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Draft Safety Analysis Report for the Vrbina Krško LILW Repository

Section 7 Safety analysis

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ABBREVIATIONS AND TERMS

CSRAO: Central radioactive waste storage facility at Brinje

DCRLs: derived consideration reference levels

FEPs: features, events, processes

IAEA: International Atomic Energy Agency

ICRP: International Commission on Radiological Protection

IDP: preliminary design

IDZ: conceptual design

IJS: Institut Jožef Stefan

FP: final package

N2b: concrete container, final package into which 2 x 2 TTCs or an appropriate number of drums of other volumes can be loaded

Krško NPP: Krško nuclear power plant

LILW: low- and intermediate-level radioactive waste

ODEs: ordinary differential equations

osnVP – Draft Safety Analysis Report

PDF: probability density function

PIEs: postulated initiating events

RW: radioactive waste

RAPs: reference animals and plants

SA.2: investment scenario with participation of Croatia (disposal of all waste from Krško NPP and Slovenian institutional waste)

SA.3: basic investment scenario (disposal of Slovenian part of waste from Krško NPP and Slovenian institutional waste)

SSCs: structures, systems and components

TTC: tube-type container

7 SAFETY ANALYSIS

7.1 GENERAL FINDINGS

Safety analysis and assessments are an integral part of the lifecycle of a nuclear facility such as the LILW repository. [1] The first iteration of the security analysis was undertaken back in 2006 [2] within the framework of the Option Study. [3] The purpose of the analysis was to identify the optimal waste disposal concept for the Vrbina site in Krško. The waste disposal concept of underground silos excavated from the surface was proposed and later confirmed in the detailed plan of national importance (DPN). [4] The concept was then developed to the phase of Revision C of the conceptual design, [5] as described in Section 2.1 of this draft safety analysis report (DSAR).

The next iteration of the safety analysis and assessments was drawn up after the adoption of the DPN for the needs of obtaining the environmental consent. At this stage the safety analysis had two main objectives:

- assisting in the optimisation of the waste disposal concept,
- supporting the acquisition of the environmental consent (providing the requisite calculations and estimates for drawing up the draft safety analysis report).

This safety analysis has been conducted for a specific site (Vrbina in Krško), and for the developed waste disposal concept, namely near-surface underground silos.

The safety analysis is divided into two basic parts. They are:

- safety analysis during operation of the LILW repository, and
- safety analysis after closure of the LILW repository.

It will also be presented as such. Since 2010 the safety analysis has been drawn up by a consortium of partners consisting of ENCO (Austria), Intera (USA), Studsvik (Sweden), Facilia (Sweden) and Irgo (Slovenia). The complete work on safety analysis is collated in reports that follow a pyramid structure of documents at three levels.

Level 1: The document represents a general overview of the safety analysis conducted, [6] and is produced in Slovene and English.

Level 2: Consists of synthesis reports for individual areas in Slovene and English:

- Post-Closure Safety Assessment, [7]
- Operational Safety Assessment, [8]
- Acceptance Criteria, [9] and
- Optimisation Report. [10]

Level 3: Detailed reports in English.

The safety analysis for the phase of obtaining the environmental consent was begun back in 2011, and was based on data and the preliminary design. The optimisation and development of the project took place within the framework of safety analysis, where individual optimisations were analysed from the perspective of the impact on nuclear and radiation safety. Only optimisations that had a positive impact on nuclear and radiation safety were adopted.

A conservative approach was taken in the production of the safety analysis, the main purpose of which was to analyse the most adverse scenarios during operation and after closure of the LILW repository. The results thus obtained represent an envelope showing the repository's maximum possible impact on people and on the environment. In later phases of the project the safety analysis will be updated with new input data, which, given the project optimisation and the reduced unreliability and conservativeness of the safety analysis, will demonstrate a smaller potential environmental impact from the repository.

7.2 SAFETY ANALYSIS DURING OPERATION OF THE LILW REPOSITORY

The following section provides an overview and description of safety analysis during the operation of the LILW repository. The safety analysis is described in detail in the reports drawn up within the framework of the safety analysis conducted in 2012:

- Operational Safety Assessment Context Report, [11]
- System Description for Operational Safety Assessment, [12]
- Operational Safety Assessment Report on Scenarios, Models and Results of Calculations. [13]

A report summarising the aforementioned reports was also drawn up:

- Operational Safety Assessment. [8]

Because optimisation of the LILW repository project was undertaken during the preparation of the project documentation, [14], [15], [16], and the main key to operational safety is the optimisation that the conditioning of the waste for disposal is no longer undertaken at the repository site, the safety analysis was upgraded with regard to the operational safety of the repository and the following report was drawn up:

- Revised Operational Safety Assessment. [17]

Section 7.2 of this draft safety analysis report summarises the aforementioned reports.

7.2.1 ASSESSMENT CONTEXT: ASSESSMENT OF THE CONTENT OF THE SAFETY ANALYSIS DURING OPERATION OF THE REPOSITORY

The main objective of the safety analysis for the period of operation of the repository is to provide sufficient and adequate assurance for all stakeholders (government departments, regulatory bodies, the general public, the profession) in the LILW repository project that it has been planned and will be built and operated such that the facility's impact on the environment, on the public and on employees will be acceptable in all phases of its operation.

The key assumptions of the safety analysis during operation are:

- The repository site (presented in detail in Section 4 of this draft safety analysis report) of Vrbina lies 300 to 400 m east of Krško NPP. The area is currently used for agriculture.

It is bordered in the west by an apple orchard, and in the east by a closed municipal waste repository, where a municipal waste sorting facility is currently operating.

- The repository site is enclosed by a fence during the operational phase, and is envisaged to be so during the phase of long-term controls. The reference population group during the operation of the repository consists of individuals working in a field directly next to the perimeter for 2,016 hours per calendar year (252 eight-hour working days).
- The repository is designed as a near-surface LILW repository in low-permeability Miocene silt lying below a thin Quaternary aquifer. The repository will be built on an elevated plateau that protects the repository from the probable maximum flood. Only the remediation of potentially damaged FPs is conducted at the technological facility at the repository site; the conditioning of waste for disposal (conditioning of FPs) is not conducted at the repository (conditioning is envisaged at Krško NPP).
- All LILW is conditioned for disposal at Krško NPP, which is also responsible for transporting containers that have been conditioned for disposal to the repository. Disposal containers are used to prepare waste for disposal, as these provide for relatively easy transport and handling. The conditioning of all LILW at Krško NPP for disposal at the repository is allowed under Article 95 of the ZVISJV, [18] which permits the operator of a nuclear facility to store and process radioactive waste and spent fuel for the needs of the provider of the compulsory national public utility service of radioactive waste management, provided that it obtains the relevant licence from the authority responsible for nuclear safety. The process of agreeing these activities with Krško NPP has begun, but the agreement has not yet been formalised.
- The waste will begin being disposed of in the disposal silo at a depth of 55 m below the surface. All waste will be disposed of in N2b concrete containers. The design envisages ten layers of 99 containers.
- The Vrbina LILW repository is designed for the disposal of LILW generated in the operation and decommissioning of Krško NPP, from the Central Radioactive Waste Storage Facility, and from the decommissioning of the TRIGA research reactor.
- The repository is planned for the disposal of half of the waste from Krško NPP and other waste generated in Slovenia, with the option of expanding the repository for the disposal of all waste from Krško NPP.
- The inventory for the safety analysis was drawn up on the basis of the inventory report of 2015, [19], which was drawn up by the same consortium that drew up the safety analysis. It consists of the entire disposal inventory in line with the SA.2 scenario (all waste from Krško NPP and Slovenian institutional waste). In the case of disposal under the SA.2 scenario, this entails the disposal of all waste from the operation of Krško NPP that is generated by that time, the disposal of Slovenian institutional waste, the sealing of the silo, and the transition of the repository into the idle phase. A second silo is to be constructed a few years before the beginning of the second phase The first silo is sealed, and has no influence on the construction of the second silo.
- The principal technological processes that will be undertaken at the LILW repository are:
    - takeover of waste conditioned for disposal,
    - inspection of FPs (review of transport documentation, visual controls),
    - radiological controls of FPs (controls of package and vehicle at entry to supervised area),
    - transportation of FPs to disposal silo,

- disposal of FPs in silo (unloading of FPs from vehicle, release of FPs into silo, monitoring of disposal phase),
- filling of voids in silo (between individual FPs, and between FPs and silo wall), and construction of levelling layer for every two layers of containers, monitoring of void filling, and controls of levelling layer execution,
- replacement of drainage pipes, and controls of installation.

The operation of the repository is described in detail in Section 9.1.1 of this draft safety analysis report.

7.2.1.1 Legislative framework

The Ionising Radiation Protection and Nuclear Safety Act [18] is the principal legislative framework taken into account in the preparation of the safety analysis. The Decree on dose limits, radioactive contamination and intervention levels (UV2) [20] sets out the effective dose limits for employees and the public (and was amended in 2018 to harmonise the effective dose limits with decree BSS95/2013; the new decree will be taken into account in the safety report for obtaining a building permit). The dose limits taken into account are as follows:

- 20 mSv/year for employees,
- 1 mSv/year for a member of the public,

where the following equivalent doses should also be taken into account for employees:

- 500 mSv/year for hands, forearms, feet and ankles,
- 150 mSv/year for eye lenses, and
- 500 mSv/year for the skin,

together with the following equivalent doses for the public:

- 15 mSv/year for eye lenses, and
- 50 mSv/year for the skin.

Article 10 of the decree then stipulates that it is necessary to optimise protection against ionising radiation such that the exposure of the public as a whole and individual members of the public is as low as reasonably possible, having regard for economic and social factors.

7.2.1.2 Objectives of safety analysis during operation

The safety analysis during operation follows the recommendations summarised in the previous section, and compiled from calculations of the effective dose to which employees and members of the public (individually and collectively) could be exposed, for the normal evolution scenario during operation and scenarios of potential accidents during operation.

The calculated impacts were compared with the limits of 20 mSv/year for employees and 0.2 mSv/year for the public. The JV5 rulebook [21] sets a limit of 0.3 mSv/year for the public, although in its guidelines the local community required a limit on the impact of the repository (external radiation) at the perimeter of 0.2 mSv/year, which was also taken into account in the safety analysis (the impact of Krško NPP at the outside fence of Krško NPP was taken to be less than 0.5 µSv/year [22] and a total value of 0.2 mSv/year was therefore taken for the repository). In accordance with the JV5 rulebook, [21] the two values above represent the permitted dose for an individual after the closure of the repository, and the same values were taken for the period of operation in the safety analysis.

The basic objectives of the safety analysis during the operation of the repository were as follows:

- determining the annual effective dose for an individual employee as a result of the planned activities,
- determining the collective dose for employees within the framework of the activities planned for one year,
- assessing the effective dose for a member of the public as a result of external irradiation during normal operation, which takes account of the skyshine dose from the unsealed, uncovered silo, FPs during transport to the repository site and handling at the repository site, and waste that could be stored in the technological facility,
- determining the effective dose for an individual employee in the event of scenarios of altered evolution during operation of the repository,
- determining the effective dose for an individual member of the public in the event of scenarios of altered evolution during operation of the repository,
- determining the maximum permitted gaseous and liquid releases from the repository.

7.2.1.3 General methodological approach in the safety analysis during operation of the repository

The production of the safety analysis made use of the SADRWMS [23] (Safety Assessment Driven Radioactive Waste Management Solutions) method developed from the ISAM [24] methodology. Use was also made of the SAFRAN tool, [25] which was developed within the framework of this method.

The selected approach stipulated that the data used to produce the safety analysis should be as realistic as possible (where available), while conservative estimates were used when it was not available.

7.2.2 DESCRIPTION OF SYSTEM

The actual repository site is described in Section 4 of this draft safety analysis report, while operation is described in Section 9. The remainder of this section provides only basic information about the repository that is of importance to the production of the safety analysis.

7.2.2.1 Radioactive waste: inventory

A revalued inventory [19] was taken in 2015 for the production of the safety analysis for the operation of the repository. The updated inventory includes new findings, and a reduction in certain unreliable elements. For this reason it differs to a lesser extent from the inventory used to draw up the safety analysis after closure of the repository. The change in the inventory does not have a significant impact on the final results of the safety analysis. In the next phase (preparation of the safety report) the inventory will be additionally updated. There are six basic streams of LILW in Slovenia, which are defined in Section 7.3.3.2 of this draft safety analysis report. The most active of them is operational waste from Krško NPP, which was conservatively taken as the basis for assessing the inventory for the safety analysis during operation of the repository. The sole exception is certain waste from decommissioning (the reactor vessel), for which the method of disposal and transport will be defined later and taken into account in future safety analysis; the assumption is that the analysed cases are conservative enough to encompass these types of waste.

On this basis the inventory presented in the following table (Table 7-1) was taken for the purposes of the safety analysis (operational period).

The values in columns A and B represent the average total activity of individual radionuclides for all packages and all waste streams from the operation of Krško NPP, where column A represents the total activity of 99 FPs (each containing four TTCs or 12 standard drums, or an appropriate combination of TTCs and drums), which represents one layer of waste disposed in the silo. Column B corresponds to the average radioactivity of one TTC or three standard drums. Column C represents the total activity of an individual radionuclide in the "hottest" Type-2 TTC from the Krško NPP base (waste stream PRH2).

Column D represents the total activity of individual radionuclides in an FP under the assumption that it contains one hottest TTC with activity as presented in column C, and three TTCs with activity as presented in column B.

Column E represents the total activity (the average of the most active [hottest] wastes from the Krško NPP base) for individual radionuclides in one hot TTC.

Table 7-1: Isotopic composition and activity for individual radionuclides used in safety analysis during operation of the repository

Isotope	Activity [Bq]				
	A	B	C	D	E
Ag-108m	2.02E+08	5.11E+05		1.53E+06	
Ag-110m	7.23E+10	1.83E+08		5.48E+08	9.81E+08
Am-241	4.07E+08	1.03E+06	6.49E+07	6.80E+07	3.23E+07
Ba-140	3.64E+06	9.19E+03		2.76E+04	
Ce-141	3.58E+09	9.04E+06		2.71E+07	1.04E+11
Ce-144	2.45E+07	6.19E+04		1.86E+05	
Cm-242	3.73E+07	9.41E+04	5.94E+06	6.22E+06	1.86E+06
Cm-244	9.66E+08	2.44E+06	1.49E+08	1.56E+08	7.27E+07
Co-57	2.27E+12	5.74E+09		1.72E+10	7.46E+09
Co-58	2.47E+12	6.25E+09	2.11E+11	2.29E+11	9.00E+10
Co-60	3.85E+12	9.72E+09	5.52E+11	5.81E+11	1.09E+11
Cr-51	6.54E+09	1.65E+07		4.95E+07	
Cs-134	7.57E+11	1.91E+09	1.32E+11	1.37E+11	2.05E+10

Isotope	Activity [Bq]				
	A	B	C	D	E
Cs-137	2.36E+12	5.97E+09	3.98E+11	4.16E+11	1.96E+11
Fe-59	4.77E+11	1.21E+09		3.62E+09	1.67E+08
I-131	2.15E+06	5.42E+03		1.63E+04	
Mn-54	1.04E+11	2.63E+08	7.09E+10	7.17E+10	1.27E+10
Nb-94	1.17E+08	2.95E+05		8.85E+05	
Nb-95	1.27E+13	3.21E+10		9.64E+10	1.16E+09
Pu-238	1.82E+09	4.60E+06	2.97E+08	3.10E+08	1.47E+08
Pu-239	3.43E+08	8.66E+05	5.58E+07	5.83E+07	2.77E+07
Ru-103	1.46E+07	3.68E+04		1.10E+05	5.30E+08
Ru-106	3.68E+11	9.29E+08		2.79E+09	
Sb-124	1.59E+08	4.01E+05		1.20E+06	
Sb-125	3.36E+10	8.48E+07		2.54E+08	5.69E+09
Sn-113	2.19E+11	5.52E+08		1.66E+09	8.28E+08
Te-132	4.97E+07	1.25E+05	2.95E+08	2.95E+08	2.95E+08
Zn-65	4.02E+08	1.02E+06		3.05E+06	
Zr-95	1.15E+12	2.89E+09		8.68E+09	1.66E+09

The isotopic composition from column A was used to evaluate the impact of different layers of waste in the silo on employees' direct exposure in the silo and in the hall above the silo, and the exposure of a member of the public.

The isotopic composition from columns B, C and D was used to evaluate the impact of the FP on the external exposure of employees working at the repository, and to calculate the external exposure of a member of the public as a result of an FP disposed in the silo, or one damaged FP stored in the technological facility.

The isotopic composition from columns B and E was used to evaluate the impact of the FP on the external irradiation of a member of the public as a result of the transport of an FP inside the LILW repository site.

For the purposes of calculations of accident scenarios (abnormal operation), an "average" FP and a "hot" FP were defined similarly as for the normal operation scenario. Because under the accident scenarios (abnormal operation) other pathways of irradiation in addition to direct exposure arise, e.g. inhalation, ingestion, certain other radionuclides from the estimated inventory [19] were included in the safety analysis. These were taken from measurements from the Krško NPP base (31 December 2015), while for those that were not measured, correlation factors [19] were used on the values for Co-60, Cs-137 and Pu-239 measured at Krško NPP (radioactive decay was not taken into account). Radioactive decay until 2014 was taken into account for all data on the activity of individual radionuclides measured in the Krško NPP base and used for the safety analysis. A projection of the future generation of radioactive waste and individual radionuclides was also drawn up on the basis of data from the Krško NPP base. The isotopic composition of an average FP containing four TTCs (column 3 in Table 7-3, which is the same as four times column 2 in the same table) or their equivalent in standard drums, and a hot FP consisting of one hot TTC (column E in Table 7-1) and three average TTCs (column 2 in Table 7-2) or their equivalents in standard drums was determined on the basis of this data. The isotopic composition for individual radionuclides for accident scenarios (abnormal operation) is presented in the table below (Table 7-2).

Table 7-2: Isotopic composition of TTC and FP for accident scenarios (abnormal operation)

Isotope	Activity of average TTC [Bq]	Activity of average FP [Bq]	Activity of hot FP containing one TTC with PRH2T2 waste [Bq]
	2	3	4
H-3	2.27E+05	9.08E+05	1.16E+07
Ag-108m	5.10E+02	2.04E+03	1.53E+03
Ag-110m	8.67E+05	3.47E+06	9.84E+08
Am-241	1.42E+06	5.67E+06	3.65E+07
Ba-133	1.87E+04	7.49E+04	1.15E+06
Ba-140	1.22E+01	4.90E+01	3.67E+01
C-14	6.07E+08	2.43E+09	1.05E+10
Ce-141	3.08E+05	1.23E+06	1.04E+11
Ce-144	2.45E+03	9.79E+03	7.34E+03
Cl-36	4.60E+03	1.84E+04	7.92E+04
Cm-242	3.38E+03	1.35E+04	1.87E+06
Cm-244	1.16E+06	4.63E+06	7.62E+07
Co-57	5.82E+05	2.33E+06	7.46E+09

Isotope	Activity of average TTC [Bq]	Activity of average FP [Bq]	Activity of hot FP containing one TTC with PRH2T2 waste [Bq]
	2	3	4
Co-58	2.39E+07	9.58E+07	9.01E+10
Co-60	8.20E+08	3.28E+09	1.11E+11
Cr-51	1.24E+05	4.95E+05	3.71E+05
Cs-134	4.56E+07	1.82E+08	2.06E+10
Cs-135	8.41E+04	3.36E+05	2.21E+06
Cs-137	5.60E+09	2.24E+10	2.13E+11
Eu-152	2.24E+05	8.96E+05	1.44E+07
Eu-154	2.39E+08	9.54E+08	2.03E+10
Eu-155	1.06E+08	4.24E+08	1.40E+10
Fe-55	3.93E+08	1.57E+09	1.10E+11
Fe-59	7.54E+04	3.02E+05	1.67E+08
I-129	2.52E+04	1.01E+05	6.64E+05
I-131	3.37E+03	1.35E+04	1.01E+04
Mn-54	1.32E+07	5.38E+07	1.27E+10
Nb-94	2.28E+04	9.12E+04	6.84E+04
Nb-95	5.96E+05	2.39E+06	1.16E+09
Ni-59	2.30E+08	9.20E+08	3.96E+09
Ni-63	3.07E+10	1.23E+11	5.38E+11
Np-237	4.78E+03	1.91E+04	1.25E+05
Pd-107	8.41E+03	3.36E+04	2.21E+05
Pu-238	6.29E+06	2.52E+07	1.66E+08
Pu-239	1.19E+06	4.78E+06	3.13E+07
Pu-241	5.42E+07	2.17E+08	2.93E+09

Isotope	Activity of average TTC [Bq]	Activity of average FP [Bq]	Activity of hot FP containing one TTC with PRH2T2 waste [Bq]
	2	3	4
Ru-103	1.00E+03	4.02E+03	5.30E+08
Ru-106	3.20E+03	1.28E+04	9.59E+03
Sb-124	6.46E+01	2.58E+02	1.94E+02
Sb-125	4.83E+06	1.93E+07	5.70E+09
Se-79	3.36E+04	1.35E+05	8.85E+05
Sm-151	2.52E+07	1.01E+08	6.64E+08
Sn-113	2.11E+03	8.43E+03	8.28E+08
Sr-90	4.52E+08	1.81E+09	2.10E+10
Tc-99	1.45E+07	5.79E+07	3.18E+08
Te-132	8.55E+01	3.42E+02	2.95E+08
U-234	1.19E+04	4.78E+04	3.13E+05
U-235	2.39E+02	9.55E+02	6.26E+03
U-238	4.78E+03	1.91E+04	1.25E+05
Zn-65	4.14E+11	1.65E+02	1.24E+02
Zr-95	2.60E+05	1.04E+06	1.66E+09

7.2.2.2 Structures, systems and components

The structures, systems and components of the repository are presented in more detail in Section 6 of this draft safety analysis report. Only the key summaries of relevance to the production and understanding of the safety analysis during the operation of the repository are illustrated below.

In accordance with the conceptual design, [5] the safety analysis (operation of repository) envisages that one silo will initially be constructed at the repository, with the option of expansion (construction of a second silo) in the event of the realisation of the SA.2 scenario. Should the SA.3 scenario be realised, only one silo will be constructed at the repository.

Within the framework of the Investment Programme for the LILW Repository, [26] it was established that the SA.3 and SA.2 scenarios are the optimal variants for disposal at the LILW repository. The key features of the scenarios are as follows:

- SA.3
    - only LILW repository disposal is conducted at the repository site; conditioning for disposal is conducted at Krško NPP,
    - half of the waste from Krško NPP and all Slovenian institutional waste will be disposed of at the repository site (one silo is required for the disposal of this waste),
    - in the determination of the quantity of waste, it was assumed that Krško NPP would operate until 2043;

- SA.2
    - the same assumptions apply as for the SA.3 scenario, except that the Croatian half of the waste will also be disposed of at the repository site (two silos are required for the disposal of all the waste).

It was therefore decided that it would be reasonable to conduct an environmental impact assessment that addresses the total quantity of disposed waste, i.e. two silos. Revision C of the conceptual design [5] sets out a basic project design for the SA.3 scenario, which in the *Development potentials of the repository to be taken into account in elaboration of the EIA*, which is part of the conceptual design, addresses the execution of the SA.2 scenario, i.e. the disposal of all operating and decommissioning waste from Krško NPP and all other Slovenian LILW. In this case two disposal silos will be constructed at the repository site.

The safety analysis is therefore drawn up for the entire inventory of waste that could be disposed of at the LILW repository site. In addition to the disposal unit, the repository site will also be home to a technological facility (second phase, see Section 9 *Operation* of this draft safety analysis report), monitoring systems, flood defences, and other buildings and systems not related to nuclear and radiation safety.

The following will be located at the technological facility:

- a storage area, where minor work on repairing damaged FPs can be carried out when necessary,
- a hot workshop (where decontamination can be carried out),
- a measurement room,
- a control room, and other areas not related to nuclear and radiation safety.

As envisaged in the conceptual design, [5] the hall above the silo will be used in the first phase for the temporary storage and remediation of damaged FPs. To this end a temporary cabin with controlled ventilation will be set up. A storage area will be constructed alongside the technological facility in the second phase.

The technological facility will be equipped with a ventilation system that will have two operating regimes corresponding to the technological facility's requirements and activities: normal (the majority of the time) and technological (during works). The design envisages air filtration, at input and output alike, for both areas. The gas outflow from the controlled area will be filtered with M6 type filters (with effectiveness of 60% to 80% at particle sizes of 0.4 µm) and HEPA13 filters (with effectiveness of 99.95%). The system will also be equipped with an automatic radiation counter, which in the event of detecting elevated radiation levels will close the flaps for the inflow and outflow of air and will alert the control room. The outflow will be 10 m above the floor in this phase.

The technological facility can also generate industrial and municipal wastewater, which could potentially be contaminated (water from the decontamination areas, floor washing, etc.). A drainage system will also be installed in the disposal silo. Its task will be to collect any percolating water, and to remove it in controlled fashion. All of this wastewater will be captured, checked and, provided that the prescribed activity concentrations are not exceeded, released into the public sewerage system. If the limits are exceeded, the water will be properly treated in a suitable plant.

The technological facility also houses an input/output control point with contamination controls and decontamination capability.

The disposal silo will be covered by a canopy, the main task of which is to protect the silo from weather conditions. The stairway that will lead to the bottom of the silo (access to the sump below the silo) will be equipped with a ventilation system with a capacity of 1,000 m^3/h. The system will have flaps and automatic radiation counter in the outflow, which will close in the event of an emergency. To prevent any buoyancy force the construction of the silo will be locally thickened in the lower part with cross-laminate inserts to provide protection against the flooding of the silo in the event of full hydrostatic pressure (flotation). The impact of construction on the surrounding hills will be minimalised to practically zero, thanks to the excavation approach (diaphragms). The construction of the silo is described in detail in Section 6.2.1 of this draft safety analysis report.

The FPs used for disposal are reinforced concrete containers in which four TTCs or 12 standard drums (200 l, 300 l) can be installed. In accordance with the requirements for the carriage of dangerous goods (the ADR), the container will comply with the requirements for Type IP-2 industrial packages.

7.2.3 SCENARIOS DURING OPERATION OF THE REPOSITORY

The scenarios for the preparation of the security analysis were developed on the basis of events and states described below, in accordance with the LILW repository project documentation, [5] the design bases [27] and the JV 5 rulebook. [21]

The following operational states were recorded within the framework of the reference documentation: [28], [29]

- Operational state 1 – Acceptance and disposal of LILW,
- Operational state 2 – Readiness for acceptance and disposal of LILW,
- Operational state 3 – Non-disposal-related work in the area of the silo, and
- Operational state 4 – Idle phase.

The following events may occur within the framework of the aforementioned operational states:

- normal operational occurrences:
    - takeover of waste at the repository, including visual controls of FPs, measurement of radiation on the surface of FPs, documentation check,

    - disposal (if all requirements are met), as the FP is transported to the roof and the disposal silo, where it is unloaded using a gantry crane into its predetermined position in the disposal silo,
    - drainage (a drainage system is in operation in the silo during all of the aforementioned operational states to collect and remove any percolating water),
    - in the case of Scenario SA.3, after the Slovenian half of the operational waste is disposed of, the standby phase commences as a sub-phase of operation (although without FPs being disposed of),
    - filling of voids (after the silo has been filled and during operation, once two layers of waste have been disposed, the voids between the containers and the wall of the silo are filled with backfill material. A concrete slab is placed on top of the vault),
    - sealing (a clay layer is then placed on top of this, providing an additional barrier between the silo and the Quaternary aquifer);
- anticipated operational occurrences (abnormal operation), which will be managed via internal rulebooks and instructions, and are assessed as having no impact on nuclear and radiation safety:
    - authorised dose limit exceeded,
    - loss of off-site power supply,
    - failure of a LILW transport vehicle at the repository site,
    - failure of the crane above the silo,
    - failure of the pumping station in the silo and by the control pool,
    - failure of the fire alarm system,
    - failure of the fire protection system,
    - failure of the LILW data recording system,
    - failure of devices for measuring releases and radiological monitoring devices, and
    - rejection of an LILW shipment;
- emergency design-basis events and accidents:
    - fire,
    - container drop,
    - airplane crash (including explosion and fire),
    - terrorist attack,
    - earthquake (followed by operational shutdown and checking of SSCs).

The development of the scenarios for the safety analysis and the models used with the individual parameters during the operation of the repository are described in detail in the safety analysis report. [13] A summary is presented below.

Within the framework of the design bases, [27] and in accordance with the international recommendations, [23] certain scenarios were defined to cover the facility stages set out by the JV5 rulebook. [21] In accordance with the recommendations, [23] these scenarios were then verified within the framework of the safety analysis, where they were evaluated with regard to the IAEA's recommended postulated initiating events (PIEs). [23] The engineering assessment method [13] was used.

Practical guideline 1.03 Content of the safety report for the LILW repository [30] stipulates that it is necessary to select the PIEs, to classify them into categories and then to further classify them with regard to their frequency.

The safety analysis for the LILW repository also included analysis and risk identification for initiating events, out of which scenarios of abnormal operation (design-basis accidents) during the operation of the LILW repository were then developed. The initiating events were used as the basis, in accordance with the IAEA recommendations. [23] Under the method described in the report, [31] the following were determined for all initiating events:

- the severity of the event,
- the probability of the event, and
- the risk of the event

The severity of the PIEs is defined with regard to the consequences that each PIE could have on SSCs. The severity of events was assessed on the basis of the assumptions below and was scored from 1 to 4, where the scores represent the following categories:

- **1** a PIE of low significance
- **2** a PIE of medium severity
- **3** a PIE of high severity
- **4** a critical PIE

The PIEs were classified into categories according to the following assumptions:

- **PIE of low significance**: an event that is not possible, or that has no impact on the working and functionality of SSCs. It may cause discomfort to the operator of the repository. It has no (discernible) impact on the environment or on people.
- **PIE of medium severity**: an event that requires attention and that without actions being taken could compromise the working of SSCs at the repository. SSCs still perform their postulated safety functions, but without actions being taken there could be long-term disruption of safety functions. The impact on the environment and on people is very small, but discernible (measurable, determinable).
- **PIE of high severity**: an event that without immediate actions being taken could lead to non-functional SSCs, individual SSCs still perform their safety functions during the event, but the performance of a safety function could cease. The facility is damaged, but there are no emissions from the facility. The impact on the environment and on people is small.
- **Critical PIE**: an event that requires immediate action, SSCs cannot perform their safety functions, damage may occur to the facility, which could lead to environmental emissions. The impact on the environment and on people is substantial.

The probability of a PIE is defined with regard to the timescale over which the event might occur. The probability is assessed on the basis of the assumptions presented below, according to which events are classified into five probability categories. Categories 1 to 5 represent the probability of a PIE as follows:

- **5 (very high)**: the PIE might occur approximately every five years
- **4 (high)**: the PIE might occur approximately every five to 20 years
- **3 (high)**: the PIE might occur approximately every 20 to 50 years
- **2 (low)**: the PIE might occur approximately every 50 to 500 years
- **1 (low)**: the PIE might occur within a period of more than 500 years

The risk of individual PIEs from the set was assessed by multiplying the severity of the event by the probability of the event:

risk = severity x probability

If risk assessment was sufficiently low (less than 12), the PIE was excluded from the analysis, and no scenario of abnormal operation was developed for it.

The analysis of initiating events was conducted on the basis of an engineering assessment, which was conducted by a group of experts in various fields (geology, hydrogeology, geotechnology, biology, mechanical engineering, materials science, nuclear and radiation safety, transport, chemistry, etc.) who have good knowledge of the planned LILW repository project.

A table of analysis of initiating events is given below, based on which scenarios of accidents (abnormal operation) were developed.

Table 7.5: Analysis of initiating events: external (human) PIEs sets out analysis of explosion events. On the basis of NUREGa CR-7201, [32] the impact of an explosion at the surface (peak pressure) was assessed and calculated, under the assumption of the explosion of 27,216 kg TNT on a road, which is an extremely large quantity, under the most unfavourable conditions (no buildings between the explosion and the silo, worst possible soil conditions), at a distance of 167 m. The closest road to the repository silos is more than 200 m away, for which reason no scenario of a traffic accident involving inflammable or explosive substances was developed.

Table 7.3: Analysis of initiating events: PIEs (Annex I, GSG-3)

PIE from Annex I of IAEA GSG-3 (2013)	PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.1)	Severity of event	Probability of event	Risk of event	Remarks
SF (1) Acceptance of packages not in compliance with acceptance criteria LTSF (1) Waste accepted that is not in compliance with facility acceptance criteria, leading to exposure scenarios of workers and the public no longer being valid	-	4	2	8	The specifications for the preparation of FPs were developed under the assumption that the FPs will be inspected before disposal (at the NPP, after conditioning) and also upon arrival at the repository, and therefore the acceptance of an FP that does not comply with the WACs is highly unlikely (low probability).
SF (2) Incorrect determination or no determination of chemical characteristics and other characteristics of waste in FPs. This could result in: (i) The presence of free liquids. (ii) The degradation or corrosion of containers faster than anticipated.	–	4	2	8	The N2 container type is currently defined in the final project documentation as the only container for the disposal of LILW at Vrbina. After the proper processing and preliminary storage, all waste streams in Slovenia are conditioned for disposal in N2 containers, by virtue of which the FPs created will be acceptable for disposal at the LILW repository. The FP specifications were developed with regard to the nature, content and behaviour of the FP, and ensure a connection between the safety assessment (SA), research and development, and the manufacturing of the FP, thereby proving compliance with the WACs. For all wastes generated in Slovenia to date and currently stored in dedicated storage areas (the CSRAO in Brinje and the storage areas at the NPP), radiological, chemical

PIE from Annex I of IAEA GSG-3 (2013)	PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.1)	Severity of event	Probability of event	Risk of event	Remarks
(iii) Generation and release of toxic gases. (iv) Generation of gases (hydrolysis) leading to damage to the matrix. (v) Variation of pressure due to chemical reaction inside containers. (vi) Fire due to vapours on surface of matrix material (e.g. bitumen). (vii) Biological contamination. LTSF (3) Waste containers not in compliance with requirements.					and physical properties are defined for the majority of parameters of importance to the operational and long-term safety of the LILW repository at Vrbina. The management approach and methods that may be applied to wastes generated in the future are also assessed. An assessment is also made of the waste disposal capacity, for the purpose of determining which wastes (after final conditioning) comply with the acceptance criteria for disposal at the LILW repository. When necessary and relevant, recommendations were issued for further processing for the purpose of ensuring that packages of waste meet the WACs provided that the requirements, processes and controls set out by the FP specification are taken into account in practice. This means that all measures have been taken to prevent any situation in which the incorrect determination of the chemical or other properties of the wastes in containers, wrapping and FPs that could lead to the production of unacceptable packaging might occur. Actions in cases of identified non-compliance with the WACs were nevertheless envisaged in the design of the plant (e.g. temporary storage and repair of damaged packaging that does not comply with certain requirements of WACs for disposal). The FP specification also sets out the need for multi-level controls and checking, to ensure that waste packages complies with the requirements of the WACs for disposal, including the possibility for the operator of the repository to check the quality and properties of the production of waste packages, the acceptance/delivery of waste packages from the final conditioning plant to the repository, etc.

PIE from Annex I of IAEA GSG-3 (2013)	PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.1)	Severity of event	Probability of event	Risk of event	Remarks
					Furthermore, with regard to waste packages (that comply with the WACs), the OpSA and PCSA take account of all relevant scenarios regarding operation and post-closure in connection with the degradation or corrosion of the FPs, such that, in the event that they would lose integrity on this basis prematurely or would begin generating and releasing toxic gases or generating gases (by hydrolysis) that could damage the FPs and lead to changes in pressure owing to chemical reactions in the FPs, all safety requirements have been met. The results of the safety assessment were taken into account for the determination of the WACs for the relevant properties of waste packages. No substance that could cause a fire hazard owing to surface vapours, and no biologically or microbiologically hazardous materials or pathogenic bacteria that could lead to biological contamination will be disposed of at the LILW repository.
SF (3) Loss of power, which could lead to various issues such as lack of ventilation or interruption in transport of containers leading to long exposure times.	–				The repository site can be managed safely without power (this means that its primary function [disposal] does not depend on the electricity supply). In addition, a back-up electrical system (generator) is available for all important systems.
SF (4) Vehicle collision (e.g. fork-lift trucks damaging	a) Forces caused by load drops or collisions	4	5	20	The consequences of the collapse of waste disposed of in the silo are no greater than a container drop into the silo. A vehicle collision could lead to a container drop into the

PIE from Annex I of IAEA GSG-3 (2013)	PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.1)	Severity of event	Probability of event	Risk of event	Remarks
shielding, safety equipment or FPs).					silo or in the technological facility. Both scenarios are analysed in the SA.
SF (5) Loss or malfunction of instrumentation, which, specifically with regard to storage, could result in loss of temperature control and failure of effective air monitoring.	e) Various temperature fluctuations (overheating or underheating).	1	3	3	Various temperature fluctuations [...] are not possible, as the repository is designed for the disposal of LILW (which does not generate heat, and contains practically no fissile material). The monitoring and ventilation systems are regularly inspected. In the event of faulty operation, remedial measures are envisaged. Even the non-functioning of the system would not have an immediate impact on the environment or on people.
SF (6) Ineffective personal monitoring.	–	2	2	4	In a situation of this type it is possible to reach or even exceed a dose limit. A situation of this type is nevertheless very unlikely (low-probability event), given the legal requirements to be upheld and the regulatory oversight.
SF (7) Faulty or ineffective security monitoring.	–	4	3	12	In a situation of this type unauthorised entry would be possible. A terrorist attack was postulated and analysed, as one of the potential scenarios.
SF (8) Faulty calibration of instruments, leading to quality assurance and safety issues.	–	2	3	6	The calibration of all instruments will be undertaken in accordance with legal requirements within the repository operator's management system. A situation of this type is therefore treated as a medium-probability event.
SF (9) Maintenance activities not well managed.	–	3	2	6	All inspections, tests and maintenance must be planned in advance, and monitored within the management system put in place by the repository operator. A situation of this type is therefore treated as highly unlikely (low-probability event).

PIE from Annex I of IAEA GSG-3 (2013)	PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.1)	Severity of event	Probability of event	Risk of event	Remarks
SF (10) Malfunction of lifting equipment leading to falling or dropping of waste packages. LTSF (2) Dropping or damage of waste containers during handling or loss of content, which could compromise individual SSCs.	a) Forces caused by load drops	4	3	12	Two scenarios are considered: • drop of waste container in the technological facility, • drop of waste container into the silo.
LTSF (6) Collapse or damage of structures during offload of waste packages.		4	3	12	Two scenarios are considered: • drop of waste container in the technological facility, • drop of waste container into the silo.
SF (11) Loss of shielding (leading to overexposure of workers). LTSF (8) Loss of shielding (e.g. damage to concrete drums during transport).	–	4	3	12	Loss of shielding leading to overexposure of workers could occur as a result of an accident (drop, fire or explosion). The radiological consequences for workers have been assessed in all possible accident situations.
SF (12) Criticality due to violation of storage arrangements.	g) criticality	4	1	4	Criticality is not possible, as the repository is designed for the disposal of LILW (which does not generate heat, and contains practically no fissile material).

PIE from Annex I of IAEA GSG-3 (2013)	PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.1)	Severity of event	Probability of event	Risk of event	Remarks
SF (13) Fire (due to, for example, sparks, cigarette smoking). SF (16) Spontaneous combustion of materials.	f) fire	4	3	12	Fire scenario.
SF (14) Improper inspection or inappropriate inspection frequency. LTSF (5) Inspections being neglected.	–	/	/	/	N/A
SF (15) Failure of emergency equipment (e.g. malfunction of fire extinguishers).	–	4	3	12	There will be no protection system in the storage area that could affect the functioning of the repository in the event of a fault. For accidents covered by the safety assessment during the operation of the repository, no measures and responses that could mitigate fire (for example) are considered, as a result of which the consequences of the non-availability of such measures can be no worse than those already assessed.
SF (17) Failure to control natural phenomena, such as a rising water table. LTSF (9) Effects due to natural weather	-	3	2	6	There are no external natural phenomena that could affect the repository, and therefore there is no need for controlling natural phenomena.

PIE from Annex I of IAEA GSG-3 (2013)	PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.1)	Severity of event	Probability of event	Risk of event	Remarks
conditions not managed (e.g. erosion after heavy rain).					
SF (18) Loss of or insufficient ventilation, which could lead to internal contamination and surface contamination.	–	2	2	4	A fault in the ventilation system will not have a direct impact on the safe operation of the repository. Like in all nuclear facilities, atmospheric radioactivity will be monitored. In the event of the accumulation of radioactivity (owing to poor ventilation of the workplace), work at the repository will be adjusted to ensure that the legal restrictions will not be exceeded.
LTSF (10) Intrusion of animals, such as rabbits or rats, not controlled.	–	1	4	4	There are no events of this type that could affect the repository, and therefore there is no need to monitor such phenomena.
LTSF (7) Leaking of waste containers.	b) internal flooding or dripping owing to leaking or broken pipes, pumps or valves for steam, water, hydraulic fluid or other liquids used in the process.	4	2	8	There is no danger of internal flooding or dripping owing to leaking or broken pipes, as the repository has a water management system that has been designed to collect all wastewater, drained water and rainwater. An adequate removal system has been provided everywhere where leaks might occur. No waste that might cause leaking from the waste containers during the repository's operational phase meets the WACs, and therefore no such waste can occur in the repository.
LTSF (4) Loss of or compromise or deterioration of engineering controls.	–	4	2	8	There are two possible situations with regard to the loss of or compromise or deterioration of engineering controls: - partial loss of safety function of FP barrier, - rapid deterioration in safety functions of disposal system barrier.

PIE from Annex I of IAEA GSG-3 (2013)	PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.1)	Severity of event	Probability of event	Risk of event	Remarks
					The first situation may arise because of an operational occurrence of a fault in the remote control system and/or the equipment for handling waste packages in the hall above the silo and/or during lifting of waste packages during temporary storage in the technological facility. In both cases the dropping of a waste package may occur. The scenarios envisaged in the safety analysis include a container drop into the silo for the first case, and a container drop in the technological facility for the second. The second situation may arise after the closure of the repository: scenario of early failure of engineering barriers.

Table 7.4: Analysis of initiating events: external (natural) PIEs

External (natural) PIE from Annex I of IAEA GSG-3	External natural PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.3)	Severity of event	Probability of event	Risk of event	Remarks
(1) Extreme meteorological conditions: (i) Strong winds, dust, sand storms (causing abrasive effects, damage to roofs or structures). (ii) Cyclones (causing damage and flying objects). (iii) Tornadoes.	a) Meteorological and climatic conditions on site and in the region: precipitation, rain, hail, snow, glaze ice, drought, wind, tornadoes, hurricanes, cyclones, quantity and duration of direct solar radiation, temperature, permafrost, cyclical freezing/melting of soil,	2	4	8	In accordance with the conceptual design, all construction materials for repository structures comply with the relevant construction regulations, and with the special requirements for the durability of structures. The rules with regard to protection of buildings against damp, and the rules with regard to thermal and sound insulation will be upheld. The conceptual design sets out the relevant wind loads,

External (natural) PIE from Annex I of IAEA GSG-3	External natural PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.3)	Severity of event	Probability of event	Risk of event	Remarks
(iv) Hurricanes. (vi) Lightning. (vii) Snow. (viii) Rain. (ix) Drought. (x) Extreme temperatures (causing heating or freezing). (xiii) Humidity and high salt content. (xiv) Hail. (xv) Frost. (xvi) Fog.	barometric pressure, humidity, fog, frost, lightning.				snow loads and temperature loads (for uninsulated parts) that were taken into account in sizing the storage structure. Slovenia occasionally reports local cyclonic storms, but not in the Krško Basin; neither are there any reports of hurricanes in the Krško Basin. High salt content is impossible given the geographical position of the storage facility. Extreme drought is not typical of the region where the storage facility is located. Because the structure is underground, lightning, heavy snow, strong winds, hail, fog and heavy rain have no effect on the repository; lightning may affect the power supply, but the storage facility can be safely operated without power (its primary function of storage does not depend on the power supply); the permeable bedrock of the Krško Polje area, where the construction of the storage facility is planned, means that precipitation drains into groundwater.
(v) Tsunamis. (xi) Floods. (xii) Extremely high or low tides.	b) Hydrological and hydrogeological conditions on site: surface run-off, flooding, speed of erosion, groundwater state, impact of waves, high tides, storm surges, tsunamis, floods, speed of coastal erosion.	2	1	2	In accordance with the conceptual design the structures of the repository have been designed to ensure adequate protection against the probable maximum flood (PMF), and against the accumulation of rainwater in the event of strong local storms; there are no tides, storm surges or tsunamis in the Krško Basin. The Sava

External (natural) PIE from Annex I of IAEA GSG-3	External natural PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.3)	Severity of event	Probability of event	Risk of event	Remarks
					floods frequently, but a flood defence was built on the left bank to protect against floods, from the Vipap cellulose plant to a point 750 m downstream from Krško NPP. All facilities important to nuclear and radiation safety are built on an embankment that takes account of the PMF. The flooding of the Sava will thus have no impact on the repository.
(2) Seismic conditions. (3) Ground instability. (7) Volcanism.	c) Geology on site and in the region: lithography, stratigraphy, seismology, volcanology, past mining and excavation.	4	2	8	In accordance with the conceptual design, the structures of the repository have been properly sized, and seismic loads have been taken into account, as required by the applicable standards; there is no information on local volcanology or mining in the past, but PIEs of this type will be investigated subsequently should such information come to light. Thus, after all possible external events that could cause the abnormal operation of the LILW repository have been reviewed, the only identified PIE that could have a non-negligible impact on the operation of the storage facility is an earthquake of higher magnitude than that envisaged in the design. Furthermore, the only possible consequence is the dropping of an FP (during disposal in the silo) or waste package from a lifting device, and both scenarios have been addressed.

External (natural) PIE from Annex I of IAEA GSG-3	External natural PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.3)	Severity of event	Probability of event	Risk of event	Remarks
(4) Landslides (e.g. due to ice melting). (5) Erosion.	d) Geomorphology and topography on site: stability of natural materials, surface erosion, influences of terrain (topography), weather conditions or consequences of extreme weather.	1	1	1	In accordance with the data on the geological structure of the terrain, no natural instability was identified. On this basis it can be concluded that in the natural conditions, there is no identification of landslides, karstic porosity, subsurface faults or natural subsidence owing to diagenesis or owing to natural compression of strata.
(8) Biological phenomena (e.g. algae or marine growth, fauna and flora invasion, and biological contamination).	e) Land and aquatic flora on site (with regard to impact on the plant): vegetation (land, aquatic), rodents, birds, other wildlife.	2	2	4	In accordance with the conceptual design, events such as blockages of entrances and exits or damage to the structure of the repository do not constitute a potential risk to the working of the repository; direct damage owing to the excavation, gnawing or deposition of material by rodents and birds can be avoided through the correct planning of the landscaping of the storage site, and by a maintenance programme. The silo will be sealed (covered by the hall above the silo) so that local flora and fauna will not have a direct impact on the disposal process. There are no algae or marine growth in the area. No biological wastes or wastes contaminated with microbes or pathogenic bacteria meet the WACs. This means that no waste of this type can be handled and disposed of in Vrbina.
(6) Natural fires.	f) Potential for: natural fires, explosions in the work site, methane or natural toxic gas	2	2	4	At this stage there is no information about natural fires, dust storms or sand storms in the area. The storage facility site is

External (natural) PIE from Annex I of IAEA GSG-3	External natural PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.3)	Severity of event	Probability of event	Risk of event	Remarks
	(from wetlands or landfills), dust storms or sand storms (including the potential blockage of entrances and exits).				surrounded by residual forest in the midst of agricultural land, where the possibility of natural fires is deemed to be very low. Because the Kostak landfill is closed, the potential impact of releases of methane or natural gas from the landfill site will be investigated in later phases, if necessary.
- glaze ice	–	3	2	6	The highest glaze ice thickness recorded in the area is 13 cm. Given the dimensions of the hall above the silo, the total mass of ice that could collect above it exceeds the maximum design load. In such an event, the roof of the hall could collapse, and ice could fall to the floor. Because the area of the silo is only a third of the area of the roof of the hall, not all of the ice would fall on top of the silo; only a third or, at most, a half would do so. The maximum impact would be expected to come from ice falling to the bottom of the silo (because speed would be highest). Assuming the ice falls from 68 m (the hall height of 18 m plus the 50 m depth of the silo to the first layer of containers), the calculated collision energy is an order of magnitude lower than the collision energy in the event of an airplane crashing into the silo. Thus the airplane accident scenario also covers this scenario (the consequences would be less severe than an airplane crash).

Table 7.5: Analysis of initiating events: external (human) PIEs

External (human) PIE from Annex I of IAEA GSG-3	External human PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.2)	Severity of event	Probability of event	Risk of event	Remarks
Explosions.	a) Explosion (solid substances, liquids, gas, powder cloud or aerosol).	4	3	12	Low probability of impact on the repository on account of the distance and the features of underground disposal. An explosion in the vicinity of the repository would have no impact on waste already disposed, which would be disposed of below the surface. Includes the scenario of a terrorist attack (including explosion at the technological facility). The airplane crash scenario (into the silo) also includes an explosion component.
Fire from: (i) The sea after oil spill from a vessel, (ii) Uncontrolled bush or veld fires.	a) Fire (solid substances, liquids, gas, powder cloud or aerosol).	4	2	8	Low probability of impact on the repository. The airplane crash scenario (into the silo) includes a fire component. Furthermore, an internal fire at the technological facility has been assessed, and there are no grounds for believing that an external fire would have consequences different from an internal fire.
Aircraft crashes and other unpredicted mobile sources.	b) Aircraft accident.	4	1	4	Very low probability, as the repository will be located in the vicinity of Krško NPP, and there are no civil aviation corridors across the site; however, because it is in the vicinity of Cerklje military airfield (approximately 4 km south), and the scenario examines the maximum impact of the event on the environment and on people, this scenario was also assessed at the request of the ARAO.

External (human) PIE from Annex I of IAEA GSG-3	External human PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.2)	Severity of event	Probability of event	Risk of event	Remarks
Projectiles, sources of high energy from machines and flying objects.	c) Projectiles caused by structural/mechanical faults in machinery in the vicinity.	4	2	8	Low probability of impact on the repository, given the distance to the nearest machinery.
Floods due to dam failures.	d) Floods (due to structural defects in dams or weirs).	3	2	6	In accordance with the conceptual design, all repository facilities of importance to nuclear and radiation safety are built on an embankment that will protect the structures against the probable maximum flood, including the destruction of dams on the Sava.
Mining activities.	a) Sinkholes or subsidence caused by tunnel construction, mining.	3	1	3	Low probability, as no works of this type are envisaged during the operation of the repository. The repository will be built on a plain on which no tunnel construction is envisaged. In addition, there are no natural resources suitable for mining in the area.
	f) Soil vibrations.				Low probability of impact on the repository, as no works that would include vibrations are envisaged in the area. The construction of the road links with the road to Vrbina will be completed before the repository begins operation, while other work to develop and modernise the local (road) infrastructure will have no impact on the working of the repository.
Nearby industrial activities (toxic gases, corrosion, smoke).	g) Release of corrosive, toxic and/or radioactive substances.	3	2	6	Accidental radioactive emissions from Krško NPP could lead to the radioactive contamination of the repository, and therefore the plan of emergency actions and procedures needs to include safety measures for workers at the repository,

External (human) PIE from Annex I of IAEA GSG-3	External human PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.2)	Severity of event	Probability of event	Risk of event	Remarks
					including early warning procedures in the event of a release that could have an impact on the repository. The working of the repository could be suspended after such an event, but accidental releases from Krško NPP will not have any impact on operational safety. In the vicinity of the repository there are no industrial zones that could discharge toxic gases, corrosive products or smoke in quantities that could affect the working of the repository.
Transport infrastructure. Nearby military activities. Electromagnetic interference (e.g. caused by a power station close by).	h) Geographic and demographic data (population density and expected changes in the plant lifetime, industrial and military facilities and related activities, and consequences of accidents at these facilities, in traffic, and in transport infrastructure to the storage facility (motorways, airports and/or flightpaths, railways, rivers and canals, pipelines, etc., and potential for influence or accidents that include hazardous materials).	4	2	8	The only external accident that could have an impact on the repository is a radiological accident at Krško NPP that results in radioactive releases. Traffic accidents involving explosive or inflammable materials or tanks in the vicinity of the repository are also seen as unlikely, as transport in Slovenia complies with European legislation. However, in the event of an accident of this type, the impact on the repository would be smaller than in the event of an explosion or a fire in the repository; in addition the waste is disposed of below the surface, and surface-level accidents of this type would have no impact on it. There will always be a maximum of one FP on the surface that is included in the explosion or fire scenario. Consequently the only human PIE identified at this stage is a traffic accident involving a

External (human) PIE from Annex I of IAEA GSG-3	External human PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.2)	Severity of event	Probability of event	Risk of event	Remarks
					truck loaded with LILW for transport. A traffic accident up to the site of the storage facility is deemed extremely unlikely. At the repository there is no technical system whose safety function could be compromised by electromagnetic interference. Nevertheless, the only potential adverse impact of electromagnetic interference is the failure of the remote control system for handling waste packages in the hall above the silo. The worst case scenario in the event of a failure would be a container dropping into the silo, which has been addressed.
	i) Power supply and potential interruption of power supply	3	2	6	As described in the conceptual design, the repository's electrical system envisages a back-up power system (diesel generator). If both systems are out at the same time, the only systems that could compromise the normal working of the repository are the lifting systems. In the event of any of these appliances failing, the worst case scenario is a an FP drop.
Sabotage. Theft. Civil strife and war.	j) Civil strife	4	3	12	Given the current political situation in Slovenia, it is unlikely that unrest that might endanger the working of the repository could occur. The design of the repository and the FPs means that there is no concern in the event of unrest (it is LILW). In addition, a terrorist attack has been included and assessed, and there are no grounds for believing that unrest

External (human) PIE from Annex I of IAEA GSG-3	External human PIE covered by OpSA report EISFI-TR-(11)-11 Vol 3, Rev 2, October 2012 (Section 2.2.2)	Severity of event	Probability of event	Risk of event	Remarks
					or war could have a larger impact than an airplane crash or terrorist attack at the technological facility.

7.2.3.1 Normal evolution scenario

The safety analysis for the operation of the repository in the normal evolution scenario encompasses all repository states where all SSCs are working as planned.

The activities that will proceed at the repository during normal operation are defined as normal operational occurrences in Section 7.2.3.

The aforementioned activities were addressed within the framework of the safety analysis as the normal evolution scenario. The doses (individual and collective) for the employees that will carry out these activities have been assessed, and the impact on members of the public has also be assessed. [17] With regard to international practice, the impact on an adult member of the public who could be most exposed given the properties of the repository and the scenario is assessed.

During operation there is no expectation that a major quantity of gases could be generated, mainly because there will be no free water present and the waste will be in aerobic conditions. Nevertheless, certain reactions could occur between individual materials, and could produce gases as a result. They are:

- corrosion of metallic materials (corrosion of aluminium – generates hydrogen),
- degradation of organic materials (degradation of cellulose – generates carbon dioxide),
- radiolysis (radiolysis of water – generates hydrogen).

It is assumed that carbon dioxide generated by the degradation of cellulose would dissolve in water and would participate in carbonation reactions in alkaline (cement) pore water. It is therefore envisaged that carbon dioxide will not be present in the mixture of gases generated in the silo during operation. Within the framework of the safety analysis, [33] it was also assessed that the quantity of gas generated by all of the aforementioned reactions is small, and therefore was not considered within the framework of the repository's operational safety.

The investigations of the container [34] established that research showed that the prototype container satisfies the water permeability requirements, as its permeability is a class lower than the requirement (< 10^{-12} m/s). The tests also established that the materials used ensure gas permeability.

Models were developed from the scenarios and events described above, and are presented below.

7.2.3.2 Accident scenarios: abnormal operation (design-basis accidents)

Within the framework of safety analysis, [13] a list of PIEs was analysed in accordance with the IAEA recommendations. [23] The following accident scenarios (abnormal operation during the operation of the repository) were recorded on this basis.

7.2.3.2.1 Scenario of container drop (does not apply to standby phase)

The FP is planned as a Type IP-2 industrial package, which means that within the framework of the testing to determine its ability to tolerate the usual transportation conditions and given its mass, it must withstand a drop from a height of 30 cm. In the handling of FPs at the LILW repository, there is also the possibility of FPs being dropped from a greater height. Within the framework of safety analysis, [17] it was assessed that the probability of such an event is very low, but the impact of such an event was nevertheless assessed and evaluated.

Within the framework of the safety analysis the following sub-scenarios of the dropping of an FP were evaluated:

- An FP drop could occur in the hall above the silo, or in the technological facility. For both facilities it is assessed that the maximum height from which it could drop is 9 m. The volume of the hall above the silo is larger than that of the storage area in the technological facility, and there would therefore be a higher concentration and consequently a higher load on employees in the latter in the event of a drop (given equal release of radioactive substances). It was therefore decided to address the case of an FP dropping from a height of 9 m at the technological facility. It was also assumed that the walls of the technological facility would not be damaged by the drop, and there would be releases because of a fault in the ventilation system, at a height of 10 m.
- An FP drop may occur during unloading (the dropping of an FP into the silo). The two most conservative (in the sense of radiological impact) sub-scenarios were assessed:
    - An FP drop from a height of 50 m onto a disposed layer of containers covered by a levelling layer 10 cm thick, where the drop represents a container dropping onto the silo bottom without any disposed waste. The assessment [17] was that the radiological consequences would be largest in this case, both for employees and for members of the public. It was also assessed that in the event of a drop into the concrete layer there could be greater dispersal (the FP could disintegrate into several parts, which would have a larger area [the total silo area] for dispersal) than in the case of an FP dropping onto disposed FPs (in this event there would be damage to multiple FPs, but the damage would be minor and the dispersal of parts would be smaller).
    - An FP dropping from a height of 35 m onto the fifth layer of disposed waste covered by 10 cm of levelling concrete.

FPs will be brought to the repository conditioned for disposal. Because transportation to the repository is not part of this safety report, it was assessed that given the very short distances at the repository and the very light traffic, the scenario of a traffic accident and a container drop during transport around the repository is extremely unlikely, and was therefore not addressed. The transport of waste will be undertaken in FPs, which will be subject to tests to prove their ability to tolerate the usual transport conditions, and will be certified as IP-2 containers. [35] Once the requirements for Type IP-2 containers have been met, transport of radioactive substances is allowed, in containers of this type, by public road, without further analysis and safety assessments.

The calculation of the aforementioned three scenarios made use of an average FP, a hot FP and inventory as defined in Section 7.2.2.1 of this draft safety analysis report. Use was also made of the release fractions and other data necessary for an impact assessment from the literature addressing the drop of a similar container (the WAGR container was developed in the UK for the purpose of disposing of waste from the decommissioning of the Windscale Advanced Gas-Cooled Reactor, and is of similar dimensions to the N2bV). [36], [37], [38], [39] With regard to the release fraction, the WAGR container was tested for a drop from a height of 7 m, and the release fractions for other heights were modelled and estimated on the basis of the results of the drop. In accordance with the above references, the release fractions stated in the table below were also estimated.

Table 7.6: Release fractions used for FP drop scenario from various heights

Drop height, m	Release fraction, for assessment of inhalation and external irradiation
9	0.0003
35	0.001
50	0.0015

Within the framework of the safety analysis, doses for employees as a result of external irradiation and inhalation were estimated for the FP drop scenario, with the conservative assumption of no release into the environment (all particles remain in the facility). The impact on a representative member of the public was estimated for the case of a certain level of release via the unsealed flaps of the ventilation system in the technological facility. This results in inhalation, resuspension, ground scattering and external irradiation. The impact was estimated for various distances from the repository and for various weather conditions. In the case of an FP dropping into the silo, it was assumed that there would be no release into the environment (the ventilation system of the access shaft is closed in the event of increased activity in the silo, and is closed during the disposal of FPs).

The scenario of a container dropping into the silo included the study of the fall of a container together with the crane. Due to the robustness and size of the structure such a fall is highly unlikely, however, a detailed analysis of a drop will be given in the next phase of the project. The impact of the scenario of a container drop together with the crane is estimated to be lower than the impact of the airplane crash scenario.

7.2.3.2.2 Fire scenario

For the fire scenario, within the framework of the safety analysis [17] the impact of a fire that could occur in the technological facility or in the hall above the silo was assessed, although the smaller volume of the storage area in the technological facility means that the conservative approach was to address a fire in the technological facility only, as this would have a greater impact on employees. The fire was assumed to last one hour, with either one average FP or one hot FP involved in the fire.The container properties were taken from the research and testing conducted for the WAGR container, which is presented in the previous section. [36], [37], [38], [39] The release fractions for individual radionuclides in case of fire are illustrated in the table below (Table 7.7), and were taken from a study [36] of a fire lasting one hour at 1000°C.

Table 7.7: Atmospheric release fractions for individual radionuclides in case of fire

Radionuclides	Release fraction in case of fire
C, Cl, H, I, Se	1
Cs, Sb, Sn, Tc, Te, Ag, Ba, Ru, Zn	7E-04
Co, Cr, Eu, Fe, Mn, Ni, Pd, Sr, U	6E-05
Am, Ce, Cm, Np, Pu, Zr, Nb	3E-05

The impacts on employees from direct exposure, inhalation and external irradiation, and on members of the public from inhalation and external irradiation were estimated within the framework of the scenario. The impacts were estimated for various distances from the repository and for various weather conditions.

7.2.3.2.3 Terrorist attack scenario

The terrorist attack scenario is identified as sensitive from the perspective of physical security, and therefore only the non-sensitive information is provided in the draft safety analysis report. The scenario is given in full in a documentation classified as FOR INTERNAL USE ONLY. [39] For the scenario of a terrorist attack at the repository, it was assumed within the framework of the safety analysis [17] that terrorists would set a large quantity of explosives at the facility, where an average FP or hot FP is stored. The impact on members of the public and on employees from inhalation and external irradiation was estimated for various distances, exposure times and breathing rates.

7.2.3.2.4 Aircraft crash scenario

It was assumed in the first phase of the preparation of the safety analysis [13] that an airplane crashes into the technological facility in which waste is being conditioned for disposal. Because it was decided during the project optimisation [15] that conditioning for disposal would be carried out at Krško NPP, it was assumed in this phase of the safety analysis [17] that an airplane crash into the silo would occur at a time when the silo is completely full, but not yet sealed, and that the crash would be followed by a fire caused by the airplane's fuel. The crash is assumed to involved a 30-tonne military airplane at a speed of 150 m/s. There would be severe damage to the roof, and the resulting release of radioactivity would enter the atmosphere in full. After the crash, a fire lasting one hour would break out, caused by 20 tonnes of fuel in the airplane. For the source of radioactivity in the airplane crash scenario, the assumption was that there are 99 average FPs disposed of in the upper layer, as defined in Section 7.2.2.1 of this draft safety analysis report. It was assumed that not all of the FPs would be damaged in the crash. The crash scenario assumes damage to one FP, while three FPs burn in the resulting fire. HotSpot software was used to estimate the effective doses caused by the explosion and fire from external irradiation, inhalation, resuspension and ground scattering in various weather conditions for employees and for members of the public. It was assessed that a variety of damage to FPs would occur as a result of the explosion and the fire. Various release fractions were taken for the explosion and the fire.

7.2.4 MODELS

The conceptual and mathematical models for assessing the scenarios given in the previous sections of this draft safety analysis report are presented in detail in Section 3 of the safety analysis report of 2012 [40] and Section 5 of the safety analysis report of 2016. [17] In this section of the draft safety analysis report the models are presented at the level necessary for understanding the safety analysis as a whole.

7.2.4.1 External irradiation

MICROSHIELD and MICROSKYSHINE software was used to assess the doses from external exposure for the normal evolution scenario for employees and for members of the public. This allows for the modelling of sources: wastes of various forms and compositions. The wastes installed in TTCs and then in FPs were modelled within the framework of the safety analysis, during internal transportation around the repository, during disposal in the silo, and then once disposed of into the repository unit (the silo). In the modelling the wastes were assumed to be homogenous, and a build-up effect was also taken into account, i.e. the contribution to radiation by photon scattering.

In all models the approximation used for the wastes (sources) was that they are cylindrical forms composed of a homogenous mixture of materials:

- concrete (FP, backfill material in FP and in silo),
- metals (drums and TTCs),
- wastes.

The proportion of an individual material in the mixture of materials was calculated on the basis of the volumes of individual materials in one layer of waste.

To evaluate the contribution of waste disposed of in the silo to the total dose at the perimeter of the repository owing to the skyshine effect, an analytical approach was used, which was verified with the MICROSKYSHINE software package.

For the transportation of waste, the form of the waste was assumed to be cubic, containing waste in cylindrical form. A detailed description of the geometry and parameters used in the models is given in Appendix A of the safety analysis report. [17]

7.2.4.2 Gas generation model

The quantity of gas generated during the operation of the repository was estimated on the basis of an estimate of gas generated by the corrosion of aluminium. The model takes account of single linear corrosion and the area of deposited aluminium. The quantity of gas generated is estimated on the basis of the equation (Equation 7.1) for the gas generation rate at standard conditions (m^3/year) in the corrosion of surface aluminium (the ends of individual components are not taken into account in the calculation). [41]

$$\dot{V} = \frac{22.4 \rho x C A_{metal}}{M}$$

Equation 7.1: Quantity of gas generated by corrosion of aluminium during operation of the repository

Where:

C = corrosion rate [m/year] = 1E-03 m/year for Al

A_{metal} = surface area of metal that can be corroded [m^2]

ρ = density of metal [kg/m^3] = 2,700 kg/m^3 for Al

M = atomic mass of metal [kg/kmol] = 27 for Al

x = stochiometric factor [kmol of gas per kmol of metal] = 1.5 for Al

22.4 = volume of 1 kmol gas at standard conditions [m^3]

The general assumption in the use of the above formula is that all metal components have planar geometry, and the area of the metal is expressed by the specific surface area of the metal.

7.2.4.3 Model of impact of atmospheric releases inside buildings

To estimate the dose that would be received by employees in buildings in the event of atmospheric releases, it was assumed that mixing is instantaneous and complete. The received doses owing to radioactive material in the air were estimated on the basis of the following formula (Equation 7.2): [42]

$$E_{inh} = \sum_j C_j * e_{inh,j} * t_{exp} * B,$$

Equation 7.2: Estimated dose from exposure to radioactive releases into atmosphere

Where:

C_j = concentration of radionuclide *j* in the air (Bq/m^3)

$e_{inh,j}$ = effective dose coefficient for dose obtained by inhalation of radionuclide *j* by employee [Sv/Bq], in case of moderate absorption in blood and median particle diameter of 5 µm [44]

t_{exp} = exposure time (h)

B = breathing rate = 1.2 m^3/h [44]

The concentration of radionuclide *j* was calculated by dividing the source of potential contamination (defined for individual scenarios in Section 7.2.3 of this draft safety analysis report) by the volume of air available for dispersion for the scenario in question.

In the case of the FP drop scenario, it was assumed that there is an immediate and homogenous spread of contamination in the air available in the silo and up to a height of 2 m above the silo. After a certain time the contamination spreads throughout the hall above the silo. The concentration of atmospheric contamination was calculated such that the release was divided according to the volume of air in the silo (V_S) and the volume of air in the hall (V_H). There are two possible sub-scenarios for an FP drop in the silo: a drop of 50 m, and a drop of 35 m.

$V_{s,35}$ = 18,848.3 m^3 (drop of 35 m)

$V_{s,50}$ = 26,809.0 m^3 (drop of 50 m)

Two sub-scenarios are also considered for the volume of air in the hall above the silo:

a) immediately after the drop, contamination spreads through the hall to a height of 2 m above the silo ($V_{H,i}$ = 1,348.3 m^3), which applies to the estimation of the impact of releases on employees who are currently located in the hall,
b) after a certain time, the contamination spreads throughout the hall above the silo ($V_{H,r}$ = 35,593.7 m^3), which applies to the estimation of the impact on employees who come into the hall after the event to remediate the situation.

Under the above assumptions, the volume of air in which the contamination could spread after a container drop was estimated thus:

a) for employees located at the edge of the silo at the moment of the drop, $V_{i,35} = V_{s,35} + V_{H,i}$ = 20,196.5 m^3 for an FP drop of 35 m, and $V_{i,50} = V_{s,50} + V_{H,i}$ = 28,157.5 m^3 for an FP drop of 50 m,
b) for employees involved in the remediation of an FP drop, $V_{i,35} = V_{s,35} + V_{H,i}$ = 55,790.2 m^3 for a drop of 35 m, and $V_{i,50} = V_{s,50} + V_{H,i}$ = 62,402.7 m^3 for a drop of 50 m.

