

RETENTION, RELEASE, AND MEASUREMENT OF VOLATILE RADIONUCLIDES IN THE MORSLEBEN REPOSITORY PLANT (ERAM)

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ABSTRACT

For the operational and shut-down phases of a repository, compliance with protection goals has to be demonstrated by means of a site-specific safety analysis. With respect to the release of volatile radionuclides from the waste, semi-empirical release rates have been used. To gain confidence in these rates under actual conditions of a repository and to study possible retention and delay mechanisms, an experimental program was initiated at the Morsleben repository plant (ERAM). Measurements and data evaluations are still in progress, preliminary results and conclusions are presented.

INTRODUCTION

The Morsleben repository plant is located in a former salt mine within the federal state of Saxony-Anhalt. Some basic dates and facts concerning the history of this mine are as follows:

- 1912 The first salt hauling at the shaft Bartensleben occurs.
- 1970 The former German Democratic Republic (GDR) selects the mine as a repository for radioactive waste.
- 1971 The first emplacement of radioactive waste into the mine takes place.
- 1986 A permanent operational permit is received.
- 1990 After the German reunification the following occurs:
 - Ownership is given to the Federal Republic of Germany with BfS (Federal Office for Radiation Protection) as the operator and license owner.
 - An operational management contract is established between BfS and DBE (Deutsche Gesellschaft zum Bau und Betrieb von Endlagern für Abfallstoffe mbH).
 - A license restriction of ten years up to 06/30/2000 is established.
- 1991 The disposal is interrupted and a subsequent safety analysis by GRS (Gesellschaft für Anlagen- und Reaktorsicherheit mbH) on behalf of the Federal Ministry for the Environment, Nature Conservation & Nuclear Safety is initiated.
- 1994 Radioactive waste disposal is resumed.
- 1998
 - A prolongation of five years of the license up to 06/30/2005 is established.
 - Disposal is interrupted due to a Higher Administrative Court decision.

In September of 1998 the emplacement of radioactive waste was stopped due to a Higher Administrative Court decision. Despite the fact that formal juridical reasons and no safety-related aspects were responsible for this decision, it became evident that compliance with protection goals in the post-operational phase of the repository is difficult to demonstrate. Therefore, BfS regards further emplacement of radioactive waste to be no longer justifiable and now strives for the decommissioning of the plant. In the former German Democratic Republic (GDR) the licensing procedure did not include a long-term safety analysis equal to that of western standards. Instead, a step-wise process of repository

development accompanied by research and development work was performed. Currently, difficulties for the long-term safety analysis result from the complexity of the mine layout, the grade of rock penetration, and the occurrence of potash seams that might be dissolved as a result of brine intrusion. A three-dimensional view of the mine around shaft Bartensleben is given in fig. 1.

