INTERMEDIATE STORAGE AND LONG-TERM EVALUATION FOR GRAPHITE AND ALUMINUM WASTES RESULTED FROM A VVR-S TYPE RESEARCH REACTOR DECOMMISSIONING

A. O. Pavelescu[1*,2] and C. Tuca[1,2]

[1]Horia Hulubei National Institute for Physics and Nuclear Engineering, IFIN-HH, 30 Reactorului Street, P.O.B. MG-6, RO-077125, Magurele, Romania

[2]Academy of Romanian Scientists, (AOSR), Splaiul Independenței, No. 54, Sector 5, Bucharest, Romania

Abstract

The aim of our studies consists in the assessment of the options for intermediate storage of the graphite and aluminium radioactive waste resulted from VVR-S reactor decommissioning. The VVR-S nuclear research reactor from “Horia Hulubei” National Institute of Physics and Nuclear Engineering (IFIN-HH), Bucharest, Romania was put in operation in 1957 and finally shut-down in 1997 in order to be decommissioned. The main purpose was the radioisotope production in the thermal column for medical and industrial purposes as well as other research activities. Based on the preliminary radiological characterisation the appropriate decommissioning strategy was chosen consisting in the immediate dismantling of the contaminated and activated components and structures. Presently, for low and intermediate activity wastes such as aluminium (from reactor block) and graphite (from the thermal column), there are no suitable final storage technologies, therefore according to national legislation it is mandatory to store them into an intermediary waste repository. Based on our research the best option for radioactive waste intermediate storage until 2055 is the former spent fuel assemblies’ pools situated in the proximity of the reactor building. The nuclear spent fuel assembly’s repatriation in Russian Federation was completed in 2012 and the radioactive liquid effluents emptied in order to be treated. Consequently, the graphite rings were stored inside the pools which were previously decontaminated. The activated aluminium wastes were put into 220 l stainless steel containers hosted inside of 260 l cast iron containers, for safety reasons and stored inside of the spent fuel pools hall. The evaluation of the interim storage option is made according to the IAEA methodology for low and intermediate activity waste repository using the AMBER computer code.

Key words: decommissioning, aluminium/ graphite radioactive waste, intermediate storage, long term repository evaluation.

1. INTRODUCTION

The VVR-S nuclear research reactor with thermal neutrons from IFIN-HH, Romania (see Figure 1) was operated between 1957 and 1997 at a nominal thermal power of 2 MW and $2x10^{13}$ n/cm^2s maximum neutron flux and generated 9.59 GWd thermal energy. The reactor was designed for radioisotope production for medical and industrial purpose and research. Distillate water was used as coolant, moderator and reflector and low-enriched uranium (10%, EK-10 type) and the high-enriched uranium (36%, S-36 type) were used as fuel [1]. Due to the safety reasons the reactor was permanently shut-down in 2002 for decommissioning, which started in 2010 and will be completed in 2020. One of the main tasks in the last phase of decommissioning, consists of reactor block biological shielding dismantling, performed between May 2017 and March 2018. The shielding is a concrete cylinder with a 6 m height above the ground level, 0.5 m below ground and a diameter of 6m, containing embedded steel and cast-iron structures.

2. CURRENT STATE OF THE DISPOSAL

Radioactive wastes generated from decommissioning activities are considered low or intermediate activity level wastes (LILW) according to the nuclear legislation [2]. Every decommissioned component is dismantled and cut into pieces small enough to be securely placed in a barrel (waste form) of stainless steel or carbon steel with a volume of 220 litters considering the level of activity, material type and the selected treatment/disposal solution. The barrels are measured using gamma spectrometry in a dedicated lab of the Reactor Decommissioning Department (DDR) before being intermediately stored or transferred to the Radioactive Wastes Management Department (DMDR). Aluminium wastes are stored after dismantling in 220 l stainless steel barrels (Fig. 1), which are later placed in special cast iron containers of EKOL Rad type due to radiological security reasons

Fig. 1 - Stainless steel barrel (220 l) inserted in EKOL Rad cast iron barrel (260 l)

