# Detached plasma regimes in innovative long-legged divertor configurations

**M.V. Umansky[1], B. LaBombard[2], D. Brunner[2], T. Golfinopoulos[2], A.Q. Kuang[2], M.E. Rensink[1], J.L. Terry[2], M. Wigram[3], D.G. Whyte[2]**

[1] *Lawrence, Livermore National Laboratory, Livermore, CA 94550, USA*
[2] *MIT Plasma Science and Fusion Center, Cambridge, MA 02139, USA*
[3] *York Plasma Institute, Dept. of Physics, Univ. of York, York, YO10 5DD, UK*

*Presented at Sherwood Fusion Theory Conference, April 23-25, 2018, Auburn, AL*

This work was performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Security, LLC, Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344.
LLNL-PRES- 749279

# Outline

- **Motivation:**
  - Tokamak divertor challenge prompts search for innovative divertors
- **Simulations of long-legged divertors:**
  - Stable fully-detached regime found in computational study of long-legged divertors
- **Analysis of simulations results:**
  - Understanding physics mechanisms in numerical solutions and assessment of sensitivity to various model assumptions lends confidence in modeling results
- **Summary & Conclusions:**
  - Detached regime found in simulations of long-legged divertors looks promising for high-power tokamaks

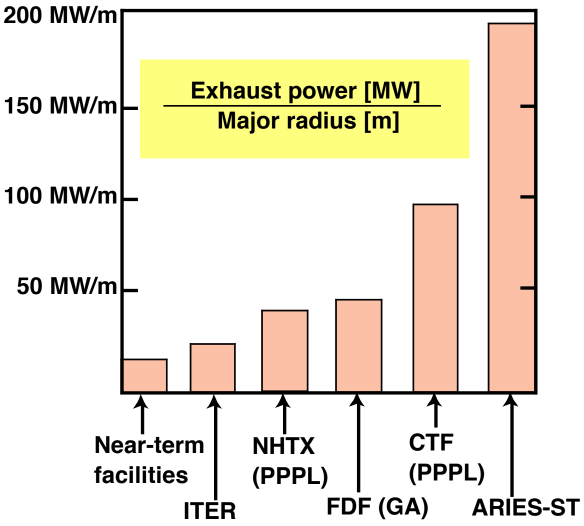

Fusion Energy Sciences

# Divertor heat exhaust is going to be a major challenge for next generation of tokamaks

- **SOL width $\lambda_{SOL}$ small (~1mm)**
  **$\Rightarrow$ divertor heat flux large**

- **For constant $\lambda_{SOL}$ an important figure of merit is P/R**

$$q_{div} = \frac{P}{2\pi R \lambda_{SOL}} \propto \frac{P}{R}$$






Fusion Energy Sciences

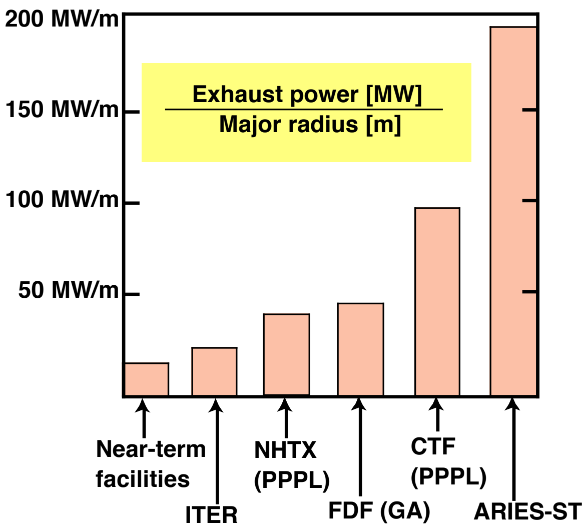
# Divertor heat exhaust is going to be a major challenge for next generation of tokamaks

- **SOL width $\lambda_{SOL}$ small (~1mm)**
  **$\Rightarrow$ divertor heat flux large**

- **For constant $\lambda_{SOL}$ an important figure of merit is P/R**

$$q_{div} = \frac{P}{2\pi R \lambda_{SOL}} \propto \frac{P}{R}$$

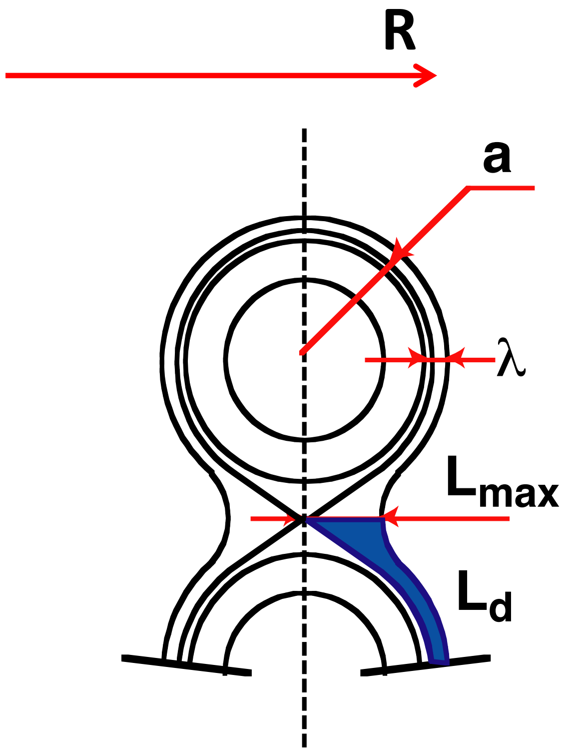
- **Moreover, the recently found scaling $\lambda_{SOL} \sim 1/I_p$ independent of machine size* is unfavorable for large tokamaks**




- **For next generation tokamaks innovative divertor solutions are needed!**

*Eich et al., 2013 Nucl. Fusion 53 093031

# Divertor parameters are constrained by overall tokamak design; but there are a few degrees of freedom left for divertor



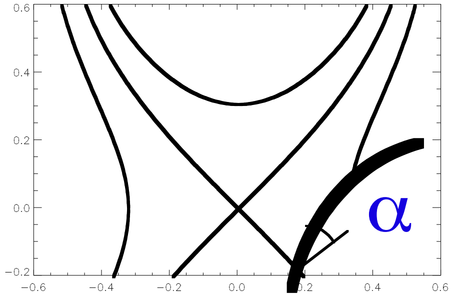

**For given tokamak design one cannot change:**

- Exhaust power
- Major radius
- Minor radius
- SOL width

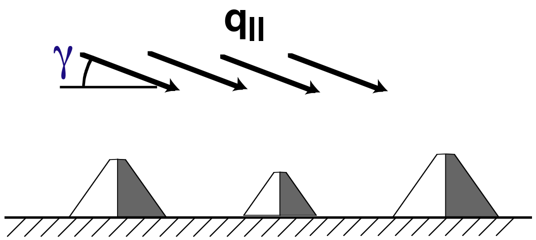
**But (within some limits) one can change:**

- Divertor plate tilt/shaping
- Divertor leg length
- Divertor poloidal flux expansion
- Divertor magnetic field topology

# Increasing plasma-wetted area $A_w$ geometrically is limited for given target major radius $R_t$




- **$A_w$ can be increased by plate tilting and poloidal flux expansion**
- **For either method, grazing angle $\gamma$ between surface and total B becomes small**




- **For $\gamma<\gamma_0\approx1^o$ - hot-spot formation due to surface roughness**
- **At minimum angle $\gamma_0$: $A_w \approx 2\pi R_t\, [(\lambda_{mid}/\gamma_0)\, (B_{pol}/B_{tor})_{mid}]$ [1]**
- **For further increase of $A_w$ larger target radius $R_t$ needed (super-X idea) [2]**

**Purely geometric solutions for innovative divertor are limited!**

[1] Ryutov et al., Proc. 2008 Fusion Energy Conf., Paper IC/P4-8
[2] Valanju et al., Phys. Plasmas 16, 056110, 2009



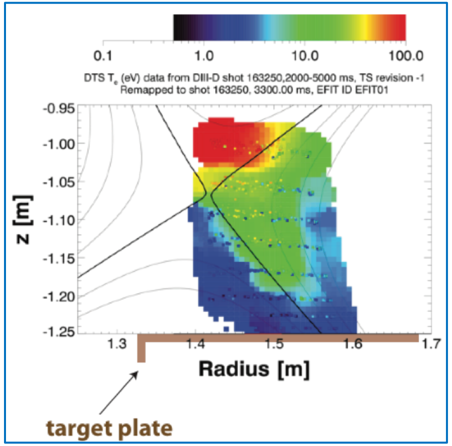
Fusion Energy Sciences

# Divertor detachment: plasma stays away from plasma-facing components, cushioned by neutral gas

- Detached divertor will be <u>required</u> for reactor
  - reduction of heat flux at target
  - suppression of erosion

- Challenge of detached operation:
  - Need stable, controllable state
  - Need large operating window
  - Need to maintain good pedestal/core

$T_e$ from divertor Thomson




A. McLean et al., APS DPP 2015



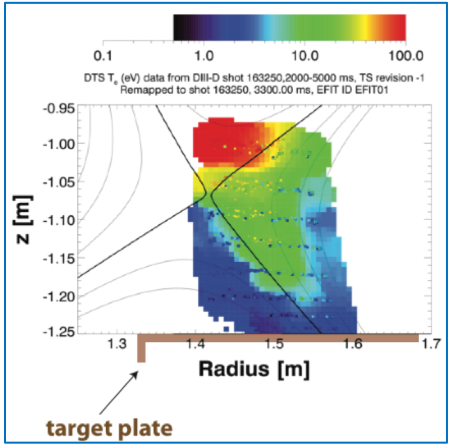
Fusion Energy Sciences

# Divertor detachment: plasma stays away from plasma-facing components, cushioned by neutral gas

- Detached divertor will be <u>required</u> for reactor
  - reduction of heat flux at target
  - suppression of erosion

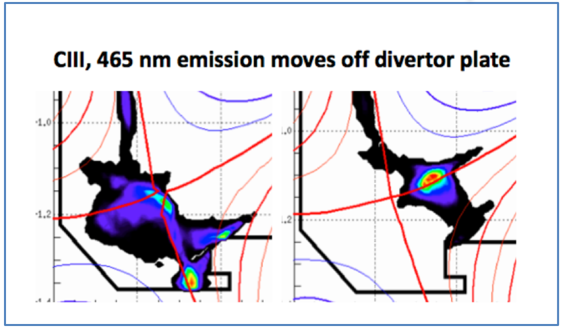
- Challenge of detached operation:
  - Need stable, controllable state
  - Need large operating window
  - Need to maintain good pedestal/core
  - Full detachment often leads to MARFE
    - radiation near main X-point
    - degradation of core confinement

