Need for Inboard Shield to Protect the Center Post of ST Power Plants

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ABSTRACT

The center post (CP) is the most critical in-vessel component in spherical tokamaks (ST). Advanced ST power plant designs normally call for high neutron wall loads (>5 MW/m^2) forcing the CP to operate in a high radiation environment. Radiation degrades the physical properties of the current carrying conductor and severely affects the overall performance of the CP. An unshielded CP does not appear to offer an attractive design. This paper presents the rationale for shielding the CP of ARIES-ST, the reasons for the design choices, and the consequences of the choices on the power plant design.

I. INTRODUCTION

The remarkable results from the existing experimental spherical tokamak devices have stimulated worldwide interest in this concept. STs offer the potential for significant improvements over conventional tokamaks due to their high beta and high bootstrap current. Several nations are currently building small ST physics experiments to pursue this concept further and develop the physics database. The motivation for developing the ST concept is to ultimately offer an attractive source of energy. The ARIES team has recently launched a study to identify the key physics and engineering issues and to investigate the potential of STs for achieving a cost competitive and environmentally attractive fusion power plant.

The STs are compact, high power density machines, having a unique configuration with a copper TF coil in the center. The compactness of the STs forces the in-vessel components to operate in a high radiation environment. This raises several engineering issues and concerns regarding the performance of the highly irradiated components of ST machines. The TF magnet is a key component that influences the performance of STs due to its large power requirement that is dominated by the center post. The protection of the CP against radiation and its influence on the performance of the ST power plants are critical issues that receive special attention during the course of the ARIES-ST study.

In this section, an overview is given of the currently expected overall configuration of the ARIES-ST design. For a project not completed, the final results cannot be stated with certainty. The interim results reported here are based on neutronics, safety, and economics analyses performed before May 1998 that are subject to change as the design evolves. The ARIES-ST design implements the top-level requirements developed for the U.S. fusion power plants and the essential objective of using low activation materials capable of resisting high neutron fluence. As illustrated in Figure 1, the continuous TF magnet, which serves also as a vacuum vessel, surrounds the internal components. The plasma of the 1.6 aspect ratio machine produces 4.4 GW of fusion power and the plant delivers 1 GW of net electric power. The CP consumes ~700 MW of dissipation power. It is composed of 85% DS GlidCop AL15 conductor and 15% water coolant. Flaring the CP at both ends will reduce the resistive power losses below 400 MW. The outer TF legs are electrically attached to the CP through sliding joints at the top and bottom. The present design employs a helium-cooled shield for the inboard and divertor regions. More efficient, high performance shields are being investigated for the inboard side. The design utilizes a dual coolant LiPb/He blanket with ferritic steel structure

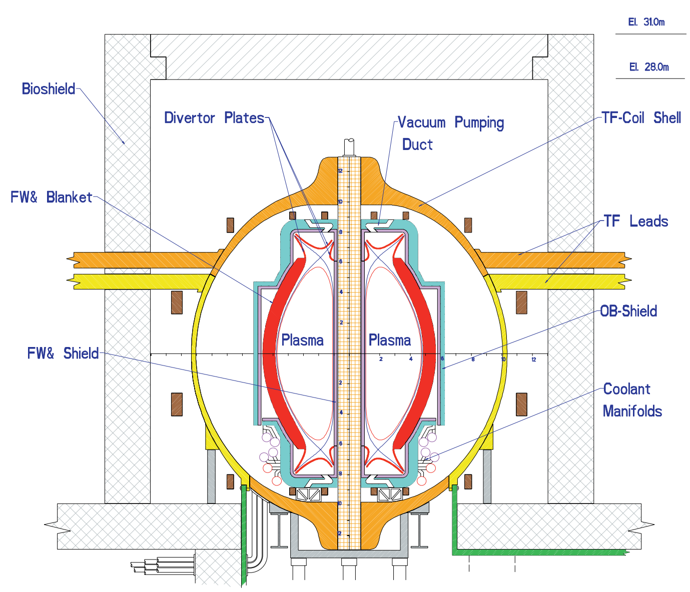

Fig. 1. Vertical cross section of ARIES-ST.

to breed the tritium needed for plasma operation. The blanket system occupies the entire outboard side and is packaged in a single toroidal module. The design allows the removal of the internals (CP, inboard shield, blanket and divertor) vertically downward for replacement and maintenance.

