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Overview of ARIES-RS neutronics and radiation shielding: key issues and main conclusions

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Abstract

The neutronics and radiation shielding issues were assessed for the ARIES-RS power plant. The assessment addresses the breeding level, service lifetime, radiation damage, shielding requirements and design, and personnel protection. Major efforts were devoted to fulfil the top-level requirements which include the demonstration of a closed tritium fuel cycle, generation of no radioactive waste greater than Class C, and production of electricity at a competitive cost. The material optimization was one of the themes of this study as it influences the neutronics results and helps meet the requirements while minimizing cost. An important outcome of the neutronics and shielding analyses is the specification of the radial builds that contain key component parameters in terms of sizes and optimal compositions. The shielding system comprises a major element of the fusion power core (FPC). The primary function of the shield is radiation protection: protection of superconducting magnets, vacuum vessel, workers and the public. As an element of the power core, the bulk shield meets other requirements for power production and service lifetime. In addition, it serves as a heat sink for the FW/blanket decay heat during a loss-of-coolant accident. All shielding elements are integrated with the remainder of the FPC to meet the assembly, mechanical support and attachments, and maintenance requirements. The system requirements developed for the shield stem from these essential shielding functions. The work reported herein illustrates the strong impact of numerous factors (such as service lifetime, blanket segmentation, shield optimization, radial builds, and unit costs of materials) on the economics of power production.

Keywords: ARIES-RS; Fusion power core; Radiation shielding

1. Introduction

The ARIES-RS project has evaluated several candidate engineering design options and finally adopted features of the engineering systems of the ARIES-II design [1] as a starting point [2]. Thus, the ARIES-RS study is built upon the ARIES-II design but addresses the engineering issues in greater detail. An extensive assessment of improvements to the ARIES-II in-vessel components has been performed along with the economic impact of the various options on the overall cost of the machines. The safety and economic requirements were constantly factored into the assessment to ensure that the most desirable safety features were integrated in the design in a cost-effective manner. The safety requirements severely limit the material choices for all in-vessel compo-

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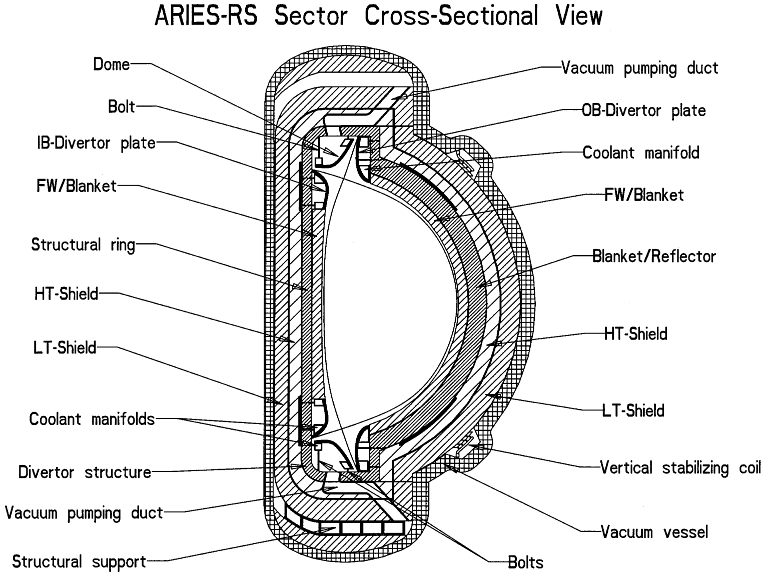

Fig. 1. Vertical cut through ARIES-RS showing the latest divertor configuration.

nents, while economics-based performance requirements provide strong incentives to operate at high coolant temperature and thermal efficiency. The economic requirements constrain many aspects of the design and urge designers to optimize the performance of all in-vessel components, employ low cost materials, and extend the useful life of all components as much as practically possible.

The neutronics analyses performed for the individual components are summarized here with brief descriptions of the subsystems. Detailed technical information on each component can be found in the accompanying papers for the individual components. The shielding analysis forms the basis for the shield design, using material optimization as one of the central themes of the study. Several shielding components of ARIES-II were modified to improve the performance and economics. Section 6 highlights the improvements made to the bulk shield, the rationale for making the changes, and the design optimization activities that enhanced the performance and reduced the cost of the shield. It also highlights the ARIES-RS safety issues that were solved by design improvements and the constraints on the design imposed by the maintenance scheme.

The neutronics and shielding calculations reported herein were performed using the three-dimensional (3-D) MCNP code [3] and the discrete-ordinates DANTSYS code [4] with the P_3-S_8 approximation and the FENDL-1 cross-section library [5] in the 46n-21g group structure. A close attention was paid to the evolution of the nuclear cross section data library. The FENDL-1 library, used to evaluate the ARIES-RS study, was updated and improved through an international process [5] and was checked against existing experimental data on tritium production, nuclear heating, and induced activation. When compared to the US ENDF/B-V library [6,7], used in the ARIES-II design, important differences were found between the two libraries.

The 1-D model is a toroidal cylindrical geometry where the inboard and outboard sides are modeled simultaneously to simulate properly the neutron reflection and spectral effects between the two sides. The model includes all components comprising the fusion power core (see Fig. 1). Inside the bore of the TF magnets is the vacuum vessel (V.V.) which surrounds the internal, removable components; these include the first wall (FW), blanket, reflector, shield and divertor. The FW/blanket is the innermost component of the

ARIES-RS machine. Its functions are to breed most of the tritium necessary for plasma fueling, to remove the surface heat flux and the volumetric nuclear heating, to protect the shield against radiation for the entire plant life, and, along with the shield and V.V., to protect the magnets from excessive radiation damage. The shield, V.V., and magnets are lifetime components designed to perform properly for ~ 50 years. As indicated later, the FW/blanket has limited life of ~ 3 years determined by the attainable radiation damage to the structure. Some in-vessel components have an intermediate lifetime of ~ 10 years. The internals are divided toroidally into 16 sectors. One sector of the outboard contains the penetrations needed for the auxiliary RF heating and current drive systems. The entire torus is housed in a cryostat surrounded by buildings containing the necessary support systems. The torus and its ancilliary system within the bioshield are maintained with robotic and remote handling systems. Personnel access into the building is not feasible at any time after shutdown.

2. Neutron wall loading distribution

An essential element in the neutronics calculations is the poloidal distribution of the neutron wall loading (Γ). The distribution provides the peak values which are used to size the shield, determine components' lifetimes, estimate damage levels, and assess the radiation environment around the torus. Furthermore, an accurate estimate for the wall loading at the critical shielding region behind the inner divertor plate where the shield recesses to accommodate the divertor system is essential to warrant adequate protection for the inner legs of the TF magnets. The 3-D MCNP code was used to accurately determine the wall loading distribution over the FW and the complex divertor surface.

The divertor shape, particularly the inner plate, was developed iteratively with guidance from the neutron wall loading analysis. The old divertor design had an almost flat inner surface that faces a large portion of the plasma and thus was subject to a high neutron wall loading. It was thus recommended to curve the inner plate outward in the final design to help protect the surface of the recessed shield against source neutrons. The two divertor shapes were modeled for the MCNP code to quantify the drop in wall loading for the curved inner plate.

Fig. 1 shows a vertical cut through the FPC, dis playing the latest divertor configuration where the inner plate is curved outward and the outer plate is curved inward. Only the upper half of the machine was modeled for the MCNP code due to symmetry. A combination of cones, tori, and cylinders was used to accurately model the first wall. The FW was divided into 40 segments and particles crossing each segment were tallied. The 14 MeV neutrons were sampled from the plasma zone according to the source distribution provided by the ARIES Systems Code (ASC) [8]. The direction of the source neutons was sampled from an isotropic angular distribution. One million particles were sampled yielding a statistical uncertainty less than 1% at any FW segment.

