

10. ARIES-I SAFETY DESIGN AND ANALYSIS

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Contents

10.1.	INTRODUCTION	10-1
10.2.	SAFETY GOALS IN THE ARIES-I DESIGN	10-1
10.3.	ACCIDENT ANALYSES	10-2
10.3.1.	Afterheat	10-2
10.3.2.	Loss-of-Coolant Analysis	10-5
10.3.3.	Superconducting Magnet Transients	10-6
10.4.	RELEASABLE INVENTORIES AND PUBLIC DOSE	10-8
10.4.1.	Lithium-Zirconate Breeding Material	10-9
10.4.2.	Tungsten Coating on the Divertor	10-13
10.4.3.	Tritium Inventories	10-18
10.4.4.	Silicon-Carbide Structure	10-19
10.4.5.	Role of Impurities in Materials	10-20
10.5.	WASTE DISPOSAL RATINGS	10-22
10.6.	ECONOMIC IMPACT OF SAFETY ISSUES	10-22
10.7.	SUMMARY AND CONCLUSIONS	10-29
	REFERENCES	10-31

10. ARIES-I SAFETY DESIGN AND ANALYSIS

10.1. INTRODUCTION

Safety—of the general public, of the operators, and of the investors in a fusion power plant—has been of paramount importance in the ARIES-I design effort. The ARIES-I design team has made low radioactive inventories and the avoidance of toxic materials central to the reactor design. For this reason, silicon-carbide composite is used as the structural material for the blanket and divertor as well as for the shield. Helium, which is chemically inert and is not neutronically activated, is used as the coolant. The zirconium in the lithium-zirconate breeder and the tungsten coating on the divertor have been isotopically tailored to reduce, though certainly not eliminate, the production of radioactive materials. This section will first review the safety goals for the ARIES-I reactor design (Sec. 10.2), and then identify the energy sources that could contribute to an accidental release of hazardous materials (Sec. 10.3). The off-site doses resulting from releases of activation products and of tritium are presented in Sec. 10.4. The waste disposal ratings of the reactor materials are calculated in Sec. 10.5. Section 10.6 discusses the economic benefits of using low-activation materials and the avoidance of nuclear-grade equipment. Summary and conclusions are given in Sec. 10.7.

10.2. SAFETY GOALS IN THE ARIES-I 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 of safety assurance, as adopted by Piet [1] and the ESECOM study [2]. A reactor that 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. Other authors have defined inherent safety solely in terms of the inventories of radioactive materials, without considering whether energy sources are available to release those inventories. This section will consider both radioactive inventories and energy sources in defining inherent safety. A reactor that 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. A 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 [3].

Investor safety is enhanced by using factory fabrication of large components and extensive 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.

10.3. ACCIDENT ANALYSES

Two ingredients are necessary for an accident involving radioactive or toxic materials to endanger the public: (1) an inventory of a hazardous substance and (2) 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 a chemically inert coolant, by using low-afterheat material (SiC), by isotopically tailoring materials to reduce afterheat, and by arranging the magnets to prevent a transfer of energy to more radioactive components.

The two principal energy sources that could cause the accidental release of hazardous inventories are the afterheat in the breeder and structural materials (which could cause overheating in case of a loss-of-coolant accident), and the energy stored in the toroidal-field (TF) magnets.

10.3.1. Afterheat

The induced radioactivities in the structural silicon-carbide (SiC) composite and breeder materials (although relatively small compared to metals) are the main sources of afterheat for the blanket component. Most of the afterheat in the SiC composite is due to Si. The major induced radioisotopes from Si that are contributing to the afterheat are ^{28}Al (half-life 2.24 min), ^{29}Al (6.56 min), ^{27}Mg (9.46 min), ^{24}Na (15 h), ^{28}Mg (20.9 h), and ^{31}Si (2.62 h). These radioisotopes are significant for both beta and gamma-ray energies. The major induced radioisotopes from carbon, however, are ^{13}B (17.4 ms), ^6He

(807 ms), and ^{12}B (20.2 ms), indicating the insignificant contribution made by these very short half-life radioisotopes to the afterheat.

Induced afterheat in the breeder, Li_2ZrO_3 , is primarily due to the element Zr. Lithium and oxygen produce little afterheat because of the very short half-life of induced radioisotopes. Isotopically-tailored zirconium is used for the Li_2ZrO_3 breeder in order to retain the low-activation characteristics of the ARIES-I blanket. The zirconium is tailored to contain 99.9% ^{92}Zr and, thus, only 0.1% (compared to their 72% natural abundance) of the high afterheat isotopes ^{90}Zr , ^{94}Zr , and ^{96}Zr . Such tailored-Zr element will produce induced afterheat levels a factor of 6.7 lower than the natural Zr. 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) for beta decay energy, and ^{89}Zr , ^{95}Zr , $^{89*}\text{Zr}$ (4.18 min), $^{91*}\text{Y}$ (49.7 min), $^{90*}\text{Y}$ (3.19 h), and ^{95}Nb (35.0 d) for gamma-ray energy.

The beta decay energies are considered to be deposited at the site of the decay. However, the gamma-ray energies will be transported and deposited throughout the materials in the reactor components. The combined beta and gamma-ray energies deposited in the ARIES-I reactor were estimated after four full power years (FPY) of operation at 5 MW/m² neutron wall loading. The afterheat and neutron activation in the ARIES-I reactor components was estimated using the ANISN neutron-transport code [4] and the REAC*2 activation-analysis code [5]. The results are given in Figs. 10.3-1 for the ARIES-I first wall and blanket.

Figure 10.3-1(A) shows the afterheat level at the ARIES-I first wall as a function of time after shutdown. Beta decay energies constitute the primary afterheat source for the first wall from shutdown to about one hour after shutdown. Thereafter, until about two days after shutdown, beta and gamma-ray energies contribute equally to the afterheat. After two days, the first-wall afterheat is principally due to gamma-rays from the Li_2ZrO_3 breeder in the breeding zone. At shutdown, the afterheat level is about 0.4 MW/m³. At one hour after shutdown, the afterheat level at the first wall drops to about 0.004 MW/m³, a factor of 100 reduction compared to that at shutdown. After two days the level of heating drops to three orders of magnitude less than at shutdown. Figure 10.3-1(B) shows the afterheat level at the ARIES-I blanket breeding zone.

