ACTIVATION, DECAY HEAT, AND WASTE DISPOSAL ANALYSES FOR THE ARIES-AT POWER PLANT

D. Henderson, L. El-Guebaly, P. Wilson, A. Abdou, and the ARIES Team
University of Wisconsin-Madison, Fusion Technology Institute
1500 Engineering Drive, Madison, WI 53706
(608) 263-0808

ABSTRACT

Detailed activation, decay heat and waste disposal calculations of the ARIES-AT design are performed to evaluate the safety aspects of the device. The high initial activity of the SiC highly irradiated components translates directly into a higher initial decay heat for these structures than for the well-protected steel-based components. However, after a one-hour cool-down period, the SiC decay heat drops by two decades to levels comparable to the steel-based components. The decay heat of the LiPb coolant was found to exceed that of the SiC components for several days after shutdown. This implies that a loss of flow accident (LOFA) event is more critical than a loss of coolant accident (LOCA) event for LiPb/SiC systems. Regarding waste disposal, all structures can easily meet the Class C Low-Level Waste (LLW) requirements established for the ARIES power plants. Many components could qualify as Class A LLW after a 100-year storage period after selection of low activation materials and control of the Nb and Mo impurities in ferritic steel. A purification system will be required to remove the ^{210}Po and ^{203}Hg generated by Pb during operation.

I. INTRODUCTION

The ARIES-AT device is a 1000 MW_e conceptual fusion power plant and the most recent design in a series of tokamak facilities investigated by the multi-institutional ARIES team.[1] The novel features of the device include the use of SiC/SiC composite structures that are low-activation, high-temperature resilient materials, and the use of LiPb that serves both as coolant and breeder. The machine has a lifetime design of 40 full power years (FPY), which except for the first wall, is also the lifetime of most of the components. The plasma facing first wall must be replaced after 4 FPY because of the damage caused by the high neutron wall loading. The average inboard (IB), outboard (OB), and divertor wall loadings are 2.2, 4, and 1 MW/m^2, respectively. The operation schedule of the facility consists of a ~10-month steady-state operational period followed by a two-month extended maintenance period. This amounts to an 80% availability factor for the facility.

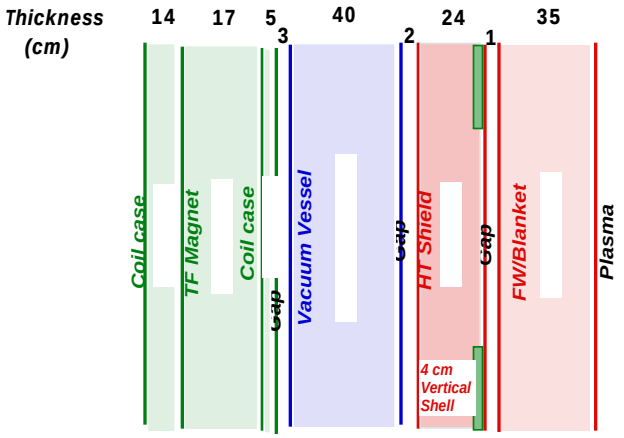

Figure 1. Inboard radial build.

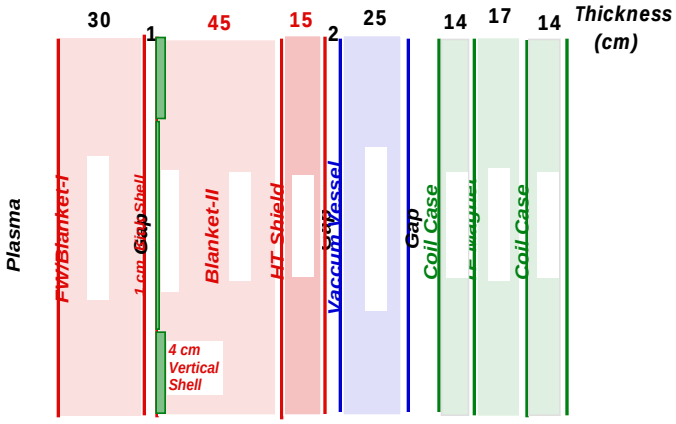

Figure 2. Outboard radial build.

Detailed activation calculations for ARIES-AT are performed to determine the decay heat distribution throughout the device and to compute the Fetter and NRC 10CFR 61 waste disposal ratings (WDR) for various components of the tokamak. These quantities are used to evaluate the safety aspects of the plant. The analysis will determine whether a LOCA or LOFA is the critical accident event and whether the radioactive waste generated during the operation of the facility can be released as cleared metals or qualify for Class A or Class C disposal.

