Overview of ARIES-CS In-vessel Components: Integration of Nuclear, Economics, and Safety Constraints in Compact Stellarator Design

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Vienna, Austria
Multi-Institution ARIES Project
Six Stellarator Power Plants Developed Worldwide Over Past 25 y

- ARIES-CS Compact Stellarator
  - US
- HSR Helias Stellarator Reactor
  - Germany
- FFHR Force Free Helical Reactor
  - Japan
- SPPS Stellarator Power Plant Study
  - US
- ASRA6C Modular Advanced Stellarator Reactor
  - US
- UWTOR-M Modular Stellarator Power Reactor
  - US

Calendar year
Six Stellarator Power Plants Developed Worldwide Over Past 25 y (Cont.)

- UWTOR-M
- SPPS
- ASRA-6C
- FFHR
- HSR
- ARIES-CS
Stellarators Offer Unique Features and Engineering Challenges

**Advantages:**
- Inherently steady-state devices
- No need for large plasma current
- No external current drive
- No risk of plasma disruptions
- Low recirculating power due to absence of current-drive requirements
- No instability and positional control systems.

**Challenges:**
- Complex geometry
- Maintainability and component replacement
- Highly constrained local shielding areas
- 3-D modeling
- Managing large volume of active materials.
ARIES Compact Stellarator

Study aimed at reducing stellarators’ size by:
- Developing compact configuration with advanced physics & technology
- Optimizing minimum plasma-coil distance ($\Delta_{\text{min}}$) through rigorous nuclear assessment.

### 3 Field Periods Configuration

- **Average Major Radius**: 7.75 m
- **Average Minor Radius**: 1.7 m
- **Aspect Ratio**: 4.5
- **Fusion Power**: 2400 MW
- **Average NWL**: 2.6 MW/m²
- **Net Electric Power**: 1000 MW$_e$
- **COE ($\text{2004}$)**: 78 mills/kWh
ARIES-CS Nuclear Areas of Research

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

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

**High-Performance Shielding Module at $\Delta_{\text{min}}$**

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

**Activation Issues:**
- Activity and decay heat
- Thermal response during LOCA/LOFA events
- Radwaste classification & management

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

1-D Nuclear Analysis
($\Delta_{\text{min}}$, TBR, $M_n$, damage, lifetime)

Radial Build Definition
@$\Delta_{\text{min}}$ 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

Prelim. Physics
($R$, $a$, $P_f$, $\Delta_{\text{min}}$, plasma contour, magnet CL)

NWL Profile
($\Gamma$ peak, average, ratio)

no $\Delta_{\text{min}}$ match
or insufficient breeding

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

Design Requirements

Blanket Design

Systems Code
($R$, $a$, $P_f$)

CAD Drawings

Safety Analysis
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><strong>LiPb (reference)</strong></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>
<tr>
<td><strong>External VV</strong>:</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LiPb</td>
<td>–</td>
<td>FS</td>
<td>He/LiPb</td>
<td>He or H₂O</td>
<td>He</td>
</tr>
<tr>
<td>Li</td>
<td>–</td>
<td>FS</td>
<td>He/Li</td>
<td>He</td>
<td>He</td>
</tr>
</tbody>
</table>

* VV inside magnets.
# VV outside magnets.
## ARIES-CS Requirements Guide
### In-vessel Component Design

<table>
<thead>
<tr>
<th>Calculated Overall TBR</th>
<th>1.1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Net TBR (for T self-sufficiency)</td>
<td>~1.01</td>
</tr>
</tbody>
</table>

### Damage to Structure
(for structural integrity)

<table>
<thead>
<tr>
<th>Damage to Structure</th>
<th>200 dpa - advanced FS</th>
</tr>
</thead>
</table>

### Helium Production @ Manifolds and VV
(for reweldability of FS)

<table>
<thead>
<tr>
<th>Helium Production</th>
<th>1 He appm</th>
</tr>
</thead>
</table>

### S/C Magnet (@ 4 K):

<table>
<thead>
<tr>
<th>S/C Magnet (@ 4 K):</th>
<th>Peak Fast $n$ fluence to Nb$_3$Sn ($E_n &gt; 0.1$ MeV)</th>
<th>$10^{19}$ n/cm$^2$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Peak Nuclear heating</td>
<td>$2$ mW/cm$^3$</td>
</tr>
<tr>
<td></td>
<td>Peak dpa to Cu stabilizer</td>
<td>$6\times10^{-3}$ dpa</td>
</tr>
<tr>
<td></td>
<td>Peak Dose to electric insulator</td>
<td>$&lt; 10^{11}$ rads</td>
</tr>
</tbody>
</table>

