Nuclear Performance Assessment of ARIES-AT

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UWFDM-1180

To be published in *Fusion Engineering and Design.*

FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

MADISON WISCONSIN

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L.A. El-Guebaly and the ARIES Team

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

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Abstract

The transition to a new ARIES design always involves a significant change in the engineering system with emphasis on high performance. Compared to ARIES-RS, numerous improvements were noted in ARIES-AT, ranging from physics improvements to a focus on advanced engineering for an economically and environmentally attractive source of energy. A key engineering aspect of ARIES-AT is its compactness and the high-conversion efficiency of the LiPb/SiC blanket that is capable of performing at high temperature. Certain features of the nuclear activity were focused on areas unique to ARIES-AT, including breeding potential of the LiPb/SiC blanket containing tungsten stabilizing shells, high-performance shielding components to protect the high-temperature superconducting magnet, and compact radial builds to minimize the volume of solid waste requiring geological burial. During the study, we closely monitored the key nuclear parameters and called for measures to enhance the engineering and physics aspects of the design. Optimization of components' constituents, characterization of the radiation environment, and meeting the ARIES-AT specific design needs were also given considerable attention.

1. Introduction

Promising advances in tokamak physics and fusion technology (derived from national and worldwide developmental and experimental programs) enhanced the performance of the most recent ARIES design and ensured its competitiveness with other power generation systems. The 1000 MWe ARIES-AT advanced tokamak design [1] is safer, economical, more compact, and therefore generates less radioactive waste compared to predecessor designs. As Fig. 1 indicates, the internals of ARIES-AT are the standard tokamak components, namely the blanket, shield, vacuum vessel, and magnets. The design is built on the ARIES-RS study, focusing on more advanced physics and technology. Several important features have been incorporated to improve the performance of ARIES-AT. For instance, the high-temperature SiC/SiC structure operates at 1000°C, the high-performance blanket achieves ~60% efficiency, the near plasma stabilizing shells allow high plasma beta exceeding 10%, the radiation-resistant high-temperature superconducting magnets offer less stringent radiation limits, and the high-performance shield results in more compact radial builds that help minimize the radwaste.

Table 1 summarizes the key design parameters and radiation limits that are essential for the nuclear assessment. The neutron wall loading profile resembles that of ARIES-RS [2] with slightly lower values mainly due to the increased thermal conversion efficiency. The distribution peaks at the outboard (OB) midplane at ~4.8 MW/m^2 and drops to ~ 1 MW/m^2 at the divertor plates. In absence of firm experimental data for advanced structural materials and high-temperature magnets, optimistic criteria determined the limits and accounted for future improvements to existing materials. This certainly represents an important issue open to debate between materials experts and designers.

During the two-year term of the study, the nuclear analysis has been a fundamental element of the design process. In addition to the continuous updates of the neutronics and shielding performance, we were actively involved in the design development and provided attractive options for a self-sufficient tritium breeding blanket, well-optimized shield, compact radial build, acceptable radiation damage to structural components, and reasonable service lifetime for in-vessel components.

Table 1. Key design parameters and radiation limits for ARIES-AT

Fusion power	1755 MW
FW location at midplane – IB, OB	3.85, 6.55 m
at top/bottom – IB, OB	3.85, 4.7 m
Neutron wall loading: Peak at IB, div., OB	3.1, 2, 4.8 MW/m^2
Average at IB, div., OB	2.2, 1, 4 MW/m^2
Machine lifetime	40 FPY
Availability	80%
SiC burnup limit	3%
FS dpa limit	200 dpa
FS reweldability limit	1 He appm
HT magnet fluence limit	10^{19} n/cm^2 (E_n> 0.1 MeV)
GFF polyimide insulator limit	10^{11} rads

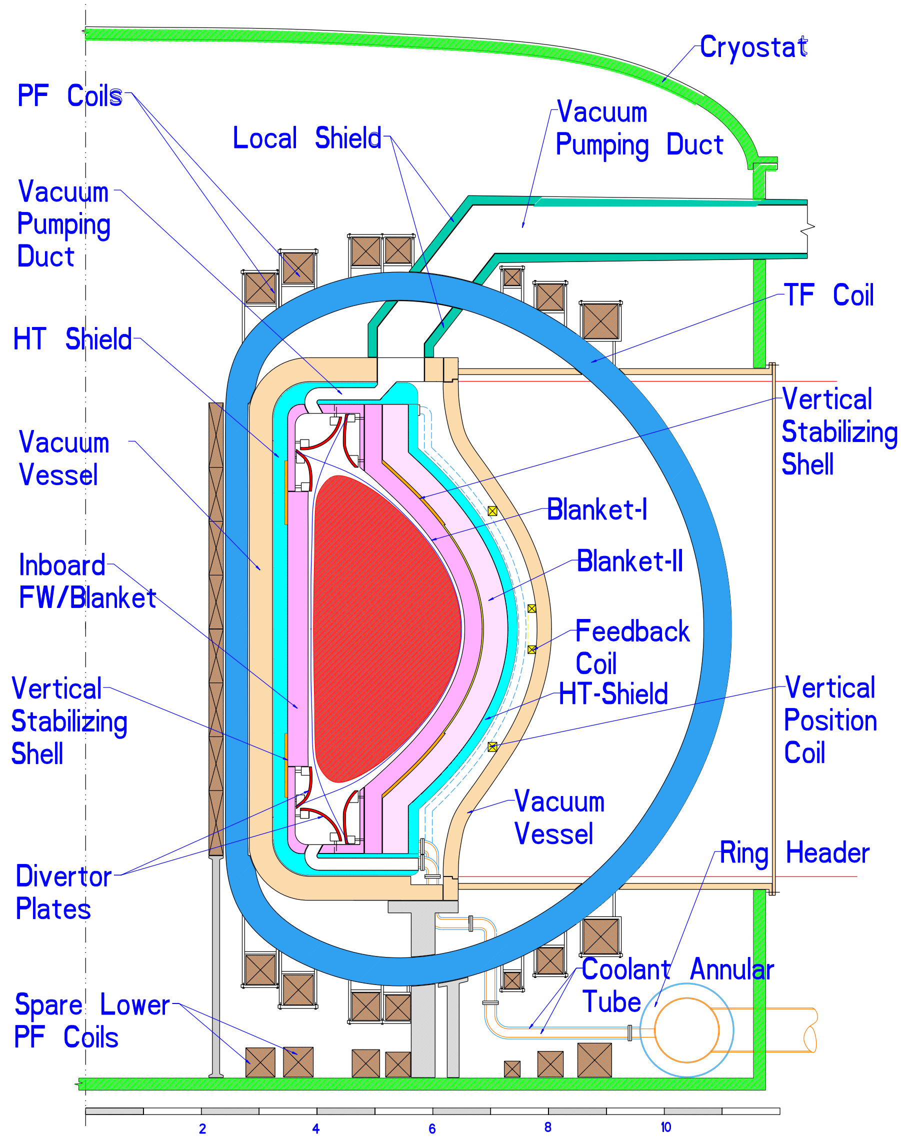

Figure 1. ARIES-AT vertical cross section.

