THE ARIES-CS COMPACT STELLARATOR
FUSION POWER PLANT

F. NAJMABADI* and A. R. RAFFRAY  Center for Energy Research
University of California, San Diego, MC 0417, La Jolla, California 92093-0417

ARIES-CS TEAM: S. I. ABDEL-KHALIK,a L. BROMBERG,b L. CROSATTI,a L. EL-GUEBALY,c
P. R. GARABEDIAN,a, A. A. GROSSMAN,c D. HENDERTON,c A. IBRAHIM,c T. IHLI,f
T. B. KAISER,g B. KIEDROWSKI,c L. P. KU,h J. F. LYON,i R. MAINGI,i S. MALANG, j C. MARTIN,c
T. K. MAJ, j B. MERRILL,k R. L. MOORE,k R. J. PEIPERT, Jr., l D. A. PETTI,k D. L. SADOWSKI,a
M. SAWAN,c J. H. SCHULTZ,b R. SLAYBAUGH, k K. T. SLATTERY,l G. SVIATOSLAVSKY,c
A. TURNBULL,m L. M. WAGANER, l X. R. WANG,e J. B. WEATHERS, p P. WILSON,c
J. C. WALDROP III, 1 M. YODA,a and M. ZARNSTORFFh

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An integrated study of compact stellarator power plants, ARIES-CS, has been conducted to explore attractive compact stellarator configurations and to define key research and development (R&D) areas. The large size and mass predicted by earlier stellarator power plant studies had led to cost projections much higher than those of the advanced tokamak power plant. As such, the first major goal of the ARIES-CS research was to investigate if stellarator power plants can be made to be comparable in size to advanced tokamak variants while maintaining desirable stellarator properties. As stellarator fusion core components would have complex shapes and geometry, the second major goal of the ARIES-CS study was to understand and quantify, as much as possible, the impact of the complex shape and geometry of fusion core components. This paper focuses on the directions we pursued to optimize the compact stellarator as a fusion power plant, summarizes the major findings from the study, highlights the key design aspects and constraints associated with a compact stellarator, and identifies the major issues to help guide future R&D.

KEYWORDS: fusion power plant, compact stellarator

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

I. INTRODUCTION

ARIES research aims at establishing the economic, safety, and environmental potential of fusion power plants and at identifying physics and technology areas with the highest leverage for achieving attractive and competitive fusion power in order to guide fusion research and development (R&D). The ARIES Team is a U.S. national effort with participation from national laboratories, universities, and the industry and with strong international collaborations. The Team performs detailed physics and

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*E-mail: fnajmabadi@ucsd.edu
aGeorge W. Woodruff School of Mechanical Engineering, Georgia Institute of Technology, Atlanta, Georgia 30332-0405
bMassachusetts Institute of Technology, Plasma Science and Fusion Center, 167 Albany Street, Cambridge, Massachusetts 02139
cUniversity of Wisconsin, Fusion Technology Institute, 1500 Engineering Drive, Madison, Wisconsin 53706-1687
dNew York University, Courant Institute of Mathematical Sciences, New York 10012
eCenter for Energy Research, University of California, San Diego, MC 0417, La Jolla, California 92093-0417
fForschungszentrum Karlsruhe, IKET, P.O. Box 3640, 76012 Karlsruhe, Germany
gLawrence Livermore National Laboratory, Livermore, California
hPrinceton Plasma Physics Laboratory, Princeton University, Princeton, New Jersey 08520
iOak Ridge National Laboratory, Oak Ridge, Tennessee 37831
jConsultant, Fliederweg 3, D 76351 Linkenheim-Hochstetten, Germany
kIdaho National Laboratory, Idaho Falls, Idaho
lThe Boeing Company, P.O. Box 516, St. Louis, Missouri 63166
mGeneral Atomics, San Diego, California 92186
engineering analyses using the most current and detailed models available and then uses the results to perform optimization and trade studies via a cost-based systems code. Our latest study, ARIES-CS, explores attractive compact stellarator configurations as fusion power plants and identifies key R&D areas.

In a stellarator, most of the confining field is produced by external coils (the poloidal field is generated by the external coils as well as the bootstrap current). The absence of a large externally driven plasma current leads to many attractive features: Stellarators are inherently steady state, stable against external kink and axisymmetric modes, and resilient to plasma disruptions.

Earlier stellarator power plant studies led to devices with large sizes compared to tokamaks. The Helias Reactor (HSR) study is based on the Wendelstein 7-X (W7-X) plasma configuration (linked mirrors). It has an average major radius \( R = 22 \) m for a five-field-period configuration (HSR-5), and \( R = 18 \) m for a recent four-field-period (HSR-4) configuration.\(^1\) The Force-Free Helical Reactor\(^2\) (FFHR) is a ten-field-period heliotron/torsatron (\( l = 2 \) stellarator). FFHR-1 and several variants [MHR-S (Ref. 3) and FFHR2m2 (Ref. 4)] have \( R = 10 \) to 20 m. The ARIES Stellarator Power Plant Study (SPPS), completed in 1996, was based on a four-field-period Modular Helias-like Heliac (MHH) configuration and led to an \( R = 14 \) m device that was the first step toward a smaller stellarator power plant.\(^5\) The major radii of these earlier stellarator designs are typically two to four times larger than those of recent tokamak power plant studies, ARIES-RS (Ref. 6) and ARIES-AT (Ref. 7), which have \( R \sim 5.5 \) m.

The large size and mass predicted by earlier stellarator power plant studies had led to cost projections much higher than those of the advanced tokamak power plant. As such, the first major goal of the ARIES-CS research was to investigate if stellarator power plants can be made to be comparable in size to advanced tokamak variants while maintaining desirable stellarator properties. Stellarator fusion core components would have complex shapes and geometry. This complexity imposes severe constraints on fusion core components such as nonuniform heat, particle, and neutron fluxes; accessibility for assembly and maintenance; and feasibility and cost of manufacturing. The second major goal of the ARIES-CS study was to understand and quantify, as much as possible, the impact of the complex shape and geometry of fusion core components.

In Sec. II, we explore the design directions that we pursued to optimize the compact stellarator as a fusion power plant. Sections III and IV present overviews of our detailed physics and engineering design and analysis and highlight the key design aspects and constraints associated with a compact stellarator. Section V examines the extent to which ARIES-CS has met the top-level requirements for fusion power plants and describes the key R&D topics. A complete discussion of ARIES-CS systems can be found in the accompanying papers in this special issue (see Refs. 8 through 17). Figures 1 and 2 show the ARIES-CS fusion power core and plasma/coil configuration, respectively. The major device parameters are listed in Table I.

