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ARIES-ST nuclear analysis and shield design

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Abstract

A power plant design based on the spherical torus (ST) concept has been developed by the ARIES team. This paper documents the results of the nuclear and radiation protection analyses carried out for the ARIES-ST design. The nuclear analysis addresses the key neutronics issues, such as the neutron wall loading profile, radiation damage to structural components and their lifetimes, tritium breeding ratio (TBR), and nuclear heat loads to in-vessel components. The main theme of the shielding analysis is to develop guidance and recommendations on radiation protection for the TF magnet, in particular, the center-post. The need for an inboard shield, the selection of an optimal shield, the rationale for the shielding material choices, and the consequences of the shielding material choices on the overall design are reported herein. During the course of the ARIES-ST study, the design has been analyzed rigorously with a set of 3-D nuclear analyses to guide the developing design. The results of the assessment had a major impact on the design choices. For instance, the insufficient breeding along with other design constraints have ruled out the use of several attractive solid breeder blankets, the excessive neutron damage to the center-post provided strong incentives to shield the center-post, and the high radiation damage to the plasma facing components has limited the service lifetime of the ferritic steel (FS) structure to a few full power years. Furthermore, the high heat load to the inboard shield (400 MW, 10% of thermal power) forced the design to recover the inboard heating as high-grade heat to enhance the power balance. Also, the sensitivity of the outboard-only LiPb breeding blanket to the inboard shielding materials has limited the shielding options and excluded several high performance inboard-shielding materials. The performed analyses and results are reported and discussed in relation to the integrated design and the established top-level requirements for the ARIES power plants.
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Keywords: ARIES-ST; Nuclear analysis; Shield design

1. Introduction

The ARIES team has developed a conceptual design for the spherical torus (ST) concept to assess its potential as a commercial power plant. The design is now completed and a write-up of the final report is currently in progress. The ARIES-ST design delivers 1 GW of net electric power, has an aspect ratio of 1.6, and a major radius of 3.2 m [1]. It is assumed that the machine 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 rad-waste. The design employs ferritic steel (FS) as the primary

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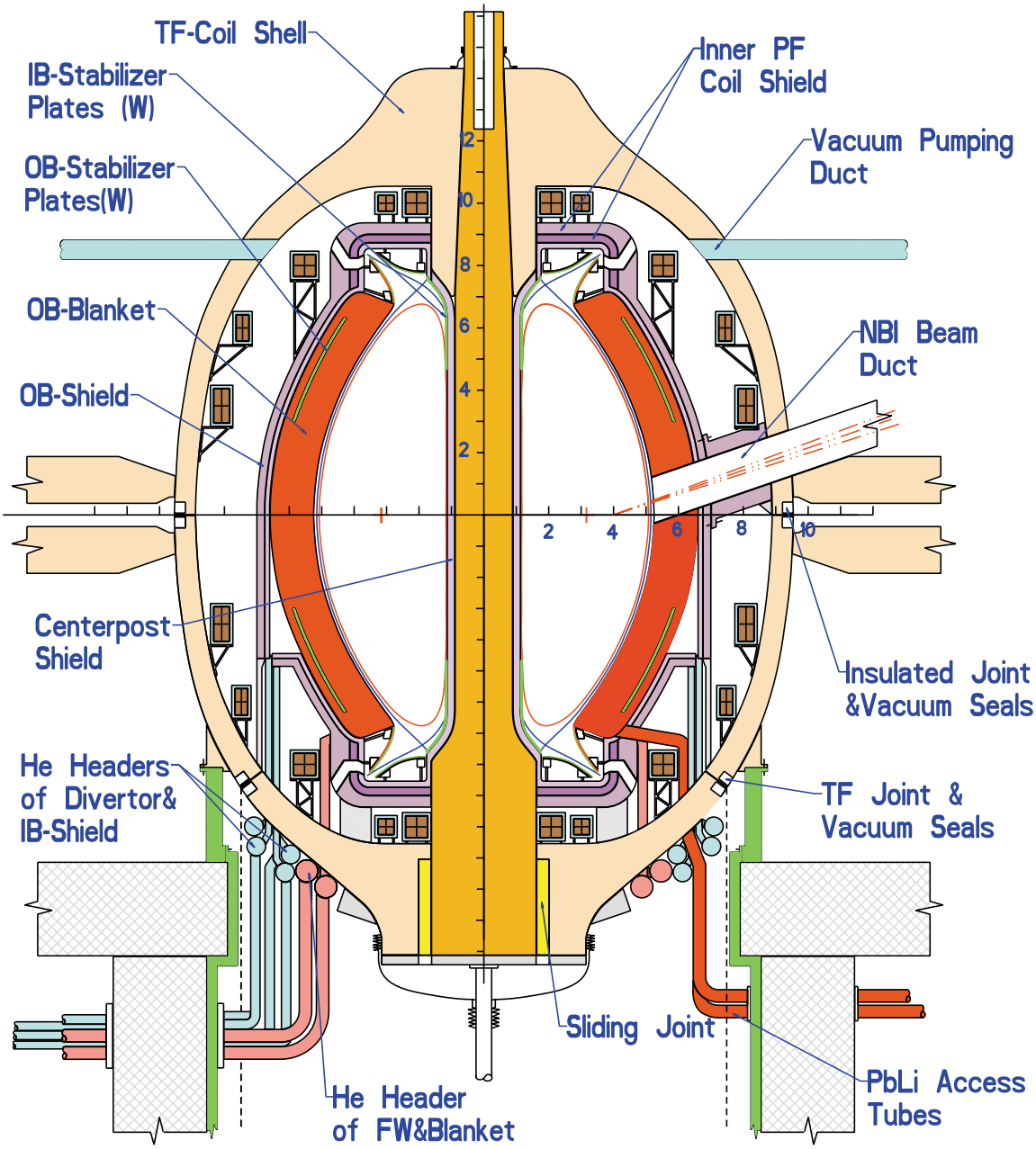

Fig. 1. Vertical cross-section of ARIES-ST.

structure and helium gas as the main coolant. Fig. 1 shows the in-vessel components comprising the ARIES-ST fusion power core (FPC). The demountable, resistive TF magnet surrounds the internal, removable components; those include the blanket, divertor, and inboard shield. The TF-coil outer shell is electrically attached to the center-post through sliding joints at the top and bottom. Two sets of 5-cm-thick tungsten stabilization shells are located at the extremities of the

inboard first wall and outboard blanket. The outboard-only blanket breeds all the tritium necessary for plasma operation and protects the magnet from excessive radiation damage. The TF-coil shell serves also as a vacuum vessel and along with the blanket provides adequate shielding for the superconducting PF coils. The divertor shield is a lifetime component and is integrated with the remainder of the FPC to meet the assembly and maintenance requirements. Local shield surrounds all penetrations to protect the magnets and outer components against streaming neutrons. The FPC is housed in a cryostat and building (not shown in Fig. 1) containing the necessary support systems. The machine is vertically, remotely maintained and personnel access into the building is not permitted at any time during operation or after shutdown.

ST has a unique configuration. The in-vessel components are subjected to a high radiation level. This raised several engineering issues and concerns regarding the performance of the highly irradiated components. 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 center-post against radiation and the influence of the inboard shield on the performance of the machine are critical issues that received special attention during the ARIES-ST nuclear study. A special aspect of ARIES-ST is that the overall design performance strongly favors the need for an inboard shield. This, however, raised some concerns as the shield competes with the center-post for a valuable space and the space taken by the shield is quite likely to increase the power dissipation in the center-post. The open questions, however, were whether a bare center-post can handle the high heat loads and radiation damage and whether the economic gain and design benefits of having an inboard shield that captures useful thermal power offset the incremental increase in the center-post Joule losses. An extensive study was carried out to investigate these issues, to optimize the inboard shield design, and to select the shielding parameters that optimize the overall design. Prior to initiating the study, a set of subsystem requirements was established to guide the design process of the inboard shield. Those requirements stem from the top-level requirements that constitute a common basis for evaluating all fusion concepts and US power plant designs [2].

Achieving tritium self-sufficiency in ST devices is a challenging task. It appears unlikely that breeding blankets could be installed in the inboard and divertor areas because of space limitations. As the aspect ratio of STs increases, the coverage fraction of the outboard blanket decreases and it becomes difficult to depend entirely on the outboard blanket to provide all the tritium needed for plasma operation. During the initial phase of the ARIES-ST study, a number of blanket concepts were developed and a neutronics scoping assessment was made of the breeding potential of the candidate breeders (LiPb, Li, Li_2O, and Li_2TiO_3). A reference LiPb/FS design was then selected for further development. The selection process emphasized the ability to provide the tritium needed for plasma operation, the compatibility of the blanket with the water-cooled center-post, the desire for high thermal conversion efficiency, and the capability of handling high heat loads.

A three-dimensional Monte Carlo analysis was performed using the MCNP code [3] for the final design to confirm the key nuclear parameters. 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 used in both analyses is based on the FENDL-1 evaluation [5]. The results reported herein pertain to the final design. Previous 1- and 3-D supporting analyses performed for interim designs are included in the references. Due to several design modifications, the final results differ from assessments made in the past.

