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Engineering design of the ARIES-RS power plant

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Abstract

ARIES-RS is a 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. A steady-state reversed shear tokamak currently appears to offer the best combination of good economic performance and physics credibility for a tokamak-based power plant. In this article, the engineering design features of the ARIES-RS power core are described, with an emphasis on features adopted to meet the top-level requirements, on design integration advances, and on the key issues for further research and development.

1. Introduction

ARIES-RS is a commercial 1000-MW power plant based on a reversed-shear tokamak, DT-burning plasma. The engineering design emphasized the attainment of a set of top-level power plant requirements developed in the early part of the study in a collaborative effort between the ARIES Team and representatives from US electric utilities and industry [1,2]. As compared with earlier studies which aimed to explore new concepts, the ARIES-RS efforts concentrated on refining the level of detail and demonstrating feasibility as self-consistently as possible.

Prior to initiating the design phase of the ARIES-RS study, various classes of engineering design options were examined and their potential to meet the power plant requirements assessed [3]. These options include material choices for the structure, breeder and coolant. The design space for an attractive tokamak fusion power core is not unlimited; previous studies have shown that advanced low activation ferritic steel, vanadium alloy, or SiC/SiC composites are the most promising candidates for the primary in-vessel structural material.

Following this assessment, high-performance vanadium-alloy structures cooled by lithium were chosen for the plasma-facing components. The blanket and shield designs evolved from ARIES-II [4]. Cost-optimization led to the use of steel, helium and efficient shielding materials in locations away from the plasma. Radial segmentation minimizes the waste stream and the replacement cost of life-limited in-vessel components. All components requiring scheduled maintenance are assembled as an integral replacement unit, which is removed as a single piece using straight horizontal motions.

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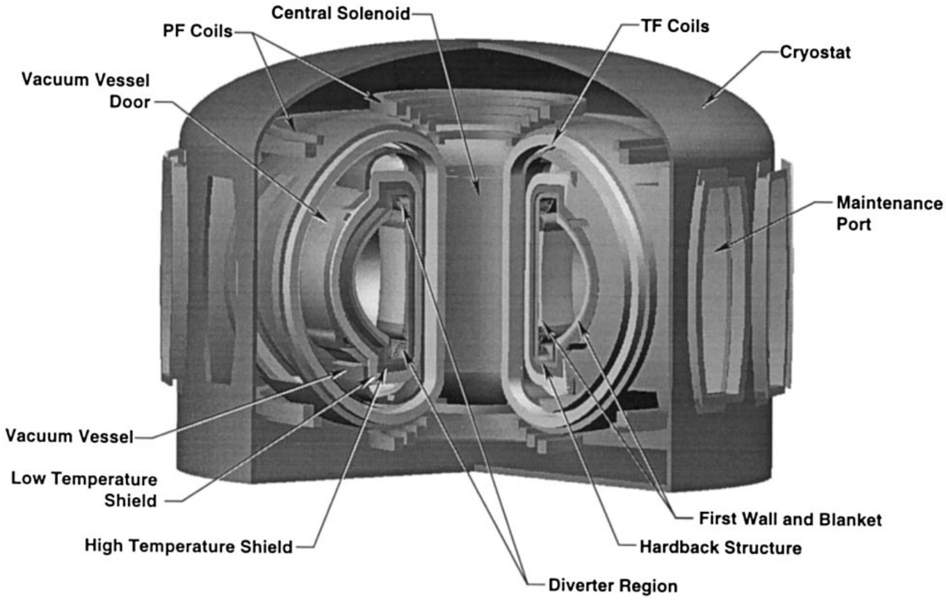

Fig. 1. Cutaway of the ARIES-RS power core.

The adoption of insulating coatings for the Li coolant opens up a wide design space, allowing operation of the first wall at 0.5 MW m^{-2} surface heat flux and 5.6 MW m^{-2} peak neutron wall loading, with a very simple thermal-hydraulic design. Highly radiative zones at the top and bottom of the machine distribute plasma transport energy such that credible design solutions are possible for the divertor plates using solid surfaces. Many of these advantages hinge on the ability to develop effective and reliable insulating coatings on vanadium.

