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ABSTRACT

Activation and safety analysis has been performed for the chamber, shield and Li_2O coolant of the inertial confinement fusion (IFE) reactor SOMBRERO. The total activities generated in the reactor graphite chamber and steel-reinforced concrete shield at shutdown are 0.054 and 10.12 MCi, respectively. The biological dose rate at the back of the shield drops to 1.6 mrem/hr after one day of shutdown allowing for hands-on maintenance. Radwaste classification has shown that both the chamber and shield would easily qualify as Class A low level waste (LLW) according to the 10CFR61 waste disposal concentration limits (WDL). At the same time, the Li_2O granules would qualify as Class C LLW. The maximum public dose from atmospheric effluents is 0.93 mrem/yr. The dose is due to tritium and its maximum value occurs at the reactor site boundary which is 1 km away from the point of tritium release. Only a small fraction (0.44%) of the graphite first wall would be mobilized during a loss of coolant accident (LOCA). During such an accident, the shield temperature would only increase by a few degrees releasing a very small fraction of its radioactive inventory. The total tritium inventory in the containment building which is assumed to be released at the onset of a severe accident is 182.6 grams. The estimated whole body (WB) early dose from a severe accident resulting in the failure of the reactor containment is 2.22 rem. The very low off-site dose eliminates the need for N-stamp nuclear grade components in SOMBRERO.

I. INTRODUCTION

SOMBRERO is a conceptual design of a 1000 MW_e KrF laser driven IFE power reactor utilizing direct drive targets with near symmetric illumination.[1] 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 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_0), the blanket is divided into 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 behind it. The steel-reinforced concrete shield is made of 70% concrete, 20% mild steel and 10% helium coolant.

In this paper a detailed activation and safety analysis is performed in order to show the favorable environmental safety characteristics of SOMBRERO.

II. CALCULATIONAL PROCEDURE

Neutron transport calculations have been performed using the one-dimensional discrete ordinates neutron transport code ONEDANT.[2] 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 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 has been 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 has been calculated for the 40 year reactor lifetime. A separate calculation has been performed for the coolant. 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 decay gamma source produced by the DKR-ICF code is used with the adjoint neutron flux to calculate the biological

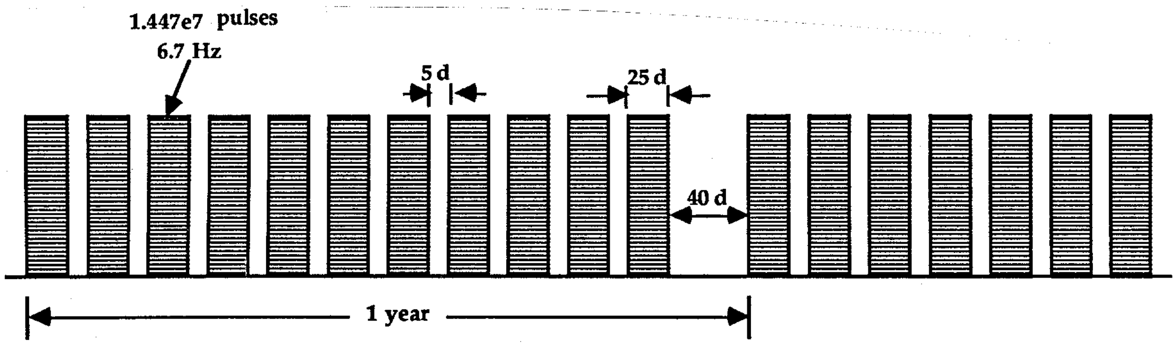

Fig. 1. Pulse sequence used in the activation calculation.

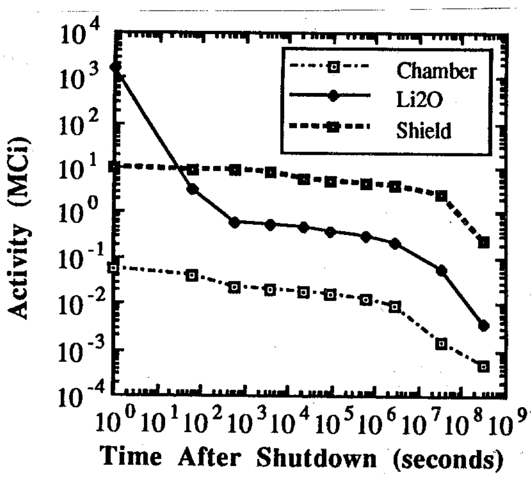

Fig. 2. Activity after shutdown in different SOMBRERO regions.

dose rate after shutdown using the DOSE[3] code. The dose rate calculations have been performed at two different locations. The first location is in the intermediate heat exchanger (IHX) enclosures behind the concrete shield while the second location is in the space between the chamber and shield. The activation results have been also utilized in the radwaste classification and the off-site dose calculations performed by the FUSCRAC3[4] code. The off-site doses are produced by the accidental release of the radioactive inventory from the reactor containment building assuming the worst case weather conditions. Finally, the EPA code AIRDOS-PC[5] has been used to estimate the off-site dose due to the routine release of tritium.

III. ACTIVITY, DECAY HEAT AND BHP

The total activity generated in the different regions of SOMBRERO as a function of time following shutdown is shown in Fig. 2. 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 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 I shows the dominant contributors to the activity generated in SOMBRERO during different time periods following shutdown. Table II 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 U.S. Nuclear Regulatory Commission (NRC) regulations specified in 10CFR20.[6] The temporal variation of the decay heat and BHP after shutdown are 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 I. One value which is useful for predicting the thermal response of the shield to a loss of coolant accident is the integrated decay heat. 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.

