CONTENTS

5.0 ENVIRONMENTAL AND SAFETY ASSESSMENT

5.1 INTRODUCTION

A strong emphasis has been given to the environment and safety issues in both the SOMBRERO and Osiris reactor designs. Carbon/carbon composite has been used as the chamber material to avoid a high level of induced radioactivity in both reactor structures. Similarly, the use of Li_2O in SOMBRERO and Flibe in Osiris as coolant and breeder materials eliminates the hazard posed by the energy-producing chemical reactions usually associated with the use of lithium and, hence, reduces the risk of mobilizing the radioactive inventory present in both reactors. The methodology used in this analysis does not depend on the probability of accident-initiating scenarios; we have adopted the principle of considering the worst possible accident scenario. To evaluate the possible radiological hazard to the public, we used a two-step approach in calculating the possible off-site dose. The first step in our approach is the identification of the sources and locations of the radioactive materials inside the reactor building. However, since the existence of radioactivity does not in itself represent a safety hazard, the second step in our approach was to consider a set of pessimistic, but rather credible, accident scenarios for mobilizing and releasing the radioactive material.

A detailed activation analysis was performed in order to calculate all possible radioactive inventories for each of the two reactor designs. Results of the radioactivity calculations are used to evaluate the following:

1. The biological dose rate at different locations inside the reactor building following shutdown to assess the feasibility of hands-on maintenance
2. The radwaste classification for each region of the reactor
3. The maximum public dose from routine operational effluents
4. The off-site doses from accidental release of the radioactive inventories present in the reactor building, target factory, and fuel reprocessing facility.

5.2 SAFETY DESIGN GOALS

The main safety goals pursued for both the SOMBRERO and Osiris reactor designs are:

1. Limiting the need for remote maintenance and allowing for hands-on maintenance by reducing the biological dose rate following shutdown below 2.5 mrem/hr by increasing the biological shield where it is possible.

2. Disposing the reactor structure and coolant as either Class A or Class C low-level wastes as regulated by the Nuclear Regulatory Commission's (NRC) 10CFR61 guide lines.
3. Limiting the public dose to the maximally exposed individual (MEI) from routine operational effluents to less than 5 mrem/yr.
4. Limiting the whole-body (WB) early dose during a conservative accident scenario to 25 rem, which was recommended for this study by the study guidelines. The low off-site dose will allow for the avoidance of early fatalities in case of an accidental release of radioactivity.
5. Eliminating the need for the use of N-Stamp nuclear grade components.

5.3 OFF-SITE DOSE DEFINITIONS

Off-site dose is used to predict the degree of radiological hazard to the public posed by any routine or accidental release of radioactivity from the reactor. However, the health effects to the various human organs are dependent on both the length and method of exposure. While dose from external exposure (cloudshine and groundshine) is only limited to the length of the exposure, decay of the radionuclides inside the irradiated body (from inhalation and ingestion) leads to a continuous internal exposure. In this chapter we used the following of dose definitions:

Prompt Dose at 1 km: This is the dose delivered to a particular organ at 1 km from the release. It includes the dose from cloudshine during plume passage, the dose from 7 days of groundshine, and the dose commitment over an organ-dependent, critical acute time period from inhalation during plume passage.

WB:	Whole body, $t_{acute} = 2$ days
BM:	Bone marrow, $t_{acute} = 7$ days
Lung:	Lung, tacute = 1 year
LLI:	Lower large intestine, $t_{acute} = 7$ days

WB Early Dose: The whole body early dose is the dose from initial exposure: cloudshine during plume passage, 7 days of groundshine, plus the 50-year dose commitment from radioactivity inhaled during plume passage.

WB Chronic Dose at 1 and 10 km: These are the whole body doses at 1 and 10 km from the release due to both initial and chronic (50-year) exposures.

Inh + grd: Chronic exposure considers the 50-year groundshine exposure plus the 50-year dose commitment from inhaled resuspended radioactivity.

Ing: Chronic exposure considers the ingestion pathway only.

Total: Chronic exposure considers all three pathways: groundshine, resuspension, and ingestion.

Cancers: This is the total number of cancers in a 50-mile radius from initial and chronic exposure.

Sum Organs: The number of cancers where the body is treated as a sum of individual organs and calculations are based on organ-specific dose factors and dose responses.

WB: The number of cancers where the body is treated as a single organ and the whole body dose conversion factors and dose response are used.

Population Dose WB (Man-Rem): This is the total whole body man-rem due to both initial exposure plus an 80-year chronic exposure to the whole body.

5.4 CALCULATIONAL PROCEDURE

Neutron transport calculations have been performed using the one-dimensional discrete ordinates neutron transport code ONEDANT.[5.1] The analysis uses a P_3 approximation for the scattering cross sections and S_8 angular quadrature set. The problem has been 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 has been used in the activation calculations. The calculations have been performed using the computer code DKR-ICF[5.2] with the ACTL[5.3] 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. 5.1. In order to achieve 75% availability, the reactor has been assumed to shutdown 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 has been calculated for the 40-year reactor life-time.

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[5.2] code. The dose rate calculations have been performed at different locations inside the reactor building. The activation results have also been utilized in the radwaste classification and the off-site dose calculations

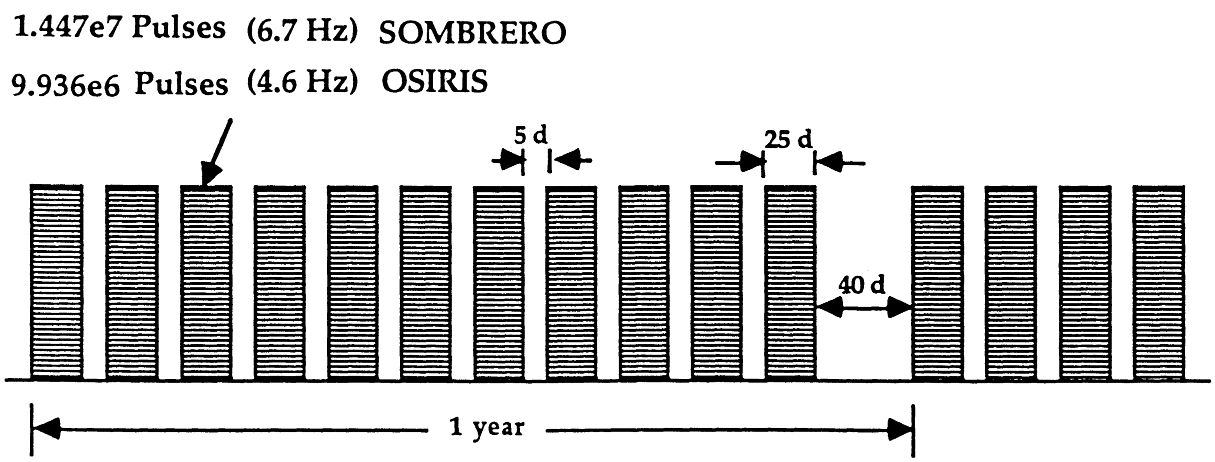

Fig. 5.1. Pulse sequence used in activation calculations.

performed by the FUSCRAC3 code.[5.4] 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[5.5] has been used to estimate the off-site dose due to the routine release of tritium.

5.5 SOMBRERO SAFETY ANALYSIS

Activation and safety analysis has been performed for the chamber, shield, and Li_2O coolant of SOMBRERO. The reactor chamber is made of a low activation carbon/carbon composite and the blanket consists of a moving bed of solid Li_2O granules (90% density factor) flowing through the chamber by gravity. The particles are transported in a fluidized state by helium gas at 0.2 MPa. There are 60 laser beams in near symmetric distribution. The laser energy is 3.4 MJ, the gain is 118, and the rep-rate is 6.7 Hz. The reactor first wall is 1 cm thick and is made of 100% graphite. To maximize the tritium breeding ratio (TBR) and overall energy multiplication M_o, the blanket is divided to three different regions. The first region is 19 cm thick and consists of 52.4% Li_2O and 3% C. Each of the second and third regions is 40 cm thick. However, the second region consists of 43.2% Li_2O and 20% C while the third region is made of 27% Li_2O and 50% C. The increase in the graphite fraction in the blanket with distance from the target obviates the need for a separate reflector behind the blanket. The chamber is surrounded by a 170-cm thick shield to allow for hands-on maintenance at selected locations behind it. The steel-reinforced concrete shield is made of 70% concrete, 20% mild steel and 10% helium coolant. The radial build used in the calculations is shown in Fig. 5.2.