In the following sections, the overall design features of the power core are summarized and the principle features and conclusions from the individual components described. These include the first wall and blanket, radiation shielding, divertor, heating and current drive systems, and magnet systems.

2. Fusion power core configuration

The layout of the ARIES-RS fusion power core is shown in Fig. 1. The figure shows the in-vessel sectors together with the magnets, vacuum vessel and cryostat. Each sector has its own horizontal maintenance port, allowing replacement of the entire sector without opening the cryostat or disassembling other components such as the coil system. Each maintenance port is sealed by two doors that enclose a separate port vacuum. The inner door is formed by the outboard low temperature shield, while the outer door is a separate component located at the radius of the cryostat.

A separate vacuum is created inside this region, such that any coolant leak or dirt caused by handling operations will not contaminate the plasma chamber or the building atmosphere. This feature provides the flexibility to employ either welded or mechanical connectors for the coolant access tubes as well as for the maintenance doors, depending on the shortest maintenance time. For replacement of a sector, only 6–8 coolant access tubes must be disconnected and reconnected to the new sector. All coolant access tube connections are made in the ante-chamber inside the port. More detail on the maintenance procedure is found in Refs. [5,6].

Another distinctive features of this design is the integration of the sectors. The first wall, blanket, divertor, parts of the shield and stability shells

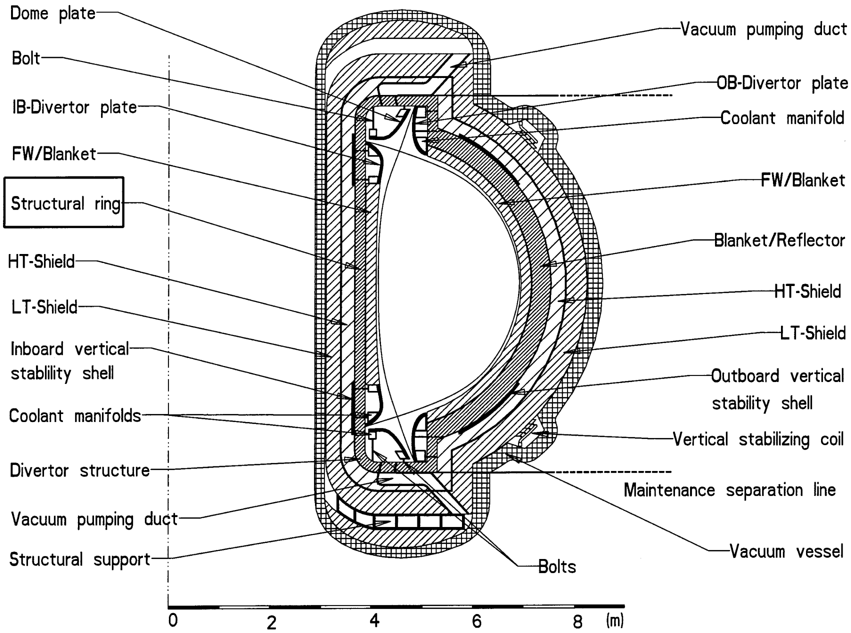

Fig. 2. Elevation view of the sector, showing the maintenance separation line.

form an integral unit within each sector. Fig. 2 shows the cross section of an individual sector, including the 'replacement units' and the maintenance separation line. The integrated sector construction eliminates time-consuming maintenance operations inside the plasma chamber and provides a very sturdy continuous structure able to withstand large loads. The unit is attached only to the bottom structure of the vacuum vessel and can be connected and disconnected by working from the port area. Sectors are disassembled and reuseable parts maintained in hot cells after the plant returns to operation. Furthermore, 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. The relative increase in capital cost as compared with a design having TF-coils closely fitting the outboard blankets is only about 5% of the power core cost. Since the cost of electricity is proportional to the availability, the net savings can be large. A more quantitative comparison of this trade-off requires more detailed studies on replacement time and availability.