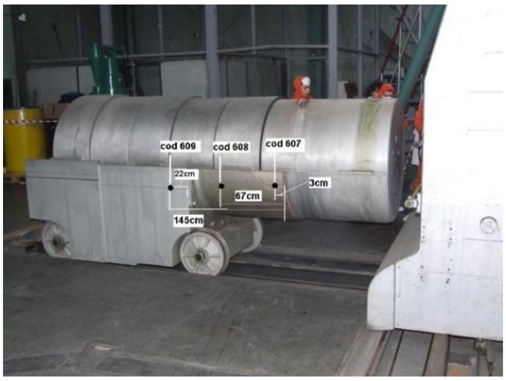

Fig. 2 - Graphite column made of graphite discs

The activated graphite wastes consist of the discs of the thermal column (Fig. 2). They are intermediary stored in the former Nuclear Spent Fuel Disposal Facility (DCNU), located near the reactor building. Presently, DCNU is authorized by the national regulatory body for nuclear activities (CNCAN) for Phase 1 of the Decommissioning. Starting with December 2012, the Spent Fuel Disposal Building (DCNU) (Fig. 3) is completely free of spent fuel, which has been transferred back to the Russian Federation. The storage water from the spent fuel pools has also been evacuated and transferred to the Radioactive Wastes Management Department for treatment and the pools have been decontaminated (Fig. 4).

Fig. 3 - Former spent fuel disposal building

Fig. 4 - Former spent fuel pool

3. ACTIVITY INVENTORY OF THE SOLID RAD WASTES FROM THE REACTOR BLOCK

The activity inventory for the solid wastes resulted from the Reactor Block in the 2010÷2017 period is presented in Table 1 [3].

Table 1. Activity inventory of the solid rad wastes from the reactor block

Material	**Mass [kg]**											**Activity [Bq]**									
Radionuclide		^{3}H	^{14}C	^{60}Co	^{55}Fe	^{63}Ni	^{59}Ni	^{108m}Ag	^{125}Sb	^{134}Cs	^{137}Cs	^{143}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{192}Ir	^{196}Au	^{235}U+D	^{238}U+D	^{241}Am	$A_{material}$ [Bq]
Aluminium	**2.92 E+03**	-	-	**2.14 E+09**	**1.31 E+10**	**6.43 E+10**	**7.07 E+08**	**3.08 E+03**	**2.85 E+05**	**2.35 E+04**	**1.75 E+08**	-	**3.25 E+07**	**3.03 E+07**	**1.79 E+06**	-	-	**5.74 E+05**	**4.99 E+06**	-	**8.05 E+10**
Metallic wastes	3.38 E+03	-	-	7.52 E+10	1.65 E+10	8.27 E+09	-	6.20 E+03	-	9.97 E+03	1.54 E+10	2.74 E+03	9.29 E+04	1.43 E+04	-	-	1.51 E+07	-	-	-	1.15 E+11
Plastic	3.47 E+03	-	-	3.45 E+06	-	-	-	1.01 E+04	-	2.27 E+04	5.69 E+05	-	6.52 E+04	2.91 E+04	-	4.36 E+03	-	-	3.78 E+04	3.74 E+03	4.19 E+06
Graphite	**5.30 E+03**	**8.62 E+09**	**2.65 E+09**	**1.63 E+08**	**1.03 E+08**	**2.58 E+07**	-	-	-	-	-	-	-	**3.13 E+06**	-	-	-	-	-	-	**1.16 E+10**
Total [Bq]	1.51 E+04	8.62 E+09	2.65 E+09	7.75 E+10	2.97 E+10	7.26 E+10	7.07 E+08	1.94 E+04	2.85 E+05	5.61 E+04	1.56 E+10	2.74 E+03	3.27 E+07	3.35 E+07	1.79 E+06	4.36 E+03	1.51 E+07	5.74 E+05	5.03 E+06	3.74 E+03	2.07 E+11

