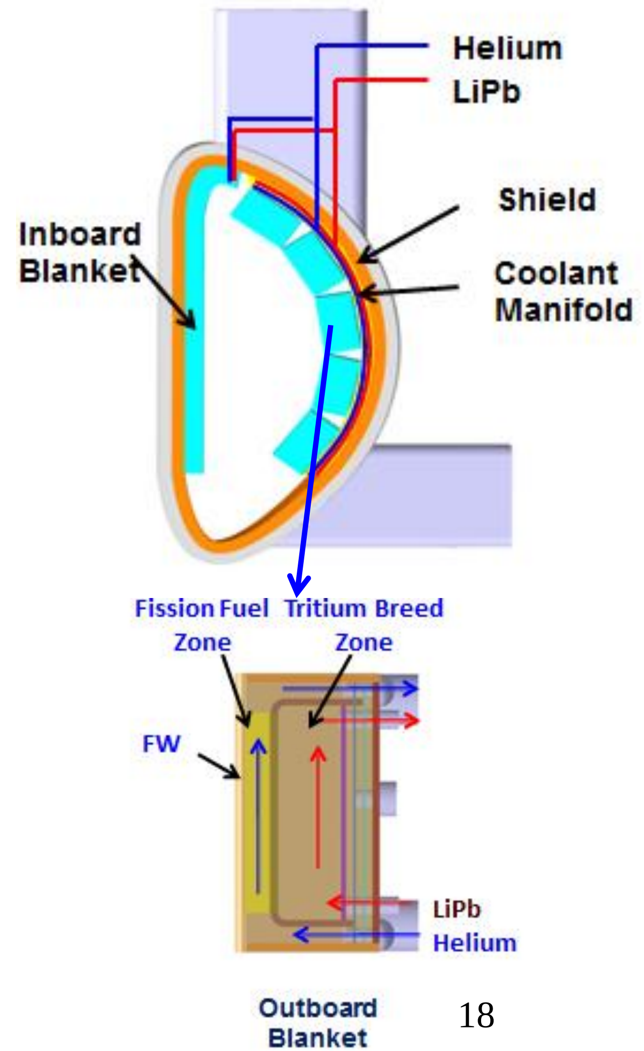
- Multi module segment;
- 32 segments in toroidal;
- 5 blanket module in one segment;
- Blanket is divided into two zones along the radial direction: fission zone and the tritium breeding zone.

■ Inboard blanket

- Banana type;
- 24 segments in toroidal;
- Only for tritium breeding.

■ Material

- Structure: RAFM (e.g. CLAM);
- Tritium Breeder: LiPb;
- Coolant of Fission Zone: Helium.

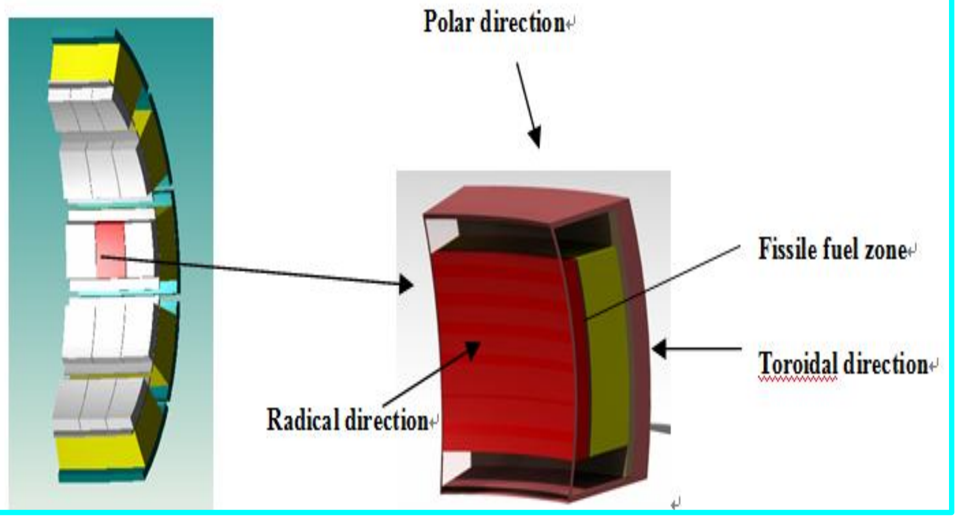


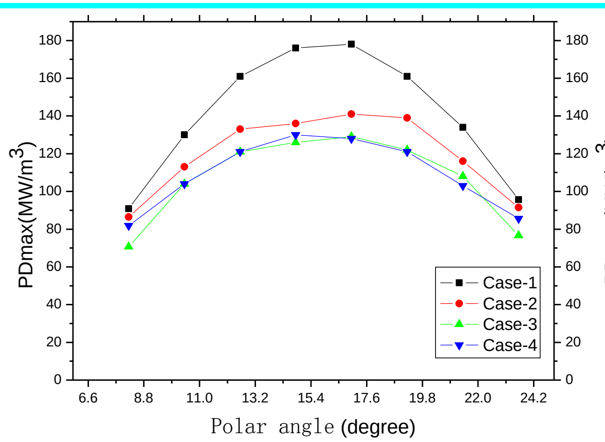
ASIPP · USTC

Neutronics Analysis

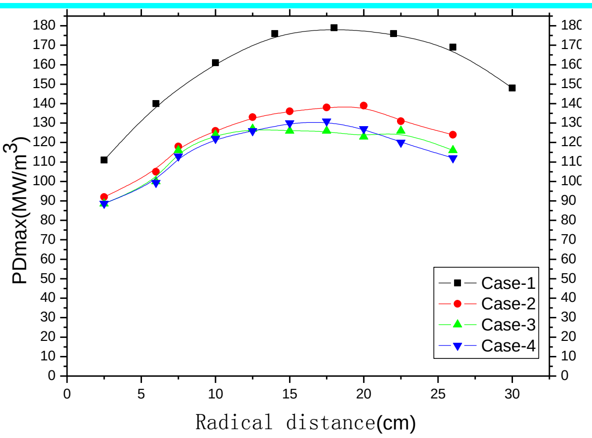
Enriched Uranium Phase

- **Power density profile**
  - ✓ Toroidal profile
  - ✓ Polar profile (Coolant flow direction)
  - ✓ Radical profile
- **Case 3 is chosen as the typical design considering fissile fuel loaded mass and power density flattening**

