Design Concept of K-DEMO for Near-Term Implementation

Fusion Energy Conference

Oct 16th, 2014

National Fusion Research Institute

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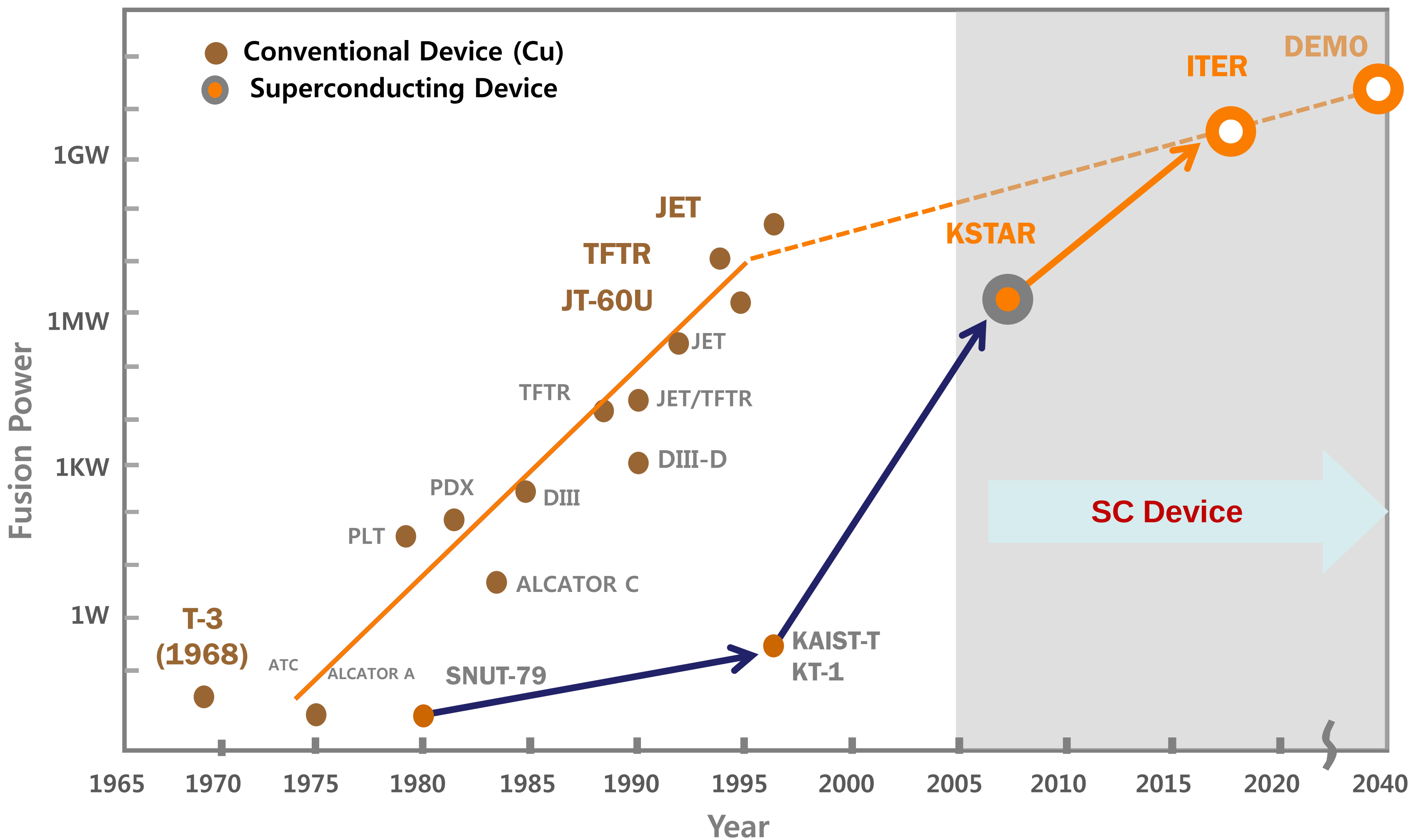
Background

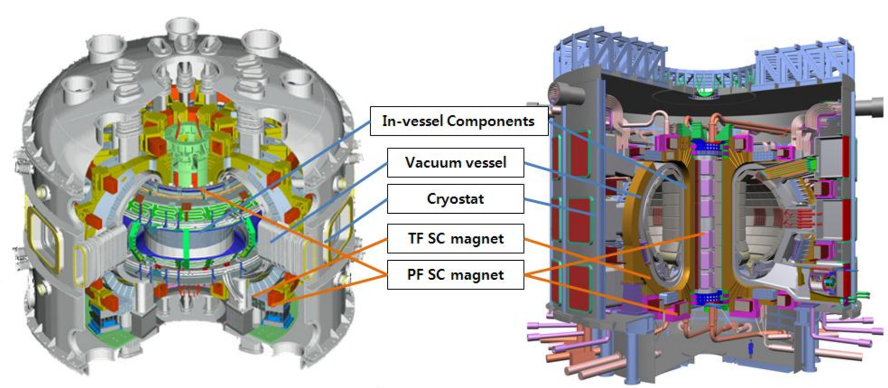
Mid-Entry Strategy in 1995



KSTAR Mission and Parameters

KSTAR Mission

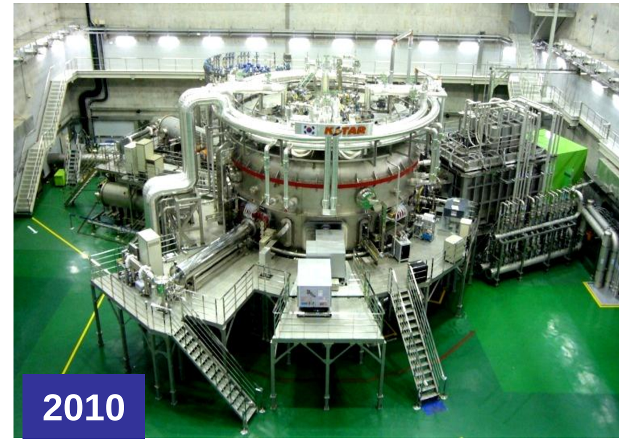
- To achieve the superconducting tokamak construction and operation experiences, and
- To develop high performance steady-state operation physics and technologies that are essential for ITER and fusion reactor development

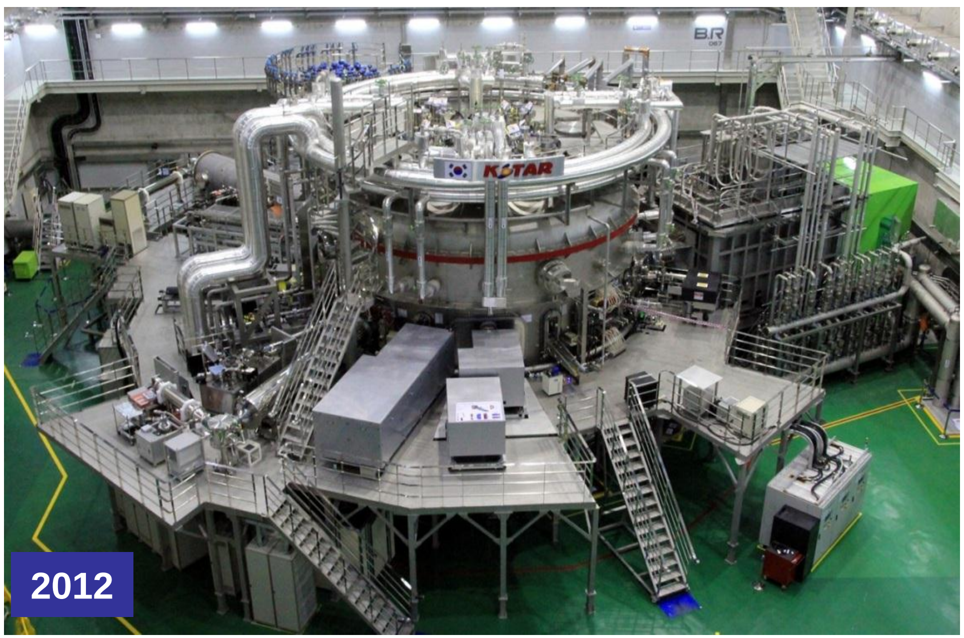


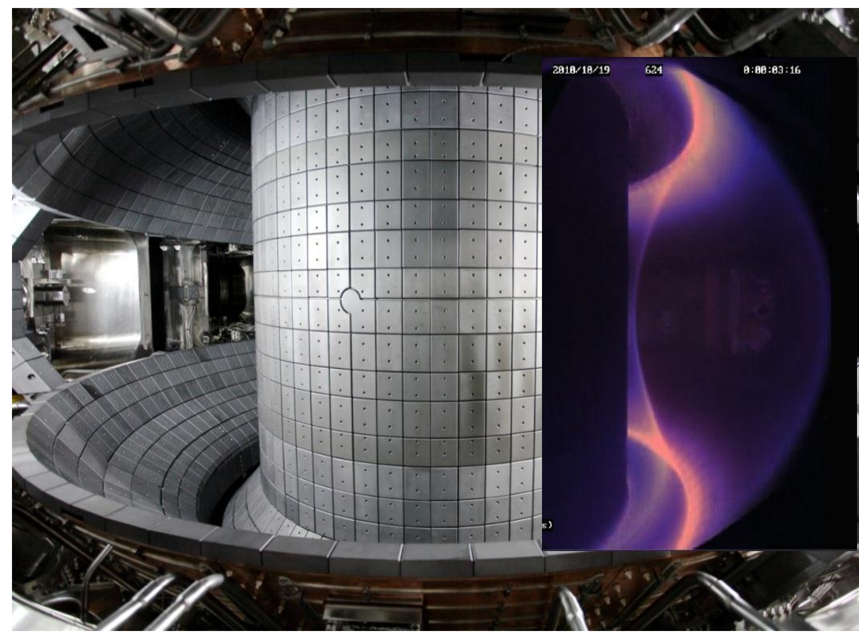
KSTAR Parameters

PARAMETERS	Designed	Achieved
Major radius, R_0	1.8 m	1.8 m
Minor radius, a	0.5 m	0.5 m
Elongation, κ	2.0	2.0
Triangularity, δ	0.8	0.8
Plasma volume	17.8 m^3	17.8 m^3
Bootstrap Current, f_{bs}	> 0.7	-
PFC Materials	C, CFC (W)	C
Plasma shape	DN, SN	DN
Plasma current, I_P	2.0 MA	1.0 MA
Toroidal field, B_0	3.5 T	3.6 T
Pulse length	300 s	10 s
β_N	5.0	> 1.5
Plasma fuel	H, D	H, D, He
Superconductor	Nb_3Sn, NbTi	Nb_3Sn, NbTi
Auxiliary Heating / CD	~ 28 MW	~ 6 MW
Cryogenic	9 kW @4.5K	5 kW @4.5 K

KSTAR Superconducting Tokamak

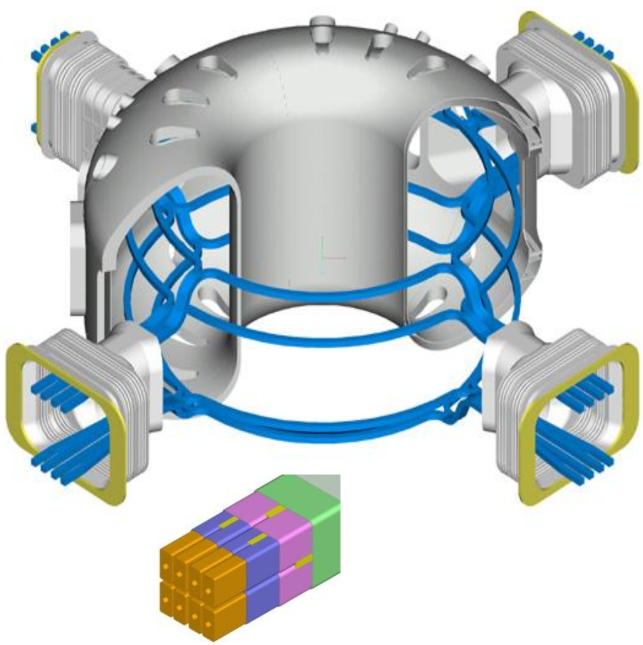






KSTAR In-vessel Control Coil System (3-D)

- **Modular 3D field coils (3 poloidal x 4 toroidal)**
  - all internal and segmented with saddle loop configurations
  - 8 conductors in each coil
  - Control capability : vertical control, radial control, error correction, RMP, RWM
- **Wide spectra of Resonance Magnetic Perturbations (RMP) are possible**
  - n=1 RMP (phase angles : +90, -90, 180, 0) and n=2 RMP (even or odd parity)



Schematics of IVCC and its conductor

n=1, +90 phase

θ

B_P

+	+	-	-	top
-	+	+	-	mid
-	-	+	+	bot

ϕ

n=2, even parity

+	-	+	-	top
-	+	-	+	mid
+	-	+	-	bot

n=1, 0 phase

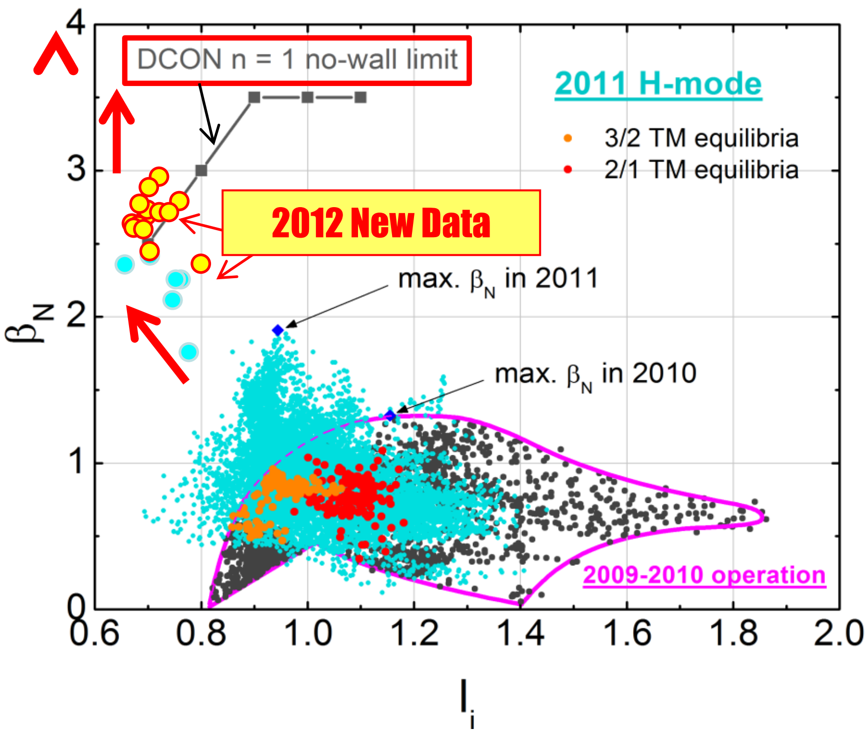
+	+	-	-	top
+	+	-	-	mid
+	+	-	-	bot

n=2, odd parity

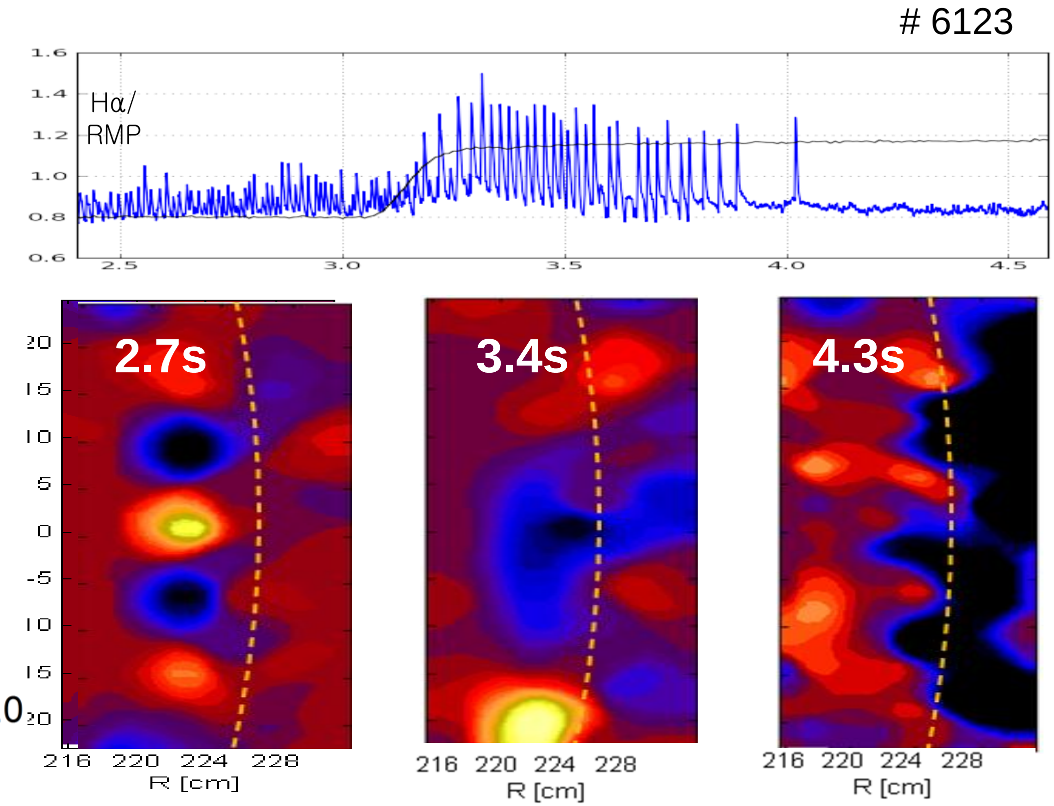
+	-	+	-	top
				mid
-	+	-	+	bot

β_N-limit and ELM Suppression

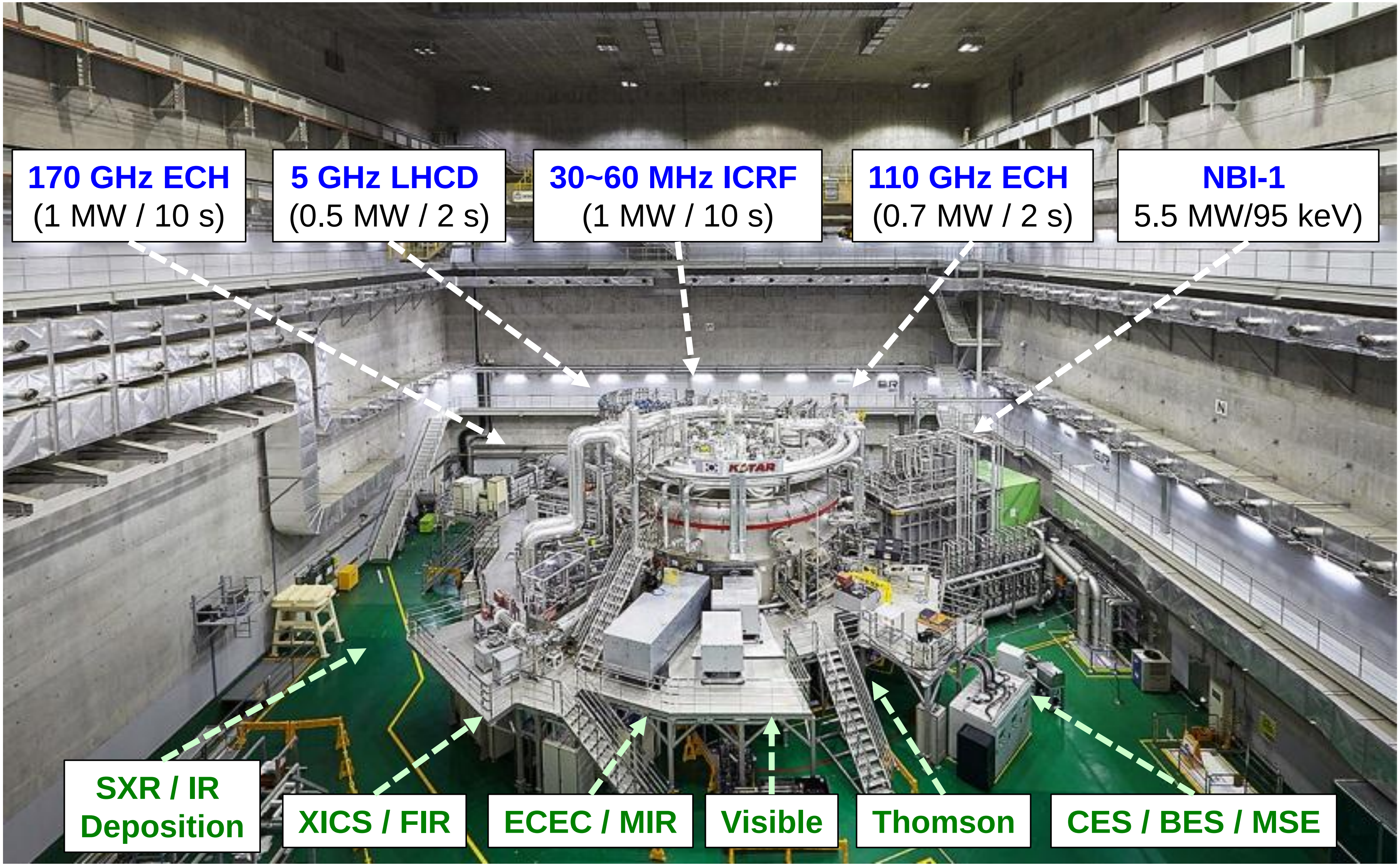
- KSTAR reached β_N〉2.5 and β_N/li〉 4.0 "no-wall limit" in H-mode Operation



- ELM suppressed by n=1 RMP (I_p = 600 kA, B_T=1.6~2.3T)



KSTAR 2014 Campaign (H-mode > 30 sec)



Fusion Energy Development Promotion Law (FEDPL)

- To establish a long-term and sustainable legal framework for fusion energy development phases.
- To promote industries and institutes which participating the fusion energy development by supports and benefit.
- The first country in the world prepared a legal foundation in fusion energy development.

History of the FEDPL

- 1995. 12 : National Fusion R&D Master Plan
- 2005. 12 : National Fusion Energy Development Plan

- 2007. 3 : Fusion Energy Development Promotion Law
- 2007. 4 : Ratification of ITER Implementation Agreement
- 2007. 8 : Framework Plan of Fusion Energy Development (The first 5-Year Plan)
- 2012. 1 : The 2nd 5-year plan has begun

Vision and Goal of Fusion Energy Development Policy

Vision: Secure sustainable new energy source by technological development and the commercialization of fusion energy

Phase	Phase 1 ('07~'11)	Phase 2 ('12~'21)	Phase 3 ('22~'36)
Policy Goal	Establishment of a foundation for fusion energy development	Development of core technology for DEMO	Construction of DEMO by acquiring construction capability of fusion power plants
Basic Directions	▪ Acquisition of operating technology for the KSTAR ▪ Participation in the international joint construction of ITER ▪ Establishment of a system for the development of fusion reactor engineering technology	▪ High-performance plasma operation in KSTAR for preparations for the ITER ▪ Completion of ITER and acquisition of core technology ▪ Development of core technology for the design of DEMO	▪ DEMO design, construction, and demonstration of electricity production ▪ Undertaking of a key role in ITER operations ▪ Completion of reactor core and system design of the fusion power reactor ▪ Commercialization of fusion technology
Basic Promotion Plan	Basic Promotion Plan 1 ('07~'11)	Basic Promotion Plan 2 ('12~'16) Basic Promotion Plan 3 ('17~'21)	Basic promotion plan 4 ('22~'26) Basic promotion plan 5 ('27~'31) Basic promotion plan 6 ('32~'36)

Policy Goal for Plan-2: R&D for DEMO Technology based on KSTAR and ITER

Primary Strategy for Plan-2:

- Attainment of KSTAR high-performance plasma and development of DEMO basic technology
- Basic research in fusion and cultivation of man power
- International cooperation and improvement of status in ITER operations
- Commercialization of fusion/plasma technology and promotion of social acceptance

Introduction

Two Stage Operation

- The operation stage I K-DEMO is not considered as the final DEMO. It is a kind of test facility for a commercial reactor.
- But the operation stage II K-DEMO will require a major up-grade by replacing the blanket & diverter system and others if required.
- The operation stage I K-DEMO
  - At initial stage, many of ports will be used for diagnostics for the operation and burning plasma physics study, but many of them will be transformed to the CTF (Component Test Facility).
  - At least more than one port will be designated for the CTF including blanket test facility.
  - It should demonstrate the net electricity generation ($Q_{eng} > 1$) and the self-sufficient Tritium cycle (TBR > 1.05).
- The operation stage II K-DEMO
  - Though there will be a major upgrade of In-Vessel-Components, at least one port will be designated for CTF for future studies.
  - It is expected to demonstrate the net electricity generation larger than 450 MWe and the self-sufficient Tritium cycle.
  - Overall all plant availability > 70%.
  - Need to demonstrate the competitiveness in COE.

