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ENVIRONMENTAL AND SAFETY ASPECTS OF "OSIRIS": A HEAVY ION BEAM DRIVEN IFE REACTOR

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ABSTRACT

A detailed safety analysis was performed for the inertial confinement fusion reactor OSIRIS. The radioactivity induced in the carbon fabric chamber concrete shield and Flibe breeder is very low allowing for their disposal at the end of the reactor life as Class A low level waste (LLW). The biological dose rate after shutdown behind the reactor biological shield shield is very low (0.11 μ rem/hr) allowing only for hands-on maintenance. A total of 91.5 Ci/day are routinely released to the environment producing an off-site dose to the maximally exposed individual (MEI) of 2.43 mrem/yr at the reactor site boundary. Only a small fraction (0.2%) of the reactor first wall would be mobilized during a loss of coolant/loss of flow accident. The decay heat generated in the concrete shield is very low such that its temperature would only increase by less than 2 degrees during such an accident. OSIRIS contains 660 tonnes of liquid Flibe as a coolant and breeder. A severe accident including a breach of the reactor building and chamber is estimated to release less than 0.5 kg of the activated Flibe to the environment. The total whole body (WB) early dose at the reactor site boundary during a pessimistic accident that includes the potential release of radioactive products from the chamber, shield and Flibe coolant would amount to 13.5 mrem. In addition, a 100% release of all the tritium steady state inventory (12.7 g) inside the reactor building during operation would result in an additional 114 mrem of off-site dose. The total of 128 mrem off-site dose produced from OSIRIS eliminates the need for using N-stamp nuclear grade components in the reactor.

I. INTRODUCTION

OSIRIS is a conceptual design study of a 1000 MW_e heavy ion beam (HIB) driven inertial fusion energy (IFE) power reactor[1]. It utilizes indirect-drive targets with two-sided illumination and auto-neutralized HIB focusing. The target yield is 432 MJ and the rep-rate is 4.6 Hz resulting in 1987 MW of fusion power. The reactor has a wetted-wall chamber that uses Flibe coolant and has a carbon fabric structure. A thin layer of the Flibe leaks through the carbon fabric and is renewed on each shot. The reactor uses a blanket which is constructed of a porous carbon fabric filled with the molten salt Flibe (67% LiF and 33% BeF_2). The reactor pool-type configuration helps contain the radioactive Flibe in a concrete pool with a double-layer steel liner. The OSIRIS chamber is surrounded by a 3 meter thick shield for protection from direct neutrons and gammas during operation. The steel-reinforced concrete shield is made of 70% concrete, 20% mild steel and 10% helium coolant.

In this paper a detailed safety analysis is presented in order to evaluate the favorable safety characteristics of the OSIRIS design.

II. CALCULATIONAL PROCEDURE

Neutron transport calculations were performed using the one-dimensional discrete ordinates neutron transport code ONEDANT[2]. The problem was modeled in spherical geometry with a point source at the center of the chamber. The source emits neutrons and gamma photons with energy spectra determined from target neutronics calculations for a generic single shell target. The neutron flux obtained from the neutron transport calculations was used in the activation calculations. The calculations were performed using the DKR-ICF[3] computer code with the ACTL activation cross section library. The DKR-ICF code allows for accurate modeling of the pulsing schedule. The pulse sequence used in the activation calculations is shown in Fig. 1. In order to achieve 75% availability, the reactor was assumed to shut down for a period of 5 days following every 25 days of operation for routine maintenance and for the last 40 days of each calendar year for an annual extended maintenance. The radioactivity generated in the reactor chamber and shield was calculated for the 40 year reactor lifetime. A second calculation was performed to determine the amount of radioactivity induced in the tantalum high-Z target material. The Ta debris which is soluble in Flibe is continuously removed from the Flibe, recycled, refabricated and reinjected in the chamber. Since Ta was assumed to go through this cycle once a week, Ta was only exposed to 1560 (number of weeks in 30 years) shots during the reactor lifetime. Even though Ta is continuously extracted from Flibe, a steady state concentration of 10 wppm of radioactive Ta in the Flibe was assumed at all times during op-

