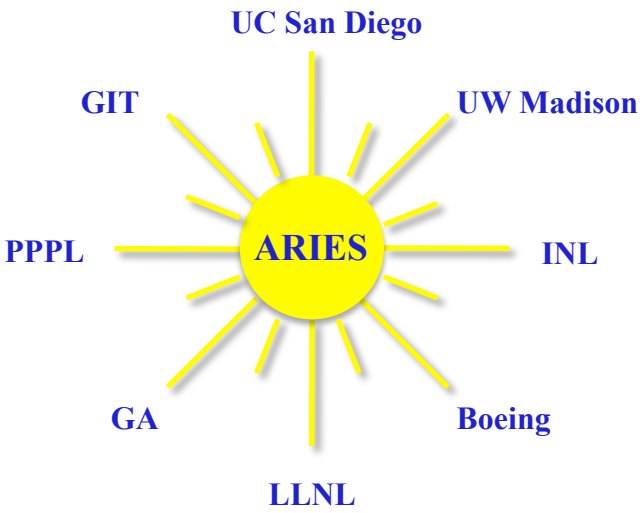


ARIES Advanced & Conservative Tokamak Power Plant Study: Impacts of Physics and Technology Assumptions

C. E. Kessel for the ARIES Team

US-JA Workshop on Fusion Power Plants and Related Advanced Technologies, UCSD, March 13-14, 2014

"Conservative" and "Advanced" (and Common) Assumptions – P_{elec} = 1000 MW

Physics:	Conservative	Advanced
β_N^{total}	2.6 (no wall)	5.8 (with wall)
H_{98}	1.25	1.65
n/n_{Gr}	as low as possible	

Technology:	Conservative	Advanced
blanket concept	DCLL (Fe-steel, He, LiPb) (η_{th}~0.44)	SCLL (SiC-comp) (SiC , LiPb) (η_{th}~0.58)
q_{div}^{peak}, MW/m^2	$\leq$10 MW/m^2	$\leq$15 MW/m^2

Common Assumptions

A = 4.0

Up-down sym DN

κ_X = 2.2
δ_X = 0.625

$f_{rad,div}$ = 0.90 ($P_{rad,div}/P_{SOL}$)

Ip = 100% non-inductive (CS for rampup assist and equilibrium)

Nb_3Sn superconductors, TF and PF

He-cooled, tungsten divertor

$Li_{17}Pb_{83}$ breeder

Systems Analysis Provides the Integrated Configuration Operating Point(s)

The database approach solves for large numbers of viable points by scanning all the critical plasma configuration variables (R, B_T, q_{95}, β_N, κ, δ, ε, Q, l_i, T profile, n profile, impurity fraction, n/n_{Gr}, η_{CD}, τ_p^*/τ_E)

Physics configurations are passed through all the primary engineering assessments with input from neutronics for the IB radial build of FW, blanket, shield (adjustable to $\langle N_W \rangle$), and VV (FW heat flux, divertor heat flux, fusion core and overall plant power balance, TF coil, bucking cylinder/superstructure, CS/PF coils)

Either results can be filtered OR buildout and costing are done, and then the results are filtered

$975 \leq P_{elec} \leq 1025$ MW ← primary filter

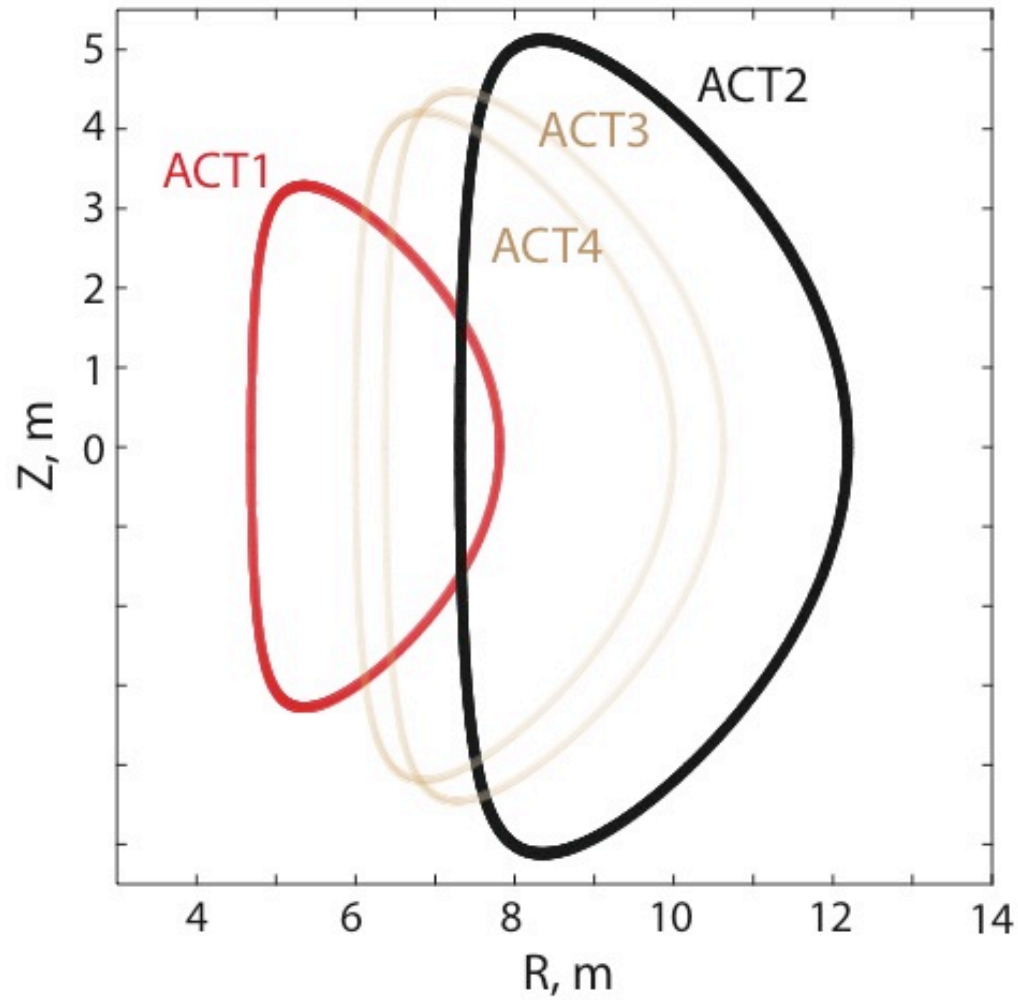
For example, $\beta_N^{total} \leq 5.75$, $n/n_{Gr} \leq 1.0$, $H_{98} \leq 1.6$, $q_{div}^{peak} \leq 15$ MW/m^2….

Secondary filters can include Q, TF Coil J/B, minimizing R, etc.

Physics operating points → Viable engineering points → Full device build-out and costing

Concentrate on ACT1 (adv/adv) and ACT2 (cons/cons)

	ACT1	ACT2	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/m^2	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



Operating Point Variants Around ACT1 (adv phys / adv tech) Reference

	ACT1	ACT1-a	ACT1-b	ACT1-c	ACT1-d	ACT1-e
R, m	6.25	6.25	6.0	6.75	6.25	6.75
Ip, MA	10.9	11.1	11.6	11.5	11.3	11.6
B_T, T	6.0 (11.8)	5.75 (10.2)	7.0 (12.5)	6.5 (11.2)	7.25 (12.8)	5.25 (9.05)
β_N^{th}, β_N^{fast}	4.75, 0.85	5.0, 0.96	4.50, 0.83	4.25, 0.82	4.0, 0.77	5.0, 0.96
q_{95}	4.5	4.25	4.75	5.0	5.25	4.0
H_{98}	1.65	1.62	1.65	1.65	1.57	1.52
n/n_{Gr}	1.0	1.0	1.0	1.0	1.0	1.0
q_{div}^{peak}, MW/m^2	13.3	12.8	14.7	10.5	14.0	13.1
$P_{H/CD}$, MW	42.7	42.7	49.3	50.5	53.6	67.1
P_{fus}, MW	1813	1919	2096	1894	2009	2012
f_{BS}	0.91	0.90	0.91	0.90	0.89	0.85
$\langle N_w \rangle$, MW/m^2	2.45	2.60	3.08	2.20	2.72	2.33
$P_{rad,core}$	115.5	106.8	156.7	132.1	129.4	102.9
COE	64.3	64.4	66.3	67.0	66.5	66.6

Time-Dependent Simulations are Done for the in 4-Corners Activity for ACT1 and ACT2

Tokamak Simulation Code (TSC) free-boundary simulation is used to show plasma growth, volt-second consumption, current profile evolution, temperature profile evolution, heating and current drive, radiated powers, etc….coupled to TRANSP