2. Neutron wall loading profile

An accurate evaluation of the neutron wall loading, Γ, is essential for determining the radiation damage to the structure and magnet, for designing the various components of the power core, and for assessing the radiation environment around the torus. Due to symmetry, the upper half of the first wall and divertor was modeled for the

3-D MCNP code to generate the poloidal Γ distribution. The geometry is taken from Fig. 1 and the plasma parameters are based on data provided by the ARIES systems code (ASC) [6]. A combination of cones, cylinders, and planes was used to accurately model the plasma facing surfaces. The surfaces were divided into small segments to obtain the detailed poloidal Γ distribution. One million source neutrons were sampled isotropically from three plasma zones (refer to Section 3.4) with different intensities provided by the ASC. Neutron currents incident on each segment were tallied for the Γ distribution. The statistical uncertainty in Γ at any segment is below 1%. The machine produces a fusion power of 2860 MW. The total surface area of first wall and divertor plates amounts to 700 m^2, resulting in a machine average Γ of 3.3 MW/m^2. Fig. 2 illustrates the poloidal variation of Γ for the inboard, outboard, inner divertor plate, and outer divertor plate. The neutron wall loading peaks at 3.7 MW/m^2 at the mid-plane of the inboard first wall, at 6.4 MW/m^2 at the mid-plane of the outboard first wall, and 2 MW/m^2 at the inner divertor plate. While the peak values are essential for assessing the peak radiation damage and components lifetimes, the average Γ over the various surfaces are needed to determine the global neutronics, thermal, and activation parameters. The average values are 2.1 MW/m^2 for the inboard first wall, 4.6 MW/m^2 for the outboard first wall, 0.9 MW/m^2 for the inner divertor plate, and 0.1 MW/m^2 for the outer divertor plate. Therefore, the peaking factors are 1.8 for the inboard and 1.4 for the outboard.

Tritium breeding depends strongly on the blanket coverage fraction. Plasma support systems (e.g. divertor, current-drive and heating, etc.) could cover 10–20% of the first wall area. In most cases, breeding is low or nil behind these systems. In STs, the center-post prohibits breeding in the inboard side, raising concerns about achieving tritium self-sufficiency. The sensitivity of the coverage fraction to the aspect ratio (A) was examined to estimate the losses in the first wall coverage and breeding due to incremental changes in the aspect ratio. In absence of detailed 3-D neutronics analysis, the global values (such as overall tritium breeding ratio (TBR) and neutron energy multiplication) 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 first wall segments. The NCF depends primarily on the aspect ratio, the location and vertical extent of the segment, and several plasma parameters. Accurate evaluation of the NCF ensures that the error associated with the approximate 1-D estimate during the early stage of the design is within a few percent of the more accurate 3-D results. Fig. 3 illustrates the variation of the coverage fraction with the aspect ratio. The data were generated by the 3-D MCNP code using the design parameters for STs ($A = 1.2$–1.8) and for conventional tokamaks ($A = 4$). The assumption is that the coverage of the inboard and outboard seg-

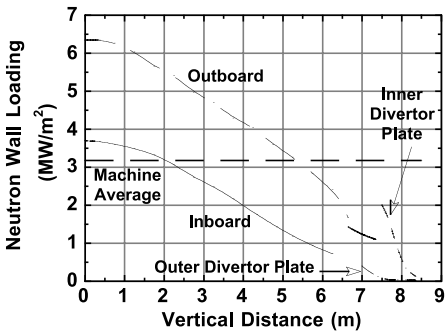

Fig. 2. Variation of neutron wall loading with vertical distance (measured from mid-plane).

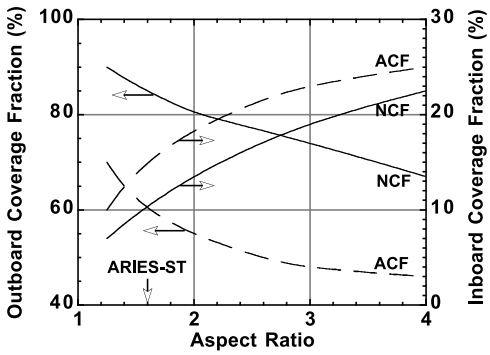

Fig. 3. Inboard an outboard coverage fraction as a function of aspect ratio.

ments end vertically at the cross point of the magnetic field lines. The area coverage fraction (ACF) represents the fraction of surface area covered by the first wall segments. The NCF of the outboard is larger than the ACF. This confirms the importance of the outboard for breeding. On the other hand, the inboard side displays the opposite trend, de-emphasizing the importance of the inboard for breeding. Changing A within the range of interest for STs from 1.25 to 1.8 drops the outboard NCF from 90 to 80%. This translates to $\sim 10\%$ losses in outboard breeding, which could be a problem for high aspect ratio STs, in particular for blankets with marginal breeding.

3. Tritium breeding issues

3.1. Breeding requirements

The breeding blanket must provide an overall TBR of 1.1 or more. The 10% breeding margin is required to account for the calculational uncertainties in cross section data and modeling, tritium losses, and tritium supply for new plants. A detailed discussion of the breeding requirements for the ARIES designs is documented in [7]. A generic breeding-related issue encountered in all power plant design is whether the integral blanket system will over-breed or under-breed after plant operation. Due to the nature of the uncertainties in the cross section data and modeling ($\pm 9\%$), the net TBR achievable after operation could range between 1.01 and 1.19. It appears unlikely that the achievable net TBR could be verified without building and operating the machine. Under-breeding (net TBR < 1.01) will place ARIES-ST at risk as the tritium bred may not suffice for machine operation. In this case, the plant will require an external tritium supply, which violates the top-level requirements. It is, therefore, necessary to conservatively design ARIES-ST with an over-breeding blanket (overall TBR ≥ 1.1) providing that design solutions for reducing the breeding level by up to 20% are established for the reference design. The adjustment in breeding can take place during the second period of operation after the first blanket change out. Supporting calculations should be performed for the preferred solution prior to plant operation to assess the impact of design the changes on breeding and key design parameters.

3.2. Blanket options

The choices of ST design parameters and blanket materials could easily reduce the overall TBR to a marginal level of 1.1 or less. Four solid breeder and liquid metal blankets were proposed for ARIES-ST. Those are the LiPb/FS/SiC/He, Li/V, Li_2O/SiC, and Li_2TiO_3/FS blankets [8,9]. The breeding performance of the proposed blankets was evaluated by a series of 1-D analyses and the overall TBR was obtained by coupling the 1-D results with the NCF. In the toroidal cylindrical 1-D model, the inboard and outboard sides were modeled simultaneously to properly simulate the neutron reflection and spectral effects between the two sides. No blanket was considered for the inboard side due to the limited inboard space. This restriction placed severe constraints on the outboard blanket for providing all the T needed for plasma operation. In the 1-D analysis, a water-cooled Cu center-post surrounded by a helium-cooled FS shield was considered for the inboard side.

References [8,9] cover the details of the proposed blanket options. Here, only basic information on the four blankets and the related breeding issues are given. The LiPb/FS/SiC/He blanket is based on the European Dual-Coolant Demo blanket design [10]. The blanket is packaged within two toroidal modules. This eliminates the need for many side-walls and assembly gaps compared with the usual practice of 16–32 individual modules. A cross section of the four poloidal cells comprising the blanket is shown in Fig. 4. The helium coolant flows radially inward 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 other three cells. The cells are lined with SiC inserts that separate the LiPb breeder from the FS structure and help maintain the FS temperature below 600 °C. The design choices have reduced

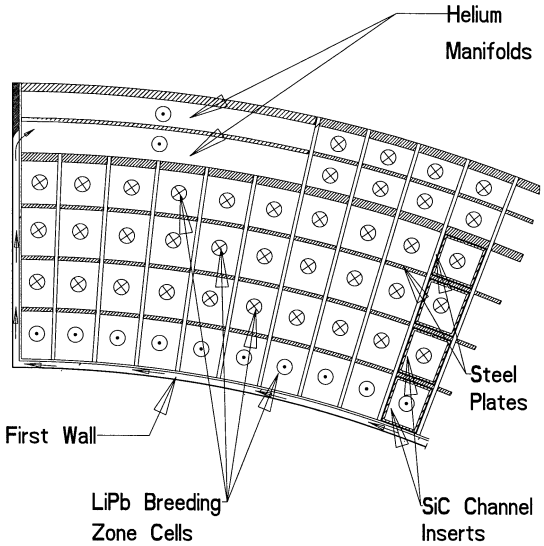

Fig. 4. Cross-section of the reference LiPb blanket at the outboard mid-plane.

the breeding of this blanket to a marginal level. The breeding is sensitive to the amount of SiC inserts, first wall thickness, Li enrichment, blanket thickness, and type of inboard shielding materials. Several modifications to augment the breeding were identified and implemented in the final design.

The Li/V option is an ARIES-RS type blanket with lithium flowing poloidally from bottom to top. A 60–70 cm-thick blanket can easily provide an overall TBR of 1.1. Even though this blanket is fairly simple, can stand high neutron wall loadings (7–10 MW/m^2), and offers high thermal conversion efficiency (45%), it was ruled out for safety reasons. The use of water coolant in the center-post prohibited the use of the chemically reactive Li breeder.

