Design Study of NCT and its Shape and Aspect Ratio Controllability

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JAERI-Naka

IEA/LT Workshop (W59) combined with DOE/JAERI Technical Planning of Tokamak Experiments (FP1-2) :
Shape and Aspect Ratio Optimization for High Beta Steady-State Tokamak

GA, San Diego, 15 Feb., 2005

Contents

1. Superconducting conductor developments

1.1 Selection of superconductor

The device size of NCT is limited by present JT-60 facilities such as NBI units.

In order to attain compact coils, Nb_3Al, Nb_3Sn and NbTi strand with
(1) High critical current density (Jc),
(2) High Cu/non-Cu ratio for stability
have been developed.

NbTi: High Jc at low B, low AC loss.

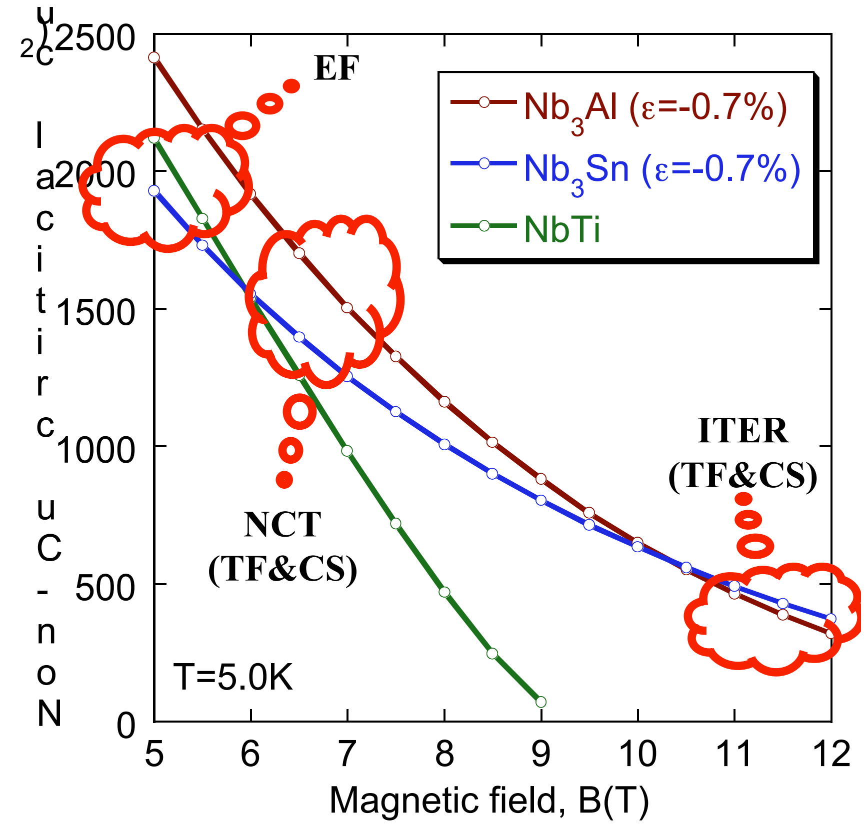
Suitable for EF coils

Nb_3Al: High Jc at high B

Suitable for TF coil

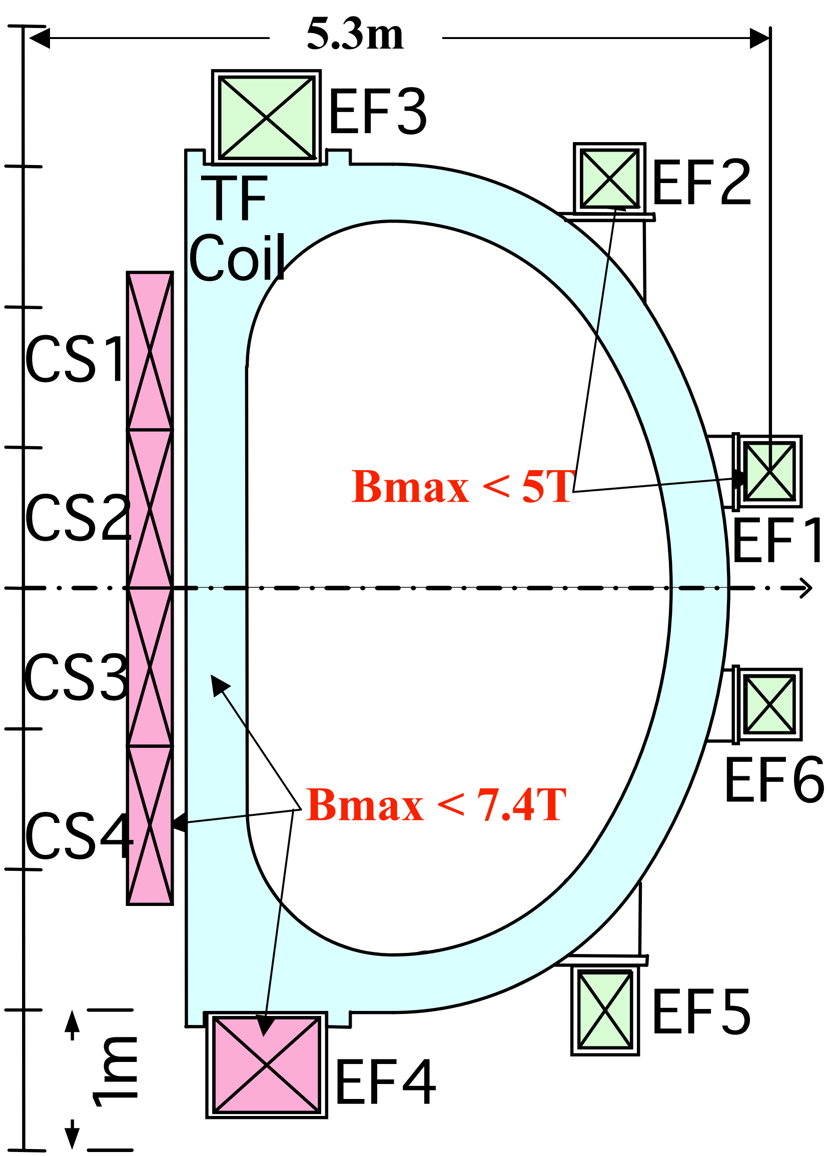
Nb_3Sn: High Jc at high B, low AC loss

Suitable for CS & divertor coil



Strand	Cu/non-Cu ratio
Nb_3Al for NCT	4.0
Nb_3Sn for NCT	2.3
NbTi for NCT	7.0
Nb_3Sn for ITER	1.0-1.5

1.2 Developed coil technologies



Design example of superconducting coil system for NCT

Toroidal Field (TF) coil

Nb_3Al conductor

Demonstration of coil fabrication by react-and-wind (R&W) method.

Central solenoid (CS), Divertor coil

Nb_3Sn conductor

Novel coil winding technique to attain low AC loss from beginning of operation.

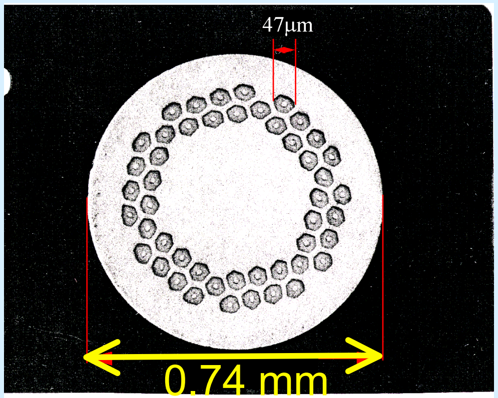
Equilibrium field (EF) coil

NbTi conductor

Development of low cost and low AC loss NbTi conductor with Ni plating strands

1.3 Design of Superconducting Conductors

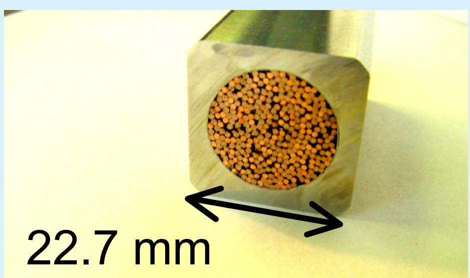
TF coil



Nb_3Al

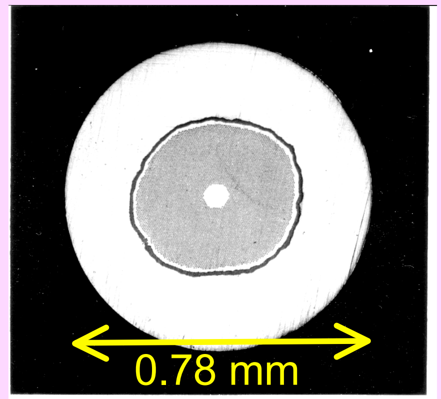
SC 216
(+Cu108)

Cu Ratio 4.0



Full size Nb_3Al CICC

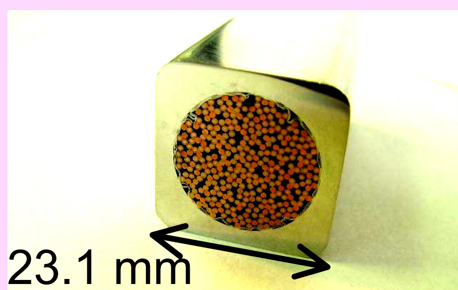
CS, EF4



Nb_3Sn

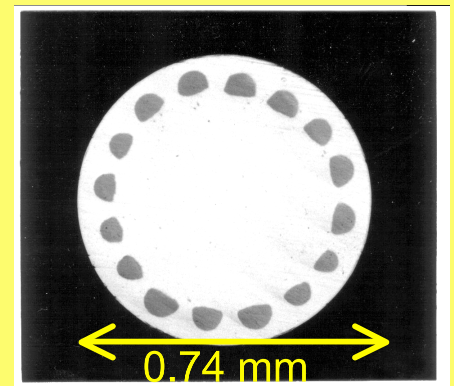
SC 216
(+Cu 108)

