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Institute for **Plasma Research**

EXPLORATION OF A COMPACT DEMO REACTOR: CONSTRAINTS ON SHIELDING MATERIALS AND HTS MAGNETS FROM PARAMETER-SPACE SCANS

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DEMO Strategy Needs to be Revised (*)

- Need to look for compact, cost-effective solutions
- Industrial involvement
- Flexibility to adopt emerging technologies

- Leveraging from ITER
- Collaboration, synergistic & complementary programs
- Structures that enable a wide resource-network

R&D Phase	**FEST**	**Pilot Plant**	**DEMO**
Achieving Parity with Peers in Hi-tech	Integrated Test facility for **F**usion **E**ngineering **S**cience & **T**echnology	Pre-DEMO Q ~ 5	Net Electricity (250 MW)
Addressing the Gaps	**Crucial Decision Making Step**		Q ~ 20

(*) R. Srinivasan, S. P. Deshpande, Fusion Engineering and Design 83 (2008) 889–892

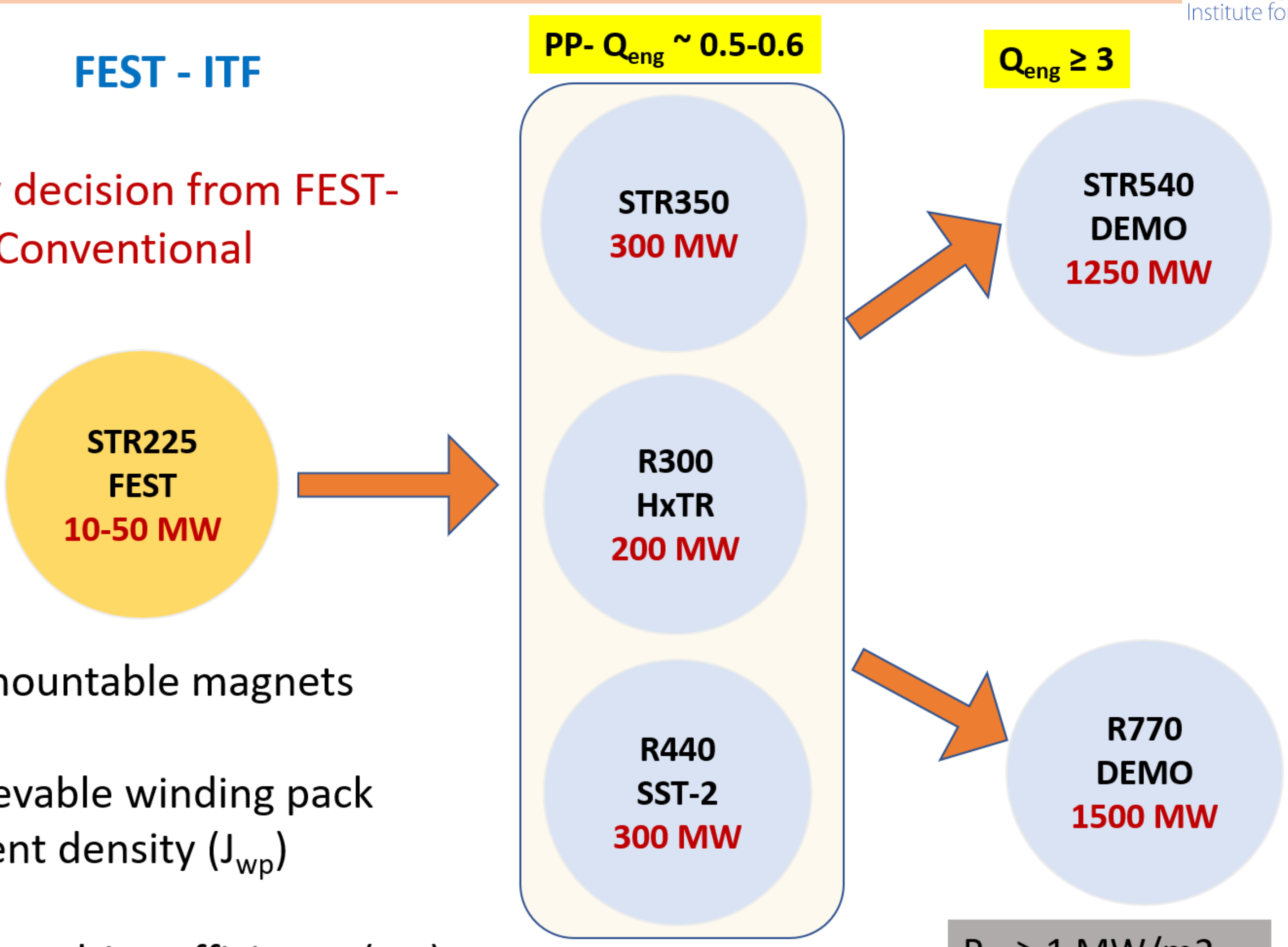
Potential Candidates for PP & DEMO

R&D Gaps

- **Physics**
  - **Equilibrium & Shaping**
  - **Current Drive**
  - **Confinement**
  - **α-heating**
  - **Divertor**
- **Engineering & Materials**
  - **Magnets**
  - **Blanket**
  - **Structural**
- **Maintenance & RH**

