Studies of Plasma Exhaust Scenario and Divertor Design for JA DEMO

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1. Introduction:
   DEMO special design team in Japan
   BA DEMO design activity, and power exhaust concepts

2. Power exhaust in JA DEMO, and the divertor simulation

3. Engineering design for JA DEMO divertor

4. Summary, future work and issues
1. DEMO design activity in Japan: **BA DDA, Special design team**

Joint Special Design Team for Fusion DEMO

- **BA DDA**
  - Phase I
  - Common design issues on DEMO
  - Develop DEMO pre-conceptual designs
  - Phase II (tbd)

- **Domestic activity**
  - Special Design Team
    - Overall design of Japan’s DEMO plant
    - Strategy of DEMO incl. cost, resources, and PA

- **Conceptual Design**
  - Pre-conceptual
  - Conceptual

- **Engineering Design**

- **6th Plenary meeting (June 2017)**

- Total 83 members (incl. site & visiting, 2017 June)
  - QST 28, NIFS 3, Univ. 28, NIMS 1, Makers/Industries 23
### Similar size (8-9m), $P_{\text{fusion}}$ (1.7-2GW), $P_{\text{heat}}$ (430-460MW)

**Common missions for JA and EU DEMOs:**
- Electric generation,
- Fuel generation,
- High duty-cycle,
- Remote-maintenance in high neutron dose,
- Safety,
- Plasma control & Power handling in long-pulse/steady-state

<table>
<thead>
<tr>
<th>Parameters</th>
<th>JA DEMO</th>
<th>EU DEMO1</th>
<th>ITER (inductive)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$R_p / a_p$ (m)</td>
<td>8.5 / 2.42</td>
<td>9.1 / 2.94</td>
<td>6.2 / 2.0</td>
</tr>
<tr>
<td>$A$</td>
<td>3.5</td>
<td>3.1</td>
<td>3.1</td>
</tr>
<tr>
<td>$\kappa_{95}$</td>
<td>1.75</td>
<td>1.59</td>
<td>(1.70)</td>
</tr>
<tr>
<td>$q_{95}$</td>
<td>4.1</td>
<td>3.2</td>
<td>3</td>
</tr>
<tr>
<td>$I_p$ (MA)</td>
<td>13.5</td>
<td>19.6</td>
<td>14</td>
</tr>
<tr>
<td>$B_T / B_{T\text{max}}$ (T)</td>
<td>5.94 / 12.1</td>
<td>5.7 / 12.3</td>
<td>5.3 / 12</td>
</tr>
<tr>
<td>Operation</td>
<td>Steady-state</td>
<td>Pulsed 2 hrs</td>
<td>~400 s</td>
</tr>
<tr>
<td>$P_{\text{fusion}}$ (MW)</td>
<td>1694</td>
<td>2037</td>
<td>500</td>
</tr>
<tr>
<td>$P_{\text{gross}}$ (MWe)</td>
<td>588</td>
<td>914</td>
<td>--</td>
</tr>
<tr>
<td>$P_{\text{aux}}$ (MW)</td>
<td>96</td>
<td>50</td>
<td>73</td>
</tr>
<tr>
<td>$Q$</td>
<td>18</td>
<td>41</td>
<td>10</td>
</tr>
<tr>
<td>$P_{\alpha} + P_{\text{aux}} (=P_{\text{heat}})$ (MW)</td>
<td>435</td>
<td>457</td>
<td>173</td>
</tr>
<tr>
<td>Neutron (MWm$^{-2}$)</td>
<td>~1</td>
<td>~1</td>
<td>0.5</td>
</tr>
<tr>
<td>$HH_{98y2}$</td>
<td>1.3</td>
<td>1.1</td>
<td>1.0</td>
</tr>
<tr>
<td>$\beta_N$</td>
<td>3.4</td>
<td>2.6</td>
<td>1.8</td>
</tr>
<tr>
<td>$f_{BS}$</td>
<td>0.61</td>
<td>0.35</td>
<td>0.15</td>
</tr>
<tr>
<td>$n_e/n^{GW}$</td>
<td>1.2</td>
<td>1.2</td>
<td>~0.9</td>
</tr>
</tbody>
</table>

**Japan DEMO (steady-state)**
(increasing $\kappa_{95}$ and seeding)

**EU DEMO1 (pulsed 2hrs)**
Power exhaust scenario and design concept in JA and EU

Power exhaust scenario for DEMO plasmas and the baseline divertor design have been studied with a high priority in JA and EU.