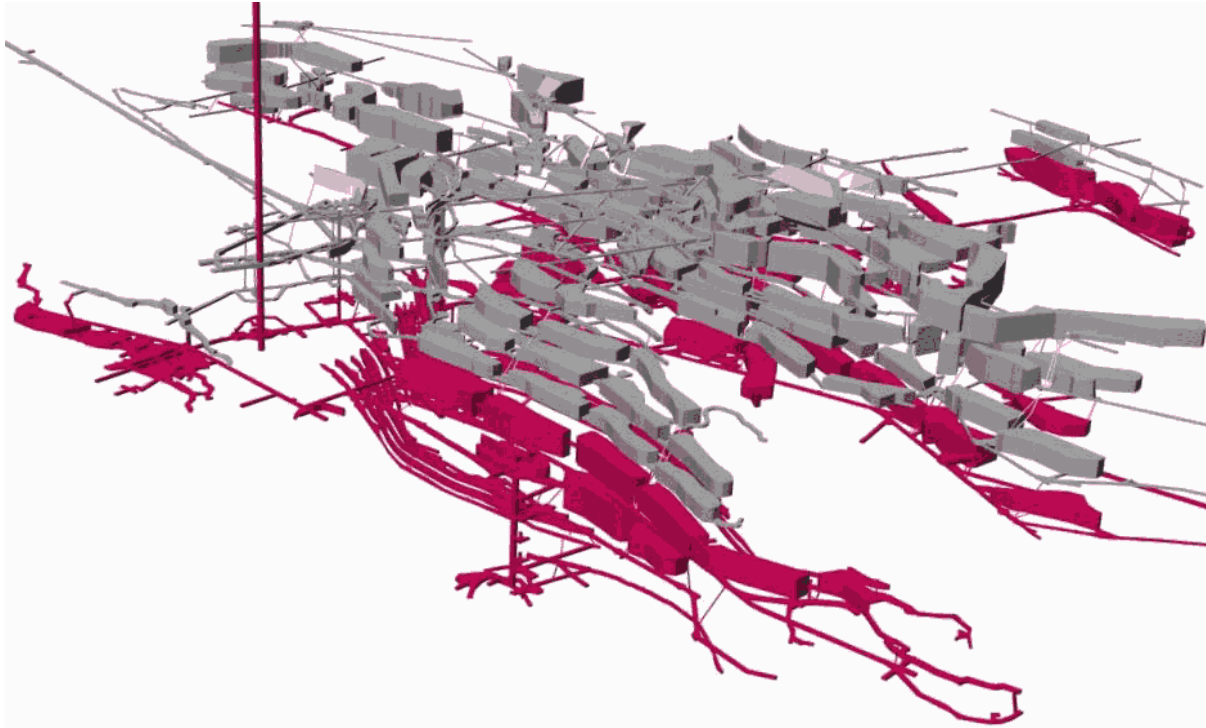


Fig. 1: Three-dimensional view of the Morsleben repository mine around the Bartensleben shaft. The restricted access area (4th level and below) is coloured in red. Conventional mine openings are grey.

As of 09/28/1998, 36,752m³ of low and intermediate-level waste with low concentrations of alpha emitters and 6,617 sealed radiation sources with an α -activity of $2.6 \cdot 10^{11}$ Bq and a β/γ -activity of $1.7 \cdot 10^{14}$ Bq have been disposed of. This was accomplished by stacking drums and boxes, dumping of solid waste, and, before 1991, by solidifying liquid wastes with filter ashes. Several emplacement fields with different characteristics, disposal techniques, and types of backfill have been used for the disposal of different waste types. These different waste types significantly influence the amount of volatile radionuclides released and/or accumulated inside the residual voids. An overview of the different emplacement fields and the amount and types of waste is given in table 1.

Table 1: Types and amount of waste disposed of in ERAM from 1971 to 1991 and since 1994.

Waste characteristics	Emplacement fields				
	Western mine field	Southern mine field	Northern mine field	Eastern mine field	
Solid waste [m³]	≤ 1991	3,431	849	1,701	
	≥ 1994	15,145	1,037	11 ¹⁾	6,139
Disposal technique	Stacking	Dumping	Stacking	Stacking	
Liquid waste [m³]	≤ 1991		8,234		
Disposal technique		Solidification with filter ashes			
Radiation sources	≤ 1991		6,223		
	≥ 1994		394		
Disposal technique		Dumping			
Total volume [m³]		18,576	10,119 ²⁾	1,712	6,139
Total activity [Bq]		3.3E+13	1.1E+14	1.0E+12	1.6E+13

1) Operational waste of ERAM

2) Without radiation sources

PRINCIPLE OF SAFETY ANALYSIS

Compliance with protection goals has to be demonstrated in the safety analyses for the operational and shut-down phases. An important aspect of these analyses is the behaviour of the volatile radionuclides, especially tritium (H-3 or T), radiocarbon (C-14), and radon (Rn-222), under the site-specific conditions of the repository. This concerns both the release from the waste matrix and the activity transport through the barriers. Rn-220 is of minor importance because of its short half life ($T_{1/2} = 55.6s$), and will not be discussed here.