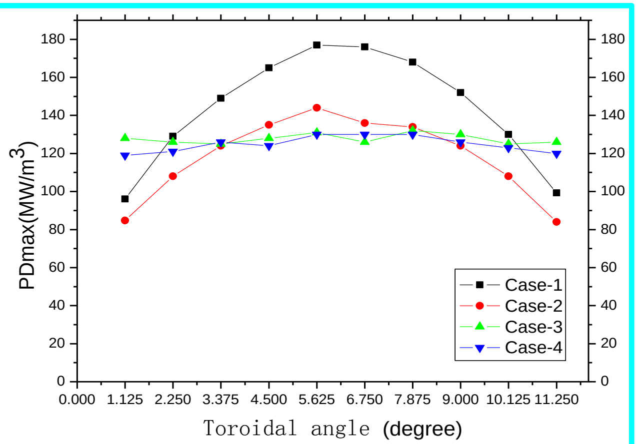




Toroidal Distribution of Power Density



Radial Distribution of Power Density



Polar Distribution of Power Density

Irradiation Damage Analysis

Uranium Phase

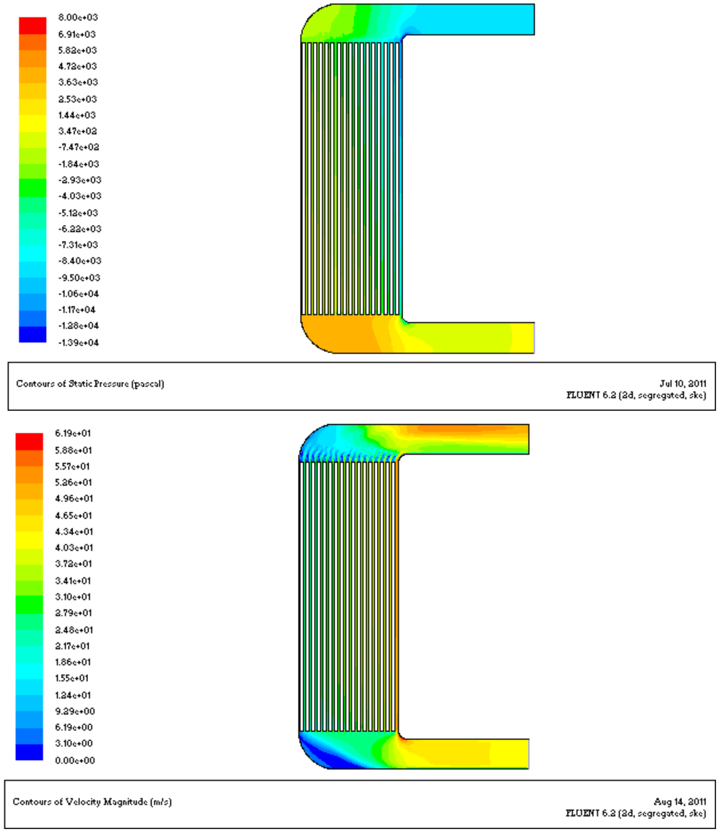
	Natural Uranium Blanket FW	Enriched Uranium Blanket FW
Full Power Years	5	3
DPA Production Rate (10^{-7}dpa/s)	0.61	1.3
Helium Production Rate (10^{-7}dpa/s)	5.12	5.3
DPA	9.6	12.5
Helium Production (appm)	81	50.6

The structure material can bear the irradiation damage

Thermal-hydraulics Analysis

Enriched Uranium Phase

Input Conditions		
Zones	Fission	Tritium Breeding
Coolant	Helium	LiPb
Power density	102MW/m^3	0.579MW/m^3
Coolant volume fraction	40%	100%
Operation Pressure	8MPa	/
Inlet Temperature	300°C	300°C
Outlet Temperature	500°C	320°C
Results		
Velocity	41m/s	0.009m/s
Pressure drop	0.028MPa	/



Temperature and Helium Velocity Distributions of Fission Zone

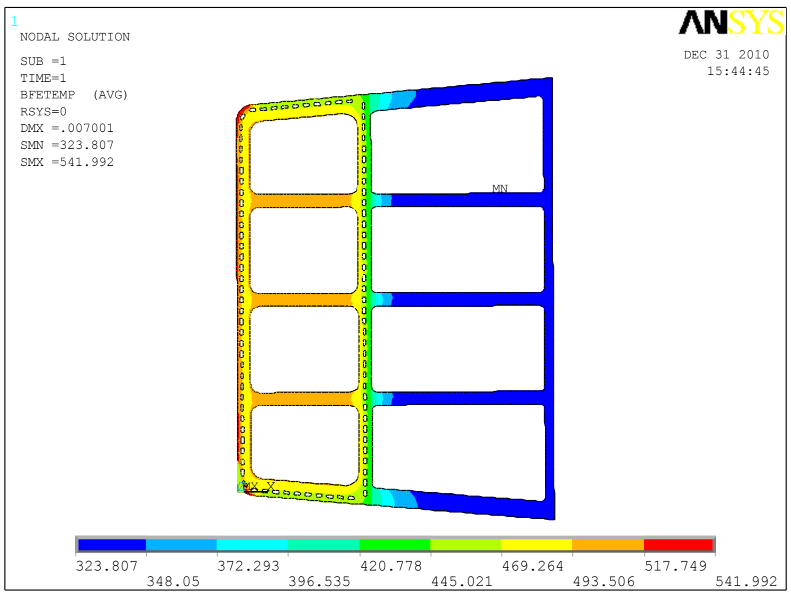
Heat can be removed effectively, the pressure drop and max velocity of He is acceptable.
The velocity of LiPb is very slow.
The pressure drop of He in turnings, manifold will be investigated.

Thermo-mechanical Analysis

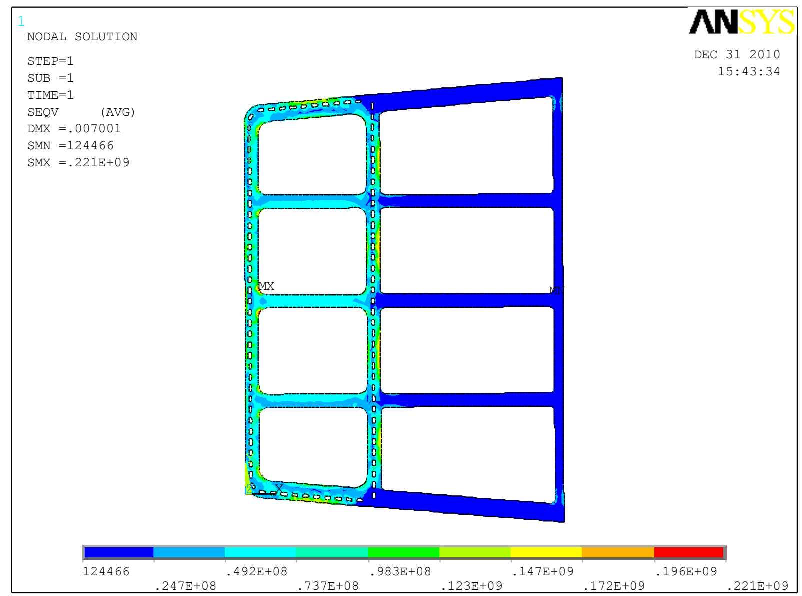
Enriched Uranium Phase

2D calculation model: Radial-toroidal plane. Including FW, fission fuel zone and tritium breeding zone.

Boundary conditions: FW heat flux, coolant convection, fuel heat source and structure heat source.



Temperature Distributions



Stress Distributions

The maximum temperature is 541ºC , under the limited temperature of the structure material.
The maximum thermal stress is 221MPa, satisfying 3Sm criteria of structure material.

