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

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Abstract

The recent development of the compact stellarator concept delivered ARIES-CS – a compact stellarator with 7.75 m average major radius, approaching that of tokamaks. In stellarators, the most influentual engineering parameter that determines the machine size and cost is the minimum distance between the plasma boundary and mid-coil (Δ_{min}). Accommodating the breeding blanket and necessary shield within this distance to protect the superconducting magnet represents a challenging task. Selecting the nuclear and engineering parameters to produce an economic optimum, modeling the complex geometry for 3-D nuclear analysis to confirm the key engineering parameters, and minimizing the radwaste stream received considerable attention during the ARIES-CS design process. This paper provides a perspective on the successful integration of the nuclear activity, economics, and safety constraints into the final ARIES-CS configuration.

1. Introduction

Over the past 5-6 decades, the stellarator concept has been studied in the U.S., Europe, and Japan as an alternate to the mainline magnetic fusion tokamaks, offering steady state operation and eliminating the risk of plasma disruptions. The earlier 1980s studies suggested large-scale stellarator power plants with an average major radius exceeding 20 m. The most recent development of the compact stellarator concept has led to the construction of the National Compact Stellarator Experiment (NCSX) in the U.S. [1] and the 3-year power plant study of ARIES-CS [2] – a compact, low-aspect-ratio machine with three field periods and 7.75 m average major radius, approaching that of tokamaks. Stellarators in general are quite complex machines. For instance, the ARIES-CS first wall (FW) 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 degrees toroidally, the configuration changes from a bean-shape at 0^o to a D-shape at 60^o, then back to a bean-shape at 120^o, continually switching the surfaces from convex to concave over a toroidal length of ~17 m. This means the FW 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 Δ_{min} where the magnets move closer to the plasma, constraining the space between the plasma edge and mid-coil. Δ_{min} should accommodate the scrapeoff layer (SOL), FW, blanket, shield, vacuum vessel, assembly gaps, coils case, and half of the winding pack. Being the most influential parameter for the stellarator's size and cost, Δ_{min} optimization was crucial to the overall design.

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An innovative approach was developed to downsize the blanket at Δ_{min} and utilize a highly efficient tungsten carbide (WC)-based shield [3]. This approach placed a premium on the full blanket to supply the majority of the tritium needed for plasma operation.

