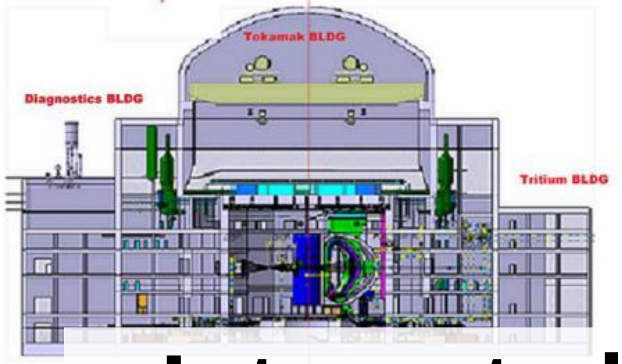
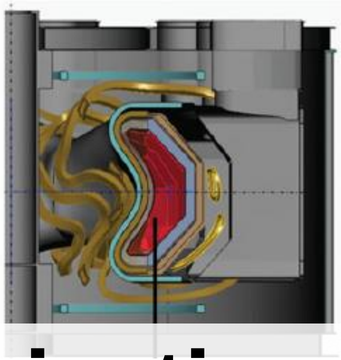
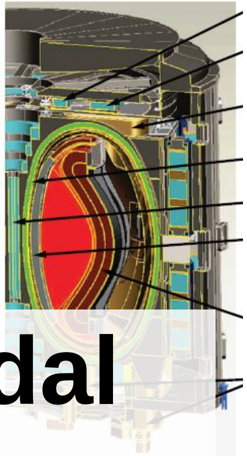






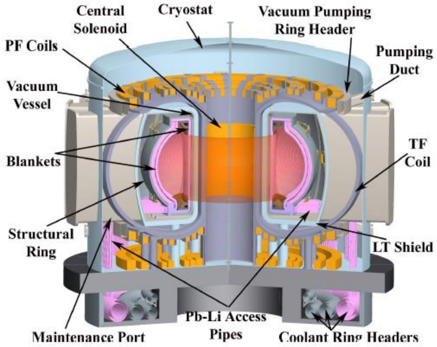
Integrated optimization of (toroidal magnetic) fusion power plants

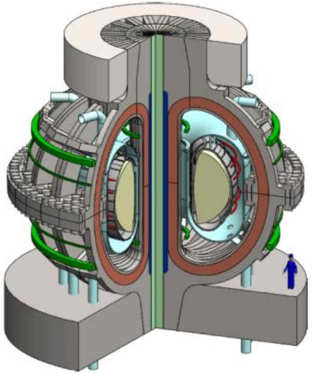
Walter Guttenfelder

Princeton Plasma Physics Laboratory

SULI 2020

June 23, 2020





Goals of this talk

- Summarize and compile key constraints (nuclear, engineering, plasma physics) and inter-dependencies that influence conceptual design of fusion power plants & pilot plants
  - Use that information to inform where innovations can make a difference
- Provide examples of 0D "systems analysis" and conceptual design studies (as time allows)
  - Highlight some of the front-end choices and assumptions that influence results

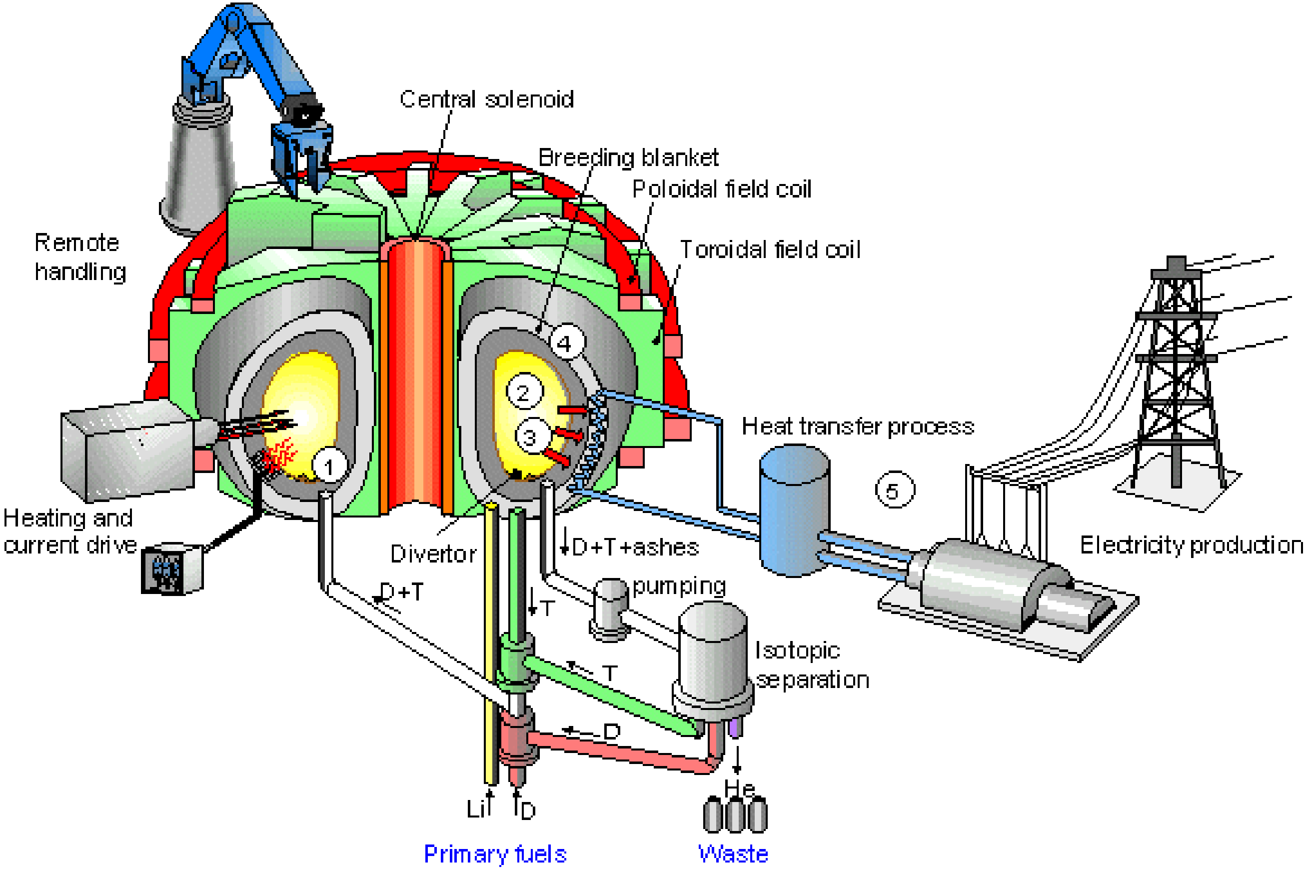
Further reading (& many figures taken from the following)**:**

- J. Friedberg, Phys. Plasmas **22**, 070901 (2015) ← **tutorial style introduction**
- C. Kessel, Fusion Sci. Tech **67**, 1 (2015) (and the entire January 2015 issue) [*ARIES-ACT power plant study*]
- All ARIES studies (http://aries.ucsd.edu/ARIES/) [*Many ARIES power plant studies*]
- B. Sorbom, Fusion Engineering and Design **100**, 378 (2015) [*ARC HTS pilot plant*]
- A. Kuang, Fusion Engineering and Design **137**, 221 (2018) [*ARC HTS pilot plant*]
- J.E. Menard, Nuclear Fusion **56**, 106023 (2016), Phil. Trans. R. Soc A (2019) [*Low-A HTS pilot plant*]
- M. Kovari, Fusion Engineering and Design **89**, 3054 (2014) [*PROCESS systems code - physics*]
- M. Kovari, Fusion Engineering and Design **104**, 9 (2016) [*PROCESS systems code - engineering*]
- H. Zohm, Nuclear Fusion **57**, 086002 (2017) (+ many others) [*ITER → EU-DEMO analysis*]
- G. Federici, Nuclear Fusion 59, 066013 (2019) (+ many others) [*EU-DEMO considerations*]

A non-engineered, turbulent path: From engineering, to turbulence, to fusion plasmas, to plasma turbulence (and maybe back to engineering … one day?)

- “I come from Des Moines. Somebody had to.” (*The Lost Continent*, B. Bryson)
- Studied electrical engineering at Milwaukee School of Engineering – I loved analog circuits (I was a wannabe audiophile)
- Discovered the beauty & magic of Maxwell’s equations → definitely going to grad. school
- But I got distracted by fluid dynamics, thermodynamics & aerodynamics
- Studied turbulent flames using laser induced fluorescence (Purdue University) → intro to turbulence
- Turbulent flames are fascinating, but I missed my Maxwell’s Equations → Plasma!
- Did my Ph.D. at the University of Wisconsin – Madison on HSX stellarator (see Bader talk, Day 4)
- Realized turbulence was an important research topic in magnetized fusion plasmas → **HOOKED!**
- Post-docs at U. Warwick (UK) & PPPL → spherical tokamak research (MAST, NSTX)
- I spend a lot of time babysitting supercomputer simulations solving nonlinear 5D gyrokinetic-Maxwell equations, analyzing data, and comparing the two to validate predictions, develop transport models
- But … sometimes I miss engineering “realities” (I suppose this is how I got myself roped into giving this talk … Arturo!)

Schematic of a fusion power plant



- Fusion core (magnets, plasma)
- Blanket (neutron capture, tritium breeding)
- Divertor/PFCs
- Heating & current drive
- Tritium processing and fueling
- Power conversion
- Maintenance scheme and waste

Conceptual design studies make front-end decisions and assumptions, then optimize remaining inter-dependencies

- Define mission deliverables like $P_{e,net}$, availability, cost metrics, e.g.
  - Nth-of-a-kind power plant with competitive COE
  - Demonstration reactor (DEMO), validate all systems expected for power plant
  - Pilot plant that produces net electricity, establishes capability for high average power output, demonstrates safe production and handling of tritium as well as feasibility of a closed fuel cycle (2019/2020 Community Planning Process Report)
- Choose a core architecture (steady-state tokamak, pulsed tokamak, stellarator, inertial fusion, …),
- Other elements might be assumed up front (blanket materials, heating scheme, …), ideally perform "trade" study to quantify impact

- **Methodology: Start with 0D "systems studies" (today's talk), validate design points with higher fidelity analysis, iterate**

