Safety and Environment Assessment of ARIES-AT

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ABSTRACT

ARIES-AT is a 1000 MWe conceptual fusion power plant design with a very low projected cost of electricity. The design contains many innovative features to improve both the physics and engineering performance of the system. From the safety and environmental perspective, there is greater depth to the overall analysis than in past ARIES studies. For ARIES-AT, the overall spectrum of off-normal events to be examined has been broadened. They include conventional loss of coolant and loss of flow events, an ex-vessel loss of coolant, and in-vessel off-normal events that mobilize in-vessel inventories (e.g. tritium and tokamak dust) and bypass primary confinement such as a Loss of Vacuum and an in-vessel loss of coolant with bypass. This broader examination of accidents improves the robustness of the design from the safety perspective and gives additional confidence that the facility can meet the no-evacuation requirement. We also provide a systematic assessment of the design to address key safety functions such as confinement, decay heat removal, and chemical energy control. In the area of waste management, waste is classified by both the volume of the component and its hazard. In comparison to previous ARIES designs, the overall waste volume is less because of the compact design.

I. BACKGROUND AND OBJECTIVES

The DOE Fusion Safety Standard[1] was developed in 1996 to enumerate the safety requirements and to provide corresponding safety guidance related to the hazards associated with D-T magnetic fusion facilities. Two fusion-specific requirements that were developed are:

- The need for an off-site evacuation plan shall be avoided
- Wastes, especially high-level radioactive wastes, shall be minimized

The ARIES program has adopted these last two requirements for their design studies. These stringent requirements have impacts in terms of materials selection and design decisions, and were adopted to demonstrate the safety and environmental potential of fusion power. The no-evacuation requirement translates into a dose limit of 10 mSv (1 rem) in a worst case accident. The waste minimization requirement is currently interpreted in the ARIES context as requiring that all waste meet current US low-level waste (Class C or better) disposal requirements. In this paper, we evaluate the ARIES-AT design against these requirements.

II. RADIOLOGICAL INVENTORIES AND RELEASE LIMITS

The major radiological inventories in the ARIES-AT design are tritium and activation products in plasma facing components, in structural materials and in the coolant. We present the major inventories and the corresponding release limits to meet the no-evacuation limit.

A. Tritium

Tritium is found in three major locations in the in-vessel components in the ARIES-AT: in the W plasma facing components in the divertor, in the LiPb coolant, and in the SiC structure. Experimental measurements on tritium uptake in tungsten indicated that the amount of tritium in the W armour of the divertor is small. Based on analysis using the TMAP4 computer code, the inventory of implanted tritium in the tungsten is on the order of 0.4 g/m^2. With a divertor area of 67.7 m^2, this surface coverage translates into ~ 27 g of tritium.

In addition, about 1-2 g of tritium are bred in the SiC in the divertor via nuclear reactions. Thus the total inventory in the divertor is ~ 30 g.

Tritium in the LiPb coolant is extremely small (< 1 g) because of the very low solubility of tritium in LiPb. The rather low solubility results in a partial pressure of tritium over the LiPb of about 20 Pa.

Tritium in the first wall and SiC comes from implantation/uptake from the plasma and production via nuclear reactions in the SiC. Analysis of tritium uptake in the plasma chamber indicates ~ 534 g in the first wall and ~ 150 g that could be formed as co-deposited layers in the vacuum vessel due to sputtering of the SiC and subsequent C:H re-deposition in cooler regions of the plasma chamber such as the pumping ducts. In addition, nuclear reactions are expected to generate ~ 7 g of tritium in the SiC structure. The overall inventories are summarized in Table I below.