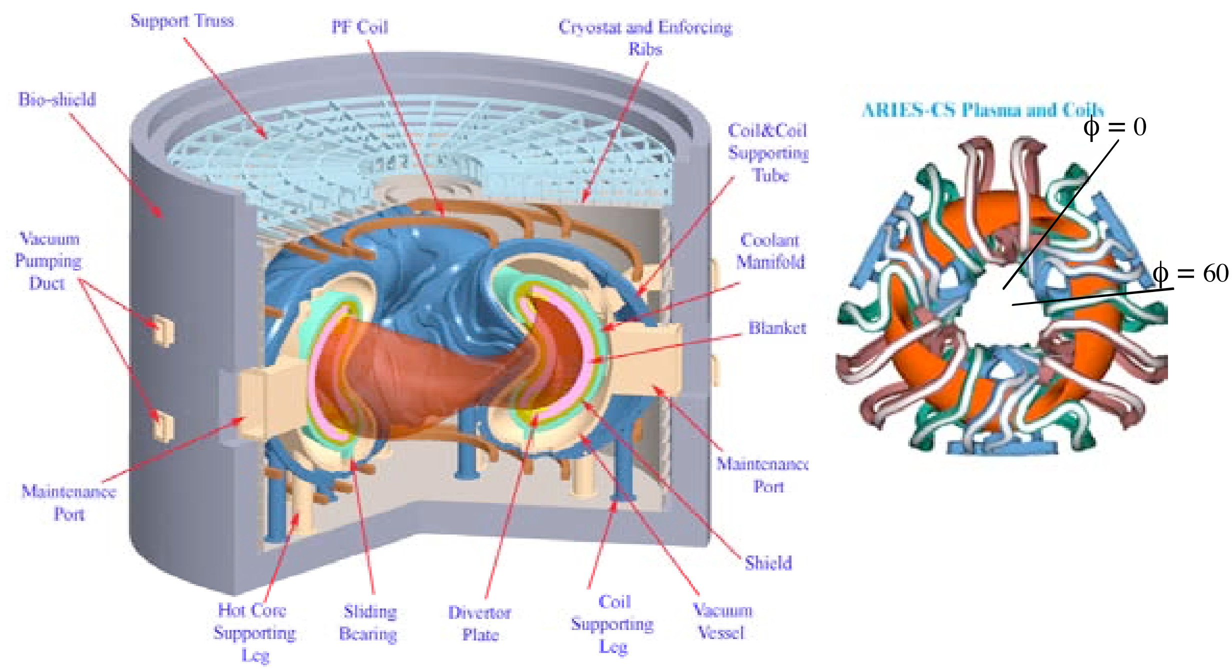

Figure 1. Isometric view of ARIES-CS three field period configuration with <R> = 7.75 m.

During the 3-y study period, several blanket/shield systems have been considered employing advanced ferritic steel (FS) structure (such as IEA MF82H) and SiC/SiC composites [4]. The list of candidates includes four liquid breeder-based systems and one solid breeder-based system:

- Self-cooled Flibe with beryllium multiplier and advanced oxide dispersion strengthening (ODS)-FS structure,
- Self-cooled LiPb with SiC/SiC composites,
- Dual-cooled LiPb (or Li) with He and FS structure, and
- He cooled Li_4SiO_4 with beryllium multiplier and FS structure.

Each blanket concept offers advantages and drawbacks. An integrated study with guidance from the nuclear analysis (neutronics, shielding, and activation) and blanket design identified the preferred concept (the dual cooled LiPb/He/FS with 42% η_{th}) 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 ^{o}C), offering high thermal conversion efficiency (56%) to enhance the economics. The reference ARIES-CS design employs dual coolants (LiPb and He) to recover the heat from the power producing components (FW, blanket, shield, manifolds, and divertor). One of the advantages of using dual coolants is to provide redundancy in case of accident and to ultimately protect the design from off-normal scenarios, such as loss of either coolant or flow events [5].
The following sections highlight the nuclear analysis and results for the FS-based LiPb/He/FS reference design. Throughout this study, the nuclear assessments were evaluated with the

MCNPX 3-D Monte Carlo code, DANTSYS discrete ordinates transport code, ALARA pulsed activation code, and IAEA FENDL-2 cross-section library.

2. Neutron wall loading distribution

Calculating the spatial variations of the NWL for the complex geometry of ARIES-CS requires advanced neutronics methods. The CAD-based MCNPX-CGM [6] tool provides just such a capability. Using the Fourier description of the last closed magnetic surface, a plasma surface for the <R>= 7.75 m reference design was created. One third of the toroidal extent of the first wall surface was segmented to allow for spatial resolution of the NWL calculation. In the toroidal direction, a spacing of 7.5° gave 16 segments, while 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 offset. The uncollided neutron current through each surface was tallied using MCNPX-CGM (F1 tally) and divided by the surface area for each segment, normalized to the 2355 MW total fusion power of the system, and reported at the midpoint of each segment. The NWL contour map for the 5 cm scrape-off layer is shown in Fig. 2 along with the detailed poloidal distributions of the NWL at various toroidal angles. The NWL averages at 2.6 MW/m^2. The 5.3 MW/m^2 peak is identified near the outboard midplane while the minimum occurs near the divertor region.