FEST - ITF

Key decision from FEST-ST/Conventional



PP- Q_{eng} ~ 0.5-0.6

$Q_{eng} \geq 3$

STR225
FEST
10-50 MW

STR350
300 MW

R300
HxTR
200 MW

R440
SST-2
300 MW

STR540
DEMO
1250 MW

R770
DEMO
1500 MW

De-mountable magnets

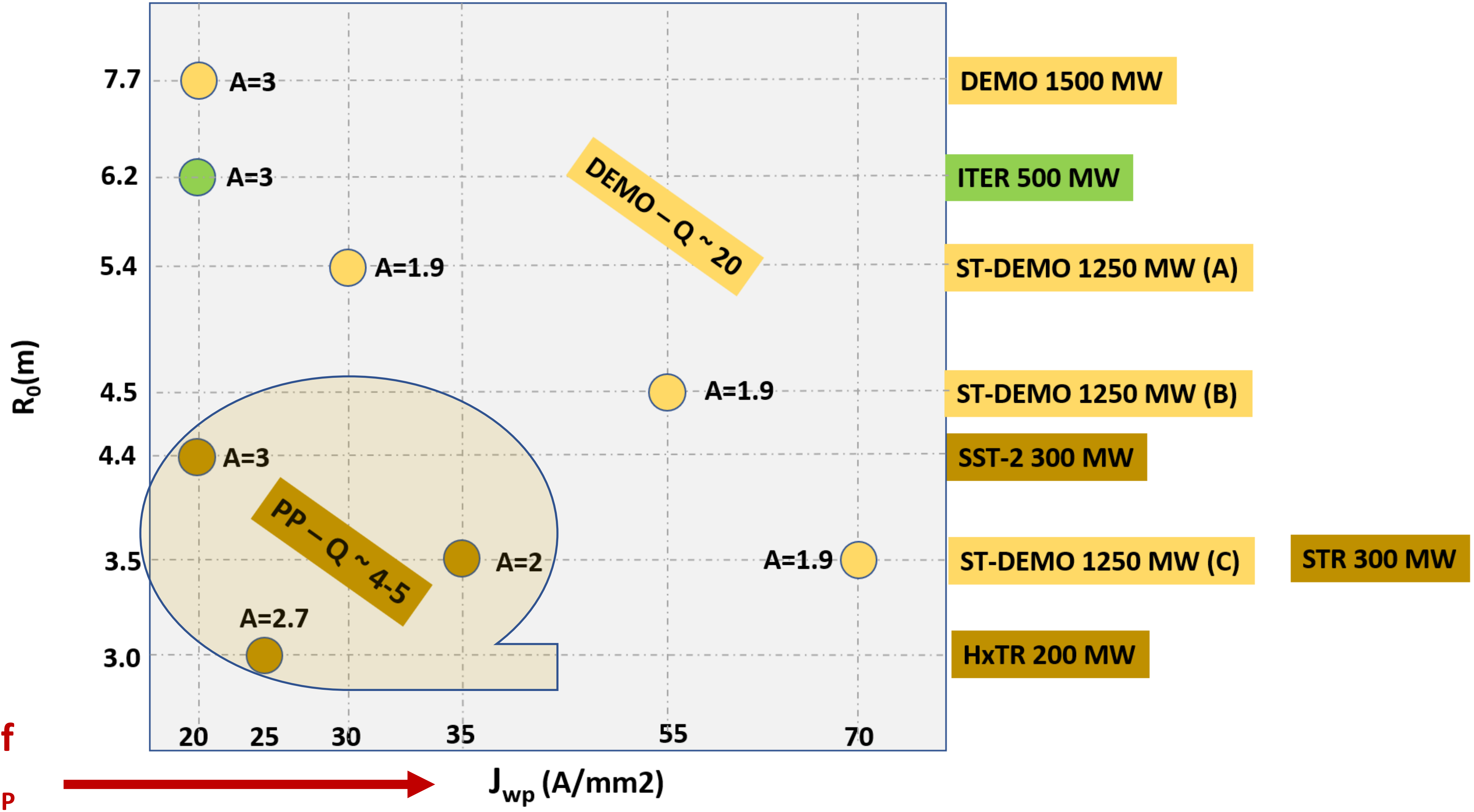
Achievable winding pack current density (J_{wp})

Current drive efficiency (γ_{CD})

P_n <1 MW/m^2, 1- 3 FPY
P_{div} < 8 MW/m^2 , Q ~ 4-5

$P_n \geq$ 1 MW/m2
10-15 FPY
Q ~ 19-20

J_{WP} Limits Minimum Machine Size



Current density of winding pack - J_{WP} →

SARAS – Systems Code

Systems **A**nalysis for Fusion **R**eactor **A**nd **S**coping Code

0-D code with 1-D profiles for κ, δ, n_e, T_e, n_j, T_j, n_j, P_f

Target

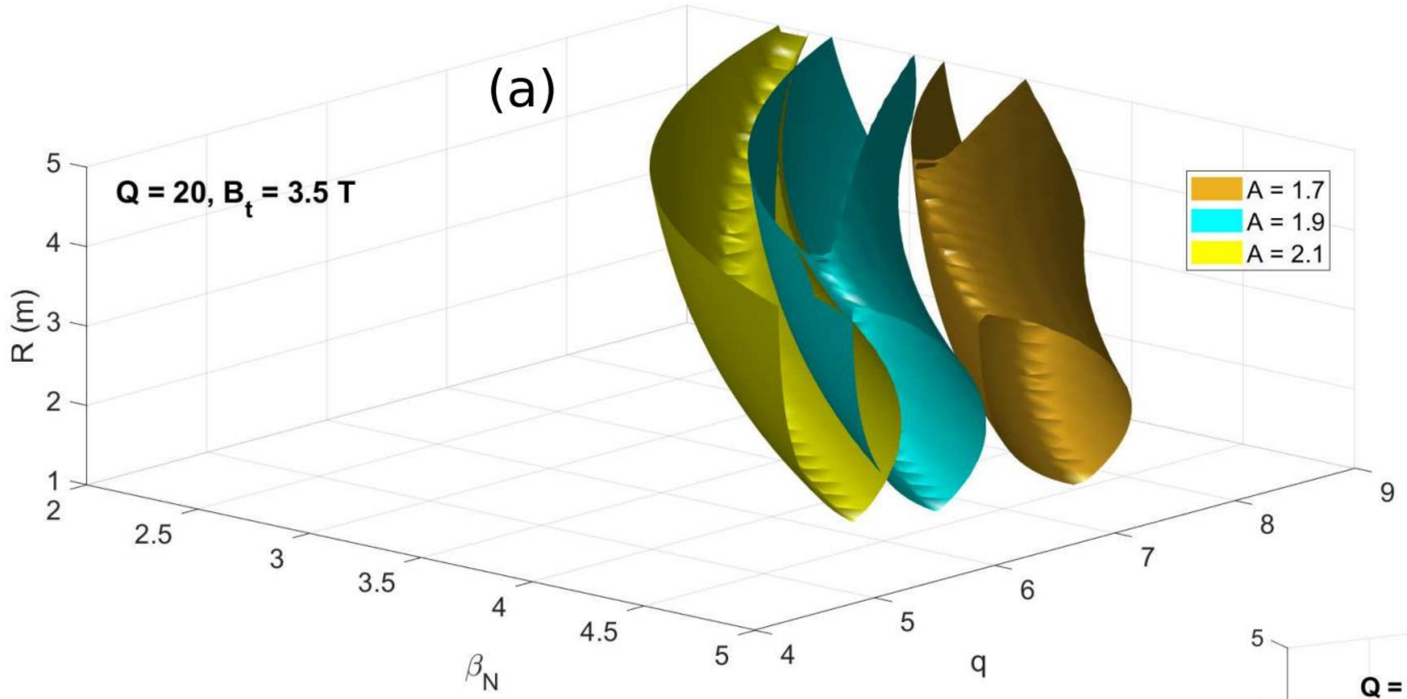
Quick turnaround in configuration changes

