COMPARISON OF OPTIONS FOR A PILOT PLANT FUSION NUCLEAR MISSION

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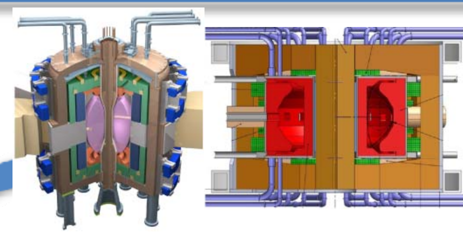
Outline

- What is a Pilot Plant (PP)?
- What are the candidate options being studied?
- What is the pilot plant mission, design goals and basic requirements?
- What are the basic design details of each option?
- What comparisons can be made between the candidate PP options?

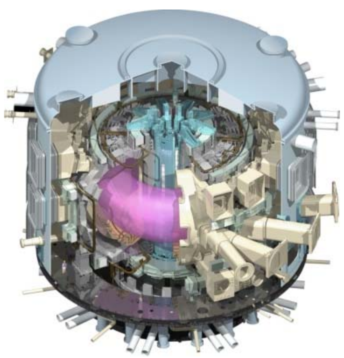
Pathways to Fusion Power



Component Test Facility

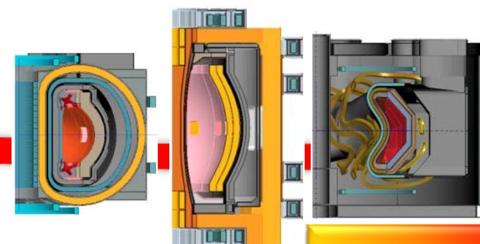
Blanket R&D, T self-sufficiency

ITER



Supporting Physics and Technology

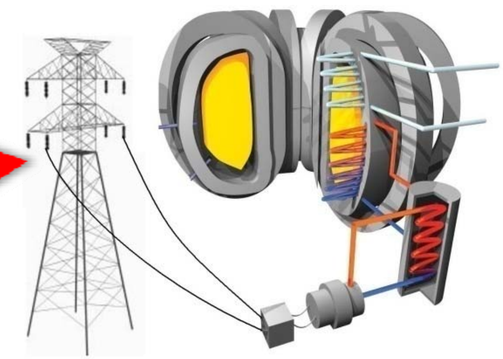
- Core Physics
- Materials R&D
- Plasma Material Interface



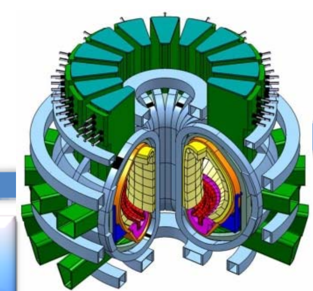
Pilot Plant

CTF + Power plant like maintenance, $Q_{eng} \geq 1$

Practical Power Plant



2-Stage Demo



CTF + Power plant like maintenance, $Q_{eng} \sim 5$

Three PP options were studied

- **Spherical Tokamak (ST)**
  - Copper TF Coils,
  - Potential for simplified maintenance, reduced cost
- **Advanced Tokamak (AT)**
  - Most mature confinement physics, technology
- **Compact Stellarator (CS)**
  - Low re-circulating power, low/no disruptions

The Pilot Plant Mission

Assess the feasibility of integrating key science and technology capabilities of a fusion power plant in a reduced device size

Targeted capabilities:

- Fusion Nuclear Science research, Component Testing
  - Steady-state plasma operating scenarios
  - Neutron wall loading ≥ 1MW/m^2
  - Tritium self-sufficiency
- Applicable power plant maintenance scheme
  - Capable of fast replacement of in-vessel components
- Small net electricity production
  - Bridge the gap between ITER and first-of-a-kind power plan

Major component features

Components requirements set to meet power plant mission

- S/C magnets sized for reduced cycles of steady state operations
  - No advantage taken for improvements in quench protection or conductor grading
- Copper ST magnets sized for Q_{eng}~1 operation
- A blanket strategy involved operating with “low tech” DCLL blankets in de-rated mode with planned upgrade to advanced blankets based on test results.
- Diagnostic installation set by measurements control and evaluation function only

PP configuration definition

A major PP design goal was to define configuration arrangements that could achieve high availability

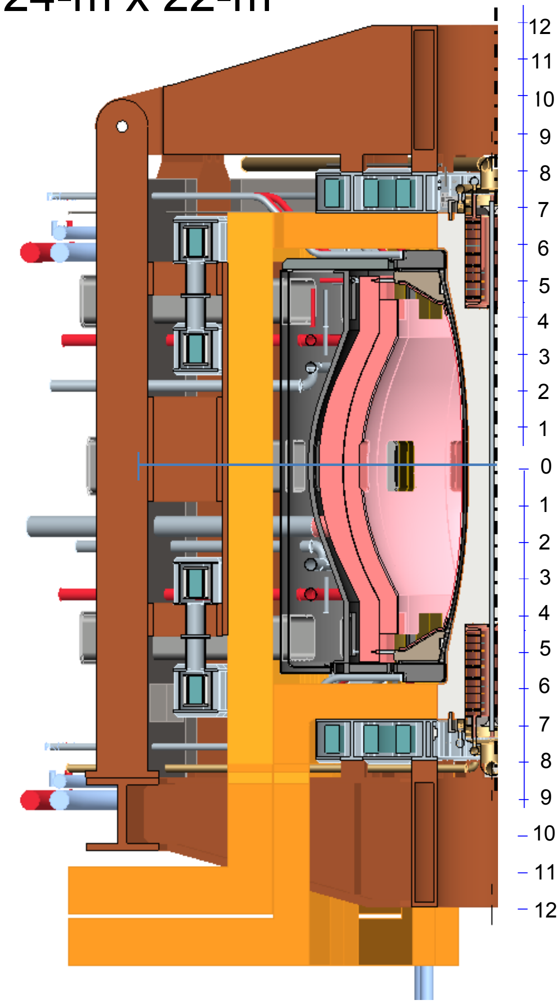
Reviewed passed community studies:

- ST: Followed vertical maintenance approach of past ST studies with added design variations
- AT: ARIES-AT, EU studies, JAEA DEMO and other S/C defined high availability solutions
- CS: The configuration was developed around the ARIES-CS design, with improved maintenance features

Relative Size Comparison

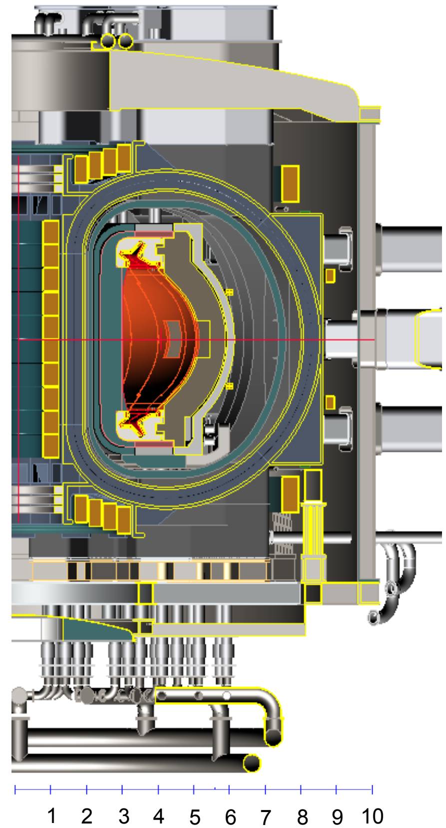
overall height x outside diameter

24-m x 22-m



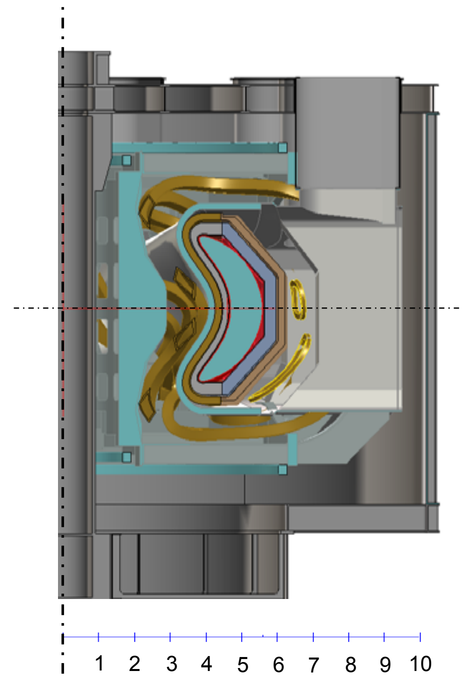
ST – 2.2 m R_0,

17-m x 20-m



AT – 4 m R_0,

15-m x 22-m



CS – 4.75 m R_0,

A brief review of the design details of each pilot plant option

ST option design details

External support structure

S/C external PF coils housed in a vacuum enclosures

Contoured blanket / shield system

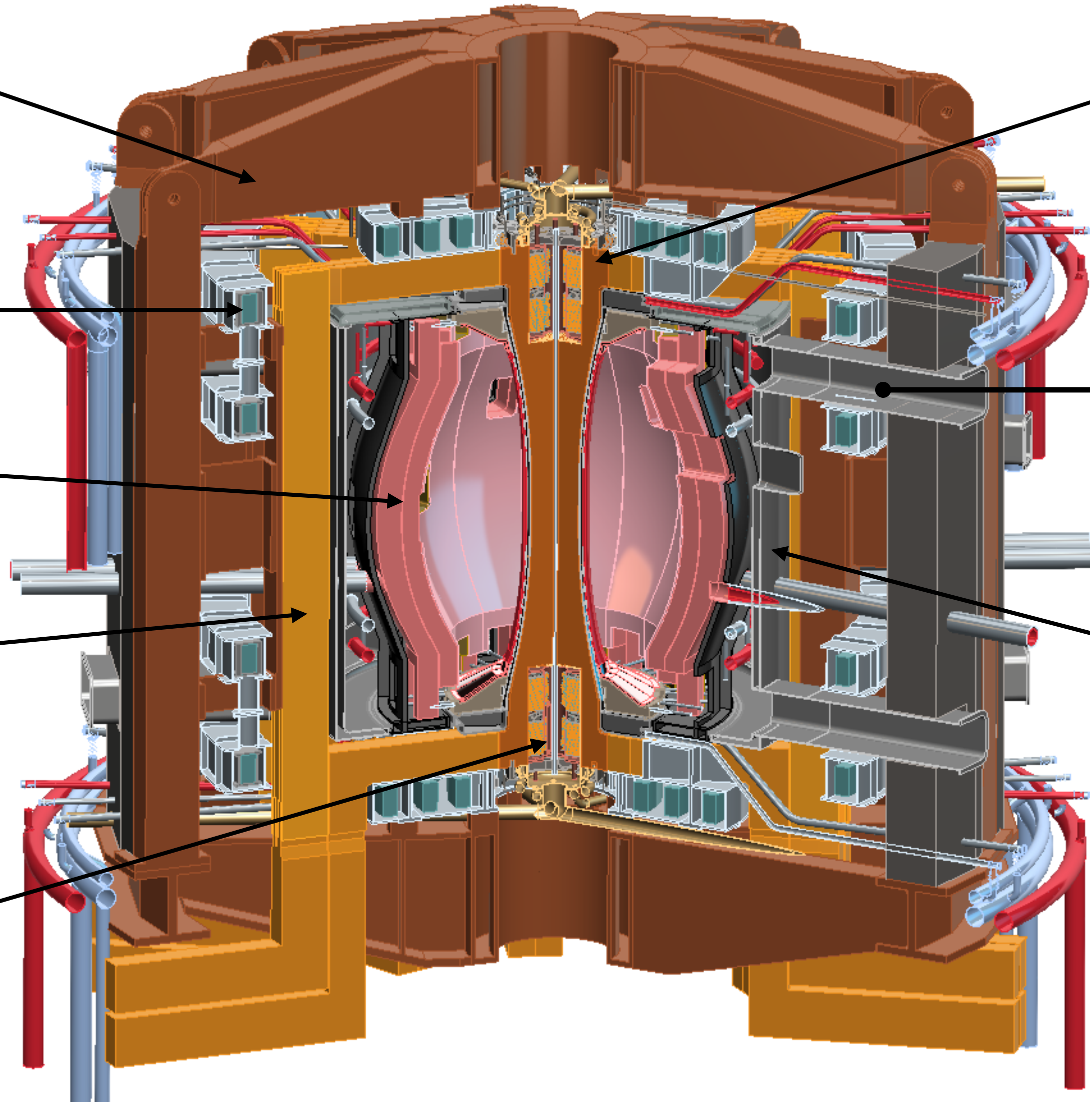
Copper TF coil system with discrete legs