For the scenario of an FP drop in the technological facility, there was a similar assumption that the spread of radioactivity in the air is instantaneous and homogenous. Immediately after the drop the spread is in the shape of a hemisphere with a diameter of 9 m, the height of the technological facility. After a certain time the radioactivity spreads throughout the space. The following volumes were taken:

a) $V_{i,TO}$ = 1,526 m^3
b) $V_{r,TO}$ = 5,174.1 m^3

The employee exposure time was estimated at 10 minutes for workers present at the time of the drop, and 30 mins for workers involved in the remediation.

In addition to internal exposure owing to inhalation, in the FP drop scenario there is also external exposure owing to external irradiation in the contamination plume, which was taken into account for workers involved in the remediation. The impact was estimated with the following equation (Equation 7.1).

$$E_{sub} = \sum_j C_j * e_{sub,j} * t_{exp},$$

Equation 7.3: Impact on employees owing to external irradiation

Where:

C_j = concentration of radionuclide *j* in the air (Bq/m^3)

$e_{sub,j}$ = effective dose coefficient owing to atmospheric external irradiation [Sv/Bq·s·m^{-3}], taken from [45] and corrected by incorporating the equivalent effective dose for skin
t_{exp} = exposure time (s)

7.2.4.4 Model of impact of atmospheric releases outside buildings

HotSpot software, which was developed by the Department of Energy in the USA and is presented in a report within the framework of the safety analysis, [46] was used for the calculation of atmospheric releases outside buildings in the event of accidents.

HotSpot uses a Gaussian dispersion equation to estimate the dispersal of atmospheric releases (of radioactivity), which is as follows:

$$C(x,y,z,H) = \frac{Q}{2\pi\sigma_y\sigma_z u} exp\left(\frac{y^2}{2\sigma_y^2}\right)\left[exp\left(-\frac{(z-H_e)^2}{2\sigma_z^2}\right) + exp\left(-\frac{(z+H_e)^2}{2\sigma_z^2}\right)\right] exp\left(-\frac{\lambda x}{u}\right) DF(x)$$

Equation 7.4: Gaussian dispersion equation

Where:

C = atmospheric concentration integrated over time [Bqs/m^3]

Q = source emission [Bq]

λ = radioactive decay constant [s^{-1}]

x = downwind distance [m]

y = crosswind distance [m]

z = distance on vertical axis [m]

H = effective release height [m]

σ_y = standard deviation in integrated concentration distribution in crosswind direction [m]

σ_z = standard deviation in integrated concentration distribution in vertical direction [m]

u = average wind speed at effective release height [m/s]

DF[X] = plume dispersion factor

HotSpot uses the methodologies recommended by the ICRP to calculate doses. The dose was estimated at various distances from the repository at time intervals of 10 minutes at a height of 1.5 m above the ground, which is the average height at which breathing occurs. Details of the algorithms used by HotSpot are given in the software report. [46] Details of the parameters used in the model are presented in Appendix B of the safety analysis report. [17]

The estimation of the doses owing to exposure to atmospheric release outside the buildings took account of inhalation, external irradiation, resuspension and ground scattering. The exposure time was taken to be 1 day after the initial release. The emission source was then determined and calculated in the software as follows:

$$\text{source that can be inhaled} = MAR \times DR \times LPF \times ARF \times RF,$$

$$\text{source that cannot be inhaled} = MAR \times DR \times LPF \times ARF \times (1-RF)$$

Where:

MAR ....is material at risk, the total quantity of radionuclides included in the release scenario

DR ....is the damage ratio, the fraction of the MAR that is directly included in the release scenario

LPF ....is the leak path factor, the fraction of the MAR that can escape the containment system, e.g. filters. When there is no containment system, LPF = 1

ARF ....is the airborne release fraction, the fraction of the MAR that can enter the atmosphere in aerosol form

RF ....is the respirable fraction, the fraction of aerosolised material that can be inhaled; the aerodynamic diameter of the particles must be less than 10 μm

The analysis then made use of meteorological data (wind and solar radiation). The most common atmospheric stability classes were used (determined on the basis of incoming solar radiation per unit land area), which consist of:

- A very unstable atmospheric conditions
- B moderately unstable atmospheric conditions
- D neutral conditions
- F relatively stable conditions

Wind speeds were determined for each class, as follows:

- A – 0.1 m/s, 1 m/s
- B – 1 m/s, 2 m/s
- D – 1 m/s, 2 m/s, 3 m/s
- F – 1 m/s, 2 m/s, 3 m/s.

Each calculation was made once without precipitation, and a second time with precipitation of 10 mm/h.

The impact was estimated and calculated for various distances from the emission source. The doses were calculated for 20 distances ranging from 30 m to 80 km from the site. The receptor height was taken to be 1.5 m, while the breathing rate was taken to be 1 m^3/h. [43]

The algorithms for estimating the dose using HotSpot software are presented in detail within the framework of the software report. [46]

7.2.4.5 Evaluation of DCFs for calculating effective doses that can be received by individuals in particular age groups for the LILW repository in the event of accidents

Within the framework of the Evaluation of DCFs document, [47] factors were analysed for the calculation of effective doses that can be received by individuals in particular age groups in the event of accidents that might occur at the LILW repository. It was established that in the case of inhalation, the DCFs for babies could be up to four times higher for certain radionuclides than the DCFs for adults, but the breathing rate of adults is almost eight times higher than that of babies. It can therefore be concluded that the estimated effective received doses for adults are representative for all members of individual age groups.

In the case of ingestion, the DCFs for babies are 12 times higher than the DCFs for adults on average. However, the doses that could occur by ingestion are very low. Publications that provide the basis for determining DCFs [43] judge that the use of biokinetic parameters for adults to calculate the factors for children overstates the estimated dose because the substance elimination rate is higher in the young.

7.2.5 RESULTS OF SAFETY ANALYSIS DURING OPERATION

The results of the safety analysis during operation are presented in detail in the safety analysis reports for operational safety and in Appendices A and B of this report. [17] All the main results demonstrating the safety of the operation of the planned LILW repository are summarised below.

In the estimation and calculation of doses for individual groups of people (employees and members of the public), the following assumptions were used:

- the isotopic composition and activity of specific radionuclides used to calculate doses were estimated for a hot TTC or FP on the basis of package number 15798 from the Krško NPP database, i.e. spent ion exchange resins. The average package was defined as the average of all packages,
- the maximum number of FPs disposed of in a year is 200,
- wastes are assumed to be disposed of five years after being generated, but calculations were also made for fresh wastes and 10-year-old wastes,
- the basic radioactive material used for modelling was spent ion exchange resins (maximum activity at minimum density: 0.8 g/cm^3, which entails the minimum self-absorption of radioactivity),
- radioactivity is always modelled as homogenous,
- the build-up effect is taken into account in dose estimation.

7.2.5.1 *Estimation of dose for employees under normal evolution scenario*

A list of the activities that will be carried out at the repository and details of their scope are given in the following table (Table 7.8).

Table 7.8: List and scope of activities at the LILW repository, taken from the conceptual design [5]

No.	Workplace	Worker	Activity	Duration (hours per container)
1	Entrance to repository: reception at administrative and service building	S	Acceptance and review of transport documentation in reception (at distance of 10 m from vehicle)	0.10
		S	Vehicle security check (at distance of 1 m from vehicle)	0.05
2	Entrance control point: entry to CRA	R	Measurement of contamination and dose rate of arriving vehicle and cargo	0.10
3	Platform in hall	L / O	Loading of containers into disposal silo: • removal of container attachments on vehicle at distance of 1 m • monitoring of unloading of container at distance of 10 m	 0.15 0.05
3	Disposal silo	O	Monitoring and supervision of filling of voids between containers and construction of levelling layer (for every two layers disposed of: 198 containers) from top of silo or from platform in hall	0.18
		E	Filling of voids between containers and construction of levelling layer from top panels of disposed containers (performed by two workers)	0.10
		O	Monitoring and guidance of installation of drainage pipes for every two disposed layers (from top of silo)	0.10
		E	Installation of drainage pipes (performed by four workers)	0.40

Key to worker abbreviations:

- S: security guard (receptionist)
- O: operator
- L: logistics officer
- R: radiologist (entrance and exit radiological characterisation and other radiological measurements and analysis)
- E: external contractor for construction work

On the basis of the above table (Table 7.8) and other assumptions and models presented in the previous sections of this draft safety analysis report, conservative (they could also be termed maximum) doses for employees were estimated. For all workplaces and positions other than external contractor for construction work it was conservatively assumed that the work would be performed by a single worker, even though the use of several workers could be envisaged for certain work. For external contractors for construction work, it was envisaged that the filling of voids between containers and the construction of the levelling layer from the top panel of disposed containers is undertaken by two workers, while the installation of drainage pipes is undertaken by four workers. The results thus illustrate the maximum dose received in a year (disposal of 200 containers). The results are presented in Table 7.9 below.

All workers who work on the disposal of waste are treated as exposed workers. Those not working in this area are not treated as exposed workers. The maximum dose that exposed workers could receive is estimated in this phase of the project. In future phases procedures will be optimised in line with the ALARA principle, to minimise doses for all workers.

Table 7.9: Estimated dose per employee

	Maximum total individual dose per year [mSv/year]	Maximum total dose per year per activity [mSv/year]
security guard (receptionist)	2.7	2.7
radiologist	3.3	3.3
operator	7.7	7.7
logistics officer	7.7	7.7
external contractor for construction work (multiple workers envisaged)	0.07	0.2
TOTAL		21.6 person-mSv/year

The highest dose is received by a logistics worker and an operator when disposing of waste in the silo, when other employees are not present. A smaller fraction is contributed by supervision of the filling of voids and the installation of drainage pipes, when external contractors are also involved, although there are more of them (two in void filling, four in drainage installation), which means that their dose is lower.

7.2.5.2 Estimation of dose for public under normal evolution scenario

On the basis of the assumptions and models presented in the previous sections of this draft safety analysis report, conservative (they could also be termed maximum) doses for members of the public during the operation of the repository were estimated. Dose contributions from external irradiation and skyshine were taken into account.

The following assumptions were taken into account:

- the silo is completely filled with waste (all ten layers),
- the voids between the disposed FPs have been filled, but no levelling layer has been constructed,
- the representative member of the public will spend 2,016 hours by the perimeter of the repository.

The calculation of the dose for a member of the public at the perimeter of the LILW repository under the aforementioned conservative assumptions was made for three different options for disposed waste: fresh waste, waste held in storage for five years, and waste held in storage for ten years. The results of the calculations are illustrated in Table 7.10 below.

Table 7.10: Calculated dose for member of public at perimeter of repository during disposal of waste of various ages

age of waste [years]	dose [mSv/year]
0	0.011
5	0.005
10	0.003

The estimated dose from skyshine was 0.3 pSv/year, and can be treated as negligible.

7.2.5.3 Estimation of dose from FP drop in technological facility

In terms of the impact on employees, two situations were addressed in the case of an FP being dropped in the technological facility:

- exposure of employees immediately after the FP drop. The assumption was exposure for 10 minutes, during which time the employees withdraw from the technological facility,
- after a certain time, when the contamination has fully spread throughout the entire area where the drop occurred, employees enter the area to attend to urgent initial remediation of the event. In this case the exposure lasts 30 minutes.

It was assumed in the case of a drop that the FP disintegrates into three parts, and the direct exposure of employees was estimated on this basis at various distances from the FP (from 0.5 m to 2 m). The estimated doses for employees are presented in Table 7.11 below.

Table 7.11: Estimation of dose for employees from FP drop in technological facility

Brief description of situation during FP drop in technological facility	Average estimated dose for employee (for analysed situations at distance of 0.5 m to 2 m) [µSv per event]	Maximum estimated dose for employee (for analysed situations) [µSv per event]
10 min of exposure in immediate vicinity of disintegrated average FP, main contribution via inhalation	79.2	94.4
10 min of exposure in immediate vicinity of disintegrated hot FP	979	1,380
30 min of exposure, full spread of contamination, average FP	100	1,650
30 min of exposure, full spread of contamination, hot FP	146	2,840

The dose for a representative member of the public was also estimated within the framework of the FP drop scenario. For all cases it was assumed that the member of the public is exposed to a contamination plume for one day. The estimated doses for members of the public are presented in Table 7.12 below.

Table 7.12: Estimated dose for member of public from FP drop in technological facility

Brief description of parameters assumed for FP drop in technological facility	Estimated dose for member of public (exposure time: 1 day) [µSv per event]	
Distance from technological facility [m]	100	1,000
Ventilation system works at 99.95% efficiency, average FP, worst weather conditions (wind 3 m/s, precipitation 10 mm/h)	0.0036	0.0002
Ventilation system works at 99.95% efficiency, hot FP, worst weather conditions (wind 3 m/s, precipitation 10 mm/h)	0.032	0.0014
Ventilation system not working, entire contamination released, average FP, worst weather conditions (wind 1 m/s, precipitation 10 mm/h)	4.8	0.2
Ventilation system not working, entire contamination released, average FP, worst weather conditions without precipitation (wind 0.1 m/s, no precipitation)	1.2	0.027

It is evident from the above results that the impacts of an FP drop in the technological facility are acceptable, i.e. below the allowed loads. In the case of the estimated impact on a member of the public, the estimated doses are lower than the doses received from the natural environment for all situations assessed.

In the estimation of the dose for employees in the event of an FP drop in the technological facility, it can be seen that the doses depend primarily on the exposure time and distance from the radiation source. The received doses have been estimated very conservatively, and could be reduced substantially, for example by using means of protection, appropriate procedures, etc.

7.2.5.4 Estimation of dose from FP drop into silo

The results of the estimation of the dose from an FP drop into the silo are presented in Table 7.13 below.

Table 7.13: Estimation of dose for employee immediately after FP drop into silo

Brief description of parameters assumed for FP drop into silo	Estimated dose for employee [µSv per drop]	
Drop height, m	35	50
Employee remains at edge of silo for 10 min after drop, average FP	16.7	17.8
Employee remains at edge of silo for 10 min after drop, hot FP	161	169

The main contribution to the dose from an FP drop into the silo comes from inhalation. This is 100 times higher than the dose from direct exposure to the damaged FP. The dose from waste already disposed of is even lower than that (three orders of magnitude).

The estimated dose received by employees who remediate the situation after a certain time (when the contamination has fully mixed with the available air in the silo and in the hall above the silo) is presented in Table 7.14 below. It is assumed that the employee is in a position where remediation is carried out for 30 minutes.

Table 7.14: Estimation of dose for employee during remediation of FP drop into silo

Brief description of parameters assumed for FP drop in silo	Estimated dose for employee [µSv per drop]	
Drop height, m	35	50
After certain time, employee remains 30 min at edge of layer of disposed waste, average FP	34.6	40.7
After certain time, employee remains 30 min at top of layer of disposed waste, average FP	52.1	58.2
After certain time, employee remains 30 min at edge of layer of disposed waste, hot FP	580	640
After certain time, employee remains 30 min at top of layer of disposed waste, hot FP	1,002	1,007

In this case the key contributions to the dose for an employee come from direct exposure and inhalation.

The estimated doses for all analysed cases of an FP drop into the silo are lower than the limit of 20 mSv.

7.2.5.5 Estimation of dose in fire scenario in technological facility

The results of the estimation of the dose from a fire in the technological facility in which an FP is involved are presented in Table 7.15 below.

Table 7.15: Estimation of dose for employee in fire scenario in technological facility

Brief description of parameters assumed for fire in technological facility	Estimated dose for employee [µSv per event]	
Distance from FP involved in fire [m]	0.5	2
During fire, employee remains for 10 min at defined distance from average FP	36.2	28
During fire, employee remains for 10 min at defined distance from hot FP	562	349
After fire (all contamination spreads throughout space), employee remains for 30 min at defined distance from average FP	58.7	34
After fire (all contamination spreads throughout space), employee remains for 30 min at defined distance from hot FP	1,190	552

The estimated doses for members of the public are presented in Table 7.16 below.

Table 7.16: Estimation of dose for member of public in fire scenario in technological facility

Brief description of parameters assumed for fire scenario in technological facility	Estimated dose for member of public (exposure time: 1 day) [µSv per event]	
Distance from technological facility [m]	100	1,000
Ventilation system works at 99.95% efficiency, average FP, worst weather conditions (wind 1 to 3 m/s, precipitation 10 mm/h)	0.053	0.0023
Ventilation system works at 99.95% efficiency, hot FP, worst weather conditions (wind 1 to 3 m/s, precipitation 10 mm/h)	0.240	0.01
Ventilation system not working, entire contamination released, average FP, worst weather conditions (wind 3 m/s, precipitation 10 mm/h)	100	4.6

7.2.5.6 Estimation of dose in terrorist attack scenario

Only the main results are presented for the terrorist attack scenario (Tables 7.17 and 7.18). More on the scenarios and calculations is given in a separate report, [39] which is classified as FOR INTERNAL USE ONLY for security reasons.

Table 7.17: Estimation of dose for member of public in terrorist attack scenario

Brief description of parameters assumed for terrorist attack scenario	Estimated dose for member of public (exposure time: 1 day) [µSv per event]	
Distance from technological facility [m]	100	1,000
Average FP, worst weather conditions (wind 1 to 3 m/s, precipitation 10 mm/h)	64	3
Hot FP, worst weather conditions (wind 1 to 3 m/s, precipitation 10 mm/h)	570	27

Table 7.18: Estimation of dose for employee in terrorist attack scenario

Brief description of parameters assumed for terrorist attack scenario	Estimated dose for employee (exposure time: 1 day) [µSv per event]
Distance from technological facility [m]	30
Average FP, worst weather conditions (no wind, precipitation 10 mm/h)	780

7.2.5.7 Estimation of dose in airplane crash scenario

As described in Section 7.2.4.4 of this draft safety analysis report, the estimation of the dose for members of the public in the event of an airplane crash was carried out for various weather conditions and various distances from the repository. Two impacts were assessed:

- the impact of an airplane crash caused by the collision of the airplane with the silo,
- the impact of an airplane crash caused by the fire after collision.

The estimated doses from an airplane crash into the silo and the resulting fire are presented in Table 7.19 below. The assumptions were that an employee would be 30 m away from the silo, while the impact at other distances would apply to the representative member of the public.

Table 7.19: Estimation of dose in airplane crash scenario including fire after impact

Brief description of parameters assumed for airplane crash scenario at silo	Estimated dose [mSv per event]		
Distance from silo [m]	30 (exposure of 8 hours)	100 (exposure of 1 day)	1,000 (exposure of 1 day)
Crash of airplane (30 tonnes), 99 average FPs, worst weather conditions	17	15	0.7
Fire after airplane crash (20 tonnes of fuel), 99 average FPs, duration of fire 1 hour, worst weather conditions	2.1	2.4	0.3

The total estimated dose in the airplane crash scenario (sum of crash and fire) is presented in Table 7.20 below. The total estimated dose cannot be taken as the simple sum of the doses for the crash and the fire, as the summation of the crash and the fire takes account of different weather conditions from those applying to the dose estimates in Table 7.19.

Table 7.20: Estimation of dose in airplane crash scenario (crash and fire combined)

Brief description of parameters assumed for airplane crash scenario at silo	Estimated dose [mSv per event]		
Distance from silo [m]	30 (exposure of 8 hours)	100 (exposure of 1 day)	1,000 (exposure of 1 day)
Crash of airplane (30 tonnes), 99 average FPs, worst weather conditions	17	16	0.9

The calculated dose in the airplane crash scenario is the highest of all the scenarios of accidents and abnormal operations, but the calculated values are below the limit of 100 mSv, which in line with the European standards represents the upper reference limit for the public during extraordinary events.

7.2.6 SENSITIVITY ANALYSIS OF SAFETY ANALYSIS DURING OPERATION

The next section presents sensitivity analysis of the safety analysis undertaken for the time of the operation of the repository, which was conducted within the framework of the preparation of the safety analysis, and is taken from the safety analysis report during operation. [17] Various parameters that were found to have a significant impact on the final results of dose estimation were examined within the framework of the sensitivity analysis. The main purpose of the sensitivity analysis is to determine and check the impact of uncertainties in individual parameters on the final estimate.

7.2.6.1 Change in inventory data under the normal evolution scenario

According to the safety analysis conducted for the normal evolution scenario, the radionuclides that contribute most to the estimated dose (direct exposure) under the normal evolution scenario are Co-58, Co-60 and Cs-137. These account for 92% to 99.5% of the dose, depending on the age of the radioactive waste. The data used for the calculation applies to fresh waste, and waste held in storage for five years and ten years. The contribution of significant radionuclides to the final estimated dose depending on the distance from the container and the age of the waste is illustrated in Figure 7.1 below. The contribution of individual radionuclides to the estimated dose owing to direct exposure depending on the distance from the FP and the age of the waste is illustrated in Figure 7.2.

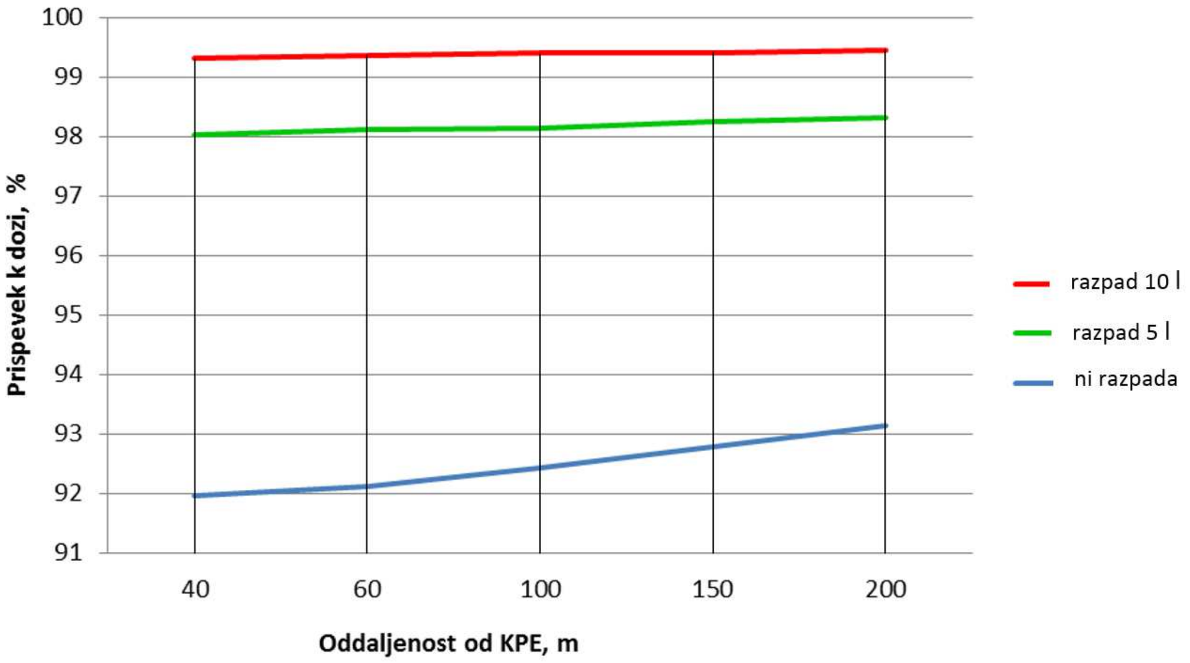

Figure 7.1: Contribution of significant radionuclides (Co-58, Co-60 and Cs-137) to final estimated dose depending on distance from container and age of waste

Prispevek k dozi, %	Contribution to dose (%)
razpad 10 l	decay 10 years
razpad 5 l	decay 5 years
ni razpada	no decay
Oddaljenost od KPE, m	Distance from FP (m)

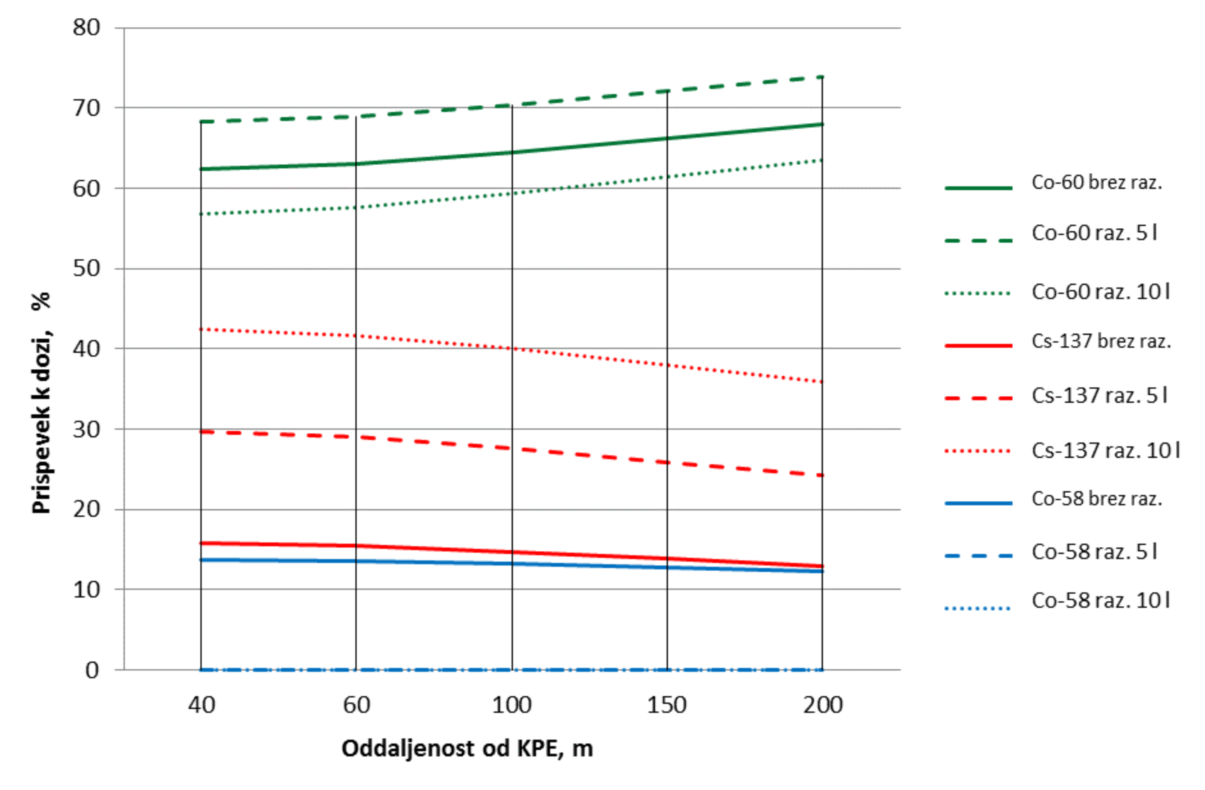

Figure 7.2: Contribution of individual radionuclides to estimated dose owing to direct exposure depending on distance from FP and age of waste

Prispevek k dozi, %	Contribution to dose (%)
Co-60 brez raz.	Co-60 excl. decay
Co-60 raz. 5 l	Co-60 decay 5 years
Co-60 raz. 10 l	Co-60 decay 10 years
Cs-137 brez raz.	Cs-137 excl. decay
Cs-137 raz. 5 l	Cs-137 decay 5 years
Cs-137 raz. 10 l	Cs-137 decay 10 years
Co-58 brez raz.	Co-58 excl. decay
Co-58 raz. 5 l	Co-58 decay 5 years
Co-58 raz. 10 l	Co-58 decay 10 years
Oddaljenost od KPE, m	Distance from FP (m)

On the basis of the above graphs (Figure 7.1 and Figure 7.2), it can be concluded that the contribution of the key radionuclides (Co-58, Co-60 and Cs-137) increases relative to others with the distance from the FP. The relatively short half-life of Co-58 means that its contribution to the estimated dose becomes negligible after four years. The contribution of Co-60 is dominant for all the waste ages, and is eight to nine times larger than the contribution of Co-58 and Cs-137 for fresh waste and three times larger than that of Cs-137 for 10-year-old waste. Given the assumed initial activity and half-life of Co-60 and Cs-137, it can be concluded that for waste more than 15 years old the key contribution to the final estimated total dose will come from Cs-137.

The impact of the decay or aging of waste before disposal on the estimated dose at the perimeter of the repository (for a member of the public present at the perimeter for 8 hours a day on every working day in the year) is illustrated by Figure 7.3 and Figure 7.4. The figures illustrate the dose at the perimeter caused by direct external irradiation and skyshine (total impact). The dose was estimated using conservative analytical estimates and MicroSkyshine software. [17] The software estimate was taken into account in Section 7.2.5.2, as the analytical method was judged to be overly conservative. The results of the two methods are nevertheless comparable. From the analysis it can be concluded that the waste ages used (from fresh to ten years) are sufficient for analysing the impact on the final contribution to the estimated dose.

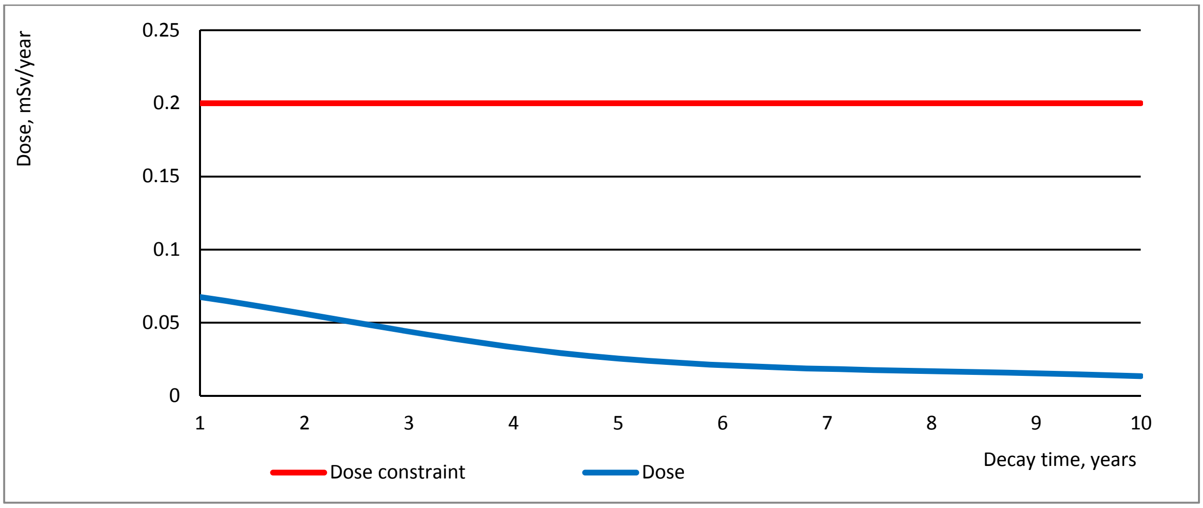

Figure 7.3: Impact of waste aging before disposal on estimated dose at perimeter of repository (Dose), taking account of analytical calculation of skyshine (the red line denotes the envisaged constraint at the perimeter of the repository [Dose constraint])

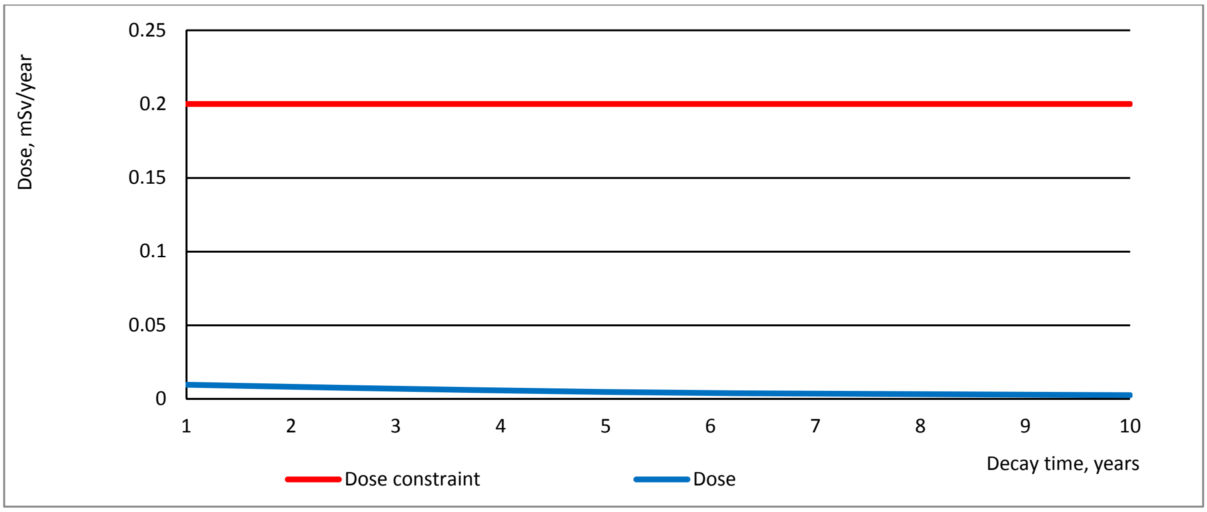

Figure 7.4: Impact of waste aging before disposal on estimated dose at perimeter of repository (Dose), taking account of calculation of skyshine with MicroSkyshine software (the red line denotes the envisaged constraint at the perimeter of the repository [Dose constraint])

7.2.6.2 Variation in number of FPs disposed of in one year

The sensitivity analysis calculated the estimated dose for employees and for members of the public under various assumptions for the number of FPs disposed of in one year (5, 50, 100, 150 and 200). The results are illustrated in the three figures below (Figure 7.5, Figure 7.6 and Figure 7.7). The estimated doses can be seen to be below the limits in all cases. It can also be seen that the number of FPs disposed of does not have an impact on the dose for a member of the public at the perimeter, but does have an impact on the dose for an employee.

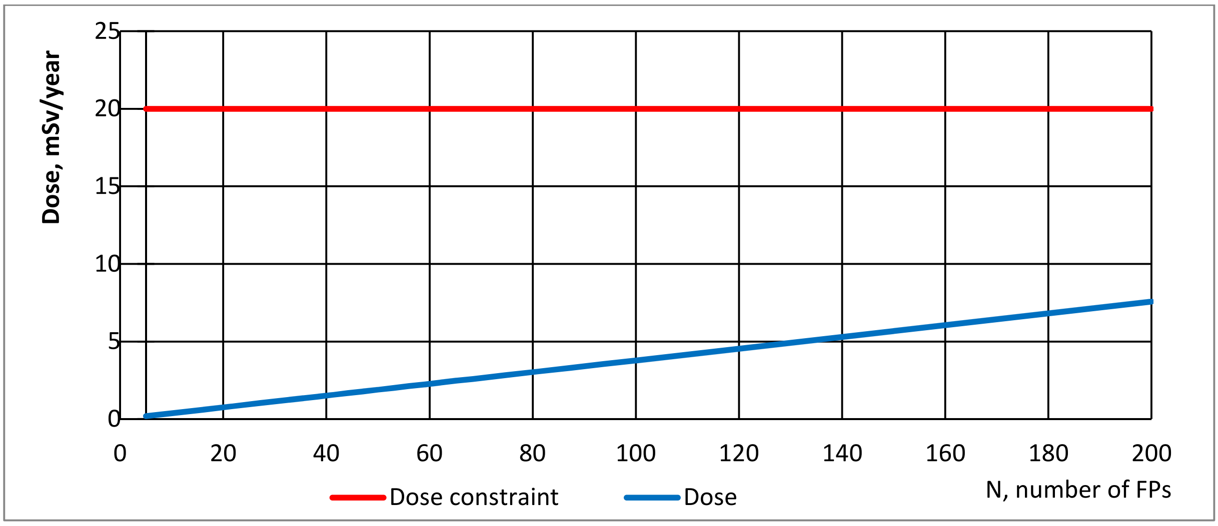

Figure 7.5: Dose received by operator (Dose) versus number of FPs disposed of in one year, waste aged for five years (the red line represents the constraint with regard to the received dose for employees [Dose constraint])

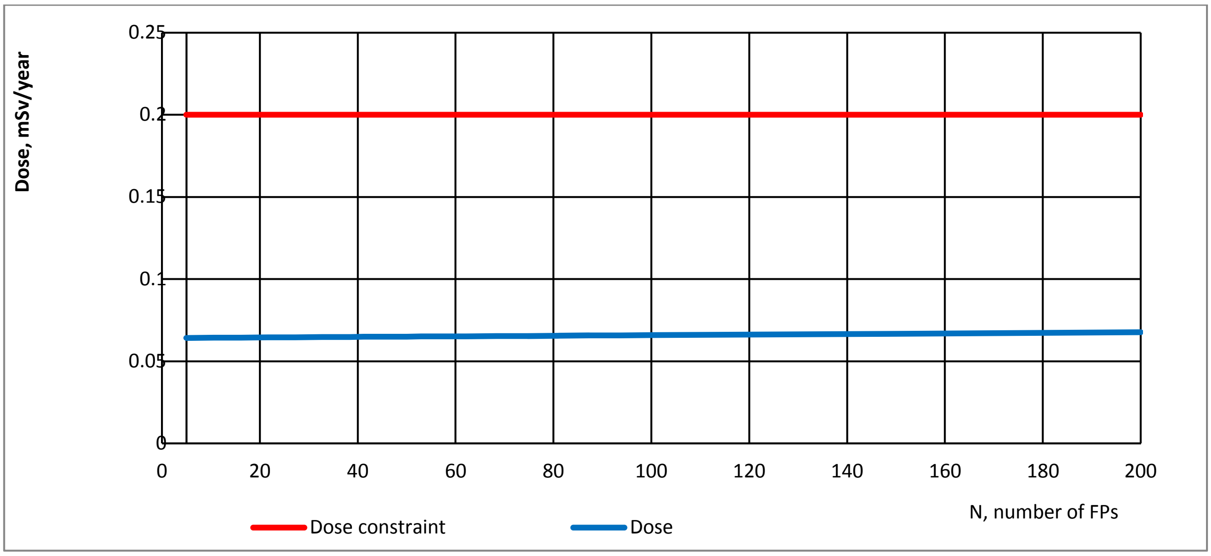

Figure 7.6: Dose received by member of public (Dose) versus number of FPs disposed of in one year, fresh waste, skyshine determined analytically (the red line represents the constraint with regard to the received dose at the perimeter [Dose constraint])

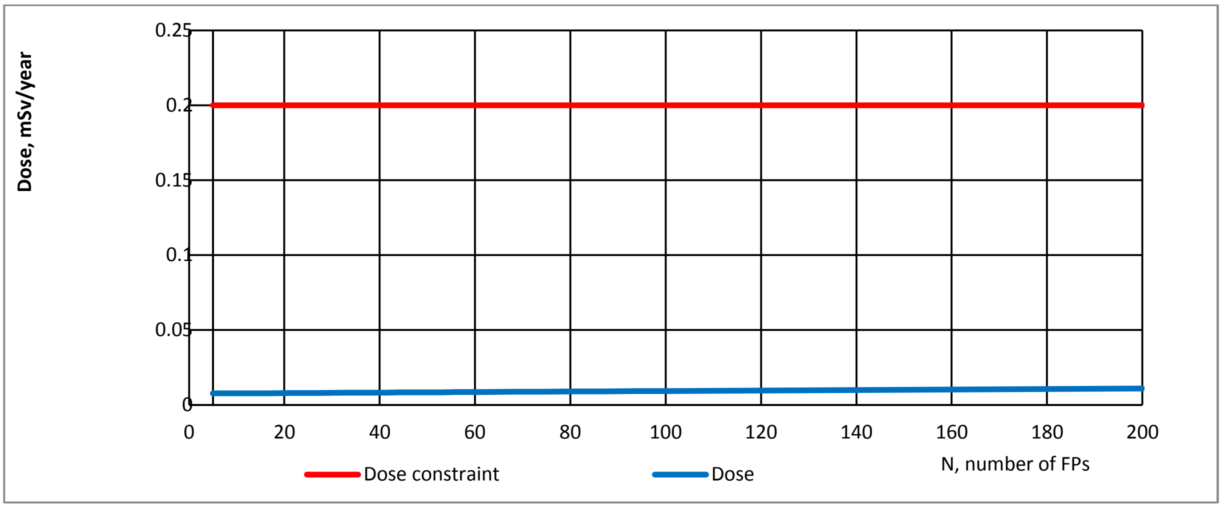

Figure 7.7: Dose received by member of public (Dose) versus number of FPs disposed of in one year, fresh waste, skyshine determined by MicroSkyshine software (the red line represents the constraint with regard to the received dose at the perimeter [Dose constraint])

7.2.6.3 Impact on estimated dose of duration of activity at repository

The time needed to carry out individual activities in the disposal of FPs in the silo is taken from the conceptual design, [5] where it is estimated on the basis of the practical experience of other repositories or nuclear facilities where similar activities are carried out. The precise times of individual activities will only be known in later phases of the repository (trial operation, operation) on the basis of the practical experience of the workers who perform the tasks in question. The approach taken in the sensitivity analysis was to use the time assumed for each individual activity (T_i) and times increased or reduced by 50% (T_i ± 50%). The analysis is presented in Figure 7.8 below. It is evident from the results that all estimated doses are below the limit prescribed for employees.

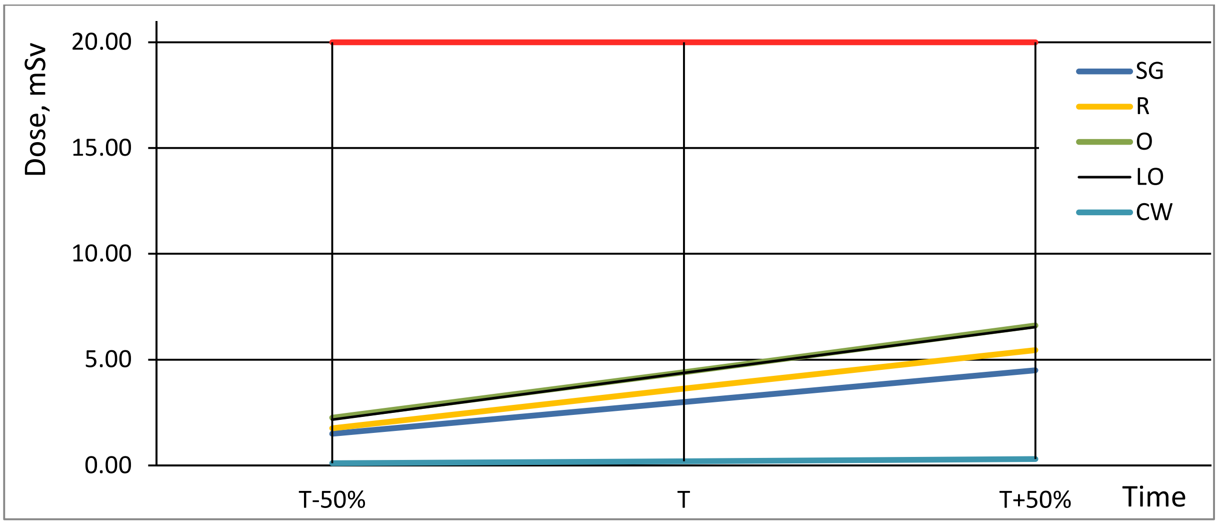

Figure 7.8: Impact of time taken to carry out activities on dose received by employee (SG: security guard; R: radiologist; O: operator; LO: logistics officer; CW: external contractor for construction work)

7.2.6.4 Impact of meteorological conditions on accident scenarios (abnormal operation)

As stated in the previous sections of the draft safety analysis report, the spread of a contamination plume in the event of an accident scenario (abnormal operation) also depends on the weather conditions. The four most common atmospheric stability classes for the Vrbina site (determined on the basis of incoming solar irradiation per unit land area) were examined in the sensitivity analysis. They consist of:

- A very unstable atmospheric conditions
- B moderately unstable atmospheric conditions
- D neutral conditions
- F relatively stable conditions

The scenario of an FP drop in the technological facility with a wind speed of 1 m/s and no precipitation was taken for the calculation. The results of the calculations are illustrated in Figure 7.9 below.

As explained in Section 7.2.3.2.1, the release of contamination is envisaged at 10 m. Because of this, and the increased dispersion, very low concentrations occur in the actual vicinity of the facility under relatively stable conditions. The mechanism is explained in detail in the user instructions for Hot Spot software. [46]

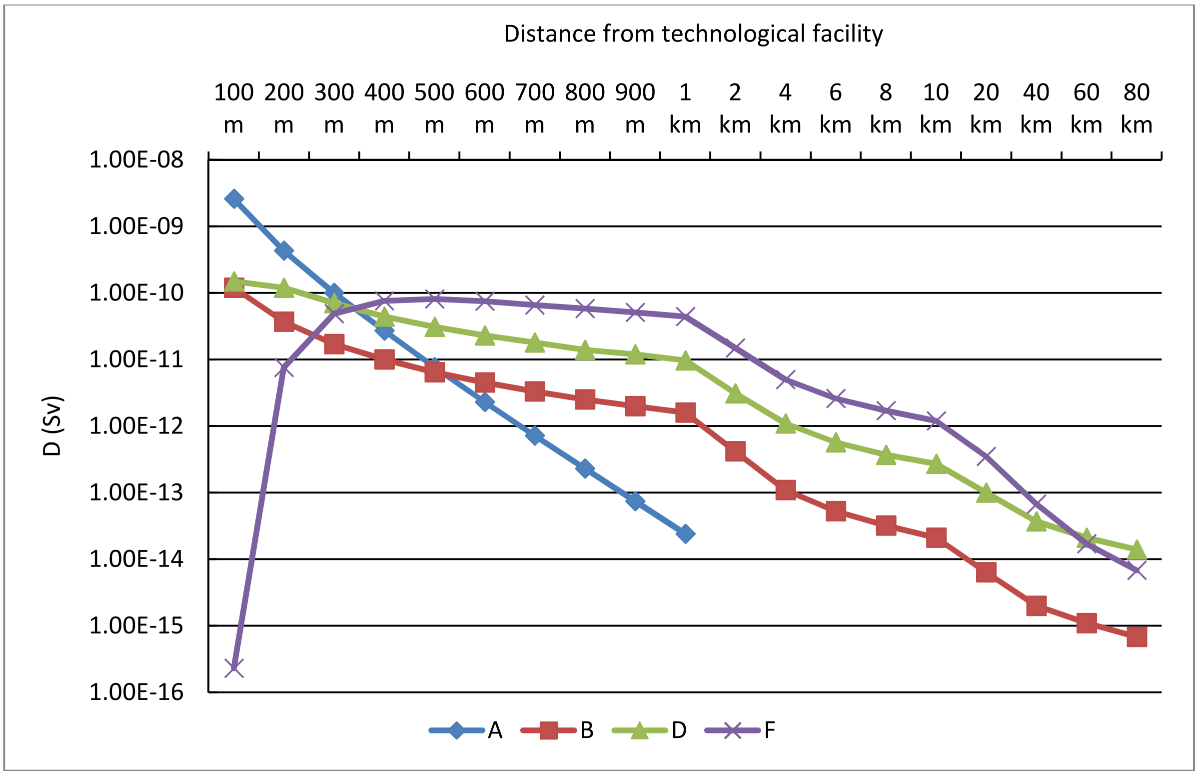

Figure 7.9: Impact of atmospheric stability on effective dose from FP drop in technological facility at various distances from repository

On the basis of the results, the worst atmospheric conditions were selected in the models for the estimation of the effective dose for individual distances.

In addition to the atmospheric conditions, the wind speed also has an impact on the spread of contamination. Calculations of the impact of wind speed for various atmospheric conditions were therefore made, and are illustrated in the following figures (Figure 7.10, Figure 7.11, Figure 7.12 and Figure 7.13).

The results show that in the case of unstable atmospheric conditions (class A), high speed wind reduces the effective dose by approximately a power of ten at smaller distances and by approximately 25% at larger distances. According to the results, the wind speeds taken for certain scenarios were the least favourable (the highest effective doses were calculated).

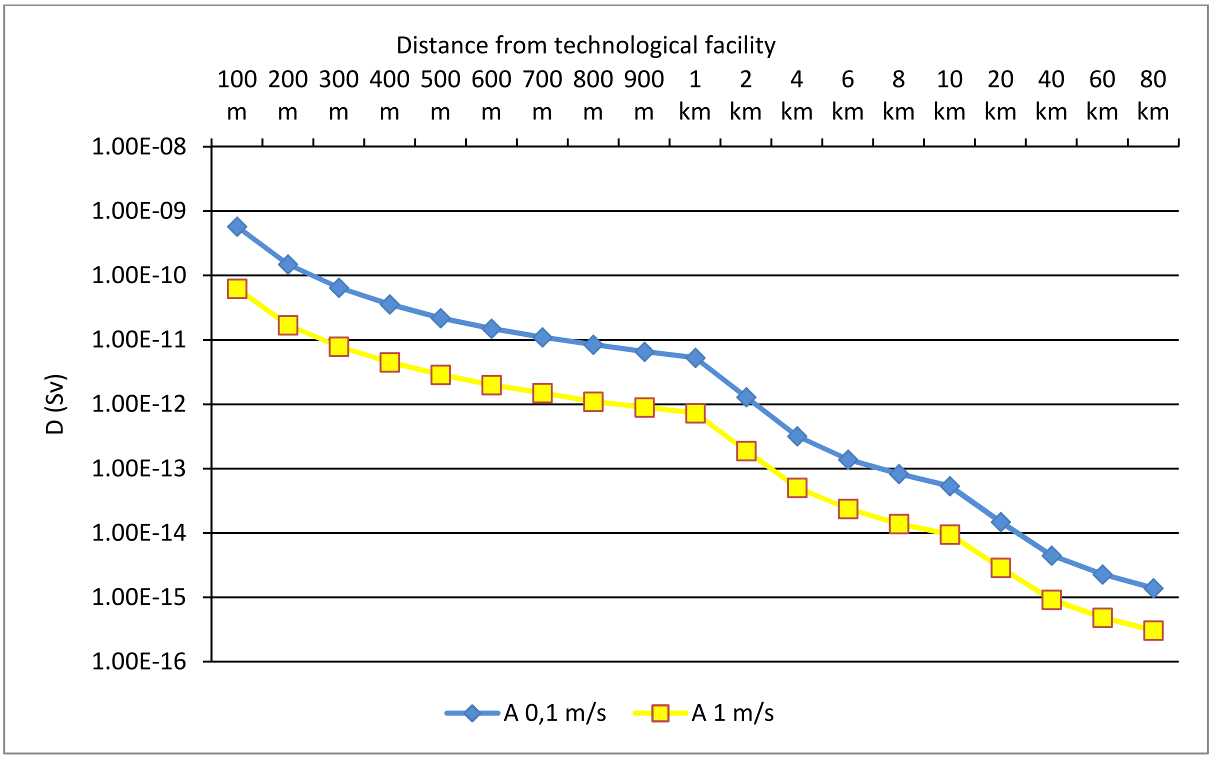

Figure 7.10: Impact of wind on effective dose from FP drop in technological facility at various distances from repository (class A atmospheric stability)

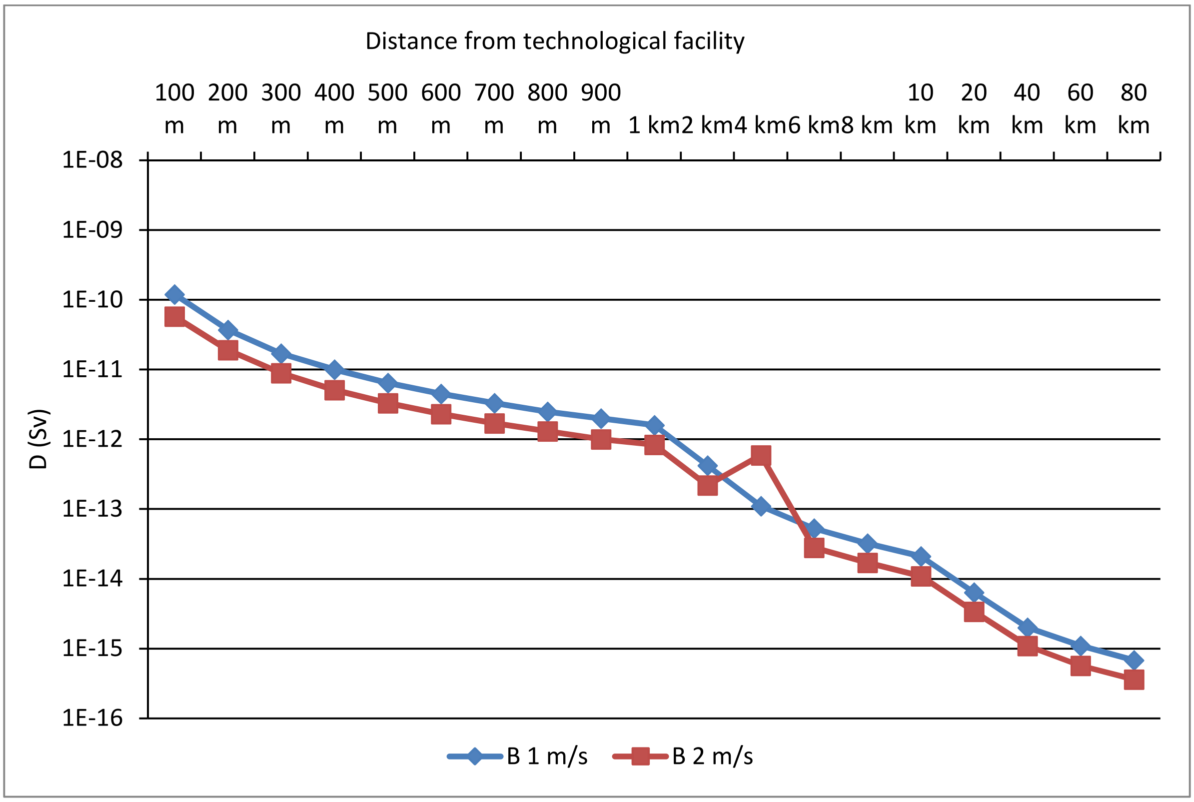

Figure 7.11: Impact of wind on effective dose from FP drop in technological facility at various distances from repository (class B atmospheric stability)

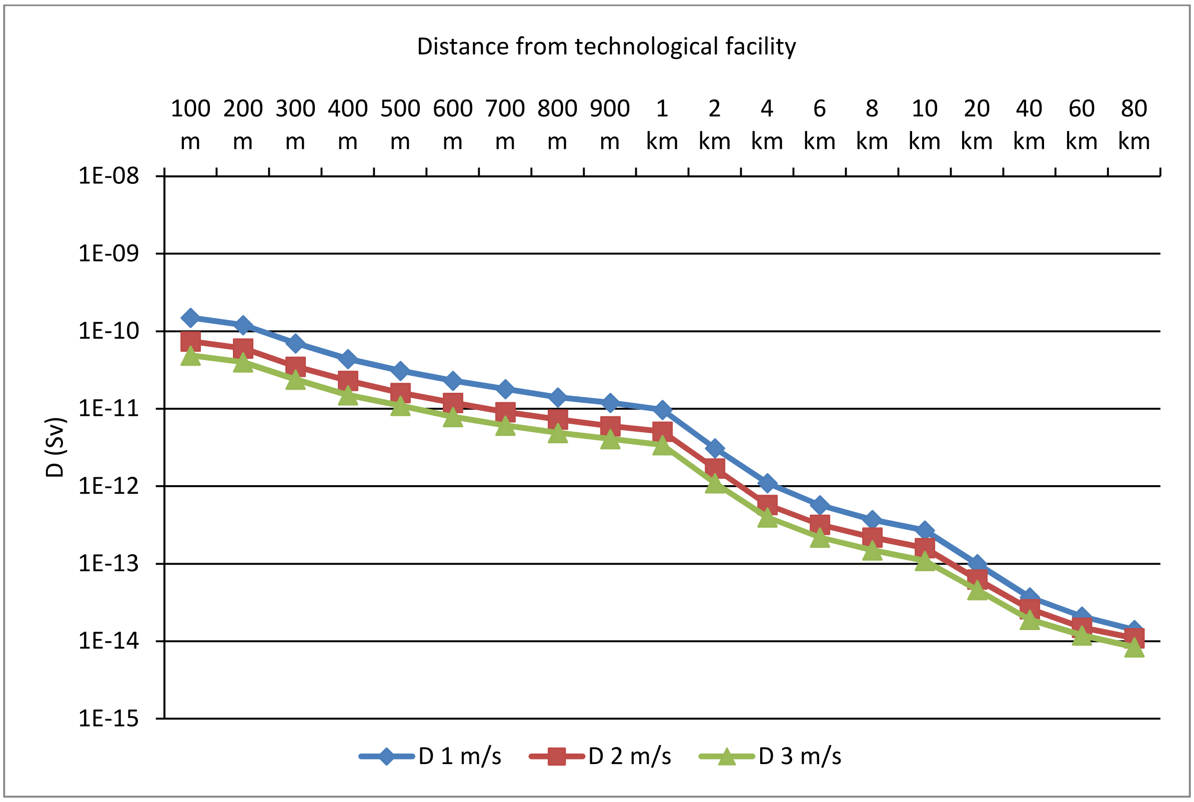

Figure 7.12: Impact of wind on effective dose from FP drop in technological facility at various distances from repository (class D atmospheric stability)

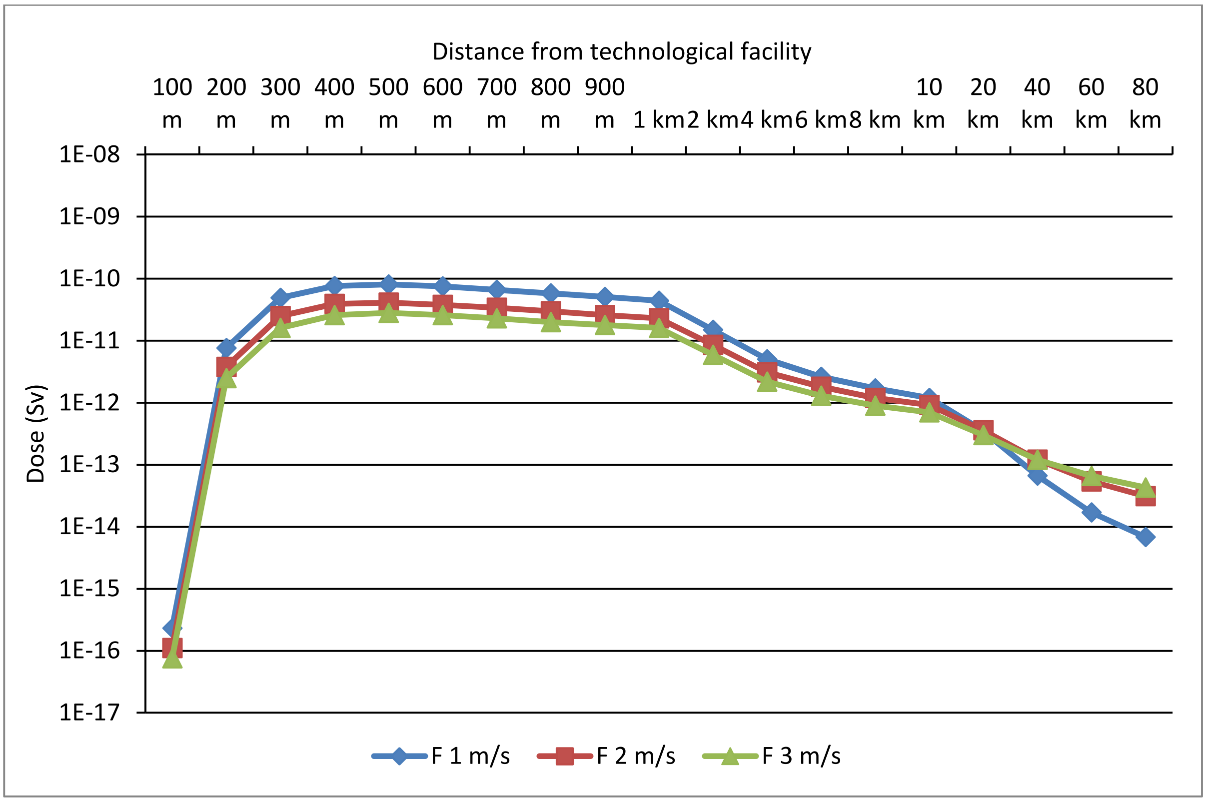

Figure 7.13: Impact of wind on effective dose from FP drop in technological facility at various distances from repository (class F atmospheric stability)

The impact of precipitation on the results of the spread of contamination in the atmosphere in the event of an FP drop in the technological facility was also examined within the framework of the sensitivity analysis. Various quantities of precipitation were taken for individual atmospheric classes and wind speeds, and their impact on the final results (effective dose) was estimated. The results are illustrated in the following figures (Figure 7.14, Figure 7.15, Figure 7.16 and Figure 7.17).

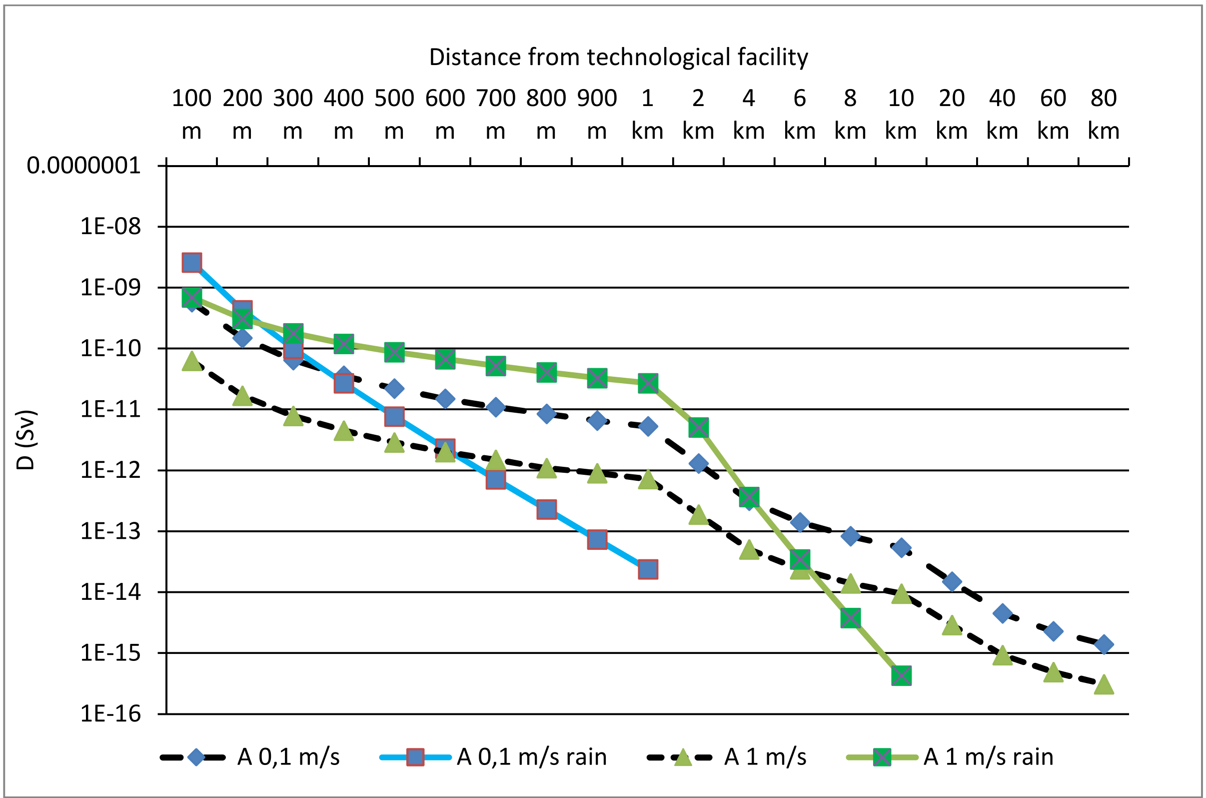

Figure 7.14: Impact of precipitation in combination with wind on effective dose from FP drop in technological facility at various distances from repository (class A atmospheric stability)

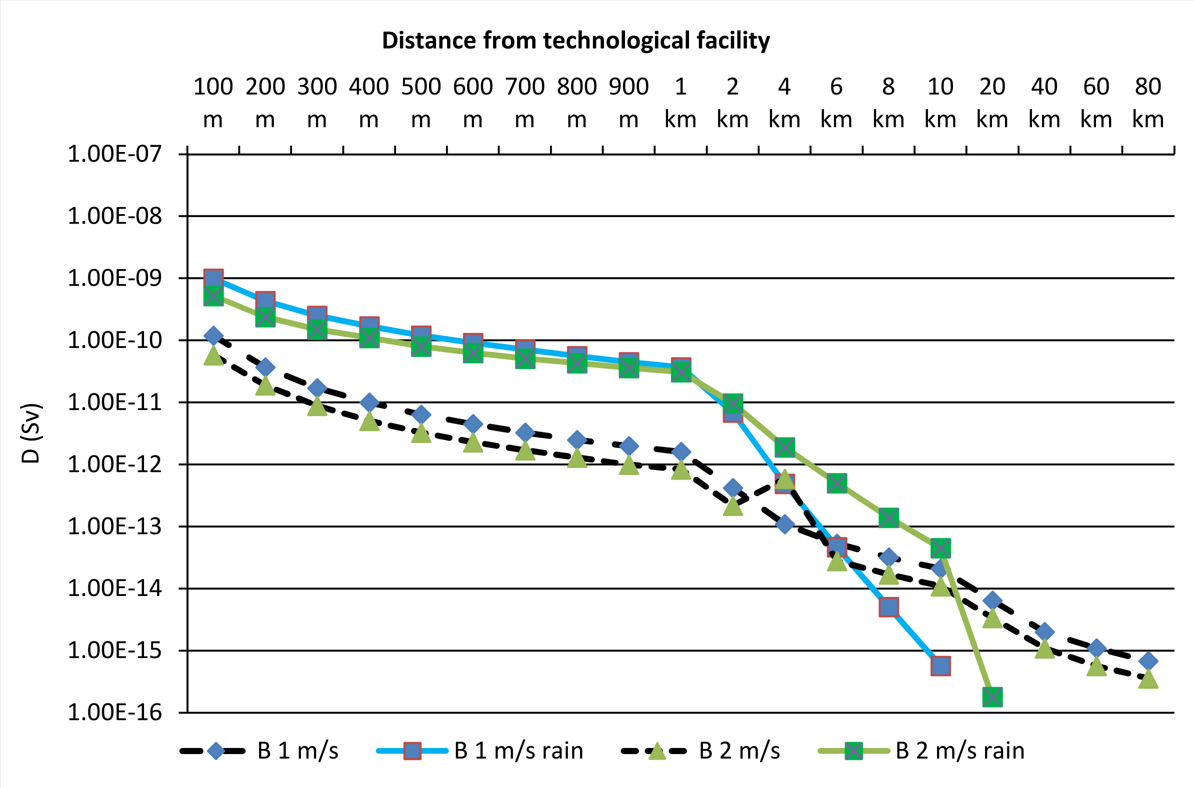

Figure 7.15: Impact of precipitation in combination with wind on effective dose from FP drop in technological facility at various distances from repository (class B atmospheric stability)

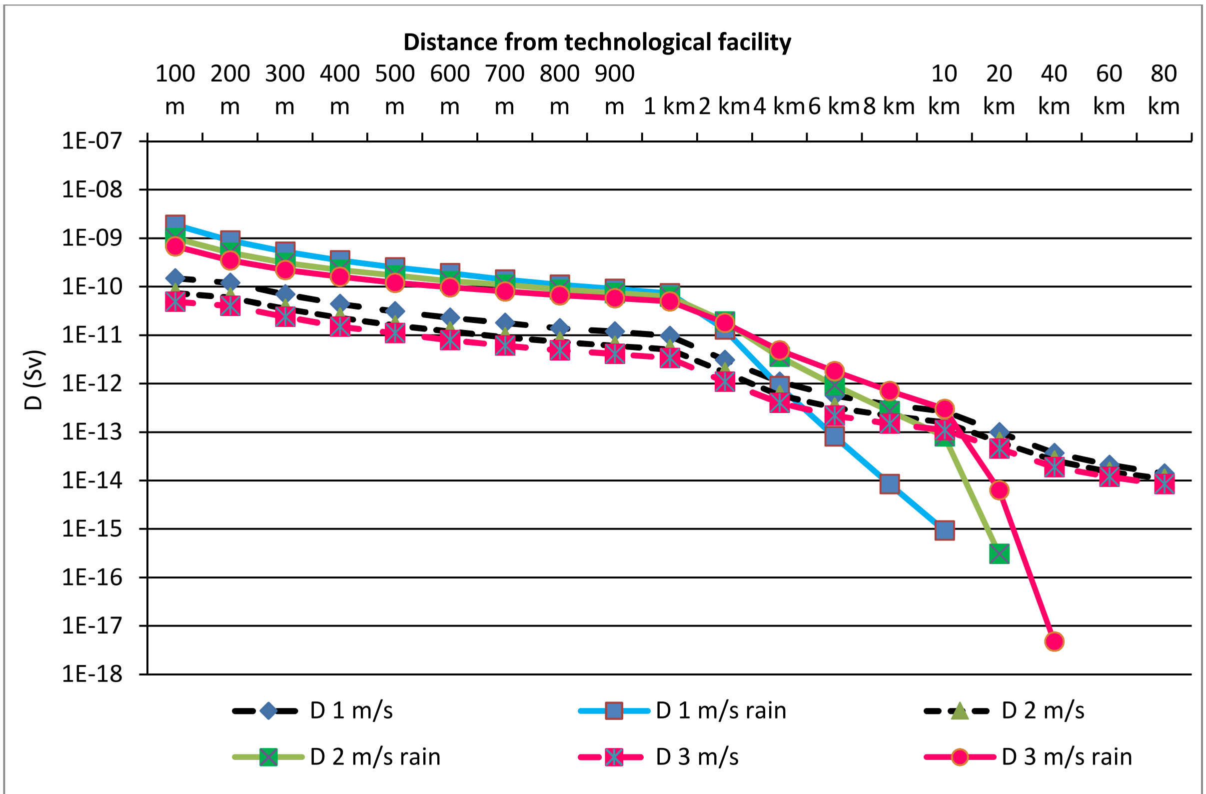

Figure 7.16: Impact of precipitation in combination with wind on effective dose from FP drop in technological facility at various distances from repository (class D atmospheric stability)

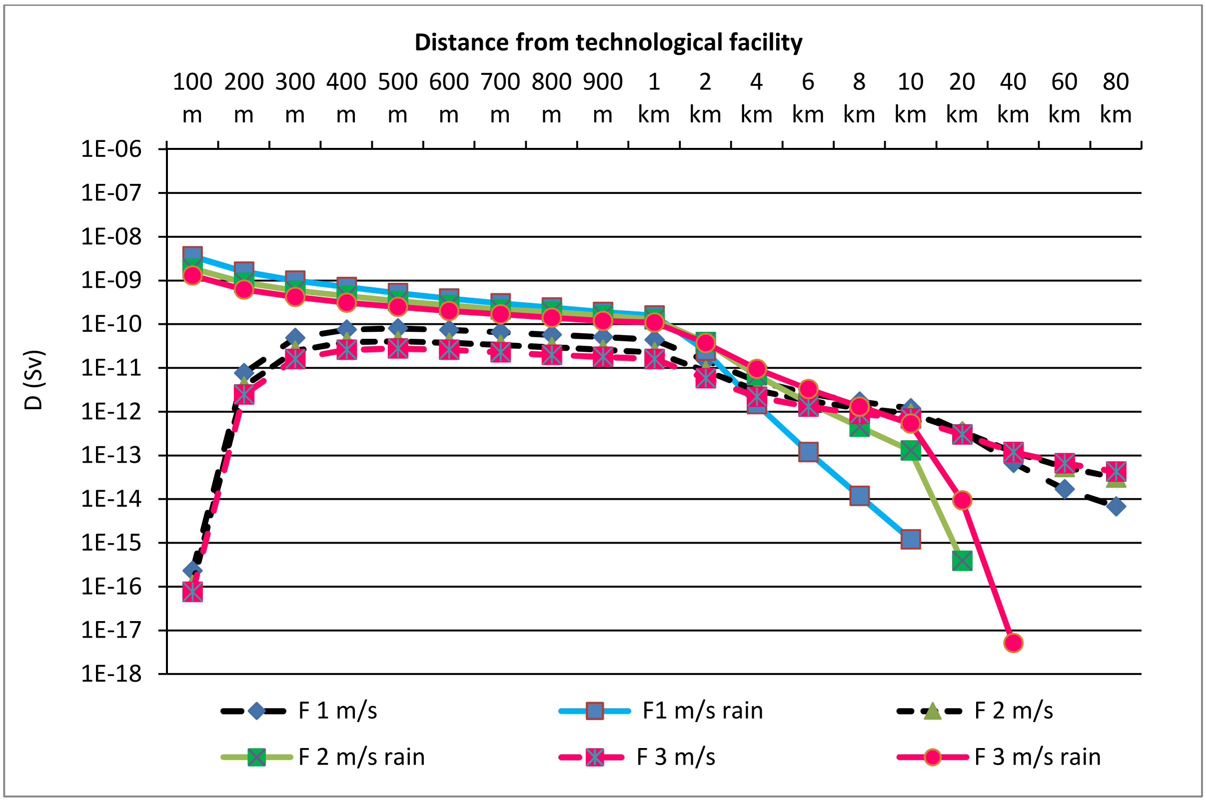

Figure 7.17: Impact of precipitation in combination with wind on effective dose from FP drop in technological facility at various distances from repository (class F atmospheric stability)

The results show that the presence of precipitation in very unstable atmospheric conditions with weak wind (class A) slightly increases the dose close to the repository (up to 400 m away), before the dose reduces, and reaches almost zero at approximately 2 km distant from the repository. In the case of wind of 1 m/s in these conditions, the dose increases close to the repository, then falls, and becomes negligible at a distance of approximately 6 km. In the case of class B, the estimated doses are slightly higher in the combination of wind and precipitation at smaller distances, while at larger distances (between 10 km and 20 km from the repository) they fall rapidly to virtually zero. The results are similar in the case of neutral atmospheric conditions (class D). In the case of relatively stable atmospheric conditions (class F), the results show that the estimated doses increase even more in the actual vicinity of the repository because of the combination of wind and precipitation, but then fall faster as the distance from the source of contamination increases.

