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ARIES-III**

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ABSTRACT

Activation analysis has been performed for the D-³He fueled reactor ARIES-III. The activity, decay heat and biological hazard potential (BHP) have been calculated for the low activation steel (modified HT-9) first wall and shield as a function of time following the reactor shutdown. The total activity produced in the reactor at shutdown is 1549 MCi. The total activity produced in the reactor organic coolant following 30 full power years of operation without reprocessing is 458 Ci. The modified HT-9 shield qualifies for shallow land burial as Class A low level waste. The biological dose rate after shutdown at the back of the outboard shield is too high to allow hands-on maintenance. Burning all the tritium in the plasma chamber results in increasing the radioactivity generated in ARIES-III by 65% to 85% at different times following the reactor shutdown.

I. INTRODUCTION

ARIES-III is a D-³He fueled tokamak conceptual reactor design which has a major radius of 7.5 m and an aspect ratio of 3. The reactor is operated for 30 full power years and produces a net electric power of 1000 MW_e. The average and peak neutron wall loadings are 0.069 and 0.087 MW/m² for the inboard region, and 0.093 and 0.114 MW/m² for the outboard region. The inboard shield thickness is 64.5 cm and the outboard shield thickness is 80 cm. The shield is made of the low activation ferritic steel (modified HT-9) and cooled with the organic coolant HB-40.

Activation analysis has been performed to identify the possible safety, environmental and radwaste advantages of using the D-³He fuel cycle. We investigated several activation-related issues for both the reactor steel structure and organic coolant. 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 is needed to calculate the potential effects of radioactive inventory release in the event of an accident.¹ 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). Another issue that has been examined in this paper is the waste disposal rating (WDR) of the reactor structure at the end of its lifetime. The waste disposal rating is needed to de-

termine if the structure would satisfy the regulations criteria for shallow land burial as a low level waste (LLW). To assess the feasibility of hands-on maintenance, we calculated the biological dose rate at the back of the outboard shield. The amount of radioactivity generated in the organic coolant has been calculated in order to estimate the effluent dose outside the reactor from the routine release of radioactive nuclides (particularly tritium and ¹⁴C) which are continuously removed by incinerating the decomposed coolant waste.¹

II. CALCULATIONAL PROCEDURE

Calculations for a one-dimensional toroidal cylindrical geometry model were conducted using the DKR-ICF computer code² with activation cross sections taken from the ACTL³ 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 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.

The neutron flux used for the activation calculations was generated by the one-dimensional discrete ordinates neutron transport code ONEDANT⁴ using the ENDF/B-V cross section data. Since the one-dimensional model used in the calculations represents the inboard and outboard first walls by right cylinders with the midplane radii, effective heights for these cylinders were used as the effective height for the inboard and outboard regions. These effective heights were determined such that the first wall area and hence the neutron power incident on the wall is preserved. The effective heights for the inboard and outboard cylinders were taken as 8 and 12.52 meters, respectively.

The total neutron flux resulting from both the DD and DT components of the source and normalized to the proper values of the neutron wall loadings in the inboard and outboard regions is used in the calculations. Using the flux resulting from the mixed neutron source spectrum is essential due to the nonlinearity of the rate equations governing the activation calculations. Performing two separate calculations using flux components resulting from the DD and DT parts of the neutron source spectrum and adding the results yields conservative results due to underestimation of the destruction of the parent nuclides.⁵ The overestimation becomes more pronounced for short lived nuclides, long

