Radiative divertor simulation and advanced divertor study for Demo tokamak reactor

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1. Introduction: power handling in DEMO reactor

**Power handling** is the most important issue for reactor design.

*Demo reactor aiming* $P_{el} = 1GW \Rightarrow P_{FP}$ of *3GW level* $[P_{heat}(\alpha + \text{ad.}) = 600-700 \text{ MW}]$

$\Rightarrow$ Power exhaust to SOL ($P_{out}$) is 5-6 times larger.

$\Rightarrow$ Radiation loss ($P_{rad}$) at the edge and divertor is required 10 times larger.

Energy (plasma, radiation, neutral) dissipation in the divertor to reduce peak $q_{target}$

**SlimCS Demo tokamak for** $P_{fus} = 2.95 \text{ GW}[1]$

<table>
<thead>
<tr>
<th>Power handling</th>
<th>SlimCS</th>
<th>ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td>$P_{heat}(\alpha + \text{ad.})$ [MW]</td>
<td>650</td>
<td>150</td>
</tr>
<tr>
<td>$P_{out} = P_{heat} - P_{rad}^{\text{core}}$ [MW]</td>
<td>550</td>
<td>100</td>
</tr>
<tr>
<td>$P_{out}/R_p$ [MW/m]</td>
<td>100</td>
<td>16</td>
</tr>
<tr>
<td>$P_{div} = (P_{out} - P_{rad}^{\text{div&amp;edge}})$ &lt; 50</td>
<td>~50</td>
<td>~50</td>
</tr>
<tr>
<td>$P_{rad}^{\text{div&amp;edge}}$ [MW]</td>
<td>~500</td>
<td>~50</td>
</tr>
<tr>
<td>$P_{rad}^{\text{div&amp;edge}}/P_{out}$ &gt; 90%</td>
<td>~50%</td>
<td>~50%</td>
</tr>
</tbody>
</table>

- Major radius : $R_p=5.5 \text{ m}$
- Minor radius : $a_p=2.1 \text{ m}$
- Plasma current : $I_p=16.7 \text{ MA}$
- Toroidal field : $B_t=6.0 \text{ T}$
- Plasma volume : $V_p=941 \text{ m}^3$

DEMO divertor design study for huge power handling

- Long divertor leg
- Increase *Wet area* with shallow target angle

<table>
<thead>
<tr>
<th>Design parameters</th>
<th>ITER</th>
<th>SlimCS</th>
</tr>
</thead>
<tbody>
<tr>
<td>$f^{\text{exp}}_{\text{div}}$ (flux expansion)</td>
<td>4</td>
<td>3.7</td>
</tr>
<tr>
<td>$L_{\text{div}}$ (outer leg length)</td>
<td>1.05m</td>
<td>1.73m</td>
</tr>
<tr>
<td>$\theta_{\text{div}}$ (target pol. angle)</td>
<td>24°</td>
<td>18°</td>
</tr>
<tr>
<td>$A$ (wet area $\lambda_{q \text{mid}}=5\text{mm}$)</td>
<td>1.9 m²</td>
<td>3.0 m²</td>
</tr>
</tbody>
</table>

⇒ increase radiation and recombination (dissipation) upstream of the target.

- Impurity (Ar) seeding
  ⇒ enhance edge and divertor radiation.

- V-shaped corner
  ⇒ further enhance neutral and impurity recycling near the strike-point.

SONIC code incl. MC impurity transport self-consistently simulates radiation distribution and plasma detachment, for the first time, under DEMO divertor condition.