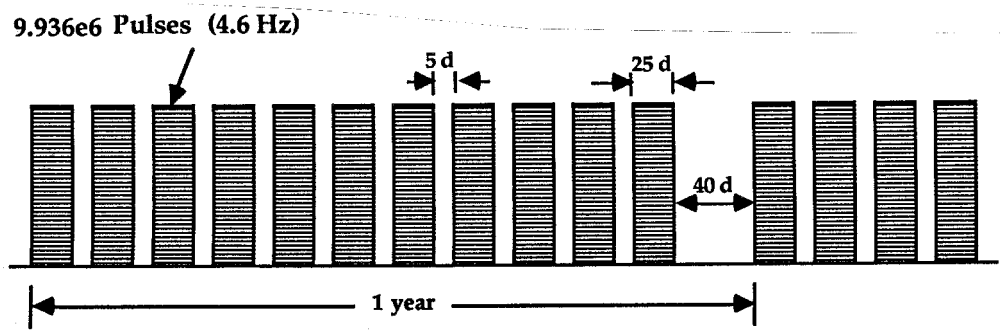

Fig. 1. Pulse sequence used in the activation calculation.

eration. A third calculation was performed for the coolant. The residence time of the Flibe coolant in the chamber is 60 seconds. However, the 660 tonnes of Flibe take 90 seconds to go through the reactor chamber. Therefore, the coolant activity was calculated to allow for the fact that Flibe spends only 67% of the time exposed to neutrons in the reactor chamber. In addition to the tantalum impurities, we used a Flibe composition which contains eleven other impurity elements[4].

The decay gamma source produced by the DKR-ICF code is used with the adjoint neutron flux to calculate the biological dose rate after shutdown using the DOSE[3] code. The dose rate calculation was performed in the space between the chamber and shield. The activation results were also utilized in the radwaste classification and the off-site dose calculations performed by the FUSCRAC3[5] code. The off-site doses are produced by the accidental release of the radioactive inventory from the reactor building assuming the worst case weather conditions. Finally, the EPA code AIRDOS-PC[6] was used to estimate the off-site dose due to the routine release of tritium.

III. ACTIVATION ANALYSIS

A. Chamber and Shield

A small amount of activity is induced in the OSIRIS chamber during the reactor lifetime. The total activity generated in the carbon fabric structure at shutdown is only 12,326 Ci. It drops to 3,512 and 364 Ci, respectively, at one day and one year after shutdown. During the first day after shutdown, the activity is dominated by radionuclides such as ^{28}Al ($T_{1/2}$ = 2.25 m), ^{37}Ar ($T_{1/2}$ = 35 d) and ^{24}Na ($T_{1/2}$ = 14.96 hr) which are induced from the impurity elements, aluminum (4 wppm), calcium (22 wppm) and sodium (10 wppm), respectively. The intermediate and long-term activities are dominated by ^{10}Be ($T_{1/2}$ = 1.6 $\times$ 10^6 yr) and ^{14}C ($T_{1/2}$ = 5730 yr), produced from the main constituent element, carbon. Figure 2 shows the activity generated in the different regions of OSIRIS as a function of time following shutdown. The activity generated in the steel-reinforced concrete shield is dominated by contribution from its steel component (20% of the shield). At shutdown, the total activity is 2.33 MCi and drops to 1.2 MCi within a day and 0.69 MCi after one year. The products of iron, ^{54}Mn ($T_{1/2}$ = 312.2 d), ^{56}Mn ($T_{1/2}$ = 2.6 hr) and ^{55}Fe ($T_{1/2}$ = 2.7 yr) are the major sources of activity present in the shield during the first year following shutdown. In the meantime, the long-term activity ($>$ 10 yr) is dominated by ^{39}Ar ($T_{1/2}$ = 269 yr), ^{63}Ni ($T_{1/2}$ = 100 yr) and ^{14}C which are all induced from impurities in both the steel and concrete used in this analysis.