4. SECURITY EVALUATION FOR A PROPOSED LILW REPOSITORY FOR THE ALUMINIUM AND GRAPHITE

4.1. Conceptual model and mathematical modelling

Considering that the aluminium and graphite cannot be sent to the current Romanian radioactive waste geological disposal facility from Baita-Bihor located in former uranium mine due to impending legislation, they are intermediately stored in the DCNU facility near the reactor building. Therefore, in the paper we proposed a reconversion of the DCNU into a LILW near surface repository using multiple engineering barriers (Fig. 5) according to AIEA ISAM methodology [4]. The wastes barrels will be stabilized by cementing into a matrix formed of disposal cells. The cells are placed on an individual plate, having a pillow of compacted loess beneath. The cells will be covered with a covering system made of several isolating and draining layers having a long-term protection role against water infiltration from precipitations. [5]

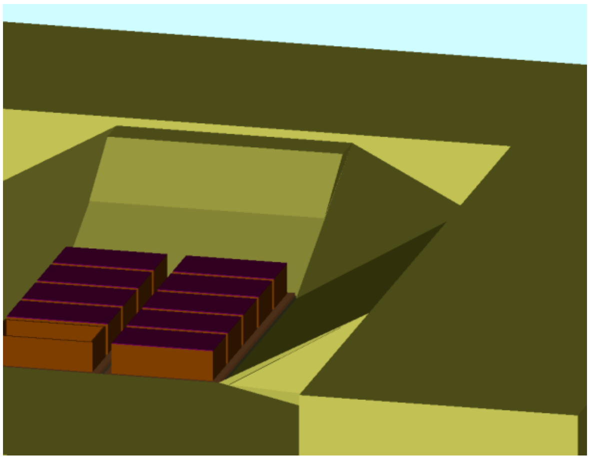

Fig. 5 - General view of the proposed LILW repository for the aluminium and graphite from VVR-S research reactor decommissioning

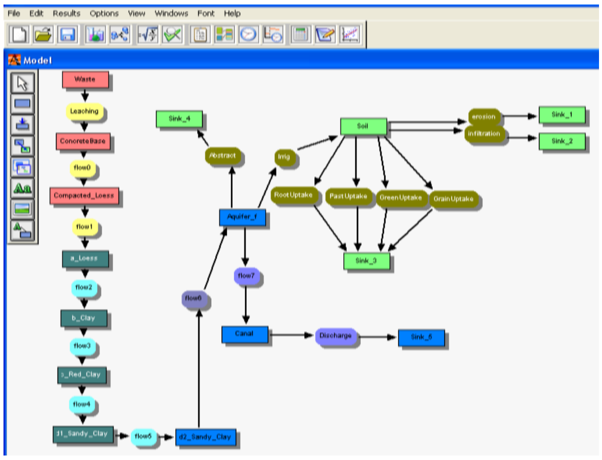

Fig.6 - Repository model used in AMBER 5.5 code.

For a reference repository post-closure scenario, a general conceptual model has been developed using AMBER 5.5 code (Fig. 6) The AMBER code is using an approach based on compartmental modelling for radionuclides migration and environmental behaviour.

For the i compartment, the speed with which radionuclides inventory is changing in time is given by the equation system 1 below: [6

$$\frac{dN_i}{dt} = \left(\sum_{j \neq i} \lambda_{ji} N_j + \lambda_N M_i + S_i(t) \right) - \left(\sum_{j \neq i} \lambda_{ij} N_i + \lambda_N N_i \right) \qquad (1)$$

where:

i and j – compartments;

N; M – radionuclides inventory (Bq) in compartment (M is the precursory of N in the decay chain);

S(t) – an external source, dependent on time of N radionuclide [Bq year^{-1}];

λ - transfer loss rate;
λ_N – decay constant of N radionuclide (year^{-1});
λ_{ji}; λ_{ij} – transfer coefficients (year^{-1}), representing contribution/loss of N radionuclide for i and j compartments. [7]

Solution of equation matrix (1), if solved for all the compartments and transfers of the system, is offering a time dependent inventory for each compartment. The hypothesis regarding compartment volumes allow the concentration evaluation for the given environment, and consequently the estimation of the received dose rates.

