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**ACTIVATION PRODUCT SAFETY
IN THE ARIES-I REACTOR DESIGN**

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Abstract

The ARIES design effort has sought to maximize the environmental and safety advantages of fusion through careful selection of materials and careful design. Three goals are that the reactor achieve inherent or passive safety, that no public evacuation plan be necessary and that the waste be disposable as 10CFR61 Class C waste. The ARIES-I reactor consists of a SiC composite structure for the first wall and blanket, cooled by 10 MPa He.

The breeder is Li_2ZrO_3 , although Li_2O and Li_4SiO_4 were also considered. The divertor consists of SiC composite tubes coated with 2 mm of tungsten. Due to the minimal afterheat of this blanket design, LOCA calculations indicate maximum temperatures will not cause damage if the plasma is promptly extinguished.

Two primary safety issues are the zirconium in the breeder and tungsten on the divertor. Li_2ZrO_3 was chosen because of its demonstrated high-temperature stability. The other breeders have lower afterheat and activation. Use of zirconium in the breeder will necessitate isotopic tailoring to remove ^{90}Zr and ^{94}Zr . The 5.8 tonnes of W on the divertor would also have to be tailored to remove ^{186}W and/or to concentrate ^{183}W .

Thus the ARIES-I design achieves the passive safety and low-level waste disposal criteria with respect to activation products. Development of low activation materials to replace zirconium and tungsten is needed to avoid requiring an evacuation plan.

Introduction

Safety to the general public, to the operators and to the investors in a fusion power plant, has been of paramount importance in the ARIES-I design effort. The design team has made low radioactive inventories and the avoidance of toxic materials central in the reactor design. This paper will first review the safety goals for the ARIES-I reactor design, then identify the energy sources which could contribute to an accidental release of hazardous materials. The off-site doses due to releases of activation products and of tritium are shown and the waste disposal ratings of the reactor materials are calculated.

Safety Goals in the ARIES Design

In order to protect the public from accidents at the reactor, the goal in the design has been to achieve Level 1 or 2 safety assurance, as adopted by Piet [1] and the ESECOM study [2]. A reactor which has level 1 safety is "inherently safe," that is, the inventories of radioactive materials and stored energy are insufficient to cause prompt fatalities among the general public at the site boundary. A reactor which has Level 2 safety is "passively safe," that is, it requires no active safety systems to protect the general public as long as the large-scale passive features of the plant are maintained.

A second safety goal is to maintain inventories of radioactive and toxic materials in the plant low enough so that an off-site evacuation plan is unnecessary. The third goal for protecting the public in the long term is that all wastes produced by the plant be disposable as Class C waste in shallow land burial as regulated by 10CFR61. Investor safety is enhanced by using factory fabrication and testing whenever possible, by keeping the construction time short and by achieving a level of public safety such that the viability of the utility would not be threatened by any event within the plant.

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Accident Analyses

Two ingredients are necessary for an accident involving radioactive or toxic materials to endanger the public: an inventory of a hazardous substance and an energy source sufficient to release that substance. The amount of energy available in the accident determines the maximum temperature reached by the radioactive material and thus the release fraction. The product of release fraction and the radioactive inventory determine the dose to the general public.

In the ARIES-I design, radioactive inventories have been reduced by choosing low activation materials such as silicon carbide and beryllium and by isotopically tailoring zirconium and tungsten to reduce easily activated isotopes. Energy sources have been reduced by using an inert coolant, by isotopically tailoring materials to reduce afterheat and by arranging the magnets to prevent transfer of energy to more radioactive components.

Afterheat

The afterheat in the ARIES-I reactor is due to the zirconium in the Li_2ZrO_3 (about 85%), secondly, due to the tungsten coating on the divertor (about 12 %) and lastly, due to silicon in the structural ceramics composite, SiC. Lithium and oxygen in the breeder produce little afterheat due to their short half-lived activation products.