With respect to the release of volatile radionuclides, the protection goal for the environment is the dose limit of 0.3mSv/a (effective dose) as established by the German Radiation Protection Ordinance. Alternatively, nuclide-specific upper limits for an annual release via the air path, which in sum result in the above-mentioned dose limit, may also be taken as protection goals. From the release rate limits G_i [Bq/a], the disposable activities, A_i [Bq], of waste containing the nuclide i (where $i = H-3, C-14, Ra-226$) were derived with the general relation

$$f_{o,i} \cdot A_{o,i} + f_{b,i} \cdot A_{b,i} < G_i \quad (\text{Eq. 1})$$

where f_i [Bq·a⁻¹/Bq = 1/a] are the relative annual release rates with respect to the inventory, with the indices "o" and "b" indicating open and backfilled disposal rooms, respectively.

For the release from the waste matrix ($f_{o,i}$) empirical values have been used (1). The tightness of the packaging was not taken into account. In the case of backfilled disposal rooms, a 90% reduction was

assumed for H-3 in the chemical form of HTO (tritium containing water) and Rn-222 was expected to decay entirely during the time of its migration through the backfill. This decay is almost complete since its half-life is only 3.8 days. For all other volatile species no retention by backfill was considered.

In general, the relative annual release rates depend on the following waste and barrier properties:

- The physico-chemical waste form and type of waste conditioning affect the release from the waste matrix ($f_{o,i}$).
- The tightness of the packaging² affects the release from the waste packages ($f_{o,i}$).
- The physico-chemical properties of the backfill and sealing materials and,
- the type and structure of the host rock affect the release from backfilled/closed disposal rooms ($f_{b,i}$).

The pore volume of the backfill material (crushed salt and electrostatic filter ashes in ERAM) is one of the characteristic features determining the radionuclide retention and sorption properties. Another characteristic factor is the pH-level of the backfill.

During operation, the emplacement of waste was accompanied or interrupted by backfilling of the residual voids with filter ashes or crushed salt. For the beginning of the shut-down phase that may last for some decades it is assumed that all emplacement fields are closed by walls and have been backfilled. Up to the time of final closure, predictions need to be made for the time dependent concentration of volatile radionuclides and their release through the barriers until long-term seals will be installed.

It is assumed that the release of volatile radionuclides from the waste into the residual air or the backfill takes place at a constant rate. Additional driving forces for further radionuclide transport are

- gas generation due to corrosion, microbial disintegration, and (possibly) radiolysis³,
- host rock convergence, and
- pressure fluctuations in the underground atmosphere.

Pressure fluctuations with different time scales are caused by changes of the outer atmospheric pressure as well as by conventional activities in the mine, i.e. shaft hoisting and car driving. An example for the release of radon decay products due to these mining activities is given in fig. 2.