- EU and JA studies have covered divertor physics and engineering aspects: Water-cooled single-null divertor, operating with Plasma detachment.
- Both concepts require highly radiation ($f_{rad} = P_{rad}/P_{heat}$ $\sim$ 80%) to handle with ITER-level peak heat load (10 MWM$^{-2}$), while balance of $f_{rad}^{main}$ and $f_{rad}^{div}$ is different:

**JA Power exhaust (steady-state):**

*Lower $I_p = 13.5$MA* due to higher $q_{95}$ (4.1), and expecting good plasma performance,

$\implies$ **Large power handling in the divertor:** $P_{sep}/R \sim 30$ MW/m

**EU Power exhaust (pulse 2hours):**

*Higher $I_p = 20$MA*, and $n_e = 8.7 \times 10^{19}$m$^{-3}$,

$\implies$ **Large radiation loss in main plasma using high-Z impurity seeding, to reduce $P_{sep}/R$ to 17 MW/m (ITER-level)**

$\implies$ Two approaches will provide important case-studies for future decision of the divertor design.

<table>
<thead>
<tr>
<th>Parameters</th>
<th>JA DEMO</th>
<th>EU DEMO1</th>
<th>ITER (Q=10)</th>
</tr>
</thead>
<tbody>
<tr>
<td>line-ave. $n_e$ ($10^{19}$m$^{-3}$)</td>
<td>8.5</td>
<td>8.7</td>
<td>$\sim$10</td>
</tr>
<tr>
<td>$n_{imp}/n_e$ (%)</td>
<td>0.60 (Ar)</td>
<td>0.039 (Xe)</td>
<td>N$_2$, Ne, Ar...</td>
</tr>
<tr>
<td>$P_{\alpha}+P_{aux} (=P_{heat}$ MW)</td>
<td>435</td>
<td>457</td>
<td>173</td>
</tr>
<tr>
<td>$P_{rad}^{main}$ (MW)</td>
<td>177</td>
<td>306</td>
<td>$\sim$70</td>
</tr>
<tr>
<td>$P_{rad}^{main}/P_{heat}$</td>
<td>0.41</td>
<td>0.67</td>
<td>0.40</td>
</tr>
<tr>
<td>$P_{sep}$ (MW)</td>
<td>258</td>
<td>154</td>
<td>$\sim$100</td>
</tr>
<tr>
<td>$P_{sep}/R_p$ (MWm$^{-1}$)</td>
<td>30</td>
<td>17</td>
<td>16</td>
</tr>
</tbody>
</table>

Common and Complemental works were summarized in Asakura, et al. ISFNT-13 2017.

System code predictions (JA:TPC, EU:PROCESS)
2. Power exhaust and Impurity seeding in JA DEMO scenario

Impurity seeding into JA DEMO 2014 ($I_p = 12\,\text{MA}$, $\kappa_{95} = 1.65$, $P_{\text{fusion}} \sim 1.5\,\text{MW}$, $H_{H98(y2)} = 1.3$):

- Fusion power ($P_{\text{fusion}}$) was reduced $<1.5\,\text{GW}$ due to fuel dilution.
- Higher $H_{H98(y2)} (>1.4)$ was required to maintain $W_{\text{th}}$ and $\beta_N$.