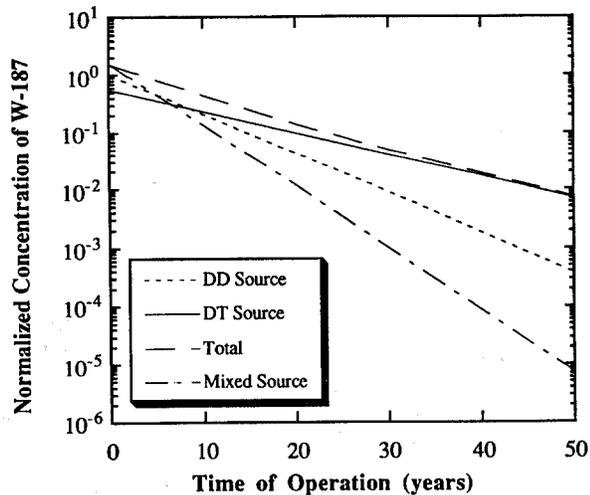


Fig. 1. Overestimation of ^{187}W activity resulting from adding contributions from DD and DT components of the neutron source in the first wall of ARIES-III.

operation time, large neutron flux and large destruction cross sections for the parent nuclides. The effect is shown in Fig. 1 for ^{187}W ($T_{1/2} = 23.9$ hr) which is produced from (n,γ) reactions with ^{186}W . The results shown were obtained using the first wall flux and indicate that the activity will be overestimated by nearly two orders of magnitude if calculations are done separately for the DD and DT source components. Notice that the overestimation will be only by a factor of three for the total ^{187}W inventory in the shield due to the reduced neutron flux in the bulk of the shield.

The coolant activation calculations were performed for the worst case with the organic coolant saturated with nitrogen and argon from the air. The saturation concentration levels at atmospheric pressure and 400 °C have been determined to be 80 and 1.3 $\mu\text{g}/\text{cm}^3$ for nitrogen and argon, respectively. The HB-40 organic coolant used in ARIES-III has a density of 0.85 g/cm^3 and consists of hydrogen and carbon at equal atom densities. Since at any moment only half of the coolant exists inside the reactor, the calculations were performed with an irradiation time of 15 FPY which is half of the reactor lifetime. The results were then multiplied by two to yield the total radioactivity inventory in the coolant system after 30 FPY of reactor operation. The decomposed coolant waste is continuously removed leading to a coolant residence time in the loop of less than a year. Since the radionuclides produced in the coolant will be also continuously removed along with the coolant waste, assuming the activity to build up over the whole reactor lifetime (30 FPY) without coolant reprocessing leads to overestimating the coolant radioactive inventory particularly for those nuclides with half-lives greater than the coolant residence time.

The structure activation results were utilized in the rad-waste classification. The decay gamma source file generated by the DKR-ICF code was used along with the adjoint flux to calculate the biological dose rate after shutdown using the DOSE² code. The elemental composition of the low activation ferritic

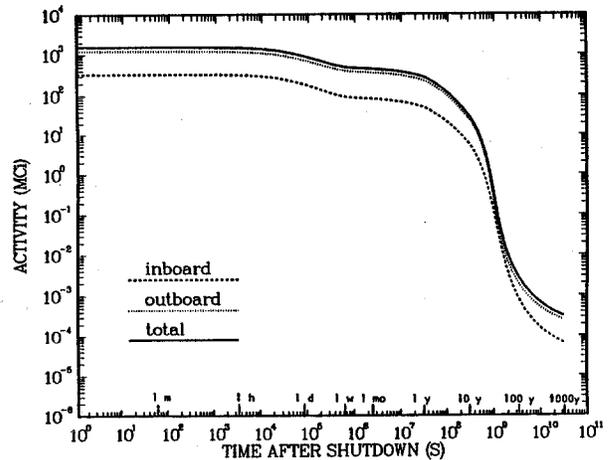


Fig. 2. Activity induced in ARIES-III shield.

steel structural material (modified HT-9) used in the analysis is that presented in the Blanket Comparison and Selection Study (BCSS) report.⁶