4.2. Evolution scenario

It is considered that near the repository location, in the post closure period there is a farm where plants are cultivated (e.g. cereals, green leaves vegetables and roots) and animas are grown (e.g. cow for milk). The scenario involves the following processes and phenomena: migration of radionuclides from the repository after its closure, through geological layers down to a local aquifer which feeds a fountain [8]. The water from the fountain can be used for drinking or irrigation in the vegetable garden, animals are taken to a non-contaminated pasture, but are given partially contaminated water from the farm fountain. The scenario critical group consists of farmer’s family. The reference person considered in evaluations is an adult male. Exposure pathways are: contaminated water consumption, contaminated vegetables, meat intake and contaminated milk drinking. [9]

5 Results

Evolution of total dose for the reference person in the proposed scenario calculated for every relevant radionuclide for the activity inventory of aluminium and graphite resulted from the IFIN-HH VVR-S research reactor decommissioning for all the exposure pathways has been evaluated for a 500-year time period since repository closure. Results are presented in Fig. 7-10. As it can be seen in Fig. 7, the exposure to C-14, an important component of activated graphite, is reaching a maximum towards the end of the life time of the repository after 500 years, but the maximum value of circa 30 µSv/year is lower that the legal dose constrain for the reference person of 50 µSv/year allowed for the decommissioning activities of the VVR-S reactor [1]. Therefore, we can conclude that the exposure is not significant, and the public is not affected by long term storage of this type of wastes. A similar situation is found related to the effects of environmental release of H-3 (tritium), another component of activated graphite. In this case the maximum dose (Fig. 8) for the reference person is registered after 150 years form the repository closure. The obtained value of 0.25 µSv/year is insignificant compared with the above-mentioned dose constrain of 50 µSv/year. Exposure of the critical group to the gradual environmental release of Co-60, a key nuclide for the activation products resulted from decommissioning (Fig. 9) and of Cs-137 (Fig. 10), a key nuclide for the fission products, both present in activated aluminium is insignificant for the entire lifetime of the repository.