Key Idea of K-DEMO Design

Current Drive and Magnetic Field

- Considering the size, a steady state Tokamak is selected as a K-DEMO.
- Because of high neutron irradiation on ion sources, NBI is not practical for the main off-axis current drive of K-DEMO.
- Because of high density of K-DEMO plasma, high frequency ECCD systems (> 240 GHz) are required in order to minimize the deflection of wave.
- In order to match with the high frequency ECCD, a high toroidal magnetic field Tokamak is required and the magnetic field at plasma center requires > 6.5 T.
- Also, $I_{p,limit} \propto B$, $n_{e,\ limit} \propto B$, and **Power $\propto R^3B^4$** [**Reactor Cost $\propto R^3B^2$**]

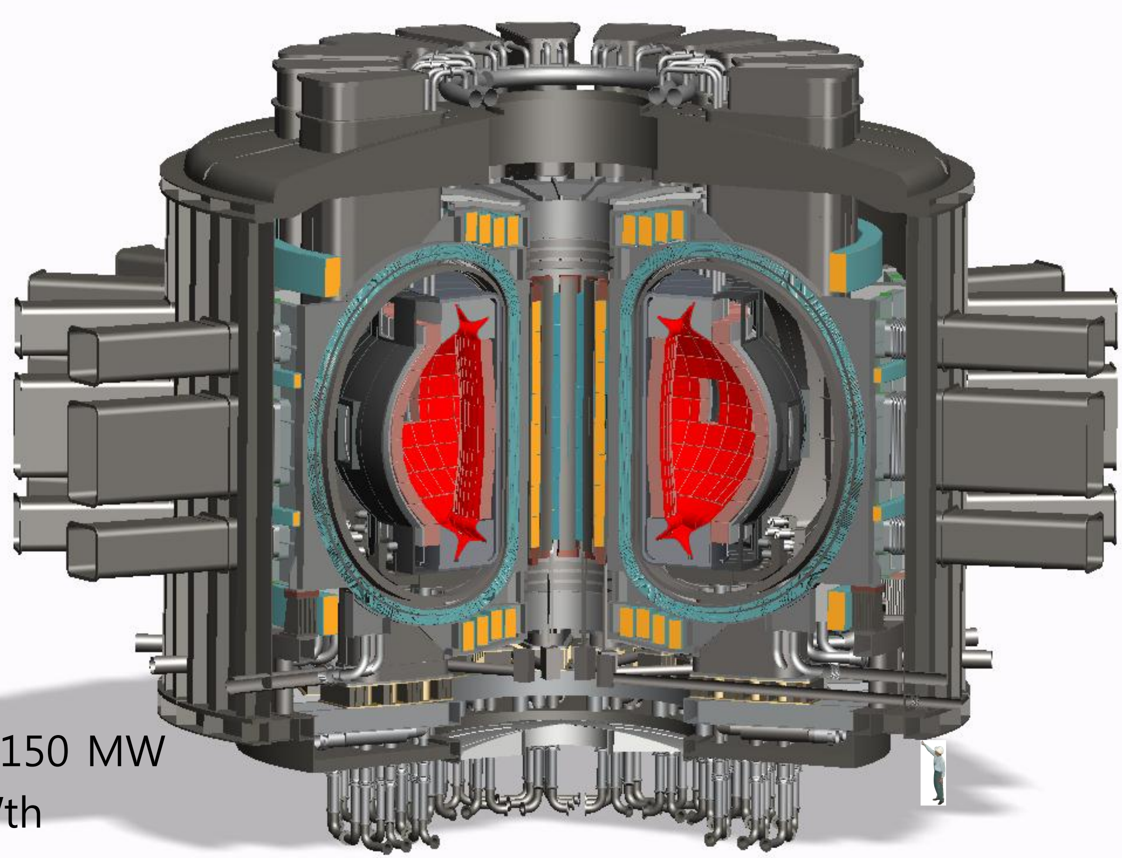
Choice of Coolant and Blanket System

- Helium is not considered as a coolant of K-DEMO because of its low heat capacity and a required high pumping power.
- Supercritical water is not considered as a coolant of K-DEMO because of its serious corrosion problem.
- Pressurized water (superheated water) is considered as a main coolant of K-DEMO considering BOP(Balance of Plant).
- Both of ceramic and liquid metal blanket system is considered at this stage. But even in the liquid blanket system, the liquid metal will not be used as a main coolant and a water cooling system will be installed inside the liquid metal blanket.

K-DEMO Parameters

Main Parameters

- R = 6.8 m
- a = 2.1 m
- B-center = 7.0~7.4 T
- B-peak = 16 T
- κ_{95} = 1.8
- δ = 0.625
- Plasma Current > 12 MA
- Te > 20 keV



Other Feature

- Double Null Configuration
- Vertical Maintenance
- Total H&CD Power = 110~150 MW
- P-fusion = 2200~3000 MWth
- P-net > 400 MWe at Stage II
- Number of Coils : 16 TF, 8 CS, 12 PF

K-DEMO Tokamak Design

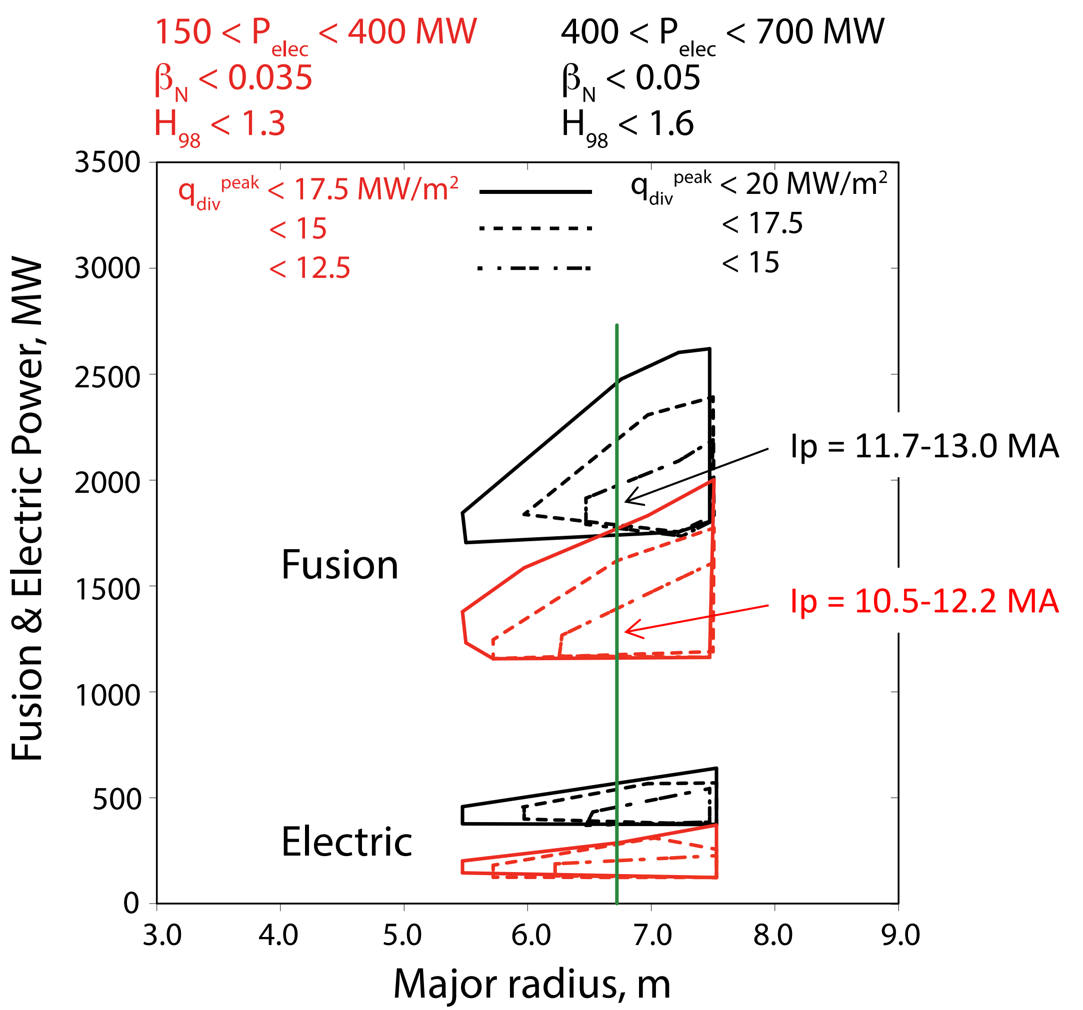
Systems Analysis to Explore Configurations

- Scan plasma parameters R, B_T, q_{95}, κ, δ, n/n_{Gr}, β_N, Q ($=P_{fus}/P_{input}$), $n(0)/\langle n \rangle$, $T(0)/\langle T \rangle$, τ_p^*/τ_E, η_{CD}, f_{imp}
- Solve 0D plasma power and particle balance
- Pass solutions of viable plasma operating points through engineering and inboard radial build assessments
  - Radiated power to first wall and transported and radiated power in the divertor
  - Plant power balance
  - First wall, blanket, shield, vacuum vessel inboard radial build
  - Toroidal field coil
  - Bucking cylinder, TF superstructure
  - Central solenoid
- Filtering viable engineering solutions to meet P_{elec}, q_{div}^{peak}, β_N, H_{98}, etc

※In collaboration with PPPL PRINCETON PLASMA PHYSICS LABORATORY

Determination of Plasma Geometry

- The peak heat flux on the divertor poses a significant limitation
- The q_{div}^{peak} is reduced as the device grows in size
- KDEMO will be a first of a kind, desire to reduce the cost
- A compromise between the high power and low power operating regimes at R = 6.8 m is chosen as the reference, and B_T ~ 7.4 T at the plasma

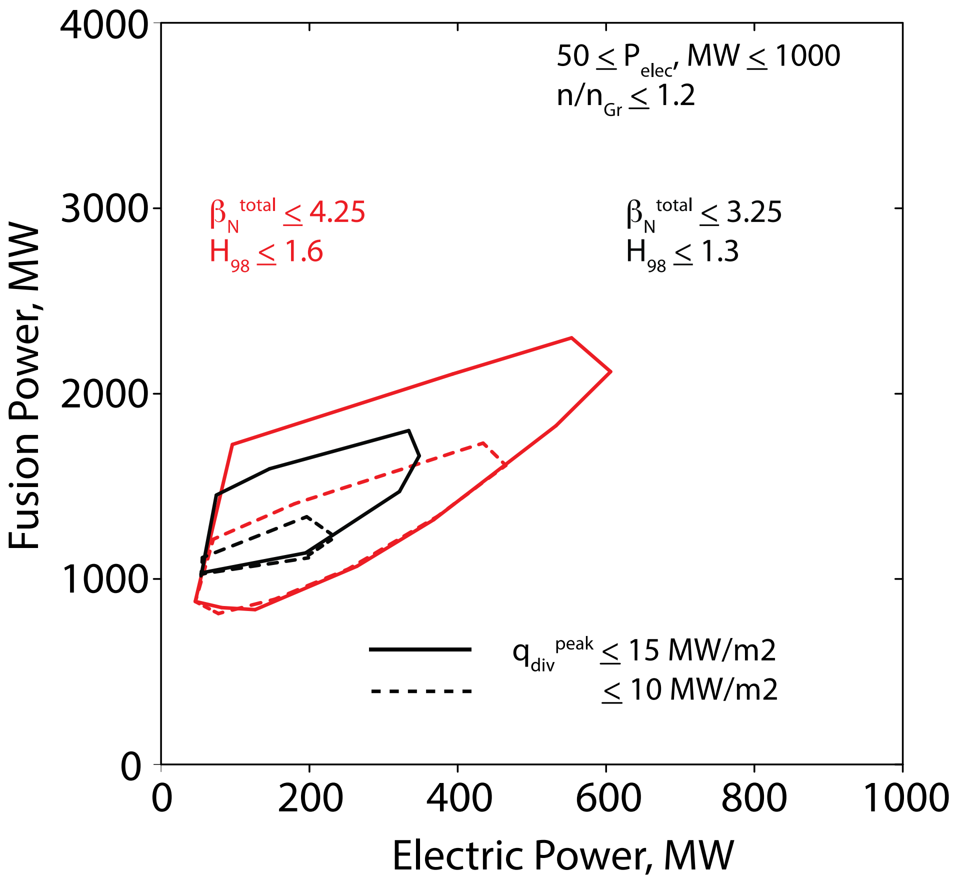


Operating Spaces, Restricting q_{div}^{peak}

- Stage I operated at lower P_{elec} and β_N, with lower plasma energy confinement reaches up to 350 Mw_e

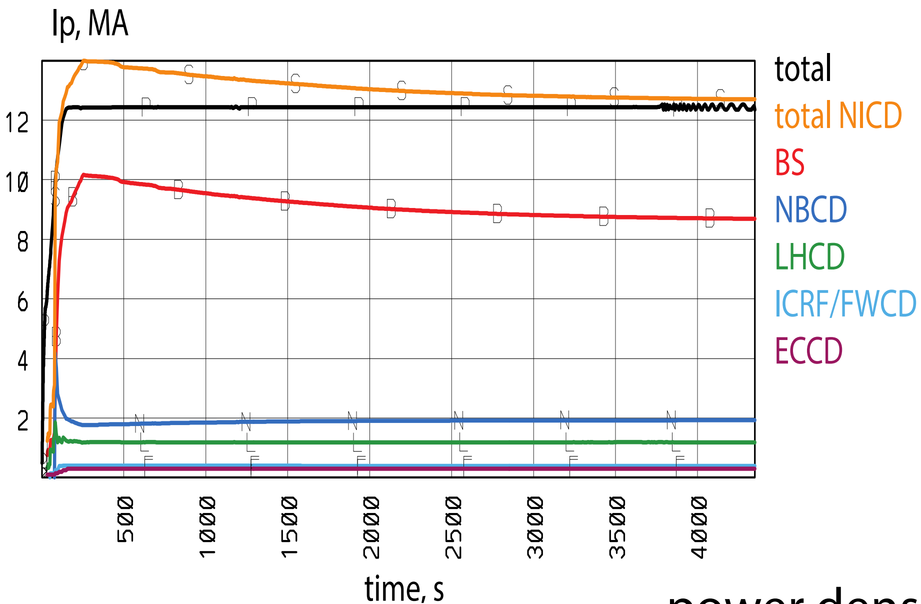
- Stage II operated at higher P_{elec} and β_N, with higher plasma energy confinement reaches up to 600 Mw_e

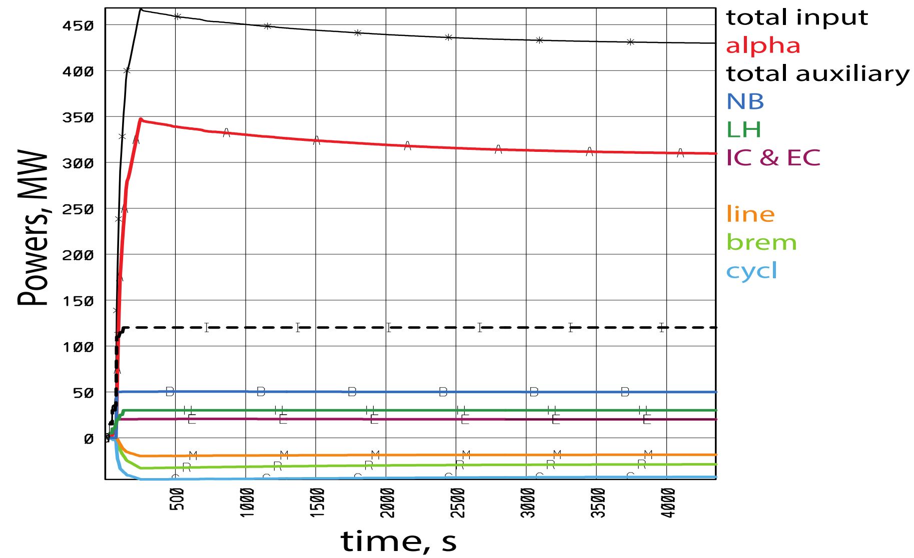
- Restricting $q_{div}^{peak} \leq 10$ MW/m^2 shrinks accessible operating space

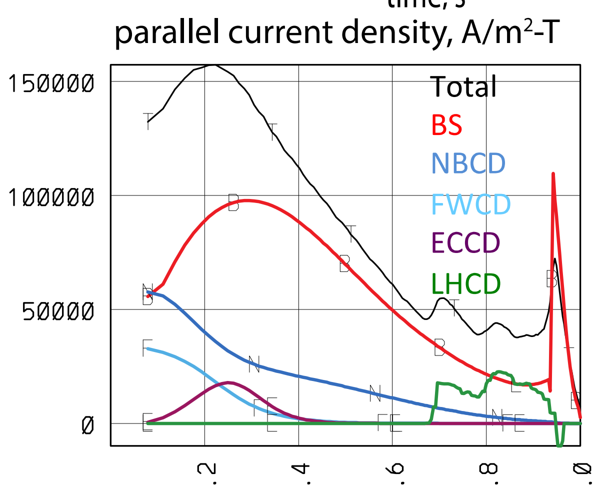


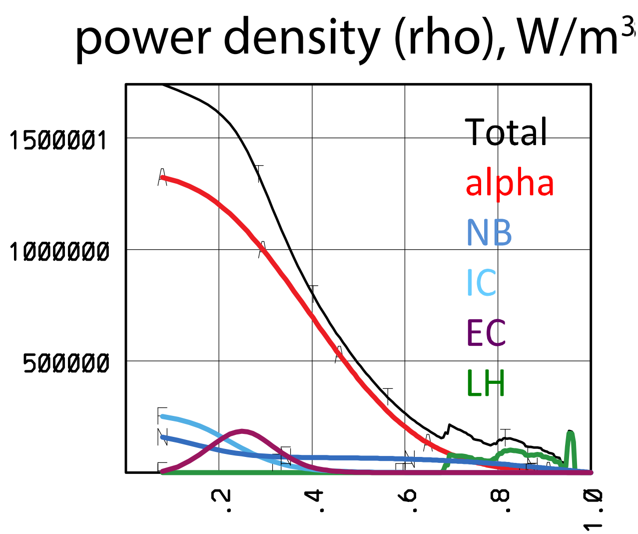
Heating and Current Drive

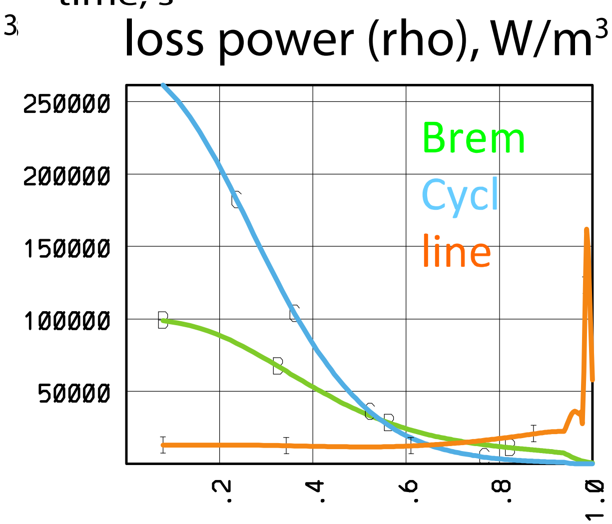
- TSC, TRANSP and other codes used (P_{NB} ~50, P_{LH} ~30, P_{IH}~20, P_{EC} ~20)







