Updated Nuclear Parameters, Radial/Vertical Build, and Activation Analysis for ARIES-AT

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With inputs from:
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Web address:

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Design Parameters

Fusion power 1737 MW

FW location at midplane – OB, IB 6.05, 3.55 m
at top/bottom – OB, IB ~4.5, 3.55 m

\( \Gamma \): Peak OB, IB, div. 6.1, 4, 2 MW/m^2
Average OB, IB 5.2, 2.8 MW/m^2

FW poloidal length* – OB, IB ~5.5, 4.5 m

SiC burnup limit 3% (1.5 atom% He)

FS dpa limit 200 dpa

Machine lifetime 40 FPY

HT magnet

ARIES-RS’ vacuum vessel configuration

# 10/14/99 Strawman
* Between X points
Updated Nuclear Parameters*

• Key features of FW/Blanket:
  – 1/2000 FW/blanket design
  – 1.9 cm thick FW: 51% SiC, 49% LiPb
  – IB and OB blankets only (no blanket behind divertor):
    – 30 cm thick IB FW/blanket
    – 65 cm thick OB FW/blanket segmented into:
      – 30 cm FW/Blanket -I
      – 35 cm Blanket-II
  – 90% enriched LiPb
  – Vertical stabilizing shell not included
  – Penetrations:
    – 0.6 m² for ICRF, 1.1 m² for NBI, 1 m² for LH on OB, per TK
    – 2 cm radial gaps between 16 blanket modules

• Reference nuclear parameters:
  Overall TBR 1.16
  Overall $M_n$ 1.1
  SiC Burnup rate 1% per FPY#
  FW EOL Fluence 18.5 MWy/m²
  FW Lifetime 3 FPY

• Comments:
  – More SiC content in FW degrades breeding
  – Thicker blanket increases breeding slightly (~ 3%)
  – Higher enrichment (> 90%) is expensive and has insignificant impact on breeding
  – More penetrations/gaps reduce breeding
  – Vertical stabilizing shells degrade breeding
  – Blanket thickness and/or enrichment will be adjusted to meet breeding requirement of 1.1 after including stabilizing shell

* Using FENDL-2 cross section data library
# 0.7% Si, 0.3% C
Two continuously toroidal shells placed at top/bottom of IB and OB sides

IB shells embedded in HT shield

OB shells embedded in B-II and cover 50% of poloidal length

Shells have insignificant impact on breeding of FW/B-I

Shells degrade breeding of upper/lower parts of B-II behind shells

Based on 3-D calculations, B-II parts behind shells contribute ~10% to TBR
Al and Cu shells have < 1% impact on breeding

W shells could reduce breeding by 5-8%, depending on thickness
## Nuclear Heating Deposited in OB Stabilizing Shells

<table>
<thead>
<tr>
<th>Material</th>
<th>Operating Temperature</th>
<th>W Shells</th>
<th>Al Shells</th>
<th>Cu Shells</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>400 °C</td>
<td>6 cm</td>
<td>4 cm</td>
<td>2 cm</td>
</tr>
<tr>
<td></td>
<td>1000 °C</td>
<td>12 cm</td>
<td>11 cm</td>
<td>4 cm</td>
</tr>
<tr>
<td>W Shells</td>
<td>36 MW*</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Al Shells</td>
<td>5 MW</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cu Shells</td>
<td>7 MW</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* Heating in both upper and lower shells

**W shells contain 4-7 X nuclear heating deposited in Al or Cu shells**
• Passive cooling of all shells is not feasible. **Shells should be actively cooled, preferably with He**, per Rene and Malang,

• **Recommended operating temperature** for Al and Cu shells is < 700 °C.

• **W shell** should operate above 800 °C to avoid embrittlement. W shells will be ~10 cm thick and heavy

• Al shell (~5 cm thick) need W or TZM cladding (compatible below 700 °C)

• **3-4 cm thick Cu shell is recommended.** It is thin, light, and has negligible impact on breeding

• **Effect of disruption forces** on structural integrity of Cu should be assessed (may need to support Cu by steel cables or provide strong casing)

• Will **transmutations** increase Cu resistivity significantly? TBD

• Impact of Cu shells on decay heat, LOCA/LOFA temperature, and WDR will be assessed
Components’ Lifetimes

- Service lifetimes are based on:
  - 3% burnup limit for SiC structure of FW, blanket, HT shield
  - 200 dpa limit for FS structure of LT shield and V.V.

