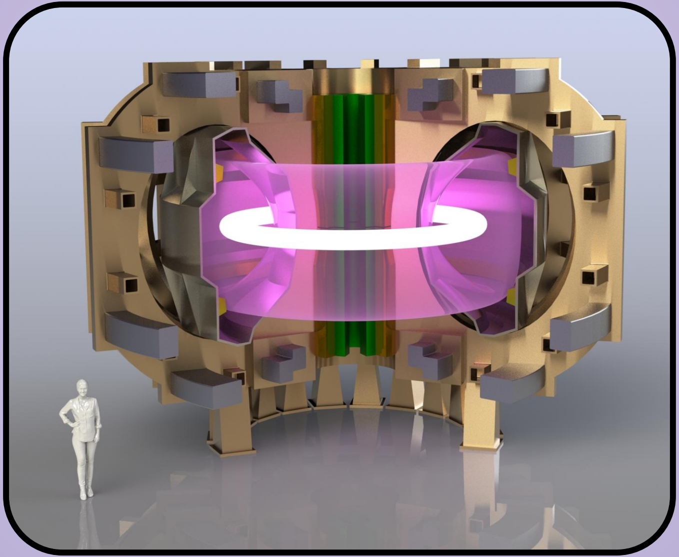
CENTAUR

Compact Experimental Negative TriAngUlarity Reactor

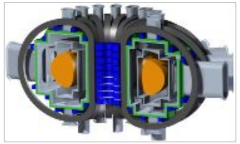
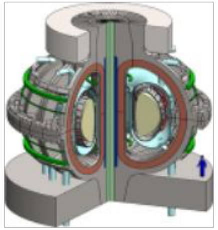


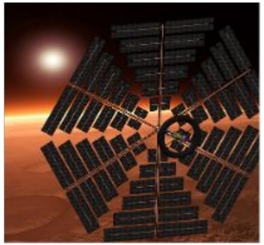
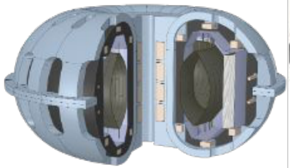
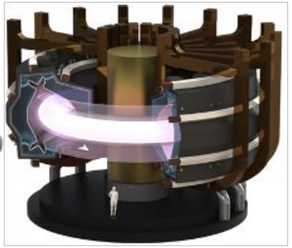
Community Webinar Presentation

July 31, 2025

Columbia
Fusion Research Center

Fusion Systems Design: Essential Fusion Education

MIT+ Princeton

MIT+ Columbia

2012 — 2014 — 2016 — 2018 — 2020 — 2022

VULCAN
- PMI science facility
- D-D fuel
- Demountable HTS joints
- HFS current drive

G. Olynyk et al., *Fusion Eng. and Design*, 87(3), 2012.
Y. Podpaly et al., *Fusion Eng. and Design*, 87(3), 2012.
D. G. Whyte et al., *Fusion Eng. and Design*, 87(3), 2012.
H. Barnard et al., *Fusion Eng. and Design*, 87(3), 2012.

ARC
- D-T fusion pilot plant
- FLiBe liquid immersion blanket
- Demountable HTS joints
- Steady state AT operation

B. N. Sorbom et al., *Fusion Eng. and Design*, 100, 2015.

Commonwealth Fusion Systems

ARC+divertor
- Added advanced divertor solution to ARC design
- Studied ARC diagnostics

A. Q. Kuang et al., *Fusion Eng. and Design*, 137 (2018).

Fission-fusion interplanetary spacecraft
- D-T mirror-driven subcritical fission assembly

ARCH
- High-temperature ARC for hydrogen production
- Disruption-tolerant vacuum vessel (LSVV)
- High energy density L-mode operation with radiative heat exhaust
- ***Joint with Princeton***

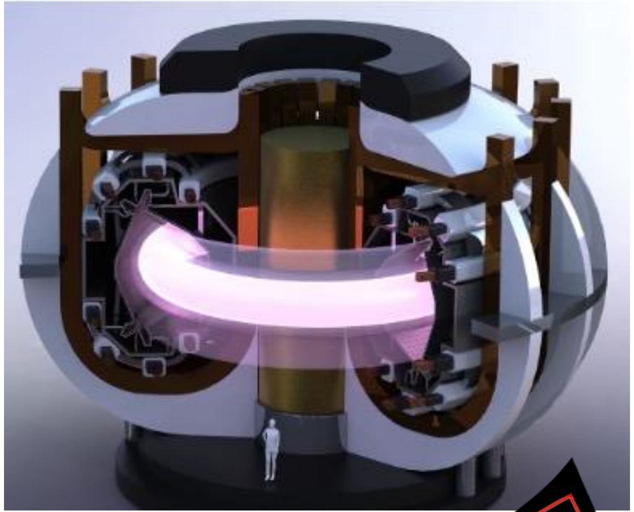
MANTA
- NASEM-compliant fusion pilot plant
- Negative triangularity
- FLiBe liquid immersion blanket
- Demountable HTS joints
- ***Joint with Columbia***

Credit: Dennis Whyte

Columbia Continued the Excellent MIT Tradition with NT

2022

w/ MIT

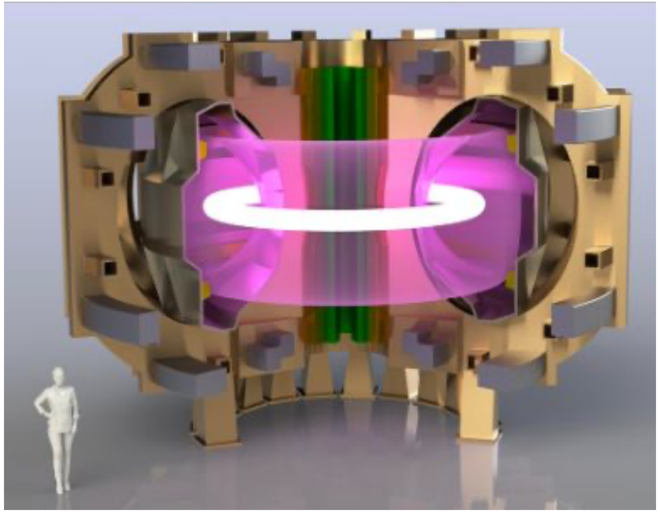


MANTA

Modular Adjustable Negative Triangularity ARC

NASEM-Compliant FPP! P_{elec}=50 MW

2024



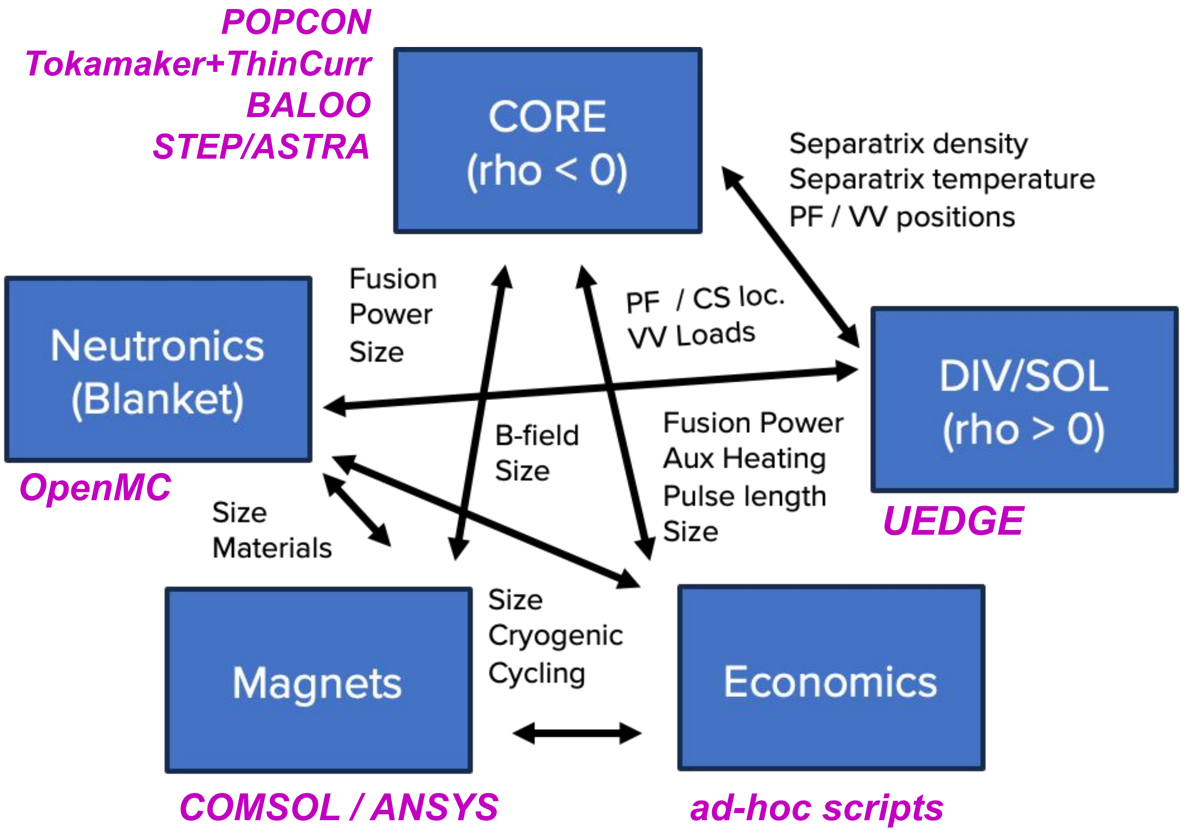
CENTAUR

Compact Experimental Negative TriAngUlarity Reactor

SPARC-like: Soonest Possible Q>1

Structure of the Fusion Design Class

- One semester course!
- ~20+ students participated (mostly grad students)
- Students divided into 5 sub-teams
- Weekly sub-team meetings
- Weekly plenary interface meeting
- After class concludes, continue to refine results

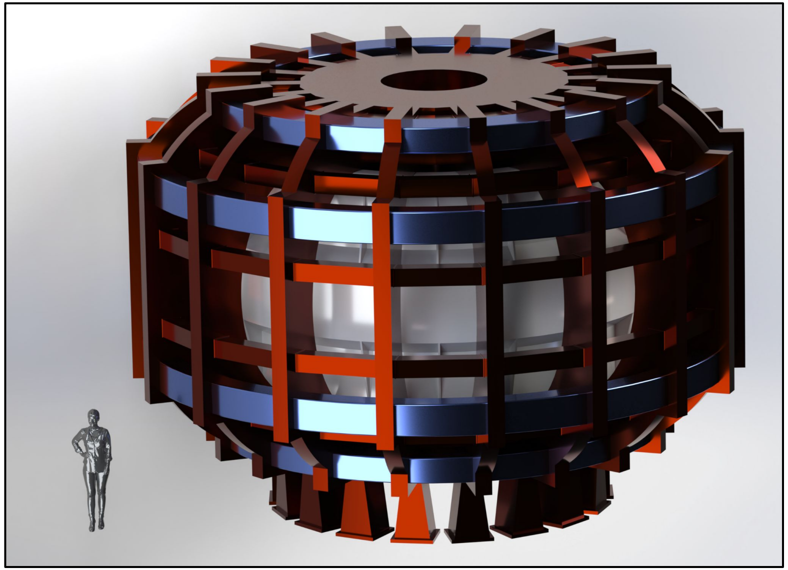


Design targets input by Prof. Carlos Paz-Soldan

- Produce net energy from the plasma: **Q > 1** (plasma gain)
- Lowest possible capital cost, goal: **< $2B**, estimate operational cost
- No tritium breeding requirement, **<10 g tritium site inventory**
- Pulsed operation with **10 s flat-top**
- Survive **10,000 full-performance non-nuclea**r (H/He) pulses
- Survive **3,000 full-performance nuclear** (D/T) pulses
- Survive **100 full-performance unmitigated disruptions**
- Only **mature technologies** (i.e., cannot use jointed HTS, FliBe blankets)

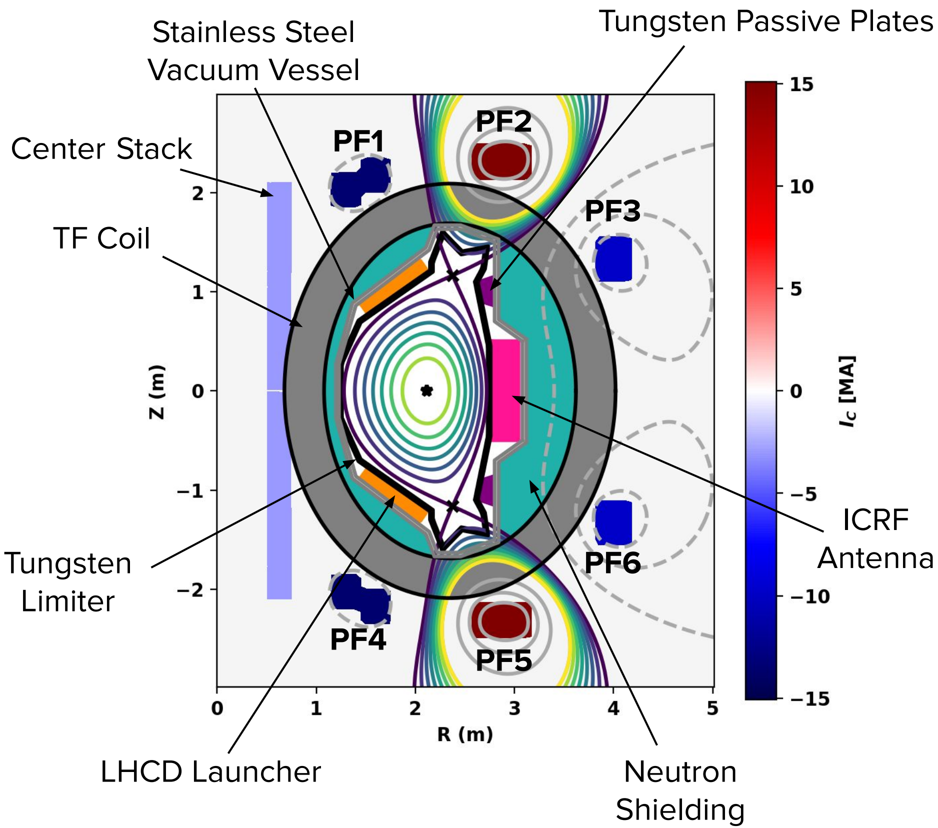
Outline

- Overview and Key Parameters
- **Core scenario**
  - Plasma performance, stability, power balance, disruption EM loads
- **Power handling**
  - Divertor heat and particle loads, strike plate cooling
- **Neutronics**
  - Neutron TF heating and shielding requirements
- **Magnetics**
  - TF, PF, CS designs and EM loads
- **Economics**
  - Construction and operating costs
- Conclusion and Summary



Topology incorporates constraints from every team

Key Parameters		
Design	Major radius – R	2.0 m
	Minor radius – a	0.72 m
	Toroidal Field – B_T	10.9 T
Plasma	Plasma current – I_p	9.6 MA
	Elongation – κ_{edge}	1.63
	Triangularity – δ	-0.52
	Safety Factor – q_{95}	2.49
	Normalized Beta – β_N	1.65
	Physics gain – Q	1.3
	H-factor – H_{98}	0.64
	Greenwald fraction – f_{GW}	0.6



Core Scenario

Abdullah Hyder, Alexa Lachmann, Anson Braun
Avigdor Veksler, Kian Orr (PU), Hiro Farre (PU), Jamie Xia
Mentors: Chris Hansen, Nils Leuthold, Matthew Pharr
Orso Meneghini (GA), Oak Nelson, Tim Slendebroek (GA),
Benedikt Zimmermann

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GENERAL ATOMICS
PRINCETON UNIVERSITY

Design a Q > 1 plasma scenario within engineering constraints

OpenPOPCON[1]
- 0-D fitted scaling laws
- Initial operating space and plasma parameter scoping

TokaMaker[4]
- Time-dependent Grad-Shafranov solver
- Required coil drive for vertical stability
- Vertical displacement event forces

