Within the ARIES-CS project, design activities have focused on developing the first compact device that enhances the attractiveness of the stellarator as a power plant. The objectives of this paper are to review the nuclear elements that received considerable attention during the design process and provide a perspective on their successful integration into the final design. Among these elements are the radial build definition, the well-optimized in-vessel components that satisfy the ARIES top-level requirements, the carefully selected nuclear and engineering parameters to produce an economic optimum, the modeling—for the first time ever—of the highly complex stellarator geometry for the three-dimensional nuclear assessment, and the overarching safety and environmental constraints to deliver an attractive, reliable, and truly compact stellarator power plant.

KEYWORDS: stellarator, nuclear analysis, fusion power plants

Note: Some figures in this paper are in color only in the electronic version.

I. INTRODUCTION

In recent years the stellarator concept has emerged as a competitive source of fusion energy, offering a steady-state, nondisruptive operation. The most recent development of compact stellarators has led to the construction of the National Compact Stellarator Experiment (NCSX) at the Princeton Plasma Physics Laboratory and the three-year power plant study of ARIES-CS (Ref. 2). During the ARIES-CS design process, the principle of compactness drove the physics, engineering, and economics. The three design disciplines proceeded interactively and the systems code determined the reference parameters. The code varied the physics and engineering parameters, subject to preassigned physics and technology limits, to produce an economic optimum. Basically, the ARIES-CS study aimed at reducing the stellarator size by

1. developing a compact configuration with low aspect ratio (≈4.5) and combination of advanced physics and technology

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2. optimizing the minimum plasma-coil distance ($\Delta_{\text{min}}$) through rigorous nuclear assessment since $\Delta_{\text{min}}$ significantly impacts the overall size and cost.

An integral approach considering the design configuration, materials choice, design requirements, and component optimization was deemed necessary during the ARIES-CS design process. Based on lessons learned from previous ARIES studies, special care was taken in selecting the materials so that even in the unlikely accident events, any environmental or release effect would be minimal or nonexistent. The collective impact of materials on the activation, safety, and waste management characteristics has done much to shape the design development during the three-year period of the study. As such, the nuclear analyses (neutronics, shielding, and activation) have been a fundamental element of the ARIES-CS study and have received considerable attention. Out of numerous nuclear-related questions, we identified the following top five sets of questions based on how fundamental they are to the stellarator concept and how much of an impact their solutions will have on the overall design:

1. How do the neutron wall loading (NWL) and radiation heating vary poloidally and toroidally? Where do they peak? Do the peaks occur at the same blanket module? What is the peak-to-average NWL ratio?

2. How compact can the design be? How far can we push $\Delta_{\text{min}}$? Can the design tolerate a nonbreeding blanket at $\Delta_{\text{min}}$? What other components can be excluded at $\Delta_{\text{min}}$ to reduce its size? What is the impact of a truly compact $\Delta_{\text{min}}$ on the overall tritium breeding, machine size, and economics? Would a design with uniform blanket/shield everywhere be equally attractive?

3. What role does the overall tritium breeding ratio (TBR) play in defining the machine average major radius ($\langle R \rangle$)?

4. How can the complex stellarator configuration be modeled for the three-dimensional (3-D) nuclear analysis without geometrical approximations? Can the actual source strength be represented in real 3-D space?

5. Does a solution exist for the large radwaste volume generated by stellarators? How can the radwaste volume be minimized by design? Can the geological disposal option be avoided and replaced by recycling/clearance?

Stellarators promise disruption-free, steady-state operation with reduced recirculating power due to the absence of current drive requirements. However, such advantages could be offset by the challenging engineering issues. Stellarators are quite complex machines. In ARIES-CS, the first wall and surrounding in-vessel components conform to the plasma, as shown in Fig. 1, and deviate from the uniform toroidal shape in order to achieve compactness. Within each field period that covers 120 deg toroidally, the configuration changes from a bean shape at 0 deg to a D shape at 60 deg, then back to a bean shape at 120 deg, continually switching the surfaces from convex...
to concave over a toroidal length of \( \sim 17 \) m. Figure 2 displays nine cross sections over half a field period, showing the plasma boundary and the midcoil filament. This means the first-wall and in-vessel component shapes vary toroidally and poloidally, representing a challenging 3-D modeling problem. In each field period, there are four critical regions of \( \Delta_{\text{min}} \) (at \( \sim 11 \) and 33 deg in Fig. 2) where the magnets move closer to the plasma, constraining the space between the plasma edge and midcoil. \( \Delta_{\text{min}} \) should accommodate the scrape-off layer (SOL), first wall, blanket, shield, vacuum vessel, assembly gaps, coils case, and half of the winding pack. The penalty associated with increasing \( \Delta_{\text{min}} \) by 10 cm is 50 to 60 cm in the major radius and 1 to 2 mills/kWh (in 2004 U.S. dollars) in the cost of electricity.\(^5\) Being the most influential parameter for the stellarator’s size and cost, \( \Delta_{\text{min}} \) optimization was crucial to the overall design. An innovative approach was developed to downsize the blanket at \( \Delta_{\text{min}} \) and utilize a highly efficient tungsten carbide (WC)-based shield. This approach placed a premium on the full blanket to supply the majority of the tritium needed for plasma operation.

Modeling ARIES-CS for the 3-D nuclear analysis was a challenging engineering task. A novel approach based on coupling the computer-aided design (CAD) model with the MCNPX Monte Carlo code was developed to model, for the first time ever, the complex stellarator geometry for nuclear assessments (Secs. II and IV). Because of the complexity of the 3-D geometry, we relied heavily at the early stages of the design on the

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**Fig. 2.** Nine plasma and midcoil cross sections covering one-half a field period. Dimensions are in meters. The toroidal angle (\( \Theta \)) is measured from the beginning of the field period.
simple one-dimensional (1-D) poloidal cylindrical model, using the average plasma minor radius to predict the overall TBR and $M_{R}$ (combining 1-D estimates with blanket coverage fractions) and the average NWL (from the systems code$^{6}$) with 1.5 to 1.6 peaking factor for the shielding analysis. As the design progressed, a 3-D analysis was judged essential to generate the exact NWL profile and peaking factor and to confirm the key nuclear parameters.

Accommodating the breeding blanket and necessary shield to protect the superconducting magnet at $\Delta_{\text{min}}$ represented another challenging task. Utilizing a nonuniform blanket combined with a highly efficient WC shield in this highly constrained region helped reduce the machine size and cost of the reference design (Secs. III and V). Because stellarators generate more radwaste than tokamaks, managing ARIES-CS active materials during operation and after plant decommissioning was essential for the environmental attractiveness of the machine (Sec. VII). Several additional nuclear-related tasks received considerable attention during the ARIES-CS design process. These included the radial build definition, the well-optimized in-vessel components that satisfied the top-level requirements, the streaming of neutrons through the helium access tubes and pipes, the carefully selected nuclear and engineering parameters to produce an economic optimum, and the overarching safety constraints to deliver a safe and reliable power plant. This paper describes the successful integration of the nuclear design elements into the reference three-field-period ARIES-CS design. Previous supporting nuclear analyses performed for interim designs are included in Refs. 4, 6, and 7.

At the outset, the design process itself took into consideration the fabricability, constructability, operability, and maintainability of the machine.$^{8,9}$ To ensure that the top-level requirements$^{10}$ are fully incorporated, a subset of nuclear-related requirements was established for the ARIES-CS design, as summarized in Table I. ARIES-CS has a fusion power level of $\sim 2400$ MW and delivers $1000$ MW(electric) net output power. As is discussed in Sec. IV, a calculated overall TBR of 1.1 ensures tritium self-sufficiency. An on-line adjustment of the $^{6}$Li enrichment for the LiPb breeder is necessary to adjust the net TBR during operation to $\sim 1.01$ in case of overbreeding or underbreeding blanket.$^{11}$ The life-limiting criteria for the structural components and magnets are key factors to accurately determine their lifetimes. We adopted high radiation limits in concert with similar ground rules considered in the past for advanced ARIES designs.$^{12-14}$ The nuclear heat leakage from the shield to the surroundings must remain below 1% to enhance the power balance. If there is a need to cut and reweld the manifolds and vacuum vessel, the helium production level should not exceed 1 atom part per million (ppm) at any time during operation. No high-level waste should be produced to avoid deep geological burial. The disposal option could be replaced with more environmentally attractive scenarios, such as recycling and clearance.