II. DESIGN DIRECTIONS

II.A. Reducing the Size

The impact of machine size on the cost of electricity (COE) is well known. Starting from a large device, the COE decreases substantially as the machine size is reduced. The reduction in the COE then levels out when the average plasma radius becomes roughly similar in size to the plasma-coil spacing; i.e., further reduction in size would only lead to a small reduction in the COE. Smaller devices typically require a larger extrapolation from current technology as for a given power output; reducing the size would lead to higher heat and particle loads on components, tighter spaces (e.g., for T breeding and assembly/maintenance), the need for a higher-performance plasma, and a host of other engineering constraints. As such, in most modern tokamak studies, such as ARIES-AT, the minimum-COE point is not chosen as the reference design point. Rather, the operating point is chosen near the “knee” of the curve of the COE versus the machine size where the COE is slightly larger than the minimum value predicted by the systems code. In this manner, the slightly larger machine size relaxes engineering constraints and allows for a more robust design. On the other hand, ARIES-CS is the first major study of a compact stellarator power plant. As such, we chose the minimum-COE point as the reference design in order to explore the design space and identify engineering constraints and trade-offs.

The first major goal of the study was to investigate if stellarator power plants can be made to be similar in size to advanced tokamak variants. The large size of the earlier stellarators is generally dictated by the minimum required distance between the plasma [i.e., the last closed magnetic surface (LCMS)] and the middle of the coil winding pack (denoted by \( \Delta_{\text{min}} \)). This space is occupied by the scrape-off layer, first wall, breeding blanket, shield that protects the superconducting coils, coolant manifolds, vacuum vessel, assembly gaps, coil case, and half the radial depth of the coil winding pack—typically, these components require a distance of 1.5 to 2 m between the LCMS and the middle of the coil winding pack. Because the external coils generate a multipolar field and the high-order harmonics of the magnetic field decay rapidly with distance from the coils, the coil currents and irregular shape increase drastically as the distance between the coils and plasma is increased. As such, for a given stellarator plasma/coil configuration (and aspect ratio), the average minor radius of the plasma (and the plasma-coil distance) should be increased until sufficient space is
available for the above components; i.e., the radial build of the device dictates a minimum average plasma minor radius. As the earlier stellarator power plant studies all had a relatively large plasma aspect ratio, the radial built constraint had led to large devices.

![](image1)

**Fig. 1.** The ARIES-CS fusion power core.

It should be noted that while $\Delta_{\text{min}}$ (plasma-to-midcoil distance) is the relevant parameter for arriving at the coils that generate the desired stellarator configuration, the distance between the plasma and the front of the coils is a more relevant engineering constraint. As such, a coil winding pack with a rectangular cross section (thinner in the radial direction) would lead to a smaller $\Delta_{\text{min}}$ for a given plasma to the front of the coil distance.

**TABLE I**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius, $\langle R_{\text{axis}} \rangle$ (m)</td>
<td>7.75</td>
</tr>
<tr>
<td>Minor radius, $\langle a \rangle$ (m)</td>
<td>1.70</td>
</tr>
<tr>
<td>On-axis field, $\langle B_0 \rangle$ (T)</td>
<td>5.7</td>
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<tr>
<td>Peak field on the coil, $B_{\text{max}}$ (T)</td>
<td>15.1</td>
</tr>
<tr>
<td>Plasma current (MA)</td>
<td>4</td>
</tr>
<tr>
<td>Plasma ($\beta$) (%)</td>
<td>6.4</td>
</tr>
<tr>
<td>Plasma temperature, $\langle T \rangle$ (keV)</td>
<td>6.6</td>
</tr>
<tr>
<td>Electron density ($\text{m}^{-3}$)</td>
<td>$4 \times 10^{20}$</td>
</tr>
<tr>
<td>ISS95 confinement multiplier</td>
<td>2</td>
</tr>
<tr>
<td>Alpha-particle loss fraction (%)</td>
<td>5</td>
</tr>
<tr>
<td>Fusion power (MW)</td>
<td>2440</td>
</tr>
<tr>
<td>Thermal power (MW)</td>
<td>2920</td>
</tr>
<tr>
<td>Thermal conversion efficiency (%)</td>
<td>43</td>
</tr>
<tr>
<td>Average neutron wall load (MW/m$^2$)</td>
<td>2.6</td>
</tr>
<tr>
<td>Peak neutron wall load (MW/m$^2$)</td>
<td>5.4</td>
</tr>
</tbody>
</table>
Based on the above observation, three avenues were pursued to reduce the size:

1. Develop stellarator configuration with lower aspect ratios.

2. For a given stellarator configuration (i.e., plasma aspect ratio), increase the coil-plasma spacing by developing a coil design with a lower coil aspect ratio and optimizing the coil cross section.

3. Develop engineering options to reduce the required minimum distance between the plasma and the midpoint of the coil ($\Delta_{\text{min}}$).

II.A.1. Quasi-Axisymmetric Configuration

In recent years drift orbit optimization techniques have been widely used to develop stellarator configurations. In particular, the ability to achieve topological symmetry as seen by particles can be exploited to optimize the properties of the configuration such as quasi-helical symmetry utilized in the Helically Symmetric Experiment (HSX); quasi-axisymmetry utilized in the National Compact Stellarator Experiment (NCSX), which is under construction in the United States; quasi-isodynamicity and linked mirrors utilized in W7-X under construction in Germany; and quasi-poloidal symmetry utilized in the proposed Quasi-Poloidal Stellator (QPS) experiment. In addition to the desired symmetry, these techniques can be utilized to optimize other plasma properties (such as magnetic shear, magnetic well depth, and the amount of external rotational transform), measures of magnetohydrodynamic (MHD) stability (such as external kinks, vertical displacement, and infinite-n ballooning modes), and figures of merit for transport (such as the effective ripple or diffusion coefficient evaluations).

We focused our analysis on QAS configurations as they are able to operate at a lower plasma aspect ratio (4 to 5) compared to other stellarator configurations. Some efforts on optimizing a configuration with quasi-helical symmetry were made; however, we could not achieve desirable low-aspect-ratio configurations. We did not investigate quasi-poloidal-symmetric configurations because of the lack of resources.

The particle drift orbits in a QAS configuration are similar to that of a tokamak. As such, it is argued that a QAS configuration would have good confinement similar to tokamaks. Also, because of the quasi-axisymmetry, the neoclassical bootstrap current is also similar to tokamaks, but the magnitude is reduced by the higher rotational transform. The low-aspect-ratio QAS configurations, therefore, tend to be like hybrids between tokamaks and conventional stellarators.

Because of the large research effort spent in developing the NCSX configuration, the ARIES-CS configurations naturally evolved from NCSX equilibrium. The major critical issue of the NCSX configuration for a fusion plasma is its high alpha-particle loss rate. Our efforts to reduce the alpha-particle loss rate led to new criteria for optimizing the QAS configuration and led to the baseline plasma configuration used for engineering analysis and trade-off studies, N3ARE (see Fig. 3 and Sec. III).