$T_e$ from divertor Thomson




A. McLean et al., APS DPP 2015

M. Fenstermacher et al., reconstructed CIII data from TV cameras on DIII-D






Fusion Energy Sciences

# Outline

- Motivation:
  - Tokamak divertor challenge prompts search for innovative divertors

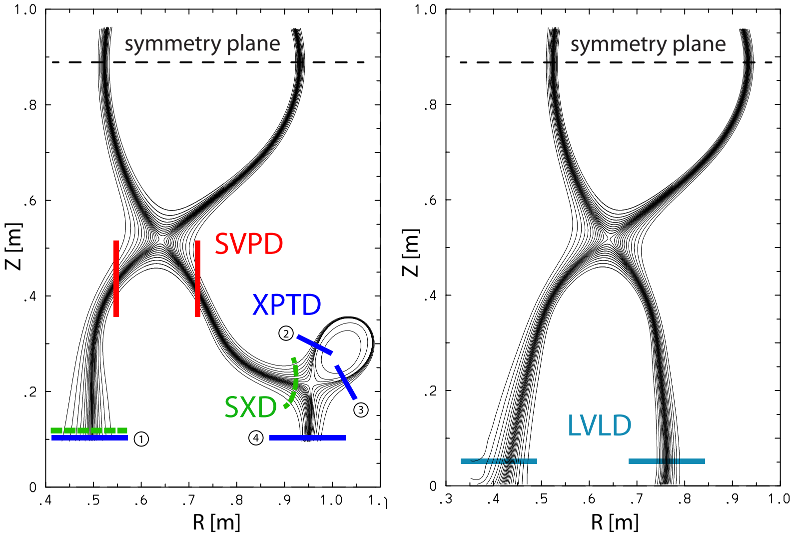
- Simulation of long-legged divertors:
  - **Stable fully-detached regime found in computational study of long-legged divertors**

- Analysis of simulations results:
  - Understanding physics mechanisms in numerical solutions and assessment of sensitivity to various model assumptions lends confidence in modeling results

- Summary & Conclusions:
  - Detached regime found in simulations of long-legged divertors looks promising for high-power tokamaks


Fusion Energy Sciences

# UEDGE is applied to four tokamak divertor arrangements based on same (or similar) magnetic configurations




- **SVPD – Standard Vertical Plate Divertor**
- **SXD – Super-X Divertor**
- **XPTD – X-point Target Divertor**
- **LVLD – Long Vertical Leg Divertor**

- **Tokamak edge transport code UEDGE [1] finds a steady state solution of plasma fluid equations in edge domain**
- **UEDGE domain can include a secondary X-point – important for modeling innovative divertors**
- **UEDGE setup used here is based on design of ADX tokamak [2]**


[1] Rognlien et al., J. Nucl. Mater. 196–198, 347 (1992)
[2] LaBombard et al., Nucl. Fusion 55, 053020 (2015)




Fusion Energy Sciences

# Enhanced radial transport in far SOL and strong interaction with outer wall is essential part of our physics model

- **Non-diffusive nature of edge plasma transport established since late 1990s [1,2,3]**
- **Time-average SOL transport can be represented by radially growing $D_r$ or radial outward pinch $V_r$**
- **Directly connected to ballistic dynamics of filamentary plasma structures in the edge**
- **Overwhelming evidence from tokamaks and other devices**

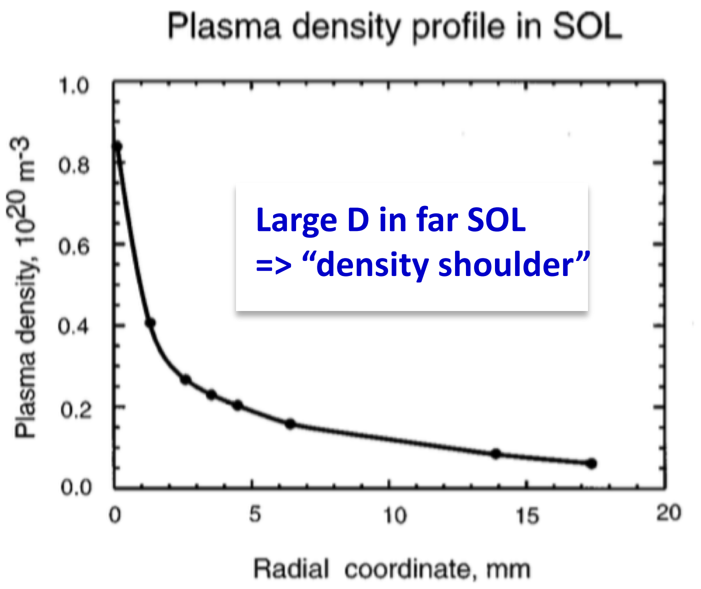


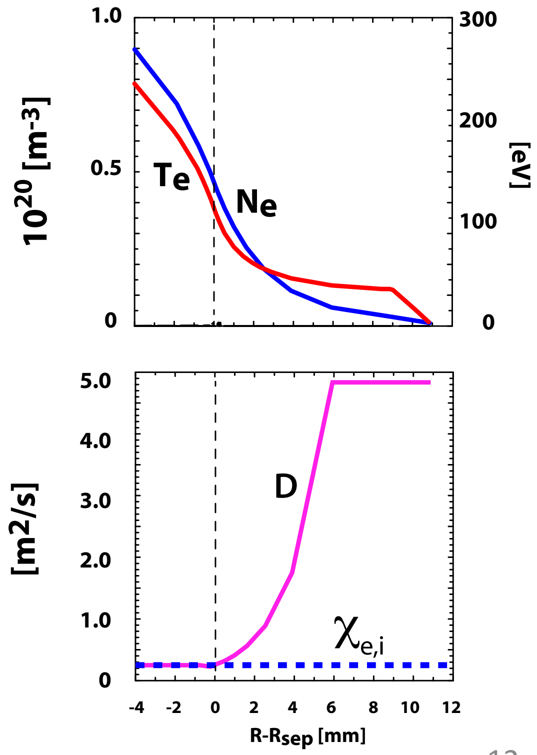

FIG. 3. Typical experimental plasma density profile in the Alcator C-Mod.


[1] Umansky et al., Phys. Plasmas, 5(9):3373, 1998
[2] LaBombard et al., Nucl. Fusion, 40(12):2041–2060, 2000
[3] Krasheninnikov, Phys. Lett. A 283, 368, 2001

# UEDGE model is set to match projected ADX design

- **Modeled cases based on geometry & parameters from ADX tokamak design [1]**
  - **MHD equilibrium**
  - **Power into lower half-domain $P_{1/2}$**
  - **Density at separatrix ~0.5e20 $m^{-3}$**
  - **SOL profiles width**
- **Using fully recycling wall B.C. on all material surfaces**
- **Using radially growing diffusion coefficient $D_{\perp}$ to match expected $N_{e,i}$ profile width ~5 mm**
- **Spatially constant $\chi_{\perp e,i}$ is sufficient to achieve ~3 mm width of mid-plane $T_{e,i}$**

Mid-plane profiles

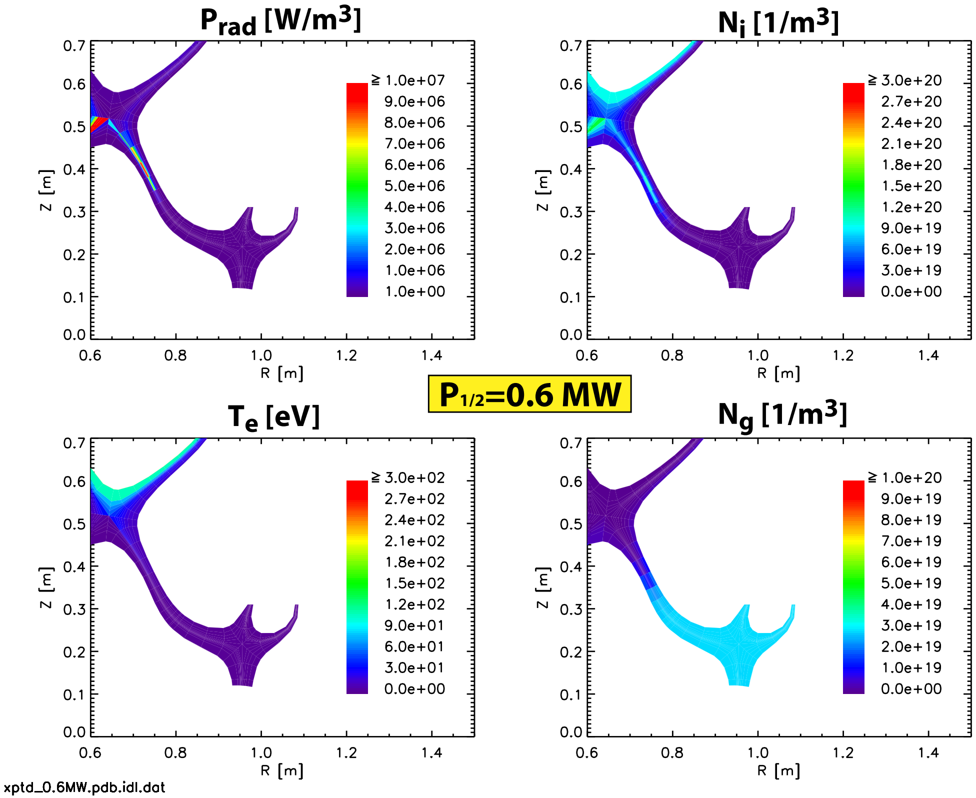



[1] LaBombard et al., Nucl. Fusion 55, 053020, 2015.