The Li_2TiO_3 blanket is cooled with helium gas that flows radially between SiC channels containing the Li_2TiO_3 breeder. The titanate requires a large amount of beryllium multiplier to achieve an overall TBR of 1.1. A Be to Li_2TiO_3 ratio of 9:1 provides the highest breeding. A 50-cm thick Li_2TiO_3/Be/SiC blanket with 90% enriched Li could satisfy the breeding requirement. The large amount of Be raised some concerns regarding safety (interaction of Be with water), economics (Be cost of \$600 per kg), and limited US resources. Furthermore, the excessive nuclear heating generated in the Li_2TiO_3/Be channels has placed severe constraints on the maximum allowable neutron wall loading (<3 MW/m^2) [9].

The Li_2O blanket calls for granules of Li_2O flowing by gravity from top to bottom. A 60–80 cm-thick Li_2O breeder could provide the required breeding, depending on the size and density of the Li_2O granules. In the absence of metallic structures, this blanket could breed with natural lithium and without a beryllium multiplier. Even though this concept has some salient features, the free fall of granules has many design issues that need further investigation [9].

3.3. Selection of reference blanket

The LiPb blanket was judged the preferred option for ARIES-ST. It is compatible with the core components, particularly the water cooled center-post, and avoids the safety problems associated with the other concepts (e.g. interaction of Li or Be with water). It offers high thermal conversion efficiency of 45%, can handle high wall loadings of 6–8 MW/m^2. There is a class of low-activation FS alloys that is attractive for this application such as the internationally recognized modified F82H FS and the 9Cr–2WVTa FS developed by ORNL [11]. The latter alloy, 9Cr–2WVTa, was used in this analysis. The final blanket design calls for a 3.6-cm thick first wall (25% steel and 75% He by volume), a meter thick blanket (7% steel, 7% He, 12% SiC, and 74% LiPb), and 34-cm-thick He manifolds (10% steel and 90% He). The $Li_{17}Pb_{83}$ eutectic is the preferred breeder. Other LiPb eutectics offer a few percent higher breeding, but require a more complex blanket design. The SiC inserts that separate LiPb and steel [8] were originally envisioned as a 1 cm thick, 75% dense element with closed surface. A preliminary scoping 1-D analysis has indicated that the first cell provides $>50\%$ of the overall breeding and the thickness of the SiC inserts of the first cell in particular controls the breeding level of the blanket [12]. Moreover, a blanket without SiC inserts could provide a 10% higher TBR. However, such a blanket is expected to have a lower LiPb

exit temperature (~400 °C) and a lower thermal conversion efficiency (<40%). From a practical point of view, 0.3 cm is the minimum desirable thickness for the SiC inserts. Therefore, a combination of 0.3-cm thick SiC in Cell 1 and 1-cm thick SiC insert in the remaining cells was selected for the reference blanket.

Even though the inboard side represents a small NCF of only 11%, the material comprising the center-post and inboard shield have a much larger impact on the outboard breeding than originally thought. From a shielding standpoint, a water-cooled steel shield is superior to a He-cooled steel shield in protecting the center-post. However, the water slows down and absorbs the neutrons, resulting in less reflection to the outboard side. The marginal breeding of the blanket indicated the desirability of removing water from the inboard shield. For this reason and for power balance considerations, the inboard shield is cooled with helium rather than water as discussed in detail later in the shielding section.

3.4. Description of 3-D model

Three-dimensional calculations for the TBR and key nuclear parameters were made for several preliminary designs [12] to confirm the approximate 1-D estimates and guide the design toward the final configuration. The 3-D analysis and results reported in this paper pertain to the final configuration given in Fig. 1. A vertical cut through the 3-D model is shown in Fig. 5. The major components in the model can be identified easily by comparing Figs. 1 and 5. The model includes the essential components comprising the power core; the first wall, blanket, shield, divertor, TF magnet, and NBI and ECH penetrations (not shown in the figure), and the 5-cm-thick tungsten stabilizing shell. Homogenization was used when the fine structures within each component were too costly to model. Some small structures that would not affect the calculations were omitted. The first cell of the blanket was modeled separately because it controls the breeding level [12]. Reflective boundary conditions were used to create one-fourth of the model that is equivalent to the full geometry. The neutron source was sampled

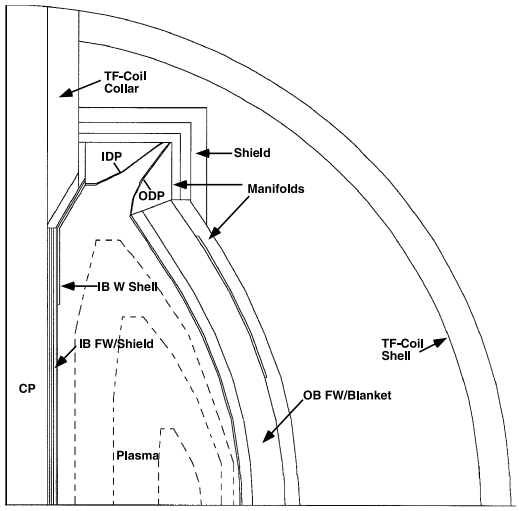

Fig. 5. Three-dimensional MCNP model of ARIES-ST for nuclear analysis (an output from MCNP plotting routine).

from three nested plasma regions as illustrated in Fig. 5. The source intensity is 30% in the inner region, 55% in the middle region, and 15% in the outer region. The source subroutine selects the source region at a frequency appropriate with the source strength. A 14-MeV source neutron-starting position is then chosen uniformly from within that region, neglecting the slight variation across the region. Within selected zones and across the boundaries, the geometry splitting variance reduction technique was used heavily to improve the accuracy of the nuclear responses. The 20 000 source particles results in low statistical errors of < 5% for local values and < 1% for global values.

3.5. Overall tritium breeding ratio

The 3-D results indicated that the overall TBR is 1.1 for a 60% enriched Li. The variation of the 3-D TBR with the Li enrichment is demonstrated in Fig. 6. The blanket has an excess breeding capability of 5% if the Li is enriched to 90%. The 1-D TBR, which is based on coupling the 1-D results with the outboard NCF, is in excellent agreement with the 3-D results. This proves the accuracy of the simple 1-D approach (originally developed for tokamaks) in estimating the overall values for ST devices. A more detailed breakdown

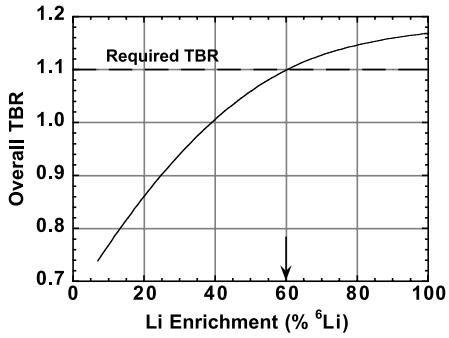

Fig. 6. Dependence of TBR on enrichment of LiPb breeder.

of the breeding is illustrated in Fig. 7. The first cell provides 53% of the overall breeding. Most of the tritium is bred around the mid-plane, emphasizing the need to keep the mid-plane free of penetrations to minimize the losses in breeding.

The ARIES-ST blanket offers several approaches to adjust the net TBR to the required 1.01 level after plant operation. In the case of over-breeding blanket (net TBR > 1.01), lower enrichment of ~20% ^{6}Li (refer to Fig. 6), thinner blanket of ~60 cm, or shorter blanket of ~10 m could reduce the overall breeding by ~20%. The adjustment to the blanket design could take place during the second period of operation after the first blanket replacement. On the other hand, there are two approaches to augment the breeding in the case of under-breeding blanket (net TBR < 1.01). If 0.95 < net TBR < 1.01, the solution is simply to increase the Li enrichment to 90%. If net TBR < 0.95, one or more of the following options should be adopted: install a thin blanket (~25 cm)

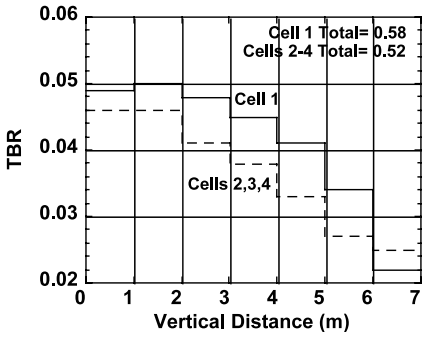

Fig. 7. Detailed breakdown of the TBR in the upper half of the blanket.

on the inboard side, add beryllium multiplier to the first cell of the outboard blanket, and reduce the aspect ratio of the machine to allow higher outboard coverage. Clearly, those solutions require major changes to the entire machine and the addition of Be that may interact with water during an accident could jeopardize the safety feature of the design. It is, therefore, less problematic to design ARIES-ST with an over-breeding blanket that offers an excess breeding margin.

4. Power production

The FPC includes both high-temperature components that are cooled with helium gas and LiPb breeder and low-temperature components (TF magnet and low-temperature shield) that are cooled with water. It is desirable to recover the nuclear power deposited in all components. However, a preliminary study has indicated that it is cost effective to dump the energy of the water-cooled components as low-grade heat because of the low operating temperature. Therefore, only the helium and LiPb-cooled components of ARIES-ST are power production units. Those are the first wall, blanket, divertor, and high-temperature shield.