Table I compares the activity, decay heat and biological hazard potential (BHP) in the chamber and shield of OSIRIS. The biological hazard potential has been calculated using the maximum permissible concentration limits in air for the different isotopes according to the Nuclear Regulatory Commission (NRC) regulations specified in 10CFR20[7]. For the most part, the decay heat and biological hazard potential are dominated by the same nuclides that dominate activity. The integrated decay heat generated in the chamber structure is insignificant, and poses no safety concern in a case of loss of coolant accident (LOCA). In the same time, the shield has an integrated decay heat of only 1.5 GJ, one month after shutdown. This amount of decay heat can not increase the shield temperature by more than 1 to 2 degrees.

B. High-Z Target Material

Tantalum is used in the OSIRIS target because of its high solubility in Flibe. The Ta was assumed to have a 4 mm inner radius and 90 μm thickness. Ta debris was assumed to be continuously removed from the Flibe, returned to the target factory to be reused in the fabrication of new targets and finally reinjected into the reactor. Hence, the time cycle assumed for this process is one week. The radioactivity calculations were performed using a total of 2.78 $\times$ 10^6 targets (847 kg of Ta) which represents the number of targets used in OSIRIS every week. As shown in Table II, the activity at shutdown is dominated by ^{180m}Ta ($T_{1/2}$ = 8.15 hr) produced from (n,2n)* reaction with ^{181}Ta. The intermediate-term activity is dominated by ^{179}Ta ($T_{1/2}$ = 1.8 yr) and ^{182}Ta ($T_{1/2}$ = 114.43 d). The only remaining source of activity, 100 years after shutdown is ^{180}Ta ($T_{1/2}$ = 1.2 $\times$ 10^{15} yr).

C. Coolant

Flibe is used in OSIRIS as the coolant and tritium breeder. The Flibe composition analyzed in this paper contains a total of eleven impurities in addition to 10 wppm of Ta. The 10 wppm of Ta represents the steady state concentration which exists in

Table I

Radioactivity After Shutdown in Different OSIRIS Regions

Time After	Activity (MCi)		Decay Heat (MW)		BHP (km^3 air)	
Shutdown	Chamber	Shield	Chamber	Shield	Chamber	Shield
0	1.23e-2	2.33	1.92e-4	2.29e-2	1.04e+3	2.63e+5
1 hour	4.88e-3	1.91	5.97e-5	1.30e-2	758.1	2.45e+5
1 day	3.51e-3	1.19	2.53e-5	2.32e-3	413	2.04e+5
1 week	2.58e-3	1.13	6.80e-6	1.65e-3	270	1.96e+5
1 month	1.78e-3	1.05	4.31e-6	1.41e-3	260.1	1.83e+5
1 year	3.64e-4	0.69	2.01e-7	5.99e-4	204.8	9.03e+4
10 years	1.06e-4	6.44e-2	3.62e-8	2.59e-5	172.8	4.64e+3
100 years	3.81e-5	3.23e-4	2.46e-8	3.00e-7	170.5	2.07e+3

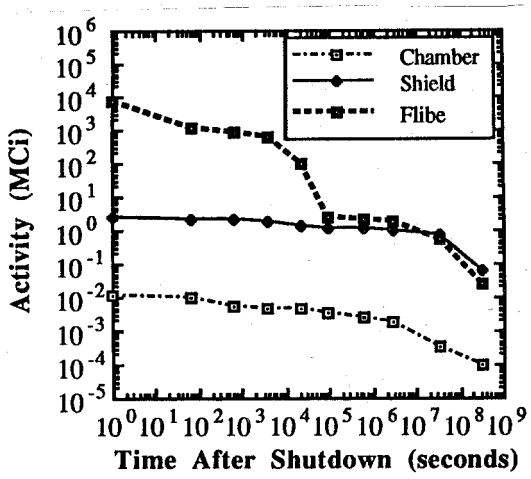

Fig. 2. Activity after shutdown in different OSIRIS regions.

the Flibe at all times and is determined by assuming a 50% extraction efficiency on 10% of the Flibe flow. Therefore, unlike the rest of the Ta inventory which is only exposed to the neutron flux only during one shot every week, this amount of Ta (6.7 kg) is also continuously exposed to the neutron flux throughout the 30 full power years.

