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Fusion Reactor Systems Studies

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Progress Report for the Period
November 1, 1992 to October 31, 1993

Fusion Technology Institute
University of Wisconsin-Madison

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1 Introduction

Fusion Technology Institute personnel actively participated in the ARIES/PULSAR project during the present contract period. Numerous presentations were made at PULSAR project meetings, major contributions were written for the ARIES-II/IV Final Report [1], presentations and papers were given at technical conferences [2, 4, 5], contributions were written for the ARIES Lessons Learned report [6], and a very large number of electronic-mail and regular-mail communications were sent. The participation of Fusion Technology Institute engineers and scientists in PULSAR meetings is shown in Fig. 1 and a list of presentations is given in Table 1. The remaining sections of this progress report will summarize the work accomplished and in progress for the PULSAR project during the contract period (November 1, 1992 to October 31, 1993). The main areas of effort are

- PULSAR Research

- ☐ Neutronics
- ☐ Shield Optimization and Design
- ☐ Activation
- ☐ Materials Stress and Fatigue
- ☐ Energy Storage
- ☐ Safety

- ARIES-II/IV Report Contributions

- ☐ Chapter 7: Startup (contributions)
- ☐ Chapter 14: ARIES-II Neutronics (authored whole chapter)
- ☐ Chapter 16: ARIES-IV Fusion Power Core Overview (contributions)
- ☐ Chapter 18: ARIES-IV Divertor, First Wall, Blanket, and Power Conversion (contributions)
- ☐ Chapter 19: ARIES-IV Neutronics (authored chapter, minor non-UW contributions)

- ARIES Lessons Learned Report Contributions

- ☐ Advanced Fuels
- ☐ Neutronics
- ☐ Activation

- Stellarator Study

- ☐ Neutronics
- ☐ Engineering

Figure 1: *University of Wisconsin participation in ARIES project meetings.*

Table 1: *University of Wisconsin Fusion Technology Institute presentations at PULSAR project meetings.*

MEETING	UW AUTHORS	TITLE
December 9-11, 1992	Blanchard	Thermal Stress and Fatigue under Cyclic Loads
March 17-19, 1993	El-Guebaly Khater	PULSAR Blanket and Shield Neutronics
	Blanchard	Transient Thermal and Thermal Stress Issues
May 18-21, 1993	El-Guebaly	PULSAR Shielding
	Blanchard	Fatigue Lifetime of Plasma-Facing Components
July 22-23, 1993	El-Guebaly Khater Mogahed	Blanket and Shield Neutronics
	Khater	PULSAR's Activation Analysis
	Crowell Blanchard	Fatigue Behaviors of SiC and V-Alloy First Walls
Sept 27-29, 1993	El-Guebaly	Neutronics Issues
(planned)	Blanchard	First Wall Fatigue Analysis

2 Summary of Tasks Accomplished During FY93

2.1 Neutronics

The extensive, comprehensive calculations necessary for a credible design are being performed for the PULSAR first wall, blanket, shield, and vacuum vessel. Neutronics activities for PULSAR included

- Updating the ARIES-II and ARIES-IV blankets for PULSAR parameters,
- Analysing the effects of pulsing on activation,
- Determining the number of cycles at the end of the blanket life, which is determined by radiation damage,
- Discovering that about 0.3 kg of tritium will be generated and trapped in the SiC structure at the end of blanket life,
- Pointing out the importance of including the impurities in SiC and that the SiC unit cost should be consistent with the impurity level considered.
- Proposing the use of the shield as an energy storage unit, and
- Proposing titanium hydride for the shield in the SiC-blanket case, potentially saving 5-10 mill/kWh.

L.A. El-Guebaly is leading the PULSAR neutronics effort and University of Wisconsin personnel are calculating the nuclear heating, radiation damage, and tritium breeding for the PULSAR design. The nuclear heating is necessary to perform the power cycle thermal hydraulics analysis. Tritium breeding requirements must be accurately evaluated in order to assure that the blanket would provide sufficient tritium to support the fuel cycle.

2.2 Shield Optimization and Design

For the PULSAR design, we are performing a complete optimization and design of the magnet shields, including evaluation of candidate materials, variation of composition, and computation of radiation effects on the magnets. Parameters for metallic (V/SS) and SiC/SiC-composite shields are being scoped out. Corresponding algorithms for shield thickness have been given to the ARIES systems code group. Optimizing the shield will be an ongoing effort throughout the course of the PULSAR design.

2.3 Activation

H.Y. Khater is leading the PULSAR activation analysis and performing the main calculations. The activation of the first wall, blanket, shield, and vacuum vessel are being calculated in order to identify the possible safety, environmental and radwaste advantages. Several activation-related issues are also being investigated for the reactor structure. The effect of impurities and tritium production in SiC is also being calculated. The activity, decay heat, and biological hazard potential (BHP) are being calculated for up to 1000 years to assess the feasibility of hands-on maintenance following shutdown. Evaluation of the structure activity and biological hazard potential are needed to calculate the potential effects of radioactive inventory release in the event of an accident. In addition, results of the decay heat calculation are essential to examine the thermal response of the reactor shield following a loss-of-coolant accident (LOCA). The waste disposal rating (WDR) of the reactor structure at the end of its lifetime is also being evaluated. The waste disposal rating is needed to determine if the structure would satisfy the criteria for shallow land burial as a low-level waste.

2.4 Materials Stress and Fatigue

J.P. Blanchard is leading the materials stress and fatigue analyses for PULSAR, and is leading University of Wisconsin personnel in evaluating first-wall and blanket fatigue. He is also directing McDonnell Douglass in their analysis of disruption effects on reactor structures. This is a critical area for PULSAR, as some materials may undergo up to 100,000 cycles.

2.5 Energy Storage

University of Wisconsin are contributing to the vigorous discussion and design effort for the PULSAR energy-storage system. L.A. El-Guebaly contributed the idea of using the radiation shield for energy storage—an idea which is presently the main PULSAR energy-storage option.

2.6 Safety

Fusion Technology Institute personnel have been involved in the analysis of PULSAR safety, which is a critical fusion reactor issue. Activities include providing background information and detailed safety analyses. Activation, waste disposal ratings, afterheat, LOCA, and other safety issues are being investigated.

2.7 Contributions to the ARIES-II/IV Final Reports

The ARIES-II/IV Final Report will be published in 1994 and UW authors contributed primarily to Chapter 14, ARIES-II Neutronics (Appendix A), and Chapter 19, ARIES-IV Neutronics (Appendix B). Contributions were also made to Chapter 7, Startup, Chapter 16, ARIES-IV Fusion Power Core Overview, and Chapter 18, ARIES-IV Divertor, First Wall, Blanket, and Power Conversion. Numerous general contributions were made throughout the the ARIES-II/IV conceptual design process.

2.8 Contributions to the ARIES Lessons Learned Report

A summary document for the lessons learned from all of the ARIES studies is being put together. University of Wisconsin authors provided several contributions, and these are given in Appendix C.

2.9 Stellarator Study

University of Wisconsin personnel participated heavily in the stellarator study being led by J. Lyon of ORNL. We estimated the shield needed in critical areas and gave a range for blanket thicknesses to be used in the less-constrained spaces. This information was required to run a preliminary strawman for the continuous-coil ASR design option [7]. Contributions are being made to the magnet design many other engineering areas for this stellarator reactor. University of Wisconsin personnel attended and actively participated in four stellarator study meetings at ORNL (February, 1993), UCLA (May, 1993), GA (May, 1993), and UW (July, 1993).

References

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- [6] R.A. Krakowski, et al., "ARIES Lessons Learned," Los Alamos National Laboratory Report to be published in 1994.
- [7] J. Lyon, "Overview of the U.S. Stellarator Reactor Study," *Proceedings of the IAEA Symposium on Fusion Reactor Design and Technology* (to be published, 1993).

A Appendix: ARIES-II Neutronics

FIGURES AT END

Chapter 14

ARIES-II NEUTRONICS

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14. ARIES-IV NEUTRONICS

14.1. INTRODUCTION

Since the ultimate goal of this study is to design an attractive, reliable, safe, and cost competitive tokamak reactor, the environmental constraints and safety features were integrated in the blanket and shield design from the beginning with a view to cost, complexity, fabricability, reliability and integrability. The severe neutron environment associated with the ARIES-II device indicates that radiation-related issues are significant considerations in the blanket and shield design. The major neutronics concerns that require quantification include the determination of the tritium breeding potential of the blanket, radiation damage to the structural components, shielding requirements for the magnets, peaking due to streaming through penetrations, activation and waste disposal characteristics, and requirements for occupational personnel protection during reactor operation.

A blanket is required to breed a sufficient amount of tritium that ensures the self-sustaining operation of the DT fueled reactor. The blanket utilizes the liquid lithium as a coolant and breeder and the low activation vanadium alloy V5Cr5Ti as a structural material. Of significance is that there is no need for a neutron multiplier with the Li breeder to achieve tritium self-sufficiency and the required breeding is provided by the inboard and outboard blankets only. Section 14.2 documents the details of the design requirements for tritium breeding and the optimal blanket parameters in terms of Li enrichment and thicknesses. The radiation damage to the V structure of the first wall, blanket, reflector, and shield and the lifetime of the different components are covered in Section 14.3.

The blanket is followed by the shield which is primarily used to protect the superconducting toroidal field (TF) magnets against radiation. Since the dimensions of the shield directly affect the size and cost of the machine, an extensive analysis was performed to optimize the shield composition and performance. A variety of shield options was examined and the ability of various materials to protect the different reactor components was assessed, as explained in Section 14.4

The radiological issues of the design are addressed in Section 14.5. The decay heat, radioactive inventory, and waste disposal level are quantified. Finally, the conclusions to be drawn from this work are summarized in Section 14.6.

14.2. BLANKET DESIGN

14.2.1. Tritium Breeding Requirements

Since tritium (T) is one of the materials which has a very limited resource, it must be produced in a lithium blanket that surrounds the plasma. Thus, the primary goal of the blanket design is to attain tritium self-sufficiency that ensures the self-sustaining operation of the DT fueled reactors. The requirements to achieve this goal depend on some plasma physics parameters, several uncertainties in the design, and the technical performance of specific reactor components. The determination of these uncertainties and parameters is essential in order to define the excess margin in the breeding potential necessary to achieve tritium self-sufficiency. In ARIES-II, a serious effort was developed to lower the breeding requirements by increasing T burn-up in the plasma, ensuring extremely low T losses, achieving low T inventory in all subsystems, and requiring high reliability and fix rate for the T processing systems. It appears that with the existing technology, a system with negligible T leak can be achieved to guarantee a safe and reliable design.

The tritium breeding ratio (defined as the amount of T generated in the blanket per fusion reaction) in a self-sustained power reactor must exceed unity by a margin G_o . Thus, the required tritium breeding ratio (TBR) can be expressed as [1]:

$$\text{TBR} \geq (1 + G_o) + \sqrt{\Delta_G^2 + \Delta_p^2}$$

where G_o is the breeding margin for the reference design, Δ_G is the uncertainty in estimating $(1 + G_o)$, and Δ_p is the uncertainty due to nuclear data, calculational methods, and geometrical models. Here, it is assumed that the design is well defined and no more physics or engineering changes to the reference design are anticipated.

The breeding margin G_o is necessary to mainly supply the inventory for startup of other fusion reactors, maintain the equilibrium hold-up inventory, provide adequate reserve storage inventory, and compensate for the radioactive decay of T between production and use. The T inventory depends largely on several system parameters such as T fractional burn-up in plasma, T thru-put to the plasma, number of days of reserve inventory, and doubling time. The latter is defined as the time at which the T inventory in the storage is equal to the sum of the initial T inventory required for starting a new ARIES-type reactor and the minimum T inventory that must be kept in storage for continued reactor operation during interruption of any part of the tritium cycle. In the ARIES-II design, some approaches were identified to minimize the T inventory in

the fusion power core and on-site in the storage and plasma processing facilities. For instance, a high T burn-up fraction of 18% and a low T thru-put of 1.7 kg per full-power day were considered in order to lower the T inventory in the plasma exhaust processing system. The low T thru-put to the plasma and the short repair time (<1 day) for the T processing system help reduce the reserve storage inventory to ~ 0.5 kg. The equilibrium hold-up inventory in the fusion power core (first wall structure, divertor plates, and blanket) is estimated to be ~ 2 kg. On this basis, the supply inventory to start a new reactor should be ~ 2.5 kg. A doubling time of 5 y seems acceptable for the first-of-a-kind ARIES-type reactors and it can be increased to 7-10 y for the tenth-of-a-kind reactors.

To calculate the breeding margin G_o , a simple model was employed that accounts for the radioactive decay of T . In order to have a T inventory of 3 kg at the end of a 5 y period, the blanket production rate (P) for T can be calculated using the following expression

$$N(t) = \frac{P}{\lambda} (1 - e^{-\lambda t})$$

where $N(t)$ is the required T inventory (3 kg), λ is the T decay constant (0.562 y^{-1}), and t is the doubling time (5 y). In this case, the blanket production rate is estimated to be 1.9 g/day. Using a T thru-put of 1.3 kg/day (@ 76% availability), the breeding margin G_o ($P/\text{thru-put}$) amounts to 0.015 for the ARIES-II reference design.

As noticed, the quantity $(1 + G_o)$ is a function of several plasma and engineering components parameters that play a key role in the tritium cycle. To accurately quantify the uncertainty Δ_G associated with $(1 + G_o)$, a rigorous statistical/uncertainty analysis is required for the reactor parameters of interest. Unfortunately, the early R&D stage the fusion program is in now does not permit the development of such analysis. Alternatively, a reasonable approach that relies on "engineering" judgment was employed. Based on the analysis presented in Ref. 1, the uncertainty Δ_G for the ARIES-II parameters is estimated to be $<4\%$.

As mentioned before, the calculated TBR for the ARIES-II design is subject to uncertainties (Δ_P) due to approximations and/or errors in the various elements of the calculations such as nuclear data, calculational method, and geometric representation. The uncertainties in the nuclear data include the uncertainties for the basic data, data processing and representation, and tritium production cross sections for lithium. The largest source of uncertainty in nuclear data is that for the basic nuclear data, which is $\sim 6\%$ for the Li/V blanket [2, 3], and the other uncertainties are small ($\sim 1\%$). The transport calculations and the 1-D modeling of the blanket involved some geometrical approximations and the associated error is estimated to be in the order of $\sim 3\%$. Hence,

the uncertainties in the calculational elements total to 10% which is significant and will have a major impact on the breeding requirements.

In this regard, the ARIES-II design will attain fuel self-sufficiency if the calculated TBR is equal to or exceeds 1.12. It should be stressed that the TBR must be calculated for the reference design and the effects of the blanket coverage, Li burn-up, penetrations, assembly gaps and side walls should all be included in the computed TBR. Also, it is of importance to point out the potentials the blanket design might have to enhance the breeding capability because of the possible existence of uncovered uncertainties in the system definition.

14.2.2. Optimum Blanket Parameters

A scoping analysis using the one-dimensional (1-D) code ONEDANT [4] was performed for the V/Li blanket to evaluate the impact of the blanket parameters on both the tritium breeding ratio (TBR) and neutron energy multiplication (M). A simple poloidal cylindrical model was utilized where a centimeter thick first wall (FW) was followed by a V/Li blanket of variable thickness and, then, a meter thick steel reflector/shield. In the analysis, the Li enrichment was varied between 7.42% (natural) and 50% ^6Li and the blanket thickness was gradually increased from 10 to 80 cm. The V structural content was kept fixed at 10% by volume. The results are illustrated in Figures 14.2-1, 14.2-2, 14.2-3, and 14.2-4. As Figure 14.2-1 indicates, the TBR increases with Li enrichment and peaks at higher enrichment for thinner blankets. In addition, M decreases as the enrichment increases for all blanket thicknesses and the effect of enrichment on the TBR and M is more pronounced for the thinner blankets. It should be mentioned that M includes the energy generated in both blanket and reflector/shield. As illustrated in Figure 14.2-3, the breeding level can effectively be controlled by adjusting the thicknesses of the inboard (i/b) and outboard (o/b) blankets. Since the overall size of the machine is much more sensitive to the dimensions of the i/b components, it was decided to consider a 20 cm thick blanket for the space limited inboard side and adjust the outboard blanket thickness to satisfy the breeding requirements.

The ARIES-II 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 incident on the top and bottom parts of the reactor is small relative to the i/b and o/b and these neutrons will be attenuated by the divertor plates leading to a further reduction in their contribution to tritium breeding.

FIGURES AT ENO

Figure 14.2-1. Effect of Li enrichment on TBR for different blanket thicknesses.

Figure 14.2-2. Effect of Li enrichment on M for different blanket thicknesses.

Figure 14.2-3. Effect of blanket thickness on TBR for different Li enrichments.

Figure 14.2-4. Effect of blanket thickness on M for different Li enrichments.

The cost of the breeder is an important factor that should be considered in selecting the Li enrichment. Currently, the natural Li cost ranges between 45 and 50 \$/kg. Enriching the Li to 20%, 50%, and 90% ^6Li increases the unit cost by factors of about 7, 20, and 40, respectively. Enriched Li will surely have a major impact on the blanket cost and the associated replacement cost. Thus, Li enrichment should be avoided and alternate ways must be pursued to achieve the breeding goal.

14.2.3. Overall Tritium Breeding Ratio and Energy Multiplication

The configuration of the reactor dictates that a realistic model to compute the overall TBR and M must be three-dimensional. The Monte Carlo codes (e.g., MCNP or MORSE) are capable of handling such geometry. However, due to lack of resources and time limitations for this study, an approximate method based on 1-D analyses has been established with the objective of ensuring that the errors associated with the final results are as low as can reasonably be achieved. Thus, the overall TBR and M have been estimated by coupling the 1-D results with the coverage fractions of the i/b and o/b blankets. This approach has proven to reasonably predict the overall values within a few percent [5] and, as mentioned before, provisions were made in the tritium breeding requirements for such uncertainties in the results.

The i/b and o/b blankets extend vertically up to ~ 3 and ~ 3.5 m from the midplane, respectively. The coverage fraction, representing the fraction of source neutrons going directly to each region, was determined by the NEWLIT code [6]. These fractions amount to $\sim 15\%$ and $\sim 70\%$ for the i/b and o/b blankets, respectively, and 15% for both the top and bottom divertor regions. Losses in available areas due to the presence of the plasma control ports and assembly gaps imply that the blanket coverage fraction should be modified by excluding the areas occupied by the various penetrations. The assembly gaps between adjacent blanket modules are 2 cm wide. For 16 blanket modules, the space taken by these gaps amounts to $\sim 2\%$ of the FW area. The outboard side has heating and current drive ports that occupy $\sim 5.8 \text{ m}^2$ ($< 2\%$) of the o/b FW area. The net effect of the ports and gaps on the overall TBR is about 2% reduction.

Calculations of the TBR were made for several preliminary configurations. The purpose of these calculations was to guide the design towards the final configuration that provides T self-sufficiency. The analysis and results given in this section, however, pertain only to the final design. In these calculations, the neutron reflection and spectral effects between the i/b and o/b blankets were taken into account by using a toroidal cylindrical geometry where both blankets were modeled simultaneously. In the model, the blankets

were followed by a meter thick metallic reflector/shield. The FW is composed of a centimeter thick Li flowing between 0.3 and 0.1 cm thick V layers. The blanket and shield are represented by homogeneous mixtures of Li/V (90/10 vol%) and SS/V/Li (80/15/5 vol%) respectively. The Li is natural and its density was taken to correspond to a mean temperature of 450°C. The i/b blanket thickness was kept fixed at 20 cm while the o/b blanket was gradually thickened till the breeding requirements are met. The initial calculations indicate that a 50 cm thick o/b blanket should result in an acceptable TBR. This was later confirmed by coupling the 1-D results with the actual coverage of each region, as shown below.

