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SAFETY IN THE ARIES-III D-³He TOKAMAK REACTOR DESIGN

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Abstract

The ARIES-III reactor study is an extensive examination of the viability of a D-³He-fueled commercial tokamak power reactor. Because neutrons are produced only through side reactions ($D+D \rightarrow ^3\text{He}+n$; and $D+D \rightarrow T+p$ followed by $D+T \rightarrow ^4\text{He}+n$), the reactor has the significant advantages of reduced activation of the first wall and shield, low afterheat and Class A or C low level waste disposal. Since no tritium is required for operation, no lithium-containing breeding blanket is necessary. A ferritic steel shield behind the first wall protects the magnets from gamma and neutron heating and from radiation damage.

The ARIES-III reactor uses an organic coolant to cool the first wall, shield and divertor. The organic coolant has a low vapor pressure at the operating temperature required for good thermal efficiency. Radiation damage requires processing the coolant to remove and crack radiolytic products that would otherwise foul cooling surfaces. The cracking process produces waste, which must be disposed of through incineration or burial. We estimated the offsite doses due to incineration at five candidate locations.

The plasma confinement requirements for a D-³He reactor are much more challenging than those for a D-T reactor. Thus, the demands on the divertor are more severe, particularly during a disruption. We explored the potential for isotopically tailoring the 4 mm tungsten layer on the divertor in order to reduce the offsite doses should a tungsten aerosol be released from the reactor after an accident.

We also modeled a loss-of-cooling accident (LOCA) in which the organic coolant was burning in order to estimate the amount of radionuclides released from the first wall. Because the maximum temperature is low, < 600°C, release fractions are small. We analyzed the disposition of the 20 g/day of tritium that is produced by D-D reactions and removed by the vacuum pumps. For our reference design, the tritium will be burned in the plasma. These results re-emphasize the need for low activation materials and advanced divertor designs, even in reactors using advanced fuels.

Introduction and Design Summary

The safety analysis of the ARIES-III commercial tokamak design has considered several of the important implications in the use of ³He as a fusion fuel. Helium-3 has a definite advantage over tritium when used in a fusion plasma in that the reaction $^3\text{He}(d,p)^4\text{He} + 18.3 \text{ MeV}$ produces no neutrons and releases all of the energy via charged particles. Neutrons are produced by the side reactions $D(d,n)^3\text{He} + 3.1 \text{ MeV}$ and

$D(d,p)T \rightarrow T(d,n)^4\text{He} + 17.6 \text{ MeV}$. As a result, the neutron activation in a D-³He reactor is one to two orders of magnitudes lower than in a similar D-T-fueled reactor.

The fusion reaction rate for D-³He is about 12-17 times smaller than the D-T reaction rate, thus requiring a higher plasma current and more thermal energy resident in the plasma. The chief safety implication of the higher plasma current is in the amount of tungsten vaporized from the divertor due to a plasma disruption. The tungsten, which is 4 mm thick on the divertor, would be evaporated to a maximum depth of 3.9 mm in the most severe disruption. We considered the use of natural tungsten and of two isotopically tailoring tungsten mixtures for the divertor coating.

About 50 percent of the tritium produced in the side reactions shown above will be burned in the plasma before reaching the divertor or first wall. The remaining half of the tritium produced can either be returned to the plasma or recovered for storage or use in D-T reactors. The ARIES team felt that the additional activation of the first wall and shield posed a smaller danger than the 10 grams per day of tritium that would accumulate and have to be stored and/or transported. Therefore the reference design for ARIES-III assumes that all of the tritium will be burned in the plasma.

In addition to the tritium produced in the plasma, small amounts of tritium are produced in the first wall, shield and coolant. About 78 Ci/day of tritium is also produced via $(n,2n)$ reactions with the thin (about 1 mm) layer of beryllium used for impurity control on the first wall. (Because of the small amount of beryllium involved, we have not considered it further in the analysis.) We assume that one third (26 Ci) of the tritium in the beryllium diffuses into the organic coolant and the remainder diffuses back into the plasma and is burned. In addition, 3.6 Ci/day of tritium is produced via beryllium reactions, all of which diffuses into coolant. Finally about 8 Ci/day of tritium is produced in the steel of the shield itself, which also diffuses into the coolant. Thus, the total tritium diffusing into the coolant is about 37 Ci/day, all of which is assumed to be eventually incinerated with the degraded coolant.