FUSE[2]
- Whole-device modeling framework
- Integrated core-edge transport solution

ASTRA[3]
- Modular transport framework
- TGLF turbulent transport calculations [Staebler POP 2007]
- FUSE ICRF heating profiles

[1] github hansec/OpenPOPCON
[2] Menghini arXiv 2024
[3] Pereverzev IPP Report 2002
[4] Hansen Comp. Phys. Com. 2024

Initial scoping using POPCONs defined our initial parameter space

Plasma **OP**eration **CON**tour Plot

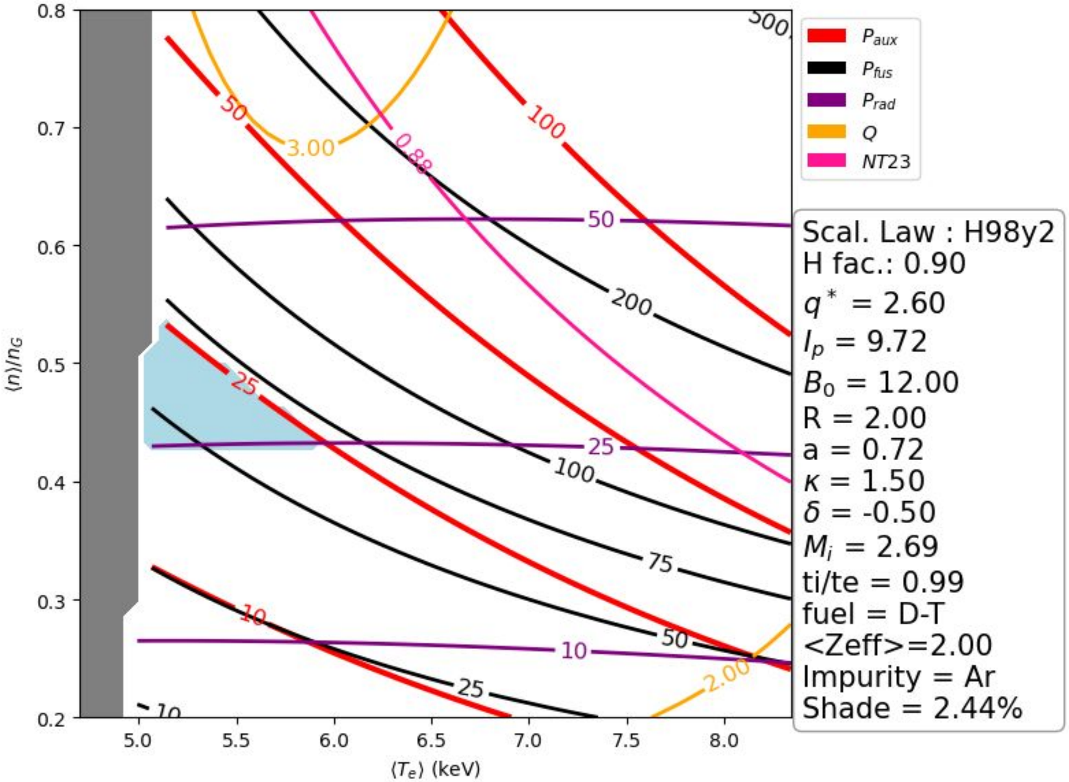
Conservative constraints chosen for a factor of safety

Initial Constraints:

- Q > 2
- P_{aux} < 25 MW
- P_{rad} > 25 MW

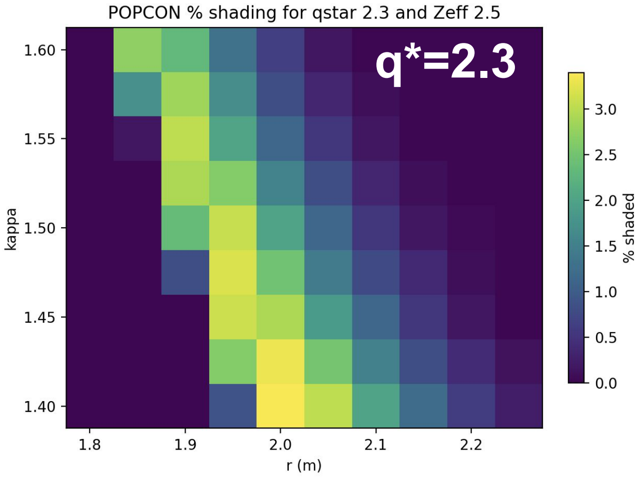
$n_G = 5.968 \times 10^{20}\ m^{-3}$

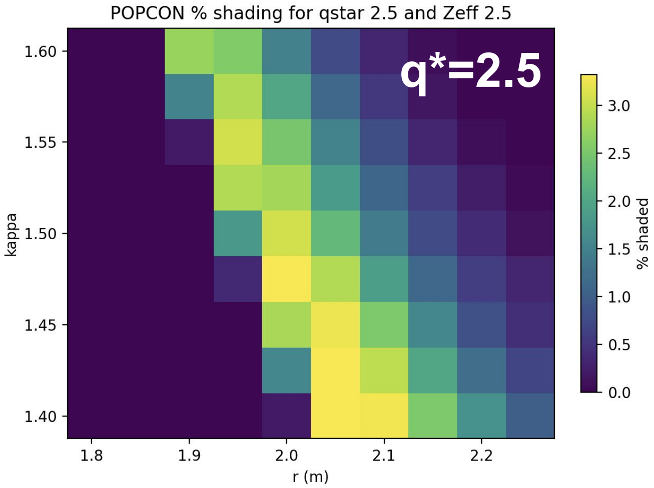
Maximize “accessible area”

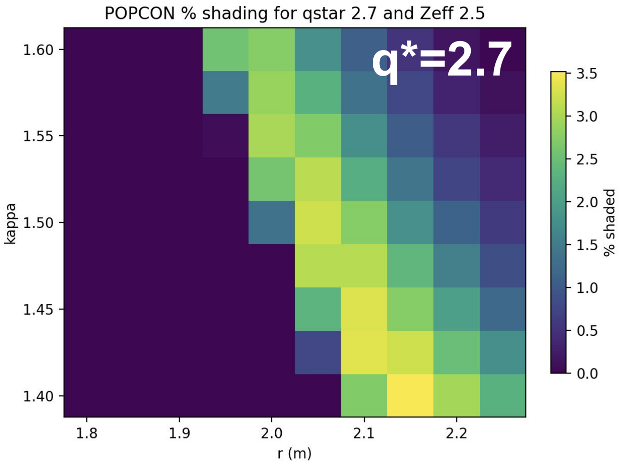


OpenPOPCON is used to select device parameters with a large possible operating space

For every POPCON, we calculated the “accessible area” metric, which measures how much of the (n_e, T_e) operating space meets our constraints







Increasing q*, fixed Z_{eff}=2.5

We picked an initial starting point within our POPCON to plug into higher fidelity codes to verify and begin design process.

Plasma **OP**eration **CON**tour Plot

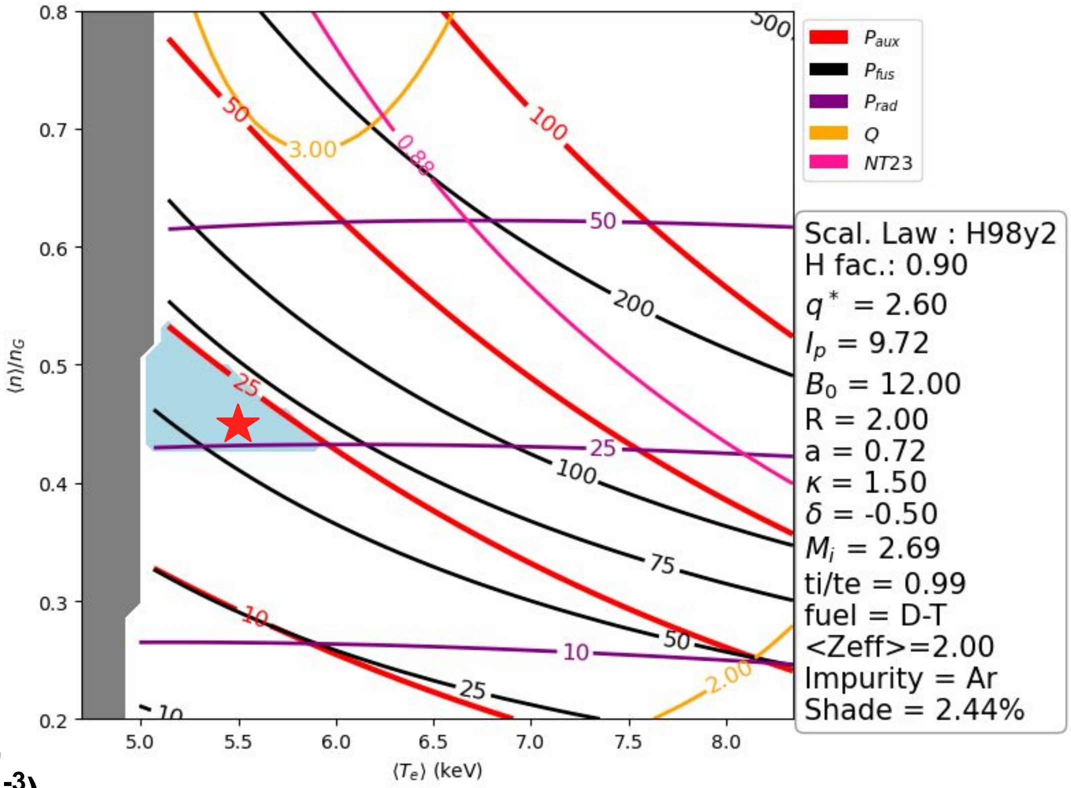
Conservative constraints chosen for a factor of safety

Initial Constraints:

- $Q > 2$
- $P_{aux} < 25$ MW
- $P_{rad} > 25$ MW

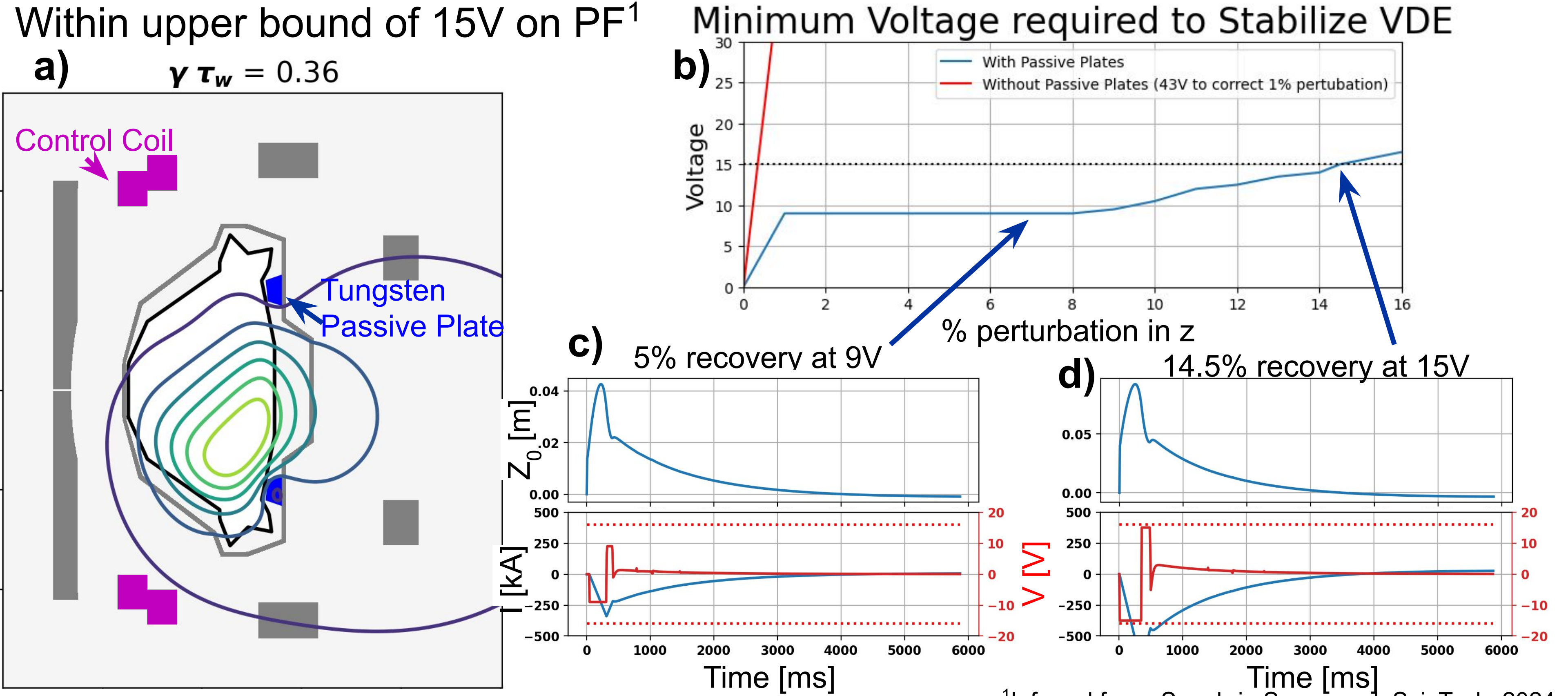
$n_G = 5.968 \times 10^{20}\ m^{-3}$

Starting Point at: ★ **($\langle T_e \rangle = 5.5$ keV, $\langle n_e \rangle = 2.7\text{e}20\ m^{-3}$)**



Tokamaker finds active VDE recovery from a max ~15% perturbation

Within upper bound of 15V on PF[1]



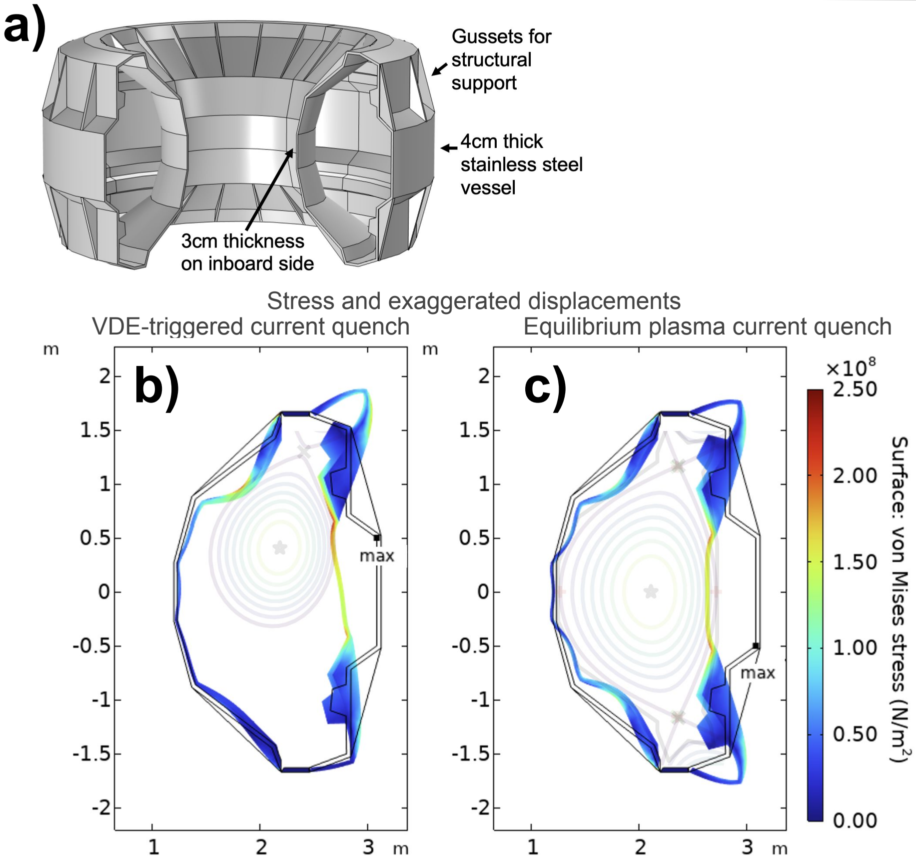
[1]Inferred from: Sanabria Supercond. Sci. Tech. 2024

[2] Hansen Comp. Phys. Com. 2024

COMSOL modeling indicates vacuum vessel will safely withstand disruptions

Disruptive current quench (CQ) simulations were run on Tokamaker to calculate eddy current JxB forces.

