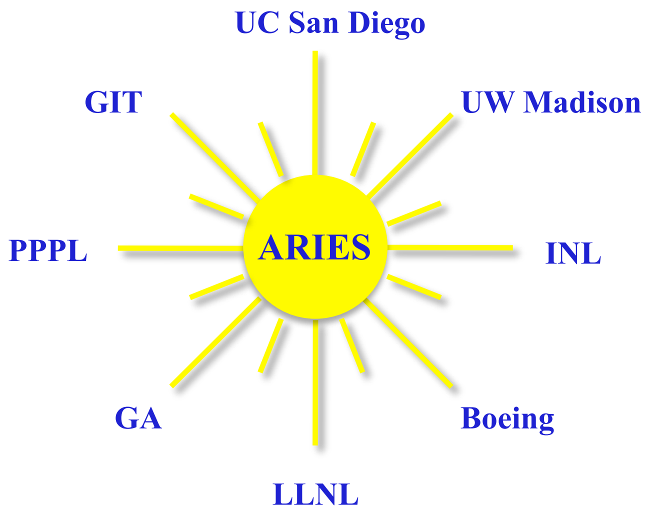

ARIES Physics Analysis

C. E. Kessel

Princeton Plasma Physics Laboratory

Systems Studies COV Program Review, Wash DC, August 29, 2013

ARIES is an Integrated Analysis Activity

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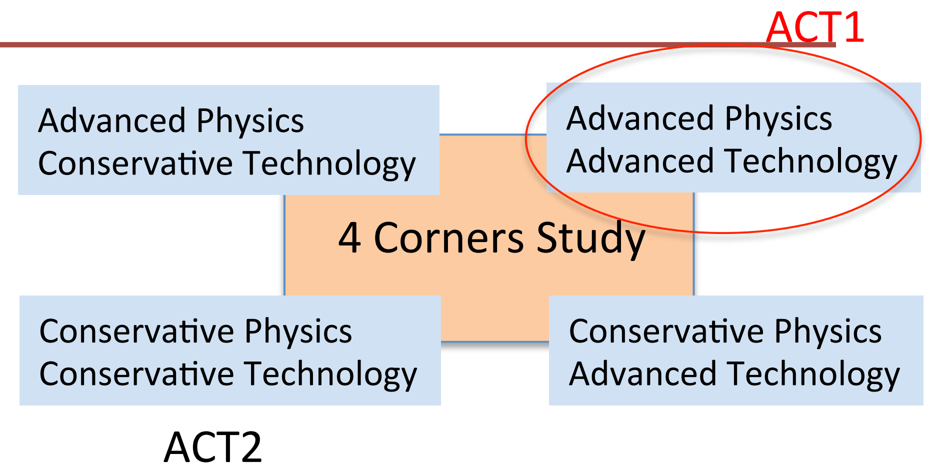
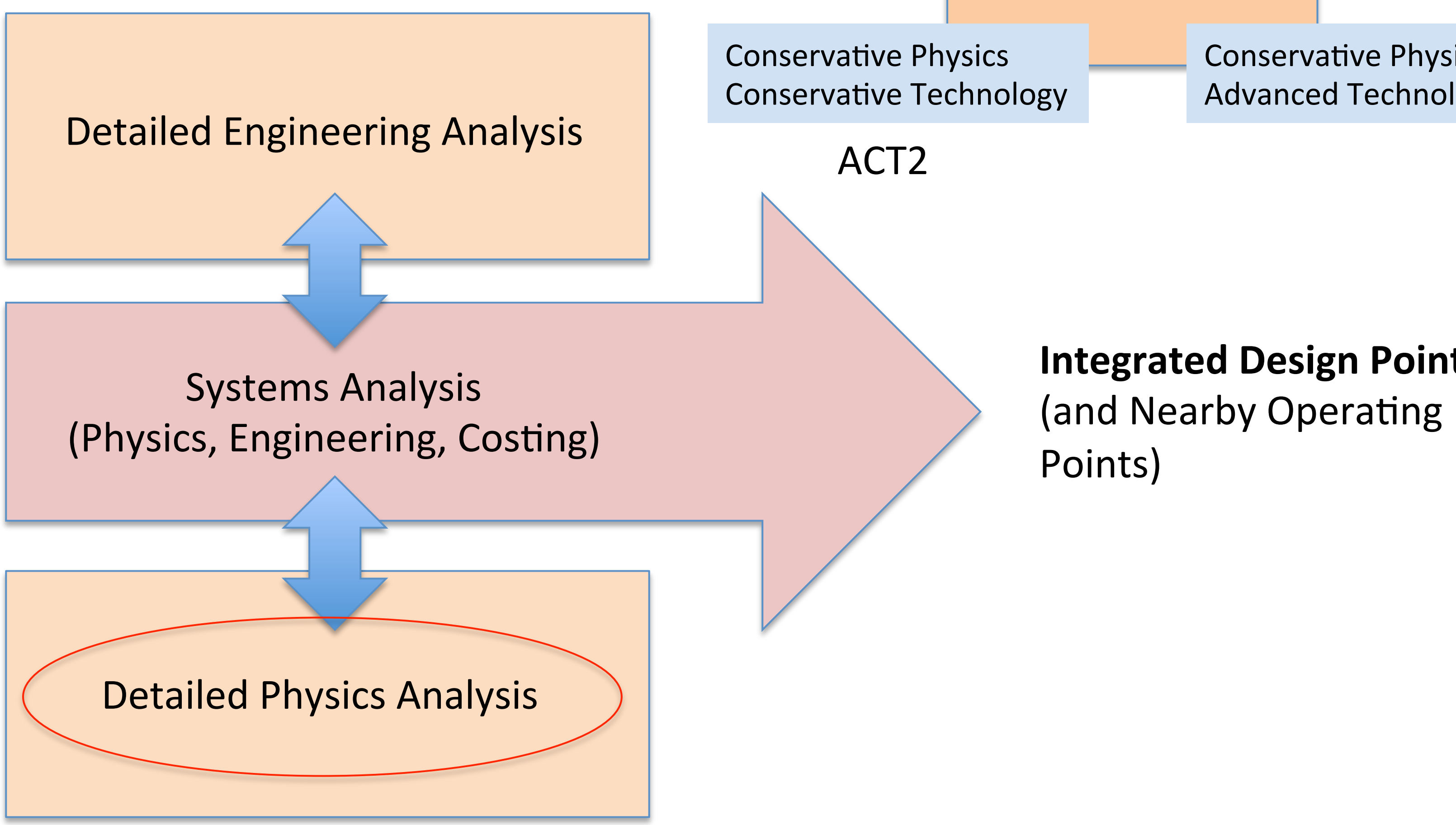




Physics Analysis is Used to Define Attractive Plasma Operating Points

Systems Analysis originally identifies a plasma configuration that is consistent with physics philosophy, engineering constraints, power production, and cost

The operating point is developed with (2D equilibrium/1D transport) free-boundary time-dependent transport simulations including heating and current drive

- Tokamak Simulation Code (TSC)
- TRANSP

Ideal MHD stability is assessed for low-n external kink modes, high-n ballooning modes, vertical stability, and peeling-ballooning modes (for pedestal)

Fast particle stability (alpha particles)

Poloidal field coil design

New physics treatments/methods denoted in blue

Scrape-off Layer and Divertor Plasma Simulations

Heat Flux Descriptions for First Wall and Divertor

Other physics issues: Greenwald density, disruption data, tritium burnup, etc.

Systems Physics and Detailed Physics

Systems physics analysis solves 0D power and particle balance in steady state, along with several physics relationships

1.5D (2D/1D) analysis allows us to compare our 0D configuration with the more detailed one

0D uses:

$$n(\rho) = n(0) \left[(1-f_n)(1-\rho^2)^{\alpha n} + f_n \right]$$

$$T(\rho) = T(0) \left[(1-f_T)(1-\rho^2)^{\alpha T} + f_T \right]$$

Global integrated quantities like radiation

Artificial flux surfaces for fusion reactivity and line radiation

Simple correlations, for example, for bootstrap current based on equilibrium analysis

ACT1	Sys Op Point	broad p TSC $B_T = 6.0$ T
Ip, MA	10.9	11.1
I_{BS}, MA	9.89	9.75
I_{LH}, MA	1.04	1.12
I_{IC}, MA		0.125
q_{min}, q(0)		2.83, 3.60
li	0.5 (input)	0.47
n/n_{Gr}	1.0	1.0
W_{th}, MJ	690	673
n(0), $/m^3$ $x10^{20}$	1.65	1.67
$\langle n \rangle_v$, $/m^3$ $x10^{20}$	1.3	1.33
n(0)/<n>	1.27	1.27
β_N^{th}, β_N^{total}	**4.75, 5.75**	**4.9, 5.79**
τ_E, s	2.26	1.94
$H_{98(y,2)}$	1.65	1.50
$T_{e,i}(0)$, keV	40.4	40, 35.6
Te,i(0)/<T>	2.15	2.09,2.05
P_{alpha}, MW	363	389
P_{LH}, MW	39	40
P_{IC}, MW	3.0	15
P_{cycl}, MW	35.0	23
P_{line}, MW	24.2	32.7
P_{brem}, MW	56.3	48.4
$P_{L-H,thr}$, MW	109	119
$P_{net}/P_{L-H,thr}$	2.66	2.86
Z_{eff}	2.11	2.0
n_{He}/n_e	0.097	0.076
n_{DT}/n_e	0.752	0.802
n_{Ar}/n_e	0.003	0.003

Time-Dependent Simulations are Done for the First Time in 4-Corners Activity

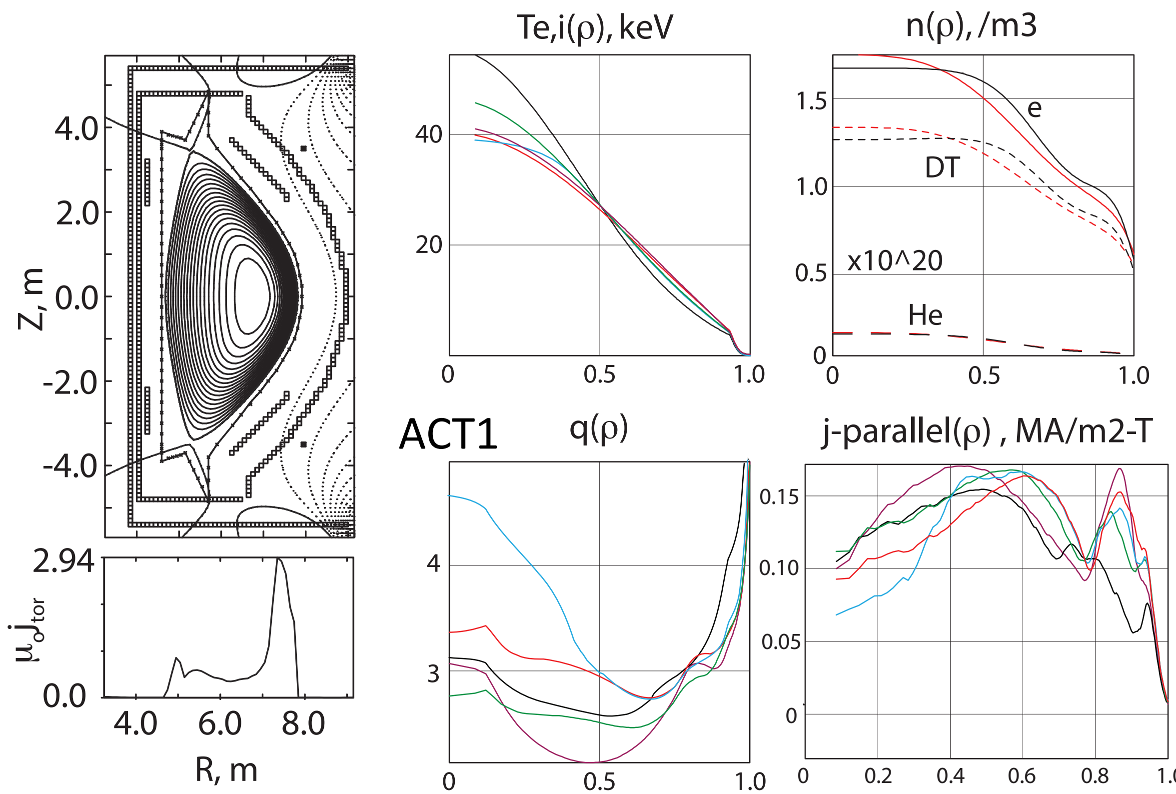
Tokamak Simulation Code (TSC) free-boundary simulation is used to show plasma growth, volt-second consumption, current profile evolution, temperature profile evolution, heating and current drive, radiated powers, etc….coupled to TRANSP

