Environmental and Safety Assessment of the Inertially Confined Direct Drive Laser Fusion Power Reactor SIRIUS-P

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ABSTRACT

Environmental and safety analyses have been performed for SIRIUS-P, and its target factory and fuel reprocessing facilities. Both the c/c composite chamber and steel-reinforced concrete shield would easily qualify as Class A low level waste (LLW). Due to the high ^{14}C activity, the Li_2O solid breeder and TiO_2 coolant would only qualify for Class C LLW. The radiological dose to the population in the vicinity of the reactor site due to the routine release of tritium is 0.56 mrem/yr. During a loss of coolant accident (LOCA) or loss of flow accident (LOFA), the whole body (WB) early dose at the site boundary (1 km) only amounts to 1.55 and 58.2 mrem for the chamber and shield, respectively. The WB early dose at the site boundary due to the Li_2O and TiO_2 are 93.5 μrem and 93 mrem, respectively. A 100% release of the 156.3 g of tritium contained inside the reactor containment at any moment would produce a WB early dose on the order of 1.4 rem. Even though the target factory processes a total of 580,000 targets/day, the total tritium inventory along the production line is limited to only 285 g. The maximum WB early dose projected as a result of a severe accident involving the target factory of SIRIUS-P would be limited to 2.57 rem. In addition, a 100% release of the tritium contained in the fuel reprocessing facility would only result in a WB early dose of 640 mrem. The use of N-stamp nuclear grade components in SIRIUS-P can be avoided due to the low off-site doses.

I. INTRODUCTION

SIRIUS-P is a 1000 MW_e fusion power reactor based on a near symmetrically illuminated configuration provided by a KrF laser [1]. 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 chamber is divided into two distinct parts: (1) First wall assembly, constructed from a carbon/carbon composite and cooled with a flowing granular bed of TiO_2; (2) Blanket assembly, constructed from a SiC/SiC composite and cooled with a moving bed of solid Li_2O granules (60% density factor) flowing through the chamber by gravity. The particles are transported in a fluidized state by helium gas at 0.2 MPa. The chamber is surrounded by a biological shield to allow for hands-on maintenance at selected locations behind it. The steel-reinforced concrete shield is made of 70% boron fritsbarytes concrete, 20% mild steel and 10% helium coolant.

A strong emphasis has been given to the environment and safety issues in the SIRIUS-P reactor design. c/c composite is used as chamber material to avoid a high level of induced radioactivity in the reactor structure. Similarly, the use of TiO_2 and Li_2O as coolant and breeder materials eliminates the hazard posed by the energy producing chemical reactions usually associated with the use of lithium and hence reduce the risk of mobilizing the radioactive inventory present in the reactor. The methodology used in this safety analysis does not depend on the probability of accident initiating scenarios. We have rather 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 inventories inside the reactor building. However, since the existence of radioactivity does not in itself represent a safety hazard, the second step is to consider a set of pessimistic but rather credible accident scenarios for mobilizing and releasing the radioactive inventory.

II. CALCULATIONAL PROCEDURE

Neutron transport calculations have been performed using the one-dimensional discrete ordinates neutron transport code ONEDANT [2] using the ENDF/B-V cross section data. 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 reactor has a neutron wall loading value of 3.43 MW/m^2. 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 [3] with the ACTL [4] activation cross section library. The decay and gamma source data is taken from the table of isotopes [5] with the gamma source data being in 21 group structure format. The DKR-ICF code allows for accurate modeling of the pulsing schedule. 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 steel-reinforced shield has been calculated for the 40 year reactor lifetime with 75% availability. In the mean time a separate calculation has been performed for the coolant and breeder. The residence time of the coolant and breeder in the chamber is 76 seconds. The total inventories of TiO_2 and Li_2O take about 135 seconds to go through the reactor chamber. Therefore, their activities have been calculated to allow for the fact that the TiO_2 and Li_2O granules spend only 57% of the time exposed to neutrons in the reactor chamber. The activation results have also been utilized in the radwaste classification and the off-site dose calculations performed by the FUSCRAC3 [6] 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 [7] has been used to estimate the dose from routine atmospheric effluents.

