ARIES-CS
Radial Build Definition and Nuclear System Characteristics

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Nuclear Areas of Research

Radial Build Definition:
- Dimension of all components
- Optimal composition

Neutron Wall Loading Profile:
- Toroidal & poloidal distribution
- Peak & average values

Blanket Parameters:
- Dimension
- TBR, enrichment, $M_n$
- Nuclear heat load
- Damage to FW
- Service lifetime

High-performance shielding module at $D_{\text{min}}$

Activation Issues:
- Activity and decay heat
- Thermal response to LOCA/LOFA events
- Radwaste management

Radiation Protection:
- Shield dimension & optimal composition
- Damage profile at shield, manifolds, VV, and magnets
- Streaming issues
Nuclear Task Involves Active Interaction with many Disciplines

- **Prelim. Physics**
  - (R, a, P_f, Δ_min, plasma contour, magnet CL)

- **NWL Profile**
  - (peak, average, ratio)
  - no Δ_min match
  - or insufficient breeding

- **1-D Nuclear Analysis**
  - (Δ_min, TBR, M_n, damage, lifetime)

- **Design Requirements**

- **Radial Build Definition**
  - @ Δ_min and elsewhere
  - (Optimal dimension and composition, blanket coverage, thermal loads)

- **3-D Neutronics**
  - (Overall TBR, M_n)

- **Blanket Concept**
- **Init. Magnet Parameters**
- **Init. Divertor Parameters**

- **Activation Assessment**
  - (Activity, decay heat, LOCA/LOFA, Radwaste classification)

- **Blanket Design**
- **Systems Code**
  - (R, a, P_f)

- **CAD Drawings**

- **Safety Analysis**
Stellarators Offer Unique Engineering Features and Challenges

- Minimum radial standoff at $\bar{D}_{\text{min}}$ controls machine size and cost.
  - Well optimized radial build particularly at $\bar{D}_{\text{min}}$

- Sizable components with low shielding performance (such as blanket and He manifolds) should be avoided at $\bar{D}_{\text{min}}$.

- Could design tolerate shield-only module (no blanket) at $\bar{D}_{\text{min}}$? Impact on TBR, overall size, and economics?

- Compactness mandates all components should provide shielding function:
  - Blanket protects shield
  - Blanket and shield protect manifolds and VV
  - Blanket, shield, and VV protect magnets

- Highly complex geometry mandates developing new approach to directly couple CAD drawings with 3-D neutronics codes.

- Economics and safety constraints control design of all components from beginning.
# ARIES-CS Requirements Guide

## In-vessel Components Design

<table>
<thead>
<tr>
<th>Requirement</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Overall TBR</strong></td>
<td>1.1</td>
</tr>
<tr>
<td>(for T self-sufficiency)</td>
<td></td>
</tr>
<tr>
<td><strong>Damage to Structure</strong></td>
<td>200 dpa - advanced FS</td>
</tr>
<tr>
<td>(for structural integrity)</td>
<td>3% burn up - SiC</td>
</tr>
<tr>
<td><strong>Helium Production @ Manifolds and VV</strong></td>
<td>1 appm</td>
</tr>
<tr>
<td>(for reweldability of FS)</td>
<td></td>
</tr>
<tr>
<td><strong>S/C Magnet (@ 4 K):</strong></td>
<td></td>
</tr>
<tr>
<td>Fast $n$ fluence to Nb$_3$Sn ($E_n &gt; 0.1$ MeV)</td>
<td>$10^{19}$ n/cm$^2$</td>
</tr>
<tr>
<td>Nuclear heating</td>
<td>2 mW/cm$^3$</td>
</tr>
<tr>
<td>dpa to Cu stabilizer</td>
<td>$6 \times 10^{-3}$ dpa</td>
</tr>
<tr>
<td>Dose to electric insulator</td>
<td>$&gt; 10^{11}$ rads</td>
</tr>
<tr>
<td><strong>Machine Lifetime</strong></td>
<td>40 FPY</td>
</tr>
<tr>
<td><strong>Availability</strong></td>
<td>85%</td>
</tr>
</tbody>
</table>
Reference Dual-cooled LiPb/FS Blanket Selected with Advanced LiPb/SiC as Backup

<table>
<thead>
<tr>
<th>Breeder</th>
<th>Multiplier</th>
<th>Structure</th>
<th>FW/Blanket Coolant</th>
<th>Shield Coolant</th>
<th>VV Coolant</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Internal VV:</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Flibe</td>
<td>Be</td>
<td>FS</td>
<td>Flibe</td>
<td>Flibe</td>
<td>H₂O</td>
</tr>
<tr>
<td>LiPb (backup)</td>
<td>–</td>
<td>SiC</td>
<td>LiPb</td>
<td>LiPb</td>
<td>H₂O</td>
</tr>
<tr>
<td>LiPb (reference)</td>
<td>–</td>
<td>FS</td>
<td>He/LiPb</td>
<td>He</td>
<td>H₂O</td>
</tr>
<tr>
<td>Li₄SiO₄</td>
<td>Be</td>
<td>FS</td>
<td>He</td>
<td>He</td>
<td>H₂O</td>
</tr>
</tbody>
</table>

| **External VV:** |          |           |                    |                |            |
| LiPb          | –         | FS        | He/LiPb            | He or H₂O      | He         |
| Li            | –         | FS        | He/Li              | He             | He         |
FW Shape Varies Toroidally and Poloidally - a Challenging 3-D Modeling Problem

