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# RADIATION DOSES ASSESSMENT IN SOME OPERATIONAL SCENARIOS FOR RADIOACTIVE WASTE DISPOSAL IN A NEAR SURFACE REPOSITORY, BY USING MONTE CARLO METHODOLOGY*

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*Abstract.* Near surface disposal is an option used by many countries for the disposal of radioactive waste containing mainly short lived radionuclides and low concentrations of long lived radionuclides. This disposal option requires design and operational measures to be provided for the human health and the environment protection, both during disposal facility operation and following its closure. The goal of this paper is to apply the Monte Carlo methodology to radiological impact estimation for some scenarios characterizing the near surface repository operational phase. Two operational scenarios were considered, namely: a normal operation scenario (workers exposure to external irradiation due to the stored waste packages) and an abnormal operation scenario (workers exposure to external irradiation due to the dropping and crushing of a waste package during storage into the disposal cell). For both scenarios, the paper follows to estimate, by means of the Monte Carlo MORSE-SGC shielding code, the radiation doses collected by the operator sitting in the gantry cab and transporting the waste package, and, also, by the workers sitting on the ground level in the disposal cell and operating the waste package, due to the external exposure.

*Key words:* near surface disposal, waste package, disposal cell, operational scenario, external exposure, radiation dose.

## 1. INTRODUCTION

At present, in the Romanian Government policy, the nuclear energetic is considered as optimal solution for the national energetic needs. In 2002, the Government decided to approve the Nuclear National Strategy for the next 50 years [1]. The Fundamental Objective specifies that between 2025–2050, Romanian NPPs must provide (20–40)% from the total electricity generated in Romania, while respecting both of the competitive costs conditions and the nuclear safety assurance at international agreed standards.

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Romania has only one nuclear power plant, Cernavoda NPP, equipped with 5 reactors PHWR CANDU 6 type, CANDU 6 standard series – 706 MW(e) each. Unit 1 is in commercial operation since December, 1996, Unit 2 reactor reached first criticality on May 7, 2007 and Unit 3 is under construction, the rest of two units being under preservation stage. In almost 11 years of commercial operation, Cernavoda NPP Unit1 has given ~10% of the total electricity produced in Romania (Fig. 1) [2], this percent assuming to be around 18% after 2007 (with both Unit1 and Unit 2 in operation).

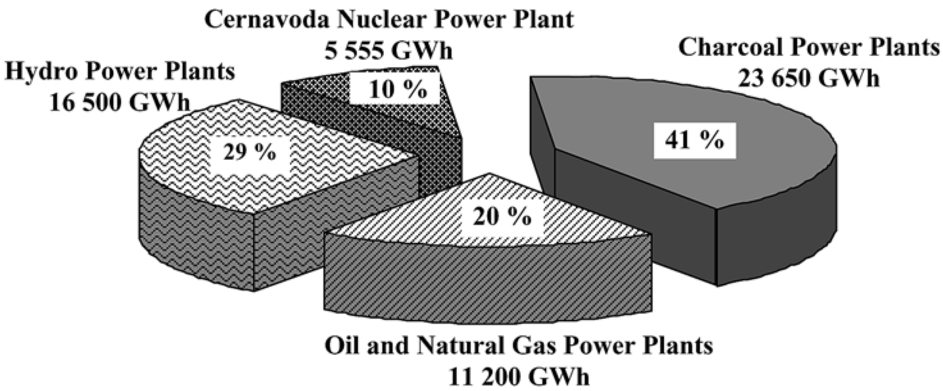


Fig. 1 – Coarse Power structure in Romania, end of 2005 [2].

At the end of 2006, Unit 1 registered an average capacity factor of 91.37%, being the forth with the other worldwide CANDU NPPs. Over the year 2006 the Cernavoda NPP-Unit 1 generated an amount of electricity of 5,631 GWh, out of which 5,118 GWh were supplied to the electric power [3]. The electric energy price was 23 USD/MWh, being competitive with the one corresponding to the classic power plants.

The National Agency for Radioactive Waste – ANDRAD was set-up in 2003 and its activity since then has been an important step in the start-up of the radioactive waste decisional process.