2. Key nuclear objectives and requirements

We began the nuclear assessment by identifying a set of nuclear objectives in support of the top-level requirements developed in the mid-90s for the ARIES power plants [3]. These are summarized below:

Top-Level Requirements	Nuclear-Related Impact
Closed tritium fuel cycle	Calculated TBR $\geq$ 1.1 $\Rightarrow$ Net TBR $\geq$ 1.01
Low-level waste	Careful choice of low activation materials
Structural integrity	W-based materials in outer components only
	No hydrides in shield
Minimum waste	Well-optimized components
	Compact radial build
	Segmented blanket
Competitive COE	Efficient WC filler for IB side to reduce machine size
	SiC structure for inner HT components only
	Less expensive FS structure for outer LT components.

A tritium-breeding ratio (TBR) of 1.1 assures tritium self-sufficiency for ARIES-AT. The 10% breeding margin accounts for the uncertainties in the cross section data (~7%), approximations in geometric model (~2%), losses during T reprocessing (~1%), and T supply for future power plants (~1%) [4]. The net TBR at the beginning of plant operation may range between 1.01 and 1.2 and a flexible blanket design could adjust the net TBR to 1.01 after the first blanket changeout. In case of over-breeding (net TBR > 1.01), the TBR can be reduced by lowering the enrichment or replacing breeding modules by shielding components. In case of under-breeding (net TBR < 1.01), major changes are needed to adjust the breeding. These include replacing the W stabilizing shells by Al or Cu shells, lowering the SiC structure in the blanket, or adding beryllium to the blanket.

Addressing the nuclear issues, it was prudent to routinely check if the requirements are met when the design choices are made and whether there is still further improvement potential. We started the analysis by examining the breeding capacity of the candidate breeders, and then defined the blanket parameters (thickness, composition, and Li enrichment). Next, the shield and vacuum vessel were simultaneously designed to essentially protect the magnets. Finally, we specified the radial build and identified the key nuclear parameters: TBR, neutron energy multiplication (M_n), nuclear heat load to all components, radiation damage to structural components and their service lifetimes. The nuclear assessment proceeded interactively with guidance from the thermo-mechanical analysis.

3. Blanket system modeling and breeder selection

From its very inception, ARIES-AT selected the SiC/SiC composites [5] as the main structure for the high-temperature components as they can withstand very high temperature (~1000°C), a property of high payoff. Moreover, it is an inherently low activation material with very low decay heat and can easily satisfy the low-level waste requirement [6]. The selection criteria for the breeder include several design parameters that play an essential role in the acceptability of the breeder. These are the compatibility with the SiC structure, tritium-breeding capacity, radiation stability, safety characteristics, operating temperature window, and projected thermal efficiency for the overall system. Two liquid metal (LM) breeders and a molten salt seem to be compatible with the SiC/SiC composites. Those are $Li_{17}Pb_{83}$, $Li_{25}Sn_{75}$, and F_4Li_2Be. Each breeder has its own design issues that are addressed in detail elsewhere [7,8,9]. This nuclear assessment examines the breeding potential of the three breeders. Two blanket designs were originally proposed for ARIES-AT: a self-cooled LM concept and a dual-coolant (He and LM) concept. Further development and investigation were carried out to underpin the design choices and led to the selection of the self-cooled concept as the reference design as it outperformed the dual-cooled concept. In the reference design, the LM flows poloidally upward through the FW, returns downward in the blanket cells, and exits the blanket at ~1100°C, allowing a high thermal conversion efficiency of ~60% with the Brayton cycle [7].

Three-dimensional Monte Carlo analysis was performed using the MCNP code [10] to confirm the key nuclear parameters. Preceding the 3-D analysis, a series of parametric 1-D analysis using the DANTSYS code [11] was established to guide the design process. The 1-D estimate is based on coupling the 1-D results with the neutron coverage fraction of the blanket segments. Use is made of the 3-D results to re-normalize the neutron source for the 1-D analysis. The FENDL-2 data library [12] was employed for both analyses, point-wise data for the 3-D analysis and 175 neutron and 42 gamma group structure for the 1-D analysis with P_3-S_8 approximation. The 3-D model included the essential components comprising the power core: the first wall (FW), blanket, shield, stabilizing shells, divertor, and magnets (see Fig. 1). The innermost cell of the blanket was modeled separately as it controls the breeding level. The 10,000 source particles resulted in acceptable statistical errors of < 5% for FW local damage and heating and < 1% for overall TBR and M_n parameters. Excellent agreement was obtained between the 1-D and 3-D estimates for TBR and M_n [13].

The breeding potential of all breeders (90% enriched $Li_{17}Pb_{83}$, 90% enriched $Li_{25}Sn_{75}$, and natural F_4Li_2Be) was examined using the LiPb/SiC FW and blanket configuration [7]. The results showed that the TBR of the LiSn/SiC and FLiBe/SiC blankets remained below 0.9 and the breeding requirement (1.1) could not be met even with a fairly thick blanket. The lack of breeding has ruled out the LiSn and FLiBe breeders from further consideration. Both breeders will certainly need a beryllium multiplier to provide a TBR of 1.1. Beryllium may raise safety, economic, and resource concerns that need further investigation.

There is only one 35 cm thick blanket segment on the inboard (IB) side, whereas on the outboard side there are two cells: 30 and 45 cm thick. The outer cell is designed to be a lifetime component. The segmentation helps reduce the cumulative blanket waste and replacement cost by a factor of two. With the reference configuration and material selection (20% SiC structure in blanket, 90% enriched LiPb, 4 cm thick W vertical shells, 1 cm thick RWM shell, necessary assembly gaps between 16 modules, and 2.5 m^2 penetrations), the breeding requirement (TBR= 1.1) is achieved and the blanket approached its full breeding capacity. The IB blanket, OB blanket, and both divertor system and shield supply 25%, 70%, and 5% of the breeding, respectively. Increases in SiC content of the FW in particular, thicker stabilizing shells, and more penetrations or gaps will reduce the TBR below the 1.1 requirement. An excess breeding margin of 2-3% is still available with a thicker OB blanket. Enriching the LiPb above 90% changes the TBR by less than 1% and would not be cost effective.