Releasable Inventories and Public Dose Calculations

The final step in determining the off-site dose is the modeling of the transport and deposition of various radionuclides from the release point to the person at the site boundary. The worst case weather conditions used in calculating off-site doses following accidental release of tritium, zirconium or tungsten are shown in Table 1.

Table 1. Worst case weather conditions used in the FUSECRAC code

Stability	Class F
Windspeed	1 m/s
Inversion Layer Height	250 m
Population Density	50 / km ²
Plume Dimensions	
sigma-y	100 m
sigma-z	50 m
Ground Level Release	
Puff Release Duration	3 min
Deposition Velocity	0.01 m/s

Lithium Zirconate Breeder

Three tritium breeders were considered for use in the ARIES-I design: Li_4SiO_4 , Li_2ZrO_3 and Li_2O . Initial analyses focused on the Li_4SiO_4 because of its low activation and low afterheat. However, because the database on the performance of Li_4SiO_4 under fusion irradiation conditions is sparse, Li_2ZrO_3 was chosen as the reference breeder material. The use of zirconium poses much more severe activation concerns because of the production of ^{89}Zr , resulting in high doses due to accidental releases and because of the production of ^{93}Zr , $t_{1/2} = 1.5$ Myr, a concern in radioactive waste management.

The ARIES-I design is not inherently dependent on any of these three breeders and the orthosilicate or the oxide could be used. However, because of its larger database and resistance to radiation damage, the zirconate was chosen as the reference breeder.

In order to reduce the zirconium activation in Li_2ZrO_3 , varying degrees of isotopic tailoring were invoked to enrich the zirconium in ^{91}Zr and/or ^{92}Zr . The isotopically-tailored zirconium employed for the Li_2ZrO_3 breeder is attractive for the low activation SiC composite structured ARIES-I blanket. The tailored zirconium contains increased proportions of ^{91}Zr and ^{92}Zr , while the ^{90}Zr , ^{94}Zr , and ^{96}Zr have been removed. The isotopic abundances are shown in Table 2. Case 4 produced afterheat a factor of 6.7 lower than that of natural zirconium. The major induced radioisotopes from the tailored Li_2ZrO_3 breeder, that are contributing to the afterheat, are ^{90}Y (64.1 h), ^{91}Y (58.5 d), ^{92}Y (3.54 h), ^{89}Zr (78.4 h), ^{89}Sr (50.6 d), and ^{95}Zr (64.0 d); and ^{89}Zr (3.27 d), ^{95}Zr (64 d), ^{89m}Zr (4.18 min), ^{91m}Y (49.7 min), ^{90m}Y (3.19 h), and ^{95}Nb (35.0 d), respectively, for beta and gamma-ray energies. The tungsten coating on the divertor was enriched in ^{183}W and depleted in ^{186}W to avoid the production of ^{187}W ($t_{1/2} = 23.9$ h), the major source of afterheat. A summary of those enrichments, their costs and the reduction in off-site doses are shown in Table 2.

The off-site dose resulting from accidental releases of 0.1 percent of the Li_2ZrO_3 using natural zirconium is shown in Table 2, Case 1. Case 6 summarizes the off-site doses for an accidental release of 0.1 percent of the Li_2ZrO_3 inventory using a moderately tailored zirconium. In Case 6 the concentrations of ^{90}Zr , ^{94}Zr , and ^{96}Zr have been reduced by a factor of 50 from their natural concentrations. Note that the 1 km early dose has been reduced by about a factor of 20 from the natural zirconium case. A much more strongly tailored mixture, which was used in the design and in which only ^{92}Zr is selected, is shown as Case 7.