After 30 years of irradiation (1 shot/week), the Ta composition changes to 2.5% ^{180}Ta and 97.5% ^{181}Ta replacing the original composition of 0.012% ^{180}Ta and 99.998% ^{181}Ta. Using this Ta composition with Flibe yields a shutdown activity of 7000 MCi. By far the major source of activity at shutdown is ^{18}F ($T_{1/2}$ = 1.83 hr). As shown in Fig. 2, the Flibe activity drops to only 2.55 MCi during the first day following shutdown. ^{179}Ta and ^{182}Ta produced from the Ta impurities dominate the activity during the first 5 years following shutdown. The tritium steady state inventory in the Flibe is kept at a low level of 1 g or 10,000 Ci.

IV. RADWASTE CLASSIFICATION

Table II

Tantalum Activity After 30 Years of Irradiation

Nuclide	Activity (Ci)
^{177}Lu	1.65e+4
^{181}Hf	5.46e+4
^{179}Ta	1.50e+5
^{180}Ta	4.14e-8
^{180m}Ta	3.43e+8
^{182}Ta	7.63e+5
^{182m}Ta	5.64e+6
^{183}Ta	7.17

The radwaste classifications of the OSIRIS chamber, shield and coolant were evaluated according to both the NRC 10CFR61[8] and the Fetter[9] waste disposal concentration limits (WDL). The specific activities calculated by the DKR-ICF code were used to calculate the different waste disposal ratings (WDR). The waste disposal rating is defined as the sum of the ratio of concentration of a particular isotope to the maximum allowed concentration of that isotope taken over all isotopes and for a particular class. If the calculated WDR $\leq$ 1 when Class A waste disposal limits (WDL) are used, it qualifies as Class A low level waste (LLW). If the WDR is $>$ 1 using Class A WDL but $\leq$ 1 when Class C limits are used instead, it qualifies for shallow land burial as Class C waste. A WDR of $>$ 1 implies that the radwaste does not qualify for shallow land burial.

Table III shows the waste disposal ratings (WDR) for each of the reactor regions. Both the chamber and shield would qualify as Class A low level waste. The ^{14}C ($T_{1/2}$ = 5730 yr) generated from ^{13}C (n,γ) reaction is the major contributor to the WDR of the chamber if Class A limits are used and ^{3}H ($T_{1/2}$ = 12.3 yr) produced from the boron impurities in the carbon fabric via the ^{10}B (n,2α) reaction is the other contributor. If Class C waste disposal limits are used, ^{14}C and ^{26}Al ($T_{1/2}$ = 7.3 $\times$ 10^5 yr) produced from ^{27}Al (n,2n) reaction are the major dominant nuclides if the 10CFR61 and Fetter limits are used, respectively. Sixty five percent of the Class A waste disposal rating of the shield is contributed by tritium due to the high boron content

Table III

Waste Disposal Ratings (WDR) for Different OSIRIS Regions

WDR	Chamber	Shield	Flibe
Class A (10CFR61 limits)	0.023 (0.021 ^{14}C, 1.7e-3 ^{3}H)	6.09e-3 (4.e-3 ^{3}H, 1.1e-3 ^{63}Ni)	0.23 (0.21 ^{63}Ni, 0.01 ^{14}C)
Class C (10CFR61 limits)	2.15e-3 (2.15e-3 ^{14}C)	1.01e-4 (5.3e-5 ^{94}Nb, 4.3e-5 ^{14}C)	2.3e-3 (0.001 ^{63}Ni, 0.001 ^{14}C)
Class C (Fetter)	3.00e-4 (2.4e-4 ^{26}Al, 2.9e-5 ^{14}C)	8.78e-5 (5.3e-5 ^{94}Nb, 3.0e-5 ^{26}Al)	0.048 (0.047 ^{26}Al)

of the concrete. ^{63}Ni ($T_{1/2}$ = 100 yr) produced from ^{63}Cu and ^{94}Nb ($T_{1/2}$ = 20,000 yr) produced from ^{93}Nb and ^{94}Mo are the other major contributors. Both ^{63}Ni and ^{94}Nb are generated in the steel component of the shield.