4. Impact of stabilizing shells on breeding

A specific topic that has been investigated jointly by the engineers and physicists is the location of the stabilizing shells and their impact on the plasma parameters (beta, elongation, plasma current, and triangularity) and nuclear parameters (TBR, nuclear heating, activation, and decay heat). So, the stabilizing shells have been incorporated into the nuclear analysis from the beginning. We suggested moving the OB shells closer to the plasma and placing them between the two outboard blanket segments instead of behind the blanket, as defined in the previous ARIES-RS design [14]. This suggestion had a favorable impact on the plasma physics resulting in a triangularity of 2.2 and a high beta of 10% or more [15]. However, the OB shells deteriorate the breeding. We have evaluated the three candidate vertical shell materials (W, Cu and Al) and determined that a blanket without shells would yield an overall TBR of 1.16. Aluminum conductor has the least impact on breeding followed by Cu, then W as illustrated in Fig. 2. The thickness of the shell depends on the type of conductor, operating temperature, and time constant. Further optimization analysis indicated that W is the preferred material for the shells that operate hot and are passively cooled.

The toroidally continuous OB vertical stabilizing shells are placed at the upper/lower extremity of the OB blanket (z = 1.8 - 2.8 m). The local breeding in the OB blanket behind the 4 cm thick shells dropped by up to ~40%, which is a 4% reduction in the overall TBR. The 1 cm thick resistive wall mode (RWM) shell located at the OB midplane between the vertical shells drops the breeding further by ~2%. To compensate for the breeding losses, the OB blanket thickness has been adjusted to 75 cm. The final design is in accordance with these changes. No outstanding safety issues were identified for the W shells [6]. The decay heat is minimal due to the small volume of the shells and the outer blanket segment containing the shells still qualifies as Class C waste at the end of plant operation.

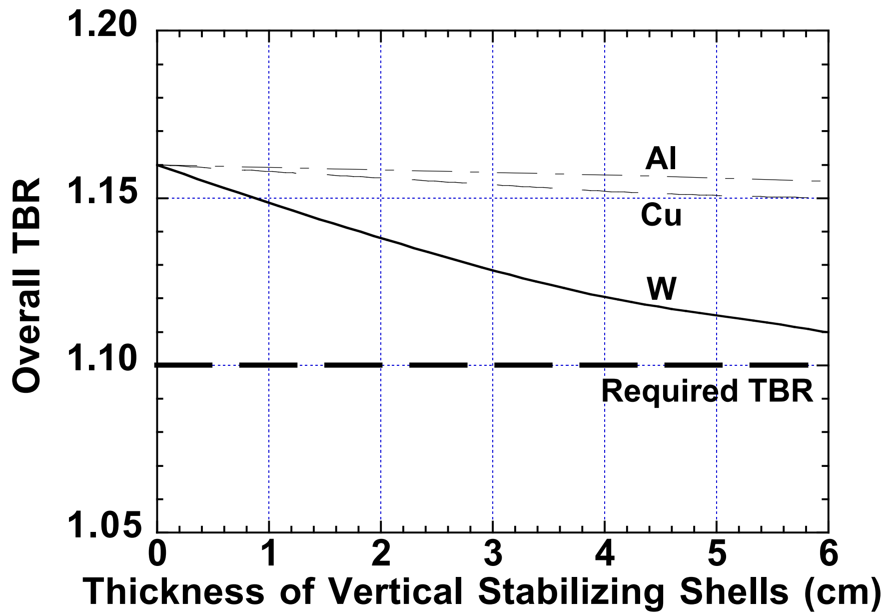

Figure 2. Degradation of TBR with thickness of vertical stabilizing shell.

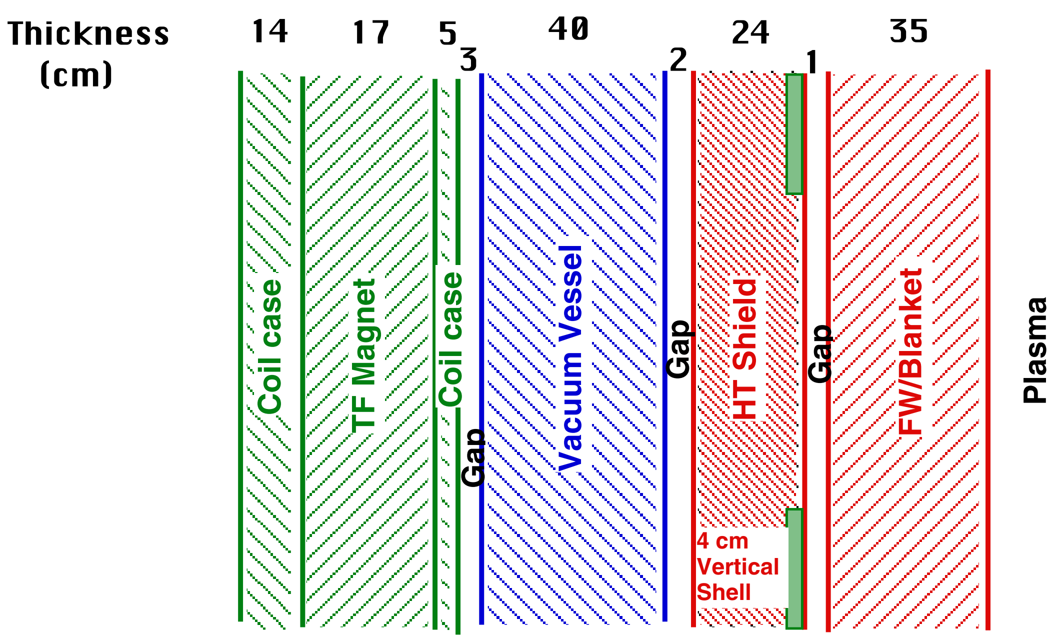

Figure 3. Inboard radial build of ARIES-AT.

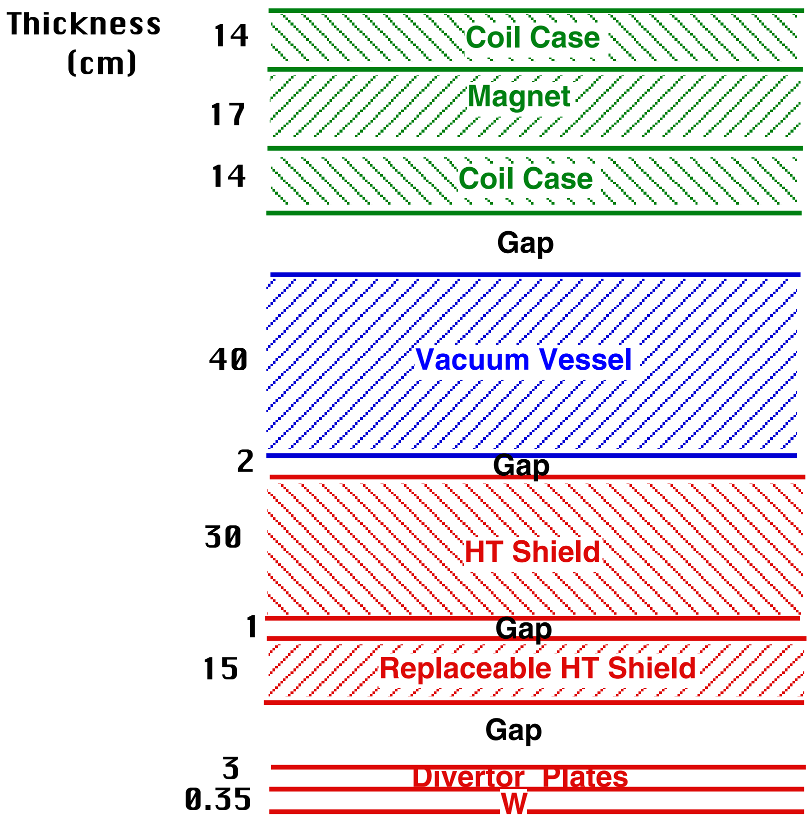

Figure 4. Vertical build of ARIES-AT.