Cu Ratio 2.3



Full size Nb_3Sn CICC

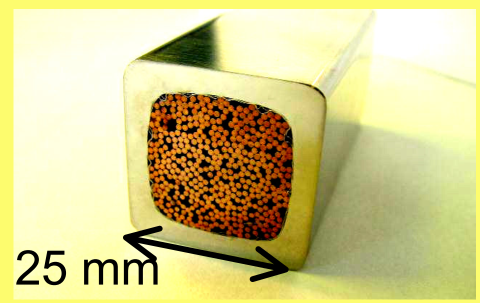
EF coil



NbTi

SC 432

Cu Ratio 7.0



Full size NbTi CICC

1.4 TFC design using the developed Nb_3Al superconducting conductor for NCT

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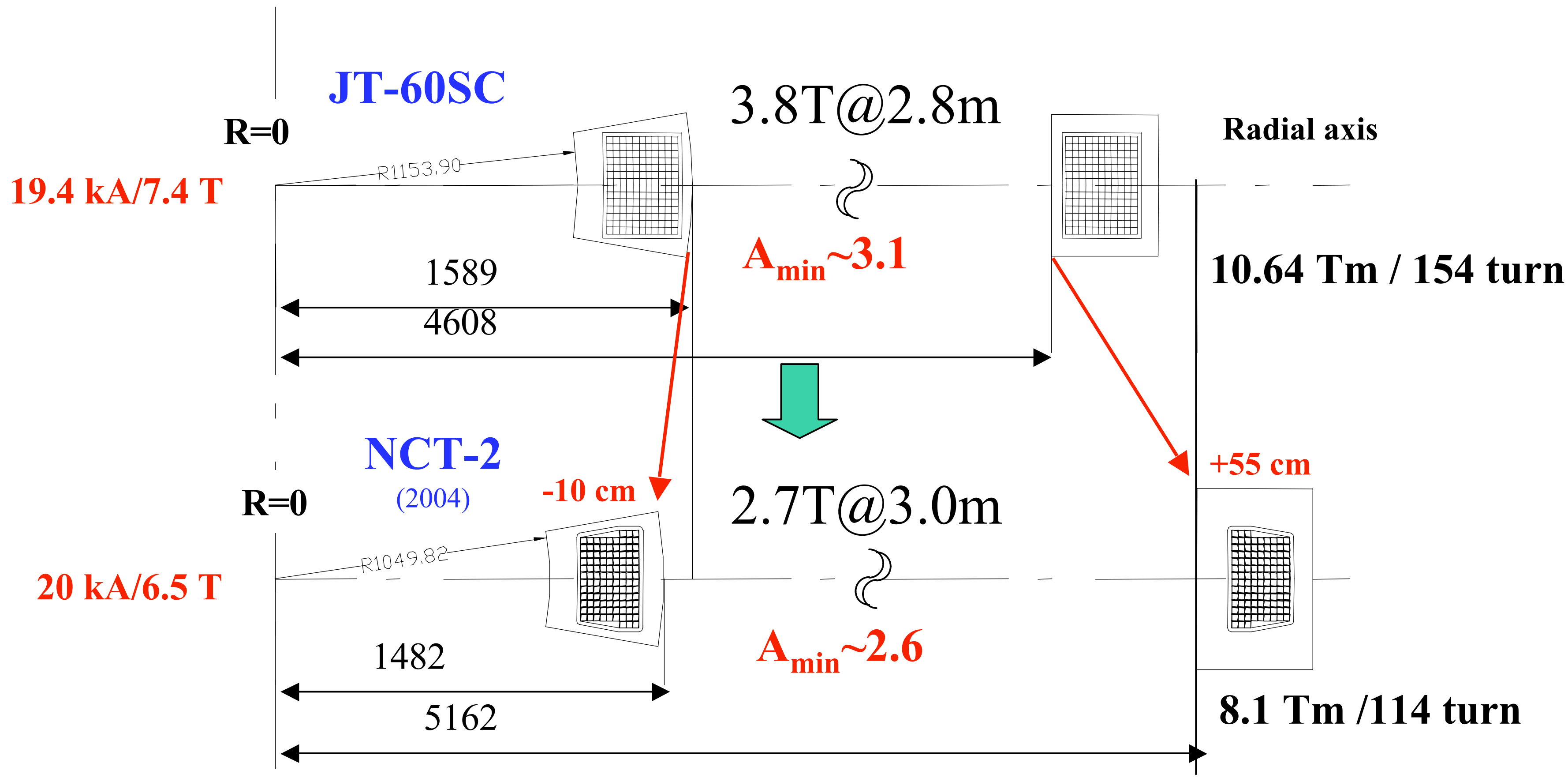
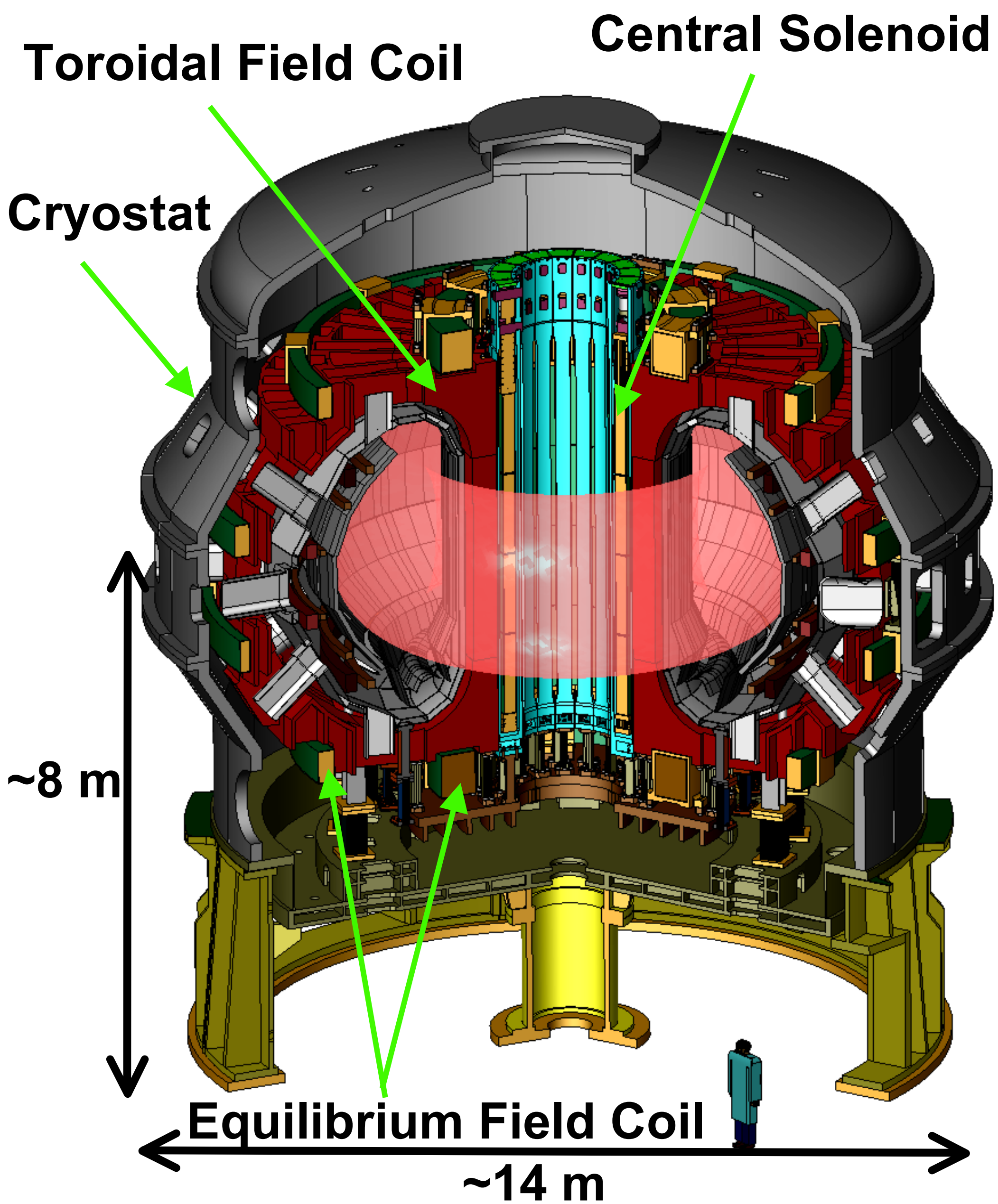


Since the Cu/Non-Cu ratio is limited to around 4 by the manufacturing technology, the developed superconductor is still optimal for NCT-2 because of lower Bmax.

[1] K. Kizu, IEEE Trans. Applied Superconductivity, 14, No.2, 1535-1538 (2003)

2. Plasma shaping capability of NCT-2 (2003)

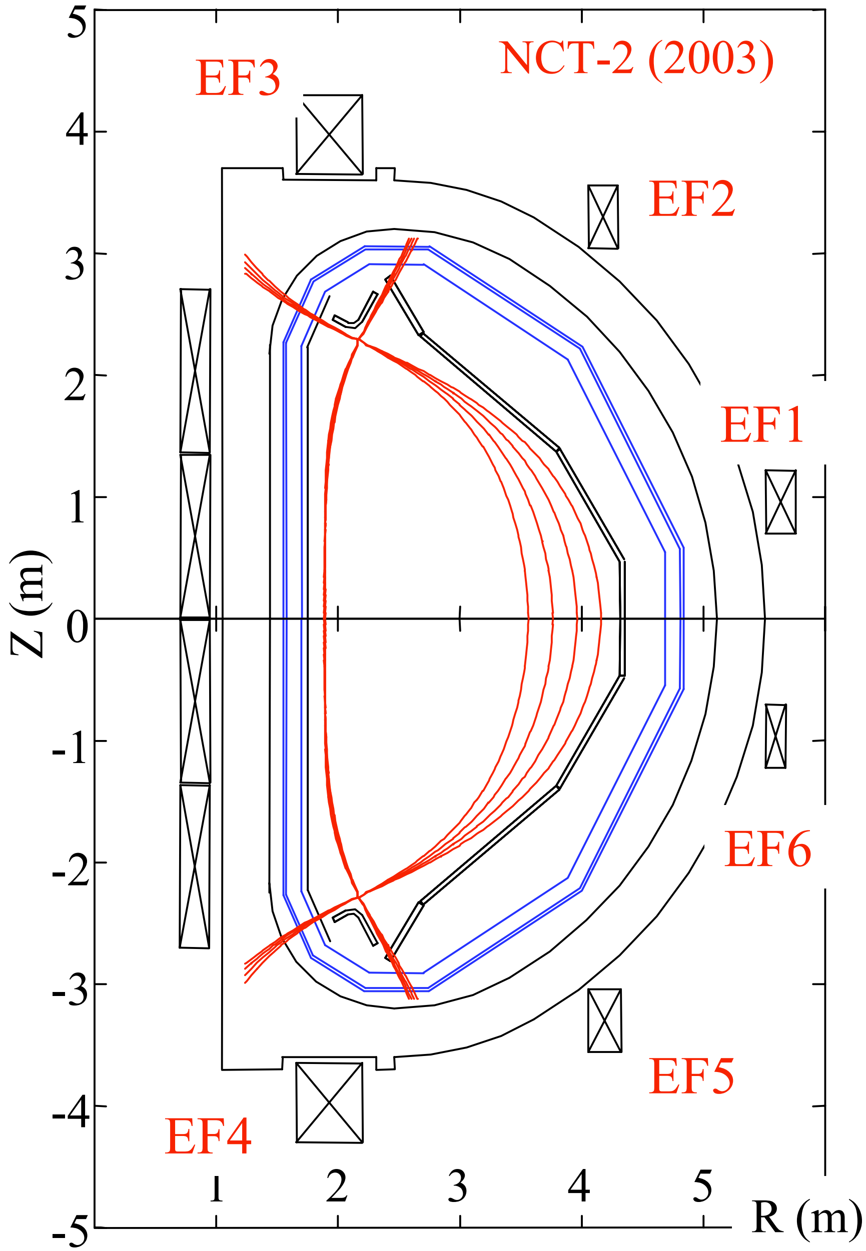
2.1 Superconducting Tokamak NCT-2 (2003)