$\kappa_{95} = 1.75$ is proposed to increase $I_p = 13.5\,\text{MA}$, $\langle n_e \rangle = 7.2 \times 10^{19}\,\text{m}^{-3}$, $P_{\text{fusion}} > 1.5\,\text{GW}$

$\Rightarrow$ Conducting shell design (vertical stability control) is improved (Utoh, ISFNT-13, 2017)

Power exhaust by Ar seeding:
increasing $n_{\text{Ar}}/n_e = 0.5\text{-}0.75$

$\Rightarrow P_{\text{rad\,main}} = 150\text{-}205\,\text{MW}$ and $P_{\text{fusion}} = 1.75\text{-}1.65\,\text{GW}$

$f_{\text{rad\,main}} = P_{\text{rad\,main}}/P_{\text{heat}} = 0.35\text{-}0.45$ (ITER-level)

$\Rightarrow P_{\text{sep}} = 285\text{-}205\,\text{MW}$, i.e $P_{\text{sep}}/R = 34\text{-}25\,\text{MW/m}$

Large power handling is required for JA divertor, while $H_{H98(y2)} \sim 1.3$, $\beta_N = 3.4$, $P_{\text{fusion}} > 1.5\,\text{GW}$ is maintained.

(assuming $n_{\text{He}}/n_e = 7\%$, $n_{\text{W}}/n_e = 10^{-5}$)

Note: L-H $P_{\text{th}}$ (Martin, J. Phys.: Conf. Ser. 2008):
need database of H to L threshold power at $Z_{\text{eff}} > 2\text{-}3$, and $P_{\text{th}}$ at $f_{\text{GW}} > 1$? (Huber, Nucl. Metter. Energy 2017)
**Divertor physics study by SONIC simulations**

**SONIC simulation:** self-consistently solved by iteration of plasma fluid (SORDOL), neutral (D) MC (NEUT2D) and impurity (Ar) MC (IMPMC) codes

- At core-edge boundary $r/a=0.95$: exhausted power ($P_{out} = 250$ MW), particle ($\Gamma_{out}^{D+} = 1 \times 10^{22}$ s$^{-1}$)
- Transport coefficients: $\chi_e=\chi_i=1$ m$^2$/s, $D=0.3$ m$^2$/s (same as ITER calc.), Drifts are not incorporated.
- Covering the connecting SOL: $r_{mid} < 3.2$ cm.

**Long leg divertor:** $L_{div-out} = 1.6$ and 2.0 m was investigated to determine the size

$T_e^{sep}$ & $T_i^{sep}$ are increased: 2-3 times larger than ITER
$\Rightarrow \lambda_{q//}$ near-sep. becomes small while $\chi$ is the same:
$\lambda_{q//} = 1.9$ mm is still larger than Eich’s scaling (1.3 mm). $\chi$ & D to 1/2-1/4 (Kikushkin JNM 2013) is in progress.

Optimization of Ar seeding rate at $P_{rad}^{\text{edge+sol+div}}/P_{out} = 0.8$

$n_e^{\text{sep}}$ is increased (1.8 to $2.4 \times 10^{19} \text{m}^{-3}$), Ar puff rate is minimized (4.2 to $3.3 \times 10^{20}/\text{s}$), where radiation power fraction, $(P_{rad}^{\text{edge}}+P_{rad}^{\text{sol}}+P_{rad}^{\text{div}})/P_{out} = 0.8$, is fixed.

$\Rightarrow P_{rad}^{\text{sol}}$ is increased from 26 to 36MW, and $P_{rad}^{\text{div}}$ is decreased from 162 to 150MW.

Inner target: Total power $P_{\text{target}} = 52-66 \text{ MW}$, and peak $q_{\text{target}} = 5-8 \text{ MW/m}^2$

Outer target: $P_{\text{target}}$ is reduced from 80 to 60 MW and peak $q_{\text{target}}$ from 8 to 4 MW/m$^2$, since the plasma detachment is extended to upstream and wider (next).

$n_{Ar}/n_e$ in SOL decreases to 0.42% (comparable to main $n_{Ar}/n_e = 0.5-0.7\%$: system code), imp. shielding, $(n_{Ar}/n_e)^{\text{div}}/(n_{Ar}/n_e)^{\text{SOL}}$, is improved $\Rightarrow$ appropriate fuel & Ar puff rates.
Power exhaust for \( P_{\text{rad}}^{\text{edge+sol+div}} / P_{\text{out}} = 0.8 \), \( D_2 \) puff: 100Pam³/s

