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May 1996

UWFDM-1023

Presented at the 12th Topical Meeting on the Technology of Fusion Power, 16–20 June 1996, Reno NV

FUSION TECHNOLOGY INSTITUTE

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ENVIRONMENTAL AND SAFETY ASSESSMENT OF LIBRA-SP: A LIGHT ION FUSION POWER REACTOR DESIGN

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ABSTRACT

LIBRA-SP is a 1000 MWe light ion beam power reactor design study. The reactor structure is made of a low activation ferritic steel and uses LiPb as a breeder. The total activities in the blanket and reflector at shutdown are 721 MCi and 924 MCi, respectively. Hands-on maintenance is impossible anywhere inside the reactor chamber. The biological dose rates near the diode are too high at all times following shutdown allowing only for remote maintenance. The blanket and reflector could qualify for disposal as Class C low level waste. The dose to the maximally exposed individual in the vicinity of the reactor site due to the routine release of tritium is about 2.39 mrem/yr. Ten hours after a loss of coolant accident, the reflector produces a whole body (WB) early dose at the site boundary of 253 mrem. The blanket would produce a WB early dose of 8.91 rem. The potential off-site dose produced by the mobilization of LiPb during an accident is 142 mrem. A 100% release of the vulnerable tritium inventory present in the containment at any moment results in a WB early dose of 459 mrem. Release of the vulnerable tritium inventories present in the target factory and fuel reprocessing facility during an accident would result in WB early doses of 1.3 and 0.95 rem, respectively.

I. INTRODUCTION

The LIBRA-SP[1] reactor is driven by 7.2 MJ of 30 MeV Li ions. The target yield is 589 MJ and the repetition rate is 3.88 Hz. A strong emphasis has been given to the environment and safety issues in the LIBRA-SP reactor design. Modified HT-9 has been used in the blanket and reflector to avoid a high level of induced radioactivity in both regions. Similarly, LiPb has been used as a coolant and breeder to eliminate 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 the reactor. To evaluate the possible radiological hazard to the public, we used a two step approach in calculating the possible off-site doses. 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 in our approach was to consider a set of pessimistic but rather credible accident scenarios for mobilizing and releasing the radioactive inventory.

Neutron transport calculations have been performed using the discrete ordinates neutron transport code TWODANT.[2] The neutron flux obtained from the neutron transport calculations has been used in the activation calculations using the computer code DKR-ICF.[3] The DKR-ICF code allows for accurate modeling of the pulsing schedule. 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. The decay gamma source produced by the DKR-ICF code is used to calculate the biological dose rates after shutdown using the DOSE[3] code. The activation results have been also utilized in the radwaste classification, and in the off-site dose calculations performed by the FUSCRAC3[4] code. Finally, the EPA code AIRDOS-PC[5] has been used to estimate the off-site dose due to the routine release of tritium.

TABLE I

Radioactivity After Shutdown

Time After Shutdown	Activity (MCi)		Decay Heat (MW)		BHP (km^3 air)	
	Blanket	Reflector	Blanket	Reflector	Blanket	Reflector
0	721	924	2.99	3.34	3.7×10^8	1.63×10^9
1 hour	620	684	2.18	2.88	8.9×10^7	8.02×10^7
1 day	498	407	0.46	1.09	8.2×10^7	5.54×10^7
1 week	480	175	0.42	0.18	8.01×10^7	3.2×10^7
1 month	449	147	0.39	0.14	7.5×10^7	2.7×10^7
1 year	307	63	0.22	5.63×10^{-2}	3.7×10^7	9.24×10^6
10 years	29	6.5	1.23×10^{-2}	1.05×10^{-2}	1.05×10^6	2.0×10^6
100 years	2.78×10^{-3}	8.2×10^{-3}	1.39×10^{-6}	2.72×10^{-6}	1.02×10^4	4.1×10^4

II. CHAMBER ANALYSIS

The reactor has been modeled in spherical geometry with a point source at the center of the chamber. The reactor blanket and reflector are made of a modified HT-9. The chamber is surrounded by a steel-reinforced concrete biological shield. The shield is made of 70% concrete, 20% mild steel and 10% helium coolant. A separate activation calculation has been performed for the LiPb coolant. The residence time of the LiPb coolant in the chamber is 5 seconds. The total inventory of LiPb takes 15 seconds to go through the reactor chamber. Therefore, the coolant activity has been calculated to allow for the fact that LiPb spends only 33% of the time exposed to neutrons inside the reactor chamber.