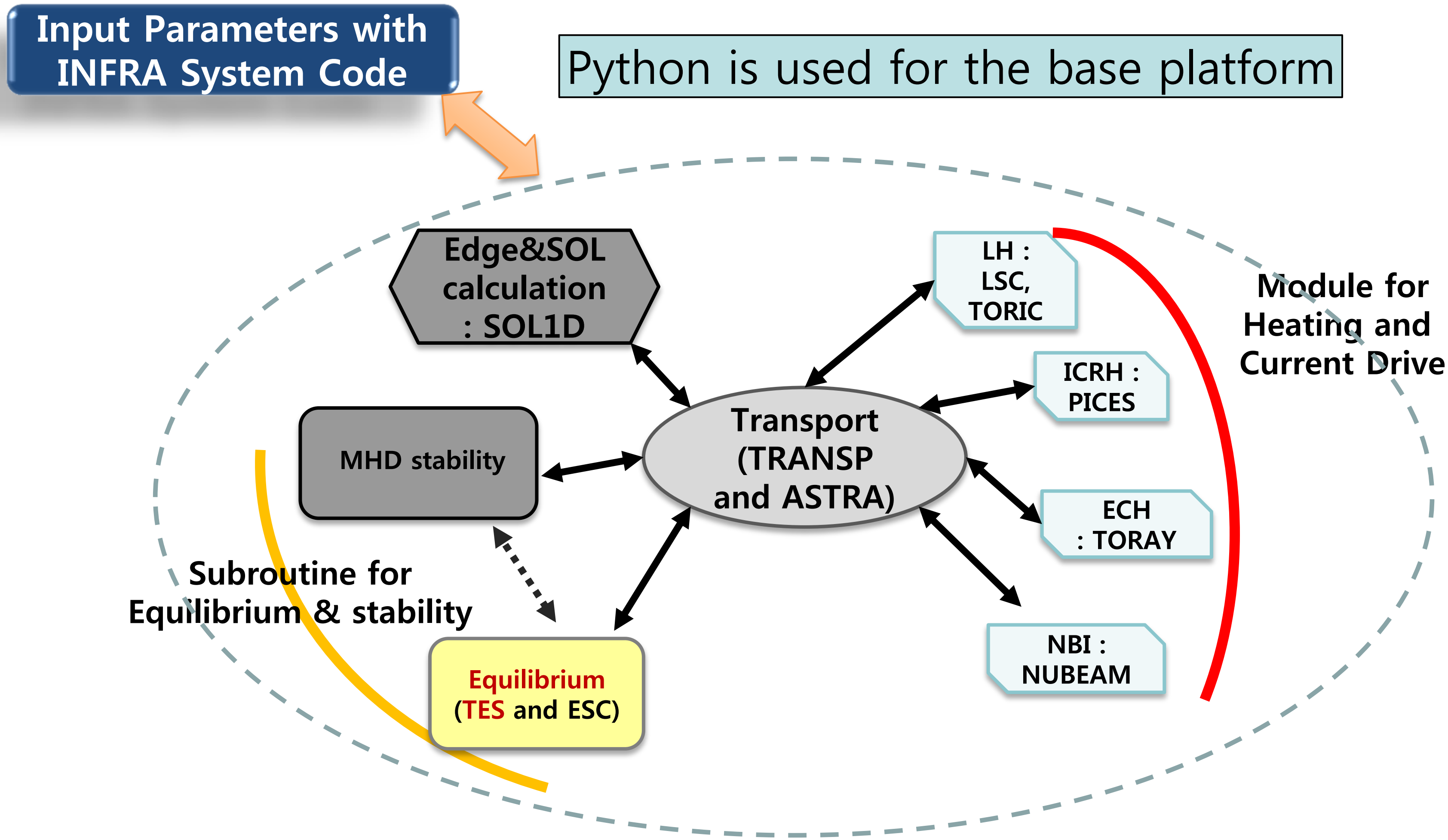




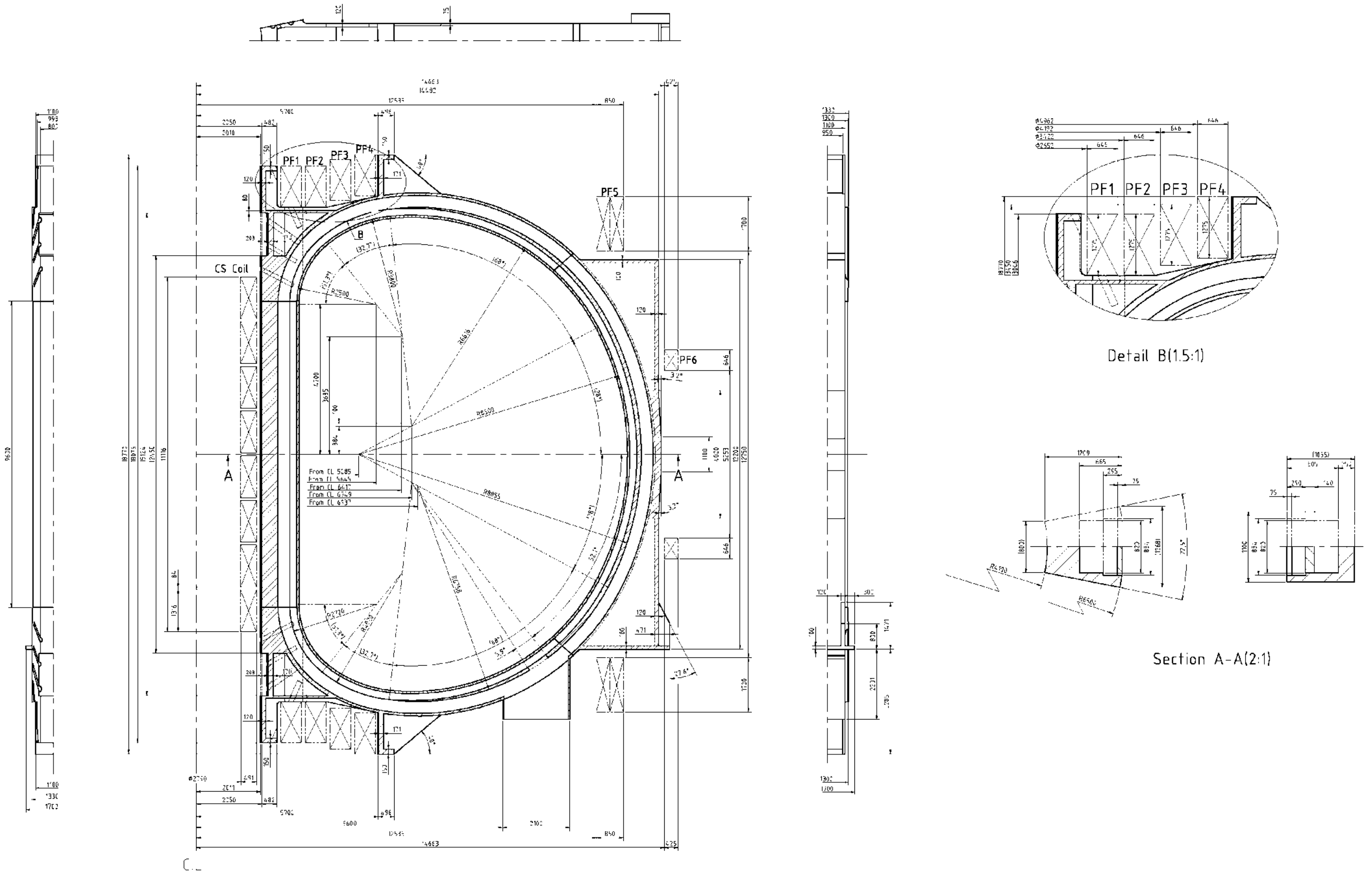
Development of Tokamak Simulator, INFRA

■ Structure of the INFRA Code

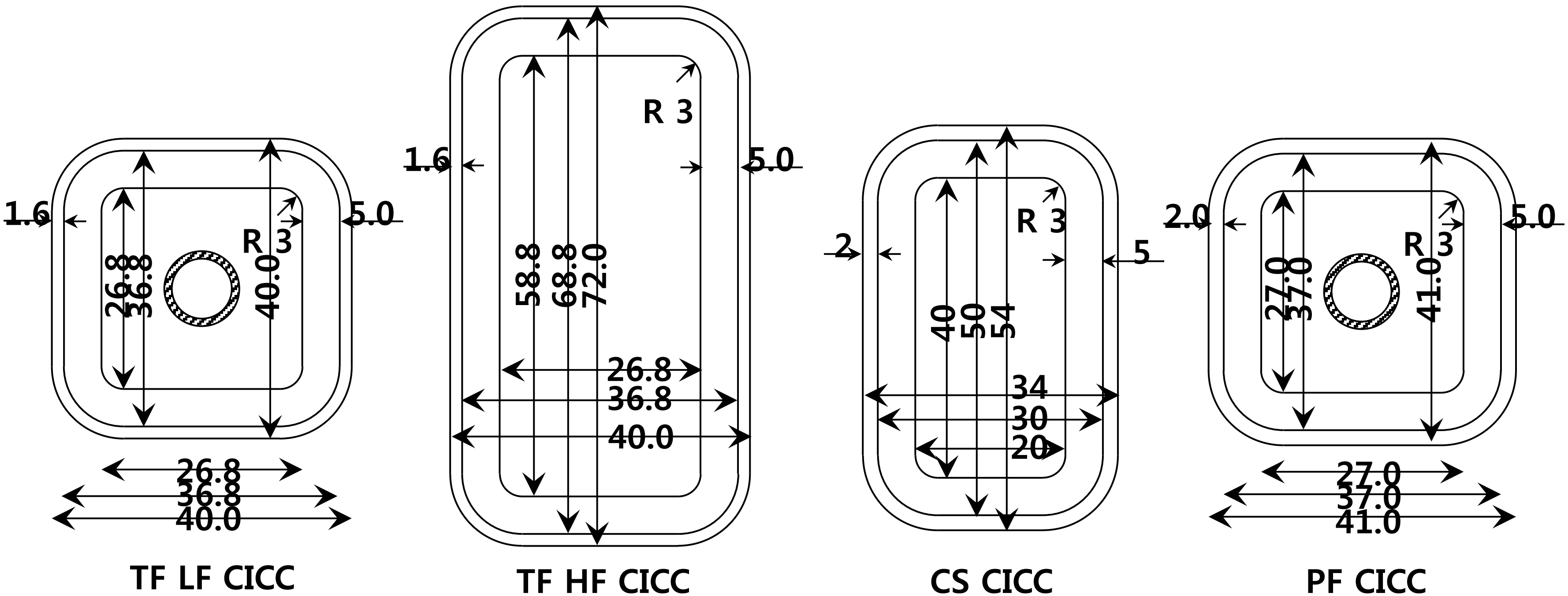
(INFRA : INtegrated Fusion Reactor Analysis)

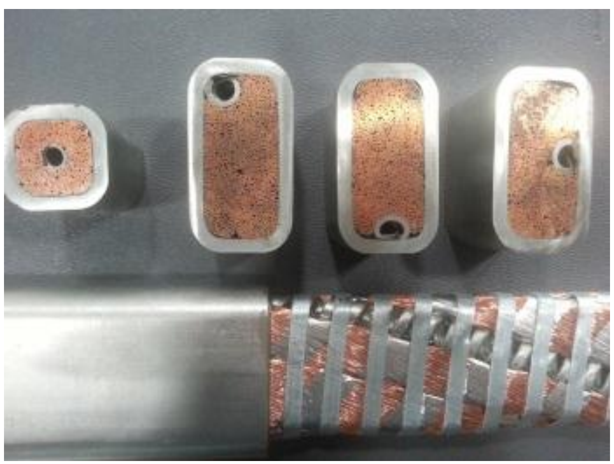


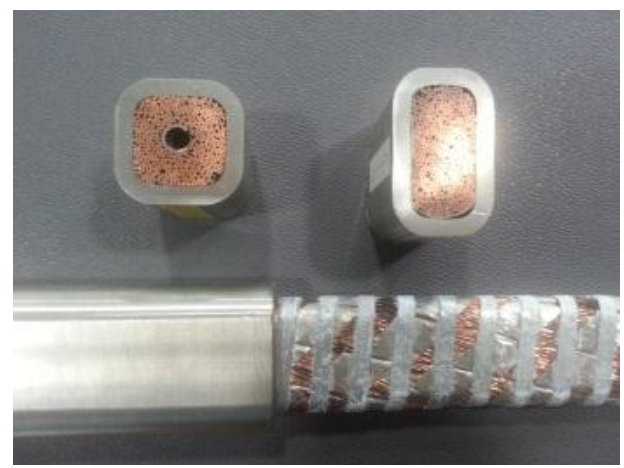
2D Drawing of Magnet System



CICC Dimensions and Trial Fabrication





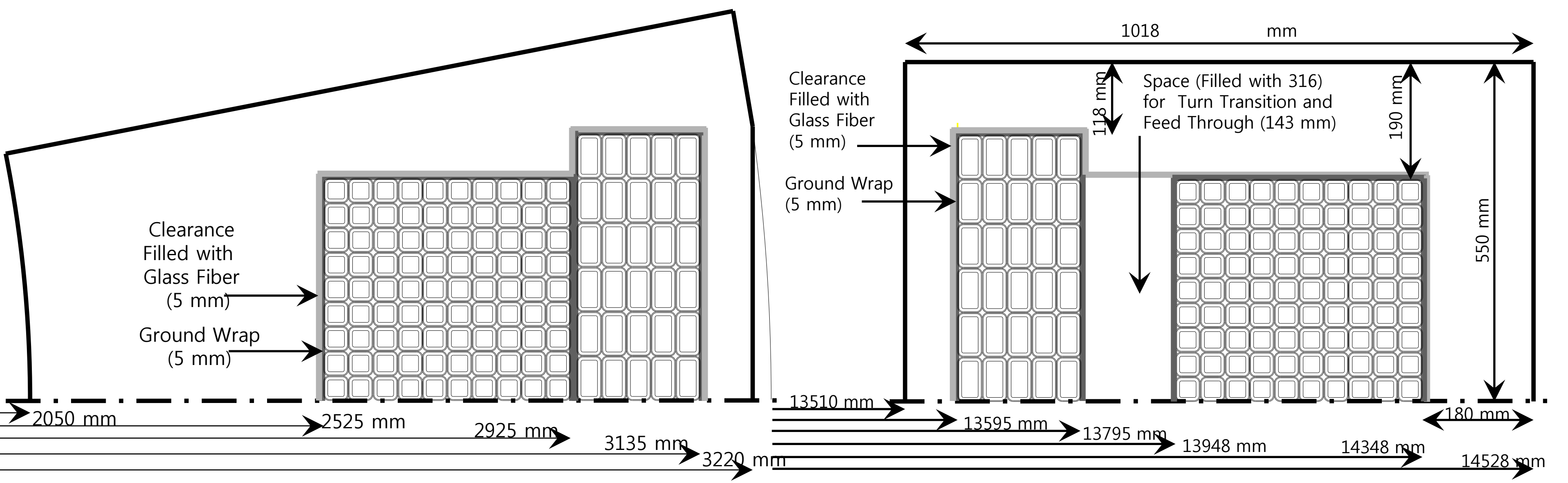


CICC Parameter

Parameter	TF HF	TF LF	CS	PF1-4	PF5-6
■ Cable pattern No. of SC strands No. of Copper strands Spiral Dimension (mm)	(3SC)x4x5x6x5 + Helical Spiral 1800 - ID 7 / OD 12	(((2SC+2Cu)x5)x6+7 Cu)x6 + Central Spiral 360 432 ID 7 / OD 9	(2SC+1Cu) x3x4x4x6 No Cooling Spiral 576 288 -	(2SC+1Cu)x3x4x4x5+Central Spiral 480 240 ID 7 / OD 9	
■ Void Fraction (%)	27.1	26.0	36.6	32.5	
■ Strand Type	High Jc (> 2600 A/mm2) Nb3Sn Strand 0.82 mm diameter (~450 ton + ~280 ton)		ITER type (Jc ~ 1000 A/mm2) Nb3Sn Strand 0.82 mm diameter (~102 ton + ~90 ton)		NbTi Strand 0.82 mm diameter (~90 ton)
■ Cu/non-Cu of Strand	1.0				
■ Insulation	1.6 mm (including Voltage Tap) (0.1 mm Kapton 400% + 0.3 mm S glass 400%)		2.0 mm (including Voltage Tap) (0.1 mm Kapton 400% + 0.4 mm S glass 400%)		
■ Jacket Thickness	5.0 mm				
■ Twist Pitch (mm) 1st stage 2nd stage 3rd stage 4th stage 5th stage	 80 ± 5 140 ± 10 190 ± 10 245 ± 15 415 ± 20	 80 ± 5 140 ± 10 190 ± 10 300 ± 15 -	 27 ± 5 45 ± 10 85 ± 10 150 ± 15 385 ± 20	 35 ± 5 75 ± 10 135 ± 10 285 ± 15 410 ± 20	
■ Wrapping Tape Sub-cable wrap thickness Sub-cable wrap width Cable wrap thickness Final wrap width	 0.08 mm, 40% coverage 15 mm 0.4 mm, 60% coverage 7 mm				

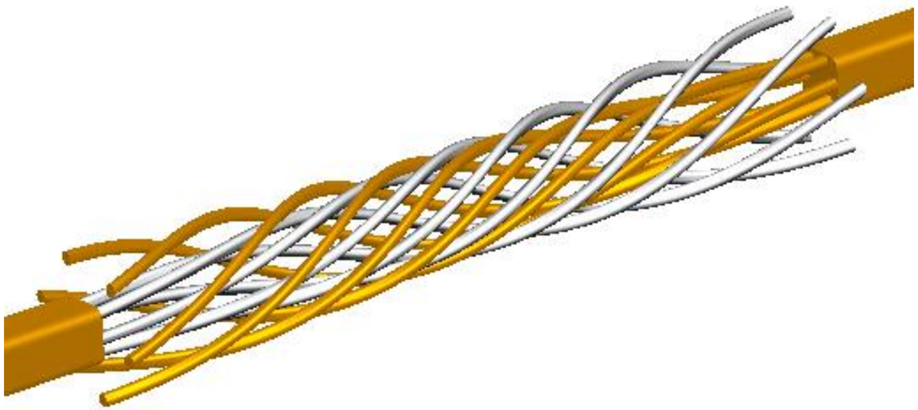
Overview of TF Coil

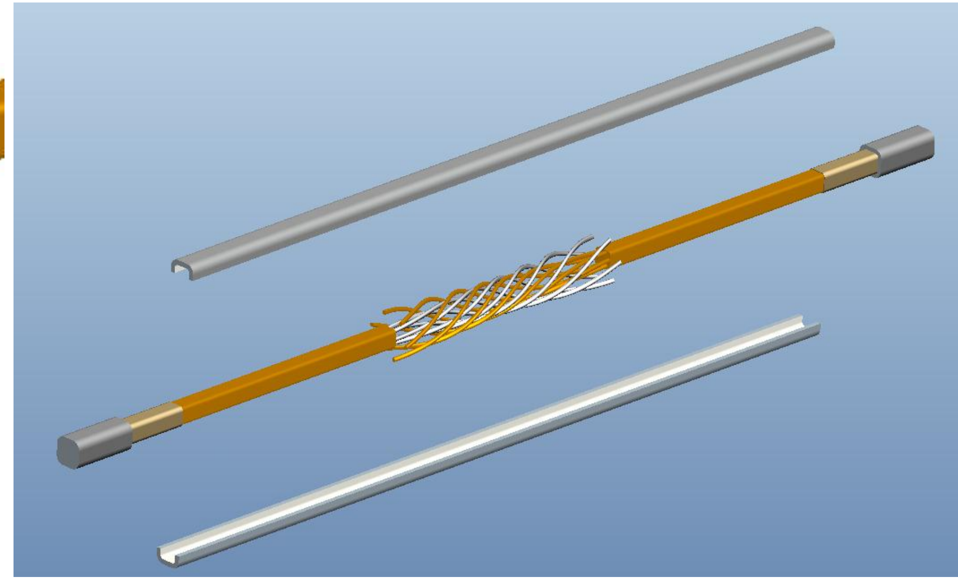
- **<u>Selected for Detailed Study (Maintenance Space = 2.5 m)</u>**
- **<u>Considering Vertical Maintenance Scheme</u>**
- R = 6.8 m, a = 2.1 m
- Small CICC Coil : 18 x 10 turns Large CICC Coil : 12 x 5 turns (Total : 240 turns)
- Magnetic Field at Plasma Center : ~7.4 Tesla (Bpeak ~ 16 Tesla, T-margin > 1 K)
- Nominal Current : 65.52 kA
- Conductor Length :
  - LQP = ~900 m (Quadruple Pancake) (Total : ~450 ton)
  - SDP = ~930 m (Double Pancake) (Total : ~280 ton)

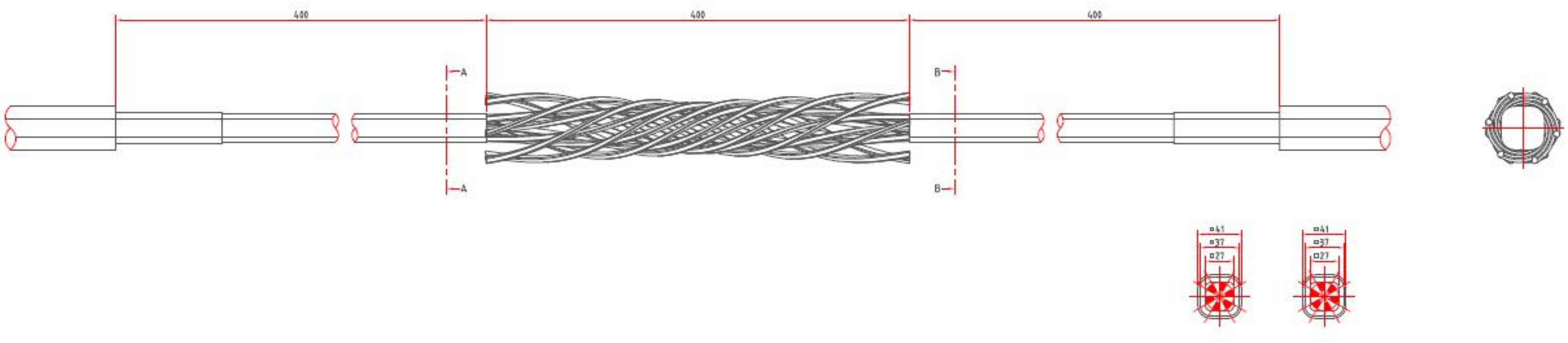


Inter-coil Joint Scheme of Magnet

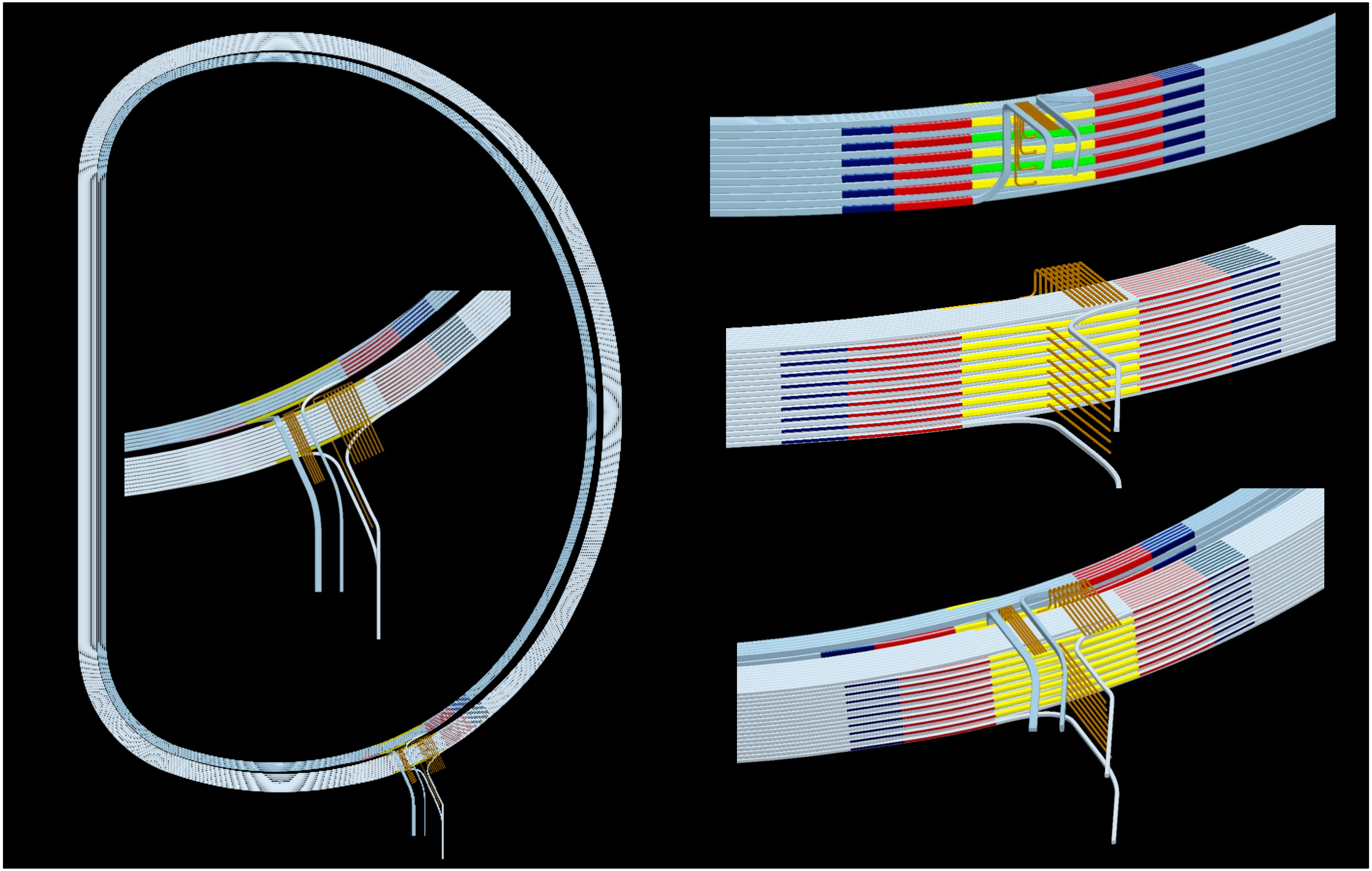
- ITER CS Inter-coil Joint Scheme used
  - Joint Resistance ~0.2 n-ohm/joint