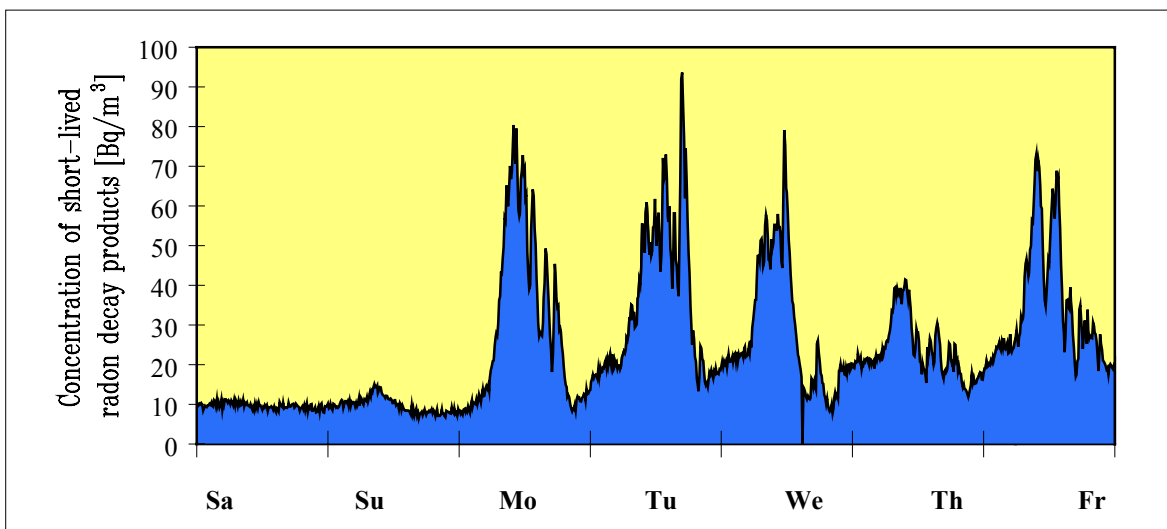


Fig. 2: Concentration of short-lived radon decay products in the mine air above room 1 of the southern mine field for a typical week.

Besides retardation by the more or less permeable seals, there may be a permanent retention inside the backfill. In the case of a non-delayed convective/diffusive transport, the retardation factor is $R = 1$. If chemical bindings or isotopic exchange between the volatile radionuclides and the backfill material occur, two cases have to be considered:

1. For the case of a reversible binding there is a certain delay until a sorption/desorption equilibrium with an equilibrium constant K is reached. The retardation factor is then given by $R = 1 + K$.
2. For the case of an irreversible binding, a permanent retention is achieved, thus $R = \infty$.

EXPERIMENTAL PROGRAM

In order to verify the release rates to be used in the licensing procedure for the shut-down, the activity concentrations of tritium (H-3), radiocarbon (C-14), and radon (Rn-222) inside the residual voids were measured. Additional measurements in the return air flow from the corresponding emplacement fields were performed as well.

The HTO and (C-14)O₂ concentrations were determined by using washing bottles. The measurements were usually carried out for some weeks with typical sampling times of 3 days. The molecular sieve measurements were not necessarily performed at the same time and there was normally only one sampling period of one or two days. Therefore, the latter measurements are less representative and deviations between the two techniques are likely to be attributed to the different times of measurement.

With regard to the release of volatile radionuclides, different disposal techniques and waste types in different emplacement fields have to be considered. An overview of the results of the above-mentioned measurements and the properties of the disposal rooms is given in tables 2 and 3. From all these measurements the following information is obtained:

- The retention and adsorbing properties of the backfill for the different physico-chemical species of the volatile radionuclides,
- The tightness of the barriers to the different disposal rooms (wall, dumping sluice), and
- The relative annual release rates, f_i , for the volatile species.

Based on this information the following conclusions can be drawn:

RETENTION BY PHYSICO-CHEMICAL EFFECTS

From practical experience H-3 and C-14 are mainly expected to be released from the waste matrix as HTO and (C-14)O₂. This is indicated by measurements in the residual cavities of the northern mine field. Up to now, this mine field is not backfilled and the emplacement drifts are only closed by walls. In contrast, the residual air volume of room 1s of the western mine field has a ratio of HTO/HT that is drastically reduced to less than 1:100. It is assumed that HTO reacts with the hygroscopic filter ashes that require water for their setting reaction. The waste packages are completely covered and only small residual cavities are left between the backfill and the roof of room 1s. For this reason, only a small amount of tritium containing water seems to be able to penetrate the ash layer and reach the residual volume. Furthermore, HTO is effectively captured by isotopic exchange with the residual humidity of the

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waste, the backfill and the sealing material. Examples for this have been measured for different materials (2).

Measurements of both chemical forms of tritium (oxidised and non-oxidised) in the return air of the western mine field were carried out at the end of 1997. Thus, the values are not to be compared with the newly derived washing bottle results (see Table 2). Due to the large amount of backfilled waste packages in this emplacement area, the ratio of HTO/HT is measured to be about 1:3. In contrast, the return air of the whole mine has a ratio of 2:1.