IV. MAINTENANCE

Biological dose rate calculations have been performed at selected locations behind the concrete shield and in the space between the chamber and shield. At shutdown, the dose rate is 22 mrem/hr and drops to 1.6 mrem/hr after one day 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 following shutdown. A limit of 2.5 mrem/hr for hands-on maintenance is used in this paper 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. In the meantime, the dose rate between the chamber and shield is quite high. At shutdown, the dose rate 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.

Table I

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 year	^{3}H, ^{37}Ar, ^{55}Fe	^{55}Fe, ^{54}Mn, ^{37}Ar	^{55}Fe, ^{35}S, ^{37}Ar
1 year–10 years	^{3}H, ^{55}Fe, ^{10}Be	^{55}Fe, ^{3}H, ^{54}Mn	^{55}Fe, ^{14}C, ^{39}Ar
$>$ 10 years	^{10}Be, ^{14}C, ^{39}Ar	^{39}Ar, ^{63}Ni, ^{14}C	^{14}C, ^{39}Ar, ^{63}Ni

V. WASTE DISPOSAL

The waste disposal ratings for SOMBRERO have been evaluated according to both the NRC 10CFR61[7] and Fetter[8] waste disposal concentration limits (WDL). According to the 10CFR61, there are two classes of low level waste (LLW) that are acceptable for near surface or shallow land burial. The first class is Class A waste which is also known as segregated waste is not considered to be hazardous after 100 years from its disposal, a time at which a loss of institutional control on the disposal site is assumed. The second class of LLW is known as Class C or intruder waste. This class of waste must be packaged and buried such that it will not pose a hazard to an inadvertent intruder after the 100 year institutional control period is over.

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 III 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. Both the chamber and shield would easily qualify as Class A low level waste. ^{14}C ($T_{1/2}$ = 5730 yr) generated from ^{13}C (n,γ) 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,2α) 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 are the other major contributors. Both ^{63}Ni and ^{94}Nb are generated in the steel component of the shield.

As shown in Table III, 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 the ^{17}O (n,α) 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 4 times during the reactor life.

VI. 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. We considered the routine release of tritium from the reactor system, containment 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 containment 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 results in a routine release of 15 Ci/day. Finally, 14 Ci/day of tritium are released from the target factory as it processes about 580,000 targets containing over 1375 grams of tritium.

Assuming the release parameters listed in Table IV and using meteorological conditions at different cities, we calculated the dose expected at typical locations near Boston, Chicago, Albuquerque and Los Angeles. A summary of the results is shown in Table V. 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 in ITER. It is important to keep in mind that the estimated dose values strongly depend 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 metrological conditions were used. Actually, the rule of thumb for determining the necessary stack

Table II

Radioactivity After Shutdown in Different SOMBRERO Regions

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	0.11	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	878.3	2.18e+4
100 years	1.46e-4	1.70e-3	1.10e-7	1.49e-6	869.3	1.12e+4

Table III

Waste Disposal Ratings (WDR) for Different SOMBRERO Regions

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)

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[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 V.

VII. ACTIVATION PRODUCT MOBILIZATION

Another source of potential off-site doses which are of concern in SOMBRERO are the doses produced by an accidental release of the radioactive inventory in the containment building. In this section we calculated the potential off-site doses using the ESECOM[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 a significant off-site dose even though the probability of such a failure is very low.

A. The Chamber

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 containment building by one torr. The higher carbon vapor pressure would prevent the laser beam from propagating to the target 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 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. ^{24}Na, ^{48}Sc and ^{54}Mn are the major contributors to the off-site dose.

B. The Shield

The decay heat generated in the steel-reinforced concrete shield is very low. The decay heat generated within the first 2 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.[11] 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 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.

C. 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 tones. 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.

D. 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 containment 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 containment 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 containment 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 IV

Routine Atmospheric Effluents Release Parameters

• Emission Information	
Year-Round Averaging	
Stack Height	125 m
Stack Diameter	0.3 m
Momentum	1 m/s
• Tritium Pathways	
Reactor System	26 Ci/day
Containment Building	23 Ci/day
Fuel Reprocessing	30 Ci/day
Target Factory	14 Ci/day
Total (adjusted for 75% availability)	25,460 Ci/yr

Table V

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

Table VI shows the potential off-site doses produced by simultaneous occurrence of the four previous scenarios. The total whole body dose at the site boundary amounts only to 2.22 rem 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. SUMMARY

The SOMBRERO reactor has distinct favorable safety characteristics. The biological dose rate behind the steel-reinforced concrete shield is low allowing for hands-on maintenance within a day after shutdown. However, only remote maintenance is allowed in the space between the reactor chamber and shield. The chamber and shield qualify for near surface burial as Class A low level waste. After 30 FPY, the Li_2O solid breeder could qualify for shallow land burial as Class C waste. The dose from the atmospheric routine release of tritium to the maximally exposed individual is 0.93 mrem/yr which is 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 estimated off-site whole body early dose released from SOMBRERO due to a highly unlikely sequence of simultaneous accident scenarios is 2.22 rem which is below the 5 rem level where evacuation plans are needed. The 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.

Table VI

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

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