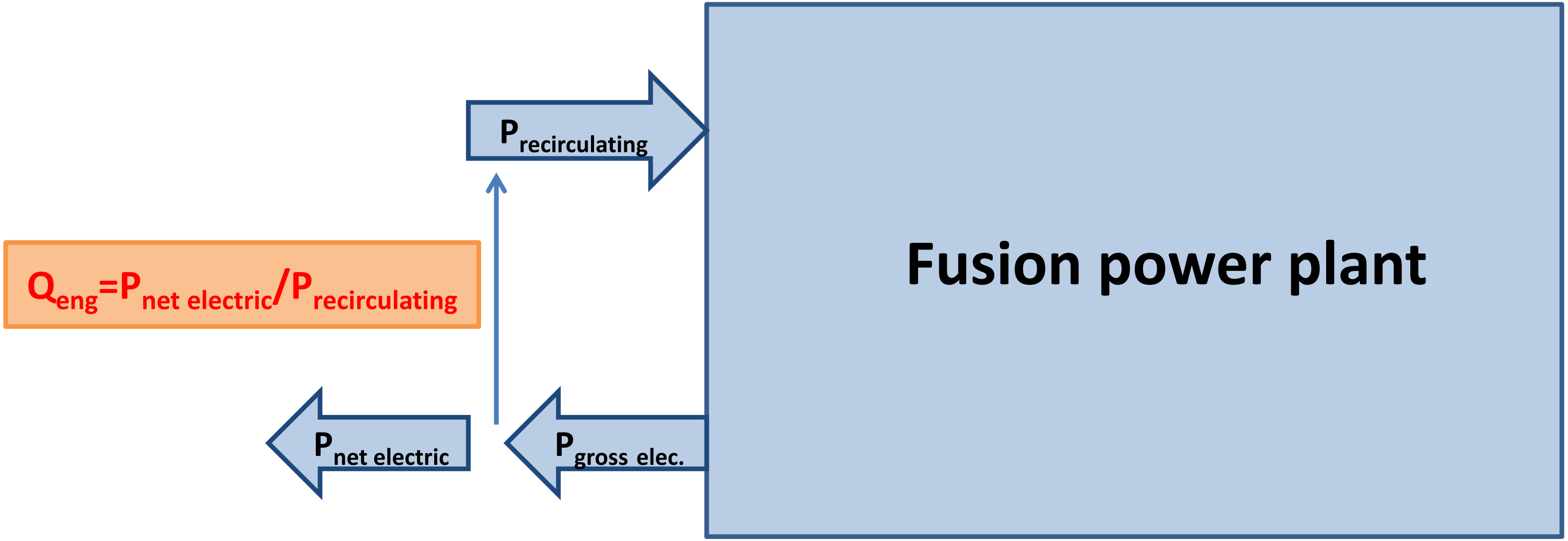
Power plant vs. pilot plant considerations

- A number of essential criteria for attractive power plants have been identified [El-Guebaly]
  - Economically competitive cost-of electricity (COE); load-following capacity and range of unit sizes; High system availability; Tritium self-sufficiency with closed fuel cycle; Radiation-resistant materials for long lifetime; RAMI (Reliability, availability, maintainability, inspectability); Easy to license; Intrinsic safety; Integral radioactive waste management and decommissioning plan
- Many international partners are pursuing "DEMO" reactors to demonstrate many of the above
  - Using most mature (least risky) technologies and physics assumptions generally leads to very large power plants (R~9 m tokamaks, up to R~20 m stellarators)
- Recent US vision* to pursue as a mission a pilot plant *at low capital cost*, perceived to be more attractive to development within the US energy market **Umstattd, Day 1**
  - Emphasis on raising "Technical Readiness" of low-maturity innovations to lower capital cost
- Recent call by DOE for NAS to perform a "fast-track" U.S. Compact Pilot Plant study to: (1) identify key goals for pilot plant; (2) list principle innovations needed for private sector to address, perhaps in concert with DOE efforts

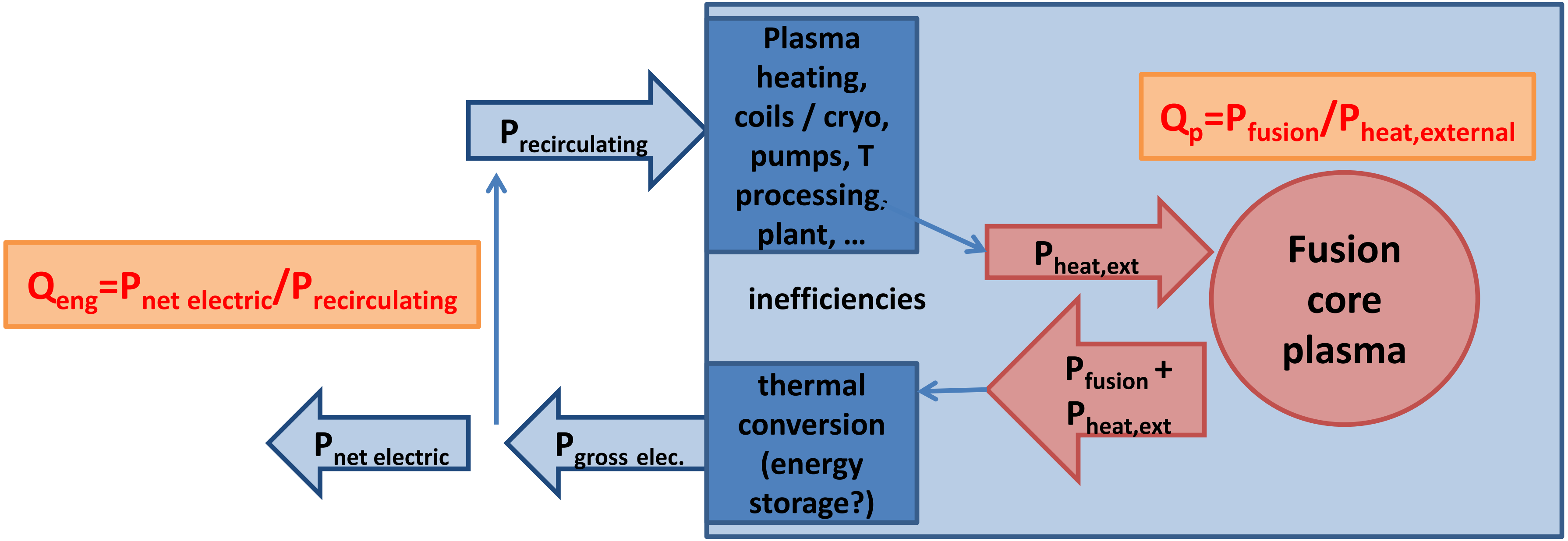
Mumgaard, Day 8

*NAS Burning Plasma report (2018), APS-DPP Community Planning Process report 2019/2020

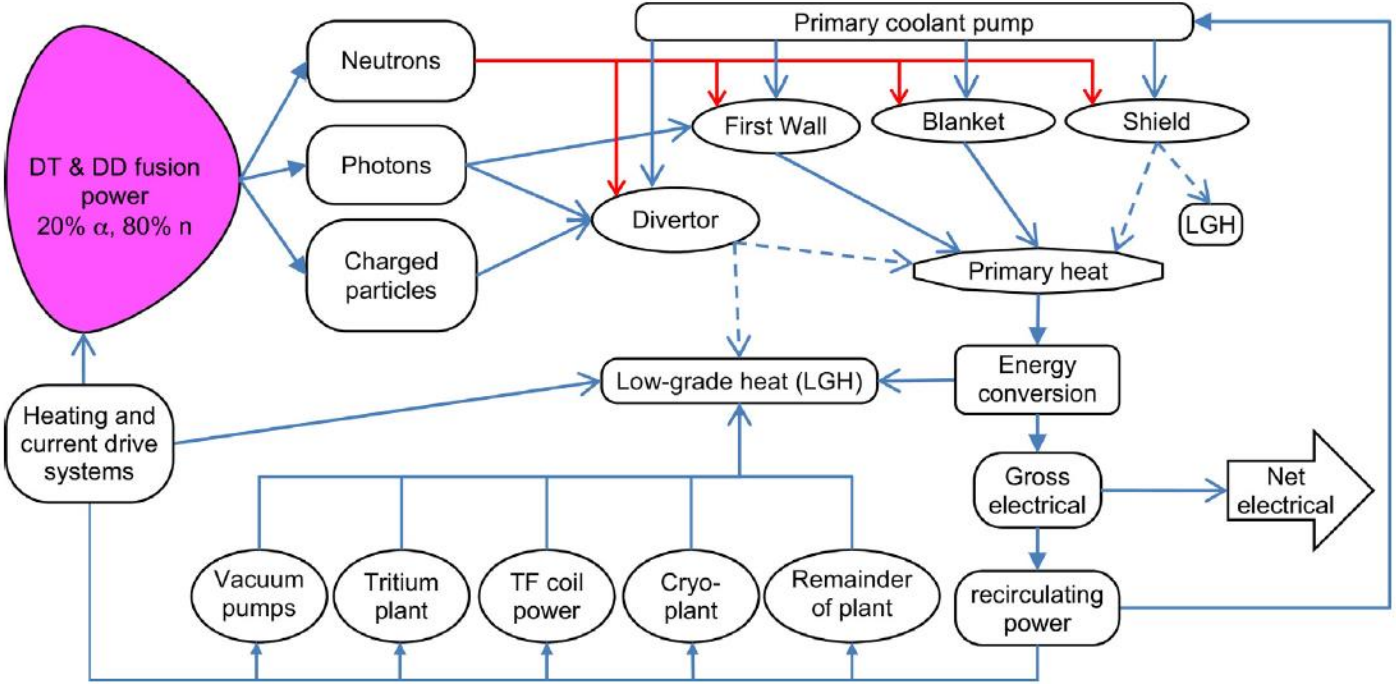
Simple block diagram of plant power balance



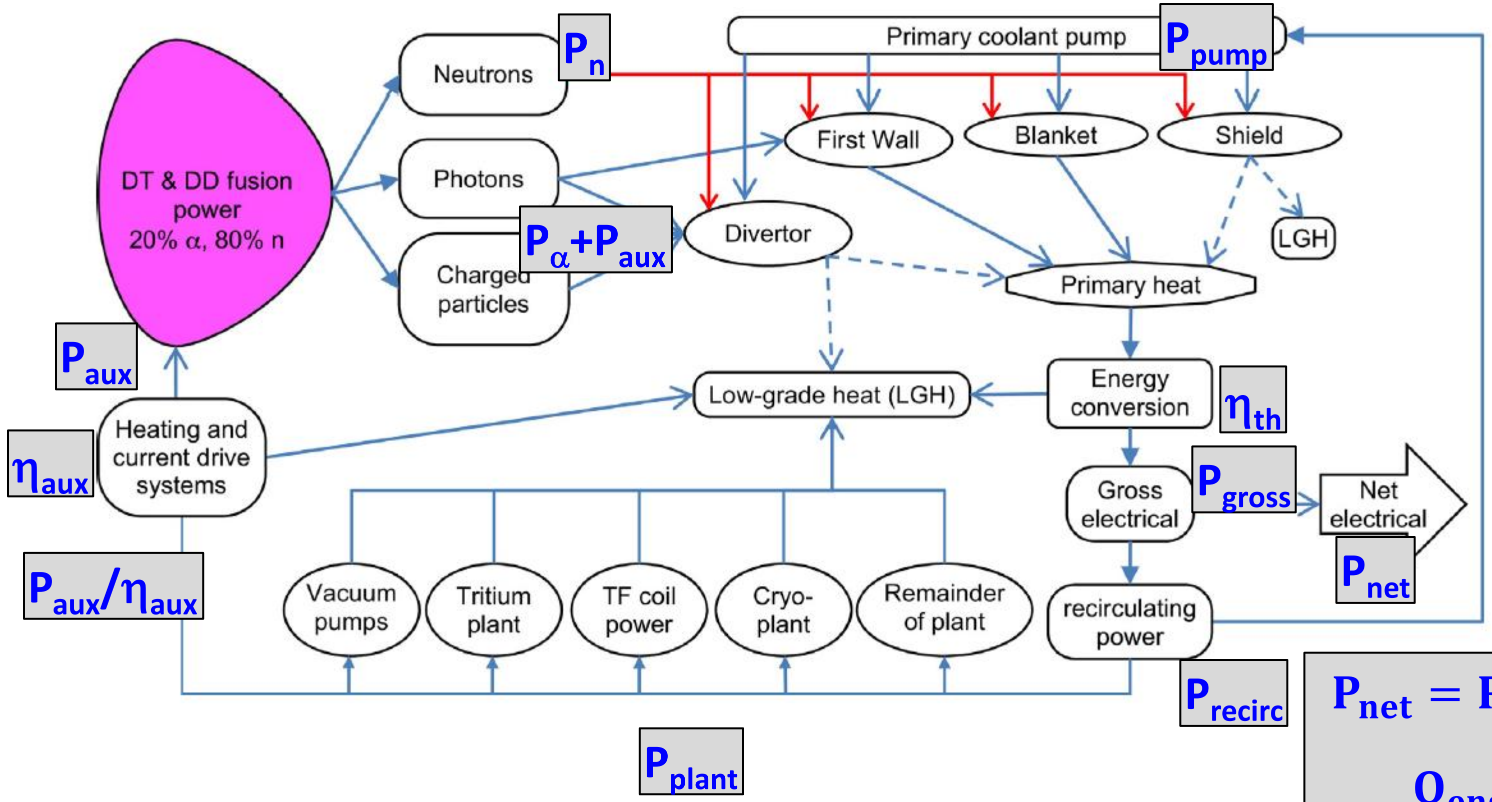
Simple block diagram of plant power balance