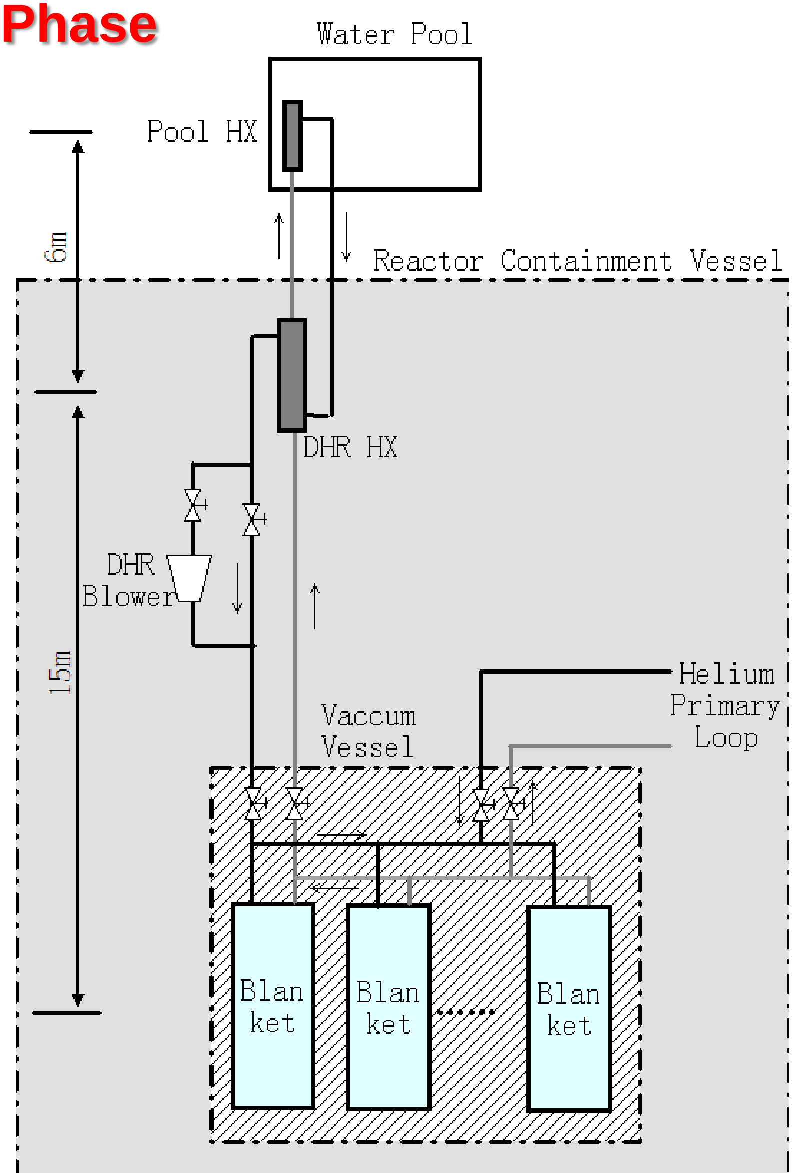
Safety Analysis

Enriched Uranium Phase

❑ Decay Heat Removal (DHR) system

- There are two DHR system (2x100% redundancy)
- Taking into account depressurization accident, auxiliary pumping power devices are needed

	DHR Loop	Secondary Loop
Coolant	Helium	Pressurized Water (~10bar)
Height	15m	6m
Driving Force	Power(Grid/ Diesel/ Batteries) or Natural	Natural



FDS-MFX Design Summary

1. A hybrid experimatal system are conceptually designed based on available or very limitedly extrapolated fusion (a fusion power of 50MW) and fission technologies (Gas-cooled Fast Reactor technologies).

2. Preliminary design of blanket and tritium systems has been carried out.

3. Preliminary neutronics/ thermal-hydraulics/ thermo-mechanical/ safety analyses have been carried out to assess the feasibility, and the results showed those designs can be conceptually achievable.

4. Further engineering design and analysis are needed/underway.

What we have done

for MFR blanket technology R&D

@FDS Team

Material & Blanket Technology R&D

1. China Low Activation Martensitic Steel (CLAM)

   and TBM Fabrication

2. Functional Materials and components:

   Anti-corrosion/tritium barrier/electrical insulation;

   SiC_f /SiC composite (FCI/Loop)

3. PbLi-Loops & PbBi-loops

4. High Intensified Neutron Generator (HINEG)

⇧

Key Technologies for Subcritical System

Ton Level Melting of CLAM Steel

■ 4 ton ingot smelting (2011)

- **Main compositions**

Elements	Fe	C	Mn	Cr	W	V	Ta
Content (wt%)	Bal.	0.10	0.45	9.0	1.5	0.20	0.15

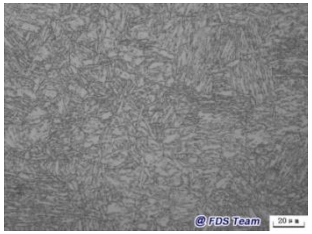
- **Procedure in production**
  - VIM (Vacuum Induction Melting)
  - VAR (Vacuum Arc Remelting)

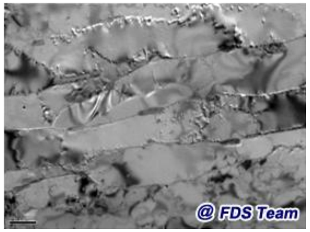

A half of 4 ton ingot after VIM

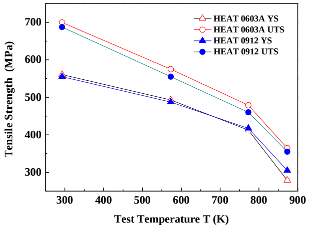

Furnace for VIM

- VAR (Vacuum Arc Remelting) underway

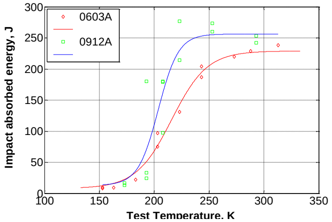
■ 1.2 ton ingot smelting (2009)







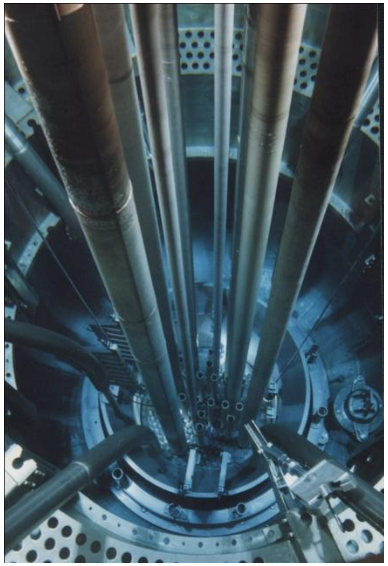
Tensile properties



Charpy impact properties

Composition and mechanical properties agree with the requirements of design

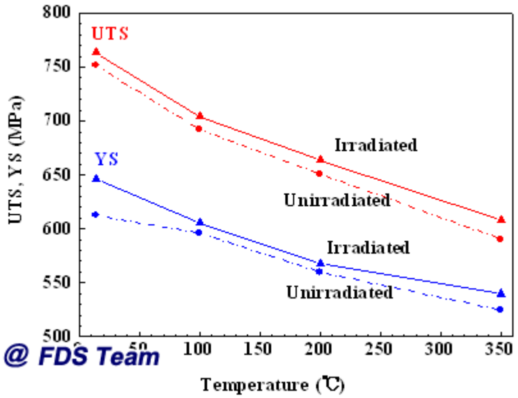
Irradiation Test of CLAM Steel



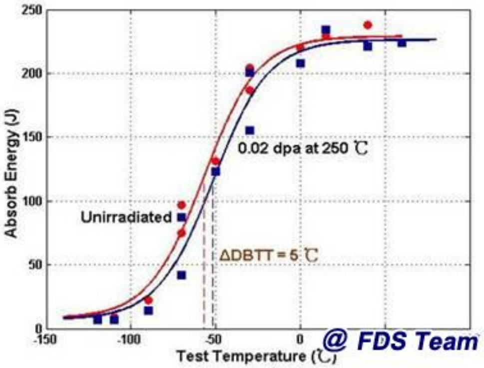
High Flux Engineering Test Reactor (HFETR) in China

Neutron Irradiation tests (~2dpa, ~300℃) are underway.