4.1. Heat load to in-vessel components

Table 1 details the nuclear heating deposited in the various components as calculated by the 3-D MCNP code. The design produces 2860 MW of fusion power. Excluding the low-grade heat, ARIES-ST has an overall nuclear energy multiplication (M_n) of 1.1. Including the surface heating, about 72% of the power is deposited in the blanket system. The inboard first wall and shield carry 13% of the nuclear heating, which is significant. This heating must be recovered to improve the power balance and enhance the economics of the ST concept. The low-grade heat amounts to only 6% of the total thermal power.

A more detailed breakdown of the nuclear heating was also calculated by the 3-D code as input to the blanket and shield thermal hydraulic analysis. The poloidal variation of the peak

Table 1
Nuclear heat load (in MW) to in-vessel components

	Inboard	Outboard	Divertor	Total
He/LiPb cooled components				
Nuclear heating				
Inboard W shells	71	N/A	N/A	71
First wall or divertor plates	18	100	42	160
Blanket	N/A	1929	N/A	1929
Manifolds	N/A	15	64	79
Shield	199	N/A	95	294
Total	288	2044	201	2533
Surface heating	120	200	250	570
Total thermal power	408	2244	451	3103
Water cooled components				
Center-post	164	N/A	N/A	164
TF-coil shell	N/A	39[a]	N/A	39
Low-temperature shield	N/A	N/A	8	8
Total low-grade heating	164	39	8	211

[a] Include 21 MW in the collar and 18 MW in the outer shell.

heating is displayed in Fig. 8 for the outboard first wall, indicating a factor of ~ 1.7 drop in intensity over a 7 m height. The corresponding drop in Γ is 3.5 (refer to Fig. 2). This means a simple approach that assumes the heating closely follows Γ distribution tends to underestimate the heating at the upper/lower ends in particular (see Fig. 8). Other data of interest is the total heating segmented in the vertical direction. This is illustrated in Figs. 9 and 10 for the upper half of the first wall, inboard shield, center-post, and blanket.

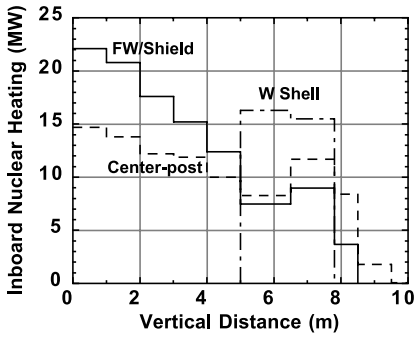

Fig. 9. Nuclear heating deposited in segments of inboard first wall, shield, tungsten shell, and center-post.

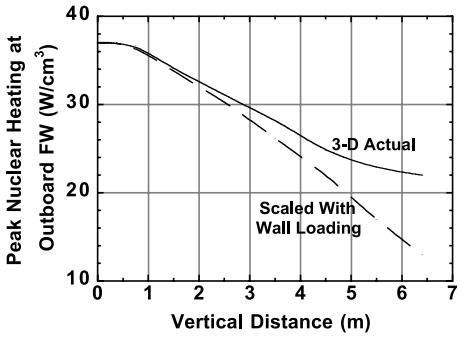

Fig. 8. Variation of peak nuclear heating with vertical distance (measured from mid-plane).

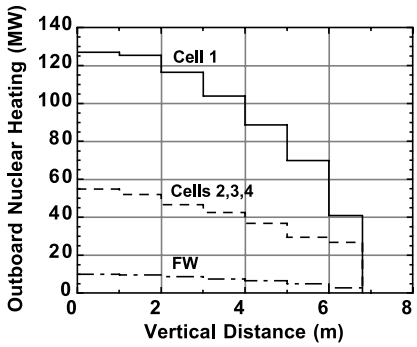

Fig. 10. Nuclear heating deposited in segments (1 m each) of outboard first wall and blanket.

4.2. Thermal power plant split between He and LiPb coolants

The heat load to the helium and LiPb coolants is an essential input to the power conversion system and to the systems code for the purpose of costing the He and LiPb heat transfer/transport systems. The distribution of power is summarized in Table 2. The 430-MW power carried by the helium of the outboard first wall/blanket includes the 110 MW heat transferred through the SiC insulator from the hot LiPb to the colder He. Most of the He pumping power (~ 90 MW) will be recovered by the helium coolant as thermal power. The end result indicates that the total thermal power splits almost equally between the He and LiPb coolants.

5. Radiation damage and lifetime assessment

The high radiation environment of ARIES-ST is expected to degrade the performance of the FPC. The core includes permanent components that perform properly during the entire life of the plant (40 FPY) and replaceable components that must be replaced every few years due to radiation damage. The permanent components are the divertor shield, TF-coil shell, and PF coils. The inboard first wall and shield, outboard first wall and blanket, and divertor plates and manifolds are replaceable components that will be removed vertically from beneath the machine as a single unit. The center-post is not a life of plant component, and depending on its lifetime, will be removed separately or with the plasma facing components. 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 [13] is desirable for the FS structure of the first wall, shield, blanket, and divertor. The DS Glid-Cop copper alloy of the center-post becomes brittle at 0.1 dpa based on recent irradiation tests that showed rapid embrittlement and complete loss of work hardening for all Cu alloys at low dpa level [14]. However, a thorough magnet analysis indicated that the thermal and mechanical stresses are lower than the allowables for brittle material [15], meaning that embrittlement is not a life-limiting factor for the ARIES-ST center-post. A less restrictive activation limit is then imposed on the center-post based on the Class C waste disposal criteria. As discussed later in Section 6.2.7, the center-post lifetime is designed to be equal to or multiple of the first-wall lifetime.

The 3-D results for the peak atomic displacement, He and H production, and nuclear heating are reported in Table 3 for the reference ARIES-ST design in which inboard and outboard neutron wall loadings peak at 3.7 and 6.4 MW/m^2, respectively. The analysis assumes the gas production continues to build up in the structure with

Table 2
Summary of thermal power load (in MW) to helium and LiPb coolants

	He coolant	LiPb coolant
Nuclear heating		
Inboard first wall and shield	288	N/A
Outboard first wall and blanket	430	1614
Divertor system and shield	201	N/A
90% of He pumping power	80	N/A
Surface heating	570	N/A
Total	1569	1614

Table 3
Peak radiation damage to first wall and center-post at mid-plane

Outboard first wall (6.4 MW/m^2)	
dpa rate	69 dpa per FPY
He production	690 appm per FPY
H production	2990 appm per FPY
Nuclear heating	37 W/cm^3
Inboard first wall (3.7 MW/m^2)	
dpa rate	47 dpa per FPY
He production	430 appm per FPY
H production	1870 appm per FPY
Nuclear heating	27 W/cm^3
Center-post	
dpa rate	12 dpa per FPY
He production	18 appm per FPY
H production	130 appm per FPY
Nuclear heating	19 W/cm^3

time. In reality, some of the gases will diffuse out of the hot structure and the actual gas inventories will be below the reported values. Based on the 200 dpa limit and maximum outboard wall loading of 6.4 MW/m^2, the end-of-life fluence for the FS structure is 18 MWy/m^2. This means the plasma facing components should be replaced every 3 FPY, requiring 13 power core replacements during the plant life. Even though the inboard and divertor components 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.

It should be mentioned that the less accurate 1-D analysis tends to overestimate the radiation effects at the first wall by 40–80% and underestimates the damage behind the shield. The discrepancy in the results is primarily attributed to the angular distribution of the source neutrons incident on the first wall [12]. In the 3-D model, the mostly perpendicular primary neutrons produce lower front damage and higher back damage. On this basis, the 3-D results were used to renormalize the neutron source of subsequent 1-D calculations conducted specifically for the thermal and activation analyses of ARIES-ST. Clearly, the normalization depends on the response function of interest and the location of the component.

6. Shielding analysis and design

6.1. Key objectives and requirements

The primary function of shielding system is radiation protection of magnets and workers. As will be shown later, the safety, tritium breeding, and economic requirements have limited the choices of the shielding materials, particularly for the inboard shield. Besides having good shielding characteristics, other features for the shielding materials include low activation, low cost, stability at high temperature and radiation, and compatibility with materials comprising the blanket and center-post. The shield contains $\sim 15\%$ of the thermal power, which must be recovered to enhance the power balance and economics. As a power production component, the inner part of the shield (facing the plasma) should, therefore, be cooled with primary helium or LiPb coolants rather than with low-temperature water.

The outer part of the shield could operate at a lower temperature than the rest of the power core. The amount of heat deposited in the outer shield is relatively small ($\leq 1\%$ of the total nuclear heating) and could be dumped as low-grade heat without having an adverse effect on the overall power balance. One of the reasons for intentionally keeping the outer shield cold is to act as a heat sink and help remove the decay heat generated by the first wall/blanket during a loss-of-coolant accident (LOCA). More importantly, the low-temperature shield could have its own cooling system, and, therefore, employ a coolant (e.g. water) with a good shielding performance to help reduce the radial build.

The outer shield is the closest component to the magnet and its composition plays an important role in controlling the radiation damage level at the TF and PF coils. Four shielding components comprise the shielding system of ARIES-ST. Those are the inboard shield, divertor shield, outboard shield, and penetration shield. Several features of ARIES-ST have influenced the design of the shield. These include the sensitivity of TBR to inboard materials, the constrained inboard space, the need for reducing the height of the machine, the necessity of recovering the nuclear heating deposited in the shield, and the desire to locate the magnets close to the plasma boundary. Those features have placed severe constraints on the material selection and dimensions of the shield. In some cases, the shield design was a compromise between several contradicting requirements and an integrated, optimum solution for key economic, safety, and performance parameters were necessary. During the shielding study, much of the effort was devoted to the inboard shield design because of the strong impact of the inboard shield on the performance of adjacent components and on the overall power balance.