Figure 10.3-2 gives the afterheat in the ARIES-I blanket and shield. The maximum heating rate is shown to be immediately behind the first wall. The heating rate is dominated by the decay energies resulting from SiC for only the first 10 minutes after shutdown. At one hour after shutdown and thereafter, the heating rate will be primarily the result of decay energies from the Li_2ZrO_3 breeder. At shutdown the heating rate is

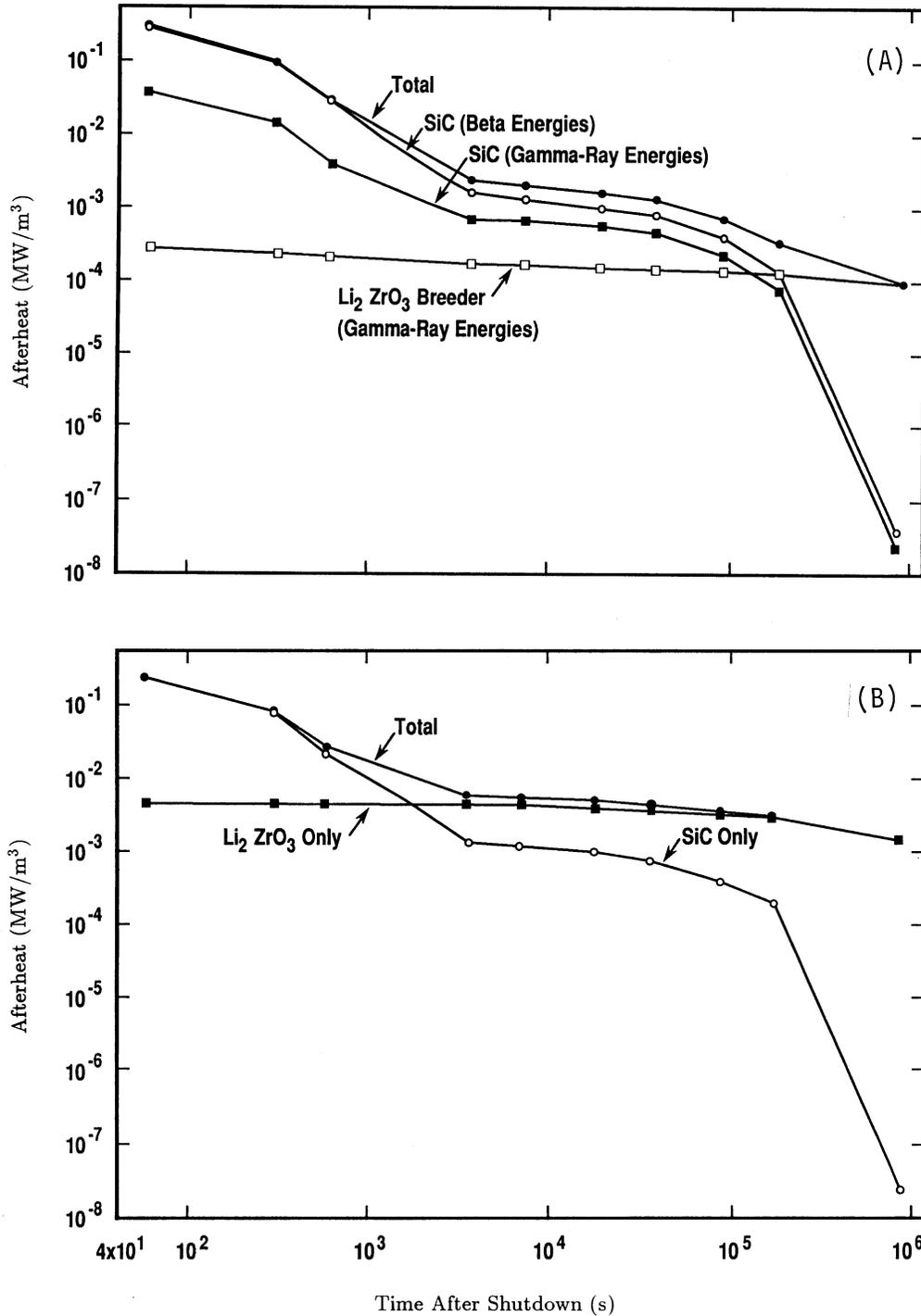


Figure 10.3-1. Afterheat in the ARIES-I (A) first wall and (B) blanket breeding zone as functions of time after shutdown (after 4 FPY at 5-MW/m² wall loading).

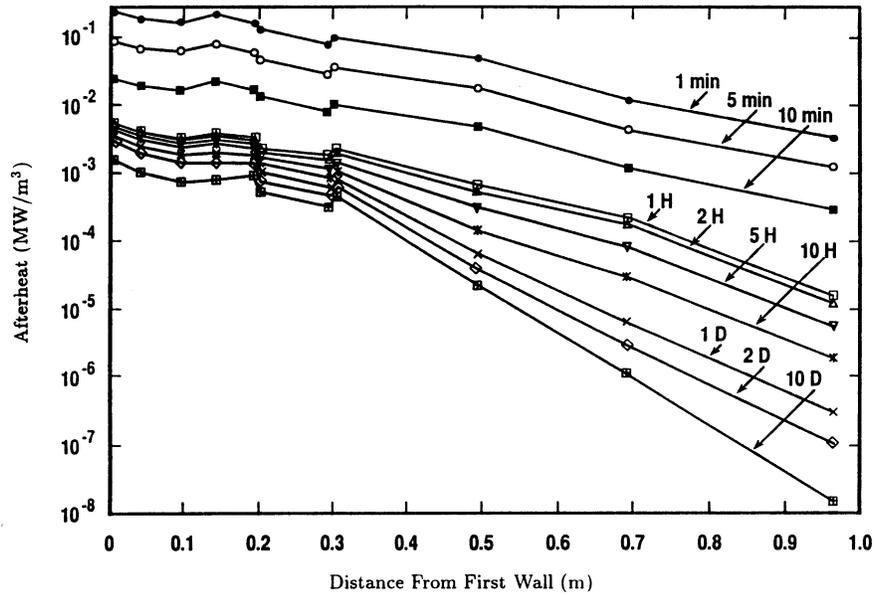


Figure 10.3-2. Afterheat in the ARIES-I blanket and shield as a function of distance from first wall (after 4 FPY at 5-MW/m² wall loading).

about 0.25 MW/m³, and drops to about 0.03 MW/m³ after 10 minutes. At one hour after shutdown and thereafter, the heating level is maintaining at about 0.005 MW/m³.

