Nuclear challenges and progress in designing stellarator power plants

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Abstract

Over the past 2-3 decades, stellarator power plants have 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 stellarators with an average major radius exceeding 20 m. The most recent development of the compact stellarator concept delivered ARIES-CS – a compact stellarator with 7.75 m average major radius, approaching that of tokamaks. For stellarators, the most important engineering parameter that determines the machine size and cost is the minimum distance between the plasma boundary and mid-coil. Accommodating the breeding blanket and necessary shield within this distance to protect the ARIES-CS superconducting magnet represents a challenging task. Selecting the ARIES-CS nuclear and engineering parameters to produce an economic optimum, modeling the complex geometry for 3-D nuclear analysis to confirm the key parameters, and minimizing the radwaste stream received considerable attention during the design process. These engineering design elements combined with advanced physics helped enable the compact stellarator to be a viable concept. This paper provides a brief historical overview of the progress in designing stellarator power plants and a perspective on the successful integration of the nuclear activity into the final ARIES-CS configuration.

Keywords: Stellarator; nuclear analysis; fusion power plant; radwaste management

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1. Introduction

As an alternative to the mainline magnetic fusion tokamaks, the stellarator concept offers salient features including inherently steady-state devices with no need for large plasma current, no external current drive, no risk of plasma disruptions, low recirculating power due to the absence of current-drive requirements, and no instability and positional control systems. For these attractive features, stellarator power plants have been studied since the early 1980s in the U.S., Europe, and Japan to enhance the physics and engineering aspects and optimize the design parameters that are subject to numerous constraints. The earlier 1980s studies delivered large stellarators 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 7.75 m average major radius, approaching that of tokamaks.

During the 3-y design phase of the ARIES-CS project, several nuclear-related issues and concerns emerged as stellarator-specific challenges, calling for innovative design solutions to mitigate the pertinent engineering problems:

- The most important parameter that determines the stellarator size and cost is the minimum distance between the plasma boundary and mid-coil. Accommodating the breeding blanket and necessary shield to protect the superconducting magnet represented a challenging task. An innovative approach utilizing a non-uniform blanket combined with a highly efficient WC shield for this highly constrained area reduced the radial standoff (and machine size and cost) by 25-30%.
- The ARIES-CS first wall configuration deviates from the standard practice of uniform toroidal shape in order to achieve compactness. Modeling such a complex geometry for the 3-D nuclear analysis is a challenging engineering task. A novel approach based on coupling the CAD model with the MCNP Monte Carlo code was developed to model, for the first time ever, the complex stellarator geometry for nuclear assessments.
- As stellarators generate more radwaste than tokamaks, a smart management of the ARIES-CS active materials during operation and after plant decommissioning was essential for the environmental attractiveness of the machine. The geological disposal option could be replaced with

more attractive scenarios, such as recycling (within the nuclear industry) and clearance (or unconditional release to the commercial market).

Several additional nuclear-related tasks received considerable attention during the ARIES-CS design process. These include the radial build definition, the well-optimized in-vessel components that satisfy the top-level requirements, 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. The following sections provide a brief historical overview of the progress in designing stellarator power plants followed by a perspective to the successful integration of the nuclear activity into the final ARIES-CS design.

2. Historical overview

Although the stellarator concept has been around for almost three decades, very little in the way of conceptual design studies has been performed compared to tokamaks, of which many studies have taken place in the U.S. and abroad. During the decade of the 1980s and continuing to the present, six stellarator power plants have been developed: UWTOR-M [3], ASRA-6C [4], SPPS [5], and ARIES-CS [2] in the U.S., HSR [6] in Germany, and FFHR [7] in Japan. The timeline of these studies is given in Fig. 1. The six studies vary in scope and depth and encompass a broad range of configuration options. Even though stellarators promise salient physics features, such advantages could be offset by the more complex configurations, shown in Fig. 2, and challenging maintenance schemes.