Engineering & geometric constraints

Features

- Self-consistent power balance, fuel dilution, different τ_E & L-H scaling
- Impurity model – Corona model, options to include others
- Assumptions on current drive efficiency (γ_{cd}), shielding thickness
- PWI model - erosion & re-deposition
- Geometrical modules
- 50 input variables and 100 output variables
- Evolving capabilities

Constant Q surfaces

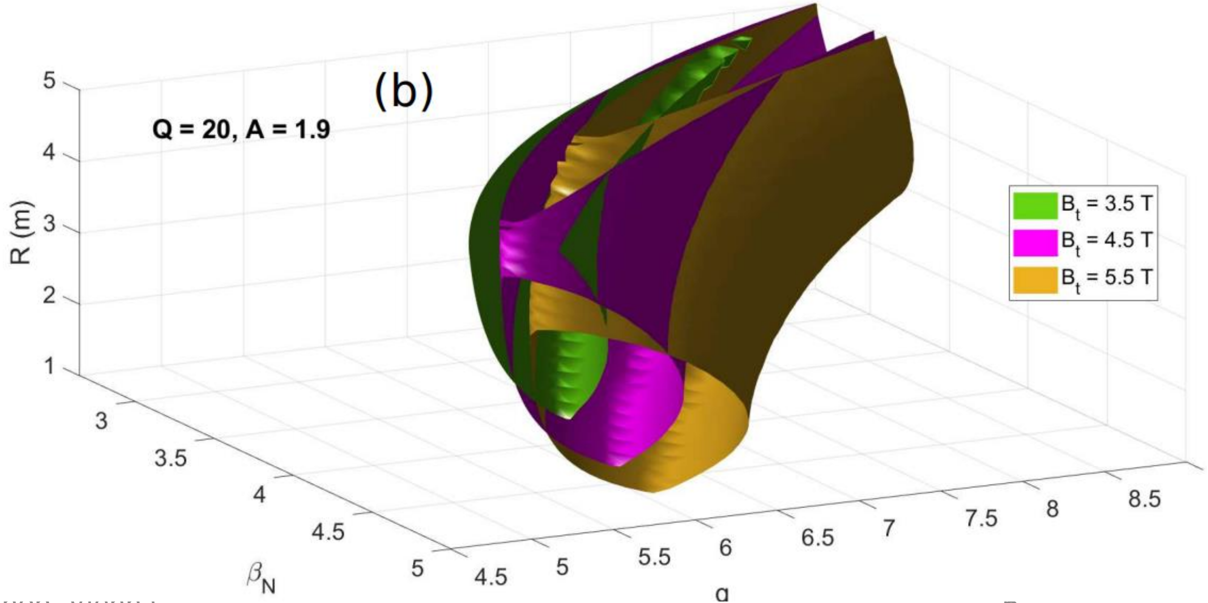


Magnetic field dependence

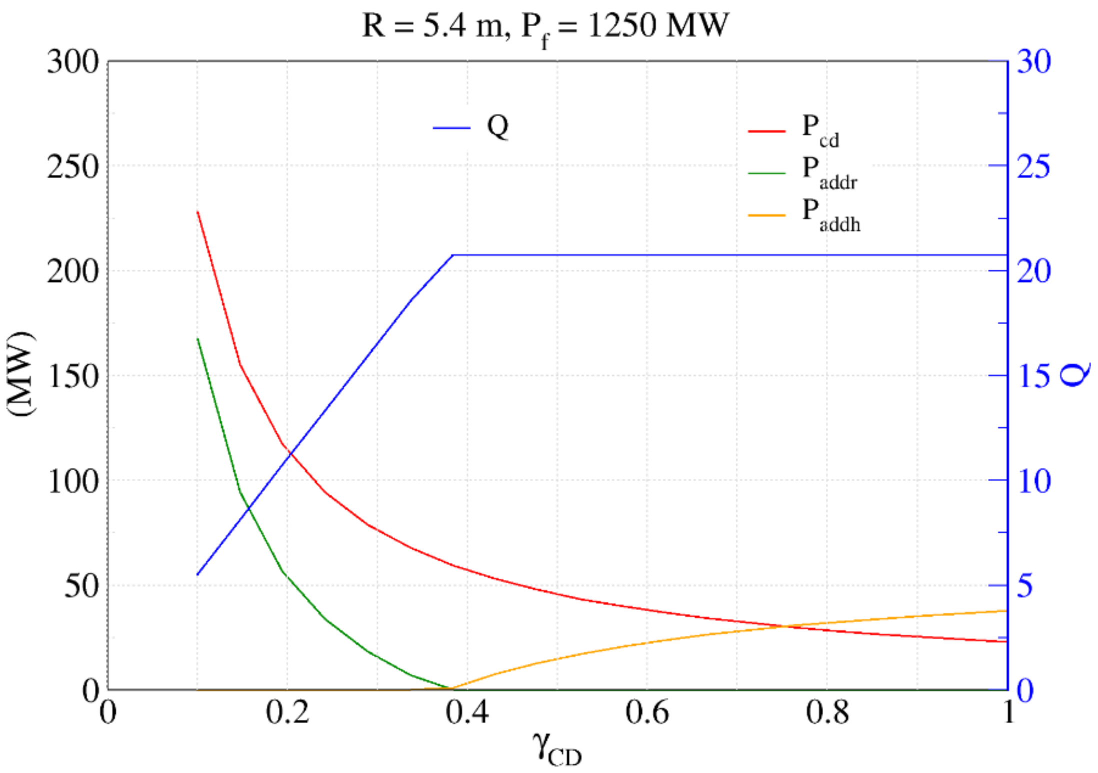
Parameter space shrinks!