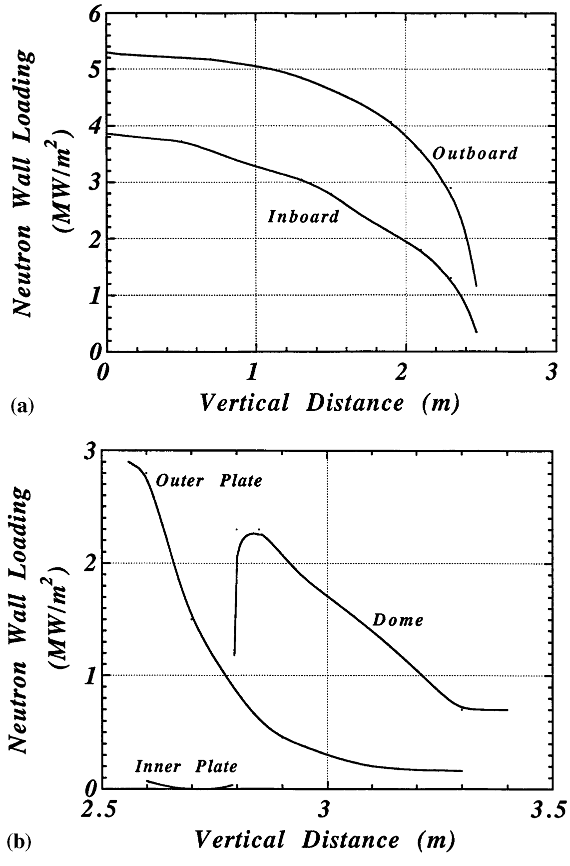

Fig. 2. Poloidal variation of neutron wall loading.

The MCNP results are plotted in Fig. 2. The machine average neutron wall loading is 3.1 MW m^{-2} for the case where the actual shape of the divertor is included in the FW area. Other parameters of interest are the average Γ over the inboard (i/b) FW, outboard (o/b) FW, and divertor surface. Those are 2.9, 4.5 and 1.1 MW m^{-2}, respectively. The wall loadings at the innermost surface of the divertor is 0.5 MW m^{-2} for the flat inner divertor plate and 0.035 MW m^{-2} for the curved inner divertor plate. As expected, the curved surface provides better protection for the shield surface against source neutrons.

3. Tritium self-sufficiency

The top-level requirements state that the ARIES-RS design must demonstrate a closed tritium fuel cycle. This means the in-vessel components should supply all the tritium needed for the ARIES-RS operation (0.33 kg day^{-1}). An external tritium supply is only needed at the start of operation and for a short period of time until steady-state production of tritium is reached. The tritium breeding ratio (TBR) in a self-sustained power plant must exceed unity by a small margin. This breeding margin is necessary mainly to supply the inventory for startup of other fusion reactors, to maintain the equilibrium holdup inventory, to provide adequate reserve storage inventory, and to compensate for the radioactive decay of T between production and use. A serious effort was developed to lower the breeding requirements by increasing T burnup in the plasma (∼ 30%), ensuring extremely low T losses, achieving low T inventory in all subsystems, and requiring high reliability and short repair time for the T processing systems [1]. It appears that, with existing technology, a system with negligible T leakage can be achieved to guarantee a safe and reliable design.

In addition to the breeding margin, a provision should be made in the calculated TBR to account for uncertainties due to approximations and/or errors in the various elements of the calculations such as nuclear data, calculational method, and geometric representation [1]. It was found that the breeding margin amounts to only 0.01 for the ARIES-RS plasma parameters and T processing system whereas the largest source of uncertainty in the calculated TBR is that for the basic nuclear data (∼ 6%) [9–11] and the geometrical approximations associated with the calculational model (∼ 3%). These uncertainties constitute a large portion (0.09) of the excess breeding. In this regard, the ARIES-RS design will attain fuel self-sufficiency if the calculated TBR is equal to or exceeds 1.1. It is pertinent to mention that the TBR will be calculated for the reference design and the effects of the elements that degrade the breeding, such as penetrations, assembly gaps, and side walls, will all be included in the computed TBR.

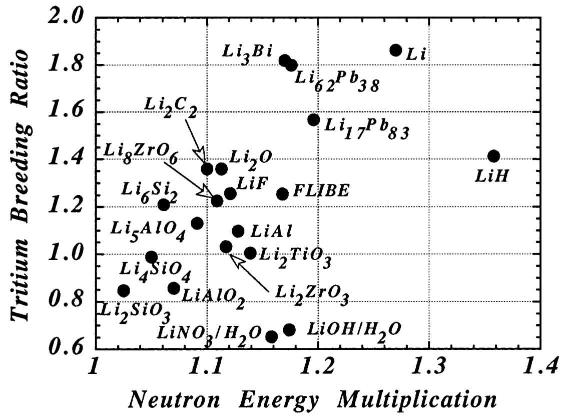

Fig. 3. Tritium breeding ratio versus neutron energy multiplication for candidate breeders (100% dense at room temperature).

Unlike solid breeders, the lithium breeder can potentially achieve tritium self-sufficiency without a neutron multiplier. A comparative 1-D analysis was conducted at the beginning of the study to compare the breeding potential of the lithium breeder relative to others. All elements that impact the neutronics performance of the breeding zone (such as the size, neutron multipliers, structural materials, etc) were excluded from the analysis. As illustrated in Fig. 3, lithium provides the highest breeding among all breeders. Further analysis has indicated that vanadium structure has the least impact on the breeding level of all breeders when compared to ferritic steel and SiC composites structures [12]. This study has confirmed that the Li/V system has the highest potential for breeding without a neutron multiplier. Therefore, achieving a tritium self-sufficiency is not expected to be a critical issue for the ARIES-RS design.

An approximate method based on 1-D analysis has been established for the ARIES-RS design with the objective of ensuring that the error associated with the final results is within a few percent of the 3-D estimates [13]. As mentioined above, provisions were made in the tritium breeding requirements for such uncertainties in the results. Thus, the overall TBR has been estimated by coupling the 1-D results with the neutron coverage fractions of the i/b and o/b blankets. The neutron coverage fractions, representing the fraction of source neutrons going directly to each region, are 20 and 68% for the i/b and o/b blankets, respectively, and 12% for both the top and bottom divertor regions. These fractions were then modified for losses in available areas due to the presence of the plasma control ports and assembly gaps. The net effect of the ports (which occupy $\sim 1\%$ of the o/b FW area) and the 2 cm wide gaps on the overall TBR is estimated to be about a 2% reduction.

The ARIES-RS design will rely on the i/b and o/b blankets to provide most of the required tritium and, therefore, no blanket will be installed in the divertor region. The reason is that the fraction of neutrons impinging on the divertor region is small relative to the i/b and o/b. Furthermore, these neutrons will be attenuated by the divertor plates and their support structures, leading to a further reduction in their contribution to the breeding. Minimizing the i/b radial build is well-known to be important for the overall size and cost of the machine. Therefore, it was decided to consider a thin blanket (20 cm) in the space limited inboard side and adjust the outboard blanket thickness to fulfill the breeding requirements. The blanket design is relatively simple. The lithium coolant/breeder flows poloidally from bottom to top. About 10% V structure is needed to support the blanket. The blanket is followed by a Li cooled reflector and shield. The analysis shows that 20 cm i/b blanket and 50 cm o/b blanket result in an acceptable TBR. The o/b blanket supplies $\sim 70\%$ of the tritium. The Li cooled reflector, divertor, and shielding components, other than FW and blanket, contribute $\sim 12\%$ to the overall breeding.