On the basis of meteorological data for the Krško region, [49] conclusions can be drawn about the frequency of temperature inversions at an altitude of 90 m to 110 m in the area. The impact of a temperature inversion at an altitude of 100 m on the spread of contamination in the scenario of a fire in the technological facility was therefore calculated for the atmospheric classes A and B, for which this phenomenon is most typical. The results are illustrated in the two figures below (Figure 7.18 and Figure 7.19). It is evident that the temperature inversion has a certain impact on the results, but it is not very substantial. It has a larger impact (the estimated dose is slightly higher) in the vicinity of the event up to about 1 km from the repository, after which the impact of the event is smaller with the temperature inversion than it is without the inversion.

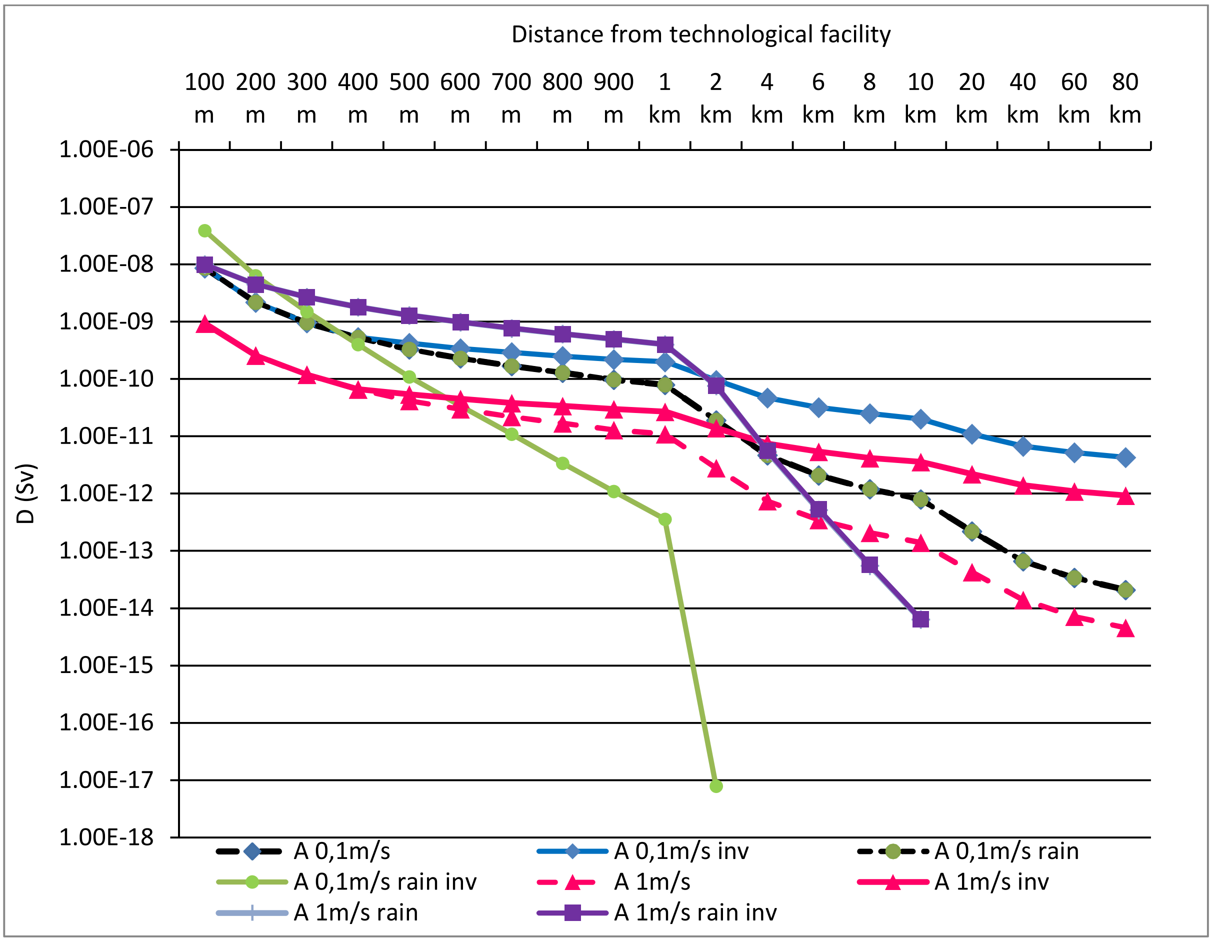

Figure 7.18: Impact of temperature inversion (inv) and rain on effective dose in fire scenario at technological facility in very unstable atmospheric conditions (class A)

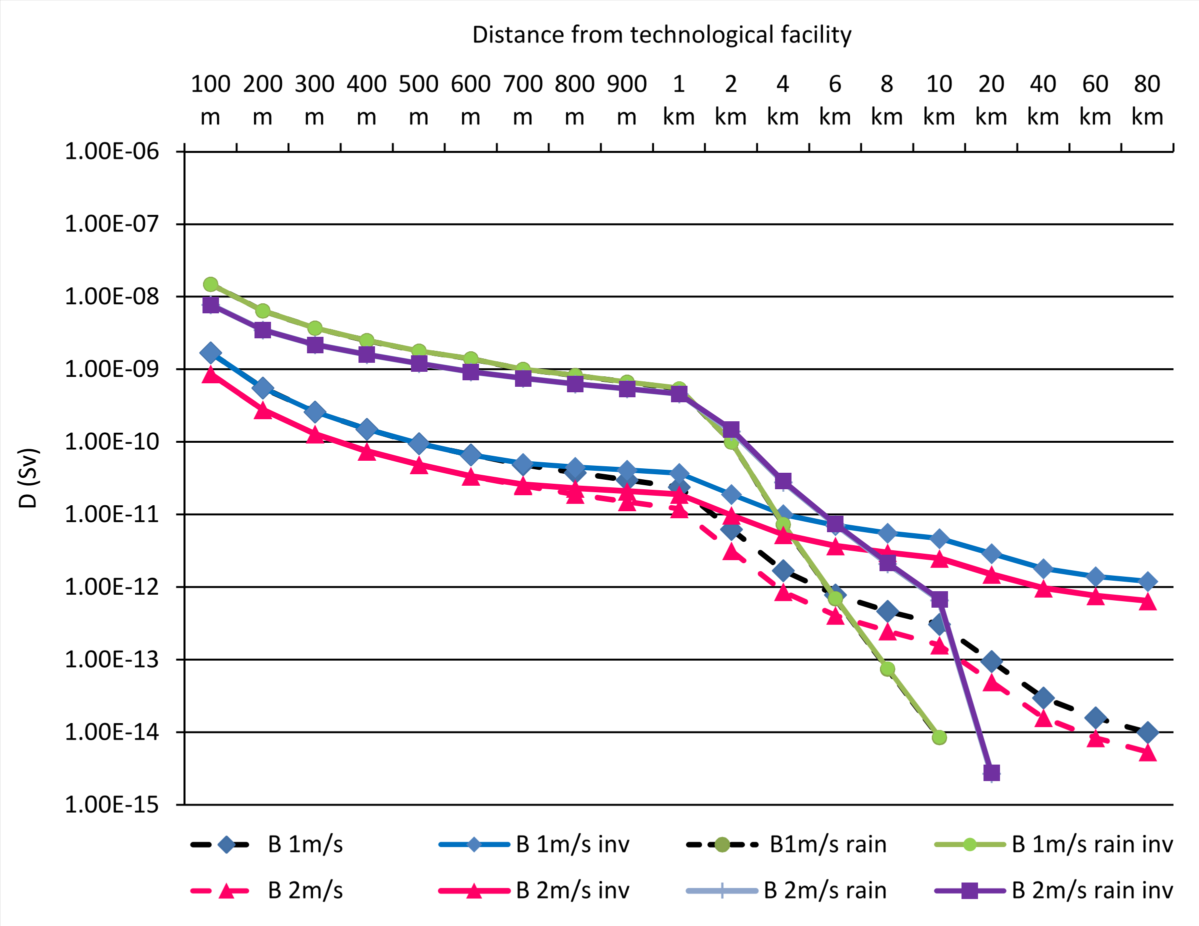

Figure 7.19: Impact of temperature inversion (inv) and rain on effective dose in fire scenario at technological facility in unstable atmospheric conditions (class B)

On the basis of the results of the sensitivity analysis, conservative combinations of weather conditions were taken for the estimation of the effective dose for employees and members of the public during accident scenarios (abnormal operation).

7.2.7 CONCLUSIONS OF THE SAFETY ANALYSIS FOR THE OPERATION OF THE REPOSITORY

The safety analysis for the operation of the LILW repository [17] showed that the proposed concept and design satisfy the safety criteria and that the impact of repository during operation is within the prescribed limits. Certain assumptions were made very conservatively, and the estimated impact of the repository could therefore be even smaller subsequently, when data with less uncertainty is used.

For the scenario of normal evolution during operation, the assessment was that in no case will the dose for an employee exceed the allowed limits. The collective dose for all employees at the repository should not exceed 21.6 person-mSv/year.

In the safety analysis it was conservatively estimated that the dose for a representative member of the public at the perimeter of the repository will not exceed 5 µSv/year for the scenario of normal evolution during operation, if it is assumed that the waste is disposed of after being held in storage for five years. In the case of immediate disposal (fresh waste), the estimated dose for a member of the public at the perimeter of the repository is 11 µSv/year.

The impact of the repository for abnormal evolution scenarios was also estimated, and it was established that in this case too the impact of the repository on employees and members of the public in the worst weather conditions and the worst working conditions is lower than the required minimum reference values under European standards. [50]

The assessment from the safety analysis for the operation of the repository is that the planned LILW repository at the Vrbina site can be operated safely.

7.3 SAFETY ANALYSIS AFTER CLOSURE OF THE LILW REPOSITORY

The following section provides an overview and description of safety analysis for the period after the closure of the LILW repository.

7.3.1 GENERAL METHODOLOGICAL APPROACH

The safety analysis was conducted in accordance with the IAEA recommendations and guidelines, [1] which recommend the use of the methodology sometimes known as the ISAM methodology, which was published as a result of the ISAM project, [50], [24] and was developed along the lines of international best practice. One of the main features of the ISAM methodology is its iterative nature, which means that individual components can be modified as necessary, while it also encourages an improvement in individual assessments by its very nature. Each iteration represents one step in the stage-by-stage approach, and is as long as needed to meet the objective for the phase of the development of repository. The results can also serve as a good tool in optimising the repository and all aspects of its operation. These improvements include changes to the description of the project for the facility (e.g. WACs, design), the scenarios, improvements to the models, and the use of additional data.

The key components of the methodology are illustrated in the figure below. The main feature of the methodology is the identification and evaluation of uncertainty, which connects the components of the methodology. The use of the methodology forces a focus on individual components in each phase (iteration), thereby increasing the confidence in the final decisions, which are thus properly supported, properly documented and coherent.

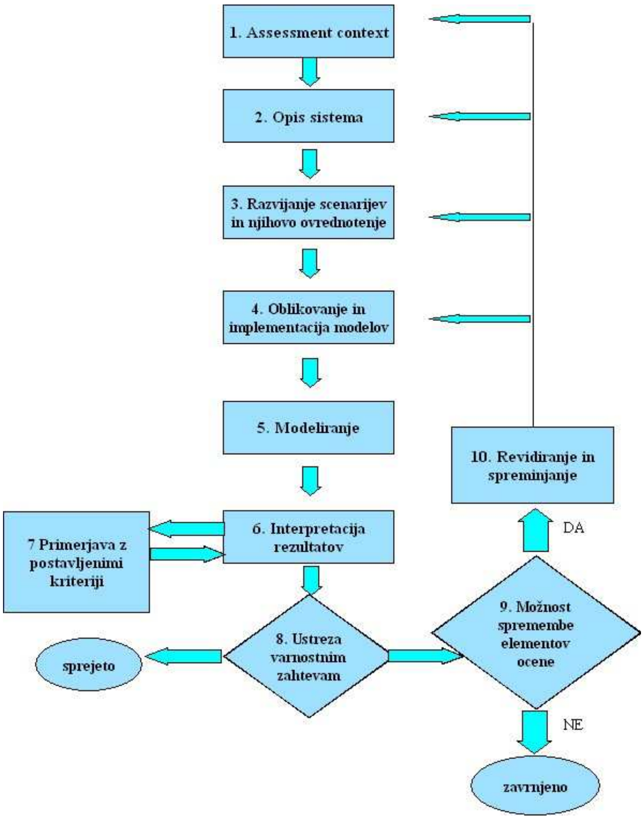

Figure 7.20: Internationally adopted IAEA ISAM methodology taken from [50]

2. Opis sistema	2. Description of system
3. Razvijanje scenarijev in njihovo ovrednotenje	3. Scenario development and evaluation
4. Oblikovanje in implementacija modelov	4. Formulation and implementation of models
5. Modeliranje	5. Modelling
6. Interpretacija rezultatov	6. Interpretation of results
7. Primerjava s postavljenimi kriteriji	7. Comparison with set criteria
8. Ustreza varnostnim zahtevam	8. Fulfilment of security requirements
sprejeto	accepted
9. Možnost spremembe elementov ocene	9. Possibility of modifying evaluation elements
NE	NO
zavrnjeno	rejected
da	yes
10. Revidiranje in spreminjanje	10. Revision and amendments

The purpose of the safety analysis and safety assessments is to develop and convey appropriate assurance that the repository has a negligible long-term environmental impact, in line with the limits defined in legislation. Experience has shown that this is most easily done through analysis and evaluation of uncertainties and sensitivities that is clearly and unambiguously presented. The uncertainties that need to be included are:

- uncertainties in the properties of the site in the future (presented via the use of alternate scenarios [modified evolution scenarios]),

- uncertainties that may arise in the description of various concepts important to the working of the repository (ensuring the safety functions of individual SSCs, use of models),
- uncertainties in the values of parameters used in models.

The general structure of the approach to analysis of uncertainties is set out in the following figure, and leads to a larger number of alternative calculations for the purpose of developing sufficient understanding of the behaviour of the entire system, on the basis of which it is possible to reliably assess its long-term behaviour.

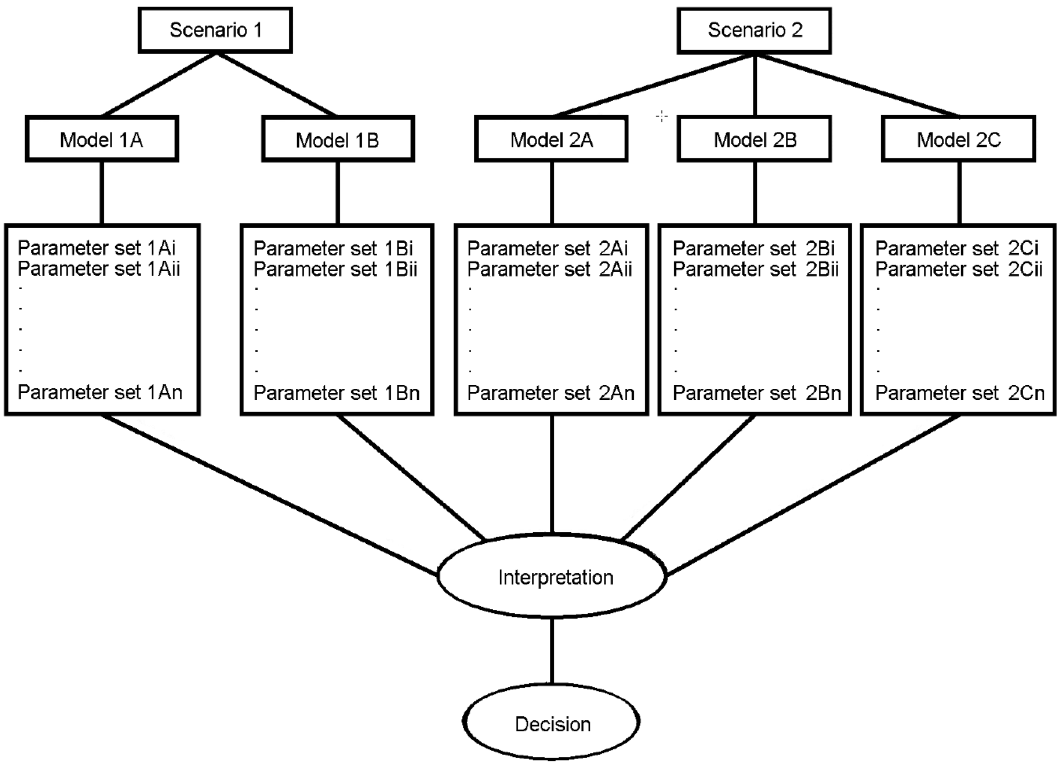

Figure 7.21: General structure of approach to analysis of uncertainties within the framework of safety analysis and safety assessments; taken from Performance Assessment of Low-Level Waste Disposal Facilities [51]

The approach taken in the safety analysis for the LILW repository was one in which a wide range of scenarios, sub-scenarios and parameters were analysed. Credible combinations were identified on the basis of this analysis and these calculations, and were then evaluated. Both deterministic and probabilistic assessments were employed, and together served as the basis for evaluating uncertainties.

In line with the recommendation, [51] the safety analysis in this phase was used for:

- an assessment of environmental impact of the repository and a comparison with legal limits,
- the possibility of improving and optimising design solutions in subsequent phases of the project.

7.3.2 ASSESSMENT CONTEXT: ASSESSMENT OF THE CONTENT OF THE SAFETY ANALYSIS AFTER CLOSURE OF THE REPOSITORY

This section (the assessment context) gives the boundary conditions of the safety analysis, the aim of which is to present the purpose and content to stakeholders. This section aims to clarify what the subject of the safety analysis is, and why the safety analysis is carried out. The IAEA [1] stipulates that the assessment context provides information about the purpose, the administrative framework, the results of safety analysis, the philosophy of assessment, the properties of the disposal system and the timeframes.

The assessment context for the draft safety analysis report is discussed in detail in the safety analysis report. [51]

It is particularly important that there is an awareness that the safety analysis and safety assessments have been designed to take account of Slovenian legal requirements for the long-term assessment of the radiological protection of the public after closure, which are clear and unambiguous. In addition, an assessment was also made of the impact of non-radioactive contaminants (which may be present in LILW) on the environment and on people, in accordance with the Slovenian regulations for the protection of groundwater. Furthermore, international recommendations and approaches have also been taken for certain areas (most notably environmental protection, where Slovenian regulations set out fewer guidelines). The legal framework taken into account by the safety analysis and safety assessments therefore includes all elements that provide for the sufficiently reliable appraisal of the disposal system within the framework of safety analysis, even in areas where Slovenian regulations are ambiguous or non-existent.

The basis and the key administrative constraints that need to be taken into account in the safety analysis are given by the JV5 rulebook: [21]

“Following its closure, the LILW repository may not impose a burden exceeding 0.3 mSv/year on a member of the public under the normal evolution scenario. In cases of alternate evolution scenarios for the repository, the following measurements shall be taken into account for the implementation of measures depending on the burden imposed on a member of the public:

a. up to 10 mSv/year: no measures to optimise the repository are required,
b. above 10 mSv/year: measures are required to minimise the probability of an alternate evolution scenario, and
c. above 100 mSv/year: measures are required to minimise the consequences of the alternate evolution scenario.”

The JV5 rulebook [21] defines the normal evolution and alternate evolution scenarios as follows:

“The normal evolution scenario of a repository is the expected degradation of the state of the facility long after its closure as a result of natural processes or human activities, based on an extrapolation of current conditions into the future.”

“An alternate evolution scenario comprises undesired events or states following the closure of a repository, caused by natural events or by human, animal or plant activity, that accelerate the long-term degradation of the repository and the migration of radioactive substances, and increase radiation. Examples of this scenario are inadvertent human intrusion, water and mineral boreholes, the effects of greenhouse gases, the activation of fractures, global icing, failure of facility seals, and migration that produces gases."

The requirements are consistent with the ICRP recommendations [52] and are clearly defined.

Environmental protection regulations

Annex 4 of the JV5 rulebook [21] states:

"The repository site shall ensure appropriate environmental protection over the entire period of operation and following the closure of the facility. Potential harmful impacts shall be mitigated to acceptable levels, considering economic, social and environmental aspects."

There are no specific requirements of how such an assessment is to be made in relation to the remainder of the biosphere (not just humans). The safety analysis therefore takes on the ICRP recommendations, which state that the estimated concentrations in the environment should be compared with the derived consideration reference levels (DCLRs) for a specific set of reference flora and fauna. [53]

Regulations for non-radioactive toxic materials

In Slovenian legislation it is not clearly defined which recommendations or standards should be taken into account in the production of safety analysis for non-radioactive toxic materials, and the timeframe for this analysis is also undefined. In general (international practice), regulations in the area of toxic materials are drawn up on a different basis to that for radioactive waste, and are therefore difficult to combine and consolidate. [54], [55]

Within the framework of the safety analysis, the Rules on drinking water, [56] which contain limits for heavy metals that are present in individual radioactive waste streams, were used to determine the limit concentrations of individual contaminants. The timeframes used were the same as those applying to radioactive waste.

7.3.2.1 Dose for member of public

The JV5 rulebook [21] stipulates that within the framework of the safety analysis it is necessary to estimate the dose for an individual member of the public, but does not stipulate the type of dose, or how to select the individual or group of individuals in question for this estimate, or how to determine the area where potentially exposed individuals are located. The recommendations from the ICRP [52], [57] were therefore used in the safety analysis.

It was assumed that the potential radioactive contamination of the biosphere would be relatively constant over a period that is much longer than the average human lifespan. It is therefore reasonable to estimate the annual dose for a member of the public that is the average over their lifetime, which means that it is not necessary to calculate doses for individual age groups, as their average is represented by the annual dose for an adult. In this context it was assumed that the "dose" refers to the "effective dose" that an individual could receive over 70 years owing to ingestion and external irradiation, which is also in line with the ICRP recommendations. [57] In line with the recommendations, [52] a critical group of individuals was designated for the purposes of the safety analysis, as presented below (Section 7.3.6.1).

7.3.2.2 Scenario of inadvertent human intrusion

The use of a scenario of inadvertent human intrusion into the repository for the assessment of the concentrations suitable for a near-surface facility comes from the 1970s. [58] In line with this approach, radioactive waste is acceptable for disposal in a near-surface facility if the concentrations are low enough that the potential doses received by a person who intrudes into the repository are still acceptable. If the concentrations of radioactive waste are so high that the potential doses during an intrusion are not acceptable, the waste cannot be disposed of in such a facility, and a different (deeper) concept is required for the repository. The subsection entitled Scenario of inadvertent human intrusion(Section 7.3.6.6) gives the estimated doses for a member of the public in the scenario of inadvertent human intrusion.

The basic principles for preventing a scenario of inadvertent human intrusion are the following: [59]

- controlling the disposal of specific waste streams,
- waste form, packaging and planning,
- natural barriers (primarily depth), and
- institutional controls.

Various national programmes have co-opted and adapted the strategy of protecting inadvertent intruders by placing greater emphasis on one of the aforementioned principles.

These assumptions are also important for the LILW repository at Vrbina, primarily for the following reasons:

- certain waste streams that will be disposed of at the repository could have higher concentrations than is typical for near-surface repositories,
- the attributes of the disposal concept are such that the probability of the scenario of inadvertent human intrusion is significantly lower than for other standard repositories,
- the repository is planned such that the consequences of any inadvertent intrusion are less than for standard repositories.

On the basis of these observations, the scenario of inadvertent human intrusion was also assessed for the LILW repository. In line with the ICRP recommendations, [60] within the framework of the safety analysis people living in the environs of the repository site were considered potential dose recipients under this scenario. The potential doses for the intruder were also estimated. It is necessary to distinguish between the normal behaviour of people living in the area, and events of short duration and/or low probability that can have an impact on a small number of people only. This means that, similarly to industrial accidents, it is not reasonable to apply the same criteria to the limits on the potential dose received by an inadvertent intruder as to the doses received by those only living in the vicinity of the repository. Accordingly, the scenario of inadvertent human intrusion was treated as an alternate evolution scenario [54] and the limits applying to such events were taken into account. The same limits were also taken for intruder and for those living in the vicinity of the repository after the intrusion.

7.3.2.3 Impact on non-human biota

Until recently the approach to repositories after closure was that it was sufficient to assess the impact on the safety of people alone, under the assumption that the protection of humans entailed the protection of other organisms and the environment. Under the new ICRP recommendations, [60] it is now necessary to directly and explicitly demonstrate that the impact of the LILW repository on the environment is negligible. The ICRP [54] recommended the use of an updated approach to environmental impact assessment. A list of reference animals and plants (RAPs) was drawn up (see Table 7.21) and a DCRL determined for each of them. The DCRL is defined as the dose rate at which there could be a harmful impact on an individual RAP.

Table 7.21: List of reference animals and plants

land organisms	freshwater organisms
invertebrate detritivores	amphibians
invertebrates (worms)	molluscs
snails	snails
amphibians	crustaceans
birds, bird eggs	benthic fish
flying insects	pelagic fish
reptiles	insect larvae
small mammals (rats)	aquatic birds (ducks)
large mammals (deer)	aquatic mammals
mosses and lichens	zooplankton
grasses and herbs	phytoplankton
bushes and shrubs	higher plants
trees	

The safety analysis for the Vrbina LILW repository made use of the Erica tool [61] to calculate doses for individual RAPs, which were compared with the DCRLs. [54]

7.3.2.4 Safety assessment for non-radioactive toxic materials

In their safety assessments some national programmes were already taking account of the risks inherent in non-radioactive toxic elements, where it should be noted that the level of the impact assessment of toxic elements is well below the level of the impact assessment of radioactive materials. The main reason is that it is extremely difficult to define the toxic components in radioactive waste.

The safety analysis for the LILW repository includes an evaluation of the impact of toxic components on people. The toxic components known to be in radioactive waste were used for this purpose. Calculations of the impact on people were made for them, and compared with the criteria for drinking water [56] (see previous section on regulations). The location of the well and the contaminated water was the same as for the estimation of the effective dose for a member of the critical population group in the drinking of contaminated water.

7.3.2.5 Timeframe of safety analysis

In Slovenian legislation the timeframe of the safety analysis is defined in the JV5 rulebook, [21] which states:

"Potential risks to the existing and envisaged future population of the site region arising from the disposal facility shall be acceptable."

This recommendation was taken into account in the safety analysis through the application of international recommendations and experience in connection with the timeframes, which are presented below.

7.3.2.5.1 Approach to period of institutional control

As a nuclear facility, the repository will be controlled by a government authority. At some point in the future the facility will lose the status of a nuclear facility, and will thus come into free use. Given the length of the period for which radioactive waste can be dangerous, and the recommendations [21] on the acceptability of risk for future generations, it is therefore necessary to envisage that at some point in the future a combination of events will cause institutional controls to cease, and knowledge of the repository site to be lost. This is one of the principal assumptions in international regulations and in the recommendation for LILW repositories.

Because Slovenian legislation contains no specific requirements with regard to the duration of institutional controls, the duration needs to be defined, particularly for the purpose of conducting the safety analysis. The main purpose of institutional controls is to ensure that there is time for the short-lived radionuclides in waste to decay, which influences the scenarios that might occur only after the cessation of institutional controls (e.g. the inadvertent intrusion scenario).

For the purposes of the safety analysis it was assumed that the repository would be subject to institutional controls for 300 years after its closure. This period is divided into periods of active and passive controls, which are presented in Section 12 of this draft safety analysis report, which is not of significance for the safety analysis.

7.3.2.5.2 Approach to long-term safety analysis

The timeframe for conducting safety analysis must take account of certain key concepts. The first is the IAEA concept [62] that requires the protection of present and future generations. This principle is the reason that safety analysis is conducted over the long term. There needs to be an awareness that the long timeframes mean that we are dealing with very large uncertainties, which hugely reduces the significance (the value) of safety analysis. [63], [64], [65]

In its Publication 81 [53] the ICRP therefore presented revised proposals of the timeframes for repositories, and introduced the concept of the critical group. ICRP 81 recognises the use of alternative, complementary indicators as an addition to the calculation of dose and risk. The BIOMASS report [66] highlights the use of societal assumptions. The following societal assumptions were thus used within the framework of the safety analysis for the LILW repository:

- the level of technological development is similar to the current level, and exploitation of the environment (food supply) has also developed consistently with this, and the emphasis is on the exploitation of local resources (in contrast to imports), which is closer to the past than the present,
- there is no reliance on improvements in radiation detection techniques or any other scientific development that could help to reduce exposure to radiation (in terms of probability and in terms of magnitude),
- there is no reliance on improvements in the diagnostics and treatment of cancer and other adverse effects caused by radiation.

The following assumptions were made for the Vrbina LILW repository:

- For the purposes of the preparation of the safety analysis, it was assumed that the repository would be subject to institutional controls for 300 years after its closure. After this period there will no longer be controls (neither active nor passive), and there will no longer be any knowledge of the repository or the site.
- The doses estimated in the safety analysis for the first 10,000 years after closure were compared directly with the legal limit. All calculations were made for a time period during which the maximum total dose and dose for individual radionuclides was reached. When the maximum calculated dose (peak) is estimated in a period of more than 10,000 years after the closure of the repository, it is taken into account as a qualitative estimate, which in practice means that the maximum doses that occur several hundred thousand years after the closure of the repository are also diligently interpreted and compared with the legal limits.

7.3.3 DESCRIPTION OF SYSTEM

7.3.3.1 Site of the LILW repository

The repository site and its properties as used in the safety analysis are presented in Section 4.

7.3.3.2 Radioactive waste: inventory

The following section summarises the description of the radioactive waste (inventory) that will be disposed of at the LILW repository. The inventory is described in detail in reports. [19] The main purpose of this section is to present the key properties of the waste that affected the repository modelling process.

The radioactive waste has four sources, which are examined below:

- waste generated in the decommissioning of Krško NPP,
- waste from the operation of Krško NPP,
- waste stored in the central radioactive waste storage facility in Brinje,
- waste generated in the decommissioning of the TRIGA reactor at the IJS,
- operational waste from the LILW repository,
- decommissioning waste from the LILW repository.

The properties of the waste under the disposal conditions (swelling, potential chemical reactions in the saturated environment, corrosion, etc.) were taken into account in the safety analysis in the concrete degradation model. Swelling is taken into account in the sense that waste whose swelling will not damage engineered barriers will be disposed of at the repository. This can be ensured, essentially, in two ways:

- ensuring that the packages containing waste that can swell have sufficient space to be able to swell, and that the swelling process will take place in a way that can be adequately predicted and described, or

- ensuring that waste that can swell is appropriately processed (conditioned for disposed so that it will not swell any further).

The WACs will stipulate in which of the two proposed methods waste that can swell will be disposed of.

The ARAO will try to recycle as much radioactive waste as possible in the future. This will entail a reduction in the quantity of the inventory. The maximum possible inventory was assumed in the safety analysis, and the impact was determined for this inventory. Should the inventory be smaller, the impact of the repository on the environment and on people will be smaller.

The latest estimated inventory will be taken into account in each revision of the safety report.

7.3.3.2.1 Breakdown of wastes by type of material

Decommissioning of Krško NPP

The planned waste from the decommissioning of Krško NPP is expected to include:

- activated components generated by neutron activation of materials in or in proximity to the reactor vessel, the reactor vessel itself, the internal parts of the reactor vessel, and the concrete biological shield,
- two steam generators made of nickel alloy and carbon steel,
- contaminated components inside the controlled area and supervised area of Krško NPP (metals and concrete),
- supercompacted combustible wastes,
- supercompacted non-combustible wastes,
- liquids: effluents (grouted),
- concrete rubble.

In terms of volume and radionuclide activity, the largest contribution comes from the reactor vessel waste.

Operation of Krško NPP

This waste includes radioactive waste generated during the operation of the power station, regular refits and maintenance, including the replacement and upgrade of various components. The key waste streams are generated in the cleaning of the primary cooling water from the reactor. The following waste streams are identified from the operation of Krško NPP:

- activated carbon,
- combustion products (including residues of aluminium and heavy metals),
- evaporator concentrates and dried sludge (sediment),
- spent ion exchange resins,

- spent filters,
- compressible waste,
- non-compressible waste.

Two previously replaced steam generators are also classed as operational waste.

Brinje central radioactive waste storage facility

Smaller quantities of radioactive waste (in terms of volume and in terms of activity) are located at the central radioactive waste storage facility in Brinje. Similar quantities can be expected in the future, and will be stored at the storage facility before disposal. This waste incudes:

- spent sealed sources,
- combustible and compressible operational waste from the IJS (the TRIGA reactor and hot cell),
- smoke detectors,
- other waste generated in industry, research, etc.

The key materials are carbon steel, stainless steel, aluminium, iron, depleted uranium, cellulose, wool, clothing and other combustible wastes.

IJS

The dominant wastes will be generated in the decommissioning of the TRIGA reactor, and generally include concrete, steel and aluminium.

7.3.3.2.2 Inventory of radioactive waste

The inventory of radioactive waste taken for the preparation of the safety analysis was divided into two parts. The first classifies waste in terms of the type of material (mass and volume), and the second in terms of the radionuclides that the waste contains (activity).

7.3.3.2.3 Inventory in terms of type of material

The assessment of the inventory in terms of type of material is given in Table 7.22.

Table 7.22: Mass of various materials in radioactive waste

Material	Krško NPP operational waste, 2043 decommissioning [kg]	Krško NPP decommissioning waste, 2043 decommissioning [kg]	IJS (TRIGA) decommissioning waste [kg]	CSRAO 2050 forecast [kg]	repository operation [kg]	repository decommissioning waste [kg]	total [kg]
aluminium	7.49E+04	4.96E+04	1.53E+03	1.16E+04	7.40E+02	4.95E+02	1.38E+05
antimony	3.80E+01	6.10E+01	0.00E+00	1.00E+00	0.00E+00	0.00E+00	1.00E+02
arsenic	5.20E+01	2.67E+02	1.00E+00	4.00E+00	1.00E+00	2.00E+00	3.24E+02
ash	3.28E+04	1.40E+04				1.40E+02	4.68E+04
boron	1.66E+04	3.39E+03	4.24E+02	0.00E+00	0.00E+00	1.70E+01	2.05E+04
brass		3.40E+03				3.50E+01	3.40E+03
cadmium	2.20E+01	8.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.00E+01
cellulose	7.09E+04	9.20E+03		7.00E+03	7.10E+02	9.50E+01	8.71E+04
carbon	1.81E+04						1.81E+04
chromium	3.12E+04	1.07E+05	3.90E+01	8.06E+02	3.00E+02	6.91E+02	1.39E+05
concrete		1.70E+06	2.12E+05			8.73E+03	1.91E+06
copper	1.41E+05	1.03E+04	1.30E+01	3.70E+01	1.41E+03	1.00E+02	1.52E+05

Material	Krško NPP operational waste, 2043 decommissioning [kg]	Krško NPP decommissioning waste, 2043 decommissioning [kg]	IJS (TRIGA) decommissioning waste [kg]	CSRAO 2050 forecast [kg]	repository operation [kg]	repository decommissioning waste [kg]	total [kg]
carbon steel / iron / other ferrous materials	3.57E+05	2.25E+06	9.00E+03	3.30E+04	3.58E+03	1.70E+04	2.65E+06
depleted uranium				1.50E+02			1.50E+02
glass fibre	4.11E+03				4.00E+01		4.11E+03
spent ion exchange resins (organic fraction)	1.13E+05	4.41E+03				4.50E+01	1.17E+05
lead	2.15E+02	3.74E+03	0.00E+00	3.00E+00	1.00E+00	4.20E+01	3.95E+03
mercury	6.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.00E+00
nickel	1.84E+04	1.59E+05	1.00E+01	4.56E+02	1.77E+02	9.50E+02	1.78E+05
nickel alloys		1.62E+05				9.20E+02	1.62E+05
organic (mixture of plastic/cellulose)	4.80E+02					9.50E+01	4.80E+02
organic (mixture of plastic/rubber)	4.89E+05	9.20E+03			4.90E+03		4.99E+05
organic (other)	4.11E+03				4.00E+01	2.00E+01	4.11E+03

Material	Krško NPP operational waste, 2043 decommissioning [kg]	Krško NPP decommissioning waste, 2043 decommissioning [kg]	IJS (TRIGA) decommissioning waste [kg]	CSRAO 2050 forecast [kg]	repository operation [kg]	repository decommissioning waste [kg]	total [kg]
other	7.20E+04	2.17E+04	5.07E+03		1.43E+03		9.88E+04
selenium	3.00E+01	7.50E+01	0.00E+00	1.00E+00	0.00E+00	1.00E+00	1.06E+02
sludges/concentrates	9.51E+05						9.51E+05
stainless steel	1.52E+05	3.74E+05		3.70E+03	1.47E+03	2.50E+03	5.30E+05
undefined metals				1.33E+04			1.33E+04
total	2.69E+06	4.90E+06	2.28E+05	7.01E+04	1.62E+04	3.20E+04	7.88E+06

Table 7.23: Volume of various materials in radioactive waste

waste stream	Krško NPP operational waste, 2043 decommissioning [m^3]	Krško NPP decommissioning waste, 2043 decommissioning [m^3]	IJS (TRIGA) decommissioning waste [m^3]	CSRAO 2050 forecast [m^3]	repository operation [m^3]	repository decommissioning waste [m^3]	total [m^3]
Evaporator concentrates and sludges	3.90E+03						3.90E+03
Spent ion exchange resins (SIRs)	1.40E+03						1.40E+03
Spent filters (SFs)	1.50E+02						1.50E+02
Compressible waste (CPW)	2.80E+03				2.70E+01		2.83E+03
Non-compressible waste (NCW)	1.60E+03				1.60E+01		1.62E+03
Special waste (SW)	1.00E+02						1.00E+02
Movable internal parts of reactor vessel		5.70E+01					5.70E+01
Steam generators		4.50E+02[1]					4.50E+02
Other activated material		4.55E+02					4.55E+02
Concrete		2.41E+03	3.00E+02			3.60E+00	2.72E+03
Contaminated components		2.42E+03[2]				6.50E+01	2.49E+03

[1] Includes two steam generators included in decommissioning
[2] Includes two previously replaced steam generators

waste stream	Krško NPP operational waste, 2043 decommissioning [m^3]	Krško NPP decommissioning waste, 2043 decommissioning [m^3]	IJS (TRIGA) decommissioning waste [m^3]	CSRAO 2050 forecast [m^3]	repository operation [m^3]	repository decommissioning waste [m^3]	total [m^3]
Combustible wastes		2.40E+02				5.00E+00	2.45E+02
Non-combustible wastes		3.87E+02				1.00E+01	3.97E+02
Grouted liquids (total)		1.39E+03					1.39E+03
Grouted quantities (spent ion exchange resins, assessment of unprocessed waste)		9.00E+00					9.00E+00
Steel			2.40E+01				2.40E+01
Aluminium			1.60E+01	1.60E+02			1.76E+02
Other			1.60E+01				1.60E+01
Stainless steel				3.00E+01			3.00E+01
Steel				1.00E+02			1.00E+02
Undefined metals				1.00E+02			1.00E+02
Iron and cast iron				1.00E+02			1.00E+02

waste stream	Krško NPP operational waste, 2043 decommissioning [m^3]	Krško NPP decommissioning waste, 2043 decommissioning [m^3]	IJS (TRIGA) decommissioning waste [m^3]	CSRAO 2050 forecast [m^3]	repository operation [m^3]	repository decommissioning waste [m^3]	total [m^3]
Depleted uranium				0.00E+00			0.00E+00
Cellulose				2.00E+02			2.00E+02
Total	9.95E+03	4.95E+03	3.56E+02	6.90E+02	4.30E+01	8.36E+01	1.61E+04

7.3.3.2.4 Inventory in terms of radionuclides

Table 7.24 cites the inventory of radionuclides with regard to the six separate waste streams that will be disposed of at the LILW repository. The reference year for the operational waste from Krško NPP is taken to be 2043, when the plant is due to shut down operations. The reference year for decommissioning waste is taken to be 2049, six years after shutdown. In the assessment of the inventory, the values cited in the table are conservative, in line with the inventory report. [67] Only radionuclides that have a half-life longer than one year, and consist primarily of fission products, products of neutron activation and transuranic elements are included in the inventory. The calculations themselves also take account of radionuclides in decay chains that have a half-life of less than one year.

With regard to the inventory breakdown according to activity, the (assessed) inventory from the decommissioning of Krško NPP is three orders of magnitude larger than the operating inventory, which is in turn approximately two orders of magnitude larger than the other two waste streams. Approximately 90% of the activities of operational waste is contained in the spent ion exchange resins, while waste from the reactor vessel is the dominant form of decommissioning waste in terms of activity.

Within the framework of the safety analysis it was assumed that the entire inventory is disposed of at the LILW repository. Under the methodological approach taken to the safety analysis, the number of silos built has no impact on the results. The nearfield and farfield models in the safety analysis are first used to calculate flows, over which the number of silos has no effect. These parameters are then taken into account in the system model, together with the entire inventory.

Table 7.24: Radionuclides in individual waste streams

radionuclide	half-life [y]	Krško NPP operational waste, 2043 decommissioning [Bq]	Krško NPP decommissioning waste, 2043 decommissioning [Bq]	IJS (TRIGA) decommissioning waste [Bq]	CSRAO 2050 forecast [Bq]	repository operation [Bq]	repository decommissioning waste [Bq]	total [Bq]
H-3	1.20E+01	9.86E+08	1.23E+15		5.41E+11	9.45E+03		1.23E+15
C-14	5.70E+03	2.86E+12	4.30E+13	3.51E+10	9.87E+07	2.55E+07	2.40E+07	4.59E+13
Na-22	2.60E+00				5.22E+06			5.22E+06
Cl-36	3.00E+05	2.15E+07	2.54E+11		1.20E+03	1.92E+02	3.00E+06	2.54E+11
Ar-39	2.65E+02			1.39E+08	0.00E+00			1.39E+08
Ca-41	1.00E+05		6.21E+12	1.64E+07	0.00E+00			6.21E+12
Mn-54	<1		1.30E+15				1.60E+09	1.30E+15
Fe-55	2.70E+00	1.88E+12	1.10E+17	6.25E+11	4.10E+09	1.96E+07	5.30E+10	1.10E+17
Ni-59	7.50E+04	1.07E+12		8.51E+08	5.28E+09	9.58E+06		1.08E+12
Co-60	5.30E+00	3.74E+12	9.00E+16	4.36E+12	4.48E+12	3.83E+07	3.10E+10	9.00E+16
Ni-59	7.50E+04		2.10E+14				2.30E+08	2.10E+14
Ni-63	9.60E+01	1.19E+14	3.00E+16	9.71E+10	6.41E+09	1.07E+09	2.60E+10	3.01E+16
Kr-85	1.08E+01				1.30E+11			1.30E+11
Se-79	3.30E+05	1.75E+08				6.70E+02		1.75E+08

radionuclide	half-life [y]	Krško NPP operational waste, 2043 decommissioning [Bq]	Krško NPP decommissioning waste, 2043 decommissioning [Bq]	IJS (TRIGA) decommissioning waste [Bq]	CSRAO 2050 forecast [Bq]	repository operation [Bq]	repository decommissioning waste [Bq]	total [Bq]
Sr-90	2.90E+01	2.58E+12	5.99E+11		2.34E+10	1.01E+07	1.80E+07	3.20E+12
Nb-93m	1.36E+01			1.80E+05	0.00E+00			1.80E+05
Nb-94	2.00E+04	3.58E+10	3.00E+12	2.00E+06	0.00E+00	3.19E+05	1.70E-02	3.04E+12
Tc-99	2.10E+05	7.13E+10	2.31E+11		1.06E+07	4.38E+05	1.30E+06	3.02E+11
Pd-107	6.50E+06	4.35E+07				1.67E+02		4.35E+07
Ag-108m	4.20E+02	2.06E+09	3.70E+13		3.64E+06	1.84E+04		3.70E+13
Ag-110m	<1		2.10E+13					2.10E+13
Cd-109	1.30E+00				3.84E+08			3.84E+08
Cd-113m	1.41E+01		1.10E+13		2.52E+08			1.10E+13
Sb-125	2.80E+00	1.93E+11	1.02E+13		0.00E+00	2.03E+06	4.20E+08	1.04E+13
I-129	1.60E+07	1.30E+08	1.85E+06		1.44E+04	5.02E+02	1.80E+03	1.32E+08
Ba-133	1.05E+01	8.51E+07	2.00E+11	9.36E+09	3.69E+06	8.25E+02		2.09E+11
Cs-134	2.10E+00	2.52E+12	1.61E+11		9.36E+06	1.05E+07	5.10E+07	2.68E+12
Cs-135	2.30E+05	4.35E+08	8.21E+06		0.00E+00	1.67E+03	8.00E+03	4.43E+08
Cs-137	3.00E+01	2.61E+13	2.57E+12		1.77E+12	1.02E+08	1.20E+09	3.04E+13

radionuclide	half-life [y]	Krško NPP operational waste, 2043 decommissioning [Bq]	Krško NPP decommissioning waste, 2043 decommissioning [Bq]	IJS (TRIGA) decommissioning waste [Bq]	CSRAO 2050 forecast [Bq]	repository operation [Bq]	repository decommissioning waste [Bq]	total [Bq]
Sm-151	9.00E+01	1.09E+11	1.40E+11			4.23E+05		2.49E+11
Eu-152	1.30E+01	1.05E+09	4.50E+12	9.36E+08	7.67E+11	4.19E+03	0.00E+00	5.27E+12
Eu-154	8.80E+00	1.03E+12	2.00E+11	9.24E+07	1.70E+10	4.18E+06	1.30E+05	1.25E+12
Eu-155	5.00E+00	3.96E+11	7.80E+10		2.23E+06	1.64E+06	2.30E+04	4.74E+11
Tl-204	3.80E+00				5.64E+06			5.64E+06
Pb-210	2.10E+01				2.61E+08			2.61E+08
Ra-226	1.60E+03				6.31E+10			6.31E+10
Ra-228	5.70E+00				4.28E+06			4.28E+06
Th-228	1.90E+00				3.72E+08			3.72E+08
Th-232	1.40E+10				8.00E+07			8.00E+07
U-234	2.40E+05	6.09E+07	4.64E+08		0.00E+00	2.34E+02	4.00E+05	5.25E+08
U-235	7.00E+08	1.22E+06	1.85E+07		1.81E+08	4.69E+00	1.80E+04	2.01E+08
Np-237	2.10E+06	2.44E+07	2.05E+06		0.00E+00	9.37E+01	8.50E+02	2.65E+07
U-238	4.50E+09	2.44E+07	3.94E+08		2.31E+10	9.37E+01	3.90E+05	2.35E+10
Pu-238	8.80E+01	2.70E+10	2.11E+09		0.00E+00	1.05E+05	2.00E+06	2.91E+10

radionuclide	half-life [y]	Krško NPP operational waste, 2043 decommissioning [Bq]	Krško NPP decommissioning waste, 2043 decommissioning [Bq]	IJS (TRIGA) decommissioning waste [Bq]	CSRAO 2050 forecast [Bq]	repository operation [Bq]	repository decommissioning waste [Bq]	total [Bq]
Pu-239	2.40E+04	6.08E+09	1.53E+10		4.28E+09	2.34E+04	5.30E+06	2.57E+10
Pu-240	6.50E+03		3.27E+08		0.00E+00		3.10E+05	3.27E+08
Pu-241	1.40E+01	2.23E+11	1.54E+12		0.00E+00	8.91E+05	2.40E+08	1.76E+12
Am-241	4.30E+02	6.83E+09	3.20E+10		3.23E+11	2.63E+04	3.00E+06	3.62E+11
Am-241/Be	4.30E+02				6.58E+11			6.58E+11
Cm-244	1.80E+01	1.30E+10	1.27E+09		1.06E+10	5.16E+04	1.20E+06	2.49E+10
Totals		1.62E+14	2.33E+17	5.13E+12	8.83E+12	1.30E+09	1.14E+11	2.33E+17

*the highest contributions of individual waste streams are in bold

7.3.3.3 *Inventory of toxic substances within inventory of radioactive waste*

The toxic substances that have been addressed in the safety analysis are taken from the report on the inventory of radioactive waste. [68]

More detailed data on the content of toxic substances within the inventory of radioactive waste is not available. The toxic substances (primarily heavy metals) were therefore assessed with regard to the possibility of their occurrence within other recognised materials, and the recognised quantities of these toxic substances within other recognised materials. The uncertainty of this assessment is acceptable, unless the impact assessments of the quantities of these substances are close to or exceed the given limits.

The basis for drawing up the list of toxic substances within the inventory was the Rules on drinking water, [57] which cites a list of toxic substances. The estimated quantities of these toxic substances in individual waste streams are cited in Table 7.25.

Table 7.25: Toxic substances (mass) in individual waste streams (all figures in kg)

	waste stream mass	antimony	arsenic	boron	cadmium	chromium	copper	lead	mercury	nickel	selenium
stainless steel	551,375	6	55	0	11	110,275	551	441	0	66,165	110
carbon steel	6,616,305	165	662	0	0	13,233	6,616	0	0	2,382	0
concrete	19,313,081	4	39	38,626	2	1,931	386	0	0	579	2
IDDS concentrate	821,160	0	0	16,423	0	0	0	0	0	0	0
vermiculite dryer concentrate	129,600	0	0	220	0	0	0	0	0	0	0
plastic	511,205	23	0	0	15	20	0	5	5	0	0
spent ion exchange resins	117,525	5	0	0	4	5	0	1	1	0	0
ash	46,890	0	2	0	1	1	170	125	0	26	0
nickel alloys	162,720	0	0	0	0	27,662	814	0	0	113,904	0
brass	3,435	0	0	0	0	0	3,435	0	0	0	0
copper	71,290	0	0	0	0	0	71,290	0	0	0	0
lead	1,720	0	0	0	0	0	0	1,720	0	0	0
total		203	757	55,270	32	153,128	83,262	2,292	6	183,056	112

7.3.3.4 Engineered barriers

All the engineered barriers are described in more detail in Section 6 of this draft safety analysis report. Because some of the safety analysis dates back to 2011, when the conceptual design [5] was not yet available, it was produced on the basis of the preliminary design. [69] The differences in the project taken into account in the production of the safety analysis and later optimised (the solutions are presented in Section 6 of this draft safety analysis report) are presented below. The differences are minor with regard to their impact on the results of the safety analysis, and will be taken into account in the next phase of the preparation of the safety report.The disposal unit for the disposal of radioactive waste is a silo with an interior diameter of 27.3 m, while the silo bed (lower elevation, where waste disposal begins) is at a depth of 55 m below the surface. The secondary liner is 1 m thick. Waste is packed into drums and cemented, and loaded into (N3a) concrete containers (FPs), into which nine TTCs or 27,200 l or 320 drums may be loaded. The basic data about the N3a container that was used in the safety analysis calculations is presented in the table below.

Table 7.26: Basic data about N3a container

external dimensions (d x b x h) (m)	2.55 x 2.55 x 3.25
internal dimensions (d x b x h) (m)	2.05 x 2.05 x 2.85
wall thickness (m)	0.2 (top) to 0.23 (at bottom)
bottom thickness (m)	0.2
dimensions of cover (d x b x h) (m)	2.25 x 2.25 x 0.2
total volume (m^3)	21.13
net volume (m^3)	12.81

The containers are to be disposed of in ten layers of 70 containers, with a gap of 10 cm. The voids are to be filled with drainage backfill. The containers are to be disposed of to a height of 33 m, and the silo is then to be sealed with a concrete slab 1.2 m thick. A layer of clay 5 m thick is to be installed on the slab, and covered with sandy gravel similar to natural material as far as the surface. After sealing, saturation is expected to occur (the silo is naturally or artificially saturated). The differences taken into account in the preparation of the safety analysis are presented comparatively in the table below, where remarks are also added, and are taken from the safety analysis report. [70]

Table 7.27: Comparison of optimisations taken into account in safety analysis and in conceptual design

solution used in safety analysis of preliminary design [69]	solution from conceptual design [5]	remarks (impact on long-term safety)
Use of N3a concrete container (for description, see Table 7.26)	Use of N2bV container (for description, see Table 6.1)	Essentially, the containers differ only in size, in that the N3a container can take 3 x 3 TTCs, while the N2b takes 2 x 2 TTCs. In the study of the relevance of design optimisations, [70] this change was evaluated as neutral from the perspective of operational safety and post-closure safety. The sole difference will be in the new model of the nearfield of the repository, which will take account of the different dimensions of the container. Given the smaller container, there will be more concrete barriers.
10 cm intermediate space between FPs	20 cm intermediate space between FPs	More concrete will be used: extra safety owing to additional barriers.
Drainage material is used for backfilling intermediate spaces	Low-permeability material is used for backfilling intermediate spaces	In the study of the relevance of design optimisations, [70] this change was evaluated as neutral from the perspective of operational safety and post-closure safety, as the backfill does not contribute to safety during the operation of the repository, and as a positive from the perspective of safety post closure: owing to the lower permeability of the backfill material, the rate of potential contamination in the nearfield of the repository is lower, and the estimate dose on the environment and on people is consequently lower.
Clay top 5 m thick above sealed silo, covered with sandy gravel to top of silo	Clay top virtually to surface	In the study of the relevance of design optimisations, [70] this change was evaluated as neutral from the perspective of operational safety and post-closure safety. It is envisaged that the thicker clay top and the reduced permeability will lead to a reduced impact on people and the environment.

The primary mechanism by which radionuclides can enter the environment from the repository is water movement, for which reason the development of an adequate conceptual model requires knowledge of the basic geometry of individual hydrogeological units, their hydrological parameters and the gradients that result in groundwater movements.

The hydrogeological units and their properties are described in detail in Section 4 of this draft safety analysis report. In essence there is an open Quaternary aquifer 6 to 9 m thick, under which lies a Miocene aquiclude, which is the layer in which the disposal facility will be positioned.

In the safety analysis it was assumed that the geometry of the entire system would remain unchanged in the post-closure period within the framework of normal evolution. Given the age of the Miocene strata (23.0 to 5.3 million years ago), it can be anticipated with great certainty that these layers will remain undisturbed in the future. There could potentially be a change in the level of the groundwater, which is covered by an alternate evolution scenario. As far as the physical parameters (permeability, porosity) are concerned, it is not anticipated that changes will occur in the future, and it is also not anticipated that there will be any processes that could result in changes.

The parameters used for modelling individual systems (nearfield, farfield, biosphere) are presented below in the description of individual sub-systems.

7.3.3.4.1 Safety functions

The safety functions for the LILW repository are presented in Section 5.2.8 of this draft safety analysis report. Their use and interpretation within the framework of the safety analysis and safety assessments are presented below.

In the consideration of engineered barriers in the safety analysis, the following basic safety functions were taken into account:

- **P (physical containment)**: prevention of the migration of radionuclides by means of physical barriers,
- **C (chemical containment)**: prevention of the migration of radionuclides by means of chemical barriers and by using sorption and solubility limits,
- **H (hydrological)**: natural and man-made barriers that reduce the flow of groundwater through the repository,
- **I (intrusion)**: natural and man-made barriers that reduce the likelihood or impact of human intrusion into the repository,
- **S (structural stability)**: the use of primarily concrete barriers that ensure the structure/geometry of the repository.

Table 7.28 summarises how individual safety functions are used, interpreted and classified in greater detail within the framework of the safety analysis. The table makes evident that:

- the safety function of physical containment is used twice, one of which depends on the design of the repository,
- the safety function of chemical containment is used four times, and is a design function in all cases,
- the hydrological safety function is used three times, and is a consequence of design in two cases,
- the safety function of intruder prevention is used three times, two of which are as a result of the design of the repository, and
- the safety function of structural stability is used twice, and is a consequence of design in both cases.

It is evident from the above that more than ten support operational safety functions proceed from the design of the repository, and accordingly the IAEA's recommendation on the use of multiple safety functions, [71] which replaces the older concept of multiple barriers, has been met. The requirement to use multiple different types (physical and chemical properties) has also been met.

Table 7.28: Support operational safety functions in the form of engineered barriers

code	description of barrier	duration of barrier
P1	physical containment of radionuclides by steel containers (drums)	assessed on the basis of corrosion
P2	physical containment by concrete containers (FPs)	assessed and provided in description of models
P3	physical containment by silo	assessed and provided in description of models
C1	chemical containment by sorption inside conditioned waste (concrete, vermiculite, etc.)	assessed and provided in description of models
C2	chemical containment by sorption inside FP	assessed and provided in description of models
C3	chemical containment by sorption inside silo	assessed and provided in description of models
C4	chemical containment by high pH water in vicinity of silo and sorption of radionuclides in concrete	assessed and provided in description of models
C5	chemical containment by sorption of radionuclides in clay top above sealed silo	assessed and provided in description of models
C6	chemical containment by sorption in Miocene sediments	assessed and provided in description of models
H1	reduction in flow of water from the geosphere through the repository by concrete barriers (vertical flow is conservatively envisaged)	assessed and provided in description of models
H2	reduction in flow of water from the geosphere through the repository by the clay top (vertical flow is conservatively envisaged)	assessed and provided in description of models
H3	low flow through the Miocene strata (and consequently through disposed waste) in combination with greater flow in the Quaternary aquifer	assessed and provided in description of models
I1	reduced probability of inadvertent human intrusion owing to physical (concrete) barriers	assessed and provided in description of models
I2	inadvertent human intrusion into repository prevented by institutional controls	300 years
I3	reduced probability of inadvertent human intrusion for the purpose of seeking drinking water owing to the contrast in permeability between the Miocene and Quaternary strata	in the safety analysis it was assumed that inadvertent intrusion would not occur until 300 years after the closure of the repository; the probability of such an event is extremely low
S1	structural/seismic resilience of monolithically disposed FPs and backfilled voids	assessment from concrete degradation model
S2	structural/seismic resilience of silo	assessment from concrete degradation model

7.3.4 SCENARIO DEVELOPMENT AND EVALUATION

The development of the post-closure scenarios for the LILW repository is taken from the scenario development report [72] within the framework of the safety analysis.

The scenarios within the framework of the safety analysis represent the potential states of the entire disposal system in the future, and as such contain the uncertainties included in the future states of the system. In the past the scenario development procedure for radioactive waste repositories was developed on the basis of international best practice. This consists of four parts:

- identification of a detailed list of features, events and processes (FEPs),
- screening of the detailed list of FEPs until an appropriate selection is obtained,
- description of the relations between the FEPs,
- development of scenarios and computing models.

FEPs represent a sound basis for upgrading the safety analysis and safety assessments in each lifetime of the repository, or phase of the preparation and upgrade of the safety analysis. It can happen that new information results in the identification of new FEPs in the scenario development procedure, and new scenarios and models on this basis.

A start was previously made on developing lists of FEPs, which were taken up by the IAEA within the framework of the ISAM project [50], and it then prepared a detailed list of FEPs for surface radioactive waste repositories.

The techniques for screening FEPs can be divided into three groups:

- those based on probability,
- those based on consequences, and
- those based on expert judgment.

A method based on expert judgment was selected as the most suitable for the Vrbina LILW repository. [72] The FEPs were selected to best represent the potential evolution of the site and the repository, which is described by the scenarios. Once the FEPs have been selected, they need to be combined into scenarios. Various methods have been developed to this end (use of lists, tree structures, diagrams, matrices). The scenarios for the LILW repository were identified on the basis of expert judgment, and were developed into conceptual models. To ensure the traceability and transparency of the process, a special FEP database was developed. It is accessible to all those involved in the project, and serves as the basis for selecting FEPs and developing scenarios. The database is presented in detail in Annexes B and C to the report on scenario development within the framework of the safety analysis. [72]

On the basis of the internationally recognised ISAM FEP database (the international database) [50] and the database from the SAFE project (the experience from the Swedish repository), a database was prepared as the first step in screening the FEPs to allow various users, primarily the experts drawing up the safety analysis at this stage, to examine individual FEPs and to include them in or exclude them from further analysis. The exclusion of FEPs is based on four categories of exclusion. They are:

- FEPs that are evidently not relevant to the safety assessment. An example of these FEPs for the LILW repository consists of FEPs related to releases into a marine environment.

- FEPs that are not relevant because of the chosen content of the safety assessment. This category includes FEPs that are related to a certain collective estimated dose, as it is primarily individual doses that are estimated in this case. In principle these FEPs may become relevant in the following phases, if, of course, the content of the safety assessment changes.
- FEPs that are assessed as immaterial. Immateriality is assessed on the basis of the selection of the disposal concept, or because other FEPs incorporate the consequences in the sense of the behaviour of the entire disposal system. An example of an FEP of this type is FEP 1.4.04 mining and other underground activities. Screening an FEP of this type requires greater judgment, and is therefore subject to greater attention.
- FEPs that are not taken into account because there is no information about them, and information about them cannot reasonably be expected to be available in the future. An example of an FEP of this type is FEP 2.1.10 biological and biochemical processes and conditions in the nearfield of the repository. Screening an FEP of this type requires the highest expert judgment, and is therefore subject to the greatest attention.

The individual FEPs included in or excluded from scenario development are described in Annex A to the report on scenario development within the framework of the safety analysis. [72] A total of 969 FEPs were analysed, of which 86 were included in scenarios, 75 were not taken into account, and 808 were identified as irrelevant.

The scenarios for the LILW repository were developed by taking into account the most important consequences, both in the period of normal evolution and in the period of alternate evolution. The expectation is that even more detailed analysis of the FEPs and scenario development in the future will determine only minor deviations from the scenarios developed in this phase. In the development of scenarios for the LILW repository, the following scenarios were identified:

- Normal evolution scenario:
    - Nominal scenario,
        - variant of nominal scenario with alternate degradation of engineered barriers,
        - variant of nominal scenario without well,
        - variant of nominal scenario with conservative assumption of use of well for drawing water,
- Alternate evolution scenarios:
    - scenario of early failure of engineered barriers (failure of all man-made barriers: waste packages, FPs, backfill, silo, clay top),
    - scenario of early failure of concrete barriers,
    - scenario of river meandering and surface erosion,
    - scenario of change to hydrological conditions.

The assumptions of the individual scenarios are illustrated in the table below, and are presented in greater detail within the framework of the description of the models for the individual scenarios.