The spent nuclear fuel sources in Romania are, as follows [4]: CANDU reactor Cernavoda NPP, TRIGA reactor and Post-Irradiation Examinations Laboratory from the Institute for Nuclear Research Pitești, VVR-S reactor from the "Horia Hulubei" Nuclear Institute for Physics and Nuclear Engineering Bucharest, IFIN-HH Bucharest (permanently shut-down in 1997, the Romanian Government decided in 2002 the decommissioning).

Other representative facilities for the Spent Nuclear Fuel management in Romania are: the National Repository for Low and Intermediate Radioactive Waste, DNDR Baita-Bihor; Fuel Fabrication Plant, FCN Pitești; Spent Fuel Intermediate Dry Storage Facility, DICA Cernavoda; the National Agency for Radioactive Waste, ANDRAD.

Near surface disposal is an option used by many countries for the disposal of radioactive waste containing mainly short lived radionuclides and low concentrations of long lived radionuclides. This disposal option requires design and operational measures to be provided for the human health and the environment protection, both during disposal facility operation and following its closure.

The near surface disposal is based on the multibarrier isolation concept in order to assure the imposed safety level. The multibarrier system consists of three major components: radioactive waste package, technical structures and geological environment.

The waste form and the waste package constitute the first barrier protecting against exposure to, or release of, radioactivity. The waste form usually consists of the waste material encapsulated in a stabilization matrix, inside of a stainless steel or concrete waste package; it serves to immobilize the radioactive material.

The second barrier consists of all the engineered structures realized between the waste package and the geological environment. In the near surface disposal case, this barrier includes: the concrete vault, the facility dome and the filling material (cementitious matrix or other matrices, depending upon the particular characteristics of the waste).

The natural geological environment surrounding the disposal facility is the third and most important component of the multibarrier system. If the radionuclides are released outside the disposal facility, their transport is controlled by the pathway length combined with the underground water speed and the physico-chemical characteristics of the geological environment, particularly the rock ability to delay the radionuclides displacement. The adequate site selection for the disposal facility implies the finding of an environment whose properties can furnish a good balance between all these factors.

After the disposal facility closure, the institutional control represents the main safety factor, decreasing the human intrusion probability.

The radiological protection objective during the operational phase of a disposal facility are the same as for any nuclear facility, namely: "The radiation doses to workers and members of the public exposed as a result of operations at the disposal facility shall be as low as reasonably achievable, and the exposures of individuals shall be kept within applicable dose limits and constraints."

The radiation dose limits and constraints for the authorized personnel and public have been settled out in Basic Safety Standards [5]:

- The occupational exposure of any worker shall be controlled so that the following limits are not exceeded: a ***20 mSv/y effective dose***, averaged on 5 consecutive years consecutively (meaning 100 mSv in 5 consecutive years) or a ***50 mSv effective dose*** in any single years.
- The averaged doses estimated for the public belonging to the critical group, must not exceed ***the effective dose of 1 mSv/y***.

The public exposure may be caused by many sources. Looking the above mentioned limit, the disposal facility must be designed such as the estimated dose to the public belonging the critical group exposed to radiations during the disposal facility operation, not exceed 0.3 mSv/y.

In all the stages of the disposal facility development the protection optimization is needed, in order to maintain the exposure radiation doses for the workers as low as possible.

Two operational scenarios were considered: OSce1 – a normal operation scenario (workers exposure to external irradiation due to the stored waste packages) and OSce2 – an abnormal operation scenario (workers exposure to external irradiation due to the dropping and crushing of a waste package during storage into the disposal cell), respectively.

The waste is immobilized in a concrete matrix, packaged in a 400 l standard metallic drum or a 290 l fiber reinforced-concrete container (CBF), and was modeled as a uniform radiation source, consistent with radioactive waste management national regulations.

After the waste package dropping and crushing on the ground, the concrete and the biological shielding are supposed to be destroyed and the waste spreads on a 1 $m^2$ area.

For both scenarios, the paper follows to estimate, by means of Monte Carlo MORSE-SGC shielding code [6], the radiation doses due to the external exposure in two exposure cases: EXP1 – the operator sitting in the gantry cab and transporting the waste package, and EXP2 – the workers sitting on the ground level in the disposal cell and operating the waste package, respectively.

## 2. SHIELDING PROBLEM GENERAL DESCRIPTION

The paper assumes that the wastes have been already treated and conditioned before their arrival to the disposal facility, taking into account just for the transport and storage in the repository disposal cells.