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

Density profiles are prescribed

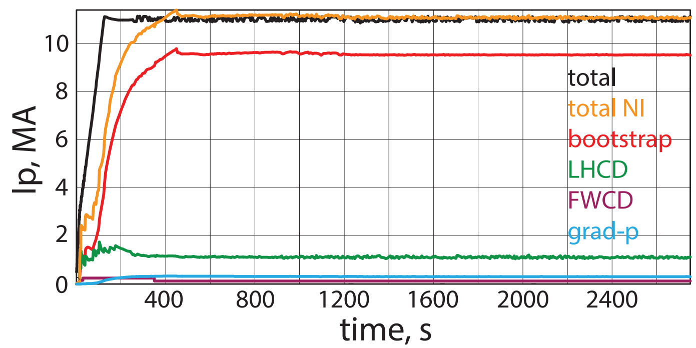
Energy transport model is L-mode with enforced pedestal height from EPED1…scaled to provide target stored energy (β_N)

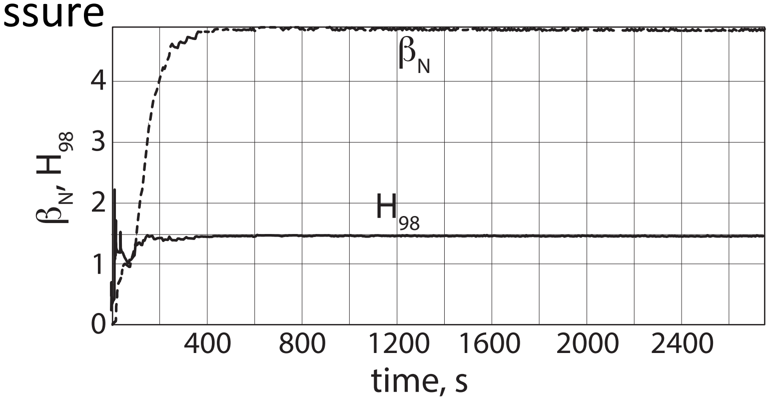
5 cases examined with different density and temperature profiles

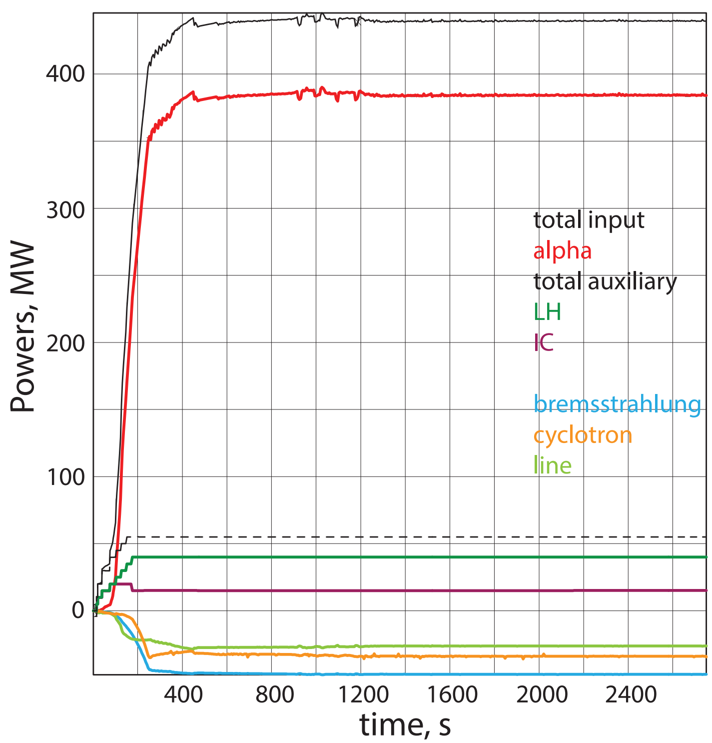


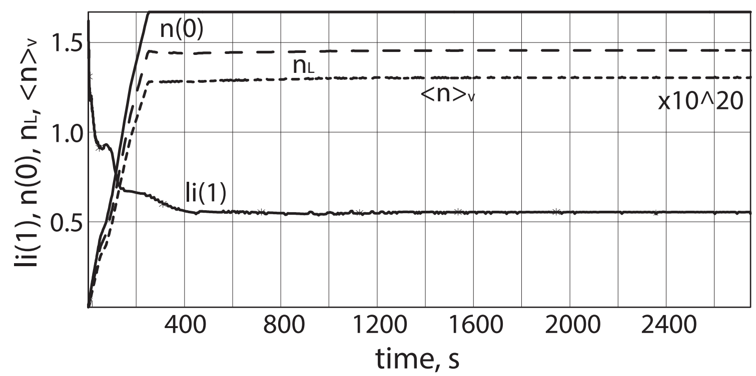
Relaxation to Steady State of an Advanced Tokamak (ACT1)

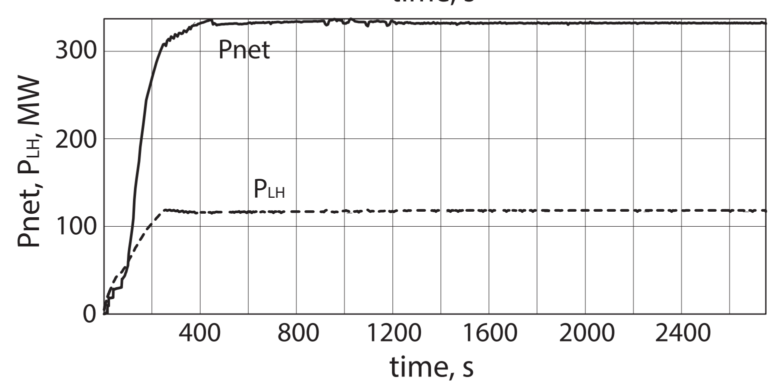
Broad pressure











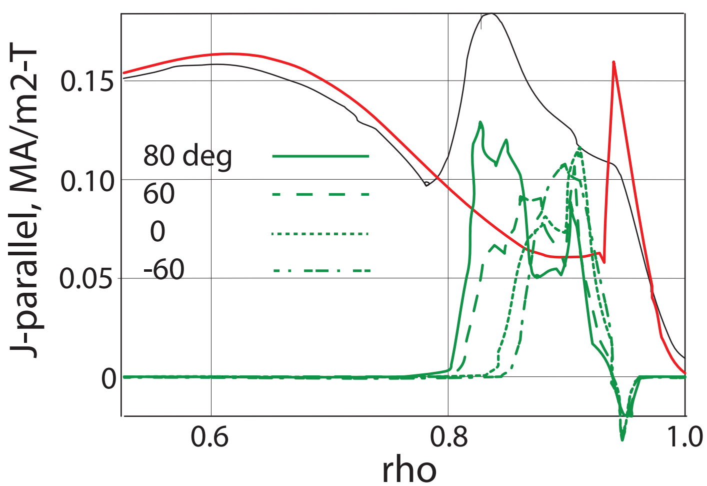
Data Interface with TRANSP Allows Us to Use High Fidelity Heating/Current Drive Models

The entire simulation from TSC can be reproduced in TRANSP, but with different heating and current drive models and fast particle treatments

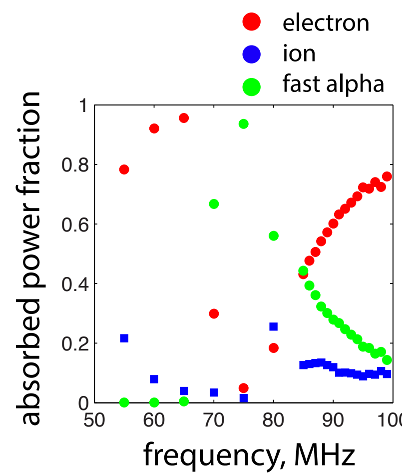
Ion cyclotron (ICRF) was analyzed with TORIC full wave model showing high CD efficiency → 0.045 A/W

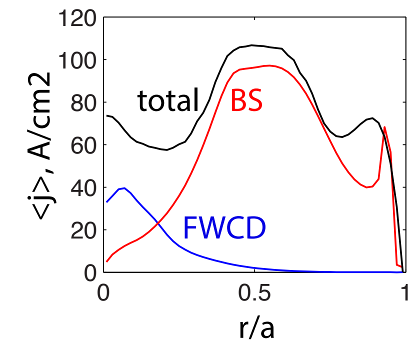
Lower hybrid was analyzed with LSC (corrected by 2D Fokker Planck analysis) → 0.028 A/W

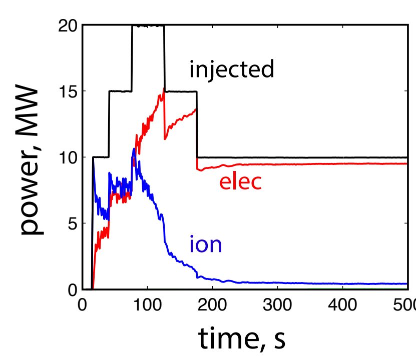
LH

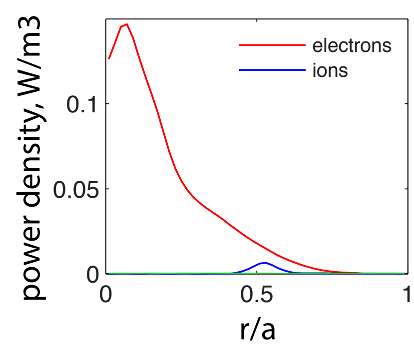


IC

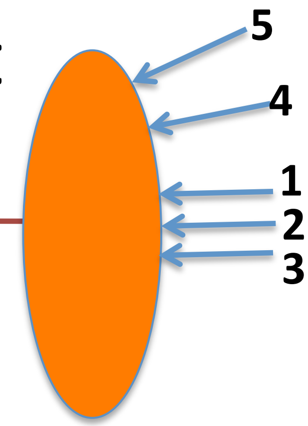






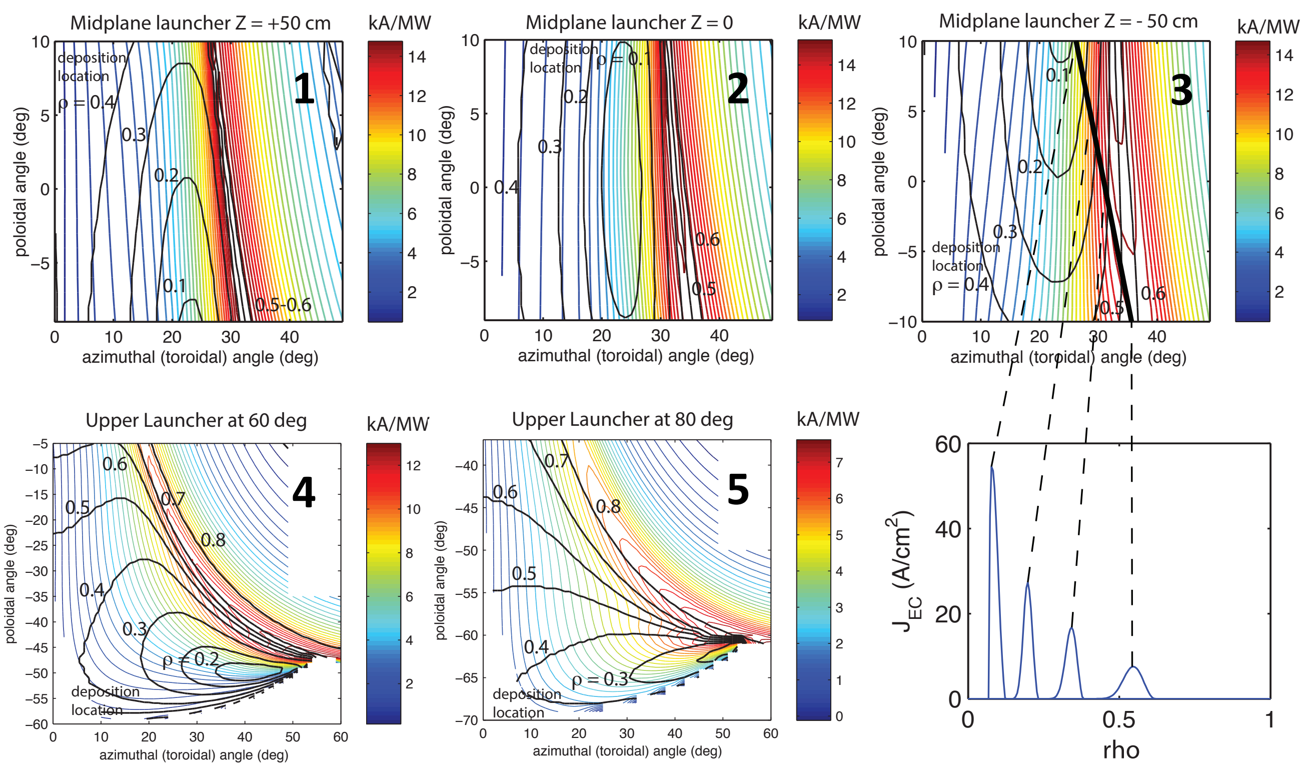


First Assessment of EC in the Power Plant Regime



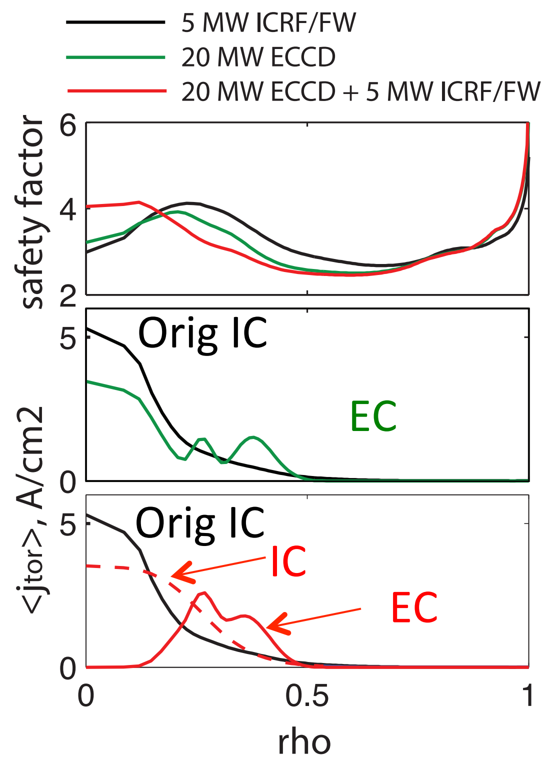
Electron cyclotron (EC) is analyzed with TORAY-GA in time-dependent simulations and GENRAY for time-slice analysis → 0.012 A/W