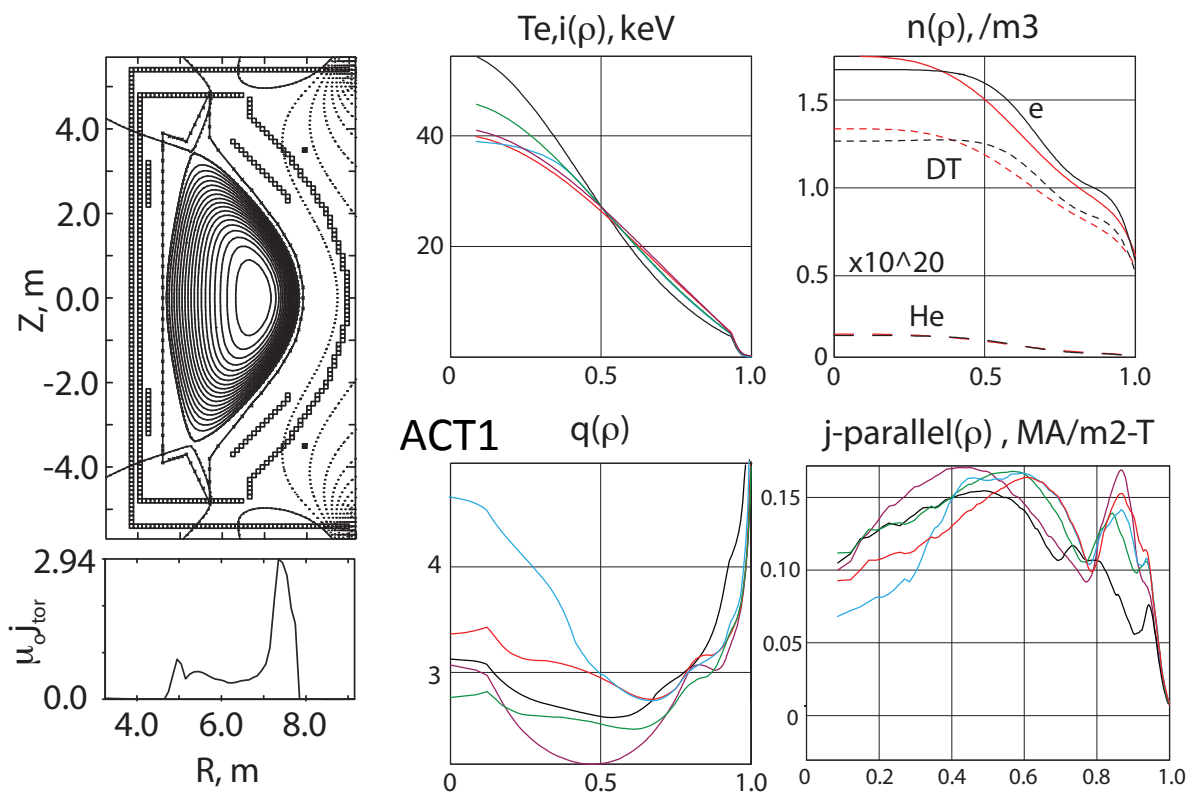
This includes conducting structures, internal feedback coils, and poloidal field coils

ACT1

Density profiles are prescribed

Energy transport model is L-mode with enforced pedestal height from EPED1…scaled to provide target stored energy (β_N)

5 cases examined with different density and temperature profiles



We are Continuing and Expanding the Ideal MHD assessments

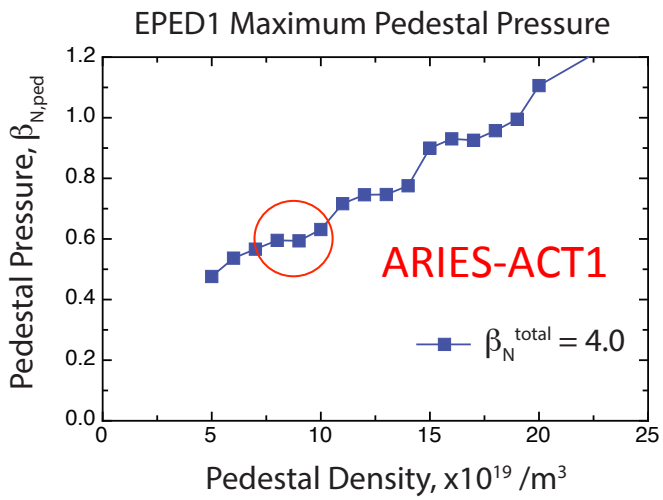
Low-n external kink stability (PEST1), and incorporating both a W stabilizing shell in the blanket, and feedback coils on each sector behind shield to assist in stabilization

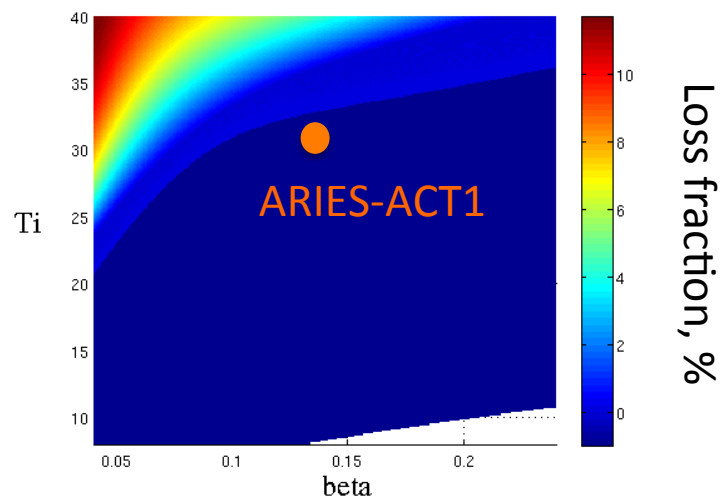
High-n ballooning (BALMSC) to constrain pressure profiles to those that are stable

Peeling-ballooning stability (EPED1) to constrain the H-mode pedestal pressure height and location

Vertical stability obtained by W stabilizing shell in blanket, and feedback coils located behind the shield

Fast particle stability (quasi-linear and NOVA-K) to address whether alpha particles are lost or redistributed with advanced tokamak profiles, which can aggravate these effects





Heating and Current Drive

ACT1: Advanced Physics

At high β_N and sufficiently high q_{95} the f_{BS} is high ~ 90%

Wall stabilization allows low li configurations

10% of 11 MA must be driven

On-axis CD, ICRF/FWCD (65 MHz)

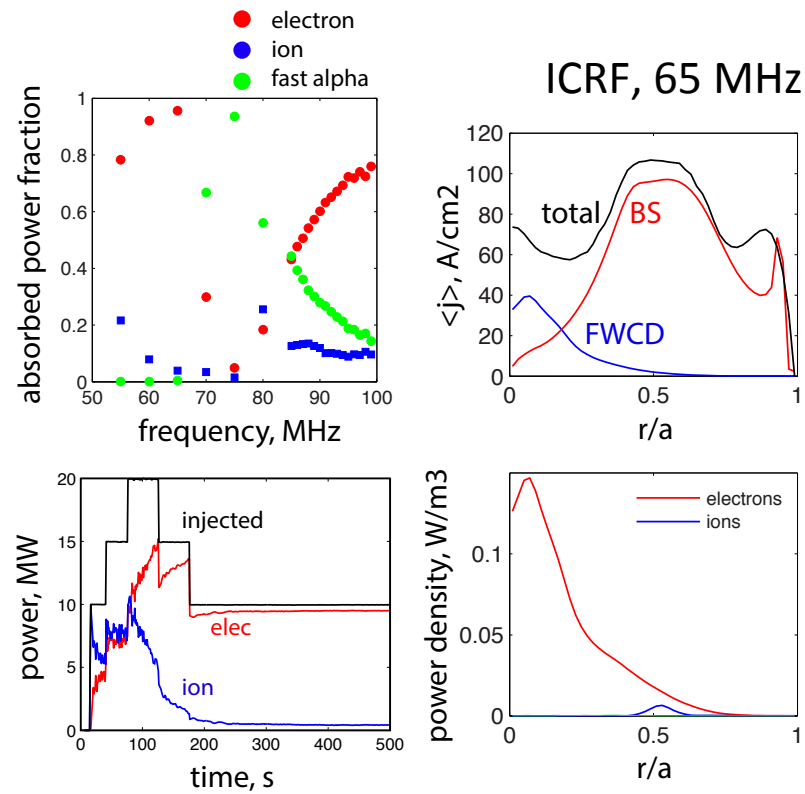
η_{CD} = 0.048 A/W, central deposition

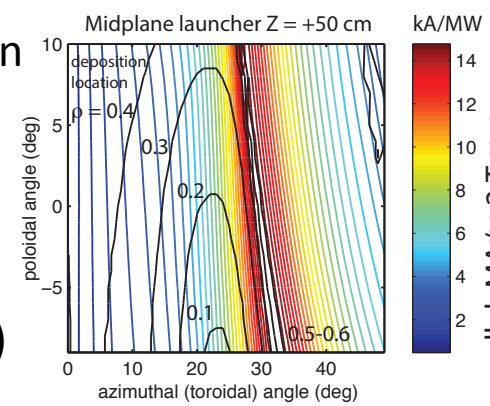
Off-axis CD, LHCD (5 GHz, $n_{||}^{+}$~2.0)