We also developed two new classes of the QAS configuration in which strict adherence to linear, ideal MHD stability constraints was relaxed. This was due to the recent experimental results from the Wendelstein 7-AS (W7-AS) and Large Helical Device (LHD) stellarators. Average beta values of 3.2 and 4.2% have been achieved on W7-AS and LHD, respectively, limited only by the available heating power and perhaps the integrity of the equilibrium flux surfaces. These beta values were maintained for 80 to 100 energy confinement times, and while MHD activity apparently existed and was active in some cases, the plasmas nevertheless were quiescent and remained quasi-stationary. In both cases, the achieved experimental values were higher than those predicted from linear stability theory (e.g., 3.2% achieved beta in W7-AS versus 2% for the theoretical prediction).

The first class of the new configurations is MH2, which aims at developing a very low-aspect-ratio geometry with relatively simpler coils (see Sec. III.B). The second class, SNS, is aimed at developing a configuration with excellent flux surface quality and nearly flat rotational transform (Sec. III.C).

II.A.2. Coil Design

Coils that produce the designed target plasma shape may be constructed by requiring that the normal component of the magnetic field on the LCMS produced by the coils cancels that due to the plasma current. Several coil designs with different coil aspect ratios were produced in order to provide scaling of the system size for trade-off studies. In addition to different aspect ratios, the coil cross section (shape and size) was varied. These coil designs included certain additional constraints of coil separation ratio and minimum radius of curvature. As discussed before, reducing the coil aspect ratio would increase the available space for fusion core components. However, as the coil aspect ratio defined as $A_c = (R)/\Delta_{\text{min}}$ is reduced (i.e., $\Delta_{\text{min}}$ is increased for a given $\langle R \rangle$), the coils generally become more complicated, and the peak field on the superconducting coil ($B_{\text{max}}$) is increased (for a given field in the plasma center $\langle B \rangle$). For example, we found that for the N3ARE configuration, $B_{\text{max}}/\langle B \rangle$ increases rapidly if the coil aspect ratio is reduced below ~5.8 (see Fig. 9 of Ref. 8). The baseline coil design has a coil aspect ratio of 5.8 with rectangular cross-section conductors (~0.19 m radially and ~0.74 m toroidally). The maximum field in the winding pack of the coils is ~15 T for an on-axis plasma field of 5.7 T (see Fig. 2).
II.A.3. Reducing Required $\Delta_{\text{min}}$

A key parameter is the minimum required plasma-to-midcoil distance ($\Delta_{\text{min}}$). This space accommodates the scrape-off layer, first wall, blanket, shield, manifolds, vacuum vessel, assembly gaps, coil case, and half of the winding pack. Typically, the distance between the plasma and the front of the coils is set by the requirements of adequate breeding and shielding of the coils (the coil radial thickness is optimized separately as discussed in Sec. III.B). An innovative approach was developed to downsize the blanket and utilize a highly efficient tungsten-carbide (WC)–based shield in the space-constrained regions where plasma is close to the coil. The special modules in these regions utilize a non-uniform blanket and a WC shield, optimized to provide shielding comparable to a regular breeding module but with a much reduced module radial thickness, as shown in Fig. 4. From Fig. 4, the typical plasma-to-midcoil radial thickness for a regular breeding module is $\sim 1.79$ m but is only $1.31$ m for an optimized module with a reduced breeding zone but with similar shielding properties. The final design of the blanket modules in those regions with minimum plasma-to-midcoil distances (for example, see the “midplane” inboard region in Fig. 1) consists of a tapered breeding region with a thickness ranging from $25$ cm at the minimum space location to $54$ cm in regions where more space is available as is shown in Fig. 5. The total tritium breeding ratio including all modules is $\sim 1.1$, found from detailed three-dimensional (3-D) neutronics analysis of the computer-aided design (CAD) model of ARIES-CS (see Sec. II.B.1 and Ref. 11). Using a tapered blanket has had a major impact on the machine size: A device with a uniform blanket and shield at all locations would have a major radius $>10$ m compared to $7.75$ m for ARIES-CS (see Sec. V.A.2).

II.B. Complex Shape and Geometry

Stellarator fusion core components will have complex shapes and geometry. This complexity imposes severe constraints on fusion core components such as nonuniform heat, particle, and neutron fluxes; accessibility for assembly and maintenance; and feasibility and cost of

Fig. 3. LCMS of the ARIES-CS baseline plasma. (a) Top and perspective view and (b) four cross sections at equal intervals over half a period.
manufacturing. The second major goal of the ARIES-CS study was to understand and quantify, as much as possible, the impact of the complex shape and geometry of fusion core components.

II.B.1. Need for 3-D Analysis

It became evident early on that the 3-D shape of the plasma and the coil (and the components between them) necessitates 3-D analysis of various components—typical correlations and insight developed for axisymmetric fusion devices are not appropriate for stellarator geometry. As such, we directly used 3-D CAD models in many of our analyses. Moreover, we found that the results are quite sensitive to the details of the 3-D shape of components, and slight variations can result in substantial changes.
For example, nuclear assessment of the system (breeding, shielding, blanket thermal recovery) requires utilization of 3-D neutron transport codes such as the MCNPX Monte Carlo code. We developed a new tool that uses CAD information to provide 3-D source terms for both fusion neutrons generated in the plasma and radiation from the plasma (bremsstrahlung and impurity line radiation). This CAD/MCNPX interface also uses CAD models for in-vessel components (i.e., first wall, blanket, etc.) to generate various zones for MCNPX (Ref. 11).

An important result from exercising this tool is the distribution of the neutron and heat loads on the first wall, which greatly influence the thermal performance and lifetime of the blanket. Figure 6 shows the contours of neutron wall loading on the first wall of the reference ARIES-CS design (the first wall follows the LCMS shape and accommodates a uniform 5-cm scrape-off-layer thickness). For this case, the peak neutron wall loading is 5.3 MW/m², and its location is denoted by a white circle in Fig. 6. Similar to other toroidal devices (e.g., tokamaks), the neutron wall loading is lowest at the top and the bottom (around ±90-deg poloidal angle) and highest on the midplane with the peak value located at the outboard location (0 poloidal angle) rather than the inboard location (±180-deg poloidal angle). However, Fig. 6 also shows a large spatial variation in the neutron wall loading (e.g., compared to that of a tokamak) and underlines the need for 3-D analysis of nuclear components. Similarly, we have found large spatial variation in the heat flux incident on the first wall11 but with a different profile compared to that of the neutron load. For example, the heat flux on the inboard side (±180-deg poloidal angle) is comparable to that of the outboard side in many locations.