The first wall and shield in ARIES-III is low-activation ferritic steel. Since there is no need to breed tritium, the primary roles of the first wall and shield are to remove the radiant heat from the plasma and to shield the superconducting magnets from the (much reduced) neutron and gamma heating. Ferritic steel was chosen because of its established database and industrial sources.

Dose Definitions for Accidental Releases

We have considered both accidental and routine releases of radionuclides for the ARIES-III reactor. For accidental releases, the early whole body doses were computed. The early whole body dose is to the most exposed individual (MEI), who lives 100 percent of the seven days following an accident at a

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fixed location, 1 km from the radioactivity release point. We do not assume any short-term evacuation in calculating the dose. The atmospheric stability is Class F, the windspeed is 1 m/s, the inversion layer is 250 m, the release is at groundlevel and the release duration is 0.05 hour. The deposition velocities for tritium was 0.001 m/s, for the noble gases zero, and for all other nuclides 0.01 m/s. The dose coefficients, i.e. the dose in rem per Curie of a given isotope released, are those calculated by Porter¹. The doses due to routine effluent releases of small amounts of radionuclides were calculated according to assumptions described later in this paper.

Levels of Safety Assurance Definitions

In this safety analysis we have adopted the levels of safety assurance (LSA) as defined in the ESECOM study². Those definitions give a reactor a safety rating ranging from LSA=1 (the most desirable level) to LSA=4, a reactor requiring active safety systems. Those definitions, quoted from the ESECOM study are:

- In a level 1 reactor, safety is assured by passive mechanisms of release limitation no matter what the accident sequence. The radioactive inventories and material properties in such a reactor preclude a fatal release regardless of the reactor's condition.
- In a level 2 reactor, safety is assured by passive mechanisms of release limitation, as long as severe reconfiguration of large-scale geometry is avoided, and escalation to fatality-producing reconfigurations from less severe initiating events can plausibly be precluded by passive design features. In such a reactor, natural heat-transfer mechanisms suffice to keep temperatures below those needed - its radioactivity inventories and materials properties - to produce a fatal release unless large-scale geometry is badly distorted.
- In a level 3 reactor, safety is assured by passive mechanisms of release limitation as long as severe violations of small-scale geometry - such as a large break in a major coolant pipe - are avoided, and escalation to fatality-capable violations from less severe initiating events can plausibly be precluded by passive design features. In such a reactor, sufficiency of natural heat-transfer mechanisms to keep temperature low enough to avoid fatal release - given the radioactivity inventories and materials properties - can only be assured while the coolant boundary is substantially intact.
- In a level 4 reactor, there are credible initiating events that can lead to fatalities without any severe violation of small-scale or large-scale geometry, or that can only be prevented from escalating to fatality-capable boundary violations or reconfigurations by means of active safety systems.

Logic in Determining the LSA Rating

The analysis of a reactor's LSA rating follows the logic used in the ESECOM study and begins by calculating the fractional release of a reactor radioactive inventory that would result in a dose of 200 rem to the most exposed individual (MEI). If the entire inventory of the plant could be released without causing a dose of 200 rem, then the reactor clearly has an LSA=1. If the amount of radioactivity in the plant is sufficient to cause a 200 rem early dose, then one proceeds to the analysis of the stored energies and material properties. Using the possible mobilization fractions for the materials involved at the temperatures that could be reached through release of the stored energy (plasma, chemical, magnetic, etc.) one then calculates whether the early dose to the MEI is >200 rem. If

it can clearly be shown that there is insufficient energy available to cause a dose >200 rem, then the reactor has LSA=1 on the basis of energy. Mobilization fractions include volatilization as well as transport of particles out of the failed reactor. Note that, within the definition of LSA=1 above, no credit can be taken in this analysis for the reactor or its building retaining its geometry, including filter systems and the integrity of the vacuum vessel or containment building.