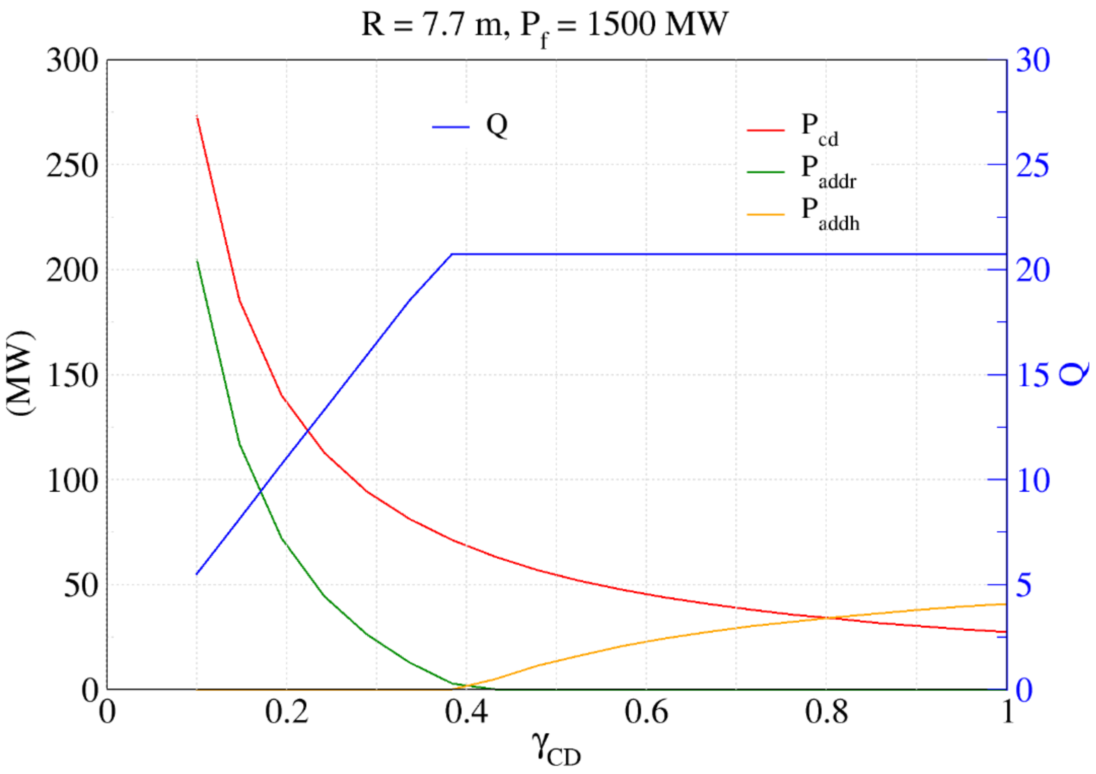
Aspect ratio dependence

At higher A, surfaces shift to the lower q, β_N



Maximum Required γ_{CD} ~ 0.4 – 0.5

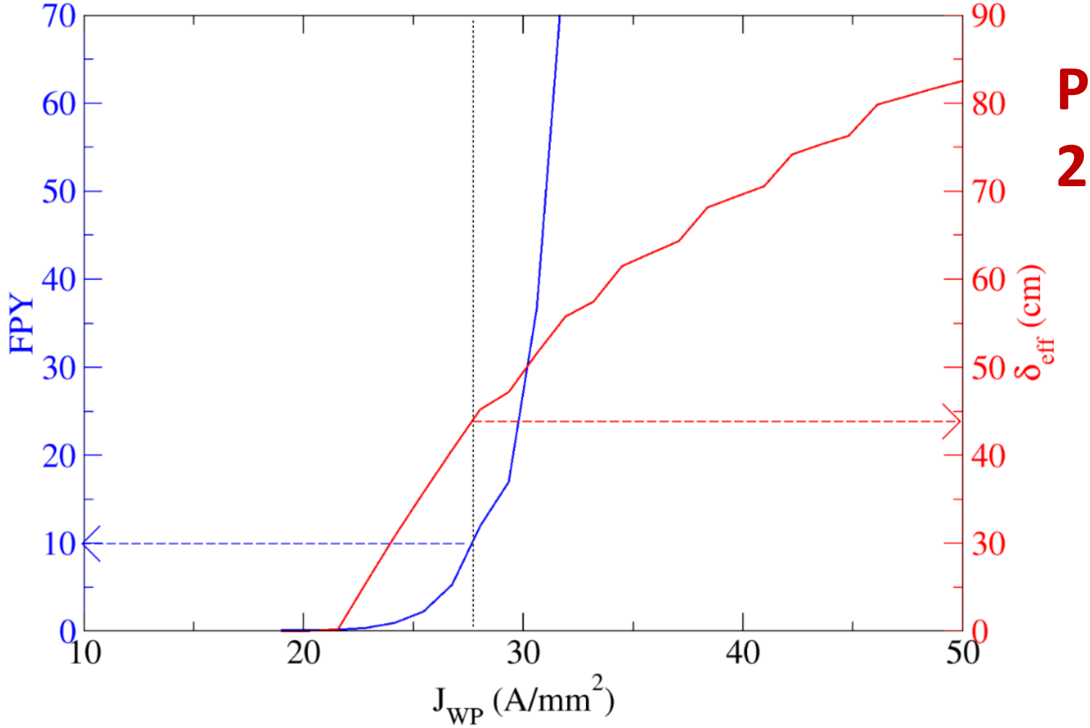




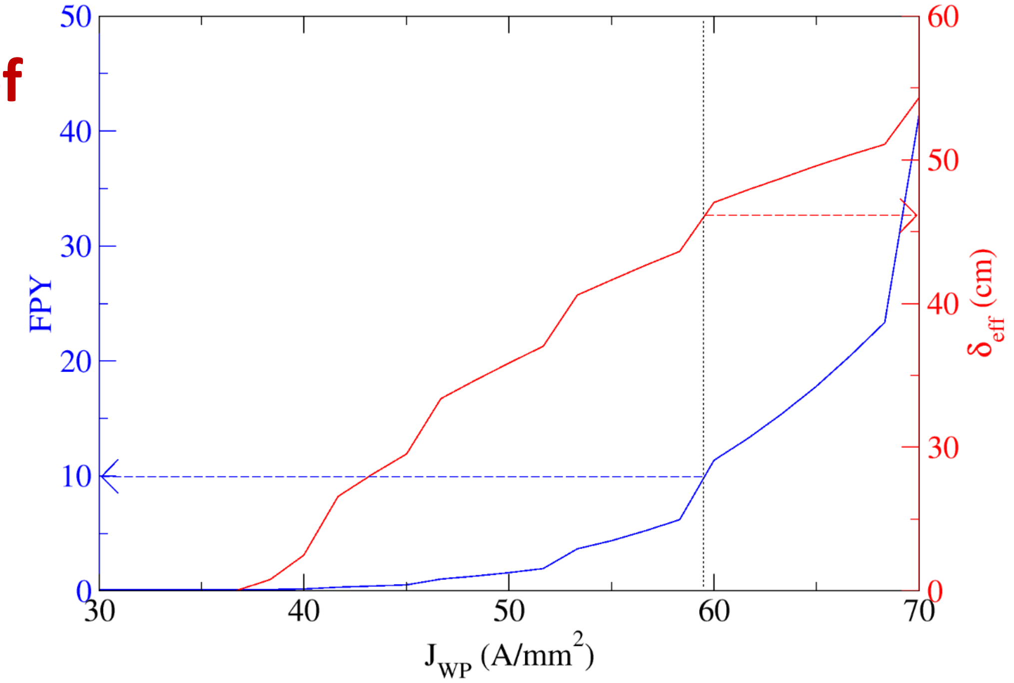
However, wall-plug efficiency does improve Q_{eng}