During the three-year period of the study, several blanket/shield systems were considered employing advanced ferritic steel (FS) structures (such as IEAMF82H) and SiC/SiC composites.$^{4,15-17}$ The list of candidates includes four liquid breeder–based systems and one solid breeder–based system:

1. self-cooled Flibe with beryllium multiplier and advanced oxide dispersion strengthening-FS structure
2. self-cooled LiPb with SiC/SiC composites

### TABLE I

<table>
<thead>
<tr>
<th>ARIES-CS Design Requirements and Radiation Limits</th>
</tr>
</thead>
<tbody>
<tr>
<td>Calculated overall TBR</td>
</tr>
<tr>
<td>Net TBR (for $T$ self-sufficiency)</td>
</tr>
<tr>
<td>Damage to structure (for structural integrity)</td>
</tr>
<tr>
<td>Helium production at manifolds and vacuum vessel (for reweldability of FS)</td>
</tr>
<tr>
<td>Superconducting magnet (at 4 K)</td>
</tr>
<tr>
<td>Peak fast $n$ fluence to Nb$<em>3$Sn ($E</em>{n} &gt; 0.1$ MeV)</td>
</tr>
<tr>
<td>Peak nuclear heating</td>
</tr>
<tr>
<td>Peak dpa to Cu stabilizer</td>
</tr>
<tr>
<td>Peak dose to electric insulator</td>
</tr>
<tr>
<td>Plant lifetime</td>
</tr>
<tr>
<td>Availability</td>
</tr>
<tr>
<td>Operational dose to workers and public</td>
</tr>
<tr>
<td>Radwaste level</td>
</tr>
<tr>
<td>Radwaste minimization approach</td>
</tr>
<tr>
<td>Recycling and/or clearance</td>
</tr>
</tbody>
</table>


3. dual-cooled LiPb (or Li) with He and FS structure
4. He-cooled Li$_4$SiO$_4$ with beryllium multiplier and FS structure.

Each blanket concept offers advantages and drawbacks. An integrated study with guidance from the nuclear analysis and blanket design identified the preferred concept (the dual-cooled LiPb/He/FS with 42% $\eta_{nb}$) and a more advanced LiPb/SiC concept as a backup. The rationale for the latter is that as new developments occur, a future hope is the prospect of using SiC/SiC composites as the main structure for a high-temperature blanket (>1000°C), offering high thermal conversion efficiency (56%) to enhance the economics.

The following sections document the detailed analyses and results for the FS-based LiPb/He/FS reference design and to a lesser extent for the SiC-based LiPb/SiC backup system. Throughout this study, the neutronics and shielding assessments were evaluated with the MCNPX 3-D Monte Carlo code and its IAEAFENDL-2 data library and the DANTSYS discrete ordinates transport code with the IAEA FENDL-2 175-neutron 42-gamma group coupled cross-section library. The activation results were computed with the ALARA pulsed activation code and the IAEA FENDL-2 175-neutron group transmutation cross-section library.

II. NWL AND RADIATION FLUX PROFILES

To take full advantage of the 3-D neutron transport modeling capability enabled by the CAD-based MCNPX-CGM tool, it is necessary to generate the neutron source with a full representation of the 3-D variations in the source intensity. Traditional methods of defining a neutron source for Monte Carlo analysis of a fusion system assume varying degrees of symmetry. For example, for tokomaks there is toroidal symmetry that can be treated by many codes, but stellarators do not have such symmetry. For ARIES-CS, not only is there no toroidal symmetry, but the complex mathematical source representation does not lend itself to a standard R-Z-$\Theta$ cylindrical mesh either. A new strategy for representing the 3-D variations in neutron source strength (i.e., fusion power) for use in Monte Carlo transport calculations was developed with three steps:

1. A regular, uniform mesh is generated in flux coordinate space ($\Theta$, $\Phi$, $s$), where $\Theta$ is the toroidal angle, $\Phi$ is the poloidal angle, and $s$ is the flux surface number. The Fourier description of the plasma flux surfaces can then be used to transform these points to a cylindrical coordinate system ($R$, $Z$, $\Theta$), in real space, which can then be transformed to a Cartesian coordinate system ($X$, $Y$, $Z$). Care must be taken to properly account for the fact that the hexahedra (“hexes”) adjacent to the magnetic axis are degenerate, manifested as prisms rather than true hexes.

2. The flux surfaces represent iso-surfaces of constant plasma temperature and density, and therefore constant fusion power and neutron source density. Therefore, since every point in the ($X$, $Y$, $Z$) mesh represents a point on a known flux surface, the neutron source density can be evaluated at each point and a hex-averaged source density can be determined. The relative probability that a source is born in a given hex is simply the product of the hex-averaged source density and the hex volume. A cumulative distribution function (CDF) over the entire set of hexes can be found from the normalized probabilities and provided to a Monte Carlo code for sampling. In particular, the mesh vertices and hex-CDF values are written to a file for use by the Monte Carlo code.

3. Within the Monte Carlo code, MCNPX-CGM in this case, finding the birth location of each source neutron begins by sampling the discrete CDF that represents the set of hex probabilities. Once the particular hex is known and its eight vertices have been identified (six vertices of hexes adjacent to the magnetic axis), the hex is sampled uniformly to determine a position ($X$, $Y$, $Z$) within that hex.

This scheme has proved useful in modeling the ARIES-CS source distribution (see Fig. 3) for the reference design ($R$ = 7.75 m) but could form the basis for an improved coupling of the neutron source to the transport calculation for all magnetically confined fusion systems. Ongoing improvements are being incorporated to allow higher-order mesh spacing, source averaging, and subhex position sampling.

II.A. NWL Distribution

As with all fusion systems, the NWL is a valuable quantity to assess the scaling of neutron responses on local system components. Calculating the spatial variations of the NWL for the complex geometry of the ARIES-CS system requires advanced neutronics methods. The CAD-based MCNPX-CGM tool provides just such a capability.

Using the Fourier description of the last closed magnetic surface (LCMS), a plasma surface for the ($R$) = 7.75 m reference design was created using the CUBIT solid modeling tool. From this, two separate geometries were generated to represent different offsets between the plasma and first wall. In one case, a 5-cm SOL was modeled by creating a first-wall surface with a 5-cm offset from the LCMS surface. In the other case, a 30-cm offset was used to model the regions of the first wall that would be moved farther away from the plasma. As is the convention for NWL calculations, only the uncollided neutron current is measured at the first wall. To accommodate this, the space outside the first-wall surface was defined to be a perfect absorber (zero importance in the vernacular of MCNPX).
For each geometry, one-third of the toroidal extent of
the first-wall surface, representing one field period, was
segmented using CUBIT to allow for spatial resolution
of the NWL calculation. In the toroidal direction, a spacing
of 7.5 deg gave 16 segments (see Fig. 2 for plasma cross
sections), and a spacing of 40 cm in the axial (Z) direction
gave a varying number of axial segments, depending on
the toroidal position and magnitude of the first-wall off-
set (see Fig. 4). This resulted in 352 segments for the
5-cm SOL case and 472 segments for the 30-cm SOL case.

The uncollided neutron current through each surface
was tallied using MCNPX-CGM (F1 tally) and divided
by the surface area for each segment, as reported by
CUBIT, normalized to the 2355-MW total fusion power
of the system and reported at the midpoint of each seg-
mant. Postprocessing, performed with MATLAB, mapped
those results onto a coordinate system based on the to-
roidal and poloidal angles (\( \Theta, \Phi \)) relative to the magnetic
axis. The results were interpolated onto a 600 \( \times \) 600 grid,
uniform in \( \Theta = [-60, 60 \text{ deg}] \) and \( \Phi = [-180, 180 \text{ deg}] \),
to generate two-dimensional NWL maps and extract po-
loidal variations in NWL with poloidal angle at specific
toroidal angles.

