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UWFDM-886

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ABSTRACT

An efficient organic cooled low activation ferritic steel first wall and shield has been designed for the D-^{3}He power reactor ARIES-III. The inboard shield is 65 cm thick and provides adequate magnet protection. The steel structure has a peak end-of-life damage of only 63 dpa and is expected to last for the whole reactor life. The total absorbed dose rate in the organic coolant is 10^{26} eV/s resulting in a radiolytic decomposition rate of 171 kg/hr. Tritium production leads to a release rate of 33.5 Ci/d and an acceptable off-site effluent dose $<$ 1.3 mrem/yr.

I. INTRODUCTION

ARIES-III is a conceptual design for a D-^{3}He tokamak fusion power reactor.[1] The 1000 MWe reactor operates in the second stability regime. The plasma major radius is 7.5 m and the aspect ratio is 3. The plasma current is 30 MA and the toroidal field on axis is 7.5 T. Only 4% of the fusion power is carried by neutrons with the rest being carried by charged particles that deposit energy as surface heat on the first wall (FW) and divertor. The average neutron wall loading is only 0.08 MW/m^2 with the peak value at midplane being 0.114 MW/m^2. Since no tritium breeding blanket is required in a D-^{3}He reactor, only a simple relatively thin shield is needed for magnet radiation protection. On the other hand, the peak surface heat flux in the FW is 1.86 MW/m^2. This has placed a premium on the design of the FW which should be capable of removing the high surface heat without exceeding temperature and stress limits.

The large fraction of D-^{3}He power produced in the form of charged particles and synchrotron radiation makes direct conversion very attractive. However, due to the tokamak configuration and the selection of the second stability regime, a thermal conversion system is used. Organic coolants have been used in fission reactors[2] and useful experience of operating an organic cooled system was generated. Organic coolants can be used at a higher temperature than water (450°C vs. 350°C) with a much lower pressure (1 MPa vs. 20 MPa). In a D-^{3}He reactor, the concern regarding radiolysis is much alleviated. The high surface heat flux in ARIES-III requires a coolant with good heat removal capability. In addition, the high capital cost and recirculation power require good thermal conversion efficiency. These requirements resulted in the selection of the organic coolant for the ARIES-III FW and shield. Since the organic coolant is limited to about 450°C, advanced structural materials are not required. The low activation ferritic steel (modified HT-9) was, therefore, selected as the structural material.

Neutronics calculations have been performed to determine the different nuclear parameters for the ARIES-III first wall and shield. The primary objective of the neutronics analysis is to determine the optimum shield design that provides adequate magnet protection. The nuclear heating profiles which provide the input for the thermal-hydraulics analysis and the neutron damage rate in the ferritic steel structural material utilized in the structure lifetime analysis are also determined. The neutronics analysis is aimed also at calculating the tritium production rate in the different components of the first wall and shield. This information is useful for determining the off-site dose resulting from routine tritium release. Another objective of the neutronics analysis is to determine the absorbed dose rate in the organic coolant to be used for calculating the radiolytic decomposition rate of the organic coolant. In this paper, the results of the neutronics analysis for ARIES-III are presented.

II. CALCULATIONAL PROCEDURE

One-dimensional toroidal cylindrical geometry, in which the inboard (IB) and outboard (OB) regions are modeled simultaneously, is used in the neutronics calculations. Hence, the neutronics coupling (reflection and spectral effects) between the outboard and inboard regions is taken into account. A 1.5-mm-thick beryllium coating is used in front of the first wall. A 100-micron-thick tungsten barrier is used between the beryllium and front surface of the first wall to prevent Be-steel interactions. The detailed first wall layered configuration as well as the coolant content in the different zones are used in the calculations.

The one-dimensional discrete ordinates code ONEDANT[3] was used to perform the transport calculations. The analysis uses a P_3 approximation for the scattering cross sections and a S_8 angular quadrature set. A 67-coupled group nuclear data library (46 neutron and 21 gamma) based on the ENDF/B-V evaluation is used in the calculations. This library is based on

TABLE I

Peak and Average Neutron Wall Loading Values (Γ) in the Different Reactor Regions

	Inboard Region	Outboard Region	Divertor Region	Total Reactor
Area (m^2)	246.3	857.5	236	1339.8
Peak Γ (MW/m^2)	0.087	0.114	0.073	0.114
Average Γ (MW/m^2)	0.069	0.093	0.052	0.082
Peaking Factor	1.27	1.23	1.41	1.40

VITAMIN-E[4] for the transport cross sections and KAOS/LIB[5] for the nuclear responses. The neutron source used in the calculations has two components, a DT 14.1 MeV component and a DD 2.45 MeV component. Based on the ARIES-III reference parameters, 36% of the source neutrons have a 14.1 MeV energy and 64% have a 2.45 MeV energy. The neutron source intensity was normalized to yield the appropriate neutron wall loading.

