Final Nuclear Analysis for ARIES-AT

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Web address:

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UW - Madison
Major Conclusions

Neutronics:
- **Blanket satisfies breeding requirement** (TBR \( \geq 1.1 \)) with minor adjustment to accommodate W shells
- **Blanket segmented** to lower replacement cost, reduce volume of waste, and increase repository capacity
- **4 FPY** service lifetime for inner blanket and divertor system and **40 FPY** for all other components
- **1.1** overall energy multiplication

Shielding:
- **Shield and V.V.** are well optimized for design constraints
- **Compact radial build** compared to previous ARIES designs
- **Well protected magnets** for 40 FPY
- **Need higher reweldability limit for FS**, relocate V.V. welds in low radiation zones, or avoid welding in front 15 cm of V.V.
- **Remotely maintained** components; no hands-on maintenance

Activation:
- **Low activation materials** for all components \( \Rightarrow \) low level waste
  (SiC, FS with impurity control, and LiPb with Bi purification system)
- **Design solutions developed** to reduce decay heat and mitigate effect during accident
- **Relatively small volume of waste** compared to previous ARIES designs
- **Because of compactness**, all components have clearance index \( > 1 \)
  \( \Rightarrow \) **no “Free Release” of metals** even after 100 y storage period
- **Waste management options:**
  - Recycle and reuse in nuclear facilities (to increase repository capacity)
  - Dispose near surface as **low level waste**:
    - All components easily qualify as **Class C LLW**
    - Most components qualify as **Class A LLW**
    - Waste classification: **90% Class A and 10% Class C**
Design Parameters and Radiation Limits

Fusion power 1719 MW

**FW location**
- at midplane – OB, IB: 6.55, 3.85 m
- at top/bottom – OB, IB: 4.7, 3.85 m

Γ: Peak OB, IB, div.
- 4.75, 3.1, 2 MW/m²
- Average OB, IB, div.
- 4.0, 2.2, 1 MW/m²

Machine lifetime 40 FPY

Availability 75%

SiC burnup limit 3% (1.5 atom% He*)

FS dpa limit 200 dpa

Steel reweldability limit
1 He appm for FS
5 - 30 He appm for 316SS**

HT magnet fluence limit $10^{19}$ n/cm² ($E_n > 0.1$ MeV)

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# 3/10/2000 Strawman

* up to 2 atom% He is acceptable, per R. Jones

** Ref.: W. Daenner (Germany), 1999 EU experimental results
Final Nuclear Parameters

• Key features of FW/Blanket:
  – 1.5 cm thick FW: 68% SiC, 32% 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 (replaceable)
      – 35 cm Blanket-II (permanent)
  – 90% enriched LiPb @ 770 °C
  – Penetrations:
    – 2.5 m² on OB for plasma control
    – 2 cm wide radial gaps between 16 blanket modules

• Nuclear parameters:
  Overall TBR: 1.11 w/o shells
  1.07 w/ 4 cm thick W shells**
  Overall Mn: 1.1
  SiC Burnup rate: 0.77% per FPY#
  FW EOL Fluence: 18.5 MWy/m²
  FW Lifetime: 4 FPY

Thicken OB blanket-II by 10 cm to meet breeding requirement (1.1) in case of 4 cm thick W vertical stabilizing shells

• Comments:
  – Lower SiC content increases breeding
  – Thicker OB blanket (> 45 cm) does not increase breeding
  – Higher enrichment (> 90%) is expensive and has insignificant impact on breeding
  – More penetrations/gaps reduce breeding

Breeding level is marginal ⇒ reduce SiC in FW/blanket below 20%

* Using FENDL-2 cross section data library
** Vertical stabilizing shells
# 0.54% Si, 0.23% C
Nuclear Heat Load to In-vessel Components

\( P_f = 1719 \text{ MW} \), \( P_n = 1375 \text{ MW} \)