The NWL from the 3-D neutron source was analyzed
for both cases (5- and 30-cm SOLs), and contour maps
are shown in Fig. 5 for a fusion power of 2355 MW. Both
exhibit the same general distribution, with similar loca-
tions of the maximum and minimum NWL values. In
addition to reducing the peak NWL from 5.3 to 4.4 MW/
m\(^2\), the minimum NWL increases from 0.32 to 0.42 MW/
m\(^2\). In both cases, the peak is identified near the outboard
midplane (\( \Phi = -18 \text{ or } -25 \text{ deg} \)) at a toroidal angle of
-11 deg. However, given the statistical error of the re-
sults, it is more important to recognize the large area
within 10% of the peak NWL, extending approximately
60 deg (-30 to 30 deg) in the toroidal direction and
140 deg (-70 to 70 deg) in the poloidal direction. The
minimum NWL occurs near the divertor region at approximately ±120 deg in the poloidal direction and extending approximately 20 deg (−10 to 10 deg) in the toroidal direction. Figure 6 shows the detailed poloidal distributions of the NWL at various toroidal angles. In each case, the poloidal distribution at the location of the peak NWL (−11 deg) is included. Notice that the maximum NWL at a toroidal angle of 0 deg is close to the overall maximum, consistent with the earlier observation that a substantial wall area is exposed to an NWL near the maximum.

The reference configuration selected for ARIES-CS deviates somewhat from the standard practice of uniform SOLs. It calls for 5-cm SOL everywhere except at the divertor, where the SOL expands to 30 cm to maintain the heat flux at the divertor surface within a tolerable level. The first-wall area, including the divertor, is ~730 m², and for 1885-MW neutron power, the average NWL amounts to 2.6 MW/m². This means the peaktaverage NWL ratio is approximately 2. Of interest is the drop of this ratio with the plasma aspect ratio (defined as the average major radius divided by the average, circularized minor radius). Figure 7 displays the less steep variation for stellarators compared to tokamaks.

II.B. Radiative Heating Distribution

In addition to the NWL, the core radiation distribution on the first wall was calculated using the same methodology but using the source profiles for the 354-MW bremsstrahlung radiation being emitted by the plasma (refer to Fig. 3). Since this source distribution is much closer to a uniform distribution, the peak location moves to a toroidal angle of −34 deg at a poloidal angle of −17 deg. The maximum first-wall heat flux due to core radiation is 0.68 MW/m², and the average value is 0.48 MW/m². Figure 8 shows that the region where the maximum heating from core radiation occurs is different from the regions of maximum NWL. As with the NWL, Fig. 9 shows the poloidal distributions of the core radiative heating at a number of toroidal angles. The final design values for the first-wall heat flux should also include other smaller edge radiation components, resulting in overall
maximum and average first-wall heat flux values of 0.76
and 0.57 MW/m², respectively.5,17

III. RADIAL BUILD DEFINITION

The reference ARIES-CS design employs dual cool-
ants (LiPb and He) to recover the heat from the power-
producing components (first wall, blanket, shield, 
manifolds, and divertor). One of the advantages of using
dual coolants is to provide redundancy in case of acci-
dent and ultimately to protect the design from off-normal 
scenarios, such as loss of either coolant or flow events. 
Although the FS-based blanket17 is based on the same 
concept developed earlier by the ARIES team for the 
ARIES-ST spherical tokamak13 and later considered as 
an ITER blanket testing module by many ITER parties, 
the unique blanket safety features were thoroughly ex-
amined and analyzed to provide assurance of their 
effectiveness.26–28 A coolant with more efficient shield-
ing performance such as water26,27 was employed for the 
vacuum vessel—a nonproducing power component. Be-
cause of the high reliability of the vacuum vessel cooling 
system, water can flow naturally, carrying the decay heat 
out of the in-vessel components during accidents, en-
hancing the safety features of the design.26,27

The compactness of the machine mandates that all 
components (blanket, shield, manifolds, and vacuum ves-
sel) provide a shielding function. We started the nuclear 
assessment by defining the first-wall, blanket, and back 
wall parameters (thickness, composition, and ⁶Li enrich-
ment). Besides breeding sufficient tritium for plasma op-
eration and recovering >90% of the neutron energy, the 
blanket along with the first and back walls protects the 
shield [<200 displacements per atom (dpa)] for the en-
tire plant life [40 full-power years (FPY)]. Next, the 
shield was designed to protect the welds of the manifolds

Fig. 6. Poloidal distributions of NWL at various toroidal angles, including the location of the peak NWL for (a) the 5-cm and 
(b) the 30-cm SOL geometries. The poloidal angle is measured from the outboard midplane at the level of the magnetic axis.

Fig. 7. Reduction of peak-to-average NWL ratio with stel-
larator’s plasma aspect ratio. The tokamak curve is 
shown for comparison.
Finally, the vacuum vessel composition and dimension were optimized essentially to protect the superconducting magnets that operate at 4 K. All materials were carefully chosen to enhance the shielding performance and minimize the long-term environmental impact. We periodically checked and determined the key nuclear parameters with a series of 1-D and 3-D analyses, and the results were constantly reviewed for potential design modifications. All components have been sized for the maximum NWL and designed to provide adequate performance margins compared to requirements. The reference radial builds are shown schematically in Fig. 10 for two cross sections through the nominal, full blanket (designed for a peak NWL of 5.3 MW/m²) and at \( \Delta_{\text{min}} \) (designed for 3.3 MW/m² NWL—the maximum at the nonuniform blanket region). The first wall, blanket, back wall, and divertor system are replaceable components, whereas all components outside the back wall are permanent, with 40 FPY lifetimes. Figure 11 demonstrates a toroidal cross section through the nonuniform blanket as envisioned for the transition region between \( \Delta_{\text{min}} \) and the full blanket. Table II lists the compositions of all components, and the alloying elements and impurities are given in Ref. 29.

Because of the absence of He coolant, the backup LiPb/SiC system offers a more compact nominal radial build, as illustrated in Fig. 12. Note that the radial standoff at \( \Delta_{\text{min}} \) was kept fixed at 1.3 m to keep the major radius at 7.75 m in order to meet the breeding requirement and first-wall heat load limit. Filling the 1.3-m space with LiPb/SiC blanket and shielding materials provides more protection for the vacuum vessel and magnet than actually needed and suggests borated FS filler for the shield of the backup design, instead of WC.

The main deliverables are summarized in Table III, and Secs. IV and V provide the details of the supporting analyses and the rationale for the shield design. It should be mentioned that the 50-cm reduction resulting from the compact radial build at \( \Delta_{\text{min}} \) is estimated to save \( \sim 30\% \) in the major radius (7.75 versus 10.1 m) and \( \sim 12\% \) in the cost of electricity (78 versus 87 mills/kWh in 2004 U.S. dollars), which is significant. The benefit of the compact feature can be fully recognized when comparing ARIES-CS to all five large-scale stellarators developed to date (see Fig. 13). The most recent advanced physics and technology and innovative means of radial dimension control helped reduce the major radius by more than threefold, approaching that of advanced tokamaks.
IV. KEY NUCLEAR PARAMETERS

IV.A. Initial 1-D TBR Estimate

For each blanket concept, we developed three radial builds at $\Delta_{\text{min}}$, divertor region, and nominal area everywhere. The nominal radial build varies widely with blanket concepts (1.8 m for LiPb/He/FS and 1.4 m for LiPb/SiC). As for the blanket itself, we sized it essentially to meet the breeding requirement and protect the shield for the 40-FPY plant life. The nominal first wall, breeding zone, and back wall of the LiPb/He/FS and
LiPb/SiC systems are 63 and 50 cm thick, respectively. The analysis assumes a few blanket modules can be installed behind the divertor plates (20-cm-thick FS/W/He, 33/4/63 by volume). The nonuniform, tapered blanket (25 cm thick at $\Delta_{\text{min}}$) expands and joins the full blanket (see Fig. 11). Depending on the average major radius, it covers 15 to 35% of the first-wall area. We had to make an educated assumption that is essential for the accuracy of the breeding level. We assumed that the penetrations and divertor plates/baffles cover 1 and 15% of the first-wall area, respectively. To estimate the overall TBR, we combined the 1-D local TBR with the coverage fraction of the three regions: nominal, nonuniform, and divertor blankets. Figure 14 shows the contours that bound the nonuniform blanket within each field period for a wide range of average major radii. The larger the machine, the lower the nonuniform blanket coverage and the higher the breeding. Using the coverage fraction and the 1-D TBR estimates, we examined the sensitivity of the TBR to the machine size. It is customary when using 1-D