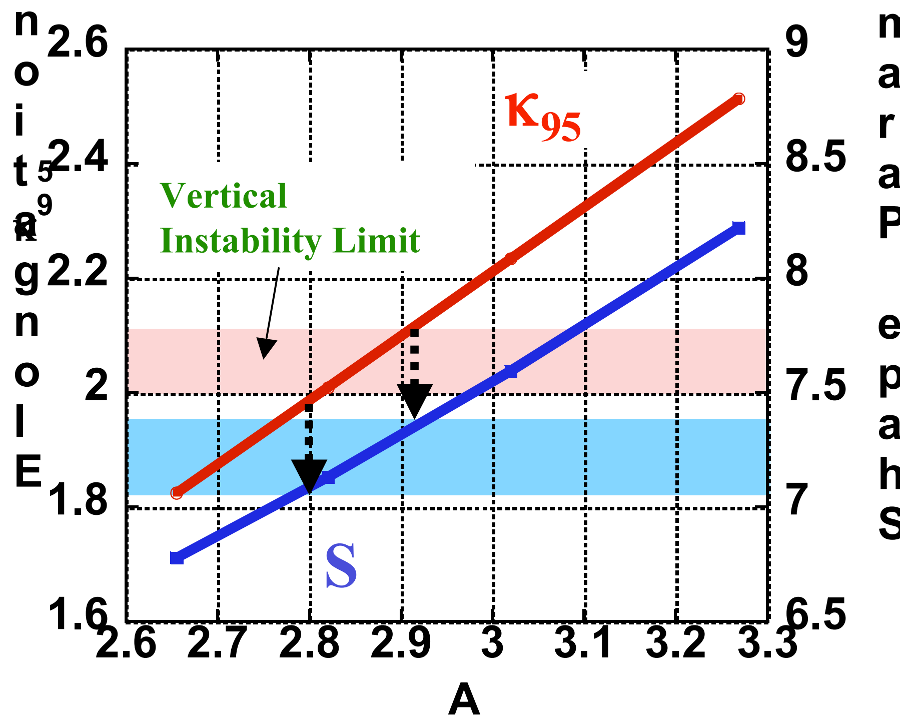


Major Parameters of NCT

Rated Plasma Current	5.5 MA
Discharge Pulse Length (Flattop)	~150 sec (100 sec)
Toroidal Field at the Plasma Center	2.7 T
Plasma Major Radius	~3.0 m
Plasma Minor Radius	~1.15 m
Aspect Ratio	~2.6

2.2 Plasma shaping capability at DN configuration in NCT-2 (2003)

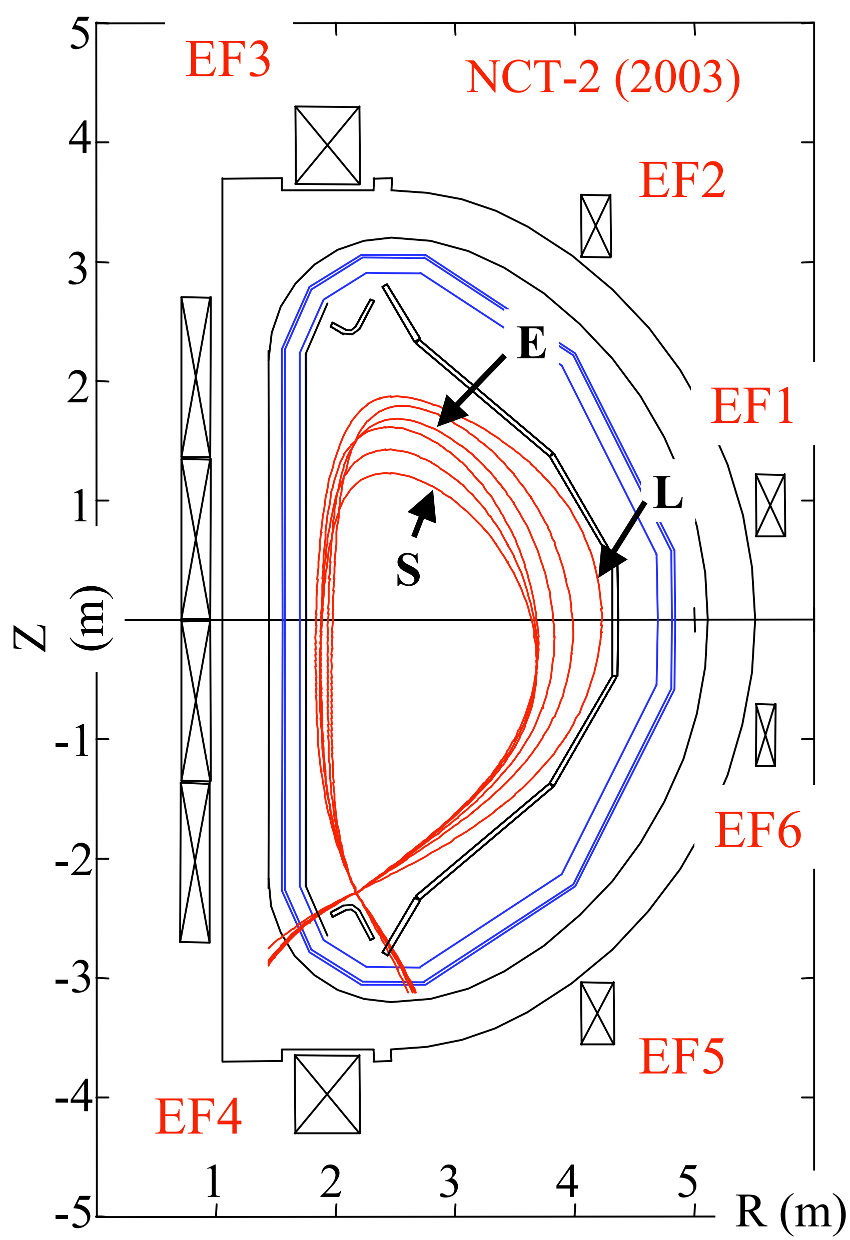


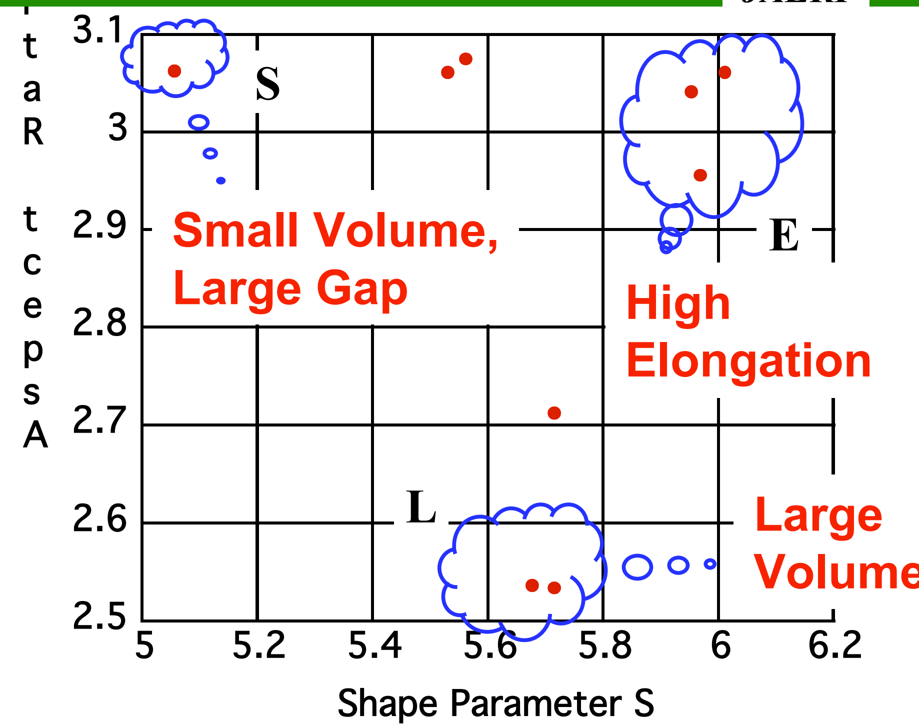


(1) Very poor shaping capability is expected in DN, because the up/down X-point fix rigidly inboard side plasma shape.

(2) Increasing of elongation κ may support to enhance plasma shape parameter S, but it will be limited to **S ~ 7.3** due to the vertical positional instability.

2.3 Plasma shaping capability at SN configuration in NCT-2 (2003)



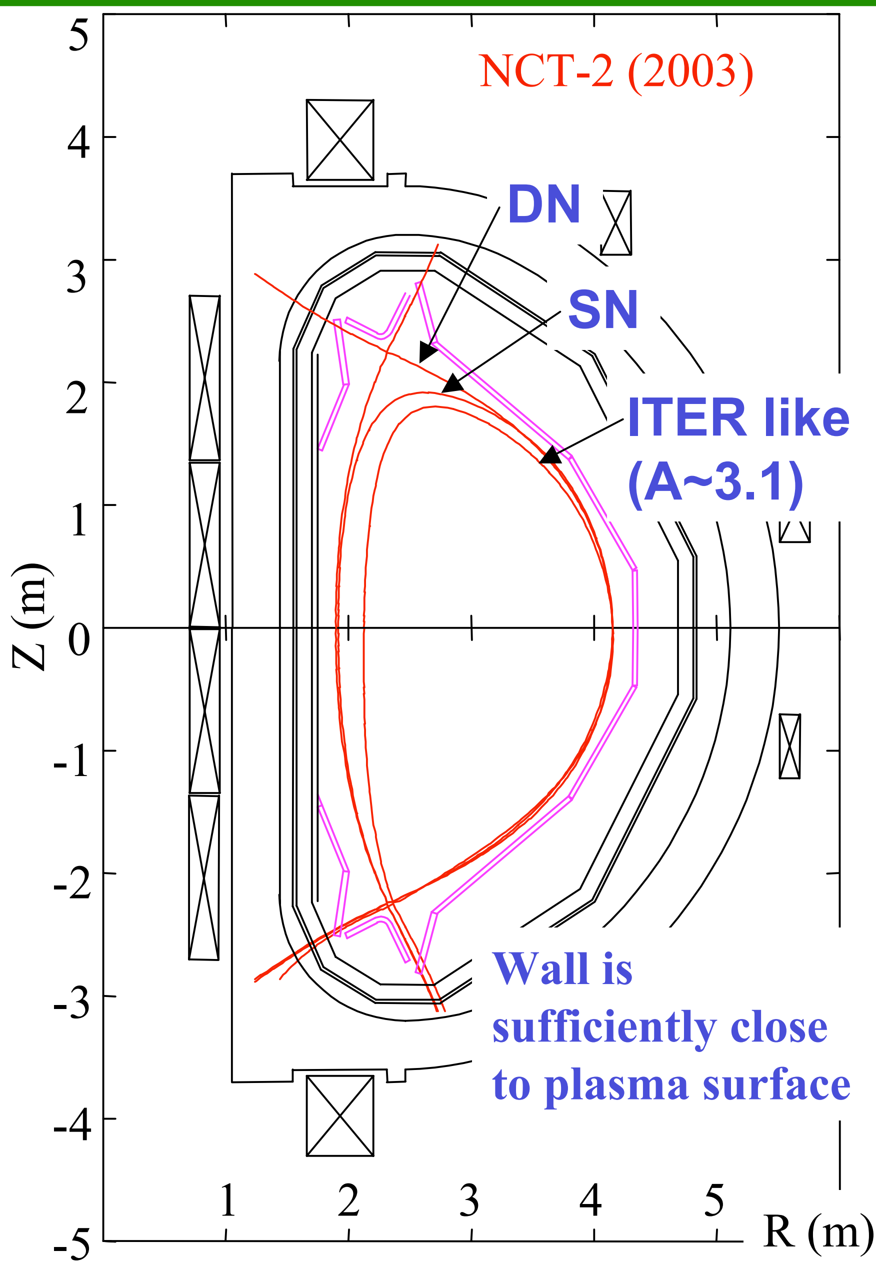


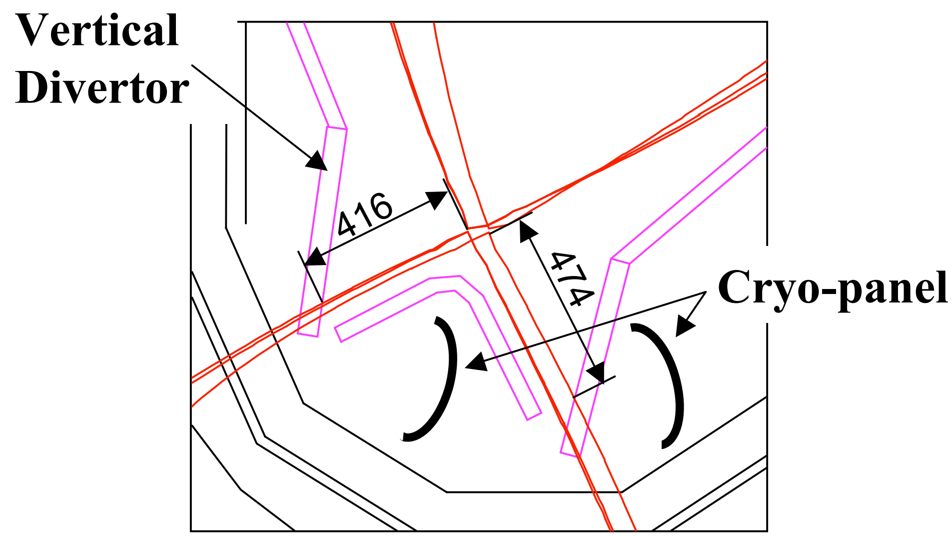
(1)The maximum shape parameter is S~6 in the SN configuration using the divertor geometry optimized for DN configuration.