6.2. Inboard shield design

6.2.1. Basic functions

The severity of the radiation damage to a bare center-post has indicated that an unshielded center-post does not offer an attractive design [16]. To the contrary, an inboard shield could improve the power core performance and the resultant economics. This view generated a great deal of interest and discussion within the fusion community. An important part of this study was to understand the trade-offs between the center-post and shield and to translate that understanding into improvements in the design. Throughout the course of the study, the trade-off analysis was periodically updated by closely coupled cost-based, integrated systems analysis. In principle, the main issue was not how large is the incremental increase in Joule losses ascribed to the inboard shield, but whether the inboard shield improves the overall power balance and design performance. As anticipated, an inboard shield offers a potential for prolonging the center-post lifetime, meeting the Class C top-level requirement, reducing the radiation damage to the Cu conductor, limiting the power dissipation caused by neutron-induced resistivity, minimizing the rad-waste stream and replacement cost, reducing the change-out frequency of the center-post and, thus, increasing the overall system availability. 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 center-post. This will certainly improve the power balance, reduce the nuclear heat load to the center-post, and, therefore, mitigate the center-post thermal stress problems.

6.2.2. Main issues and concerns

During the inboard shield study, several issues and concerns were raised regarding the candidate shielding materials, coolants, and size of inboard shield. The concerns relate to the compatibility of the shield constituents with adjacent components (mainly center-post and blanket), the influence of the shielding materials on TBR, the effectiveness of the various shielding materials in protecting the center-post, and the impact of the inboard shield size on the center-post Joule losses, overall power balance, and economics. A trade study was initiated to determine the sensitivity of the machine performance to those factors and to recommend an optimal shield for the final design. Throughout the study, it was mandatory to retain the basic design philosophy of optimizing the overall design, not only a single component, minimizing the cost of electricity (COE), which is the nominal Figure-of-Merit for all ARIES designs, and including the safety and economic factors from the beginning to ensure that the most desirable safety features are integrated in the design in a cost-effective manner.

6.2.3. Performance requirements

An important part of this work was to develop a set of subsystem requirements to guide the design process of the inboard shield. The requirements stem from the essential functions of the shield and from the top-level requirement [2] that constitute a common basis for evaluating the ARIES power plant designs. The identified subsystem requirements for the inboard shield include:

1) design requirements:
   - protect center-post against radiation for 3 FPY or more.
   - Enhance outboard breeding.
   - Reduce heat load and thermal stresses to center-post.
   - Meet stress and temperature limits.
   - Replaceable, reliable, and maintainable.
2) Safety requirements:
   - compatible with center-post and blanket materials.
   - Class C low-level waste.
   - Low decay heat.
   - No degradation during LOCA/LOFA.
3) Economic requirements:
   - prolong center-post lifetime.
   - Maximize neutron energy multiplication.
   - Reduce center-post Joule losses.

It is important to note that the majority of these requirements favor a thick shield except the reduction of center-post Joule losses would favor

a thin shield. This remark contradicts the feeling that compactness of the inboard shield is an obvious means of high performance and cost reduction. It is true that a high performance, thin water-cooled shield is important for the center-post Joule losses. However, the highly efficient shielding materials degrade the breeding performance of the outboard LiPb blanket, reduce M_n, and jeopardize the safety features of the design. This means a thin, high-performance shield may not necessarily lead to an attractive design. On the other hand, a sizable, less efficient He-cooled shield would be desirable to fulfill the breeding requirements, capture most of the inboard heating, and prolong the lifetime of the massive center-post ($\sim$900 tonnes). Those contradicting factors 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 center-post.

6.2.4. Shielding options

Specific work on the inboard shield has focused on the selection of the optimal composition and dimension of the shield, description of the rationale for the choices, and implications of the choices on the overall design. The main goal is to search for the inboard shield that satisfies the design constraints and meets the requirements. Several inboard shielding options have been identified for evaluation. The candidate shielding materials include FS, W, WC, and WB. SiC, Be, C, and V were also examined but found to offer insignificant shielding and breeding enhancements. Water was considered for its superior shielding characteristics relative to other coolants such as heavy water and He gas. LiPb coolant was excluded because the reference outboard blanket provides all the tritium needed for plasma operation. The main structure for the inboard shield is FS, occupying 15% of the volume. No structural role has been envisioned for the W, WC, and WB fillers. It is assumed that the helium fraction for the He-cooled shields is 20% and the water fraction for the water-cooled shield is 10%, by volume. The water density was taken to correspond to a mean temperature of 200 °C and a mean pressure of 2 MPa. The same conditions were applied to the heavy water. Parametric calculations have been carried out to examine the performance of the various shielding options. It was not possible to analyze every option using 3-D analyses. Rather, a representative set of 1-D calculations has been performed for all options. The toroidal, cylindrical 1-D model included the outboard blanket to assess the impact of the inboard shielding materials on the breeding level. In the analysis, the thickness of the homogeneous inboard shield was assumed to range from 5 to 20 cm, trading center-post for shield, and the outer boundary was kept fixed at 110 cm. The strong dependence of the damage to the center-post on the choice of materials within the shield is displayed in Figs. 11 and 12. The results must be regarded as only indicative of the relative effect because of the limited precision of the 1-D calculations. Tungsten and WB (not shown in the figure) led to a slightly higher damage compared with WC while water offered a slightly better shielding performance than D_2O. Figs. 13 and 14 demonstrate the sensitivity of TBR to the inboard shielding materials for the reference LiPb blanket design with 60% Li enrichment. Admittedly, the water coolant would help reduce the size of the shield. However, it slows down and absorbs the neutrons resulting in less reflection to the outboard and degradation of the breeding. Clearly, the LiPb blanket will not provide sufficient breeding with a water-cooled inboard shield or with a bare center-post. A heterogeneous model containing more coolant in the outermost layer of

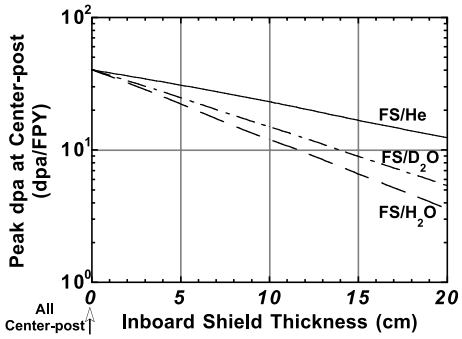

Fig. 11. Sensitivity of center-post damage to inboard FS-based shield with various coolants.

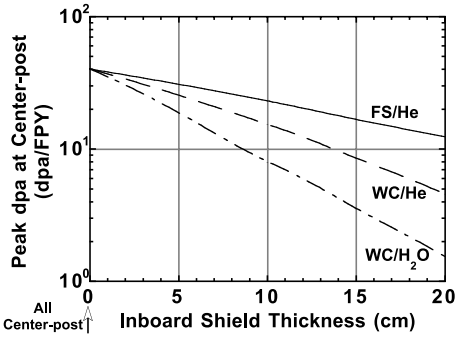

Fig. 12. Sensitivity of center-post damage to inboard FS- and WC-based shields.

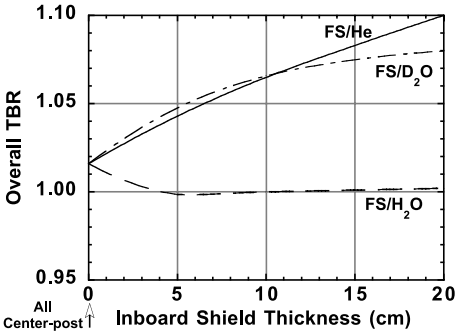

Fig. 13. Sensitivity of TBR to inboard shield coolants.

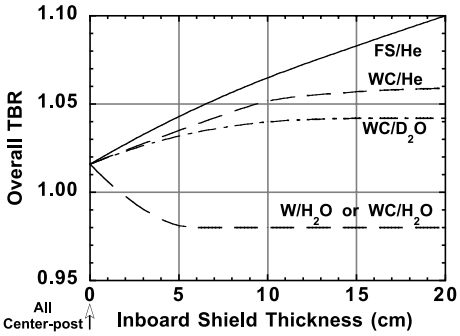

Fig. 14. Sensitivity of TBR to inboard shielding materials.

the inboard shield led to degradation of breeding. The use of heavy water, which has a lower neutron absorption cross section, in all the inboard components (first wall, shield, and center-post) along with raising the Li enrichment to 90% would certainly alleviate the breeding problem associated with the W and WC materials.