\[ \text{Power to divertors } (P_{\text{div}}) : 202 \text{MW} (\geq 2x\text{ITER}) \]
\[ \text{Divertor radiation } (P_{\text{rad}}^{\text{div}}) : 153 \text{MW} (\sim 3x\text{ITER}) \]

inner target: Full detachment \( (T_{e,i} \sim 1 \text{eV}) \)
outer target: Partial detachment \( (r < 14 \text{cm}) \)

\[ \frac{P_{\text{rad}}^{\text{tot}}}{P_{\text{heat}}} = 0.84 \] (system code & SONIC)

\[ P_{\text{rad}}\text{main} = 180 \text{MW} \]
\[ (P_{\text{rad}}\text{main} / P_{\text{heat}} = 0.41) \]

\[ P_{\text{sep}} = 235 \text{MW} \]

\[ P_{\text{heat}} (= P_{\alpha} + P_{\text{add}}) = 430 \text{MW} \]

\[ P_{\text{rad}}\text{SOL} = 33 \text{MW} \]

\[ P_{\text{div}} = 202 \text{MW} \]
\[ (P_{\text{rad}}\text{SOL+div} / P_{\text{heat}} = 0.43) \]

\[ P_{\text{rad}}\text{div} = 153 \text{MW} \]

Target heat load

Radiation and electron temperature profiles for \( L_{\text{div}} = 1.6 \text{m divertor case} \) \( (D \text{ puff: 100Pam}^3/s) \)

\[ P_{\text{rad}}\text{div-In} = 72 \text{MW} \]
\[ P_{\text{rad}}\text{div-Out} = 81 \text{MW} \]

\[ W_{\text{rad}} \]

inner target: Full detachment \( (T_{e,i} \sim 1 \text{eV}) \)
outer target: Partial detachment \( (r < 14 \text{cm}) \)
Control of radial peak and detachment: Simulation vs Experiments

- Control of the strong radiation (detachment) in the divertor leg ($L_{\text{div}} > 1\text{m ITER-level}$) is necessary to propose the power exhaust concept and divertor design.

- In DEMO (and ITER) with high $T_e$ and $q_{\text{||}}$, radiation peak can be maintain in the divertor.

- Pressure loss at the detachment is small: $n(T_e+T_i) + m_i n_i (V_i)^2$: (Xp)$3.4\times10^3$ -> (div) $1.6\times10^3\text{Pa}$

Radiation is enhanced near the X-point and detachment expands at the target $\Rightarrow$ reducing in $T_e^{\text{ped}}$, $H_{H98(y2)}$

$P_{\text{rad}}^{\text{div}}/P_{\text{heat}} > 0.6$

N seeding:
- $H_{98} = 0.9$
- $f_{\text{rad}} \leq 90%$
- Type-III ELMs

Type-I/no ELMs

Ar seeding:
- $H_{98} = 0.7$
- $f_{\text{rad}} \leq 75%$

Kr seeding:
- $H_{98} = 0.65$
- $f_{\text{rad}} \leq 60%$

H/L transitions

M. Bernert, Nucl. Mater. Energy 2017
Detachment plasma at inner and outer targets

**Inner target:** plasma heat load is reduced, while ionization still occurs at $T_e = 1\text{-}2$ eV.
\[ \Rightarrow \text{peak } q_{\text{target}} \sim 5 \text{ MW/m}^2: \text{ surface recombination load is a dominant component.} \]

**Outer target:** “partial” detachment is produced in $r_{\text{target}} < 14\text{ cm}$.
\[ \Rightarrow \text{peak } q_{\text{target}} \sim 5 \text{ MW/m}^2: \text{ plasma heat load is dominant at the attached region, and radiation load is also large due to significant radiation loss near the target (above } < 10 \text{ cm).} \]

**Note:** $W$-target erosion in the outer attached area is an issue for Steady-State operation.
Lower $P_{rad}$ case: $P_{rad \text{ edge+sol+div}} / P_{out} = 0.75$ \hspace{1cm} ($P_{rad \text{ tot}} / P_{heat} \sim 0.8$)

Ar seeding in $P_{out} = 250\text{MW}$, $f_{rad} = 0.75$, i.e. increasing $P_{out} - P_{rad}$ from 50 to 62.5MW

Ar puff rate is decreased (4 to $3 \times 10^{20} / \text{s}$) $\Rightarrow$ $P_{sep} \sim 240\text{MW}$ is comparable, $P_{rad \text{ div}} = 150\text{MW}$ is lower

Larger $q_{target}$ (due to surface recom.) is seen at the inner target?