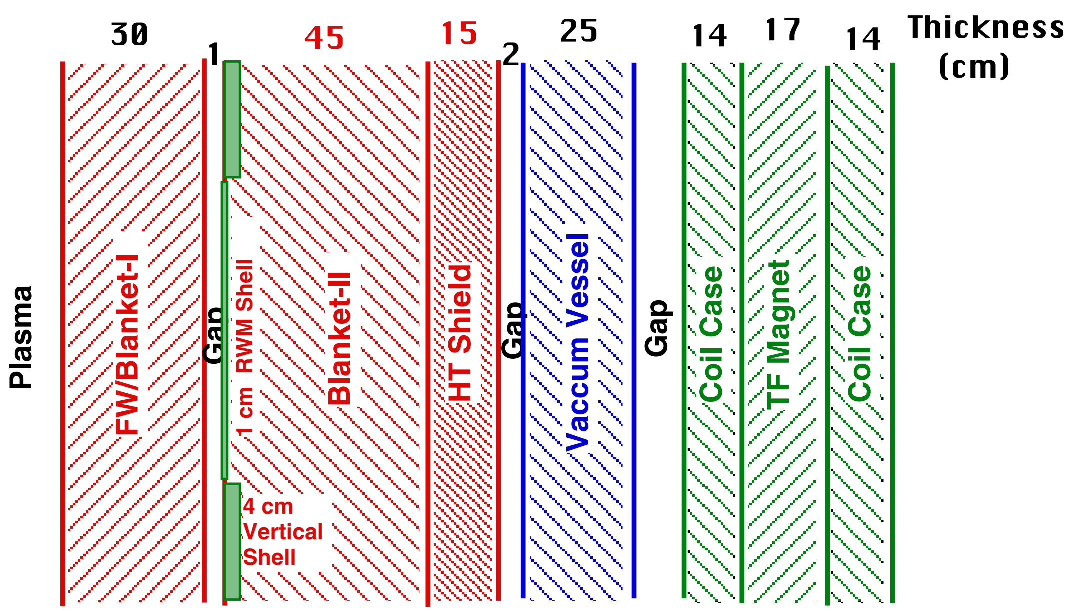

Figure 5. Outboard radial build of ARIES-AT.

5. Radial build definition

Discussed in the following sections is the in-vessel component optimization process that helped define the operational space of the machine, therefore minimizing the burden of unnecessary waste generated by non-optimized radial builds. During this process, we focused our shielding effort on the inboard area where a better shielding performance makes a notable difference to the overall machine size. We placed less emphasis on the outboard and divertor areas where the shielding space is not constrained and no economic and design enhancements are gained with high-performance, compact shields.

The reference radial builds for the inboard, divertor, and outboard regions are shown in Figs. 3, 4, and 5, respectively, highlighting the main features of the blanket, shield, vacuum vessel, and TF magnet. The water-cooled vacuum vessel (VV) is a low-temperature (LT) component and the LiPb-cooled shield is a high-temperature (HT) component. The gap between the VV and magnet contains 15% thermal superinsulation. An interim design had a thin LT shielding component located between the HT shield and a thin VV, analogous to ARIES-RS [4]. With close collaboration with the VV designers, we suggested combining the LT shield and VV in a single unit due to common materials, similar lifetimes, and structural connections. This would also simplify the plumbing connections and increase the structural integrity of the VV. The only problem is that if there is a need to cut and weld the VV during the plant life, the welds must be protected from radiation.

6. Magnet protection requirements

We revisited the magnet radiation limits as they strongly influence the compositions and size of the shielding components. All in-vessel components (blanket, shield, and VV) provide a shielding function for the magnets. As discussed later, the composition of the outermost component (VV) is driven partly by magnet shielding considerations. The selected conductor material for both the TF and PF coil sets [16] is the YBCO because of the low material unit cost and inexpensive construction technique ($50/kg). The liquid nitrogen-cooled coils are dominated by the Inconel structure (72%) and can operate at high fields (> 16 T) without a quench requirement. The cooling requirement for the YBCO HT magnet is greatly relaxed compared to the liquid helium-cooled magnets of ARIES-RS because of the higher operating temperature (70-80 k). This means the nuclear heat limit to the HT magnets could be quite high and the cryogenic heat load is not a concern [17]. The limit for the GFF polyimide insulator is 10^{11} rads [2]. Therefore, the predominant radiation limit is the fast neutron fluence to the YBCO conductor. With the limited irradiation data presently available, the magnet designers predict the radiation limit for the YBCO conductor to be comparable to the limit for the Nb_3Sn conductor, that is, 10^{19} n/cm^2 for $E_n > 0.1$ MeV [17].

7. Vacuum vessel design and trade study

Being the closest component to the magnet, the VV composition greatly influences the magnet radiation effect. The skeleton of the double-walled VV with internal ribs provided by the VV designers [18] was filled with shielding materials and optimized to achieve the necessary requirements for magnet shielding. Tradeoff analyses of the water and filler materials were conducted for the VV. We had two candidates for consideration: a high-performance design using tungsten carbide (WC) filler and a more typical borated ferritic steel (B-FS; FS with 3 wt% B) design. The WC-based inboard shield and VV is 15 cm thinner than a B-FS-based design. An optimal combination for the VV would be to use WC for the inboard to reduce the size of the overall machine, B-FS for the divertor, and only water for the outboard to simplify the VV design. The IB/divertor and

Table 2. Peak radiation effects at TF magnet

	IB	OB	Divertor	Limit
Fast neutron fluence (E_n > 0.1 MeV) (n/cm^2 @ 40 FPY)	9×10^{18}	1×10^{19}	9×10^{18}	10^{19}
Nuclear heating (mW/cm^3)	0.2	1.7	0.2	--
Dose to GFF polyimide (rads @ 40 FPY)	7×10^{9}	3×10^{10}	6×10^{9}	10^{11}

OB sections of the VV call for 13% and 30% FS structure, respectively, dictated by the structural requirements of the VV [18]. Figures 6 and 7 demonstrate the VV optimization curves, showing the tradeoff between water and filler. About 22% water is optimal for the IB VV section, driven by shielding considerations. For simplicity, the same water content is used in the divertor section. Even though the OB VV section optimizes at 50-70% water (see Fig. 7), a decision was made to eliminate the filler to simplify the OB VV design. The 70% water content results in notably higher magnet heating and VV activity [6]. It is noted that the elimination of the B-FS filler has increased the VV neutron flux with E < 1 keV by several orders of magnitude.