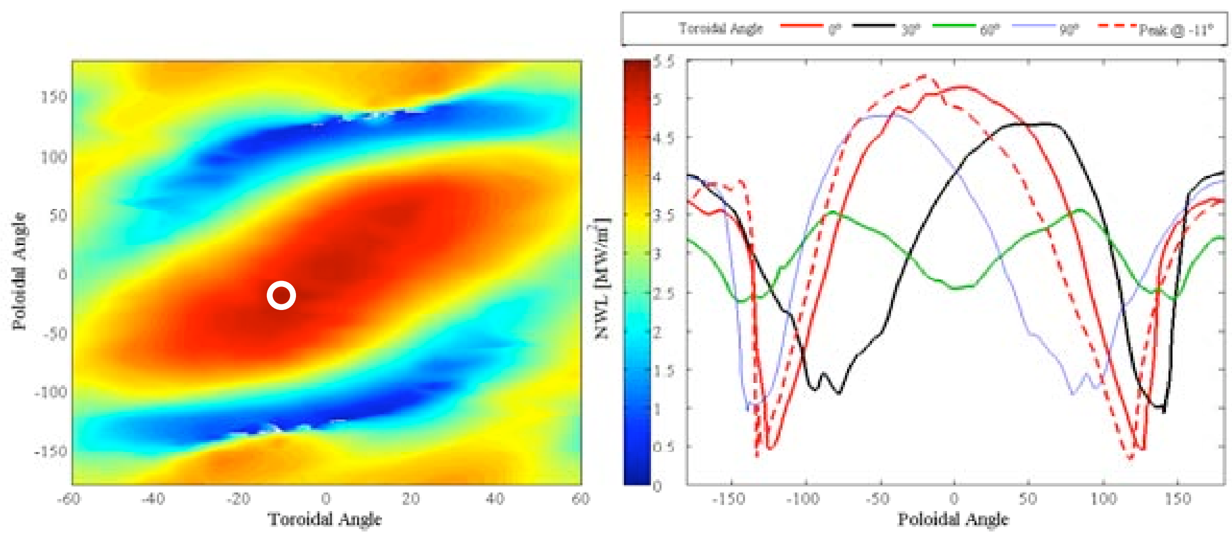

Figure 2. Contour map of NWL [in MW/m^2]. The approximate location of the peak is indicated with circles. The toroidal angle is measured from the beginning of the field period while the poloidal angle is measured from the outboard midplane at the level of the magnetic axis.

3. Radial build definition and key nuclear parameters

The compactness of ARIES-CS mandates that all components provide a shielding function. We focused our shielding activity on Δ_{min} where a superior shielding performance makes a notable difference to the machine size and cost. This feature is unique to stellarators. Thus, this topic has been investigated jointly by the engineers and physicists to examine the location, size, and FW coverage of Δ_{min} [3] and their impact on the machine parameters (major radius, field at coil, etc.), nuclear-related issues (tritium breeding, magnet protection, activation, and decay heat), and economics.

The blanket, along with the back wall, provides an important shielding function as it protects the shield for the entire plant life (40 full power years (FPY)). An additional shielding criterion relates to the reweldability of the manifolds and VV. The blanket and shield must keep the neutron-induced helium at the manifolds and VV below the reweldability limit (1 appm) at any time during plant operation. The VV, along with the blanket, shield, and manifolds, protects the superconducting magnets that operate at 4 K. A coolant with an efficient shielding performance (such as water) was employed for the VV – a non-producing power component. Because of the high reliability of the VV cooling system, water can flow naturally, carrying the decay heat out of the in-vessel components during accidents, enhancing the safety features of the design [7].

All materials were carefully chosen to enhance the shielding performance and minimize the long-term environmental impact [3]. 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 5.3 MW/m^2 maximum NWL and designed to provide adequate performance margins compared to the design requirements. The reference radial builds are shown schematically in Fig. 3 for two cross sections through the nominal, full blanket (designed for a peak NWL of 5.3 MW/m^2) and at Δ_{min} (designed for 3.3 MW/m^2 NWL – the maximum at the non-uniform blanket region). The main idea behind the compact, high-performance radial build of Fig. 3 is to use a reduced size blanket with more efficient shielding materials at local spots around Δ_{min} and deploy the nominal blanket elsewhere. The detailed composition of all components along with the alloying elements and impurities are given in Ref. 8. It should be mentioned that the 50 cm reduction resulting from the compact radial build at Δ_{min} saved 25-30% in the major radius and cost of electricity, which is significant.