II. KEY SHIELDING ISSUES AND CONCERNS

The top-level requirements for ARIES power plants provide the designers with a number of general design guidelines to consider in the ARIES-ST study. These include the need to optimize the overall design, not only a single component, minimize the cost of electricity (COE), which is the nominal Figure-of-Merit for all ARIES designs, and factor in the safety and economic requirements from the beginning to ensure that the most desirable safety features are integrated in the design in a cost-effective manner. For the inboard components, a number of design, safety, and economic issues and concerns were identified and scoping assessments were performed to develop a self-consistent design that fulfills the top-level requirements. The main issues and concerns for the inboard side are:

1. Compatibility of inboard shield with in-vessel components (mainly CP and blanket)
2. Impact of shielding materials and coolant of inboard side on outboard breeding
3. Influence of inboard side on overall power balance
4. Radiation damage to CP, radwaste level, and lifetime
5. Joule losses in CP

III. SUBSYSTEM REQUIREMENTS FOR SHIELD

The strong impact of the inboard shield on the performance of the adjacent components and on the overall power balance had led to the development of a set of subsystem requirements to guide the designers of ST power plants. The requirements stem from the essential function of the inboard shield and relate to its performance, economic, and safety features:

1. Design requirements for inboard shield
   - must be compatible with CP and blanket
   - enhance outboard breeding
   - maximize neutron energy multiplication (Mn)
   - protect CP against radiation
   - reduce heat load to CP
   - meet stress and temperature limits
   - must be replaceable, reliable, and maintainable

2. Safety requirements
   - Class C low level waste for shield and CP (with impurity control)
   - no damage during LOCA/LOFA
   - low afterheat

3. Economic requirements
   - prolong CP lifetime
   - reduce Joule losses
   - recover inboard heating

While the ohmic losses in the CP provide strong incentives to operate without the inboard shield, the design performance, safety, and economic requirements suggest that an inboard shield would be beneficial to the overall ARIES-ST design. The inboard shield competes with the CP for valuable space in the inboard side. A high performance, thin shield is important for the Joule losses in the CP. However, the highly efficient shielding materials degrade the breeding performance of the outboard blanket and may jeopardize the safety features of the design. This means a thin, high performance shield may not necessarily lead to an attractive ARIES-ST design. The constrained inboard space normally limits the size of the shield in order to allow for a larger CP, and therefore, lower ohmic losses. On the other hand, a sizable shield would be desirable to fulfill the breeding requirements, capture most of the inboard heating, and prolong the lifetime of the massive CP (300 tonnes). Those contradicting requirements imply that the inboard shield design is a compromise between several design constraints and its parameters should be chosen to optimize the overall design, not only to minimize the power dissipation in the CP. The economic impact of the inboard shielding parameters can only be assessed self-consistently using integrated systems analysis.

Several inboard shielding options have been identified for evaluation:

1. Helium-cooled shield
2. Water-cooled shield
3. He-Cooled LiPb blanket
4. High-performance shield (He or H_2O cooled)

In all options, the main structural material is ferrritic steel. The advanced option is currently being pursued to assess the impact of the more efficient shielding materials on the outboard breeding, safety, and power balance. The third option could be needed if the outboard blanket does not provide sufficient tritium for plasma operation. Neutronics calculations showed that the first shielding option enhances the breeding of the outboard blanket and yields an acceptable 1.1 overall tritium breeding ratio (TBR)[1]. The water coolant of the second option helps reduce the size of the shield. However, the water slows down and absorbs the neutrons resulting in less reflection and degradation of the outboard breeding. Figure 2 shows the variation of TBR with inboard shield thickness for the helium and water-cooled options. One-dimensional modeling has been used for the comparative analysis. The TBR reaches the required 1.1 value at a thickness of 20 cm FS/He shield. Even though the water-cooled option offers

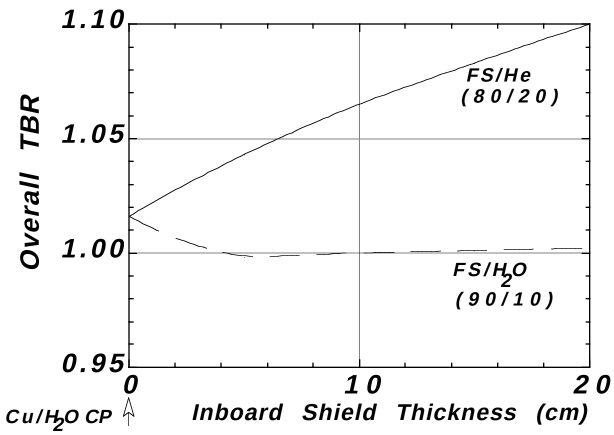

Fig. 2. Effect of inboard shielding materials on outboard breeding.

a thinner shield and lower CP Joule losses, it raises a serious breeding concern. The LiPb blanket will not breed with a water-cooled inboard shield or with a bare CP. Blanket designs employing solid breeders with beryllium multiplier may not encounter this problem, although no solid breeder blanket has been identified that can withstand the high power density in the STs.