II. COMPUTATIONAL MODEL

The IB and OB ARIES-AT radial builds for the neutronic and activation calculations are depicted in Figures 1 and 2. The constituent material compositions for the various components used in the analysis are provided in Table 1. Impurities for all materials are included in the analysis. The IB high-temperature (HT) shield and OB blanket-II contain the W stabilizing shells required for plasma control. The IB side vacuum vessel (VV) employs tungsten carbide (WC) to reduce the radial standoff.

Table 1. Composition of Various Major Components Comprising the Radial Builds

Inboard Blanket	19% SiC, 81% LiPb*
Outboard Blanket-I	20% SiC, 80% LiPb*
Outboard Blanket-II	19% SiC, 77% LiPb*, 3% W
Inboard HT Shield	15% SiC, 10% LiPb*, 70% B-FS, 5% W
Outboard HT Shield	15% SiC, 10% LiPb*, 75% B-FS
Inboard side V.V.	13% FS, 22% H_2O, 65% WC
Outboard side V.V.	30% FS, 70% H_2O
TF Magnet#	72% Inconel-625, 0.5% Ag, 7% $YBa_2Cu_3O_5$, 7% CeO_2, 13.5% GFF Polyimide

* 90% enriched Li
# Electric insulator not included

The neutron flux throughout the facility was computed using the FENDL-2 175 neutron 42 gamma group coupled cross section library and the DANTSYS[2] discrete ordinates, deterministic transport code. For the activation calculations, use is made of the ALARA[3] activation code, the most recent activation code developed at the Fusion Technology Institute at the University of Wisconsin. The FENDL-2 175 neutron group transmutation cross-section library is employed for these calculations.

The activation and subsequent decay heat and radiological analyses of LiPb coolant/breeder is more complex than that of the other major components. This is because it circulates in and out of the neutron radiation field in five separate coolant flow paths through the device. Each channel carries a portion of the coolant through a sequence of different structural components, each having a different residence time. The LiPb is modeled by considering a given control volume as it circulates through the system. The irradiation history of the control volume can be represented as a pulsed history with one pulse for each pass through the flow channels. The fluence that a given control volume of LiPb receives is a function of the flux spectrum and residence time of the flow path it follows. Several flow paths are modeled with residence times ranging from 1 to 240 s. Finally, the flows from the different flow paths spend approximately two minutes outside the blanket region and mix in various sub-systems (e.g. heat transfer, chemistry control). Hence a given control volume does not necessarily follow the same flow path each time through the device.[4]

III. ACTIVITY AND DECAY HEAT ANALYSIS

The specific activity results for several major components in the inboard and outboard sides of the ARIES-AT tokamak are depicted in Figures 3 and 4. One notes immediately that the IB and OB FW/Blanket-I have the highest activity at shutdown. The activity drops by nearly two orders of magnitude below the level of the ferritic steel components after a one-hour shutdown and by three orders of magnitude over the course of a day. This drop in activity is due to the decay of the short half-life (< 10 min) radionuclides ^{28}Al and ^{27}Mg and the radionuclide ^{31}Si (2.62 hr half-life), which are generated in the SiC structures of the FW/Blanket-I components. The activity level of the vacuum vessel eventually drops to the level of the SiC containing FW/Blanket-I components after 10-100 years after shutdown of the device.

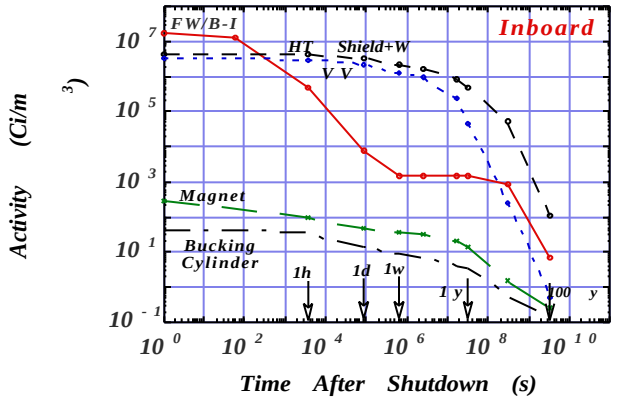

Figure 3. Inboard activity results.