III. NEUTRON WALL LOADING DISTRIBUTION

The poloidal distribution of the neutron wall loading in ARIES-III has been determined using the NEWLIT code.[6] The reference plasma parameters are used to describe the plasma shape and neutron source distribution. The plasma major radius is 7.5 m and the plasma minor radius is 2.5 m. The plasma vertical elongation is 1.842 and the plasma triangularity is 0.814. The magnetic axis, where the neutron source peaks, is located at a major radius of 8.8 m. The total neutron power generated in the plasma is 109.5 MW. Taking into account the plasma density and temperature profiles, the neutron source density is considered to vary as $(1-(a/a_p)^2)^{1.753}$, where a is the minor radius for the flux surface of interest and a_p is the plasma minor radius. Table I gives the neutron wall loading values in the different regions of the reactor. While the inboard scrape-off zone at midplane is 10 cm thick, the outboard scrape-off zone thickness at midplane is increased to 90 cm in order to reduce the peak surface heat flux to less than 2 MW/m^2 as required for achieving tolerable structure temperatures. The coverage fractions for the inboard, outboard, and divertor regions are 15.7%, 73%, and 11.3%, respectively. The coverage fraction of a region is the fraction of source neutrons going directly to that region.

The peak surface heat flux was estimated taking into account the power carried by the bremsstrahlung and synchrotron radiations as well as the transport power from the plasma edge and the divertor. It is assumed that the contribution of bremsstrahlung radiation to the surface heat flux follows the same profile as the neutron wall loading. On the other hand, the synchrotron contribution is assumed to be uniform because it is absorbed during multiple reflections. The peak surface heat flux in the first wall is 1.86 MW/m^2 at the midplane in the outboard region. The peak inboard value is 1.52 MW/m^2 at the midplane.

IV. MAGNET SHIELDING

The intent of the shielding analysis was to design an efficient low activation shield that protects the superconducting magnets against radiation. In order to insure the proper performance of the TF coils of ARIES-III, the radiation effects must be below certain limits. 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/cm^2 to avoid degradation of the critical properties of the Nb_3Sn superconductor material. It is undesirable to subject the magnet to nuclear heating above 50 kW to avoid excessive cryogenic load. A limit of 2 mW/cm^3 is imposed on the peak nuclear heating in the winding pack. The end-of-life dose to the polyimide insulator is limited to 10^{11} rads to ascertain its mechanical and electrical integrity. Our neutronics calculations indicate that the predominant magnet radiation limits are the end-of-life fast neutron fluence and the nuclear heat load to the magnets. Hence, the shield is optimized to primarily minimize these effects.

Many shielding design options were evaluated to assess the ability to protect the TF magnets. The candidate low activation materials, in order of increasing safety attractiveness but decreasing assurance of technological feasibility, are Tenelon, modified HT-9, SiC composites, and C/C composites. Helium, water and organic coolants (OC) were considered in the analysis. The compatibility of the coolants with the various structural materials was taken into account. Certainly, each coolant has its merits and drawbacks. From the shielding viewpoint, the water is superior in slowing down the fast neutrons with the OC having less shielding capability because of the lower hydrogen content. Although the He gas has poor shielding characteristics, the possibility of achieving a high thermal efficiency makes it an attractive coolant for the SiC and C/C composites. The overall size and cost of the reactor is significantly influenced by the size of the inboard space between the plasma and the magnet. Hence, an extensive optimization study was performed to determine the optimal composition and thinnest inboard shield that minimizes the magnet damage for each of the design options considered. Figure 1 gives the minimum inboard shield thickness needed for each of the optimized shield design options to satisfy the fluence limit. The thinnest shield can be achieved using water cooled Tenelon. However, the MHT-9/OC shield is chosen as the reference shield because of the ability to achieve a relatively high thermal efficiency of ~44% compared to ~35% for a water cooled shield.