Initiated in the early 1980s, the University of Wisconsin's (UW) first stellarator design (UWTOR-M [3]) has 18 modular twisted coils with only two different coil geometries arranged in a toroidal configuration. The blanket employs ferritic steel (FS) as the main structure and LiPb for cooling and tritium breeding. UWTOR-M was followed by the ASRA-6C study [4] performed in collaboration between UW and two German laboratories: FZK at Karlsruhe and IPP at Garching. All 30 coils of ASRA-6C and the internal components (FW, FS/LiPb blanket, and shield) have identical elliptical bores as shown in Fig. 2. Next came the Stellarator Power Plant Study (SPPS) [5] initiated in 1995 by the multi-institutional ARIES team to address key issues for stellarators based on the modular Helias-like Heliac approach. As Fig. 2 indicates, the baseline configuration has four field periods produced by 32 modular coils of four distinct types. Vanadium structure and lithium breeder are the reference materials for SPPS. On the international level, a Helias Stellarator Reactor (HSR) study was initiated in Germany in the late 1990s based on the Wendelstein 7-X experiment that is under construction in Greifswald, Germany. The most recent HSR4/18 design [6] has four field periods with 40 coils and LiPb/FS blanket. Alternatively, the stellarator configuration can be produced using continuous helical coils. An example of this approach is the Force Free Helical Reactor (FFHR) presently under study in Japan [7]. Vanadium structure, Flibe breeder, and beryllium multiplier are the materials of choice for FFHR. All designs developed to date employed liquid breeders (Flibe, LiPb, or Li) for breeding and cooling to cope with the complex geometry of stellarators. In summary, the very few stellarator studies developed thus far (compared to tokamaks) provide some fertile ground for innovation in ARIES-CS, as will be discussed shortly.

Earlier stellarator studies led to large power plants. The UWTOR-M design [3] had an average major radius (R_{av}) of 24 m in a six field period (FP) configuration. Moving toward smaller sizes, the ASRA-6C study [4] suggested 20 m R_{av} with four FPs. The most recent German HSR4/18 study [6] delivered 18 m R_{av} with four FPs. The ARIES SPPS study [5] was the first step toward a smaller size stellarator, proposing 14 m R_{av} with four FPs. Japan developed a series of FFHR designs [7], recently calling for 14 m R_{av} with ten FPs.

After two decades of stellarator power plant studies, it was evident that a new design that reflects the advancements in physics and improvements in technology was needed. To realize this vision, the ARIES team launched the ARIES-CS study [2] to provide perspective on the benefits of optimizing the physics and engineering characteristics of the so-called compact stellarator power plants. The primary goal of the study is to develop a more compact machine that retains the cost savings associated with the low recirculating power of stellarators, and benefits from the higher beta, smaller size, and higher power density, and hence lower cost of electricity (COE), than was possible in earlier studies. The benefit of the compact feature can be fully recognized when comparing ARIES-CS to all five stellarators developed to date (see Fig. 3). The most recent advanced physics and technology and innovative means of radial dimension control helped reduce the major radius by more than 3-fold, approaching that of advanced tokamaks.

3. ARIES-CS brief description

The FW and surrounding in-vessel components conform to the plasma, as shown in Fig. 4, deviating 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. A novel approach based on coupling the CAD model with the 3-D neutronics code was developed to model, for the first time ever, the complex stellarator geometry for nuclear assessments. 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, coil case, and half of the winding pack. The penalty associated with increasing Δ_{min} by 10 cm is ~60 cm in the major radius and ~1 mill/kWh in the cost of electricity. Being the most influential parameter for the stellarator's size and cost, its optimization was crucial to the overall design. An innovative approach was developed to downsize the blanket at Δ_{min} and utilize a highly efficient WC-based shield. This approach placed a premium on the full blanket to supply the majority of the tritium needed for plasma operation.

To guide the design process, a set of nuclear-related requirements, summarized in Table 1, was established at the outset of the ARIES-CS study. For instance, a tritium breeding ratio (TBR) of 1.1 assures tritium self-sufficiency. 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. The nuclear heat leakage from the power producing components 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 (VV), the helium production level should not exceed 1 appm 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.

4. Radial build definition and key nuclear parameters

The nuclear assessment began by generating the neutron wall loading (NWL) profile in the toroidal and poloidal directions, using the newly developed CAD-MCNP coupling approach [8]. For a fusion power of 2355 MW, the NWL averages at 2.6 MW/m^2 and peaks at 5.3 MW/m^2 near the outboard mid-plane, close to $\Phi=0^o$ cross section. 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) [9,10]. One of the advantages of using dual coolants is to provide redundancy in case of accident and to ultimately protect the design from offnormal scenarios, such as loss of either coolant or flow events. While the ferritic steel-based blanket is based on the same concept developed earlier by the ARIES team for the ARIES-ST spherical tokamak [11], and later considered as an ITER blanket testing module by many ITER parties, the unique blanket safety features were thoroughly examined and analyzed to provide assurance of their effectiveness [12]. A coolant with more 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 [13].