6.2.5. *Rationale for shield selection*

An ideal combination from the shielding point of view would be WC (or W) filler and H_2O (or D_2O) coolant. This option represents the thinnest possible shield that minimizes the center-post Joule losses. However, it has several shortcomings. First, the reference design makes no provision for recovering the heat of the low-temperature water loop. This means the 400 MW thermal power deposited in the water-cooled inboard first wall and shield will be dumped as low-grade heat. Recovering such heat would complicate the design due to requirement for an independent high-temperature water circuit. It seems that the added complexity and cost of the water loop will offset the perceived benefits of both recovering the inboard heating at 30% thermal conversion efficiency and the slight drop in center-post Joule losses. Considering the overall power balance, the water-cooled inboard shield does not seem to offer an attractive alternative to the He-cooled inboard shield option. Second, the drop in breeding to a marginal level is a major concern. Third, the D_2O coolant, in particular, generates appreciable amounts of tritium from (n, γ) reactions. A possible design selection for this problem is to add a tritium removal system to prevent the continuous build up of tritium in the D_2O coolant. This will certainly present additional complexity and cost for including the D_2O loop and the T removal system in the design. Fourth, depending on the operating conditions, the radiolysis of water by energetic neutrons and γ rays may lead to high decomposition rates and, therefore, radiolytic corrosion/erosion of the steel structure. Fifth, the high decay heat generated by WC (and W) filler presents a serious safety concern under accident conditions. During LOCA, the temperature of the WC shield is likely to exceed the 700–800 °C limit that warrants the structural integrity and re-usability of the inboard shield. Admittedly, the heavy water allows the TBR to approach the required 1.1 value when Li is enriched to 90%. However, the highly enriched blanket will have no excess breeding capability to account for the overlooked details in the breeding calculations. It is, therefore, desirable to remove the water and tungsten-based material from the inboard shield to

avoid paying an unacceptable price to fix the TBR of under-breeding blankets.

6.2.6. Optimal inboard shield

All the water and helium-cooled W-based shields have breeding, economic, and/or safety problems. There are possible design solutions for some problems but they are likely to be complex and expensive. The helium-cooled FS shield offers a potential for superior performance compared with the other options. Table 4 identifies the advantages and disadvantages of the He-cooled steel shield. Even though the water-cooled steel shield does not meet the design constraints, it is included for comparison. There is a small space penalty paid for the He coolant. Nevertheless, the gain due to the recovered inboard heating more than offsets the incremental Joule losses due to the thicker He-cooled shield. As Table 4 indicates, the drawbacks of the FS/He shield are far outweighed by its benefits. It exhibits outstanding features under the ARIES-ST operating conditions and, therefore, has been selected as the reference shield for the inboard side.

Table 4
Advantages and disadvantages of inboard shielding options

Advantage	Disadvantage
20-cm-thick helium-cooled FS shield	
400 MW nuclear heating recovered	~25 MW higher center-post Joule losses
Higher $M_n = 1.1$	40 MW He pumping power (90% recovered as thermal heat)
Acceptable outboard breeding	–
Safety barrier between center-post and blanket	–
12-cm-thick water-cooled FS shield	
25 MW lower center-post Joule losses	Marginal breeding
–	400 MW nuclear heating dumped as waste heat
–	Lower $M_n = 1.0$
–	Need tritium removal system for D_2O loop
–	Larger center-post and higher magnet cost
–	Water radiolysis could be a problem

Due to concerns related to the optimal thickness of the shield, a trade-off study was conducted for the inboard side. It has been often argued that better performance can be obtained if the shield is kept as thin as practically possible. Initial results have indicated that a 10–15 cm thick FS/He shield would satisfy the breeding requirements (with higher enrichment than 60%) and protect the center-post for 3 FPY (same as first wall lifetime). However, such thin shield may not necessarily improve the overall power balance. The shield thickness that minimizes the inboard power losses will certainly enhance the power balance, and, thus, reduce the overall cost. Therefore, the net power losses from the inboard side have been assessed as a possible means of determining the optimal inboard shield thickness. The 'lost' thermal power in the center-post includes the nuclear heating in the center-post ($P_{n,CP}$) as well as the equivalent thermal power to energize the TF coils and run the helium pumps. The 'recovered' thermal power is the surface heating ($P_{s,IS}$) and volumetric heating ($P_{n,IS}$) of the inboard shield as well as 90% of the He pumping power (P_{pump}). The net inboard power losses (P_{IB}) is defined as the difference between the 'lost' and 'recovered' thermal powers:

$$P_{IB} = \left(P_{n,CP} + \frac{P_{\Omega} + P_{pump}}{\eta}\right) - (P_{s,IS} + P_{n,IS} + 0.9P_{pump}) \quad (1)$$

where P_{Ω} is the center-post Joule losses (150–215 MWe for 0–20 cm thick shield). The helium pumping power is ~40 MWe, mostly for the first wall. The surface heat load to the inboard side is ~120 MW. A thermal conversion efficiency $\eta =$ 45% is used.

An essential input to this assessment is the change in nuclear heating with shield thickness shown in Fig. 15 for both center-post and first wall/shield. The data was generated by a set of 3-D calculations. The center-post was traded for shield and the outer boundary was kept fixed at 110 cm. The net power losses are plotted against the inboard shield thickness in Fig. 16. Clearly, there is a strong reduction in losses (from 1000 to 300 MW_{th}) with increasing shield thickness. The re-

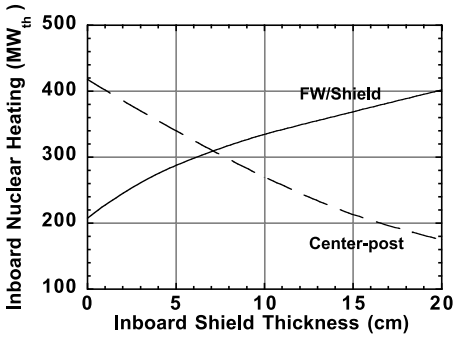

Fig. 15. Dependence of nuclear heating deposited in inboard components on shield thickness, trading shield for center-post.

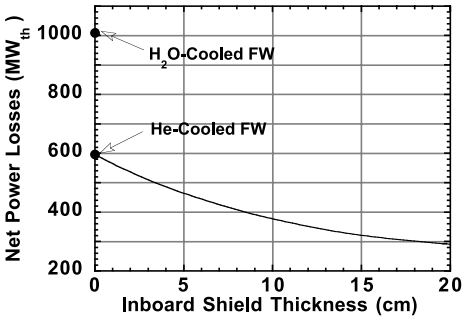

Fig. 16. Dependence of inboard thermal power losses on shield thickness.

covered inboard heating more than offsets the incremental increase in center-post Joule losses ascribed to the shield. The curve peaks at no shield, meaning that a bare center-post (with He or H_2O-cooled inboard first wall) will lead to the most expensive design. The optimal inboard FS/He shield that minimizes the losses and cost is 20 cm thick. This shield helps the design meet the breeding, safety [17] and economic requirements and is utilized in ARIES-ST final design. Specific work on the mechanical design and thermal analysis for the inboard first wall and the 20 cm thick shield are given in [8].

6.2.7. Center-post radiation damage and service lifetime

The center-post employs the DS GlidCop AL15 copper alloy as the primary candidate conductor and the CuCrZr alloy as an alternative. The water coolant comprises 15% of the center-post volume and flows vertically through tubes embedded in the single turn Cu conductor [15]. Radiation degrades the performance and limits the lifetime of the center-post. Even though protected by 20 cm thick FS/He shield, the center-post will be subjected to high radiation damage. It is important to identify potential radiation-induced effects and failure mechanisms so that the center-post lifetime could be reasonably predicted. The radiation effects include the embrittlement of Cu conductor, neutron-induced change in Cu resistivity due to transmutations, activation of Cu, neutron-induced swelling of Cu, and radiolysis of water coolant. To illustrate how severe the radiation damage is for an unshielded center-post, a case was examined early in the design process with no intervening shield between the plasma and center-post [18]. Besides having detrimental effect on the breeding and power balance (refer to Figs. 13 and 16), an unshielded center-post suffers from severe thermal stresses due to the high nuclear heat load (~ 500 MW), becomes extremely brittle after 1 day of operating because of the high radiation environment, and must be replaced every 2 months in order to dispose of the highly activated Cu as Class C low-level waste. The frequent replacement of the center-post will degrade the economics by increasing the replacement cost of the massive center-post (~ 900 tonnes), increasing the rad-waste stream and disposal cost, and lowering the system availability. Overall, a bare center-post does not seem to offer an attractive design.

Three radiation effects were thoroughly investigated to identify the potential failure mechanism, the lifetime limiting radiation damage, and the service lifetime of the center-post. The three radiation effects considered in this investigation are the activation of Cu, embrittlement, and change in resistivity due to transmutations. Other radiation effects such as swelling of Cu and radiolysis of water seem to cause no problem at the relatively low operating temperature of the center-post (< 120 °C) [15]. It appears that the activation of Cu is the predominant, life limiting radiation effect for the center-post. It is more critical than the embrittlement of Cu and the change in resistivity due to transmutations.