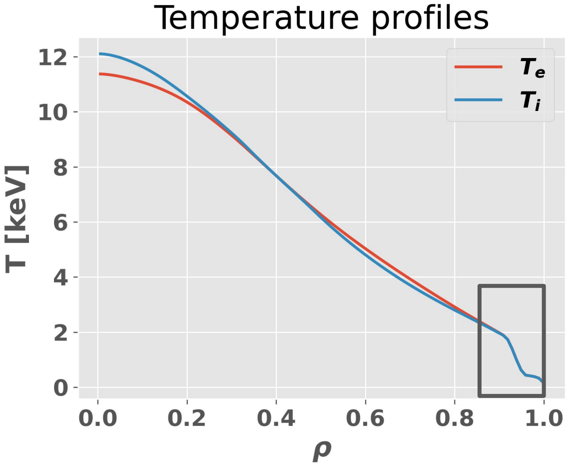
- **3 ms CQ times** were used following the ITPA disruptions database scaling[1]
- Max stress from VDE CQ = **249 MPa; 100,000 disruptions** before failure
- Max stress from central CQ = **230 MPa; 230,000 disruptions** before failure
- Using experimental steel fatigue studies[2] we conclude the VV will withstand CQ forces through its lifetime

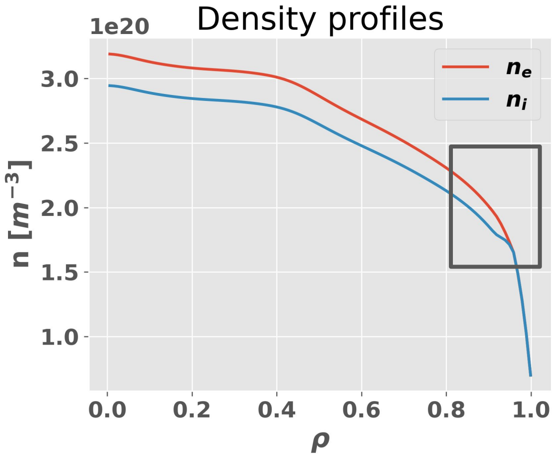


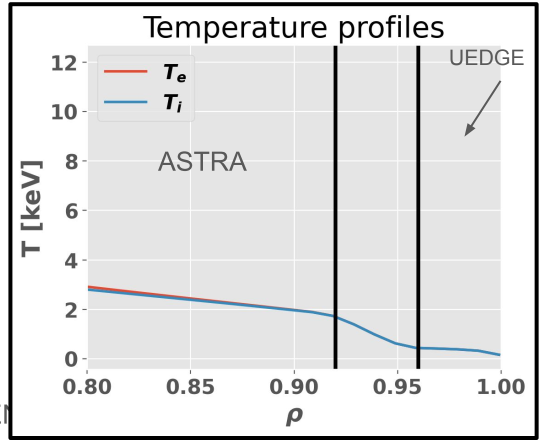
[1] Eidietis NF 2015
[2] Mohamad Mater. Sci. Eng. 2012

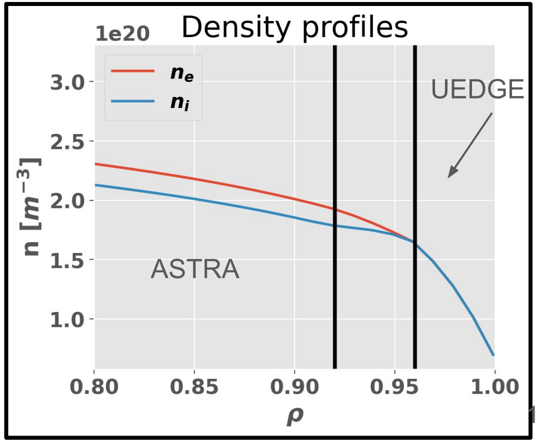
Transport codes using ASTRA[1] reveal Q > 1, and reasonable power balance

Plasma gain – Q	1.3
H factor (vs. H-mode) – H_{98}	0.64
H factor (vs. DIII-D NT) - H_{NT}	0.24
Fusion power – P_{fusion}	38 MW
Auxiliary power – P_{aux}	29 MW
Power thru SOL – P_{SOL}	13.5 MW
Radiated power – P_{rad}	26.8 MW
Effective ion mass – Z_{eff}	2.4









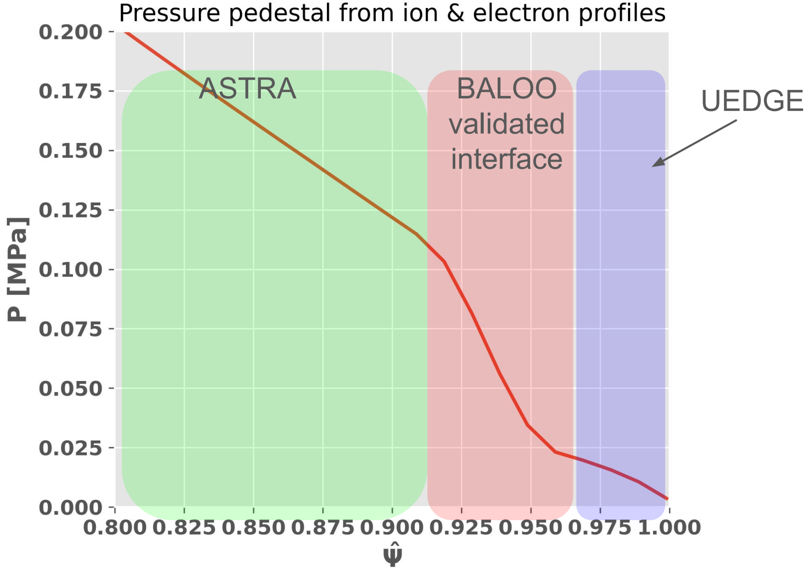
[1] Pereverzev and Yushmanov IPP Report 2002
[2] Fable PPCF 2013

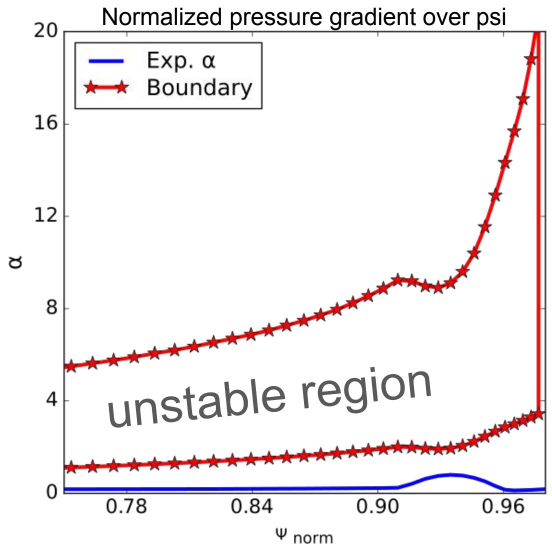
BALOO results suggest pedestal is well within available upper bound

ASTRA core profiles were matched with UEDGE edge profiles in scrape-off-layer power, and connected with a theorized pedestal.

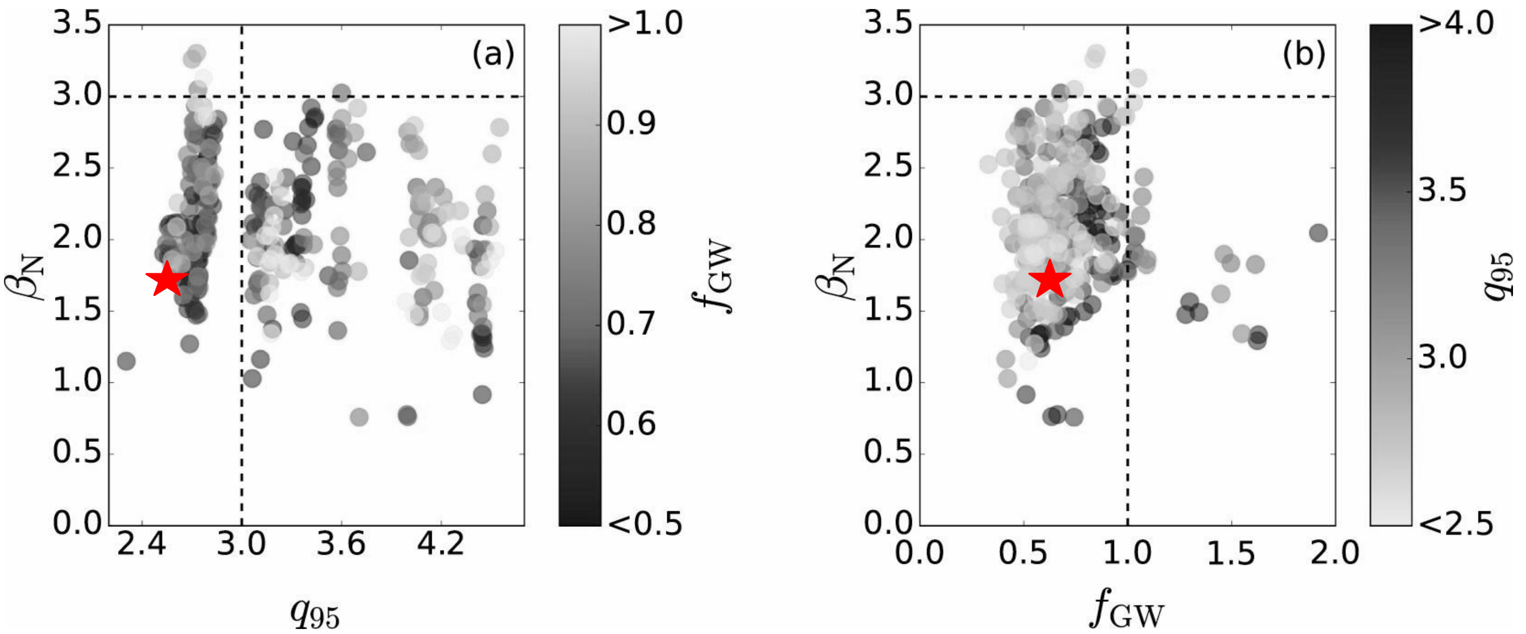
BALOO[1] (infinite-n kinetic ballooning mode stability code) shows that this pedestal is well-within the stability limits.

[1]R. L. Miller, et Al. *Phys. Plasmas* 1 April 1997





CENTAUR falls within DIII-D NT campaign operational points[1]



At this operating point, the main stability challenge at high normalized current will be the resistive tearing instability (similarly to SPARC).

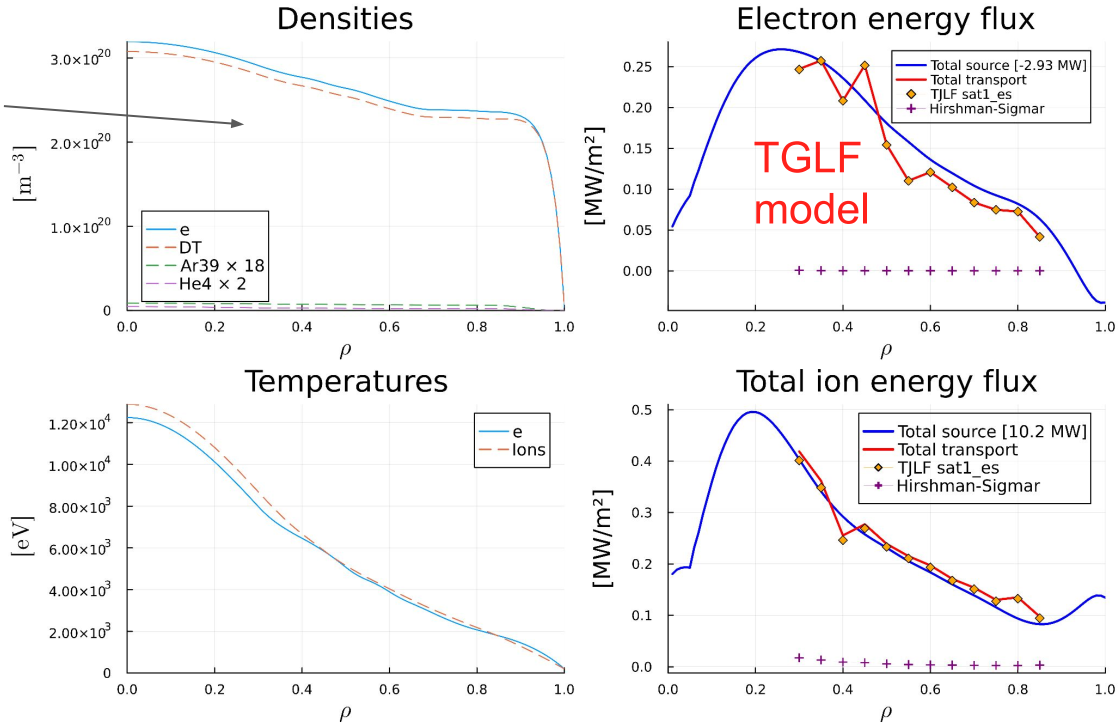
Operating point ★ at β_N = 1.65, f_{GW}=0.6, q_{95}=2.57

[1] Paz-Soldan NF 2024, plots adapted with permission

FUSE transport provides more optimistic case in comparison to ASTRA results

FUSE[1] results are comparable to ASTRA, but edge conditions differ, giving an alternative case

Physics gain – Q	2.95
H factor – H_{98}	0.834
Fusion power – P_{fus}	32.5 MW
Auxiliary power – P_{aux}	11 MW
Power thru SOL – P_{SOL}	7.23 MW
Radiated power – P_{rad}	11.5 MW
Effective ion mass – Z_{eff}	2



[1] Meneghini arXiv 2024

Power Handling and Edge Integration

Eliot Felske, Freddie Sheehan, Mohammed Haque,
Samuel Freiberger, Shreyas Seethalla
Mentors: Oak Nelson, Chris Hansen
Filippo Scotti (LLNL), Andreas Holm (LLNL)

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Lawrence Livermore National Laboratory

UEDGE and COMSOL used to form overall topology edge physics, and divertor heat load capabilities

Critical design parameters

- Match edge profile to core profile
- Construct divertor plate geometry and edge profiles using UEDGE
- Model heat transport in PFCs using COMSOL to ensure divertor plates can withstand pulse heat fluxes