Recent design study:

- Seeding impurity selection: radiation distribution and imp. screening

- Longer leg design (in geometry studies): further energy dissipation in divertor Ref.6

- Effects of plasma diffusion (transport studies): detachment and energy dissipation
SONIC: self-consistent coupling with impurity Monte Carlo has been developed for the divertor simulation

MC impurity simulation has advantage:
most impurity transport processes are incorporated in original formula such as
- Radiation & Recombination at multi-charge states
- Kinetic effect $\Rightarrow$ Thermal force (temperature gradient force)

SONIC: self-consistent coupling with impurity Monte Carlo

Seeding impurity transport
Ionization processes: ionization, charge exchange
Surface/volume recombination

Ion transport processes: thermal force – friction force
$\text{Ar}^0$ atom dynamics

Development:

Simulation for SlimCS Demo divertor:
Input parameters: \( P_{\text{out}} = 500 \text{ MW}, \ n_e = 7\times10^{19} \text{ m}^{-3} \ (r/a=0.95), \ \chi_i = \chi_e = 1 \text{ m}^2\text{s}^{-1}, \ D = 0.3 \text{ m}^2\text{s}^{-1} \)  

(same diffusion coefficients for ITER simulation[7])

- Conversion of plasma solution was rather unstable/oscillating under Demo condition due to large \( q_{\parallel} \) and high \( T_e \) in SOL, and high flux and low \( T_{\text{div}} \) in divertor.

- Conversion became stable by recent improvements of SONIC simulation, 
  (1) using distribution of impurity atom and particle balance under the divertor (exhaust route), which were calculated up to a steady-state condition (see below),  
  \[ \Rightarrow \text{ particle balance calculation was incorporated self-consistently in SONIC V3,} \]

  (2) smoothing source terms near the target (by shorter time-step calculations)  
  to reduce the MC noise,  

  (3) correcting thermal force term for long mean-free-path (at high \( T_i \)), etc.

Self-consistent coupling solution of the fluid plasma, MC neutral and impurity was obtained “in steady-state”.

Ar backflow from exhaust slots to the divertor plasma was handled as gas puff:  
\[ \Gamma_{\text{back}}^{\text{Ar (in)}} = 9\times10^{21} \text{ and } \Gamma_{\text{back}}^{\text{Ar (out)}} = 14\times10^{21} \text{ Ar/s.} \]
2. Impurity seeding and Long-leg for Demo power handling

Under Demo edge condition, $T_{e}^{SOL} = 300-400$ eV, $T_{e}^{Edge} = $ a few keV are expected.

- Noble impurities radiate photon efficiently, enhancing at high $T_e (> 100$ eV) with $Z$.

⇒ appropriate radiators for Demo, which requires large radiation at SOL and edge.

- Long leg divertor ($L_{//}=L_{div}B_{//}/B_p$) decreases $T_{div} \propto q_{//}^{10/7}/n_u^2L_{//}^{4/7}$ (from 2-point model), and enhances particle & impurity recycling and produces detachment efficiently.

⇒ appropriate length to produce full detachment and dissipation to reduce peak $q_{target}$.

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**Radiation loss rate coefficients (coronal equilibrium)**

![Graph showing radiation loss rate coefficients](image)

**$T_e$ distribution in Long leg divertor**

![Graph showing $T_e$ distribution](image)

ITER physics Basis, Chap. 4, Nucl. Fusion, 39 (1999) 2391
2.1 Detachment in high radiative divertor

$P_{\text{rad}}^{\text{tot}}/P_{\text{out}}$ is increased to $\sim92\%$ ($P_{\text{rad}}^{\text{tot}} = 460\text{MW}$) by Ar puff rate of $1.45\times10^{21}\text{ s}^{-1}$, radiation is distributed at edge and divertor: $P_{\text{rad}}^{\text{div}}/P_{\text{out}} = 61\%$, $P_{\text{rad}}^{\text{edge}\&\text{SOL}}/P_{\text{out}} = 31\%$.

Partial detachment ($T_e < 1-2\text{ eV}$) is seen near the outer strike-point ($< 5\text{ cm}$), and Plasma is attached ($T_e = 10\text{ eV}$, $T_i = 80\text{ eV}$) at outer flux surfaces due to low density and low collisionality.

**Ar seeding**

**$T_e$, $n_i$ distributions in outer divertor**

Distance from separatrix (m)
Power load profile at the outer target in detached divertor

- Plasma heat load (cond. + conv. + surface rec.) is reduced to ~9MWm$^{-2}$

- Radiation power load is large (3-5 MWm$^{-2}$) over a wide area in the divertor

$\Rightarrow$ peak heat load ($q_{\text{target}}$) is ~16 MWm$^{-2}$ due to radiation source near above target.