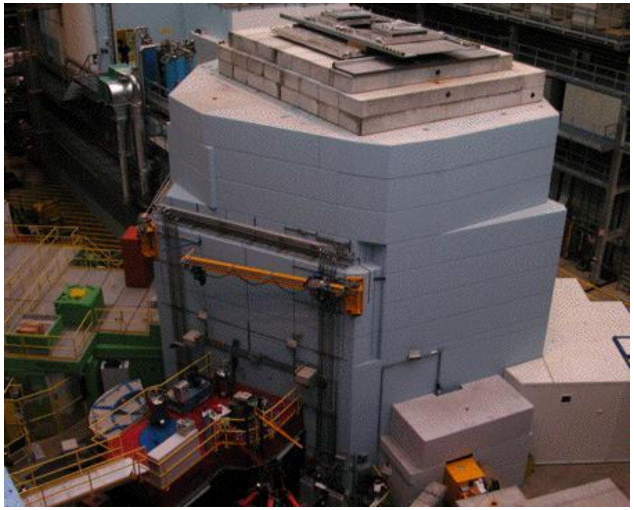
Irradiation Temp.: 250℃, DPA: 0.02



Tesile properties



Charpy impact properties



Spallation Neutron Source, SINQ, in PSI, Switzerland

Spallation irradiation tests (10~20 dpa, 100~500℃) was finished.

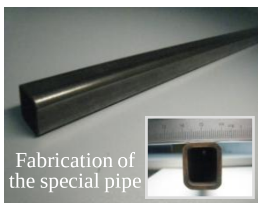
PIE is underway.

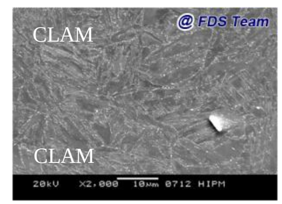
Ion and electron irradiation tests were aslo done to investigate irradiation effects.

Fabrication and Manufacture of TBM

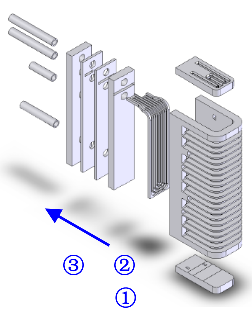
- **Following the design & test strategy of DFLL-TBM, exploration for the fabrication and manufacture technique of TBM are being performed.**

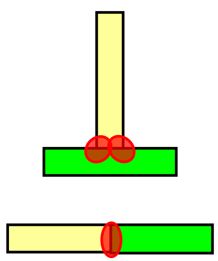
⌖ Exploration for fabrication of the FW and CP by HIP welding

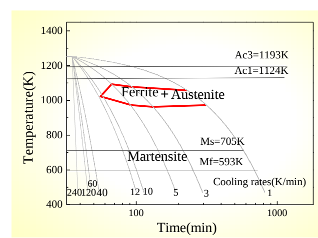




⌖ Exploration for assembly of the key components by EB Welding







⌖ Small mockups

Fabrication of SiC_f/SiC Composites

Requirements:

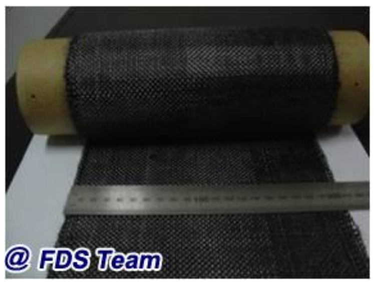
- **Low / high thermal conductivity**
- **Low electrical conductivity**
- **Good compatibility with LiPb under elevated temp.**
- **Stable under neutron irradiation**

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

SiC fiber braid tube

Connection of metal and SiC composite

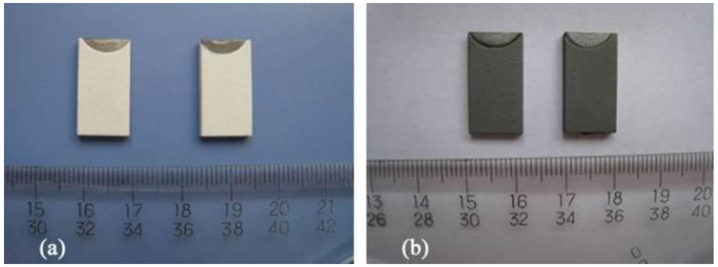
SiC_f/SiC composites were fabricated by CVI + PIP + CVD.

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

Development of Functional Coatings

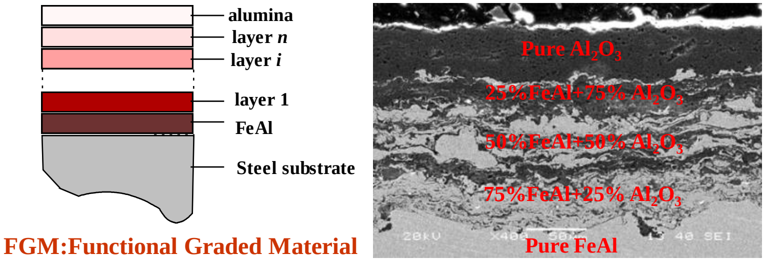
Coating fabrication

✓ Al_2O_3 / SiC Coatings



(a) Al_2O_3 (APS)   (b) SiC (MS) on Al_2O_3

✓ FeAl/Al_2O_3 Coatings

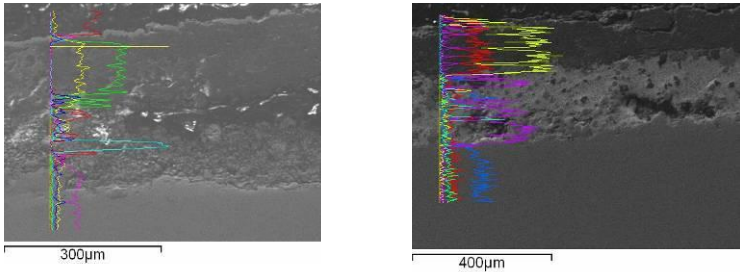


FeAl/Al_2O_3 FGM coatings (VPS)

Coating compatibility

✓ Experiment in the static isothermal capsule

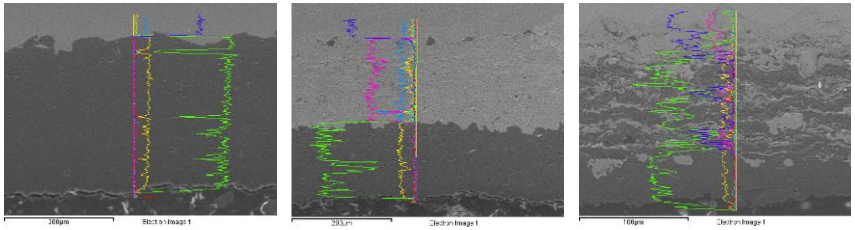
- Al_2O_3, Al_2O_3/SiC coatings (550℃, 5000h)



(a) Al_2O_3 (APS)   (b) Al_2O_3/SiC (MS)

✓ Experiment in the revolving isothermal capsule

- Al_2O_3, Al_2O_3/SiC, FeAl/Al_2O_3 coatings (550 ℃, 0.16m/s, 300h)



(a) Al_2O_3 (APS)   (b) Al_2O_3/SiC (MS)   (c) FeAl/Al_2O_3 (VPS)

- Both coatings showed a higher bond strength with CLAM steel, porosity was controlled at a low level;
- Al_2O_3/SiC coating showed excellent electrical resistivity;
- FeAl/Al_2O_3 coating showed excellent shock resistance.

