Three-Dimensional Neutronics Study for ARIES-ST Power Plant

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THREE-DIMENSIONAL NEUTRONICS STUDY FOR ARIES-ST POWER PLANT

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ABSTRACT

The ARIES team is presently studying a fusion power plant based on the spherical tokamak (ST) concept. This paper addresses the key nuclear issues for spherical tokamaks and illustrates the impact of the neutronics factors on the ARIES-ST design. A three-dimensional analysis was carried out for an interim design to determine the key nuclear parameters. Preceding the 3-D analysis, a series of parametric 1-D analyses were performed to guide the design toward the final configuration. Comparing the 1-D and 3-D results, important differences were identified and attributed mainly to the angular distribution of the incident source neutrons on the first wall. Those differences are unique to spherical tokamaks.

I. INTRODUCTION

ARIES-ST[1] is a low aspect ratio spherical tokamak that delivers a gigawatt of net electric power and operates for over 50 years with an availability of ~75%. Structural components that wear out due to radiation damage during the plant life will be replaced and disposed of as Class C low level radwaste. The demountable, resistive TF magnet surrounds the internal, removable components of ARIES-ST; those include the outboard-only breeding blanket, divertor, and inboard shield. The machine is vertically, remotely maintained and personnel access into the building is not feasible at any time during operation or after shutdown.

Several nuclear parameters had a major impact on the design choices for ARIES-ST. Those parameters are the radiation damage to the structural components, lifetime of in-vessel components, overall tritium breeding ratio (TBR), and nuclear heat loads to the various components. One- and three-dimensional studies were performed to assess those parameters. Most of the analysis was carried out for an interim design of ARIES-ST that had an aspect ratio of 1.6, a major radius of 2.7 m, and a minor radius of 1.7 m. The design is still evolving and it is likely that parameters will change, so they should be taken as preliminary.

II. DEPENDENCE OF TBR ON ASPECT RATIO

In the absence of detailed 3-D neutronics analysis, the overall TBR can be estimated by coupling the 1-D results with the neutron coverage fraction (NCF) that is defined as the fraction of source neutrons incident directly on the breeding blanket. The NCF depends primarily on the aspect ratio, the location and vertical extent of the blanket segments, and the plasma parameters. Accurate evaluation of the NCF ensures that the error associated with the approximate 1-D TBR estimate is within a few percent of the more accurate 3-D results. Figure 1 illustrates the variation of the coverage fraction with the aspect ratio (A). The data were generated by the 3-D MCNP code[2] using the design parameters for STs and tokamaks. It is assumed that the coverage of the outboard segment ends vertically at the X point. The area coverage fraction (ACF) represents the fraction of the first wall surface area covered by the blanket segment. The NCF of the outboard side is larger than the ACF. This confirms the importance of the outboard side for breeding. Changing A within the range of interest for ST designs from 1.25 to 1.8, drops the outboard NCF from 90% to 80%. This 10% drop could represent a problem for high A ST machines, in particular for blankets with marginal breeding.

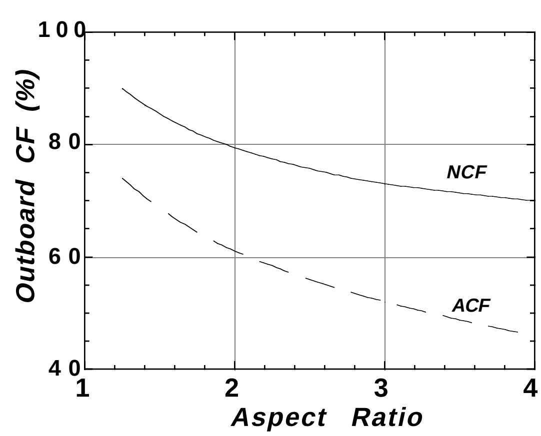

Fig. 1. Variation of coverage fraction with aspect ratio.