Copper Bitter coils embedded within the TF centerpost

Felt metal sliding joint

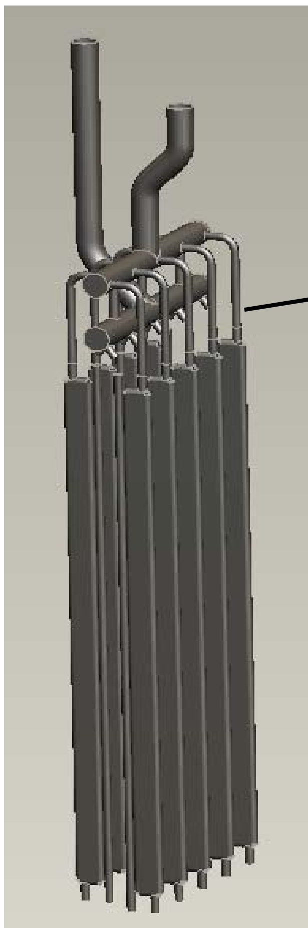
Horizontal replacement of divertor segment

VV located inside the TF

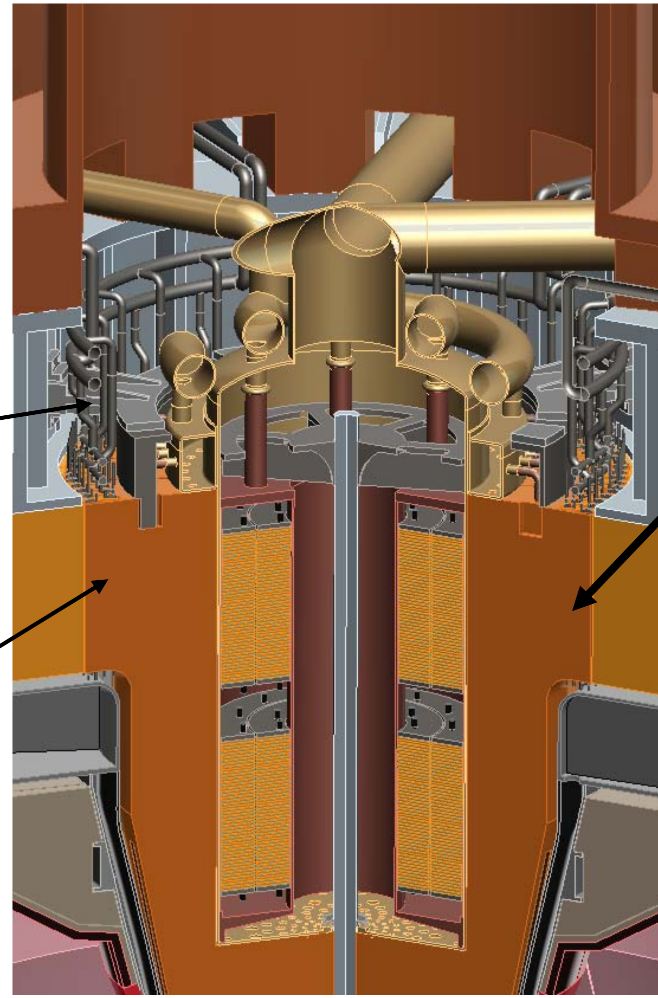


ST centerstack region

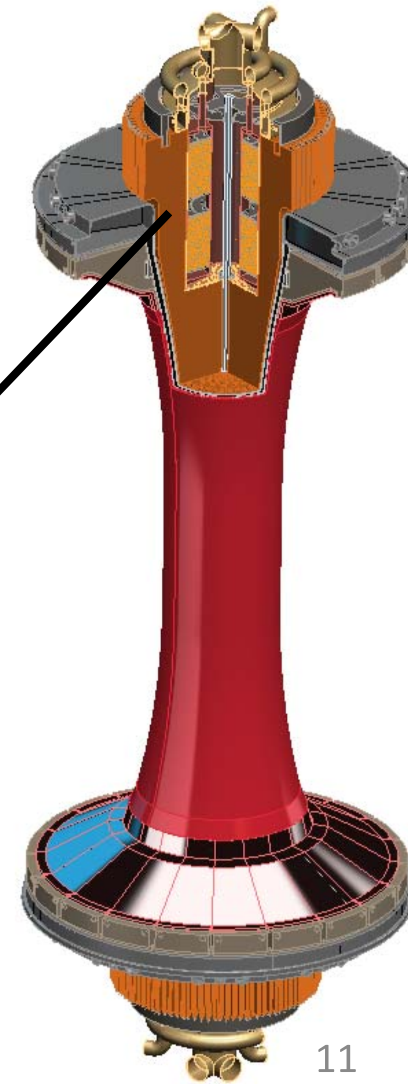
Felt metal
pressurized bladder
