Progress on Conceptual Design of the K-DEMO Magnet System

October 20th, 2015

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National Fusion Research Institute
Mid-Entry Strategy: Korea, Year 1995

- Conventional Device (Cu)
- Superconducting Device

**Fusion Power**
- 1GW
- 1MW
- 1KW
- 1W

**Year**
- 1965
- 1970
- 1975
- 1980
- 1985
- 1990
- 1995
- 2000
- 2005
- 2010
- 2015
- 2020
- 2040

**Devices**
- T-3 (1968)
- SNUT-79
- ATC
- ALCATOR A
- PLT
- ALCATOR C
- PDX
- JT-6U
- TFTR
- JET/TFTR
- DIII-D
- KSTAR
- ITER
- DEMO

**Strategies**
- FIRST MOVER
- FAST FOLLOWER
KSTAR and ITER Project
KSTAR Mission and Key Parameters

- **KSTAR missions**
  - To achieve the superconducting tokamak construction and operation experiences
  - To explore the physics and technologies of high performance steady-state operation that are essential for ITER and fusion reactor

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### Achieved key parameters

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Designed</th>
<th>Achieved (~2014)</th>
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<tbody>
<tr>
<td>Major radius, $R_0$</td>
<td>1.8 m</td>
<td>1.8 m</td>
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<tr>
<td>Minor radius, $a$</td>
<td>0.5 m</td>
<td>0.5 m</td>
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<tr>
<td>Elongation, $\kappa$</td>
<td>2.0</td>
<td>1.8</td>
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<td>Triangularity, $\delta$</td>
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<tr>
<td>Plasma shape</td>
<td>DN, SN</td>
<td>DN, SN</td>
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<tr>
<td>Plasma current, $I_p$</td>
<td>2.0 MA</td>
<td>1.0 MA</td>
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<tr>
<td>Toroidal field, $B_0$</td>
<td>3.5 T</td>
<td>3.5 T</td>
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<tr>
<td>H-mode duration</td>
<td>300 s</td>
<td>45 s</td>
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<tr>
<td>$\beta_N$</td>
<td>5.0</td>
<td>4.0</td>
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<tr>
<td>Superconductor</td>
<td>Nb$_3$Sn, NbTi</td>
<td>Nb$_3$Sn, NbTi</td>
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<tr>
<td>Heating/CD</td>
<td>~ 28 MW</td>
<td>~ 7 MW</td>
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<tr>
<td>PFC</td>
<td>C, CFC, W</td>
<td>C</td>
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</table>
KSTAR Final Assembly (2007. 1)
KSTAR Device and Key Components

**Heating & CD (‘15)**
- **NBI**
  - 5.5 MW on-axis
- **ECH/CD**
  - 1 MW 170 GHz
  - 1 MW 105/140 GHz
- **LHCD**
  - 0.5 MW 5 GHz
- **ICRF**
  - 2 MW 30-60 MHz
- **Helicon CD**
  - 0.3 MW 0.5 GHz

**Diagnostics**
- Magnetic & Probes
  - Thomson / ECE
  - Dα / Bolometer
  - Interferometer / Reflect.
  - Visible TV / IRTV
  - ECEI / MIR / RF
  - Soft X-ray & Hard X-ray
  - CES / BES (D & Li)
  - MSE / XICS

**PWI research**
- Graphite PFC
- W-bond marker tiles
- Deposition & erosion

**3D field research**
- In-vessel coils
  - (4 x 3 module)
- Broadband power supply
Recent operation results toward long-pulse large current H-mode discharge

- Long-pulse H-mode discharge: 48s (0.5 MA), 43s (0.6 MA),
  - $I_p = 0.6$ MA, $t_{H\text{mode}} \approx 43$ s, $B_T = 2$ T, $P_{\text{NBI}} \approx 4.3$ MW,
  - $W_{\text{dia}} \approx 0.4$ MJ, $<n_e> \approx 2 \times 10^{19}/m^3$,
  - $\beta_N \approx 2.1$, $H_{89} \approx 1.7$, $f_{\text{NI}} \approx 0.8$
  - Limited by electric MVA interlock

- Large plasma current H-mode discharge 1 MA (9s), goal: 2 MA (300s)

$\beta_N \approx 2.1$, $H_{89} \approx 1.7$, $f_{\text{NI}} \approx 0.8$

Limited by electric MVA interlock
Exploring the advanced high performance scenarios development