The results of the optimization analysis indicate that the magnet damage is minimized at 30% OC as shown in Fig. 2. A concern for the OC is the radiolysis of the coolant particularly in the high radiation zone of the shield. A viable solution is to reduce the OC content in the front layer of the shield from 30 to 5%. The effect of the thickness of the 5% OC front layer on the magnet damage was investigated. The results shown in Fig. 3 indicate that the fluence slightly reduces at a 20 cm thick front zone and there is no significant change in heating. Therefore, a 20 cm thick front zone with 5% OC is used in the reference ARIES-III design. Calculations performed for the reference shield design indicated that the end-of-life fast neutron

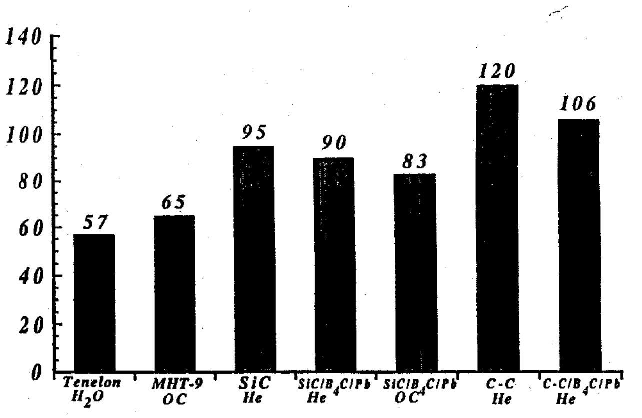

Fig. 1. Minimum inboard shield thickness (cm) required for different shield design options.

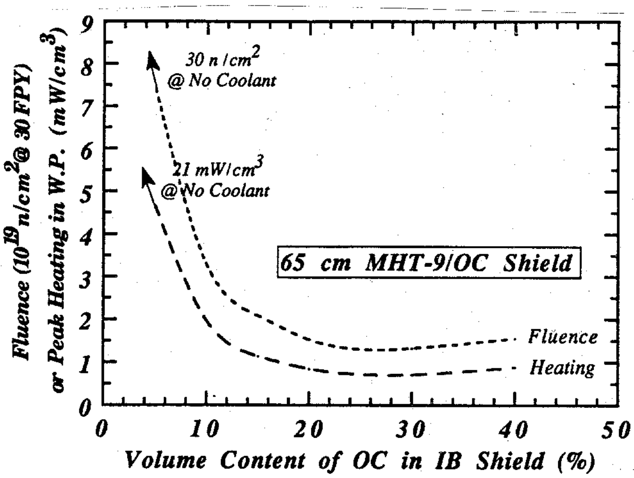

Fig. 2. Impact of organic coolant content on magnet damage.

fluence limit is satisfied and the total nuclear heating in the TF coils is 25 kW. Figure 4 gives the radial build for both the inboard and outboard FW and shield.

V. NUCLEAR HEATING

Accurate determination of energy deposition in the different regions of the FW and shield is essential for performing the thermal-hydraulics analysis. Over 90% of the reactor thermal power is in the form of surface heat deposited at the FW surface. Hence, the thermal performance of the FW is influenced primarily by the surface heat flux. On the other hand, the nuclear heating deposited volumetrically will be the main driver for the thermal performance of the shield. The neutron energy multiplication for the ARIES-III FW and shield was calculated to be 2.2. This value was calculated using the mixed neutron source spectrum with 74.7% of the neutron power being carried by 14.1 MeV neutrons and the rest carried by 2.45 MeV neutrons. Notice that the energy multiplication in ARIES-III is higher than that achieved in D-T reactors. This is attributed to the large energy multiplication for the 2.45 MeV component of the neutron source spectrum calculated to be 4.37. The energy