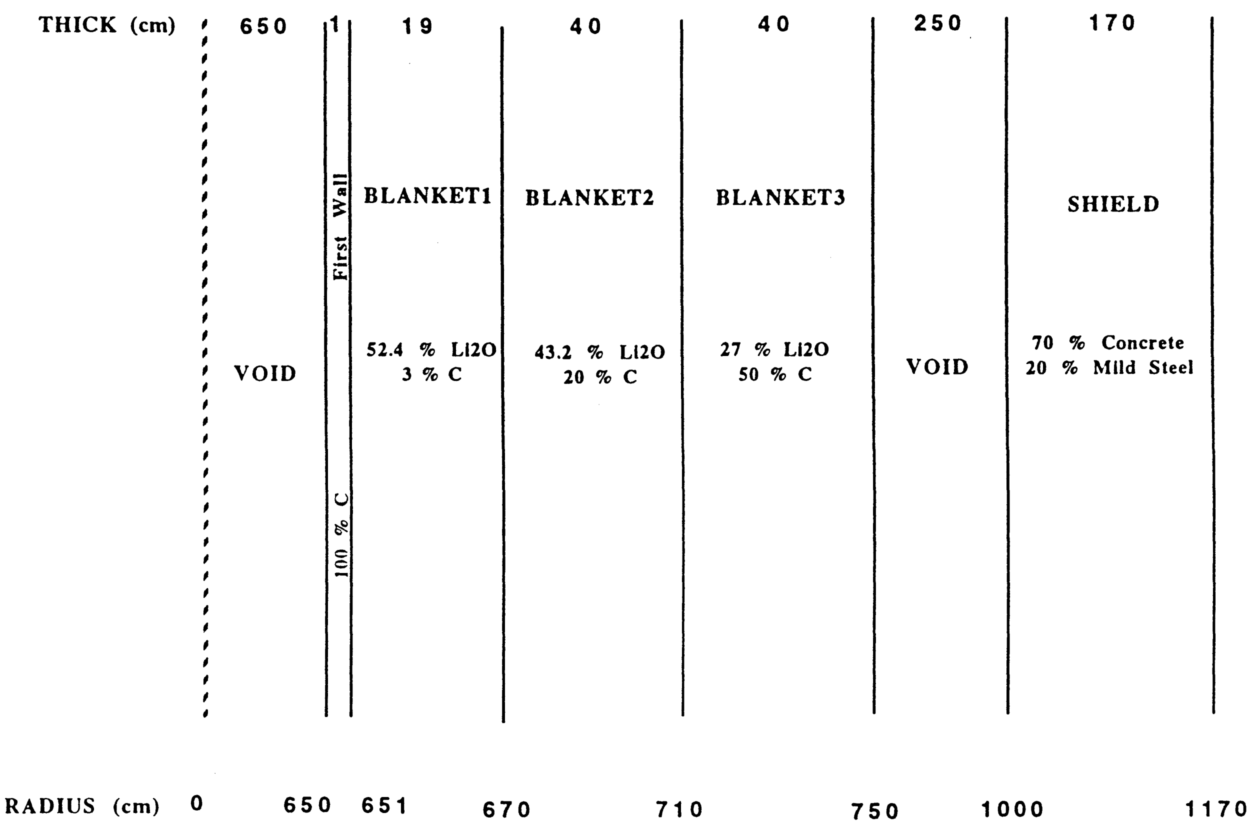

Fig. 5.2. Radial build used in activation calculations.

5.5.1 Activation Analysis

The radioactivity generated in the reactor carbon/carbon composite chamber and steel-reinforced shield was calculated for the 40-year reactor life-time with 75% availability. The elemental composition of the chamber and shield materials are shown in Table 5.1. A separate calculation was performed for the coolant. The composition of the Li_2O coolant used in this analysis is given in Table 5.2. The residence time of the Li_2O coolant in the chamber is 100 seconds. The total inventory of Li_2O takes 300 seconds to go through the reactor chamber. Therefore, the coolant activity has been calculated to allow for the fact that the Li_2O granules spend only 33% of the time exposed to neutrons in the reactor chamber. The total activity generated in the different regions of SOMBRERO as a function of time following shutdown is shown in Fig. 5.3. The total activity in the chamber at shutdown is 0.054 MCi and drops to 0.016 MCi in one day and 0.0015 MCi in one year. Most of the steel-reinforced concrete shield activity is due to

Table 5.1. Elemental Composition of the Carbon/Carbon Composite, Concrete, and Mild Steel Used in the Calculations.

Nuclide	Graphite	Concrete	Steel
	(wppm unless wt% indicated)		
H	----	0.56 wt%	----
B	2	1.04 wt%	----
C	99.999 wt%	----	200
N	----	----	70
O	----	33.8 wt%	----
F	----	0.23 wt%	----
Na	10	1.21 wt%	----
Mg	1	0.23 wt%	----
Al	4	0.64 wt%	----
Si	21	3.31 wt%	0.31 wt%
P	----	----	160
S	1	9.15 wt%	400
K	----	100	4
Ca	22	6.26 wt%	----
Ti	1	----	----
V	1	----	----
Mn	----	200	0.52 wt%
Fe	3	2.19 wt%	98.747 wt%
Ni	----	1.32	60
Cu	----	0.22	0.16 wt%
Zn	----	0.66 wt%	----
Ba	----	40.13 wt%	2
Nb	----	200	1
Mo	----	800	3
Pb	7	----	----

Table 5.2. Elemental Composition of Li_2O

Nuclide	wppm unless wt% indicated
^6Li	39.61991 wt%
^7Li	4.86447 wt%
Be	1
B	1
C	100
N	2
O	55.46493 wt%
F	0.1
Na	60
Mg	10
Al	50
Si	50
P	1
S	0.1
Cl	10
K	20
Ca	100
Ti	10
V	1
Cr	1
Mn	1
Fe	50
Co	0.2
Ni	10
Cu	10
Zn	10
As	0.1
Br	0.1
Zr	1
Mo	0.1
Cd	0.1
Sn	1
Sb	1
Ba	5
Pb	0.1

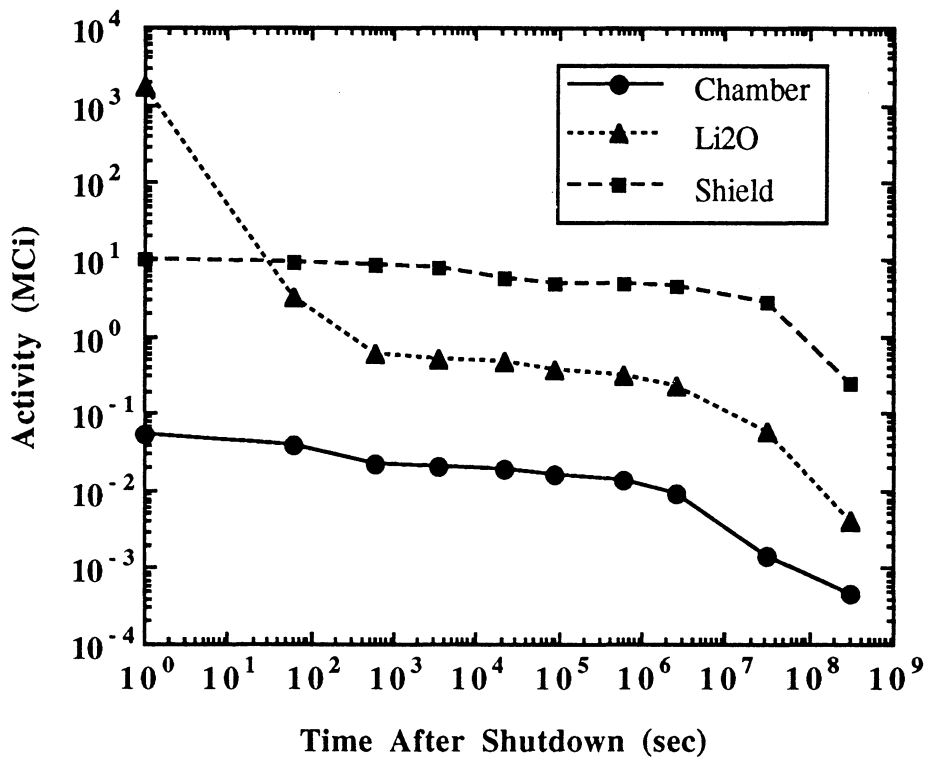

Fig. 5.3. Activity after shutdown in different regions of SOMBRERO.