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. No economic and design enhancements are gained with a high-performance, compact shield at any place, but at Δ_{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 FW coverage of Δ_{min} and their impact on the machine parameters (major radius, field at coil, etc.), nuclear parameters (TBR, 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. All materials were carefully chosen to enhance the shielding performance and minimize the long-term environmental impact [10]. 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 of Table 1. The reference radial builds are shown schematically in Fig. 5 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 detailed composition of all components along with the alloying elements and impurities are given in Ref. 14. 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.

The main idea behind the compact, high-performance radial build of Fig. 5 is to use a reduced size blanket with more efficient shielding materials at local spots around Δ_{min} and deploy the nominal blanket elsewhere. 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 such limited space. Components with poor shield 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 an overall TBR ≥ 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-MCNP model included the FW/blanket/back wall, shield, manifold, and divertors as shown in Fig. 6. A number of features were incorporated in the model to account for the design elements. A homogenized material definition was used throughout. To accommodate their impact on TBR, the three electron cyclotron heating ducts were also included in the model. 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. The target 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.

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 200 dpa limit for the ferritic steel structure, ending its service lifetime. 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.

5. Radiological characteristics of active materials

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. It is becoming evident that future regulations for geological burial will be upgraded to assure tighter environmental controls. Along with the political difficulty of constructing new repositories, the current reality 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. We applied the three scenarios to ARIES-CS components:

– Disposal: To classify the waste, we evaluated the waste disposal rating (WDR) for a fully compacted waste using the most conservative waste disposal limits. Like all ARIES power plants developed to date, ARIES-CS generates only low-level waste (WDR < 1) that requires near-surface, shallow-land burial according to the U.S. waste classification. The WDRs of the VV and external components are very low (< 0.1), to the extent that these components could qualify as Class A LLW – the least hazardous type of waste. Excluding the cryostat and bioshield, ~ 70% of the waste (blanket, shield, and manifolds) is Class C LLW. The remaining ~30% (VV and magnet) would fall under the Class A LLW category.

- Clearance: By definition, it is the unconditional release of materials from radiologically controlled areas to the commercial market at the end of an interim storage period. After plant decommissioning, individual materials could be stored for a specific period (< 100 years), segregated, then released to the commercial market if the clearance index (CI) falls below one. Because of the compactness of ARIES-CS, the CIs 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 [12]. This means the in-vessel components should be recycled or disposed of in repositories as LLW. Of interest is the 2 m thick external concrete building (bioshield) that surrounds the torus. It represents the largest single component of the decommissioned waste. Fortunately, the bioshield along with the 5 cm thick cryostat and some magnet constituents qualify for clearance, representing ~80% of the total active material volume [12].
- Recycling: We applied the recycling approach to the non-clearable in-vessel components (blanket, shield, divertor, and vacuum vessel). All components can potentially be recycled [12] using conventional and advanced remote handling (RH) equipment that can handle 10 mSv/h (1000 times the hands-on dose limit) and high doses ≥ 3000 Sv/h, respectively. Recycling is an essential step toward achieving the goal of radwaste minimization. It should be pursued despite the lack of details on how to implement it now. We expect significant advancements in recycling technologies some 50-100 y in the future based on current accomplishments and near-term developments in the rapidly growing area of fission fuel reprocessing.

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.

Over the past three decades, the radwaste volume aspect of fusion in general continued to be a concern. As such, the ARIES project has been committed to the achievable 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 UWTOR-M [3]. Figure 7 demonstrates this impressive trend and illustrates the 30% reduction in ARIES-CS volume achieved even during the 3 year timeframe 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 non-clearable materials can potentially be recycled indefinitely and therefore, will not be assigned for geological disposal.

6. Conclusions

We reviewed the nuclear-related elements that received considerable attention during the ARIES-CS design process and provided a perspective on their successful integration into the final design. 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 highly constrained areas, developing a new CAD/MCNP tool to model, for the first time ever, such a complex geometry for 3-D nuclear analyses, and establishing 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 2-3 decades ago. ARIES-CS has benefited substantially from its compactness, using advanced physics and engineering performance.

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Figure Captions:

Figure 1. History timeline for stellarator power plants.

Figure 2. Isometric view of the six stellarator power plants developed in the U.S., Europe, and Japan.

Figure 3. Evolution of stellarator size. Advanced tokamak and spherical torus included for comparison.

Figure 4. Isometric view of ARIES-CS.

Figure 5. Radial builds for full and reduced blanket/shield.

Figure 6. Three-dimensional neutronics model of ARIES-CS.

Figure 7. Comparison of in-vessel components and magnet volumes for U.S. stellarator power plants (actual volumes, no compactness, no replacements).