Scan of EC steering angle



EC to replace IC

EC to add to IC



Advantages of EC are deposition location, and does not require coupling to the plasma, however it does reduce the fusion gain

We are Continuing and Expanding the Ideal MHD assessments

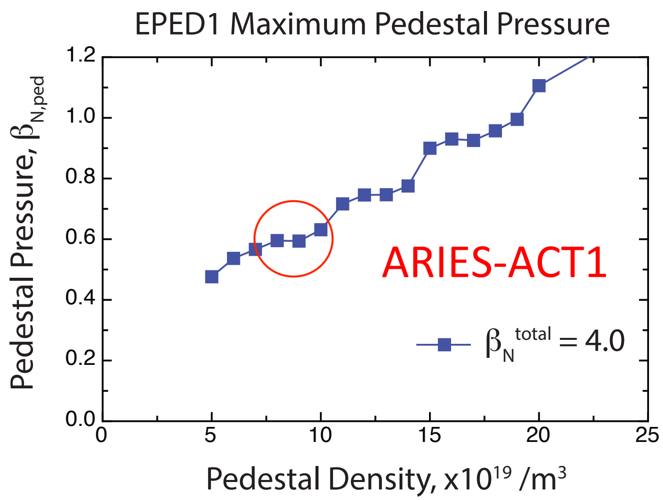
Low-n external kink stability (PEST1), and incorporating both a W stabilizing shell in the blanket, and feedback coils on each sector behind shield to assist in stabilization

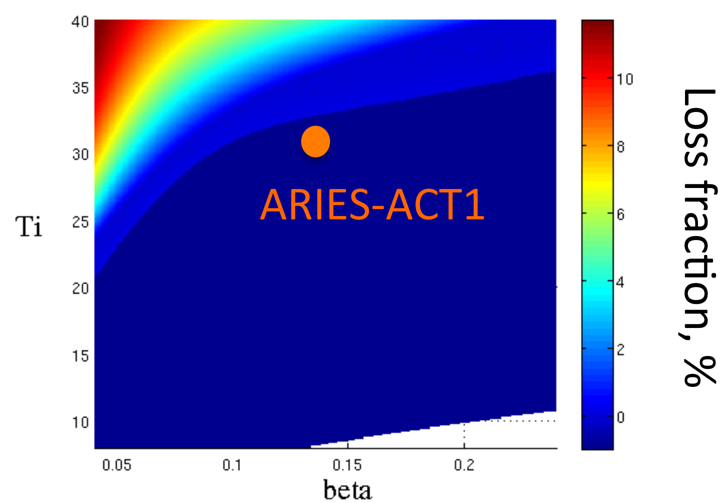
High-n ballooning (BALMSC) to constrain pressure profiles to those that are stable

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

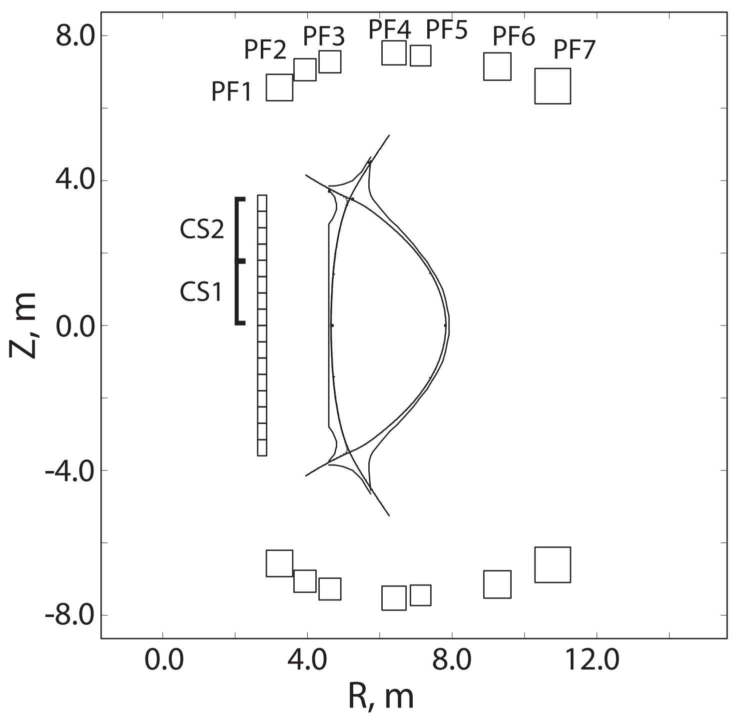




Poloidal Field Coil Layout

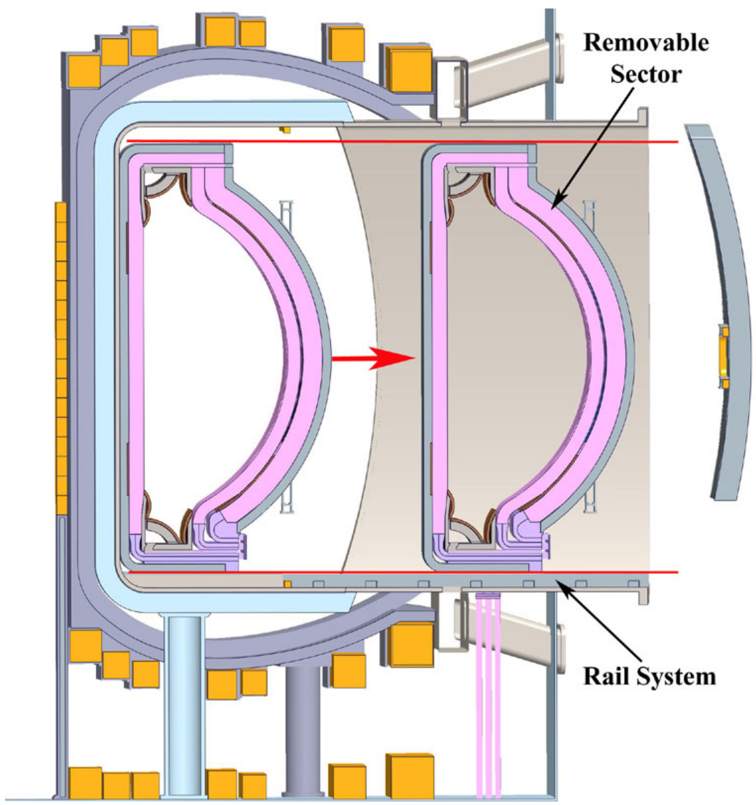
Poloidal field coils drive plasma current in the rampup and they provide the equilibrium force balance

This analysis is done with an equilibrium code

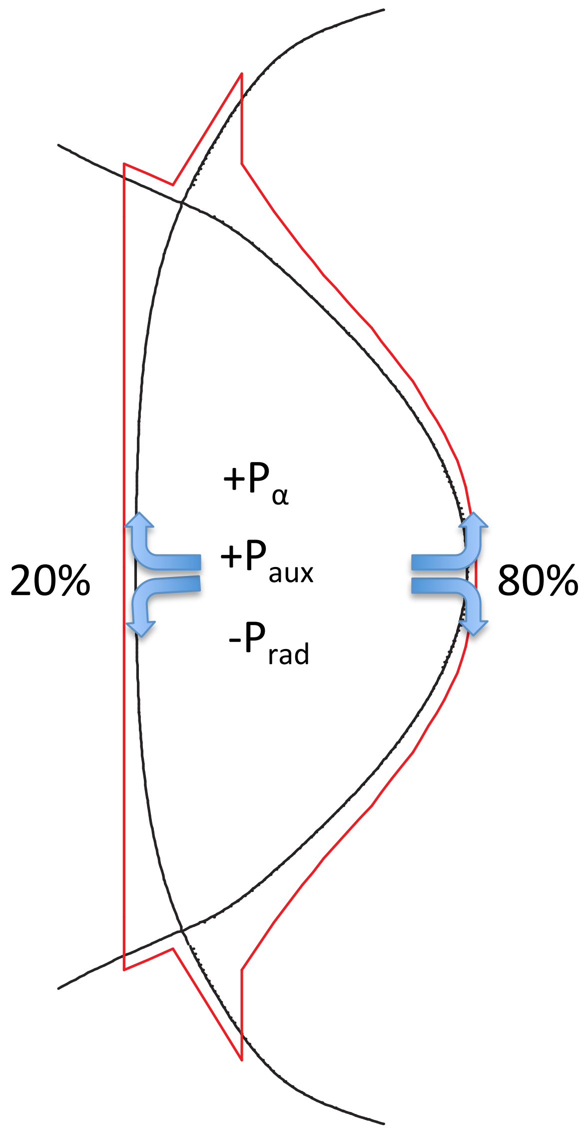


Major contraints include

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



Steady State Divertor Heat Loading



$P_{SOL} = P_{\alpha} + P_{aux} - P_{rad}$

The SOL power flows to the divertor within a very narrow layer called the power scrape-off width

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

~ 4 mm for ARIES-ACT1 at the OB midplane

The width expands with the magnetic flux as it travels to the divertor

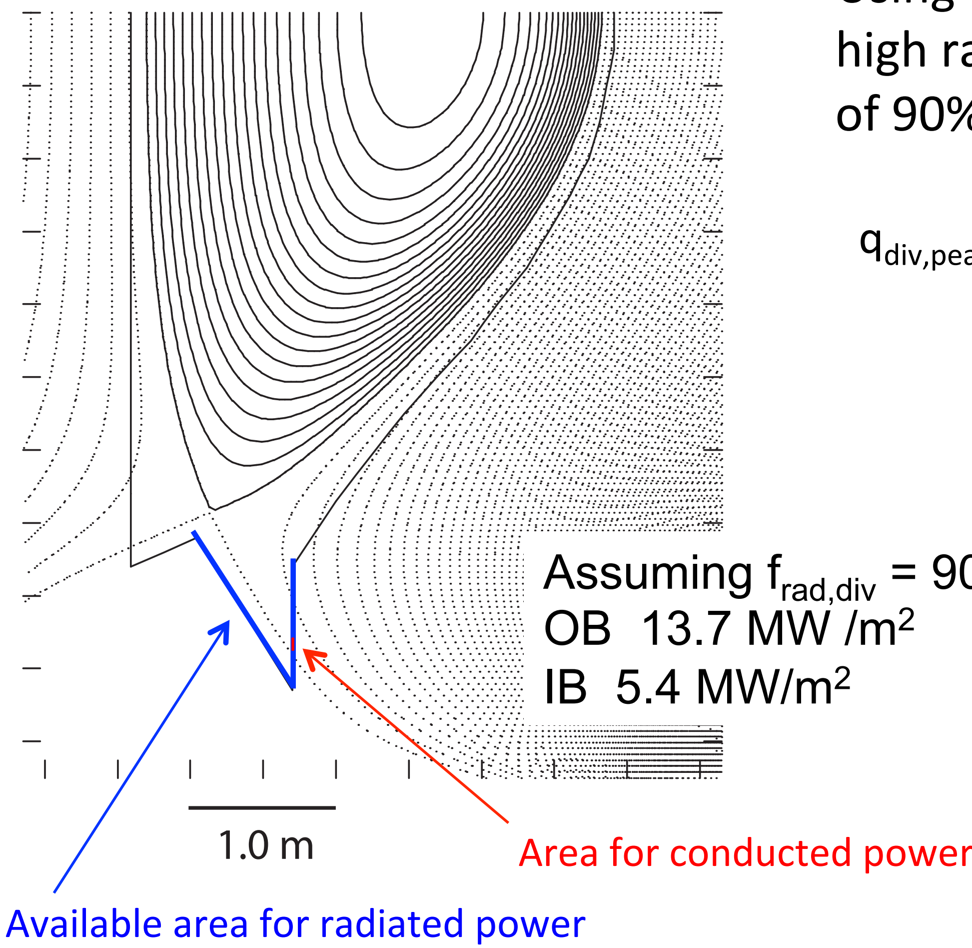
The final area which the power impinges on is ~ 1.38 m^2 OB and 1.17 m^2 IB