<table>
<thead>
<tr>
<th><strong>Nuclear Heating (MW)</strong></th>
<th><strong>Inboard</strong></th>
<th><strong>Outboard</strong></th>
<th><strong>Divertor</strong></th>
<th><strong>Total</strong></th>
</tr>
</thead>
<tbody>
<tr>
<td>FW or DP</td>
<td>41</td>
<td>100</td>
<td>39#</td>
<td>180 (12%)</td>
</tr>
<tr>
<td>Blanket:</td>
<td></td>
<td></td>
<td></td>
<td>1150 (76%)</td>
</tr>
<tr>
<td>B-I (28.5 cm)</td>
<td>280</td>
<td>710</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td>B-II (35 cm)</td>
<td>---</td>
<td>135</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td>W Shells (4 cm)</td>
<td>---</td>
<td>20</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td>16 Wedges*</td>
<td>---</td>
<td>5</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td>HT Shield/W Shells</td>
<td>50 / 4</td>
<td>20</td>
<td>113##</td>
<td>187 (12%)</td>
</tr>
<tr>
<td>Total</td>
<td>375</td>
<td>990</td>
<td>152</td>
<td>1517 (25%) (65%) (10%)</td>
</tr>
</tbody>
</table>

Overall neutron energy multiplication is 1.1

**Low Grade Heat:**

- Vacuum Vessel (MW) 7 3 1 11*** (< 1% of total htg)
- Magnet (kW) 0.6 48 0.3 ~50###

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** upper and lower divertor regions
# 25 MW in dome, 8 MW in outer divertor plates, 6 MW in inner divertor plates
* 3\% of B-II
## 58 MW in replaceable shield, 27 MW in vertical shield, 28 MW in IB shield above/below X point
*** does not include thermal heat leak from HT shield (5-10 MW)
### requires 0.5 MW of cryogenic load @ 10 W/W
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 V.V.
  – $10^{19}$ n/cm$^2$ fast n fluence to YBCO conductor of HT magnet
Inboard Radial Build*

Component Composition#

FW (1.5 cm) 68% SiC, 32% LiPb
Blanket (28.5 cm) 18% SiC, 82% LiPb
HT Shield 14% SiC, 10% LiPb, 72% B-FS, 4% W
Vacuum Vessel 13% FS, 22% H₂O, 65% WC
HT Magnet 87% SS, 10% LN, 2.5% Y₁Ba₂Cu₃O₇, 0.5% Ag
Bucking cylinder 95% SS, 5% LN

• 29 cm thick HT shield helps reduce heat leakage and V.V. decay heat
• FS 1 He appm reweldability limit is NOT met at outer surface of V.V. (10 He appm, 46 H appm, 20 dpa)
• Could higher reweldability limit for 316SS (5-30 He appm) apply to FS? If not, locate cut/weld areas away from high radiation zones
• Magnet composition does not contain CeO₂ insulator nor impurities

* Safety factor of 3 considered in all shielding calculations
# SiC and WC are 95% dense
Outboard Radial Build* without Shells

Component Composition#
FW/Blanket-I:  
FW (1.5 cm) 68% SiC , 32% LiPb  
B-I (28.5 cm) 18% SiC , 82% LiPb  
Blanket-II 20% SiC , 80% LiPb  
HT Shield 15% SiC , 10% LiPb , 75% B-FS  
Vacuum Vessel** 30% FS , 70% H₂O  
HT Magnet 87% SS, 10% LN, 2.5% Y₃Ba₂Cu₃O₇, 0.5% Ag  
Coil Case 95% SS, 5% LN  

- Along with blanket/shield/V.V., 10 cm thick port enclosures provide shielding for sides of magnets  
- Wedges underneath magnets must be carefully designed to protect magnets  
- FS 1 He appm reweldability limit is met at inner surface of V.V.  
  (1 He appm, 5 H appm, 3 dpa)  
- Is 5 cm thick outer coil case acceptable?

* Safety factor of 3 considered in all shielding calculations  
# SiC and WC are 95% dense  
** Composition is slightly of-optimum to simplify port design
Outboard Radial Build*
With 4 cm W Vertical Stabilizing Shell

<table>
<thead>
<tr>
<th>Component</th>
<th>Composition#</th>
</tr>
</thead>
<tbody>
<tr>
<td>FW/Blanket-I:</td>
<td></td>
</tr>
<tr>
<td>FW (1.5 cm)</td>
<td>68% SiC, 32% LiPb</td>
</tr>
<tr>
<td>B-I (28.5 cm)</td>
<td>18% SiC, 82% LiPb</td>
</tr>
<tr>
<td>Blanket-II</td>
<td>19% SiC, 78% LiPb, 3% W</td>
</tr>
<tr>
<td>HT Shield</td>
<td>15% SiC, 10% LiPb, 75% B-FS</td>
</tr>
<tr>
<td>Vacuum Vessel</td>
<td>30% FS, 70% H₂O</td>
</tr>
<tr>
<td>HT Magnet</td>
<td>87% SS, 10% LN, 2.5% Y₁Ba₂Cu₃O₈, 0.5% Ag</td>
</tr>
<tr>
<td>Coil Case</td>
<td>95% SS, 5% LN</td>
</tr>
</tbody>
</table>