It is important to keep in mind that the waste disposal concentration limits used to calculate the waste disposal ratings of the chamber and shield are those assigned for the disposal of solid waste. As shown in Table III, the Flibe coolant could qualify for shallow land burial as Class A LLW. However, Flibe has to be in solid form before such disposal can take place and the feasibility/practicality of such a process has to be determined. Almost all of the contributors to the Flibe waste disposal rating are induced by the impurities included in the Flibe composition used in this analysis.

V. BIOLOGICAL DOSE RATE

The biological dose rate calculations were performed at selected locations behind the concrete shield and in the space between the chamber and shield. The dose rate between the chamber and shield at shutdown is 7.95 rem/hr and only drops to 191 mrem/hr one year after shutdown. The biological dose rate is dominated by ^{56}Mn ($T_{1/2}$ = 2.6 hr) and ^{54}Mn ($T_{1/2}$ = 313 day) during the first day and by ^{54}Mn and ^{55}Fe ($T_{1/2}$ = 2.7 yr) within the first few years. A limit of 2.5 mrem/hr for hands-on maintenance is used in this study assuming that maintenance personnel work for 40 hours a week and 50 weeks a year. Therefore, only remote maintenance would be feasible in the space between the chamber and shield. In the mean time, the dose rate behind the shield is quite low. The dose rate at shutdown is only 0.11 μrem/hr allowing for hands-on maintenance on the IHX behind the concrete shield.

VI. ROUTINE ATMOSPHERIC EFFLUENTS

The EPA AIRDOS-PC code was used to calculate the offsite dose to the maximally exposed individual (MEI) as a result of the routine release of tritium to the environment. The code calculates the effective dose equivalent (EDE) as mandated by 40 CFR 61.93 and 61.94 at several distances from the point of release. Dose values are computed from ingestion, inhalation, air immersion and ground surface pathways. The amount of tritium estimated for routine release is based upon the daily flow rate of tritium through each system, based upon the TSTA experience that about 1 Ci/day of tritium is released per 100 grams of tritium processed (i.e, a barrier factor of 10^6 is considered). We considered the routine release of tritium from the reactor system, containment building, fuel reprocessing facility and the target factory.

In OSIRIS, the major sources of tritium release from the reactor system are the breeder, the heavy-ion beam lines and the steam generator. The total amount of Flibe in the reactor, vacuum disengager, IHX and connecting pipes is 330 m^3, resulting in a tritium flow rate in the breeder of 1045 g/day. Hence, a well-enclosed system would only release 10 Ci/day of tritium to the environment. At the same time, a total of 156 g/day of tritium are expected to effuse into the heavy-ion beam ports where they get absorbed by cryogenic adsorption traps. Since most of these traps would be recycled every hour, only a 1.5 Ci/day of tritium is projected to be lost into the beam facility assuming a tightly controlled adsorption and degassing of these adsorbers. In addition, the tritium permeation through the steam generator is 40 Ci/day giving a total daily routine release of tritium from the reactor system of 51.5 Ci. An analysis of the containment building identified the target delivery system as the major source of tritium release. The system handles 1020 grams of tritium per day and therefore is projected to release 10 Ci/day. The third source of tritium is the fuel reprocessing system. The OSIRIS fuel reprocessing system has high tritium inventories in both the vacuum pumps of the vacuum disengager and the cryogenic distillation system. Each of the two systems handles 1024 grams of tritium per day and results in a routine release of 10 Ci/day. The last source of tritium considered in this analysis is the target factory. The factory processes some 400,000 targets a day with a total of 1020 grams of tritium and hence would be expected to routinely release about 10 Ci/day.