$\gamma_{cd} * \eta_w$ ~ 0.17-0.18

Optimized WP designs reduce machine size



Peak field of 20 T at CP



R = 5.4 m, B_t = 2.59 T, n_W= 0.9 MW/m^2

R = 4.5 m, B_t = 3.5 T, n_W = 1.3 MW/m^2

Shielding thickness ($\boldsymbol{\delta}_{eff}$) for a decadal length of 13 cm

A = 1.9, κ = 2.7, δ = 0.3, P_f ~ 1250 MW, Q - 19-20, BBZ – 60 cm, Φ_n = 10^{22} n/m^2

Optimizing J_{wp} should be R&D priority

Parameter Space for 250 MWe DEMO

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ST-DEMO – 250 MWe

R = 5.4 m | A = 1.9 | a = 2.8 m
κ = 2.7 | δ = 0.3 | B_t = 2.6 T | q = 7.5
I_p = 19 MA | β_N = 3.25
$\langle n_{20}\rangle$ = 0.60 | $\langle T_{keV}\rangle$ = 9.5 | f_{bs} = 0.65

P_f = 1250 | Q ~ 20
P_{cd} = 63 MW | f_{He} = 4%

$P_{n\text{-}wall}$ = 0.88 MW/m^2
P_{div*} = 6.50 MW/m^2
J_{WP} = 29 A/mm^2 | Δ_{SB_in} = 87cm

DEMO – 250 MWe

R = 7.7 m | A = 3 | a=2.6 m
κ =1.9 | δ = 0.4 | B_t = 4.9 T | q = 5
I_p = 14.3 MA | β_N = 2.80
$\langle n_{20}\rangle$ = 0.63 | $\langle T_{keV}\rangle$ = 12.2 | fbs = 0.6

P_f = 1500 | Q ~ 20
P_{cd} = 78 MW | f_{He} = 5.7%

$P_{n\text{-}wall}$ = 1.10 MW/m^2
P_{div*} = 6.15 MW/m^2
J_{WP} = 19.3 A/mm^2 | Δ_{SB_in} = 80cm

Key assumptions:

γ_{cd} = 0.35 | f_G = 0.9 | H_h = 1.2-1.3 | η_{th} = 0.33 | η_w = 0.5 | S_n = 0.5 | S_T = 1.0 | τ_{He}/τ_E = 5

Blanket & Maintenance Considerations for DEMO

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- Breeding zone of 60 cm and an effective shielding zone of 47 cm at the inboard
- Pilot plant will require to test blanket materials under neutron loads ≥ 1MW/m^2 (1 – 5 MW/m^2), tritium breeding & extraction.
- Liquid & solid breeder concepts → high neutron loads make liquid breeder attractive

- High temperature heat extraction (~500 C) with RAFM steels require helium based coolant. Advanced cooling concepts to optimize power consumption.

- De-mountable magnets decide the maintenance scheme. Reasonably large vertical port, taking into account the PF coil design, will be needed for blanket RH.

- Performance of magnet insulation, structural materials & shielding materials under neutron environment is an R&D priority. Irradiation using ions, fission neutrons and g-rays are ongoing – A careful extrapolation of the data to reactor regimes is urgently needed →statistical/AI-based techniques for credible extrapolation?

Acknowledgements and Credits

The authors acknowledge many useful discussions with the members from fusion blanket division and the magnet division of IPR.

For further information on R&D: www.ipr.res.in

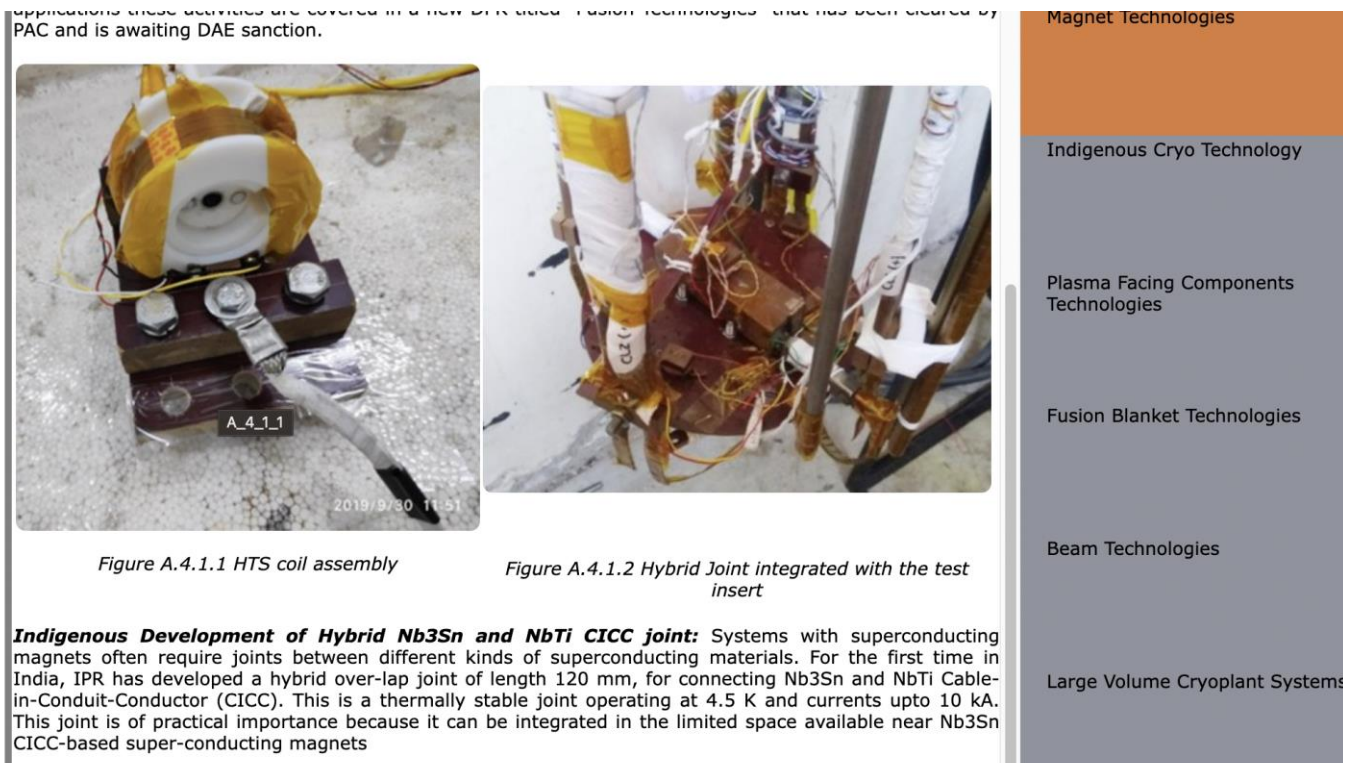
Thankyou to the participants of the DEMO workshop and the IAEA

Backup

Magnets and HTS R&D is in progress

Fusion & Related Technologies

To realise thermonuclear fusion, a vast majority areas of different technologies have to work in tandem: High current electromagnets, advanced material technologies, robotic technologies, cryogenic technologies, beam technologies, Radio-Frequency wave technologies etc. Also these technologies have to work in a very hostile environments which includes nuclear radiations. Continuous progress related to these technologies are being made with relevant science and engineering.