its steel component. At shutdown, the total activity amounts to 10.12 MCi and drops to 4.95 MCi within a day and 2.68 MCi within a year. ^{6}He ($T_{1/2}$ = 807 ms) produced from $^{6}Li(n,p)$ and $^{7}Li(n,d)$; and ^{16}N ($T_{1/2}$ = 7.13 s) produced from $^{16}O(n,p)$ are the major contributors to the high coolant activity at shutdown. The activity of Li_2O drops from 1700 MCi to 0.38 MCi during the first day following shutdown. Table 5.3 shows the dominant contributors to the activity generated in SOMBRERO during different time periods following shutdown. Table 5.4 compares the activity, decay heat, and biological hazard potential (BHP) in the chamber and shield of SOMBRERO. 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.[5.6]

The temporal variation of the decay heat and BHP after shutdown is similar to that of the activity. In general, the decay heat and biological hazard potential are dominated for the most part by the same nuclides shown in Table 5.3. One value useful for predicting the thermal response of the shield to a loss of coolant accident is the integrated decay heat. Fig. 5.4 shows the integrated decay heat generated during the first two months following shutdown in the different regions of SOMBRERO. The integrated decay heat generated in the reactor shield during the first two months following shutdown is 23 GJ, which will only increase the shield temperature by less than 3 °C.

Table 5.3. Dominant Contributors to Radioactivity in SOMBRERO

Time After Shutdown	Chamber	Shield	Li_2O
< 1 day	^{28}Al, ^{37}Ar, ^{24}Na	^{56}Mn, ^{54}Mn, ^{55}Fe	^{6}He, ^{16}N, ^{37}Ar
1 day - 1 yr	^{3}H, ^{37}Ar, ^{55}Fe	^{55}Fe, ^{54}Mn, ^{37}Ar	^{55}Fe, ^{35}S, ^{37}Ar
1 yr - 10 yr	^{3}H, ^{55}Fe, ^{10}Be	^{55}Fe, H^{3}, ^{54}Mn	^{55}Fe, ^{14}C, ^{39}Ar
> 10 yr	^{10}Be, ^{14}C, ^{39}Ar	^{39}Ar, ^{63}Ni, ^{14}C	^{14}C, ^{39}Ar, ^{63}Ni

Table 5.4. Radioactivity After Shutdown in the Different Regions of SOMBRERO

Time After	Activity (MCi)		Decay Heat (MW)		BHP (km^3 air)	
Shutdown	Chamber	Shield	Chamber	Shield	Chamber	Shield
0	5.36e-2	10.12	7.69e-4	1.10e-1	4.12e+3	1.31e+6
1 hour	2.02e-2	7.98	1.77e-4	5.55e-2	2.84e+3	1.23e+6
1 day	1.63e-2	4.95	8.64e-5	1.09e-2	1.61e+3	1.06e+6
1 week	1.31e-2	4.69	3.60e-5	8.24e-3	1.21e+3	1.02e+6
1 month	8.87e-3	4.29	2.28e-5	7.05e-3	1.17e+3	9.61e+5
1 year	1.48e-3	2.68	8.89e-7	2.89e-3	1.00e+3	4.72e+5
10 years	4.67e-4	0.24	1.57e-7	9.56e-5	8.78e+2	2.18e+4
100 years	1.46e-4	1.70e-3	1.10e-7	1.49e-6	8.69e+2	1.12e+4

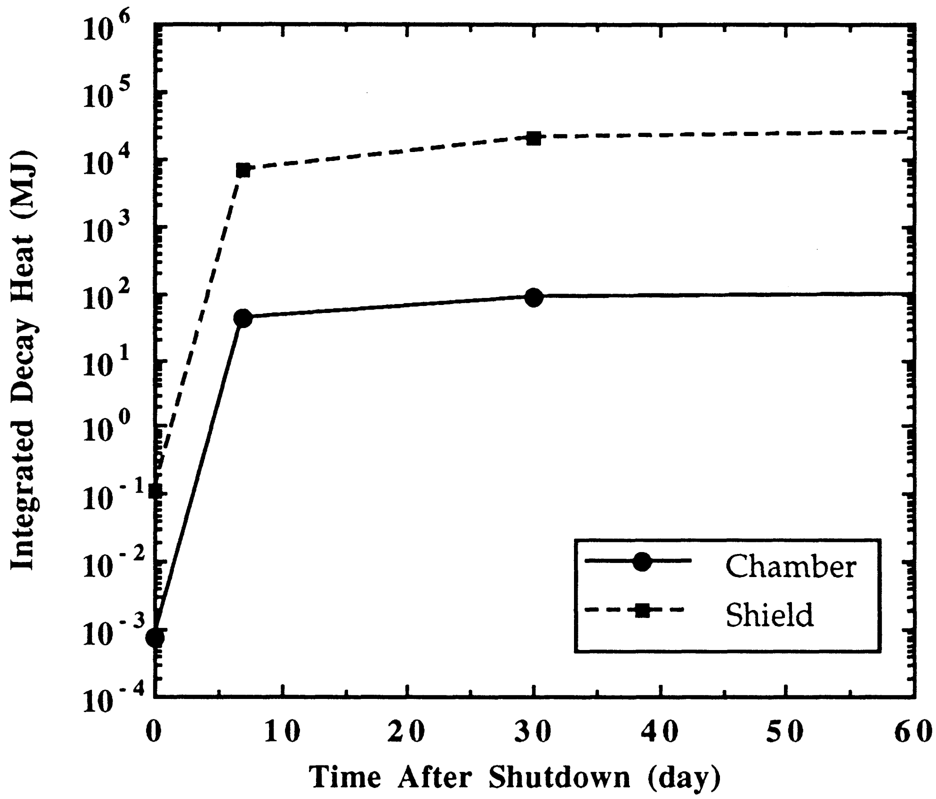

Fig. 5.4. Integrated decay heat generated in SOMBRERO.

5.5.2 Maintenance

Biological dose rate calculations have been performed at selected locations behind the concrete shield and in the space between the chamber and shield. Fig. 5.5 shows the calculated dose rate behind the shield as a function of time following shutdown. ^{56}Mn ($T_{1/2}$ = 2.6 hr) and ^{54}Mn ($T_{1/2}$ = 313 day) dominate the biological dose rate during the first day. The dose is dominated by ^{54}Mn and ^{55}Fe ($T_{1/2}$ = 2.7 yr) within the first few years. As shown in the figure, the dose rate drops to 1.6 mrem/hr after one day following shutdown. A limit of 2.5 mrem/hr for hands-on maintenance is used in this analysis assuming that the maintenance personnel work for 40 hours a week and 50 weeks a year. Hence, hands-on maintenance will definitely be allowed on the intermediate heat exchangers (IHX) behind the concrete shield within a day following shutdown. The dose rate between the chamber and shield is quite high. As shown in Fig. 5.6, the dose rate at shutdown is 25 rem/hr and drops to 1.1 rem/hr one year after shutdown. Therefore, only remote maintenance is feasible in the space between the chamber and shield of SOMBRERO.