III. RADWASTE CLASSIFICATION

Waste disposal ratings have been evaluated according to both the Nuclear Regulatory Commission 10CFR61 [8] and Fetter [9] waste disposal concentration limits (WDL). The specific activities calculated for the different radionuclides have been used to evaluate the radwaste classification of the chamber, shield, TiO_2 coolant and Li_2O solid breeder. Table I 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, noncompacted values are based on averaging over the total volume of a particular region implying that internal voids will be filled with concrete before disposal.

As shown in the table, 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 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 x 10^5 yr), produced from ^{27}Al (n,2n) reaction, are the major dominant nuclides for the 10CFR61 and Fetter limits, respectively. About 65% of the Class A waste disposal rating of the shield is contributed by tritium due to the concrete's high boron content. ^{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. ^{63}Ni and ^{94}Nb are generated in the steel component of the shield.

Table I. Waste Disposal Ratings of the different SIRIUS-P Components

Class	Chamber	Shield	Li_2O	TiO_2
A (NRC)	0.032	0.235	1.21	6.56
C (NRC)	2.78e-3	4.55e-3	0.117	0.656
C (Fetter)	0.67	5.41e-3	6.23e-3	8.78e-3

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 chamber, this ^{14}C is generated by ^{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 once during the reactor life. Finally, the TiO_2 coolant would only qualify for Class C LLW regardless of the limits used due to its high ^{14}C activity.

IV. 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 [10] which indicated that only 1.5 Ci of tritium were released through the stack during the processing of 100 grams within 38 hours. Hence, 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.

Table II. Routine Atmospheric Release Parameters

Emission Information	
Year-Round Averaging	
Stack Height	125 m
Stack Diameter	0.3 m
Momentum	1m/s
Tritium Pathways	
Reactor System	0.01 Ci/d
Containment Building	14 Ci/d
Fuel Reprocessing	21 Ci/d
Target Factory	21 Ci/d
Total (75% availability)	15,330 Ci/yr

Assuming the release parameters listed in Table II 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 III. The worst dose was in the Albuquerque area but was only 0.56 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 dose values strongly depend on the stack height. For example, using a 30 meter stack height results in an EDE of 11 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 the plume into the wake of the building [11]. A shorter stack must be justified with appropriate analysis. If one were to apply the rule of thumb to SIRIUS-P 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 III.

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

Site	Dose (mrem/yr)	Distance (m)
Albuquerque	0.56	1000
Boston	0.14	3000
Chicago	0.22	1000
Los Angeles	0.42	3000

V. ACCIDENT ANALYSIS

Off-site doses could be produced at the onset of an accidental release of the radioactive inventory present inside the containment building of SIRIUS-P. In this section we calculated the potential off-site doses using the ESECOM [12] methodology due to the release of some of the radioactive inventory of the chamber, shield and coolants. 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. 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 inside the containment 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 (class F wind stability, 1 m/s wind speed, etc.), we calculated the off-site doses produced by the release of 0.44% of the graphite first wall. The whole body (WB) early dose at the site boundary (1 km) only amounts to 1.55 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.

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, 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 [13]. 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 58.2 mrem. Most of the dose is produced by the manganese isotopes, ^{54}Mn and ^{56}Mn.

B. TiO_2 and Li_2O

The blanket of SIRIUS-P consists of a moving bed of solid TiO_2 and 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 TiO_2 and Li_2O in the reactor is 380 and 734 tonnes, respectively. The off-site doses were calculated by using experimental values for the vapor pressure of TiO_2 and Li_2O at 1850 K and 1600 K, respectively, and assuming a one hour release of activated TiO_2 and Li_2O through a hole in the containment building. The most probable scenario of a major containment failure would be caused by the crashing of a fighter aircraft because of their high momentum per unit frontal area and because of the possible detonation of ordnance on board. Experimental studies at Sandia National Laboratory of the crash of an F-4 fighter aircraft in a concrete wall about 1 m thick have only produced several inches of spalled concrete [14]. A containment hole area of 1 m^2 was chosen in order to estimate conservative values of the off-site doses. The whole body early dose at the site boundary due to Li_2O is 93.5 μrem. ^{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. On the other hand, the whole body early dose at the site boundary due to TiO_2 is 93 mrem. The major contributors to the dose, ^{48}Sc, ^{46}Sc, ^{47}Sc, and ^{45}Ca, are products of neutron interactions with titanium.