η_{CD} = 0.028 A/W, ρ ~ 0.8-0.95

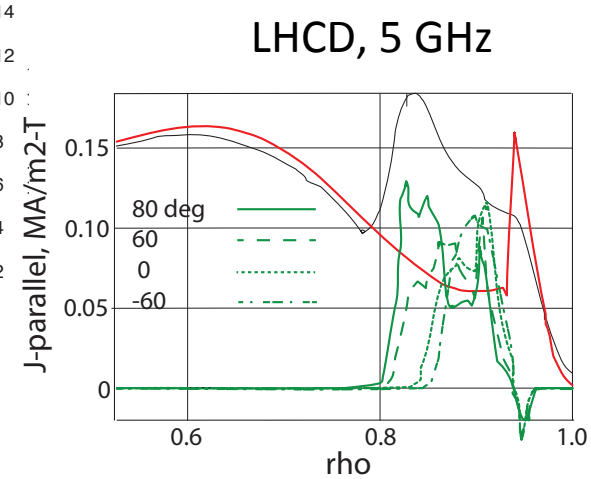
Off-axis, inner ½ radius, ECCD (170 GHz)

η_{CD} = 0.012 A/W, but flexibility in deposition

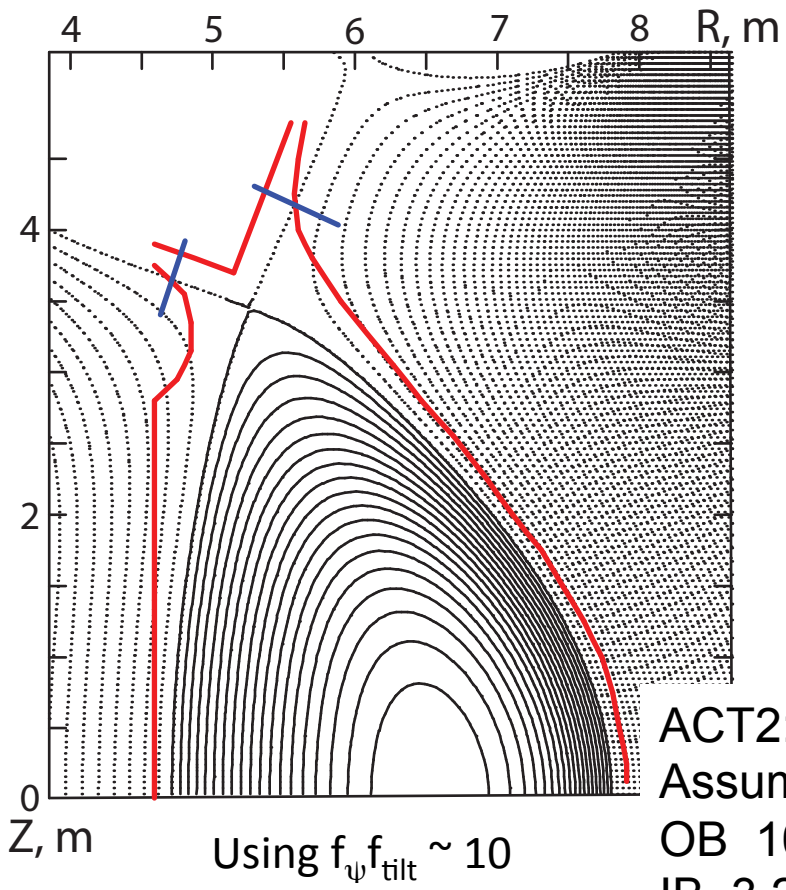




ECCD, 170 GHz



Steady State Divertor Heat Loading – Used in Systems Analysis to Find Operating Points



Using $f_\psi f_{tilt}$ ~ 10

Assuming detached divertor solution to reach high radiated powers in the divertor slot of 90%

Fundamenski, 2005

$$\lambda_{pow} \sim 7.5e\text{-}2\ q_{95}^{0.75}\ n_u^{0.15} / (P_{SOL}^{0.4}\ B_T)$$

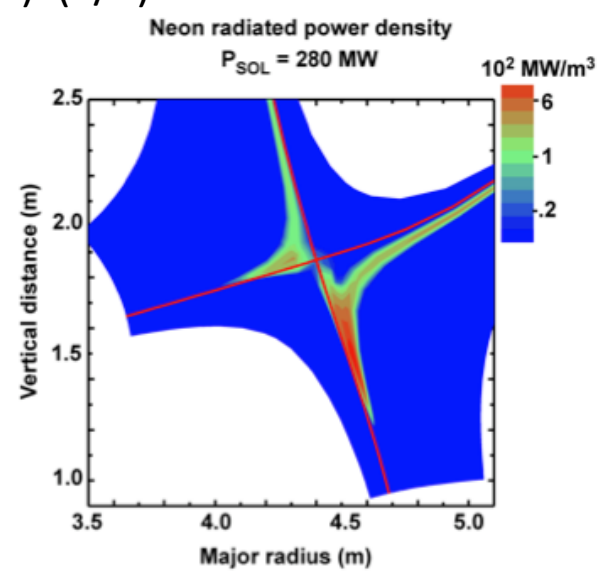
$$q_{div,peak}\ (MW/m^2) = P_{SOL}\ f_{IB/OB}\ f_{vert}\ x\ [\ (1-f_{div,rad})/A_{div,cond} + f_{div,rad}/A_{div,rad}\ ]$$

$$A_{div,cond} = 2\ \pi\ (R-a/2)\ \lambda_{pow} f_\psi f_{tilt}$$

$$A_{div,rad} = 2\ \pi\ (R-a/2)x(a/2)x2$$

ACT2:
Assuming $f_{rad,div}$ = 90%
OB 10.0 MW /m^2
IB 3.28 MW/m^2

ACT1:
Assuming $f_{rad,div}$ = 90%
OB 13.7 MW /m^2
IB 5.4 MW/m^2



UEDGE analysis, LLN

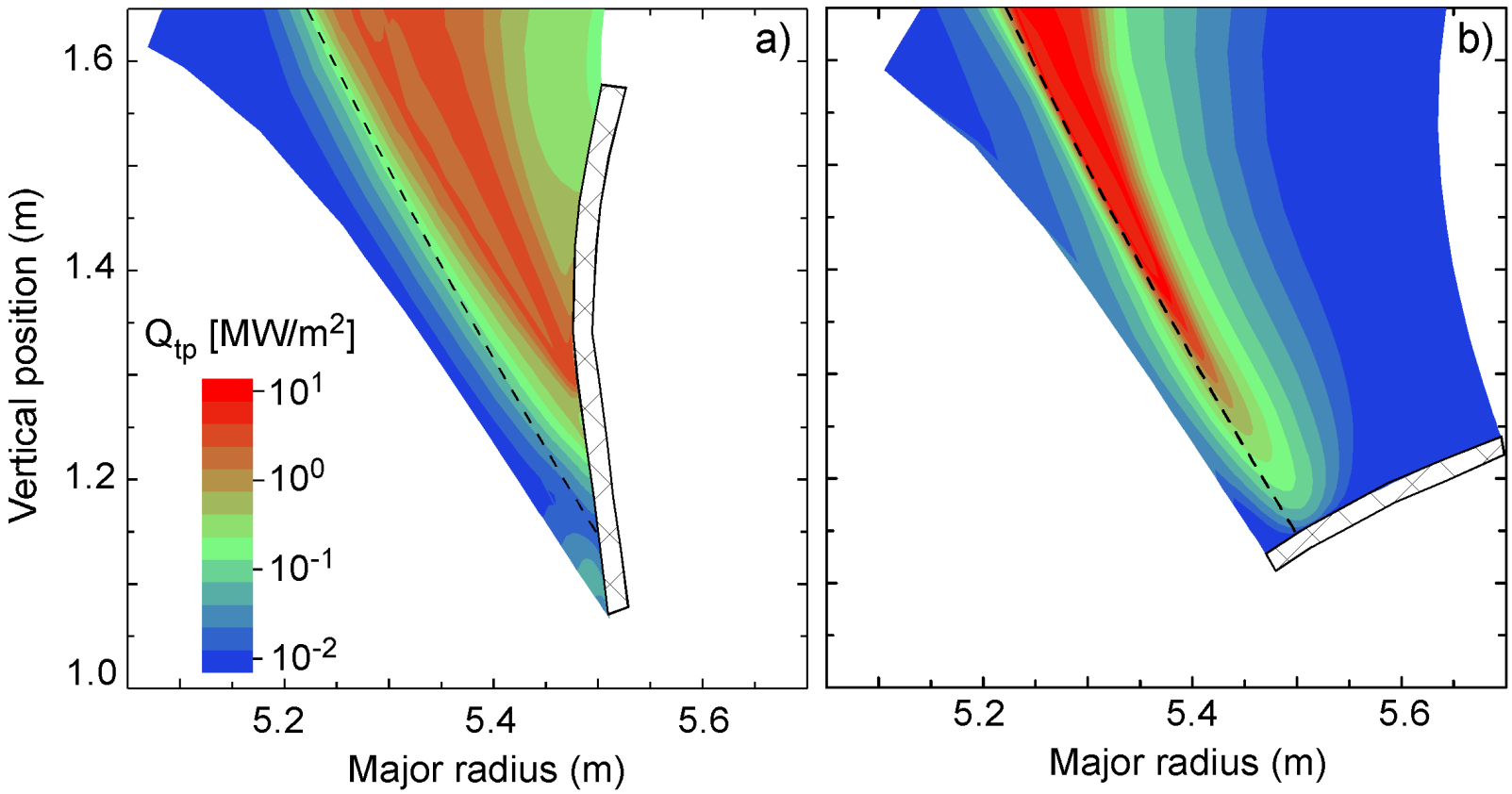
Differences in Outer Leg Plasmas for Two Geometries Show Partial and Full Detachment

Order-of-magnitude estimate of local poloidal plasma heat flux is $Q_{tp} = (B_p/B)nTv_t$

LLNL

Tilted-plate partial detachment has strong in/out asymmetry

Flat-plate full detachment provides gas cushion on both sides of sep.