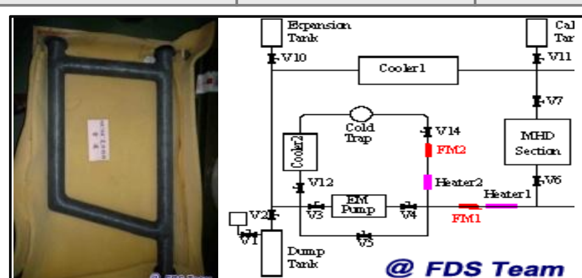
- There's no obvious thinning of external Al_2O_3, SiC layers after 5000hrs exposure in static LiPb.
- The coatings showed good compatibility with flowing liquid LiPb after 300hrs exposure.

Development of DRAGON Series LiPb Loops

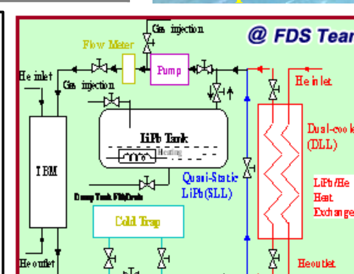
Loops	Type	Experimental functions	Parameters	Construction period
DRAGON-I	TC	Compatibility experiments	420~480℃	2001-2005
DRAGON-II	TC	Compatibility experiments	550~700℃	2004-2006
DRAGON-III	TC	Compatibility experiments	800~1000℃	2011-2012
DRAGON-IV	FC	Compatibility, thermal hydraulics, reference blanket module, MHD experiments	480~800℃	2007-2009
DRAGON-V	FC	Blanket module for dual cooling experiment, complex channel MHD experiment	300~700℃	2011-2013
DRAGON-VI	FC	EAST-TBM auxiliary systems	-	2013-2015
DRAGON-VII	FC	ITER-TBM auxiliary systems	-	2015-2018
DRAGON-VIII	FC	DEMO blanket auxiliary systems		

□ DRAGON-I

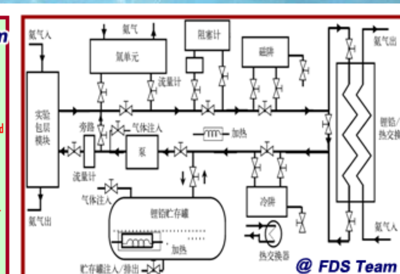
□ DRAGON-II



□ DRAGON-III

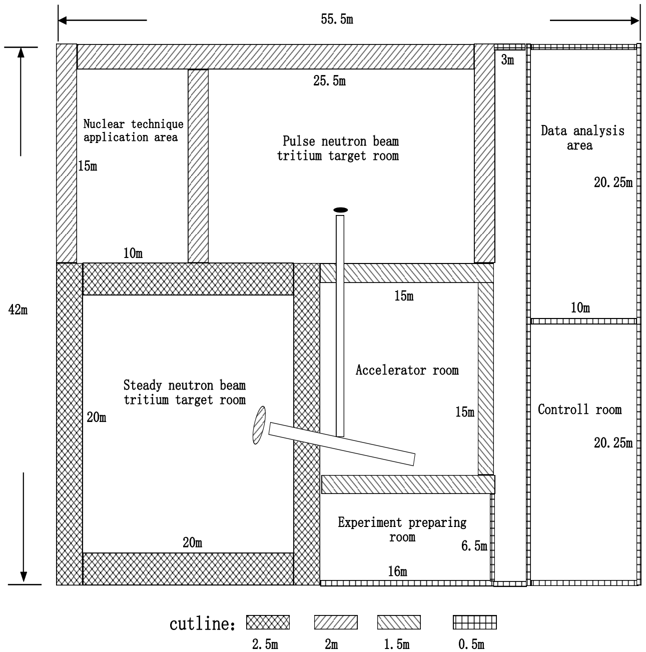


□ DRAGON-V

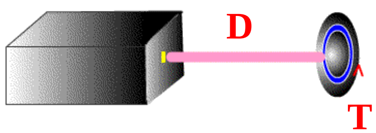


□ DRAGON-VI

Accelerator-Based Fusion Neutron Source & Zero-Power Subcritical System



Functions:



1. **Neutronics experiments:**
   - ＊ Validation of codes and data
   - ＊ Nuclear data measurement
   - ＊ Materials activation and irradiation
2. **Radiation protection studies:**
   - ＊ Neutron shielding
   - ＊ Neutron detection
   - ＊ Neutron biology influence
3. **Nuclear technique applications**

Neutron dose detector

Intensity survey equipment

Energy spectrum survey equipment

Tritium monitor

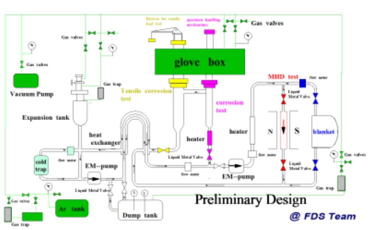
Photon energy spectrum detector

Neutron scout device

Testing Strategy of Blanket to ITER/MFX

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

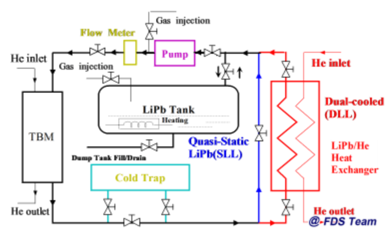
Thermal convection loop
Forced convection loop



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

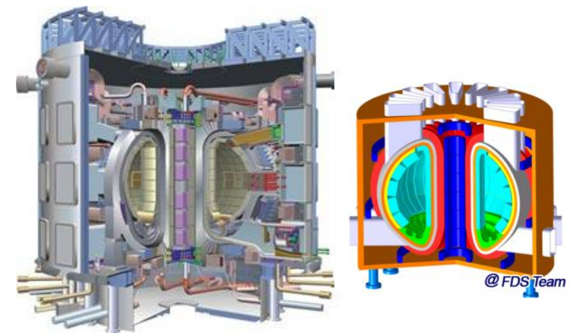
Stage II: Test in EAST
(1/2 size)

LiPb/He system
for TBM in EAST

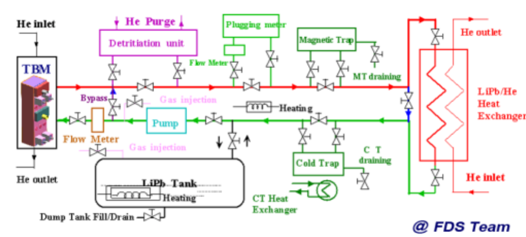


- EM and thermo-mechanics
- Partially neutronics performances
- Influence on plasma confinement
- Thermaldynamics and MHD
- Diagnostic and measurement