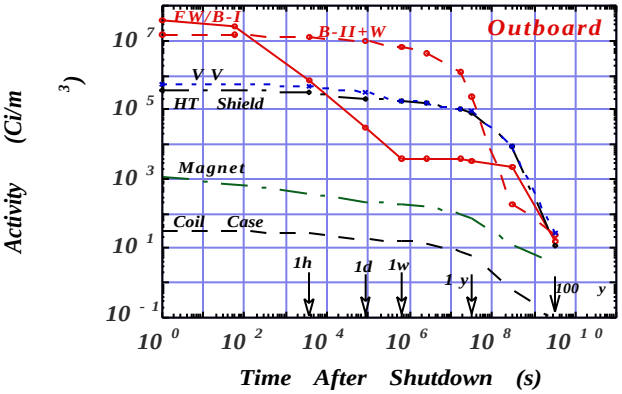

Figure 4. Outboard activity results.

Note that the activity of the OB blanket-II component containing the W stabilizing shells remains fairly level

over several weeks before dropping nearly one order of magnitude one year after shutdown of the facility. Its activity reaches levels comparable to that of the SiC containing FW/Blanket-I structures 5-6 years after shutdown.

The high initial activity of the SiC first wall component translates directly into the initial high decay heat of the first wall components. The time dependent behavior of the decay heat for various components is depicted in Figures 5 and 6. Note that the initial high decay heat of the first wall components drops to levels below that of the ferritic steel components within 20 minutes to one day after shutdown of the facility. This means that SiC contributes to the initial heat load after a LOCA/LOFA event and that the long-term heat sources are the steel and W components. In order to determine which accident event (LOCA or LOFA) is the more crucial event for the facility, the effect of the LiPb decay heat on these accident scenarios must be analyzed.[5]

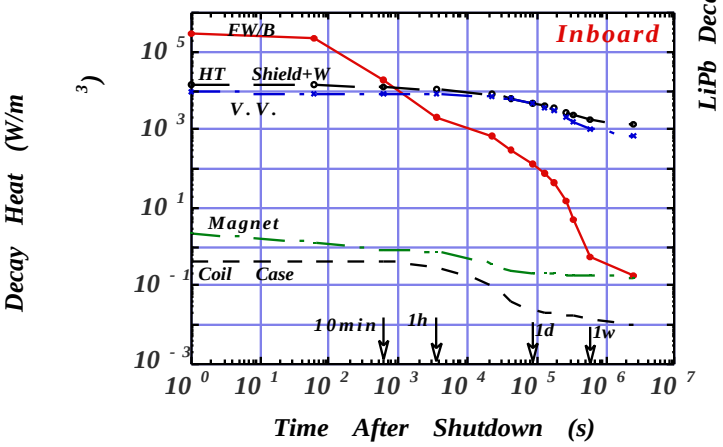

Figure 5. Inboard side decay heat.

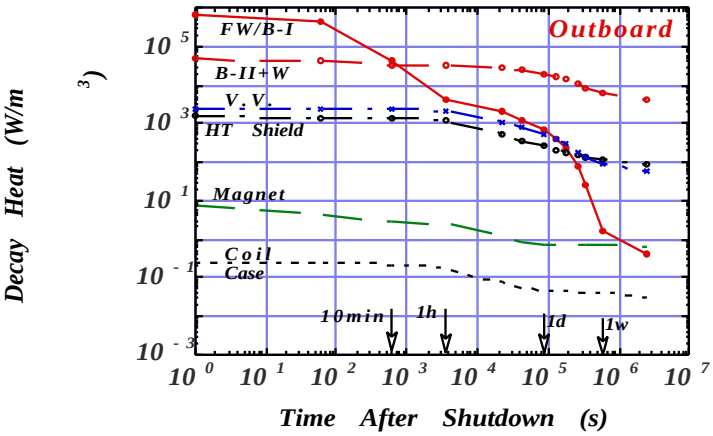

Figure 6. Outboard side decay heat.

Figure 7 provides a decay heat comparison of the LiPb and OB FW/Blanket-I structure. Note that the initial decay heat loading of the OB blanket region is driven by the SiC containing FW structure. As noted earlier this is due to the high activity of the SiC components. Within one hour after shutdown, the decay heat of the FW/Blanket-I structure drops to the same level as that of the LiPb. The decay heat of LiPb exceeds that of the FW/Blanket structure after several hours to weeks after an accident meaning that the dominant high source during this period in the blanket region is LiPb. This implies that the LOFA event is more critical than a LOCA event because the heat load to the structure is higher in the LOFA case than the LOCA case.