An economic factor that will influence the selection process of the reference design is the impact of the inboard shield on the overall power balance. One of the main functions of the inboard shield is to reduce the heat load to the CP in order to alleviate the CP thermal stress problems. The inboard FW and shield contain about 450 MW of surface and volumetric heating. This corresponds to ~10% of the total thermal power of ARIES-ST. If dumped as low-grade heat, the 450 MW will have a negative impact on the power balance. Helium and LiPb are the two main coolants for the ARIES-ST power cycle. For the water-cooled option, the heat removed from the inboard shield cannot be recovered as high-grade heat unless a water cycle is added to the system. If feasible, the economic gain due to the ~150 MW additional electric power will be nearly offset by the cost of the added water cycle. For the He-cooled option, the recovered inboard heating will more than offset the incremental increase in dissipation power associated with the thicker shield. In this regard, the He-cooled option is attractive as it offers high Mn with net power gain.

The pros and cons of the three shielding options are summarized below.

I. Key features of helium cooled shield option:
- \+ 21 cm thick FS/He FW/shield
- \+ ~450 MW deposited in i/b FW/shield will be recovered
- \+ highest nuclear energy multiplication (1.1)
- \+ low impact on outboard breeding
- \+ simple inboard design
- – 10 cm He manifold behind shield
- – Joule losses in CP ~400 MW
- – He pumping power ~60 MW

II. Key features of water cooled shield option:
- \+ ~16 cm thick FS/H_2O FW/shield
- \+ no need for water manifold behind shield
- \+ relatively lower Joule losses in CP (~300 MW)
- \+ simple inboard design
- – degrades outboard breeding (TBR < 1.1)
- – ~450 MW deposited in i/b FW/shield will be dumped as low grade heat
- – low nuclear energy multiplication (1.0)
- – large and costly TF coils
- – high CP replacement cost

III. Key features of LiPb/He cooled blanket option:
- \+ ~25 cm thick LiPb/FS/SiC FW/blanket
- \+ ~450 MW deposited in i/b FW/blanket will be recovered
- \+ highest overall breeding (TBR > 1.1)
- \+ allow for thinner o/b blanket (75 cm instead of 1 m)
- \+ lower inboard afterheat
- – need 5-10 cm inboard He manifold behind blanket
- – complex inboard design
- – Joule losses in CP 400-500 MW
- – high He and LiPb pumping power

Even though the water-cooled option does not meet the breeding requirements, it is included for comparison. The heat recovered from the inboard side, the breeding level, and the waste disposal rating of the CP determine the size of the inboard shield (or blanket). A thinner inboard FS/He shield will lower Mn and more importantly, drop the breeding below the acceptable level. In all cases, the CP meets the Class C low-level waste requirement after three full power years (FPY) of operation.

IV. RADIATION DAMAGE TO CENTER POST

Radiation degrades the performance and limit the lifetime of the ARIES-ST CP. Unless shielded, the CP will be subject to excessive radiation damage[2]. The high neutron wall loading results in high heat load to the unshielded CP (~600 MW), causing high temperature and thermal stresses. Other radiation effects include radiolysis of water coolant, severe embrittlement of Cu conductor, large neutron-induced change in Cu resistivity due to transmutations, and highly activated CP. The latter severely limits the CP lifetime, lowering the system availability and increasing the replacement cost. Overall, a bare CP does not appear to offer an attractive design.

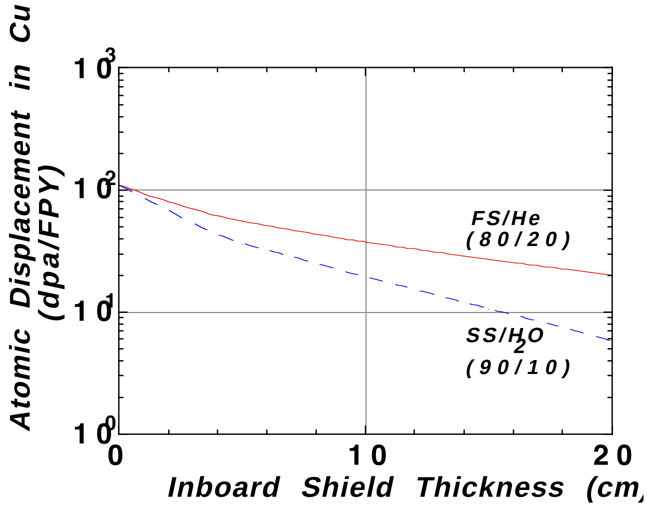

Fig. 3. Variation of peak damage to CP with inboard shield thickness.