III. STRUCTURE ACTIVITY, DECAY HEAT AND BIOLOGICAL HAZARD POTENTIAL (BHP)

The total activity in ARIES-III at shutdown is 1549 MCI. Since the outboard region volume is about 4.75 times the inboard volume and the neutron wall loading on the outboard is about 30% higher than its value on the inboard region, the outboard activity dominates the total activity in ARIES-III at all times following shutdown. The short-term activity after shutdown (≤ 1 day) is dominated by ^{55}Fe ($T_{1/2} = 2.7$ yr), ^{188}Re ($T_{1/2} = 16.95$ hr), ^{56}Mn ($T_{1/2} = 2.6$ hr) and ^{51}Cr ($T_{1/2} = 27.7$ day). In the period between 1 day and 1 year after shutdown, ^{55}Fe and ^{54}Mn in addition to ^{60}Co ($T_{1/2} = 5.27$ yr) and ^{185}W ($T_{1/2} = 75.1$ day) dominate the activity induced in the structure. ^{55}Fe and ^{60}Co are the major contributors up to 10 years following shutdown. The long-term activity comes from ^{63}Ni ($T_{1/2} = 100$ yr), ^{14}C ($T_{1/2} = 5730$ yr) and ^{53}Mn ($T_{1/2} = 3.8 \times 10^6$ yr). Figure 2 shows the total activity induced in the different regions of ARIES-III as a function of time following shutdown. The total structure activity drops to 863 and 254 MCI in one day and one year following shutdown, respectively.

The decay heat generated in ARIES-III is almost dominated by the same isotopes that dominate the level of activity in the reactor after shutdown. ^{56}Mn produces most of the decay heat within the first 8 hours. Within the first year after shutdown, ^{56}Mn , ^{54}Mn , ^{55}Fe and ^{60}Co are the major sources of decay heat. The long-term decay heat is governed by the decay of ^{53}Mn and ^{59}Ni ($T_{1/2} = 80,000$ yr). The total decay heat at shutdown is 4.91 MW and drops to 1.43 MW in one day and 0.21 MW in one year. Figure 3 shows the total integrated decay heat in the dif-

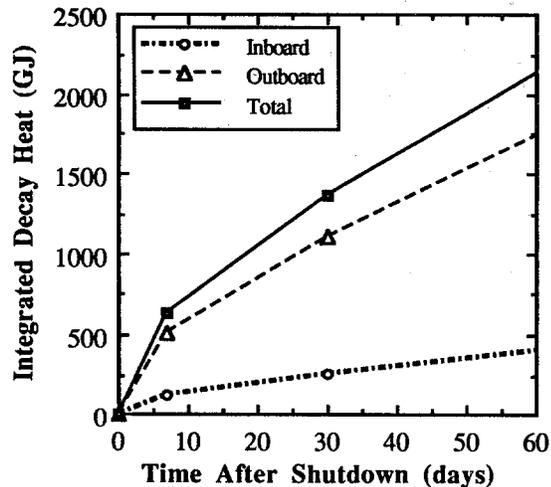


Fig. 3. Integrated decay heat in ARIES-III shield.

ferent regions of ARIES-III during the first 2 months following shutdown. One week after shutdown the values of the integrated decay heat generated are 126 and 509 GJ for the inboard and outboard regions, respectively. These results are useful for predicting the thermal response of the shield to a LOCA.¹

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 10CFR20.⁷ The total BHP in ARIES-III at shutdown is 204×10^6 km³ air with about 80% of it contributed by the outboard region. The short-term BHP is dominated by ¹⁸⁸Re, ⁵⁶Mn and ¹⁸⁷W ($T_{1/2} = 23.9$ hr). ⁶⁰Co and ⁵⁵Fe are the major sources of mid-term BHP (≤ 10 years). Finally, the long-term BHP is produced by ⁵³Mn and ^{108m}Ag ($T_{1/2} = 130$ yr).

