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ABSTRACT

Activation and safety analyses were performed for the ARIES-ST design. The ARIES-ST power plant includes a water cooled copper center post and uses a SiC/LiPb blanket. The first wall and shield are made of low activation ferritic steel and cooled with helium. The center post, first wall, inboard shield and blanket were assumed to survive for 2.6 full power years (FPY). On the other hand, the outboard shield and vacuum vessel were assumed to stay in place for 40 FPY. Neutron transmutation of copper resulted in the production of several nickel, cobalt and zinc isotopes. The production of these isotopes resulted an increase of the time-space average resistivity of the center post by about 6% after 2.6 FPY. All of the plant components met the limits for disposal as Class C low level waste (LLW). The off-site doses produced at the onset of an accident are caused by the mobilization of the radioactive inventory present in the plant. Analysis of a Loss of Coolant Accident (LOCA) indicated that the first wall and shield would reach a maximum temperature of less than 700°C during the accident. The calculated temperature profiles and available oxidation-driven volatility experimental data were used to calculate the dose at the site boundary under conservative release conditions. The current design produces an effective whole body early dose of 1.77 mSv at the site boundary.

I. INTRODUCTION

This paper reports work in progress in which a detailed activation and safety analyses were performed for the latest design[1] of the ARIES-ST power plant. ARIES-ST is a low-aspect-ratio spherical tokamak power plant which is one of several fusion power plant designs being assessed within the ARIES project. The ARIES-ST power plant produces a 1000 MW of net electric power and is assumed to operate for 40 full power years. The plant includes a water cooled, DS GlidCop Al15 copper center post (CP) and uses a SiC/LiPb blanket. The first wall and shield are made of low activation ferritic steel (9Cr-2WVTa) and cooled with helium. Activation analysis was performed assuming average neutron wall loadings of 3 and 5 MW/m^2 for the inboard and outboard sides, respectively. The high neutron wall loadings are the limiting factors for the lifetime of the different components of the power plant. The center post, first wall, inboard shield and blanket were assumed to survive for 2.6 FPY. The outboard shield and vacuum vessel were assumed to stay in place for 40 FPY due to the fact that they will be exposed to a lower neutron environment.

A major goal of the ARIES-ST design has been achieving the highest level of safety while maintaining its economic attractiveness. Taking this into account, the design aimed at achieving the following goals:

1. Minimizing the increase in the center post resistivity.
2. Disposal of the plant structure as low level waste.
3. Significantly reducing the off-site doses during Design Basis Accidents (DBA).

In this paper, it is shown that the first goal could be achieved by adequate shielding of the center post. The other goals could be achieved by using low activation materials in the first wall and shield of the plant. The 9Cr-2WVTa ferritic steel was selected because it produces a low level of long-term radioactivity and acceptable levels of short and intermediate-term radioactivity. The disposal of the structure as low level waste is dependent on producing low levels of long-term radioactivity. On the other hand, off-site doses during an accident are dominated by nuclides with short and intermediate lifetimes. In addition, nuclides with intermediate lifetimes are the major contributors to the decay heat and hence, the temporal variation of the structure temperature during an accident.

II. CALCULATIONAL PROCEDURE

The neutron flux used for the activation calculations was generated by the one-dimensional discrete ordinates neutron transport code ONEDANT.[2] The plant structure calculations used toroidal cylindrical geometry models with the inboard and outboard sides modeled simultaneously. The average neutron wall loadings on the inboard and outboard sides are 3 and 5 MW/m^2, respectively. The activation analysis was performed using

the latest version of the activation code DKR-PULSAR2.0.[3] The code combined the neutron flux with the FENDL/A-2.0 data library[4] to calculate the activity and decay heat generated in the different regions of the plant. The plant was assumed to operate continuously for 40 FPY. The center post, first wall, inboard shield and blanket were assumed to survive for 2.6 FPY. The outboard shield and vacuum vessel were assumed to stay in place for 40 FPY. The structure activation results were utilized in a radwaste classification. The decay heat results were used in a Loss of Coolant Accident analysis.[5] The structure and the $Li_{17}Pb_{83}$ breeder activation results were used in the off-site dose calculations following the LOCA. The materials used in the different regions of the plant are presented in Table I.