### Plant Lifetime

<table>
<thead>
<tr>
<th>Plant Lifetime</th>
<th>40 FPY</th>
</tr>
</thead>
</table>

### Availability

<table>
<thead>
<tr>
<th>Availability</th>
<th>85%</th>
</tr>
</thead>
</table>

### Operational dose to workers and public

<table>
<thead>
<tr>
<th>Operational dose to workers and public</th>
<th>$&lt; 2.5$ mrem/h</th>
</tr>
</thead>
</table>
FW ShapeVaries Toroidally and Poloidally: Challenging 3-D Modeling Problem
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.
Neutron Wall Loading Distribution

<table>
<thead>
<tr>
<th>Peak (Min) [MW/m²]</th>
<th>Toroidal Angle (degrees)</th>
<th>Poloidal Angle (degrees)</th>
</tr>
</thead>
<tbody>
<tr>
<td>5.26 (0.32)</td>
<td>-11 (-4)</td>
<td>-18 (-116)</td>
</tr>
</tbody>
</table>

Peak/Ave. NWL = 2
Well-Optimized Blanket & Shield Protect Vital Components
(5.3 MW/m² Peak $\Gamma$)

Replaceable FW/Blkt/BW

<table>
<thead>
<tr>
<th>Thickness (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Full Blanket &amp; Shield</td>
</tr>
<tr>
<td>Non-uniform Blanket &amp; Shield @ $\Delta_{min}$</td>
</tr>
</tbody>
</table>

$\Delta \geq 179$ cm
High Performance Components at $\Delta_{\text{min}}$ Help Achieve Compactness, Minimize Major Radius, and Enhance Economics

- Non-uniform, Tapered Blanket/Shield (24% of FW area)
- Full Blanket/Shield and Divertor (61%+15%= 76% of FW area)

$\Delta_{\text{min}} = 130.7\,\text{cm}$
• Large machines breed more T as non-uniform blanket coverage decreases with R.
• Designs with \( R < 7.5 \) m will not provide T self-sufficiency.
R=7.75 m Reference Design Provides Tritium Self-Sufficiency

**3-D model** includes essential components for TBR:
- Non-uniform and full blanket/shield
- Homogenized: FW/Blanket/BW Shield
  Manifolds
  Divertor.

Calculated Overall **TBR = 1.1**
with 70% Li enrichment
Neutron Streaming Through Penetrations Compromises Shielding Performance

- **7 types of penetrations:**
  - 198 He tubes for blanket (32 cm ID)
  - **24 Divertor He access pipes (30-60 cm ID)**
  - 30 Divertor pumping ducts (42 x 120 cm each)
  - 12 Large pumping ducts (1 x 1.25 m each)
  - 3 ECH ducts (24 x 54 cm each).
  - 6 main He pipes - HX to/from blanket (72 cm ID each)
  - 6 main He pipes - HX to/from divertor (70 cm ID each)

- **Potential solutions:**
  - Local shield behind penetrations
  - He tube axis oriented toward lower neutron source
  - Penetration shield surrounding ducts
  - Replaceable shield close to penetrations
  - Avoid rewelding VV and manifolds close to penetrations
  - Bends included in some penetrations.
3-D Assessment of Streaming Through Divertor He Access Pipe

Attila 3-D Model

Shield inserts help protect surrounding components
# Key Nuclear Parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Peak NWL</td>
<td>5.3 MW/m²</td>
</tr>
<tr>
<td>Average NWL</td>
<td>2.6 MW/m²</td>
</tr>
<tr>
<td>Peak to Average NWL</td>
<td>2</td>
</tr>
<tr>
<td>Calculated Overall TBR</td>
<td>1.1 with 70% Li enrichment</td>
</tr>
<tr>
<td>Net TBR</td>
<td>~1.01</td>
</tr>
<tr>
<td>FW/blanket Lifetime</td>
<td>3 FPY</td>
</tr>
<tr>
<td>Shield/manifold/VV/magnet Lifetime</td>
<td>40 FPY</td>
</tr>
<tr>
<td>Overall Energy Multiplication</td>
<td>1.16</td>
</tr>
<tr>
<td>$\Delta_{\text{min}}$</td>
<td>1.3 m</td>
</tr>
<tr>
<td>$\Delta_{\text{max}}$</td>
<td>1.8 m</td>
</tr>
</tbody>
</table>
ARIES-CS Major Radius Approaches R of Advanced Tokamaks

Well optimized radial build along with advanced physics and technologies helped reduce ARIES-CS size
ARIES Project Committed to Radwaste Minimization

Stellarator waste volume dropped by 3-fold over 25 y study period

* Actual volumes (not compacted, no replacements).
Highlights of ARIES-CS
Safety Features

Environmental impact:
- **Low activation materials** with strict impurity control
  ⇒ minimal long-term environmental impact.
- **No high-level waste.**
- **Minimal radioactive releases** during normal and abnormal operations.