Of particular importance is the large peaking factor of ~2 in the neutron wall loading (peak value of 5.3 MW/m² and average value 2.6 MW/m²), which is considerably larger than tokamaks (1.5 for ARIES-AT). This rather large peaking factor and the similarly large peaking factor for heat fluxes have a dramatic impact on the engineering design of the system. For a given power, the size of the device sets the average power loading on the components. However, components should be designed to withstand the peak loadings, which typically leads to a lower performance for these components (e.g., the radiation lifetime of the component is set by the peak neutron flux).

II.B.2. Configuration, Assembly, and Maintenance

We have found that the engineering configuration and the assembly and maintenance procedure are key elements in optimizing a compact stellarator; in some cases, these issues determine the choice of technologies that can be utilized.

We considered several assembly and maintenance procedures early in the study. Sector-based maintenance, similar to that envisioned for ARIES-AT, was ruled out because of the irregular shape of the coils. A field-period maintenance scheme, i.e., the replacement of integral units based on a field period including disassembly of the modular coil system, was also considered.20 We judged this scheme not to be feasible because it involves movement of massive cryogenic structures as well as the need for fine realignment during reassembly. As such, port-based maintenance, i.e., the replacement of the blanket and other in-vessel components through a number of designated ports, appears to be the only feasible assembly and maintenance scheme. This is a complicated and time-consuming procedure as the size of the ports as well as the weight limit of the maintenance boom limits the size of a typical blanket module to ~3 to 4 m² on the side facing the plasma (about 200 such modules in ARIES-CS). Furthermore, module removal and replacement require the cutting and rewelding of the coolant pipes, which necessitates removal of a neighboring module first (as discussed in Sec. IV and Ref. 10).

The need for a large number of modules with coolant pipes that can be cut and rewelded in a relatively

![Fig. 6. Contour maps of the neutron wall load (in MW/m²) of ARIES-CS. The approximate location of the peak is indicated with a circle. The toroidal angle is measured from the beginning of the field period while the poloidal angle is measured from the outboard midplane at the level of the magnetic axis.](image-url)
short time drove us toward the choice of a ferritic steel–
based blanket as a reference option instead of a higher-
performance blanket utilizing SiC composite as structural 
material [e.g., SiC/Pb-17Li blanket of ARIES-AT (Ref. 7)], 
which was kept as a possible backup. The reference 
blanket concept is a dual-coolant concept with 
self-cooled Pb-17Li zones and a He-cooled ferritic steel 
structure.

The choice and design of superconducting magnets 
are another example of the impact of the configuration on 
the choice of available technologies. The candidate su-
perconductor materials for operation at high field (up to 
16 T) are variants of Nb$_3$Sn or MgB$_2$. Both are glassy 
materials, and their current-carrying capability is drasti-
cally reduced with induced strain. For these types of 
superconductors, the superconductor material is wound 
around a large spool (with roughly the same radius of 
curvature as the coil itself) and is heat treated (700°C for 
a few hundred hours) to attain superconducting prop-
erties. The resultant cable is then gently unwound from 
the spool, insulated, and wound onto the coil. The reason 
for this two-step process is that organic insulators cannot 
survive the heat treatment process and need to be added 
afterward. Because of the irregular shape of the coils, this 
method cannot be used in a compact stellarator because 
of excessive strain induced during the winding process. 
In order to keep the strain down, the superconductor 
material should be wound as a complete cable (including 
the insulator) on the coil and then heat treated. This method 
requires development of inorganic insulators that can 
withstand the heat treatment process. These issues are 
discussed in more detail in Ref. 14. It should be noted 
that the strain induced during the winding process is not 
an issue for ductile NbTi superconductors. However, the 
maximum field $B_{\text{max}}$ is limited to ~7 to 8 T for 4 K 
operation (up to 9 T can be achieved by operation at 2 K, 
although there are issues with temperature margin). This 
reduction in maximum field strength will lead to a sub-
stantial increase in the machine size.

II.B.3. Costing and Manufacturing Feasibility

As most of the cost of various components in a fu-
sion system is due to the manufacturing process (and not 
the cost of the raw material), it is expected that there will 
be additional costs associated with the irregular and com-
plex shape of the stellarator components. Our attempt to 
get guidance from industry on cost penalties associated 
with the complexity of components were unsuccessful 
because of the lack of industrial experience in manufac-
turing complex components on a large scale. It was gen-
erally reported that manufacturing of these components 
with conventional techniques would be quite expensive 
and challenging. However, advanced manufacturing tech-
niques, under development, would allow for the compo-
nent size not to be a limiting factor and the process to be 
highly automated with minimal labor. In principle, ad-
vanced manufacturing techniques would result in a much 
lower additional cost for complex components. These 
advanced manufacturing techniques are more suitable 
for a monolithic structure such as the superconducting 
coil structure as opposed to heterogeneous structures such 
as blanket modules. As such, we have assumed such an 
advanced manufacturing technique in designing our coil 
support structure and in estimating the COE. In addi-
tion, we did not include any cost penalties for manufac-
turing of other components or for machine assembly.

While this approach still allows us to use the COE as a 
good optimization parameter for understanding the trade-
off among stellarator features and optimizing the design, 
care should be taken in cross-comparison with other fu-
sion concepts and/or other sources of energy. Table II 
provides a cost estimate of various components of ARIES-
CS. The parametric dependence of the COE on addi-
tional cost penalties to complexity for heterogeneous 
components (e.g., blankets) is reported in Ref. 9. For 
example, applying a 25% cost penalty to each major com-
ponent separately increases the COE by 0.37 $\$/kWh 
(0.1 $\$/kWh for the blankets, 0.14 $\$/kWh for the shields, 
and 0.13 $\$/kWh for the coils and support structure).

III. PLASMA PHYSICS

We have found that in many cases, a slight increase 
in the major radius (to ~8.25 m) would lead to a more 
robust engineering design for the components.

In recent years, the technique of drift-orbit optimi-
ization has been widely used to arrive at modern stellar-
ator configurations. These techniques are based on the 
observation that particle drift trajectories in “Boozer” 
coordinates depend on only the strength of the local mag-
netic field (and not on the vector components of the 
field). These extra degrees of freedom mean that many 
3-D plasma geometries with well-confined particle tra-
jectories can be constructed and they can be optimized to 
have other described properties. For example, QAS con-
figurations aim at achieving an axially symmetric field 
strength (in Boozer coordinates) that leads to particle 
orbits being similar to those in a tokamak. As such, this 
class of configurations has the potential to combine the 
desirable features of tokamaks (good confinement and 
moderate aspect ratio) with those of large-aspect-ratio 
stellarators (steady-state operation, stability against ex-
ternal kinks and axisymmetric modes, and resilience to 
disruptions).