10.3.2. Loss-of-Coolant Analysis

Selecting materials that generate low levels of afterheat is expected to result in a blanket design that should suffer few adverse consequences from a loss-of-coolant accident (LOCA). To verify this assumption, a two-dimensional (2-D) finite-element model of the ARIES-I first wall, blanket, and shield was prepared and analyzed using the computer code, TOPAZ [6]. The level of afterheat in the first wall is quite low, initially 0.4 MW/m³, at the onset of the LOCA. The afterheat levels throughout the blanket drop by about an order of magnitude after five minutes. The following assumptions are made:

1. At the onset of the LOCA, the plasma is quenched instantly.
2. The heat removal capacity of the coolant is neglected as the gas pressure drops.
3. All of the structure is initially at a uniform temperature of 700 °C.
4. The vacuum boundary remains intact and no conduction through it is provided.

The results of this analysis show that the peak temperature during the LOCA occurs in the first wall at about 280 seconds into the event. The peak temperature is 780 °C. The temperature rise of ~80 °C is quite low and is primarily due to the large volume and heat capacity of the beryllium in the blanket. Of particular interest is the determination of the maximum plasma quench time as determined by temperature excursions in the ARIES-I first wall.

The divertor is the one component of the fusion power core (FPC) that does not have low levels of afterheat and has also been treated to the LOCA analysis. The divertor consists of a 2-mm-thick coating of tungsten on a helium-cooled substrate of SiC composite tubes. At shutdown, the level of afterheat in the tungsten is ~5 MW/m³ and is approximately constant for periods in excess of one hour. The divertor target faces into the plasma chamber and radiates the power to the first wall. Assuming the first wall reaches the calculated LOCA temperature of 780 °C and remains constant, then the peak temperature reached by the divertor target is conservatively estimated to be 1020 °C.

10.3.3. Superconducting Magnet Transients

In general, a large amount of energy is stored in the TF magnets. Release of this energy may potentially cause damage to the FPC and result in accidental release of hazardous inventories. The ARIES-I reactor utilizes high-field TF coils with a peak field of 21 T. Several transient scenarios were studied to determine their safety implications. The characteristics of the TF coils as used in the transient analysis are shown in Table 10.3-I. Each of the coils has been divided into two windings of 87 turns each. The TF-coil set is powered by two power supplies so that the windings in each coil are powered by two independent supplies. Dump resistors and circuit breakers are located between each of the windings.

The safety analysis, using MSCAP [7], considered a variety of faults, as shown in Table 10.3-II. In the analysis, the coils containing the fault and its immediately adjacent coils (and their associated resistors and circuit breakers) were individually modeled, while the remaining 13 coils were combined into two sets of components. The worst case for arcing, shorts, and winding-to-winding voltages occurs when those faults are adjacent to the power supplies.

Table 10.3-I.
ARIES-I Toroidal-Field Coils^(a)

Number of coils	16
Turns per coil	174
Conductor current (kA)	126
Maximum field at the conductor (T)	21
Stored energy (GJ)	130
Dump resistors (m Ω /winding)	5
Coupling coefficient between adjacent coils	0.90

^(a)With internally cooled, conduit conductors.

Table 10.3-II.
Toroidal-Field-Magnet Transients Considered

-
1. Routine discharge to establish baseline;
 2. Turn-to-turn voltages in adjacent windings due to 10-s delay in opening by one circuit breaker;
 3. Turn-to-turn voltages due to complete failure of single circuit breaker;
 4. A 10- $\mu\Omega$ short between adjacent windings, with 10-s delay in opening by one breaker;
 5. Arcs of 20 V and 35 V between adjacent windings with 10-s delay in breaker opening;
 6. Arcs of 20 V and 35 V between adjacent windings with complete failure of one breaker;
 7. Arcs of 20 V and 35 V within one winding, normal breaker operation.
-

The most severe of the transients, in terms of energy deposited, are the arcs within one winding. A model of the quench propagation was developed and used to account for the partitioning of energy between the arc and the resistive heating of the copper stabilizer. This model assumes that a quench is initiated by overcurrent in the winding in which the arc is occurring. The rest of the coil set continues to discharge routinely through the dump resistors. Quench fronts begin to propagate in both directions along each turn of the conductor, beginning from the location of maximum field. The heat deposition in the copper stabilizer was modeled, thus partitioning the energy between the arc and the stabilizer. The temperature-dependence of copper resistivity and specific heat were included. The results of this transient showed that the maximum arc power was 1.2 MW, the total deposited arc energy was 17 MJ, the maximum quench power (*i.e.*, resistive heating of the stabilizer) was 90 MW, and the total deposited quench energy was 2 GJ which is not sufficient to raise the temperature of a TF magnet to room temperature.

These analyses have shown that a severe magnet transient would not lead to a release of radioactive material and the afterheat is the dominant energy source in terms of radioactive releases. Severe magnet transients, however, have potential economic consequences because of the plant downtime for repair.

10.4. RELEASABLE INVENTORIES AND PUBLIC DOSE

In determining the dose to the general public, the analysis must first establish the radioactive inventories and then estimate release fractions based on the postulated accident and the energy available to promote that release. Finally, the transport and deposition of the various radionuclides from the release point to the person at the site boundary have to be calculated. The hazard inventories were calculated using the ANISN neutron-transport code [4] coupled to the REAC*2 activation-analysis code [5]. Estimates were made of the maximum-possible release fractions, and the public dose was calculated using the FUSECRAC code [8]. The worst case weather conditions used in calculating off-site doses [9] following accidental release of tritium, zirconium, or tungsten are shown in Table 10.4-I. Radioactive inventories, release fractions, and the resulting public doses are described in this section for the various radioactive reactor components.

Table 10.4-I.
Worst-Case Weather Conditions
Used in the FUSECRAC Code^(a)

Stability	Class F
Windspeed (m/s)	1
Inversion layer height (m)	250
Population density (km ²)	50
Plume dimensions (m)	
σ_y	100
σ_z	50
Puff release duration (min)	3
Deposition velocity (m/s)	0.01

^(a)Ground level release.

10.4.1. Lithium-Zirconate Breeding Material

During the scoping phase of the ARIES-I study, 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. Lithium-zirconate, however, was selected as the reference solid breeder because the data base demonstrates long-term and high-temperature stability. Unfortunately, the use of zirconium poses much more severe activation concerns because of the production of ^{89}Zr (half-life 78.4 h), resulting in high doses due to accidental releases and because of the production of ^{93}Zr (1.5 My), a concern in radioactive waste management. Extensive isotopic tailoring of zirconium would be necessary to reduce the level of decay heat and improve the waste disposal rating of the blanket. Laser isotope separation [10] is the only feasible technique for this tailoring since both light and heavy isotopes are to be removed. Table 10.4-II summarizes the reduction in the off-site dose for several level of enrichment (tailoring) and the associated costs.