In order to maximize the useful lifetime of all elements as well as to minimize the waste stream, all power core elements are subdivided into radial zones characterized by different lifetimes. Three lifetime classes were selected: 2.5 FPY, 7.5 FPY and 45 FPY. The first class with the shortest lifetime is composed of the plasma facing components—the first wall and divertor plates. Underneath those components are the elements with intermediate lifetime. They constitute an integral hardback support structure, or 'skeleton ring' capable of withstanding large loads caused by gravity and disruption forces. The two shielding zones on the outboard side are life-of-plant components, but are also part of the replacement unit in order to facilitate replacement of a power core sector. High-and low-temperature shields in the inboard region as well as in the divertor region are life-of-plant components too, and can remain at their normal position during the removal of a replacement unit. All zones which are not at the end of their lifetime at scheduled maintenance intervals will be separated in the hot cell and reused.

Another advantage of radial segmentation is the thermal-hydraulic decoupling of the zones. In both the blanket and divertor, the rear zone (farthest from the plasma) is used as a 'superheater' to maximize the coolant bulk outlet temperature. The coolant always enters in the regions of highest surface heat flux, where surface temperatures are difficult to maintain within their design limits, and exits in regions heated exclusively by volumetric heating. This configuration works especially well with self-cooled blankets which absorb volumetric heating directly in the coolant. The primary disadvantage is the reduction in radial heat conduction, which is important in loss-of-coolant accidents. Conduction to the shield and vacuum vessel is an important mechanism to limit the peak temperatures in the first wall. Detailed analyses have shown that the safety requirements can be met even in worst-case loss-of-coolant accidents.

3. First wall, blanket and shield

The ARIES-RS blanket uses a self-cooled lithium design with vanadium alloy as the structural material. The V-alloy has low activation, low afterheat, high temperature capability and can handle high heat flux. A self-cooled liquid lithium blanket is simple, and with the development of an insulating coating, has low operating pressure. Also, this blanket gives excellent neutronic performance and the potential for high reliability and long lifetime.

With the assumption of reliable insulating coatings, the MHD pressure drop is no longer a major concern. The design of the FW/B/S can be optimized to improve heat transfer and to simplify the configuration. 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 Fig. 3. The development of insulating coatings is at a very early stage and much more R&D is required. However, the improvements and design flexibility that coatings provide for selfcooled liquid metal blankets make this a very high leverage item.

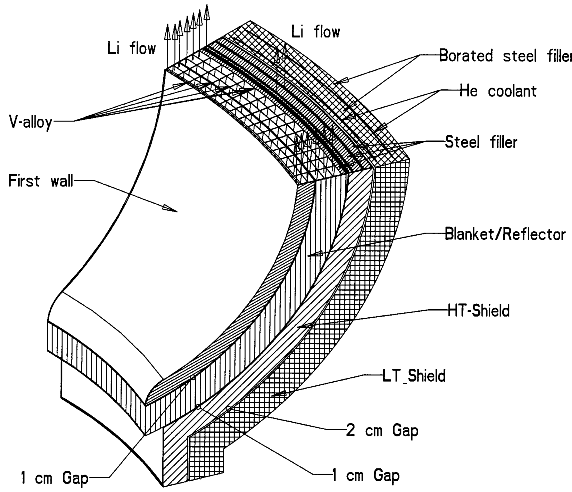

Fig. 3. Outboard blanket and shield.

Table 1 summarizes the heat loads and peak temperatures in the blanket and divertor. The wall loading is determined from Monte Carlo neutron transport analysis of the power core, whereas the surface heat flux is based on a combination of the core plasma radiation distribution plus thermal radiation transport emanating from the radiative divertor slots.

Multiple flow passes in the blanket provide the capability for removing a minimum of 0.5 MW m^{-2} of surface heat flux. 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

Table 1
Power flows and peak temperatures

Average neutron wall load (MW m^{-2})	4.0
Peak neutron wall load (MW m^{-2})	5.6
Average FW surface heat flux (MW m^{-2})	0.42
Peak FW surface heat flux (MW m^{-2})	0.46
Blanket and divertor bulk inlet temperature (°C)	330
Blanket and divertor bulk outlet temperature (°C)	610
Peak V temperature (°C)	700

heating the coolant away from the high heat flux region.