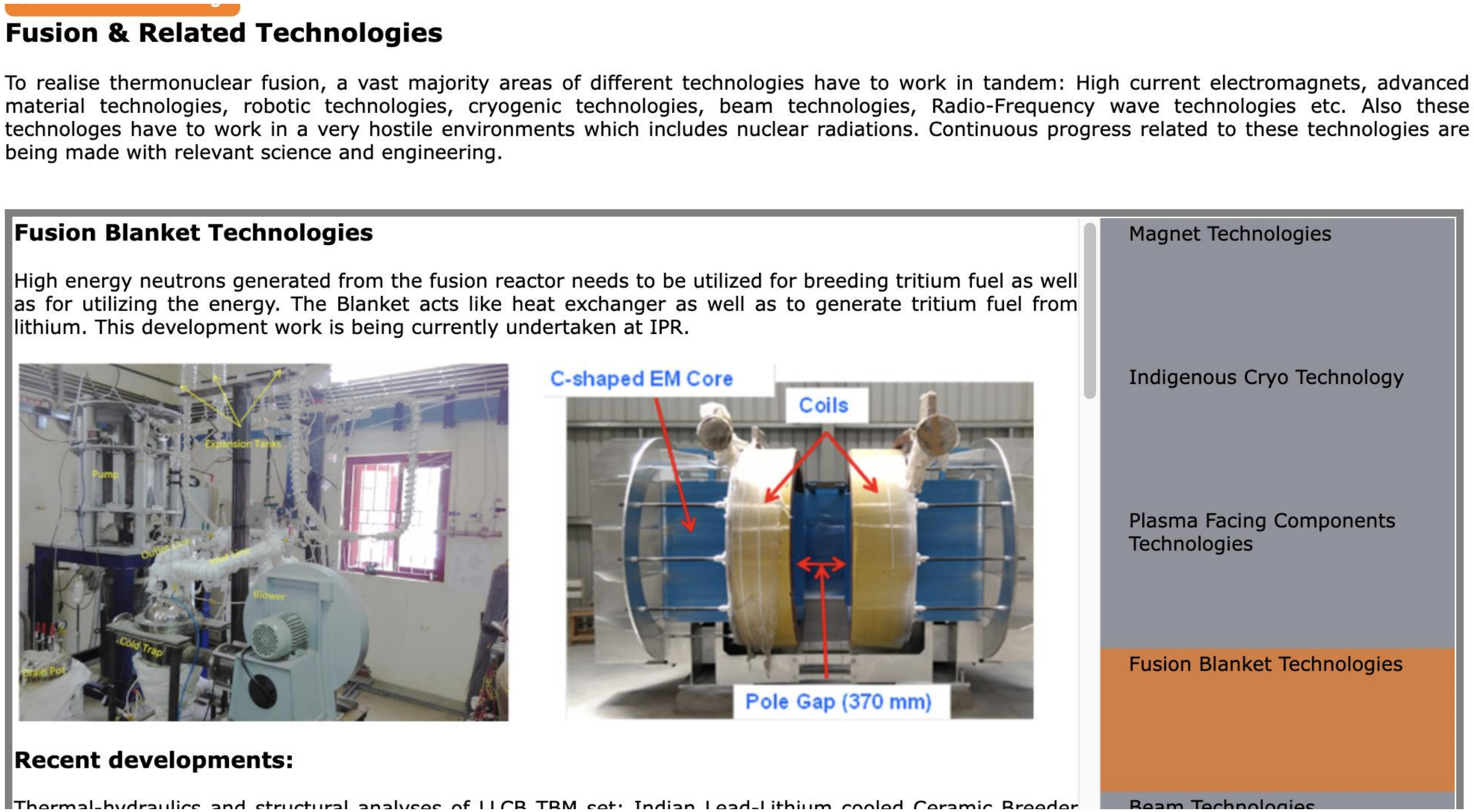
applications these activities are covered in a new DPR titled "Fusion Technologies" that has been cleared by PAC and is awaiting DAE sanction.

Figure A.4.1.1 HTS coil assembly

Figure A.4.1.2 Hybrid Joint integrated with the test insert

Indigenous Development of Hybrid Nb3Sn and NbTi CICC joint: Systems with superconducting magnets often require joints between different kinds of superconducting materials. For the first time in India, IPR has developed a hybrid over-lap joint of length 120 mm, for connecting Nb3Sn and NbTi Cable-in-Conduit-Conductor (CICC). This is a thermally stable joint operating at 4.5 K and currents upto 10 kA. This joint is of practical importance because it can be integrated in the limited space available near Nb3Sn CICC-based super-conducting magnets

Blanket R&D is in progress



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Fusion Blanket Technologies

High energy neutrons generated from the fusion reactor needs to be utilized for breeding tritium fuel as well as for utilizing the energy. The Blanket acts like heat exchanger as well as to generate tritium fuel from lithium. This development work is being currently undertaken at IPR.

Recent developments:

Thermal-hydraulics and structural analyses of LLCB TBM set: Indian Lead-Lithium cooled Ceramic Breeder

Blanket Considerations for DEMO

- A critical decision on whether demountable HTS magnets can be made and reliably re-assembled will determine the ST vs. conventional path. FEST or its equivalent (called by various names) will therefore prove a crucial step.
- Heat extraction at high temperatures (~500 C) allowed for the RAFM steels will require the use of helium as a coolant. Significant pumping power will play a role in net electricity gain.
- The breeder zone of about 60 cm and effective shielding thickness of about 45 cm on the inboard-side is expected. Better shielding for magnet insulation and longer replacement time should be a priority in R&D.
- Tritium breeding and extraction R&D needs to be carried out
- Reasonably large vertical port, taking into account the PF coil design, will be needed for blanket RH.