Table I compares the activity, decay heat and biological hazard potential (BHP) in the blanket and reflector regions. Most of the steel-reinforced concrete shield activity is due to its steel component. At shutdown, the total activity amounts to 41.2 MCi and drops to 6.9 MCi within a day. On the other hand, due to the rapid decay of ^{207m}Pb ($T_{1/2}$ = 0.8 s), the activity of LiPb drops from 17830 MCi to 370 MCi within a minute from shutdown. The LiPb activity drops to 110 and 0.35 MCi during the first day and first year following shutdown, respectively. In general, the decay heat and biological hazard potential are dominated for the most part by the same nuclides that dominate activity. The integrated decay heat generated during the first day following shutdown in the blanket and reflector are 78 and 150 GJ, respectively. Only 13 GJ of decay heat is generated in the shield during the first month following shutdown.

Biological dose rate calculations have been performed at selected locations to assess the possibility of hands-on maintenance. Figure 1 shows the calculated dose rates as a function of time following shutdown. At all locations, ^{56}Mn ($T_{1/2}$ = 2.6 hr) and ^{54}Mn ($T_{1/2}$ = 313 day) dominate the dose rates 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, hands-on maintenance is impossible anywhere inside the reactor chamber. Only remote maintenance is possible behind the biological shield. Increasing the shield thickness should allow for the possibility of hands-on maintenance. A limit of 25 μSv/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.

III. DIODE ANALYSIS

Two-dimensional coupled neutron-gamma transport calculations are performed using the code TWODANT. Figure 2 shows the two-dimensional model used in the calculations. The diode casing, cathodes and anodes are assumed to be made of type 304 stainless steel. The magnet is made of copper and uses epoxy as an organic insulator.

The biological dose rate values due to streaming neutrons at the outer casing of the diode (z = 950 cm) and near the transmission lines (z = 980 cm) are shown in Table II. The dose rates during the first few minutes following shutdown are dominated by ^{28}Al and ^{52}V ($T_{1/2}$ = 3.76 min) produced from ^{51}V (n,γ), ^{52}Cr (n,p), and ^{55}Mn (n,α) reactions. The high content of manganese in the steel chamber results in ^{56}Mn ($T_{1/2}$ = 2.578 hr) being the major contributor to the dose rate up to one day. Even though most of the ^{56}Mn is produced as a result of the ^{55}Mn (n,γ) reaction, a significant amount is also produced by the ^{56}Fe (n,p) reaction. In the period between 1 day and 10 years, ^{54}Mn and ^{60}Co dominate the dose rate. Be-

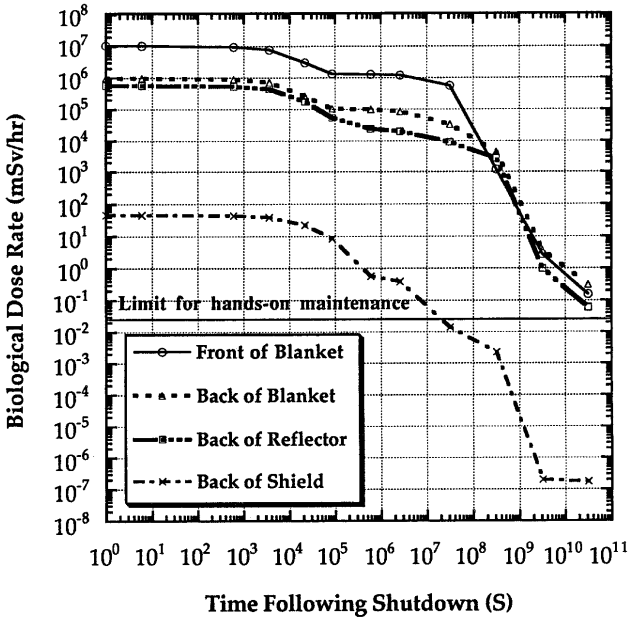

Fig. 1. Biological dose rates following shutdown.

yond ten years after shutdown, the dose rate is primarily dominated by radionuclides induced from the steel impurities. The two major contributors are ^{94}Nb ($T_{1/2} = 2 \times 10^4$ yr) produced from ^{93}Nb (n, γ) and ^{94}Mo (n, p), and ^{93}Mo ($T_{1/2}$ = 3,500 yr) produced from ^{92}Mo (n, γ) and ^{94}Mo $(n, 2n)$ reactions. Due to the high dose rates near the diode, access should be limited to remote maintenance.

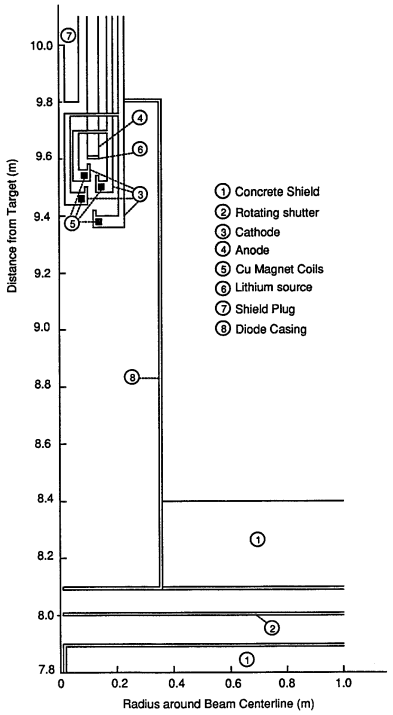

Fig. 2. The R-Z two-dimensional neutronics model.