More detailed block diagram of plant power balance



More detailed block diagram of plant power balance



$$P_{net} = P_{gross} - P_{recirc}$$

$$Q_{eng} = \frac{P_{gross}}{P_{recirc}}$$

Electricity output & gain depend on fusion power & gain, thermal efficiency, and external heating efficiency

$$P_{gross} = \eta_{th} \cdot [M_n P_n + P_\alpha + P_{aux}]$$

$$\mathbf{P_{gross} = \eta_{th} \cdot Q_p \cdot P_{aux} \cdot \left[0.8M_n + 0.2 + 1/Q_p\right]}$$

$M_n \approx 1.1$, neutron energy multiplier

$$P_{recirc} = \frac{P_{aux}}{\eta_{aux}} + P_{coils} + P_{sub} + P_{control} + P_{pump}$$

$$\mathbf{P_{recirc} = \frac{P_{aux}}{\eta_{aux}}\left[1 + \eta_{aux}\frac{P_{coils}}{P_{aux}} + \eta_{aux} Q_p \frac{\left(P_{sub} + P_{control} + P_{pump}\right)}{P_{fus}}\right]}$$

Electricity output & gain depend on fusion power & gain, thermal efficiency, and external heating efficiency

- Assuming SC coils ($P_{coils} \rightarrow 0$), $(P_{sub}+P_{con}+P_{pump})/P_{fus}=0.07$, $M_n=1.1$

$$\mathbf{Q_{eng} = \eta_{th} \cdot Q_p \cdot \eta_{aux} \cdot \frac{[0.8M_n + 0.2 + 1/Q_p]}{[1 + 0.07\eta_{aux}Q_p]}}$$

$$P_{net} = P_{fus} \cdot [\eta_{th} \cdot (0.8M_n + 0.2 + 1/Q_p) - 0.07] - \frac{P_{aux}}{\eta_{aux}}$$

$$\mathbf{P_{net} \approx P_{fus} \cdot \eta_{th} \cdot 0.94 - \frac{P_{aux}}{\eta_{aux}}}$$

Cost-of-electricity (COE) and capital cost are also of great interest

- Not considering cost metrics here (quantitatively)

- There have been multiple attempts to quantify Nth-of-a-kind power plant COE, depending on:
  - Capital cost (generally expect $$$ ~ volume or mass) + “learning curve”
  - Construction, licensing and operating costs
  - Availability
  - Waste disposal
  - Contingency

- Capital cost & development costs driven by risks & unknowns, complexity, engineering, design, prototyping ← hard to quantify

Remainder of this talk: assemble expressions, relations and constraints for P_{fusion}, P_{aux}, Q_p, η_{th}, η_{aux}, …

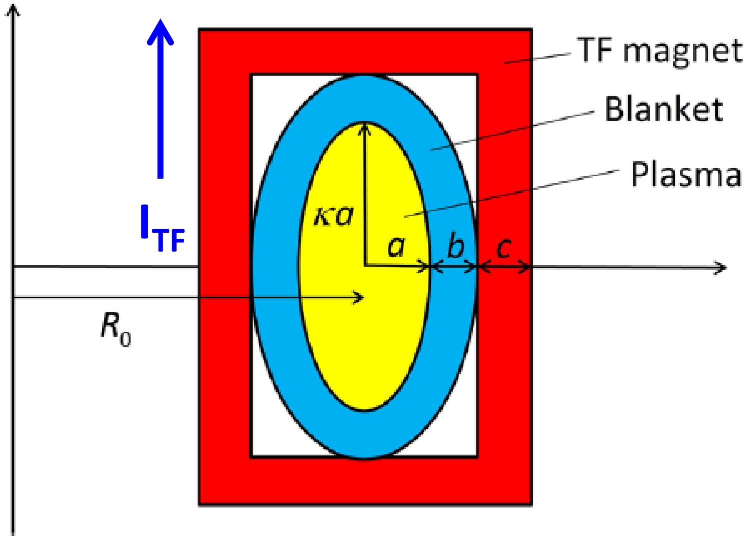
- **Geometry and size**
- **Nuclear physics**
- **Engineering**
- **Plasma physics**

→ **Highlight interactions in 0D systems analysis that influence design points**

- I'll focus on tokamak, but can do similar for others (stellarators, RFPs, FRCs, MCF, ICF, …) ← <u>good homework problem for you! (and me)</u>

Some geometry definitions

Major radius: $\mathbf{R_0}$
Minor radius: $\mathbf{a}$
Aspect ratio: $\boldsymbol{A} = \frac{\mathbf{R_0}}{\mathbf{a}}$
Inverse aspect ratio: $\boldsymbol{\epsilon} = \frac{\mathbf{1}}{\mathbf{A}} = \frac{\mathbf{a}}{\mathbf{R_0}}$
Plasma elongation: $\boldsymbol{\kappa}$
Blanket thickness: $\mathbf{b}$
Coil thickness: $\mathbf{c}$



Plasma volume: $\boldsymbol{V = 2\pi R_0 \pi a^2 \kappa}$

Plasma surface area: **S**$\approx \mathbf{2\pi} \boldsymbol{R_0} \cdot \mathbf{2\pi a}\left[(\mathbf{1} + \boldsymbol{\kappa}^{\mathbf{2}}/\mathbf{2})\right]^{\mathbf{1/2}}$

Fusion power depends on pressure and volume

$$P_{fus} = n_D n_T \langle \sigma v \rangle \mathcal{E}_{DT} \cdot V$$

$$P_{fus} \sim (nT)^2 V \sim p^2 \cdot Ra^2 \kappa$$

$$\boldsymbol{P_{fus} \sim \beta^2 B^4 R a^2 \kappa}$$

$$\boldsymbol{P_{fus} \sim \beta^2 B^4 R^3 \epsilon^2 \kappa}$$

Cowley, Day 1

Only highlighting 0D relations, can also accommodate n, T profile shapes if desired

Fusion gain depends on the "triple product" $nT\tau_E$ (power balance)

Fusion plasma gain

$$Q = \frac{P_{fusion}}{P_{heat,external}}$$

$$Q \sim (nT) \cdot \left(\frac{nTV}{P_{loss}}\right)$$

$$\mathbf{Q \sim nT\tau_E \sim p \cdot \tau_E}$$

Energy confinement time:

$$\tau_E = \frac{\text{stored energy}}{\text{rate of energy loss}} \sim \frac{nTV}{P_{loss}}$$

More accurately

From power balance:

$$P_{loss} = P_\alpha + P_{heat,ext}$$

$$\mathbf{nT\tau_E \sim \frac{Q}{Q+5}}$$

Should also solve particle balance (→ dilution of fuel):

$$N_{D,T} = N_e - 2N_{He} - \sum_{imp} Z_{imp} n_{imp}$$

$$\frac{dN_{He}}{dt} = \frac{P_\alpha}{E_\alpha} - \frac{N_{He}}{\tau_{He}}$$

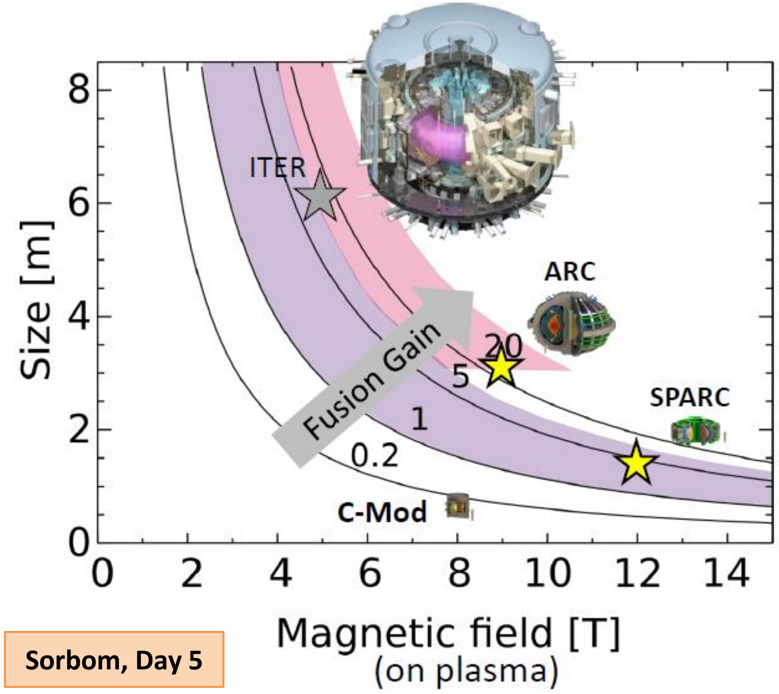
Cowley, Day 1

Achievable fusion gain tied to global plasma stability limits, engineering limits and energy confinement time

$$\boxed{\mathbf{Q \sim nT\tau_E \sim \beta \cdot B^2 \cdot \tau_E}}$$

For fixed geometry assumptions and physics constraints, fusion gain and power set largely by ~ (R_0,B_0)

- $\beta = p / (B^2/2\mu_0)$ limited by global MHD stability
- Magnetic field, B, determined by superconductor technology, mechanical stress & strain limits, and blanket & shield thickness
- Energy confinement time, τ_E, dominated by turbulent losses, some room for manipulation (flow shear, plasma shape)



Sorbom, Day 5

Energy confinement time characterized by various empirical or semi-empirical scalings

- Empirical confinement scalings inferred from multi-machine database, e.g. 1998 ITER Physics Basis H-mode scaling (IPB98)

$$\boldsymbol{\tau_{\mathrm{E,IPB98}} = 0.0562 \cdot \mathrm{I_p^{0.93}\ B_T^{0.15}\ n^{0.41}\ P_{loss}^{-0.69}\ R^{1.97}}\ \epsilon^{0.58}\ \kappa^{0.78}\ A^{0.19}}$$