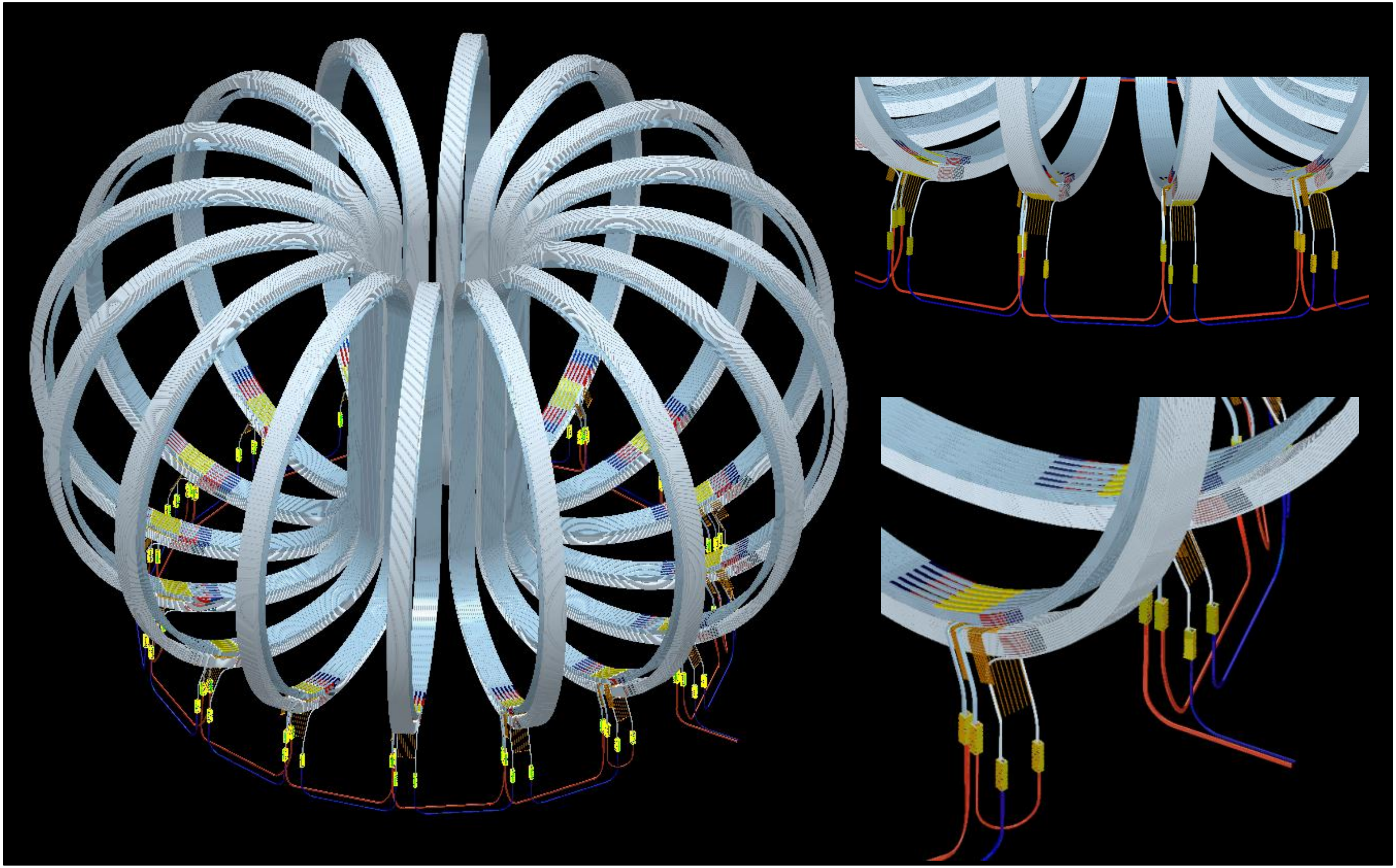




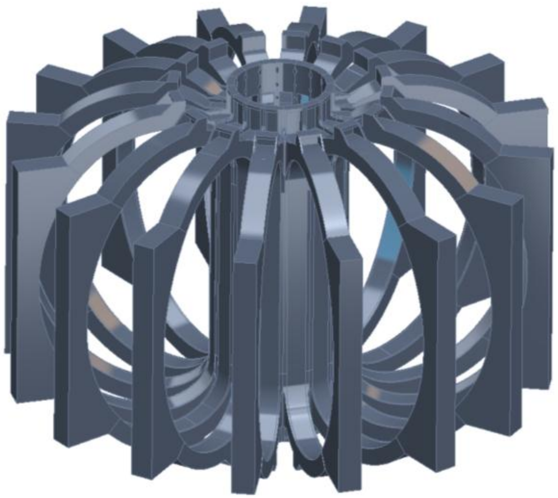
3D Modeling of TF Magnet

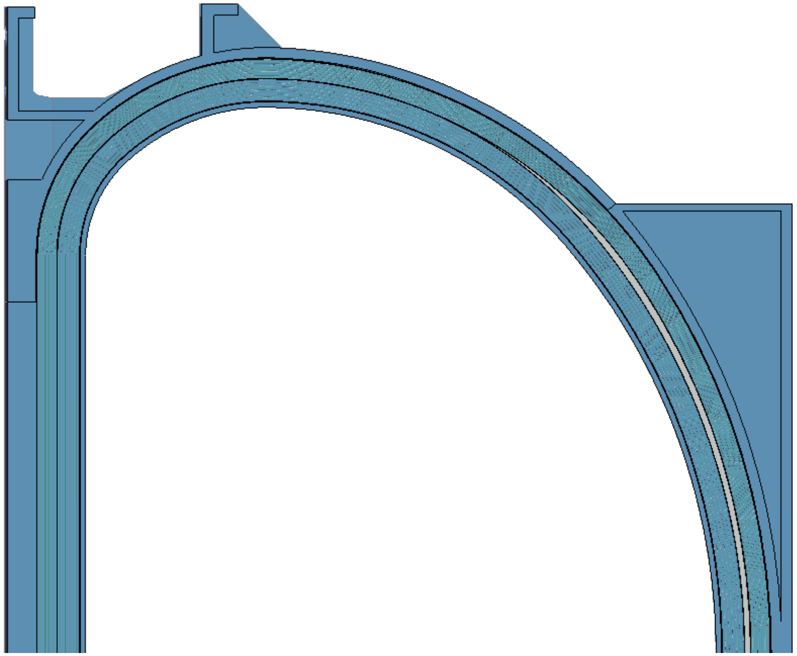


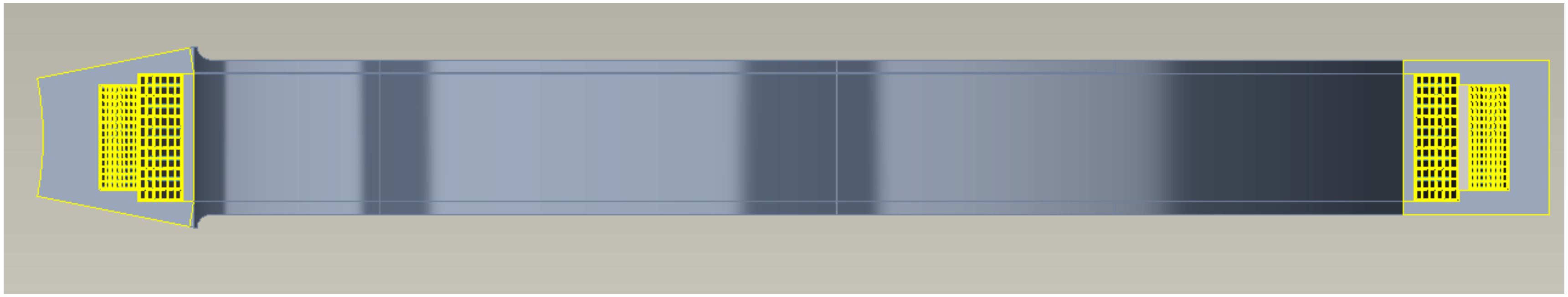
3D Modeling of TF Assembly



TF Coil Structure

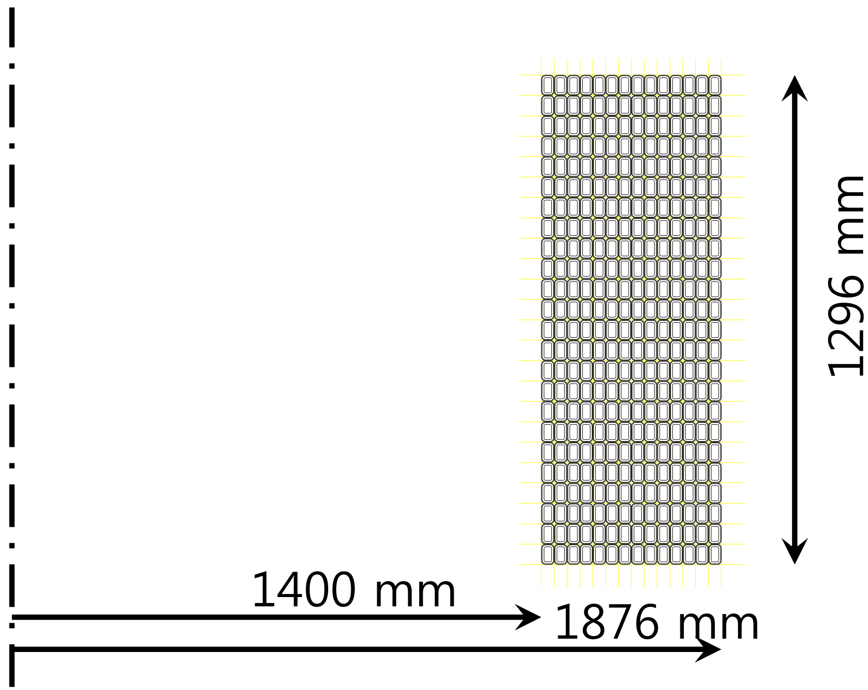


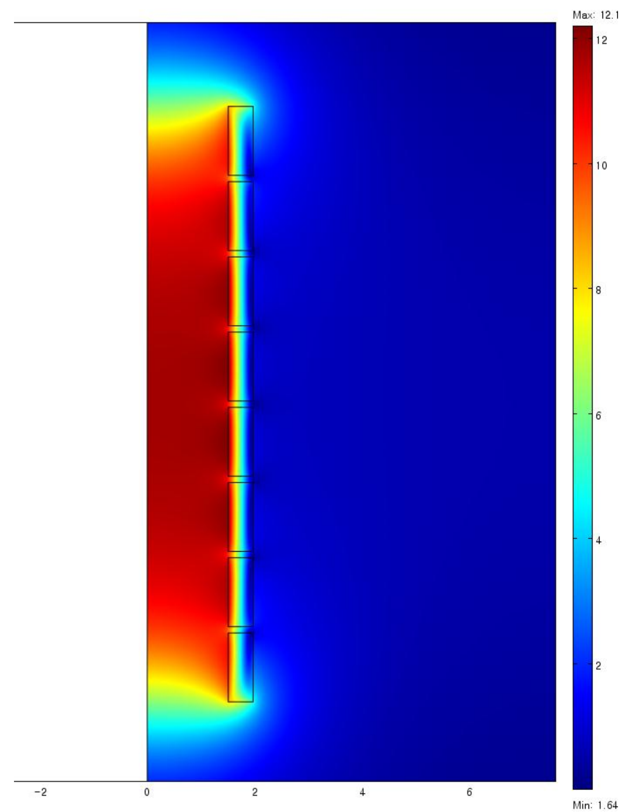




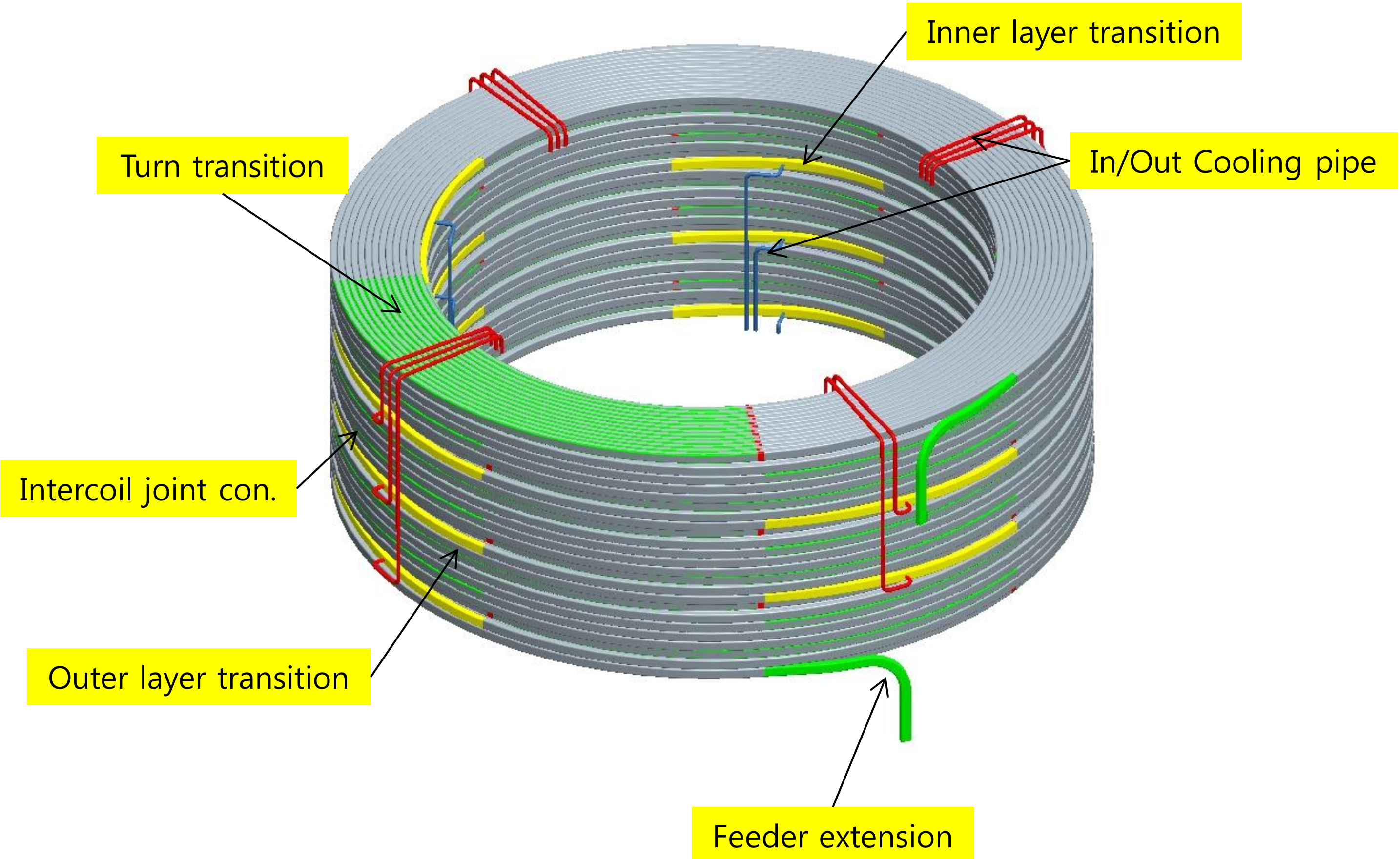
Overview of CS Coils

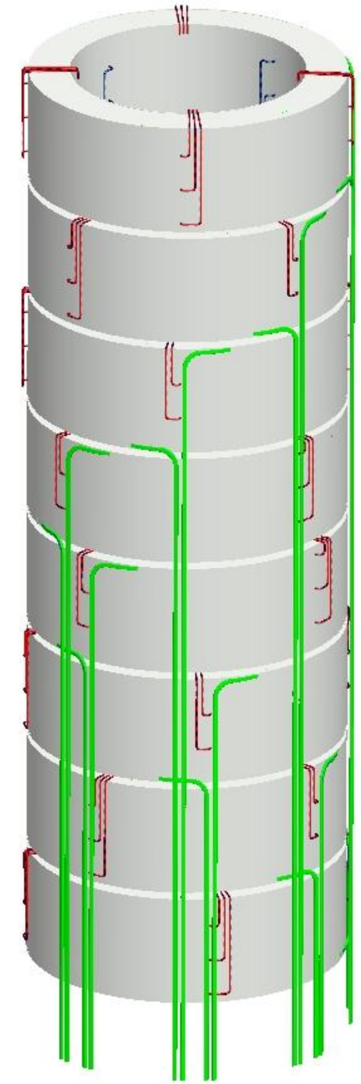
- Number of Turns : 14 (Total SC strand weight : ~102 tons)
- Number of Layers : CS1, CS2, CS3 & CS4 : 24 layers
- Magnetic Field at Center : ~11.8 T (Bpeak < 12.194 T, Half Flux Swing ~ 85 Wb)
- Conductor Unit Length : 885 m (CS1, CS2, CS3 & CS4 : UL x 4)
- Gap Between Coils : 104 mm
- Magnet Center Position : (1638, 700), (1638, 2100), (1638, 3500), (1638, 4900)
- Nominal Current : 42 kA (Current can be increased)
- Temperature Margin ~ 1.3 K





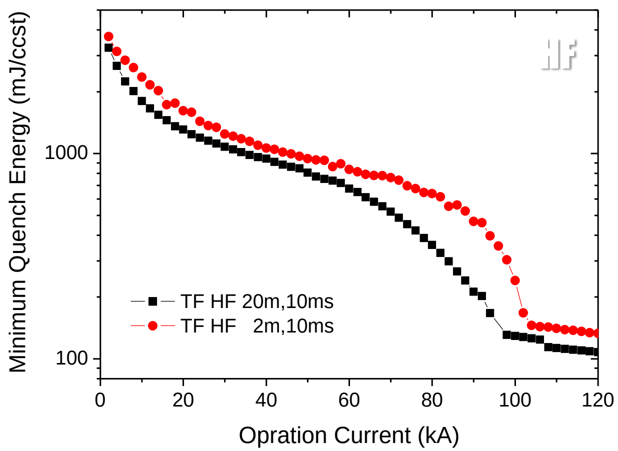
3D Modeling of CS Coils

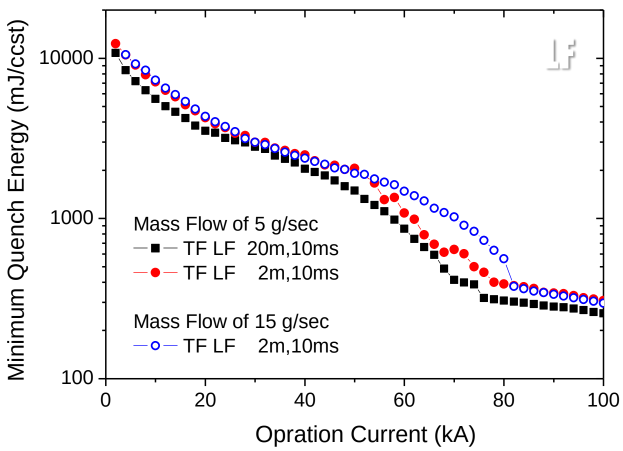


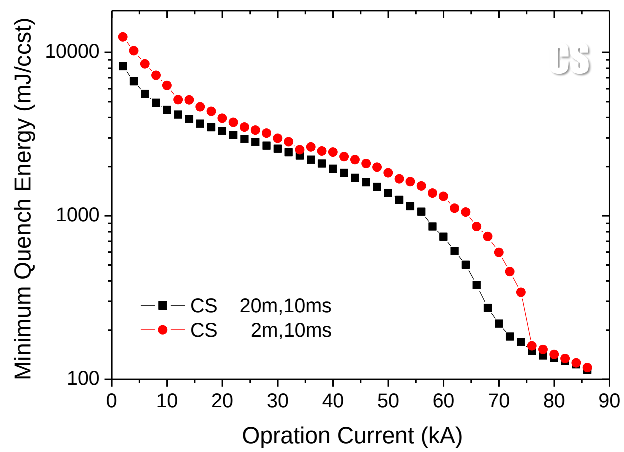


Stability Analysis of TF and CS CICC

- Gandalf Code has been used for the estimation.
- Assumption & Result
  - Gaussian shape DC heat pulse was applied for 10 ms at the center of the CICC's.
  - The nominal strain of -0.5% was assumed for the superconducting wires.
  - The field, temperature and strain dependence of the critical current density was estimated by the scaling law based on strong-coupling theory.
  - The percentage perforation of the separation perimeter between the bundle and hole He channels was set to 0.5 and the inlet pressure of 0.5 MPa case was studied.
  - For the HF CICC, the energy margin at an operation current of 65.52 kA is well above 500 mJ/ccst whether the heating zone is 2 or 20 m, even for the stagnant flow condition.
  - But for the LF conductor, the energy margin at the operation current is above 500 mJ/ccst, when there is a He mass flow of 5 g/sec at the flow path inlet. The energy margin was increased almost twice as the He mass flow increased to 15 g/sec,

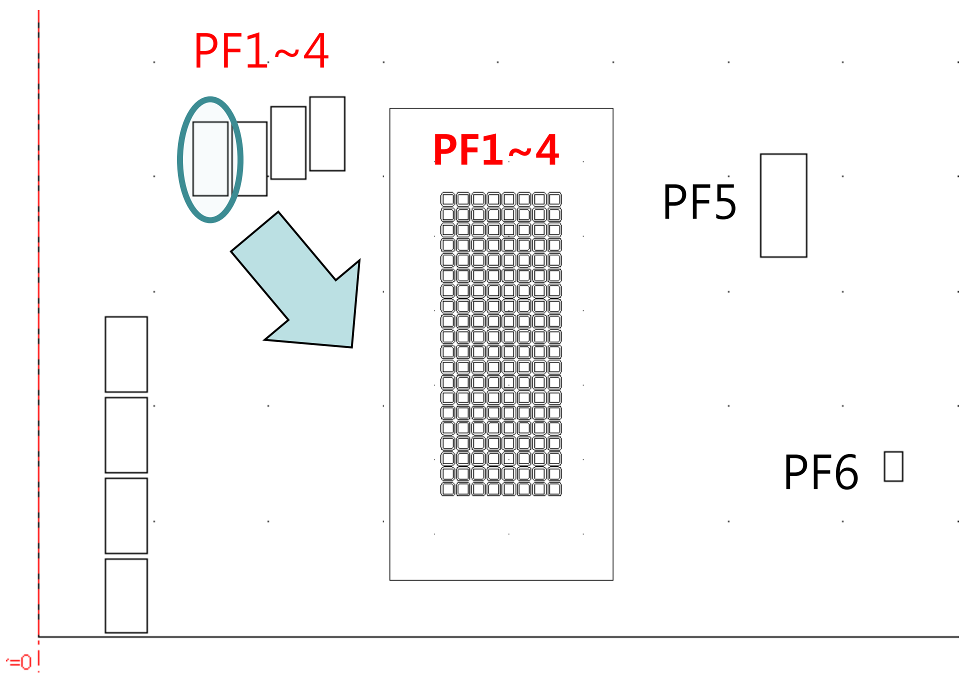


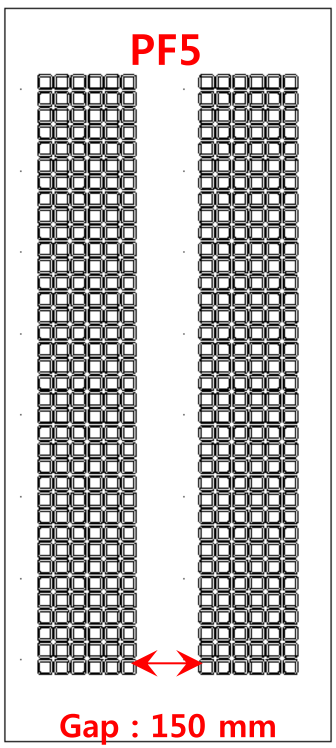


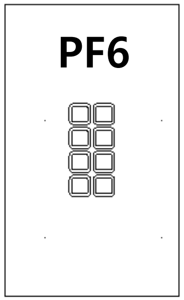


Overview of PF Coils

- Number of Turns : 8 turns for PF1~4, 12 for PF5, and 2 for PF6
- Number of Layers : 20 layers for PF1~4, 36 for PF5 and 4 for PF6
- Nominal Current : 36, 50, 50, 44, 37, 28 kA for PF1 to 6, respectively.
- Conductor Unit Length : 620, 755, 890 and 1030 m for PF1~4
  980 & 1010 m for PF5 and 770 m for PF6
- Coil Center Position : (2980, 8310), (3660, 8310), (4340, 8590) – PF1~3
  (5020, 8750), (12762 & 13158, 7500), (14880, 2950) – PF4~6
- Temperature Margin > 1.5 K

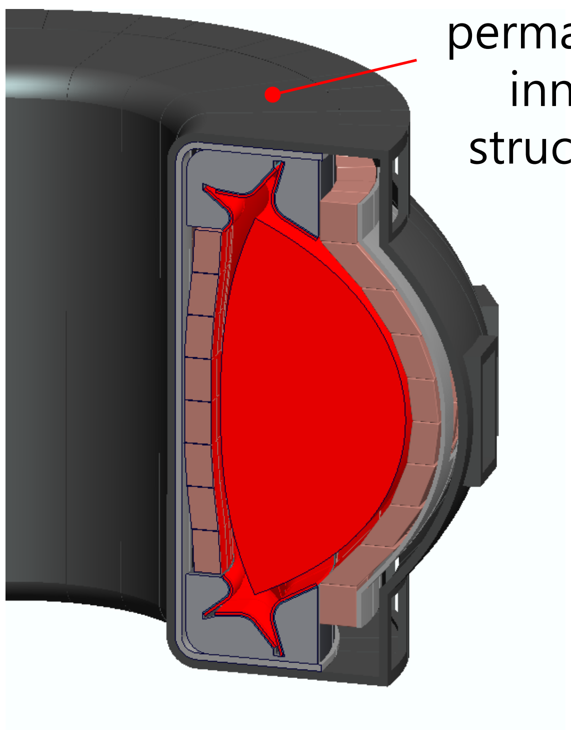




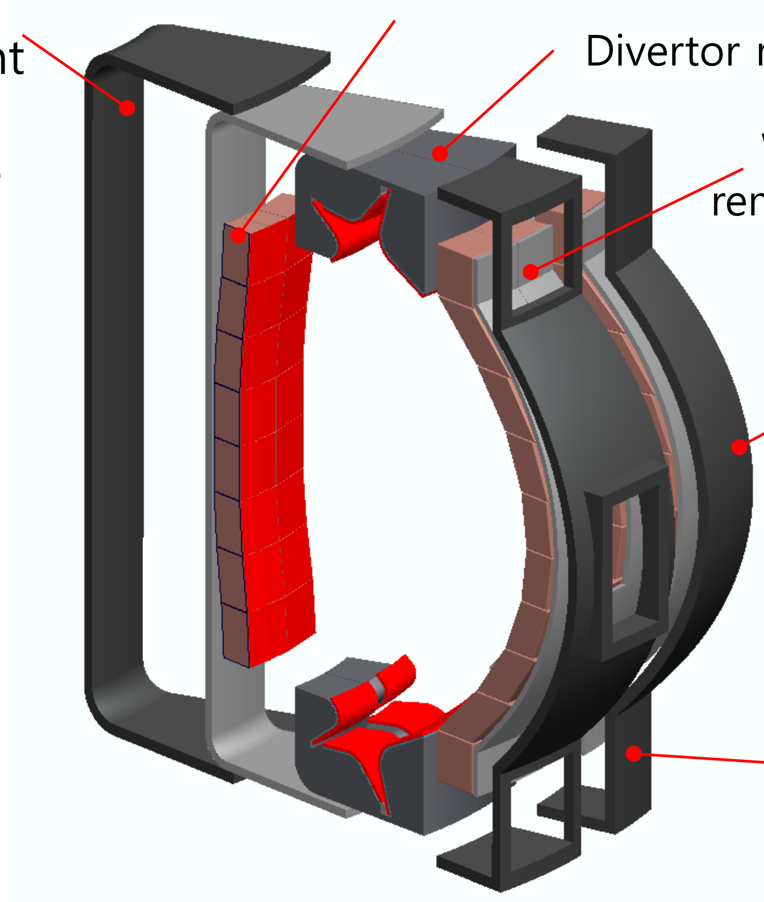


In-Vessel Components

- In-Vessel Component Segmentation 22.5°



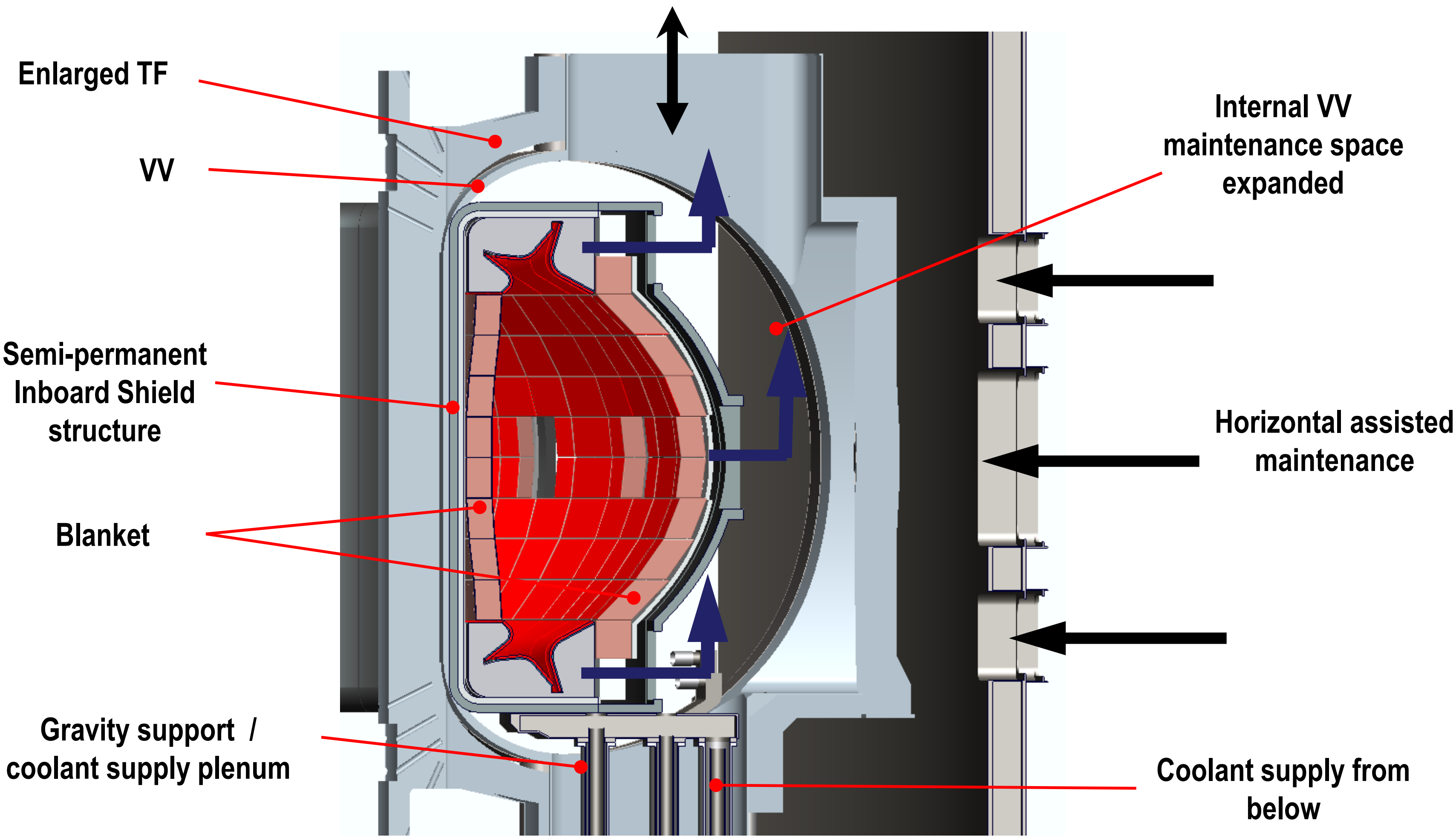
In-vessel components



In-vessel component segmentation

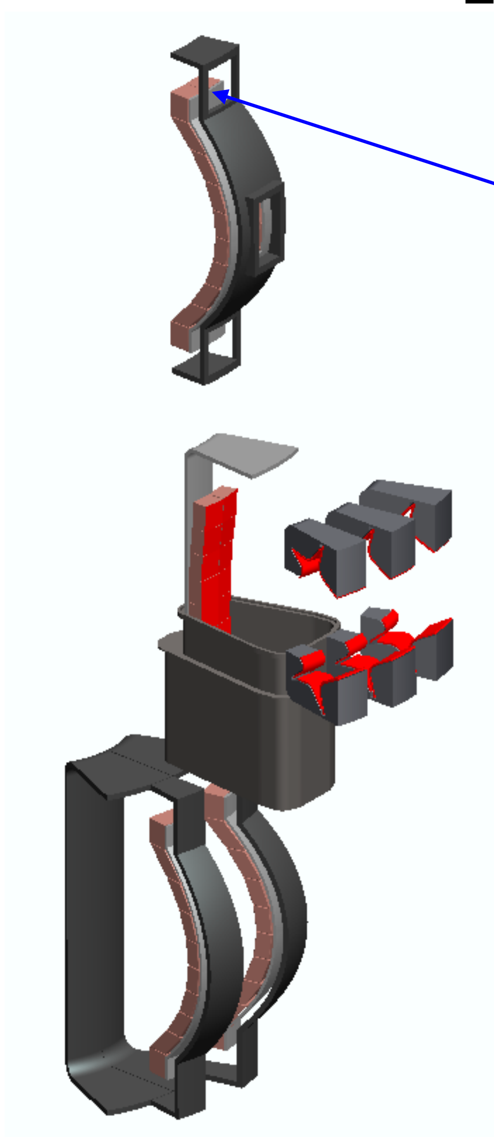
Concept of Vertical Maintenance [I]