**Radiation distribution**

- Total heat load on the target:
  $$q_{\text{target}} = \gamma n_d C_{sd} T_d + n_d C_{sd} E_{ion} + f_1(P_{\text{rad}}) + f_2 \left( \frac{1}{2} m v_0^2 n_0 v_0 \right)$$

- Plasma transport (conduction/convection of electron & ion)

- Surface-recombination

- Radiation power load

- Neutral load

- Plasma heat load (cond. + conv. + surface rec.)

- Neutral load

- Radiation load

- Surface rec.

- Plasma conv.

- & cond. (ion+el)
Influences of seeding impurity on radiation and target plasma are apparent for Kr.

**All cases:**

\[ P_{\text{rad\,tot}} = 460\text{MW} \]  
\[ (P_{\text{rad\,tot}}/P_{\text{out}} \sim 92\%) \]

| puff | \(10^{21}\) atm/s |
| Ne   | 3.8 |
| Ar   | 1.5 |
| Kr   | 0.93 |

Detachment width \((T_e < 2\text{eV})\) increases from 4cm (Ne), 5cm (Ar) to 10cm (Kr).
Detachment is efficient (high-Z) with decreasing power to divertor

For higher Z seeding, $P_{\text{rad}}^{\text{edge+SOL}}$ increases from 86MW (Ne), 153MW (Ar), 229MW (Kr), where $P_{\text{rad}}^{\text{edge}} = 39$MW(Ne), 53MW(Ar), 121MW(Kr).

⇒ Peak heat load appears at the detach boundary, and it is reduced with plasma heat load ($T_e$ & $T_i$).

$P_{\text{rad}}^{\text{div-OUT}}$ distribution changes from “near separatrix” (Ne) to “wide above target”(Kr), while $P_{\text{rad}}^{\text{div-OUT}}$ slightly decreases.

⇒ Radiation power load increases from ~2 MW/m$^2$ to 4-5 MW/m$^2$ over wide target area.
Impurity transport in SOL and edge

Impurity transport determines $n_z$ distribution $\Rightarrow P_{\text{rad div-OUT}}, P_{\text{rad div-IN}}$ and $P_{\text{rad div-SOL}}$, which are also determined by plasma transport $(n_e, T_e, T_i, v_i)$ and rad.-process $L(T_e)$.

- Impurity screening for Ne, Ar, Kr is comparable $[(n_z/n_i)_{\text{div}}/(n_z/n_i)_{\text{SOL}}] \approx 1.5-2$.
- Concentration $(n_z/n_i)_{\text{SOL}} \approx 0.1(\text{Ne}), 0.02(\text{Ar}), 0.01(\text{Kr})$, is determined by that in divertor. Issue of the core plasma dilution remains for all.

### (MW) Radiation at SOL & edge

- Large cooling rate $L(T_e)$, and Concerning confinement degradation
- Lower $n_e$ & $L(T_e)$

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2.3 Effects of the Long-leg divertor

Divertor leg ($L_{\text{div}}$) is extended from 1.7 to 2.5 m, while flux expansion is reduced: “Long leg divertor” decreases $T_{\text{div}}$ and enhances particle & impurity recycling and produces detachment efficiently.

- Strong radiation region moves upstream, and still stays in the V-shaped corner ⇒ producing full detachment.

Radiation power density in Reference divertor

Radiation power density in Long-leg divertor

Ar puff: $1.5 \times 10^{21} \text{ s}^{-1}$

Ar puff: $0.77 \times 10^{21} \text{ s}^{-1}$
Effects of divertor length on radiation and detachment

- $P_{\text{rad}}^{{\text{div-OUT}}}$ largely increases in full detached divertor, while $P_{\text{rad}}^{{\text{edge+SOL}}}$ is reduced
  $\Rightarrow$ radiation region is extended in the long-leg geometry
- **Peak heat load decreases** from 16 to 12 MW/m$^2$ in full detached divertor:
  *Radiation load* as well as *plasma heat load* decrease significantly.
  $\Rightarrow$ *surface recombination due to low-temperature plasma* and *neutral flux by volume recombination* increase, which may be caused by small flux expansion.