Table 7.29: Overview of assumptions for individual scenarios addressed in the safety analysis

Scenario	Nearfield of repository (Section 7.3.5.2 of this draft safety analysis report)	Farfield of repository (Section 7.3.5.3 of this draft safety analysis report)	Biosphere (Section 7.3.5.4 of this draft safety analysis report)
Nominal scenario	Simultaneous onset of degradation of engineered barriers (see model in Section 7.3.5.1.2 of this draft safety analysis report)	Transport of potential contamination to the aquifer that drains into the Sava, and a well is sunk 100 m from the repository in the direction of the contamination	100% of all drinking water (public) comes from the well. 100% of all consumed fish are from the Sava. The irrigation of arable crops with water from the river is envisaged, but irrigation with water from the well is not. Livestock drink water from the river. All arable crops, meat and milk is from areas that are irrigated. People spend 100% of their time in the area
Variant of nominal scenario with an alternate degradation of engineered barriers	Parallel degradation of engineered barriers (see model in Section 7.3.5.1.1 of this draft safety analysis report)	Same as in nominal scenario	Same as in nominal scenario
Variant of nominal scenario without well	Same as in nominal scenario	Transport of potential contamination to the aquifer that drains into the Sava, without a well that draws water from the aquifer	Same as in nominal scenario, except that 100% of people's drinking water comes from the river
Variant of nominal scenario with conservative assumption of use of well for drawing water	Same as in nominal scenario	Same as in nominal scenario	Same as in nominal scenario, except that people get 100% of their drinking water from the well, and 100% of their edible fish are caught in the Sava. The well is used for two additional functions: irrigating vegetables

Scenario	Nearfield of repository (Section 7.3.5.2 of this draft safety analysis report)	Farfield of repository (Section 7.3.5.3 of this draft safety analysis report)	Biosphere (Section 7.3.5.4 of this draft safety analysis report)
			and providing water for livestock. In the case of irrigation from the well, the assumption is that the most exposed individual spends 500 hours a year in the irrigation zone
Early failure of engineered barriers	It is assumed that all components of engineered barriers are subject to rapid decay at the end of the institutional controls (not merely the disintegration of the concrete, but also faster corrosion and other degradation processes; the calculations take account of permeability and the radionuclide leaching rate). The scenario could also be named the scenario of no engineering barriers	Same as in nominal scenario	Same as in nominal scenario
Early failure of concrete barriers	It is assumed that solely the concrete components are subject to rapid decay at the end of the institutional controls. The calculations take account of high permeability	Same as in nominal scenario	Same as in nominal scenario
River meandering and surface erosion	Same as in nominal scenario	It is assumed that the farfield model is not taken into account in the joint model, which indicates that erosion removes all natural material around the repository. The potential releases are direct into the Sava. There is no well	Same as in nominal scenario, except that 100% of drinking water is taken from the river
Change to hydrological conditions	Same as in nominal scenario, except that the gradient in the nearfield vicinity model is modified (the water flow rates around and through the repository are different)	Same as in nominal scenario	Same as in nominal scenario

Scenario	Nearfield of repository (Section 7.3.5.2 of this draft safety analysis report)	Farfield of repository (Section 7.3.5.3 of this draft safety analysis report)	Biosphere (Section 7.3.5.4 of this draft safety analysis report)
Inadvertent human intrusion	The depth of the silo means that the only possible scenario of intrusion is drilling. The probability of such a scenario is very low, as there are very few reasons to drill to the depth of the silo. In drilling, very small quantities of waste would be excavated, during which the drillers would be irradiated. It is envisaged that after drilling the excavated material would be distributed over an area 29 m in diameter, and that the thickness of the contaminated zone would be 15 cm. After intrusion, the area is left without controls, and a family moves into the area and builds a farm at the site. Farming activities spread the contamination to cover 2,500 m^2. Family members are exposed to contaminated food that they ingest, contaminated air that they inhale, and external irradiation from radionuclides in the soil		

The table below illustrates which FEPs were included in the development of individual scenarios.

Table 7.30: FEPs from which scenarios were developed

Scenario	FEPs from which scenarios were developed (scenario number from international list of FEPs is cited in brackets) [73]
Early failure of engineered barriers	Seismicity (1.2.03) Waste form materials and characteristics, and degradation processes (2.1.02) Container materials and characteristics, and degradation processes and errors (2.1.03) Buffer/backfill materials and characteristics, and degradation processes (2.1.04) Engineered barrier system characteristics, and degradation processes (2.1.05) Hydraulic/hydrogeological processes and conditions (in wastes and engineered barriers) (2.1.08) Gas sources and effects (in wastes and engineered barriers) (2.1.12)
River meandering and surface erosion	Erosion and sedimentation (1.2.07) Hydrological/hydrogeological response to geological changes (1.2.10)
Inadvertent human intrusion	Drilling activities (human intrusion) (1.4.03)
Change to hydrological conditions	Erosion and sedimentation (1.2.07) Hydrological/hydrogeological response to geological changes (1.2.10) Climate change, global (1.3.01) Climate change, regional and local (1.3.02) Hydrological/hydrogeological response to climate changes (1.3.07) Human influences on climate (1.4.01) Water management (wells, reservoirs, dams) (1.4.10)

7.3.5 FORMULATION AND IMPLEMENTATION OF MODELS

The formulation and implementation of models is described in detail in the report on models, [74] which the following section merely summarises.

A modelling approach is required for the normal evolution scenario, and for the alternate evolution scenarios, mainly so that the potential doses received because of the repository in the future can be compared with the legal limits, and to better understand the evolution of uncertainty in the future. The modelling approach was divided into two parts, each of which contributes to the aforementioned objectives. The first part is an aid to understanding how the system behaves in detail, when expert knowledge of the physical and chemical processes and the functions of the disposal system are considered. This is achieved with the help of detailed process models for individual areas of the disposal system (nearfield, farfield, biosphere). The second part needs to provide an complete overview of the system of uncertainty under given boundary and initial conditions. To this end it requires system models that are more effective numerically, but still describe the knowledge of the complete system to a sufficient degree. The relationship between the two parts is described in Figure 7.22 below. Detailed process models provide for an understanding of individual elements of the system. The specific output data from the process models is then appropriately collated and used as input data for the system model.

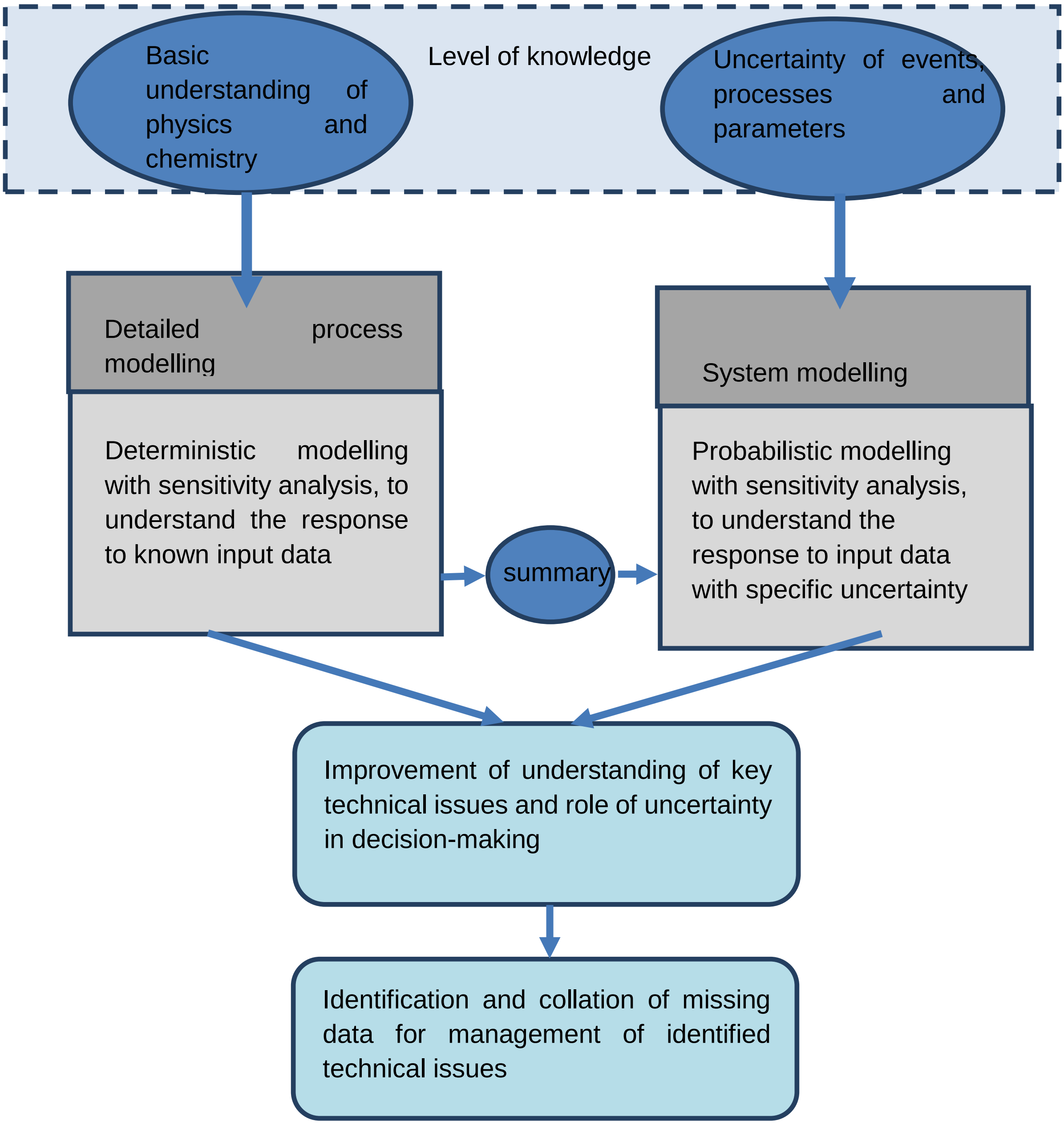

Figure 7.22: Combined use of data from detailed process models and system model [74]

The approach described above was also used for the LILW repository, and the following were used within the framework of the process models:

- various possible models of the degradation of engineered barriers (Section 7.3.5.1 of this draft safety analysis report),
- Hydrus software, for modelling the flow through the silo for various levels of degradation of engineered barriers,
- FEFLOW software for modelling the flow in the nearfield of the repository,
- Ecolego software for modelling the biosphere.

The results of the process models were taken and appropriately implemented as input data for the system model, which was installed in an Ecolego software environment.

A description of the individual software tools and the proof of their correct use in the preparation of the safety analysis for facilities such as the LILW repository (software validation) is given in a separate report. [45]

7.3.5.1 Model of degradation of engineered barriers

The evolution of the degradation of engineered barriers is presented in detail in the report [75] drawn up within the framework of the safety analysis. The report is summarised below.

When speaking of engineered barriers (primarily concrete and similar materials), the basic conceptual model is that the concrete is initially undamaged and of low permeability to water, as planned in the project. The hydraulic conductivity of the silo, the FPs and the backfill material increases over time, as a result of various degradation mechanisms. This means that the planned permeability taken initially increases over time because of degradation, and after a certain time reaches the permeability of gravel or aggregate from the perspective of water flow. During this time (while it is degrading), the concrete retains certain chemical properties (notably sorption, high pH), and still retards the flow of radionuclides from the repository.

The Vrbina LILW repository envisages underground disposal facilities, which after closure will be in a saturated zone, below the level of the groundwater, which from a chemical perspective means that the engineered barriers and waste will be in low-acidity conditions for the long term.

The assumed disintegration of the concrete was extremely conservative. An alternate evolution scenario was taken as the worst case of failure.

The concrete used for the engineered barriers, most notably the containers and the silo, will be very high in quality and also robust. The model that was developed and applied to determine the degradation of the concrete is also used by other countries for safety analysis for LILW repositories, and is based on the degradation processes that can occur in concrete. [75] In the case of the Vrbina LILW repository, concrete will be used as part of the underground facility, and will be in saturated state after closure, which is ideal conditions for concrete. Belgium, for example, is planning a surface repository, where the failure of the engineered barriers is envisaged after 800 years.

Because the resilience of concrete (except for the standard freeze-thaw tests) is hard to prove, a scenario of early failure of engineered barriers was also developed alongside the models.

The use of concrete at the LILW repository is not solely for the physical containment of radionuclides. Another key property of concrete is that its high pH means that it has good sorption of the majority of radionuclides. This property of concrete (high pH) will be retained despite any degradation, and is one of the key safety functions of the LILW repository.

In later phases the concrete mixes that will be used for the LILW repository will be properly tested and investigated. In this phase of the project, the parameters for the concrete used to produce the safety analysis were conservative in the opinion of the producers of the safety analysis and the designer.

A programme of research, development, modelling, testing and monitoring to understand the evolution of the repository will be drawn up and carried out in the next phases of the project.

The basic initial properties of the concrete, as taken from the preliminary design, [69] are given in the table below.

Table 7.31: Basic initial properties of concrete, taken from preliminary design [69]

Parameter	Value	Reference
Strength class	C25/30	SIST EN 1992
Exposure class	XC4, XA3	SIST EN 1992, SIST EN 206-1 and SIST 1026
Resistance to water penetration	PV-III	SIST EN 206-1 and SIST 1026
Freeze-thaw resistance	Min 100 cycles	SIST 1026
Chloride content	CL 0,10	SIST EN 206-1
Resistance to abrasion	Min XB1	SIST EN 206-1 and SIST 1026
Thickness of top layer	Min 40 mm	SIST EN 1992
Cracking limit	Max 0.1 mm	76/76
W/C ratio	< 0.45	[76][76]
Aggregate D max	64 mm	
Reinforcement	S 500 C	SIST EN 1992, Appendix C

The parameters of the initial (undamaged) concrete and the degraded concrete were determined on the basis of the initial properties. [75] These parameters are as follows for the initial concrete:

- porosity n_0 = 0.33
- effective diffusion coefficient for non-sorptive solutions D_{eff0} = 1.9 E-12 m^2/s
- initial hydraulic conductivity K_0 = 1 E-09 m/s

The concrete was assumed to have similar properties to sand when it reached its final state of degradation. They are as follows:

- porosity n_d = 0.3
- effective diffusion coefficient for non-sorptive solutions D_{effd} = 1 E-09 m^2/s
- initial hydraulic conductivity K_{0d} = 1 E-04 m/s

The following degradation mechanisms were examined within the framework of the safety analysis:

Sulphate-magnesium attack

- Sulphate-magnesium attack is caused by aqueous sulphates and magnesium that come into contact with concrete. Chemical reactions occur where the sulphate and magnesium penetrate the surface of the concrete and weaken it. There is corrosion of the surface of the concrete, when the surface loses its strength and is displaced, making new surface available for attack. This reduces the thickness of the concrete over time. On the basis of the model presented in the report on the evolution of the system of engineered barriers, [75] the rate of degradation of concrete from sulphate-magnesium attack was estimated at 8.7 E-02 mm/year, which means that it would take approximately 11,000 years for concrete 1 m thick to disintegrate on account of this degradation mechanism. As described above, during the degradation the concrete spalls, but remains intact below the surface, and retains its overall permeability and diffusivity as long as the thickness of the concrete itself is not significantly reduced. The safety analysis therefore assesses that 10,000 years of sulphate-magnesium attack will not have a significant impact on the safety functions performed by the disposal silo.

Calcium hydroxide leaching

- When groundwater penetrates the concrete, leaching can occur, which can have a significant impact on the properties of the concrete over the long term. Leaching of water-soluble components, such as calcium hydroxide, can occur in this process. During leaching of this type, the concrete loses compressive strength, while its porosity and thus permeability increase. The leaching model used in the safety analysis was a shrinking core model, [77] which assumes that leaching and removal of calcium from the interior of the concrete is faster that the transport of calcium through the concrete. The depth of leaching in the concrete is estimated on the basis of Equation 7.5:

$$X = \left(2D \frac{C_i - C_{gw}}{C_s} t \right)^{1/2},$$

Equation 7.5: Depth of leaching in concrete [78]

Where: [79]

X .... depth

D .... coefficient of diffusion of Ca^{2+} ions in concrete

C_i .... concentration of Ca^{2+} ions in pores

C_{gw} .... concentration of Ca^{2+} ions in groundwater

C_s .... concentration of Ca^{2+} ions in solid concrete

Within the framework of the safety analysis [75] it was assessed that this process in the case of the LILW repository is relatively slow, as it would take 10,000 years for the leaching to reach a depth of 2 cm in the concrete.

Alkali-silica reaction

- This reaction is highly complex, and leads to swelling and stretching, which destroys the structure of the concrete and brings an increase in permeability. This can lead to structural failure because of other degradation processes (such as sulphate-magnesium attack). All aggregates used in the production of concrete react chemically with the cement paste to a certain extent. The chemical processes in the alkali-silica reaction are well-known. [80] The main components of the reaction are hydroxide alkali salts in the cement paste and potentially reactive silica in the aggregate. When reaction occurs because of swelling, the concrete cracks. In the safety analysis [75] it was assumed that best practice in the use of concretes (particularly in the choice of aggregate) would be used in the construction of the repository, and that the consequences of alkali-silica reactions in the degradation of the concrete would be negligible.

Carbonation

- Carbonation is a process in which there is a reduction in the pH of pore water owing to the conversion of calcium hydroxide into calcium carbonate caused by reaction with carbon dioxide. The reaction is accompanied by a change in the microstructure of the concrete. Carbonation is more common when the concrete is in a water-saturated environment, and peaks at 50% relative humidity, before slowing. Carbon dioxide must also be present for the reaction to occur. Most carbonation will thus occur during the operation of the silo. The safety analysis [75] finds that carbonation could occur to a depth of 7 to 8 mm during operation, which is little compared with the thickness of the silo wall. The properties of the groundwater at the repository site are given in Section 4 of this draft safety analysis report.
- After filling and sealing, the silo will be completely saturated with water, and carbonation will depend on the concentration of carbon dioxide in the vicinity. Carbonation will therefore be slow, but will contribute to additional cracking of the concrete. The reduction of the local pH will give rise to an environment with greater potential for corrosion of the reinforcement, which is described in one of the sections below.

Acid attack

- Acid attack can occur when the concrete is exposed to groundwater or other sources of water with low pH. Acid leaches out soluble components in concrete (calcium hydroxide), and the concrete then loses structural resilience and sees an increase in porosity. Research in the field [81] shows that the pH of the groundwater in the vicinity of the repository is close to neutral, and degradation through acid attack therefore cannot be expected. [75]

Corrosion

Carbon steel

- When corrosion occurs to the reinforcement (carbon steel), the corrosion products have a greater volume than the corroded metal. This causes swelling and cracking of the concrete. It also increases the permeability of the concrete.
- The corrosion rate depends mainly on the pH, and to a lesser extent on the chloride concentration. Because there will be a great deal of concrete in the repository, the pH of the pore water will be relatively high, and only the hydration process described above will reduce it. Chloride concentration in the Miocene water around the repository is also low. The safety analysis [75] assumes a corrosion rate of 10^{-7} m/year. Because corrosion causes swelling and thus damage to the concrete, it is assumed that the concrete degrades completely when the reinforcement is 25% to 50% corroded. This means that full degradation of the concrete owing to corrosion of the reinforcement (thickness 1 cm) will occur after 12,500 to 25,000 years.

Stainless steel

- Some of the waste disposed of at the repository will be made of stainless steel. This corrodes differently to carbon steel. The corrosion products do not occupy a greater volume than the base metal, and there is therefore no swelling or cracking of the concrete. In addition, the corrosion rate is much lower. This means that the corrosion of stainless steel is not a factor in the disintegration of the concrete, and leads more slowly to the elimination of activation products in waste made of stainless steel. The corrosion rate of stainless steel assumed for the purposes of the safety analysis was 0.01 to 2 μm/year. [75]
- Special attention was focused on the corrosion of the reactor vessel (see Annex 1 of the report on the evolution of engineered barriers within the framework of the preparation of the safety analysis [75]), as the release of radionuclides from activated metals depends on it. It is envisaged that the reactor vessel will be cut up before disposal, and loaded into 200 l drums before being packed into FPs. In the report it was estimated that corrosion would lead to a 50% loss in the mass of metal (carbon steel) with activated radionuclides over a period of 8,250 to 165,000 years after the closure of the repository, compared with a period of 165,000 to 165 million years after the closure of the repository for stainless steel. The corrosion rates assumed in this estimate were conservative.

Based on studies of the degradation mechanisms described above, two models for the degradation of engineered barriers were developed within the framework of the safety analysis. [75] In the first (sequential degradation of engineered barriers), it was assumed that the concrete barriers (silo, FPs, filling in KPs) begin to fail from the outside in, once the groundwater from the Miocene strata has reached them. When the failure of the silo occurs, water can penetrate through to the FPs, and when they fail, to the waste. In this model internal barriers remain intact until the external barriers fail.

In the second model (simultaneous degradation of engineered barriers), it is assumed that all the barriers begin degrading immediately after the closure of the repository at the same rates. This model is more conservative than the first, is easier to analyse, and most likely better represents degradation in anaerobic conditions.

7.3.5.1.1 Sequential degradation of engineered barriers

The sequential degradation of engineered barriers model is illustrated schematically in Table 7.32 below, and is described in detail in the safety analysis report. [75] The time periods were estimated on the basis of the mechanism of carbonation in concrete. After the sealing of the silo, there is no longer any chance of an uplift force acting, and therefore the silo will not be put in a tensile state. The silo will be subject to compression loading only. It is assumed that the barriers begin to degrade after the closure of the repository. The operating and standby phases were not considered. It is assumed that during operation and the standby phase there is monitoring of the properties of the silo and, in the event that the evolution of events does not follow the envisaged path and there is degradation of the engineered barriers, appropriate action is taken (remedial measures, etc.). That such errors could occur, or be identified more than 10 years after construction, is unlikely. For this reason it is necessary to plan construction extremely carefully, and then to monitor it closely. Should this nevertheless occur, there is potential for injection, for example from beneath.

Table 7.32: Degradation of engineered barriers assumed in sequential degradation of engineered barriers

Time period [years]	Silo	FPs	Waste conditioned for disposal
t < 220	undamaged	undamaged	undamaged
220 < t < 660	gradual degradation until t = 12,700	gradual degradation until t = 1,040	undamaged
660 < t < 1,040	gradual degradation until t = 12,700	gradual degradation until t = 1,040	undamaged
1,040 < t < 6,000	gradual degradation until t = 12,700	complete degradation	undamaged
6,000 < t < 6,650	gradual degradation until t = 12,700	complete degradation	gradual degradation until t = 6,650
t > 6,650	gradual degradation until t = 12,700	complete degradation	complete degradation

7.3.5.1.2 Simultaneous degradation of engineered barriers

In this case corrosion of the reinforcement causes degradation of the concrete barriers immediately after the closure of the LILW repository. The model assumes a corrosion rate of 10^{-7} m/year, in line with the high pH and anaerobic conditions that will be present in the repository after closure. The model does not take account of carbonation, primarily because:

- in the Miocene water around the repository, the concentration of carbon dioxide is low,
- carbonation causes pore clogging, and generates calcite-brucite layers, which further improve the structural properties of the concrete.

The complete degradation of the concrete occurs 12,500 to 25,000 years after the closure of the repository. This means that the safety analysis assumed that the properties of the concrete gradually (on a liner basis) evolve from the initial design values to the final values in place when the concrete has completely degraded and has similar properties to sand.

7.3.5.2 Nearfield model

For the draft safety analysis report, for the phase of obtaining the environmental consent, the nearfield model took the concept from the preliminary design phase, which differs from the optimised concept from the construction permit project phase in terms of the backfill between the containers and the thickness of the clay layer above the silo. The differences are illustrated in detail in Table 7.27. The old concept is more conservative in the opinion of the producers of the safety analysis, and represents an upper limit on impact. Analysis is conducted that subsequently will be more realistic and will take account of the latest concept of the repository. The presented analysis assumes the maximum possible gradients, which occur when groundwater is low. It follows that the high water table entails less impact.

The nearfield model of the repository consists of a disposal silo and several metres of natural material around the silo, and was modelled with Hydrus 2D/2D software, which solves the Richards flow equation and the advection equation (dispersion of heat and dissolved substances in a changing saturated underground medium) using a final elements method. Given the assumption that the silo is fully saturated at time zero (immediately after sealing), the Richards equation simplifies to the Darcy equation, which describes groundwater flow. The nearfield model is described in detail in the report on the nearfield model, [82] a summary of which is presented below.

The nearfield model of the repository encompasses the disposal silo itself and several metres around it. The model is illustrated conceptually in Figure 7.23 below.

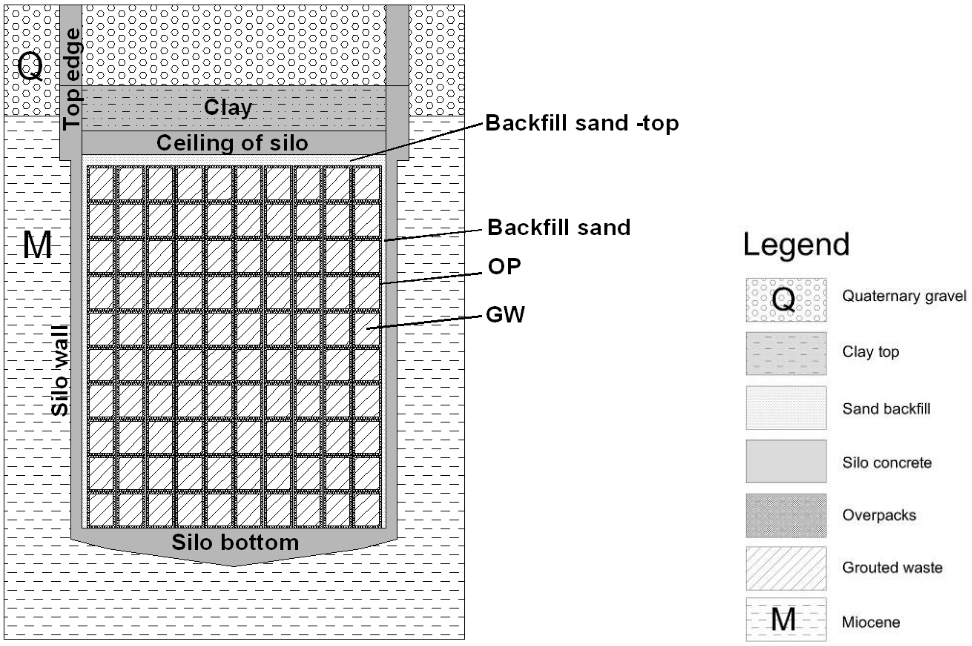

Figure 7.23: Nearfield conceptual model

The nearfield model was run for four phases of degradation of engineered barriers, where the time component (the models for the degradation of engineered barriers are presented in detail in Section 7.3.5.1) is not significant, as time (and the modelled degradation of engineered barriers) is treated as a variable in the system model. The key task of the model was to obtain the flow velocity of substances under various assumptions for the degradation of engineered barriers and other boundary and initial conditions.

The initial state (K1 in Table 7.33) represents the repository in the initial intact state with the material possessing its design characteristics. The second state (K2 in Table 7.33) covers the degradation of the silo, the third state (K3 in Table 7.33) covers the degradation of the FPs, and the fourth state (K4 in Table 7.33) covers the complete degradation of all engineered barriers. The hydraulic conductivities of intact and degraded materials for individual engineered barriers used in the nearfield model are presented in Table 7.33 below.

Table 7.33: Hydraulic conductivity of materials of engineered barriers for various states used in safety analysis [82]

Material	K_1 [m/s]	K_2 [m/s]	K_3 [m/s]	K_4 [m/s]	Reference
Miocene 1 (8 to 31 m)	5.0×10^{-7}				[81]
Miocene 2 (31 to 46 m)	6.4×10^{-8}				[81]
Miocene 3 (46 to 49 m)	6.9×10^{-7}				[81]
Silo concrete	1.0×10^{-9}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	[69]
Sand backfill	1.0×10^{-4}				[estimate]
FPs	1.0×10^{-9}	1.0×10^{-9}	1.0×10^{-4}	1.0×10^{-4}	[69]
Clay top	1.0×10^{-9}				[estimate]
Grouted waste	1.0×10^{-9}	1.0×10^{-9}	1.0×10^{-9}	1.0×10^{-4}	[estimate]

The individual states (first to fourth, from left to right and top down) are also illustrated in Figure 7.24 below, where degraded barriers are denoted in red.

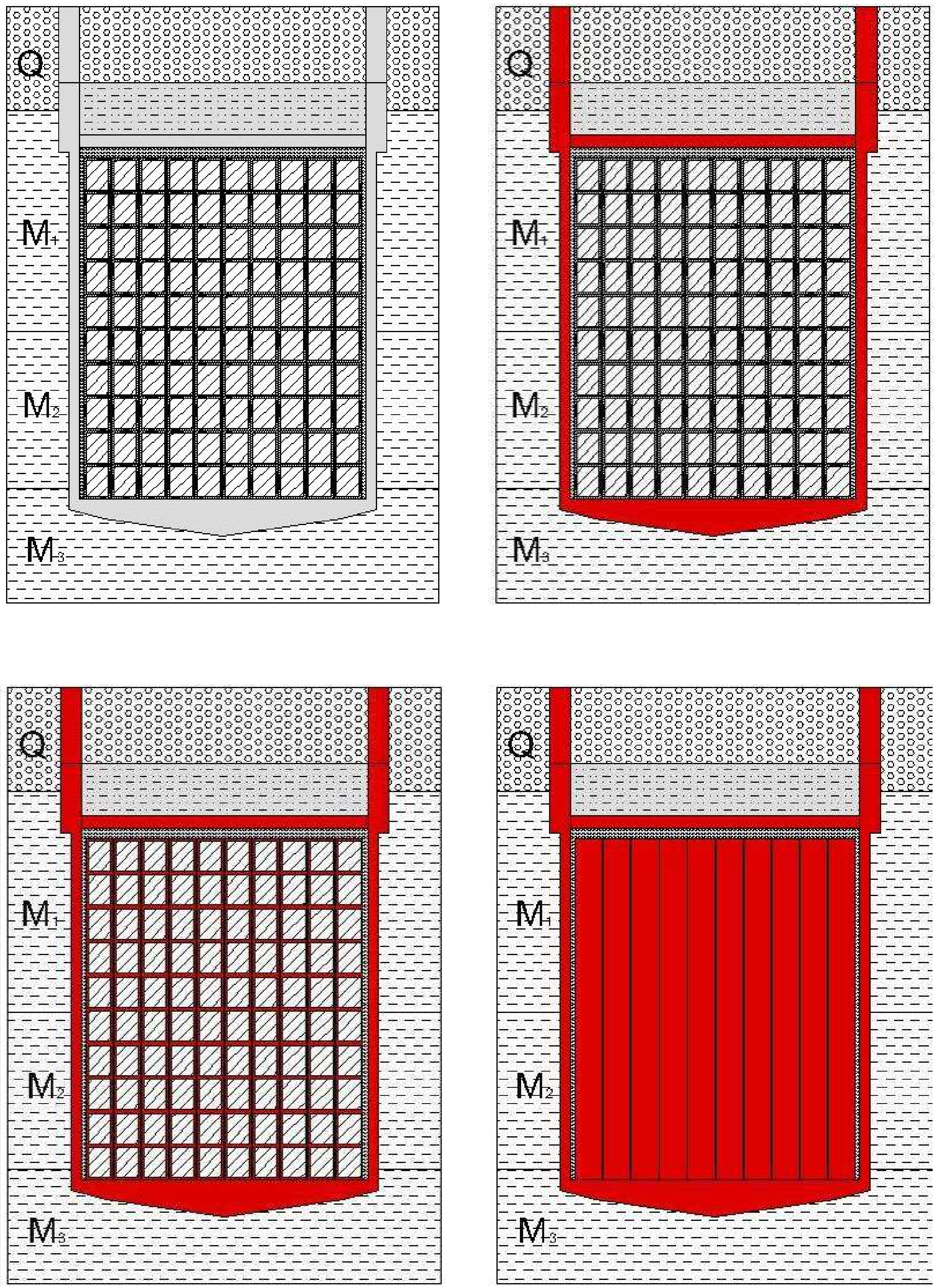

Figure 7.24: Four states of degradation of engineered barriers, degraded barriers denoted in red (Q: Quaternary stratum; M_1, M_2, M_3: Miocene strata)

The nearfield model takes account of a constant pressure head of 50.75 m at the bottom of the model (depth of 57 m) and 3.47 m at the top of the model (depth of 10 m). The boundary condition without flow was taken into account in the boundaries of the model. The height difference of 47 m takes account of a pressure head of 47.28 m, which represents a vertical gradient of 0.006. The calculations are also made for gradients ranging from 0.001 to 0.02. The Quaternary aquifer at the top of the silo was represented in the model with a pressure head at the top limit of the model. The nearfield model thus represents the silo and the site in the immediate vicinity of the silo. It is conservatively assumed that the flow of water through the silo is upward at all times. The boundary conditions are also presented in Figure 7.25 below.

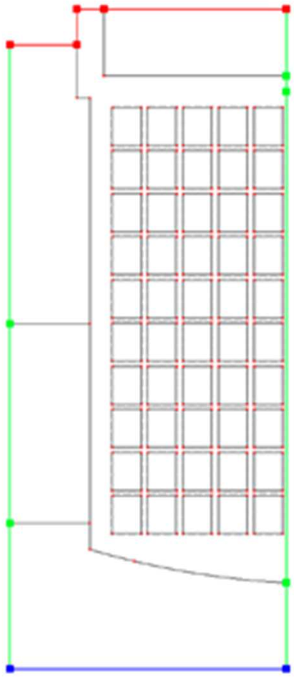

Figure 7.25: Boundary conditions of nearfield model (green line: no flow; blue line: pressure head of 50.75 m; red line: pressure head of 3.47 m)

The basic results of the nearfield model are illustrated in Tables 7.34 and 7.35 below. The first illustrates the water velocity [m/year] through individual engineered barriers, while the second illustrates the water flow rate [m^3/year] through individual engineered barriers of the repository. The results are illustrated for a time when the model reaches a static state, i.e. the data no longer changes over time. The velocity and flow rate are the key pieces of data used in the system model of safety analysis for the LILW repository.

Table 7.34: Darcy velocity through various parts of the repository, as result of the nearfield model

Repository part	Darcy velocity [m/y]			
	State 1	State 2	State 3	State 4
Clay layer	8.17E-05	6.78E-05	7.79E-05	9.68E-05
Top edge of clay layer	8.17E-05	5.59	7.27	7.29
Silo ceiling	8.17E-05	0.738	0.958	0.962
Top edge of backfill	9.41E-05	0.505	0.846	0.962
Silo wall (outflow)	1.05E-04	-	-	-
Silo wall (vertical)	-	2.27	1.70	0.962
Grouted waste	1.95E-04	1.62E-04	3.72E-05	0.962
FPs	1.95E-04	1.62E-04	1.70	0.962
Backfill	1.20E-03	2.27	1.70	0.962
Silo bottom	3.63E-04	0.738	0.958	0.962

Table 7.35: Flow through various parts of the repository, as result of the nearfield model

Repository part	State 1 Flow rate [m^3/y]	State 2 Flow rate [m^3/y]	State 3 Flow rate [m^3/y]	State 4 Flow rate [m^3/y]
Clay layer	0.048	0.040	0.046	0.057
Top edge of clay layer	0.007	497.30	646.07	648.30
Silo ceiling	0.055	497.34	646.12	648.35
Top edge of backfill	0.055	295.53	494.98	562.86
Silo wall (outflow)	0.190	-	-	-
Silo wall (vertical)	-	201.81	151.14	85.49
Grouted waste	0.057	0.048	0.011	282.88
FPs	0.031	0.026	273.70	154.82
Backfill	0.156	295.46	221.27	125.16
Silo bottom	0.245	497.34	646.12	648.35

The results show that the flow rates and Darcy velocities increase as a result of the degradation of individual barriers, until they reach the maximum possible values under the assumption of the degradation of all engineered barriers.

The other results of the nearfield model that were taken into the system model are presented in the report on parameters used. [83]

7.3.5.3 Farfield model

The farfield model for the LILW repository is presented in detail in the safety analysis report on the farfield model [84] and the model verification report. [87] A summary alone is given below.

Within the framework of the safety analysis a farfield model was also developed with the help of FEFLOW software, with the aim of obtaining the Darcy velocities of geological strata in the farfield of the repository. These velocities serve as input data for the system model of safety analysis for the LILW repository. The model encompasses the conceptual model, [88] which relies on data from investigations in the field, [81] as disclosed in Section 4 of this draft safety analysis report. The basic concept of the farfield model is illustrated in Figure 7-26 below.

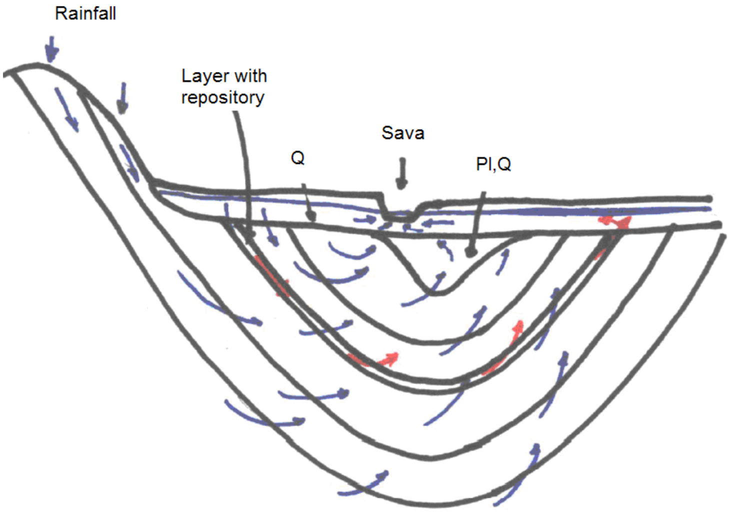

Figure 7-26: Conceptual model of flow in the LILW repository farfield

The farfield model covers the wider zone of the LILW repository, as illustrated in Figure 7.27 below.

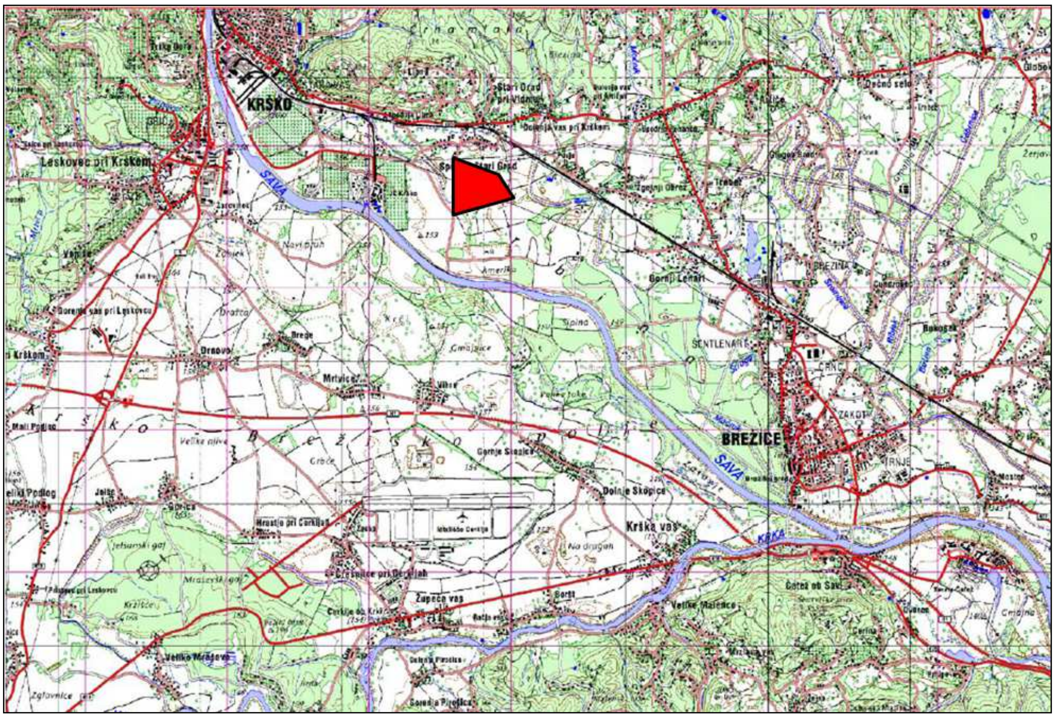

Figure 7.27: Farfield of LILW repository

The model was divided into 25 layers and 260,494 hubs. The model was calibrated for 27 points with groundwater level as at 2 October 2008. The construction of Brežice hydroelectric power plant occurred during the creation of this document, although the final quasi-stationary state has not yet been attained: after filling of the reservoir, groundwater levels are still changing, and rising. The farfield model will be recalibrated in the next phase of safety analysis, although it is assessed that the change will have no impact on the assessment of the impact of the repository on people and the environment. The extreme case of a change in the hydrology of the site is the river meandering scenario, which is presented in Section 7.3.6.4 of this document and assumes direct discharge from the biosphere into the Sava. The farfield model primarily has an impact on the long-term security of the repository, which exceeds the planned lifetime of the constructed hydroelectric plant.

The results of the model are illustrated in Figure 7.28, Figure 7.29 and Figure 7.30, which illustrate the hydraulic head of groundwater in the farfield of the LILW repository, the hydraulic head of groundwater in the nearfield of the LILW repository, and the results of the modelling of particle transport. In this case this is merely a randomly chosen concentration for the purpose of illustrating the direction and decrease of the concentration with distance from the repository site.

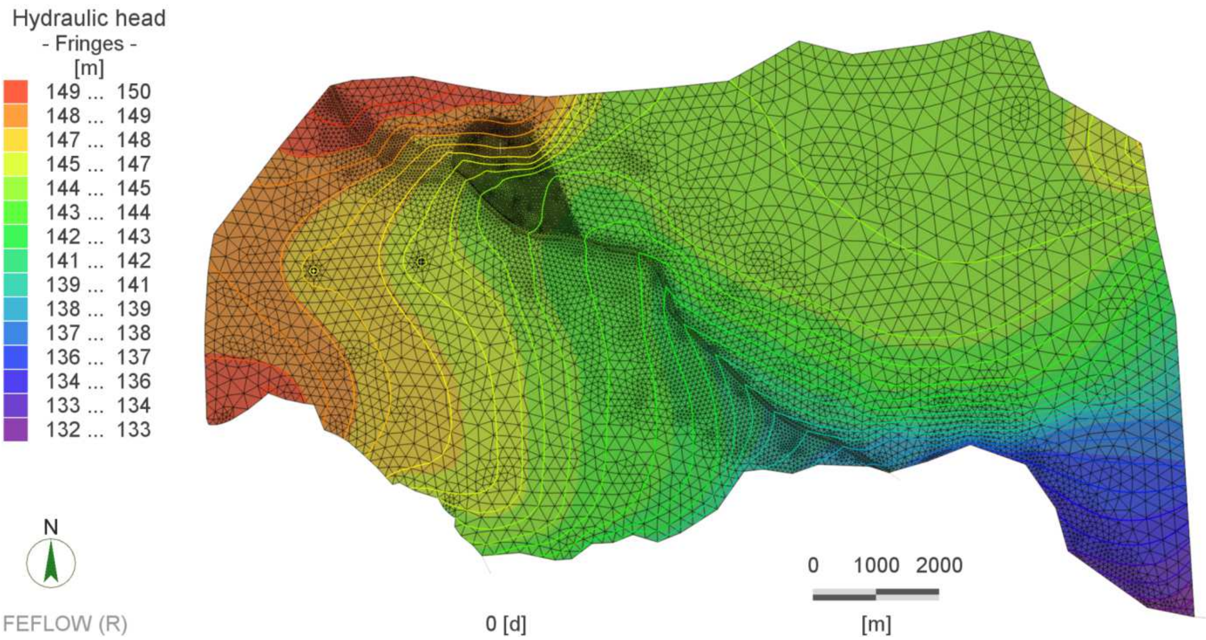

Figure 7.28: Hydraulic head of groundwater in farfield of LILW repository

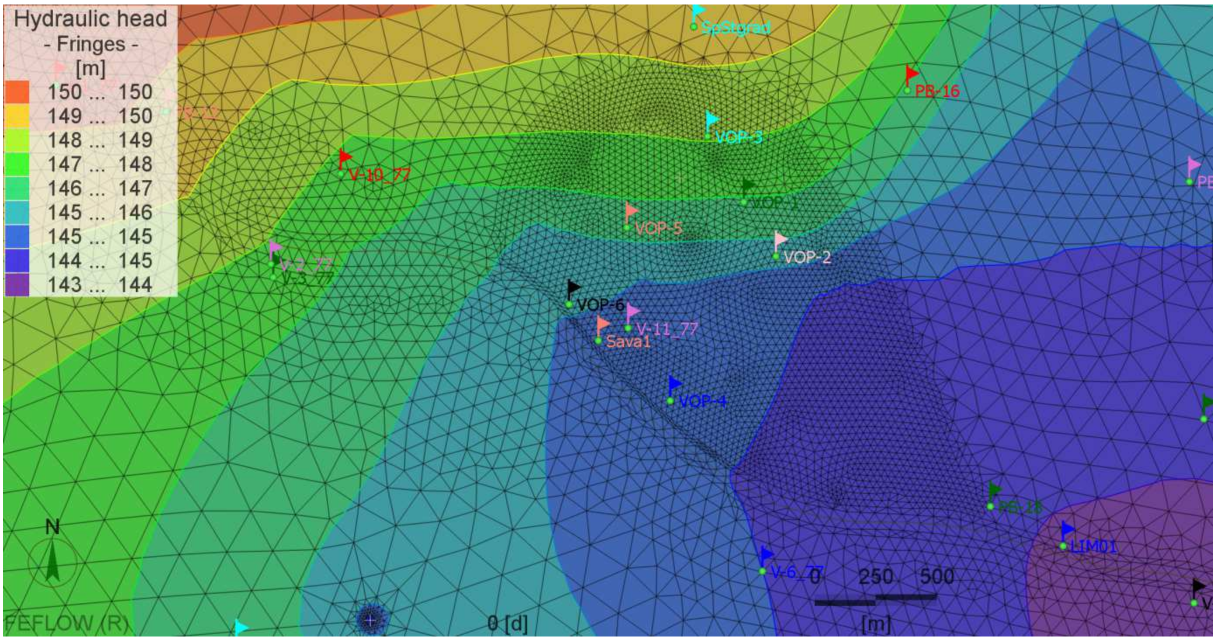

Figure 7.29: Hydraulic head of groundwater in nearfield of LILW repository

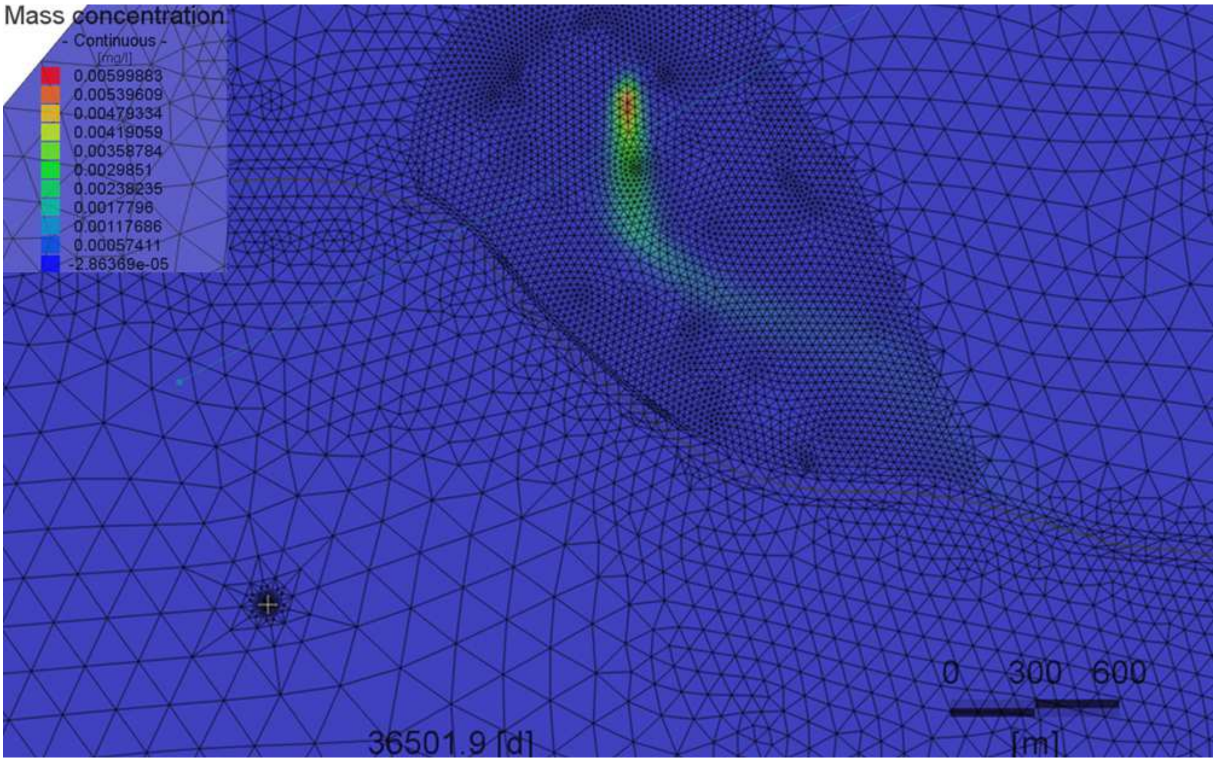

Figure 7.30: Results of particle transport modelling

The results of the farfield model that were transferred into the system model are as follows:

- Darcy velocity in Quaternary strata v = 58.1 m/year
- Hydraulic conductivity in Quaternary strata K = 1.02 E-03 m/s

The other results of the farfield model that were taken into the system model are presented in the report on parameters used. [83]

7.3.5.4 Biosphere model

The biosphere model is described in detail in the report on models [74] within the framework of the safety analysis. A summary of the development of the biosphere model is given below.

The biosphere model assumes that radionuclides can enter the biosphere via two pathways. They are:

- drawing of water from the well located in the vicinity of the repository (in the centre of potential contamination, 100 m from the repository),
- leaching of radionuclides into the river.

The potential exposure pathways for a member of the critical group are as follows:

- drinking water from the well and/or the river,
- eating food from the river (fish),
- eating plant-based food contaminated by irrigation from the well or river, and indirectly contaminated plants and soil,
- drinking milk and eating meat from livestock pastured in areas contaminated by irrigation with water from the well or the river,

- external irradiation and irradiation from inhalation as a result of presence in contaminated fields and pastures.

Water for irrigation may be drawn from the well or the river, and is used in fields and pasture.

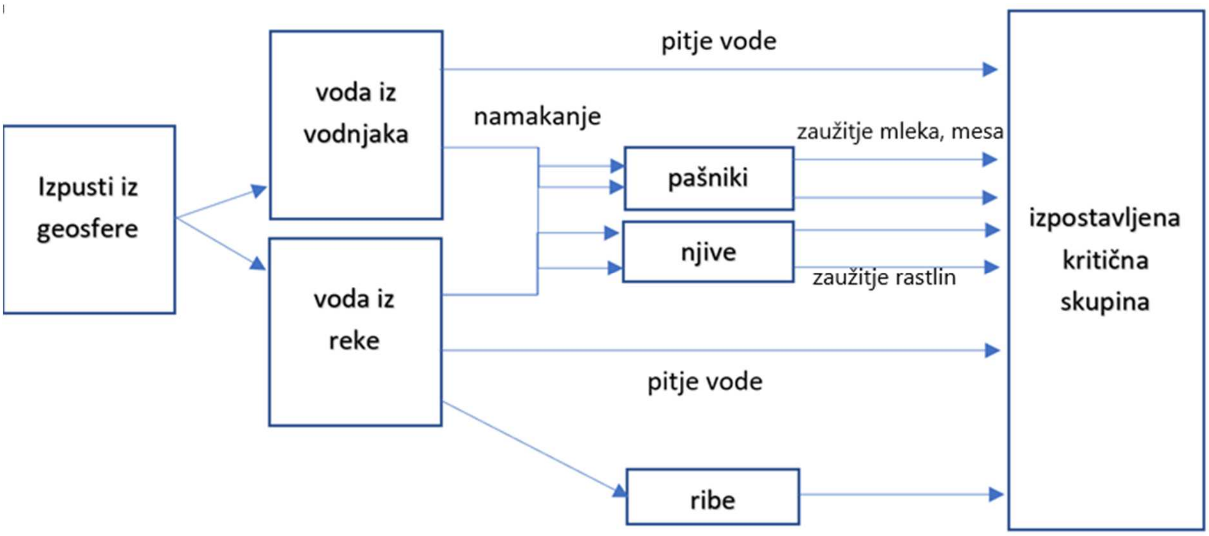

Figure 7.31: Biosphere sub-model

Izpusti iz geosfere	Releases from geosphere
voda iz vodnjaka	water from the well
voda iz reke	water from the river
pitje vode	drinking of water
namakanje	irrigation
zaužitje mleka, mesa	ingestion of milk, meat
pašniki	pastures
njive	fields
zaužitje rastlin	ingestion of crops
ribe	fish
izpostavljena kritična skupina	exposed critical group

The parameters used in the biosphere model that were also used in the system model are presented in the report on parameters used. [83]

7.3.5.5 System model

The system model for the LILW repository is divided into three sub-models: the nearfield model, the farfield model and the biosphere model. The transport of radionuclides from the nearfield through the farfield to the biosphere is illustrated schematically in Figure 7.32 below.

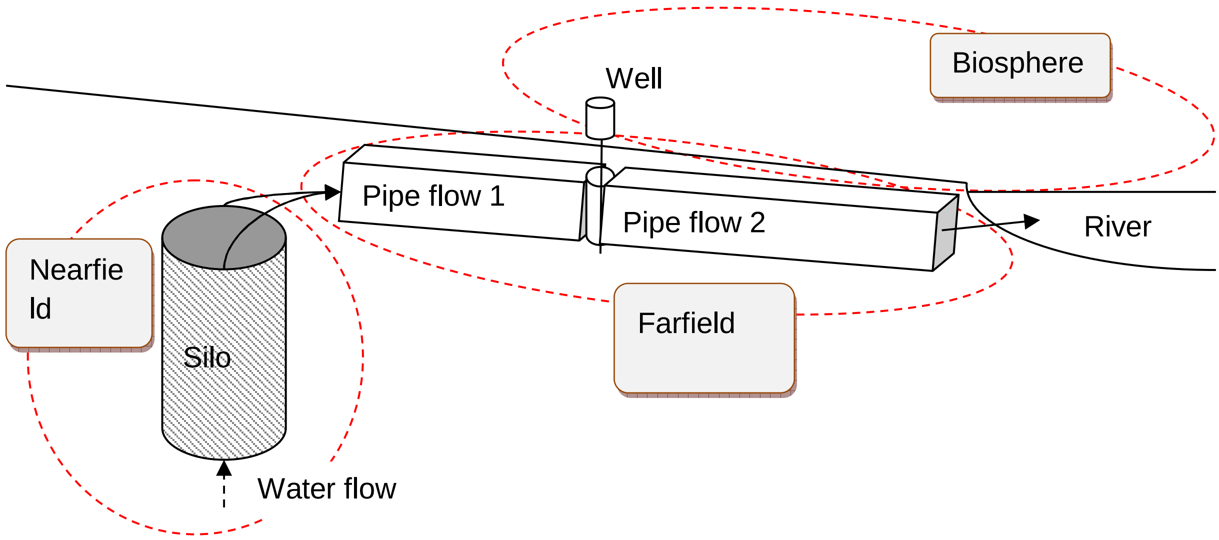

Figure 7.32: Transport of radionuclides from nearfield through farfield to biosphere, as used in system model

In system terms, all three models are represented in Ecolego software as a single integrated system. As stated above, the input data is prepared on the basis of process sub-models, which are presented in detail in the reports on degradation of engineered barriers [75] and on the flow of water through the silo, [82] the report on the geosphere model, [84] and the report on the biosphere sub-model, [74] which are summarised below in this section.

A compartment model approach is used in Ecolego, as illustrated in Figure 7.33 below. Under this approach, the model describes a physical system and is illustrated in space with individual compartments. Each of the compartments is homogenous, and corresponds to physically separate parts of the whole system. In mathematical terms, this represents a system of ordinary differential equations, where each represents the state of a particular variable, e.g. the inventory of radionuclides in an individual compartment. Multiple states are defined for each compartment: one for each radionuclide present in waste. The compartments are interconnected by using transfer rates corresponding to radionuclides' transport processes between compartments. These transport processes represent the physical and chemical behaviour of the system over time, and include parameters that may be dependent on a particular radionuclide, its chemical form or other properties of the system. The three main compartments are connected via simple mass transfer equations.

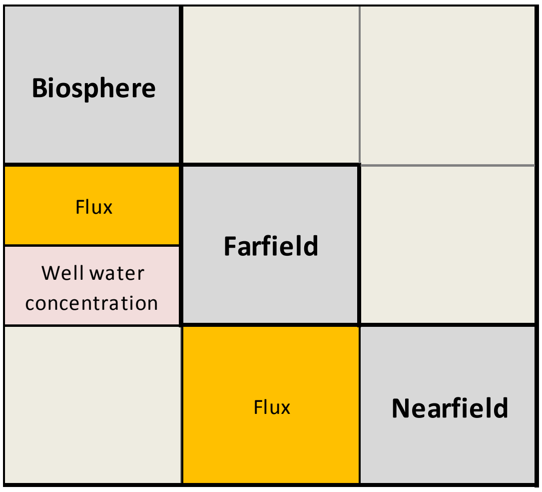

Figure 7.33: Three compartments and their interaction in an Ecolego matrix (where well water concentration refers to the activity of radionuclides in water from the well)

The rate of change (of concentration) for each state of a variable in a particular compartment is described by the following differential equation:

$$\frac{dA_i^k}{dt} = \sum_j TC_{ji}^k A_j^k - \sum_j TC_{ij}^k A_i^k + \lambda^p A_i^p - \lambda^k A_i^k$$

Equation 7.6:

where:
A_i^k is the quantity of radionuclide *k* in compartment *i* [mol]

A_j^k is the quantity of radionuclide *k* in compartment *j* [mol]

TC_{ij}^k is the transfer coefficient for radionuclide *k* from compartment *i* to compartment *j*

[y^{-1}]

TC_{ij}^k is the transfer coefficient for radionuclide k from compartment *j* to compartment *i* [y^{-1}]

λ^k and λ^prepresent the decay rates of radionuclides *k* and *p*, where radionuclide *p* is the parent of radionuclide *k* in the decay chain [y]

The equations for individual transfer coefficients are presented in detail in the report on models. [74]

7.3.5.6 Treatment of scenario of inadvertent human intrusion

As defined previously in Section 7.3.2.3 of this draft safety analysis report, the scenario of inadvertent human intrusion into the repository after the end of institutional controls is highly unlikely. The reasons for this are:

- practically the only possible scenario that can reach the depth of the repository is geotechnical drilling,

- there is very little motivation for drilling to the depth of the repository at the repository site (there are no mineral ores at the depth of the repository, and groundwater in the aquifer is above the level of the repository),
- the repository will be constructed very robustly, with thick concrete walls, and waste will be conditioned for disposal in concrete containers and metal drums, which practically prevents drilling, even if a driller wishes to drill at such depth.

The scenario of inadvertent human intrusion is therefore treated as an alternate evolution scenario for the repository, and criteria for the implementation of measures to respond to the burden imposed on a member of the public are taken into account for it in accordance with the JV5 rulebook. [21]

The safety analysis thus addressed a scenario of intrusion that could occur because of planned geotechnical drilling at the repository site after the end of institutional controls.

7.3.5.6.1 Intrusion into repository because of drilling

In the event of drilling into the repository, there is very little quantity of waste excavated (core drills have a diameter of just a few inches). The waste in the core obtained with a core drill would be mixed with clean material from above the repository. Assuming that the drill fully penetrates the repository in the intrusion scenario, the concentration of waste in the mixed material is determined as follows:

$$Conc_{surface}^{i} = \frac{Conc_{waste}^{i}}{\rho_{surface}} \frac{V_{exh}}{V_{exh} + V_{clean}}$$

Equation 7.7:

Where:

$Conc^{i}_{surface}$ is the concentration of radionuclide *i* in the mixed material [Bq/kg]

$\rho_{surface}$ is the density of the earth above the repository (at the surface) [kg/m^3]

$Conc^{i}_{waste}$ is the concentration of radionuclide *i* in waste reached by the drill [Bq/m^3]

V_{exh} is the volume of waste excavated by the drill [m^3]

V_{clean} is the assumed quantity of clean material with which waste has been mixed [m^3]

It was assumed in the safety analysis that the volume containing disposed waste (the height of waste disposed of in the silo) is 33 m in height, and the volume of clean earth that can be mixed with the waste is 100 m^3 (an area of 100 m^2 to a depth of 1 m). All the parameters used to estimate the dose were assumed on a conservative basis. The dose that a driller may obtain in the intrusion scenario depends primarily on the time spent in the vicinity of the drilled core (when core drilling is used), and on the concentration of radionuclides in the soil in the vicinity (when a core is not obtained, and the excavated sediment is deposited in the vicinity of the drilling site). The activity in the waste was estimated from the total activity of all disposed waste, and the estimated disposal volume (the mean volume assumed from various estimates was 14,800 m^3, which in later phases will be aligned with the latest data, which will only be finalised after the decommissioning of Krško NPP). The volume of waste excavated by drilling was estimated as follows:

$$V_{exh} = \frac{\pi d^2}{4} depth_{silo}$$

Equation 7.8:

Where:

d is the diameter of the borehole [m]

$depth_{silo}$ is the depth of waste disposed of in the silo [m]

7.3.5.6.1.1 Model for calculating dose from inadvertent intrusion into repository

The total exposure of the driller (total dose, Sv/year) under the scenario of inadvertent human intrusion is estimated as the sum of the doses received by the driller owing to ingestion, inhalation and external irradiation.

<u>Dose from ingestion</u>

The dose from ingestion relates primarily to the unintentional ingestion of contaminated soil, caused by the ingestion of dirt on the driller's hands. The dose was estimated as follows:

$$Dose_{ing}^{i} = Conc_{surface}^{i} \cdot ingSoil \cdot TimeOnsite \cdot DCC_{ingFood}^{i}$$

Equation 7.9:

Where:

ingSoil is the soil ingestion rate [kg/h]

TimeOnsite is the time during which the driller is exposed to excavated waste [h/y]

$DCC^{i}_{ingFoodi}$ is the dose conversion coefficient of radionuclide *i* when ingested [Sv/Bq]

<u>Dose from inhalation</u>

In the preparation of the model it was assumed that the driller is exposed to dust caused by drilling work. The estimated dose from inhalation was calculated as follows:

$$Dose_{inh}^{i} = Conc_{surface}^{i} \cdot Conc_{Dust} \cdot inhRate \cdot TimeDrilling \cdot DCC_{inh}^{i}$$

Equation 7.10:

Where:

$Conc_{Dust}$ is the concentration of dust in atmospheric air [kg/m^3]

InhRate is the breathing rate of the exposed person [m^3/h]

TimeDrilling is the duration of the drilling work [h/year]

DCC^i_{inhi} is the dose conversion coefficient of radionuclide *i* when inhaled [Sv/Bq]

Dose from external irradiation

The dose received by the driller as a result of external irradiation was calculated as follows:

$$Dose^i_{ext} = Conc^i_{surface} \cdot \rho_{surface} \cdot TimeOnsite \cdot DCC^i_{ext} \cdot F_{ext}$$

Equation 7.11:

Where:

DCC_{ext}^i is the dose conversion coefficient of radionuclide *i* owing to external irradiation [Sv/Bq]

F_{ext} is the area correction factor

The area correction factor is introduced to the estimated dose because the dose conversion coefficients have been developed for an infinite area. In the inadvertent intrusion scenario the driller will only be exposed to radiation in a limited area, which reduces the dose that the driller receives as a result of external irradiation. A highly complex approach to determining the impact of area on the dose received as a result of external irradiation has been developed, [89] where the factor is dependent on the energy of the radiation, its depth, etc. A slightly simpler approach [90] was used for the purposes of the safety analysis, and was recognised by the previously mentioned study [89] as a good approximation to the estimation of the area correction factor, other than for very low energy radiation. Because this radiation is the least significant in terms of consequences, this approach [90] was recognised in the safety analysis as satisfactory. The area correction factor is introduced to the estimated dose because the dose conversion coefficients have been developed for an infinite area. In the inadvertent intrusion scenario the driller will only be exposed to radiation in a limited area, which reduces the dose that the driller receives as a result of external irradiation.

The area correction factor is determined with the help of Table 7.36.

Table 7.36: Determination of area correction factor

case	F_{ext}
$0 < Area_{cont} \leq 25\ m^2$	$0.016 * Area_{cont}$
$25 < Area_{cont} \leq 100\ m^2$	$0.35 + 0.002\ Area_{cont}$
$100 < Area_{cont} \leq 500\ m^2$	$0.48 + 0.00065\ Area_{cont}$
$500 < Area_{cont} \leq 1{,}222\ m^2$	$0.67 + 0.00027\ Area_{cont}$
$Area_{cont} > 1{,}222\ m^2$	1

Where the contaminated area was determined as:

$$Area_{cont} = \frac{V_{exh} + V_{clean}}{depth_{cont}}$$

Equation 7.12:

Where:

$depth_{cont}$ is the depth of the contaminated area [m]

7.3.5.6.2 Scenario of inadvertent human intrusion: settlement of area after intrusion

This section describes the approach to evaluating the dose for an inhabitant who settles in the area of inadvertent intrusion after the event.

As a result of inadvertent intrusion through the repository (drilling in the repository area), as described in Section 7.3.5.6.1 of this draft safety analysis report, it is assumed that there is contamination of an area 29 m in diameter around the borehole. It is assumed that the soil is contaminated to a depth of 15 cm. The site and the borehole are abandoned after work is completed, without additional safety measures. Contamination occurs because the borehole was abandoned, and spreads as a result of weather phenomena. Shortly after, the area of the borehole is settled by a family, who establish a self-sufficient farm in the area, where they grow field crops and vegetables. Farming activities spread the contamination over a wider area (2,500 m^2). [91] In the scenario and the model it was assumed that the members of the family are exposed to radiation from the ingestion of contaminated food, the inhalation of contaminated dust, and external irradiation caused by radionuclides in the soil in the area. [91]

In the area where there is exposure to radiation, the concentration of an individual radionuclide in the soil [Bq/kg] is estimated with Equation 7.13: below:

$$Conc_{soil}^{i} = Conc_{surface}^{i} \frac{Area_{exp}}{Area_{cont}} \exp\left(-\log(2)\frac{T1}{halflife^{i}}\right)(1 - \exp\left(-\log(2)\frac{T2}{halflife^{i}}\right))/(\log(2)\frac{T2}{halflife^{j}})$$

Equation 7.13:

Where:

$Area_{exp}$ is the contaminated area where exposure occurs [m^2]

T1 is the time between the contamination and the beginning of exposure [years]

T2 is the assumed time of exposure after time T_1 [years]

$halflife^i$ is the half-life of radionuclide *i* [years]

The above equation (Equation 7.13:) takes account of the decrease in contamination because of radioactive decay, but conservatively does not take account of other processes that could lead to a reduction in contamination (erosion, leaching, harvesting). A period of one year was assumed for exposure time T_2.

The concentrations of radionuclides in the air and the food chain, and the doses received by the inhabitants from ingestion, inhalation and direct irradiation were calculated with the biosphere model presented in Section 7.3.5.4**Error! Reference source not found.** of this draft safety analysis report. The sole exceptions were concentrations of H-3 and C-14, which were calculated using the equations presented in the safety analysis report on models. [74] The parameters for evaluating the scenario of inadvertent human intrusion into the repository are presented in detail in Section 5 of the safety analysis report on parameters. [83]

7.3.5.7 Possibility of generation of tritium (H-3) and/or carbon-14 (C-14), and radon (Rn-222) in gases generated in the repository, and their impact on estimated effective dose

The generation of gases is defined in detail in the safety analysis report, [33] and a summary is given below.

H-3 and C-14 could also be generated at the repository in the gas generation reactions presented in Section 7.2.3.1 of this draft safety analysis report. As is evident from Table 7.21, the main waste streams containing H-3 and C-14 come from the decommissioning of Krško NPP. These wastes are expected to contain 1.23E+15 Bq of H-3 and 4.30E+13 Bq of C-14.

Ignoring the half-life, it was estimated [33] that the total disposed inventory could emit 4.34E+09 Bq/year of C-14, which would be in the form of methane. The methane containing radioactive carbon (C-14) will be highly diluted with non-radioactive methane generated by the degradation of organic waste.

In the case of tritium, ignoring the half-life, 2.0E+08 Bq/year of H-3 could be generated, although it will decay relatively quickly given the relatively short half-life (12 years).

Small quantities of radon (Rn-222), which has a half-life of 3.82 days, could also be generated at the repository. In the case of radon, the half-life is a key factor in the radon decaying before it can reach the surface.