Table I. Tritium inventories in ARIES-AT

Component	Inventory from plasma interactions	Inventory from nuclear activation	Total Inventory
First wall	534 g	7 - 8 g	~ 540 g
Divertor	27 g	1 - 2 g	~ 30 g
LiPb Coolant	< 1 g	--	< 1 g
Co-deposited Layers	150 g	---	150 g
Total	711 g	8 - 10 g	~ 720 g
Total Mobilizable in an accident			180 g

Of these inventories, the major source of mobilizable tritium would be from the co-deposited layer. The greatest concern would be the levitation of this material and its potential for being carried to the surrounding in the event of a breach of the vacuum vessel confinement boundary. We assume in the analysis presented here that the entire 150 g of co-deposited tritium is mobilized in any event in which the vacuum vessel is breached. In addition to the co-deposited material, the tritium in the tungsten divertor plate will come out at a fairly high rate even if the plates are at ambient temperature. Thus, an additional 30 g is considered to be mobilizable in an accident. The bulk of the tritium in the SiC is tightly held in the SiC matrix under the accidents considered in Section III. The low diffusivity in the SiC results in very little release in accidents. For example, after one year at 750°C only 6% of the tritium is released. Much higher temperatures (in excess of 1400°C) would be required to mobilize this inventory. No accidents have been identified in the ARIES-AT design that could result in a temperature of 1400°C in SiC (see Section III).

B. Activation Products

Neutron activation calculations have been performed for the structures and coolant in ARIES-AT using the ALARA activation code.[2] The lifetime of most components is 40 FPY, except for the FW and divertor which, due to the high neutron wall loading (3-5 MW/m^2), must be replaced after 4 FPY. Calculations assumed 75% availability. Of specific interest to the safety assessment are the Po-210 and Hg-203 in the LiPb coolant and the activated tungsten divertor plate, which could produce tungsten dust via a disruption. Inventories in the coolant have been estimated based on an estimated residence time of the LiPb in each circuit and assuming cleanup of the bismuth impurity to 1 ppm.[3] This results in about 0.1 ppb of Po-210 in the coolant.[i] The results are shown in the Table II. (If there were no cleanup of bismuth in the coolant then the Po-210 inventory would be 190 kCi.) The specific activities of the dose-dominant isotopes in tungsten are shown in Table III.

Table II. Inventories of Po-210 and Hg-203 in LiPb

Isotope	Concentration (Ci/m^3)	Inventory
Hg-203	2000	1.2 MCi
Po-210	4.167	2500 Ci

Table III. Specific activity of dose-dominant isotopes in tungsten

Isotope	Specific Activity (Ci/g)
Ta-182	2.30E-01
W-185	7.89E+00
W-187	3.14E+00
Re-186	8.68E-01
W-181	3.22E+00
Re-184	1.33E-01
Re-188	3.16E-01
Re-184m	2.25E-02
Sc-46	1.14E-03
Sc-48	1.93E-03

C. Allowable Release Limits

Radiological release targets for the major inventories have been established to meet the 10 mSv

[i] Design of a cleanup system to remove Po-210 and Hg from the LiPb coolant is currently under study and could greatly reduce these inventories.

(1 Rem) no-evacuation limit consistent with the requirements of the DOE Fusion Safety Standard.[1] Radiological dose calculations have been performed using the MACCS2 code for radionuclide release from a 100-m stack and at ground level for a 1-km site boundary[4] using average weather conditions (stability class D and wind speed 4 m/s). Based on these calculations, the allowable releases in Table IV were established.[ii] In the case of tungsten dust, the radionuclide inventories of the predominant tungsten, tantalum and rhenium isotopes produced in the tungsten were combined with the allowable release for each isotope to obtain a release target for dust in terms of total mass. Tritium is assumed to be HTO. Ground level release limits are used because all of the accidents identified in Section III result in releases that are not routed up the stack.

Table IV. Radiological Release Targets for ARIES-AT to meet the No-evacuation Objective*

Release Point	Ground	Elevated via 100 m stack
Site Boundary	1 km	1 km
Tritium as HTO	150 g	1.3 kg
Activated W dust	6 kg	69 kg
Hg-203	25,000 Ci	250,000 Ci
Po-210	25 Ci	250 Ci

*No-evacuation dose limit is 10 mSv (1 rem) per off normal event using best-estimate or average weather

By comparing the mobilizable inventories with these limits, we can determine the degree of radiological confinement that we need during accidents to meet the no-evacuation limit. A factor of only 20% confinement is needed in tritium, approximately a factor of 10 for tungsten dust and a factor of 50 to 100 for events in which Hg-203 and Po-210 are mobilized from the LiPb coolant.