Stage III: Test in ITER/MFX
(full size)



LiPb/He system
for TBM in ITER



- To confirm results of EM, thermo-mechanics test in EAST
- To test neutronics, tritium production, fission and integration performances in ITER/MFX

Advanced Reactor Simulation Software R&D

- Multi-functional integrated 4D neutronics simulation system: VisualBUS
- Multi-physics (neutronics/thermohydraulics/MHD) coupled simulation codes: NTC/MTC
- Tritium Analysis Program for Fusion System: TAS
- System (safety/economics) analysis codes: RiskA, RiskAngel, SYSCODE
- Database Management System for Fusion： FusionDB
- Integrated Design and Simulation of Advanced Reactors： 4DS

> 500 person-year investment

Key Tools for Design and Analysis

ASIPP · USTC

VisualBUS

Multi-Functional 4D Neutronics Simulation System

Main Functions:

CAD-based/Imaged-based Modeling

- Monte Carlo (MC) geometries
- Discrete Ordinates (SN) geometries
- MC-SN coupled geometries
- CT/MRI/Color images

4D Coupled Multi-Process Calculation

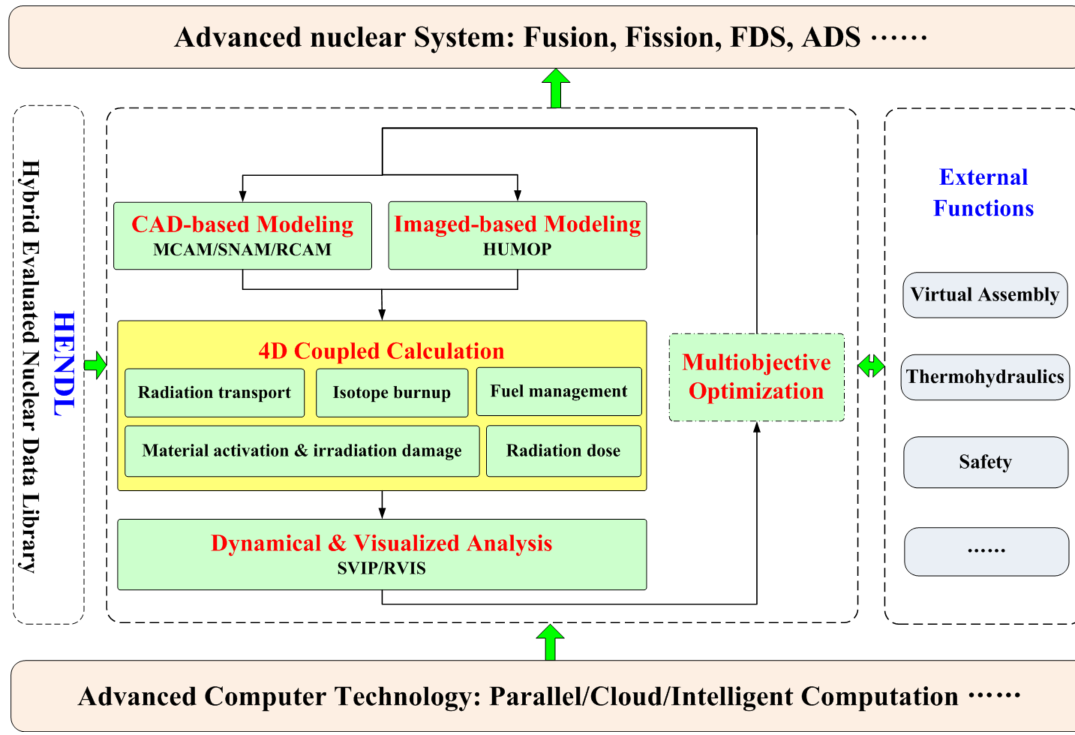
- Radiation Transport
- Isotope Burnup
- Material Activation & Irradiation Damage
- Radiation Dose
- Fuel management

Dynamical &Visualized Analysis

- Static / dynamic physical data fields
- Human virtual roaming & dosimetry assessment

Multi-objective Optimization

- Artificially intelligent algorithms
- Space optimization of irregular complex solutions

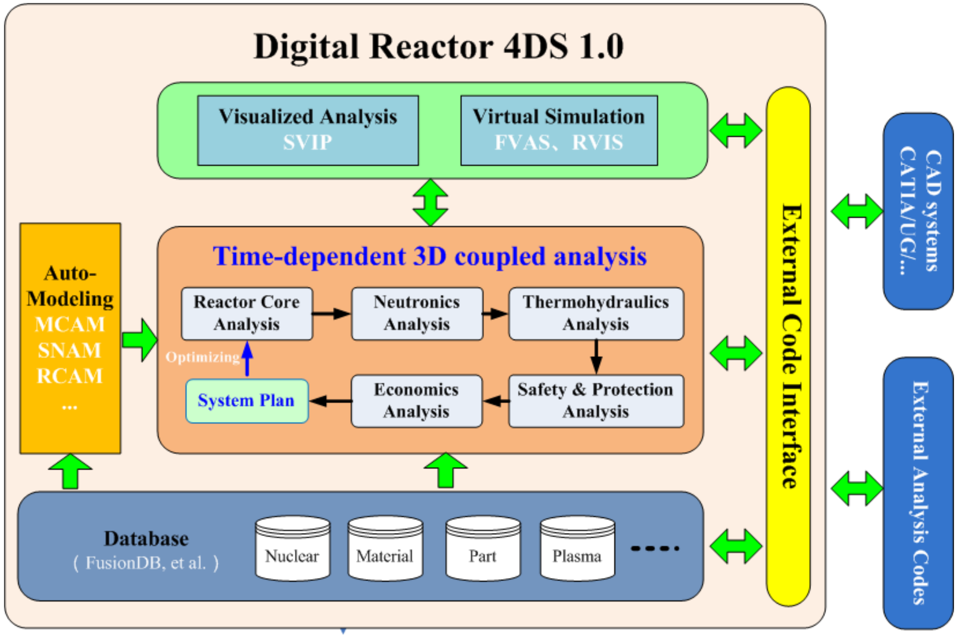


- **Hybrid Evaluated Nuclear Data Library** for fusion/fission/hybrid systems
- **External functions for other physics process simulations** such as virtual assembly, thermal-hydraulics, safety, environmental impact and economics etc.