<table>
<thead>
<tr>
<th>TABLE II</th>
<th>Compositions of ARIES-CS Components</th>
</tr>
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<tbody>
<tr>
<td><em><em>LiPb</em>/He/FS</em>*</td>
<td><em><em>LiPb</em>/SiC</em>*</td>
</tr>
</tbody>
</table>
| First wall | 34% FS structure  
66% He coolant |
| Divertor system | 32.6% FS structure  
4.0% W  
63.4% He coolant |
| Full blanket | 79% LiPb (70% enriched $^{6}\text{Li}$)  
7% SiC inserts  
6% FS structure  
8% He coolant |
| Back wall | 80% FS structure  
20% He coolant |
| FS or SiC shield | 15% FS structure  
10% He coolant  
75% borated steel filler |
| WC shield | 15% FS structure  
10% He coolant  
75% WC filler |
| Manifolds | 52.0% FS structure  
22.7% LiPb (70% enriched $^{6}\text{Li}$)  
24.0% He coolant  
1.3% SiC inserts |
| Vacuum vessel | 28% FS structure  
49% water  
23% borated steel filler |
| Inner coil case, strong back, and intercoil structure | 95% JK2LB structure  
5% liquid He coolant |
| Winding pack | 18.5% JK2LB structure  
48.2% Cu  
12.8% Nb$_3$Sn  
10.0% insulator  
10.5% LHe coolant |
| Cryostat | 100% Type 304 stainless steel structure |
| Bioshield | 85% concrete  
10% mild steel  
5% He coolant |

*17 at.% Li and 83 at.% Pb.
models to make conservative assumptions and add a safety margin. Figure 15 shows the conservative 1-D estimate of the TBR. A design with a major radius less than 7.5 m even with 90% $^6$Li enrichment cannot provide tritium self-sufficiency. This clearly demonstrates the important role the breeding requirement plays in determining the smallest major radius of compact stellarators—a feature unique to this concept.

The reference LiPb/He/FS design calls for a 7.75-m machine that meets the overall design space requirements and barely satisfies the limit on the heat load accommodation of the LiPb/He/FS blanket. The final divertor design suggests a slight change in the divertor coverage fraction (~12% of the first-wall area, instead of 15%). It seems likely that the 7.75-m design with 90% $^6$Li enrichment will breed more tritium than needed for plasma operation, allowing a larger breeding margin. To achieve a calculated overall TBR of 1.1, a $^6$Li enrichment lower than 90% could be considered. However, a decisive action to adjust the enrichment could not be taken without establishing a 3-D model for the entire machine to confirm the 1-D TBR estimate. Section IV.C presents the final 3-D analysis and results.

### IV.B. Heat Load to In-Vessel Components

The power deposited in the first wall, blanket, shield, manifolds, and divertor components will be recovered by the He and LiPb coolants as a high-grade heat. Table IV details the breakdown of the volumetric nuclear heating deposited in these in-vessel components, assuming the divertor covers 15% of the first-wall area. As the table indicates, most of the power (94%) goes to the first wall, divertor, and blanket. The He/LiPb-cooled shield and manifolds carry 6% of the nuclear heating, which is significant and recovered to improve the power balance and enhance the economics. The small heat leakage to the vacuum vessel (~3 MW) will be dumped as a low-grade heat. The heat load to the winding pack and intercoil structure is ~12 kW, corresponding to a liquid He cryogenic load of ~5 MW(electric). For a neutron power of

### TABLE III

<table>
<thead>
<tr>
<th>Key Nuclear Parameters for the Reference and Backup Systems</th>
<th>LiPb/He/FS</th>
<th>LiPb/SiC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Peak NWL</td>
<td>5.3 MW/m²</td>
<td>0.5 MW/m²</td>
</tr>
<tr>
<td>Average NWL</td>
<td>2.6 MW/m²</td>
<td>2.6 MW/m²</td>
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<tr>
<td>Peak to average NWL</td>
<td>2</td>
<td>2</td>
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<tr>
<td>Calculated overall TBR</td>
<td>1.1</td>
<td>1.1</td>
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<tr>
<td>$^6$Li enrichment</td>
<td>70%</td>
<td>&lt;90%</td>
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<tr>
<td>First-wall end-of-life fluence</td>
<td>15.7 MW/m²</td>
<td>18 MW/m²</td>
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<tr>
<td>First-wall/blanket lifetime</td>
<td>3 FPY</td>
<td>3.4 FPY</td>
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<tr>
<td>Shield/manifold/blanket lifetime</td>
<td>40 FPY</td>
<td>40 FPY</td>
</tr>
<tr>
<td>Overall energy multiplication</td>
<td>1.16</td>
<td>1.1</td>
</tr>
<tr>
<td>$\Delta_{\text{min}}$</td>
<td>1.3 m</td>
<td>1.3 m</td>
</tr>
<tr>
<td>$\Delta_{\text{max}}$</td>
<td>1.8 m</td>
<td>1.4 m</td>
</tr>
</tbody>
</table>

![Fig. 12. Radial builds for the backup, advanced LiPb/SiC blanket.](image-url)
1884 MW (80% of the 2355 fusion power), the overall neutron energy multiplication ($M_n$) amounts to 1.16. The total heat deposition along with the power density that peaks at 44 W/cm$^3$ served as a source term for the detailed thermal analysis carried out for the ARIES-CS design.\textsuperscript{17}

The power split between the He and LiPb coolants is an essential input to the power conversion system and to the systems code for the purpose of costing the He and LiPb heat transfer/transport system. The distribution of power including the surface heating is summarized in Table V. The 1377 MW recovered by the helium coolant includes the 111 MW transferred through the SiC insulator from the hot LiPb to the colder He (Ref. 17). About 90% of the He pumping power [183 MW(electric)] will be recovered by the helium coolant as a friction power. The bottom line result is that the thermal power split between He and LiPb is 49:51.

**IV.C. Final 3-D TBR and $M_n$ Analysis**

A model of the full ARIES-CS system was developed based on a one-sixth toroidal solid model. The primary goals of this analysis were to determine the TBR and neutron energy multiplication for the LiPb/He/FS system in the complex ARIES-CS geometric configuration. The solid model was generated with blanket, shield, manifold, and divertors, as shown in Fig. 16. A number of features were incorporated in the model to account for design elements. Since the blanket model was not constructed to distinguish between the front wall, back wall, and central breeding region, a homogenized material...
A definition was used throughout. However, this central breeding region is not of uniform thickness throughout the blanket. To accommodate this, the blanket was divided into two regions, one region with a material description consistent with the nominal blanket thickness ("uniform" region with 68% LiPb, 13.6% FS, 6% SiC, and 12.4% He) and one region with a material description based on the varying blanket thickness ("nonuniform" region with 54% LiPb, 17% FS, 6% SiC, 10% B-FS, and 13% He). In addition, the blanket regions behind the divertors were modeled as separate regions with different homogenized mixtures (65.5% LiPb, 17.2% FS, 8% SiC, and 9.3% He). To accommodate their impact on the TBR, the electron cyclotron heating (ECH) ducts were also included in the model as rectangular penetrations (24 cm × 54 cm) through the blanket, shield, and manifolds at 35 deg in each field period. Because of the complexity of the model, the vacuum vessel was not included in this analysis since its impact on the TBR and Mn is negligible. Since periodic boundary conditions were not available for use on this MCNPX-CGM model, it was necessary to replicate the one-sixth model (~8.5 m long) with appropriate rotations to generate a full toroidal model. Axial asymmetries in the model prevented the generation of a true union of these sectors and an approximately 1-cm-thick region was introduced at each boundary to alleviate this problem. Each of these regions is modeled as a void and assumed to be a small perturbation to the system. Finally, one limitation of the source function described above is that the source can be generated in only a single region. Since the model is made from six sectors, there are six different source regions. This was overcome by modeling the full toroidal system with only one-sixth of the source. Consider a component $C$ in sector $i$. The response of this component to a full source $R_{C,T}$ is the superposition of the responses from the source in each sector $R_{C,j}$, where $j = [1, 6]$. However, by symmetry, the response of component $C$ in sector $i$ due to a source in sector $j$ is equivalent to the response of