For the reference configuration, Δ_{min} occurs at four locations per field period and the transition region between Δ_{min} and the full nominal blanket covers ~24% of the FW area. 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, replacing the B-FS filler. Costing roughly the same as the steel filler, the WC cost difference is not prohibitive for the transition region. Components with poor shielding performance, such as the manifolds, have been avoided at Δ_{min}. Considering the positive impact on the overall machine and economics, it pays to incorporate the compact radial build at Δ_{min}. Challenging engineering tasks that received considerable attention during the design process include the heat removal mechanism and the integration of the non-uniform blanket/shield with the surroundings [9].

Addressing the breeding issue, the blanket must breed sufficient tritium for plasma operation, meaning a calculated overall TBR $\geq$ 1.1. Due to the complexity of the geometry, the 3-D neutronics analysis was judged essential to predict the key nuclear parameters (overall TBR and energy multiplication (M_n)). The 3-D CAD-MCNPX model included the FW/blanket/back wall, shield, manifold, and divertors as shown in Fig. 4. A number of features were incorporated in the model to account for the design elements. A homogenized material definition was used throughout. To simplify the model, the vacuum vessel was not included since its impact on the TBR and M_n is negligible. Using this methodology, the results for TBR and M_n were determined for each major component and for the entire device [3]. The required TBR of 1.1 suggests a ^{6}Li enrichment of at least ~70%. The majority of the tritium breeding (> 77%) occurs in the uniform blanket region and approximately 2.5% occurs in the blanket region behind the divertors.