IV. COOLANT ACTIVITY

The activity produced in the organic coolant was determined for the worst case with the coolant saturated with nitrogen and argon from the air with concentrations of 80 and 1.3 $\mu\text{g}/\text{cm}^3$, respectively. The total activity produced in the coolant following 30 FPY of operation without reprocessing is shown in Fig. 4. While short-term activity is dominated by ⁴¹Ar ($T_{1/2} = 1.83$ hr), intermediate and long-term activities are dominated by ³H ($T_{1/2} = 12.33$ yr) and ¹⁴C ($T_{1/2} = 5730$ yr), respectively. The total coolant activity at shutdown is 458 Ci and drops to 151 Ci in one year and 44 Ci after 100 years.

It is important to keep in mind that the radioactive inventory of the organic coolant calculated after 30 FPY operation is overestimated as the radionuclides are continuously removed by incinerating the coolant waste resulting from decomposition. Since the coolant residence time in the loop is only 262 days, this effect is not significant only for radionuclides with half-lives much less than a year. This effect will be very pronounced if one examines the calculated ¹⁴C activity. Calculations for 30 FPY of operation without reprocessing yield a ¹⁴C activity of

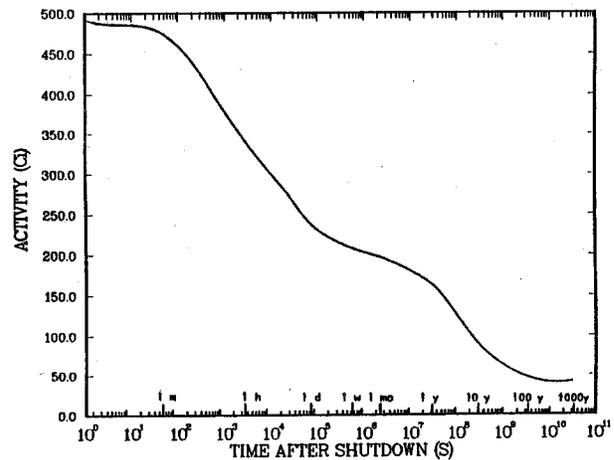


Fig. 4. Activity of the organic coolant.

44 Ci at the end of life. However, ¹⁴C is produced at the rate of 1.1 Ci/year and reaches an equilibrium level of 0.83 Ci in less than a year as a result of the continuous removal of decomposed coolant. This corresponds to a ¹⁴C equilibrium concentration of less than 2.5 mCi/m³ in the coolant. In addition, this coolant activation analysis yields very conservative results due to the fact that many of the impurities used in the calculations that had undetectable concentrations were assigned a concentration corresponding to the experimental detection limits.

V. RADWASTE CLASSIFICATION

The radwaste of ARIES-III structure has been evaluated according to both the NRC 10CFR61⁸ and Fetter⁹ waste disposal concentration limits (WDL). 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 a particular class. If the calculated $WDR \leq 1$ when Class A limits are used, the radwaste should qualify for Class A segregated waste. 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. Using Class C limits, a $WDR > 1$ implies that the radwaste does not qualify for shallow land burial.

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 Fig. 5. The results in the figure are given for both Class A and Class C with the activities averaged over the total volume of the first wall and shield of both the inboard and outboard regions. The 10CFR61 Class A WDR is given after a waiting period of 15 years which is required for the specific activity of short-lived nuclides ($T_{1/2} \leq 5$ years) to drop below 7000 Ci/m³. The 7000 Ci/m³ 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

^{94}Nb as the second major contributor.

VI. BIOLOGICAL DOSE RATE

The biological dose rate has been calculated as a function of time following shutdown at the back of the outboard shield. The results showed that the dose rate at the back of the shield and away from penetrations through the shield is still too high to allow hands-on maintenance. The dose is dominated by ^{56}Mn ($T_{1/2} = 2.6$ hr) and ^{58}Co ($T_{1/2} = 70.8$ day) in the first day. ^{54}Mn ($T_{1/2} = 313$ day) and ^{60}Co ($T_{1/2} = 5.27$ yr) dominate the biological dose in the first few years following shutdown. Figure 6 shows the calculated dose rate at the back of the outboard shield as a function of time following shutdown. The drop in the dose rate at about 8 hours following shutdown is the result of the decay of ^{56}Mn . The dose rate at the back of the outboard shield one day after shutdown is 906 mrem/hr. This implies that the shield required for magnet protection does not allow for hands-on maintenance. If hands-on maintenance is desirable, the dose rate should be kept below 2.5 mrem/hr. Further analysis showed that an additional 50 cm of shielding is needed in the outboard region to allow for hands-on maintenance.