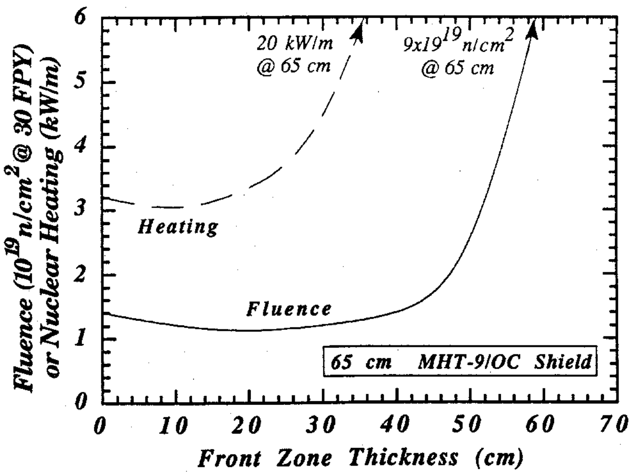

Fig. 3. Effect of varying thickness of the 5% OC front layer of shield.

multiplication for the 14.1 MeV DT component of the spectrum is 1.47. Based on a total neutron power of 109.5 MW generated in the plasma, the amount of nuclear heating in the FW and shield is estimated to be 241 MW. 84.5% of the total nuclear heating is contributed by gamma heating with the rest resulting from neutron heating. About half of the total nuclear heating results from the DT component of the neutron source. The total nuclear heating represents only 8% of the total thermal power of ARIES-III. The contribution to nuclear heating from the different zones of the FW and shield is given in Fig. 5. Most of the nuclear heating is deposited in the first wall and front 20 cm of shield. The peak power densities in the MHT-9 first wall are 2.37 and 2.18 W/cm^3 in the outboard and inboard regions, respectively. The peak power densities in the beryllium coating

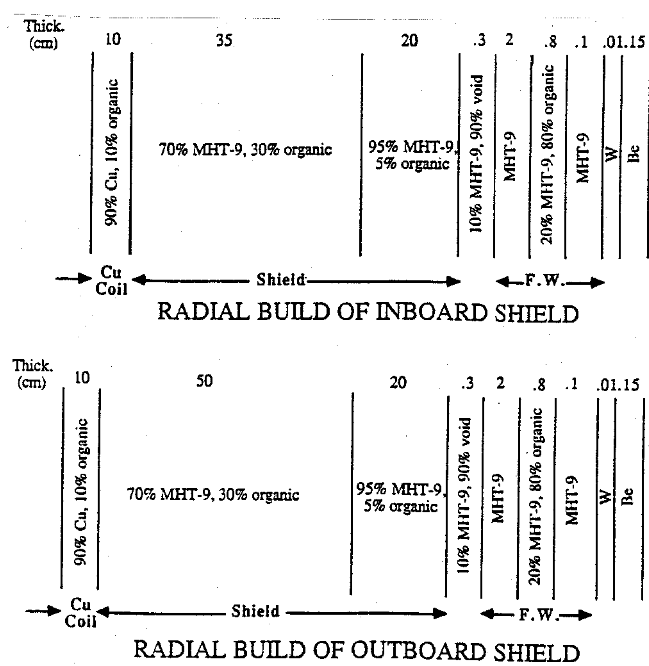

Fig. 4. Radial build of the reference first wall and shield.

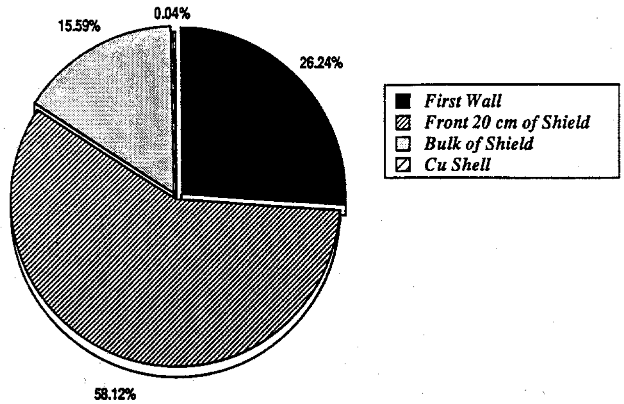

Fig. 5. Contribution to nuclear heating from different first wall and blanket zones.

TABLE II

Radiation Damage to the Modified HT-9 Ferritic Steel First Wall

	Inboard FW	Outboard FW
Peak neutron wall loading (MW/m^2)	0.087	0.114
Peak dpa/FPY	1.93	2.10
Peak He appm/FPY	12.04	13.14
Peak H appm/FPY	41.81	43.49

are 2.06 and 1.93 W/cm^3 for the outboard and inboard regions, respectively.