Designing the ARIES-RS plant for an overall TBR of 1.1 means that the actual achievable breeding ratio after the start of operation could range between 1.01 and 1.2 for a $\pm 9\%$ uncertainty. The achievable TBR in a realistic system will not be verified until after the operation of a demonstration plant. Underbreeding (overall TBR < 1.1) will generally place the plant operation at risk as the tritium bred may not suffice for machine operation. In this case, the plant will require an external tritium supply which is unacceptable for the ARIES designs as it violates the top-level requirements. It is therefore, imperative to conservatively design the ARIES plants with overbreeding blankets (overall TBR ≥ 1.1) providing that design solutions for reducing the breeding level by up to 20% should be established for all blanket designs. The adjustment in the design could take place during the second period of operation after changing out the blanket. For the Li/V system, a feasible solution would be to thin the o/b blanket by up to 20 cm. Alternatively, 2 or 4 o/b blanket modules could be converted into shielding modules. Support calculations should be performed prior to the plant operation for the proposed solutions in order to assess the impact of the changes on the breeding level and related neutronics parameters.

4. Radiation environment

The radiation levels at the ARIES-RS in-vessel components (FW, blanket, reflector, divertor, shield, and V.V.) and out-of-vessel components (magnet, cryostat and bioshield) are documented in this section. Also, the radiation effects on the various components (such as radiation damage, limited lifetime, needed protection, limited personnel access, etc.) are discussed. ARIES-RS is a high power density machine that operates at a high level of neutron wall loading (4–6 MW m^{-2}) and thus has a relatively intense radiation environment inside and outside the torus. The vanadium structure exhibits high radiation damage and requires frequent replacement over the 50 years planned operation of ARIES-RS. The permanent components need a sizable shield to provide the necessary protection against radiation. For instance, in addition to the breeding blanket, about a meter of shield/V.V. is needed to adequately protect the magnets. The blanket and shield should protect the V.V. and allow its reweldability at any time during operation. Besides breeding tritium, the blanket and reflector must protect the permanent shield for the entire plant life. Hands-on maintenance is not feasible in the reactor hall as the biological dose outside the V.V. is quite high after shutdown. A 2.5 m thick concrete wall should surround the torus to protect the workers and general public by limiting the biological dose to <2.5 mrem h^{-1} during operation.

4.1. Radiation damage to structures

Two main structural materials are employed for ARIES-RS, vanadium and stainless steel (SS) alloys. V4Cr4Ti is the candidate vanadium alloy as it possesses high radiation resistance to neutron damage. Among the different types of steels, Tenelon was chosen for having superior shielding performance compared to modified HT-9 and other steels. For economic reasons, the use of V is limited to the plasma facing components where it is absolutely necessary for high temperature operation. The less expensive steel is used for components exposed to relatively lower radiation levels such as the back of the shield, V.V., magnet, and cryostat.

The atomic displacement, helium, and hydrogen production levels have the most impact on the neutron-induced effects in the V and steel alloys. The lifetimes of V and SS structures are determined by the dpa level attainable during operation. The criterion adopted in this study is that no more than 200 and 150 dpa are desirable for V and SS alloys, respectively [14]. A more restrictive limit is imposed on the steel of the V.V. To assure the reweldability of the V.V., the helium production level should not exceed 1 appm at any time during operation.

For a peak outboard neutron wall loading of 5.6 MW m^{-2}, the FW flux peaks at a value of 6×10^{14} n cm^{-2} s^{-1}. The o/b blanket and shield provide five orders of magnitude attenuation for the flux. The peak neutron-induced damage to the o/b FW structure at the end of one full power year (FPY) of operation are 77 dpa/FPY, 315 He appm/FPY, and 6660 H appm/FPY. The FENDL-1 library (which is based mainly on ENDF/B-VI evaluation) and the ENDF/B-V library (which was used in previous ARIES designs) were compared for the radiation damage to the vanadium structure of the o/b first wall. The FENDL-1 library predicts 10% higher dpa rate, 33% lower He production rate, and a factor of three higher H production rate for V. The latter raised some concerns within the materials community as hydrogen may embrittle the vanadium particularly during shutdown when the temperature drops below 500°C. Further investigation has revealed that among the various (n, H) cross sections for V, the (n, $n'p$) cross section has increased significantly in the FENDL-1 evaluation. This inconsistency has been brought to the attention of the LANL data processing group to validate the ENDF/B-VI data for V. The identified discrepancy will hopefully be removed in the forthcoming FENDL-2 data library.

The radial variation of damage is illustrated in Fig. 4 for the outboard components which are

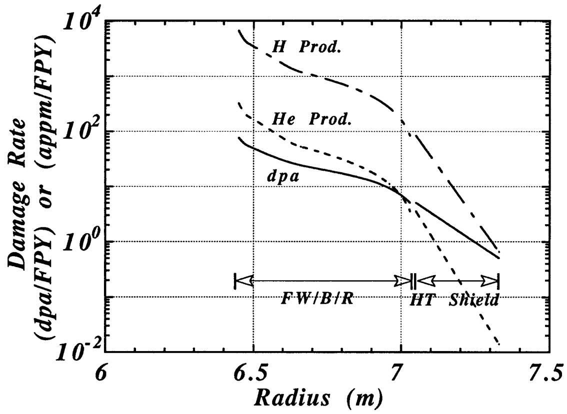

Fig. 4. Radial variation of dpa, helium production, and hydrogen production rates in vanadium structure of outboard components for 5.6 MW m^{-2} peak wall loading.

Table 1
Radiation limits for ARIES-RS superconducting magnets

Fast neutron fluence to NB_3Sn conductor ($E_n > 0.1$ MeV)	10^{19} n cm^{-2}
Peak nuclear heating in winding pack	2 mW cm^{-3}
Total nuclear heating in magnets	50 kW
Peak dose to GFF polyimide insulator	10^{11} rad
Peak dpa to Cu stabilizer	6×10^{-3} dpa

subjected to the highest wall loading. The damage to the V structure falls off rapidly as one moves radially away from the FW. Although the flux decreases by a factor of ~ 3 across the 50 cm thick blanket, the drop in damage, which is a spectrum dependent, ranges from 8 to 25, depending on the radiation effect. The analysis assumes that the gas production continues to build up within the structure with time. In reality, some of the gases will diffuse out of the hot structure (500–700°C) and the gas levels will be below the reported values. However, the MHD coating on the V structure will certainly affect the dynamics of the diffusion process. The available data are very limited and, therefore, the actual amount of gases retained in the irradiated V is not known at the present time.

The He production level at the V.V. at the end of the 40 FPY plant life is below the 1 appm limit meaning that any part of the V.V. can be rewelded at any time during operation. Furthermore, the shield, which is primarily designed to protect the magnet, will also provide lifetime protection for the V.V. On the other hand, there is no experimental data on the limit for the reweldability of irradiated V alloys. It is predicted that the welding environment could be more important than the neutron-induced gas production for V [14]. Due to lack of data, a decision was made to assemble the in-vessel components using mechanical attachments and avoid welding the V components inside the high radiation environment.

4.2. Radiation damage to magnet

The radiation limits provided by the magnet designers are listed in Table 1. These limits must be met in order to avoid degradation of the magnet critical properties due to radiation at any time during plant operation. The shield is designed primarily to protect the magnet. The blanket, reflector, and V.V. provide additional shielding for the magnet. Section 8 documents the radial dimensions of the various in-vessel components needed to reduce the radiation level below the specified limits. Details of the magnet damage profile and the basis for the shield design are documented in Section 6.