If the mobilization fractions still indicate that an early dose >200 rem could result, then we proceed to the analysis of conceivable accident scenarios. If early doses >200 rem result only if the large-scale geometry of the reactor and its building fail and if this failure can be prevented by passive means, then the reactor has LSA=2. If early doses >200 rem result from small-scale violation of the geometry and if that violation can be prevented by passive means, then the design has LSA=3. If active means are necessary to keep the early doses >200 rem, then LSA=4.

Radioactive Inventory

The early whole body dose at 1 km due to the release of 52 percent of the 6000 kg of tungsten would be 200 rem, if either isotopically tailored tungsten mixture, shown in Table 1, were used. If natural tungsten is used in the divertor, release of 6.5 percent of the tungsten of the divertor would result in a dose of 200 rem. [Note: Khater³ has recently developed an isotopically tailored mixture for tungsten which further reduces the 1 km early dose by a factor of about six.]

The ferritic steel shield in the ARIES-III reactor weighs about 7000 t. If 0.5 percent of the radioactive inventory in the shield is released, the early dose at 1 km is 200 rem. Thus the ARIES-III design cannot be designated LSA=1 on the basis of radioactive inventory alone.

The D-³He fuel cycle has a definite safety advantage because of its very low tritium inventory in the first wall, divertor and shield. The tritium is produced only through side reactions with the beryllium and shield and is not bred.

Stored Energy

The reactor contains 187 m³ of organic coolant with a density of 0.8 kg/l and heat of combustion of 40 MJ/kg for a stored energy of 6000 GJ. The magnetic fields contain about 165 GJ and the plasma thermal energy is about 8 GJ. The decay heat for one week is about 600 GJ. This energy is sufficient to raise the temperature of the first wall, shield, and divertor by 850° C, *if the energy is uniformly distributed over the entire shield*⁴. However, we cannot make such an assumption in an LSA=1 analysis. A fire within the plasma chamber could release the 0.5 percent of the shield which would result in an early dose of 200 rem. About one GJ will melt one tonne of steel and about five GJ will vaporize a tonne of steel. While the radioactivity in the shield is distributed throughout its 7000 t mass, the organic coolant contains enough energy to mobilize more than 0.5 percent of the steel, especially considering the potential for producing volatile oxides.

Thus we cannot clearly show that ARIES-III has LSA=1 on the basis of inventory and stored energy. We then proceed to a set of conceivable accidents to show LSA=2.

Conceivable Accidents

We considered several accidents in analyzing the safety of the ARIES-III design. Those were a disruption during which

part of the divertor tungsten is vaporized, a loss of cooling accident (LOCA) in the divertor, a LOCA in the shield and a LOCA in the shield combined with a fire outside the reactor. In these accidents, we assume that the large-scale geometry of the reactor and its building remains intact, although we take no credit for the vacuum vessel, reactor building or filters retaining radionuclides once they are released from the parent steel or tungsten.

Divertor Accidents

The most probable accident is a disruption in which 7.7 GJ of the plasma thermal energy is deposited on the plates of the bottom divertor only and is proportionally divided between the inner and outer plates. The maximum depth of evaporation, including allowance for vapor shielding, is 3.9 mm, which results in a triangular trough in the divertor 300 mm wide and 3.9 mm deep, evaporating 720 kg, i.e. 14 percent of the tungsten on the divertor. It is important to note that the evaporated material has the same elemental composition as the base material and none of the elements or their oxides are preferentially vaporized. This is not the case in the LOCA analysis discussed below.

Two isotopically tailored forms of tungsten and natural tungsten were analyzed for the dose calculations. Their compositions are shown in Table 1. The early whole body doses at the site boundary, 1 km from the groundlevel release point, are shown in the last line of the table. It is important to note that we have assumed no redeposition of the tungsten vapor or aerosol within the reactor or the reactor building, consistent with the ESECOM methodology.

Table 1.

Tungsten Release due to a Disruption

	Natural	2.86% ¹⁸⁶ W	90% ¹⁸³ W
¹⁸⁰ W	0.13 %	0.18 %	0.02 %
¹⁸² W	26.30 %	35.78 %	3.23 %
¹⁸³ W	14.30 %	19.46 %	90.00 %
¹⁸⁴ W	30.67 %	41.73 %	3.76 %
¹⁸⁶ W	28.60 %	2.86 %	3.51 %

Early Dose (rem), 1 km, Whole Body:

423.	55.7	52.6
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A second accident involving the divertor would be a fire fed by the organic coolant within the plasma chamber. We assume that the plasma chamber has been breached at two locations, allowing some flow of air to sustain the fire. The tungsten mass on the divertors is about 6000 kg and the thermal energy available from the organic coolant inventory is about 6000 GJ.