III. Assessment of Safety Implementation in Design

A. Introduction

Because of the use of tritium and the presence of activated materials in ARIES-AT, some degree of radiological confinement is needed to protect the public and the workers at the facility. As outlined the DOE Fusion Safety Standard, radiological confinement is implemented as the key safety function in ARIES-AT to ensure that releases during normal operation are kept as low as reasonably achievable and that releases during accidents are below the no-evacuation release limits discussed in Section II.

Double confinement is implemented in ARIES-AT around all of the large inventories of tritium and activation products. For in-vessel inventories, the vacuum vessel and its extensions are the primary confinement, and the cryostat and its extensions are the second confinement boundary. In the heat transfer systems, the coolant piping forms the primary boundary and the vaults or rooms that house the coolant systems form the second boundary.

Demonstration of compliance with the no-evacuation safety requirements[1] requires examination of a broad range accidents that could challenge these radiological confinement boundaries to determine if any could lead to releases in excess of the no-evacuation limits. These accidents fall into three major categories:

- events that directly breach a confinement boundary (e.g. loss of vacuum, overpressure failure of the vacuum vessel, in-vessel loss of coolant with bypass of vacuum vessel),
- events related to decay heat removal (e.g., complete loss of coolant or loss of flow), and
- events associated with the chemical reactivity of materials (e.g., ex-vessel spill of LiPb coolant).

In the following sections, an overview of each of these events is provided, followed by a summary of the detailed analysis of the progression of the accident including the accident's consequences in terms of releases to the environment.

B. Challenges to Radiological Confinement

The major challenges to radiological confinement fall into two main categories: a loss of vacuum event induced by failure of the confinement boundary and in-vessel coolant breaches that result in overpressure and subsequent failure of the confinement boundaries.

1. Loss of Vacuum Event (LOVA). In this event it is assumed that the two confinement barriers surrounding the tokamak (VV and its penetrations and cryostat and its penetrations) fail at some location (most likely at a penetration) allowing air to enter the plasma chamber. The ingress air would instantly extinguish the plasma and cause a density-limit disruption. The disruption, based on ITER estimates,[5]

[ii] More recent DOE rules require that worst case meteorology should be used. This would results in release limits that are ten times more restrictive than in Table 4. The safety analysis results presented later in this paper indicate that ARIES-AT can still meet the no-evacuation limit of 10 mSv under worst case weather.

could mobilize 10 kg of tungsten dust in the divertor with a diameter of 0.1 μm depending on the magnitude of the disruption and the surface area of the divertor affected. The 180 g of easily mobilizable tritium (corresponding to 1.2 kg of HTO) would also become airborne in the plasma chamber. The mobilization time is 100 seconds, which approximately corresponds to the time required for the VV to pressurize during this LOVA.

To analyze this event, a MELCOR[6,7] model of the ARIES-AT design was developed. The model includes: 1) the free volume within the vacuum vessel (VV) and pumping ducts, 2) the in-vessel first wall (FW), high temperature shield (HTS), and divertor plate (DP) components, 3) the VV walls, 4) the confinement building upper functional areas (UFA), the heating-ventilation-air-conditioning (HVAC) system of the UFA, and 5) a duct that connects the VV to the UFA. The bypass duct that connects the VV to the UFA is assumed to have a cross-sectional area of 0.02 m^2 (diameter of 0.16 m), and a length of 10 m. The size of the duct has been selected to simulate a diagnostic or heating port in the VV. Because the UFA are non-nuclear rooms of the confinement building, the HVAC air exchange rate was assumed to be one volume per hour and a leak rate of one volume per day at an overpressure of 400 Pa was assumed for the UFA. These specifications are based on the ITER design of rooms around the tokamak.[iii] The bypass duct was divided vertically into two flow paths to allow for the prediction of a counter-current flow pattern expected to develop within this duct due to natural convection during this event. The HVAC is assumed to turn off after one hour, isolating the UFA, and a VV detritiation system starts that circulates one VV volume per hour. These assumptions are similar to those developed for the ITER EDA safety study.[5]