Table I. Materials Used in the Analysis

Center Post	85% Cu, 15% water
Inboard Shield	80% steel, 20% He
Inboard First Wall	40% steel, 60% He
Outboard First Wall	40% steel, 60% He
Outboard Blanket	6% steel, 6% He, 12% SiC, 76% $Li_{17}Pb_{83}$
He Manifold	30% steel, 70% He
Outboard Low Temperature Shield	15% steel, 25% borated steel, 60% water

III. CHANGE IN CENTER POST RESISTIVITY

Interactions between high energy neutrons and the copper CP lead to the production of several nickel, cobalt, and zinc isotopes as transmutation products. Production of these isotopes leads to an increase in the center post electrical resistivity. The center post resistivity increases linearly with the increase in time of operation. Increase in the center post resistivity would lead to an increase in the recirculating power and lower net efficiency. The following are the most important reactions:

- $^{63}Cu\ (n,2n)^{62}Cu(\beta^{+})^{62}Ni$, $^{65}Cu\ (n,2n)^{64}Cu(\beta^{+})^{64}Ni$, and $^{63}Cu\ (n,\gamma)^{64}Cu(\beta^{+})^{64}Ni$
- $^{63}Cu\ (n,\gamma)^{64}Cu(\beta^{-})^{64}Zn$, $^{65}Cu\ (n,2n)^{64}Cu(\beta^{-})^{64}Zn$, and $^{65}Cu\ (n,\gamma)^{66}Cu(\beta^{-})^{66}Zn$
- $^{63}Cu\ (n,\gamma)^{64}Cu(n,a)^{60}Co$, and $^{65}Cu\ (n,2n)^{64}Cu(n,\gamma)^{60}Co$

Figure 1 shows the radial distribution of the percentage increase in the center post resistivity. As shown in the figure, the outermost 30 cm of the 80 cm-thick center post exhibits the bulk of transmutation. This is due to the fact that the change in the copper transmutation is mostly due to the production of the ^{64}Ni isotope. As already shown, ^{64}Ni is mostly produced via high energy threshold reactions. The electric current will redistribute within the center post to avoid the region with high resistivity. The space-time average increase in resistivity over the entire center post is about 6% which is considered as a tolerable value.

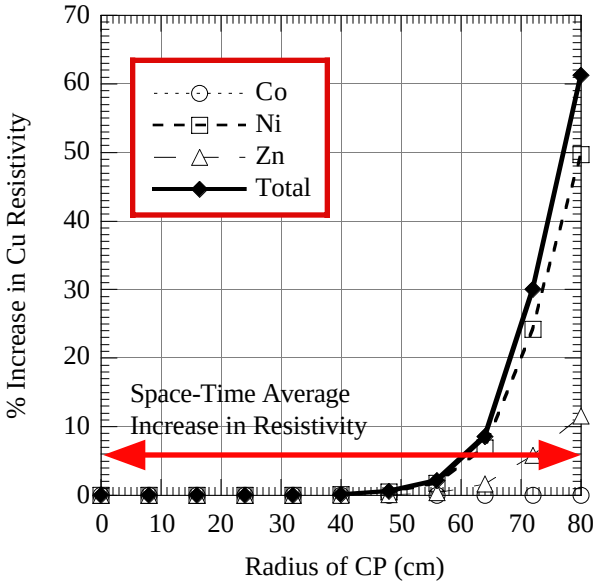

Fig. 1. Radial distribution of the percentage increase in the center post resistivity.