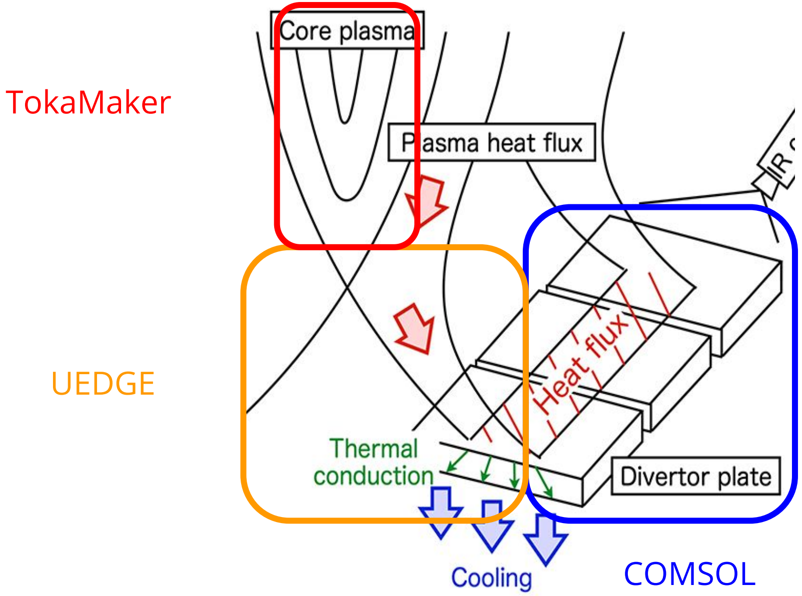
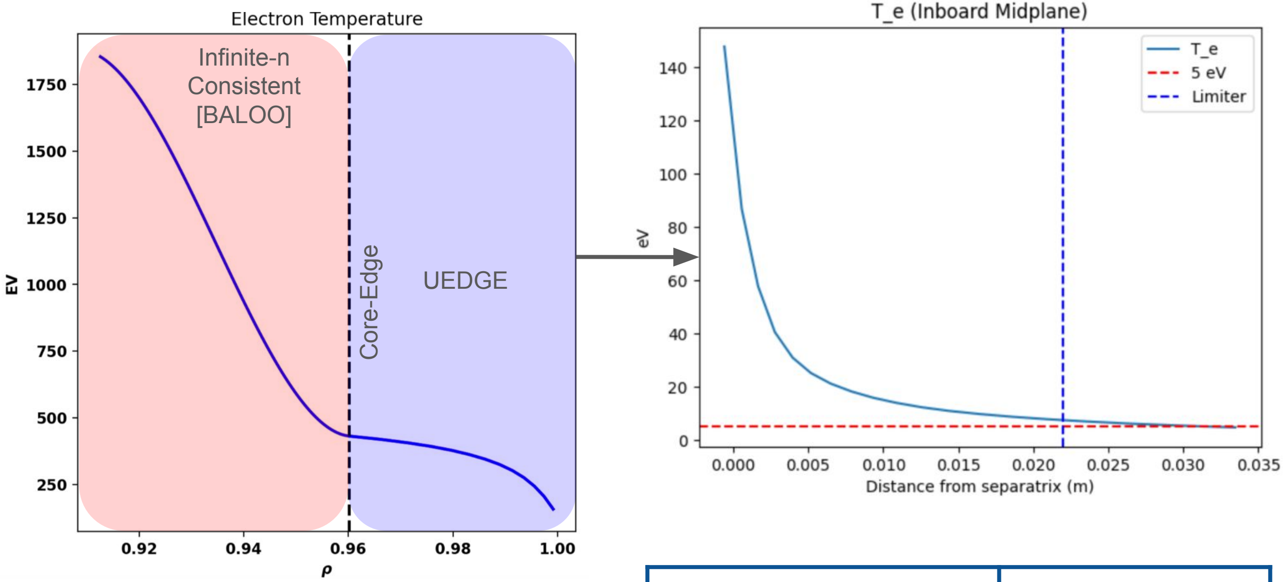


Fig. from Y. Hayashi Fusion Eng. Des. 2021

Characterizing Radial Heat Falloff

- Set radially stepped, poloidally constant diffusivity coefficient profiles around separatrix
- λ_q = 1.1 mm from exponential fit to UEDGE case T_e falloff
- Inboard midplane limiter 2.1 cm from LCFS
- T_e = 7.01 eV at midplane wall



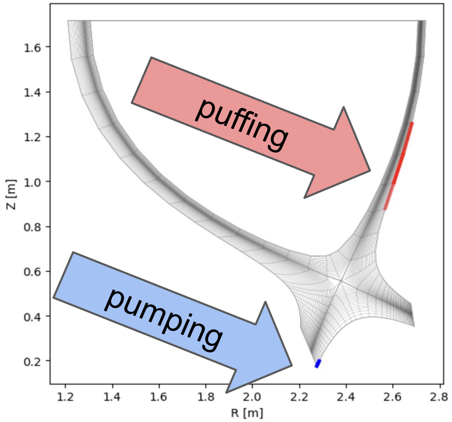
Scaling	λ_q (mm)
Eich H-Mode[1]	0.36
Horacek L-Mode[2]	4.8

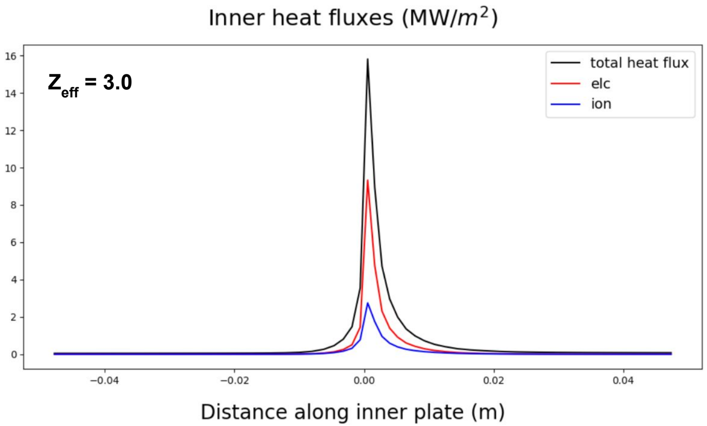
[1] Eich NF 2013
[2] Horacek NF 2020

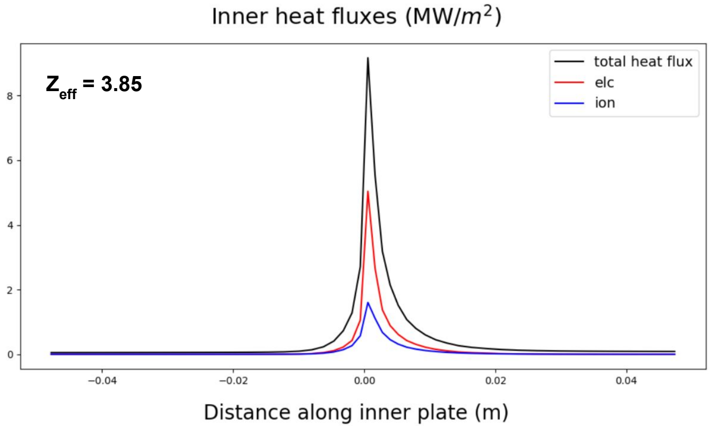
Neon impurities are used to radiate power

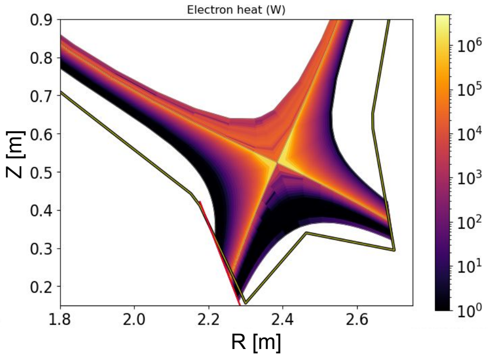
- Baseline case Ne impurity: **2.0%** just above X-point
- Z_{eff} (fixed fraction model): **3.09** - can be tuned with pumping (blue) / puffing (red)
- f_{rad} in divertor: Inner = **0.213**, Outer = **0.211** (line integrated radiation over total power)
- Z_{eff} = 3.85 brings inner plate below 10 MW/m^2 steady-state material limit

Gas puffing and pumping location



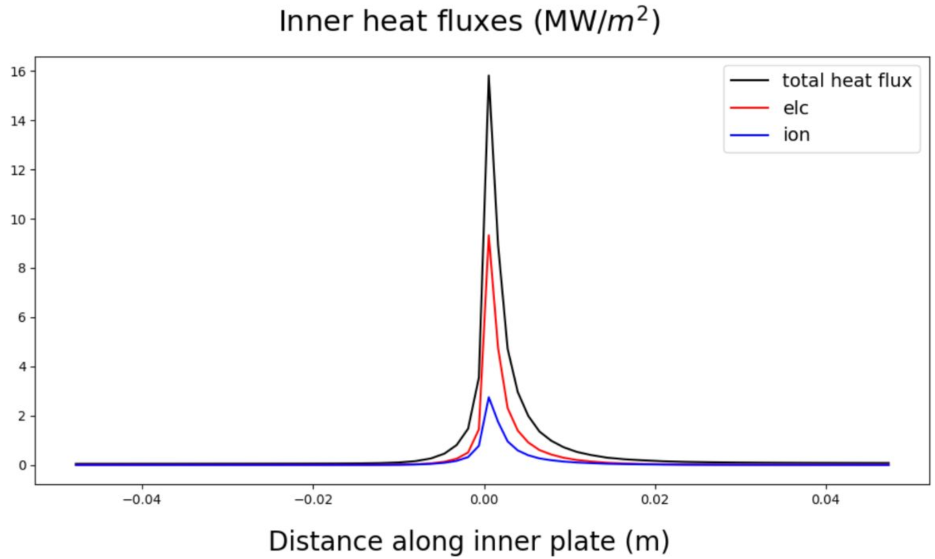


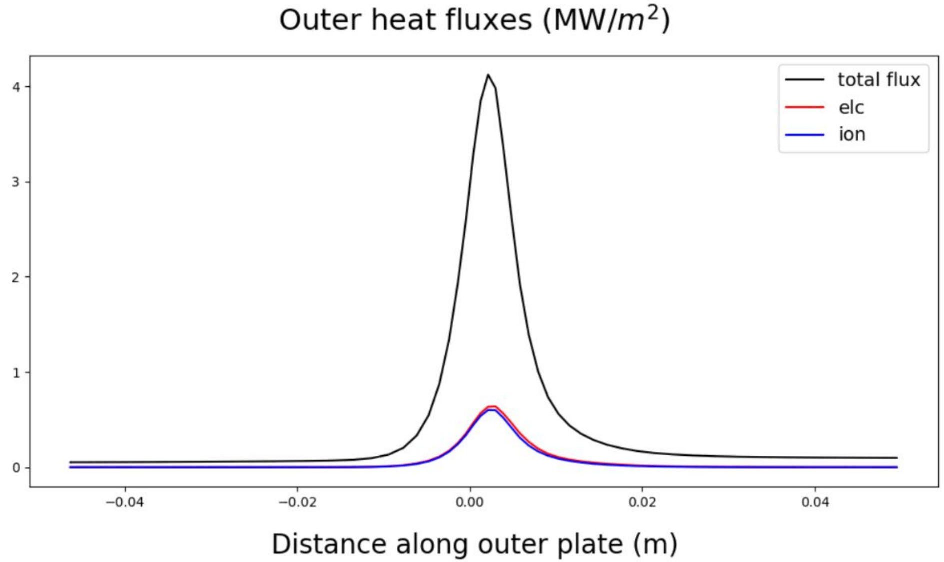




Final values are near to steady-state limits

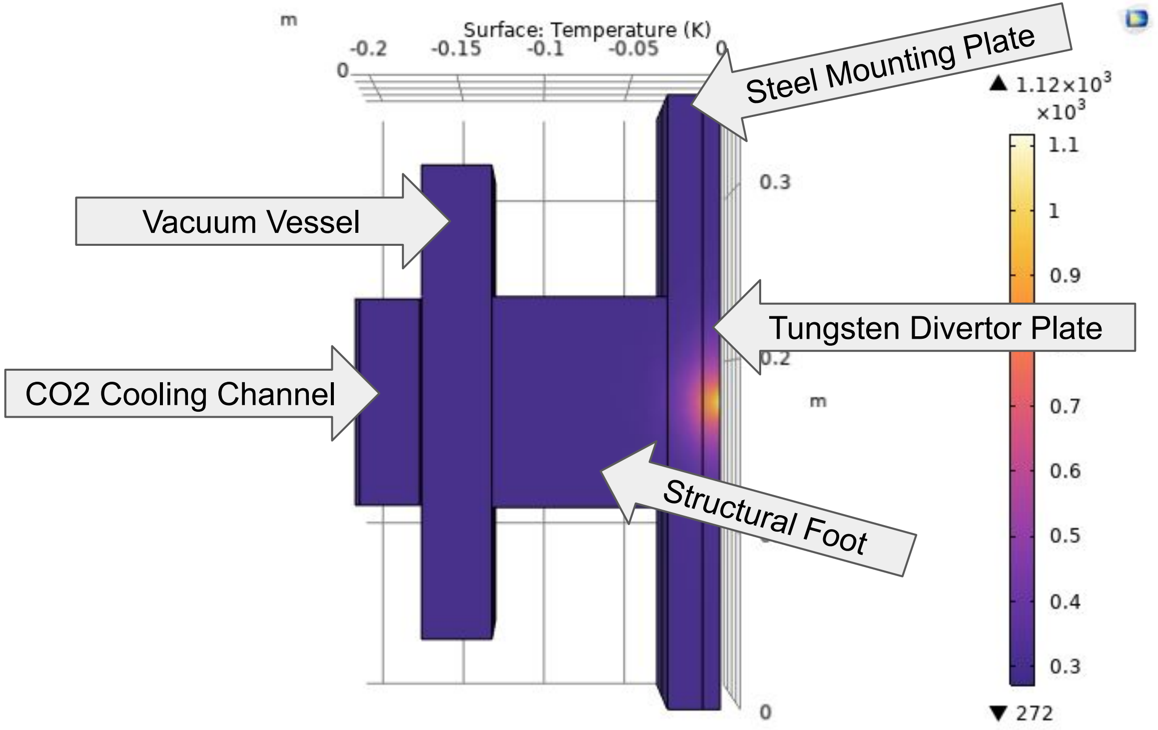
- High Ne seeding, Z_{eff}: **3.09**
- High field side to low field side heat flux ratio: **3.95**
- P_{rad} between ρ = 0.92 and ρ = 1.0: **2.72 MW** (39.8%)
- Inner plate angle: **45°**, Outer plate angle: **58°** (mitigates backstreaming in electron density)
- Peak heat flux: Inner = **16.03 MW/m^2**, Outer = **4.05 MW/m^2**





3D COMSOL models show good safety margin in divertor

- Divertor plate model geometry corresponds to simplified radial build
- Peak flux of ~16 MW/m^2 corresponds to current operating point
- Tungsten re-crystallizes at ~1800 K[1]
- Max temperature on plate from 30s pulse with 10s flat-top of ~1100 K by conservative estimates



[1]Suslova Sci. Rep. 2014

Power Handling & Edge Modeling Summary

- λ_q is within our range of physically relevant scaling calculations (1.1 mm)
- Divertor plate geometry is optimized for **strikeline spreading** and **neutral backstream mitigation**
- We are able to radiate enough heat with **high impurity seeding** in the edge region to keep tungsten divertor plates well below than their recrystallization temperature
- Divertor plate heating **does not necessitate** strike point sweeping or other advanced divertor designs

Neutronics

Daniel Burgess, Jake Halpern, Evan Bursch
Mentors: Matt Tobin, Benedikt Zimmermann

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The Fu Foundation School of Engineering and Applied Science