In the 1-D toroidal model, the neutron balance calculations show that 24% and 76% of the source neutrons impinge on the i/b and o/b regions, respectively. As a result, the 20 cm thick i/b blanket produces 0.27 T/fusion whereas the 50 cm thick o/b blanket has a larger production of 1.0 T/fusion. Based on these numbers, the local TBR is 1.15 for the i/b blanket and 1.3 for the o/b blanket. Modifying these values by the appropriate coverage fraction yields the contributions to the overall TBR of 0.17 from the i/b and 0.9 from the o/b. Even though the divertor region has no blanket, the Li coolant of the divertor shield will contribute to the breeding. A poloidal calculation for the divertor region indicates that the local TBR is 0.33 for the divertor shield which is made out of double-sized SS balls. For 15% coverage fraction, the divertor shield contributes 0.05 to the overall TBR. On this basis, the overall TBR amounts to 1.12 as detailed in Table 14.2-I. It should be mentioned that the ARIES-II design has the potential for achieving higher breeding by thickening the blankets, installing blanket in the divertor region, and/or enriching the Li to 15-20% ^6Li .

The same methodology used to estimate the overall TBR was followed to evaluate the neutron energy multiplication of the system. The end results show that the overall energy multiplication is 1.38. The reactor produces 1910 MW of fusion power and 80% of this power is carried by neutrons. Neutrons and gamma rays deposit their energies in the blanket/shield generating an additional 580 MW of thermal power, resulting in 2110 MW_{th} of total nuclear heating. The breakdown of the heating deposited in the FW, blankets, reflectors, and shields is given in Table 14.2-II. The reflector and shield designs are described in detail in the sections to follow. As noticed from the table, a large fraction of the power (55%) is deposited in the o/b blanket/reflector whereas the i/b blanket/reflector carries ~15% of the heat. This is due to the fact that the i/b blanket is thinner and has less coverage compared to the o/b blanket. It is also of interest to note that about 30% of the heating is generated in the shields. This is a significant amount of heat which cannot be discarded as a low grade heat without having a negative impact

Table 14.2-I.

TRITIUM BREEDING IN BLANKETS AND Li COOLED SHIELDS

	Inboard Blanket	Outboard Blanket	Divertor Shield
Blanket thickness (cm)	20	50	0
Coverage fraction	15%	70%	15%
Local TBR	1.15	1.30	0.33
Contribution to TBR	0.17	0.9	0.05
Overall TBR		1.12	

on the design. Therefore, a decision was made to run all the in-vessel components hot and to recover the nuclear heating deposited in all blanket/shield components as a high grade heat.

Table 14.2-II.

NUCLEAR HEATING GENERATED IN THE
VARIOUS REACTOR COMPONENTS

Nuclear Heating (MW)	Inboard	Outboard	Divertor	Total
FW/Blanket/Reflector	325	1153	—	1478
Shield	60	197	375	632
Total	385	1350	375	2110

14.3. RADIATION DAMAGE AND LIFETIME OF V STRUCTURE

There is some concern that the V structural material will not survive the intense radiation environment and will require frequent replacement over the 40 years of operation. From the radiation damage viewpoint, the lifetime of the V structural components of the FW, blanket, reflector and shield is determined by the level of swelling and embrittlement attainable during reactor operation. The atomic displacement level and the amount of helium generated have the most impact on these neutron-induced effects in the V alloys. Unfortunately, the information on radiation effects for V-base alloys is much more limited than for other metallic alloys. Most of the data exists for V alloys developed for use in fission reactors and efforts are continuing to optimize the V-base alloys for fusion reactors. The V5Cr5Ti alloy is the candidate structural material for the ARIES-II design. This alloy seems to possess high radiation resistance to fusion neutron damage. Even though insufficient data exist for this particular alloy, projections based on available information are used, as discussed in Section 12.2.

In ARIES-II, the lifetime of V is determined by the dpa level attainable during operation. The criterion adopted in this study is that no more than 200 dpa is desirable. The dpa, helium, and hydrogen production rates were calculated for both the i/b and o/b sides of the reactor and are listed in Table 14.3-I for the various components. As noticed, the o/b side exhibits higher damage due to the higher neutron wall loading. Although the i/b FW/blanket/reflector (FW/B/R), which are combined in a single unit, should be replaced on a less frequent basis, a decision was made to change out the i/b modules along with the o/b modules. Certainly, there will be an incremental cost associated with the early replacement of the i/b side, but this will be offset by the gain due to the higher availability of the system. The 200 dpa limit for V implies that the FW/B/R should be replaced every 4.2 y and the corresponding end-of-life (EOL) fluence is 16.4 MWy/m^2 . The maximum He and H levels attainable at the EOL of the FW/B/R are 1300 and 6300 appm, respectively. This will result in a few percent swelling and will not pose significant problems for the dimensional stability of the V structure.

The results indicate that at the reactor end-of-life (40 years), the shield meets the 200 dpa limit and, on this basis, the shield is considered a lifetime component and does not need replacement due to radiation damage. In fact, the lifetime protection of the shield was achieved by adjusting the thickness of the reflector so that the first layer of the shield has an acceptable damage level at end-of-life. This mandates the reflector to be 15 cm and 5 cm thick behind the 20 and 50 cm thick i/b and o/b blankets, respectively.

Table 14.3-I.

RADIATION DAMAGE TO V STRUCTURE

COMPONENTS	FW/B	Reflector	Shield
<u>Inboard</u>			
He production rate (appm/FPY*)	278	46	5
H production rate (appm/FPY)	1,345	247	22
Atomic displacement rate (dpa/FPY)	50	20	<6
Lifetime (FPY)	4	10	30
(CY)	5.3	13	40
EOL FW/B/R Fluence (MWy/m ²)	13		
<u>Outboard</u>			
He production rate (appm/FPY)	410	14	6
H production rate (appm/FPY)	1,960	77	34
Atomic displacement rate (dpa/FPY)	63	8	<6
Lifetime (FPY)	3.2	25	30
(CY)	4.2	33	40
EOL FW/B/R Fluence (MWy/m ²)	16.4		
*full power year (FPY)			

14.4. SHIELD DESIGN

The intent of this section is two folds: to design a radiation shield to protect the TF magnets and to design a biological shield around the torus for personnel protection during operation. In the analysis, the impact of shield performance on the design is assessed with a view to cost, complexity, fabricability, and safety. The shield is required to keep the radiation level below certain limits. As a matter of fact, these limits have a strong effect on the characteristics of the shield, in terms of thicknesses and compositions. To reduce the size of the shield, it is highly recommended to employ materials with high radiation tolerance in the viable components of the reactor. Moreover, it is essential to optimize the composition of the shield. Significant savings in the overall cost of the machine can result from the design of an efficient shield as it helps reduce the size of the reactor. Specifically, a centimeter less shield in the inboard side reduces the overall cost by 5-10 M\$, depending on the major radius of the device.

14.4.1. Magnet Radiation Limits

Besides the basic requirements, special considerations due to the radiation environment confront the magnet designers. The nuclear heating in the coils generated by the neutrons and gamma rays must be transported out of the winding packs with very little elevation of the conductor temperatures. This heating also determines the required capacity of the helium refrigeration system. Cumulative irradiation effects over the reactor life (40 years) are also important: degradation of critical current/temperature/field of superconductivity, increase in the resistance of the stabilizing Cu, and deterioration of the mechanical properties of the electrical insulation material in the windings.

As the radiation adds many problems to the magnet design, sufficient shielding should be placed between the plasma and magnet to keep the radiation effects below certain limits. These limits are set by the magnet designers to insure the proper performance of the TF coils. For instance, at the end of 30 full power years (FPY) of operation, the fast neutron fluence ($E_n > 0.1$ MeV) should not exceed 10^{19} n/cm² to avoid degradation of the critical properties of the Nb₃Sn superconductor material. It is undesirable to subject the magnets to a total nuclear heating above 50 kW to avoid excessively high cryogenic load to the cryoplant. A limit of 2 mW/cm³ is imposed on the peak nuclear heating in the winding pack. The end-of-life dose to the glass-fiber-filled (GFF) polyimide is limited to 10^{11} rads to ascertain the mechanical and electrical integrity of the insulator. The neutron-induced atomic displacement in the Cu stabilizer should not exceed 6×10^{-3}

dpa to avoid high increase in the Cu electric resistivity. It should be mentioned that the fluence and dose limits are at least a factor of 2-3 lower than the experimental values at which degradation of properties was observed. Our neutronics calculations indicate that the predominant magnet radiation limits are the EOL fast neutron fluence and the nuclear heat load to the magnets. Hence, the shield is optimized to primarily minimize these effects.

14.4.2. Optimization Analysis

The intent of this section is to design an efficient low activation shield to protect the TF magnets against radiation. This involves the selection of the shielding materials, assessment of the shielding capability of the various candidate materials, optimization of the composition of the shield, and determination of the thickness of the shield required to keep the radiation damage at the magnet below the permissible level.

The analysis employed the 1-D transport code ONEDANT with the P_3 legendre expansion for the scattering cross section and the S_8 angular quadrature set. The associated 46 neutron and 21 gamma group cross section library was derived from the ENDF/B-V evaluation. The analysis was mainly carried out for the i/b shield as it strongly affects the overall size of the machine. In the calculational model, the geometric configuration of ARIES-II was maintained. Special effort was devoted to include in the neutronics model all components, such as the blanket, vacuum vessel (V.V.) and cryostat, that have significant impact on the shield design. The radial build of the various components are shown schematically in Figure 14.4-1. All the in-vessel components are Li cooled. The V.V. is located outside the shield and is made of a low activation steel (Tenelon). Outside the V.V. is the TF magnet surrounded by the cryostat. The winding pack of the TF magnet includes Incaloy structure, Nb_3Sn superconductor, Cu stabilizer, Cu/Nb conduit, GFF polyimide insulator, and liquid He coolant at 57, 6, 24, 5, 4, and 4 vol.%, respectively. The cryostat consists of a 3 cm thick thermal insulator sandwiched between two SS plates, 1 cm thick each.

The different components of the blanket and shield were modeled in toroidal cylindrical geometry around the machine axis, permitting the representation of the inboard and outboard sides simultaneously. It is pertinent to mention that a poloidal model should not be employed, particularly for the inboard side, as it underestimates the damage at the magnets. The plasma shift toward the outboard is taken into account in modeling the neutron source. The distribution of the neutron wall loading in the poloidal direction is taken from the system code output. The average wall loading is 3.1 MW/m^2 and the

Figure 14.4-1. Schematic of inboard blanket/reflector/shield, V.V., and magnet.

distribution peaks at the outboard midplane at a value of 5.1 MW/m^2 . The inboard wall loading maximizes at the midplane at 3.15 MW/m^2 . The divertor region has the lowest wall loading of 2 MW/m^2 . The neutron source strength used in the transport calculations is consistent with these wall loadings.

The optimization analysis has focused on optimizing the major factors that influence the shield performance. These factors are the composition of the shield, material arrangement, and coolant content within the shield. To insure the reliability of the shield design, it is essential to select shielding materials that can stand the high radiation environment of the design. The shield is considered a semi-permanent component and should be designed to perform successfully over the entire lifetime of the machine (40 y). It should be quite unlikely that the shield requires repair or replacement but must be replaceable by practical means if that should become necessary. Thus, the shield is segmented toroidally into 48 modules and designed to be easily disassembled in order to provide ready access for maintenance and repair to the internal 32 blanket modules.

Besides V and steel, many materials have the potential to protect the magnets. Each material has some merits and drawbacks. The behavior of the material, particularly in a high radiation environment, may limit its use in the shield. For example, the tungsten is known as the best shielding material but its high decay heat and the production of the ^{187}W isotope cause some safety problems. The hydrogen compounds (like titanium hydride, boron hydride, organic, water, and polyethylene) are superior in slowing down the fast neutrons. Nevertheless, all hydrides decompose under irradiation and the high pressure of the dissociated hydrogen jeopardizes the integrity of the shield. An ideal situation from the shielding point of view is to use water (or borated water) in the shield. However, this would complicate the cooling system considerably due to the requirement for two independent primary cooling circuits. The borated materials (such as boron carbide, boron hydride, boronized carbon, and borated water) possess remarkably high neutron absorption cross sections particularly for low energy neutrons, a property of great importance in magnet shielding. Among these borides, boron carbide is the best for having the combined features of excellent shielding capability, stability under irradiation, and ease of fabrication. One concern regarding the B_4C (as well as all other borides) is the tritium generated in the boron. To decrease the in-shield tritium inventory, the B_4C should be situated close to the back of the shield. This constraint turned out to be beneficial for the B_4C since the neutron spectrum therein is relatively soft and the neutron absorption capability of B_4C is high. The lead has been known for its good ability to reduce magnet heating by absorbing the gamma radiation. However, lead has a few drawbacks of which one should be aware. Lead increases the neutron fluence (compared

to steel) because it multiplies the neutrons, starts to deform at $\sim 220^{\circ}\text{C}$ and, thus, has to be canned in a structural material, and generates polonium, which is a safety concern.

The first set of calculations demonstrates the shielding performance of several materials relative to V. To integrate the environmental constraints into the design from the beginning, it is important to select shielding materials that do not become strongly radioactive when bombarded by neutrons. This will eliminate the need for deep geological burial of the radioactive waste and avoid high radiation exposures at the site boundary during an accident. There are a few low activation metallics that are readily available to be used in fusion reactors. These are Tenelon, modified HT-9 (MHT-9), and Fe1422 steels. The option of using these materials as filler materials with a V structure was investigated. At the present time, there is no structural role envisioned for the filler material in the shield and, therefore, a substantial cost reduction will result from the use of such less costly filler materials. The impact of these materials on magnet damage, TBR, and M is shown in Table 14.4-I. As noticed, the Tenelon has the best shielding capability as it gives the lowest magnet damage. Furthermore, it results in the highest energy multiplication since it contains a substantial amount of manganese (14.5 wt%). On the other hand, MHT-9 gives a slightly higher TBR but the thicker shield needed to protect the magnet will more than offset the gain from higher breeding. Clearly, Tenelon is the best candidate filler material for the shield and it is anticipated that the combined effect of the good shielding properties, low unit cost, and high M will have a positive economic impact on the overall design. It should be mentioned that Tenelon has a slightly higher decay heat compared to others, but this does not seem to present a problem even in the event of a loss of coolant accident, as discussed in Chapter 13.

The i/b zone was configured into 3 separate regions: blanket, reflector, and shield. V is used as the primary structural material in all components where 10 and 15 vol% structure are foreseen enough to support the blanket and reflector/shield, respectively. The 20 cm thick i/b blanket is followed by a 15 cm thick reflector composed of 15 vol% V, 10 vol% Li, and 75 vol% Tenelon filler. The Li content in the reflector is dictated by the cooling requirements. The thickness of the reflector is determined such that the V structure of the shield has an acceptable neutron-induced damage at EOL (see Section 14.3). The composition of the shield was then optimized to reduce the radiation damage at the magnet. First, the Tenelon filler in a 70 cm thick shield was traded for Li coolant and the results are illustrated in Figure 14.4-2. As mentioned earlier, the fast neutron fluence and the peak nuclear heating, which is an indicative for the total heating in the magnets, are the most demanding radiation limits. In other words, once these limits are met all the other limits will automatically be satisfied. Although the fast neutron fluence and

Figure 14.4-2. Impact on magnet damage of the Li coolant content in the steel shield.

Table 14.4-I.

**NUCLEAR PERFORMANCE OF SOME LOW ACTIVATION
MATERIALS RELATIVE TO TENELON**

Filler Material	Peak Magnet Heating	Peak Magnet Fluence	TBR	M
Tenelon	1	1	1	1
MHT-9	2.6	2.4	1.05	0.93
Fe 1422	1.3	1.4	1.003	0.995
V	4.8	1.6	1.09	0.94

nuclear heating minimize at no coolant in the shield, 5 vol% Li was considered to meet the cooling requirements.

B_4C was then introduced to enhance the performance of the SS shield. At the beginning, there are some issues related to the B_4C that need to be discussed. Firstly, the SS shield and the B_4C shield must be modeled as separate layers, the same way they will actually be manufactured and installed in the reactor. The homogenization of the various layers of the shield underpredicts magnet damage and leads to a nonoptimum design. Secondly, it is advisable to use B_4C bricks (230 \$/kg) instead of B_4C powder (62-80% dense, 45 \$/kg) despite its lower cost, particularly if B_4C is used in the i/b side. The reason is that the B_4C powder needs a larger shielding space. This implies bigger magnets and a bigger overall machine. The net effect is that the cost savings for the cheaper B_4C powder will be more than offset by the increase in shield thickness (~10 cm), and, thus, overall machine size, due to its low density. Furthermore, from the reliability viewpoint, the solid form is preferable as the uniformity of the shield is assured. Lastly, the option of enriching the B_4C in ^{10}B should be avoided. The slight improvement in the shielding performance and the corresponding thinning in shield will not justify the significant increase in the cost of the enriched material (factor of 10 higher cost for 90% ^{10}B). The implications of including B_4C in the shield are shown in Figure 14.4-3. The total shield thickness was kept fixed at 70 cm, and the back layer of the SS shield was gradually replaced by B_4C shield ($B_4C/V/Li$ at 80/15/5 vol% and 90% dense natural B_4C). As is evident, the B_4C significantly reduces the magnet damage and a 47 cm thick SS shield followed by a 23 cm thick B_4C shield is an optimum arrangement for the shield.

Figure 14.4-3. Effect of replacing the back layer of the steel shield by B_4C shield.

An attempt was made to include a thin layer of lead at the back of the V/SS/B₄C/Li shield, keeping the total shield thickness fixed at 70 cm. Our results, shown in Figure 14.4-4, indicate that the Pb acts to raise the fluence and decrease the heating. Since the fluence is the design driver for the shield and the heating is already well below the design limit, it is clear that there is no benefit for adding Pb to the shield. As a final point, it is of interest to show the effect the blanket has on magnet damage. Since the blanket has less shielding capability compared to the shield, the damage at the magnet decreased upon replacing the 20 cm thick i/b blanket by shielding materials, as illustrated in Figure 14.4-5. The presence of the 20 cm thick i/b blanket raised the magnet damage by a factor of 3-5. This means that about 10 cm of the i/b space could be saved if the i/b blanket is eliminated.

14.4.3. TF Magnet Shielding

The optimal shield design presented thus far essentially pertains to the inboard side of the reactor, but found to be applicable to the other regions. However, the outboard and divertor shielding spaces are less constrained relative to the inboard. The reason is that the maintenance scheme requires the outer legs of the TF magnets to be located at a large enough distance so that the blanket modules can be pulled out radially for replacement. This resulted in ambient space behind the blanket (see Chapter 10 for a vertical cross section of the reactor). It is, therefore, not necessary to utilize a high efficient shield therein and, instead, a pure SS shield can protect the magnet. Admittedly, the SS shield will be thicker and heavier but certainly less expensive than the SS/B₄C shield. In the o/b side, the 50 cm thick blanket is followed by a 5 cm thick reflector, and then a meter thick SS shield (SS/V/Li at 80/15/5 vol%) to virtually zero out the nuclear heating in the outer legs of the TF coils. The magnet damage levels are listed in Table 14.4-II. The o/b shield is made out of consecutive SS layers and Li cooling channels. An alternative option, judged easier from the mechanical point of view, could be using SS balls instead of SS blocks. This option was abandoned to mainly reduce the Li inventory and to limit the volume of the bulky o/b shield. On the contrary, the use of double-size SS balls (with 80% packing fraction) is more suitable for the divertor region to simplify the design of the critical area behind the divertor plates. The effective thickness of the Li cooled V divertor plates and their support structure is estimated to be 5 cm. The analysis shows that the divertor shield (SS/V/Li at 67/15/18 vol%) must be 105 cm thick to adequately protect the magnets. The damage levels at the top/bottom parts of the TF coils are given in Table 14.4-II.