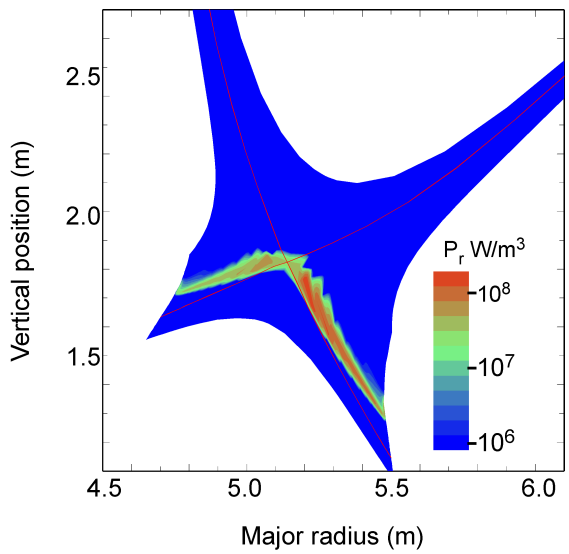


Partial detachment provides $f_{div,rad}$ ~ 0.75

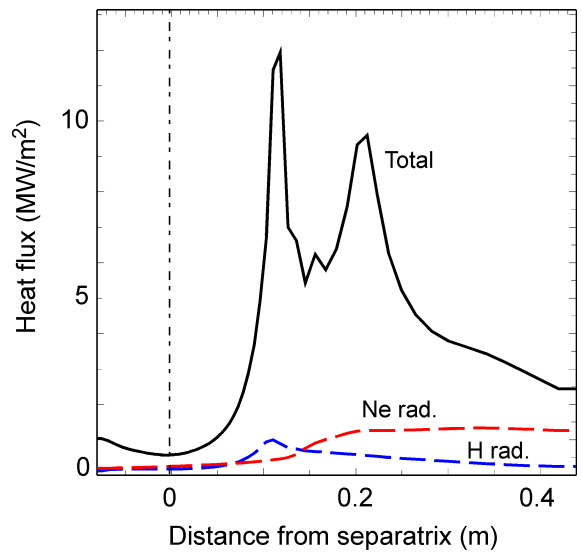
Full detachment provides $f_{div,rad}$ ~ 100%

Heat Flux to Target Plates and Side Walls Are Close to 10 MW/m^2 or Less

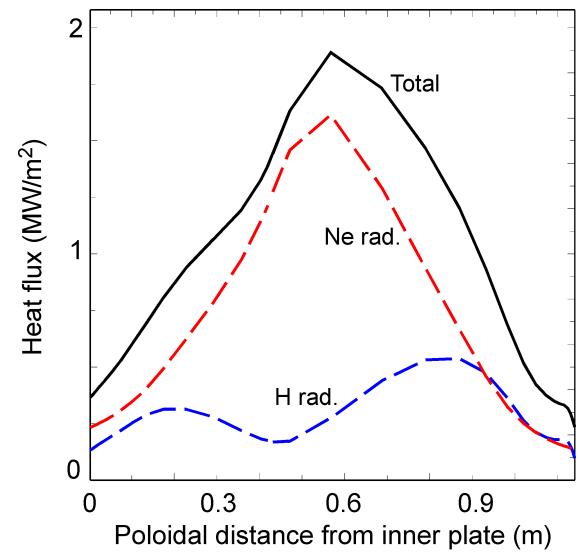
Radiation from seeded neon (or Ar) concentrated in divertor legs



Heat flux to outer target plate is ~12 MW/m^2, mostly from plasma



Heat flux to private flux dome is dominated by line radiation as is outer wall



For alternate wide "slot" divertor: fully detached plasmas with radiation-dominated heat flux ~2 MW/m^2 on target plates and walls; stable operating window can be expanded via impurity feedback on ~1 sec timescale

Comparison of ARIES-ACT1 and ARIES-AT

Inclusion of physics developments since 1999 and better treatments

Narrow power scrape-off width, higher divertor heat load

Peeling-ballooning consistent pedestal

Lower triangularity to accommodate engineering space and shielding

Improved ICRF, LH, and EC modeling

1.5D consistent configurations providing limits to profile broadness

Lower wall plug efficiencies

	ARIES-ACT1	ARIES-AT
R, m	6.25	5.20
a, m	1.56	1.30
κ_x	2.2	2.2
δ_x	0.63	0.90
β_N^{max}	5.75	6.00
B_T, T	6.0	5.86
$l_i(3)$	0.47	0.29
$q_{div,Ob}^{peak}$, MW/m^2 $f_{div,rad}$ = 0.9	13.7	22.6
Ip, MA	11.0	12.8
q_{95}	4.5	3.3
$\langle n \rangle_v$, /m^3	1.33	2.15
n/n_{Gr}	1.0	1.0
P_{aux}, MW	45	37
P_{fusion}, MW	1856	1758

ARIES-AT originally calculated q_{div}^{peak} = 5 MW/m^2 in 1999

Operating Point Variants Around ACT2 (cons phys / cons tech) Reference

	ACT2	ACT2-a	ACT2-b	ACT2-c	ACT2-d	ACT2-e
R, m	9.75	9.25	9.75	10.0	10.0	8.75
Ip, MA	13.98	13.3	11.6	13.9	14.5	12.95
B_T, T	8.75 (14.4)	8.75 (14.6)	8.75 (14.4)	8.75 (14.3)	8.0 (13.1)	8.75 (14.8)
β_N^{th}, β_N^{fast}	2.25, 0.35	2.25, 0.37	2.25, 0.37	2.25, 0.36	2.25, 0.38	2.5, 0.48
q_{95}	8.0	8.0	8.25	8.25	7.25	7.75
H_{98}	1.22	1.13	1.22	1.28	1.06	1.29
n/n_{Gr}	1.3	1.3	1.3	1.3	1.25	1.3
q_{div}^{peak}, MW/ m^2	10.0	14.0	11.0	9.0	14.9	10.5
$P_{H/CD}$, MW	105.5	106.4	95.7	94.8	140.5	77.3
P_{fus}, MW	2639	2663	2635	2609	2814	2516
f_{BS}	0.77	0.77	0.79	0.79	0.69	0.82
$\langle N_w \rangle$, MW/m^2	1.46	1.64	1.46	1.38	1.49	1.73
$P_{rad,core}$	289.8	216.0	249.5	288.8	229.5	255.1
Q	25.0	25.0	27.5	27.5	20.0	32.5
COE	86.2	82.0	84.7	86.9	87.2	77.4

Time-Dependent Free Boundary Analysis with TSC – Reproduce the Systems Target

Energy transport modeled with modified Coppi-Tang to provide different T-profiles with varying peakedness

- Peaking parameter
- H-mode pedestal
- Scaling to reach H_{98}/W_{th} target

EPED1 gives 220 kPa at n_{ped} = 0.8×10^{20} /m^3 (P. Snyder)