<table>
<thead>
<tr>
<th>TABLE IV</th>
<th>Nuclear Heat Load (in MW) to In-Vessel Components of LiPb/He/FS System</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Full Blanket/Shield</td>
</tr>
<tr>
<td>First wall</td>
<td>116</td>
</tr>
<tr>
<td>Divertor</td>
<td>1137</td>
</tr>
<tr>
<td>Blanket</td>
<td>9</td>
</tr>
<tr>
<td>Back wall</td>
<td>59</td>
</tr>
<tr>
<td>Shield</td>
<td>6</td>
</tr>
<tr>
<td>Total</td>
<td>1327</td>
</tr>
</tbody>
</table>

(⇒$M_n = 1.16$)

<table>
<thead>
<tr>
<th>TABLE V</th>
<th>Summary of Thermal Power Load (in MW) to Helium and LiPb Coolants of LiPb/He/FS System</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Helium</td>
</tr>
<tr>
<td>Surface heating</td>
<td>471</td>
</tr>
<tr>
<td>90% of He pumping power</td>
<td>165</td>
</tr>
<tr>
<td>First wall</td>
<td>162</td>
</tr>
<tr>
<td>Divertor</td>
<td>149</td>
</tr>
<tr>
<td>Blanket</td>
<td>179</td>
</tr>
<tr>
<td>Back wall</td>
<td>18</td>
</tr>
<tr>
<td>Shield</td>
<td>120</td>
</tr>
<tr>
<td>Manifolds</td>
<td>2</td>
</tr>
<tr>
<td>Leakage from LiPb to He</td>
<td>+111</td>
</tr>
<tr>
<td>Total</td>
<td>1377</td>
</tr>
</tbody>
</table>

(49%) (51%)
component $C$ in sector $j$ due to a source in sector $i$: $R_{C_{i,j}} = R_{C_{j,i}}$. Therefore,

$$R_{C_{i,T}} = \frac{1}{6} \sum_j R_{C_{i,j}} = \frac{1}{6} \sum_j R_{C_{j,i}}.$$

Using this methodology, the results for the TBR and $M_n$ were determined for each major component and for the whole device. Figure 17 shows a summary of the variations in these integral responses as a function of the $^6$Li enrichment in the blanket. The target TBR of 1.1 is indicated in the figure and suggests a $^6$Li enrichment of at least 65%. Approximations introduced by the homogenization of the material for the blanket introduce some uncertainties (confirmed later with 1-D estimates) and suggest a slightly higher enrichment of ~70%. As discussed below, an online adjustment of the $^6$Li enrichment is mandatory to meet the breeding requirement. Table VI summarizes the results for the TBR, showing the distribution of tritium breeding among the major blanket regions. In all cases, the majority (>77%) of the tritium breeding occurs in the uniform blanket region and approximately 2.5% occurs in the blanket region behind the divertors.

The energy multiplication is seen to be independent of the $^6$Li enrichment, with an approximately 16% increase in the neutron source energy occurring in the system due to neutron multiplication and exothermic nuclear reactions. Approximately half of the nuclear heating is from photons: 46, 48, and 52% for $^6$Li enrichments of 90, 60, and 30%, respectively. The distribution of the energy generation is shown in Table VII, with the majority occurring in the blanket and a significant fraction (10 to 12%) in the divertor and shield.

### IV.D. Potential Solutions for Overbreeding and Underbreeding Blankets

A calculated overall TBR of 1.1 ensures tritium self-sufficiency for ARIES-CS. The 10% breeding margin mainly accounts for the uncertainties in the cross-section data (~6%), approximations in the 3-D geometric modeling (~3%), and excess breeding to supply T for future power plants (~1%). The achievable net TBR during operation could be as low as 1.01, much lower than the calculated TBR. An extensive research and development program is needed to reduce the gap between the calculated TBR and net TBR (Ref. 11). Expectedly, the ~9% breeding margin will decrease with time as nuclear data improve, more sophisticated 3-D neutronics modeling tools develop, and detailed engineering designs become

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**TABLE VI**

Summary of 3-D TBR Results for $^6$LiPb/He/FS System

<table>
<thead>
<tr>
<th>Blanket Region</th>
<th>$^6$Li Enrichment</th>
<th>30%</th>
<th>60%</th>
<th>90%</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uniform</td>
<td>0.73</td>
<td>0.85</td>
<td>0.91</td>
<td></td>
</tr>
<tr>
<td>Nonuniform</td>
<td>0.15</td>
<td>0.21</td>
<td>0.24</td>
<td></td>
</tr>
<tr>
<td>Behind divertor</td>
<td>0.022</td>
<td>0.028</td>
<td>0.029</td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>0.91</td>
<td>1.08</td>
<td>1.18</td>
<td></td>
</tr>
</tbody>
</table>

*The 1σ statistical error is indicated for each case.

**TABLE VII**

Summary of Energy Multiplication Results for $^6$LiPb/He/FS System

<table>
<thead>
<tr>
<th>Component/Region</th>
<th>$^6$Li Enrichment</th>
<th>30%</th>
<th>60%</th>
<th>90%</th>
</tr>
</thead>
<tbody>
<tr>
<td>Blanket</td>
<td>0.99</td>
<td>1.01</td>
<td>1.03</td>
<td></td>
</tr>
<tr>
<td>Uniform</td>
<td>0.74</td>
<td>0.75</td>
<td>0.77</td>
<td></td>
</tr>
<tr>
<td>Nonuniform</td>
<td>0.24</td>
<td>0.25</td>
<td>0.25</td>
<td></td>
</tr>
<tr>
<td>Behind divertor</td>
<td>0.013</td>
<td>0.013</td>
<td>0.013</td>
<td></td>
</tr>
<tr>
<td>Shield</td>
<td>0.065</td>
<td>0.052</td>
<td>0.043</td>
<td></td>
</tr>
<tr>
<td>Main shield</td>
<td>0.045</td>
<td>0.034</td>
<td>0.026</td>
<td></td>
</tr>
<tr>
<td>Behind divertor</td>
<td>0.020</td>
<td>0.018</td>
<td>0.017</td>
<td></td>
</tr>
<tr>
<td>Manifold</td>
<td>0.0014</td>
<td>0.0014</td>
<td>0.0012</td>
<td></td>
</tr>
<tr>
<td>Divertor plates</td>
<td>0.10</td>
<td>0.091</td>
<td>0.086</td>
<td></td>
</tr>
<tr>
<td>ECH duct</td>
<td>$5.7 \times 10^{-5}$</td>
<td>$5.4 \times 10^{-5}$</td>
<td>$4.8 \times 10^{-5}$</td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>1.16</td>
<td>1.16</td>
<td>1.16</td>
<td></td>
</tr>
</tbody>
</table>

*The 1σ statistical error is shown for the total $M_n$ in each case.

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Fig. 17. Variation of 3-D TBR and $M_n$ with $^6$Li enrichment.
available. References 11 and 33 provide a more detailed breakdown of the breeding margin.

Consuming a large T amount (55.6 kg per year, per 1 GW of fusion power), the calculated overall TBR should be fairly accurate since an uncertainty as small as 1% translates into 1.3 kg/FPY of T for 2.355-GW fusion power of ARIES-CS. This has a significant impact that is equally important for an excess of T and a shortage of T. It is less risky to produce more T (i.e., overbreeding blanket) with the understanding that an on-line adjustment of the TBR is a requirement for fusion. This is easily achieved for liquid breeders with fine-tuning of the $^6$Li enrichment.\textsuperscript{11} In case of overbreeding (net TBR > 1.01), the TBR could be decreased by lowering the enrichment below 70% or replacing a few breeding modules by shielding components. In case of underbreeding (net TBR < 1.01), an easy fix would be to increase the Li enrichment to 80 or 90%. Major changes would require thickening the breeding zones (see Fig. 18), adding beryllium to the blanket, and/or increasing the major radius above 7.75 m.