III. BLANKET NEUTRONICS

As a top-level requirement, the ARIES blanket must provide an overall TBR of 1.1 to supply the tritium needed to start new fusion plants and to account for 9% uncertainties in the nuclear data and approximations in the calculational model[3]. Four blanket options were proposed for ARIES-ST. Those options are LiPb/FS/SiC/He, Li/V, Li_2O/SiC, and Li_2TiO_3/FS/SiC. The LiPb blanket was judged the preferred option. It is compatible with the core components, particularly the water cooled center post (CP), and avoids most of the safety problems associated with the other concepts. A cross-section of the proposed blanket design with 3 cells, 25 cm each, is shown in Figure 2. The helium coolant of the FS first wall and the 1.8 cm thick grid plates (53% FS and 47% He) flows radially from He manifolds attached to the back of the blanket. The $Li_{17}Pb_{83}$ breeder flows poloidally upward in the first cell and returns downward in the second and third cells. The cells are lined with SiC inserts that separate the LiPb breeder from the FS structure. The He and LiPb coolants exit the blanket at 500°C and 700°C, respectively. A gross thermal efficiency of 45% can be achieved with a power conversion system based on a Brayton cycle with a helium gas turbine.

The preliminary layout of the blanket system calls for a 4.4 cm thick first wall (44% FS and 56% He), a 75 cm blanket (15% FS/He, 15% SiC, and 70% LiPb), and 25 cm thick He manifolds (10% FS and 90% He). The low-activation structure employed in the design is the ORNL 9Cr-2WVTa ferritic steel[4]. The SiC inserts have no structural role and were originally envisioned as a 1 cm thick, 75% dense element with closed surface. The 1-D model included all the elements comprising the FW, blanket, and manifolds as defined in Figure 2. The middle section of each cell was homogenized to take into account the effect of the radial elements on breeding. A 30 cm thick helium-cooled FS shield was assumed on the inboard side[5]. The 1-D model is a toroidal cylindrical geometry with inboard and outboard sides modeled simultaneously to properly simulate the neutron reflection and spectral effects between the two sides. The study was performed using the discrete-ordinates DANTSYS code[6] with P_3-S_8 approximation and the FENDL-1 cross-section library[7] in the 46n-21g group structure.

The 1-D results show that the overall TBR is 1.03 for an A = 1.6 machine with an outboard NCF of 84%. The first cell provides most (63%) of the breeding. A blanket design without SiC inserts could provide an overall TBR of 1.15. However, it offers a lower LiPb exit temperature (~400°C) and a lower thermal efficiency (<40%). Options

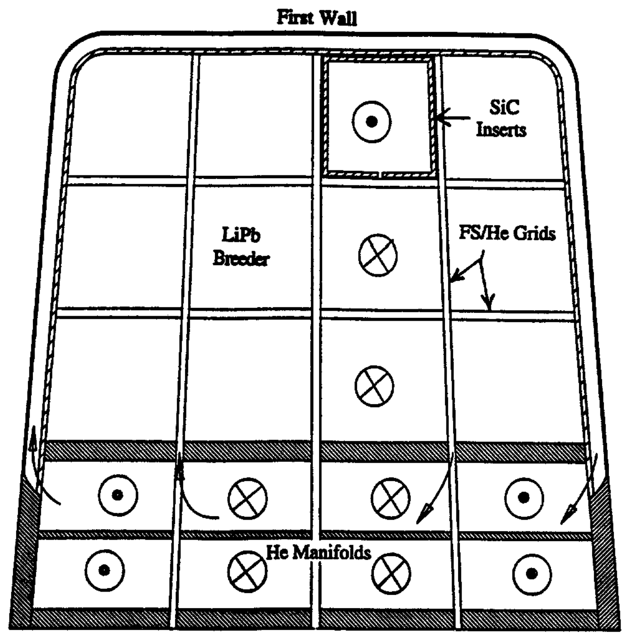

Fig. 2. Cross section of the LiPb blanket with 3 cells.

for minimizing the impact of SiC on breeding were investigated. Table 1 lists the overall TBR for various thicknesses of SiC inserts in the three cells. The results assume a 2% reduction in breeding due to penetrations.