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- Need to look for compact, cost-effective solutions
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R&D Phase

- Design innovation
- Material selection, testing & qualification
- Component dev., testing
- Facility and infrastructure dev.
- Tokamak performance enhancements

FEST:

Integrated Test facility for Fusion Engineering Science & Technology

- Integrated performance test of various systems and technology validation
- Pulsed, low-power fusion demonstration
- long-pulse non-fusion demonstration
- Decision on ST vs conventional

Pilot plant

- Achieving Q ~ 5
- Power extraction
- T-breeding
- Reliability & Availability
- Fuel cycle tech. demo.
- Maintenance & RH

DEMO

- Achieving Q ~ 20
- Net power generation
- T-sufficiency
- Reliability & Availability
- Maintenance & RH

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Basic Equations

Equilibrium & Stability	$\beta = \frac{2\mu_0\langle p\rangle}{B_t^2}$ $\beta_N = \frac{\beta[\%]aB_t}{I_p}$ $q = \frac{5a^2B_tS_k}{RI_p}$
Profiles	$R_i = R + r_i\cos(\theta + \delta_i\sin(\theta)), Z_i = \kappa_i r_i \sin(\theta)$
Ion Density	$n_i = n_e\left(1 - \sum\langle Z\rangle_j f_j\right) = n_e(1 - 2f_{he} - \epsilon_z)$
Fusion Power & Gain	$P_f = n_D n_T\langle\sigma v\rangle E_f$ $Q = \frac{P_f}{P_{aux}}$ $P_{aux} = P_h + P_{cd}$
Power Balance	$P_h = P_L + P_r^{core}$
Confinement time	$\tau_E = 0.0562 H_h I_p^{0.93} B_t^{0.15} n_{20}^{0.41} 10^{0.41} R^{1.97} \kappa^{0.78} \epsilon^{0.58} M^{0.19} P_L^{-0.69}$
L-H transition threshold	$P_{LH} = 0.042\, n_{20}^{0.73} B_t^{0.74} S^{0.98}$
α-confinement time	$\frac{\tau_\alpha}{\tau_E} = 5 - 10$

FEST – Integrated Test Facility

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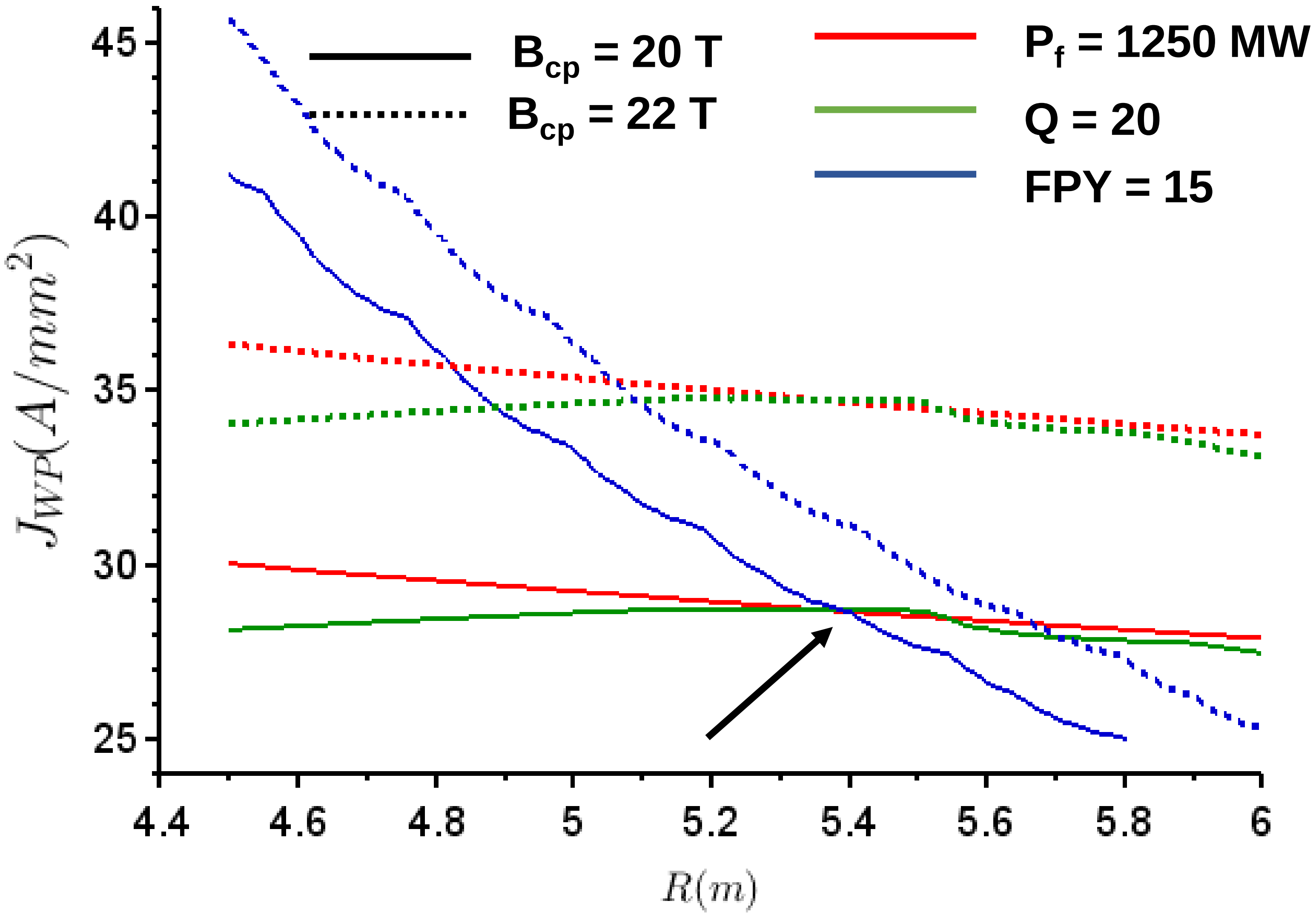
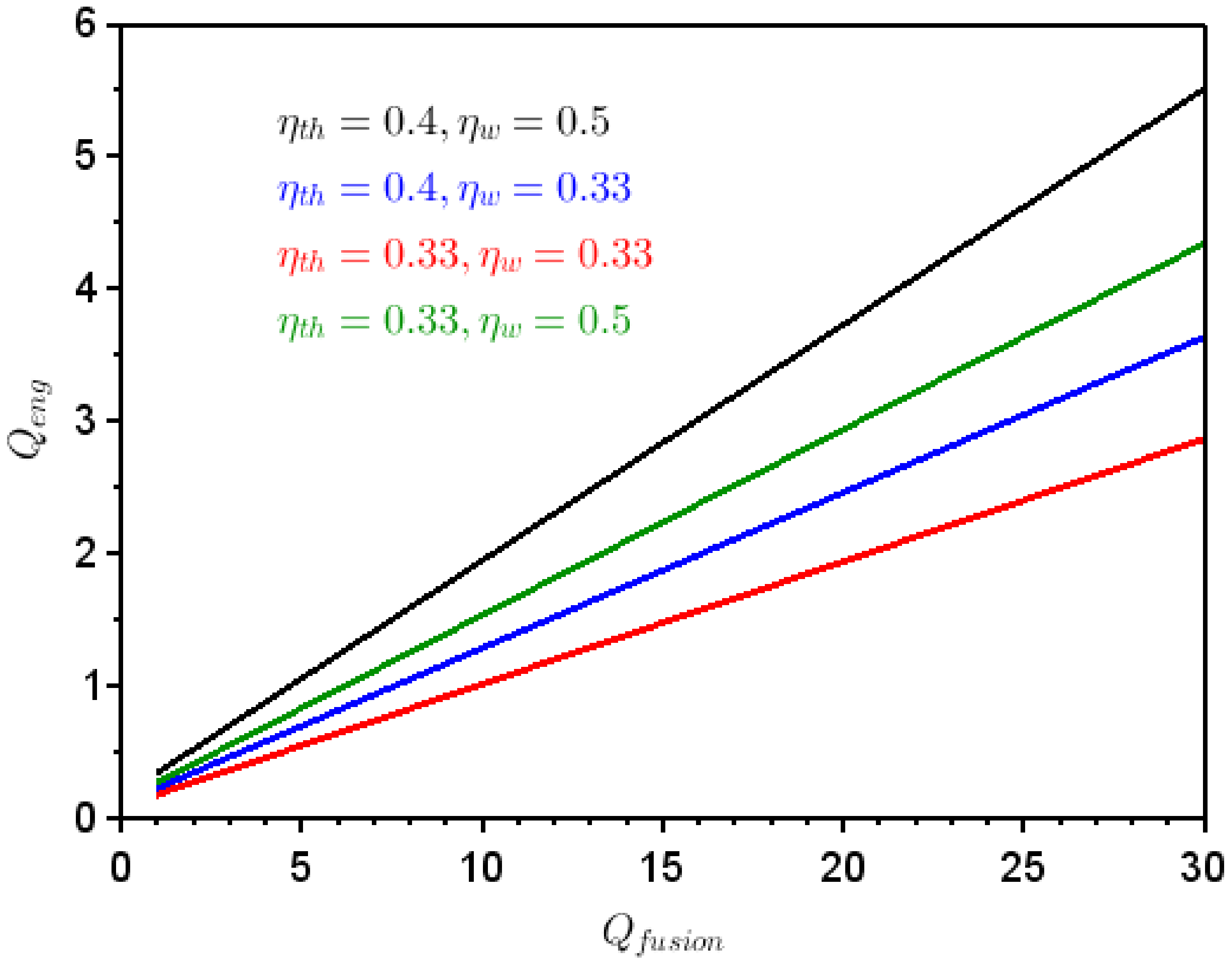
- Integrated testing of HTS coils, different blanket concepts, structural materials & divertor
- Integrated testing of current drive systems
- Testing of maintenance and RH schemes