Assuming the release parameters listed in Table IV and using meteorological conditions at different cities, we calculated the dose expected at typical locations. The meteorological conditions at Boston, Chicago, Albuquerque and Los Angeles were used in this paper. A summary of the results is shown in Table V. The worst case occurs in the Los Angeles area but is only 2.43 mrem/yr. More than 85% of the doses at all sites are incurred via

Table IV

Routine Atmospheric Effluents Release Parameters

• Emission Information	
Year-Round Averaging	
Stack Height	75 m
Stack Diameter	30 cm
Momentum	1 m/s
• Tritium Pathways	
Reactor System	51.5 Ci/day
Containment Building	10 Ci/day
Fuel Reprocessing	20 Ci/day
Target Factory	10 Ci/day
Total (adjusted for 75% availability)	25,050 Ci/yr

Table V

Dose to the Maximally Exposed Individual (MEI)

Site	Dose (mrem/yr)	Distance (m)
Albuquerque	1.82	300
Boston	0.76	1000
Chicago	1.11	1000
Los Angeles	2.43	1000

the ingestion pathway. Notice that these results are obtained assuming a 75 meter stack height which is 2.5 times the height of the OSIRIS containment building. The rule of thumb for determining the necessary stack height is to use 2.5 times the height of the nearest tall building in order to avoid downwash of the plume into the wake of the building[10]. The estimated dose values strongly depend on the stack height. For example, using a 35 meter stack height results in an EDE of 11 mrem/yr at the Los Angeles site boundary (1 km). However, a shorter stack must be justified with appropriate analysis. The estimated doses at all the sites are within the current EPA effluent limit of 10 mrem/yr and less than the 5 mrem/yr limit adopted in ITER if the 75 m stack height is assumed.

VII. ACTIVATION PRODUCT MOBILIZATION

Another source of potential off-site dose which is of concern in OSIRIS is the dose produced by an accidental release of the radioactive inventory in the reactor containment. In general, the existence of highly radioactive products does not in itself pose a radiological hazard without a credible accident scenario for mobilizing and releasing it to the environment. Even though it is quite unlikely that any of the radioactive products would escape the building under reasonable conditions, we calculated the potential off-site doses using the ESECOM[11] methodology assuming a sequence of severe accidents. In addition, we have calculated the doses produced by the release of all the tritium contained in the containment during an accident. To account for the worst possible accident, a reactor containment failure is postulated in order to produce a significant off-site dose even though the probability of such a failure is very low.

A. Chamber

During a loss of coolant accident (LOCA) or loss of flow accident (LOFA), the chamber first wall surface would still be protected with Flibe as long as there is Flibe in the blanket. However, should the Flibe drain out altogether then as much as 2 to 3 kg of the carbon first wall would evaporate from a single shot. This is equal to the evaporation of about 0.2% of the first wall which is 0.5 cm thick. At the same time, the high Flibe vapor pressure would stop beam propagation and hence shut down the reactor. Using the worst release characteristics as defined by the ESECOM methodology (wind speed class F, 1 meter/second wind speed, etc.), we have calculated the off-site dose produced by the release of 0.2% of the first wall (FW). The whole body (WB) early dose at the site boundary (1 km) only amounts to 0.28 mrem and is dominated by ^{24}Na, ^{48}Sc and ^{54}Mn which are produced from the sodium, titanium and iron impurities in the carbon fabric.

B. Shield

The decay heat generated within the first month in the steel-reinforced concrete shield following a LOCA would only increase the shield temperature by < 2°C. Since the shield average operating temperature is much less than 500 °C, the full mobilization of the shield radioactive products is impossible. The highest temperature the shield would reach determines the release fraction of its radioactive products. Since most of the radioactive inventory is contributed by the mild steel (20% of the shield), off-site dose calculations were performed using steel experimental volatility rates.[12] Adjusted PCA volatility rates at 600 °C in dry air were used in this paper. To estimate conservative release fractions, we assumed a 10 hour LOCA in which the 1 hour release rates have been used for the full 10 hours, to account for any possible loss of iron oxide protection. At 600 °C, the whole body early dose at the site boundary is only 5.69 mrem which comes from ^{54}Mn and ^{56}Mn mainly produced from the iron in the shield.