Steady State Divertor Heat Loading, cont'd

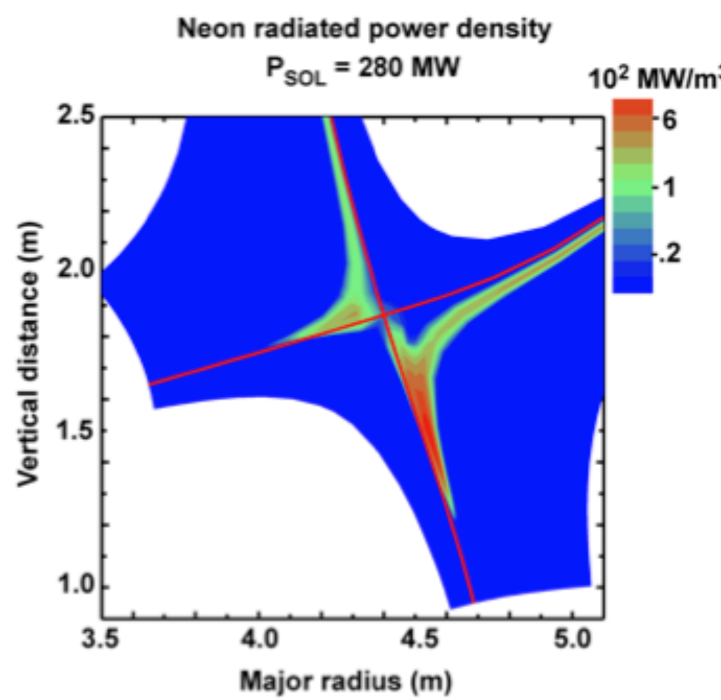
Using detached divertor solution to reach high radiated powers in the divertor slot of 90%

$$q_{div,peak}\ (MW/m^2) = P_{SOL}\ f_{IB/OB}\ f_{vert}\ \times$$

$$[\ (1-f_{div,rad})/A_{div,cond} + f_{div,rad}/A_{div,rad}\ ]$$



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



UEDGE analysis, LLNL

New Direction to Include Routine Analysis of Scrape-Off Layer and Divertor Plasmas

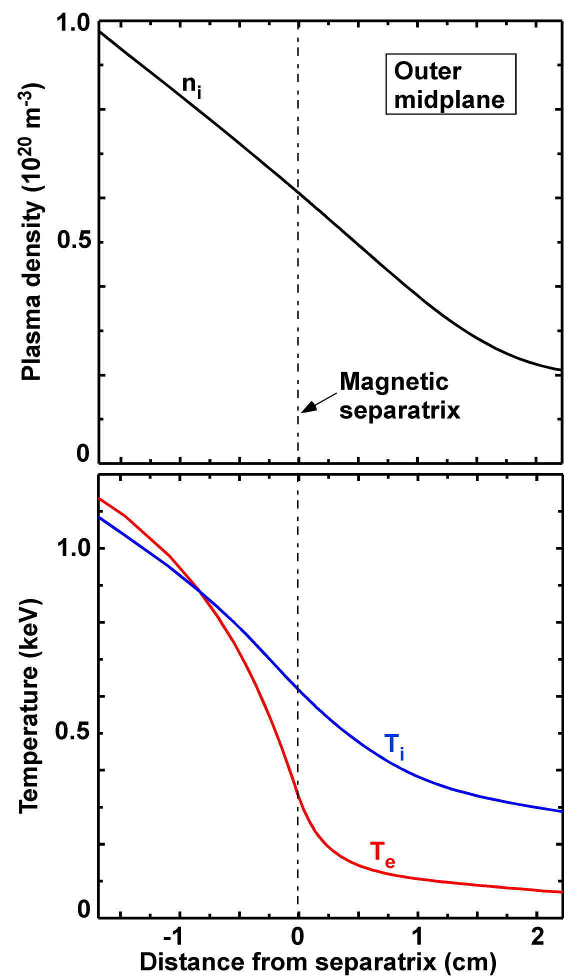
2D analysis of plasma beyond separatrix with UEDGE and fluid or Monte Carlo neutrals

Power plants require strong radiation in the divertor to dissipate the high powers flowing there

Solutions are sought with high density and impurities to provide this distributed energy loss ….detached plasmas that are stable

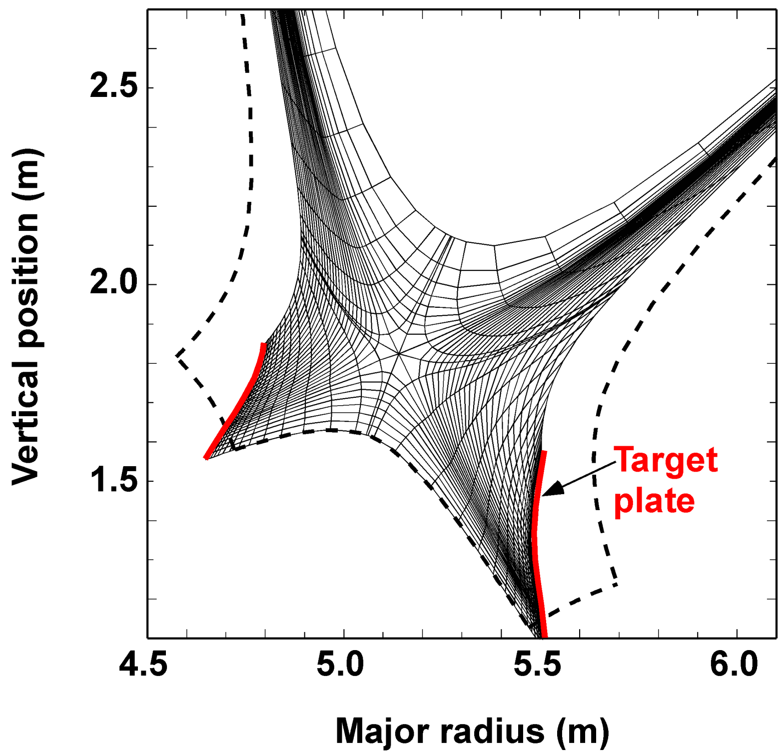
Simulation model and radial transport assumptions similar to that used for ITER divertor

Fluid neutral model used to efficiently survey geometric options; limited Monte Carlo work

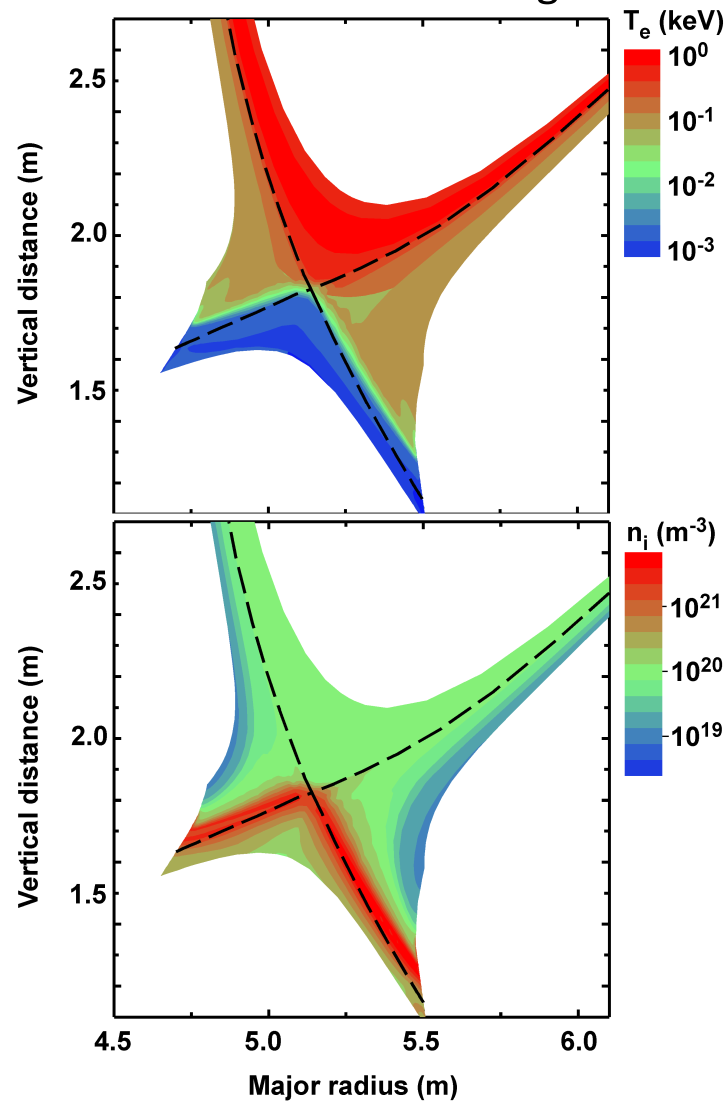


Divertor Configurations Defined by Location of Target Plates and Side Walls

Two options for divertor geometry: ITER-style tilted target plates for partial detachment & wide slot (dashed line) for full detachment

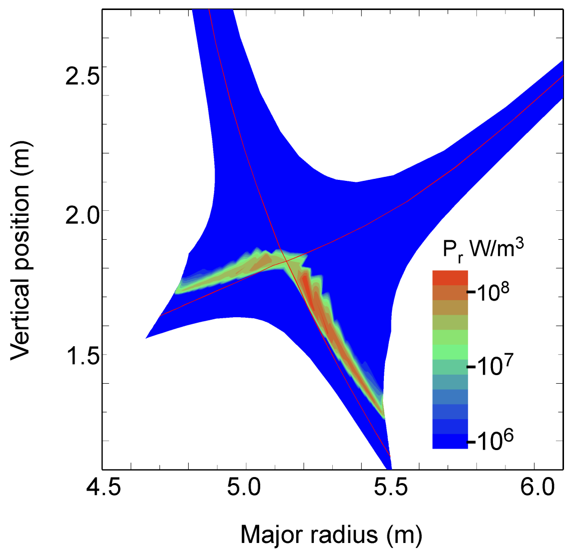


For ITER-like tilted target

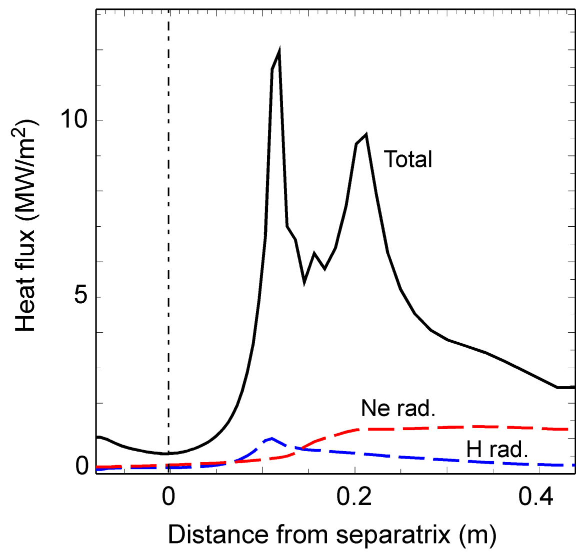


Heat Flux to Target Plates and Side Walls Are Close to 10 MW/m² or Less

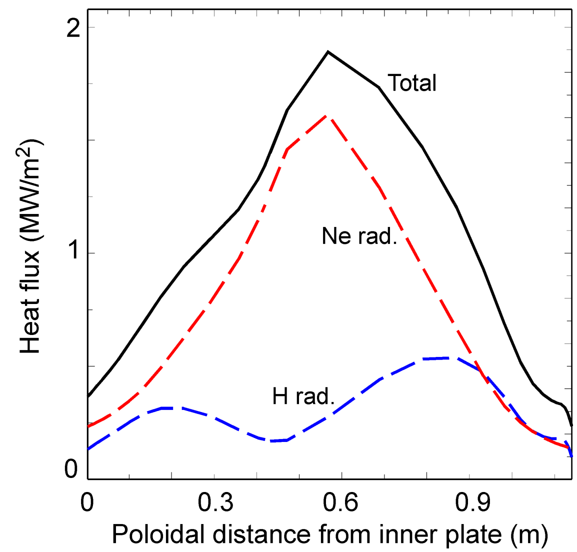
Radiation from seeded neon (or Ar) concentrated in divertor legs



