ACTIVATION ANALYSIS FOR A He/LiPb DUAL COOLANT BLANKET FOR DEMO REACTOR

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The objective of the Spanish national project TECNO_FUS is to generate a conceptual design of a DCLL (Dual-Coolant Lithium-Lead) blanket for the DEMO fusion reactor. The dually-cooled breeding zone is composed of He/Pb-15.7 ^{6}Li and SiC as liquid metal flow channel inserts. Structural materials are ferritic-martensitic steel (Eurofer-97) for the blanket and austenitic steel *(316LN) for the Vacuum Vessel (VV). The goal of this work is to analyze the radioactive waste production by the neutron-induced activation and the back-end of the blanket and the VV (SS316LN) materials (Eurofer, SiC, LiPb, and SS316LN). Furthermore, the radioactive waste production in the cryostat (SS316LN) and the bioshielding (concrete) has been estimated. Following the current approach to the back-end of the materials in fusion facilities, the radioactive waste has been subdivided according to the activity-level classification (EW, exempted waste, LILW, low and intermediate level waste, and HLW, high level waste) and according to the radiological complexity of operations (handling and cooling). The activation calculations have been carried out with the ACAB code.*

I. INTRODUCTION

The framework of this study is the Spanish national project TECNO_FUS, whose objective is to carry out a design of a Dual-Coolant Lithium-Lead (DCLL) blanket for a DEMO fusion reactor. Particularly, in this work the neutron-induced activation of the reactor materials is analyzed from the point of view of the waste management.

The blanket design is based on the C model of the European Fusion Power Plant Conceptual Study (PPCS[1]) with modifications to reactor parameters and plant systems specifications. The neutron wall loading (NWL) average is 2.1 MW/m^2 for the blanket first wall with 3450 MW of fusion power.

The blanket has a dually-cooled breeding zone with Pb-15.7 ^{6}Li (90% ^{6}Li enrichment) serving as breeder and coolant, and with pressurized helium as primary coolant. The reduced activation ferritic-martensitic (RAFM) steel, Eurofer-97, is used as structural material in the blanket and 316-LN austenitic steel in the vacuum vessel (VV). The flow channel inserts of the LiPb liquid metal are made of SiC serving as electrical and thermal insulator.

Following the current approach to the back-end of the fusion materials,[2] the activation calculations have been performed to obtain contact dose rate and decay heat. These quantities allow us to categorize the different materials according to the radiological complexity of operation and the activity-level classifications. Also the activity limit for the recycling in foundries and the clearance index has been taken into account for the cryostat and the bioshielding where these limits can be fulfilled.

The activation calculations have been carried out with the ACAB[3] code with EAF2007 libraries. The neutron fluxes from MCNPX have been processed with the MC2ACAB[4] code (modified version of TARTREAD[5]) that generates the ACAB input automatically.

II. METHODOLOGY

II.A. Neutronic Calculation and Geometry Model.

The neutronic model[6] (Fig. 1) is a simplified layered model generated with an Excel/CATIA interface and transferred to MCNP geometry format by MCAM.[7] Actually the divertor and coil materials of the design are not accurately enough defined and are not considered in the activation calculations.

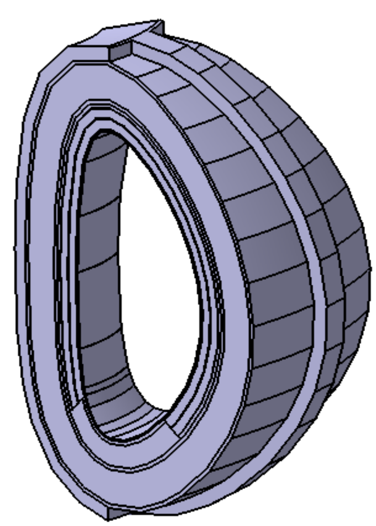

Fig. 1. Geometry of the MCNP neutronic model.

II.B. Activation Calculations.

II.B.1. Irradiation Scenarios.