Figure 14.4-4. Effect of replacing the back layer of the SS/B₄C shield by Pb shield.

Figure 14.4-5. Reduction in magnet damage due to replacing the inboard blanket by shielding materials.

Another data of interest is the total nuclear heating deposited in the 16 TF coils. This heat will be removed from the magnets at 4.2 K by a cryogenic plant. The nuclear heating in the magnets was estimated taking into account the changes in the shield thickness, the poloidal variation in the neutron wall loading, and the toroidal coverage of the magnets. Table 14.4-III details the nuclear heating in the various parts of the magnets. Most of the inner and outer legs heating is deposited at the middle part around the midplane. The reason is that the FW curves to follow the plasma contour and, thus, the radial thickness of the shield (which is minimum at the midplane) gradually increases along the poloidal direction. Also, the neutron wall loading drops from peak values of 3.15 and 5.1 MW/m² at the midplane to minima of 1.5 and 2 MW/m² at the upper/lower ends of the i/b and o/b sides, respectively, and this helps reduce the heating even further. As indicated in the table, negligible heating is deposited in the outer legs of the magnets and the top/bottom heating is quite reasonable. The integrated nuclear heating totals to 5.5 kW. This heat is removed from the magnets at a cryoplant efficiency of 310 W per W [7] and, therefore, the required cryogenic load amounts to 1.7 MW. It appears that this is an acceptable load as it represents only 0.2% of the net electric power.

14.4.4. Penetration Shielding

The shield has been treated so far as being free of penetrations. In general, the presence of a penetration indicates a weakening in the shield performance as neutrons stream through and increase the radiation levels around the penetrations. Therefore, supplementary shielding must be added to the perimeter of these penetrations to ensure that the radiation induced effects will not lead to unacceptable degradation in the expected performance of the magnets and external components. In some cases, the penetration shielding may interface with the requirements of the device such as those for machine accessibility. This necessitates the use of highly efficient shield in some localized areas that might have restricted space or limited accessibility.

The major penetrations of the device are the divertor pumping ducts, gaseous divertor slots, and current drive ports. Assembly gaps between blanket/shield modules also play an important role in enhancing streaming. The major neutronics concerns associated with these penetrations are the local damage hot spots at the magnets, radioactivity level of the external components, radiation induced-noise in the sensitive electronic equipment, activation of the reactor covering gas, and biological dose level of occupational personnel. Proper analysis of the streaming problem requires 3-D Monte Carlo calculations which is usually an expensive and time-consuming method. Due to the lack of resources, the

Table 14.4-II.

**KEY BLANKET/SHIELD PARAMETERS AND
RADIATION EFFECTS AT THE TF MAGNETS**

Region	Inboard	Outboard	Divertor
Thicknesses (cm):			
Blanket	20	50	—
Reflector	15	5	—
Shield:	70	100	105
SS shield	47	100	105
B ₄ C shield	23	—	—
Vacuum vessel	10	10	10
Total	115	165	115
Peak neutron wall loading (MW/m ²)	3.15	5.1	2
Peak fast neutron fluence to Nb ₃ Sn (n/cm ² @ 30 FPY)	7.5×10^{18}	2×10^{18}	9×10^{18}
Peak nuclear heating in winding pack (mW/cm ³)	0.6	0.1	0.5
Peak dose to GFF polyimide (rads @ 30 FPY)	9.4×10^9	2×10^9	8×10^9
Peak dpa in Cu stabilizer (dpa @ 30 FPY)	3.3×10^{-3}	8×10^{-4}	3.3×10^{-3}

Table 14.4-III.

TOTAL NUCLEAR HEATING IN THE 16 TF MAGNETS

	Inner Legs	Outer Legs	Top and Bottom
Average wall loading (MW/m ²)	2	3.5	2
Poloidal height or length (m)	6	7	2
Nuclear heating per unit length (kW/m)	0.7*	0.2*	0.9
Integrated heating (kW)	2	0.5	3

*At midplane

most critical streaming problems were qualitatively quantified and rough estimates for the required shielding were obtained by rescaling from the 3-D results of previous reactor studies. Although more rigorous analysis may result in some adjustments to the penetration requirements, the shielding parameters will probably not change greatly.

Concerning the gaseous divertor design, there are two slots which penetrate into the divertor shield. Fortunately, there is sufficient space at the top/bottom of the reactor to add extra shield behind these slots. This is in fact true for the outer slot only and shielding of the inner slot is a real concern. As a matter of fact, the plasma parameters, specifically the triangularity, play an important role in providing enough shielding space behind the inner slot in particular. The reason is that the low triangularity provides sufficient space for shielding as it moves the crossing point of the magnetic flux lines outward. In this regard, the final design was arrived at after several iterations between the neutronics and physics in order to meet the shielding requirements, physics constraints, and assembly/maintenance demands. To satisfy the shielding requirements, it is essential to provide 105 cm of V/SS/Li shield behind the slots to protect the magnets. This shield estimate is conservative as it takes no credit for the shielding effectiveness of the divertor plates and their support structure. The latest CAD drawings showed that provision of such shielding space can be marginally achieved without difficulty. Should unforeseen shielding problems arise, the more efficient shield (V/SS/B₄C/Li) would be employed.

In the double null configuration, the plasma ashes will be pumped out from both the upper and lower divertors. The design calls for 16 ducts at each of the top and bottom of the reactor. These ducts protrude the o/b shield horizontally between the outer legs of the TF coils. The toroidal width of each duct is in the order of 20 cm. The design is characterized by several features that help reduce the streaming through these ducts. Firstly, the plasma center is shifted outward and, thus, the duct openings are out of sight of most of the source neutrons. Secondly, the top/bottom ends of the o/b blanket/shield are extended inward as much as practical to screen the duct from a direct view to the plasma. The combined effect of these features is to preclude the uncollided source neutrons from streaming directly into the ducts. Even though, some neutrons will be reflected into the ducts from the opposite surfaces and increase the flux level inside the ducts. It is estimated that about 50 cm thick lateral shield (V/SS/Li) is required around the ducts to protect the sides of the outer legs of the TF magnets against streaming radiation. This estimate is based on the results of 3-D streaming analysis from a previous reactor study [8]. It seems that there is a sufficient space to accommodate this shield without any access problem.

There are 3 RF systems for heating and current drive at steady state and startup operations. The essential features of these ducts include opening areas of 0.6, 1 and 2 m² at the FW, duct axes at the o/b equatorial plane and at 30 degree above midplane, launcher modules and antenna array replacing 20-100 cm of the o/b blanket/shield, and feeding transmission lines embedded in the o/b shield. A more detailed description of the plasma control system is given in Chapter 4. The design offers the flexibility of bending the transmission lines to greatly attenuate the streaming radiation. Nevertheless, the presence of the apertures containing the antenna array mandates the addition of localized shields behind these apertures (10-70 cm thick) to assure the proper protection of the magnets. There is enough space between the back of the shield and the V.V. to accommodate this shield.

Along the toroidal circumference, the blanket/shield are divided into modules to facilitate the disassembly and maintenance of the in-vessel components. Adjacent modules are separated by a 2 cm wide assembly gap. There is a concern that neutrons will penetrate through these gaps and give rise to significant damage levels at the magnets. To lessen the severity of this problem, the blanket is segmented into 16 modules whereas the shield is subdivided into 32 units and the centerlines of the blanket and shield gaps were misaligned to eliminate any direct line-of-sight between the plasma and the magnets. The final layout of the blanket/shield accords with these requirements (refer to Chapter 10).

14.4.5. Biological Shielding

The biological shield is required for radiation protection of the occupational personnel. A concrete shield has been designed around the periphery of the torus to permit personnel access during reactor operation. The criterion adopted for personnel access is to limit the biological dose rate outside the shield to 0.5 mrem/h during operation. This is consistent with the DOE guidelines for the protection of workers as it reflects a factor of 5 reduction in the absolute limit [9].

The operational dose analysis employed 1-D toroidal cylindrical geometry to calculate the neutron and gamma fluxes outside the biological shield. The flux-to-dose conversion factors were taken from Appendix H of Ref. 10. The maximum dose will occur at the midplane of the outboard side between the outer legs of the TF magnets. Thus, the neutronics model included the blanket, shield, and V.V. while the TF magnets were excluded. The need for a sufficient space behind the V.V. for blanket maintenance requires the biological shield to enclose the torus at a radius of 15 m. The biological shield was chosen to be steel-reinforced concrete and composed of 90 vol% concrete, 5 vol% mild steel, and 5 vol% void/He coolant. The constituents of the concrete are 52% O, 34% Si, 6% Al, 5.6% Ca, 1.4% Fe and 1% H, by weight.

The attenuation of the dose with the shield thickness is demonstrated in Figure 14.4-6. The results indicate that 2.5 m thick steel-reinforced concrete limits the biological dose outside the shield to 0.5 mrem/h during operation. As noticed, the dose inside the concrete shield is quite high ($> 10^7$ mrem/h) and the corresponding neutron flux level is $\sim 10^9$ n/cm²·s. In fact, there is no restriction on the dose level in the reactor hall since all in-vessel components will be maintained remotely and the magnet shield design does not allow for hand-on maintenance or personnel access into the reactor hall shortly after shutdown.

14.5. ACTIVATION ANALYSIS

14.5.1. Introduction

Activation analysis has been performed in this study to identify the possible safety, environmental and radwaste advantages of ARIES-II. We have investigated several activation-related issues for the reactor structure. The activity, decay heat and biological hazard potential (BHP) have been calculated for up to 1000 years following shutdown. Evaluation of the structure activity and biological hazard potential are needed to calculate the

Figure 14.4-6. Attenuation of operational biological dose with steel-reinforced concrete.

potential effects of radioactive inventory release in the event of an accident. In addition, results of the decay heat calculation are essential to examine the thermal response of the reactor blanket and shield following a loss of coolant accident (LOCA).

Another issue that have been examined in this analysis is the waste disposal rating (WDR) of the reactor structure at the end of its life-time. The waste disposal rating is needed to determine if the structure would satisfy the regulations criteria for shallow land burial as a low level waste (LLW).

14.5.2. Calculational Procedure

Calculations for a one-dimensional toroidal cylindrical geometry model were conducted using the DKR-ICF computer code [11] with activation cross sections taken from the ACTL [12] library. The neutron transmutation data used is in a 46 group structure format. The decay and gamma source data is taken from the table of isotopes [14] with the gamma source data being in 21 group structure format. The reactor is assumed to operate continuously for 30 full power years (FPY) which corresponds to 40 years of operation at 75% availability. While the reactor blanket was assumed to be replaced every three full power years, the shield was assumed to stay in place for the duration of the reactor life-time.

While, assuming continuous operation does not affect the calculated activities for radionuclides with half-lives much less than the shortest period of continuous operation or much greater than the reactor lifetime, the radioactive inventory of radionuclides with intermediate half-lives is overestimated by the inverse of the reactor availability [13]. Hence, our results for the radioactive inventory could be up to 33% conservative.

The neutron flux used for the activation calculations was generated by the one-dimensional discrete ordinates neutron transport code ONEDANT [4] using the ENDF/B-V cross section data. The peak neutron wall loadings are 3.4 and 5.75 MW/m² for the inboard and the outboard regions, respectively. The calculations were performed using the average neutron wall loadings with an average to peak ratio of 0.75.

The structure activation results were utilized in the radwaste classification performed using the WDR [15] computer code. The decay heat results were utilized in determining the thermal response of the shield following a loss of coolant accident (Chapter 13 of this report). Finally, the structure activation results were used in the safety analysis calculations presented in Chapter 15. The elemental compositions of the vanadium alloy (V-5Cr-5Ti) and the low activation austenitic steel (Tenelon) used in the calculations

are those presented in the Blanket Comparison and Selection Study (BCSS) report [16] (with minor adjustment in the case of the Vanadium alloy).

14.5.3. Structure Activity, Decay Heat and BHP

The total activities in ARIES-II inboard and outboard regions at shutdown are 1203 MCi and 2645 MCi, respectively. Since the outboard region volume is more than three times the inboard volume and the neutron wall loading on the outboard is about 70% higher than its value on the inboard region, the outboard activity dominates the total activity in ARIES-II at all times following shutdown.

The shield's short-term activity after shutdown (≤ 1 day) is dominated by ^{51}Cr ($T_{1/2} = 27.7$ day), ^{54}Mn ($T_{1/2} = 312.2$ day), ^{56}Mn ($T_{1/2} = 2.6$ hr) and ^{187}W ($T_{1/2} = 23.9$ hr). On the other hand, the blanket's short-term activity (following its replacement) is dominated by ^{48}Sc ($T_{1/2} = 43.7$ hr), ^{51}Cr , ^{47}Sc ($T_{1/2} = 3.349$ day) and ^{45}Ca ($T_{1/2} = 162.7$ day). In the period between 1 day and 1 year after shutdown, ^{54}Mn and ^{60}Co ($T_{1/2} = 5.27$ yr) dominate the activity induced in the shield. During a similar period of time, the blanket's activity is dominated by ^{49}V ($T_{1/2} = 337$ day), ^{45}Ca and ^{46}Sc ($T_{1/2} = 83.81$ day). Finally, the long-term activities induced in both the shield and blanket comes from the steel components and are dominated by ^{14}C ($T_{1/2} = 5730$ yr), $^{93\text{m}}\text{Nb}$ ($T_{1/2} = 16.1$ yr), ^{94}Nb ($T_{1/2} = 2 \times 10^4$ yr) and ^{93}Mo ($T_{1/2} = 3.5 \times 10^3$ yr).

Figure 14.5-1 shows the total activity induced in the different regions of ARIES-II as a function of time following shutdown. The total structure activity (inboard plus outboard) drops to 396 and 59 MCi in one day and one year following shutdown, respectively.

The temporal variation of decay heat after shutdown is shown in Figure 14.5-2. The decay heat generated in ARIES-II is almost dominated by the same isotopes that dominate the level of activity in the reactor after shutdown. ^{56}Mn and ^{52}V ($T_{1/2} = 3.76$ min) produce most of the decay heat generated in the shield within the first 8 hours. Within the first year after shutdown, ^{56}Mn and ^{60}Co are the major sources of decay heat. The long-term decay heat is governed by the decay of ^{94}Nb and $^{108\text{m}}\text{Ag}$ ($T_{1/2} = 130$ yr). In the mean time, the short-term decay heat generated in the blanket is due to ^{48}Sc and ^{52}V . ^{46}Sc and ^{49}V are the dominant nuclides up to one year following the blanket replacement or the reactor shutdown. ^{94}Nb and ^{14}C dominate the decay heat generated in the blanket several hundred years following the end of its life-time.

The total decay heat generated in ARIES-II at shutdown is 53 MW and drops to 3.23 MW in one day and 0.3 MW in one year. Figure 14.5-3 shows the total integrated

Figure 14.5-1. Activity induced in ARIES-II structure.

Figure 14.5-2. Decay heat induced in ARIES-II structure.

decay heat in the different regions of ARIES-II during the first 2 months following shutdown. One week after shutdown, the values of the integrated decay heat generated are 548 and 1298 GJ for the inboard and outboard regions, respectively. These results are useful for predicting the thermal response of the shield to a LOCA.

The biological hazard potential was calculated using the maximum permissible concentration limits in air for the different isotopes according to the NRC regulations specified in 10CFR20 [9]. The BHP in air as a function of time after shutdown is shown in Figure 14.5-4 with the contributions of the inboard and outboard regions indicated. The total BHP in ARIES-II at shutdown is 388×10^6 km³ air with about two-third of it contributed by the outboard region. The short-term BHP is dominated by ⁵⁴Mn, ⁵⁶Mn and ⁵²V in case of the shield and ⁴⁹V and ⁴⁸Sc in case of the blanket. While ⁶⁰Co and ⁵⁴Mn are the major sources of mid-term BHP (≤ 10 years) generated in the shield, ⁴⁹V is responsible for most of the BHP in the blanket. Finally, in addition to ⁹⁴Nb, the long-term BHP is produced by ^{108m}Ag and ⁹³Mo in case of the shield and blanket, respectively.

14.5.4. Radwaste Classification

The radwaste of ARIES-II structure has been evaluated according to both the NRC 10CFR61 [17] and Fetter [18] waste disposal concentration limits (WDL). The 10CFR61 regulations assume that the waste disposal site will be under administrative control for 100 years. The dose at the site to an inadvertent intruder after the 100 years is limited to less than 500 mrem/year. The waste disposal rating (WDR) is defined as the sum of the ratio of the concentration of a particular isotope to the maximum allowed concentration of that isotope taken over all isotopes and for particular class. If the calculated $WDR \leq 1$ when Class A limits are used, the radwaste should qualify for Class A segregated waste. The major hazard of this class of waste is to individuals who are responsible for handling it. Such waste is not considered to be a hazard following the loss of institutional control of the disposal site. If the WDR is > 1 when Class A WDL are used but ≤ 1 when Class C limits are used, the waste is termed Class C intruder waste. It must be packaged and buried such that it will not pose a hazard to an inadvertent intruder after the 100 year institutional period is over. Class C waste is assumed to be stable for 500 years. Using Class C limits, a $WDR > 1$ implies that the radwaste does not qualify for shallow land burial.

Fetter developed a modified version of the NRC's intruder model to calculate waste disposal limits for a wider range of long-lived radionuclides which are of interest for fusion researchers than the few that currently exist in the current 10CFR61 regulations. Fetter's

Figure 14.5-3. Integrated decay heat in ARIES-II structure.

Figure 14.5-4. Biological hazard potential in ARIES-II structure.

model included more accurate transfer coefficients and dose conversion factors. However, while the NRC model limits the whole body dose to 500 mrem or the dose to any single organ (one of seven body organs) to 1.5 rem, Fetter limits are based on the maximum dose to the whole body only.

The different radionuclide specific activities calculated by the DKR-ICF code were used to calculate the waste disposal ratings. The waste disposal ratings for Class A and Class C low level waste (LLW) are shown in Figure 14.5-5 and Figure 14.5-6 for the shield and blanket, respectively. The results in the figures are given for both Class A and Class C with the activities averaged over the total volume of the shield and blanket of both the inboard and outboard regions. The 10CFR61 Class A WDR is given after a waiting period of about 10 years to allow for the specific activity of short-lived nuclides ($T_{1/2} \leq 5$ years) to drop below 7000 Ci/m^3 . The 7000 Ci/m^3 limit is 10 times larger than the limit specified by the NRC for Class A disposal of short-lived nuclides where the waste form is not specified. By comparison with other isotopes for which limits are given for different waste forms, the factor of 10 is used for isotopes contained in metal waste. Since, the NRC regulations do not specify any limit for short-lived activity for Class C LLW, the Class C WDR values were calculated after 1 year cooling period for both 10CFR61 and Fetter limits.

The contributions from the different radionuclides to the WDR are shown in both figures. ^{94}Nb ($T_{1/2} = 20,000 \text{ yr}$), which is produced from ^{93}Nb or ^{94}Mo , is the major contributor to the waste disposal rating for both Class A and Class C. Other major contributor to Class A is the ^{60}Co ($T_{1/2} = 5.27 \text{ yr}$) produced from the cobalt, nickel and copper impurities in the steel. ^{14}C ($T_{1/2} = 5730 \text{ yr}$) produced from ^{14}N and ^{17}O is the other major contributor for the Class C rating if the 10CFR61 limits are used.

If Fetter limits are used, ^{108m}Ag ($T_{1/2} = 130 \text{ yr}$) produced from ^{107}Ag becomes a major contributor to the Class C waste disposal rating for the cases of the inboard blanket and the outboard shield. ^{26}Al ($T_{1/2} = 7.3 \times 10^5 \text{ yr}$) produced from ^{27}Al and ^{208}Bi ($T_{1/2} = 3.68 \times 10^5 \text{ yr}$) produced from ^{209}Bi are the other major contributors to the outboard blanket and inboard shield, respectively.