Heat flux to outer target plate is ~12 MW/m², mostly from plasma



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



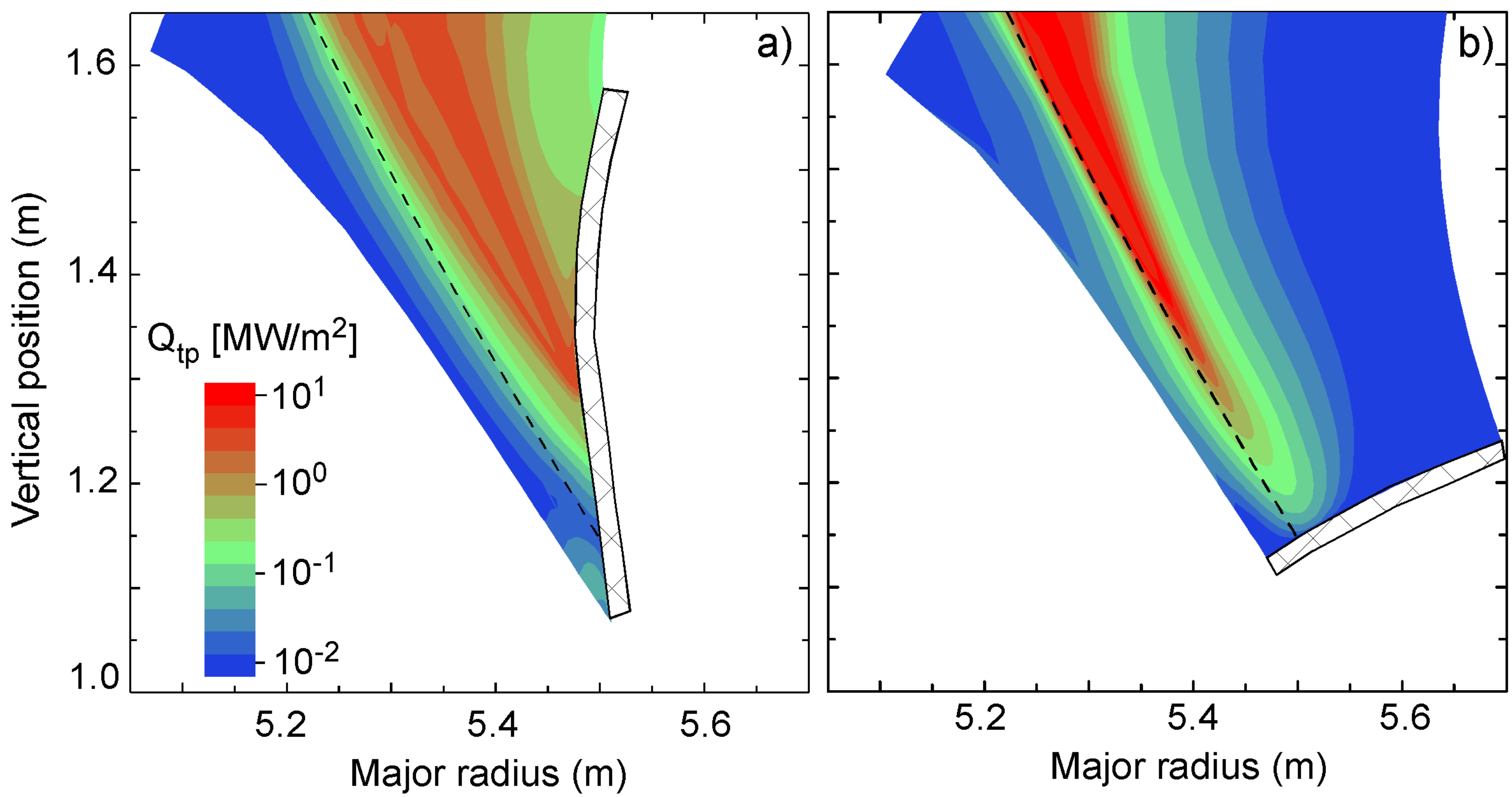
For alternate wide "slot" divertor: fully detached plasmas with radiation-dominated heat flux ~2 MW/m² on target plates and walls; stable operating window can be expanded via impurity feedback on ~1 sec timescale

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

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

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 $f_{div,rad}$ > 0.95

Provide Physics Based Data to Engineers for ELMs and Disruptions

Utilizing information derived for ITER, primarily on JET and ASDEX-U

ΔT_{rise} (^oK or ^oC) = $2/3\ (2\ \alpha^{1/2}\ \Delta W_{ELM}^{div,rise})\ /\ [\pi^{1/2}\ k\ A_{div,ELM}\ (2\ \tau_{||})^{1/2}]$,

$= 2/3\ C_{material}\ \Delta W_{ELM}^{div,rise}\ /\ A_{div,ELM}\ (2\ \tau_{||})^{1/2}$

Total ELM energy release from plasma 23.4 MJ, **100% to outboard**

- 50% to divertor
  - 65% to each divertor (ΔW_{ELM}^{div})
    - 40% rise phase, 0.44 ms ($\Delta W_{ELM}^{div,rise}$)
    - 100% over full waveform, 1.3 ms
- 50% to FW
  - 40% rise phase, 0.44 ms
  - 100% over full waveform, 1.3 ms

$A_{ELM,\ OB\ divertor}$ = 1.38 m^2 (possible expansion by 4-6x)
$A_{ELM,\ OB\ FW}$ = 396 m^2/4 = 99 m^2 (peaking)
f_{ELM} ~ 3.2 Hz (/s)

Produce ~ 10^8 cycles in a year

Physics-Engineering Interfaces are Common in the Power Plant Studies

- Vertical and low-n kink stabilizing shells in the blanket affect the tritium breeding
  - Tungsten shells, cm's thick, to slow instabilities so feedback can control them
- Wall-plug efficiencies assumed for heating and current drive systems affect the overall operating point choice (R, Ip, B_T…) through recirculating power
  - All systems wall-plug efficiencies recently reduced to 0.4 to account for source, transmission and coupling losses
- High heat loading on the divertor due to small power scrape-off width
  - Large uncertainty, but formulation used gives ~ 4 mm, requires larger R
  - Estimates for ACT1 range from 0.7 mm to 80-200 mm
- Heat loading derived from experiments for ELMs (and disruptions) affect the design of the divertor, and analysis provides input to constrain the allowed ELM size
- PF coils are strongly constrained in their locations, requiring large currents in the outer equilibrium field coils

Plasma Heating and Current Drive Systems Hardware Integration

Based on ITER power densities and launcher designs

Lower Hybrid 40 MW: 20 MW/m^2, Passive-Active Multi-junction Launcher

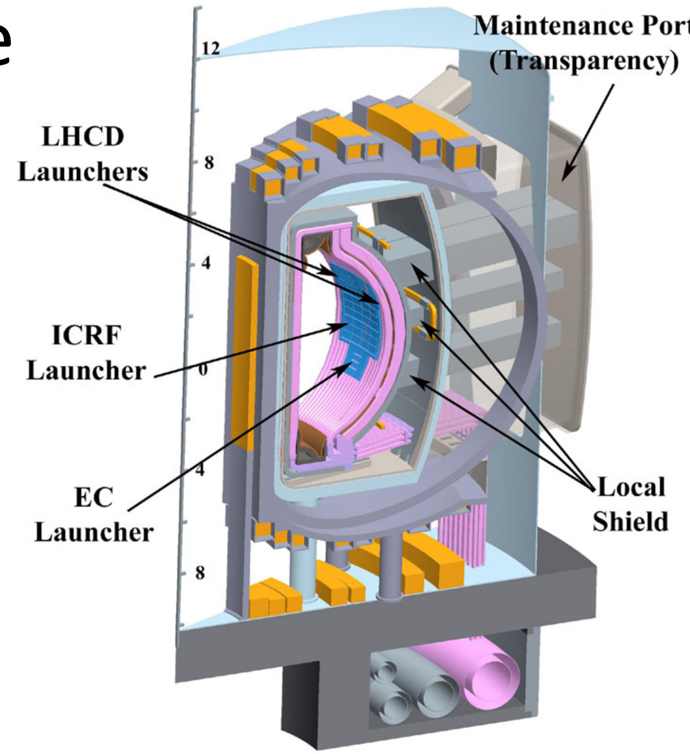
Ion Cyclotron 20 MW: 10 MW/m^2, 4 Strap Antenna

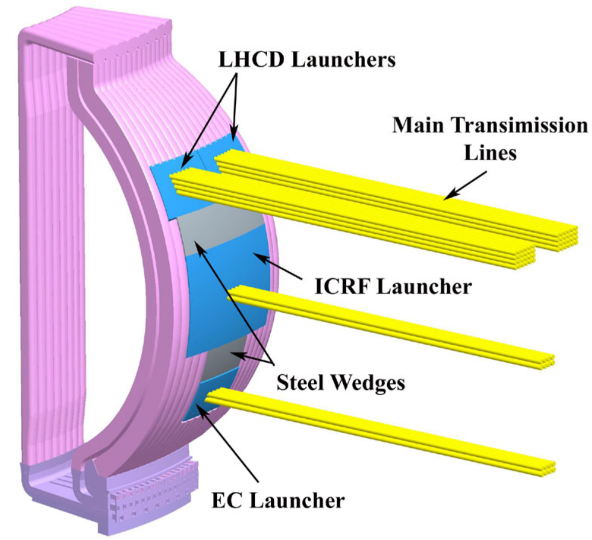
Electron Cyclotron 20 MW: 20-40 MW/m^2, 3 Slot

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
- Going to larger devices aggravates this, causing operating points to exceed it even more
- Operating at higher densities reduces 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

Divertor Solutions in the Power Plant Regime

Studies showed that the ITER tilted plate divertor solution inhibits detached divertor operation, while a long and wide slot-like geometry with orthogonal plate enhanced detached operation

Feedback solutions involving puffing gas/impurities and pumping neutrals was capable of stabilizing detached regimes…still studying this

Divertor slot length is a trade-off between engineering the divertor into the overall configuration, and achieving the lowest possible heat fluxes

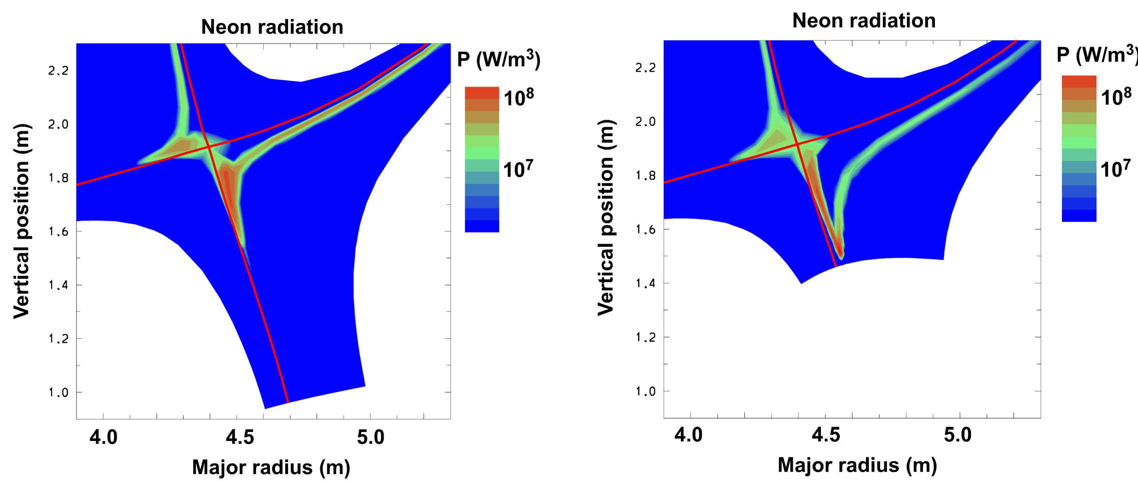
<u>Modeling studies are continuing</u>:

Comparing ACT and ITER

Monte Carlo vs fluid neutrals

Multi-charge state impurities vs fixed fraction coronal equilibrium

q_{div}^{peak} = 1.5 MW/m^2 q_{div}^{peak} = 9.5 MW/m^2



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

Future Work on 4-Corners Activity

Presently developing a conservative physics and conservative technology configuration, ACT2

P_{elec} = 1000 MW, DCLL blanket

Parameter limits:

- $\beta_N \leq 2.6$
- $q_{div}^{peak} \leq 10$ MW/m^2
- $H_{98} \leq 1.3$
- $n/n_{Gr} \leq 1.3$

Physics issues:

- H/CD sources (NB, IC, EC, LH)
- MHD stability w/o wall
- MHD stability with far-away wall

	ACT1	ACT2
R, m	6.25	9.75
a, m	1.56	2.44
κ_x	2.2	2.2
δ_x	0.63	0.63
Ip, MA	10.95	13.98
B_T, T (B_{Tcoil})	6.0 (10.6)	8.75 (14.4)
β_N^{th}, β_N^{fast}	4.75, 0.85	2.25, 0.35
β^{th}, β_p^{th}	5.54, 2.76	1.48, 2.32
q_{95}	4.5	8.0
n/n_{Gr}	1.0	1.3
H_{98}	1.65	1.22
$\langle T_{e,i} \rangle$, keV	20.6	17.8
$\langle n \rangle$, /m^3 x 10^{20}	1.3	0.86
$T(0)/\langle T \rangle$	2.15	2.15
$n(0)/\langle n \rangle$	1.27	1.41
$\langle N_w \rangle$, MW/m^2 (at plasma)	2.45	1.46
Z_{eff}	2.11	2.12
W_{th}, MJ	691	1486
V_{plasma}, m^3	582	2209
f_{BS}	0.91	0.77
P_{brem}, MW	56.3	96.5
P_{cycl}, MW	35.0	150.4
P_{line}, MW	24.2	42.9
$P_{aux(CD)}$, MW	42.7	105.5
P_{alpha}, MW	363	528
P_{fusion}, MW	1813	2637

Physics Analysis is Improving and Expanding on ARIES

ARIES-ACT 4 Corners study is the newest tokamak power plant examination, >10 years after ARIES-AT

The ACT study is examining advanced and conservative physics and technology configurations

The physics activities are utilizing higher fidelity models, more self-consistency with time-dependent transport evolution, and including more sophisticated physics assessments