In the activation calculations two irradiation scenarios have been assumed: i) a DEMO scenario with limited irradiation time, ii) a power plant scenario considering its lifetime:

- DEMO scenario: 20000 hours of continuous irradiation.
- Power Plant scenario:
  - ♦ 5 years of continuous irradiation for the blanket components (structure, Flow Channel Inserts and liquid metal) and 8 replacements during plant life time.
  - ♦ 40 years of continuous irradiation for the permanent components (Vacuum Vessel, Cryostat and Bioshielding).

II.B.2. Composition of the Materials.

The compositions of the materials (with impurities) Eurofer (specified composition), SiC and SS316LN are in Ref. 8-10. The LiPb used is Pb-15.7 ^{6}Li (using lithium enriched to 90 % ^{6}Li) with the impurities listed in Table I:

TABLE I. LiPb Impurities.

Element	ppm
Al	1
P	10
Ti	10
Cr	3
Fe	13
Ni	7
Cu	1
Ga	1
As	40
Ag	1
Cd	1
In	1
Sb	10
Au	1
Hg	1
Bi	200

II.C. Waste Management Criteria.

II.C.1. Radiological Complexity of Operation.

The actual way to analyze the radiological complexity of radioactive waste operation[2] takes into account two factors: handling and cooling. The handling limits are based on the dose limits that a worker or a machine can tolerate while handling the waste, and the quantity used for these limits is the contact dose rate (CD, in this work the semi-infinite slab approximation is assumed):

- Hands-on Handling (HOH): CD <10 μSv/h.
- Shielded Hands-on Handling (SHOH): 10 μSv/h < CD < 2mSvh.
- Remote Handling (RH): CD> 2mSv/h.

The Cooling limits are based on the cooling necessary for the safe storage of radioactive waste and the quantity of decay heat (DH):

- No-active cooling: DH* < 10 W/m^3.
- Dry cooling: 10 W/m^3 < DH < 2 kW/m^3.
- Wet cooling: DH > 2 kW/m^3.

The radiological complexity of operation can be divided in different levels (Table II) assigning different values to the previous criteria and summing them up.

TABLE II. Radiological complexity of operation.

Handling (H)	Cooling (C)	Difficulty	score (H+C)
HOH=1	NONE=0	level 1	1
SHOH=2	DRY=3	level 2	2
RH=3	WET=5	level 3	3,4,5
		level 4	6,7,8

Level 1 materials can be handled hands-on and no cooling is required. Level 2 materials can be handled using shielded hands-on methods and again no cooling is required. In level 3 materials require remote handling and/or dry cooling. Finally, level 4 materials require active cooling and no operation of material is possible.

II.C.2. Activity-level Classification.

The activity-level classification is taken from IAEA.[11] In our case we use a classification based only on the clearance index and the decay heat.

- Exempted Waste (EW): below the Clearance limit.
- Low and intermediate level waste (LILW): DH < 2 kW/m^3.
- High level waste (HLW): DH > 2 kW/m^3.

III. WASTE ASSESSMENT

III. A. Blanket and Vacuum Vessel.

In order to classify the radioactive waste within the above mentioned criteria, the contact dose and the decay heat (rates) have been analyzed for the different materials of the blanket and the VV. The critical isotopes for these quantities have been also identified at different cooling times.

In the following subsections the most critical zones for each material are analyzed. These zones are the first layers (in radial direction) of Eurofer (blanket first wall, FW), SiC (breeding zone), LiPb (breeding zone) and SS316LN (inner wall of the VV).

III.A.1. Eurofer.

For the first wall of Eurofer the contact dose meets the SHOH limit before 100 years of cooling time but does not meet the HOH limit in a reasonable time (more than 1000 years). The dry cooling limit is fulfilled after less than five years and active cooling is not needed after the time interval between 10 and 50 years.