- need improvement of A&M and/or elastic-collision processes
- in-out asymmetry in power distribution may be required.

Outer target: narrower “partial” detachment region ($r_{target} < 10\text{cm}$).
$\Rightarrow$ peak $q_{target} \sim 6.5 \text{ MW/m}^2$:
$T_{e \text{ div}}$ and $T_{i \text{ div}}$ at the attached region became higher.

Inner target: ionization still occurs at $T_e = 1-2\text{ eV}$.
$\Rightarrow$ peak $q_{target} \sim 8 \text{ MW/m}^2$:
surface recombination load is dominant.
3. Engineering design for JA DEMO divertor

Heat sink unit and cooling-pipe routing in a cassette structure are investigated:

(1) Remote maintenance: one cassette covers 7.5° toroidal area
   ⇒ 3 cassettes are replaced from 1 port (total 16 ports and TFCs) [Utoh, et al FED 2015]

(2) Fuel/He exhaust, pumping route
   ⇒ private pumping, opening at the bottom (tentative)

(3) Power exhaust design under n-irradiation
   ⇒ Application of ITER technology is investigated:
     W-monoblock & Cu-alloy cooling pipe
     ⇒ Cooling-pipe arrangement in a cassette
     ⇒ Analysis of the heat sink of ~10 MWm⁻²

(4) Cassette design and Replacement
Design concept of W and CuCrZr/F82H unit in neutron condition

- Neutronics analysis (MCNP-5 code with FENDL-2.1 nuclear database) shows W-nonoblock & CuCrZr heat sink can be applied at high heat flux and low n-flux area: ~1 dpa/fpy (inner) and ~1.5 dpa/fpy (outer) on CuCrZr pipe near the strike points.

ITER technology (W/CuCrZr-target) can be applied (ITER life time dose on W: 0.54dpa, CuCrZr: 2.5 dpa), while the replacement is 1-2 years in DEMO condition, due to increasing operation time as well as the neutron flux.
Design concept of the water-cooling routing (2015)

- Heat removal concept of W-monoblock & CuCrZr/F82H cooling-pipes was designed for large margin case on the target ($P_{\text{div,thermal}}^\text{thermal}: 380 \text{MW} + P_{\text{div,neutron}}^\text{neutron}: 118 \text{ MW}$), where $P_{\text{div,thermal}}$ is 1.7 times larger than SONIC simulation ($P_{\text{div,thermal}} = P_{\text{sep}} - P_{\text{rad}}^{\text{SOL}} = 220\text{MW}$).

- Pressurized water for the CuCrZr-pipe: 200°C, 5 Mpa, and for the F82H-pipe: 290°C, 15 MPa (similar to pressurized water reactor: PWR) is used in the different routes.

Note: PWR water will be used for the electric generation by turbine system similar to a PWR.

Asakura et al. Nucl. Fusion 2017
Heat analysis in W-monoblock & CuCrZr-pipe heat sink

- Base-Temp. (200 °C) of the pressurized water and Nuclear heat larger than ITER.
- Heat load profiles (plasma, radiation, neutral) with the peak heat load 10MWM⁻², are used for 3D FEM calculation of heat flux and thermal stress.
- Max. temperatures of W-surface 1021°C (lower than recrystalization Temp. 1200°C), and Cu-alloy-pipe 331°C are within operation range. Total peak heat load: 10MWM⁻² + Nuclear Heat

ABAQUS 3D FEM analysis of W/Cu-alloy (four) monoblock unit:
Heat transport in W and CuCrZr pipe, and max. heat flux