Parameters obtained from the system code – SARAS → **S**ystems **A**nalysis for Fusion **R**eactor **A**nd **S**coping Code

$A = 2.1$, $k = 2.5$, $d = 0.5$
$R = 2.25$ m, $B_t = 3$ T, $I_p =$ **6 MA**
$Pf = 50$ MW, $Q = 1.5$
$f_{bs} = 0.4 – 0.7$, $\beta_N = 3.5$
$J_{wp} = 50$ A/mm^2
$n_w \sim 0.5$ MW/m^2

Constraint from the Peak Field and I_c





$\eta_{th} = 0.4, \eta_w = 0.5$
$\eta_{th} = 0.4, \eta_w = 0.33$
$\eta_{th} = 0.33, \eta_w = 0.33$
$\eta_{th} = 0.33, \eta_w = 0.5$
Q_{eng}
0
1
2
3
4
5
6
0
5
10
15
20
25
30
Q_{fusion}

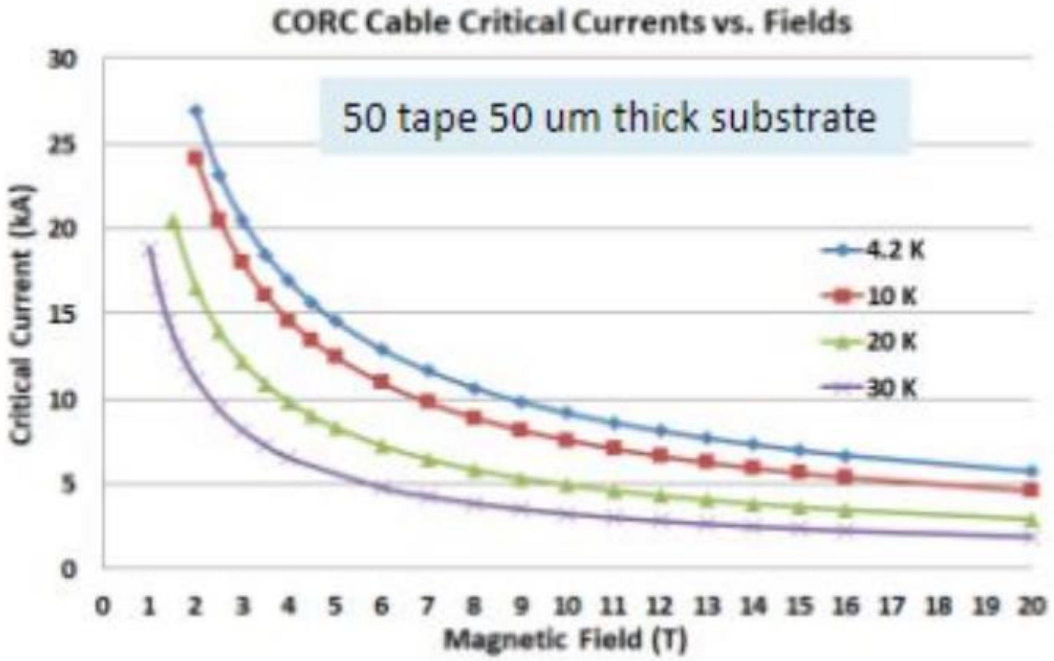


Fig. 4. The I_c performance of a 50 tape 50 μm substrate CORC

Zhai et al, 2021, FNSF