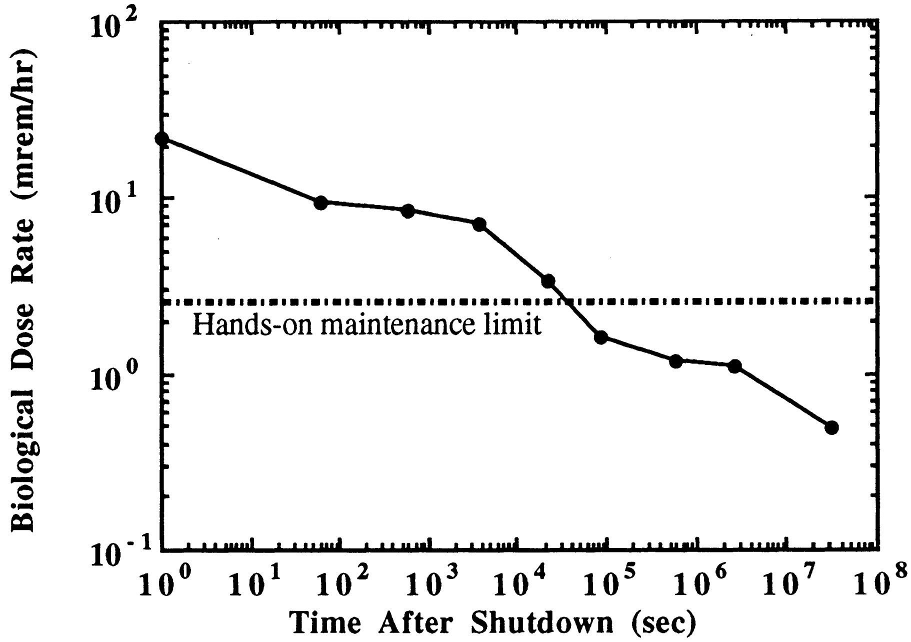

Fig. 5.5. Biological dose rate in the IHX enclosures.

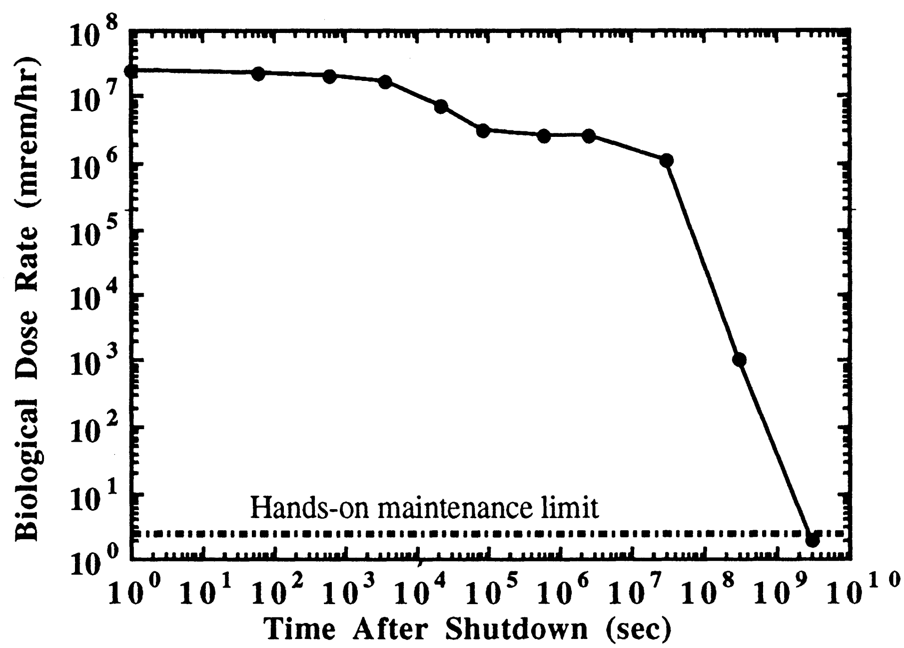

Fig. 5.6. Biological dose rate in the space between the chamber and shield.

5.5.3 Radwaste Classification

The waste disposal ratings for SOMBRERO have been evaluated according to both the NRC 10CFR61[5.7] and Fetter[5.8] waste disposal concentration limits (WDL). The 10CFR61 regulations assume that the waste disposal site will be under administrative control for 100 years. The dose at the site to an inadvertent intruder after the 100 years is limited to less than 500 mrem/year. The waste disposal rating (WDR) is defined as the sum of the ratio of the concentration of a particular isotope to the maximum allowed concentration of that isotope taken over all isotopes and for particular class. If the calculated WDR ≤ 1 when Class A limits are used, the radwaste should qualify for Class A segregated waste. The major hazard of this class of waste is to individuals who are responsible for handling it. Such waste is not considered to be a hazard following the loss of institutional control of the disposal site. If the WDR is > 1 when Class A WDL are used but ≤ 1 when Class C limits are used, the waste is termed Class C intruder waste. It must be packaged and buried such that it will not pose a hazard to an inadvertent intruder after the 100-year institutional period is over. Class C waste is assumed to be stable for 500 years. Using Class C limits, a WDR > 1 implies that the radwaste does not qualify for shallow land burial. Fetter developed a modified version of the NRC's intruder model to calculate waste disposal limits for a wider range of long-lived radionuclides, which are of more interest for fusion researchers than the few that currently exist in the current 10CFR61 regulations. Fetter's model included more accurate transfer coefficients and dose conversion factors. However, while the NRC model limits the whole body dose to 500 mrem or the dose to any single organ (one of seven body organs) to 1.5 rem, Fetter limits are based on the maximum dose to the whole body only.

The specific activities calculated for the different radionuclides have been used to evaluate the radwaste classification of the SOMBRERO's chamber, shield, and Li_2O solid breeder. Table 5.5 shows the waste disposal ratings (WDR) for each of the reactor regions in the compacted form. Compacted values correspond to crushing the solid waste before disposal. On the other hand, non compacted values are based on averaging over the total volume of a particular region implying that internal voids will be filled with concrete before disposal. Both the chamber and shield would easily qualify as Class A low level waste. ^{14}C ($T_{1/2}$ = 5730 yr) generated from $^{14}C(n,\gamma)$ reaction is the major contributor to the WDR of the graphite chamber if Class A limits were used. ^{3}H ($T_{1/2}$ = 12.3 yr) produced from the boron impurities in the graphite via the $^{10}B(n,2\alpha)$ reaction is a distant second. If Class C waste disposal limits were 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 were used, respectively. About 70% of the Class A waste disposal rating of the shield is contributed by tritium due to the high boron content 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

Table 5.5. Waste Disposal Ratings of the Different Regions of SOMBRERO

WDR	Chamber	Shield	Li_2O
Class A (10CFR61 limits)	0.043 (0.038 ^{14}C, 5.5e-3 ^{3}H)	0.058 (0.041 ^{3}H, 0.01 ^{63}Ni)	4.07 (4.07 ^{14}C)
Class C (10CFR61 limits)	3.76e-3 (3.76e-3 ^{14}C)	7.57e-4 (4.5e-4 ^{94}Nb, 2.6e-4 ^{14}C)	0.4 (0.4 ^{14}C)
Class C (Fetter)	7.05e-4 (5.6e-4 ^{26}Al, 8.8e-5 ^{10}Be)	8.17e-4 (4.5e-4 ^{94}Nb, 3.2e-4 ^{26}Al)	0.077 (0.053 ^{14}C)

are the other major contributors. Both ^{63}Ni and ^{94}Nb are generated in the steel component of the shield.

As shown in Table 5.5, the Li_2O granules would not qualify for Class A LLW even after extracting all the tritium out of the granules due to the high ^{14}C activity. Unlike the graphite chamber, this ^{14}C is generated by $^{17}O(n,\alpha)$ reaction. Using Class C waste disposal limits, the Li_2O would qualify for shallow land burial. It is important to keep in mind that this calculation is based on the Li_2O remaining for the whole 30 full power years (FPY). However, Li_2O may qualify for Class A LLW if it is replaced at least four times during the reactor life.

5.5.4 Routine Atmospheric Effluents

The radiological dose to the population in the vicinity of the reactor site due to the routine release of tritium has been estimated by using the EPA AIRDOS-PC code. The code calculates the effective dose equivalent (EDE) as mandated by 40 CFR 61.93 and 61.94 to the maximally exposed individual (MEI) and at several distances from the point of release. Dose values are computed from ingestion, inhalation, air immersion, and ground surface pathways. The routine releases from the several processing systems were based upon the quantity of tritium processed per day and followed recent experience at TSTA, which indicated that a barrier factor of 10^6 is an acceptable one. As discussed in Sec. 3.2.6.4, we considered the routine release of tritium from the reactor system, reactor building, fuel reprocessing facility, and the target factory.