The gases generated at the repository will migrate towards the surface in various ways [92] in bubbles travelling with groundwater, while some gases will also dissolve in the water. Migration towards the surface will be prevented by engineered barriers and by natural barriers (concrete, silt, clay top, etc.). The gas (C-14 in particular) will then bind to the root systems of plants growing in the vicinity of the repository, and some will migrate into the air, where it will be further diluted.

In light of the above, and given that the complete inventory is taken into account in the radionuclide transfer models, the assessment is that the contribution to the total estimated effective dose by gases is negligible.

7.3.6 RESULTS OF SAFETY ANALYSIS AFTER CLOSURE OF THE REPOSITORY: DETERMINISTIC COMPUTATIONS

7.3.6.1 Nominal scenario (normal evolution scenario)

The term "nominal" is used within the framework of the safety analysis to describe the predicted evolution (behaviour) of the repository in the absence of unusual and unexpected events and processes. The term was carefully chosen, as this scenario is not necessarily understood as "expected" or "most probable", but rather in the opinion of the group preparing the safety analysis it reflects the reasonable future evolution of events at the repository, and satisfies the purpose of the safety analysis, which is to show that the impact of the repository is smaller than the prescribed limit of 0.3 mSv/year.

The progression of events in the nominal scenario can be described as follows. It is envisaged that the silo is saturated at the time of its sealing. After the closure of the repository, the drainage system will also be closed and sealed. Because the silo lies in a saturated zone, the silo will become saturated, which will take some time. In the production of the safety analysis, it was conservatively assumed that the silo is immediately saturated upon sealing, and the potential transport of radionuclides whose main transport route is water is immediately possible. The silo saturation time will be evaluated further in the next phase of the safety analysis. In this phase this constitutes a certain level of additional safety (conservativeness), as saturation time delays the onset of the potential transport of radionuclides. After closure, active long-term controls and maintenance of the repository are envisaged for 50 years, followed by 250 years of passive controls (see Section 12 of this draft safety analysis report), which does not mean that potential releases are delayed for this time. It was assumed in the safety analysis that releases could occur irrespective of institutional controls, and that releases depend on the state of the engineered barriers. For the nominal scenario, the simultaneous onset of the degradation of all engineering barriers was assumed. [75]

The scenario envisages that there could be potential releases to the groundwater in the vicinity, which drains into the Sava, while the contamination plume is intercepted by the well. The impact of water drawn from the well on drainage into the Save is not taken into account in the sense that the potential contamination (radionuclides) would be divided into a part that goes to the well, and a part that drains into the Sava. The approach is not realistic in physical terms, but is conservative and much more justifiable than dividing the potential contamination into two parts. The model did not take account of the contamination of farmland by capillary action (and thus vertical flow), primarily because of a lack of information about how to model (assess) the contamination pathway. This scenario is included in an alternate evolution scenario where water from the well is used to irrigate farmland, thereby allowing the contamination to directly reach the upper layers of the soil.

The doses are calculated for a hypothetical representative of the potentially most-exposed population group. The following exposure pathways are considered:

- ingestion of fish from the river,
- ingestion of plants grown in an area irrigated with water from the river,
- ingestion of cow's milk and meat from cattle drinking water from the river and pasturing in areas irrigated with water from the river,
- ingestion of drinking water from the well that intercepts potential contamination; the well is located in the centre of potential contamination, 100 m from the repository. The use of the well is envisaged to begin after the end of institutional controls (300 years),
- inhalation and external exposure as a result of presence in an irrigated area.

In the calculation of the impact (dose), it was assumed that all ingested food and water is contaminated and an exposed individual spends 100% of his/her time in a contaminated area.

For the nominal scenario outlined above, deterministic analysis for a nominal set of input parameters was conducted. [83] These parameters were chosen to illustrate the nominal behaviour of the entire system. They were not chosen with the aim of illustrating the worst case or the most realistic case. The choice of parameters means that the case illustrated is somewhere in between. An assessment of the conservativeness is given in Section 7.3.7.4, where sensitivity analysis for individual parameters is disclosed.

The results of the calculations for the baseline nominal scenario are illustrated in the figures and tables below.

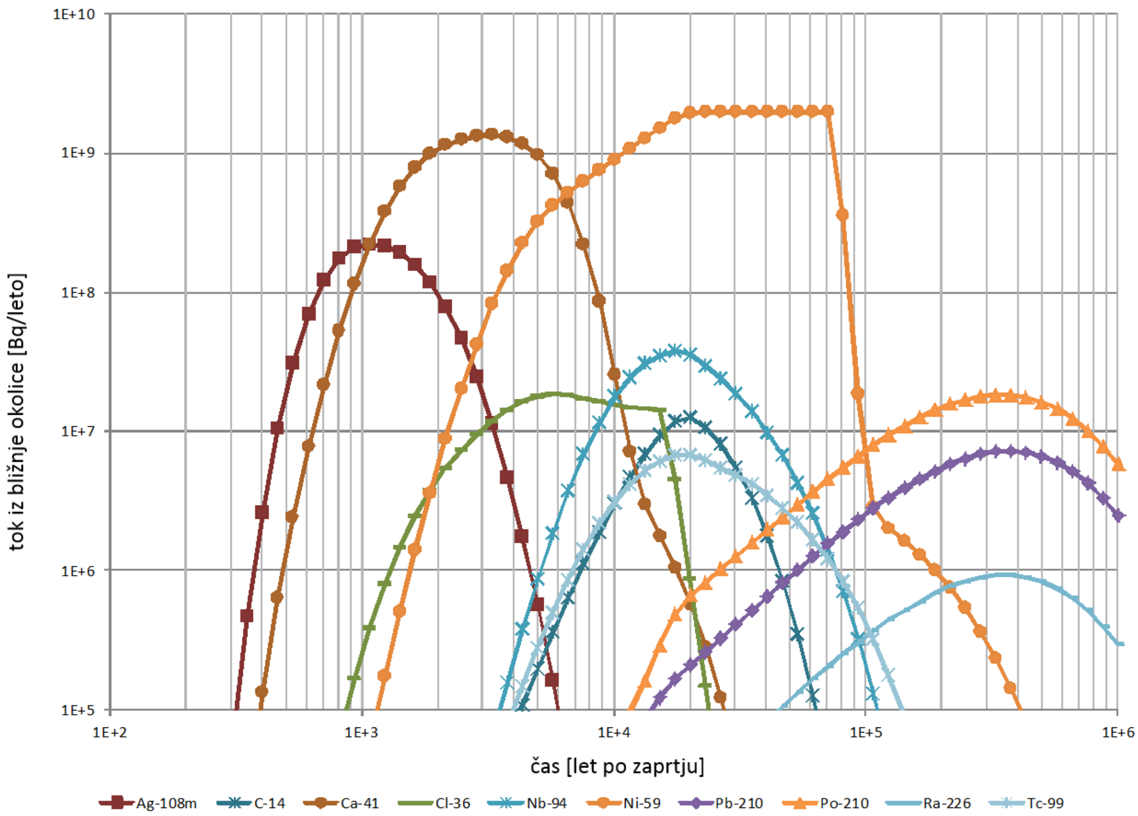

Figure 7.34: Releases of radionuclides from nearfield to farfield of repository under nominal scenario

tok iz bližnje okolice [Bq/leto]	flow from nearfield [Bq/year]
čas [let po zaprtju]	period [years after closure]

Table 7.37: Maximum releases and time of occurrence for releases from nearfield of repository (silo) to geosphere under nominal scenario

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Ac-227	4.1E+03	10,000	5.8E+03	15,199
Ag-108m	2.2E+08	1,072	2.2E+08	1,072
Am-241	8.6E-01	4,977	8.6E-01	4,977
Ba-133	5.2E-06	28	5.2E-06	28
C-14	3.0E+06	10,000	1.3E+07	20,092
Ca-41	1.4E+09	3,275	1.4E+09	3,275
Cd-109	9.7E-10	2	9.7E-10	2
Cd-113m	9.5E-04	38	9.5E-04	38

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Cl-36	1.9E+07	5,722	1.9E+07	5,722
Cm-244	5.4E-09	50	5.4E-09	50
Co-60	4.6E-02	12	4.6E-02	12
Cs-134	1.1E-02	3	1.1E-02	3
Cs-135	2.4E+04	10,000	2.4E+04	15,199
Cs-137	1.6E+01	464	1.6E+01	464
Eu-152	2.7E-08	33	2.7E-08	33
Eu-154	2.9E-09	22	2.9E-09	22
Eu-155	4.4E-10	11	4.4E-10	11
Fe-55	6.2E+02	4	6.2E+02	4
H-3	7.2E+00	305	7.2E+00	305
I-129	4.3E+04	2,154	4.3E+04	2,154
Na-22	4.0E-08	5	4.0E-08	5
Nb-93m	3.2E-14	25	3.2E-14	25
Nb-94	1.8E+07	10,000	3.8E+07	17,475
Ni-59	9.0E+08	10,000	2.0E+09	70,548
Ni-63	4.9E+03	1,233	4.9E+03	1,233
Np-237	3.2E+00	10,000	5.3E+02	70,548
Pa-231	4.4E+03	10,000	6.5E+03	15,199
Pb-210	4.1E+04	10,000	7.2E+06	376,494
Pd-107	2.4E+02	10,000	1.2E+03	20,092
Po-210	5.9E+04	10,000	1.9E+07	327,455
Pu-238	1.1E-09	933	1.1E-09	933
Pu-239	2.6E+02	10,000	1.7E+04	35,112
Pu-240	9.9E-01	10,000	2.5E+01	23,101

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Pu-241	2.0E-09	22	2.0E-09	22
Ra-226	3.4E+03	10,000	9.3E+05	376,494
Ra-228	2.0E+01	10,000	2.0E+04	376,494
Sb-125	6.0E-02	4	6.0E-02	4
Se-79	1.1E+04	8,697	1.1E+04	8,697
Sm-151	8.6E-09	705	8.6E-09	705
Sr-90	3.0E+01	464	3.0E+01	464
Tc-99	3.1E+06	10,000	6.8E+06	17,475
Th-228	3.4E-02	10,000	3.3E+01	376,494
Th-229	3.2E-03	10,000	3.6E+01	215,443
Th-230	4.3E-02	10,000	2.7E+03	376,494
Th-232	5.7E-03	10,000	3.3E+01	376,494
Tl-204	7.2E-08	8	7.2E-08	8
U-233	1.3E-01	10,000	2.2E+02	215,443
U-234	1.4E+01	10,000	1.9E+04	247,708
U-235	2.5E+00	10,000	4.9E+02	70,548
U-236	9.2E-04	10,000	2.7E-01	70,548
U-238	3.0E+02	10,000	5.7E+04	70,548

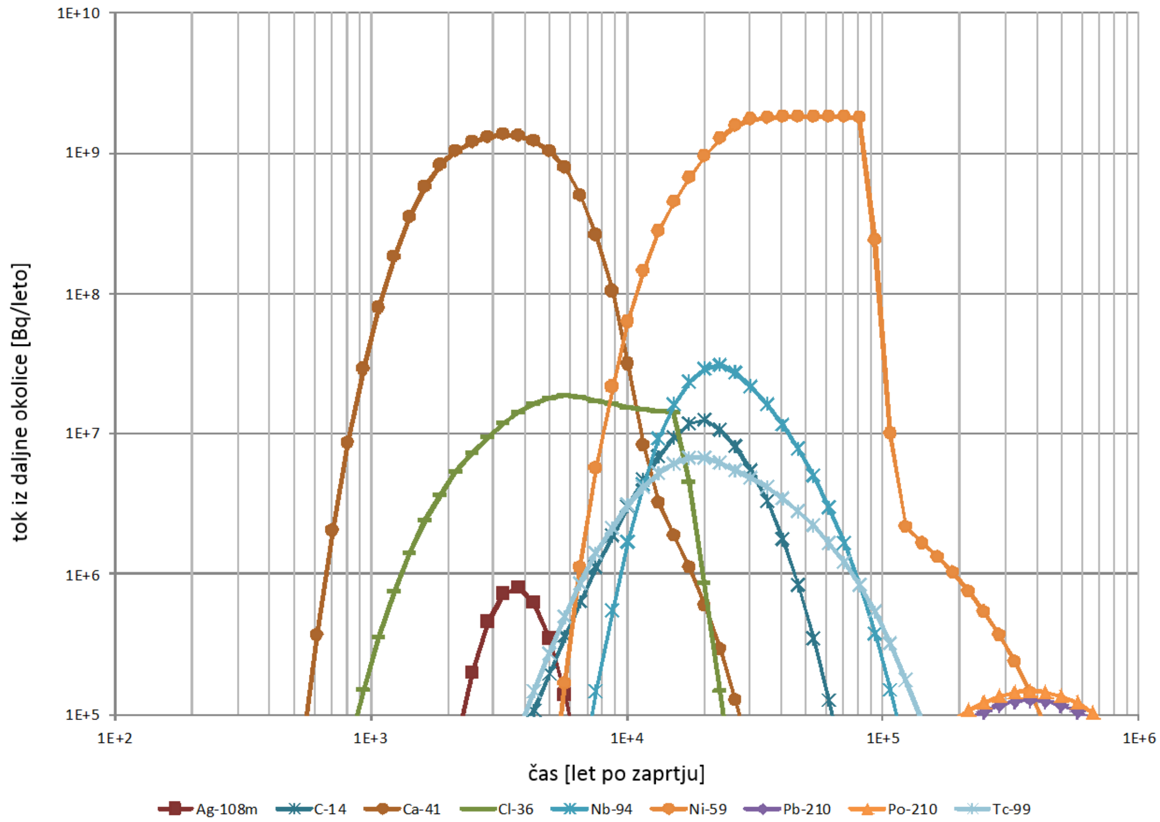

Figure 7.35: Releases of radionuclides from geosphere to biosphere (drainage into river) under nominal scenario

tok iz daljne okolice [Bq/leto]	flow from farfield [Bq/year]
čas [let po zaprtju]	period [years after closure]

Table 7.38: Maximum releases and time of occurrence for releases from geosphere model (farfield) to biosphere under nominal scenario

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Ac-227	1.5E-03	10,000	2.1E+03	93,260
Ag-108m	7.9E+05	3,765	7.9E+05	3,765
Am-241	4.7E-10	10,000	9.9E-10	13,219
Ba-133	3.4E-06	33	3.4E-06	33
C-14	3.0E+06	10,000	1.3E+07	20,092
Ca-41	1.3E+09	3,275	1.3E+09	3,275
Cd-109	2.5E-64	57	2.5E-64	57

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Cd-113m	5.8E-29	534	5.8E-29	534
Cl-36	1.9E+07	5,722	1.9E+07	5,722
Cm-244	7.1E-77	811	7.1E-77	811
Co-60	2.8E-47	231	2.8E-47	231
Cs-134	3.9E-118	100	3.9E-118	100
Cs-135	4.9E-39	10,000	3.8E+02	869,749
Cs-137	1.1E-78	1,874	1.1E-78	1,874
Eu-152	1.7E-82	614	1.7E-82	614
Eu-154	1.5E-89	404	1.5E-89	404
Eu-155	2.6E-98	231	2.6E-98	231
Fe-55	1.6E-46	115	1.6E-46	115
H-3	5.4E+00	305	5.4E+00	305
I-129	4.3E+04	2,477	4.3E+04	2,477
Na-22	7.5E-09	11	7.5E-09	11
Nb-93m	7.2E-35	534	7.2E-35	534
Nb-94	1.7E+06	10,000	3.1E+07	23,101
Ni-59	6.3E+07	10,000	1.8E+09	70,548
Ni-63	1.2E-07	3,275	1.2E-07	3,275
Np-237	2.3E+00	10,000	5.3E+02	70,548
Pa-231	1.3E-03	10,000	1.8E+03	93,260
Pb-210	8.2E-09	10,000	1.3E+05	376,494
Pd-107	3.1E+01	10,000	1.1E+03	23,101
Po-210	9.7E-09	10,000	1.5E+05	376,494
Pu-238	7.4E-23	2,848	7.4E-23	2,848
Pu-239	1.1E-01	10,000	1.2E+04	46,416

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Pu-240	2.5E-04	10,000	7.3E+00	35,112
Pu-241	2.4E-40	534	2.4E-40	534
Ra-226	1.6E-10	10,000	1.1E+04	376,494
Ra-228	2.9E-17	10,000	2.0E+02	1,000,000
Sb-125	6.2E-18	76	6.2E-18	76
Se-79	1.0E+04	10,000	1.1E+04	13,219
Sm-151	2.5E-25	3,275	2.5E-25	3,275
Sr-90	4.4E-20	1,417	4.4E-20	1,417
Tc-99	3.1E+06	10,000	6.8E+06	17,475
Th-228	2.3E-18	10,000	1.6E+01	1,000,000
Th-229	9.6E-07	10,000	8.6E+00	247,708
Th-230	1.6E-13	10,000	8.8E+02	376,494
Th-232	2.3E-18	10,000	1.6E+01	1,000,000
Tl-204	2.2E-08	14	2.2E-08	14
U-233	1.9E-04	10,000	1.8E+02	247,708
U-234	9.2E-10	10,000	2.1E+04	247,708
U-235	2.4E-08	10,000	4.7E+02	123,285
U-236	1.7E-09	10,000	2.6E-01	123,285
U-238	1.8E-08	10,000	5.5E+04	123,285

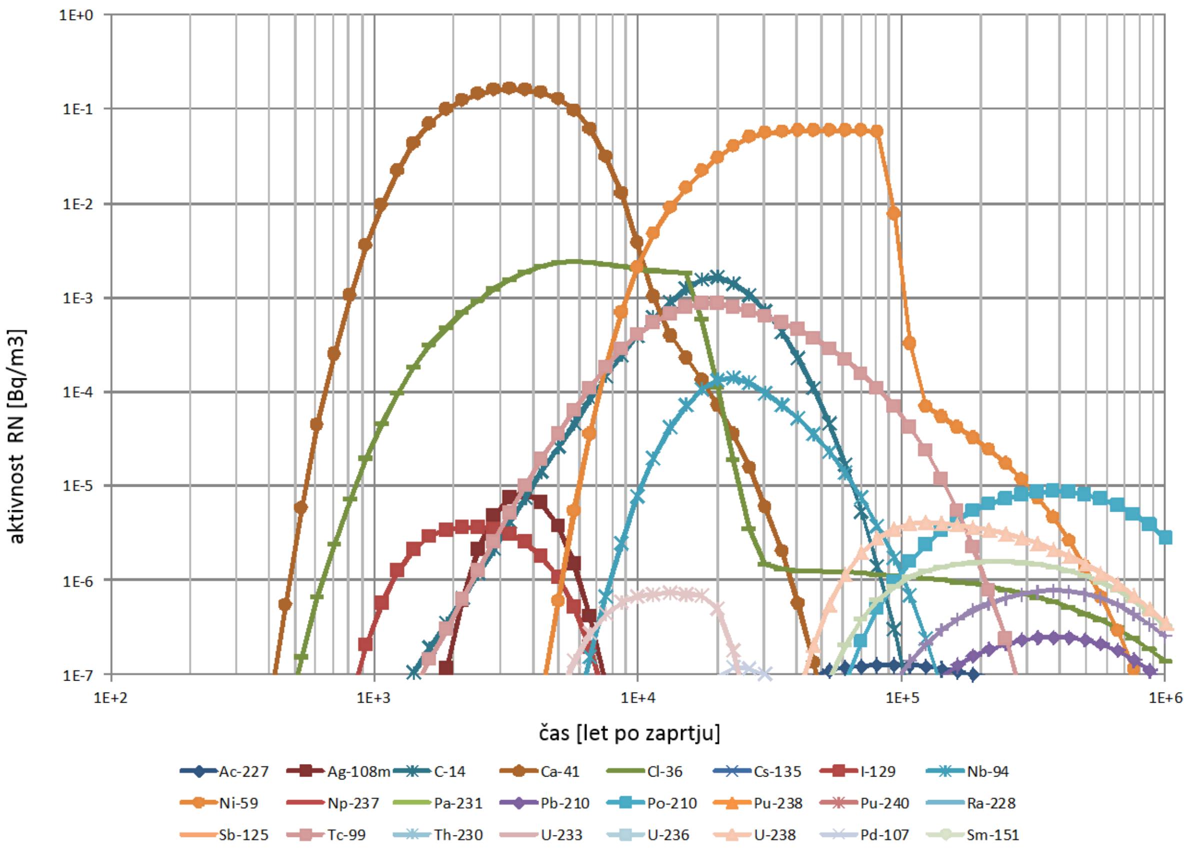

Figure 7.36: Concentration of radionuclides in river water under nominal scenario

aktivnost RN [Bq/m3]	RN activity [Bq/m3]
čas [let po zaprtju]	period [years after closure]

Table 7.39: Maximum releases and time of occurrence for concentration of radionuclides in river under nominal scenario

Radionuclide	Maximum before 10,000 years (Bq/m^3)	Time of occurrence (years)	Maximum (Bq/m^3)	Time of occurrence (years)
Ac-227	9.2E-14	10,000	1.3E-07	93,260
Ag-108m	8.4E-06	3,765	8.4E-06	3,765
Am-241	4.0E-21	10,000	8.4E-21	13,219
Ba-133	3.6E-16	33	3.6E-16	33
C-14	4.0E-04	10,000	1.6E-03	20,092
Ca-41	1.6E-01	3,275	1.6E-01	3,275
Cd-109	2.8E-75	57	2.8E-75	57

Radionuclide	Maximum before 10,000 years (Bq/m^3)	Time of occurrence (years)	Maximum (Bq/m^3)	Time of occurrence (years)
Cd-113m	6.7E-40	534	6.7E-40	534
Cl-36	2.4E-03	5,722	2.4E-03	5,722
Cm-244	5.8E-87	811	5.8E-87	811
Co-60	5.8E-58	231	5.8E-58	231
Cs-134	1.1E-128	100	1.1E-128	100
Cs-135	1.4E-49	10,000	1.1E-08	869,749
Cs-137	3.3E-89	1,874	3.3E-89	1,874
Eu-152	2.8E-93	614	2.8E-93	614
Eu-154	2.5E-100	404	2.5E-100	404
Eu-155	4.2E-109	231	4.2E-109	231
Fe-55	2.1E-56	115	2.1E-56	115
H-3	7.1E-10	305	7.1E-10	305
I-129	3.7E-06	2,477	3.7E-06	2,477
Na-22	9.8E-19	11	9.8E-19	11
Nb-93m	3.3E-46	534	3.3E-46	534
Nb-94	7.7E-06	10,000	1.4E-04	23,101
Ni-59	2.0E-03	10,000	5.8E-02	70,548
Ni-63	3.8E-18	3,275	3.8E-18	3,275
Np-237	3.0E-10	10,000	7.0E-08	70,548
Pa-231	1.3E-14	10,000	1.8E-08	93,260
Pb-210	1.6E-20	10,000	2.5E-07	376,494
Pd-107	3.3E-34	2,848	3.3E-34	2,848
Po-210	4.7E-13	10,000	5.1E-08	46,416
Pu-238	1.1E-15	10,000	3.2E-11	35,112
Pu-239	1.1E-51	534	1.1E-51	534

Radionuclide	Maximum before 10,000 years (Bq/m^3)	Time of occurrence (years)	Maximum (Bq/m^3)	Time of occurrence (years)
Pu-240	1.1E-20	10,000	7.8E-07	376,494
Pu-241	2.0E-27	10,000	1.4E-08	1,000,000
Ra-226	5.1E-28	76	5.1E-28	76
Ra-228	5.0E-30	1,417	5.0E-30	1,417
Sb-125	5.3E-18	10,000	4.7E-11	247,708
Se-79	8.9E-25	10,000	4.8E-09	376,494
Sm-151	1.3E-29	10,000	8.8E-11	1,000,000
Sr-90	2.9E-18	14	2.9E-18	14
Tc-99	1.4E-14	10,000	1.4E-08	247,708
Th-228	6.8E-20	10,000	1.6E-06	247,708
Th-229	1.8E-18	10,000	3.5E-08	123,285
Th-230	1.3E-19	10,000	1.9E-11	123,285
Th-232	1.3E-18	10,000	4.1E-06	123,285
Tl-204	3.3E-09	10,000	1.2E-07	23,101
U-233	6.7E-07	10,000	7.3E-07	13,219
U-234	1.8E-36	3,275	1.8E-36	3,275
U-235	5.8E-19	10,000	8.7E-06	376,494
U-236	4.1E-04	10,000	8.9E-04	17,475
U-238	1.3E-29	10,000	8.8E-11	1,000,000

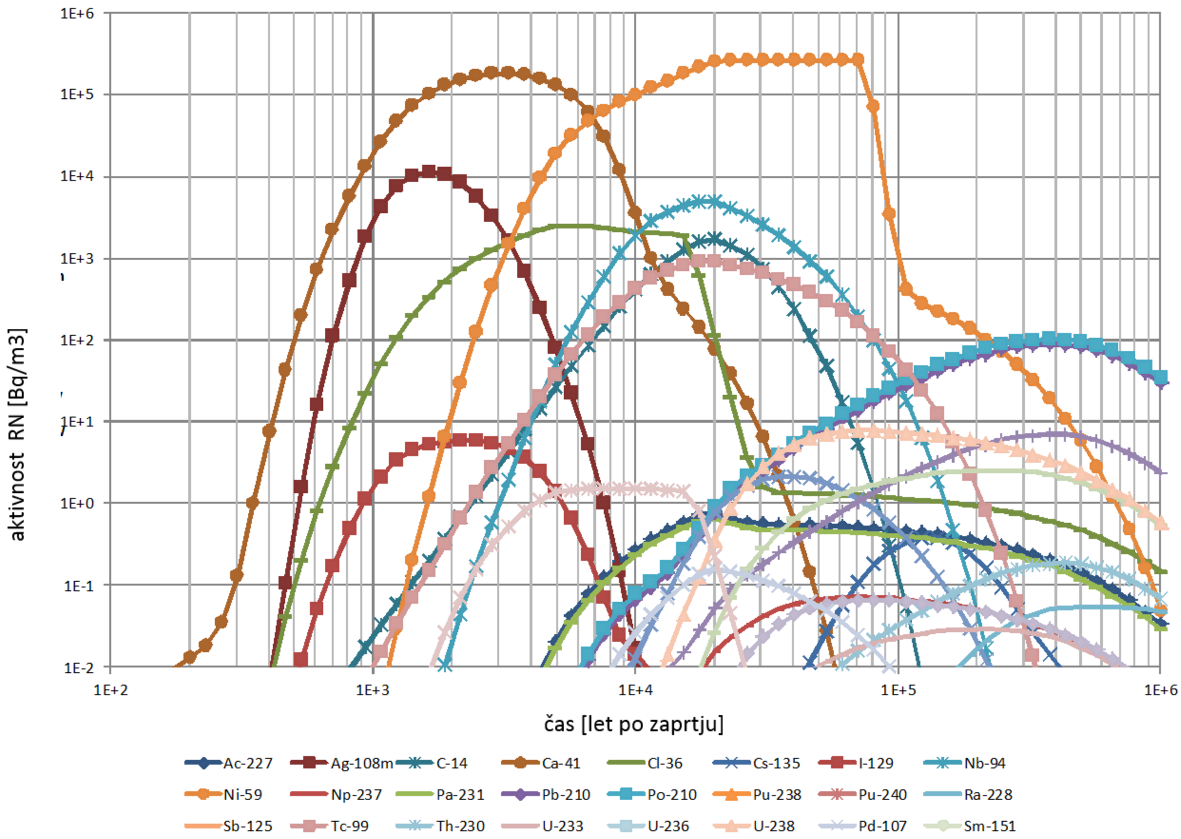

Figure 7.37: Concentration of radionuclides in well under nominal scenario

aktivnost RN [Bq/m3]	RN activity [Bq/m3]
čas [let po zaprtju]	period [years after closure]

Table 7.40: Maximum releases and time of occurrence for concentration of radionuclides in well under nominal scenario

Radionuclide	Maximum before 10,000 years (Bq/m^3)	Time of occurrence (years)	Maximum (Bq/m^3)	Time of occurrence (years)
Ac-227	2.7E-01	10,000	7.6E-01	20,092
Ag-108m	1.1E+04	1,630	1.1E+04	1,630
Am-241	1.1E-06	6,579	1.1E-06	6,579
Ba-133	6.6E-10	28	6.6E-10	28
C-14	4.1E+02	10,000	1.7E+03	20,092
Ca-41	1.8E+05	3,275	1.8E+05	3,275
Cd-109	1.0E-30	22	1.0E-30	22

Radionuclide	Maximum before 10,000 years (Bq/m^3)	Time of occurrence (years)	Maximum (Bq/m^3)	Time of occurrence (years)
Cd-113m	1.2E-14	201	1.2E-14	201
Cl-36	2.5E+03	5,722	2.5E+03	5,722
Cm-244	1.3E-34	351	1.3E-34	351
Co-60	9.7E-20	87	9.7E-20	87
Cs-134	3.1E-45	38	3.1E-45	38
Cs-135	2.5E-09	10,000	3.8E-01	123,285
Cs-137	5.0E-29	933	5.0E-29	933
Eu-152	3.7E-36	231	3.7E-36	231
Eu-154	2.8E-39	152	2.8E-39	152
Eu-155	6.4E-43	87	6.4E-43	87
Fe-55	7.8E-17	43	7.8E-17	43
H-3	9.4E-04	305	9.4E-04	305
I-129	5.8E+00	2,154	5.8E+00	2,154
Na-22	4.2E-12	6	4.2E-12	6
Nb-93m	1.0E-23	175	1.0E-23	175
Nb-94	1.9E+03	10,000	4.9E+03	17,475
Ni-59	1.0E+05	10,000	2.6E+05	70,548
Ni-63	2.2E-03	1,630	2.2E-03	1,630
Np-237	4.2E-04	10,000	7.2E-02	70,548
Pa-231	2.3E-01	10,000	6.5E-01	20,092
Pb-210	6.6E-02	10,000	8.9E+01	376,494
Pd-107	9.7E-17	1,417	9.7E-17	1,417
Po-210	1.2E-02	10,000	2.1E+00	40,370
Pu-238	4.3E-05	10,000	2.8E-03	26,561
Pu-239	2.9E-22	201	2.9E-22	201

Radionuclide	Maximum before 10,000 years (Bq/m^3)	Time of occurrence (years)	Maximum (Bq/m^3)	Time of occurrence (years)
Pu-240	4.4E-03	10,000	7.0E+00	376,494
Pu-241	2.4E-16	10,000	5.5E-02	572,237
Ra-226	5.6E-10	28	5.6E-10	28
Ra-228	1.1E-08	705	1.1E-08	705
Sb-125	1.6E-09	10,000	1.4E-03	215,443
Se-79	2.4E-08	10,000	1.9E-01	432,876
Sm-151	1.7E-17	10,000	4.3E-03	572,237
Sr-90	8.2E-12	9	8.2E-12	9
Tc-99	4.1E-07	10,000	2.9E-02	215,443
Th-228	5.8E-05	10,000	2.6E+00	247,708
Th-229	1.0E-05	10,000	6.6E-02	81,113
Th-230	3.4E-09	10,000	3.6E-05	81,113
Th-232	1.2E-03	10,000	7.7E+00	81,113
Tl-204	2.6E-02	10,000	1.6E-01	20,092
U-233	1.5E+00	10,000	1.5E+00	10,000
U-234	7.5E-17	1,630	7.5E-17	1,630
U-235	7.7E-02	10,000	1.0E+02	376,494
U-236	4.2E+02	10,000	9.1E+02	17,475
U-238	1.8E-17	10,000	4.3E-03	572,237

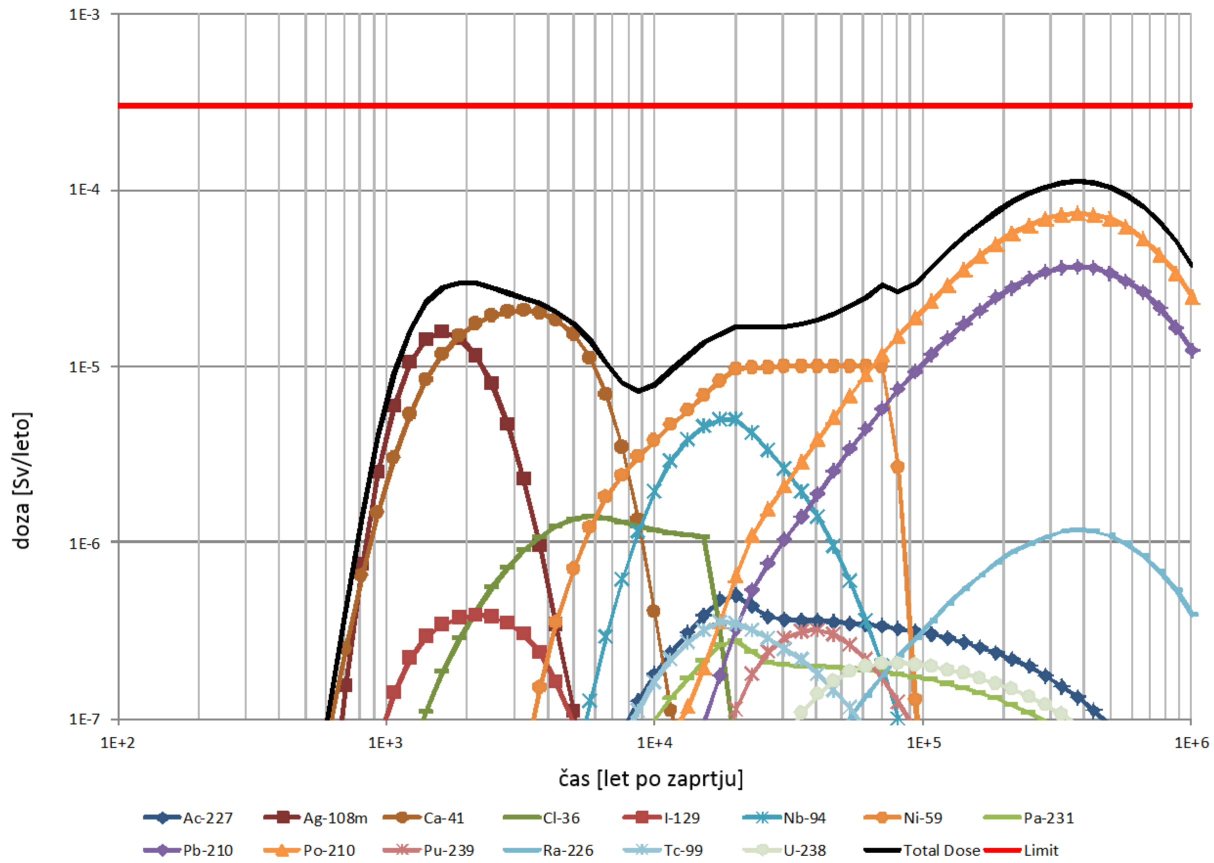

Figure 7.38: Annual dose for representative member of most-exposed population group under nominal scenario (limits also shown for comparison)

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

Table 7.41: Maximum doses and time of occurrence for annual dose under nominal scenario

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Total dose	3.0E-05	1,874	1.1E-04	376,494
Ac-227	1.8E-07	10,000	5.0E-07	20,092
Ag-108m	1.6E-05	1,630	1.6E-05	1,630
Am-241	< 1E-10	6,579	< 1E-10	6,579
Ba-133	< 1E-10	305	< 1E-10	305
C-14	7.2E-09	10,000	3.0E-08	20,092
Ca-41	2.1E-05	3,275	2.1E-05	3,275

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Cd-109	< 1E-10	57	< 1E-10	57
Cd-113m	< 1E-10	305	< 1E-10	305
Cl-36	1.4E-06	5,722	1.4E-06	5,722
Cm-244	< 1E-10	351	< 1E-10	351
Co-60	< 1E-10	305	< 1E-10	305
Cs-134	< 1E-10	305	< 1E-10	305
Cs-135	< 1E-10	10,000	4.5E-10	123,285
Cs-137	< 1E-10	933	< 1E-10	933
Eu-152	< 1E-10	305	< 1E-10	305
Eu-154	< 1E-10	305	< 1E-10	305
Eu-155	< 1E-10	305	< 1E-10	305
Fe-55	< 1E-10	305	< 1E-10	305
H-3	< 1E-10	305	< 1E-10	305
I-129	3.8E-07	2,154	3.8E-07	2,154
Na-22	< 1E-10	11	< 1E-10	11
Nb-93m	< 1E-10	305	< 1E-10	305
Nb-94	1.9E-06	10,000	5.0E-06	17,475
Ni-59	3.8E-06	10,000	9.9E-06	70,548
Ni-63	< 1E-10	1,630	< 1E-10	1,630
Np-237	< 1E-10	10,000	4.7E-09	70,548
Pa-231	9.9E-08	10,000	2.8E-07	20,092
Pb-210	2.7E-08	10,000	3.7E-05	376,494
Pd-107	< 1E-10	10,000	< 1E-10	20,092
Po-210	5.6E-08	10,000	7.4E-05	376,494
Pu-238	< 1E-10	1,417	< 1E-10	1,417

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Pu-239	1.9E-09	10,000	3.2E-07	40,370
Pu-240	< 1E-10	10,000	4.2E-10	26,561
Pu-241	< 1E-10	305	< 1E-10	305
Ra-226	7.4E-10	10,000	1.2E-06	376,494
Ra-228	< 1E-10	10,000	2.3E-08	572,237
Sb-125	< 1E-10	76	< 1E-10	76
Se-79	2.6E-09	10,000	2.6E-09	10,000
Sm-151	< 1E-10	1,630	< 1E-10	1,630
Sr-90	< 1E-10	705	< 1E-10	705
Tc-99	1.6E-07	10,000	3.5E-07	17,475
Th-228	< 1E-10	10,000	1.9E-10	572,237
Th-229	< 1E-10	10,000	4.0E-10	215,443
Th-230	< 1E-10	10,000	2.3E-08	432,876
Th-232	< 1E-10	10,000	5.9E-10	572,237
Tl-204	< 1E-10	16	< 1E-10	16
U-233	< 1E-10	10,000	8.8E-10	215,443
U-234	< 1E-10	10,000	7.5E-08	247,708
U-235	< 1E-10	10,000	1.8E-09	81,113
U-236	< 1E-10	10,000	< 1E-10	81,113
U-238	< 1E-10	10,000	2.1E-07	81,113

The results show that the dominant pathway for radionuclides is drinking water from the well, which is also evident from the concentrations, which in the well are several powers of ten higher than in the river. As stated above, the well is located in the centre of potential contamination, and is 100 m from the repository. The probability of events evolving in this manner (the most conservative site for installing a well for drinking water) was not evaluated, although it could be envisaged that it would be low, particularly if it is assumed that in this case there is no dilution with uncontaminated water, and the contamination plume would be relatively narrow. The level of dependence between the concentration and the distance of the well from the repository was assessed. The dependence proved to be very low.

The calculated annual doses are presented in Figure 7.38 and Table 7.41 above. The calculated maximum dose for the total period covered by the calculation (one million years) is 0.11 mSv/year, and is still below the limit prescribed for the repository. The radionuclides that contribute most to the calculated dose are Pb-210 and Po-210, which peak after 300,000 years. The maximum dose calculated in the period of 10,000 years after the closure of the repository is 0.03 mSv/year, which is a tenth of the limit, and occurs 1,900 years after the closure of the repository; the dominant radionuclides are Ca-41 and Ag-108m.

Ag-108m reaches its maximum release from the farfield to the geosphere 1,072 years after closure. The maximum release into the river occurs 3,765 years after closure, and produces the maximum concentration in the river at the same time. The maximum in the well, which is closer than the river, comes 1,630 years after closure. The maximum dose for the most-exposed population group (impact from well) also occurs 1,630 years after closure.

7.3.6.1.1 Variant of nominal scenario with alternate degradation of engineered barriers

This section presents the results of the deterministic analysis for the variant of the nominal scenario including an alternate model of the degradation of engineered barriers, namely conceptual model 1 for the degradation of engineered barriers [75] presented in Section 7.3.5.1.1 of this draft safety analysis report. The assumptions for the transport of radionuclides into the geosphere and the biosphere are the same as under the nominal scenario.

This scenario envisages that there is degradation of the engineered barriers from the outside in, which means that the internal barriers begin to fail later than the external barriers. Given the current state of knowledge and understanding, this degradation model is probably more realistic than that assumed under the nominal scenario. Because understanding and interpreting this model is highly demanding, and there thus remain several uncertainties in the determination of the parameters of the model, [75] on the basis of the results obtained under this scenario it was decided not to use this model of the degradation of engineered barriers in the nominal scenario (Section 7.3.6.1), but instead to use the model of simultaneous degradation of engineered barriers, which is also more conservative, in the nominal scenario.

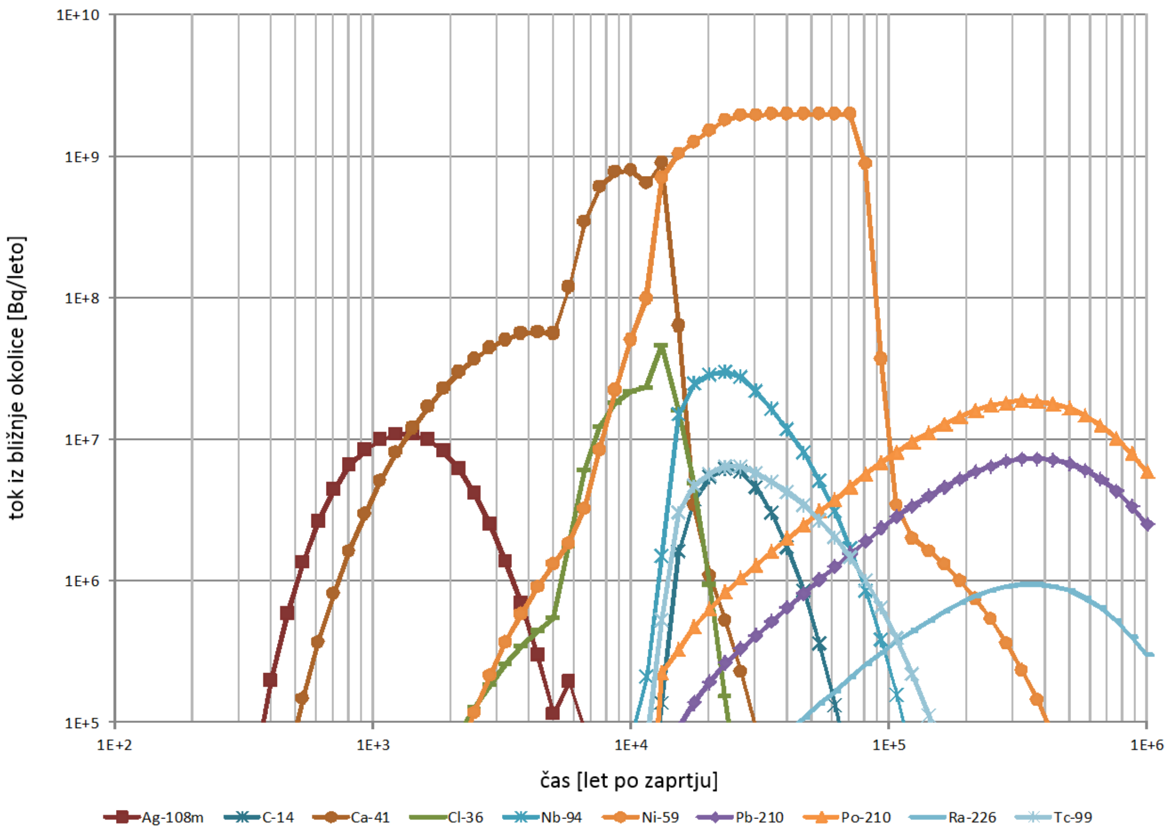

Figure 7.39: Releases of radionuclides from nearfield to geosphere under variant of nominal scenario with alternate degradation of engineered barriers

tok iz bližnje okolice [Bq/leto]	flow from nearfield [Bq/year]
čas [let po zaprtju]	period [years after closure]

Table 7.42: Maximum releases and time of occurrence for releases from nearfield to geosphere under variant of nominal scenario with alternate degradation of engineered barriers

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Ac-227	4.8E+03	10,000	6.0E+03	15,199
Ag-108m	1.1E+07	1,233	1.1E+07	1,233
Am-241	1.9E-03	4,329	1.9E-03	4,329
Ba-133	5.2E-06	28	5.2E-06	28
C-14	7.1E+03	10,000	6.2E+06	23,101
Ca-41	7.8E+08	10,000	9.0E+08	13,219
Cd-109	9.7E-10	2	9.7E-10	2

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Cd-113m	9.5E-04	38	9.5E-04	38
Cl-36	2.2E+07	10,000	4.7E+07	13,219
Cm-244	5.4E-09	50	5.4E-09	50
Co-60	4.6E-02	12	4.6E-02	12
Cs-134	1.1E-02	3	1.1E-02	3
Cs-135	7.5E+03	10,000	3.6E+04	15,199
Cs-137	3.1E+00	43	3.1E+00	43
Eu-152	2.7E-08	33	2.7E-08	33
Eu-154	2.9E-09	22	2.9E-09	22
Eu-155	4.4E-10	11	4.4E-10	11
Fe-55	6.2E+02	4	6.2E+02	4
H-3	3.1E-01	305	3.1E-01	305
I-129	2.6E+04	7,565	2.6E+04	7,565
Na-22	4.0E-08	5	4.0E-08	5
Nb-93m	3.2E-14	25	3.2E-14	25
Nb-94	7.7E+04	10,000	3.0E+07	23,101
Ni-59	5.1E+07	10,000	2.0E+09	70,548
Ni-63	8.9E+01	933	8.9E+01	933
Np-237	5.6E-03	10,000	5.3E+02	70,548
Pa-231	4.8E+03	10,000	6.5E+03	15,199
Pb-210	3.2E+03	10,000	7.3E+06	376,494
Pd-107	1.7E+00	10,000	1.1E+03	23,101
Po-210	3.4E+03	10,000	1.9E+07	327,455
Pu-238	5.1E-10	404	5.1E-10	404
Pu-239	3.6E-01	10,000	1.4E+04	40,370

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Pu-240	1.2E-03	10,000	1.6E+01	30,539
Pu-241	2.0E-09	22	2.0E-09	22
Ra-226	2.8E+02	10,000	9.4E+05	376,494
Ra-228	3.1E-02	10,000	2.0E+04	376,494
Sb-125	6.0E-02	4	6.0E-02	4
Se-79	1.1E+04	10,000	4.1E+04	13,219
Sm-151	4.7E-09	404	4.7E-09	404
Sr-90	1.1E+00	404	1.1E+00	404
Tc-99	2.7E+04	10,000	6.4E+06	26,561
Th-228	5.2E-05	10,000	3.3E+01	376,494
Th-229	5.3E-06	10,000	3.7E+01	215,443
Th-230	6.5E-05	10,000	2.7E+03	376,494
Th-232	7.2E-06	10,000	3.3E+01	376,494
Tl-204	7.2E-08	8	7.2E-08	8
U-233	2.2E-04	10,000	2.2E+02	215,443
U-234	2.4E-02	10,000	1.9E+04	247,708
U-235	4.2E-03	10,000	4.9E+02	70,548
U-236	1.5E-06	10,000	2.7E-01	70,548
U-238	4.9E-01	10,000	5.7E+04	70,548

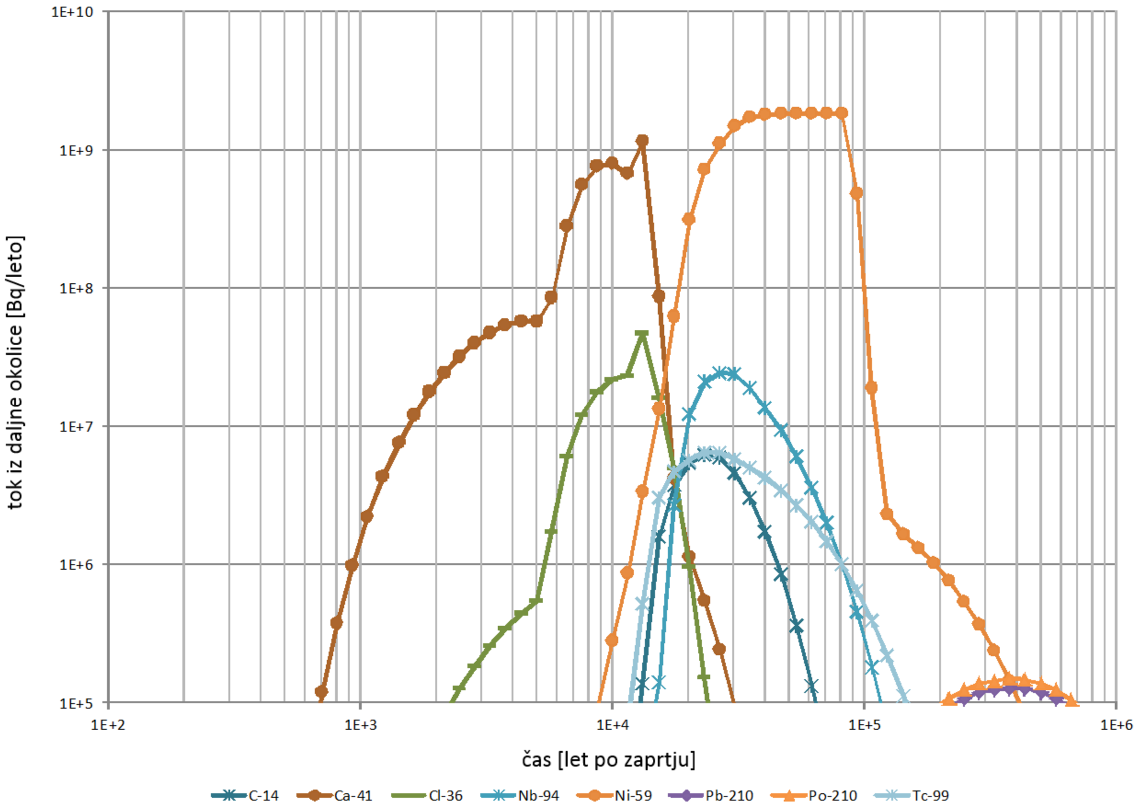

Figure 7.40: Releases of radionuclides from geosphere (drainage into river) to biosphere under variant of nominal scenario with alternate degradation of engineered barriers

tok iz daljne okolice [Bq/leto]	flow from farfield [Bq/year]
čas [let po zaprtju]	period [years after closure]

Table 7.43: Maximum releases and time of occurrence for releases from geosphere to biosphere under variant of nominal scenario with alternate degradation of engineered barriers

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Ac-227	6.2E-05	10,000	2.1E+03	93,260
Ag-108m	4.3E+04	3,765	4.3E+04	3,765
Am-241	1.5E-12	10,000	2.2E-12	11,498
Ba-133	3.4E-06	33	3.4E-06	33
C-14	7.1E+03	10,000	6.1E+06	23,101
Ca-41	7.9E+08	10,000	1.2E+09	13,219

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Cd-109	2.5E-64	57	2.5E-64	57
Cd-113m	5.8E-29	534	5.8E-29	534
Cl-36	2.2E+07	10,000	4.7E+07	13,219
Cm-244	7.1E-77	811	7.1E-77	811
Co-60	2.8E-47	231	2.8E-47	231
Cs-134	3.9E-118	100	3.9E-118	100
Cs-135	5.3E-41	10,000	3.8E+02	869,749
Cs-137	1.7E-79	1,630	1.7E-79	1,630
Eu-152	1.7E-82	614	1.7E-82	614
Eu-154	1.5E-89	404	1.5E-89	404
Eu-155	2.6E-98	231	2.6E-98	231
Fe-55	1.6E-46	115	1.6E-46	115
H-3	2.4E-01	305	2.4E-01	305
I-129	2.6E+04	8,697	2.6E+04	8,697
Na-22	7.5E-09	11	7.5E-09	11
Nb-93m	7.2E-35	534	7.2E-35	534
Nb-94	3.8E+03	10,000	2.4E+07	26,561
Ni-59	2.8E+05	10,000	1.8E+09	81,113
Ni-63	2.0E-09	2,848	2.0E-09	2,848
Np-237	3.8E-03	10,000	5.3E+02	70,548
Pa-231	5.3E-05	10,000	1.8E+03	93,260
Pb-210	4.4E-11	10,000	1.3E+05	376,494
Pd-107	6.3E-02	10,000	1.1E+03	30,539
Po-210	5.2E-11	10,000	1.5E+05	376,494
Pu-238	2.2E-23	2,477	2.2E-23	2,477

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Pu-239	2.8E-04	10,000	1.0E+04	53,367
Pu-240	5.0E-07	10,000	4.6E+00	40,370
Pu-241	2.4E-40	534	2.4E-40	534
Ra-226	8.9E-13	10,000	1.1E+04	376,494
Ra-228	7.2E-20	10,000	2.0E+02	1,000,000
Sb-125	6.2E-18	76	6.2E-18	76
Se-79	1.1E+03	10,000	2.3E+04	17,475
Sm-151	8.3E-26	2,848	8.3E-26	2,848
Sr-90	1.5E-21	1,233	1.5E-21	1,233
Tc-99	2.7E+04	10,000	6.4E+06	26,561
Th-228	5.7E-21	10,000	1.6E+01	1,000,000
Th-229	1.7E-09	10,000	8.7E+00	247,708
Th-230	6.7E-16	10,000	8.9E+02	376,494
Th-232	5.7E-21	10,000	1.6E+01	1,000,000
Tl-204	2.2E-08	14	2.2E-08	14
U-233	3.2E-07	10,000	1.8E+02	247,708
U-234	3.6E-12	10,000	2.1E+04	247,708
U-235	7.0E-11	10,000	4.7E+02	123,285
U-236	3.8E-12	10,000	2.6E-01	123,285
U-238	6.6E-11	10,000	5.5E+04	123,285

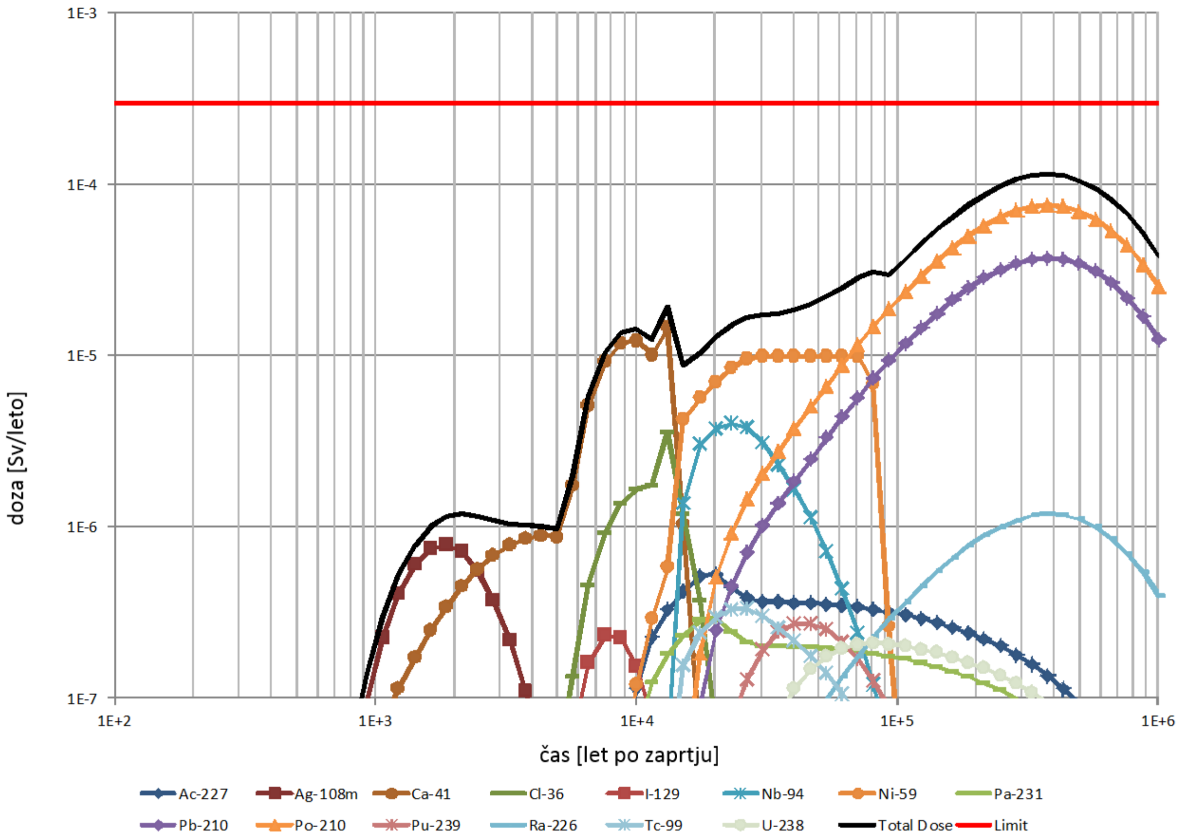

Figure 7.41: Annual dose for representative member of most-exposed population group under variant of nominal scenario with alternate degradation of engineered barriers (limits also shown for comparison)

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

Table 7.44: Maximum doses and time of occurrence for annual doses under variant of nominal scenario with alternate degradation of engineered barriers

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Total dose	1.4E-05	10,000	1.1E-04	376,494
Ac-227	1.1E-07	10,000	5.2E-07	20,092
Ag-108m	7.8E-07	1,874	7.8E-07	1,874
Am-241	< 1E-10	6,579	< 1E-10	6,579
Ba-133	< 1E-10	305	< 1E-10	305
C-14	< 1E-10	10,000	1.5E-08	23,101

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Ca-41	1.2E-05	10,000	1.4E-05	13,219
Cd-109	< 1E-10	57	< 1E-10	57
Cd-113m	< 1E-10	305	< 1E-10	305
Cl-36	1.7E-06	10,000	3.5E-06	13,219
Cm-244	< 1E-10	351	< 1E-10	351
Co-60	< 1E-10	305	< 1E-10	305
Cs-134	< 1E-10	305	< 1E-10	305
Cs-135	< 1E-10	10,000	4.5E-10	141,747
Cs-137	< 1E-10	614	< 1E-10	614
Eu-152	< 1E-10	305	< 1E-10	305
Eu-154	< 1E-10	305	< 1E-10	305
Eu-155	< 1E-10	305	< 1E-10	305
Fe-55	< 1E-10	305	< 1E-10	305
H-3	< 1E-10	305	< 1E-10	305
I-129	2.3E-07	7,565	2.3E-07	7,565
Na-22	< 1E-10	11	< 1E-10	11
Nb-93m	< 1E-10	305	< 1E-10	305
Nb-94	6.4E-09	10,000	4.0E-06	23,101
Ni-59	1.2E-07	10,000	9.9E-06	70,548
Ni-63	< 1E-10	1,630	< 1E-10	1,630
Np-237	< 1E-10	10,000	4.7E-09	70,548
Pa-231	6.3E-08	10,000	2.9E-07	20,092
Pb-210	3.4E-10	10,000	3.7E-05	376,494
Pd-107	< 1E-10	10,000	< 1E-10	26,561
Po-210	6.8E-10	10,000	7.5E-05	376,494

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Pu-238	< 1E-10	1,072	< 1E-10	1,072
Pu-239	< 1E-10	10,000	2.7E-07	46,416
Pu-240	< 1E-10	10,000	2.7E-10	30,539
Pu-241	< 1E-10	305	< 1E-10	305
Ra-226	< 1E-10	10,000	1.2E-06	376,494
Ra-228	< 1E-10	10,000	2.3E-08	572,237
Sb-125	< 1E-10	76	< 1E-10	76
Se-79	2.2E-09	10,000	6.2E-09	13,219
Sm-151	< 1E-10	1,233	< 1E-10	1,233
Sr-90	< 1E-10	705	< 1E-10	705
Tc-99	1.4E-09	10,000	3.3E-07	26,561
Th-228	< 1E-10	10,000	1.9E-10	572,237
Th-229	< 1E-10	10,000	4.1E-10	215,443
Th-230	< 1E-10	10,000	2.4E-08	432,876
Th-232	< 1E-10	10,000	5.9E-10	572,237
Tl-204	< 1E-10	16	< 1E-10	16
U-233	< 1E-10	10,000	8.9E-10	215,443
U-234	< 1E-10	10,000	7.6E-08	247,708
U-235	< 1E-10	10,000	1.9E-09	81,113
U-236	< 1E-10	10,000	< 1E-10	81,113
U-238	< 1E-10	10,000	2.1E-07	81,113

The maximum total dose for a member of the most-exposed population group is the same as under the nominal scenario, as are the radionuclides that contribute most (Pb-210 and Po-210). The maximum dose is therefore not dependent on the choice of model for the degradation of engineered barriers.

The maximum calculated dose for the period of 10,000 years after the closure of the repository is 0.014 mSv/year, half of that under the nominal scenario. Similarly to the nominal scenario, the largest contribution to the total dose again comes from Ca-41. Doses for Ag-108m are lower than under the nominal scenario, which is attributable to its shorter half-life in combination to its longer initial containment time in the silo (inside the engineered barriers), which follows from the alternate degradation model. Under this variant of the nominal scenario, it is Cl-36 that contributes most to the total dose in the first 10,000 years after the closure of the repository. Like under the nominal scenario, here the dominant pathway for radionuclides from the nearfield to a member of the most-exposed population group is again the ingestion of water from the well.

The results confirm that this model of degradation of engineered barriers is less conservative than the simultaneous model, albeit not significantly so. As stated above, some assumptions of this model are slightly uncertain, [75] and the model of simultaneous degradation of engineered barriers is therefore used in the nominal scenario.

7.3.6.1.2 Variant of nominal scenario without well

Under the nominal scenario it is assumed that a well is positioned in the middle of the contamination plume. The probability of this actually occurring can assumed to be very low. It is much more likely that a well is built that only partly intercepts the contamination plume, and is farther away from the repository. Calculations were made for the purposes of this scenario where the well would be completely excluded; the estimated doses derive solely from exposure to water from the Sava. These calculations allow for an assessment of the impact of the well in the nominal scenario and the potential reduction in doses that would result from a more realistic treatment of the well.

Releases of radionuclides from the nearfield and farfield of the repository are the same under this scenario as under the nominal scenario.

The calculated doses are presented below.

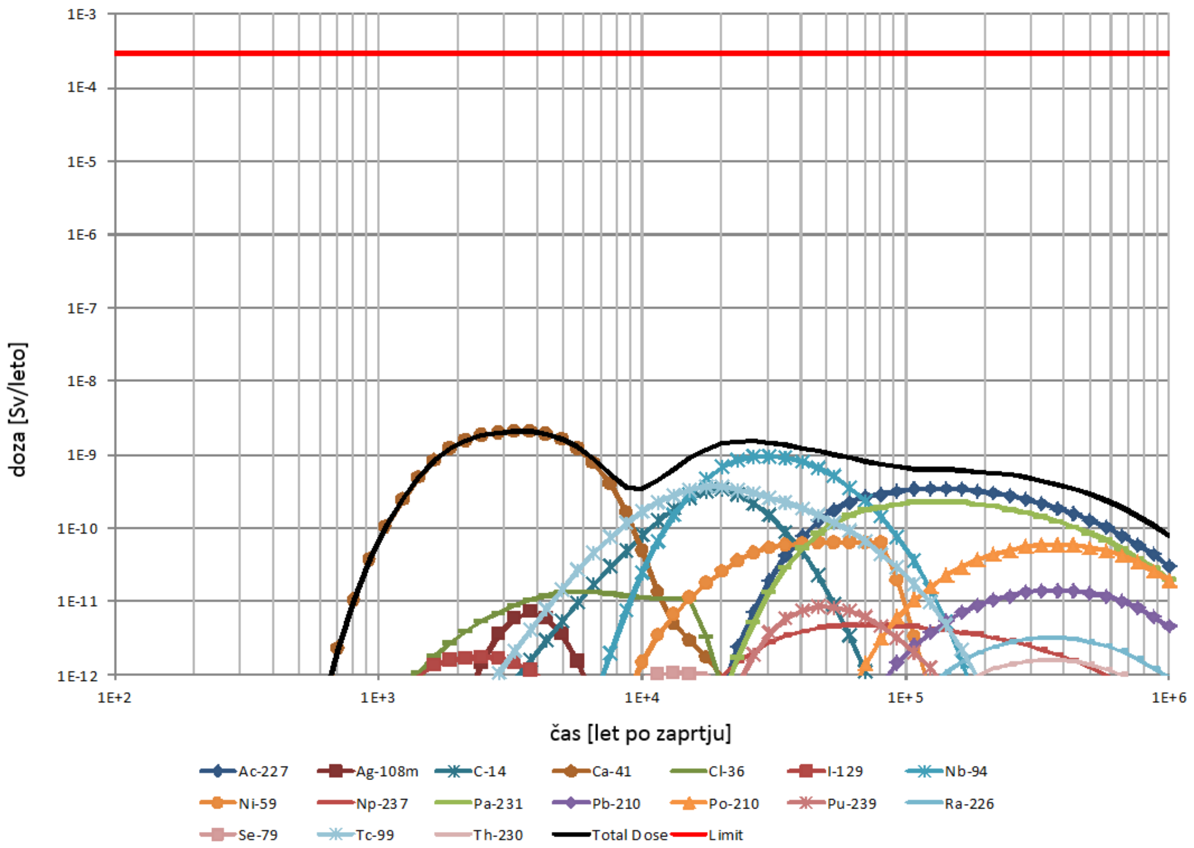

Figure 7.42: Annual dose for representative member of most-exposed population group under variant of nominal scenario without well (limits also shown for comparison)

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

Table 7.45: Maximum doses and time of occurrence for annual dose under variant of nominal scenario without well

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Total dose	2.1E-09	3,275	2.1E-09	3,275
Ac-227	< 1E-10	10,000	3.4E-10	123,285
Ag-108m	< 1E-10	3,765	< 1E-10	3,765
Am-241	< 1E-10	10,000	< 1E-10	13,219
Ba-133	< 1E-10	33	< 1E-10	33
C-14	< 1E-10	10,000	3.3E-10	20,092

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Ca-41	2.0E-09	3,275	2.0E-09	3,275
Cd-109	< 1E-10	57	< 1E-10	57
Cd-113m	< 1E-10	534	< 1E-10	534
Cl-36	< 1E-10	5,722	< 1E-10	5,722
Cm-244	< 1E-10	811	< 1E-10	811
Co-60	< 1E-10	231	< 1E-10	231
Cs-134	< 1E-10	100	< 1E-10	100
Cs-135	< 1E-10	10,000	< 1E-10	1,000,000
Cs-137	< 1E-10	1,874	< 1E-10	1,874
Eu-152	< 1E-10	614	< 1E-10	614
Eu-154	< 1E-10	404	< 1E-10	404
Eu-155	< 1E-10	231	< 1E-10	231
Fe-55	< 1E-10	115	< 1E-10	115
H-3	< 1E-10	305	< 1E-10	305
I-129	< 1E-10	2,477	< 1E-10	2,477
Na-22	< 1E-10	11	< 1E-10	11
Nb-93m	< 1E-10	534	< 1E-10	534
Nb-94	< 1E-10	10,000	9.5E-10	30,539
Ni-59	< 1E-10	10,000	< 1E-10	70,548
Ni-63	< 1E-10	3,275	< 1E-10	3,275
Np-237	< 1E-10	10,000	< 1E-10	70,548
Pa-231	< 1E-10	10,000	2.3E-10	123,285
Pb-210	< 1E-10	10,000	< 1E-10	376,494
Pd-107	< 1E-10	10,000	< 1E-10	26,561
Po-210	< 1E-10	10,000	< 1E-10	376,494

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Pu-238	< 1E-10	2,848	< 1E-10	2,848
Pu-239	< 1E-10	10,000	< 1E-10	46,416
Pu-240	< 1E-10	10,000	< 1E-10	35,112
Pu-241	< 1E-10	534	< 1E-10	534
Ra-226	< 1E-10	10,000	< 1E-10	376,494
Ra-228	< 1E-10	10,000	< 1E-10	1,000,000
Sb-125	< 1E-10	76	< 1E-10	76
Se-79	< 1E-10	10,000	< 1E-10	13,219
Sm-151	< 1E-10	3,275	< 1E-10	3,275
Sr-90	< 1E-10	1,417	< 1E-10	1,417
Tc-99	1.7E-10	10,000	3.6E-10	17,475
Th-228	< 1E-10	10,000	< 1E-10	1,000,000
Th-229	< 1E-10	10,000	< 1E-10	284,804
Th-230	< 1E-10	10,000	< 1E-10	327,455
Th-232	< 1E-10	10,000	< 1E-10	1,000,000
Tl-204	< 1E-10	16	< 1E-10	16
U-233	< 1E-10	10,000	< 1E-10	247,708
U-234	< 1E-10	10,000	< 1E-10	247,708
U-235	< 1E-10	10,000	< 1E-10	162,975
U-236	< 1E-10	10,000	< 1E-10	141,747
U-238	< 1E-10	10,000	< 1E-10	162,975

Table 7.46: Contribution of various exposure pathways to maximum dose under variant of nominal scenario without well

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External exposure	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Ac-227	3.6E-17	10,000	0%	0%	0%	73%	9%	17%	0%
Ag-108m	7.1E-12	3,765	0%	0%	86%	7%	2%	4%	1%
Am-241	1.9E-26	10,000	3%	1%	0%	64%	1%	0%	31%
Ba-133	1.1E-24	33	31%	0%	5%	61%	0%	1%	2%
C-14	8.1E-11	10,000	0%	0%	0%	2%	0%	1%	97%
Ca-41	2.0E-09	3,275	1%	0%	0%	90%	2%	6%	1%
Cd-109	3.9E-81	57	0%	0%	0%	7%	0%	0%	93%
Cd-113m	1.5E-44	534	0%	0%	0%	34%	0%	0%	66%
Cl-36	1.4E-11	5,722	10%	0%	0%	33%	10%	25%	23%
Cm-244	4.4E-91	811	0%	0%	0%	0%	0%	0%	99%
Co-60	7.1E-65	231	2%	0%	73%	18%	0%	0%	6%
Cs-134	1.8E-134	100	1%	0%	1%	5%	3%	2%	87%
Cs-135	2.5E-56	10,000	1%	0%	0%	6%	5%	4%	84%
Cs-137	3.9E-95	1,874	1%	0%	4%	6%	4%	3%	83%

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External exposure	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Eu-152	2.0E-99	614	0%	0%	16%	54%	10%	20%	1%
Eu-154	1.9E-106	404	0%	0%	13%	54%	11%	21%	1%
Eu-155	3.0E-116	231	0%	0%	2%	58%	13%	25%	2%
Fe-55	5.0E-65	115	8%	0%	0%	14%	5%	0%	72%
H-3	3.9E-20	305	46%	0%	0%	26%	7%	19%	2%
I-129	1.7E-12	2,477	14%	0%	0%	38%	7%	19%	21%
Na-22	1.9E-26	11	10%	0%	11%	16%	7%	19%	38%
Nb-93m	2.6E-53	534	0%	0%	0%	4%	0%	0%	95%
Nb-94	2.4E-11	10,000	0%	0%	64%	2%	0%	0%	34%
Ni-59	1.4E-12	10,000	5%	0%	0%	49%	35%	5%	6%
Ni-63	5.7E-27	3,275	6%	0%	0%	48%	34%	5%	6%
Np-237	2.1E-14	10,000	0%	0%	0%	0%	0%	0%	99%
Pa-231	2.8E-17	10,000	0%	0%	0%	58%	7%	13%	21%
Pb-210	8.1E-25	10,000	1%	0%	0%	94%	2%	2%	1%
Pd-107	9.4E-17	10,000	0%	0%	0%	12%	2%	4%	82%

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External exposure	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Po-210	3.6E-24	10,000	12%	0%	0%	38%	28%	1%	21%
Pu-238	5.0E-38	2,848	0%	0%	0%	4%	0%	0%	95%
Pu-239	7.7E-17	10,000	0%	0%	0%	4%	0%	0%	95%
Pu-240	1.8E-19	10,000	0%	0%	0%	4%	0%	0%	95%
Pu-241	3.3E-57	534	0%	0%	0%	4%	0%	0%	96%
Ra-226	1.1E-26	10,000	17%	0%	11%	63%	3%	2%	3%
Ra-228	5.0E-33	10,000	17%	0%	4%	70%	3%	2%	3%
Sb-125	2.9E-36	76	12%	0%	35%	30%	1%	0%	22%
Se-79	9.5E-13	10,000	0%	0%	0%	53%	6%	4%	37%
Sm-151	5.2E-43	3,275	0%	0%	0%	57%	7%	15%	21%
Sr-90	4.3E-37	1,417	19%	0%	0%	64%	3%	11%	3%
Tc-99	1.7E-10	10,000	0%	0%	0%	1%	0%	0%	98%
Th-228	5.1E-34	10,000	0%	2%	62%	6%	0%	30%	0%
Th-229	5.9E-21	10,000	0%	4%	5%	6%	0%	84%	0%
Th-230	1.8E-29	10,000	1%	2%	0%	27%	0%	70%	0%

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External exposure	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Th-232	5.6E-34	10,000	0%	4%	0%	16%	0%	79%	0%
Tl-204	1.6E-25	16	1%	0%	0%	24%	6%	11%	58%
U-233	4.0E-21	10,000	11%	3%	0%	68%	1%	17%	1%
U-234	9.5E-27	10,000	21%	0%	0%	65%	1%	12%	1%
U-235	2.8E-25	10,000	18%	1%	9%	60%	1%	11%	1%
U-236	1.8E-26	10,000	20%	1%	0%	65%	1%	12%	1%
U-238	1.7E-25	10,000	21%	0%	1%	64%	1%	12%	1%

The maximum calculated dose occurs around 3,000 years after the closure of the repository, and amounts to 2 × 10^{-6} mSv/year. Over the entire period the calculated doses are much lower (by several orders of magnitude) than the doses calculated for the nominal scenario, which is an indication of the dominant contribution by the ingestion of water from the well under the nominal scenario. The above table illustrating the contributions of various exposure pathways to the total dose shows that the contribution varies for different radionuclides. For example, for Ca-41 the main contribution is from the ingestion of crops from the irrigated area, for C-14 it is the ingestion of aquatic animals from the river, and for Ag-108m it is external exposure to radionuclides accumulated in the soil from irrigation.