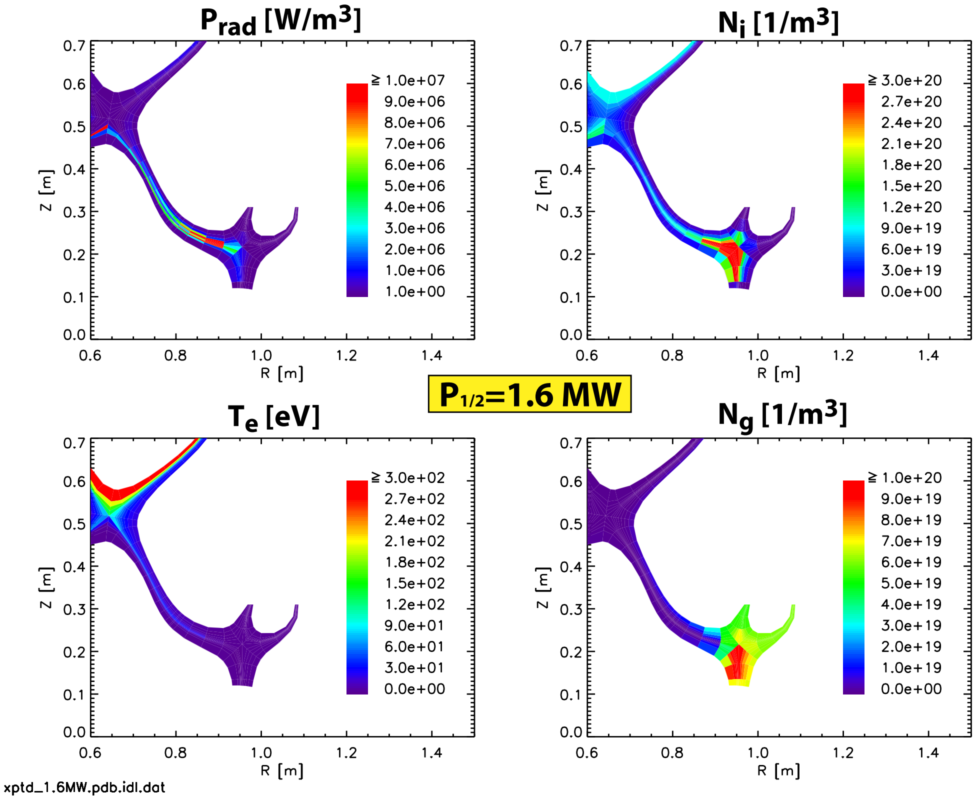


Fusion Energy Sciences

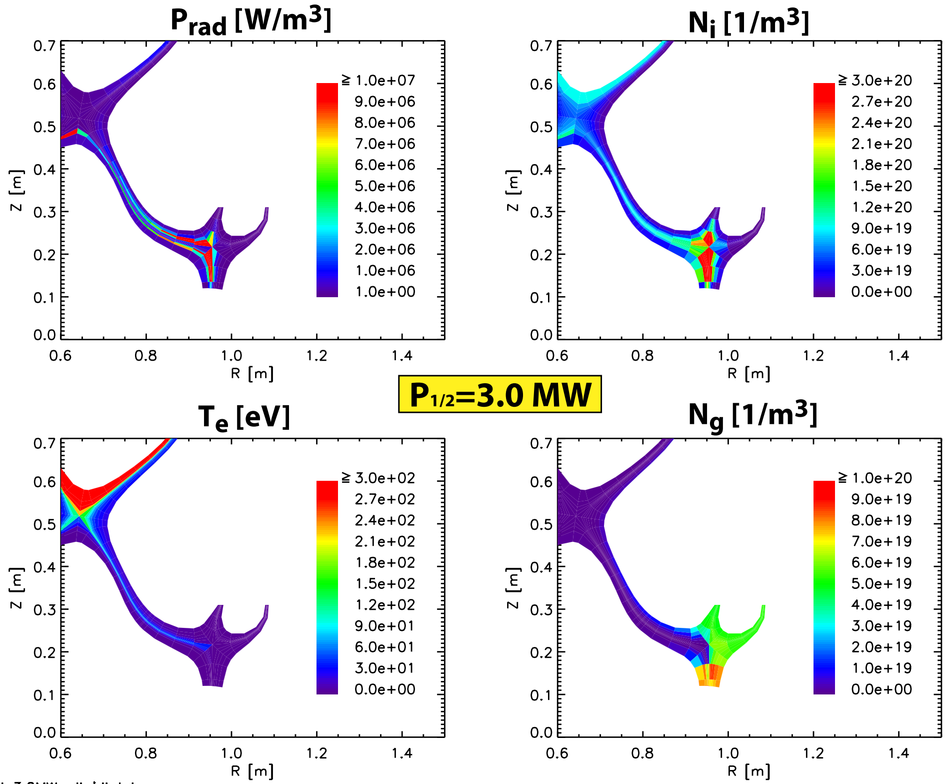
# Results for XPTD, power $P_{1/2}$ = 0.6 MW



# Results for XPTD, power $P_{1/2}$ = 1.6 MW

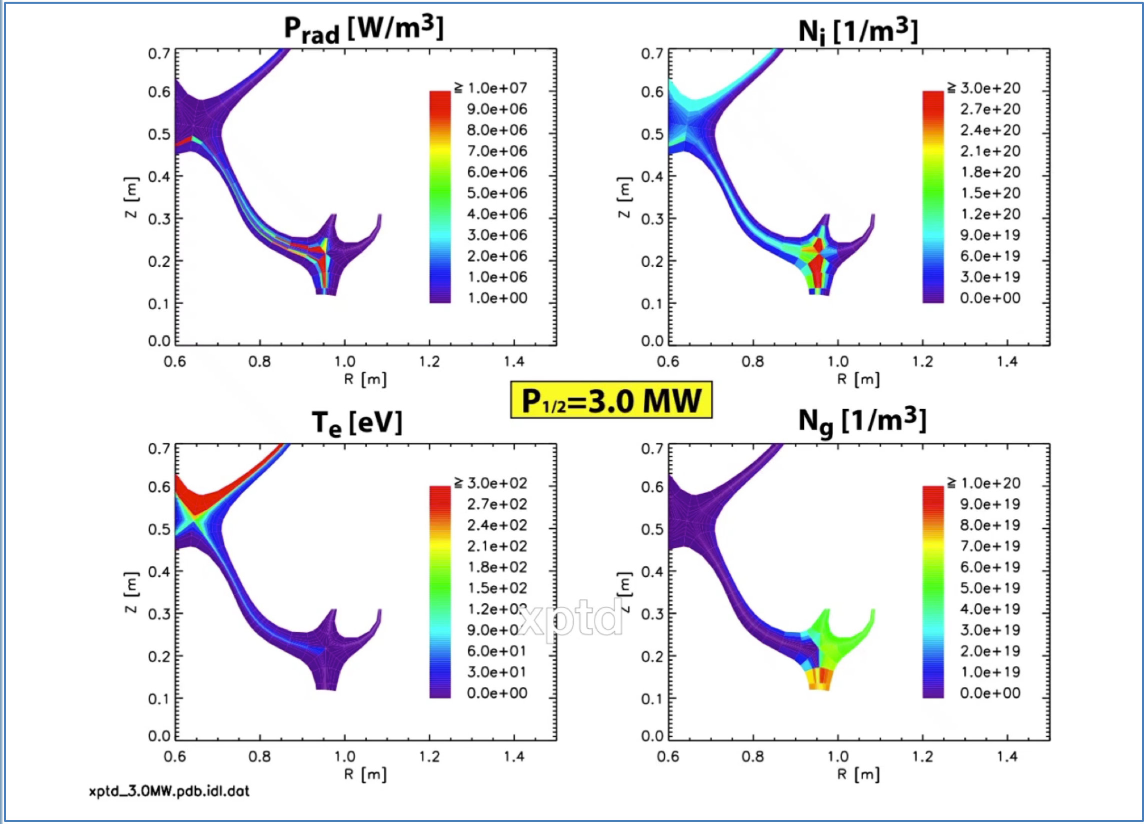


# Results for XPTD, power $P_{1/2}$ = 3.0 MW




xptd_3.0MW.pdb.idl.dat

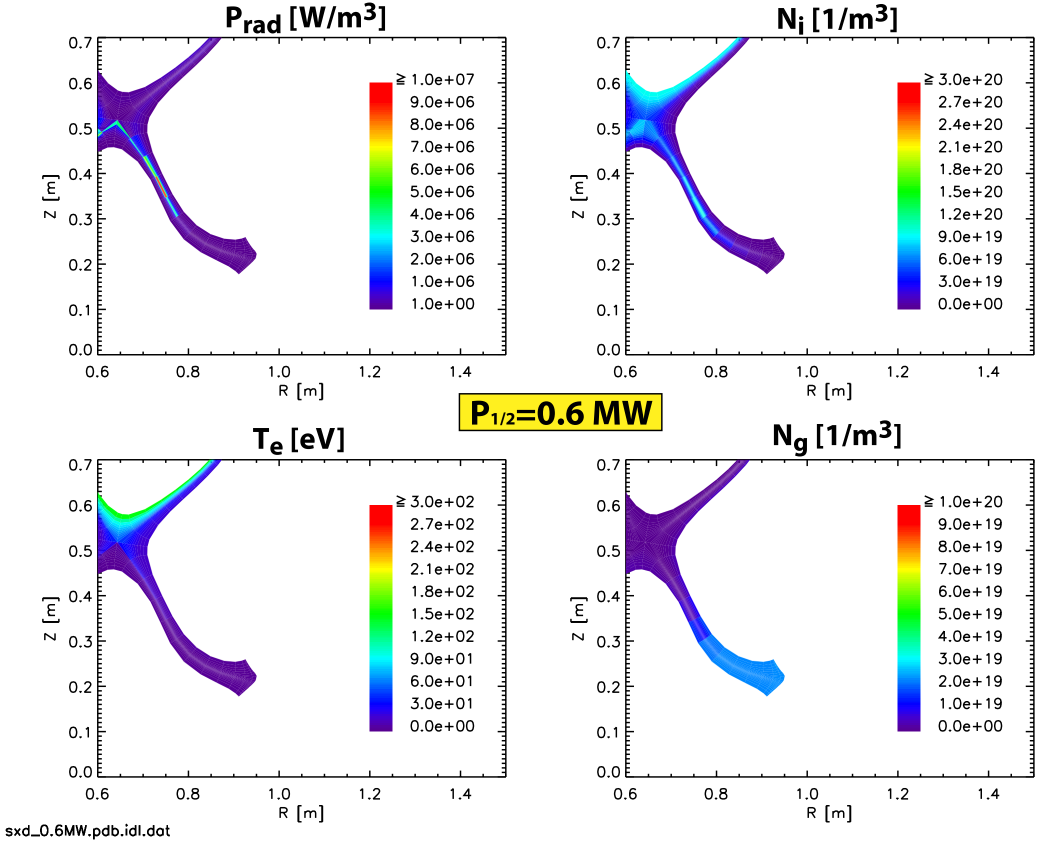
# Results for XPTD: steady-state location of detachment front shifts up/down in response to input power