Figure 1 contains the predicted vacuum vessel pressure during this event. The pressure in the vacuum vessel reaches a near equilibrium with the UFA in about 500 seconds. However, equilibrium between these two volumes does not actually occur until almost 2000 s at which time these flows in the bypass duct become stratified at about 5 g/s. Once the internal components of ARIES-AT cool, the flows in the duct equilibrate at about 2 g/s, which is a flow velocity of about 0.5 m/s. Only about 0.15 g of the 10 kg of dust in the plasma chamber makes it to the UFA, of which 0.04 g is released to the environment before the UFA is isolated. The reason for this low release is that most of the dust in the VV settles to the floor of the VV before outflow from the VV begins. In fact, only 270 g of the initial 10 kg remains airborne by 2000 seconds. In addition to the dust release, an estimated 0.06 g of HTO is stacked by way of the HVAC system during the first hour of the event, with an another 0.02 g leaked from the UFA over the course of a day.

There is some uncertainty regarding the initial inventory of dust in the VV, which could be as high as 110 kg based on detailed analysis for ITER. However, because this release scales almost linearly with initial mass mobilized, it is expected that the release from mobilizing 110 kg of tungsten dust would be small as well. A summary of the source term behavior for this event is found in Table V.

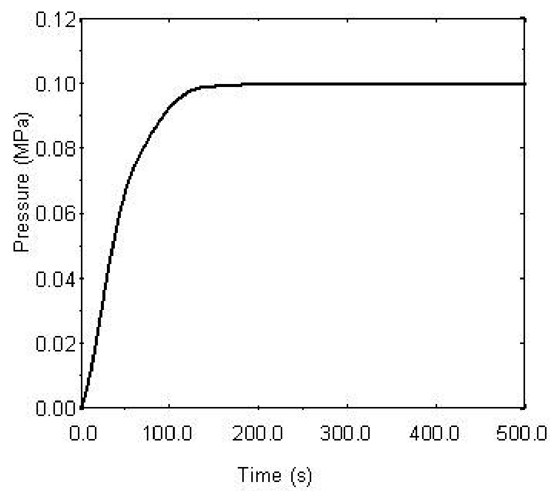

Figure 1. ARIES-AT vacuum vessel pressure during a LOVA.

Table V. Source Term Behavior for ARIES-AT LOVA

Material	Mobilizable Inventory	Released to the UFA	Released to the Environment
Tritium	180 g	0.35 g	0.08 g
Dust	10 kg	0.15 g	0.04 g

2. In-vessel LOCA with Bypass. Since the LiPb blanket coolant is not significantly pressurized and will not change phase upon pipe breach, an in-vessel blanket LOCA does not generate any significant pressure. However, the low temperature shield/vacuum vessel is cooled with pressurized water. A suppression system with a passive rupture disk will

[iii] This leak rate is less than the confinement buildings of some DOE reactors but much higher than fission reactor containment buildings.

be needed to relieve the pressure in the plasma chamber in the event of a LOCA of this system.

In this event, it is assumed that there is a small in-vessel LOCA associated with failure of the water-cooled low temperature shield/VV. The size of the break is approximately 4.5 cm in diameter. The in-vessel LOCA is assumed to cause a plasma disruption and terminate the plasma. The pressurized water from the break then sprays onto the back of the LiPb/SiC blanket where it enters film boiling. The steam that is generated is superheated because of the high temperatures of the in-vessel components and thus rapidly pressurizes the vacuum vessel. The pressure suppression system is assumed to open at a pressure of 0.2 MPa (2 atm) in the vacuum vessel. The impingement of water from the low temperature shield onto the back of high temperature shield will probably cause the high temperature shield to fail because of thermal stress or cracking of the SiC or the formation of SiO_2. LiPb would be released from the failed blanket sector and puddle on the floor of the vacuum vessel. The pool will fill up half of the plasma chamber to a depth of ~ 3.25 m. Because all of the in-vessel surfaces of ARIES-AT are above the critical temperature of water, any water released from the break will flash to steam and become superheated. Thus, no interaction between water and the LiPb pool is expected. However, the LiPb pool is expected to release any Po-210 and Hg as it cools at the bottom of the vacuum vessel. (Details of the release model are presented in Section III.D)