- High beta ($\beta_N \sim 4.0$) operation exceeding ideal $n=1$ no wall limit
- Extremely low $q_{95}$ ($q_{95} \sim 2.0$) operation verifying extremely low error field
- Fully non-inductive H-mode discharge 12s (0.4 MA, $f_{NI} \sim 1.0$)
  - $I_p = 0.4$ MA, $t_{Hmode} \sim 14$s, $B_T = 2.9$T,
  - $P_{NBI+ECH} \sim 5.6$ MW, $W_{dia} \sim 0.4$MJ,
  - $\beta_N \sim 2.1$, $l_i \sim 1.3$, $f_{NI} \sim 1.0$
  - Limited by limiter temperature interlock #13008

\[ \beta_N \sim 4.0 \]

\[ q_{95} \sim 2.0 \]

\[ \text{Vloop} \sim 0 \text{ V, 12s} \]
Validation of low intrinsic error field and potential of 3D field research

- Lowest intrinsic error field (EF) and TF ripple compared to any other tokamaks
  - Extremely low intrinsic error field measured by standard campus scan using IVCC
    - $\delta B_{2/1} / B_0 \sim 1 \times 10^{-5}$
    - TF ripple at edge $\sim 5 \times 10^{-4}$

- 3D field research capability using in-vessel control coils & broadband power supplies
  - Unique features of in-vessel coils with 3 poloidal rows
    - Magnetic perturbation at n=1, n=2, and mixed
    - Mixed mode error field perturbation study
    - Dynamic error field correction
    - NTV rotation control, RWM stabilization

- Establish limits for confinement and stability of the tokamak plasmas
**Edge Localized Mode (ELM) - crash suppression using low-n error field uniquely in KSTAR**

- KSTAR is unique device showing the ELM-crash suppression at n=1 (up to 4s) and n=2 middle coil operation (up to 5s).
  - Successful ELM suppression at low n could be related to low error field.
  - Check the possibility of the ex-vessel control as for ELM control in ITER and DEMO.

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**Graphs:**

- ELM-crash suppression at n=1 (4s)
- ELM-crash suppression at n=2 (5s)
Re-assessment of ELM crash mechanism using advanced diagnostics (2D/3D ECEI, MIR, RF, etc)

- 2D/3D observation of the edge filamentary structure using ECEI advanced diagnostics gives a new assessment of the ELM crash and suppression mechanism

- Under ELM crash suppressed phase, conventional Dα signal is suppressed.
- But Edge localized mode is remain (Higher-n coherent modes, marginally stable)

Gunsu Yun, ITER ELM workshop (2015)
1. TF Conductor (Completed)
Total Value (kIUA) : 215.01
KO Allocation : 20.2%
KO Contribution (kIUA) : 43.39

2. Vacuum Vessel Main Body
Total Value(kIUA) : 123.04
KO Allocation : 21.3%
KO Contribution (kIUA) : 26.20

3. Vacuum Vessel Port
Total Value(kIUA) : 76.96
KO Allocation : 72.7%
KO Contribution (kIUA) : 55.98

4. Thermal Shield
Total Value(kIUA) : 26.88
KO Allocation : 100%
KO Contribution(kIUA) : 26.88

5. Blanket Shield Block
Total Value(kIUA) : 56.34
KO Allocation : 49.8%
KO Contribution(kIUA): 28.07

6. Assembly Tooling
Total Value(kIUA) : 23.01
KO Allocation : 100%
KO Contribution(kIUA): 23.01

7. Tritium SDS
Total Value(kIUA) : 15.21
KO Allocation : 82.1%
KO Contribution(kIUA) : 12.48

8. AC/DC Converters
Total Value(kIUA) : 123.58
KO Allocation : 37.3%
KO Contribution(kIUA): 46.06

9. IVC Bus bars
Total Value(kIUA) : 3.98
KO Allocation : 100%
KO Contribution (kIUA) : 3.98

10. Diagnostics
Total Value(kIUA) : 142.09
KO Allocation : 3.2%
KO Contribution (kIUA) : 4.49

11. Test Blanket Module*
KO Contribution :
HCCR TBS (TBM System)
kIUA Value : N/A

* TBMA (TBM Agreement) was signed in 2014
Total Value : 270.54 kIUA
# Procurement Schedule of the KODA

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- **PA**: Procurement Action
- **Contract Award**
- **1st Delivery**
- **Last Delivery**
ITER Procurement Activities of Korea

KO TF conductors (20.18%) consist of 19 rDPs (760 m) and 8 sDPs (415 m).
- Production of strands and cabilities was completed in 2013 and in May 2014, respectively.
- All 27 TF conductors were delivered to JADA by the end of November 2014, on schedule.