Because of the desire to reduce the size of a stellar-
ator power plant and the need to reduce the plasma aspect 
ratio, the ARIES-CS study focused on the QAS config-
uration. As a large research effort has been spent in 
developing the NCSX configuration, the ARIES-CS con-
figurations naturally evolved from NCSX equilibrium. 
The most important issue in developing the ARIES-CS
reference plasma was to develop NCSX-like configurations with low alpha-particle loss (see Sec. III.A). We were also successful in increasing the separation of coils from the plasma.

We also developed two new classes of QAS configuration in which strict adherence to linear, ideal MHD stability constraints was relaxed while the quality of the flux surfaces and transport, particularly the confinement of alpha particles, were emphasized. The first class of the new configurations is MHH2, which aims at developing a very low-aspect-ratio geometry with relatively simpler coils. The second class, SNS, is aimed at developing a configuration with excellent flux surface quality and a nearly flat rotational transform to explore how good and robust flux surfaces can be designed. These configurations have not been examined in detail in the engineering designs but were evaluated with the system code.9

The ARIES-CS reference plasma, N3ARE (Fig. 3), is an NCSX-like configuration with three field periods and has a major radius of 7.75 m and a plasma aspect ratio of 4.5. The configuration has excellent quasi-axisymmetry, as measured by the effective helical ripple, \( e_{\text{eff}} \) \( (e_{\text{eff}} < 0.6\% \) at LCMS and \( \sim 0.1\% \) in the core region). The profile of the ARIES-CS rotational transform at \( \beta = 5\% \) is shown in Fig. 7. The shaping of the plasma results in a vacuum rotational transform from \( \sim 0.4 \) to \( \sim 0.5 \). The plasma current (bootstrap current) is \( \sim 4 \) MA, which raises the rotational transform to \( \sim 0.7 \) near the plasma edge.

### III.A. Alpha-Particle Loss

The distinct feature of this configuration is that a bias is introduced in the mirror and helical terms of the magnetic spectrum in order to alter the structure of the ripple and improve alpha-particle confinement.8 In addition, the ARIES-CS reference plasma operates at low temperature \( \langle T \rangle = 6.6 \) keV; \( \langle T \rangle \) is the density weighted volume-averaged temperature and high density (volume-averaged density of \( 4 \times 10^{21} \) m\(^{-3} \)) in order to decrease alpha-particle slowing-down time and further reduce alpha-particle losses. In this manner, the alpha-particle loss in ARIES-CS is reduced to \( \sim 5\% \). While the heat flux from these lost alpha particles may be handled on the divertor target plate, erosion and, in particular, exfoliation due the accumulation of He atoms in the armor are major concerns (see Sec. IV.B and Ref. 10). As such, developing configurations with lower alpha-particle loss is a key research area.

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<td>22.1.3</td>
<td>Magnets</td>
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<td>22.1.4</td>
<td>Supplemental heating/current drive systems</td>
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<td>Primary structure and support</td>
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<td>Direct cost (not including contingency)</td>
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\(^a\)No cost penalty has been assumed for manufacturing of complex components. For example, applying a 25% cost penalty to major components (blanket, shield, and coils) increases the COE by 0.37 \( \epsilon/\text{kW} \cdot \text{h} \).

\(^b\)Assumes an 85% availability (similar to ARIES-AT).
to the smallest penalty in coil manufacturing and cost as related to a power plant would require a set of poloidal field coils. In addition to the modular coils of Fig. 2, a compact stellarator was designed. These PF coils were not performed; rather, the total current and cost of superconducting magnets depends on the details of the pressure and current profiles, a sensitivity analysis was performed by considering a wide range of pressure and density profiles (and the resulting bootstrap current profile). We found that only slight modification of the plasma boundary is needed to ensure stability for a wide range of pressure profiles. This is indicative of the robust MHD properties of the baseline configuration.

As discussed before, the stellarator experiments have operated at beta values substantially larger than predicted by the linear MHD theory (nonlinear theory appears to indicate a higher stability boundary than that calculated by the linear MHD theory). Understanding the beta limit in stellarators is a critical issue that will have a large impact on the optimization of the stellarator configuration. In the absence of such an understanding, we have used linear stability theory as a guide to optimize the configuration; however, we have assumed that the beta value for the reference design is 6.5% (60% larger than predicted from linear stability theory of 4%). Variation of machine parameters with plasma beta is investigated parametrically; for example, if plasma beta is limited to 4%, the major radius of the device has to be increased to ~8.25 m with a 7% penalty in the COE.

III.D. Transport

Similar to stability, the database for transport in drift-optimized stellarators is limited. Because of the low effective ripple, neoclassical transport is expected to be small in the ARIES-CS configuration. The required energy confinement time for ARIES-CS is ~1 s and is a factor of \( H = 2 \) larger than the value predicted by the widely used ISS95 scaling for the energy confinement in stellarators. Data from W7-AS and LHD lie above the ISS-95 scaling (values for H-ISS95 up to 2.5 have been achieved). It is thought that lower values of the effective

---

**Fig. 7.** Profiles of ARIES-CS rotational transform as a function of the normalized toroidal flux \( \beta = 5\% \). Dotted line is the transform due to the deformation of the plasma surface. Solid line is the total transform including the contribution from the plasma current.

**III.B. Coil Design**

Several coil designs were produced with different \( \Delta_{\text{min}} \), the distance between the LCMS and the middle of the winding pack (parameterized by the coil aspect ratio, \( A_c = \langle R \rangle / \Delta_{\text{min}} \)). Another important parameter is the ratio of the maximum field on the coil \( B_{\text{max}} \) to the field at the plasma axis \( B_0 \) (\( B_{\text{max}} \) directly impacts the choice, design, and cost of superconducting magnets). We have found that there is an optimum coil aspect ratio of \( \sim 6 \). Above this aspect ratio, \( B_{\text{max}} / B_0 \) increases rapidly. This is an optimum point as it allows for the largest \( \Delta_{\text{min}} \) with the smallest penalty in coil manufacturing and cost as related to \( B_{\text{max}} \). In addition, we examined the impact of the coil cross section (square versus rectangular cross section). Ribbon-like coils (i.e., thin in the radial direction and wide in the toroidal/poloidal direction) lead to smaller \( B_{\text{max}} / B_0 \) as the effective distance between the coil center and plasma is reduced. A coil cross section of \( 0.194 \times 0.743 \text{ m} \) was chosen for ARIES-CS to minimize \( B_{\text{max}} / B_0 \) while ensuring sufficient separation between coils for coil cases and other components. The ARIES-CS reference coil design is shown in Fig. 2. It consists of 18 coils over the three-field period (three different coil shapes). The maximum field on the coil is \( \sim 15 \text{ T} \) (for \( B_0 = 5.7 \text{ T} \)) necessitating the use of Nb$_3$Sn superconductors. In addition to the modular coils of Fig. 2, a compact stellarator power plant would require a set of poloidal field (PF) coils to provide control of the plasma equilibrium in response to pressure changes and start-up. Trim coils may also be needed to ensure the integrity of the flux surfaces and to correct for field errors. Because of lack of resources, detailed analysis of the location and current of these PF coils was not performed; rather, the total current capability was estimated by scaling the relevant NCSX analyses.
helical ripple ($\varepsilon_{\text{eff}}$) may play a role in the improved confinement. This correlation suggests that large H-ISS factors should be possible for the very low $\varepsilon_{\text{eff}}$ quasi-symmetric compact stellarators. Assuming ISS95 scaling for energy confinement (with $H = 2$), ARIES-CS requires \sim 20 MW of auxiliary heating to achieve ignition.