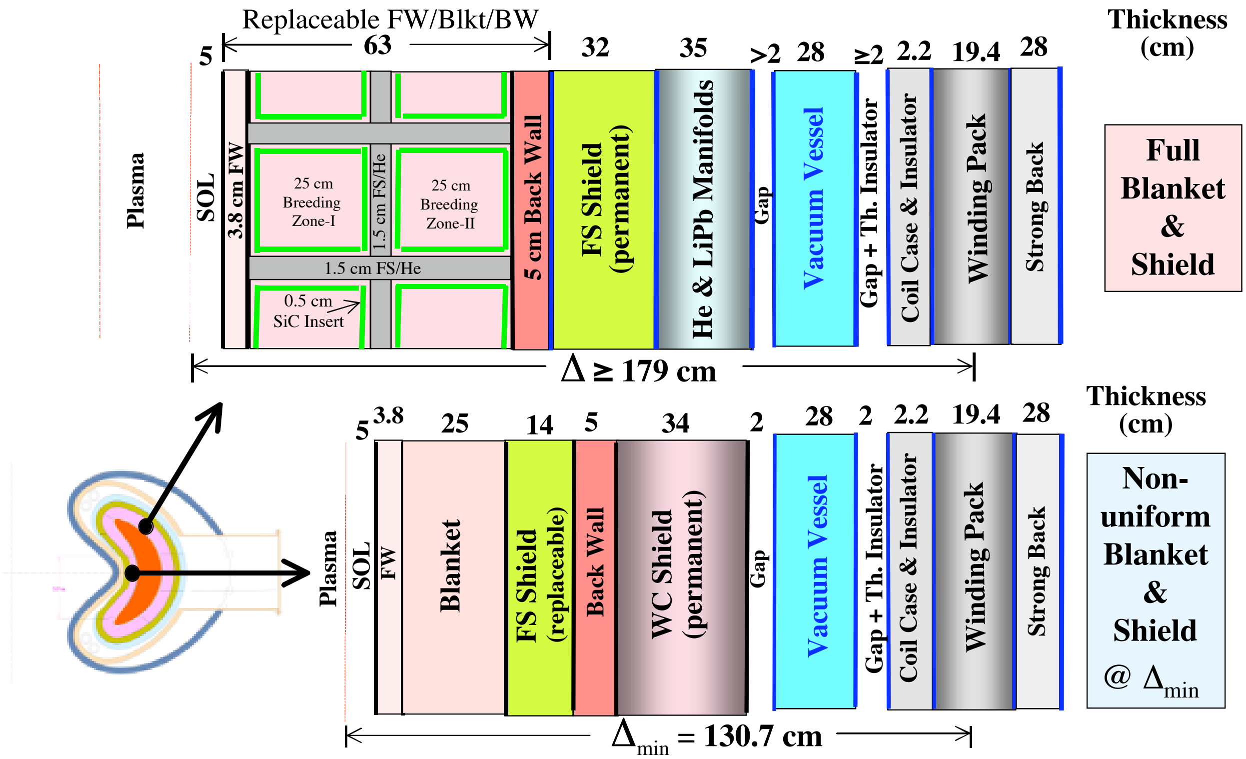

Figure 3. Radial builds for the reference LiPb/He/FS blanket.

The overall energy multiplication amounts to 1.16. The power deposited in the FW, blanket, shield, and divertor components will be recovered by the He and LiPb coolants as a high-grade heat. Most of the power (94%) goes to the FW, divertor, and blanket. The shield and manifolds carry 6% of the nuclear heating, which is significant and must be recovered to improve the power balance and enhance the economics. The small heat leakage to the VV (~ 3 MW) will be dumped as a low-grade heat.

The blanket modules are designed with replaceability as a design consideration. 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 for the ferritic steel structure, ending its service lifetime at 200 dpa. For a peak NWL of 5.3 MW/m^2, the FW lifetime is 3 FPY, requiring 13 replacements during the 40 FPY plant lifetime. 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 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.