4.3. Radiation level outside torus

Personnel accessibility to the torus and the feasibility of hands-on maintenance depend on the radiation level around the machine. A key factor in this assessment is the biological dose throughout the machine. This is shown in Fig. 5 at different times after shutdown for the outboard FW and V.V. of ARIES-RS subjected to an average outboard neutron wall loading of 4.5 MW m^{-2}. According to DOE guidelines for protection of workers [15], personnel access is allowed in areas where the dose falls below 2.5 mrem h^{-1}. The results indicate that the area inside the cryostat is non-accessible and should be maintained by robotics and/or remote handling equipment. If hands-on maintenance or personnel access requirements are mandatory, then major changes to the shield design should be implemented in order to meet the 2.5 mrem h^{-1} limit.

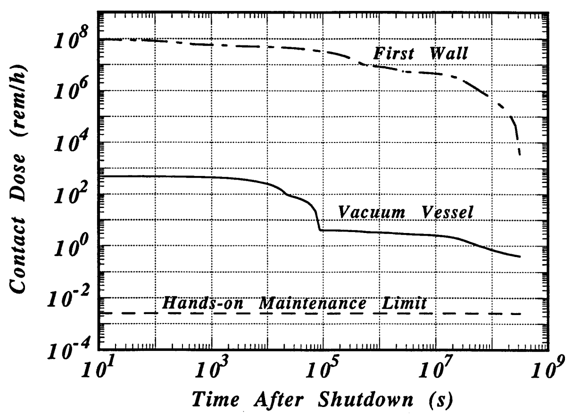

Fig. 5. Contact dose at outboard first wall and vacuum vessel (Courtesy of H. Khater (UW)).

In this case, the machine will be heavily shielded as additional bulk shield, sizable shield around penetrations, local shields behind assembly gaps, and well shielded transporters and flasks are needed to meet the limit in the area outside the V.V. The extra shield will increase the cost of the machine by several mill/kW_eh, which is significant. However, once the machine is opened and the FW/blanket and shield are moved out for replacement, the outer areas will be highly contaminated. It is judged that the added cost and complexity offset the benefit of permitting hands-on maintenance in the limited area outside the V.V. for a short period of time (a few hours) before opening the machine for maintenance. For these reasons, the ARIES-RS machine will be maintained by robotics and remote handling equipments.

The biological shield surround the cryostat to protect the work-force and general public from radiation hazards within the plant. It controls the radiation level outside the plant and adds a confinement barrier against radiological release. A 2.5 m thick steel-reinforced concrete is required to limit the dose to 0.5 mrem h^{-1} outside the bioshield during machine operation. The attainable dose level reflects a safety factory of five reduction in the absolute limit (2.5 mrem h^{-1}). A general guideline for the buildings containing the support systems is to limit as much as possible the areas where personnel access is not possible. Rooms for RF launchers and hot cells are non-accessible. The use of separation walls will create intermediate accessible areas close to these systems. Transition tunnels to hot cells should be well shielded. If personnel access is desirable outside the hot cell building, the walls should be 2 m thick. Other rooms containing less radioactive components (such as coils, cryostat dome, etc.) may have thinner walls ($\sim$0.5 m thick).

5. Components lifetimes

For previous ARIES designs, the plant service lifetime was 40 years, consistent with the NRC fission plant license period. It is prudent for advanced fusion designs to anticipate possible life extension by designing to a higher value, for example 40 FPY, as has been the case for ARIES-RS. For a plant factor of 80–90%, the service life could be extended to 45–50 years. This is a reasonable assumption as extended life of even 60–80 years are suggested for advanced fission plants. If desirable, a longer ARIES-RS life ($>$40 FPY) will require a few centimeter additions to the radial dimensions of the replaceable components.

The lifetimes of the individual in-vessel components vary with the radiation damage level attainable during operation. Plasma facing components (PFC) are subjected to the highest flux and thus have the shortest lifetime. Larger cost items, such as shield and magnet, are designed to last for the plant life. The lifetime of the V structure is set by the 200 dpa limit. This limit implies that the PFC should be replaced every $\sim$2.5 FPY at an end-of-life (EOL) fluence of 15 MWy m^{-2}, requiring 15 replacements during the machine lifetime. The EOL fluence is based on a peak o/b neutron wall loading of 5.6 MW m^{-2} that remains nearly constant over a poloidal height of a few meters around the midplant. Even though the i/b FW/blanket is subjected to a lower wall loading, it will be replaced with the o/b blanket on the same basis. There will certainly be an incremental cost associated with the early replacement of the i/b blanket, but this will be offset by the gain due to the fewer maintenance processes for the combined i/b and o/b replacement scheme, shorter down time, and thus higher availability for the overall system. The replacement rate for the divertor plates is not determined by their longer radiation damage lifetimes, but rather by the W coating erosion rate. The divertor design was able to achieve an erosion lifetime similar to the radiation damage lifetime of the o/b FW/blanket. This achievement was essential to simplify the maintenance scheme and to meet the high availability goal.

The replacement components include the first wall, divertor, blanket, reflector, divertor support structure, and outermost part of the i/b shield. The dpa level achieved after one FPY for all in-vessel components are listed in Table 2. Note that all shields are lifetime components as they meet the 200 dpa limit at the 40 FPY end of plant life. It is pertinent to mention that the lifetime

Table 2
Peak atomic displacement in the vanadium structure of in-vessel components

	Peak dpa @1 FPY	Lifetime for 200 dpa	
		FPY	MWy m^{-2*}
Outboard components (5.6 MW m^{-2}):			
FW and blanket-1	77	2.6	15
Blanket-2 and reflector	24	8.2	46
HT shield	5	40	225
Inboard components (3.7 MW m^{-2}):			
FW/blanket	62	3.2	18
Replaceable shield	23	8.7	49
HT shield	4.8	42	235
Divertor components (2.3 MW m^{-2}):			
Divertor plates	28	7.1	40
Support structure	17	11.8	66
HT shield	5	40	225

* Relative to o/b peak wall loading.

protection of the shield was achieved by adjusting the thicknesses of the o/b reflector, i/b replaceable shield, and divertor support structures so that the first layer of the shield has an acceptable damage level at EOL.

The desire for economic improvement led to an innovative replacement scheme. A cost saving was achieved by radially segmenting the replaceable in-vessel components into two zones so that the large and/or heavy outer segments can be reused. In addition to the cost savings, the segmentation will maximize the useful lifetime of the components and thus minimize the radwaste stream. The cost saving associated with this approach is evaluated with the perspective that the overall replacement cost is minimized and the radial partitioning is not too complex or time consuming. The replacement schemes for ARIES-RS are scheduled at $\sim$2.5 FPY and 7.5 FPY. The second lifetime is three times the first in order to keep to 2.5 FPY components as thin as practically possible. Hence, the o/b blanket is divided into two zones (blanket-1 and blanket-2), the i/b blanket is separated from the i/b replaceable shield, and the divertor plates are attached to the divertor support structure. The inner components (i/b FW/blanket, o/b FW/blanket-1, and divertor plates) are replaced at $\sim$2.5 FPY, while the outer components (i/b replaceable shield, divertor support structure, and o/b blanket-2/reflector) are replaced every $\sim$7.5 FPY. Even though some components (like the o/b reflector and divertor support structure) have longer life, they will be replaced at 7.5 FPY to increase the availability of the system. This novel replacement scheme has reduced the annual replacement cost by $\sim$2 mills/kW$_e$h ($\sim$3% of COE) and decreased the cumulative radwaste by $\sim$2000 tonnes ($\sim$13% of FPC mass). The effect of the smaller frequency of replacement on the availability of the system was not quantified. Fig. 6 displays the two sets of replaceable components. The outer 7.5 FPY components comprise a structural ring that provides poloidal continuity and attachment points for the inner 2.5 FPY components. During maintenance, 1/16th of the ring is moved out radially as a single unit to a hot cell [16] where the inner components are replaced and the outer components are reused until they reach their own useful life.