6.2.7.1. Center-post service lifetime. As a top-level requirement, the center-post should be disposed of at the end of its useful lifetime as low-level waste, not greater than Class C. Another design-specific requirement is that the center-post lifetime should match or exceed the power core lifetime (3 FPY) in order to enhance the overall system availability. Therefore, the flared center-post is designed to be physically independent from the power core and to be replaced either separately or with the replaceable plasma facing components. According to the activation limits, an unshielded center-post could qualify as Class C low-level waste after a very short service lifetime of 2 months [18]. Since it is not feasible to replace 1000 tonnes of Cu every couple of months, the economics mandates shielding the center-post to prolong its lifetime to 3 FPY (or more), reduce the rad-waste stream, minimize the replacement cost, and improve the system availability. A 10–15 cm-thick FS/He shield would satisfy the radiological limits and allow the flared center-post to be replaced on the same time basis (every 3 FPY) as the plasma facing components. However, as noted previously, a 20-cm shield is needed to enhance the power balance and fulfill the breeding requirements. Based on waste disposal considerations, the 20-cm shield extends the lifetime of the flared center-post to more than 3 FPY. The details of supporting analysis (waste-disposal limits, dominant radionuclides, impurity levels, etc.) are given in [16,17]. The results reported in [16] for an interim design indicate a center-post lifetime of 3 FPY and a waste disposal rating (WDR) of 0.83 for the 22-m-high straight center-post with 80 cm radius. The larger center-post of the reference design (30 m high with 90 cm radius flared at the lower end to 170 cm) calls for more than twice the volume of the previous straight center-post. For the same radiation level, the larger volume reduces the WDR and prolongs the service life of the center-post. Moreover, the additional protection of the center-post by the 5-cm-thick W stabilizing shell and TF-coil collar helps reduce the WDR further. Fig. 17 indicates the significant drop in the neutron flux along the center-post outer surface as computed by the 3-D code for the reference design. The drop in the flux is attributed to the poloidal reduction in the

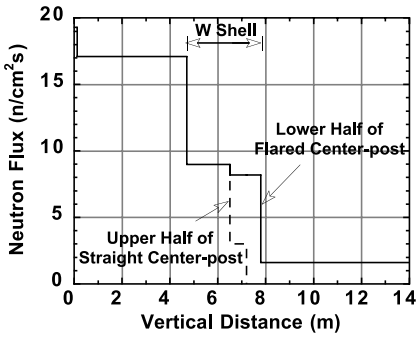

Fig. 17. Variation of neutron flux with vertical distance (measured from mid-plane).

neutron wall loading and the additional shielding provided by the W shell and A1 collar. The combined benefit of the larger center-post and the drop in the axial flux is expected to prolong the center-post service life to 6 FPY or more. Of interest is that adopting less restrictive radiological limits has insignificant impact on the overall cost of the device. In the reference design, the replacement cost of the center-post amounts to ~ 1 mills/kW h (1% of COE). For a fixed shield thickness, less restrictive waste disposal limits result in longer center-post lifetime, more transmutation products, and, therefore, higher Joule losses. The incremental change in the overall COE due to those contradicting effects is very small ($\pm 1\%$).

6.2.7.2. Embrittlement of Cu. There is a serious concern regarding the embrittlement of the Cu conductor under irradiation. 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 [14]. In ARIES-ST, the Cu conductor becomes brittle shortly after operation. Table 3 reveals that the shielded center-post reaches 0.1 dpa after a few days and ~ 70 dpa at the end of its service lifetime of 6 FPY. Due to the self-shielding effect, the central region of the center-post ($r < 45$ cm) is not embrittled (see Fig. 18). It is unlikely that the ST designs could employ a fairly thick shield to obviate the Cu embrittlement. This means the center-post design is not driven by the attainable dpa level. Furthermore, the center-post lifetime will not be limited by embrittlement considera-

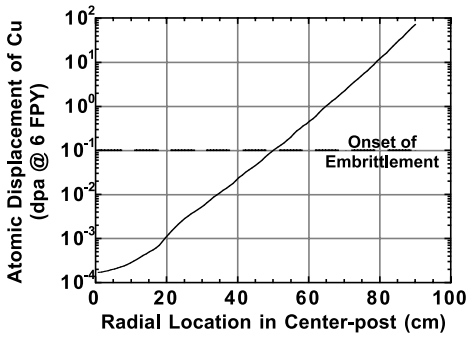

Fig. 18. Radial distribution of dpa at the mid-plane of the center-post at the end of service life (6 FPY).

tions. Rather, the center-post design is driven by the allowable stresses to avoid coil failure due to the rapid crack growth expected with brittle Cu [15]. Designing a center-post with brittle Cu conductor seems to be a challenging task for the magnet designers. Structural design criteria for embrittled materials need to be applied to the center-post design of ARIES-ST [15]. Also, the materials community should develop more radiation resistant Cu alloys for high radiation environments to meet the ST-specific needs.

6.2.7.3. Change in Cu resistivity due to transmutations. Neutrons could severely degrade the electric performance of the center-post. Copper interacts with neutrons and produces transmutations, defects, and dislocations that could change the Cu resistivity significantly, leading to higher Joule losses and recirculating power fraction. The change in resistivity due to radiation damage is small compared with those due to transmutations because large fraction of the defect-induced resistivity is expected to anneal out at the operating temperature of the center-post (<110 °C). Furthermore, the defect-induced resistivity saturates at low fluence while the transmutation-induced resistivity scales somewhat linearly with fluence.

At any instant of time, the two stable isotopes of copper (69.2% ^{63}Cu and 30.8% ^{65}Cu) interact with neutrons and give rise to Ni, Zn, and Co transmutation products. The reactions of interest are Cu(n, H)Ni, Cu(n, α)Co, Cu(n, γ), and Cu(n, 2n). The last two reactions produce unstable Cu isotopes that emit β particles leading to Zn and Ni stable isotopes. The transmutation products continue to build up with time. The outermost layer of the center-post exhibits a large inventory of ^{62}Ni, ^{64}Ni and Co produced by (n, 2n) and (n, α) reactions which have threshold energies of 9 and 5 MeV, respectively. These reactions decrease rapidly within the coil. The change in Cu resistivity is less than 10% when averaged over the flared center-post radial dimension (90–150 cm), height (30 m), and irradiation time ($\sim$6 FPY). The details of the transmutation calculation are given in a companion paper [17]. The neutron-induced change in resistivity translates into a slight increase in Joule losses (<20 MW). The average value is quite lower than the peak, which occurs at the mid-plane of the center-post outermost layer. The difference is due to the exponential attenuation of the neutron flux with radial distance through the coil and the vertical drop of the neutron wall loading over the height of the center-post. In a single turn coil, the electric current will redistribute within the center-post to avoid the highly irradiated resistive areas. Of interest is the case of unshielded center-post. Due to the high (nuclear and Joule) heat load, the outer layers should have a large coolant and small conductor volume fractions. Furthermore, the excessive transmutations produced by the much harder neutron spectrum increases the resistivity of these outer layers, forcing the current to flow in the central region only. As such, the outermost layer of the unshielded center-post are ineffective in producing toroidal magnetic field. The center-post diameter can be reduced if these outermost layers of the unshielded center-post are replaced with a 'thin' but effective shield. In addition, a center-post shield recovers most of the inboard heating and mitigates the center-post radiation damage and stress problems.

6.3. Divertor shield design

STs have unique skinny and tall configuration. Comparing the overall dimensions of STs with conventional tokamaks, there is a significant difference in the height of the machine, but the radial dimensions are almost comparable. Specifi-

cally, STs can be three times taller than tokamaks while the outboard first wall radius at the mid-plane differs by only 25%. This means tokamaks are actually more compact than STs. To help reduce the height of ARIES-ST, an effort was made to design a high-performance divertor shield. The main features of the divertor shield include the use of highly efficient shielding materials, the division of the shield into an inner high-temperature component and an outer low-temperature component [19], and the selection of water for the low-temperature shield for the dual purpose of cooling and shielding.

The primary function of the divertor shield is to protect the top/bottom superconducting PF coils that are placed inside the TF-coil shell for effective plasma control (see Fig. 1). The shield is expected to last for the life of the plant ($\sim$40 FPY). The size of the divertor shield depends on the magnet radiation allowance, the shielding material used, the level of neutron wall loading, and the additional shielding provided by the adjacent components, mainly the manifolds, divertor system, and magnet cryostat. The divertor manifolds serve multiple functions. They carry the He coolant to/from the inboard and divertor regions, provide structural support for the divertor plates and along with the divertor plates, provide lifetime protection for the shield. In order to meet the 200-dpa limit for the FS structure, an equivalent thickness of 13 cm steel must be contained within the 20 cm thick manifolds to protect the divertor shield for 40 FPY. The divertor plates are constructed from small (2–3 cm OD) He-cooled tungsten tubes brazed to the steel manifolds. The neutron wall loading peaks at the inner divertor plate at a value of 2 MW/m^2 (see Fig. 2). The material choices and shield dimensions were chiefly influenced by the PF magnet radiation limits provided by the magnet designers [20]. The limits were established to avoid degradation of the superconductor critical properties by radiation during plant operation. Those limits are the 10^{19} n/cm^2 fast neutron fluence ($E_n > 0.1$ MeV) to the Nb_3Sn superconductor, 2 mW/cm^3 peak nuclear heating at the winding pack, 6×10^{-3} dpa to the Cu stabilizer, and 10^{11} rad to the GFF polyimide insulator. The fast neutron fluence and heating are the predominant radiation limits and will drive the shield design.

Both high- and low-temperature shields use FS as the main structure. The high-temperature shield and the divertor plates are cooled with helium to recover the 550 MW of nuclear heating and, therefore, enhance the power balance. The low-temperature shield, however, has different design constraints. It contains a small amount of nuclear heating ($<$10 MW) that can be dumped as low-grade heat without impacting the power balance significantly. This means coolants with better shielding characteristics could be used for the low-temperature shield instead of helium. The prime candidate is the water coolant. A combination of water coolant in the low-temperature shield and WC filler in both high- and low-temperature shields would adequately protect the PF coils with minimal shielding. Fig. 19 illustrates the beneficial effect of water to the fluence and heating for the reference divertor shield design, which consists of 25-cm-thick high-temperature shield followed by 36-cm-thick low-temperature shield. The results suggest that 15% water, by volume, is the optimum for minimizing the magnet damage. Local shields should surround the PF coils to protect the sides of the magnets against radiation streaming through the divertor pumping ducts.