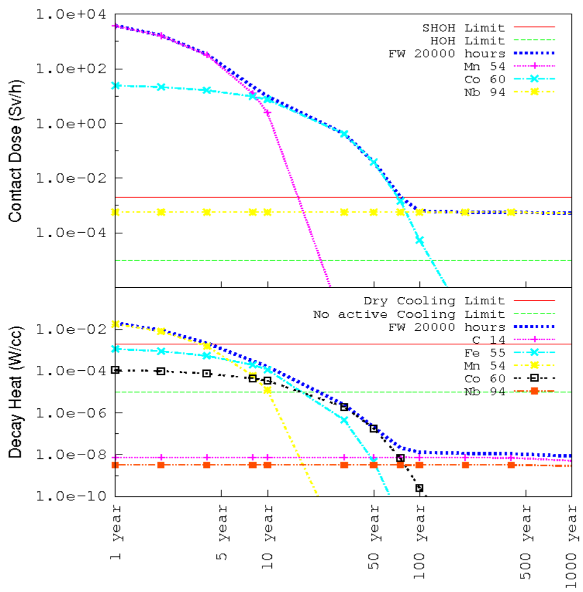

Fig. 2. Contact Dose and Decay Heat for the first wall of Eurofer under DEMO scenario.

The critical isotopes for the HOH limit are Co 60 up to 100 years and Nb 94 after this time. The latter does not allow us to fulfill the HOH limit in a reasonable time (even after 1000 years).

III.A.2. SiC.

Commercial SiC (Fig. 3) meets the SHOH limit before 50 years of cooling time and always is in the no-active cooling regime.

The critical isotopes for the HOH limit are Co 60 up to 100 years and Nb 94 after this time. Like in EUROFER, Nb 94 does not allow us to fulfill the HOH limit in reasonable time (even after 1000 years).

III.A.3. Lithium-Lead.

The contact dose (Fig. 4) does not meet the SHOH limit in a reasonable time (even after 1000 years). The decay heat is always in the dry cooling zone before 100 years of cooling time, while after that period of time it is in the no–active cooling zone.

The critical isotopes for the SHOH limit are Bi 207 up to 100 years and Bi 208 after this time. The latter is the most critical because it does not allow meeting the SHOH in a reasonable time (even after 1000 years).

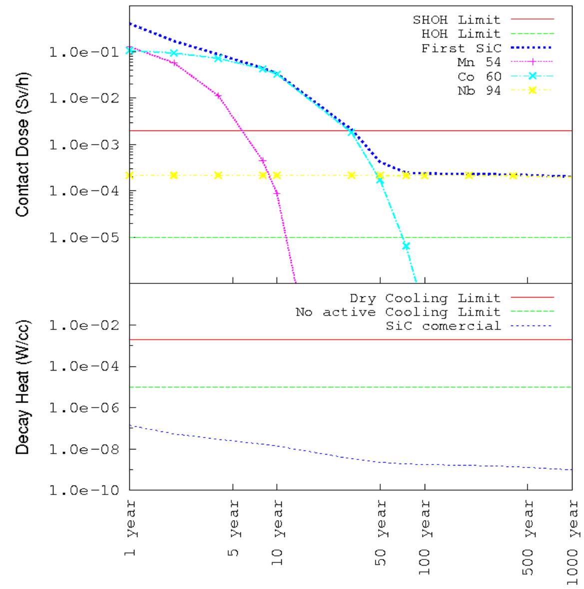

Fig. 3. Upper: Contact dose of the first layer of SiC under DEMO scenario. Lower: Decay heat for the first layer of SiC under DEMO scenario.

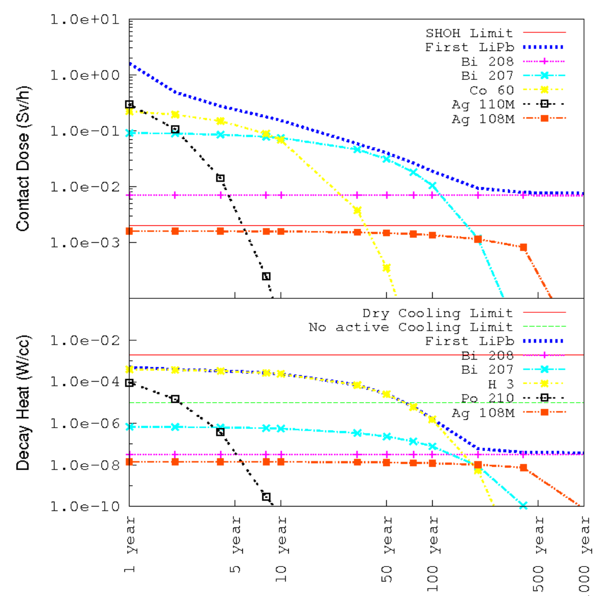

Fig. 4. Contact Dose and Decay Heat for the first layer of lithium-lead under DEMO scenario.