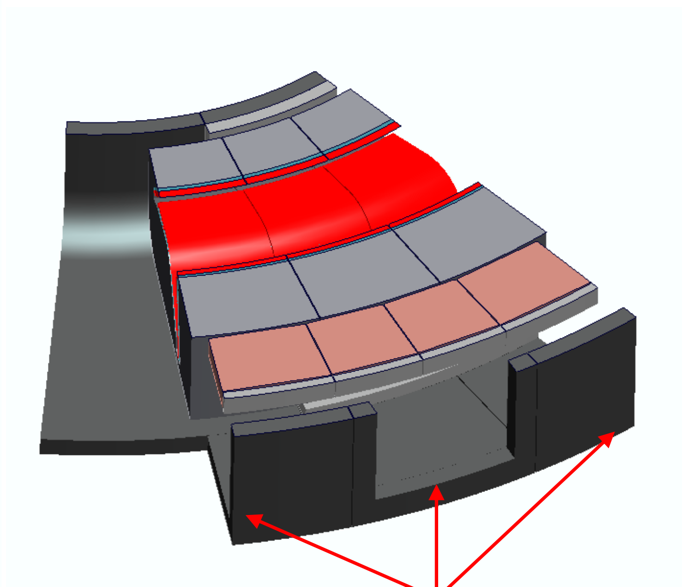
- Vertical maintenance of all in-vessel components

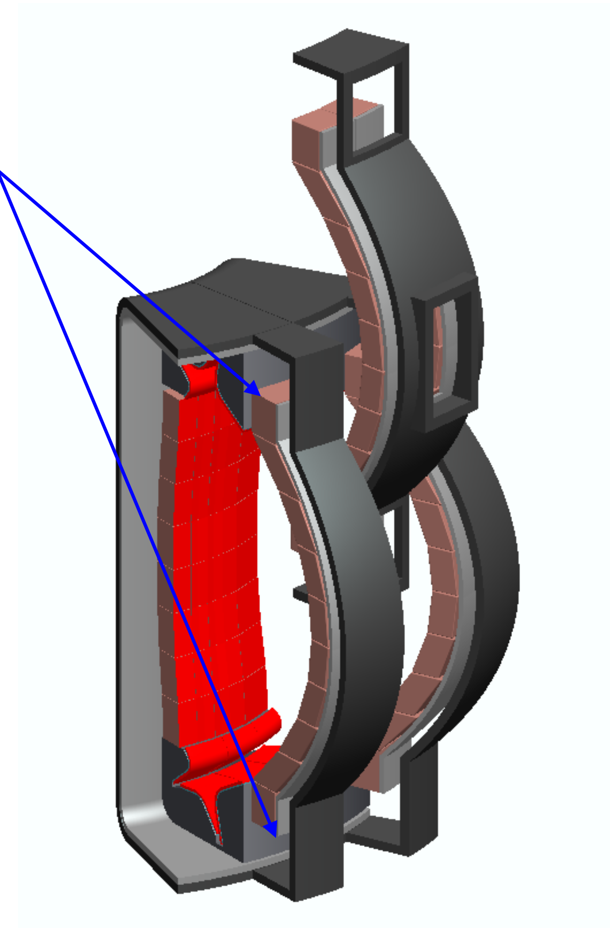


Concept of Vertical Maintenance [II]

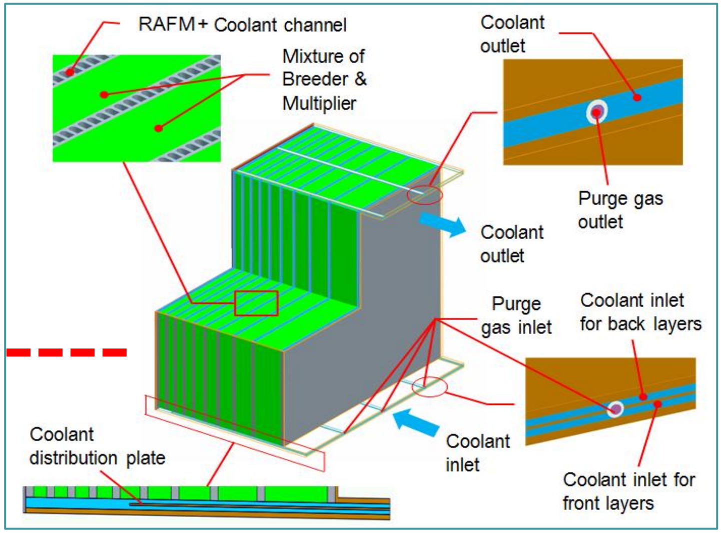
- An option to consider is to gain divertor access by removing the central outboard blanket module to gain access

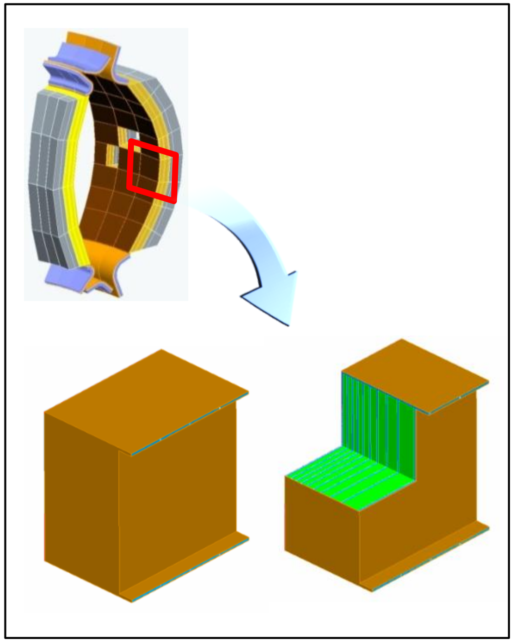




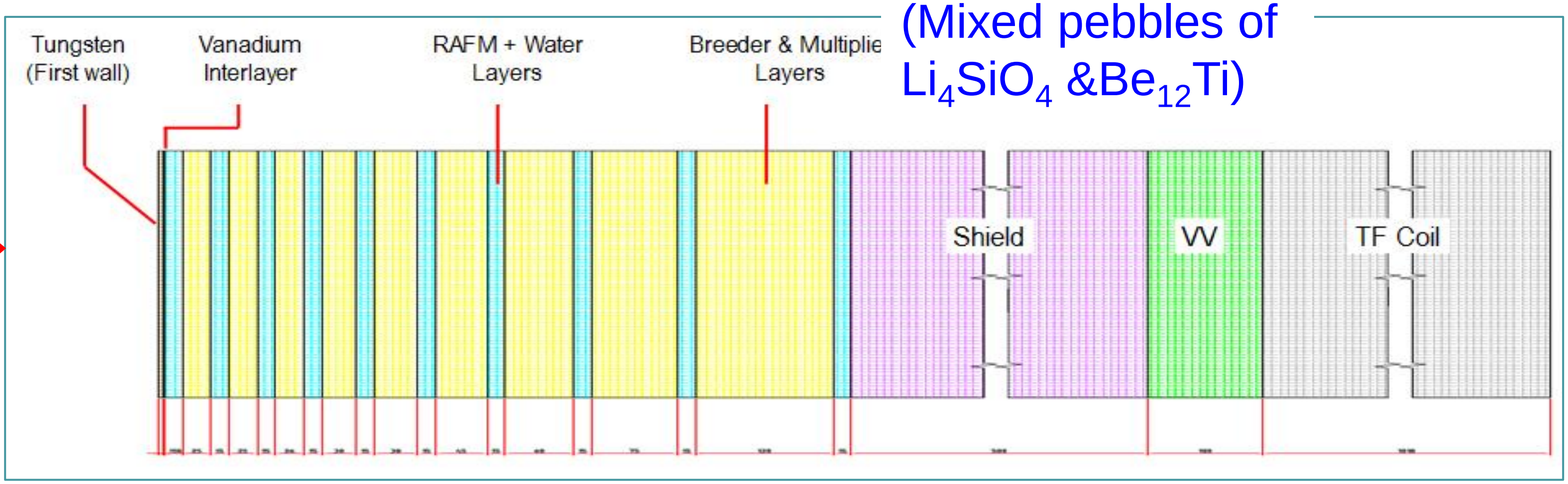


Blanket Module Configuration (water cooling)



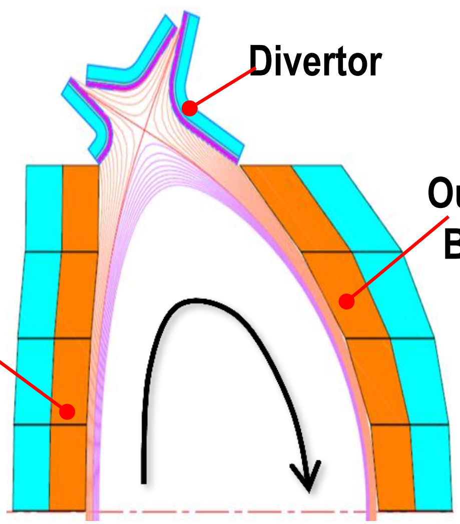


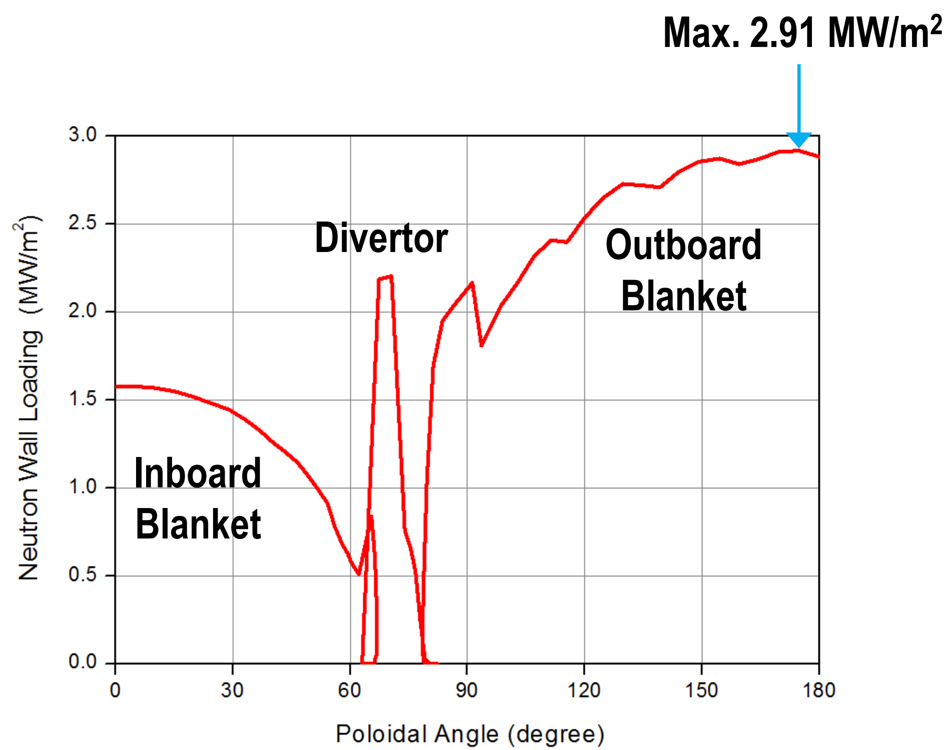
(Mixed pebbles of Li_4SiO_4 &$Be_{12}Ti$)



Neutron Wall Loading Distribution

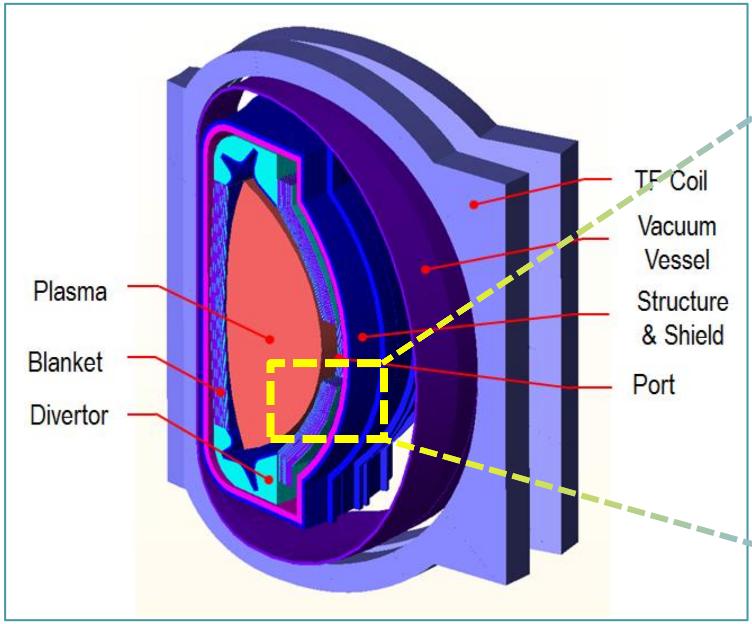
- Fusion power = 2200 MW

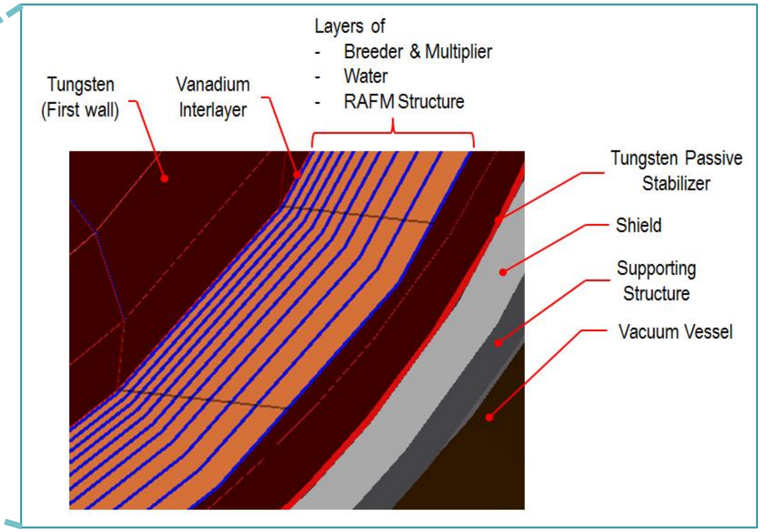




3D Global TBR (Tritium Breeding Ratio)

- MCNP model – tokamak 45° sector

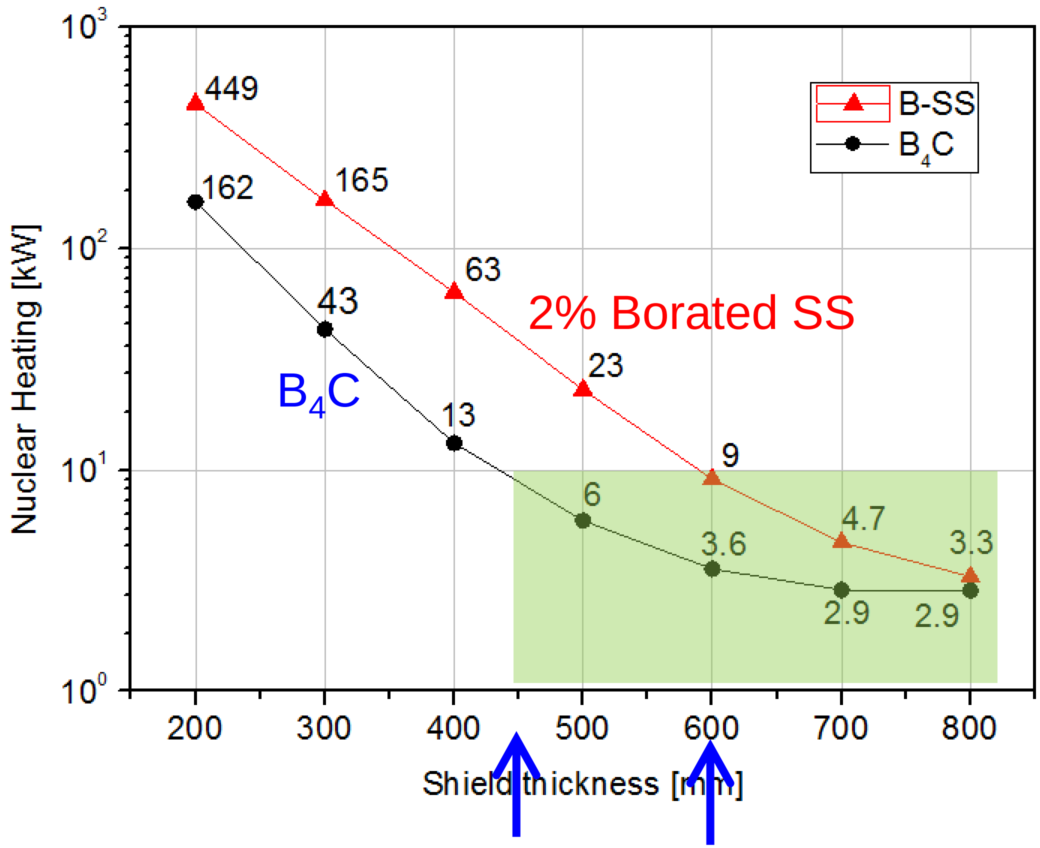




- MCNP Calculation with the help of MCAM developed by Chinese FDS team

Breeder Pebble	Multiplier Pebble	Global TBR
Li_4SiO_4	**Be**	**> 1.20**
Li_4SiO_4	**$Be_{12}Ti$**	**< 1.00 (so far)**
Li_4SiO_4 + $Be_{12}Ti$ (Mixed pebbles)		**~ 1.00 (so far)**

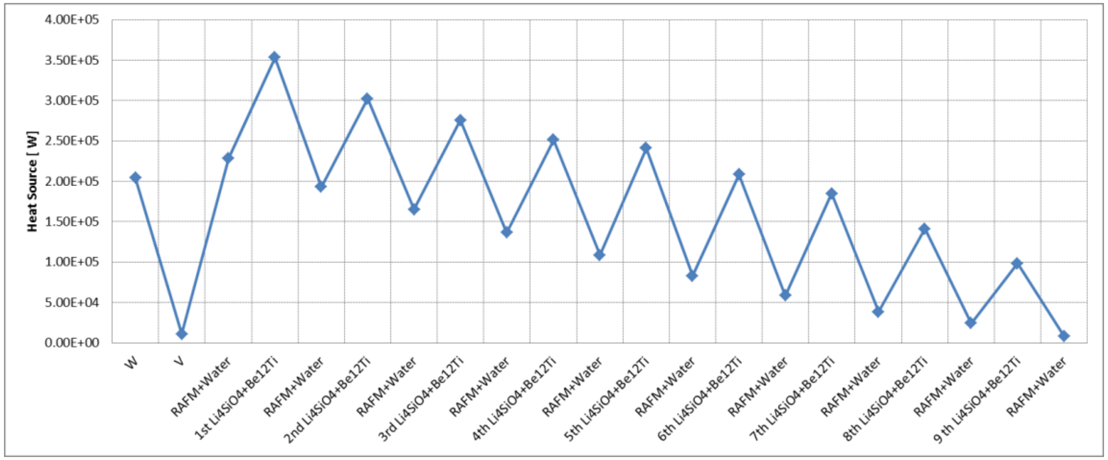
Nuclear Heating in TF Coils



- 450 mm thick B_4C or 600 mm thick 2% Borated SS could reduce the nuclear heating in 16 TF coils below 10 kW.