8. Rationale for shield design selection

As an essential element of the power core, the shield provides radiation protection for the externals and meets the power production requirements. The 12% nuclear heating carried out by the shield is recovered to enhance the power balance and economics. Therefore, the shield operates hot and employs 15% LiPb for cooling and B-FS as filler. Substitution of the B-FS with WC would have decreased the IB radial build by ~5 cm but generates higher decay heat by an order of magnitude. Correspondingly, the temperature rise during an accident is also higher by one or two orders of magnitude, depending on the time period after the onset of the accident. Therefore, we avoided the use of WC filler in the shield for safety reasons and recommended the use of B-FS filler instead. Note that the innermost component of the divertor shield employs FS filler to control the tritium inventory. Overall, the shield and VV design choices help improve the economics and reduce the radiation damage at the magnets below the allowable radiation limits. Table 2 provides the radiation level at the TF magnet for the base case design. In all 1-D shielding calculations, we have considered a safety factor of three to account for uncertainties in the computing tools and design elements. The main features of the shielding system could be summarized as follows:

- HT and LT components to reduce cost [2]
- Nuclear heating recovered (~12%) to improve power balance
- SiC structure and LiPb coolant for HT shield
- Less expensive FS structure with water coolant for LT VV
- Fillers with SiC and FS structures to improve shielding and reduce cost
- Highly efficient WC fillers for IB section of VV to reduce machine size
- Water coolant for VV for better shielding performance.

The optimum compositions for all components are listed in Table 3. The SiC-based wedges underneath the outer legs of the TF magnets should match the optimized composition of the blanket and shield to provide the necessary protection for the coils. Along with the blanket/shield/VV, the 5 cm thick port enclosures and the 5 cm thick side coil cases adequately protect the sides of the outer legs of the TF coils.

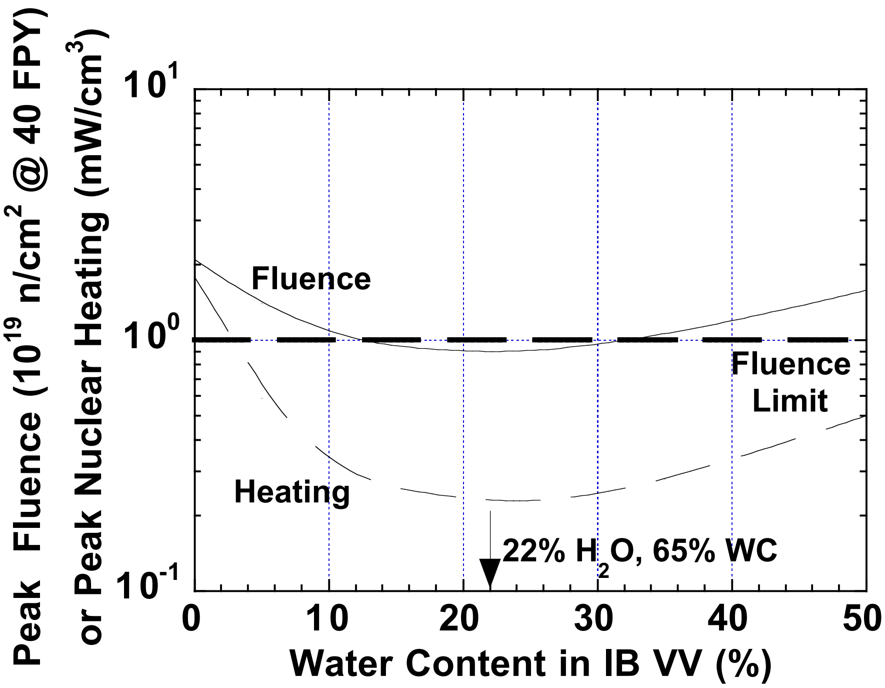

Figure 6. Variation of radiation effects at TF coil with compositions of IB section of W, substituting WC for water.

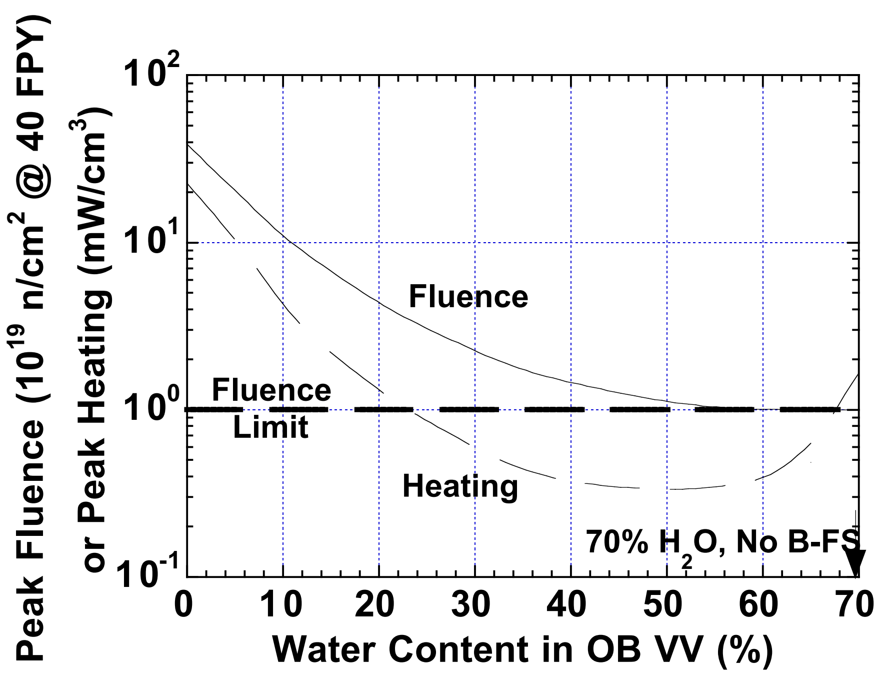

Figure 7. Impact of VV composition on radiation effects of outer legs of TF coils, substituting B-FS for water.