4DS: 4-Dimensional System
for Integrated Design and Simulation of Advanced Reactors

- Time-dependent 3D accurate calculation based on multi-physics coupling concept
- Auto-modeling & visualized analysis
- Virtual roaming & assembly

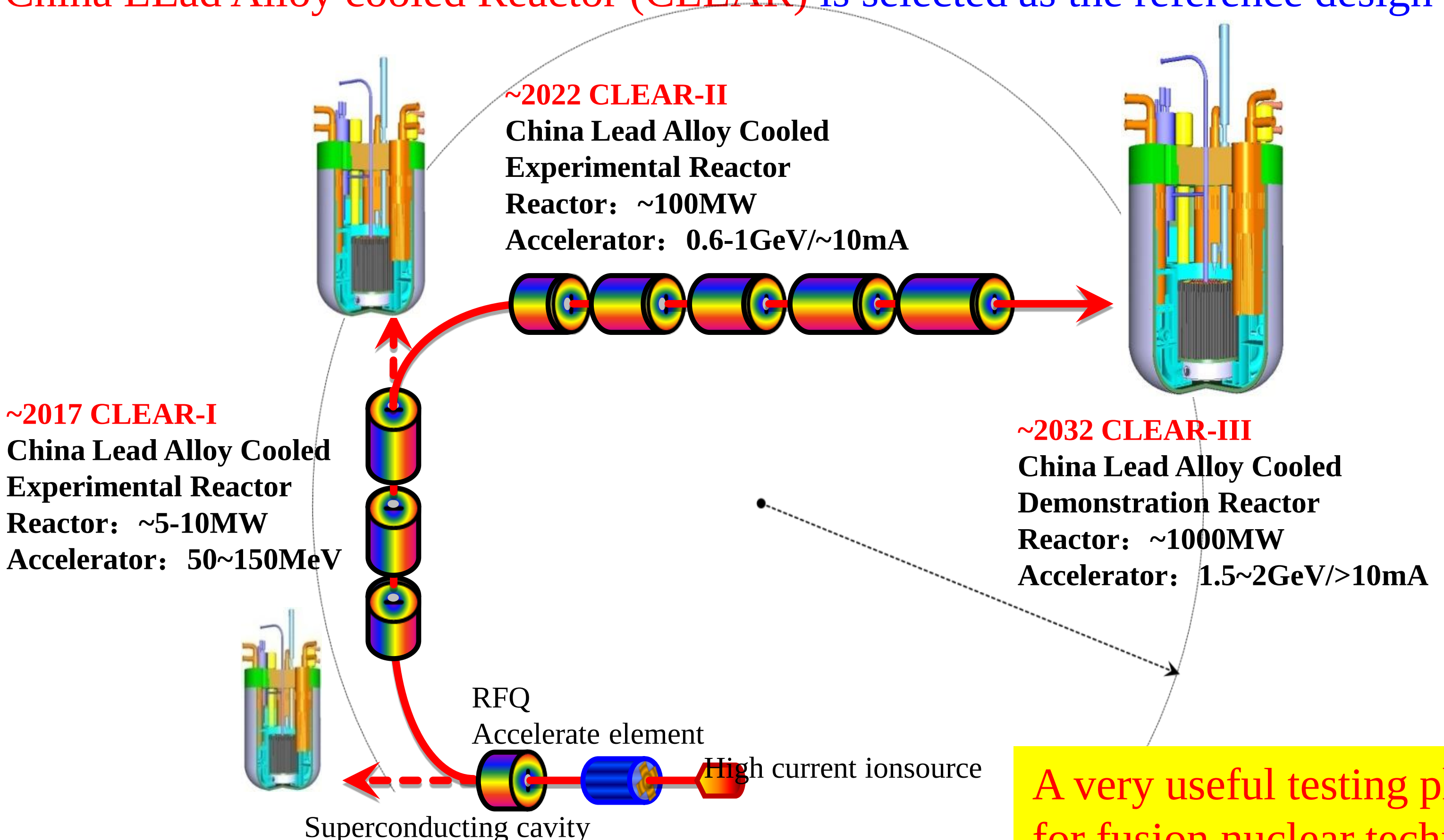
- Integration with design & simulation
- Auto coupling each process
- Easy to integrate new-developed codes, due to hierarchical design



Ideal design & simulation platform for advanced nuclear energy systems (fusion/fission reactors, FDS/ADS sub-critical systems, etc.)

Roadmap of ADS Development in China

- Chinese Academy of Sciences (CAS) has been carried out an ADS Project, and plan to construct demonstrated ADS transmutation system ~ 2032 .
- China LEad Alloy cooled Reactor (CLEAR) is selected as the reference design

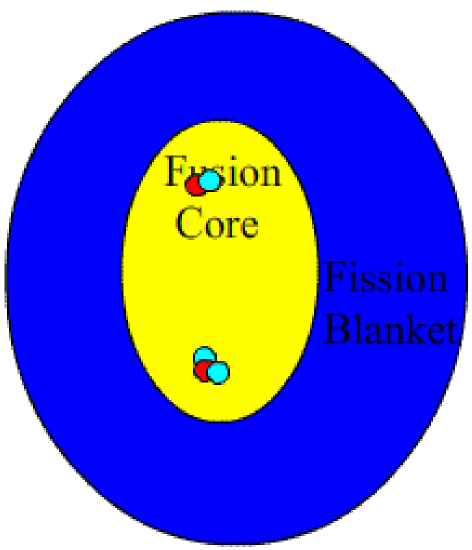


A very useful testing platform for fusion nuclear technology

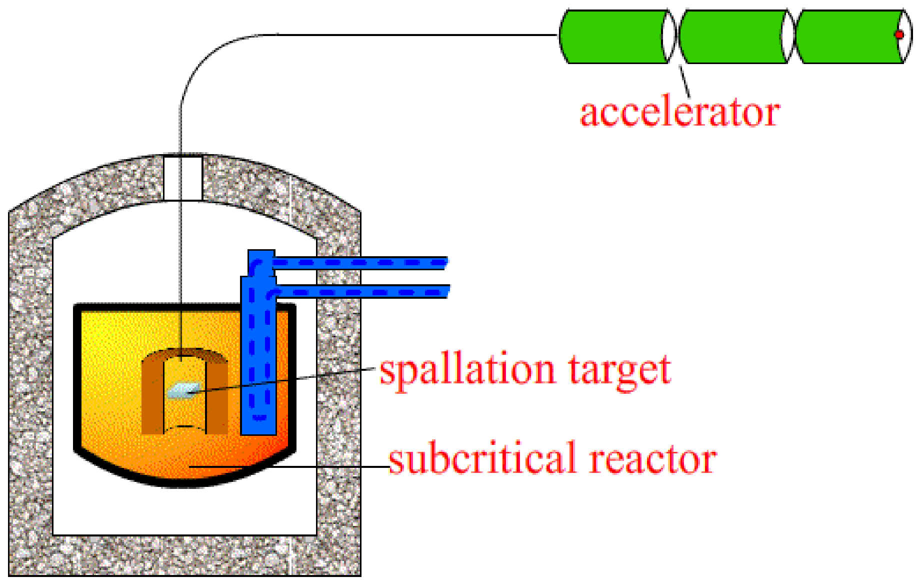
Hybrid Nuclear Energy System – FDS & ADS

Neutron Energy: ~14MeV (fusion), ~10MeV (ADS)

Coolant: PbLi/He (fusion), PbBi (ADS)