(2)The S~6 is obtained even in the ITER like aspect ratio A~3.1 with highly elongated plasma (k_{95}~2).

(3) Plasma shaping flexibility is considerably limited by X-point position if plasma surface must be to close to stabilizing plate.

2.4 Moderate geometry of the divertor and stabilizing plate compatible to DN/SN and ITER like aspect ratio



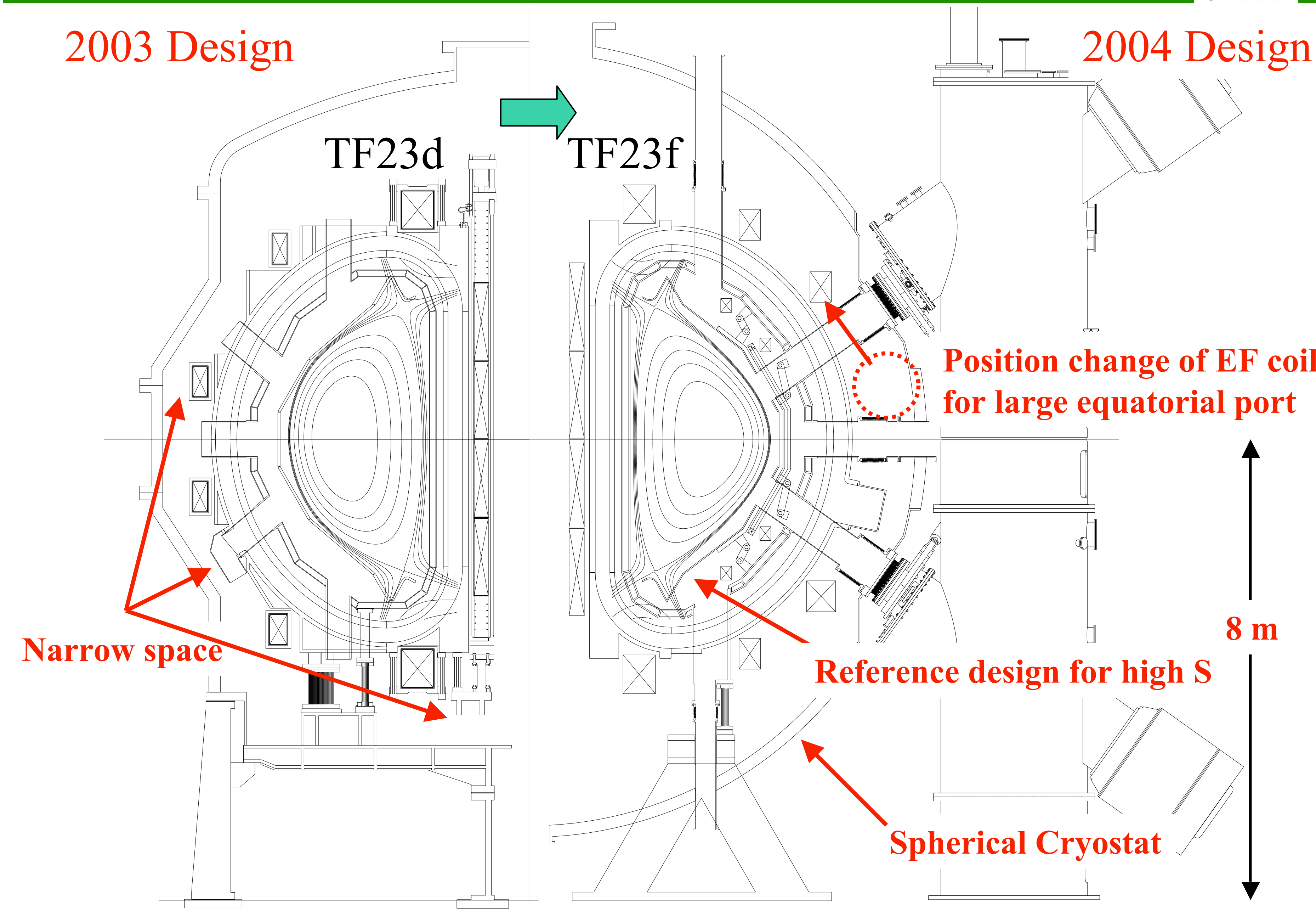


Length of divertor legs satisfied the requirements

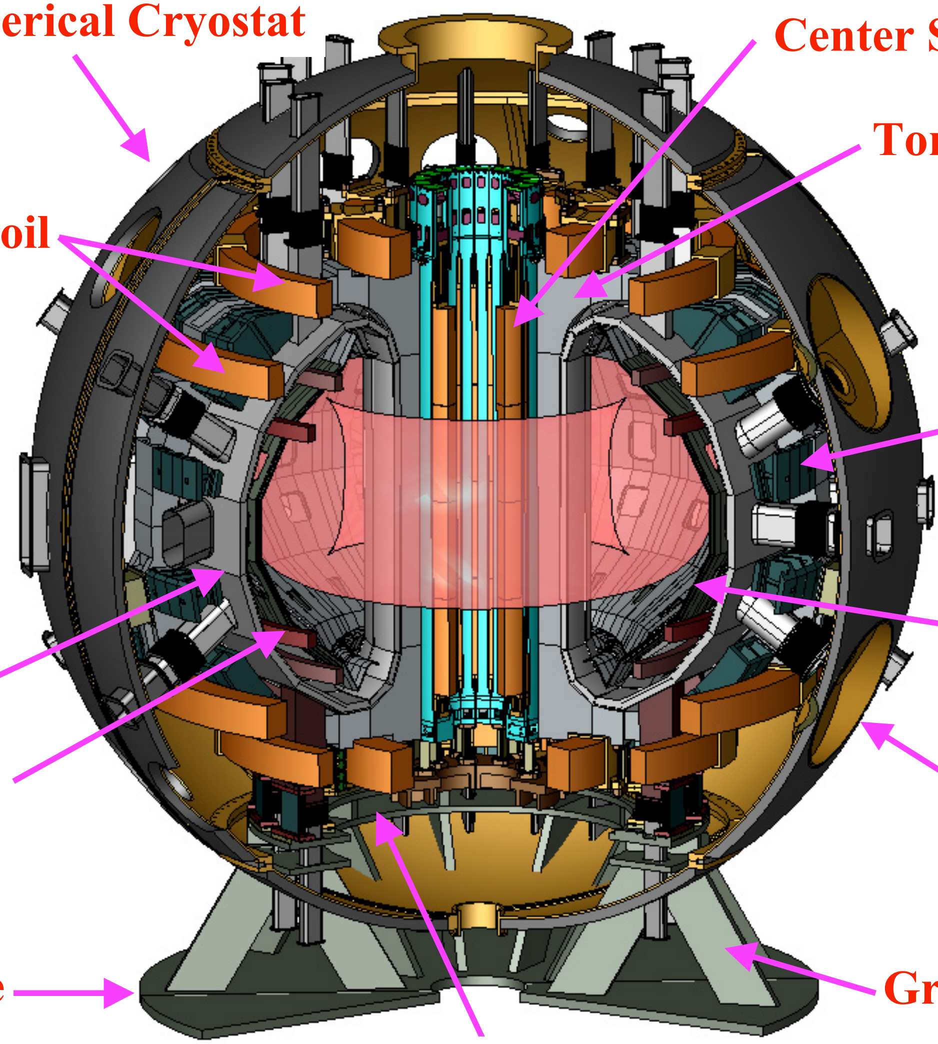
Parameter	DN	SN	ITER like
Ip (MA)	5.5	5.5	4.0
Bt (T)	2.7	2.7	2.58
κ95	1.83	1.78	1.88
δ95	0.46	0.41	0.50
q95	3.42	3.14	3.50
A	2.67	2.71	3.09
S	6.14	5.72	5.34

3. Plasma shaping capability of NCT-2 (2004)

3.1 Latest design of NCT-2



3.2 Outline of NCT-2 (2004 Design)



3.3 Comparison of the 2003 and 2004 Designs of NCT-2

Design	2003	2004	Comments
Plasma current	5.5 MA	5.5 MA	Higher triangularity of 2004 Design has advantage to increase Q_{DT} up to unit.
EF coil Ampere-turn	less	much	Large interval space between EF1,6 coils of 2004 Design caused poor efficiency to push back plasma to inboard side.
Off-axis heating of N-NBI	NG	OK	Beam line of N-NBI could be pull down up to ~1 m in the case of 2004 Design.
Maintenance space	NG	Better	Very narrow space is permitted in the case of 2003 Design. For example, the access to the near place of superconducting coil joints is impossible without a break of the cryostat.
Equatorial Ports	small	Large	Height of the port is about 3 times different