Figure 1

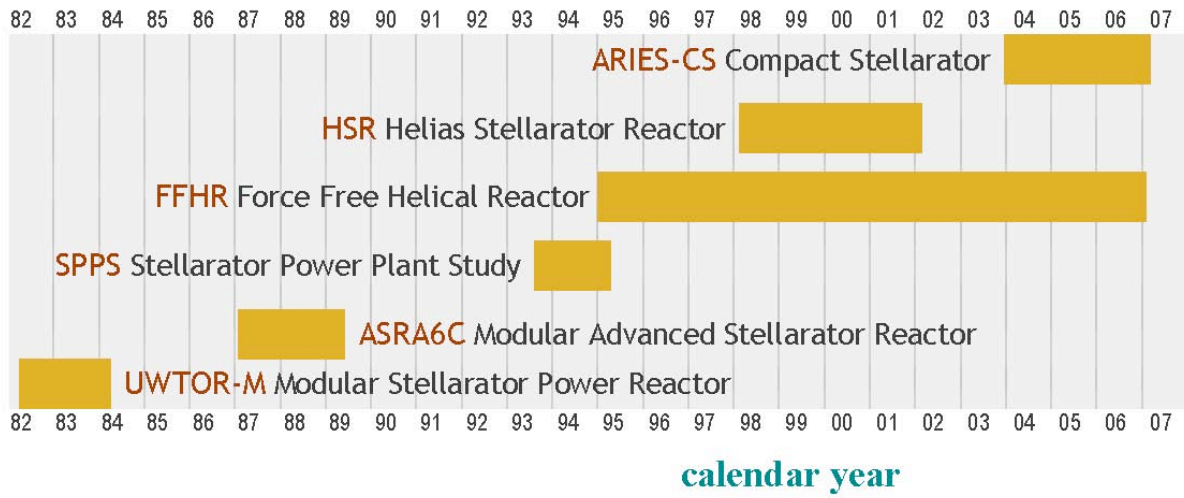

82 83 84 85 86 87 88 89 90 91 92 93 94 95 96 97 98 99 00 01 02 03 04 05 06 07
ARIES-CS Compact Stellarator
HSR Helias Stellarator Reactor
FFHR Force Free Helical Reactor
SPPS Stellarator Power Plant Study
ASRA6C Modular Advanced Stellarator Reactor
UWTOR-M Modular Stellarator Power Reactor
82 83 84 85 86 87 88 89 90 91 92 93 94 95 96 97 98 99 00 01 02 03 04 05 06 07
calendar year