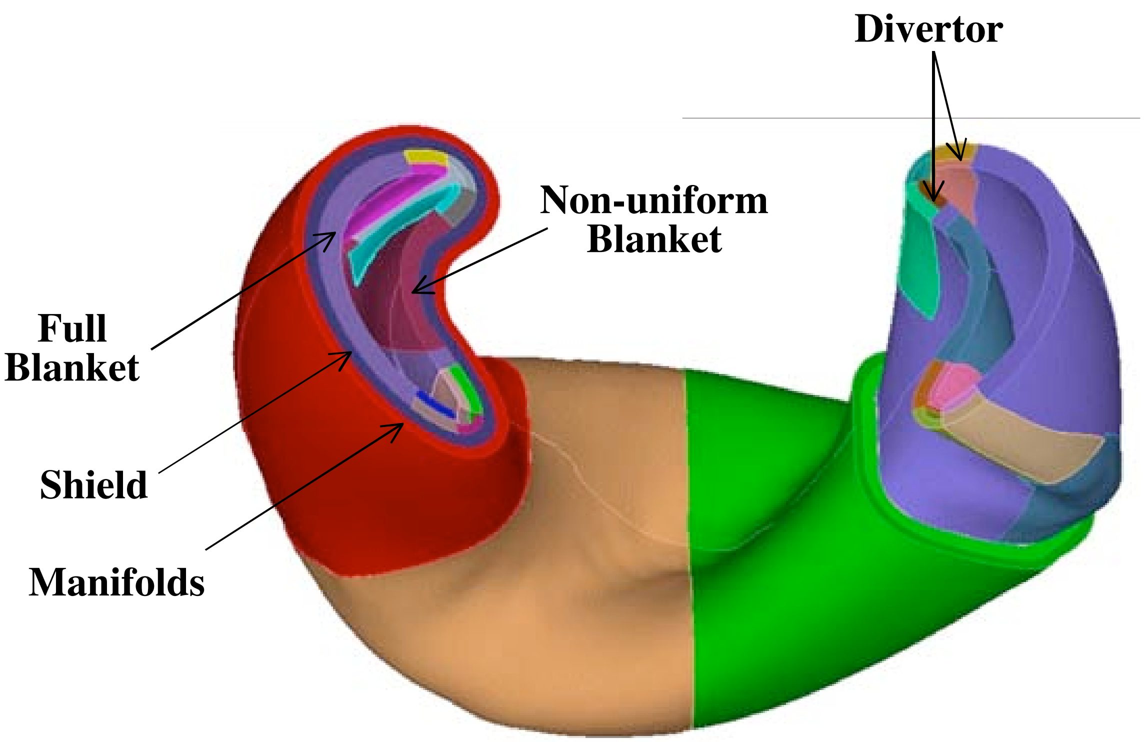

Fig. 4. 3-D neutronics model of ARIES-CS (<R>= 7.75 m).

4. Environmental issues and radwaste management

Since the inception of the ARIES project in the late 1980s, we focused our attention on the disposal of all active materials in near-surface geological repositories, as the main option for handling the replaceable and life-of-plant components, adopting the preferred fission waste management approach of the 1960s. It is becoming evident that future regulations for geological burial will be upgraded to assure tighter environmental controls. Furthermore, the political difficulty of constructing new repositories suggests reshaping all aspects of handling the continual stream of fusion active materials, replacing the disposal option with more environmentally attractive approaches such as recycling and clearance, if technically and economically feasible. These approaches became more technically feasible in recent years with the development of radiation-hardened remote handling (RH) tools and the introduction of the clearance category for slightly radioactive materials by national and international nuclear agencies [10]. We applied the three scenarios to ARIES-CS components:

– Like all ARIES power plants developed to date, ARIES-CS generates only low-level waste that requires near-surface, shallow-land burial according to the U.S. waste classification.
– The 2 m thick bioshield along with the 5 cm thick cryostat and some magnet constituents qualify for clearance, representing ~80% of the total active material volume. Because of the compactness of ARIES-CS, the clearance indices of all internal components (blanket, shield, manifolds, and vacuum vessel) exceed the

clearance limit by a wide margin even after an extended period of 100 y. This means the in-vessel components should be recycled or disposed of in repositories as LLW.

– All ARIES-CS components can potentially be recycled using conventional and advanced RH equipment that can handle 10 mSv/h (1000 times the hands-on dose limit) and high doses ≥ 10,000 Sv/h, respectively.

To enhance prospects for a successful radwaste management scheme, additional tasks should receive more attention in future studies. These include the key issues and concerns for disposal, recycling, and clearance, the capacity of existing repositories, the status of the recycling infrastructure, the development of advanced RH equipment, the need for new clearance guidelines for fusion-specific radioisotopes, the availability of a commercial market for cleared materials, and the acceptability of the nuclear industry to recyclable materials [10].

The ARIES project has been committed to the goal of radwaste minimization by design. The focus on compact devices with radwaste reduction mechanisms (such as advanced physics and technology and well-optimized components) contributed most significantly to the 3-fold reduction in ARIES-CS total radwaste volume compared to large-scale stellarators developed in the early 1980s [10]. In fact, recycling and clearance can be regarded as an 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 non-clearable materials can potentially be recycled indefinitely and therefore, will not be assigned for geological disposal.

5. Summary

A number of challenging engineering issues have been addressed during the ARIES-CS design process in order to deliver a credible design. We reviewed the nuclear-related elements that received considerable attention and provided a perspective on their successful integration into the final design. Numerous design issues stem from the compactness and complexity of the machine. Serious efforts have been made to adjust the radial standoff to accommodate the highly constrained areas, to develop a new CAD-MCNPX tool to model, for the first time ever, the complex stellarator geometry for 3-D nuclear analysis, and to establish a framework for handling the radioactive materials and minimizing the radwaste stream. With the successful completion of the 3-y study, ARIES-CS predicts a much brighter future for stellarators than had been anticipated 5-6 decades ago.

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