6. Shielding system

The ARIES-RS shield consists of the bulk (or magnet) shield, penetration shield, and biological shield. The bulk shield protects primarily the superconducting magnets, serves as a power production component, and acts as a heat sink for the FW/blanket decay heat during a loss-of-coolant accident (LOCA). The biological shield surrounds the torus and controls the radiation level outside the plant, as discussed in Section 4.3. This section documents the shielding system requirements developed for the shield of U.S. fusion power plants and describes in detail the design of the bulk shield in particular, emphasizing the attainment of the requirements.

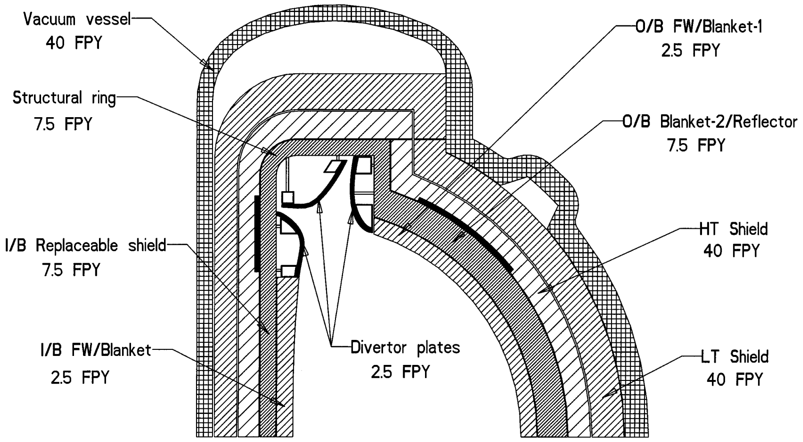

Fig. 6. Cross section of ARIES-RS showing lifetimes of various in-vessel components.

6.1. System requirements for shield

A set of system requirements for the shield of U.S. fusion power plants were developed concurrently with all plant subsystems. The requirements relate to the performance, economic, and safety features of the design as seen from the utility point of view [17]. During the course of the study, examinations were made of the ability of the reference shield design to meet the requirements. As listed in Table 3, the requirements are:

1. *Provide magnet protection*. The prime function of the bulk shield is to protect the superconducting magnet against radiation. The magnet limits provided in Table 1 are set by the magnet designers to assure the proper performance of the magnet during operation.
2. *Provide vacuum vessel protection*. Controlling the neutron-induced helium level in the steel of the vacuum vessel enables rewelding of the vacuum vessel at any time during the 50 y planned operation. It is likely that the magnet shield will also provide lifetime protection for the vacuum vessel.
3. *Lifetime component*. The shield is massive and costly and, therefore, should be a lifetime component requiring no replacement during plant operation due to radiation damage considerations. Its own lifetime protection is provided by sizing the blanket and reflector so that the atomic displacement to the V structure of the shield is below 200 dpa.
4. *Power production component*. About 20% of the nuclear heating is generated in the shield. This must be recovered as high grade heat to improve the economics, meaning that the shield should be a power production component.
5. *Safety and environmental impact*. Shielding materials that exhibit radiological disintegra-

Table 3
System requirements developed for the shield of U.S. power plants

Provide lifetime protection for magnets	
Provide lifetime protection for vacuum vessel	<1 He appm
Must be a lifetime component	<200 dpa for V <150 dpa for steel
Must be a power production component	
Must have low safety and environmental impact:	
Low level waste (Class C with impurity control)	
No hazardous materials	
No damage in case of LOCA/LOFA	
Must have a reasonable cost (<200 M$)	
Must meet stress and temperature limits	
Must be reliable, maintainable, and replaceable	
Should not expose workers to more than 2.5 mrem/h during operation	

tion during operation or decompose at high temperatures ($\sim 800°C$) during a LOCA should absolutely be excluded. As a general design requirement, the shield should qualify as a Class C low level waste at the end of plant operation. This means both vanadium and steel alloys should be produced with strict specifications to control impurities of radiological concern, such as Nb, Mo, Ag, Ir, etc.

6. *Economic requirement*. The shield is a major cost item. Reducing its cost to below 200 M$ is the most challenging item on the list of requirements. A serious effort is made in the ARIES-RS study to fulfill this cost requirement.
7. *Stress and temperature requirements*. Proper arrangement of the coolant channels within the shield assures that the stress and temperature limits for V and steel structures are not exceeded. The design limits on peak temperatures for V and steel are 700 and 550°C, respectively.
8. *Reliability, maintainability and replaceability*. The shield configuration should be simple and have the ability to be quickly maintained. Even though the shield is a permanent component, the design should allow for its removal if the magnet needs replacement due to malfunction. During shutdown, the maintenance process will be performed using robotics and remote handling equipments.
9. *Radiological protection*. The bioshield that surrounds the torus should be sized to limit the biological dose to the public and workers to 2.5 mrem h^{-1} during operation.

As highlighted throughout this section, all shielding system requirements have been met through a series of design improvements.

6.2. Cost-effective bulk shield design

6.2.1. Design choices

Vanadium alloys and low-activation ferritic steels are the most promising candidates for the bulk shielding components. V offers significant advantages in terms of low activation characteristics and high thermal performance capability. On the other hand, low-activation steels are relatively inexpensive, have a higher shielding performance than V, and have the largest database and industrial support, and hence the smallest uncertainty in performance. Vanadium alloys certainly offer the benefits of operating at higher wall loadings (making the design more compact and less expensive), operating at higher temperatures (meaning higher thermal conversion efficiency), and having lower radioactive inventory and afterheat. Special materials, such as B_4C and WC, offer advantages for local shielding applications. Their high cost limits their use in large quantities so that most of the shielding materials still would be vanadium or steel.

Previous designs traditionally employed water-cooled steel shields to protect the magnets. For safety and technical reasons, the water-cooled steel shield cannot be used in the Li/V designs. The use of lithium breeder will probably make the use of water (or organic coolant) in the shield impossible for safety reasons. No coolant other than Li is known to be compatible with V.

Even though V alloys offer several advantages over steels, the economic advantage of V is limited. Because V structure is expensive (300$ kg^{-1}), power plants made entirely out of V structure will not be competitive (all costs reported herein are quoted in 1992 dollars). Thus, some design tradeoffs are necessary in order to improve the economics of Li/V designs and minimize the impact on the safety and environmental features of the design. For instance, the space between the coolant channels of the shield could be filled with cheap filler materials, instead of being made out of fabricated structures. There is no structural role envisioned for the filler materials. Another improvement that helps reduce the cost of the shield is the use of steel filler, instead of V filler [18]. Besides having a lower unit cost, steel has better shielding performance than V and thus results in thinner radial builds and a smaller machine. The economic analysis provided by the ARIES systems code (ASC) has indicated that these modifications to the shield design have lowered the cost of electricity by tens of mills/kW_eh. However, the Li/V shield still represents a major cost item (~ 400 M$) and needs further improvements.

6.2.2. Shield improvements

This subsection illustrates the steps by which the shield was modified while satisfying the prescribed system requirements. During the study, the ASC was heavily used to quantify the cost savings of the proposed changes before incorporating them in the ARIES-RS design. The starting configuration of the shield is taken from the ARIES-II design and consists of 40 cm steel shield followed by 23 cm B_4C shield on the inboard. A less efficient, cheaper steel shield is utilized in the less constrained outboard and divertor regions. All shields contain 15% V structure and 5% Li coolant, by volume. A breakdown of the cost has revealed that the outboard shield constitutes more than 50% of the shield cost and, more importantly, the V structure comprises $\sim$60% of the overall shield cost, even though it occupies only 15% of the shielding space. These findings have prompted the need to minimize the use of V structure in the ARIES-RS shield (and external systems) and instead utilize less expensive steel structure to reduce cost. It is essential, therefore, to limit the use of V to components where high temperature performance is most needed, i.e. in the plasma facing components and blanket.