Figure 2

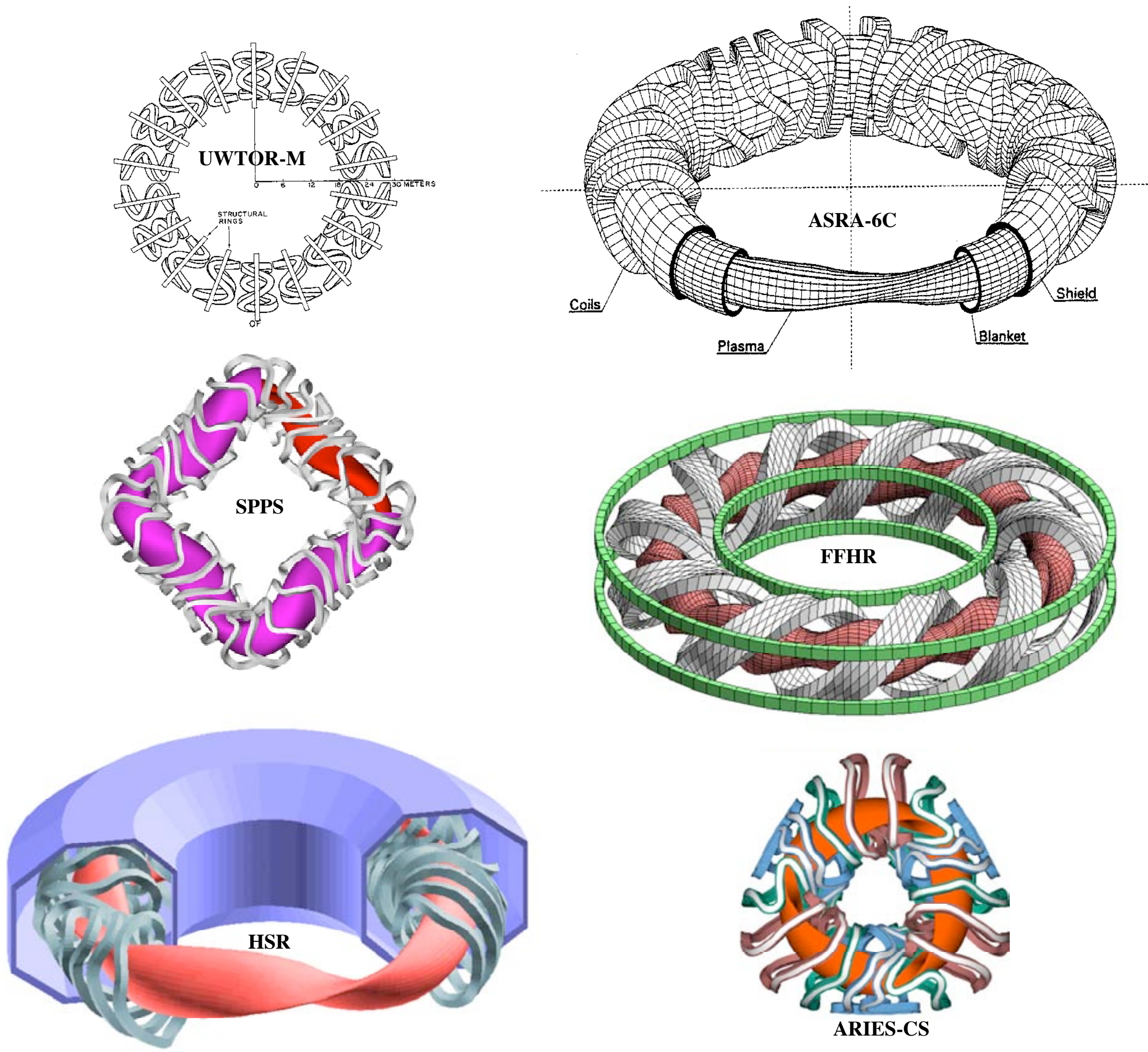
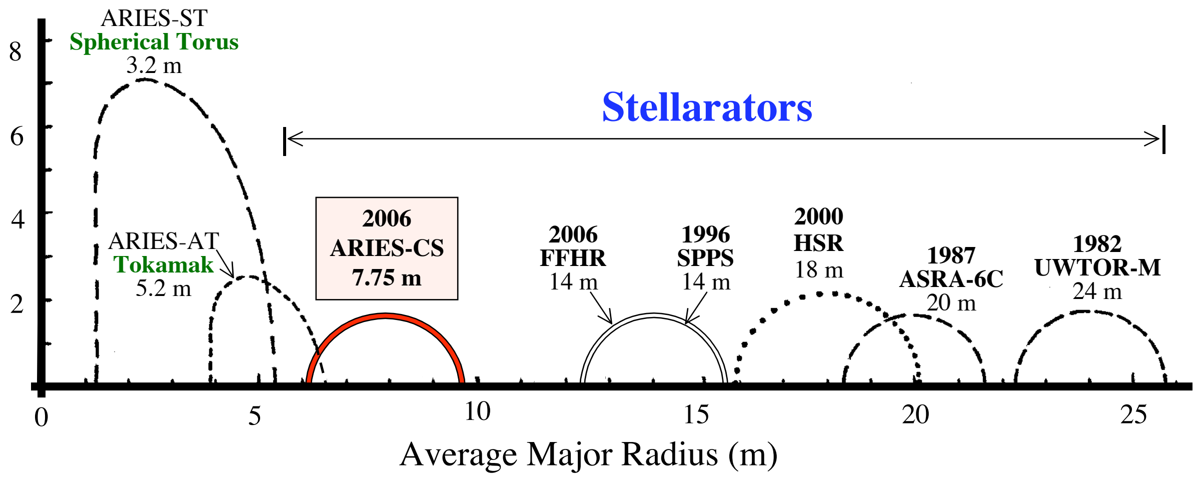

UWTOR-M
STRUCTURAL RINGS
ASRA-6C
Coils
Plasma
Blanket
Shield
SPPS
FFHR
HSR
ARIES-CS

Figure 3



ARIES-ST
Spherical Torus
3.2 m
Stellarators
ARIES-AT
Tokamak
5.2 m
2006
ARIES-CS
7.75 m
2006
FFHR
14 m
1996
SPPS
14 m
2000
HSR
18 m
1987
ASRA-6C
20 m
1982
UWTOR-M
24 m
8
6
4
2
0
5
10
15
20
25
Average Major Radius (m)