The three sources of tritium release from the reactor system are the Li_2O breeder, the helium circuits, and the steam generator. Under routine daily operation, each of the breeder and helium circuits processes 550 grams of tritium and is expected to release 5.5 Ci/day. In addition, the tritium permeation through the steam generator is 15 Ci/day, giving a total daily routine release of tritium from the reactor system of 26 Ci. A separate examination of the reactor building showed that each day both the building atmosphere of Xe and the target injector system handle 900 and 1400 grams of tritium, respectively. Hence, these two systems are also expected to release the sum of 23 Ci/day. The fuel reprocessing system has high tritium inventories in both the desiccant beds and the cryogenic distillation system. Each of the two systems handles 1500 grams of tritium per day and result in a routine release of 15 Ci/day. Finally, a 14 Ci/day of tritium are released from the target factory as it processes about 580,000 targets.

Assuming the release parameters listed in Table 5.6 and using meteorological conditions at different cities, we calculated the dose expected at typical locations near Boston, Chicago, Albuquerque, and Los Angeles.

Table 5.6. Routine Atmospheric Release Parameters

Site Information:	
Locations:	Albuquerque Boston Chicago Los Angeles
Temperature:	15 °C
Rainfall:	75 cm/yr
Emission Information:	
Year-Round Averaging	
Stack Height:	125 m
Stack Diameter:	30 cm
Momentum:	1 m/s
Tritium Pathways:	
Reactor System:	26 Ci/day
Reactor Building:	23 Ci/day
Fuel Reprocessing:	30 Ci/day
Target Factory:	14 Ci/day
Total (adjusted for 75% availability):	25,460 Ci/yr

A summary of the results is shown in Table 5.7. The worst dose was in the Albuquerque area, but was only 0.93 mrem/yr. More than 85% of the doses at all sites are incurred via the ingestion pathway. The estimated doses at all sites are far below the current EPA effluent limit of 10 mrem/yr and less than the 5 mrem/yr limit adopted by ITER. It is important to keep in mind that the estimated doses depend strongly on the stack height. For example, using a 30-meter stack height results in an EDE of 18 mrem/yr at the site boundary (1 km) if the Los Angeles meteorological conditions were used. Actually, 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

Table 5.7. Dose to the Maximally Exposed Individual (MEI)

Site	Dose (mrem/yr)	Distance (m)
Albuquerque	0.93	1000
Boston	0.23	3000
Chicago	0.36	1000
Los Angeles	0.69	3000

the plume into the wake of the building.[5.9] A shorter stack must be justified with appropriate analysis. If one were to apply the rule of thumb to SOMBRERO, the stack would be on the order of 300 m. The EDE values calculated at all sites would be one to two orders of magnitude lower than those presented in Table 5.7.

5.5.5 Accident Analysis

An accidental release of the radioactivity from the reactor building is another potential source of off-site doses. In this section we calculated the potential off-site doses using the ESECOM[5.10] methodology due to the release of some of the radioactive inventory of the chamber, shield and Li_2O granules. In addition, we calculated the doses produced by the release of all the tritium contained in the reactor building during an accident. To account for the worst possible accident, a containment failure is postulated in order to produce significant off-site doses even though the probability of such a failure is very low.

5.5.5.1 Chamber and Shield

During a loss of coolant accident (LOCA) or loss of flow accident (LOFA), the amount of evaporated graphite would not exceed 50 kg, which is equivalent to about 0.44% of the 1 cm first wall. This amount of evaporated graphite will increase the carbon partial pressure in the reactor building by one torr. The higher carbon vapor pressure would prevent the laser beam from propagating to the target and hence shutdown the reactor. Using the worst release characteristics as defined by the ESECOM methodology (Table 5.8), we calculated the off-site doses produced by the release of 0.44% of the graphite first wall (FW). The whole body (WB) early dose at the site boundary (1 km) only amounts to 1.31 mrem. The dose is dominated by radionuclides produced from the graphite impurities. As shown in Table 5.9, ^{24}Na, ^{48}Sc, and ^{54}Mn are the major contributors to the off-site dose.

Table 5.8. Activation Products Release Characteristics

Pasquill Stability Class	F
Wind Speed	1 m/s
Inversion Layer Height	250 m
Deposition Velocity	0.01 m/s
Duration of Release	0.05 hr
Population Density	50 person/km^2
Ground Level Release	
Site Boundary	1 km and 10 km
Initial Plume Dimensions	
Sigma-Y	100 m
Sigma-Z	50 m
Percentage of Land	
Crop Farming	15 %
Milk/Meat Products	15 %
Groundshine Shielding	
Prompt Dose	0.70
Chronic Dose	0.33

Table 5.9. SOMBRERO's WB Early Dose Dominant Nuclides

Chamber	Li_2O	Shield
^{24}Na ($T_{1/2}$ = 14.96 h)	^{24}Na ($T_{1/2}$ = 14.96 h)	^{54}Mn ($T_{1/2}$ = 312 d)
^{48}Sc ($T_{1/2}$ = 43.7 h)	^{60}Co ($T_{1/2}$ = 5.27 yr)	^{56}Mn ($T_{1/2}$ = 2.6 h)
^{54}Mn ($T_{1/2}$ = 312 d)	^{58}Co ($T_{1/2}$ = 70.88 d)	^{64}Cu ($T_{1/2}$ = 12.7 h)
^{46}Sc ($T_{1/2}$ = 83.81 d)	^{65}Zn ($T_{1/2}$ = 243.8 d)	^{59}Fe ($T_{1/2}$ = 44.5 d)

The decay heat generated in the steel-reinforced concrete shield is very low. The decay heat generated within the first two months following a LOCA would only increase the shield temperature by < 3 °C. Since the shield average operating temperature is 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 have been performed using steel experimental volatility rates.[5.11] Adjusted PCA volatility rates at 600 °C in dry air were used in this analysis. 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 24.7 mrem. Most of the dose is produced by the manganese isotopes, ^{54}Mn and ^{56}Mn. Even at 1000 °C, the shield would only produce a WB early dose of 167 mrem.

5.5.5.2 The Li_2O Solid Breeder

SOMBRERO's blanket consists of a moving bed of solid Li_2O particles flowing through the chamber by gravity. Tritium is continually extracted from the Li_2O granules by helium gas. The total inventory of Li_2O in the reactor is 2000 tonnes. Since the Li_2O particles are from 300-500 μm in diameter, we do not anticipate that more than 1% of the total Li_2O inventory would be released outside the reactor building in case of a failure of the containment and chamber. The Whole body early dose at the site boundary would be 551 mrem. ^{24}Na produced from the sodium impurities in the Li_2O is the major contributor to the early dose. ^{60}Co and ^{58}Co are the second and third contributors to the dose, respectively.

5.5.5.3 Tritium

The fourth and final source of potential off-site doses considered in this analysis is produced by the accidental release of the tritium contained inside the reactor building at any moment. We identified the tritium inventories in the Li_2O granules present in the reactor system as our major source of concern. The tritium solubility in the Li_2O at an average temperature of 650 °C is 0.081 wppm. For a total Li_2O inventory of 2000 tonnes, the steady state inventory is 162 g. The other two sources of tritium in the reactor system are the graphite structure and the helium circuit. The graphite reactor structure will absorb some tritium. Based upon the first wall, 165 tonnes of C, the total inventory would be 10 grams of tritium. On the other hand, the He circuit contains HTO at a partial pressure of 6 Pa and an average temperature of 918 °C, giving a total inventory of 5 grams of tritium. In addition, the reactor building atmosphere of Xe has a continuous tritium inventory of about 4.6 g. Finally, the target feed channel leading to the injector within the reactor building is about 50 m long which allows it to handle about 1400 grams of tritium per day. However, since the number of targets present inside the channel is limited to one minute fueling time, the total tritium inventory in this system is kept at about 1 g. Assuming a 100% release, the whole body early dose produced by the release of all of the 182.6 g of tritium is 1.64 rem.

Table 5.10 shows the potential off-site doses produced by simultaneous occurrence of the four previous scenarios. The total whole body early dose at the site boundary mounts only to 2.22 rem, which is far below the 25 rem value recommended for this study by the oversight committee as a threshold for avoidance of early fatalities. The WB early dose is also below the 5 rem level where evacuation plans are required.