The disposal cell, a reinforced concrete vault, partially excavated in the ground, consisting of 7.5 m height × 80 cm thick external walls and 7.5 m height × × 60 m thick internal walls, respectively, is presented in Fig. 2 [7].

The waste is immobilized in a concrete matrix, packaged in a transport/storage container (400 l standard metallic drum or 290 l fiber reinforced-concrete container, CBF) outside the repository.

For each disposal cell two stages are considered, functionally speaking:

– filling/loading stage: waste containers are disposed on horizontal layers, each layer being immobilized in grout after completion;

– final disposal stage: once the disposal cell has been filled, concrete shielding is poured on top of the cell; upon completion of filling the disposal cell is closed by placing a reinforced concrete monolith plate and completing a hydroisolation.

Exposure to workers will be principally due to direct irradiation and especially in case of damage of the waste package during repository operation.

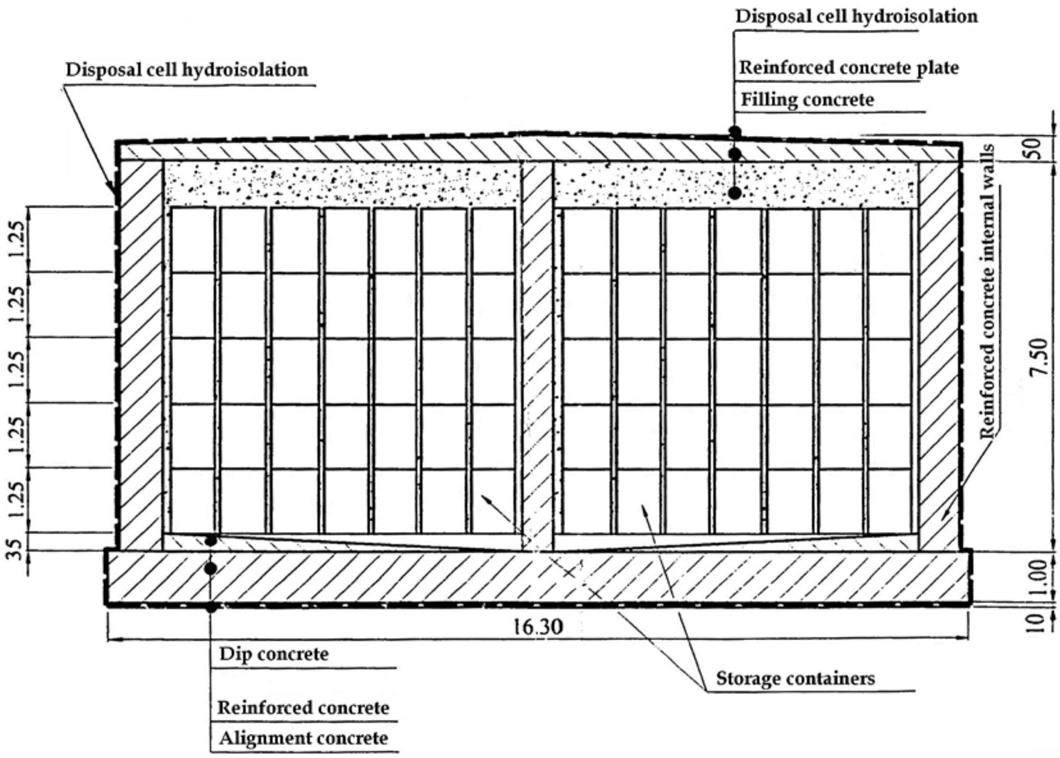


Fig. 2 – Near surface repository disposal cell [7].

## 3. CALCULATION ASSUMPTIONS

Two operational scenarios were considered:

– ***OSce1***: a normal operation scenario (workers exposure to external irradiation due to the stored waste packages)

– ***OSce2***: an abnormal operation scenario (workers exposure to external irradiation due to the dropping and crushing of a waste package during storage into the disposal cell).

The shielding problem initial assumptions were as follows:

– Waste packages that are transported and those that have been already stored in the disposal cell are intact, without any solid, liquid or gaseous contaminant releases (OSce1).