Figure 4

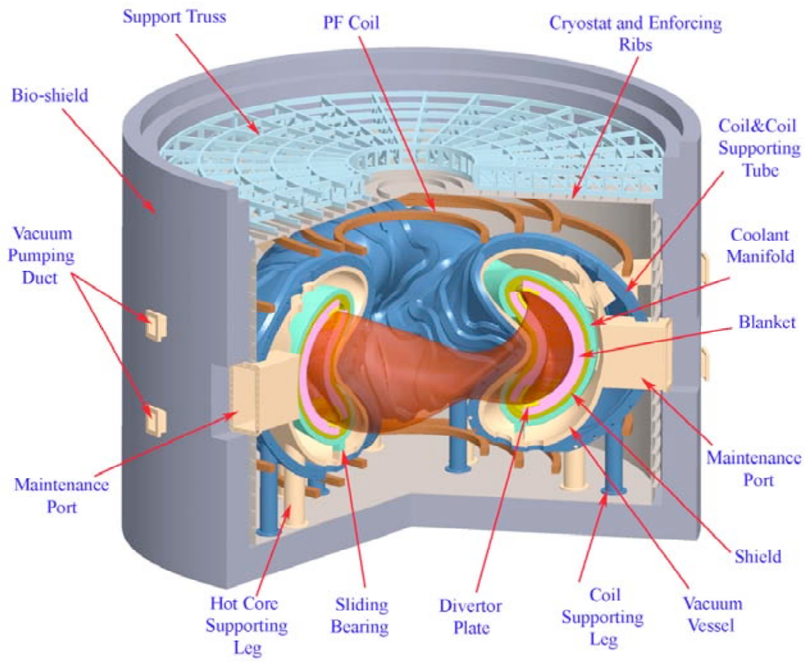
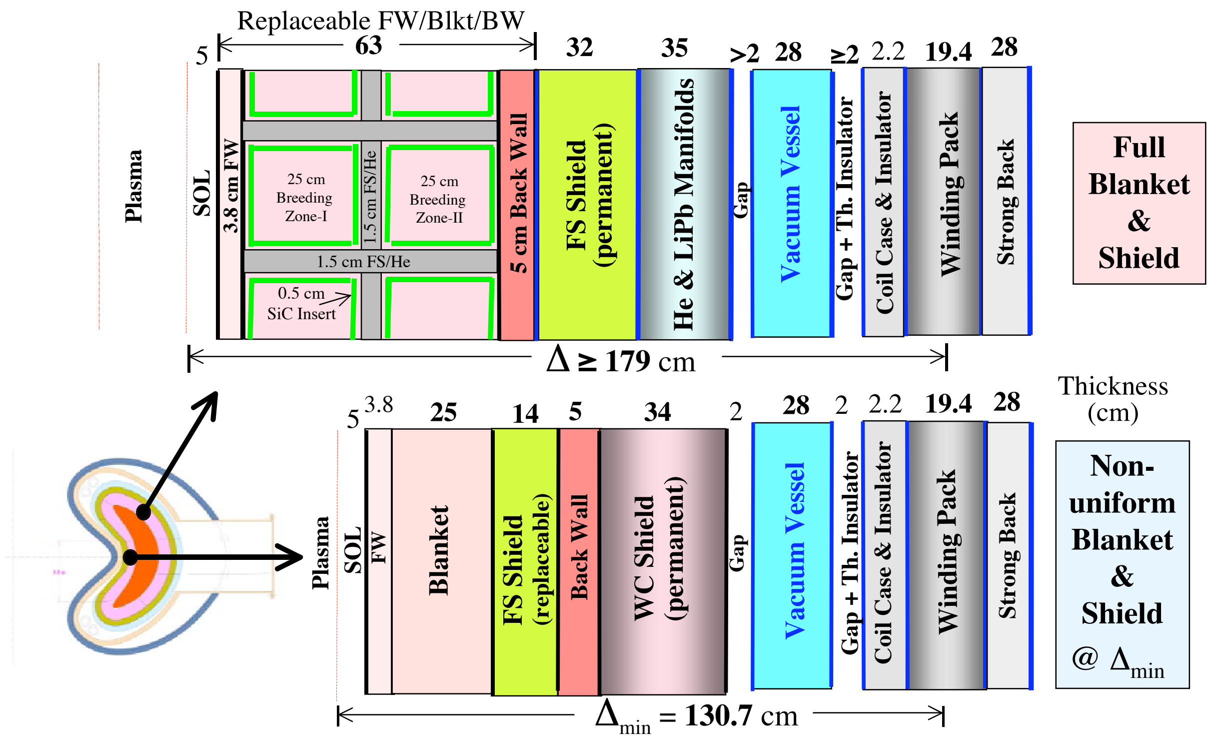