Radiation power

- $P_{\text{rad}}^{{\text{sol+edge}}}$: 18% in long-leg, 31% in reference divertor
- $P_{\text{rad}}^{{\text{div-in}}}$: 18% in long-leg, 25% in reference divertor
- $P_{\text{rad}}^{{\text{div-out}}}$: 56% in long-leg, 36% in reference divertor

<table>
<thead>
<tr>
<th>Long-leg</th>
<th>Ref. Div.</th>
</tr>
</thead>
<tbody>
<tr>
<td>$P_{\text{rad}}^{\text{sol+edge}}$: 18%</td>
<td>31%</td>
</tr>
<tr>
<td>$P_{\text{rad}}^{\text{div-in}}$: 18%</td>
<td>25%</td>
</tr>
<tr>
<td>$P_{\text{rad}}^{\text{div-out}}$: 56%</td>
<td>36%</td>
</tr>
</tbody>
</table>

Plasma & neutrals: 8% in long-leg, 13% in reference divertor

Power load density

- $T_i^{{\text{div}}}$: 6.1 eV, 1.6 eV
- $T_i^{{\text{div}}}$: 4.6 eV, 1.6 eV
- $n_i$: 30 $\times 10^{20}$ m$^{-3}$ at outer target
- $T_e$: 20 eV

Total heat load

- Radiation load
- Plasma conv. & cond. (ion+el)
- Surface rec.

Distance from separatrix (m)
3. Effect of plasma diffusion on detachment

Diffusion coefficients change radially to simulate experiments such as “blob”:

Example-1: at the outer SOL ($r_{\text{mid}} \geq 1.5\text{cm}$: 15 cm at divertor), $\chi = 3 \text{ m}^2/\text{s}$ & $D = 1 \text{ m}^2/\text{s}$

- SOL plasma near the strike-point is influenced by enhancement of diffusion in the outer SOL.
  $\Rightarrow T_i, T_e$ and $n_e$ decrease, radiation region moves upstream
  $\Rightarrow$ peak $q_{\text{target}}$ is $\sim 7\text{MWm}^{-2}$

Plasma profile at midplane

![Plasma profile diagram](image-url)
Particle and energy dissipations enhance the detachment

Example-2: enhancement of particle and energy dissipation in whole SOL & divertor
\( \chi = 2 \text{ m}^2\text{s}^{-1} \) & \( D = 0.6 \text{ m}^2\text{s}^{-1} \) (bouble) \( \Rightarrow \lambda_q^{\text{SOL}} \) is increased only from 2.2 to 2.7mm.

- Diffusion largely affects detachment and heat load profile:
  \( T_i, T_e \) and \( n_e \) decrease, radiation region moves upstream
  \( \Rightarrow \) peak \( q_{\text{target}} \) is \( \sim 5 \text{ MWm}^{-2} \) : full detachment is produced,
  surf. recombination and neutral flux are also reduced.
4. Investigation of “advanced divertor” as new options

Advanced divertors “Super-X divertor” and “Snowflake divertor” have advantage to increase both connection length ($L_{\parallel}$) and wet area ($A_{\text{wet}}$) ⇒ Reduction in peak $q_{\text{target}}$

$q_{\text{target}} = P_{\text{div}} / A_{\text{wet}}$, where $A_{\text{wet}} = f_{\text{exp \ div}} (R_{\text{div}} / R_{\text{mid}}) A_{\text{mid}} / \sin \theta_{\text{div}}$, $f_{\text{exp \ div}} \approx [B_p / B]_{\text{div}} / [B_p / B]_{\text{mid}}$

“Super X divertor”: $f_{\text{exp}}$ and $L_{\parallel}$ are increased from the divertor null to SX null, introducing SX-null: $R_{\text{SX}}$, and Flux ratio of SX-null: $f_{\text{SX}} = (\Psi_{\text{SXD}} - \Psi_{\text{mag}}) / (\Psi_s - \Psi_{\text{mag}})$

“Snowflake divertor”: large $f_{\text{exp}}$ and long $L_{\parallel}$ are produced near SF null ⇒ Enhancement of $P_{\text{rad}}$ and reduction of peak $q_{\parallel}$ are expected in a compact divertor.