Fusion Energy Sciences

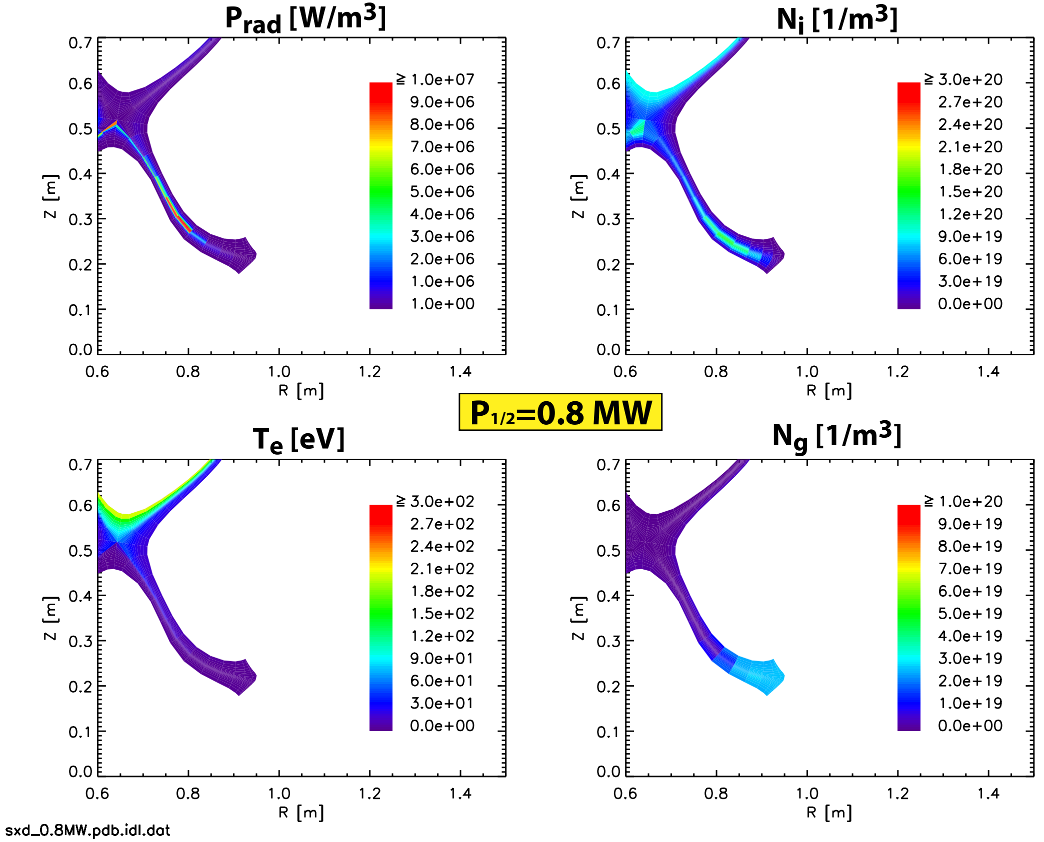
# Results for SXD, power $P_{1/2}$ = 0.6 MW