A similar but much weaker effect is observed for the ratio of $(C-14)O_2/(C-14)H_4$. This ratio is measured to be about 3:1 in the same cavern (room 1s) whereas in the northern mine field this value is about 50:1. This difference is due to the chemical reaction of CO_2 with the alkaline filter ashes used as backfill. If the waste is cemented or the backfill ensures an alkaline environment, C-14 released as $(C-14)O_2$ will be readily bound in the alkaline medium. The effectiveness of this process is only limited by the binding capability of the medium and the duration of its alkaline character.

The situation in the southern mine field is somewhat different. Unless most of the waste is backfilled, the organic and inorganic components in room 2, as well as in the return air are comparable (see table 3). In rooms 1 and 3 the ratio of $(C-14)O_2/(C-14)H_4$ is less than 1:10. In the return air of the whole mine it is 3:1. This is in accordance with the results from the western mine field. A similar discrepancy is observed for HTO and HT. Although HTO is quite dominant in the residual voids of the southern mine field both chemical forms are found to be of similar height in the corresponding return air (see table 3). These discrepancies may be attributed to the fact, that the place of sampling in the rather large residual volumes may not be representative. A potential gradient in the concentrations of the different chemical species is likely because backfilled and non-backfilled wastes are at different locations. Furthermore the slow release through the dumping sluice, as well as the connections to the undercut drifts which are periodically ventilated, may also give rise to an inhomogeneous distribution inside the caverns.

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The retention of the noble gas Rn-222 is mainly determined by the time necessary for its diffusion through the backfill. This may vary over a wide range with respect to the distance. Due to its short half life of 3.8 days, the decay is likely to occur throughout this time. Hence, the daughter activities will rest inside the backfill and no release will occur.

In times of waste emplacement there is little chance for accurately measuring the release rates from waste in open and backfilled disposal areas, since waste emplacement as well as backfilling proceeds with time. One exception in ERAM was the backfilling of the first emplacement level in room 2 of the eastern mine field. There the conditions remained stable for a sufficient time. Drums, sometimes embedded in lost concrete shieldings, have been stacked up to a maximum height of 2.8m (up to 3 drums one upon another) in a cavern about 75m by 25m in area and 28m in height. Within 3 weeks, 2040m³ of radioactive waste have been covered with about 5500 tons of crushed salt. This provided a filling of the remaining voids, as well as a salt layer of at least 1.2m above the drums.

The results of the measurements in the return air stream before and after backfilling are given in table 4. For HTO the observed retaining factor of about 1.6 is surprisingly low. For the other volatile species of H-3 and C-14 only lower limits can be given. Nevertheless, in all cases a reduction of activity release is evident. The interpretation of data is difficult because there is no information to determine if the sorption/desorption equilibrium was reached and if there was any pre-existing activity concentration in the incoming mine air.

Table 2: Measured concentrations of volatile radionuclides inside the disposal rooms of the northern and western mine field and in the corresponding return air.

	Emplacement field/room					
	Northern mine field				Western mine field	
	Northern drift		Eastern drift		Room 1s	
Volume [m ³]	1,300		1,380		6,000	
Waste [m ³]	820		880		4,080	
Type of backfill	None		None		Filter ashes	
Barrier	Wall		Wall		Wall	
Residual cavity [m ³]	480		500		5?	
Activity concentration in the air of the residual cavity (mean values) [Bq/m ³]						
	Northern drift		Eastern drift		Room 1s	
Measuring technique	Washing bottle	Molecular sieve	Washing bottle	Molecular sieve	Washing bottle	Molecular sieve
HTO	1,000,000	160,000?	4,000,000	1,200,000	15	18
HT		460,000		160,000		3,000
(C-14)O ₂	80,000	83,000	70,000	74,000	400	730
(C-14)H ₄		810		2,000		220
Rn-222 ¹⁾	16,000		50,000		90	
Activity concentration in the specific return air (mean values) [Bq/m ³]						
	Northern mine field			Western mine field		
Measuring technique	Washing bottle		Molecular sieve		Washing bottle	Molecular sieve
HTO	1,000		410		0.8	4.3 ³⁾
HT			660			14.5 ³⁾
(C-14)O ₂	20		10		0.6	1.3 ³⁾
(C-14)H ₄			3.2			0.2 ³⁾
Rn-222 ²⁾	22			-		
Mean retardation time, t_r , of volatile radionuclides according to equation (2) [h]						
HTO	460		300		-	
HT			80			
(C-14)O ₂	670		1400		-	
(C-14)H ₄			80			
Rn-222	270			-		

¹⁾ Including Rn-222 from filter ashes

²⁾ After subtraction of 8Bq/m³ Rn-222 of natural origin (fresh air, concrete etc.)