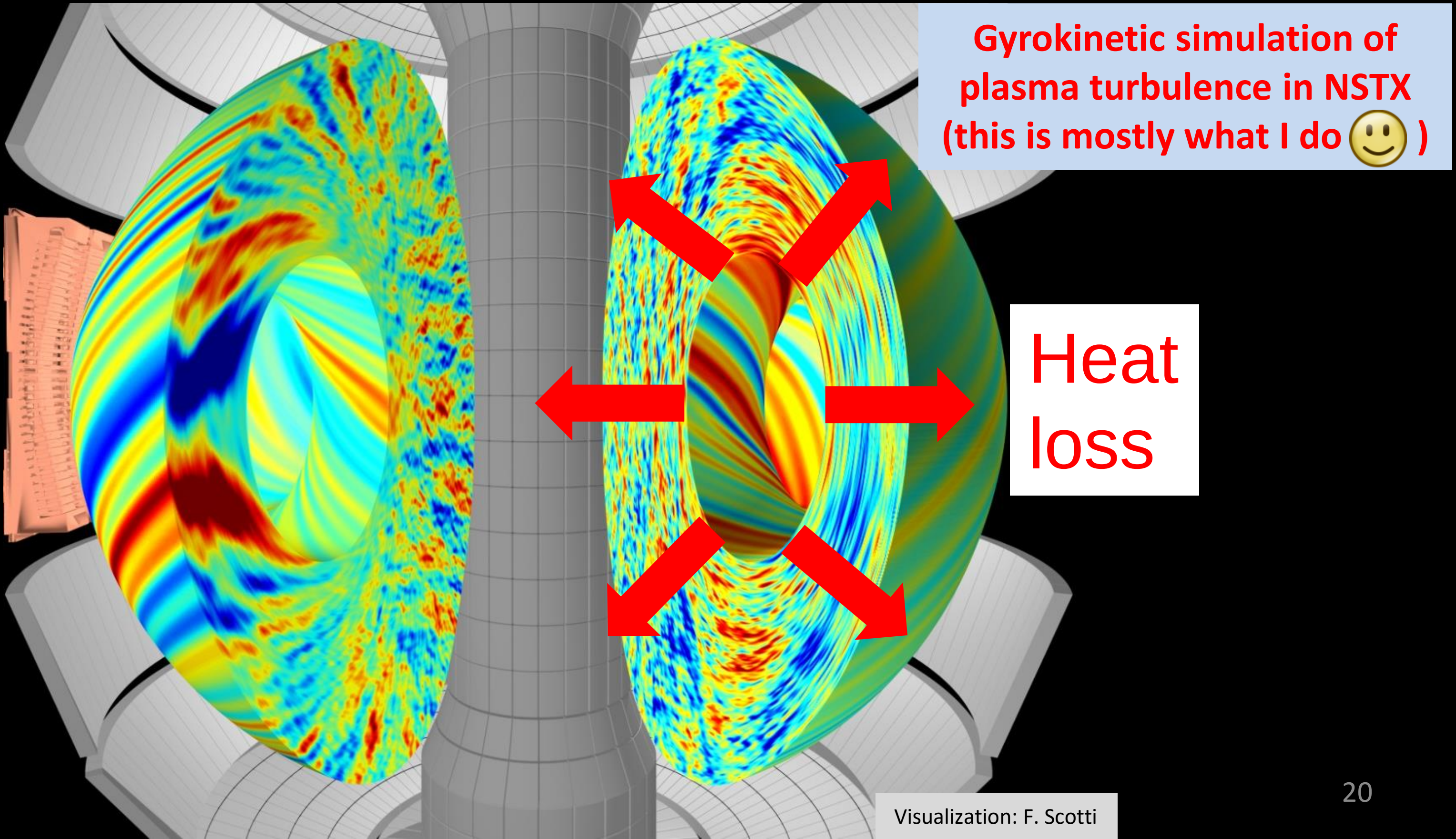
  - Scaling trends supported by theory and modeling

- Other forms exist, depending on machine & plasma state (L-mode, H-mode, I-mode, QH-mode, …), e.g. NSTX H-modes

$$\boldsymbol{\tau_{\mathrm{E,NSTX06}} = 0.095 \cdot \mathrm{I_p^{0.57}\ B_T^{1.08}\ n^{0.44}\ P_{loss}^{-0.73}\ R^{1.97}}\ \epsilon^{0.58}\ \kappa^{0.78}\ A^{0.19}}$$

- "H_{98}" factor (e.g. $H_{98}=\tau_{E,NSTX}/\tau_{E,IPB}$) to quantify what we don't understand well enough (at least to predict quantitatively) … but we're working on it

Nature of turbulent losses can vary with machine geometry and operating regimes → opportunities for improved performance



Global stability (MHD, others) provides a number of constraints

- Normalized beta limit: $\beta_N = \beta \, / \, (I_P/aB) < \beta_{N,limit} \sim 2\text{-}6$ (function of R/a, κ, proximity to conducting walls)
  - To avoid disruptions or otherwise deleterious effects
- Safety factor limit: $q_* \sim aB_{tor}/RB_{pol} \sim a^2B_{tor}/RI_p \cdot (1+\kappa^2) > 2.5$
  - To avoid "kink" modes (plasma current is limited for a given toroidal field strength)

Battaglia, Day 4

- Elongation limit: $\kappa_{limit} \sim 1.7\text{-}2.5$ (function of R/a, plasma inductance)
  - To avoid vertical instability
- Empirical density limit ("Greenwald limit"): $n < n_{GW} = I_p \, / \, \pi a^2$ ($f_{GW}<1$)
  - To avoid disruptions

Steady-state tokamaks require 100% non-inductive current

- Inductive current drive from central solenoid is limited
- Externally driven current (from heating sources) must make up the difference

$$\mathbf{I_{BS} + I_{CD} + \cancelto{0\ \text{(ideally)}}{I_{OH}} = I_P} \quad \textbf{or} \quad \mathbf{f_{BS} + f_{CD} + \cancelto{0\ \text{(ideally)}}{f_{OH}} = 1}$$

- Luckily, pressure gradient in tokamaks leads to self-generated “bootstrap current” (due to ∇B, curvature drifts + ∇p + collisions)

$$\mathbf{f_{BS}} = \frac{\mathbf{I_{BS}}}{\mathbf{I_p}} \sim \sqrt{\epsilon}\cdot \boldsymbol{\beta}_{\mathbf{pol}} \sim \sqrt{\epsilon}\,\frac{\mathbf{pa^2}}{\mathbf{I_p^2}}\left(\mathbf{1+\kappa^2}\right) \sim \frac{\boldsymbol{\beta_N q_*}}{\sqrt{\epsilon}}\left(\mathbf{1+\kappa^2}\right)$$

Battaglia Day 4

Auxiliary heating required to access high temperatures and to drive current

- External heating required to access burning plasma conditions (T~14 keV)
- Can also drive current for long-pulse or steady-state tokamak

$$\mathbf{I_{CD} = \eta_{CD} \frac{P_{CD}}{n_e R} \cdot \mathit{F}(T_e, Z_{eff}, \cdots)} \qquad \mathbf{\eta_{aux} \approx 0.3 - 0.4}$$

- Each approach has different efficiencies (η_{CD} in units of 10^{20} MA/MW-m^2) and trade-offs

  Pinsker, Day 5

  - NBI (η_{CD}≤**0.35**); well-established, needs vessel openings, impacts tritium breeding ratio (TBR), line-of-sight to neutrons; large, high voltage sources (0.5-1 MeV)
  - LHCD (η_{CD}≤**0.45**); sources available, needs internal antenna, direct exposure to plasma and neutrons
  - ECRH (η_{CD}≤**0.25**); very precise, has density cutoffs (motivates gyrotron developments 200-300 GHz)

Steady-state current in a tokamak provides a very challenging constraint

$$\mathrm{f_{BS}} + \mathrm{f_{CD}} = 1$$

$$\mathrm{A_1} \frac{\beta_\mathrm{N} \mathrm{q}_*}{\sqrt{\epsilon}} (1 + \kappa^2) + \mathrm{A_2} \frac{\eta_\mathrm{CD} \mathrm{P_{CD}}}{\mathrm{I_p} \mathrm{n_e} \mathrm{R}} = 1$$

f_{GW} ~ constant < 1 ($n \sim I_P/a^2$)

$$\mathbf{A_1} \frac{\boldsymbol{\beta}_\mathbf{N} \mathbf{q}_*}{\sqrt{\boldsymbol{\epsilon}}} \left(\mathbf{1} + \boldsymbol{\kappa}^\mathbf{2}\right) + \mathbf{A_2} \frac{\boldsymbol{\eta}_\mathbf{CD} \mathbf{P_{CD}} \mathbf{a^2}}{\mathbf{f_{GW}} \mathbf{I_p^2} \mathbf{R}} = \mathbf{1}$$

- High bootstrap fraction (depending on stability limits), current drive efficiency, and confinement quality (τ_E, at I_p as low as possible) all needed

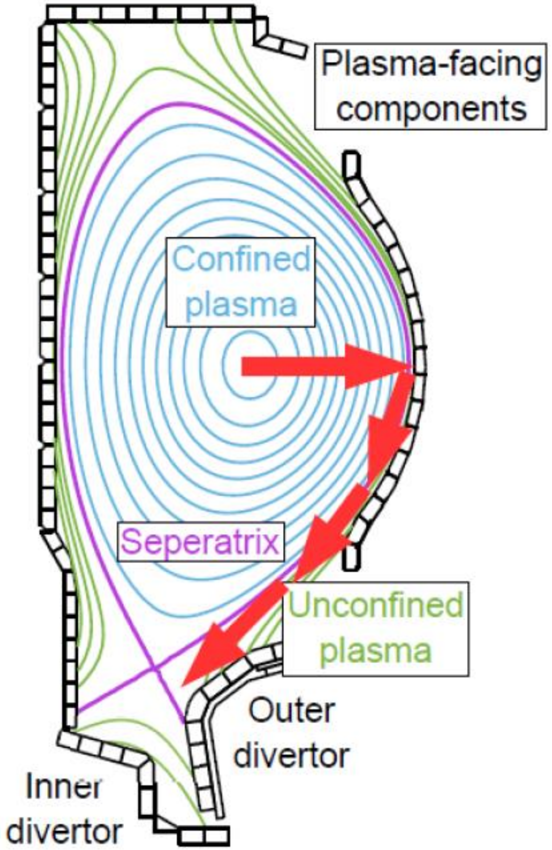
Divertor and first wall material limits constrain exhaust power & particle handling

- Must dissipate heat fluxes (steady state, transient) crossing from closed surfaces to open field lines to satisfy material limits
  - Solid PFCs, $q_{\perp,\text{solid PFC}} \leq$ **5-10 MW/m²**
  - Liquid metal PFCs (Li, Sn, SnLi), $q_{\perp,\text{LM PFC}} \leq$ **50 MW/m²**
  - Vapor shielded PFCs, $q_{\perp,\text{vapor PFC}} \leq$ **(???) MW/m²**

- 0D studies often simply evaluate a scrape-off-layer (SOL) heat flux metric to represent the "heat exhaust challenge"

$$Q_{||} = \frac{P_\alpha B_0}{R_0} = \left(\frac{E_\alpha}{E_f}\frac{P_E}{\eta_T}\right)\left(\frac{B_0}{R_0}\right).$$

- Recent analysis has clarified scaling of impurity seeding (f_{imp}) required to radiatively dissipate ($P_{rad,imp}$) large $Q_{||}$ ("detachment")
  - Reinke [2017], Goldston [2017]