C. High-Z Target Material

In this subsection we investigated the safety hazard posed by using tantalum in the target. As mentioned before, activated Ta debris is exposed to one shot a week before being recycled and reinjected into the reactor. Hence, there are 847 kg of Ta in 2.87×10^6 targets circulating through the reactor once a week. However, there are only about 300 targets (92 g of Ta) present inside the target injector at any moment (1 minute fuel). In the same time 6.7 kg (as 10 wppm impurity in the Flibe) of this Ta would also be exposed to further irradiation as they circulate around the chamber with the Flibe coolant. The potential whole body early dose caused by 100% release of all the radionuclides produced in the 300 targets and the 6.7 kg of Ta contained in

Table VI

Tantalum-Induced WB Early Off-Site Dose (Rem)

Nuclide	Ta in Targets (300 targets)	Ta in Flibe (6.7 kg)
^{177}Lu	2.54e-5	3.36e-4
^{181}Hf	1.40e-3	2.93e-2
^{179}Ta	4.17e-4	7.01e-3
^{180m}Ta	1.30e-2	2.49e-2
^{182}Ta	4.72e-2	1.15e+3
^{182m}Ta	2.38e-4	4.71e-4

the 660 tonnes of Flibe at the reactor shutdown is shown in Table VI. ^{182}Ta produced from ^{181}Ta via (n,γ) reaction is the most dominant isotope. As will be shown in the next subsection we only anticipate the mobilization of 0.5 kg of the Flibe in case of an accident. In addition, we assumed that the release of 1% of the Ta contained in the target inside the reactor containment (3 targets) is conservative enough. In such a case the WB early doses induced by the release of the Ta contained in the Flibe vapor and the release of 1% of the Ta contained in the target are 0.91 and 0.6 mrem, respectively.

D. Flibe

Flibe is used as a coolant and breeder in OSIRIS. The tritium inventory in the Flibe is kept very low by its continuous removal during the reactor operation. We calculated the potential off-site dose produced by the mobilization of the Flibe during an accident where a breach of the containment is postulated. Also, the 10 wppm of Ta contained in the Flibe as an impurity is included in this analysis. Following every fusion explosion, x-rays vaporize 2.78 mg/cm^2 or 14.1 μm of Flibe from the chamber wall. For a 3.5 m radius first wall, we calculated that 4.3 kg of vapor Flibe is produced per shot. A simultaneous breach in the containment and chamber would allow the cold air to flow into the chamber. The air starts cooling the Flibe vapor and hence reduces its vapor pressure. As Flibe vapor pressure falls, Flibe starts condensing rapidly. Condensed Flibe begins to form aerosol particles which in turn start falling into the hot pool in the bottom of the chamber. However, a fraction of the aerosol particles can be carried out by the hot air leaving the chamber. In the HYLIFE-II study[4], the ratio of the mobilized Flibe is estimated at about 10% of the total Flibe evaporated after each shot. Using a similar assumption, we performed the off-site dose calculation assuming that 0.5 kg of the vapor Flibe is mobilized in the form of aerosol particles. The whole body early dose at the site boundary would be 7.2 mrem. More than 85% of the dose is produced by the ^{18}F isotope. The rest of the dose is caused by ^{182}Ta produced from the target material impurities, and ^{24}Na and ^{54}Mn produced from the natural impurities in the Flibe.

The 0.5 kg of Flibe escaping the containment building contains 165 g of BeF_2. The BeF_2 is a major safety hazard because of the beryllium toxicity. Using the same assumptions as in the HYLIFE-II study, a one hour release of BeF_2 would result in its concentration at the reactor site boundary being about 1.5 μg/m^3. This value is below the level of concern as the recommended upper limits for continuous and peak exposures are 2 and 25 μg/m^3, respectively.

E. Tritium

Finally we considered the potential off-site doses produced by the accidental release of the tritium from both the reactor system and the containment building at any moment. The three major sources of tritium in the reactor system are the Flibe breeder, the graphite fabrics and the heavy ion beam lines. The tritium concentration in Flibe is 3.4 mg/m^3 and the total Flibe inventory in the reactor, vacuum disengager and IHX is 330 m^3. Consequently, the steady state tritium inventory in the Flibe salt is only 1 g. In the meantime, the graphite fibers forming the chamber first wall (3000 kg) are subjected to T_2 pressure from the tritium dissolved in the Flibe resulting in a maximum tritium inventory in these fibers of about 4 g. The two sources of tritium accumulating inside the heavy-ion beam ports are due to the continual evaporation of TF from the Flibe and the unburned target fuel produced by chamber blasts. Cryogenic adsorption traps installed along the internal surface of the beam tubes accumulate about 156 grams of tritium per day. However, most of them would be recycled every hour so that their total tritium inventory is only 6.5 g.