The off-site doses resulting from accidental releases of 0.1% of the Li_2ZrO_3 using natural zirconium are shown in Table 10.4-III. For a comparison, Table 10.4-IV summarizes

Table 10.4-II.
Off-Site Dose,^(a) Waste Disposal Rating, and Cost of
Isotopically Tailored Zirconium in ARIES-I

Case	Natural ^(b)	Isotopically Tailored					
	1	2	3	4	5	6	7
Isotopic 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	99.908
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 rating	7.27	3.64	1.47	0.74	0.38	0.17	0.05
Tailoring cost ^(c) (\$/kg Zr)	0	66	198	296	391	511	2097

^(a)Off-site dose calculated based on releasing 0.1% of Zr Inventory.

^(b)The dominant isotopes contributing to the dose for natural Zr are:

⁸⁹Zr (72%) and ⁸⁸Y (15%), both of which are produced by ⁹⁰Zr.

^(c)Cost based on \$50/swu (separative work unit).

the off-site doses using the tailored zirconium shown as Case 6. In this tailoring, the concentrations of ⁹⁰Zr, ⁹⁴Zr, and ⁹⁶Zr 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.

The reference ARIES-I breeder is enriched to 99.9% in ⁹²Zr at the considerable cost of ~\$2100/kg (Case 7 of Table 10.4-II). Even at this level of enrichment, the total off-site dose of the ARIES-I reactor is still dominated by the Zr. Other solid breeders such as Li₂O and Li₄SiO₄ can be used as alternative solid breeders. These are low-activation breeders that do not require isotopic tailoring, and their use would drastically reduce the off-site dose of the ARIES-I reactor.

Table 10.4-III.
Off-Site Doses
After Release of 0.1% of the Li_2ZrO_3 Inventory
Using Natural Zirconium^(a)

	Dose (rem)	Dominant Isotope (% dose)
Prompt dose at 1 km		
Whole body (WB)	669.42	⁸⁹ Zr (72.5)
Bone marrow (BM)	766.48	⁸⁹ Zr (73.4)
Lung	1065.96	⁸⁹ Zr (52.9)
Large lower intestine (LLI)	662.20	⁸⁹ Zr (63.9)
WB early dose		
1 km	691.25	⁸⁹ Zr (70.8)
10 km	47.29	⁸⁹ Zr (70.4)
WB chronic dose at 1 km (chronic plus early)		
Inhalation + groundshine	2162.58	⁸⁸ Y (43.9)
Ingestion	1187.44	⁹⁰ Sr (33.6)
Total	3349.48	⁸⁸ Y (32.4)
WB chronic dose at 10 km (chronic plus early)		
Inhalation + groundshine	148.99	⁸⁸ Y (44.3)
Ingestion	82.01	⁹⁰ Sr (33.8)
Total	230.98	⁸⁸ Y (32.6)
Cancers		
All organs	398.29	⁹⁵ Nb (31.0)
WB	133.22	⁹⁵ Nb (26.4)
Population dose		
WB ($\times 10^3$ man-rem)	843.98	⁹⁵ Nb (26.4)
Waste disposal rating	0.13	⁹³ Zr (66.6)

^(a)For 35,420 kg of inventory and irradiated for 4 FPY at 5 MW/m².

The initial isotopic composition is:

⁹⁰Zr (51.45%), ⁹¹Zr (11.27%), ⁹²Zr (17.17%), ⁹⁴Zr (17.33%), ⁹⁶Zr (2.78%).

Table 10.4-IV.
Off-Site Doses
After Release of 0.1% of the Li_2ZrO_3 Inventory
Using Moderately Tailored Zirconium^(a)

	Dose (rem)	Dominant Isotope (% dose)
Prompt dose at 1 km		
Whole body (WB)	34.41	⁸⁹ Zr (80.3)
Bone marrow (BM)	39.17	⁸⁹ Zr (81.8)
Lung	236.30	⁹¹ Y (75.4)
Large lower intestine (LLI)	63.50	⁸⁹ Zr (37.9)
WB early dose		
1 km	38.35	⁸⁹ Zr (72.7)
10 km	2.61	⁸⁹ Zr (72.6)
WB chronic dose at 1 km (chronic plus early)		
Inhalation + groundshine	99.71	⁸⁹ Y (44.4)
Ingestion	308.73	⁹⁰ Sr (49.6)
Total	408.38	⁹⁰ Sr (37.6)
WB chronic dose at 10 km (chronic plus early)		
Inhalation + groundshine	6.85	⁸⁹ Zr (44.0)
Ingestion	21.43	⁹⁰ Sr (49.7)
Total	28.27	⁹⁰ Sr (41.3)
Cancers		
All organs	97.26	⁹¹ Y (47.6)
WB	21.07	⁹⁰ Sr (41.3)
Population dose		
WB ($\times 10^3$ man-rem)	133.41	⁹⁰ Sr (41.3)
Waste disposal rating	0.03	⁹³ Zr (96.4)

^(a)For 35,420 kg of inventory and irradiated for 4 FPY at 5 MW/m².

The initial isotopic composition is:

⁹⁰Zr (1.03%), ⁹¹Zr (39.06%), ⁹²Zr (59.51%), ⁹⁴Zr (0.35%), ⁹⁶Zr (0.06%).

The cases shown in Tables 10.4-III and 10.4-IV assume a release of 0.1% of the total zirconium inventory. We have assumed that one of the 64 blanket modules fails due to overpressure and that two adjacent modules are damaged by the failure. For solid breeders, which may contain small particulates due to radiation damage, one can generally assume a release fraction of 1%-3% [11]. Thus, the release fraction assumed is $(3/64) \times 2\%$ or approximately 0.1% of the total zirconium inventory. Furthermore, 0.1% is consistent with a dust release, since the afterheat is not sufficient to volatilize the ceramic breeder.

If dust generation is determined to be a severe problem because of radiation damage, a greater enrichment would be required. As the ARIES-I reference case, we have considered a 2% release of the entire Zr inventory. This case, as summarized in Table 10.4-V, represents an accident in which all of the 64 blanket modules fail simultaneously, as they might because of helium-coolant overpressure. The zirconium has been very strongly tailored, such that the concentration of ^{92}Zr is 99.91%. The cost of tailoring is estimated at \$2,097/kg of Zr using ALVIS [10] at \$50/swu.

Although this is fairly extensive isotopic tailoring, the total cost, \$72 M for the complete reactor, is a small fraction of the total reactor cost. Note that the release fraction can be limited to 2% 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 Zr inventory is the size distribution of the lithium-zirconate particles after the rupture of the blanket modules. If further data show that the breeder ceramic maintains its geometric integrity at high fluences (*e.g.* ceramic does not disintegrate into a fine powder), the release fraction from the reactor building could decrease below the assumed 2% level, especially since the distance to the site boundary is 1 km and particles greater than 10 microns would not remain airborne.