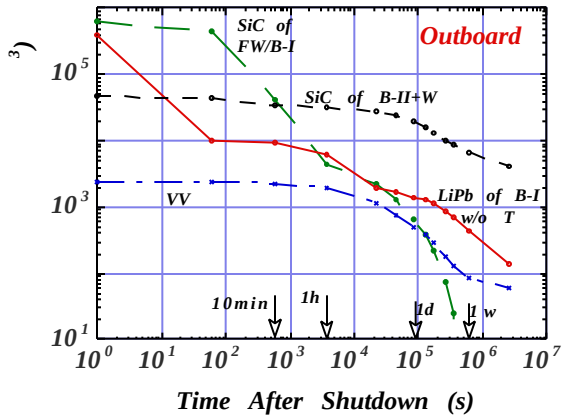

Figure 7. FW/Blanket and LiPb decay heat comparison.

IV. RADIOLOGICAL INVENTORY ASSESSMENT OF LiPb

The radiological hazardous material inventory was modeled using the complex irradiation history outlined in Section II. The primary radiological concern in the LiPb coolant is from the radionuclides ^{210}Po and ^{203}Hg (ref. 4). Both of these isotopes have well defined primary production pathways beginning with the lead isotopes ^{206}Pb and ^{208}Pb:

^{208}Pb $\xrightarrow{(n,\gamma)}$ ^{209}Pb $\xrightarrow{(8.7\,[3.25\,h])}$ ^{209}Bi $\xrightarrow{(n,\gamma)}$ …

… $\xrightarrow{(n,\gamma)}$ ^{210}Bi $\xrightarrow{(8.7\,[5.01\,d])}$ ^{210}Po,

and ^{206}Pb $\xrightarrow{(n,\gamma)}$ ^{203}Hg.

In addition to ^{208}Pb and ^{206}Pb, the LiPb coolant contains a 43 appm impurity of ^{209}Bi and that is a two-step neutron absorption and subsequent decay process leading to ^{210}Po. A number of longer Pb based pathways also contribute to the ^{203}Hg concentration over the lifetime of the coolant.

Figure 8 shows the inventories of ^{209}Bi and ^{210}Po in the full 600 m^3 volume of coolant. The ^{209}Bi inventory is of interest because, as a precursor to ^{210}Po, control of its concentration can serve as a mechanism to limit the ^{210}Po inventory. It is clear from this figure that the ^{210}Po levels

are above the radiological limit for ^{210}Po (25 Ci) very soon in the operational life of the facility, primarily due to the initial bismuth impurity. A purification system is therefore necessary to remove the ^{210}Po (ref. 6). One could imagine installing a ^{209}Bi purification system rather than a system to remove the radiological hazardous radionuclide ^{210}Po generated by Pb during operation.

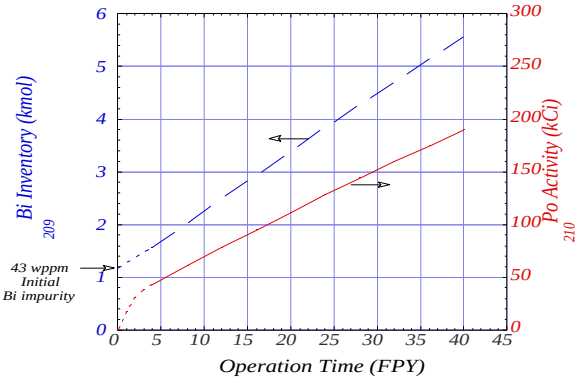

Figure 8. ^{209}Bi and ^{210}Po inventories.

Figure 9 displays the activity of ^{203}Hg over the lifetime of the facility. Note that after a few days the activity is above the radiological limit for ^{203}Hg (25 kCi) and a purification system will be necessary to control its level[6].

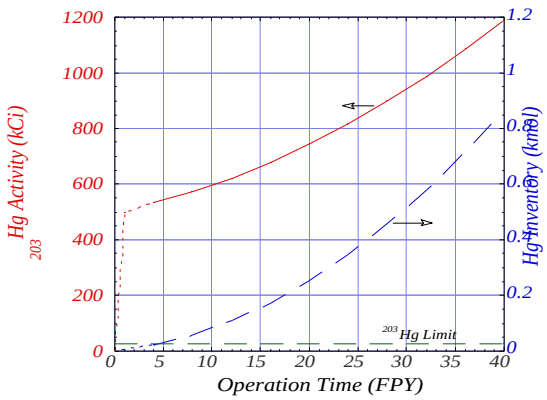

Figure 9: Mercury inventories in LiPb.