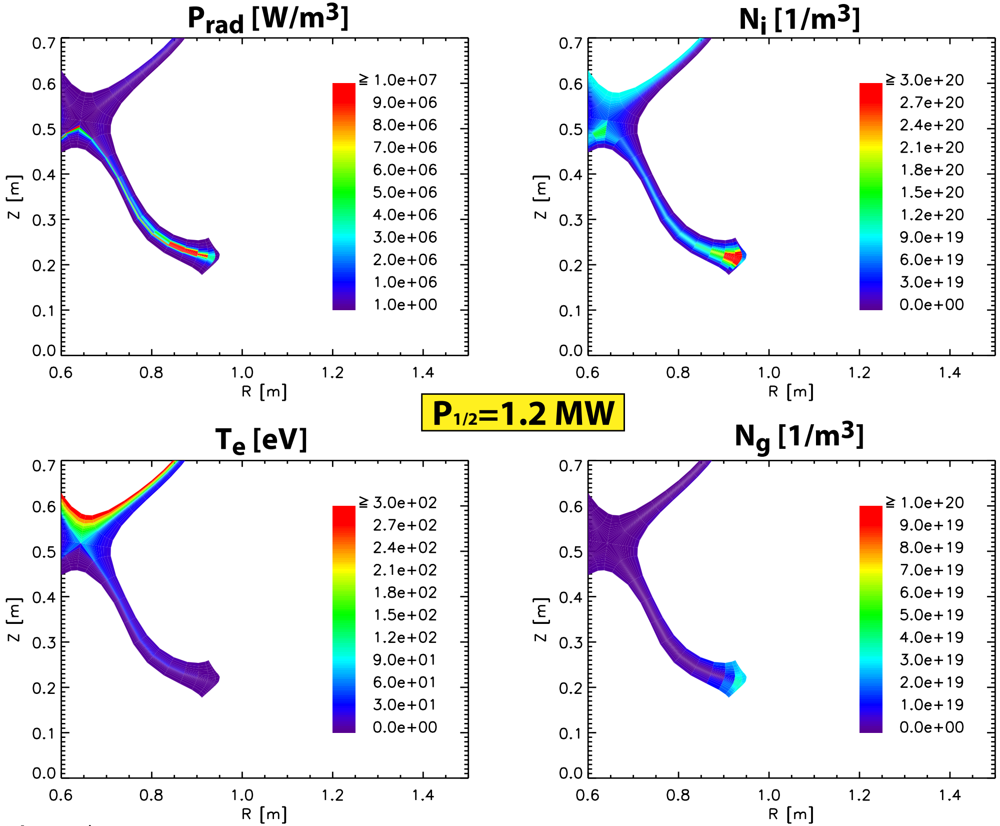



Fusion Energy Sciences

# Results for SXD, power $P_{1/2}$ = 0.8 MW

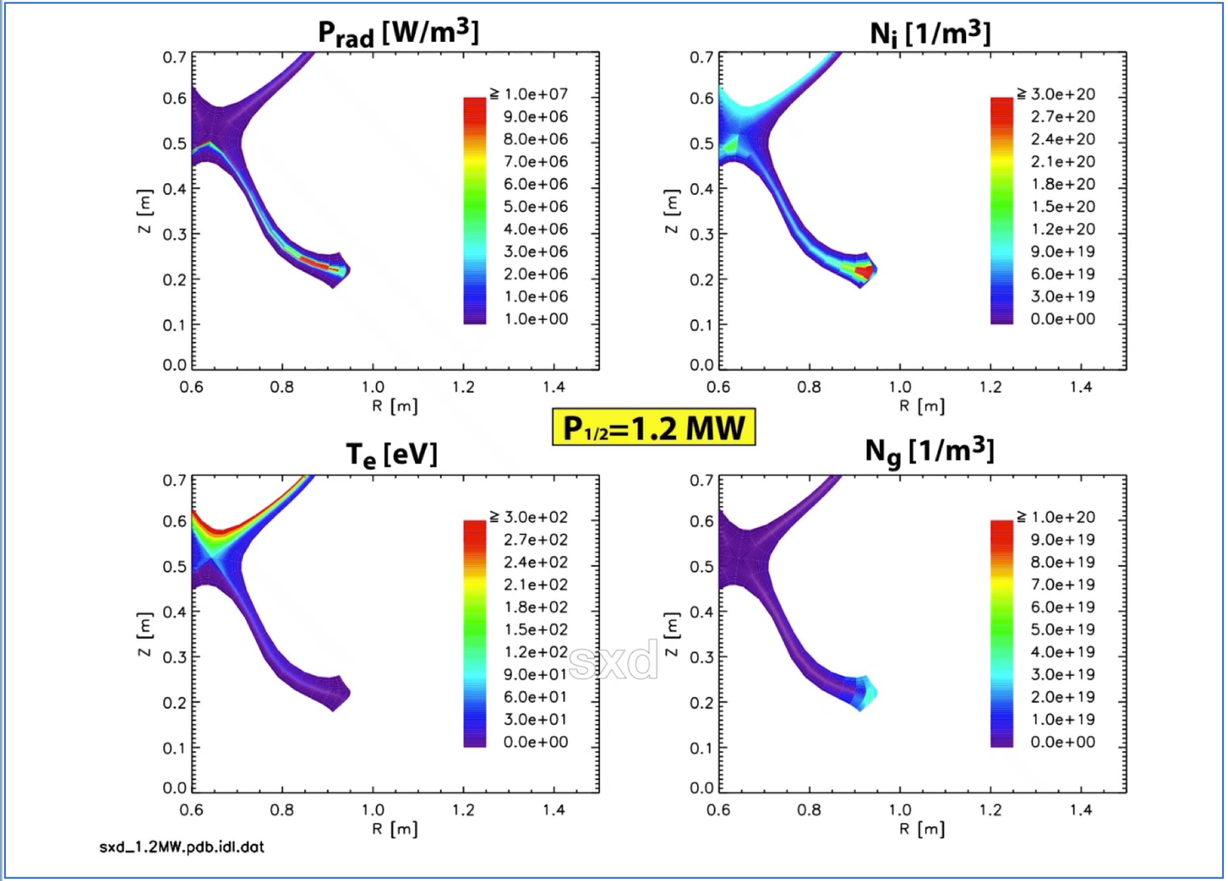


# Results for SXD, power $P_{1/2}$ = 1.2 MW




sxd_1.2MW.pdb.idl.dat

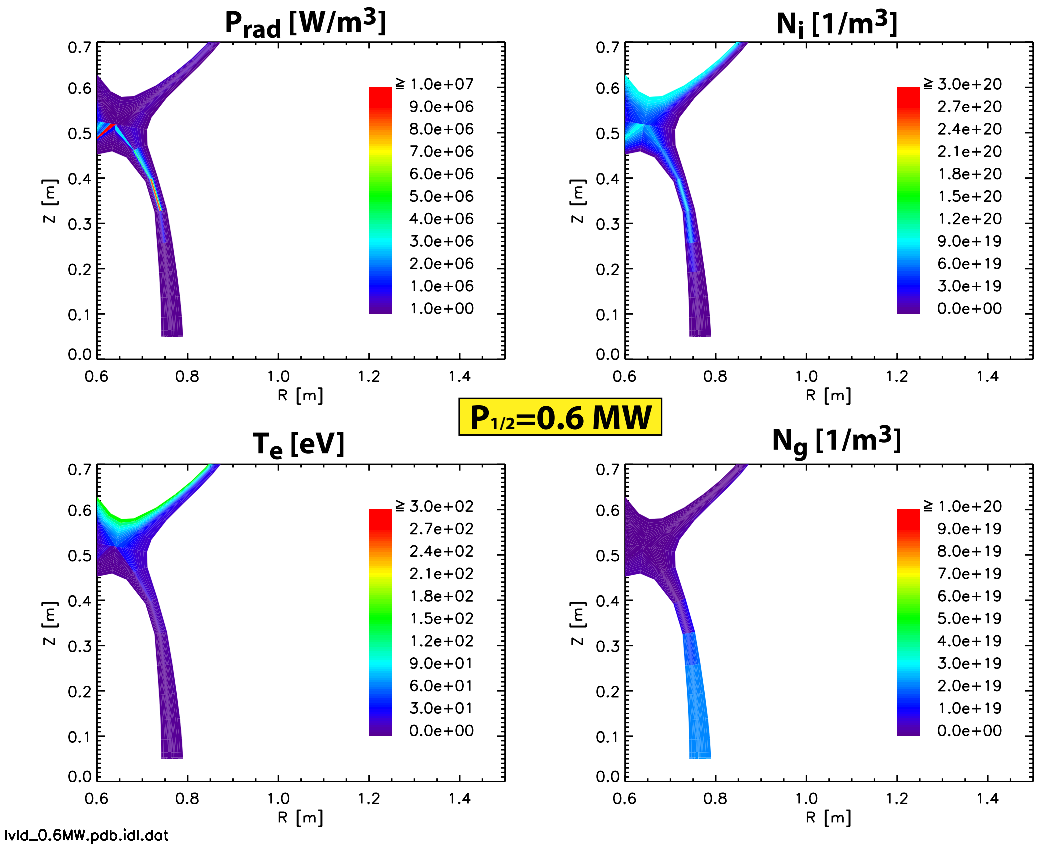
# Results for SXD : steady-state location of detachment front shifts up/down in response to input power






Fusion Energy Sciences

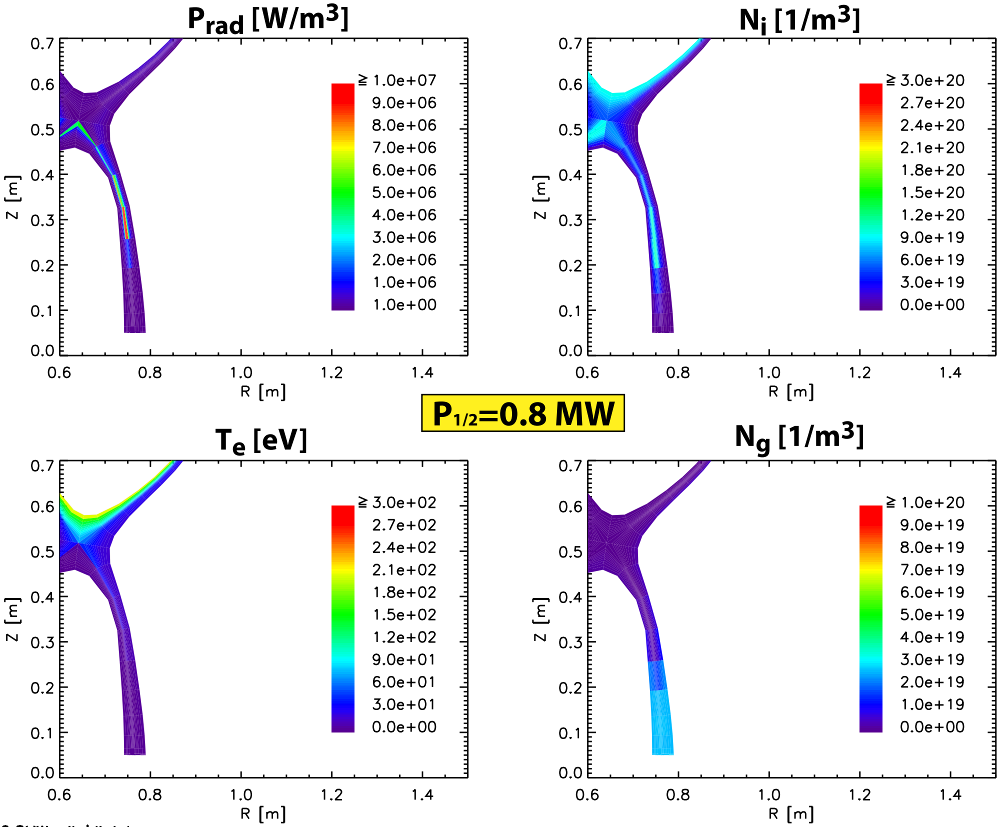
# Results for LVLD, power $P_{1/2}$ = 0.6 MW





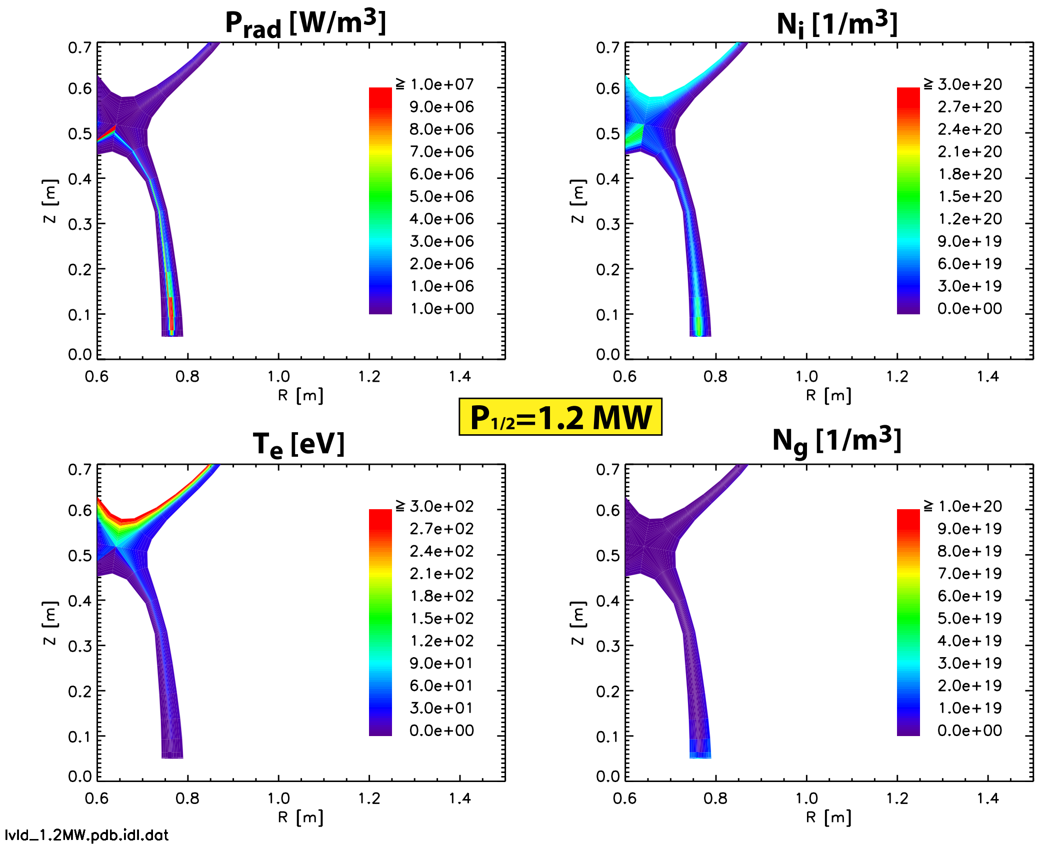

Fusion Energy Sciences

# Results for LVLD, power $P_{1/2}$ = 0.8 MW




lvld_0.8MW.pdb.idl.dat

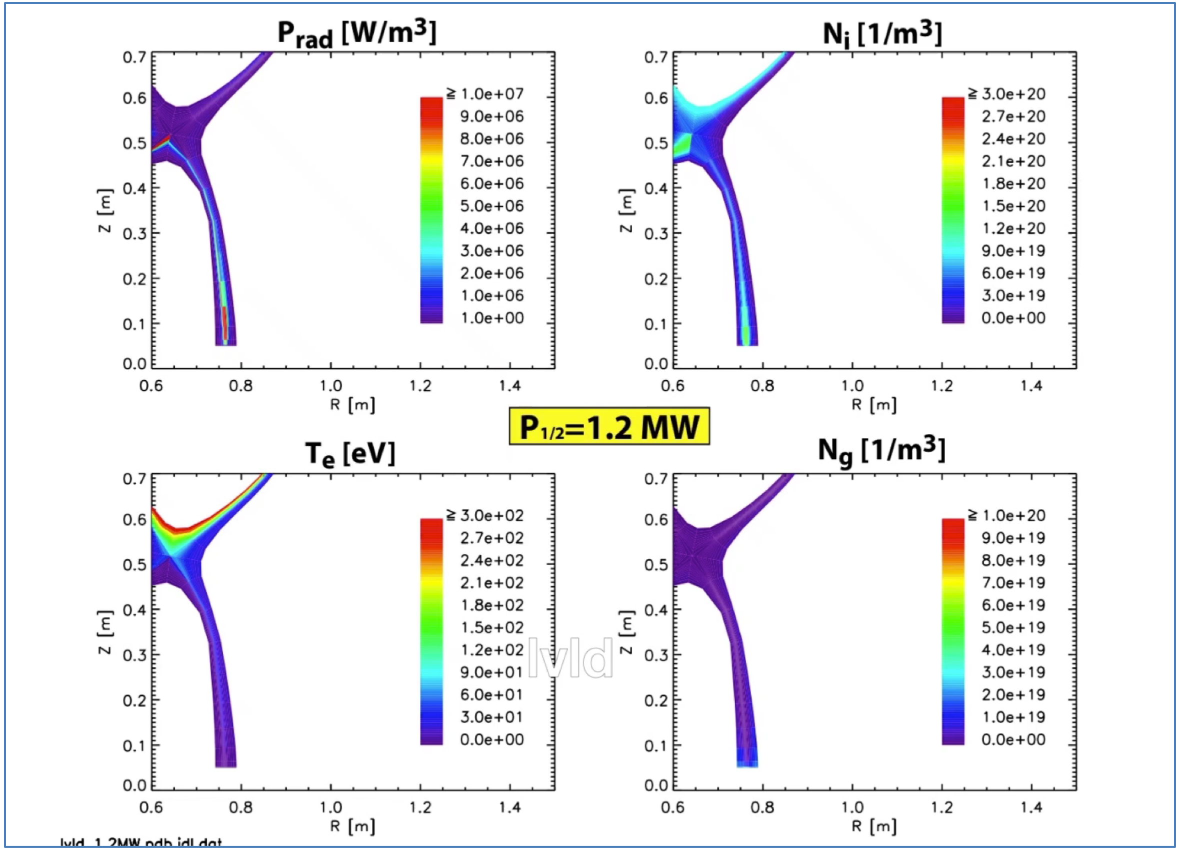
# Results for LVLD, power $P_{1/2}$ = 1.2 MW