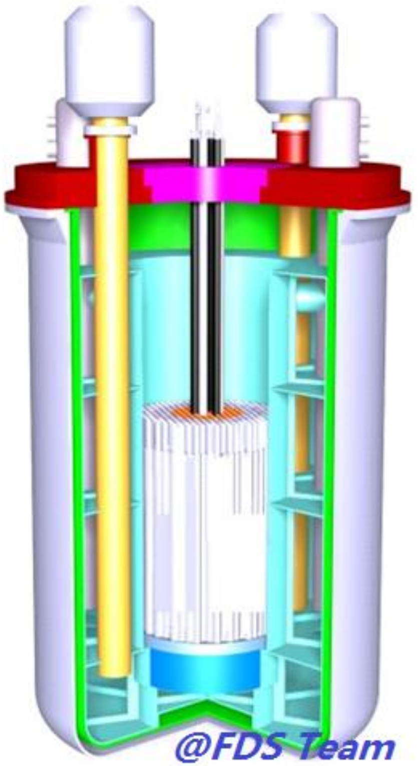
FDS: Fusion-Driven Subcritical System



ADS: Accelerator-Driven Subcritical System

CLEAR-I Design Parameters

Parameter		Values
Thermal power (MW)		**10**
Core	Activity height (m)	0.86
	Activity diameter (m)	1.5
	Fuel (enrichment)	UO_2(19.75%)
Cooling system	Primary Coolant	LBE
	Inlet/Outlet Temperature	300/363
	Coolant drive type	Natural circulation
	Heat exchanger	4.5
	Second coolant	Water
	Heat sink	Air cooler
Material	Cladding	316 Ti
	Structure	316L



A preliminary consideration on MFR development roadmap

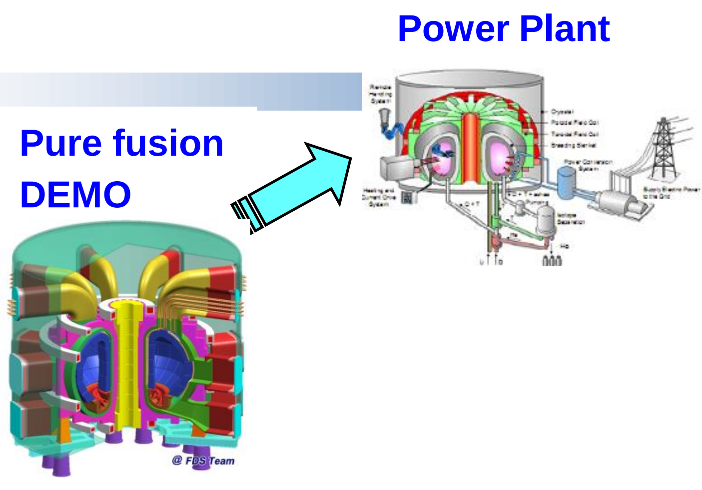
and

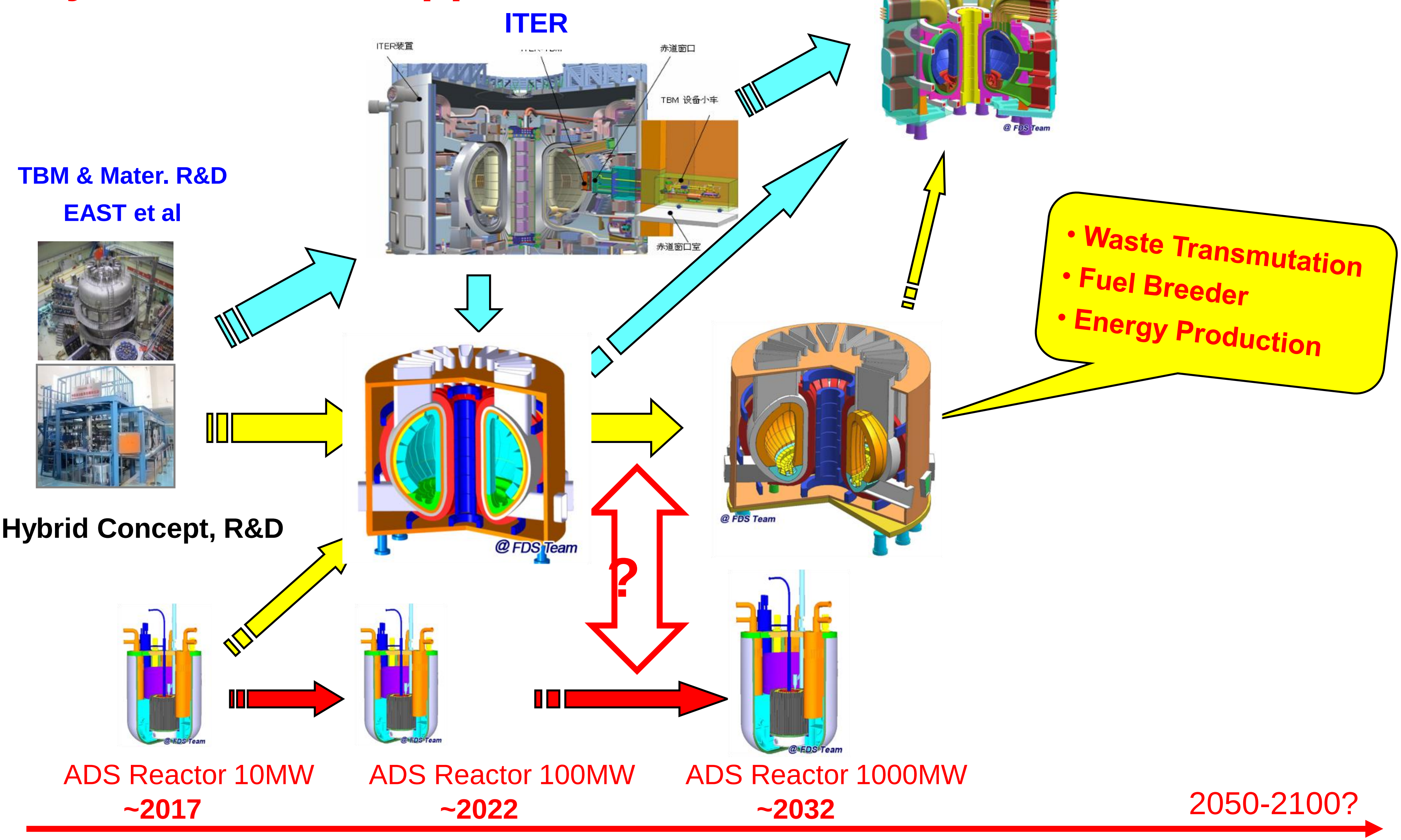
Challenges

Blanket Technology Challenges (critical)

- **Structural & functional materials (anti-irradiation etc.)**
- **Tritium technology and fuel self-sustaining**
- **Spent fission fuel processing and fuel manufacture**
- **High-irradiated and activated component remote handling in complex geometry**
- **Safety issues (LOCA, afterheat etc.) and license**
- **Others**

Roadmap proposal for Hybrid Reactor Application





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- **Conceptual Design and Testing Strategy of a Dual Functional Lithium-Lead Test Blanket Module in ITER and EAST**. *Y. Wu et al*, **Nuclear Fusion**, 2007, 47(11): 1533-1539.

- **Conceptual Design Activities of FDS Series Fusion Power Plants in China.** *Y. Wu, et al*, **Fusion Engineering and Design,** 2006, 81(23-24): 2713-2718

The End

Thanks for your attention !

FDS Website: www.fds.org.cn

Potential Advantages of Subcritical System

- **Rich neutrons to achieve multi-goals**

  (improved neutron balance by external neutron source)
- **Good passive and inherent safety performances**

  (fission reaction shutdown passively when neutron source stop)
- **Lower requirements on driver technologies**

  e.g. plasma-related technology or accelerator-related technology

  (improved energy balance by fission blanket)
- **In general, it can benefit both fusion and fission**

  (fill in the gap, solve left problems by fission, promote fusion)