- To meet breeding requirement, OB B-II should be 45 cm thick, trading shield for blanket
- Total OB radial standoff remains fixed

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

<table>
<thead>
<tr>
<th>Component</th>
<th>Composition#</th>
</tr>
</thead>
<tbody>
<tr>
<td>W coating</td>
<td>100 W-0.2%TiC alloy</td>
</tr>
<tr>
<td>Divertor Plates</td>
<td>46% SiC, 54% 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>Vacuum Vessel</td>
<td>13% FS, 22% H₂O, 65% B-FS</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>

- FS reweldability limit (1 He appm) is **NOT met** at front of V.V.
  ⇒ Locate cut/weld areas away from high radiation zones or adopt higher limits
- **15 cm thick local shield** must be provided behind divertor pumping ducts. Cool it with LN to act as heat sink during LOCA/LOFA
- **No shielding problem** to inner legs of magnets behind inner divertor plates

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* Safety factor of 3 considered in all shielding calculations
* SiC and WC are 95% dense
• 13% FS structural content dictated by structural requirements, per L. Waganer

• V.V. composition optimized by trading WC filler for water

• Optimal VV composition for minimum fluence is:
  13% FS structure, 22% H₂O, and 65% WC filler
Activation Analysis

• Codes, parameters, and assumptions:
  – 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 4 and 2.2 MW/m$^2$, respectively
    (20-30% lower $\Gamma$ compared to previous strawman)
  – IB and OB sides as defined by radial builds
    (29 cm thick IB HT shield helps reduce V.V. decay heat)
  – 4 cm W vertical stabilizing shells embedded in IB HT shield and in 35 cm OB B-II
  – Operation time: 4 FPY for FW/B-I and 40 FPY for all other components
  – Latest magnet composition with impurities not included
  – 75% system availability included in analysis

• Reported results are for:
  – IB and OB at end of service life:
    – Activity
    – Decay heat
    – Radwaste level
  – 100% dense compacted waste (coolants and void excluded)
  – SiC, WC, W, and LiPb with impurities
  – FS with impurity control: 1 wppm Nb, 20 wppm Mo

• Activation results are posted on Wilson’s web site:
  http://marie.ep.wisc.edu/~wilson/ALARA/aries_at_results/
• 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.
Impact of 75% Availability on Decay Heat

- **Availability credit** = decay heat with 100% availability / decay heat with 75% availability

- **Assumption**: 9 months of operation and 3 months of down time

- **Results for 1h-1w time period**:
  - 1-7% effect on SiC components
  - 1-12% effect on FS filler and structure
  - 5-16% effect on magnet

System availability will be included in future activation analysis
Continuous irradiation overestimates decay heat of flowing LiPb by more than factor of 10 ⇒ consider LiPb residence time in torus and in outer loop

Assumptions:
- Same LiPb used for 40 FPY (Li can be refurbished if needed)
- On-line removal of all tritium generated
- Per René, LiPb residence times for in-vessel components are:
  - 1 s in IB FW
  - 2 s in OB FW
  - 3 s in DP
  - 35 s in channel of IB Blanket
  - 70 s in channel of OB Blanket-I
  - 240 s in channel of OB Blanket-II
  - 10 s in side and back walls of blankets
  - 60 s in HT shield
- LiPb spends ~2 min in outer loop for heat recovery, T extraction, and Po/Bi/Hg purification
- LiPb returns to same location inside torus (conservative results)
- 75% system availability

If LOFA temperature is excessive, less conservative assumptions will be considered to reduce LiPb decay heat
- After one hour, LiPb of Blanket-I generates higher decay heat than SiC

In LiPb/SiC system, LOFA is more critical than LOCA
• Bi and Po inventories continue to rise during operation