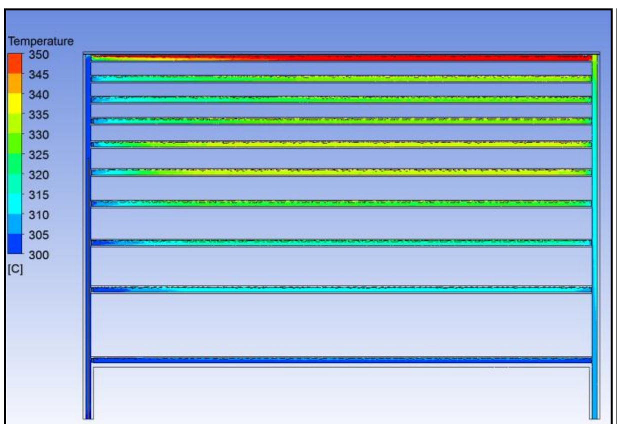
Thermo-hydraulic Analysis on Blanket Module

- Heat generation vs. Pebble Layers (9 layers)



Total Heat Generation
: 3.36 MW
(incl. water)

- Temperature distribution of blanket module (coolant speed = 4 m/s)



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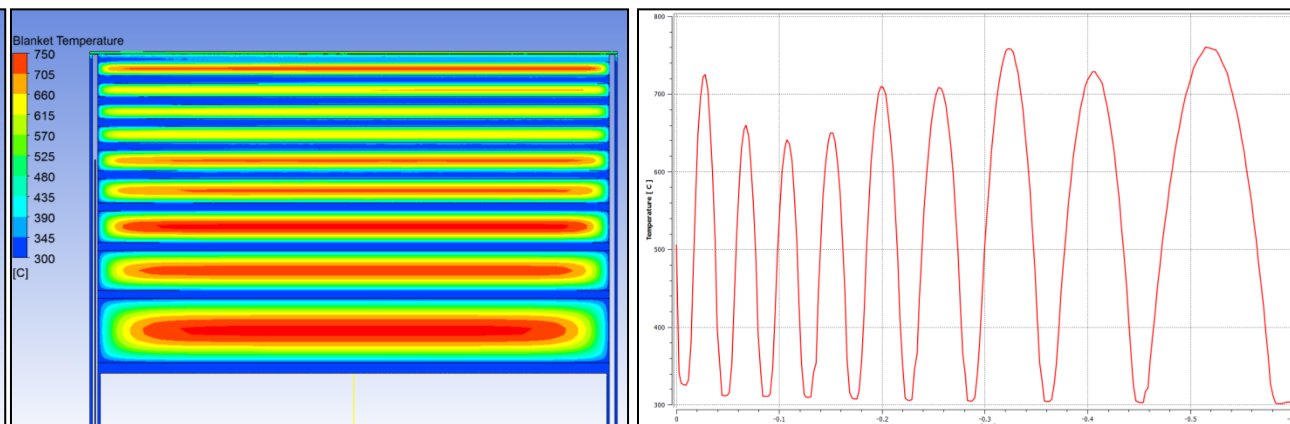

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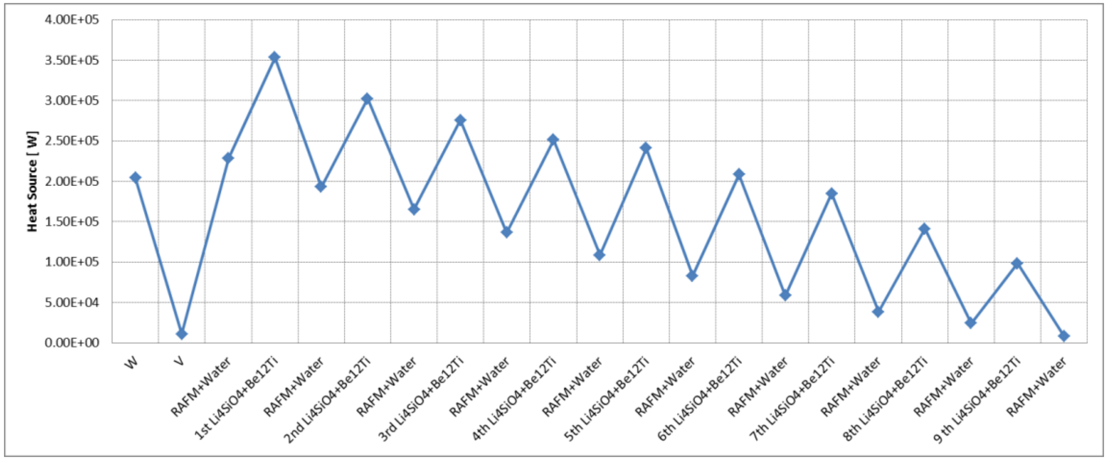
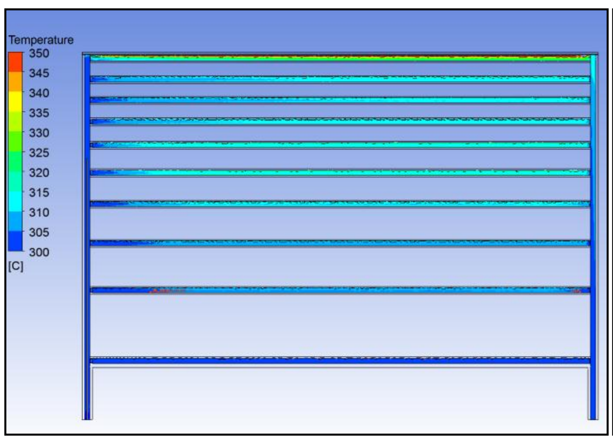
Thermo-hydraulic Analysis on Blanket Module

- Heat generation vs. Pebble Layers (9 layers)



Total Heat Generation
: 3.36 MW
(incl. water)

- Temperature distribution of blanket module (coolant speed = 10 m/s)



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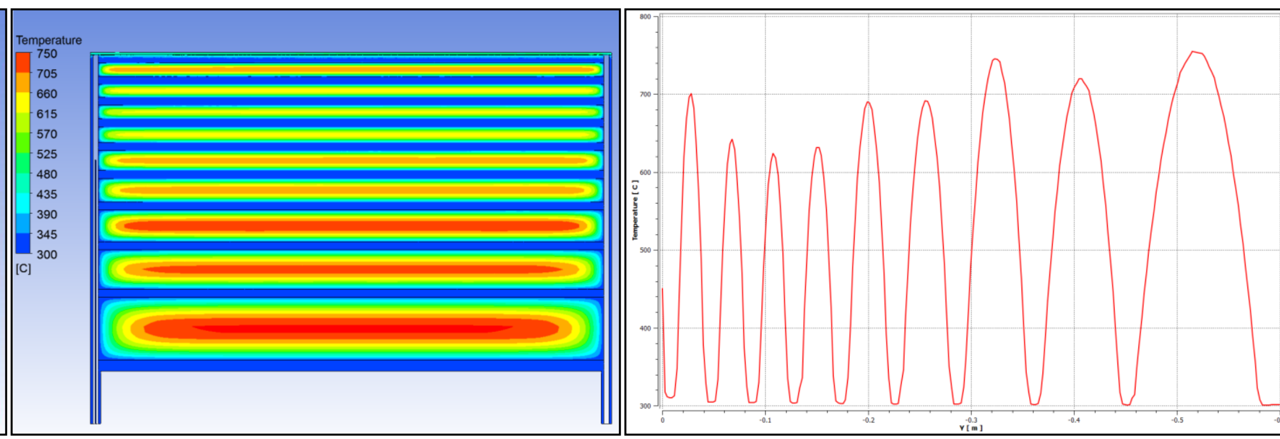

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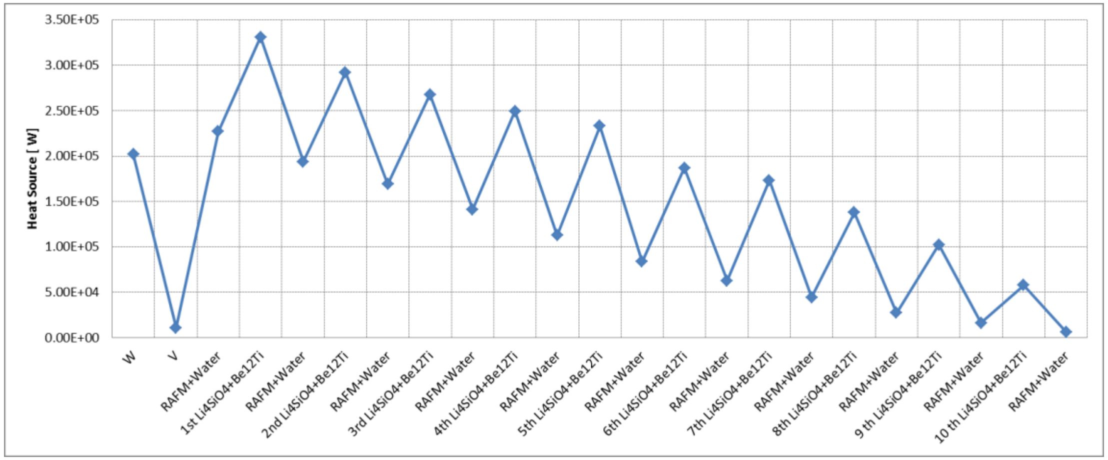
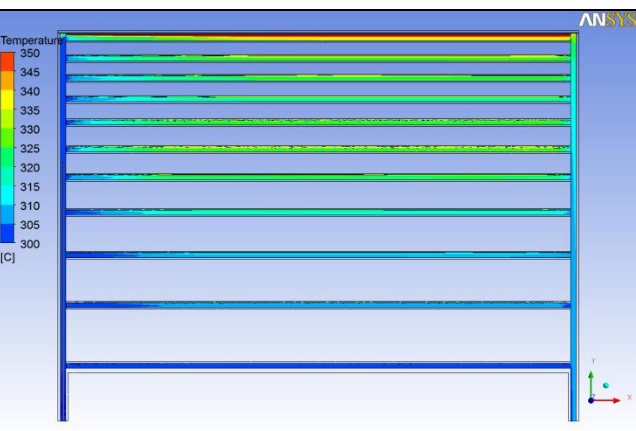
Thermo-hydraulic Analysis on Blanket Module

- Heat generation vs. Pebble Layers (10 layers)



Total Heat Generation
: 3.37 MW
(incl. water)

- Temperature distribution of blanket module (coolant speed = 4 m/s)



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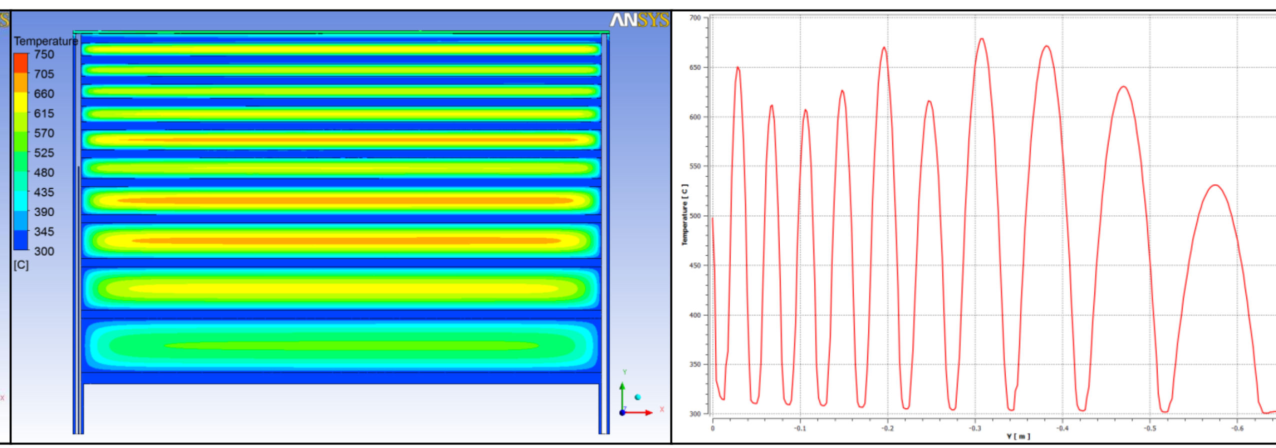

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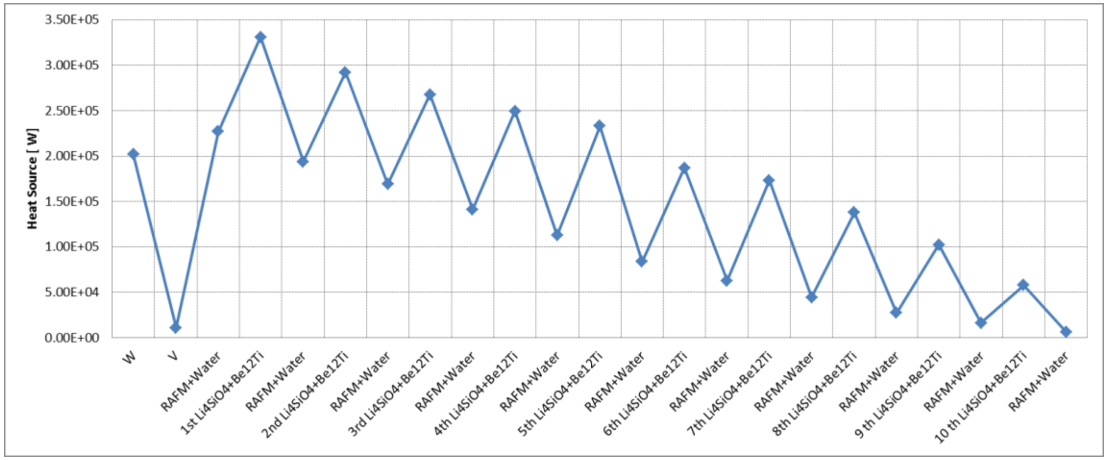
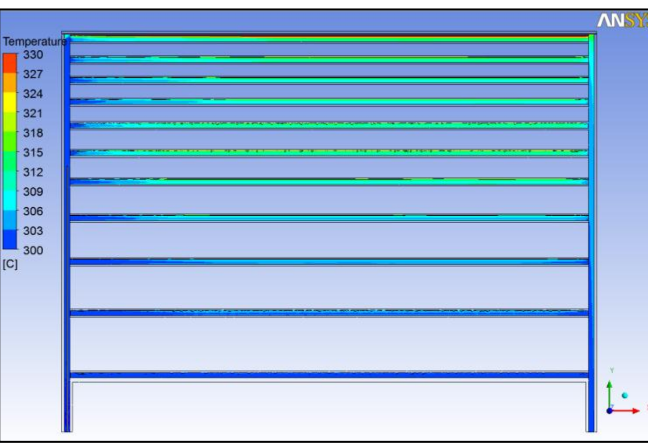
Thermo-hydraulic Analysis on Blanket Module

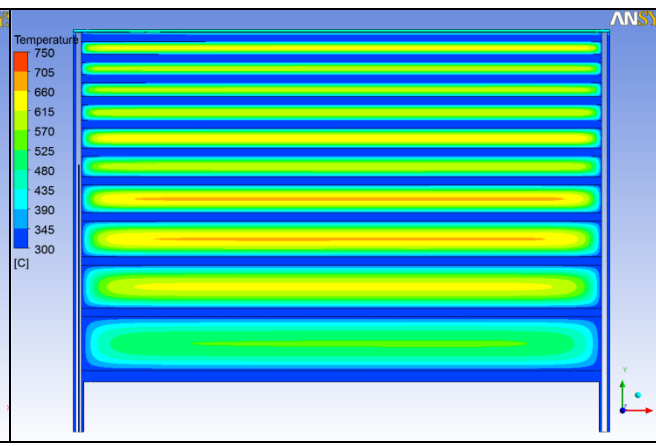
- Heat generation vs. Pebble Layers (10 layers)

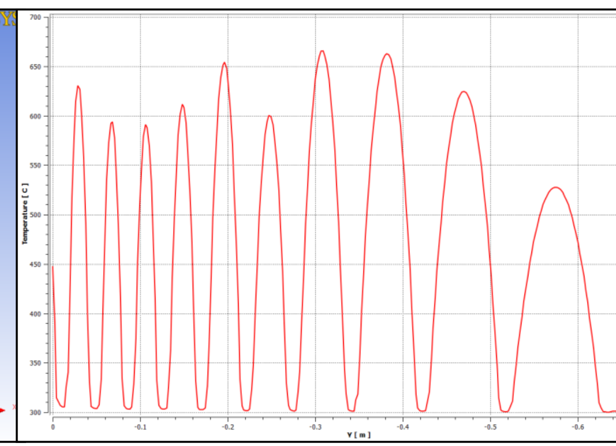


Total Heat Generation
: 3.37 MW
(incl. water)

- Temperature distribution of blanket module (coolant speed = 10 m/s)







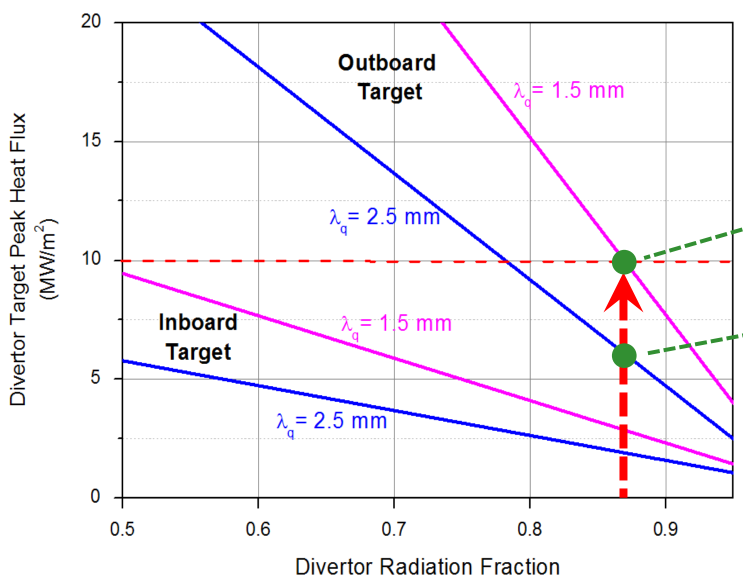
< Water Temperature >

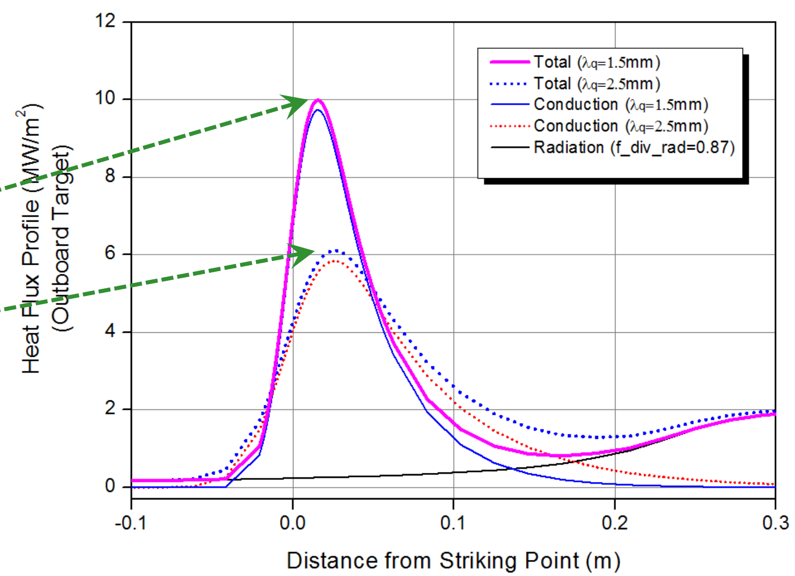
< Blanket Temperature >

Divertor Peak Heat Flux & Distibution

■ Convection + radiation heat load (for 2.2 GW of fusion power)
- Core radiation fraction = 40%
- λq = 1.5~2.5 mm
- flux expansion = 4.3

■ Divertor Radiation ~87% is required to meet Peak Heat flux = 10 MW/m^2 (λq = 1.5 mm)





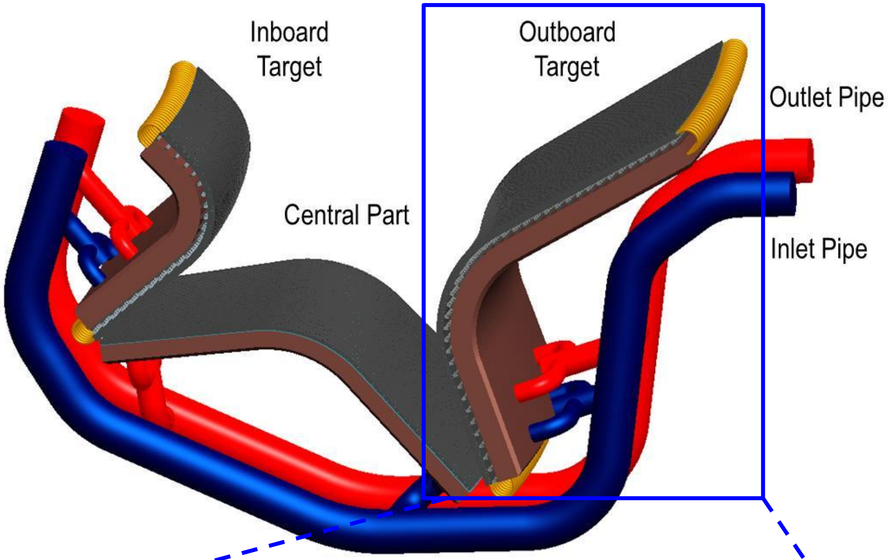
Peak Heat Flux vs. Divertor Radiation Fraction

Heat Flux Distribution at Outboard Divertor Target (f_{rad_div} = 0.87)

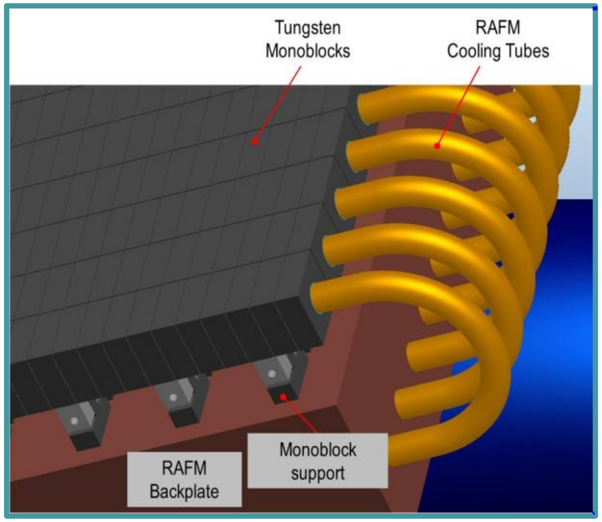
Concept of Divertor

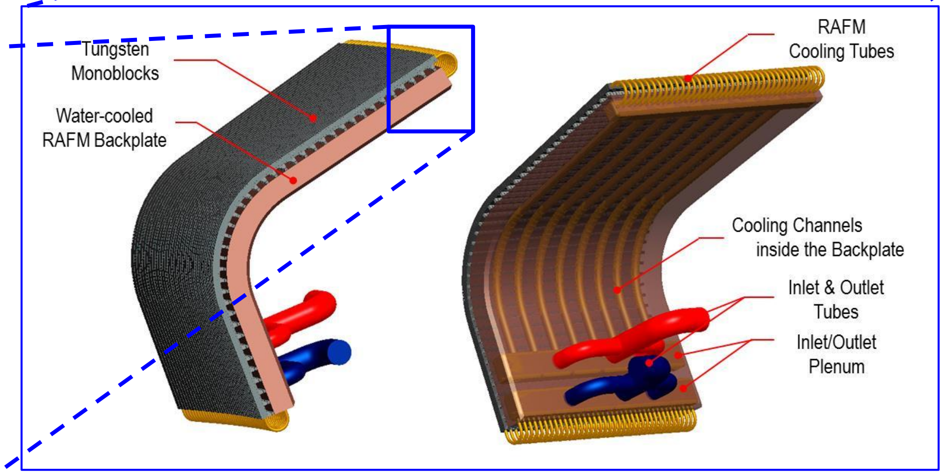
- Each divertor module is consisted of a central part, an inboard target, and an outboard target.

- High Heat Flux (HHF) unit:
  - Tungsten mono-blocks
  - RAFM cooling tube
  - Vanadium interlayer



Detail View of HHF Units

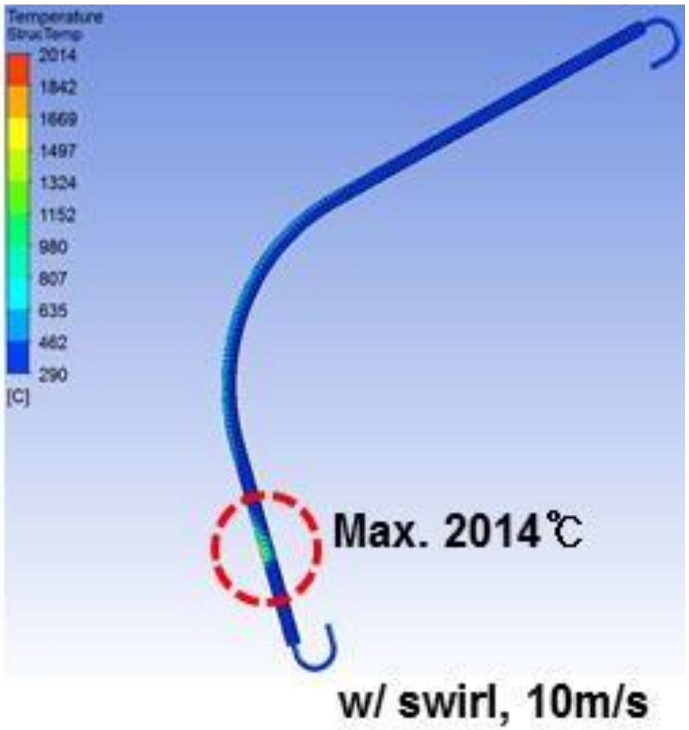


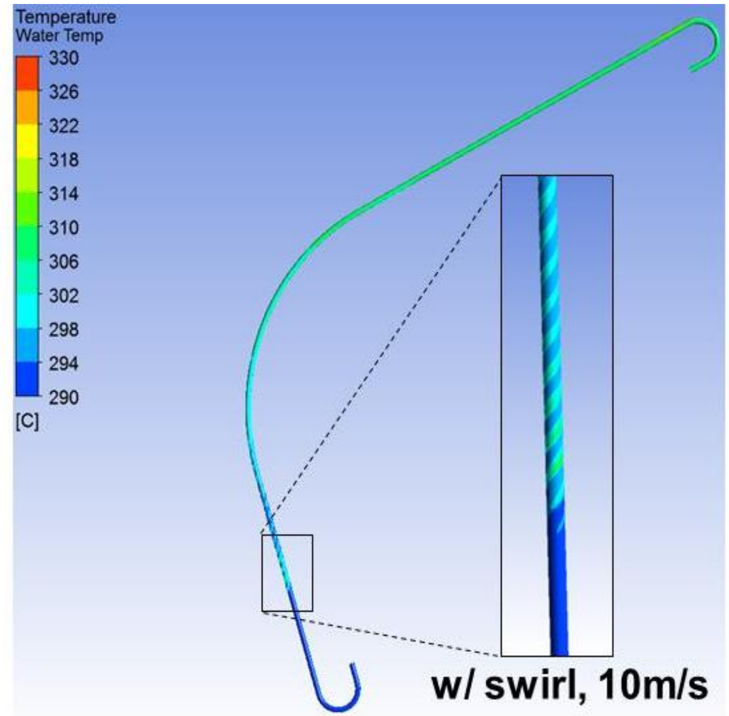


Outboard target

Preliminary Thermo-hydraulic Analyses

■ Thermo-hydraulic Analyses for the HHF unit

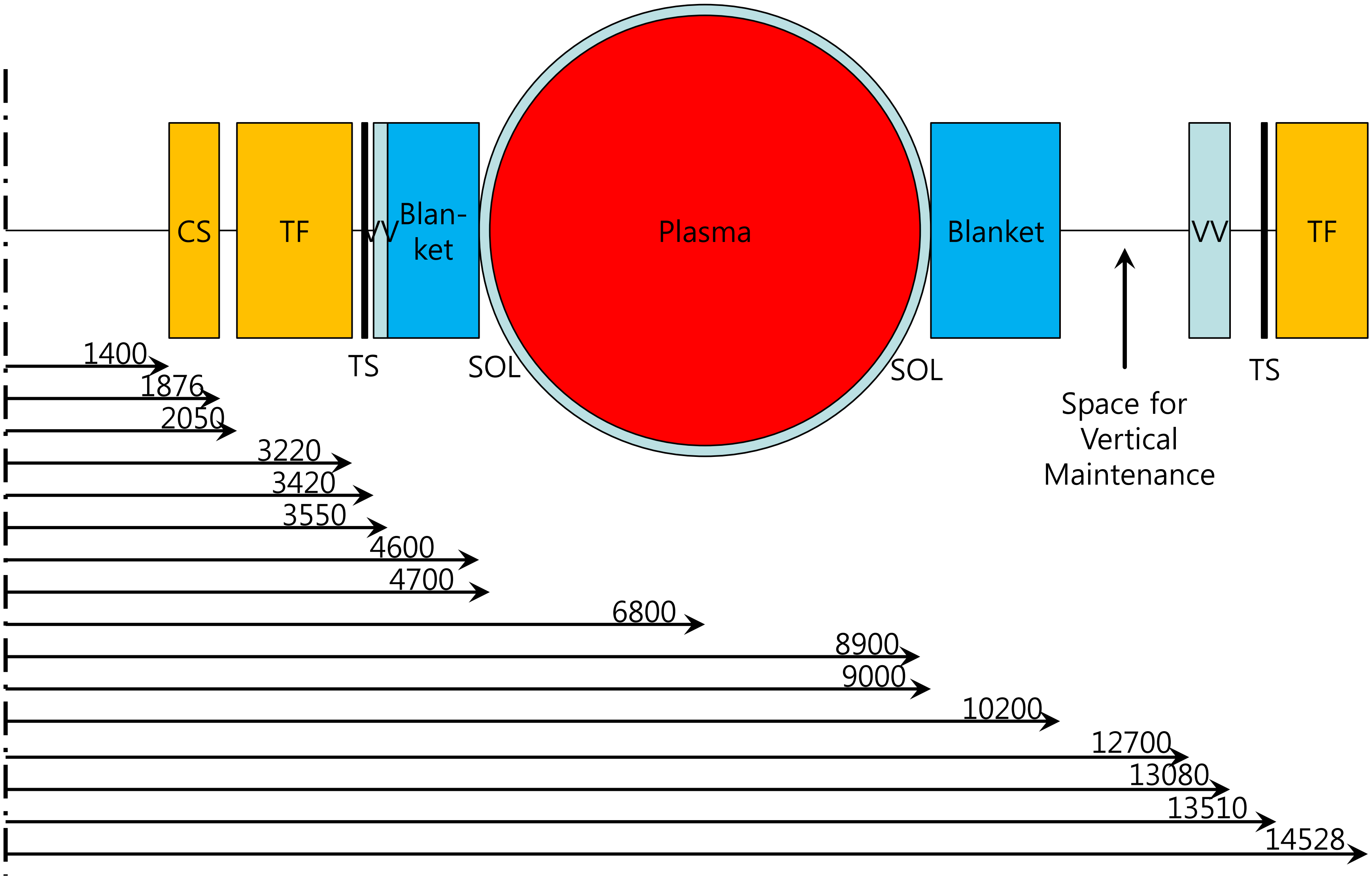




- Peak temperature of tungsten
  ~ 2000 °C

- Coolant temperature:
  290~315 °C (15 MPa)
  ➔ meet a PWR-like coolant conditions