3.4 Moderate geometry of the divertor and stabilizing plate compatible to DN/SN and ITER like shape

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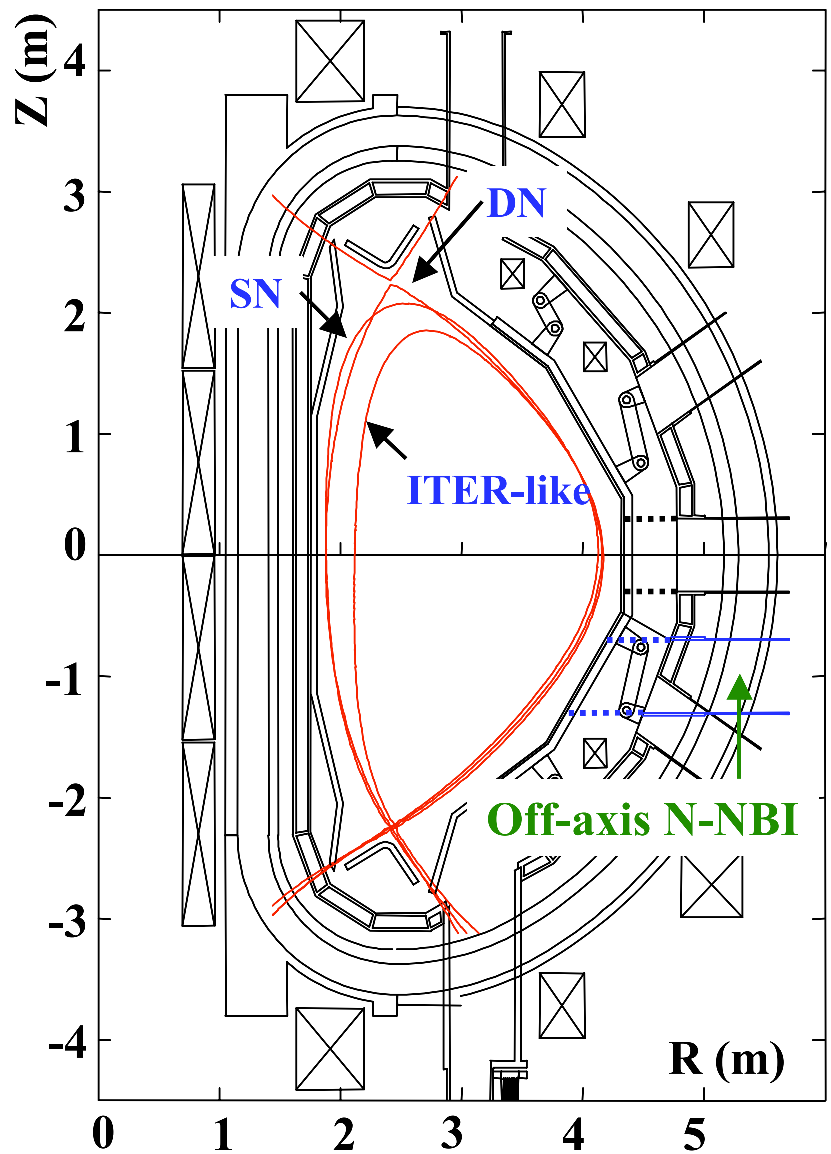

Parameter	DN	SN	ITER like
Ip (MA)	5.5	5.5	4.0
Bt (T)	2.7	2.7	2.58
κ95	1.81	1.82	1.89
δ95	0.44	0.42	0.47
q95	3.22	3.24	3.49
A	2.65	2.67	3.06
S	5.78	5.86	5.28

1. The radial position of X-point was shifted to outboard side by 10 cm, because the plasma surface near the equatorial plane naturally expanded significantly.
2. Off-axis Heating by N-NBI is possible for 3MA fully non-inductive current drive.
3. High triangularity plasma shape is easily obtained, but low triangularity is difficult.

3.5 Plasma curvature control at the low field side with help of extra two small EF coils (Rectangularity Control)

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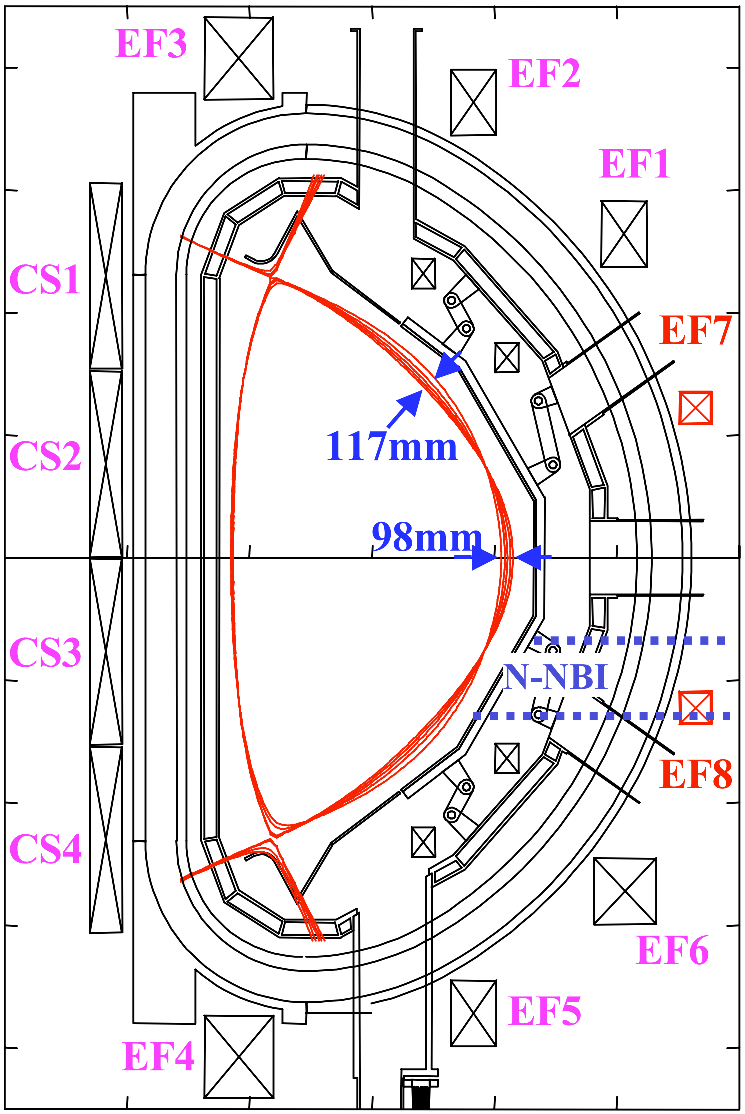

1. The radial position of plasma outermost surface at the equatorial plane can be controlled by the range of ~10cm with help of two extra EF coil **EF7** and **EF8**.

2. It should support to enhance the plasma shaping capability.

3. The preparation of required new DC power supplies and feeders is not so easy due to the lack of space.

4. The EF coil connection change at the out of cryostat may be one of the possible solution.

5. The new **EF8** coil just below the equatorial port conflicts with the off-axis injection of the negative ion-based NBI.

3.6 ITER Plasma Simulation using NCT-2 of 2004 Design

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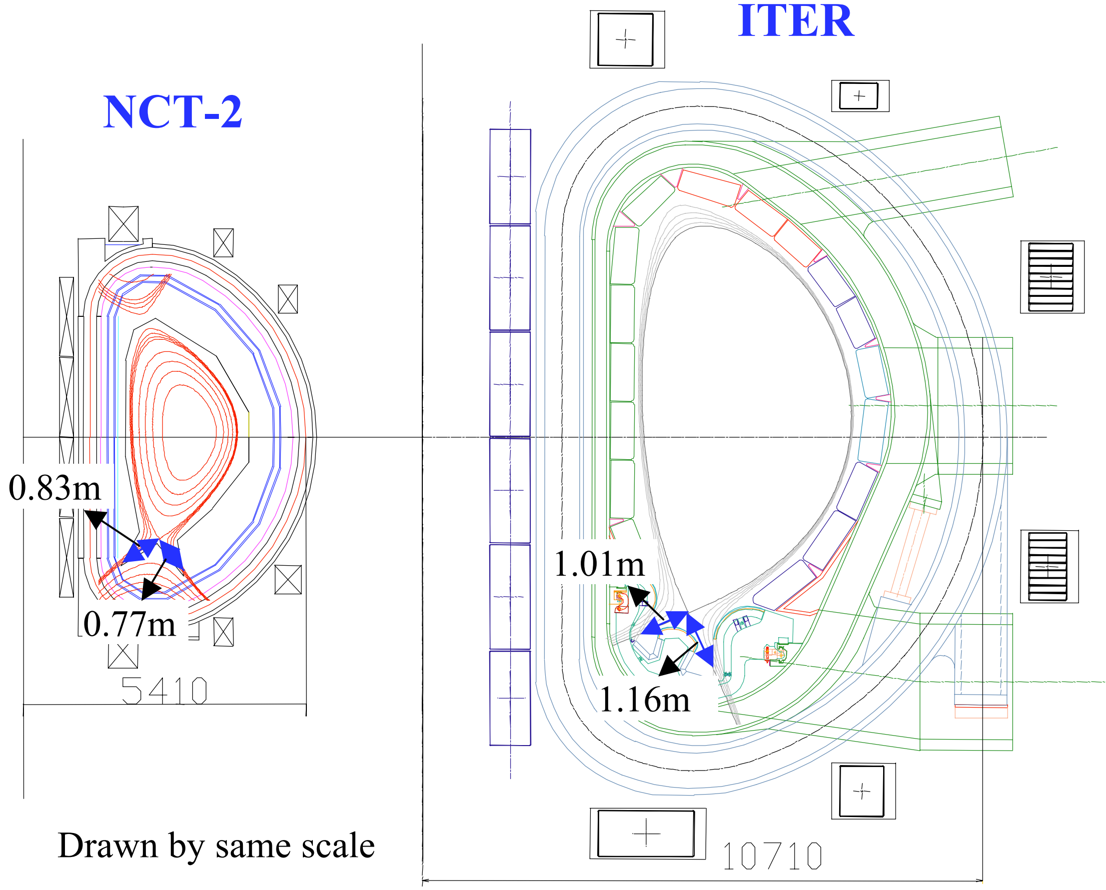
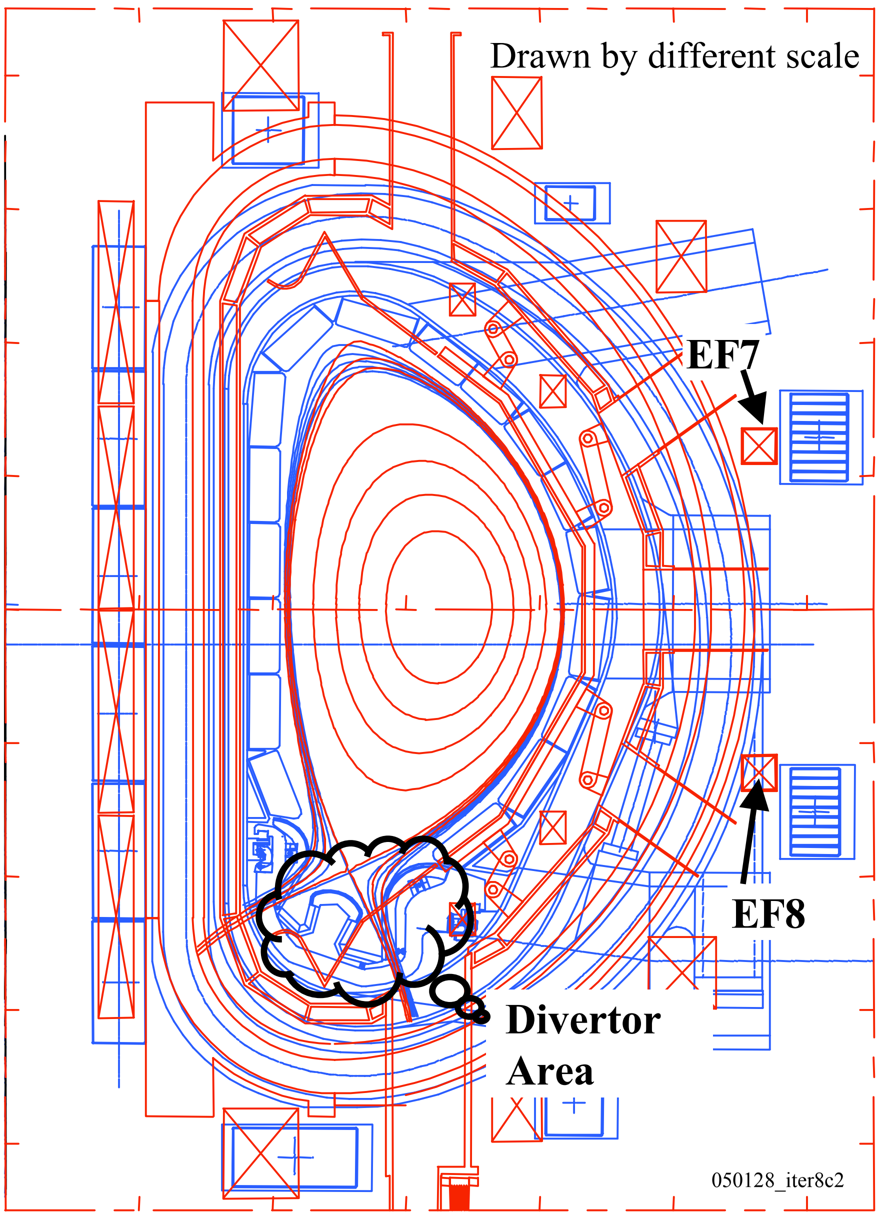


Parameter	NCT-2	ITER
Ip (MA)	4	15
Rp (m)	3.1	6.2
a (m)	1.0	2.0
Aspect Ratio A	3.1	3.1
q_{95}	3.0	3.0
κ_{95}	1.83	1.70
δ_{95}	0.43	0.33
Magnetic Shear	NS	PS

NCT-2 can produce a plasma which has the just same aspect ratio of 3.1, safety factor of 3.0 with that of ITER. A very nice divertor performance could be realized with very long divertor legs [1], but their elongation, triangularity are significantly different from that of ITER standard plasma.