Radial build depends on TF coils, blanket, minor radius and central solenoid

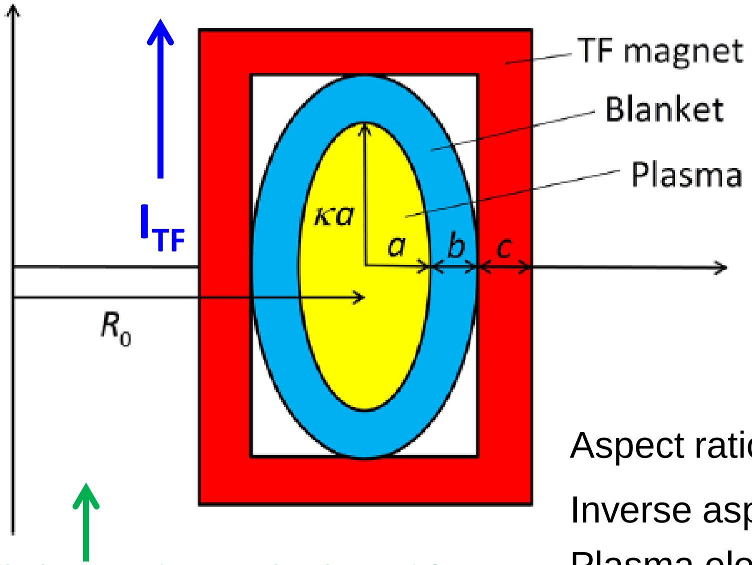
- Magnetic field in plasma determined by magnet technology & engineering limits, blanket thickness, and aspect ratio

$$B(R) = \frac{\mu_0 I_{TF}}{2\pi R} = \frac{B_0 R_0}{R}$$

$$B_0 R_0 = B_{TF} R_{TF}$$

$$R_{TF} = R_0 - a - b = R_0 \left(1 - \epsilon - \epsilon_b\right)$$

$$\mathbf{B_0 = B_{TF}(1 - \epsilon - \epsilon_b)}$$



Will also need central solenoid for inductively driven current ramp-up

Aspect ratio: $\mathbf{A} = \frac{\mathbf{R_0}}{\mathbf{a}}$

Inverse aspect ratio: $\boldsymbol{\epsilon} = \frac{\mathbf{1}}{\mathbf{A}} = \frac{\mathbf{a}}{\mathbf{R_0}}$

Plasma elongation: $\boldsymbol{\kappa}$

Normalized blanket thickness: $\boldsymbol{\epsilon_b} = \frac{\mathbf{b}}{\mathbf{R_0}}$

Breeder-blanket thickness requirement largely determined by neutron absorption issues

- Breeder-blanket required for neutron multiplication (w/ Be, Pb), neutron moderation / slowing down, tritium breeding (w/ Li), shielding (e.g. magnets)
- **Blanket thickness, b ~ 1 m to satisfy slowing down, breeding & shielding requirements**
  - Must be validated by neutronics with sufficiently detailed geometry & materials

Kessel, Day 6

- Device size (R,a,b) and power density also determines neutron wall loading, $W_n = P_{neutrons}/S_{wall}$, $S_{wall} \approx 2\pi R_0 \cdot 2\pi a\left[(1+\kappa^2/2)\right]^{1/2}$
  - Higher W_n enables more compact configuration, but reduces component lifetime which impacts availability (→COE)
- **W_n~1-4 MW/m^2**

Garrison, Day 7

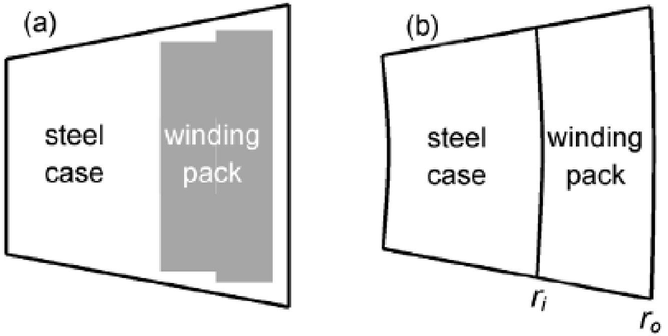
Thermal efficiency depends on achievable temperatures in blanket coolant and thermodynamic cycle

- A number of blanket concepts proposed with different materials & coolants, affects achievable temperatures and thermal conversion efficiencies Kessel, Day 6
  - Sub-critical water cooled (T_{out}~350 °C), η_{th}**~0.3** (Rankine / steam cycle)
  - Super-critical CO2 (T_{out}~500 °C), η_{th}**~0.36-0.45** (Brayton / gas cycle)
  - DCLL, He cooled (T_{out}~650 °C), η_{th}**~0.45** (Brayton / gas cycle)
  - SCLL, He cooled (T_{out}~1000 °C), η_{th}**~0.58** (Brayton / gas cycle)

- Above, along with other functional aspects, validated by more sophisticated modeling (neutronics for shielding, TBR; MHD for pump power; etc.) - **If I ever switched research areas, I'd probably analyze fusion breeder-blankets!**

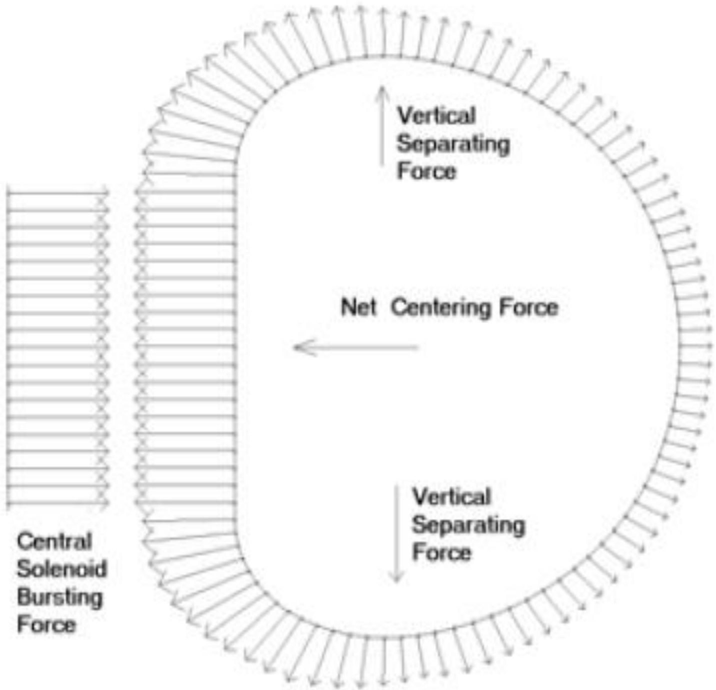
Coil sizing & achievable B_0 field depend on required structural material to support J×B forces + winding pack to support current

TF coil cross section (inboard side)



- Need structural support to manage J×B forces
  - Vertical TF force
  - Centering TF force
  - Out-of-plane TF bending force (from interaction with PF coils)
  - Central solenoid bursting force
- Need sufficient winding pack area to support field, $B_{TF} \sim I_{TF} = J_{WP} \cdot A_{WP}$

Coil forces



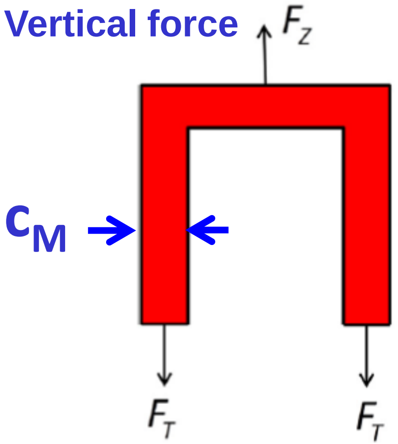
P. Titus

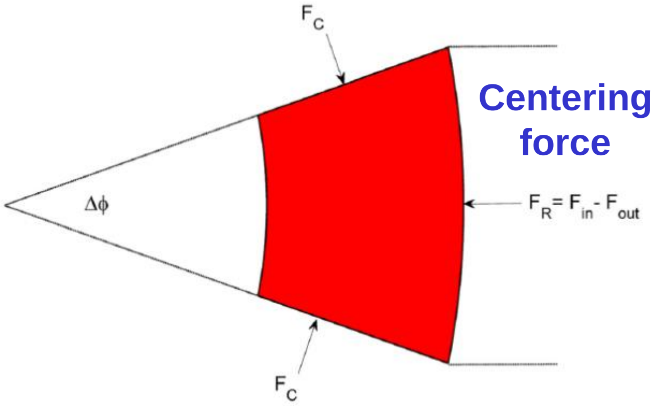
Width of supporting coil structure determined by material stress limits, can limit B_{TF} (B_0)

- TF vertical force ($F_Z \sim B_0^2 R_0^2$) balanced by tensile force ($2F_T = 2\sigma_T A_T$)
- TF centering force ($F_R \sim B_0^2 R_0$) balanced by compression from vaulting / wedging ($F_C = 2\sigma_C A_C$)
  - Can also "buck" on central solenoid or bucking cylinder (JET)

- Coil structure thickness (**c_M**) constrained by total stress limit (e.g. Tresca stress **$\sigma_T + \sigma_C \leq \sigma_{max} \sim 660$ MPa** for steels)
- Must also consider strain limits, $\sigma = E_{young}\varepsilon$
  - E.g. **$\varepsilon < 0.3\text{-}0.45\%$** for HTS, (Sorbom, Day 5)

- More detailed models include discrete number of TF coils (TF ripple from non-axisymmetry), # of WP turns, structural breakdown of WP layers, …





Width of winding pack depends on achievable current density, can limit B_{TF} (B_0)

- $B_{TF} = \frac{\mu_0 I_{TF}}{2\pi R_{TF}}$
- $I_{TF} = J_{WP} \cdot A_{WP}$
- $A_{WP} = \pi[(R_0 - a - b)^2 - (R_0 - a - b - c_{WP})^2]$

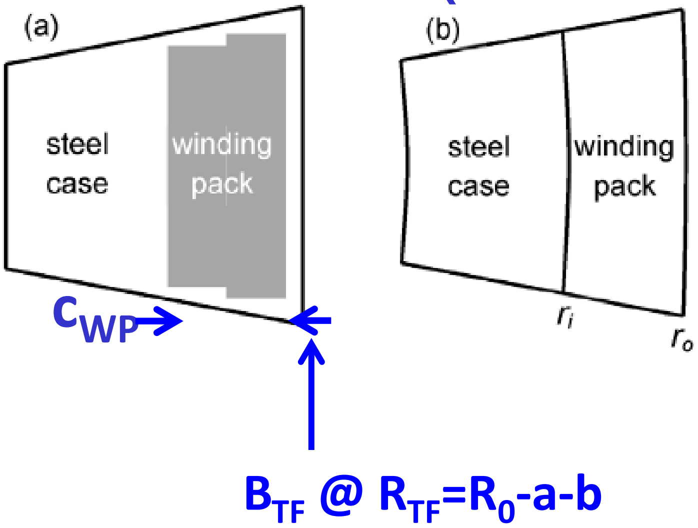
Constrained by:

- **$B_{TF,max}$ < 12-20 T**
- **$J_{WP,crit}$ < 12-100+ MA/m^2**

depending on LTS / HTS technology & configuration

$$B_{crit} = B_{c,0} \cdot F(J/J_{crit}, T/T_{crit})$$
$$J_{crit} = J_{c,0} \cdot F(B/B_{crit}, T/T_{crit})$$
$$T_{crit} = T_{c,0} \cdot F(B/B_{crit}, J/J_{crit})$$

TF coil cross section (inboard side)



Sorbom Day 5

Let's put it all together to identify integrated self-consistent solutions

- Choose targets and constraints
- Solve remaining equations
- Test sensitivity of solution to parameter variations