Support Truss
PF Coil
Cryostat and Enforcing Ribs
Bio-shield
Coil&Coil Supporting Tube
Vacuum Pumping Duct
Coolant Manifold
Blanket
Maintenance Port
Maintenance Port
Shield
Hot Core Supporting Leg
Sliding Bearing
Divertor Plate
Coil Supporting Leg
Vacuum Vessel

Figure 5



Replaceable FW/Blkt/BW
5
63
32
35
>2
28
≥2
2.2
19.4
28
Plasma
SOL
3.8 cm FW
25 cm Breeding Zone-I
1.5 cm FS/He
25 cm Breeding Zone-II
1.5 cm FS/He
0.5 cm SiC Insert
5 cm Back Wall
FS Shield (permanent)
He & LiPb Manifolds
Gap
Vacuum Vessel
Gap + Th. Insulator
Coil Case & Insulator
Winding Pack
Strong Back
Full Blanket & Shield
Δ ≥ 179 cm
Thickness (cm)
5
3.8
25
14
5
34
2
28
2
2.2
19.4
28
Plasma
SOL
FW
Blanket
FS Shield (replaceable)
Back Wall
WC Shield (permanent)
Gap
Vacuum Vessel
Gap + Th. Insulator
Coil Case & Insulator
Winding Pack
Strong Back
Non-uniform Blanket & Shield @ Δmin
Δmin = 130.7 cm