It is concluded that except for the outboard blanket which would qualify as Class A low level waste, the rest of the reactor structure would only qualify for Class C rating. The outboard blanket Class A rating value of 0.95 is based on allowing it to cool down for about 10 years following the end of its life-time.

Figure 14.5-5. Waste disposal rating for the ARIES-II shield.

Fig 14-2.1

Fig 14-2.2

Fig 14-2.4

Fig 14.4-1

Thickness (cm)	
50	Magnet
5	Cryostat
5	Gap
10	V.V.
2	Gap
	Shield
2	Gap
	Reflector
	Blanket
	FW
	Plasma

Fig. 14.4-2

Fig 14-4-3

Fig 14.4-5

Fig 14.4-6

Fig. 14.5-1

Activity Induced in ARIES-II Structure

Fig. 14-5-2

Decay Heat Induced in ARIES-II Structure

Fig. 14.5-3

Integrated Decay Heat in ARIES-II

Fig. 14.5-4

Biological Hazard Potential Induced in ARIES-II

Fig. 14.5-5

Waste Disposal Rating for ARIES-II Shield

Fig. 14.5-6

Waste Disposal Rating for ARIES-II Blanket

B Appendix: ARIES-IV Neutronics

FIGURES IN PROGRESS

Chapter 19

ARIES-IV NEUTRONICS

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19. ARIES-IV NEUTRONICS

19.1. INTRODUCTION

The issues of attaining tritium self-sufficiency, achieving adequate magnet protection, and integrating the environmental constraints into the design are the dominant themes of the ARIES-IV blanket and shield design. Since the ultimate goal of this study is to design an attractive, reliable, safe, and cost competitive tokamak reactor, the environmental constraints and safety features were integrated in the blanket and shield design from the beginning with a view to cost, complexity, fabricability, reliability and integrability. The major neutronics concerns that require quantification include the determination of the tritium breeding potential of the blanket, radiation damage to the structural components, shielding requirements for the magnets, peaking due to streaming through penetrations, activation and waste disposal characteristics, and requirements for occupational personnel protection during reactor operation.

A blanket is required to breed a sufficient amount of tritium that ensures the self-sustaining operation of the DT fueled reactor. The blanket utilizes Li_2O as a breeder and the low activation SiC/SiC composites as a structural material. The dimensions of the blanket were dictated by the breeding requirements. It is found that the uncertainties due to nuclear data and modeling are significant and have a major impact on the breeding requirements. Allowances have been made to accommodate these uncertainties and to meet other constraints and an overall tritium breeding ratio of 1.12 seems adequate for ARIES-IV. The Li_2O /SiC blanket will not provide tritium self-sufficiency unless a neutron multiplier is incorporated in the design. Section 19.2 documents the details of the design requirements for tritium breeding and the optimal blanket parameters.

The severe neutron environment associated with the ARIES-IV device indicates that radiation-related issues are significant considerations in the blanket and shield design. There are some radiation damage concerns for the SiC/SiC composites structure and the Be neutron multiplier. The tendency to produce large amount of helium and hydrogen gases and the subsequent burn-up and swelling of the materials are major factors in determining the lifetime of the different components, as explained in Section 19.3.

The blanket is followed by the shield which is primarily used to protect the superconducting toroidal field (TF) magnets against radiation. Since the dimensions of the shield

directly affect the size and cost of the machine, an extensive analysis was performed to optimize the shield composition and performance. A variety of shield options was examined and the ability of various materials to protect the different reactor components was assessed, as explained in Section 19.4.

The radiological issues of the design are addressed in Section 19.5. The decay heat, radioactive inventory, and waste disposal level are quantified. Finally, the conclusions to be drawn from this work are summarized in Section 19.6.

19.2. BLANKET DESIGN

19.2.1. Tritium Breeding Requirements

Since tritium (T) is one of the materials which has a very limited resource, it must be produced in a lithium blanket that surrounds the plasma. Thus, the primary goal of the blanket design is to attain tritium self-sufficiency that ensures the self-sustaining operation of the DT fueled reactors. The requirements to achieve this goal depend on some plasma physics parameters, several uncertainties in the design, and the technical performance of specific reactor components. The determination of these uncertainties and parameters is essential in order to define the excess margin in the breeding potential necessary to achieve tritium self-sufficiency. In ARIES-IV, a serious effort was developed to lower the breeding requirements by increasing T burn-up in the plasma, ensuring extremely low T losses, achieving low T inventory in all subsystems, and requiring high reliability and fix rate for the T processing systems. It appears that with the existing technology, a system with negligible T leak can be achieved to guarantee a safe and reliable design.

The tritium breeding ratio (defined as the amount of T generated in the blanket per fusion reaction) in a self-sustained power reactor must exceed unity by a margin G_o . Thus, the required tritium breeding ratio (TBR) can be expressed as [1]:

$$\text{TBR} \geq (1 + G_o) + \sqrt{\Delta_G^2 + \Delta_p^2}$$

where G_o is the breeding margin for the reference design, Δ_G is the uncertainty in estimating $(1 + G_o)$, and Δ_p is the uncertainty due to nuclear data, calculational methods, and geometrical models. Here, it is assumed that the design is well defined and no more physics or engineering changes to the reference design are anticipated.

The breeding margin G_0 is necessary to mainly supply the inventory for startup of other fusion reactors, maintain the equilibrium hold-up inventory, provide adequate reserve storage inventory, and compensate for the radioactive decay of T between production and use. The T inventory depends largely on several system parameters such as T fractional burn-up in plasma, T thru-put to the plasma, number of days of reserve inventory, and doubling time. The latter is defined as the time at which the T inventory in the storage is equal to the sum of the initial T inventory required for starting a new ARIES-type reactor and the minimum T inventory that must be kept in storage for continued reactor operation during interruption of any part of the tritium cycle. In the ARIES-IV design, some approaches were identified to minimize the T inventory in the fusion power core and on-site in the storage and plasma processing facilities. For instance, a high T burn-up fraction of 18% and a low T thru-put of 1.7 kg per full-power day were considered in order to lower the T inventory in the plasma exhaust processing system. The low T thru-put to the plasma and the short repair time (<1 day) for the T processing system help reduce the reserve storage inventory to ~0.5 kg. The equilibrium hold-up inventory in the fusion power core (first wall structure, divertor plates, and blanket) is estimated to be ~2 kg. On this basis, the supply inventory to start a new reactor should be ~2.5 kg. A doubling time of 5 y seems acceptable for the first-of-a-kind ARIES-type reactors and it can be increased to 7-10 y for the tenth-of-a-kind reactors.

To calculate the breeding margin G_0 , a simple model was employed that accounts for the radioactive decay of T . In order to have a T inventory of 3 kg at the end of a 5 y period, the blanket production rate (P) for T can be calculated using the following expression

$$N(t) = \frac{P}{\lambda} (1 - e^{-\lambda t})$$

where $N(t)$ is the required T inventory (3 kg), λ is the T decay constant (0.562 y^{-1}), and t is the doubling time (5 y). In this case, the blanket production rate is estimated to be 1.9 g/day. Using a T thru-put of 1.3 kg/day (@ 76% availability), the breeding margin G_0 ($P/\text{thru-put}$) amounts to 0.015 for the ARIES-IV reference design.

As noticed, the quantity $(1 + G_0)$ is a function of several plasma and engineering components parameters that play a key role in the tritium cycle. To accurately quantify the uncertainty Δ_G associated with $(1 + G_0)$, a rigorous statistical/uncertainty analysis is required for the reactor parameters of interest. Unfortunately, the early R&D stage the fusion program is in now does not permit the development of such analysis. Alternatively, a reasonable approach that relies on "engineering" judgment was employed. Based on the analysis presented in Ref. 1, the uncertainty Δ_G for the ARIES-IV parameters is estimated to be <4%. Interestingly, a factor of 2 uncertainty in the estimated Δ_G leads

to a few percent variation, at most, in the TBR requirements. Every 1% uncertainty in the prediction of the TBR results in a discrepancy of ~ 2 kg of T requirement for ARIES-IV.

As mentioned before, the calculated TBR for the ARIES-IV design is subject to uncertainties (Δ_P) due to approximations and/or errors in the various elements of the calculations such as nuclear data, calculational method, and geometric representation. The uncertainties in the nuclear data include the uncertainties for the basic data, data processing and representation, and tritium production cross sections for Li_2O . The largest source of uncertainty in nuclear data is that for the basic nuclear data, which is $\sim 6\%$ for the Li_2O blanket [2, 3], and the other uncertainties are small ($\sim 1\%$). The transport calculations and the 1-D modeling of the blanket involved some geometrical approximations and the associated error is estimated to be in the order of $\sim 3\%$. Hence, the uncertainties in the calculational elements total to 10% which is significant and will have a major impact on the breeding requirements.

In this regard, the ARIES-IV design will attain fuel self-sufficiency if the calculated TBR is equal to or exceeds 1.12. It should be stressed that the TBR must be calculated for the reference design and the effects of the blanket coverage, Li burn-up, penetrations, assembly gaps, and side walls should all be included in the computed TBR because the overall breeding ratio is likely to be close to the required minimum and any small changes in the neutron balance could be sufficient to interrupt the tritium fuel cycle. Also, it is of importance to point out the potentials the blanket design might have to enhance the breeding capability because of the possible existence of uncovered uncertainties in the system definition.

19.2.2. Preliminary Blanket Parameters

A scoping analysis using the one-dimensional (1-D) code ONEDANT [4] was performed for the $\text{SiC}/\text{Li}_2\text{O}/\text{He}$ blanket to evaluate the impact of the blanket parameters on the tritium breeding ratio. A simple poloidal cylindrical model was utilized where a centimeter thick first wall (FW) was followed by a $\text{SiC}/\text{Li}_2\text{O}$ homogeneous blanket of variable thickness and, then, a meter thick SiC/He shield. In the analysis, the Li enrichment was varied between 7.42% (natural) and 50% ^6Li and the blanket was gradually increased in thickness from zero to one meter. The SiC (95% dense), Li_2O (90% dense), and He/void contents were kept fixed at 13%, 60%, and 27% by volume, respectively. As Figure 19.2-1 indicates, the TBR hardly increases with Li enrichment and peaks at a slightly higher enrichment ($\sim 15\%$) for thinner blankets. Figure 19.2-2, shows that, for

FIGURES IN
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Figure 19.2-1. Effect of Li enrichment on TBR for different blanket thicknesses.

Figure 19.2-2. Effect of blanket thickness on TBR.

natural Li_2O , the breeding level can effectively be controlled by adjusting the thickness of the blanket. It was also important to investigate the effect of variation in the volume proportions of the SiC structures to provide guidance for the engineering design. For a 60 cm thick blanket utilizing natural Li_2O breeder, the SiC content was varied between zero and 25%, trading SiC/He for Li_2O . As illustrated in Figure 19.2-3, the SiC content has a significant impact on the breeder capability of the blanket and must be kept as low as practically possible.

The above analysis concludes that since Li enrichment, besides being costly, has little effect on breeding, it should be avoided and alternate ways must be pursued to achieve the breeding goal. Furthermore, in designing the blanket, an effort should be made to lower the structural content as much as practical in order to enhance the breeding. Before proceeding to a more detailed analysis, some design decisions were made to guide the blanket design toward the final configuration. Since the overall size of the machine is much more sensitive to the dimensions of the i/b components, it was decided to consider a 35 cm thick blanket for the space limited inboard side and adjust the outboard blanket thickness to satisfy the breeding requirements. The design will thus 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 in a double-null configuration, the gaseous divertors cover $< 15\%$ of the FW area and the fraction of neutrons impinging on the divertor region is small relative to the i/b and o/b. In addition, these neutrons will be attenuated by the divertor plates and their support structures, leading to a further reduction in their contribution to breeding.

19.2.3. Overall Tritium Breeding Ratio and Energy Multiplication

The configuration of the reactor dictates that a realistic model to compute the overall TBR and M must be three-dimensional. The Monte Carlo codes (e.g., MCNP or MORSE) are capable of handling such geometry. However, due to lack of resources and time limitations for this study, an approximate method based on 1-D analyses has been established with the objective of ensuring that the errors associated with the final results are as low as can reasonably be achieved. Thus, the overall TBR and M have been estimated by coupling the 1-D results with the coverage fractions of the i/b and o/b blankets. This approach has proven to reasonably predict the overall values within a few percent [5] and, as mentioned before, provisions were made in the tritium breeding requirements for such uncertainties in the results.

Figure 19.2-3. Effect of SiC structural content on TBR.

The i/b and o/b blankets extend vertically up to ~ 3 and ~ 3.5 m from the midplane, respectively. The coverage fraction, representing the fraction of source neutrons going directly to each region, was determined by the NEWLIT code [6] for a preliminary configuration for the ARIES-IV design. These fractions amount to $\sim 15\%$ and $\sim 70\%$ for the i/b and o/b blankets, respectively, and 15% for both the top and bottom divertor regions. Losses in available areas due to the presence of the plasma control ports and assembly gaps imply that the blanket coverage fraction should be modified by excluding the areas occupied by the various penetrations. The assembly gaps between adjacent blanket modules are 3 cm wide. For 32 blanket modules, the space taken by these gaps amounts to 2-3% of the FW area. The outboard side has heating and current drive ports that occupy $\sim 5.8 \text{ m}^2$ (2% of the o/b FW area). The net effect of the ports and gaps on the overall TBR is about 3% reduction.

Calculations of the TBR were made for several preliminary configurations. The purpose of these calculations was to guide the design towards the final configuration that provides T self-sufficiency. The blanket design has been optimized by a series of neutronics and thermal hydraulics calculations, subject to constraints such as nuclear heating generation, heat removal capacity, and breeder and structure operating temperatures. The analysis and results given in this section, however, pertain only to the final design. In these calculations, the neutron reflection and spectral effects between the i/b and o/b blankets were taken into account by using a toroidal cylindrical geometry where both blankets were modeled simultaneously. In the model, the FW (~ 2 cm thick) and blanket were followed by a meter thick He cooled SiC shield. It was found necessary to model in detail the consecutive layers of SiC shells and Li_2O breeder. The 1.2 cm thick shells contain 30-40% He coolant and He purge gas. The breeding zone starts with ~ 0.7 cm thick layer near the FW and gradually increases toward the back of the blanket. The 90% dense sphere-pack breeder has a packing fraction of 80%. These parameters and the results presented herein pertain to the reference "poloidal-flow" blanket design of ARIES-IV.

With the most fundamental aspect of the optimization investigated using poloidal model and homogeneous blanket (refer to Section 19.2), a parametric survey was initiated to estimate the overall TBR and propose necessary design changes that allow the 1.12 TBR limit to be approached. A 1-D toroidal model was developed with 35 cm i/b and 60 cm o/b heterogeneous blankets. The neutron balance shows that, for the ARIES-IV geometry, about 25% and 75% of the source neutrons impinge on the i/b and o/b regions, respectively. As a result, the 35 cm thick i/b blanket produces 0.29 T/fusion whereas the 60 cm thick o/b blanket has a larger production of 0.76 T/fusion. Based on these

numbers, the local TBR is 1.16 for the i/b blanket and 1.01 for the o/b blanket. Modifying these values by the appropriate coverage fraction yields the contributions to the overall TBR of 0.17 from the i/b and 0.7 from the o/b, totaling to a net TBR of 0.86. Even with 35 cm thick blanket installed behind the gaseous divertor, the overall TBR could not exceed unity. Scaling from Figure 19.2-3, a full coverage blanket can only breed if the SiC structure fraction is reduced from 17% to less than 5%, which is not practical. It appears that there is no credible geometric improvement which leads to an acceptable TBR. These results indicate that the present $\text{Li}_2\text{O}/\text{SiC}$ blanket configuration will not breed without a multiplier. Noteworthy, this conclusion was confirmed in the past by many Li_2O blanket designers and it was hoped that the use of SiC structure, instead of metallics, might alter this conclusion.

The above analysis indicates the absolute requirement to incorporate a neutron multiplier within the blanket to improve the breeding. The use of beryllium or lead to enhance the neutron population has been investigated in previous reactor studies. Pb is less effective than Be and generates polonium which raises some safety concerns. The major problems with the use of Be are the neutron-induced swelling, chemical toxicity, high cost (600\$/kg), and limited resources. To alleviate these problems, ways should be found to minimize the amount of Be used in the blanket and recycle the irradiated Be. From a safety viewpoint, beryllium compounds, such as BeO or Be_2C , have advantages over Be metal. The main reason is that BeO and Be_2C are less chemically reactive, whereas Be metal contains a large amount of chemical energy. Thus, it is more difficult to release BeO or Be_2C in the case of an accident. Further, Be is less toxic in the form of BeO or Be_2C . A safety study revealed that to achieve a level of safety assurance (LSA) rating of 1, the maximum allowable inventories of Be and BeO (or Be_2C) in the blanket should be about 0.5 and 50 tonnes, respectively. Although BeO and Be_2C have safety advantages, they are not as effective as Be metal in providing neutron multiplication. This is evident by the ability of the blanket to breed tritium using these multipliers. As will be shown shortly, the minimum amount of Be needed for T self-sufficiency ($\text{TBR} \geq 1.12$) is approximately 30 tonnes. The use of a larger amount of BeO or BeC (> 50 tonnes) resulted in a less than unity breeding ratio, and even with some geometric modifications, the maximum achievable TBR is unlikely to exceed 1.05. This indicates that attaining $\text{TBR} \geq 1.12$ with BeO or Be_2C multipliers is not possible unless the blanket design is modified and/or the SiC structure content is significantly reduced.

There are three principle options for Be arrangement within the blanket:

1. Be distributed throughout the blanket and mixed with breeder (ARIES-I type blanket),

2. Alternate layers of Be and breeder (ITER-CDA type blanket).
3. Be in a separate zone located behind the FW.

It is recognized that an ideal situation from the neutronics viewpoint would be to mix the Be and breeder in a homogeneous manner rather than separate them into zones. However, the compatibility problem between Be and Li_2O dictated the need for separate zones as opposed to a homogeneous mixture. This is the main reason for precluding the first option from consideration. The second option would result in an expensive blanket as it requires a large Be inventory (> 200 tonnes) and that raises safety concerns. The third option is attractive as it utilizes the least amount of Be thus mitigating some of the Be problems. In this option, the beneficial effect of Be is expected to be confined to the front zone where multiplication rates are significant.

Three locations for the Be front zone were thus investigated:

1. Be zone behind the i/b FW only
2. Be zone behind the o/b FW only
3. Be zone behind both i/b and o/b FW.

The impact on the overall TBR of trading Li_2O breeder for Be in the front zones is demonstrated in Figures 19.2-4, 19.2-5, and 19.2-6 for the three cases. The abscissa represents the total thickness of Be (80% packing fraction and 90% dense) in the Be zones, excluding the cooling shells. The Be mass is included in the figures to provide guidance in the design selection. It is pertinent to mention that appropriate cooling shells should be modeled separately within the Be zones rather than homogenized with Be as homogenization overestimates the TBR significantly. Furthermore, iterations of neutronics and thermal hydraulics calculations are mandatory during the analysis to establish a mutually compatible design and detailed distributions of the nuclear heating in the blanket should be calculated as an input for the thermal hydraulics analysis. Figure 19.2-4 reveals that the additional breeding obtained through the use of > 6 cm Be in the i/b is marginal and even if a large amount of Be is used in the i/b, the TBR will not exceed unity, meaning that the blanket will not provide T self-sufficiency ($\text{TBR} > 1.12$) if Be is confined to the i/b side only. Higher breeding is obtained by employing Be in the o/b side, as shown in Figure 19.2-5, and a large amount of Be (~ 35 tonnes) is needed to meet the breeding requirements. It appears that considerable reduction in

Figure 19.2-4. Variations in TBR with beryllium used behind inboard FW only.