– The worker's exposure to radiation will be due to the direct irradiation due to the waste packages that have been already stored in the disposal cell (OSce1).

– After the waste package drops, only the considered waste package have been crushed, without damaging any other waste packages, already loaded in the disposal cell (OSce2).

– The worker's exposure to radiation will be due to the direct irradiation and, mainly only in the case of package damaging during the manipulation and radioactive material spreading inside the disposal cell (OSce2).

– Following the package crushing, the emergency has been duly solved by the authorized workers so that OSce2 doesn't imply the public leaving near the near surface repository.

– No intervention from the implied workers is present, in order to solve the emergency situation; after the package crushing, the workers urgently leave the area, allowing the authorized personnel to move in.

## 4. THE WASTE TRANSPORT/STORAGE CONTAINER

The radioactive material transport regulations [8, 9] specify that the radiation level under routine conditions of transport shall not exceed 2 mSv/h at any point on, and 0.1 mSv/h at 2 m from, the external surface of the conveyance.

The transport/storage containers used for the present shielding problem calculation were a 400 l metallic drum and a 290 l fiber reinforced-concrete container, CBF, respectively.

Both types of containers were modeled as a uniform source of radiation, in agreement with the national regulations regarding the safe transport of radioactive material. The radionuclides inventory is dominated by Co60 and the radiation source specific radioactivity was considered according to [10] by taking into account for the containers storage volume: $14.76 \times 10^{10}$ Bq/m$^3$ for CBF (about 660 kg radioactive wastes) and $20.36 \times 10^{10}$ Bq/m$^3$ for the metallic drum (about 910 kg radioactive wastes), respectively [11].

After the dropping, the waste package is broken, the concrete and biological shielding is damaged, and the waste is spreading on a 1 m$^2$ area; air density was considered $1.29 \times 10^{-3}$ g/cm$^3$ [7].

## 5. SCENARIO CHARACTERISTIC PARAMETERS

The paper follows to estimate the radiation doses collected by the operator sitting in the gantry cab and transporting the waste package, and, also, by the workers sitting on the ground level in the disposal cell and operating the waste package, due to the external exposure.

For OSce1, the following operations have been identified: container clamping in the gantry elevation mechanism; container transport to the disposal place, container unclamping from the gantry elevation mechanism. For each operation, corresponding minimum elevation height, minimum distance from the container and maximum exposure time have been estimated [11] (see Table 1).

For OSce2, the following operations have been identified: container clamping in the gantry elevation mechanism; container transport to the disposal place, dropping of the container, concrete and biological shield damage and waste spreading on the ground. For each operation, corresponding maximum elevation height, minimum distance from the container, minimum distance from the spread waste and maximum exposure time have been estimated [11] (see Table 1).

The exposure considered cases were: the gantry operator exposure (EXP1) and the workers exposure (EXP2), both due to external exposure.

Total radiation dose will be sum of collected radiation doses during the operations considered for each scenario. For different operations, the exposure case will be indexed by a, b c (*e.g.*, $EXP1_a$, $EXP1_b$ or $EXP1_c$ for exposure of the gantry operator during operation 1, operation 2 or operation 3, respectively).

*Table 1*

Operational scenarios characteristic parameters

| **Operation** | **Operational Scenarios** | |
|---|---|---|
| | *OSce1* | *OSce2* |
| *Operation 1* | Workers<br>$d_{min}$ = 0.3 m<br>$t_{max}$ = 60 sec<br>Gantry operator<br>$d_{min}$ = 2 m | Workers<br>$d_{min}$ = 0.3 m<br>$t_{max}$ = 60 sec<br>Gantry operator<br>$d_{min}$ = 2 m |
| *Operation 2* | Workers and gantry operator<br>$d_{min}$ = 2 m<br>$t_{max}$ = 6 min | Workers and gantry operator<br>$d_{min}$ = 2 m<br>$t_{max}$ = 5 min |
| *Operation 3* | Workers<br>$d_{min}$ = 0.3 m<br>$t_{max}$ = 60 sec<br>Gantry operator<br>$d_{min}$ = 2 m | Workers and gantry operator<br>$d_{min}$ = 1 m<br>$t_{max}$ = 10 sec<br>(before waste spreading)<br>$d_{max}$ = 7 m<br>$t_{max}$ = 20 sec<br>(after the waste spreading) |

## 6. RESULTS

The radiation doses have been obtained by means of the Monte Carlo MORSE-SGC shielding code, included in ORNL's SCALE5 package [6].