Radial Build of K-DEMO [unit : mm]



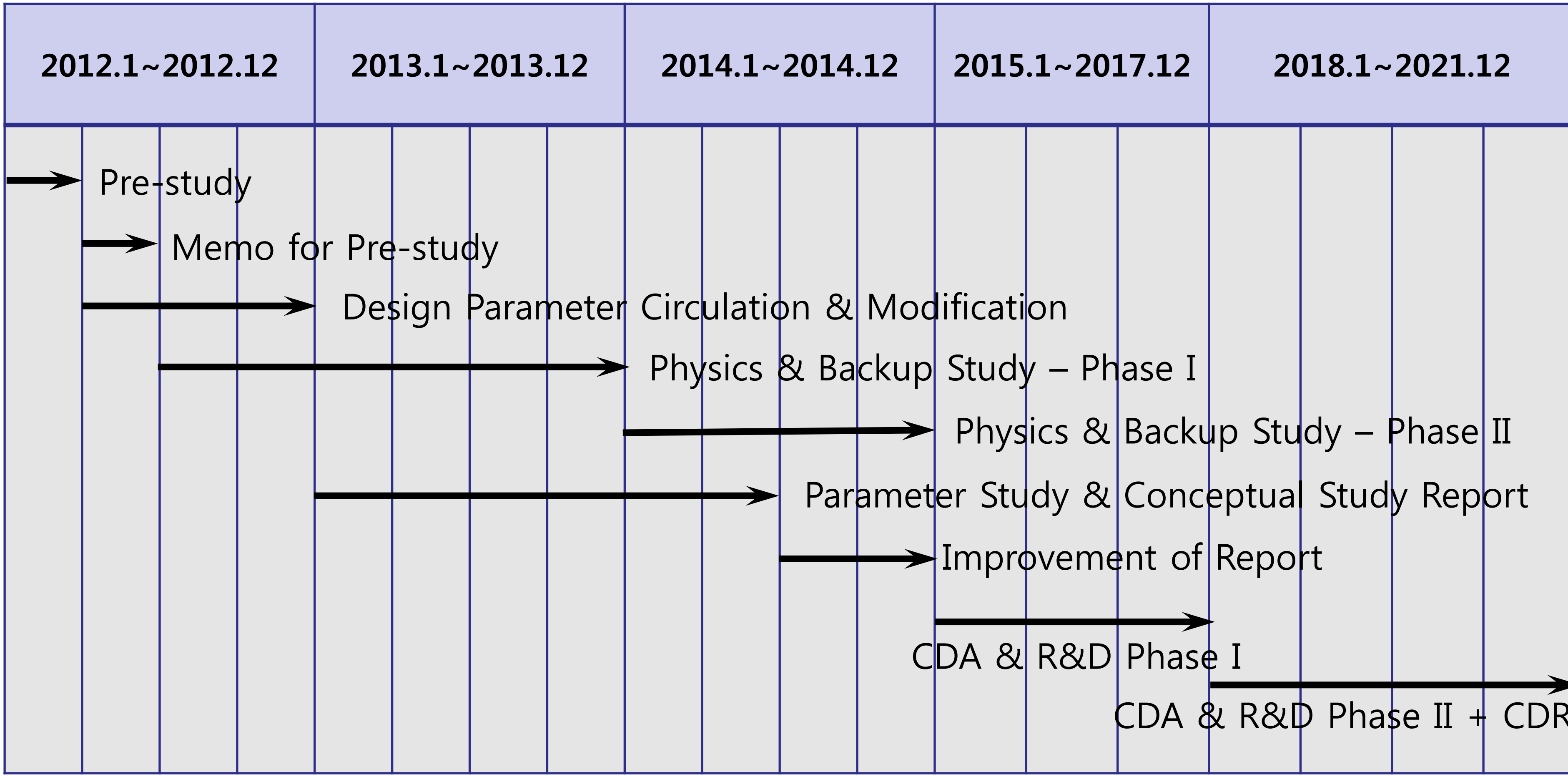
Details of Radial Build of K-DEMO

	COMP BUILD, Z=0 (in)	(mm)	(in)	(mm)	TOTAL (mm)
	Machine Center				0
	Solenoid Center	1390	54.724		1390
CS	ground wrap	10	0.394		
	Winding pack thickness	476	18.740		
	ground wrap	10	0.394	496	1886
	Gap	10	0.394		
	Tie Plate / lead support	114	4.488		
	OH TPT*	6	0.236		
	TF TPT	9	0.354		
	Min OH/TF Gap	10	0.394		
	OH/TF deflection	10	0.394		
	Wedged coil asmbly fit up	5	0.197		
	Bucking Cylinder	0	0.000	164	2050
INBD TF	Ext structure	505	19.882		
	Clearance	5	0.197		
	ground wrap	5	0.197		
	OC Winding pack thickness	360	14.173		
	ground wrap	5	0.197		
	ground wrap	5	0.197		
	IC Winding pack thickness	200	7.874		
	ground wrap	5	0.197		
	Clearance	5	0.197		
	Ext structure	75	2.953	1070	3220
Thermal Insul	TF winding tolerance	10	0.394		
	Wedged coil assembly fit up	10	0.394		
	Trapezoidal Effect	30	1.181		
	TF TPT	10	0.394		
	cold wall / thermal insul	120	4.724		
	Min VV/TI Gap	10	0.394		
	VV TPT	10	0.394	200	3420
Inboard VV	VV shell thickness	40	1.575		
	Shell gap	50	1.969		
	VV shell thickness	40	1.575	130	3550
Backbone	VV TPT	5	0.197		
	EM load displacement	9	0.354		
	Backbone TPT	5	0.197		
	Min VV/Backbone Gap	5	0.197		
	Thermal Shield	5	0.197		
	Backbone shield structure	100	3.937		
	Gap+TPT	5	0.197		
	Diagnostic mounting space	**25**	**0.984**	**159**	**3709**

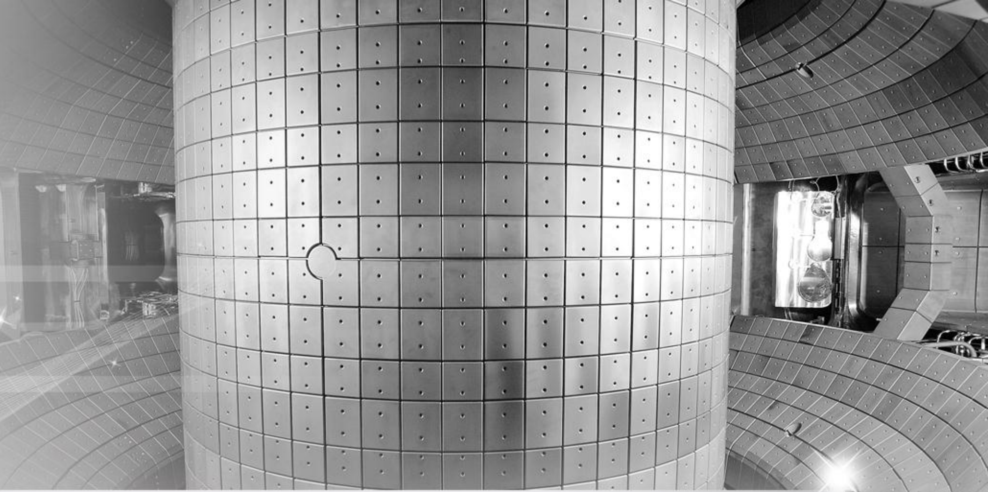
***TPT : True Position Tolerance**

	COMP BUILD, Z=0 (in)	(mm)	(in)	(mm)	TOTAL (mm)
Blanket/Shld	Manifold space	217	8.543		
	Back wall-Shield	120	4.724		
	tolerances	6	0.236		
	Breading Zone	517	20.354		
FW	FW	31	1.220	891	
Plasma R0	Plasma SO	100	3.937		4600
	Plasma minor radii	2100	82.677		4700
	--				6800
	Plasma minor radii	2100	82.677		8900
	Plasma SO	100	3.937		9000
FW/Blanket	FW	31	1.220		
	Blanket - segment 1	517	20.354		
	Passive Plates	10	0.394		
	Blanket - segment 2	0	0.000		
	attachment space	50	1.969	608	9608
Shield	Skeleton Ring Shield	200	7.874	300	9808
	Manifold space	392	15.433	220	10200
Space	Thermal Shield	10	0.394		
	EM load displacement	10	0.394		
	Shield / Bkt TPT	10	0.394		
	VV Shield Gap	2460	96.850		
	VV TPT	10	0.394	2500	12700
Outboard VV	VV shell thk	40	1.575		
	Shell gap	300	11.811		
	VV shell thk	40	1.575	380	13080
Thermal Insul	cold wall / thermal insul	120	4.724		
	TF/VV Gap	290	11.417		
	VV TPT	10	0.394		
	TF TPT	10	0.394	430	13510
OUTBD TF	Ext structure	75	2.953		
	winding tolerance	5	0.197		
	ground wrap	5	0.197		
	IC Winding pack thk	200	7.874		
	ground wrap	5	0.197		
	Space for layer transition	143	5.630		
	ground wrap	5	0.197		
	OC Winding pack thk	360	14.173		
	ground wrap	5	0.197		
	winding tolerance	5	0.197		
	Ext structure	210	8.268	1018	14528

K-DEMO Conceptual Study & CDA Schedule



Target Date for K-DEMO Construction : End of 2037



DEMO Core Technology Development Plan

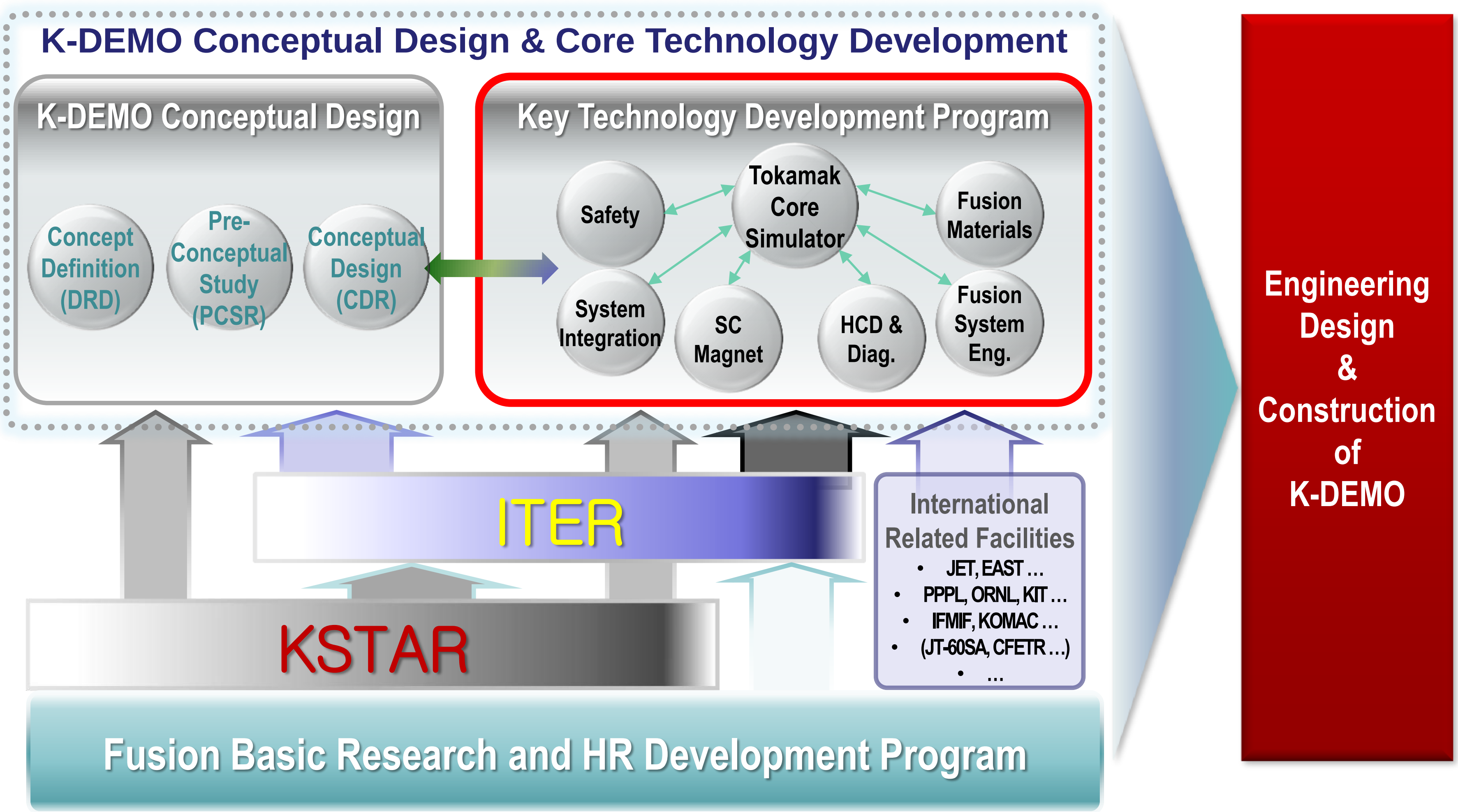
DEMO Core Technology Development Plan

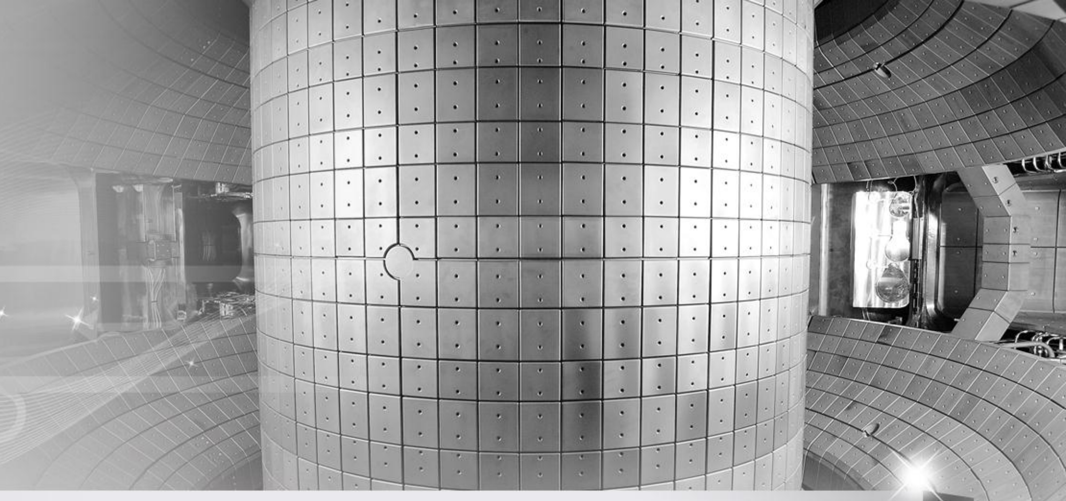
Development of Core Technology

- 3 Major Research Fields, 7 Core Technologies, 18 Detail Technologies and 6 Major Research Facilities
- Through the complete technical planning process with the full participation of experts from all fields covering fusion, fission, physics, computing, mechanics, material, electrics, electronics, and so on.

K-DEMO 3 Major Research Fields	K-DEMO 7 Core Technologies	Major Research Facilities
Design Basis Technology	Tokamak Core Plasma Technology	• Extreme Scale Simulation Center
	Reactor System Integration Technology	
	Safety and Licensing Technology	
Material Basis Technology	Fusion Materials Technology	• Fusion Materials Development Center • Fusion Neutron Irradiation Test Facility • SC Conductor Test Facility
	SC Magnet Technology	
Machine and System Engineering Basis Technology	H&CD and Diagnostics Technology	• Blanket Test Facility • PMI Test Facility
	Heat Retrieval System Technology	

K-DEMO Design & Core Technology Development





Major Facilities

Nation-wide DEMO R&D Center Planning

Chonbuk Province

Daegu Province

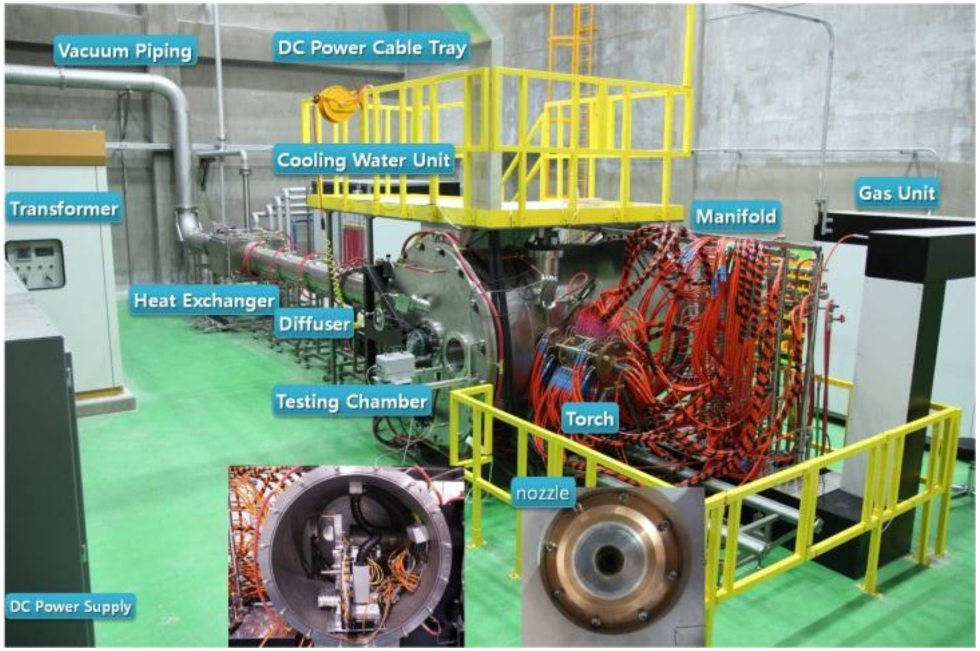
Busan Province

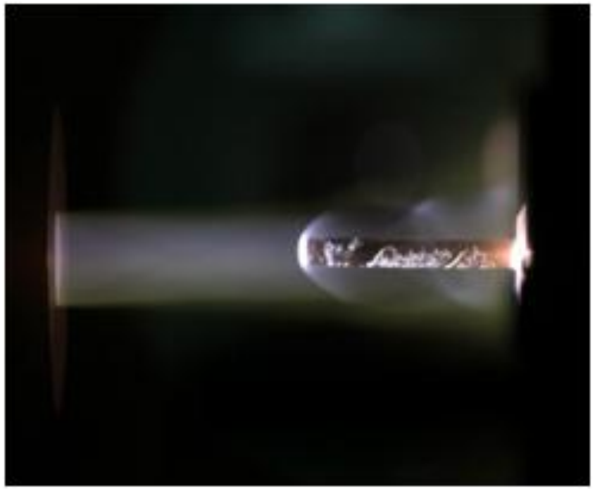
Extreme Environment Material R&D Hub
- Fusion Reactor Materials R&D

Advanced Magnetic Field Center
- Superconductor Test Facility (SUCCEX)

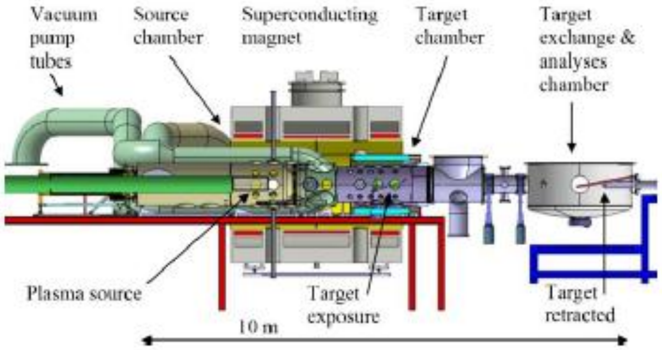
High Enthalpy Plasma Application R&D Center
- Plasma-Material Interaction Test Facility etc.

PMI Test Facility (Chonbuk Province)





- **400kW High-Temperature Plasma Test Facility**
  - Upgrade Plasma Facility for PMI Test
  - Additional, Blanket Test Facility

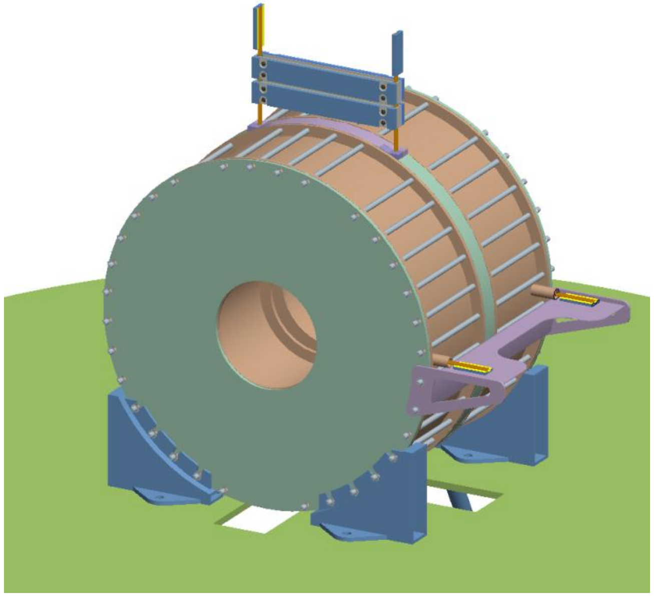


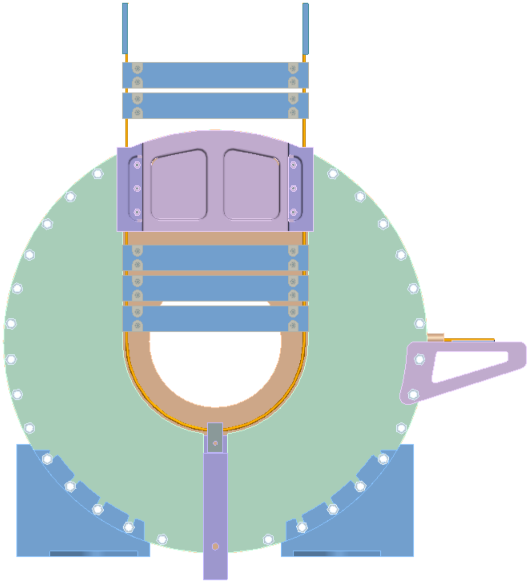
MAGNUM-PSI (Cf.)