VI. RADIATION DAMAGE

The damage rate profiles for the modified HT-9 structural material have been determined. The radial variation of dpa and helium production rates as well as nuclear heating at midplane in the outboard FW and shield is shown in Fig. 6. The peak damage rate is 2.1 dpa/FPY and occurs at the front surface of the outboard FW at midplane. For a 30 FPY reactor lifetime, the peak accumulated damage will be 63 dpa implying that FW and shield change-out is not required during the whole reactor life. Table II lists the peak damage parameters for the ferritic steel in both inboard and outboard first walls. The peak helium production rate in the beryllium coating is 441 appm/FPY yielding a burnup rate of only 0.023% per FPY. At reactor end-of-life the peak beryllium burnup is less than 0.7% and is not of concern.

The contribution of the DT neutrons to the different FW damage parameters is shown in Fig. 7. While DT neutrons represent only 36% of the source neutrons, they carry 75% of the neutron power. The DT source neutrons contribute 46%, 95%, and 100% of the FW atomic displacements, hydrogen production and helium production, respectively. The contribution of DT neutrons to the dpa rate is less than their contribution to the neutron wall loading. This is due to the larger number of the low energy DD neutrons that can still produce atomic displacements.

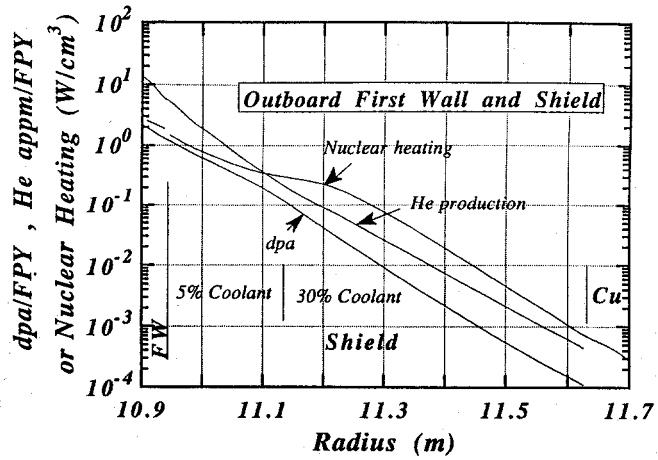

Fig. 6. Damage and nuclear heating profiles in the outboard FW and shield.

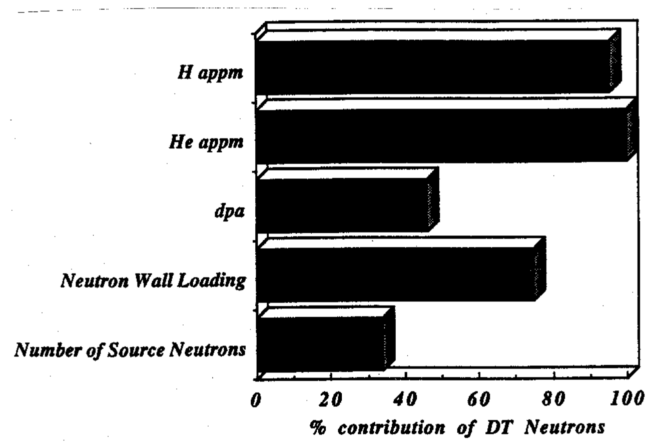

Fig. 7. Contribution of DT neutrons to FW damage parameters.

On the other hand, these DD neutrons have little contribution to the high energy helium and hydrogen production. It is interesting to note that, in ARIES-III, one gets about 20 dpa in steel per 1 $MW\cdot y/m^2$ FW fluence. This is about twice the value obtained in D-T reactors and is due to the fact that a larger number of neutrons are incident on the FW of a D-^{3}He reactor for the same fluence. Hence, one should be very careful in defining the material lifetime when different fuel cycles are used. The accumulated dpa should be used rather than the neutron fluence in $MW\cdot y/m^2$.

VII. TRITIUM PRODUCTION

Tritium is produced in the FW and shield of ARIES-III as a result of neutron interactions with the different materials used. Neutronics calculations have been performed to determine the tritium production rates in the different components of the FW and shield. These values are used to determine the off-site dose resulting from routine tritium release. The results of the calculations were normalized to the average neutron wall loadings of the inboard and outboard regions.