- Lifetime of stabilizing shell is unknown. TBD
• ARIES-AT HT S/C magnet radiation limits:

\[ 10^{19} \text{ n/cm}^2 \] Peak fast n \text{fluence}\# to HT S/C

--- Radiation resistant thermal insulator

• ARIES-RS LT S/C magnet radiation limits:

\[ 10^{19} \text{ n/cm}^2 \] Peak fast n \text{fluence} to Nb\textsubscript{3}Sn S/C

\[ 2 \text{ mW/cm}^3 \] Peak nuclear \textit{heating}

\[ 6 \times 10^{-3} \text{ dpa} \] Peak atomic \textit{displacement} to Cu stabilizer\*

\[ 10^{11} \text{ rad} \] Peak \textit{dose} to GFF polyimide

--- Radiation resistant thermal insulator

\* \( E_\text{n} > 0.1 \text{ MeV} \)

\* 85\% of dpa can be annealed out by warming up magnets during maintenance
## Inboard Radial Build

### Component Composition

<table>
<thead>
<tr>
<th>Component</th>
<th>Composition</th>
</tr>
</thead>
<tbody>
<tr>
<td>FW (1.9 cm)</td>
<td>51% SiC, 49% LiPb</td>
</tr>
<tr>
<td>Blanket (28.1 cm)</td>
<td>12% SiC, 88% LiPb</td>
</tr>
<tr>
<td>HT Shield</td>
<td>15% SiC, 10% LiPb, 75% B-FS</td>
</tr>
<tr>
<td>LT Shield</td>
<td>15% FS, 10% H₂O, 75% WC</td>
</tr>
<tr>
<td>Vacuum Vessel</td>
<td>35% FS, 65% H₂O</td>
</tr>
<tr>
<td>HT Magnet</td>
<td>87% SS, 10% LN, 2.5% Y₁Ba₂Cu₃O₇, 0.5% Ag</td>
</tr>
<tr>
<td>Bucking cylinder</td>
<td>95% SS, 5% LN</td>
</tr>
</tbody>
</table>

- LT shield and V.V. are combined in a single component. Reweldability limit (1 He appm) for FS is not met at front of LT shield
  ⇒ Locate cut/weld areas away from high radiation zones
- LT shield and TF magnet radiation limits are all met* for peak $\Gamma = 4$ MW/m$^2$ (200 dpa at LT shield and $10^{19}$ n/cm$^2$ at magnet @ EOL)

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* SiC and WC are 95% dense
* Safety factor of 3 considered in all shielding calculations
# Outboard Radial Build

## Component Composition

<table>
<thead>
<tr>
<th>Component</th>
<th>Composition</th>
</tr>
</thead>
<tbody>
<tr>
<td>FW/Blanket-I:</td>
<td></td>
</tr>
<tr>
<td>FW (1.9 cm)</td>
<td>51% SiC , 49% LiPb</td>
</tr>
<tr>
<td>B-I (28.1 cm)</td>
<td>12% SiC , 88% LiPb</td>
</tr>
<tr>
<td>Blanket-II</td>
<td>14% SiC , 86% LiPb</td>
</tr>
<tr>
<td>HT Shield</td>
<td>15% SiC , 10% LiPb, 75% B-FS</td>
</tr>
<tr>
<td>Vacuum Vessel</td>
<td>25% FS , 75% H₂O</td>
</tr>
<tr>
<td>HT Magnet</td>
<td>87% SS, 10% LN, 2.5% Y₁Ba₂Cu₃O₇, 0.5% Ag</td>
</tr>
<tr>
<td>Coil Case</td>
<td>95% SS, 5% LN</td>
</tr>
</tbody>
</table>

- Blanket-II and HT shield could be combined in a single lifetime component
- FS, V.V., and TF magnet radiation limits are all met* for peak $\Gamma = 6$ MW/m²
  (200 dpa for FS, 1 He appm at V.V., and $10^{19}$ n/cm² at magnet @ EOL)

---

* SiC and WC are 95% dense  
* Safety factor of 3 considered in all shielding calculations
Vertical Build

<table>
<thead>
<tr>
<th>Component</th>
<th>Composition</th>
</tr>
</thead>
<tbody>
<tr>
<td>Divertor Plates</td>
<td>19% W, 5% SiC, 76% LiPb</td>
</tr>
<tr>
<td>Replaceable HT Shield</td>
<td>15% SiC, 10% LiPb, 75% FS</td>
</tr>
<tr>
<td>HT Shield</td>
<td>15% SiC, 10% LiPb, 75% B-FS</td>
</tr>
<tr>
<td>LT Shield</td>
<td>15% FS, 10% H₂O, 75% B-FS</td>
</tr>
<tr>
<td>Vacuum Vessel</td>
<td>35% FS, 65% H₂O</td>
</tr>
<tr>
<td>HT Magnet</td>
<td>87% SS, 10% LN, 2.5% Y₁Ba₂Cu₃O₈, 0.5% Ag</td>
</tr>
</tbody>
</table>