Table 2 Off-site dose, Waste Disposal Rating and Cost of Isotopically Tailored Zirconium in ARIES-I, Releasing 0.1 % of the Zr Inventory

Case	1	2	3	4	5	6	7
Isotopic (nat)(-----isotopically tailored-----)							
Composition (%)							
⁹⁰ Zr	51.45	25.73	10.29	5.15	2.57	1.03	0.057
⁹¹ Zr	11.27	25.45	33.96	36.79	38.21	39.06	0.013
⁹² Zr	17.17	38.77	51.73	56.05	58.21	59.51	99.91
⁹⁴ Zr	17.33	8.67	3.47	1.73	0.87	0.35	0.019
⁹⁶ Zr	2.78	1.39	0.56	0.28	0.14	0.06	0.003
1 km Dose (rem)							
	694.1	356.6	155.9	89.03	55.47	35.52	4.55
Waste Disposal Ratings							
	7.27	3.64	1.47	0.74	0.38	0.17	0.05
Tailoring Cost (per kg of Zr product at \$50/swu)							
	0	\$66	\$198	\$296	\$391	\$511	\$2097

Note: The dominant isotopes contributing to the dose for natural Zr are: ⁸⁹Zr (72%) and ⁸⁸Y (15%), both of which are produced by ⁹⁰Zr.

To evaluate the level of safety assurance we have considered a 2 percent release of the entire Zr inventory. This case, as summarized in Table 3, represents an accident in which all the 64 blanket modules fail simultaneously, as they might due to helium coolant overpressure. For solid breeders, which may contain small particulate due to radiation damage, one can generally assume a release fraction of 1 - 3 percent [4]. Because of its low vapor pressure, the afterheat is not sufficient to volatilize the ceramic breeder. The zirconium has been very strongly tailored, such that the concentration of ⁹²Zr is 99.91 percent.

Although this design requires heroic isotopic tailoring, the cost, approximately \$72 million for the complete reactor, is a small fraction of the total reactor cost. Note that the release fraction can be limited to 2 percent of the total zirconium inventory because the energy sources available to cause the release, i.e. the decay heat and the compressed helium, are insufficient to drive the inventory out of the containment. Neither SiC nor the breeder react with the coolant or air at the temperatures that can be attained in a LOCA. A determining factor in the transport of the zirconium inventory is the size distribution of the lithium zirconate particles after the rupture of the blanket modules. If radiation damage does not cause the ceramic to be

Table 3 Off-site doses resulting from an accidental release of 2 percent of the Li₂ZrO₃ inventory, using strongly tailored zirconium.

Isotope	Abundance	
⁹⁰ Zr	0.057 %	
⁹¹ Zr	0.013 %	
⁹² Zr	99.908 %	
⁹⁴ Zr	0.019 %	
⁹⁶ Zr	0.003 %	
Cost of tailoring: \$2097/kg _{Zr} using AVLIS at \$50/swu.		
Release Fraction: 2.0000 percent		
Dominant Isotopes		
Prompt Dose at 1 km		percentage:
WB	63.68 rem	⁸⁹ Zr 57.6
BM	71.05 rem	⁸⁹ Zr 59.9
Lung	1056.97 rem	⁹¹ Y 71.3
LLI	319.74 rem	⁹⁰ Y 42.6
WB Early Dose		
1 km	91.04 rem	⁸⁹ Zr 40.6
10 km	6.12 rem	⁸⁹ Zr 41.2
WB Chronic Dose at 1 km		
Inh+grd	261.63 rem	⁸⁸ Y 53.2
Ingestion	2918.39 rem	⁸⁹ Sr 63.8
Total	3179.96 rem	⁸⁹ Sr 59.3
WB Chronic Dose at 10 km		
Inh+grd	17.88 rem	⁸⁸ Y 54.1
Ingestion	202.74 rem	⁸⁹ Sr 63.7
Total	220.61 rem	⁸⁹ Sr 59.3
Cancers		
Sum organs	783.60	⁸⁹ Sr 52.9
WB	199.54	⁸⁹ Sr 66.4
Population dose		
WB	1,262,210. man-rem	⁸⁹ Sr 66.3
WDR		
	0.05	⁹³ Zr 99.8

Note that the off-site dose is still dominated by ⁸⁹Zr.

divided into a fine powder, the release fraction from the reactor building could be limited to less than 2 percent, especially since the distance to the site boundary is 1 km and particles greater than 10 microns would not remain airborne.