• For safety reasons, Bi and Po should be controlled during operation

• Pb and Bi impurity (43 wppm) generate 90% and 10% of Bi$^{208}$, respectively, via the following reactions:

\[
\text{Pb}^{208} (n,\gamma) \rightarrow \text{Pb}^{209} (\beta^- \text{ decay}) \rightarrow \text{Bi}^{209} (n,2n) \rightarrow \text{Bi}^{208} \\
\text{Bi}^{209} (n,2n) \rightarrow \text{Bi}^{208}
\]

• Bi generates Po$^{210}$:

\[
\text{Bi}^{208} (n,\gamma) \rightarrow \text{Bi}^{209} (n,\gamma) \rightarrow \text{Bi}^{210} (\beta^- \text{ decay}) \rightarrow \text{Po}^{210}
\]

• Purification system should be designed to keep average Bi$^{208}$ and Po$^{210}$ inventories below permissible level (~25 Ci)
Waste Management

• Options:
  
  – Clear or “free release” of materials to industrial facilities

  – Dispose near surface as Class A or C low level waste (LLW)

  – Recycle waste and reuse in nuclear facilities

• Clearance and disposal options have been addressed in details in
  ARIES-AT study

• Waste could be recycled at a cost (unknown). No effort devoted to
  recycling of ARIES-AT waste
In absence of national NRC standards, IAEA clearance limits are used
• All ARIES-AT components have clearance index > 1 based on IAEA clearance limits
  ⇒ Components/constituents cannot be released as cleared metals
• At present, US industry has no tolerance for slightly contaminated materials
  ⇒ No market for cleared metals
• NRC limits could be more restrictive than IAEA’s (dose ~1 mrem/y)

ARIES-AT waste will be disposed near-surface as Class A or C LLW or could be recycled

* Defined as unrestricted release of items and materials from radiologically controlled areas
Waste Disposal Rating

- WDR reported for **compacted waste** (void excluded)

- WDR < 1 means component qualifies as LLW

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

- All components should meet BOTH Fetter’s and NRC-10CFR61 WD limits for Class A or C LLW

- **Waste disposal limits:**
  - **NRC (10CFR61):**
    - **Official** U.S. WD limits
    - NRC has developed Class A and Class C WD limits for 9-10 isotopes beside actinides.
    - NRC limits not available for ~90 isotopes of interest to fusion
    - Class A has low limit for tritium ($T_{1/2} \approx 12.3$ y)
  - **Fetter’s:**
    - **Not** in regulations form
    - Approved by U.S. Fusion Safety Standing Committee
    - NRC has not endorsed Fetter’s limits
    - No limits available for Class A LLW
    - Fetter developed Class C WD limits for 101 isotopes of interest to fusion. 19 isotopes have range of limits rather than single value due to uncertainties in corrosion assumptions. Those beta emitters are: $^{14}C$, $^{32}Si$, $^{36}Cl$, $^{41}Ca$, $^{63}Ni$, $^{79}Se$, $^{90}Sr$, $^{97}Tc$, $^{98}Tc$, $^{99}Tc$, $^{107}Pd$, $^{129}I$, $^{151}Sm$, $^{148}Gd$, $^{150}Gd$, $^{154}Dy$, $^{210}Pb$, $^{226}Ra$, $^{227}Ac$
    - Fetter-L and Fetter-H WDRs are calculated using Fetter’s low and high limits, respectively.
    - Fetter-L limits were not considered in previous ARIES designs
    - Which limit should we continue to report, Fetter-H or Fetter-L? (Fetter-L is more conservative)
Fetter’s Waste Disposal Rating

<table>
<thead>
<tr>
<th>Inboard Components:</th>
<th>Fetter-H</th>
<th>Fetter-L</th>
</tr>
</thead>
<tbody>
<tr>
<td>FW/B</td>
<td>0.015</td>
<td>0.017</td>
</tr>
<tr>
<td>HT Shield</td>
<td>0.45</td>
<td>0.47</td>
</tr>
<tr>
<td>V.V.</td>
<td>0.03</td>
<td>0.06</td>
</tr>
<tr>
<td>Magnet</td>
<td>0.015</td>
<td>0.024</td>
</tr>
<tr>
<td>Bucking Cylinder</td>
<td>0.005</td>
<td>0.01</td>
</tr>
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</table>