Table 5.10. SOMBRERO's Potential Off-Site Doses

	Chamber (0.44%FW)	Shield (600°C)	Li_2O (1 %)	Tritium (100 %)	Total
Prompt Dose at 1 km (Rem)					
WB	1.24e-3	2.41e-2	4.84e-1	2.12e-1	7.21e-1
BM	1.29e-3	2.81e-2	5.22e-1	7.77e-1	1.33
Lung	2.06e-3	5.44e-2	9.95e-1	1.69	2.74
LLI	1.09e-3	2.55e-2	4.11e-1	2.70e-1	7.08e-1
WB Early Dose (Rem)					
At 1 km	1.31e-3	2.47e-2	5.51e-1	1.64	2.22
At 10 km	8.31e-5	1.53e-3	3.62e-2	3.81e-1	4.19e-1
WB Chronic Dose at 1 km (Rem)					
Inh + Grd	3.72e-3	1.34e-1	7.22	2.26	9.62
Ingestion	7.97e-3	1.69e-1	22.5	84.71	107.4
Total	1.17e-2	3.03e-1	29.72	86.97	117
WB Chronic Dose at 10 km (Rem)					
Inh + Grd	2.47e-4	9.04e-3	4.99e-1	5.24e-1	1.03
Ingestion	5.52e-4	1.17e-2	1.56	19.61	21.19
Total	7.99e-4	2.07e-2	2.06	20.13	22.22
Cancers					
Sum Organs	3.62e-3	1.77e-2	2.71	25	27.73
WB	1.23e-3	1.59e-2	3.59	50.73	54.34
Population Dose (Man-Rem)					
WB	7.78	101	2.27e+4	3.21e+5	3.44e+5

5.6 OSIRIS SAFETY ANALYSIS

A separate activation and safety analysis was performed for the chamber, shield, and Flibe coolant of Osiris. The reactor target yield is 432 MJ and the rep-rate is 4.6 Hz, resulting in 1987 MW of fusion power. Osiris 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 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.

5.6.1 Activation Analysis

The radioactivity generated in the reactor chamber and shield was calculated for the 40 year reactor life-time. 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 has been assumed to go through this cycle once a week, Ta has been only exposed to only 1560 (number of weeks in 30 years) shots during the reactor life-time. Even though Ta is continuously extracted from Flibe, a steady state concentration of 10 wppm of radioactive Ta in the Flibe has been assumed at all times during operation. A third calculation has been 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 has been 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 (Table 5.11). The radial build used in this analysis is shown in Fig. 5.7.

5.6.1.1 Chamber and Shield

A small amount of activity is induced in the Osiris chamber during the reactor life-time. 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, produced from the

Table 5.11. Elemental Composition of Flibe

Nuclide	wppm unless wt% indicated
Li	14.038 wt%
Be	8.975 wt%
C	91
O	987
F	76.848 wt%
Mg	5.5
Al	77
Si	27
Ti	19
Cr	9
Mn	11
Fe	139
Ni	13
Cu	7

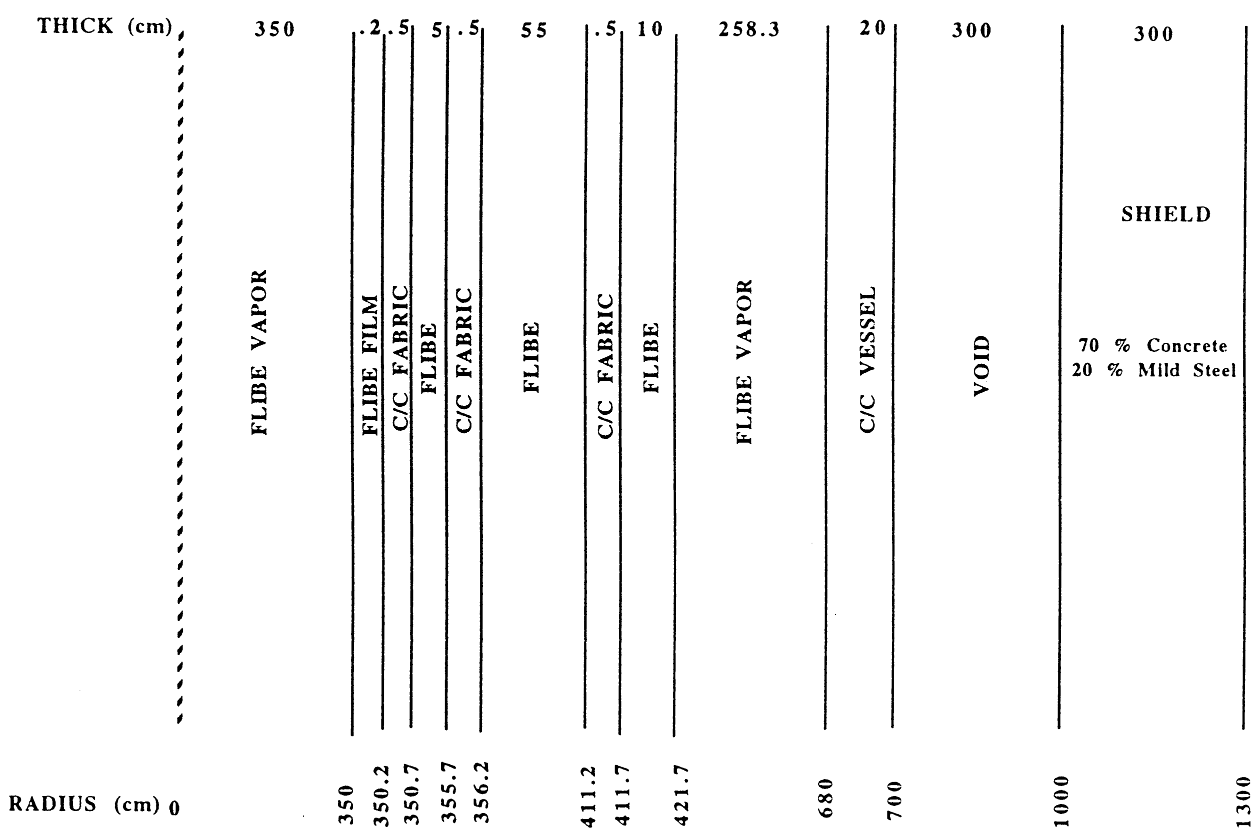

Fig. 5.7. Radial build used in activation calculations for Osiris.

main constituent element, carbon. Fig. 5.8 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, ^{56}Mn, and ^{55}Fe are the major sources of activity present in the shield during the first year following shutdown. The long-term activity (> 10 yr) is dominated by ^{39}Ar ($T_{1/2}$ = 269 yr), ^{63}Ni, and ^{14}C which are all induced from impurities in both the steel and concrete used in this analysis.

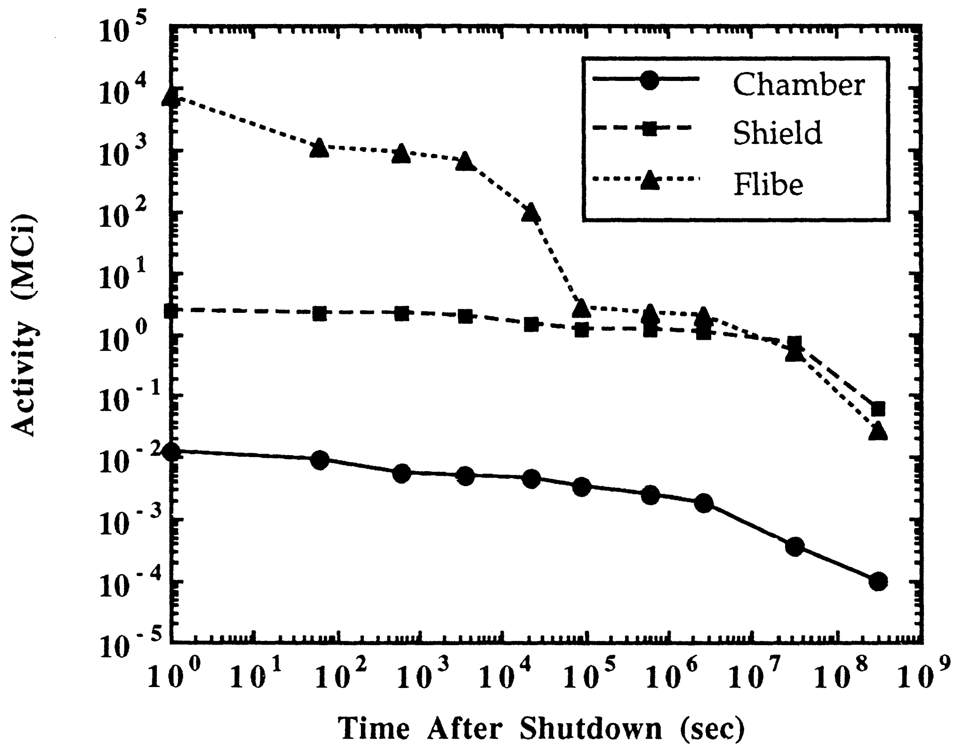

Fig. 5.8. Activity after shutdown in different regions of Osiris.