- Divertor coil location (inside or outside TFC) and target location are determined.

Restrictions of the poloidal coil location for Demo design:
(1) Minimum number of PFCs
(2) PFC currents and size (inside $I_{\text{PF}} < 15\text{MA}$, 1m outside $I_{\text{PF}} < 50-60\text{MA}$, 2m)
(3) Suitable for sector (Blanket & Divertor) replacement
Magnetic and Divertor geometries are investigated for Super-X divertor (SXD):

- **Flux expansion** is increased from Divertor-null to SX-null
  
  \[ R_{\text{div}} = 6.7m, \quad f_{\text{SX}} = 0.95 \]

  \[ \Rightarrow \text{Divertor opening becomes wider to handle SOL field lines of } r_{\text{mid}} < 5\text{cm}. \]

- **Outer target** (\( R_{\text{div}} = 6.7m \)) is designed outer the Super-X null (\( R_{\text{SX}} = 6m \)), where **Target angle** (\( \theta_{\text{div}} \)) is increased \( \Rightarrow \) further inclination of the target is required

  **Effect of** \( f_{\text{SX}} \) (0.95 to 0.99) **on flux expansion appears near the separatrix**: \( < 1\text{ cm} \).

\[ R_{\text{SX}} = 6m, \quad f_{\text{SX}} = 0.95 \]

\[ f_{\text{SX}} = 0.99 \]

\[ r_{\text{mid}} = 0, 1, 3, 5, 10\text{cm} \]
Divertor coils and Magnetic configurations for SXD and SFD

Key divertor coils for SXD and SFD should be inside TFC, due to restriction of \( I_{pFC} \): engineering issues (PFC shield, feedthrough, appropriate cassette design) should be solved.

\( L_{//} \) is increased near SX- or SF-null (1.4-1.6 larger than conv. div.) for advanced divertors

**SFD**: Divertor size will be compact, compared to the long-leg and SX divertors, but \( f_{exp}^{div} \) and \( A_{wet} \) are smaller than SXD. And more ...

- Divertor geometry and the divertor coil locations are largely modified for SFD.
- Current distribution of SFD coils largely affects the lower plasma shaping.

**Issues for SFD**: Larger currents of the divertor and some CS coils are required, and **Control scenario for the SF-null and plasma shaping must be developed.**

<table>
<thead>
<tr>
<th></th>
<th>Conv. long-leg</th>
<th>SXD Int. TFC</th>
<th>SFD Int. TFC</th>
</tr>
</thead>
<tbody>
<tr>
<td>PFC-7 (MA)</td>
<td>-2 (ext)</td>
<td>+10</td>
<td>+42</td>
</tr>
<tr>
<td>PFC-8 (MA)</td>
<td>+36(ext)</td>
<td>-4</td>
<td>-18</td>
</tr>
<tr>
<td>PFC-9 (MA)</td>
<td>+17(ext)</td>
<td>+18</td>
<td>+6</td>
</tr>
<tr>
<td>( R_{div} ) (at target)</td>
<td>6.4 m</td>
<td>6.7m</td>
<td>6.2m</td>
</tr>
<tr>
<td>( f_{exp}^{div} ) at target</td>
<td>2.4</td>
<td>3.9</td>
<td>2.3</td>
</tr>
<tr>
<td>Wet area, ( A_{wet} )</td>
<td>1.6 m²</td>
<td>1.8m²</td>
<td>1.0m²</td>
</tr>
<tr>
<td>( L_{//} ) (Xp to target)</td>
<td>41m</td>
<td>54m</td>
<td>57m</td>
</tr>
</tbody>
</table>
5. Summary and Future work for Demo power handling (1/2)

- Enhancement of $P_{\text{rad}}^{\text{div}}$ and $P_{\text{rad}}^{\text{SOL+edge}}$ and “Full detachment” are necessary to increase energy (plasma, radiation, neutral) dissipation in the divertor and to reduce peak $q_{\text{target}}$.