- LT shield and V.V. are combined in a single component. Reweldability limit (1 Heppm) for FS is not met at front of LT shield
  ⇒ Locate cut/weld areas away from high radiation zones
- LT shield and TF magnet radiation limits are all met* for peak $\Gamma = 2 \text{ MW/m}^2$
  (200 dpa at LT shield and $10^{19} \text{ n/cm}^2$ at magnet @ EOL)
- Shielding behind inner divertor plates will be assessed

---

* SiC and WC are 95% dense

* Safety factor of 3 considered in all shielding calculations
• V.V. composition optimized by trading WC filler for water

• Eliminating WC filler simplifies V.V. design but results in high magnet heating, high thermal neutron flux, and high activity at V.V.

• Optimum VV composition for fluence:
  35% FS structure, 40% H₂O, 25% WC filler

• Optimum VV composition for heating:
  35% FS structure, 25% H₂O, 40% WC filler
• V.V. composition optimized by trading B-FS filler for water

• Eliminating B-FS filler simplifies V.V. design but results in high magnet heating, high thermal neutron flux, and high activity at V.V.

• Optimum VV composition for fluence:
  25% FS structure , 60% H$_2$O,  15% B-FS filler

• Near optimum VV composition for heating that meets fluence limit:
  25% FS structure , 40% H$_2$O,  35% B-FS filler
Comparison Between ARIES-AT and ARIES-RS Radial/Vertical Builds

<table>
<thead>
<tr>
<th>Thickness (cm):</th>
<th>Inboard</th>
<th>Outboard</th>
<th>Divertor</th>
</tr>
</thead>
<tbody>
<tr>
<td>DP</td>
<td>AT</td>
<td>RS</td>
<td>AT</td>
</tr>
<tr>
<td>DP</td>
<td>---</td>
<td>---</td>
<td>---</td>
</tr>
<tr>
<td>FW/Blanket-I</td>
<td>30</td>
<td>20</td>
<td>30</td>
</tr>
<tr>
<td>Blanket-II</td>
<td>---</td>
<td>---</td>
<td>35</td>
</tr>
<tr>
<td>Replaceable shld</td>
<td>---</td>
<td>20</td>
<td>---</td>
</tr>
<tr>
<td>HT shield</td>
<td>22</td>
<td>26</td>
<td>25</td>
</tr>
<tr>
<td>LT shield</td>
<td>27</td>
<td>28</td>
<td>---</td>
</tr>
<tr>
<td>Vacuum vessel</td>
<td>20</td>
<td>20</td>
<td>30</td>
</tr>
<tr>
<td>Subtotal</td>
<td>99</td>
<td>114</td>
<td>120</td>
</tr>
<tr>
<td>Reduction in thickness</td>
<td>15</td>
<td>0</td>
<td>35</td>
</tr>
<tr>
<td>Magnet &amp; cryostat</td>
<td>25</td>
<td>55</td>
<td>25</td>
</tr>
<tr>
<td>Total</td>
<td>124</td>
<td>169</td>
<td>145</td>
</tr>
<tr>
<td>Net reduction in thickness</td>
<td>45</td>
<td>0</td>
<td>65</td>
</tr>
</tbody>
</table>

- Thinner ARIES-AT radial/vertical builds are due to:
  - Superior shielding
    - better LiPb shielding performance compared to Li
    - use of water in LT shield and V.V. instead of He
  - thin HT magnet
Activation Analysis

- **Codes and model:**
  - Activation: ALARA code; FENDL-2 activation library
  - Flux: 1-D DANTSYS code; FENDL-2 Xn data
  - 175 n and 42 g group structure
  - 3-D neutron flux used to re-normalize 1-D flux for all components
  - Average OB and IB $\Gamma$ are 5.2 and 2.8 MW/m$^2$, respectively
  - Operation time: 3 FPY for FW/B-I, 40 FPY for all other components
  - Continuous operation, unless indicated (this overestimates radioactivity of intermediate $T_{1/2}$ nuclides by inverse of availability [10-25%])

- Activity, decay heat, WDR, and clearance index depend strongly on materials, flux level, neutron spectrum, operation time, and cooling period

- **Results reported here are for:**
  - 100% dense compacted waste (coolants and void excluded)
  - IB and OB sides as defined by radial builds
  - SiC, WC, and LiPb compositions with impurities.
  - FS with impurity control (IC) to qualify as Class C waste

<table>
<thead>
<tr>
<th>Elements</th>
<th>Original FS wppm</th>
<th>FS w/ IC wppm</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nb</td>
<td>4</td>
<td>1</td>
</tr>
<tr>
<td>Mo</td>
<td>70</td>
<td>20</td>
</tr>
</tbody>
</table>

- Impurities for magnet constituents are not available yet. Will be included in future calculations

- **Results include:**
  - Activity
  - Decay heat
  - Fetter-L and Fetter-H waste disposal ratings for Class C waste
  - NRC (10CFR61) waste disposal ratings for Class C and A waste
Unlike metals, SiC activity drops by several orders of magnitude shortly after shutdown.