IV.E. First-Wall/Blanket Service Lifetime

As noted earlier, ARIES-CS blanket modules were designed with replaceability as a design consideration.\textsuperscript{8} The 198 blanket modules would be built in factories and then shipped to the plant for installation. Failure mechanisms in the structure are influenced by the atomic displacement in the case of ferritic steels and by the burnup of Si and C atoms in the case of SiC/SiC composites, ending their service lifetime. In this study, we adopted lifetime limits of 200 dpa for the FS structure and 3% burnup for the SiC structure, in concert with similar ground rules considered for advanced ARIES designs.\textsuperscript{13,14} For a peak NWL of 5.3 MW/m$^2$, the first-wall lifetimes are 3 and 3.4 FPY for the FS and SiC structures, respectively, requiring 11 to 13 replacements during the 40-FPY plant lifetime. Within the blanket, the SiC burnup rate drops faster than the dpa rate, calling for a thinner replaceable SiC first wall/blanket (25 cm) compared to the 63-cm-thick FS first wall, blanket, and back wall. To help reduce the radwaste stream and the annual replacement cost, we segmented the 50-cm-thick SiC blanket into two equal segments: replaceable and permanent (refer to Fig. 12). Even though the majority of the blanket modules are subject to NWLs less than 5.3 MW/m$^2$, they will all be replaced every 3 to 3.4 FPY. There is certainly an incremental increase in cost and radwaste volume associated with the early replacement, but this will be offset by the high gain due to the fewer maintenance processes, shorter down time, and therefore higher system availability.

V. RADIATION PROTECTION AND SHIELDING

V.A. Nominal Shield Design

We focused our shielding activity on $\Delta_{\text{min}}$, where a superior shielding performance makes a notable difference to the machine size and cost. No economic and design enhancements are gained with a high-performance, compact shield at any place, but at $\Delta_{\text{min}}$ as the nominal shielding space is not constrained elsewhere. This feature is unique to stellarators. Thus, the topic has been investigated jointly by the engineers and physicists to examine the location, size, and first-wall coverage of $\Delta_{\text{min}}$ and their impact on the machine parameters (major radius, field at coil, etc.), nuclear issues (T breeding, magnet protection, activation, and decay heat), and economics.

The selection criteria for the shielding materials included several design parameters that play an essential role in the acceptability of the materials. These are the compatibility with the main structure and the constituents of other components, radiation stability, safety characteristics, and operating temperature windows. The magnet radiation limits strongly influence the compositions and size of the shielding components. Being the closest component to the magnet, the composition of the vacuum vessel affects the magnet radiation damage significantly. The double-walled vacuum vessel was filled with shielding materials and optimized to achieve the necessary requirements for magnet protection. Several fillers have been identified for evaluation: water, borated water, FS, and B-FS (FS with 3 wt% B). No structural role has been envisioned for the fillers. Water was considered for its superior shielding characteristics relative to other coolants such as liquid breeders and He gas. In fact, liquid breeders were excluded since the blanket and, to a lesser extent, the shield provide all the tritium needed for plasma operation.

Nb$_3$Sn and JK2LB (Japanese austenitic steel) have been selected as the superconductor and coil structure

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Fig. 18. Local TBR versus thickness of full blanket.
The rationale for selecting the JK2LB steel over Incoloy-908 relates to the activation characteristics of the two candidate alloys, as highlighted in Sec. VII and detailed in Ref. 28. The maximum radiation damage to the winding pack is limited by the $10^{19}$ fast neutron fluence to the Nb$_3$Sn superconductor, dose to the insulator, and displacement of atoms for the copper stabilizer ($6 \times 10^{-3}$ dpa). The peak nuclear heating limitation, determined by the refrigeration requirement, is $2 \text{ mW/cm}^3$ for the low-temperature $4 \text{ K}$ magnet.

The proposed insulation for ARIES-CS is an inorganic tape impregnated with a ceramic binder applied to the tape prior to application to the cable. Several types of tapes have been considered, including S2-glass tape that has been sized. The ceramic-based tape is wrapped around the conductor during the winding process and prior to the heat treatment using an inorganic clay-glass insulator. The lack of organic materials reduces the sensitivity of the proposed insulation to radiation damage. Inorganic insulators have a typical fluence limit on the order of $10^{11}$ rads ($10^9$ Gy) (Ref. 9). The actual composition of the ceramic-based inorganic insulator is not available—it is a proprietary property of a European manufacturing company. Instead, the composition of the glass fiber–filled polyimide has been used throughout the shielding analysis.

Our results show that the fast neutron fluence to the Nb$_3$Sn superconductor and peak nuclear heating are the predominant radiation limits for the magnet. The strong dependence of fluence and heating on the choice of the vacuum vessel fillers is displayed in Fig. 19 for the LiPb/He/FS concept, showing the impact of the trade-off between water and B-FS filler. The water content could range from 30 to 70%, by volume. We selected 65% water content to minimize the mass of the vacuum vessel and solid radwaste without exceeding the heating and fluence limits. Sandwiched between 3-cm-thick face sheets, the central part of the vacuum vessel consists of 5% FS ribs (dictated by the structural requirement), 65% water, and 30% borated FS filler, by volume. Table VIII summarizes the peak radiation damage at the three distinct regions of ARIES-CS: full blanket, nonuniform blanket, and divertor. The results reflect a safety factor of 3 that accounts for the uncertainties in the cross-section data, approximations in the 1-D model, and the presence of the assembly gaps between adjacent modules. In other words, the reported results are three times higher than the computed values. The analysis assumes perfect shield with no penetrations. Admittedly, neutrons streaming through the helium supply pipes will enhance the damage, but hopefully it will not exceed the limit. Means to

![Fig. 19. Sensitivity of peak radiation effects at magnet to vacuum vessel composition, trading B-FS for water.](image-url)
alleviate the streaming problems have been investigated and are discussed in Sec. V.C.

The compact radial build for the critical area surrounding $\Delta_{\text{min}}$ can withstand up to 4 MW/m$^2$ peak NWL. Looking beyond conventional materials (such as steel, water, and borides), tungsten and its compounds possess superior shielding performance. Tungsten carbide, in particular, offers the most compact radial build when used in the shield of the reference design, replacing the B-FS filler. Costing roughly the same as the steel filler, the WC cost difference is not prohibitive for such limited space. Figure 10 displays the LiPb/He/FS radial builds that meet the design requirements. Components with poor shield performance, such as the manifolds, have been avoided at $\Delta_{\text{min}}$. The compact blanket/shield helps reduce the radial standoff at $\Delta_{\text{min}}$ by 50 cm while maintaining the radiation level below the design limit (refer to Table VIII). A challenging task would be the heat removal mechanism and the integration of the nonuniform blanket/shield with the surroundings. Blanket and shield with variable thicknesses have been envisioned for the transition region, as demonstrated in Fig. 11.

V.B. Bioshield

Surrounding the magnet is the 5-cm-thick Type 304 stainless steel cryostat, followed by the 2-m-thick steel-reinforced concrete bioshield (85% concrete, 10% mild steel structure, and 5% He coolant). To size the bioshield, we took advantage of the intercoil structure between the magnet winding packs. Along with the in-vessel components, the 15- to 30-cm-thick intercoil structure provides a shielding function that helps reduce the bioshield dimension. The 2-m-thick bioshield limits the biological dose during operation to 0.25 mrem/h (2.5 $\mu$Sv/h) (see Fig. 20). This operational dose is consistent with the U.S. guidelines for the protection of workers and public and reflects a 10-fold reduction in the absolute limit of 2.5 mrem/h (25 $\mu$Sv/h) in order to keep the dose as low as reasonably achievable. The need for a sufficient space outside the magnet to conduct the maintenance operation mandates the bioshield to be placed at a radius of 13 m or more. The dose inside the bioshield is quite high even after shutdown, meaning that all ARIES-CS components should be maintained remotely, with no personnel access into the hall surrounding the magnets.