[1] H. Kawashima et al., to be presented at ISFNT-7 at Tokyo on coming May in 2005.

3.7 ITER like configuration in NCT-2 with 8 EF coils

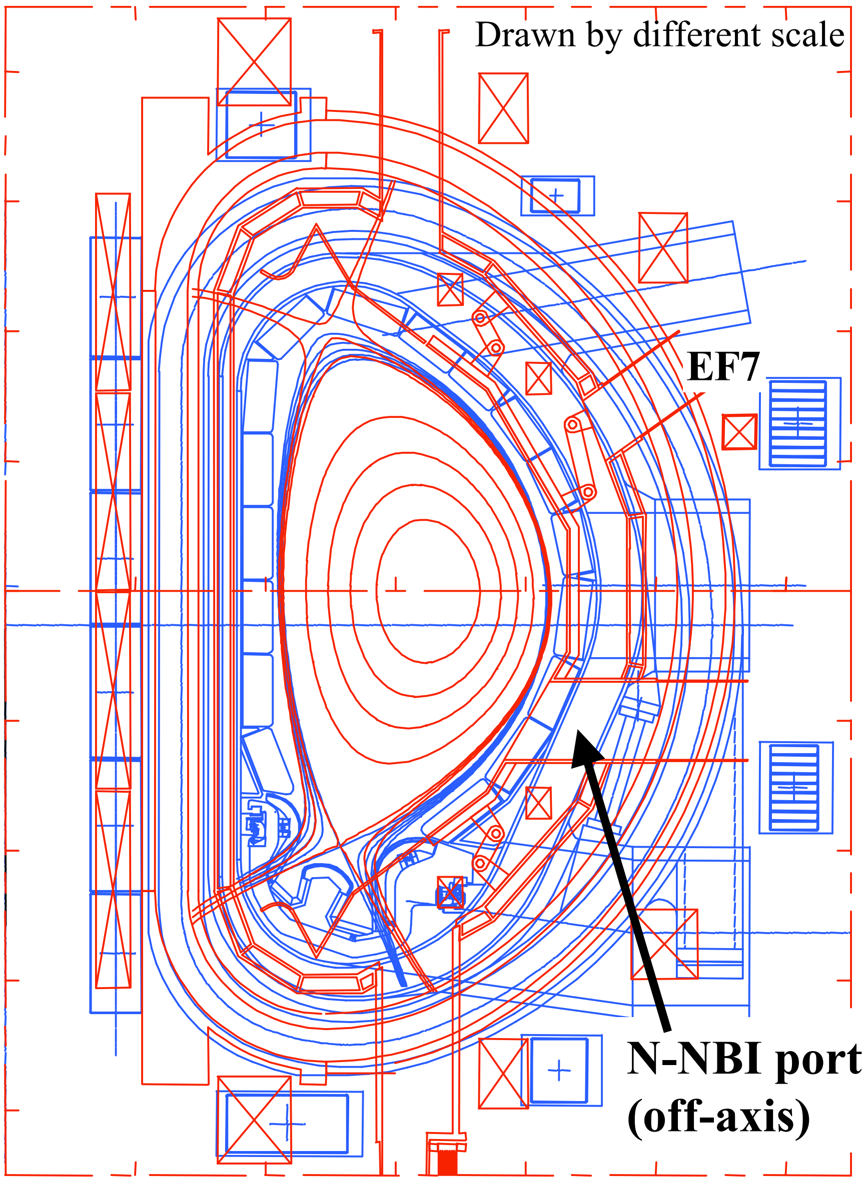


Overlaid plot of ITER and NCT-2 with 8 EF coils

Parameter	NCT-2	ITER
Ip (MA)/Bt (T)	3.4 / 2.59	15 / 5
Rp / ap (m)	3.13 / 1.03	6.2 / 1.0
κ_{95} / δ_{95}	1.68 / 0.30	1.70 / 0.33
q_{95} / S	3.0 / 3.9	3.0 / 4.25

1. Plasma shape just close to ITER can be produced using eight EF coils.
2. Replacing only divertor module enables to simulate ITER plasmas.
3. Off-axis heating is impossible due to the presence of EF8 coil.

3.8 ITER like configuration in NCT-2 with 7 EF coils



Overlaid plot of ITER and NCT-2 with 7 EF coils

1. Plasma shape parameter similar to ITER can be obtained also using 7 EF coils.
2. Curvature of the separatrix may change slightly due to the lack of EF8 coil.
3. Off-axis heating by N-NBI is possible.
4. Since the SOL width may not be proportional to plasma size, the divertor geometry of NCT will be different from that of ITER.
5. The NCT-2 of 2003 Design may be suitable for ITER plasma simulation.

4. Modification design of power supplies

4.1 Common configuration of the DC power supplies

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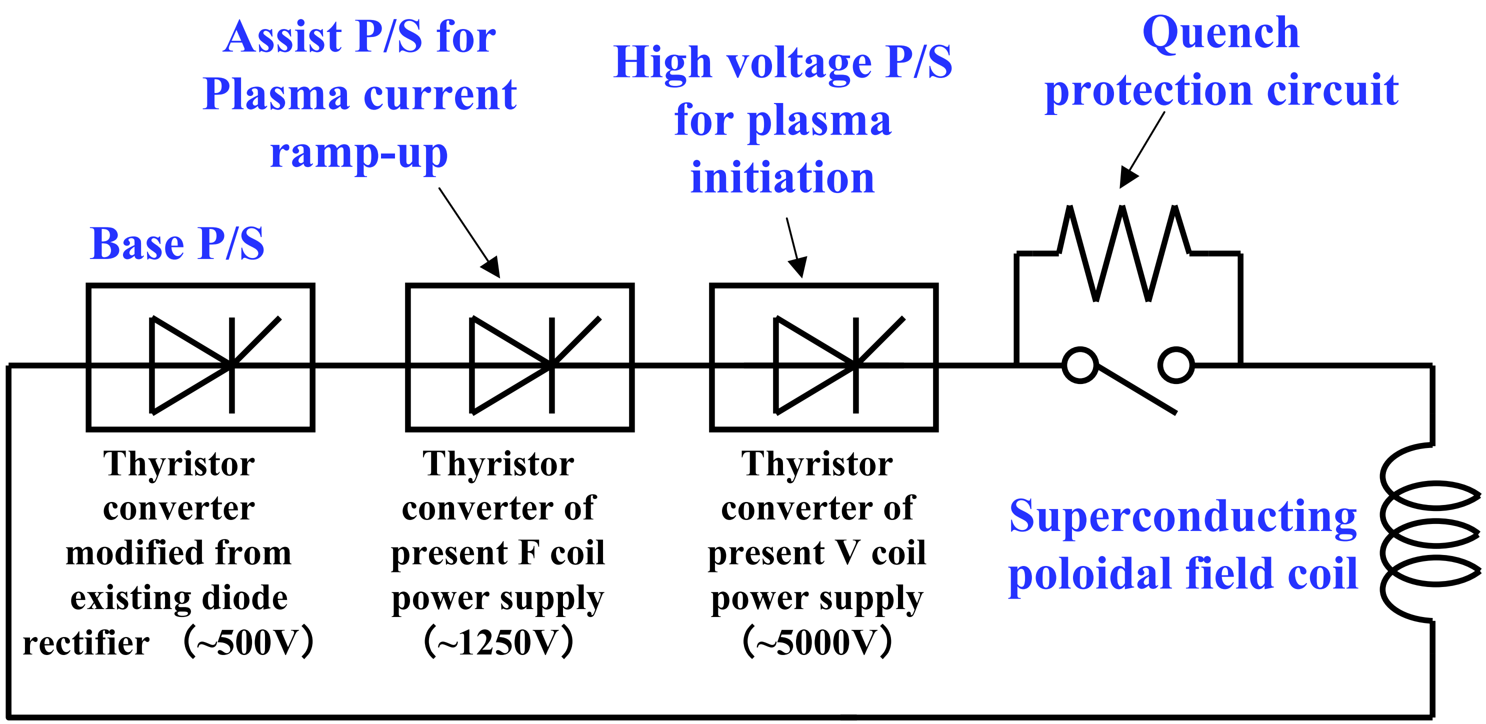

1. Almost DC power supplies are reuse and/or modified converters from the existing power supply system of JT-60U.

2. The base power supply and the quench protection circuit will be installed to every DC power supplies, but auxiliary power supplies for plasma initiation and plasma current ramp-up will be installed according to their operational pattern.

4.2 Configuration of Base Power Supply

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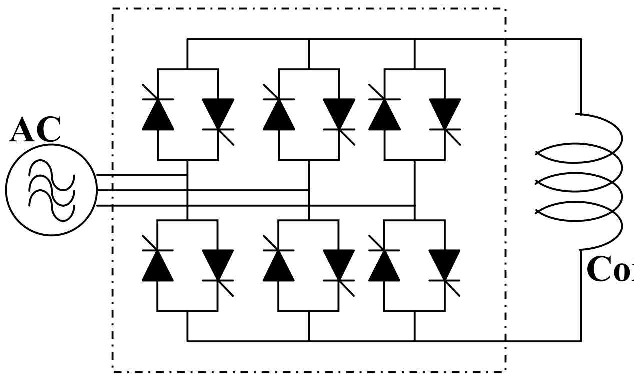
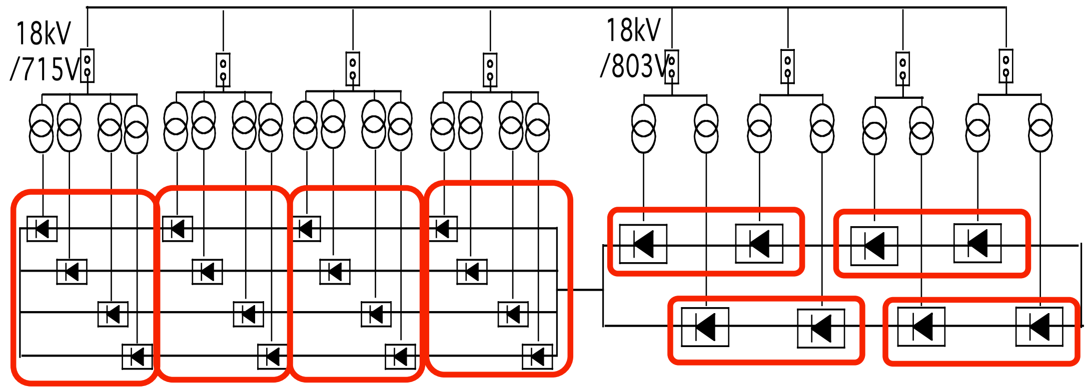
Bi-directional Unit Converter



