Nuclear issues and analysis for ARIES spherical and advanced tokamaks

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Abstract

During the past few years, the ARIES team has developed advanced designs to identify the physics and technology areas of high leverage for achieving environmentally attractive and economically competitive fusion power plants. This paper documents the key neutronics parameters and identifies the nuclear issues for the most recent spherical and advanced tokamak designs. The nuclear assessment played an important role during the design process. For instance, the excessive radiation damage to the resistive magnet of the spherical tokamak design provided strong incentives to shield the center post in order to prolong its lifetime and enhance the overall power balance. At present, a number of advanced blanket concepts based on the SiC/SiC composites are being investigated for the advanced tokamak study. An assessment of the breeding performance of the candidate breeders (LiPb, LiSn, and FLiBe) was conducted. The design implications are discussed along with issues related to the limited breeding potential of the LiSn and FLiBe breeders. © 2000 Elsevier Science B.V. All rights reserved.

1. Introduction

Over the past 10 years, the ARIES team has examined the critical design issues for tokamak, RFP, and stellarator fusion power plants. From among the broad range of alternate concepts, the spherical tokamak (ST) has also been studied as the remarkable results from existing experiments have stimulated interest in this concept. The 1000 MW_e ARIES-ST design [1] is now completed and the write-up of the report is currently in progress. The design employs the low-activation ferritic steel (FS) as the main structural material. The blanket is based on the European dual coolant concept with flowing $Li_{17}P_{83}$ breeder through helium-cooled FS structure. As Fig. 1 illustrates, the demountable, resistive TF-coil shell surrounds the internal, removable components; those include the first wall (FW), outboard-only breeding blanket, divertor, inboard shield, and center post (CP).

Recently, the ARIES team has launched a new study based on the advanced tokamak (AT) concept. The study is built on the ARIES-RS design [2] and focuses on more advanced technology and higher plasma performance. The advanced technology includes the high-temperature blanket with $\approx 60\%$ efficiency, high-performance shield, and high-temperature, high-field superconducting magnet. In principle, the ARIES-AT study offers a more compact, less-expensive power plant.

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Three-dimensional Monte Carlo neutronics analysis was performed using the MCNP code [3] to confirm the key nuclear parameters for both designs. Preceding the 3-D analysis, a series of parametric 1-D analysis using the DANTSYS code [4] was established to guide the design process. The data library for both analyses is based on the FENDL-1 evaluation [5]. Comparing the 1-D and 3-D results, important differences were identified for both designs.

2. Breeding requirement and blanket system

The required tritium breeding ratio (TBR) for ARIES-ST and ARIES-AT designs is 1.1. The 10% breeding margin is needed to account for the uncertainties in the cross-section data and geometrical modeling, T losses, and T supply for new power plants. The achievable TBR depends on the blanket coverage fraction. Conventional tokamaks rely on both inboard (IB) and outboard (OB) blankets for breeding. In ARIES-ST, the presence of the central column prohibited the breeding on the IB side. This restriction placed severe constraints on the OB blanket for providing all the T needed for plasma operation.
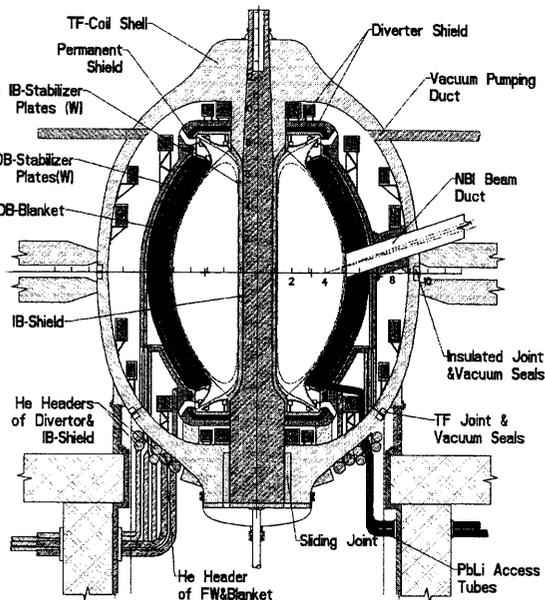

Fig. 1. Vertical cross-section of ARIES-ST

2.1. ARIES-ST blanket system

The LiPb/FS blanket can handle a high heat flux (≈ 0.8 MW/m^2) and neutron wall loading (6–8 MW/m^2). Four radial cells comprise the blanket [6]. The Li$_{17}$Pb$_{83}$ breeder flows poloidally upward in the first cell and returns downward in the other three cells. All cells are lined with SiC inserts that separate the hot LiPb from the FS structure. The helium coolant flows radially inward from manifolds behind the blanket to cool the FW and the blanket FS structure. The He and LiPb coolants exit the blanket at 550 and 700°C, respectively, offering a thermal conversion efficiency of 45% with the Brayton cycle.