- First consider power plant ($P_{e,net}$=1000 MWe)
- Then pilot plant ($P_{e,net}$ ~ 100-200 MWe)

Power plant example: $P_{e,net}$=1000 MWe , steady-state using well-established (~conservative) physics limits [Freidberg tutorial]

- **Choose a target power**: P_E=1,000 MWe, $f_{recirc} \leq 0.15$ (f_{CD}=0.1, η_{aux}=0.4 → P_{aux}), η_{th}=0.4, T_{plasma}=14 keV
- **Choose geometry**: R/a=4, κ=1.7, $W_n \sim P_\alpha$/Area = 4 MW/m^2
- **Solve for**: R, a, pressure, density

- **Choose blanket thickness**: $b_{blanket}$=1.2 m (n-moderation, T-breeding)
- **Choose coil technology**: Nb_3Sn coil model ($B_{TF,crit}$=13 T, $J_{WP,crit}$=20 MA/m^2, σ_{max}=600 MPa)
- **Solve for**: coil thickness & B_0 (from radial build)

- **Solve for**: confinement time and required current ($P_{loss} \approx P_\alpha$ → τ_E → I_P)
- **Solve for**: externally driven current (assumed LHCD, $\eta_{CD} \approx 0.43$ MA/MW·m^2)

- **Evaluate stability and criteria:**
  - (1) beta limit (β_N<2.8)
  - (2) kink limit (q*>2)
  - (3) greenwald density limit (n/n_{GW}<1)
  - (4) 100% non-inductive ($f_{BS}+f_{CD}$=1)

Not possible to satisfy 100% non-inductive or kink stability limit due to large required current

- Base assumptions (H_{98}=1) require large I_p=14 MA to achieve necessary Q~τ_E

$$\tau_{E,IPB98} \sim H_{98} \cdot I_p^{0.93}$$

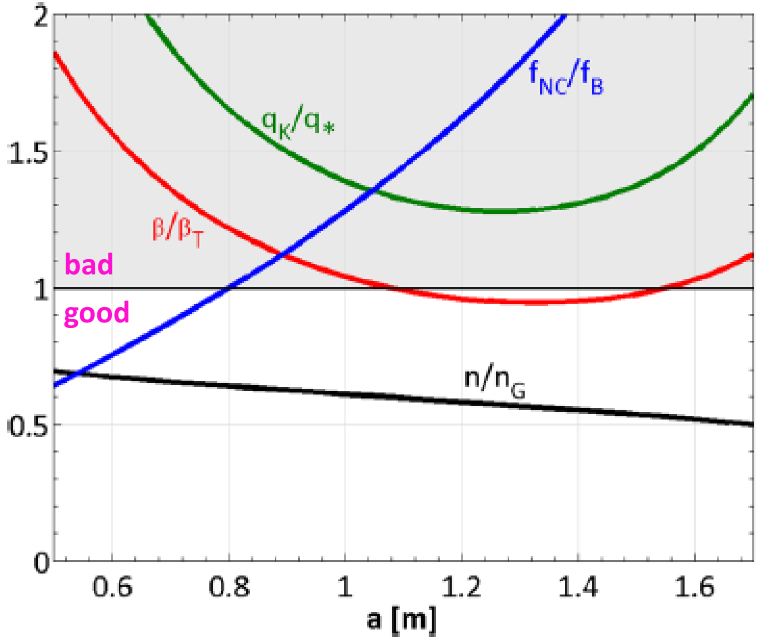
- Insufficient bootstrap current and external current drive for 100% non-inductive

$$A_1 \frac{\beta_N q_*}{\sqrt{\epsilon}}(1+\kappa^2) + A_2 \frac{\eta_{CD} P_{CD}}{I_p n_e R} = 1$$

- Also fails to meet kink stability due to large I_p

$$q_* \sim a^2 B_{tor}/R I_p \cdot (1+\kappa^2)$$

Base case: a=1.34 m, R=5.34 m



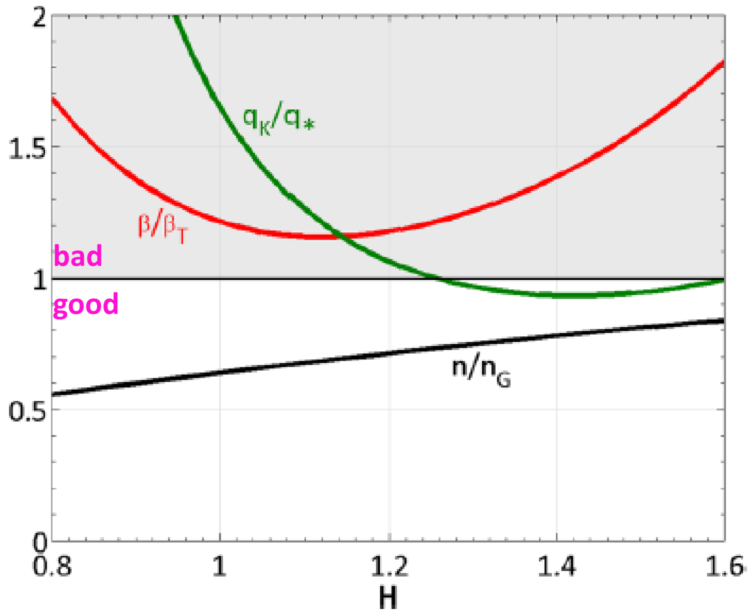
Enhanced confinement enables 100% non-inductive scenario

- Enforce 100% non-inductive constraint and vary confinement enhancement (H98)

$$\tau_{E,IPB98} \sim H_{98} \cdot I_p^{0.93}$$

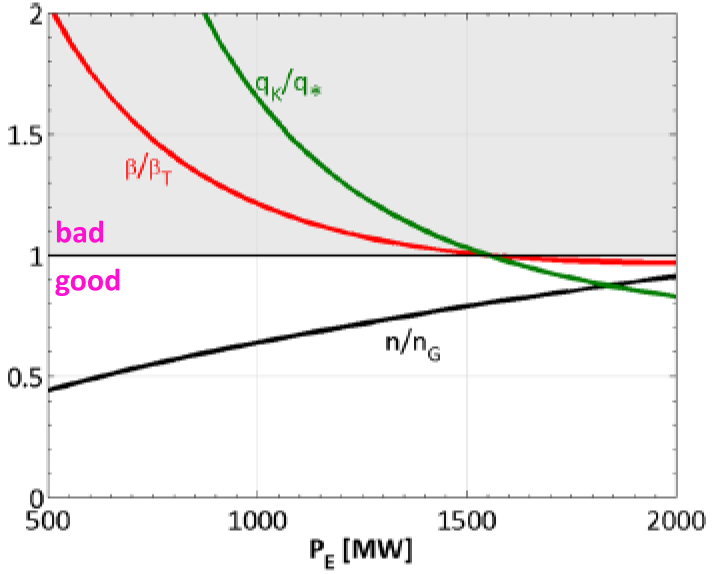
- Can achieves sufficient confinement (τ_E) at lower I_p → enables 100% non-inductive & avoids kink instability

- Still violates beta limit → if we can simultaneously operate at higher beta and confinement, we have a solution
  - Higher β_{pol} gives higher $f_{BS} \sim q^* \beta_N / \varepsilon^{1/2}$
- Achieving high β_N and H at low-disruptivity are major research priorities DIII-D (beta limited) and NSTX-U (confinement limited)



Higher capacity enables 100% non-inductive, eventually satisfying all stability constraints at very large scale

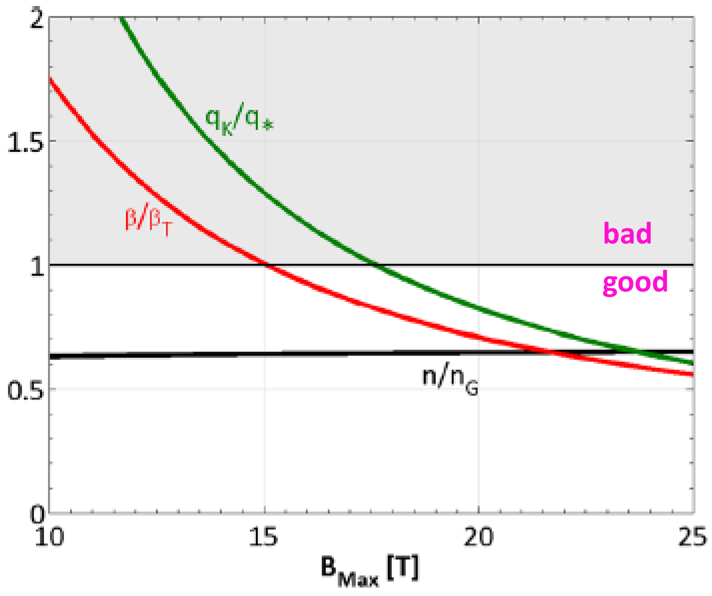
- Favorable for COE, but not for capital cost



Higher field on-axis satisfies all constraints

- More aggressive technology (e.g HTS, Sorbom, Day 5) relaxes most plasma physics challenges
  - Avoids β and q* kink limit

- Higher power density & wall loading
- Exacerbates boundary heat flux mitigation challenge, $Q_{||}$ = PB/R
  - But radiative detachment solution for plasma exhaust also scales with B [Reinke, 2017]



ARIES-ACT study performed trade study in assumed physics and technology limits

- "ACT": Advanced and Conservative Tokamak power plant study
  - 1000 MWe
  - 100% non-inductive
  - PbLi breeder
  - Nb_3Sn
  - A=4, κ=2.2

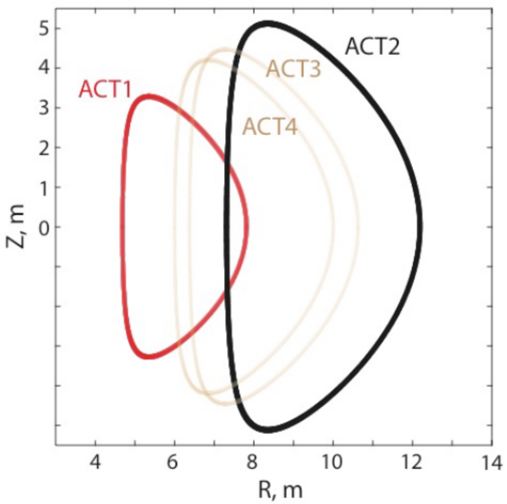
- ACT1: More aggressive physics and technology
  - Much smaller (1000 MWe at size of ITER)
  - Requires elevated confinement (H_{98}=1.65) and good stability (β_N=4.75)

- ACT2: Conservative physics and technology
  - Larger ~ EU-DEMO, higher I_p and P_{aux}, lower f_{BS}

	ACT-adv	ACT-cons	ACT3	ACT4
	A phys A tech	C phys C tech	A phys C tech	C phys A tech
R, m	6.25	9.75	8.50	8.00
Ip, MA	11.0	14.0	16.4	11.1
B_T, T	6.0	8.75	6.25	9.0
β_N^{th}	4.75	2.25	4.00	2.25
q_{95}	4.5	8.0	4.25	8.5
H_{98}	1.65	1.25	1.65	1.25
n/n_{Gr}	1.0	1.3	1.0	1.3
f_{BS}	0.91	0.77	0.86	0.81
q_{div}^{peak}, MW/m²	13.3	10.0	9.55	8.6
P_{fus}, MW	1813	2637	2538	1848
Q	42.5	25.0	32.5	27.5
Q_{engr}	6.5	3.1	3.9	4.7
$P_{H/CD}$, MW	42.7	105.5	78.1	67.2
$P_{rad,core}$, MW	115.5	289.8	311.4	171.2
η_{th}	0.58	0.44	0.44	0.58