Table 5.12 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 NRC regulations specified in 10CFR20. 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

Table 5.12. Radioactivity After Shutdown in the Different Regions of Osiris

Time After shutdown	Activity (MCi) Chamber	Activity (MCi) Shield	Decay Heat (MW) Chamber	Decay Heat (MW) Shield	BHP (km^3 air) Chamber	BHP (km^3 air) Shield
0	1.23e-2	2.33	1.92e-4	2.29e-2	1040	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

1.5 GJ, one month after shutdown. This amount of decay heat cannot increase the shield temperature by more than 1 to 2 degrees.

5.6.1.2 High-Z Target Material

Tantalum is used in the Osiris target because of its high solubility in Flibe. The Ta has been assumed to have a 4-mm inner radius and 90 μm thickness. Ta debris has been 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 have been performed using a total of 2.782 x 10^6 targets (847 kg of Ta), which represents the number of targets used in Osiris every week. As shown in Table 5.13, 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 x 10^{15} yr).

5.6.1.3 Coolant

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

Table 5.13. **Total Tantalum-Induced Activity at Osiris Shutdown**

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

inventory, which is only exposed to the neutron flux only during one shot every week, we conservatively assume that 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. 5.8, 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 five years following shutdown. The tritium steady state inventory in the Flibe is kept at a low level of 1 g or 10,000 Ci.

5.6.2 Maintenance

Biological dose rate calculations have been performed at selected locations behind the concrete shield and in the space between the chamber and shield. Fig. 5.9 shows the calculated dose rate between the chamber and shield as a function of time following shutdown. At shutdown the dose rate is 7.95 rem/hr and only drops to 191 mrem/hr after one year. The biological dose rate is dominated by ^{56}Mn and ^{54}Mn during the first day and by ^{54}Mn and ^{55}Fe 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. The dose rate behind the shield, however, 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.

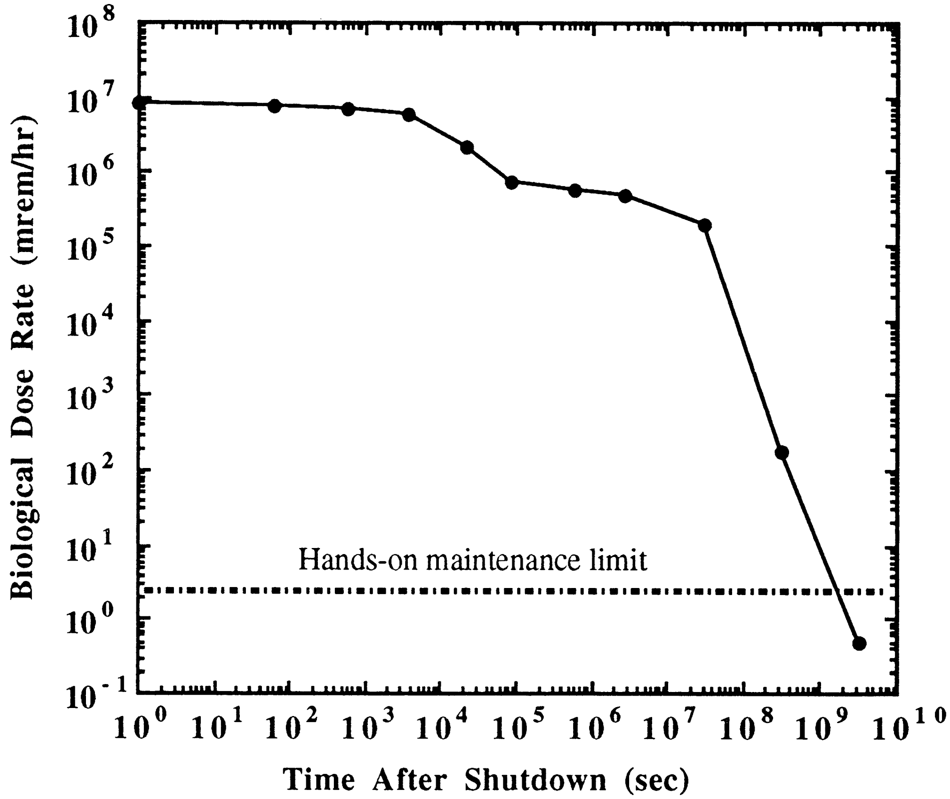

Fig. 5.9. Biological dose rate in the space between Osiris chamber and shield.

5.6.3 Radwaste Classification

The radwaste classifications of the Osiris chamber, shield, and coolant were evaluated according to both the NRC 10CFR61 and the Fetter waste disposal concentration limits (WDL). The specific activities calculated by the DKR-ICF code were used to calculate the different waste disposal ratings (WDR). Table 5.14 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 generated from $^{13}C(n,\gamma)$ reaction is the major contributor to the WDR of the chamber if Class A limits are used and ^{3}H produced from the boron impurities in the carbon fabric via the $^{10}B(n,2\alpha)$ reaction is the other contributor. If Class C waste disposal limits are used, ^{14}C and ^{26}Al produced from $^{26}Al(n,2n)$ 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 of the concrete. ^{63}Ni produced from ^{63}Cu and

Table 5.14. Waste Disposal Ratings (WDR) of the Different Regions of Osiris

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)
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.e-5 ^{26}Al)	0.048 (0.047 ^{26}Al)

^{94}Nb 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 5.14, 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 and 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.

5.6.4 Routine Atmospheric Effluents

The EPA AIRDOS-PC code was used to calculate the off-site dose to the maximally exposed individual (MEI) as a result of the routine release of tritium to the environment. Once again 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). As discussed in Sec. 2.2.7.2 and Sec. 2.2.7.3, we considered the routine release of tritium from the reactor system, reactor 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 reactor 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. 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 5.15 and using meteorological conditions at different cities, we calculated the dose expected at typical locations. A summary of the results is shown in Table 5.16. 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 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 reactor building. As mentioned before, 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. 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 by ITER if the 75 m stack height is assumed.

5.6.5 Accident Analysis

In this subsection, we calculated the potential off-site dose produced in Osiris during an accident. 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 methodology assuming a sequence of sever accidents. In addition, we have calculated the doses produced by the release of all the tritium contained in the reactor building 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 low.

Table 5.15. Routine Atmospheric Release Parameters

Site Information:	
Locations:	Albuquerque
	Boston
	Chicago
	Los Angeles
Temperature:	15 °C
Rainfall:	75 cm/yr
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
Reactor Building:	10 Ci/day
Fuel Reprocessing:	20 Ci/day
Target Factory:	10 Ci/day
Total (adjusted for 75% availability):	25,050 Ci/yr

Table 5.16. 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

5.6.5.1 Chamber and Shield

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, shutdown the reactor. Using the worst release characteristics as defined by the ESECOM methodology (Table 5.8), 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.

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. Adjusted PCA volatility rates at 600 °C in dry air were used in this analysis. 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.

5.6.5.2 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 2.78 x 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 the 660 tonnes of Flibe at the reactor shutdown is shown in Table 5.17. ^{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 building (3 targets) is conservative

Table 5.17. 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
^{81}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

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.

5.6.5.3 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 reactor building 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 are produced per shot. A simultaneous breach in the reactor building and chamber would allow the cold air to flow into the chamber. The air starts cooling the Flibe vapor and hence reducing 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[5.12], 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 reactor building contains 165 g of BeF_2. The BeF_2 is a major source of 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 $\mu g/m^3$. This value is below the level of concern as the recommended upper limits for continuous and peak exposures are 2 and 25 $\mu g/m^3$, respectively (US Federal Register, 1975; National Institute of Safety and Health, 1972).