10.4.2. Tungsten Coating on the Divertor

The ARIES-I divertor plates are coated with a 2-mm-thick 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 INEL [12], 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 following elements (concentrations

Table 10.4-V.
Off-Site Doses
After Release of 2% of the Li_2ZrO_3 Inventory
Using Strongly Tailored Zirconium^(a)
(ARIES-I reference case)

	Dose (rem)	Dominant Isotope (% dose)
Prompt dose at 1 km		
Whole body (WB)	63.68	⁸⁹ Zr (57.6)
Bone marrow (BM)	71.05	⁸⁹ Zr (59.9)
Lung	1,056.97	⁹¹ Y (71.3)
Large lower intestine (LLI)	319.74	⁹⁰ Y (37.9)
WB early dose		
1 km	91.04	⁸⁹ Zr (40.6)
10 km	6.12	⁸⁹ Zr (41.2)
WB chronic dose at 1 km (chronic plus early)		
Inhalation + groundshine	261.63	⁸⁸ Y (53.2)
Ingestion	2,918.39	⁸⁹ Sr (63.8)
Total	3,179.96	⁸⁹ Sr (59.3)
WB chronic dose at 10 km (chronic plus early)		
Inhalation + groundshine	17.88	⁸⁸ Y (54.1)
Ingestion	202.74	⁸⁹ Sr (63.7)
Total	220.61	⁸⁹ Sr (59.3)
Cancers		
All organs	783.60	⁸⁹ Sr (52.9)
WB	199.54	⁸⁹ Sr (66.3)
Population dose		
WB ($\times 10^3$ man-rem)	1,262.21	⁸⁹ Sr (66.3)
Waste disposal rating	0.05	⁹³ Zr (99.8)

^(a)For 35,420 kg of inventory and irradiated for 4 FPY at 5 MW/m².

The initial isotopic composition is:

⁹⁰Zr (0.06%), ⁹¹Zr (0.01%), ⁹²Zr (99.91%), ⁹⁴Zr (0.02%), ⁹⁶Zr (0.0%).

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

given in wt %): W (95%), Ni (2.3%), Re (1.1%), Fe (1.0%), Co (0.55%), Ta (0.15%), Os (0.06%), Cu (0.05%), and Mn (0.05%). The measured volatilization rates are given in Table 10.4-VI.

In assessing the off-site doses from an air ingress accident on the ARIES-I reactor, we have assumed the worst-case tungsten release to be a 1273-K, 10-h LOCA in which the 1-h release rates are used for the full 10 hours to account for the flaking off of any protective oxide resulting from 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 10.4-VII for the case of tungsten enriched to 90% ^{183}W and irradiated for two full power years (FPY) at 1 MW/m². The release fractions of 0.03% for the tungsten and 29.2% for the rhenium are based on the volatilization rates shown in Table 10.4-VI. All other elements are assumed to be volatilized at the detection limit in the experiment. The off-site doses for other isotopic compositions and other irradiation times are shown in Table 10.4-VIII.

Table 10.4-VI.
Experimental Tungsten-Alloy Volatilization Rates

Temperature (°C)	Duration (h)	Volatilization Rates (g/h-m ²)	
		W	Re
600	20	1.4×10^{-2}	1.9×10^{-2}
800	1	0.12	5.2
	5	6.8×10^{-4}	2.4
	20	0.14	1.3
1000	1	1.2	12.4
	5	5.4×10^{-2}	4.4
	20	0.51	2.6
1200	1	36.4	11.2
Maximum release		0.2%	14%

Table 10.4-VII.
Off-Site Doses
After Accidental Release of Tungsten-Rhenium^(a)
(ARIES-I reference case)

	Dose (rem)	Dominant isotope (% dose)
Prompt dose at 1 km		
Whole body (WB)	9.34	¹⁸⁶ Re (90.2)
Bone marrow (BM)	13.96	¹⁸⁶ Re (91.9)
Lung	140.71	¹⁸⁶ Re (98.2)
Large lower intestine (LLI)	26.48	¹⁸⁶ Re (95.4)
WB early dose		
1 km	11.20	¹⁸⁶ Re (91.2)
10 km	0.76	¹⁸⁶ Re (91.4)
WB chronic dose at 1 km (chronic plus early)		
Inhalation + groundshine	18.98	¹⁸⁶ Re (77.1)
Ingestion	294.07	¹⁸⁶ Re (95.1)
Total	312.99	¹⁸⁶ Re (94.0)
WB chronic dose at 10 km (chronic plus early)		
Inhalation + groundshine	1.30	¹⁸⁶ Re (77.0)
Ingestion	20.07	¹⁸⁶ Re (95.1)
Total	21.36	¹⁸⁶ Re (94.0)
Cancers		
All organs	128.23	¹⁸⁶ Re (82.8)
WB	56.22	¹⁸⁶ Re (93.6)
Population dose		
WB ($\times 10^3$ man-rem)	356.28	¹⁸⁶ Re (93.6)
Waste disposal rating	0.10	^{178*} Hf (65.0)

^(a)For 5,790 kg of inventory and irradiated for 2 FPY at 1 MW/m².

Accident duration: 10 h at 1000 °C.

Release fractions: 0.0327% of W and 29.2% of Re.

The initial isotopic composition is:

¹⁸⁰W (0.02%), ¹⁸²W (3.23%), ¹⁸³W (90.00%), ¹⁸⁴W (3.76%), ¹⁸⁵W (3.51%).