We focused on an operation with high density because of the $\bar{n}_e$ dependence in ISS95 scaling and also the belief that high-density operation would help in reducing the heat flux on the divertor plates. The design point was chosen to be thermally stable (via POPCON plots) but very close to the thermal stability limit. As such, the ARIES-CS plasma density is 1.5 times higher than the Sudo density limit\textsuperscript{31} but is consistent with recent experimental results.\textsuperscript{32} Toward the end of the study with the engineering design of the component completed, we found that the plasma is thermally unstable (slightly) for $H = 2$ constraints (mainly due to the field on axis being slightly lower than estimated). The ARIES-CS plasma can be made thermally stable either by assuming a slightly larger $H (> 2.1)$ and/or slightly increasing the size of the device ($\sim 8$ m).

### III.E. Divertor

The heat and particle load on the divertor plates were calculated by adding the contributions of the thermalized particles crossing the LCMS and the escaping fast alpha particles. The analysis was based on tracing field lines from the plasma edge to the divertor plates and assuming that the parallel transport can be represented by field-line mapping and that cross-field transport can be modeled with a prescribed field-line diffusion scheme.\textsuperscript{12} In ARIES-CS (similar to NCSX), the field lines outside the LCMS make a transition from ergodically covering a volume to becoming moderately stochastic as one moves away from the LCMS. As there is a significant flux expansion at the top and bottom of the bean-shaped cross section, target plates are located mainly in these regions. The ARIES-CS divertor concept consists of two pairs of target plates per field period, one pair each at the top and bottom of the plasma (see Figs. 1 and 8). The poloidal and toroidal extent of the plates and their shape and distance to the plasma are designed to intercept all the heat flux and to minimize the peak thermal heat load.

Similar to advanced tokamaks, a large portion of plasma power should be radiated in order to obtain a reasonable heat flux on the divertors. In the absence of a detailed physics model for stellarator edge plasma and the complexity of 3-D analysis, we assumed that 75% of the core plasma is radiated by the addition of impurities (e.g., 0.0008% iron). In addition, 75% of power entering the scrape-off layer is also radiated through impurity injection similar to methods proposed for tokamaks. In this manner, only 5% of plasma power (23.6 MW) appears as conduction power on divertor plates (in addition to an alpha-particle loss power of 23.6 MW). An engineering design of the divertor plates was also produced that can handle 10 MW/m$^2$ of heat flux (see Sec. IV.B and Ref. 10).

To assess the divertor heat flux due to thermalized particles crossing the LCMS, an iterative process was used in which for a given divertor geometry, the STELLA field-line tracing code was utilized to find the distribution of the heat load on the divertor plates. The divertor plate geometry was adjusted, and the process was repeated to further reduce the peaking factor of the heat flux on the divertor plates. This is a complex procedure as a large number of field lines (e.g., 128,000) should be traced in order to achieve good statistics. Furthermore, because of
the 3-D nature of the LCMS, many iterations of divertor plate location, orientation, and plate topology are needed to find an optimum position.

To assess the divertor heat flux due to the escaping energetic alpha particles, we first used the ORBIT3D code to determine the footprint and the energy distribution of the alpha particles leaving the LCMS (Ref. 8). The STELLA field-line tracing code was then used to follow these energetic alpha particles and compute the resultant heat flux on the divertor plates.12

Because of the lack of resources, only a limited number of cases were examined.12 By tailoring the divertor plate, we were able to reduce the peak combined conducted plasma and escaping energetic alpha particle to \( \sim 18 \text{ MW/m}^2 \), which far exceeds the design limit of divertor plates of \( \sim 10 \text{ MW/m}^2 \). However, we believe that further refinement of divertor target shapes can reduce the peak heat flux to the desired value.

IV. FUSION TECHNOLOGIES

The engineering design of the fusion power core components and the coil structural design are briefly described in the next sections, and key stellarator-specific challenges affecting the design are highlighted. These include the impact of the minimum plasma-coil distance, the coil design requirements, and the need for alpha-particle power accommodation. More details on the engineering design and analysis can be found in Refs. 10 through 17.

IV.A. Blanket

A dual-coolant configuration with a self-cooled Pb-17Li zone and He-cooled reduced-activation ferritic steel (RAFS) structure was selected as the reference concept based on its relatively good combination of performance, design simplicity, safety features, and modest R&D needs.10 A modular concept was adapted for the ARIES-CS compact stellarator geometry compatible with the port-based maintenance scheme. Figure 9 shows details of the module layout.10 The helium coolant is routed to first cool the first wall and then all other structural walls (see Fig. 10). The Pb-17Li flows slowly in the large inner channels in a two-pass poloidal configuration (see Fig. 11). The Pb-17Li channels (and manifolds) are lined with a SiC insulating layer. This insulating layer together with He cooling of the structure maintains the Pb-17Li/FS interface temperature below its compatibility limit while allowing for a higher Pb-17Li temperature in the channel. This SiC layer also provides the electrical insulation needed to minimize the MHD effects on the Pb-17Li pressure drop. Cooling of the first-wall region of the blanket (where the heat load is highest) with He (instead of Pb-17Li) avoids the need for an electrically insulating coating in this high coolant velocity region. Such an insulating coating would be needed in order to prevent the large MHD pressure drop associated with liquid metal flow. In addition, the He coolant allows for preheating of the blanket and serves as an independent and redundant afterheat removal method and enhances the safety performance of the blanket.

The blanket is coupled to a Brayton cycle through a heat exchanger where both coolants (He and Pb-17Li) transfer their energy to the cycle working fluid (He). Detailed thermal-hydraulic analyses were performed to optimize the power production within the given material constraints (including a maximum Pb-17Li/FS interface temperature <500°C and an RAFS maximum temperature <550°C). These are described in Ref. 10, along with all the supporting design and analysis results for the blanket. As an illustration of the parametric analysis results, Fig. 12 shows the variation of the cycle efficiency with the first-wall surface area for a given fusion power. Also shown are the blanket He pumping power and the corresponding net efficiency (based on the gross electrical power minus the coolant pumping power). The corresponding maximum neutron wall load is indicated on the upper x-axis. Stress analyses, described in detail in Ref. 10, indicate that the design has been pushed to its limit and that it would be beneficial to relax the heat load to provide more margin (for example, by increasing the machine size). Some of the major blanket parameters for the reference design case are listed in Table III.