The following three main modifications would offer potential improvements to the shield design:

1. The unit cost of steel should be revisited and compared with estimates from other designs for components with similar level of complexity.
2. Further enhancement to the shielding performance is needed by optimizing the composition and employing more efficient, inexpensive shielding materials.
3. The use of advanced V material must be limited to those regions where it is absolutely necessary for high temperature operation.

The impact of the shield improvements on the overall size and cost was assessed using the ASC. Several runs were made to quantify the cost saving of each change made to the shield design. The changes were implemented one at a time sequentially and then the incremental cost reduction for each sequential change was determined. As discussed shortly, by making these changes the cost of the shield has reduced below 200 M$ ($\sim$ factor of 2 reduction) and the overall cost of electricity has dropped by 10%, which is significant [8,19].

The impact of changing the unit costs of fabricated steel structure and filler was investigated first. The ARIES-II design was based on unit costs for steel structure of 68$ kg^{-1} and steel filler of 25$ kg^{-1}. Since the shield and vacuum vessel (V.V.) are well protected and not subjected to as high a radiation level as the FW/blanket, any type of steel could be used in these components. There are several low activation steels (MHT-9, Tenelon, Fe1422, 316SS, ORNL 9Cr-2WVTa, F82H-M, MANET, etc.) that are readily available for use in fusion power plants. As shown later, steel will be employed in the outer part of the shield, V.V., and all ex-vessel components. The steel shield and V.V. will run at low temperatures relative to the FW/blanket; will not suffer as much radiation damage (meaning simpler welds or less inspection); will have much simpler configuration, fewer attachments, and less plumbing; and will thus have the ability to be quickly maintained. These features should translate into lower costs for such moderate-complexity components. Other designs [20,21] have quoted lower cost estimates such as $\sim$35$ kg^{-1} for steel structures and $\sim$10$ kg^{-1} for steel fillers. These values are based upon industrial experience with large steel structures and cost estimates for similar shielding and V.V. components. Using these lower unit costs, the shield and V.V. cost has been reduced substantially by $\sim$100 M$ and the overall cost of electricity (COE) has dropped by 3 mills/kW_eh.

The second change enhances the performance of the shield by employing more efficient shielding materials such as tungsten carbide and borated-steel [22]. Specifically, boron carbide was replaced by tungsten carbide (65$ kg^{-1}) in the inboard side and the steel filler was replaced by borated-steel filler (10$ kg^{-1}) in the outboard and divertor regions. These modifications have reduced the radial builds by 8–20 cm, the major radius by 8 cm, and the COE by 2 mills/kW_eh.

The third change is to limit the use of the vanadium structure in the shield. Since steel cannot operate at temperatures as high as V, the use of steel structure in the entire shield will certainly

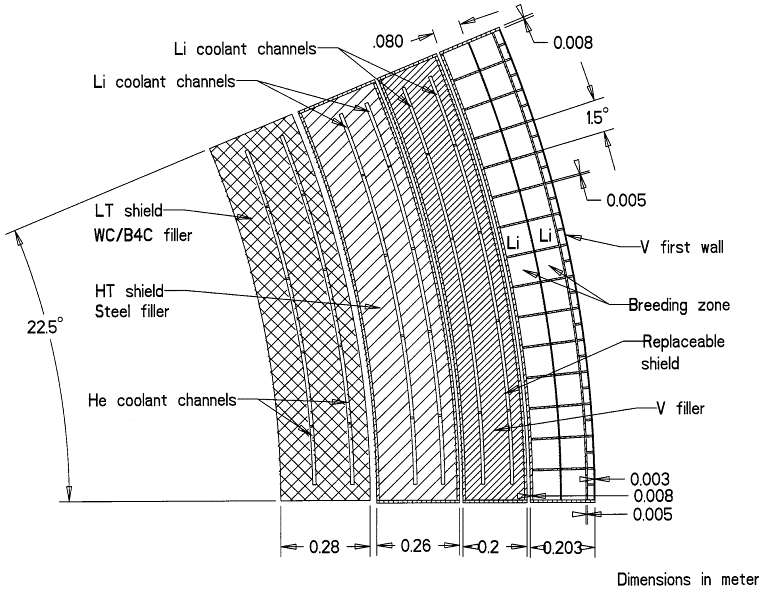

Fig. 7. Cross section of inboard blanket and shield.

degrade the overall thermal conversion efficiency of the system. In addition, the option of running the steel shield at very low temperature is not viable because the nuclear heating generated in the shield is significant ($\sim$ 20% of the total heating) and cannot be dumped as low grade heat without negatively impacting the power balance of the machine. For these reasons, V structure should be used in all or part of the shield to increase the ability to extract the heat with Li coolant at high efficiency for the purpose of generating electricity. An attractive solution is to divide the shield into two parts: the inner part follows the blanket and operates at a high temperature (300–700°C), while the outer part operates at a relatively lower temperature ($\sim$ 200°C). Hence, the high temperature (HT) shield along with the FW and blanket employs V structure whereas the low temperature (LT) shield utilizes stainless steel as the main structural material. The heat extracted from the LT shield will not be recovered. The tradeoffs between stainless steel and advanced materials will thus depend on the dividing boundary between the two layers of the shield and on how much power could be dumped as low grade heat without overly affecting the power balance. It is anticipated that the allowable reduction in the useful thermal power to be on the order of 1–5%. The cost saving due to segmenting the shield is 2 mills/kW$_e$h. This modification should not impact the safety characteristics of the design as the shield is subjected to low radiation flux, generating low levels of radioactivity and afterheat. In fact, segmenting the shield into two zones helps the design in case of an accident. It is true that the gap between the two zones will slow down the conduction of decay heat from the HT shield to the LT shield, but more importantly, the LOCA analysis [23] has shown that the He cooled LT shield, which operated at a temperature below 200°C, acts as a heat sink and helps the FW temperature to drop faster after the first day following any accident.

6.3. ARIES-RS shield design

Figs. 7 and 8 display cross sections of the i/b and o/b in-vessel components. An elevation view of ARIES-RS is displayed in Fig. 1. In response to the safety analysis, which proceeded iteratively with guidance from the neutronics analysis, V filler is used in the inboard replaceable shield to

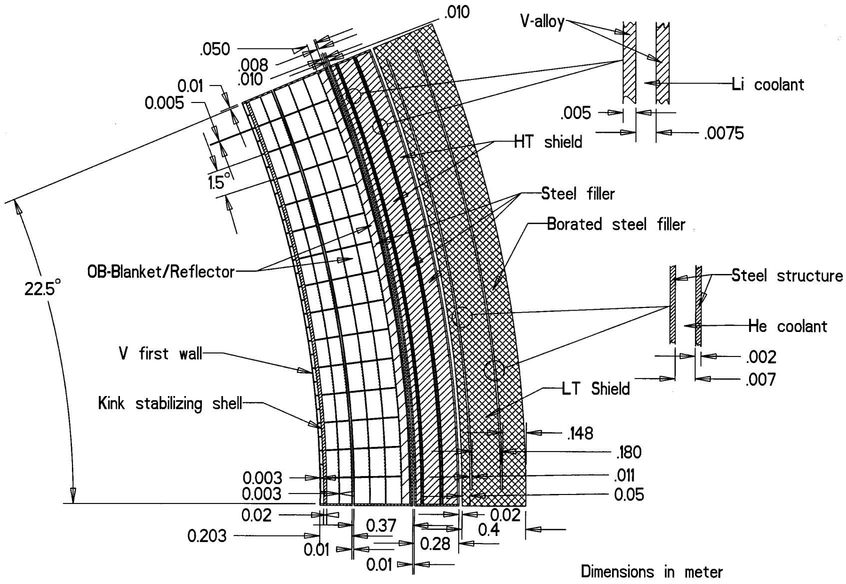

Fig. 8. Cross section of outboard blanket and shield.

reduce the decay heat and the peak FW temperature attainable during LOCA. The HT shield contains steel (Tenelon and MHT-9) filler while the use of more efficient shielding materials (WC, B_4C, and borated steel) is limited to the LT shield where the radiation environment is less severe. To reduce the size and cost of the overall machine, the highly efficient and more expensive WC and B_4C fillers are utilized in the space-limited inboard side while borated steel is used elsewhere. WC filler is used in the 2 m high central zone around the i/b midplane whereas B_4C fills the remaining space of the i/b LT shield. Tenelon has a superior shielding performance compared to other steels. It is thus employed as the main filler and structural material for all steel-based components except in the divertor support structure. Instead, modified HT-9 is used for having lower decay heat, a property of great importance for highly irradiated components. Near the end of the study, new information became available from the materials community on the ORNL 9Cr-2WVTa steel and the international IEA modified F82H steel [24]. Due to the near completion stage of the ARIES-RS design, it would have been difficult to replace the Tenelon and MHT-9 by the recommended steels and complete the redesign within the remaining time of the study. Nevertheless, the low activation steels will be considered in future designs.