Figure 2 is a plot of pressures in the VV/Shield cooling system, the plasma chamber and the UFA as a function of time. The pressure in the VV/shield cooling system decreases as a result of the break. The water entering the vacuum vessel is heated by impingement onto the hot high temperature shield and the pressure in the plasma chamber increases. At ~ 10 s, the pressure suppression system opens when the pressure in the plasma chamber exceeds 0.2 MPa. Although the pressure suppression system opens at 0.2 MPa, the peak pressure in the vessel reaches ~ 0.24 MPa. This high pressure is assumed to cause a failure in the duct that connects the VV to the upper functional area (UFA). Thus, this in-vessel LOCA is assumed to cause a bypass of the confinement. The size of the bypass duct has been selected to simulate a diagnostic or heating port in the VV, similar to that in the LOVA analysis. Ventiliation in the UFA is the same as that used in the LOVA analysis. The source term for this event includes the 180 g of mobilizable tritium in the vessel, 10 kg of tungsten dust generated from the disruption, and the release of Po-210 and Hg from the pool.

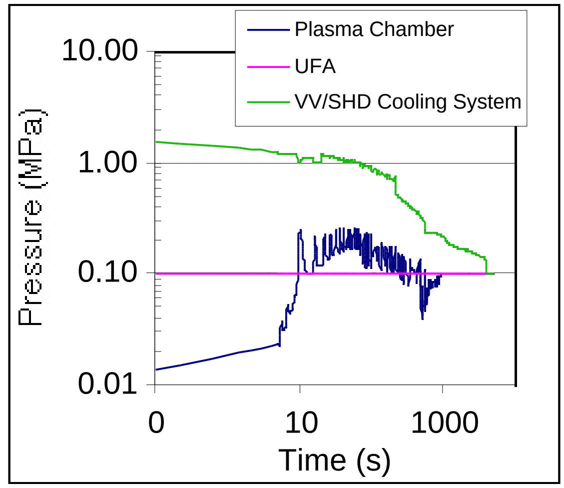

Figure 2. Pressure response to in-vessel LOCA with bypass

The overall releases in the event are shown in Table VI. The pressure suppression system is quite effective at capturing most of the material that is mobilized in the plasma chamber. Thus, the overall releases are small and well below the no-evacuation limits.

Table VI. Source Term Behavior for In-vessel LOCA with bypass

Material	Mobilizable Inventory	Released to the Environment
Tritium	180 g-T	6.7 g-T
Dust	10 kg	207 g
Po-210	500 Ci	0.021 Ci
Hg-203	250000 Ci	88.5 Ci

C. Decay Heat Removal

A series of complete loss of coolant accidents (LOCA) and total loss of flow accidents (LOFA) were analyzed for ARIES-AT to examine the ability to passively remove decay heat from the machine.[8] The activation calculations indicate that within one hour after shutdown the activity of the SiC structure drops by several order of magnitudes below the activity of the steel-based shielding components. The higher initial activity of the highly irradiated SiC components translates directly into a higher initial decay heat for SiC. However, within an hour, the SiC decay heat drops by two orders of magnitude to levels comparable to that of the well-protected steel-based components.

A realistic pulsed operation analysis was also used to investigate the decay heat of the LiPb as a function of both the time spent in the fusion power core (1-3 minutes) and the time spent in the outer loop (2 minutes). The LiPb decay heat drops by two orders of magnitude at one minute after shutdown and remains almost flat for several days, exceeding that of the SiC structure. This means LOFA is more critical than LOCA for LiPb/SiC systems. Of all possible LOCA and LOFA events, a combined LOFA in the blanket and a LOCA in the shield resulted in the most severe event.