assembly

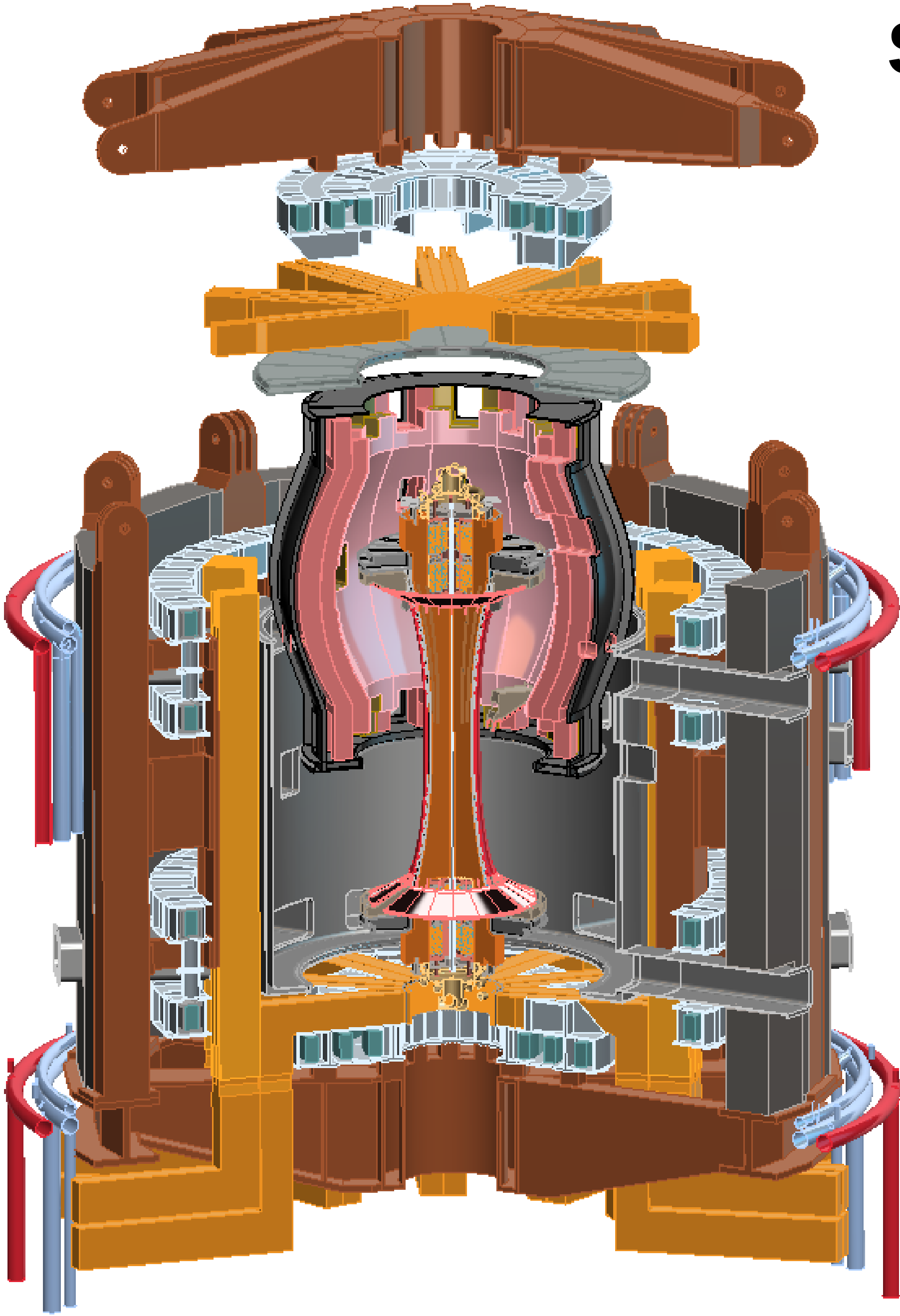


Felt metal
sliding joint
region



centerstack

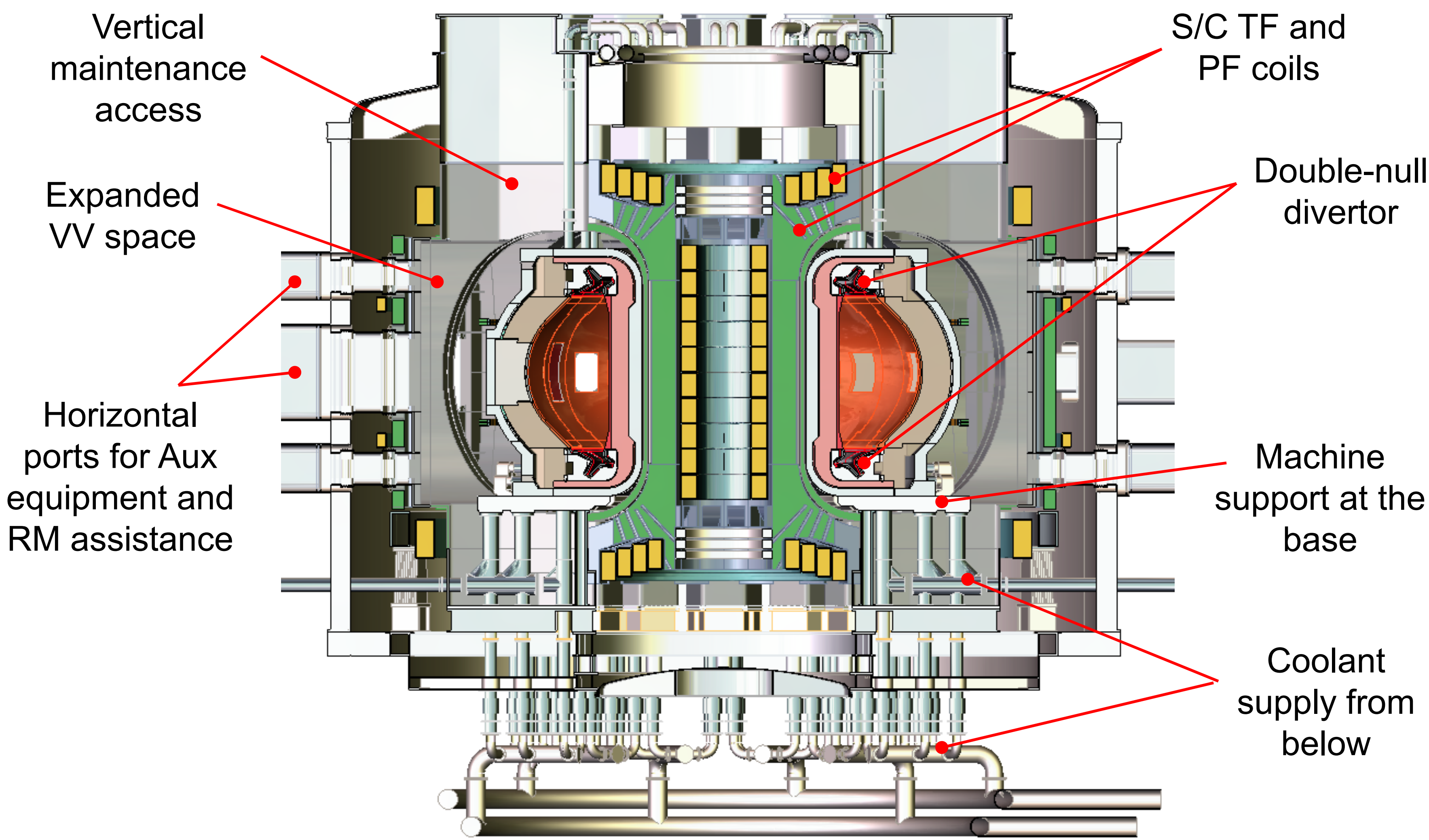




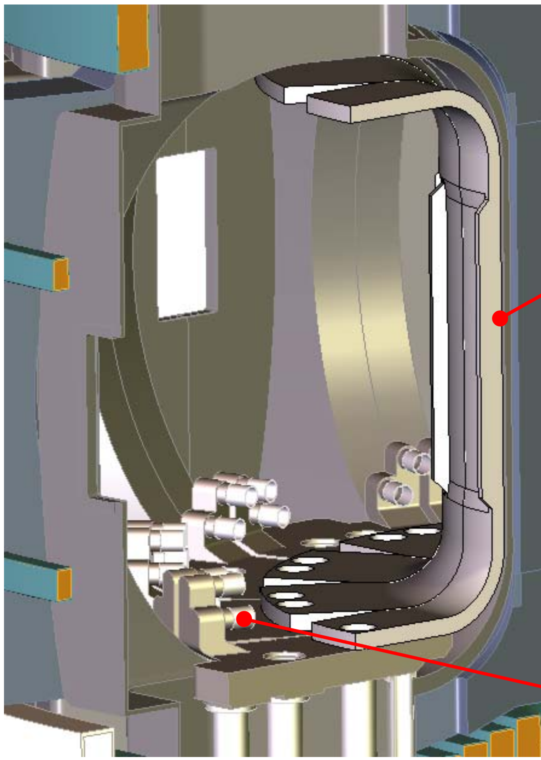
ST maintenance scheme

- Vertical maintenance of all components
- Capability for horizontal maintenance of segmented divertors