IV. RADWASTE CLASSIFICATION

The radwaste of the different regions of the plant were evaluated according to both the NRC 10CFR61[6] and Fetter[7] waste disposal concentration limits (WDL). The 10CFR61 regulations assume that the waste disposal site will remain under administrative control for 100 years. The dose at the site to an inadvertent intruder at the end of the 100 year period is limited to less than 5 mSv/year. The waste disposal rating (WDR) is defined as the sum of the ratio of the concentration of a particular isotope to the maximum allowed concentration of that isotope taken over all isotopes and for the particular class. If the WDR is ≤ 1 when Class C WDL are used, the waste is termed Class C intruder waste. It must be packaged and buried such that it will not pose a hazard to an inadvertent intruder after the 100 year institutional control period is over. Class C waste is assumed to be stable for 500 years. Using Class C limits, a WDR > 1 implies that the radwaste does not qualify for shallow land burial. Fetter developed a modified version of the NRC's intruder model to calculate waste disposal limits for a wider range of long-lived radionuclides than the few that currently exist in the current 10CFR61 regulations.

Specific activities calculated by the DKR-PULSAR2.0 code were used to calculate the waste disposal ratings. The waste disposal ratings for the

10CFR61 and Fetter limits are shown in Tables II and III. Results in the tables are given for compacted wastes. Compacted waste corresponds to crushing the solid waste before disposal and thus disallowing artificial dilution of activity. The Class C WDR values were calculated after a one year cooling period. As shown in Table II, according to the 10CFR61 limits, the center post WDR is dominated by ^{63}Ni, which is produced via the ^{63}Cu (n,p) reaction. Since this reaction is a high energy threshold reaction, the amount of ^{63}Ni generated in the CP could be reduced further by providing extra shielding on the inboard side. On the other hand, ^{94}Nb, produced from the 0.5 wppm niobium impurities in the 9Cr-2WVTa steel, is the dominant source of waste hazard. As shown in Table III, ^{108m}Ag produced from the 20 wppm silver impurities contained in the GlidCop Al15 copper alloy, is the major waste hazard in the CP according to Fetter limits. In addition to ^{94}Nb, ^{192m}Ir is the other waste hazard associated with ferritic steel. These results show that the waste classification of the CP is controlled by its 10CFR61 WDR as it is entirely due to direct transmutation of copper rather than impurities included in the Cu alloy. All other WDR could be limited by controlling the level of impurities in the copper and steel alloys regardless of the waste disposal limits used.

Table II. WDR Using 10CFR61 Limits

Zone	FPY	WDR	Dominant Nuclides
CP	2.86	0.83	^{63}Ni
i/b Shield	2.86	0.097	^{94}Nb
i/b FW	2.86	0.1	^{94}Nb
o/b FW	2.86	0.1	^{94}Nb
o/b Blanket	2.86	0.011	^{94}Nb
o/b Manifold	2.86	1.4×10^{-3}	^{94}Nb
o/b Shield	40	1.3×10^{-3}	^{94}Nb

Table III. WDR Using Fetter Limits

Zone	FPY	WDR	Dominant Nuclides
CP	2.86	0.44	^{108m}Ag
i/b Shield	2.86	0.44	^{192m}Ir, ^{94}Nb
i/b FW	2.86	0.29	^{192m}Ir, ^{94}Nb
o/b FW	2.86	0.28	^{192m}Ir, ^{94}Nb
o/b Blanket	2.86	0.025	^{192m}Ir, ^{94}Nb
o/b Manifold	2.86	7.2×10^{-3}	^{192m}Ir
o/b Shield	40	0.03	^{192m}Ir