V.C. Streaming Issues

The radiation levels reported so far pertain to a perfect configuration without penetrations. Penetrations are necessary for vacuum pumping, coolant supply lines, plasma control, and maintenance ports. Such penetrations jeopardize the effectiveness of the shield since neutrons streaming through these penetrations enhance the damage at the shield, manifolds, vacuum vessel, and magnet. Helium-cooled systems in particular raise a specific concern since designing penetration shields for the He access tubes/pipes represents a challenging problem. Eight types of penetrations have been identified for the reference LiPb/He/FS design:

1. 198 He tubes for blanket [32-cm inner diameter (i.d.)]
2. 24 divertor He access pipes (30- to 60-cm i.d.)
3. 30 divertor pumping ducts (42 cm $\times$ 120 cm each)
4. 12 large pumping ducts (1 cm $\times$ 1.25 m each)
5. 5 ECH ducts (24 cm $\times$ 54 cm each)
6. 6 main He pipes connecting heat exchanger to blanket and shield (72-cm i.d. each)
7. 6 main He pipes connecting heat exchanger to divertor (70-cm i.d. each)
8. 4 access holes (3 cm diameter) for each blanket module.

To improve the prospects for ARIES-CS with reduced neutron streaming, we developed a list of practical solutions that could be selectively applied to each penetration:

1. local shield behind penetrations
2. He tube axis oriented toward lower neutron source
3. penetration shield surrounding ducts
4. replaceable shield close to penetrations
5. avoid rewelding vacuum vessel and manifolds close to penetrations
6. several bends along penetration lines.

Two-dimensional (2-D) and 3-D analyses need to be performed to address the streaming issues and concerns.
However, the definition of several penetrations came too late in the design process, suggesting a qualitative assessment for some penetrations:

1. The large penetrations for ECH and He supply lines should be surrounded with 0.5- to 1-m-thick penetration shield to protect the surroundings. The ECH hardware is embedded at the front of the blanket in an equatorial port located toroidally at $\Theta = 35$ deg. A few 90-deg bends along the deep penetration line help attenuate the streaming neutrons.

2. Three or more bends are recommended for the vacuum pumping ducts located between the inboard and outboard divertor plates. A 50-cm-thick local shield behind each duct helps protect the magnet against streaming radiation.

3. The blanket access holes raise a streaming concern since the 3-cm-diam, 50-cm-deep holes provide a clear path for the 14-MeV source neutrons to reach the shield. Long nuts (30 to 50 cm) screwed in from the plasma side are necessary to protect the four bolts located at the back of each blanket module. The holes occupy a small fraction of the blanket ($<1\%$) and will have insignificant impact on the TBR.

4. There is a close-fitting shielding plug inserted into the maintenance ports during operation. These shielding components will be neutronically equivalent to the vacuum vessel and intercoil structure that are missing in the port region.

More serious streaming issues related to the blanket He access tubes and divertor He access pipes were investigated in detail using 2-D and 3-D DANTSYS (Ref. 19), MCNPX (Ref. 18), and Attila$^{35}$ codes, as briefly discussed below. Note that these particular streaming problems are unique to the dual-cooled LiPb/He/FS reference design, that is, inapplicable to the LiPb/SiC backup system. Of specific interest are the impacts of radiation streaming on the dpa, He production, and nuclear heating at the surrounding components (blanket, shield, manifolds, vacuum vessel, and magnet). If the damage level in the structure is too high, these components will have to be replaced in order to maintain the integrity of the machine. For example, the dpa and He production levels at the shield and manifolds/vacuum vessel should not exceed 200 dpa and $1$ appm, respectively, at any time during operation. Also, the neutron-induced swelling at the outer screws that adjust the divertor plates (see Fig. 21) should be minimal.

### V.C.1. Streaming Through Blanket He-Access Tubes

The aim of this study is to estimate the radiation damage at the manifolds, vacuum vessel, and magnet due to neutron streaming through the 32-cm-diam He tubes that supply the He from the manifolds to the blanket modules. One of the 198 tubes is shown schematically in Fig. 11 for the nonuniform blanket module. The results of the 2-D model indicate a high damage at the manifolds, exceeding the 1 appm reweldability limit at 40 FPY. To protect the manifolds, the analysis suggests increasing the nominal shield thickness to 32 cm and orienting the tubes away from the high-neutron source that peaks at the plasma center. In addition, an innovative idea suggests protecting the welds further with a removable 10-cm-thick WC shielding ring.$^{34}$ In this case, the welds will not be in a direct line of sight with the streaming neutrons, but rather embedded in the manifold structure, away from the tube surface.

The damage profile at the vacuum vessel and magnet indicates the streaming results in hot spots peaking behind the tube centerline, and the damage exceeds the limit by more than an order of magnitude. To protect the vacuum vessel and magnet, approximately 25-cm-thick local shield should be placed behind each tube, as illustrated in Fig. 11. The toroidal and poloidal extents of these local shields need to exceed the 32-cm diameter of the tube to adequately protect the vacuum vessel and magnet against streaming radiation.

### V.C.2. Streaming Through Divertor He Access Pipes

A set of four pipes is required to supply the He coolant to each divertor system. The pipe begins behind the divertor and extends outward through the blanket, shield, manifolds, vacuum vessel, and coil structure. Our preliminary 2-D analysis for a simple, straight pipe with 40-cm i.d. indicated a notable peaking in damage at all components surrounding the pipe, with more pronounced peaking at the magnet than at the shield. Furthermore, the neutron flux behind the pipe outside the magnet increases by orders of magnitude due to neutron streaming. These results suggest redesigning the pipe with internal WC shielding plug and inserts, hoping to control the streaming radiation. Figure 21 displays a first-attempt design approach that would be maintainable while offering several features that may alleviate the streaming problem. The tools utilized to analyze this 3-D problem are the Attila deterministic code$^{35}$ and DAG-MCNPX Monte Carlo code.$^{36}$ Both codes import a CAD model of the divertor and four pipes, eliminating modeling errors and allowing faster design iterations,$^{37}$ if needed.

Our 3-D analysis indicated that neutron attenuation through the WC shielding plug and inserts is not sufficient to eliminate the streaming problems entirely. The main results are as follows:

1. Bulk shield is well protected, so no need for blanket/shield extension.

2. Peaking in damage is more pronounced at magnet than at shield.

3. Helium production at manifolds and vacuum vessel exceeds reweldability limit by 2-8 fold, so avoid rewelding within 20 to 40 cm from pipe.

4. Winding pack should be placed at least 40 cm from pipe.
5. Neutron-induced swelling in screws (that adjust divertor plates during operation) is negligible (<0.05%).

6. Front 60 to 120 cm of pipe wall is not reweldable and should be replaced along with divertor system every 3 FPY.

The final ARIES-CS design accords with these recommendations, calling for no rewelding of the manifolds and vacuum vessel near the pipe and ensuring that the coil is located at least 40 cm from the pipe. The neutron flux behind the pipe is excessive, mandating additional local shield (~1 m) to protect the externals. This fairly thick local shield is missing in the final ARIES-CS CAD drawings.

Comparing the 2-D results (for simple, straight pipe) with the 3-D results (for the pipe with WC shielding plug and inserts) indicates that the addition of WC shield is ineffective since all streaming problems at manifolds, vacuum vessel, magnet, and outside pipe still exist, except at the shield. Past studies should develop a more effective scheme to attenuate the streaming neutrons and reduce the flux outside pipes. For example, a simple pipe with i.d. smaller than 60 cm and several right-angle bends represents a better approach, eliminating the need for the massive WC shielding plug and inserts (170 tons for the 24 pipes of ARIES-CS).

VI. COMPARISON BETWEEN LiPb/He/FS AND LiPb/SiC SYSTEMS

There are two primary aspects for this comparison: nuclear performance and economics. The first aspect relates to the radial build, breeding capacity, shielding performance, and activation level. The second aspect must also be considered to assess the economic impact of introducing the helium coolant to the LiPb/FS system in order to solve the magnetohydrodynamic problem. Other aspects outside the scope of this assessment, such as the thermal and mechanical performances and technological readiness, could be important issues, but they are not addressed here.