The physics activities are attempting to bring the power plant physics regime to light more clearly, by discussing these features to encourage research activities

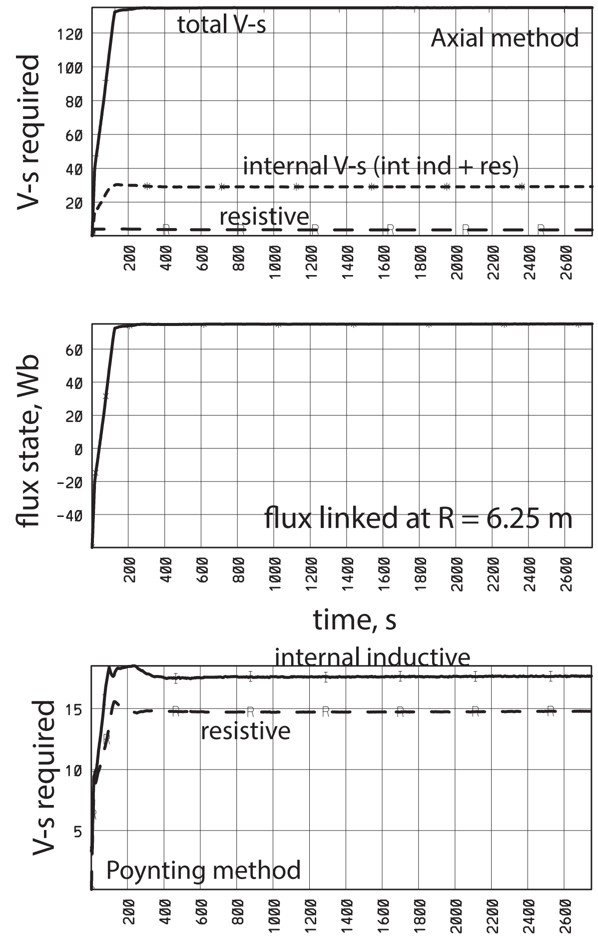
C. E. Kessel, M. S. Tillack, J. P. Blanchard, "Evaluation of the Heat Loading from Steady, Transient, and Off-Normal Conditions in ARIES Power Plants", Fus Sci Tech, 2013.

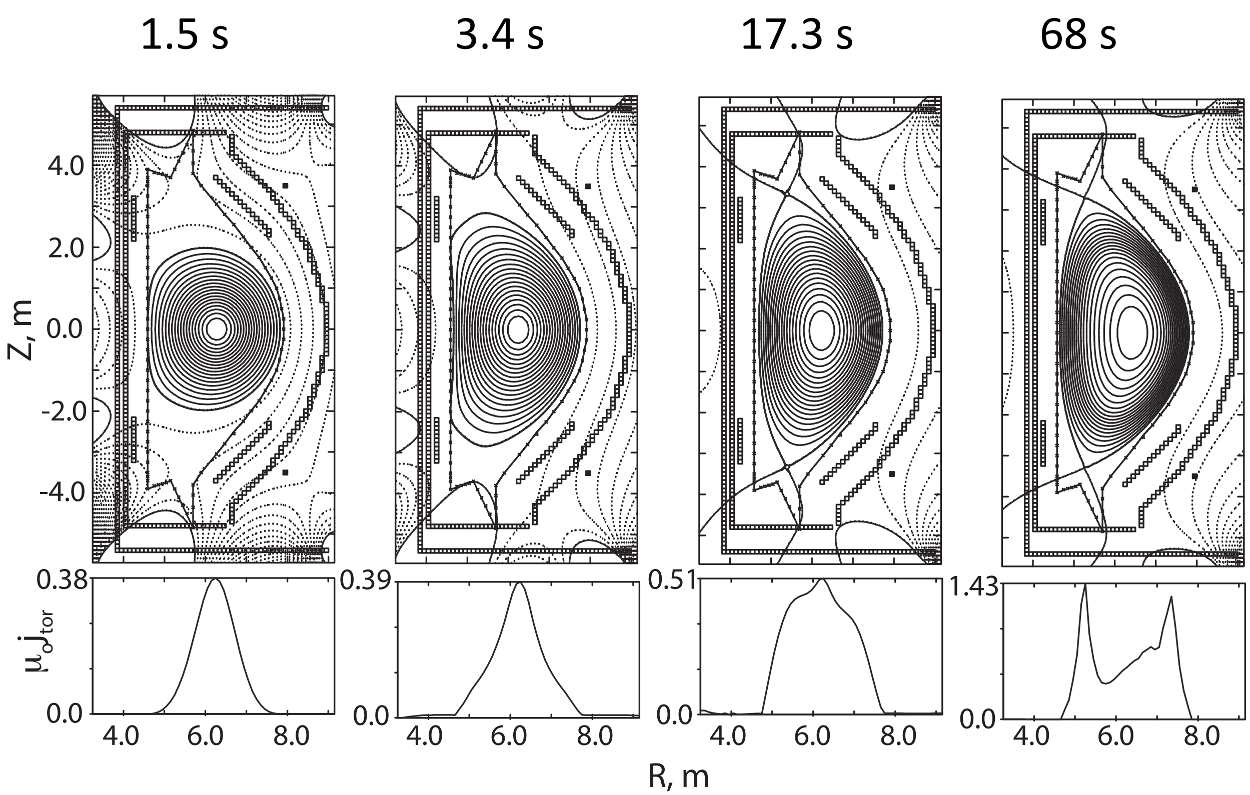
C. E. Kessel, F. M. Poli, K. Ghantous, N. N. Gorelenkov, M. E. Rensink, T. D. Rognlien, P. B. Snyder, H. St. John, A. D. Turnbull, "The Physics Basis for an Advanced Physics and Advanced Technology Tokamak Power Plant Configuration, ARIES-ACT1", to be submitted 2013.

T. D. Roglien and M. E. Rensink, "Edge Plasma and Neutral Modeling for the ARIES-ACT1 Power Plant", to be submitted 2013.

Backup Slides

ACT1 plasma growth evolution and V-s requirement





ACT1 cases

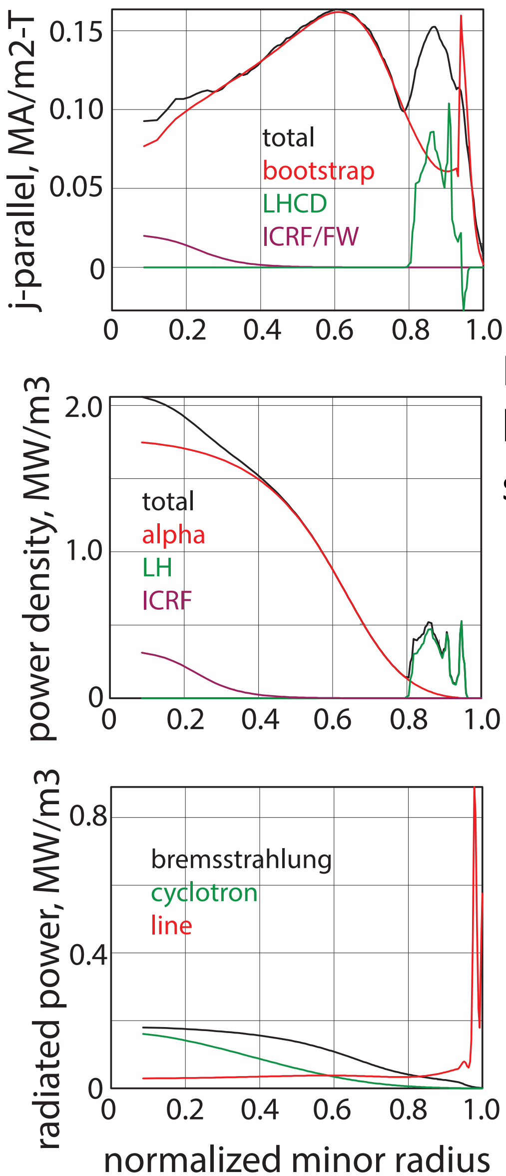
Systems ref case

5 separate 1.5D TSC simulation cases

ACT1 ref case is the broad p at 6.0 T

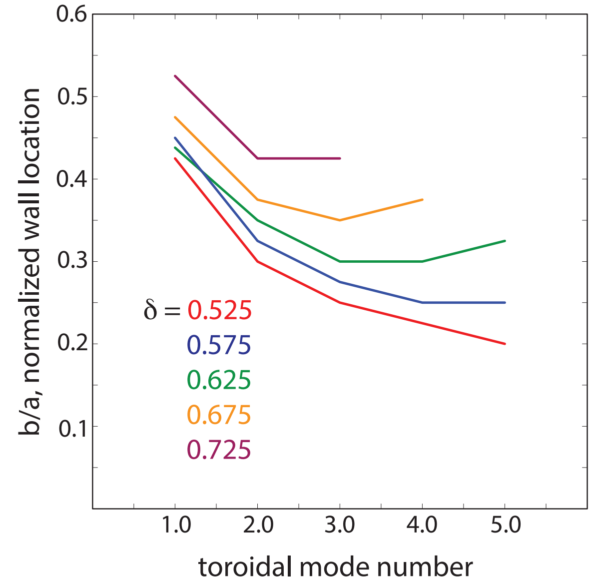
	Sys Op Point	peak p B_T = 7.0 T	med p B_T = 6.75 T	broad p B_T = 6.0 T	broad p 2 B_T = 6.0 T	dens, broad p B_T = 6.0 T
Ip, MA	10.9	11.0	11.1	11.1	11.1	11.0
I_{BS}, MA	9.89	9.49	9.57	9.75	9.20	9.64
I_{LH}, MA	1.04	1.21	1.14	1.12	1.08	1.21
I_{IC}, MA		0.125	0.125	0.125	0.35	0.125
q_{min}, q(0)		2.56, 3.05	2.73, 3.09	2.83, 3.60	2.63, 3.80	2.14, 3.05
li	0.5 (input)	0.60	0.56	0.47	0.51	0.57
n/n_{Gr}	1.0	1.0	1.0	1.0	1.0	1.0
W_{th}, MJ	690	700	687	673	669	638
n(0), /m^3 x10^{20}	1.65	1.67	1.67	1.67	1.67	1.75
$\langle n\rangle_v$, /m^3 x10^{20}	1.3	1.28	1.30	1.33	1.33	1.25
$n(0)/\langle n\rangle$	1.27	1.30	1.29	1.27	1.27	1.40
β_N^{th}, β_N^{total}	**4.75, 5.75**	**4.4, 5.15**	**4.45, 5.28**	**4.9, 5.79**	**4.8, 5.67**	**4.6, 5.49**
τ_E, s	2.26	2.25	2.05	1.94	1.95	1.98
$H_{98(y,2)}$	1.65	1.56	1.50	1.50	1.50	1.45
$T_{e,i}(0)$, keV	40.4	54.7, 50.0	46, 41	40, 35.6	38.5,34.4	40.0, 35.6
Te,i(0)/<T>	2.15	2.79, 2.79	2.42, 2.33	2.09,2.05	2.05, 2.0	2.18, 2.13
P_{alpha}, MW	363	382	385	389	389	357
P_{LH}, MW	39	40	40	40	40	40
P_{IC}, MW	3.0	15	15	15	15	15
P_{cycl}, MW	35.0	46	27	23	23	22.0
P_{line}, MW	24.2	36	35	32.7	34.6	29.6
P_{brem}, MW	56.3	46	48	48.4	48.3	44.9
$P_{L-H,thr}$, MW	109	119	119	119	119	119
$P_{net}/P_{L-H,thr}$		3.06	2.8	2.86	2.8	2.70
Z_{eff}	2.11	2.0	2.0	2.0	2.0	2.0
n_{He}/n_e	0.097	0.070	0.077	0.076	0.066	0.075
n_{DT}/n_e	0.752	0.79	0.79	0.802	0.82	0.80
n_{Ar}/n_e	0.003	0.003	0.003	0.003	0.003	0.003
High-n stability		U	S	S	S	S
Low-n b/a			0.375	0.30	0.30	0.275