The thermal shields will be provided by Korea.
- Preparation for fabrication of Vacuum Vessel Thermal Shield is on going;
  - Manufacturing drawing
  - Manufacturing procedure
- After manufacturing of VVTS 10-degree prototype.
  - Water-jet cutting
  - 2D bending
  - 3D forming
  - In-board welding
  - Out-board welding
  - Cooling tube fit-up and welding
  - Machining of in-board and out-board
  - 3D scanning
  - VVTS 10-degree prototype & its test

Manufacturing of vacuum vessel sector 6 is on-going at HHI of Korea.
- Welding of inter-modular & centering keys of inner shell of upper segment (PS2)
- Forming of inner shell & machining of divertor rail, port stub corner, 4-pipe & penetration of lower segment (PS4)
- Start welding after inner and outer jigs for inner shell of equatorial segment (PS3)
- Inner and outer jigs for welding after forming of inner shell of inboard segment (PS1)

Manufacturing sequence of upper segment (PS2) of VV sector 6
- Fit-up for welding of 8 keys
- In & outside welding
- VT & PT
- RT
- Fit-up for welding of 6 keys
- Completion of inside welding

This is the first successful procurement item from KO.
Fusion Energy Development Roadmap in Korea
To establish a long-term and sustainable legal framework for fusion energy development phases.

To promote industries and institutes participating fusion energy development by support and benefit.

The first country in the world that 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 (First 5-Year National Plan)
- **2012. 1**: The 2nd 5-year National Plan has started.
Vision and Goal of Fusion Energy Development Policy

**Vision**

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

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

**R&D for DEMO Technology based on KSTAR and ITER**

- § 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
## K-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.

<table>
<thead>
<tr>
<th>K-DEMO 3 Major Research Fields</th>
<th>K-DEMO 7 Core Technologies</th>
<th>Major Research Facilities</th>
</tr>
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<tbody>
<tr>
<td><strong>Design Basis Technology</strong></td>
<td>Tokamak Core Plasma Technology</td>
<td>• Extreme Scale Simulation Center</td>
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<td></td>
<td>Reactor System Integration Technology</td>
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<td>Safety and Licensing Technology</td>
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<tr>
<td><strong>Material Basis Technology</strong></td>
<td>Fusion Materials Technology</td>
<td>• Fusion Materials Development Center</td>
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<td>SC Magnet Technology</td>
<td>• Fusion Neutron Irradiation Test Facility</td>
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<tr>
<td><strong>Machine and System Engineering Basis Technology</strong></td>
<td>H&amp;CD and Diagnostics Technology</td>
<td>• Blanket Test Facility</td>
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<tr>
<td></td>
<td>Heat Retrieval System Technology</td>
<td>• PMI Test Facility</td>
</tr>
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</table>
K-DEMO Magnet System
**Mission & Strategy**

**Mission:** To demonstrate the sustainable generation of electricity from fusion power

**Strategy:**

- **Natural Path:** KSTAR $\Rightarrow$ ITER $\Rightarrow$ DEMO (tokamak)

- **To mitigate risks in the course of DEMO development** $\Rightarrow$ Two-Phased Operation strategy

- **The operation Stage I** $\Rightarrow$ not considered as a final DEMO
  - At least one port will be designated for the CTF including blanket test facility.
  - To demonstrate the net electricity generation ($Q_{\text{eng}} > 1$) and the self-sufficient Tritium cycle ($TBR > 1.05$).

- **The operation Stage II**
  - Major upgrade of In-Vessel-Components
  - To demonstrate the net electricity generation $> 400$ Mwe.
  - 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, \( I_{p,\text{limit}} \propto B \), \( n_{e,\text{limit}} \propto B \), and \( \text{Power} \propto R^3 B^4 \) \[ \text{[Reactor Cost} \propto R^3 B^2] \]

### Choice of Coolant and Blanket System
- Pressurized water (superheated water) is considered as a main coolant of K-DEMO considering BOP(Balance of Plant).
- Supercritical \( \text{CO}_2 \) is also considered as a future coolant.
- Helium is also a candidate as a coolant of K-DEMO, but there are concerns about its low heat capacity, a required high pumping power and BOP.
- 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
- $R = 6.8\ m$
- $a = 2.1\ m$
- $B_{\text{center}} = 7.0\sim7.4\ T$
- $B_{\text{peak}} = 16\ T$
- $\kappa_{95} = 1.8$
- $\delta = 0.625$
- Plasma Current $> 12\ MA$
- $T_e > 20\ keV$