V. HAZARD ASSESSMENT

A strong emphasis was given to the environmental and safety issues in the ARIES-ST design. Low activation ferritic steel (9Cr-2WVTa) was used in the first wall and shield to avoid generating high levels of induced radioactivity. Similarly, the use of LiPb as a breeder eliminates the hazard posed by the energy producing chemical reactions usually associated with the use of lithium and hence reduces the risk of mobilizing the radioactive inventory present in the plant. To evaluate the possible radiological hazard to the public, a two step approach was used in calculating the possible off-site dose. The first step in the approach is the identification of the sources and locations of the radioactive inventories inside the plant. However, since the existence of radioactivity does not in itself represent a safety hazard, the second step in the approach was to consider a pessimistic but rather credible accident scenario for mobilizing and releasing the radioactive inventory. The methodology used in this safety analysis depends on the probability of accident initiating scenarios. The analysis assumed a total loss of coolant accident as the worst case credible accident. This LOCA is also considered as a design basis accident where heat from all in-vessel components is transported to the massive copper busbars.[5] Such an accident is expected to have a probability of about one in a million years.

A. Structure Hazard

During a LOCA, a large increase in the structure temperature could lead to the mobilization and partial release of the radioactive inventory. In calculating the release fraction, only release caused by oxidation-driven volatility was considered. Analyses of other pathways like corrosion products and tokamak dust are still in progress. The decay heat generated during the first day following a LOCA would on average increase the structure temperature by < 700°C. Under these conditions, the full mobilization of the structure radioactive products is impossible. The highest temperature the structure would reach determines the release fraction of its radioactive products. Most of structure radioactivity is generated in its steel component. Off-site dose calculations were performed using ferritic steel experimental volatility rates.[8] HT-9 volatility rates at 700°C in dry air were used in this analysis. To estimate conservative release fractions, a 24-hour LOCA was assumed. One hour release rates were used for the full 24 hours to account for any possible loss of iron oxide protection.

B. Breeder Hazard

The two sources of radiological hazard in a LiPb blanket are tritium and the activation products of LiPb. The steady

state tritium inventory in LiPb is kept very low, in the order of 10 g, by its continuous removal during the plant operation. The activation products of major radiological hazard in a LiPb blanket are the two isotopes, ^{203}Hg ($T_{1/2}$ = 46.61 d) and ^{210}Po ($T_{1/2}$ = 138.38 d). Both ^{203}Hg (γ and β emitter) and ^{210}Po (α emitter), are highly volatile materials. ^{203}Hg produces a high prompt bone marrow dose and ^{210}Po results in high values of prompt, early as well as chronic doses. ^{210}Po is produced via nuclear transmutation of bismuth and is considered as the main safety hazard in LiPb blankets. Bismuth is a major impurity of commercial lead and is also produced as a transmutation product of Pb. Commercial Pb contains 500-1500 wppm of Bi and high-purity Pb contains less than 10 wppm of Bi. The LiPb used in this analysis contains 43 wppm of Bi impurities. It is desirable to keep the Bi impurity in lead below 10 wppm. The amount of Po generated can be controlled by limiting the Bi impurities initially present in Pb as well as the on-line continual removal of Bi atoms produced by neutron-lead reactions. Fortunately, Po evaporates into the form of an intermetallic compound PbPo, whose evaporation rates are very small because of the low vapor pressure of this polonide.[9] Similarly, Hg evaporates into the form of an intermetallic compound LiHg, whose evaporation rates are orders of magnitude lower than Hg. It is estimated that Po retention in a $Li_{17}Pb_{83}$ melt is in the range of 96.4% to 99.2%.[10] In addition, under accidental spill conditions, the dilution of Po is such that α and γ radiation will be shielded by the large amount of lead atoms surrounding Po atoms.

A major advantage of using LiPb as a blanket is its low chemical reactivity. During an accident, a leak of water into the LiPb region will result in a chemical reaction between water and the Li in the molten LiPb. The reaction potential is much smaller than a water/liquid Li reaction. A LiPb/water reaction tends to be self-limiting due to the fact that the liquid metal is formed by 83% Pb which does not react with water and which after initial depletion of Li, tends to shield the remaining amount of alloy from further interaction with water.[9] In addition, solid products Li_2O and LiOH are produced and provide shielding for the remaining liquid metal from the rest of the water. The LiPb/water reaction is an exothermic reaction which leads to an increase in temperature on the order of 200-400°C. A complete reaction between water and LiPb would lead to the production of 55.6 mole of H_2 per kg of water. However, because the Li oxidation is the source of H_2 production, no oxygen is present and therefore explosion cannot occur.[11]