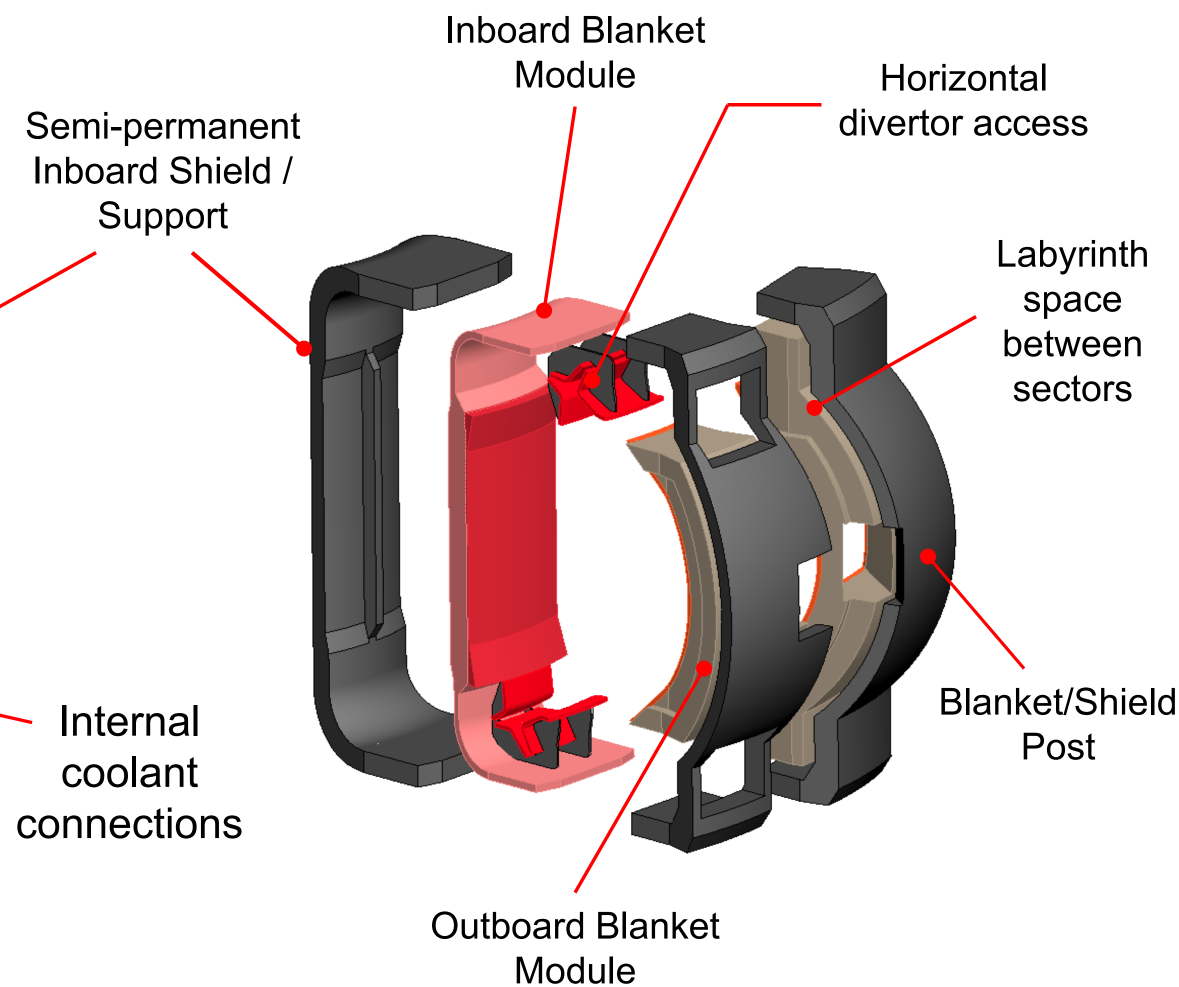
AT option design details



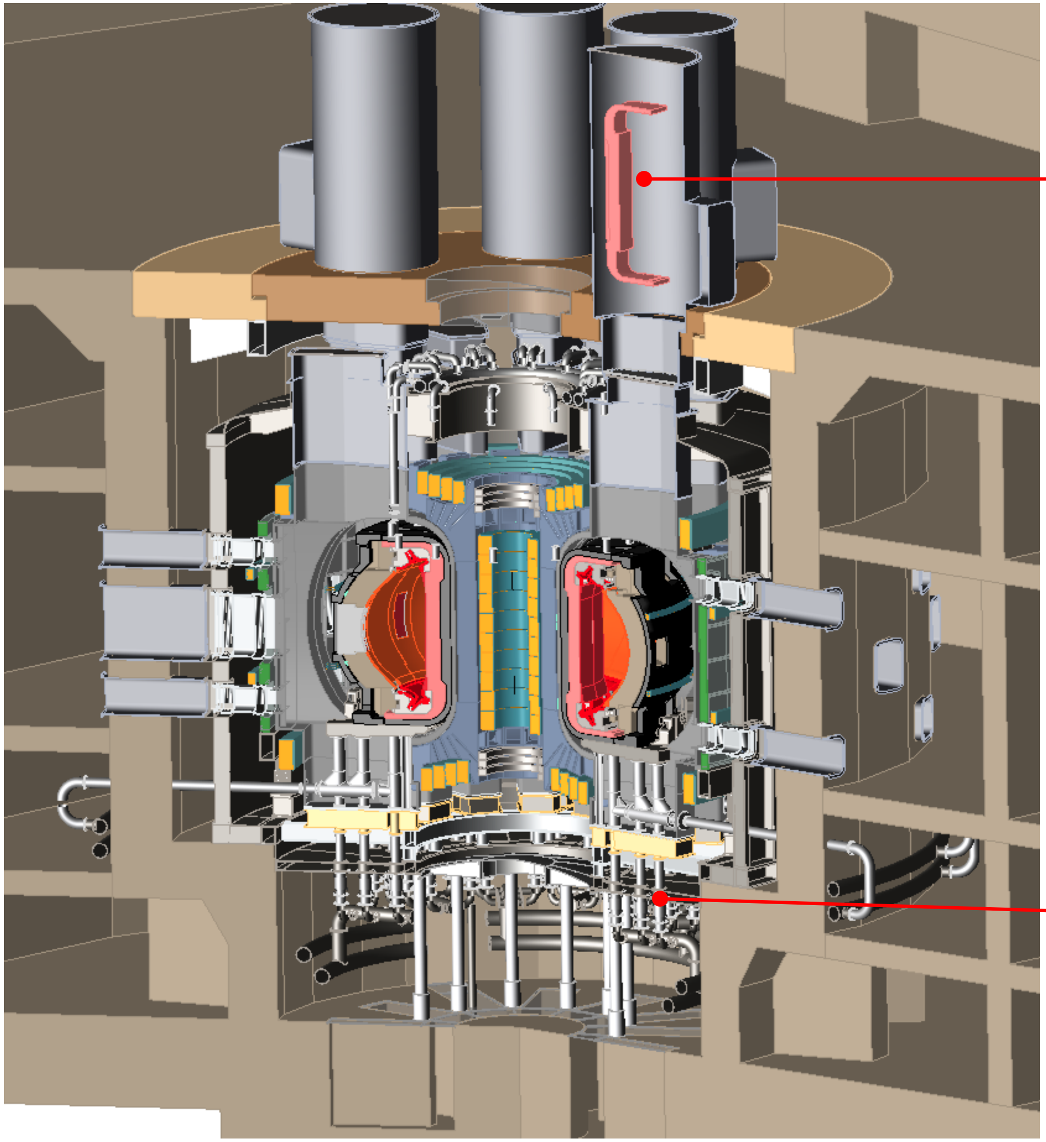
AT option in-vessel segmentation for vertical maintenance scheme



Inboard semi-permanent shield assembled



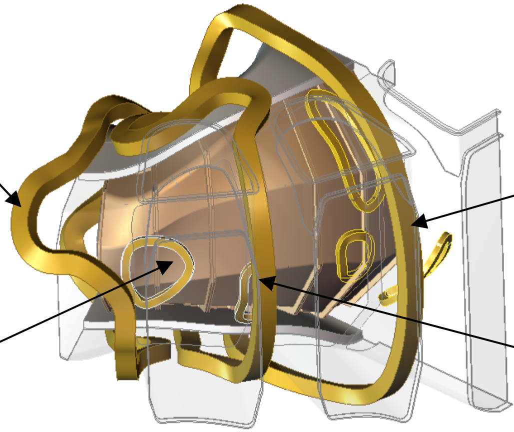
AT device core / facility integration



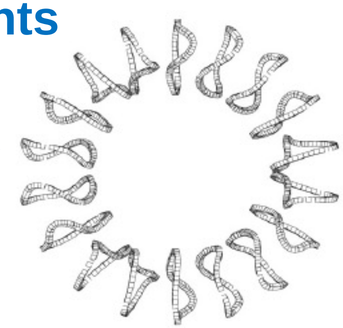
Space provided to service half of the in-vessel core components at one time