TABLE II

Biological Dose Rates Near the Diode (mSv/h)

Time After Shutdown	Near Diode Casing	Near Transmission Lines
At shutdown	3.4×10^3	5.96×10^3
1 min	3.34×10^3	5.87×10^3
10 min	3.13×10^3	5.51×10^3
1 h	2.67×10^3	4.71×10^3
6 h	1.37×10^3	2.48×10^3
1 day	8.6×10^2	1.62×10^3
1 week	7.56×10^2	1.45×10^3
1 month	6.48×10^2	1.26×10^3
1 year	1.85×10^2	4.35×10^2
10 year	1.83×10^1	6.7×10^1
100 year	3.77×10^{-2}	5.57×10^{-2}

IV. RADWASTE CLASSIFICATION

The waste disposal ratings for LIBRA-SP have been evaluated according to both the NRC 10CFR61[6] and Fetter[7] waste disposal concentration limits (WDL). Table III shows the waste disposal ratings (WDR) for each of the reactor regions in the compacted and non-compacted (between brackets) forms. 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.

As shown in the table, the blanket may only qualify for disposal as Class C low level waste if it were disposed in its non-compacted form and only according to the 10CFR61 limits. The reflector could qualify for Class C waste in both forms according to 10CFR61 limits. Due to the low induced activity in the biological shield, it could qualify as Class A waste. 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. According to the 10CFR61 limits, LiPb could qualify for shallow land burial as Class A waste after extracting all the tritium. On the other hand, if Fetter limits are used, LiPb will not qualify for disposal as Class C waste. It is important to keep in mind that the waste disposal concentration limits used to calculate the WDR of the LiPb breeder are those assigned for the disposal of solid waste. Hence, LiPb has to be in solid form before such disposal can

TABLE III

Waste Disposal Ratings (WDR) of the Different Regions of LIBRA-SP

WDR	Blanket	Reflector	LiPb	Shield
Class A (10CFR61 limits)	40.5 (1.62) (^{94}Nb, ^{3}H)	8.5 (7.66) (^{94}Nb, ^{60}Co)	0.2 (0.064) (^{63}Ni, ^{60}Co)	(0.125) (^{14}C, ^{94}Nb
Class C (10CFR61 limits)	2.57 (0.103) (^{94}Nb, ^{14}C)	0.68 (0.61) (^{94}Nb, ^{14}C)	9.6e-4 (3.2e-4) (^{63}Ni)	(8.64e-3) (^{14}C, ^{94}Nb
Class C (Fetter limits)	41.5 (1.66) (^{192m}Ir, ^{158}Tb)	28.4 (2.56) (^{192m}Ir, ^{108m}Ag)	40 (13.1) (^{108m}Ag, ^{208}Bi)	(2.78e-3) (^{94}Nb, ^{14}C)

take place and the feasibility/practicality of such a process has to be determined.

V. OFF-SITE DOSES

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) to the maximally exposed individual (MEI). 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. Assuming the release parameters listed in Table IV, the dose at the site boundary was only 2.39 mrem/yr. Most of the dose is incurred via the ingestion pathway. The estimated dose is far below the current EPA effluent limit of 10 mrem/yr and less than the 5 mrem/yr limit adopted by ITER.

The other main source of potential off-site dose could be produced by the accidental release of part of the radioactive inventory in the containment building. In this paper 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.

During a loss of coolant accident (LOCA), a large increase in the steel structure temperature could lead to the mobilization and partial release of its radioactive inventory. Under adiabatic conditions, the decay heat generated within the first ten hours following a LOCA would increase the blanket temperature by < 400°C. Under the same conditions, the decay heat generated in the reflector would increase the temperature by < 100°C. Since the blanket and reflector peak operating temperatures are less than 625 and 400°C,

TABLE IV

Routine Atmospheric Release Parameters

• Emission Information	
Stack Height	75 m
Stack Diameter	30 cm
Momentum	1 m/s
• Tritium Pathways	
Reactor Building	12 Ci/day
Steam Generator	40 Ci/day
Fuel Reprocessing	26 Ci/day
Target Factory	12 Ci/day
Total (75% availability)	24,640 Ci/yr

respectively, the full mobilization of the structure radioactive products is extremely unlikely. The highest temperature a structure would reach determines the release fraction of its radioactive products. Off-site dose calculations have been performed using HT-9 experimental volatility rates[8] at 600 and 1000°C in dry air for the reflector and blanket, respectively. 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 reflector produces a whole body (WB) early dose at the site boundary of 253 mrem. Most of the dose is produced by the manganese isotopes, ^{54}Mn and ^{56}Mn. On the other hand, at 1000°C, the blanket would produce a WB early dose of 8.91 rem. In addition to the manganese isotopes, ^{32}P is the third major contributor to the dose. The decay heat generated within the first 2 months following a LOCA would only increase the steel-reinforced concrete shield temperature by < 3°C. The shield average operating temperature is less than 500°C. Since most of the radioactive inventory is contributed by mild steel, off-site dose calculations have been performed using adjusted PCA volatility rates at 600°C in dry air. At 600°C, the whole body early dose at the site boundary is 37.1