Table 4 lists the various shielding components and the radiation levels at the shield, V.V., and magnet. The replaceable components (FW, blanket, reflector, and divertor support structure) are sized to result in ≤ 200 dpa at the V structure of the shield. All shields are thus lifetime components. The magnet radiation effects are below the limits, meaning that the magnets will perform properly for 40 FPY and there is no need for magnet annealing. The integral nuclear heating to the 16 TF magnets amounts to 6 kW. This corresponds to an acceptable cryogenic load of ~ 2 MW. The shield will also provide lifetime protection for the V.V. as evidence by the fact that the neutron-induced helium level is below the 1 appm limit. This warrants the ability to reweld the V.V. at any time during the 50 years of operation. It should be mentioned that the V.V. and magnet radiation damage reported in Table 4 include the safety factors that accounts for uncertainties in nuclear data, calculational model, and effect of radiation streaming through the 2 cm wide radial

Table 4
Key blanket and shield dimensions and radiation effects at shield, V.V. and magnet

	Inboard	Outboard	Divertor
Thickness (cm):			
FW or divertor plates	0.3	0.3	5
Blanket	20	50	0
Replaceable shield, reflector, or divertor support structure	20	7	20
Permanent shields:			
HT shield	26	28	35
LT shield	28	40	45
Vacuum vessel	20	30	20
Total (excluding gaps)	114	155	125
Peak neutron wall loading (MW m^{-2})	3.7	5.6	2.3
Peak dpa to V of shield (dpa@40 FPY)	190	200	200
Peak He production in V.V. (appm@40 FPY)	0.2	3×10^{-2}	6×10^{-3}
Damage to magnet:			
Peak fast neutron fluence to Nb_3Sn (n/cm^2@40 FPY, $E_n > 0.1$ MeV)	1×10^{19}	1×10^{19}	8×10^{18}
Peak nuclear heating (mW cm^{-3})	0.4	0.2	0.17
Peak dose to GFF polyimide (rad@40 FPY)	1.4×10^{10}	7×10^{9}	6×10^{9}
Peak dpa to Cu stabilizer (dpa@40 FPY)	5.4×10^{-3}	4×10^{-3}	3.3×10^{-3}

gaps between modules [25,26]. The streaming effect has been mitigated by offsetting the blanket gaps from the shield gaps. The recommended safety factors for the ARIES-RS design are 2 for the integral nuclear heating and 3 for the local radiation effects, such as fast neutron fluence, peak nuclear heating, peak dose to insulator, and peak dpa to Cu stabilizer. The impact of including the safety factors in the analysis is to thicken the shield by ~ 6–7 cm.

In standard diverted tokamaks, a critical shielding area exists above/below the i/b blanket where the shield recesses to accommodate the inner divertor plate. The shapes of the i/b blanket and divertor plate were designed to protect the recessed surface and the magnet against source neutrons. 3-D analysis indicated that the neutron wall loading at the recessed surface is relatively low (0.035 MW m^{-2}) due to the shadowing effect of the inner plate (see Section 2). The radial space therein is approximately one meter thick. It consists of 20 cm structural ring, 26 cm HT shield, 28 cm LT shield, 20 cm V.V., and a few 1–2 cm wide assembly gaps. The calculations showed that the one meter shielding space adequately protects the inner legs of the magnets providing that WC filler is used in both V.V. and LT shield.

7. Nuclear heat loads

All in-vessel components (FW, blanket, divertor, reflector and shield) are power production units except the rear portion of the shield. The exclusion of the back of the shield from producing power was made for economic and safety reasons as described in Section 6. The breakdown of the heat deposited in the FW, divertor, blanket, reflector, and shield is given in Table 5. The machine produces 2167 MW of fusion power and 80% of this power is carried by neutrons. Neutrons and gammas deposit their energies in the various components generating an additional 430 MW of thermal power. Hence, the design has an overall neutron energy multiplication (M) of 1.2. A large fraction of the power (70%) goes to the blanket.

One of the items that contributes to the attractiveness of the design is the neutron energy multiplication of the system. From an economic point of view, it is desirable to develop a system with high M that leads to a combination of high gross thermal efficiency, compact machine, and low cost of electricity. The energy multiplication of ARIES-RS is evaluated with the International FENDL-1 Data Library [5] following the same

Table 5
Nuclear heating deposited in the various in-vessel components

	Inboard	Outboard	Divertor	Total
FW or divertor plates	10	23	54	87
Blanket-1	285	680	—	965
Blanket-2	—	445	—	445
Reflector, replaceable shield, or divertor support structure	150	115	102	367
HT shield	48	122	46	216
Total	493	1385	202	2080

methodology used to estimate the overall TBR. Even though the two systems are quite similar, M of ARIES-RS is smaller than that of ARIES-II (1.37) which was evaluated with the U.S. ENDF/B-V data library [6,7]. The difference in M is strictly attributed to the cross section library evaluation. The 12% reduction in M corresponds to a 1.5 mill/kW_eh increase in the ARIES-RS overall cost of electricity. This shows the important role of the cross section libraries in power plant designs and the need for validation and checking against experimental data.

8. Radial builds

This section contains a description of the radial builds emphasizing the impact of the neutronics and shielding analyses on the dimensions and constituents of the in-vessel components. Tables 6 and 7 summarize the final dimensions and compositions provided by the engineering group with guidance from neutronics, thermal hydraulic, structural, economic, and safety analyses. The given dimensions herein are at the i/b and o/b midplanes and for a vertical cross section through the divertor region.

As indicated throughout the preceding sections, the neutronics analysis forms the basis for most of the in-vessel component dimension/composition. The blanket is sized to supply the plant with the tritium needed for self-sufficient operation (TBR $\geq$ 1.1). Moving away from the i/b midplane, the i/b blanket fans out and the i/b FW conforms to the plasma boundary. The i/b shield has vertically straightened surfaces. The i/b blanket is 35 cm thick at the top/bottom end. The curved blanket provides an additional shielding for the magnet and helps reduce the integral nuclear heating in the inner legs of the coil by a factor of ~ 2, compared to a straightened blanket.

The radiation limits for the structure and magnet played a key role in determining the radial dimensions of the various components. For instance, the i/b replaceable shield and reflector are sized to result in 200 dpa to the V structure of the shield at the end of the 40 FPY plant life. The specified dimensions of the shields fulfill the radiation limits listed in Tables 1 and 3 and protect both magnet and V.V. for 40 FPY. The shield is divided into HT and LT shields for economic and safety reasons. The He cooled LT shield is sized to contain 1% (~ 20 MW) of the total nuclear heating deposited in all components. The space between the coolant channels of the shield is filled with inexpensive filler materials, instead of solid structure, to reduce cost. The i/b replaceable shield is made out of V, instead of steel, to reduce the decay heat in case of an accident. The W stabilizing shells are embedded in the shield and occupy 4–7% of the i/b and o/b HT shield.