III.A.4. SS316LN.

For DEMO irradiation scenario simulated for the SS316LN steel located in the inner wall of the VV (Fig. 5 and 6) the SHOH limit is fulfilled after around 100 years of cooling time, the no-active cooling requirement is achieved after 50 years (Fig. 6). The value of decay heat in the steel is always below the dry cooling limit. For plant scenario (Fig. 7) the SHOH is not fulfilled in a reasonable time (even after 1000 years) while the no-active cooling is fulfilled around 50 years.

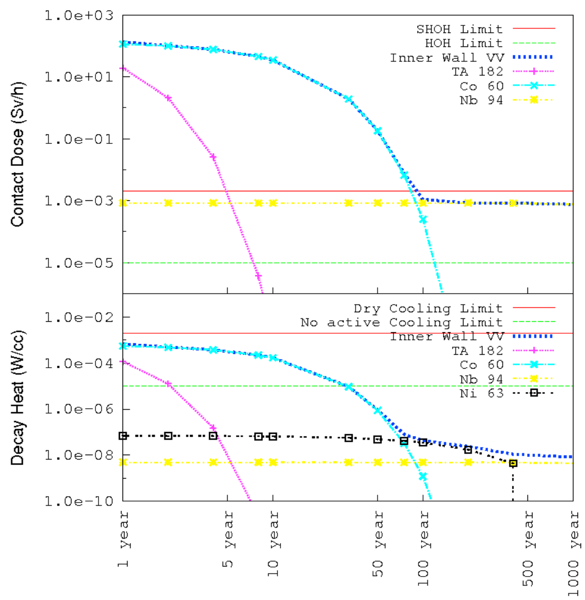

Fig. 5. DH and CD in the VV inner wall under DEMO scenario.

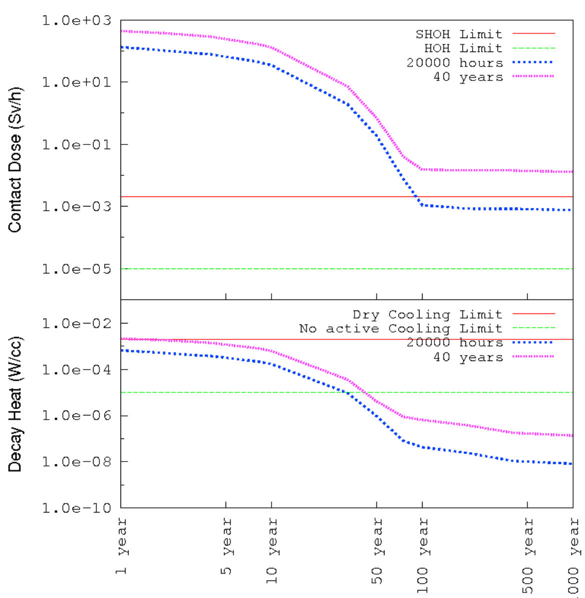

Fig. 6. The Vacuum Vessel inner wall in both scenarios DEMO and Power Plant.

For the contact dose, the most critical isotopes are Co 60 up to 100 years and Nb 94 beyond this time. For the decay heat, they are Co 60 up to 100 years, Ni 63 from 100 years to 400-500 years and Nb 94 beyond this time.

III.B. Cryostat and Bioshielding.

The cryostat and the bioshielding have been analyzed separately because they can be assessed using more aggressive waste management options (recycling in foundries and clearence) than for the blanket and VV components. The cryostat is made of SS316LN and fulfills the limit of 1000 Bq/g for recycle in foundries within 100 years of cooling time (10-50 years for DEMO scenario and at 100 years for power plant scenario, Fig.8).