Table 3. Composition* of ARIES-AT components (in volume %)

	Inboard	Divertor	Outboard
Divertor plates		40% SiC, 50% LiPb,	
		10% W	
FW/Blanket-I	19% SiC, 81% LiPb		20% SiC, 80% LiPb
Blanket-II			19% SiC, 77% LiPb,
			3% W
Replaceable HT		15% SiC, 10% LiPb,	
shield		75% FS	
HT shield	15% SiC, 10% LiPb,	15% SiC, 10% LiPb,	15% SiC, 10% LiPb,
	70% B-FS, 5% W	75% B-FS	75% B-FS
Vacuum vessel	13% FS, 22% H_2O,	13% FS, 22% H_2O,	30% FS, 70% H_2O
	65% WC	65% B-FS	
Coil case	95% 304-SS, 5% LN		
TF coil	72% Inconel-625, 0.5% Ag, 7% $YBa_2Cu_3O_5$, 7% CeO_2 insulator,		
	13.5% GFF polyimide insulator		

* SiC, W, and WC are 95% dense

1.4 cm thick FW contains 73% SiC and 27% LiPb

Table 4. Radial dimensions (in cm) of ARIES-AT components compared to ARIES-RS

	ARIES-AT			ARIES-RS		
	IB	Div.	OB	IB	Div.	OB
Divertor plates		3.35			5	
FW/Blanket-I	35		30	20.3		20.3
Gap	1	> 5	1		1	1
Blanket-II			45			30
Replaceable HT shield		15		20	20	7
Gap		1		1	1	1
HT shield	24	30	15	26	35	28
Gap	2	2	2	2	2	2
LT shield				28	45	40
Gap				2	> 2	> 2
Vacuum vessel	40	40	25	20	20	30
Subtotal*	102	96	118	120	131	161
Gap	3	> 5	> 5	5	> 5	> 5
Inner coil case	5	14	14	3	3	3
Coil	17	17	17	50	50	50
Outer coil case	14	14	14	5	5	5
Total*	141	146	168	183	194	224

* Minimum gap width considered

Table 5. Nuclear heat load (in MW) to ARIES-AT components

	Inboard	Divertor	Outboard	Total
FW and Divertor	39	43	96	178
Blanket:				1207
Blanket-I	302	---	727	
Blanket-II	---	---	178	
HT shield	40	112	9	161
Total	381	155	1010	1546

With the higher triangularity, the IB divertor slot is virtually eliminated, impacting the design of the IB divertor region and the shield behind. More accurate definition of the headers and plumbing in the divertor region helped estimate the damage to the magnet where the IB shield recesses to accommodate the inner divertor plate (see Fig. 1). Our analysis showed no excessive damage or shielding problems behind the inner divertor plates.

A comparison between ARIES-AT and ARIES-RS [2,14] showed an appreciable reduction of 18-43 cm in the dimension of the in-vessel components and an even higher reduction of 42-56 cm when the magnet is included. The superior shielding performance of the LiPb breeder compared to Li, the ability to use water in the VV, and the thinner HT magnet contributed to the compactness of ARIES-AT. Table 4 tabulates the radial dimensions of the components for both designs. The alloying and impurity elements of the ARIES-AT materials are documented in Reference 19.

9. Nuclear heat load to in-vessel components

The power production components include the FW, blanket, shield, and divertor. Table 5 details the breakdown of the nuclear heating deposited in the in-vessel components. As the table indicates, most of the power (88%) goes to the FW, divertor, and blanket. The small heat leakage to the VV (< 1%) is dumped as a low-grade heat. The heat load to the TF magnets is ~50 kW, corresponding to a LN cryogenic load of ~0.5 MW. The IB, divertor, and OB regions generate 25%, 10%, and 65% of the total heating, respectively. For a neutron power of 1404 MW (80% of the fusion power), the neutron energy multiplication amounts to 1.1. The total heat deposition along with the power density distribution shown in Fig. 8 served as a source term for the detailed thermal analysis carried out for the ARIES-AT design [7].

10. Lifetime assessment

The ability to identify the life-limiting criteria for the structural materials is a key factor to determine accurately the service lifetime of the in-vessel components. Historically, the thermal and mechanical stresses, thermal creep, burnup of atoms, atomic displacement, and activation products have led to either a failure mechanism or a violation of the waste requirement, therefore prematurely ending the service lifetime of structural components. There are no firm guidelines for the SiC/SiC composites [5] while for the FS components, the life-limiting criterion has traditionally been the displacement of atoms, ranging between 100 and 200 dpa. In the ARIES study, we have adopted the limits of 3% burnup for the SiC structure and 200 dpa for the FS structure [2].

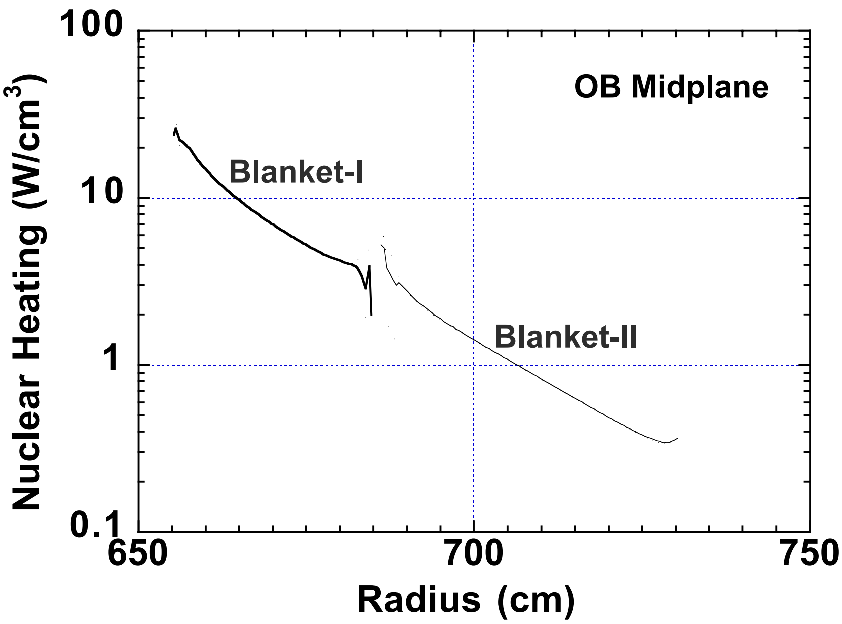

Figure 8. Radial distribution of nuclear power density in OB blanket.

