Engineering Overview of ARIES-RS Tokamak Power Plant *

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The ARIES-RS tokamak power plant is based on the reversed-shear plasma mode of operation and uses vanadium alloy as the high-temperature structural material and Li as the coolant. The design process emphasized the attainment of the top-level mission requirements developed in the early part of the study in a collaborative effort between the ARIES Team and representatives from U.S. electric utilities and industry. Major efforts were devoted to develop a credible configuration which allows rapid removal of full sectors followed by disassembly in the hot cells during plant operation. This was adopted as the only practical means to meet availability goals. Use of an electrically insulating coating for the self-cooled blanket and divertor provides a wide design window and simplified design. Optimization of the shield, which is one of the larger cost items, significantly reduced the power core cost by using ferritic steel where the power density and radiation levels are low. An additional savings is made by radial segmentation of the blanket, such that large segments can be reused. Design features and critical issues for the first wall, blanket, divertor, heating and current drive, and magnet systems are summarized.

1. INTRODUCTION

ARIES-RS is a U.S. fusion power plant design study which has examined the ability of an advanced tokamak based power plant to compete with future energy sources and play a significant role in the future energy market. "Top-level" system-wide requirements were established through extensive consultation between the ARIES team and U.S. electric utilities and industry [1]. Various physics and engineering concepts were assessed to determine their ability to meet these requirements. Based on the assessment phase of the project, a more detailed design activity was initiated to examine the reversed-shear mode of plasma operation, coupled to a fusion power core that uses high-performance lithium-cooled vanadium components.

The overall tokamak configuration is described here briefly, together with each of the major power core components: the first-wall, blanket and shield; divertor; heating and current drive; and magnet systems. A brief summary of the design features and operating characteristics is given, and the principal conclusions summarized.

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** Institutions participating in the ARIES Team, in addition to UC San Diego, include Argonne National Laboratory, General Atomics, Los Alamos National Laboratory, Massachusetts Institute of Technology, McDonnell Douglas Aerospace Co., Princeton Plasma Physics Laboratory, Raytheon Engineers and Constructors, Rensselaer Polytechnic Institute, and the University of Wisconsin-Madison.

2. DESIGN SUMMARY

Table 1 summarizes the key parameters for the final design point of ARIES-RS, obtained in a self-consistent manner with the best available physics, engineering, and systems models. The attractiveness of the reversed-shear mode of operation comes from a combination of high plasma β together with good matching of the bootstrap current density profile to that needed for plasma equilibrium and stability, resulting in moderate current-drive power for a steady-state system. The power density in the design is moderately high, but remains within design limits for the Li/V first wall and blanket.

Table 1. Key System Parameters

Major radius	5.52	m
Minor radius	1.38	m
Elongation (x-point)	1.9	
Plasma current	11.3	MA
Bootstrap current fraction	0.88	
Toroidal field on axis	8.0	T
Toroidal β	5.0	%
Average neutron wall load	4.0	MW/m^2
Fusion power	2167	MW
Gross electric power	1204	MW
Net electric power	1000	MW
Gross cycle efficiency	46	%

Figure 1 shows a cutaway view of the fusion power core. The plant availability goal was a primary influence on the overall configuration.

The vacuum vessel is a double-walled steel structure with reinforcing ribs that closely surrounds the shield. A "bell jar" cryostat was chosen over the close-fitting design adopted in earlier ARIES studies to simplify the configuration. Horizontal maintenance of full sectors is performed through large ports using transporter casks [2]. Port doors on the back of the shield and at the cryostat prevent the spread of radioactive contamination to the confinement building. All coolant connections are made in the evacuated port area where the radiation field is low. This design allows rapid disconnections of the piping, which could be either mechanically sealed or cut and rewelded.

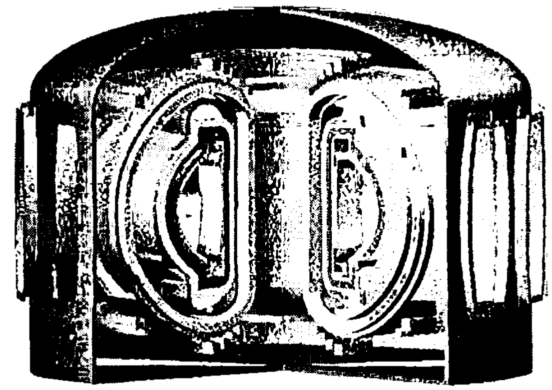

Figure 1. ARIES-RS fusion power core

One of the distinctive features of this design is the integration of the sectors. The first wall, blanket, parts of the shield, divertor and stability shells form an integral unit within each sector. These "replacement units" are shown in Figure 2. The integrated sector construction eliminates in-vessel maintenance operations and provides a very sturdy continuous structure able to withstand large loads. Gravity loads are supported at the bottom through the vacuum vessel. Sectors are disassembled and reuseable parts maintained in hot cells after the plant returns to operation. No rewelding is needed for elements located within the radiation environment.