The HT shield contains steel (Tenelon and MHT-9) filler while the use of more efficient materials (WC, B_4C, and borated steel) is limited to the LT shield where the radiation environment is less severe. To reduce the size and cost of the overall machine, the highly efficient and more expensive WC and B_4C fillers are utilized in the space-limited inboard side while borated steel is used elsewhere. WC filler is used in the 2 m high central zone around the i/b midplane whereas B_4C fills the remaining space of the i/b LT shield. Tenelon has a superior shielding performance compared to other steels. It is thus employed as

Table 6
Radial dimensions of ARIES-RS components

	Inboard	Outboard	Divertor
Thicknesses (cm):			
FW or divertor plates	0.3	0.3	5
Blanket-1	20	20	0
Gap	1	1	1
Blanket-2*	—	30	—
Replaceable shield*, reflector*, or divertor support structure*	20	7	20
Assembly gap	1	1	1
Permanent shield:	56	70	82
HT shield	26	28	35
Gap	2	2	2
LT shield	28	40	45
Assembly gap	2	>2	>2
Vacuum vessel	20	30	20
Total FW/B/R/S/V.V./gaps	120	161	130
V.V.-magnet gap	5	>5	>5
Winding pack	~50	~50	~50
Cryostat	24	24	24

* Comprise the structural ring.

the main filler and structural material for all steel-based components except in the divertor support structure. Instead, modified HT-9 is used for having lower decay heat, a property of great importance for highly irradiated components. The divertor support structure contains coolant tubes for divertor system, i/b blanket, and replaceable shield. Above/below the i/b blanket, the surface of the shield is recessed to accommodate the inner divertor plates. The shielding space therein is constrained and thus WC filler is used in the LT shield and V.V. to adequately protect the inner legs of the magnet.

The composition of the vacuum vessel is driven by shielding considerations. The V.V. is the closest component to the magnet and thus its composition affects the radiation level at the magnet. It is a double wall He cooled steel structure stiffened with steel ribs. For shielding considerations, the 13–23 cm thick space between the face sheets (3 cm thick each) is filled with borated steel plates except on the space-limited i/b side where more efficient shielding materials (WC and B_4C) are employed.

The elements surrounding the winding packs were included in the neutronics model as they provide additional shielding for the coils. The winding pack is surrounded by a 2 cm steel plate followed by 0.2 cm electric insulation, 1 cm steel jacket for handling protection, and 4 cm thermal insulation (Al foils separated by fiberglass papers). To reduce the radial standoff on the inboard side, the thermal insulation is placed in the gap between the V.V. and magnet. The cryostat surrounds the entire torus. It consists of 2 cm thick steel face sheets, inter-connected with steel ribs. The steel comprises 18% of the cryostat volume.

The radial builds include the gaps between the various in-vessel components. In addition to the 2 cm wide radial assembly gaps between the blanket/shield modules, toroidal/poloidal gaps are provided between the various components within each module. Generally, the toroidal/poloidal gaps between the same-temperature components are 1 cm thick while that between the permanent shielding components with different operating temperatures are at least 2 cm thick. The V.V.-magnet gap is 5 cm thick on the i/b and much wider elsewhere.

Table 7
Main compositions of ARIES-RS components

First wall	100% V structure
Blanket	10% V structure
	90% Li breeder–coolant
i/b replaceable shield	15% V structure
	10% Li coolant
	75% V filler
o/b reflector	15% V structure
	10% Li coolant
	75% Tenelon filler
i/b permanent shield:	
HT shield	15% V structure
	5% Li coolant
	76% Tenelon filler
	4% W structure
LT shield	15% Tenelon structure
	5% Helium coolant
	53% B4C filler (90% d.f.)
	27% WC filler (95% d.f.)
o/b permanent shield:	
HT shield	15% V structure
	5% Li coolant
	73% Tenelon filler
	7% W structure
LT shield	15% Tenelon structure
	5% Helium coolant
	80% borated Tenelon filler
Divertor plates	90% V structure
	6% Li coolant
	4% W coating
Divertor support structure	15% V structure
	25% Li coolant
	60% MHT-9 filler
Divertor permanent shield:	
HT shield	15% V structure
	5% Li coolant
	75% Tenelon filler
	5% WC filler (95% d.f.)
LT shield	15% Tenelon structure
	5% Helium coolant
	77% B-Tenelon filler
	3% WC filler (95% d.f.)
Vacuum vessel:	
Inboard	35% Tenelon structure
	5% Helium coolant
	40% B4C filler (90% d.f.)
	20% WC filler (95% d.f.)
Outboard	25% Tenelon structure
	5% Helium coolant
	70% B-Tenelon filler
Divertor	35% Tenelon structure
	5% Helium coolant
	57% B-Tenelon filler
	3% WC filler (95% d.f.)
Magnet	73% 316SS structure
	14% Cu
	6% Nb_3Sn/NbTi
	3% GFF polyimide
	4% LHe
Cryostat	18% 316SS structure
	82% void

9. Conclusions

Detailed neutronics and shielding analyses raise no serious difficulties with the ARIES-RS design. The neutronics analysis provided the primary nuclear inputs for the thermal, mechanical, and safety analyses. The neutronics assessment included the neutron wall loading distribution, tritium production level, nuclear heat loads, and radiation environment which affects damage parameters (hence lifetime) and activation. The radial build of the in-vessel components is based on this assessment. The desire for economic improvements has led to several innovative schemes. For instance, a remarkable cost saving was achieved by radially segmenting the blanket and shield in order to reduce the direct cost of components, maximize their useful lifetimes, and minimize the radwaste stream.

A dedicated effort was devoted to the bulk shield in particular as it represents a major cost item for advanced designs. Shielding and economics considerations were used iteratively to guide the shield toward an optimal configuration while maintianing the attractive safety feature of the design. The ARIES-RS shield emphasizes the attainment of the top-level requirements. For instance, the bulk shield provides lifetime protection for the vacuum vessel and magnet, costs <200 M$, produces 20% of the power, generates Class C low level waste, and suffers no damage in case of LOCA.

Significant savings in shield cost were obtained by implementing several cost-effective improvements. Following are the principle findings of the shielding analysis:

1. The use of expensive V alloy should be limited to those regions where high temperature operation is absolutely necessary.
2. Segmentation of the shield into high temperature and low temperature zones offers a major reduction in shield cost.

3. The nuclear heat deposited in the HT shield (~ 20% of total heating) must be recovered as high grade heat to enhance the overall power balance of the machine.
4. LT shield, vacuum vessel, and other external components (having low levels of nuclear heating that can be dumped without significantly affecting the power balance) can employ a cheap stainless steel as the main structural material, instead of V.
5. The use of cheap steel filler (rather than V filler) in the shield reduces the cost tremendously. Power plants made entirely out of solid V structure will not be economically competitive. Fillers have no structural role and thus have lower unit costs compared to structures.
6. Careful attention should be paid to the arrangement of the high performance materials (normally expensive) within the shield. Highly efficient, expensive materials, such as WC and B_4C, could be used only in the space-constrained inboard side to reduce the overall size and cost of the machine while less efficient, cheaper materials could be used in the divertor and outboard sides.
7. Designing the shield to last for the entire plant life without replacement due to radiation damage considerations is essential to reduce the overall cost and to minimize the radwaste stream. Lifetime protection of the shield can be achieved by properly sizing the blanket and reflector.
8. The shield is a moderate-complexity component and thus should not be costed on the same basis as the blanket and plasma facing components. The shield should have lower fabrication, installation, inspection, and quality assurance costs.

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