Density profile is prescribed

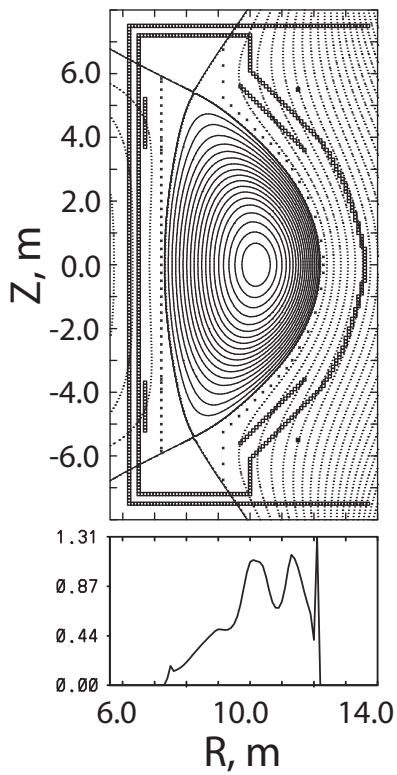
- n(0)/<n> ~ 1.4-1.5

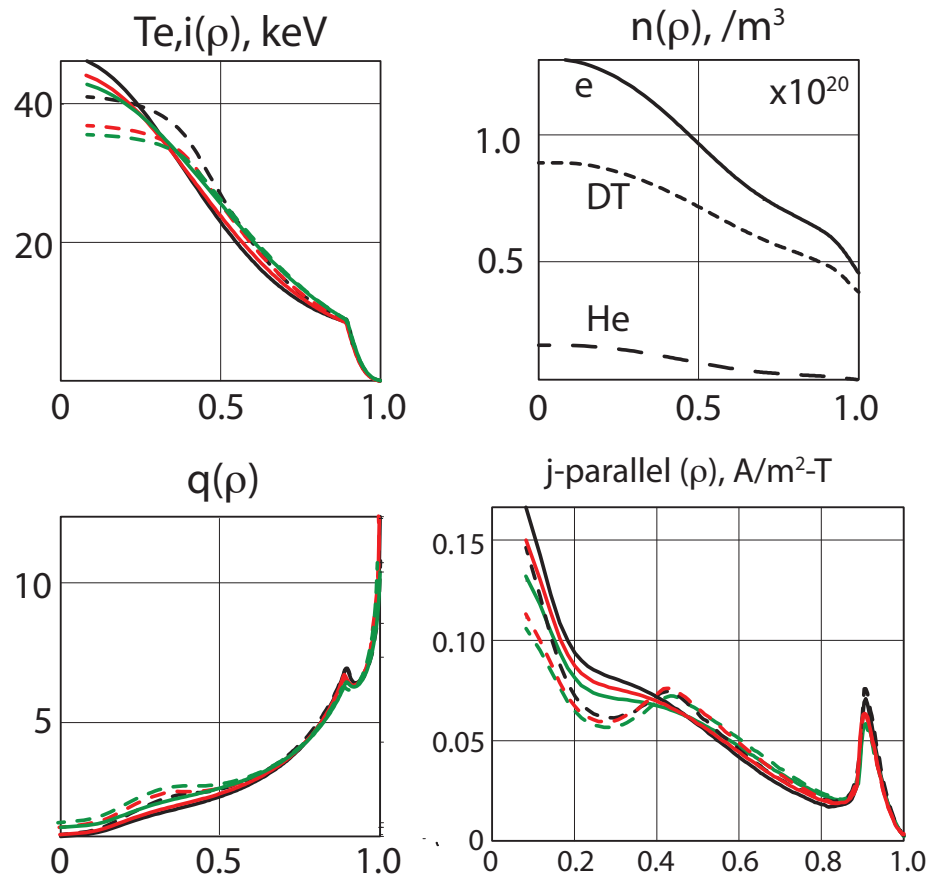
6 cases with p(0)/<p> ~ 2.5-3.2

q(0) = 1.06-1.73

$l_i(1)$ = 0.73-0.88

H_{98} = 1.25-1.37





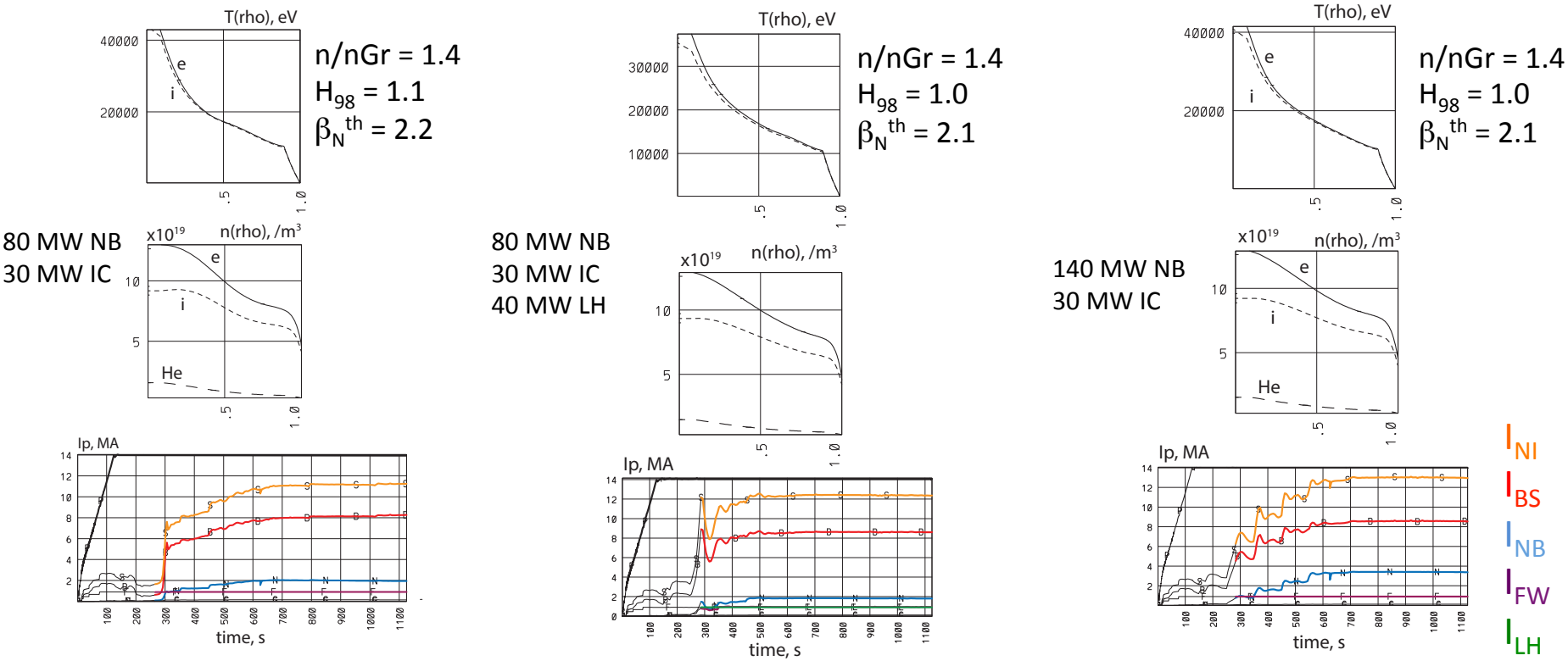
Using GLF23 in Scenario to See Energy Transport Prediction for ACT2

Solving for Ti and Te, n is prescribed, alpha-stab off, rotational stab off (examining low rotation now, not much effect)

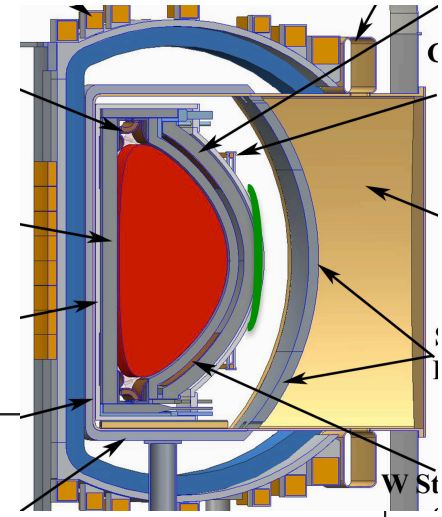
So far, H_{98} ~ 1-1.1
n/n_{Gr} ~ 1.4, keeping $n_{ped} \leq n_{Gr}$
$I_{NI} < Ip$

Density profile and magnitude variations

Injection of LH and more NB to drive more current



Ideal MHD of n=1 external kink mode with no wall and with far-away wall*



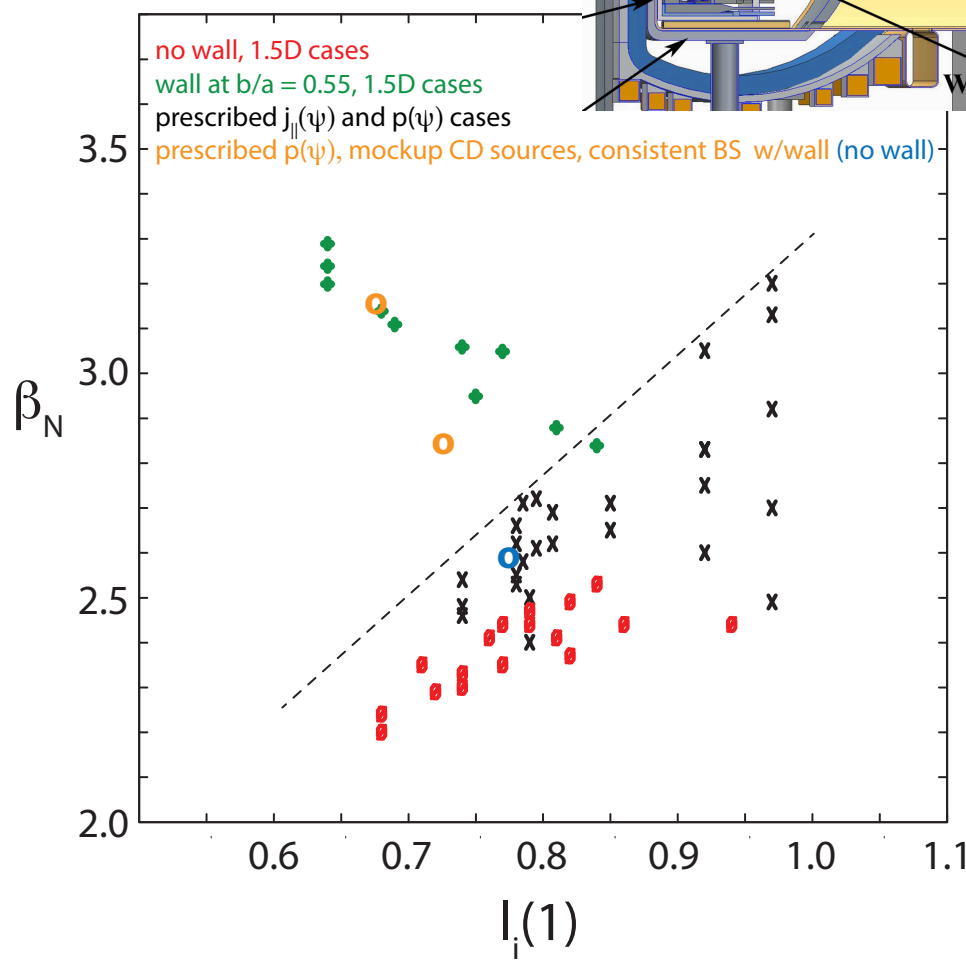
All cases were found to be high-n ballooning stable