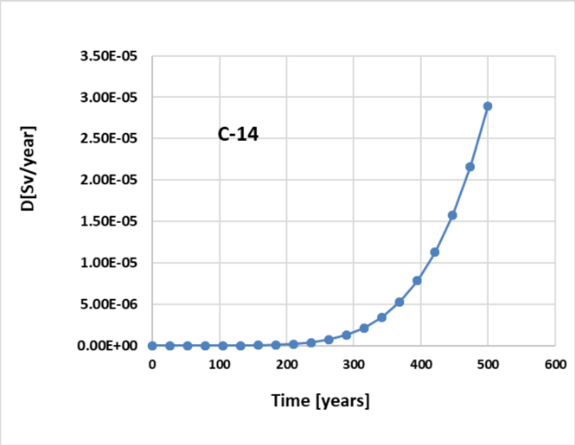

Fig. 7 - Total Dose for the refence person due to released C-14 exposure

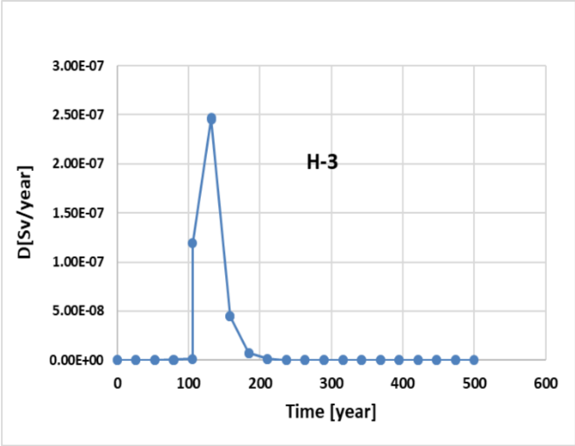

Fig. 8 - Total Dose for the refence person due to the released H-3 exposure

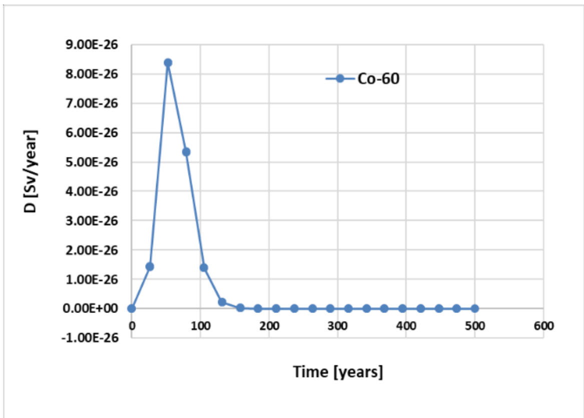

Fig. 9 - Total Dose for the refence person due to released Co-60 exposure

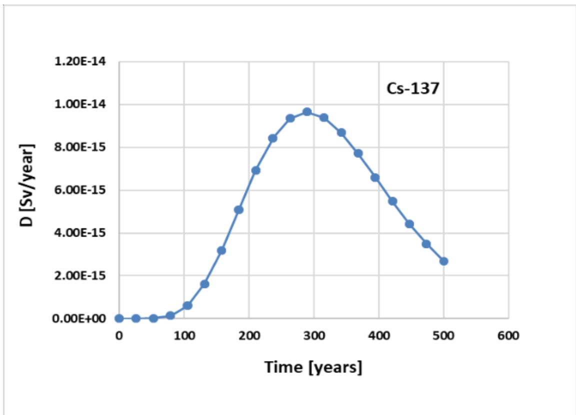

Fig. 10 -Total Dose for the refence person due to the released Cs-137 exposure

6. CONCLUSION

In a normal evolution scenario presented in the paper, the presence of a farm in the vicinity of the repository location, the reference person from the critical group, as well as the general public will not be affected in anyway. The potential annual doses, received by the exposed, calculated based on the real radioactive material inventory with the help of AMBER 5.5 computer code validated by AIEA in various inter-comparison exercises, are much lower that the decommissioning admissible limits for all the reference radionuclides contained in wastes of irradiated graphite and activated aluminium resulting from the decommissioning of the IFIN-HH VVR-S nuclear research reactor. Therefore, it can be concluded that proposed long storage solution of ILLW from graphite and aluminium resulted from decommissioning of a nuclear research reactor is viable, does not represent a danger for the public and could be implemented. In a highly improbable scenario involving a major disruptive event such as a powerful earthquake, major flooding or airplane impact explosion a supplemental safety analysis would be required.

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REFERENCES

[1]. C. Tuca, et.al. (2015) Assessment of derived emission limits for radioactive effluents resulted from the decommissioning activities of the VVR-S nuclear research reactor, J. Environ. Rad. 148 , 130-136*;
[2]. C.Tuca, A.Stochioiu, D.Gurau (2016) Analysis of radionuclides inventory contained in liquid effluents resulted from decommissioning of VVR-S nuclear research reactor, Rom. Rep. Phys. 68, no. 3 (2016), 1048–1059;
[3]. E. Ionescu, D. Gurau, D. Stanga, O.G. Duliu (2012) Decommissioning of the VVR-S research reactor – Radiological characterization of the reactor block, Romanian Reports in Physics, Vol. 64, no. 2, P. 387–398;
[4]. Safety Assessment Methodologies for Near Surface Disposal Facilities, Results of a co-ordinated research project, IAEA, 2004, ISAM Project;
[5]. AEA TECDOC No.1380, Derivation of activity limits for the disposal of radioactive waste in near surface disposal facilities, Vienna, 2003;
[6]. AMBER Computer Code v.5, Reference Guide, Enviros and Quintessa Ltd, UK, 2006;
[7]. ICRP, 2012. Compendium of Dose Coefficients based on ICRP Publication 60. ICRP Publication 119. Ann. ICRP 41(Suppl.);
[8]. Improvement of Safety Assessment Methodologies for Near Surface Disposal Facilities. Results of a co-ordinated research project. Volume 1 Review and enhancement of safety assessment approaches and tools; Volume 2: Test Cases. IAEA-ISAM-2, Vienna, 2004;
[9]. Jin Beak Park, Joo-Wan Park, Eun-Young Lee and Chang-Lak Kim, Experiences from the source-term analysis of a Low and Intermediate Level Radwaste Disposal Facility, WM'03 Conference, February 23-27, 2003, Tucson, Arizona, USA.

* Presenting author, e-mail: alexandru.pavelescu@nipne.ro