Fusion Energy Sciences

# Results for LVLD : steady-state location of detachment front shifts in response to input power

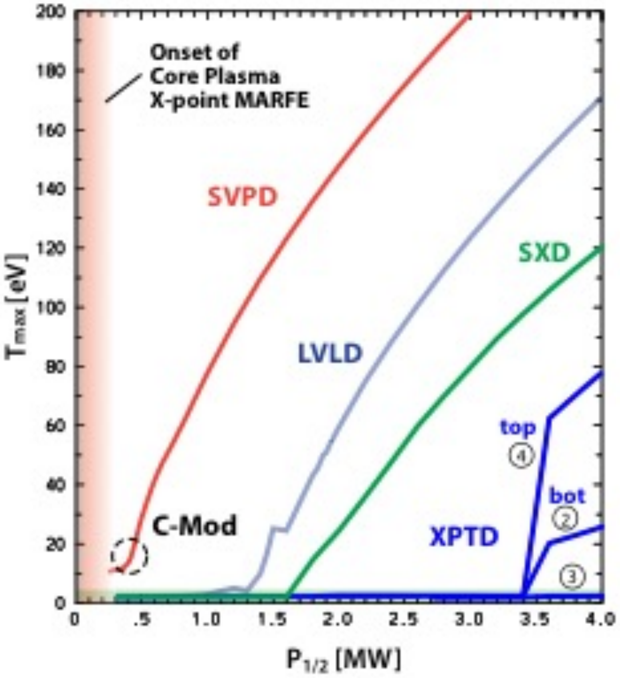





Fusion Energy Sciences

# Varying input power into SOL shows how transition to detachment depends on divertor configuration

$T_{max}$ on outer plate[s] vs. $P_{1/2}$




$P_{1/2}$ = power into the lower half-domain

XPTD: plate 2 - top, plate 4 - bottom

- Large operational window with detached divertor ($T_{max}$<5 ev) found for all three long-legged configurations [1]
- For standard divertor (SVPD), detached plasma solution may exist but at rather low input power
- Radially or vertically extended outer leg is good for detached operation
- Long vertical leg (LVLD) achieves detachment at about same power as radially extended leg (SXD)
- Secondary X-point in outer leg (XPTD) extends detached operation window

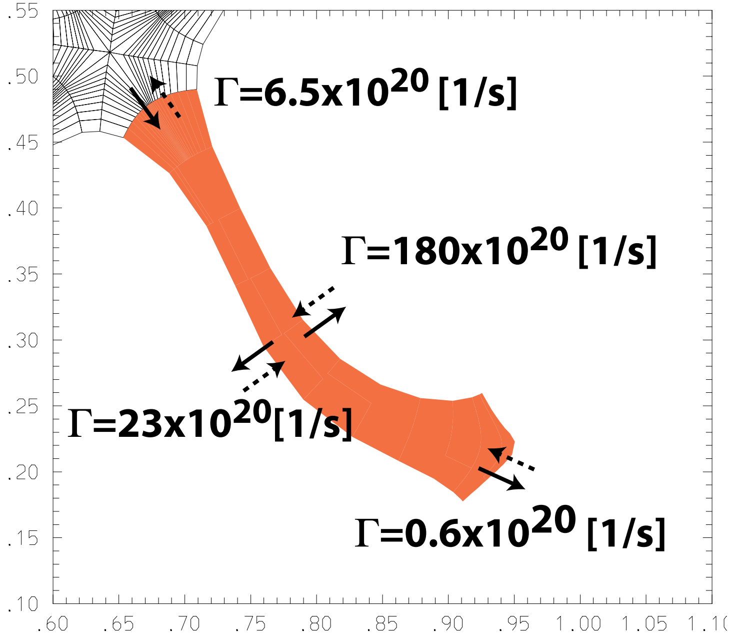

[1] Umansky et al., Phys. Plasmas, 24(5), 2017

# Outline

- Motivation:
  - Tokamak divertor challenge prompts search for innovative divertors
- Simulation of long-legged divertors:
  - Stable fully-detached regime found in computational study of long-legged divertors
- **Analysis of simulations results:**
  - **Understanding physics mechanisms in numerical solutions and assessment of sensitivity to various model assumptions lends confidence in modeling results**
- Summary & Conclusions:
  - Detached regime found in simulations of long-legged divertors looks promising for high-power tokamaks

# Analysis of plasma & neutral density fluxes in divertor leg show stagnant poloidal flow picture

**Analyzing a representative case:**




**SXD, $P_{1/2}$=0.6 MW**

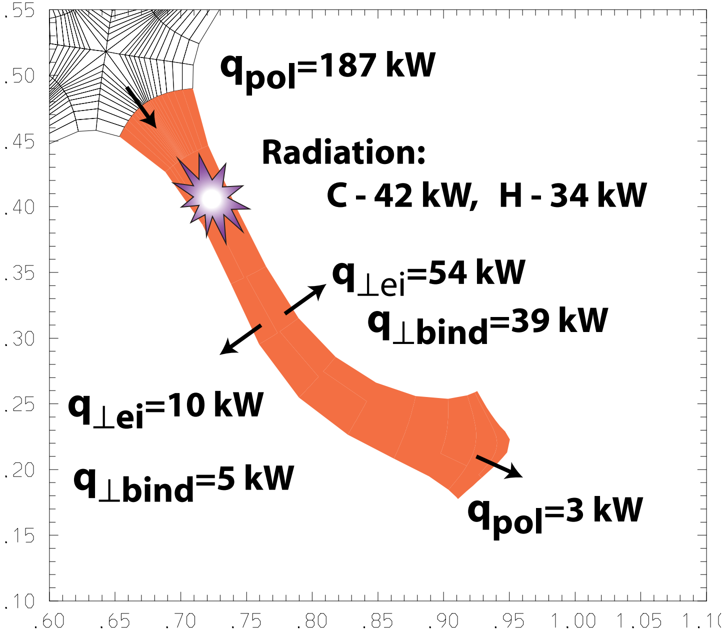
- **On each surface bounding the leg domain ion flux is matched by opposing neutral flux**
- **Poloidal fluxes entering/leaving domain are tiny compared to radial fluxes**
- **SOL flow is stagnant**
- **Poloidal plasma pressure gradient is balanced by CX interaction with neutrals in divertor leg**



Fusion Energy Sciences

# Analysis of energy fluxes in divertor leg shows that most entering energy ends up on outer divertor wall

**Analyzing a representative case:**

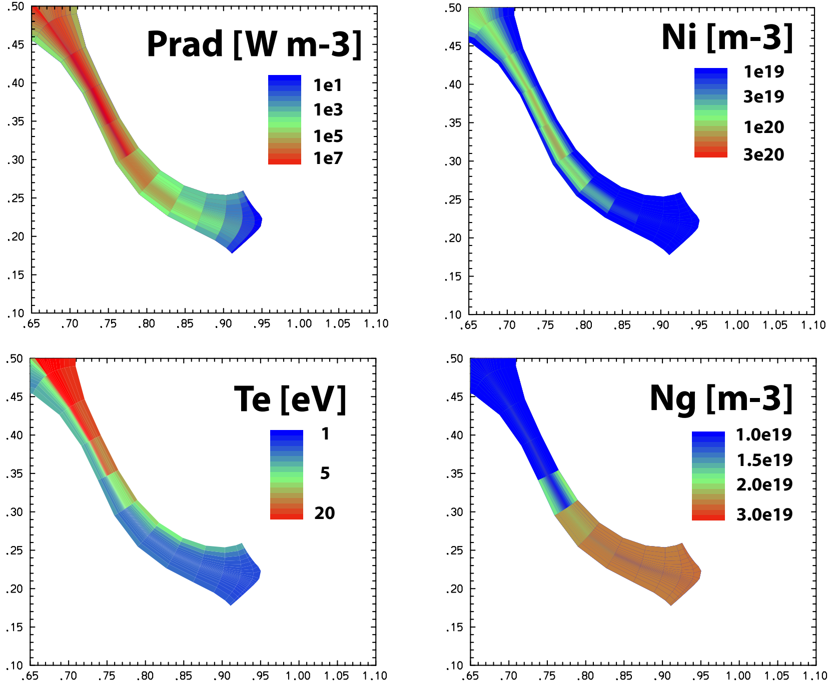



- About 1/2 of power entering outer leg goes to outer wall with plasma and neutral energy flux
- The rest of power entering outer leg is lost with impurity and hydrogen radiation
- Most of power entering divertor leg is lost on its outer wall