2.2. ARIES-AT blanket system

This design is at an early stage of development and is still evolving. The interim results reported here are based on analyses performed before June 1999, and are subject to change. The design adopts the SiC/SiC composites as the main structural material [7]. The combination of SiC structure and liquid metal (LM) breeder offers the potential for operating at high temperatures. Two LMs and a molten salt seem to be compatible with the SiC/SiC composites. Those are Li$_{17}$Pb$_{83}$, Li$_{25}$Sn$_{75}$, and F$_4$Li$_2$Be. Each breeder has its own design issues that will be addressed in detail during the course of the study. This nuclear assessment examines the breeding potential of the three breeders. So far, two blanket designs have been developed for detailed analysis: a self-cooled LM concept and a dual-coolant (He and LM) concept. The self-cooled design was selected for the scoping nuclear assessment. A novel FW design capable of handling high heat fluxes was developed for this concept. The LM flows poloidally upward through the FW, returns downward in the blanket cells, and exits the blanket at 1000–1100°C, allowing a high thermal conversion efficiency of $\approx 60\%$ with the Brayton cycle [7].

2.3. Tritium breeding ratio and energy multiplication

Three-dimensional calculations were performed for the ARIES-ST final design and the ARIES-

Table 1
FW and blanket parameters

	ARIES-ST		ARIES-AT	
	IB	OB	IB	OB
FW				
Thickness (cm)	3.1	3.6	5	5
Composition	26% FS	25% FS	30% SiC	30% SiC
	74% He	75% He	70% LiPb	70% LiPb
Blanket				
Thickness (cm)	–	100	25	50
Composition	–	76% LiPb	92% LiPb	92% LiPb
		6% FS	8% SiC	8% SiC
		6% He		
		12% SiC		
Li enrichment	60%		90%	
Overall TBR — 3-D/1-D	1.10/1.12		1.10/1.13	
Overall Mn — 3-D/1-D	1.1/1.1		1.09/1.1	

AT interim design to confirm the approximate 1-D estimates for the key nuclear parameters. The 3-D model included the essential components comprising the power core: the FW, blanket, shield, divertor, and magnet. The innermost cell of the blanket was modeled separately as it controls the breeding level. The 10 000 source particles resulted in acceptable statistical errors of $< 5\%$ for local values and $< 1\%$ for global values. The nuclear assessment helps determine the thicknesses of the blanket segments, the Li enrichment of the breeder, and the optimal content of the structural materials. The latter proceeded interactively with guidance from the thermo-mechanical analysis. Table 1 lists the key FW and blanket parameters and the overall TBR and neutron energy multiplication (Mn). Excellent agreement was obtained between the 1-D and 3-D TBR and Mn estimates. The 1-D estimate is based on coupling the 1-D results with the neutron coverage fraction of the blanket segments. This method has been used previously in a large number of tokamak studies and has recently proven highly accurate for the ST concept.

The breeding potential of the other breeders (90% enriched $Li_{25}Sn_{75}$ and natural F_4Li_2Be) was assessed using the LiPb/SiC FW and blanket configuration. The estimated 1-D TBR is 0.93 for the LiSn/SiC blanket and 0.88 for the FLiBe/SiC blanket. The breeding requirement could not be met even with fairly thick blankets. Both breeders will certainly need a beryllium multiplier to provide a TBR of 1.1. Beryllium will raise serious safety, economic, and resource issues that need further investigation.

3. Shielding system

The shielding system is an important element of the fusion power core. Unless carefully designed, it could represent a major cost item, particularly for advanced designs. The prime function of the shield is magnet protection. It contains 15–20% of the thermal power, which must be recovered to enhance the overall power balance. As a power production component, the inner part of the shield (facing the plasma) should be cooled with the primary coolant (He or LM). The outer part of the shield, which contains $< 1\%$ of the heating, could operate at a lower temperature and employ its own coolant. Water, for instance, offers superior shielding characteristics and helps reduce the radial build substantially.

3.1. ARIES-ST shield

The protection of the CP against radiation and the influence of the inboard shield on the performance of STs were identified as critical design issues that received special attention during the course of the ARIES-ST shielding study. The severity of the radiation damage to a bare CP has indicated that an unshielded CP does not offer an attractive design. The inclusion of the IB shield was controversial and raised several concerns. Within the limited IB space, the IB shield competes with the CP and is likely to increase the Joule losses of the CP. However, the benefits of having an IB shield may offset the incremental increase in CP Joule losses. The shield offers the advantages of reducing the radiation damage to the CP, limiting the power dissipation caused by neutron-induced resistivity, prolonging the CP lifetime to $> 3y$, meeting the Class C low level waste, and reducing the nuclear heat load and thermal stresses to the CP. Furthermore, an economic benefit is the potential of removing the nuclear heating deposited in the inboard shield as high-grade heat and converting it into electricity instead of being dumped in the low-temperature CP.