³⁾ Measured values from 1997, not to be compared with washing bottle results

Table 3: Measured concentrations of volatile radionuclides inside the disposal rooms of the southern mine field and in the corresponding return air.

	Emplacement field/room					
	Southern mine field					
	Room 1		Room 2		Room 3	
Volume [m ³]	17,200		27,200		27,000	
Waste [m ³]	1,000		1,500		7,600	
Type of backfill	Filter ashes		Filter ashes		Filter ashes	
Barrier	Dumping sluice		Dumping sluice		Dumping holes	
Residual cavity [m ³]	8,000		1,500		10?	
Activity concentration in the air of the residual cavity (mean values) [Bq/m ³]						
Measuring technique	Washing bottle	Molecular sieve	Washing bottle	Molecular sieve	Washing bottle	Molecular sieve
HTO	20,000	18,000	2,800	3,200	2,000	7,000
HT		< 0.3		30		140
(C-14)O ₂	10	8.5	2	8.4	0.4	0.8
(C-14)H ₄		96?		5.7		12
Rn-222 ¹⁾	2,500		500		400	
Activity concentration in the specific return air (mean values) [Bq/m ³]						
Measuring technique	Washing bottle			Molecular sieve		
HTO	1.6 (15 ³⁾)			3.2		
HT				5.5		
(C-14)O ₂	0.3			0.1		
(C-14)H ₄				0.1		
Rn-222 ²⁾	17					

¹⁾ Including Rn-222 from filter ashes

²⁾ After subtraction of 8Bq/m³ Rn-222 of natural origin (fresh air, concrete etc.)

³⁾ Including times of undercut drift ventilation

Table 4: Measured tritium and C-14 concentrations in oxidised and non-oxidised form in the return air of room 2 in the eastern mine field before and after backfilling with crushed salt.

	Activity concentration in the return air [Bq/m ³]			
	HTO	HT+...	(C-14)O ₂	(C-14)H ₄ +...
Before backfilling	7.2 ± 1.1	2.6 ± 0.5	0.34 ± 0.09	0.14 ± 0.06
After backfilling	4.6 ± 1.0	< 0.68	< 0.11	< 0.09
Retaining factor	~1.6	> 3.1	> 2.3	> 1.0

RETARDATION BY BARRIERS

After closing the disposal rooms, the barriers (i.e. wall or dumping sluice) are not completely tight. However, in addition to the backfill material (if present), they also give rise to a retardation of the volatile radionuclides. Assuming an equilibrium, this delay can be regarded as mean retardation time, t_r , inside the cavern which is calculated according to the relation

$$t_r = \frac{c_i \cdot V_{res}}{c_o \cdot V_{air}/t} \quad (\text{Eq. 2})$$

where c_i and c_o are the concentrations inside the disposal room and outside in the corresponding return air. V_{res} is the residual volume in the disposal room and, V_{air}/t is the air flow rate from the specific disposal area or room.

Comparable retardation times of a few hundred hours (see table 2) are observed for HTO and (C-14)O₂ in the northern mine field. In general, the Rn-222 delay is shorter. This indicates that besides the simple delay of the noble gas Rn-222 by diffusion/convection through the barriers, there are additional retention mechanisms for the other volatile species. For the combustible/organic species HT and (C-14)H₄ the measured delay of about 80 hours is even shorter.