ARIES-RS uses both active and passive stabilization systems for vertical displacement and plasma kink-mode stabilization. These systems were integrated into the sectors. Passive, radiatively-cooled tungsten shells are located between the blanket and shield on the inboard and outboard sides and actively-cooled coils are placed outboard between the low-temperature shield and the vacuum vessel for vertical stability. A thickened vanadium 'second wall' behind the first wall cooling channel was shown to provide sufficient conductivity to stabilize kink modes. The vertical stability shells must be toroidally continuous. Therefore, intersector 'bridges' have been designed in the space between the maintenance ports. Electrical contact is made through compliant materials using a mechanically-inserted wedge.

The design of the primary loop and power conversion system are critical to the attractiveness of the power plant. This system is responsible for efficient power conversion, isolation of radioactive products within the nuclear island, and reliable operation of the power plant. The choice of an advanced Rankine cycle 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.

Detailed analyses identified no critical issues with the shields. A dedicated effort was devoted to the bulk shield in particular, as it represents a major cost item for advanced tokamak designs. Significant savings in shield cost were obtained by implementing several cost-effective improvements: (1) the use of steel in the LT shield, vacuum vessel, and other external components having low levels of nuclear heating that can be dumped without significantly affecting the power balance; (2) the use of cheaper filler materials rather than fabricated structures; (3) segmentation into high temperature (HT) and low temperature (LT) zones; and (4) optimized use of WC and B_4C in the space-constrained inboard side to reduce the overall size and cost of the machine while less efficient, cheaper materials are used in the divertor and outboard sides.

4. Divertor systems

The divertor region of the sector consists of two principal parts: the target plates and the structures. The structures fulfill several essential functions: (1) mechanical attachment of the plates; (2) shielding for the magnets; (3) coolant routing paths for the plates as well as the inboard blanket and replaceable shield; (4) additional heating 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.

The target plates are shown in Fig. 4. A 2-mm-thick castellated 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.

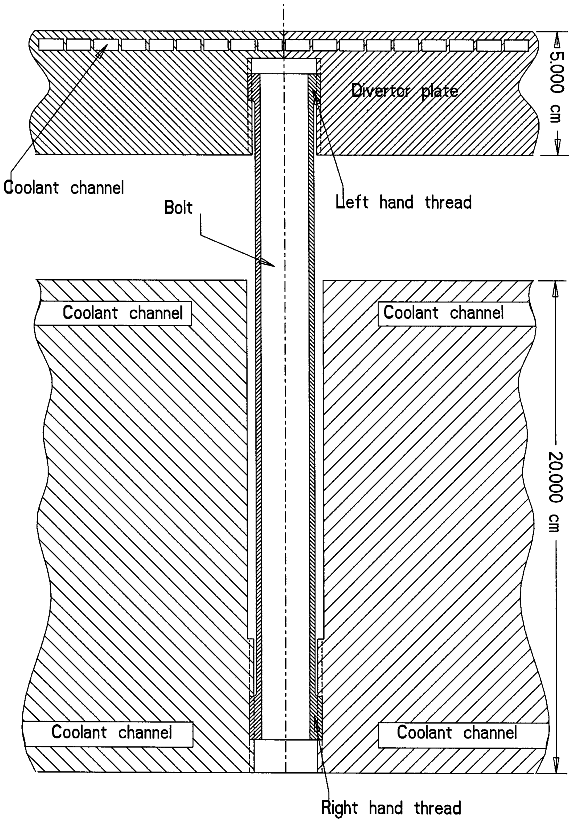

Fig. 4. Divertor plate and attachment.

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 systems

With 88% bootstrap current, the current drive requirements for the reversed shear plasma are modest. However, three RF current-drive systems, operating in different frequency regimes, are required to drive and control the equilibrium current density profile needed to maintain MHD stability at the design point. The power from these systems also is expected to heat the plasma from startup to its final operating conditions. The three RF systems make use of ICRF fast waves (98 MHz), high-frequency fast waves (1.0 GHz), and lower hybrid waves (3.0–4.6 GHz) to drive seed currents in the on-axis, off-axis (inside the RS region), and edge regions, respectively. The total power delivered to the plasma from these systems is 102 MW, of which 81 MW is used in driving useful currents.