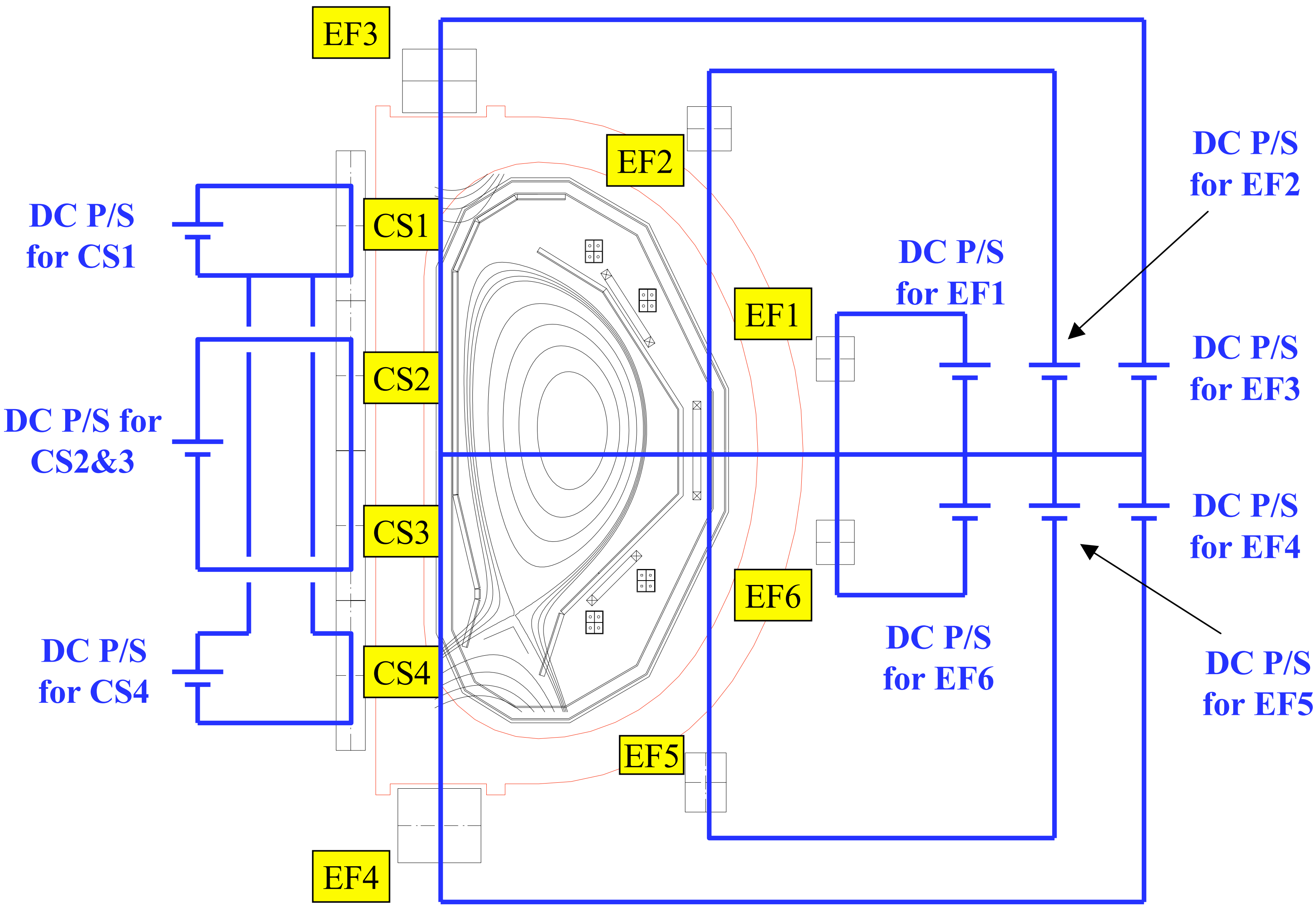


Present connection of JT-60 Toroidal Field Coil Power Supply (6.5kV-52.1kA)

1. **Four quadrant operation**
2. **Parallel connection is assumed for smooth change of output current polarity.**

1. **The existing 24 diode rectifier will be re-arranged to 8 thyristor converter blocks of ~700V/±20kA by replacing the semiconductor power devices.**
2. **The parallel operation is the key to decrease the heat loss of transformer, bus-bar and etc. (20kA-250s)**

4.3 Connection of DC power supplies and PF coils



Hybrid connection between PF coils and their DC power supplies are adopted to minimize the number of feeders.

4.4 Outline of the whole power supply system for NCT

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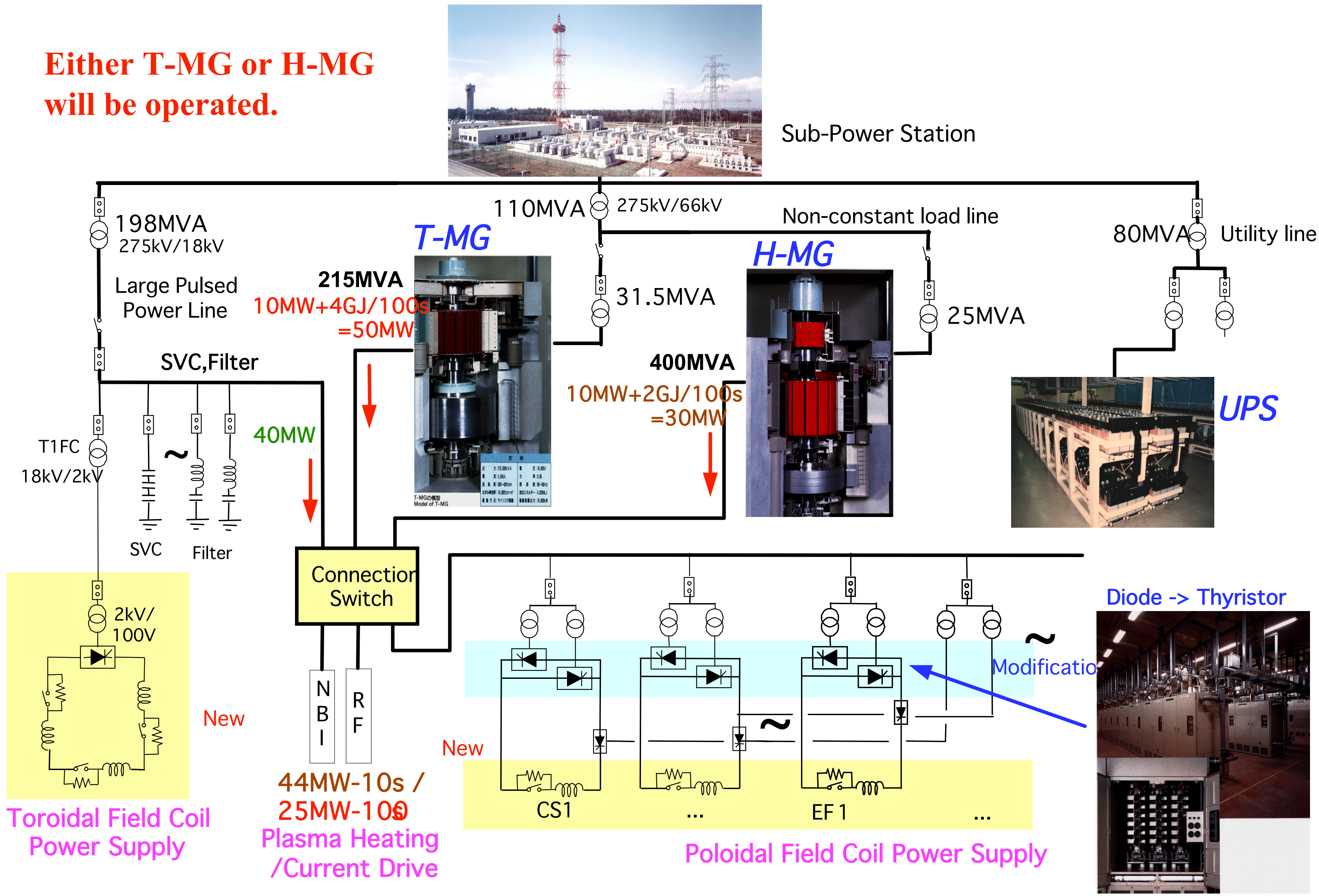
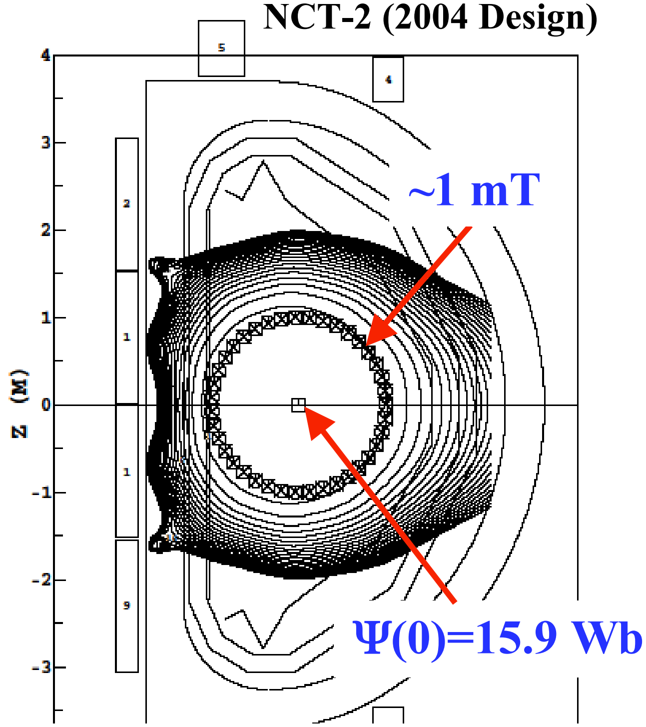
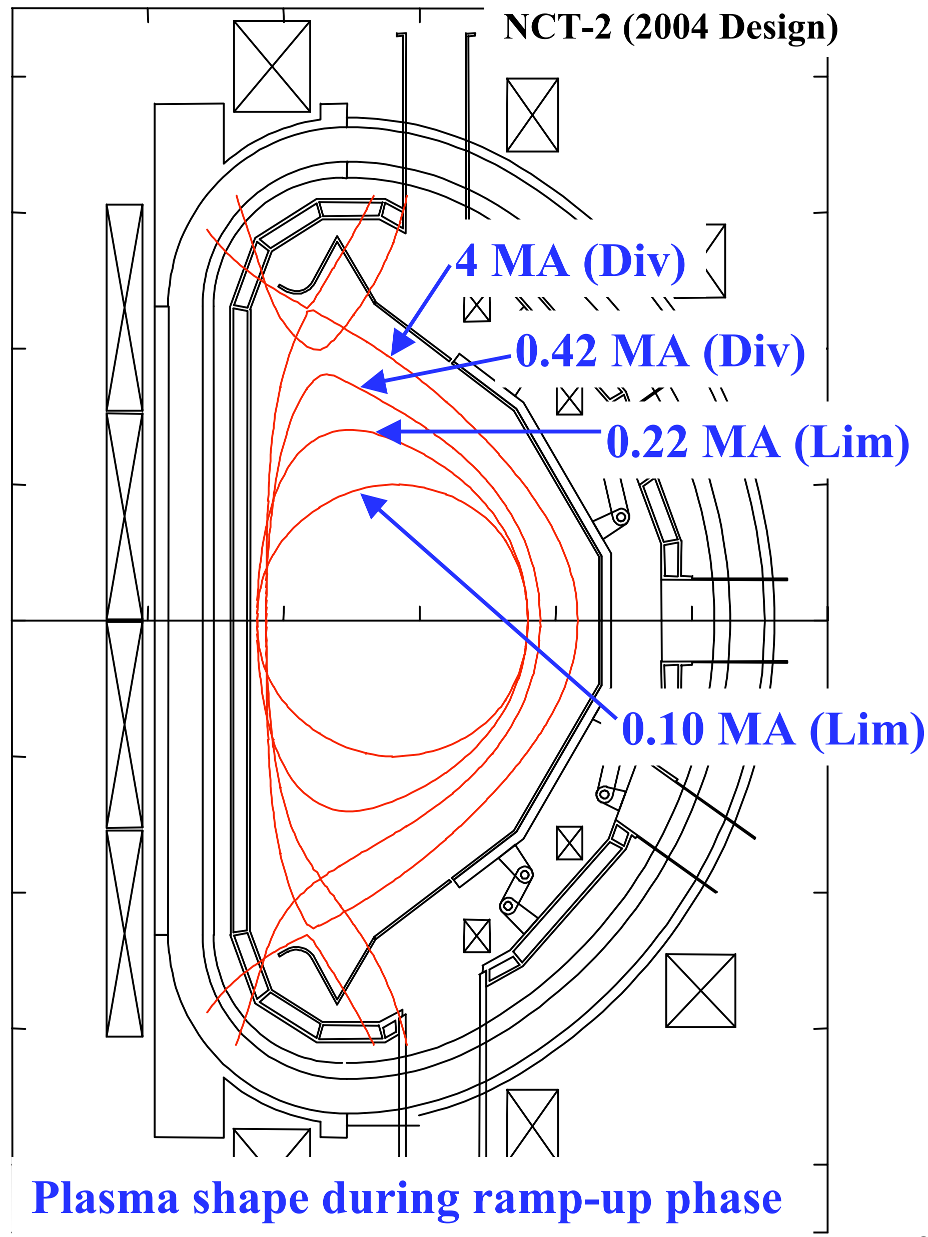