The most severe penalty of single-piece sectors is the increased size of both the TF and PF coil systems, needed to allow adequate space for sector removal. With an optimized design, the cost of the increased size of the TF and PF coils is 2~3 mill/kWeh, which is substantially smaller than the cost savings due to the increased availability.

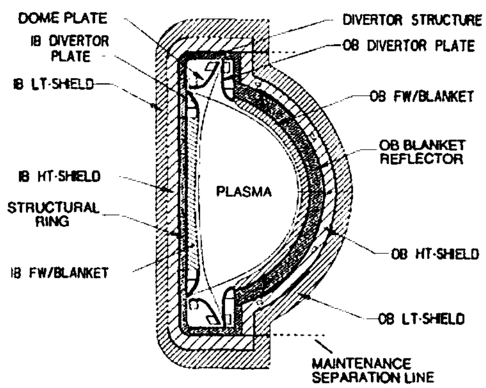

Figure 2. Sector elevation view

3. FIRST WALL, BLANKET & SHIELD

The first wall and breeding blanket use a simple box-like structure, with lithium coolant flowing in simple poloidal paths. The outboard cross section is shown in Figure 3. An effective insulating coating is necessary to reduce the MHD pressure drop to an acceptable level. With the reduced MHD pressure drop, improved performance and a relatively simple design are obtained. The choice of a CaO coating, maintained by adding 0.5% Ca in the flowing Li is based on recent experimental results [3]. The development of insulating coatings is at a very early stage and much more R&D is required. The improvements and design flexibility that coatings provide for self-cooled liquid metal blankets make this a very high leverage item.

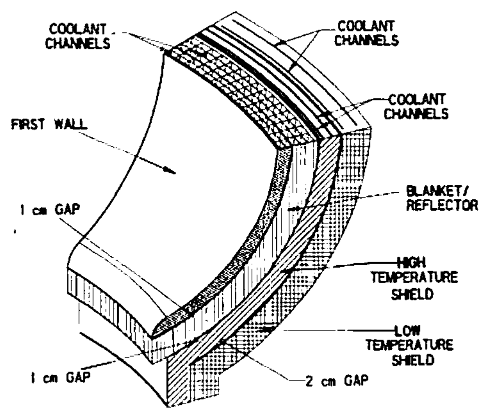

Figure 3. Outboard blanket and shield

An important feature of the blanket and shield is the radial segmentation into four zones: two blanket zones and two shield zones. The blanket is divided into two regions to maximize the lifetime of the structures, reduce the replacement cost, and minimize the waste stream. Scheduled replacement occurs after 2.5 full power years (FPY), or when the V-alloy reaches 200 dpa. At that time, the front portion is disposed, but the rear portion can be used until it reaches its own useful life, at about 7.5 FPY. The rear portion also serves as the structural ring, which provides poloidal continuity to the sectors and attachment points for the inner blanket segments. Location of the coolant connections outside the vacuum vessel allows for easy disassembly of the segments. Radial segmentation creates safety concerns, since radial heat transport pathways are critical in loss-of-coolant scenarios. Design solutions have been proposed; however blanket response to coolant loss remains an important concern.

Table 2 summarizes the heat loads and peak temperatures in the blanket and divertor. Multiple flow passes in the blanket provide the capability for removing at least 0.5 MW/m^2 of surface heat flux, which may be necessary with a highly-radiative divertor mode of operation. The full coolant flow is passed first through the front zone, where the surface heat flux creates large temperature gradients, and then through the back zones where the bulk temperature can be raised by volumetric heating without exceeding any structure temperature limits. Segmentation of the shield into a hot and cold zone allows partial utilization of the heat deposited, and also provides further capability for superheating the coolant away from the high heat flux region.

Table 2. Power flows and peak temperatures

Multiplied neutron heating	2092 MW
Total transport power	431 MW
Bremstrallung power	56 MW
Core line radiation	25 MW
Power reradiated in divertor	341 MW
First wall surface heating	165 MW
Divertor total surface heating	348 MW
Divertor particle power	88 MW
Blanket bulk outlet temperature	610 °C
Divertor bulk outlet temperature	610 °C
Peak V temperature in first wall	~700 °C
Peak V temperature in divertor	681 °C

An advanced Rankine cycle conservatively offers 46% gross thermal conversion efficiency. High thermal efficiency is desirable to partially offset the high capital cost of fusion. A double-walled IHX with a Na secondary loop is used to isolate the activated Li primary coolant from the steam side. The IHX is also the location where the transition from V to SS is made. The piping which connects the blanket to the IHX uses a double-walled structure with a thin V liner to minimize the added cost of vanadium.

4. DIVERTOR

The divertor region of the sector, highlighted in Figure 4, consists of two principal parts: the target plates and the structures. The structures fulfill several essential functions: (1) mechanical attachment of the plates through adjustable screw-bolts which provide for module alignment and offer lateral flexibility for thermal expansion together with strong support against EM events (*e.g.*, disruptions), (2) shielding for the magnets, (3) coolant routing paths for the plates as well as the inboard blanket and replaceable shield, (4) "superheat" of the coolant to optimize surface heat removal while maintaining high outlet temperature, and (5) a contribution to the breeding ratio, since the coolant is Li.