We considered a simple LOCA in which the temperature of the divertor reached 800° C, as estimated by Grotz and Khater. We found that the early dose at 1 km was 8.1 rem for the natural tungsten and 2.0 and 2.5 rem for the two isotopically tailored alloys. Note that in this analysis, we have used the volatilization data in air, which results in a release of 0.003 percent of the tungsten and 12 percent of the rhenium, which is a transmutation product of the original tungsten.

Note that these volatilization experiments show the amount of material that is evaporated from the surface of the alloy. Additional radionuclides could be transported out of the reactor as aerosols or small dust particles if the oxide layer on the surface is continually broken up by vibrations or turbulence in

the gas stream. Although the flame temperature of the organic coolant fire is above 1000° C, recent experiments with tungsten in air⁵ and steam⁶ show the formation of loosely attached oxides which could be carried out of the breached plasma chamber by the turbulence of the combustion gases. These experiments have shown that, at 1000° C to 1200° C, the tungsten would be completely oxidized in about 12-42 hours.

Shield LOCA and Fire

Steve Grotz⁷ calculated the maximum temperature in the first wall/shield due to a LOCA and due to a LOCA plus an external organic coolant fire. First, for the LOCA with no fire, he found that the maximum temperature in the first wall/shield was 504° C and that temperature peak occurred 3.7 days after the onset of the LOCA.

For the LOCA with organic coolant fire he assumed that the coolant burns with a flame temperature of 1000° C in a pool beneath the reactor for 10 hours. Furthermore, the flame was exposed to the back of the shield, which could also radiate to the surrounding containment walls. Two temperature peaks occurred: 598° C 10 hours after the LOCA, at the end of the coolant burn, and 515° C 3.5 days after the LOCA.

To assess the release of radionuclides from the first wall and shield resulting from the LOCA and fire we used data from ferritic steel activation product volatility experiments at the INEL⁸. These data give the release of the constituent elements in terms of g/m²-hr, where the area is that exposed to a slow, laminar gas flow. It is important to note that these experiments were performed for samples in air and steam, but not under combustion conditions. Assuming the average thickness of steel in the first wall and shield is 2 mm, the doses for various maximum temperatures are shown in Figure 1. The curves indicate doses at 1 km from the release point and for activation product inventories resulting from burning 50 percent and 100 percent of the tritium produced in D-D reactions. The temperatures shown were assumed to last for 10 hours and the volatilization rates were those found in one-hour experiments where no protective oxide had formed.

Figure 1. Off-site dose due to release of radionuclides from steel shield

Thus, after considering these four conceivable accidents we can state that the ARIES-III design is clearly large-scale passively safe (i.e. LSA=2), since none of the accidents result in early doses >200 rem.

Offsite Doses due to Routine Effluent Releases

The ARIES-III design uses a diphenyl organic coolant in the first wall and shield because of its low vapor pressure at the 450° C operation temperature. Because of neutron and gamma irradiation, the coolant is degraded, forming a high-boiling point fraction which would plug the 4 mm diameter cooling tubes if not removed. About 85 percent of the 'high boiler' can be reclaimed through catalytic cracking, but the remaining 15 percent is a waste which is both chemically and radioactively hazardous. We analyzed the offsite doses due to incineration of the non-reclaimable degraded coolant, releasing 47 Ci of tritium per day. We also included 1.2 Ci/yr of ^{14}C , produced by $^{14}\text{N}(n,p)^{14}\text{C}$ reactions assuming that the organic coolant is saturated with air at the 400°C operating temperature. Finally we included 18.9 mCi/yr of ^{60}Co , due to activation of natural ^{59}Co , which is used on valve facing. The cobalt content was based on the Canadian experience in operating reactors using organic coolants.