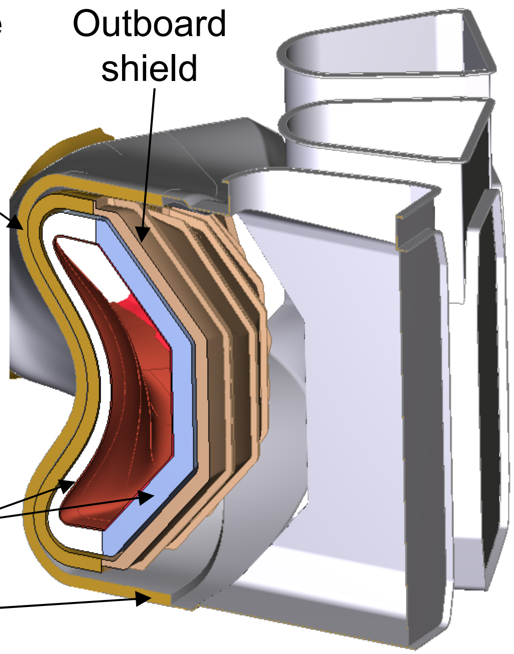
Coolant supply from below

CS design details

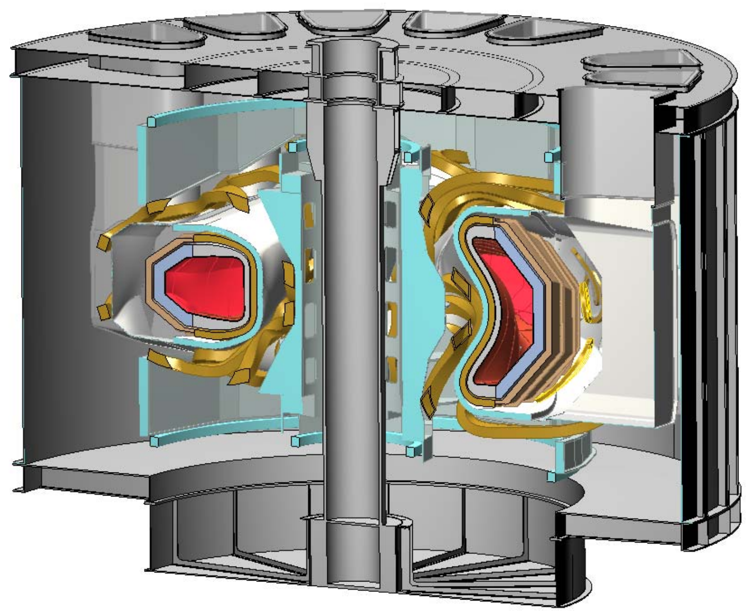


Additional improvements may be possible with aspect-ratio scans and HTS monoliths





Revised ARIES-CS design



Vertically maintained ARIES-CS design

Cautiously optimistic assumptions were made with diagnostics

The PP has different requirements for plasma measurements and hence require different diagnostic systems.

Under the assumptions;

- **that diagnostics are required only to support plasma set-up and optimization, and real-time control (i.e not a comprehensive scientific program), and that**
- **diagnostic developments currently underway are successful,**

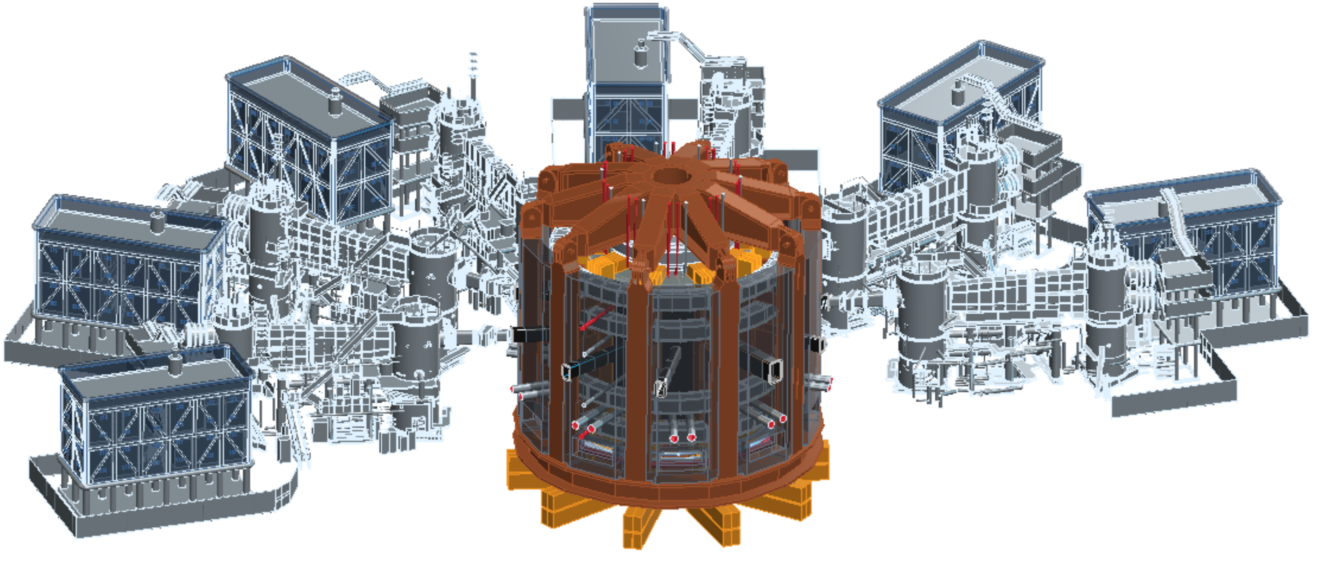
Plasma measurements have been defined, diagnostic systems have been selected, and the requirements for in-vessel, ex-vessel and in-port installations have been determined.

It is concluded that the ST & AT will require ~ 27 diagnostic systems installed in-vessel, ex-vessel, and in 4 upper, 2 mid-plane, and 4 lower ports, and probably some systems integrated in the divertor tiles and structures.