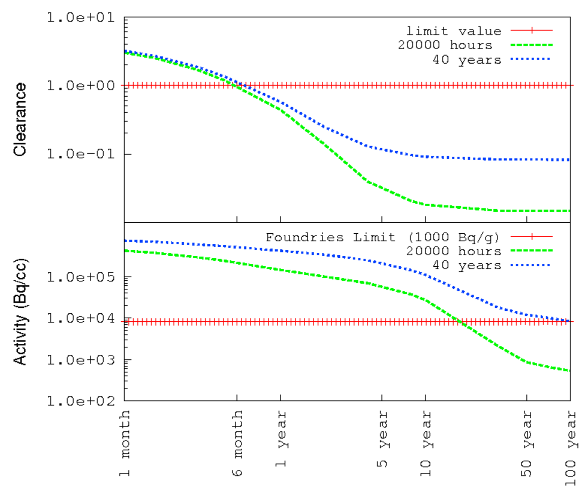

Fig. 7. Upper: clearance in the bioshielding. Lower: activity in the cryostat.

The bioshielding is made of conventional concrete with 50 cm of thickness and fulfills the clearance level before the first year in both irradiation scenarios (Fig. 8).

III.C. Management.

III.C.1. Radiological complexity of operation.

In Fig. 9 and 10 the distribution of the total mass for each level of radiological complexity of operation at 3 cooling times is showed for both irradiation scenarios.

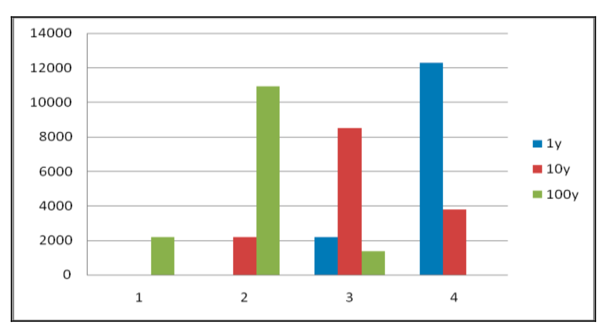

Fig. 8. Distribution of the total mass (in tons) for each level of radiological complexity of operation under DEMO scenario at 3 cooling times (1, 10 and 100 y).

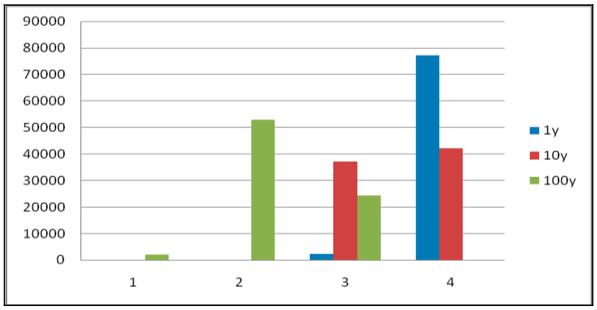

Fig. 9. Distribution of the total mass (in tons) for each level of radiological complexity of operations under plant scenario and 3 times of cooling time (1, 10 and 100 y).

All produced wastes are in 1-3 levels after 100 years of cooling time in both irradiation scenarios, avoiding the troublesome level 4.

III.C.2. Activity-level classification.

Following the IAEA classification above described the table III shows the mass in tons of LILW and HLW for the materials of the blanket and the VV. It can be seen that beyond 10 years of cooling time all the radioactive wastes are in LILW.

TABLE III. Mass in tons for both irradiation scenarios.

Level	1 y	10 y	100 y	1000 y
LILW (DEMO)	12256	12443	12443	12443
HLW (DEMO)	187	0	0	0
LILW (Plant)	61112	64335	64335	64335
HLW (Plant)	3223	0	0	0

IV. CONCLUSIONS

Assessment of the radioactive waste generated in a conceptual design of a DCLL blanket has been performed taking into account its radiological complexity of operations and its activity-level classification.

For both DEMO and power plant scenarios, all the radioactive wastes of the blanket and VV are LILW after one year of cooling time, and all the wastes are in 1-3 complexity of operation levels after 100 years of cooling time, avoiding the troublesome level 4. None of the materials that compound the blanket fulfills the limits of recycling in foundries and unconditional clearance.

The bioshielding is allowed for unconditional clearance before the first year of cooling for both irradiation scenarios and the cryostat can be recycled in foundries between 15-50 years for DEMO scenario and at 100 years for the power plant scenario.

Ongoing work on the study of this design with respect to waste management is focused on performing a refined analysis using a more realistic neutronic model and a thorough study about the recycling and disposal routes under a particular legislation like the Spanish one.

ACKNOWLEDGMENTS

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