Figure 19.2-5. Variations in TBR with beryllium used behind outboard FW only.

Figure 19.2-6. Variations in TBR with beryllium used behind inboard and outboard FW.

Be mass can be achieved by dividing the Be between the i/b and o/b, as illustrated in Figure 19.2-6. In this case, Be was added to the i/b side first, then to the o/b till the TBR reaches 1.12. Clearly, case 3 combined the benefits of satisfactory breeding and minimal Be inventory and it has been selected as the reference case for ARIES-IV.

A set of calculations was performed to examine the effect of Li enrichment on TBR while retaining the composition of the $\text{Li}_2\text{O}/\text{Be}/\text{SiC}/\text{He}$ blanket. The TBR for several enrichments in the range 7-90% ^6Li is shown in Figure 19.2-7, demonstrating that 20% enrichment gives < 1% increase in TBR. Accordingly, Li enrichment should be avoided since the gain in breeding is marginal and the cost penalty is significant (factor of ~ 4 higher).

Time-dependent neutron induced effects, such as Li burn-up, could influence the tritium breeding capability of the blanket. In this regard, burn-up calculations were carried out to assess the impact of the temporal variation in Li concentration on the space-integrated TBR. Since the blanket will be replaced every 3.7 years (see Section 19.3), the calculations have been limited to this period and the results are normalized to 2.9 and 4.7 MW/m^2 for the i/b and o/b wall loadings, respectively. The burn-up in ^6Li and ^7Li were calculated for each breeding zone of the i/b and o/b blankets and the changes in the atomic densities of the original Li isotopes were respecified to the code in order to quantify the loss in breeding due to burn-up. Figure 19.2-7 shows the contribution of the 9 i/b and 15 o/b breeding zones to the overall breeding. The ^6Li contributes 80-95% to the breeding, depending on the location of the breeding zone. The calculations indicate that the Li burn-up is more pronounced for the i/b, compared to the o/b, and is enhanced at the front breeding zones, relative to the back. The end-of-life TBR using the burnt-up ^6Li and ^7Li atom densities are compared with the reference values in Figures 19.2-8 and 19.2-9. The decrease in the overall TBR is < 1%. This insignificant effect is the result of the redistribution of the neutrons between the breeding zones.

Several minor modifications were made to align the neutronics model with the developing parameters of the design:

1. The dimensions of the blanket were increased to correspond to the final configuration.
2. The cooling shells and breeder zones were modified slightly to accord with the latest fusion power and subsequent nuclear heating evaluation.
3. The blanket coverage fractions are 20% for the i/b and 68% for the o/b according to the latest configuration.

Figure 19.2-7. Effect of Li enrichment on overall TBR.

Figure 19.2-8. Impact of Li burn-up on breeding of the 9 i/b breeding zones.

Figure 19.2-9. Impact of Li burn-up on breeding of the 15 o/b breeding zones.

Table 19.2-I.

TRITIUM BREEDING IN BLANKETS

	Inboard Blanket	Outboard Blanket
Blanket thickness (cm)	35	60
Coverage fraction	20%	68%
Local TBR	1.68	1.23
Contribution to TBR	0.33	0.79
Overall TBR	1.12	

4. The thicknesses of the Be zones were adjusted to maintain the overall TBR at a value of 1.12.

The final blanket design calls for a total of four Be zones: 2 and 2.3 cm thick zones in the i/b blanket and 2 and 3.5 cm thick zones in the o/b blanket. The i/b side contributes 30% to the breeding while the o/b blanket supplies the balance. A detailed breakdown of the breeding is illustrated in Table 19.2-I. It should be mentioned that the ARIES-IV design has the potential to achieve higher breeding by increasing the Be content, thickening the blankets, installing blanket in the divertor region, and/or enriching the Li_2O to 20% ^6Li .

The same methodology used to estimate the overall TBR was followed to evaluate the neutron energy multiplication of the system. The beryllium multiplier has enhanced the energy multiplication by $\sim 15\%$ while variations in the other blanket parameters resulted in a very slight effect. The end results show that the overall energy multiplication is 1.22. The reactor produces 2022 MW of fusion power and 80% of this power is carried by neutrons. Neutrons and gamma rays deposit their energies in the blanket/shield generating an additional 351 MW of thermal power, resulting in 1968 MW_{th} of total nuclear heating. The breakdown of the heating deposited in the FW, blankets, and shields is given in Table 19.2-II. As noticed a large fraction of the power (82%) is deposited in the blankets. The i/b blanket carries only 21% of the heat. This is due to the fact that

Table 19.2-II.

**NUCLEAR HEATING GENERATED IN THE
VARIOUS REACTOR COMPONENTS**

Nuclear Heating (MW)	Inboard	Outboard	Divertor	Total
FW/Blanket	410	1200	–	1610
Shield	65	57	236	358
Total	475	1257	236	1968

the i/b blanket is thinner and has less coverage compared to the o/b blanket. It is also of interest to note that about 18% of the heating is generated in the shields. This is a significant amount of heat which cannot be discarded as a low grade heat without having a negative impact on the design. Therefore, a decision was made to run all the in-vessel components hot and to recover the nuclear heating deposited in all blanket and shield components as a high grade heat.

19.3. RADIATION DAMAGE AND LIFETIME ANALYSIS

19.3.1. SiC/SiC Composites

There are some radiation damage concerns for the SiC/SiC composites structure. The neutron-induced transmutation products can detrimentally affect the mechanical properties of the material. The high energy neutrons (3-14 MeV) interact with and, thus, burn the Si and C atoms generating Al, Mg, Na, Be, Li, and large amounts of helium and hydrogen gases. A high level of these products will deteriorate the thermo-mechanical properties of the composites. Furthermore, the gaseous products cause volumetric swelling, decrease the strength, and eventually lead to the failure of the material [7]. Presumably, the gases accumulate and form bubbles at the grain boundaries and at the interfacial areas between the fiber and matrix causing a loss of integrity of the composites. The lifetime of the composites will thus depend on the production level of these gases. Hence, the burn-up of the SiC atoms is used to determine the usefulness and lifetime of the FW/blanket in ARIES-IV.

It has been postulated that the SiC/SiC composites can stand up to 3% burn-up without failure. The effect of such burn-up level on the structural properties of the composites is totally unknown. Work is underway in Italy to simulate the effect of the transmutation products using alpha-particle irradiation [8]. Unfortunately, these tests will not fully simulate the fusion reactor environment. Since this issue is of enormous importance for the credibility of the SiC/SiC composites as a structural material for fusion reactors, 14 MeV neutron irradiation experiments need to be conducted to study the effect of transmutation products on the thermo-mechanical properties of the composites. In parallel, the manufacturing technique of SiC/SiC composites must be developed to meet the fusion-specific needs. It is pertinent to mention that while composites can be tested in fission reactor facilities, such facilities will not provide adequate tests because most of the transmutation reactions have relatively high neutron threshold energies (> 3 MeV) and will not occur under neutron irradiation with fission spectrum.

The radiation damage level attainable during operation was determined for the i/b and o/b sides of ARIES-IV. The criterion adopted is that no more than 3% burn-up and 200 dpa are allowed in the structural components. It has been found that the burn-up limit is more demanding than the dpa limit. As such, the burn-up limit will determine the lifetime of the FW/blanket and shield. The dpa rates at the i/b and o/b FW amount to 47 and 57 dpa/FPY, respectively. It should be mentioned that the dpa results obtained in this analysis agree within 10% with assessments made by new theoretical models [9] and suggest that the present nuclear data adequately represent the atomic displacement cross sections for SiC. The peak helium and hydrogen production at the first wall are listed in Table 19.3-I. As noticed, the o/b side exhibits higher damage due to the higher neutron wall loading. Figures 19.3-1, 19.3-2, and 19.3-3 illustrate the radial variations of radiation damage at the cooling shells of the blanket. The drop in damage across the 60 cm thick blanket ranges from 30 to 330, depending on the radiation effects. The thinner i/b blanket exhibits a lesser decrease in damage (7-20). Large fraction ($\sim 90\%$) of the He production rate in C is due to the $(n, 3\alpha)$ reaction caused by high energy neutrons ($E > 8$ MeV). This reaction diminishes considerably toward the back of the blanket as depicted in Figure 19.3-2.

The burn-up rates of Si and C atoms are listed in Table 19.3-I. Postulating that any (n, α) or (n, p) reaction transmutes the Si and C atoms into Al, Mg, or Be and destroy a SiC molecule, the burn-up of SiC would be the direct sum of the Si and C burn-up rates. However, it is possible that some neutrons would react with the Si and C atoms of the same molecule. Furthermore, some of the free Si and C atoms may recombine and reform SiC molecules. As such, the burn-up of SiC is reduced by $\sim 20\%$ to account for these

Figure 19.3-1. Radial variation of dpa rate in SiC structure of blankets.

Figure 19.3-2. Radial variation of He production rate in SiC structure of blankets.

Figure 19.3-3. Radial variation of H production rate in SiC structure of blankets.

Table 19.3-I.

PEAK RADIATION DAMAGE TO SiC STRUCTURE

	Inboard FW		Outboard FW	
	C	Si	C	Si
He production rate (appm*/FPY)	8,290	2,080	12,300	3,080
H production rate (appm*/FPY)	0	3,930	0	5,750
Burn-up rate (% per FPY)	0.3	0.6	0.5	0.9
SiC burn-up rate (% per FPY)	0.7		1.1	
Blanket lifetime @ 3% burn-up (FPY)	4.3		2.8	
(year)	5.6		3.7	
FW/B EOL fluence (MWy/m ²)	12.3		13	

*atom parts per million atoms of C or Si.

effects yielding a total of 0.7 and 1.1% per full power year (FPY) at the i/b and o/b FW, respectively. The 3% burn-up limit dictates that the FW/blanket (~ 400 tonnes) should be replaced every 3.7 years. Although the i/b FW/blanket could be replaced on a less frequent basis, a decision was made to change out the i/b modules along with the o/b. Certainly, there will be an incremental cost associated with the early replacement of the i/b side, but this will be offset by the gain due to the higher availability of the system. The end-of-life (EOL) fluence of the FW/blanket amounts to 13 MWy/m². This corresponds to a peak outboard wall loading of 4.8 MW/m² which remains nearly constant along a poloidal height of a few meters around the midplane. The shield suffers less damage compared to the blanket and the low burn-up level at the front layers indicates that the shield is a lifetime component and requires no replacement during the 40 y of operation.

19.3.2. Beryllium Neutron Multiplier

As previously mentioned, the present design of the Li₂O/SiC blanket will not breed unless a Be neutron multiplier is incorporated in the blanket. The major problem associated with the use of Be is the production of large amounts of helium gas and the

subsequent swelling induced by that gas. This requires that the initial space containing the beryllium be wide enough to accommodate the swelling without imposing undue stresses on the cooling shells. The impact of such requirements on the heat removal and mechanical design of the Be zone is a serious concern and unfortunately, this issue has not been given much attention in this study due to time limitations.

The effect of He production on the swelling and thermal properties of Be at the operating conditions of ARIES-IV (1000°C , 4.7 MW/m^2 , 13 MWy/m^2 , $10^{23} E_n > 0.1 \text{ MeV n/cm}^2$, $53,000 \text{ He appm}$, and 600 H appm) needs to be understood. The presently available data are very limited and, as shown shortly, is somewhat ambiguous in that some experimental data indicate high volumetric swelling ($> 100\%$) while other expectations suggest lower effect ($\sim 30\%$ swelling). Irradiation experiments need to be conducted to resolve this issue particularly for ARIES-IV where the design relies on the structural integrity of Be for an electrical conducting shell.

The available data on beryllium irradiation effects (namely the formation of He bubbles and the resulting swelling) originated from samples irradiated in fission reactors in the temperature range of $50\text{-}700^{\circ}\text{C}$ at much softer neutron spectrum and lower fluence level than the design values for ARIES-IV. Higher temperature irradiation effects were evaluated analytically and experimentally. The analytical models [10] agreed well with in-reactor irradiation results for $T < 500^{\circ}\text{C}$. Higher temperature irradiation behavior have been inferred largely from post-irradiation heat treatment observations which tend to underestimate the actual swelling that result from in-reactor irradiation. Several high temperature ($550\text{-}970^{\circ}\text{C}$) experiments were conducted on Be specimens at the Russian fission reactor CM-2 [11, 12]. The results reveal that the swelling increases with temperature and fluence. Scaling from reported data and adjusting for the harder fusion spectrum and higher fluence, the volumetric Be swelling in ARIES-IV is expected to be $> 100\%$. On the other hand, some theories predict that at high temperatures the He should diffuse out of Be like tritium and the resulting swelling should not exceed 30% . This argument is somewhat controversial because, besides the Russian experimental results, the bubbles in uranium (which behaves like Be due to closeness of melting points) did not anneal out [13] even at higher homologous temperatures than postulated for Be.

Another concern for Be, which is already a brittle material, is that it becomes more brittle under irradiation and will undergo significant cracking during irradiation. Micro-cracks were observed in Be specimens irradiated at a 10 times lower fluence than that of ARIES-IV and at even lower temperatures [13]. This means that the Be layer may not hold its integrity during reactor operation and will be divided into many pieces. These pieces may not be able to sinter even at elevated temperature since sintering can only be

used for small sphere. Therefore, relying upon the structural integrity of the beryllium layer for an electrical conducting shell would be highly risky because Be properties may severely change under irradiation.

19.4. SHIELD DESIGN

The intent of this section is two folds: to design a radiation shield to protect the TF magnets and to design a biological shield around the torus for personnel protection during operation. In the analysis, the impact of shield performance on the design is assessed with a view to cost, complexity, fabricability, and safety. The shield is required to keep the radiation level below certain limits. As a matter of fact, these limits have a strong effect on the characteristics of the shield, in terms of thicknesses and compositions. To reduce the size of the shield, it is highly recommended to employ materials with high radiation tolerance in the viable components of the reactor. Moreover, it is essential to optimize the composition of the shield. Significant savings in the overall cost of the machine can result from the design of an efficient shield as it helps reduce the size of the reactor. Specifically, a centimeter less shield in the inboard side would reduce the overall cost by 5-10 M\$, depending on the size of the device.

19.4.1. Magnet Radiation Limits

Besides the basic requirements, special considerations due to the radiation environment confront the magnet designers. The nuclear heating in the coils generated by the neutrons and gamma rays must be transported out of the winding packs with very little elevation of the conductor temperatures. This heating also determines the required capacity of the helium refrigeration system. Cumulative irradiation effects over the reactor life (40 years) are also important: degradation of critical current/temperature/field of superconductivity, increase in the resistance of the stabilizing Cu, and deterioration of the electrical and mechanical properties of the insulation material in the windings.

As the radiation adds many problems to the magnet design, sufficient shielding should be placed between the plasma and magnet to keep the radiation effects below certain limits. These limits are set by the magnet designers to insure the proper performance of the TF coils. For instance, at the end of 30 full power years of operation, the fast neutron fluence ($E_n > 0.1$ MeV) should not exceed 10^{19} n/cm² to avoid degradation of the critical properties of the Nb₃Sn superconductor material. It is undesirable to subject

the magnets to a total nuclear heating above 50 kW to avoid excessively high cryogenic load to the cryoplant. A limit of 2 mW/cm³ is imposed on the peak nuclear heating in the winding pack. The end-of-life dose to the glass-fiber-filled (GFF) polyimide is limited to 10¹¹ rads to ascertain the electrical integrity of the insulator. Such high dose is allowed due to the absence of shear forces on the insulator. The neutron-induced atomic displacement in the Cu stabilizer should not exceed 6×10^{-3} dpa to avoid high increase in the Cu electric resistivity. It should be mentioned that the fluence and dose limits are at least a factor of 2-3 lower than the experimental values at which degradation of properties was observed. Our neutronics calculations indicate that the predominant magnet radiation limits are the EOL fast neutron fluence and the nuclear heat load to the magnets. Hence, the shield is optimized to primarily minimize these effects.

19.4.2. Optimization Analysis

The intent of this section is to design an efficient low activation shield to protect the TF magnets against radiation. This involves the selection of the shielding materials, assessment of the shielding capability of the various candidate materials, optimization of the composition of the shield, and determination of the thickness of the shield required to keep the radiation damage at the magnet below the permissible level.

The analysis employed the 1-D transport code ONEDANT with the P_3 legendre expansion for the scattering cross section and the S_8 angular quadrature set. The associated 46 neutron and 21 gamma group cross section library was derived from the ENDF/B-V evaluation. The analysis was mainly carried out for the i/b shield as it strongly affects the overall size of the machine. Outside the shield is the TF magnet surrounded by the cryostat. The winding pack of the TF magnet includes Incaloy structure, Nb₃Sn superconductor, Cu stabilizer, Cu/Nb conduit, GFF polyimide insulator, and liquid He coolant at 57, 6, 24, 5, 4, and 4 vol.%, respectively. The cryostat consists of a 3 cm thick thermal insulator sandwiched between two SS plates, 1 cm thick each.

The different components of the blanket and shield were modeled in toroidal cylindrical geometry around the machine axis, permitting the representation of the inboard and outboard sides simultaneously. It is pertinent to mention that a poloidal model should not be employed, particularly for the inboard side, as it underestimates the damage at the magnets. The plasma shift toward the outboard is taken into account in modeling the neutron source. The distribution of the neutron wall loading in the poloidal direction is taken from the system code output. The average wall loading is 2.8 MW/m² and the distribution peaks at the outboard midplane at a value of 4.7 MW/m². The inboard

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wall loading maximizes at the midplane at 2.9 MW/m^2 . The divertor region has the lowest wall loading of 2 MW/m^2 . The neutron source strength used in the transport calculations is consistent with these wall loadings.

The optimization analysis has focused on optimizing the major factors that influence the shield performance. These factors are the composition of the shield, form of materials, and material arrangement within the shield. To insure the reliability of the shield design, it is essential to select shielding materials that can stand the high radiation environment of the design. The shield is considered a semi-permanent component and should be designed to perform successfully over the entire lifetime of the machine (40 y). It is quite unlikely that the shield requires repair or replacement but must be replaceable by practical means if that should become necessary. Thus, the shield is segmented toroidally into 48 modules and designed to be easily disassembled in order to provide ready access for maintenance and repair to the internal 32 blanket modules.

Besides SiC and steel, many materials have the potential to protect the magnets. Each material has some merits and drawbacks. The behavior of the material, particularly in a high radiation environment, may limit its use in the shield. For example, the tungsten is known as the best shielding material but its high decay heat and the production of the ^{187}W isotope cause some safety problems. The hydrogen compounds (like titanium hydride, boron hydride, organic, water, and polyethylene) are superior in slowing down the fast neutrons. Nevertheless, all hydrides decompose under irradiation and elevated operating temperatures. Subsequently, the high pressure of the dissociated hydrogen may jeopardize the integrity of the shield. However, for designs with low afterheat generation this problem is greatly mitigated particularly if the temperature of the hydride layer is controlled during operation. An ideal situation from the shielding point of view is to use water (or borated water) in the shield. However, this would complicate the cooling system considerably due to the requirement for two independent primary cooling circuits. The borated materials (such as boron carbide, boron hydride, boronized carbon, and borated water) possess remarkably high neutron absorption cross sections particularly for low energy neutrons, a property of great importance in magnet shielding. Among these borides, boron carbide is the best for having the combined features of excellent shielding capability, stability under irradiation, and ease of fabrication. One concern regarding the B_4C (as well as all other borides) is the tritium generated in the boron. To decrease the in-shield tritium inventory, the B_4C should be situated close to the back of the shield. This constraint turned out to be beneficial for the B_4C since the neutron spectrum therein is relatively soft and the neutron absorption capability of B_4C is high. The lead has been known for its good ability to reduce magnet heating by absorbing

the gamma radiation. However, lead has a few drawbacks of which one should be aware. Lead increases the neutron fluence (compared to steel) because it multiplies the neutrons, starts to deform at $\sim 220^{\circ}\text{C}$ and, thus, has to be canned in a structural material, and generates polonium, which is a safety concern.