Tritium is produced in the 1.5 mm thick beryllium coating at the rate of 3.29 g per full power year (FPY) with 80% of it produced in the outboard region. Assuming a reactor availability of 75%, the tritium production in the beryllium coating corresponds to a rate of 67.5 Ci per day. Tritium is produced also as a result of (n,t) reactions with the constituent elements of the modified HT-9 structure. The total tritium production rate in the steel structure is 0.36 g/FPY with 83% contributed by the outboard region. This corresponds to a daily production rate of 7.4 Ci/d. Deuterium and helium-3 particles from charge exchange of the edge plasma will impinge on the FW. It was estimated that the FW structure will saturate with inventories of about 20 grams of deuterium and 0.14 grams of helium-3. Neutron interactions with deuterium and helium-3 will produce tritium via the reactions $D(n,\gamma)T$ and $^3He(n,p)T$. The total tritium production rate from the deuterium in the FW is only 80 μCi/d. Due to the large (n,p) reaction cross section for ^{3}He, tritium is produced at a relatively large rate of 3.6 Ci/d from the helium-3 in the FW.

In a nuclear environment, tritium is produced in the organic coolant as a result of neutron interactions with the constituent elements of the coolant. This is mainly due to (n,γ) reactions with the naturally existing deuterium in the organic coolant. In addition, tritium can be produced as a result of neutron double capture in the hydrogen by converting it first to deuterium via the (n,γ) reaction followed by (n,γ) reactions with the produced deuterium. In ARIES-III, 6.9 kg of deuterium exist initially in the HB-40 organic coolant based on the natural abundance of deuterium in hydrogen. Additional deuterium is produced during reactor operation at the rate of 0.12 kg/FPY due to the $H(n,\gamma)D$ reaction. Deuterium builds up in the coolant to 10.6 kg at end-of-life of the ARIES-III reactor (30 FPY) if no coolant reprocessing is employed. The maximum tritium production rate in the coolant at reactor end-of-life is determined to be 7.6×10^{-5} g/FPY which corresponds to 1.6 mCi/d. This is a very small amount compared to tritium production rates in the beryllium coating and steel structure. Hence using the organic coolant in ARIES-III results in negligible tritium production.

Figure 8 shows the tritium production rates in the different components of the FW and shield. Tritium will be released to the atmosphere as a result of incinerating the decomposed organic coolant waste. The tritium release rate to the environment will depend on tritium permeation into the coolant. A conservative estimate was made by assuming that all tritium produced in the FW and shield will diffuse to the coolant except for the beryllium coating where two-thirds of the tritium produced is assumed to go back to the plasma. Hence, the total tritium release rate is estimated to be 33.5 Ci/d.

VIII. ABSORBED DOSE IN THE ORGANIC COOLANT

Using organic coolants in a nuclear environment can lead to decomposition due to radiolysis. The decomposition rate depends on the rate of energy deposition (dose rate). In a fusion reactor, energy is deposited in the coolant as a result of neutron and gamma photon interactions with the constituent elements of the organic coolant. These elements are hydrogen and carbon

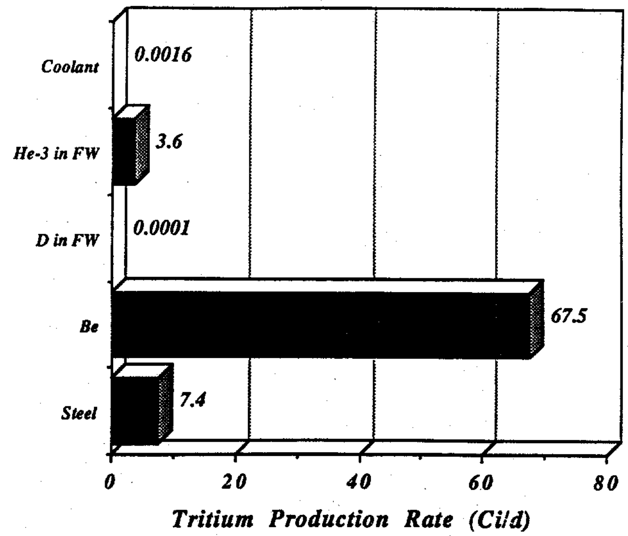

Fig. 8. Tritium production rate in the first wall and shield.

for the HB-40 organic coolant. Neutrons come directly from the fusion reactions in the plasma while the gamma photons result from neutron reactions with the reactor material. The energy deposition rate per unit volume of the coolant varies spatially in the FW, blanket and shield of a fusion reactor. It is highest in the high neutron flux FW and decreases as one moves deeper in the blanket and shield away from the plasma due to the reduction and softening (reduced energy) of the neutron flux. Hence, in order to reduce the total absorbed dose rate and consequently the radiolytic decomposition rate, it is essential to reduce the organic coolant content in the high flux zones without jeopardizing heat removal.