**Seeding impurity selection:** higher Z (Ar/Kr) is preferable to increase $P_{\text{rad}}^{\text{SOL}}$
- $P_{\text{rad}}^{\text{edge}}$ restriction due to confinement degradation
- dilution in core plasma

**Longer leg design and divertor geometry study:**
- effective for full detachment and $P_{\text{rad}}^{\text{div}}$ enhancement
- reduction in ion and neutral fluxes
- appropriate size and exhaust slot location

**Plasma (and impurity) diffusion** – large impact on detachment and energy dissipation, suggesting that *global/local enhancement* promotes full detachment.
- database and extrapolation to Demo condition
- development of techniques

- Conceptual design [$P_{FP} = 3\text{GW} \ P_{el} = 1\text{GW}$] is now revised from many viewpoints in BA Demo Design Activity ⇒ Divetor operation window will be investigated for lower $P_{FP}$.

- SONIC is now improved to V3 & systematic scan in Rokkasho CSC (Helios) is planned.

improvement/development of plasma & impurity transport modelling is necessary
- detachment of ion flux
- thermal force in low collisionality SOL
- SOL flow modelling
- coupling with edge transport (plasma&impurity), etc.
5. Summary and Future work for Advanced divertor study

“Advanced divertor” study started to provide new options of magnetic configuration. Magnetic geometry and target location were investigated with minimal number of PFCs (9), using TOSCA equilibrium code.

(1) Super-X divertor (SXD) equilibrium was produced by introducing 2 parameters: SX-null location and Flux ratio of SX-null: $f_{SXD} = (\Psi_{SXD} - \Psi_{mag})/(\Psi_{s} - \Psi_{mag})$

- Connection length from divertor-null to target ($L_{//}$) was increased compared to the conventional long-leg divertor: 1.4 - 1.9 times with $f_{SXD}$ (0.95 to 0.99).

(2) Formation of Snowflake divertor (SFD) equilibrium was developed:

- $L_{//}$ was largely increased near SF null (1.5-1.7 times),
  $\Rightarrow$ SFD is compact compared to conv. and SX divertors, but key issues remain: control scenario for SF-null and plasma shaping should be developed, and appropriate SFD geometry design is necessary.

(3) Inter-TFC divertor coils (engineering issues) are required both for SFD and SXD, $\Rightarrow$ appropriate design for PFC shield, feedthrough and cassette is necessary. coil winding and horizontal maintenance should be developed/improved.

Divertor simulation (SONIC) is developed to calculate “advanced divertor” plasma:

- Plasma detachment and radiation distribution are investigated.
- Divertor geometry appropriate for detachment and pumping is determined.
Code development:

3.1 Backflow model for impurity MC calculation

MC approach for impurity transport modeling
flexibility in modeling $\Leftrightarrow$ long computational time

**Full calculation:** trajectories of injected impurity particles are traced in the plasma and sub-divertor region.

**Backflow model:** amount of the backflow (B, C and D) is evaluated in advance, then simulating impurity flux injected from the exhaust slot to the divertor region like a backflow.

Calculation time is reduced significantly, and iterative calculation of SONIC for DEMO divertor simulation becomes possible.
3.2 Optimization of SONIC code on HELIOS

Calculation time and accuracy are improved by increase in computer cores.
- **wide-range parameter survey**: divertor geometry, impurity species …
- **stable calculation** due to reduction in MC noise

**BX900 (JAEA)**: 40 hours
128PE, 170000 MC test particles

**Helios**: 15 hours
1024 PE, 340000 MC test particles

Calculation time becomes less than half while double of test particles is treated.