Highly irradiated SiC components generate lower intermediate activity (1d-5y) than well protected FS and WC components.
Unlike metals, SiC decay heat drops fast after one minute, meaning slight increase in temperature of SiC components during LOCA/LOFA.

Detailed decay heat for individual constituents of all components (including coolants) provided for LOCA/LOFA analysis.
LiPb Decay Heat for LOFA Analysis

- **Assumptions:**
  - Same LiPb is used for 40 FPY (Li can be refurbished if needed)
  - LiPb spends 1 min in both divertor and FW/B-I and 3.4 min in both OB B-II and HT shield, per Rene
  - LiPb spends $t_{\text{out}} \sim 2$ min in outer loop for heat recovery, T extraction, and Po/Bi/Hg purification
  - LiPb returns to same location inside torus (conservative)
  - 100% system availability (conservative)

- **Pulsed analysis** performed to determine:
  - Sensitivity of LiPb decay heat to $t_{\text{out}}$
  - Variation of LiPb decay heat with operation time (3,10,20,40 y)

- Among all LiPb cooled components, highest LiPb decay heat is generated in LiPb of OB FW considered for sensitivity analysis
Selected parameters for LOFA analysis:

- $t_{in} = 1\text{ min}$ for B-I and $3.4\text{ min}$ for B-II
- $t_{out} = 2\text{ min}$
- Irradiation time = $40\text{ y}$ (7x10$^6$ irradiation periods for B-I and 4x10$^6$ for B-II)
Continuous irradiation overestimates decay heat of flowing LiPb by factor of 10
• LiPb of OB FW/B-I contains highest decay heat compared to other blankets
• 1 h after shutdown, LiPb generates higher decay heat than SiC
  ⇒ LOFA is more critical than LOCA
• Less conservative assumptions reduce decay heat by 10-30%
Waste Disposal Rating

- WDR reported for compacted waste (void excluded)

- WDR < 1 means component qualifies as low level waste (LLW)

- WDR remains constant for 100’s of years after shutdown, unless indicated

- All components should meet BOTH Fetter’s and NRC WD limits

- Fetter developed limits for 101 isotopes. 19 isotopes have range of limits rather than single value. Those (beta emitters) are: C$^{14}$, Si$^{32}$, Cl$^{36}$, Ca$^{41}$, Ni$^{63}$, Se$^{79}$, Sr$^{90}$, Tc$^{97}$, Tc$^{98}$, Tc$^{99}$, Pd$^{107}$, I$^{129}$, Sm$^{151}$, Gd$^{148}$, Gd$^{150}$, Dy$^{154}$, Pb$^{210}$, Ra$^{226}$, Ac$^{227}$

- Fetter-L and Fetter-H WDRs are evaluated for low and high limits, respectively, for Class C LLW. Fetter-L limits were not considered in previous ARIES designs.

- NRC has Class C and Class A WD limits for 9-10 isotopes. Class A has low limit for tritium
## Fetter’s Waste Disposal Rating

### Class C Waste:

<table>
<thead>
<tr>
<th>Component</th>
<th>Fetter-H</th>
<th>Fetter-L</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Outboard Components:</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FW/B-I</td>
<td>0.092</td>
<td>0.095</td>
</tr>
<tr>
<td>B-II</td>
<td>0.004</td>
<td>0.02</td>
</tr>
<tr>
<td>HT Shield</td>
<td>0.2</td>
<td>0.3</td>
</tr>
<tr>
<td>V.V.</td>
<td>0.05</td>
<td>0.055</td>
</tr>
<tr>
<td>Magnet</td>
<td>0.016</td>
<td>0.023</td>
</tr>
<tr>
<td><strong>Inboard Components:</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FW/B</td>
<td>0.019</td>
<td>0.021</td>
</tr>
<tr>
<td>HT Shield</td>
<td>0.7</td>
<td>1.0*</td>
</tr>
<tr>
<td>LT Shield</td>
<td>0.030</td>
<td>0.046</td>
</tr>
<tr>
<td>V.V.</td>
<td>0.0014</td>
<td>0.0017</td>
</tr>
<tr>
<td>Magnet</td>
<td>0.015</td>
<td>0.022</td>
</tr>
<tr>
<td><strong>Bucking Cylinder</strong></td>
<td>0.003</td>
<td>0.008</td>
</tr>
</tbody>
</table>