1) Parametric scan to find β_N for systems analysis (black X's), prescribe j and p profiles

2) Using H/CD mockup profiles, self-consistent bootstrap current (blue and yellow circles)

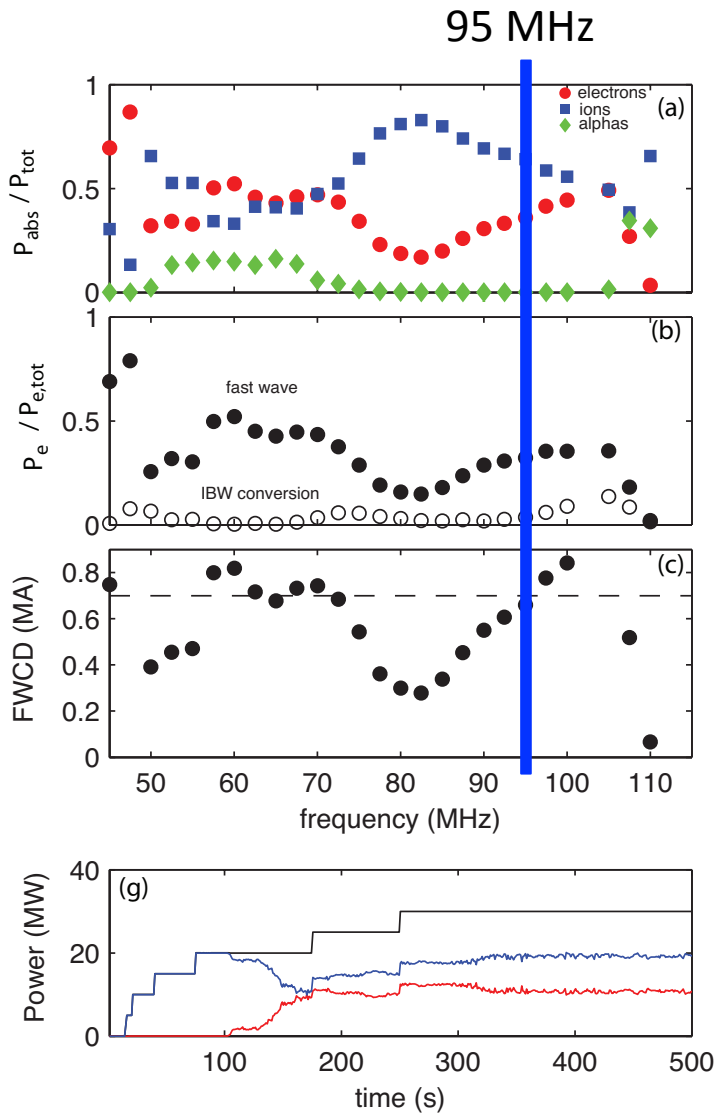
3) Examine 1.5D plasma configurations, adjust p to obtain stability/instability (both fixed j and self-consistent j) (red and green points)

*Wall stabilization requires rotation, feedback, and/or kinetic stabilization



ICRF FW for Ion and Electron H/CD, and LH for CD

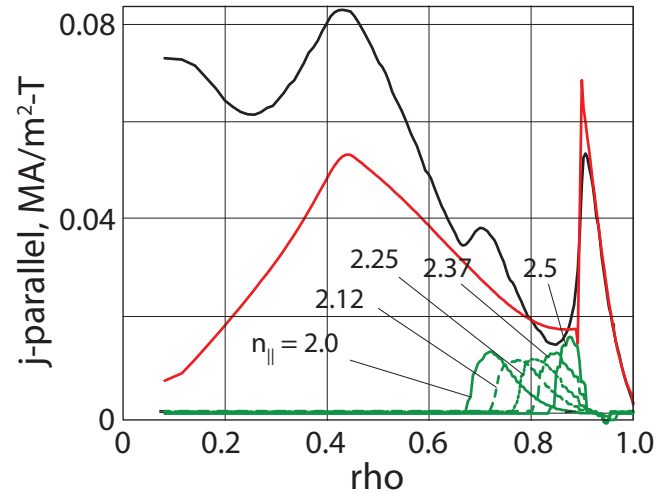
Driving ~ 0.7 MA for 30 MW (nRI/P = 0.26 A/W-m^2)

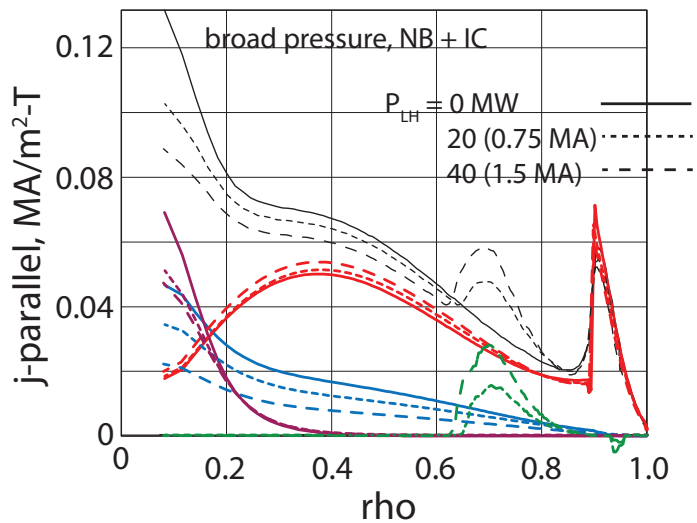


Ideal MHD restricts the total LHCD to ≤ 1.0 MA for no wall, and up to at least 1.5 MA for far-away wall stabilization

Strong electron Landau damping at 15-17 keV, so that the ~ 9keV pedestal did not hinder LH penetration

5 GHz, $n_{||}$ = 2, (nRI/P = 0.28 A/W-m^2)



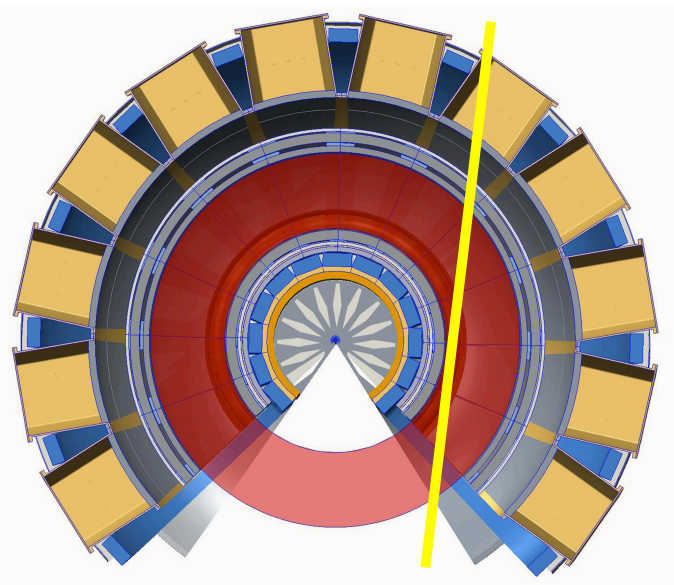


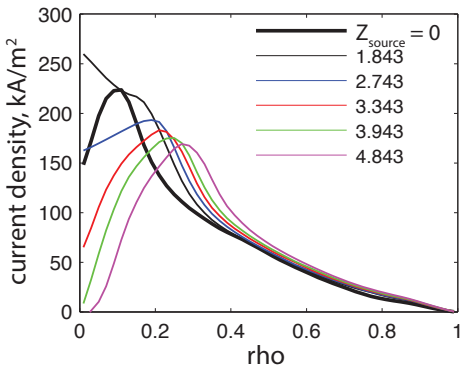
Negative Ion Neutral Beams for Broad Current Profile, Large Bulk CD ~ 3 MA