C. Off-Site Dose Calculations

The radioactive inventory calculated by the DKR-PULSAR2.0 code was used as an input to the MACCS2 code[12] to calculate effective whole body off-site dose inventory (dose caused by 100% release of radioactivity) under worst release conditions. These conditions are: ground release, atmospheric stability class F, 1 km site boundary and 1 m/s wind speed. Doses calculated are produced through all of the following pathways:

- Inhalation of radionuclides during plume passage.
- Inhalation of resuspended radionuclides.
- External exposure to the plume.
- External exposure from ground deposition.
- Cloudshine or groundshine.
- Ingestion of contaminated food.

Table IV. Early Doses Released during a DBA

Zone	Inventory (Sv)	Released (mSv)	Dominant Nuclides
CP	2,668	----	^{64}Cu
i/b Shield	4,175	0.17	^{60}Co, ^{54}Mn, ^{56}Mn
i/b FW	306	0.02	^{54}Mn, ^{56}Mn, ^{60}Co
O/b FW	5,000	0.2	^{54}Mn, ^{56}Mn, ^{60}Co
Blanket	2,257	1.38	^{210}Po, ^{203}Hg, ^{54}Mn
Manifold	167	0.004	^{60}Co
o/b Shield	53	0.0008	^{60}Co
Total	14,626	1.77	^{210}Po, ^{203}Hg, ^{54}Mn

The off-site doses were calculated by combining the total off-site dose inventory with the ferritic steel volatility data under the LOCA condition discussed previously. Since no volatility data are available for LiPb, very conservative release rates were adopted for the release of ^{3}H, ^{203}Hg and ^{210}Po. 100% of the ^{3}H, 30% of the ^{203}Hg and 10% of the ^{210}Po were assumed to mobilize during an accident. Air ingress into the coolant channel results in the volatilization of in-vessel materials as previously discussed. Once airborne, these particles could be transported to the site boundary. Assuming that the vacuum vessel and the containment would stay intact during accidents, they would be expected to act as release barriers. For a vacuum vessel leak rate of 1% per day, a containment factor of 99% could be considered. Considering the vacuum and containment boundaries as two independent barriers leads to an overall radioactivity containment factor of 99.99%.[13] As shown in Table IV, the current design produces an effective whole body early dose of 1.77 mSv at the site boundary.

VI. SUMMARY

Detailed activation and safety analyses were performed for the ARIES-ST spherical tokamak power plant. ARIES-ST includes a water cooled copper center post and uses a SiC/LiPb blanket. The plant is assumed to operate for 40 full power years. The center post, first wall, inboard shield and blanket were assumed to survive for 2.6 FPY. The outboard shield and vacuum vessel were assumed to stay in place for 40 FPY. Neutron irradiation resulted in the increase of the center post copper resistivity due to the production of neutron-induced transmutation. Neutron transmutation of copper resulted in the production of several nickel, cobalt and zinc isotopes. The production of these isotopes resulted in an increase of the time-space average resistivity of the center post by as much as 6% after 2.6 FPY. Waste disposal limits were calculated for the different plant components using the NRC 10CFR61 and Fetter waste disposal limits. All of the plant components met the limits for disposal as Class C low level waste. Analysis of a Loss of Coolant Accident indicated that the structure would reach a maximum temperature of less than 700°C during the accident. The calculated temperature profiles and available oxidation-driven volatility experimental data were used to calculate doses at the site boundary under conservative release conditions. The vacuum vessel and the containment were assumed to stay intact during accidents and hence act as release barriers. A leak rate of 1% per day and a containment factor of 99% were considered. The current design produces an effective whole body early dose of 1.77 mSv at the site boundary.

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