Neutronics optimizes shielding for neutron damage and heating

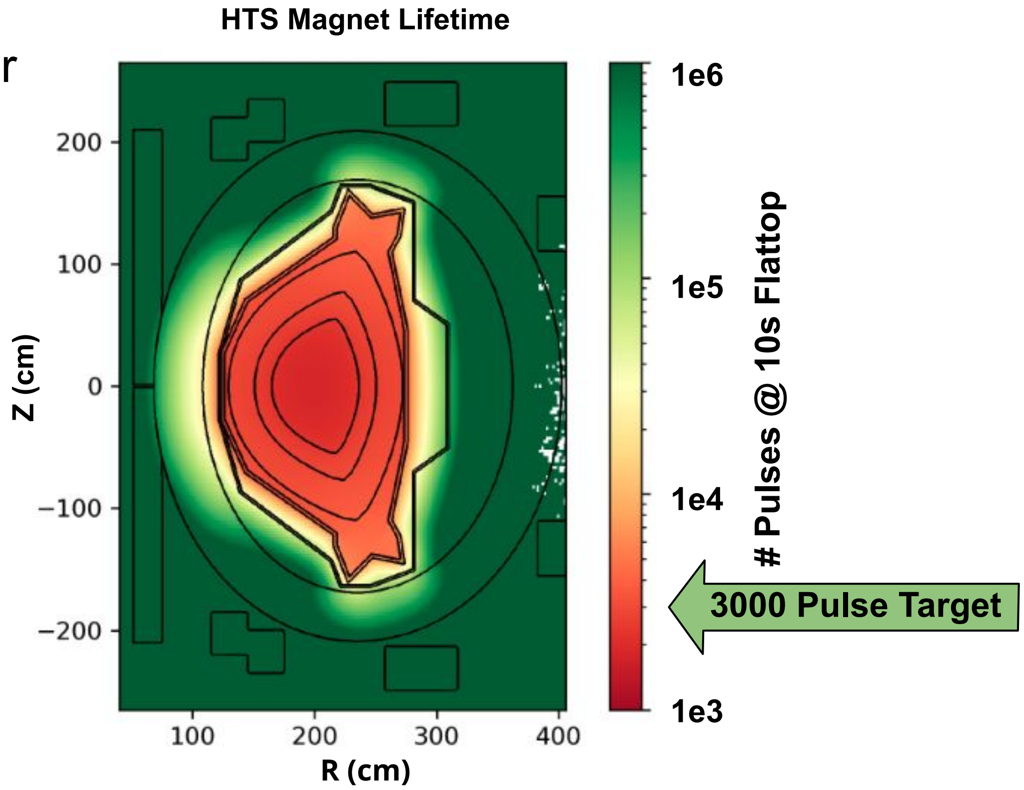
- Design Targets:
  - Neutron damage: high energy neutron fluence allows for **3000 DT full-power pulses**[1]
  - Neutron heating: superconducting magnets to **stay under 30K**, based on magnet current values, **to avoid quenching**[1]
- Tools
  - OpenMC: open source Monte Carlo photon/neutron simulation code[2]
  - Calculations for neutron flux, displacement per atom (DPA), and heating for HTS magnets, vacuum vessel, and shielding
  - Final modeling was done with **10^7 particles** (app. 100 cpu-hours)
  - These inform neutron shielding thickness and material choice

OpenMC

[1]Fischer Supercond. Sci. Technol. 2018
[2]Romano Ann. Nucl. Energy 2015

Magnet lifetime does not strongly direct shielding considerations

- All HTS components are an order of magnitude above 3000 pulse target lifetime
- This corresponds to 3×10^{22} neutrons/m^2
- Pulsed nature of device keeps total neutron fluence low for damage/activation



B_4C was chosen to minimize HTS neutron heating

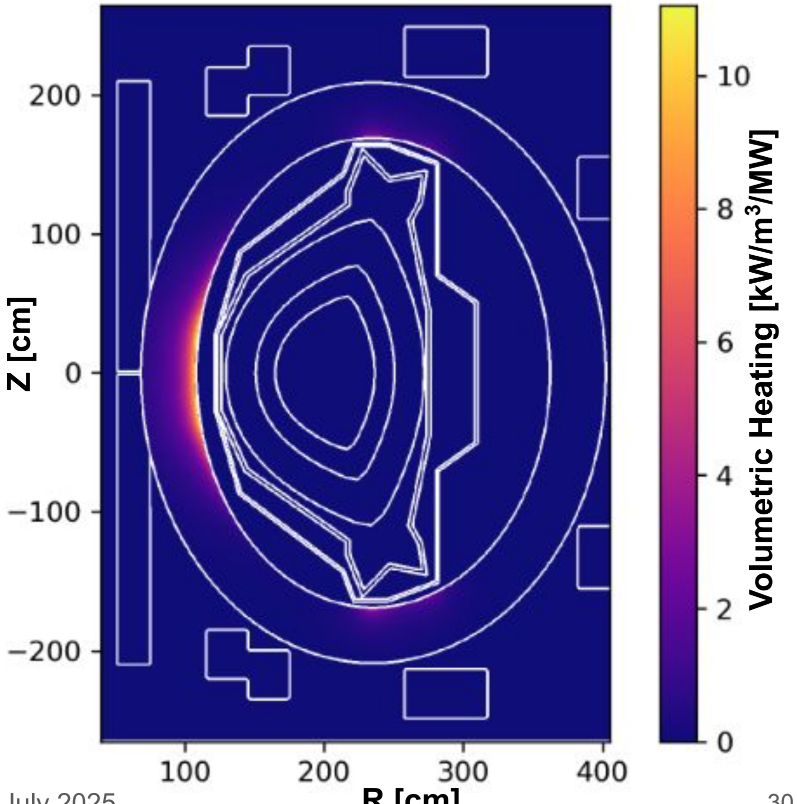
- To avoid quenches, the HTS cannot be heated above 30K during a shot
- B_4C reduces neutron heating to 42.2% of vacuum case
- Other materials did not limit heating to acceptable operating regimes

Shielding Material	Percent of Vacuum Neutron Heating
B_4C	42.2%
WC	58.5%
HDPE (high density polyethylene)	73.5%

Inboard midplane TF heating sets shielding requirements

- Max heating strongly localized to inboard side midplane
- Under the chosen operating conditions, the target max volumetric heating in the HTS was 12.2 kW/m^3/MW
- The chosen configuration had a max heating in the HTS of 8.33 kW/m^3/MW

Local Volumetric Heating w/ 12 cm B4C (kW/m^3/MW)



Radial build constrained by required shielding at inboard midplane

Inboard midplane shielding (purple) = 12 cm

Inboard midplane VV (gray) = 3 cm (4 cm elsewhere)

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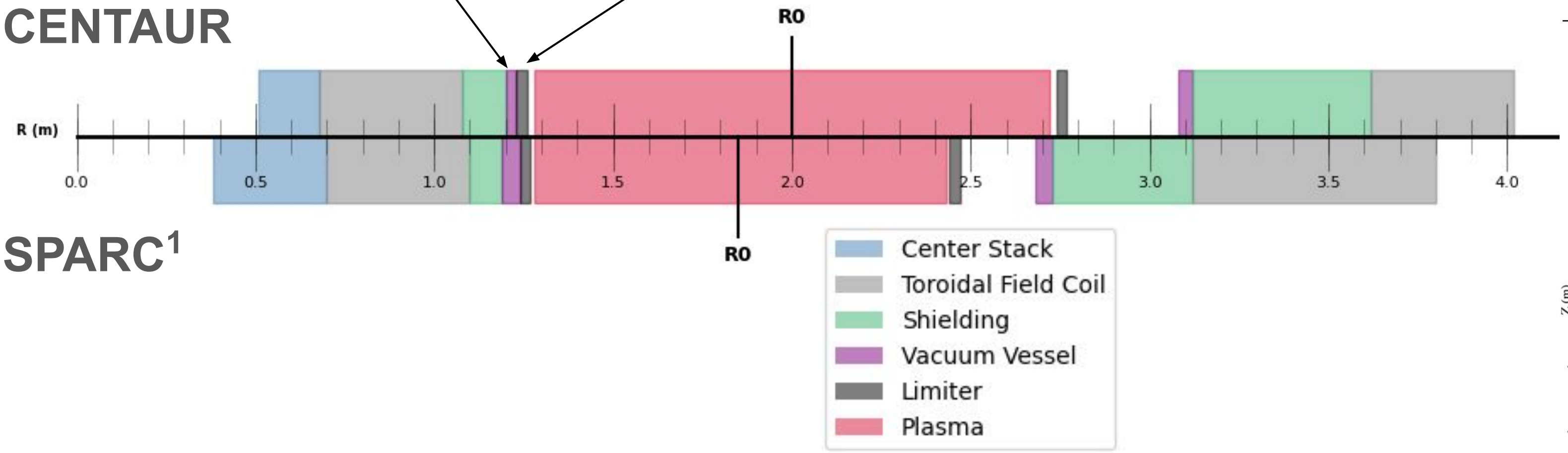
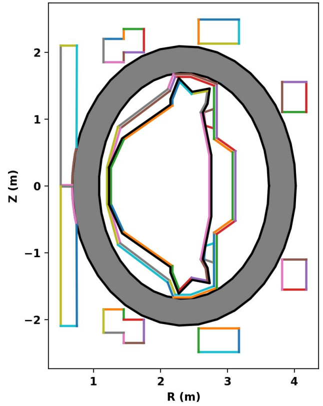
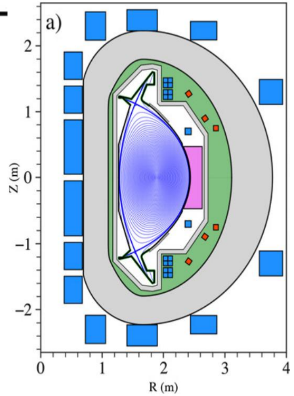
SPARC[1]



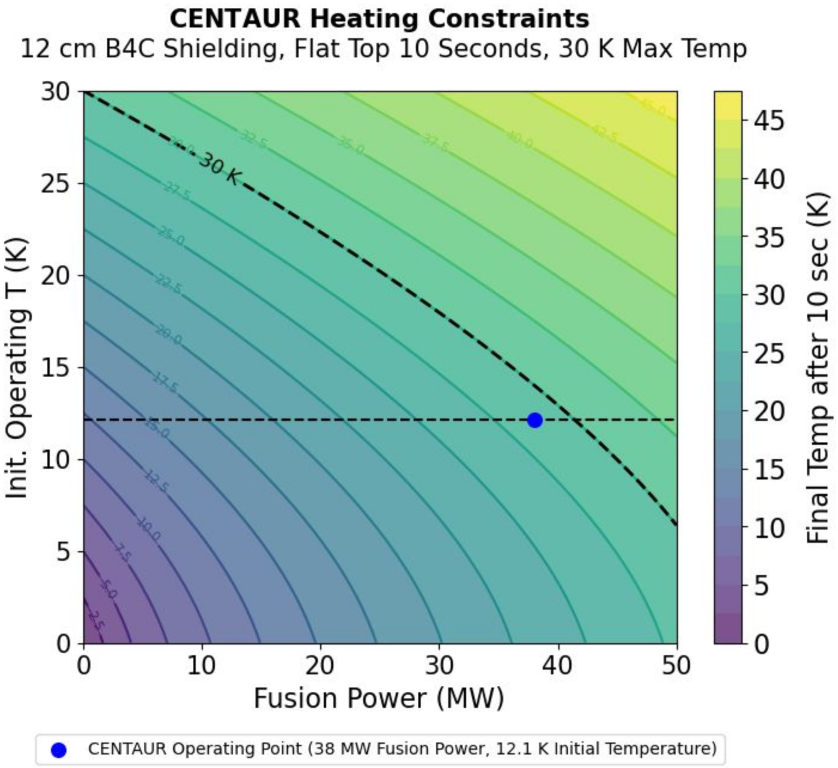




[1] SPARC Dimensions and Fig. from Rodriguez-Fernandez NF 2

TF heating contour plot identified viable operating points

- Scanning flat-top, shielding, fusion power, and HTS operating temperatures informed design choice
- Temperature dependent HTS specific heat from Drotziger[1]
- The goal was to have 10 s flat-top at 38 MW of fusion power
- This led to 12.1 K initial HTS temperature assuming 12 cm of B_4C shielding



[1] Drotziger IEEE Trans. Appl. Supercond. 2016

Shielding is effective for magnet heating and lifetime considerations

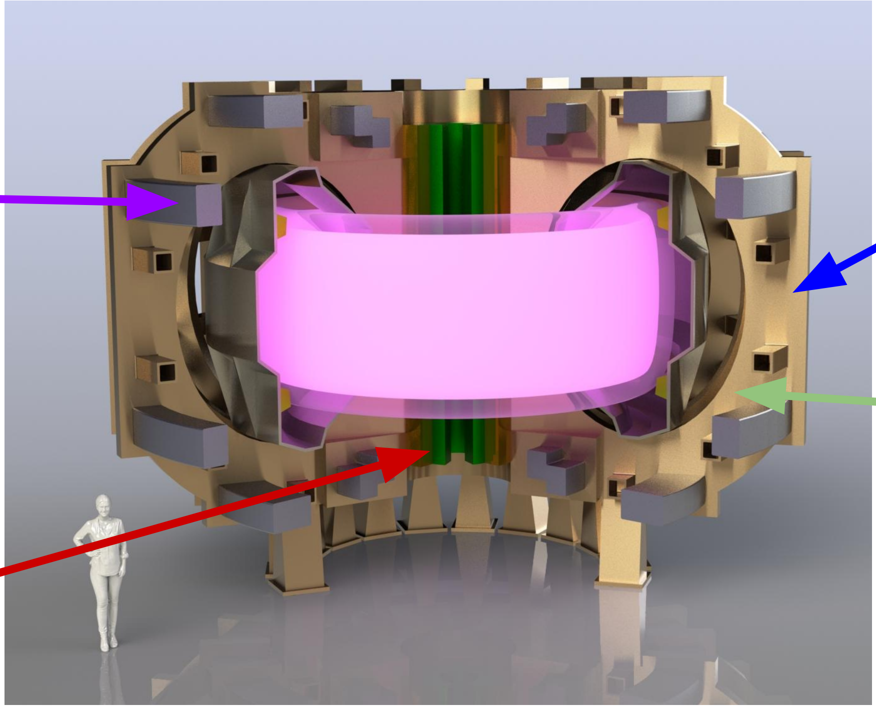
- Lifetime of the magnets based on Monte Carlo neutron displacement per atom simulations are predicted to be **well above the 3000 full power DT shot limit**
- **B_4C shielding** was chosen due to it being the most effective shielding material considered
- By operating the **HTS at 12 K with 12 cm of B_4C**, the HTS magnets are below quenching temperatures during 10 second flat-top at full 38 MW fusion power
- At our **minimum operating temperature of 8 K,** we can tolerate a max of 42.3 MW of fusion power for a ten second flattop

Magnets

Sophia Guizzo, Kalen Richardson, John Labbate

Mentor: Haley Wilson

CENTAUR achieves high-field with HTS magnets



Solenoid flux requirement set by plasma core scenario

- Startup flux estimated with analytic formulas, using core plasma parameters[1]
    - Full, time-dependent start-up simulation is needed
- Plasma loop voltage estimated using Spitzer resistivity with classical and neoclassical corrections
- Experiments on DIII-D demonstrate that poloidal field coils **detract, rather than add,** to the available flux **for NT**[2]

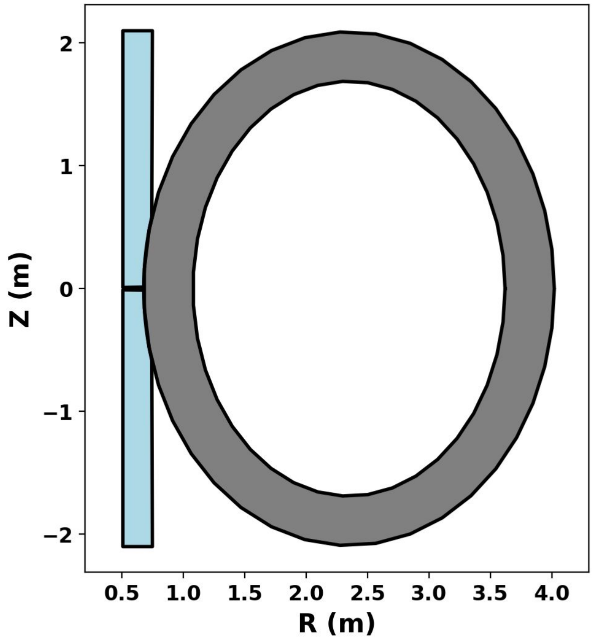
Startup flux	40.9 Wb
Loop voltage	0.29 V
10 s flat-top flux	2.9 Wb
Flux requirement	**43.8 Wb**

[1] Sugihara J Nucl. Sci. Technol. 1982
[2] Leuer Presentation 2020

“Curved” solenoid required to achieve necessary flux swing

Pit VIPER J_{eng}	~95 A/mm^2
Maximum field on HTS	~25 T
Flux swing	~52 Wb
Flux swing with PFs	~48 Wb

- Leverages PIT VIPER technology, which has demonstrated **50 kA per cable** at **25 T** and **20 K**[1]
- **Exceeds 43.8 Wb flux** required for plasma startup and flat-top
- Extra windings away from midplane produce ~30% of the flux