- Al\(^{26}\) is dominant nuclide for Fetter’s WDR of SiC components
  - Si\(^{28}\) (n , np) Al\(^{27}\) (n , 2n) Al\(^{26}\)

### All components qualify as Class C LLW @ EOL

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* Dictated impurity control for FS. Value was 2 without impurity control.
### NRC Waste Disposal Rating

<table>
<thead>
<tr>
<th>Outboard Components</th>
<th>NRC Class A</th>
<th>NRC Class C</th>
</tr>
</thead>
<tbody>
<tr>
<td>FW/B-I</td>
<td>12*</td>
<td>0.03</td>
</tr>
<tr>
<td>B-II</td>
<td>3*</td>
<td>0.1</td>
</tr>
<tr>
<td>HT Shield</td>
<td>8</td>
<td>0.2</td>
</tr>
<tr>
<td>V.V.</td>
<td>3*</td>
<td>0.03</td>
</tr>
<tr>
<td>Magnet</td>
<td>0.1</td>
<td>0.005</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Inboard Components</th>
<th>NRC Class A</th>
<th>NRC Class C</th>
</tr>
</thead>
<tbody>
<tr>
<td>FW/B</td>
<td>5*</td>
<td>0.02</td>
</tr>
<tr>
<td>HT Shield</td>
<td>128</td>
<td>0.6</td>
</tr>
<tr>
<td>LT Shield</td>
<td>0.3</td>
<td>0.02</td>
</tr>
<tr>
<td>V.V.</td>
<td>0.08</td>
<td>0.001</td>
</tr>
<tr>
<td>Magnet</td>
<td>0.1</td>
<td>0.005</td>
</tr>
<tr>
<td>Bucking Cylinder</td>
<td>0.04</td>
<td>0.003</td>
</tr>
</tbody>
</table>

- For SiC components, T and C\(^{14}\) are dominant nuclides for NRC-A WDR and C\(^{14}\) is dominant nuclide for NRC-C WDR
- \(C^{12}\ (n, \gamma)\ C^{13}\ (n, \gamma)\ C^{14}\)
- Some components qualify as Class A LLW

**All components qualify as Class C LLW @ EOL**

* Could qualify as Class A LLW after 100 y of storage period
LiPb Waste Disposal Rating

Class C

Fetter-H,L  5.5

NRC  0.0002

- LiPb does not qualify as Class C waste unless Bi is controlled during operation
- Bi$^{208}$ is dominant nuclide for Fetter’s WDR (95%)
- Pb and Bi impurity (43 wppm) generate 90% and 10% of Bi$^{208}$, respectively, via the following reactions:
  \[
  \text{Pb}^{208} (n, \gamma) \text{Pb}^{209} (\beta^- \text{ decay}) \text{Bi}^{209} (n, 2n) \text{Bi}^{208}
  \]
  \[
  \text{Bi}^{209} (n, 2n) \text{Bi}^{208}
  \]
- Also, Bi generates Po$^{210}$ which raises safety concerns:
  \[
  \text{Bi}^{209} (n, 2n) \text{Bi}^{208} (n, \gamma) \text{B}^{209} (n, \gamma) \text{Bi}^{210} (\beta^- \text{ decay}) \text{Po}^{210}
  \]
- Bi production continues to rise during operation

- **Purification system** should be designed to keep average Bi$^{208}$ and Po$^{210}$ inventories below permissible level
Conclusions

• **Neutronics:**
  – Blanket meets breeding requirement (TBR ≥ 1.1) with adequate margin
  – 3% burnup limit (1.5 atom% He) results in lifetime of 3 FPY for SiC components. If 2 atom% He is acceptable, burnup limit could be raised to 4%

• **Shielding:**
  – Radial builds are well optimized for the design constraints
  – May need to incorporate WC/B-FS in V.V. to reduce V.V. activation

• **Activation:**
  – Unlike metals, SiC activity and decay heat drop rapidly by 3 orders of magnitude in one day
  – LOFA is more critical than LOCA
  – All components qualify easily as Class C LLW
  – LiPb does not qualify as Class C waste unless purification system removes Bi^{208} during operation
  – Nb and Mo impurity control is needed for FS
  – Less conservative assumptions will result in lower activation

• **Stabilizing shells** will be included in future analysis