V. WASTE MANAGEMENT

The primary options for managing the ARIES waste include the near-surface disposal as Class A or C LLW, recycling and reuse in nuclear facilities, and clearing of materials containing very low radioactivity to mainly reduce the volume of waste. By definition, clearance is the unconditional release of materials from radiologically controlled areas after an interim storage period of 50 or 100 years. Currently, the NRC has not yet defined standards for clearance of solid materials. Due to the absence of official U.S. guidelines, we have temporarily adopted the IAEA nuclide specific clearance limits developed for 1650 radioisotopes by various international organizations.[7]

The WDR's in the U.S. are defined in the 10CFR 61 document that sets specific activity limits for Class A and Class C waste.[8] While both are considered low-level waste, Class C waste requires engineered intruder barriers and a minimum disposal depth of 5 m, while Class A must meet only the minimum packaging requirements[8]. The NRC waste classification is based largely on radionuclides that are important in fission facilities. In the early 90's, Fetter and others performed analyses to determine the Class C specific activity limits for all long-lived radionuclides of interest to fusion using a methodology similar to that used in 10CFR 61 (ref. 9). Although the calculations carry no regulatory acceptance, they are useful because they include fusion-specific isotopes. The ARIES approach requires all components to meet both NRC and Fetter's limits until the NRC develops official guidelines for fusion waste.

Because of the compactness of the machine, all components have clearance indices greater than one, meaning none of the ARIES-AT fusion power core components can be released as cleared solids even after a 100 year storage period.[10] The Fetter and NRC Class C WDR for the blanket and divertor structures of the device are presented in Table 2. All components easily meet the Class C LLW requirements. For the SiC components, the long-lived radionuclide ^{14}C is the main contributor to the NRC Class C WDR and ^{26}Al is the main contributor for the Fetter Class C WDR. Nb and Mo impurities must be controlled below 1 and 20 appm, respectively, in ferritic steel for the IB HT shield to qualify as Class C waste after 40 FPY of operation.

Table 2. Fetter and NRC Waste Disposal Ratings

Class C Limits:	Fetter	NRC
Inboard Components:		
FW/B	0.019	0.017
HT Shield	0.7	0.4
Vacuum Vessel	0.08	0.008
Magnet	0.09	0.07
Outboard Components:		
FW/B-I	0.09	0.03
B-II	0.6	0.4
HT Shield	0.2	0.1
Vacuum Vessel	0.07	0.04
Magnet	0.1	0.09
Divertor Components:		
DP + 0.35 cm W	0.4	0.07
Manifolds	0.006	0.01
Replaceable HT shield	0.2	0.1
HT Shield	0.3	0.2
Vacuum Vessel	0.02	0.008
Magnet	0.04	0.03

If low activation materials were utilized and impurity levels of Nb and Mo were controlled in the ferritic steel structure, 90% of the ARIES-AT waste could qualify for Class A LLW after a 100 year storage period. The remaining 10% of the waste that consists of the OB blanket-II + W, IB HT shield, and divertor HT shield would still fall under the Class C waste category.

VI. CONCLUSIONS

Detailed activation calculations of the ARIES-AT tokamak design are performed to evaluate the decay heat and waste disposal aspects of the device. The initial activity level of the SiC in the FW/Blanket components was found to be higher than the ferritic steel components. However, after a one-hour cooling period the SiC activity level dropped approximately two orders of magnitude below the level of the ferritic steel components. Over the course of a day, the SiC activity level dropped by three orders of magnitude. The initial high decay heat levels of the SiC components are directly related to their initial activity level. After a one-hour decay period, the SiC decay heat level dropped to a level comparable to that of the ferritic steels. Detailed LiPb activity calculations were performed to investigate the LiPb decay heat and radiological hazardous material inventory. It was found that the LiPb decay heat level exceeded that of the SiC components several days after shutdown. This implies that a LOFA accident is more critical than a LOCA accident for LiPb/SiC systems. Due to the compactness of the design, all components have clearance indices greater than one. This means that no metals are released even after a 100 year storage period. The WDR of the major components in the device were computed. It was found that all components could easily meet the Class C requirements after the selection of low activation materials and the control of Nb and Mo impurity levels in ferritic steel. Class A LLW requirements could be met by many components after a 100 year storage period. The recycling option will be assessed in future ARIES studies to increase the repository capacity. The radionuclides ^{210}Po and ^{203}Hg are generated by Pb during operation. In order to reduce the buildup of these radiologically hazardous materials in the LiPb coolant/breeder, a purification system will be required to remove these two radionuclides shortly after operation begins.

ACKNOWLEDGEMENT

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