[1] Sanabria Supercond. Sci. Technol. 2024

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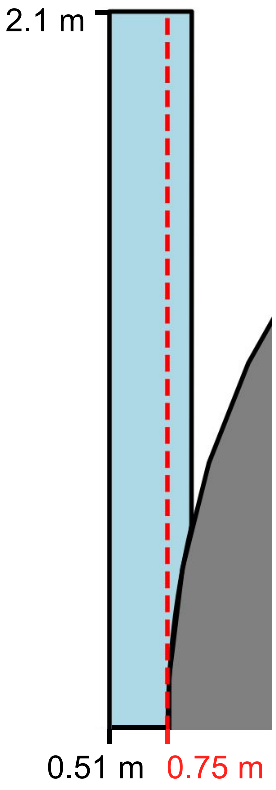


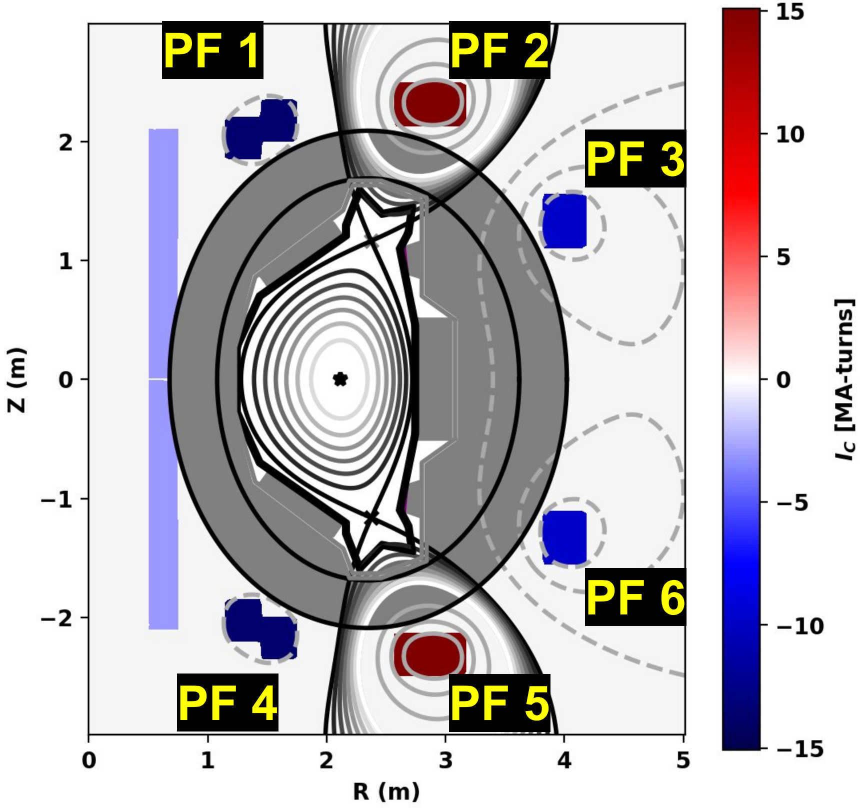
[1] Sanabria Supercond. Sci. Technol. 2024

PIT VIPER poloidal field coils achieve equilibrium currents

- Number of turns per poloidal field coils set by equilibrium currents and 50 kA per cable PIT VIPER[1] limit
- Coils sized according to number of turns

	Equilibrium Current (MA-turns)	Number of Turns
PF 1, 4	-13.54	271
PF 2, 5	15.07	302
PF 3, 6	-9.88	198

[1] Sanabria Supercond. Sci. Technol. 2024



Toroidal field coils meet core plasma requirements

Toroidal field coil achieves desired field with a **maximum HTS current density of 440 A/mm²** → 2x margin of safety

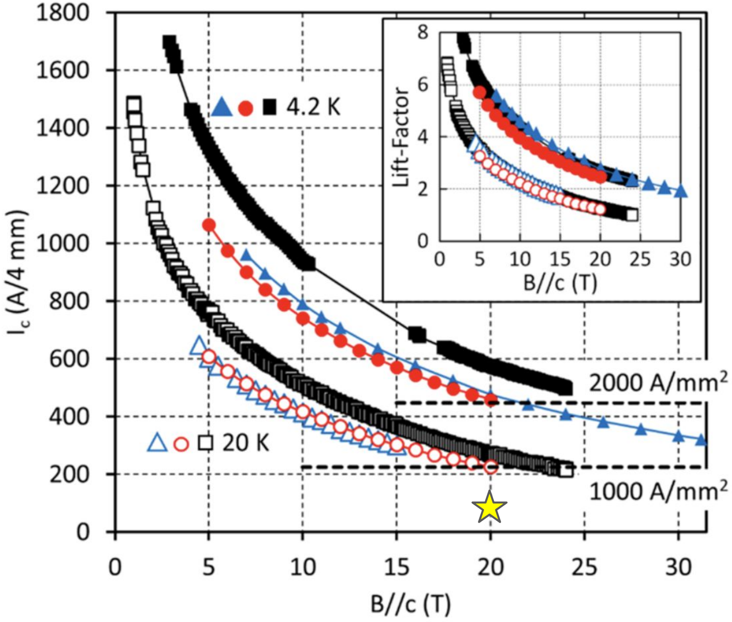
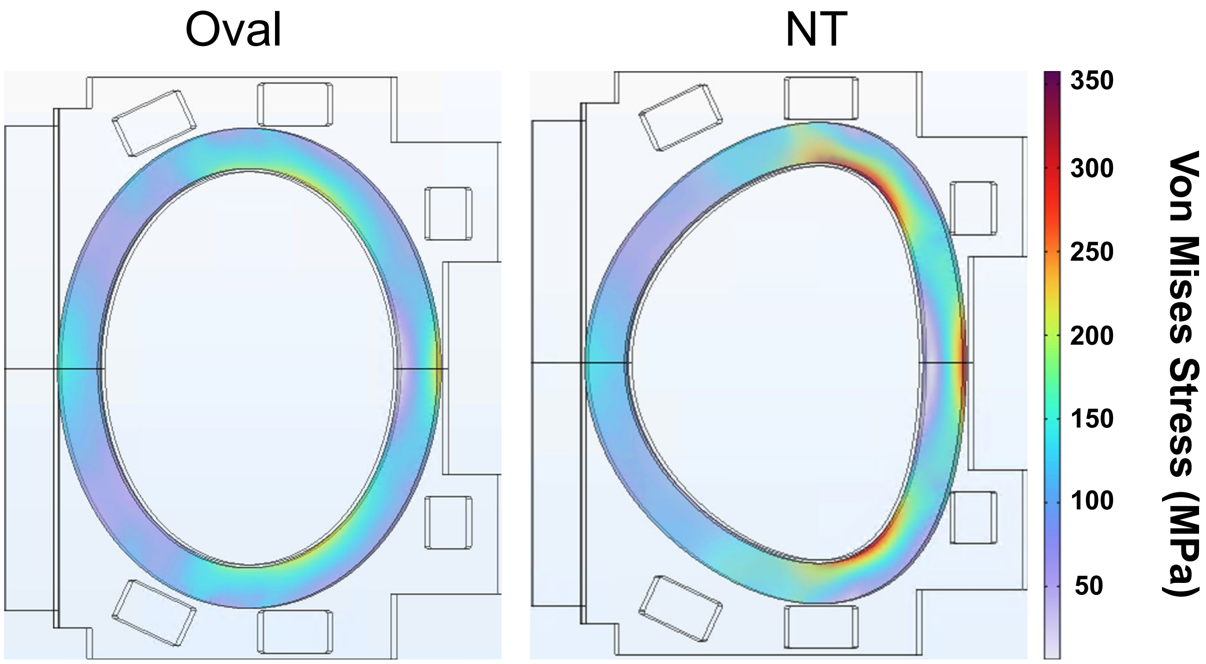


Fig. from Molodyk Sci. Reports 2021

# of TFs	18
Maximum ripple	0.34 %
Field on-axis	10.9 T
Maximum field on TF	21 T
# of pancakes per TF	15
# of turns per pancake	16

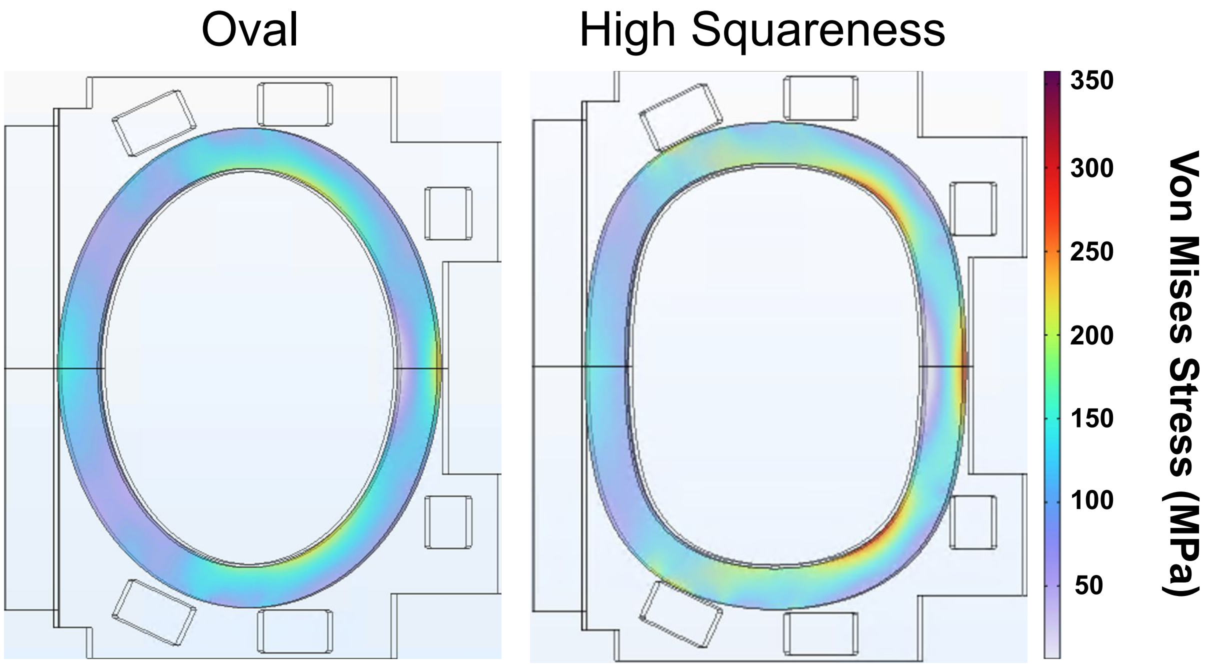
COMSOL stress simulations identify optimal magnet shape

- Princeton D magnet topology (not shown) results in large **magnet volumes** and **unused space** for an NT device
- Oval magnet shape shows **lower peak stress** than a conformal, NT magnet shape

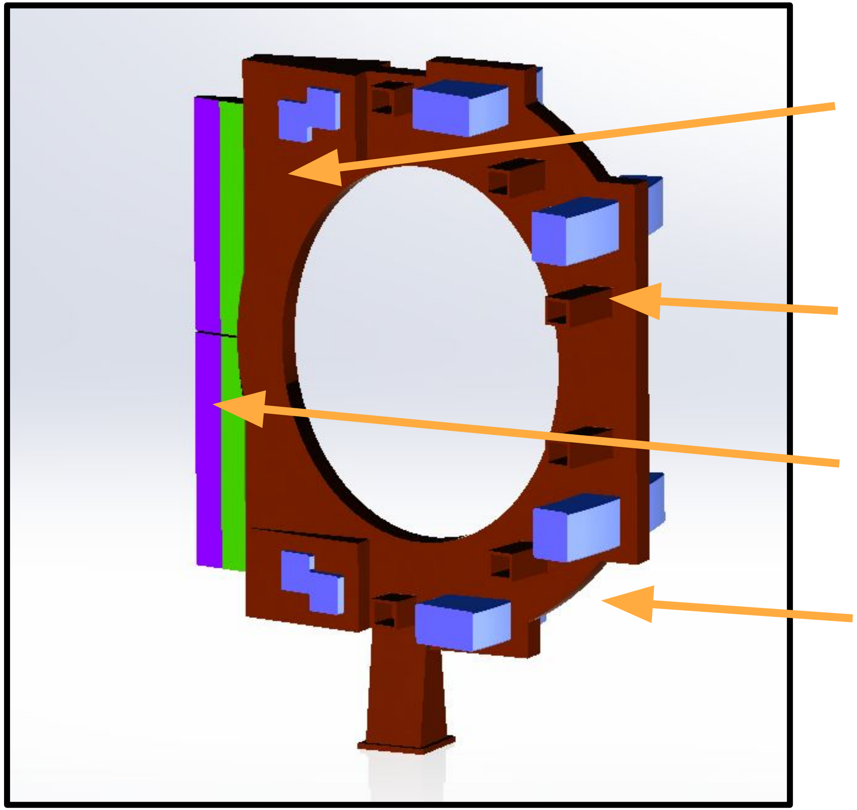


COMSOL stress simulations identify optimal magnet shape

- **Oval magnet** has **lower peak stresses** and requires less magnet volume
- Elongation chosen to accommodate vacuum vessel

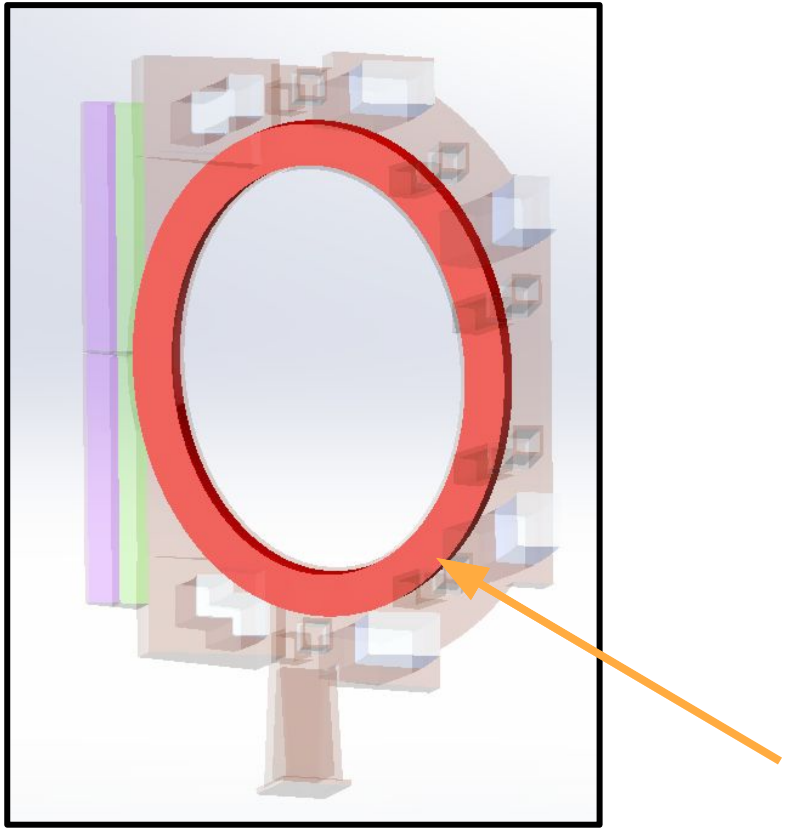


Stress simulations inform structural support design



- Structure providing mechanical connection between CS and TF to combat inward radial force
- Brackets between toroidal field coils prevent overturning
- Plug in central solenoid combats large radial force at zero current
- Structure removed from areas of low stress in initial simulations
- TF coil embedded in Nitronic 40 case