The maximum achievable overall TBR is 1.1 for a LiPb blanket with SiC inserts in all three cells. Reducing or removing the SiC of Cells 2 and 3 has no impact on breeding, and the thickness of the SiC inserts of Cell 1 controls the breeding level. This is attributed to the sensitivity of the Pb multiplier to the spectrum of the incident neutrons. Besides occupying a space that would otherwise be available for the LiPb breeder, the SiC inserts moderate the neutrons and reduce their energies below the ~7 MeV threshold of the Pb (n, 2n) reaction. The softer spectrum leads to a lower neutron multiplication by Pb and subsequently lower breeding. From a practical point of view, 0.3 cm is the minimum desirable thickness for the SiC inserts. So a combination of 0.3 cm SiC in Cell 1 and 1 cm SiC in Cells 2 and 3 was selected for further analysis.

In ARIES-ST, the inboard side represents a small NCF of only 10%. Nevertheless, it was discovered that the materials comprising the CP and shield have a much larger impact on the outboard breeding than previously thought. The severity of the radiation damage to the CP suggests that an inboard shield is beneficial to the overall design[5]. From a shielding standpoint, a water-cooled steel shield is superior to a He-cooled steel shield in protecting the water-cooled CP (70% DS GlidCop AL15 copper

Table 1. Effect of Thickness of SiC Inserts in the Three Cells on Overall TBR

SiC Thickness (cm)			Overall
Cell 1	Cell 2	Cell 3	TBR
1	1	1	1.03
0	1	1	1.15
0	0	1	1.15
0	0	0	1.15
1	0.3	0.3	1.03
0.3	0	0	1.10
0.3	1	1	1.10

alloy and 30% H_2O). However, the water slows down and absorbs the neutrons, resulting in less reflection to the outboard side. The impact of both shields on breeding is illustrated in Figure 3. In this analysis, the water-cooled shield contains 30% H_2O. Both the unshielded CP and the water-cooled shield degrade the breeding by ~10%. Evidently, the LiPb blanket cannot tolerate such a large drop in breeding because of the small breeding margin. For this reason, the inboard shield should be cooled with helium rather than water.

IV. THREE-DIMENSIONAL ANALYSIS

Following the scoping 1-D analysis, a 3-D study was conducted to assess the service lifetime of the components based on the attained radiation damage, to confirm that the blanket fulfills the top-level requirement for tritium self-sufficiency, and to determine the heat loads to the in-vessel components. The 3-D study was carried out using the MCNP code[2] with the pointwise cross-section data library based on the FENDL-1 evaluation[7]. Figure 4 displays the 3-D model for the upper half of the machine. The FW, LiPb blanket, He manifolds, FS/He inboard shield, and Cu/H_2O CP were all homogenized for the 3-D model. A representative outboard and divertor shields were included to simulate the neutron reflection. The model

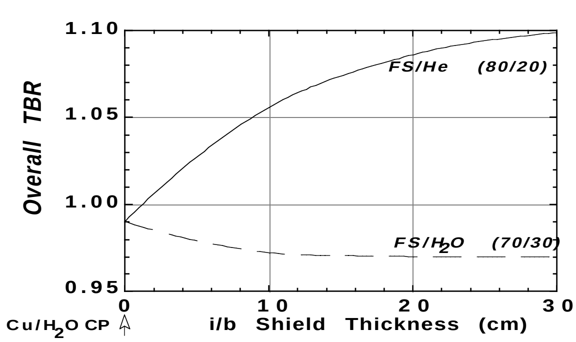

Fig. 3. Impact of inboard shield on outboard breeding.

assumes no penetrations on the outboard side. The non-uniform neutron source is represented by three zones of different intensities. The 100,000 source particles resulted in acceptable statistical errors of <10% for local values and <1% for global values. Geometry splitting and Russian Roulette variance reduction techniques were used to improve the accuracy of the nuclear responses.