We used the EPA code AIRDOS-PC⁹, which determines compliance with the Clean Air Act per 40 CFR 61.93 and 61.94, to calculate the dose to the maximally exposed individual, 300 and 1000 m from the release point, due to routine release of the above radionuclides from typical locations near Chicago, Albuquerque, Boston, Butte, Montana and Los Angeles. The stack height was 75 m. The worst doses were in the Los Angeles area, but were only 1.3 mrem/yr, primarily due to the tritium, much less than the 10 mrem/yr EPA effluent limit.

Conclusions

Helium-3 has several definite advantages as a fusion fuel in that it produces less activation and radiation damage than D-T, thus allowing components to last the lifetime of the reactor.

We have used the Level of Safety Assurance logic developed in the ESECOM study to analyze the ARIES-III design. First, we found that the radioactive inventory in the divertor and shield were too large to declare LSA=1 on the basis of inventory alone. We determined the amount of energy available to melt and mobilize the shield and divertor, about 6000 GJ. We found that the combustion energy in the organic coolant, if not uniformly distributed through the shield, was sufficient to mobilize enough of the shield and divertor to produce an early dose at 1 km >200 rem.

We then looked at four conceivable accidents in which the large-scale geometry of the reactor was maintained. All of the conceivable accidents which have been analyzed result in early doses at 1 km of >200 rem, if isotopically tailored tungsten is used on the divertor. Therefore, we conclude that the use of D- ^3He in the ARIES-III reactor has resulted in a design that is clearly passively safe (LSA=2).

The routine releases due to incineration of the degraded organic coolant are manageable and less than the 5 mrem/year limit imposed on present nuclear facilities. The burning of all the D-D produced tritium in the plasma increases the 1 km early dose by about 30 percent.

This analysis has been limited to the operation of the reactor alone and did not consider the environmental and safety impacts of recovering and transporting the ^3He from lunar or planetary sources. A more detailed analysis of the D- ^3He fuel cycle should consider those risks.

Recommendations

Low activation ferritic steel was chosen for the first wall/shield because of its ductility, database, cost and the existence of manufacturing facilities. Other materials, such as SiC, might be used in the shield and result in lower activation. Granted, most alternative materials do not have the ductility, database, cost or manufacturing advantages held by ferritic steel. Nevertheless, those alternatives should be considered.

The choice of a divertor material is especially difficult because of the activation of tungsten and because of the severe disruption characteristics of a fusing D- ^3He plasma. Perhaps alternative gas or liquid metal film divertors could withstand the disruptions, remove the heat and still have a lower activation inventory than the present tungsten design.

References

1. Lisa J. Porter, "Upgrade of a Fusion Accident Analysis Code and its Application to a Comparative Study of Seven Fusion Reactor Designs," PFC/RR-89-10, Massachusetts Institute of Technology, June 1989.
2. John P. Holdren, D. H. Berwald, R. J. Budnitz, J. G. Crocker, J. G. Delene, R. D. Endicott, M. S. Kazimi, R. A. Krakowski, B. G. Logan and K. R. Schultz, *Report of the Senior Committee on Environmental, Safety, and Economic Aspects of Magnetic Fusion Energy*, UCRL-53766, September 25, 1989, p. 69.
3. Hesham Khater, e-mail message to ARIES team, September 23, 1991.
4. Hesham Khater, private communication, September 12, 1991.
5. G. R. Smolik, R. M. Neilson, Jr. and S. J. Piet, "Oxidation-Driven Volatility from Copper and Tungsten Alloys for Fusion Reactor Applications," Conf-981007-35, IEEE 13th Symposium on Fusion Energy, Knoxville, October 2-6, 1989.
6. G. R. Smolik, S. J. Piet and R. M. Neilson, Jr., "Predictions of Radioactive Tungsten Release from Hypothetical ITER Accidents," *Fusion Technology*, 19, (3, part B), May 1991, pp. 1342-1348.
7. Steven P. Grotz, private communication, March 19, 1991.
8. S. J. Piet, et al., *Initial Experimental Investigation of the Elemental Volatility from Steel Alloys for Fusion Safety Applications*, EGG-FSP-8459, April 1989.
9. User's Guide for AIRDOS-PC, Version 3.0, EPA 520/6-89-035, U. S. Environmental Protection Agency, Office of Radiation Programs, Las Vegas, NV, December 1989.

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