4.5 Plasma Current ramp-up scenario of NCT



Error field strength at Plasma initiation

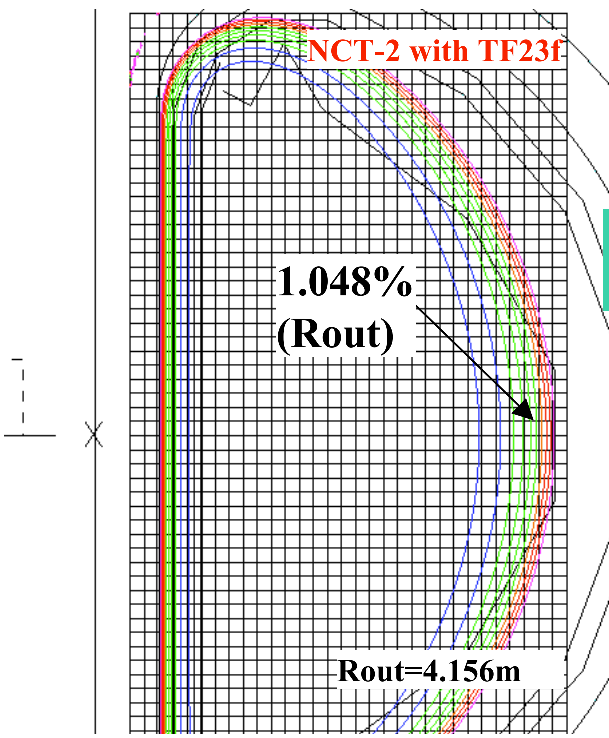


Plasma shape during ramp-up phase

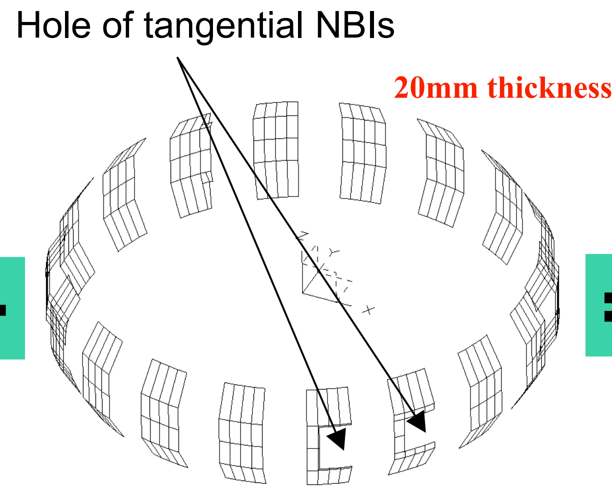
The plasma current ramp-up rate of 0.4 MA/s necessary for the negative shear plasma can be achieved using the originally reserved power supplies, and the early divertor formation is possible at Ip~0.4 MA.

5. Other important issues in the NCT design

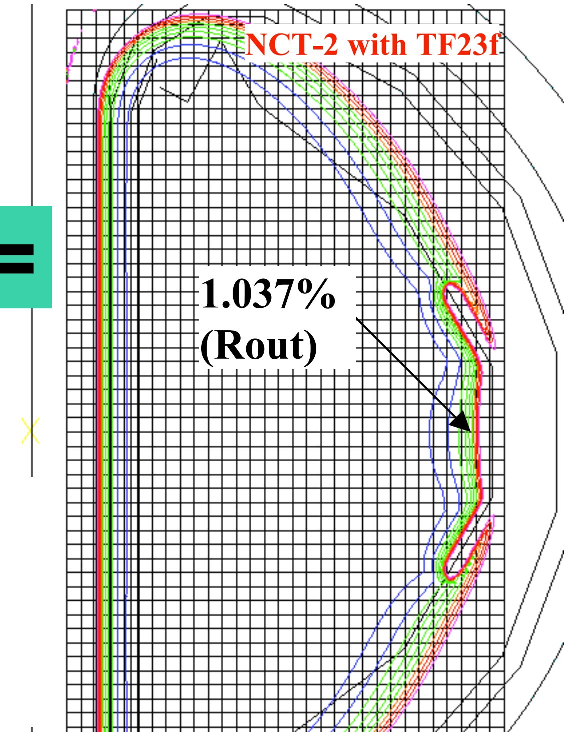
5.1 Estimated toroidal field ripple and the reduction of fast ion losses using ferritic steel F82H



Toroidal field ripple without compensation



Arrangement of ferritic steel F82H attached on the stabilizing plates for ripple compensation



Toroidal field ripple with compensation

Results of OFMC code

Compensation	Without	With
Fast Ion Losses	5.06%	2.42%

Ferritic steel is effective for reducing fast ion losses in the case of rated toroidal filed operation.

5.2 Latest design of in-vessel coil for RWM stabilization

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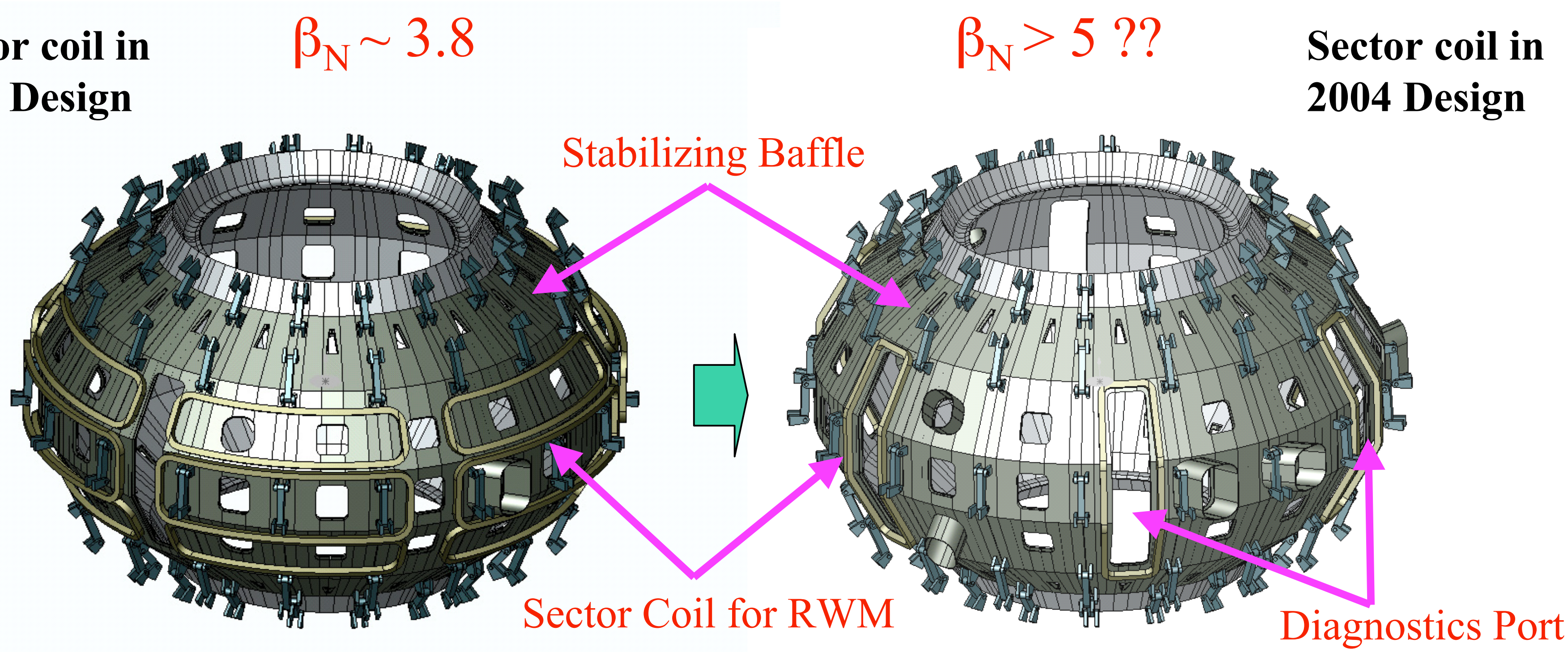
Sector coil in 2003 Design

Sector coil in 2004 Design



Achievable ultimate beta-value calculated by Valen-code was limited to β_N ~3.8 due to the magnetic field shielding effect of stabilizing plates [1]. In the latest design, the in-vessel sector coil is planned to be attached on the large port of stabilizing baffle plate to improve the response time.

[1] G. Kurita et al., "Critical b analyses with ferromagnetic and plasma rotation effects and wall geometry for a high β steady state tokamak" IAEA 2004, FT/P7-7

6. Conclusion

1. New superconductors which makes the magnet system compact are successfully developed.
2. NCT-2 can produce wide aspect ratio plasmas in the range of $2.6<A<3.4$, but the rectangularity control using extra two EF coils is a trade-off with off-axis heating of N-NBI.
3. For the ITER plasma simulation, at least one extra EF coil will be necessary if the similar plasma shape were required in NCT to support ITER.
4. Many of coil power supplies will be prepared by reuse and modification of the present JT-60 facilities.
5. Ferritic steel is not absolutely necessary for toroidal field ripple compensation (but it will be used for compatibility test with high-performance plasmas).
6. Design of in-vessel coil for RWM suppression will be continued.