Table 10.4-VIII.
Off-Site Doses (rem)
After Accidental Release of Tungsten-Rhenium^(a)
(including ARIES-I tungsten-air reaction)

	Natural W ^(b)	2.86% ¹⁸⁶ W ^(c)	90% ¹⁸³ W ^(d)
After 0.5 FPY at 1 MW/m²			
Early 1 km (rem)	8.71	3.62	1.14
Population dose ($\times 10^3$ man-rem)	119.230	101,696	17.496
Waste disposal rating	0.11	0.14	0.01
After 1.0 FPY at 1 MW/m²			
Early 1 km (rem)	24.06	16.88	3.45
Population dose ($\times 10^3$ man-rem)	638.121	550.529	95.376
Waste disposal rating	0.24	0.29	0.03
After 2.0 FPY at 1 MW/m²			
Early 1 km (rem)	57.63	46.81	11.2
Population dose ($\times 10^3$ man-rem)	1,789.443	1,587.206	356.280
Waste disposal rating	0.57	0.65	0.10
After 3.0 FPY at 1 MW/m²			
Early 1 km (rem)	85.57	73.15	21.61
Population dose ($\times 10^3$ man-rem)	2,754.533	2,510.282	704.716
Waste disposal rating	0.99	1.08	0.20
After 4.0 FPY at 1 MW/m²			
Early 1 km (rem)	108.53	95.59	33.79
Population dose ($\times 10^3$ man-rem)	3,558.365	3,296.640	1,110.428
Waste disposal rating	1.50	1.58	0.35
After 5.0 FPY at 1 MW/m²			
Early 1 km (rem)	126.86	114.72	47.1
Population dose ($\times 10^3$ man-rem)	4,198,197	3,965,330	1,552.344
Waste disposal rating	2.05	2.13	0.54

^(a)For divertor area of 15 m², tungsten coating thickness of 2 mm.

Accident duration: 10 h at 1000 °C.

Release fractions: 0.0327% of W (1.2 g/h-m²) and 29.2% of Re (12.4 g/h-m²), and 0.20% of others (0.0114 g/h-m²) using Ta data.

^(b)The initial isotopic composition is:

¹⁸⁰W (0.13%), ¹⁸²W (26.30%), ¹⁸³W (14.30%), ¹⁸⁴W (30.67%), ¹⁸⁶W (28.60%).

^(c)The initial isotopic composition is:

¹⁸⁰W (0.18%), ¹⁸²W (35.78%), ¹⁸³W (19.46%), ¹⁸⁴W (41.73%), ¹⁸⁶W (2.86%).

^(d)The initial isotopic composition is:

¹⁸⁰W (0.02%), ¹⁸²W (3.23%), ¹⁸³W (90.00%), ¹⁸⁴W (3.76%), ¹⁸⁶W (3.51%).

10.4.3. Tritium Inventories

The tritium inventories in the plasma-facing components are calculated by using the DIFFUSE code [13], developed by Sandia National 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 effects 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 structural material, before it has enough time to diffuse forward.

For the ARIES-I first-wall design, the tritium penetration depth is about 2 nm, and diffusion in the SiC is extremely slow, while the sputtering rate of the first wall is estimated to be 0.2 mm/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. Therefore, the total tritium inventory is only 6 g, compared to a few kg predicted without considering sputtering.

The tritium production within the Be is estimated to be 1.15 kg/FPY and the recoil deposition (implantation of energetic tritons recoiling out of the adjacent Li_2ZrO_3 particles into the Be) is 1.4 kg/FPY. Therefore, the total tritium production and deposition rates in the Be are estimated to be 2.5 kg/FPY. Recent data (Billone [14] and recent tests of the ATR reflector Be [15]) 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. Thus, the tritium inventory in Be after 1 FPY is ~ 0.64 kg. For ARIES-I, it is proposed that the entire blanket temperature be raised to above 883 K at least once per year, thereby driving out the entire tritium inventory. Therefore, the maximum tritium inventory in Be would be 0.64 kg.

The ARIES-I tritium inventories are summarized in Table 10.4-IX. The total tritium inventory is about 700 g. Releasing the entire tritium inventory, using Porter's FUSE-CRAC calculations for approximately the same conditions, results in an off-site dose of about 7.1 rem. It should be noted, however, that the tritium inventory in Be is highly uncertain and can be higher (or lower) by a factor of 3 to 5.

Table 10.4-IX.
Tritium Inventories in the ARIES-I Design

Components	Max. Tritium Inventory (g)	Comments
Fuel cycle		
Vacuum pumping	Small	Using ceramic turbomolecular pump
Helium separation	Small	Using Pd diffuser
Cryogenic distillation	50	Only removing H
First wall	6	Sputtering removes implanted T
Divertor plates	10	Using W on SiC
Blanket		
Breeding material	1	Selection of breeding material and temperature
Be	640	1. Complete tritium release at > 883 K 2. Blanket is heated to > 883 K once a year
Total	707	

10.4.4. Silicon-Carbide Structure

The first wall of the ARIES-I blanket is made of SiC-composite structural material. Table 10.4-X gives the induced radioactivities and associated latent dose ratings resulting from exposure of the SiC composite in the fusion neutron environment after 4 FPY of operation at 5 MW/m² wall loading. The principal contributors to the shutdown activity are ²⁸Al (half-life 2.24 min), ²⁴Na (15 h), ³¹Si (2.6 h), ²⁸Mg (20.9 h), and ²²Na (2.2 y), whose decay rates primarily produce the shutdown radioactivities shown in Table 10.4-X. The major contributors to the potential accident dose from the SiC-composite first wall are ²⁴Na and ²²Na.

Table 10.4-X.
Induced Radioactivities in SiC-Composite First-Wall^(a)

Time After Shutdown	Activity (Ci/cm ³)
1 min	1.26×10^2
10 min	1.31×10^1
1 h	1.20
10 h	6.47×10^{-1}
1 d	3.36×10^{-1}
10 d	2.31×10^{-4}
1 y	3.60×10^{-5}

^(a)After 4 FPY of operation at 5 MW/m².

10.4.5. Role of Impurities in Materials

The impact of impurities in the first wall and shield has been included in the afterheat analysis in Sec. 10.3, the dose calculations in Sec. 10.4, and the waste disposal evaluation in Sec. 10.5. The shield consists of SiC and B₄C, while the first wall is an SiC composite. The impurities in the SiC composite structure and shield are those assumed in the ESECOM study [2]. The impurities in the lithium zirconate were assumed to be the same as those present in lithium oxide. The impurity concentrations are given in Table 10.4-XI.

The effect of impurities on afterheat is negligible. Impurities will have impact on the prompt dose that could be received by the public under worst-case accident conditions. The dose was estimated by S. K. Ho at the University of California at Berkeley using the release fractions suggested by ESECOM [2] and the FUSEDISE code [16]. The total prompt dose due only to impurities could be as high as 21 rem.

Table 10.4-XI.
Impurity Concentration in SiC and Li₂ZrO₃

Impurity	ppm
<u>SiC^(a)</u>	
Fe	11
Co	3
<u>Li₂ZrO₃^(b)</u>	
K	370
Ca	210
Cl	100
Fe	100
Pb	80
Na	50
Al	20
Mn	20
Ni	20
Si	10
Cu	5

^(a)From ESECOM study [2].

^(b)Assumed to be the same as those in Li₂O.