Unfortunately, due to the above-mentioned difficulties, no reliable retardation times can be extracted from the measurements in the southern mine field. Because the undercut drifts being connected to the disposal rooms require ventilation for monthly inspections, the ventilation conditions are not stable and a significant amount of volatile radionuclides is removed from the disposal rooms.

RELEASE RATES

Relative release rates have been derived from the measurements in the return air of specific mine openings and the upcast shaft of the Morsleben repository. These rates were measured by comparing the annual release with the actual activity in the corresponding disposal rooms. These values can be compared with the semi-empirical results determined in the earlier safety analyses of Morsleben and Konrad. Open and backfilled, as well as old (≤ 1991) and new (≥ 1994), disposal rooms have been investigated. The results are given in table 5.

Although our measured values for specific mine fields sometimes differ by one order of magnitude, no annual release rates higher than the values assumed in the safety analysis have been observed (see columns 3 and 5 of table 5). When considering the release rates derived for the whole mine (see column 6 of table 5) the HTO and C-14 release rates are lower by a factor of 5. In the case of HT, the release is

lower by about a factor of 15.

However, the HT release cannot be explained by radiolytic decomposition, but only by the fact that the waste itself contains tritium in this specific chemical form. Due to backfilling and closing the older disposal rooms, the 1997 measurements with molecular sieves in the return air now reveal that the amount of HT is similar to HTO. This was not the case earlier, when a higher tritium release, mainly as HTO, was observed (3). This observation is consistent with the measurements in the residual cavity of the western mine field.

Table 5: Comparison of the relative annual release rates taken for the safety analysis with new experimental data from the Morsleben repository.

Nuclide	Chemical form/ waste matrix	Relative annual release rates [%/a]			
		Used in the safety analysis		Measured values	
		For open rooms	For backfilled rooms	For specific mine fields	For the whole mine
H-3	HTO	5	0.5	0.7...2.2	1.0
	HT	≤ 5	≤ 5	0.2...1.9	0.3
C-14	Unspecified	0.5	0.5	0.01...0.5	0.08
Rn-222 ¹⁾	Immobilized	330 (5)	0	} 80...590 { (1.2...9)	} 320 (5) {
	Non-immobilized	5,300 (80)	0		

¹⁾ Release rates with respect to the Ra-226 activity in the waste, values in brackets are referred to the Rn-222 production rate

The measured Rn-222 release cannot completely be related to the activity disposed of because there is a significant amount of natural Rn-222 (typically 5 to 10Bq/m³) taken into the mine with the downcast air. Moreover, the filter ashes used for backfilling and the concrete structures in the mine contain a significant amount of natural Ra-226 and, thus, give rise to radon emanation. With this in mind, the measured values can only be regarded as upper limits which cannot directly be compared with the waste-specific release rates used for the safety analysis. Nevertheless, the deduced values agree with the release rate for immobilised waste and are lower by one order of magnitude if non-immobilised waste is assumed. Thus, the conservatism of the safety analysis is demonstrated.

ANALYSIS OF ACTIVITY TRANSPORT

When proceeding with the emplacement of waste, changes in the ventilation system had to be carried out throughout the years. In order to clarify if some releases of volatile radionuclides still occur via the old return air drifts or to detect a possible accumulation in non-ventilated areas, additional measurements were performed. Due to the complexity of the mine layout and lack of continuous ventilation at some locations, several measurements had to be performed at the same time in order to get a consistent picture of the potential activity transport. Because the necessary number of sampling devices as well as electric power at all locations of interest was not available calciumchloride (CaCl₂) samples were used to collect the humidity from the air. As such, tritium as HTO was taken as an indicator for potential continuing releases. Additional samples of kieserite from the sidewall or roof and, if present, brine samples from pools of condensed aerial humidity were taken and analysed for their tritium content.

The evaluation is based on the assumption that the tritium concentration in the humidity of the air (Bq/ml) is in equilibrium with that in the brine, kieserite, or the calciumchloride samples. To calculate the activity concentration in the air (Bq/m³) the absolute humidity needs to be known. It was measured at the beginning and the end of sampling. If necessary, the variation of the aerial humidity with time was reconstructed from the measurements at the above ground meteorological station. The results are summarised in table 6.