A key engineering parameter affecting the size of a compact stellarator is the minimum coil-plasma distance. A novel approach has been developed where a highly efficient WC shield is used in the critical area to minimize this distance, as discussed in Sec. II.A.3. In this way, the
normal blanket/shield/manifold thickness of 1.76 m can be reduced to ~1.3 m by optimizing the shield while the loss of breeding in these regions can be compensated by optimizing the breeding in other regions (to maintain the
the physics modeling results with the capability to accommodate a heat flux of 10 MW/m² (as a reasonable initial goal in anticipation of the physics modeling results) and that can be integrated with the in-reactor component design. In addition, the alpha-particle loss fraction is fairly high, ~5%, and concerns exist as to the accommodation of the resulting heat load and particle flux on local regions of the first wall.

The divertor configuration consists of a He-cooled “T-tube” illustrated in Fig. 13. The T-tube is ~15 mm in diameter and ~100 mm long and is made up of a W-alloy inner cartridge and outer tube. A W armor layer is attached to the top of the outer tube. The design provides some flexibility in accommodating the divertor area since a variable number of such T-tubes can be connected to a common manifold to form the desired divertor target.

As illustrated in Fig. 13, the helium coolant (10 MPa) is routed through the inner cartridge first and then pushes through thin slots (~0.4 mm) to cool the heat-loaded outer tube surface. A two-dimensional-shaped impinging slot jet is created, leading to high heat transfer at reasonable pressure drop. After impingement, the coolant flows as a highly turbulent wall jet along the large inside surface of the tube and then returns in the lower section of the annular gap between the tube and cartridge. The inlet and outlet He temperatures are ~573 and ~700°C, which fit within the overall heat exchanger and Brayton cycle scheme. A stress analysis was also performed indicating that the total stress intensity (primary and secondary stresses) is <370 MPa for the entire geometry, which is assumed to be less than the 3σm limit of an anticipated W alloy at the corresponding temperatures and is considered to be acceptable. Some of the major divertor parameters for the reference design are also listed in Table III. The thermal-hydraulic performance of the concept has been verified through laboratory-scale experiments. Other R&D issues include the development of the W-alloy and fabrication techniques.

The peak heat flux due on the divertor plate due to the combined conduction and alpha-particle power is estimated at ~18 MW/m², far exceeding the design limit for the T-tube divertor design. We believe that further work on shaping the divertor plates may be able to reduce the peak heat flux to an acceptable level. However, a major concern is the possible erosion due to the energetic He flux, in particular, any exfoliation that could result from the accumulation of He atoms in the armor. A possible solution would be to use an engineered W armor with a low-porosity nanostructure that could enhance the release of He. Clearly, this is a key issue for compact stellarators.

### IV.B. Plasma-Facing Components

A concerted effort on the divertor has been launched as part of the ARIES-CS study. On the physics side, it involves adapting and using codes to better assess the location of the divertor and to estimate the corresponding heat loads. On the engineering side, the effort focused on evolving a design well suited to the compact stellarator with the capability to accommodate a heat flux of 10 MW/m² (as a reasonable initial goal in anticipation of the physics modeling results) and that can be integrated with the in-reactor component design. In addition, the alpha-particle loss fraction is fairly high, ~5%, and concerns exist as to the accommodation of the resulting heat load and particle flux on local regions of the first wall.

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### IV.C. Maintenance Scheme and Machine Layout

A port-based maintenance scheme has been selected due in great part to the 3-D geometry constraints associated with a compact stellarator. The scheme involves the replacement of blanket modules and divertor plates using an articulated boom through designated ports (one major port and one auxiliary port per field period for the three-field-period configuration). As illustrated in Fig. 1, the

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**TABLE III**

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<td>Thermal power removed by He (MW)</td>
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<td>Pumping power (MW)</td>
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*Including ~111-MW reduction due to conducted power to He.*
*Including 141 MW of friction power + 111 MW of conducted power from Pb-17Li.*
*Including 23.6 MW of alpha-particle loss power and 24 MW of friction power.*

3-D tritium breeding ratio at ~1.1). More details on the neutronics analysis in support of the power plant design can be found in Ref. 11.

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vacuum vessel is internal to the coils and serves as an additional shield for the protection of the coils from neutron and gamma irradiation. During maintenance, no disassembling and rewelding of the vacuum vessel is required, and the modular coils are kept at cryogenic temperatures. While maintenance through ports is expected to lead to a longer maintainer period and a lower availability, we have assumed that the power plant availability is similar to that of tokamaks in our costing analysis.

### IV.D. Coil Design

The complex 3-D geometry of the coil configuration in a compact stellarator introduces engineering constraints affecting a number of design features such as the distance between the plasma and coil, the minimum coil bend radius, the coil support requirement, and assembly and maintenance. The preferred superconductor is Nb$_3$Sn, operating at ~4 K. It is installed with a wind and react method and heat treated. JK2LB, with a yield strength of 1420 MPa at 4 K, is used as structural material.

The overall coil system consisting of the intercoil structure, coil cases, and winding packs is enclosed in a common cryostat. The coils are wound into grooves at the inside of a strong supporting toroidal tube for each field period, as shown in Fig. 14. These tubes are then connected to each other using a bolt arrangement to provide a continuous ring structure to react the large centering force that pulls the coil toward the center of the torus. The out-of-plane forces acting between neighboring coils inside a field period are reacted by the intercoil structure, while the weight of the cold coil system is transferred to the foundation by three legs per field period. Because of the complex geometry of the coils, 3-D magnetic forces and stress analyses were performed using ANSYS to characterize the stress and displacement of the coil structure. The results indicated that the stress and deformation can vary over a substantial range in different regions of the structure, offering the possibility of reducing unnecessary structure in regions where the stresses are low in order to minimize costs. These are described in detail in Ref. 14.

### IV.E. Safety Analysis

Safety studies were performed to demonstrate that a site evacuation plan is not needed and that the radioactive waste shall be minimized. The results indicate that through the use of reduced-activation materials, a passive decay heat removal system, and the defense-in-depth confinement strategy, the radioactive source terms of ARIES-CS are contained during the design-basis accidents analyzed to the degree that no site evacuation plan is needed for ARIES-CS. Activation analyses indicate that the internal components of ARIES-CS could be disposed of in low-level waste repositories. In addition, portions of the magnets,
cryostat, and bioshield could meet the national and international clearance criteria for release to private industrial markets.