Other Feature
- Double Null Configuration
- Vertical Maintenance
- Total H&CD Power $= 80\sim120\ MW$
- $P_{\text{fusion}} = 2200\sim3000\ MWth$
- $P_{\text{net}} > 400\ MWe\ at\ Stage\ II$
- Number of Coils: 16\ TF, 8\ CS, 12\ PF
2D Drawing of Magnet System
CICC Dimensions and Trial Fabrication

TF LF CICC

TF HF CICC

CS CICC

PF CICC
### CICC Parameter

<table>
<thead>
<tr>
<th>Parameter</th>
<th>TF HF</th>
<th>TF LF</th>
<th>CS</th>
<th>PF1-4</th>
<th>PF5-6</th>
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</thead>
<tbody>
<tr>
<td>Cable pattern</td>
<td>(3SC)x4x5x6x6 + Helical Spiral</td>
<td>(((2SC+2Cu)x5)x6+7 Cu)x6 + Central Spiral</td>
<td>(2SC+1Cu) x3x4x4x6 No Cooling Spiral</td>
<td>(2SC+1Cu)x3x4x4x5 + Central Spiral</td>
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<tr>
<td>No. of SC strands</td>
<td>1800</td>
<td>360</td>
<td>576</td>
<td>480</td>
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<tr>
<td>No. of Copper strands</td>
<td>-</td>
<td>432</td>
<td>288</td>
<td>240</td>
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<tr>
<td>Spiral Dimension (mm)</td>
<td>ID 7 / OD 11</td>
<td>ID 7 / OD 9</td>
<td>360</td>
<td>480</td>
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<td>Void Fraction (%)</td>
<td>27.6</td>
<td>26.0</td>
<td>36.6</td>
<td>32.5</td>
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<td>Strand Type</td>
<td>High Jc (&gt; 2600 A/mm²) Nb3Sn Strand 0.82 mm diameter</td>
<td>ITER type (Jc ~ 1000 A/mm²) Nb3Sn Strand 0.82 mm diameter</td>
<td>NbTi Strand 0.82 mm diameter (~90 ton)</td>
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<td>Cu/non-Cu of Strand</td>
<td>1.0</td>
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<td>Insulation</td>
<td>1.6 mm (including Voltage Tap) (0.1 mm Kapton 400% + 0.3 mm S glass 400%)</td>
<td>2.0 mm (including Voltage Tap) (0.1 mm Kapton 400% + 0.4 mm S glass 400%)</td>
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<td>Jacket Thickness</td>
<td>5.0 mm</td>
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<tr>
<td>Twist Pitch (mm)</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>1st stage</td>
<td>80 ± 5</td>
<td>80 ± 5</td>
<td>27 ± 5</td>
<td>35 ± 5</td>
<td></td>
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<tr>
<td>2nd stage</td>
<td>140 ± 10</td>
<td>140 ± 10</td>
<td>45 ± 10</td>
<td>75 ± 10</td>
<td></td>
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<tr>
<td>3rd stage</td>
<td>190 ± 10</td>
<td>190 ± 10</td>
<td>85 ± 10</td>
<td>135 ± 10</td>
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<tr>
<td>4th stage</td>
<td>245 ± 15</td>
<td>300 ± 15</td>
<td>150 ± 15</td>
<td>285 ± 15</td>
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<tr>
<td>5th stage</td>
<td>415 ± 20</td>
<td>-</td>
<td>385 ± 20</td>
<td>410 ± 20</td>
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<tr>
<td>Wrapping Tape</td>
<td></td>
<td></td>
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<tr>
<td>Sub-cable wrap thickness</td>
<td>0.08 mm, 40% coverage</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>Sub-cable wrap width</td>
<td>15 mm</td>
<td></td>
<td></td>
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<tr>
<td>Cable wrap thickness</td>
<td>0.4 mm, 60% coverage</td>
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<tr>
<td>Final wrap width</td>
<td>7 mm</td>
<td></td>
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</tbody>
</table>
TF Winding and Structure

- Dual winding packs with 2 types of CICC
- High magnetic field with huge Cost savings
- High Jc Strands
- No Radial Plate
Cross-Section of TF Coil

- **Selected for Detailed Study (Maintenance Space = 2.5 m)**
- **Considering Vertical Maintenance Scheme**
- \( R = 6.8 \text{ m}, \ a = 2.1 \text{ m} \)
- Small CICC Coil: 18 x 10 turns, Large CICC Coil: 12 x 5 turns (Total: 240 turns)
- Magnetic Field at Plasma Center: \( \sim 7.4 \text{ Tesla} \) (\( B_{\text{peak}} \sim 16 \text{ Tesla}, T_{\text{-margin}} > 1 \text{ K} \))
- Nominal Current: 65.52 kA
- Conductor Length:
  - LQP = \( \sim 900 \text{ m} \) (Quadruple Pancake) (Total: \( \sim 450 \text{ ton} \))
  - SDP = \( \sim 930 \text{ m} \) (Double Pancake) (Total: \( \sim 280 \text{ ton} \))
Structural Analysis of TF Case

- TF coil case stresses contoured to 900 MPa maximum !!!