An extensive study was carried out to determine the optimal IB shielding parameters that benefit the overall design. Several shielding options were investigated and an assessment was made of the shielding potential of the candidate materials (FS, W, WC, WB) and coolants (He, H_2O, D_2O). It was found that the breeding of the outboard LiPb/FS/He blanket is sensitive to the inboard shielding materials. This placed an additional limitation on the material selection and along with several safety constraints and power balance considerations have discouraged the use of high performance shielding materials, such as WC, H_2O, and D_2O. From an economic viewpoint, helium gas is the preferred coolant as it offers the advantage of recovering the IB heating as high-grade heat. A 20-cm-thick helium cooled FS shield was selected as the reference IB shield. Further analysis has indicated that this shield is optimal for ARIES-ST and the economic gain of recovering the IB heating more than offsets the incremental

increase in the CP Joule losses due to the space taken by the shield.

The divertor and OB shield design is chiefly influenced by the radiation limits of the PF coils. The fast neutron fluence (10^{19} n/cm^2) and heating (2 mW/cm^3) are the predominant limits. The main features of the divertor shield include the use of highly efficient shielding material (WC) to help reduce the height of the machine, the division of the shield into an inner He-cooled high-temperature (HT) component and an outer low-temperature (LT) component, and the selection of water for the LT shield for the dual purpose of cooling and shielding. The reference divertor shield consists of a 25-cm-thick HT shield followed by a 36-cm-thick LT shield. On the OB side, the 1-m-thick blanket is followed by a 25-cm-thick LT shield to reduce the PF coils' radiation damage below the permissible level.

3.2. ARIES-AT shield

The IB, divertor, and OB shields consist of LM-cooled HT components followed by water-cooled LT components. The IB shield employs a highly efficient shielding material (WC) to help reduce the overall size of the machine. Borated FS is used in the divertor and OB shields. The vacuum vessel has the same composition as the LT. For a high neutron wall loading level of 4–7 MW/m^2, the thicknesses of the IB, divertor, and OB HT/LT components required to protect the HT superconducting magnet are 20/45, 50/40, and 15/50 cm, respectively. The dominant radiation effect is the 10^{19} n/cm^2 fast neutron fluence to the superconductor. Further optimization of the shielding components is ongoing.

4. Radiation damage to structure and service lifetime

The FW, blanket, and divertor plates are replaceable components that will be replaced every few years due to radiation damage considerations. The service lifetime of the structural components is determined by the radiation damage attainable during operation. The criterion adopted in this

study is that no more than 200 dpa and 3% burn-up fraction are desirable for the FS and SiC structures, respectively. The 1-D and 3-D results for the radiation effects are reported in Table 2. The damage level depends strongly on the neutron wall loading, which peaks at 4–5 and 6–7 MW/m^2 at the midplane of the IB and OB FWs, respectively. For the same wall loading, the IB FW exhibits higher damage compared to the OB. Based on radiation damage considerations, the lifetime of the structural components is $\approx$ 3 FPY, requiring 13 power core replacements during the plant life (40 FPY). Even though the inboard and divertor are subject to lower wall loadings, they will be replaced with the outboard components. There is certainly an incremental cost associated with the early replacement, but this will be offset by the gain due to the fewer, less complex maintenance process, shorter down time, and therefore higher availability. The less accurate 1-D analysis tends to overestimate the peak damage to the FW and thus underestimates the service lifetime of the structural components. The discrepancy in the results is primarily attributed to the angular distribution of the source neutrons incident on the FW. In the 3-D model, the mostly perpendicular primary neutrons produce lower damage. On this basis, the 3-D results are used to re-normalize the neutron source of the 1-D calculations conducted specifically for the thermal and activation analyses of the ARIES designs.

5. Conclusions

The nuclear issue that raised the most concern in the ARIES-ST design is the protection of the center post against radiation. An unshielded CP does not appear to offer an attractive design due to the high radiation damage and degraded power balance. The analysis showed that the economic gain and design benefits of having a 20-cm-thick FS/He inboard shield more than offset the incremental increase in the CP Joule losses due to the space taken by the shield. The IB shield parameters were carefully chosen to optimize the overall design, not only to minimize the CP Joule losses. Among several breeders, only LiPb can provide tritium self-sufficiency for the SiC-based ARIES-AT design. The other candidate breeders (LiSn and FLiBe) will not meet the breeding requirement unless beryllium multiplier is incorporated in the blanket. With regard to the service lifetime issue, the structural components will require frequent replacement, every 3 years, based on radiation damage considerations. The ARIES-ST shield is well optimized for the design constraints and requirements. The ARIES-AT shield adequately protects the magnet and further enhancement to the shield design is being investigated.

Table 2
Peak radiation damage to FW per unit neutron wall loading

	ARIES-ST		ARIES-AT	
	1-D	3-D	1-D	3-D
Inboard				
dpa/FPY	25	13	28	15
He appm/FPY	210	120	1730	1200
H appm/FPY	830	510	1220	480
Nuclear heating (W/cm^3)	16	7	12	6
Outboard				
dpa/FPY	15	11	18	11
He appm/FPY	170	110	1410	870
H appm/FPY	660	470	990	350
Nuclear heating (W/cm^3)	10	6	9	5

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