VII. IMPACT OF BURNING ALL TRITIUM IN PLASMA

A major concern about burning only 50% of the tritium produced in the ARIES-III plasma is handling and storing the excess tritium. To avoid the safety implication of tritium handling, the option of a 100% burnup of tritium is considered. Burning all the tritium in the plasma results in doubling the number of 14.1 MeV neutrons generated and hence yielding more high energy threshold reactions which consequently produce more intermediate and long-lived nuclides. Since intermediate and long term activity in the ARIES-III shield is dominated by radionuclides produced by (n,2n) and (n,p) reactions, such activity will almost double as a result of burning all the tritium in the plasma. A comparison between the induced activity, decay heat and biological hazard potential (BHP) for the two options (50% and 100% tritium burnup) is shown in Table I. Burning all the tritium in the plasma results in increasing the induced activity, decay heat and BHP at all times following shutdown. The highest increase is detected within one year after shutdown. During this period of time, the radioactivity induced in ARIES-III is dominated by ^{55}Fe produced mainly by the ^{56}Fe (n,2n) reaction and ^{54}Mn produced by the ^{54}Fe (n,p) and ^{55}Mn (n,2n) reactions. At one year following shutdown, the increase in the induced activity is as high as 85%. At the same time, the increase in the decay heat and BHP is about 65%.

As shown in Fig. 7, burning all the tritium in the ARIES-III plasma would not result in changing the waste disposal ratings. However, a waiting period of 35 years will be needed to allow nuclides with half-lives ≤ 5 years (mostly ^{55}Fe) to decay to a value below 7000 Ci/m³ in order for the structure to qualify as Class A low level waste. The waste disposal ratings for the 100% tritium burnup case are dominated by the same radionuclides that dominate the 50% burnup case if either 10CFR61 or Fetter's limits are used.

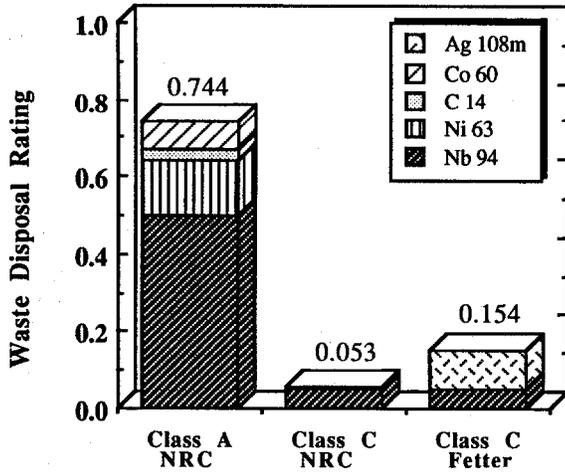


Fig. 5. Waste disposal rating for the ARIES-III shield.

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 a 1 year cooling period for both 10CFR61 and Fetter limits.

The contributions from the different radionuclides to the WDR are shown in Fig. 5. ^{94}Nb ($T_{1/2} = 20,000$ 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 if 10CFR61 limits are used. Other major contributors to Class A are ^{63}Ni ($T_{1/2} = 100$ yr) produced from ^{63}Cu and ^{14}C ($T_{1/2} = 5730$ yr) produced from ^{14}N and ^{17}O . If Fetter limits are used, ^{108m}Ag ($T_{1/2} = 130$ yr) produced from ^{107}Ag becomes the major contributor to the Class C waste disposal rating, followed by