TF stress simulations inform structural support design

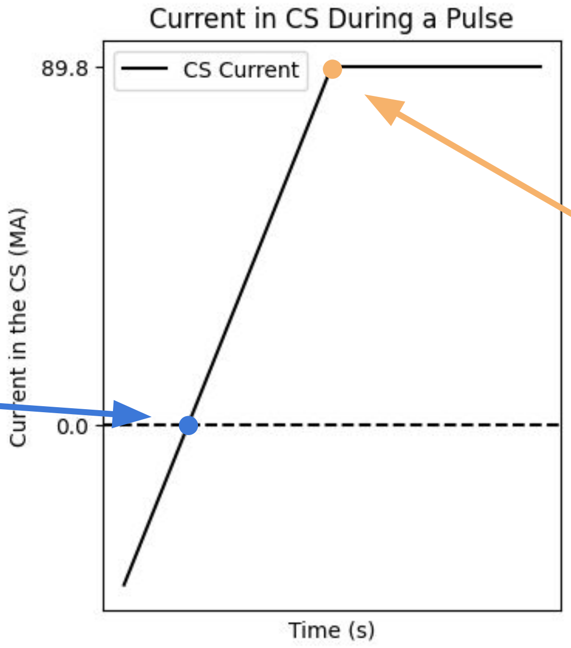


- Structure providing mechanical connection between CS and TF to combat inward radial force
- Brackets between toroidal field coils prevent overturning
- Plug in central solenoid combats large radial force at zero current
- Structure removed from areas of low stress in initial simulations
- TF coil embedded in Nitronic 40 case

Two loading conditions considered for stress simulations

Loading Condition 1 (LC1)

- No CS magnetic field
- Full TF, PF,
  plasma magnetic field



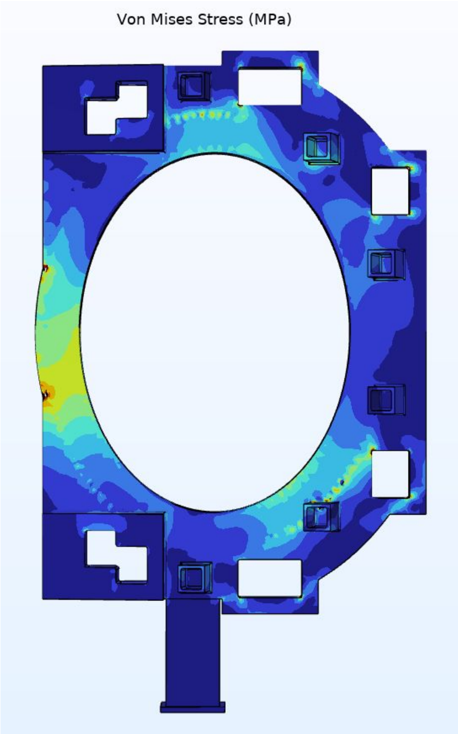
Loading Condition 2 (LC2)

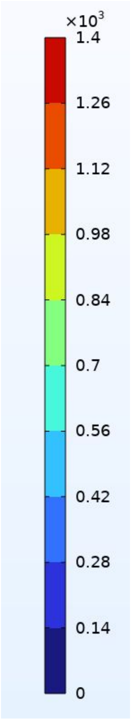
- Full CS, TF, PF,
  plasma magnetic field

These two loading conditions are expected to yield the largest stresses on magnet and structural components

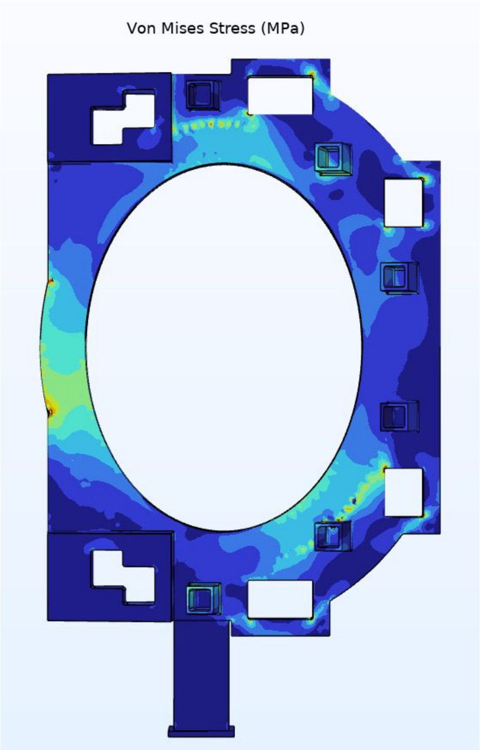
Stresses below yield strength of Nitronic 40 TF case

No CS field, full TF, PF, plasma B field (LC1)



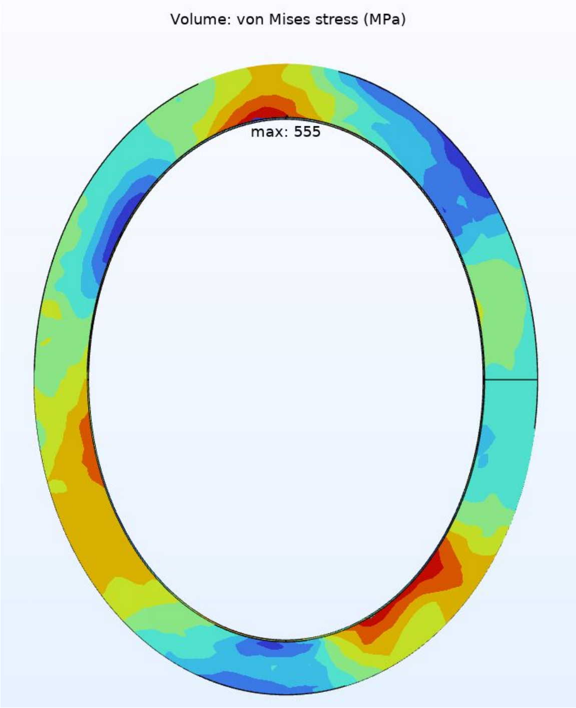


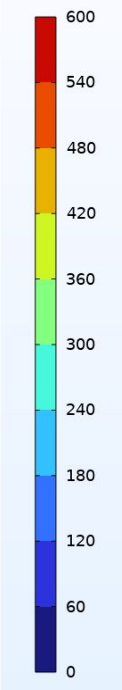
Full TF, CS, PF, plasma B field (LC2)



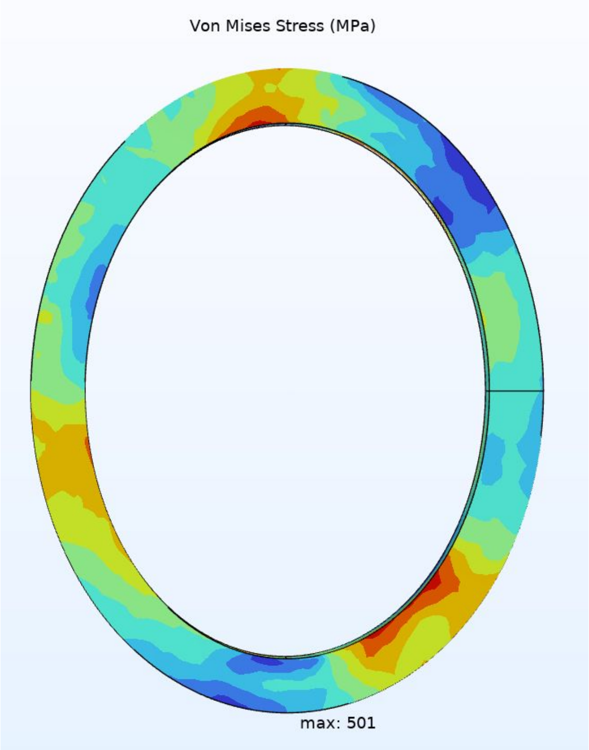
Stresses below limit for HTS pancake TF magnet

No CS field, full TF, PF, plasma B field (LC1)





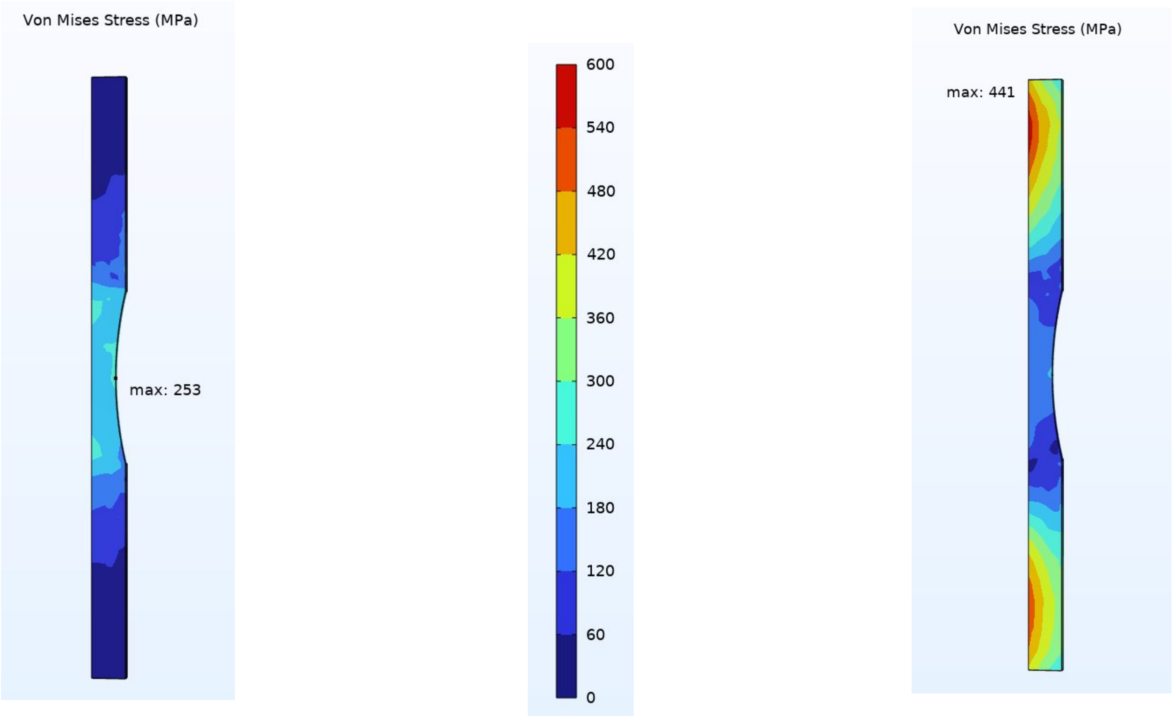
Full TF, CS, PF, plasma B field (LC2)



Stresses below limit for PIT VIPER CS magnet

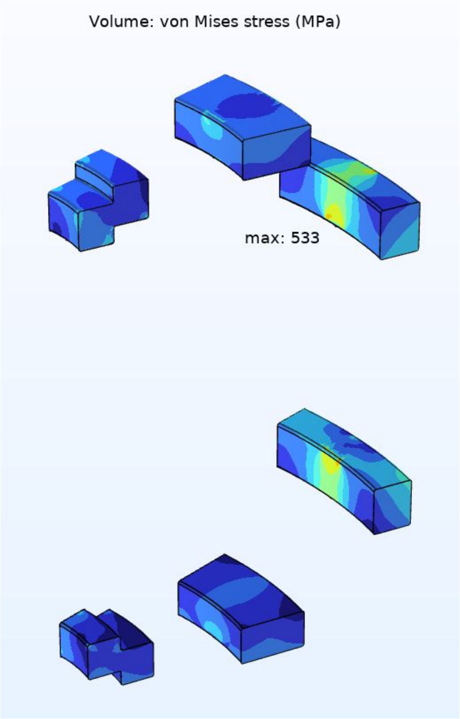
No CS field, full TF, PF, plasma B field (LC1)

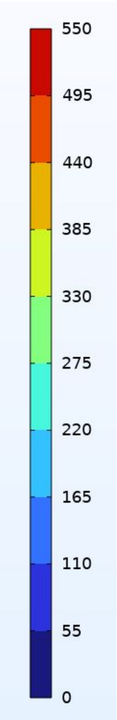
Full TF, CS, PF, plasma B field (LC2)



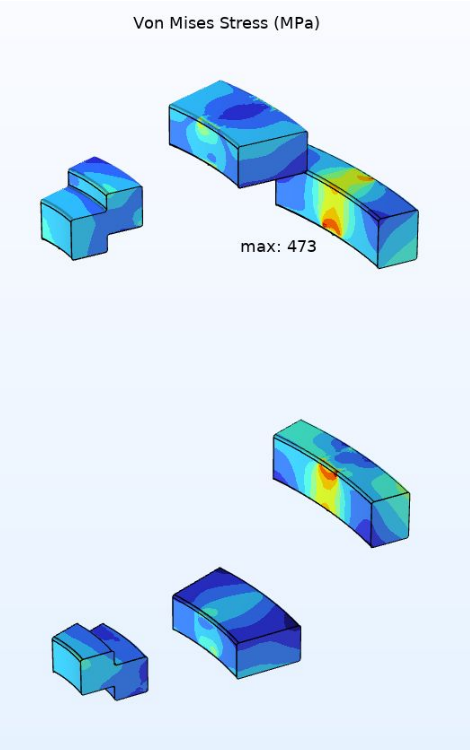
Stresses below limit for PIT VIPER PF magnet

No CS field, full TF, PF, plasma B field (LC1)

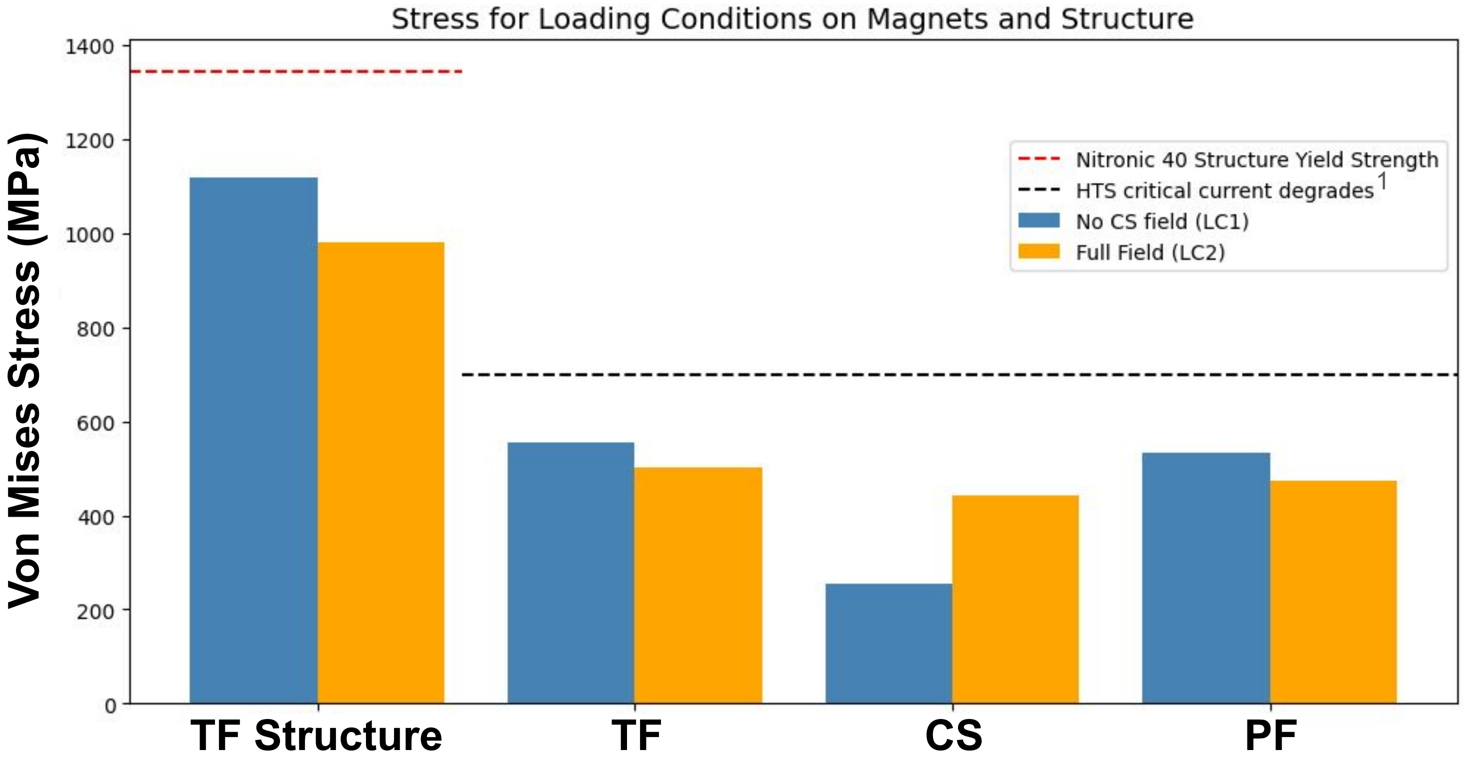




Full TF, CS, PF, plasma B field (LC2)