7.3.6.1.3 Variant of nominal scenario with conservative assumption for use of well

Wells are not typical for the area of the LILW repository, primarily on account of the unfavourable hydrological conditions. The use of a well for drinking water has nevertheless been taken into account in the nominal scenario. The use of this water (from the well) for irrigation and for watering livestock was assessed as unlikely (mainly because of the proximity of the river), and was therefore not taken into account in the nominal scenario. To estimate the potential impact of these excluded uses, a variant of the nominal scenario with a conservative assumption for the use of the well for drawing water was drawn up, where the water from the well is used for irrigating field crops and watering beef cattle. The results of the calculations of the two sub-scenarios (albeit highly unlikely) are presented below.

7.3.6.1.3.1 Use of well water for irrigating field crops and vegetables

All assumptions in this sub-scenario are the same as for the nominal scenario, except that water from the well is used for irrigating field crops and vegetables in addition to being used for drinking water. The calculation was made for the case of a small field being irrigated with water from the well and used for the cultivation of vegetables. It was assumed that all the vegetables (137 kg/year) ingested by a member of the most-exposed population group are produced in the aforementioned field, and that he/she spends 500 hours a year in an area that is contaminated.

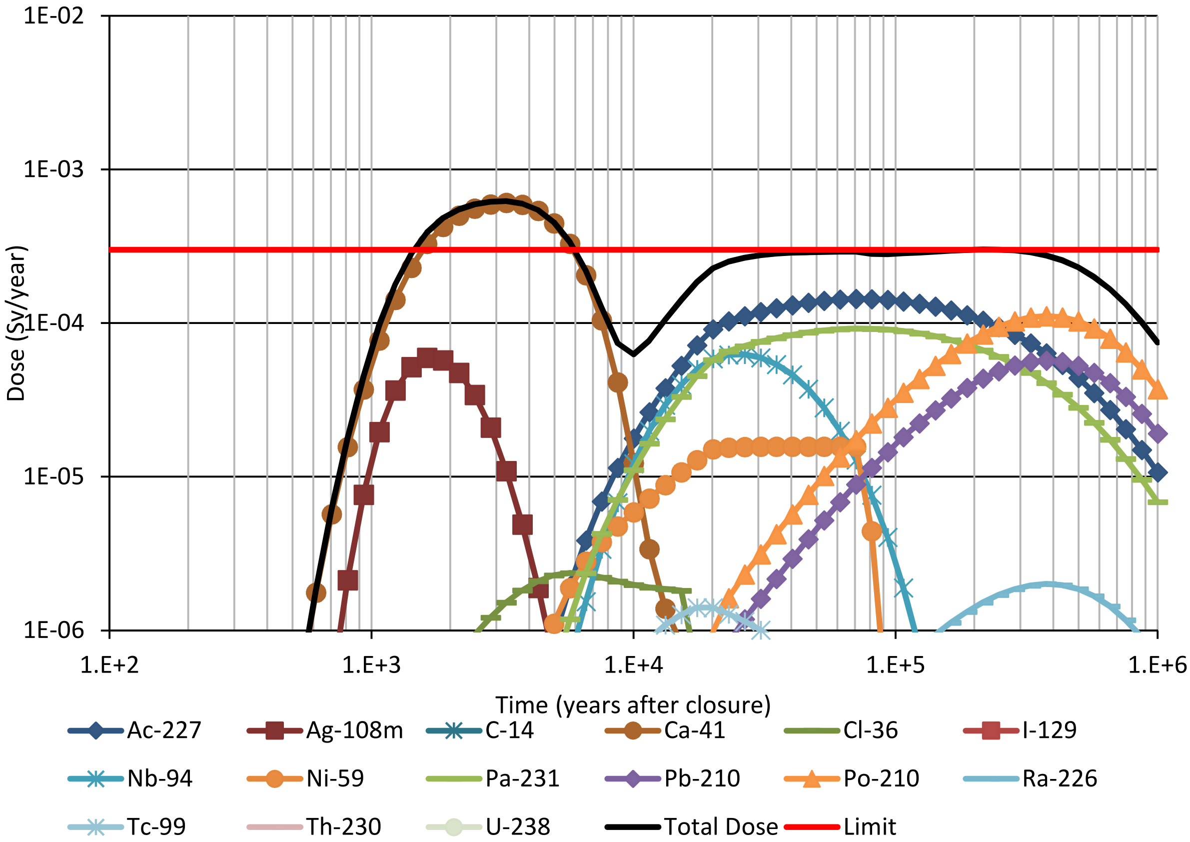

Figure 7.43: Annual dose for representative member of most-exposed population group under variant of nominal scenario where well water is used for irrigation (limits also shown for comparison)

Table 7.47: Maximum doses and time of occurrence for annual dose under nominal sub-scenario of use of well water for irrigating field crops and vegetables

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Total dose	6.2E-04	3,275	6.2E-04	3,275
Ac-227	1.8E-05	10,000	1.4E-04	70,548
Ag-108m	6.0E-05	1,630	6.0E-05	1,630
Am-241	< 1E-10	6,579	< 1E-10	6,579
Ba-133	< 1E-10	305	< 1E-10	305
C-14	2.2E-07	10,000	9.0E-07	20,092
Ca-41	6.1E-04	3,275	6.1E-04	3,275
Cd-109	< 1E-10	57	< 1E-10	57
Cd-113m	< 1E-10	305	< 1E-10	305
Cl-36	2.4E-06	5,722	2.4E-06	5,722
Cm-244	< 1E-10	351	< 1E-10	351
Co-60	< 1E-10	305	< 1E-10	305
Cs-134	< 1E-10	305	< 1E-10	305
Cs-135	< 1E-10	10,000	9.0E-10	141,747
Cs-137	< 1E-10	933	< 1E-10	933
Eu-152	< 1E-10	351	< 1E-10	351
Eu-154	< 1E-10	305	< 1E-10	305
Eu-155	< 1E-10	305	< 1E-10	305
Fe-55	< 1E-10	305	< 1E-10	305
H-3	< 1E-10	305	< 1E-10	305
I-129	6.3E-07	2,154	6.3E-07	2,154
Na-22	< 1E-10	11	< 1E-10	11
Nb-93m	< 1E-10	305	< 1E-10	305

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Nb-94	1.2E-05	10,000	6.3E-05	26,561
Ni-59	5.9E-06	10,000	1.6E-05	70,548
Ni-63	< 1E-10	1,630	< 1E-10	1,630
Np-237	< 1E-10	10,000	7.4E-09	70,548
Pa-231	1.1E-05	10,000	9.2E-05	70,548
Pb-210	4.3E-08	10,000	5.7E-05	376,494
Pd-107	< 1E-10	10,000	< 1E-10	20,092
Po-210	8.3E-08	10,000	1.1E-04	376,494
Pu-238	< 1E-10	1,417	< 1E-10	1,417
Pu-239	2.9E-09	10,000	4.9E-07	40,370
Pu-240	< 1E-10	10,000	6.5E-10	26,561
Pu-241	< 1E-10	305	< 1E-10	305
Ra-226	1.2E-09	10,000	2.0E-06	376,494
Ra-228	< 1E-10	10,000	3.7E-08	572,237
Sb-125	< 1E-10	76	< 1E-10	76
Se-79	1.8E-07	10,000	1.8E-07	10,000
Sm-151	< 1E-10	1,630	< 1E-10	1,630
Sr-90	< 1E-10	705	< 1E-10	705
Tc-99	6.5E-07	10,000	1.4E-06	17,475
Th-228	< 1E-10	10,000	8.4E-10	657,933
Th-229	< 1E-10	10,000	9.7E-10	247,708
Th-230	< 1E-10	10,000	4.2E-08	432,876
Th-232	< 1E-10	10,000	9.7E-10	572,237
Tl-204	< 1E-10	14	< 1E-10	14
U-233	< 1E-10	10,000	1.6E-09	215,443

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
U-234	< 1E-10	10,000	1.4E-07	247,708
U-235	< 1E-10	10,000	3.4E-09	93,260
U-236	< 1E-10	10,000	< 1E-10	81,113
U-238	< 1E-10	10,000	3.6E-07	81,113

Table 7.48: Contribution of various exposure pathways to maximum dose under nominal sub-scenario of use of well water for irrigating field crops and vegetables

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External irradiation	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Total dose	6.2E-04	3,275	4%	0%	1%	95%	0%	0%	0%
Ac-227	1.8E-05	10,000	1%	0%	0%	99%	0%	0%	0%
Ag-108m	6.0E-05	1,630	26 %	0%	60%	14%	0%	0%	0%
Am-241	2.1E-13	6,579	65 %	0%	0%	35%	0%	0%	0%
Ba-133	1.3E-20	305	65 %	0%	1%	35%	0%	0%	0%
C-14	2.2E-07	10,000	3%	0%	0%	97%	0%	0%	0%
Ca-41	6.1E-04	3,275	3%	0%	0%	97%	0%	0%	0%
Cd-109	3.6E-81	57	0%	0%	0%	1%	0%	0%	99%
Cd-113m	8.6E-23	305	52 %	0%	0%	48%	0%	0%	0%
Cl-36	2.4E-06	5,722	59 %	0%	0%	41%	0%	0%	0%

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External irradiation	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Cm-244	1.4E-41	351	65 %	0%	0%	35%	0%	0%	0%
Co-60	1.6E-34	305	55 %	0%	13%	32%	0%	0%	0%
Cs-134	2.9E-109	305	65 %	0%	1%	34%	0%	0%	0%
Cs-135	4.8E-18	10,000	64 %	0%	0%	36%	0%	0%	0%
Cs-137	6.3E-37	933	62 %	0%	4%	34%	0%	0%	0%
Eu-152	3.2E-44	351	4%	0%	4%	92%	0%	0%	0%
Eu-154	7.6E-49	305	13 %	0%	3%	84%	0%	0%	0%
Eu-155	1.1E-58	305	13 %	0%	0%	86%	0%	0%	0%
Fe-55	5.9E-55	305	63 %	0%	0%	37%	0%	0%	0%
H-3	2.8E-14	305	83 %	0%	0%	17%	0%	0%	0%
I-129	6.3E-07	2,154	60 %	0%	0%	40%	0%	0%	0%
Na-22	1.5E-26	11	12 %	0%	1%	7%	8%	24%	47%
Nb-93m	2.4E-34	305	65 %	0%	0%	35%	0%	0%	0%
Nb-94	1.2E-05	10,000	16 %	0%	75%	9%	0%	0%	0%
Ni-59	5.9E-06	10,000	64 %	0%	0%	36%	0%	0%	0%

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External irradiation	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Ni-63	3.0E-13	1,630	65 %	0%	0%	35%	0%	0%	0%
Np-237	4.3E-11	10,000	64 %	0%	0%	36%	0%	0%	0%
Pa-231	1.1E-05	10,000	1%	0%	0%	99%	0%	0%	0%
Pb-210	4.3E-08	10,000	65 %	0%	0%	35%	0%	0%	0%
Pd-107	9.2E-13	10,000	63 %	0%	0%	37%	0%	0%	0%
Po-210	8.3E-08	10,000	67 %	0%	0%	33%	0%	0%	0%
Pu-238	2.1E-23	1,417	65 %	0%	0%	35%	0%	0%	0%
Pu-239	2.9E-09	10,000	65 %	0%	0%	35%	0%	0%	0%
Pu-240	9.9E-12	10,000	65 %	0%	0%	35%	0%	0%	0%
Pu-241	4.3E-31	305	65 %	0%	0%	35%	0%	0%	0%
Ra-226	1.2E-09	10,000	63 %	0%	2%	36%	0%	0%	0%
Ra-228	1.5E-22	10,000	65 %	0%	0%	35%	0%	0%	0%
Sb-125	1.3E-36	76	25 %	0%	4%	21%	2%	0%	47%
Se-79	1.8E-07	10,000	1%	0%	0%	99%	0%	0%	0%
Sm-151	2.2E-25	1,630	2%	0%	0%	98%	0%	0%	0%

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External irradiation	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Sr-90	3.2E-16	705	59 %	0%	0%	41%	0%	0%	0%
Tc-99	6.5E-07	10,000	25 %	0%	0%	75%	0%	0%	0%
Th-228	1.6E-24	10,000	49 %	1%	24%	26%	0%	0%	0%
Th-229	8.1E-16	10,000	59 %	4%	4%	34%	0%	0%	0%
Th-230	4.7E-15	10,000	64 %	1%	0%	35%	0%	0%	0%
Th-232	3.6E-24	10,000	64 %	1%	0%	35%	0%	0%	0%
Tl-204	1.3E-25	14	2%	0%	0%	8%	6%	12%	72%
U-233	2.0E-14	10,000	63 %	0%	0%	37%	0%	0%	0%
U-234	2.7E-12	10,000	64 %	0%	0%	36%	0%	0%	0%
U-235	4.5E-13	10,000	63 %	0%	1%	36%	0%	0%	0%
U-236	1.5E-16	10,000	64 %	0%	0%	36%	0%	0%	0%
U-238	5.0E-11	10,000	64 %	0%	0%	36%	0%	0%	0%

The contribution of various exposure pathways to the maximum dose differs for different radionuclides. For all radionuclides other than Ca-41, the doses are below the prescribed limit of 0.3 mSv/year throughout the simulation. The maximum concentration of Ca-41 reaches a value approximately double that of the prescribed limit 3,000 years after closure of the repository. It should be noted that the calculation of releases of Ca-41 did not take account of the presence of large quantities of stable calcium in the concrete, which could result in the immobilisation of Ca-41 via exchange with stable calcium, which means that the calculated doses are substantially overstated. The phenomenon of calcium sorption is being investigated, and will be examined in more detail in the next phases of the project, when it will also be included in the safety analysis and the safety report. For this phase the conservative assumption was that there is no exchange. It is not reasonable to use the doses calculated for this variant of the scenario for the purpose of demonstrating compliance with the prescribed limit, as the probability of the occurrence of this scenario is extremely low.

7.3.6.1.3.2 Use of well water for watering livestock

In this sub-variant of the scenario all the assumptions are the same as in the nominal scenario, except that the water from the well is used as drinking water for people, and as water for livestock. This means that livestock drink water from the well instead of water from the river. The results are presented below.

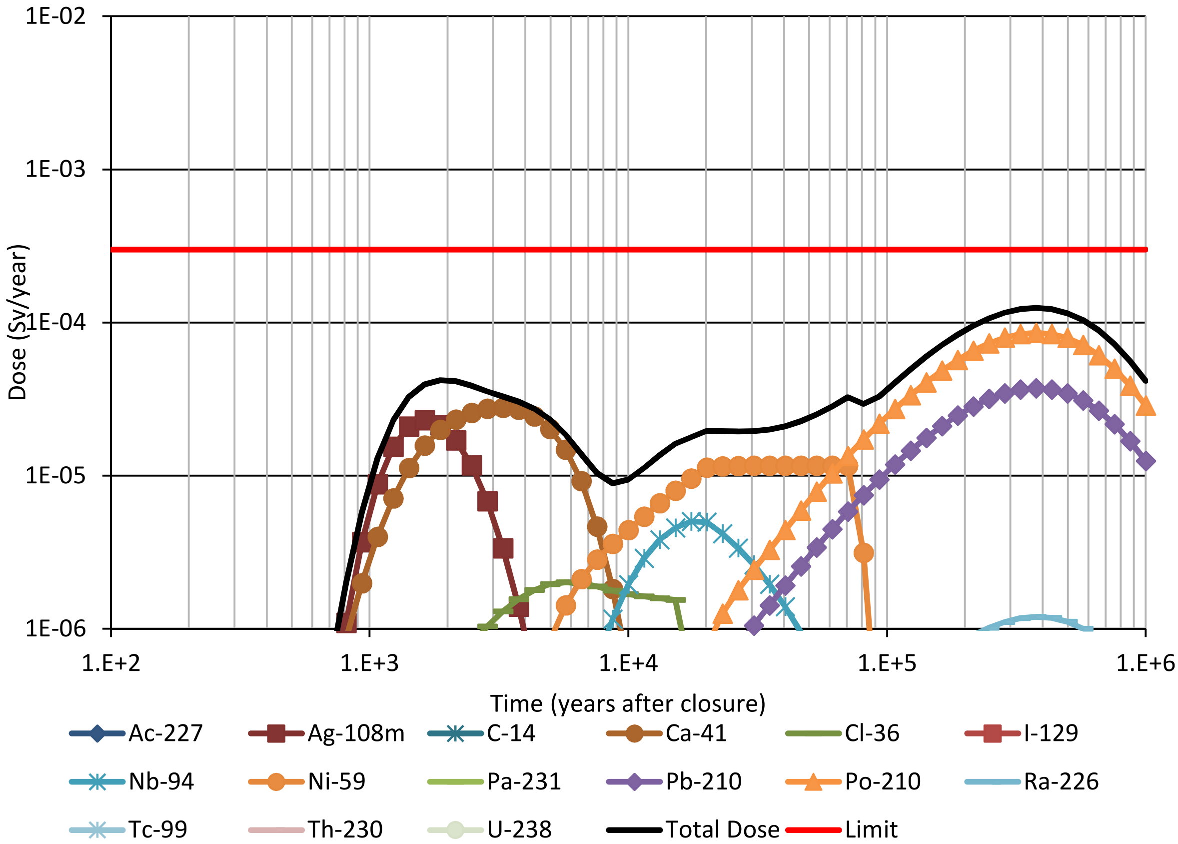

Figure 7.44: Annual dose for representative member of most-exposed population group under variant of nominal scenario where well water is also used for watering livestock (limits also shown for comparison)

Table 7.49: Maximum doses and time of occurrence for annual dose under nominal sub-scenario of use of well water for watering livestock

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Total dose	4.2E-05	1,874	1.2E-04	376,494
Ac-227	2.6E-07	10,000	7.4E-07	20,092
Ag-108m	2.3E-05	1,630	2.3E-05	1,630
Am-241	< 1E-10	6,579	< 1E-10	6,579
Ba-133	< 1E-10	305	< 1E-10	305
C-14	1.0E-07	10,000	4.3E-07	20,092
Ca-41	2.8E-05	3,275	2.8E-05	3,275
Cd-109	< 1E-10	57	< 1E-10	57
Cd-113m	< 1E-10	305	< 1E-10	305
Cl-36	2.0E-06	5,722	2.0E-06	5,722
Cm-244	< 1E-10	351	< 1E-10	351
Co-60	< 1E-10	305	< 1E-10	305
Cs-134	< 1E-10	305	< 1E-10	305
Cs-135	< 1E-10	10,000	5.7E-10	123,285
Cs-137	< 1E-10	933	< 1E-10	933
Eu-152	< 1E-10	305	< 1E-10	305
Eu-154	< 1E-10	305	< 1E-10	305
Eu-155	< 1E-10	305	< 1E-10	305
Fe-55	< 1E-10	305	< 1E-10	305
H-3	< 1E-10	305	< 1E-10	305
I-129	4.5E-07	2,154	4.5E-07	2,154
Na-22	< 1E-10	11	< 1E-10	11
Nb-93m	< 1E-10	305	< 1E-10	305

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Nb-94	1.9E-06	10,000	5.0E-06	17,475
Ni-59	4.4E-06	10,000	1.2E-05	70,548
Ni-63	< 1E-10	1,630	< 1E-10	1,630
Np-237	< 1E-10	10,000	7.0E-09	70,548
Pa-231	1.5E-07	10,000	4.1E-07	20,092
Pb-210	2.8E-08	10,000	3.7E-05	376,494
Pd-107	< 1E-10	10,000	< 1E-10	20,092
Po-210	6.4E-08	10,000	8.6E-05	376,494
Pu-238	< 1E-10	1,417	< 1E-10	1,417
Pu-239	1.9E-09	10,000	3.2E-07	40,370
Pu-240	< 1E-10	10,000	4.2E-10	26,561
Pu-241	< 1E-10	305	< 1E-10	305
Ra-226	7.5E-10	10,000	1.2E-06	376,494
Ra-228	< 1E-10	10,000	2.3E-08	572,237
Sb-125	< 1E-10	76	< 1E-10	76
Se-79	3.3E-09	10,000	3.3E-09	10,000
Sm-151	< 1E-10	1,630	< 1E-10	1,630
Sr-90	< 1E-10	705	< 1E-10	705
Tc-99	2.4E-07	10,000	5.2E-07	17,475
Th-228	< 1E-10	10,000	2.5E-10	572,237
Th-229	< 1E-10	10,000	5.4E-10	215,443
Th-230	< 1E-10	10,000	3.1E-08	432,876
Th-232	< 1E-10	10,000	7.9E-10	572,237
Tl-204	< 1E-10	16	< 1E-10	16
U-233	< 1E-10	10,000	9.2E-10	215,443

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
U-234	< 1E-10	10,000	7.9E-08	247,708
U-235	< 1E-10	10,000	1.9E-09	81,113
U-236	< 1E-10	10,000	< 1E-10	81,113
U-238	< 1E-10	10,000	2.2E-07	81,113

Table 7.50: Contribution of various exposure pathways to maximum dose under nominal sub-scenario of use of well water for watering livestock

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External irradiation	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Total dose	4.2E-05	1,874	71%	0%	0%	0%	8%	20%	0%
Ac-227	2.6E-07	10,000	68%	0%	0%	0%	10%	22%	0%
Ag-108m	2.3E-05	1,630	68%	0%	0%	0%	10%	22%	0%
Am-241	1.4E-13	6,579	100 %	0%	0%	0%	0%	0%	0%
Ba-133	8.4E-21	305	100 %	0%	0%	0%	0%	0%	0%
C-14	1.0E-07	10,000	7%	0%	0%	0%	41%	52%	0%
Ca-41	2.8E-05	3,275	75%	0%	0%	0%	6%	19%	0%
Cd-109	3.9E-81	57	0%	0%	0%	7%	0%	0%	93%
Cd-113m	4.7E-23	305	96%	0%	0%	0%	4%	0%	0%
Cl-36	2.0E-06	5,722	70%	0%	0%	0%	8%	23%	0%
Cm-244	1.4E-41	351	68%	0%	0%	0%	10%	22%	0%
Co-60	9.0E-35	305	99%	0%	0%	0%	0%	0%	0%

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External irradiation	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Cs-134	7.2E-109	305	79%	0%	0%	0%	12%	9%	0%
Cs-135	3.8E-18	10,000	79%	0%	0%	0%	12%	9%	0%
Cs-137	4.9E-37	933	79%	0%	0%	0%	12%	9%	0%
Eu-152	3.5E-45	305	68%	0%	0%	0%	10%	22%	0%
Eu-154	1.4E-49	305	68%	0%	0%	0%	10%	22%	0%
Eu-155	2.1E-59	305	68%	0%	0%	0%	10%	22%	0%
Fe-55	4.1E-55	305	91%	0%	0%	0%	9%	0%	0%
H-3	3.5E-14	305	67%	0%	0%	0%	9%	24%	0%
I-129	4.5E-07	2,154	85%	0%	0%	0%	4%	11%	0%
Na-22	1.9E-26	11	10%	0%	11 %	16%	7%	19%	38%
Nb-93m	1.5E-34	305	100 %	0%	0%	0%	0%	0%	0%
Nb-94	1.9E-06	10,000	100 %	0%	0%	0%	0%	0%	0%
Ni-59	4.4E-06	10,000	85%	0%	0%	0%	13%	2%	0%
Ni-63	2.3E-13	1,630	85%	0%	0%	0%	13%	2%	0%
Np-237	4.0E-11	10,000	68%	0%	0%	0%	10%	22%	0%
Pa-231	1.5E-07	10,000	68%	0%	0%	0%	10%	22%	0%
Pb-210	2.8E-08	10,000	99%	0%	0%	0%	0%	0%	0%
Pd-107	8.5E-13	10,000	68%	0%	0%	0%	10%	22%	0%
Po-210	6.4E-08	10,000	87%	0%	0%	0%	13%	0%	0%
Pu-238	1.3E-23	1,417	100 %	0%	0%	0%	0%	0%	0%

Radionuclide	Maximum dose (Sv/year)	Time (years)	Drinking water	Inhalation	External irradiation	Ingestion of crops	Ingestion of meat	Ingestion of milk	Ingestion of fish (aquatic animals)
Pu-239	1.9E-09	10,000	100 %	0%	0%	0%	0%	0%	0%
Pu-240	6.4E-12	10,000	100 %	0%	0%	0%	0%	0%	0%
Pu-241	2.8E-31	305	100 %	0%	0%	0%	0%	0%	0%
Ra-226	7.5E-10	10,000	98%	0%	0%	0%	1%	1%	0%
Ra-228	1.0E-22	10,000	98%	0%	0%	0%	1%	1%	0%
Sb-125	2.9E-36	76	12%	0%	35 %	30%	1%	0%	22%
Se-79	3.3E-09	10,000	80%	0%	0%	0%	12%	8%	0%
Sm-151	6.5E-27	1,630	68%	0%	0%	0%	10%	22%	0%
Sr-90	1.9E-16	705	96%	0%	0%	0%	1%	3%	0%
Tc-99	2.4E-07	10,000	68%	0%	0%	0%	10%	22%	0%
Th-228	1.1E-24	10,000	75%	0%	0%	0%	0%	24%	0%
Th-229	6.3E-16	10,000	75%	0%	0%	0%	0%	25%	0%
Th-230	4.0E-15	10,000	75%	0%	0%	0%	0%	24%	0%
Th-232	3.1E-24	10,000	75%	0%	0%	0%	0%	24%	0%
Tl-204	1.6E-25	16	1%	0%	0%	24%	6%	11%	58%
U-233	1.3E-14	10,000	95%	0%	0%	0%	0%	4%	0%
U-234	1.8E-12	10,000	95%	0%	0%	0%	0%	4%	0%
U-235	3.0E-13	10,000	95%	0%	0%	0%	0%	4%	0%
U-236	1.0E-16	10,000	95%	0%	0%	0%	0%	4%	0%
U-238	3.4E-11	10,000	95%	0%	0%	0%	0%	4%	0%

The total dose of all radionuclides is below the prescribed limit of 0.3 mSv/year for the entire period of the simulation.

7.3.6.2 Scenario of early failure of engineered barriers

The scenario of the early failure of engineered barriers is evaluated in the same way as the nominal scenario, with the difference that individual (built) components of the engineered barriers are subject to rapid degradation with regard to their prescribed and required properties, immediately after the end of institutional controls. This scenario does not solely comprise faults and failures in the concrete, but also includes accelerated corrosion and other degradation processes. As such it could be named the scenario of worst-case evolution, where all safety functions for the nearfield of the repository related to the containment and mitigation of potential releases fail. The scenario is not directly related to any specific FEP, but is included as a marginal case. As such it could be treated as analysis without barriers, and not as a genuine scenario.

It is assumed that a certain initial event could occur at a time between 300 and 10,000 years after the closure of the repository. Under the conservative approach, it is assumed in the evaluation of the scenario that there is an event immediately after the end of institutional controls, i.e. 300 years after the closure of the repository. At the time when the event occurs, the properties of the engineered barriers quickly move from their initial values to a degraded state. The high speed means that the first stage of the change in the behaviour of the engineered barriers occurs over the course of one year. All other properties of the system are the same as under the nominal scenario.

The results of the calculations are illustrated below.

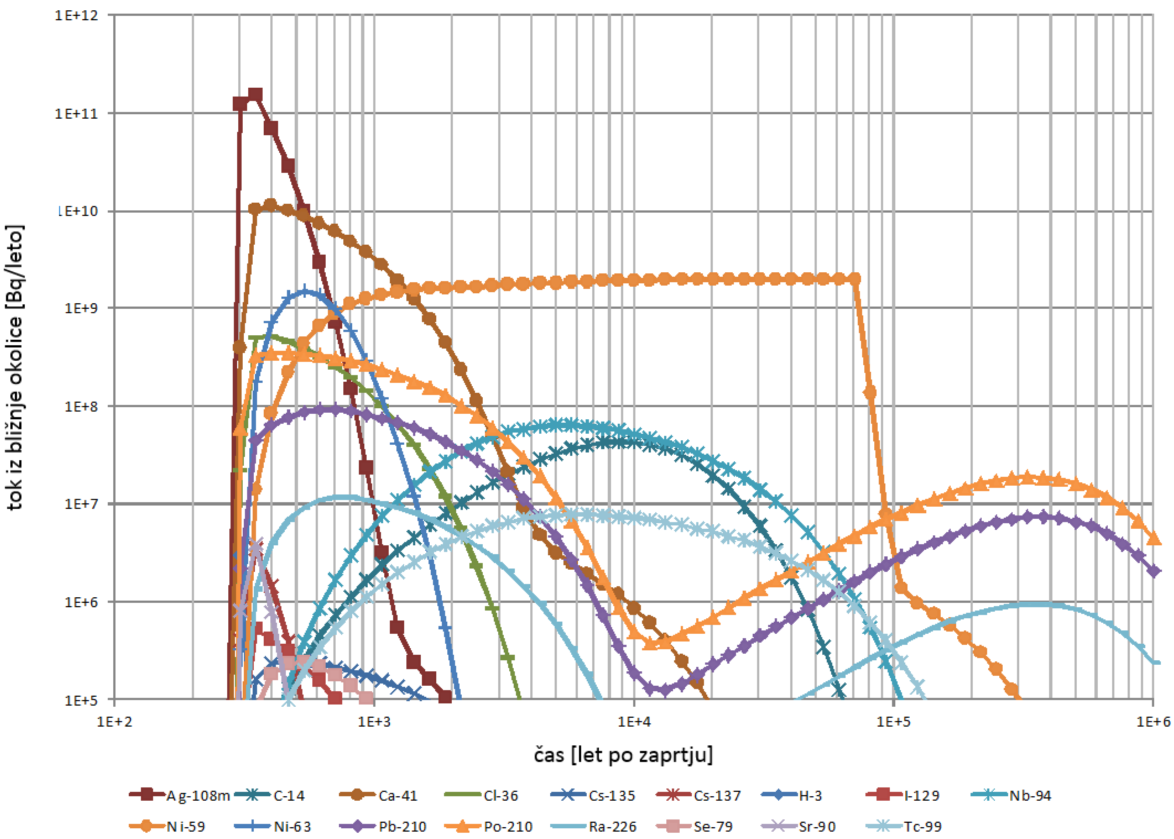

Figure 7.45: Releases of radionuclides from nearfield to farfield of repository under scenario of rapid failure of engineered barriers; it is assumed that the failure occurs immediately after the end of institutional controls (300 years after the closure of the repository)

tok iz bližnje okolice [Bq/leto]	flow from nearfield [Bq/year]
čas [let po zaprtju]	period [years after closure]

Table 7.51: Maximum releases and time of occurrence for releases from silo under scenario of rapid failure of engineered barriers

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Total release	1.6E+11	351	1.6E+11	351
Ac-227	1.0E+04	351	1.0E+04	351
Ag-108m	1.5E+11	351	1.5E+11	351
Am-241	3.4E+04	1,874	3.4E+04	1,874
Ba-133	2.1E+00	305	2.1E+00	305
C-14	4.3E+07	8,697	4.3E+07	8,697

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Ca-41	1.1E+10	404	1.1E+10	404
Cd-109	9.7E-10	2	9.7E-10	2
Cd-113m	1.5E+01	351	1.5E+01	351
Cl-36	5.2E+08	404	5.2E+08	404
Cm-244	1.5E-04	351	1.5E-04	351
Co-60	4.6E-02	12	4.6E-02	12
Cs-134	1.1E-02	3	1.1E-02	3
Cs-135	2.5E+05	464	2.5E+05	464
Cs-137	3.4E+06	351	3.4E+06	351
Eu-152	2.8E-06	351	2.8E-06	351
Eu-154	2.9E-09	22	2.9E-09	22
Eu-155	4.4E-10	11	4.4E-10	11
Fe-55	6.2E+02	4	6.2E+02	4
H-3	2.9E+06	305	2.9E+06	305
I-129	5.2E+05	351	5.2E+05	351
Na-22	4.0E-08	5	4.0E-08	5
Nb-93m	8.3E-11	351	8.3E-11	351
Nb-94	6.4E+07	4,977	6.4E+07	4,977
Ni-59	1.9E+09	10,000	2.0E+09	70,548
Ni-63	1.5E+09	534	1.5E+09	534
Np-237	1.9E+02	10,000	5.7E+02	53,367
Pa-231	1.1E+04	351	1.1E+04	351
Pb-210	9.1E+07	705	9.1E+07	705
Pd-107	1.4E+03	4,977	1.4E+03	4,977
Po-210	3.5E+08	404	3.5E+08	404

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Pu-238	3.7E-02	705	3.7E-02	705
Pu-239	1.2E+04	10,000	2.4E+04	26,561
Pu-240	8.5E+01	10,000	9.2E+01	13,219
Pu-241	5.7E-08	351	5.7E-08	351
Ra-226	1.1E+07	705	1.1E+07	705
Ra-228	5.9E+02	10,000	2.1E+04	327,455
Sb-125	6.0E-02	4	6.0E-02	4
Se-79	2.4E+05	534	2.4E+05	534
Sm-151	2.4E-01	705	2.4E-01	705
Sr-90	3.9E+06	351	3.9E+06	351
Tc-99	7.8E+06	5,722	7.8E+06	5,722
Th-228	9.9E-01	10,000	3.5E+01	327,455
Th-229	3.0E-01	10,000	3.7E+01	215,443
Th-230	4.4E+00	10,000	2.7E+03	376,494
Th-232	4.4E-01	10,000	3.5E+01	327,455
Tl-204	7.2E-08	8	7.2E-08	8
U-233	7.4E+00	10,000	2.2E+02	187,382
U-234	7.9E+02	10,000	1.9E+04	215,443
U-235	1.4E+02	10,000	5.2E+02	61,359
U-236	5.0E-02	10,000	2.9E-01	61,359
U-238	1.6E+04	10,000	6.1E+04	61,359

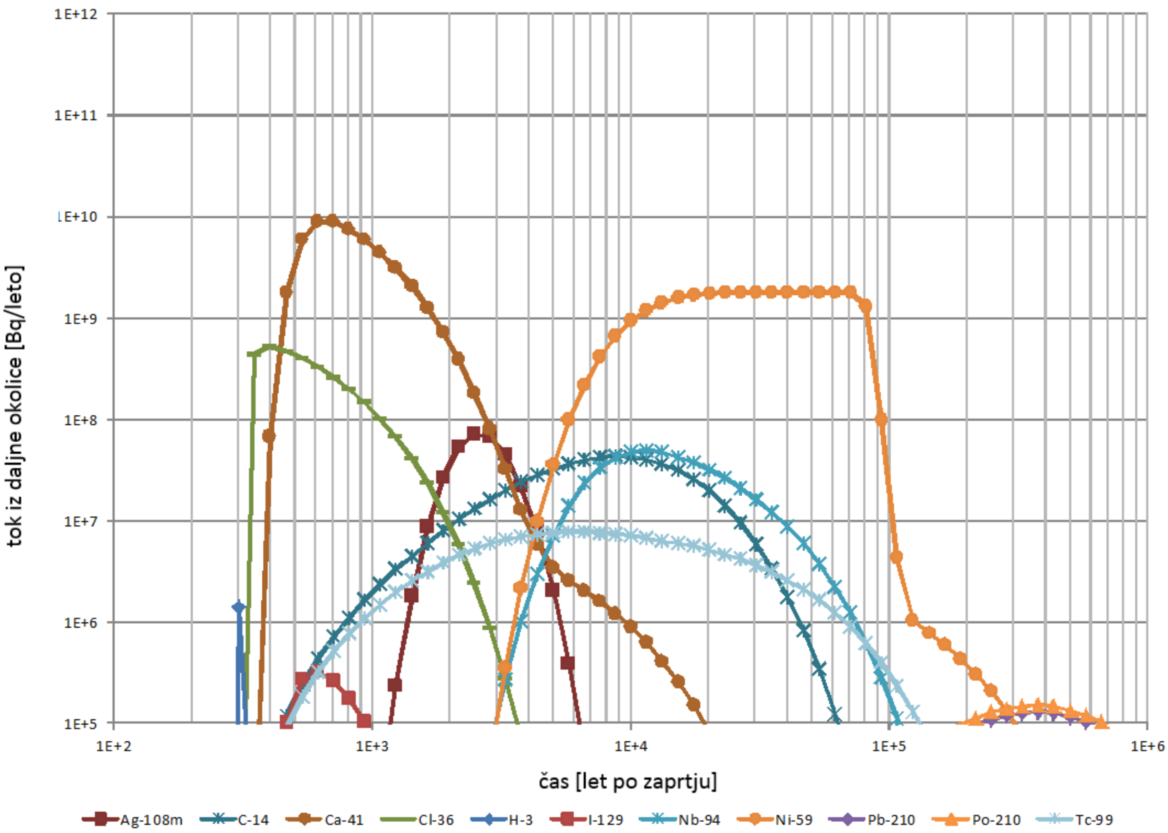

Figure 7.46: Releases of radionuclides from geosphere (drainage into river) under scenario of rapid failure of engineered barriers; it is assumed that the failure occurs immediately after the end of institutional controls (300 years after the closure of the repository)

tok iz daljne okolice [Bq/leto]	flow from farfield [Bq/year]
čas [let po zaprtju]	period [years after closure]

Table 7.52: Maximum releases and time of occurrence for releases from farfield model under scenario of rapid failure of engineered barriers

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Total release	9.1E+09	705	9.1E+09	705
Ac-227	6.9E-02	10,000	2.3E+03	81,113
Ag-108m	7.2E+07	2,477	7.2E+07	2,477
Am-241	2.7E-05	10,000	2.7E-05	10,000
Ba-133	1.5E-01	351	1.5E-01	351
C-14	4.3E+07	8,697	4.3E+07	8,697

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Ca-41	8.9E+09	705	8.9E+09	705
Cd-109	2.5E-64	57	2.5E-64	57
Cd-113m	9.3E-25	811	9.3E-25	811
Cl-36	5.3E+08	404	5.3E+08	404
Cm-244	1.8E-72	1,233	1.8E-72	1,233
Co-60	2.8E-47	231	2.8E-47	231
Cs-134	3.9E-118	100	3.9E-118	100
Cs-135	1.3E-35	10,000	4.0E+02	869,749
Cs-137	9.9E-74	1,630	9.9E-74	1,630
Eu-152	1.9E-80	933	1.9E-80	933
Eu-154	1.5E-89	404	1.5E-89	404
Eu-155	2.6E-98	231	2.6E-98	231
Fe-55	1.6E-46	115	1.6E-46	115
H-3	1.4E+06	305	1.4E+06	305
I-129	3.3E+05	614	3.3E+05	614
Na-22	7.5E-09	11	7.5E-09	11
Nb-93m	2.8E-31	933	2.8E-31	933
Nb-94	4.8E+07	10,000	5.0E+07	11,498
Ni-59	9.4E+08	10,000	1.8E+09	70,548
Ni-63	1.9E-02	2,477	1.9E-02	2,477
Np-237	1.7E+02	10,000	5.7E+02	53,367
Pa-231	5.8E-02	10,000	1.9E+03	81,113
Pb-210	2.8E-03	10,000	1.3E+05	376,494
Pd-107	1.3E+03	10,000	1.3E+03	10,000
Po-210	3.3E-03	10,000	1.5E+05	376,494

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Pu-238	1.5E-15	2,848	1.5E-15	2,848
Pu-239	2.6E+02	10,000	1.6E+04	40,370
Pu-240	1.8E+00	10,000	2.6E+01	23,101
Pu-241	1.2E-38	933	1.2E-38	933
Ra-226	9.1E-05	10,000	1.1E+04	376,494
Ra-228	6.9E-13	10,000	2.2E+02	1,000,000
Sb-125	6.2E-18	76	6.2E-18	76
Se-79	4.4E+04	4,329	4.4E+04	4,329
Sm-151	4.0E-18	3,275	4.0E-18	3,275
Sr-90	2.4E-15	1,233	2.4E-15	1,233
Tc-99	7.8E+06	5,722	7.8E+06	5,722
Th-228	5.4E-14	10,000	1.7E+01	1,000,000
Th-229	4.2E-04	10,000	8.8E+00	247,708
Th-230	7.9E-09	10,000	9.0E+02	376,494
Th-232	5.4E-14	10,000	1.7E+01	1,000,000
Tl-204	2.2E-08	14	2.2E-08	14
U-233	4.1E-02	10,000	1.8E+02	247,708
U-234	3.6E-05	10,000	2.2E+04	247,708
U-235	1.1E-04	10,000	5.0E+02	107,227
U-236	2.4E-05	10,000	2.8E-01	123,285
U-238	7.5E-04	10,000	5.9E+04	107,227

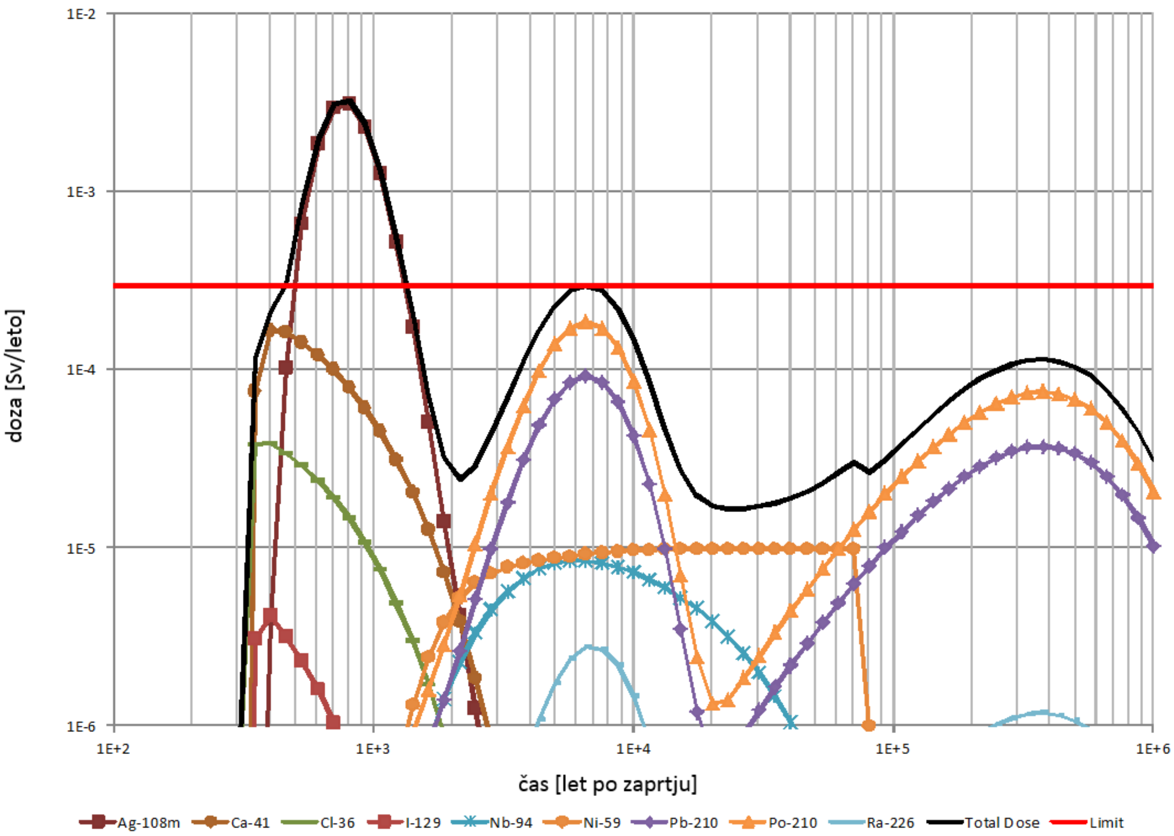

Figure 7.47: Annual dose for representative member of most-exposed population group under scenario of rapid failure of engineered barriers; it is assumed that the failure occurs immediately after the end of institutional controls (300 years after the closure of the repository). The limit of 0.3 mSv/year applying to the normal evolution scenario is illustrated for comparison; under the alternate evolution scenario, it is the case that measures to optimise the repository are not required up to 10 mSv/year

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

Table 7.53: Maximum doses and time of occurrence for annual dose under scenario of rapid failure of engineered barriers

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Total dose	3.2E-03	811	3.2E-03	811
Ac-227	3.6E-07	10,000	3.8E-07	23,101
Ag-108m	3.1E-03	811	3.1E-03	811
Am-241	4.4E-09	3,765	4.4E-09	3,765

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Ba-133	< 1E-10	305	< 1E-10	305
C-14	1.0E-07	8,697	1.0E-07	8,697
Ca-41	1.7E-04	404	1.7E-04	404
Cd-109	< 1E-10	57	< 1E-10	57
Cd-113m	< 1E-10	464	< 1E-10	464
Cl-36	3.9E-05	404	3.9E-05	404
Cm-244	< 1E-10	614	< 1E-10	614
Co-60	< 1E-10	404	< 1E-10	404
Cs-134	< 1E-10	305	< 1E-10	305
Cs-135	< 1E-10	10,000	5.0E-10	123,285
Cs-137	< 1E-10	811	< 1E-10	811
Eu-152	< 1E-10	534	< 1E-10	534
Eu-154	< 1E-10	305	< 1E-10	305
Eu-155	< 1E-10	305	< 1E-10	305
Fe-55	< 1E-10	305	< 1E-10	305
H-3	1.1E-08	305	1.1E-08	305
I-129	4.2E-06	404	4.2E-06	404
Na-22	< 1E-10	11	< 1E-10	11
Nb-93m	< 1E-10	534	< 1E-10	534
Nb-94	8.5E-06	5,722	8.5E-06	5,722
Ni-59	9.7E-06	10,000	9.9E-06	70,548
Ni-63	4.2E-08	1,072	4.2E-08	1,072
Np-237	1.7E-09	10,000	5.1E-09	53,367
Pa-231	2.0E-07	10,000	2.1E-07	23,101
Pb-210	9.2E-05	6,579	9.2E-05	6,579

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Pd-107	< 1E-10	5,722	< 1E-10	5,722
Po-210	1.9E-04	6,579	1.9E-04	6,579
Pu-238	< 1E-10	1,233	< 1E-10	1,233
Pu-239	1.8E-07	10,000	4.6E-07	26,561
Pu-240	1.3E-09	10,000	1.6E-09	15,199
Pu-241	< 1E-10	534	< 1E-10	534
Ra-226	2.8E-06	6,579	2.8E-06	6,579
Ra-228	< 1E-10	10,000	2.4E-08	497,702
Sb-125	< 1E-10	76	< 1E-10	76
Se-79	3.4E-08	1,072	3.4E-08	1,072
Sm-151	< 1E-10	1,417	< 1E-10	1,417
Sr-90	< 1E-10	614	< 1E-10	614
Tc-99	4.0E-07	5,722	4.0E-07	5,722
Th-228	< 1E-10	10,000	2.0E-10	497,702
Th-229	< 1E-10	10,000	4.1E-10	215,443
Th-230	< 1E-10	10,000	2.4E-08	432,876
Th-232	< 1E-10	10,000	6.3E-10	497,702
Tl-204	< 1E-10	16	< 1E-10	16
U-233	< 1E-10	10,000	9.0E-10	215,443
U-234	8.0E-10	10,000	7.7E-08	215,443
U-235	1.4E-10	10,000	2.0E-09	70,548
U-236	< 1E-10	10,000	< 1E-10	70,548
U-238	1.5E-08	10,000	2.2E-07	61,359

The results show a maximum dose of 3.2 mSv/year, which occurs 800 years after the closure of the repository. For the sake of consistency with other figures in this section of the draft safety analysis report, the figure also illustrates the limit of 0.3 mSv/year for the normal evolution scenario. The scenario presented is an alternate evolution scenario that does not need to be compared with the prescribed dose limit. In accordance with the JV5 rulebook, [21] optimisation is necessary if the estimated dose in the alternate evolution of the repository exceeds 10 mSv/year. The calculated dose under the scenario of the early failure of engineered barriers is below 10 mSv/year, and further optimisation of the repository is not necessary.

7.3.6.3 Scenario of early failure of concrete barriers

The scenario of the early failure of concrete barriers is complementary to the previous scenario, where in this case only the concrete barriers fail. As such it represents a large number of potential FEPs that could have an impact on the capacity of the repository to isolate waste from the environment, and to contain it in the repository. These FEPs include seismic events that are not envisaged in the project (beyond-design-basis events), faults in manufacture and construction, anomalous operation and maintenance, etc. All these FEPs are evaluated in the generic sense without precise specifications of what actually occurs. Under this approach there is no avoiding conservativeness in the interpretation of failure, which would be smaller if for each FEP a process model were to be drawn up for how the event impacts the facility. For example, a larger seismic event could result in minor localised cracks, if the actual process of the generation of such cracks were modelled by mechanical tension process models. Analysis of this type would rely on a large number of assumptions, and it is therefore much more efficient to assume a more conservative approach, which is also easier to argue. The scenario thus addresses a situation in which all concrete barriers fail, although the probability of this occurring is very low or negligible, and there is no expectation of the scenario occurring. The scenario thus represents all possible combinations of events that could lead to the failure of the concrete barriers.

In the scenario it is assumed that the concrete barriers fail very quickly, i.e. their properties evolve quickly from intact to disintegrated, which means they no longer constitute a hydrological barrier. It is highly unlikely that such an event would really occur in totality, for which reason this scenario needs to be understood as marginal analysis of the faster failure of concrete barriers than that envisaged under the nominal scenario.

The scenario is evaluated in the same way as the nominal scenario, except that it is envisaged that the concrete barriers are subject to rapid failure (a change in properties) after the end of institutional controls. It is assumed that an initial event could occur at a time between 300 and 10,000 years after the closure of the repository.[3] Under the conservative approach, it is assumed in the evaluation of the scenario that there is an event immediately after the end of institutional controls, i.e. 300 years after the closure of the repository. After the initial event occurs, the properties of the concrete barriers evolve from their initial values to a degraded state over the course of one year. All other properties of the system evolve the same way as under the nominal scenario.

In the event of an earthquake during operation of the repository, an inspection of SSCs will be conducted. An earthquake that is significant from the aspect of verifying the status of SSCs is a seismogenic movement of the soil with acceleration in excess of 0.15 g, measured at one of

[3] If the initial event occurs within a period of 300 years after the closure of the repository, this is during the period of institutional controls, and faults in the disposal system could be rectified. A further scenario of early failure immediately after the closure of the repository will be added in the next phase of the safety analysis.

the nearest earthquake monitoring stations within the national network (LEGS, CRES, GOLS or GCIS). The impact of such an earthquake is addressed in detail in Section 6 of this draft safety analysis report.

The results of the analysis are presented below.

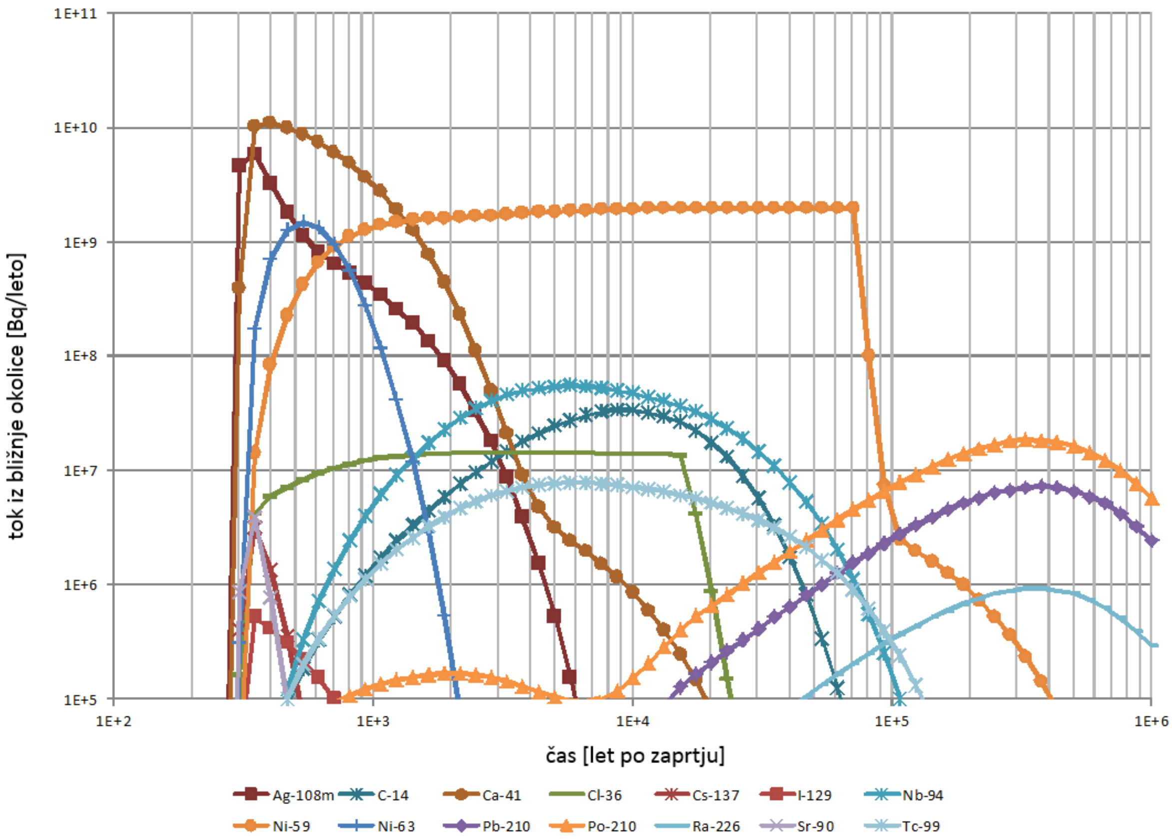

Figure 7.48: Releases of radionuclides from nearfield to farfield of repository under scenario of rapid failure of concrete barriers; it is assumed that the failure occurs immediately after the end of institutional controls (300 years after the closure of the repository)

tok iz bližnje okolice [Bq/leto]	flow from nearfield [Bq/year]
čas [let po zaprtju]	period [years after closure]

Table 7.54: Maximum releases and time of occurrence for releases from silo under scenario of rapid failure of concrete barriers

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Total release	1.6E+10	351	1.6E+10	351
Ac-227	3.8E+03	10,000	5.6E+03	15,199
Ag-108m	5.8E+09	351	5.8E+09	351

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Am-241	2.6E+03	2,154	2.6E+03	2,154
Ba-133	1.4E-02	305	1.4E-02	305
C-14	3.4E+07	10,000	3.4E+07	10,000
Ca-41	1.1E+10	404	1.1E+10	404
Cd-109	9.7E-10	2	9.7E-10	2
Cd-113m	5.7E-01	351	5.7E-01	351
Cl-36	1.4E+07	2,848	1.4E+07	2,848
Cm-244	5.6E-06	351	5.6E-06	351
Co-60	4.6E-02	12	4.6E-02	12
Cs-134	1.1E-02	3	1.1E-02	3
Cs-135	2.1E+04	10,000	2.2E+04	15,199
Cs-137	3.2E+06	351	3.2E+06	351
Eu-152	1.0E-07	351	1.0E-07	351
Eu-154	2.9E-09	22	2.9E-09	22
Eu-155	4.4E-10	11	4.4E-10	11
Fe-55	6.2E+02	4	6.2E+02	4
H-3	1.7E+04	305	1.7E+04	305
I-129	5.2E+05	351	5.2E+05	351
Na-22	4.0E-08	5	4.0E-08	5
Nb-93m	3.0E-12	351	3.0E-12	351
Nb-94	5.5E+07	5,722	5.5E+07	5,722
Ni-59	1.9E+09	10,000	2.0E+09	61,359
Ni-63	1.5E+09	534	1.5E+09	534
Np-237	4.4E+01	10,000	5.3E+02	70,548
Pa-231	4.4E+03	10,000	6.5E+03	15,199

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Pb-210	4.9E+04	2,154	7.2E+06	376,494
Pd-107	6.4E+02	10,000	1.0E+03	17,475
Po-210	1.6E+05	1,874	1.8E+07	327,455
Pu-238	1.5E-03	705	1.5E-03	705
Pu-239	6.1E+03	10,000	1.9E+04	30,539
Pu-240	1.9E+01	10,000	3.7E+01	20,092
Pu-241	2.8E-08	351	2.8E-08	351
Ra-226	6.3E+03	2,154	9.2E+05	376,494
Ra-228	1.9E+02	10,000	2.0E+04	376,494
Sb-125	6.0E-02	4	6.0E-02	4
Se-79	1.1E+04	614	1.1E+04	614
Sm-151	9.9E-03	705	9.9E-03	705
Sr-90	3.8E+06	351	3.8E+06	351
Tc-99	7.8E+06	5,722	7.8E+06	5,722
Th-228	3.2E-01	10,000	3.3E+01	376,494
Th-229	6.0E-02	10,000	3.6E+01	215,443
Th-230	7.6E-01	10,000	2.6E+03	376,494
Th-232	9.5E-02	10,000	3.3E+01	376,494
Tl-204	7.2E-08	8	7.2E-08	8
U-233	1.8E+00	10,000	2.2E+02	215,443
U-234	1.8E+02	10,000	1.8E+04	247,708
U-235	3.1E+01	10,000	4.9E+02	70,548
U-236	1.2E-02	10,000	2.7E-01	70,548
U-238	3.6E+03	10,000	5.7E+04	70,548

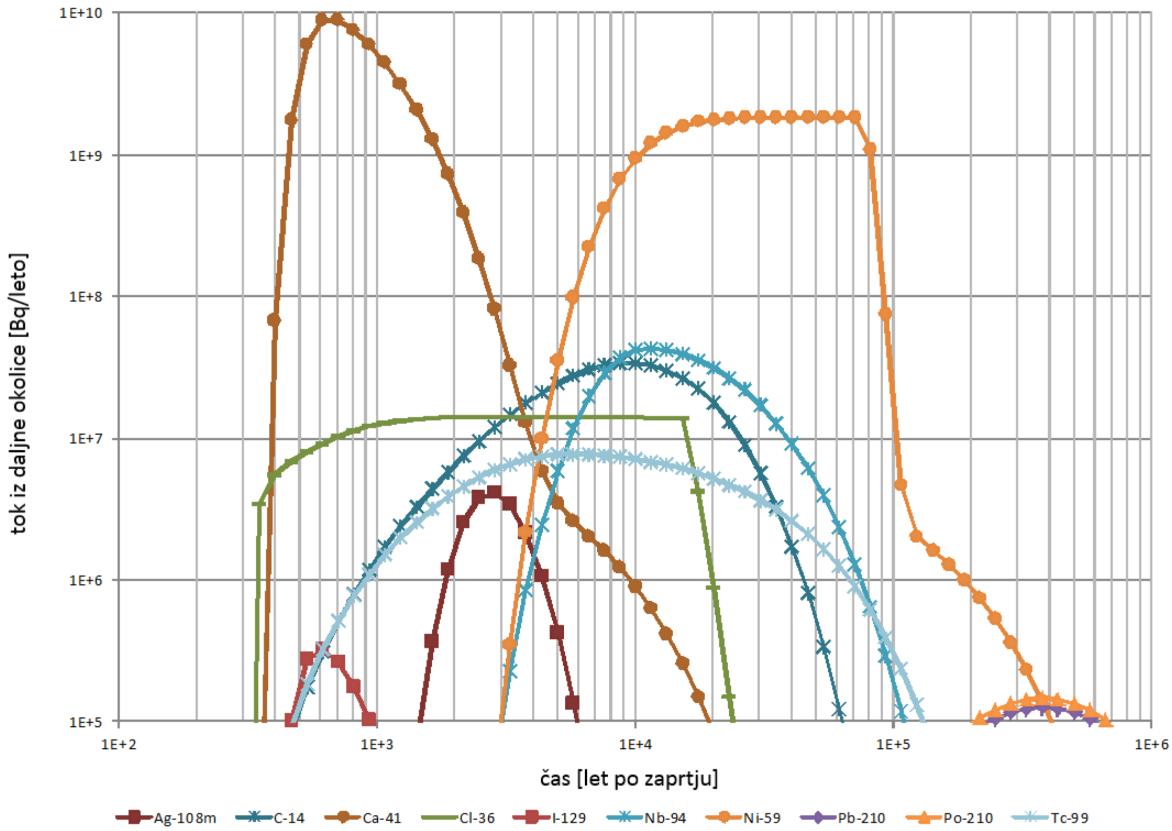

Figure 7.49: Releases of radionuclides from geosphere to groundwater and then into Sava and well under scenario of rapid failure of concrete barriers; it is assumed that the failure occurs immediately after the end of institutional controls (300 years after the closure of the repository)

tok iz daljne okolice [Bq/leto]	flow from farfield [Bq/year]
čas [let po zaprtju]	period [years after closure]

Table 7.55: Maximum releases and time of occurrence for releases from farfield model to groundwater and then into Sava and well under scenario of rapid failure of concrete barriers

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Total release	8.9E+09	705	8.9E+09	705
Ac-227	4.5E-03	10,000	2.1E+03	93,260
Ag-108m	4.2E+06	2,848	4.2E+06	2,848
Am-241	2.3E-06	10,000	2.3E-06	10,000
Ba-133	1.3E-03	351	1.3E-03	351
C-14	3.4E+07	10,000	3.4E+07	10,000

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Ca-41	8.9E+09	705	8.9E+09	705
Cd-109	2.5E-64	57	2.5E-64	57
Cd-113m	3.4E-26	811	3.4E-26	811
Cl-36	1.4E+07	2,848	1.4E+07	2,848
Cm-244	6.7E-74	1,233	6.7E-74	1,233
Co-60	2.8E-47	231	2.8E-47	231
Cs-134	3.9E-118	100	3.9E-118	100
Cs-135	6.7E-37	10,000	3.8E+02	869,749
Cs-137	9.3E-74	1,630	9.3E-74	1,630
Eu-152	7.1E-82	933	7.1E-82	933
Eu-154	1.5E-89	404	1.5E-89	404
Eu-155	2.6E-98	231	2.6E-98	231
Fe-55	1.6E-46	115	1.6E-46	115
H-3	9.0E+03	305	9.0E+03	305
I-129	3.3E+05	614	3.3E+05	614
Na-22	7.5E-09	11	7.5E-09	11
Nb-93m	1.0E-32	933	1.0E-32	933
Nb-94	4.1E+07	10,000	4.3E+07	11,498
Ni-59	9.4E+08	10,000	1.8E+09	70,548
Ni-63	1.8E-02	2,477	1.8E-02	2,477
Np-237	3.9E+01	10,000	5.3E+02	70,548
Pa-231	3.8E-03	10,000	1.8E+03	93,260
Pb-210	1.1E-06	10,000	1.3E+05	376,494
Pd-107	3.5E+02	10,000	1.0E+03	23,101
Po-210	1.3E-06	10,000	1.5E+05	376,494

Radionuclide	Maximum before 10,000 years (Bq/year)	Time of occurrence (years)	Maximum (Bq/year)	Time of occurrence (years)
Pu-238	6.3E-17	2,848	6.3E-17	2,848
Pu-239	1.1E+02	10,000	1.3E+04	46,416
Pu-240	1.7E-01	10,000	1.1E+01	30,539
Pu-241	5.7E-39	933	5.7E-39	933
Ra-226	3.2E-08	10,000	1.1E+04	376,494
Ra-228	5.3E-14	10,000	2.0E+02	1,000,000
Sb-125	6.2E-18	76	6.2E-18	76
Se-79	9.2E+03	10,000	9.4E+03	15,199
Sm-151	1.7E-19	3,275	1.7E-19	3,275
Sr-90	2.4E-15	1,233	2.4E-15	1,233
Tc-99	7.8E+06	5,722	7.8E+06	5,722
Th-228	4.2E-15	10,000	1.6E+01	1,000,000
Th-229	5.8E-05	10,000	8.5E+00	247,708
Th-230	3.0E-10	10,000	8.7E+02	376,494
Th-232	4.2E-15	10,000	1.6E+01	1,000,000
Tl-204	2.2E-08	14	2.2E-08	14
U-233	6.7E-03	10,000	1.8E+02	247,708
U-234	1.4E-06	10,000	2.1E+04	247,708
U-235	4.3E-05	10,000	4.7E+02	123,285
U-236	2.0E-06	10,000	2.6E-01	123,285
U-238	2.5E-05	10,000	5.5E+04	123,285

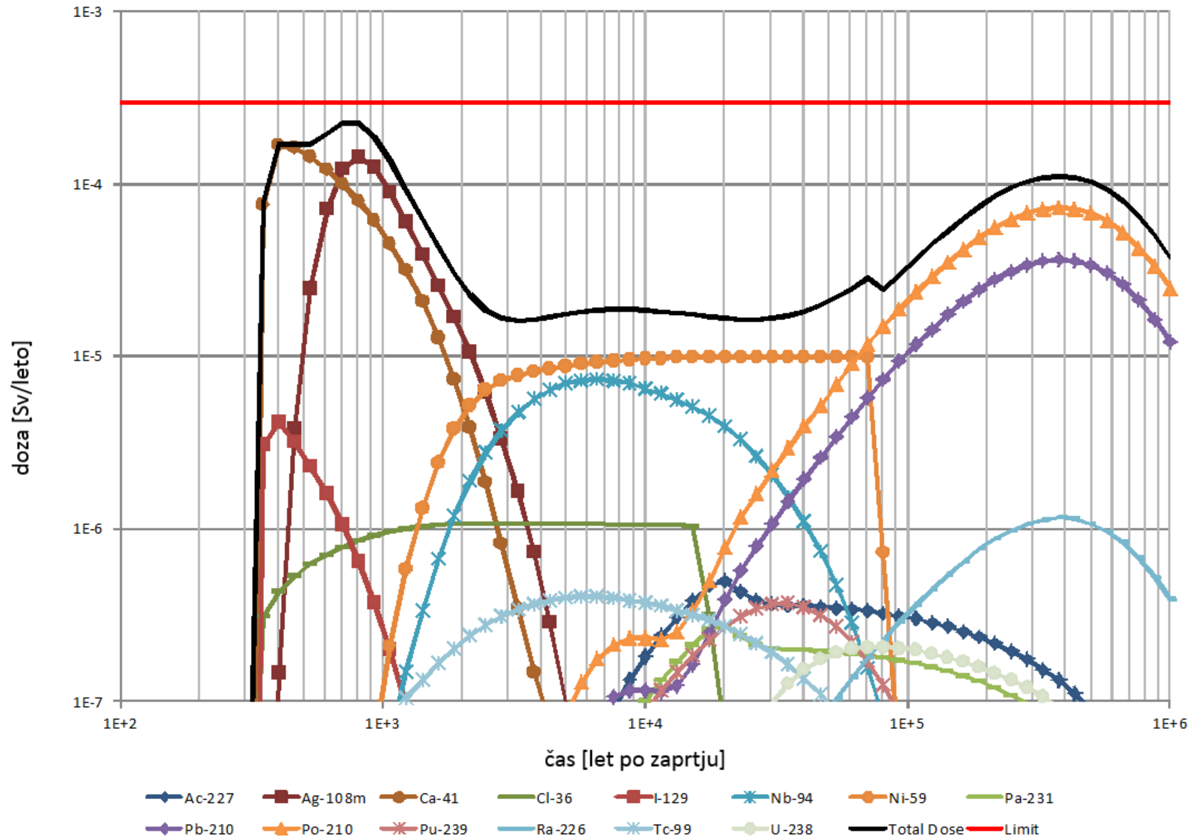

Figure 7.50: Annual dose for representative member of most-exposed population group under scenario of rapid failure of concrete barriers; it is assumed that the failure occurs immediately after the end of institutional controls (300 years after the closure of the repository). The limit of 0.3 mSv/year applying to the normal evolution scenario is illustrated for comparison; under the alternate evolution scenario, it is the case that measures to optimise the repository are not required up to 10 mSv/year

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

Table 7.56: Maximum doses and time of occurrence for annual dose under scenario of rapid failure of concrete barriers

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Total dose	2.3E-04	705	2.3E-04	705
Ac-227	1.8E-07	10,000	5.0E-07	20,092
Ag-108m	1.4E-04	811	1.4E-04	811
Am-241	3.6E-10	4,329	3.6E-10	4,329

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Ba-133	< 1E-10	305	< 1E-10	305
C-14	8.0E-08	10,000	8.0E-08	10,000
Ca-41	1.7E-04	404	1.7E-04	404
Cd-109	< 1E-10	57	< 1E-10	57
Cd-113m	< 1E-10	464	< 1E-10	464
Cl-36	1.1E-06	2,848	1.1E-06	2,848
Cm-244	< 1E-10	614	< 1E-10	614
Co-60	< 1E-10	305	< 1E-10	305
Cs-134	< 1E-10	305	< 1E-10	305
Cs-135	< 1E-10	10,000	4.6E-10	123,285
Cs-137	< 1E-10	811	< 1E-10	811
Eu-152	< 1E-10	534	< 1E-10	534
Eu-154	< 1E-10	305	< 1E-10	305
Eu-155	< 1E-10	305	< 1E-10	305
Fe-55	< 1E-10	305	< 1E-10	305
H-3	< 1E-10	305	< 1E-10	305
I-129	4.1E-06	404	4.1E-06	404
Na-22	< 1E-10	11	< 1E-10	11
Nb-93m	< 1E-10	534	< 1E-10	534
Nb-94	7.3E-06	6,579	7.3E-06	6,579
Ni-59	9.7E-06	10,000	9.9E-06	70,548
Ni-63	4.2E-08	1,072	4.2E-08	1,072
Np-237	3.9E-10	10,000	4.7E-09	70,548
Pa-231	1.0E-07	10,000	2.8E-07	20,092
Pb-210	1.1E-07	8,697	3.6E-05	376,494

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Pd-107	< 1E-10	10,000	< 1E-10	20,092
Po-210	2.3E-07	8,697	7.4E-05	376,494
Pu-238	< 1E-10	1,233	< 1E-10	1,233
Pu-239	8.7E-08	10,000	3.7E-07	35,112
Pu-240	2.4E-10	10,000	6.2E-10	23,101
Pu-241	< 1E-10	534	< 1E-10	534
Ra-226	3.5E-09	10,000	1.2E-06	376,494
Ra-228	< 1E-10	10,000	2.3E-08	572,237
Sb-125	< 1E-10	76	< 1E-10	76
Se-79	2.2E-09	10,000	2.2E-09	10,000
Sm-151	< 1E-10	1,417	< 1E-10	1,417
Sr-90	< 1E-10	614	< 1E-10	614
Tc-99	4.0E-07	5,722	4.0E-07	5,722
Th-228	< 1E-10	10,000	1.9E-10	572,237
Th-229	< 1E-10	10,000	4.0E-10	215,443
Th-230	< 1E-10	10,000	2.3E-08	432,876
Th-232	< 1E-10	10,000	5.9E-10	572,237
Tl-204	< 1E-10	16	< 1E-10	16
U-233	< 1E-10	10,000	8.7E-10	215,443
U-234	< 1E-10	10,000	7.4E-08	247,708
U-235	< 1E-10	10,000	1.8E-09	70,548
U-236	< 1E-10	10,000	< 1E-10	70,548
U-238	1.8E-09	10,000	2.1E-07	70,548

The maximum calculated dose under the scenario of the rapid failure of the concrete barriers is 0.2 mSv/year, and occurs 700 years after the closure of the repository.