The shielding calculation will consist in:

• Dose rates estimation to the metallic drum/CBF wall and in air at different distances from the waste package, before the package drop and crush accident

• Radiation doses estimation for the exposure considered cases, in OSce1 and OSce2 conditions.

The dose rates values evolution with the measuring point in OSce1 conditions for both considered waste packages are presented in Table 2 and Fig. 3.

*Table 2*

Estimated dose rates in OSce1 conditions

| **Measuring point** | **Dose rate [mSv/h]** | |
|---|---|---|
| | *Metallic drum* | *CBF* |
| Waste package wall | 2.15E–02 | 1.44E+00 |
| 0.3 m in air | 1.27E–02 | 7.79E–01 |
| 1 m in air | 6.22E–03 | 2.62E–01 |
| 2 m in air | 2.71E–03 | 9.54E–02 |
| 5 m in air | 6.13E–04 | 1.90E–02 |
| 7 m in air | 3.26E–04 | 1.03E–02 |
| 10 m in air | 1.70E–04 | 5.13E–03 |

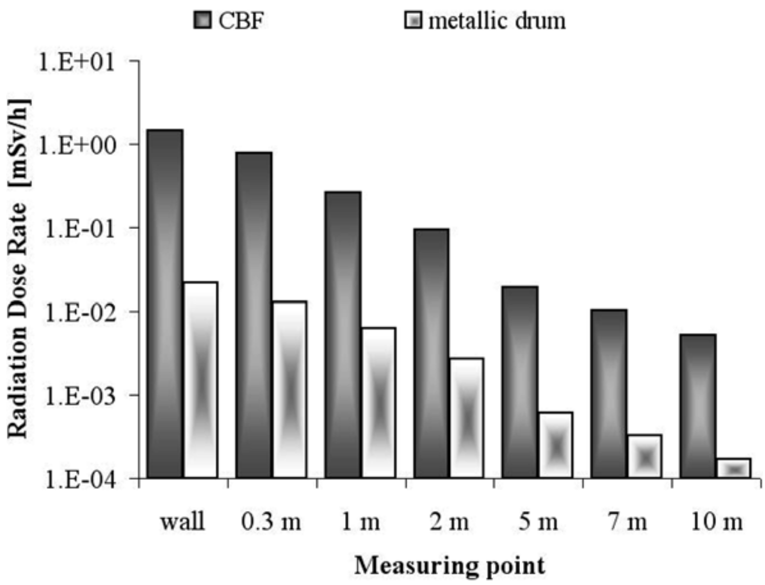


Fig. 3 – Dose rates evolution with the measuring point, OSce1.

The radiation doses estimated in OSce1 conditions, taking into account for the exposure characteristic parameters, for both considered waste packages are presented in Table 3. For different operations, the exposure case was indexed by a, b c (*e.g.*, $EXP1_a$, $EXP1_b$ or $EXP1_c$ for exposure of the gantry operator during operation 1, operation 2 or operation 3, respectively).

*Table 3*

Estimated radiation doses in OSce1 conditions

| **Exposure case** | **Radiation dose [mSv]** | |
|---|---|---|
| | *Metallic drum* | *CBF* |
| *EXP1a* | 4.52E–05 | 1.59E–03 |
| *EXP1b* | 2.71E–04 | 9.54E–03 |
| *EXP1c* | 4.52E–05 | 1.59E–03 |
| EXP1= sum of the above exposures | **3.61E–04** | **1.27E–02** |
| *EXP2a* | 2.12E–04 | 1.30E–02 |
| *EXP2b* | 2.71E–04 | 9.54E–03 |
| *EXP2c* | 2.12E–04 | 1.30E–02 |
| EXP2= sum of the above exposures | **6.94E–04** | **3.55E–02** |

The estimated dose rates values in OSce2 conditions, before the waste package drop and crush accident, are identical with those obtained in OSce1 conditions, based on the exposure characteristic parameters.

The radiation doses estimated in OSce2 conditions, taking into account for the exposure characteristic parameters, for both considered waste packages are presented in Table 4. As it was mentioned before, for different operations, the exposure case have been indexed by a, b, c (for operation 3 we have $c_1$ and $c_2$ in order to make the difference between conditions before and after waste spreading, respectively).