3.3 Development of W transport simulation

Improvement of full orbit MC particle simulation code IMPGYRO is in progress to estimate lifetime of W-plasma facing components.

In the preliminary analysis with fixed background solution (partial detachment), net erosion rate was too high (~10^{-7} m/s) compared with the required lifetime up to next maintenance, because of large self-sputtering.

Self-consistent analysis between background and W transport is necessary. Also influence on the core performance has to be estimated.
Development of TOSCA equilibrium code for Super-X divertor

(1) Same location of 9 PFCs (outside TFC) is used
(2) New input parameters for SXD are introduced,
   • Super-X null location \( (R_{SX}, Z_{SX}) \)
   • Ratio of poloidal fluxes at SX-null and separatrix:
     \[ f_{SX} = \frac{(\Psi_{SX} - \Psi_{ax})}{(\Psi_s - \Psi_{ax})} \]

Current distribution \( (I_{PFC}) \) is determined to minimize the following least square errors:

\[
E = \sum_{k=1}^{N_s} W_k (\Psi_k - \Psi_s)^2 + \sum_{j=1}^{N_c} \gamma_j (I_j - I_{Ref,j})^2 + \chi_{STD} \left( \Psi_{STD} - \left( \Psi_{Mag} - f_{STD} (\Psi_{Mag} - \Psi_s) \right) \right)^2
\]

1\textsuperscript{st} & 2\textsuperscript{nd} terms: Flux at selected plasma surface and null
3\textsuperscript{rd} term: Mag. field at divertor & SX null points
4\textsuperscript{th} term: Interlinkage flux (set to \( \Psi_{Ref} = -100 \text{ Vs} \))
6\textsuperscript{th} term: Magnetic flux at SX-null

Weight factors were optimized to obtain better accuracy of SX-null position. \( I_{PFC} \) for Conv. Divertor: +17

Note: size of PFC does not correspond to the current
Equilibrium control and divertor geometry for SFD

Equilibrium control and formation scenario of SFD have been investigated:
(1) from SXD to SFD with $f_{Sx} \sim 0.99 \ (<1)$ --- below
(2) from unbalanced-SFD (SF- or SF+) to balanced-SFD --- not shown

Divertor geometry must be design for SFD&SXD, NOT appropriate for the conventional divertor.

Divertor coil currents must be increased to obtain balanced-SFD -> need SFD coil arrangement.

<table>
<thead>
<tr>
<th>Current (MA)</th>
<th>SXD $R_{Sx}$:6m</th>
<th>SXD $R_{Sx}$:5m</th>
<th>SXD $R_{Sx}$:4.75m</th>
</tr>
</thead>
<tbody>
<tr>
<td>PFC-7</td>
<td>+10</td>
<td>+33</td>
<td>+51</td>
</tr>
<tr>
<td>PFC-8</td>
<td>-4</td>
<td>-23</td>
<td>-36</td>
</tr>
<tr>
<td>PFC-9</td>
<td>+17</td>
<td>+32</td>
<td>+39</td>
</tr>
</tbody>
</table>

Divertor coil arrangement for SXD with $R_{Sx}$=6m

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*Diagram showing divertor coil arrangements for different $R_{Sx}$ values.*
Future studies of Demo advanced divertor

[Next step for divertor physics study]
Divertor simulation (SONIC) is developed to calculate “advanced divertor” plasma:
- Plasma detachment and radiation distribution are investigated.
- Divertor geometry appropriate for detachment and pumping is determined.

[Magnetic equilibrium study]
- Divertor coil locations are optimized to reduce the PFC currents:
  Combination of Inter-PFC and outer-PFC is also considered.
- PFC arrangement for horizontal replacement of sector/blanket/divertor.
- For SFD, control scenario for SF-null and plasma shaping must be developed, and unbalanced SF configurations (SF+ and SF-) are also investigated.

[Engineering issues]
- PFC design for shield and feedthrough is investigated for inter-PFC.
- Super-Conductors and Winding procedure should be determined.
- Design work of divertor geometry and target & cassette structure and both for SXD and SFD are necessary, compatible with the divertor & sector maintenance.