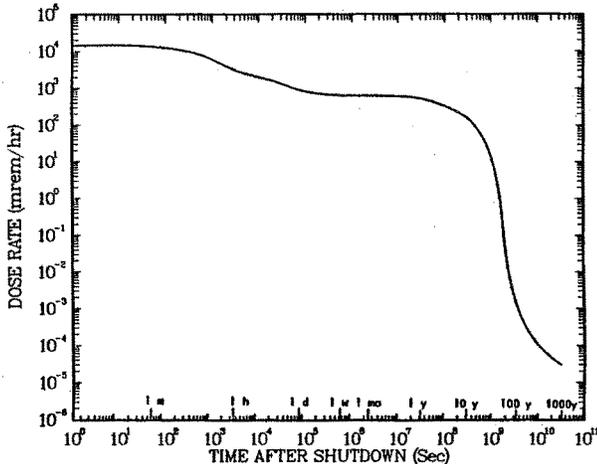


Fig. 6. Dose rate at the back of ARIES-III outboard shield.

Table I

Comparison Between the Induced Activity, BHP and Decay Heat After Shutdown for the 50% and 100% Tritium Burnup Options.

Time	Activity (MCi)		BHP ($10^6 \text{ km}^3 \text{ air}$)		Decay Heat (MW)	
	50% t	100% t	50% t	100% t	50% t	100% t
0	1549	2048	205	270	4.91	6.48
1 d	863	1217	129	180	1.43	1.58
1 w	457	796	78	125	0.45	0.71
1 mo	409	722	72	115	0.39	0.64
1 yr	254	467	35	58	0.21	0.35
10 yr	24	44	3.8	5.5	2.3e-3	3.6e-2

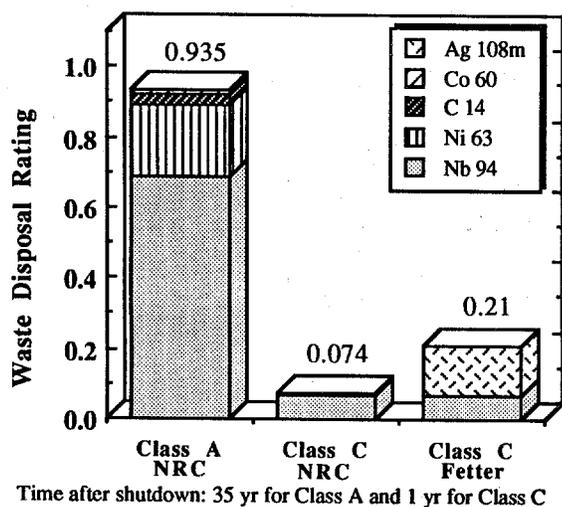


Fig. 7. Waste disposal rating for the ARIES-III shield (with 100% tritium burnup).

Finally, burning all the tritium in the plasma results in doubling the amount of tritium and ^{14}C released routinely to the atmosphere during the organic coolant waste incineration. The release rates for tritium and ^{14}C to the atmosphere in the case of 100% burnup are 67 and 1.66 Ci/day, respectively.

VIII. CONCLUSIONS

The levels of radioactivity generated in the ARIES-III structure and organic coolant are tolerable. The ARIES-III low activation modified HT-9 shield qualifies as Class A low level waste with a waste disposal rating value of 0.74 after a cooling period of 15 years. The Class C waste disposal rating value is 0.053 using the 10CFR61 limits and 0.154 using the limits of Fetter, one year after shutdown. The high level of the biological dose rate following shutdown at the back of the outboard shield

will only allow for remote maintenance. An additional 50 cm of shielding is needed in the outboard region to allow for hands-on maintenance. Burning all the tritium in the plasma results in increasing the radioactivity generated in the reactor by about 75% within the first year following shutdown. It also results in doubling the amount of tritium and ^{14}C released routinely to the atmosphere during the organic coolant waste incineration.

ACKNOWLEDGEMENT

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