*Table 4*

Estimated radiation doses in OSce2 conditions

| **Exposure case** | **Radiation dose [mSv]** | |
|---|---|---|
| | *Metallic drum* | *CBF* |
| *EXP1a* | 4.52E–05 | 1.59E–03 |
| *EXP1b* | 2.26E–04 | 7.95E–03 |
| *EXP1$c_1$* | 4.19E–02 | 3.06E–02 |
| *EXP1$c_2$* | 1.14E–02 | 8.28E–03 |
| EXP1= sum of the above exposures | **5.36E–02** | **4.84E–02** |

(continues)

*Table 4* (continued)

| **Exposure case** | **Radiation dose [mSv]** | |
|---|---|---|
| | *Metallic drum* | *CBF* |
| *EXP2a* | 2.12E–04 | 1.30E–02 |
| *EXP2b* | 2.26E–04 | 7.95E–03 |
| $EXP2c_1$ | 4.19E–02 | 3.06E–02 |
| $EXP2c_2$ | 1.14E–02 | 8.28E–03 |
| EXP2= sum of the above exposures | **5.38E–02** | **5.98E–02** |

## 7. CONCLUSIONS

The results have demonstrated that, even in the event of operation accidents involving a breach of packaging, as in the drop and crush scenario, releases are likely to be mainly contained within the facility and doses to workers are not likely to be significant.

During normal operation of a near surface repository only very minor releases of radioactive material, if any (*e.g.*, releases of gaseous radionuclides), are expected, so that significant doses to the workers and members of the public are not anticipated.

According to radiation dose limits and constraints for workers settled out in the Basic Safety Standards [5], the occupational exposure of any worker shall be controlled so that the following limits are not exceeded: a 20 mSv/y effective dose over 5 consecutive years and a 50 mSv effective dose in any single year.

The radiation doses characterizing both the normal operation scenario (OSce1) and the waste package damage scenario (OSce2) were less than the limits settled out in the IAEA Basic Safety Standards.

Important and relevant considerations from a safety point of view include: separation of construction activities from waste emplacement activities; use of remote handling and shielded equipment for waste emplacement; control of working environments, reducing the potential for accidents and their consequences; minimization of maintenance requirements in radiation and contamination areas.

## REFERENCES

1. *Decision of the Romanian Government to approve the National Strategy for the Development of Nuclear Domain in Romania and the Plan of Actions to implement this Strategy*, (in Romanian), HG No. 1259, Bucharest, (2002).
2. Cernavoda NPP Direction, CNE Prod, http://www.cne.ro.
3. *Annual performance indicators*, Semnal "N", nr. 6, Nov.–Dec. (2006).

4. *Romania National Report, Joint convention on the safety of spent nuclear fuel management and on the safety of radioactive waste management*, (2003).
5. IAEA, FAO, ILO, OECD, WHO, *International Basic Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*, Safety Series no. 115, IAEA, Viena (1996).
6. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, ORNL/TM-2005/39, Version 5, Vols. I–III, April 2005. Available from RSICC at Oak Ridge National Laboratory as CCC-725.
7. IAEA, Derivation of activity limits for the disposal of radioactive waste in near surface disposal facilities, IAEA-TECDOC-1380, IAEA, Vienna (2003).
8. ***, *Fundamental Regulations for the Safe Transport of Radioactive Material*, NTR-01, National Commission for Nuclear Activities Control – CNCAN Order no. 373 (2001).
9. ***, *Regulations for the Safe Transport of Radioactive Material – 1996 Edition (Revised)*, TS-R-1 (SF-1, Revised), IAEA Safety Standard Series (2000).
10. G. DEMETTE, D. GAY, R. STORK, D. BUHMANN, *Preparatory Measures for the Long-Term Safety Assessment of the Low-Level Radioactive Waste Repository Baita Bihor*, Ctr. 006-RO/PHARE-SCR/A6-01, IPSN-France/GRS-Germany (2001).
11. C. A. Margeanu, *Radiological impact evaluation for a LILW Repository operational scenarios, using Monte Carlo Methodology*, INR Internal Rep. no 7569 (2006).