V. DESIGN EVALUATION AND R&D NEEDS

In considering the results from the ARIES-CS study, it should be emphasized that this was the first major study of a compact stellarator. As such, in order to fully understand the technical issues and trade-offs resulting from the “compactness” of the configuration, the reference ARIES-CS design is the minimum-COE design point that was the smallest in size. In most areas, we found that increasing the machine size compared to the ARIES-CS reference design would provide more margins on space and engineering constraints such as material stress, temperature limits, etc., with small cost penalties (<1%) (Ref. 9). Also, most of the engineering research was performed on NCSX-class quasi-asymmetric configurations. Caution should be used in extrapolating to other compact stellarator configurations.

V.A. Design Evaluation

Overall, ARIES-CS demonstrates that compact stellarator power plants that are comparable in size to advanced tokamaks are feasible. By focusing on QAS configurations, we have managed to simultaneously reduce the coil aspect ratio and alpha-particle losses while maintaining desirable stellarator properties. Furthermore, the configuration space for QAS stellarators is rich with desirable configurations, and many trade-offs may lead to more attractive configurations. In particular, the mechanism governing the beta limit is not understood. Relaxing the criteria for linear MHD stability may lead to configurations with less complex coil and machine geometries.

On the engineering side, we found that not only does the irregular shape of components necessitate 3-D analyses in most cases but also the results are quite sensitive to details of the 3-D shape of components and slight variations can have a substantial impact. Because of this sensitivity, optimization of the component shape with respect to one parameter (e.g., sculpting the first wall to arrive at a more uniform neutron load) may lead to major issues in other systems. Overall, we find that a high degree of integration in the analyses and design is essential. Exploring such a 3-D optimization of all components was beyond the scope of ARIES-CS and has been left for future studies.

The device configuration, assembly, and maintenance drive the design optimization and technology choices in many cases. Examples include a port-based maintenance scheme that drives the internal design of the fusion core and led to the choice of a ferritic-steel, dual-coolant irregular shape of the superconducting coil that necessitates development of inorganic insulators for high-field magnets.

Lack of industrial guidance on feasibility and cost of manufacturing of irregularly shaped components has prevented an important optimization dimension (increase of size and mass to lower complexity as compared to more compact but also more complex devices). Lessons learned from the construction of NCSX and W7-X would be valuable information for follow-up studies of compact stellarator power plants.

Some of our more specific findings are listed below.

V.A.1. Beta Limit

The configuration space for QAS stellarators is quite rich, and many desirable configurations are possible. An important constraint in optimizing a configuration is the plasma beta limit. The mechanisms limiting the plasma beta in a stellarator are not well understood as stellarator experiments have operated at beta values substantially larger than predicted by the linear MHD theory. In the absence of such an understanding, we have used linear stability theory as a guide to optimize the configuration; however, we have assumed that the beta value for the reference design is 6.5% (60% larger than predicted from linear stability theory of 4%). Variation of machine parameters with plasma beta is investigated parametrically; for example, if plasma beta is limited to 4%, the major radius of the device has to be increased to ~8.25 m with 7% penalty in the COE. Understanding the mechanism that limits the plasma beta is an important physics topic that should be addressed in the next generation of stellarators. Not only would relaxing the linear MHD stability constraint allow optimization of other parameters (e.g., alpha-particle losses), but also a larger beta value can be used to reduce the technological requirement of the superconducting magnets.

V.A.2. Tritium Breeding

Tritium breeding played an important role in defining the ARIES-CS major radius. For compact stellarators based on an NCSX-type configuration, a major radius, \( R \geq 7.5 \text{ m} \), is needed for tritium self-sufficiency. For \( 10 \text{ m} \geq \langle R \rangle \geq 7.5 \text{ m} \), a variable thickness blanket is needed to ensure tritium self-sufficiency (a reduced thickness blanket is used at \( \Delta_{\text{min}} \), where the magnet is closest to the plasma). Designs with \( \langle R \rangle > 10 \text{ m} \) can accommodate uniform blanket/shield everywhere. Figure 15 shows the contours that bound the nonuniform blanket within each field period. The larger the machine is, the lower the nonuniform blanket coverage is, and the higher is the breeding.

V.A.3. Peaking Factors and Nonuniformities

While the average heat load on the first wall of ARIES-CS is similar to advanced tokamaks, the ratio of
peak-to-average heat flux (and neutron flux) is larger for compact stellarators. In general, for a given fusion power and first-wall area, the performance of a given blanket would be somewhat more limited in a compact stellarator than in a tokamak (e.g., lower thermal efficiency, shorter radiation lifetime).

V.A.4. Divertor Heat Loads

Similar to advanced tokamaks, a large portion of plasma power should be radiated in order to obtain a reasonable heat flux on the divertors. This mode of operation should be demonstrated in experiments. Field-line tracing analysis of the reference configuration resulted in a peak load of 18.4 MW/m² (and average heat load of 0.77 MW/m²), which exceeds the 10 MW/m² engineering limit. Because of lack resources and the 3-D nature of the problem, neither an aspect ratio scaling of the divertor heat load nor an optimized divertor design study has been carried out fully. Careful tailoring of divertor plates may lead to a reduction of the peaking factor on the divertor plates and an acceptable solution but will probably require careful alignment of the divertor plates.

We focused on operation with a high density \(4 \times 10^{20} \text{ m}^{-3}\) because of the \(n_e^{0.51}\) dependence in ISSS95 scaling and also the belief that high-density operation would help in reducing the heat flux on the divertor plates. Neoclassical theory predicts impurity accumulation in steady-state stellarator discharges with such high densities, and this is typically observed in experiments. However, both LHD and W7-AS have also operated in high-density regimes without impurity accumulation.\(^{31}\)

V.A.5. Alpha-Particle Confinement

While the alpha-particle loss does not impact the plasma power balance appreciably, it leads to serious issues for in-vessel components: (a) it can lead to localized heating and increase peak-to-average loads and (b) more importantly, energetic alpha particles can damage the material wall through blistering. Much more work is needed to reduce the “energetic” (i.e., >10 keV) alpha-particle loss fraction further.

V.B. R&D Needs

Major R&D issues specific to compact stellarators that are identified include

1. development and experimental demonstration of the compact stellarator configuration with reduced alpha-particle loss
2. understanding of beta limits in stellarators
3. demonstration of profile control in compact stellarators to ensure the achievement and control of the desired iota profile, including bootstrap current effects
4. development and experimental demonstration of “pumped” divertor geometries in compact stellarators with a highly radiative plasma including operation with a high plasma density without any impurity accumulation
5. development and experimental demonstration of plasma start-up scenarios and path to ignition with resonance-avoidance techniques.

On the engineering side, key R&D items specific to stellarators are

1. development of high-field superconducting magnets with irregular shape (e.g., in-organic insulators and/or high-temperature superconductors)
2. engineering accommodation of fast alpha-particle loss
3. development and demonstration of methods to fabricate, assemble, and maintain large superconducting stellarators free of resonance-inducing field errors
4. development of advanced manufacturing techniques.

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