As compared to other well established techniques (i.e. washing bottles, condensers) the CaCl₂-measurements yielded similar results, whereas the kieserite samples gave higher values than the corresponding CaCl₂-measurements. This may be attributed to the fact that the kieserite samples provide the tritium concentration of a longer time scale and that, due to its hygroscopicity, it may not release tritium containing humidity. That means, no equilibrium with the aerial humidity is reached. This is not the case for the brine samples, where a permanent isotopic exchange with the aerial humidity is expected. For this reason, their evaluation gave similar results as the CaCl₂-measurements.

Table 6: Comparison of HTO concentrations obtained with different sampling techniques from different media.

No. of location	Kieserite samples		Brine samples		CaCl ₂ samples	
	[Bq/ml]	[Bq/m ³]	[Bq/ml]	[Bq/m ³]	[Bq/ml]	[Bq/m ³]
5			350	4200	200	2400
7	270	3200			180	2200
8	34	260			43	200
9	3.3	25			1.4	7
10	3.1	23			3.5	17
15	17	130			36	160
19			82	620	94	400
28	4.1	31			8.2	36
33	11	80			5.4	30

CONCLUSION

Besides the retardation of volatile radionuclides by convective flow through the backfill and technical barriers, an effective retention is achieved for HTO and (C-14)O₂ when filter ashes are used as backfill. For crushed salt and newly covered waste there is also a measurable retention but no information is available about the time required for the system to reach equilibrium. In the case of a reversible binding or a limited binding capacity, the retention will only be temporary.

Retardation times for the volatile radionuclides of some hundred hours have been obtained for the northern mine field, where the emplacement drifts are closed by walls. But besides for Rn-222 (and also Rn-220) of which a significant amount decays throughout this time no durable reduction of activity release is achieved by closing disposal rooms with walls. Only backfilling ensures a retention of volatile radionuclides for longer periods of time.

The observed annual release rates were below the semi-empirical values used in earlier safety analyses.

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Since these waste specific release rates were used for the modelling of activity transport in the shut-down phase, confidence is gained for the prediction of activity release under the specific conditions when the repository will be closed.

The sampling of aerial humidity with CaCl_2 turned out to be a very simple but also effective and accurate method to simultaneously measure the tritium concentration at many different locations in the mine without needing any power connection. Within a four-weeks-sampling period, HTO detection limits below 0.2Bq/m^3 were achieved. As a result, neither a significant release via the old return air drifts nor an inadmissible accumulation of activity due to the missing ventilation was detected.

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FOOTNOTES

¹ The annual release rates for Rn-222 are referred to the activity of the mother nuclide Ra-226.

² Not considered in the ERAM safety analysis.

³ Not to be expected in ERAM

REFERENCES

- (1) D. EHRLICH, T. M. IBACH, V. KUNZE, and W. MÜLLER, "The Role of Volatile Radionuclides in the Safety Analysis of Underground Repositories", DisTec '98, Proc. Int. Conf. on Radioactive Waste Disposal, Hamburg, Germany, September 9-11 1998, Kontec Gesellschaft für technische Kommunikation mbH (1998) 470
- (2) H. BEYER, G. EIGENWILLIG, R. BECK, and E.W.HAAS, "Laborversuche zur Rückhaltefähigkeit von Versatzmaterialien aus der Grube Konrad gegenüber radioaktiven Gasen", 20. Jahrestagung des Fachverbandes für Strahlenschutz e.V., Basel, Switzerland, October 6-9 1987, FS-87-44-T (1987) 358
- (3) D. EHRLICH, V. KUNZE, and T. M. IBACH, "Release and Detection of Tritium in the Morsleben Repository Plant (ERAM)", KONTEC '97, Proc. 3rd Symposium "Conditioning of Radioactive Operational & Decommissioning Wastes", March 19- 21 1997, Kontec Gesellschaft für technische Kommunikation mbH (1997) 730