The wave launcher and transmission systems are designed and configured so as to minimize intrusions into the blanket and shield structures. All RF launchers can fit within a single blanket sector occupying 0.58% of the first wall area. As a result, the engineering impacts are modest, and the effects on shielding and waste disposal are manageable. Fig. 5 shows the layout of the launchers, which are located around the outboard midlplane. The transmission lines and waveguides are routed behind the shielding structures to min-

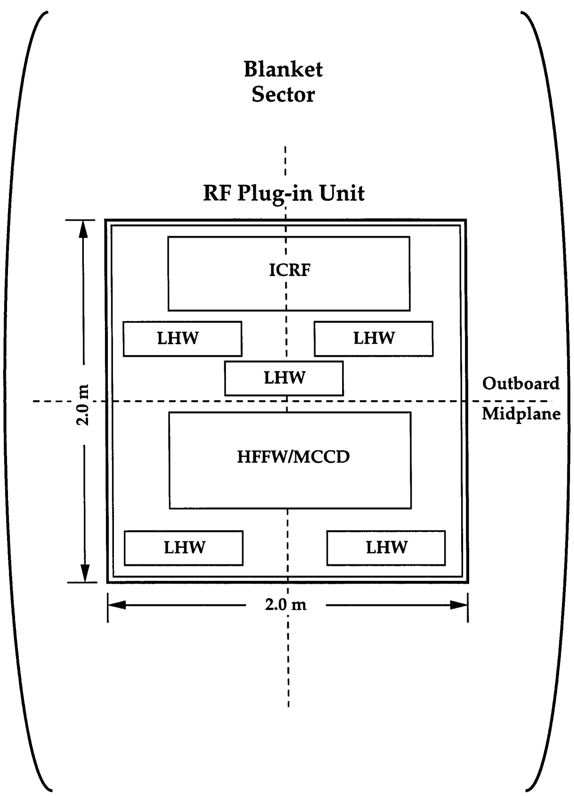

Fig. 5. RF plug-in unit that contains all RF launchers in one blanket sector. ICRF, ion cyclotron resonant frequency; LHW, lower hybrid wave; HFFW, high-frequency fast wave; MCCD, mode conversion current drive.

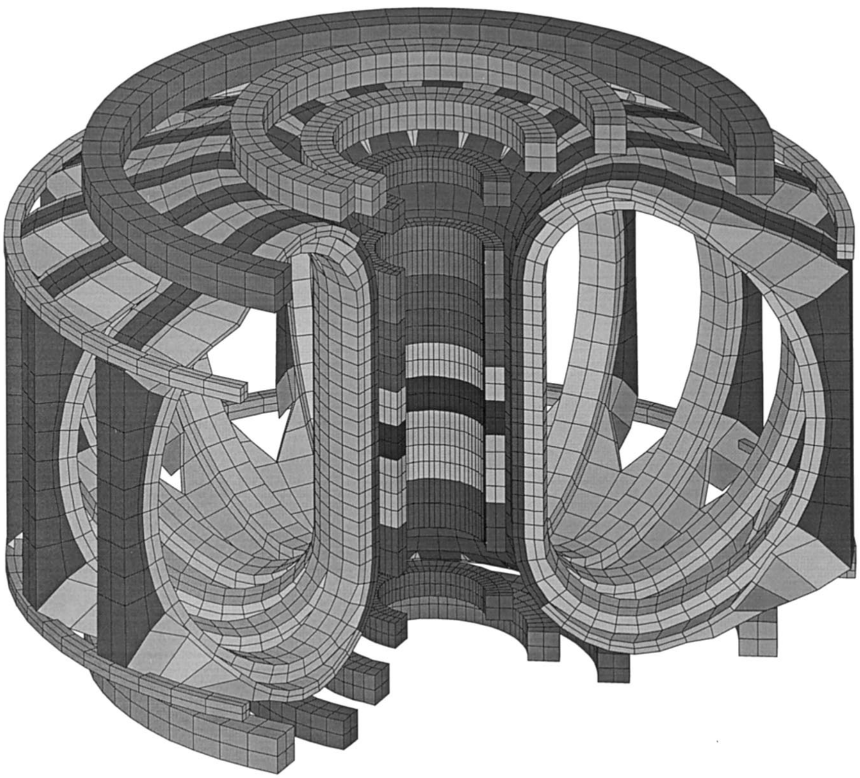

Fig. 6. TF and PF coil systems.

imize neutron streaming and irradiation of ex-vessel components.