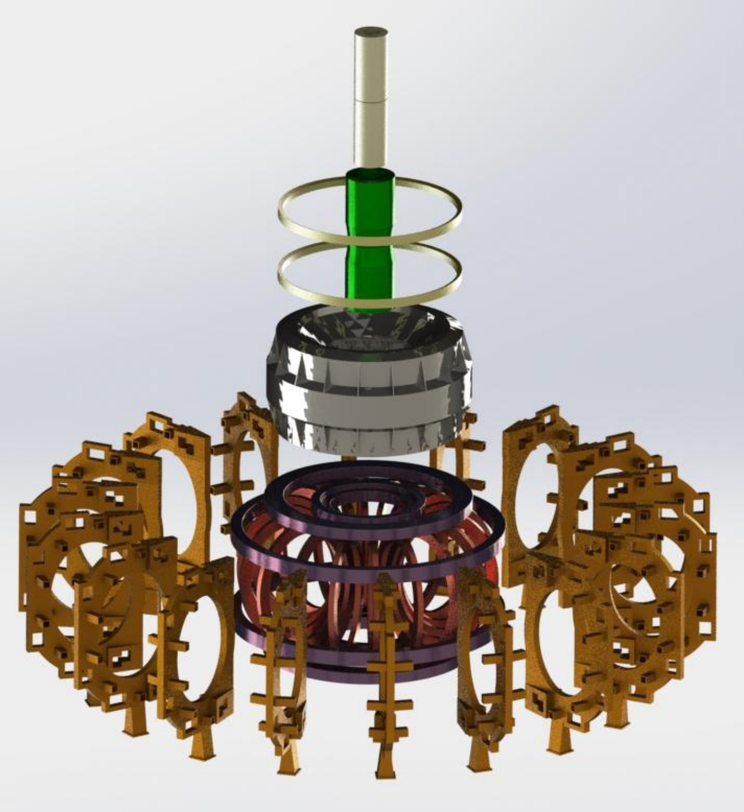
Magnets and structure will not fail during operation



[1] Barth Supercond. Sci. Technol. 2015

Magnet systems meet core requirements and withstand forces

- Superconducting magnetic structure
  - HTS wound TF coil
  - PIT VIPER CS and PFs
- On-axis magnetic field, CS flux requirements, and shaping requirements all are satisfied
- Structure designed and optimized to support magnetic forces
- Simulated all supporting structures and plasma current for extrema scenarios



Economics

Javier Chiriboga, Rohan Lopez, Mel Russo
Nathaniel Chen (PU)
Mentor: Ian Stewart

COLUMBIA | ENGINEERING
The Fu Foundation School of Engineering and Applied Science

PRINCETON UNIVERSITY

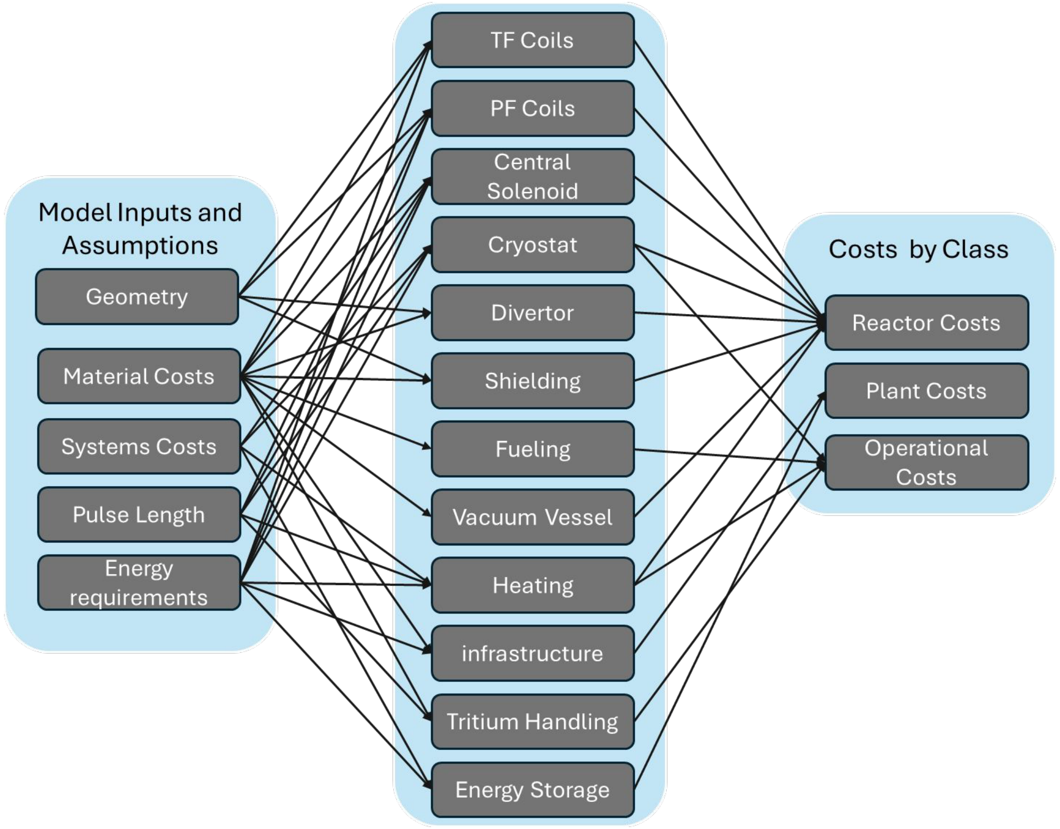
CENTAUR costing model improves upon MANTA

Goal: Determine overnight device cost and economic feasibility of CENTAUR's design

The MANTA costing model enabled wholistic system costing in parallel with the design process

CENTAUR costing model:

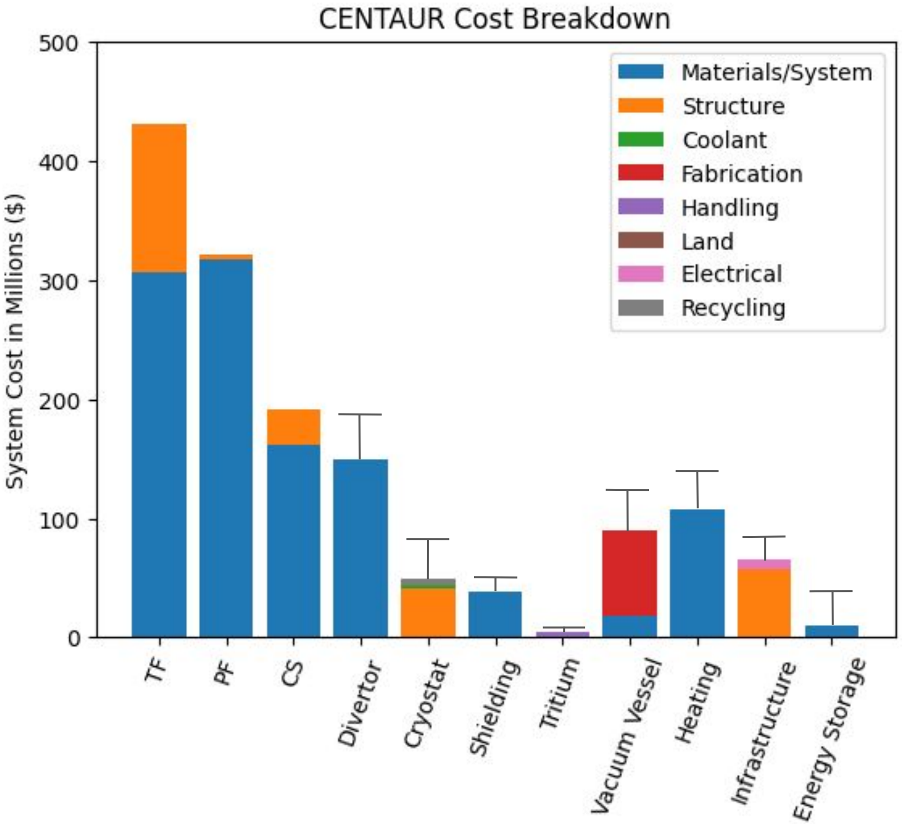
- Each system/subsystems interlink with design parameters and current materials and manufacturing pricing
- Material based systems priced volumetrically
- Increased modularity and subsystem estimates



Cost breakdown

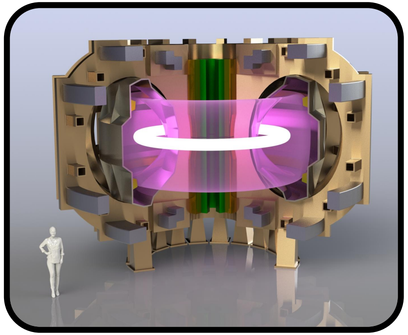
Total Cost: **$2.0 B ± 0.1**

- Material costs contribute the majority of total expenses ~$1.1B (53%)
  - HTS uncertainty at high volume is still unknown
- Magnet systems are the key drivers of overall reactor costs but do not scale with device lifetime
  - Tritium handling highly depends on operational lifetime
- Infrastructure costs are highly dependent on site location and availability of electrical grid accessibility



Direct cost is consistent with ARIES, Sheffield estimates

- **ARIES**[1] is a more modular approach. It calculates cost to a high degree of detail, differentiating the cost of spare parts, personnel salaries, inner/outer walls, etc.
- **Sheffield**[1] includes the maintenance and decommissioning costs, and accounts for inflation
  - Calculates most things as a function of $/kWh, which is less accurate for our one-time device.
  - Decommissioning costs range between **$100 million and $500 million USD**.



	CENTAUR	SPARC
CENTAUR model	**$2.0 B**	$1.9 B
ARIES	$1.9 B	$1.6 B
Sheffield	$2.4 B	$1.9 B
HTS Cost	$731 M (38%)	$410 M (21%)

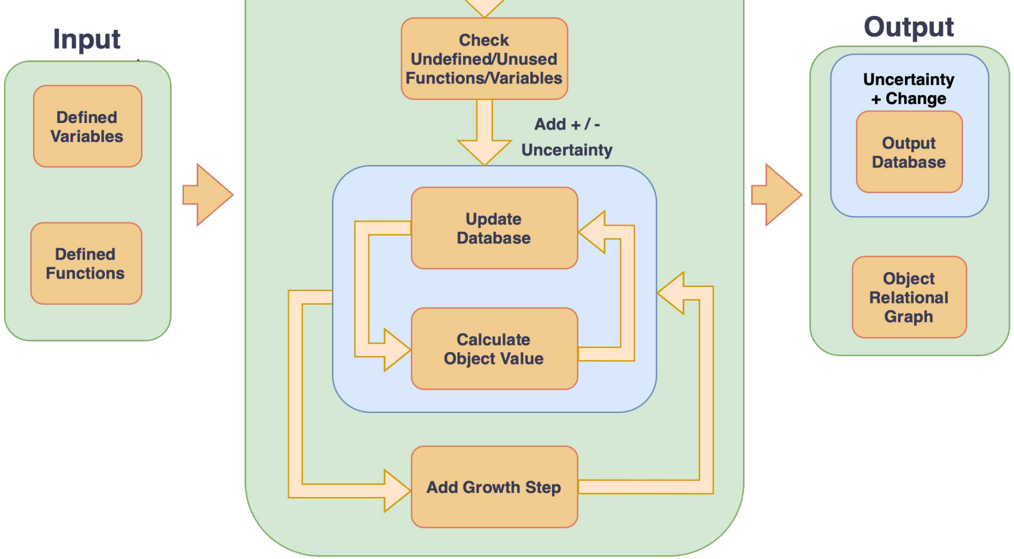
[1] Meneghini arXiv 2024

Major improvements made to costing model

Additions and updates to our model:

- Super battery energy storage
- Electrical infrastructure
- Tritium handling
- Heating systems
- Cryostat systems
- Pulse length scanning
- Shielding cost scanning

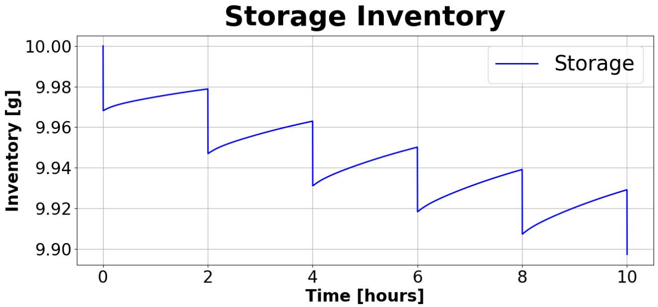
New Costing Model

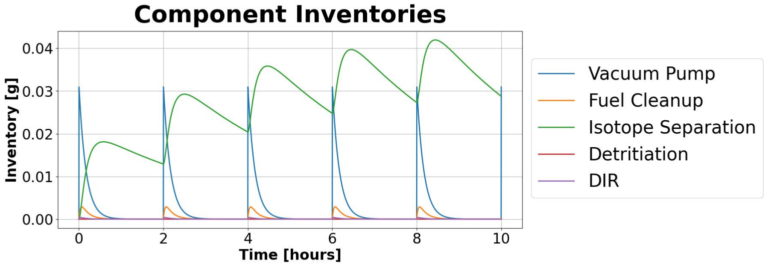


Tritium Inventory is set at 10 grams onsite by NRC

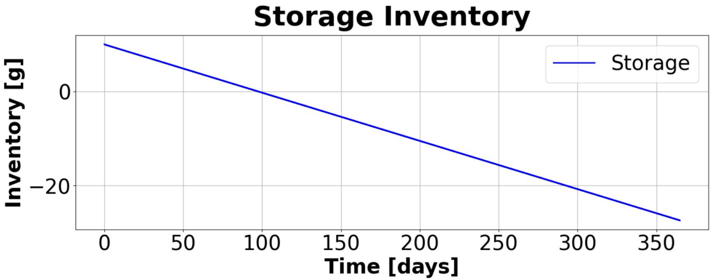
- 5 10-second flat top shots every 2 hours
- Burn rate is 6.37×10^{-8} kg/s based on 36 MW of fusion power
- Inventory model used is the Meschini 2023 model[1]
- Reprocessing will become difficult over time because of tritium trapped in wall

Single Day





One Year



[1]

Losing about 0.1 g per day on average (27 g/365 days)
Total cost over a year = 27 g × $30,000/g = **$810,000**

[1] Meschini NF 2023

Conclusions

CENTAUR advancements and innovations

- **HTS superconducting magnets designed for NT** plasma generation with comprehensive stress analyses
- **Curved CS** provides a novel efficient solution for field in a compact device
- Found cases with **high radiative fractions** that were benign to the divertor
- Beta tested **new physics codes** (FUSE, UETOOLS)
- Developed an **automated interface between Tokamaker and OpenMC** for rapid neutronics heating and damage testing
- Refactored modular costing model

CENTAUR – A feasible, high-field, energy breakeven negative triangularity tokamak

- Developed Q > 1 core scenario for high-field negative triangularity device
- Modeled a highly radiative divertor regime that avoids strike point sweeping and advanced divertor concepts
- Calculated required shielding to maintain sufficiently low neutron heating and damage to coils
- Designed high-field magnets optimized for low aspect ratio and negative triangularity
- Performed economic analysis using improved costing model