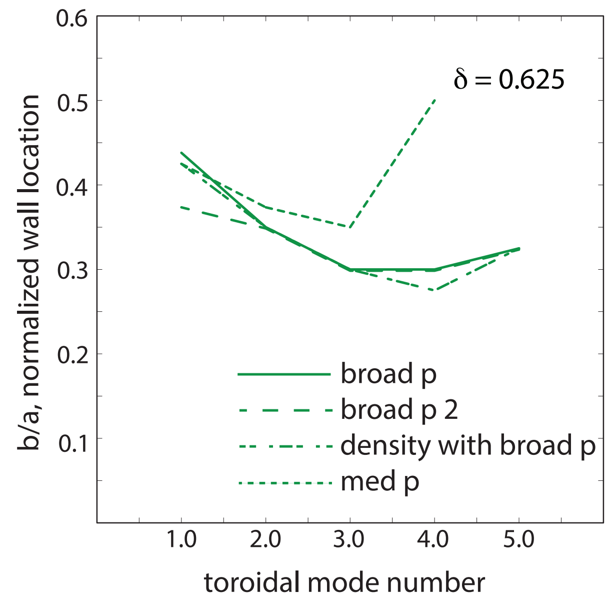
ACT1 Physics Results



Plasma profiles for broad pressure 1.5 D simulation of ACT1

Ideal MHD low-n stability



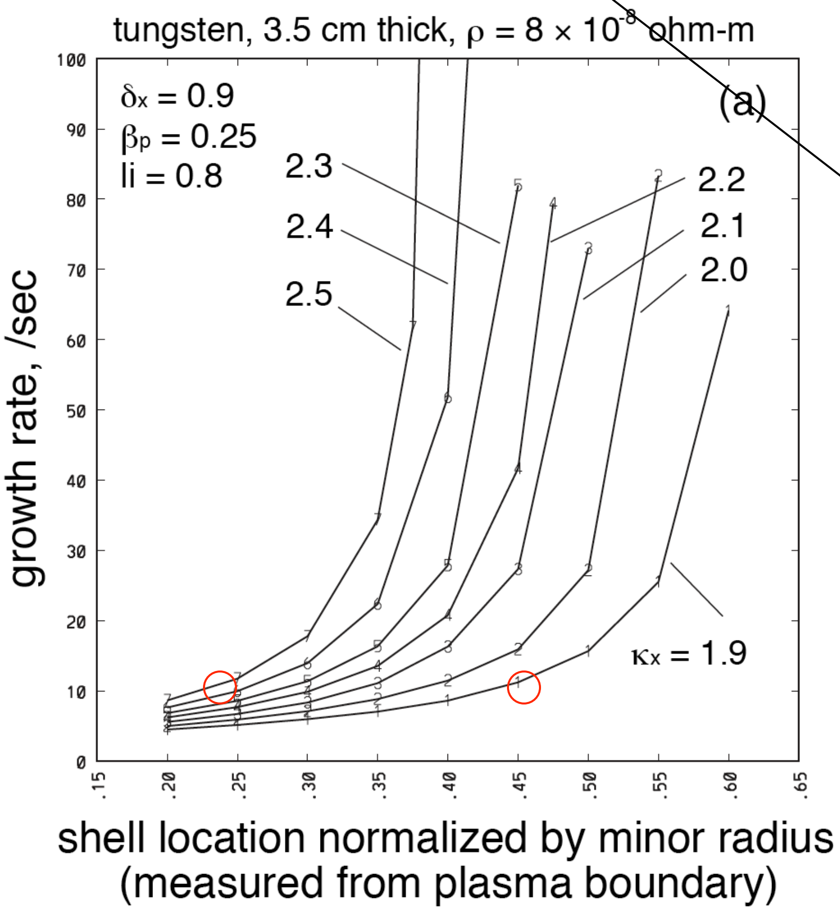


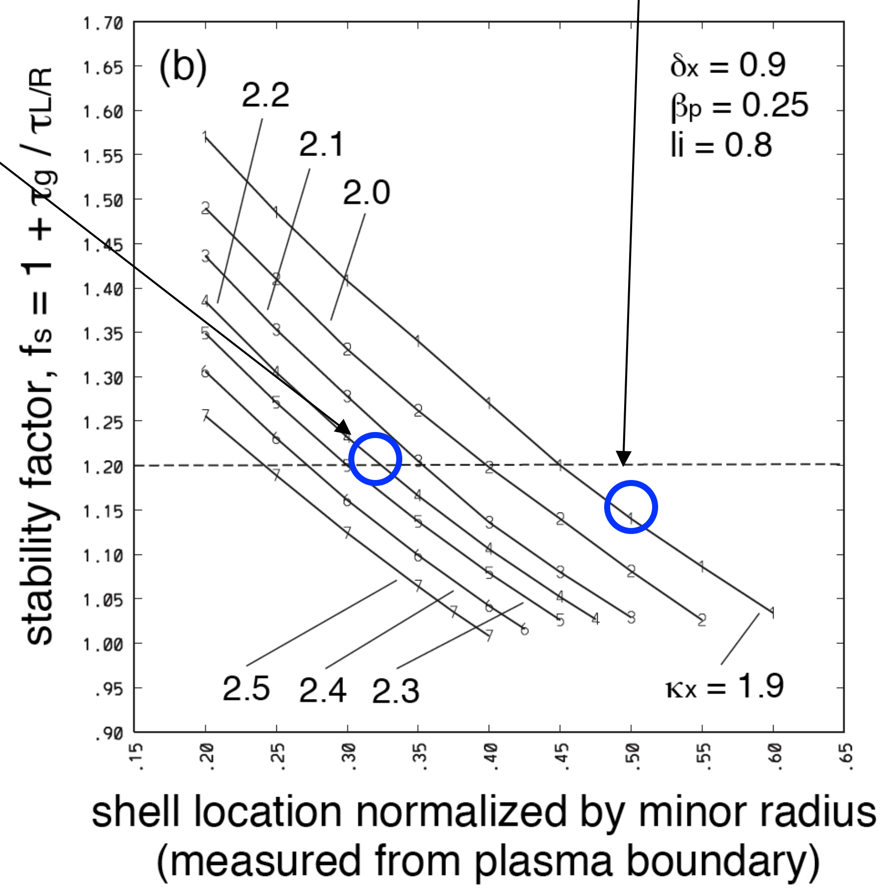
Generic Vertical Stability Study from ARIES-AT

$$\frac{\mu_o \Delta b}{\eta} \approx 0.25 s$$

ARIES-AT had κ = 2.2, b/a = 0.33

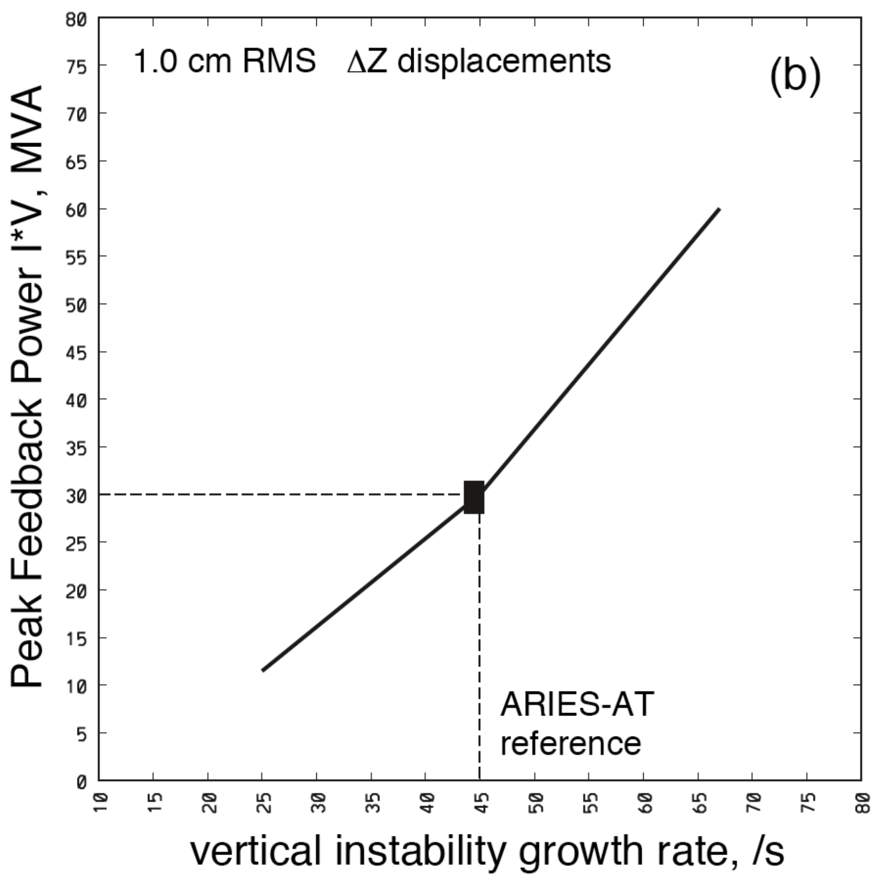
ARIES-RS has κ = 1.9, b/a = 0.5



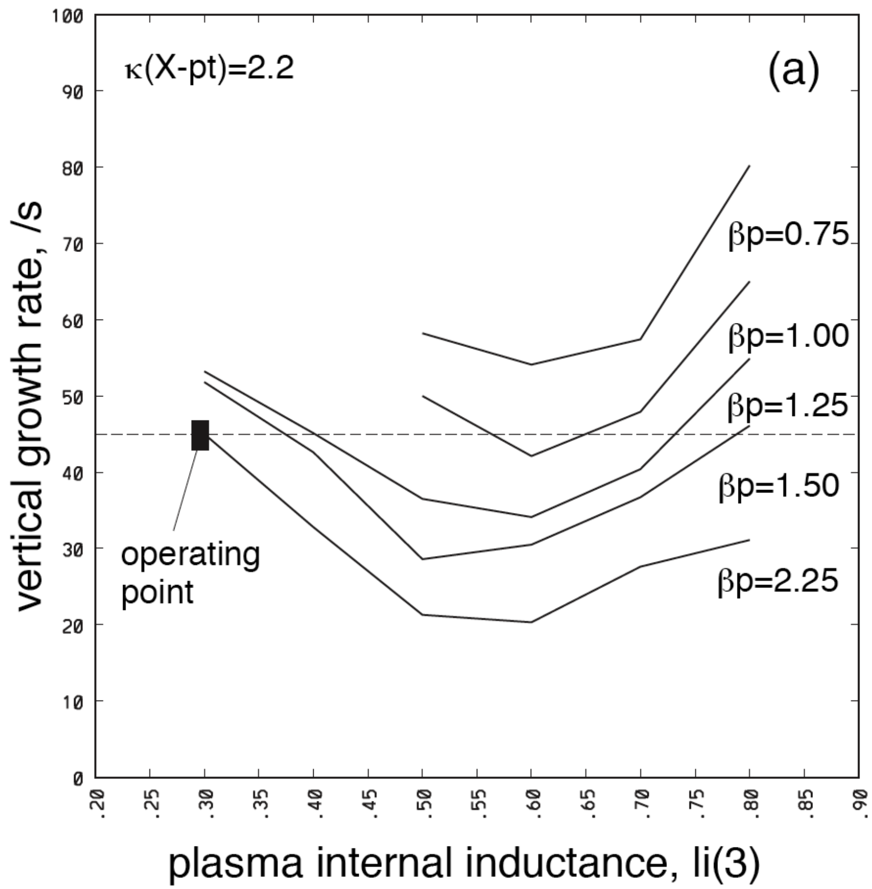


Feedback Control of Vertical Position

Analysis of the vertical control has been done with TSC to find I and V values, to give MVA requirement



Calculate vertical stability operating space as a function of current profile and pressure



Kink Instability Shell

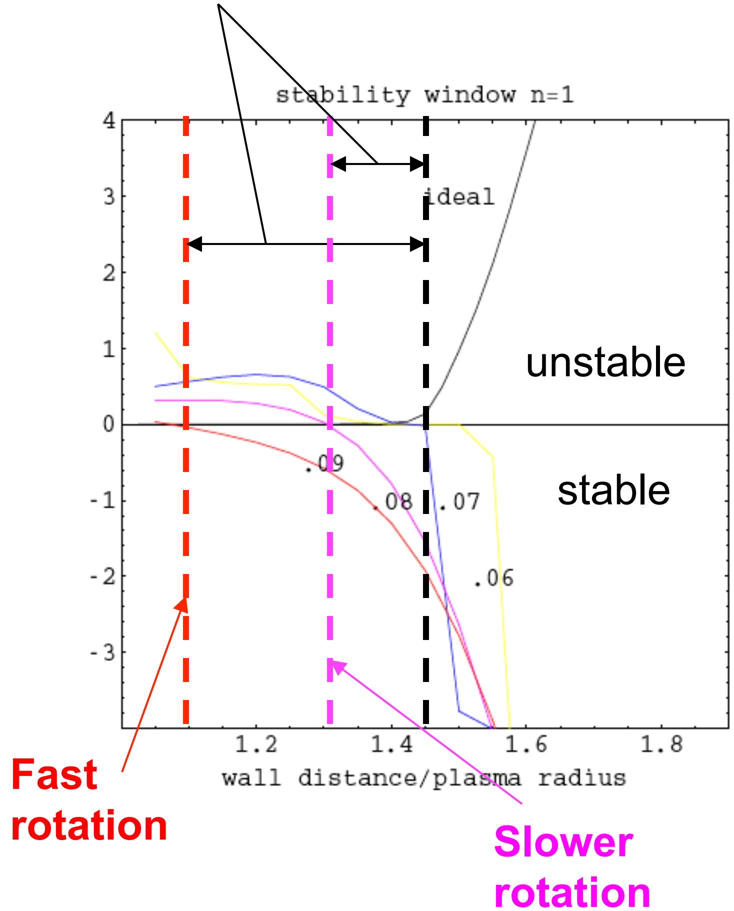
Placing conducting structures close enough to the plasma will slow the kink instability down, but **not** stabilize it

If the plasma is rotating and a damping mechanism exists then, the kink instability can be stabilized if the plasma rotates fast enough --- rotating large reactor plasmas is expected to be difficult

The alternative is to have feedback control coils to stabilize the plasma, and then plasma rotation is not required (we think) ---> this is our design choice