The structural material for the in-vessel RF system components is the same V–4Cr–4Ti alloy as that of the blanket/shield. To minimize the RF wall losses, the inside surfaces of the structures are coated with dispersion strengthened copper, GlidCop AL-25, varying in thickness from 8 μm in LH systems to 59 μm in ICRF systems. Loss of copper conductivity over the RF component lifetime due to neutron-induced transmutations does not appear to significantly degrade the system performance efficiency.

The amount of copper in the RF structures is an important concern from the perspective of waste disposal rating. The bulk of the copper is used as the structural material for passive waveguides of the LHW launchers facing the plasma, and to provide sufficient conductive cooling from the surface heat flux. The total amount of copper used in the three RF system components (surface coating and passive waveguides) has been estimated to be about 0.2% of the total structure volume, and therefore should not pose a waste disposal issue if the RF systems are replaced together with the blanket sectors.

6. Magnet systems

The ARIES-RS toroidal field 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 poloidal field (PF) coil set consists of 18 coils: eight form the center stack, and the remaining ten elongate the plasma, provide equilibrium and form the divertor magnetic configuration. The PF system, although large due to the large toroidal field coils, does not present unusual design issues. The system utilizes NbTi and Nb_3Sn

conductors with peak field of 13.5 T occuring near the divertor coils. Fig. 6 shows the overall configuration of both the TF and PF coils and structures.

Three areas were emphasized in the design of the TF coil systems for ARIES-RS:

(1) The magnet internal cross section was optimized to reduce the cross sectional area and the cost. The design uses toroidal-shell structural forms with grooves into which the conductor is wound. Due to the improved capability to support out of plane loads of the toroidal-shell magnets, it is expected that this configuration will be superior to that of case magnets, and to that of magnets with radially-oriented plates. With this toroidal shell approach, the toroidal field coils can be layer-wound, thus making possible the grading of the conductor in a simplified winding process.

(2) Modifications to the TF coils were made to minimize the impacts of the full sector maintenance scheme. Cap-like structures, similar to those of previous ARIES designs [6], are used to minimize the displacement due to the out-of-plane loads. The caps are reinforced with a set of straps located in the shadow of the outer legs of the TF coil. The amount of material added to the structure, is relatively small. The thickness of the band in the outer leg of the TF coil is comparable to that of the thickness of the TF coil.

(3) Failure modes of the TF coil were investigated under off-normal events. It was determined that failure of the poloidal field system or a disruption does not increase the out-of-plane loads on the TF system substantially. The worst case scenario was during a case with a coil shorted across its leads, followed with a dump of the other coils. In this case, the current in the shorted coil increases (by inductance coupling), while the current in the other coils decrease. The system of coils therefore cease to behave as a toroidal system (with D-shape coils as the bending free structure). The over-driven coil tends to become circular. In the case of ARIES II and PULSAR [7], the coil has little capability of carrying bending (since it is made of thin shells), and in the absence of a large support structure (present in ITER and in ARIES-RS), the coil deformations would be very large. The large cap/strap structure serve the dual purpose of minimizing the strains during these faults and to support the out-of-plane loads without the need of intercoil structure in between the outer legs of the TF coil.

7. Conclusion

The final design concept for ARIES-RS offers hope that a tokamak can achieve both the economic as well as safety and environmental characteristics required to provide a competitive product in the future energy marketplace. By optimization of the invessel power core components, the high performance aspects of Li coolant and V alloy structure were exploited, while use of ferritic steel and maximum ‘burn-up’ of the structures reduces the impact of the high unit cost of vanadium. High availability requires reliable operation as well as rapid replacement of components which fail or reach the end of their useful life. The configuration choices made for ARIES-RS accepted small increases in the plant capital cost in order to make a credible case for replacement of sectors in less than a month. One of the most important remaining uncertainties, and indeed, one of the most critical issues for fusion as an energy source is the ability of in-vessel components to operate reliably. However, confidence in reliability estimates requires testing. Until a serious program of testing and component development is undertaken for these advanced technologies, it will be impossible to make a compelling case that these systems will meet their availability goals.

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