<table>
<thead>
<tr>
<th>Outboard Components:</th>
<th>Fetter-H</th>
<th>Fetter-L</th>
</tr>
</thead>
<tbody>
<tr>
<td>FW/B-I</td>
<td>0.072</td>
<td>0.076</td>
</tr>
<tr>
<td>B-II</td>
<td>0.001</td>
<td>0.18</td>
</tr>
<tr>
<td>HT Shield</td>
<td>0.15</td>
<td>0.23</td>
</tr>
<tr>
<td>V.V.</td>
<td>0.03</td>
<td>0.04</td>
</tr>
<tr>
<td>Magnet</td>
<td>0.03</td>
<td>0.04</td>
</tr>
</tbody>
</table>

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

Based on Fetter’s limits, all components qualify as Class C LLW @ EOL

* 4 cm thick vertical stabilizing shell
# NRC Waste Disposal Rating

<table>
<thead>
<tr>
<th>NRC Inboard Components:</th>
<th>NRC Class A Limits</th>
<th>NRC Class C Limits</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>w/o Shells</td>
<td>with W Shells</td>
</tr>
<tr>
<td>FW/B</td>
<td>3.8</td>
<td>0.017</td>
</tr>
<tr>
<td>HT Shield</td>
<td>63</td>
<td>61</td>
</tr>
<tr>
<td>V.V.</td>
<td>0.08</td>
<td>0.006</td>
</tr>
<tr>
<td>Magnet</td>
<td>0.08</td>
<td>0.007</td>
</tr>
<tr>
<td>Bucking Cylinder</td>
<td>0.05</td>
<td>0.004</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>NRC Outboard Components:</th>
<th>NRC Class A Limits</th>
<th>NRC Class C Limits</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>w/o Shells</td>
<td>with W Shells</td>
</tr>
<tr>
<td>FW/B-I</td>
<td>10</td>
<td>0.03</td>
</tr>
<tr>
<td>B-II</td>
<td>1.3</td>
<td>3.1</td>
</tr>
<tr>
<td>HT Shield</td>
<td>4.7</td>
<td></td>
</tr>
<tr>
<td>V.V.</td>
<td>1.9</td>
<td>0.02</td>
</tr>
<tr>
<td>Magnet</td>
<td>0.3</td>
<td>0.009</td>
</tr>
</tbody>
</table>

- NRC-A WDR reported at shutdown and varies with time after shutdown
- For SiC components, T and C^{14} are dominant nuclides for NRC-A WDR and C^{14} is dominant nuclide for NRC-C WDR
  \[ C^{12} (n, \gamma) \rightarrow C^{13} (n, \gamma) \rightarrow C^{14} \]

Based on NRC limits, all components qualify as Class C LLW @ EOL
- Thicker OB blanket (45 cm) will reduce OB HT shield WDR by factor $> 2$

Based on NRC limits, all IB & OB components except IB HT shield and OB B-II qualify as Class A LLW after 50-100 y storage period
IB & OB Compacted Waste Volumes

* 45 cm thick OB B-II and 15 cm thick OB HT shield. 8m height for all components except IB blanket (6 m)
Total Compacted Waste Volumes

<table>
<thead>
<tr>
<th>Component</th>
<th>IB &amp; OB</th>
<th>All Components</th>
</tr>
</thead>
<tbody>
<tr>
<td>IB &amp; OB Blanket-I</td>
<td>287</td>
<td>287 (22%)</td>
</tr>
<tr>
<td>OB Blanket-II</td>
<td>33</td>
<td>33 (3%)</td>
</tr>
<tr>
<td>Shield</td>
<td>95</td>
<td>340 (27%)</td>
</tr>
<tr>
<td>V.V.</td>
<td>75</td>
<td>120 (9%)</td>
</tr>
<tr>
<td>Magnets &amp; SS</td>
<td>75</td>
<td>200 (16%)</td>
</tr>
<tr>
<td>Structure</td>
<td>150</td>
<td>150 (12%)</td>
</tr>
<tr>
<td>Cryostat</td>
<td>140</td>
<td>140 (11%)</td>
</tr>
</tbody>
</table>

* 45 cm thick OB B-II and 15 cm thick OB HT shield
Class A and C Waste Volumes* for All Components

ARIES-AT generates 90% Class A LLW and 10% Class C LLW according to NRC limits

45 cm thick OB B-II and 15 cm thick OB HT shield
# OB B-II, IB HT shield, and divertor HT shield