Unlike metals, the helium production in SiC is excessive. Another unique feature for SiC is the high He to H ratio of 2-3. Helium could have an adverse effect on the thermal conductivity and mechanical and dimensional stability of the SiC structure. Under ARIES-AT operation conditions, the peak damage to SiC reaches 4200 He appm/FPY, 1700 H appm/FPY and 56 dpa/FPY at the midplane of the OB first wall. Energetic neutrons having E > 3 MeV transmute the Si and C via (*n,* He) and (*n*, H) reactions into Al, Mg, Li and Be. Each (*n*, He) or (*n,* H) reaction with either a Si or C atom could potentially burn a SiC molecule. Our results show that Si atoms burn faster than C atoms at a 2:1 ratio. For a 3% burnup limit, the FW/blanket lifetime is ~ 4 FPY, implying an end-of-life fluence of 18.5 MWy/m^2 for the SiC structure. As Fig. 9 indicates, the burnup rate drops rapidly within the LiPb blanket due to the slowdown and interaction of neutrons with Pb generating lower energy neutrons with energy below the 3 MeV threshold for He and H production. Therefore, we segmented the OB blanket into two parts: a 30 cm thick replaceable FW/Blanket-I with a limited lifetime of 4 FPY and a permanent 45 cm thick Blanket-II that could last for the entire 40 FPY plant life. This segmentation has lowered the cumulative blanket waste by a factor of two and helped reduce the annual replacement cost.

There is a class of low activation, radiation-resistant FS alloys that offer low neutron-induced swelling, low thermal expansion coefficient, high resistance to irradiation creep, and high range of operating temperatures. An example of advanced FS suitable for fusion applications includes MHT-9, F82H, MANET, ORNL-9Cr-2WVTa, and nanocomposited FS. The reference FS for this study is the ORNL-9Cr-2WVTa [20]. The more advanced nanocomposited FS [21] will be considered in future studies. The only FS-based component in ARIES-AT is the vacuum vessel, operating at a nominal temperature of 100°C. Design measures have been incorporated in ARIES-AT to assure that the maximum temperature during the worst-case accident does not exceed the 700°C limit, permitting the reusability of the VV following an accident [22]. Regarding the radiation damage, the 200 dpa limit is met at the innermost surface of the VV at the 40 FPY end-of-plant life. However, the FS structure will not survive the high radiation environment if there is a need to cut and reweld the VV

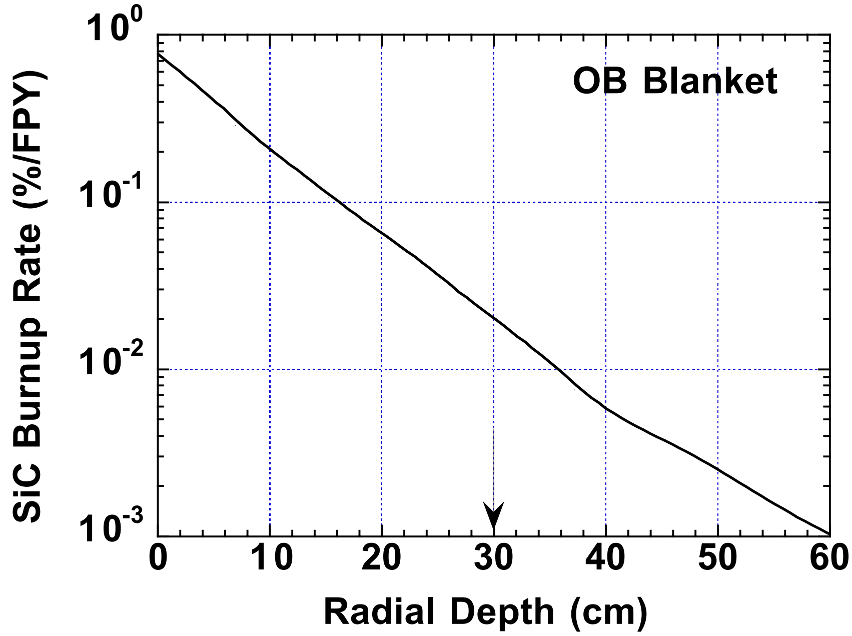

Figure 9. Radial variation of SiC burnup within the OB blanket reaching the 3% limit at the end of plant life (40 FPY) at a depth of 30 cm.

during plant operation. For instance, the helium production level at the inner surfaces of the IB and divertor sections of the VV is excessive (~15 He appm @ 40 FPY) and exceeds the project-assumed reweldability limit of 1 He appm for FS. Recently, experimental data suggest higher limits for 316-SS ranging from 5 to 30 He appm [23]. Considering today's knowledge of the FS irradiated material properties, the VV welds should be hidden and located away from the high-radiation zones. We recommended a recessed weld joint design that offers sufficient protection if the VV joints need to be rewelded within the nominal service life. Alternatively, the VV could be replaced with a new component in case of a failure. Given the projected low cost of the VV [18], the replacement would not be a prohibitive expense but add to the waste stream.

11. Waste minimization

Designing compact radial builds has been recognized as a means to reduce the volume of radioactive waste. Other options for waste reduction include recycling and clearing of materials and components. In advanced designs, only the confinement concrete building would be cleared from regulatory control, not the in-vessel and magnet components [24]. With proper optimization, ARIES-AT has achieved a substantial reduction in the size of the components that contribute to the total volume of radioactive waste. Shown in Fig. 10 is the evolution of the waste volume since the inception of the ARIES designs in the early 1990s. The waste volume belongs to the fusion power core components: blanket, shield, divertor, VV, and magnets. Design measures such as high-power density machines, well-optimized radial builds, extended service lifetimes, and segmented blankets have all contributed to the persistent trend in waste minimization. During the 10 year period of the ARIES studies, a factor of two reduction in waste has been achieved, which is impressive.