10.5. 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 federal radioactive waste-disposal regulation 10CFR61 (low-level waste disposed of by near-surface, shallow-land burial) [3]. Since 10CFR61 was developed for fission reactors, most of the fusion-specific isotopes are not explicitly listed. The WDRs for each of the materials used in ARIES-I are based on an extension of the methodology of 10CFR61 to all radionuclides with $Z < 88$ by Fetter, Cheng, and Mann [17]. The specific activity limits developed in that extension are shown in Table 10.5-I. The WDR is defined as the sum over all nuclides of the ratio of the actual activity for each nuclide, α_i , to the specific activity for that nuclide for 10CFR61, Class-C low-level waste disposal, $\alpha_{i,S}$,

$$\text{WDR} = \sum_i \left(\frac{\alpha_i}{\alpha_{i,S}} \right). \quad (10.5-1)$$

The WDRs of the various components for the reference ARIES-I design are shown in Table 10.5-II. The WDR of the SiC-composite first wall is 0.12 after 4 full power years of operation at 5 MW/m²-neutron wall loading. As can be seen from Table 10.5-II, all components of ARIES-I can be disposed of as low-level waste under 10CFR61 regulations at the end of their full design life. Impurities could also affect the WDR. However, it is not expected that any of the materials that have very low waste-disposal limits (*e.g.*, Nb, Ag, Tb) will be found in the ARIES-I materials.

10.6. ECONOMIC IMPACT OF SAFETY ISSUES

Nuclear-grade materials are used in certain parts of fission power plants. Because of the excellent safety features of the ARIES-I reactor, we considered which components in ARIES-I should be designated as nuclear grade. A preliminary analysis of the applicability of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Nuclear Plant Components (hereafter referred to as the Code), has been completed. The Code provides requirements for the design of pressure-retaining components used in nuclear power plants.

Topics covered by the Code include items such as material traceability and testing; minimum strength requirements and allowable stresses; component fabrication, installation, examination, and testing; overpressure protection; and documentation and quality

Table 10.5-I.
Specific Activity Limits for Class-C Disposal
Using 10CFR61 Methodology [3]

Radionuclide	Half-Life (y)	Specific Activity Limit (Ci/m ³)
³ H	1.23×10^1	TMSA ^(a)
¹⁰ Be	1.60×10^6	3,000
¹⁴ C	5.70×10^3	700–7,000
²⁶ Al	7.20×10^5	0.09
³² Si	1.04×10^2	900–4,000
³⁶ Cl	3.01×10^5	10–100
³⁹ Ar	2.69×10^2	10,000
⁴² Ar	3.30×10^1	20,000
⁴⁰ K	1.30×10^9	1.5
⁴¹ Ca	1.03×10^5	8,000–20,000
⁴⁴ Ti	4.70×10^1	200
⁵³ Mn	3.70×10^6	TMSA
⁶⁰ Fe	1.00×10^5	0.1
⁶⁰ Co	5.30×10^3	3×10^8
⁵⁹ Ni	7.50×10^4	900
⁶³ Ni	1.00×10^2	7×10^5 – 7×10^6
⁷⁹ Se	6.50×10^4	100–1,000
⁸¹ Kr	2.10×10^5	30
⁸⁵ Kr	1.07×10^1	TMSA
⁸⁷ Rb	4.80×10^{10}	TMSA
⁹⁰ Sr	2.85×10^1	1×10^6 – 9×10^6
⁹³ Zr	1.50×10^6	2,000
⁹¹ Nb	6.80×10^2	200
⁹² Nb	3.60×10^7	0.2
^{93m} Nb	1.36×10^1	TMSA
⁹⁴ Nb	2.00×10^4	0.2

Table 10.5-I. (cont'd)

Radionuclide	Half-Life (y)	Specific Activity Limit (Ci/m ³)
⁹³ Mo	3.50×10^3	300
⁹⁷ Tc	2.60×10^6	1-10
⁹⁸ Tc	4.20×10^6	0.03-0.1
⁹⁹ Tc	2.13×10^5	0.2-2
¹⁰⁷ Pd	6.50×10^6	TMSA ^(a)
^{108m} Ag	1.27×10^2	3
^{113m} Cd	1.37×10^1	TMSA
^{121m} Sn	5.50×10^1	10,000
¹²⁶ Sn	1.00×10^5	0.1
¹²⁹ I	1.57×10^7	30
¹³⁵ Cs	3.00×10^6	TMSA
¹³⁷ Cs	3.00×10^1	50,000
¹³³ Ba	1.05×10^1	2×10^8
¹³⁷ La	6.00×10^4	30
¹³⁸ La	1.06×10^{11}	TMSA
¹⁴⁵ Pm	1.77×10^1	TMSA
¹⁴⁶ Pm	5.50	TMSA
¹⁴⁶ Sm	1.03×10^8	TMSA
¹⁴⁷ Sm	1.06×10^{10}	TMSA
¹⁵¹ Sm	9.00×10^1	TMSA
^{150m} Eu	3.60×10^1	3,000
¹⁵² Eu	1.33×10^1	300,000
¹⁵⁴ Eu	8.80	5×10^6
¹⁴⁸ Gd	9.80×10^1	$7 \times 10^5 - 7 \times 10^6$
¹⁵⁰ Gd	1.80×10^6	TMSA

Table 10.5-I. (cont'd)

Radionuclide	Half-Life (y)	Specific Activity Limit (Ci/m ³)
¹⁵⁷ Tb	1.50×10^2	1,000
¹⁵⁸ Tb	1.50×10^2	4
¹⁵⁴ Dy	2.90×10^6	TMSA ^(a)
^{166m} Ho	1.20×10^3	0.2
¹⁷⁶ Lu	3.59×10^{10}	TMSA
^{178m2} Hf	3.10×10^1	9,000
¹⁸² Hf	9.00×10^6	0.2
^{186m} Re	2.00×10^5	9
¹⁸⁷ Re	4.00×10^{10}	TMSA
¹⁹⁴ Os	6.00	TMSA
^{192m2} Ir	2.41×10^5	2
¹⁹⁰ Pt	6.00×10^{11}	TMSA
¹⁹³ Pt	5.00×10^1	9×10^6
¹⁹⁴ Hg	5.20×10^2	0.5
²⁰² Pb	5.30×10^4	0.6
²⁰⁵ Pb	1.90×10^7	TMSA
²¹⁰ Pb	2.23×10^1	$9 \times 10^6 - 8 \times 10^7$
²⁰⁷ Bi	3.22×10^1	8,000
²⁰⁸ Bi	3.68×10^5	0.09
^{210m} Bi	3.00×10^6	1
²¹⁰ Po	1.02×10^2	3,000

^(a)Theoretical maximum specific activity allowed (*i.e.*, no limit).