We identified the tritium contained in the fuel targets present in the target delivery system as the major source of potential tritium release from the containment building during an accident. The target delivery system handles 1020 grams of tritium every day, out of which 1.2 g of tritium (contained in the number of targets needed for the order of one minute of fueling) are vulnerable to any accidental release. An accident releasing 100% of the specified tritium inventory (12.7 g) would produce a whole body early dose of 114 mrem.

Table VII shows the potential off-site doses produced by simultaneous occurrence of the previous accident scenarios. The total whole body dose at the site boundary amounts only to 128 mrem which is far below the 200 rem value recommended by the ESECOM committee as a threshold for avoidance of early fatalities. A separate analysis for the accidental release of tritium from both the fuel reprocessing facility and the target factory is not included in this paper.

VIII. CONCLUSIONS

The OSIRIS reactor has distinct favorable safety characteristics. The chamber and shield qualify for near surface burial as Class A low level waste. The Flibe coolant could qualify for shallow land burial as Class A waste. The biological dose rate after shutdown behind the 3 meter biological shield of OSIRIS is only 0.11 μrem/hr allowing for hands-on maintenance. The dose from the atmospheric routine release of tritium to the maximally exposed individual is 2.43 mrem/yr which is far below the 10 mrem/yr EPA current effluent limit. The estimated off-site whole body early dose at the reactor site boundary (1 km) due to the mobilization of some of the radioactive products from

Table VII

OSIRIS Potential Off-Site Doses

	Chamber (0.2% FW)	Shield (600 °C)	Flibe (0.5 kg)	Tantalum (3 targets)	Tritium (12.7 gm)	Total
Prompt dose at 1 km (Rem)						
WB	2.70e-4	5.57e-3	7.08e-3	5.58e- 4	1.49e-2	2.83e-2
BM	2.78e-4	6.50e-3	8.57e-3	6.72e-4	5.41e- 2	7.01e-2
Lung	4.15e-4	1.29e-2	1.23e-2	3.66e-3	1.19e- 1	1.48e-1
LLI	2.36e-4	6.01e-3	5.38e-3	9.17e-4	1.85e-2	3.11e-2
WB Early Dose (Rem)						
At 1 km	2.81e-4	5.69e-3	7.20e-3	6.03e-4	1.14e-1	1.28e-1
At 10 km	1.79e-5	3.50e-4	2.41e-4	4.10e-5	2.65e-2	2.71e-2
WB Chronic Dose at 1 km (Rem)						
Inh + Grd	7.19e-4	2.55e-2	1.75e-2	4.52e-3	1.57e-1	2.05e-1
Ingestion	1.43e-3	3.45e-2	2.71e-2	1.41e-2	5.90	5.97
Total	2.15e-3	6.00e-2	4.46e-2	1.86e-2	6.06	6.18
WB Chronic Dose at 10 km (Rem)						
Inh + Grd	4.74e-5	1.71e-3	8.56e-4	3.12e-4	3.64e-2	3.93e-2
Ingestion	9.88e-5	2.37e-3	1.87e-3	9.86e-4	1.37	1.38
Total	1.46e-4	4.08e-3	2.73e-3	1.30e-3	1.40	1.42
Cancers						
Sum Organs	6.19e-4	3.79e-3	4.99e-2	2.73e-2	1.74	1.82
WB	2.23e-4	3.53e-3	8.53e-3	4.55e-3	3.54	3.56
Population Dose (Man-Rem)						
WB	1.41	22.33	54.1	28.93	2.24e+4	2.25e+4

OSIRIS during a highly unlikely sequence of simultaneous accident scenarios is 128 mrem. This dose is far below the 5 rem level where evacuation plans are needed. In the meantime, this very low off-site dose eliminates the need for N-stamp nuclear grade reactor components which is only required if the dose exceeds the 25 rem limit.

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