Concentrate on ACT-adv and ACT-cons

$P_{electric}$ = 1000 MW_e



Kessel et al. (2015)

More recent focus to target pilot plant parameters

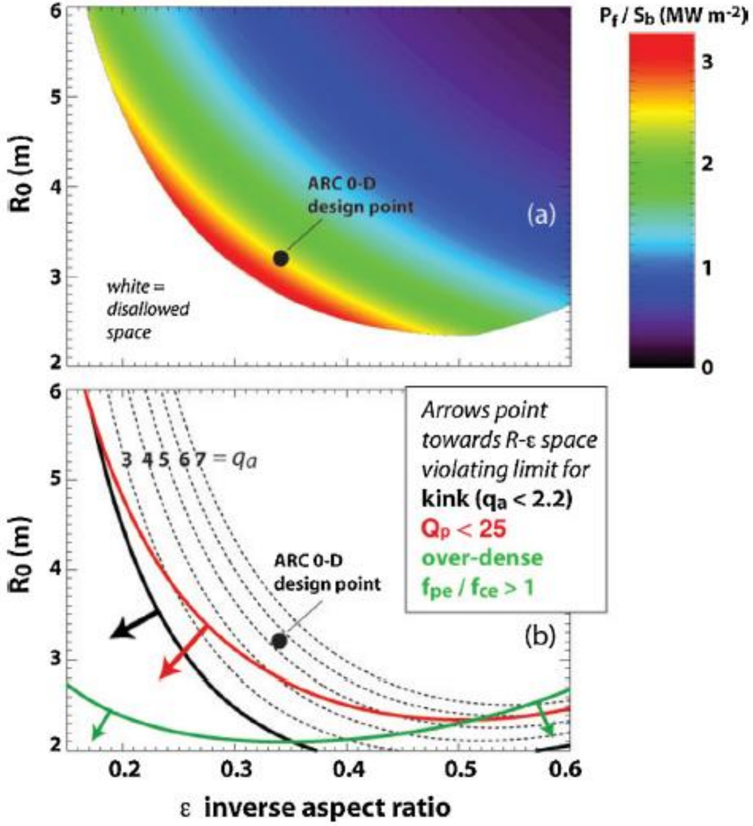
- Lower capacity, P_{net}~100-200 MWe (COE not the immediate concern)
  - Don’t need to demonstrate 100% of all “essential criteria”, as long as solutions perceived to scale

- Target aggressive technology and physics to push for pilot plant at low capital cost (e.g. compact tokamaks, stellarators)

High field HTS magnets enables smaller, net electric pilot plants (ARC design)

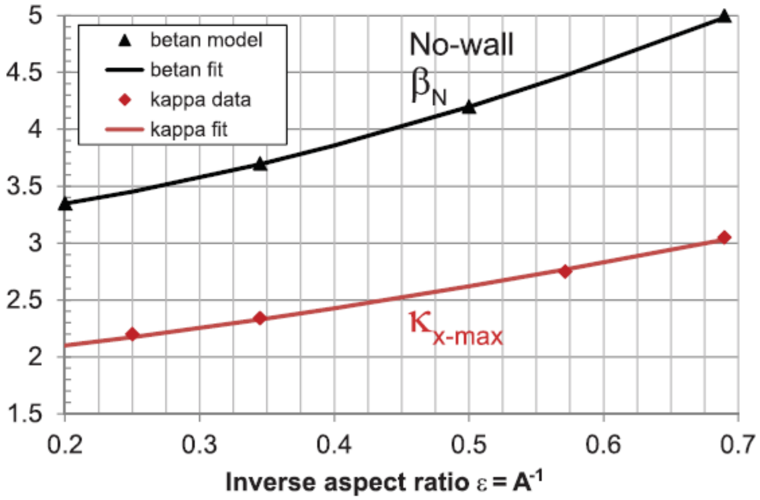
- Targeting fixed fusion power: P_{fusion} = 500 MW
- HTS coil properties: B_{max}=18 T
- Shielding: Δ_b=0.5 m (<1 m blankets)
- Constraints: P_{fusion} = 500 MW, Q_P>25, q*>2.2, β_N=<3, RF heating cutoff, W_n>2.5 MW/m^2

- Enforcing 100% non-inductive → P_{net}=190 MW, f_{BS}=0.63
  - Still requires elevated confinement (H_{98}~1.8) for steady-state



Lower aspect ratio provides opportunities to achieve improved plasma performance

Lower $A=R/a=1/\varepsilon$ can access larger stable β_N and κ



Re-write fusion power & gain in terms of B_{TF}, R_0, ε and β_N, κ stability parameters

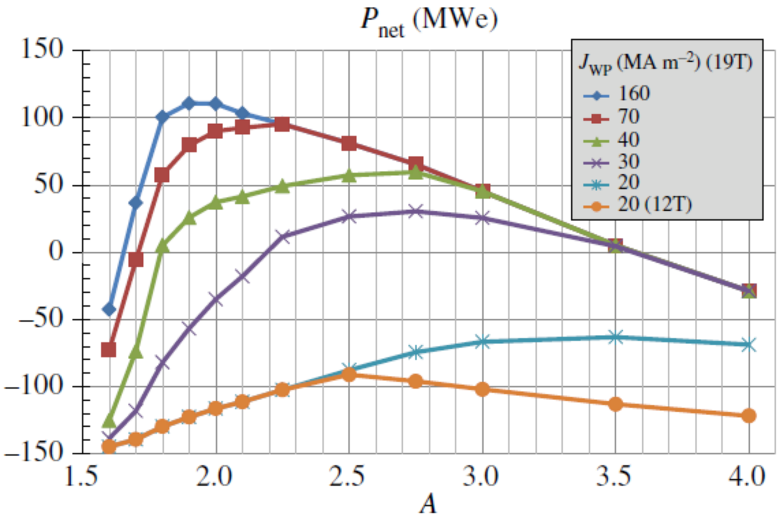
$$P_{fus} \sim B_{TF}^4 R_0^3 \cdot \boldsymbol{\varepsilon^3(1-\varepsilon-b/R_0)^4 \cdot \beta_N^4 \kappa^4} \cdot (C_{BS}/f_{BS})^2$$

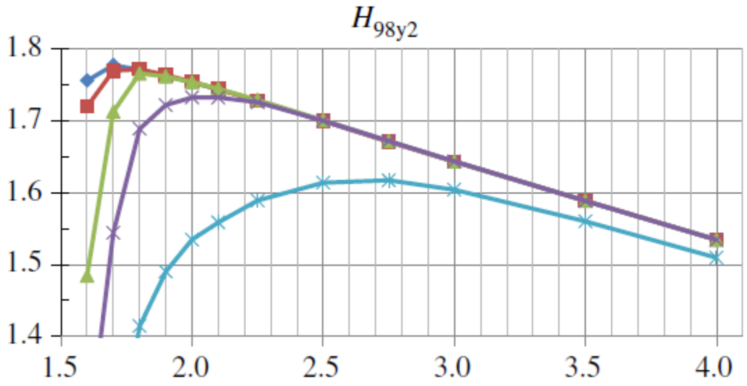
$$Q_{DT}^* \propto f_{gw}^{2/3} \epsilon^{5/3} \left(\frac{HR\beta_N(\epsilon) C_{BS}}{f_{BS}} \right)^2 B_T(\epsilon)^3 \kappa(\epsilon)^4$$

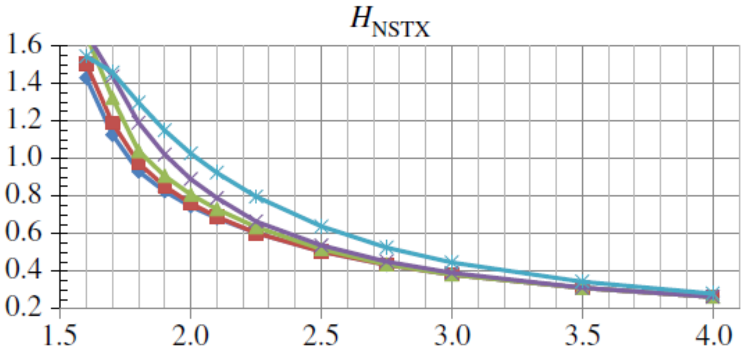
(Petty 08 τ_E scaling)

P_{net} maximized at lower A~2-2.4 if coil current density (J_{WP}) high enough and shielding not too thick

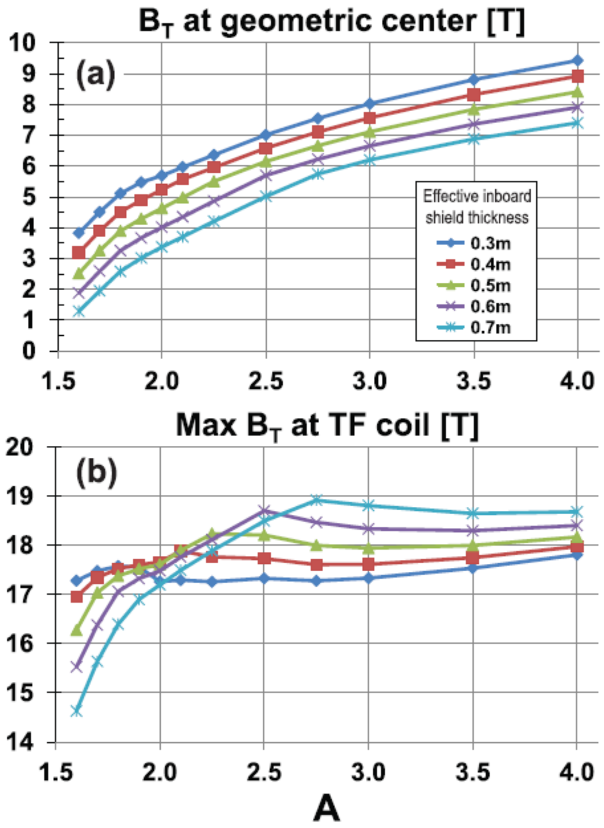
- Required confinement enhancement H_{98} is still large
- Minimal confinement enhancement *if* NSTX confinement scaling remains ← **key NSTX-U research priority**

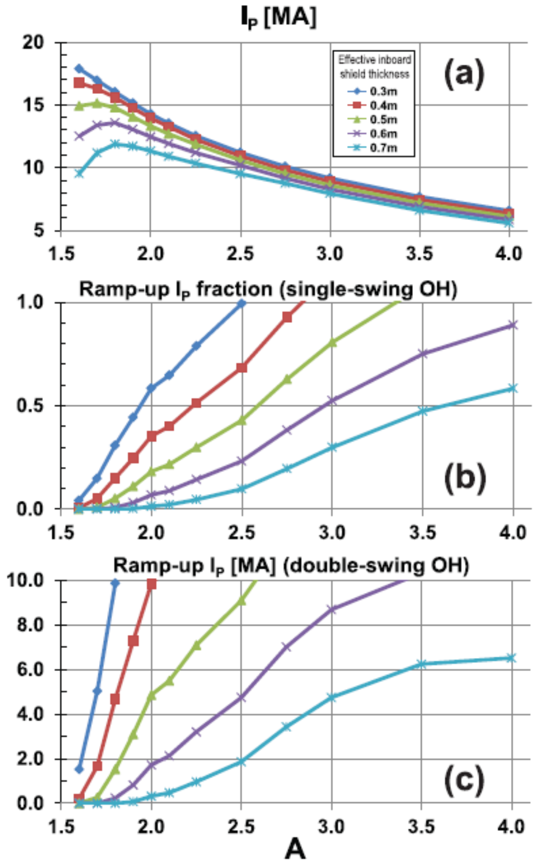






Achievable $B_{TF,max}$ eventually drops with aspect ratio; progressively less central solenoid flux available for ramp-up