The target plates include three pieces: inboard, outboard, and "dome" plates. The plasma flows through the scrape-off layer and enters the divertor, where enhanced line radiation from injected neon impurity allows much of the power to be distributed along the plates and also partially redirected out to the first wall. Most of the unradiated particle energy strikes the outboard plates, but the peak heat flux has been maintained below 6 MW/m^2. The strike points are located close to the coolant inlet in order to maintain the vanadium structures below 700°C.

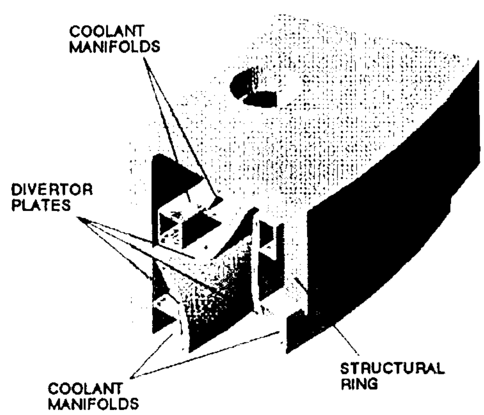

Figure 4. Divertor geometry

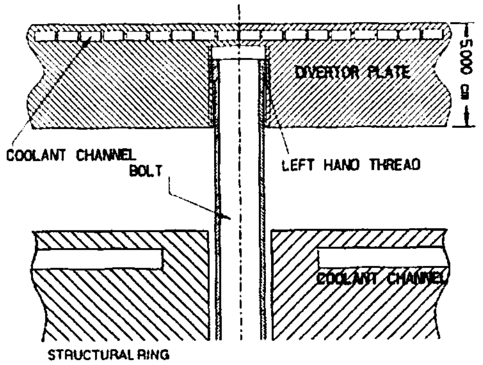

Figure 5. Divertor plate and attachment

The target plates are shown in Figure 5. A 2-mm thick catellated W coating is applied to the coolant channel front surface, which is only 1-mm thick V to satisfy temperature limits. Thermal stresses are reduced by using a relatively thick solid back on the target plates. The plates are connected to the rear zone via strong adjustable screw-type attachments. These attachments can be designed to react the full force of disruptions and also accommodate thermal expansion. They also permit precise alignment to adjacent surfaces and removal of individual plates in the hot cells.

Vacuum pumping ducts are placed behind the dome near the strike points for efficient exhaust. Radial channels then direct the gas to a single set of cryopumps at the bottom of the machine. Top-to-bottom conductance connecting both divertors is achieved by using the inter-sector vessel volume underneath TF coils.

5. HEATING AND CURRENT DRIVE

With 80%-90% bootstrap current, the current drive requirements for the reversed shear plasma are modest. However, three rf systems are needed for adequate current profile control: ICRF fast wave at the plasma axis, lower hybrid (LH) near the plasma edge, and high frequency fast wave (HFFW) near the q_{min} surface. All three of these systems are integrated into a single special sector, occupying only 0.5% of the total first wall surface area.

Folded waveguides (see Figure 6) were chosen for the ICRF system as the most compact, robust mechanical structures. Thin copper coatings on all waveguides are used to minimize the dissipated losses.

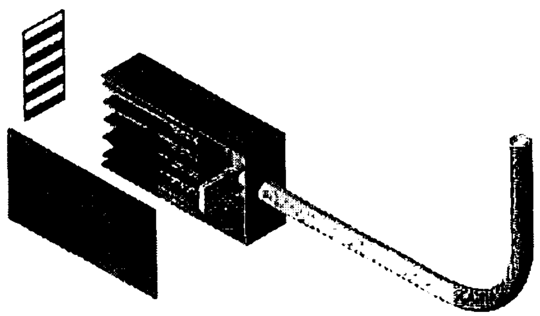

Figure 6. ICRF folded waveguide launcher

6. MAGNET SYSTEMS

The TF coil set consists of 16 coils using multi-filamentary Nb_3Sn and NbTi superconductors with a peak field of 15.8 T at the coil. The TF coils are flattened vertically away from a constant-tension D-shape in order to reduce the size of both TF and PF coils and the peak fields. The design optimizes the superconductor, copper, helium, insulation, and structural ratios by using 4 grades of conductor. Advanced magnet design techniques maximize the utilization of structural materials by using structural forms with grooves into which the conductor is wound. Rather than winding all of the materials in the magnet, only the conductor requires winding.

Support of out-of-plane loads has been provided without intercoil structure in the outer legs of the TF coils, using caps and outer straps. This makes full-sector maintenance possible. The cap and strap structures also have been shown to accommodate off-normal events (e.g., a single coil short during a dump) and the bending stresses due to the shaping.

The PF coil set consists of 24 coils: 10 form the center stack, and the remaining 14 elongate the plasma, provide equilibrium and form the divertor magnetic configuration. An attempt has been made to keep the field at the PF coils <8 T, such that less expensive NbTi conductors can be used. This is accomplished by shaping the TF coils and by allowing a larger number of closely-spaced PF coils.

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