Thus, we calculated the detailed thermal response of each component during the LOCA/LOFA using transient axi-symmetric 2-D finite element model. We assume that the plasma is immediately quenched at the onset of the LOCA/LOFA and then the temperature of the in-vessel components begins to rise due to the decay heat. The results are shown in Figure 3.

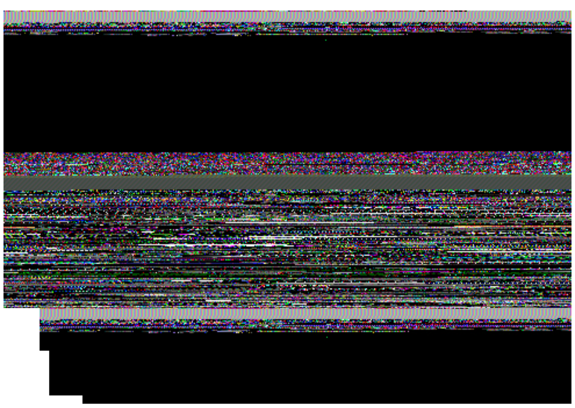

Figure 3. Inboard Temperature History of Key Components

The water-cooled vacuum vessel (VV) operates at a much lower temperature (50-100°C) compared to the breeding blanket (700-1100°C). Consequently, the ferritic steel-based VV acts as a heat sink for the high temperature SiC-based components during LOCA/LOFA events. We conservatively assumed an adiabatic condition at the innermost surface of the IB VV. On the other hand, the OB VV is allowed to radiate the heat to the maintenance ports located on the OB side. Among several accident scenarios, a combination of loss of LiPb flow and loss of water coolant was judged the most severe LOCA/LOFA event. Our results indicate that the highest temperature occurs at the W-coated SiC divertor plates, reaching 1050°C at 2-3 hours following an accident. Due to the inherent rapid drop of decay heat, the temperature of the SiC components drops quickly to 500-600°C within several hours. The peak VV temperature occurs at the IB mid-plane, reaching 686°C at ~2.4 days following the accident. Those temperatures are within the limits for the SiC and FS structures, meaning there is no need for a dedicated separate decay heat removal system for the ARIES-AT design.

D. Chemical Reactivity

The key chemical reactivity issue with the ARIES-AT design is reaction of the blanket coolant with air and water. In terms of LiPb/water interactions, the cooling systems are separate and no credible ex-vessel interactions between LiPb and water have been identified.

The Li in the LiPb is buffered to a large degree by the large heat capacity of the Pb. As a result, reaction with air is not a serious issue in terms of chemical energy production or combustible gas generation. At temperatures up to 900°C, no violent reaction has been observed in experiments.[9,10] The key concern is the release of Po-210 and Hg-203 from the LiPb during a spill. As a result, an ex-vessel spill of LiPb coolant in the room housing the coolant manifold was examined to determine the potential for Po-210 and Hg-203 release.[iv]

A version of the MELCOR code that simulates coolants other than water, in this case LiPb, was used for this ex-vessel LOCA calculation.[11] A model that represents one quadrant of ARIES-AT was developed, and as such the total LiPb inventory of 150 m^3.

The selected location for the ex-vessel LOCA to occur was the lower functional areas (LFA). This region of the confinement building is the lowest portion of the building adjacent to the reactor vault, and as such should be the location where the largest spill of coolant for the ARIES-AT design would occur. It was assumed for this event that the LFA was divided into quadrants. An instantaneous double-ended-off-set

[iv] Releases of Po-210 and Hg-203 can be mitigated in ARIES-AT through the use of a drain tank system. The drain tank system (one system for each loop) is foreseen for maintenance activities in which the loop must be drained. However, the system could also be used in a spill event if needed to minimize the overall amount of material spilled and the potential for releases to the environment. In the analysis presented here, the effect of the drain tank is not considered to be conservative.

shear of an outlet coolant pipe at the location it enters the LFA from the reactor vault has been assumed to be the initiator of this accident. The PHTS pump coast down is assumed to occur in 30 seconds. Because the LFA is expected to be a non-nuclear area, the operation of the HVAC system for this area during the accident is the same as that described for the UFA in Section III.B.