The nominal radial build varies widely, with up to 42 cm difference, as Figs. 10 and 12 indicate. This translates into more materials and sizable components for the LiPb/He/FS system. In addition, the He and LiPb...
manifolds that surround the shield represent another hefty component. Sizable helium tubes and pipes connecting the manifolds to the blanket, shield, and divertor raise neutron streaming concerns and require numerous patches of local shields behind the pipes to protect the vacuum vessel and magnet. As far as breeding is concerned, both systems satisfy the breeding requirement with ~50-cm-thick blanket. The SiC system offers a slightly longer first-wall/blanket lifetime, slightly lower energy multiplication, and remarkably higher thermal efficiency but a more expensive structure [$510/kg (U.S.) for SiC/SiC composites versus $103/kg for FS]. Cost credits are anticipated for the SiC system due to the absence of He pumping power [$\sim 180$ MW(electric)] and He heat transfer/transport loop. The very low activation of SiC translates into more attractive safety features expressed in the level of safety assurance (LSA = 1 for SiC system and LSA = 2 for FS system; the lower the LSA, the lower the cost). An integrated economic analysis\textsuperscript{5} assessed self-consistently the impact of all of these features and parameters on the overall cost of ARIES-CS, and Table IX shows the evaluation of the two systems. Overall, the cost of electricity (COE) metric used to evaluate all ARIES concepts indicates a 20 to 25\% better economic performance for the SiC system.\textsuperscript{5} In other words, the He-cooled FS system increments the COE by $\sim 17$ mills/kWh.

### VII. HIGHLIGHTS OF ACTIVATION AND WASTE MANAGEMENT ASSESSMENTS

Since the safety assessment frequently requires knowledge of the activation parameters, we estimated the highest possible activity, decay heat, and radwaste level on the 100-yr timescale after shutdown. As a source term, the activity has been used to generate the decay heat for the loss of coolant/flow accident (LOCA and LOFA) analysis\textsuperscript{26,27} and to evaluate the radiological hazards of the individual components.\textsuperscript{28} For all components except the divertor system, the decay heat is manageable, resulting in a LOCA/LOFA temperature below the 740°C reusability limit of FS (Ref. 26). The W of the divertor generates high decay heat and the LOCA/LOFA temperature exceeds 800°C, necessitating replacement of the divertor system following an accident or installing an active heat removal system operating for the first 24 h.

Since the inception of the ARIES project, we focused our attention on the disposal of all active materials in geological repositories. To classify the waste, we evaluated the waste disposal rating for a fully compacted waste using the most conservative waste disposal limits developed in the United States. In general, ARIES plants generate only low-level waste (class A or C) that requires near-surface, shallow land burial according to the U.S. waste classification. The recent introduction of the clearance category for slightly radioactive materials and the development of radiation-resistant remote-handling equipment opened the possibility to recycle and clear the majority of the ARIES-CS radwaste. Scenarios for fusion radwaste management now include disposal in geological repositories, recycling and reuse within the nuclear industry if technically and economically feasible, and clearance or release to the commercial market if the materials contain traces of radioactivity.\textsuperscript{38} There is a growing international effort in support of this new trend to essentially avoid the repository disposal and eliminate the long-term radwaste burden on future generations.\textsuperscript{39}

We applied the disposal, recycling, and clearance scenarios to all ARIES-CS components. The recycling and clearance options appeared technically attractive and judged necessary to control the ARIES-CS radwaste stream. Note that this activation assessment limited the material choices

<table>
<thead>
<tr>
<th>TABLE IX</th>
<th>Comparison Between Reference and Backup Systems</th>
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<tbody>
<tr>
<td></td>
<td>LiPb/He/FS</td>
</tr>
<tr>
<td>Overall TBR</td>
<td>1.1</td>
</tr>
<tr>
<td>First-wall/blanket lifetime</td>
<td>3 FPY</td>
</tr>
<tr>
<td>Overall energy multiplication</td>
<td>1.16</td>
</tr>
<tr>
<td>$\eta_{th}$</td>
<td>42%</td>
</tr>
<tr>
<td>Structure unit cost$^a$</td>
<td>$103/kg $</td>
</tr>
<tr>
<td>Blanket/divertor/shield/manifolds cost$^a$ (millions)</td>
<td>$288 $</td>
</tr>
<tr>
<td>Cost$^a$ of heat transfer/transport system (millions)</td>
<td>$475 $</td>
</tr>
<tr>
<td>Pumping power</td>
<td>$183$ MW(electric) $</td>
</tr>
<tr>
<td>LSA factor</td>
<td>2</td>
</tr>
<tr>
<td>COE$^a$</td>
<td>78 mills/kWh</td>
</tr>
<tr>
<td>Reference design ($\langle R \rangle = 7.75$ m)</td>
<td></td>
</tr>
<tr>
<td>Full blanket/shield everywhere ($\langle R \rangle = 10.1$ m)</td>
<td>87 mills/kWh</td>
</tr>
</tbody>
</table>

$^a$In 2004 U.S. dollars.
for one of ARIES-CS components. It provided a definitive answer to “what is the best structural material for ARIES-CS magnet: JK2LB or Incoloy-908?” The superior recycling and clearance characteristics of the JK2LB Japanese steel provided a strong incentive to use it as the reference magnet structure. For more detailed discussion, the reader is directed to the ARIES-CS safety paper in this issue.28

During the past three decades, the radwaste volume aspect of fusion continued to be of concern.40 As such, the ARIES project has been committed to the achievable goal of radwaste minimization by design. Figure 22 displays the breakdown of ARIES-CS radwaste. The focus on compact devices with radwaste reduction mechanisms (such as well-optimized components) contributed most significantly to the threefold reduction in ARIES-CS total radwaste volume compared to previous stellarators30,31 developed prior to 1990. Figure 23 demonstrates this impressive trend and illustrates the 30% reduction in ARIES-CS volume achieved even during the three-year time frame of the study. In fact, recycling and clearance can be regarded as a more effective means to diminish the radwaste stream. The reason is that clearable materials will not be categorized as waste and the majority of the remaining nonclearable materials can potentially be recycled and therefore will not be assigned for geological disposal.40

VIII. CONCLUSIONS

The three unique aspects of ARIES-CS, namely, advanced physics, advanced technology, and advanced manufacturing techniques, driven by the guiding principle of compactness, have resulted in an economically competitive stellarator power plant. A number of challenging engineering issues have been addressed in order to deliver a credible design. Among other factors, these issues stem from the compactness and complexity of the machine. Serious efforts have been made to address the nuclear-related issues in particular, by adjusting the radial standoff to accommodate the constrained areas where the magnet moves close to the plasma, by developing a new CAD/MCNPX tool to model, for the first time, such a complex geometry for 3-D nuclear analyses, and by establishing a framework for handling the radioactive materials and minimizing the radwaste stream.

The first and foremost effort has been on defining the components of the minimum plasma-coil space, which is the most influential engineering element that greatly impacts stellarators’ overall size and cost. Through rigorous nuclear analysis, we demonstrated the following:

1. The novel shielding approach developed specifically for $\Delta_{\text{min}}$ helps reduce the radial standoff by $\sim 28\%$, the major radius by $\sim 30\%$, and the COE by $\sim 12\%$. Equally important is the consequence of the substantial reduction in ARIES-CS radwaste volume compared to previous stellarator designs.

2. The tritium breeding plays an important role in determining the minimum major radius of ARIES-CS—a unique feature to compact stellarators.

3. The reference blanket and shield design satisfies the design requirements, breeding sufficient tritium and
protecting vital components with adequate margin. The first wall/blanket and divertor need frequent replacement every 3 FPY, whereas the shield, manifolds, vacuum vessel, and magnet are life-of-plant components.

4. Exact modeling of any stellarator for 3-D nuclear assessment would not be possible without the CAD/MCNP coupling approach. Its development proved to be a requirement to accurately generate the NWL profile and confirm the overall nuclear parameters (TBR and $M_n$).

5. Streaming through divertor He access pipes causes notable damage to the surrounding components. Such penetrations should be redesigned with several right-angle bends to alleviate the streaming problem.

6. Attention should be paid to radwaste management issues. A recycling/clearance strategy to control stellarators’ waste stream must be developed in concert with present U.S. regulation and growing international effort in support of this new trend.

With technologies shifting constantly for advantages as new developments occur, the prospect of using SiC/SiC composites as the main structure offers high operating temperature, high thermal conversion efficiency, and salient safety features—all are potential enhancers for the economic performance of the LiPb/SiC backup system. The absence of the helium coolant and consequences related to streaming problems and use of numerous local shields around the He access pipes represent an additional advantage for the LiPb/SiC system.

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REFERENCES