Figure 6

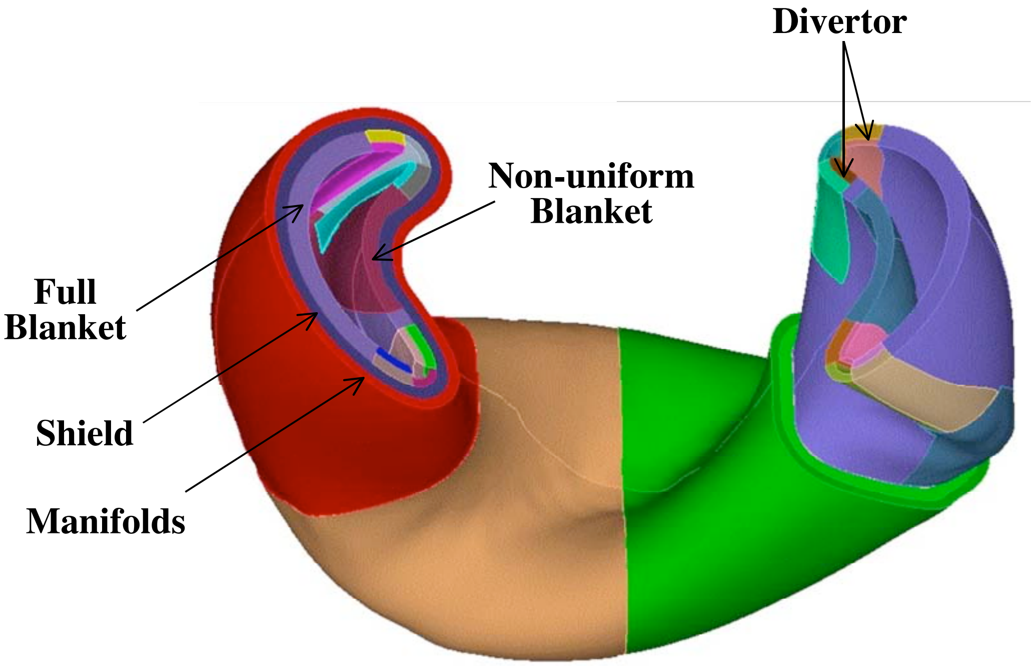

Divertor
Non-uniform Blanket
Full Blanket
Shield
Manifolds

Figure 7

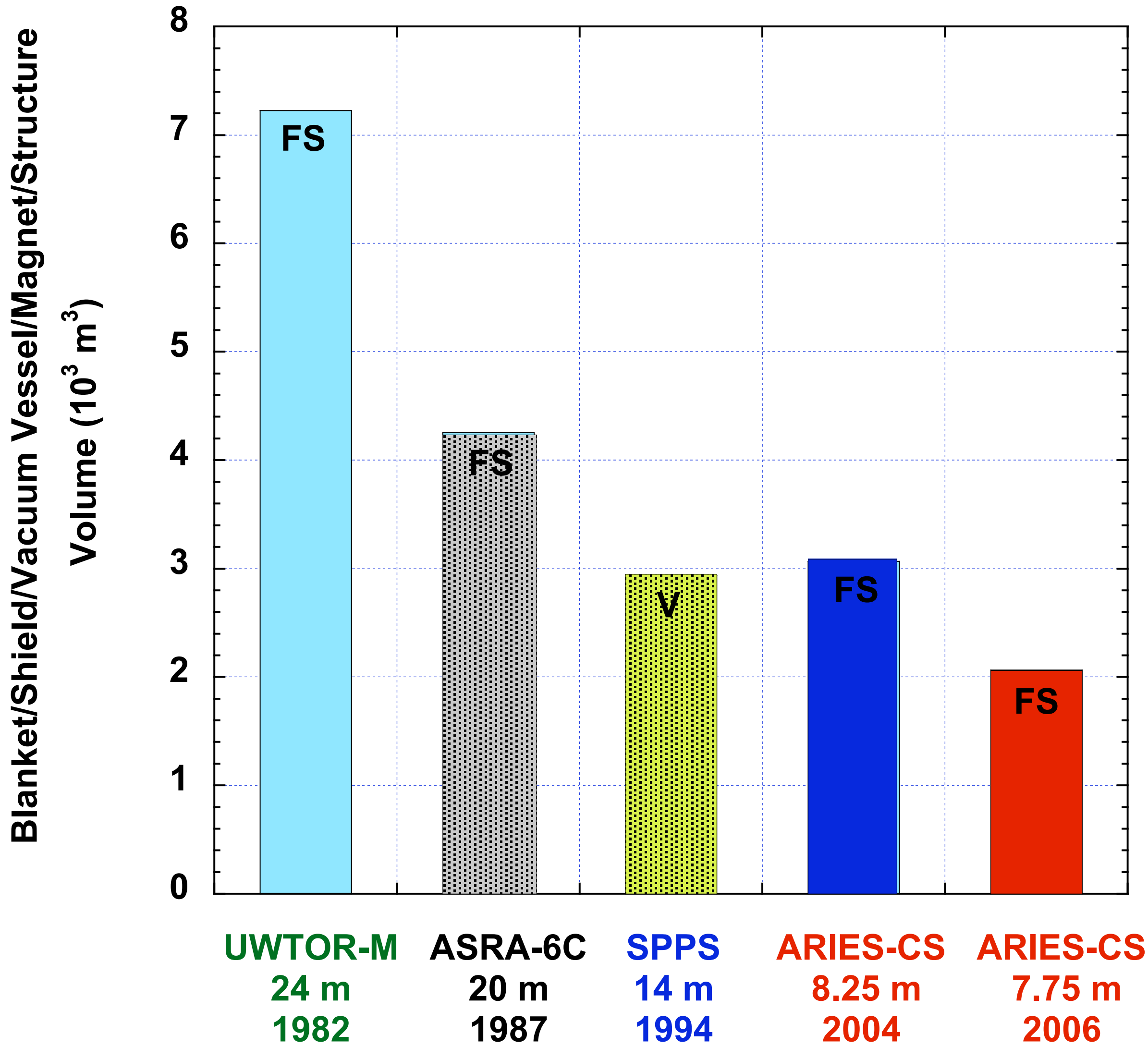

Blanket/Shield/Vacuum Vessel/Magnet/Structure
Volume (10^3 m^3)
0
1
2
3
4
5
6
7
8
FS
FS
V
FS
FS
UWTOR-M
24 m
1982
ASRA-6C
20 m
1987
SPPS
14 m
1994
ARIES-CS
8.25 m
2004
ARIES-CS
7.75 m
2006

Table 1
ARIES-CS design requirements and radiation limits

Overall TBR (for T self-sufficiency)	1.1	
Damage to structure	200	dpa
Helium production @ manifolds and VV (for reweldability of FS)	1	He appm
Nuclear heat leakage	< 1%	
S/C magnet (@ 4 K):		
Peak fast n fluence to Nb_3Sn (E_n > 0.1 MeV)	10^{19}	n/cm^2
Peak nuclear heating	2	mW/cm^3
Peak dpa to Cu stabilizer	$6x10^{-3}$	dpa
Peak dose to electric insulator	10^{11}	rads
Plant lifetime	40	FPY
Availability	85%	
Operational dose to workers and public	< 2.5	mrem/h
LLW level	Class A or C	
Radwaste minimization	Recycle and/or clear	

*Acronyms: TBR for tritium breeding ratio, dpa for displacement per atom, appm for atom part per million, LLW for low-level waste, FPY for full power year.