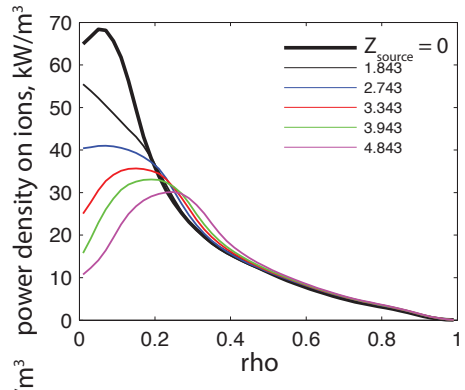
Start with minimal engineering impact geometry

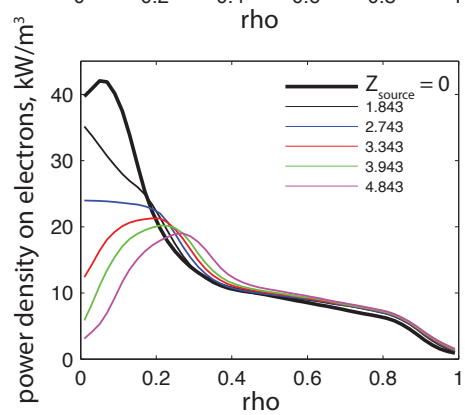
This forces us to ~ R-a tangency radii

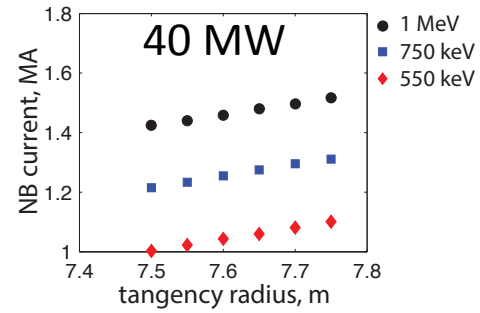
Examined ITER-like NB, same size footprint, energies, and vertical position of source above midplane

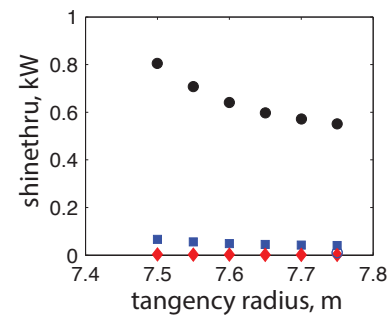












Reference is 1 MeV, sources ~ 2 m above midplane, (nRI/P = 0.37 A/W-m^2), injecting 80 MW

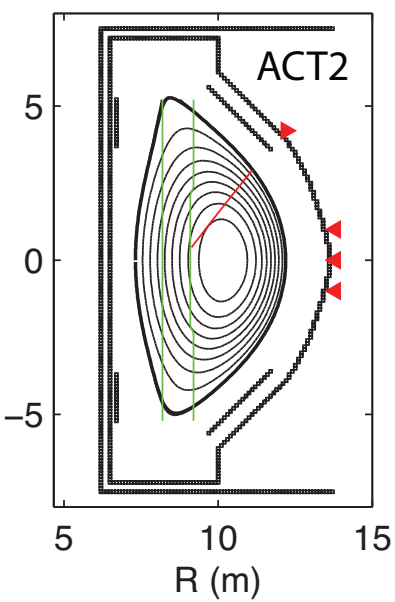
To avoid shinethru in ramp phase , wait until ramp is done

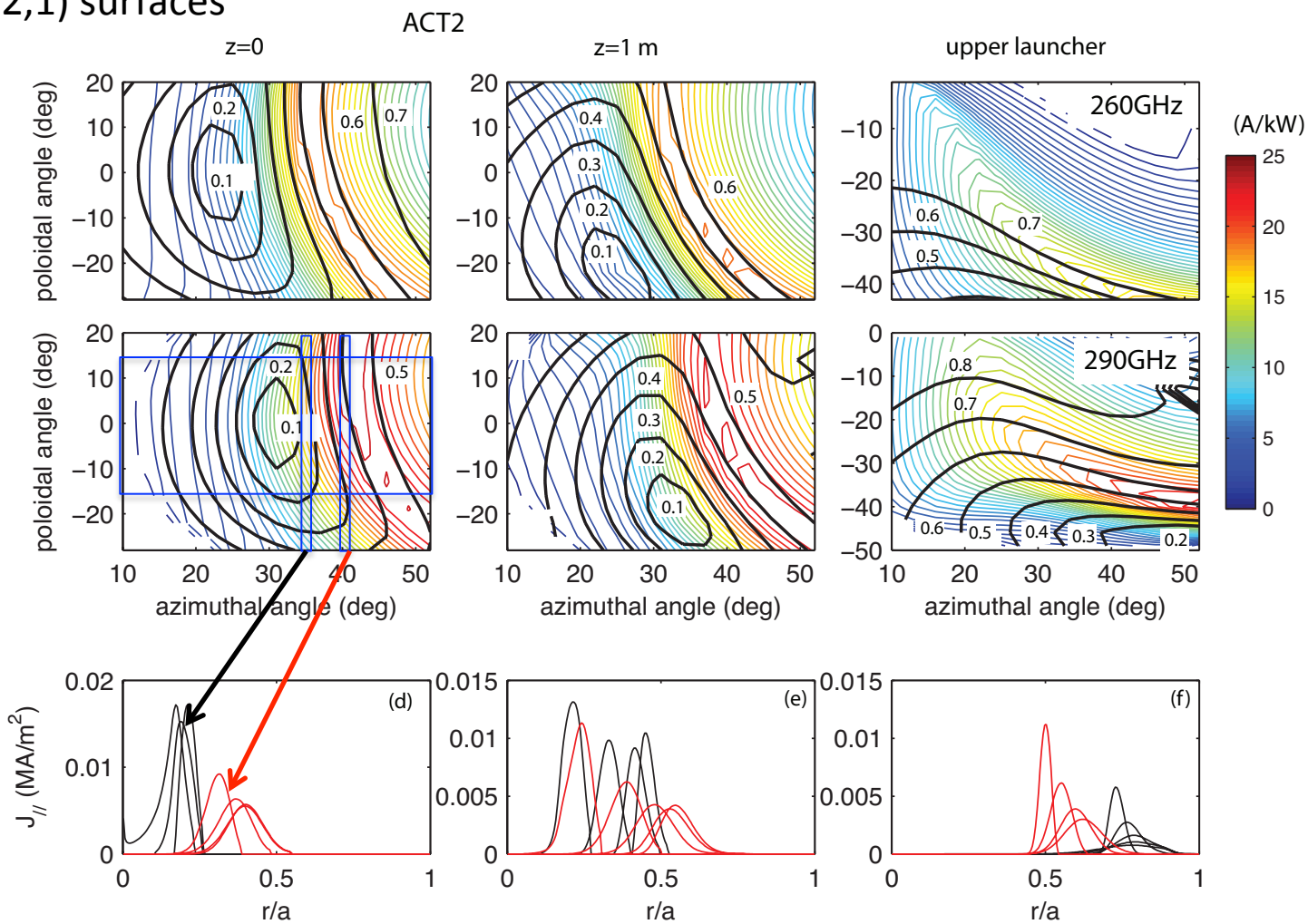
ECCD Has Flexibility Over ρ = 0.2-0.6+ For Adjusting q-profile

For 290 GHz
0.16 A/W-m^2 @ ρ=0.2
0.23 A/W-m^2 @ ρ=0.6

Did not assess need for NTM control, however midplane and upper launchers can reach (3,2) and (2,1) surfaces