δ ~ 1 mm

- Elastic deformation occurs from the top-right corner of TF inboard side and the maximum stress at the top-left corner can be reduced.

- A detail analysis required to make the stress almost uniform (average stress)
Problem in the Helical Cooling Spiral Shape

- One sub-cable between two smaller cooling spirals
- (3SC)x4x5x6x5 + 2 Helical spirals
- Spiral size: ID=6 mm, OD=9 mm
- CICC height increased to 74 mm (Reduction of peak field)
- Void Fraction = 28%

Elliptic !!!
Reduction of Void Fraction
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
Cross-Section 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 Tesla (Bpeak < 12.194 Tesla, Half Flux Swing ~83 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

- Inner layer transition
- Turn transition
- Intercoil joint con.
- Outer layer transition
- In/Out Cooling pipe
- Feeder extension
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,
Cross-Section 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

Gap: 150 mm
Radial Build of K-DEMO [unit : mm]

- CS
- TF
- Blanket
- Plasma
- Blanket
- VV
- TF

Space for Vertical Maintenance

- 1400
- 1876
- 2050
- 3220
- 3420
- 3550
- 4600
- 4700
- 6800
- 8900
- 9000
- 10200
- 12700
- 13080
- 13510
- 14528
Superconducting Conductor Experiment Facility [SUCCEX]
SUCCEX Facility

**SUCCEX (SUperConducting Conductor EXperiment)**

- **Background field**: 16 Tesla
- **Split-pair Solenoid Magnet System**
- **Inner-bore Size**: ~ 1 m
- **Two Test Modes**:
  - Sultan-like sample test mode
  - Semi-circle type conductor sample test mode

(Cf.) **SULTAN**

- **Background field**: 11 Tesla
- **100 kA SC Transformer** for the short sample test

SULTAN-type  Semi-circle-type
**Conductor Parameter of SUCCEX Magnets**

- **IC (Inner Coil) CICC**: $(3SC)\times4\times5\times6$ [360 SC strand], $VF = 27.62\%$
- **MC (Middle Coil) CICC**: $(2SC+1Cu)\times3\times4\times6$ [144 SC strand], $VF = 26.96\%$
- **OC (Outer Coil) CICC**: $(1SC+2Cu)\times3\times4\times6$ [72 SC strand], $VF = 26.96\%$
- **Strand**: High $J_c$ (> 2600A/mm²) Nb3Sn (total ~ 6.8 ton)
- **Twist Pitch**: 50 mm - 110 mm - 170 mm - 290 mm
- **No Sub-Cable Wrapping**

![Cross-section diagrams of IC, MC, OC CICC with dimensions and annotations]
SUCCEX Magnet Cross-Section (Upper Coil)
Magnetic Field & Stress of SUCCEX Magnets
Stability Analysis of SUCCEX Magnets

Graph a:
- Mass flow of 10 g/sec
- IC 1m, 10 ms
- MC 1m, 10 ms

Graph b:
- OC 8m, 10 ms
- OC 1m, 10 ms

Operation Current (kA)
Minimum Quench Energy (mJ/ccst)
**Conclusion**

<table>
<thead>
<tr>
<th>KSTAR : Physics Machine</th>
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<tbody>
<tr>
<td>• High Performance ($\beta_N &gt; 4$) Plasma Research for Fusion Power Plant</td>
</tr>
<tr>
<td>• Fusion Physics Validation</td>
</tr>
<tr>
<td>• Tokamak Simulator Development</td>
</tr>
<tr>
<td>• Design Requirement for Fusion Plant</td>
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<tr>
<th>ITER : Fusion Engineering</th>
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<tbody>
<tr>
<td>• Stable ($\beta_N \sim 2$) Burning Plasma &amp; Fusion Nuclear Science Research (14 MeV Neutron Effect)</td>
</tr>
<tr>
<td>• Confirmation of Engineering Feasibility for Fusion Power Plant</td>
</tr>
</tbody>
</table>

**Construction & Operation of Artificial SUN (Fusion Power Plant)**

[$\beta_N > 4$ required]