The first set of initial calculations demonstrates the shielding performance of several materials relative to SiC. To integrate the environmental constraints into the design from the beginning, it is important to select shielding materials that do not become strongly radioactive when bombarded by neutrons. This will eliminate the need for deep geological burial of the radioactive waste and avoid high radiation exposures at the site boundary during an accident. It should be noted that these initial calculations were completed for a preliminary blanket model where the 35 cm thick i/b blanket was followed by 20 cm thick SiC reflector, 13 cm He manifold and, then, a shield. The blanket, reflector, and shield contain an assumed He coolant fraction of 5% by volume. The He manifold has 50% SiC structure and 50% He gas. The shield was optimized to minimize the damage at the magnet. The analysis shows that in order to satisfy the magnet radiation limits, an all SiC shield would be fairly thick (> 1 m). Therefore, other materials should be incorporated in the shield. B_4C and Pb have proven to be very effective at improving the performance of the SiC shield in particular. A 76 cm thick SiC/ B_4C /Pb shield is sufficient to protect the inner legs of the TF magnets with an optimal composition of 48 cm SiC shield, 23 cm B_4C shield, and 5 cm Pb shield. The effect of the SiC and B_4C arrangement within the shield was investigated. As an alternative to the two thick layers, the SiC/ B_4C shield was configured into alternating thinner layers of SiC and B_4C , about 5 cm thick each. The other possibility of using a homogeneous mixture of SiC and B_4C was also examined.

Table 19.4-I lists the shielding requirements for each option. The multilayered and homogeneous SiC/ B_4C shields save approximately 2 and 3 cm in shield thickness, respectively. There is, however, a price paid for these savings in terms of design complexity and higher in-shield T inventory, which adversely affects safety. As expected, the T production considerably increases as the B_4C is brought up closer to the plasma. As these two options exhibit no clear outstanding advantages, the two separate SiC and B_4C layers option was selected as the preferred option for the SiC-based shield. An attempt was made to reduce the size of the SiC shield by including some low activation metallics in the shield. These metallics are used only as filler materials and there is no structural load envisioned at the present time for these materials. It was found that a considerable reduction in the shield thickness of 8 and 15 cm can be achieved by including V and modified HT-9 fillers, respectively, in the SiC shield. Table 19.4-I details the composition

Table 19.4-I.

**RADIATION
EFFECTS AT TF MAGNETS FOR VARIOUS SHIELD DESIGN OPTIONS**

Shield Type	SiC/B ₄ C/Pb	V/B ₄ C/Pb	SS/B ₄ C/Pb	Heterogeneous H Multilayers
Composition	48 cm SiC shield 23 cm B ₄ C shield 5 cm Pb shield	36 cm V shield 29 cm B ₄ C shield 3 cm Pb shield	34 cm SS shield 26 cm B ₄ C shield 1 cm Pb shield	14 layers of SiC & B ₄ C 1 cm Pb shield
Inboard shield thickness (cm)	76	68	61	74
Total inboard FW/B/R/M/S* thickness (cm)	145	137	130	143
T inventory in shield (g @ 40 y)	3	3	3	30
<u>Radiation Effects at Magnet:</u>				
Peak fast neutron fluence to Nb ₃ Sn (10 ¹⁹ n/cm ² @ 30 FPY)	1	1	1	1
Peak nuclear heating in W.P. (mW/cm ³)	2.0	1.5	1.8	1.4
Peak dose to insulator (10 ¹⁰ rads @ 30 FPY)	2.8	2.2	2.4	2.0
Peak dpa in Cu stabilizer (10 ⁻³ dpa @ 30 FPY)	5.3	5.6	5.5	5.6

*First wall/blanket/reflector/manifold/shield.

of the different options. Each layer of the shield is supported by a 15 v/o SiC structure and the magnet radiation limits are met for all options. An appreciable cost reduction ($\sim 5\%$) would result from the use of such cheaper steel filler materials. However, it was recognized that the relatively higher afterheat of the steel shield may affect the safety features of the SiC shield and, thus, a decision was made to choose the SiC shield as the reference shield for ARIES-IV. This decision was based on approximate adiabatic afterheat calculations which indicated that the shield temperature may exceed 500°C in case of a loss of coolant accident. A more rigorous afterheat analysis should be performed in future designs to evaluate the safety impact of the steel option as it provides some economic advantages.

Several major modifications were made to the blanket and vacuum vessel components and, therefore, a detailed shielding optimization study was carried out for the most up-to-date design. For the reference "poloidal-flow" blanket design, the helium manifolds will not be located behind the blanket but rather placed at the top/bottom of device. The blanket is followed directly by the shield and the reflector was eliminated from the design. This has reduced the blanket replacement cost considerably. To support this decision, the radiation-induced damage to the SiC structure of the shield was checked and found acceptable even at the end of the reactor life (40 y). This means the blanket provides the required protection for the permanent shield. The latest blanket design calls for a much higher helium/void fraction than originally assumed in the initial analysis (5%). A homogenized i/b blanket consists of 17 v/o SiC structure, 10 v/o Be, 37 v/o Li_2O , and 36 v/o He/void. A 10 cm thick vacuum vessel surrounds the shield and is made of a low activation steel (modified HT-9).

Special effort was devoted to include in the neutronics model all components, such as blanket, vacuum vessel (v.v.), and cryostat, that have significant impact on the shield design. The radial build of the various components are shown schematically in Figure 19.4-1. SiC is the primary structural material in all components and 15 v/o structure is foreseen enough to support the shield. All the in-vessel components are helium cooled. The composition of the shield was optimized to reduce the radiation damage at the magnet. As mentioned earlier, the fast neutron fluence and the peak nuclear heating, which is an indicative for the total heating in the magnets, are the most demanding radiation limits. In other words, once these limits are met all the other limits will automatically be satisfied. In the neutronics analysis, the 35 cm thick i/b blanket was followed by a 95 cm thick all SiC shield. The results indicate that although the fluence limit is met, the nuclear heating at the magnet is excessive ($10 \text{ mW}/\text{cm}^3$). Thus, other materials, such as B_4C and Pb were then introduced to enhance the performance

Figure 19.4-1. Schematic of inboard blanket/shield, V.V., and magnet.

of the SiC shield. At the beginning, there are some issues related to the B_4C that need to be discussed. Firstly, the SiC shield and the B_4C shield must be modeled as separate layers, the same way they will actually be manufactured and installed in the reactor. The homogenization of the various layers of the shield underpredicts magnet damage and leads to a nonoptimum design. Secondly, it is advisable to use B_4C bricks (230 \$/kg) instead of B_4C powder (62-80% dense, 45 \$/kg) despite its lower cost, particularly if B_4C is used in the i/b side. The reason is that the B_4C powder needs a larger shielding space. This implies bigger magnets and a bigger overall machine. The net effect is that the cost savings for the cheaper B_4C powder will be more than offset by the increase in shield thickness (~ 10 cm), and, thus, overall machine size, due to its low density. Furthermore, from the reliability viewpoint, the solid form is preferable as the uniformity of the shield is assured. Lastly, the option of enriching the B_4C in ^{10}B should be avoided. The slight improvement in the shielding performance and the corresponding thinning in shield will not justify the significant increase in the cost of the enriched material (factor of 10 higher cost for 90% ^{10}B).

The implications of including B_4C in the shield are shown in Figure 19.4-2. The total shield thickness was kept fixed at 95 cm, and the back layer of the SiC shield was gradually replaced by B_4C shield ($B_4C/SiC/He$ at 80/15/5 vol% and 90% dense natural B_4C). As is evident, the B_4C significantly reduces the magnet damage and a 65 cm thick SiC shield followed by a 30 cm thick B_4C shield is an optimum arrangement for the SiC/ B_4C /He shield. Near the end of the study, it was discovered that TiH_2 is superior to B_4C in reducing magnet damage. For 30 cm thick TiH_2 shield, a factor of ~ 5 less fluence and a slightly less heating were obtained. This means it is possible to save 10-15 cm in the i/b shield by employing TiH_2 . It is likely that the TiH_2 will require some design modifications to ensure the integrity of the TiH_2 layer during operation and after shutdown. Due to time limitation, these issues as well as the savings in reactor size and cost will be addressed in future studies.

An attempt was made to include a thin layer of lead at the back of the SiC/ B_4C /He shield, keeping the total shield thickness fixed at 95 cm. Our results indicate that the Pb acts to raise the fluence and decrease the heating. Since the fluence is the design driver for the shield and the heating is already well below the design limit, it is clear that there is no benefit for adding Pb to the shield. As a final point, it is of interest to show the effect the blanket has on magnet damage. Since the blanket has less shielding capability compared to the shield, the damage at the magnet decreased upon replacing the 35 cm thick i/b blanket by shielding materials, as illustrated in Figure 19.4-3. The presence

Figure 19.4-2. Effect of replacing the back layer of the SiC shield by B₄C shield.

Figure 19.4-3. Reduction in magnet damage due to replacing the inboard blanket by shielding materials.

of the 35 cm thick i/b blanket raises the fluence by a factor of ~ 5 . This means about 13 cm of the i/b space could be saved if the i/b blanket is eliminated.

19.4.3. TF Magnet Shielding

The optimal shield design presented thus far essentially pertains to the inboard side of the reactor, but found to be applicable to the other regions. However, the outboard and divertor shielding spaces are less constrained relative to the inboard. The reason is that the maintenance scheme requires the outer legs of the TF magnets to be located at a large enough distance so that the blanket modules can be pulled out radially for replacement. This resulted in ambient spaces at the top/bottom and behind the o/b blanket (see Chapter 10 for a vertical cross section of the reactor). It is, therefore, not necessary to utilize a high efficient shield therein and, instead, a pure SiC shield can protect the magnet. Admittedly, the SiC shield will be thicker but certainly less expensive than the SiC/B₄C shield. In the o/b side, the 60 cm thick blanket is followed by a 1.12 m thick SiC shield (SiC/He at 95/5 vol%) to diminish the nuclear heating in the outer legs of the TF coils. The neutron flux attenuation throughout the o/b blanket and shield is illustrated in Figure 19.4-4. The 60 cm blanket reduces the neutron flux by a factor of 10 while the attenuation through the 122 cm thick shield/V.V. is ~ 4 orders of magnitude. The divertor plates and their support structure contain an equivalent thickness of 5 cm of SiC. The analysis shows that the divertor shield must be 1.25 m thick to adequately protect the magnets. The damage levels at the various parts of the TF coils are listed in Table 19.4-II.

Another data of interest is the total nuclear heating deposited in the 16 TF coils. This heat will be removed from the magnets at 4.2 K by a cryogenic plant. The nuclear heating in the magnets was estimated taking into account the changes in the shield thickness, the poloidal variation in the neutron wall loading, and the toroidal coverage of the magnets. Table 19.4-III details the nuclear heating in the various parts of the magnets. Most of the inner and outer legs heating is deposited at the middle part around the midplane. The reason is that the FW curves to follow the plasma contour and, thus, the radial thickness of the shield (which is minimum at the midplane) gradually increases along the poloidal direction. Also, the neutron wall loading drops from peak values of 2.9 and 4.7 MW/m² at the midplane to minima of 1.3 and 2 MW/m² at the upper/lower ends of the i/b and o/b sides, respectively, and this helps reduce the heating even further. As indicated in the table, the integrated nuclear heating totals to 13.5 kW. This heat is removed from the magnets at a cryoplant efficiency of 310 W per W [14] and, therefore,

Figure 19.4-4. Neutron flux level at outboard blanket and shield.

Table 19.4-II.

**KEY BLANKET/SHIELD PARAMETERS AND
RADIATION EFFECTS AT TF MAGNETS**

Region	Inboard	Outboard	Divertor
Peak neutron wall loading (MW/m ²)	2.9	4.7	2
Thicknesses (cm):			
Blanket	35	60	—
Shield:	95	112	125
SiC shield	65	112	125
B ₄ C shield	30	—	—
Vacuum vessel	10	10	10
Total	140	182	135
Peak fast neutron fluence to Nb ₃ Sn (n/cm ² @ 30 FPY)	7.5×10^{18}	6.4×10^{17}	2×10^{18}
Peak nuclear heating in winding pack (mW/cm ³)	1.5	0.5	1.5
Peak dose to GFF polyimide (rads @ 30 FPY)	2.0×10^{10}	5.5×10^9	1.7×10^{10}
Peak dpa in Cu stabilizer (dpa @ 30 FPY)	3.5×10^{-3}	2.8×10^{-4}	8×10^{-4}

Table 19.4-III.

TOTAL NUCLEAR HEATING IN THE 16 TF MAGNETS

	Inner Legs	Outer Legs	Top and Bottom
Average wall loading (MW/m ²)	2	3.3	2
Poloidal height or length (m)	6	7	2
Nuclear heating per unit length (kW/m)	1.8*	0.6*	1.5
Integrated heating (kW)	6	1.5	6

*At midplane

the required cryogenic load amounts to 4.2 MW. It appears that this is an acceptable load as it represents only 0.4% of the net electric power.

19.4.4. Penetration Shielding

The shield has been treated so far as being free of penetrations. In general, the presence of a penetration indicates a weakening in the shield performance as neutrons stream through and increase the radiation levels around the penetrations. Therefore, supplementary shielding must be added to the perimeter of these penetrations to ensure that the radiation induced effects will not lead to unacceptable degradation in the expected performance of the magnets and external components. In some cases, the penetration shielding may interface with the requirements of the device such as those for machine accessibility. This necessitates the use of highly efficient shield in some localized areas that might have restricted space or limited accessibility.

The major penetrations of the device are the divertor pumping ducts, gaseous divertor slots, and current drive ports. Assembly gaps between blanket/shield modules also play an important role in enhancing streaming. The major neutronics concerns associated with these penetrations are the local damage hot spots at the magnets, radioactivity level of the external components, radiation induced-noise in the sensitive electronic equipment, activation of the reactor covering gas, and biological dose level of occupational personnel.

Proper analysis of the streaming problem requires 3-D Monte Carlo calculations which is usually an expensive and time-consuming method. Due to the lack of resources, the most critical streaming problems were qualitatively quantified and rough estimates for the required shielding were obtained by rescaling from the 3-D results of previous reactor studies. Although more rigorous analysis may result in some adjustments to the penetration requirements, the shielding parameters will probably not change greatly.

Concerning the gaseous divertor design, there are two slots which penetrate into the divertor shield. Fortunately, there is sufficient space at the top/bottom of the reactor to add extra shield behind these slots. This is in fact true for the outer slot only and shielding of the inner slot is a real concern. As a matter of fact, the plasma parameters, specifically the triangularity, play an important role in providing enough shielding space behind the inner slot in particular. The reason is that the low triangularity provides sufficient space for shielding as it moves the crossing point of the magnetic flux lines outward. In this regard, the final design was arrived at after several iterations between the neutronics and physics in order to meet the shielding requirements, physics constraints, and assembly/maintenance demands. To satisfy the shielding requirements, it is essential to provide 125 cm of SiC/He shield behind the slots to protect the magnets. The latest CAD drawings showed that provision of such shielding space can be marginally achieved without difficulty. Should unforeseen shielding problems arise, the more efficient shield (SiC/B₄C/He) would be employed.

In the double null configuration, the plasma ashes will be pumped out from both the upper and lower divertors. The design calls for 16 ducts at each of the top and bottom of the reactor. These ducts protrude the o/b shield horizontally between the outer legs of the TF coils. The toroidal width of each duct is in the order of 20 cm. The design is characterized by several features that help reduce the streaming through these ducts. Firstly, the plasma center is shifted outward and, thus, the duct openings are out of sight of most of the source neutrons. Secondly, the top/bottom ends of the o/b blanket/shield are extended inward as much as practical to screen the duct from a direct view to the plasma. The combined effect of these features is to preclude the uncollided source neutrons from streaming directly into the ducts. Even though, some neutrons will be reflected into the ducts from the opposite surfaces and increase the flux level inside the ducts. It is estimated that about 70 cm thick lateral shield (SiC/He) is required around the ducts to protect the sides of the outer legs of the TF magnets against streaming radiation. This estimate is based on the results of 3-D streaming analysis from a previous reactor study [15]. It seems that there is a sufficient space to accommodate this shield without any access problem.

There are 3 RF systems for heating and current drive at steady state and startup operations. The essential features of these ducts include opening areas of 0.6, 1 and 2 m² at the FW, duct axes at the o/b equatorial plane and at 30 degree above midplane, launcher modules and antenna array replacing 20-100 cm of the o/b blanket/shield, and feeding transmission lines embedded in the o/b shield. A more detailed description of the plasma control system is given in Chapter 4. The design offers the flexibility of bending the transmission lines to greatly attenuate the streaming radiation. Nevertheless, the presence of the apertures containing the antenna array mandates the addition of localized shields behind these apertures (10-90 cm thick) to assure the proper protection of the magnets. There is enough space between the back of the shield and the V.V. to accommodate this shield.

Along the toroidal circumference, the blanket/shield are divided into modules to facilitate the disassembly and maintenance of the in-vessel components. Adjacent modules are separated by a 3 cm wide assembly gap. There is a concern that neutrons will penetrate through these gaps and give rise to significant damage levels at the magnets. To lessen the severity of this problem, the blanket is segmented into 32 modules whereas the shield is subdivided into 48 units and the centerlines of the blanket and shield gaps were misaligned to eliminate any direct line-of-sight between the plasma and the magnets. The final layout of the blanket/shield accords with these requirements (refer to Chapter 10).

19.4.5. Biological Shielding

The biological shield is required for radiation protection of the occupational personnel. A concrete shield has been designed around the periphery of the torus to permit personnel access during reactor operation. The criterion adopted for personnel access is to limit the biological dose rate outside the shield to 0.5 mrem/h during operation. This is consistent with the DOE guidelines for the protection of workers as it reflects a factor of 5 reduction in the absolute limit [16].

The operational dose analysis employed 1-D toroidal cylindrical geometry to calculate the neutron and gamma fluxes outside the biological shield. The flux-to-dose conversion factors were taken from Appendix H of Ref. 17. The maximum dose will occur at the midplane of the outboard side between the outer legs of the TF magnets. Thus, the neutronics model included the blanket, shield, and V.V. while the TF magnets were excluded. The need for a sufficient space behind the V.V. for blanket maintenance requires the biological shield to enclose the torus at a radius of 15 m. The biological shield was chosen to be steel-reinforced concrete and composed of 90 vol% concrete,

5 vol% mild steel, and 5 vol% void/He coolant. The constituents of the concrete are 52% O, 34% Si, 6% Al, 5.6% Ca, 1.4% Fe and 1% H, by weight.

The attenuation of the dose with the shield thickness is demonstrated in Figure 19.4-5. The results indicate that 2.8 m thick steel-reinforced concrete limits the biological dose outside the shield to 0.5 mrem/h during operation. As noticed, the dose inside the concrete shield is quite high ($\sim 10^8$ mrem/h) and the corresponding neutron flux level is $\sim 10^{10}$ n/cm²·s. In fact, there is no restriction on the dose level in the reactor hall since all in-vessel components will be maintained remotely and the magnet shield design does not allow for hand-on maintenance or personnel access into the reactor hall shortly after shutdown.

19.5. ACTIVATION ANALYSIS

Activation analysis was performed for the ARIES-IV design using the REAC2 based activation cross section and decay data library, and activation calculation code [18]. The one-dimensional toroidal neutronics model employed for the nuclear analysis of blanket and shield was adopted for the activation analysis. As a result of using this model, the neutron wall loading values in the inboard and outboard first wall regions are 2.8 and 4.7 MW/m², respectively, reflecting a consistent peak to average ratio of 1.7 at o/b midplane of the reactor, as indicated in Section 19.2.