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

The steady state tritium inventory in LiPb is kept very low (3 g) by its continuous removal during the reactor operation. Following every fusion explosion, x-rays vaporize about 6.62 kg of LiPb. A simultaneous breach in the containment and chamber would allow the cold air to flow into the chamber. The air starts cooling the LiPb vapor and hence reduces its vapor pressure. As LiPb vapor pressure falls, it starts condensing rapidly. The LiPb alloy was selected because it does not react vigorously with air or water. Nevertheless, we performed the off-site dose calculation assuming that all of the 6.62 kg of the vapor LiPb is mobilized and released to the environment. The whole body early dose at the site boundary is 142 mrem. More than 65% of the dose is produced by tritium present in the LiPb at the onset of an accident. The rest of the dose is caused by ^{203}Pb, ^{110m}Ag and ^{210}Po.

The final source of potential off-site doses considered in this analysis is produced by the accidental release of the tritium from the reactor containment. The two sources of tritium inside the reactor containment are the target injector and breeder loops. While the target injector contains as much as 49 grams of tritium (one hour supply), the LiPb breeder present in the reactor system has a steady state inventory of only 2 g. The maximum amount of tritium that may be released from the reactor degassing unit and piping during an accident is 2 g. The whole body early dose produced by the release of all of the 51 g of tritium is 459 mrem.

The target factory produces a total of 337,000 targets/day. The daily amount of tritium processed in the factory is 1200 g. The total vulnerable inventory present in the factory at any moment and which could be released during a severe accident is only 147 grams of tritium. Similarly, the fuel reprocessing facility contains a 2-hr supply of vulnerable inventory, or about 106 grams of tritium, which might be released during an accident. A 100% release of vulnerable tritium would result in WB early doses of 1.3 and 0.95 rem for the target factory and fuel reprocessing facility, respectively.

VI. CONCLUSIONS

The LIBRA-SP power reactor design has attractive environmental and safety characteristics. The reactor structure and breeder qualify for disposal as low level waste. Routine tritium release is low, resulting in an off-site dose on the order of 2.39 mrem/yr. During a severe loss of coolant accident, the maximum whole body early dose at the site boundary is 9.8 rem. The blanket results in 90% of the total off-site dose. Postulated accidents involving the fuel reprocessing facility and the target factory results in off-site doses of 0.95 and 1.3 rem, respectively.

ACKNOWLEDGEMENT

Funding was provided by Sandia National Laboratory and Forschungszentrum Karlsruhe.

REFRENCES

1. B. Badger et al., "LIBRA-SP — A Light Ion Fusion Reactor Design Study Utilizing a Self-Pinched Mode of Ion Propagation, Report for the Period Ending June 30, 1995," University of Wisconsin Fusion Technology Institute Report UWFDM-982, June 1995.

2. R. Alcouffe et al., "User's Guide for TWODANT: A Code Package for Two-Dimensional, Diffusion – Accelerated, Neutral Particle Transport," LA-10049-M, Los Alamos National Laboratory (March 1982).

3. D. L. Henderson and O. Yasar, "DKR-ICF: A Radioactivity and Dose Rate Calculation Code Package," UWFDM-714, Vol. 1, University of Wisconsin (April 1987).

4. L. J. Porter, "Upgrade of a Fusion Accident Analysis Code and Its Application to a Comparative Study of Seven Fusion Reactor Designs," PFC/RR-89-10, Massachusetts Institute of Technology (June 1989).

5. User's Guide for AIRDOS-PC, Version 3.0, EPA 520/6-89-035, U. S. Environmental Protection Agency, Office of Radiation Programs, Las Vegas, NV (December 1989).

6. Nuclear Regulatory Commission, 10CFR part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," Federal Register, FR 47, 57446 (1982).

7. S. Fetter, E. Cheng and F. Mann, "Long Term Radioactive Waste from Fusion Reactors," *Fusion Engineering and Design*, **13**, 239-246 (1990).

8. S. J. Piet et al., "Initial Experimental Investigation of the Elemental Volatility from Steel Alloys for Fusion Safety Applications," EGG-FSP-8459, Idaho National Engineering Laboratory (April 1989).