No energy and pressurization threats to confinement barriers (VV and cryostat):
- Decay heat problem solved by design
- Chemical reaction avoided
- No combustible gas generated
- Chemical energy controlled by design
- Overpressure protection system
- Rapid, benign plasma shutdown.

Occupational and public safety:
- **No evacuation plan** following abnormal events (early dose at site boundary < 1 rem*) to avoid disturbing public daily life.
- **Low dose** to workers and personnel during operation and maintenance activity (< 2.5 mrem/h*).
- **Public safety** during normal operation (bio-dose << 2.5 mrem/h*) and following credible accidents:
  - External events (seismic, hurricanes, tornadoes, etc.).
  - LOCA, LOFA, LOVA, and by-pass events.

# Such as T, volatile activated structure, corrosion products, and erosion dust. Or, from liquid and gas leaks.
* 1 rem (= 10 m Sv) accident dose stated in Fusion Safety Standards, DOE report, DOE-STD-6002-96 (1996).
In-vessel Components Exhibit Structural Integrity during LOCA/LOFA Event

- **Design Base Accident scenario**: He LOCA and LiPb LOFA in all modules and water LOFA in VV.
- Plasma stays on for 3 seconds after onset of LOCA/LOFA.
- Peak FW temperature remains below 740°C – reusability limit for ferritic steel.
Radwaste Management Approach

• Three options examined:
  – Disposal in repositories: LLW (WDR < 1)
  – Recycling – reuse within nuclear facilities (dose < 10,000 Sv/h)
  – Clearance – release slightly-radioactive materials to commercial market if CI < 1.

• Lack of geological repositories and tighter environmental controls will force fusion designers to promote recycling and clearance, avoiding disposal*  
  ⇒ minimize radwaste burden for future generations.

• There’s growing international effort in support of this new trend.

* L. El-Guebaly, “Environmental Aspects of Recent Trend in Managing Fusion Radwaste: Recycling and Clearance, Avoiding Disposal,” This IAEA TM, Wednesday @ 9 AM.
## Comparison Between Reference and Backup Systems

<table>
<thead>
<tr>
<th></th>
<th>LiPb/He/FS</th>
<th>LiPb/SiC</th>
</tr>
</thead>
<tbody>
<tr>
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<td>1.1</td>
<td>1.1</td>
</tr>
<tr>
<td>FW/blanket lifetime</td>
<td>3 FPY</td>
<td>3.4 FPY</td>
</tr>
<tr>
<td>Overall energy multiplication</td>
<td>1.16</td>
<td>1.1</td>
</tr>
<tr>
<td>(\eta_{th})</td>
<td>42%</td>
<td>56%</td>
</tr>
<tr>
<td>Structure unit cost*</td>
<td>103 $/kg</td>
<td>510 $/kg</td>
</tr>
<tr>
<td>Blanket/divertor/shield/manifolds cost*</td>
<td>$288M</td>
<td>$282M</td>
</tr>
<tr>
<td>Cost* of heat transfer/transport system</td>
<td>$475M</td>
<td>$175M</td>
</tr>
<tr>
<td>Pumping power</td>
<td>183 MW(_e)</td>
<td>---</td>
</tr>
<tr>
<td>LSA factor</td>
<td>2</td>
<td>1</td>
</tr>
<tr>
<td><strong>Cost of Electricity</strong>*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reference design (R=7.75 m)</td>
<td>78 mills/kWh</td>
<td>60 mills/kWh</td>
</tr>
<tr>
<td>Full blanket/shield everywhere (R=10.1 m)</td>
<td>87 mills/kWh</td>
<td></td>
</tr>
</tbody>
</table>

* in 2004 \$. 

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Conclusions

• Nuclear assessment received considerable attention during ARIES-CS design process.

• First time ever complex stellarator geometry modeled for nuclear assessment using UW newly developed CAD/MCNP coupling approach.

• Radial build satisfies design requirements in terms of breeding sufficient tritium and shielding vital components.

• Novel shielding approach developed for ARIES-CS helped reduce radial standoff by 40%, major radius by 30%, and overall cost by 10%.

• ARIES-CS demonstrates adequate performance in several safety and environmental areas.

• Successful integration of well-optimized radial build into final design, along with carefully selected engineering parameters and overarching safety and environmental constraints, delivered attractive and truly compact stellarator power plant.