EC can be used to locally heat and modify the bootstrap current

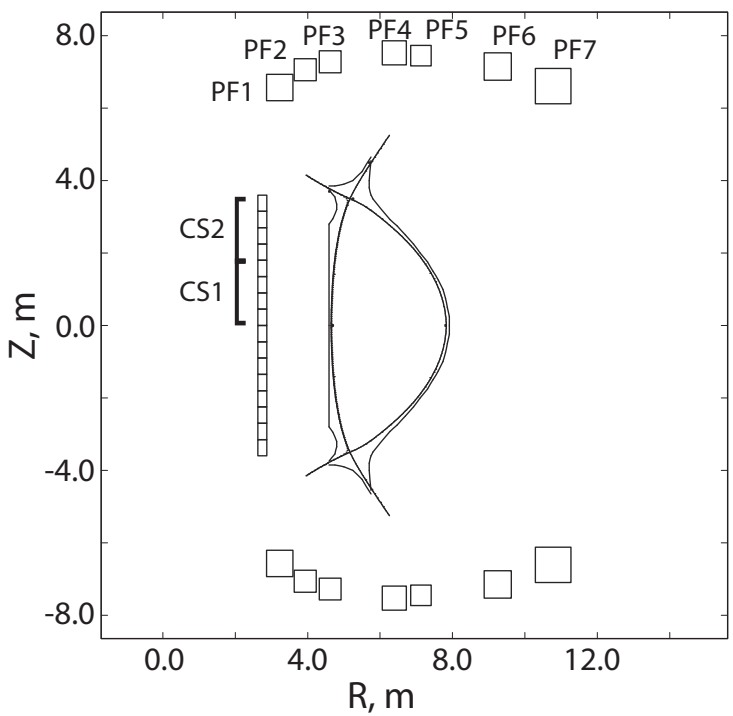




Poloidal Field Coil Layout – ACT1, similar for ACT2

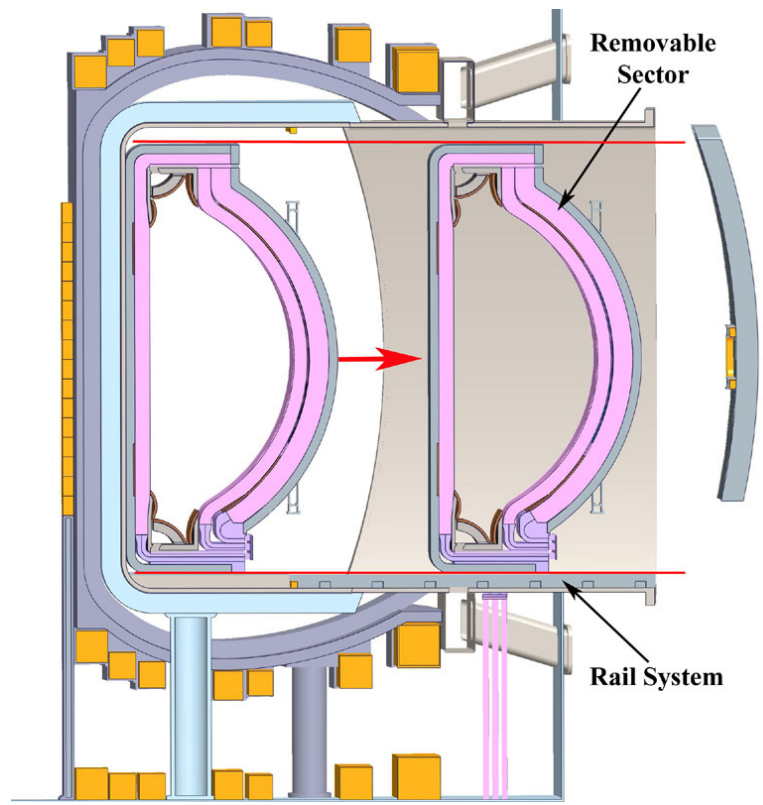
Poloidal field coils drive plasma current in the rampup and they provide the equilibrium force balance

This analysis is done with an equilibrium code

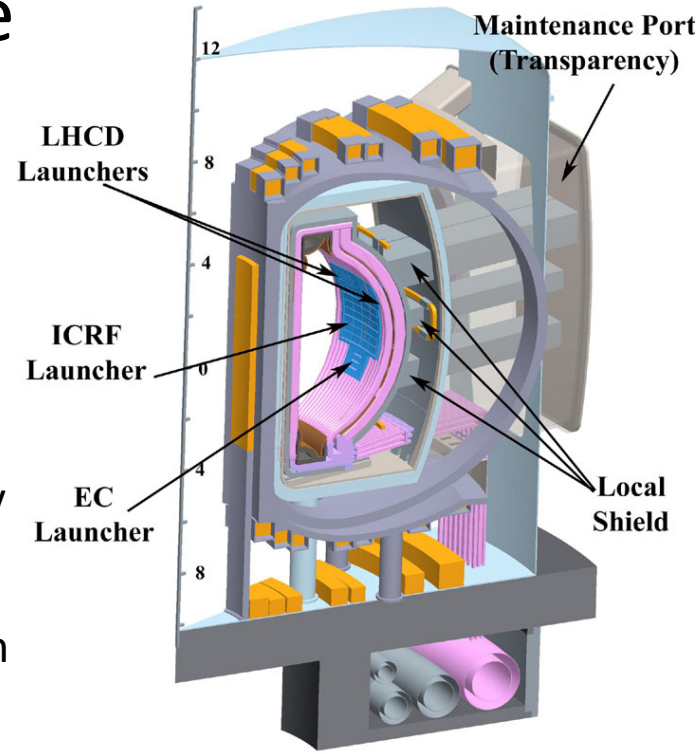


Major contraints include

1) Radial maintenance
2) Lower supports
3) TF coil & support, build

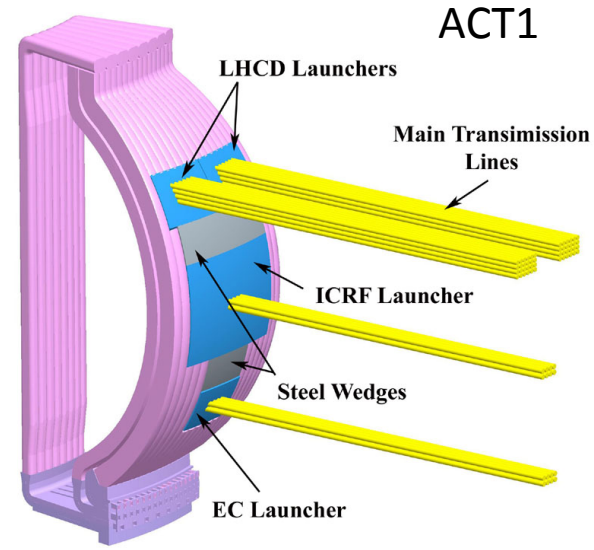


Plasma Heating and Current Drive Systems Hardware Integration



Based on ITER power densities and launcher designs

- Lower Hybrid : 20 MW/m^2, Passive-Active Multi-junction Launcher
- Ion Cyclotron: 10 MW/m^2, 2 Strap Antenna (recently reduced to 5 MW/m^2)
- Electron Cyclotron: 20-40 MW/m^2, 3 Slot Housing on midplane
- Negative Ion Neutral Beam: 25 MW/m^2



Additional volume reserved for support and cooling

Additional shielding is required since these launching structures can have large void fractions

These structures would have to be built out of neutron resistant materials, operated at high temperature, and resist plasma exposure

Physics in the Power Plant Regime

The plasma density in power plants is routinely found to be at or above the Greenwald density limit ($n_{Gr} = Ip/\pi a^2$)

- Actually tokamak experiments have exceeded the limit, but it is not routine to operate there….actually $n_{ped} < n_{Gr}$
- Going to larger devices aggravates this, causing operating points to exceed it even more
- Operating at higher densities weakens fast particle instabilities
- Pellet fueling will be used on power plants and this will aid in operating above the n_{Gr}, but the compatibility with a high density divertor is unknown

Tritium burnup has been a lingering issue for the power plant studies

- Particle transport inside and outside the plasma is not well understood
- The interaction of the core and edge plasmas will be different than present tokamaks…..particles will not penetrate the plasma efficiently as they do today
- The "residence time" of tritium in the core plasma may be strongly reduced, leading to low burnup fractions
- A short residence time for tritium also means a low He residence time, and so a low He concentration in the plasma which is good, and visa versa

Impacts of Physics and Technology Assumptions/ Philosophies

Peak divertor heat flux with narrow power scrape-off width formula (Fundamenski + $f_{rad,div}$) resulted in increasing device size

- ACT1 λ_{pow} = 4 mm
- ACT2 λ_{pow} = 3.8 mm
- Larger devices have reduced the $\langle N_W \rangle$ in all cases

Generally we have not pursued very high Z_{eff} since it strongly degrades RF H/CD sources, values of 1.9-2.3 have been typical

Lowering β_N from "with wall" to "no wall" values also increased device size and increased toroidal field simultaneously

- Leads to high cyclotron losses depending on reflectivity of the FW, and would affect level of intentional high Z impurity

Change in thermal conversion efficiency from SCLL (η_{th}~58%) to DCLL (η_{th}~44%) AND using $\eta_{aux}^{wall\text{-}plug}$ (source, transmission and coupling) = 0.4, also drives us to higher device sizes

- Combining with better estimates of $\eta_{H/CD}$ from detailed analysis

Impacts of Physics and Technology Assumptions/ Philosophies, cont'd

Database approach to systems analysis has helped us see that there are numerous solutions near what we might call an optimum or reference with small changes in the COE

These variations allow one to identify an operating space where we can consider the uncertainties associated with many of our "limits" and benefits of small parameter pushing

ACT1 variants

1) showed a lower $\beta_N^{th} = 4$ case at higher $B_T = 7.25$ T, while still generating 1000 MW_e, indicating this regime should be explored to identify an operating point closer to DIII-D expts ($\beta_N \sim 3.5$-3.7, $I_{NI} \sim I_P$, for 1-1.5 τ_R)
2) Showed a lower $q_{div}^{peak} = 10.5$ MW/m^2 was accessible, albeit at a slightly larger major radius 6.75 m, also at reduced β_N

ACT2 resulted in high $q_{95} \sim 8$ configurations, in order to provide 100% non-inductive current and keep external CD power down….these are not examined on present experiments very often (although JT-60U have many discharges like this)

The large devices in ACT2 results also force us to high n/n_{Gr} largely due to geometry, $n_{Gr} = Ip/\pi a^2$, and defining accessible densities that are consistent with core, edge and divertor is critical