Tungsten coating on divertor

The plasma-facing surfaces of the ARIES-I divertor are coated with a 2 mm layer of tungsten. The area of

the tungsten coating is 150 m². That tungsten, and the rhenium resulting from neutron activation, will be released if the hot surfaces of the divertor are exposed to air during an accident. In recent experiments at the INEL [5], an inductively heated sample of a tungsten alloy was exposed to upward flowing air in a vertical tube. The sample, 10.4 mm in diameter by 4.7 mm thick, contained the elements shown in Table 4.

Table 4 Experimental tungsten alloy composition (weight percent)

W	Ni	Re	Fe	Co	Ta	Os	Cu	Mn
95.	2.3	1.1	1.0	0.55	0.15	0.06	0.05	0.05

The volatilization rates at 1273 K, the maximum temperature of the divertor during a LOCA with plasma shutdown, are shown in Table 5.

Table 5 Experimental tungsten alloy volatilization rates, g/m²-h

Temperature	time	Tungsten Rhenium	
600 C	20 h	1.4E-2	1.9E-2
	1 h	0.12	5.2
	5 h	6.8E-4	2.4
800 C	20 h	0.14	1.3
	1 h	1.2	12.4
	5 h	5.4E-2	4.4
1000 C	10 h	0.51	2.6
	1 h	36.4	11.2
	Max Release	0.2 %	14. %

Ta: below detachability for 20 h duration

In assessing the off-site doses from an air ingress accident on the ARIES-I reactor, we have assumed that the worst-case tungsten release is a 1273 K, 10-hour LOCA in which the 1-hour release rates are used for the full 10 hours to account for any flaking of any protective oxide due to vibration during the accident. The divertor temperature during the LOCA is based on an analysis in which the plasma is shut off within 15 seconds of the loss of coolant. The off-site doses are shown in Table 6 below for the case of tungsten enriched to 90 percent ¹⁸³W irradiated for 2 years at 1 MW/m². The release fractions of 0.03 percent for the tungsten and 29.2 percent for the rhenium are based on the volatilization rates shown in Table 5. All other elements are assumed to be volatilized at the detection limit in the experiment.

Table 6 Tungsten-Rhenium Accidental Release Reference Case

Release Fraction: 0.0327 percent W
29.2040 percent Re
Detection limit: 0.1969 percent
(based on Ta, used for non-Re-W elements)
Accident duration: 10 hours at 1000° C,
using 1 hr release rate
Irradiation: 1 MW/m² for 2.0 years
Inventory: 5790 kg

Initial Isotopic Composition
atom fraction

¹⁸⁰ W	0.0001
¹⁸² W	0.0306
¹⁸³ W	0.9000
¹⁸⁴ W	0.0359
¹⁸⁶ W	0.0334
Total	1.0000

Prompt Dose at 1 km

WB	9.34 rem	¹⁸⁶ Re	90.2
BM	13.96 rem	¹⁸⁶ Re	91.9
Lung	140.71 rem	¹⁸⁶ Re	98.2
LLI	26.48 rem	¹⁸⁶ Re	95.4

WB EARLY Dose

1 km	11.20 rem	¹⁸⁶ Re	91.2
10 km	0.76 rem	¹⁸⁶ Re	91.4

WB Chronic Dose at 1 km

Inh+grd	18.98 rem	¹⁸⁶ Re	77.1
Ingestion	294.07 rem	¹⁸⁶ Re	95.1
Total	312.99 rem	¹⁸⁶ Re	94.0

WB Chronic Dose at 10 km

Inh+grd	1.30 rem	¹⁸⁶ Re	77.0
Ingestion	20.07 rem	¹⁸⁶ Re	95.1
Total	21.36 rem	¹⁸⁶ Re	94.0

Cancers

Sum organs	128.23	¹⁸⁶ Re	82.8
WB	56.22	¹⁸⁶ Re	93.6

Population dose

WB	356,280. man-rem	¹⁸⁶ Re	93.6
WDR	0.10	¹⁷⁸ Hf	65.0

The off-site doses for other isotopic compositions and other irradiation times are shown in Table 7.