5.6.5.4 Tritium

Finally, we considered the potential off-site doses produced by the accidental release of the tritium from both the reactor systems and the reactor building at any moment. The three major sources of tritium in the reactor system are the Flibe breeder, the graphite fabric 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. 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, as mentioned before (Sec. 2.2.7.2) most of the adsorption traps would be recycled every hour so that their total tritium inventory is only 6.5 g. We identified the tritium contained in the targets present in the target delivery system as the major source of potential tritium release from the reactor 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 5.18 shows the potential off-site doses produced by simultaneous occurrence of the previous accident scenarios. The total whole body dose at the site boundary mounts to only 128 mrem, which is far below the 25 rem value recommended for this study by the oversight committee as a threshold for avoidance of early fatalities. The WB early dose is also far below the 5 rem level where evacuation plans are required.

Table 5.18. Osiris' Potential Off-Site Doses

	Chamber (0.2%FW)	Shield (600°C)	Flibe (0.5 kg)	Tantalum (3 targets)	Tritium (12.7 g)	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

5.7 TARGET FACTORY ANALYSIS

The target factory facility processes a total of 400,000 and 580,000 targets per day for Osiris and SOMBRERO, respectively. Hence, the facility is expected to handle a daily flow of tritium of 1020 grams for Osiris and 1400 grams for SOMBRERO. For both reactors, the rate of target production is maintained at the rate of usage to minimize the amount of stored tritium in the fabricated fuel targets. The total tritium inventory along the production line is limited to only 300 g. Even though 200 g of this tritium is stored in two liquid cryogenic containers, surrounded by evacuated chambers making it very unlikely for the tritium to be released in case of an accident, we still assumed that a worst accident scenario should involve the release of the total 300 grams of tritium. Table 5.19 shows the potential off-site doses produced during such an accident. As shown in the table, the maximum WB early dose projected as a result of a severe accident involving the target factory of either reactor designs would be 2.7 rem. Therefore, no evacuation plans are required.

Table 5.19. Off-Site Doses Due to Tritium Release from Target Factory

Prompt Dose at 1 km (Rem)	
WB	0.35
BM	1.28
Lung	2.80
LLI	0.44
WB Early Dose (Rem)	
At 1 km	2.70
At 10 km	0.63
WB Chronic Dose at 1 km (Rem)	
Inh + Grd	3.72
Ingestion	139.24
Total	142.95
WB Chronic Dose at 10 km (Rem)	
Inh + Grd	0.86
Ingestion	32.25
Total	33.11
Cancers	
Sum Organs	41.11
WB	83.40
Population Dose (Man-Rem)	
WB	5.29e+5

5.8 FUEL REPROCESSING FACILITIES

Most of the tritium present in SOMBRERO's fuel reprocessing facility is located in its cryogenic distillation system and the desiccant bed used to absorb the HTO from He. The tritium inventory in the distillation system during continuous operation is 13 g and the inventory of the desiccant beds during two-hours of operation is 61 g. At the onset of an accident, the tritium released from the two systems is vented to an evacuated tank and hence disallowing any tritium release. However, a failure in the venting system and 100% release of the tritium contained in SOMBRERO's fuel reprocessing facility would result in a WB early dose of 676 mrem at the site boundary (1 km).

On the other hand, in the fuel reprocessing facility of Osiris constant tritium inventories exist in the cryogenic distillation system (9.5 g) and the vacuum pumps from the vacuum disengager. The vacuum disengager is used to separate T_2 from Flibe by the use of vacuum pumps capable of maintaining low pressure. If the pumps are on stream for one-hour, their tritium inventory would be 44 g. Once again, assuming a venting system failure and 100% release of the 53.5 grams of tritium during an accident would result in a WB early dose of 482 mrem. Table 5.20 shows the different off-site doses expected during accidents involving the fuel reprocessing facilities of both SOMBRERO and Osiris.

5.9 NUCLEAR GRADE COMPONENTS

N-Stamp nuclear grade components are only required if the estimated off-site dose released is above the 25 rem limit. As shown in the previous analysis, none of the two reactor components would produce an off-site whole body early dose in access of 25 rem during a conservative accident scenario. However, a total release of the Flibe or Li_2O radioactive inventories would produce an off-site dose which exceeds the 25 rem limits. In such a case, some N-Stamp components would be required. A total release is quite impossible due to the lack of energy sources sufficient to mobilize most of the Flibe or Li_2O. Therefore, we conclude that none of the components (from either reactor) would require nuclear grade materials. Also, none of the fuel reprocessing facilities proposed for SOMBRERO and Osiris would produce more than 1 rem at the onset of an accident, allowing them to avoid the N-Stamp requirements. Similarly, due to the low tritium inventory present in the target factory at any moment (300 g), we can also avoid the use of nuclear grade components in the proposed target factory.

Table 5.20. Off-Site Doses Due to Tritium Release from Fuel Reprocessing Facility

	SOMBRERO	Osiris
Prompt Dose at 1 km (Rem)		
WB	0.09	0.06
BM	0.32	0.23
Lung	0.70	0.50
LLI	0.11	0.08
WB Early Dose (Rem)		
At 1 km	0.68	0.48
At 10 km	0.16	0.11
WB Chronic Dose at 1 km (Rem)		
Inh + Grd	0.93	0.66
Ingestion	34.81	24.83
Total	35.73	25.49
WB Chronic Dose at 10 km (Rem)		
Inh + Grd	0.22	0.15
Ingestion	8.07	5.75
Total	8.28	5.90
Cancers		
Sum Organs	10.28	7.33
WB	20.85	14.87
Population Dose (Man-Rem)		
WB	1.32e+5	9.42e+4

5.10 COMPARISON OF SOMBRERO AND OSIRIS

The SOMBRERO and Osiris reactor designs have distinct favorable safety characteristics. Because of the double wall layout used in SOMBRERO, the biological dose rate behind the steel-reinforced concrete shield is low enough to allow hands-on maintenance inside the IHX enclosures within a day after shutdown. The dose rate after shutdown behind the 3 meter biological shield of Osiris is only 0.11 μrem/hr allowing for hands-on maintenance. However, only remote maintenance is allowed in the space between the chamber and shield of both reactors. The chamber and shield of both reactor designs qualify for near surface burial as Class A low level waste. Using the NRC waste disposal limits for solid waste, both the Li_2O solid breeder and Flibe could qualify for shallow land burial as Class C and Class A low level wastes, respectively. 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.

Some tritium does reach the off-site environment during normal operation. The reactor system, the reactor building, the fuel reprocessing facility, and the target factory are the major sources of routine release of tritium. Assuming a barrier factor of 10^6, the doses from the atmospheric routine release of tritium from SOMBRERO and Osiris to the maximally exposed individual are 0.93 and 2.43 mrem/yr, respectively. Both values are far below the 10 mrem/yr EPA current effluent limit. The site boundary is assumed to be at 1 km from the point of release. The off-site doses caused by an accidental release of radioactivity from both reactor designs are dominated by the dose resulting from the off-normal release of tritium. During an accident, the maximum vulnerable inventory of tritium in SOMBRERO is 182.5 g. Most of the tritium (162 g) is contributed by the Li_2O granules. On the other hand, due to the small tritium inventory in Flibe salt (1 g), the maximum vulnerable inventory of tritium in Osiris is only 12.7 g. The estimated off-site whole body (WB) early dose released from SOMBRERO due to a highly unlikely sequence of simultaneous accident scenarios involving, the reactor chamber, biological shield, breeder, and tritium is 2.22 rem. This dose is below the 5 rem level where evacuation plans are needed and far below the 25 rem value recommended for this study by the oversight committee as a threshold for avoidance of early fatalities. Assuming similar accident scenarios, the Osiris design would result in a WB early dose of only 128 mrem.

An accident analysis involving the target factory facility showed that a 100% release of the 300 grams of tritium expected to be present inside the facility at any moment would result in a WB early dose at the site boundary of only 2.70 rem, which again is below the limits required for public evacuation. Finally, an accident resulting in the release of the total inventory of tritium existing in the fuel reprocessing facilities of SOMBRERO and Osiris would produce off-site doses of only 676 and 482 mrem, respectively. The very low off-site dose for either reactor designs eliminates the need for N-Stamp nuclear grade reactor components, which are only required if the dose exceeds the 25 rem limit.

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