C. Tritium

The 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 Li_2O breeder particles produce about 450 grams of tritium per day; however, at steady state all the tritium is released as HTO to the helium circuit inside the reactor chamber. Based upon the partial pressure of HTO in the gas phase, the solubility of tritium in the oxide particles varied from nearly zero to a maximum of 0.323 wppm, for an average solubility of 0.162 wppm. For a total Li_2O inventory of 734 tonnes, the steady state tritium inventory is 129 g. The other two sources of tritium in the reactor system are the graphite structure and the helium circuit. The graphite structure will absorb some tritium. Based upon the first wall, 12 tonnes of carbon, the steady state inventory would be 22 grams of tritium. On the other hand, the helium gas flowing counter-currently to the breeder particles has an average tritium concentration of 2 mg/m^3. The volume of helium in the reactor is about 245 m^3 and up to twice this volume may exist in equipment external to the reactor. The tritium inventory which could be released by a rupture in the circuit is 1.6 g.

In addition, the containment building atmosphere has a volume of 1.8×10^5 m^3 and continues Xe, which is mixed with the unburned target fuel, at 0.5 torr pressure. The tritium inventories in this atmosphere and the building walls are about 1.7 and 1 g, respectively. 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 156.3 g of tritium is 1.40 rem.

A simultaneous occurrence of all previous accident scenarios, involving the chamber, shield, coolant, solid breeder and tritium, would produce a total whole body dose of 1.55 rem. This WB early dose is significantly below the 200 rem value recommended by the ESECOM committee as a threshold for avoidance of early fatalities. It is also below the 5 rem level where evacuation plans are required.

VI. TARGET FACTORY AND FUEL REPROCESSING FACILITIES

The target factory facility processes a total of 580,000 targets per day. Hence, the facility is expected to handle a daily flow of 1400 grams of tritium. Since 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 285 g. Even though 171 g of this tritium (3 hour supply) 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 would involve the release of the total 285 grams of tritium. The maximum WB early dose projected as a result of a severe accident involving the target factory of SIRIUS-P is 2.57 rem. Most of the tritium present in the fuel reprocessing facility is located in its cryogenic distillation system and the desiccant bed used to absorb HTO from He. The tritium inventory in the distillation system during continuous operation is 12 g and the inventory of the desiccant beds during two hours of operation is 59 g. At the onset of an accident, the tritium released from the two systems is vented to an evacuated tank and hence disallows any tritium release. However, a failure in the venting system and 100% release of the tritium contained in the fuel reprocessing facility would result in a WB early dose of 640 mrem at the site boundary.

VII. 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 safety analysis, none of the reactor components would produce an off-site whole body early dose in excess of 25 rem during a conservative accident scenario. However, a total release of the TiO_2 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. Since such total release is quite impossible due to the lack of sources of energy that are sufficient to mobilize most of the TiO_2 or Li_2O, we concluded that none of the reactor components would require nuclear grade materials. Similarly, the fuel reprocessing facility would only produce less than 1 rem at the onset of an accident, allowing it to avoid the N-Stamp requirements. Finally, due to the low tritium inventory present in the target factory at any moment (285 g), the use of nuclear grade components in the proposed target factory can also be avoided.

VIII. CONCLUSIONS

The SIRIUS-P reactor design has distinct favorable environmental and safety characteristics. The chamber and shield structure qualify as Class A low level waste (LLW). The Li_2O solid breeder can qualify for Class A if it is reprocessed once in the reactor lifetime, otherwise it is Class C. The TiO_2 coolant would only qualify for Class C LLW due to its high ^{14}C activity. Routine tritium release is low, 65 Ci/day, resulting in a dose to the Maximally Exposed Individual (MEI) on the order of only 0.56 mrem/year. During a LOCA/LOFA, the whole body (WB) early dose at the site boundary (1 km) only amounts to 1.55 and 58.2 mrem for the chamber and shield, respectively. The off-site doses caused by the accidental release of Li_2O and TiO_2 were calculated by using experimental values of their vapor pressure and assuming a 1 hour release through a 1 m^2 hole in the containment building. The WB early dose at the site boundary due to the Li_2O and TiO_2 are 93.5 μrem and 93 mrem, respectively. The dose due to any major accidental release of tritium from the reactor, target factory and fuel reprocessing facility is below the 5 rem level where evacuation plans are required. The use of N-stamp nuclear grade components in SIRIUS-P can be avoided due to the low off-site doses.

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