# Neutral particles confinement has strong effect on position of detachment front

**Analyzing a representative case:**

- **SXD, $P_{1/2}$=0.6 MW**




**Plasma recycling coef: 100%**

**Neutral albedo: 100%**

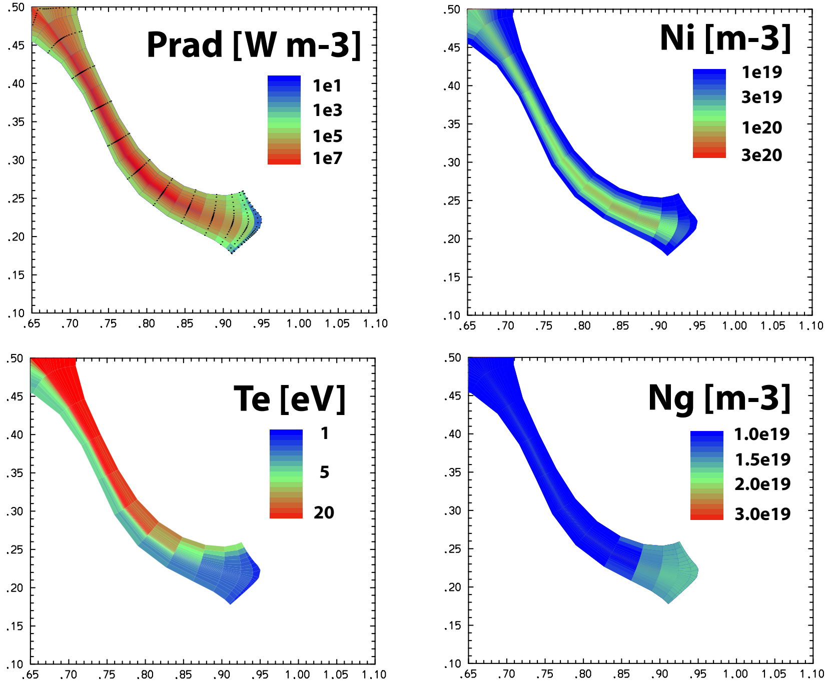
# Neutral particles confinement has strong effect on position of detachment front

**Analyzing a representative case:**

- **SXD, $P_{1/2}$=0.6 MW**

**Plasma recycling coef: 100%**

**Neutral albedo: 99.5%**



# Varying assumptions in the model leads to quantitative changes; but overall picture appears to hold

- **Increasing impurity fraction 1%C to 2%C =>**
  **detachment power threshold $P_{det}$ increases by 10-20%**

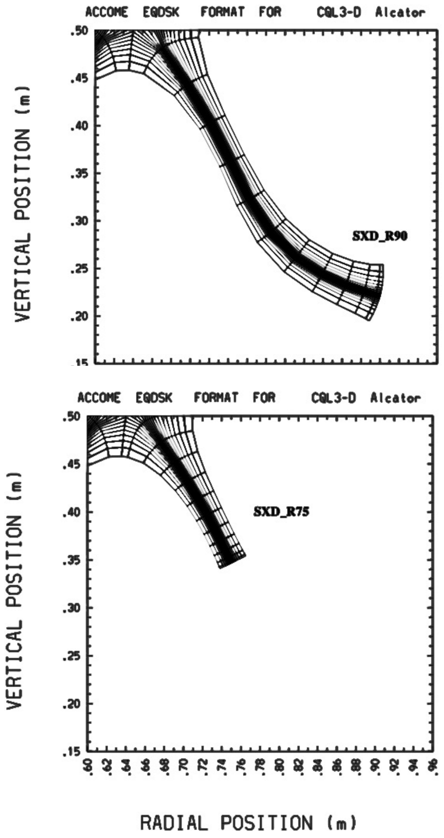
- **Using 1% Ne impurity instead of 1%C =>**
  **$P_{det}$ decreases by ~50%**

- **Setting uniform $\chi_\perp$ =$D_\perp$ =1.0 in outer leg only =>**
  **$P_{det}$ decreases by 50%**

- **Changing Dirichlet boundary conditions to ”extrapolation” at outer wall =>**
  **still seeing similar fully detached divertor solutions**