H&CD system impacts

H&CD requirements were defined and design implications were assessed

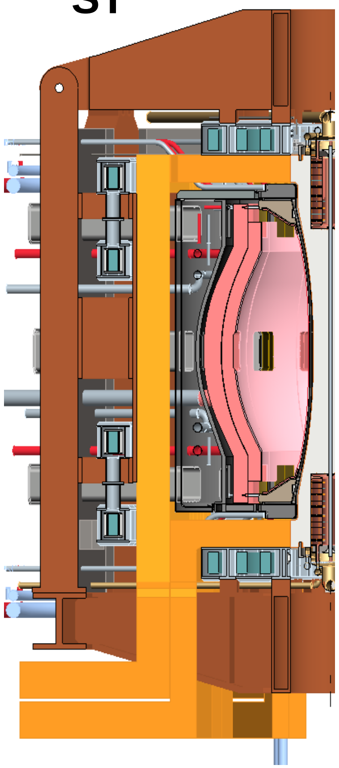
- 6 JT-60SA NNB injectors are shown on the ST device



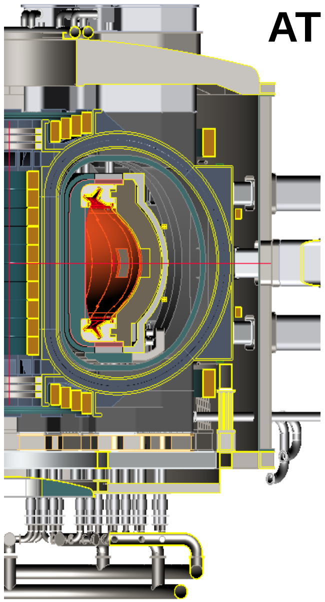
- The AT option has a different mix of H&CD systems ranging from 50 MW EC and NNB's in 4-5 ports to 25 MW IC and 75 MW LH in 3-4 ports depending on the power densities.

Selected PP Comparison Data

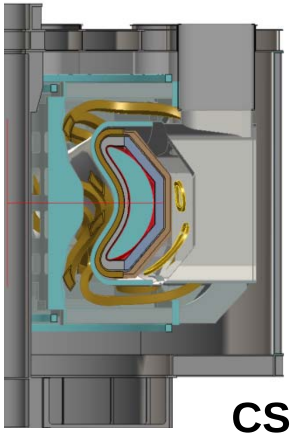
ST



	ST	AT	CS
Major Radius, R_0, m	2.20	4.00	4.75
Plasma volumn, m^3	192	146	104
Plasma surface area, m^2	227	234	266
P_{fus} (MW) - 0.45 thermal eff n_{th}	645	510	313
P_{aux} (MW) - 0.45 thermal eff, n_{th}	60	100	18
$P_{alpha+aux}$ / S, MW/m^2	0.83	0.86	0.30
$P_{alpha+aux}$ / R_0, MW/m	86	51	17
Blanket/shield - tonne	1364	1370	
Divertor (upper) - tonne	40	76	
PF winding - tonne	910	921	
TF winding - tonne	5893	360	
TF structure - tonne	3033	915	
Cryostat -tonne	-	1055	



AT



CS

A review of the dominant pros and cons each pilot plant option follows

Pro / Con for each option - ST

Pro:

- ST physics offers a special class of low-aspect-ratio, wall-stabilized high-β, high-bootstrap fraction tokamak equilibrium.
- The ability to assemble a full blanket system before installation in the device core simplifies alignment.
- The external assembled blanket system may benefit development of a simplified disruption support system.
- Low-aspect-ratio enables higher wall loads to be developed in a given size.
- Jointed TF coils allow the replacement of in-vessel components located within the TF boundary.

Pro / Con for each option - ST

Con:

- Low-aspect-ratio plasmas allow little inboard space for shielding, preventing use of S/C TF & OH magnets.
- Copper TF coils result in high circulating power and the need to size the device to compensate for its use.
- Lack of inboard shielding prevents the use of in-board plasma control diagnostics requiring the need to develop alternate plasma control solutions.
- Jointed coils operated in steady state conditions may have higher failure rates; reliable steady state operation of jointed TF coils needs to be demonstrated.
- Copper TF coils sized for power balance and sliding joints results in heavy components and support superstructure.

Pro / Con for each option - ST

Con - continued:

- Maintenance of a full blanket assembly within the test cell and interfacing cask is complicated by the size of the component.
- To minimize power losses for large conductor currents requires power supplies (conventional or homo-polar generators) to be located very near the device, complicating interfacing details of competing auxiliary equipment and services.

Pro / Con for each option - AT

Pro:

- The AT option has a large physics database.
- The plasma can be sized to allow sufficient inboard space for blanket/shield, plasma control diagnostics and S/C TF and PF coils.
- Plasma physics allows a device to be sized with wall loading and divertor heat loads that is more amenable to material limits.
- Continuous S/C TF coils should afford high reliability operation with technology advancement offering further improvements.
- The AT configuration developed allows in-vessel components to be sized for easier integration with maintenance cask and facility.

Pro / Con for each option - AT

Con:

- Although an intermediate disruption support shell structure has been added to the AT option, the ability to survive disruption loads needs to be demonstrated.
- Developing a viable current drive system remains an open issue.

Pro / Con for each option - CS

Pro:

- The stellarator has the potential of solving two limiting impediments of the tokamak design - high-beta disruption-free operation without plasma control and operation with low circulating power without the need for current drive.
- Operates with lower surface and divertor heat loads.
- A design based on quasi-axisymmetric shaping results in a smaller stellarator device, more in line with tokamak sizing.

Con:

- The coil system geometry used to form non-axisymmetric shaping of the plasma result in complex configuration designs with more complex maintenance approaches. Alternate concepts as described in the PP study need to be pursued.
- 3-D shaping results in more difficult design practices and more complex part manufacturing.

General PP study conclusions

- Availability calculations were not made, however there was no evidence to believe that any one option presents a superior availability advantage.
- All three options confront divertor issues. The higher divertor heat flux of the ST will need the integration of new divertor concepts (Super X, snowflake…) to bring the heat loads to manageable levels. Although the CS has more tenable divertor heat loads it can expect greater complexity in their design and maintenance features.
- Developing a viable current drive system for the ST and AT option is an open issue with respect to demonstrating a credible tokamak scenario with very high bootstrap fractions and economic efficiencies of external H&CD systems.
- In moving from a pilot plant sized device to full power plant facility the AT and CS options appear more feasible than the ST option.
- Further study of the ST option sized to meet a strict fusion nuclear science mission is warranted.

A Pilot Plant fosters new challenges

Design requirements expand beyond ITER:

- Demonstrates electricity breakeven
- Incorporates power plant relevant technologies
- Establishes tritium self sufficiency
- Operates the plasma core components in a neutron / thermal environment prototypical of a power plant, and
- Operates with high availability prototypical of a fusion power plant with the desired flexibility to make in-vessel modifications

THANK YOU FOR YOUR ATTENTION!