Tritium Inventories

The tritium inventory in the plasma facing components are basically calculated by using the DIFFUSE code developed by Sandia National

Table 7 ARIES-I Tungsten-Air Reaction Summary

Tungsten thickness:	2.0 mm	
Divertor area:	150. m ²	
LOCA conditions:		
Temperature:	1000° C	
Accident Duration:	10 hr	
W rel rate	1.200 g/m ² -hr	0.0327 percent
Re rel rate	12.400 g/m ² -hr	29.2040 percent
other	0.0114 g/m ² -hr	0.1969 percent
(using Ta detection limit)		

Isotopic Mixtures (atom percent)			
Natural	2.86% W-186	90.0% W-183	
W-180	0.13	0.18	0.02
W-182	26.30	35.78	3.23
W-183	14.30	19.46	90.00
W-184	30.67	41.73	3.76
W-186	28.60	2.86	3.51

Doses

1 MW/m ² for 0.5 yr			
Early 1 km	8.71	3.62	1.14 rem
Pop dose	119,230	101,696	17,496 man-rem
WDR	0.11	0.14	0.01
1 MW/m ² for 1.0 yr			
Early 1 km	24.06	16.88	3.45 rem
Pop dose	638,121	550,529	95,376 man-rem
WDR	0.24	0.29	0.03
1 MW/m ² for 2.0 yr			
Early 1 km	57.63	46.81	11.2 rem
Pop dose	1,789,443	1,587,206	356,280 man-rem
WDR	0.57	0.65	0.10
1 MW/m ² for 3.0 yr			
Early 1 km	85.57	73.15	21.61 rem
Pop. dose	2,754,533	2,510,282	704,716 man-rem
WDR	0.99	1.08	0.20
1 MW/m ² for 4.0 yr			
Early 1 km	108.53	95.59	33.79 rem
Pop dose	3,558,365	3,296,640	1,110,428 man-rem
WDR	1.50	1.58	0.35
1 MW/m ² for 5.0 yr			
Early 1 km	126.86	114.72	47.1 rem
Pop. dose	4,198,197	3,965,330	1,552,344 man-rem
WDR	2.05	2.13	0.54

Laboratories. The DIFFUSE code uses diffusion as the only mechanism for the tritium mass transfer inside the

solid. In the ARIES-I design, we investigated the effect of sputtering on the tritium diffusion mechanism. The sputtering process changes the mass transfer problem at the first wall into a problem with a moving boundary. If the sputtering removal rate is much faster than the mass diffusion rate, the tritium within the deposition layer will be removed by the sputtering process, together with the structure material, before it has enough time to diffuse forward.

For the ARIES first wall design, the tritium penetration depth is about 2 nm, while the sputtering rate of the first wall is estimated to be 0.2 nm/y. Therefore, the life of the deposition depth (2 nm) is only about 100 seconds and the maximum tritium inventory within the first wall cannot exceed the tritium incident on the first wall during 100 seconds. The total tritium inventory thus calculated is only 6 g, comparing a few kg's predicted without considering sputtering. Similar calculation indicates tritium inventory in the ITER divertor to be in the 10's g, although consideration of trapping may increase the estimate of the inventory dramatically. The tritium inventory summary is shown in Table 8.