Attempts to self-consistently include divertor / plasma heat exhaust solution have recently been made

$$\mathbf{P}_{\alpha} + \mathbf{P}_{\mathbf{aux}} = \mathbf{P}_{\mathbf{loss,conduction}} + \mathbf{P}_{\mathbf{rad,Brehm}} + \mathbf{P}_{\mathbf{rad,sync}} + \mathbf{P}_{\mathbf{rad,imp}}$$

- Additional constraints on required impurity seeding for radiative divertor (detachment) provide upper bound on (R_0,B_0)
  - Plus synchrotron radiation increases at higher B field

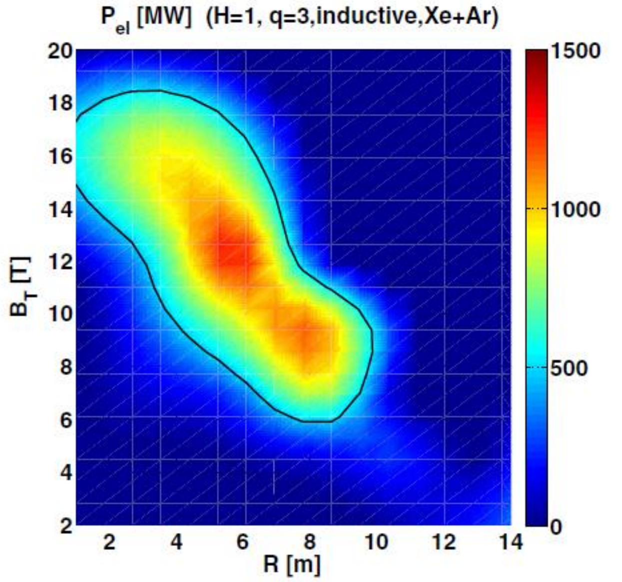


Figure 5. Net electrical power for Xe + Ar cases (i.e. with divertor compatibility enforced). The black line highlights the $P_{el} = 500$ MW level curve. For these simulations, we have set $H = 1$ and $q = 3$ (inductive machines).

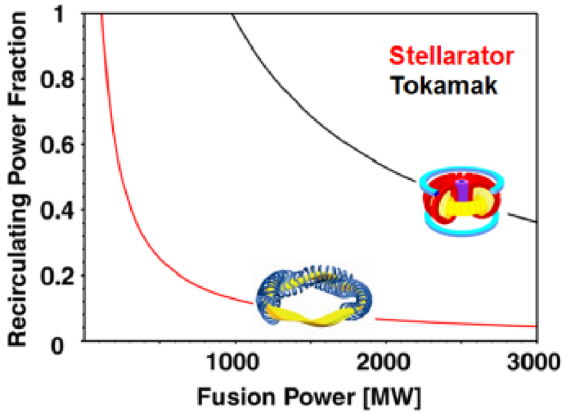
Siccinio, Nucl. Fusion 58, 016032 (2018)

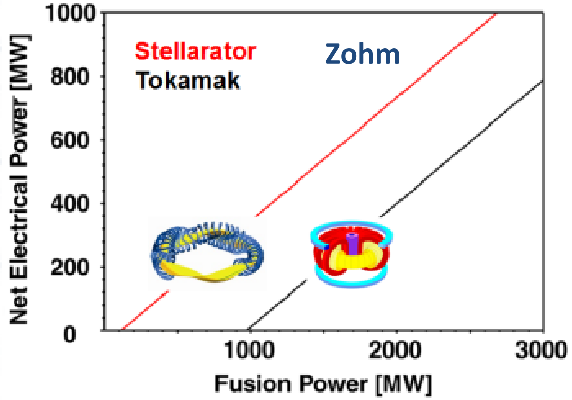
Stellarators offer path to steady-state $P_{net}>0$, $Q_{eng}>1$ at lower P_{recirc} & P_{fusion}

- Intrinsically steady-state, don't need to sustain internal plasma current, P_{aux} can be smaller → much lower recirculating power: $\mathbf{P_{net} \approx P_{fus} \cdot \eta_{th} \cdot 0.94 - \frac{P_{aux}}{\eta_{aux}}}$
  - Avoids current-driven "disruptions" → eases control needs, alleviates some availability/risk concerns
  - Different global MHD stability characteristics → can operate at higher density, potentially relaxing some plasma facing component (PFC) constraints

Bader Day 4

- 3D shape adds complexity to coils, blanket, plumbing, but also provides freedom for optimization ← critical research area
- Modular coils can be constructed and shipped; biggest tokamak PF coils must be wound on-site (e.g. ITER PF coil building)





Summary

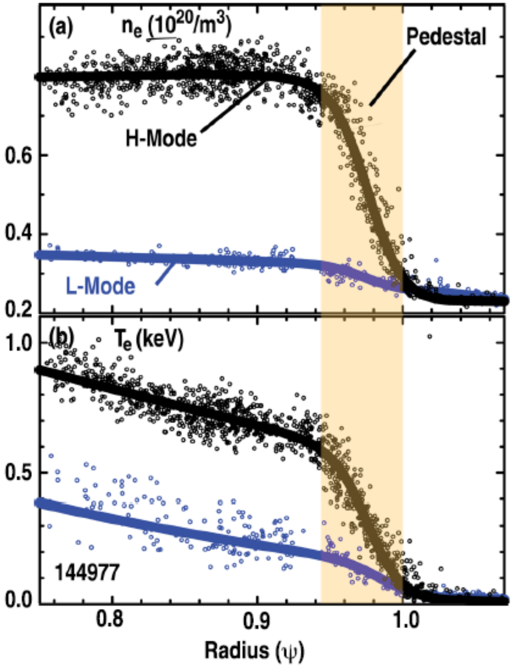
- Conceptual power/pilot plant design studies must consider self-consistent integrated solution including nuclear, engineering and plasma physics constraints
  - Neutron absorption, blanket concepts, superconductor technology, current drive efficiencies, confinement, stability limits, …

- Studies have been performed highlighting key sensitivities that influence performance → **motivates critical, innovative research in US & worldwide fusion programs**

Tokamak plasmas exhibit a few different operating regimes

- L-mode (“low” confinement mode)
- H-mode (“high” confinement mode): with sufficient power passing through the edge, an edge transport barrier spontaneously develops → considerably improved confinement
  - There are others (QH-mode, I-mode, VH-mode, EP-H mode, …)

- Access to H-mode depends on a “L-H threshold power”, $\mathbf{P_{L/H} \sim 0.049 \cdot B^{0.80} n^{0.72} S^{0.94}}$
- Power crossing the last closed flux surface (P_{sep} = $P_{conduction,loss}$ – P_{rad}) must be bigger then $P_{L/H}$ to remain in H-mode: $\mathbf{P_{sep} > P_{L/H}}$



Inductive current drive (for ramp-up and "flat top" in tokamaks) requires a central solenoid (CS)

$$\frac{\partial}{\partial t}\int \mathrm{B}\cdot d\mathrm{S} = -\oint \nabla\times \mathrm{E}\cdot d\mathrm{S} = -\oint \mathrm{E}\cdot d\mathrm{l}$$

$$\frac{\partial}{\partial t}\Phi_{\mathrm{CS}} = -\mathrm{V}_{\mathrm{loop}}$$

$$\boldsymbol{\Delta\Phi_{CS} = \Delta\Phi_{res} + \Delta\Phi_{ind}}$$

- Can estimate how much central solenoid flux $\Delta\Phi_{CS}$ (V-s) required to ramp-up plasma current (prior to 100% non-inductive flat-top):
  - Resistivity: $\Delta\Phi_{res} \sim \mu_0 I_p R_0$
  - Self-inductance: $\Delta\Phi_{ind} \sim L_P I_p$

$$\boldsymbol{\Delta\Phi_{CS} \sim I_p}$$

- Motivates research on non-inductive plasma break-down and current ramp-up schemes (Diem, Day 2; Battaglia, Day 4)

Shrinking aspect ratio ultimately constrained by stress limits, current density limits, central solenoid requirements

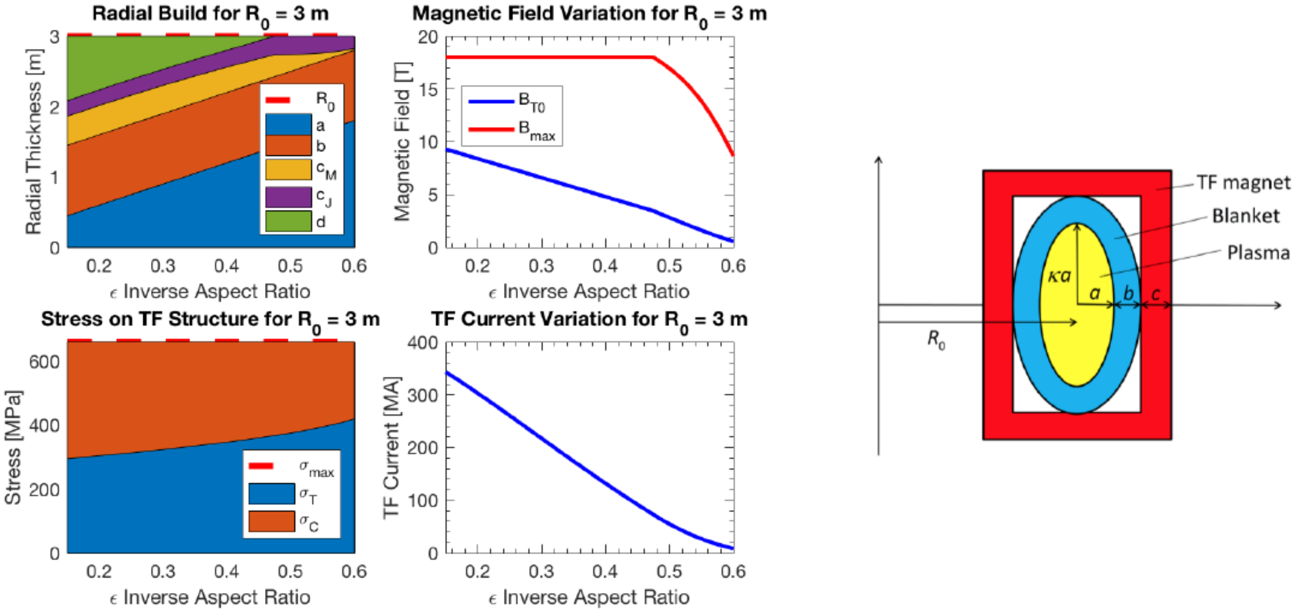


Figure 1: (left) Performance characteristic of REBCO magnets ($B_{max} = 18T, J_{WP} = 70\frac{MA}{m^2}$) with $\sigma_{max} = 660MPa$ and $b = 1.0m$. (right) Simple depiction of radial build where $c = c_J + c_M$.

C. Perks, SULI 2018 project