A. Overall Tritium Breeding Ratio

For a 75 cm thick homogeneous blanket (77% $Li_{17}Pb_{83}$, 11% SiC, 6% FS, and 6% He), the overall 3-D TBR is 1.05, assuming complete blanket coverage and no penetrations on the outboard side. The corresponding 1-D TBR is 1.08 for a homogeneous 1-D model. The 1.1 1-D value reported earlier is for a heterogeneous blanket that assumes a 2% reduction due to penetrations. The homogeneous model underestimates the breeding level of the blankets as the front zone contains more SiC (11%) than in the actual design (4.5% SiC in Cell 1). Because the SiC of the first cell controls the breeding level of the blanket, it is anticipated that a heterogeneous 3-D model with 4.5% SiC in Cell 1 would result in a higher breeding (~1.08). Based on these findings, the 1-D and 3-D TBR agree within a few percent. This agreement indicates the accuracy of the fairly simple approach used at the early stage of the design to estimate the overall TBR by coupling the 1-D results with the NCF of the blanket.

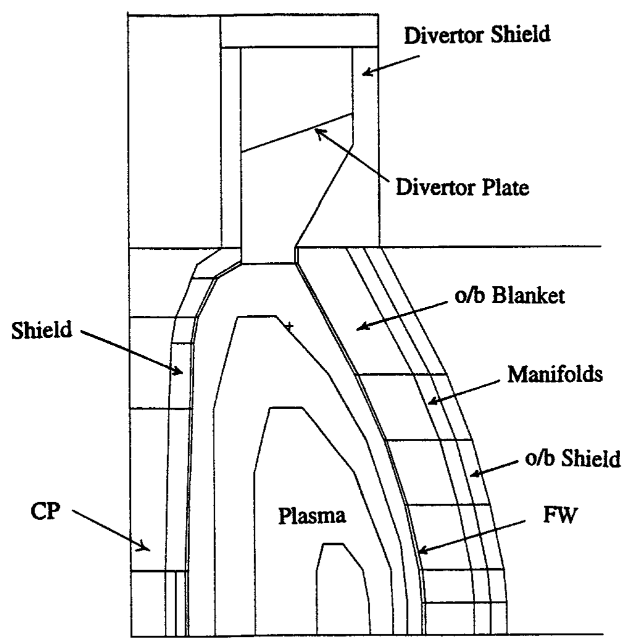

Fig. 4. Vertical cross section of ARIES-ST plotted by the MCNP graphic routine.

According to the 3-D analysis, the 75 cm thick LiPb blanket provides slightly lower breeding than the requirement (1.1). Options to enhance the breeding were identified to augment the TBR by a few percent and to compensate for the losses in breeding due to the most recent changes in design. The latest design calls for a 20 cm thick inboard shield (instead of 30 cm) to allow for a larger CP and lower dissipation power. The thinning of the inboard shield will reduce the breeding by ~2% (refer to Figure 3). Another ~2% reduction in breeding is expected due to the RF penetrations located at the midplane of the outboard. Design changes that enhance the breeding include thickening the blanket by adding a fourth cell and thinning the steel of the outboard FW by 3 mm. Those agreed upon changes will be implemented in the final blanket design to achieve an overall TBR of 1.1. A more effective option to enhance the breeding is to install a thin blanket on the inboard side. This option will jeopardize the simplicity feature of the inboard side and results in higher Joule losses in the CP.

B. Overall Neutron Energy Multiplication

All the helium and LiPb cooled components of ARIES-ST are power production units. Those are the FW, blanket, divertor, and shield. Because of the low operating temperature, the energy deposited in the water-cooled CP will be dumped as low-grade heat. The ARIES-ST design has an overall neutron energy multiplication (Mn) of 1.1. About 75% of the nuclear power is generated by the blanket system. The inboard FW and shield carry 10% of the nuclear heating. This heating should be recovered to improve the power balance and enhance the economics of the ST concept. The low-grade heat of ARIES-ST amounts to only 5% of the total. The 3-D results agree with the 1-D Mn estimate evaluated with the same methodology used for calculating the overall 1-D TBR.