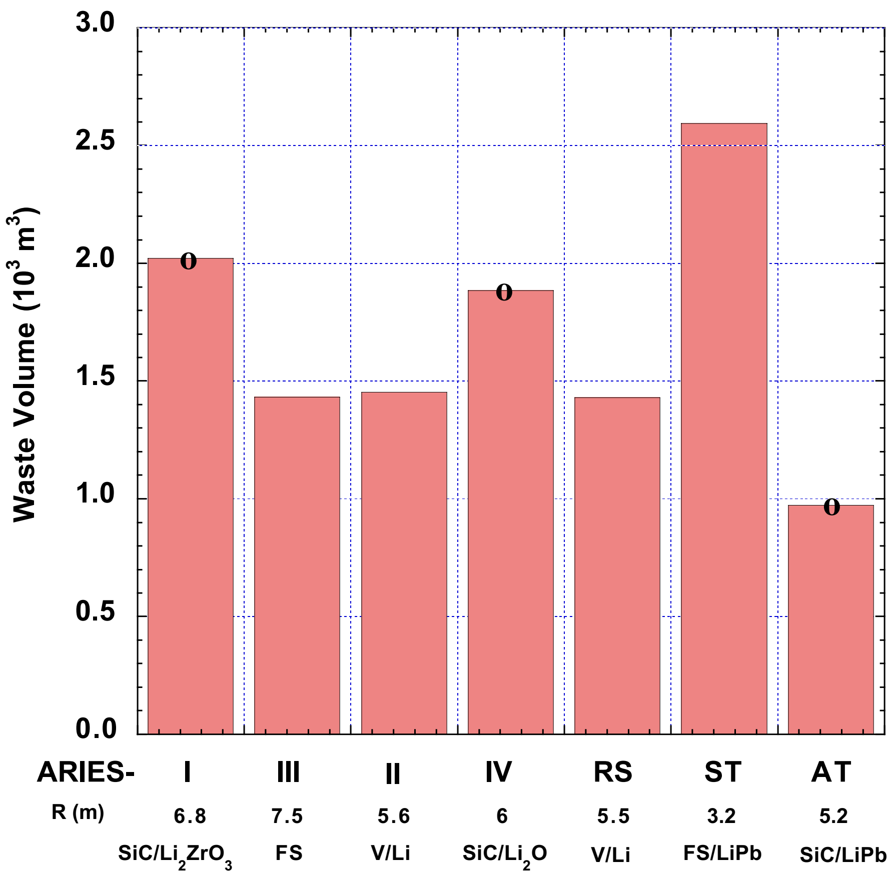

Figure 10. Evolution of ARIES waste showing a persistent trend in volume reduction for tokamaks. SiC designs are marked with small circles.

12. Conclusions

The integrated nuclear activity guided the ARIES-AT design during the design process and interactions with subsystem designers improved the nuclear integration. Most of the nuclear activity was aimed at achieving the desired requirements by focusing effort on the blanket and shielding systems while carefully monitoring the economic and safety impacts of the material choices. A number of breeders were initially considered for ARIES-AT: LiPb, LiSn, and FLiBe. As the design began to develop, it was soon realized that LiSn and FLiBe provide insufficient tritium breeding for the SiC-based system. The reference LiPb/SiC configuration satisfies the minimum breeding requirements (TBR = 1.1). As the stabilizing shells were added to the blanket, the excess breeding margin diminished considerably.

All three major components (blanket, shield, and vacuum vessel) provide a shielding function and protect the magnets against radiation. Because of the growing interest in minimizing the volume of waste generated by fusion machines, enhanced shielding features are invoked in this design to avoid a large radial standoff between the plasma and magnet. As a result, ARIES-AT generates the least amount of waste among all ARIES-like tokamaks.

Regarding the radiation damage and service lifetimes, all components except the innermost FW/blanket, divertor system, and front divertor shield are permanent life-of-plant components. The life-limited components need replacement every 4 FPY. The radiation level at the magnet is below the limit. Overall, the material choices for the internals of ARIES-AT satisfy the desired performance requirements and all components are well optimized to achieve the design constraints.

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References

[1] F. Najmabadi et al., Overview of the ARIES-AT advanced tokamak, advanced technology power plant study, Fus. Eng. and Des., this issue.

[2] L.A. El-Guebaly, Overview of ARIES-RS neutronics and radiation shielding: key issues and main conclusions, Fus. Eng. and Des., 38 (1997) 139-158.

[3] F. Najmabadi et al., The STARLITE study: assessment of options for tokamak power plants, University of California-San Diego, UCSD-ENG-005 (1997).

[4] L. El-Guebaly, Tritium breeding issues for MFE and IFE, Presentation given at tritium workshop held on March 6-7, 2001 at Livermore, California, http://fti.neep.wisc.edu/FTI/ARIES/MAR2001/tritium_lae.pdf

[5] A.R. Raffray, R. Jones, G. Aiello et al., Design and material issues for high performance SiC/SiC-based fusion power cores, Fus. Eng. Des. 55 (2001) 55-95.

[6] D. Henderson, L. El-Guebaly, P. Wilson, and A. Abdou, Activation, decay heat, and waste disposal analysis for ARIES-AT power plant, Fusion Technology, 39, No. 2, Part 2 (March 2001) 444-448.

[7] R. Raffray et al., Advanced power core system for ARIES-AT, Fus. Eng. Des., this issue.

[8] D.K. Sze, R. Mattas, Z. Wang et al., Sn-Li, a new coolant/breeding material for fusion applications, Argonne National Laboratory, ANL/TD/CP-98464 (1999).

[9] D.K. Sze, FLiBe assessment, Argonne National Laboratory, ANL/TD/CP-103408 (2000).

[10] MCNP – A general Monte Carlo code for neutron and photon transport, Version 4B, Los Alamos National Laboratory Report LA-12625 (1993).

[11] DANTSYS: A Diffusion Accelerated Neutral Particle Transport Code System, Los Alamos National Laboratory, LA-12969-M (June 1995).

[12] M. Herman and H. Wienke, FENDL/MG-2.0 and FENDL/MC-2.0, the processed cross-section libraries for neutron-photon transport calculations, Report IAEA-NDS-176, International Atomic Energy Agency (March 1997).

[13] L. El-Guebaly, Nuclear issues and analysis for ARIES spherical and advanced tokamaks, Fus. Eng. Des. 51-52 (2000) 325-330.

[14] F. Najmabadi et al., Overview of ARIES-RS reversed shear power plant study, Fus. Eng. Des. 38 (1997) 3-25.

[15] C. Kessel et al., Plasma equilibrium and MHD stability analysis for ARIES-AT, Fus. Eng. Des., this issue.

[16] T. Brown et al., ARIES-AT magnet system, Fus. Eng. Des., this issue.

[17] L. Bromberg, M. Tekula, L. El-Guebaly, and R. Miller, Options for the use of high temperature superconductor in tokamak fusion power plant, Fus. Eng. Des. 54 (2001) 167-180.

[18] L.A. Waganer, L. El-Guebaly, and X. Wang, Vacuum vessel design approach and cost assessment, Fus. Eng. Des., this issue.

[19] D. Petti, B. Merrill, R. Moore et al., ARIES-AT safety design and analysis, Fus. Eng. Des., this issue.

[20] R. Klueh, M.L. Grossbeck, and E.E. Bloom, Impurity content of reduced-activation ferritic steels and vanadium alloy, Fusion Materials Semiannual Progress Report for Period Ending December 31, 1996, U.S. Department of Energy Office of Fusion Energy Sciences, DOE/ER-0313/21 (April 1997).

[21] G.R. Odette and D.T. Hoelzer, Development of nanocomposited ferritic alloys for high performance fusion first wall and blanket structures, article submitted to ANS FED newsletter, http://fti.neep.wisc.edu/FTI/FED/news0602.pdf (June 2002).

[22] E. Mogahed, L. El-Guebaly, A. Abdou, P. Wilson, and D. Henderson, Loss of coolant and loss of flow accident analyses for ARIES-AT power plant, Fusion Technology, 39, No. 2 (2001) 462.

[23] W. Daenner, Max-Planck-Institut für Plasmaphysik, Garching, Germany, private communications (March 2000).

[24] L. El-Guebaly, D. Henderson, A. Abdou, and P. Wilson, Clearance issues for advanced fusion power plants, Fusion Technology, 39, No. 2 (2001) 986-990.