$3 \text{ FP}$

$R = 8.25 \text{ m}$
UW Developed CAD/MCNP Coupling Approach to Model ARIES-CS for Nuclear Assessment

- Only viable approach for ARIES-CS 3-D neutronics modeling.
- Geometry and ray tracing in CAD; radiation transport physics in MCNPX.
- This unique, superior approach gained international support.
- Ongoing effort to benchmark it against other approaches developed abroad (in Germany, China, and Japan).
- DOE funded UW to apply it to ITER - relatively simple problem.
3-D Neutron Wall Loading Profile Using CAD/MCNP Coupling Approach

- \( R = 8.25 \) m design.
- Neutrons tallied in discrete bins:
  - Toroidal angle divided every 7.5°.
  - Vertical height divided into 0.5 m segments.
- Peak \( G \sim 3 \) MW/m\(^2\) at OB midplane of \( \theta = 0° \)
- Peak to average \( G = 1.52 \)
Novel Shielding Approach Helps Achieve Compactness

**Benefits:**
- Compact radial build at $d_{\text{min}}$
- Small R and low $B_{\text{max}}$
- Low COE.

**Challenges:**
- Integration of shield-only and transition zones with surrounding blanket.
- Routing of coolants to shield-only zones.
- Higher WC decay heat compared to FS.
Toroidal / Radial Cross Section

(R = 8.25 m )

Vacuum Vessel

LiPb & He Manifolds

FS-Shield

Back Wall

Blanket

Non-uniform Blanket

WC-Shield-II

WC-Shield-I

Divertor System
(10% of FW area)

Plasma

Nominal Blanket/shield Zone (~80%)

Transition Region (~13.5%)

WC-Shield only Zone (~6.5%)

D_{min} = 119 cm
Radial Build Satisfies Design Requirements

(3 MW/m² peak □)

Replaceable

Permanent Components

Thickness (cm)

Blanket/ Shield Zone

Thickness (cm)

Shield Only Zone

@ □_{min}

D_{min} = 119 cm
Blanket Design Meets Breeding Requirement

- Local TBR approaches 1.3
- 3-D analysis confirmed 1-D local TBR estimate for **full blanket coverage**.
- Uniform and non-uniform blankets sized to provide **1.1 overall TBR** based on 1-D results combined with blanket coverage. To be confirmed with detailed 3-D model.
Preliminary 3-D Results Using CAD/MCNP Coupling Approach

| Model* | 1-D | 3-D | ±
|--------|-----|-----|-----
| Local TBR | 1.285 | 1.316 | 0.61%
| Energy multiplication ($M_n$) | 1.14 | 1.143 | 0.49%
| Peak dpa rate (dpa/FPY) | 40 | 39.4 | 4.58%
| FW/B lifetime (FPY) | 5 | 5.08 | 4.58%
| Nuclear heating (MW): | | | |
| FW | 156 | 145.03 | 1.33%
| Blanket | 1572 | 1585.03 | 1.52%
| Back wall | 13 | 9.75 | 6.45%
| Shield | 71 | 62.94 | 2.73%
| Manifolds | 18 | 19.16 | 5.49%
| Total | 1830 | 1821.9 | 0.49%


Future 3-D analysis will include blanket variation, divertor system and penetrations to confirm 1.1 overall TBR and $M_n$. 
He Access Tubes Raise Streaming Concern

- **Blanket/Shield Zone**
  - Thickness (cm)
  - Blanket
  - Shield-only or Transition Region
  - Magnet
  - VV

- **Manifolds**
  - 25 cm Breeding Zone-I
  - 25 cm Breeding Zone-II
  - 0.5 cm SiC Insert
  - 1.5 cm FS/He

- **Local Shield**
  - He Tube (32 cm ID)
  - 5 cm BW

- **Vacuum Vessel**
  - 3.8 cm Blanket
  - 3.8 cm Shield-only
  - 10 cm External Structure