A. Embrittlement of Cu

Designing a CP with brittle Cu conductor is a challenging task for the magnet designers. Irradiation tests at temperatures below 150°C indicated hardening accompanied by dramatic embrittlement of all Cu alloys at damage levels as low as 0.1 dpa[3]. In ARIES-ST, the shielded CP reaches 0.1 dpa after a few days and ~60 dpa after 3 FPY of operation. The dpa rate falls with increasing shield thickness and the water-cooled shield offers a lower dpa level than the helium-cooled shield as illustrated in Figure 3. The difference in damage reflects the fact that water is an efficient shielding material. However, the negative impact of water on the LiPb breeding has ruled out its use in the inboard shield. For all shielding options, the Cu conductor of ARIES-ST CP becomes brittle shortly after operation. It is unlikely that the design will employ a fairly thick shield to meet the 0.1 dpa embrittlement limit. Therefore, structural design criteria for embrittled materials need to be applied to the CP design of ARIES-ST. Also, the materials community should develop more radiation resistant Cu alloys for high radiation environments to meet the ST-specific needs.

B. Change in Electric Resistivity of Cu

Copper will interact with neutrons and produce Ni, Zn, and Co transmutations that build up with time and could significantly change the resistivity of the Cu conductor. The inventory of the transmutations in ARIES-ST CP is determined by the DKR-PULSAR2.0 code[4]. The increase in resistivity is the sum of $\rho_i A_i$, where ρ_i and A_i are the specific resistivity[2] and atom percentage of the individual transmutations, respectively. The radial distribution of the increase in resistivity at the end of 3 FPY is shown in Figure 4 for an 80 cm radius CP with a 20 cm FS/He inboard shield. The change in resistivity increases linearly with time. The results reflect the axial drop in wall loading from a peak of 5 MW/m^2 at the midplane to an average of 2 MW/m^2 over the 20 m high CP. The space and time average change in resistivity over a 3 FPY time period amounts to 6%. This translates into an acceptable ohmic heating of 20-30 MW. The outermost 20 cm thick layer of the CP exhibits large transmutations produced by the highly energetic neutrons (E > 5 MeV). In a single turn coil, the electric current will redistribute within the CP to avoid the highly irradiated resistive regions. Of interest is the case of the unshielded CP. The current can hardly flow in the outermost layers of the unshielded CP. The high heat load and the excessive transmutations produced by the much harder neutron spectrum will force the current to flow into the central region. This turns the more resistive outermost layers of the unshielded CP into ineffective space for the electric current. It is, therefore, cost effective to utilize the plasma facing region for shielding to mitigate the CP radiation damage problems and to recover the inboard heating.

C. Radiological Waste of CP

The waste disposal rating (WDR) of the CP appears to be a more critical concern than the neutron-induced embrittlement and transmutations. As a top-level requirement, the ARIES design should generate only low level waste, not greater than Class C, in order to demonstrate that fusion is an environmentally attractive source of energy. It is likely that ARIES power plants would be licensed and regulated by the Nuclear Regulatory Commission (NRC). So far, the NRC guidelines for Class C radwaste, as stated in 10CFR61, were basically developed for nuclear waste generated by medical, industrial, and fission facilities. The NRC eventually will develop a complete list of radwaste limits for fusion

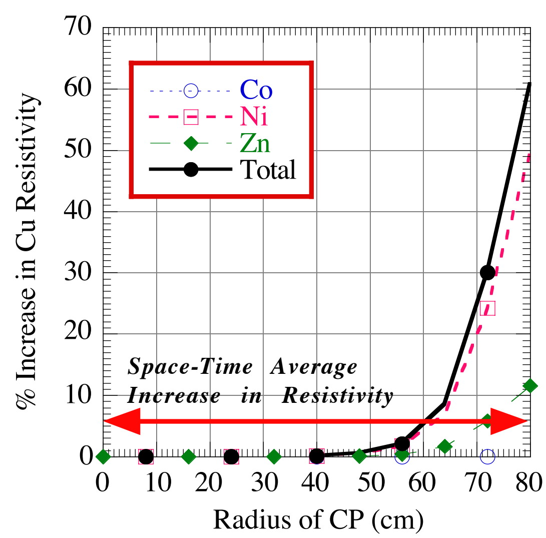

Fig. 4. Radial distribution of neutron-induced change in Cu resistivity.