As result of the spill, a 0.238 m deep pool, (with a surface area of 525 m^2 and a total volume of 125 m^3), forms on the floor of the LFA. Figure 4 shows the temperature evolution of the pool during the course of this accident.

The peak temperature is 980°C, down from the reactor outlet temperature of 1125°C due to contact cooling of the LiPb with the LFA floor and walls. This figure also shows the concrete floor of the LFA reaching a maximum temperature of 580°C at 500 seconds. The pool continues to cool until the triple point temperature of LiPb is reached nearly nine hours after the spill occurs. The cooling of the pool is by conduction through the floor and outer wall of the LFA into the ground. It takes two additional days for this pool to completely solidify given these heat transfer assumptions.

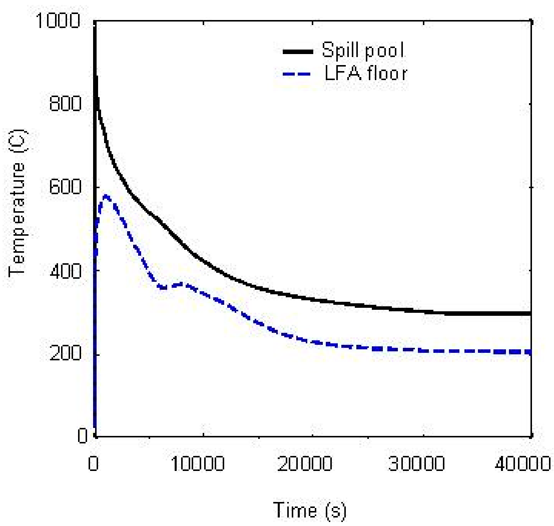

Figure 4. Coolant spill temperatures during an ex-vessel LOCA in ARIES-AT

For a spill of 150 m^3 of LiPb, 500 Ci of Po-210 and 2.5×10^5 Ci of Hg-203 will be in the pool. Two key processes are involved in the release to the room atmosphere: diffusion from the bulk pool to the pool surface and vaporization off of the pool surface. Diffusion to the pool surface is simply given by:[12]

$$FR = 1 - \frac{8}{\pi^2}\sum_{n=0}^{\infty}\frac{1}{(2n+1)^2}\exp\left[-\frac{D(2n+1)^2\pi^2}{4L^2}\right] \qquad (1)$$

where

FR = fraction of initial inventory of species that reaches the surface of the pool

D = diffusion coefficient (m^2/s)

L = depth of pool (m).

The diffusion coefficient in the liquid metal can be estimated using the Scheibel modification of the Wilke-Change correlation.[13] It relates diffusivity, D, in cm^2/s to the viscosity of the liquid in Poise (μ), the temperature of the pool in K, and the molar volumes of the species (V_a) and the liquid (V_m) in cm^3/gmole:

$$D(cm^2/s) = 8.2E\text{-}10\ [1 + (3V_a/V_m)2/3]\ T/\ [\mu/V_m)^{1/3}] \quad (2)$$

The depth of the pool is ~ 24 cm. For Hg and Po, the values of D range from 1.5×10^{-5} cm^2/s at 200°C to 1.5×10^{-4} cm^2/s at 1000°C.

Once at the surface the Hg and Po behave quite differently. The Hg will be well above its boiling point of 360°C for most of the transient and thus we assume it completely vaporizes once it reaches the surface. For Po, the vaporization is based on an assessment of release data from laboratory experiments. The data are from experiments in Russia, Germany and the US conducted over the past two decades.[14] The measured release from the experiments is part aerosol and part vapor. The aerosol is condensed PbPo and the vapor is an hydroxide of Po based on RF work. In air some PoO_2 is expected but it decomposes above 500°C. The experimental results also indicate that more aerosol and less vapor are produced at higher temperatures compared to lower temperatures. Thus, we assume that the release of PbPo is aerosol and we use the vaporization rate recommended by Schipakin:

$$J\ (Ci/cm^2\text{-}hr) = 5.85 \times 10^{-6}\ P_{sat}(T)\ (1000/T)^{0.5}\ [x/x_o] \quad (3)$$

where $P_{sat}(T)$ is the saturation vapor pressure for PbPo in mm of Hg and the ratio $[x/x_o]$ is a linear correction factor equal to 4.7 to account for the fact that the Po concentration in the LiPb in ARIES-AT is 4.7 times higher than the 1.95×10^{-11} mole fraction in Shipakin's experiments. This rate is based on small-scale experiments where there is no limit to transport of PbPo to the surface of the melt. As discussed previously, in the pool there is a significant mass transport resistance to the surface. Thus, for the PbPo, the mass flux of the PbPo arriving at the surface (derivative of Equation 1 converted to mass flux) is compared to the rate of vaporization from the surface. The mobilization rate is assumed to be the minimum of these two rates at a given time step in the calculation.

With these models, about 17.4 Ci of Po-210 is mobilized, and 11.7 Ci released to the environment prior to isolating the LFA (shutting down the HVAC system) after one hour. In one hour, 8870 Ci of Hg-203 are mobilized and nearly 5770 Ci are stacked to the environment. Of this release, 260 Ci is predicted to leak through the walls of the LFA to the environment before the pool completely solidifies. The quantity of PbPo leaked was negligible. These results are well below the no-evacuation limits. Table VII summarizes the results of the source term analysis.

Table VII. Source Term Analysis for Ex-vessel Spill of LiPb

Isotope	Inventory in Spilled LiPb (Ci)	Amount Mobilized (Ci)	Released to the Environment (Ci)
Hg-203	250000	8770	5770
Po-210	500	17.4	11.7

IV. WASTE MANAGEMENT

Based on prudent selection of materials in the ARIES-AT design, all components can easily meet the Class C low level waste (LLW) requirement established for the ARIES power plants (see Table VIII). Some components could even qualify as Class A LLW after a 100 y storage period. The Class A/C LLW was achieved by selecting low activation materials for the individual components, controlling the Nb and Mo impurities for ferritic steel to less than 1 and 20 wppm, respectively, and installing a LiPb purification system to remove the Bi-208 generated by Pb during operation.[2]

Due to the lack of official U.S. guidelines for cleared metals, we also applied the IAEA clearance criterion to ARIES-AT.[15] The IAEA clearance limits developed for about 1,700 radioisotopes were used to evaluate the in-vessel and ex-vessel components of ARIES-AT. The results indicate that, because of the compactness of the design, none of the ARIES-AT components can be cleared at the end of a 100-year interim storage period.

An attempt was made to clear the outboard vacuum vessel and magnet by thickening the outboard shield. The net effect of incorporating the additional shielding and removing the vacuum vessel and magnet from the waste stream is a 15% reduction in the total volume of waste generated by the ARIES-AT design. However, the thickening of these outboard components increased the overall radial build and hence cost of electricity from ARIES-AT. Given that the extremely compact radial build of the design does reduce the overall volume of waste compared to other tokamak designs, no additional optimizations were made to try to specifically clear any of the components in the design.

Table VIII. Waste Disposal Results for ARIES-AT Components

	Fetter Class C Limits	NRC Class C Limits
Inboard Components		
FW/B	0.019	0.017
HT Shield	0.73	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		
Divertor plates	0.4	0.07
Manifold, Shield, VV, Magnet	<<1	<< 1

V. SUMMARY AND CONCLUSIONS

ARIES-AT is a 1000 MWe conceptual fusion power plant design with a low projected cost of electricity. The design contains many innovative features to improve both the physics and engineering performance of the system. From the safety and environmental perspective, we have performed a more robust safety analysis than in past ARIES studies. The results indicate that in all cases, the ARIES-AT design can meet the no-evacuation limit. In the area of waste management, waste is classified by both the volume of the component and hazard. In comparison to previous ARIES designs, the overall waste volume is less because of the compact design.

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