C. Radiation Environment and Components Lifetimes

ARIES-ST operates in a high radiation environment that requires frequent replacement of the plasma facing components during operation. The lifetime of the FS structure is determined by the radiation damage attainable during operation. The criterion adopted in this study is that no more than 200 dpa are desirable for the FW structure. A more restrictive embrittlement limit of 0.1 dpa is imposed on the DS GlidCop alloy of the normal TF magnet. The peak atomic displacement, He and H production, and nuclear heating are reported in Table 3 for inboard and outboard neutron wall loadings of 6.4 and 10.5 MW/m^2, respectively. The most recent ARIES-ST design calls for a lower wall loading level of 5-7 MW/m^2. Based on the 3-D analysis, the end-of-life fluence of the FS structure is 20 MWy/m^2 meaning that the FW/blanket/manifolds should be replaced every 2 FPY. The lower wall loading of the most recent design will extend the service life to ~3 FPY, permitting ~13 replacements during the plant life. Even though the inboard components are subjected to a lower wall loading, they will be replaced with the outboard components. There is certainly an incremental cost associated with the early replacement of the inboard, but this will be offset by the gain due to the fewer maintenance processes, shorter down time, and therefore, higher availability.

Table 2. Peak Radiation Effects at First Wall and CP

	3-D Results	1-D Results
Outboard FW (10.5 MW/m^2)		
dpa/FPY	100	160
He appm/FPY	1060	1730
H appm/FPY	4590	6950
Nuclear Heating (W/cm^3)	80	100
Inboard FW (6.4 MW/m^2)		
dpa/FPY	90	160
He appm/FPY	890	1310
H appm/FPY	3860	5280
Nuclear Heating (W/cm^3)	60	100
Center Post		
dpa/FPY	14	14
He appm/FPY	10	8
H appm/FPY	90	60
Nuclear Heating (W/cm^3)	25	35

V. COMPARISON BETWEEN 1-D AND 3-D ANALYSES

Good agreement was obtained between the global 1-D and 3-D values (TBR and Mn). The agreement between the local values (such as peak radiation damage and nuclear heating) depends on the nuclear responses. As Table 2 indicates, the 1-D analysis tends to overestimate the radiation effects at the first wall by 40-80% and underestimates the damage behind the shield. Table 3 identifies the differences between the two sets of calculations. 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 front damage and higher back damage. The difference in the angular distribution of the source neutrons is more pronounced in ST designs than in tokamaks. Another

Table 3. Differences between 3-D and 1-D Analyses

	3-D	1-D
Model	actual	toroidal cylindrical
Angular distribution of incident 14 MeV n's on FW	mostly perpendicular	perpendicular and tangential components
Plasma shape	actual	cylindrical
n source distribution	actual	uniform, shifted outward
Reflection from i/b, o/b, div.	actual	no div. effect
Vertical variation of wall loading	non-uniform	uniform

factor that contributes to the lower front damage in the 3-D analysis is the vertical variation of the neutron wall loading. This results in a lower reflection from the regions above and below the midplane, compared to the 1-D analysis that assumes a uniform distribution. Based on this comparison, the 3-D results will be used to renormalize the neutron source of subsequent 1-D calculations conducted for the ARIES-ST design. Clearly, the normalization depends on the response function of interest and the location of the component. The key results for the final ARIES-ST design will be confirmed by detailed 3-D analysis.

VI. SUMMARY AND CONCLUSIONS

The neutronics and shielding assessment for ARIES-ST addressed the key nuclear issues such as the breeding potential, radiation damage, service lifetime, and shielding requirements. No serious neutronics issues have been identified as the design fulfills the top-level requirements and meets the radiation limits. Achieving tritium self-sufficiency is one of the challenging tasks in ARIES-ST. As the aspect ratio increases, it becomes more difficult to depend entirely on the outboard blanket to provide all the tritium needed for plasma operation. For an aspect ratio of 1.6, the LiPb blanket has a marginal breeding. Options to enhance the breeding were identified and incorporated in the final design. The 20 cm He-cooled steel shield protects the center post and helps maximize the breeding. Due to radiation damage, the plasma facing components require frequent replacement every 3-4 years of operation.

Comparing the 3-D and 1-D results, important differences were identified between the two sets of calculations. The 1-D results tend to overestimate the radiation effects at the plasma facing components. This difference is more pronounced in ST designs than in tokamaks. On the other hand, good agreement was obtained between the 1-D and 3-D analyses for global parameters, such as overall TBR and Mn. This indicates the accuracy of the simple approach used to estimate the global values by coupling the 1-D results with the neutron coverage fraction of the plasma facing components.

ACKNOWLEDGMENT

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