The calculations show that even if an event that seriously damages the concrete barriers of the repository occurs 300 years after its closure, the doses are still below those prescribed for the normal evolution scenario. The analysis assumed that the initial event occurs immediately after the end of institutional controls at the repository. The sensitivity analysis also addresses the sensitivity of the maximum calculated dose to the time at which the initial event occurs. The results are illustrated in Section 7.3.7.4.3 of this draft safety analysis report.

7.3.6.4 Scenario of river meandering and surface erosion

Natural and man-made future events and processes could lead to a change in the course of the Sava, which in the worst case scenario could flow over the repository. The result would be erosion of the Quaternary stratum of the aquifer, and a change in the speed and direction of the flow in the aquifer itself. In geological terms, it does not seem credible that erosion would reach the depth of the silo within 10,000 years, judging by the depth of the paleochannels of the Sava. [72] The primary effect of a change in the course of the Sava from its current bed would be a change in the speed and direction of the flow of groundwater in the vicinity of (and through) the repository.

Under the scenario it is assumed that the potential meandering of the river in the future does not constitute a threat to the physical integrity of the repository. This claim is based on the following facts:

- the Sava only partly penetrates the Quaternary sediments in the Krško Polje / Brežiško Polje area. Figure 7-26 shows that if the river were to flow above the disposal silo now, its depth would not reach the disposal facility. With regard to future river profiles, it is assumed that they will be similar to today, which means that in the future the river will not penetrate the Miocene sediments,
- the entire field consists of an area of sediment deposits, and accordingly it is envisaged that the thickness of the Quaternary strata will increase in the future,
- the construction of Brežice hydroelectric plant has tended to reduce the flow of the Sava, which will reduce the river's potential for carving a bed to the depth of the repository,
- under these conditions, in the event of the river meandering, the river bed over the repository would be at least 5 to 10 m above the top edge of the sealed disposal silo.

The following assumptions were made for the scenario of river meandering and surface erosion:

- the geosphere (farfield) model is eliminated, so that potential releases from the repository travel directly into the river, which is assumed to flow directly over the repository,
- releases into the well are not envisaged, as they go directly into the river,
- analysis of the biosphere model is limited by the assumption of releases into the river only,
- the flow of groundwater around and through the repository is the same as under the nominal scenario.

The results are presented below.

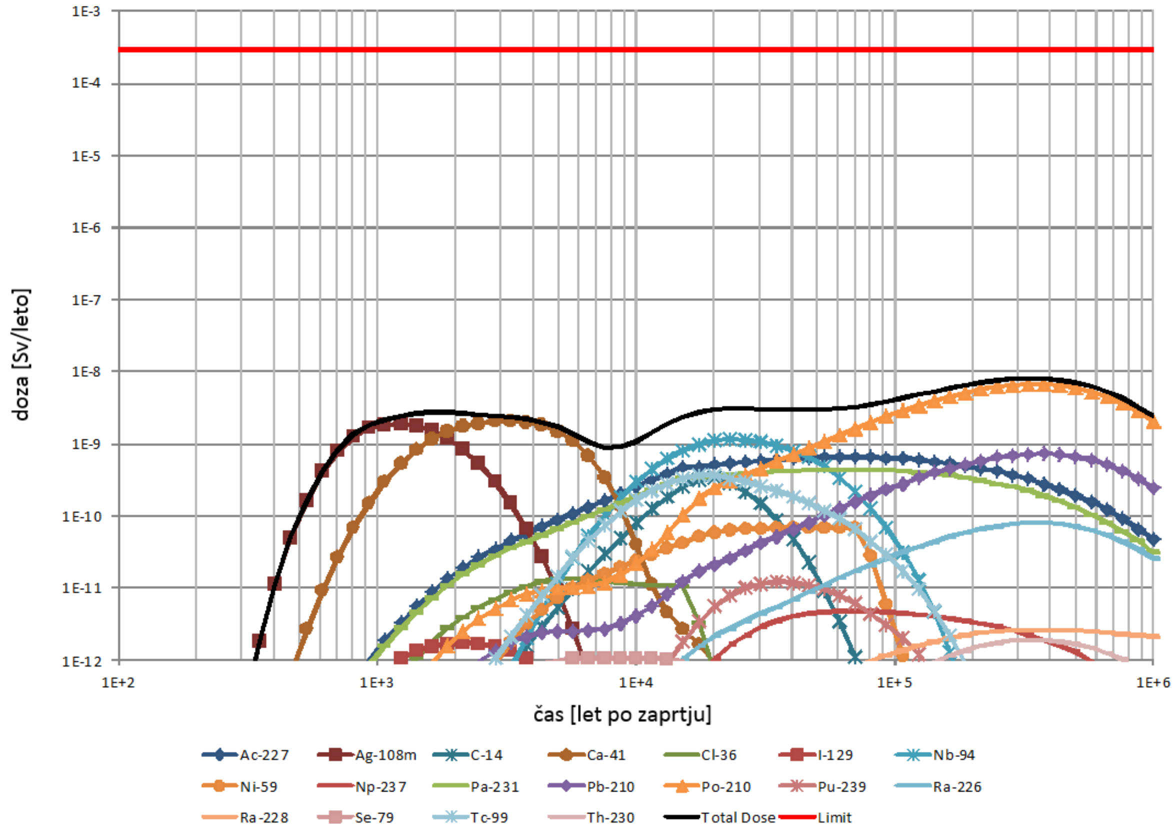

Figure 7.51: Annual dose for representative member of most-exposed population group under scenario of river meandering and surface erosion; the limit of 0.3 mSv/year applying to the normal evolution scenario is illustrated for comparison; under the alternate evolution scenario, it is the case that measures to optimise the repository are not required up to 10 mSv/year

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

Table 7.57: Maximum doses and time of occurrence for annual dose under scenario of river meandering and surface erosion

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Total dose	2.7E-09	1,630	8.0E-09	327,455
Ac-227	2.5E-10	10,000	6.5E-10	70,548
Ag-108m	1.9E-09	1,233	1.9E-09	1,233
Am-241	< 1E-10	4,977	< 1E-10	4,977
Ba-133	< 1E-10	28	< 1E-10	28

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
C-14	< 1E-10	10,000	3.3E-10	20,092
Ca-41	2.1E-09	3,275	2.1E-09	3,275
Cd-109	< 1E-10	2	< 1E-10	2
Cd-113m	< 1E-10	43	< 1E-10	43
Cl-36	< 1E-10	5,722	< 1E-10	5,722
Cm-244	< 1E-10	50	< 1E-10	50
Co-60	< 1E-10	16	< 1E-10	16
Cs-134	< 1E-10	3	< 1E-10	3
Cs-135	< 1E-10	10,000	< 1E-10	15,199
Cs-137	< 1E-10	464	< 1E-10	464
Eu-152	< 1E-10	50	< 1E-10	50
Eu-154	< 1E-10	28	< 1E-10	28
Eu-155	< 1E-10	16	< 1E-10	16
Fe-55	< 1E-10	4	< 1E-10	4
H-3	< 1E-10	305	< 1E-10	305
I-129	< 1E-10	2,154	< 1E-10	2,154
Na-22	< 1E-10	5	< 1E-10	5
Nb-93m	< 1E-10	25	< 1E-10	25
Nb-94	3.0E-10	10,000	1.1E-09	23,101
Ni-59	< 1E-10	10,000	< 1E-10	70,548
Ni-63	< 1E-10	1,233	< 1E-10	1,233
Np-237	< 1E-10	10,000	< 1E-10	70,548
Pa-231	1.8E-10	10,000	4.4E-10	61,359
Pb-210	< 1E-10	10,000	7.2E-10	376,494
Pd-107	< 1E-10	10,000	< 1E-10	20,092

Radionuclide	Maximum dose before 10,000 years (Sv/year)	Time of occurrence (years)	Maximum dose (Sv/year)	Time of occurrence (years)
Po-210	< 1E-10	10,000	6.7E-09	327,455
Pu-238	< 1E-10	933	< 1E-10	933
Pu-239	< 1E-10	10,000	< 1E-10	35,112
Pu-240	< 1E-10	10,000	< 1E-10	23,101
Pu-241	< 1E-10	22	< 1E-10	22
Ra-226	< 1E-10	10,000	< 1E-10	376,494
Ra-228	< 1E-10	10,000	< 1E-10	376,494
Sb-125	< 1E-10	5	< 1E-10	5
Se-79	< 1E-10	8,697	< 1E-10	8,697
Sm-151	< 1E-10	705	< 1E-10	705
Sr-90	< 1E-10	464	< 1E-10	464
Tc-99	1.7E-10	10,000	3.6E-10	17,475
Th-228	< 1E-10	10,000	< 1E-10	432,876
Th-229	< 1E-10	10,000	< 1E-10	247,708
Th-230	< 1E-10	10,000	< 1E-10	327,455
Th-232	< 1E-10	10,000	< 1E-10	572,237
Tl-204	< 1E-10	9	< 1E-10	9
U-233	< 1E-10	10,000	< 1E-10	215,443
U-234	< 1E-10	10,000	< 1E-10	247,708
U-235	< 1E-10	10,000	< 1E-10	107,227
U-236	< 1E-10	10,000	< 1E-10	93,260
U-238	< 1E-10	10,000	< 1E-10	107,227

The maximum calculated annual dose for the scenario of river meandering and surface erosion is 3 x 10^{-6} mSv/year, which occurs 1,600 years after the closure of the repository.

7.3.6.5 Scenario of change to hydrological conditions

A large number of FEPs could result in a regional change to the hydrological conditions in the vicinity of the LILW repository. These FEPs include natural and man-made climate changes, the construction of dams or other projects on the Sava, and other indirect human interventions in the aquifer. The result of all of these interventions could be a change to the direction and speed of the flow of groundwater in the nearfield of the repository and the aquifer. All of these can be simulated by a change in the hydraulic gradient in the nearfield of the repository. The dose was estimated under the assumption of a vertical gradient of 2 x 10^{-2} (m/m), which represents a gradient four times higher than that applied under the nominal scenario.

All other assumptions remain the same as under the nominal scenario.

The results of the analysis are presented below.

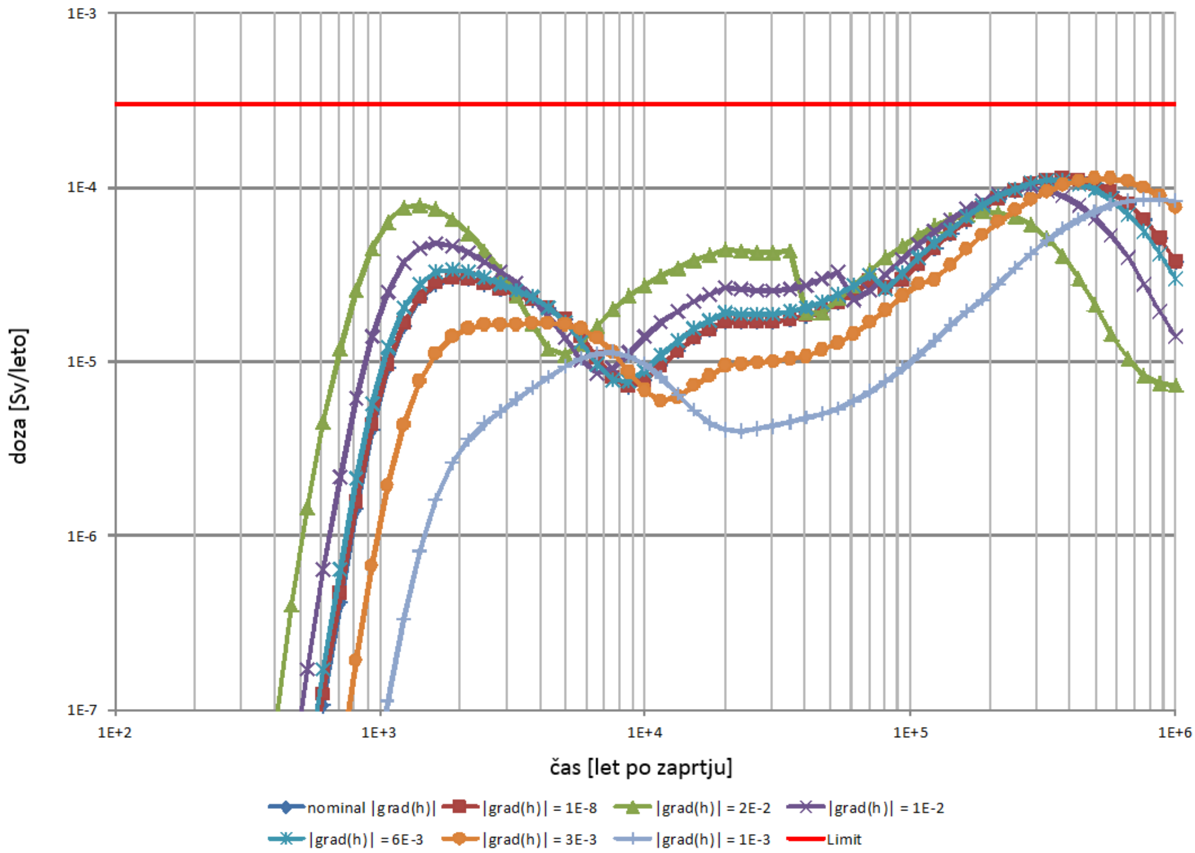

Figure 7.52: Annual dose for representative member of most-exposed population group under scenario of change to hydrological conditions. The limit of 0.3 mSv/year applying to the normal evolution scenario is illustrated for comparison; under the alternate evolution scenario, it is the case that measures to optimise the repository are not required up to 10 mSv/year

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

The maximum calculated annual dose under the scenario of a change to the hydrological conditions is 0.08 mSv/year, and occurs approximately 1,400 years after the closure of the repository. This takes account of an assumption that the dose is estimated quantitatively for the first 10,000 years, while a qualitative estimate only is given for later periods. In this case the maximum dose, which occurs 1,400 years after the closure of the repository, is taken as the result. The maximum dose after this period is comparable. The approach to dose estimation is defined in detail in Section 7.3.2.

7.3.6.6 Scenario of inadvertent human intrusion

As examined previously in Section 7.3.2 of this draft safety analysis report, the probability of the scenario of inadvertent human intrusion into the repository is extremely low, primarily owing to the disposal concept, according to which waste is disposed of below the water table, which could be an alternative source of drinking water, and there is minimal motivation for deep drilling in the area. The doses calculated in connection with the scenario of inadvertent human intrusion include an estimate of the dose for the intruder, and for a person who settles in the area soon after. The model is described in detail in the safety analysis report, [74] and is summarised in Section 7.3.5.6 of this draft safety analysis report.

It is assumed the inadvertent human intrusion occurs as a result of drilling through the repository, and the excavation of waste to the surface occurs, which results in exposure to radiation. Here the conservative assumption is that the drill bit is capable of drilling through concrete and metal (e.g. the reactor vessel).

The estimated dose for the intruder is presented in Figure 7.53 below. Intrusion may occur 300 years after the closure of the repository, as institutional controls are envisaged for the first 300 years after closure, during which time intrusion cannot occur. The estimated doses from intrusion are relatively low, owing to the short exposure time and the relatively low activity of the excavated waste. The maximum dose occurs 300 years after closure, and amounts to 5E-05 Sv/year, the largest contributions to which come from Ag-108m and Nb-94. The contribution by Ag-108m decreases over time as a result of decay, while Nb-94 is a long-lived radionuclide and its contribution is almost constant over the first 10,000 years. The red line denotes the limit of 10 mSv/year (the criterion for taking action), in line with Slovenian legislation. [21]

The scenario of inadvertent human intrusion was assessed conservatively, and the results slightly exceed the limits. The scenario will be addressed more realistically in subsequent safety analysis. It is also possible to constrain it via the acceptance criteria themselves.

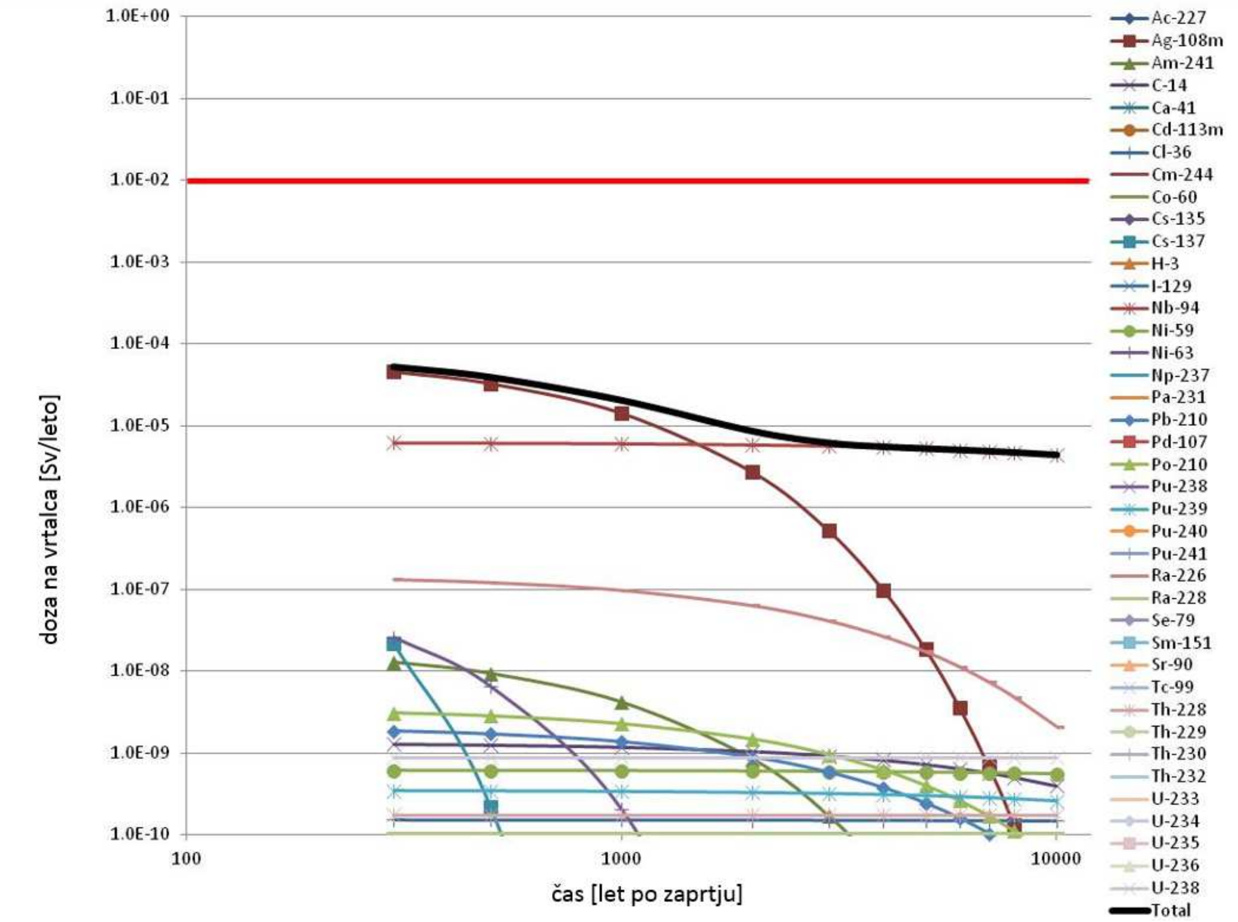

Figure 7.53: Estimated dose for driller under scenario of inadvertent human intrusion

doza na vrtalca [Sv/leto]	dose per driller [Sv/year]
čas [let po zaprtju]	period [years after closure]

The estimated dose under the scenario of inadvertent human intrusion into the repository for a member of the population who settle in the area of the repository after the intrusion is presented in Figure 7.54 below. The red line denotes the limit of 10 mSv/year (the criterion for taking action). [21] The maximum is attained in the first year after the end of institutional controls, when the scenario may occur. It amounts to just over 10 mSv/year, the largest contribution to which comes from Ag-108m. The dose falls below 10 mSv/year 500 years after the closure of the repository, as the Ag-108m decays. The estimated dose 1,000 years after closure is of the order of 1 to 2 mSv/year.

As stated previously, there are two basic sources of conservativeness taken into account in the calculation. The first is that the probability of such an event is extremely low, while the second is that the equipment for geotechnical drilling is capable of penetrating concrete and metal (usually metal is resistant to such drilling). Here it should be emphasised that the two radionuclides that contribute the most to the estimated dose are activation products, and are located in activated metal materials. On the basis of these facts, the safety analysis assesses that the estimated doses for the scenario of inadvertent human intrusion into the repository can be considered sufficiently low for there to be no need for additional measures to optimise the repository.

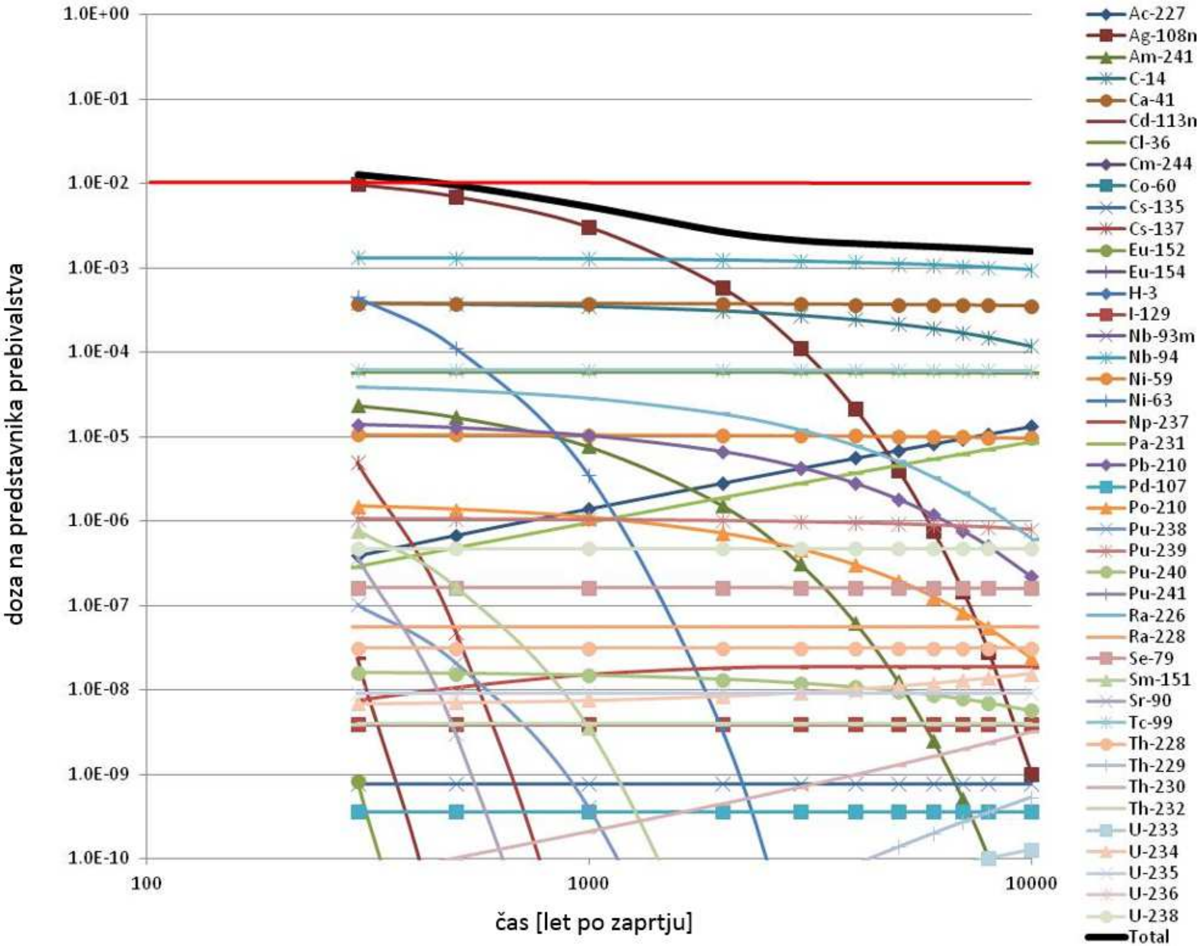

Figure 7.54: Dose for member of population settling in area of intrusion under scenario of inadvertent human intrusion

doza na predstavnika prebivalstva	dose per member of public
čas [let po zaprtju]	period [years after closure]

7.3.7 RESULTS OF SAFETY ANALYSIS AFTER CLOSURE OF THE REPOSITORY: PROBABILISTIC COMPUTATIONS AND SENSITIVITY ANALYSIS

The probabilistic calculations within the framework of the safety analysis were made using Ecolego software [93] and the Latin Hypercube Sampling method [94] that the software tool supports. The probabilistic calculations within the framework of the safety analysis were made using Ecolego software [93] and the Latin Hypercube Sampling method [94] that the software tool supports. The main purpose of using probabilistic calculations was to evaluate the impact of the uncertainty of individual parameters on the calculated final doses. The deterministic calculations for the scenarios described in Section 7.3.6 of this draft safety analysis report constitute a set of possible future events for the repository system, and include the best available expert and engineering judgements of the FEPs to which the repository system might be subject. Meanwhile the probabilistic calculations constitute a tool for evaluating the uncertainty of the parameters for a chosen scenario. This approach provides supplementary information about uncertainties within the framework of the safety assessment, as envisaged in the assessment context report. [51]

7.3.7.1 Nominal scenario: probabilistic calculations

The nominal scenario (described in Section 7.3.6.1 of this draft safety analysis report) represents the conservative basis on which the probabilistic analysis rests. Within this framework a thousand calculations were made using the Latin Hypercube Sampling method. This took account of a 95% and 5% confidence interval (95th percentile and 5th percentile). The calculations used probability density functions for the following parameters:

- corrosion times for stainless steel and carbon steel,
- concrete degradation time,
- Darcy velocity in the repository farfield model,
- thickness of the saturated zone of the alluvial aquifer,
- length of flow pathway from repository to river,
- flow rate of Sava,
- concentration ratios,
- sorption coefficients.

The parameters were chosen on the basis of best practice and the experience of those drawing up the safety analysis, and on the basis of their assessment it was determined which parameters have the largest impact on the variability of the final result. The values for individual parameters taken into account in the analysis and their probability density functions are presented in detail in the report on parameters. [83]

Applying the confidence intervals to the following parameters:

- concrete degradation time, and
- sorption coefficient,

the impact of the varying composition of the concrete on the results of the analysis was indirectly reviewed. The results obtained will aid in the preparation of recipes for particular concretes in future phases.

The doses calculated within the framework of the group of probabilistic calculations are presented in Figure 7.55 below, while Figure 7.56 illustrates the contribution of each parameter to the change in the final calculated dose. The contributions are illustrated as absolute contributions to the change in the final result, and not as the sensitivity of the calculated dose to a change in the individual parameter. The parameters that have an impact on the key radionuclides in the analysis and have greater uncertainty are illustrated. They are: the Darcy velocity in the geosphere model, the sorption coefficient in the nearfield model for Ca-41 (concrete), C-14 (concrete), Ni-59 (sand), U-238 (concrete), Nb-94 (concrete), Ra-226 (concrete) and Ag-108m (concrete), the sorption coefficient in the geosphere model for Pb-210, Po-210, Ag-108m, Ca- 41 and Ra-226, the corrosion times for carbon steel, and the concrete degradation time.

The maximum estimated doses at the 95% confidence interval do not exceed the limit of 0.3 mSv/year in the first 10,000 years after the closure of the repository. The maximum during this period is attained approximately 3,000 years after closure, and amounts to 0.05 mSv/year. During the first 10,000 years after closure, the highest estimated doses occur between 1,000 and 5,000 years after closure, and the most sensitive parameter is the Darcy velocity in the farfield model, i.e. this parameter contributes the most to the change in the final result.

The first maximum (0.05 mSv/year) is primarily produced by Ca-41 and Ag-108m. The two radionuclides are taken into account very conservatively in the calculations, particularly in terms of their geochemical behaviour. The evaluation of Ca-41 did not take account of solubility or isotopic dilution, despite the large quantities of stable calcium in the concrete in engineered barriers (no account was taken of leaching of stable calcium and exchange for Ca-41). The most important (most influential) parameter in the first period is the Darcy velocity in the farfield model, while the sorption coefficient in the nearfield model (K_d) later becomes the most influential parameter. The difference arises because the most influential radionuclides in the first period are Ca-41 and Ag-108m, which have a small range in K_d values, while over the longer term the most influential radionuclides are Ni-59 and Ra-226 and their daughters, which have greater uncertainty in the determination of K_d. This effect is also evident from the area of uncertainty (Figure 7.55), i.e. the difference between the 95th percentile and the 5th percentile, which amounts to one order of magnitude in the early period and three orders of magnitude later.

The uncertainty of individual parameters over the longer term after the closure of the repository is reflected in the wide distribution of the results (doses). At the 95th percentile this even exceeds the limit of 0.3 mSv/year, and reaches a value of 0.6 mSv/year, which is the result of the highly conservative approach in the application of parameters to the calculation of the 95th percentile.

Figure 7.55 also illustrates the result of the deterministic calculations. This was made with moderate, not excessive conservativeness, which is evident from the results illustrated in comparison with the probabilistic calculations.

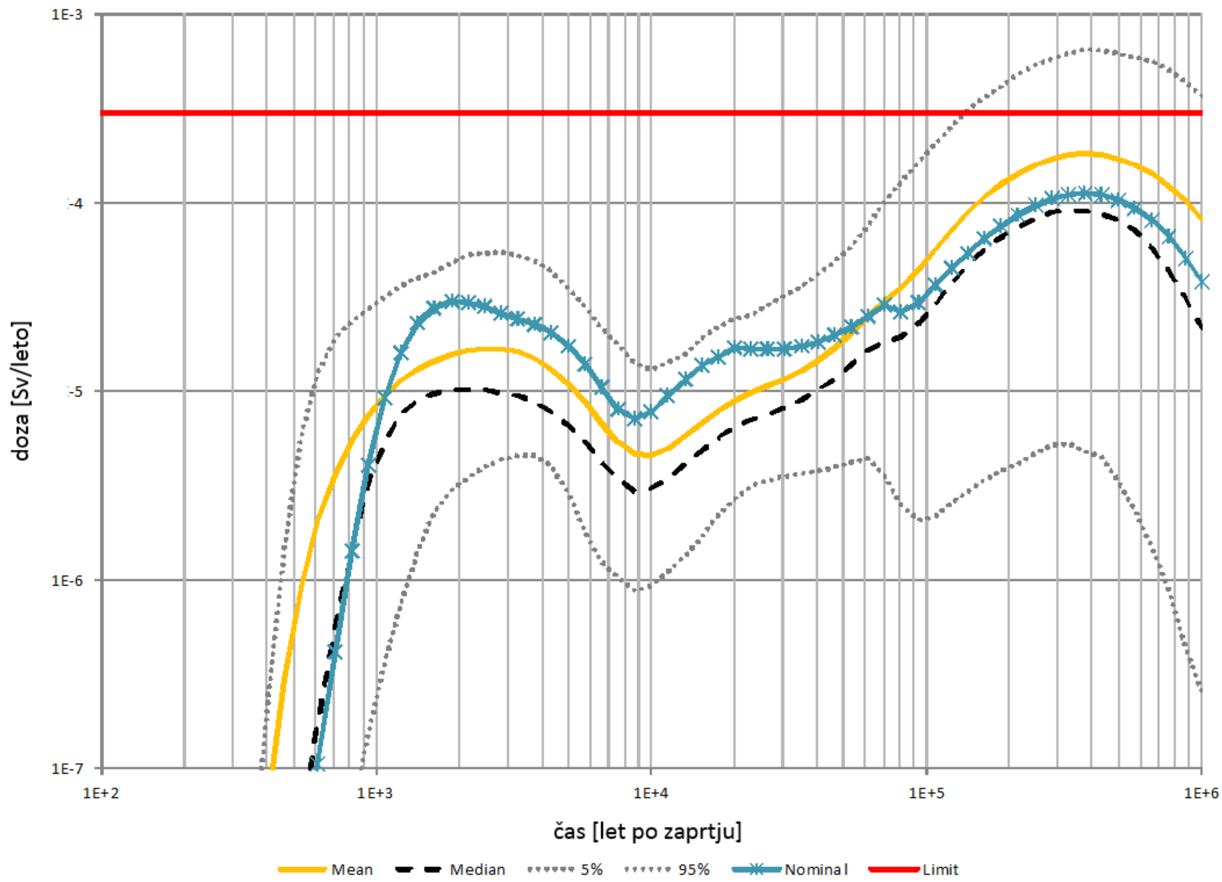

Figure 7.55: Doses for member of critical population group estimated by probabilistic analysis under nominal scenario. The results calculated by deterministic estimates are also illustrated (labelled “nominal” in the figure). The limit of 0.3 mSv/year is also illustrated. (lower dotted line denotes 5%, upper denotes 95%)

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

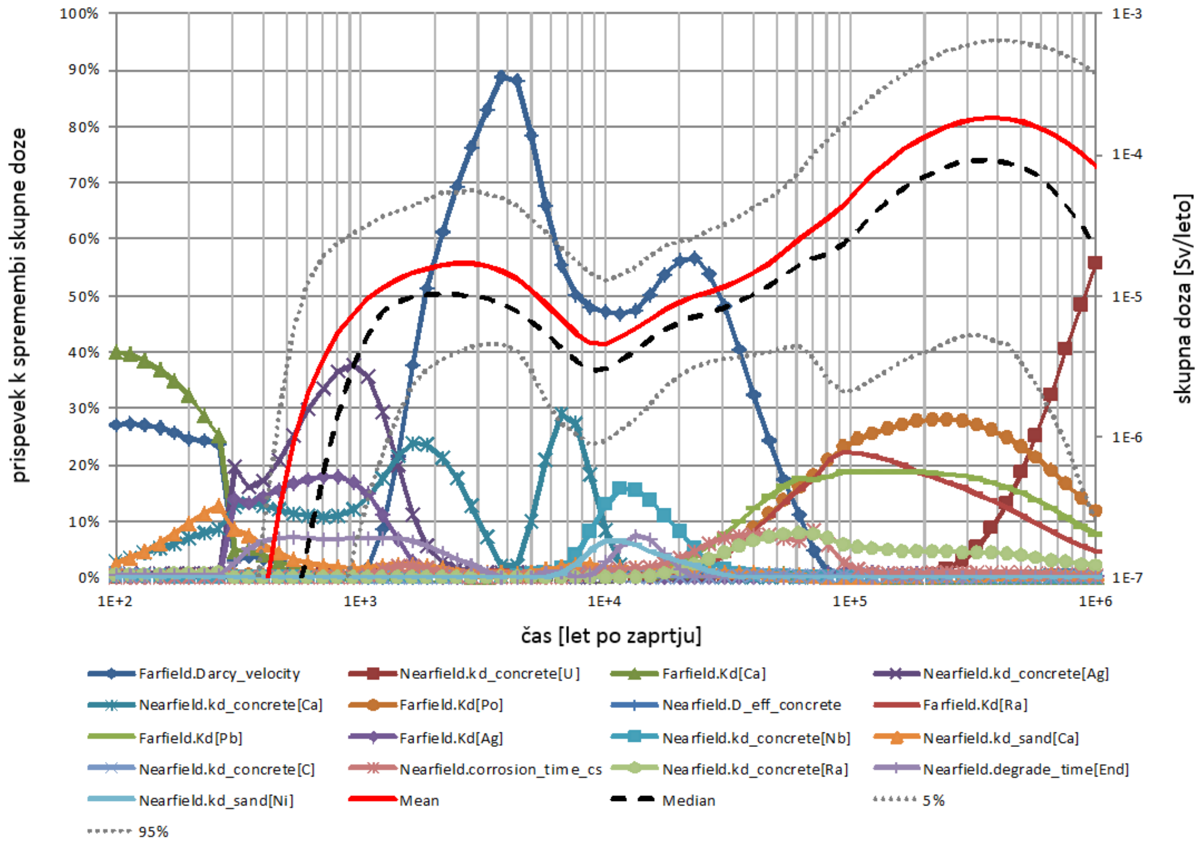

Figure 7.56:Contribution of individual parameters to change in total dose, having regard for uncertainty obtained from probabilistic calculations of nominal scenario (lower dotted line denotes 5%, upper denotes 95%)

prispevek k spremembi skupne doze	contribution to change in total dose
skupna doza [Sv/leto]	total dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.1.1 Variant of nominal scenario without well

Similarly to the nominal scenario, probability density functions were used in the calculations for the following parameters:

- corrosion times for stainless steel and carbon steel,
- concrete degradation time,
- Darcy velocity in the repository farfield model,
- thickness of the saturated zone of the alluvial aquifer,
- length of flow pathway from repository to river,
- flow rate of Sava,
- concentration ratios,
- sorption coefficients.

The group of results calculated within the framework of the probabilistic simulations is illustrated in Figure 7.57, while the sensitivity of the final result to a change in individual parameters is illustrated in Figure 7.58. As already presented in Section 7.3.6 of this draft safety analysis report, the key radionuclides under this scenario are the same as under the nominal scenario, although the final estimated dose is much lower. There is, however, a change in the key parameters that have the largest impact on the final estimated dose. The flow in the farfield model no longer has such a large impact in the first period. Instead the most influential parameters are in the biosphere model, particularly those determining the pathways of the ingestion of radionuclides through food. In the calculated scenario the maximum dose at the 95th percentile does not exceed the limit of 0.3 mSv/year over the entire period of the simulation. The maximum is reached approximately 3,000 years after the closure of the repository, and amounts to 3 x 10^{-5} mSv/year. In the first period the parameter with greatest influence on the final result is the concentration ratio of calcium in crop roots, which is attributable to Ca-41 being the dominant radionuclide that contributes most to the total dose in the first period (up to 10,000 years after closure).

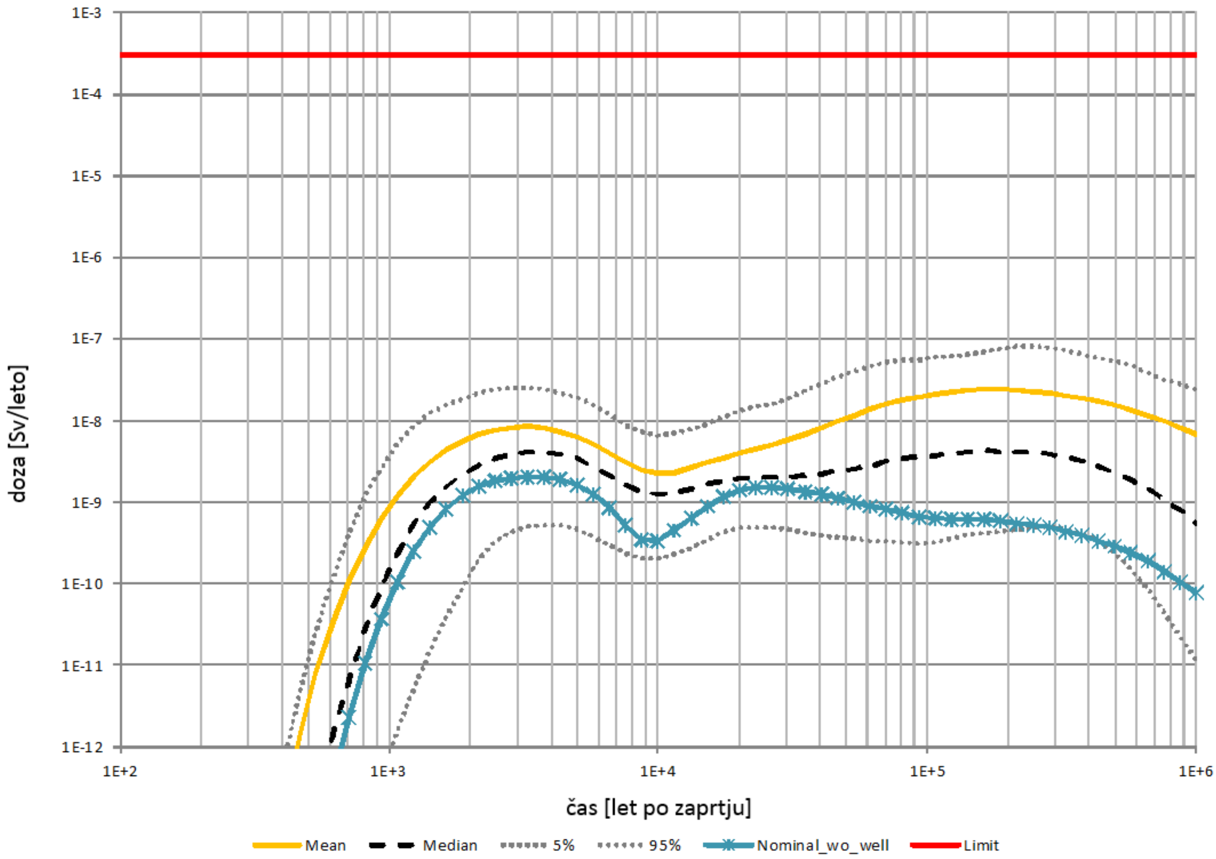

Figure 7.57: Doses for member of critical population group estimated by probabilistic analysis under variant of nominal scenario without well. The results calculated by deterministic estimates are also illustrated (labelled “nominal” in the figure). The limit of 0.3 mSv/year is also illustrated. (lower dotted line denotes 5%, upper denotes 95%)

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

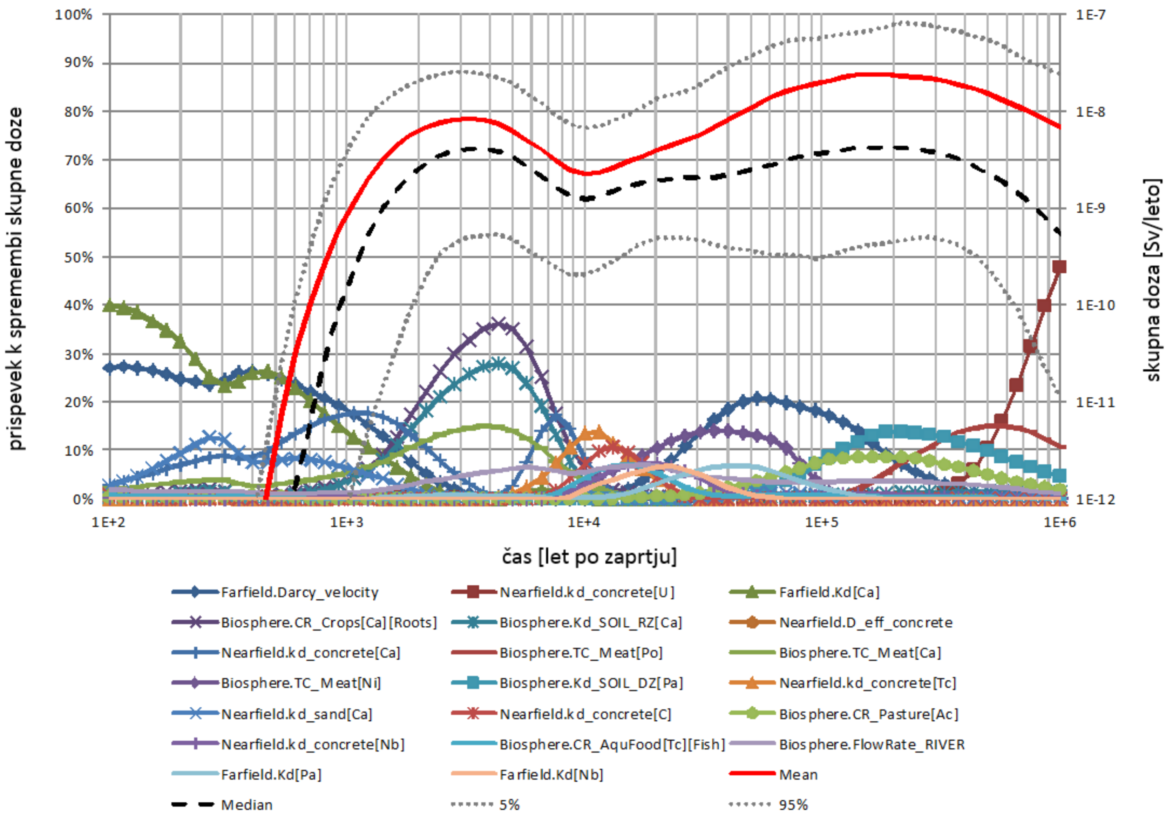

Figure 7.58: Contribution of individual parameters to change in total dose, having regard for uncertainty obtained from probabilistic calculations of variant of nominal scenario without well (lower dotted line denotes 5%, upper denotes 95%)

prispevek k spremembi skupne doze	contribution to change in total dose
skupna doza [Sv/leto]	total dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.2 Early failure of concrete barriers: probabilistic calculations

The results of the probabilistic calculations for the scenario of the early failure of concrete barriers are illustrated in Figure 7.59 and Figure 7.60. They are similar to the results for the nominal scenario. The flow velocity parameters in the farfield model have a slightly larger impact on the final result of the analysis. This is primarily because in the case of the failure of concrete barriers (the nearfield), the farfield represents the key hydrological barrier to the migration of radionuclides. In the later period the most important parameters are the sorption coefficients (K_d), and those radionuclides that have greater uncertainty in the determination of their K_d values.

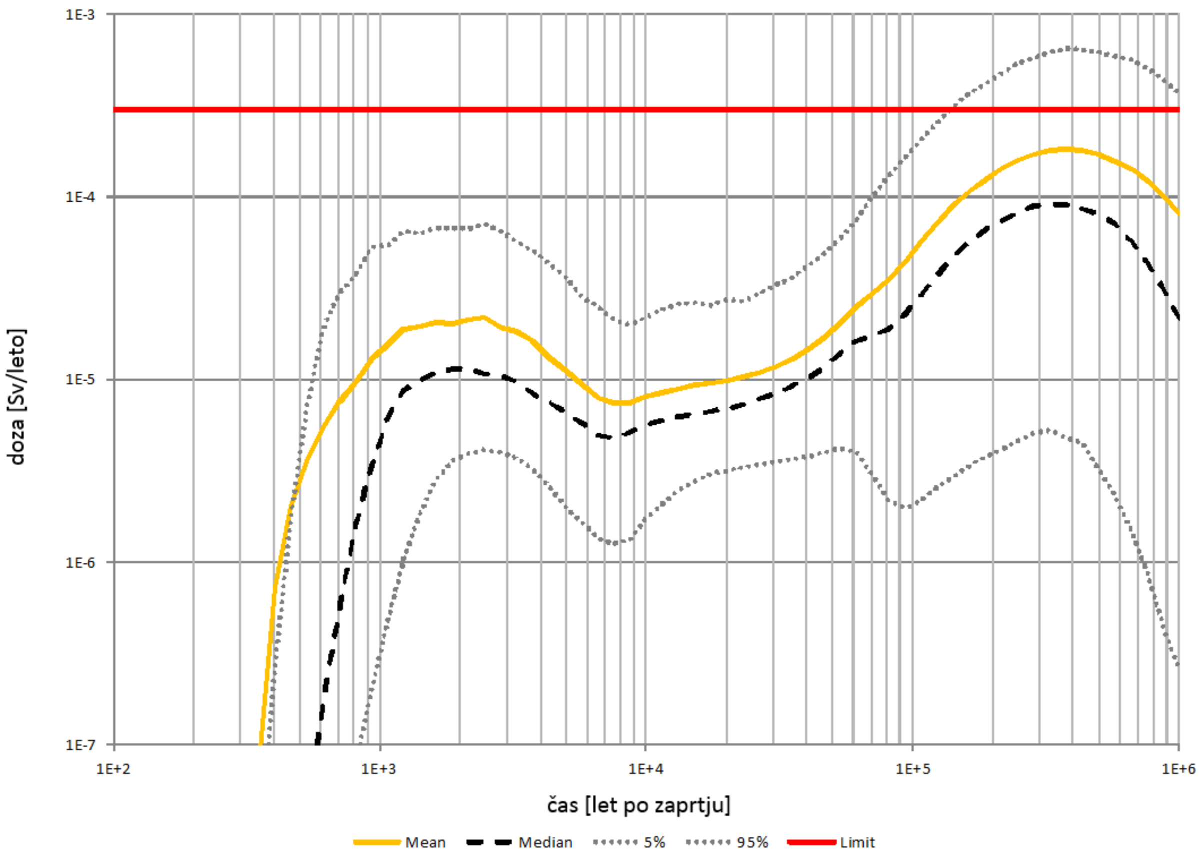

Figure 7.59: Doses for member of critical population group estimated by probabilistic analysis under scenario of early failure of concrete barriers. The limit of 0.3 mSv/year is also illustrated (lower dotted line denotes 5%, upper denotes 95%)

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

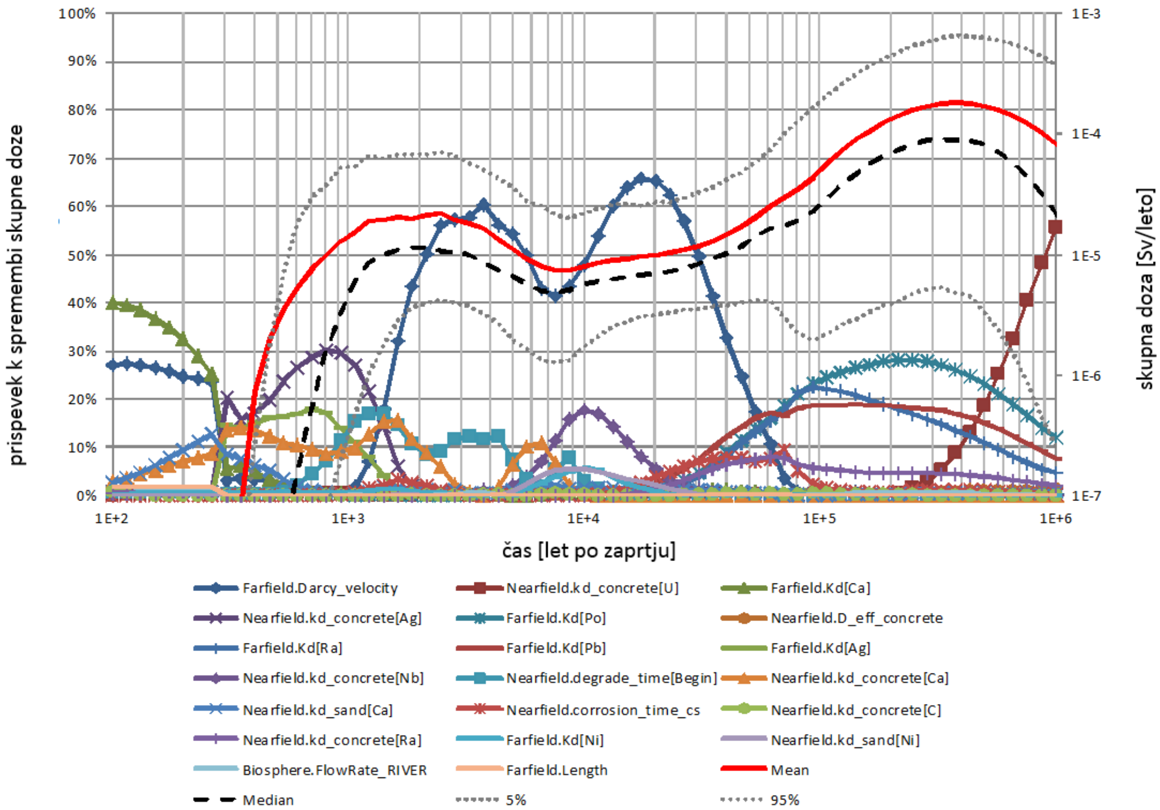

Figure 7.60: Contribution of individual parameters to change in total dose, having regard for uncertainty obtained from probabilistic calculations of scenario of early failure of concrete barriers (lower dotted line denotes 5%, upper denotes 95%)

prispevek k spremembi skupne doze	contribution to change in total dose
skupna doza [Sv/leto]	total dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.3 Scenario of river meandering and surface erosion: probabilistic calculations

Similarly to the nominal scenario, probability density functions were used in the calculations for the following parameters:

- corrosion times for stainless steel and carbon steel,
- concrete degradation time,
- Darcy velocity in the repository farfield model,
- thickness of the saturated zone of the alluvial aquifer,
- length of flow pathway from repository to river,
- flow rate of Sava,
- concentration ratios,
- sorption coefficients.

The group of results calculated within the framework of the probabilistic simulations is illustrated in Figure 7.61, while the sensitivity of the final results to a change in individual parameters is illustrated in Figure 7.62. The result of the calculation at the 95th percentile does not exceed the limit of 0.3 mSv/year over the entire period of the simulation. The maximum dose of 0.05 mSv/year is attained in the first period at the 95th percentile 10,000 years after the closure of the repository. In the period when the estimated doses are highest (5,000 to 10,000 years after closure), the largest contribution to the change in the final dose comes from the concentration ratio of calcium in crop roots. For this reason Ca-41 also makes the largest contribution to the maximum dose in the first 10,000 years after the closure of the repository. Comparing the deterministic calculations with the probabilistic calculations, it can be seen that the deterministic results show a central trend of sorts throughout, and are slightly below the calculated medians and means of the assumed parameters. The results of the deterministic analysis are like this because the best estimates of parameters were assumed *a priori*. However, the maximums for all calculations are of the same order of magnitude.

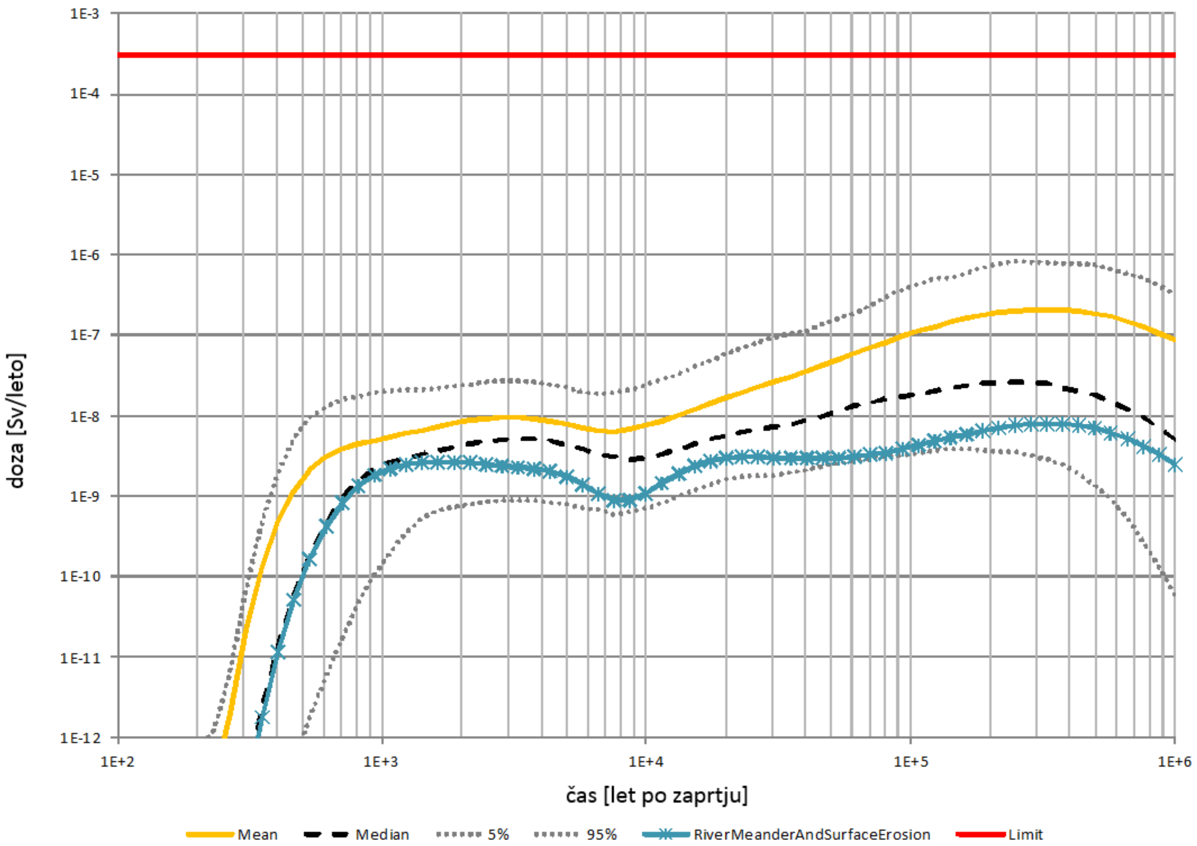

Figure 7.61: Doses for member of critical population group estimated by probabilistic analysis under scenario of river meandering and surface erosion. The results calculated by deterministic estimates are also illustrated (labelled "RiverMeanderAndSurfaceErosion" in the figure). The limit of 0.3 mSv/year is also illustrated (lower dotted line denotes 5%, upper denotes 95%)

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

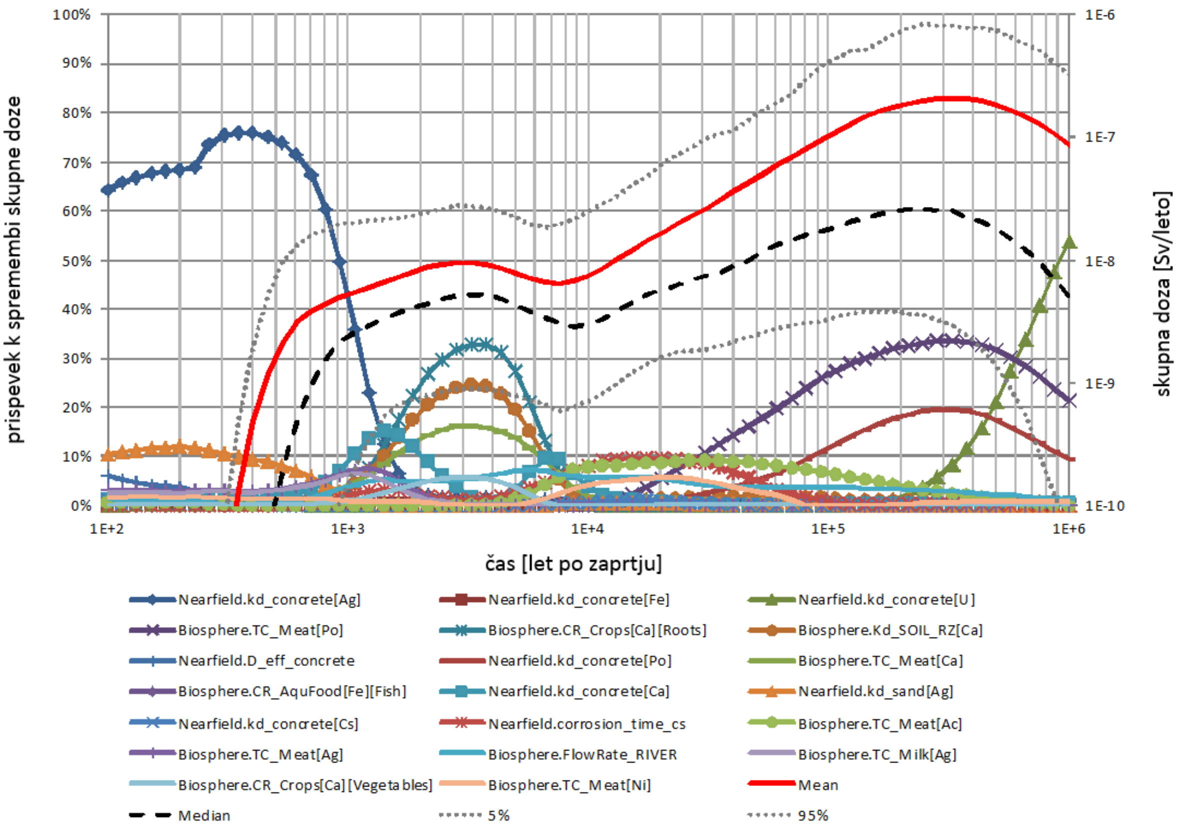

Figure 7.62: Contribution of individual parameters to change in total dose, having regard for uncertainty obtained from probabilistic calculations of scenario of river meandering and surface erosion (lower dotted line denotes 5%, upper denotes 95%)

prispevek k spremembi skupne doze	contribution to change in total dose
skupna doza [Sv/leto]	total dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.4 Sensitivity analysis for individual parameters

This section presents the results of sensitivity analysis for individual chosen parameters. They represent a particular focus on specific parameters and radionuclides that were identified as significant within the framework of deterministic and probabilistic calculations. The main purpose of this section is to give a view of the impact and consequences of uncertainty in individual parameters on the final results.