- **Max. heat flux to coolant:** $18\text{MWm}^{-2}$ is $2/3$ of the Critical heat flux ($27\text{MWm}^{-2}$).
- **Heat flux to cooling pipe** is localized at side surface: $25\text{MWm}^{-2} \Rightarrow$ acceptable level
- **Thermal stress** is increased at *upper inner surface* of CuCrZr-pipe by elastic analysis: maximum stress-strain is evaluated with *elasto-plastic stress analysis*, considering residual stress-strain after the braze process from $950^\circ$: not apparent in $10\ \text{MWm}^{-3}$.
Summary 1: Power exhaust and divertor design for JA DEMO

**Power handling with Ar seeding** has been investigated in steady-state JA DEMO: \( P_{\text{heat}} = P_\alpha + P_{\text{add}} \approx 435 \text{MW}, R_p = 8.5 \text{m}, P_{\text{sep}} \approx 258 \text{MW}, P_{\text{sep}}/R \approx 30 \text{MW/m}, \) where \( n_{\text{Ar}}/n_e \approx 0.6\% \).

The concept was revised to increase \( \kappa_{95} \) from 1.65 to 1.75 (and increase \( I_p \)), and to improve the plasma performance for the impurity seeding (\( P_{\text{rad}}^{\text{main}}/P_{\text{heat}} \approx 0.4 \)).

(1) **Divertor (larger than ITER)** has been investigated using SONIC with Ar seeding.

Increasing \( P_{\text{rad}}^{\text{SOL+div}}/P_{\text{heat}} = 0.43 \) in long-leg divertor (\( L_{\text{leg}} = 1.6 \text{m} \)) \( \Rightarrow (n_{\text{Ar}}/n_e)^{\text{SOL}} = 0.4-0.5\% \).

The power exhaust scenario provides peak-\( q_{\text{target}} \) is reduced less than 10 MWm\(^{-2} \) \( \Rightarrow \) ITER divertor technology (W-monoblock&Cu-alloy pipe) is applicable.

• Improvement of plasma & neutral modelling in the detachment, and the divertor operation (lower \( f_{\text{rad}}, \chi \) and D, geometry, impurity selection) are high priority issues.

**Physics issues of the JA DEMO power exhaust design (at this stage):**

• Possibility and its margin of high \( f_{\Gamma \text{T}} \) in impurity seeding (\( n_e \) is within exp. database).
• \( P_{\text{th}}^{\text{L-H and H-L}} \) and its margin in \( Z_{\text{eff}} = 2-3 \) (and \( f_{\Gamma \text{T}} > 1 \)).
• Control of radiation peak and detach front in divertor leg (leg length, power flux?)
• Scenario (model) of the edge plasma when the X-point radiation is enhanced.
• Appropriate \( \chi \) & D profiles for DEMO, and dissipation in the detachment (\( S_{\text{div}} \)).
• Impurity concentration (\( n_{\text{imp}}/n_e \)) at edge-SOL –divertor, and the shielding efficiency.
Heat removal of W-monoblock/cooling-pipe and water-cooling arrangement in a cassette were studied for severe case ($P_{\text{div \, thermal}}: 300\text{MW} + P_{\text{div \, neutron}}: 128 \text{ MW}$):

- Neutronics analysis showed W/CuCrZr heat sink can be applied at high heat flux and low n-flux area $\Rightarrow$ frequent remote-maintenance (1-2 year) will be required.
- 3D heat transport analysis showed Max. temp. on W ($1021^\circ\text{C}$) and Cu-alloy ($331^\circ\text{C}$) and Heat flux were in operation range: appropriate for peak $q_{\text{target}} = 10\text{MWm}^{-2}$-level.

- **Cassette design**, considering nucl. heat cooling, n-shield, exhaust opening and target unit replacement, is high priority issue in 2016-2017.

Engineering and material issues of the JA DEMO divertor design (at this stage):

- Application of CuCrZr at high irradiation dose ($>2\text{dpa}$) as a heat sink (structure)
- Appropriate water temperature (150-200°C) for W/Cu-alloy heat exhaust unit.
- Evaluation of **RM period (life time)** even without melting: erosion, re-crystallization of both W and CuCrZr and thermal fatigue by transient events.
- Replacement design of CuCrZr cooling unit in the cassette (in hot cell)
- Divertor cassette design as a neutron shield for the vacuum vessel.