power plants. In the late 80's, Fetter[5] had used and modified the NRC methodology to develop waste disposal limits for a wide range of radionuclides that are of interest to fusion researchers. Nevertheless, Fetter's limits have not been endorsed by NRC, are not in the form of regulations, and cannot be considered as official limits for U.S. fusion waste. Due to uncertainties in the waste disposal limits, the ARIES team conservatively requires all in-vessel components to meet both 10CFR61 and Fetter's limits for Class C waste for all design studies until NRC develops official guidelines for fusion power plants.

The CP WDR after 3 FPY of operation is provided in Figure 5 as a function of the CP radius for the helium-cooled shield option. Class C waste should have a WDR below one. According to the predominant 10CFR61 limits, an 18 cm thick shield (82 cm CP) would provide a WDR of unity and allow the CP to be replaced on the same time basis (every 3 FPY) as the plasma facing components. As mentioned earlier, at least a 20 cm shield is needed to fulfill the breeding requirements. On the basis of these findings, the baseline design employs a 20 cm thick inboard shield to meet the waste disposal and breeding constraints. Clearly, an unshielded CP will not meet the Class C low-level waste requirement.

Adopting less restrictive radiological limits has insignificant impact on the overall cost of the machine. In the present design, the replacement cost of the CP amounts to ~2 mills/kWh (2% of COE). For a fixed shield thickness, less restrictive WD limits imply longer CP lifetime, meaning higher Joule losses. The incremental change in the overall cost of electricity due to those contradicting effects is very small (<1%).

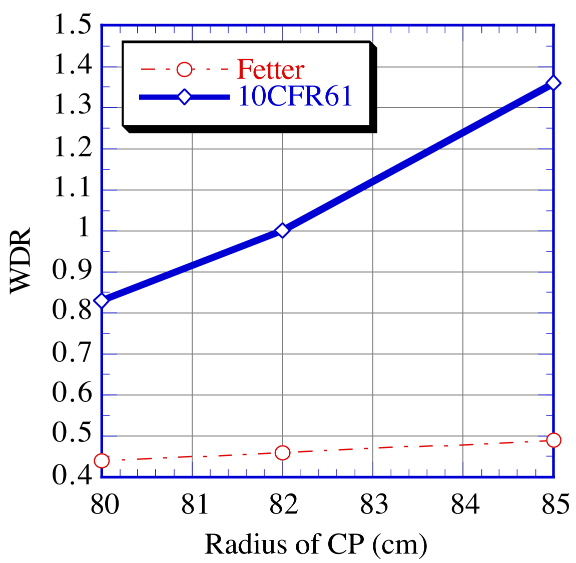

Fig. 5. Variation of WDR with CP radius, trading shield for CP. Dominant nuclides are ^{63}Ni for 10CFR61 and ^{108m}Ag for Fetter.

V. CONCLUSIONS

The ARIES-ST design constraints and requirements provided a number of issues and concerns to consider in designing the inboard shield. The design proceeded iteratively with guidance from neutronics, safety, and economics analyses to ensure that the most desirable features are integrated in a cost-effective manner. Options were investigated to protect the CP and to increase the net electric power by reducing the CP Joule losses and increasing Mn. Because of several contradicting design requirements, the inboard shield design was a compromise between many constraints. The sensitivity of the outboard LiPb blanket to the inboard shielding materials along with the safety constraints have limited the design choices and excluded several high performance inboard shielding options, although further investigation of this issue is ongoing. Based on economic performance and breeding considerations, the optimal shield for the water-cooled CP of ARIES-ST is a 20 cm thick FS structure cooled with helium gas. The selected FS/He inboard shield offers advantages and drawbacks. It is relatively simple, satisfies the breeding and safety requirements, captures useful thermal power, protects the CP for 3 FPY, but results in ~25% higher CP Joule losses due to the space needed for the He coolant. Less restrictive safety requirements and blanket designs with a higher breeding margin could allow the use of a high performance shield and reduce the dissipation power in the CP below 400 MW.

ACKNOWLEDGMENT

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