19.5.1. Induced Radioactivity

Table 19.5-I lists the types, half-lives, and inventories (Ci/cc) of radionuclides induced in pure SiC due to neutron interaction immediately after 4 years of operation. The results shown in Table 19.5-I were calculated for both the first wall and breeding zone components in the inboard and outboard region of the reactor. The types of radionuclides induced in pure SiC include the following:

(a) very short-lived (half-life less than 10 minutes) radionuclides - Al28, Al29, and Mg27. The contribution of their activities is important only within the first hour after shutdown;

(b) short-lived (half-life less than one day) radionuclides - F18, Na24, Mg28, and Si31. The activities from this group of radionuclides will dominate the radioactivity shortly (hours) after the decay of the very short-lived radionuclides. They will remain dominating until a few days after shutdown;

Figure 19.4-5. Attenuation of operational biological dose with steel-reinforced concrete.

Table 19.5-I.

SHUTDOWN ISOTOPIC INVENTORIES (Ci/cm³) IN PURE SiC
IRRADIATION TIME: 4 YEARS
WALL LOADING: INBOARD 2.80 MW/m²
OUTBOARD 4.73 MW/m²

Isotope	Half-Life	Outboard	Inboard	Outboard	Inboard
		First Wall	First Wall	Breeding Zone	Breeding Zone
H3	12.3 y	6.15e-01	5.52e-01	6.81-02	9.66e-02
Be10	1.6 My	3.09e-07	2.71e-07	4.81e-08	6.50e-08
C14	5.7 ky	1.19e-06	1.06e-06	3.16e-07	4.29e-07
F18	110 min	5.26e-08	3.30e-08	1.66e-11	6.08e-11
Na22	2.6 y	4.94e-05	3.57e-05	9.45e-08	2.60e-07
Na24	15 h	3.87e-01	3.04e-01	7.19e-03	1.36e-02
Mg27	9.5 min	1.91	1.71	2.03e-01	2.91e-01
Mg28	21 h	2.87e-02	2.64e-02	2.98e-03	4.35e-0
Al26	720 ky	3.46e-07	2.85e-07	4.60e-09	9.31e-09
Al28	2.3 min	59.6	52.1	9.42	11.1
Al29	6.5 min	2.79	2.43	4.46e-01	6.01e-01
Si31	2.6 h	4.13e-02	4.01e-02	2.20e-02	3.00e-02
Si32	330 y	1.08e-14	8.98e-15	2.37e-16	8.52e-16
P32	14.3 d	2.17e-05	2.04e-05	6.19e-06	1.15e-05

(c) medium-lived (half-life less than 20 years) radionuclides- P32, Na22, and H3. H3 activity is the principal contributor to the total radioactive inventory if the entire induced tritium inventory is to retain in the SiC structure. P32 and Na22 activities are derived from second generation reactions and are naturally much less (a few orders of magnitude), than H3 activity. If H3 can be diffused out of the structural SiC material significantly, the activity in SiC will then be dominated by Na24 which decays away rather quickly because of the relatively short half-life (15 h); and

(d) long-lived (half-life longer than 20 years) radionuclides- Be10, C14, Al26, and Si32. These radionuclides and their concentrations will determine the acceptable waste management scenario for long-term environmental considerations.

The total radioactivities averaged over the first wall and breeding zone components in both outboard and inboard regions are depicted in Figure 19.5-1 as a function of time after 4 years of continuous operation.

Note that the activities between a few hours and a few tens of years after shutdown are primarily due to H3 inventories. If the H3 activity is reduced because of diffusion, the total radio- activity within the above mentioned time regime will drop significantly, as revealed in Figure 19.5-2 for the afterheat where H3 does not contribute to it drastically. Figure 19.5-2 gives the afterheat heating rates for the first wall and breeding zone components as a function of time after shutdown. As seen in Figure 19.5-2, the afterheat heating values representing the two humps shortly (minutes) after shutdown are due to Na24 (15 h) and Na22 (2.6 y) activities.

Table 19.5-II gives the afterheat heating rates (W/cc) and integrated decay energies (J/cc) at 1 min, 1 h, 1 d, 10 d, 0.1 y and 1 y after shutdown. The corresponding temperature increases in the various first wall and breeding zone components due to adiabatic afterheat heating are shown in Table 19.5-III. As shown in Table 19.5-III, the maximum adiabatic temperature increases, which occur at the outboard first wall, are 66, 190, and 405 C at 1 h, 1 d, and 1 y after shutdown, respectively.

The issue of impurity elements is discussed below. Due to the very low activation nature of pure SiC, the shutdown radio- activity will be dominated by the impurities if the impurity content is significant. However, a recent analysis shows that the SiC composite material can be controlled to reduce the level of impurity elements. Table 19.5-IV shows the measured impurity elements and concentration levels. Note that, as shown in Table 19.5-IV, the concentrations of most impurity elements except iron are less than 0.5 ppm. The iron impurity is found to be about 3.4 ppm, also too small to be significantly considered in the safety and radiation hazard analyses of a fusion power reactor.

Figure 19.5-1.

Figure 19.5-2.

Table 19.5-II.

**SHUTDOWN AFTERHEAT HEATING RATES (W/cc) AND
INTEGRATED DECAY ENERGIES (J/cc)
FOR SiC IN THE FIRST WALL AND BREEDING ZONE**

Time After Shutdown	First Wall				Breeding Zone			
	Outboard		Inboard		Outboard		Inboard	
	A	E	A	E	A	E	A	E
1 min	0.846	59.3	0.739	51.8	0.132	9.22	0.178	12.5
1 h	0.0109	283	0.00865	245	0.000313	38.9	0.000531	53.0
1 d	0.00358	817	0.00283	667	0.000079	51.0	0.000142	74.7
10 d	0.000022	1110	0.000019	899	0.000002	59.0	0.000003	88.4
0.1 y	0.000021	1160	0.000019	942	0.000002	64.3	0.000003	95.9
1 y	0.000020	1740	0.000018	1460	0.000002	128	0.000003	186

A: Afterheat heating rate

E: Integrated Decay Energy

19.5.2. Waste Disposal Rating

The ratings of ARIES-IV first wall and breeding zone SiC components for 10 CFR 61 Class C waste disposal are primarily dominated by Al26 (720,000 y half-life) and impurity elements such as Ag and Nb. Al26 is produced by the following multistep reactions:

Table 19.5-V shows the waste disposal ratings due to Al26 in the first wall and breeding zone SiC components. The waste disposal ratings averaged over these two components are also given in Table 19.5-V. As shown in Table 19.5-V, the mixture over first wall and breeding zone components, both inboard and outboard, will qualify as a 10 CFR 61 Class C waste after 4 years of continuous operation, although the first wall component alone

Table 19.5-III.

ADIABATIC TEMPERATURE RISE (C) IN SiC DUE TO AFTERHEAT
(SiC: DENSITY = 3.1 g/cc; C_p = 1.4 J/g/K)

Time After Shutdown	First Wall		Breeding Zone	
	Outboard	Inboard	Outboard	Inboard
1 min	14	12	2.2	2.9
1 h	66	57	9.1	13
1 d	190	155	12	17
10 d	258	209	14	20
0.1 y	270	219	15	22
1 y	405	340	30	43

Table 19.5-IV.

IMPURITY CONCENTRATION IN SiC [19]

Element	Concentration (ppm)
Cd	0.158
Co	0.0039
Cr	0.416
Eu	0.0097
Fe	3.4
Hf	0.0016
Ir	0.000013
Ni	0.016
Sb	0.0133
Ta	0.0011

Table 19.5-V.

**WASTE DISPOSAL RATINGS (10 CFR 61 CLASS C) FOR VARIOUS SiC COMPONENTS AFTER 4 YEARS OF CONTINUOUS OPERATION
(100% DENSE SiC MATERIAL)**

Outboard			Inboard		
First Wall	Breeding Zone	Average*	First Wall	Breeding Zone	Average*
1.5	0.021	0.13	1.3	0.042	0.15

*Average value over first wall and breeding zone.

Table 19.5-VI.

**ALLOWABLE Nb AND Ag IMPURITY CONCENTRATIONS (ppm)
IN SiC STRUCTURE FOR MEETING 10 CFR 61
CLASS C WASTE DISPOSAL REQUIREMENT
(AFTER 4 YEARS OF CONTINUOUS OPERATION)**

Outboard			
Impurity Element	First Wall	Breeding Zone	Average*
Ag	0.82	2.7	2.3
Nb	2.3	3.3	3.2

*Average value over first wall and breeding zone.

may not. However, if the operation time is 2.8 years instead of 4, the first wall component alone will become a Class C waste. At this operating time, the corresponding maximum exposure at the first wall is 13 Mw-y/m².

The allowable concentrations for representative impurity elements whose long-lived activity may impact the waste disposal rating are given in Table 19.5-VI. The two representative elements given in Table 19.5-VI are Ag and Nb, whose induced radioactivities favor high energy (n,p), and low energy (n,g) reactions, respectively. The maximum allowable concentrations at the first wall (outboard) and averaged over the first wall and breeding zone components (also outboard) are 0.8 and 2.3 ppm for Ag, respectively; and 2.3 and 3.2 ppm for Nb, respectively. Note these impurity levels were estimated assuming 4 years of continuous operation. If the operating time is reduced, the allowable impurity level will increase accordingly.

19.6. CONCLUSIONS

The major neutronics issues for the ARIES-IV design are the tritium breeding potential of the Li₂O/SiC/He blanket, radiation damage to Be and SiC composites structure, shielding requirements, and radioactivity level. An overall tritium breeding ratio of 1.12 seems adequate for ARIES-IV. The 12% excess breeding margin is necessary to achieve tritium self-sufficiency. Several uncertainties in the design, technical performance of specific reactor components, and some plasma physics parameters were found to determine the excess breeding margin. In all ARIES designs, a serious effort was developed to lower the breeding requirements by increasing the tritium (T) burn-up in the plasma, ensuring extremely low T losses, achieving low T inventory in all subsystems, and requiring high reliability and fix rate for the T processing systems. The uncertainties in the cross section data determined by several recent US-Japan integral experiments and the approximate 1-D geometry used to model the blanket for the radiation transport code constitute a large portion (~ 10%) of the excess breeding. The neutronics analysis indicates that a blanket utilizing Li₂O breeder and SiC composites structure will not provide tritium self-sufficiency unless a neutron multiplier is incorporated in the design. Among several Be compounds (Be, BeO, and Be₂C), only Be metal was found to meet the ARIES-IV breeding requirements. For economic and safety reasons, an effort was made to reduce the amount of Be utilized in the blanket. A minimum amount of 33 tonnes of Be is necessary to achieve tritium self-sufficiency in ARIES-IV, compared to 180 tonnes in ARIES-I. The Be is placed in separate layers behind the first wall of the 35 cm thick inboard and 60 cm thick outboard blankets. A simple approach, proven to

reasonably predict the overall tritium breeding, was applied where the 1-D results were coupled with the coverage fractions of the inboard and outboard blankets. A 3-D model must be considered in future designs to reduce the errors associated with the final results.

The major problem with the use of Be as a neutron multiplier is the production of large amounts of helium gas and the subsequent swelling induced by that gas. The effect of helium production on the swelling and thermal properties of Be at the operating conditions of ARIES-IV (1000°C, 13 MWy/m², and 53,000 He appm) needs to be understood. The presently available data are very limited and is somewhat ambiguous in that some experimental data show high volumetric swelling (> 100%) while other expectations indicate lower effect (~ 30% swelling). Irradiation experiments need to be conducted to resolve this issue particularly for ARIES-IV where the design relies on the structural integrity of Be for an electrical conducting shell. Modifications to the present blanket design are needed to accommodate the swelling of Be. The high cost of Be and its scarce availability implies that ways should be found to recycle the radioactive Be.

There are some radiation damage concerns for the SiC/SiC composites structure. The burn-up of SiC atoms have the most impact on the integrity and strength of the SiC structure. The high energy neutrons (4-14 MeV) interact with and, thus, burn the Si and C atoms generating large amounts of helium and hydrogen gases. The effect of these gases on the interfacial areas and strength of the composites are totally unknown. It has been postulated that the SiC composites can stand up to 3% burn-up without significant damage. This implies that the SiC structure will not survive the intense radiation environment of ARIES-IV and will require frequent replacement (every 3.7 years) over the 40 years of reactor operation. This means that a total of 4,000 tonnes of Class C waste from changing out the blanket will be disposed of during reactor operation. To strengthen the credibility of the SiC composites as structural material for fusion reactors, 14 MeV neutron irradiation experiments need to be conducted to study the effect of transmutation products on the thermo-mechanical properties of the SiC composites. In parallel, the manufacturing technique of SiC composites must be developed to meet the fusion-specific needs. It should be mentioned that while SiC composites can be tested in fission reactor facilities, such facilities will not provide an adequate test because the neutron flux spectrum is much softer than that of fusion reactors, meaning the absence of He and H production.

Sufficient shield was placed behind the blanket to protect the TF magnets. High performance SiC/B₄C shield is employed for the inboard side to reduce the overall size and cost of the machine. A less efficient and cheaper SiC shield is utilized in the divertor and outboard regions where the space is less constrained. The SiC has a less shielding

performance compared to metals, requires a relatively thicker shield to protect the magnet, and thus results in a bigger machine. It is highly recommended to optimize the shield in order to reduce the radiation damage at the magnet. It is also mandatory to employ realistic models in the shielding analysis such as toroidal geometry specially for the inboard region, heterogeneous layers, forms of materials, and appropriate coolant and structural fractions in the shield. The option of using low activation stainless steel filler materials in the SiC shield should be assessed as it provides some economic advantages. A rigorous afterheat analysis should be performed to evaluate the safety impact of this option. During the ARIES-IV study, considerable emphasis was placed on the neutronics performance while very little attention was paid to the mechanical design of the shield and this area requires further work to better define the structural requirements of the shield modules. A rough estimate for the radiation streaming into the various penetrations reveals that, for most cases, there is sufficient space to accommodate local shields around the penetrations. However, the space behind the inner divertor slot is marginal. This appears to be a shortcoming for the gaseous divertor configuration. Calculations of the flux level outside the torus indicates that the biological dose is quite high and exceeds the limits for hands-on-maintenance meaning that a remote maintenance capability should be fully integrated in the designs for repair of module failures or changing of blankets. This had a negative impact on the maintenance scheme and the availability of the machine.

The SiC possesses low activation and very low afterheat characteristics compared to metals. This was confirmed by the detailed activation analysis which indicates that both blanket and shield easily qualify as Class C low level waste. An independent FW barely qualifies as Class C waste and should be integrated with the blanket for this reason. The Class C waste of the FW, blanket, and shield was achieved by strictly controlling the SiC impurities to the ppb level (especially Nb and Ag). The feasibility of manufacturing SiC composites with such low impurity levels and the impact on the cost of the SiC composites need further assessment. Significant amounts of tritium (~ 300 g) will be generated in the gas-leak-tight coated SiC structure of the FW/blanket and the safety impact of such tritium level should be addressed. The analysis also indicates that the activation and afterheat are mainly generated by the FW and blanket while the shield has a minor impact on the overall radioactivity level. The very small afterheat is spread out over a large mass of the blanket and shield and does not present a safety problem in the event of a loss of coolant accident.

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C Appendix: ARIES Lessons Learned Contributions

Lessons Learned – ARIES-II

The major neutronics issues for the ARIES-II design are the tritium breeding potential of the Li/V blanket, radiation damage to the V structure, shielding requirements, and radioactivity level. An overall tritium breeding ratio of 1.12 seems adequate for ARIES-II. The 12% excess breeding margin is necessary to achieve tritium self-sufficiency. Several uncertainties in the design, technical performance of specific reactor components, and some plasma physics parameters were found to determine the excess breeding margin. The uncertainties in the cross section data determined by several recent US-Japan integral experiments and the approximate 1-D geometry used to model the blanket for the radiation transport code constitute a large portion ($\sim 10\%$) of the excess breeding. The neutronics analysis indicates that it is possible to breed sufficient tritium with 20 cm thick inboard and 50 cm thick outboard Li/V blankets with no need for a neutron multiplier. A simple approach, proven to reasonably predict the overall tritium breeding, was applied where the 1-D results were coupled with the coverage fractions of the inboard and outboard blankets. A realistic 3-D model must be considered in future designs to reduce the errors associated with the final results.

There are some radiation damage concerns for the V structure. The neutron-induced atomic displacement and helium generation have the most impact on the structural integrity of V. It has been postulated that the candidate V5Cr5Ti alloy for ARIES-II can stand up to 200 dpa without significant damage. It should be mentioned that there is no neutron irradiation data to support such high atomic displacement level. The 200 dpa limit implies that the V structure will not survive the intense radiation environment of ARIES-II and will require frequent replacement (every 4 years) over the 40 years of reactor operation. This means that a total of 4000 tonnes of Class C waste from changing out the blanket will be disposed of during reactor operation. The high level of helium production is another concern for V. The 1300 appm attained at the end of the 4 y FW/blanket life hopefully will not pose significant problems for the dimensional stability of the V structure. Irradiation experiments are needed to confirm the radiation stability of the V5Cr5Ti alloy at the 200 dpa and 1300 He appm levels attainable in commercial fusion power reactors. Another concern is the radiation damage to the V insulating coatings used to alleviate MHD problems. The degradation of the electric resistance, level of swelling, and cracking of the coatings are not known for the 10^{23} n/cm² fluence level of ARIES-II. Test facilities are needed to prove that such coatings are viable under the ARIES-II operating conditions.

Sufficient shield was placed behind the blanket to protect the TF magnets. High performance shield is employed for the inboard side to reduce the overall size and cost of

the machine. A less efficient and cheaper shield is utilized in the divertor and outboard regions where the space is less constrained. It is highly recommended to optimize the shield in order to reduce the radiation damage at the magnet. It is also mandatory to employ realistic models in the shielding analysis such as toroidal geometry specially for the inboard region, heterogeneous layers, forms of materials, and appropriate coolant and structural contents for the shield. The use of low activation stainless steel filler materials in the V shield had a positive economic impact while the safety of the device was maintained. It has been found that the less costly steel yields higher radiation attenuation and energy multiplication than V resulting in a significant reduction in COE. During the ARIES-II study, considerable emphasis was placed on the neutronics performance while very little attention was paid to the mechanical design of the shield and this area requires further work. A rough estimate for the radiation streaming into the various penetrations reveals that, for most cases, there is sufficient space to accommodate local shields around the penetrations. However, the space behind the inner divertor slot is marginal. This appears to be a shortcoming for the gaseous divertor configuration. Calculations of the flux level outside the torus indicates that the biological dose is quite high and exceeds the limits for hands-on-maintenance meaning that a remote maintenance capability should be fully integrated in the designs for repair of module failures or changing of blankets. This had a negative impact on the maintenance scheme and the availability of the machine.

The vanadium possesses low activation and afterheat characteristics compared to other metallic structural materials. This was confirmed by the detailed activation analysis which indicates that both blanket and shield easily qualify as Class C low level waste. This was made possible by utilizing low activation steel filler in the shield since an all V shield, irradiated for 30 FPY, will hardly qualify for Class C waste. The analysis also indicates that the activation and afterheat are mainly generated by the FW and blanket while the shield has a minor impact on the overall radioactivity level. The small amount of afterheat (2% of the thermal power) is spread out over a large mass of the blanket and shield and does not seem to present a safety problem even in the event of a loss of coolant accident. It should be mentioned that the nitride coating and the tritium-barrier coating needed for V were ignored in the activation calculations. The effect of these coatings on the radioactive inventory of the V system needs to be assessed.