Table 8 Tritium Inventories in the ARIES-I Design

Component	Tritium inventory,g	comments
Fuel cycle		
Vacuum pumping	small	Using ceramic TMP
He separation	small	Using Pd diffuser
Cryogenic distillation	50	Using CD to remove H only
First wall	6	Sputtering removes Tritium
Divertor wall	10	Using W on SiC
Blanket		
Breeding material	1	Selection of breeding material and temperature
Beryllium		
	640	1. Complete tritium released at >883 K
		2. Blanket is heated to >883 K yearly
Total	707 g T	

The total tritium inventory is about 700 g. The tritium inventory in Be is highly uncertain. It can be easily higher (or lower) by a factor of a 3 to 5.

Tritium Release Estimates

The total tritium production and deposition rates in the Be is estimated to be 2.5 kg/fpy. The tritium production within the Be is estimated to be 1.15 kg/fpy and the recoil deposition is 1.4 kg/fpy. Recent data (Billone, ANL, 1988 and recent tests at HEDL of ATR reflector Be) indicate that the tritium is released from the beryllium at temperatures above 883 K. Less than one quarter of the beryllium in the blanket will have a normal operating temperature below 880 K. The entire blanket will be raised to a temperature greater than 883 K at least once per year, thereby driving out the entire tritium inventory. Thus, a maximum of 0.64 kg of tritium would be released from the blanket in a LOCA transient in which the plasma was NOT shut-down within 15 s. Using Porter's FUSECRAC calculations [3] for approximately the same conditions, also with a quality factor of 1, the dose is about 7.1 rem.

Waste Disposal Ratings

The Waste Disposal Rating (WDR) is the ratio of the total weighted concentration of radionuclides in a material to that allowable for Class C waste under 10CFR61. Since 10CFR61 was developed for fission reactors, most of the fusion-specific isotopes are not explicitly listed. The WDRs for each of the materials shown in the tables above is based on an extension of the methodology of 10CFR61 to all radionuclides with $Z < 88$ by Fetter, Cheng and Mann [6].

Conclusions

The ARIES-I design achieved its goals of being passively safe as defined in the ESECOM study and that all waste be disposable by shallow land burial under 10CFR61 Class C.

By using isotopic tailoring of the zirconium and tungsten and by keeping the tritium inventory low, the 1 km early doses are as shown in Table 9.

Table 9 Summary of ARIES-I Off-site Doses

Source	Dose (rem)
Zr	91. (using 99.91 percent ^{92}Zr , 2 percent release)
W	11.2 (using 90 percent ^{183}W , 10 hour LOCA at 1000 C)
Tritium	6.4 (FUSECRAC analysis, $qf=1$, 1/4 of Be < 883 K)
FW & shield	21. (impurities)
Total	130. rem

Thus the design is passively safe because of its use of 99.91 percent ^{92}Zr in the Li_2ZrO_3 breeder and 90 percent W-183 as the tungsten coating. The doses shown above assume that 2 percent of the entire Li_2ZrO_3 inventory is released from the reactor due to the failure of all 64 modules. No credit is assumed for retention of the ceramic breeder material with the reactor building. The off-site doses due to an air-ingress accident and tungsten release are based on experimental data on the volatilization tungsten and rhenium at 1000° C for 10 hours, using the one hour release rates to account for the loss of protective oxides. The LOCA analyses on the divertor assume retention of the divertor geometry and passive (i.e. radiative) cooling paths which limited the maximum divertor temperature to 1000° C. Without passive cooling, the divertor coating would have reached much higher temperatures and release fractions of tungsten and rhenium would have been substantially higher.

The doses due to impurities in the first wall, blanket, and shield are based on impurity concentrations assumed in the ESECOM study.

An evacuation plan will probably be necessary, since the off-site dose due to any of the components in Table 9 exceeds the 1-5 rem typically adopted by state and local authorities as a threshold level. The off-site doses could be further reduced by using another breeder, such as Li_4SiO_4 or Li_2O .

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