The reduced neutron yield in a D-^{3}He reactor compared to DT reactors results in reduced radiolytic decomposition and allows use of organic coolants. In ARIES-III, the organic coolant HB-40 is used to remove the heat from the ferritic steel first wall and shield. Although the shielding optimization indicated that the optimum coolant content is 30%, only 5% coolant is used in the front 20 cm of the shield to reduce the total absorbed dose rate. Because of the high surface heat flux in D-^{3}He reactors, a relatively large coolant volume fraction of 20% is used in the FW. Neutronics calculations have been performed to determine the absorbed dose rate in the organic coolant for the different FW and shield zones of ARIES-III.

Table III gives the absorbed rate per unit volume of the organic coolant in the different zones. The irradiated coolant volumes in the different zones are also indicated. The total coolant volume inside the reactor is 188 m^3. The total absorbed dose rate in the organic coolant is 1.05×10^{26} eV/s. Only 15% of this dose results from gamma photon interactions with the coolant and the rest is due to neutron interactions. About 80% of the total dose rate is contributed by the outboard region. Although the coolant in the FW represents only 4% of the total coolant volume inside the reactor, about 60% of the total absorbed dose is contributed by the FW. The front 20 cm-thick zone of the shield contributes 25% of the total dose. The radiolytic decomposition rate is 171 kg/hr. The pyrolytic decomposition is mini-

TABLE III

Absorbed Dose Rate in First Wall and Shield Zones

Zone	Coolant Volume (m^3)	Dose Rate ($eV/cm^3 \cdot s$) Neutron	Gamma	Total
FW	7.95	6.26×10^{18}	9.57×10^{17}	7.22×10^{18}
Front of shield (@ 5% OC)	10.52	2.19×10^{18}	3.13×10^{17}	2.50×10^{18}
Bulk of shield (@ 30% OC)	158.02	6.75×10^{16}	4.48×10^{16}	1.12×10^{17}
Cu coil (@ 10% OC)	11.34	4.67×10^{14}	3.44×10^{14}	8.11×10^{14}
Total	187.83	4.65×10^{17}	9.57×10^{16}	5.61×10^{17}

mized in ARIES-III by minimizing the time spent by the coolant in the high temperature regions. The pyrolytic decomposition rate amounts to 307 kg/hr resulting in a total decomposition rate of 478 kg/hr. A hydrocracking process was developed for processing the decomposition waste.[7] It has been demonstrated that 90% of the waste can be recovered. Therefore, only 48 kg/hr of the organic coolant decomposed waste has to be disposed of by incineration.

The radioactivity produced in the coolant is very small[8] and the off-site effluent dose rate resulting from incinerating the coolant waste is determined primarily by the tritium release rate. An acceptable dose rate <1.3 mrem/yr is obtained at 1 km from the reactor.[9]

IX. SUMMARY

The reduced neutron yield for a D-^{3}He fuel cycle reduces radiolytic decomposition and allows organic coolants to be used. A high performance low activation ferritic steel shield cooled with organic coolant has been optimized to provide adequate radiation protection for the superconducting TF coils. The total inboard first wall and shield thickness is 65 cm. The peak neutron wall loading is 0.11 MW/m^2 in the outboard region at reactor midplane yielding an end-of-life peak steel damage of only 63 dpa implying that no change-out is required. About half the damage is contributred by the DT component of the neutron source. The radiolytic decompositon of the organic coolant is reduced by decreasing the coolant content in the front high neutron flux zone of the shield. The total absorbed dose rate in the coolant is 10^{26} eV/s leading to a radiolytic decomposition rate of 171 kg/hr. Tritium is produced in the different FW and shield components with the Be coating being the largest contributor. The total tritium release rate is 33.5 Ci/d yielding an acceptable off-site effluent dose rate < 1.3 mrem/yr.

ACKNOWLEDGEMENT

Support for this work was provided by the U.S. Department of Energy.

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