Recent expts show only slow rotation may be necessary, and kinetic stabilization is possible

Only for rotating plasmas, the wall must be within this distance from the plasma



Kink Feedback Control

If we assume the shell is close enough to the plasma and feedback coils are behind shield, then we can estimate its properties based on the feedback control

$$I = \pi Z |B_r| / \mu_o$$

$$V = 3N\mu_o RI / \tau_w$$

$$\tau_w = \mu_o \Delta b / \eta_w$$

$\tau_w \approx 3/(2\pi f)$, $f \approx 5$ Hz

$\tau_w \approx 0.1$ s

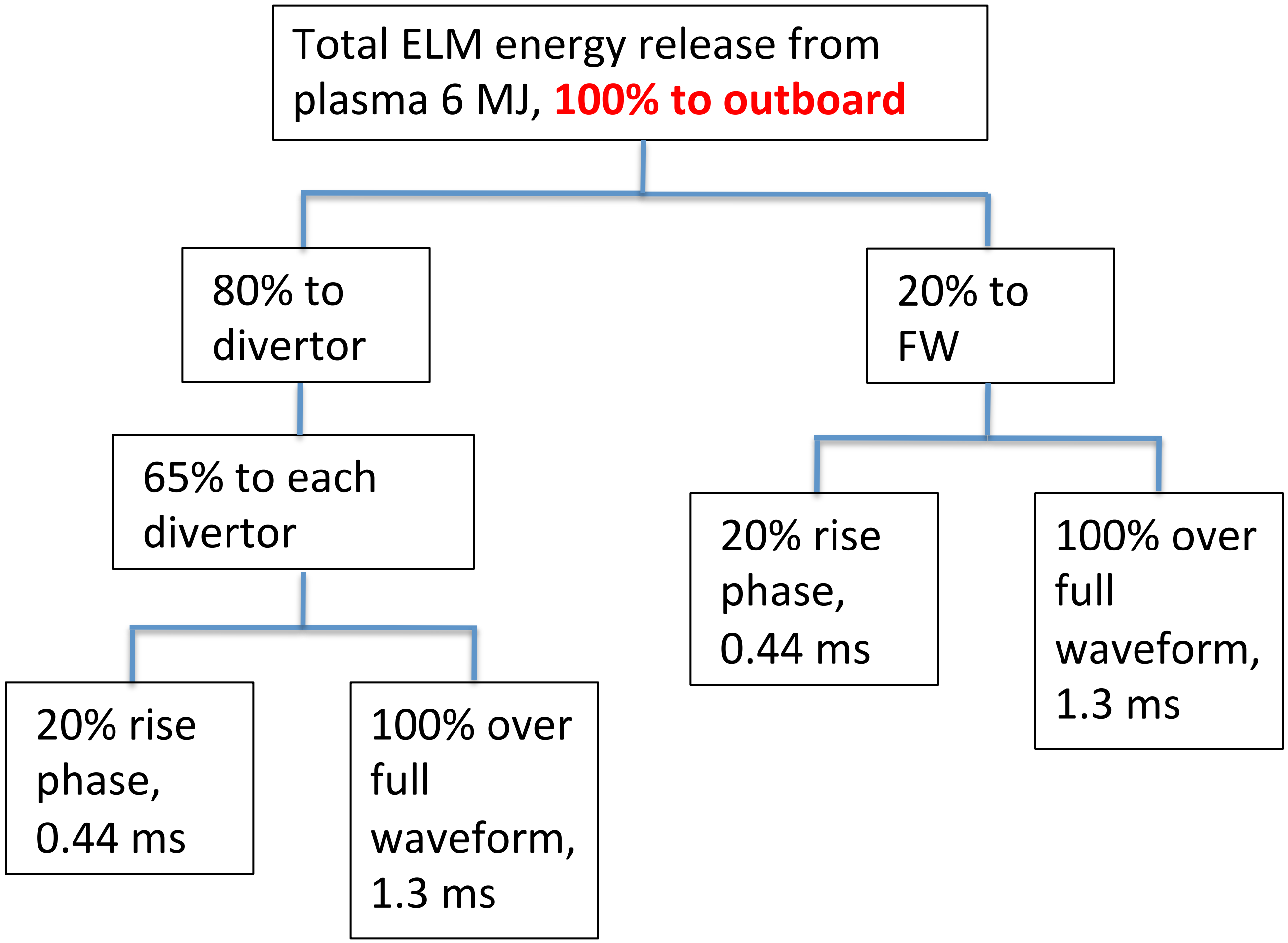
B_r = smallest detectable perturbation (then assume that coil should produce 20-50 times this)

Z = height of coil above midplane
R = major radius of coil
N = number of turns in coil

τ_w = shell time constant (approx)
Δ= shell thickness
b = minor radial shell distance
η_w = shell resistivity (function of T)

Leads and other parts of circuit are likely to make the coil performance worse, so keep τ_w large and f small

Small ELM Loading Description



$A_{ELM,\ OB\ divertor}$ = 1.38 m^2 (possible expansion by 1.5x)

$A_{ELM,\ OB\ FW}$ = 396 m^2/4 = 99 m^2 (peaking)

f_{ELM} ~ 20 Hz (/s)

Inter-ELM loading Specification

Both Large and Small ELM regimes, have ~ 205 MW (P_{SOL}) released from the plasma continuously between ELMs

This power follows the 80% to the outboard, and 20% to the inboard ($f_{OB/IB}$)

Then apply 65% for each divertor leg

- Upper OB divertor 0.8*0.65
- Lower OB divertor 0.8*0.65
- Upper IB divertor 0.2*0.65
- Lower IB divertor 0.2*0.65

Assuming 90% radiated power fraction in each divertor, multiply by 0.9 for radiated heat loading and 0.1 for conducted heat loading (f_{rad})

Areas are

- $A_{OB\ div,conduct} = 1.38\ m^2$
- $A_{OB\ div,rad} = 53.6\ m^2$
- $A_{IB\ div,conduct} = 1.18\ m^2$
- $A_{IB\ div,rad} = 23.0\ m^2$

Assuming no ELMs, the steady heat flux would be the same formula with P_{SOL} = 290 MW

$$q'' = P_{SOL} * f_{OB/IB} * 0.65 * [(1-f_{rad}) / A_{OB\ div,conduct} + f_{rad} / A_{OB\ div,rad}]$$

Loading Prescription for TQ

Plasma stored energy, W_{th} =
345 (VDE) - 690 (MD) MJ
is released in TQ

10-50% goes to divertor,
time scale of 3 x Δt_{TQ}
Δt_{TQ} ~ 2 ms
10x area increase
Full energy level

0-15% radiated
OB/IB, 80/20%
ignored

90-50% goes to FW,
outboard only
time scale of 3 x Δt_{TQ}
Δt_{TQ} ~ 2 ms
2x peaking on OB area
Full energy level

65% to each divertor

Plasma Current Quench Energy Flow

Plasma magnetic energy, $1/2L_pI_p^2$

$1/2L_{int}I_p^2 + 1/2L_{ext}I_p^2$

Only track magnetic energy release while plasma exists

CQ is 25 ms while the time constant of the VV is seconds

$1/2L_{int}I_p^2 + f\, x1/2L_{ext}I_p^2$, use $f \sim 0.2$

Regular CQ

- 20-30% into eddy currents
- 80-40% radiated to FW
- 0-30% cond/conv to FW

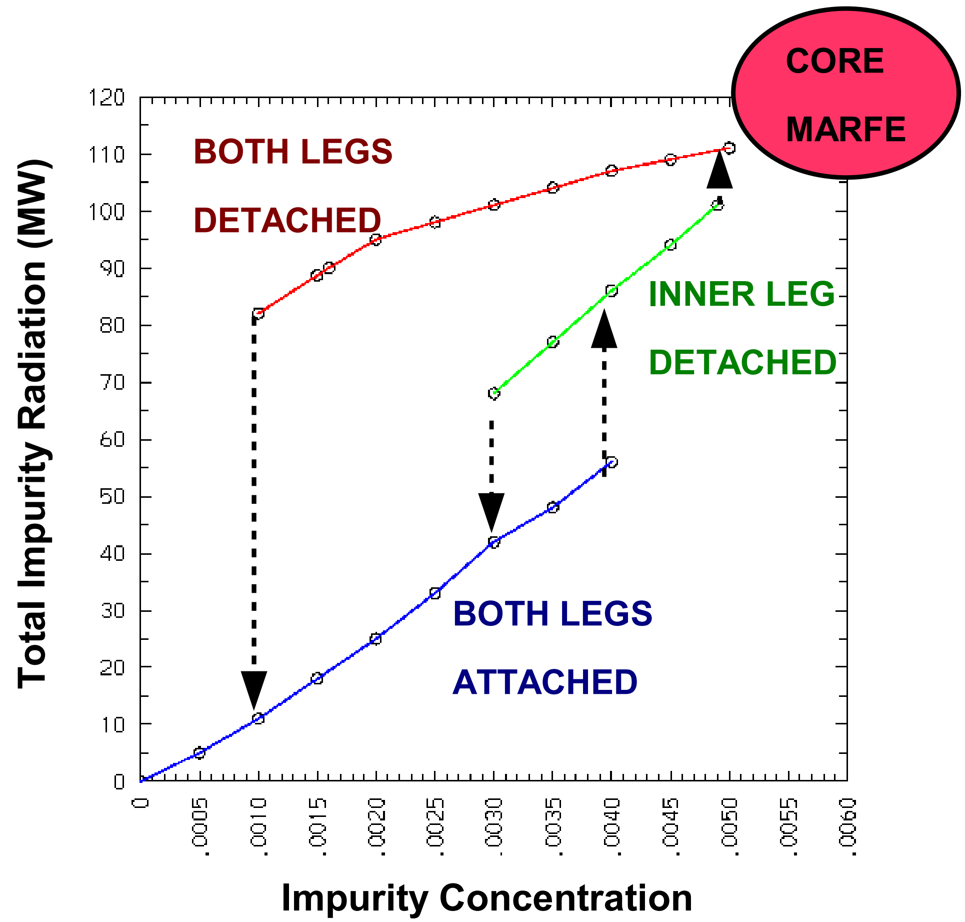
- (IB/OB 20/80) for rad
- OB only for cond/conv
- 2x peaking rad, cond/conv
- Time scale 25 ms

Runaway Electron (RE) CQ

- 50% radiated to FW in initial CQ
- 20-30% into eddy currents
- 30-10% kinetic RE energy and cond/conv to FW
- 0-10% radiated post RE

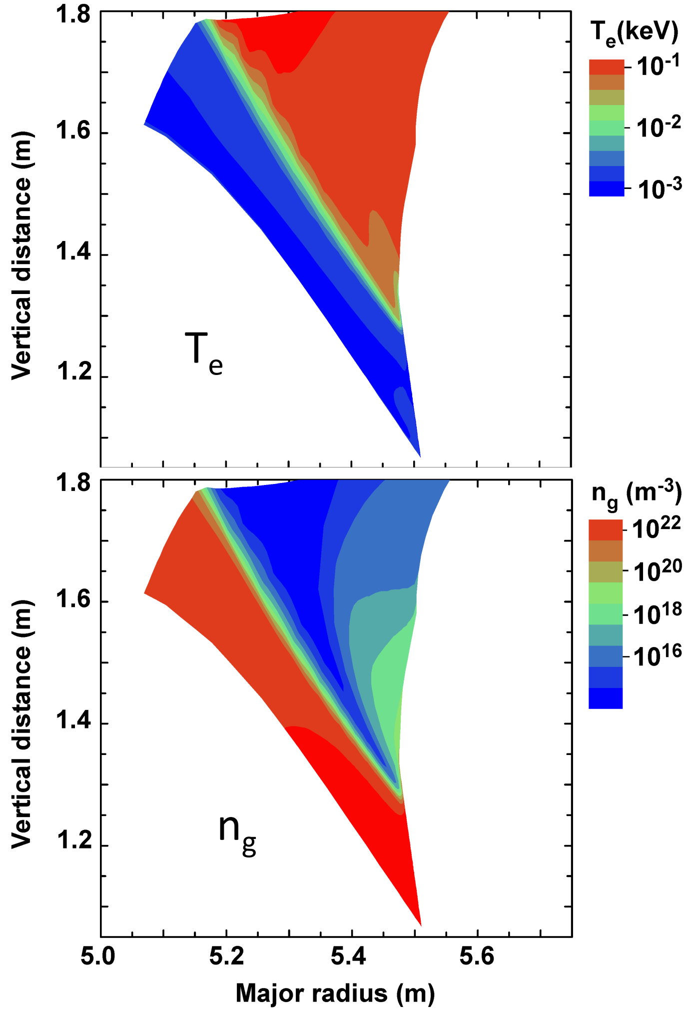
Multiple solutions can be observed when finding stable operating regimes for the divertor

- Uniform concentration (Neon)
- Orthogonal divertor plates
- Pedestal density 1.5×10^{20} m^{-3}
- Power from pedestal 160 MW
  (lower half of double null)



Backup – Differences in Outer Leg Plasmas and Neutrals for Partially and Fully Detached Cases

Tilted plate partial detachment has strong in/out asymmetry



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