7.3.7.4.1 <u>Change in sorption</u>

To test the model in response to a change in sorption for individual radionuclides, the nominal scenario was used with various values of the sorption coefficient K_d for radium in the farfield model (see Figure 7.63). Radium was chosen because it is a key radionuclide in the longer term after the closure of the repository, but does not make a key contribution in the first period (10,000 years after closure).

Another reason for the choice of radium is that there are two different radium (Ra-226) sources in the repository. The first maximum dose for radium is associated with Ra-226 in the inventory, and its decay time is 1,600 years, which means that it does not contribute greatly to the total dose in the period more than 10,000 years after closure. The maximum to which Ra-226 makes a key contribution is the result of radium generated in the decay chain of U-238. In this case it can be observed how changing the sorption coefficient for an individual radionuclide (Ra-226) impacts the final change in the maximum dose. When K_d = 1 m^3/kg, the calculated maximum is 0.2 mSv/year, and occurs approximately 375,000 years after the closure of the repository. When K_d = 302.5 m^3/kg, the calculated maximum is 0.03 mSv/year, and occurs approximately 400,000 years after closure. The results show that the calculated maximum is not linear with regard to the change in the value of K_d. The highest maximum is attained with an intermediate K_d somewhere between its lowest and highest modelled values. This non-linear effect can be explained as follows: when the sorption coefficient is low, there is faster removal of Ra-226 and thus greater dilution, while when the sorption coefficient is high, more Ra-226 binds to the environmental material and the values of the maximum are therefore lower. The highest maximum is thus attained at a value of K_d between the minimum and maximum values.

The main contributions to the final dose from the uranium decay chain come from the daughters of Ra-226, namely Po-210 and Pb-210, which similarly to Ra-226 are relatively mobile in the entire system of the LILW repository when generic values for the sorption coefficients are used. Consequently, the non-linear part of the calculation of the dose is the result of multiple contributions to the final dose, which result in a significantly higher dose than would be the case if only Ra-226 is taken into account.

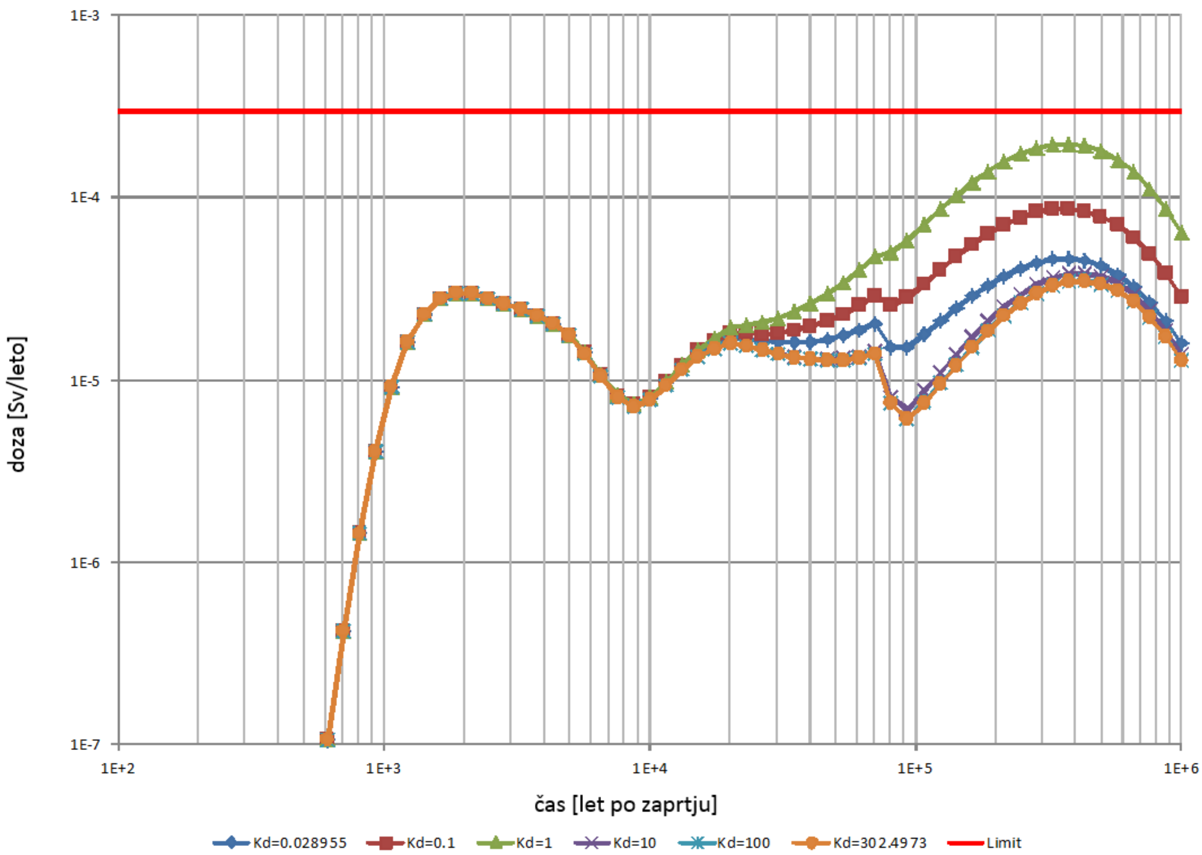

Figure 7.63: Impact of change in sorption coefficient of radium in farfield model on final dose

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.4.2 Change in degradation rate of engineered barriers

A nominal scenario in which various degradation times of engineered barriers were applied was used to analyse the impact of the degradation rate of engineered barriers on the final result. The following degradation times were used: 12,500 years, 15,000 years, 17,500 years, 20,000 years, 22,500 years and 25,000 years. These times represent a range of possible times for the degradation of engineered barriers as presented in Section 7.3.5.1 of this draft safety analysis report and taken from the safety analysis report, [75] where the time taken for the nominal scenario is 18,750 years. The results of the sensitivity analysis are illustrated in Figure 7.64 below.

It is evident from the figure that the model depends very little on the change in degradation time of engineered barriers. In no case does the dose exceed the limit of 0.3 mSv/year. In the case of a degradation time of 12,500 years, the calculated maximum is 0.04 mSv/year, 1,600 years after closure. In the case of a degradation time of 25,000 years, the calculated maximum is 0.02 mSv/year, 2,000 years after closure.

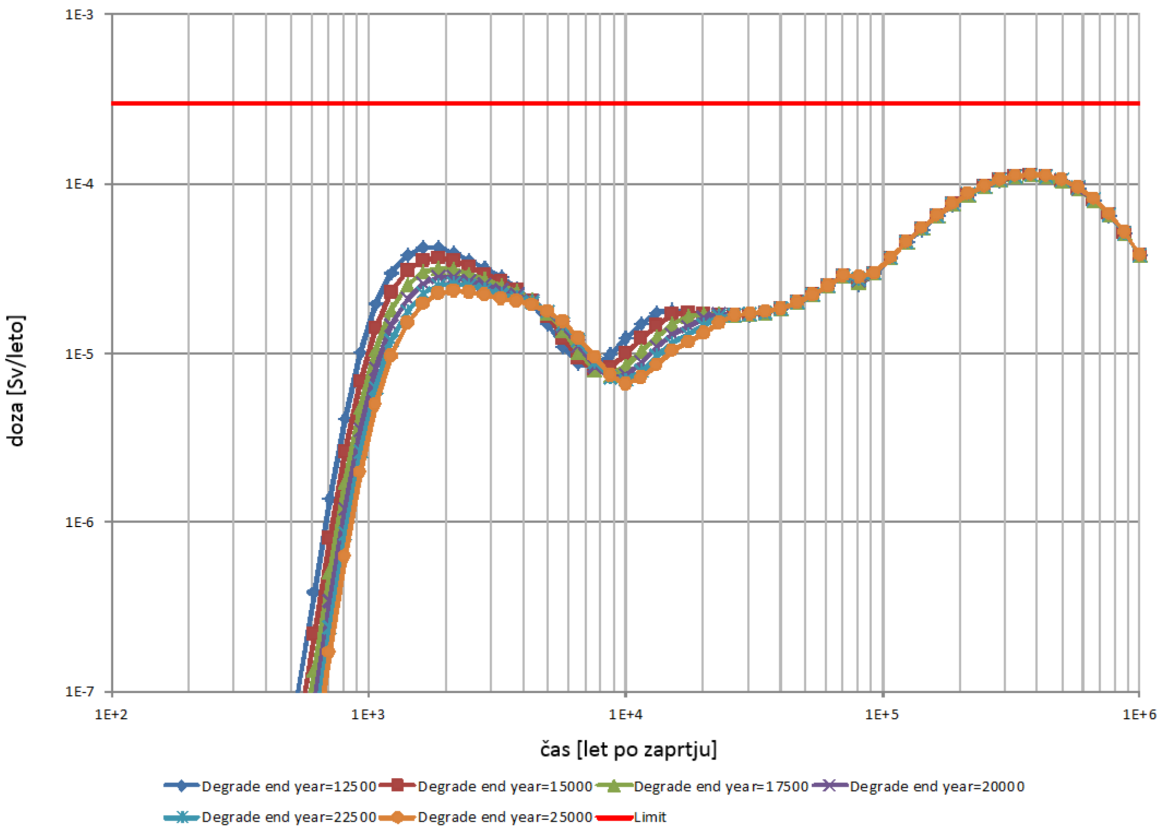

Figure 7.64: Impact of change in degradation time of engineered barriers on final calculated dose under nominal scenario

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.4.3 Change in initial degradation time under scenario of early failure of concrete barriers

Within the framework of the probabilistic calculations, analysis was conducted of the sensitivity of the final result under the scenario of the early failure of the concrete barriers to a change in the initial degradation time. The following initial degradation times were evaluated in the calculation: 300 years, 500 years, 1,000 years, 3,000 years and 5,000 years after the closure of the repository. The results are presented in Figure 7.65 below. It can be seen that the maximum dose for the first few times is virtually insensitive to this parameter, but then slowly declines. The calculated maximums are given in in Table 7.58 below.

Table 7.58: Calculated maximums for different initial degradation times under scenario of early failure of concrete barriers

Initial degradation time [years after closure of repository]	Maximum [mSv/year]
300	0.23
500	0.22
1,000	0.18
3,000	0.14
5,000	0.05

Under all initial times the maximum occurs approximately 400 years after the initial failure, which represents the time that the key radionuclides need to reach a member of the critical population group.

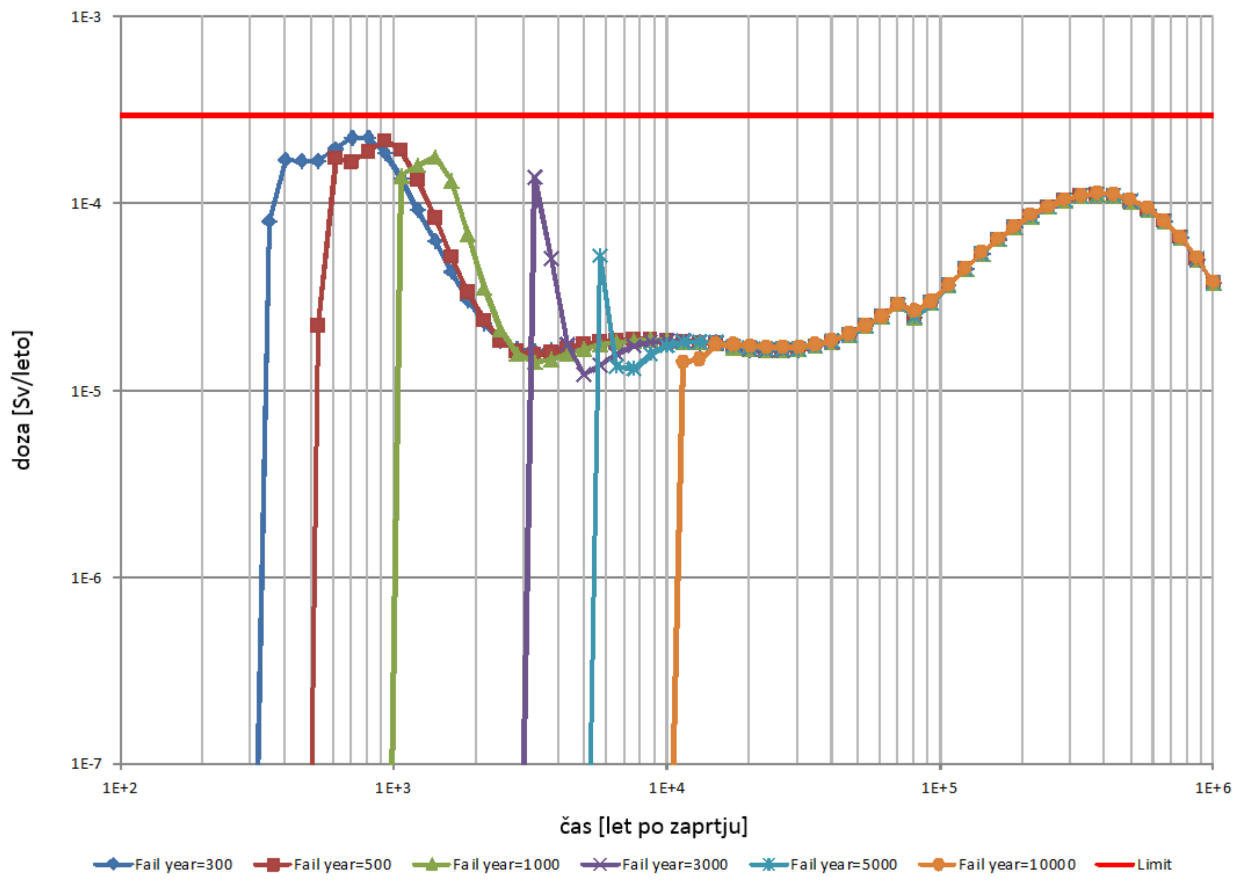

Figure 7.65: Impact of change in initial time for failure of concrete barriers on final calculated dose under scenario of early failure of concrete barriers

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.4.4 Change in flow velocity

The impact of a change in the flow velocity in the Quaternary alluvium on the final estimated dose is presented in Figure 7.66 below. The nominal velocity was determined at 58.1 m/year. [84] With regard to measurements of hydraulic conductivity, [95] possible flow velocities of 11.1 to 246.8 m/year were defined. Values of 11.1, 20, 50, 100, 150, 200 and 246.8 m/year were used in the sensitivity analysis. The nominal scenario was assumed for the calculation.

Under a flow velocity of 11.1 m/year, the calculated maximum during the first 10,000 years is 0.12 mSv/year, which occurs 3,200 years after closure. Under a flow velocity of 246.8 m/year, the maximum is attained 1,400 years after closure, and amounts to 0.01 mSv/year.

During the first period (10,000 years), higher water flow velocities in the aquifer lead to a maximum that is earlier but smaller. This occurs because the faster flow causes faster transport, and greater dilution in the geosphere and consequently in the well. The flow velocity used in the nominal scenario constitutes a central trend with regard to all the results.

A reverse correlation is observed in the later time period (after 10,000 years). Higher flow velocities yield slightly higher maximums, but the trend slows above 100 m/year and the maximums are comparable.

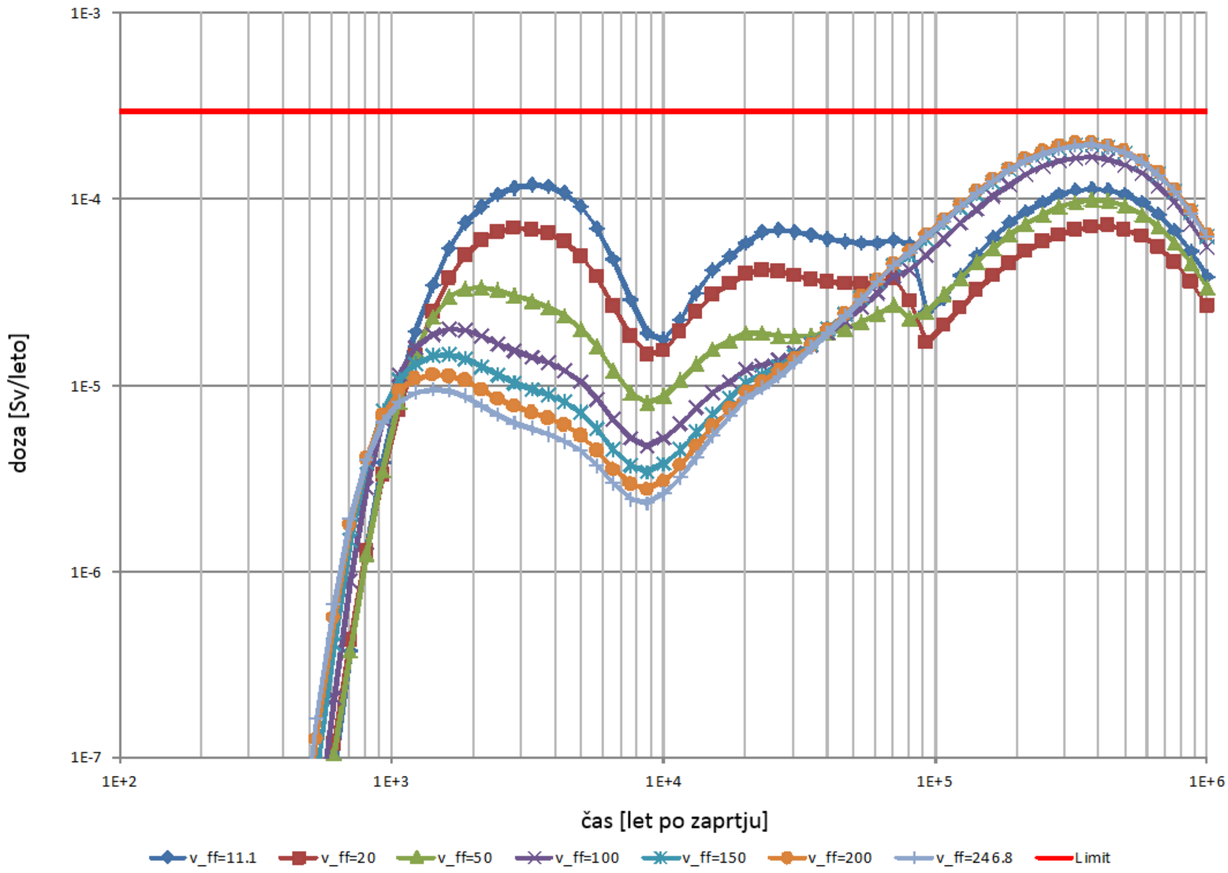

Figure 7.66: Impact of change in groundwater flow velocity in aquifer on final calculated dose under nominal scenario

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.4.5 Change in space discretisation density in model

As defined in the repository model verification report, [87] the behaviour of the model depends on the discretisation of the modelled space (the size and the number of cells into which the model is divided). As defined in the repository model verification report, [87] the behaviour of the model depends on the discretisation of the modelled space (the size and the number of cells into which the model is divided). In the Ecolego model, it is assumed in the nominal scenario that the flow from the repository to the well is discretised into ten parts (n = 10). Other values from n =1 to n = 30 were taken in the sensitivity analysis. The results of the analysis are presented in Figure 7.67 below. A small impact on the final result from a change in the parameter can be observed. For a discretisation of n = 1, the first maximum is 0.02 mSv/year and occurs 2,800 years after closure, while for n = 30, the first maximum is 0.03 mSv/year and occurs 1,900 years after closure. The results virtually match for all values of the parameter n greater than one.

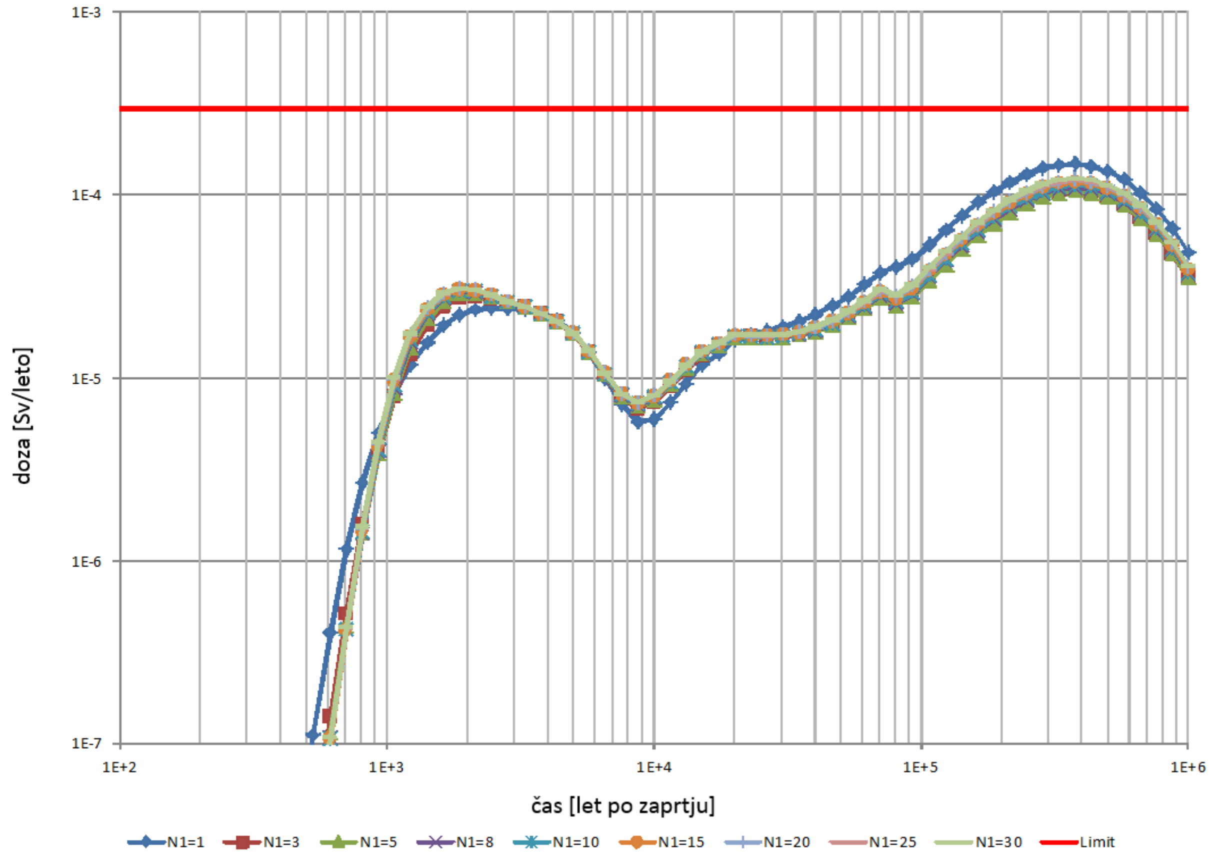

Figure 7.67: Impact of change in discretisation on final calculated dose under nominal scenario

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.7.4.6 Change in initial inventory

To conduct the analysis of the sensitivity of the calculated final dose to the initial disposal inventory, the nominal scenario was used, where the inventory in the silo was multiplied by factors of 0.5, 2, 5 and 10. The results are presented in Figure 7.68 below. The analysis was conducted primarily because, given the complexity of the nearfield model, it was unclear whether the final results are proportional to the initial inventory, which is usually the case in simple nearfield models. The results in Figure 7.68 below show this is truly the case here, and that the final doses are linearly dependent on the initial inventory. The first maximum occurs 1,900 years after the closure of the repository in all cases. The individual maximums are given in Table 7.59 below.

Table 7.59: First maximums for various initial inventories under nominal scenario

Size factor of initial inventory	Calculated first maximum [mSv/year]
0.5	0.015
1	0.03
2	0.06
5	0.15
10	0.3

The linear dependence between the inventory and the maximum indicates that the impact of solubility limits for key radionuclides is small, and that conservatively high values were assumed in the calculations.

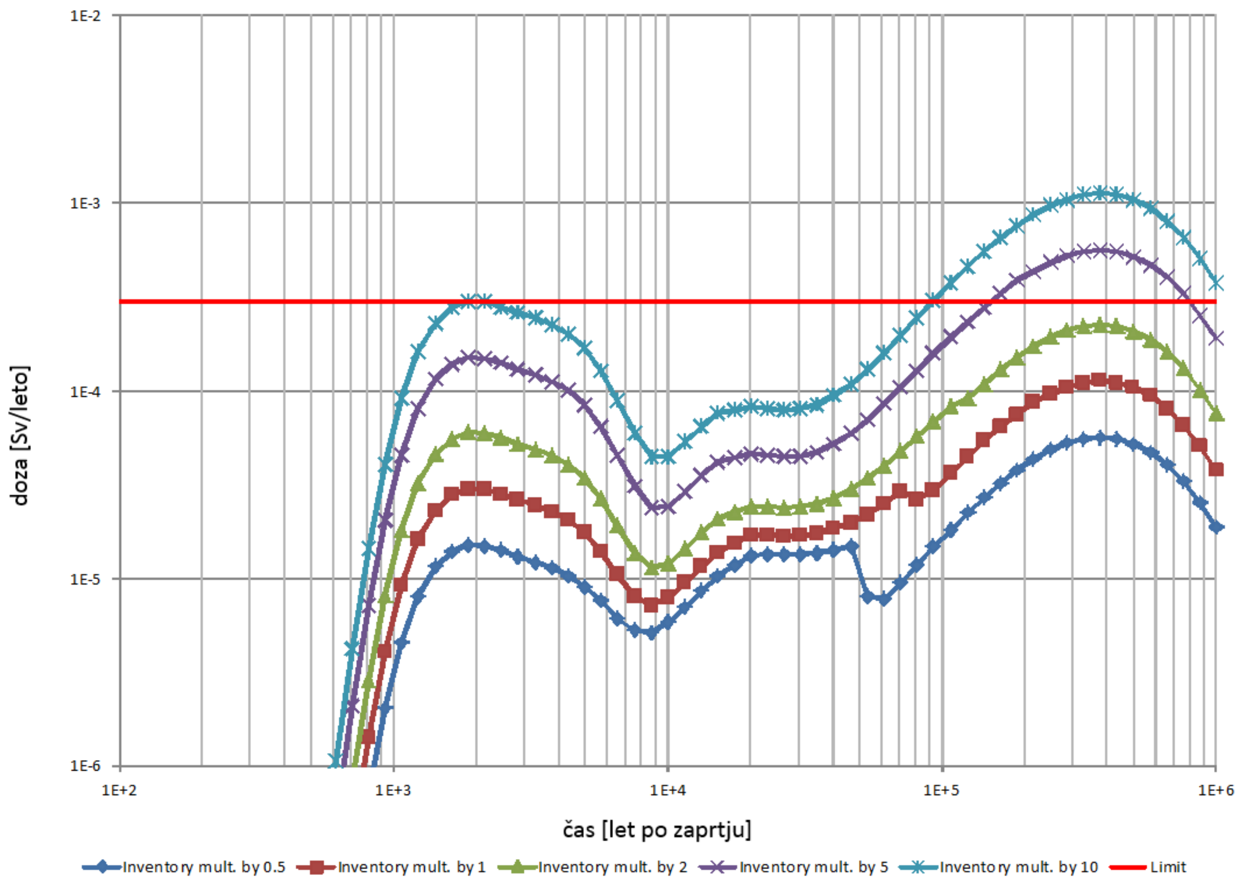

Figure 7.68: Impact of change in initial inventory on final calculated dose under nominal scenario

doza [Sv/leto]	dose [Sv/year]
čas [let po zaprtju]	period [years after closure]

7.3.8 ASSESSMENT OF IMPACT OF TOXIC METALS

As defined in Section 7.3.2 of this draft safety analysis report, the impact of toxic metals from the repository on a member of the critical population group was also evaluated within the framework of the safety analysis. The impact was compared with the drinking water standards. [56] The limits for individual toxic substances results are presented in Table 7.60 below.

Table 7.60: Legal limits for toxic substances that could be released from the repository

Toxic metal	Legal limit [µg/l]
chromium	50
lead	10
nickel	20
cadmium	5
selenium	10

Toxic metals in waste are mainly found in waste containing stainless steel. [68] In the safety analysis report on engineered barriers, [75] the corrosion factor (R_c) in the conditions of the repository was taken to be between 0.01 and 0.2 µm/year, for a specific area (SA) of 3.4E-3 m^2/kg and density (ρ) of 8,000 kg/m^3 for the materials in question. The release fractions for toxic metals can be estimated from the above parameters as follows:

$$R_C \cdot SA \cdot \rho$$

Equation 7.14: Release fraction for toxic metals

It ranges from 2.7E-7 to 5.4E-6 $year^{-1}$. Using these values, and the estimates of the inventory, [68] the rate of release of toxic metals from the repository can be estimated. The estimates are given in Table 7.61 below.

Table 7.61: Rate of release of individual toxic metals from repository

Toxic metal	Rate of release [g/year]
chromium	38 – 750
lead	0.6 – 12
nickel	48 – 970
cadmium	0.008 – 0.2
selenium	0.03 – 0.7

From the above data it was estimated whether the releases are below the limits cited in Table 7.60. The approach to calculating the concentrations of metals in groundwater from the well was as described in the nominal scenario. The approach includes several conservative assumptions, such as toxic metal releases owing to corrosion coming directly into the aquifer without taking into account dilution, dispersion, sorption or any other reaction. The sorption coefficients taken into account in the alluvial aquifer are cited in Table 7.62 below. All the coefficients other than those for chromium were obtained during on-site investigation at the site itself, [95] while the sorption coefficient for chromium was taken from the recommendation of the EPA in the USA. [96], [97]

Table 7.62: Sorption coefficients used to estimate concentrations of toxic elements from repository in alluvial aquifer

Toxic element	**K_d [m³/kg]**
chromium	0.018
lead	0.22
nickel	0.31
cadmium	0.24
selenium	0.14

The concentrations of individual contaminants in the groundwater were calculated using Equation 7.15: as follows:

$$C = \frac{RF_{sol}}{Q}$$

Equation 7.15:

Where:

$C = concentration$ [μg/l]
$R = rate\ of\ release$ [μg/year]
$F_{sol} = mass\ fraction\ in\ solution$
$Q = volumetric\ flow\ rate$ [l/year]

The rates of release were taken from Table 7.61, while the mass fractions in solution can be determined from the sorption coefficients by means of the following equation:

$$F_{sol} = \frac{1}{\left((1-\phi)\rho_p K_d/\phi + 1\right)}$$

Equation 7.16:

Where:

$\rho_p = particle\ density\ of\ absorbing\ material$ [kg/m^3]
$K_d = sorption\ coefficient$ [m^3/kg]
$\phi = porosity$

The volumetric flow rate is given by the following equation:

$$Q = qhW_0$$

Equation 7.17:

Where:

$$q = Darcy\ velocity\ [\mathrm{m/s}]$$
$$h \equiv \text{thickness of saturated zone [m]}$$
$$W_0 \equiv \text{width of contamination plume [m]}$$

If Equation 7.17: and Equation 7.16: are plugged into Equation 7.15:, the concentration can be calculated.

$$C = \frac{R}{\left((1-\phi)\rho_p K_d/\phi + 1\right) q h W_0}$$

Equation 7.18

The following values were used to calculate the concentrations:

- rate of release .... taken from Table 7.61
- particle density of material .... 2,690 kg/m^3 (taken from [95])
- width of contamination plume .... 29.3 m (approximate diameter of silo)
- Darcy velocity .... 11.1 m/s (farfield model)
- thickness of saturated zone .... 3.7 m (taken from [95])

A conservative approach was taken to determining all parameters. The results of the calculations of concentrations of toxic metals are presented in Table 7.63 below.

Table 7.63: Calculated concentrations of toxic elements from repository and legal limits for drinking water

Toxic element	Calculated concentration [µg/l]	Legal limit [µg/l]
chromium	5.5	50
lead	0.0072	10
nickel	0.42	20
cadmium	0.00011	5
selenium	0.00066	10

According to the results and limits cited in Table 7.63 above, the conservatively estimated releases of toxic elements from the repository are below the prescribed limits for drinking water, as expected.

7.3.9 ASSESSMENT OF IMPACT ON NON-HUMAN BIOTA

The estimated concentrations of radionuclides under the nominal scenario were also used within the framework of the safety analysis to assess the consequences that these concentrations could have for non-human biota. The maximum concentrations for the river (Figure 7.69) were taken into account in the estimation of the doses for aquatic organisms, while the maximum concentration in soil (in the area of root systems) was taken into account for other organisms (Figure 7.70).

The ERICA software package, which is described in detail in the safety analysis report on software, [45] was used to assess the impact. The assessment was made for reference organisms.

The ICRP recommendations [60] for derived consideration reference levels (DCRLs) recommend value of between 4 and 40 µGy/h for reference animals and plants. The DCRLs do not constitute strict limits, but values at which radiation could have a certain impact on organisms. The calculated dose concentrations presented in Table 7.64 below are several orders of magnitude below the DCRLs.

land organisms	total dose rate [µGy/h]	freshwater organisms	total dose rate [µGy/h]
invertebrate detritivores	1.74E-08	amphibians	1.18E-07
invertebrates (worms)	1.76E-08	molluscs	2.88E-06
snails	6.66E-09	snails	2.97E-06
amphibians	9.57E-09	crustaceans	2.99E-06
bird eggs	1.56E-08	benthic fish	2.53E-06
birds	9.91E-09	insect larvae	5.81E-06
flying insects	6.64E-09	birds	1.21E-07
reptiles	9.32E-09	aquatic mammals	1.21E-07
small mammals (rats)	1.93E-08	zooplankton	2.03E-07
large mammals (deer)	8.29E-09	phytoplankton	7.71E-10
mosses and lichens	2.17E-09	higher plants	3.06E-06
grasses and herbs	1.38E-08		
bushes and shrubs	7.83E-09		
trees	6.02E-09		

Table 7.64: Estimated doses for non-human biota

land organisms	total dose rate [µGy/h]	freshwater organisms	total dose rate [µGy/h]
invertebrate detritivores	1.74E-08	amphibians	1.18E-07
invertebrates (worms)	1.76E-08	molluscs	2.88E-06
snails	6.66E-09	snails	2.97E-06
amphibians	9.57E-09	crustaceans	2.99E-06
bird eggs	1.56E-08	benthic fish	2.53E-06
birds	9.91E-09	insect larvae	5.81E-06
flying insects	6.64E-09	birds	1.21E-07
reptiles	9.32E-09	aquatic mammals	1.21E-07
small mammals (rats)	1.93E-08	zooplankton	2.03E-07
large mammals (deer)	8.29E-09	phytoplankton	7.71E-10
mosses and lichens	2.17E-09	higher plants	3.06E-06
grasses and herbs	1.38E-08		
bushes and shrubs	7.83E-09		
trees	6.02E-09		

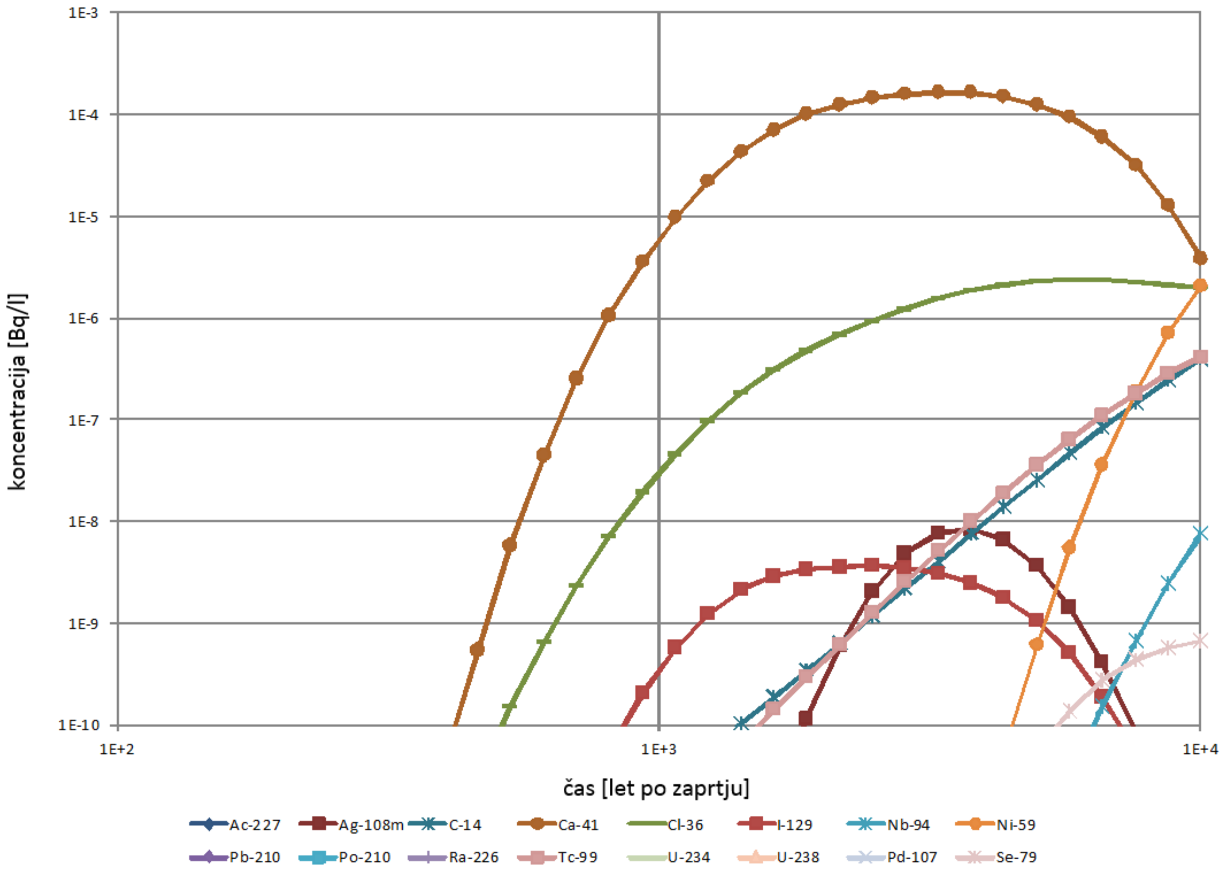

Figure 7.69: Maximum concentrations in river used to assess impact on aquatic organisms

koncentracija [Bq/l]	concentration [Bq/l]
čas [let po zaprtju]	period [years after closure]

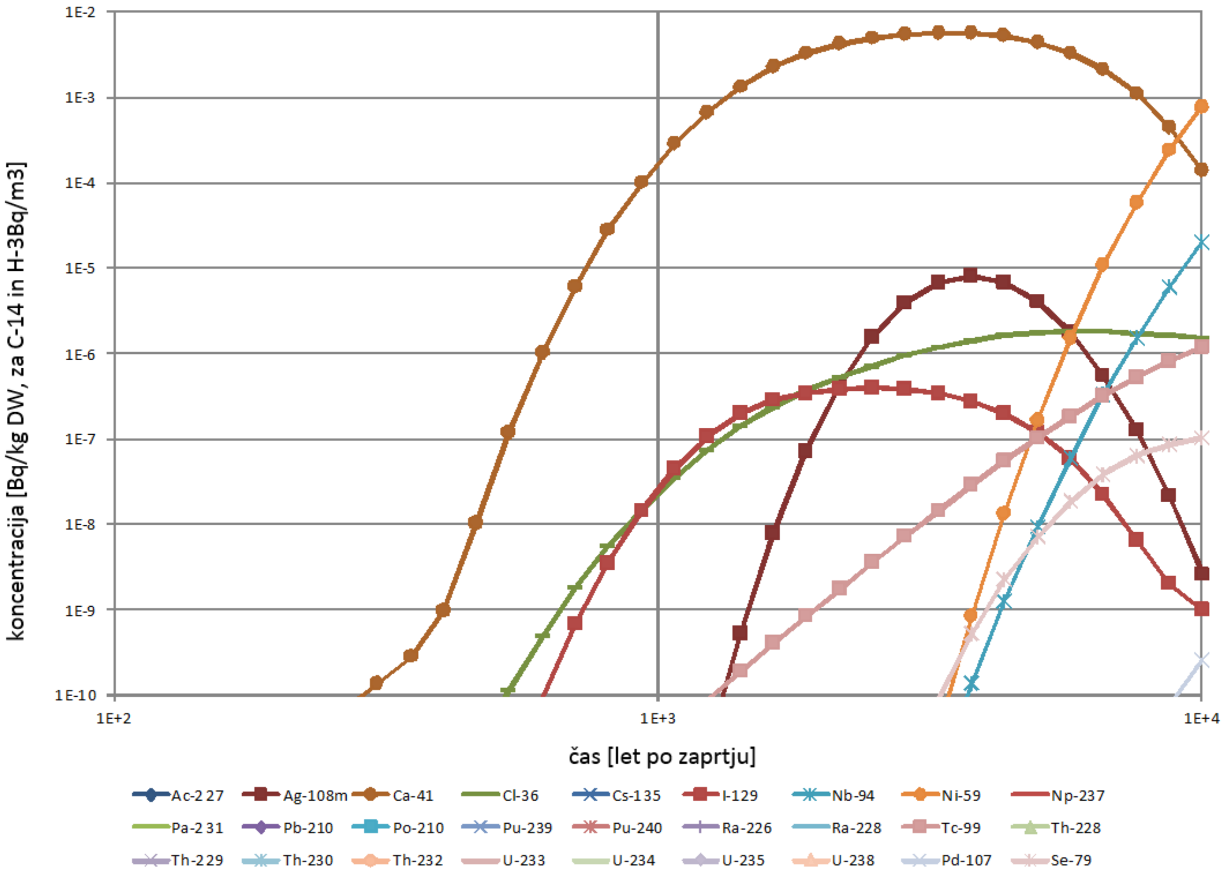

Figure 7.70: Concentrations in soil used to assess impact on land organisms

koncentracija [Bq/kg DW, za C-14 in H-3Bq/m3]	concentration [Bq/kg DW, for C-14 and H-3Bq/m3]
čas [let po zaprtju]	period [years after closure]

7.3.10 CONCLUSIONS OF SAFETY ANALYSIS FOR POST-CLOSURE PERIOD

The safety analysis for the LILW repository was conducted for the entire inventory of LILW generated in Slovenia (Scenario SA.2), but the impact of the disposal of smaller or larger quantities of waste was also assessed in the section on uncertainty. The parameters of the water flow through the disposal system and the site were obtained through the use of detailed models of the nearfield and farfield. These were then taken in the system model, which was used to conduct the sensitivity analysis. In the system model it was assumed that the entire inventory is disposed of in one disposal silo, where the following assumptions also applied:

- given the conservative use of one-dimensional vertical flow through the repository, the impact of the construction of a second silo on the vertical flow is negligible,
- distributing the waste between the two silos would lower the specific concentration i.e. specific activity, but in the models none of the radionuclides are restricted by any solubility limits, so that this does not have any impact on the distribution model,

- distributing the waste between the two silos would increase the surface area for potential releases (it would double), which would entail a reduction in concentrations in both the well and the contamination plume.

In light of the above reasoning, it can be concluded that the assumed disposal of the entire inventory in a single silo is the conservative approach, and the results thus obtained constitute the upper envelope of the impact of the repository on the environment and on people.

From the results of the deterministic safety analysis, it follows that:

- The results of the safety assessment under the nominal scenario are below the prescribed limit of 0.3 mSv/year for a member of the critical population group, despite the conservative assumption of the use of water from the well.
- The results obtained under the scenario without the use of water from the well disclose much lower doses, of the order of 10^{-6} mSv/year. This illustrates the major contribution of well water to the total estimated dose. With the more realistic use of water in the nominal scenario, the estimated doses under the nominal scenario would be much lower, meaning that the safety margin for the repository would be much larger than indicated by the nominal scenario at first glance.
- The results of the scenario of alternate failure of engineered barriers (sequential failure of barriers) show later and lower maximum doses than under the nominal scenario. It is thought that the use of this sub-scenario is more realistic than the nominal scenario itself, which further increases the safety margin of the repository.
- Under the scenario of the early failure of engineered barriers, the maximum estimated dose is 3.24 mSv/year, which is still below the legal limit of 10 mSv/year for which additional measures are required under alternate evolution scenarios. This scenario could also be described as a scenario without all engineered barriers, as no combination of FEPs leads to the realisation of such a scenario.
- The result of the river meandering and surface erosion scenario is a very low estimated dose of 10^{-6} mSv/year.
- The dose results under the scenario of change to the hydrological conditions are comparable to the doses under the nominal scenario at the time of 10,000 years after the closure of the repository. These results indicate that the nominal scenario represents a good basis for decision-making by the regulatory body.

The probabilistic analysis and calculation for the period of 10,000 years after closure show the following:

- The deterministic results for the nominal scenario constitute a conservative safety assessment, but not extremely conservative, given the uncertainty in the parameters used. Moreover, the calculation under the nominal scenario is also below the permissible limits for the 95th percentile for the analysed parameters.
- The results under the scenario without the use of the well show very low doses throughout the period covered in the safety analysis, and constitute a large safety margin in comparison with the regulatory limit.
- The results for the river meandering and erosion scenario show a large safety margin in the calculations for the 95th percentile for the analysed parameters.

Within the framework of the sensitivity analysis, the following conclusions can be drawn:

- During the period more than 10,000 years after the closure of the repository, a change in the sorption coefficient for Ra-226 shows a non-linear relationship between the maximum and sorption.
- The impact of the degradation rate of the engineered barriers on the maximums for individual radionuclides in the period more than 10,000 years after the closure of the repository was shown to be only slightly significant.
- The groundwater flow velocity in the aquifer was a significant parameter in the period up to 10,000 years after the closure of the repository. Higher velocities lead to greater dilution and lower doses. The velocity used in the nominal scenario was shown to be conservative and credible.
- For the period more than 10,000 years after the closure of the repository, the sorption of individual radionuclides into the materials of the engineered barriers in the nearfield of the repository is a significant parameter.
- The discretisation of the system model of the farfield proved to be an insignificant parameter.
- The dose rate and the impact of the repository are directly proportional to the size of the inventory, which is the result of the choice of generic solvency parameters, which yield a conservative assessment of the solvency limits used and a conservative assessment of the final results.

Analysis of the impact of non-radioactive toxic elements showed that the repository and its potential releases meet the Slovenian standards for drinking water. The evaluation of the repository’s impact on non-human biota shows a very low dose rate in comparison with the current ICRP recommendations.

The results of the analysis of inadvertent human intrusion show that the highest dose rates in the event of intrusion are faced immediately after the end of institutional controls. In this case the estimated dose for the intruder is 0.05 mSv/year, and the maximum dose for a person living in the area after the intrusion is slightly over 10 mSv/year. Intrusions that occur after the end of institutional controls result in lower dose rates. The estimated results show that radionuclides contained in activated metals at the repository make the greatest contribution to the dose in the event of intrusion. The analysis, however, does not take account of the problems (unfeasibility) of intrusion (drilling through metal with equipment used for geotechnical drilling). Given the repository design concept, the probability of an intrusion event is also extremely low, and the analysis can be defined as conservative. The estimated impact is still below the limit of 100 mSv/year at which additional optimisation would have to be undertaken in line with the requirements. It is assumed on this basis that additional optimisation (measures to mitigate the consequences) is not required for the Vrbina LILW repository, and that 300 years is a suitable duration of institutional controls for the LILW repository.

The safety margin in the security analysis comes from the conservative approach taken to safety analysis. This is reflected in several areas:

- the scenarios were selected with conservative assumptions despite a low probability of occurrence,
- multiple sub-scenarios were selected,
- tested models were used,

- the parameters used in the models were chosen conservatively.

The minimum safety margins deriving from the above assumptions are thus presented for the deterministic calculations in the individual sections, which represent the results for the individual scenarios.

In the safety analysis it was therefore assessed that the impact of the repository after closure is below the prescribed limits, and is negligible.

7.3.11 **UNCERTAINTY MANAGEMENT AND SENSITIVITY ANALYSIS**

Uncertainty and sensitivity analysis are addressed in detail in Section 7.3.7 of this draft safety analysis report. The uncertainties addressed can be divided into the following:

- uncertainties inherent in the **models used** to produce the safety analysis:
  - discretisation density in the model,
- uncertainties inherent in the **scenarios of the evolution** of the repository after closure:
  - probabilistic analysis of the nominal scenario with regard to various parameters (Section 7.3.7.1),
  - probabilistic analysis of the nominal scenario without the well, with regard to various parameters (Section 7.3.7.1.1),
  - probabilistic analysis of the scenario of the early failure of concrete barriers, with regard to various parameters (Section 7.3.7.2),
  - probabilistic analysis of the scenario of river meandering and surface erosion, with regard to various parameters (Section 7.3.7.3),
- uncertainties inherent in the **input data**:
  - corrosion times for stainless steel and carbon steel,
  - concrete degradation time,
  - Darcy velocity in the repository farfield model,
  - thickness of the saturated zone of the alluvial aquifer,
  - length of flow pathway from repository to river,
  - flow rate of Sava,
  - concentration ratios,
  - sorption coefficients,
  - initial inventory.

Given the above uncertainties, it is necessary to devote particular attention to this issue in the next phases, and to reduce uncertainty.

The following conclusions can be drawn from the analysis:

- During the period more than 10,000 years after the closure of the repository, a change in the sorption coefficient for Ra-226 shows a non-linear relationship between the maximum and sorption.
- The impact of the degradation rate of the engineered barriers on the maximums in the period more than 10,000 years after the closure of the repository was shown to be only slightly significant.
- The groundwater flow velocity in the aquifer was a significant parameter in the period up to 10,000 years after the closure of the repository. Higher velocities lead to greater

dilution and lower doses. The velocity used in the nominal scenario was shown to be conservative and credible.

- For the period more than 10,000 years after the closure of the repository, sorption in the nearfield of the repository is a significant parameter.
- The discretisation of the system model of the farfield proved to be an insignificant parameter.
- The dose rate is directly proportional to the size of the inventory, which is the result of the choice of generic solvency parameters, which yield a conservative assessment of the solvency limits used and a conservative assessment of the final results.

The following need to be clearly defined in the next phase of the analysis:

- the impact of the non-homogeneity of the waste on doses for employees and members of the public (under the scenario, operational waste will first be disposed of, and only after will decommissioning waste follow),
- the impact of doses on different age groups will be defined in detail.

REFERENCES

[1] The Safety Case and Safety Assessment for the Disposal of Radioactive Waste, SSG-23. IAEA, 2012.

[2] Posebna varnostna analiza za umestitev odlagališča NSRAO, Lokacija Vrbina v občini Krško (*Special safety analysis for the siting of the LILW repository, Vrbina site in the Municipality of Krško*), December 2006. ARAO, DDC, ZVD, ZAG and Imos Geateh.

[3] Odlagališče NSRAO Vrbina, Študija variant (*Vrbina LILW repository, Study of alternatives*), Rev. 1, NSRAO-Vrb.ŠV/ŠV 01/06, T-2136, December 2006. Acer Novo mesto d.o.o., Savaprojekt d.d.

[4] Državni prostorski načrt za odlagališče NSRAO na lokaciji Vrbina v občini Krško (*Detailed plan of national importance for the LILW repository at the Vrbina site in the municipality of Krško*), Adopted document, 07-180-00, NSRAO - Vrb-pDPN 01-09, 02-01-067-006, December 2009. Acer Novo mesto d.o.o., Savaprojekt d.d.

[5] Odlagališče NSRAO Vrbina, Krško, Idejna zasnova (*Vrbina LILW repository, Conceptual Design*), Rev. C. 2016.

[6] Splošen pregled poročila o varnostni oceni (*General safety assessment context*) O2-SWG-003-01-slo, ARAO - EISFI - TR-(11)-15 Vol 5. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[7] Varnostna ocena po zaprtju (*Post-closure safety assessment*), Summary report, Issue 1, NSRAO2-PCS-001-01-slo, ARAO, EISFI-TR-(11)-15 Vol. 1, Rev. 1. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[8] Obratovalna varnostna ocena (*Operational safety assessment*), Summary report, Issue 2, NSRAO2-OPS-001-01, ARAO, EISFI-TR-(11)-15 Vol. 4, Rev. 2. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[9] WAC zbirno poročilo – povzetek in priporočila za optimizacijo (*WACs summary report: summary and recommendations for optimisation*), Issue 1, NSRAO2-WAC-001-01-slo, ARAO, EISFI-TR-(11)-15 Vol. 3, Rev. 1. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[10] Poročilo o optimizaciji varnosti po zaprtju (*Report on post-closure optimisation*), Issue 1, NSRAO2-PCS-002-01-slo, ARAO, EISFI-TR-(11)-15 Vol. 2, Rev. 1. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[11] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Operational Safety Assessment Context Report, ARAO, EISFI-TR-(11)-11 Vol. 1, NSRAO2-OPS-002-01. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[12] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, System description for operational safety assessment, ARAO, EISFI-TR-(11)-11 Vol. 2, NSRAO2-OPS-003-001-eng. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[13] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Operational Safety Assessment Report on Scenarios, Models and Results of Calculations, ARAO, EISFI-TR-(11)-11 Vol. 3, NSRAO2-OPS-004-01-e. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[14] Revizija in optimizacija projektnih rešitev iz IDP – SILOS (*Revision and optimisation of design solutions from the preliminary design: silo*). NSRAO2-ŠTU-002-01, IBE d.d., 2011.

[15] Revizija in optimizacija projektnih rešitev iz IDP - Tehnologija odlaganja (*Revision and optimisation of design solutions from the preliminary design: disposal technology)*, NSRAO2-ŠTU-003-01, NRVB 3x1060. IBE d.d., 2011.

[16] Optimizacija neodlagalnega dela odlagališča (*Optimisation of the non-disposal part of the repository*). NSRAO2-ŠTU-014-01, IBE d.d., 2014.

[17] Safety analysis and waste acceptance criteria preparation for low and intermediate level waste repository in Slovenia, Phase II and III, Revised Operational Safety Assessment, ARAO, EISFI-TR-(15)-37 Vol. 1, NSRAO2-PCS-019-01-eng. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2016.

[18] Radiation Protection and Nuclear Safety Act (ZVISJV) (Official Gazette of the RS, 102/04 [official consolidated version], 70/08 [ZVO-1B], 60/11 and 74/15), 2015.

[19] EISFI consortium, Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Inventory Report. Report No. EISFI-TR-(11)-12 Vol. 1, Rev. 4, NSRAO2-WAC-002-01-eng. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2015.

[20] Decree on dose limits, reference levels and radioactive contamination (UV2). (Official Gazette of the RS, 18/18).

[21] Rules on radiation and nuclear safety factors (JV5). (Official Gazette of the RS, 74/16).

[22] Razširjeno poročilo o varstvu pred ionizirajočimi sevanji in jedrski varnosti v Republiki Sloveniji za leto 2015 (*Expanded annual report on ionising radiation protection and nuclear safety in the Republic of Slovenia for 2015*). SNSA, 2016.

[23] The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste, GSG-3. Vienna: IAEA, 2013.

[24] Safety Assessment Methodologies for Near Surface Disposal Facilities, Results of a Coordinated Research Project, Volume 2: *Test cases, IAEA - ISAM*.

[25] "SAFRAN 2.3.2.0. http://www.safran.facilia.se/safran/show/HomePage".

[26] Odlagališče NSRAO Vrbina, Krško, Investicijski program (*Vrbina LILW repository, Feasibility Study*) Rev. *C.* IBE d.d., 2013.

[27] Projektne osnove za odlagališče NSRAO Vrbina, Krško - faza presoje vplivov na okolje (*Design bases for the Vrbina LILW repository: environmental impact assessment phase*), Revision 1, 02-08-011-001/NSRAO2-POR-013-01. ARAO, 2016.

[28] Obratovanje (*Operation*), reference document for draft safety analysis report, NSRAO2-POR-020-01, 02-08-001-003, NRVB 5X/M23. IBE d.d., 2016.

[29] Obratovalni pogoji in omejitve (*Operating conditions and constraints*), reference document for draft safety analysis report, NSRAO2-POR-027-00 02-08-011-003. IBE, 2016.

[30] Praktične smernice - Vsebina varnostnega poročila za odlagališče nizko in srednje radioaktivnih odpadkov (*Practical Guidelines – Content of the safety report for the low- and intermediate-level radioactive waste repository*). 2012.

[31] Določitev tveganj za začetne dogodke obratovalne faze odlagališča NSRAO (*Determination of risks for initial events of operating phase of LILW repository*), Revision 0, 02-08-011-003. ARAO, 2019.

[32]NUREG/CR-7201: Characterizing Explosive Effects on Underground Structures, 2015

[33] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Gas Generation Processes and Design Implications, ARAO, EISFI-TR-(11)-08 Vol. 4, Rev. 1. NSRAO2-PCS-010-01-eng, 2012. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO).

[34] Decision on the Publication of amendments to Annexes A and B to the European Agreement Concerning the International Carriage of Dangerous Goods /ADR/. Official Gazette of the RS, 9/03, including amendments (Official Gazette of the RS, 9/05, 9/07, 125/08, 97/10, 14/13 and 10/15).

[35] Geological Disposal – Waste package accident performance status report. Report NDA/RWMD/032. NDA, 2010.

[36] Geological Disposal – Upstream Optioneering. Overview and uses of the 6 cubic metres concrete box. NDA Technical Note no. 18959097. NDA, 2013.

[37] Waste Package Impact Release Fraction Data. Report 24857-14-01. NDA, 2010.

[38] Release Fractions from Vault Impacts at 15 metres. Report 124857-08. NDA, 2011.

[39] Terrorist Attack Scenario Analysis Supplementary Report to Revised Operational Safety Assessment, Technical Report ARAO, EISFI-TR-(15)-37 Vol. 1 - FOR INTERNAL USE ONLY. EISFI consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2016.

[40] EISFI Consortium, Preparation for Low and Intermediate Level Waste Repository in Slovenia - Operational safety assessment report on Scenarios, Models and Results of Calculations, Rev. 1, 2012.

[41] W. Rodwell, GASNET - A Thematic Network on Gas Issues in Safety Assessment of Deep Repositories for Radioactive Waste: Final Report on the Treatment in Safety Assessments of Issues Arising from Gas Generation. European Commission Report EUR EN.

[42] IAEA, Generic Models for Use in Assessing the Impact of Discharges of Radioactive Substances to the Environment, SRS No. 19, 2001.

[43] Compendium of Dose Coefficients based on ICRP Publication 60. ICRP Publication 119. Ann. ICRP 41 (Suppl.). ICRP, 2012.

[44] K.F. Eckerman and J.C. Ryman, External Exposure to Radionuclides in Air, Water, and Soil. Federal Guidance Report No. 12. EPA-402-R-93-081. 1993.

[45] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Software quality assurance report, ARAO, EISFI-TR-(11)-08, Vol. 9, Rev. 1, NSRAO2-PCS-015-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[46] S.G. Homann and S.G. Aluzzi, HotSpot Health Physics Codes Version 3.0 User's Guide. LLNL-SM-636474. National Atmospheric Release Advisory Center, Lawrence Livermore National Laboratory, Livermore. 2014.

[47] Ovrednotenje DCF faktorjev za izračun efektivnih doz, ki jo lahko prejme posameznik v

posamezni starostni skupini prebivalstva za odlagališče (*Evaluation of DCFs for calculating effective doses that can be received by individuals in particular age groups for the LILW repository in the event of accidents*). ARAO. NSRAO 02-08-030 NSRAO2-OPS-006-00. 2018.

[48] Z. Petkovšek, Emisijski potencial SO2 za večino kotlin Slovenije (*Emission potential of SO_2 for majority of basins in Slovenia*), Proceedings of the Slovenian Meteorologists' Society, 23, 2, pp 37-49. [COBISS.SI-ID 1327716]. 1979.

[49] Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom a. Official Journal of the EU, Volume 57, January 2014.

[50] IAEA, Safety Assessment Methodologies for Near Surface Disposal Facilities, Results of a Coordinated Research Project, Volume 1: Review and Enhancement of Safety Assessment Approaches and Tools, IAEA – ISAM. 2004.

[51] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate level Waste Repository in Slovenia, Assessment context report, ARAO, EISFI-TR-(11)-04, Rev. 2, NSRAO2-PCS-003-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[52] Radiation protection recommendations as applied to the disposal of long-lived solid radioactive waste. ICRP Publication 81, Ann. ICRP, Vol. 28, 4. ICRP, 2000.

[53] Environmental Protection: the Concept and Use of Reference Animals and Plants. ICRP Publication 108, Ann. ICRP Vol. 38 Issue 4-6. ICRP, 2009.

[54] Risk-Based Classification of Radioactive and Hazardous Chemical Wastes. Report 139 of the US National Council on Radiation Protection and Measurements (NCRP). Bethesda. NCRP, 2002.

[55] *Management of low and intermediate level radioactive wastes with regard to their chemical toxicity.* IAEA-TECDOC-1325. International Atomic Energy Agency, Vienna. IAEA, 2002.

[56] Rules on drinking water. Official Gazette of the RS, 19/04.

[57] Assessing Dose of the Representative Person for the Purpose of the Radiation Protection of the Public. ICRP Publication 101a Ann. ICRP 36 (3). ICRP, 2006.

[58] J.A. Adam and V.L. Rogers, A Classification System for Radioactive Waste Disposal – What Waste Goes Where. Status Report on Waste Classification 1 October 1977 to 31 May 1978, US Nuclear Regulatory Commission Report NUREG-0456.

[59] Draft Environmental Impact Statement: Licensing Requirements for Land Disposal of Radioactive Waste. NUREG-0782 Vol. 1. US Nuclear Regulatory Commission, Washington, DC. USNRC, 1981.

[60] The 2007 Recommendations of the International Commission on Radiological Protection, Publication 103, Elsevier. ICRP, 2007.

[61] http://www.project.facilia.se/erica/download.html.

[62] Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1. Vienna: IAEA, 2006.

[63] Safety Indicators in Different Time Frames for the Safety Assessment of Underground Disposal of Radioactive Waste Repositories. IAEA-TECDOC-767. International Atomic Energy Agency, Vienna. IAEA, 1994.

[64] The Handling of Timescales in Assessing Post-Closure Safety. Nuclear Energy Agency. NEA, 2004.

[65] Nuclear Energy Agency Radioactive Waste Management Committee Integration Group for the Safety Case (IGSC), Workshop on Handling of Time Scales Assessing Post-Closure Safety, NEA/RWM/IGSC(2002)6. NEA, 2002.

[66] Reference Biospheres for Solid Radioactive Waste Disposal. Report of BIOMASS Theme 1 of the BIOsphere Modelling and ASSessment (BIOMASS) Programme IAEA-BIOMASS-6. International Atomic Energy Agency Safety, Vienna. IAEA, 2003.

[67] Inventory Screening for Post-Closure Safety Assessment. Report No. EISFI-TR-(11)-08, Vol. 5, Rev.1, May 2012, NSRAO2-PCS-011-01-eng. ARAO (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[68] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Report on inventory of radionuclides and other toxic materials, EISFI-TR-(11)-05 Rev. 2, NSRAO2-PCS-004-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[69] Odlagališče NSRAO Vrbina, Krško, Idejni projekt (*Vrbina LILW repository, Preliminary Design*), Rev. A. NSRAO-Vrb-IDP 01/09, IBE d.d., 2009.

[70] Consortium EISFI, Report on safety relevance of the design optimisation proposals, NSRAO2-SWG-004-01-eng. 2015.

[71] Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4. IAEA.

[72] Report on initial scenarios under post-closure conditions, EISFI-TR-(11)-07, Rev 1, NSRAO2-PCS006-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[73] NEA, Features, Events and Processes (FEPs) for Geologic Disposal of Radioactive Waste - An International Database. 1999.

[74] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Post-closure safety assessment model report, ARAO, EISFI-TR-(11)-08 Vol. 6, Rev. 1, NSRAO2-PCS-012-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[75] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Evolution of the Engineered Barrier System, ARAO, EISFI-TR-(11)-08 Vol. 3, NSRAO2-PCS-009-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[76] Izdelava metodologije za presojo varnosti inženirskih oregrad odlagališča nizko in srednje radioaktivnih odpadkov (Methodology for safety assessment of engineering barriers for low and intermediate level waste repository), Third phase, Final report. University of Ljubljana, FGG, 2006.

[77] J.C. Walton, L.E. Plansky, and R.W. Smith, Models for Estimation of Service Life of

Concrete Barriers in Low-Level Radioactive Waste Disposal, NUREG/CR-5542. 1990.

[78] USDOE, *The US Department Of Energy Idaho National Laboratory Site Draft Section 3116 Waste Determination For Idaho Nuclear Technology And Engineering Center Tank Farm Facility. Appendix E. Degradation Analysis of the Grouted Tank/Vault and Piping System*. 2006.

[79] K.H. Subramaninan, Life Estimation of High Level Waste Tank Steel for F-Tank Farm Closure Performance Assessment, Rev.1. WSRC-STI-2007-00061, Rev. 1, Savannah River National Laboratory. 2007.

[80] R.N. Swamy, Alkali-Aggregate Reaction - The Bogeyman of Concrete, Concrete Technology: Past, Present and Future. The Proceedings of the V. Mohan Malhotra Symposium ACI SPI-144, P.K. Mehta (ed.), American Concrete Institute, pp 105-140. 1994.

[81] Poročilo o izvedbi programa dopolnilnih začetnih terenskih in laboratorijskih raziskav geosfere in hidrosfere za potencialno lokacijo Vrbina-Krško (*Report on supplementary initial field and laboratory geosphere and hydrosphere research for the potential Vrbina-Krško site*), Rev. 1. J.V. GeoZS, ZAG, Geoinženiring, IRGO, ZZVMB, 2009.

[82] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Near Field Flow Modelling Report, ARAO, EISFI-TR-(11)-08, Vol. 1, NSRAO2-PCS-007-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[83] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Post-Closure Safety Assessment Parameters Report, ARAO, EISFI-TR-(11)-08, Vol. 7, NSRAO2-PCS-013-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[84] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Far Field Flow Modelling Report, ARAO, EISFI-TR-(11)-08, Vol. 2, NSRAO2-PCS-008-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[85] Safety Analysis and Waste Acceptance Criteria Preparation for Low and Intermediate Level Waste Repository in Slovenia, Verification and Testing of the Safety Assessment Model Report, ARAO, EISFI-TR-(11)-08, Vol. 8, NSRAO2-PCS-014-01-eng. EISFI Consortium (ENCO, INTERA, STUDSVIK, FACILIA, IRGO), 2012.

[86] Conceptual Hydrodynamic and Transport Checking Model of the LILW Repository Impact Area, Final report - Revision 1, 02-01-069-014. HGEM d.o.o. et al., 2010.

[87] S. Kamboj, C. Yu, and D.J. LePoire, External Exposure Model Used in the RESRAD Code for Various Geometries of Contaminated Soil. ANL/EAD/TM-84, Argonne National Laboratory. 1984.

[88] B.A. Napier, Intruder Dose Pathway Analysis for the Onsite Disposal of Radioactive Wastes: The ONSITE/MAXI1 Computer Program. NUREG/CR-3620, PNL-4054, prepared by Pacific Northwest Laboratory, Richland, Washington, for US Nuclear Regulatory Commission, Division of Waste. 1984.

[89] R.L. Aaberg, W.E.J. Kennedy, and V.W. Thomas, Definition of Intrusion Scenarios and Example Concentration Ranges for the Disposal of Near-Surface Waste at the Hanford Site. PNL-6312. 1990.

[90] A.J. Baker, D.A. Lever, J.H. Rees, M.C. Thorne, C.J. Tweed, and R.S. Wikramaratna,

Nirex 97: An Assessment of the Post-Closure Performance of a Deep Waste Repository at Sellafield. Volume 4: The Gas Pathway, Nirex Science Report S/97/012, United Kingdom Nirex Ltd, Harwell, Oxfordshire. 1997.

[91] http://ecolego.facilia.se/ecolego/show/HomePage

[92] M.D. McKay, R.J. Beckman, and W.J. Conover, A Comparison of Three Methods for Selecting Values of Input Variables in the Analysis of Output from a Computer Code, Technometrics (JSTOR Abstract). American Statistical Association, pp 239-245. 1979.

[93] Glavne raziskave geo in hidrosfere za potrebe graditve odlagališča NSRAO (*Main geosphere and hydrosphere research for the construction of the LILW repository*), Rev. 1, 2015. J.V. IRGO Consulting d.o.o., GeoZS, NLZOH Maribor, Geoinženiring d.o.o., ZAG.

[94] J.D. Allison and T.D. Allison, Partition Coefficients for Metals in Surface Water, Soil, and Waste. EPA/600/R-05/74. Washington, DC. 2005.

[95] US EPA, Understanding Variation in Partition Coefficient, K_d, Values, Volume II: Review of Geochemistry and Available K_d Values for Cadmium, Cesium, Chromium, Lead, Plutonium, Radon, Strontium, Thorium, Tritium (3H), and Uranium. EPA 402-R-99-004B. US EPA. 1999.