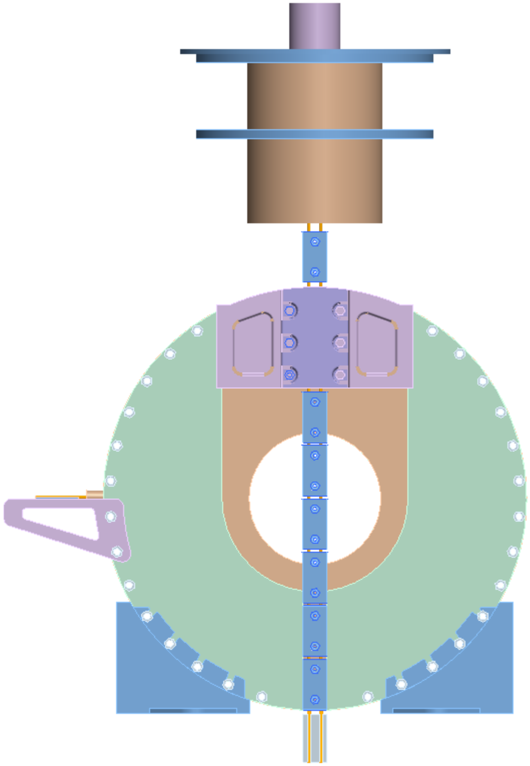
Superconductor Test(SUCCEX) Facility

- SUCCEX : SUperConducting Conductor EXperiment
- Split pair solenoid magnet system
- Inner bore size : ~1 m
- B_{peak} ~ 16.39 Tesla, B_{center} ~ 15.33 Tesla
- Maximum Helium flow channel length : < 100 m
  - One magnet of the split pair consists of three coil (IC, MC, OC) and the maximum of Helium channel length should be maintained below 100 m
  - Every double pancake of each coil will have Helium inlet and outlet
  - Each coil have a number of inter magnet joints because of the maximum fabrication capability in the length of CICC
- Test Mode
  - Semi-circle type conductor sample test mode
    - ✓ U-shape sample with the bottom radius of 0.5 m
    - ✓ DEMO TF conductor will have a rectangular cross-section to reduce the strain effect & will have capability of a minimum bending radius of 0.5 m
    - ✓ No joint : no question regarding voltage arising from the joint
    - ✓ There are enough distance for the voltage relaxation from the sample terminal to voltage taps
  - Sultan like sample test mode
    - ✓ For the case of CS conductor, the size of the is expected to be a range of 50 mm. And it is not easy to make the U-shape sample because of the minimum bending radius
    - ✓ Also the facility could support the joint technology development

Conceptual View of SUCCEX

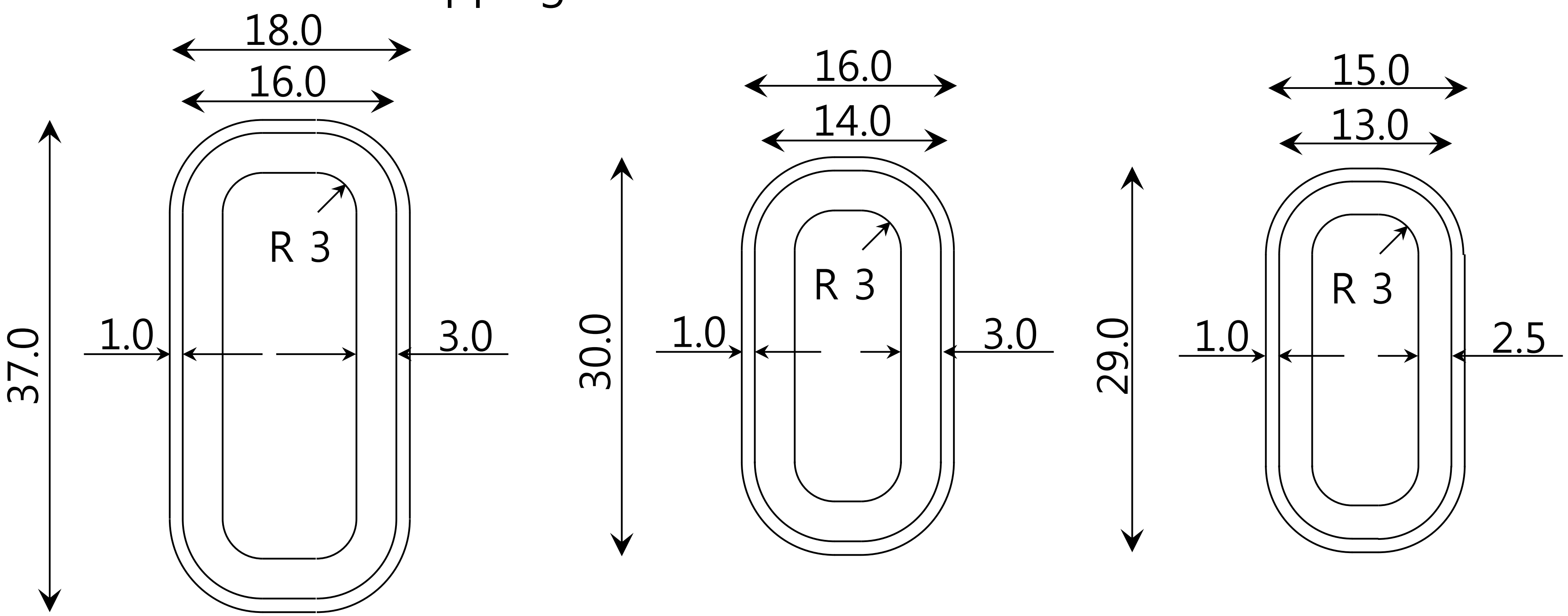






Conductor Parameter of SUCCEX Magnets

- IC (Inner Coil) CICC : (3SC)x4x5x6[360 SC strand], VF = 27.62%
- MC (Middle Coil) CICC : (2SC+1Cu)x3x4x6[144 SC strand], VF = 26.96%
- OC (Outer Coil) CICC : (1SC+2Cu)x3x4x6[72 SC strand], VF = 26.96%
- Strand : High Jc (> 2600A/mm2) Nb_3Sn (total ~ 6.8 ton)
- Twist Pitch : 50 mm – 110 mm – 170 mm – 290 mm
- No Sub-Cable Wrapping

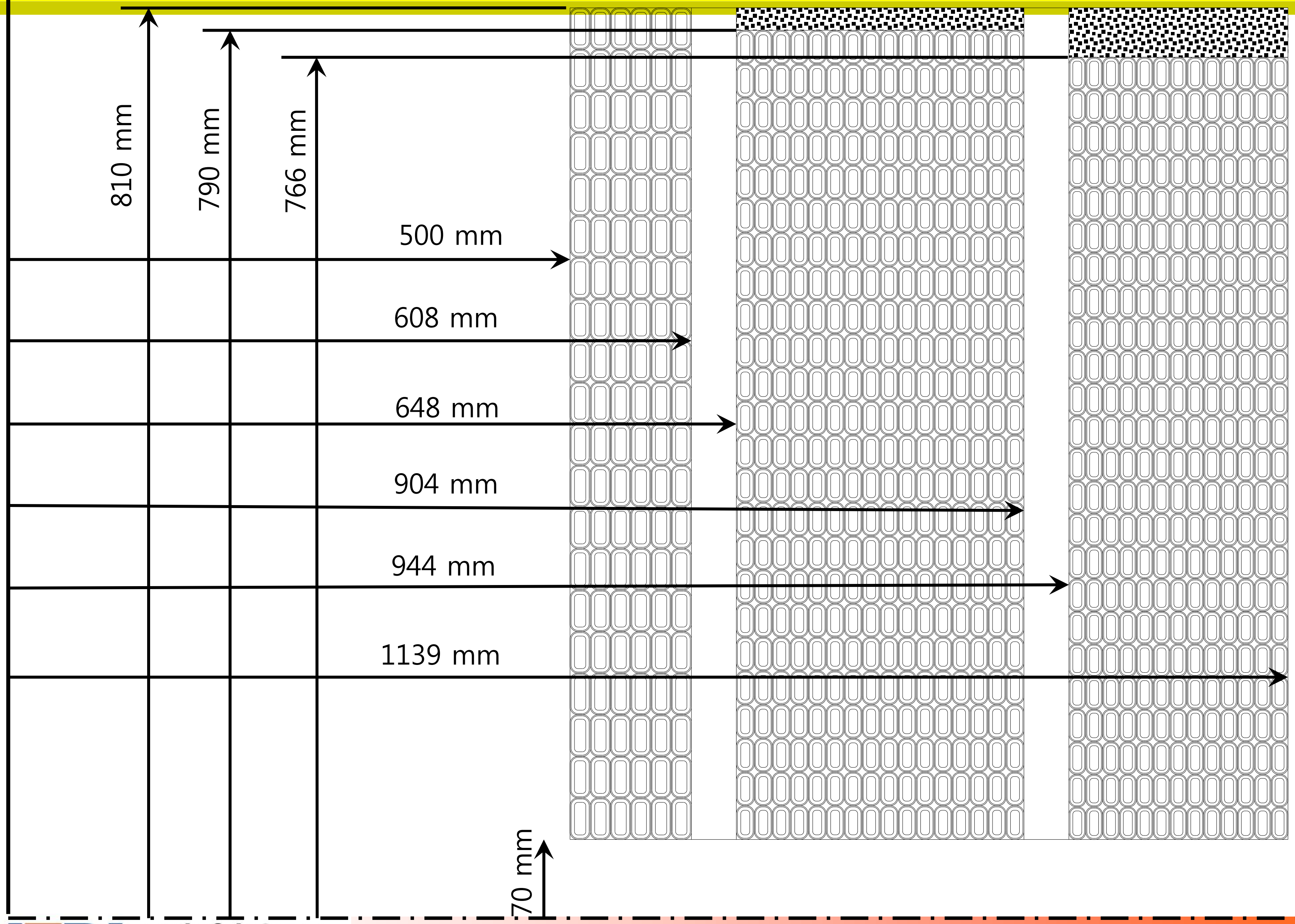


IC CICC Cross-section

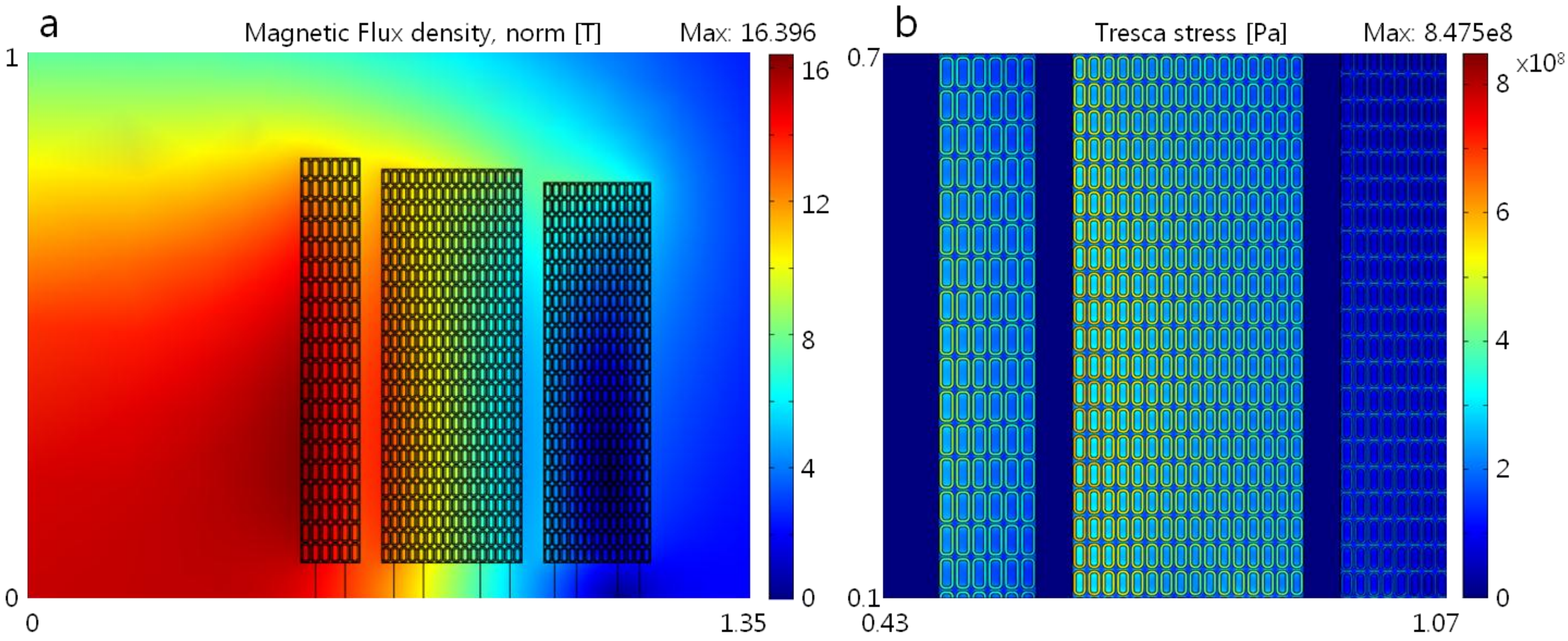
MC CICC Cross-section

OC CICC Cross-section

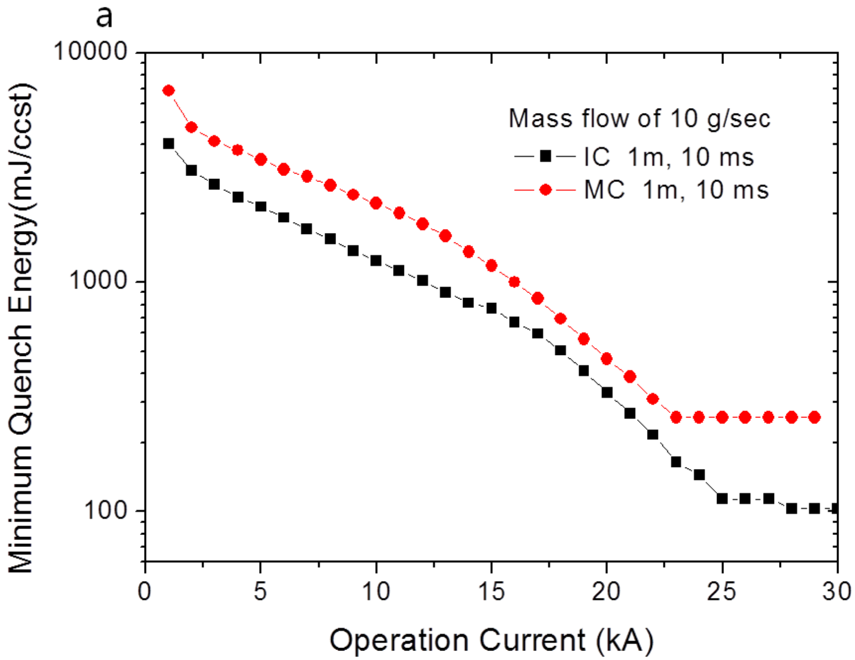
SUCCEX Magnet Cross-Section (Upper Coil)

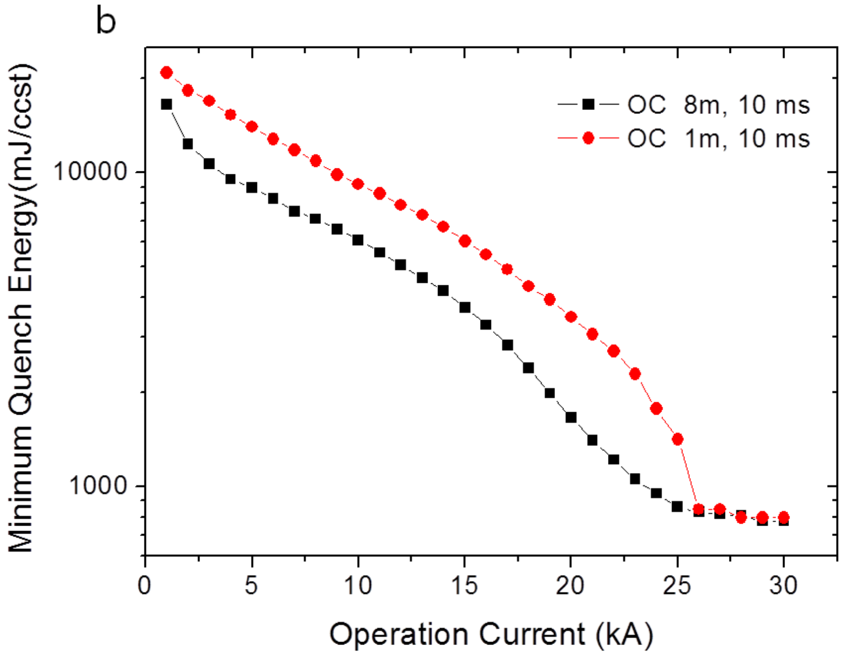


Magnetic Field & Stress of SUCCEX Magnets



Stability Analysis of SUCCEX Magnets

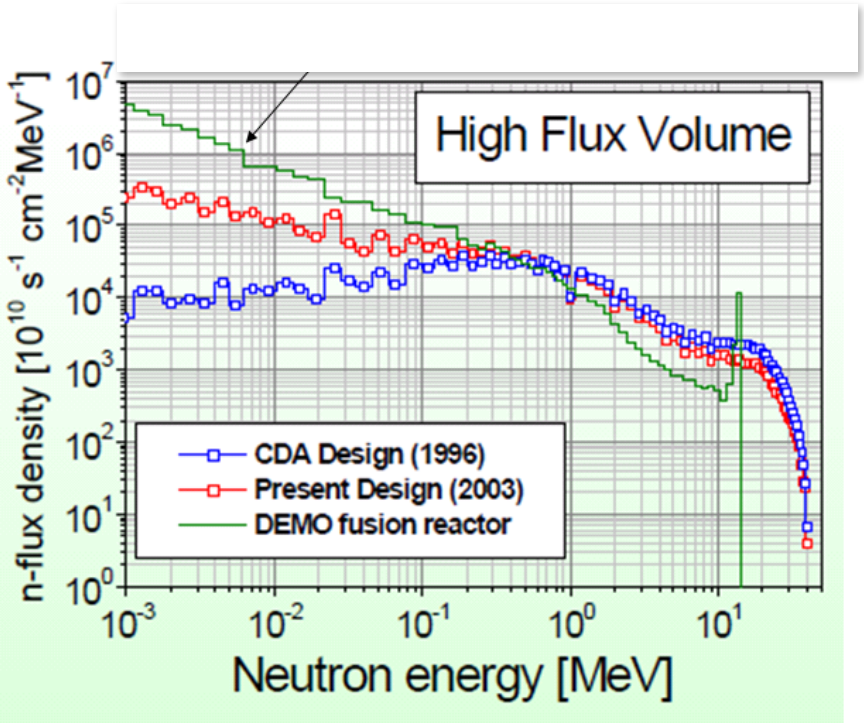




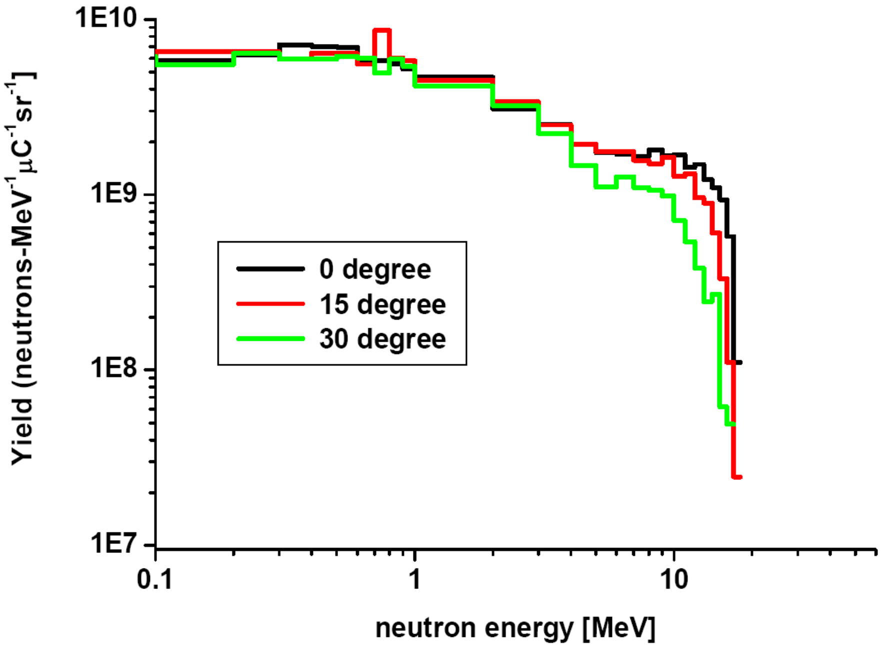
KOMAC Site (Gyeong-ju)

Neutron Energy Spectrum in KOMAC

Fusion Neutron similar Spectrum by Pulse-type Proton beam on Be-target (>1dpa/y)

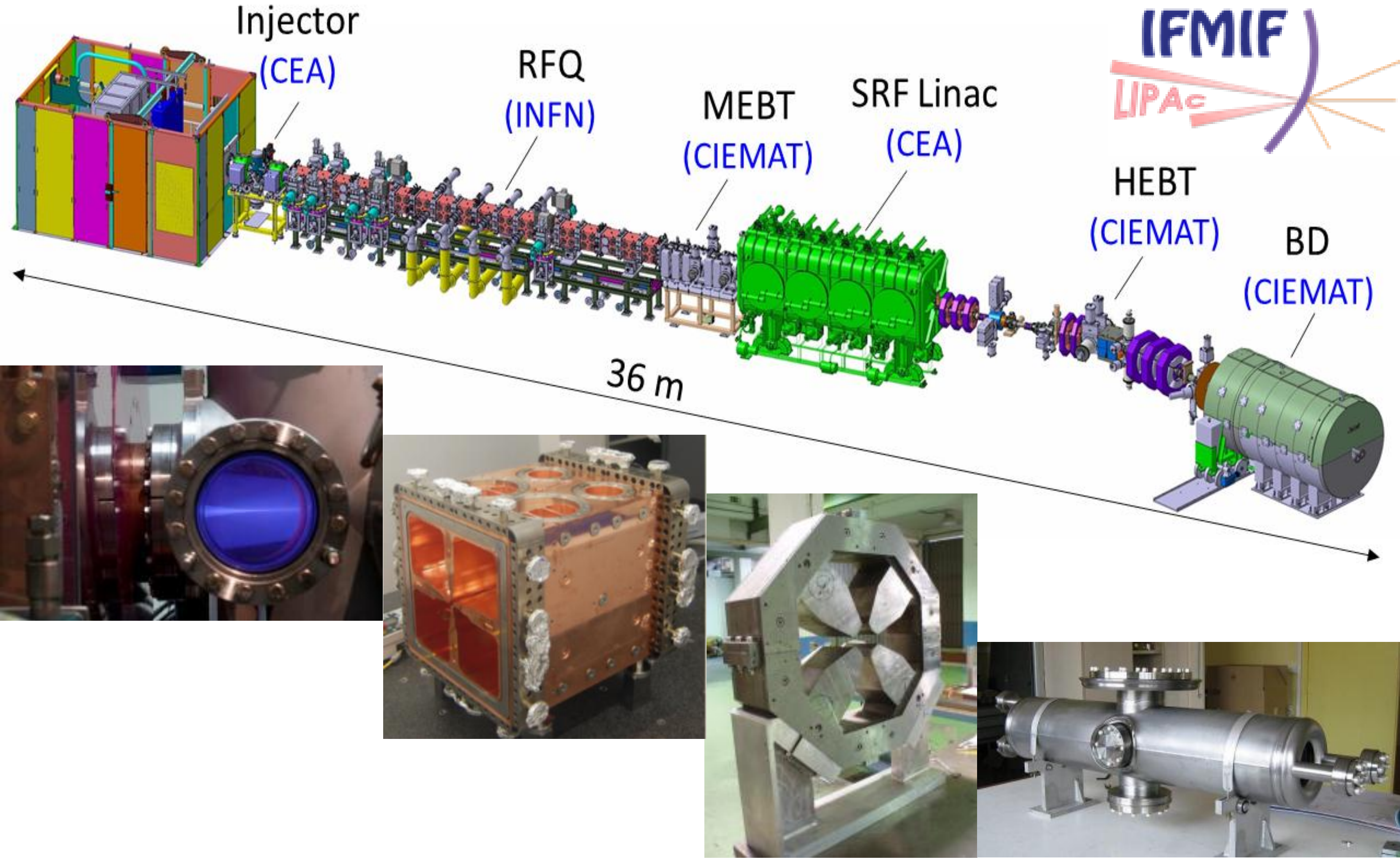


(Ref.) Institute for Materials Research, KIT
I A. Moslang



Neutron Energy Spectrum in KOMAC

IFMIF(International Fusion Material Irradiation Facility)



Summary

- The conceptual study on the Korean fusion demonstration reactor (K-DEMO) has been started in 2012, based on the National Fusion Roadmap released in 2005 and Korean Fusion Energy Development Promotion Law (FEDPL) enacted in 2007.
- After the thorough 0-D system analysis, the major radius and minor radius of K-DEMO are chosen to be 6.8 m and 2.1 m, respectively
- For matching the high frequency ECCD, the designed K-DEMO TF magnet system provides the magnetic field at the plasma center above 7 T with a peak field of ~16 T by using high performance Nb_3Sn-based superconducting strand.
- For a high availability operation, K-DEMO incorporates a vertical maintenance design.
- Pressurized water is the most prominent choice for the main coolant of K-DEMO when considering balance of plant development details.
- A global TBR greater than 1 is achieved using a water cooled ceramic breeder blanket system.
- Considering the plasma performance and the peak heat flux in the divertor system, a double-null divertor system becomes the reference choice of K-DEMO.