Lessons Learned – ARIES-IV

The major neutronics issues for the ARIES-IV design are the tritium breeding potential of the Li/V blanket, radiation damage to Be and SiC composites structure, shielding requirements, and radioactivity level. An overall tritium breeding ratio of 1.12 seems adequate for ARIES-IV. The 12% excess breeding margin is necessary to achieve tritium self-sufficiency. Several uncertainties in the design, technical performance of specific reactor components, and some plasma physics parameters were found to determine the excess breeding margin. In all ARIES designs, a serious effort was developed to lower the breeding requirements by increasing the tritium (T) burn-up in the plasma, ensuring extremely low T losses, achieving low T inventory in all subsystems, and requiring high reliability and fix rate for the T processing systems. The uncertainties in the cross section data determined by several recent US-Japan integral experiments and the approximate 1-D geometry used to model the blanket for the radiation transport code constitute a large portion ($\sim 10\%$) of the excess breeding. The neutronics analysis indicates that a blanket utilizing Li_2O breeder and SiC composites structure will not provide tritium self-sufficiency unless a neutron multiplier is incorporated in the design. Among several Be compounds (Be, BeO, and Be_2C), only Be metal was found to meet the ARIES-IV breeding requirements. For economic and safety reasons, an effort was made to reduce the amount of Be utilized in the blanket. A minimum amount of 33 tonnes of Be is necessary to achieve tritium self-sufficiency in ARIES-IV, compared to 180 tonnes in ARIES-I. The Be is placed in separate layers behind the first wall of the 35 cm thick inboard and 60 cm thick outboard blankets. A simple approach, proven to reasonably predict the overall tritium breeding, was applied where the 1-D results were coupled with the coverage fractions of the inboard and outboard blankets. A realistic 3-D model must be considered in future designs to reduce the errors associated with the final results.

The major problem with the use of Be as a neutron multiplier is the production of large amounts of helium gas and the subsequent swelling induced by that gas. The effect of helium production on the swelling and thermal properties of Be at the operating conditions of ARIES-IV (1000°C , 13 MWy/m^2 , and $53,000 \text{ He appm}$) needs to be understood. The presently available data are very limited and is somewhat ambiguous in that some experimental data show high volumetric swelling ($> 100\%$) while other expectations indicate lower effect ($\sim 30\%$ swelling). Irradiation experiments need to be conducted to resolve this issue particularly for ARIES-IV where the design relies on the structural integrity of Be for an electrical conducting shell. Modifications to the present blanket design are needed to accommodate the swelling of Be. The high cost of Be and its scarce availability implies that ways should be found to recycle the radioactive Be.

There are some radiation damage concerns for the SiC composite structure. The burn-up of SiC atoms have the most impact on the integrity and strength of the SiC structure. The high energy neutrons (4-14 MeV) interact with and, thus, burn the Si and C atoms generating large amounts of helium and hydrogen gases. The effect of these gases on the interfacial areas and strength of the composites are totally unknown. It has been postulated that the SiC composites can stand up to 3% burn-up without significant damage. This implies that the SiC structure will not survive the intense radiation environment of ARIES-IV and will require frequent replacement (every 3.7 years) over the 40 years of reactor operation. This means that a total of 4,000 tonnes of Class C waste from changing out the blanket will be disposed of during reactor operation. To strengthen the credibility of the SiC composites as structural material for fusion reactors, 14 MeV neutron irradiation experiments need to be conducted to study the effect of transmutation products on the thermo-mechanical properties of the SiC composites. In parallel, the manufacturing technique of SiC composites must be developed to meet the fusion-specific needs. It should be mentioned that while SiC composites can be tested in fission reactor facilities, such facilities will not provide an adequate test because the neutron flux spectrum is much softer, meaning the absence of He and H production, than that of fusion reactors.

Sufficient shield was placed behind the blanket to protect the TF magnets. High performance SiC/B₄C shield is employed for the inboard side to reduce the overall size and cost of the machine. A less efficient and cheaper SiC shield is utilized in the divertor and outboard regions where the space is less constrained. The SiC has a less shielding performance compared to metals, requires a relatively thicker shield to protect the magnet, and thus results in a bigger machine. It is highly recommended to optimize the shield in order to reduce the radiation damage at the magnet. It is also mandatory to employ realistic models in the shielding analysis such as toroidal geometry specially for the inboard region, heterogeneous layers, forms of materials, and appropriate coolant and structural contents for the shield. The option of using low activation stainless steel filler materials in the SiC shield should be assessed as it provides some economic advantages. A rigorous afterheat analysis should be performed to validate the safety impact of this option. During the ARIES-IV study, considerable emphasis was placed on the neutronics performance while very little attention was paid to the mechanical design of the shield and this area requires further work to better define the structural requirements of the shield modules. A rough estimate for the radiation streaming into the various penetrations reveals that, for most cases, there is sufficient space to accommodate local shields around the penetrations. However, the space behind the inner divertor slot is marginal. This appears to be a shortcoming for the gaseous divertor configuration. Calculations

of the flux level outside the torus indicates that the biological dose is quite high and exceeds the limits for hands-on-maintenance meaning that a remote maintenance capability should be fully integrated in the designs for repair of module failures or changing of blankets. This had a negative impact on the maintenance scheme and the availability of the machine.

The SiC possesses low activation and very low afterheat characteristics compared to metals. This was confirmed by the detailed activation analysis which indicates that both blanket and shield easily qualify as Class C low level waste. An independent FW barely qualifies as Class C waste and should be integrated with the blanket for this reason. The Class C waste of the FW, blanket, and shield was achieved by strictly controlling the SiC impurities to the appm level. The feasibility of manufacturing SiC composites with such low impurity levels and the impact on the cost of the SiC composites need further assessment. The analysis also indicates that the activation and afterheat are mainly generated by the FW and blanket while the shield has a minor impact on the overall radioactivity level. The very small afterheat is spread out over a large mass of the blanket and shield and does not present a safety problem in the event of a loss of coolant accident.

I. THE IMPACT OF REACTOR OPERATING TIME ON THE LEVEL OF INDUCED ACTIVITY

The amount of radioactivity present in any reactor at the end of its life will determine the level of radioactive waste that has to be disposed. Disposal of the radioactive waste as a low level waste (LLW) is one of the goals envisioned by the U.S. fusion community in any reactor design. In the mean time, safety analysts are equally concerned with the potential release of the radioactive inventory of any fusion reactor during possible accidents. A worst accident scenario will have to involve the release of the highest amount of radioactivity to the environment. The two main factors that control such a process are the amount of radioactivity present in the reactor at the onset of the accident and the amount of energy available to release it. In most loss of coolant accidents (LOCA), the decay heat generated in the reactor structure represents the major source of energy available to mobilize its radioactive inventory. Keeping in mind the need to quantify the highest possible levels of activity and decay heat generated in any fusion reactor during an accident, it has been common to use the radioactivity values calculated at the end of the reactor life in worst case scenarios. Assuming that the highest radioactivity generated in any fusion reactor occurs at its end-of-life is only accurate if one is mainly concerned with identifying the level of radioactive waste generated during the reactor life-time. On the other hand, safety analysts can not assume that the end-of-life radioactivity generated by radionuclides with short or intermediate half-lives represent the worst case values especially for a typical fusion reactor with as long as 30 year lifetime.

The level of radioactivity generated by any radionuclide at the end of irradiation is controlled by the destruction rate of its parent atom. Assuming that longer irradiation times yield higher shutdown activity might not be correct especially if the parent atom has high destruction cross section. This behavior is more pronounced for radionuclides with short or intermediate half-lives. The peak radioactivity for such radionuclides is usually observed within the first year of irradiation and the level of activity starts to drop with increasing the irradiation time. Hence, using the reactor end-of-life activity in worst case safety analysis of any fusion reactor could result in underestimating the radiological hazard associated with each of the radionuclides present in the reactor at the onset of the accident. In addition to the daughter radionuclide half-life, the level of underestimation is also dependent on the reactor lifetime, the destruction cross section of the parent atoms and the magnitude of the neutron flux. One would expect to observe this behavior more clearly in the reactor first wall and the first few layers of its shield than in layers at the back of the shield which are exposed to a much lower neutron flux (Ref. 1).

Fig. 1 shows the shutdown activity generated in ARIES-III structure during different operation times. The activity induced in the reactor after 6 months of operation is more than twice its value after 30 years of operation. Fig. 2 shows the level of activity present in the reactor during the first 10 years following its shutdown after different operation times. As shown in the figure, the impact of operation time on the level of activity in the reactor starts to disappear after a week from shutdown. Similarly, the decay heat generated in the reactor during the first week of operation is about three times its value after 30 years. As with the activity, the underestimation ratio can not be detected after a week from the reactor shutdown regardless of the duration of its operation. Since the full mobilization of the radioactive inventory in ARIES-III is highly unlikely, the amount of radioactivity released to the environment in an accident would be determined by the highest temperature of the shield. The whole body (WB) early off-site dose were calculated using steel experimental volatility rates at 1200 C (Ref. 2) as a conservative assumption. During a one hour LOCA, the maximum WB early dose at the reactor site boundary (1 km) is 19.2 rem and occurs after only one year of operation (Fig. 3). Using the shutdown activity at the end of the

reactor life (30 years) to calculate the off-site dose result in a WB early dose of only 9.3 rem.

Figure 1. ARIES-III Shutdown Activity as a Function of Operation Time.

Figure 2. Activity in ARIES-III After Shutdown as a Function of Operation Time.

Figure 3. Whole Body Early Dose at 1 km from ARIES-III as a Function of Operation Time.

II. THE IMPORTANCE OF USING THE MIXED NEUTRON FLUX IN ACTIVATION ANALYSIS OF D-³He REACTORS

The D-D and D-T secondary reactions in D-³He reactors provide the neutron source term for most of the radioactivity produced in the reactor structure. The amount of activity produced by any radionuclide depends on the number of its parent atoms present at any given time. One approach to account for the activity induced by both neutron sources in any activation analysis is to conduct two sets of activity calculations using each of the two neutron sources separately. From the nonlinearity of the rate equation, it is clear that performing two separate activity calculations for the D-D and D-T neutron source components and adding the results yields conservative results due to underestimating the destruction of the parent atoms. The overestimation ratio depends on the half-life of the daughter radionuclide, the time of operation, the magnitude of the neutron flux and the destruction cross section of the parent atoms (Ref. 3).

The overestimation ratios are expected to decrease as the neutron spectrum gets softer as a result of slowing down in the reactor shield. Fig. 4 shows the activity induced in the ARIES-III structure by each of the neutron source components in addition to the activity induced if the mixed source is used in the calculations. At shutdown, the overestimation ratio is about 40%. As shown in Fig. 5, in case of the accidental release of 100% of the radioactive inventory of ARIES-III, the overestimation in the calculated whole body early offsite dose is 27%.

Figure 4. Total activity in ARIES-III structure.

Figure 5. Whole body early dose at 1 km from ARIES-III.

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FIGURE 1

FIGURE 2

FIGURE 3

FIGURE 4

FIGURE 5

DRAFT

INPUT FOR ARIES LESSONS LEARNED REPORT

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May 26, 1993

4.6 ADVANCED FUELS

Introduction

This section summarizes the key advanced-fuel lessons learned during the ARIES-III design process---recognizing that there will be some overlap with other sections of this report. The main objectives of the ARIES-III design were to study the trade-offs involved in burning D-3He fuel in a tokamak reactor and to compare a reference D-3He design with the D-T ARIES designs. Based on earlier studies, the expectation was that the physics requirements would be more stringent for a D-3He reactor, but that the engineering constraints would predominantly be relaxed. The ARIES-III design quantified several such considerations in tokamak geometry.

Generally, satisfying the physics constraints for fusion plasmas is more difficult for D-3He fuel than for D-T fuel, because of a much lower fusion power density in the plasma for D-3He. This is shown in Fig. AF-1, where the relative fusion power density is plotted as a function of the plasma ion temperature. The engineering benefits of the D-3He fuel cycle stem from a low neutron production. This is illustrated in Fig. AF-2, where the fraction of fusion power produced in neutrons is plotted as a function of temperature for D-T and D-3He fuel. A primary goal of the ARIES-III study was to examine these benefits in the areas of engineering, environment, and safety.

The scarcity of 3He on the earth has historically kept the interest in this fuel cycle at a relatively low level, and the decision to study a D-3He fueled tokamak reactor was motivated by the recent identification of a large 3He resource on the moon [1]. Although the source of 3He fuel was not addressed as part of the ARIES project, this is certainly a key issue for D-3He reactors.

Physics

The physics analysis of ARIES-III experienced unanticipated problems in achieving MHD kink-mode stability, but settled upon a reference-case beta value of 24%. This value requires the use of a fast-response-time set of feedback coils. Such coils have not been implemented experimentally, and they are considered challenging. In the second-stability, D-T, ARIES-II/IV designs, the choice was made to design for kink stability without feedback coils, resulting in a beta value of 3.4%.

Another important finding was that the bootstrap current, which is generally beneficial for first-stability tokamaks, drove too little current near the axis and too much near the edge in the second-stability regime, thus constraining the design space. The eventual reference point, although optimized to reduce this mismatch, left 25% of the current to be driven. Because a better understanding of the physics of current drive over the past decade has lowered the expected efficiency by a factor of about 20, the 7.5-MA current required to be driven externally in ARIES-III resulted in a neutral-beam power of 160 MW and a recirculating power fraction of 23%. The high plasma temperature of a D-3He plasma reduces the plasma collisionality and enhances the current-drive efficiency. However, the inherently low efficiency of the main technologies considered for the ARIES designs, neutral-beam and radio-frequency current drive, led to the high input power. Several innovative current-drive concepts, including compact-toroid and helicity injection, are presently under consideration, but they were considered too unproven at this time for inclusion in the ARIES designs. Furthermore, these methods tend to drive a

relatively broad current profile, and second stability requires that the current profile be highly peaked. Should these or another advanced current-drive technique prove feasible, it would greatly benefit a D-3He tokamak reactor, as it would considerably open up the available design space.

The ARIES-III study quantified the requirements for the energy confinement time, τ_E , and the fusion-ash particle confinement time, τ_p , for D-3He tokamak reactors in comparison to experimental results. Both first- and second-stability D-3He reactors require energy confinement that is 2--4 times better than present experiments. Achieving this will require operating at high plasma current and high magnetic field. To avoid choking the plasma with fusion ash, low values of τ_p/τ_E compared to most of the present experimental data base must be attained. Figure AF-3 shows the present data base, the ratio of fusion-ash particle confinement time to the total-plasma energy confinement time, and the multiplier for the energy confinement time compared to the 'ITER-89P' scaling relation [2]. The assumed high energy confinement time for ARIES-III is partly motivated by the expectation that transport will be reduced in the second-stability regime due to stabilization by magnetic field shear of modes that potentially are causing the large radial transport in present tokamak experiments. If the indicated values for ARIES-III or a first-stability D-3He reactor do not prove feasible, active pumping of fusion ash from the plasma core will be required. Techniques for this have been proposed, but they have not been demonstrated experimentally.

Pellet fueling would be more difficult for D-3He than for D-T, because the high temperature D-3He plasma quickly ablates a pellet and thus only allows edge fueling. This may be acceptable for the flat radial profiles of the second-stability regime in ARIES-III. For a first-stability reactor, however, radial profiles tend to be peaked, and it may be necessary to use other, more efficient techniques---such as compact-toroid fueling---that are presently under development.

Although most of the ARIES-III design work focused on the second-stability regime, some issues related to a first-stability, D-3He tokamak were investigated early in the study. The decision to choose the second-stability regime for the reference ARIES-III case was primarily motivated by several characteristics of the first-stability regime: the probable need for actively removing fusion ash from the plasma core, a higher power for externally driven plasma current, a higher cost of electricity, and the reliance upon the experimentally unverified use of synchrotron radiation for partial current drive and direct energy conversion. For second stability, the main issues were operation in the second-stability regime, the high H-mode energy-confinement multiplier, a large bootstrap current overdrive, and the difficulty of stabilizing kink-modes.

Engineering

Early in the ARIES-III study, the decision was made to use structural materials that would require minimal development. Therefore, a low-activation ferritic steel, modified HT-9, was chosen for the first wall and shield. The low-activation steels require the replacement of refractory metals, such as molybdenum, nickel, and niobium, by other refractory metals, such as tungsten. Their development is not expected to be difficult. Due to the low neutron wall loading, calculations predict that the first wall and shield will last the reactor lifetime (30 full-power years). This is a major advantage for a D-3He reactor compared to a D-T reactor---where the difficult task of remotely replacing highly radioactive reactor-core components must be performed several times during the reactor lifetime. Although the ARIES project did not have the resources to address the critical question of reactor availability, the full-lifetime first wall and structure would likely have a significant effect.

The low neutron fluence (3 MW-y/m²) not only minimizes maintenance requirements, but it also greatly reduces the amount of radioactive waste material requiring disposal, both during the operating years of the reactor and

in total. The amounts of material involved are shown in Fig. AF-4. Furthermore, the waste material from a D-3He reactor will qualify as Class A, necessitating only shallow burial and minimal monitoring, whereas the D-T reactor waste material will be Class C, requiring encasement in a strong container, deeper burial, and longer monitoring.

With regard to disruptions, the difficult problem of analyzing the forces generated by image currents in the structure during the current quench phase was not analyzed in depth. Nevertheless, preliminary calculations indicate that these forces are manageable, because the lack of a tritium-breeding blanket allows the forces to be taken in a shield that can be much more robust than the blanket in a typical D-T design. A more fundamental problem is the thermal quench phase, because the plasma stored energy for a given fusion power in a D-3He reactor will be much higher than in a D-T reactor. Thus, heat deposition and erosion during disruptions may be more constraining for D-3He plasmas.

The high plasma temperature and magnetic field required for burning D-3He fuel in a tokamak lead to a very large synchrotron radiation power generation unless the first wall is highly conducting. A consideration that had not been fully appreciated previously was that the increased resistivity of many materials at the high temperatures required for efficient power cycles makes those materials incompatible with this requirement. For ARIES-III, this problem was solved by using a thin coating of beryllium on the steel first wall.

It was originally hoped that the startup of ARIES-III could be accomplished without the use of a D-T boost phase. However, to reach the second-stability operating plasma parameters, it is necessary to pass through a first-stability region where the energy confinement is not expected to be as good, and 200 MW of additional input power would be required. Therefore, a D-T phase lasting about 100 s was included in the startup scenario. This phase generated a peak 14-MeV neutron power of about 1000 MW, but it required no input power beyond the steady-state values. The extra heat load on the shields and magnets was found to be manageable, although some concern remains about the thermal stresses during the neutron-power rise time, and the additional neutron fluence was negligible.

The low neutron flux in D-3He reactors allows the consideration of organic coolants, which can operate at high temperature and give better efficiency (44%) than water coolant. In a D-T reactor, the radiolytic decomposition of an organic coolant by neutrons is excessive. In ARIES-III, this process could be kept to the same rate as pyrolytic decomposition and an acceptable amount of waste material would be generated. Unfortunately, the use of an organic coolant introduces an energy source if a fire should occur. In combination with the desire to use nearly off-the-shelf structural materials, this led to sufficient concerns about the release of radioactivity that an LSA-2 level of safety (passive) was assigned to the design, although unanimous agreement on the ARIES-III LSA designation was not achieved. This designation occurred late in the design process, and there was not time to address the question of the trade-offs against a water-cooled system (with lower thermal efficiency but a thinner shield and more safety credits).

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Figure Captions:

AF-1. Relative fusion power density in the plasma for D-T and D-3He (1:1 density

ratio) fuels.

AF-2. Fraction of fusion power produced in neutrons as a function of temperature for D-T and D-3He fuel for several values of the

AF-3. Comparison of the present data base for tokamak energy and particle confinement with the requirements for a first-stability D-3He tokamak reactor (FSR) and the ARIES-III second-stability D-3He tokamak reactor.

AF-4. Amounts of waste material requiring disposal for D-T and D-3He tokamak reactors.

[IN PROGRESS]

FIGURE AF-1

FIGURE AF-2

Figure AF-3

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