6.4. Outboard shield design

In addition to the main functions of tritium breeding and heat removing, the blanket assists in

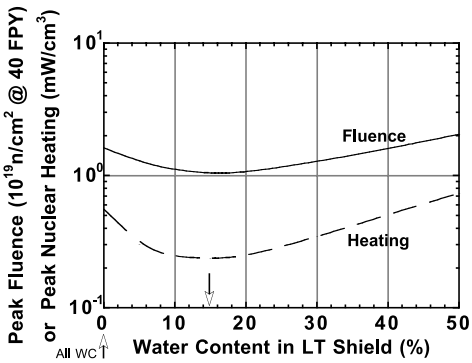

Fig. 19. Effect of water coolant in divertor low-temperature shield on damage to PF coils: trading WC for water.

protecting the magnet from excessive radiation damage. Compared with superconducting PF magnets, the resistive TF-coil shell has less restrictive shielding requirements determined by the radiological limits. Even though the outboard blanket over-protects the shell, it does not provide full protection for the PF magnets. The additional shielding requirement depends on the location of the PF coils; inside or outside the TF-coil shell. If the PF coils are located outside the TF-coil shell, the 1-m-thick LiPb blanket along with the TF-coil shell could adequately protect the PF coils. This means the shell will provide a shielding function. In this case, there is no need for a separate shielding component between the blanket and shell. The shell is cooled with water flowing through tubes embedded in the conductor. The volumetric content of the water coolant and the minimum thickness of the shell at the mid-plane are determined by the PF coil shielding requirements.

A shielding analysis was performed to demonstrate the relative merits of the candidate conductors (Cu and Al) and coolants (H_2O and borated H_2O) as a function of shell thickness. The results indicate that Cu has superior shielding performance compared with Al. The PF coil radiation limits can be met with 30-cm-thick water-cooled Cu shell, 50-cm-thick borated water-cooled Al shell, or 70-cm-thick water-cooled Al shell. The sensitivity of the magnet radiation damage to the coolant content of the shell is shown in Figs. 20 and 21. The results suggest that a 15% coolant would be useful not only for lessening the

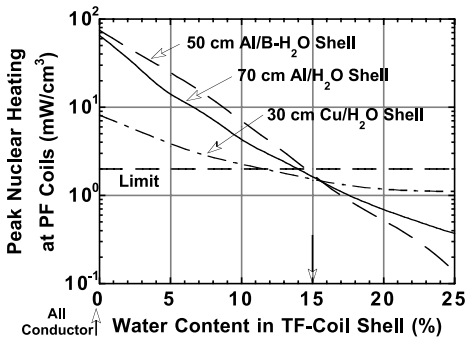

Fig. 20. Sensitivity of nuclear heating at PF coils to water content in TF-coil shell, trading conductor for water.

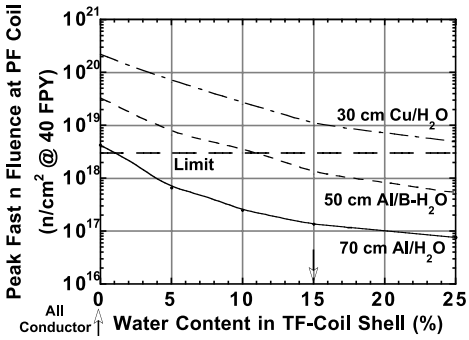

Fig. 21. Sensitivity of fast neutron fluence at PF coils to water content in TF-coil shell, trading conductor for water.

severity of the nuclear heating, but also for reducing the fast neutron fluence below the 3×10^{18} n per cm^2 limit for NbTi superconductor. The heating limit is more demanding as 85% of the damage to NbTi can be annealed out by warming up the PF magnets during shutdown periods. The 70-cm-thick Al/H_2O shell was judged the preferred option for offering lower Joule losses, lighter weight, and, therefore, lower cost [15].

If the PF coils are placed inside the TF-coil shell, an additional shielding component will be needed between the blanket and PF coils. The thickness of the shield is determined by the permissible radiation level at the PF coils and type of shielding materials. Several materials could be used in the outboard shield. Those are FS, borated FS, WC, and B_4C with helium or water coolant. Unless carefully selected, the materials of the massive outboard shield could represent a major cost item. From an economic viewpoint, it is cost effective to employ low cost, less efficient shielding materials (such as FS, borated FS, and H_2O) rather than highly efficient, expensive materials [19]. Excluding the costly W and B_4C materials limits the material choices to FS, borated FS, He, and H_2O. Borated FS (with 3 wt% B) offers a lower magnet heating compared with FS. As anticipated, water has a superior shielding performance relative to He coolant. Figs. 22 and 23 and Table 5 detail the impact of coolants on magnet damage and on size and cost of outboard shield. The water-cooled shield is notable for its effectiveness, compactness, lightweight, and low cost. Fig. 24 displays the substantial reduction in

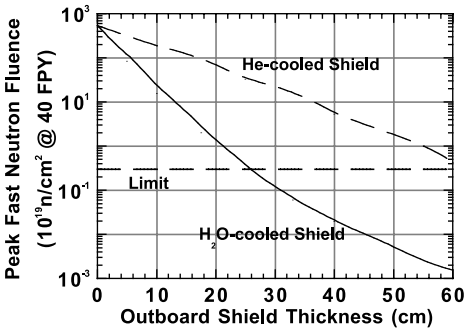

Fig. 22. Dependence of fast neutron fluence at PF coils on thickness and coolant of outboard shield.

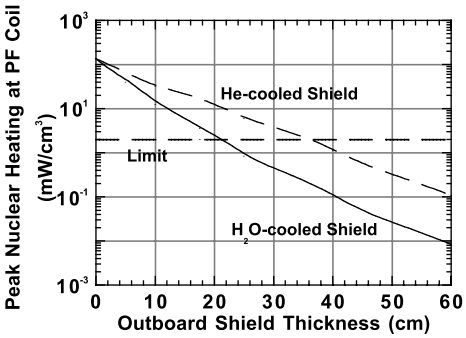

Fig. 23. Dependence of nuclear heating at PF coils on thickness and coolant of outboard shield.

Table 5
Comparison of size and cost of candidate outboard shields

	H_2O coolant	He coolant
Outboard shield thickness (cm)	25	65
Outboard shield mass (tonnes)	700	3800
Outboard shield cost (million \$)	30	120

damage with water content. No additional benefit was obtained with borated water. A 25-cm-thick outboard water-cooled shield reduces the radiation damage to the permissible level at the PF coils. The optimal shield consists of 15% FS structure, 35% borated FS filler, and 55% water coolant. The only concern for the water-cooled shield is the impact of the coolant on the power balance. In other words, throwing away $<1\%$ of the power to make a thin, less expensive shield actually pays off.

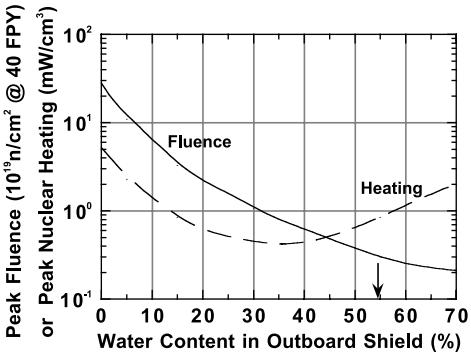

Fig. 24. Variation of fluence and heating with water content in outboard shield.

7. Conclusions

The nuclear assessment proceeded interactively with the economic and safety analyses to determine how the various nuclear parameters influence the performance of ARIES-ST. The areas that raised most concern were the tritium breeding and center post protection. Prior to initiating the nuclear analysis, subsystem requirements were developed to ensure meeting the top-level design goals. In relation to the breeding issue, several modifications have been incorporated in the final LiPb/FS/SiC/He blanket design to achieve tritium self-sufficiency. The marginal breeding of the blanket indicated the desirability of removing water and W from the inboard side. An unshielded center-post does not appear to offer an attractive design. The main concern is that a bare center-post will not survive the intense radiation environment and will require replacement every 2 months based on waste disposal considerations. The center-post of STs exhibit severe embrittlement problem that imposes strict requirements on the center-post design.

The nuclear analysis confirmed the necessity of shielding the center-post and considering all design aspects, particularly the safety and economics, during the design process of the shield. Selecting the optimum shielding parameters was a complex task involving trade-offs between numerous competing factors. Of interest is that an effort to reduce the center-post Joule losses and neglecting the benefits of the inboard shield is likely to lead to

non-optimum designs. The significance of the 20-cm-thick He-cooled FS inboard shield is in its positive impact on the breeding, overall power balance, economics, and safety of ARIES-ST. An alternative water (or heavy water) cooled inboard shield would offer some advantages to other designs employing water as the primary coolant. The overall conclusion is that the nuclear profile is well optimized for the design constraints and requirements for ARIES-ST.

Acknowledgements

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