Table 10.5-II.
Waste Disposal Ratings of ARIES-I Material

Component	WDR
Li ₂ ZrO ₃ breeder	0.05
Tungsten divertor target	0.10
Tritium	0
SiC first wall	0.12
SiC/B ₄ C shield	<0.10

assurance requirements. Applicable components are designated as Class I, II, or III, depending on their function and importance, and experience and regulatory interpretation have provided clear guidance on this classification. Because the Code was written specifically for fission reactors, it is not directly applicable to a fusion facility.

For nonreactor nuclear facilities, including tritium and fusion facilities, U.S. regulations contained in Department of Energy (DOE) Order 6430.1A require consideration of the ASME Code requirements in the design and fabrication of safety class components. Safety class systems are: (1) Those whose failure could produce exposure consequences that would exceed the guidelines for limiting exposure to the public, which is defined as 5 mSv (500 mrem) effective dose equivalent from early exposure; and (2) those required to achieve and maintain the facility in a safe shutdown condition. The DOE Order only mentions design of safety class components to Class II rules. This indicates that design of any components to the more restrictive Class I rules would not be required for a fusion facility. Although it is not specifically spelled out in the DOE Order, it is assumed that designation of some components to the less restrictive Class III would be acceptable if justified based on risk to the public. Based on a comparison with fission-reactor component functions and classifications, a general classification scheme has been developed for ARIES-I. Components will be designed to the Code requirements for Class II (1) if they provide a primary confinement function for significant quantities of radioactive inventories whose release would produce an individual public dose over 5 mSv (500 mrem),

(2) if failure could result in a loss of confinement function of these primary boundaries, or (3) if they are required to achieve and maintain a safe shutdown condition.

Components providing support that is necessary for ensuring proper functioning of Class II components (*i.e.*, secondary cooling systems, and other components providing a secondary containment function) could be designated as Class III. Special consideration must also be given to other components, such as the magnet sets, that may not fall under the strict Code definition of a nuclear-grade component but that may require application of Code criteria in order to prevent events that could cause physical damage to Class II components.

Based on application of these criteria, the preliminary assignment of various ARIES-I components to nuclear-grade classification is summarized in Table 10.6-I. This preliminary assignment is not intended to be a final or all encompassing effort, but is intended primarily to provide preliminary guidance and to initiate a dialog on the subject to obtain a consensus among all participants including regulatory agencies. Refinement of these categories will continue as fusion designs progress and the safety implications of various component failures is better understood.

Department of Energy Order 6430.1A3 requires the use of Section III (Nuclear) of the ASME Code in nonreactor nuclear facilities, including fusion and tritium facilities. As the Code was written specifically for fission reactors, it is not directly applicable to a fusion facility such as ARIES-I and the question has been raised as to the applicability of the Code to a fusion facility. One important concern in this questioning relates to the cost of a fusion facility. Looking at overall system costs, savings of up to 60% could be realized if N-stamped components were not required. The Nuclear Regulatory Commission regulations for fission reactors (10CFR100 [18]) imply that ASME Section III, Class I requirements must be met if a component failure could result in a site boundary dose in excess of 0.25 Sv (25 rem). This is 50 times less restrictive than the DOE 6430.1A3 limit of 5 mSv, but requires more restrictive Class I rules. The relationship between these rules must be clarified as fusion advances from experiments to power plants.

Table 10.6-I.
Summary of System Classification Based on
Preliminary Interpretation of U.S. Regulations

System	Class	Comments
Vacuum vessel system	II	
Vacuum pumping system	II	
First-wall heat-transfer system	II	Portions of the system could be downgraded if alternate safety-class shutdown-cooling system is provided.
Divertor heat-transfer system	II	Portions of the system could be downgraded if alternate safety-class shutdown-cooling system is provided.
Breeder heat-transfer system	II	Same as above except that blanket-coolant radioactive inventories must be addressed.
Tritium fueling system	II	At least one primary or secondary system must be Class II.
Isotope separation system	II	At least one primary or secondary system must be Class II.
Fuel storage system	II	
Plasma diagnostics attachments	II	Classification depends on extent of isolation from the torus.
Tritium cleanup system	III	
Secondary cooling systems	III	
Magnets		Magnet systems are not directly covered by Code but may require design to Code requirements to avoid gross coil movement.

10.7. SUMMARY AND CONCLUSIONS

The ARIES-I study had three specific safety goals:

1. That the design achieve either Level of Safety Assurance 1 or 2. Level of Safety Assurance 1 indicates that the reactor is inherently safe, having an insufficient radioactive inventory and energy source to release that inventory to cause prompt fatalities among the general public. Level of Safety Assurance 2 indicates that the reactor is "large-scale passively safe," requiring no active safety systems to protect the general public as long as the large features of the plant are maintained.
2. Maintain low enough inventories of radioactive and toxic materials in the plants so that an off-site evacuation plan is unnecessary.
3. All wastes produced by the plant must be disposable as Class C waste in shallow land burial as regulated by 10CFR61 [3].

Table 10.7-I.
Summary of ARIES-I Off-Site Doses

Source	Dose (rem)	Comments
Zr	91.	Using 99.91% ^{92}Zr , 2% release
W	11.2	Using 90% ^{183}W , 10-h LOCA at 1000 °C
Tritium ^(a)	6.4	FUSECRAC analysis, $qf = 1$
Impurities ^(b)	21.	In SiC, B ₄ C, and Li ₂ ZrO ₃
Total	130.	

^(a)Based on a maximum tritium inventory in Be of 640 g (Sec. 10.4).

^(b)Using impurity concentrations from ESECOM [2].

By using isotopic tailoring of the zirconium and tungsten and by keeping the tritium inventory low, the total 1-km early dose is ~ 130 rem (Table 10.7-I). A dose of 200 rem is the minimum required to cause a prompt fatality to the most sensitive individual. Thus the design is passively safe because of its use of 99.91% ^{92}Zr in the Li_2ZrO_3 breeder and 90% ^{183}W as the tungsten coating. The doses shown assume that 2% 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 in the reactor building. The off-site doses due to an air-ingress accident and tungsten release are based on experimental data on the volatilization of tungsten and rhenium at 1000°C for 10 hours, using the 1-h release rates to account for the loss of protective oxides. The loss-of-coolant accident analyses of 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 this 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 and release fractions 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 10.7-I exceeds the 1 to 5 rem typically adopted by state and local authorities as a threshold level for evacuation. The off-site doses could be further reduced by using another breeder, such as Li_4SiO_4 or Li_2O .

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