- **Magnet**
  - 3 cm FW
  - 3 cm BW

- **Coil Case & Insulator**
  - 2.2 cm

- **Winding Pack**
  - 30 cm

- **Gap + Th. Insulator**
  - >2 cm

- **Gap**
  - 2 cm

- **FW**
  - 5 cm

- **SOL**
  - 3.8 cm

- **28 cm**

- **17 cm**

- **38 cm**

- **min = 119 cm**

- **> 181 cm**
Local Shield Helps Solve Neutron Streaming Problem

Ongoing 3-D analysis will optimize dimension of local shield

Hot spots at VV and magnet
# Key Design Parameters for Economic Analysis

<table>
<thead>
<tr>
<th>Parameter</th>
<th>LiPb/FS/He (reference)</th>
<th>LiPb/SiC (backup)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\Delta_{\text{min}}$</td>
<td>1.19</td>
<td>1.14</td>
</tr>
<tr>
<td>Overall TBR</td>
<td>1.1</td>
<td>1.1</td>
</tr>
<tr>
<td>Energy Multiplication ($M_n$)</td>
<td>1.14</td>
<td>1.1</td>
</tr>
<tr>
<td>Thermal Efficiency ($\vartheta_{\text{th}}$)</td>
<td>40-45%*</td>
<td>55-63%*</td>
</tr>
<tr>
<td>FW Lifetime (FPY)</td>
<td>5</td>
<td>6</td>
</tr>
<tr>
<td>System Availability</td>
<td>~85%</td>
<td>~85%</td>
</tr>
</tbody>
</table>

* Depending on peak $\Delta$.

Integral system analysis will assess impact of $\Delta_{\text{min}}$, $M_n$, and $\vartheta_{\text{th}}$ on COE.
Activation Assessment

• **Main concerns:**
  
  – **Decay heat** of FS and WC components.
  
  – **Thermal response** of blanket/shield during LOCA/LOFA.
  
  – **Waste classification:**
    
    - Low or high level waste?
    
    - Any cleared metal?
  
  – **Radwaste stream.**
Decay Heat

WC decay heat dominates at 3 h after shutdown
Thermal Response during LOCA/LOFA Event

Nominal Blanket/Shield
\[ T_{\text{max}} \sim 711 \, ^\circ\text{C} \]

Shield-only Zone
\[ T_{\text{max}} \sim 1060 \, ^\circ\text{C} \]

- 3-coolant system \[ \square \] rare LOCA/LOFA event \(< 10^{-8}/\text{y}\)
- Severe accident scenario: He and LiPb LOCA in all blanket/shield modules & water LOFA in VV.
- For blanket, FW temperature remains below 740 \(^\circ\text{C}\) limit.
- WC shield modules may need to be replaced. More realistic accident scenario will be assessed.
Waste Management Approach

**Options:**
- **Disposal** in LLW or HLW repositories
- **Recycling** – reuse within nuclear facilities
- **Clearing** – release to commercial market, if CI < 1.

**Repository capacity is limited:**
- **Recycling** should be top-level requirement for fusion power plants
- **Transmute** long-lived radioisotopes in special module
- **Clear** majority of activated materials to minimize waste volume.
ARIES-CS Generates Only Low-Level Waste

<table>
<thead>
<tr>
<th>Component</th>
<th>WDR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Replaceable Components:</td>
<td></td>
</tr>
<tr>
<td>FW/Blanket-I</td>
<td>0.4</td>
</tr>
<tr>
<td>WC-Shield-I</td>
<td>0.9</td>
</tr>
<tr>
<td>Permanent Components:</td>
<td></td>
</tr>
<tr>
<td>WC-Shield-II</td>
<td>0.3</td>
</tr>
<tr>
<td>FS Shield</td>
<td>0.7</td>
</tr>
<tr>
<td>Vacuum Vessel</td>
<td>0.05</td>
</tr>
<tr>
<td>Magnet</td>
<td>&lt; 1</td>
</tr>
<tr>
<td>Confinement building</td>
<td>&lt;&lt; 0.1</td>
</tr>
</tbody>
</table>

LLW (WDR < 1) qualifies for near-surface disposal or, preferably, recycling
Majority of Waste (74%) can be Cleared from Regulatory Control

In-vessel components cannot be cleared
Building Constituents can be Cleared 1-4 y after Plant Decommissioning

Building composition: 15% Mild Steel, 85% Concrete
Stellarators Generate Large Radwaste Compared to Tokamaks

Means to reduce ARIES-CS radwaste are being pursued (more compact machine with less coil support and bucking structures)
Well Optimized Radial Build Contributed to Compactness of ARIES-CS

Over past 25 y, stellarator major radius more than halved by advanced physics and technology, dropping from 24 m for UWTOR-M to 7-8 m for ARIES-CS, approaching R of advanced tokamaks.
Concluding Remarks

• **Novel shielding approach** developed for ARIES-CS. Ongoing study is assessing benefits and addressing challenges.

• **CAD/MCNP coupling approach** developed specifically for ARIES-CS 3-D neutronics modeling.

• Combination of shield-only zones and non-uniform blanket presents **best option** for ARIES-CS.

• Proposed **radial build** satisfies design requirements.

• **No major activation problems** identified for ARIES-CS.

• At present, ongoing ARIES-CS study is examining more compact design (R < 8 m) that needs further assessment.
ARIES-CS Publications