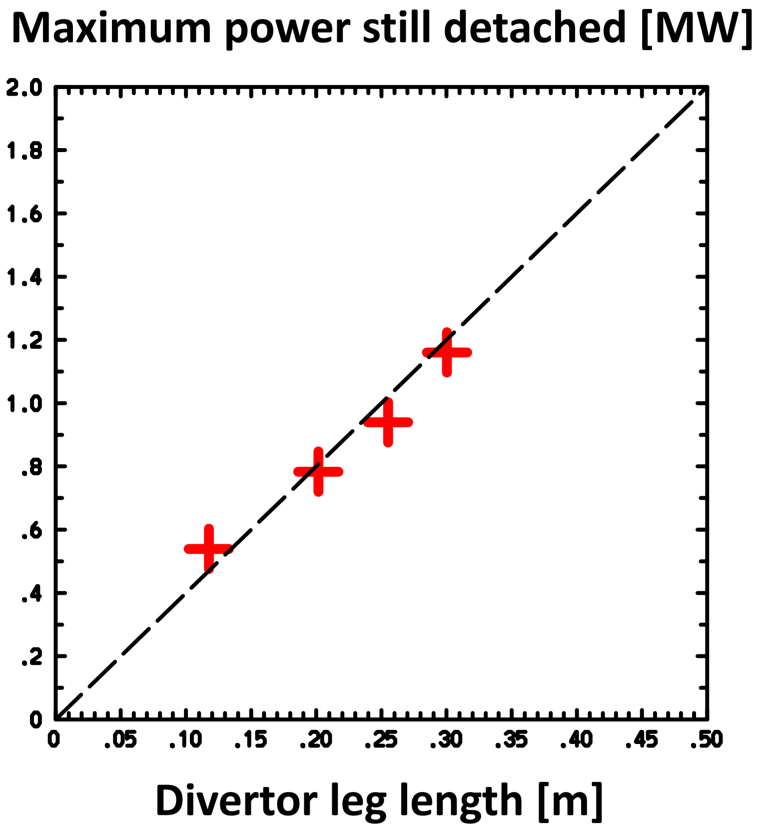

Fusion Energy Sciences

# Power dissipation limit scales linearly with leg length: why?



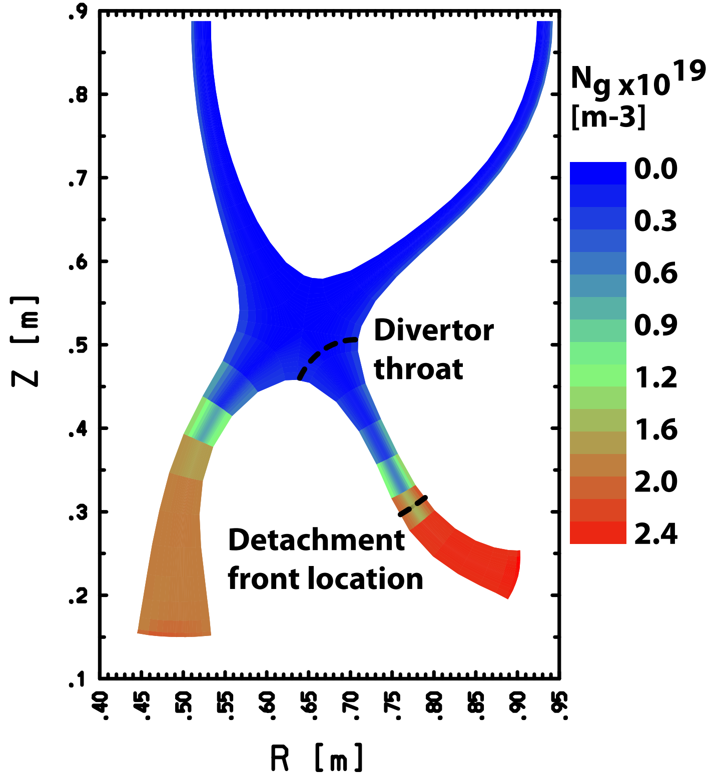

Long leg

Short leg

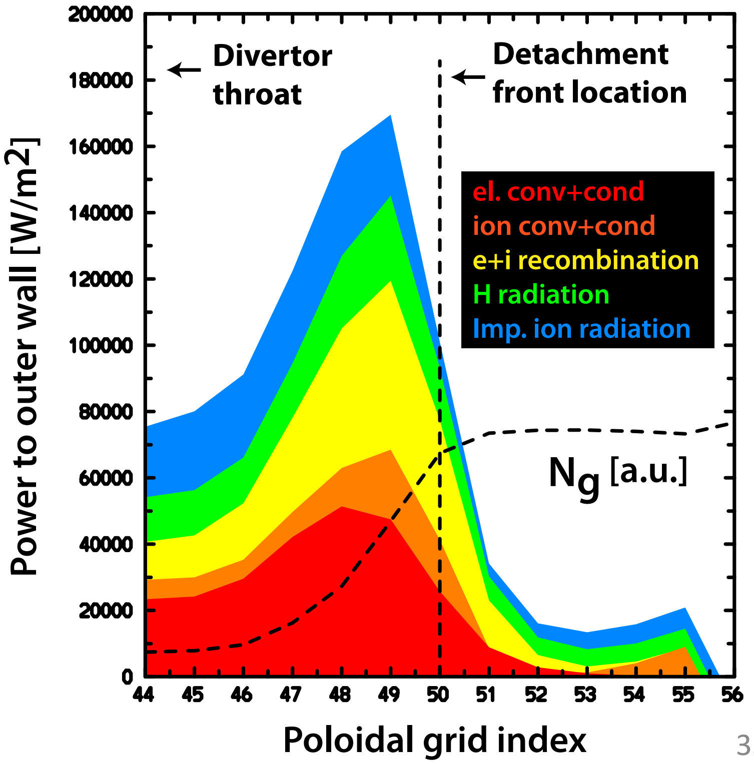


# Energetics provide stability of detachment front: power to wall is small beyond detachment front location

## A representative SXD detached case




## Poloidal distribution of radial power flux



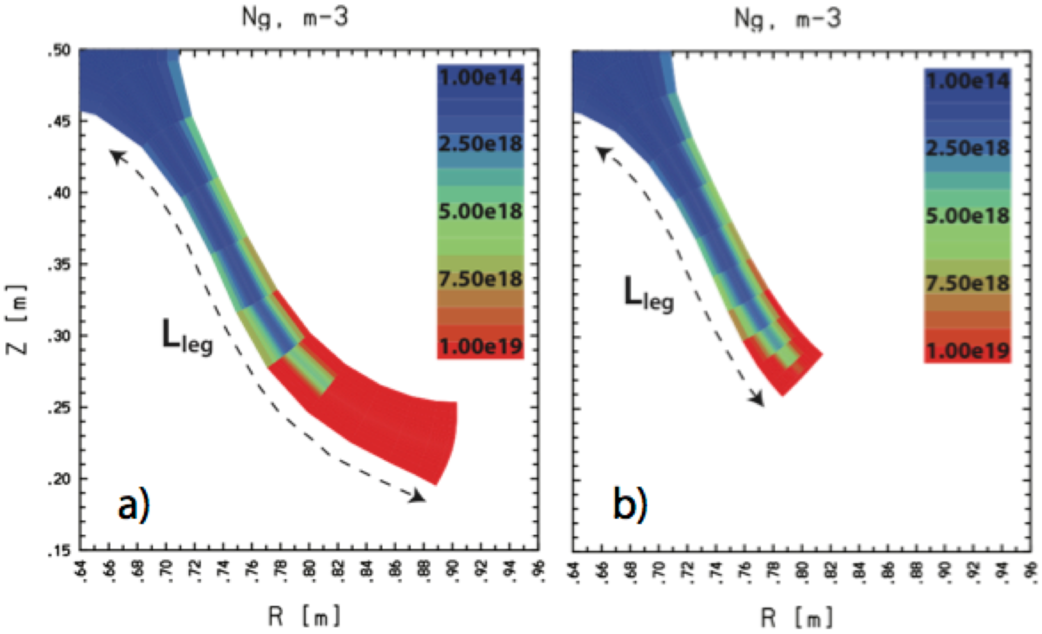



Fusion Energy Sciences

# Amount of power that can be accommodated by divertor leg (still detached) is proportional to leg length

**For same power and other parameters, compare longer leg and shorter leg**

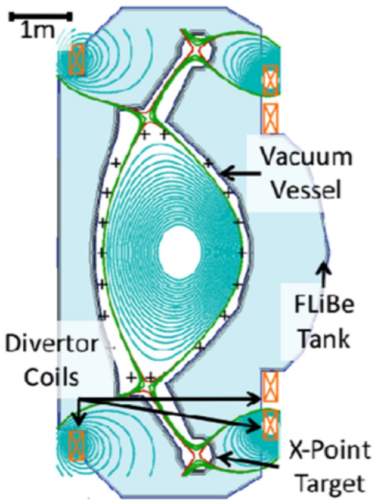
**Longer leg can take more power still staying detached**



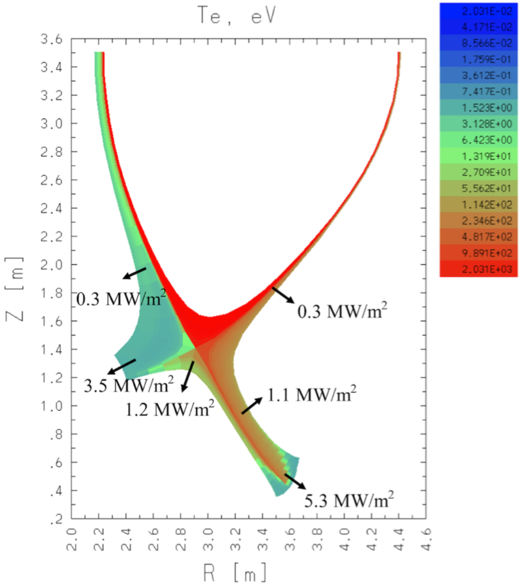

**Shorter leg is already at the limit of power dissipation**

**Main-chamber-recycling physics in long divertor leg => stability of fully detached divertor!**

# Does this regime scale to a reactor? Yes!
# Robust detached long-legged divertor solutions for ARC




From [1]




From [2]

- ARC exhaust power into SOL is projected to be ~100 MW [1]
- In the first modeling study [2], fully detached ARC divertor solution is found for exhaust power 88 MW
- Peak power to PFC ~5 MW/m$^2$
- 0.5% Ne impurity used

[1] Sorbom et al., Fusion Engineering and Design, 100, 378–405 (2015)
[2] Wigram, LaBombard, Umansky et al., CPP (2018)


Fusion Energy Sciences

# Long-legged divertor is attractive for detachment monitoring and control

- **Problem of detached divertor control, particularly in a neutron environment, is unresolved challenge for all traditional divertor solutions**
- **ARIES, DEMO, ITER etc. - no viable fast feedback control scheme identified**
- **For long-legged divertor, detection of detachment front location (with neutron tolerant diagnostics) can be used to monitor and control divertor plasma conditions [1]**
- **Long-legged divertor is potentially a viable robust solution for divertor power handling and control in a reactor, currently considered for high-field designs (ADX, ARC, SPARC)**

---

[1] Brunner et al., Nucl. Fusion 57 86030 (2017)

# Summary and conclusions

- Several tightly baffled long-legged divertor configurations studied with UEDGE for parameters of ADX tokamak design
- Stable steady state fully detached regime found for long-legged divertors
  - Up to 10x higher detachment power threshold than for standard divertor
  - Detachment front stays far away from main X-point, no problem for core
- Key physics for this detached divertor regime combines strong convective plasma transport to outer wall, confinement of neutral gas in divertor volume, geometric effects including secondary X-point, and atomic radiation
- Energetics of PMI in long divertor leg provide stability of detachment front
- With several essential model assumptions varied, the overall picture of stable fully-detached regime in tightly-baffled long-legged divertor still holds
- Long-legged divertor holds promise of stable high-power fully-detached operation

# Backups

# UEDGE* (Unified EDGE code) time-evolves collisional plasma fluid equations in axisymmetric tokamak geometry

plasma density

ion ∥ momentum

electron thermal energy

ion thermal energy

neutral density

ad-hoc radial transport

neutral ∥ momentum

charge conservation

sheath bound. cond.

$$\frac{\partial}{\partial t}(n_i) \quad + \quad \nabla \bullet (n_i \vec{u}_i) = -S_r + S_i$$

$$n_i u_\perp = -D_{\perp i} \nabla_\perp n_i$$

$$\frac{\partial}{\partial t}(m n_i u_{\parallel i}) \quad + \quad \nabla \bullet (m n_i u_{\parallel i} \vec{u}_i - \eta_i \nabla u_{\parallel i}) = -\nabla_\parallel P_i + m n_N n_i K_{cx}(u_{\parallel N} - u_{\parallel i}) + m S_r u_{\parallel N} - m S_i u_{\parallel N}$$

$$\frac{\partial}{\partial t}(3/2 n_e T_e) \quad + \quad \nabla \bullet (\frac{5}{2} n_e T_e \vec{u}_e + \vec{q}_e) = \vec{u}_e \bullet \nabla(3/2 n_e T_e) - \Pi_e \bullet \nabla \vec{u}_e + Q_e$$

$$\frac{\partial}{\partial t}(3/2 n_i T_i) \quad + \quad \nabla \bullet (\frac{5}{2} n_i T_i \vec{u}_i + \vec{q}_i) = \vec{u}_i \bullet \nabla(3/2 n_i T_i) - \Pi_i \bullet \nabla \vec{u}_i + Q_i$$

$$q_\perp = -n\, \chi_\perp \nabla_\perp T$$

$$\frac{\partial}{\partial t}(n_N) \quad + \quad \nabla \bullet (n_N \vec{u}_N) = S_r - S_i$$

$$n_N u_{\perp N} = -D_{\perp N} \nabla_\perp n_N$$

$$\frac{\partial}{\partial t}(m n_N u_{\parallel N}) \quad + \quad \nabla \bullet (m n_N u_{\parallel N} \vec{u}_N - \eta_N \nabla u_{\parallel N}) = -\nabla_\parallel P_N - m n_N n_i K_{cx}(u_{\parallel N} - u_{\parallel i}) - m S_r u_{\parallel N} + m S_i u_{\parallel N}$$

$$\nabla \bullet J(\phi) = 0$$

$$J_\parallel = \frac{en}{0.51 m\nu} \frac{B_x}{B} \left( \frac{1}{n} \frac{\partial P_e}{\partial x} - e \frac{\partial \phi}{\partial x} + 0.71 \frac{\partial T_e}{\partial x} \right)$$

$$J_r = \sigma_\perp E_r$$

$$\phi = \frac{-Te}{e} \ln \left[ 2\sqrt{\pi} \left( \frac{J_\parallel - enu_{\parallel i}}{env_{te}} \right) \right]$$

*T. D. Rognlien et al., J. Nucl. Mater. 196–198, 347 (1992)



Fusion Energy Sciences

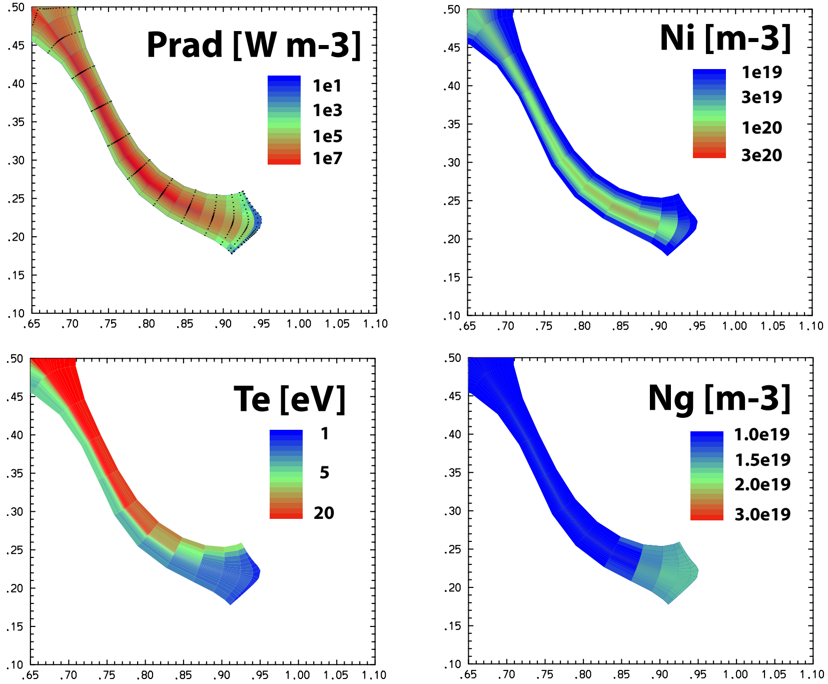
# Neutral particles confinement has strong effect on position of detachment front

**Analyzing a representative case:**

- SXD, $P_{1/2}$=0.6 MW

**Plasma recycling coef: 100%**

**Neutral albedo: 99.5%**




- **Similar results if reducing plasma recycling coefficient**
- **Neutral particles confinement in the leg appears to control detachment front location**



Fusion Energy Sciences